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NUREG-2215

Standard Review Plan for Spent Fuel Dry Storage Systems and Facilities

Final Report

Office of Nuclear Material Safety and Safeguards

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Manuscript Completed: February 2020
Date Published: April 2020

Office of Nuclear Material Safety and Safeguards

ABSTRACT

This Standard Review Plan (SRP) provides guidance to the U.S. Nuclear Regulatory Commission (NRC) staff for reviewing safety analysis reports (SARs) for (1) a Certificate of Compliance (CoC) for a dry storage system for use at a general license facility and (2) a specific license for a dry storage facility that is either an independent spent fuel storage installation (ISFSI) or a monitored retrievable storage installation (MRS). This SRP does not apply to wet storage ISFSIs or MRSs (e.g., GE-Morris). NUREG-2215 is a consolidation of existing guidance for staff's use when reviewing applications for licenses and certificates for spent fuel dry storage systems and facilities, and as such, it is not intended to offer new or differing guidance.

The objectives of this SRP are to assist the NRC staff in its reviews by doing the following:

- promoting a consistent regulatory review of a SAR for an ISFSI or MRS license, or for a CoC
- promoting quality and uniformity of these reviews across each technical discipline
- presenting a basis for the review's scope
- identifying acceptable approaches to meeting regulatory requirements
- suggesting possible evaluation findings that can be used in the safety evaluation report

This SRP was published for public comment and the responses to those comments are available at ML19303C896. This NUREG is a rule as defined in the Congressional Review Act (5 U.S.C. 801-808). However, the Office of Management and Budget has not found it to be a major rule as defined in the Congressional Review Act.

This SRP may be revised and updated as the need arises on a chapter-by-chapter basis to clarify the content, correct errors, or incorporate modifications approved by the Director of the Division of Fuel Management. Comments, suggestions for improvement, and notices of errors or omissions should be sent to and will be considered by the Director, Division of Fuel Management, Office of Nuclear Material Safety and Safeguards, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001.

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ABBREVIATIONS AND ACRONYMS

ACI	American Concrete Institute
ADAMS	Agencywide Documents Access and Management System
AISC	American Institute of Steel Construction
ALARA	as low as is reasonably achievable
ANO	Arkansas Nuclear One
ANS	American Nuclear Society
ANSI	American National Standards Institute
APSR	axial power shaping rod
ASCE	American Society of Civil Engineers
ASD	allowable stress design
ASME	American Society of Mechanical Engineers
ASNT	American Society for Nondestructive Testing
ASTM	American Society for Testing and Materials
AWS	American Welding Society
B_4C	boron carbide
B&PV	boiler and pressure vessel
BPR	burnable poison rod
BPRA	burnable poison rod assembly
BR	breathing rate
BWR	boiling-water reactor
CDE	committed dose equivalent
CEDE	committed effective dose equivalent
CFD	computational fluid dynamics
CFR	Code of Federal Regulations
CISCC	chloride-induced stress-corrosion cracking
CoC	certificate of compliance
CR	control rod
CRC	commercial reactor critical
DBA	design-basis accident
DCF	dose conversion factor
DDE	deep dose equivalent
DOE	U.S. Department of Energy
DP	decommissioning plan
D/Q	deposition parameter
DSF	dry storage facility
DSS	dry storage system
EALF	energy of average neutron lethargy causing fission
EDEX	effective dose equivalent from external exposure

EIA	Energy Information Administration
EP	emergency plan
EPA	U.S. Environmental Protection Agency
EPRI	Electric Power Research Institute
FEA	finite element analysis
FPP	fire protection program
GBC	generic burnup credit
GCI	grid convergence index
GTCC	greater-than-Class-C (waste)
GTRF	grid-to-rod fretting
HLW	high-level radioactive waste
HPS	Health Physics Society
H/X	hydrogen-to-fissile atom ration
I&C	instrumentation and controls
IBA	integral burnable absorber
IBC	International Building Code
ICRP	International Commission on Radiological Protection
IEEE	Institute of Electrical and Electronics Engineers
ISFSI	independent spent fuel storage installation
ISG	Interim Staff Guidance
k_{eff}	effective neutron multiplication factor
LDE	lens (eye) dose equivalent
LWR	light-water reactor
MMS	metal matrix composite
MofS	margin of safety
MOX	mixed-oxide
MPC	multipurpose cask
MRS	monitored retrievable storage installation
MT	magnetic particle testing
MTHM	metric ton heavy metal
MTU	metric ton of uranium
NCRP	National Council on Radiation Protection and Measurements
NDE	nondestructive examination
NFH	nonfuel hardware
NRC	U.S. Nuclear Regulatory Commission
NRR	Office of Nuclear Reactor Regulation

NSA	neutron source assembly
OFA	optimized fuel assembly
O/M	oxygen to metal
ORNL	Oak Ridge National Laboratory
P&ID	piping and instrumentation diagram
PAR	protective action recommendation
PM	project manager
PMF	probable maximum flood
PMP	probable maximum precipitation
PRA	poison rod assembly
PT	liquid (dye) penetrant testing
PWR	pressurized-water reactor
QA	quality assurance
QAPD	quality assurance program description
RCA	radiochemical assay
RES	NRC Office of Nuclear Regulatory Research
RG	regulatory guide
RT	radiographic examination
SAE	Site Area Emergency
SAR	safety analysis report
SDE	shallow (skin) dose equivalent
SER	safety evaluation report
SFA	spent fuel assembly
SFPO	NRC Spent Fuel Project Office
SFST	NRC Division of Spent Fuel Storage and Transportation
SI	système international d'unités (International System of Units)
SNF	spent nuclear fuel
SRP	Standard Review Plan
SSCs	structures, systems, and components
TEDE	total effective dose equivalent
TLAA	time-limiting aging analysis
TSUNAMI	Tools for Sensitivity and Uncertainty Methodology Implementation
U_3O_8	triuranium octoxide
UO_2	uranium dioxide
UT	ultrasonic testing
X/Q	atmospheric dispersion

UNITS

Bq	becquerel
°C	degrees Celsius
Ci	curie
cm	centimeter
cm^2	square centimeter
cm^3	cubic centimeter
°F	degrees Fahrenheit
ft	foot
ft^2	square foot
ft^3	cubic foot
g	gram
GWd/MTHM	gigawatt days per metric ton heavy metal
GWd/MTU	gigawatt days per metric ton of uranium
hr	hour
in.	inch
K	Kelvin
kg	kilogram
kgf	kilograms force
km	kilometer
ksi	thousand pounds per square inch
lb	pound
m	meter
m^2	square meter
m^3	cubic meter
mb	millibar
MeV	mega electron volt
mCi	milliCurie (one-thousandth of a Curie)
mg	milligram (one-thousandth of a gram)
mi	mile
mJ	millijoule
mm	millimeter (one-thousandth of a meter)
MPa	megapascal (million pascals)
mph	miles per hour
mrem	millirem (one-thousandth of a rem)
ms	millisecond
mSv	millisievert (one-thousandth of a sievert)
MWd/MTHM	megawatt days per metric ton heavy metal
MWd/MTU	megawatt days per metric ton of uranium
Pa.	Pascal
ppm	parts per million
psf	pounds per square foot
psi	pounds per square inch

psig	pounds per square inch gauge
s	second
Sv	sievert
μCi	microcurie (one-millionth of a curie)
yr	year

GLOSSARY

The U.S. Nuclear Regulatory Commission (NRC) staff has defined the terms provided in this section for the purposes of this Standard Review Plan (SRP).

Acceptance Test. Tests conducted by the applicant to ensure that the material or component produced was fabricated in compliance with the material or component design requirements of the application. Acceptance tests are also used to ensure that the process is operating in a satisfactory manner by using statistical data for selected measurable parameters.

Accident Condition. The extreme level of an event or condition, which has a specified resistance, limit of response, and requirement for a given level of continuing capability, which exceeds off-normal events or conditions. Accident conditions include both design-basis accidents and conditions caused by natural and manmade phenomena. These conditions include events that are Design Events III and IV in American National Standards Institute/American Nuclear Society (ANSI/ANS) 57.9, "Design Criteria for an Independent Spent Fuel Storage Installation (Dry Storage Type)."

Aging Management Program. See definition in Title 10 of the *Code of Federal Regulations* (10 CFR) 72.3, "Definitions."

Amendment of a License or CoC. An application for amendment of a license or a CoC is generally submitted when a holder of a specific license or CoC desires to change the license or CoC (including a change to the technical specifications that accompany the license or CoC). The application must fully describe the desired change(s) and the reason(s) for such change(s), and follow as far as applicable the form prescribed for original applications. See 10 CFR 72.56, "Application for Amendment of License," and 10 CFR 72.244, "Application for Amendment of a Certificate of Compliance".

Areal Density. Mass per unit area, usually expressed in grams per square centimeters (g/cm^2). In this SRP, this term is used to describe the distribution of neutron absorber content in a material.

Assembly Defect. Any change in the physical as-built condition of the SNF assembly except for normal in-reactor changes such as elongation from irradiation growth or assembly bow. Examples of assembly defects include (a) missing rods, (b) broken or missing grids or grid straps (spacers), and (c) missing or broken grid springs.

As Low As Is Reasonably Achievable (ALARA). See 10 CFR 20.1003, "Definitions," and 10 CFR 72.3, "Definitions."

Basic Safety Criteria. The following are considered the basic safety criteria for design of the spent fuel storage system or facility:

- Maintain subcriticality.
- Prevent the release of radioactive material above amounts that ensure compliance with regulatory dose requirements, including ALARA.
- Ensure that doses do not exceed the levels that ensure compliance with regulatory dose requirements, including ALARA.

Benchmarking. Establishing a predictable relationship between calculated results and reality. The main goal of benchmarking is to gain a quantitative understanding of the difference, or "bias," between calculated and expected results and the uncertainty in this difference (bias uncertainty). Also known as code or method "validation."

Breached Spent Nuclear Fuel (SNF) Rod. An SNF rod with cladding defects that permit the release of gases or solid fuel particulates from the interior of the fuel rod. SNF rod breaches include pinhole leaks, hairline cracks or gross ruptures.

Burnable Poison Rod (BPR). A rod containing neutron-absorbing material that, during long-term neutron flux exposure, loses its absorbing capability at a controlled rate.

Burnable Poison Rod Assembly (BPRA). An assembly of BPRs used to absorb neutrons created in the nuclear reactor to control the power produced in the associated fuel assembly during the early core life. The BPRs are inserted into the assemblies through the upper end fittings of the assembly and held in place against lift forces in the core by a retainer mechanism. BPRAs may be approved for storage with SNF assemblies when stored within the assembly envelope.

Burnup. The measure of the thermal power produced in a specific amount of nuclear fuel through fission, usually expressed in units of gigawatt days per metric ton of uranium (GWd/MTU). For the purpose of assessing the allowable contents, the maximum burnup(s) of the fuel should be specified in terms of the average burnup of the entire fuel assembly (i.e., assembly average). Additionally, for SNF criticality analyses that rely on burnup credit, a minimum required assembly average burnup will be specified. For the purpose of assessing fuel cladding integrity in the materials review, the rod with the highest burnup within the fuel assembly should be specified in terms of peak rod average burnup. For assemblies with mixed oxide (MOX) or thoria rods, the units will usually be megawatt days per metric ton heavy metal (MWd/MTHM).

Can for Damaged Fuel (aka Damaged Fuel Can). A metal enclosure that is sized to confine damaged SNF contents. A can for damaged fuel must satisfy fuel-specific and dry storage system (DSS)-related functions for undamaged SNF, as required by the applicable regulations.

Canister. In a DSS for SNF, a metal cylinder that is sealed at both ends and may be used to perform the function of confinement. Typically, a separate overpack performs the radiological shielding and physical protection functions during storage on the storage pad, while a separate transfer cask performs these functions during operations such as canister loading, preparation for storage, and transfer into the storage overpack.

Canning. One method to store damaged or consolidated SNF or nuclear fuel debris, placing it in a separate container (e.g., can for damaged fuel), and confine it in such a way that degradation of the fuel during storage will not pose operational safety problems with respect to its removal from storage (per 10 CFR 72.122(h)(1)).

Cask. See Spent Fuel Storage Cask.

Certificate of Compliance (CoC). See 10 CFR 72.3.

Certificate of Compliance Holder (CoC Holder). See 10 CFR 72.3.

Certificate of Compliance User (CoC User). The general licensee that has loaded a DSS, or purchased a DSS and plans to load it, in accordance with a CoC issued under 10 CFR Part 72, "Licensing Requirements for the Independent Storage of Spent Nuclear Fuel, High-Level Radioactive Waste, and Reactor-Related Greater than Class C Waste."

Collective Dose. See 10 CFR 20.1003.

Committed Dose Equivalent (H_T 50). See 10 CFR 20.1003.

Committed Effective Dose Equivalent (H_E 50). See 10 CFR 20.1003.

Co-locate. To locate a 10 CFR Part 72 facility on the same site as another fuel cycle or other radioactive materials facility. Facilities that are co-located may share common facilities. For example, a specific license ISFSI may be co-located at a power reactor site licensed under 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities," or 10 CFR Part 52, "Licenses, Certifications, and Approvals for Nuclear Power Plants." General license ISFSIs must be located at a power reactor site that is authorized to possess or operate nuclear power reactors under 10 CFR Parts 50 or 52. These co-located ISFSIs may share the storage pad (as a common facility) with materials stored under the 10 CFR Part 50 or 52 license (e.g., reactor-related greater-than-Class-C (GTCC) waste) also being stored on the same storage pad as the SNF that is stored under the 10 CFR Part 72 license.

Confinement Boundary. In a DSS for SNF, the outer boundary of the confinement system that prevents the release of radioactive material to the environment.

Confinement. The ability to limit or prevent the release of radioactive substances into the environment.

Confinement System. See 10 CFR 72.3.

Confirmatory Calculations. Independent calculations performed by the NRC reviewer to confirm the adequacy of the applicant's analyses. These calculations do not replace, nor do they endorse, the applicant's design calculations.

Construction. Includes materials, design, fabrication, installation, examination, testing, inspection, maintenance, and certification as required in the manufacture and installation of structures, systems, and components (SSCs).

Controlled Area. See 10 CFR 72.3. See also10 CFR 20.1003. The definition in 10 CFR 20.1003 is broader in scope and allows for, or includes, establishment of access controls to areas within the site for any reason (for radiation protection).

Critical. The state of a fissile material system where the rate of production of neutrons, from fission and other sources, is equal to the rate of loss, from absorption and leakage. A system that is exactly critical will have a constant population of neutrons.

Damaged Spent Nuclear Fuel. Any fuel rod or fuel assembly that cannot meet the pertinent fuel-specific, DSS, or dry storage facility (DSF)-related regulations in 10 CFR Part 72. See Chapter 8 of this SRP.

Deep-Dose Equivalent (H_D). See 10 CFR 20.1003.

Degradation. Any change in the properties of a material that adversely affects the performance of that material; adverse alteration.

Design Bases. See 10 CFR 72.3.

Design Criteria. The criteria the facility or cask designer uses to show that the design meets all of the requirements in 10 CFR Part 72. Design criteria can include, but is not limited to, safety margins, maximum stresses, maximum or minimum material temperatures, dose rates, and k-effective (k_{eff}).

Design-Basis Earthquake. The design earthquake ground motion for a site where a DSF may be used, or where a DSF may be sited. DSF siting requirements for a specific license are determined in accordance with 10 CFR 72.102 or 10 CFR 72.103.

Design Event (I, II, III, or IV). Conditions and events as defined and used for an ISFSI in ANSI/ANS 57.9.

Dry Storage System. A system that typically uses a cask or canister in an overpack as a component in which to store SNF in a dry environment. A DSS provides confinement, radiological shielding, sub-criticality control, structural support, and passive cooling of its SNF during normal, off-normal, and accident conditions.

Dry Storage. The storage of SNF in a DSS, which typically involves drying the DSS cavity and backfilling with an inert gas.

Emergency Power. The power supply that is selected to furnish electric energy to instruments, utility service systems, the central security alarm station, and operating systems in amounts sufficient to allow safe storage conditions to be maintained and to permit continued functioning of all systems essential to safe storage when the primary power supply is not available.

Exemption. The request for an exception from application of a specific regulatory requirement that otherwise is required. The NRC must explicitly approve an exemption and will only do so if the applicable regulatory requirements are met. See, for example, 10 CFR 72.7, "Specific exemptions".

General License. Authorizes the storage of SNF in an ISFSI at power reactor sites to persons (i.e., general licensee) authorized to possess or operate nuclear power reactors under 10 CFR Part 50 or 10 CFR Part 52. The general license is limited to (1) that SNF which the general licensee is authorized to possess at the site under the specific 10 CFR Part 50 or 10 CFR Part 52 license for the site, and (2) storage of SNF in casks approved under the provisions of 10 CFR Part 72, Subpart L, "Approval of Spent Fuel Storage Casks." See 10 CFR 72.210, "General license issued," and 10 CFR 72.212(a)(1)–(2).

Gross Breach. A breach in the spent fuel cladding that is larger than either a pinhole leak or a hairline crack and allows the release of particulate matter from the spent fuel rod.

Hairline Crack. A minor SNF cladding defect that will not permit significant release of particulate matter from the spent fuel rod and therefore presents a minimal as low-as-is-reasonably-achievable concern during fuel handling operations.

High Burnup Fuel. SNF with assembly average burnup (see "Burnup") that exceeds 45 GWd/MTU.

Hoop Stress. The tensile stress in cladding wall in the circumferential orientation of the fuel rod.

Insolation. Exposure of a material to sunlight; the rate of solar radiation received per unit area.

Intact Spent Nuclear Fuel. Any fuel rod or fuel assembly that can meet the pertinent fuel-specific or system-related regulations for the transportation package (10 CFR Part 71, "Packaging and Transportation of Radioactive Material") or dry storage system (10 CFR Part 72). Intact SNF rods may not contain pinholes, hairline cracks, or gross breaches. Intact SNF assemblies may have assembly defects if able to meet the pertinent fuel-specific or DSS-related regulations.

Intended Function. A design-bases function defined as either (1) important to safety or (2) failure of which could impact a safety function.

_K_{eff}. "k-effective_." Effective neutron multiplication factor including all biases and uncertainties at a 95-percent confidence level for indicating the level of subcriticality relative to the critical state. At the critical state, $k_{eff} = 1.0$.

Lens Dose Equivalent. See 10 CFR 20.1003.

Low Burnup Fuel. SNF with an assembly average burnup (see "Burnup") that does not exceed 45 GWd/MTU.

Margin of Safety (Safety Margin) (MofS). This term may be defined, through a factor of safety, f.s. = capacity/demand, as MofS = F.S. (capacity/demand) – 1 (with minimum acceptable MofS > 0.0)."

Member of the Public. See 10 CFR 20.1003.

Misloading. The placement of SNF in a DSS or DSF storage container in a configuration not supported by the design basis or authorized by the certificate or license and technical specifications for the DSS or DSF container. For reactor-related GTCC waste and solidified high-level radioactive waste (HLW) containers at a DSF, the placement of waste in these containers that do not meet the characteristics of the container's allowable contents.

Monitored Retrievable Storage Installation. See 10 CFR 72.3.

Monitoring. Data collection to determine the status of a DSS or DSF SSC and to verify the continued efficacy of the SSC on the basis of measurements of specified parameters, including temperature, direct radiation, radioactive effluents, functionality, and characteristics of the SSC. With respect to direct radiation and radioactive effluents, according 10 CFR 20.1003, monitoring means the measurement of radiation levels, concentrations, surface area concentrations, or quantities of radioactive material and the use of the results of these measurements to evaluate potential exposures and doses.

Neutron Absorber. Also known as "poison." Materials that have a high neutron absorption cross section and are used to absorb neutrons to make a fissile material system less reactive. They are used to ensure subcriticality during normal, off-normal, and accident conditions in containers of fissile materials.

Nondestructive Examination (NDE). Testing, examination, or inspection of a component that does not affect the functionality and performance of the component. NDE can be broadly divided into three categories: visual, surface, and volumetric examinations. Additional information may be found in the American Society of Mechanical Engineers Boiler and Pressure Code, Section V, "Nondestructive Examination," Appendix A.

The following NDE-related terms are presented in order of increasing severity:

Discontinuity: An interruption in the normal physical structure of a material. Discontinuities may be unintentional (such as those formed inadvertently during the fabrication process) or intentional (such as a drilled hole).

Indication: Sign of a discontinuity observed when using an NDE method.

Flaw: An imperfection in an item or material that may or may not be harmful.

Defect: A flaw that, because of its size, shape, orientation, location, or other properties, is rejectable to the applicable construction code. Defects may be detrimental to the intended service of a component, and the component must be repaired or replaced.

Common NDE examination methods include the following:

LT leak testing

MT magnetic particle examination

PT liquid penetrant testing

RT radiographic examination

UT ultrasonic examination

VT visual examination

Non-Fuel Hardware. Hardware that is not an integral part of a fuel assembly. This is the term used to identify what the regulation refers to as "other radioactive materials associated with fuel assemblies" (see SNF definition in 10 CFR 72.3). While not integral to the assembly, it includes those items that are designed to operate and are positioned or operated within the envelope of the fuel assembly during reactor operation and are stored within the assembly envelope in the storage container. Typical examples of non-fuel hardware include: BPRAs, control element assemblies, thimble plug assemblies, and boiling-water reactor (BWR) fuel channels. Examples of items that do not meet this definition include boron sources, BWR in-core instruments, and BWR control blades.

Non-Mechanistic Event. An event, such as cask tipover, that should be evaluated for acceptable system capability, although a cause for such an event is not identified in the analyses of off-normal and accident events and conditions.

Normal Events and Conditions. Conditions that are intended operations, planned events, and environmental conditions that are known or reasonably expected to occur with high frequency during storage operations. "Normal" refers to the maximum level of an event or condition that is expected to routinely occur (similar to Design Event I in ANSI/ANS 57.9). The DSS and DSF SSCs are expected to remain fully functional and to experience no temporary or permanent degradation of that functionality from normal operations, events, and conditions. Specific normal conditions to be addressed are evaluated for the DSS or DSF and are documented in the SAR for that system or facility.

Normal Means. The ability to move a fuel assembly with a crane and grapple used to move undamaged assemblies at the point of cask loading. The addition of special tooling or modifications to the assembly to make the assembly suitable for lifting by crane and grapple does not preclude the assembly from being considered moveable by normal means.

Off-Normal Events or Conditions. An event or condition that, although not occurring regularly, can be expected to occur with moderate frequency and for which there is a corresponding maximum specified resistance, limit of response, or requirement for a given level of continuing capability. Off-normal events and conditions are similar to Design Event II in ANSI/ANS 57.9. The DSS and DSF SSCs are expected to experience off-normal events and conditions without permanent degradation of capability to perform its full function (although operations may be suspended or curtailed during off-normal conditions) over the full storage term (the license period for a specific license facility or the storage period equivalent to the certificate term for a DSS). Off-normal events or conditions are referred to as anticipated occurrences in 10 CFR 72.104, “Criteria for Radioactive Materials in Effluents and Direct Radiation from an ISFSI or MRS.”

Overpack. A heavy-walled concrete, metal, or combined concrete and metal structure designed to store SNF, HLW, or reactor-related GTCC in canisters. The overpack provides physical protection of canisters and radiological shielding, while allowing passive cooling. For the purposes of this SRP, the term overpack will be used generically in the horizontal, vertical, and underground storage of canisters.

Pinhole Leak. A minor cladding defect that will not permit significant release of particulate matter from the SNF rod and therefore present a minimal ALARA concern during fuel-handling operations.

Preferential Loading. A non-uniform loading configuration of SNF assemblies within a DSS that typically is specified by assigning a fuel zone designation to each basket cell and specifying limiting nuclear and physical parameters of SNF assemblies that can be loaded into each zone. Preferential loading is often used as a means to optimize allowable SNF parameters (e.g., burnup, cooling time, decay heat) while satisfying the shielding, criticality, and thermal performance objectives of the storage container or system.

Qualification Test. A test, or series of tests, conducted at least once for a given manufacturing process and set of material specifications to demonstrate the quality and durability of the component, such as neutron absorber product, over the licensed/certified service life of the facility/storage container.

Rad. The special unit of absorbed dose, which is defined in 10 CFR 20.1004, “Units of Radiation Dose.”

Ready Retrieval. The ability to safely remove SNF, reactor-related GTCC waste, or HLW from storage for further processing or disposal.

Real Individual. Any individual who lives, works, or engages in recreation or other activities close to the DSF for a significant portion of the year. The requirements in 10 CFR 72.104 include annual dose limits for real individuals located beyond the controlled area boundary. For the purposes of these limits, doses to nuclear or radiation workers while they are working are excluded.

Recovery. The capability of returning the stored radioactive materials from an accident to a safe condition without endangering public health and safety or causing significant or unnecessary exposure to workers. Any potential release of radioactive materials during recovery operations must not result in doses or radiation exposures that exceed the limits in 10 CFR Part 20, "Standards for Protection Against Radiation." Doses during recovery operations are included in the dose estimates for accidents, the total of which must not exceed the limits in 10 CFR 72.106, "Controlled Area of an ISFSI or MRS."

Restricted Area. An area to which access is limited by the licensee for the purpose of protecting individuals against undue risks from exposure to radiation and radioactive materials. Restricted areas do not include areas used as residential quarters, but separate rooms in a residential building may be set apart as a restricted area (10 CFR 20.1003).

Retrievability. See *Ready Retrieval*. Storage systems must be designed to allow ready retrieval of SNF, HLW, and reactor-related GTCC waste for further processing or disposal (10 CFR 72.122(l)).

Safety Analysis Report (SAR). In the context of this SRP, the report submitted to the NRC staff by an applicant for a CoC for a DSS, or for a specific license for a DSF, to present information related to the design and operations of the system or facility. The SAR provides the justification and analyses to demonstrate that the design meets regulatory requirements and acceptance criteria (10 CFR 72.24, "Contents of Application: Technical Information," 10 CFR 72.230(a)). The SAR is submitted to obtain approval for the DSF or DSS. The final SAR is defined in 10 CFR 72.48(a)(5).

Safety Evaluation Report (SER). In the context of this SRP, the report prepared by the NRC staff that describes the basis for the NRC's approval and issuance of a specific license for a facility or a CoC for a DSS. The SER also identifies the recommended license/CoC conditions and technical specifications ("operating controls and limits" or "conditions of use") and the bases for those conditions and technical specifications.

Safety Functions. The functions that DSS and DSF SSCs important to safety (see 10 CFR 72.3) are designed to maintain, perform, or both, include the following:

- protection against environmental conditions
- content temperature control
- radiation shielding
- confinement
- subcriticality control

Shallow Dose Equivalent (H_S). See 10 CFR 20.1003.

Spent Nuclear Fuel. See 10 CFR 72.3.

Spent Fuel Storage Cask. See 10 CFR 72.3.

Standby Power. The power supply that is chosen to furnish electric energy to select electrical equipment that is not important to safety when the primary (i.e., normal) power supply is not available. Standby power cannot be used interchangeably with emergency power.

Storage Container. The generic term used to refer to the containers of radioactive materials for which the DSS or DSF is certified or licensed for storage. This term covers canister-based and non-canister-based DSSs. For canister-based DSSs, it can be used to refer to the canister alone or the configuration of the canister in an overpack or transfer cask. The term also refers to non-DSS SNF storage containers, storage containers for GTCC waste, and storage containers for HLW. If storage of these wastes involves canister-based designs that include transfer casks and overpacks, the term is applied in the same manner as for canister-based DSSs.

Structures, systems, and components important to safety. See 10 CFR 72.3.

Subcritical. The state of a fissile material system where the rate of production of neutrons, from fission and other sources, is less than the rate of loss, from absorption and leakage. A system that is subcritical will have a decreasing population of neutrons.

Supplemental Cooling. Additional temporary external forced cooling (circulating water or air flow) of a DSS or DSF storage container during loading operations or during transfer operations.

Supplemental Shielding. Shielding that is not an integral part of DSS or DSF SSCs used to handle, transfer, or store SNF, GTCC waste, or HLW. There are three general types of supplemental shielding. The first type consists of engineered features, such as earthen berms or shield walls that are used to ensure compliance with the 10 CFR Part 72 dose limits. The second type consists of items that are used in operations for ALARA purposes but are generally not credited in the SAR dose rate and dose analyses. These items include, for example, lead blankets. The third type consists of items that are necessary for personnel to safely perform storage activities and meet relevant dose limits and which are credited in the SAR dose rate and dose analyses. Examples of storage activities for this third type include canister welding and decontamination. These items include, for example, thick steel shields that surround the transfer cask during activities to prepare the canister for storage or to transfer the canister to the storage overpack. The SRP may also refer to the second and third types as temporary shielding.

Thimble Plug Assembly. An assembly of short rods inserted into the assembly's guide tubes to restrict the flow of coolant through a fuel assembly. This component is designed for operations within the fuel assembly envelope and, when stored with SNF, fits within that envelope.

Total Effective Dose Equivalent. See 10 CFR 20.1003.

Undamaged Spent Nuclear Fuel. Any fuel rod or fuel assembly that can meet the pertinent fuel-specific or DSS-related regulations. Undamaged SNF rods may contain pinholes or hairline cracks, but may not contain gross breaches. Undamaged SNF assemblies may have assembly defects if able to meet the pertinent fuel-specific or DSS-related regulations.

Unrestricted Area. An area to which access is neither limited nor controlled by the licensee (10 CFR 20.1003).

Validation. See Benchmarking.

Volume Percent. The percent of a mole of the material that is present in a volume equal to the standard volume for the material as a gas; the volume occupied by 1 mole of the material as a gas at standard conditions for gases (760 millimeters of mercury (760 torr) for pressure and 0 degree Celsius (32 degrees Fahrenheit) for temperature).

INTRODUCTION

Purpose of the Standard Review Plan

This Standard Review Plan (SRP) is intended to provide guidance to the U.S. Nuclear Regulatory Commission (NRC) staff for reviewing safety analysis reports (SARs) for the following:

- Certificate of Compliance (CoC) for a dry storage system (DSS) for use at a nuclear power reactor authorized to possess or operate under Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, "Domestic Licensing of Production and Utilization Facilities," or 10 CFR Part 52, "Licenses, Certifications, and Approvals for Nuclear Power Plants"

- specific license for a dry storage facility (DSF) that is either an independent spent fuel storage installation (ISFSI) or a monitored retrievable storage installation (MRS)

This SRP does not apply to wet storage ISFSIs or MRSs (e.g., GE Morris), but does have information related to pools for repackaging at a DSF. Refer to NUREG-1567, "Standard Review Plan for Spent Fuel Dry Storage Facilities," for information regarding the review of wet pools (such as for spent fuel repackaging, loading, un-loading).

Note that the guidance for specific license applications is intended to cover all specific license DSFs, including those co-located with 10 CFR Part 50 and 10 CFR Part 52 facilities and those that are not co-located with these other facilities. For specific license DSFs that are co-located with 10 CFR Part 50 and 10 CFR Part 52 facilities, technical discipline reviews should appropriately factor this condition into the evaluation. The applicant may refer to documents submitted to the Commission in connection with applications for a license under 10 CFR Part 50 or 10 CFR Part 52, as long as the applicant can demonstrate that the information is applicable to the requirements in 10 CFR Part 72, "Licensing Requirements for the Independent Storage of Spent Nuclear Fuel, High-Level Radioactive Waste, and Reactor-Related Greater than Class C Waste" and still be factual.

This introduction provides an overview of DSSs and DSFs along with the function of the SAR in the review process to assist the NRC project manager coordinate the review effort. It is also intended to assist individual technical reviewers understand how specific evaluations should be coordinated and integrated across disciplines to produce a comprehensive safety evaluation report (SER). In accomplishing their evaluations, the reviewers should coordinate their efforts to achieve a determination of the sufficiency of the application.

This SRP may be revised and updated as the need arises on a chapter-by-chapter basis to clarify the content, correct errors, or incorporate modifications approved by the Director of the Division of Spent Fuel Management. Comments, suggestions for improvement, and notices of errors or omissions should be sent to and will be considered by the Director, Division of Spent Fuel Management, Office of Nuclear Material Safety and Safeguards, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001.

Types of Licenses for Use of Dry Storage Systems and Dry Storage Facilities

A license is required for the receipt, handling, storage, and transfer of spent nuclear fuel (SNF), high-level radioactive waste (HLW), and reactor-related greater-than-Class-C (GTCC) waste. There are two types of ISFSI licenses: specific and general. An MRS license is a specific license.

The regulations in 10 CFR Part 72 also provide for issuance of a Certificate of Compliance (CoC) for use with the general license.

A specific license authorizes a person (see the definition in 10 CFR 72.3, “Definitions”) to receive, handle, store, and transfer SNF, and reactor-related GTCC. A specific-license ISFSI may be co-located with a reactor facility or may be located away from a reactor facility.

A specific license for an MRS (see the definition in 10 CFR 72.3) authorizes DOE to construct and operate a DSF to receive, transfer, package, possess and safeguard SNF, HLW, and reactor-related GTCC waste. HLW is only authorized for storage in an MRS and not in a specifically licensed or generally licensed ISFSI (see 10 CFR 72.2, “Scope”).

The second type of ISFSI license is a general license. A general license authorizes storage of SNF in an ISFSI at power reactor sites to persons authorized to possess or operate a power reactor under 10 CFR Part 50 or 10 CFR Part 52, “Licenses, Certifications, and Approvals for Nuclear Power Plants” (see 10 CFR 72.210). A general licensee may construct and operate an ISFSI and store SNF using NRC-approved DSSs (see 10 CFR 72.210 and 10 CFR 72.212, “Conditions of General License Issued Under § 72.210”). The NRC approves the DSSs through the issuance of a CoC to the vendor of the systems, which allows the general licensee to use the systems (see 10 CFR 72.214, “List of Approved Spent Fuel Storage Casks”). A general license provided in 10 CFR 72.210 is effective without the filing of an application with the Commission or the issuance of a licensing document to a particular person.

The safety review conducted for a specific license or CoC is primarily based on the information the applicant provides in a SAR to show that the design and operation meet the appropriate requirements in 10 CFR Part 72. Note that 10 CFR 72.13, “Applicability,” states which regulations apply to a specific licensee, general licensee, and a CoC holder. Each application for approval and issuance of a CoC for a DSS design or a specific license for a DSF must include an accompanying SAR (see 10 CFR 72.230, “Procedures for Spent Fuel Storage Cask Submittals,” and 10 CFR 72.24, “Contents of Application: Technical Information,” respectively).

Before submitting an SAR, the applicant should have evaluated the DSS or DSF in sufficient detail to conclude that it can be properly fabricated, constructed, and safely operated without endangering the health and safety of the public. The SAR is the principal document in which the applicant provides the information on the design and operations and their associated technical bases and demonstrates that the design meets all the applicable requirements in 10 CFR Part 72. The NRC reviewers should understand the facility design and operations and their technical bases, including but not limited to the selection of materials and geometries, mathematical models and equations used, and computer models and calculated results, in order to draw conclusions that the DSS or DSF does in fact meet the regulatory requirements in 10 CFR Part 72.

This SRP is divided into 17 chapters, several of which also include appendices. This SRP discusses regulatory requirements, staff positions, industry codes and standards, acceptance criteria, and other information.

Technical Review Oversight

CoC holders are responsible for demonstrating that the DSS design and fabrication meet the requirements in 10 CFR Part 72, Subpart L, “Approval of Spent Fuel Storage Casks,” (see 10 CFR 72.234(a)). Licensees are responsible for the safety of the DSF design and for DSS and DSF construction, safe operation, and for complying with appropriate regulations. The mission of the NRC as the regulator is to certify, license, and provide inspection oversite on the operation of each DSS and DSF to ensure adequate protection of public health and safety and the environment.

The staff's review should evaluate the proposed DSS or DSF design, contents, operations, and, for a DSF only, the proposed site to ensure that the application provides reasonable assurance that the design and operations meet the regulations in 10 CFR Part 72. In addition to the requirements in 10 CFR Part 72, an application for a DSF must also address other pertinent regulations, such as the standards for protection against radiation in 10 CFR Part 20, "Standards for Protection Against Radiation." Chapter 10A, "Radiation Protection Evaluation for Dry Storage Facilities (SL)," describes the evaluation approach regarding the 10 CFR Part 20 requirements, including the use of dose assessments in the applicant's SAR.

The NRC review team uses its independent expertise to identify and resolve potential design or operational deficiencies, analytical errors, significant uncertainties or non-conservatisms in design approaches, or other issues which might hinder the review team's ability to ensure compliance with the regulations. If otherwise left unchecked by the CoC holder or licensee and the regulator, these issues could potentially lead to the unsafe or noncompliant use or operation of the DSS or DSF.Several considerations may influence the depth of review that is needed for a reasonable assurance determination that the applicable regulations have been met. These include, but are not limited to, the uniqueness of the design (as compared to existing designs), safety margins, operational experience, defense-in-depth, and the relative risks that have been identified for normal operations and potential off-normal conditions (or anticipated occurrences) and accident conditions. Reviewers should also consider the design parameters and methods the applicant describes in the SAR and their possible use, upon approval of the DSS or DSF design (i.e., issuance of a CoC or specific license) in subsequent 10 CFR 72.48(c) changes to the design or procedures by the CoC holder or licensee. Any aspect of the design or procedures that the NRC determines should not be changed by the CoC holder or licensee, without NRC approval beforehand, must be placed in the CoC or license conditions or the technical specifications of the CoC or license.

Review Process

The reviews are performed by members of the NRC review team with expertise in the technical areas described in this SRP. Because of the dependencies in the technical information in different chapters of the SAR, reviewer coordination among the different disciplines is important to ensure a comprehensive, consistent, uniform, and quality review. Each chapter includes a flow chart that diagrams the technical issues that cross disciplines; as such, many reviews rely on input from multiple areas.

When reviewing an amendment to, or a new application for, a DSS or DSF, the NRC review team should consult the SERs of previous amendments, as well as the SERs for similar, approved DSSs and DSFs to understand past NRC determinations regarding analyses affecting or similar to those in the SAR under review. The staff should also consult other relevant sources, such as generic communications, on issues that describe the staff's current position(s) on an issue(s) pertinent to the DSS and DSF review. The staff also relies on published industry standards to support its review. The guidance in this SRP, along with any regulatory guides that endorse industry standards, identifies industry standards that are acceptable to the staff and, where needed, the specific version(s) of the standards the staff finds acceptable. While some of these standards have been withdrawn, they may still be appropriate to use. In some cases, no suitable replacement has been issued for a withdrawn standard.

For amendments, the staff should review the entire amendment to ensure that the applicant has identified all of the proposed changes. Amendments may range from minor changes in the DSS or DSF design, contents, or operations to adding new major component designs or contents. Some amendments are based on the design and methods previously reviewed by the NRC for that same DSS or DSF. Evaluations of amendment changes are often based on the performance of the DSS or DSF as an integrated system. As a result, the staff may examine portions of previously approved components, contents, or methods in the SAR to assess the impact on the proposed amendment.

If the information provided in the SAR does not demonstrate that the new or revised DSS or DSF design meets the regulations, the staff may develop and then forward to the applicant a request for additional information, which contains questions requesting clarification of technical issues. The staff should refer to the updated SAR when reviewing the applicant's response to the request for additional information, for acceptability. The process is repeated as necessary (i.e., additional requests for information and applicant responses), until the SAR shows that the design meets the requirements in 10 CFR Part 72, or until the review is closed by the NRC or the applicant.

For review and issuance of a CoC, once the technical review of a DSS is complete, the NRC prepares a draft SER that summarizes the results of the review. If the NRC intends to authorize use of a new or amended CoC, the NRC staff prepares the *Federal Register* notices for a direct final rule and a companion proposed rule. The rulemaking notices identify the Agencywide Documents Access and Management System (ADAMS) Accession numbers for the draft CoC, technical specifications, and SER. During the rulemaking process, stakeholders and members of the public are allowed to comment on the draft CoC, technical specifications, and preliminary SER. If there are no significant adverse comments, the NRC publishes a notice of confirmation of the effective date of the rulemaking in the *Federal Register*. If the NRC receives a significant adverse comment, then the staff will withdraw the direct final rule and address the public comment in the companion proposed rule process. After addressing the comment, the NRC staff will either modify the proposed CoC, technical specifications, and preliminary SER, if necessary, and publish a final rulemaking in the *Federal Register* or withdraw the rulemaking. The rulemaking, when completed, leads to an update of 10 CFR 72.214 to add the new or amended CoC to the list of approved cask designs.

For review and issuance of a license for a DSF, if no adjudicatory hearing is requested and granted, the technical review of a DSF is complete when the staff issues the license (and associated technical specifications), and an SER documenting the results of the safety review and the staff's findings of compliance. The staff must also issue an environmental assessment (or environmental impact statement) that identifies the environmental impacts of the proposed licensing action. The NRC regulations require that a *Federal Register* notice be published upon issuance of the license and the publishing of the environmental assessment. NUREG-1748, "Environmental Review Guidance for Licensing Actions Associated with NMSS Programs," provides guidance to staff on conducting an environmental review for a DSF.

<u>Safety Evaluation Report and Content</u>

The SER documents the results of the staff's evaluation. The structure typically follows the applicant's SAR or this SRP and contains the following information:

- a general description of the system or facility, operational features, and content specifications

- a summary of the approach the applicant used to demonstrate compliance with the regulations, and a description of the reviews the NRC staff performed to confirm compliance

- a comparison of systems, components, analyses, data, or other information important in the review analysis for comparison with the acceptance criteria, in addition to conclusions regarding the acceptability, suitability, or appropriateness of this information to provide reasonable assurance the acceptance criteria have been met; the staff should clearly state its basis for approval or acceptance of the applicant's design, analyses, results, and conclusions

- a summary of aspects of the review that were selected or emphasized, aspects of the design or contents that the applicant modified, aspects of the design that deviated from the criteria stated in the SRP, and the bases for any deviations from the SRP

- summary statements for evaluation findings at the end of each chapter

<u>Content of SRP</u>

Each chapter of the SRP is organized into the following sections:

- Review Objective
- Applicability
- Areas of Review
- Regulatory Requirements and Acceptance Criteria
- Review Procedures
- Evaluation Findings
- References

<u>Review Objective</u>. This section provides the purpose and scope of the review and establishes the major review objectives for the chapter. The reviewer should obtain reasonable assurance during the review that the objectives are met.

<u>Applicability</u>. This section describes the scope of each chapter in terms of whether a chapter, or a portion of a chapter, is applicable to the review of SARs for both DSSs and DSFs, or only DSSs, or only DSFs.

<u>Areas of Review</u>. This section lists the areas of review. Each area of review encompasses systems, components, analyses, data, or other information. This section provides the organizational structure for the rest of the chapter.

<u>Regulatory Requirements and Acceptance Criteria</u>. This section summarizes the regulatory requirements pertaining to the review and specifies either regulatory or self-imposed acceptance criteria. Generally, the requirements for a given SAR chapter will be in 10 CFR Part 72, but the chapter can also list other significant regulatory requirements, such as those in 10 CFR Part 20. The reviewer should refer to the regulations to ensure the SAR addresses all relevant requirements. Sections of 10 CFR Part 72 that are applicable to the review of an application for a new or an amendment to a DSF specific license or a DSS CoC are listed in 10 CFR 72.13(b) and (d), respectively. The reviewer should read the complete language of the current version of 10 CFR Part 72 to determine the proper set of regulations for the section being reviewed for the application (CoC or specific license).

The acceptance criteria portion of this section addresses the design criteria and, in some cases, addresses specific analytical methods that NRC staff reviewers have found to be acceptable for meeting the regulatory requirements that apply to the given SAR chapter. Most chapters organize the acceptance criteria in accordance with the review areas established in the "Areas of Review" section of the specific chapter and identify the type and level of information that should be in the SAR.

This section typically sets forth the solutions and approaches that staff reviewers have previously determined to be acceptable for demonstrating compliance with the regulations and addressing specific safety concerns or design areas that are important to safety. These solutions and approaches are discussed in the SRP so that the reviewers can implement consistent and well-understood positions as similar safety issues arise in future cases. These solutions and approaches are acceptable to the staff, but they are not the only possible method for meeting the regulations.

Substantial staff time and effort has gone into developing these acceptance criteria. Consequently, a corresponding amount of time and effort may be required to review and accept new or different solutions and approaches. Thus, applicants proposing new solutions and approaches to safety issues or analytical techniques other than those described in the SRP may experience longer review times. An alternative for the applicant is to propose new methods on a generic basis, independent from a CoC or license application, possibly as a topical report.<u>Review Procedures</u>. This section presents a general approach that reviewers should typically follow to establish reasonable assurance that the applicable regulations have been met. As an aid to the reviewer, this section may also provide information on what has been found acceptable in past reviews. This section identifies standards that have been found acceptable in particular reviews, or that are desirable but not specifically identified in existing regulatory documents. Since many of the reviews are interdisciplinary, the reviewers should coordinate with each other, as necessary, to identify issues in other SAR chapters. The section includes a flow chart to depict the coordination across disciplines that may be necessary to conduct reviews. In addition, the reviewer may identify conditions of the approval. In these cases, the reviewer should include a discussion of each condition and the reasons for the addition of the condition in the relevant sections of the SER.

<u>Evaluation Findings</u>. This section provides example evaluation findings and summary statements to be incorporated into the SER. The reviewer prepares the evaluation findings based on how satisfactorily the application meets the regulatory requirements. The NRC publishes the findings in the SER.

<u>References</u>. This section lists the NRC documents, codes, specifications, standards, regulations, and other technical documents referenced in the chapter.

1 GENERAL INFORMATION EVALUATION

1.1 Review Objective

The objective of this U.S Nuclear Regulatory Commission (NRC) general information evaluation is to verify that the applicant's safety analysis report (SAR) includes a description (proprietary information may be provided as described in this chapter) of major components and operations adequate to familiarize reviewers with the pertinent features of the dry storage system (DSS) or dry storage facility (DSF) and to ensure that the applicant for a specific license has the relevant technical qualifications and quality assurance program. In addition, if the applicant submits an amendment application during or after a renewal of the license or certificate of compliance (CoC), the evaluation should ensure that all relevant aspects of aging management have been considered.

1.2 Applicability

This standard review plan (SRP) chapter applies to the review of applications for specific licenses for an independent spent fuel storage installation (ISFSI) or a monitored retrievable storage installation (MRS), categorized as DSFs. This chapter also applies to the review of applications for a DSS CoC for use by a general licensee. Sections of this chapter that apply only to specific license applications are identified with "**(SL)**" in the heading. Sections that apply only to DSS CoC applications have "**(CoC)**" in the heading. A subsection without an identifier applies to both types of applications. Applicants for a CoC will describe how their storage system was designed to ensure that a general licensee who chooses this system will be able to meet the applicable regulatory requirements.

1.3 Regulatory Requirements and Acceptance Criteria

This section summarizes those parts of Title 10 of the *Code of Federal Regulations* (10 CFR) Part 72, "Licensing Requirements for the Independent Storage of Spent Nuclear Fuel, High-Level Radioactive Waste, and Reactor-Related Greater than Class C Waste," that are relevant to the review areas addressed by this chapter. The NRC staff reviewer should refer to the exact language in the regulations. Table 1-1a matches the relevant regulatory requirements to the areas of review for an SL review. Table 1-1b matches the relevant regulatory requirements to the areas of review for a CoC review.

Table 1-1a Relationship of Regulations and Areas of Review for a DSF (SL)

Areas of Review	10 CFR Part 72 Regulations				
	72.24 (a)(b)(c)(f)(j)(l)(n)	72.28(a)	72.42	72.56	72.120 (b)(c)
Site Description	(a)		●	●	
DSF Description and Operational Features	(b)(c)		●	●	●
Engineering Drawings	(c)		●	●	●
Contents	(b)		●	●	●
Amendment Applications Submitted during the Renewal Review or after the Renewal Is Issued			●	●	
Qualifications of the Applicant	(j)	●		●	
Quality Assurance Program Description	(n)			●	

Table 1-1b Relationship of Regulations and Areas of Review for a DSS (CoC)

Areas of Review	10 CFR Part 72 Regulations		
	72.230 (a)	72.236 (a)(c)(g)(h)(m)	72.240
DSS Description and Operational Features	●	(g)(h)	●
Engineering Drawings	●	(g)	●
Contents		●	●
Amendment Applications Submitted during the Renewal Review or after the Renewal Is Issued		●	●
Conditions for DSS Transportability		●	

The bulk of this chapter focuses on the general description of the DSS or DSF designs, the DSF site, and the consistency of the general description with the contents of the remaining chapters of the SAR. All reviewers should evaluate the general description, regardless of their specific review assignments, to obtain an overall understanding of the DSS or DSF and DSF site; its structures, systems, and components (SSCs); and the protections afforded for public health and safety. The other chapters of this SRP present this information in more detail.

The general description should contain sufficient information to enable all reviewers to obtain an understanding of the principal functions and design features of the proposed DSS or DSF. The NRC staff should review the SAR for adequacy of descriptions and drawings of the DSS or DSF and its respective support systems. In addition, the staff should review the SAR for a DSF for the adequacy of the site description.

The following paragraphs briefly describe the acceptance criteria for the material provided in the general information evaluation.

1.3.1 Site Description (SL)

The SAR should contain a general description (including engineering drawings, sketches, and illustrations) of the site on which the proposed facility would be located, as well as a proposed schedule for construction and operations. This description should identify the geographical location and discuss the suitability and demography of the site in broad terms. It should contain sufficient information to enable all reviewers, regardless of their specific review assignments, to gain a general understanding of the proposed site.

1.3.2 DSS or DSF Description and Operational Features

The application should contain a broad overview and a general description (including engineering drawings, sketches, and illustrations) of the DSS or DSF. This information should clearly identify the functions of all principal components and principal auxiliary equipment and provide a list of those components classified as "important to safety." Important aspects from all of the disciplinary areas should be summarized. If there are several versions of the DSS or DSF storage containers, the application should delineate the differences between the versions. The application should briefly describe typical operational sequences and procedures for loading and unloading the radioactive materials to be stored.

The application should include an index of any documents submitted to the NRC in other applications that are incorporated in whole or in part in the SAR, as well as provide a summary of such documents in the appropriate section of the SAR. The applicant should provide clear and specific references to the information incorporated by reference to ensure all relevant and intended information is clearly identified and irrelevant and unintended information is not incorporated from the referenced documents.

1.3.3 Engineering Drawings

Engineering drawings should be included in the first chapter of the SAR. The drawings should contain sufficient detail to allow the reviewer to understand the general arrangement and dimensions of the overall DSS or DSF design and various components and features so that the reviewer can verify the models used in the various safety calculations for the design. The drawings will also support the staff's understanding of the key operation features of the DSS or DSF and any special equipment used for loading, unloading, transfer, or storage of or at the DSS or DSF. Also, the drawings should provide sufficient detail to allow the reviewer the option of developing a model for confirmatory calculations. The drawings should include allowable tolerances to support safety analyses.

1.3.4 Contents

The SAR should provide specifications for the contents expected to be stored in the DSS or DSF. For spent nuclear fuel (SNF) contents, these specifications may include, but are not limited to, type of SNF (i.e., boiling-water reactor (BWR), pressurized-water reactor (PWR), or both); number of SNF assemblies the DSS or DSF storage container can accommodate; maximum and minimum allowable enrichment of the fuel before irradiation; maximum burnup; minimum acceptable cooling time of the SNF before storage in the DSS or DSF (e.g., aged at least 1 year); maximum heat designed to be dissipated; maximum mass of SNF authorized for loading; condition of the SNF (e.g., intact, undamaged, damaged); weight and nature of nonfuel hardware; and inert atmosphere requirements. For specific license applications requesting approval to store reactor-related greater-than-Class-C (GTCC) waste and high-level radioactive waste (HLW) (MRS only), these specifications should also include, but are not limited to, radionuclides and their maximum quantities, maximum mass of the GTCC waste, physical properties, and chemical compositions.

1.3.5 Amendment Applications Submitted during the Renewal Review or after the Renewal Is Issued

By regulation (10 CFR 72.42, “Duration of License; Renewal,” and 10 CFR 72.240, “Conditions for Spent Fuel Storage Cask Renewal”), applicants must demonstrate that SSCs important to safety will continue to perform their intended function(s) for the requested period of extended operation as a part of the renewal request. For *concurrent amendment and renewal applications*, the amendment application should include a scoping evaluation and an aging management review for that amendment to document the evaluation of the amendment’s SSCs (and associated subcomponents) for extended operation, or the renewal application should be supplemented to address the proposed amendment to document the evaluation of the amendment’s SSCs (and associated subcomponents) for extended operation. Any amendment application submitted after the renewal has been issued (post-renewal amendment applications) should include a scoping evaluation and an aging management review for that amendment.

For post-renewal amendment applications or concurrent amendment applications that include a scoping evaluation and an aging management review, the amendment application should either: (1) show that the in-scope SSCs (and associated subcomponents) described in the amendment are already encompassed in the time-limited aging analyses (TLAAs) or aging management programs included in the specific-license or CoC renewal application, or (2) include revised or new TLAAs or aging management programs to address aging effects of any new in-scope SSCs (and associated subcomponents) proposed in the amendment application.

The project manager (PM) and technical reviewers should verify that Chapter 8 of the application, “Materials Evaluation,” includes details on the amendment with regard to scoping evaluation, aging management review, and appropriate SAR changes to incorporate the results of this review (see also Section 1.4.4, Application Content,” of NUREG-1927, Revision 1, “Standard Review Plan for Renewal of Specific Licenses and Certificates of Compliance for Dry Storage of Spent Nuclear Fuel,” issued June 2016.

For concurrent amendment and renewal applications, if there are different PMs assigned to the renewal review and the amendment review, the PMs and technical reviewers should coordinate across the reviews to ensure that renewal aspects are covered for the amendment. Note that, before proceeding with the review of an amendment submitted *during* the renewal review, the PMs should consider how each review may affect the other, and decide, in conjunction with branch and division management, whether to proceed with both reviews or to delay one review until the other is complete. For additional guidance, refer to Regulatory Issue Summary 2004-20, “Lessons Learned from Review of 10 CFR Parts 71 and 72 Applications,” dated December 16, 2004 (NRC 2004).

The NRC staff may include a condition in the renewed license or CoC noting all future amendments would need to address aging management.

1.3.6 Qualifications of the Applicant (SL)

As required in 10 CFR 72.24(j) and 10 CFR 72.28(a), the SAR must include the technical qualifications of the applicant to engage in the proposed activities, including any contractors that the applicant may employ (e.g., for design, construction, fabrication, aspects of facility operations). Qualifications should include training and experience.

1.3.7 Quality Assurance (SL)

The application should briefly describe the proposed quality assurance (QA) program and cite the applicable implementing procedures. Details of the QA program should be discussed in Chapter 15, "Quality Assurance Evaluation," of this SRP. This description should discuss how the QA program satisfies all requirements of Subpart G, "Quality Assurance," to 10 CFR Part 72. Chapter 15 of this SRP addresses NRC's detailed review of the QA program.

1.3.8 Consideration of Dry Storage System Transportability (CoC)

The application should include information on how the DSS design considered compatibility with removal of the stored SNF from a reactor site, transportation, and ultimate disposition by the Department of Energy per the requirement in 10 CFR 72.236(m).

1.4 Areas of Review

This chapter addresses the following areas of review:

- site description **(SL)**
- DSS or DSF description and operational features
- engineering drawings
- contents to be stored in the DSF or DSS
- amendment applications submitted during the renewal review or after the renewal is issued
- qualifications of the applicant **(SL)**
- quality assurance program description **(SL)**
- consideration of DSS transportability **(CoC)**

1.5 Review Procedures

Figure 1-1 shows the interrelationship between the general information evaluation and the other chapters described in this SRP.

Chapter 1 – General Information Evaluation

- Site Description (SL)
- DSS or DSF Description and Operational Features
- Engineering Drawings
- Contents
- Amendment Applications After and During Renewal
- Qualifications of the Applicant (SL)
- Quality Assurance Program (SL)
- Consideration of DSS Transportability (CoC)

Chapter 2 – Site Characteristics Evaluation (SL)

- Geography and Demography
- Surface and Subsurface Hydrology
- Nearby Facilities
- Geology and Seismology
- Meteorology

Chapter 3 – Principal Design Criteria Evaluation

- Classification of SSCs
- Design Criteria for Safety Protection Systems
- Design Bases for SSCs Important to Safety
- Design Criteria for Other SSCs (SL)

Chapter 4 – Structural Evaluation

- Description of the SSCs
- Normal and Off-normal Conditions
- Design Criteria
- Accident Conditions
- Loads

Chapter 5 – Thermal Evaluation

- Decay Heat Removal System
- Analytical Methods, Models, and Calculations
- Material and Design Limits
- Thermal Loads and Environmental Conditions
- Surveillance Requirements

Chapter 6 – Shielding Evaluation

- Shielding Design Description
- Shielding Analyses
- Radiation Source Definition
- Reactor-Related GTCC Waste Storage (SL)
- Shielding Model Specification

Chapter 7 – Criticality Evaluation

- Criticality Design Criteria/Features
- Criticality Analysis
- Fuel Specification
- Burnup Credit
- Model Specification
- Reactor-Related GTCC Waste and HLW (SL)

Chapter 8 – Materials Evaluation

- General Review Considerations
- Fuel Cladding Integrity and Retrievability
- Material Properties
- Environmental Degradation; Chemical and Other Reactions
- Code Use and Quality Standards

Chapter 9 – Confinement Evaluation

- Confinement Design Characteristics
- Nuclides with Potential for Release
- Confinement Analyses
- Confinement Monitoring Capability
- Supplemental Information

Chapter 10A (SL)/10B (CoC) – Radiation Protection Evaluation

- ALARA
- Design Features
- Radiation Exposures
- Dose Assessment
- Health Physics Program (SL)

Chapter 11 – Operation Procedures and Systems Evaluation

- Operation Description
- Storage Container Handling and Storage Operations
- Other Operating Systems (SL)
- Analytical Sampling (SL)
- Storage Container Loading
- Repair and Maintenance (SL)
- Operation Support Systems (SL)
- Fire and Explosion Protection (SL)
- Storage Container Unloading
- Control Room and Control Area (SL)

Chapter 12 – Conduct of Operations Evaluation

- Organizational Structure (SL)
- Normal Operations (SL)
- Acceptance Tests
- Personnel Selection (SL)
- Maintenance Program
- Emergency Planning (SL)
- Preoperational Testing and Startup (SL)
- Physical Security/Safeguards (SL)

Chapter 13 – Waste Management Evaluation (SL)

- Waste Sources and Facilities
- Solid Wastes
- Off-Gas Treatment and Ventilation
- Waste Stream Radiological Characteristics and Dose Analyses
- Liquid Waste Treatment/Retention

Chapter 14 – Decommissioning Evaluation (SL)

- Proposed Decommissioning Plan
- Operational Features
- Design Features
- Decommissioning Funding Plan

Chapter 15 – Quality Assurance Evaluation

- Organization and Program
- Document Control
- Design and Nonconformance
- Procurement and Test Control
- Procedures and Drawings
- Inspections and Audits

Chapter 16 – Accident Analysis Evaluation

- Cause of Event
- Detection of Event
- Definition of Operating Environment and Physical Parameters
- Summary of Event Consequences and Regulatory Compliance
- Corrective Course of Action

Chapter 17 – Technical Specifications Evaluation

- Functional and Operating Limits, Monitoring Instruments, and Limiting Control Settings
- Design Features
- Surveillance Requirements
- Limiting Conditions
- Administrative Controls

Figure 1-1 Overview of General Description Evaluation

The following sections delineate review procedures applicable to the general description evaluation. Because the review of the general description of the DSS or DSF is interdisciplinary, coordinate with other reviewers (e.g., structural, thermal, shielding, criticality, materials) as necessary.

1.5.1 Site Description (SL)

Verify that the SAR presents the location of the ISFSI or the MRS and schedules for construction. Verify that the SAR provides an overview of the geographical location and discusses the site's suitability and the demography of the area around the site. Verify that this overview is consistent with the detailed assessment provided in the "Site Characteristics Evaluation" chapter of the SAR.

1.5.2 DSS or DSF Description and Operational Features

Verify that the application provides a broad overview of the DSS or DSF design that the reviewers and other stakeholders can use to become familiar with the features of the proposed DSS or DSF. Confirm that the description does the following:

- presents the principal characteristics of the DSS or DSF SSCs and features including their dimensions, weights, and construction materials and, for a DSF, physical locations relative to each other and site boundaries (e.g., controlled area boundary, restricted area boundary)

- clearly identifies all SSCs and features considered important to safety and those SSCs that are not important to safety but are relied upon by SSCs that are important to safety or that (in the event of failure) could impact the performance of SSCs important to safety

- identifies and describes features such as the confinement vessel, vessel internals (e.g., fuel basket, GTCC liner), valves, lids, seals, penetrations, trunnions or other items used for lifting, closure mechanisms, shielding design features, criticality control features, and impact limiters

- discusses special design features of the DSS or DSF such as a heat-removal system, neutron poisons, or monitoring instrumentation

Ensure that the application includes a clear definition of the primary confinement system of the DSS or DSF's storage containers.

Compare the sketches and diagrams provided throughout the SAR with the detailed drawings presented in the SAR chapter on general information. If the application includes proprietary drawings and descriptions that will remain proprietary upon approval of the license or CoC, the sketches, drawings, and diagrams that provide the general description and operational features need not show the proprietary features. This may be achieved by depicting less detail or by illustrating generic components that fulfill the design functions. However, these representations should show the operational concept and features important to safety in sufficient detail to form an acceptable basis for public review and comment.

In addition to information on an individual DSS or DSF storage container, ensure that the application describes any limitations on the arrangement of DSS or DSF storage container arrays. For a particular DSS or DSF storage container, these limitations may include the minimum spacing between adjacent DSSs and DSF storage containers or the maximum density of DSSs or

DSF storage containers in an array. For specific licenses, the limitations include the total number of storage containers for each content type (SNF, reactor-related GTCC waste, HLW) or the total amount of SNF, GTCC waste, or (for MRS) HLW that may be stored at the DSF. The reviewer should describe the acceptable limitations in the safety evaluation report (SER) and ensure that the CoC or license conditions or technical specifications include the necessary limitations (see Chapter 17, “Technical Specifications Evaluation,” of this SRP). Ensure that the SAR has provided information on the configuration(s) of the DSS or DSF storage containers. Configuration information includes items such as above-ground or below or ingrade storage, use of a metal confinement vessel in a concrete overpack or vault, bolted or welded closures of the confinement vessel, canister-based or noncanister-based storage system or container, as appropriate, and orientation of the stored contents (e.g., horizontal or vertical). Ensure that the application describes the operational sequences for loading and unloading the DSS or DSF storage containers.

Damaged fuel may require canning for storage and transportation. The purpose of canning is to confine gross fuel particles to a known, subcritical volume during off-normal and accident conditions, and to facilitate handling and ready retrieval of contents. Canning of damaged fuel also provides geometry control of the SNF to avoid relocation, concentration, or both, of radiation sources that may create problems for radiation shielding. Therefore, verify that the application includes a description of how damaged fuel would be canned, the characteristics of the can, and the means in which the can would be placed in the storage container and either readily retrieved during normal operations or off-normal conditions or recovered after an accident condition (see Chapter 16, “Accident Analysis Evaluation,” Section 16.4.5, “Recovery and Retrievability,” of this SRP for further discussion).

(SL) Verify that the SAR provides a brief description of the operating systems, including fuel, reactor-related GTCC waste, HLW handling (MRS), or all three; decay heat removal; site-generated waste treatment; and auxiliary systems. Determine whether the application provides sufficient detail to allow for an understanding of the systems involved.

(SL) Verify that the application presents the principal function and design features of the installation. Ensure that the SAR describes the DSF facilities (e.g., administrative building, health physics facilities) needed to support DSF operations. Ensure that the description includes a layout of the DSF with all features clearly identified and appropriate distances between facilities and features marked.

(SL) Note that a specific license application may involve use of a DSS certified under Subpart L, “Approval of Spent Fuel Storage Casks,” to 10 CFR Part 72 and include the final SAR (FSAR) for the certified DSS by reference. In this case, verify that the SAR for the DSF provides additional information relating to the DSS, including the applicant’s evaluations that establish that site and design parameter limits and facility operations for the DSF are within the bounds of those established as limiting conditions as set forth in the referenced CoC and FSAR. The applicant does not need to re-perform the evaluations that were done for the certified DSS that are being incorporated by reference into the SAR for the DSF. Ensure that references are clear and specific (i.e., point to specific relevant pages or sections of a specific revision of the DSS FSAR and CoC, including the specific amendment number, that describe the information or analyses the applicant is including by reference).

1.5.3 Engineering Drawings

Engineering drawings are usually presented in the chapter of the SAR covering general information. Reviewers should be familiar with NUREG/CR-5502, "Engineering Drawings for 10 CFR Part 71 Package Approvals," issued May 1998. Although NUREG/CR-5502 was written for transportation packages, the criteria in NUREG/CR-5502 for drawings are also applicable to applications for DSSs or DSF storage containers.

Verify that all SSCs important to safety are sufficiently detailed to enable reviewers to evaluate their effectiveness. In addition, review information about items not important to safety to ensure they do not impede the systems that are important to safety.

Each reviewer should evaluate the level of detail furnished with the application. Ensure that the drawings specify those details of the DSS or DSF design that affect its evaluation.

Devote particular attention to ensuring that dimensions, materials, and other details on the drawings are consistent with those described in the text of the SAR as well as those used in safety analyses. Confirm that the dimensions shown on the general arrangement drawing(s) specify the overall size of the DSS or DSF storage containers, the location and configuration of the contents within the DSS or DSF storage containers, and the configurations and locations of the containers on the site for DSFs. Verify that all dimensions indicated on drawings include tolerances that are consistent with the DSS or DSF evaluation and that the tolerances are consistent with the assumptions used in the safety analyses.

1.5.4 Contents

Confirm that the application presents a general description of the contents proposed for storage in the DSS or DSF. Because a very detailed description of the proposed DSS or DSF contents is typically provided in the chapter of the SAR on principal design criteria, the general information discussion in the SAR is important only to the extent that it permits overall familiarization with the DSS or DSF.

1.5.4.1 *Spent Nuclear Fuel*

Verify that the application contains the key parameters for SNF, including the type of fuel (i.e., PWR, BWR, or both), number of fuel assemblies, parameters that adequately characterize the radiation source terms associated with these fuel assemblies, any nonfuel hardware stored with the assemblies (e.g., maximum burnup, minimum enrichment, minimum cooling time, hardware material specifications), preferential loading, and condition of the fuel assemblies (i.e., intact, damaged, or consolidated). The general information may also include additional characteristics such as maximum burnup, initial enrichment, heat load, and cooling time as well as the assembly vendor and configuration (e.g., Westinghouse 17 x 17). These characteristics may also be repeated in the principal design criteria. In addition, verify that the application identifies the cover gas, as applicable.

If the applicant proposes the storage of damaged fuel, confirm that the SAR defines the range of permissible conditions for the stored material. The regulation in 10 CFR 72.122(h)(1) allows for "canning" or use of other acceptable means for storing fuel with cladding that is not or may not remain intact and for unconsolidated assemblies (without intact cladding). Consistent with 10 CFR 72.236(c), the damaged fuel must be maintained in a subcritical condition, while 10 CFR 72.236(h) requires the damaged fuel to be compatible with wet or dry loading and unloading facilities. If damaged fuel is to be stored, ensure that the application addresses how the following basic requirements will be met:

- Maintain subcriticality.
- Prevent unacceptable release of contained radioactive material.
- Avoid excessive radiation dose rates and doses.
- Ensure the application describes how the design will protect the fuel for a specific licensee, or for a CoC application, facilitate a general licensee's ability to protect the cladding against gross rupture or the fuel must be otherwise confined such that degradation of the fuel during storage will not pose operational safety problems with respect to its removal from storage and (10 CFR 72.122(h)(1)).
- Maintain ready retrieval of the contents.

If the application requests approval to store nonfuel hardware together with the SNF assemblies (e.g., control spiders, burnable poison rod assemblies, control rod elements, thimble plugs, fission chambers, and primary and secondary neutron sources, or BWR channels that are an integral part of the fuel assembly that do not require special handling), confirm that the application presents summary descriptions of those components in the SAR's chapter on general information. The NRC has made a practice of carefully characterizing components as being "associated with or integral to" the fuel assembly, which the NRC refers to as nonfuel hardware. Chapter 3, "Principal Design Criteria," of this SRP covers the evaluation of nonfuel hardware in more detail.

1.5.4.2 *Greater-Than-Class-C Waste and High Level Waste* ***(SL)***

Verify that the application lists the key parameters for reactor-related GTCC waste, such as the physical and chemical form of the GTCC waste (e.g., activated metal, process waste), the maximum quantity of GTCC waste to be stored at the DSF, and the radionuclide inventory. Confirm that the application lists the key parameters for HLW proposed for storage at an MRS, including the physical and chemical properties of the HLW as well as the radionuclides present and the quantities of these radionuclides present in the HLW.

Ensure the application reflects 10 CFR 72.2, "Scope," and 10 CFR 71.120(b) and (c), which require reactor-related GTCC wastes, if stored at an ISFSI or an MRS, to be in solid form and stored in a separate container (i.e., GTCC wastes may not be stored together with SNF in the same storage container). Verify that the application reflects that liquid reactor-related GTCC wastes may not be stored at an ISFSI or an MRS.

1.5.5 Amendment Applications Submitted during the Renewal Review or after the Renewal Is Issued

Ensure that, for concurrent amendment and renewal applications, the amendment application includes a scoping evaluation and an aging management review for that amendment to document the evaluation of the amendment's SSCs (and associated subcomponents) for extended operation, or that the renewal application is supplemented to address the proposed amendment to document the evaluation of the amendment's SSCs (and associated subcomponents) for extended operation. Verify that any amendment application submitted after the license or CoC has been renewed includes a scoping evaluation and an aging management review for that amendment.

For post-renewal amendment applications or concurrent amendment applications that include a scoping evaluation and an aging management review, verify that the amendment application either: (1) shows that the in-scope SSCs (and associated subcomponents) described in the amendment are already encompassed in the TLAAs, aging management programs included in the specific-license, or CoC renewal application, or (2) includes revised or new TLAAs or aging management programs to address aging effects of any new in-scope SSCs (and associated subcomponents) proposed in the amendment application.

The PM and technical reviewers should verify that Chapter 8 of the application, “Materials Evaluation,” includes details on the amendment with regard to scoping evaluation, aging management review, and appropriate SAR changes to incorporate the results of this review (see also Section 1.4.4 of NUREG-1927, Revision 1).

For concurrent amendment and renewal applications, if there are different PMs assigned to the renewal review and the amendment review, the PMs and technical reviewers should coordinate across the reviews to ensure that renewal aspects are covered for the amendment. Note that, before proceeding with the review of an amendment submitted during the renewal review, the PMs should consider how each review may affect the other, and decide, in conjunction with branch and division management, whether to proceed with both reviews or to delay one review until the other is complete. For additional guidance, refer to Regulatory Issue Summary 2004-20.

The NRC staff may include a condition in the renewed license or CoC noting all future amendments would need to address aging management.

1.5.6 Qualifications of the Applicant (SL)

Confirm that the SAR clearly designates the applicant and the prime agents, consultants, and contractors, if known, for design, fabrication, and testing of the proposed DSF SSCs and features. In addition, verify that the SAR clearly defines the division and assignment of responsibilities among those parties. Although specific subcontractors may not be known at the time the SAR is submitted, the SAR should clearly identify any activities the applicant will not perform. In addition, verify that the SAR describes the technical qualifications, previous experience, and suitability of all organizations participating in the proposed activities.

1.5.7 Quality Assurance (SL)

Confirm that the application describes the proposed QA program, citing all implementing procedures in a manner that satisfies the 18 criteria defined in Subpart G to 10 CFR Part 72. The description only needs to refer to procedures that implement the QA program, and these procedures do not need to be explicitly included in the application. Verify that the QA program addresses design, fabrication, construction, testing, operation, and modification activities for the SSCs that are important to safety. Verify that the application also discusses the activities to be performed under the QA program and how these activities will be controlled to ensure compliance with all of the requirements of Subpart G. These controls may be applied to the various activities using a graded approach as presented in NUREG/CR-6407, “Classification of Transportation Packaging and Dry Spent Fuel Storage System Components According to Importance to Safety,” issued February 1996 (i.e., QA efforts expended for a given activity should be consistent with that activity’s system classification and function).

In accordance with 10 CFR 72.140(d), a QA program previously approved by the NRC and established, maintained, and executed for another DSF will be accepted as satisfying the requirements for a QA program for the purpose of this application. Additionally, previously approved QA programs that meet the requirements of Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," to 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities," or Subpart H, "Quality Assurance," to 10 CFR Part 71, "Packaging and Transportation of Radioactive Material," will be acceptable provided they also meet the recordkeeping requirements in 10 CFR 72.174, "Quality Assurance Records." Ensure that any reference to a previously approved QA program identifies the program by date of submittal to the NRC, docket number, and date of NRC approval. Coordinate with the review under SRP Chapter 15, "Quality Assurance Evaluation."

1.5.8 Consideration of Dry Storage System Transportability (CoC)

Coordinate the review with all of the technical disciplines to ensure the SAR demonstrates that the DSS design includes, to the extent practicable, consideration of transportation of the SNF from the licensees' sites per 10 CFR 72.236(m). For most DSS designs, this evaluation is fairly simple and straightforward and does not require significant effort on the part of the reviewers; the adequacy of the consideration of transportation should be fairly obvious from the design. For DSS designs with uncommon or unusual features or construction, that lack common important features, or that exhibit unusual responses to off-normal or accident conditions, use more care when evaluating the design's adequacy in this regard. Consider whether or not the applicant may need to provide further justification or analyses to demonstrate appropriate consideration of transportation to meet 10 CFR 72.236(m).

1.6 <u>Evaluation Findings</u>

The reviewer should prepare evaluation findings upon satisfaction of the regulatory requirements in Chapter 1. If the documentation submitted with the application fully supports positive findings for each of the regulatory requirements, the statements of findings should be similar to the following:

F1.1 **(SL)** The site description and a discussion of the suitability of the site for a DSF, as presented in SAR Section(s) _____, are sufficient to familiarize a reviewer or stakeholder with the site and its suitability.

F1.2 The general description and discussion of the [DSS or DSF] presented in SAR Section(s) _____, with special attention to the design and operating characteristics, unusual or novel design features, and principal considerations important to safety, are sufficient to familiarize a reviewer or stakeholder with the design.

F1.3 Drawings for the SSCs important to safety are presented in SAR Section _____. A listing of those drawings (including dates and revision numbers) that were relied upon as a basis for approval appears in SER Section _____.

F1.4 The specifications for the [SNF/HLW/reactor-related GTCC waste] to be stored [in the DSS/at the DSF] provided in SAR Section _____ are sufficient to familiarize a reviewer or stakeholder with the contents to be stored. Additional details concerning these specifications are presented in SAR Section _____ and SER Section _____.

F1.5 **(SL)** The technical qualifications of the applicant to engage in the proposed activities are identified and described in SAR Section _____ and determine that the applicant has the technical qualifications to design, build, and operate a DSF.

F1.6 **(SL)** The QA program and implementing procedures are sufficiently described in SAR Section _____.

The reviewer should provide a summary statement similar to the following:

> The staff concludes that the general information presented in the SAR satisfies the requirements for the general description under 10 CFR Part 72. This finding is reached on the basis of a review that considered the regulation, itself, applicable regulatory guides, and accepted practices.

1.7 References

10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities."

10 CFR Part 71, "Packaging and Transportation of Radioactive Material."

10 CFR Part 72, "Licensing Requirements for the Independent Storage of Spent Nuclear Fuel, High-Level Radioactive Waste, and Reactor-Related Greater than Class C Waste."

NUREG-1927, U.S. Nuclear Regulatory Commission, "Standard Review Plan for Renewal of Specific Licenses and Certificates of Compliance for Dry Storage of Spent Nuclear Fuel." Revision 1, June 2016, (ADAMS Accession No. ML16179A148).

NUREG/CR-5502, U.S. Nuclear Regulatory Commission, "Engineering Drawings for 10 CFR Part 71 Package Approval," UCRL-ID-130438, Lawrence Livermore National Laboratory, May 1998.

NUREG/CR-6407, U.S. Nuclear Regulatory Commission, "Classification of Transportation Packaging and Dry Spent Fuel Storage System Components According to Importance to Safety," INEL-95/0551, Idaho National Engineering Laboratory, February 1996.

Regulatory Issue Summary 2004-20, "Lessons Learned from Review of 10 CFR Parts 71 and 72 Applications," December 16, 2004, https://www.nrc.gov/reading-rm/doc-collections/gen-comm/reg-issues/2004/ri200420.pdf.

2 SITE CHARACTERISTICS EVALUATION FOR DRY STORAGE FACILITIES (SL)

2.1 Review Objective

The objective of the U.S. Nuclear Regulatory Commission's (NRC's) review of the site characteristics for dry storage facilities (DSFs) is to provide reasonable assurance that the applicant's safety analysis report (SAR) (1) properly identifies the external natural and human-induced phenomena for inclusion in the design basis and that the design basis levels are adequate, (2) adequately characterizes local land and water use and population so that the reviewer can identify important individuals and populations likely to be affected, and (3) adequately characterizes the transport processes that could move any released contamination from the facility to the maximally exposed real individuals and populations, in compliance with Title 10 of the *Code of Federal Regulations* (10 CFR) Part 72, "Licensing Requirements for the Independent Storage of Spent Nuclear Fuel, High-Level Radioactive Waste, and Reactor-Related Greater than Class C Waste." The results of this review will determine the acceptability of site-derived design bases.

2.2 Applicability

This chapter applies to the review of applications for specific licenses for an independent spent fuel storage installation (ISFSI) or a monitored retrievable storage (MRS) facility, categorized as DSFs. The title of this chapter is denoted as "**(SL)**" to make it readily apparent that this chapter applies only to the review of an application for a specific license.

2.3 Areas of Review

This chapter addresses the following areas of review:

- geography and demography
- nearby industrial, transportation, and military facilities
- meteorology
- surface hydrology
- subsurface hydrology
- geology and seismology

2.4 Regulatory Requirements and Acceptance Criteria

This section summarizes those parts of 10 CFR Part 72 that are relevant to the review areas addressed by this chapter. The NRC staff reviewer should refer to the exact language in the regulations. Table 2-1 matches the relevant regulatory requirements to the areas of review covered in this chapter. The reviewer should refer to the language in the regulations and verify the association of the regulatory requirements with the areas of review presented in the table to ensure that no requirements are overlooked as a result of unique applicant design features.

Table 2-1 Relationship of Regulations and Areas of Review for a DSF

Areas of Review	10 CFR Part 72 Regulations				
	72.24 (a)(c)(e)	72.40 (a)(1)(2)(3)	72.90	72.92	72.94
Geography and Demography	●	●	●		
Nearby Industrial, Transportation, and Military Facilities	●	●			●
Meteorology	●	●	●	●	
Surface Hydrology	●	●	●	●	
Subsurface Hydrology	●	●			
Geology and Seismology	●	●	●	●	

Areas of Review	10 CFR Part 72 Regulations (cont.)							
	72.96	72.98	72.100	72.102	72.103	72.104 (a)	72.106	72.122 (b)(c)(e)
Geography and Demography	●	●	●			●	●	●
Nearby Industrial, Transportation, and Military Facilities		●	●					(e)
Meteorology		●						●
Surface Hydrology		●						●
Subsurface Hydrology		●						●
Geology and Seismology		●		●	●			●

2.4.1 Geography and Demography

2.4.1.1 *Site Location*

The SAR should provide information on the site location of the proposed ISFSI or MRS and nearby facilities, including the site's host State and county and the site's latitude and longitude. Maps and aerial photographs of the site should be presented with radial coverage extending a minimum of 16 kilometers (km) (10 miles (mi)) from the site. A detailed map of the site area should show adjacent buildings, roads, railroads, transmission lines, wetlands, and surface water bodies. The reviewer should be aware of the limitations on ISFSI and MRS siting that are listed in 10 CFR 72.96, "Siting Limitations," and the potential changes to these limitations that may have been enacted by Congress.

2.4.1.2 *Site Description*

The SAR should include a site map that shows the site boundary and the controlled area boundary, controlled area access points, and the distances from the boundary to significant features of the installation. The SAR should discuss the applicant's legal responsibilities for the properties described, such as ownership, lease, or easements. Topographic maps should depict the site topography and surface drainage patterns, as well as roads, railroads, transmission lines, wetlands, and surface water bodies on the site. The SAR should describe vegetative cover and surface soil characteristics to facilitate evaluation of fire hazards and erosion. Other activities the applicant conducts within the controlled area should be identified, as well as the potential interactions with ISFSI or MRS operations.

2.4.1.3 *Population Distribution and Trends*

The SAR should present current population data and projections. This information may include such items as a sector map of the population in the surrounding area, extending to an adequate distance from the DSF. If appropriate, the sector map may divide the area within a 16-km (10-mi) radius of the site by concentric circles with radii of 1.5, 3, 5, 6.5, and 16 km (approximately 1, 2, 3, 4, and 10 mi), and by 22.5-degree segments, each centered on one of the 16 compass points. The map should provide current and projected populations in each sector. The population data should overlay a base map that shows cities or towns.

The maximally exposed real individual(s) should be specifically identified with a rationale for their selection (e.g., nearest well, closest person downwind in the predominant wind direction).

2.4.1.4 *Land and Water Use*

The SAR should describe the use of land and water within the surrounding area. It should present residential, farming, dairy, industrial, and recreational uses of land and water in sufficient detail to allow estimates of radiation doses to populations from any airborne or liquid effluents.

2.4.2 Nearby Industrial, Transportation, and Military Facilities

As required by 10 CFR 72.94, “Design basis external man-induced events,” the SAR must include an examination of past and present man-made facilities and activities that might endanger the proposed ISFSI or MRS. Therefore, the SAR should indicate the locations of nearby industrial, transportation, military, nuclear, and radioactive materials installations on a map that shows their distance and relationship to the ISFSI or MRS. All facilities within the surrounding nearby area and all relevant facilities at greater distances should be included. The SAR should describe the products or materials produced, stored, or transported for each facility, and any potential hazards to the ISFSI or MRS from activities or materials at the facilities. Finally, the SAR should discuss any effect of these facilities on the specific ISFSI or MRS design basis.

2.4.3 Meteorology

As required by 10 CFR 72.92, “Design basis external natural events,” the SAR must include an evaluation of any natural phenomena that may exist or that can occur in the region of a proposed site. Therefore, the SAR should describe the meteorological conditions at the DSF and vicinity and identify the conditions that could influence the design and operation of the facility. The SAR should state the sources of all information cited. Sufficient information should be provided to permit the NRC staff to independently evaluate atmospheric diffusion characteristics of the site area. The SAR should also provide sufficiently detailed information to permit the NRC staff to determine the basis for the high winds (either straight line or tornado winds) and high temperature used in the design basis.

2.4.3.1 *Regional Climatology*

The SAR should describe the climate of the region, including temperature, precipitation, relative humidity, general airflow, pressure patterns, cloud cover, average wind speeds, and prevalent wind direction, as well as the ranges and seasonal variations of these parameters. The SAR should mention climate characteristics attributable to terrain and present data on the frequency, intensity, and duration of severe weather. For example, the SAR should address temperature, wind, and precipitation extremes; hurricanes, tropical storms, tornadoes, lightning strikes; and snow, ice, and hail storms. The SAR should discuss all data sources and the reliability of the sources. The SAR should present the design-basis winds and temperature and explain a rationale for their selection.

2.4.3.2 *Local Meteorology*

The SAR's description of local meteorology should summarize data on temperature, wind speed and direction, and relative humidity collected on site as well as at nearby weather stations. The SAR should discuss any data collected offsite and whether the data are representative of the onsite conditions. If such offsite data adequately represent onsite conditions, then onsite data may not be necessary. For the purpose of evaluating atmospheric diffusion, the SAR should provide topographic maps at two different scales: One should show detailed topographic features, as modified by the facility, within an 8-km (5-mi) radius around the site; a smaller-scale map should show topography out to a 16-km (10-mi) radius around the site. This map should be accompanied by profiles of maximum elevation over distance from the center of the installation out to 16-km for each of the 22.5-degree compass-point sectors.

2.4.3.3 *Onsite Meteorological Measurement Program*

Unless offsite data adequately represent onsite conditions, the SAR should include meteorological data collected onsite, adequate for the NRC staff to conduct independent atmospheric dispersion estimates for both postulated accidents and expected routine releases of gaseous effluents. The meteorological data should be provided in the form of joint frequency distributions of wind speed and wind direction by atmospheric stability class. The SAR should state the measurements made, the locations and elevations of measurements, descriptions of the instruments used, instrument performance specifications, calibration and maintenance procedures, and data analysis procedures. Any onsite program and any programs to be used during operations to estimate offsite concentrations of airborne effluents should be described. Regulatory Guide (RG) 1.23, "Meteorological Monitoring Programs for Nuclear Power Plants," provides guidance related to an acceptable onsite meteorological measurements program and the format for presenting stability class data.

If no onsite measurement program exists, the SAR should provide justification for using data from nearby stations as long as those stations conform to the criteria of RG 1.23.

2.4.4 Surface Hydrology

As required by 10 CFR 72.98, "Identifying regions around an ISFSI or MRS site," the SAR must include an evaluation of the regional extent of external phenomena, man-made or natural, that are used as a basis for the design of the ISFSI or MRS. Therefore the SAR should contain adequate information for an independent review of all surface hydrology-related design bases, performance requirements, and operating procedures important to safety.

2.4.4.1 *Hydrologic Description*

The SAR should characterize the surface hydrologic features of the region, area, and site because this information is the basis for hydrologic engineering analyses. Specifically, the SAR should describe the location, size, and hydrologic characteristics of all streams, rivers, lakes, and adjacent shore regions that influence or may influence the site or facilities under severe hydrologic conditions. It should include topographic maps of the area and the site to give a clear understanding of these features. A map of the site area should indicate any proposed change to the natural drainage features. If the site is vulnerable to river flooding, any river control structures, upstream or downstream of the site, should be identified.

The SAR should identify the sources of the hydrologic information, the types of data collected, and the methods and frequency of collection. The SAR should also list the structures important to safety, including their exterior accesses, and equipment and systems that may be affected by hydrologic features. The SAR should note any surface waters that could potentially be affected by normal or accident effluents from the site. A listing of any population groups that use such surface waters as a potable water supply should be provided, as well as the size of these population groups, their location, and water-use rates.

2.4.4.2 *Floods*

The SAR should adequately support any claim that the proposed site is flood-dry, that is, with structures important to safety so high above potential sources of flooding that safety is obvious or can be documented with little analysis, as indicated in American National Standards Institute/American Nuclear Society (ANSI/ANS) 2.8, "Determining Design Basis Flooding at Power Reactor Sites."

If the DSF site is not flood-dry, then the SAR should identify the design-basis flood and provide a rationale for this specific design basis. Such a rationale should contain a synopsis of the flood history of the site, including dates and maximum water levels. Causes of past and potential future flooding, such as river or stream floods, surges, tsunami, dam failures, and ice jams, should be provided. The remainder of Section 2.4.4 of this SRP describes the required detailed analysis of the flooding potential of the site. This information should be detailed enough for the NRC staff to perform an independent flood analysis of the site, as described in RG 1.59, "Design Basis Floods for Nuclear Power Plants," and referenced in RG 3.48, "Standard Format and Content for the Safety Analysis Report for an Independent Spent Fuel Storage Installation or Monitored Retrievable Storage Installation (Dry Storage)," RG 3.62, "Standard Format and Content for the Safety Analysis Report for Onsite Storage of Spent Fuel Storage Casks," and RG 1.102, "Flood Protection for Nuclear Power Plants."

2.4.4.3 *Probable Maximum Flood on Streams and Rivers*

As required by 10 CFR 72.122(a), the applicant must evaluate the structures, systems, and components (SSCs) important to safety to withstand the effect of floods. Therefore, the SAR must consider the effects of the probable maximum flood (PMF) on adjacent streams and rivers in its detailed flood analysis. If the SAR did not follow the approach in ANSI/ANS 2.8 for assessing PMFs, then it should describe the alternative approach used. The SAR should describe the steps taken to derive the probable maximum precipitation (PMP) over the applicable drainage area, the precipitation losses, the amount of runoff, and the PMF, and include a topographic map that identifies drainage basins. The SAR should include the estimated discharge hydrograph for the PMF at the site and, if applicable, a similar hydrograph without the effects of an upstream reservoir. The conversion of the PMF peak discharge into water elevation at the site should be described. Wind-wave activity that could coincide with the PMF should be discussed. Finally, the SAR should summarize the locations and associated water levels for which PMF determinations have been made.

2.4.4.4 *Potential Dam Failures (seismically induced)*

If potential dam failures are necessary to identify flood design bases, then the SAR should discuss the effects of potential seismically induced dam failures (both upstream and downstream) on the water levels of streams and rivers. The SAR should describe existing or proposed dams and reservoirs that could influence conditions at the site and include seismic design criteria for dams. The potential dam failure modes that lead to the most critical consequences for the site (flood or low reservoir level) should be described, and domino-type or cascading dam failures from floodwaves should be considered when applicable. Finally, the SAR should address the reliability of the water-level estimate.

2.4.4.5 *Probable Maximum Surge and Seiche Flooding*

If the site is at risk of inundation from surge or seiche flooding, the SAR should describe these hazards. It should describe water bodies that could impact the site and provide the surge and seiche history of the site. The SAR should describe the frequency and magnitudes of potential causes of surges, such as hurricanes, wind storms, squall lines, and other mechanisms and include a graph of the calculated maximum surge hydrograph. The potentially coincident wind-generated waves and the possibility of wave oscillation at natural frequencies should be described. The SAR should provide estimates of potential wave run-up, erosion, and sedimentation and any site facilities designed to guard against these processes.

2.4.4.6 *Probable Maximum Tsunami Flooding*

If the site abuts a coastal area, the SAR should analyze the hazards associated with tsunami. The SAR should include an analysis of the history of tsunami in the region, whether recorded, translated, or inferred from the geologic record. The analysis should include all potential tsunami generators, such as specific faults, fault zones, volcanoes, and potential landslide areas. The maximum tsunami height from these causes should be estimated at the source, in deep water, offshore from the site, and onshore. A probable maximum tsunami should be derived from these analyses. Near-shore routing, wave breaking, bore formation, and resonance effects of the probable maximum tsunami should be discussed. The SAR should describe any structures designed to protect against tsunami flooding.

2.4.4.7 *Ice Flooding*

The SAR should indicate whether the site is subject to flooding caused by ice jams. If it is, the SAR should provide an analysis of this hazard. The SAR should describe the history of ice-jam formation in the region and the location of ice-generating mechanisms relative to the facility, as well as any structures designed to protect against flooding from ice jams. If the site is not subject to flooding from ice jams, the SAR should provide a brief statement of explanation.

2.4.4.8 *Flood Protection Requirements*

The SAR should describe the static and dynamic consequences of all types of flooding on each facility structure and component important to safety if the previous flooding analyses indicate that the structure or component is subject to flooding. The design bases required to ensure that all structures and components can survive all design flood conditions should be included.

2.4.4.9 *Environmental Acceptance of Effluents*

The SAR should describe the ability of the surface water and ground water environment to disperse, dilute, or concentrate normal and inadvertent liquid releases of radioactive effluents for the full range of anticipated operating conditions, including accident scenarios leading to worst-case releases. The SAR should identify all potential surface water and ground water pathways by which radionuclides could reach existing and potential water users. Any potential for water recirculation, sediment concentration, or hydraulic short-circuiting of cooling ponds should be assessed in anticipation of normal or accidental releases of radionuclides.

2.4.5 Subsurface Hydrology

As required in 10 CFR 72.122(b)(4), if the ISFSI or MRS is located over an aquifer which is a major water resource, measures must be taken to preclude the transport of radioactive materials to the environment through this potential pathway. Therefore, the SAR should contain adequate information for an independent review of all subsurface hydrology-related design bases and compliance with radiological dose and exposure standards.

If the site is located over an aquifer that is a source of well water, the SAR should describe the ground water aquifer(s) beneath the site, the associated hydrologic units, and their recharge and discharge areas. The SAR should provide the results of a survey of ground water users, well locations, source aquifers, water uses, static water levels, pumping rates, and drawdown. A water table contour map showing surface water bodies, recharge and discharge areas, and locations of monitoring wells to detect leakage from storage structures should also be provided. Information on monitoring wells should include wellhead elevation, screened interval, installation method, and representative hydrochemical analyses. In addition, the SAR should provide an analysis bounding the potential ground water contamination from site operations and a graph of time versus radionuclide concentration at the closest existing or potential downgradient well.

2.4.6 Geology and Seismology

The SAR should identify conditions that may influence the design and operation of the facility and state the sources of all information. It should provide enough information for an independent evaluation of the potential ground vibrations and the seismic and fault displacement hazards at the site area, in accordance with 10 CFR 72.102, “Geological and seismological characteristics for applications before October 16, 2003 and applications for other than dry cask modes of storage,” and 10 CFR 72.103, “Geological and seismological characteristics for applications for dry cask modes of storage on or after October 16, 2003.” Design bases for ground vibration, surface faulting, subsurface material stability, and slope stability should also be provided. Information on nearby and recent volcanic activity should also be identified, if appropriate or applicable.

2.4.6.1 *Basic Geologic and Seismic Information*

The SAR should provide basic geologic and seismic characteristics of the site and vicinity. The description of the geologic history of the area should include its lithologic, stratigraphic, and structural conditions. A large-scale geologic map of the site area showing the surface geology and the location of major facilities should be provided, as well as a stratigraphic column and cross sections. A geologic map showing bedrock surface contours should identify planar and linear features of structural significance such as folds, faults, synclines, anticlines, basins, and domes. The SAR’s description of the site geomorphology should include areas of potential landsliding or subsidence and include a topographic map showing geomorphic features and principal site facilities. It should provide the results of pertinent geophysical investigations in the area, such as seismic refraction, seismic reflection, aeromagnetic, or geoelectrical surveys.

The SAR should evaluate geologic features from an engineering geology perspective. Detailed static and dynamic engineering properties of soil and rock underlying the site should be provided, with the results integrated to provide a comprehensive understanding of the surface and subsurface conditions. A small-scale map should show major features of the installation and the locations of all borings, trenches, and excavations. Small-scale cross sections should demonstrate relationships between major foundations and subsurface materials, structures, and the water table. Finally, the SAR should present any physical evidence concerning the behavior of surficial site materials during previous earthquakes.

2.4.6.2 *Ground Vibration*

The SAR should present the design-basis ground vibration and explain a rationale for its selection. The rationale should list historical earthquakes that could have affected the site and their dates, epicenter locations, and magnitudes. This listing of events is not constrained by distance and may include entries for distant structures, such as the New Madrid fault system. All faults and epicenters should be displayed on maps of appropriate scales. The fault map should include all potentially significant faults or parts of faults within 161 km (100 mi) of the site, regardless of capability. The SAR should identify and adequately describe all capable faults (as defined in Appendix A, “Seismic and Geologic Siting Criteria for Nuclear Power Plants,” to 10 CFR Part 100, “Reactor Site Criteria”) that may be of significance in establishing the design-basis ground vibration for the site. The maximum ground vibration at the site should be derived from the potential earthquakes from all capable faults and from floating earthquakes (i.e., those not associated with a previously identified structure).

2.4.6.3 *Surface Faulting*

The SAR should describe surface faulting at the site and any underlying tectonic structures that have caused or might cause faulting. In addition, the SAR should describe the capability of any mapped faults 300 meters (1,000 feet) or longer within 8 km (5 mi) of the site. The SAR should describe in detail those faults judged to be capable, with special attention to their displacement history and their relationship to any regional tectonic structures.

2.4.6.4 *Stability of Subsurface Materials*

The SAR should describe the stability of the rock, defined as having a shear wave velocity of at least 1,166 meters per second (3,500 feet per second) and soil beneath the foundations of the facility structures while subjected to the design-basis ground vibration. The description should include the geologic features that could affect the foundations, such as areas of potential uplift or collapse, or zones of deformation, alteration, structural weakness, or irregular weathering. The SAR should describe the static and dynamic engineering properties of the materials underlying the site, as well as the physical properties of foundation materials. A plot plan showing the locations of all borings, trenches, seismic lines, piezometers, geologic cross sections, and excavations, with all installation structures superimposed, should be provided. Plans and profiles showing the extent of excavations and backfill, as well as compaction criteria, should be provided. Further, the water table history and anticipated ground water conditions beneath the site during facility construction and operation should be described. The SAR should provide analyses of soil and rock responses to dynamic loading and discuss potential liquefaction beneath the site. It should discuss criteria, references, or methods of design used, along with safety factors.

2.4.6.5 *Slope Stability*

The SAR should describe the stability of all natural and human-made slopes, both cut and fill, whose failure could adversely affect the site. The description should provide cross sections of the slopes and a summary of the static and dynamic properties of embankment and foundation soil and rock underlying the slopes. The design criteria and analyses used to determine slope stability should be described.

2.5 Review Procedures

Figure 2-1 shows the interrelationship between the site characteristics evaluation and the other areas of review described in this SRP.

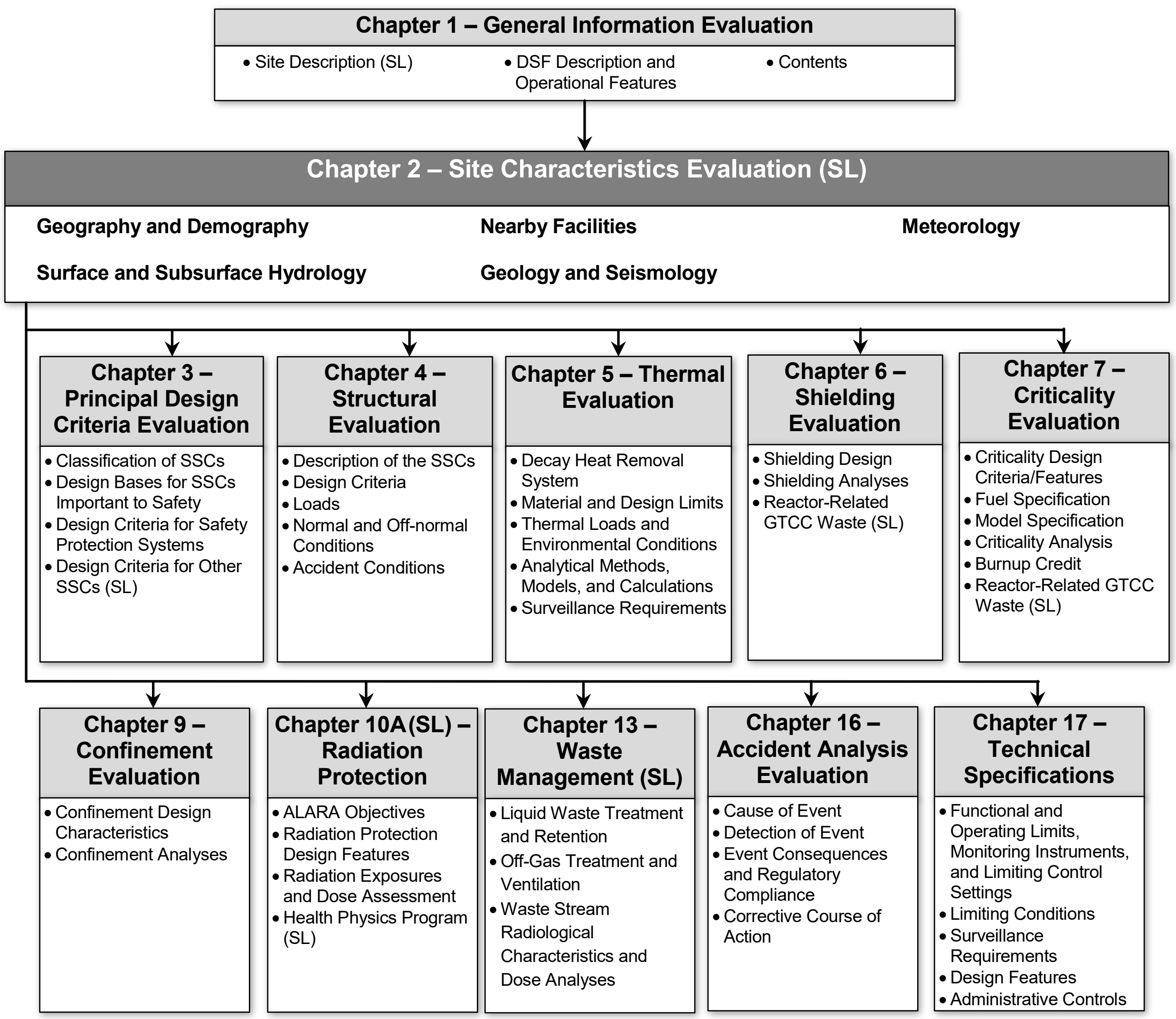

Figure 2-1 Overview of Site Characteristics Evaluation

2.5.1 Geography and Demography

2.5.1.1 *Site Location*

Confirm that the site location, its relationship to political boundaries, and the natural and anthropogenic features of the area are properly described. Use U.S. Geological Survey (USGS) topographic maps, aerial photos, or other verifiable methods (obtained either independently or from the applicant) to verify the location described in the SAR.

2.5.1.2 *Site Description*

Ensure that the site maps clearly delineate the site, controlled areas, and their boundaries. Confirm that the SAR accurately reports distances between the controlled area boundaries and the facility structures, including the storage location, as well as other possible effluent release points. These distances should agree with those used in the SAR discussion of accident analyses. Verify that the SAR indicates that the minimum distance from the DSF to the controlled area boundary is at least 100 meters (328 feet) per the requirements in 10 CFR 72.106, "Controlled area of an ISFSI or MRS." Check that the SAR indicates that access to the controlled area will be adequately restricted to protect members of the public, consistent with the requirements in 10 CFR Part 72.104. Ensure that the orientation of facility structures with respect to nearby roads, railways, and waterways is shown, and that there are no obvious ways by which transportation routes within the controlled areas can interfere with normal ISFSI or MRS operations

2.5.1.3 *Population Distribution and Trends*

Confirm that the source of the population data used in the SAR is appropriate and that the basis for population projections is reasonable. The population data can be compared with other data available from local or State agencies, councils of government, U.S. Census Bureau records and projections, or any Bureau of Economic Analysis special census. Note significant differences from SAR data that may require clarification.

Determine whether the rationale for identifying the maximally exposed real individual located at or beyond the controlled area boundary is consistent with local meteorology and patterns of land and water use.

2.5.1.4 *Land and Water Use*

Compare land use information provided in the SAR to existing data on land use, land use controls such as zoning, potential for growth, and other factors that may encourage or hinder population growth between the facility and the nearest population. Confirm the identification of any bodies of water or aquifers used by humans, livestock, or farms within the region surrounding the site. Compare SAR information with available independent data on water use and any projections of future water use in the vicinity of the site. Consider the level of detail appropriate to the projected distance of the nearest future population center to the site and the level of projected water withdrawal within the region surrounding the site.

2.5.2 Nearby Industrial, Transportation, and Military Facilities

Review the potential hazards associated with nearby facilities. In addition to obvious industrial, nuclear, or radioactive materials facilities in the area, consider other anthropogenic features that could conceivably pose a hazard, such as transportation routes, railroads, and airports. Confirm the accuracy of the information provided in the SAR by referring to USGS maps, aerial photos, or other documents, such as applications from any nearby nuclear plants. Use contacts with local, State, and other Federal agencies.

Review specific information relating to types of potentially hazardous material expected to be transported in the area, including distance, quantity, and frequency of shipment. The hazards from nearby facilities may include, but are not limited to, explosions of chemicals, flammable material, or munitions; detonation of explosives stored at mines or quarries; structure, petrochemical, brush, or forest fires; and release of toxic gases. Consider aircraft size, velocity, weight, and fuel load in assessing the hazards of aircraft crashes on an installation near an airport. Analyze the effects of any airborne pollutants from nearby facilities and the effects of a possible collapse of any discharge stacks on site. Determine if the methods documented in the application to quantify offsite hazards are consistent with the guidance in Chapter 16, "Accident Analysis Evaluation," of this SRP. Identify potential accidents that cannot be eliminated from consideration as design-basis events because the consequences could affect facility safety features. Ensure that such accidents are adequately considered in the design criteria of described in the SAR.

2.5.3 Meteorology

2.5.3.1 *Regional Climatology*

Review the SAR's description of climate parameters against standard references listed in NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition," Section 2.3.1(II), under the heading "SRP Acceptance Criteria," for verifying meteorological discussions and data. Confirm that the data sources are reliable and that the level of detail in the database is appropriate. Ensure that climate data are based on long-term data gathered at National Weather Service stations and other sites with reliable meteorological monitoring equipment. Review the information on severe weather, especially strong wind and wind-borne missiles, and check for consistency with the values used to develop structural design criteria in the SAR.

Ensure the regional meteorological conditions identified as site characteristics for ISFSI or MRS license applications include the following:

- the weight of the 100-year return period snowpack and the weight of the 48-hour probable maximum winter precipitation for use in determining the weight of snow and ice on safety-related structures

- the tornado parameters (including maximum wind speed, translational speed, rotational speed, and maximum pressure differential with the associated time interval) to be used in establishing pressure and tornado missile loadings on structures, systems, and components (SSCs) important to safety

- the 100-year return period (straight-line) 3-second gust wind speed to be used in establishing wind loading on safety-related structures

2.5.3.2 *Ambient Temperature and Humidity*

Ambient temperature and humidity statistics (e.g., 2 percent and 1 percent annual exceedance and 100-year maximum dry bulb temperature and coincident wet bulb temperature; 2 percent and 1 percent annual exceedance and 100-year maximum wet bulb temperature (noncoincident); 98 percent and 99 percent annual exceedance and 100-year minimum dry bulb temperature) for use in establishing heat loads for the design of heat sink systems Local Meteorology

Use maps and site visits to become familiar with the locations of all primary meteorological stations. Review the topographic maps for the accurate location of features and confirm the accurate portrayal of topography on the topographic profiles. Review summaries of the meteorological data for adequacy and completeness of the database. Whenever possible, review the onsite wind speed and atmospheric stability data that are used to model atmospheric diffusion because airflow and vertical temperature structure can vary substantially over short horizontal distances. If only offsite data are available, determine how well the data represent site conditions. Consult references in NUREG-0800, Section 2.3.2(II), under the heading "Acceptance Criteria," to evaluate whether the meteorological data from the weather stations and periods of record adequately represent onsite conditions. Data summaries from nearby stations with long periods of records should well represent long-term meteorological extremes. Ensure consistency between these extreme values and those used to develop structural and thermal design criteria.

2.5.3.3 *Onsite Meteorological Measurement Program*

Review two areas in this section, the instruments gathering the meteorological data and the data itself, by examining instrument siting, meteorological sensors, recordings of meteorological sensor output, instrument surveillance, and data acquisition and reduction, as discussed in detail in RG 1.23 and NUREG-0800, Section 2.3.3, "Onsite Meteorological Measurements Programs."

Review the joint frequency distributions of wind speed, wind direction, and atmospheric stability. Ensure that measurement heights and data recording periods are appropriate. In addition, determine the climatic representativeness of the joint frequency distribution by comparing with data from nearby stations that have collected reliable meteorological data over a long period, such as 10–20 years. Ensure that the meteorological measurement program is consistent with gaseous effluent release structures and systems design. Verify that the effluent release structures and systems design are commensurate with the degree of risk to public health and safety.

2.5.4 Surface Hydrology

2.5.4.1 *Hydrologic Description*

Ensure that the SAR addresses and properly describes all relevant hydrologic features by using USGS topographic maps and available independent hydrologic reports for this verification. Determine whether hydrologic features that influence or may influence the site under severe hydrologic conditions (e.g., a flood) have been adequately described. Review the criteria governing the operation of any upstream or downstream river control structures for scenarios of problems in river management. Examine any proposed alterations to the natural drainage pattern of the site. Ensure that the design of any SSCs important to safety can accommodate the effects of these alterations. Review local hydrologic reports to confirm the SAR's identity of population groups getting potable water from the described hydrologic features. Use references in NUREG-0800, Section 2.4.1(II), under the heading "Acceptance Criteria," to verify information in the application.

2.5.4.2 *Floods*

Review any claim that the site is flood-dry. Consider that a descriptive statement of circumstances and relative elevations may be enough to complete such a review. Evaluate the bases of any analogy with comparable watersheds for which PMF levels have been determined or approximations of PMF levels used. Require details only to the level needed to prove that SSCs important to safety are safe from flooding. Ensure that conservatism is used in all methods and assumptions. Consult ANSI/ANS 2.8 for descriptions of acceptable procedures to demonstrate flood-dry status.

If the site is not clearly flood-dry, review in detail the flood analyses. Determine whether the SAR chapter on principal design criteria adequately addresses the design-basis flood.

2.5.4.3 *Probable Maximum Flood on Streams and Rivers*

Review the SAR derivation of the PMF. Rely on information from actual storms in the region of the drainage basin. Consider storm configurations, maximum storm precipitation amounts (compare these with National Weather Service and U.S. Army Corps of Engineers determinations), time distributions, orographic effects, storm centering, seasonal effects, antecedent storm sequences, and antecedent snowpack. Confirm by calculations that the maximum storm precipitation distribution for the drainage basin is conservative. Review the SAR analysis of the absorption capability of the drainage basin. Ensure that assumptions of initial losses, infiltration rates, and antecedent precipitation are reasonable and justified. Review the SAR model for calculating runoff, as well as the input data such as hydrologic response characteristics of the watershed. Check that subbasin drainage areas and topographic features are mapped properly, and review the tabulation of drainage areas, runoff, and reservoir and channel-routing coefficients. Confirm that the PMF hydrograph represents the flow from the PMP and any possible coincident snowmelt.

Determine whether the PMF analysis considers any existing or proposed upstream dams or river structures and their ability to withstand a PMF. Confirm the maximum water flows from breaches if they are not designed to withstand a PMF. Review the PMF stream course response model and its ability to compute floods of various magnitudes up to the severity of a PMF. Review any reservoir and channel-routing assumptions, and the assessment of initial conditions, outlet works, spillways, coincident wind-wave action, wave protection, and reservoir design capacity. Review the process of translating PMF discharge to peak water level at the site by such means as topographic profiles, reconstitution of historical floods, standard step methods, roughness coefficients, bridge and other losses, extrapolation of coefficients for the PMF, estimates of PMF water surface profiles, and flood outlines. Review the SAR discussion of the effects on structures from run-up and the static and dynamic effects of wave action that may occur coincidentally with the PMF peak water level.

Perform an independent analysis of the PMF by using alternative data and interpretations when available. Request additional justification if the SAR analyses are more than 5 percent less conservative than independent NRC estimates.

Consult the following documents in reviewing SAR data and analyses:

- RG 1.59 for guidance on estimating the PMF design basis
- RG 1.102 for a description of acceptable flood protection for safety-related facilities

- National Weather Service and Army Corps of Engineers (USACE) documents (e.g., NWS 1978, 1982; USACE 1984, 1987, 1991, 1998) for estimating PMF discharge and water-level conditions at the site

2.5.4.4 *Potential Dam Failures (seismically induced)*

Review the SAR to determine whether the applicant considered all relevant dams and reservoirs that could affect the site in the event of failure. Review the drainage areas above reservoirs, and ensure that all dam structures, appurtenances, and ownership are completely described. Review the reservoir elevation and storage relationships and short- and long-term storage allocations. Ensure that the discussion of dam failures considers all factors, including landslides, antecedent reservoir levels, domino-type multiple dam failures, and base-river flow coincident with the flood peak, but not necessarily the simultaneous occurrence of the PMF with a seismic dam failure. Ensure that the applicant used a conservative analysis and that the analysis assumes that the maximum earthquake (based on historical seismicity) coincides with full reservoirs and either a flood half the size of the PMF or a standard-project flood as defined by the Army Corps of Engineers. Review for conservatism the basis for selecting the maximum earthquake that can lead to dam failure.

Review the calculations used to derive the peak flow rate and water level at the site that could result from the worst-possible dam failure. Examine all methods and coefficients used in these calculations, and ensure that the analytical methods apply to such artificially large floods. Review the discussion of static and dynamic effects of the floodwave at the site. Examine the assumptions used to attenuate the wave if credit is taken for downstream attenuation of a floodwave. Ensure that wind waves that may coincide with the flood are properly considered.

Conduct a more refined analysis, as described in NUREG-0800, Section 2.4.4(III), if this flooding analysis indicates a potential flooding problem. To the extent possible, conduct an independent analysis of the flooding effects from a seismically induced dam failure by using simplified, conservative procedures according to guidance in ANSI/ANS 2.8. Require additional justification if the SAR analyses are more than 5 percent less conservative than independent NRC estimates.

2.5.4.5 *Probable Maximum Surge and Seiche Flooding*

Review the descriptions of potential surge and seiche sources, ensuring that they address the most severe combination of reasonable meteorological parameters, including storm track, wind fields, wind fetch, and bottom effects. Use NUREG-0800, Section 2.4.5(III), for its discussion of methods to develop the maximum hurricane parameters for a site, to estimate the maximum surge still water elevations at coastal sites, and to estimate coincident wind-generated waves and run-up. Use National Oceanic and Atmospheric Administration Technical Report NWS-23 (NWS 1979), ““Meteorological Criteria for the Standard Project Hurricane and Probable Maximum Hurricane Windfields, Gulf and East Coasts of the United States,” for its descriptions of the meteorological characteristics of the probable maximum hurricane for the East and Gulf Coasts, the most severe combination of meteorological parameters of moving squall lines for the Great Lakes, and the most severe combination of meteorological parameters capable of producing high storm-induced tides for the West Coast.

Confirm that ambient water levels, including tides and sea-level anomalies, are conservatively estimated. Use NUREG-0800, Section 2.4.5(III), for its discussion of water-level estimation methods that follow the National Oceanic and Atmospheric Administration and USACE guidance. Ensure that the method of developing the surge hydrograph from the meteorological, hydrological, and site-specific information is appropriate. Review the information on wave action that may coincide with surges. Ensure that estimates of wave height and run-up are adequately conservative and, if appropriate, include breaking waves. Review the analysis of wave resonance within any lakes or harbors near the site.

To the extent possible, conduct an independent analysis of the water level and wave height for surges and seiches by using alternative data and interpretations when available. Request additional justification if the SAR analyses are more than 5 percent less conservative than independent NRC estimates.

2.5.4.6 *Probable Maximum Tsunami Flooding*

Review the historical tsunami information provided in the SAR for completeness. Review for completeness the tabulation of source areas capable of generating tsunami at the site. Evaluate the seismic characteristics of the tsunami generators, including fault location and orientation, as well as amplitude and areal extent of potential vertical displacement to ensure the application uses conservative values. Examine this information for consistency with that provided in the SAR geology and seismology section. Review the tabulation of maximum tsunami wave heights that can be generated at each local source and the maximum deep-water heights generated by distant sources. Review the process used to identify the source of the probable maximum tsunami for transparency. Examine the method used to translate tsunami waves from deep-water, offshore locations to the site. Review the analysis of local factors that may affect the magnitude of tsunami flooding, such as coastline shape, offshore land areas, hydrography, and stability of the coastal area. Ensure the reasonableness of assumptions and the inclusion of appropriate bathymetric data in the analysis. For the probable maximum tsunami, review the analysis of potential breaking wave formation, bore formation, resonance effects, or other factors that can affect the maximum height of the tsunami water level. Use NUREG-0800, Section 2.4.6(III), for references for evaluating ambient tide and wave conditions, oscillation of waves at natural periodicity, and the adequacy of protection from flooding.

To the extent possible, conduct an independent analysis of the source of the probable maximum tsunami and its resulting water height at the site by using alternative data and interpretations when available. Request additional justification if the SAR analyses are more than 5 percent less conservative than independent NRC estimates.

2.5.4.7 *Ice Flooding*

Determine whether ice flooding poses a threat to the site on the basis of a review of the applicable literature describing historical occurrences of icing in the region, and, if so, ensure the adequacy of the SAR historical description. Use NUREG-0800, Section 2.4.7(III), for references in researching the history and potential for ice formation in the region. Ensure that the SAR properly considers all ice-related hazards, such as ice-jam floods, wind-driven ice ridges, and ice-produced forces that could affect the site. If feasible, conduct an independent analysis of the ice flooding hazard by using independent data and assumptions.

2.5.4.8 *Flood Protection Requirements*

Compare the estimated design-basis flood level (both SAR and any independent estimates) with the locations and elevations of SSCs important to safety to confirm whether flood protection at the site is necessary and, if so, to what levels. If flood protection is necessary, review the facility flood design basis for compatibility with the positions in RG 1.59. Appropriate flood protection measures must protect against both static and dynamic flooding effects; RG 1.102 provides guidance for implementing 10 CFR 72.92(a). Review the SAR for flood protection measures based on standard engineering practices, such as those developed by the Federal Emergency Management Agency (e.g., FEMA 1999, FEMA 2013), in positive flood control and shoreline protection.

2.5.4.9 *Environmental Assessment of Effluents*

Evaluate scenarios for routine, anticipated (or off-normal), and accidental releases to ensure consideration of worst-case releases of radionuclides into surface water or ground water. Examine the physical parameters used in calculating the transport paths and times of liquid effluent between the release point and receptors downstream or downgradient. Confirm that mathematical models used in the application to analyze flow and transport have been verified by field data and have used conservative input parameters. Ensure that any site-specific data sources used in modeling the transport of radionuclides through water are adequately described and referenced.

Use independent data and assumptions to the extent possible to assess the transport capabilities and potential contamination pathways of the surface water and ground water environments. Focus this independent assessment on transport to existing and possible future water users under normal, anticipated (or off-normal), and accident conditions. Use NUREG-0800, Section 2.4.13(III) for its descriptions of simplified, calculation procedures for models used to assess effluent transport through surface water and ground water.

2.5.5 Subsurface Hydrology

Review the descriptions of hydrogeologic units beneath the site. For each hydrogeologic unit, ensure the proper representation of potentiometric level, hydraulic gradient and conductivity, effective porosity, storage coefficient, recharge and discharge areas, and potential for ground water flow reversal. For the water table aquifer, ensure that the application has conservatively bounded seasonal fluctuations in the water level. Compare the SAR chemical analyses, including major ions, acidity/alkalinity, electrical potential, and presence of radionuclides, with independent analyses.

Review the information on existing ground water use, such as withdrawal points, pumping rates, source aquifers, and drawdown. Use reports by USGS or a State geological survey in reviewing site hydrogeology and water withdrawal downgradient of the site.

Review the analysis of the potential effects of the facility on any ground water recharge areas within the site, including dewatering during construction. Ensure that this analysis uses conservative assumptions and input values. Confirm that estimated ground water withdrawal volumes during facility operation are conservative and that drawdown or other effects on the aquifer(s) are addressed.

Review the transport characteristics of aquifers that are subject to radionuclide contamination. Ensure that the application adequately describes any contamination pathways and that the models and codes used to predict radionuclide migration are appropriate for the site. Ensure that potential future ground water uses are conservatively estimated. If warranted, conduct an independent analysis of radionuclide migration by using an alternative transport model or independent data.

2.5.6 Geology and Seismology

2.5.6.1 *Basic Geologic and Seismic Information*

Verify the documentation of the results from all independent surveys, geophysical studies, borings, trenches, and other investigations. Review the descriptions of techniques, graphic logs, photographs, laboratory results, and identification of principal investigators. Review the reports cited in the SAR, such as published reports and dissertations, as well as other relevant reports on local geology.

Review the SAR discussion of basic site characteristics that may be problematic in siting a DSF, such as high seismic activity or recent volcanic activity. Scrutinize any SAR statement that the presence of unstable geologic characteristics will not have a deleterious effect on the facility or that the effects are within the design bases of all facility components important to safety.

Examine the geologic maps, cross sections, and stratigraphic columns in the SAR. For each lithologic unit, review the origin, unit thickness, physical characteristics, mineral composition, and degree of consolidation. Use the summary logs of borings, excavations, and trenches in reviewing the lithology. Compare the geologic map for the site area with other available published maps. If the SAR interpretations differ substantially from the published literature, ensure that the differences are noted and that the SAR interpretations are adequately justified. Review the bedrock contour map to confirm that the application accurately represents all relevant structural features. Review the description of the site geomorphology to ensure that all significant landforms, including the geologic processes that engendered them, are properly described. Ensure that the application identifies all locations of potential landsliding, subsidence, or uplift resulting from natural or anthropogenic processes and evaluates any associated hazards.

Review the results of any geophysical surveys, paying particular attention to the methods by which the data were gathered. Compare the interpretations of stratigraphy and structures with other cross sections. Require that discrepancies be explained. Examine any values of compressional and shear wave velocities for reasonableness.

Review the plan showing the locations of all major features of the facility, as well as the locations of all borings, trenches, and excavations. Examine the cross sections showing the relationships of engineered structures to subsurface material. Ensure that the application accurately represents the water table (and fluctuation range) and that ground water cannot have an adverse effect on these structures. Review the profile drawings that show the extent of excavation and backfill, as well as the compaction criteria for the engineered backfill. Ensure that compaction criteria meet appropriate engineering standards. Determine whether the SAR conservatively evaluates the effects of deformation zones, such as shears, joints, fractures, faults, or folds, on structural foundations. Ensure that the SAR addresses alteration zones, irregular weathering profiles, and zones of structural weakness composed of crushed or disturbed materials in terms of engineering geology.

Examine the tabulation of the static and dynamic engineering soil and rock properties of the various materials underlying the site, including grain size classification, Atterberg limits, water content, unit weight, shear strength, relative density, shear modulus, Poisson’s ratio, bulk modulus, damping, consolidation characteristics, seismic wave velocities, density, porosity, strength characteristics, and strength under cyclic loading. Ensure that the data are substantiated with appropriate representative laboratory test records. Give extra attention to mechanical properties of aquifer materials and any fine-grained materials associated with the uppermost confined or semiconfined aquifer. Scrutinize any site materials that may have an adverse response to seismic shaking, as well as any rocks or soils that may be unstable because of their mineral composition, lack of consolidation, or water content. For those that may respond adversely to seismic shaking, ensure that the SAR uses conservative estimates for seismic response characteristics, such as liquefaction, thixotropy, differential consolidation, cratering, and fissuring. Review the SAR for the inclusion of available data on the behavior of site geologic materials during previous earthquakes. Review the analytical techniques and safety factors used in evaluating the stability of foundations for all structures and embankments under normal operating and extreme environmental conditions.

2.5.6.2 *Ground Vibration*

Examine the provided maps of earthquake epicenters and faults in the region. Confirm that the epicenter map adequately represents the locations of the tabulated historical earthquakes. Ensure that the earthquake tabulation comes from a credible source; compare it with an alternative earthquake catalog if available. Confirm that the SAR uses sound practices in estimating the magnitudes of historical earthquakes that predate seismological instrumentation. Consider differences in soil and bedrock properties between the site and the location where earthquake intensity was reported. Review the descriptions of any capable faults, including length, relationship to regional tectonic structures and the regional stress regime, and the nature and amount of the maximum displacement per event during the Quaternary. Ensure that the SAR uses suitable methods, such as those outlined by Slemmons (1977), to determine fault capability. Ensure that fault studies used photogeologic work and field investigations. Compare the SAR findings to any published alternative interpretations. Review any justification of noncapability for any fault within 161 km (100 mi) of the site that, if it produced its maximum magnitude earthquake at its closest distance to the site, would produce site ground acceleration greater than or equal to the design value. Confirm that field investigations and conservative assumptions justify the classification of such a fault as noncapable. Use trench excavations in determining capability if a fault is overlain by Late Pleistocene sediments.

Review the SAR calculation of the ground motion design-basis value as defined by a response spectrum corresponding to the peak horizontal ground acceleration. A standardized design-basis earthquake described by an appropriate response spectrum anchored at 0.25 g may be used for the site if it meets three criteria: (1) located east of the Rocky Mountain front; (2) not in a seismically active region (e.g., New Madrid, Missouri; Charleston, South Carolina; or Attica, New York); and (3) not subject to ground motion above 0.2 g (per an appropriate response spectrum) as shown by reconnaissance investigation. Alternatively, for sites that do not meeting the three criteria, ensure that the application references 10 CFR Part 100, Appendix A, to develop a ground motion design-basis value.

Review the ground motion value derived from the methods in 10 CFR Part 100, Appendix A, by using the following procedures.

- Ensure that all capable faults have been considered as seismic sources, with the maximum magnitude earthquake occurring on the fault at its nearest approach to the site.
- Ensure that the maximum magnitude event is based on an accepted fault length-to-magnitude relationship, such as Slemmons et al. (1982) or Bonilla et al. (1984).

- Use a next-generation attenuation (NGA) model to ensure that the peak ground acceleration at the site is calculated from the earthquake magnitude and the site-to-source distance. For the western United States, next-generation attenuation models include those of Chiou and Youngs (2014), Campbell and Bozorgnia (2014), Abrahamson et al. (2014), and Boore et al. (2014). Pending completion of the next-generation attenuation East Project, for the central and eastern United States, use the model described in Electric Power Research Institute (2013).

- Ensure that the SAR analysis considered a floating earthquake, that it based the floating earthquake magnitude on the seismological history of the tectonic province, and that it used 15 km (9 mi) as the site-to-source distance for calculating ground acceleration at the site. Ensure that the SAR considered adjacent provinces and their characteristic floating earthquakes if the site is near a tectonic province boundary. Ensure that the site-to-source distance for a floating earthquake in an adjacent province is 15 km or the closest approach of the province to the site, whichever is greater.

- Ensure that the site-specific response spectrum used to derive the peak horizontal ground acceleration from the design-basis earthquake considers the specific engineering properties of the material underlying the site, including seismic wave velocities, density, water content, porosity, and strength. Ensure that the design criteria in the SAR consider the design ground motion value.

2.5.6.3 *Surface Faulting*

Review the SAR evaluation of tectonic structures underlying the site. Consider whether the application uses boreholes or geophysical surveys to reveal buried structures. Determine the need for geophysical or other studies to establish the presence or absence of such structures if local geology investigations provide some evidence that buried, potentially active structures may underlie the site. Ensure that the SAR evaluation of onsite structures considers the effects of human activities, such as mining activity, loading effects from dams or reservoirs, and pumping fluids out of or into the subsurface, and the proclivity of faults to slip. Confirm that the SAR includes a capability assessment of all faults longer than 300 meters (1,000 feet) and passing within 8 km (5 mi) of the site. Examine these assessments to ensure that the conclusions are based on sound geologic principles and practices and, in cases where capability remains equivocal, a preponderance of the available geologic evidence. Review the information provided on fault length and relationship to regional tectonic structures, the nature and amount of Quaternary displacement, and the magnitude of the maximum Quaternary displacement event for those faults that are deemed capable. Ensure that the SAR identifies the outer limits of the fault or fault zone along the trace 16 km (10 mi) in either direction of the point where the fault makes its closest approach to the site. Ensure that any fault displacement, if the site is subject to surface faulting, does not exceed the design criteria. Ensure the safety margin is sufficient if critical facilities are to be located in areas subject to displacement because fault displacement is a difficult phenomenon to assess.

2.5.6.4 *Stability of Subsurface Materials*

Review the description of geologic features to ensure that the application has not overlooked any natural features that could affect foundation stability during ground shaking. Examine the tabulations of the physical and engineering properties for the foundation materials underlying the site. Ensure that foundation material properties include grain size classification, consolidation characteristics, water content, Atterberg limits, unit weight, shear strength, relative density, shear modulus, damping, Poisson's ratio, bulk modulus, strength under cyclic loading, seismic wave velocities, density, porosity, and strength characteristics. Compare selected values against representative laboratory test results to confirm the accuracy of the values of selected properties.

Examine the SAR plans and profiles of the locations of investigative studies and facility structures. Confirm that the plans include all appropriate boreholes, trenches, and other excavations. Ensure that the profiles accurately show the relationships between structure foundations and subsurface materials and the ground water and engineering characteristics of the subsurface materials. Review the SAR plans and profiles that show excavation and backfill activity to ensure that compaction criteria are substantiated with representative laboratory or field-test records. Examine the tables and profiles of the compressional and shear wave velocities in the soil and rock beneath the site. Ensure that these data were gathered by appropriate methods. Examine any graphic logs of boreholes, trenches, or other excavations for accuracy. Ensure that the SAR analyses of the soil and rock responses to dynamic loading are conservative.

Review the discussion of the liquefaction potential of material beneath the site. Conduct an independent analysis to verify a claim that liquefaction-susceptible soils are absent beneath the site. Ensure that the discussion of soil zones with the potential for liquefaction includes relative density, void ratio, ratio of shear stress to initial effective stress, number of load cycles, grain size distribution, degrees of cementation and cohesion, and ground water elevation fluctuations.

Ensure that the SAR analysis for soil stability uses the appropriate response spectra in determining the design ground motion from the design-basis earthquake. Ensure that the static analyses address settlement and lateral pressures and are accompanied by representative laboratory data. Review the SAR specifications for any techniques, such as grouting, vibraflotation, rock bolting, or anchors, required to improve unstable subsurface conditions. Ensure that designs follow proper engineering standards. Examine the safety factors and the criteria, references, or methods of design used in ensuring that the facility can withstand seismic ground motion and surface faulting.

2.5.6.5 *Slope Stability*

Examine the slope cross-section drawings for accuracy. Review the static and dynamic properties of the embankment and foundation soil and rock beneath the slope to ensure that the values are reasonable and substantiated with representative laboratory test data. Ensure that stability assessments address the potential effects of erosion, deposition, and seismicity, either individually or in combination. Ensure that erosional processes discuss sheet and rill flow, mass wasting, and valley widening. Ensure that the compaction specifications are based on representative laboratory analyses. Review the logs of core borings and test pits taken in these areas for any proposed borrow areas. Ensure that the analyses supporting the slope and erosional stability findings use conservative methods and assumptions.

2.6 <u>Evaluation Findings</u>

The NRC reviewer should prepare evaluation findings upon satisfaction of the regulatory requirements in Section 2.4 of this SRP. If the documentation submitted with the application fully supports positive findings for each of the regulatory requirements, the statements of findings should be similar to the following:

F2.1 The SAR provides an acceptable description and safety assessment of the site on which the [ISFSI/MRS] is to be located, in accordance with 10 CFR 72.24(a).

F2.2 The proposed site complies with the criteria in 10 CFR Part 72, Subpart E, "Siting Evaluation Factors," as required in 10 CFR 72.40(a)(2).

The reviewer should provide a summary statement similar to the following:

> As set forth above, the applicant has presented and substantiated information to establish the site characteristics. The staff has reviewed the information provided and, for the reasons given above, concludes that it is sufficient for the staff to evaluate compliance with the requirements in 10 CFR Part 72. The staff further concludes that the applicant provided sufficient details about the site characteristics to allow the staff to evaluate, as documented in this safety evaluation report, whether the applicant has met the relevant requirements of 10 CFR Part 72 with respect to determining the acceptability of the site.

2.7 <u>References</u>

10 CFR Part 72, "Licensing Requirements for the Independent Storage of Spent Nuclear Fuel, High-Level Radioactive Waste, and Reactor-Related Greater than Class C Waste."

10 CFR Part 20, "Standards for Protection Against Radiation,"

10 CFR Part 100, "Reactor Site Criteria."

Abrahamson, N.A., W.J. Silva, and R. Kamai, "Summary of the ASK14 Ground Motion Relation for Active Crustal Regions," *Earthquake Spectra*, 30:1025–1055, 2014.

American National Standards Institute/American Nuclear Society 2.8, "Determining Design Basis Flooding at Power Reactor Sites."

Bonilla, M.G., R.K. Mark, and J.J. Lienkaemper, "Statistical Relations among Earthquake Magnitude, Surface Rupture Length, and Surface Fault Displacement," *Bulletin of the Seismological Society of America*, 74:2379-2411, 1984.

Boore, D.M., J.P. Stewart, E. Seyhan, and G.M. Atkinson, "NGA-West2 Equations for Predicting PGA, PGV, and 5%-damped PSA for Shallow Crustal Earthquakes," *Earthquake Spectra*, 30:1057–1085, 2014.

Campbell, K.W. and Y. Bozorgnia, "NGA-West2 Ground Motion Model for the Average Horizontal Components of PGA, PGV, and 5%-Damped Linear Acceleration Response Spectra," *Earthquake Spectra*, 30:1087–1115, 2014.

Chiou, B-S.J. and R.R. Youngs, "Update of the Chiou and Youngs NGA Ground Motion Model for Average Horizontal Component of Peak Ground Motion and Response Spectra," *Earthquake Spectra*, 30:1117–1153, 2014.

Electric Power Research Institute, "Ground Motion Model (GMM) Review Project," Final Report, 2013.

Federal Emergency Management Agency (FEMA), "Protecting Building Utilities from Flood Damage, Principles and Practices for the Design and Construction of Flood Resistant Building Utility Systems," First Edition, FEMA 348, November 1999.

FEMA, "Floodproofing Non-Residential Buildings," FEMA P-936, July 2013.

National Weather Service (NWS), "Probable Maximum Precipitation Estimates, United States East of the 105th Meridian," Hydro-meteorological Report No. 51, National Oceanic and Atmospheric Administration, Washington, DC, June 1978.

NWS, "Meteorological Criteria for the Standard Project Hurricane and Probable Maximum Hurricane Windfields, Gulf and East Coasts of the United States," Technical Report NWS-23, National Oceanic and Atmospheric Administration, September 1979.

NWS, "Application of Probable Maximum Precipitation Estimates—United States East of the 105th Meridian," Hydrometeorological Report No. 52, National Oceanic and Atmospheric Administration, Washington, DC, April 1982.

NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition."

Regulatory Guide 1.23, "Meteorological Monitoring Programs for Nuclear Power Plants."

Regulatory Guide 1.59, "Design Basis Floods for Nuclear Power Plants."

Regulatory Guide 1.102, "Flood Protection for Nuclear Power Plants."

Regulatory Guide 3.48, "Standard Format and Content for the Safety Analysis Report for an Independent Spent Fuel Storage Installation or Monitored Retrievable Storage Installation (Dry Storage)."

Regulatory Guide 3.62, "Standard Format and Content for the Safety Analysis Report for Onsite Storage of Spent Fuel Storage Casks."

Slemmons, D.B., "State-of-the-Art for Assessing Earthquake Hazards in the United States: Report 6, Faults and Earthquake Magnitude," Miscellaneous Paper S-73-1, U.S. Army Corps of Engineers Waterways Experiment Station, Vicksburg, MS, 1977.

Slemmons, D.B., P. O'Malley, R.A. Whitney, D.H. Chung, and D.L. Bernreuter, "Assessment of Active Faults for Maximum Credible Earthquakes of the Southern California-Northern Baja

Region," Lawrence Livermore National Laboratory (LLNL), University of California, LLNL Publication No. UCID 19125, 1982.

U.S. Army Corps of Engineers (USACE), "Probable Maximum Flood Estimation—Eastern United States," Technical Paper 100, Hydrologic Engineering Center, Davis, CA, September 1984.

USACE, "HMR52 Probable Maximum Storm (Eastern United States) User's Manual," CPD-46, Hydrologic Engineering Center, Davis, CA, April 1987.

USACE, "HEC-2 Water Surface Profiles—User's Manual," CPD-2A, Hydrologic Engineering Center, Davis, CA, September 1991.

3 PRINCIPAL DESIGN CRITERIA EVALUATION

3.1 Review Objective

The objective of the U.S. Nuclear Regulatory Commission's (NRC's) review of the principal design criteria and bases related to structures, systems, and components (SSCs) and safety protection systems is to ensure that the principal design criteria comply with the requirements in Title 10 of the *Code of Federal Regulations* (10 CFR) Part 72, "Licensing Requirements for the Independent Storage of Spent Nuclear Fuel, High-Level Radioactive Waste, and Reactor-Related Greater Than Class C Waste." The result of this review will determine whether the applicant adequately defined (1) the classification of SSCs according to their importance to safety and (2) the design criteria and bases for SSCs important to safety, safety protection systems, and other SSCs. These design criteria and bases include the limiting characteristics of the spent nuclear fuel (SNF), reactor-related greater-than-Class-C (GTCC) waste, or other high-level radioactive waste (HLW) materials to be stored and external conditions during normal and off-normal operations, accident conditions, and natural phenomena events.

3.2 Applicability

This chapter applies to the review of applications for specific licenses for an independent spent fuel storage installation (ISFSI) or a monitored retrievable storage installation (MRS), categorized as a dry storage facility (DSF). It also applies to the review of applications for a Certificate of Compliance (CoC) of a dry storage system (DSS). Sections of this chapter that apply only to a DSF specific license application are identified with "**(SL)**" in the heading. Sections or tables that apply only to DSS CoC applications have "**(CoC)**" in the heading. A subsection without an identifier applies to both types of applications.

3.3 Areas of Review

This chapter addresses the following areas of review:

- classification of SSCs
- design bases for SSCs important to safety
    - SNF specifications
    - reactor-related GTCC waste specifications **(SL)**
    - HLW specifications **(SL–MRS only)**
    - external conditions
- design criteria for safety protection systems
    - general
    - structural
    - thermal
    - shielding, confinement, and radiation protection
    - criticality
    - material selection
    - decommissioning **(SL)**
    - retrievability
- design criteria for other SSCs

3.4 Regulatory Requirements and Acceptance Criteria

This section summarizes those parts of 10 CFR Part 72 that are relevant to the review areas addressed by this chapter. The reviewer should refer to the exact language in the regulations. Table 3-1a matches the relevant regulatory requirements to the areas of review for a specific license **(SL)** review. Table 3-1b matches the relevant regulatory requirements to the areas of review for a CoC review.

Table 3-1a Relationship of Regulations and Areas of Review for a DSF (SL)

Areas of Review	10 CFR Part 72 Regulations			
	72.24 (a)(b)(c)(e)(f)(l)	72.40 (a)(1)(2)(3)	72.90–94	72.98
Classification of SSCs	•			
Design Bases for SSCs Important to Safety	•		•	•
Design Criteria for Safety Protection Systems	•	•		
Design Criteria for Other SSCs	•			

Areas of Review	10 CFR Part 72 Regulations (cont.)					
	72.102–103	72.104	72.106	72.120	72.122–126	72.128–130
Classification of SSCs						
Design Bases for SSCs Important to Safety	•	•	•	•	•	•
Design Criteria for Safety Protection Systems		•	•	•	•	•
Design Criteria for Other SSCs				•	•	•

Table 3-1b Relationship of Regulations and Areas of Review for a DSS (CoC)

Areas of Review	10 CFR Part 72 Regulations		
	72.104[A]	72.106[A]	72.122 (a), (b)(1),(2)(i),(3) (c), (f), (h)(1)(4) (i), (l)[B]
Classification of SSCs			
Design Bases for SSCs Important to Safety	●	●	●
Design Criteria for Safety Protection Systems	●	●	●

Areas of Review	10 CFR Part 72 Regulations (cont.)			
	72.124 (a)(b)	72.126 (a)(1)(2)(3) (4)(5)(6)[B]	72.236 (a)(b)(c)(d)	72.236 (e)(f)(g) (h)(i)(l)(m)
Classification of SSCs			●	
Design Bases for SSCs Important to Safety			●	●
Design Criteria for Safety Protection Systems	●	●	●	●

A This requirement applies to CoCs and CoC applications through the requirement in 10 CFR 72.236(d).

B Note that while 10 CFR 72.122, "Overall requirements," and 126, "Criteria for radiological protection," are not applicable to an application for a CoC, the CoC applicant should describe how the DSS design facilitates the ability of the licensee to meet these requirements.

The reviewer should verify that the applicant has provided sufficient general or summary discussions of the SSC design features for both operational (including normal operation conditions and anticipated occurrences (that is, off-normal conditions)) and accident conditions, including natural phenomena. This demonstrates a clear and defensible case that the applicants have met the design criteria. For specific license applications, refer to Chapter 2, "Site Characteristics Evaluation," of this Standard Review Plan (SRP) for the specific methods and guidance reviewers should use to identify site characteristics to ensure the DSF design criteria are adequate for the DSF to be built and operated at that site and will meet the 10 CFR Part 72 requirements. For CoC applications, the safety analysis report (SAR) defines a bounding envelop of conditions for normal, off-normal, and accident conditions for which the DSS is designed to fulfill its design functions. A general licensee wishing to use the DSS at its site will need to show, in a 10 CFR 72.212, "Conditions of general license Issued under § 72.210," evaluation report, that its site is bounded by the conditions for which the DSS was analyzed. In evaluating the principal design criteria and bases related to SSCs and safety protection systems, reviewers should seek to ensure that the DSS or DSF design fulfills the design bases and design criteria described below.

3.4.1 Classification of Structures, Systems, and Components

The applicant must identify all SSCs important to safety and provide a rationale for the identification. Acceptance criteria for classification of SSCs important to safety are based on 10 CFR 72.24, “Contents of application: Technical information,” for a specific license review and 10 CFR 72.236, “Specific requirements for spent fuel storage cask approval and fabrication,” for a CoC review.

The structural, thermal, shielding, confinement, radiation protection, criticality, materials, and decommissioning evaluation chapters of this SRP discuss areas of review that also include SSCs important to safety that are identified as safety protection systems. The following sections discuss design bases for SSCs important to safety and safety protection systems.

3.4.2 Design Bases for Structures, Systems, and Components Important to Safety

3.4.2.1 *Spent Nuclear Fuel Specifications*

The applicant should provide information on the SNF to be stored including a complete list of SNF parameters and characteristics. This information includes, but is not limited to, the reactor type (e.g., boiling-water reactor (BWR), pressurized-water reactor (PWR)); fuel manufacturer and model designation and number; fuel physical characteristics; fuel cladding material; thermal and radiological characteristics; and history and census, including burnup, initial enrichment, and cooling time and, for specific licenses, the total amount of SNF to be stored at the DSF. The applicant should also identify if components associated with or integral to fuel assemblies (e.g., rod cluster control assemblies, thimble plug assemblies) would be stored and provide adequate information to characterize these components. These components are also referred to as nonfuel hardware. This information includes component types, quantities, material specifications, and any other properties, including operational specifications (e.g., 10-percent insertion into the reactor core, number of cycles or duration of use in the reactor), that are important to evaluate the components’ effects on or contribution to criticality safety, heat generation, radiological source terms, and structural and confinement performance of the DSS or DSF SSCs and SNF. The applicant must also provide information on the ranges of parameters of the SNF to be stored.

The application should specify the range and types of SNF that the DSS or DSF is designed to store. These specifications should also include, but are not limited to, the following:

- type of SNF (i.e., BWR, PWR, or both)
- cladding material
- maximum assembly uranium mass loading
- bounding composition specifications for mixed-oxide SNF and SNF with thoria (includes masses of uranium, plutonium, thorium; initial enrichments of uranium and plutonium isotopes)
- assembly weights
- dimensions and configurations of the fuel

- identification and limits on amount and position of damaged fuel, if damaged fuel is to be stored, and the dimensions of the "damaged-fuel can"

- maximum allowable enrichment of the fuel before any irradiation for criticality safety and minimum enrichment for the shielding evaluation

- assigned burnup loading value (i.e., in megawatt days per metric ton of uranium or per metric ton heavy metal)

- loading curves for each set of licensing conditions if burnup credit is used (required minimum burnup versus enrichment curve)

- operational history parameters (e.g., in-core soluble boron concentration, moderator temperature) if burnup credit is used

- minimum acceptable cooling time of the SNF before storage in the DSS or DSF

- maximum heat to be dissipated

- maximum number of SNF elements

- condition of the SNF (i.e., intact assembly, damaged fuel, consolidated fuel rods)

- inerting atmosphere requirements and the maximum amount of fuel permitted for storage in the DSS or DSF

For DSSs or DSFs that will be used to store components that are associated with or integral to fuel assemblies (e.g., control rods and BWR fuel channels), the reviewer should ensure that the applicant specifies, along with the already noted parameters, the types and amounts of radionuclides, heat generation, and the relevant source strengths and radiation energy spectra permitted for storage in the DSS or DSF. For these components, the SAR should also specify and evaluate the following:

- the design-basis radiation source term

- the effects of gas generation on the cask internal pressure

- the effects of the additional weight and length of the proposed material on structural and stability analyses

- the impact of the added heat from these components, including the impact on heat transfer characteristics

- credit for any negative reactivity from residual neutron-absorbing material remaining in the control components

3.4.2.2 *Reactor-Related Greater that Class C Waste Specifications* ***(SL)***

Only solid reactor-related GTCC waste may be stored under 10 CFR Part 72, provision for which is made only for specific-license DSFs. Licensees under 10 CFR Part 50 "Domestic Licensing of Production and Utilization Facilities," are already authorized to possess and store reactor-related

GTCC waste under provisions of 10 CFR Part 30, Rules of General Applicability to Domestic Licensing of Byproduct Material," and 10 CFR Part 70, "Domestic Licensing of Special Nuclear Material"; therefore, general licensees store reactor-related GTCC waste under their 10 CFR Part 50 license and not as part of their 10 CFR Part 72 general license. Solid reactor-related GTCC waste is typically activated metals, such as reactor vessel internals, and in-core instrumentation.

There are two general categories of reactor-related GTCC waste:

- Activated metals—These wastes are not integral components of a fuel assembly and include control rod blades, local power range monitor strings, intermediate-range monitor strings, short-range monitor dry tubes, in-core instrument strings, top fuel guide, BWR core shroud, upper core support plate, PWR core shroud (baffle), lower core barrel, lower core support plate, and primary and secondary neutron sources that are not contained within the fuel assembly.

- Process wastes—These are wastes generated from the operation and decommissioning of reactors. These wastes are generated from mechanical filtration operations and can consist of paper, metals and plastics. Process wastes include control rod drive strainers, fuel pool and vacuum filters, PWR miscellaneous cartridge filters, crud tank filters, and ion exchange resins.

For reactor-related GTCC waste, the application should include the following information: waste form (e.g., activated metal, process waste), the maximum quantity of waste to be stored at the ISFSI or MRS, the radionuclide inventory, and the location and configuration of reactor-related GTCC waste containers with respect to the SNF storage casks. Applicants may choose to store reactor-related GTCC waste in containers designed to store SNF or in containers specifically designed to store GTCC waste. In either case, the application should describe the characteristics of the reactor-related GTCC waste containers necessary to demonstrate DSF compliance with the regulations when storing GTCC waste. Liquid GTCC waste may not be stored under 10 CFR Part 72.

3.4.2.3 *High-Level Radioactive Waste Specifications* ***(SL–MRS only)***

The regulations in 10 CFR 72.3, "Definitions," define HLW. The regulations in 10 CFR 72.2(a)(2) identify that only HLW in solid form is acceptable for storage and may only be stored under 10 CFR Part 72 at a DOE-owned MRS. Further conditions regarding the form of this waste are discussed in 10 CFR 72.120(c). Liquid HLW is not acceptable for storage. The applicant should provide information on the waste form, proposed storage package, characteristics of any encapsulation material, radionuclide characteristics, heat generation rate, and history. The SAR should include bounding ranges of parameters of the material to be stored. This information includes quantities, material specifications, and any other properties that are important to evaluate the criticality safety, heat generation, radiological source terms, and structural and confinement performance of the DSF SSCs associated with storage of HLW.

3.4.2.4 *External Conditions*

The SAR should define the bounding conditions under which the DSS or DSF is expected to operate and perform its design functions. The principal design bases should include the following items:

- normal design conditions, including external conditions such as ambient temperature, humidity, and insolation; operational parameters such as maximum load capacity of cranes and handling equipment; and maximum dimensions of the casks or other critical equipment to be handled

- off-normal design conditions, including external conditions such as ambient temperatures and insolation, and operational parameters that do not approach accident conditions

- accident conditions, including external conditions such as tornado wind velocities, tornado missiles, tornado pressure drop, maximum wind velocities, design-basis earthquake, peak explosive overpressure, peak flood elevation, and hypothetical accidents including storage container drop and tipover.

For specific license applications, the SAR only needs to address those conditions that are credible for, applicable to, or both, the DSF site. For CoC applications, the SARs should define the enveloping conditions for normal, off-normal, and accident (including natural phenomena) conditions for which the DSS is designed. The DSS SAR analyses should show that the DSS performs its design functions for these conditions. A general licensee wishing to use the DSS will need to show in a 10 CFR 72.212 evaluation report (which is subject to NRC inspection) that its site is bounded by the conditions evaluated in the DSS SAR.

For unique designs where operations may involve multiple configurations, including temporary configurations, for normal, off-normal and accident conditions, the SAR should include analyses of these conditions for the different possible configurations. For example, in cases were storage array expansion involves removal of material (or exposure of nonstructural material) relied on for shielding, the SAR should include analyses of normal, off-normal, and accident conditions for configurations where shielding material is removed as well as configurations where the shielding material is in place. Unique aspects of storage operations or site characteristics may necessitate evaluation of normal, off-normal, and accident conditions that are not usually considered in most DSS or DSF applications.

3.4.3 Design Criteria for Safety Protection Systems

3.4.3.1 *General*

The maximum certificate term for a DSS is not to exceed 40 years (see 10 CFR 72.230(b)). The maximum license term for a DSF is 40 years from the date of issuance (see 10 CFR 72.42(a)). The applicant should demonstrate that the design will last for the proposed effective certificate or license term, as applicable. The reviewer should verify that the applicant has provided a brief description of the proposed quality assurance program and of applicable industry codes and standards that will be applied to the design, fabrication, construction, and operation of the DSS or DSF. The applicant should also describe how the design considers compatibility with removal from a reactor site, transportation, and ultimate disposition of the stored SNF.

In establishing normal and off-normal conditions applicable to the design criteria for DSS or DSF designs, applicants should account for actual facility operating conditions and configurations. Therefore, design considerations should reflect normal operational ranges, including any seasonal variations or effects and any temporary configuration changes that may occur as part of normal operations.

An aspect of the DSF design criteria and design basis is fire protection. Regulatory Guide (RG) 1.189, “Fire Protection for Nuclear Power Plants,” RG 1.191, “Fire Protection Program for Nuclear Power Plants During Decommissioning and Permanent Shutdown,” and RG 1.205, “Risk-Informed, Performance-Based Fire Protection for Existing Light-Water Nuclear Power Plants,” provide guidance related to fire protection. Chapter 11, “Operation Procedures and Systems Evaluation,” of this SRP provides details on the fire protection review of the proposed DSS or DSF design.

3.4.3.2 *Structural*

The SAR should define how the DSS or DSF structural components are designed to accommodate combined normal, off-normal, and accident loads while preserving recoverability and protecting the DSS or DSF contents, including site-generated wastes for DSFs, from significant structural degradation, criticality, loss of shielding, and loss of confinement. This discussion is generally a summary of the analytical techniques and calculation results from the detailed analysis given in the SAR chapter addressing the structural evaluation, and it should be presented in a nonproprietary form. Chapter 4, “Structural Evaluation,” of this SRP details the acceptance criteria to be considered in the structural design of the proposed DSS or DSF.

RG 1.13, “Spent Fuel Storage Facility Design Basis,” provides general design guidance for SNF storage facilities and specific design guidance for pools at those facilities. RG 1.13 refers to American National Standards Institute/American Nuclear Society (ANSI)/(ANS) standard ANSI N210-1976/ANS-57.2-1983, “Design Objectives for Light Water Reactor Spent Fuel Pool Storage Facilities at Nuclear Power Stations.” RG 1.13 specifically provides guidance for licensees under 10 CFR Part 50, but can be used for those licensed under 10 CFR Part 72. Additional guidance includes the following:

- design bases guidance for tornado protection in RGs 1.76, “Design Basis Tornado and Tornado Missiles for Nuclear Power Plants,” and 1.117, “Protection Against Extreme Wind Events and Missiles for Nuclear Power Plants”

- guidance for flood protection in RG 1.59, “Design Basis Floods for Nuclear Power Plants,” and 1.102, “Flood Protection for Nuclear Power Plants,” guidance for protection against seismic events in RGs 1.29, “Seismic Design Classification,” 1.60, “Design Response Spectra for Seismic Design of Nuclear Power Plants,” 1.61, “Damping Values for Seismic Design of Nuclear Power Plants,” 1.92, “Combing Modal Responses and Spatial Components in Seismic Response Analysis,” 1.122, “Development of Floor Design Response Spectra for Seismic Design of Floor-Supported Equipment or Components,” 1.198, “Procedures and Criteria for Assessing Seismic Soil Liquefaction at Nuclear Power Plant Sites,” and 1.208, “A Performance-Based Approach to Define the Site-Specific Earthquake Ground Motion”

In addition, consider the guidance in ANSI/ANS 57.9, “Design Criteria for an Independent Spent Fuel Storage Installation (Dry Storage Type).”

3.4.3.3 *Thermal*

The SAR should contain a general discussion of the proposed heat-removal systems, including the reliability and testing of such systems, and any associated limitations. All heat-removal systems should be passive and independent of intervening actions under normal and off-normal conditions. Chapter 5, "Thermal Evaluation," of this SRP details the acceptance criteria to be considered in the thermal design of the proposed DSS or DSF.

3.4.3.4 *Shielding, Confinement, Radiation Protection*

The applicant should describe those features of the storage facility that protect occupational workers and members of the public against direct radiation doses and releases of radioactive material and minimize the dose from normal operations and from any off-normal or accident conditions.

The applicant should also identify the design criteria and design bases for the storage facility's shielding, confinement, and radiation protection design, including discussion of any appropriate regulatory guides used for those criteria and bases.

Chapters 6, 9, 10A and 10B, and 13 ("Shielding Evaluation," "Confinement Evaluation," "Radiation Protection Evaluation," "Waste Management Evaluation," respectively) of this SRP detail the acceptance criteria to be considered in the shielding, confinement, radiation protection, and waste management design, respectively, of the proposed DSS or DSF.

3.4.3.5 *Criticality*

The SAR should address the mechanisms and design features that enable the storage facility to maintain SNF, and, as applicable for a specific license DSF, the reactor-related GTCC waste and HLW in a subcritical condition under normal, off-normal, and accident conditions. Chapter 7, "Criticality Evaluation," of this SRP details the acceptance criteria to be considered in the criticality design of the proposed DSS or DSF.

3.4.3.6 *Material Selection*

The materials selected for the DSS or DSF must demonstrate adequate corrosion performance during normal operation, off-normal, and accident conditions in the environmental conditions of the storage facility for the duration of the license for DSFs and the environmental conditions to which the DSS may be exposed (or for which it was intended to be designed) for the duration of the certified period of storage.

The SNF cladding must be protected during storage against degradation that leads to gross ruptures, or the fuel must be otherwise confined such that degradation of the fuel during storage will not pose operational problems with respect to its removal from storage.

Chapter 8, "Materials Evaluation," of this SRP details the acceptance criteria to be considered in the materials selection design of the proposed DSS or DSF.

3.4.3.7 *Decommissioning* ***(SL)***

The NRC outlines the regulatory requirements for decommissioning considerations for specific licenses in 10 CFR 72.130, "Criteria for decommissioning."

DSF SSCs should be designed for ease of decontamination and eventual decommissioning. The SAR should describe the features of the design that support these two activities.

Chapter 14, "Decommissioning Evaluation," of this SRP details the acceptance criteria to be considered in the review of decommissioning proposed for the DSF design.

3.4.3.8 *Retrievability*

The regulation in 10 CFR 72.122(l) states that “storage systems must be designed to allow ready retrieval of spent fuel, high-level radioactive waste, and reactor-related GTCC waste for further processing or disposal.” The NRC interprets this regulation to require that a storage system be designed to allow for ready retrieval in the initial design, amendments to the design, and in license renewal, through the aging management of the design. Retrievability is applicable only during normal and off-normal conditions; it does not apply to accident conditions. The retrievability requirement applies to all general licensed and specific licensed ISFSIs. The requirements in 10 CFR 72.236(m) state that CoC holders should design for retrievability “[t]o the extent practicable in the design of spent fuel storage casks, consideration should be given to compatibility with removal of the stored spent fuel from a reactor site, transportation, and ultimate disposition by the Department of Energy.”

Acceptable means for removing the spent fuel from storage include the ability to perform any of the three options below. These options may be utilized individually or in any combination or sequence, as appropriate.

- Remove individual or canned SNF assemblies from wet or dry storage.
- Remove a canister loaded with SNF assemblies from a storage cask or overpack.
- Remove a cask loaded with SNF assemblies from the storage location.

Applicants for an initial ISFSI license or an ISFSI license amendment must meet the retrievability requirement in 10 CFR 72.122(l). In order to do so, the storage system design should allow for ready retrieval by the use of option A, B, or C or a combination of the options. A dry-storage system may demonstrate retrievability by the use of a known and controlled fuel selection, limits on the loading temperature, a known atmospheric environment, and transfer cask or canister temperature control. The reviewer should also verify that applications for all storage systems identify the SSCs important to safety and the SSC subcomponents that are relied upon for ready retrieval. The reviewer should further verify that the technical specifications included in the application provide for the maintenance of SSCs relied upon for ready retrieval.

When an applicant for an initial ISFSI license or license amendment relies on Option A to demonstrate ready retrieval, the reviewer should confirm that the applicant demonstrates the fuel assemblies will not exhibit gross degradation, and will be removable. Additional review will be needed in the case where there is an assembly with gross degradation or an assembly contains rods with breaches greater than a pinhole leak or a hairline crack (i.e., gross ruptures that could lead to release of fuel particulates). The reviewer should confirm that the applicant demonstrates that the fuel assembly can be placed inside a secondary container. The secondary container must confine the fuel particulate to a known volume and be capable of removal.

When an applicant for an initial dry storage ISFSI license or license amendment relies upon Option A to demonstrate ready retrieval, it is likely the storage cask or canister will, at some point, need to be moved from the storage location to a location where the SNF assemblies can be removed from the cask or canister. When the reviewer anticipates that the cask or canister will have to be moved, the reviewer should confirm the applicant relying upon Option A to demonstrate ready retrieval also demonstrates ready retrieval under Option B or Option C.

When an applicant for an initial ISFSI license or license amendment demonstrates ready retrieval with Option B or Option C, the continued ready retrieval of the storage system should be addressed in its technical specification. However, in addition to the technical specification, an applicant may also propose to implement a program to identify, monitor, and mitigate possible degradation that could impact the intended function of the dry storage system's SSCs and subcomponents of the dry storage system that are relied upon to comply with the retrievability requirements.

When the application is for renewal of an ISFSI license, verify that the 10 CFR 72.122(l) retrievability requirement is met by ensuring that the approved design bases for the item being relied upon in the option(s) chosen (e.g., fuel assembly, cask, or canister) to demonstrate ready retrieval, including any programs implemented, has not been altered. Additionally, the reviewer should verify that aging management programs and time-limited aging analysis associated with renewed licenses provide reasonable assurance that the approved design bases will be maintained during the period of extended operation. This will include reviewing operating experience, including inspections and analyses performed during the initial storage period for ensuring SSCs relied upon for ready retrieval were maintained. The reviewer should refer to NUREG-1927, Revision 1, "Standard Review Plan for Renewal of Specific Licenses and Certificates of Compliance for Dry Storage of Spent Nuclear Fuel," issued June 2016, Agencywide Documents Access and Management System (ADAMS) Accession No. ML16179A148) for additional guidance.

CoC holders and applicants for a CoC are not required by regulation to demonstrate retrievability under 10 CFR 72.122(l); however, 10 CFR 72.236(m), which applies to CoC holders, states that retrievability should be considered to the extent practicable in the design to consider removal of the SNF from storage, transportation, and ultimate disposition. When a CoC applicant for an initial certificate, amendment, or revision chooses to incorporate retrievability aspects, the reviewer should confirm that the retrievability aspects are technically justified and verify that 10 CFR Part 72 requirements affected by retrievability are evaluated and met. This may include the NRC reviewer confirming that the design for the dry storage system includes an evaluation for potential degradation mechanisms for both the storage cask or canister and the SNF to assure that the design of the system has considered removal of the SNF from storage during the storage term. Note that the general licensee must comply with the retrievability requirement in 10 CFR 72.122(l) and should demonstrate that the canister or casks meet the amendment loading requirements.

The SAR does not need to describe specific retrieval facilities, equipment, and procedures for post-accident conditions because of the wide variety of possible post-accident conditions that may occur. The design and procedures for retrieval or recovery (following design basis accident) must be such that the operations can be conducted in compliance with the requirements of 10 CFR Part 20, "Standards for Protection against Radiation."

General regulatory requirements for retrieval capability are given in 10 CFR 72.122(a), (b)(1), (b)(2), (b)(3), (c)(f)(h). Retrievability is specifically outlined in 10 CFR 72.122(l). The applicant must include design criteria and design bases for retrieval.

3.4.4 Design Criteria for Other Structures, Systems, and Components (SL)

Design criteria and bases for other SSCs (i.e., those determined as being not important to safety) should meet the general regulatory requirements in 10 CFR 72.24(a)–(h) and (l) and the appropriate requirements in 10 CFR 72.120, "General Considerations." The applicant must identify design criteria and bases for SSCs determined not important to safety. The design criteria and bases for SSCs that are not important to safety may be adequately defined by statements in the SAR identifying the design codes and standards to be met in design and construction. More extensive definition is typically appropriate for SSCs that interface with, or that could adversely affect, SSCs important to safety. Section 4.3.2 of this SRP includes some examples of the types of SSCs which may fall into this category and are within the scope of NRC review. The application should include a description of the other SSCs which are relevant to the evaluations described in the radiation protection evaluation (see Chapter 10A), including for meeting requirements such as 10 CFR 72.24(e). The descriptions should be sufficiently detailed to support those evaluations and address the relevant regulatory requirements.

3.5 Review Procedures

Figure 3-1 shows the interrelationship between the principal design criteria evaluation and the other areas of review described in this SRP.

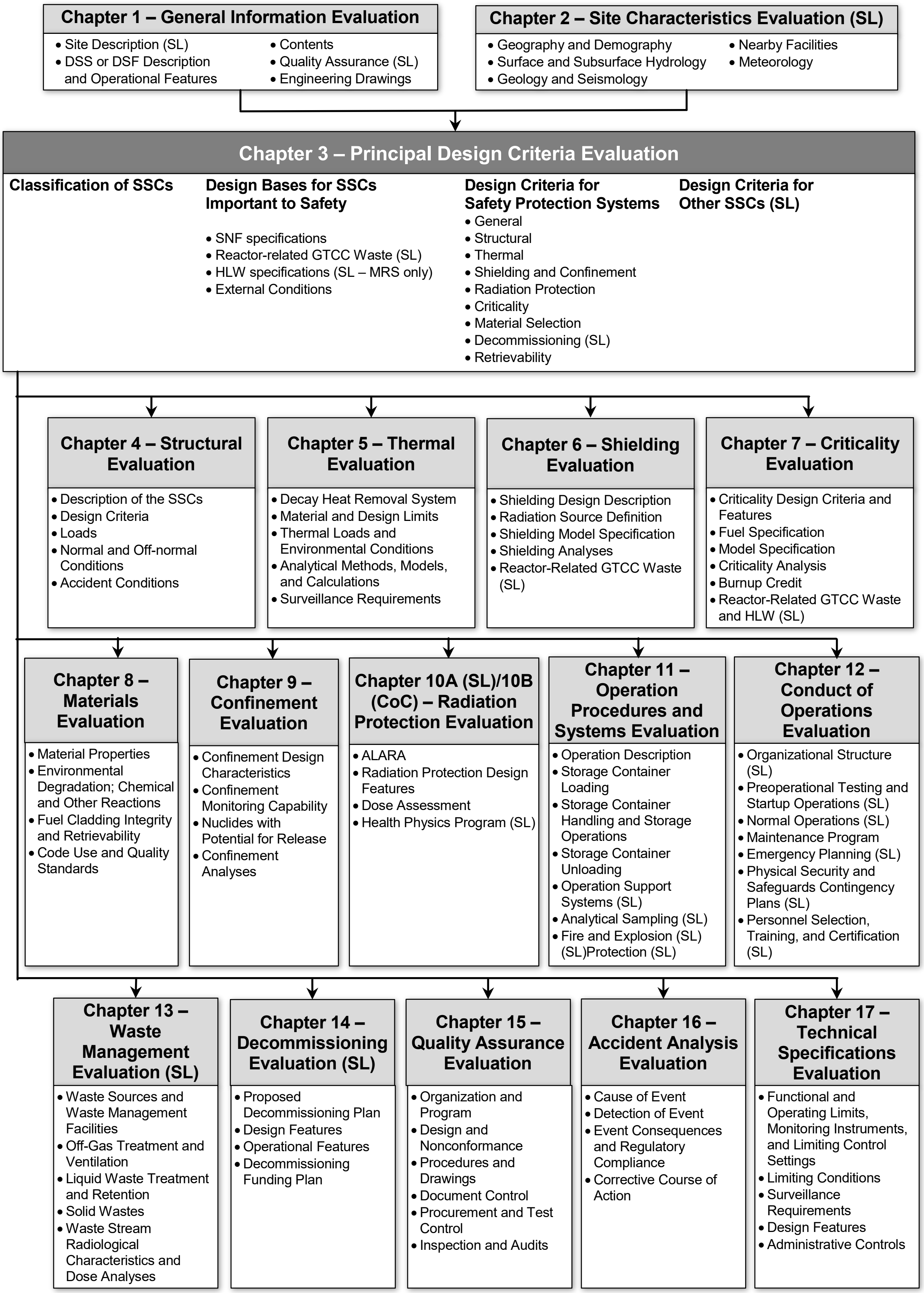

Figure 3-1 Overview of Principal Design Criteria Evaluation

Reviewers of each chapter of the SAR should consider the chapter on SSCs and principal design criteria in combination with additional details presented in their respective chapters. Evaluations of design criteria applicable to each of the relevant chapters of the SAR are discussed in detail in the respective SRP chapters. Reviewers should coordinate the review of each chapter with the applicable disciplines to ensure that multidisciplinary issues that impact more than one chapter have been addressed.

A DSF application may involve use of one or more DSSs certified under 10 CFR Part 72, Subpart L, "Approval of Spent Fuel Storage Casks," including the SARs for the certified DSSs by reference. The application should provide additional information relating to the DSSs, including the applicant's evaluations that establish that the site parameter limits are within the bounds of those established as limiting conditions as set forth in the referenced CoCs.

3.5.1 Classification of Structures, Systems, and Components

Although not an exhaustive list, determine if the application includes any of the SSCs and functions listed below that typically are considered important to safety. Determine if the application includes (or should include) other SSCs or functions that may be considered important to safety, based on the design of the DSF/DSS:

- components of the confinement boundary and integral components and structures used within the confinement boundary of the storage containers
- SSCs providing criticality control (e.g., SNF basket, neutron absorbers)
- radiation shielding
- SSCs providing capabilities for lifting, handling, and transferring SNF, reactor-related GTCC waste, or HLW, as applicable
- instrumentation and controls SSCs if they are used as the primary means for real-time recognition of off-normal conditions or accident conditions
- SSCs providing either active or passive decay heat removal
- the confinement systems to prevent the release of radioactive liquid wastes generated from site operations **(SL)**
- SSCs for retaining radioactive material within the pool building, if applicable **(SL)**

The radiation shielding includes any engineered features, such as berms or shield walls, that are used to ensure compliance with 10 CFR 72.104(a) or 10 CFR 72.106(b).

3.5.2 Design Bases for Structures, Systems, and Components Important to Safety

Verify that the types of materials to be stored comply with 10 CFR 72.120(b) or 10 CFR 72.120(c). Confirm that the SAR gives SNF, reactor-related GTCC waste, or HLW acceptance specifications, as applicable, including upper- or lower-bound limits, as appropriate, of acceptable variability. Verify that appropriate specifications are incorporated into the technical specifications for the DSS or DSF. For DSF applications, confirm that the SAR gives the criteria for procedures for testing, inspecting, and verifying wastes received for storage at the facility. Verify that the SAR defines criteria for procedures for handling; repackaging (if needed); and shipping out-of-specification wastes. For DSS applications, confirm that the SAR describes procedures for identifying and verifying that SNF and any nonfuel hardware to be loaded and stored in the DSS meet the specifications for allowable DSS contents.

3.5.2.1 *Spent Nuclear Fuel*

Review the detailed specifications for the SNF to be stored in the DSS as presented in the chapter of the SAR on principal design criteria, and ensure that the specifications are consistent with those discussed in the chapter of the SAR on general information and other locations. The descriptions of the SNF and components associated with the fuel assemblies (that is, nonfuel hardware) to be stored should include the information described in Section 3.4.2.1 of this SRP.

Examine any limitations regarding the condition of the SNF. If damaged fuel is allowed, the effects of such damage should be assessed in later sections of the SAR. Section 8.5.15.1 of this SRP provides specific conditions that define damaged fuel and identifies methods for classifying fuel. If damaged rods have been removed from a fuel assembly and they have or have not been replaced with solid dummy rods, the criticality reviewer should determine whether the intended loading configuration has been adequately analyzed to show subcriticality. The presence of additional moderating material will need to be addressed in the criticality analysis in the SAR. Coordinate the review with the structural reviewer if there are structural defects in the assembly hardware.

The release of fill and fission product gases from failed fuel rods increases the pressure in the cask cavity and the potential source term in the event of confinement failure. Verify that the application provides information regarding the fill or fission product gas present in the fuel as well as the free volume in the cask cavity to enable an evaluation of the pressure in the cask cavity resulting from cladding failure during storage. For the purpose of calculating internal cask pressures, the NRC staff has accepted the bounding assumptions presented in Section 5.5.4.6, Pressure Analysis," of this SRP on pressure analysis, as regards the minimum percentages of fuel rods that have failed (and released their gases).

Pay particular attention to the specification of burnup, cooling time, and decay-heat generation rate. These parameters are generally not independent, and the manner in which they are specified and combined can significantly affect the maximum allowed cladding temperature as discussed in Chapter 5 of this SRP.

The SARs typically list various fuel assemblies that can be stored in the DSS or at the DSF. It is not expected that one type of fuel assembly will bound all analyses. Ensure that the application justifies which specifications are bounding for each of the evaluations presented in subsequent sections of the SAR. Ensure that the SAR chapter on technical specifications and operational controls and limits clearly identifies or references the specifications used in the analyses.

If the applicant requests permission for the storage of components that are associated with or integral to the fuel assembly in the DSS or DSF storage container, examine the relevant detailed specifications, conditions, and constraints presented in the SAR. These specifications should be as detailed as the applicable information presented for the fuel designs to provide the reviewer with a basis for determining that the relevant safety functions of the DSS or DSF SSCs will be maintained. Ensure that the applicant also considers the storage of these components in the analyses.

If the applicant requests burnup credit, examine the relevant detailed specifications of the contents to which burnup credit is being applied. These specifications include those that are already considered in criticality analyses for fresh fuel (e.g., maximum initial enrichment). Additional specifications that must be reviewed include the cooling time, the burnup, the requested amount of credit (i.e., the specific actinides), and operational history parameters (e.g., core average boron concentration and assembly average moderator temperature).

3.5.2.2 *Reactor-Related GTCC Waste* ***(SL)***

Ensure that the reactor-related GTCC waste is appropriately characterized so that the reviewer has reasonable assurance that storage is in compliance with the regulations. For reactor-related GTCC waste, the applicant should provide the waste form (e.g., activated metal, process waste), the maximum quantity of waste to be stored at the ISFSI or MRS, the radionuclide inventory, and the location and configuration of reactor-related GTCC waste containers with respect to the SNF storage casks. Verify that the reactor-related GTCC waste form is solidified and that there are no liquids present in the container. The applicant should describe the means by which this verification will be done and justify that the means are sufficient to ensure that received materials meet the license requirements for storage at the facility.

Applicants may choose to store reactor-related GTCC waste in containers designed to store SNF or in containers specifically designed to store reactor-related GTCC waste. In either case, ensure that the application describes the characteristics of the reactor-related GTCC waste containers necessary to demonstrate DSF compliance with the regulations when storing reactor-related GTCC waste. Verify that the applicant has evaluated the impact(s) associated with the safe colocation of reactor-related GTCC waste and SNF at an ISFSI or MRS under normal, off-normal, and accident conditions.

3.5.2.3 *High-Level Radioactive Waste* ***(SL–MRS only)***

Determine that the HLW is appropriately characterized so that the necessary design and analytical calculations and acceptance tests may be carried out. For HLW, such characteristics include waste form, decay heat, inventory of radionuclides, and the characteristics described in Section 3.4.2.3 of this SRP.

Ensure that the waste form is solid and not liquid. If the waste form contains liquid, as in undried filter residues, the NRC staff must establish waste acceptance specifications and bounding limits of acceptability.

3.5.2.4 *External Conditions*

Verify that the SAR identifies those external conditions that significantly affect, or could potentially affect, the performance of the DSS or DSF. For a DSS, these design-basis conditions will generally restrict either the sites at which the DSS can be used for SNF storage or the manner in which the DSS can be handled. For example, by selecting the design earthquake, the SAR limits the use of the DSS being reviewed to sites that are bounded by this seismic limit. For a DSF, these design-basis conditions should be based on, or include conditions that are based on, the characteristics of the site at which the DSF will be built and operated. By establishing a design-basis drop, the SAR defines the maximum height to which a DSS or DSF storage container can be lifted without additional safety analysis or design changes (e.g., addition of impact limiters) by the applicant.

Note that movement of DSS or storage container components within a reactor building may not meet the NRC's criteria described in the NRC Bulletin 96-02, "Movement of Heavy Loads over Spent Fuel, over Fuel in the Reactor Core, or over Safety Related Equipment," dated April 11, 1996, for movement of heavy loads within the reactor building. As such, if a potential DSS user (licensee) has been identified or the DSF is co-located with a 10 CFR Part 50 or 10 CFR Part 52 licensee and involves (storage container handling) operations in a building or with SSCs licensed as part of the 10 CFR Part 50 or Part 52 facility, the reviewer should coordinate with the appropriate project manager or technical lead from the NRC's Office of Nuclear Reactor Regulation (NRR) during the early stages of the review.

At a minimum, the NRC staff has generally addressed the conditions discussed below; however, other conditions may be relevant depending on specific details of the DSS or DSF design. Pay particular attention to special design features and how these might be affected by other external conditions and other components of the DSS or DSF. Ensure that the SAR provides all required information for the design earthquake accident analysis.

"Normal" conditions (including conditions involving handling and transfer) and the extreme ranges of normal conditions are presumed to exist during design-basis accidents or design-basis natural phenomena, with the exception of irrational or readily avoidable combinations. For example, an earthquake or tornado may occur at any time and in combination with any "normal" condition. By contrast, it can be presumed that transfer, loading, and unloading operations would not be conducted during a flood.

"Off-normal" conditions and events are presumed to occur in combination with normal conditions that are not mutually exclusive. Nonetheless, the SAR is not required to analyze nor must the DSS or DSF be designed for the simultaneous occurrence of independent off-normal conditions or events, design-basis accidents, or design-basis natural phenomena.

Conditions involving a "latent" equipment or instrument failure or malfunction (that is, one that occurs and remains undetected) should be presumed to exist concurrently with other off-normal or design-basis accident conditions and events. Typical latent malfunctions include a misreading instrument that is not detected as part of routine procedures, an undetected ventilation blockage, or undetected damage from an earlier design-basis off-normal or accident event or condition if no provisions exist for detection, recovery, or remediation of such conditions.

For normal, off-normal, and accident conditions, verify that the application defines appropriate operating and accident scenarios. For these scenarios, verify that the SAR includes a comprehensive evaluation of the effects of such scenarios on the SSCs important to safety. The individual chapters of this SRP address the analyses of such events. For example, Chapter 4 addresses the analyses of an earthquake on the structural components of the DSS or DSF. Verify that the applicant's evaluations demonstrate that the requirements in 10 CFR 72.104, "Criteria for radioactive materials in effluents and direct radiation from an ISFSI or MRS," 10 CFR 72.106, "Controlled area of an ISFSI or MRS," and 10 CFR Part 20 are or will be met for DSFs and can be met for DSSs. While the requirements in 10 CFR Part 20 do not apply to DSSs, they may be useful in informing the reviews of DSS applications.

Verify that the scenarios and evaluations address all relevant configurations of the SSCs of the DSS or DSF. For example, a storage design may rely on nonstructural materials that, for that design, may be removed, exposed, or otherwise disturbed during normal, though temporary, operations such as activities to expand the existing array of storage containers. For such designs, evaluate impacts of normal, off-normal, and accident conditions for these temporary configurations as well as the long-term design configurations. Ensure that evaluations of external conditions address any conditions or events that may be unique to the design at the different stages of DSS or DSF operations. Also, for DSFs, ensure that the evaluations address SSCs in addition to the storage containers (e.g., SSCs for waste management), as applicable, and unique site characteristics and features.

If appropriate, verify that the SAR chapter on technical specifications and operational controls and limits evaluation includes the following design bases as operating controls and limits:

3.5.2.4.1 Normal Conditions

For a given SNF specification, the primary external conditions that affect DSS or DSF performance are the ambient temperatures, insolation, and the operational environment experienced by the DSS or DSF.

Ensure that the maximum and minimum "normal" temperatures are the highest and lowest ambient temperatures recorded in each year, averaged over the years of record. For a CoC SAR, the applicant may select any design-basis temperatures as long as any operational restrictions imposed are acceptable to both the applicant and the NRC. If the storage container is also designed for transportation, the temperature requirements in 10 CFR Part 71, "Packaging and Transportation of Radioactive Material," could determine the design-basis temperatures for storage. For a specific license SAR, the NRC accepts as the maximum and minimum normal temperature the highest and lowest recorded for the hottest or coldest month of each year, averaged over the years of record.

For storage containers, the NRC staff accepts a treatment of insolation similar to that prescribed in 10 CFR 71.71, "Normal conditions of transport," for transportation packages. If the applicant selects another design approach, it should justify the alternative approach in the SAR.

The operational environment experienced by the DSS or DSF under normal conditions includes the manner in which the DSS or DSF storage container is loaded, unloaded, and lifted. Occupational dose rates will, in part, depend on whether the DSS or DSF storage container is sealed in a wet or a dry environment. Fuel cladding temperatures may also be affected by these conditions. The manner in which the DSS or DSF storage container is lifted will determine the load on the trunnions, the lifting yoke, or both. The orientation of the DSS or DSF storage container (vertical or horizontal) and its height above ground during transport to the storage pad will establish initial conditions for the drop accidents discussed below.

NUREG-2174, "Impact of Variation in Environmental Conditions on the Thermal Performance of Dry Storage Casks, Final Report," issued March 2016, provides further guidance for reviewing the thermal impact and environmental conditions (e.g., ambient temperature, wind) on a DSS or DSF storage container.

3.5.2.4.2 Off-Normal Conditions

Ensure the SAR addresses several off-normal conditions, such as variations in temperatures beyond normal, failure of 10 percent of the fuel rods combined with off-normal temperatures, partial blockage of air vents, human error, out-of-tolerance equipment performance, equipment failure, and instrumentation failure or faulty calibration. Ensure that the SAR addresses retrievability of the stored SNF, reactor-related GTCC, and HLW, as applicable for the application, for these conditions.

3.5.2.4.3 Accident Conditions

The staff has generally considered that the SAR evaluates the accidents listed in this section. These do not constitute the only accidents that should be addressed if the SAR is to serve as a reference for accidents for a specific application. Other credible accidents that may be derived from a hazard analysis could include accidents resulting from operational error, instrument failure, lightning, and other occurrences. The regulations in 10 CFR 72.122 and 10 CFR 72.236 require that the storage container be designed to withstand the effects of accident conditions and natural phenomena events without impairing its capability to perform safety functions. Consequently, in the analyses for conditions resulting from design-basis accidents and natural phenomena, the NRC has asserted and the applicant should assume a release of 100 percent of the initial rod fill gases and a release of 30 percent of the fission product gases from the fuel rods into the storage container interior. The remaining 70 percent of the fission product gases is presumed to be retained within the fuel pellet. In coordination with the confinement reviewer, verify that the storage container is designed to provide the confinement safety function under all credible conditions.

Postaccident recovery of damaged fuel may require such systems as overpacks or dry-transfer systems since ready retrieval of the fuel is required only for normal and off-normal conditions. Ensure that the SAR identifies and justifies accident situations that are not credible because of design features or other reasons. Chapter 16, "Accident Analysis Evaluation," and the technical chapters of this SRP provide more detail regarding accidents.

Storage Container Drop

Verify that the SAR identifies the operating environment experienced by the storage container as well as the drop events (i.e., end, side, corner) that could result. Generally, the design basis is established either in terms of the maximum height to which the storage container may be lifted when handled by equipment not meeting the single-failure proof criteria or in terms of the maximum acceleration that the storage container could experience in a drop.

Cask Tipover

Although cask system supporting structures may be identified and constructed as important to safety (i.e., designed to prevent cask tipovers), ensure that the applicant analyzes cask tipover events. In some cases, cask tipover may be determined to be a credible hazard, and the associated analysis should reflect the conditions (e.g., heights and accelerations) associated with that hazard.

Fire

Ensure that the fire conditions postulated in the SAR provide an "envelope" for subsequent comparison with site-specific conditions for DSS applications. For DSF applications, ensure that the postulated fire conditions in the SAR are based on the site characteristics, including facility design and layout, that are described in the DSF application that may affect the fire conditions that are credible at the DSF. The NRC accepts the methods discussed in 10 CFR 71.73(c)(4). In addition, the NRC staff accepts that the availability of flammable material at a DSF may be limited such that the applicant may consider only materials such as those that are associated with vehicles transporting or lifting the storage containers or sources of nearby combustible materials. Regardless of which approach the applicant takes, the SAR should specify and justify the bounding conditions for a "design-basis" fire.

Explosive Overpressure

The conditions under which the SCCs for a DSS or DSF may be exposed to the effects of an explosion vary greatly among individual sites. Generally, explosive overpressure is postulated to originate from an industrial accident. Consequently, this SRP does not consider explosive overpressures from sabotage events.

For DSS applications, the extent to which the SAR addresses explosive overpressure directly affects the degree of site-specific review required of a general licensee to meet the requirements in 10 CFR 72.212. For DSF applications, the extent to which the SAR addresses these events should be commensurate with the site characteristics and facility design features of the DSF. The principal concern in the SAR should be the effects of explosive overpressure on the storage system and containers and, for DSFs, other important SSCs rather than descriptions of hypothesized causes. Though, for DSF applications, facility design and site characteristic information will enable the identification of possible sources of these events and the bases for estimates of the events' design parameters. Verify that the design parameters for blast or explosive overpressures identify pressure levels as reflected ("side-on") overpressure and provide an appropriate (assumed, for DSSs) pulse length and shape. For DSS applications, ensure this discussion provides sufficient information for general licensees to determine in their 10 CFR 72.212 evaluations if the effects of their site-specific hazards are bounded by the DSS design bases.

Air Flow Blockage

For storage designs with internal air flow passages, verify that the application considers blockage of air inlets and outlets in an accident condition. The NRC staff considers that the effects of such an assumption should be used in determining the appropriate inspection intervals or monitoring systems, or both, for the DSS or DSF storage containers.

3.5.2.4.4 Natural Phenomena Events

The NRC staff has generally considered that the SAR should evaluate the following events as design-basis accidents:

Flood

Ensure that the SAR establishes a design-basis flood condition. For a specific license application, verify that the design-basis flood condition is based on the site flood parameters. For a CoC application, this condition may be determined on the basis of the presumption that the DSS cannot tip over and the yield strength of the DSS will not be exceeded. Alternatively, the SAR can show that credible flooding conditions have negligible impact on the DSS design.

If the SAR establishes parameters for a design-basis flood, ensure that it recognizes all of the potential effects of flood water and ravine flood byproducts. Serious flood consequences can involve effects such as blockage of ventilation ports by water and silting of air passages. Other potential effects include scouring below foundations and severe temperature gradients resulting from rapid cooling from immersion.

Tornado

The NRC staff accepts design-basis tornado wind loading as defined by RG 1.76 and RG 1.117. Ensure that the application includes design criteria for the DSS or DSF on the basis of these wind-loading and missile-impact definitions. The DSS or DSF storage container should not tip over, and the capability to perform the confinement safety function should not be impaired. The NRC staff considers that tornados and tornado missiles may occur without warning.

Earthquake

Ensure that the SAR states the parameters of the design earthquake. For use of a DSS at reactor sites, this is equivalent to the safe-shutdown earthquake used for analysis of nuclear facilities under 10 CFR Part 50. An analysis for an operating-basis earthquake is not required for a DSS SAR prepared in accordance with 10 CFR Part 72, Subpart L. While the SAR analyzes tipover accidents, tipover caused by an earthquake may not be a credible event. Verify that the SSCs meet appropriate guidance in RG 1.29, RG 1.61, and RG 1.92.

Burial Under Debris

Debris resulting from natural phenomena or accidents that may affect storage container performance may be addressed in the SAR or left to the general licensee's site-specific 10 CFR 72.212 evaluation for DSS applications. Ensure the SAR for a DSF specific license application addresses this scenario. Such debris can result from floods, wind storms, or landslides. The principal effect typically is on thermal performance.

Lightning

Lightning typically has a negligible effect on DSFs or DSSs; however, the design of the DSF or DSS structures should adhere to the requirements of National Fire Protection Association 780, "Lightning Protection Code," and National Fire Protection Association 70, "National Electrical Code." Ensure that the applicant cites these codes as part of the general design criteria for the DSF or DSS (see Section 3.4.3.1 of this SRP). In addition, verify that the SAR addresses lightning as a natural phenomenon if DSF or DSS performance may be impacted by the effect of lightning on an SSC.

Other

The regulations in 10 CFR Part 72 identify several other natural phenomena events (including seiche, tsunami, and hurricane) that should be addressed for SNF storage. The DSS SAR may include these natural phenomena as design-basis events or show that their effects are bounded by other events. If these events are not addressed in the SAR and they prove to be applicable to a specific site, a safety analysis is required before approval for use of the DSS under a general license. Ensure that the DSF SAR addresses these other natural phenomena and their effects on the DSF's SSCs or justify why they are not applicable for the DSF site.

3.5.3 Design Bases for Safety Protection Systems

SCCs for the DSS or DSF that are to be used in facility areas subject to review under 10 CFR Part 50 should satisfy the requirements in 10 CFR Part 72 (with review guided by this SRP) and 10 CFR Part 50 (with review guided by NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LRW Edition," issued March 2007). If the application states that the DSS or DSF will be located at a specific reactor site, then the DSS or DSF project manager should inform the appropriate NRR project manager. Note that heavy loads are likely a matter of interest to NRR.

Use Table 3-2 during the initial stages of the review for both DSS and DSF applications to ensure the SAR identifies the listed design criteria (and design bases). The table also includes or identifies, as applicable, additional information that is only relevant to a DSF specific license review. The applicability of Table 3-2 may vary depending on the details of the DSS or DSF.

Table 3-2 Outline of Design Criterial and Bases

Applicability	Scope for Certificate of Compliance and Specific License Reviews	Additional Scope for Specific License Review
Design Life	Limited to the requested term in the application, not to exceed the applicable limit in either 10 CFR 72.42(a) or 10 CFR 72.230(b)	
Design Bases	SNF Specifications: (1) Assembly type(s) (2) Configuration and vendor (3) Enrichment (maximum and minimum) (4) Weight or range of weights of assemblies (5) Burnup (6) Type of cladding (7) Assemblies or cask (8) Dimensions (9) Uranium or heavy metal mass loading per assembly (10) Thoria amount or plutonium isotopic compositions for SNF with thoria and mixed-oxide (MOX) SNF, along with amount and enrichment of uranium Decay Heat Assembly: (1) Minimum decay or cooling time (e.g., 5 years, 10 years) (2) Maximum kilowatts per assembly (3) Heat-load pattern Gas Volume (at temperature) Fuel Condition or Damage Allowed Burnup Credit: (1) Credit amount (burnup and specific nuclides) (2) Operational history parameters Non-Fuel Hardware	Specifications of radioactive material to be stored (including HLW and reactor-related GTCC waste, as applicable) as described in the appropriate section of this SRP chapter. Maximum total quantities of SNF, reactor-related GTCC waste, and HLW to be stored at the facility

Applicability	Scope for Certificate of Compliance and Specific License Reviews	Additional Scope for Specific License Review
Normal Design Event Conditions	Ambient Temperature: (1) Maximum (2) Minimum Loading: (1) Wet or dry Storage and Handling (e.g., loading, transfer) Orientation: (1) Vertical or Horizontal Maximum Lift Height Maximum Cladding Temperature 1% Fuel Rod Rupture Solar Insolation Other Relevant Operational Environment Conditions (see SRP Section 3.5.2.4.1)	Other conditions or events relevant to operations of the DSF facility as described in the appropriate section of this SRP chapter. Storage containers and other DSF SSCs (e.g., waste management SSCs)
Off-Normal Design Event Conditions	• Temperature Variation Beyond Normal • 10% Fuel Rod Failure Combined with Off-Normal Temperatures • Failure of One of the Confinement Boundaries • Partial Air Flow Blockage • Human Error • Out-of-Tolerance Equipment Performance • Equipment Failure • Instrumentation Failure • Faulty Instrumentation Calibration Other Events Relevant to the Design and Operations Summarize Events Considered under External Conditions (see SRP Section 3.5.2.4.2)	Other conditions or events relevant to operations of the DSF facility as described in the appropriate section of this SRP chapter. Storage containers and other DSF SSCs (e.g., waste management SSCs)

Applicability	Scope for Certificate of Compliance and Specific License Reviews	Additional Scope for Specific License Review
Design-Basis Accident Design Events and Conditions	End Drop: (1) Lift height (or maximum acceleration) Side Drop: (1) Lift Height (or Maximum Acceleration) Tipover: (1) Acceleration (if applicable) Fire: (1) Duration (2) Temperature Complete Air Flow Blockage Explosive Overpressure Other Events Relevant to the Design and Operations (see SRP Section 3.5.2.4.3), as applicable	Other conditions or events relevant to operations of the DSF facility as described in the appropriate section of this SRP chapter. Storage containers and other DSF SSCs (e.g., waste management SSCs)
Design-Basis Natural Phenomena Design Events and Conditions	Flood Earthquake Tornado Burial Under Debris Lightning Other potentially relevant events identified in 10 CFR Part 72 (see SRP Section 3.5.2.5), as applicable	Other conditions or events relevant to operations of the DSF facility as described in the appropriate section of this SRP chapter. Storage containers and other DSF SSCs (e.g., waste management SSCs)
Structural	Design Code (e.g., ASME, AISC): (1) Containment (2) Noncontainment (3) Basket (4) Trunnions (5) Storage radiation and protective shielding and enclosure (6) Transfer radiation and protective shielding and enclosure (7) Cooling structure or system Design Weight Design Cavity Pressure: (1) Normal, off-normal, accident Response and Degradation Limits: (1) Normal, off-normal, or accident	Design Code: • Other SSCs important to safety and SSCs that affect SSCs important to safety • Radiation and protective shielding • Waste management facility SSCs important to safety • Reinforced concrete

Applicability	Scope for Certificate of Compliance and Specific License Reviews	Additional Scope for Specific License Review
Thermal	Maximum Design Temperatures: (1) Cladding (2) Other components Insolation (side, top, or bottom) Fill Gas: (1) Type (e.g., helium) (2) Initial fill pressure (at temperature) Modes of Heat Transfer Used in the Design	Maximum Design Temperatures: • Reinforced concrete • Maximum temperature gradients for structures subject to thermal stress Maximum stored materials decay heat load
Confinement	Description of Confinement Boundary Redundant Seals for Closure Maximum Leak Rate for Confinement Boundary: (1) Normal, off-normal, or accident (2) Justification of leakage rate (if not leaktight) Monitoring System Specifications	
Waste Management (SL)		Description of confinement of site-generated wastes and ventilation and treatment systems
Radiation Protection and Shielding	Storage Container: (1) Surface position (normal, off-normal, or accident) Exterior of Shielding: (1) Transfer configuration position (2) Storage configuration position (normal, off-normal, or accident) Controlled Area Boundary: (1) Dose rate (2) Annual dose (normal or off-normal) (3) Accident Dose Occupational Dose Estimates ALARA Considerations (public and occupational) in Design and Operations	DSF SSCs in addition to the storage containers ALARA policies and programs Health Physics and Radiation Protection programs Radiological Environmental Monitoring Program
Criticality	Method of Control: Geometry, Fixed Poison, Soluble Poison Minimum Boron Concentration: Fixed and Soluble Poison Maximum K_{eff} Burnable Neutron Absorber Credit Burnup Credit Analysis	

Applicability	Scope for Certificate of Compliance and Specific License Reviews	Additional Scope for Specific License Review
Decommissioning		Design for decontamination and decommissioning
Materials	Cladding Hoop Stress Corrosion	
Operating Procedures	Normal and Off-Normal Post-Accident and Natural Phenomenon Event	
Acceptance Tests and Maintenance	Industry Codes and Standards	
Technical Specifications	Operational Controls and Limits	

3.5.3.1 *General*

Check the SAR chapter on design criteria and ensure that the descriptions are consistent with the descriptions in the sections of the SAR that address confinement, cooling, subcriticality, radiation protection, decommissioning, retrieval capability, and ALARA considerations. Verify that the SAR identifies and evaluates the design criteria and bases for the system as a whole.

Determine that the criteria derived from the site characteristics **(SL)** and generic site characteristics **(CoC)** and accident analyses (accident and off-normal conditions) are consistent with the analyses used in the qualification of the SSCs. For DSFs, verify that these criteria are equivalent to those proposed in site characteristics chapter of the SAR.

Confirm that the applicant's general design criteria reflect consideration of ALARA as applicable and appropriate. For specific license applications, the criteria should reflect any stated applicant ALARA goals and policies.

Verify that criteria defining the response of SSCs to normal, off-normal, and accident conditions are satisfactory.

Determine that the application presents design criteria for normal conditions and operations that do not result in or allow any degradation of the capabilities of the DSS or DSF. Ensure that the SAR sufficiently describes routine maintenance that would correct any "wear and tear" from normal conditions and operations that would degrade the capabilities of the DSS or DSF.

(SL) Determine that the application presents design criteria for off-normal conditions that do not permit any degradation of the capabilities of the DSF, assuming contingency operations during and following off-normal conditions. The NRC does not require that radioactive material handling or waste processing functions or capabilities at a storage facility continue during an off-normal condition or that such operations resume immediately. The licensee may impose inspections and system checkouts following any event or condition.

Determine that the application presents design criteria for accident conditions that do not permit the degradation of SSCs important to safety, including, but not limited to, (1) reduced radioactive material handling and waste processing capability **(SL)**, (2) reduced capability to withstand further accident conditions without excess response and without remedial action, and (3) reduced ability to provide functions for the full system or facility life time without remedial action. Determine that design criteria for accident conditions prevent (1) criticality, (2) unacceptable releases of radioactive material, (3) unacceptable radiation doses for the public and workers, and (4) loss of retrieval capability.

The NRC does not require the assumption of multiple failure scenarios of SSCs important to safety unless these multiple failure scenarios are credible consequences of the initiating event.

The NRC requires analysis or testing of SSCs for some events (e.g., cask drop or tipover) even though the events may be determined as noncredible in the accident analysis. Verify that the application presents criteria for the survival of SSCs important to safety for these “nonmechanistic” events as the same as the criteria for the survival of credible accidents.

3.5.3.2 *Other Safety Protection Systems*

Review procedures for the evaluations of design criteria for other safety protection systems applicable to each of the relevant chapters of the SAR are discussed in detail in the respective chapters of this SRP. Coordinate the review of each chapter with the applicable disciplines to ensure that multidisciplinary issues that impact more than one chapter are addressed.

Regardless of where the descriptions and associated criteria are located in the SAR, include a description and evaluation of the safety protection systems in the chapter of the safety evaluation report on principal design criteria. The system descriptions should address the functions of the various system components in providing confinement, cooling, subcriticality, radiation protection of the public and workers, and SNF retrievability. Also, ensure the SAR describes summary criteria for the performance of the system as a whole in providing for these capabilities or functions. Verify that the design-basis assumptions presented in the SAR are consistent with and reasonable for actual site and facility conditions. Include a description and evaluation of the DSS or DSF storage container(s) design’s compatibility with removal from a reactor site or from the DSF, transportation, and ultimate disposition of the stored SNF.

Verify that the SAR describes and evaluates criteria relating to redundancy and allowable levels of response by the DSS or DSF SSCs under normal, off-normal, and accident conditions and events. In general, no unacceptable degradation in physical condition or functional performance should result from normal or off-normal conditions. Verify that the design criteria regarding limits of permissible response and degradation resulting from an accident condition are evaluated against SSC capabilities to perform the principal safety functions. Considerations of permissible responses should include detectability and corrective actions that may be proposed as conditions of system use.

The NRC staff accepts that both routine surveillance programs and active instrumentation meet the intent of “continuous monitoring” as required in 10 CFR 72.122(h)(4).

Note that some DSS or DSF designs may contain a component or feature for which continued performance over the license or certified storage period has not been demonstrated to the staff with a sufficient level of confidence (e.g., rubber “O” rings). Therefore, the NRC may require the use of active instrumentation if the failure of that system or component causes an immediate threat to the public health and safety and if that failure would not be detected by any other means. In some cases, to demonstrate compliance with 10 CFR 72.122(h)(4), the applicant or the NRC may propose a technical specification requiring such instrumentation as part of the first use of a DSS for a CoC application, or as part of operations of the DSF for a specific license application. For DSSs, after first use, and if warranted and approved by the NRC, such instrumentation may be discontinued or modified.

Verify that the applicant has met the intent of continuous monitoring so that the applicant can determine when corrective action needs to be taken to maintain safe storage conditions.

3.5.4 Design Criteria for Other Structures, Systems, and Components (SL)

Verify that the design bases and criteria for other SSCs not important to safety meet the general regulatory requirements in 10 CFR 72.24(a)–(h) and (l) and 10 CFR 72.120.

Typical concerns for general design criteria reviews of other SSCs not important to safety include, but are not limited to, adequate functional performance, interfacing with other SSCs, potentially adverse interactive effects, and recognition of appropriate site characteristics. Confirm with the other reviewers that the application includes descriptions of the other SSCs that are relevant to their review of the facility and that the descriptions are sufficient to enable evaluation of facility compliance (design and operations) with the relevant regulatory requirements. In determining the SSCs and level of detail needed for the review, consider the descriptions provided in the final safety analysis reports for facilities that the NRC has previously licensed, as appropriate. For example, for a facility that is not co-located with a 10 CFR Part 50 or 52 facility, a review of the information in the final safety analysis reports for similar previously licensed facilities would provide useful insights.

3.6 <u>Evaluation Findings</u>

The NRC reviewer should prepare evaluation findings on satisfaction of the regulatory requirements in Section 3.4 of this SRP. If the documentation submitted with the application fully supports positive findings for each of the regulatory requirements, the statements of findings should be similar to the following:

F3.1 The SSCs have been classified as important to safety or not important to safety and meet the requirements given in 10 CFR 72.24(b) for specific licenses and 10 CFR 72.236 for CoCs.

F3.2 The SAR and docketed materials adequately identify and characterize the SNF to be stored in a DSS or DSF, reactor-related GTCC waste to be stored at a specific license DSF, high-level radioactive waste to be stored at a specific license MRS, as applicable. The acceptable form of the reactor-related GTCC waste and HLW is only solid and meets the requirements given in 10 CFR 72.120(b) and (c).

F3.3 The SAR and docketed materials adequately define the bounding conditions under which the DSF or DSS is expected to operate in accordance with the requirements of 10 CFR 72.24(a), 10 CFR 72.92, 10 CFR 72.94, and 10 CFR 72.122(b)(c) for specific license applications, and 10 CFR 72.236 for CoC applications.

F3.4 The SAR and docketed materials relating to the design bases and criteria for structures categorized as important to safety meet the requirements given in 10 CFR 72.24(c); 10 CFR 72.102; 10 CFR 72.103; 10 CFR 72.104(a); 10 CFR 72.106(b); 10 CFR 72.120(a)(b)(c)(d); 10 CFR 72.122(a), (b)(1), (b)(2), and (b)(3), (c)(d)(f); 10 CFR 72.126(a)(d) for specific license applications; and 10 CFR 72.236 for CoC applications.

F3.5 The SAR and docketed materials meet the regulatory requirements for design bases and criteria for thermal consideration as given in 10 CFR 72.122(a), (b)(1), (b)(2), and (b)(3), (c)(d)(f)(g)(h)(i);

10 CFR 72.128(a)(4) for specific license applications; and
10 CFR 72.236(f) for CoC applications.

F3.6 The SAR and docketed materials relating to the design bases and criteria for shielding, confinement, radiation protection, and ALARA considerations meet the regulatory requirements as given in 10 CFR 72.24(c), 10 CFR 72.104, 10 CFR 72.106, 10 CFR 72.122(a–i), 10 CFR 72.126, 10 CFR 72.128 for specific license applications, and 10 CFR 72.236(b)(d) for CoC applications.

F3.7 The SAR and docketed materials relating to the design bases and criteria for criticality safety meet the regulatory requirements as given in 10 CFR 72.124 and, for CoC applications, 10 CFR 72.236(c).

F3.8 The SAR and docketed materials relating to materials selection meet the regulatory requirements as given in 10 CFR 72.24(c)(3), 10 CFR 72.120(d), 10 CFR 72.122(a)(b)(c), 10 CFR 72.124(a)(b), 10 CFR 72.128(a)(2) for special license applications, and 10 CFR 72.124(a)(b) and 10 CFR 72.236(b)(c)(d)(g)(m) for CoC applications.

F3.9 **(SL)** The SAR and the docketed materials relating to the design bases and criteria meet the general requirements as given in 10 CFR 72.24(c)(1), (c)(2), (c)(4); 10 CFR 72.104; 10 CFR 72.106; 10 CFR 72.120(a)(b)(c)(d); 10 CFR 72.122; 10 CFR 72.124; and 10 CFR 72.126(a)(d).

F.3.10 **(SL)** The SAR and docketed materials relating to design criteria for decommissioning of the facility comply with the regulatory requirements in 10 CFR 72.130 and the guidance in applicable portions of RGs 1.184 and 1.191.

F3.11 **(SL)** The SAR and docketed materials relating to the design bases and criteria for retrieval capability meet the regulatory requirements in 10 CFR 72.122(a), (b)(1), (b)(2), and (b)(3), (c)(f)(h)(l).

F3.12 **(SL)** The SAR and docketed materials relating to the design bases and criteria for other SSCs not important to safety, but subject to NRC approval, meet the general regulatory requirements in 10 CFR 72.24(a–h) and (l) and the appropriate requirements in 10 CFR 72.120 and 10 CFR 72.122.

F3.13 **(CoC)** The SAR and the docketed materials relating to the design bases and criteria meet the general requirements as given in 10 CFR 72.236(b).

The reviewer should provide a summary statement similar to the following:

The staff finds that the descriptions of the DSF or DSS characteristics are such that appropriate design criteria and bases for the DSF or DSS could be defined and evaluated. The staff concludes that the principal design criteria for the DSF or DSS are acceptable with regard to meeting the regulatory requirements in 10 CFR Part 72. This finding is reached on the basis of a review that considered the regulation, itself, appropriate regulatory guides, applicable codes and standards, and accepted engineering practices. Chapters 3 through 16 of the safety evaluation report present a more detailed evaluation of the design criteria and an assessment of compliance with those criteria.

3.7 References

10 CFR Part 20, “Standards for Protection against Radiation.”

10 CFR Part 50, “Domestic Licensing of Production and Utilization Facilities.”

10 CFR Part 71, “Packaging and Transportation of Radioactive Material.”

10 CFR Part 72, “Licensing Requirements for the Independent Storage of Spent Nuclear Fuel, High-Level Radioactive Waste, and Reactor-Related Greater than Class C Waste.”

10 CFR Part 73, “Physical Protection of Plants and Materials.”

American National Standards Institute (ANSI) N210-1976/American Nuclear Society (ANS) 57.2-1983, “Design Objectives for Light Water Reactor Spent Fuel Pool Storage Facilities at Nuclear Power Stations.”

ANSI/ANS 57.9-1992, “Design Criteria for an Independent Spent Fuel Storage Installation (Dry Storage Type).”

U.S. Nuclear Regulatory Commission Bulletin 96-02, “Movement of Heavy Loads over Spent Fuel, over Fuel in the Reactor Core, or over Safety Related Equipment,” dated April 11, 1996 (ADAMS Accession No. ML082590698).

National Fire Protection Association (NFPA) 780, “Standard for the Installation of Lightning Protection Systems.”

NFPA 70, “National Electrical Code.”

NUREG-0800, U.S. Nuclear Regulatory Commission, “Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition,” March 2007 (ADAMS Accession No. ML070660036 (package)).

NUREG-1927, U.S. Nuclear Regulatory Commission, “Standard Review Plan for Renewal of Specific Licenses and Certificates of Compliance for Dry Storage of Spent Nuclear Fuel,” Revision 1, June 2016, (ADAMS Accession No. ML16179A148).

NUREG-2174, U.S. Nuclear Regulatory Commission, “Impact of Variation in Environmental Conditions on the Thermal Performance of Dry Storage Casks, Final Report,” issued March 2016 (ADAMS Accession No. ML16081A181).

Regulatory Guide 1.13, “Spent Fuel Storage Facility Design Basis,” (ADAMS Accession No. ML070310035).

Regulatory Guide 1.29, “Seismic Design Classification,” (ADAMS Accession No. ML16118A148).

Regulatory Guide 1.59, “Design Basis Floods for Nuclear Power Plants,” (ADAMS Accession No. ML003740388).

Regulatory Guide 1.60, “Design Response Spectra for Seismic Design of Nuclear Power Plants,” (ADAMS Accession No. ML13210A432).

Regulatory Guide 1.61, “Damping Values for Seismic Design of Nuclear Power Plants,” (ADAMS Accession No. ML070260029).

Regulatory Guide 1.76, “Design Basis Tornado and Tornado Missiles for Nuclear Power Plants,” (ADAMS Accession No. ML070360253.pdf).

Regulatory Guide 1.86, “Termination of Operating Licenses for Nuclear Reactors,” (ADAMS Accession No. ML16099A267).

Regulatory Guide 1.92, “Combing Modal Responses and Spatial Components in Seismic Response Analysis,” (ADAMS Accession No. ML12220A043).

Regulatory Guide 1.102, “Flood Protection for Nuclear Power Plants,” (ADAMS Accession No. ML003740308).

Regulatory Guide 1.117, “Protection Against Extreme Wind Events and Missiles for Nuclear Power Plants,” (ADAMS Accession No. ML15356A213).

Regulatory Guide 1.122, “Development of Floor Design Response Spectra for Seismic Design of Floor-Supported Equipment or Components,” (ADAMS Accession No. ML003739367).

Regulatory Guide 1.184, “Decommissioning of Nuclear Power Reactors,” (ADAMS Accession No. ML13144A840).

Regulatory Guide 1.189, “Fire Protection for Nuclear Power Plants,” (ADAMS Accession No. ML092580550).

Regulatory Guide 1.191, “Fire Protection Program for Nuclear Power Plants During Decommissioning and Permanent Shutdown,” (ADAMS Accession No. ML011500010)

Regulatory Guide 1.198, “Procedures and Criteria for Assessing Seismic Soil Liquefaction at Nuclear Power Plant Sites,” (ADAMS Accession No. ML033280143).

Regulatory Guide 1.205, “Risk-Informed, Performance-Based Fire Protection for Existing Light-Water Nuclear Power Plants,” (ADAMS Accession No. ML092730314).

Regulatory Guide 1.208, “A Performance-Based Approach to Define the Site-Specific Earthquake Ground Motion,” (ADAMS Accession No. ML070310619).

4 STRUCTURAL EVALUATION

4.1 Review Objective

The objective of the U.S. Nuclear Regulatory Commission's (NRC's) structural review is to ensure that the structural integrity of structures, systems, and components (SSCs) of the dry storage facility (DSF), which includes independent spent fuel storage installations (ISFSIs) and monitored retrievable storage installations (MRSs), or of a dry storage system (DSS), emphasizing SSCs important to safety (identified in Chapter 3 of the safety analysis report (SAR)). These SSCs may provide confinement, subcriticality, radiation shielding, support, and retrievability safety functions of the stored materials, and therefore, should be appropriately maintained under all credible loads and their combinations for normal, off-normal, and accident conditions and natural phenomena effects. These SSCs include pool and pool confinement facilities. Because the pool and pool confinement facilities are not routinely part of a storage facility application, they are not included in the standard review, but are presented in Appendix 4B. The evaluation should result in a reasonable assurance that storage systems and associated facilities will maintain their intended function.

4.2 Applicability

This chapter applies to the review of applications for specific licenses for an ISFSI or a MRS facility, categorized as a DSF. It also applies to the review of applications for a certificate of compliance (CoC) of a DSS for use at a general license facility. Sections that apply only to specific license applications have "**(SL)**" in the heading. Sections that apply only to CoC applications have "**(CoC)**" in the heading. In this chapter, these designations only appear in Table 4-1b and Section 4.6, "Evaluation Findings." All other sections apply to both types of applications, as specified in the text.

4.3 Areas of Review

This chapter applies to the evaluation of structural integrity for SSCs important to safety and other SSCs. It broadly categorizes the applicable regulatory requirements, acceptance criteria, and review procedures into features common to all SSCs, followed by areas of review for site-specific SSCs, outlined as follows:

4.3.1 Structures, Systems, and Components Important to Safety

- confinement canister (shell and associated welds and bolts)
  - fuel basket
  - fuel and cladding
  - racks for positioning stored fuel or waste material within the canister or cask (including lifting components)
  - closure lids
  - closure welds
- transfer cask
- storage overpack (horizontal, vertical, or underground)
- storage cask

4.3.2 Other Structures, Systems, and Components Subject to NRC Approval

- concrete pads for placement of storage systems. Concrete storage pads may be classified important to safety depending on the application
- SSCs associated with the transfer of confinement and transfer casks on site, including cask loading and extraction equipment, trailers, prime movers, crane, and equipment unique to the cask system whose failure would not jeopardize the basic safety requirements of the confinement system
- SSCs including cranes and other equipment for intermodal transfer of containers holding nuclear materials, such as truck, rail, and barge and ship docks whose failure would not jeopardize the basic safety criteria
- onsite SSCs associated with facilities other than for the ISFSI or MRS but which are shared by the ISFSI and MRS, or that are physically connected to SSCs supporting the ISFSI or MRS, or both, and that have safety or safeguards and security-related functions
- onsite radioactive material transfer route structures, such as bridges, roads, rail crossings and heavy-haul paths
- structures and earthworks to prevent facility flooding on site
- SSCs, including equipment, that provide fire protection or that may be required to mitigate the effects of accident events
- other SSCs required for compliance with code safety requirements, such as for lightning protection

4.4 Regulatory Requirements and Acceptance Criteria

This section summarizes those parts of Title 10 of the *Code of Federal Regulations* (10 CFR) Part 72, "Licensing Requirements for the Independent Storage of Spent Nuclear Fuel, High-Level Radioactive Waste and Reactor-Related Greater Than Class C Waste," that are relevant review areas. Tables 4-1a and 4-1b match the relevant regulatory requirements to the areas of review this chapter covers. Table 4-1a matches the relevant regulatory requirements to the areas of review for specific license applications. Table 4-1b matches the relevant regulatory requirements to the areas of review for a CoC. Refer to the language in the regulations and verify the association of the regulatory requirements with the areas of review presented in the table to ensure that no requirements are overlooked as a result of unique applicant design features.

Table 4-1a Relationship of Regulations and Areas of Review for a DSF (SL)

Areas of Review	10 CFR Part 72 Regulations					
	Subpart B	Subpart C	Subpart F			
	72.24	72.40	72.120	72.122	72.124	72.128
SSCs Important to Safety	(b)(c)(d)(i)	(a)(1)	(a)	(a)(b)(c)(d)(l)	(a)(b)	(a)(2), (a)(3)
Other SSCs	(b)(c)	(a)(1)		(b)(2)(ii), (d)	(a)	
Pool and Facilities (see Appendix 4B)	(b)(c)(d)(i)	(a)(1)	(a), (b)(3)	(a)(b)(c)(d)(l)	(a)(b)	(a)(2), (a)(3)

Table 4-1b Relationship of Regulations and Areas of Review for a DSS (CoC)

Areas of Review	10 CFR Part 72 Regulations		
	Subpart F	Subpart L	
	72.124	72.234	72.236
SSCs Important to Safety	(a)(b)	(a)	(b)(c)(d)(e)(g)(h)(l)(m)
Other SSCs	(a)(b)		

Acceptability of the design of the SSCs as described in the SAR is based on compliance with the requirements in 10 CFR Part 72 and regulatory guidance as determined by independent calculations and staff judgment. The designs of the SSCs are acceptable if they meet general or specific design criteria discussed in this Standard Review Plan (SRP).

DSS or DSF applications have a one-step license approval process. Thus, the evaluation of the SAR and the supporting materials is the sole occasion during which the NRC staff comprehensively reviews the design and proposed construction.

SSCs important to safety are required to have sufficient structural capacity so that the structure can withstand the postulated worst-case loads under normal, off-normal, and accident conditions described in Section 4.5, “Review Procedures,” of this SRP, while performing their required function (confinement, shielding, subcriticality). The NRC does not accept breach of the storage confinement.

SSCs important to safety are expected to withstand the postulated worst-case loads under postulated accident conditions to successfully prevent preclude the following events:

- unacceptable risk of criticality

- unacceptable release of radioactive materials to the environment

- unacceptable radiation dose to the public or workers

- significant impairment of retrievability or recovery, as applicable, of stored nuclear materials for postulated normal and off-normal conditions.

This position does not necessarily require that the confinement system and other structures important to safety survive every postulated design-basis accident condition without any permanent deformation or other damage. Some load combination expressions for the design-basis conditions for structures important to safety permit stress levels that exceed the yield strength of the material. The SAR should include computations of the maximum extent of potentially significant accident deformations and any permanent deformations, degradation, or other damage that may occur.

Similarly, the review of the other SSCs should ensure their structural integrity under the loading resulting from postulated normal, off-normal, and accident conditions, as defined in the glossary to this SRP. Section 4.5.2 of this SRP provides a more detailed discussion for the review requirements and acceptance criteria for the SSCs.

4.5 Review Procedures

Review the entire SAR, particularly the sections that describe the overall design and operations, the design criteria including the site characterization and bases, the structural evaluation information, the accident analysis, and the operating controls and limits. Coordinate with the materials reviewer to ensure that the materials and their associated structural properties are consistent with those used in the structural evaluations. Review any drawings and calculation packages submitted with the SAR for the particular SSC being evaluated. Figure 4-1 shows the interrelationship between the structural evaluation and the other areas of review described in this SRP.

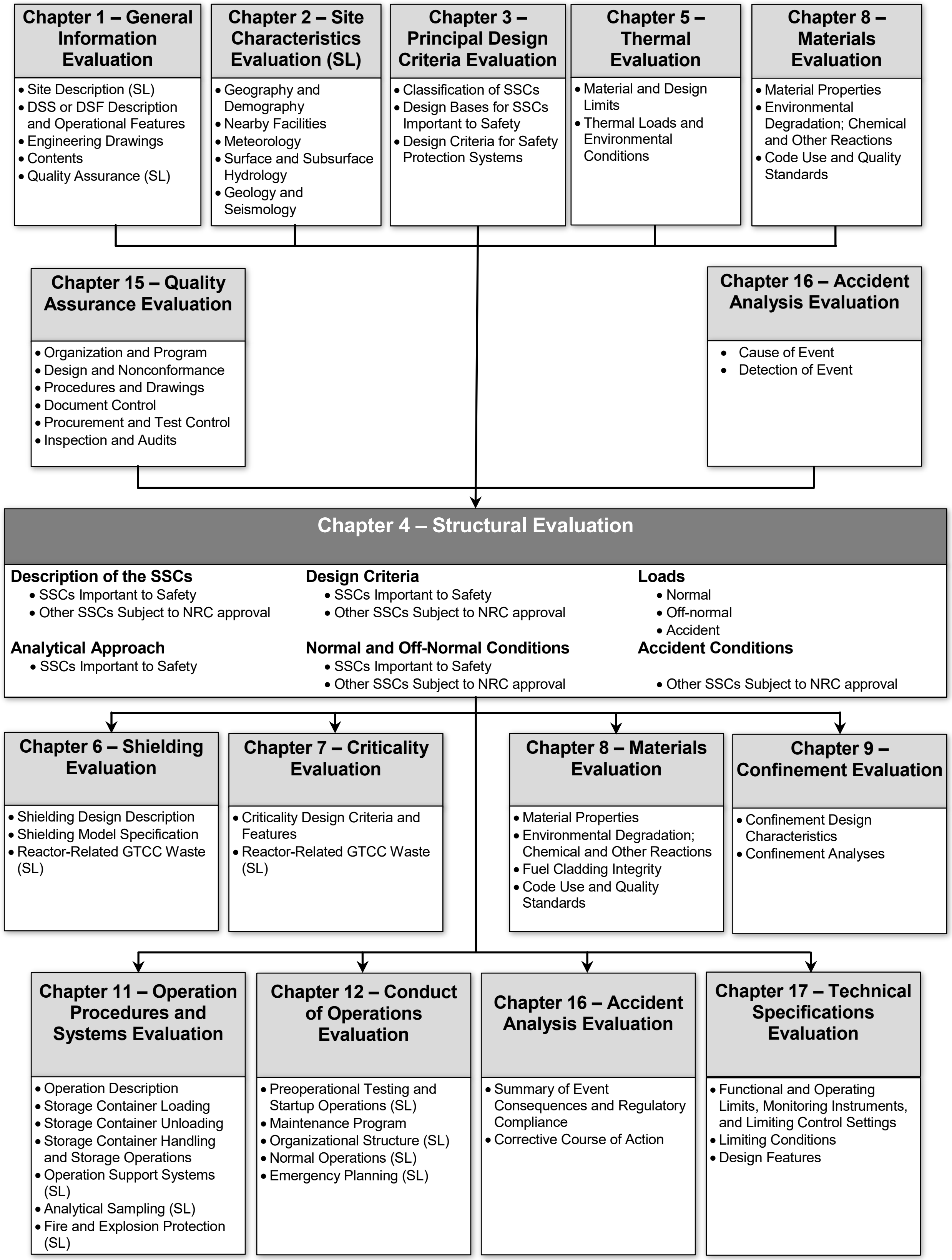

Figure 4-1 Overview of Structural Evaluation

Ensure that the application includes descriptions, design criteria, and safety analyses for site facilities and infrastructure of concern to the NRC, as appropriate to safety. These could include the waste facilities and other elements of the same infrastructure.

SSCs important to safety are not required to survive accidents to the extent that they remain suited for use for the life of the storage system without inspection, repair, or replacement. Ensure the SAR includes procedures for determining and correcting degradation and performing other acceptable remediation of SSCs if the service life of SSCs important to safety become degraded by accident conditions. The accident analysis evaluation chapter of the SAR addresses this.

Review the proposed technical specifications to ensure that they include adequate restrictions on cask handling and operations to preclude the possibility of damage to the structure or the confined nuclear material. Both the SAR and the NRC's safety evaluation report (SER) should include the operating controls and limits of the technical specifications. The SAR and SER should describe actions to be taken and inspections to be conducted upon the occurrence of events that may cause such damage.

Verify that the SAR clearly identifies the proposed structural design and construction of SSCs necessary for effective functional performance and safety. Review the SAR and supplemental material the applicant submitted to assess compliance with the applicable scope and content requirements defined in 10 CFR Part 72. Focus in particular on requirements and conditions of use related to design, construction, implementation, operation, and maintenance of SSCs.

Ensure the SAR identifies the design codes and standards used for all SSCs and their components. The structural design, fabrication, and testing of the SSCs should comply with an acceptable code or standard. Using codes and standards that have been accepted by the NRC may expedite the evaluation process.

Verify that the SAR defines the loads and load combinations. If the applicant has not adequately justified any deviations from the acceptance criteria for loads and load combinations, identify the deviations as unacceptable and transmit them to the applicant for further justification. If components associated with or integral to the fuel assembly are to be stored in the cask, ensure that the applicant's structural analysis has considered these components.

The SAR should include a comprehensive table of load combinations and safety margins for selected structural components important to safety (or otherwise subject to NRC evaluation). Ensure that the summary table includes sufficient forms of loadings (e.g., shear, flexure, axial, and combined stress situations) to verify that the lowest margins of safety are listed for the various components. In addition, the applicant can use this table to summarize the structural capacity evaluation.

Determine whether the applicant's design and analysis procedures and assumptions are conservatively defined on the basis of accepted engineering practice. Review the behavior of the structure under various loads and the manner in which these loads are treated in conjunction with other coexistent loads, and assess compliance with the acceptance criteria defined in this chapter of the SRP.

Evaluate the proposed limitations on allowable stresses and strains in the canister and steel parts important to safety and subject to review by comparison with those specified in applicable codes and standards. Where certain proposed load combinations will produce values that exceed the accepted limits for localized points on the structure, ensure the application provides adequate justification to show that a deviation will not affect the functional integrity of the SSC. Under certain conditions, limiting strains and limiting deformations may form part of the acceptance criteria.

Review the structural evaluation to determine whether conditions of use or technical specifications should be associated with the structural design. Determine the appropriateness of and need for any proposed technical specifications related to structural design and construction. Determine whether any additional technical conditions related to structural performance are needed, and, if so, provide input to the conditions of use discussed in the SER. Describe the basis for the suggested conditions in the structural evaluation section of the SER. Structure-related conditions of use may be linked to evaluations performed under other sections (such as a field verification that maximum concrete temperatures predicted from thermal analysis will not be exceeded).

4.5.1 Description of the Structures, Systems, and Components

4.5.1.1 *Structures, Systems, and Components Important to Safety*

The SSCs that are important to safety are those whose function provides for the general design criteria of confinement, subcriticality, shielding, and retrievability. Ensure that the SAR provides drawings, plans, sections, supporting computations, and specifications for those structural components important to safety in sufficient detail to allow meaningful reviews, as required by 10 CFR Part 72. Ensure that the application includes the year of all codes or standards that are referenced on the drawings.

Ensure the applicant describes the SSCs important to safety in sufficient detail to allow evaluation of their structural behavior and effectiveness under the imposed design conditions. In addition, ensure the SAR identifies all codes and standards applicable to the components.

4.5.1.1.1 Canister or Storage Cask and Metallic Internals

Review the canister or storage cask descriptive information presented in the SAR chapter on general information, as well as any related information provided in the SAR chapter on structural evaluation. These may include the canister or metal storage cask system that could include a shell, inner and outer lids, and welds or bolts; port covers and bolts; vent port covers to be welded in place; and fuel basket.

Coordinate with the confinement reviewer (SRP Chapter 9, “Confinement Evaluation”) to verify that the SAR clearly identifies the confinement boundaries. These boundaries include the primary confinement vessel; its penetrations, seals, welds, and closure devices; and the redundant sealing system as provided by the system.

Ensure that the canister or cask assembly drawing, figures, tables, and specification in the SAR fully show geometry and material used for analysis and fabrication. Canister and cask shells are normally constructed from stainless steel. Appropriate numbers of plugs are provided to drain and vent the shell assembly. Ensure the canister or cask is designed to provide confinement in an inert environment, structural support, and criticality control for the fuel assemblies. The canister or cask is equipped with design features for shielding and heat rejection capabilities. Verify that the application reflects that the spent nuclear fuel (SNF) storage cask provides redundant sealing of the confinement system.

Review the SAR to verify that the canister top and bottom cover plates are properly located and welded with full or partial penetration welds. With the exception of the top cover plates, ensure that the canister is fabricated with full penetration welds. Ensure that the closure system consists of redundant lids that are attached with partial penetration welds.

Review the SAR for any details on lifting attachments used to handle the canister or cask loaded with SNF into and out of the storage overpack and transfer cask respectively.

4.5.1.1.2 Fuel Basket

Review the SAR for the fuel basket design to ensure that it locates and confines the fuel assemblies inside the canister. Ensure the SAR describes the type and number of fuel assemblies (pressurized-water reactor or boiling-water reactor) to be stored in the fuel basket. Ensure the basket design is adequate to withstand the combined effects of weight, thermal stresses, and cask-drop impact forces that could arise during SNF transfer and storage operations. The weight supported by the basket should be the maximum or design weight of the SNF to be stored. In addition, ensure the applicant evaluates all credible potential orientations of the cask and basket during cask transfer and handling drops while transferring the SNF into storage.

4.5.1.1.3 Fuel and Cladding

Review the SAR for the design, specifications, and geometry of the fuel rod and cladding. While the fuel assembly is not necessarily an SSC, the cladding does provide defense in depth by containing fission products within its boundary.

4.5.1.1.4 Transfer cask

Review the transfer cask descriptive information presented in the SAR chapter on general information, as well as any related information provided in the SAR chapter on structural evaluation. Ensure the transfer cask is examined for normal, off-normal, handling, and accident conditions. The geometry of the transfer cask design should be such as to provide shielding and protection from potential hazards during canister loading and closure operations as well as during transfer to the storage overpack. The transfer cask is not required to be a pressure-retaining vessel. Ensure the design incorporates features to provide circulation of cooling air in the annular space between the canister and transfer cask inner diameter.

The transfer cask is usually manufactured from steel with welded bottom assemblies and a bolted top cover plate. Verify that the neutron and gamma shields are fabricated from appropriate materials. For ease of handling and transportation, lifting trunnions are usually provided on the transfer cask. The transfer cask for the vertical cask system may also have doors and associated rail or attachments on the bottom to facilitate the transfer of the canister into the storage overpack.

If impact limiters are used during the transfer and storage operations, ensure the applicant thoroughly evaluates and verifies the nonlinear impact characteristics of the limiters. In addition, ensure that the applicant tabulates and describes the crush characteristics and properties of the limiters (if any) in the directions that are to be used.

4.5.1.1.5 Storage Overpack (horizontal, vertical, or underground)

Ensure that the SAR provides a detailed description, specification, materials of construction, and drawings showing the geometry and structure arrangement of the storage overpacks. The storage overpack should be designed as a freestanding or underground structure (normally of concrete, steel, or both), designed to provide environmental protection and radiological shielding for the canister. Ensure the drawings in the SAR clearly show how the canister will be inserted and stored inside the cask. In addition, ensure the drawings show the location of reinforcing steel and embedment required to attach other components, such as heat shields and shield walls.

The concrete may be cast in place, on site, or elsewhere. Concrete overpacks may also be combinations of cast-in-place and precast sections that are integrated by bolting, welding, fitting, grouting, or placing additional concrete at the site.

4.5.1.1.6 Independent Spent Fuel Storage Installations Concrete Pad (as applicable)

If the concrete storage pad is classified as important to safety, ensure that the SAR provides a detailed description, specification, and materials of construction to be used for the ISFSI concrete pad. In addition, ensure that the drawings show the layout and cask transportation route on the pad. Verify that the SAR describes how the casks will be arranged on top of the ISFSI concrete pad.

4.5.1.2 *Other Structures, Systems, and Components Subject to NRC Approval*

Ensure that the SAR text descriptions, drawings, figures, tables, and specifications fully define the other SSCs subject to NRC approval. Ensure that the specifications reference the codes that govern the design details. Verify that the combinations of drawings, specifications, appropriate codes and standards, and supporting calculations are sufficient.

Confirm that, at a minimum, the SAR documentation provides (1) the dimensions of all sections that have a structural role including locations, sizes, configuration, and spacing; (2) structural materials with defining standards or specifications; (3) location and specifications for assembly and weld joints; (4) location of all reinforcing steel; and (5) fabrication codes and standards.

Verify that these SSCs are described sufficiently to provide an adequate basis for their approval. Typically, this would include descriptive information about the function, applicable codes, and standards for design and manufacture or procurement.

4.5.2 Design Criteria

Review the design criteria that the applicant is using to qualify the structural performance of each of the SSCs. This review should include the codes and standards and applicable loading conditions (i.e., normal, off-normal, and accident). Ensure the SAR identifies the design criteria (code, code case, or standard) used for the design, fabrication, and testing of each SSC component and any alternatives to those design criteria. Ensure the year of the code or standard is included for all codes and standards referenced in the application.

Applicants should propose a condition to the CoC or technical specification in a site license, either directly or by reference, describing the alternatives to the referenced codes. Ensure the condition or technical specification also describes a process to address one-time alternatives from the code that may occur during fabrication. Verify that the application identifies the component, references the code (code edition, addenda, section, or article), describes the code requirement, and describes the alternative. In addition, ensure the applicant justifies the alternative, including a description of how the alternative would provide an acceptable level of quality and safety. Confirm that the application describes how compliance with the code provisions would result in hardship or difficulty without a compensating increase in the level of quality or safety.

An applicant should justify the use of new criteria if no staff position has been established. However, use of codes and standards previously accepted by the NRC expedites the evaluation process. Use of other codes and standards, definition of criteria composed of extracts from multiple codes and standards with overlapping scopes, or substitution of other criteria, in whole or in part, in place of acceptable published codes or standards may require a more detailed review.

Review the identification of structural materials in coordination with the materials discipline as described in Chapter 8, "Materials Evaluation," of this SRP to the extent appropriate to determine if the materials are adequate for their intended function(s). Determine the required level of review and extent of information in relation to the possibility and consequences of secondary effects on components that are important to safety. Ensure the materials are permitted or specified in the applicable code(s).

Radiation shielding in the cask system is required to protect the public and workers involved with SNF handling and storage. Ensure such shielding will not degrade under normal or off-normal conditions or events. The shielding function may degrade as a result of an accident (e.g., displacement of source or shielding, reduction in shielding). However, the loss of function should be readily visible, apparent, or detectable. Ensure that the application shows that any permissible degradation in shielding will result in dose rates sufficiently low to permit recovery of the damaged cask including unloading, if necessary. Further, ensure that the applicant clearly identified the necessary structural criteria to assure adequate shielding remains in place.

The NRC has accepted the American Society of Mechanical Engineers (ASME) Boiler and Pressure Valve (B&PV) Code, Section III, "Rules for Construction of Nuclear Facility Components," Division 1, "Metallic Components," as the basic reference for metallic SSCs and has equated normal conditions of loading with Service Level A, off-normal loading with Service Level B, and accident condition loading with Service Level D. The ASME B&PV Code defines the requirements for categorizing stresses and determining allowable stress limits for the SSC or component in question. The NRC has also accepted the analytical approaches given in the ASME B&PV Code, Section VIII, "Rules for Construction of Pressure Vessels," for pressure systems, vessels, and casks that do not form elements of the confinement cask. In accordance with these references, stress intensity is defined on the basis of the maximum shear stress theory for ductile materials. Since the maximum shear stress is not identical to the maximum octahedral shear stress, verify that the octahedral shear stresses are not compared with the stress intensity limits. Appendices I and III to the ASME B&PV Code define values for the stress intensity limits. Verify that the applicant considers stresses resulting from inertial and pressure loads as primary stresses and that thermal stresses resulting from temperature gradients are considered secondary stresses if they are self-limiting and do not cause structural failure. Stresses caused by thermal gradients in fuel baskets may not be self-limiting; ensure the applicant considers these stresses because of the possibility of uneven heat loadings of adjacent assemblies as well as the effects of asymmetry in the basket structure. The NRC has accepted the use of American Concrete Institute (ACI) 349, "Code Requirements for Nuclear Safety-Related Concrete Structures and Commentary," as the basic reference for concrete structures important to safety that are not designed in accordance with ASME B&PV Code Section III, Division 1 or Division 2, "Code for Concrete Containments."

In general, the NRC accepts the use of the most recent code year for the design of SSCs for new applications. ASME B&PV Code Section III, Division 1, Subsection NCA-1140 has provisions for the use of ASME code editions, addenda and cases that applies to both new applications and amendments. The NRC may consider alternatives to this guidance on a case-by-case basis.

4.5.2.1 *Structures, Systems, and Components Important to Safety*

Ensure that the SAR indicates that the SSCs will not experience any permanent deformation or loss of safety function capability (i.e., confinement, subcriticality, shielding, and retrievability) during normal or off-normal operating conditions. However, the system may experience some permanent deformation, but no loss of safety function capability, in response to an accident.

4.5.2.1.1 Canister and Storage Cask Confinement Shell

A canister serves to confine and maintain safe storage conditions throughout its service life. Ensure that the SAR reflects that the confinement structures have sufficient structural capability so that every cross section of the structure can withstand the worst-case loads and successfully preclude the unacceptable risk of criticality, unacceptable release of radioactive materials to the environment, unacceptable radiation dose to the public or workers, and significant impairment of ready retrievability of the stored nuclear material. Ensure the SAR indicates that confinement of radioactive material is maintained under normal, off-normal, and accident conditions.

Design and construction codes (e.g., ASME B&PV Code, Section III) give reasonable assurance that the as-fabricated material will provide the necessary integrity. ASME B&PV Code Section III, Division 1 applies specifically to maintaining pressure boundaries and supporting structures in nuclear power plants and may not necessarily be totally applicable to all confinement SSCs. However, designers may choose to cite it as the code to which selected components are to be fabricated. Codes such as the ASME B&PV Code are not likely to address all the potential performance conditions (e.g., cracking, creep, corrosion) that may arise from environmental, electrochemical, or dynamic loading. Ensure the SAR addresses these and other effects specific to the individual application in order to meet the guidance in Chapter 8 of this SRP.

For the canister and associated welds, the NRC has accepted the use of ASME B&PV Code Section III, Division 1, Subsection NB or Subsection NC as the design criteria for normal and off-normal loading (Service Levels A and B, respectively) and Appendix F to ASME B&PV Code Section III, Division 1 for accident or natural phenomenon loading (Service Level D).

ASME B&PV Code Section III, Division 1 does not allow partial penetration welds for containment (confinement) boundaries. Because of fabrication considerations for the final canister closure weld, a full penetration weld is not always feasible. The NRC has accepted a partial penetration weld as an alternative to a full penetration weld for the closure weld, provided a stress reduction factor of 0.8 is applied to the strength of the weld to account for imperfections or flaws that may be missed by the allowed progressive surface examinations. Verify that the applicant applied a stress reduction factor of 0.8 to the allowable stress values for the design criteria. See Chapter 8 of this SRP for more information on weld design and examination.

4.5.2.1.2 Fuel Basket

For the fuel basket, the NRC staff has accepted the use of ASME B&PV Code Section III, Division 1, Subsection NG for the design criteria for normal and off-normal loading (Service Levels A and B, respectively) and Appendix F to ASME B&PV Code Section III, Division 1, for accident and natural phenomenon loading (Service Level D).

Ensure that the SAR includes an evaluation of the buckling capacity of the cask basket materials. Acceptable guidance for this evaluation is provided in Section III of the ASME B&PV Code and NUREG/CR-6322, “Buckling Analysis of Spent Fuel Basket,” issued May 1995. Ensure the applicant selects the appropriate end conditions used in the buckling capacity equations on the basis of sensitivity studies. These studies can bound the range of conditions that typically are either fixed for a welded connection or free if there is no rigid connection.

4.5.2.1.3 Fuel and Cladding

Review the design fuel cladding to ensure that it is adequately protected against gross rupture caused by degradation resulting from design or accident conditions. The combined stresses in cladding should remain below the yield strength of the material or justified otherwise. Confirm that the design ensures that the SSCs will not experience accelerations or decelerations, or both, that would damage their structural integrity or jeopardize their subcritical condition or retrievability under normal and off-normal design conditions.

Ensure that the applicant has evaluated fuel rod integrity by demonstrating that it will not buckle under the effects of the canister bottom-end drop condition.

4.5.2.1.4 Transfer Cask

For the transfer cask, the NRC has accepted the use of ASME B&PV Code Section III, Division 1, Subsection NF for the design criteria for normal and off-normal loading (Service Levels A and B respectively) and ASME B&PV Code Section III, Division 1, Appendix F for accident and natural phenomenon loading (Service Level D). For the neutron shield tank design, the NRC has accepted the use of ASME B&PV Code Section III, Division 1, Subsection ND for the design criteria for normal and off-normal loading (Service Levels A and B, respectively) and Appendix F to ASME B&PV Code Section III, Division 1 for accident and natural phenomenon loading (Service Level D).

Ensure the lid bolts that attach the lid to the body of the transfer cask are designed to the same standard as the transfer cask itself or to NUREG/CR-6007, “Stress Analysis of Closure Bolts for Shipping Casks,” issued April 1992. The NRC has accepted both standards.

The NRC has typically accepted American National Standards Institute (ANSI) N14.6 (1978), “Special Lifting Devices for Shipping Containers Weighing 10,000 Pounds (4500 kg) or More,” for governing transfer cask lifting device design and inspection requirements. This applies to lifting trunnions, their connections with the cask body, and the cask body localized around the trunnions, as modified by NUREG-0612, “Control of Heavy Loads at Power Plants: Resolution of Generic Technical Activity A-36.” NUREG-0612 stipulates that the weight of the lifting device consider a dynamic load factor. In addition, these design criteria may also apply to any doors and associated rails and attachments on the bottom of the transfer cask that facilitate the transfer of the canister into the storage overpack. Ensure the SAR reflects that the trunnions are tested to 300 percent of the design load during fabrication.

4.5.2.1.5 Storage Overpack (horizontal, vertical, underground)

The overpack should withstand the effects of credible accident conditions without impairing their ability to perform safety functions. The principle safety functions include maintaining subcriticality, containing radioactive material, providing radiation shielding for the public and workers, and maintaining retrievability of the stored material.

Concrete Storage Overpacks

For a concrete storage overpack, the NRC has accepted the use of the latest edition of ACI 349, supplemented with ACI 318, “Building Code Requirements for Structural Plain Concrete and Commentary,” for normal, off-normal, and accident loading combinations. In addition, for any structural steel elements that are part of the concrete overpack, the NRC has accepted the use of the latest edition of ANSI/American Institute of Steel Construction (AISC) 360, “Specification for Structural Steel Buildings,” or ASME B&PV Code Section III, Division 1, Subsection NF.

For welding of structural steel, the NRC has accepted American Welding Society (AWS) D1.1, “Structural Welding Code–Steel’” or ASME B&PV Code, Section IX, “Welding, Brazing, and Fusing Qualifications.”

Ensure steel embedments in the storage cask satisfy the requirements of the design code applicable to the reinforced concrete structure. Similarly, ensure structural steel satisfies the requirements of the applicable steel design code (e.g., ASME B&PV Code, AISC standard, or other identified code).

The ACI codes are intended to ensure ductile response beyond initial yield of structural components. ACI 349 also imposes conditions on design (beyond those of ACI 318) that effectively increase ductility. In particular, review the proposed reinforced concrete design to ensure that it provides code levels of ductility by satisfying the pertinent provisions in ACI 349. Seismic loads are considered to be “impulsive” and, therefore, are subject to the additional design constraints of Appendix F to ACI 349. Other accident conditions may also produce impulse or impact loading, necessitating the additional requirements of Appendix F to ACI 349.

Check the location and size of the steel reinforcement in the drawings to ensure that they are consistent with the design analysis.

Consider the following aspects of the design:

- limit on the amount (cross-section area) of compressive reinforcement in flexural members
- requirements on continuation and development lengths of tensile reinforcement
- specifications for confinement and lateral reinforcement in compression members, in other compressive steel, and at connections of framing members
- aspects of the design that ensure flexure controls (and limits) the response
- requirements for shear reinforcement

- limitations on the amount of tensile steel in the flexural members relative to that which would produce a balanced strain condition

- projected maximum responses to design-basis loads within the permissible ductility ratios for the controlling structural action

- reinforcement embedment designed for ductile failure where steel fails before pulling out from the concrete

Review the design to ensure that substitution of materials, use of larger sizes, or placement of larger quantities of steel will be precluded (to avoid changes in structural response), and that provisions for splicing or development of reinforcing steel will not reduce ductility of the members.

Metallic Storage Overpacks

For metal overpacks, or composite concrete structure overpack liners, the NRC has accepted the use of ASME B&PV Code Section III, Division 1, Subsection NF for the steel components.

If the overpack will be handled while loaded with fuel (i.e., transported to the storage pad while loaded with SNF), it should be considered a special lifting device. As such, ensure any trunnions or lifting attachments are also designed in accordance with the provisions of the lifting devices for a transfer cask in Section 4.5.2.1.4 of this SRP.

4.5.2.1.6 Independent Spent Fuel Storage Installation Concrete Storage Pad

Unless otherwise classified, the concrete storage pad is not generally classified as important to safety. In cases where the concrete pad serves a safety function (i.e., the storage cask is attached to the pad, or the pad has bollards around the cask), ensure it is classified as important to safety.

Verify that the ISFSI is designed to adequately support the static and dynamic loads of the stored casks, considering the potential amplification of earthquakes through soil structure interaction and soil liquefaction potential or soil instability due to vibratory motion. See NUREG/CR-6865, “Parametric Evaluation of Free Standing Spent Fuel Dry Cask Storage Systems,” issued February 2005 for further guidance.

Concrete storage pads that support the storage casks are not “pavements.” They should be designed and constructed as foundations under the applicable code. If the pad is classified as important to safety, the NRC has accepted ACI 349 for design and ACI 318 for construction. If the pad is not classified as important to safety, the NRC has accepted ACI 318 or the International Building Code (IBC) for design and construction.

Ensure the ISFSI concrete storage pad has sufficient capacity to withstand the worst-case loads under normal, off-normal, and accident loading combinations. Such capacity ensures that these structures will not experience permanent deformation or degradation of the ability to withstand any future loadings.

Vertical cask storage systems are evaluated against tipover during initial licensing, and all cask storage systems are evaluated against credible handling accidents during licensing. Although there is not a regulatory requirement of evaluating the system against a non-mechanistic event (i.e., non-credible tipover), performing the tipover and handling accident analysis, as documented in the SAR accident analyses chapter, provides additional assurance that the design will maintain confinement, criticality, and shielding during storage. The tipover analysis is performed by using a concrete compressive strength f'_c achieved at 28 days (see Tripathi 2007).

4.5.2.2 *Other Structures, Systems, and Components Subject to NRC Approval*

Details specific to certain codes and standards that may apply to other SSCs are listed below:

- ANSI/AISC 360–If the NRC receives an application using Load and Resistance Factor Design, or LRFD, the staff would evaluate the proposal for compliance with the loads and load combinations summarized in Tables 4-2 and 4-3, respectively, and for consistent application of the load and resistance factor design methodology.

- To date, the NRC has not required applicants to design or build structural steel components of a cask system important to safety in compliance with ANSI/ANS N690, “Nuclear Facilities–Steel Safety-Related Structures for Design Fabrication and Erection.”

- AWS D1.1

- ASCE 7, “Minimum Design Loads for Buildings and Other Structures”

- IBC

- ASME B&PV Code, Section VIII

- ACI 318

4.5.3 Loads

Review the loads that the applicant is considering for each SSC. In some cases, the loads may change based on the orientation of the SSC, such as the canister in the vertical position, down-ending into a horizontal position, in a horizontal position. Not all of the loads may apply to each SSC. For instance, a confinement canister inside a horizontal overpack may not be subject to tornado winds or tornado-generated missiles because it is protected by the overpack. It is, however, subject to seismic accelerations that may be amplified because of the dynamic response of the overpack to the seismic accelerations. Ensure that the applicant indicates all loads that are applied to each component and the manner in which they are applied.

Ensure that the design of the SSCs accommodates the full spectrum of load conditions, including all anticipated normal, off-normal, and accident or natural phenomena conditions. Coordinate with the appropriate NRC reviewer associated with Chapter 16, “Accident Analysis Evaluation,” of this SRP to verify that the accidents identified in that chapter correspond to the accident conditions evaluated in this chapter.

4.5.3.1 *Normal Conditions*

Normal conditions and events are those associated with canister system operations, including storage of nuclear material, under the normal range of environments. Ensure that the SAR states the assumed limits of normal-use environments to support an evaluation by a user of the certified cask system of its suitability for use at a licensed specific site under a general license or at a site with a specific license.

Loads normally applicable to the SSCs include weight, internal and external pressures, and thermal loads associated with operating temperature. The loads experienced may vary during loading, preparation for storage, transfer, storage, and retrieval operations. The weight is the maximum or design weight (including tolerances) of the cask in storage and loaded with SNF. However, depending on the operation and procedures, the weight should also include water fill. Confirm that the applicant evaluated all orientations of the cask body and closure lids during normal operations and storage conditions, including loads associated with loading, transferring, positioning, and retrieving the confinement cask.

Internal pressures result from hydrostatic pressure, cask drying and purging operations, filling with nonreactive cover gas, out-gassing of fuel, refilling with water, radiolysis, and temperature increases. Temperature variations and thermal gradients in the structural material may cause additional stresses in the canister, closure lids, and associated welds. Coordinate with the thermal reviewer (SRP Chapter 5, “Thermal Evaluation”) to determine the enveloping values and combinations of the cask internal pressures and temperatures for both hot and cold conditions. Use the temperature gradients calculated in the SAR chapter on thermal evaluation to determine thermal stresses. If the confinement system has several enclosed areas, all areas may not have the same internal pressures. In some canisters, enclosed areas consist of the canister cavity and the region between the inner and outer lids.

Required evaluations include weight plus internal pressures and thermal stresses from both hot and cold conditions. Verify that the applicant included the maximum thermal gradient, as determined in the thermal analysis, when evaluating thermal stresses.

For lifting and handling operations, ensure that the applicant applies an appropriate dynamic load factor to the load. See NUREG-0612, Section 5.1.1(4) for the appropriate application of the dynamic load factor for lifting operations.

For handling conditions, verify that the SAR reflects application of appropriate additional loads in vertical, transverse, and axial to fuel assemblies in normal conditions. As a minimum, the NRC considers loads of 1 g (in addition to self-weight) in all directions to be acceptable unless detailed analysis is performed otherwise.

Other loads during normal conditions may include the following:

- hydrostatic pressure in the neutron shield tank from the weight of the water and any applied pressure

- live and dynamic loads associated with the transfer of the confinement cask to and from its storage position and in its storage location for its service lifetime

- load or support conditions associated with potential differential settlement of foundations supporting the ISFSI pad over the life of the cask system

- thermal gradients associated with the normal range of operations and ranges of ambient temperature

- dead, live, and lateral soil loads defined in Table 4-3 of this SRP and ASCE 7 or the IBC for facilities

4.5.3.2 *Off-Normal Conditions*

Identify and evaluate all off-normal events and conditions described in Chapter 16 of this SRP. Review the off-normal conditions and events for those that affect the SSC. The SSCs should satisfy the same structural criteria required for normal conditions, as discussed above.

Ensure that the SAR clearly identifies anticipated off-normal conditions and events that may reasonably be expected to occur during the life of the SSC at the proposed site. In addition, verify that the SAR states the environmental limits to support comparison of the DSS design bases with specific site environmental data. Off-normal conditions and events can involve potential mishandling, simple negligence of operators, equipment malfunction, loss of power, and severe weather (short of extreme natural phenomena).

Other off-normal loads may include the following:

- live and dynamic loads associated with equipment or instrument malfunctions, or accidental misuse during transfer of the confinement cask to and from its storage position

- situations in which a confinement cask is jammed or moved at an excessive speed into contact with a reinforced concrete or steel structure

- the impact to reinforced concrete structures by a suspended transfer, confinement, or storage cask

- off-normal ambient temperature conditions; while they may be less severe than accident conditions, these may be of concern because of different sets of factors in the off-normal and accident load combinations, and because concrete temperature limits for off-normal conditions are the same as for normal conditions. Note that elevated concrete temperatures above those allowable by the code may be allowed for accident conditions in accordance with ACI 349, Section A.4. Consult Chapter 8 of this SRP for more information on elevated concrete temperatures

- dead, live, lateral soil pressure and wind loads defined in Table 4-3 of this SRP and ASCE 7 or the IBC for facilities

4.5.3.3 *Accident Conditions*

Ensure the SAR addresses, at a minimum, each of the following accidents or states why they are not credible. SRP Chapter 16 addresses the identification of credible accident conditions and any postevent inspection and remedial actions that may be necessary.

Ensure that the SAR considers the following accident scenarios:

4.5.3.3.1 Cask Drop and Tipover

A cask drop (including the transfer cask) or tipover scenario could result from cask handling during the loading and transfer process, an earthquake, flood, and wind effects. Ensure that the SAR includes a drop and tipover analysis. Ensure the SAR identifies the operating environment experienced by the SSC and the drop events (end, side, tipover) that could result. Generally, applicants establish the design basis in terms of the maximum height to which the cask is lifted or the maximum deceleration that the cask could experience in a drop. The design-basis drops should be determined on the basis of the actual potential handling and transfer accidents.

Although cask system supporting structures may be identified and constructed as important to safety (i.e., designed to preclude cask tipovers), the NRC considers that cask tipover events should be analyzed. For such analysis, the NRC has accepted cask tipover about a lower corner onto a receiving surface from a position of balance with no initial velocity. The NRC has also accepted analysis of cask drops with the longitudinal axis horizontal (side drop), together with a drop with the longitudinal axis vertical (top or bottom-end drop), if this combination bounds a non-mechanistic tipover analysis.

The applicant may use prototype or scale-model testing to obtain more realistic SSC deceleration or equivalent load for quasi-static analyses when applicable. Alternatively, applicants can develop an analytical model to calculate cask deceleration loads. In the analytical approach, the hard-receiving surface for a drop or tipover accident need not be an unyielding surface, and its flexibility may be included in the modeling. In general, using an unyielding surface will produce higher decelerations in a drop or tipover since the storage pad will, in reality, bend and deform. If the pad is treated as being other than an unyielding surface, the applicant should consider concrete hardening with time. Specifically, NUREG/CR-6424, “Report on Aging of Nuclear Power Plant Reinforced Concrete Structures,” issued March 1996, states that the majority of concrete hardening occurs within the first 10 years of service life. Compressive strength (f_c’) can be assumed to have increased on average by 65 percent, while Young’s Modulus (E) can be calculated with this value using ACI-318 for normal weight concrete.

Ensure that the applicant evaluated all credible potential orientations of the cask during cask transfer and handling drops while transferring the SNF into storage. End or side drops typically produce the greatest structural demand on various basket components. Often in an end drop, the basket is supported by the bottom of the confinement cask cavity upon impact. In the side drop, ensure the basket structure and points of contact with the confinement cask support the mass of the basket and loaded fuel.

4.5.3.3.2 Earthquake

Review the applicant’s evaluation of the cask design with regard to the structural consequences of the earthquake event. Ensure that the cask designs satisfy the load combinations that encompass earthquake, including those for sliding and overturning. Ensure that the applicant demonstrated that no tipover or drop will result from an earthquake. In addition, impacts between casks should either be precluded or should be considered an accident event for which the cask is shown to be structurally adequate. In most cases, impacts between casks are bounded by the non-mechanistic tipover analysis.

The DSS or DSF concrete pad, supported by soil, behaves as a rigid mat and therefore possesses no out-of-plane flexibility. This is valid for the majority of nuclear power plant structures, where relatively thick mats support integral reinforced concrete walls. However, pads are usually relatively thin structures (i.e., small thickness-to-length ratio) and generally do not incorporate integral walls to stiffen the pad. While the cask itself is relatively rigid, the rigid cask resting on a flexible pad has a lateral mode frequency that is generally low enough to fall within the amplified range of most design earthquake spectra. Thus, in determining the inertia forces that act at the center of gravity of the cask for the purpose of evaluating the onset of sliding or tipping, ensure that the applicant has either accounted for the out-of-plane flexibility of the pad in the seismic analysis or demonstrated that it is not an important parameter in determining the response of the cask (see Bjorkman et al. 2001).

Verify that the cask system design meets appropriate guidance in Regulatory Guide (RG) 1.29, “Seismic Design Classification,” RG 1.60, “Design Response Spectra for Seismic Design of Nuclear Power Plants,” RG 1.61, “Damping Values for Seismic Design of Nuclear Power Plants,” and RG 1.92, “Combining Modal Responses and Spatial Components in Seismic Response Analysis,” for protection against seismic events.

Ensure that the SAR includes an analysis of the potential for impacts between components of the cask system. These could include contact between the confinement canister and its inner components or outer shield and the rocking and falling back of a vertically or horizontally oriented confinement cask on its supports.

Cask systems are not required to survive a design earthquake without permanent deformation. However, ensure the SAR includes a prediction of the maximum extent of damage from a design earthquake and shows that the ability to provide the safety functions will not be degraded.

4.5.3.3.3 Tornado Winds

Verify that the SAR addresses the potential structural consequences of design-basis tornado or extreme wind effects. Review the load combination analyses for acceptable inclusion of tornadoes and tornado missiles. The guidance in RG 1.76, “Design-Basis Tornado and Tornado Missiles for Nuclear Power Plants,” recognizes three regions in the contiguous United States, each with distinct design-basis tornado parameters. Ensure that the applicant for a CoC has clearly defined the boundary conditions of the proposed cask system with respect to these regions or uses Region 1.

Confinement casks may be vulnerable to overturning or translation caused by the direct force of the drag pressure while in storage or during transfer operations. Ensure that the SAR provides criteria for resistance to overturning or sliding.

Confinement casks are generally not vulnerable to damage from overpressure or negative pressure associated with tornadoes or extreme winds. However, they may be vulnerable to secondary effects, such as windborne missiles or collapse of a weather enclosure, if used. Ensure that the SAR identifies the capability and behavior of the cask system under the collapse of any such external structure.

Tornadoes typically produce the greatest “design-level” wind effects for U.S. sites. However, there are some potential U.S. sites at which high hurricane winds may be more severe than the credible tornado. The SARs for a limited set of potential sites could reflect high wind effects as a basis for structural analysis. If the CoC is to include proven design resistance to tornadoes or extreme winds, ensure that the SAR identifies the wind levels (in miles or kilometers per hour), source (tornado or high hurricane wind), and specific wind-driven missiles (shape, weight, and velocity) against which the design is to be evaluated.

RG 1.76 provides applicable tornado-related parameters. The NRC has accepted the use of ASCE 7 for conversion of wind speed to pressure and for typical building shape factors. In sections that discuss conversion of tornado or other wind speeds to pressure, ensure that the SAR assumes that the cask system is at sea level.

Verify that the cask system design is consistent with guidance in RG 1.76; RG 1.117, "Tornado Design Classification," and NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LRW Edition," Section 3.3.2, "Tornado Loadings," for tornado protection.

Ensure the SAR considers that tornadoes and high winds can produce a significant negative pressure differential between interior spaces and the outside in a storage cask system. This is a function of wind speed and factors relating to the structure. The magnitude of negative pressure depends on other parameters of the tornado or wind, and on wall pressure coefficients (as expressed in ASCE 7). The SAR does not need to separately state negative pressure to establish an envelope for approval since negative pressure is insignificant with regard to confinement cask accident pressure analysis.

The NRC does not accept the presumption that there will be sufficient warning of tornadoes so that operations, such as transfer between the fuel transfer facility and storage site, may never be exposed to tornado effects. The staff considers overturning during onsite transfer to be a design-basis event. The tornado analysis may determine that tornado-induced overturning is bounded by drop and tipover cases. Ensure that the SAR shows that the cask system will continue to perform its intended safety functions (i.e., criticality, radioactive material release, heat removal, radiation exposure, and retrievability).

4.5.3.3.4 Tornado Missiles

Review the applicant's evaluation of the cask system design with regard to the structural consequences of wind-driven missile impact (RG 1.76 and Sections 3.5.1.4, "Missiles Generated by Tornadoes and Extreme Winds," and 3.5.3, "Barrier Design Procedures." of NUREG-0800 describe the effects of tornado missiles). Ensure that the SAR defines the missile parameters against which the cask system is to be evaluated based on the three tornado regions identified in RG 1.76.

Among the possible missile effects, the SAR should address those that may result in a tipover and those that may cause physical damage as a result of impact. Ensure that the damage does not result in unacceptable radiation dose or significantly impair criticality control, heat removal, or the retrievability of the fuel.

The NRC has accepted the use of the analytical approaches given in Cottrell and Savolainen (1965) for estimating the potential effects of missile impact on steel sheets, plates, and other structures. Section 3.5.3 of NUREG-0800 provides further guidance on analytical acceptable approaches for use in DSS or DSF design.

Cask systems are not required to survive missile impacts without permanent deformation. However, ensure that the maximum extent of damage from a design-basis event is predicted and sufficiently limited. Moreover, ensure that the ability of the SSCs to perform their safety functions is not impaired.

4.5.3.3.5 Flood

Review the applicant's evaluation of the design of SSCs with regard to the structural consequences of a flood event. The SAR may stipulate an assumption that the cask system not be used at any site where there is the potential for flooding. In this case, the cask would have to be placed at an ISFSI or MRS above the maximum probable flood level (the accident analysis in the SAR should state this condition).

If a design flood event is defined for the CoC, verify that the SSCs meet the appropriate guidance in RG 1.59, “Design Basis Floods for Nuclear Power Plants,” and RG 1.102, “Flood Protection for Nuclear Power Plants,” for that level of flood protection.

One possible structural consequence of a flood is that a vertically stored cask may tip over or translate horizontally (slide) because of the water velocity. Another possible consequence is that external hydrostatic pressure will exceed the capacity of the cask. Verify that the application states that the critical water velocity and hydrostatic pressure bound the flood analysis.

The NRC has accepted the evaluation for flooding events when the flood conditions for overturning and sliding of stored confinement casks and other cask system structures have been applied. Ensure that the application states the basis for estimation of lateral pressure on a structure is a result of water velocity.

Confirm that the SAR includes a calculation of drag coefficients and net lateral water pressure. An approach for calculating the velocity corresponding to the cask stability limit is to assume that the cask is pinned at the outer edge of the cask bottom and rotates about that outer edge, and the pinned edge does not permit sliding. The overturning moment from the velocity of the flood water can be compared to the stability moment of the cask (with buoyancy considered). The structural consequences of the flood event typically are bounded by analyses for the drop or tipover accident cases.

Additional flood conditions could lead to such consequences as potential scouring under a foundation, damage to access routes, temporary blockage of ventilation passages with water, blockage of ventilation passages and interstitial spaces between the confinement cask and shielding structure with mud, and steep temperature gradients in the shielding structure and confinement cask. Confirm that the applicant analyzed the consequences of these conditions and that the CoC or specific license identifies the consequences of these conditions so a licensee will be able to consider these factors when siting a DSS or DSF.

4.5.3.3.6 Fire

Verify that the SAR evaluation includes fire-related structural considerations, such as increased pressures in the confinement cask, changes in material properties, stresses caused by different coefficients of thermal expansion or temperatures in interacting materials (or both), and physical destruction. Chapter 5 of this SRP addresses potential fire conditions. Coordinate with the thermal reviewer to ensure that the criteria used (pressure, temperature) are consistent with accident conditions such as wild fire.

Evaluate the discussion in the SAR concerning the treatment of structural effects associated with the presumed fire and those structural effects for the assumed parameters of the postulated fire. Confirm that the applicant defined the confinement cask pressure capacity on the basis of the cask material properties at the temperature resulting from the fire. Spalling of concrete that may result from a fire is generally considered acceptable and need not be estimated or evaluated. Such damage is readily detectable, and appropriate recovery or corrective measures may be presumed. The NRC has accepted concrete temperatures that exceed the temperature limits of ACI 349 for accidents, provided the temperatures result from a fire. However, corrective actions may need to be taken for continued safe storage.

4.5.3.3.7 Explosive Overpressure

External explosion-induced overpressure and reflected pressure may result from explosives and chemicals transported by rail or on public highways, natural gas pipelines, and vehicular fires of equipment used in the transfer of casks. Explosions may result from detonation of an air-gaseous fuel mixture. With the exception of transfer vehicle accidents, the explosion hazards typically are similar to those for facilities subject to reviews under 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities."

Ensure the SAR states the level of overpressure that the cask system can withstand for this accident condition. This overpressure level would then serve as the quantitative envelope for future comparison with hazards for specific site installations. The pressure criteria for the assumed design-basis wind or tornado may also serve as an envelope for the explosive pressures for comparison with actual site hazards of a general licensee's facility, but this needs to be demonstrated in the SAR.

If the SAR includes bounding explosion effects for which the cask system is to be approved, verify that the SAR also includes structural analyses of those effects for cask system structures that may be affected. Ensure that the SAR identifies the maximum determined response. The maximum response includes pressure-induced maximum stresses at critical cask locations and governing structural performance modes for the cask components important to safety.

4.5.4 Analytical Approach

Review the structural analysis of various loading combinations and the calculated resulting stresses, strains, and deformations from different loads. Verify the applicant's proper use of acceptable analytical approaches and tools. The scope of the staff's review may include evaluating sensitivity analyses (such as finite element analyses) to validate submitted computations or their results.

Ensure that the SAR reflects analytical methods that are appropriate for the proposed type of materials and construction. In certain instances, however, the applicant may have had to adapt existing analytical methods, codes, and models for highly specialized storage-system equipment designs. Such instances require special review attention. In particular, ensure that the adapted approach is fully documented, supported, and acceptable. Consider the potential for safety-related risk associated with a possible error in the design of special cask system equipment. Appendix 4A, "Computational Modeling Software Technical Review Guidance," to this SRP chapter addresses the application of computational modeling software.

Ensure that the analysis of loads and load combinations is consistent with the code or criteria requirements used in designing the component. Material properties used in an analysis should be consistent with the approach being used.

4.5.4.1 *Hand Calculations*

This type of calculation can be used for analyses involving principles of conservation of energy and comparisons of overturning moments. Hand calculations can come in the form of spreadsheets or computer software such as Mathcad, where variables and intermediate solutions are stored within the program for later use in the calculation. The applicant has to define the equation and provide the necessary variables for its use.

Ensure that use of a particular equations or formulations for the load conditions is justified. The most important aspect of the calculations to evaluate is the basis for the assumptions used in the calculations. Check that calculations include applicable portions of the cask and appropriate load conditions. NUREG/CR-6007, "Stress Analysis of Closure Bolts for Shipping Casks," issued January 1993, provides acceptable analytical methods for closure bolts.

4.5.4.2 *Finite Element Analyses*

Because of the complexity of many structural design considerations and load conditions, structural design computations are often performed using finite-element analysis (FEA). Ensure that the applicant performed the FEA using a general-purpose program that is well benchmarked and widely used for many types of structural analyses.

Ensure that the FEA reflects appropriate element types, material properties, boundary conditions, consistent applied loading, and ability to accurately the behavior desired based on meshing and element type. Ensure the potential temperature of the material provides the basis for the elastic modulus and limit used for lead in the elastic analysis. An appropriate plasticity model of lead can be used to account for its inelastic behavior. Often, the applicant will create a partial model because of symmetry. Pay attention to the constraints introduced at the symmetry planes to ensure the proper symmetry conditions are applied to the model.

Finite element models do not generally include nonstructural components of the canister. However, check that the models include any influence these nonstructural components may have on the structural performance of the cask. Possible influences include inertial weight, restraint to motion of the structural components, and localized influence on load applications because of geometrical effects.

The NRC has accepted two approaches for analyses of the cask internal components undergoing cask drop scenarios. The first approach uses a two-step process. In step 1, the applicant performs a dynamic analysis of the cask body and its internal mass and stiffness equivalent impacting a target surface and assesses the performance of the cask body, including determining the time-history response. In step 2, this time-history response is translated into a forcing function and applied to the supporting contact points of an appropriate model of the internal components. This approach recognizes a commonly observed condition of the existence of a substantive stiffness difference between the cask body and its internals so that they can be dynamically uncoupled.

The second NRC-accepted approach uses a quasi-static analysis (assuming the quasi-static response dominates the response) of the basket subjected to the equivalent acceleration inertial load derived from the cask-drop impact analysis. If this analysis is used, ensure that the applicant applies the equivalent acceleration inertial load using an appropriate model of the internal components with the location(s) most vulnerable to the impact. Support provided by the inside surface of the cask cavity should be represented by the appropriate boundary conditions on the outside edge of the basket. In addition, ensure the applicant conservatively selects the equivalent acceleration inertial load such that it bounds the possible inertial loads resulting from a cask-drop accident onto the bounding target surfaces. If applicable, ensure the inertial load also accounts for dynamic amplification effects by using a dynamic amplification factor.

Review validation of the analytical model. The staff has completed a series of low-velocity impact tests of a steel billet from which a model validation approach and corresponding acceptance criteria have been developed. These tests and analytical evaluations are summarized in

NUREG/CR-6608, “Summary and Evaluation of Low-Velocity Impact Tests of Solid Steel Billet onto Concrete Pads.” On the basis of that report, the following model validation acceptance criteria apply to a cask-pad-soil analytical model for predicting impact responses of the cask:

When a solid steel billet is used to replace the cask in the cask-pad-soil analytical model, it should predict a pulse amplitude slightly higher than the cask. The calculated pulse duration and shape should be similar, but not necessarily identical, to those recorded from the cask. The validated billet-pad-soil model is considered adaptable to a cask-pad-soil analysis model if relevant attributes of the cask are used to replace those of the billet.

The FEA impact analysis for cask drop may consider the ISFSI concrete pad as rigid or a concrete pad underlain with soil. The material properties of the soil should be consistent with NUREG/CR-6608.

Verify that the applicant has provided information on any computer-based modeling as described in Appendix 4A to this SRP chapter and review the structural analyses the applicant submitted in accordance with Appendix 4A.

Alternatively, the draft guidance documents “Use of Explicit Finite Element Analysis for the Evaluation for Nuclear Transport and Storage Packages in Energy-Limited Impact Events” and the associated Attachment A, “Examples Demonstrating Modeling Principles for Explicit Finite Element Analysis,” may be useful in determining the quality of the applicant’s FEA model. Although the document is still in draft form at the time of publication of this NUREG, the guidance that has been developed by the Special Working Group on Computational Modeling for Explicit Dynamics may be relevant. The guidance document was submitted for ASME review in August 2017 and will be published if approved.

4.5.5 Normal and Off-Normal Conditions

Verify that the load combinations that the applicant considers to be normal and off-normal conditions are acceptable. Review the analysis on how the applicant’s results compare to the design criteria. The applicant may present the results in the form of factors of safety, stress ratios, or margins of safety. Confirm that the comparisons of calculated capacity versus demand for the various applicable loading conditions are presented in the same terms used in the design code (e.g., type of stress, bending moments, strains). Ensure the capacity values are larger than the allowed values for different load combinations. If they are not, ensure the applicant provided a defensible explanation as to how the design provides reasonable assurance against failure.

The NRC has accepted the load combinations and definitions shown in Tables 4-2 and 4-3 for analysis of non-confinement steel and reinforced concrete components. Load combinations are included in or derived from and ANSI/ANS 57.9, “Design Criteria for an Independent Spent Fuel Storage Installation (Dry Storage Type).”

4.5.5.1 *Structures, Systems, and Components Important to Safety*

4.5.5.1.1 Canister and Associated Welds and Bolts

Verify that the calculated stress in the canister and associated welds and bolts for the various normal and off-normal condition load combinations and each stress category are within the limits of allowable stress of the stated ASME B&PV Code that the applicant cited as the design criteria.

Ensure the allowable stresses are based on the temperature of the material in the loading condition considered and determined in accordance with ASME B&PV Code.

Verify that the applicant considered whether fatigue analysis of the canister is required in accordance ASME B&PV Code Section III, Division 1, Subsection NB-3222.4.

Review the design analysis for the canister's closure-lid bolts to ensure that it properly includes the combined effects of weight, internal pressure(s), thermal stress, O-ring compression force, cask impact forces, and bolt preload. Typically, applicants specify the preload and bolt torque for the closure bolts on the basis of bolt diameter and the coefficient of friction between the bolt and the lid. Externally applied loads (such as the internal pressure and impact force) produce direct tensile force on the bolts as well as an additional prying force caused by lid rotation at the bolted joint. The tensile bolt force obtained by adding together the pressure loads, impact forces, thermal load, and O-ring compression force should then be compared with the tensile bolt force computed from the preload and operating temperature load alone. The larger of the two calculated tensile forces should control the design. The maximum design bolt force should then be obtained by combining the larger direct tensile bolt force with the additional prying force. The weight is derived from the maximum or design weight of the closure lids and any cask components supported by the lids.

Review the bolt engagement lengths. If the lids are fabricated from relatively non-hardened materials, threaded inserts may be used in the closure lids to accommodate the hardened material of the bolts.

4.5.5.1.2 Fuel Basket

Verify that the calculated stress in the Fuel basket and associated welds for the various normal and off-normal condition load combinations and each stress category are within the limits of allowable stress of the stated ASME B&PV Code that the applicant cited as the design criteria. Ensure that the allowable stresses are based on the temperature of the material in the loading condition considered and determined in accordance with ASME B&PV Code.

Ensure that the applicant evaluated the buckling capacity of the cask basket materials. Acceptable guidance for this evaluation is provided in Section III of the ASME B&PV Code and NUREG/CR-6322, "Buckling Analysis of Spent Fuel Basket." issued May 1995. For this evaluation, confirm that the applicant selected the appropriate end conditions used in the buckling capacity equations on the basis of sensitivity studies. These studies can bound the range of conditions that are typically either fixed for a welded connection or free if there is no rigid connection.

Review the fuel basket design to assess the applicant's analysis of the combined effects of weight, thermal stresses, and cask-drop impact forces that could arise during spent fuel transfer and storage operations. Ensure the weight supported by the basket is the maximum or design weight of the SNF to be stored

4.5.5.1.3 Spent Fuel Assemblies and Cladding

Verify that the applicant has considered, at a minimum, dead load and internal pressure during normal condition of loading for spent fuel assemblies (SFA) and cladding and that the calculated stresses are within the allowable limits.

Verify that the SAR includes an analysis of SFA integrity for a cask-drop accident. If the analytical approach described in Chun et al. (1987) for axial buckling is used to assess fuel integrity for the cask drop accident, verify that the analysis uses the irradiated material properties and includes the weight of fuel pellets.

Alternatively, an analysis of fuel integrity that considers the dynamic nature of the drop accident and any restraints on fuel movement resulting from cask design is acceptable if it demonstrates that the cladding stress remains below yield. If a finite element analysis is performed, the analysis model may consider the entire fuel rod length with intermediate supports at each grid support (spacer). Ensure that the analysis includes irradiated material properties and the weight of fuel pellets. Coordinate with the materials reviewer (SRP Chapter 8) to verify the material properties of the irradiated fuel cladding.

4.5.5.1.4 Transfer Cask

Verify that the calculated stresses in the transfer cask components as a result of the load combinations for normal and off-normal conditions are within the limits specified in the ASME B&PV Code, or other design criteria the applicant cited. Ensure that the allowable stresses are based on the temperature of the material in the loading condition considered and determined in accordance with the ASME B&PV Code.

As a part of the transfer cask, ensure the SAR includes an analysis for the neutron shield tanks and lifting trunnions, as applicable. The appropriate factors of safety from NUREG-0612 apply to the trunnions when they are used to lift the transfer cask as a special lifting device.

Review the design of the trunnions of the transfer cask, their connections to the cask body, and the cask body in the local area around the trunnions. The design basis for the trunnions can be either nonredundant or redundant. In either case, ensure the design meets the requirements of NUREG-0612.

For a typical trunnion design, the maximum stress occurs at the base of the trunnion as a combination of bending and shear stresses. A conservative technique for computing the bending stress is to assume that the lifting force is applied at the cantilevered end of the trunnion, and that the stress is fully developed at the base of the trunnion. If other assumptions are considered, including ASME B&PV Code Section III stress limits by the finite element design analysis and slight material yielding at localized regions, ensure that the SAR includes adequate justifications.

4.5.5.1.5 Storage Overpack

The NRC has accepted the load combinations shown in Table 4-2 for an analysis of steel and reinforced concrete DSS or DSF structures that are important to safety and not within the jurisdiction of ASME B&PV Code Section III, Division 1.

Definitions of terms used in Tables 4-2 and 4-3 are as accepted by the NRC. Definitions of terms used in the load combination expressions for reinforced concrete and steel are derived from ANSI 57.9, ACI 349, AISC specifications, or other sources. Many definitions are expanded with their intended applications more fully described and implemented than in the referenced sources.

Capacities (“S” and “U” terms) and demand (factored or unfactored) loads may be loads, forces, moments, or stresses caused by such loads. Ensure that the usage is consistent among the terms used in the load combination. Units of force, rather than mass, are to be used for loads.

The load combinations defined on the basis of allowable stress apply to total stresses (that is, combined primary and secondary stresses). The load and stress factors do not change if secondary stresses are included.

Table 4-2 lists two load combinations each for reinforced concrete structures and steel structures acting during normal and off-normal conditions.

Verify that the SAR includes the thermal analysis of the storage cask on the reinforced concrete components that are not designed to permit thermal growth. Friction forces should be at the ISFSI storage pad interface.

4.5.5.2 *Other Structures, Systems, and Components Subject to NRC Review*

The NRC has accepted but does not require use of the normal and off-normal condition load combinations from Table 4-2 for steel and reinforced concrete structures that are not important to safety, including the concrete ISFSI pad that is classified as not important to the safety. If Table 4-2 is not used, the load combinations from the IBC, ASCE 7, or ACI 318, as appropriate, should be used. If load combinations other than those from Table 4-2 are used, it is not necessary to distinguish between normal, off-normal, and accident condition load combinations. The applicant can report the results of the governing load combination for the structural component in question. The NRC has accepted steel analyses that reflect allowable stress design or plastic strength design. Steel load combinations may be determined on the basis of the set of load combination expressions involving either “S” or “U” terms. Ensure the demand-to-capacity ratio for shear, axial, and bending moment at all locations in the concrete and steel structures is less than 1.0.

If the concrete ISFSI pad is important to safety, the load combinations for the pad for normal conditions listed in Table 4-2 under “Reinforced Concrete Footings” should be used. Ensure the demand-to-capacity ratio for shear, axial, and bending moment at all locations of the concrete pad is less than 1.0. In addition, ensure the soil reaction is less than the allowable bearing pressure.

Coordinate with the thermal review in Chapter 5 of this SRP to verify that the temperatures and pressures (where applicable) for other SSCs presented in the SAR, and subject to NRC approval, correspond to the same temperatures and pressures given in the thermal loads analysis.

Coordinate with the operation systems review in Chapter 3, “Principal Design Criteria Evaluation,” of this SRP to verify that the configuration of the other SSCs corresponds to the same configuration used in the various load combinations.

The information and evaluation required for these SSCs is typically to lesser levels than that required for SSCs important to safety, as described in the respective part of this section. For example, the structural capacities or design and construction codes may be stated and evaluated, but there typically is no review of structural analyses or other analyses supporting selection or assessment of projected performance.

4.5.6 Accident Conditions

Verify that the load combinations that the applicant considers to be accident or natural phenomenon conditions of loading are acceptable. Review the analysis and how the applicant’s results compare to the design criteria. The SAR may present factors of safety, stress ratios, or margins of safety. Ensure that the calculated values are less than the allowed values for different load combinations.

4.5.6.1 *Structures, Systems, and Components Important to Safety*

Review the SAR's structural analyses to assess the information regarding margins of safety or compliance with the ASME B&PV Code stress limits, overturning margins, and other design criteria as appropriate. Ensure that the applicant presented the comparisons of capacity versus demand for the various applicable loading conditions in the same manner as presented in the same terms used in the design code (e.g., type of stress). In addition, ensure the margins of safety are included on the basis of comparisons between capacity and demand for each structural component analyzed. Ensure the minimum margin of safety for any structural section of a component is included for the different load conditions.

4.5.6.1.1 Canister and Associated Welds and Bolts

Verify that the calculated stress in the canister and associated welds and bolts for each stress category, the stress allowable, stress intensity, and stress ratios are within the limits specified in the ASME B&PV Code. Ensure that the allowable stresses are based on the temperature of the material in the loading condition considered and determined in accordance with the ASME B&PV Code.

During a load drop, the canister will be subjected to compressive forces; therefore, ensure that the applicant evaluated buckling of the canister in accordance with ASME B&PV Code, Appendix F-1331.5, and NUREG/CR-6322, as applicable.

4.5.6.1.2 Fuel Basket

Verify that the applicant has considered, at a minimum, the following loading combinations on the fuel basket for the following accident conditions of loading:

- axial end drop of the transfer cask
- side drop of the transfer cask
- side drop of canister on rails in storage overpack
- side drop of the canister away from rails

During a load drop, the fuel basket will be subjected to compressive forces; therefore, ensure that the applicant evaluated buckling of the fuel basket plates in accordance with ASME B&PV Code, Appendix F-1331.5, and NUREG/CR-6322, as applicable.

4.5.6.1.3 Spent Fuel Assemblies and Cladding

Verify that the applicant has considered, at a minimum, SFA and cladding buckling during accidental side drop and corner drop of the transfer cask or storage cask. The calculated onset of buckling does not necessarily imply cladding failure. Ensure that the stress in the SFA cladding is less than the yield stress of the material. Ensure also that the maximum principal strain is less than allowable strain.

Confirm that the analytical approach used for buckling to assess fuel rod integrity for the cask drop accident uses irradiated material properties and includes the total weight of the fuel.

Alternately, the NRC accepts an analysis of fuel rod integrity that considers the dynamic nature of the drop accident and any restraints on fuel rod movement resulting from cask design. If a finite element analysis is performed, the analysis model may consider the entire fuel rod length with intermediate supports at each grid spacer. Confirm that the SAR includes the irradiated material properties and total weight of the fuel. For further guidance, see Bjorkman (2004, 2009).

4.5.6.1.4 Transfer cask

Verify that the calculated stress in the transfer cask components, the stress allowable, stress intensity, and stress ratios are within the limits specified in the ASME B&PV Code.

Confirm that the transfer cask shell and cover plates are evaluated for penetration by different missiles specified in RG 1.76. Ensure that the maximum penetration depth is not greater than the shell or cover plate thickness.

Confirm that the transfer cask, while sitting on a trailer, is evaluated for overturning from design-basis wind, seismic, and missile impact loads. Ensure the factor of safety against overturning is greater than 1.1.

4.5.6.1.5 Storage Overpack

Table 4-2 lists four load combinations for reinforced concrete structures, and nine load combinations for steel structures (six for applied stress design and three for strength design) occurring during accident conditions. For storage overpacks, ensure the SAR reflects the accident condition loads as weight of the storage overpack, live load, thermal loads, earthquake or seismic loads, accident loads from load drop, and tornado or hurricane loads.

Ensure that the demand-to-capacity ratio for shear, axial force, and bending moment for different individual components is less than 1.0.

Ensure that the applicant evaluated the transfer overpack or cask for overturning and sliding from seismic loads, tornado wind loads, combined tornado effects (wind force in combination with tornado generated missile force), and flood loads. The load combinations from Table 4-2 should be used for this evaluation.

4.5.6.2 *Other Structures, Systems, and Components*

The NRC has accepted but does not require use of the accident condition load combinations from Table 4-2 for steel and reinforced concrete structures that are not important to safety, including the concrete ISFSI pad that is classified as not important to the safety. If Table 4-2 is not used, ensure the analysis uses load combinations from the IBC, ASCE 7 or ACI 318, as appropriate. If load combinations other than those from Table 4-2 are used, it is not necessary to distinguish between normal, off-normal, or accident condition load combinations. The applicant can report the results of the governing load combination for the structural component in question. The NRC has accepted steel analyses that reflect allowable stress design or plastic strength design. Steel load combinations may be determined on the basis of the set of load combination expressions involving either "S" or "U" terms. Ensure that the demand-to-capacity ratio for shear, axial, and bending moment at all locations in the concrete and steel structures is less than 1.0.

If the concrete ISFSI pad is important to safety, ensure the SAR reflects the load combinations for the pad for normal conditions listed in Table 4-2 under the reinforced concrete footings column.

Ensure the demand-to-capacity ratio for shear, axial, and bending moment at all locations of the concrete pad is less than 1.0. In addition, ensure that the modulus of subgrade

Table 4-2 Loads and Their Descriptions

Symbol	Capacity or Load	Capacity or Load (or Demand) Description
S	Steel allowable strength design (ASD)	Strength of a steel section, member, or connection computed in accordance with the "allowable stress method" of ANSI/AISC 360.
S_v	Steel ASD shear	Shear strength of a section, member, or connection computed in accordance with the "allowable stress method" of ANSI/AISC 360.
U_s	Steel plastic strength	Strength (capacity) of a steel section, member, or connection computed in accordance with the "plastic strength method" of ANSI/AISC 360.
U_c	Reinforced concrete available strength	Minimum available strength (capacity) of reinforced concrete section, member, or embedment to meet the load combination, calculated in accordance with the requirements and assumptions of ACI 349 and, after application of the strength reduction factor, Φ, as defined and prescribed in Section 9.2, "Design Strength," of ACI 349. If strength may be reduced during the design life by differential settlement, creep, or shrinkage, those effects should be incorporated in the dead load, D (instead of by subtraction from minimum available strength). Reinforced concrete footing and foundation sections whose demand loads are dominated by the maximum soil reaction may be designed and evaluated using U_f.
U_f	Strength of foundation sections	Minimum available strength of reinforced concrete footing and foundation sections whose demand loads are dominated by the maximum soil reaction, and after the strength reduction factor, Φ, as defined and prescribed in Section 9.3 of ACI 349 is applied. Structural elements interface with columns, walls, grade beams, or footings and foundations should be evaluated by using load factors and load combinations for U_c. These interface elements include anchor bolts and other embedments, dowels, lugs, keys, and reinforcing extended into the footing or foundation.
U_g	Soil reaction or pile capacity	Minimum available soil reaction or pile capacity is determined by foundation analysis (expressed in a SAR for approval of a cask system as a required minimum for the cask system design). U_g is derived using the same load factors and load combinations as shown for determination of U_c.
O/S	Overturning or sliding resistance	Required minimum available resistance capacity of structural unit against both overturning and sliding. Capacities for resistance of overturning and sliding are checked against the factored load combination separately, although the minimum margins of safety may occur concurrently. O/S is not determined by strength capacities of structural elements. Stress or strength demands resulting from an overturning or sliding situation are evaluated in load combinations involving S, S_v, U_s, U_c, and U_f.
D	Dead load	Dead load of the structure and attachments including permanently installed equipment and piping. The weight and

Symbol	Capacity or Load	Capacity or Load (or Demand) Description
		static pressure of stored fluids may be included as dead loads when these are accurately known or enveloped by conservative estimates. Loads resulting from differential settlement, creep, or shrinkage, if they produce the most adverse loading conditions, are included in dead load. If differential settlement, creep, or shrinkage would reduce the combined loads, they should be neglected. D includes the weight of soil vertically over a footing or foundation for the purposes of determining U_g, U_f, and O/S. Regardless of the load combination factor applied, D is to be varied by +5 percent if that produces the most adverse loading condition.
L	Live loads	Live loads, including equipment (such as a loaded storage cask) and piping not permanently installed, and all loads other than dead loads that might be experienced that are not separately identified and used in the load combination, and that are applicable to the situation addressed by the load combination. Typically includes the gravity and operational loads associated with handling equipment and routine snow, rain, ice, and wind loads, and normal and off-normal impacts of equipment. Loads attributable to piping and equipment reactions are included. Depending on the case being analyzed, may include normal or off-normal events not separately identified, as may be caused by handling (not including drop), equipment or instrument malfunction, negligence, and other manmade or natural causes. Live loads attributable to casks with stored fuel need only be varied by credible increments of loading an individual cask. Live loads attributable to multiple casks should be varied for the presence and positioning of one or more cask(s), as necessary, and varied to determine the lowest margins of safety.
L	Live load for precast structures before final integration is in place	Live loads for precast structures should consider all loading and restraint conditions from initial fabrication to completion of the structure including form removal, storage, transportation, and erection. The NRC is concerned with the analysis of loading of reinforced concrete structures before use to the extent that the structures should not have suffered hidden damage from construction live loads, thereby jeopardizing the capacity of the structures when in use. If the damage would be visibly obvious before installation, analysis of capacity versus precompletion demands is not required.
DB	Design-basis (accident) loads	Design-basis loads are controlling bounds for the following external event estimates: • Extreme credible natural events to be used for deriving design bases that consider historical data or rated parameters, physical data, or analysis of upper limits of the physical processes involved. • Extreme credible external man-induced events used for deriving design bases on the basis of analysis of human activity in the region, taking into account the site characteristics and associated risks.

Symbol	Capacity or Load	Capacity or Load (or Demand) Description
		Design-basis loads include credible accidents and extreme natural phenomena. Presumption of concurrent, independent accidents or severe natural phenomena producing compounding design-basis loads is not required. Capacity to resist design-basis loads can be assumed to be that of a structure that has not been degraded by previous design-basis loads unless significant degradation in structural capacity may credibly occur and remain undetected. The retrievability of individual fuel assemblies is not required for design-basis accident conditions that include natural phenomena hazards effects.
T	Thermal loads	Thermal loads, including loads associated with normal condition temperatures, temperature distributions, and thermal gradients within the structure; expansions and contractions of components; and restraints to expansions and contractions with the exception of thermal loads that are separately identified and used in the load combination. Thermal loads should presume that all loaded fuel has the maximum thermal output allowed at the time of initial loading in the cask system. Thermal loads should be determined for the most severe of both steady-state and accident conditions. For multiple cask storage facilities, thermal loads should be determined for the worst-case loadings on potentially critical sections (e.g., all in place, only one cask in place, alternating casks in place).
T_a	Accident condition thermal loads	Thermal loads produced directly or as a result of off-normal or design-basis accidents, fires, or natural phenomena. (Note: Although off-normal and design-basis thermal loads are treated the same in the load combinations, there is a distinction between off-normal and design-basis temperature limits for concrete. Off-normal temperature limits are the same as for normal conditions.) For multiple cask storage facilities, thermal loads should be determined for the worst-case loadings on potentially critical sections.
A	Accident condition loads	Loads attributable to the direct and secondary effects of an off-normal or design-basis accident, as could result from an explosion, crash, drop, impact, collapse, gross negligence, or other man-induced occurrences, or from severe natural phenomena not separately defined. Loads attributable to direct and secondary effects may be assumed to be non-concurrent unless they might be additive. The capacity for resistance to the demand resulting from secondary effects would be that residual capacity following any degradation caused by the direct effect.
H	Lateral soil pressure	Loads caused by lateral soil pressure, as would exist in normal, off-normal, or design-basis conditions corresponding to the load combination used. H includes lateral pressure resulting from ground water, the weight of the earth, and loads external to the structure transmitted to the structure by lateral earth pressure (not including earthquake loads, which are included in E). H does not include soil reaction associated with attempted lateral movement of the structure or structural element in contact with the earth.
G	Loads attributable to soil reaction	Used only in load combinations for footing and foundation structural sections for which demand is limited by the soil reactions. G represents loads attributable to the maximum soil reaction (horizontal (passive pressure limit) and vertical (soil or

Symbol	Capacity or Load	Capacity or Load (or Demand) Description
		pile bearing limit) that would exist in normal, off-normal, or design-basis conditions corresponding to the load combination used. G is a function of U_g (i.e., G = f (U_g)).
W	Wind loads	Wind loads produced by normal and off-normal maximum winds. Pressure resulting from wind and with consideration of wind velocity, structure configuration, location, height above ground, gusting, importance to safety, and elevation may be calculated as provided by ASCE 7.
W_t	Tornado loads	Loads attributable to wind pressure and wind-generated missiles caused by the design-basis tornado or design-basis wind (for sites where design-basis wind rather than tornado produces the most severe pressure and missile loads). Pressure resulting from wind velocity and elevation may be calculated as provided for these factors in ASCE 7. Tornado wind velocity or pressure does not have to be increased for structure importance, gusting, location, height above ground, or importance to safety (these do apply for design-basis wind).
E	Earthquake loads	Loads attributable to the direct and secondary effects of the design earthquake.
F	Flood loads	Loads attributable to the static and dynamic effects of a flooding event. This includes flooding caused by severe and extreme natural phenomena (e.g., seismic, tsunamis, storm surges), dam failure, fire suppression, and other accidents.

NOTE: If any load reduces the effects of the combination of the other loads and that load would always be present in the condition of the specific load combination, the net coefficient (factor) for that load should be taken as 0.90. If the load is not always present, the coefficient for that load should be taken as zero. Each load that may not always be present in the load combinations is to be varied from 0 to 100 percent to simulate the most adverse loading conditions (to the extent of proving that the lowest margins of safety have been determined).

Table 4-3 Load Combinations for Steel and Reinforced Concrete Nonconfinement Structures

Load Combination	Acceptance Criteria
Reinforced Concrete Structures—Normal Events and Conditions	
U_c > 1.4 D + 1.7 L	Capacity/demand >1.00 for all sections.
U_c > 1.4 D + 1.7 (L + H)	Capacity/demand >1.00 for all sections.
Reinforced Concrete Structures—Off-Normal Events and Conditions	
U_c > 1.05 D + 1.275 (L + H + T)	Capacity/demand >1.00 for all sections.
U_c > 1.05 D + 1.275 (L + H + T + W)	Capacity/demand >1.00 for all sections.
Reinforced Concrete Structures—Accidents and Conditions	
U_c > D + L + H + T + (E or F)	Capacity/demand >1.00 for all sections.
U_c > D + L + H + T + A	Capacity/demand >1.00 for all sections. An overturning accident for a cask in transfer or in separate storage on a pad is to be assumed unless more severe overturning also occurs as a result of a natural phenomenon.
U_c > D + L + H + T_a	Capacity/demand >1.00 for all sections.
U_c > D + L + H + T + (W_t or W_h)	The load combination (capacity/demand >1.00 for all sections) should be satisfied without missile loadings. Missile loadings are additive (concurrent) to the loads caused by the wind

Load Combination	Acceptance Criteria
	pressure and other loads; however, local damage may be permitted at the area of impact if there will be no loss of intended function of any structure important to safety.
Reinforced Concrete Footings/Foundations—Normal Events and Conditions	
U_f > D + (L + G)	Capacity/demand >1.00 for all sections. For footing and foundation sections with load limited by soil reaction.
U_f > D + (L + H+ G)	Capacity/demand >1.00 for all sections. For footing and foundation sections with load limited by soil reaction.
Reinforced Concrete Footings/Foundations—Off-Normal Events and Conditions	
U_f > D + (L + H + T + G)	Capacity/demand >1.00 for all sections. For footing and foundation sections with load limited by soil reaction.
U_f > D + (L + H + T + W + G)	Capacity/demand >1.00 for all sections. For footing and foundation sections with load limited by soil reaction.
Reinforced Concrete Footings/Foundations—Accident Events and Conditions	
U_f > D + L + H + T + E + G	Capacity/demand >1.00 for all sections. For footing and foundation sections with load limited by soil reaction.
U_f > D + L + H + T + A + G	Capacity/demand >1.00 for all sections. For footing and foundation sections with load limited by soil reaction.
U_f > D + L + H + T_a + G	Capacity/demand >1.00 for all sections. For footing and foundation sections with load limited by soil reaction.
U_f > D + L + H + T + (W_t or W_h) + G	Capacity/demand >1.00 for all sections. For footing and foundation sections with load limited by soil reaction.
U_f > D + L + H + T + F + G	Capacity/demand >1.00 for all sections. For footing and foundation sections with load limited by soil reaction.
Steel Structures Allowable Stress Design—Normal Events and Conditions	
(S and S_v) > D + L	Factored strength/demand >1.00 for all sections.
(S and S_v) > D + L + H	Factored strength /demand >1.00 for all sections.
Steel Structures Allowable Stress Design—Off-Normal Events and Conditions	
1.3 (S and S_v) > D + L + H + W	Factored strength /demand >1.00 for all sections.
1.5 S > D + L + H + T + W	Factored strength/demand >1.00 for all sections. Thermal loads may be neglected when analysis shows that they are secondary and self-limiting in nature, and when the material is ductile.
1.4 S_v > D + L + H + T + W	Factored strength/demand >1.00 for all sections. Thermal loads may be neglected when analysis shows that they are secondary and self-limiting in nature, and when the material is ductile.
Steel Structures Allowable Stress Design—Accidents and Conditions	
1.6 S > D + L + H + T + (E or W_t or W_h or F)	Factored strength/demand >1.00 for all sections. Thermal loads may be neglected when analysis shows that they are secondary and self-limiting in nature, and when the material is ductile.
1.4 S_v > D + L + H + T + (E or W_t or W_h or F)	Factored strength (allowable stress design)/demand >1.00 for all sections. Thermal loads may be neglected when analysis shows that they are secondary and self-limiting in nature, and when the material is ductile.

Load Combination	Acceptance Criteria
1.7 S > D + L + H + T + A	Factored strength/demand >1.00 for all sections. Thermal loads may be neglected when analysis shows that they are secondary and self-limiting in nature, and when the material is ductile.
1.4 S_v > D + L + H + T + A	Factored strength/demand >1.00 for all sections. Thermal loads may be neglected when analysis shows that they are secondary and self-limiting in nature, and when the material is ductile.
1.7 S > D + L + H + Ta	Factored strength/demand >1.00 for all sections.
1.4 S_v > D + L + H + Ta	Factored strength/demand >1.00 for all sections.
Steel Structures Plastic Strength Design—Normal Events and Conditions	
U_s > 1.7 (D + L)	Plastic capacity/demand >1.00 for all sections.
U_s > 1.7 (D + L + H)	Plastic capacity/demand >1.00 for all sections.
Steel Structures Plastic Strength Design—Off-Normal Events and Conditions	
U_s > 1.3 (D + L + H + W)	Plastic capacity/demand >1.00 for all sections.
U_s > 1.3 (D + L + H + T + W)	Plastic capacity/demand >1.00 for all sections. Thermal loads may be neglected when analysis shows that they are secondary and self-limiting in nature, and when the material is ductile.
Steel Structures Plastic Strength Design—Accidents and Conditions	
U_s > 1.1 (D + L + H + T + (E or W_t or W_h or F))	Plastic capacity/demand >1.00 for all sections. Thermal loads may be neglected when analysis shows that they are secondary and self-limiting in nature, and when the material is ductile. The load combination (capacity/demand >1.00 for all sections) should be satisfied without missile loadings. Missile loadings are additive (concurrent) to the loads caused by the wind pressure and other loads; however, local damage may be permitted at the area of impact if there will be no loss of intended function of any structure important to safety.
U_s > 1.1 (D + L + H + T + A)	Plastic capacity/demand >1.00 for all sections. An overturning accident for a cask in transfer or in separate storage on a pad is to be assumed unless more severe overturning also occurs as a result of a natural phenomenon. Thermal loads may be neglected when analysis shows that they are secondary and self-limiting in nature, and when the material is ductile.
U_s > 1.1 (D + L + H + T_a)	Plastic capacity/demand >1.00 for all sections.
Overturning and Sliding—Normal and Off-Normal Events and Conditions	
O/S ≥ 1.5 (D + H)	Capacity/demand ≥1.00 for structure to be satisfied for both overturning and sliding.
Overturning and Sliding—Accidents and Conditions	
O/S ≥ 1.1 (D + H + E or F)	Capacity/demand ≥1.00 for structure to be satisfied for both overturning and sliding.
O/S ≥ 1.1 (D + H + W_t or W_h)	Capacity/demand ≥1.00 for structure to be satisfied for both overturning and sliding.

4.6 Evaluation Findings

The structural evaluation must provide reasonable assurance that the DSF or DSS will allow safe storage of SNF. The reviewer prepares evaluation findings on satisfaction of the regulatory requirements relating to the design and structural evaluation of the DSF or DSS as identified in Section 4.4 of this SRP. Based on the review of the applicant's description, proposed design criteria, appropriate use of material properties, and adequate structural analysis of the two categories of SSCs (important to safety or not important to safety, as applicable), the staff concludes that the SSCs are in conformance with NRC regulations. Because the regulatory requirements are different for a specific license and a general license, the findings for each of these license types are listed separately. Ensure the SER addresses each acceptance criteria provided in Section 4.4 of this SRP similar to the following (finding numbering is for convenience in referencing within the SRP and SER), and that the SER evaluation provides clear bases for any regulatory conclusions:

Specific License **(SL)**

F4.1 The SAR and docketed materials adequately describe the ISFSI structures, and therefore meet the requirements in 10 CFR 72.24(b) with respect to technical information.

F4.2 The SAR and docketed materials describe the design of the ISFSI structures in sufficient detail to support findings in 10 CFR 72.40, "Issuance of License," for the term requested in the application, including the design criteria pursuant to Subpart F, the design bases, and the relation of the design to the design criteria, utilize applicable codes and standards, and therefore meet the requirements in 10 CFR 72.24(c)(1), 10 CFR 72.24(c)(2), and 10 CFR 72.24(c)(4) with respect to technical information.

F4.3 The SAR and docketed material contain information relative to materials of construction, general arrangement, dimensions of principal structures, and descriptions of all SSCs important to safety in sufficient detail to support a finding that the ISFSI will satisfy the design bases with an adequate margin of safety, and therefore meets the requirements in 10 CFR 72.24(c)(3) with respect to technical information.

F4.4 The SAR and docketed material contain an analysis and evaluation of the design and performance of SSCs important to safety, with the objective of assessing the impact on public health and safety resulting from operation of the ISFSI, and therefore meet the requirements in 10 CFR 72.24(d) with respect to technical information.

F4.5 The SAR identifies the SSCs important to safety whose functional adequacy or reliability had not been demonstrated for that purpose or cannot be demonstrated by reference to performance data in related applications or to widely accepted engineering principles, and the applicant has provided a satisfactory schedule showing how safety questions will be resolved before the initial receipt of SNF, HLW, or reactor-related GTCC waste, as appropriate, for storage at the ISFSI, and therefore meets the requirements in 10 CFR 72.24(i).

F4.6 The SAR and docketed materials adequately describe the design criteria for the SSCs important to safety and other SSCs, and therefore meet the requirements in 10 CFR 72.120(a).

F4.7 Each SSC important to safety is designed to the quality standards commensurate with the importance to safety of the function to be performed, and therefore meets the requirements in 10 CFR 72.122(a).

F4.8 The SSCs important to safety are designed to withstand the normal and off-normal conditions associated with the site and can withstand postulated accidents, and therefore meet the requirements in 10 CFR 72.122(b)(1).

F4.9 The SSCs important to safety are designed to withstand the natural phenomena associated with the site without impairing their ability to perform their intended safety functions (with consideration for the most severe natural phenomena reported for the site and in the appropriate combination of normal and accident conditions), and therefore meet the requirements in 10 CFR 72.122(b)(2)(i).

F4.10 All ISFSI structures are designed to prevent massive collapse or dropping of heavy objects onto an SSC important to safety, and therefore meet the requirements in 10 CFR 122(b)(2)(ii).

F4.11 SSCs important to safety are designed and located to continue to perform their safety functions effectively under credible fire and explosion exposure conditions, and therefore meet the requirements in 10 CFR 72.122(c).

F4.12 SSCs important to safety are not shared between the ISFSI and other facilities, or the SAR indicates that such sharing does not impair the capability of either facility to perform its safety functions, including the ability to return to a safe condition in the event of an accident, and therefore meet the requirements in 10 CFR 72.122(d).

F4.13 Storage systems are designed to allow ready retrieval of SNF, HLW, and reactor-related GTCC waste for further processing or disposal, and therefore meet the requirements in 10 CFR 72.122(l).

F4.14 SNF handling, packaging, transfer, and storage systems are designed to ensure subcriticality, in that at least two unlikely, independent, and concurrent or sequential changes must occur before a nuclear criticality accident ensues. The margins of safety of these systems are adequate for the nature of the immediate environment under accident conditions, and therefore meet the requirements in 10 CFR 72.124(a).

F4.15 SSCs important to safety are designed to provide favorable geometry and permanently fixed neutron-absorbing materials, and therefore meet the requirements in 10 CFR 72.124(b).

F4.16 SSCs important to safety that contain SNF, HLW, reactor-related GTCC waste, and other related radioactive waste are designed to ensure adequate safety with respect to suitable shielding and confinement under normal and accident conditions, and therefore meet the requirements in 10 CFR 72.128(a)(2) and 10 CFR 72.24(a)(3).

Certificate of Compliance **(CoC)**

F4.17 SNF handling, packaging, transfer, and storage systems are designed to ensure subcriticality, in that at least two unlikely, independent, and concurrent or sequential changes must occur before a nuclear criticality accident ensues. The margins of safety of these systems are adequate for the nature of the immediate environment under accident conditions, and therefore meet the requirements in 10 CFR 72.124(a).

F4.18 SSCs important to safety are designed to provide favorable geometry or permanently fixed neutron-absorbing materials, and therefore meet the requirements in 10 CFR 72.124(b).

F4.19 The design bases and design criteria are provided for SSCs important to safety that meet the requirements in 10 CFR 72.236(b).

F4.20 The SNF storage cask is designed so that the SNF is maintained in a subcritical condition under credible conditions, and therefore meets the requirement in 10 CFR 72.236(c).

F4.21 The radiation shielding and confinement features are sufficient to meet the requirements of 10 CFR 72.124(a), 10 CFR 72.124(b), and 10 CFR 72.236(d).

F4.22 The SNF storage cask is designed to provide redundant sealing of confinement systems, and therefore meets the requirements in 10 CFR 72.236(e).

F4.23 The SNF storage cask is designed to store the SNF safely for the term proposed in the application, and therefore meets the requirements in 10 CFR 72.236(g).

F4.24 The SNF storage cask is compatible with wet or dry SNF loading and unloading facilities, and therefore meets the requirements in 10 CFR 72.136(h).

F4.25 The SNF storage cask and its systems important to safety have been evaluated by appropriate test or other acceptable means and have demonstrated that they will reasonably maintain confinement or radioactive material under normal, off-normal, and credible accident conditions, and therefore meet the requirements in 10 CFR 72.236(l).

F4.26 To the extent practicable, the SAR has given consideration to the design of the SNF storage cask for compatibility with the removal of the stored SNF from a reactor site, transportation, and ultimate disposition by the Department of Energy, and therefore meets the requirements in 10 CFR 72.236(m).

Provide a summary statement similar to the following:

> The staff concludes that the structural properties of the SCCs of the [cask designation] are in compliance with 10 CFR Part 72 and the applicable design and acceptance criteria have been satisfied. The evaluation of the structural properties provides reasonable assurance that the [cask designation] will allow safe storage of SNF for a licensed (certified) life of years. This finding is reached on the basis of a review that considered the regulation itself, appropriate regulatory guides, applicable codes and standards, and accepted engineering practices.

4.7 References

10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities."

10 CFR Part 72, "Licensing Requirements for the Independent Storage of Spent Nuclear Fuel, High-Level Radioactive Waste, and Reactor-Related Greater Than Class C Waste."

American Concrete Institute (ACI) 318-11, "Building Code Requirements for Structural Plain Concrete and Commentary," 2011.

ACI 349-06, "Code Requirements for Nuclear Safety-Related Concrete Structures and Commentary," 2006.

AISC 303-16, "Code of Standard Practice for Steel Buildings and Bridges," 2016.

American National Standards Institute (ANSI)/AISC 360-10, "Specification for Structural Steel Buildings," 2010.

American Society of Mechanical Engineers (ASME) Boiler and Pressure (B&PV) Code, 2015.
- Section III, "Rules for Construction of Nuclear Facility Components."
  - Division 1, "Metallic Components"; Subsections NB, NC, ND, NF, and NG
  - Appendix F
  - Division 3, "Containments for Transportation & Storage of Spent Nuclear Fuel and High Level Radioactive Material & Waste" (no NRC position on this has been established)
- Section VIII, "Rules for Construction of Pressure Vessels"
- Appendix I
- Appendix III

ANSI/ASME B16.34, "Valves–Flanged, Threaded and Welding End," 2013.

ANSI/ASME B31.1, "Power Piping," 2016.

ANSI/ASME NQA-1, "Quality Assurance Program for Nuclear Facility Applications," 2015.

ANSI/American Nuclear Society (ANS) N690-12, “Nuclear Facilities—Steel Safety-Related Structures for Design Fabrication and Erection.”

ANSI/ANS 57.9-1992, “Design Criteria for an Independent Spent Fuel Storage Installation (Dry Storage Type).”

American Petroleum Institute (API) 620, “Recommended Rules for Design and Construction of Large, Welded, Low-Pressure Storage Tanks,” 2013.

American Society for Testing and Materials (ASTM) C33, “Standard Specification for Concrete Aggregates.”

ASTM C150, “Standard Specification for Portland Cement.”

American Society of Civil Engineers (ASCE) 4-98, “Seismic Analysis of Safety-Related Nuclear Structures,” 1998.

ASCE 7-10, “Minimum Design Loads for Buildings and Other Structures,” 2010.

American Water Works Association (AWWA) D100, “Welded Carbon Steel Tanks for Water Storage,” 2011.

American Welding Society (AWS) D1.1, “Structural Welding Code—Steel,” 2011.

Bjorkman, G.S. et al., 2001, “Influence of ISFSI Design Parameters on the Seismic Response of Dry Storage Casks,” Transactions SMIRT 16, Washington DC, August 2001, Paper #1601.

Bjorkman, G., 2004, “The Buckling of Fuel Rods in Transportation Casks under Hypothetical Accident Conditions,” 14th International Symposium on the Packaging and Transportation of Radioactive Materials (PATRAM-2004), Berlin, Germany.

Bjorkman, G., 2009, "The Buckling of Fuel Rods under Inertia Loading," Proceedings of the ASME Pressure Vessel and Piping Conference (PVP 2009), Prague, Czech Republic.

Chun, R., Witte, M., and Schartz, M., “Dynamic Impact Effects on Spent Fuel Assemblies,” UCID-21246, Lawrence Livermore National Laboratory, October 20, 1987.

Cottrell, W.B., and A.W. Savolainen, “U.S. Reactor Containment Technology,” in ORNL-NSIC-5, Volume 1, Chapter 6, Oak Ridge National Laboratory, August 1965.

Washa, G.W., J.C. Seamann, and S. M. Cramer, "Fifty Year Properties of Concrete," pp. 367-371 in *Materials Journal,* 86(4), American Concrete Institute, Detroit, Michigan, July-August 1989.

Hoerner, S.F., *Fluid-Dynamics Drag*, Hoerner Fluid Dynamics, Brick Town, New Jersey, 1965.

International Code Council, International Building Code, 2015.

NUREG-0612, “Control of Heavy Loads at Power Plants: Resolution of Generic Technical Activity A-36,” July 1980 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML070250180).”

NUREG-0800, “Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition.”

NUREG/CR-6007, “Stress Analysis of Closure Bolts for Shipping Casks,” Kaiser Engineering, January 1993.

NUREG/CR-6322, “Buckling Analysis of Spent Fuel Basket,” UCRL-ID-119697, Lawrence Livermore National Laboratory, May 1995.

NUREG/CR-6424, "Report on Aging of Nuclear Power Plant Reinforced Concrete Structures," March 1996.

NUREG/CR-6608, “Summary and Evaluation of Low-Velocity Impact Tests of Solid Steel Billet onto Concrete Pads,” Lawrence Livermore National Laboratory, February 1998.

NUREG/CR-6865, “Parametric Evaluation of Free Standing Spent Fuel Dry Cask Storage Systems,” Sandia National Laboratories, February, 2005.

NUREG/CR-7004, “Technical Basis for Regulatory Guidance on Design-Basis Hurricane-Borne Missile Speeds for Nuclear Power Plants,” February 2011.

Regulatory Guide 1.26, “Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants.”

Regulatory Guide 1.29, “Seismic Design Classification.”

Regulatory Guide 1.59, “Design Basis Floods for Nuclear Power Plants.”

Regulatory Guide 1.60, “Design Response Spectra for Seismic Design of Nuclear Power Plants.”

Regulatory Guide 1.61, “Damping Values for Seismic Design of Nuclear Power Plants.”

Regulatory Guide 1.76, “Design-Basis Tornado and Tornado Missiles for Nuclear Power Plants.”

Regulatory Guide 1.92, “Combining Modal Responses and Spatial Components in Seismic Response Analysis.”

Regulatory Guide 1.102, “Flood Protection for Nuclear Power Plants.”

Regulatory Guide 1.136, “Design Limits, Loading Combinations, Materials, Construction, and Testing of Concrete Containments.”

Regulatory Guide 1.221, “Design-Basis Hurricane and Hurricane Missiles for Nuclear Power Plant”.

Regulatory Guide 3.73, “Site Evaluations and Design Earthquake Ground Motion for Dry Cask Independent Spent Fuel Storage and Monitored Retrievable Storage Installation.”

Roark, R.J., *Formulas for Stress and Strain*, McGraw Hill, 1965.

Tripathi, B.P., “Examining Opposing Requirements for ISFSI Pads When Evaluating Cask Tipover,” U.S. Nuclear Regulatory Commission, 2007.

APPENDIX 4A COMPUTATIONAL MODELING SOFTWARE TECHNICAL REVIEW GUIDANCE

4A.1 Computational Modeling Software Application

The staff does not endorse the use of any specific type or code vendor of computational modeling software (CMS). Any appropriate CMS application could be used for analyses of cask or package components; however, for any CMS to demonstrate that a particular cask design satisfies regulatory requirements, the applicant should demonstrate adequate validation of that CMS. Descriptions of CMS validations can be contained within a given application or incorporated by reference.

Verify that the SAR or related documentation (such as proprietary calculation packages or benchmark reports) provides the following information:

- details of the methodology used to assemble the computational models and the theoretical basis of the program used

- a description of benchmarking against other codes or validation of the CMS against applicable published data or other technically qualified and relevant data that are appropriately documented

- standardized verification problems analyzed using the CMS, including comparison of theoretically predicted results with the results of the CMS

- release version and applicable platforms

Once the information described above has been docketed, it need not be submitted with each subsequent application but can be referred to in subsequent safety analysis reports (SARs) or related documents. If an applicant changes its analysis methodology or changes the type or vendor of the CMS used, the applicant should submit either a revision of previously submitted information or include a clear explanation of the methodology changes, and their effects on the analysis in question, in subsequent SAR submittals.

4A.2 Modeling Techniques and Practices

The staff may need to verify the modeling techniques and practices the applicants used to demonstrate adequacy of the model.

Verify that the CMS and the options the applicant used are appropriate for adequately capturing the behavior of a cask, package, or any components.

The original application should include relevant input and results files or an equivalent detailed model description and output.

4A.3 Computer Model Development

Verify that the computer model used for the analysis is adequately described, either in the SAR or in other documentation; is geometrically representative of the cask design being analyzed; has addressed how material and manufacturing uncertainties might affect the analysis; has appropriate boundary conditions; and has no significant analysis errors.

Verify that the model description includes an adequate basis for the selection of parameters and components, as appropriate, used in the analysis model (e.g., the reason a particular element type was applied in the analysis model).

Verify that the models sufficiently represent cask or package geometry and that adequate justification is provided for simplifications used. Models created with CMS are often simplified to reduce computer processing time. Models can often omit geometric details or use homogenized or smeared material properties to represent complex geometry or material combinations and still retain analytic accuracy. If smeared or homogenized properties are used, verify that the applicant has provided adequate justification for this approach, as the response of the problem can be dramatically altered.

Verify that the applicant has discussed how manufacturing or assembly tolerances and contact resistances will affect the analyses that have been conducted, if at all, in both the structural and thermal disciplines. Verify that the applicant has described how tolerances or contact resistances are accounted for, if applicable, in the cask or package analysis models that are submitted for review.

Verify that the applicant has provided a general discussion of how error, warning, or advisory messages generated by the software affect the analysis result (if applicable). When processing a computer model developed using CMS, the software will frequently provide error, warning, or advisory messages indicating a possible problem with the model that may or may not be sufficient to terminate processing. If the error or warning function has been disabled during processing, ensure the applicant provides an explanation of why this is appropriate.

Verify that, within the specific disciplines, the dimensions and physical units used in the models developed are clearly labeled and mutually consistent. The fundamental units of time, mass, and length should be clearly identified. All other physical units derived should be consistent with the basic units adopted. For example, if the unit of length is the millimeter, time in milliseconds, and mass in gram, then the mechanical force should have units of Newton, energy in millijoule, and stress in megapascal. Verify that the input parameters are expressed in the units as assigned. If an applicant chooses not to adopt this uniformity of units, the appropriate conversion should be applied before processing input into CMS. Similar assurances should be provided for the output for the analysis solution.

4A.4 Computer Model Validation

Verify that model validation done with applicable experiments or testing is properly documented and appropriate references are provided. For example, an analytical model's ability to capture relevant model output such as g-loads, and plastic deformations could be demonstrated by comparing the physical test data of a similar package that was instead drop tested.

The test data used to validate or benchmark the analytical model should be similar in regard to the expected package behavior of interest. For instance, a package with impact limiters should be used to benchmark a package that also has impact limiters. Plastic strain data used for validation, for instance, should come from areas of the package where such data are crucial or relaxant to the performance of the package such as the containment boundary. Other details to consider when benchmarking and validating physical data include whether the package is bolted or welded, and whether the response will be dominated primarily by a quasi-static, wave or impulse-type response. The data source should be readily available or included, as appropriate, in the application and should describe all the assumptions and simplifications made during physical testing so that staff can weigh its relevance to the design of interest.

4A.5 Justification of Bounding Conditions and Scenario for Model Analysis

Ensure the applicant determines the most damaging orientation and worst-case conditions for a given design and document how the analytic model was configured for the scenario. Verify that the applicant provided sufficient justification for selecting the most damaging orientation and worst-case conditions.

4A.6 Description of Boundary Conditions and Assumptions

Verify, as necessary, that the textual description included in the SAR or other documents address boundary conditions such as an unyielding surface in a drop scenario. The textual description should also include justifications and bases for such items. Confirm that the application reflects appropriate material (temperature dependent) properties.

4A.7 Description of Model Assembly

Verify that the SAR lists the types of elements used in the model along with the corresponding materials or components in which they are used in the analysis model. The reviewer should quickly be able to discern what elements and materials are associated with specific components of the analysis model.

Verify that a sufficient explanation of the logic behind the creation of each specific computer model (such as the mesh) is provided so that effective confirmatory calculations can be performed.

The applicant should provide the input files for the models used in the analysis. If input files are not provided or do not adequately describe model assembly, the applicant should provide in the appropriate SAR chapters or related documents an adequate explanation of how computer models were assembled using the CMS.

4A.8 Loads, Time Steps, and Impact Analyses

Verify that the applicant has clearly explained the loads, load combinations, and, if used by the analytical code, the load steps used in the computer model. Evaluate all loads, how they are placed on the computer models, load combinations, and, if used, the time steps applied in the analysis.

Verify that the time steps specified for the solution of the analysis are sufficiently small to accurately capture the behavior of the structures, systems, or components being modeled.

For impact analyses using software such as LS-DYNA, examine the output files for hour-glassing energy in each part of the system in addition to the package as a whole. Verify that the impact analyses output is realistic. Parts of a model should not pass each other without deformation or through one another unrealistically. Disassemble the model by component and examine them for breaches or other unseen damage. For instance, components can be perforated, but this damage may be hidden from view by other components in the model.

4A.9 Sensitivity Studies

The discussion of the general development of the computer model should cover sensitivity studies, with relevant references to examples included in the SAR or related documents.

Verify that the applicant has completed sensitivity studies for relevant CMS modeling parameters. This includes element type and mesh density, load step size, interfacing gaps or contact friction, material models and model parameters selection, and property interpolation, if applicable. For example, a mesh sensitivity study should be conducted not only for mesh density but also for mesh density and refinement in areas of thermal or structural concern or where performance of the material is crucial, such as seal areas and lid bolts. A mesh sensitivity is also needed to make sure the analysis results are mesh independent.

Verify that the results of applicable sensitivity studies are clearly described in the SAR or related documentation and can be independently verified, if necessary.

Verify that the applicant's documentation includes at least a brief discussion of the different models used in its mesh sensitivity studies.

4A.10 Results of the Analysis

Verify that the SAR or related document(s) includes all relevant results (tabular and computer plots) for applicable load cases and load combinations evaluated for design code compliance, and that all governing results (stresses and deformations) are clearly identified in the tables and on plots.

Verify that the results are consistent throughout the SAR, and that the correct results are used in calculations of other cask or package performance parameters (e.g., verify calculated temperatures used in the internal pressure calculation).

APPENDIX 4B POOL AND POOL CONFINEMENT FACILITIES

The pool and pool confinement facilities provide a capability that may be essential to the conduct of independent spent fuel storage installation (ISFSI) and monitored retrievable storage installation (MRS) loading for storage and unloading functions and that may be needed for retrievability. The pool and pool confinement facilities are considered to include those systems important to safety that provide for wet transfer, loading, unloading, and temporary holding or long-term storage of spent nuclear fuel (SNF), high-level radioactive waste (HLW), and other radioactive materials associated with SNF or HLW storage. Other ISFSI or MRS equipment that may be used within and outside the pool facility, or that are used for lifting or transfer within the facility but are not installed cranes or conveyance systems, are addressed as structures, systems, and components (SSCs) important to safety or “other” SSCs.

The safety function of the pool and associated equipment is to maintain the SNF assemblies in a safe and subcritical array during all credible storage conditions and to provide a safe means of loading the assemblies into shipping casks.

The ISFSI and MRS pools and pool facilities should be designed as though they were to be in constant use for in-pool storage and wet transfer for the life of the ISFSI or MRS license. However, it is anticipated that the actual use of the ISFSI or MRS pool facility may differ from the use of the SNF pool at a reactor facility. Therefore, the SAR should thoroughly describe the limited or part-time use of the pool. The use status of the pool facility may have a major impact on the generation of radioactive and other waste. The design may also need to provide for conversion to standby mode or decontamination and decommissioning while the rest of the ISFSI or MRS remains in use for dry storage.

4B.1 Description of Pool Facilities

Regulations at Title 10 of the *Code of Federal Regulations* (10 CFR) 72.24(a), 72.24(b), 72.40(a)(3), and 72.106(a)(b)(c) address the descriptive information to be included in a license application. The application must describe pool facilities in sufficient detail to support a detailed review and evaluation. This includes text, descriptions, drawings, flow diagrams, figures, tables, and specifications to fully define the systems and features of the pool facilities.

The NRC accepts use of existing pool and pool confinement facilities that are licensed under 10 CFR Part 50, “Domestic Licensing of Production and Utilization Facilities,” for an ISFSI or MRS, if concerns for possible sharing of SSCs between separately licensed facilities are satisfied (10 CFR 72.3, “Definitions,” (included with the definition of an ISFSI), 72.24(a), 72.40(a)(3), and 72.122(d)). The existing pool and pool confinement facilities may continue to be licensed under 10 CFR Part 50, or they may be relicensed as elements of a wet storage or dry storage ISFSI, as appropriate.

4B.2 Design Criteria

The regulatory requirements given in 10 CFR 72.24(c)(1), (c)(2), and (c)(4); 10 CFR 72.40(a)(1); 10 CFR 72.120(a)(b); 10 CFR 72.122 (a)(b)(c)(d)(f)(g)(h)(i)(j)(k)(l); 10 CFR 72.128(a)(b); 10 CFR 72.236(b)(e)(f)(g)(k) identify acceptable design criteria.

Design criteria for important to safety facilities in 10 CFR Part 72, “Licensing Requirements for the Independent Storage of Spent Nuclear Fuel, High-Level Radioactive Waste, and Reactor-Related

Greater Than Class C Waste," are fully applicable to pool and pool confinement facilities. Pool and pool confinement facilities should meet the criteria for structural integrity for similar facilities constructed at a power reactor, which must comply with 10 CFR Part 50. These criteria are principally as stated in 10 CFR Part 50, Appendix A, "General Design Criteria for Nuclear Power Plants," General Design Criterion (GDC) 61, "Fuel Storage and Handling and Radioactivity Control." Some portions of GDC 62, "Prevention of Criticality in Fuel Storage and Handling," and GDC 63, "Monitoring Fuel and Waste Storage" apply. GDC 2, "Design Bases for Protection Against Natural Phenomena," 4, "Environmental and Dynamic Effects Design Bases," and 5, "Sharing of Structures, Systems, and Components," apply to the design of pool facilities. See NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LRW Edition," Sections 9.1.2, "New and Spent Fuel Storage," and 9.1.3, "Spent Fuel Pool Cooling and Cleanup System," for specific acceptance criteria, which derive from 10 CFR Part 50, Appendix A.

The intended usage of the pool and pool facilities may be used in the development of design requirements. Should the intended usage be long-term storage of SNF, the NRC accepts design of elements of the pool facility in accordance with American National Standards Institute (ANSI)/American Nuclear Society (ANS) 57.2, "Design Requirements for Light Water Reactor Spent Fuel Storage Facilities at Nuclear Power Plants." Should the intended usage be short term or primarily to facilitate wet transfer operations, the NRC accepts design of elements of the pool facility in accordance with ANSI/ANS 57.7, "Design Criteria for an Independent Spent Fuel Storage Installation (Water Pool Type)." Regardless of whether ANSI/ANS 57.2 or ANSI/ANS 57.7 is used, it should be noted that 10 CFR 72.2, "Scope," requires that SNF be aged for at least 1 year after discharge from the core.

The NRC accepts design of the pool liquid containment structures, systems, and components (SSCs) as required for Quality Group B (as described in Regulatory Guide (RG) 1.26, "Quality Group Classifications and Standards for Water-, Steam-, and Radioactive Waste-Containing Components of Nuclear Power Plants") that are licensed under 10 CFR Part 50. This quality group requires design to not less than the requirements of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Valve (B&PV) Code, Section III, "Rules for Construction of Nuclear Facility Components," Division 1, Subsection NC.

The NRC accepts the design of ISFSI and MRS pool facility cooling and makeup water systems (as required) for Quality Group C, as described in RG 1.26. This quality group requires design to not less than the requirements of ASME B&PV Code Section III, Division 1, Subsection ND.

The NRC accepts the guidance for reactor facility pools provided by RG 1.13, "Spent Fuel Storage Facility Design Basis," for ISFSI and MRS pool facilities. RG 1.13 lists the following principal criteria for pool facility design:

- prevent loss of water from the pool that would uncover the radioactive material
- protect the radioactive material from mechanical damage
- provide capability for limiting the potential offsite exposures in the event of a significant release of radioactivity from the subject materials

4B.3 Review Procedures

4B.3.1 Description of Pool Facilities

Review the descriptive material in Chapter 1 of the SAR and the descriptive information in Chapter 3 of the SAR. Ensure that the text descriptions, drawing figures, tables, flow diagrams, and specifications included in the application fully define the pool facilities.

Review the description of SSCs important to safety and verify that there is enough detail to proceed with the evaluation of the structural integrity and functional suitability. The configurations should be defined by drawings and fabrication specifications. Ensure that the specifications include references to the codes that govern the design details. Verify that the combination of the drawings, specifications, appropriate codes and standards, and supporting calculations are sufficient.

A pool and pool confinement facilities involve a broader range of components and systems than the confinement structures. However, the staff anticipates a diversity of pool facilities ranging from existing conventional pools designed under 10 CFR Part 50 requirements to site-specific designs used for limited, short-duration, wet-transfer operations. The facilities may contain some of the following elements that will require verification of structural integrity:

- pool structure, structural supports, and components that form the primary hydraulic confinement, water level control, cooling, and clean-up systems, such as piping, valves, pumps, filters, monitoring stations, and feeders

- pool components that provide for positioning the radioactive materials within the pool to ensure subcriticality (racks), accessibility, and compatibility with lifting interfaces

- pool components that ensure against improper movement of transfer or storage casks during wet-loading and unloading operations

- secondary hydraulic containment that precludes releases to the surface or subsurface environment that might result from leaks or rupture of elements of the primary hydraulic containment, including equipment and floor drainage system

- SSCs associated with lifting, loading, unloading, transfer, or other handling of ISFSI or MRS vessels, transfer or transportation casks, other shielding vessels, or radioactive material to be stored

- enclosure(s) of the pool and operations that involve loading, unloading, and handling of the subject radioactive materials and other SSCs forming structural elements of the confinement boundary

- emergency power capability necessary to maintain safe conditions and monitor radioactivity

- internal waste collection or confinement, demineralized water makeup system, and compressed air system for cask dewatering system (if used)

- SSCs providing compartmentalization and secondary confinement boundaries within (or coincident with) a pool facility's tertiary confinement barrier, such as for control room, electrical and machinery rooms, cask system component holding and inspection, personnel changing and showers, personnel decontamination and monitoring, health physics, and technical and administrative spaces.

Other ISFSI or MRS equipment that may be used within and outside the pool facility or that is used for lifting or transfer within the facility, but is not installed in the facility, such as cranes or conveyance systems, is addressed as “other SSCs important to safety” or “other SSCs.”

Coordinate with the confinement review, Chapter 9 of this SRP, to verify that the SAR clearly identifies the confinement boundaries associated with the pool and pool facilities.

4B.3.2 Design Criteria

For each of the SSCs being reviewed, determine what the design criteria and design bases are from the SAR. Confirm that the design criteria comply with acceptance criteria as outlined in Section 4.5.2.2 of this SRP.

Depending on the type of usage, that is, long-term storage or short-term wet transfer, verify that the appropriate criteria are applied. ANSI/ANS 57.2 is appropriate for long-term, as well as short-term storage, whereas ANSI/ANS 57.7 may be more appropriate for short-term storage or wet-transfer operations.

Verify that the following sections of NUREG-0800 (Section 9.1.2) are adequately addressed:

- GDC 2—as it relates to structures housing the facility and that the facility is capable of withstanding the effects of natural phenomena such as earthquakes, tornadoes, and hurricanes

- GDC 4—as it relates to structures housing the facility and that the facility is capable of withstanding the effects of environmental conditions and external missiles such that safety functions are not precluded

- GDC 5—as it relates to shared SSCs

- GDC 61—as it relates to the facility design for fuel storage and handling of radioactive materials

- GDC 62—as it relates to the prevention of criticality of the fuel by means of physical systems.

4B.3.3 Material Properties

Coordinate with the thermal review, Chapter 5 of this SRP, to verify that the material properties used in the structural analysis are appropriate for the load conditions and that the appropriate temperature at which the stress limits are defined is consistent with service temperatures. For each of the SSCs being reviewed, determine what structural materials are specified (e.g., reinforced concrete, steel), and verify that the material properties conform to the accepted design codes and standards. Section 4.5.2.2 of this SRP gives references to acceptable codes. Review structural and other materials, and verify that they will produce no significant chemical or galvanic action or cause corrosion degradation that could adversely affect the safety function.

4B.3.4 Structural Analysis

Confirm that the design analysis includes codes and standards, design documentation, and design conditions for (1) the SNF storage and cask handling pools; (2) the SNF cask and fuel assembly handling systems; (3) SNF storage racks; (4) fuel pool water makeup, cooling, and cleanup systems; (5) heating, ventilating, and air conditioning equipment; (6) fuel-storage buildings; and (7) electrical power, instrumentation and control, and communications, as described in ANSI/ANS 57.2 and ANSI/ANS 57.7, as appropriate.

If ANSI/ANS 57.2 is used, verify that the SSCs meet the following GDC from Appendix A to 10 CFR Part 50:

- GDC 2—Confirm that regulatory position C.2 of RG 1.13, applicable portions of RG 1.29 and RG 1.117, and appropriate paragraphs of ANSI/ANS 57.2 are met.

- Review supporting documentation and appropriate staff confirmatory calculations and verify that position C.2 of RG 1.13 is met. Position C.2 states that the pool facility should be designed to keep tornado winds and missiles generated by tornado winds from causing significant loss of watertight integrity of the fuel storage pool and to prevent tornado-driven missiles from contacting the fuel stored in the pool.

- GDC 4—Confirm that regulatory position C.2 of RG 1.13, RG 1.115, "Protection Against Low-Trajectory Turbine Missiles," and 1.117, as well as appropriate paragraphs of ANSI/ANS 57.2 are met.

- GDC 5—Confirm that SSCs important to safety are capable of performing the required safety function.

- GDC 61—Confirm that regulatory positions C.1 and C.4 of RG 1.13 and appropriate paragraphs of ANSI/ANS 57.2 are met.

- Review supporting calculations or independent staff confirmatory calculations and verify that regulatory positions C.1 and C.4 of RG 1.13 are satisfied. Position C.1 states that the fuel storage facility, including its structures and facilities (with some exceptions in regulatory position C.6), should be designed to Category I seismic requirements. Position C.4 states that a controlled leakage building should enclose the fuel pool. It should be equipped with an appropriate ventilation and filtration system to limit the potential release of radioactive materials. Although the building does not need to be designed to withstand extremely high winds, leakage should be suitably controlled during fuel-transfer operations. The ventilation and filtration system should be based on the assumption that the cladding of all the fuel rods in one fuel bundle might be breached.

- GDC 62—Confirm that regulatory positions C.1 and C.4 of RG 1.13 and appropriate paragraphs of ANSI/ANS 57.2 are met.

- Confirm that the handling of heavy loads (e.g., a SNF storage cask or SNF shipping cask) conforms to the guidance given in NUREG-0612.

Drop of a confinement cask may include secondary effects with safety implications, such as: deformation of interior structural SSCs that may preclude ready retrievability of the stored materials, structural damage and possible rupture of the pool (without loss of coolant that would uncover the fuel), damage to radioactive materials in the pool, and damage to the transfer cask, radiation shielding, or both. Secondary effects may also involve analyses addressed under the other structural evaluation categories such as the pool and pool facilities, reinforced concrete structures, and other SSCs important to safety.

RG 1.120, "Fire Protection Guidelines for Nuclear Power Plants," provides guidance for fire protection, where applicable, to some confinement systems such as the SNF pool area.

4B.4 Evaluation Findings

F4B.1 The SAR and docketed materials adequately describe the ISFSI structures, and therefore meet the requirements in 10 CFR 72.24(b) with respect to technical information.

F4B.2 The SAR and docketed materials describe the design of the ISFSI structures in sufficient detail to support findings in 10 CFR 72.40, "Issuance of License," for the term requested in the application, including the design criteria pursuant to Subpart F, the design bases, and the relation of the design to the design criteria and utilizes applicable codes and standards, and therefore meets the requirements in 10 CFR 72.24(c)(1), (c)(2), and (c)(4) with respect to technical information.

F4B.3 The SAR and docketed material contain information relative to materials of construction, general arrangement, dimensions of principal structures, and descriptions of all SSCs important to safety in sufficient detail to support a finding that the ISFSI will satisfy the design bases with an adequate margin of safety, and therefore meets the requirements in 10 CFR 72.24(c)(3) with respect to technical information.

F4B.4 The SAR and docketed material contain an analysis and evaluation of the design and performance of SSCs important to safety, with the objective of assessing the impact on public health and safety resulting from operation of the ISFSI, and therefore meet the requirements in 10 CFR 72.24(d) with respect to technical information.

F4B.5 The SAR identifies the SSCs important to safety whose functional adequacy or reliability had not been demonstrated for that purpose or cannot be demonstrated by reference to performance data in related applications or to widely accepted engineering principles, and the applicant has provided a satisfactory schedule showing how safety questions will be resolved before the initial receipt of SNF, HLW, or reactor-related GTCC waste, as appropriate, for storage at the ISFSI, and therefore meets the requirements in 10 CFR 72.24(i)

F4B.6 The SAR and docketed materials adequately describe the design criteria for the SSCs important to safety and other SSCs, and therefore meet the requirements in 10 CFR 72.120(a).

F4B.7 Any reactor-related GTCC waste that is stored is in a durable solid form with demonstrable leach resistance, and therefore meets the requirements in 10 CFR 72.120(b)(3).

F4B.8 Each SSC important to safety is designed to the quality standards commensurate with the important to safety of the function to be performed, and therefore meets the requirements in 10 CFR 72.122(a).

F4B.9 The SSCs important to safety are designed to withstand the normal and off-normal conditions associated with the site and can withstand postulated accidents, and therefore meet the requirements in 10 CFR 72.122(b)(1).

F4B.10 The SCCs important to safety are designed to withstand the natural phenomena associated with the site without impairing their capability to perform their intended safety functions (with consideration for the most severe natural phenomena reported for the site and in the appropriate combination of normal and accident conditions), and therefore meet the requirements in 10 CFR 72.122(b)(2)(i).

F4B.11 All ISFSI structures are designed to prevent massive collapse or dropping of heavy objects onto an SSC important to safety, and therefore meet the requirements in 10 CFR 122(b)(2)(ii).

F4B.12 SSCs important to safety are designed and located to continue to perform their safety functions effectively under credible fire and explosion exposure conditions, and therefore meet the requirements in 10 CFR 72.122(c).

F4B.13 SSCs important to safety are not shared between the ISFSI and other facilities, or have been shown that such sharing does not impair the capability of either facility to perform its safety functions, including the ability to return to a safe condition in the event of an accident, and therefore meet the requirements in 10 CFR 72.122(d).

F4B.14 Storage systems are designed to allow ready retrieval of SNF, HLW, and reactor-related GTCC waste for further processing or disposal, and therefore meet the requirements in 10 CFR 72.122(l).

F4B.15 SNF handling, packaging, transfer, and storage systems are designed to ensure subcriticality, in that at least two unlikely, independent, and concurrent or sequential changes must occur before a nuclear criticality accident ensues. The margins of safety of these systems are adequate for the nature of the immediate environment under accident conditions, and therefore meet the requirements in 10 CFR 72.124(a).

F4B.16 SSCs important to safety are designed to provide favorable geometry or permanently fixed neutron absorbing materials, as applicable, and therefore meet the requirements in 10 CFR 72.124(b).

F4B.17 SSCs important to safety that contain SNF, HLW, reactor-related GTCC waste, and other related radioactive waste are designed to ensure adequate safety with respect to suitable shielding and confinement under normal and accident conditions, and therefore meet the requirements in 10 CFR 72.128(a)(2) and (a)(3).

4B.5 References

10 CFR Part 50, “Domestic Licensing of Production and Utilization Facilities.”

10 CFR Part 72, “Licensing Requirements for the Independent Storage of Spent Nuclear Fuel, High-Level Radioactive Waste, and Reactor-Related Greater Than Class C Waste.”

American National Standards Institute (ANSI)/American Nuclear Society (ANS) 57.2-1983; W1993 (W=Withdrawn), “Design Requirements for Light Water Reactor Spent Fuel Storage Facilities at Nuclear Power Plants.”

ANSI/ANS 57.7-1988; R1997; W2007 (R=Reaffirmed, W=Withdrawn), “Design Criteria for an Independent Spent Fuel Storage Installation (Water Pool Type).”

American Society of Mechanical Engineers (ASME) Boiler and Pressure (B&PV) Code, 2015.
Section III, “Rules for Construction of Nuclear Facility Components.”
Division 1, “Metallic Components”; Subsection NC and ND

Regulatory Guide 1.13, “Spent Fuel Storage Facility Design Basis.”

Regulatory Guide 1.26, “Quality Group Classifications and Standards for Water-, Steam-, and Radioactive Waste-Containing Components of Nuclear Power Plants.”

NUREG/CR-6322, “Buckling Analysis of Spent Fuel Basket,” UCRL-ID-119697, Lawrence Livermore National Laboratory, May 1995.

5 THERMAL EVALUATION

5.1 Review Objective

The objective of the U.S. Nuclear Regulatory Commission's (NRC's) thermal review with regard to heat transfer and flow characteristics is to ensure that the storage container and fuel material temperatures of a dry storage system (DSS) or dry storage facility (DSF) will remain within the allowable limits for normal, off-normal, and accident conditions. The review will confirm that the temperatures of the fuel cladding (fission product barrier) will be maintained throughout the storage period to protect the cladding against degradation that could lead to gross rupture. The review will also confirm that the applicant uses acceptable analytical and testing methods, as applicable, in the safety analysis report (SAR) when evaluating the DSS or DSF thermal design.

Another objective of the thermal review is to ensure that the decay heat removal system is capable of reliable operation so that the temperatures of materials used for structures, systems, and components (SSCs) important to safety, and solidified high-level radioactive waste (HLW) containers remain within the allowable limits under normal, off-normal, and accident conditions. The NRC staff evaluate the wet and dry fuel assembly transfer systems for adequate decay heat removal under normal, off-normal, and accident conditions. In addition to storage container design, the reviewer considers siting and facility design.

The approach to thermal review and evaluation for a specific license builds upon the guidance provided for the certification review of storage containers. The guidance of this chapter unique to specific licenses is necessary because site-specific SARs will contain site-specific features (e.g., ambient temperature and wind speed limits) and other systems (e.g., pools, structures using reinforced concrete). If the DSF uses a storage container that has received a certificate of compliance (CoC), the review will address key assumptions, bounding site characteristics, environmental conditions, and storage container or facility interface requirements identified in the storage container SAR and CoC and compare them with the DSF design and environmental conditions. This review will confirm that the systems in the DSF support the assumptions used in the evaluation of the storage containers.

5.2 Applicability

This chapter applies to the review of applications for specific licenses for an independent spent fuel storage installation (ISFSI) or a monitored retrievable storage facility (MRS) categorized as DSF. It also applies to the review of applications for a DSS CoC for use at a general-license facility. Sections or paragraphs of this chapter that apply only to specific license applications are identified with "**(SL)**." Sections or paragraphs that apply only to DSS CoC applications are identified with "**(CoC)**." A section or paragraph without an identifier applies to both types of license applications.

5.3 Areas of Review

This chapter addresses the following areas of review:

- decay heat removal system
  - general considerations **(SL)**
  - DSSs **(SL)**
  - dry transfer systems **(SL)**

- material and design limits
  - general considerations
  - considerations for specific licenses **(SL)**
- thermal loads and environmental conditions
  - general considerations
  - considerations for specific licenses **(SL)**
- analytical methods, models, and calculations
  - configuration
  - material properties
  - boundary conditions
  - computer codes
  - temperature calculations
  - pressure analysis
  - confirmatory analysis
- surveillance requirements

5.4 Regulatory Requirements and Acceptance Criteria

This section summarizes those parts of Title 10 of the *Code of Federal Regulations* (10 CFR) Part 72, "Licensing Requirements for the Independent Storage of Spent Nuclear Fuel, High-Level Radioactive Waste, and Reactor-Related Greater than Class C Waste," that are relevant to the review areas this chapter addresses. The NRC staff reviewer should refer to the exact language in the regulations. Tables 5-1a and 5-1b match the relevant regulatory requirements to the areas of review covered in this chapter.

Table 5-1a Relationship of Regulations and Areas of Review for a DSF (SL)

Areas of Review	**10 CFR Part 72 Regulations**					
	72.26	72.44	72.92	72.120	72.122	72.128
Decay Heat Removal Systems					(h)	(a)(4)
Material and Design Limits				(a)(d)		(a)
Thermal Loads and Environmental Conditions			(a)		(b)	
Analytical Methods, Models, and Calculations					•	(a)
Surveillance Requirements	•	(c)			(f), h(4), (i), (j)	

Table 5-1b Relationship of Regulations and Areas of Review for a DSS (CoC)

Areas of Review	10 CFR Part 72 Regulations
	72.236
Decay Heat Removal Systems	(f)(h)
Material and Design Limits	(a)(b)
Thermal Loads and Environmental Conditions	(b)
Analytical Methods, Models, and Calculations	(g)(l)(m)
Surveillance Requirements	(b)(g)

5.4.1 Decay Heat Removal System

The spent fuel cladding must be protected during storage against degradation that leads to gross fuel rupture (10 CFR 72.122(h)). Decay heat removal systems shall have testability and reliability consistent with their importance to safety (10 CFR 72.128(a)(4)). The spent fuel storage cask must be designed to provide adequate heat removal capacity without active cooling systems (10 CFR 72.236(f)). The spent fuel storage cask must be compatible with wet or dry spent fuel loading and unloading facilities (10 CFR 72.236(h)).

The applicant should provide a detailed description of the proposed storage container heat removal system and its passive cooling characteristics. The SAR should clearly identify all major components and thoroughly explain their contribution to the removal of heat from the fuel. The SAR should also discuss the mechanism of heat removal (i.e., conduction, convection, radiation) for each component.

The applicant should provide evidence that the decay heat removal system will operate reliably under normal and loading conditions. The applicant should provide evidence that under off-normal and accident conditions, the decay heat removal system will not exceed allowable thermal limits and that the applicant will take adequate actions to bring the decay heat removal system to normal cooling.

The SAR should also describe all instrumentation used to monitor storage container thermal performance.

5.4.2 Material and Design Limits

An application to store spent fuel or reactor-related greater-than-Class-C (GTCC) waste in an ISFSI or to store spent fuel, HLW, or reactor-related GTCC waste in an MRS must include the design criteria for the proposed storage installation (10 CFR 72.120(a)). The ISFSI or MRS must be designed, made of materials, and constructed to ensure that there will be no significant chemical, galvanic, or other reactions between or among the storage system components, spent fuel, reactor-related GTCC waste, and/or high level waste including possible reaction with water during wet loading and unloading operations or during storage in a water-pool type ISFSI or MRS. The behavior of materials under irradiation and thermal conditions must be taken into account (10 CFR 72.120(d)). SSCs important to safety shall be maintained within their minimum and maximum temperature criteria for normal, off-normal, and accident conditions so as to support the performance of the intended safety function (10 CFR 72.128(a)). Specifications must be provided for the spent fuel to be stored in the spent fuel storage cask, such as, but not limited to, type of spent fuel (i.e., boiling-water reactor, pressurized-water reactor, or both), maximum allowable enrichment of the fuel before any irradiation, burn-up (i.e., megawatt days per metric ton of uranium (MTU)), minimum acceptable cooling time of the spent fuel before storage in the spent fuel storage cask, maximum heat designed to be dissipated, maximum spent fuel loading limit, condition of the spent fuel (i.e., intact assembly or consolidated fuel rods), the inerting atmosphere requirements (10 CFR 72.236(a)). Design bases and design criteria must be provided for SSCs important to safety (10 CFR 72.236(b)).

Storage container components and fuel materials should be maintained between their minimum and maximum temperature limits for normal, loading, off-normal, and accident conditions to enable all components to perform their intended safety function.

To guarantee the integrity of zirconium-based alloy cladding, the maximum calculated fuel-cladding temperature should not exceed 400 degrees Celsius (°C) (752 degrees Fahrenheit (°F)) for normal conditions of storage and short-term loading operations, including cask drying and backfilling. A higher temperature limit may only be used for low burnup spent nuclear fuel (SNF) (less than 45 gigawatt days MTU), as long as the applicant can demonstrate that the best estimate cladding hoop stress is equal to or less than 90 megapascals (MPa) (13.1 thousand pounds per square inch (ksi)) for the proposed temperature limit. During loading operations, repeated thermal cycling should be limited to less than 10 cycles when the cladding temperature difference exceeds 65 °C (149 °F). For off-normal and accident conditions, the maximum zirconium-based cladding temperature should not exceed 570 °C (1,058 °F).

To guarantee the integrity of stainless-steel cladding, the maximum calculated fuel cladding temperature should not exceed 570 °C (1,058 °F) for off-normal and accident conditions. The maximum calculated fuel cladding temperature should not exceed 400 °C (752 °F) for normal conditions of storage and short-term loading operations, including storage container drying and backfilling.

The applicant should clearly identify the operational temperature limits for all component materials important to safety under normal, loading, unloading, off-normal, and accident conditions. The applicant should provide a reliable basis for all the temperature limits.

The maximum internal pressure of the fuel container should remain within its design pressures for normal, off-normal, and accident conditions, assuming rupture of 1 percent, 10 percent, and 100 percent of the fuel rods, respectively. Assumptions for pressure calculations include release of 100 percent of the initial fill gas and 30 percent of the fission product gases generated within the fuel rods during operation.

The applicant should clearly identify the design pressure limits for the fuel container under normal, off-normal, and accident conditions.

5.4.3 Thermal Loads and Environmental Conditions

The applicant must identify and justify the design-basis thermal load and the insolation and ambient temperature assumptions used as boundary conditions for the normal, loading, off-normal, and accident scenarios (10 CFR 72.92(a)). The heat removal system must accommodate the decay heat of the SNF or HLW and the site normal, off-normal, and accident thermal conditions (10 CFR 72.122(b)). Design bases and design criteria must be provided for structures, systems, and components important to safety (10 CFR 72.236(b)). Further guidance to review the thermal impact of environmental conditions (e.g., ambient temperature, wind, elevation) on a DSS or DSF is provided in NUREG-2174, "Impact of Variation in Environmental Conditions on the Thermal Performance of Dry Storage Cask, Final Report," issued March 2016.

5.4.4 Analytical Methods, Models, and Calculations

SSCs important to safety must be designed to show compliance with 10 CFR 72.122, “Overall requirements.” Spent fuel and high-level radioactive waste storage and handling systems. Spent fuel storage, high-level radioactive waste storage, reactor-related GTCC waste storage and other systems that might contain or handle radioactive materials associated with spent fuel, high-level radioactive waste, or reactor-related GTCC waste, must be designed to ensure adequate safety under normal and accident conditions (10 CFR 128(a)). The spent fuel storage cask must be designed to store the spent fuel safely for the term proposed in the application, and permit maintenance as required (10 CFR 72.236(g)). The spent fuel storage cask and its systems important to safety must be evaluated, by appropriate tests or by other means acceptable to the NRC, to demonstrate that they will reasonably maintain confinement of radioactive material under normal, off-normal, and credible accident conditions (10 CFR 72.236(l)). To the extent practicable in the design of spent fuel storage casks, consideration should be given to compatibility with removal of the stored spent fuel from a reactor site, transportation, and ultimate disposition by the Department of Energy (10 CFR 72.236(m)).

The applicant should present a thermal analysis that clearly demonstrates the storage system’s ability to manage design heat loads and have the various materials and components remain within temperature limits. The analysis should be conducted for normal, loading (including storage container drying and backfilling), draindown and reflood (as applicable), off-normal, and accident conditions. Resulting temperature profile and internal pressure information are necessary to support the structural analysis and the confinement analysis in the SAR.

The applicant should specify the analytical methods used in the thermal evaluations, including any computational modeling software (i.e., finite element analysis or computational fluid dynamics (CFD) computer analysis codes) and should discuss the basis for the parameters and options selected for the analysis. All models should be clearly described. Material thermal properties for all storage container components should be provided and justified. Temperature-dependent thermal properties provided in the application should cover the expected operational range. The applicant should discuss, quantify, and report in the SAR any conservatism associated with the proposed thermal models. The level of detail of the discussion should be comparable with sections of the SAR that describe the analytical thermal models. For cases with small thermal margin, the SAR should include a table of results showing how the associated conservatisms affect the safety parameters (e.g., calculated peak cladding temperature, confinement seal temperatures). The table of results should be supported with fully documented analytical models and calculations. In order to justify a small thermal margin, the identified model conservatisms should demonstrate a positive increase in the predicted margin.

The computer codes used in the thermal evaluation should be well verified and validated. The applicant can include the code verification and validation in the application or in a separate calculation package along with applicable references. The applicant should provide acceptable basis (e.g., benchmark efforts that mimic heat transfer and flow characteristics for the proposed design and that includes well defined boundary conditions and high-quality data for validation purposes, published results that include the range of applicability of the computer codes and highlight the specific features relevant to storage container design) for the accuracy of the selected computer code or codes and justification for the code’s use in the proposed evaluation. The applicant should provide a discussion of the resulting level of convergence and conservatism achieved as a function of the modeling options (e.g., meshing, time-differencing). The applicant should provide solution verification results by calculating the grid convergence index (GCI). Guidance to calculate the GCI is provided in NUREG-2152, “Computational Fluid Dynamics Best

Practice Guidelines for Dry Cask Applications, Final Report," issued March 2013 (ADAMS Accession No. ML13086A202) and American Society of Mechanical Engineers (ASME) "Standard for Verification and Validation in Computational Fluid Dynamics and Heat Transfer."

To facilitate confirmatory analyses, the applicant should provide detailed drawings of the proposed design and electronic copies of the most significant input and output files. Further guidance on the review of analytical methods, models, and calculations provided to the staff for review is provided in Appendix 4A, "Computational Modeling Software Technical Review Guidance," to this standard review plan (SRP) and NUREG-2152.

5.4.5 Surveillance Requirements

Section 5.5.1, "Decay Heat Removal Systems," and Chapter 17, "Technical Specifications Evaluation," of this SRP provide information relevant to the review of surveillance requirements for a specific license.

Each application under this part shall include proposed technical specifications in accordance with the requirements in 10 CFR 72.44 and a summary statement of the bases and justifications for these technical specifications (10 CFR 72.26). The applicant must describe the program of surveillance to ensure satisfactory in-service performance of items and activities important to safety (10 CFR 72.122(f), 72.122(h)(4), 72.122(i), 72.122(j), 72.236(b), and 72.236(g)). The SAR should present the surveillance program for temperatures and pressures, as applicable, for SSCs important to safety, including those described in Chapter 12, "Conduct of Operations Evaluation," and Chapter 17 of this SRP.

5.5 Review Procedures

Review design features and acceptance criteria, given in the chapters of the SAR on general information and principal design criteria, for additional insight about the thermal models that are being presented. Review the appropriateness of the proposed heat loads and environmental conditions. Assess modeling details such as assumptions, simulation options, simplifications, and accuracy of results. The DSS or DSF is to be analyzed under normal, loading, off-normal, and accident scenarios. Review the resulting temperature distributions and internal pressures calculated in the SAR to verify compliance with design criteria and regulatory requirements.

One aspect of the DSS or DSF thermal evaluation is confirmation that the fuel cladding temperature will remain below a specified allowable limit to prevent degradation during storage. Another aspect of the DSS or DSF thermal evaluation is confirmation that materials used for SSCs important to safety and solidified HLW containers remain within the allowable limits.

Thermal performance of the storage container under off-normal and accident conditions is also evaluated in accordance with Chapter 16, "Accident Analysis Evaluation," of this SRP, as appropriate, in the overall accident analyses presented in the SAR.

Figures 5-1a and 5-1b show the interrelationships between the thermal evaluation review and the other areas of review described in this SRP for specific license and CoC applications, respectively.

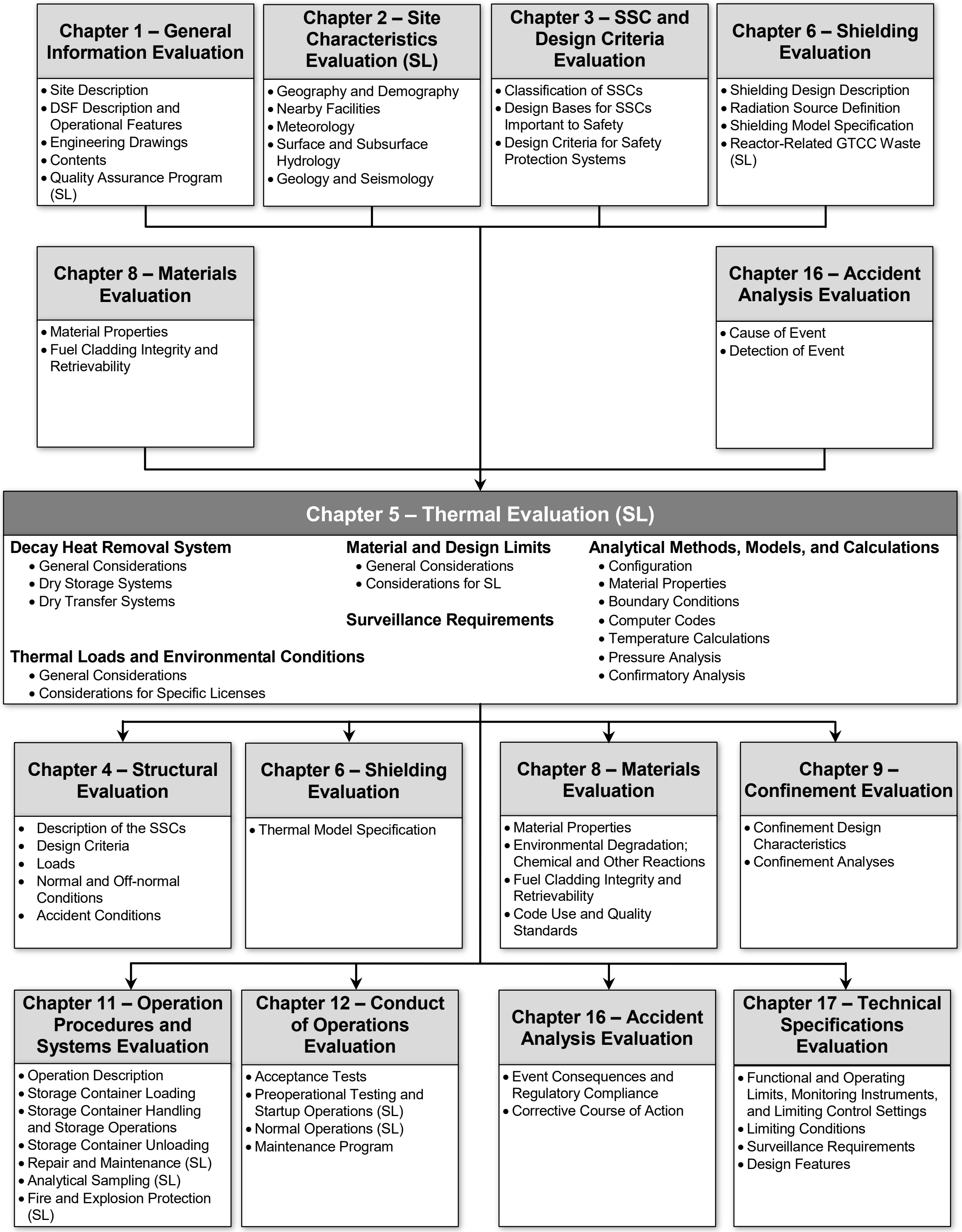

Figure 5-1a Overview of Thermal Evaluation of Specific License Applications for a DSF (SL)

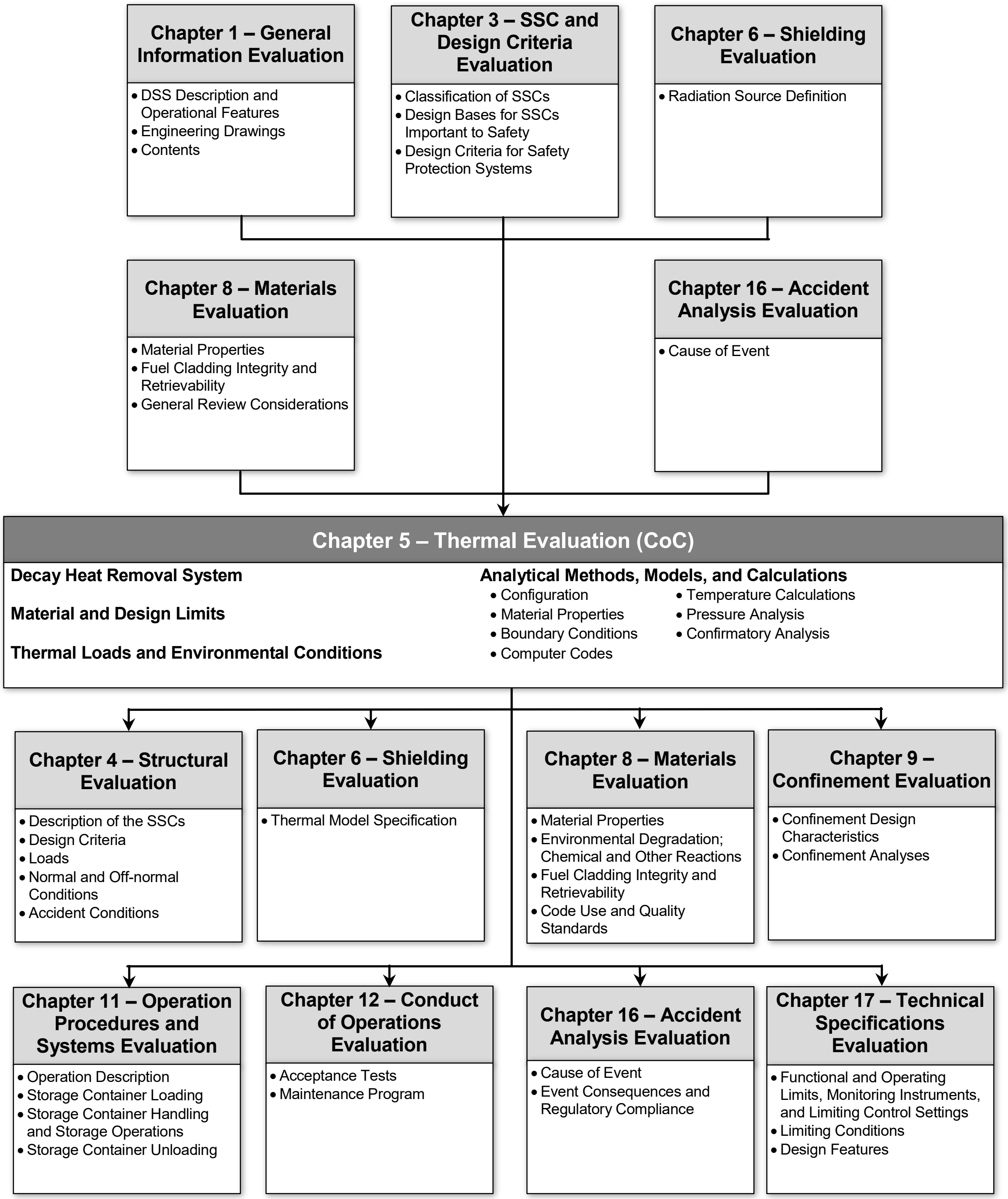

Figure 5-1b Overview of Thermal Evaluation of Applications for a DSS (CoC)

5.5.1 Decay Heat Removal Systems

Review the description of the DSS and DSF presented in the SAR chapter on general information, as supplemented by the additional information provided in the SAR on thermal evaluation. Ensure these two sources of information are consistent and supplementary. In addition to the material specifications, the dimensions of the storage container components and SNF assemblies are to be clearly indicated. Ensure all drawings, figures, and tables are sufficiently detailed to support an indepth staff evaluation.

Confirm that the applicant describes the significant thermal design features and operating characteristics of all pertinent DSS or DSF components and subsystems. Design features typically include, but are not limited to, the storage container body, thermal fins, shielding materials, fuel baskets, heat transfer disks, confinement seals, drain and vent ports, and external pressure relief devices (for the case of transfer casks). Verify that the thermal design features will adequately perform their intended safety functions during normal, loading, off-normal, and accident conditions. All thermal design features should be passive. Applicants have requested temporary supplemental cooling (circulating water or air flow) of storage container systems during loading operations or as a technical specifications action statement during transfer operations. Review such requests to ensure that they meet the original intent of the regulations—that storage container systems remain passively cooled during normal operations.

Ensure that the applicant has described any instrumentation used to monitor storage container heat removal capability in sufficient detail to support an indepth staff evaluation. Ensure that the monitoring instrumentation components have a safety classification (presented in the SAR chapter on principal design criteria) commensurate with their function and is fully justified. Verify that the SAR chapter on technical specifications and operational controls and limits clearly indicates applicable operating controls and criteria, such as temperature or pressure criteria and surveillance requirements. These should also be discussed in the safety evaluation report (SER) and included in the CoC or specific license, as appropriate.

5.5.1.1 *General Considerations* ***(SL)***

ISFSI or MRS decay heat removal systems must accommodate the decay heat of the SNF or HLW and the site normal, off-normal, and accident environmental conditions (10 CFR 72.122(b)). Verify that the SAR for the ISFSI or MRS clearly establishes that the storage system will function within the allowable thermal limits under normal, off-normal, and accident conditions. Review the specification for the design-basis fuel assembly decay heat presented in the SAR's discussion of principal design criteria and the corresponding sections of the storage container(s) SAR(s) if the storage container has received an NRC CoC. Coordinate the review with the shielding reviewer to ensure that this decay heat is consistent with the specified enrichments, burnups, and cooling times. Consider relevant generic communications (e.g., NRC information notices, regulatory guides) as part of the review.

Ensure that the decay heat removal systems have testability and reliability consistent with their importance to safety (10 CFR 72.128(a)(4)). Ensure that, during storage, the SNF cladding is protected against degradation that could lead to gross fuel rupture and is otherwise confined such that degradation of the fuel during storage will not pose operational problems with respect to its removal from storage (10 CFR 72.122(h)(1)). For each type of fuel assembly proposed for storage, confirm that the systems ensure a very low probability (e.g., 0.5 percent), per fuel rod, of cladding breach during long-term (e.g., 40-year) storage (10 CFR 72.122(h), (Levy et al. 1987).

This can be accomplished by confirming that fuel cladding temperatures will remain below recommended limits, as specified in Section 5.4.2, "Materials and Design Limits," of this SRP.

Review the thermal analysis, material temperature limits, and key assumptions of the analysis to ensure that the DSS or DSF design and environmental conditions are within the envelope of the DSS original analysis and the associated technical specifications. Confirm that the design criteria include maximum heat output of the radioactive materials (including control components or other assembly hardware such as shrouds); temperature levels for the ambient air under normal, off-normal, and accident conditions; and associated insolation. Confirm that the SAR identifies the conditions (off-normal or accident) that may result in high temperature gradients and pressures. The conditions may be time-varying and may be controllable or subject to limits (e.g., temperature, pressure, time).

Coordinate with the structural review under Chapter 4, "Structural Evaluation," of this SRP to ensure that the temperatures and pressures for all other SSCs important to safety, presented in the SAR, correspond to the same temperatures and pressures given in the thermal loads analysis in Chapter 4.

5.5.1.2 *Dry Storage Systems* ***(SL)***

Verify that the technical specifications include limiting conditions for operation and surveillance requirements to ensure that the temperature will remain within acceptable limits during dry storage and that normal cooling will begin before the temperature criterion is exceeded if the fuel cladding temperature calculation is based on heatup over a limited time period.

5.5.1.3 *Dry Transfer Systems* ***(SL)***

If the fuel cladding temperature calculation is based on heatup over a limited time period, verify that the technical specifications impose limiting conditions for the operation and surveillance requirements that ensure that the temperature will remain within acceptable limits during the process and that normal cooling will begin before the temperature criterion is exceeded.

5.5.2 Material and Design Limits

5.5.2.1 *General Considerations*

One aspect of the thermal evaluation is the confirmation that the fuel cladding temperature will prevent cladding damage or potential failure during storage. Ensure that the application complies with the criteria for cladding integrity (see Section 5.4.2 of this SRP) or provides adequate justification for any deviation from these criteria.

Ensure that the application reflects one of the following criteria: (1) the maximum calculated temperatures for normal conditions of storage and for fuel loading operations do not exceed 400 °C (752 °F), or (2) for low burnup fuel, the maximum calculated temperatures for normal conditions of storage and fuel loading operations do not exceed 570 °C (1,058 °F) and that the materials reviewer has verified that the best estimate cladding hoop stress is less than 90 MPa (13.1 ksi) for the maximum allowable temperature the applicant specified.

If the applicant uses the second approach, confirm that the materials reviewer has verified that the cladding hoop stresses are less than 90 MPa (13.1 ksi) for each fuel assembly type (e.g., 14 x 14, 17 x 17, 9 x 9) proposed for storage. Confirm that the materials reviewer evaluated cladding oxide thickness used to compute hoop stress. Because the hoop stress is dependent on the rod internal pressure, cladding geometry, and the temperature of the gases inside the rod, coordinate with the materials reviewer to verify that the applicant calculated the best estimate hoop stress corresponding to the rod internal pressure of the highest burnup fuel assemblies of the specific type of assembly.

To limit the amount of SNF that could be released from the cladding under off-normal or accident conditions, ensure that the application reflects the maximum calculated cladding temperatures is maintained below 570 °C (1,058 °F). Verify that the application clearly identifies the temperature restrictions (upper and lower allowable limits) on all components important to safety (e.g., confinement, shielding, subcriticality, heat removal) during normal, loading, off-normal, and accident scenarios and that the predicted thermal behavior of the entire DSS or DSF is indeed within the specified allowable limits. Confirm with the materials reviewer the acceptability of all proposed temperature limits.

Ensure that the maximum internal pressure of the fuel container remain within its design limits for normal, off-normal, and accident conditions assuming rupture of 1 percent, 10 percent, and 100 percent of the fuel rods, respectively. Confirm with the structural reviewer the acceptability of the proposed design pressure limits.

Ensure that any operating scenario (loading or unloading) that results in a time-dependent limiting condition (e.g., number of hours allowed for vacuum drying before fuel cladding temperature reaches its allowable limit) is also addressed in Chapter 17, “Technical Specifications Evaluation,” of this SRP and is included as a limiting condition for operation (e.g., technical specifications) in the CoC or specific license, as appropriate.

Consider the issue of storage container heatup during loading operations. If there is a loading issue, the storage container has to reach equilibrium again before reattempting loading to ensure that temperatures are not exceeded. For example, applicants may change a fill gas (i.e., helium to nitrogen) for vacuum drying. Vacuum drying may take multiple cycles, and the temperatures of the contents may therefore not fall below the contents’ initial temperature, leading to higher temperatures during subsequent cycles occurring at an earlier time during the vacuum drying process. Confirm that the applicant has provided adequate analysis and assumptions (i.e., adequate initial and boundary conditions) for subsequent drying cycles to cover loading-issue scenarios.

NRC Information Notice 2011-10, “Thermal Issues Identified During Loading of Spent Fuel Storage Casks,” dated May 2, 2011, and its supplement, Information Notice 2014-08, “Need for Continuous Monitoring of Active Systems in Loaded Spent Fuel Storage Canisters (Including Vacuum Drying Process),” dated May 16, 2014, also contain relevant information.

5.5.2.2 *Considerations for Specific Licenses **(SL)***

Verify that the SAR identifies and justifies temperature restrictions on other SSCs important to safety, including materials that are integral to confinement (e.g., storage container mechanical seals), shielding, and subcriticality functions. Verify that the applicant included the temperature limit criteria and the basis for the limits selected.

Considerations for determining temperature limits for the material of construction and the stored radioactive material can include the following, but are not limited to:

- the temperature at which the structural strength of the material is affected and the time-temperature relation required to cause the effect
- the retrievability of the radioactive material
- the temperature at which chemical or galvanic reactions that affect shielding, subcriticality assurance, or structural integrity may take place (at a significant rate)
- the temperature at which the black body characteristics of the material used for modeling may be affected
- the allowance to provide for uncertainties in the temperatures that may occur
- the temperatures that may be reached in normal, off-normal, and accident conditions and events
- the potential combinations of temperature and environment (such as may produce significant reaction with borated water)
- the outgassing of materials that produce significant amounts of either radioactive or nonradioactive gases
- the state changes of materials

Information on thermal properties may be needed for materials that are analyzed for loads on SSCs. Confirm that the source of data on thermal properties is an acceptable reference, such as the appendices to the ASME Boiler and Pressure Vessel Code, Section II, “Material,” and Section III, “Rules for Construction of Nuclear Facility Components.” Applicants may need to use other sources for nonstandard (or vendor-specific) materials such as neutron absorbers and storage container seals. Coordinate with the materials reviewer to verify the acceptability of these materials.

5.5.3 Thermal Loads and Environmental Conditions

5.5.3.1 *General Considerations*

Review the specification for the design-basis fuel decay heat presented in the chapter of the SAR on principal design criteria and coordinate with the shielding reviewer to ensure that this decay heat is consistent with the specified fuel types, burnups, enrichments, and cooling times, if included. However, some applications may provide a bounding decay heat load (kilowatt per assembly) without specifying details about the SNF (e.g., design, enrichment, cooling time). In these cases, ensure that the SER clearly specifies that the decay heat values are evaluated on the basis that NRC inspectors will verify compliance with the CoC. This verification includes reviewing the approach to determine the per-assembly decay heat and any uncertainties associated with the approach. If necessary, inspectors will coordinate with technical reviewers to determine adequacy of site-specific decay heat values and method of evaluation.

Verify that the applicant also discusses the axial distribution for the decay heat sources, with clear justification for a bounding approach. Expect a somewhat flat-at-the center axial distribution with a peak-to-average value in the range of 1.1 to 1.2, tapering to lower values toward both ends.

In general, the NRC staff accepts insolation values presented in 10 CFR Part 71, “Packaging and Transportation of Radioactive Material,” for 10 CFR Part 72 applications. Because of the large thermal inertia of a DSS, the insolation values listed in 10 CFR 71.71, “Normal conditions of transport,” may be averaged over a 24-hour day assuming steady-state conditions. Verify that insolation values specified in the thermal model are consistent with the values listed in 10 CFR 71.71. Ensure that any deviation from these values is fully justified. Confirm that the insolation values specified in the thermal model are consistent with the types of surfaces listed in 10 CFR 71.71. Verify that the applicant performed a heat balance and that the result is consistent with all heat sources specified in the thermal model.

Verify that the ambient temperatures used for normal, off-normal, and accident condition evaluations do indeed bound the available historical temperature data for any suggested storage site (current or future). Refer to the National Oceanic Atmospheric Administration National Climatic Data Center for temperature statistics for many American cities and regions (http://www.ncdc.noaa.gov/oa/ncdc.html). Further guidance to review the thermal impact of environmental conditions (e.g., ambient temperature, wind, elevation) on DSS is provided in NUREG-2174.

Ensure the loading and unloading evaluations are based on the SNF pool’s technical specifications maximum temperature limit (typically 46 °C (115 °F)).

5.5.3.2 *Considerations for Specific Licenses* ***(SL)***

Determine whether the applicant has demonstrated that reactor-related GTCC waste containers, co-located with SNF storage containers at an ISFSI or MRS, or co-located with HLW containers at an MRS, are located such that normal, off-normal, and design-basis accident conditions will not adversely impact the heat removal capability of the SNF storage containers. In general, the thermal design of reactor-related GTCC waste containers is very similar to that of SNF canisters. However, because SNF decay heat is higher than reactor-related GTCC waste, SNF canisters bound GTCC containers.

5.5.4 Analytical Methods, Models, and Calculations

For storage container system components in which material properties and performance vary with temperature, review the modeling assumptions used in determining temperature maxima, minima, gradients, and differences for the storage container system. Review the assumptions used to determine fuel cladding temperatures. The assumed temperature changes over time should result in the bounding conditions for the structural analysis. Compare the calculated temperatures in the various storage container system components to the limiting temperature criteria for the appropriate materials. Ferritic materials are subject to failure by brittle fracture at low temperatures. Verify the assumed low temperatures for storage container system handling operations for consistency with material properties. Ambient temperature restrictions may be appropriate for storage container handling operations. Ensure that any limiting conditions regarding ambient temperatures are addressed in the chapters of the SAR and SER on technical specifications and operating controls and limits. The CoC or site license should include ambient temperatures as a limiting condition for operation (e.g., technical specifications), as appropriate.

Analysis for accident conditions temperatures should not be considered to envelop the analysis of normal or off-normal temperatures. The acceptance criteria for normal and off-normal temperature demands for structural capacity will differ. Therefore, ensure that the application includes an analysis for normal, off-normal, and accident conditions. In addition, ensure the applicant evaluated the duration over which accident temperature conditions may exist.

5.5.4.1 *Configuration*

Verify that the applicant clearly described any model used in the thermal evaluation. Separate models and submodels may be used for the evaluation of different conditions (normal storage, loading, off-normal situations, and accidents). Verify that the applicant provided adequate justification when using separate models and submodels to evaluate different conditions (for example, a simplified model may be used to evaluate fire accident conditions; in which case, verify that the application provides adequate justification and shows that the results are conservative). Coordinate with the structural review as necessary to evaluate any damage that may result from accidents or natural phenomena events. All models should be shown as conservative (i.e., thermal results should include adequate margin against allowable limits).

Review the sketches or figures of all models to ensure their proper use in the thermal calculations and verify that the dimensions and materials are consistent with those in the drawings of the actual storage container, as presented in the chapter of the SAR on general information. If possible, review the computer input files to verify consistency with the model sketches and engineering drawings. The application should identify any differences between the actual storage container configuration and the model, and the model should be shown to be conservative.

Pay particular attention to gaps between storage container components. Consider tolerances so that the thermal resistance of each gap is treated conservatively. Confirm that the application describes and justifies gases (e.g., air, helium) assumed to be present in the gap. If a specific gas other than air in the cavity of the storage container or gaps between storage container components is relied upon for heat removal, verify that the applicant shows that the gas is retained and that the gas is not diluted by other gases with lower thermal conductivity during the entire storage period. For storage container components that are important to heat removal, ensure that the application adequately describes and justifies manufacturing techniques for joining components, surface roughness, contact pressures, and gap conductance values. For example, poured lead may shrink when cooled or gaps may exist if lead shielding is pounded in place as part of manufacturing.

Verify that decay heat generated in the SNF is limited to the active fuel region of the assemblies. Ensure that the decay heat model specifically accounts for peaking in the central region or provides another conservative approach. Ensure also that the heat from any other stored component (e.g., control rods), if applicable, is distributed appropriately. In addition, ensure that the position of heat sources relative to other storage container components is identified.

Confirm that the application addresses the thermal interaction among storage containers in an array by calculating the appropriate view factor. Generally, this will result in an operating control and limit in the SAR chapter on technical specifications and operating controls and limits that impose a minimum spacing between storage containers.

Coordinate with the structural reviewer to ensure that the applicant has analyzed situations that may produce the worst-case storage container loads. The greatest gradients and loadings caused by thermal expansion may occur with storage containers in alternative storage or in temporary handling positions.

Review the heat transfer processes used in the analysis. Conduction and radiation are typically defined as the primary heat transfer mechanisms within the storage container itself. In narrow regions of any orientation, little or no convective heat transfer will occur, and only conduction through the gas-filled void spaces is assumed. Larger gas volume regions can experience a significant level of convective heat transfer. Therefore, verify that the applicant has demonstrated the existence of convection in the larger gas regions and has quantified the contribution of convection heat transfer to the overall removal of heat from the package. Ensure that natural convection in enclosed cavities was validated through applicable CFD calculations or physical experiments.

5.5.4.1.1 General Guidance on CFD Analyses

Because the computational resources necessary to fully resolve flow between individual fuel pins in a storage container model with numerous fuel assemblies would be enormous, one acceptable approach would be to treat fuel assemblies as a porous media for applications seeking to credit heat removal from fuel via internal convection. Verify that any CFD approach uses realistic or bounding flow friction factors in the porous media representation of the fuel, and that friction factors are obtained for each of the limiting fuel assembly types sought as approved contents for the storage container.

An acceptable approach to calculate the friction factors would be to perform a CFD analysis for each type of fuel assembly for the expected operating conditions (pressure and average gas temperature). Verify that the application reflects that wall shear stresses were obtained separately for bare fuel rods and for fuel rods and associated grid straps. Confirm that the applicant calculated the friction factor based on the wall shear stress method. Additional details to obtain flow friction factors are provided in NUREG-2208, “Validation of Computational Fluid Dynamics Methods Using Prototypic Light Water Reactor Spent Fuel Assembly Thermal-Hydraulic Data,” issued March 2017 (ADAMS Accession No. ML17062A567).

Evaluate the method used to obtain the friction factors and ensure that the obtained values are realistic or bounding for the intended fuel assembly types. Also, since the friction factor is generally very sensitive to the geometric information (dimensions) and fuel assembly configuration, verify this information by reviewing the fuel assembly design drawings provided by the applicant.

For ventilated SNF storage systems (a canister containing the fuel within an outer overpack fitted with air vents), the mesh spacing (computational cell size) and density between an overpack liner and canister outer shell wall play an important role when selecting a turbulence model for the air flow through this annular gap, as described below.

The near-wall modeling significantly impacts the fidelity of numerical solutions, inasmuch as walls are the main source of flow mean vorticity and turbulence. After all, it is in the near-wall region that the solution variables have large gradients, and the transport of momentum and other scalar variables occurs more vigorously. Therefore, accurate representation of the flow in the near-wall region determines a successful prediction of wall-bounded turbulent flows. When dealing with wall effects on the flow, usually two modeling options are available to the analyst. The first one is the use of the semi-empirical formulas called “standard wall functions,” which are used to bridge the viscosity-affected region between the wall and the fully turbulent core region. Generally, a uniform mesh would be used when these wall functions are invoked. The use of wall functions obviates the need to modify the turbulence models to account for the presence of the wall. This modeling approach is usually applicable to flows with high Reynolds numbers. In the second approach, the viscosity-affected region is resolved with a mesh all the way to the wall, including the viscous sublayer. This type of approach is referred to as "near wall modeling." The dimensionless distance between the wall and the cell center near the wall (y+) for the mesh used for this case should generally be around 1. The documentation for the CFD program used in the application should provide guidance on how to apply any of these modeling approaches. Verify that the application fully justifies and validates any modeling approach taken.

To properly characterize the flow (e.g., internal, external, annular), Reynolds number estimates should be made using velocities from initial runs for the cooling air in the annulus and helium fill inside the canister. Reynolds numbers above 3000 based on the channel hydraulic diameter are above the critical Reynolds number of 2300 for internal flows, characterizing the flow in the transitional range between the laminar and turbulent zones. Because these are buoyancy-driven flows, both the Grashof (Gr) number, based on the hydraulic diameter of the channel, and the modified Grashof number, defined as Graetz number (Gz = Gr * W/H), where W and H are the width and height of the air channel, respectively, should also be calculated to properly characterize the annular flow. On the other hand, buoyancy-driven helium flow, cooling the inside of the canister, generally would be laminar based on both the Grashof and the Reynolds numbers because of higher kinematic viscosities and low achieved velocities within the canister. Confirm that the application provides solution verification results by calculating the GCI. Guidance to calculate the GCI is provided in NUREG-2152 and ASME's "Standard for Verification and Validation in Computational Fluid Dynamics and Heat Transfer" (ASME V&V 20).

Verify that the GCI calculation follows the assumptions used to develop the GCI method, as described in NUREG-2152 and ASME V&V 20. These are summarized below:

- Grid refinement or coarsening is performed systematically in all directions; that is, the refinement or coarsening should be structured even if the grid is unstructured.

- The observed order of accuracy should not vary greatly from the theoretical order of accuracy (i.e., the order of accuracy of the numerical method used in the analysis).

- A minimum of four grids is required to demonstrate that the observed order of accuracy is constant for a simulation series.

- A three-grid solution for the observed order of accuracy may be adequate if the values of the target variable (for example, peak cladding temperature, total heat transfer rate, or mass flow rate) predicted on the three grids are in the asymptotic region for the simulation series.

- Methods to test for asymptotic behavior of the target variable predicted values are provided in ASME V&V 20.

- The factor of safety (F_s) value is 1.25 if the target values on the three grids are in the asymptotic region and the observed order of accuracy does not vary greatly from the theoretical order of accuracy. Otherwise an F_s of 3.0 is used.

- The GCI is calculated using the observed order of accuracy if it is smaller than the theoretical value. Otherwise the theoretical order of accuracy is used.

Confirm that the application also addresses actual SNF properties and uncertainties (e.g., friction factors, crud and oxide buildup, eccentricities, nonuniform axial and radial decay heat profiles). Verify that the applicant avoided using an effective thermal conductivity for the cover gas (e.g., helium) in lieu of a specific convection model.

If applicable, confirm that the application includes an evaluation of the added heat from components stored with the SNF assemblies (e.g., control rods, fuel channels). This would ultimately affect the maximum predicted cladding temperature.

NUREG-2152 provides further guidance for the review of CFD applications. NUREG-2152 also provides additional guidance to perform CFD confirmatory analysis for dry storage container thermal evaluations.

5.5.4.1.2 General Guidance on Application of Effective Conductivity Models

In addition to a CFD method using porous media model, fuel assemblies may be modeled as a homogenous region using an effective thermal conductivity model. Review the manner in which effective conductivity is determined for each fuel assembly (see Section 5.5.4.2 below).

Use of effective thermal conductivity coefficients for regions within the confinement storage container other than the fuel (e.g., gaps) may overestimate heat transfer. If effective thermal conductivity is used in this manner, verify that the same values have been determined from test data, CFD submodels, or other appropriate sources that are representative of similar geometry, materials, temperatures, and heat fluxes used in current application. Pay particular attention to the effective thermal conductivity of neutron shield regions, such as those embedded within thermal fins. Voids or gaps typically exist as a result of either tolerances or shrinkage and should be considered in calculating effective thermal conductivity. Also, confirm that the applicant paid particular attention to the values assumed for surface emissivities and view factors, as well as the manner used to account for radiation heat transfer in determining the effective thermal conductivities.

5.5.4.2 *Material Properties*

Coordinate with the materials reviewer to verify that the material specifications and thermal properties are provided for all components used in the analytic model, the thermal properties used in the safety analysis are appropriate, and potential degradation of materials over their service life has been evaluated. Confirm that the applicant considered temperature and anisotropic dependencies of thermal properties. If regional thermal properties are determined from a combination of individual materials, ensure the manner in which these effective properties are calculated is fully described and justified.

If the thermal model is axisymmetric or three dimensional, check that the longitudinal thermal conductivity is generally limited to the conductivity of the cladding (weighted by its fractional area) within the fuel assembly. Gaps between fuel pellets and cracks in the pellets themselves can result in a considerable uncertainty regarding the contribution of the fuel to longitudinal heat transfer. Verify that the applicant considered high-burnup effects in determining the fuel region effective thermal conductivity.

5.5.4.3 *Boundary Conditions*

Verify that the applicant identified boundary conditions for normal, loading, off-normal, and accident conditions. The required boundary conditions include the total decay heat from each fuel assembly and the external conditions on the storage container surface. Ensure that the peak power factor for a fuel assembly is specified and the peak linear power ("peaking factor") of a fuel assembly is stated for a given active fuel length.

The boundary conditions on the storage container surface depend on the environment surrounding the storage container. Consequently, confirm that the application specifies the temperature of the environment for all simulated conditions, as well as the incident and absorbed insolation. Verify that the application identifies and describes the mechanisms and models for dissipating the absorbed insolation and decay heat from the surface of the storage container to the environment. The mechanisms for transferring heat from the storage container surface usually consist of natural (free) convection and thermal radiation. Confirm that the SAR presents the results of a heat balance on the surface of the storage container.

Ensure that the application establishes the initial temperature distribution of the storage container system before a fire accident based on the hottest temperature distribution during normal or off-normal storage conditions. Confirm that the application specifies the duration and flame temperature of the fire, as well as gas velocities and flame emissivity. The NRC considers the flame and storage container surface emissivities specified in 10 CFR 71.73(c)(4) for a hypothetical accident test of transportation packages as satisfactory for use with regard to a fire accident involving a storage container.

Confirm that the application identifies and describes the mechanisms and models for coupling the fire energy to the storage container surface. These mechanisms include forced convection in relation to the flame velocity (5 to 15 meters per second (16 to 49 feet per second)) as well as thermal radiation. In addition, confirm that the application justifies the convection coefficients during the fire. Verify that the application also considers the orientation of the storage container.

Following the fire, the storage container is subject to insolation and content decay heat while cooling by natural convection and thermal radiation to the environment. Confirm that the application identifies the postfire conditions of the storage container, including any changes in surface conditions or geometry (or both) that may affect radiation and convection heat losses. Confirm that the application also identifies and describes the models used for the analysis of the postfire processes.

5.5.4.4 *Computer Codes*

Verify that the applicant has provided information on any computer-based modeling as described in Appendix 4A to Chapter 4 of this SRP, and review the thermal analysis submitted by the applicant in accordance with the appendix.

5.5.4.5 *Temperature Calculations*

Confirm that the application includes a table that lists the maximum and minimum temperatures of all components important to safety under normal, loading, off-normal, and accident conditions. This table should specify the operating temperature range for each component. Verify that temperatures have been calculated for key components and that they do not exceed the allowable range for each. Ensure that the application provides justification for any material important to safety that exceeds acceptable temperature ranges. If compliance with minimum temperature criteria relies on a specific minimum heat load from the fuel, the SAR should quantify and include such a heat load as an operating control and a technical specifications criterion.

Pay particular attention to the maximum temperature of the cladding, discussed in Sections 5.4.2 and 5.5.2, "Material and Design Limits," of this chapter.

Some storage systems rely upon natural circulation of air through internal passages to remove heat from the stored confinement canister. For storage systems with internal air flow passages, blockage of inlet flow or outlet flow (or both) is an accident situation that should be evaluated. Total blockage of all inlets and outlets may result in fuel heatup, which has been assumed to approach adiabatic conditions. To ensure that blockages do not go undetected for significant periods, the NRC has required objective evidence that inlet and outlet flows are not obstructed. Consequently, for these types of storage systems, the NRC has accepted periodic visual inspection of the vents coupled with temperature measurements to verify proper thermal performance and detect flow blockages. The inspections should take place within an interval that will allow sufficient time for corrective actions to be taken before the accident temperature is reached. The inspection interval should be more frequent than the time interval required for the fuel to heat up to the established accident temperature criteria, assuming a total blockage of all inlets and outlets. Verify that the technical specifications include limiting conditions for operation and surveillance requirements to ensure that the temperature will remain within acceptable limits during dry storage and that normal cooling will begin before the temperature criterion is exceeded.

Confirm that the adiabatic heatup calculations specifically address any assumptions regarding limiting components and quasi-steady-state responses. The initial ambient temperature for the heatup calculations should bound the maximum "normal condition" temperature. Ensure that the SAR includes the resulting heatup time history, which should support the proposed inspection and monitoring intervals. This information is also useful in developing contingency operation procedures because it indicates the available time in which to take corrective actions before the fuel accident temperature criteria may be exceeded. Verify that the technical specifications include limiting conditions for operation and surveillance requirements to ensure that the temperature will remain within acceptable limits during dry storage and that normal cooling will begin before the temperature criterion is exceeded.

Some storage systems may use a transfer cask to move the loaded confinement canister from the fuel-handling building to the DSF site. When the canister is within the transfer cask, the rate of cooling is typically less than for normal operation. Therefore, fuel cladding temperatures are expected to be higher than for normal storage conditions.

Review the temperature distribution calculations for the canister inside the transfer cask and verify that heat transfer through gap regions has been treated in a conservative manner, and that material properties and dimensions of the transfer cask are consistent with the design data defined in the SAR documentation. The initial ambient temperature should be the maximum "normal condition" temperature. Storage container preparation for storage or unloading operations may include situations in which the canister is evacuated while it is in the transfer cask. If the fuel cladding temperature calculation is based on heatup over a limited time period for storage container drying operations, verify that the technical specifications impose limiting conditions for the operations. Such limiting conditions should ensure that the temperature will remain acceptable during the operations and that normal cooling will begin before the temperature criterion is exceeded.

During wet-fuel transfer operations, the liquid in the fuel canister should not be permitted to boil. This practice avoids uncontrolled pressures on the canister and the connected dewatering, purging, and recharging system(s); unacceptable discharge of liquids that may be providing radiation shielding; and a potentially unacceptable reduction in the safety margin. Ensure that, to prevent any of the above conditions, both the SAR and corresponding operating procedures identify an adequate subcooling margin to prevent boiling. This margin may be storage container specific, depending on the design of the fuel basket and key assumptions used in the criticality analysis. Ensure that the applicant performs the heatup and time-to-boil calculations and assesses whether any technical specifications or limiting conditions for operation are needed. Heatup calculations should be established on the basis of the SNF pool's technical specifications maximum temperature limit (typically 46 °C (115 °F)).

For unloading operations, ensure that the applicant evaluates temperature and pressure calculations supporting procedural steps presented in the SAR chapter on operating procedures for storage container cooldown and reflooding of the storage container internals. To ensure that the storage container does not overpressurize and that the fuel assemblies are not subjected to excess thermal stresses, confirm that the applicant's analysis specifies and justifies the appropriate temperature and flow rate of the quench fluid, assuming maximum fuel cladding temperatures in the unloading configuration. Verify that the chapter of the SAR on accident analyses also indicates that this analysis was considered in the development of thermal models for the unloading procedures, and that the technical specifications include it, as appropriate. Provide thermal profiles to the materials reviewer so that the latter can determine if the applicant has adequately addressed the issue of fuel rod response to a reflood incident as described in Chapter 8, "Materials Evaluation," of this SRP.

The most extreme thermal conditions may result from credible ambient temperatures, temperature-time histories, an adjacent fire, or any off-normal or design-basis event resulting in blockage of ventilation passages. The worst-case structural loads may occur at temperatures lower than those of design-basis accidents or natural phenomena since load combination expressions effectively require greater safety factors for normal and off-normal analyses than for any design-basis event. Typically, fire has been the worst-case accident thermal condition for storage systems without internal air flow passages.

The burning of fuel and other combustibles associated with vehicles involved in transfer operations should, at a minimum, be presumed to be a design-basis event, with the storage container in the most exposed situation during transfer or loading into storage. The NRC staff has accepted fire parameters included in 10 CFR 71.73, "Hypothetical Accident Conditions," for characterizing the heat transfer during the in-storage fire. However, the staff has also accepted a bounding analysis that limits the fuel source and thus limits the duration of the fire (e.g., by limiting the source to the fuel in the transporter).

Some SSCs may experience the most severe conditions if exposure to high temperatures is followed by dousing with water (such as rain or fire-suppression activities). A small amount of exterior concrete spalling may result from a fire, the application of fire suppression water, rain on heated surfaces, or other high-temperature condition. The damage from these events is readily detectable, and appropriate recovery or corrective measures may be presumed. Therefore, the loss of such a small amount of shielding material is not expected to cause a storage system to exceed the regulatory requirements in 10 CFR 72.106, "Controlled area of an ISFSI or MRS," and need not be estimated or evaluated in the SAR. The NRC accepts that concrete temperatures may exceed the temperature criteria of American Concrete Institute 349, "Code Requirements for

Nuclear Safety-Related Concrete Structures and Commentary," for accidents if the temperatures result from a fire. In that case, corrective action may be required for continued safe storage.

The methods that are acceptable for analyzing and reviewing the consequences of a fire depend upon the duration of the fire and the margin between the predicted temperatures and the actual thermal limits of the components. A fire of sufficient duration, or one in which material temperatures are close to the criteria of their acceptable operational range, will require a detailed model of the storage container and its contents. Storage container system components (e.g., the neutron shield) may be assumed to be intact at the start of the fire.

If a storage container tipover is a credible accident, verify that the applicant has evaluated the effect on storage container and fuel temperatures in the new configuration. An analysis may be warranted when a significant portion of heat removal capability is attributed to internal convection if a change in orientation of that storage container may have a significant effect.

5.5.4.6 *Pressure Analysis*

Pressure calculations should be performed using the ideal gas law (i.e., PV = nRT, where P is pressure, V is volume, n is the number of moles of a gas, R is the ideal gas constant, and T is the absolute temperature) and summing the partial pressures of each of the gas constituents in the storage container cavity. Confirm that the application identifies the method and all assumptions used in the pressure analysis, including the determination of the fission gas inventory.

It is necessary to consider the temperature distribution of all components within the storage container cavity and the cavity walls in calculating the gas pressure in the cavity. For the fire accident analysis, confirm that the application identifies the maximum gas temperature reached during the postfire accident phase, explains the method used to determine the average gas temperature, and specifies the time in the accident at which the peak gas temperature is attained.

This pressure also depends on the free volume in the storage container cavity, the amount (moles) of cover gas (helium) in the cavity, and the amount of gases released from ruptured fuel pins. Review the free volume calculation to determine if all components internal to the storage container cavity (e.g., fuel assemblies, basket, structural supports, spacer disks, reactor control components) have been properly considered.

The NRC accepts that normal conditions occur with less than 1 percent of the fuel rods failed, off-normal conditions occur with up to 10 percent of the fuel rods ruptured, and 100 percent of the fuel rods will have ruptured following a design-basis event. The NRC also accepts that a minimum of 100 percent of the fill gas and 30 percent of the significant radioactive gases (e.g., tritium, krypton, and xenon) within a ruptured fuel rod is available for release into the storage container cavity.

Under the conditions where any of the storage container component temperatures are close (within 5 percent) to their limiting values during an accident, or the maximum normal operating pressure is within 10 percent of its design-basis pressure, or any other special conditions, ensure that the applicant considers, by analysis, the potential impact of the fission gas in the canister (from the effect of its thermal conductivity) on the storage container component temperature limits and the storage container internal pressurization.

Coordinate with the structural reviewer to verify that the confinement pressure of the storage container is within its design limits for normal, off-normal, and accident conditions.

5.5.4.7 *Confirmatory Analysis*

Reviewers may need to perform a confirmatory analysis of the thermal performance of the storage container SSCs identified as important to safety. Confirmatory analyses are recommended if margins between the calculated temperatures and prescribed component temperature limits are small, the applicant has submitted particularly complex thermal analyses, or the applicant is submitting a new thermal methodology or analysis approach.

Ensure that the applicant made the correct assumptions and provided the correct input, and that the output is consistent with established physical (thermal) behavior. These results should specifically include steady-state temperature distributions, local heat balances, temperatures reached and temperature distributions within any reinforced concrete SSCs, and storage container cavity pressures for the bounding ambient temperatures.

To provide the most reliable confirmation, confirmatory analysis should, to the degree possible, use a different thermal analysis method than that used by the applicant. The code used for the confirmatory analysis may be the same as or different from that used by the applicant. Regardless, a review of the applicant's analytical approach and analysis models should be considered part of the overall confirmatory analysis. If necessary, include a confirmatory analysis of accident temperatures (e.g., during a fire), as applicable to the SAR analysis.

If a full confirmatory analysis is not deemed necessary, perform a minimum confirmatory review to verify that the applicant appropriately determined key design parameters and correctly expressed them as input into the computer program(s) used for the thermal analysis. Key parameters include proper dimensions, material properties (including surface emissivities and view factors for radiation), and definition of heat sources. Perform a heat balance at the outer surface of the storage container to verify that the heat from the SNF and insolation balance that removed by convection and radiation. Then assess correlations for the heat transfer coefficient to confirm that they are appropriate for the existing storage conditions. The temperature of the storage container's inner surface should be estimated by calculating the temperature distribution across the storage container body with simple heat balance approximations. Finally, compare the difference between the storage container's inner surface temperature and the maximum cladding temperature with that of similar storage containers and baskets reviewed in previous SARs.

As discussed above, a more detailed confirmatory analysis may be required and could include a model of a portion of the storage container or basket to ensure that the SAR results are realistic and conservative. A more extensive confirmatory analysis may involve the full geometry of the storage container, with relevant component details, to determine temperature distributions in the storage container system.

Appendix 4A to Chapter 4 of this SRP provides additional guidance on reviewing analytical models and conducting confirmatory analyses. NUREG-2152 also provides practical advice for reviewing CFD and heat transfer methods used in vendor applications and for achieving high-quality simulations (confirmatory analysis) of a storage container. To assist in the confirmatory analysis, the report includes procedures, analysis methods, and acceptable assumptions.

As an alternative to a confirmatory analysis, the applicant may be required to perform design-verification testing of an as-built storage container or properly scaled mockup system (when applicable) to confirm the thermal analyses presented in the SAR. Such testing may include verifying gap conductance values assumed in modeling thermal resistance. The test conditions, configuration, and type and location of instrumentation used, if any, should be sufficiently described in the SAR chapter on acceptance criteria and maintenance. Design-verification testing results should be provided in the SAR for storage container certification.

5.5.5 Surveillance Requirements

Active supplemental cooling is permitted in the cases where a limiting condition for operation is not met and an action statement of active supplemental cooling is required in the technical specification surveillance requirements. Verify that the SAR includes technical specifications relating to heat-removal capability. The applicant may have proposed these in compliance with 10 CFR 72.26, "Contents of application: Technical specifications," or they may result from the review and evaluation of submittals relating to those areas. The following is an example of a technical specification related to thermal evaluations that the NRC staff has accepted in previous applications:

> Surveillance requirement: Periodic surveillance will be performed to ensure that there is no blockage of cooling air flow in the heat removal system. This surveillance [typically based on the minimum time for stored material cladding or other material important to safety (e.g., shielding) to reach a threshold temperature in the event of a complete blockage occurring immediately following the prior surveillance and the minimum time to repair or correct the blockage condition] shall be no less frequent than _______ [insert time interval].

Other areas that are often included as part of the technical specifications include, but are not limited to, blockage of inlet ducts, burial under debris, jacket water loss, moisture removal operation (e.g., vacuum drying), multipurpose canister in a transfer cask, fuel-loading operation, fuel-unloading operation, and other short-term operations.

5.6 Evaluation Findings

The NRC reviewer should prepare evaluation findings on satisfaction of the regulatory requirements in Section 5.4 of this SRP. If the documentation submitted with the application fully supports positive findings for each of the regulatory requirements, the statements of findings should be similar to the following:

Certificate of Compliance (**CoC**)

F5.1 SSCs important to safety are described in sufficient detail in the SAR to enable an evaluation of their thermal effectiveness in accordance with 10 CFR 72.236(f) and 10 CFR 72.236(h). Storage container SSCs important to safety remain within their operating temperature ranges in accordance with 10 CFR 72.236(a) and 10 CFR 72.236(b).

F5.2 The [storage container designation] is designed with a heat-removal capability, verifiably and reliably consistent with its importance to safety. The storage container is designed to provide adequate heat removal capacity without active cooling systems in accordance with 10 CFR 72.236(f).

F5.3 The SNF cladding is protected against degradation leading to gross ruptures under normal conditions by maintaining the cladding temperature for [X] years below [X] °C ([X] °F) in an [applicable gas] environment. Protection of the cladding against degradation is expected to allow ready retrieval of the SNF for further processing or disposal in accordance with 10 CFR 72.236(g), 10 CFR 72.236(l), and 10 CFR 72.236(m).

F5.4 The SNF cladding is protected against degradation leading to gross ruptures under off-normal and accident conditions by maintaining the cladding temperature below [X] °C ([X] °F) in an [applicable gas] environment. Protection of the cladding against degradation is expected to allow ready retrieval of spent fuel for further processing or disposal in accordance with 10 CFR 72.236(g), 10 CFR 72.236(l), and 10 CFR 72.236(m).

The reviewer should provide a summary statement similar to the following:

> The staff concludes that the thermal design of the [storage container designation] is in compliance with 10 CFR Part 72, and that the applicable design and acceptance criteria have been satisfied. The evaluation of the thermal design provides reasonable assurance that the [storage container designation] will allow safe storage of SNF for a licensed (certified) life of [X] years. This conclusion is reached on the basis of a review that considered the regulation itself, appropriate regulatory guides, applicable codes and standards, and accepted engineering practices.

<u>Specific License (**SL**)</u>

F5.5 SSCs important to safety are described in sufficient detail in the SAR to enable an evaluation of their heat removal effectiveness in accordance with 10 CFR 72.122(b), 10 CFR 72.122(f), 10 CFR 72.122(i), 10 CFR 72.122(j), 10 CFR 72.122(h) and 10 CFR 72.128(a)(4). Storage container structures, systems, and components important to safety remain within their operating temperature ranges in accordance with 10 CFR 72.92(a), 10 CFR 72.120(a), 10 CFR 120(d), and 10 CFR 72.128(a).

F5.6 [If applicable] The [dry storage system designation] is designed with a heat-removal capability, testable and reliably consistent with its importance to safety in accordance with 10 CFR 72.26, 10 CFR 72.44(c), and 10 CFR 72.128(a)(4).

F5.7 [If applicable] The SNF cladding is protected against degradation leading to gross ruptures under normal conditions by maintaining the cladding temperature for [X] years below [X] °C ([X] °F) in an [applicable gas] environment. Protection of the cladding against degradation will allow ready retrieval of the SNF assembly for further processing or disposal in accordance with 10 CFR 72.122(h).

F5.8 The SNF cladding is protected against degradation leading to gross ruptures under off-normal and accident conditions by maintaining the cladding temperature below [X] °C ([X] °F) in an [applicable gas] environment. Protection of the cladding against degradation is expected to allow ready retrieval of the SNF for further processing or disposal in accordance with 10 CFR 72.122(h).

The staff concludes that the thermal design of [DSF designation] is in compliance with 10 CFR Part 72 and that the applicable design and acceptance criteria as identified in the SAR have been satisfied. The evaluation of the thermal design provides reasonable assurance that [DSF designation] will allow safe storage of SNF. This conclusion is reached on the basis of a review that considered 10 CFR Part 72, appropriate regulatory guides, applicable codes and standards, and accepted engineering practices.

5.7 References

10 CFR Part 71, "Packaging and Transportation of Radioactive Material."

10 CFR Part 72, "Licensing Requirements for the Independent Storage of Spent Nuclear Fuel, High-Level Radioactive Waste, and Reactor-Related Greater than Class C Waste."

American Concrete Institute 349, "Code Requirements for Nuclear Safety-Related Concrete Structures and Commentary."

American Society of Mechanical Engineers (ASME), "Standard for Verification and Validation in Computational Fluid Dynamics and Heat Transfer," (ASME V&V 20)

ASME, Boiler and Pressure Vessel (B&PV) Code, 2007 – Addenda 2008:
Section II, "Materials,"
Section III, "Rules for Construction of Nuclear Facility Components," Division 1, "Metallic Components."

NRC Information Notice 2011-10, "Thermal Issues Identified During Loading of Spent Fuel Storage Casks," May 2, 2011 (ADAMS Accession No. ML111090200).

NRC Information Notice 2014-08, "Need for Continuous Monitoring of Active Systems in Loaded Spent Fuel Storage Canisters (Including Vacuum Drying Process)," May 16, 2014 (ADAMS Accession No. ML14121A089).

Levy, I.S., B.A. Chin, E.P. Simonen, and A.B. Johnson, Jr., "Recommended Temperature Limits for Dry Storage of Spent Light Water Zircaloy-Clad Fuel Rods in Inert Gas," PNL-6189, Pacific (Northwest) National Laboratory, May 1987.

NUREG-2152, "Computational Fluid Dynamics Best Practice Guidelines for Dry Cask Applications: Final Report," March 2013 (ADAMS Accession No. ML13086A202).

NUREG-2174, "Impact of Variation in Environmental Conditions on the Thermal Performance of Dry Storage Cask, Final Report," March 2016 (ADAMS Accession No. ML16081A181).

NUREG-2208, "Validation of Computational Fluid Dynamics Methods Using Prototypic Light Water Reactor Spent Fuel Assembly Thermal-Hydraulic Data," March 2017 (ADAMS Accession No. ML17062A567).

6 SHIELDING EVALUATION

6.1 Review Objective

For certificate of compliance (CoC) applications, the objective of the U.S. Nuclear Regulatory Commission (NRC) shielding review is to ensure that the design features relied on for shielding provide adequate protection against direct radiation from the dry storage system (DSS) contents. The shielding features should limit the direct radiation dose to the operating staff and members of the public so that the total dose (i.e., due to direct radiation and any effluents or releases) remains within regulatory requirements during design-basis normal operating, off-normal (aka anticipated occurrences), and accident conditions (all of which are referred to as design-basis conditions in many locations in this chapter of the Standard Review Plan (SRP)). The review seeks to ensure that the shielding design is adequately defined and evaluated to support the evaluation of the following:

- the DSS's compliance with Title 10 of the *Code of Federal Regulations* (10 CFR) 72.236(d)—the DSS has shielding and confinement features sufficient to meet the requirements in 10 CFR 72.104, "Criteria for Radioactive Materials in Effluents and Direct Radiation from an ISFSI or MRS," and 10 CFR 72.106, "Controlled Area of an ISFSI or MRS."

- the occupational doses from operations with the DSS and adequate consideration of "as low as is reasonably achievable" (ALARA) in the DSS design and operations

The NRC staff conduct an assessment of compliance with these requirements and criteria in its radiation protection review (see Chapter 10B, "Radiation Protection Evaluation for Spent Fuel Dry Storage Systems," of this SRP).

For specific license applications, the objective of the NRC shielding review is to determine whether the shielding design features of the dry storage facility (DSF), whether an independent spent fuel storage installation (ISFSI) or a monitored retrievable storage installation (MRS), meet the NRC criteria for protection against direct radiation from the material to be stored. In particular, this evaluation should establish the validity of dose rate estimates made in the applicant's safety analysis report (SAR). These estimates are in turn used in the radiation protection review (described in Chapter 10A, "Radiation Protection Evaluation for Dry Storage Facilities," of this SRP) to determine (1) compliance with regulatory limits for allowable doses, and (2) conformance with criteria for maintaining ALARA with respect to radiation exposures. The scope for a specific license review is limited to evaluating the shielding for the radiation sources from spent nuclear fuel (SNF), reactor-related greater-than-Class-C (GTCC) waste, or high-level radioactive waste (HLW) to be stored. SRP Chapter 10A and Chapter 13, "Waste Management Evaluation," address other radiation sources at the ISFSI or MRS for which shielding may be required.

6.2 Applicability

This chapter applies to the review of applications for specific licenses for an ISFSI and MRS, categorized as a DSF. It also applies to the review of applications for a CoC for a DSS for use at a general license ISFSI. Sections, paragraphs, or tables that apply only to specific license applications have "**(SL)**" in the heading and apply to all relevant facility design features, operations, and contents. This includes any reactor-related GTCC waste and HLW (for MRSs only) as well as SNF to be stored at the facility and facility structures, systems, and components

(SSCs) and features in addition to the storage containers to be used at the facility. Sections, paragraphs, or tables that apply only to CoC applications have "**(CoC)**" in the heading and apply only to the DSS design features, operations, and contents, which are limited to SNF and the associated radioactive materials (referred to as nonfuel hardware (NFH)). A subsection without an identifier applies to both types of applications; however, the scope of review differs for the two application types.

6.3 Areas of Review

This chapter addresses the following areas of review:

- shielding design description
  - design criteria
  - design features
- radiation source definition
  - initial enrichment
  - computer codes for radiation source definition
  - gamma sources
  - neutron sources
  - other parameters affecting the source term
- shielding model specification
  - configuration of shielding and source
  - material properties
- shielding analyses
  - computer codes
  - flux-to-dose-rate conversion
  - dose rates
  - confirmatory analyses
- consideration of reactor-related GTCC waste storage **(SL)**
- supplementary information

6.4 Regulatory Requirements and Acceptance Criteria

This section summarizes those parts of 10 CFR Part 72, "Licensing Requirements for the Independent Storage of Spent Nuclear Fuel, High-Level Radioactive Waste, and Reactor-Related Greater than Class C Waste," that are relevant to the review areas addressed by this chapter. The NRC reviewer should refer to the exact language in the regulations. Table 6-1a provides the relevant regulatory requirements for a specific license review. Table 6-1b matches the relevant regulatory requirements for the areas of review covered in this chapter for a CoC. The NRC staff reviewer should verify the association of regulatory requirements with the areas of review presented in the tables to ensure that no requirements are overlooked as a result of unique design features.

Table 6-1a Relationship of Regulations and Areas of Review (SL)

Areas of Review	10 CFR Part 72 Regulations						
	72.24	72.104 (a)	72.106 (b)	72.120	72.122 (b)(2) (i),(c),(e)	72.126 (a)(6)	72.128 (a)(2)
Shielding Design Description	(b)(c)(e)	●	●	(a)(b)(c)	●	●	●

Radiation Source Definition	(c)	●	●	(b)(c)			
Shielding Model Specification	(b)(c)(e)	●	●	(b)(c)	●	●	●
Shielding Analyses	(m)(e)	●	●	(b)(c)	●	●	●
Consideration of Reactor-Related GTCC Waste Storage	(b)(c)(e)	●	●	(a)(b)(c)	●	●	●

Areas of Review	10 CFR Part 20 Regulations		
	20.1201 (a)(1)(2)	20.1301(a)(b)	20.1302(b)
Shielding Design Description	●	●	
Radiation Source Definition	●	●	
Shielding Model Specification	●	●	
Shielding Analyses	●	●	●
Consideration of Reactor-Related GTCC Waste Storage	●	●	●

Table 6-1b Relationship of Regulations and Areas of Review (CoC)

Areas of Review	10 CFR Part 72 Regulations		
	72.104(a)[A]	72.106(b)[A]	72.236
Shielding Design Description	●	●	(b)(d)(g)
Radiation Source Definition	●	●	(a)
Shielding Model Specification	●	●	(d)(g)
Shielding Analyses	●	●	(d)(g)

A This requirement applies to CoCs and CoC applications through the requirement in 10 CFR 72.236(d).

The regulations in 10 CFR Part 72 require that SNF (including NFH), reactor-related GTCC waste and HLW storage and handling systems be designed with adequate shielding to provide sufficient radiation protection under normal, off-normal, and accident conditions. The SAR should describe the design principles and functional features of the SSCs important to safety that are relied on for shielding in sufficient detail to allow the NRC staff to thoroughly evaluate their effectiveness. It is the responsibility of the applicant to analyze such SSCs with the objective of assessing the impact of direct radiation doses and effluent releases to the environment on public health and safety.[1] The NRC reviewer should verify the applicant's evaluations through review of the applicant's model and, as needed, through confirmatory analyses or independent modeling analysis. In addition, SSCs important to safety should be designed to withstand the effects of both credible accidents and severe natural phenomena without impairing their capability to perform their safety functions. While only applicable to licenses, 10 CFR 72.122(b) and (c) provide a list of the kinds of conditions for which a DSS should be designed and evaluated in a CoC application.

(CoC) Several technical and licensing factors should be considered during the shielding evaluation. First, 10 CFR Part 72 specifies regulatory dose limits in terms of annual doses for normal conditions and total dose from accident conditions. These limits apply to individuals located at or beyond the controlled area boundary of a DSF. The regulations do not specify dose rate limits for DSS surfaces nor at set distances from DSS surfaces, unlike the package dose rate

[1] For CoC applications, as noted in other sections of this guidance, the general licensee is responsible for the ultimate assessment of these impacts for its use of the DSS design in an approved CoC.

limits in 10 CFR Part 71, "Packaging and Transportation of Radioactive Material." Therefore, responsibility for determining compliance with the dose limits in 10 CFR 72.104(a) and 10 CFR 72.106(b) ultimately rests with the general licensee that uses the DSS at its ISFSI (see 10 CFR 72.212, "Conditions of general license issued under § 72.210," which places this responsibility with the general licensee; compliance is verified by inspection). This is because compliance with these kinds of limits considers factors that are specific to the general licensee's site. These factors include the geometric arrangement of DSS arrays, topography, distances to the controlled area boundary, distances to dose receptors, exposure times of dose receptors, actual SNF loading patterns in each DSS, and dose contributions from other surrounding fuel-cycle facilities. Because the SAR is only for a DSS design that is intended to be usable by general licensees, the SAR analyses cannot fully address these factors for sites at which the DSS might be used. This does not mean, however, that compliance with the requirements in 10 CFR 72.104 and 10 CFR 72.106 is the sole responsibility of the licensee. As stated in 10 CFR 72.234(a), the certificate holder and applicant for a certificate must ensure that the design, fabrication, testing, and maintenance of a DSS comply with the requirements in 10 CFR 72.236. This includes 10 CFR 72.236(d), which requires the CoC applicant to demonstrate that the DSS shielding, together with the DSS confinement, is sufficient to meet the requirements in 10 CFR 72.104 and 10 CFR 72.106. Given the site-specific factors that do bear upon compliance with those requirements, the typical acceptance criteria for DSS shielding define standard analyses for single DSSs, and a generic array of DSSs, to demonstrate a sufficient shielding design and compliance with 10 CFR 72.236(d).

(CoC) In general, the DSS shielding evaluation should provide reasonable assurance that the proposed design fulfills the following acceptance criteria:

- The radiation shielding features of the proposed DSS must be sufficient for it to meet the radiation dose requirements in 10 CFR 72.104. The applicant demonstrates this by providing the following:

  – a shielding analysis of the surrounding dose rates that contribute to offsite doses at appropriate distances (for a single storage overpack and transfer cask (for a canister-based DSS) or a single cask (for a non-canister-based DSS) with bounding fuel source terms at various overpack and transfer cask, or cask, locations) for normal conditions and anticipated occurrences (that is, off-normal conditions)

  – a shielding analysis of a single DSS and a generic array of DSSs at appropriate distances

- DSS contents and design features important to and relied on for shielding are adequately described for evaluating shielding effects and dose rates. Dose rates are evaluated for an adequate number of appropriate locations around the DSS for different operations configurations to enable evaluation of occupational dose estimates and evaluation of ALARA.

- Radiation shielding features must be sufficient for the design to meet the requirements in 10 CFR 72.106. The applicant demonstrates this by calculating dose rates and doses at appropriate distances for different accident conditions for appropriate DSS configurations and appropriate assumptions regarding accidents (e.g., duration, including time to recover from or repair the effects of the accidents).

- The proposed shielding features should enable a general licensee that uses the DSS to meet the regulatory requirements prescribed in 10 CFR Part 20, "Standards for Protection Against Radiation."

- Appropriate distances for the foregoing criteria are distances that are consistent with, or bounding for, the distances to the controlled area boundaries of potential DSS users. The minimum distance to the controlled area boundary is 100 meters (328 feet).

(SL) As described in the guidance for CoC applications, 10 CFR Part 72 only specifies dose limits for individuals located at or beyond the controlled area boundary; it does not specify dose rate limits for storage containers such as DSSs. Demonstration of compliance with the limits necessarily considers factors associated with the facility's site. For specific license applications, the site and its surroundings are known. Thus, site factors should be considered as part of the applicant's analysis, or the applicant should provide an analysis that is bounding for its site and describe how the analysis is bounding. The contents to be stored at the site are limited in characteristics and in quantity. Thus, the analysis should be bounding for the characteristics of what is to be stored at the site and should account for the maximum quantity to be stored at the site. This means the analysis should account for the number, configuration(s), and size(s) of the array(s) that will be employed at the site. Additionally, the SAR should describe the locations of members of the public (e.g., residences, places of work, and public access facilities or areas) and projections of changes known for the site (see Chapter 2, "Site Characteristics Evaluation for Dry Storage Facilities," of this SRP). The application should also include a description of the controlled area and any restricted areas on the facility site. Site topography is also fixed for the site. These factors should be appropriately accounted for in the analysis or bounded by the analysis.

(SL) In the radiation protection chapter (see SRP Chapter 10A), the SAR should include evaluations that demonstrate that the facility design and operations meet, or will meet, the requirements in 10 CFR 72.104 and 72.106 and 10 CFR Part 20. The shielding analysis should be adequate to support that demonstration. This includes providing dose rates for (1) the storage container (e.g., DSS) surfaces and near the containers, (2) the surfaces and vicinity of other facility SSCs used to handle or transfer the material stored at the site, and (3) locations around the facility, including within restricted areas and in facility buildings and structures where facility personnel will be, or may be, located. The dose rates should include dose rates for different phases of operations and container configurations and should address the effects of different conditions (normal, off-normal, accident, which include natural phenomena) during these operations. The dose rate estimates should be sufficient in number and location to support evaluation of occupational doses and incorporation of ALARA as well as doses to members of the public.

The acceptance criteria also help to ensure the dose rates associated with the DSS or DSF are reasonable and acceptable. The acceptance criteria also help to ensure that the methods used to calculate the dose rates are appropriate and acceptable in terms of the methods' use to demonstrate the DSS's or DSF's SSCs, as described in the application, fulfill the shielding safety function. The staff should be aware of the potential for further use of these methods and may therefore need to place additional emphasis on appropriate acceptance criteria related to methods. Such a review, however, still does not constitute approval of the methods outside of their use to demonstrate that the DSS or DSF, as described in the application, meets the shielding requirements.

In order to ensure that the shielding design of the DSS or DSF meets the regulatory requirements as defined in 10 CFR Part 72, the applicant should also include information in the SAR regarding the technical specifications that are necessary for the DSS or DSF to meet the dose limits at the controlled area boundary (see SRP Chapter 17, "Technical Specifications Evaluation"). The requirements to be included in technical specifications are described in 10 CFR 72.44(c). While only applicable to specific licenses, the information in 10 CFR 72.44(c) can be useful in determining the information needed in CoC conditions, including those referred to as technical specifications, to ensure compliance with 10 CFR 72.234(a) and 10 CFR 72.236.

6.4.1 Shielding Design Description

(SL) For a specific license, 10 CFR 72.126, "Criteria for radiological protection," and 10 CFR 72.128, "Criteria for spent fuel, high-level radioactive waste, and other radioactive waste storage and handling," require that the applicant describe the storage and handling systems requiring shielding. The SAR must provide design criteria and descriptions of design features relied on for shielding for facility features and facility SSCs that are used to store, handle, or transfer the material to be stored at the facility in accordance with 10 CFR 72.24(b) and 10 CFR 72.24(c).

6.4.1.1 *Design Criteria*

The requirements in 10 CFR 72.104 and 10 CFR 72.106 provide dose limits for the members of the public around a DSF site (i.e., offsite). The SAR chapter on principal design criteria should specify the criteria that have been used as a basis for protection against direct radiation. Design criteria should include the identification of maximum dose rates and should also be specified for occupancy areas and correlated with occupancy duration and distance to radiation sources.

The design should consider the ALARA principle. For CoC applications, the NRC reviewer should note that it is the responsibility of the general licensee using the DSS design to develop detailed procedures that incorporate the ALARA objectives of its site-specific radiation protection program. However, the DSS design should reflect appropriate consideration of ALARA to the extent practical. For specific license applications, the SAR should include sufficient information to demonstrate incorporation of ALARA into the facility design, including facility layout, and operation procedures. SRP Chapters 10A and 10B (radiation protection) provide further information on ALARA considerations that apply to the respective reviews.

(SL) In addition to the limits in 10 CFR Part 72, 10 CFR 20.1201, "Occupational dose limits for adults," and 10 CFR 20.1301, "Dose limits for individual members of the public," prescribe additional dose limits for personnel and for members of the public, respectively.

6.4.1.2 *Design Features*

The SAR should describe the material and geometric properties of all design features relied on to reduce direct radiation dose rates and may consider the following:

- self-shielding provided by the radioactive material being stored
- shielding provided by the structural and nonstructural materials forming the DSS or DSF SSCs (e.g., a SNF cask, overpack, or transfer cask)
- neutron capture provided by borated materials incorporated into the DSS or DSF SSCs

- shielding provided by the temporary placement of water into DSS or DSF SSCs (e.g., into SNF canister and transfer cask) during loading and unloading procedures

- shielding provided by temporary placement of equipment and portable shields on and around the DSS or DSF SSCs during loading and unloading procedures (for DSSs, this means only those items that are part of the DSS design)

- shielding provided by natural or human-made, engineered (e.g., berms or shield walls) barriers between the radioactive material and the area beyond the controlled area boundary; human-made, or engineered features, used for this purpose (i.e., to ensure compliance with regulatory dose limits such as 10 CFR 72.104(a)) should be classified as important to safety at the appropriate category. Such features are most likely not part of DSS designs and analyses, though they may be.

(SL) The guidance in the preceding list applies to all DSF SSCs, not just the SSCs associated with the storage containers used at the DSF. The following includes some examples of these other DSF SSCs to which the preceding guidance list applies:

- shielding provided by pool or other site facility SSCs, including interior and exterior walls
- shielding to reduce dose to personnel in site facilities such as the administrative building

The SAR should describe the geometric arrangement of shielding and include illustrations that identify the spatial relationships among sources, shielding, and design dose rate locations. For specific license applications, this description should include scaled layout and arrangement drawings of the facility that show the locations of all sources and facility SSCs and features. The SAR should clearly indicate the physical dimensions of sources and shielding materials. The SAR should also identify penetrations, voids, or irregular geometries that provide potential paths for gamma or neutron streaming. Any submitted drawings should clearly identify these potential streaming paths. The SAR should describe design features used to minimize streaming through these penetrations.

The SAR should adequately describe the material properties and specifications, including composition of the items relied on for shielding. This information is particularly needed for nonstandard or proprietary materials such as proprietary polymer-based neutron shielding. The SAR should include appropriate references for the nonstandard or proprietary materials. Additionally, the technical design (or engineering) drawings should include material specifications important to the performance of the shield materials. These specifications include items such as the industry standard for the specifications of the lead gamma shielding and the minimum mass density, hydrogen composition, and boron composition of polymer-based neutron shielding.

The SAR should clearly state any differences in shielding features (material properties, geometry, and dimensional changes) for normal, off-normal, and accident conditions. These differences may be from effects such as physical impacts and material property changes caused by temperature effects. The SAR descriptions for the different conditions should consider different operating configurations that, though temporary, affect how the different conditions may affect the DSS or DSF shielding features. For example, a DSS or DSF design that relies on soil providing shielding for the storage containers should address the impacts of normal, off-normal, and accident conditions for excavation (to expand the storage array) next to operating (i.e., loaded) storage containers.

6.4.2 Radiation Source Definition

The SAR should describe the radioactive contents to be stored. For CoC applications, the allowable contents are limited to SNF and any NFH to be stored with the SNF. That description should include the condition of the SNF (e.g., undamaged, damaged). For specific license applications, the contents may also include solid reactor-related GTCC waste and, for MRSs, HLW. The SAR should include an adequate description of these items, including the physical and chemical form(s), radionuclide content, and geometric configuration(s).

The SAR should describe each type of contained radiation source used as a basis for the shielding design calculations. The source terms should be described in a format that is compatible with the shielding calculation input. For SNF, the source terms in particles per second per metric ton of uranium (MTU) (or metric ton heavy metal (MTHM) for mixed-oxide (MOX) SNF) or per assembly (e.g., neutron per second per MTU (n/s/MTU), gamma per second per assembly (γ/s/assembly)) or, for gammas, million (mega) electron volts per second (MeV/s) per MTU (or MTHM for MOX SNF) or per assembly (i.e., MeV/s/MTU or MeV/s/assembly) should be described in the form of either a group structure or a continuous function of energy. For assembly hardware and NFH, the source can be described in terms of the nuclide(s) in the hardware and the activity (in curies or becquerels) of the nuclide(s). For reactor-related GTCC waste and HLW contents in specific license applications, the SAR should specify the isotopic composition and photon yields and, as appropriate, neutron yields for each constituent in the waste.

The SAR should clearly present the data used as input for calculating the radiation source terms and include the bases for the parameter values selected for the input. This includes any material property, physical dimension, and irradiation history values that differ from the actual properties of the radioactive contents or are derived from assumptions (e.g., assumed down time between irradiation cycles for SNF). The applicant should show that the selected input values result in appropriate or conservative results. The energy group structure from the source term calculation should correspond to that of the cross-section set of the shielding calculation. In addition, the SAR should specify the computer methodology or database application used to compute source term strength.

The SAR should include a discussion of energetic radiations created by nuclear reactions such as (n, γ) in the materials and the contents of the DSS or DSF SSCs. The SAR should also provide source-term descriptions for induced radioactivity and the bases (assumptions and analytical methods) used for their estimation. For example, high-energy (approximately 6.7-MeV) gammas may be generated by the (n, γ) reaction of thermalized neutrons and the iron in the steel shell that is typically used to contain liquid or polymer-based neutron shields. Alternatively, the SAR may describe the bases for excluding induced radioactivity source terms.

6.4.2.1 *Gamma Sources*

The SAR should specify gamma source terms for both SNF and activated materials. Most hardware source terms will be from cobalt-60; however, some NFH may include other activated nuclides that should be evaluated (e.g., materials containing hafnium or silver-indium-cadmium). For reactor-related GTCC waste and HLW contents in specific license applications, the isotopic composition and photon yields for each constituent should be specified. A tabulated form of the radiological characteristics is acceptable.

6.4.2.2 *Neutron Sources*

The SAR should also describe the neutron source terms, both total strength and spectrum, for the SNF and for neutron sources and neutron source assemblies (NSAs) included as NFH contents. The description should also include the bases used to determine the source terms. The SAR should also describe how the analysis addresses neutrons from subcritical multiplication. For reactor-related GTCC waste and HLW contents in specific license applications, similar information should be included in the SAR for neutron sources in these wastes, if applicable. The neutron source term for these wastes may be specified in terms of the constituent radionuclides with their respective neutron yields and spectra. Alternatively, contents limits in the license conditions or technical specifications may limit these wastes such that they have a negligible neutron source. In that case, the SAR should describe how the contents specifications result in a negligible neutron source from these wastes and thus neutron source information is not needed for them.

6.4.3 Shielding Model Specification

The SAR should identify the models used in the analysis and include information on materials and arrangements of sources and design features included in the models. As described in Sections 6.4.3.1 and 6.4.3.2 below, the SAR should clearly present the data used in the analyses, identifying differences between actual properties and modeled properties of SSCs and features and of material to be stored and justifying the acceptability of those differences, whether they are from simplifications or assumptions or other reasons.

6.4.3.1 *Configuration of Shielding and Source*

The SAR should include descriptions of how the sources and DSS or DSF design SSCs and features are included in the analysis models. The SAR should justify how the models adequately include the sources and DSS or DSF design SSCs and features. The SAR should also justify any simplifications of features in the model and, for features that are not represented in the models, the acceptability of not including these features in the models. The analysis should include models that represent the source and design feature configurations that are appropriate for the different stages of operations (e.g., storage at the pad, loading, draining, and drying) and are appropriate for normal, off-normal, and accident conditions for the different stages of operations. The models should consider the information in Sections 6.4.1 and 6.4.2 of this SRP and, for SNF contents, the condition of the SNF (e.g., undamaged, damaged, debris). The analysis models should also include appropriate or bounding physical distribution(s) of the source term(s). See the section of Chapter 8, “Materials Evaluation,” of this SRP that discusses the condition of SNF.

6.4.3.2 *Material Properties*

The SAR should describe how materials specifications and properties for the DSS or DSF contents and design features are included in the models. The SAR should justify that the materials properties in the models are adequate, bounding, or otherwise appropriate for representing the materials that comprise the DSS or DSF contents, SSCs and design features for different configurations (e.g., damaged SNF vs. undamaged SNF), conditions (i.e., normal, off-normal, accident conditions) and operations configurations (e.g., draining, drying, storage at the pad), considering the information in Sections 6.4.1 and 6.4.2 of this SRP chapter.

6.4.4 Shielding Analyses

The SAR should describe the computer codes, including version; computational models; data; and assumptions with their bases used in evaluating shielding effectiveness. It should provide dose rate estimates for areas of concern, as described near the beginning of the shielding evaluation acceptance criteria.

6.4.4.1 *Computer Codes*

The SAR should identify the computer codes used in the shielding evaluation, including codes for calculating the source term descriptions identified in Section 6.4.2 above and codes for calculating dose rates, and reference the appropriate documentation. For each computer code used, the SAR should provide test problem solutions that demonstrate substantial similarity to solutions from other sources (e.g., hand calculations, published literature results). The SAR should provide a summary that compares the test problem solutions in either graphical or numeric form. However, these solutions may be referenced and need not be submitted in the SAR if the references are widely available or have been previously submitted to the NRC for the same computer code and version.

The SAR should address calculational error (i.e., standard error) and uncertainties in computer codes for both radiological and thermal source terms. Because validation data are relatively limited for burnups above 45 gigawatt days/MTU (i.e., high burnup fuel), the SAR should numerically specify radiological and thermal source term uncertainties for high burnup fuels.

The SAR should determine whether and how source term values with uncertainties should be applied to the shielding analysis. The applicant may do this by making adjustments to the source term or by compensating in other aspects of the shielding analysis. In this determination, the SAR may consider the following:

- other conservative assumptions and design margins in the analysis
- the maximum fuel assembly heat loads for the design basis fuel, burnup, enrichment, and cooling time
- the maximum gamma and neutron dose rates (including relative contributions to total dose rates)
- any measurable dose rate limitations proposed in the technical specifications
- the gamma and neutron sources corresponding to the design basis decay heat limit

The applicant should calculate dose rates with a code that is capable of handling the geometries and configurations of the DSS or DSF design features and SSCs and the contents (i.e., SNF, reactor-related GTCC waste, or HLW) during the different stages of storage operations for normal, off-normal, and accident conditions. This includes storage containers (e.g., DSS) that have axial or radial variations in features relied on for shielding, inlet and outlet vents, and other features that can be streaming paths and, for a DSF, variations in facility features that can affect dose rates. This also includes configurations of contents that result in variations in the physical distribution of the contents' source term, which can also affect dose rates. The SAR should include a description of the shielding code that is sufficient to justify that it is adequate to determine dose rates for the DSS or DSF, considering the DSS's or DSF's design SSCs and features that affect shielding.

The SAR should include representative computer code input files for the different types of calculations done to support the shielding analysis.

6.4.4.2 *Flux-to-Dose-Rate Conversion*

The SAR should state the flux-to-dose-rate conversion used in the shielding analysis, including conversions that are done by a computer code using its own data library, and the basis for using that conversion(s). The SAR should include a table that shows the one-to-one conversion factor for each energy group of the source term spectra. The NRC accepts the flux-to-dose-rate conversion factors in American National Standards Institute (ANSI)/American Nuclear Society (ANS) 6.1.1-1977, "Neutron and Gamma-Ray Flux-to-Dose Conversion Factors."

6.4.4.3 *Dose Rates*

(CoC) The SAR evaluation of shielding effectiveness should include calculated or estimated dose rates in representative areas around the DSS and at appropriate distances from the DSS. The SAR should clearly indicate the locations on and around the DSSs and the distances from the DSSs for which dose rate calculations have been performed. The selected locations should be adequate to support determination of occupational dose estimates and doses to members of the public described in Chapter 10B of this SRP, demonstrating consideration of the following:

- locations on or in the immediate vicinity of DSS surfaces and at appropriate distances from the DSS where workers will perform operations during loading, retrieval, handling, maintenance, and surveillance activities

- locations of DSS features and surfaces with potentially elevated dose rates or streaming paths such as (labyrinthine) air flow passages; the SAR should include dose-rate estimates for these areas (e.g., air inlets and outlets)

- locations or distances appropriate for determining doses to individuals at or beyond the controlled area boundary (minimum distance to the boundary must be at least 100 meters (328 feet) in accordance with 10 CFR 72.106(b)); locations should be sufficient to develop dose-to-distance curves for a single DSS and a sample array of DSSs for 10 CFR 72.104 evaluations

- locations or distances appropriate for determining doses to individuals at or beyond the controlled area boundary from accidents for 10 CFR 72.106 evaluations

- potential use of some dose rates as limits in the technical specifications

(CoC) Dose rates should be calculated for the variety of DSS configurations that exist at different stages of DSS operations (e.g., storage at the ISFSI pad, DSS loading, DSS welding). Also, dose rates should be calculated for normal conditions, anticipated occurrences, and accidents and natural phenomena to enable evaluation of the doses for each of these conditions.

(CoC) For canister-based systems, the system includes a transfer cask and a storage overpack. Thus, for these DSSs, the various conditions for the different DSS configurations include the transfer cask and overpack. The overpack is a passive, engineered SSC that provides the necessary radiation shielding during storage on the DSF pad. As of the publication of this SRP, overpack designs have included vertical concrete or metal silos, concrete modules, and designs for vertical storage systems that rely on engineered fill and the surrounding soil as the “overpack.” For DSSs with the latter kind of overpack, dose rate analyses should also address normal, off-normal, and accident conditions with excavation (to expand the storage system array) next to loaded systems. This information will support any needed technical specification to limit the proximity of excavation to loaded systems. Transfer casks may also include or make use of supplemental shielding that is necessary for personnel to be able to perform some operations involving the loaded transfer cask (i.e., the third type of supplemental shielding described in the term’s definition in the SRP glossary). The applicant should consider configurations with and without this supplemental shielding in the dose rate analyses for the transfer cask. For non-canister-based systems, all configurations and conditions will involve a single cask, which is used for all operations.

(SL) The SAR evaluation of shielding effectiveness should include calculated or estimated dose rates in representative areas around the storage containers (e.g., SNF container, GTCC waste container) and at appropriate distances from the storage containers and at appropriate locations within, at, and beyond the controlled area boundary. The SAR should clearly indicate the locations on and around the containers and the distances from the containers for which dose rate calculations have been performed. The SAR should clearly indicate the locations within the facility (e.g., within the restricted area, in areas of container-handling buildings, and administrative buildings) and locations at and beyond the controlled area boundary for which dose rates were calculated. The selected locations should be adequate to support determination of occupational dose estimates and doses to members of the public described in Chapter 10A of this SRP, demonstrating consideration of the following:

- locations on or in the immediate vicinity of container surfaces and at appropriate distances from the container and the surfaces of and appropriate distances from facility SSCs used to handle, transfer, or store the containers where workers will perform operations during loading, retrieval, handling, maintenance, and surveillance activities

- locations of container features and surfaces with potentially elevated dose rates or streaming paths such as (labyrinthine) air flow passages; the SAR should include dose rate estimates for these areas (e.g., air inlets and outlets)

- locations or distances appropriate for determining doses to individuals at or beyond the controlled area boundary (minimum distance to the boundary must be at least 100 meters (328 feet) (see 10 CFR 72.106(b))); locations should be sufficient to evaluate doses for members of the public and should include residences, businesses and other places of work, recreational facilities and areas, and other public access facilities and areas around the DSF for 10 CFR 72.104 evaluations

- locations or distances appropriate for determining doses to individuals at or beyond the controlled area boundary from accidents for 10 CFR 72.106 evaluations

- Locations where personnel will be working to support DSF operations (e.g., administrative buildings)

- Locations of public access facilities and areas, including throughways (e.g., roads, highways, waterways, railways) that traverse through the controlled area

- Facility layout and locations of personnel performing DSF operations related to that layout (e.g., surveillance or maintenance conducted on a storage container within an array of containers at a single storage pad, surveillance of containers on one pad from locations surrounded by other pads for a multi-pad facility)

- Potential use of some dose rates as limits in the technical specifications

(SL) The SAR should include calculated dose rates for the variety of container configurations that exist at different stages of storage operations (e.g., storage at the DSF pad, container loading, container welding). Further, dose rates should be calculated for normal conditions, anticipated occurrences, and accidents and natural phenomena to enable evaluation of the doses for each of these conditions. The preceding CoC discussion related to canister-based systems and non-canister-based systems should also be considered, as applicable, for the storage containers to be used at the DSF. Additionally, any supplemental shielding (e.g., berms or shield walls) included in the estimates to demonstrate compliance with dose limits should be classified as important to safety at the appropriate category.

6.4.5 Consideration of Reactor-Related GTCC Waste Storage (SL)

(SL) As described in the preceding sections, an applicant that proposes to store reactor-related GTCC waste at its DSF should ensure that the shielding analysis includes the reactor-related GTCC waste. The applicant should further ensure that the SAR includes all appropriate information to support that analysis. This includes a description of the forms and compositions of different types of reactor-related GTCC waste (e.g., steel core baffle plates), the characterization of the radionuclides and their activities, the total amount of reactor-related GTCC waste to be stored at the facility, and a description of the SSCs, including the containers, used to handle, transfer, and store the reactor-related GTCC waste. The SAR should clearly state that the reactor-related GTCC waste is in solid form since only solid reactor-related GTCC waste may be stored under 10 CFR Part 72. The results of the shielding analysis should include dose rates that can be used to estimate occupational doses for operations for the reactor-related GTCC waste, including the different configurations of SSCs at the different operations stages. The shielding analysis results should include dose rates that include the impacts of reactor-related GTCC waste storage operations for evaluating the doses to members of the public for normal, off-normal, and accident conditions from DSF operations. These dose rates are used in the radiation protection evaluations (see Chapter 10A) to demonstrate facility design and operations meet, or will meet, the requirements in 10 CFR 72.104, 10 CFR 72.106, and 10 CFR Part 20.

(SL) There are multiple ways for the shielding analysis to address reactor-related GTCC waste. First, as may be done for analyses for SNF or HLW contents, the applicant may simply perform dose rate calculations for each type of reactor-related GTCC waste, or the applicant may choose to perform dose rates for a bounding reactor-related GTCC waste type. For this second option, the applicant should demonstrate that the selected waste type results in bounding dose rates for all the reactor-related GTCC waste types to be stored at the DSF. Such a demonstration would include the waste characterization, including the physical distribution of the radionuclides within the waste and any changes to that distribution resulting from the different conditions of operations. Additionally, the applicant may choose to demonstrate that the dose rates for reactor-related GTCC waste are bounded by the dose rates for the SNF or HLW to be stored at the DSF and apply the SNF or HLW dose rates to the reactor-related GTCC waste. If the reactor-related GTCC waste is handled and stored in the same containers as the SNF or HLW, then demonstrating that the bounding reactor-related GTCC waste source term is bounded by the SNF or HLW source term in total strength and across the energy spectra may be sufficient. A final option is that, in the case that the radiation protection evaluation indicates significant margins to the limits in 10 CFR 72.104 for analysis with just the SNF and HLW, as applicable, the applicant may choose to demonstrate that dose rates from GTCC waste are insignificant in comparison with the SNF or HLW dose rates. In this instance, the DSF dose rates would not need to include the reactor-related GTCC waste contribution. This last option only applies to the normal and off-normal conditions dose rates analysis for the 10 CFR 72.104 evaluation. For any one of these options, the SAR should include the appropriate information to support the selected analysis approach.

6.5 Review Procedures

Figures 6-1a and 6-1b show the interrelationship between the shielding evaluation and the other areas of review described in this SRP for specific license and CoC applications, respectively.

Coordinate with the technical specifications reviewer (SRP Chapter 17) to ensure that the license and CoC conditions and technical specifications adequately capture those items that (1) for a DSF, are necessary for the DSF to meet the regulatory dose limits, or (2) for a DSS, are necessary for the DSS to function to enable general licensees that use it to meet the regulatory dose limits. Make a determination that descriptions of the DSS or DSF in the SAR provide the information needed to evaluate the DSS or DSF shielding in the context of its proposed use and operations.

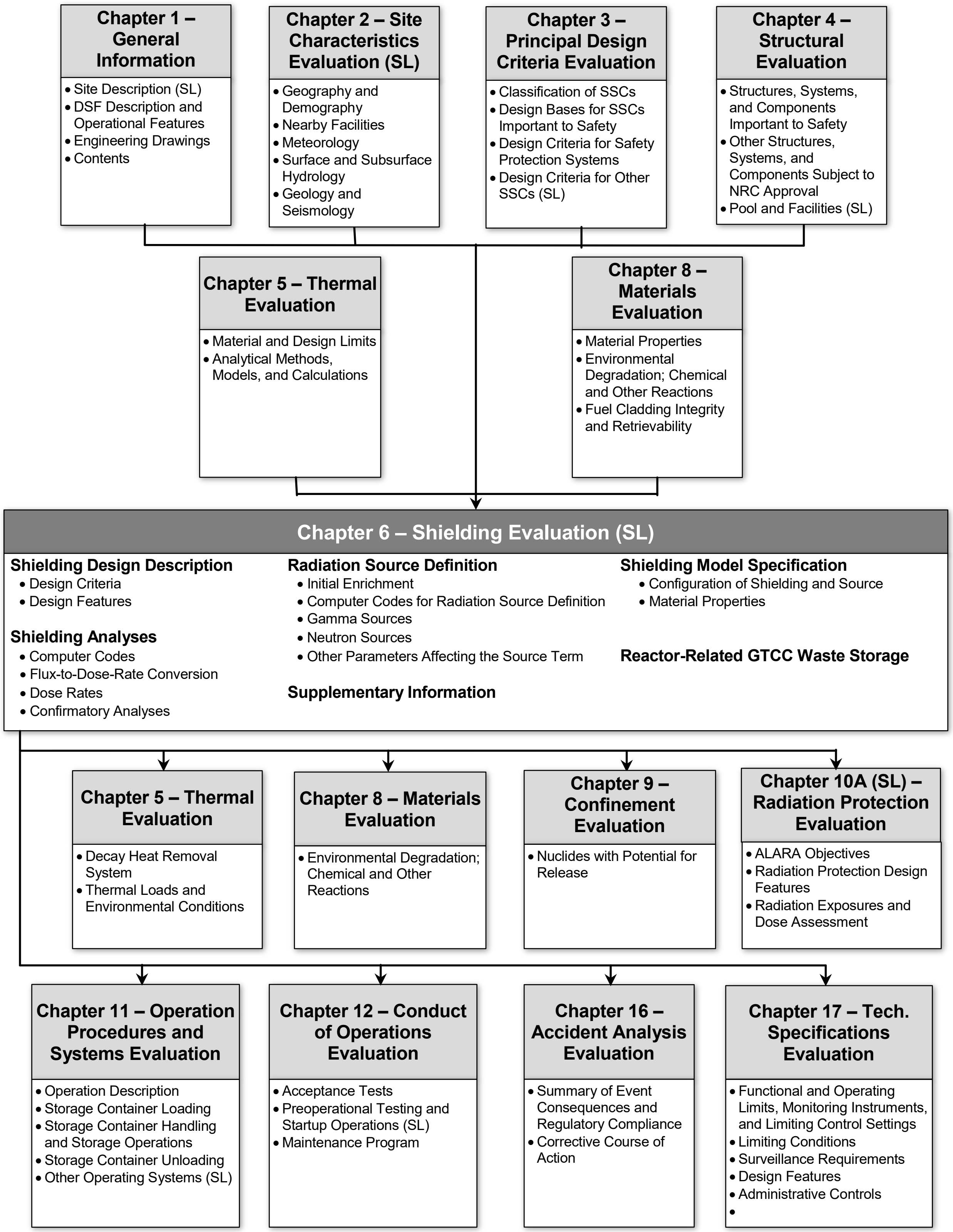

Figure 6-1a Overview of Shielding Evaluation of Specific License Applications for a DSF (SL)

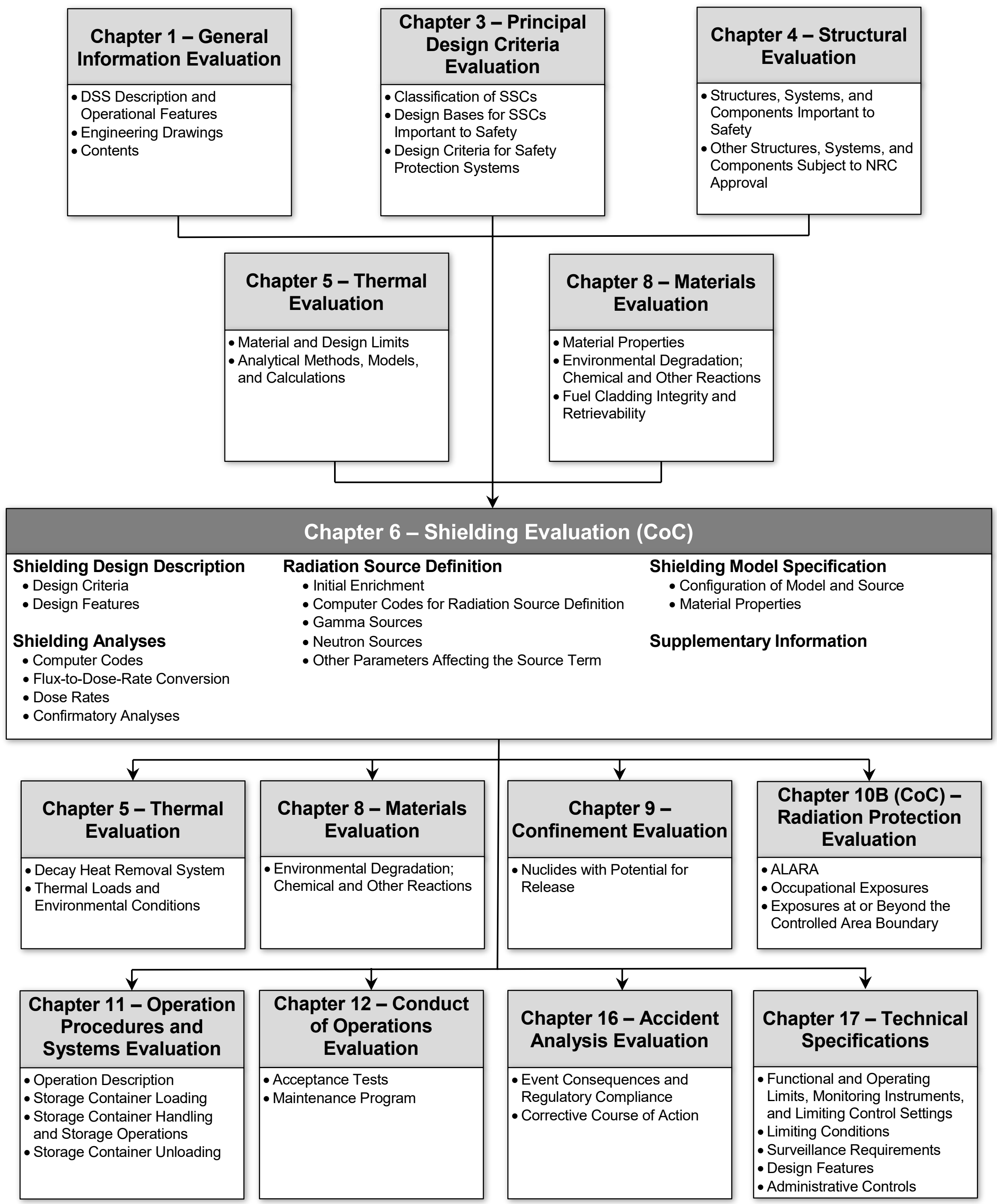

Figure 6-1b Overview of Shielding Evaluation of Applications for a DSS (CoC)

6.5.1 Shielding Design Description

6.5.1.1 *Design Criteria*

Verify that applicant has used specific design criteria and that the SAR describes the criteria as the basis for the shielding design and protection against direct radiation. These criteria may include specification of appropriate maximum dose rates for the variety of storage container (e.g., DSSs) configurations during storage operations for important and relevant container features. For specific license applications, these dose rate criteria may also include DSF SSCs involved in the handling, transfer, and storage of SNF, reactor-related GTCC waste, and HLW. Dose rates at the container surface and in the vicinity of a loaded container may vary during the different stages of storage operations (i.e., loading and unloading, activities to prepare for storage or unloading such as canister welding, canister opening, transferring to and from the storage pad, and activities conducted while the container is at the DSF storage pad).

While 10 CFR Part 72 establishes dose limits for DSFs, it does not impose specific dose rate limits on individual storage containers. The NRC has accepted SNF DSS (cask or storage overpack) storage surface dose rates from 20 to 450 millirem per hour in evaluations for previous CoC applications. For canister-based DSSs, these dose rates apply only to the storage overpack. Surface dose rates for transfer casks for these DSSs are noticeably higher. The surface dose rates for the majority of the transfer casks have not exceeded 2 rem per hour. Some instances with higher transfer cask dose rates have been accepted with technical specifications or conditions in the CoC in addition to the dose rate limit conditions, which are usually established for transfer casks. Coordinate with the radiation protection reviewer (for example, see Chapter 10A, Section 10A.5.2.3, or Chapter 10B, Section 10B.5.1, of this SRP) and technical specifications reviewer (Chapter 17 of this SRP) to determine the necessary license or CoC conditions and technical specifications for the DSS or DSF storage containers, including both the storage overpack and the transfer cask for canister-based designs.

Acceptable dose rates depend on a number of factors, including both the transfer cask and storage overpack for canister-based storage container designs. These factors include (1) the geometry of the storage array, (2) the time workers will routinely spend in the storage array for activities such as monitoring or maintenance, (3) the proximity to other areas frequently occupied by workers, (4) the proximity to the controlled area boundary or other public access areas, (5) the need for unique operation techniques (e.g., remote operations using remote optical systems to perform actions), (6) recovery from off-normal events requiring actions and proximity to SSCs significantly different from normal operations, and (7) limitations or other requirements imposed in the technical specifications for operations with the storage container design. At least some of these factors are specific to individual licensee sites and so are most directly applicable to specific license applications. However, for CoC applications, consider reasonable expectations and estimates for these factors and the implications for a licensee’s ability to meet regulatory dose limits as appropriate in determining the acceptability of the storage containers’ dose rates. This includes the dose rates for both the transfer cask and the storage overpack of canister-based designs for the different operations configurations for normal, off-normal and accident conditions. Coordinate with the radiation protection reviewer (see Chapter 10A of this SRP for specific license applications and Chapter 10B of this SRP for CoC applications) to evaluate the acceptability of the dose rates.

Coordinate with the reviewer of Chapter 3, “Principal Design Criteria Evaluation,” of this SRP and review any additional shielding-related criteria. Refer to Chapter 11, “Operation Procedures and Systems Evaluation,” of this SRP to consider any expected operating procedures that would require being close to the storage container, such as equipment that should be monitored or serviced frequently. Also, review the evaluated dose rates at the side of the same storage container to ensure that ALARA principles are either engineered into the design or evoked by specific operating procedures in the chapter of the SAR on operating procedures.

6.5.1.2 *Design Features*

Read the general description of the DSS or DSF presented in the general description chapter of the SAR, as well as any additional information provided in the shielding evaluation chapter. Review the text descriptions as well as the drawings, figures, and tables that describe the DSS or DSF SSCs and features that are relied on for shielding, or for which dose rates should be calculated, to confirm they are sufficiently detailed to allow the staff to perform an indepth evaluation. This includes any unique features or SSCs that are not commonly associated with DSS or DSF design, such as additional, or supplemental, shielding items that are necessary to enable personnel to perform some storage operations (i.e., are necessary beyond just ALARA). Confirm that the descriptions and drawings clearly identify the geometric arrangements of DSS or DSF SSCs and features and physical dimensions. Confirm that the SAR describes the differences in the configuration of the DSS or DSF SSCs and features for normal, off-normal, and accident conditions. Ensure that the information in the SAR addresses the various stages of operations for the identified conditions for all proposed contents (i.e., including any reactor-related GTCC waste and HLW to be stored at the DSF for specific licenses). For SSCs and features for which scenarios may exist that remove or expose material relied on for shielding that otherwise remains in place or unexposed (e.g., excavation near loaded storage containers that rely on the surrounding soil for shielding), ensure that the SAR addresses the effects of the conditions during such scenarios.[2]

(SL) Assess whether the SAR adequately describes the spatial relationship between sources, shielding, and the design dose rate area(s). Consider that the design of shielding can be oriented either on the radiation sources or a point to be protected. The layout of an ISFSI or an MRS typically creates the potential for direct radiation exposure of the offsite population in all directions. As a result, shielding is typically oriented on the sources, which is the most effective positioning of shielding.

Review the SAR material composition descriptions of SSCs and features relied on for shielding. Ensure that the descriptions identify and describe all materials taken into consideration in determining shielding requirements. These include the following:

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[2] Note that design features descriptions are important for ensuring compliance with regulatory dose limits, including the limits in 10 CFR 72.104(a) and 10 CFR 72.106(b). For canister-based storage container designs, these limits apply to the loaded transfer cask as well. The limits apply regardless of the storage container's location (whether in a structure under 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities," or 10 CFR Part 52, "Licenses, Certifications, and Approvals for Nuclear Power Plants," or on the DSF storage pad). This position is consistent with the November 16, 2006, rulemaking's definition of the boundary between 10 CFR Part 50 and 10 CFR Part 72 for criticality safety (see Volume 71 of the Federal Register, page 66648 (71 FR 66648)). Also note that for CoC applications, 10 CFR 72.236(d) places responsibility for designing a DSS to meet 10 CFR 72.104 and 10 CFR 72.106 with the DSS designer (the CoC applicant) and does not contemplate that this requirement regarding the DSS design can be passed to the general licensee through its 10 CFR 72.212 evaluation or 10 CFR Part 50 programs.

- materials that have other functions but their mass also provides shielding (especially gamma shielding by structural materials, gamma and neutron shielding by concrete and pool water, and building and barrier materials for DSFs)

- materials especially selected and positioned for gamma shielding, such as lead

- materials especially selected and positioned for neutron shielding, such as water, concrete, and proprietary shielding materials

Confirm that the material specifications for nonstandard materials (e.g., proprietary neutron shield materials) include appropriate references for the material's properties that are relevant to and are included in the shielding analysis. Consult with the materials evaluation and thermal evaluation reviewers (Chapters 8 and 5, respectively, of this SRP) to identify and understand the material specifications for nonstandard materials in the design. Confirm that the technical design (or engineering) drawings include material specifications important to the performance of the shield materials such as those identified in Section 6.4.1.2 of this SRP chapter, and consider whether any of these specifications should be included in the CoC or license technical specifications.

Also consult with the materials and thermal reviewers to identify and understand the impacts of normal, off-normal, and accident conditions on the properties and behavior of the DSS or DSF SSCs and features relevant to the shielding evaluation for the different stages of operations. These properties and behavior include temperature sensitivities to elevated temperatures, which may cause reduced neutron shield efficacy from the loss of bound or free water in concrete or other hydrogenous shielding materials, as well as impacts of accumulated radiation exposure. Coordinate with the materials reviewer to obtain reasonable assurance that any degradation that may occur will not impact the safe performance of the shielding materials for the term proposed in the CoC or specific license application. Confirm that the SAR includes appropriate tests with adequate acceptance criteria to ensure that components such as lead gamma shielding and neutron shielding are fabricated correctly and perform as designed and will maintain their performance for the proposed CoC or license storage term.

As part of the DSS or DSF shielding design review, also consider the items identified in Section 6.4.1.2 above, as applicable. ANSI/ANS 6.4.2, "Specification for Radiation Shielding Materials," includes information that may be useful to consider as part of this review.

6.5.2 Radiation Source Definition

Verify that all potential radiation sources have been correctly identified and quantified, even if analysis shows that they produce negligible contributions to dose.

Burnup, cooling time, initial uranium loading, and initial enrichment are parameters that affect the total source term of SNF. Examine the description of the design-basis fuel in the chapter of the SAR on principal design criteria to verify that the applicant calculated the bounding source term. Confirm that the applicant examined all designs and burnup conditions for the SNF to be stored in the DSS or at the DSF to ensure that the bounding fuel type and parameter values are used. Devote particular attention to the combined effects of gamma and neutron source terms as a function of fuel burnup, cooling times, and enrichment. In many cases, there is no single specific enrichment-burnup-cooling time combination that bounds all potential storage container loadings (see the analysis presented in NUREG/CR-6716, "Recommendations on Fuel Parameters for Standard Technical Specifications for Spent Fuel Storage Casks," issued March 2001).

Ensure that the SAR specifies cooling times and enrichments as minimum values and burnups as maximum values. For enrichments and burnups, it is acceptable for the values to be assembly average minimum and assembly average maximum values, respectively, though calculation of the assembly average may require additional consideration for fuel with axial blankets. Natural uranium blankets effectively increase the burnup in the middle of the assembly's active fuel zone, with greater effect as the length of the blankets increases. This in turn results in higher gamma and particularly neutron sources. However, the impact is insignificant for natural uranium blankets shorter than 15 centimeters (6 inches). Variations in fuel assembly type play a secondary role for pressurized-water reactor (PWR) fuel. For boiling-water reactor (BWR) fuel, void fractions and channel sizes may affect the strengths of neutron and gamma sources. Ensure that the SAR describes the condition of the SNF contents (e.g., undamaged, damaged); this information plays a role in determining the adequacy of the analysis models' representation of the physical distribution of the radiation source (see Section 6.5.3.1 below).

Pay special attention to proposed SNF contents that include MOX or thoria. Ensure that the source terms calculated for this kind of SNF properly account for unique aspects of these fuel materials, including nuclides and nuclide quantities from fuel irradiation and from natural decay of the fuel materials. For short post-irradiation cooling times, consider whether or not the source term analysis needs to account for longer times because of possible dose rate increases with time that may result from buildup of nuclides with significant radiations (e.g., Tl-208 in thoria-bearing fuel) at times longer than the analyzed post-irradiation cooling times.

NFH can contribute significantly to SNF storage container dose rates, either locally or overall depending on irradiation history and any limits regarding allowable numbers and locations within the storage containers. Thus, for storage containers that include NFH with the SNF assemblies, ensure that the SAR properly identifies the types of NFH to be stored with the assemblies. Ensure that the SAR includes the parameters needed to determine the source terms for the NFH. These parameters include the burnup (or irradiation exposure); cooling time; component materials, masses, and cobalt impurity levels in different axial zones; the neutron flux factors in the different axial zones; and neutron source types and source strengths (if NSAs are included). Be aware that some NFH may have other materials that, when activated, also can be significant sources (e.g., hafnium, silver-indium-cadmium). Ensure that the SAR addresses these materials and the NFH containing them. Also, some NFH types may have multiple configurations that can affect material amounts in different axial zones. For example, thimble plug devices may also have water displacement or absorber rods. Ensure that the NFH descriptions in the SAR appropriately account for these variations. Ensure that the design-basis NFH source term is based on a saturation value for activation of cobalt impurities or on cobalt activation from a specified maximum burnup and minimum cool time. Consider other activation products, as appropriate, as noted previously. Review the source term from the assembly hardware (e.g., upper and lower nozzles) following applicable guidance for source terms from activated NFH.

(SL) In addition to the SNF and NFH, other radiation sources at a specific license DSF may include the following:

- solid reactor-related GTCC waste
- HLW in a form ready to be stored and other activated materials to be stored with the HLW

(SL) Verify that the SAR provides the physical and chemical form, source geometry, radionuclide content, and estimated curie value and bases for estimation for each source type (i.e., the reactor-related GTCC waste, HLW, and other radioactive material referred to above). The radionuclide inventory and quantity of that inventory in each shielded container define the gamma and neutron sources for that material. The other properties of the material will be useful in defining the distribution of the radionuclide inventory within the material and how that could change under different conditions (e.g., normal vs. accident condition configurations).

Verify that the shielding analysis in the SAR uses parameter values that bound the parameter values that define the allowable SNF and NFH contents and, for specific licenses, the reactor-related GTCC waste and HLW contents in the technical specifications. The technical specification parameters for defining the SNF allowed for storage in the DSS or at the DSF should include the combination(s) of maximum burnup, minimum enrichment, and minimum cooling time that is bounded by the parameters used to define the source terms in the shielding analysis. If the applicant proposes technical specifications that limit the SNF in other ways (e.g., by decay heat only), verify the applicant's justification for that approach and that the radiation source terms used in the shielding analysis are bounding for the variety of SNF that meets the proposed technical specification limit. Verify that the applicant's justification and analysis account for the effects of uncertainty in the methods for determining a SNF assembly meets the limits on the radiation source terms (and thus the dose rates). For example, while decay heat and radiation source terms relate to each other, the relationship is such that a significant variety of burnup, enrichment, and cooling time combinations can result in a given amount of decay heat. Further, the combinations resulting in the same decay heat can vary among types and designs of fuel assemblies. Also, for a DSS, licensees using the DSS may determine their assemblies' decay heat using a different method than the applicant used, which is a source of uncertainty for the radiation source term that should be addressed. Appropriate definition of and evaluation of the source terms for the allowable contents of a DSS or DSF is an important part of the analyses for demonstrating compliance with regulatory requirements, which for a DSS includes, as discussed elsewhere in this SRP chapter, meeting the requirements in 10 CFR 72.236(d).

6.5.2.1 *Initial Enrichment*

The specifications in the chapter of the SAR on principal design criteria should indicate the maximum fuel enrichment used in the criticality analysis. For shielding evaluations, however, the neutron source term increases considerably with lower initial enrichment for a given burnup. As described in Section 3.4.1.2, "Enrichment," of NUREG/CR-6716, as the initial enrichment decreases, the fuel is exposed to a larger neutron fluence to achieve the same burnup. The larger neutron fluence generates a larger actinide content, which results in a larger neutron source term and secondary gamma source term, as illustrated in NUREG/CR-6716, Section 3.4.1.2. Therefore, confirm that the SAR specifies the minimum initial enrichment as one of the parameter limits for the SNF contents, or justifies the use of a neutron source term, in the shielding analysis, that specifically bounds the neutron sources for fuel assemblies to be placed in the storage containers, both in total source strength and strength across the energy spectrum. Because average initial enrichments typically increase with increasing burnup within the SNF population, the latter option may be used if the applicant uses low enrichments that bound the historical enrichments for fuels at the proposed burnups. However, do not attempt to use specific source terms as the bases for establishing SNF contents limits because these are not readily inspectable parameters. The fuel assembly minimum initial enrichment, maximum burnup, and minimum cooling time are more appropriate for use as loading controls and limits.

6.5.2.2 *Computer Codes for Radiation Source Definition*

Verify that the applicant determines the source terms using a computer code, such as ORIGEN-S (e.g., as a SAS2 sequence of Oak Ridge National Laboratory's "SCALE" computer code package), that is well benchmarked and recognized and widely used by the industry. If a vendor proprietary code is used, check the code validation and verification records and procedures, preferably with sample testing problems. Although easy to use, use of ORIGEN-2 and the Department of Energy, Office of Civilian Radioactive Waste Management, Characteristics Database should be discouraged. Both have energy group structure limitations. For example, for ORIGEN-2, many libraries are not appropriate for burnups exceeding 33,000 MWd/MTU. Also, ORIGEN-2 and the database are no longer maintained by the original developer and are based on outdated data that may contain errors. If the applicant uses a computer code that is designed for reactor analyses (e.g., CASMO) for source-term calculation, ensure that the code has been used in such a way that the calculations yield appropriate results to use as source terms in the shielding analysis. This includes appropriate consideration of unique aspects of any proposed SNF contents that include MOX or thoria, as described previously in this SRP.

Ensure that the applicant has provided appropriate descriptive information, including validation and verification status, and reference documentation. Determine whether the computer code is suitable for determining the source terms and if it has been correctly used. Pay particular attention to "Area of Applicability" to verify whether the application falls into the parameter ranges for which the code is validated. Determine whether the computer code is appropriately applied and that the SAR includes verification that the chosen cross section library is appropriate for the fuel specifications being considered. Many libraries are not appropriate for a burnup exceeding 45,000 MWd/MTU because validation data are limited at high burnups.

Verify that the applicant has adequately addressed calculational error and uncertainties of the computer codes used to determine the radiological and thermal source terms for the shielding analyses. As part of this determination, consider the factors described in Section 6.4.4.1 of this SRP chapter. For example, adjustments to source term values or calculation bases or other aspects of the shielding analysis may be necessary to compensate for uncertainties in the source-term calculations for fuel with high burnups. An acceptable approach to address calculation errors and uncertainties is to establish a bounding value(s) with justified conservatisms.

When reviewing the source term calculations, also consider the factor that nuclide importance changes in high burnup fuels as a function of burnup and cooling time. The data for benchmarking the calculations and computer codes is limited at high burnups. Several NRC-sponsored studies (i.e., ORNL/TM-13315, "Validation of SCALE (SAS2H) Isotopic Predictions for BWR Spent Fuel"; ORNL/TM-13317, "An Extension of the Validation of SCALE (SAS2H) Isotopic Predictions for PWR Spent Fuel"; NUREG/CR-6700, "Nuclide Importance to Criticality Safety, Decay Heating, and Source Terms Related to Transport and Interim Storage of High-Burnup LWR Fuel," issued January 2001; NUREG/CR-6701, "Review of Technical Issues Related to Predicting Isotopic Compositions and Source Terms for High-Burnup LWR Fuel," issued January 2001; NUREG/CR-6798, "Isotopic Analysis of High-Burnup PWR Spent Fuel Samples From the Takahama-3 Reactor," issued January 2003) provide additional information on high-burnup source-term issues.

Coordinate with the thermal reviewer to determine the need to evaluate the applicant's calculation of decay heat. Often, the same codes used to determine radiation source terms can also be used to calculate decay heat. Other methods are also available for determining decay heat for SNF. Regulatory Guide 3.54, "Spent Fuel Heat Generation in an Independent Spent Fuel Storage

Installation," describes a few such methods. Verify that the SAR adequately describes the calculation method and that the method is appropriate for and correctly used to determine the decay heat for the radioactive contents to be stored in the DSS or DSF. Ensure that the analysis also appropriately identifies and accounts for uncertainties in the decay heat analysis.

6.5.2.3 *Gamma Sources*

Verify that the applicant specified gamma source terms as a function of energy for both the SNF and activated hardware (both assembly hardware and NFH), and, for DSF license applications, any reactor-related GTCC waste and HLW to be stored at the facility. If the energy group structure from the source-term calculation differs from that of the cross-section set of the shielding calculation, the applicant may need to regroup the photons. Regrouping can be accomplished by using the nuclide activities from the source term calculation as input to a simple decay computer code with a variable group structure. Some applicants will convert from one structure to another using simple interpolation. In general, only gammas with energies from approximately 0.8 to 2.5 MeV will contribute significantly to the dose rate through typical types of DSS shielding; thus, regrouping outside this range is of a lesser importance for DSSs. Consider the importance of other gamma energies to dose rates for storage containers with shielding that differs from the typical DSS shielding and, for DSFs, for shielding for other SSCs for which dose rates should be calculated. Determine whether the source terms are specified per assembly, per total assemblies, per metric ton, or, for specific licenses, on some appropriate basis for any reactor-related GTCC waste and HLW. Ensure that the total source is correctly used in the shielding evaluation.

Determining the source terms for fuel assembly hardware and NFH is generally not as straightforward as for the SNF. The source term is primarily from the cobalt contained in the hardware, particularly in the steel and Inconel components. For some NFH, activation of other components such as hafnium in hafnium absorber assemblies and the silver-indium-cadmium material in some control-rod assemblies can also produce a significant gamma source. The strength and physical distribution of the hardware source term depends upon factors such as the mass of the materials, the level of cobalt impurity in the steel and Inconel components, and the axial region of the fuel assembly (i.e., top nozzle or upper end-fitting, upper plenum, fuel, lower plenum, bottom nozzle or lower end-fitting) in which the materials are irradiated. Thus, verify that the SAR identifies the materials that comprise the assembly hardware and NFH to be stored with the assemblies.

Verify that the SAR describes the masses of the materials that are located within each assembly axial zone. Ensure that the SAR includes the masses of the assembly components for steel-clad assemblies or assemblies with steel guide and instrument tubes. For NFH, such as control rod assemblies, ensure that the SAR describes the basis for the masses of the components listed for each axial region. The activation of these items is dependent upon the operation practices of the different reactors. Many may be operated with these items positioned just above the fuel region or slightly inserted into the fuel region. Thus, only the lower ends of these items are irradiated and the activation will be based on the appropriate flux factors for the axial regions in which the items were located. Ensure that the masses listed in each axial region are consistent with the extent of insertion into the assembly described in the SAR, which should be consistent with or reasonably bounding for operations practices for those items.

Ensure that the SAR identifies the cobalt impurity level used in the source-term calculation and describes the basis for that assumption. Various analyses have used impurity levels of about 800 to 1,000 parts per million (ppm), which is bounding for steel components of assemblies and NFH manufactured since the late 1980s. Data contained in PNL-6906, "Spent Fuel Assembly

Hardware: Characterization and 10 CFR 61 Classification for Waste Disposal," show that, for at least some assembly types fabricated before that time, cobalt levels may be as high as 1500 ppm in Inconel and 2100 ppm in steel. Thus, ensure that the SAR analysis uses cobalt impurity levels that are appropriate for the fuel assemblies and NFH to be stored in the DSS or DSF storage containers, given the age of the assemblies and NFH (based on their burnups and cooling times).

The nature of the flux changes in magnitude and spectrum in regions outside of the fuel region. Thus, ensure that the SAR analysis adequately accounts for the impact of these changes on hardware irradiation in these other axial regions. This may be done by the use of scaling factors such as described in Section 3.3.2, "Hardware Regional Activation" of NUREG/CR-6802, "Recommendations for Shielding Evaluations for Transport and Storage Packages," issued May 2003. Additionally, ensure that the hardware source term includes the contributions of materials such as hafnium and silver-indium-cadmium for those NFH items that include these materials. While the SAR may describe the source from cobalt in terms of curies, the source terms for these other materials likely will be described in terms of their energy spectrum.

The impacts on dose rates from the activated assembly hardware and NFH can be significant. The effort devoted to reviewing this analysis should be based on the contribution of these source terms to the dose rates presented in the shielding evaluation. Ensure that the source term analysis addresses all appropriate NFH items that are included in the proposed DSS or DSF contents, comparing the items identified in the source term analysis with those items listed in the DSS or DSF contents descriptions in the appropriate SAR chapters.

Depending on the storage container design(s), neutron interactions may result in the production of high-energy gammas near the container surface. If this source term is not treated by the shielding analysis computer code, verify that it is determined and its contribution to dose rates is addressed by other appropriate means.

Support the confinement review, as needed, by verifying the quantities of certain nuclides (e.g., krypton-85, tritium, and iodine-129) the applicant used to analyze doses from the release of radioactive material during design-basis conditions (i.e., normal, off-normal, and accident conditions). Confer with the confinement reviewer to determine the need to verify these nuclide quantities.

6.5.2.4 *Neutron Sources*

Verify that the SAR expresses the neutron source term as a function of energy. The SNF neutron source will generally result from both spontaneous fission and alpha-n reactions in the fuel. Depending on the method used to calculate these source terms, the applicant may need to define the energy group structure separately. This is often accomplished by selecting the nuclide with the largest contribution to spontaneous fission (e.g., curium-244) and using that spectrum for all neutrons, since the contribution from alpha-neutron reactions is generally small. For SNF with cooling times less than 5 years, confirm that the analysis addresses the spectra of curium-242 and californium-252.

The specification of a minimum initial enrichment may be a necessary basis for defining the allowed contents. Verify that the assumed minimum enrichments bound all assemblies the application proposes for storage. Specific limits are needed for inclusion in the CoC or license, as applicable. Lower-enriched fuel, irradiated to the same burnup as higher-enriched fuel, produces a higher neutron source. Therefore, verify that the SAR chapter on technical specifications and operational controls and limits specifies the minimum initial enrichment as an operating control and limit. Alternatively, ensure that the applicant specifically justified the use of a neutron source term, in the shielding analysis, that bounds the neutron sources for the SNF assemblies to be stored. An applicant may demonstrate that the assumed enrichment(s) bounds the proposed fuel population except for possible outliers in the SNF population. This is acceptable if the SAR specifically requires verification of the minimum enrichment with the values in the final SAR, and if there are specific dose rate limits in the technical specifications. The applicant and the NRC staff should not attempt to establish specific source terms as the operating controls and limits for SNF storage container (e.g., DSS) use.

Ensure that the SAR adequately describes the neutron source, both source strength and spectrum, for NSAs included in the NFH to be stored with the spent fuel assemblies. NSAs are divided into two main categories: primary and secondary sources. Primary sources include polonium-beryllium (PoBe), americium-beryllium (AmBe), and other sources that generate neutrons though α-n reactions or spontaneous fission. Some of these sources have significantly long half-lives and can contribute a neutron source equivalent to the source of a spent fuel assembly. It is these sources that can contribute significantly to the neutron source term in the SNF storage container and so should be included in the shielding evaluation. Secondary sources include antimony-beryllium (SbBe) and others that generate neutrons through γ-n reactions. These sources typically have very short half-lives and need to be “charged” through neutron activation of the heavier element in the source material. Thus, secondary neutron sources usually contribute negligibly to the neutron source term in the SNF storage container.

Ensure that the SAR adequately addresses contributions to the neutron source from subcritical multiplication since this contribution is not included in the results of depletion codes like SCALE’s TRITON and SAS2H or CASMO. This source can often be addressed through the use of proper options in the input to the shielding code or use of appropriate factors by which the neutron source is increased when input into the shielding code. The applicant may use such factors when the shielding model properties are such that the model would be critical or near critical (e.g., a flooded SNF container with the SNF modeled as 5-weight-percent enriched fresh fuel). Ensure that the applicant justifies the appropriateness of the selected factor(s).

(SL) The reactor-related GTCC waste and HLW to be stored at the DSF may also include neutron sources, depending on the specification of the wastes. Thus, follow the preceding guidance, as appropriate, when evaluating the neutron source terms for these wastes, considering the criteria given in Section 6.4.2.2 of this SRP chapter.

6.5.2.5 *Other Parameters Affecting the Source Term*

Ensure that the SAR contains specific information concerning reactor operations that affect the SNF source term. Several NRC technical reports (specifically, NUREG/CR-6716, but also NUREG/CR-6700, NUREG/CR-6701, and NUREG/CR-6798) discuss the potential effects of other parameters not typically included in technical specifications (e.g., moderator soluble boron concentrations, maximum poison loading, minimum moderator density (for BWR fuels), and maximum specific power). For example, the net impact of moderator density on DSS dose rates is expected to be low for PWR fuels. However, be aware that the axial variation in moderator density in BWR cores can have a measurable effect on the axial dose rate profile of a BWR spent fuel assembly. The dose rate may increase near the top of the assemblies where the moderator density was the lowest. This is particularly important for neutron sources because reduced moderator density will harden neutron spectrum and hence induce more actinide production.

6.5.3 Shielding Model Specification

Verify that the applicant adequately described the models that were used in the shielding evaluation for storage under normal, off-normal, and accident conditions. For example, if a DSS transfer cask has an external neutron shield, the SAR should reflect whether the cask would be damaged by a tipover accident or by a tornado missile or it would be degraded in a fire. Ensure that the applicant has assumed that liquid, polyesters, or other resin neutron shields are not present after an accident, unless justification is made that they remain intact. Confirm with the structural (SRP Chapter 4), thermal (SRP Chapter 5), and other reviewers, as appropriate, that the treatment of the DSS or DSF features and SSCs in the shielding analysis is consistent with or bounding for the expected operation configurations and the impacts of normal, off-normal, and accident conditions for those operation configurations. Coordinate with these reviewers to ensure the applicant has analyzed the impacts of all appropriate normal, off-normal and accident conditions, including any conditions that may be unique to the DSS or DSF design and operations. Examples include analyses of accident events with excavation adjacent to DSSs that rely on the soil for shielding and dropping onto DSS SSCs (e.g., the transfer cask) of separate shielding devices (also to be considered as part of the DSS design) that are needed to allow personnel to perform some of the DSS operations. Confirm that the shielding assumptions made in dose rate calculations, for both occupational workers and the public, are consistent with the design criteria and design drawings for the DSS or DSF. Ensure that, for DSF license applications, the analysis models address all facility SSCs and features that affect shielding. ANSI/ANS 6.4, “Guidelines on the Nuclear Analysis and Design of Concrete Radiation Shielding for Nuclear Power Plants,” includes information that may be useful to consider as part of the review of the model specifications (this SRP section) and the analysis (Section 6.5.4) for concrete DSS or DSF SSCs.

6.5.3.1 *Configuration of Shielding and Source*

Examine the sketches or figures and descriptions that indicate how the DSS or DSF SSCs and features important, or credited, for shielding are modeled. Ensure that the sketches or figures clearly indicate the geometric arrangement(s) and physical dimensions of the DSS or DSF SSCs and features. Verify that the models are consistent with the DSS or DSF design, including dimensions and materials that are consistent with those specified in the DSS or DSF drawings presented in the general information evaluation chapter of the SAR. Verify that the SAR accounts for voids, streaming paths, and irregular geometries or otherwise treats them in a conservative manner. Verify that the models address the configurations of DSS or DSF features and SSCs during the different operations stages, including any conditions that, though temporary, may affect how different conditions impact the DSS or DSF shielding features (e.g., excavation adjacent to DSSs that rely on soil for shielding) and dose rates for these different conditions. In addition, verify that the applicant clearly stated the differences, if any, between normal, off-normal, and accident conditions.

Verify that the applicant properly modeled the source term locations. For SNF storage containers, this involves properly locating the SNF source within the envelope of the assembly’s fuel zone and locating any assembly hardware and NFH sources within the proper assembly zones where this hardware and NFH source may be present. Also, verify that the applicant properly modeled the physical distribution and the material properties of the sources. In many cases, the fuel assembly materials may be homogenized within the fuel region to facilitate the shielding calculations. Watch for cases when homogenization may not be appropriate. For example, homogenization should not be used in neutron dose calculations when significant neutron multiplication can result from moderated neutrons (i.e., when significant amounts of moderating materials are present such as when the SNF container is flooded). In some, particularly early, applications, fuel and basket material homogenization may have been used; however, with improved analytical capabilities, this practice should be discouraged. If homogenization is used, ensure it is not used for configurations where significant streaming could occur between basket components or significant neutron multiplication is expected. Confirm also that the models account for any possible shifts in the position of the contents for different design-basis conditions.

If the applicant has requested storage of damaged fuel assemblies, ensure that the SAR adequately describes the proposed damage assemblies. If the fuel assemblies are damaged to the extent that reconfiguration of the fuel into a geometry different from intact fuel assemblies has occurred (e.g., fuel debris) or can occur, ensure that the SAR provides appropriate materials, geometry, and other necessary parameter specifications to calculate dose rates for normal, off-normal, and accident conditions.

SNF typically has a cosine-shaped burnup profile along its axial length. If axial peaking appears to be significant, verify that the applicant has appropriately accounted for the condition. Typically, fuel gamma source terms vary proportionally with axial burnup. Fuel neutron source terms vary exponentially by a power of 4.12 with burnup (NUREG/CR-6802), which can be applied to the axial variation in burnup. In addition, the structural support regions (e.g., top and bottom end hardware and plenum regions) of the assembly should be correctly positioned relative to the SNF. The materials in these regions may be individually homogenized. Guidance regarding homogenization in the fuel region applies to the assembly hardware regions. Generally, at least three source regions (i.e., fuel and top and bottom assembly hardware) are necessary. Some storage containers may also employ fuel spacers to maintain the axial position of the SNF inside the container.

Verify that the SAR shows or adequately describes the detector locations selected for the various dose rate calculations. Ensure that these locations are representative of all locations relevant to radiation protection issues, including for site personnel and members of the public. Pay particular attention to dose rates from streaming paths to which occupational workers would be exposed (e.g., at vent and drain port covers, lid bolts, air vents). Ensure that the applicant has noted shielding end points (such as lead in the storage container wall in relation to the assembly hardware and use of fuel spacers to center the fuel). Sections 6.4.4.3 and 6.5.4.3 of this chapter provide additional information regarding the selection of detector locations for dose rate calculations for both CoC and specific license analyses.

(SL) Ensure that the models address the design-basis conditions (i.e., normal, off-normal, and accident conditions) for the different stages of storage operations and all the proposed types of contents (any reactor-related GTCC waste and HLW, as well as SNF) in an appropriate or bounding manner. This includes storage container array configurations and maximum quantities of stored materials on the facility's storage pad(s), contents locations and orientations within containers, and cautions against homogenization of contents with container internals where that is not appropriate. This also includes other DSF SSCs, as appropriate and necessary, in addition to the storage containers.

6.5.3.2 *Material Properties*

Review the descriptions of the materials and their compositions and densities that are used in the models. Verify that the material compositions and densities are consistent with the description of design features and SSCs and the contents given in the SAR as they are geometrically represented in the models. This includes materials for SSCs and features that have other functions but that also provide shielding as well as SSCs and features that are specifically for shielding. Ensure that the materials properties in the models are consistent with or bounding for the effects on the materials properties of the different design-basis conditions (normal, off-normal, accidents). These effects include any degradation from aging, high temperature, accumulated radiation exposure, and manufacturing tolerances. Many shielding computer codes allow the densities to be input directly in grams per cubic centimeter. If densities are input into the models in atoms per barn-centimeter, pay particular attention to the conversion.

6.5.4 Shielding Analyses

6.5.4.1 *Computer Codes*

Evaluate the computer codes or programs used for the shielding analysis. There are several recognized computer codes that are widely used for shielding analysis. These include computer codes that use Monte Carlo, deterministic transport, and point-kernel techniques. The point-kernel technique is generally appropriate only for gammas since storage containers, including DSSs, and SSCs used in operations typically do not contain sufficient hydrogenous material to apply removal cross sections for neutrons.

It is important to assess whether the number of dimensions of the computer code being applied for the shielding analysis is appropriate for the dose rates being calculated. Typically, the NRC does not accept the use of one-dimensional codes for calculations other than shielding designs with simple cylindrical geometries. At the least, a two-dimensional calculation is generally necessary. One-dimensional computer codes provide little information about off-axis locations and streaming paths that may be significant to determining occupational exposure. Even a two-dimensional calculation may not be adequate for determining any streaming paths if the modeled configuration is not properly established. These considerations in applying a particular computer code also apply to the computation of dose rates at the axial ends of storage containers. In some cases, the applicant will use the flux output from a deep-penetration shielding code as input to a large distance, skyshine code. Verify that the use and interface of these codes are appropriate and done correctly.

Ensure that the codes used in the analysis have the capability to account for the effects of radiation interactions that impact dose rates. Also ensure that the applicant has used the codes to appropriately account for the impacts of those interactions. This includes gammas produced by (n, γ) reactions in the DSS or DSF SSCs and contents and subcritical multiplication. For example, for models that may be critical or near critical (e.g., assuming 5 weight percent fresh fuel for the SNF contents in a flooded storage container), use of code features to track fission may not be appropriate to account for subcritical multiplication effects on dose rates. This is because the calculation may not properly converge or finish, except due to any time limits set in the input file. In such a case, the results would not adequately represent the dose rates for the analyzed DSS or DSF SSCs and contents. For this scenario, subcritical multiplication should be addressed in another manner.

The Electric Power Research Institute (EPRI) has published a valuable primer on shielding computer codes and analysis techniques (Broadhead 1995). Computer codes that have commonly been used in CoC and specific license applications include MCNP and SCALE. Codes that have been used or may be useful include the following (grouped by code type):

- Monte Carlo codes: MORSE, MONACO/MAVRIC, MCBEND, SCALE, MCNP
- Discrete Ordinates codes: DORT, ANISN, DANTSYS, DOORS 3.2
- Point Kernel codes: QAD-CGGP, RANKERN

- Others: SKYSHINE-II, STREAMING

The NRC recognizes that there are other codes available that may also be useful for DSS or DSF shielding analyses. These codes come from a variety of sources, including government organizations and commercial vendors. Note that the previous use of these codes (in approved CoC, license, or amendment applications) does not constitute generic NRC approval of these codes.

Regardless of the code(s) used in the SAR analysis, confirm that the applicant has justified the applicability and appropriateness of the particular code(s) for the SAR analysis. The extent of the justification may vary, with codes that are well established, have a broad user base, and have capabilities to handle complex problems needing less justification than a proprietary code or a code that is limited in its capabilities. Confirm that the applicant used a computer code version that is demonstrated to be adequate for the analysis and is valid for the particular computational platform used to perform the analysis. Computer codes are periodically updated to be compatible with the latest operating system, correct errors found in previous versions, or incorporate updated methods. Therefore, consider whether additional confirmatory assessments and review are needed to validate the shielding predictions by an applicant that uses older or unsupported codes or code versions. This consideration should include a recognition that the applicant may use these codes later as the CoC holder or licensee to evaluate changes to the DSS or DSF design or operations under 10 CFR 72.48, “Changes, tests, and experiments,” and the associated implications.

Verify that the SAR describes each of the numerical models of the computer codes used in the shielding evaluation. For each computer code used, ensure that an approved, validated, and verified version of the computer code is being applied by verifying that the SAR provides the following information:

- author, source, and dated version

- description of the numerical model applied in the computer code and the extent and limitation of its application

- either (1) the evaluation of computer code solutions to a series of test problems, demonstrating substantial similarity to solutions obtained from hand calculations, analytical results published in the literature, acceptable experimental tests, a similar computer code, or benchmark problems; or (2) the specification of publicly available references for commonly used and well-established codes (e.g., SCALE and MCNP) that demonstrate validation

Examine the solution comparisons provided in the SAR and determine whether satisfactory agreement of computer and test solutions (or resolution of deviations) is evident. Ideally (though not a requirement), the applicant should have validated the computer code used for evaluation of shielded storage containers with actual dose rate measurements from similar or prototypical SNF or, for specific license applications, GTCC waste or HLW storage containers.

Be aware that applicants often use transport or point-kernel methods to calculate neutron and gamma response functions (unit of (mrem/hr)/(source particle/s/cm^2)). This technique, also known as the response function method, enables an applicant to quickly determine dose rates for different source terms by simply multiplying the source terms by the response functions instead of running a separate transport calculation for each source term. It is based on the premise that, all else being equal (e.g., source particle type, energy, origin; detector location; material and geometric properties of the system), an increase in the source strength results in a corresponding increase in dose rates. For analyses that employ this response function technique, verify the following:

- The applicant calculated a response function for each particle type and for each energy bin in the particle type's energy spectrum.
- The response functions are used only for the shielding and source configuration (geometric and material properties) for which the response functions were calculated.
- The source properties (material and geometric) are appropriate or conservative for the contents for which the functions were calculated.
- The response functions are used only for the detector location for which the functions were calculated.
- The calculations for determining the response functions are well converged and appropriately account for any errors and uncertainties resulting from calculation or use of the response functions.

Thus, multiple sets of response functions may be needed to support the shielding analysis. This includes separate sets of response functions for differences in shielding properties (material or geometric), for differences in source properties (material or geometric), and for different detector locations. Ensure that the applicant has determined a sufficient number of sets of response functions to analyze dose rates for the different stages of operations for the design-basis conditions (i.e., normal, off-normal, and accident conditions) at the locations necessary to evaluate personnel and public doses as discussed in Sections 6.4.4.3 and 6.5.4.3 of this chapter.

6.5.4.2 *Flux-to-Dose-Rate Conversion*

Review the flux-to-dose-rate conversions used in the applicant's shielding analysis and confirm that they are acceptable for the purposes for which the dose rates are used, including demonstration of compliance with regulatory dose limits, estimating occupational doses during operations, and serving as the basis for any dose rate limits in the CoC or license technical specifications, as applicable. The computer code used in the analysis may have data libraries for different conversions and options to perform these conversions automatically or require (or have an option) that conversion factors be manually included in the input file. Whichever option is used, confirm that the SAR clearly identifies the conversion factors used to determine dose rates.

While there are different conversion factors available for use, the NRC has only accepted the use of the ANSI/ANS 6.1.1-1977 conversion factors. The basis for this acceptance is explained below. Thus, unless adequately justified, confirm that the applicant used these conversion factors in its analysis. The justification should include close correspondence with the accepted conversion factors and appropriateness for the application (e.g., conversion factors are based on the same methodology as is incorporated into the limit, or usefulness for demonstration of compliance by measurement).

The requirements in 10 CFR Part 72 include two sets of dose limits to individual members of the public located at or beyond the controlled area boundary, annual dose limits for normal operations and anticipated occurrences in 10 CFR 72.104(a), and accident dose limits in 10 CFR 72.106(b).

The limits in 10 CFR 72.106(b) incorporate the methodology of 10 CFR Part 20, which incorporates the methodology from the International Commission on Radiological Protection (ICRP)-26, “Recommendations of the International Commission on Radiological Protection,” and dose calculation methods of ICRP-30, “Limits for Intakes of Radionuclides by Workers.” The limits in 10 CFR 72.104(a) are based on the methodology from ICRP-2, “Report of Committee II on Permissible Dose for Internal Radiation,” to maintain compatibility with the Environmental Protection Agency’s regulation in 40 CFR 191.03(a), which is applicable to 10 CFR Part 72 storage operations (see 63 FR 54559; October 13, 1998).

The ICRP issued a series of ICRP-30 reports that provide the means to derive doses under the dosimetry concept of ICRP-26. The dose calculation methods in the revised 10 CFR Part 20, and relevant for the 10 CFR 72.106(b) limits, do not quantify doses in terms of doses to the whole body and individual, critical organs like is done under the ICRP-2 methodology. Instead, the dose is quantified as a risk-equivalent dose that considers the relative risks of different tissues, expressed as organ or tissue weighting factors (tabulated in 10 CFR 20.1003, “Definitions”). In this manner, doses absorbed by the whole body and individual organs or tissues can be summed into a single quantity relating to risk. This method negates the need to keep track of two sets of doses, one for the whole body and another for a series of organs, as is done under the ICRP-2 methodology.

The conversion factors in the 1977 revision of ANSI/ANS 6.1.1 are derived from methodologies that are consistent with the ICRP-2 and so are appropriate for determining compliance with the limits in 10 CFR 72.104(a). For 10 CFR 72.106(b) limits, though from a different methodology, the conversion factors from the 1977 revision of the standard result in conservative dose rates versus factors derived from the methodology incorporated into 10 CFR 72.106(b) and so are acceptable for evaluating compliance with that requirement.

The 1977 ANSI/ANS 6.1.1 conversion factors are also accepted because they result in dose rates (given as dose-equivalent) that can be readily compared against dose rates measured with appropriate monitoring equipment and techniques for converting instrument readings into meaningful results. The methodology in ICRP-26 introduced dosimetry units of effective dose-equivalent, which is not a measurable quantity, at least without the aid of more sophisticated measurement techniques. Thus, dose rates determined with the 1977 ANSI/ANS 6.1.1 conversion factors are appropriate to use as a basis for dose rate limits in the CoC and license technical specifications, compliance with which is determined by measurement.

While a later revision of ANSI/ANS 6.1.1 (the 1991 revision) was issued, the conversion factors in that revision are based on determination of effective dose-equivalent. Thus, their applicability and usefulness for demonstrating compliance with 10 CFR 72.104(a) limits and for developing dose rate limits in technical specifications carries the concerns of the dosimetry bases identified above. Furthermore, while the 1991 conversion factors were intended to replace the 1977 factors, there were some issues. One basic issue is that in 1985, a recommendation was made in ICRP-45, “Quantitative Bases for Developing a Unified Index of Harm,” to double the neutron quality factors. The 1991 conversion factors, which account for body shielding, have the effect of reducing predicted neutron dose rates by about a factor of two. Had the ICRP-45 recommendation been implemented, dose rates calculated with the 1991 conversion factors and the new quality factors would have been comparable to the dose rates calculated with the current quality factors and the 1977 conversion factors (though, because the calculated dose quantities are different, a direct comparison does not have much meaning). However, the ICRP-45 recommendation was never adopted, given that the standard was later withdrawn. So, calculating dose rates with the 1991 conversion factors would result in predicted neutron dose rates that are reduced by a factor of two. If at some later time the ICRP-45 recommendation were adopted, that could mean issues with compliance with regulatory dose limits and any dose rate limits in CoC or license technical specifications. Thus, there is no regulatory advantage to use the 1991 revision of the standard, and the NRC staff has determined that it should not be used in analyses to demonstrate compliance with regulatory limits or to establish technical specifications dose rate limits.

6.5.4.3 *Dose Rates*

On the basis of experience, comparison to similar systems, or scoping calculations, make an initial assessment of whether the dose rates appear reasonable and whether their variation with location is consistent with the geometry and characteristics of the DSS or DSF contents and design features for the different configurations that exist at different operations stages for the different design-basis conditions. The models used for these calculations should be consistent with the expected condition of the DSS or DSF SSCs and features for the design-basis conditions (normal, off-normal, accident). The following guidance pertains to the selection of points at which the dose rates should be calculated.

For normal and off-normal conditions, ensure that the applicant indicated the dose rate at all locations accessible to occupational personnel during storage container loading, transfer to the DSF storage pad, and maintenance and surveillance operations. Generally, these locations include points at or near various DSS or DSF components and in the immediate vicinity of the storage container and distances from the storage container that are reasonable for the types of activities, including surveillance and maintenance, to be performed during operations, considering the likely locations of personnel involved in the system operations and activities. Examples of locations include inlet and outlet vent areas, trunnion areas, maximum dose rate locations for an SNF storage container's side and top surfaces, the canister-to-transfer cask or overpack (as applicable) gap region, top (including maximum dose rate spot) and upper radial surfaces of the canister, and the bottom of a DSS's transfer cask. Additional examples include locations of changes in shielding such as radial surface locations above and below the axial extent of radial neutron shielding and openings in the transfer cask lid as well as areas on the lid. For rectangular-shaped SSCs such as storage modules and overpacks, ensure that the locations include maximum dose rate spots on each side and on the top. Verify that the applicant calculated the dose rates at a distance of 1 meter (3.28 feet) from these locations because they typically contribute to occupational exposures.

Dose rate analyses should address potential configuration changes of the contents (e.g., reconfiguration of damaged fuel within a damaged-fuel can), if applicable, to support demonstration that the container or fuel (or both) meets the dose limits of normal, off-normal, and accident conditions of storage. The shielding analysis should assume a worst-case or bounding configuration of the contents (e.g., the canned fuel).

Verify that the dose rate estimates have appropriately considered the following:

- conservatism of simplifying assumptions and assertions that non-conservative assumptions are more than compensated for by conservative assumptions

- streaming path dose rates that include failure to offset penetrations in SSCs such as storage container lids for venting, draining, drying

- analyzed configurations consistent with or bounding for anticipated or expected configurations (e.g., water levels in canisters during welding of canister lid or canister decontamination)

- potential negative effects of radiation scattering in DSS or DSF SSCs that increase dose rates in accessible areas near the storage container

- local "hot spots" from gaps or significantly reduced shielding around the source, considering all solid angles

(CoC) Regulations in 10 CFR 72.236(d) require that the application for a DSS design demonstrate that the shielding and confinement features of the DSS are sufficient to meet the requirements in 10 CFR 72.104 and 10 CFR 72.106. Compliance with this part is evaluated as part of the radiation protection review (see Chapter 10B of this SRP).

(CoC) Ensure that the applicant calculated dose rates at appropriate and sufficient distances from the DSS. For 10 CFR 72.104 evaluations, this includes calculations for a single DSS and a sample array(s) of DSSs on a storage pad. The DSS array is typically a 2 x 10 DSS arrangement or some other array that is representative of how the system will or may be used at a DSF. For canister-based systems, ensure the calculations include the transfer cask for 10 CFR 72.106 analyses. Calculations with the transfer cask for 10 CFR 72.104 analyses may also be needed depending on the transfer cask characteristics and operations descriptions. Examples of when such calculations should be provided and evaluated for transfer casks include when dose rates at 100 meters (328 feet) from the cask indicate that transfer cask operations could result in nonnegligible, or possibly significant, doses at that distance for the estimated duration of normal operations or for an anticipated occurrence of reasonably expected time duration (e.g., crane malfunction during cask movement and associated recovery actions).[3] Coordinate with the radiation protection reviewer to determine if these calculations are needed. For both 10 CFR 72.104 and 10 CFR 72.106 analyses, ensure that the applicant calculated the dose rates at distances starting at 100 meters from the DSS and the DSS array. For 10 CFR 72.106 analyses, calculations at 100 meters have typically been sufficient to support demonstration of compliance with the regulatory limit. For 10 CFR 72.104 analyses, dose rates are typically needed at multiple distances, beginning at 100 meters.

(CoC) It is important to note that a general licensee is permitted to use distance or additional engineering features such as berms, or both, to mitigate doses to real individuals near the site. If such features are used in the DSS SAR dose rate calculations, verify that they are included in the descriptions of the DSS and their use is included in the CoC as a condition of DSS use. In addition, verify that the SAR determines the degree to which the normal condition dose rates could change for the identified off-normal conditions.

(SL) In addition to the dose rate location and estimate considerations listed above, ensure that the dose rate locations and estimates include surfaces and appropriate distances from all relevant DSF SSCs involved in the handling, transfer, or storage of radioactive materials to be stored at the site. Also ensure that the dose rate locations and estimates include other relevant site locations where facility personnel and other individuals (e.g., shippers bringing material on site) may be located, and which are needed for the radiation protection evaluation (SRP Chapter 10A) of occupational and public doses. This includes evaluation of situations such as a work station

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3 See Footnote 2 on page 6-18.

that is shielded from multiple sources of radiation. For such situations, check the solid angles about that station for potential gaps or other sources of elevated dose rates.

(SL) Consult with the radiation protection reviewer (SRP Chapter 10A) who will use the dose rate estimates (in addition to other information) to determine whether appropriately detailed SAR calculations (dose rates and collective dose estimates) show that the radiation shielding features are sufficient to meet the requirements in 10 CFR 72.104, 10 CFR 72.106, and ALARA objectives. As noted in Section 6.4.4.3 of this SRP, any supplemental shielding or feature (e.g., berms) included in the calculations to demonstrate compliance with the regulatory dose limits should be classified as important to safety.

6.5.4.4 *Confirmatory Analyses*

Perform confirmatory calculations, as necessary, to verify the results of the applicant's shielding analysis. Independently evaluate the dose rates in the vicinity of the DSS or DSF SSCs and features for normal, off-normal, and accident conditions for the different configurations at the different operations stages. In determining the level of effort appropriate for these calculations, consider the following factors:

- the degree of sophistication in the SAR analysis
- a comparison of SAR dose rates with those of similar DSS or DSF SSCs that have previously been reviewed, if applicable
- the amount of conservatism applied in the applicant's analysis
- the typical variation in dose rates expected between different computer codes and cross-section sets
- the fact that actual dose rates will be monitored and practices employed by the licensee to limit or minimize doses in accordance with the requirements in 10 CFR Part 20
- the restrictions to be placed on the DSS or DSF operations or the limits to be placed on dose rates, as documented in the CoC or license, including any technical specifications
- the applicant's experience in using the methods and computer codes in previous submittals
- the use of computational methods or computer codes that are new or that have been used in previously reviewed CoC or specific license applications
- the inclusion in the design of any significant departures from previous DSS or DSF SSC and feature designs (e.g., unusual shield geometry, new types of materials, or different source terms) or operations

Coordinate with the radiation protection reviewer in determining the need for, and level of effort to expend in, performing confirmatory calculations. At a minimum, examine the representative input files submitted in, or with, the SAR. Verify that the data for the DSS or DSF design features and contents are properly entered into the code, including proper dimensions, material properties, gamma and neutron source terms, and distributions of the sources. Verify that the applicant uses a cross section library that is appropriate for the shielding analysis, including the use of any coupled cross sections in instances where the code is used to evaluate secondary sources through modeling of the radiation interactions in the DSS or DSF shielding materials. Ensure that the applicant correctly uses appropriate code options and features to enable accurate calculations, including for secondary source contributions and neutrons from subcritical multiplication.

If a more detailed review is required (e.g., the applicant used a new shielding computer code not used in a previously approved CoC or license application, the design is unusual, dose rates are significantly high vs. other reviewed DSSs or DSFs), independently confirm the dose rates to ensure that the SAR results are reasonable and conservative. As previously noted, the use of a simple computer code for neutron calculations often does not provide results with sufficient accuracy and confidence. An extensive and more detailed evaluation may be necessary if large uncertainties are suspected. To the degree possible, the use of a different shielding computer code with a different analytical technique and cross-section set from that used in the SAR analysis will usually provide a more independent evaluation.

EPRI has published a good reference (Broadhead 1995) regarding the treatment of uncertainty in thick-shielded cask analyses.

Coordinate with the thermal and confinement reviews to determine the need to independently confirm the estimated source terms (i.e., decay heat and radionuclide quantities) and their uncertainties for these reviews. The items can be calculated with the codes used to calculate radiation source terms. Refer to the literature regarding these codes for information about the calculation uncertainties. For example, for SCALE, this information is included in various Oak Ridge National Laboratory technical reports and NUREG/CRs (e.g., ORNL/TM-13315, ORNL/TM-13317, and NUREG/CR-5625, “Technical Support for a Proposed Decay Heat Guide Using SAS2H/ORIGEN-S Data”).

(SL) In addition to the preceding guidance, consider the following in determining the appropriate level of effort:

- margin of safety in the SAR analyses
- use of the results in developing projected doses
- magnitude of estimated doses (occupational and for members of the public) under normal, off-normal, and accident conditions, as applicable, considering all radiation sources

6.5.5 Consideration of Reactor-Related GTCC Waste Storage (SL)

(SL) Review the applicant’s approach to addressing storage of solid reactor-related GTCC waste at the DSF, considering the requirements described in Section 6.4.5 of this SRP. Confirm that the applicant clearly describes the analysis approach for the reactor-related GTCC waste and the basis for that approach. Evaluate the acceptability of the approach, considering the contents, SSCs, and design features (including the storage containers), and operations descriptions. Ensure the descriptions in the SAR are adequate for the reactor-related GTCC waste, appropriately applying the guidance in the preceding review sections. For evaluating approaches that use dose rates from SNF or HLW storage to bound and represent reactor-related GTCC waste dose rates, compare the descriptions for reactor-related GTCC waste with the information for the SNF or HLW and confirm that the information supports the adequacy of the approach.

Confirm that the SAR analysis addresses all operations configurations (e.g., loading, container closure, storage at the storage pad) and all design-basis conditions (normal, off-normal, accident). Ensure that the analysis provides dose rate information that can be used in the radiation protection evaluations (see SRP Chapter 10A) to demonstrate facility design and operations meet, or will meet, the requirements in 10 CFR 72.104 and 10 CFR 72.106 and the requirements in 10 CFR Part 20, and that the storage of reactor-related GTCC waste will not have an adverse effect on the safe storage of SNF and HLW.

6.5.6 Supplementary Information

Review supplemental information, which can include copies of applicable references (especially if a reference is not generally available to the reviewer), computer code descriptions, input and output files, and any other information that the applicant deems necessary. Request any additional information needed to complete the review process.

6.6 Evaluation Findings

The NRC reviewer should prepare evaluation findings upon satisfaction of the applicable regulatory requirements in Section 6.4 of this SRP. If the documentation submitted with the application fully supports positive findings for each of the regulatory requirements, the statements of findings should be similar to the following, as separately provided for CoCs and specific licenses:

Certificate of Compliance **(CoC)**

F6.1 The SAR provides specifications of the spent fuel contents to be stored in the [DSS designation] in sufficient detail that adequately defines the allowed contents and allows evaluation of the DSS shielding design for the proposed contents. The SAR includes analyses that are adequately bounding for the radiation source terms associated with the proposed contents' specifications. (10 CFR 72.236(a))

F6.2 The SAR describes the structures, systems, and components (SSCs) important to safety that are relied on for shielding in sufficient detail to allow evaluation of their effectiveness for the proposed term of storage. [The reviewer should cite specific drawings that are used to define the SSCs relied on for shielding.] (10 CFR 72.236(b) and 10 CFR 72.236(g))

F6.3 The SAR provides reasonable and appropriate information and analyses, including dose rates, to allow evaluation of the [DSS designation's] compliance with 10 CFR 72.236(d). This evaluation is described in the radiation protection review (SRP Chapter 10B).

F6.4 The SAR provides reasonable and appropriate information and analyses, including dose rates, to allow evaluation of consideration of ALARA in the [DSS designation's] design and evaluation of occupational doses. This evaluation is described in the radiation protection review (SRP Chapter 10B).

The reviewer should provide a summary statement similar to the following:

> In summary, the staff has reasonable assurance that the design features relied on for shielding for the [DSS designation] have been adequately identified and evaluated. The evaluation includes appropriate shielding analyses for the configurations that exist during the different stages of storage operations, including the impacts of normal, off-normal, and accident conditions. The evaluation includes dose rate results that are adequate to support evaluation of the [DSS designation]'s compliance with the radiation protection requirements in 10 CFR 72.236(d), the occupational doses estimated to result from storage operations using the [DSS designation], and the adequate consideration and incorporation of ALARA principles into the [DSS designation] design and operations. The staff reached this finding on the basis of a review that considered the regulation itself, appropriate regulatory guides, applicable codes and standards, accepted engineering practices, the statements and representations in the SAR, and the staff's confirmatory analyses.

<u>Specific License **(SL)**</u>

F6.5 The SAR provides specifications of the radioactive materials to be stored at the proposed DSF in sufficient detail that adequately defines the allowed materials and allows evaluation of the DSF shielding design for the proposed materials. The SAR includes analyses that are adequately bounding for the radiation source terms associated with the proposed materials' specifications. (10 CFR 72.24(c), 10 CFR 72.120(b) and 10 CFR 72.120(c))

F.6.6 The SAR describes the DSF structures, systems, and components (SSCs), including those that are important to safety that are relied on for shielding, in sufficient detail to allow evaluation of their effectiveness for the proposed license term. [The reviewer should cite specific drawings that are used to define the SSCs relied on for shielding.] The descriptions include design criteria and design bases for the design, fabrication, construction, and performance requirements of SSCs important to safety. (10 CFR 72.24(b) and 10 CFR 72.24(c), 10 CFR 72.120(a–c))

F6.7 The DSF design includes SSCs and features to shield personnel from radiation exposure to meet 10 CFR 72.126(a)(6) and for radiation protection under normal and accident conditions to meet 10 CFR 72.128(a)(2). Evaluation of the suitability of the shielding to perform these functions is described in the radiation protection review (SRP Chapter 10A).

F6.8 The SAR provides reasonable and appropriate information, including dose rates, to allow evaluation of the DSF's compliance with 10 CFR 72.24(e). This evaluation is described in the radiation protection review (SRP Chapter 10A).

F6.9 The SAR provides reasonable and appropriate information, including dose rates, to enable performance of the evaluations required in 10 CFR 72.24(m) and to allow evaluation of the DSF's ability to meet the radiation protection requirements for members of the public in

10 CFR 72.104. 10 CFR 72.106 and 10 CFR Part 20. This information includes impacts to shielding and dose rates to support evaluations of compliance with the requirements in 10 CFR 72.122(b)(2)(i), 10 CFR 72.122(c), and 10 CFR 72.122(e) as well. These evaluations are described in the radiation protection review (SRP Chapter 10A).

The reviewer should provide a summary statement similar to the following:

> In summary, the staff has reasonable assurance that the design features relied on for shielding for the DSF have been adequately identified and evaluated. The evaluation includes appropriate shielding analyses for the configurations of DSF SSCs and features that exist during the different stages of storage operations, including the impacts of normal, off-normal, and accident conditions. The evaluation includes dose rate results that are adequate to support evaluation of the DSF's ability to meet the radiation protection requirements in 10 CFR 72.104, 10 CFR 72.106, and 10 CFR Part 20, including doses to members of the public and occupational doses estimated to result from DSF operations, and the adequate consideration and incorporation of ALARA principles into the DSF design and operations. The staff reached this finding on the basis of a review that considered the regulation itself, appropriate regulatory guides, applicable codes and standards, accepted engineering practices, the statements and representations in the SAR, and the staff's confirmatory analyses.

6.7 References

10 CFR Part 20, "Standards for Protection Against Radiation."

10 CFR Part 71, "Packaging and Transportation of Radioactive Material."

10 CFR Part 72, "Licensing Requirements for the Independent Storage of Spent Nuclear Fuel, High-Level Radioactive Waste, and Reactor-Related Greater than Class C Waste."

40 CFR Part 191, "Environmental Radiation Protection Standards for Management and Disposal of Spent Nuclear Fuel, High-Level and Transuranic Radioactive Wastes." Subpart A, Environmental Standards for Management and Storage.

American National Standards Institute (ANSI)/American Nuclear Society (ANS) 6.1.1, "Neutron and Gamma-Ray Flux-to-Dose-Rate Factors," 1977.

ANSI/ANS 6.1.1, "Neutron and Gamma-Ray Fluence-to-Dose Factors," 1991.

ANSI/ANS 6.4, "Guidelines on the Nuclear Analysis and Design of Concrete Radiation Shielding for Nuclear Power Plants," 2006.

ANSI/ANS 6.4.2, "Specification for Radiation Shielding Materials," 2006.

Broadhead, B.L. "Evaluation of Shielding Analysis Methods in Spent Fuel Cask Environments," EPRI TR-104329, Electric Power Research Institute, Palo Alto, CA, June 1995.

DeHart, M.D. and O.W. Hermann, "An Extension of the Validation of SCALE (SAS2H) Isotopic Predictions for PWR Spent Fuel," ORNL/TM-13317, Oak Ridge National Laboratory, September 1996.

Hermann, O.W. and M.D. DeHart, “Validation of SCALE (SAS2H) Isotopic Predictions for BWR Spent Fuel,” ORNL/TM-13315, Oak Ridge National Laboratory, September 1998.

International Commission on Radiological Protection (ICRP) Publication 2, “Report of Committee II on Permissible Dose for Internal Radiation,” 1959.

ICRP Publication 26, “Recommendations of the International Commission on Radiological Protection,” *Annals of the ICRP*, Vol. 1, Issue 3, 1977.

ICRP Publication 30 (Part 1), “Limits for Intakes of Radionuclides by Workers,” *Annals of the ICRP*, Vol 2, Issue 3-4, 1979. (Part 2) Vol 4, Issue 3-4, 1980. (Part 3) Vol 6, Issue 2-3, 1981. (Part 4), Vol 19, Issue 4, 1988. Supplement to Part 1 Vol 3, Issue 1-4, 1979. Supplement to Part 2, Vol 5, Issue 1-6, 1981. Supplement A to Part 3 Vol 7, Issue 1-3, 1982. Supp B to Part 3, Vol 8, Issue 1-3, 1982.

ICRP Publication 45, “Quantitative Bases for Developing a Unified Index of Harm,” *Annals of the ICRP*, Vol. 15, Issue 3, 1985.

Luksic, A. “Spent Fuel Assembly Hardware: Characterization and 10 CFR 61 Classification for Waste Disposal,” PNL-6906, Pacific Northwest Laboratory, June 1989.

NUREG/CR-5625, “Technical Support for a Proposed Decay Heat Guide Using SAS2H/ORIGEN-S Data,” ORNL-6698, Oak Ridge National Laboratory, July 1994.

NUREG/CR-6700, “Nuclide Importance to Criticality Safety, Decay Heating, and Source Terms Related to Transport and Interim Storage of High-Burnup LWR Fuel,” ORNL/TM-2000/284, Oak Ridge National Laboratory, January 2001.

NUREG/CR-6701, “Review of Technical Issues Related to Predicting Isotopic Compositions and Source Terms for High-Burnup LWR Fuel,” ORNL/TM-2000/277, Oak Ridge National Laboratory, January 2001.

NUREG/CR-6716, “Recommendations on Fuel Parameters for Standard Technical Specifications for Spent Fuel Storage Casks,” ORNL/TM-2000/385, Oak Ridge National Laboratory, March 2001.

NUREG/CR-6798, “Isotopic Analysis of High-Burnup PWR Spent Fuel Samples from the Takahama-3 Reactor,” ORNL/TM-2001/259, Oak Ridge National Laboratory, January 2003.

NUREG/CR-6802, “Recommendations for Shielding Evaluations for Transport and Storage Packages,” ORNL/TM-2002/31, Oak Ridge National Laboratory, May 2003.

Regulatory Guide 3.54, “Spent Fuel Heat Generation in an Independent Spent Fuel Storage Installation.”

U.S. Nuclear Regulatory Commission, “Minor Revision of Design Basis Accident Dose Limits for Independent Spent Fuel Storage and Monitored Retrievable Storage Installations,” *Federal Register*, Vol. 63, No. 197, October 13, 1998, pp. 54559–54562.

NRC, “Criticality Control of Fuel Within Dry Storage Casks or Transportation Packages in a Spent Fuel Pool,” *Federal Register*, Vol. 71, No. 221, November 16, 2006, pp 66648–66657.

7 CRITICALITY EVALUATION

7.1 Review Objective

The objective of the U.S. Nuclear Regulatory Commission's (NRC's) review with regard to nuclear criticality safety is to ensure that spent nuclear fuel (SNF) to be placed into the dry storage under 10 CFR Part 72, "Licensing Requirements for the Independent Storage of Spent Nuclear Fuel, High-Level Radioactive Waste, and Reactor-Related Greater than Class C Waste," remains subcritical under normal, off-normal, and accident conditions involving handling, packaging, transfer, and storage. This objective extends to the storage of high-level radioactive waste (HLW) at a specific license dry storage facility (DSF) that is a monitored retrievable storage installation (MRS). If reactor-related greater-than-Class-C (GTCC) waste is to be stored at a specific license DSF also storing SNF or HLW, then the review objective also includes ensuring that the storage of reactor-related GTCC waste does not adversely affect the safe storage of SNF and HLW and ensuring reactor-related GTCC waste remains subcritical if it includes fissile material. The objective also extends to other DSF structures, systems, and components (SSCs) in the specific license application for which criticality safety may be relevant (e.g., pools for storage or repackaging included as part of the DSF design).

7.2 Applicability

This chapter of the Standard Review Plan (SRP) applies to the review of applications for specific licenses for an independent spent fuel storage installation (ISFSI) or an MRS, categorized as a DSF. It also applies to the review of applications for a certificate of compliance (CoC) of a dry storage system (DSS). Sections, paragraphs, or tables that apply only to DSF specific license applications have "**(SL)**" in the heading. Sections, paragraphs, or tables that apply only to DSS CoC applications have "**(CoC)**" in the heading. A subsection or paragraph without an identifier applies to both types of applications.

7.3 Areas of Review

This chapter addresses the following areas of review:

- criticality design criteria and features
- fuel specification
    - fuel type
    - nonfuel hardware (NFH)
    - fuel condition
- model specification
    - configuration
    - material properties
- criticality analysis
    - computer codes and cross section data
    - neutron multiplication factor
    - benchmark comparisons
- burnup credit
    - limits for the licensing basis
    - licensing-basis model assumptions
    - code validation—isotopic depletion

- code validation—k_{eff} determination
- loading curve and burnup verification

- reactor-related GTCC waste and HLW (**SL**)
- supplemental information

7.4 Regulatory Requirements and Acceptance Criteria

This section summarizes those parts of 10 CFR Part 72 that are relevant to the review areas addressed by this chapter. The NRC staff reviewer should refer to the exact language in the regulations. Tables 7-1a and 7-1b match the relevant regulatory requirements to the areas of review this chapter covers. The reviewer should verify the association of regulatory requirements with the areas of review presented in the tables to ensure that no requirements are overlooked as a result of unique design features.

Table 7-1a Relationship of Regulations and Areas of Review for a DSF (SL)

Areas of Review	10 CFR Part 72 Regulations			
	72.24	72.40(a)(13)	72.44(c)	72.124
Criticality Design Criteria and Features	(b)(c)(g)	●	●	(a)(b)(c)
Fuel Specifications	(b)(c)(g)		●	(a)(b)
Model Specification	(d)			(a)(b)
Criticality Analysis	(d)	●		(a)(b)
Burnup Credit	(b)(c)(d)(g)		●	(a)(b)
Reactor-Related GTCC Waste and HLW	(b)(c)(g)		●	(a)

Table 7-1b Relationship of Regulations and Areas of Review for a DSS (CoC)

Areas of Review	10 CFR Part 72 Regulations		
	72.124	72.236(a)	72.236
Criticality Design Criteria and Features	(a)(b)(c)	●	(b)(c)(g)(h)(m)
Fuel Specification	(a)(b)	●	(b)(c)
Model Specification	(a)(b)	●	(b)(c)
Criticality Analysis	(a)(b)	●	(b)(c)
Burnup Credit	(a)(b)	●	(b)(c)(g)

The DSS or DSF SSCs must be designed to ensure the SNF remains subcritical under all credible conditions (10 CFR 72.124(a)). In general, the criticality evaluation seeks to ensure that a subcritical condition is maintained for the DSS or DSF design and operations by fulfilling the following acceptance criteria:

- The effective neutron multiplication factor, k_{eff}, including all biases and uncertainties at a 95-percent confidence level, should not exceed 0.95 under all credible normal, off-normal, and accident conditions for all storage operations (e.g., SNF handling, packaging, transfer, and storage).

- At least two unlikely, independent, and concurrent or sequential changes to the conditions essential to nuclear criticality safety, under normal, off-normal, and accident conditions, would need to occur before an accidental criticality is possible (i.e., double contingency principle; see 10 CFR 72.124(a)).

- When practicable, criticality safety of the design should be established on the basis of favorable geometry, permanently fixed neutron-absorbing materials (poisons), or both. Where solid neutron-absorbing materials are used, the design should provide for a positive means to verify their continued efficacy during the storage period. The continued efficacy of the neutron-absorbing materials in the DSS or DSF storage containers may be confirmed by a demonstration or analysis before use, showing that significant degradation of these materials cannot occur over the life of the DSS or DSF (i.e., the certified or licensed period of storage). In other DSF SSCs, such as a pool, the neutron absorbers may be more likely to corrode; however, they will be more accessible. Thus, appropriate periodic monitoring should be used to verify these absorbers' continued efficacy.

- Criticality safety design may credit up to 90 percent of the neutron poison material in fixed neutron absorbers when subject to adequate acceptance testing (see Chapter 8, "Materials Evaluation," Section 8.5.7, "Criticality Control Materials," of this SRP).

- **(SL)** The DSF SSCs must be designed to ensure that reactor-related GTCC waste and HLW to be stored at the DSF and containing fissile material also remain subcritical under all credible conditions. The preceding criteria for SNF should be met, as applicable.

7.5 Review Procedures

Figures 7-1a and 7-1b show the interrelationship between the criticality evaluation and the other areas of review described in this SRP for specific license and CoC applications, respectively.

To ensure that the DSS or DSF complies with 10 CFR Part 72, examine the criticality design features and criteria in the chapters of the applicant's safety analysis report (SAR) on general information and principal design criteria, in addition to the chapter on criticality evaluation, for any additional details concerning criticality design features and criteria. Assess the bounding specifications for the SNF and assure consistency with the models the applicant used in the criticality analyses. Verify that the applicant has addressed criticality safety considerations under normal, off-normal, and accident conditions. In addition, verify that the criticality calculations determine the highest k_{eff} that might occur for all loading states under normal, off-normal, and accident conditions involving handling, packaging, transfer, and storage. To the extent practicable, use independent methods to perform any k_{eff} calculations to evaluate the applicant's design. Review the operations descriptions to ensure the operations are consistent with and address the assumptions and parameters relied on in the criticality safety analyses, including any CoC or license conditions or technical specifications related to criticality safety.

(SL) The review guidance focuses mainly on the storage containers (i.e., the DSS for CoCs and the container design(s) to be used at the DSF for a specific license). However, for specific license applications, recognize that there may be other DSF SSCs for which criticality safety may be a concern and should be evaluated. Such SSCs would include any pool facilities used as part of operations for a specific license DSF (e.g., for loading, unloading, and repackaging of SNF) and included as part of the DSF design in the specific license application. Therefore, apply the guidance in this chapter to the evaluation of these other DSF SSCs as applicable and appropriate. The review guidance does address specific, unique considerations for these other DSF SSCs where necessary.

For evaluations that involve the use of industry standards, ensure the standards, including the revisions to the standards, are used in a manner consistent with the NRC's positions regarding those standards and their revisions. For example, in addition to items from specific industry standards addressed in this chapter, the NRC has documented its endorsements, including any exceptions, of various standards in Regulatory Guide 3.71, "Nuclear Criticality Safety Standards for Fuels and Materials Facilities."

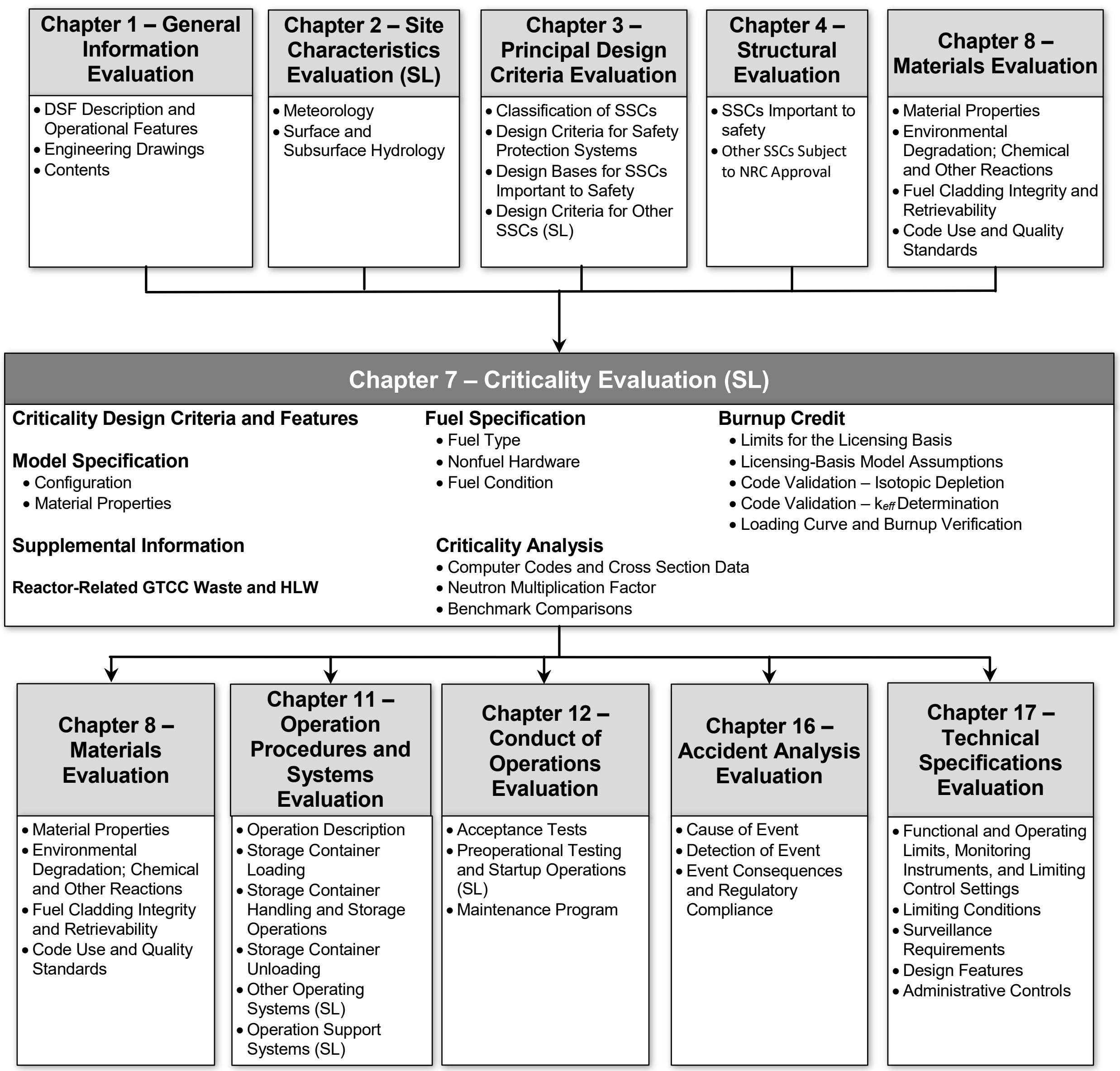

Figure 7-1a Overview of Criticality Evaluation of Specific License Applications for a DSF (SL)

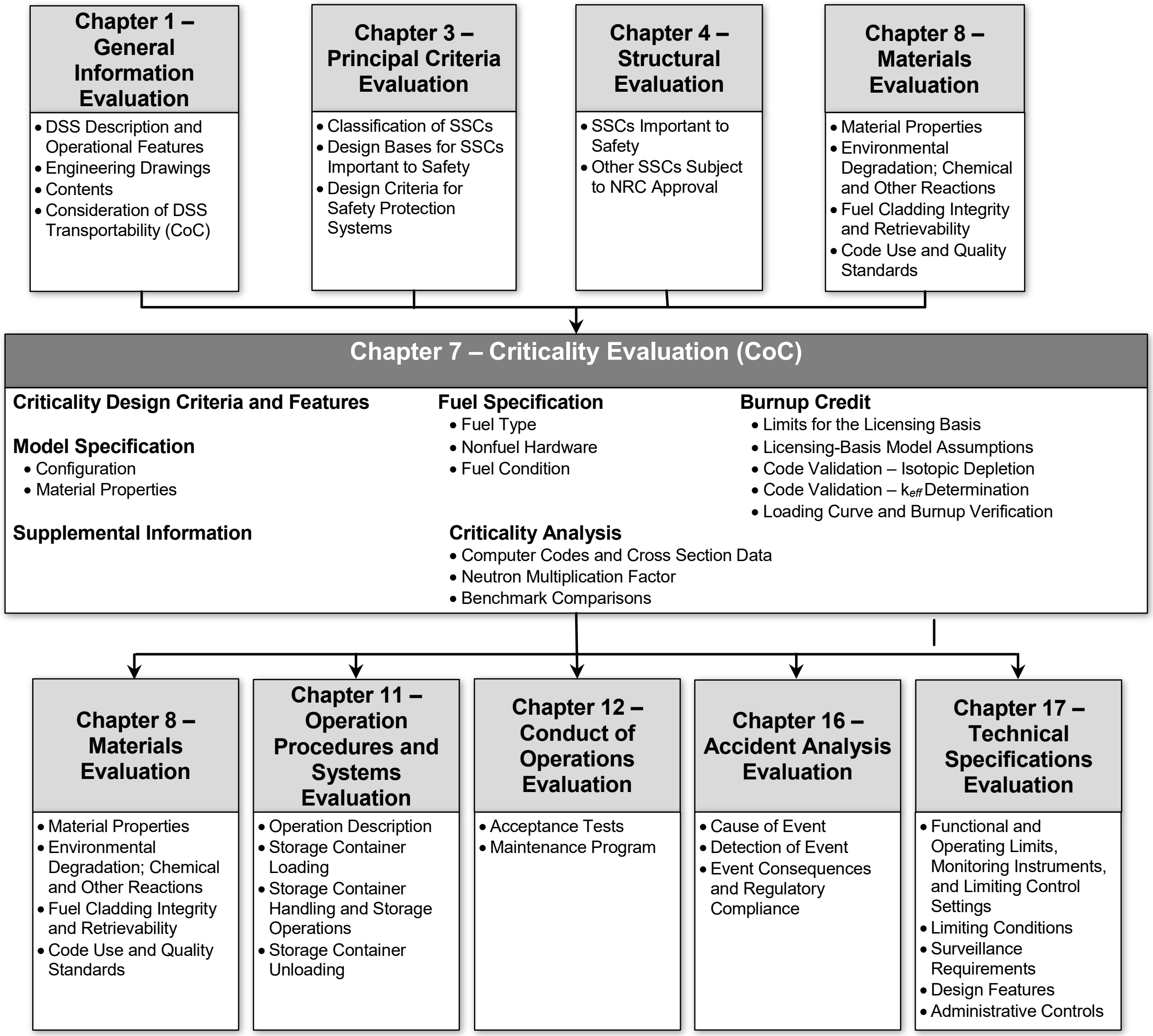

Figure 7-1b Overview of Criticality Evaluation of Applications for a DSS (CoC)

7.5.1 Criticality Design Criteria and Features

Examine the principal criticality design criteria presented in the chapter of the SAR on principal design criteria as well as any related details provided in the SAR chapter on criticality evaluation. Examine the general storage container description presented in the SAR for any relevant information. Verify that the information in the chapter of the SAR on criticality evaluation is consistent with the information in the SAR's chapters on general information and principal design criteria. Verify that all descriptions, drawings, figures, and tables are sufficiently detailed to support an indepth staff evaluation.

Criticality safety of the design must be based on favorable geometry, permanent fixed neutron absorbing materials, or both (10 CFR 72.124(b)). The criticality design of the storage container relies on the general dimensions of the container's components and the spacing of the fuel assemblies. Tolerances for the material, fabrication, and assembly of SSCs can be important in identifying worst-case (lowest margin of safety) geometries, material compositions, and densities. Ensure that the SAR uses the tolerances for the properties and construction of all SSCs involved in criticality analyses. Also ensure that the tolerances used in the analyses are identical to or conservatively bounding for the tolerances shown in the definition of the storage container design. Verify that the analyses are based on the most conservative combination of tolerances.

The criticality design often relies on neutron poisons. These may be in the form of fixed poisons in the storage container's SNF basket structure, soluble poisons in the water of the SNF pool, or both. For fixed neutron-absorbing materials, the NRC has accepted a requirement for acceptance testing of the material during fabrication as a means for verifying the continued efficacy of solid neutron-absorbing materials incorporated in the SNF storage container (see also Section 8.5.7 of this SRP). During loading and unloading operations, the NRC staff accepts the use of borated water as a means of criticality control if the applicant specifies a minimum boron content and strict controls are established to ensure that the minimum required boron concentration is maintained. This condition in turn becomes an operating control and limit in the SAR and in the CoC or license technical specifications. Include a discussion of these operation controls in the safety evaluation report (SER). Ensure that the technical specifications also include other design features significant to the criticality design, such as important basket dimensions that control the spacing of the fuel assemblies. These dimensions may be a minimum pitch for the basket cells or a minimum flux trap width.

If borated water is used for criticality control during loading and unloading operations, verify that the design and operations descriptions in the SAR include administrative controls or design features (with appropriate controls and design features included in the technical specifications), or both, to ensure that accidental flooding with unborated water is not credible. Otherwise, consider accidental flooding with unborated water. If the storage container is also intended for transport, the storage container design should not rely on borated water for criticality control. Borated water and any other liquids are not acceptable as a means of criticality control for a storage container in its dry storage configuration. This includes use of any credit in the criticality analysis for the presence of a liquid that may provide neutron shielding (and is external to the fuel basket); however, its presence and most reactive density should be assumed if it increases k_{eff}. Also, if more than one certified or licensed basket design of the same supplier could fit in the storage container, then the type of basket to be used with the container should be stamped in a location on the container that allows for easy identification of the basket. Thus, the licensee will be able to easily verify the appropriateness of the fuel contents to be loaded in the basket.

The DSS or DSF SSCs must be designed so that at least two unlikely, independent, and concurrent or sequential changes to the conditions essential to criticality safety, under normal, off-normal, and accident conditions, must occur before an accidental criticality is possible ("Double Contingency," as stated in American National Standards Institute (ANSI)/American Nuclear Society (ANS) 8.1, "Nuclear Criticality Safety in Operations with Fissionable Materials Outside Reactors"; see 10 CFR 72.124(a)). For analysis, "accidental criticality" is defined as exceeding k_{eff} of 0.95. Ensure that the criticality analysis demonstrates that k_{eff} is less than 0.95, with a 95 percent probability at the 95 percent confidence level, accounting for analysis uncertainty, bias, and bias uncertainty. Ensure that the applicant demonstrates that the double contingency criteria have been met for all configurations of the relevant DSS or DSF SSCs. For DSS or DSF storage container designs, these criteria are typically met by demonstrating a low likelihood of storage container failure and a low likelihood of flooding of the storage container to sufficient depth to cause criticality (i.e., to the height of the active fuel) in the container's dry storage configuration.

Other considerations and methods would be necessary to demonstrate that the double contingency criteria are met for other configurations of the storage container (e.g., during loading and unloading).

Under 10 CFR 72.124(c), a criticality monitoring system must be maintained in each area where special nuclear material is handled, used, or stored that will energize clearly audible alarm signals if accidental criticality occurs. This requirement does not apply while the special nuclear material is handled under water, including in a submerged storage container. The requirement also does not apply to dry storage areas where the storage container is in its dry storage configuration (i.e., drained, dried, and sealed closed). It is applicable when the storage container is removed from the pool during loading, until it is drained, dried, and sealed, and during unloading, beginning when the container's confinement barrier is no longer sealed (e.g., removal of vent or drain port covers).

Ensure that the criticality chapters of the SAR address how the criticality monitoring criteria will be met. The NRC has accepted the use of area radiation monitors, typically included in 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities," and 10 CFR Part 52, "Licenses, Certifications, and Approvals for Nuclear Power Plants," reactor facilities' spent fuel pool buildings, for meeting this requirement, provided the applicant can adequately justify that the area radiation monitors are sufficient to perform this function (in lieu of having a criticality monitoring system).

(SL) Ensure that the design of other relevant DSF SSCs, such as a pool that is part of the DSF design, are based on favorable geometry, permanently fixed absorber materials, or both. For fixed absorbers, work with the materials reviewer (SRP Chapter 8) to ensure the application provides appropriate means for verifying continued efficacy of the absorbers (e.g., periodic monitoring). If the pool also uses borated water, ensure that the facility design and operations include appropriate means to monitor and maintain the required boron concentrations for normal, off-normal, and accident conditions. Analyses of loss of soluble boron (boron dilution) may also be necessary. Ensure that the SAR demonstrates that the double contingency criteria are met for the pool and other relevant DSF SSCs under normal, off-normal, and accident conditions. Ensure appropriate controls and design features for the pool and other relevant SSCs are included in the license technical specifications. The preceding guidance should also be applied, consistent with the regulations, to all nonfuel materials to be stored at the DSF that include fissile material.

7.5.2 Fuel Specification

7.5.2.1 *Fuel Type*

Examine the specifications for the ranges or types of SNF that will be stored in the DSS or DSF storage containers as presented in the SAR chapters on general information and principal design criteria, as well as any related information in the SAR chapter on criticality. Verify that the SNF specifications given in the SAR chapter on criticality are consistent with, or bound, the specifications given in the SAR chapters on general information and principal design criteria and in the technical specifications. Keeping in mind that some specifications are more important than others, identify the specifications that are key to criticality safety, and verify that these are appropriately captured in the technical specifications. NUREG-1745, "Standard Format and Content for Technical Specifications for 10 CFR Part 72 Cask Certificates of Compliance," lists some of the fuel specifications that may be key to maintaining the system subcritical, although others may be required. While NUREG-1745 discusses an option for controlling some parameters outside of the technical specifications (i.e., in the SAR only) and obtaining NRC approval for contents alternatives regarding those parameters, the NRC has since determined that this option is not acceptable. Thus, any fuel parameters that are important to ensuring criticality safety should be captured in the license or CoC technical specifications.

Of primary interest is the type of fuel assemblies and maximum fuel enrichment that should be specified and used in the criticality calculations. Boiling-water reactors (BWRs) typically use multiple fuel pin enrichments, in which case the criticality calculations should use the maximum fuel pin enrichment present. Depending upon the fuel design, an applicant may propose use of assembly-averaged or lattice-averaged enrichments. This may be acceptable if the applicant can demonstrate that the applicant's averaging technique is technically defensible and, for the criticality calculation, produces realistic or conservative results. Because of the natural uranium blankets present in many fuel designs, use of an assembly-averaged enrichment that includes the blankets is not normally considered appropriate or conservative.

Another parameter of interest is the fuel density assumed in the analysis. Ensure that the value of the fuel density used in the calculations is justified to be realistic or conservative.

Note that, while the majority of fuel assemblies burned in commercial reactors in the United States use uranium dioxide (UO_2) as the fuel material, some fuel assemblies are made with mixed-oxide (MOX) fuel material. For MOX fuel, the material specification is typically given in terms of weight percent plutonium, and is further described by isotopic limits for the major plutonium isotopes important to criticality safety (i.e., plutonium-238, plutonium-239, plutonium-240, plutonium-241, and plutonium-242) along with the amount and maximum enrichment of the uranium in the fuel. Plutonium-239 and plutonium-241 are fissile, and should have a maximum quantity or concentration limit, while the other plutonium isotopes are neutron absorbers, and should have minimum required mass ratios or concentrations. Alternatively, the applicant may choose to conservatively assume that all plutonium is fissile plutonium-239, which is acceptable.

Some commercial fuel assemblies may include thorium fuel rods in addition to UO_2 rods. Ensure that the material specification for the thorium fuel rods includes the weight percent of the rods that is thorium oxide and uranium oxide and the maximum uranium enrichment. Note that thorium-232 will absorb neutrons to become uranium-233, a fissile nuclide. Therefore, fuel assemblies with thorium fuel material may become more reactive with irradiation in the reactor. Ensure that the SAR includes a depletion analysis for such fuel in order to determine the most reactive fuel composition.

Although the burnup of the fuel affects its reactivity, many criticality analyses have assumed the storage container to be loaded with fresh fuel (the fresh fuel assumption). Alternatively, the NRC staff has provided guidance for burnup credit for intact pressurized-water reactor (PWR) fuel. This guidance is limited to burnup credit available from specific actinide and fission product compositions associated with UO_2 fuel of 5.0 weight percent or less enrichment that has been irradiated in a PWR to an assembly-average burnup value not exceeding 60 gigawatt days per metric ton of uranium (GWd/MTU) and cooled out of the reactor for a time period between 1 and 40 years. Section 7.5.5 of this SRP chapter provides guidance for the review of a criticality analysis that involves burnup credit. Ensure that the SAR chapter on technical specifications and operational controls and limits evaluation includes specifications for the fuel that will be stored in the storage container, including those important for burnup credit (e.g., minimum burnup versus enrichment, moderator temperature, in-core soluble boron concentrations, operations under control rod banks or with control rod insertion), if applicable. Also ensure that the SER contains this same information. Ensure that the license or CoC technical specifications explicitly list those specifications determined to be key to criticality safety.

For analyses that use the fresh fuel assumption, inadvertent loading of the storage container with unirradiated or low burnup fuel is not a major concern. However, inadvertent loading of the storage container with unirradiated or low burnup fuel is a major concern for container designs that rely on criticality analyses that use burnup credit. Therefore, ensure that the detailed loading procedures for these storage containers include steps to prevent misloading of unirradiated or low burnup fuel. Regardless of which analysis is used, ensure that detailed loading procedures include steps to prevent misloading for cases when fuel exceeding the design basis for the DSS or DSF storage containers, for a DSF being licensed to store SNF from a co-located 10 CFR Part 50 or 10 CFR Part 52 reactor facility, is present in the 10 CFR Part 50 or 10 CFR Part 52 facility's pool at the time of DSS or DSF storage container loading.

(CoC) Because DSSs typically are designed to store many types and configurations of fuel assemblies, verify that the applicant has demonstrated that criticality requirements are satisfied for the most reactive case. A determination of which fuel is bounding in a criticality analysis depends on many factors and usually requires examination of several types of fuel assemblies and compositions. Note that the most reactive assembly type may be different for fresh fuel analyses in fresh water versus borated water, and if burnup of the fuel is credited according to the recommendations of Section 7.5.5 of this chapter. Therefore, verify that the applicant has demonstrated that the design-basis fuel assembly is the most reactive for the specific DSS design, including requested level of burnup credit, if applicable. Ensure that the SAR chapter on general information clearly indicates the design-basis assemblies. Also ensure that the SER contains this same information.

(SL) For specific license applications that include storage of multiple types and configurations of fuel assemblies, the considerations described above for CoC applications would also apply. However, for a specific license DSF that is co-located with a 10 CFR Part 50 or 10 CFR Part 52 reactor facility, the SNF assembly types and configurations are likely to be limited to those associated with the co-located reactor facility, which may have used only one or two fuel types with limited enrichment ranges.

7.5.2.2 *Nonfuel Hardware*

Some fuel assemblies may also have nonfuel components that are positioned or operated within the envelope of the fuel assembly during reactor operation that an applicant may seek to store with the assemblies in the SNF storage container. These items include PWR control assemblies, such as rod cluster control assemblies, control element assemblies, burnable poison rod (BPR) assemblies, and axial power shaping rods. Applicants may also seek approval for storing fuel assemblies with other items that extend into an assembly's active fuel region, such as stainless steel rod inserts used to displace water in PWR assembly guide tube dashpots. For applications that propose to load assemblies containing NFH, ensure that the analysis considers the effects of both inclusion and neglect of NFH on system reactivity. If the application relies on the presence of the NFH to meet the subcritical criterion, verify that the NFH will remain in place under all normal, off-normal, and accident conditions.

Generally, the NRC staff does not allow reliance on, or credit for, fuel-related burnable neutron absorbers. This restriction includes residual neutron-absorbing material remaining in the NFH loaded with an assembly. However, credit for any negative reactivity for this latter absorbing material may be accepted if all of the following is true:

1. The remaining absorbing material content is established through physical measurement or by calculation where a sufficient margin of safety is included commensurate with the uncertainty in the method of measurement or calculation.

2. The axial distribution of the poison depletion is adequately determined with appropriate margin for uncertainties.

3. Adequate structural integrity and placement of the nonfuel hardware under accident conditions is demonstrated.

Ensure that the fuel specifications described in the SAR chapter on technical specifications and operation controls and limits include the important details about the NFH to be stored with the fuel assemblies and the associated residual neutron-absorbing material. Also ensure that the SER contains this same information. Ensure that those details key to criticality safety are included in the CoC or license technical specifications, as appropriate. Also, verify that operating procedures are established that ensure that NFH loaded with assemblies meets the approved specifications and will remain in position under normal, off-normal, and accident conditions.

7.5.2.3 *Fuel Condition*

Determine whether the applicant has included any specifications regarding the fuel condition. To date, a number of applications have requested approval for storage of fuel that is damaged as well as intact or undamaged. Consult Section 8.5.15.1, “Spent Fuel Classification,” of this SRP for the most current staff guidance for detailed descriptions of what constitutes damaged, undamaged, and intact fuel. This guidance gives the applicant the latitude to define fuel with defects (such as missing rods but not loose rods or debris) as undamaged fuel as long as the fuel can meet all the fuel-specific or system-related functions. For purposes of criticality safety, undamaged fuel is fuel that (1) is in the form of an assembly; (2) has structural and material properties such that the assembly can withstand normal, off-normal, and accident conditions while maintaining its geometric configuration; and (3) has had any damaged or missing fuel rods replaced with solid dummy rods that displace an equal or greater amount of water as the original rods. Fuel that cannot meet these criteria is considered to be damaged. However, a fuel assembly with missing fuel rods may be considered undamaged fuel if analyses are performed that show the criterion for subcriticality will be met with the fuel rods missing.

A fuel assembly that is classified as damaged should be placed in a damaged fuel canister, or in an acceptable alternative, for loading into the DSS or DSF storage container. For a storage container that is also intended for transport, keep in mind that the more severe conditions of transport may require reanalysis of assemblies classified as undamaged under storage-only conditions before transport. Confirm that specifications concerning the condition of the fuel to be stored in the DSS or DSF storage container and the loading of damaged fuel, as applicable, are included in the chapter of the SAR on technical specifications and operation controls and limits and in the CoC or license (in the technical specifications). Also ensure that the SER contains this same information.

Verify that the criticality analysis addresses the conditions of the fuel to be stored in the storage container. Ensure that the analyses for storage containers designed to store damaged fuel bound the configuration of the damaged fuel assemblies under all credible normal, off-normal, and accident conditions. For example, some analyses have performed calculations that model the damaged fuel as arrays of bare fuel rods (i.e., the cladding is assumed to be completely removed) having an optimized rod pitch.

7.5.3 Model Specification

Verify that the applicant has specified manufacturing and fabrication tolerances. Verify that the applicant used the most reactive combination of tolerances, within the ranges of their acceptable values, in the analysis models.

7.5.3.1 *Configuration*

Verify that SAR adequately describes the criticality models used to evaluate normal, off-normal, and accident conditions. Coordinate with the structural, materials, and thermal reviewers to understand any damage that could result from accident conditions, which include natural phenomena events.

Examine the sketches or figures of the models used for criticality calculations. Verify that the dimensions and materials of the models are consistent with the engineering drawings. Ensure that the SAR identifies any differences between the actual DSS or DSF storage container configurations and the models and demonstrates that the models are conservative. Substitution of end sections and support structures of the fuel with ordinary water, or a combination of water and steel, is a common and usually conservative practice in criticality analysis. However, substitution with borated water is typically not conservative. Ensure that the applicant justified any such substitutions.

Confirm that the applicant defined tolerances for poison material dimensions and concentrations and used the most reactive conditions in the criticality analysis. In addition, ensure that the SAR identifies all important design conditions and then addresses these conditions for potential variations during normal, off-normal, and accident conditions.

Verify that the applicant has considered deviations from nominal design configurations. The evaluation of k_{eff} should not be limited to a model in which all of the fuel bundles are neatly centered in each basket compartment, with the center line of the basket coincident with the center line of the storage container. For example, a storage container with steel confinement and lead shielding may have a higher k_{eff} when the basket and fuel assemblies are positioned as close as possible to the lead. However, in some designs, the most reactive configuration may be when all fuel assemblies are shifted toward the center of the basket.

In addition to a fully flooded storage container, confirm that the SAR addresses configurations in which the container is filled with partial-density water or is partially filled with water (borated, if applicable) and the remainder of the container is filled with steam consisting of ordinary water at partial density. These configurations are considered to be possible during loading and unloading operations. Confirm that the SAR also considers the possibility of preferential or uneven flooding within the storage container, if such a scenario is credible for the container design (e.g., because of blockage in small flow or drain paths). In particular, watch for situations where there is water in the fuel regions but not in the flux traps, if applicable. Storage container designs for which this type of flooding is credible are generally unacceptable. Confirm that the SAR also considers flooding in the fuel rod pellet-to-clad gap regions with unborated water. Additionally, for damaged fuel stored in a damaged fuel canister, the tops and bottoms of the damaged fuel canister will typically have screens to allow water drainage during loading. Note that the screened damaged fuel canisters will drain slower than the rest of the storage container, resulting in the potential for preferential flooding in the damaged fuel canisters. This moderation condition could potentially be more reactive than a fully flooded condition. Verify that the applicant has evaluated this condition, if damaged fuel canisters are to be used in the DSS or DSF design and operations. Above all, ensure that the analysis demonstrates that the storage container remains subcritical for all credible conditions of moderation (10 CFR 72.124(a)).

(SL) For other relevant DSF SSCs (e.g., a pool), in addition to the preceding considerations, also ensure that the applicant has identified and addressed any unique aspects of these SSCs that may also impact criticality safety. These aspects may include addressing boron dilution in the DSF's pool.

7.5.3.2 *Material Properties*

Verify that the SAR provides compositions and densities for all materials used in the calculational models. Ensure that these compositions and densities are consistent with and account for the impacts of normal, off-normal, and accident conditions. Confirm that the SAR, in the chapter on materials evaluation, includes the source of all materials data, particularly the data for fuel and fixed poison materials. In coordination with the materials reviewer, determine the acceptability of the sources of data that are important to the criticality safety function of the storage container. Also in coordination with the materials reviewer, ensure that the applicant addressed the validation of the fixed neutron absorbers' poison concentration in the chapter of the SAR that describes the acceptance tests and maintenance programs. Criticality computer codes generally will allow the densities to be input directly in units of grams per cubic centimeter or units of atoms per barn-centimeter. In either case, pay attention to the final values used directly by the code. Confirm that the values used for neutron poisons (solid and soluble) match the minimum required values credited in the criticality analysis. Also, for the solid, fixed absorbers, confirm that the analysis does not take credit for more than the minimum amount of neutron absorber verified by the acceptance testing, subject to the criteria in Section 7.4 of this chapter (see also Section 8.5.7 of this SRP).

Among other specifications, 10 CFR Part 72 requires that the applicant provide a positive means to verify the continued efficacy of solid neutron-absorbing materials when these materials are used. Verify that the SAR indicates that the neutron flux from the irradiated fuel results in a negligible depletion of poison material over the storage period. In coordination with the materials and structural reviewers, ensure that the applicant demonstrates that the required acceptance testing of the poisons during fabrication (stated in the chapter of the SAR on acceptance tests and maintenance program evaluation) has been satisfactorily specified and, by analysis or demonstration, that the applicant has shown the poison material's durability and resistance to degradation during the certified or licensed storage period.

The neutron flux used for this analysis should be the maximum that may be produced by feasible loadings of irradiated or unirradiated fuel. Coordinate review of the applicant's acceptance testing and assessment of the poison material's durability with the materials reviewer to verify that the applicant provided a valid and accurate demonstration of the absorber material's continued efficacy. Consider the effects of physical and chemical actions as well as irradiation (gamma and neutron). There may be other ways to provide positive means of verifying the neutron absorber's continued efficacy. For applications that propose an alternative method, verify that the proposed method is reasonable (considering any effects on meeting confinement, shielding, or other system design criteria), valid, and accurate in demonstrating the absorber's continued efficacy.

(SL) When applying this guidance to absorbers used in a pool that is part of the DSF design and operations in a specific license application, ensure that the SAR appropriately considers the operating environment to which these absorbers will be exposed. Given the configuration for the pool and the operating environment, periodic monitoring of the absorbers may be necessary. Work with the materials reviewer (SRP Chapter 8) to evaluate the adequacy of the proposed monitoring.

7.5.4 Criticality Analysis

7.5.4.1 Computer *Codes and Cross-Section Data*

Both Monte Carlo and deterministic computer codes may be used for criticality calculations. Monte Carlo computer codes are better suited to three-dimensional geometry and, therefore, are more widely used to evaluate DSS and DSF storage container designs. The most frequently used Monte Carlo codes are the KENO V.a and KENO VI sequences of the SCALE code system (ORNL 2011) and MCNP (LANL 2003). These codes permit the use of either multi-group or continuous-energy cross sections. Determine whether the applicant has used a computer code that is appropriate for the particular application and has used that code correctly. Ensure that the SAR describes the code the applicant used for its analyses and provides appropriate supplemental information for codes other than those described above to enable this determination. Verify that the information regarding the model configuration, material properties, and cross sections is properly input into the code.

Determine whether the applicant has chosen an acceptable set of cross sections. Cross sections may be distributed with the criticality computer codes or developed independently from another source. Ensure that the applicant provided or referenced the source of cross section data. For user-generated cross sections, verify that the applicant specified the method used to obtain the actual data employed in the criticality analysis. For multi-group calculations, the neutron flux spectrum used to construct the group cross sections should be similar to that of the contents in the storage container. In addition to selecting a cross section set collapsed with an appropriate flux spectrum, a more detailed processing of the multi-group cross sections is necessary to properly account for resonance absorption and self-shielding. The use of multi-group KENO as part of the critical safety analysis sequences in SCALE will directly enable appropriate cross section processing.

More recent versions of Monte Carlo criticality codes can use continuous-energy cross section libraries, which require little or no cross section processing. Use continuous-energy cross sections in the confirmatory analyses, if available, particularly when the applicant has used a multi-group cross section library. This can serve as a check on the cross section processing techniques the applicant employed.

Information has been published concerning problems with some cross section libraries once commonly distributed with SCALE and KENO. One library, the "working-format" library, was used for calculations of the code manual's sample problems but is not intended for criticality calculations of actual systems (see Information Notice 91-26, "Potential Nonconservative Errors in the Working Format Hansen-Roach Cross-Section Set Provided with the KENO and Scale Codes," dated April 2, 1991). Another library, the SCALE 123-group library, is inadequate for non-thermalized, highly enriched systems (see NUREG/CR-6328, "Adequacy of the 123-Group

Cross-Section Library for Criticality Analyses of Water-Moderated Uranium Systems"), and may result in non-conservative estimates of $k_{eff.}$

Pay particular attention to the proper selection of scattering cross section data for important compounds that may be in the system. Use of a free atom cross section for nuclides in a compound may not adequately account for the scattering effects of atoms bound in molecules and lattices. This is particularly true for hydrogen bound in water, which is the most common moderator in SNF storage containers. This misrepresentation can cause the under-prediction of k_{eff}, particularly in the case of a well-moderated system where energetic up-scattering plays a significant role in the neutronics of the system.

For analyses of a storage container model with separate regions of water and steam, the use of a multi-group cross section set raises additional concerns. Verify that the applicant has addressed the differences in the flux spectra in the two regions. If the results of these calculations indicate that k_{eff} is close to 0.95, it may be necessary to conduct additional independent calculations using a different code, cross section library (a library derived from a different cross section database if possible and appropriate), or both, to confirm the applicant's calculated $k_{eff.}$ Closely examine the applicant's benchmark analysis to verify that the critical experiments the applicant considered are applicable to water- and steam-moderated systems. Note that if dissolved boron is credited for criticality control, it will not be present in the steam region.

7.5.4.2 *Neutron Multiplication Factor*

Examine the results and discussion of the k_{eff} calculations for the DSS or DSF. Verify that the calculations determine the highest k_{eff} that might occur during all operational states under normal, off-normal, and accident conditions. The applicant may have used sensitivity analyses to provide the required demonstration that the highest k_{eff}, with a 95 percent probability at a confidence level of 95 percent, has been determined. Verify that the SAR explains the variations in the results caused by differences in the models and sensitivity analyses and that such variations are reasonable.

For Monte Carlo calculations, assess whether the number of neutron histories and convergence criteria are appropriate. As the number of neutron histories increases, the mean value for k_{eff} should approach a fixed value, and the standard deviation associated with each mean value should decrease. Depending on the code the applicant used, a number of diagnostic calculations are generally available to demonstrate adequate convergence and statistical variation. For deterministic codes, a convergence limit is often prescribed in the input. Confirm that the SAR, or supporting criticality calculations, describes and demonstrates the selection of a proper convergence limit and the achievement of this limit. When burnup credit is included in the criticality analysis, confirm that proper neutron sampling and convergence have been achieved because the flux will be concentrated in the low-burnup ends of the fuel assemblies.

Because of the importance and complexity of the criticality evaluation, perform independent calculations to ensure that the applicant has addressed the most reactive conditions, the reported k_{eff} is conservative, and the applicant has appropriately modeled the storage container geometry and materials. In deciding the level of effort necessary to perform independent confirmatory calculations, consider the following factors:

- the calculation method (computer code) used by the applicant

- uniqueness and complexity of the design and analysis, compared with previously approved DSSs and DSF storage containers

- the degree of conservatism in the applicant's assumptions and analyses

- the extent of the margin between the calculated result and the acceptance criterion of k_{eff} less than 0.95

As with any design and review, a small margin below the acceptance criterion or a small degree of conservatism (or both) may necessitate a more extensive staff analysis.

Develop a model that is independent of the applicant's model. If the reported k_{eff} for the most reactive case is substantially lower than the acceptance criterion of 0.95, a simple model(s) known to produce very bounding results may be all that is necessary for the independent calculations.

If possible and appropriate, perform the independent calculations with a computer code different from the code the applicant used. Likewise, use of a different cross-section set, derived from a different cross section database, where possible and appropriate (e.g., ENDF/B, JEF, JENDL, UKNDL), can provide a more independent confirmation. The continuous-energy cross sections created for use with KENO in the SCALE code system are generated by the AMPX processing code rather than the more widely used NJOY code. Even though some cross section libraries may not have fully independent databases because they are all derived from ENDF/B data, the continuous-energy library in SCALE still can provide some level of independence and is useful for checking computations performed with libraries that were generated by using NJOY. Describe the staff's independent analysis, the analysis's general results, and the staff's conclusions in the SER.

Although a k_{eff} of 0.95 or lower meets the acceptance criterion, watch for design features or content specifications where small changes could result in large changes in the value of k_{eff}. When the value of k_{eff} is highly sensitive to system parameters that could vary, the acceptable k_{eff} limit may need to be reduced to below 0.95. When establishing a k_{eff} limit below 0.95, consider the degree of sensitivity to system parameter changes and the likelihood and extent of potential parameter variations.

7.5.4.3 *Benchmark Comparisons*

Computer codes for criticality calculations should be benchmarked against critical experiments. A thorough comparison provides justification for the validity of the computer code, its use for a specific hardware configuration, its use for the SNF to be stored, the neutron cross sections used in the analysis, and consistency in modeling by the analyst. Ultimately, the benchmarking process establishes a bias and bias uncertainty for the particular application of the code (using the benchmark results for calculations performed by another analyst does not address this last issue). Calculated k_{eff} values should then be adjusted to include the appropriate biases and bias uncertainties from the benchmark calculations.

Examine the general description of the benchmark comparisons. This examination includes verifying that the analysis of the experiments used the same computer code, computer system, cross section data, modeling methods, and code options that were used to calculate the k_{eff} values for the storage containers.

Closely examine the applicant's benchmark analysis to determine whether the benchmark experiments are relevant to the actual storage container design. No critical benchmark experiment will precisely match the fissile material, moderation, neutron poisoning, and geometric configuration in the actual storage container. However, the applicant can perform a proper benchmark analysis by selecting experiments that adequately represent storage container and fuel features and parameters that are important to reactivity. Key features and parameters that should be considered in selecting appropriate critical experiments include the type of fuel, enrichment, hydrogen-to-uranium ratio (dependent largely on rod diameter and pitch), reflector material, neutron energy spectrum, and poisoning material and placement. Confirm that the applicant discusses and properly considers the differences between the benchmark experiments and the storage containers and their contents. Ensure that the SAR addresses the overall quality of the benchmark experiments and the uncertainties in the experimental data (e.g., mass, density, dimensions). Verify that the applicant treated these uncertainties in a conservative manner (i.e., used in a way that results in a lower calculated k_{eff} for the benchmark experiment).

Verify the applicant's justification of the suitability of the critical experiments chosen to benchmark the criticality code and calculations. Techniques such as the sensitivity and uncertainty method developed by Oak Ridge National Laboratory (ORNL 2011) can be helpful when assessing the applicability of the critical experiments used to benchmark the design analysis. UCID-21830, "Determination and Application of Bias Values in the Criticality Evaluation of Storage Cask Designs," issued January 1990; the Nuclear Energy Agency's "International Handbook of Evaluated Criticality Safety Benchmark Experiments"; and NUREG/CR-6361, "Criticality Benchmark Guide for Light-Water-Reactor Fuel in Transportation and Storage Packages," issued March 1997, provide information on benchmark experiments that may apply to the storage containers being analyzed.

Assess whether the applicant analyzed a sufficient number of appropriate benchmark experiments (which is dependent, in part, on the statistical treatment used) and how the applicant converted the results of these benchmark calculations to a bias for the k_{eff} calculations. Simply averaging the biases from a number of benchmark calculations typically is not sufficient, such as when one benchmark yields results that are significantly different from the others, the number of experiments is limited, or when groups of experiments are heavily correlated. In addition, the bias may exhibit trends with respect to parameter variations (such as pitch-to-rod-diameter ratio, assembly separation, reflector material, neutron absorber material). Verify that the applicant has adequately assessed the benchmark analysis results to identify any bias trends and considered these trends in developing a bias for the k_{eff} calculations. UCID-21830 and NUREG/CR-6361 provide some guidance; however, other methods, when adequately explained, have also been considered appropriate.

For Monte Carlo codes, ensure that the applicant also addresses the statistical uncertainties of both the benchmark and the k_{eff} calculations. The uncertainties should be applied to at least the 95 percent confidence level. As a general rule, if the acceptability of the result depends on these rather small differences, question the overall degree of conservatism of the calculations.

Considering the current availability of computer resources, a sufficient number of neutron histories can readily be used so that the treatment of these uncertainties should not significantly affect the results.

Verify that the applicant has applied only biases that increase the calculated $k_{eff.}$ If the benchmark analysis results in a positive bias (i.e., one that would decrease the calculated k_{eff}), the bias should be conservatively set to zero. Only corrections that increase k_{eff} should be applied to preserve conservatism.

The reviewer may have already performed a number of benchmark calculations applicable to storage containers and may have a reasonable estimation of the bias to be applied to the independent calculation of the k_{eff} for the storage containers. If such is not the case, or if the acceptability depends on small bias differences, determine whether sufficient conservatism has been applied to the calculations.

7.5.5 Burnup Credit

The regulations in 10 CFR Part 72 require that SNF remain subcritical in storage. While unirradiated reactor fuel (or "fresh fuel") has a well-specified nuclide composition that provides a straightforward and bounding approach to the criticality safety analysis of transportation and storage systems, the nuclide composition changes as the fuel is irradiated in the reactor. Ignoring the presence of burnable poisons, this composition change will cause the reactivity of the fuel to decrease. In the criticality safety analysis, allowance for the decrease in fuel reactivity resulting from irradiation is termed "burnup credit."

This section provides recommendations to the NRC reviewer for accepting, on a design-specific basis, a burnup credit approach in the criticality safety analysis of PWR SNF storage containers. The recommendations are based on DSS-type storage container designs; however, they may also be applied to other SNF storage container designs, with appropriate consideration of the differences between container designs. For specific license applications, the recommendations may also be applied to criticality analyses for SNF in other relevant DSF SSCs (e.g., a pool that is part of the DSF design and operations), with appropriate consideration of impacts of these SSCs' features on the bases for and the application of the recommendations. The guidance represents one methodology for demonstrating compliance with the criticality safety requirement in 10 CFR Part 72 using burnup credit. Follow this guidance to determine whether the applicant has adequately demonstrated that the storage system meets the applicable criticality safety regulations in 10 CFR Part 72. Consider proposed alternative methodologies on a case-by-case basis, using this guidance to the extent practicable.

The recommendations that follow were developed with intact fuel as the basis but may also be applicable to fuel that is not intact. If an applicant requests burnup credit for fuel that is not intact, apply the recommendations provided below, as appropriate, to account for uncertainties that can be associated with fuel that is not intact and establish an isotopic inventory and assumed fuel configuration for normal, off-normal, and accident conditions that bound the uncertainties.

The recommendations in this chapter do not include burnup credit for BWR fuel assemblies, as the technical basis for BWR burnup credit in SNF storage containers has not been fully developed. The NRC has initiated a research project to obtain that technical basis. BWR fuel assemblies typically have neutron-absorbing material, typically gadolinium oxide, mixed in with the uranium oxide of the fuel pellets in some rods. This neutron absorber depletes more rapidly than the fuel during the initial parts of its irradiation, which causes the fuel assembly reactivity to increase and reach a maximum value at an assembly average burnup typically less than 20 GWd/MTU. Then reactivity decreases for the remainder of fuel assembly irradiation. Criticality analyses of BWR spent fuel pools typically employ what are known as "peak reactivity" methods to account for this behavior. NUREG/CR-7194, "Technical Basis for Peak Reactivity Burnup Credit for BWR Spent Nuclear Fuel in Storage and Transportation Systems," reviews several existing peak reactivity methods, and demonstrates that a conservative set of analysis conditions can be identified and implemented to allow criticality safety analysis of BWR spent fuel assemblies at peak reactivity in SNF storage containers. Consult NUREG/CR-7194 if the applicant used peak reactivity BWR burnup credit methods in its criticality analysis.

This SRP does not address credit for BWR burnup beyond peak reactivity; the NRC is currently evaluating this as part of a research program to investigate methods for conservatively including such credit in a BWR criticality analysis for SNF storage containers. The NRC does not recommend burnup credit beyond peak reactivity at this time. Consider conservative analyses of BWR burnup credit beyond peak reactivity on a case-by-case basis, consulting the latest research results in this area (i.e., NRC letter reports, NUREG/CRs).

The recommendations in this section also do not include burnup credit analyses for MOX or thorium fuel assemblies. Evaluate MOX burnup credit analyses on a case-by-case basis, noting that there is little MOX data available for isotopic depletion or criticality code validation. Such evaluations should include a large amount of conservatism in the representation of MOX material in the criticality model, and large k_{eff} penalties for unvalidated fuel materials. Thorium fuel criticality analyses will require a depletion analysis to determine the most reactive fuel composition with irradiation. Similar to MOX fuel, there is little code validation data available for thorium fuel, and criticality analyses should include large conservatisms and k_{eff} penalties for unvalidated materials.

Appendix 7A to this SRP chapter provides more information on the technical bases for the recommendations provided below.

7.5.5.1 *Limits for the Licensing Basis*

Available data support allowance for burnup credit where the safety analysis is based on major actinide compositions only (i.e., actinide-only burnup credit) or limited actinide and fission product compositions (see Table 7-2 below) associated with UO_2 fuel irradiated in a PWR up to an assembly-average burnup value of 60 GWd/MTU and cooled out of reactor for a period between 1 and 40 years. The range of available measured assay data for irradiated UO_2 fuel supports an extension of the licensing basis up to 5.0 weight percent enrichment in uranium-235.

Table 7-2 Recommended Set of Nuclides for Burnup Credit

Type of Burnup Credit	Recommended Set of Nuclides
Actinide-only burnup credit	^{234}U, ^{235}U, ^{238}U, ^{238}Pu, ^{239}Pu, ^{240}Pu, ^{241}Pu, ^{242}Pu, ^{241}Am
Additional nuclides for actinide-plus-fission product burnup credit	^{95}Mo, ^{99}Tc, ^{101}Ru, ^{103}Rh, ^{109}Ag, ^{133}Cs, ^{143}Nd, ^{145}Nd, ^{147}Sm, ^{149}Sm, ^{150}Sm, ^{151}Sm, ^{152}Sm, ^{151}Eu, ^{153}Eu, ^{155}Gd, ^{236}U, ^{237}Np, ^{243}Am

Within this range of parameters, exercise care in assessing whether the analytic methods and assumptions used are appropriate, especially near the limits of the parameter ranges recommended here for the licensing basis. Verify that the use of actinide and fission product compositions associated with burnup values or cooling times outside these specifications is accompanied by the measurement data or justifies extrapolation techniques, or both, necessary to extend the isotopic validation and quantify or bound the bias and bias uncertainty. If the applicant credits neutron-absorbing isotopes other than those identified in Table 7-2, ensure that the applicant gives assurance that such isotopes are nonvolatile, nongaseous, and relatively stable, and provides analyses to determine the additional depletion and criticality code bias and bias uncertainty associated with these isotopes.

A certificate or license condition indicating the time limit on the validity of the burnup credit analysis may be necessary in light of the potential need for extended dry storage. Such a condition would depend on the type of burnup credit and the credited post-irradiation decay time.

7.5.5.2 *Licensing-Basis Model Assumptions*

Confirm that the applicant calculated the actinide and fission product compositions used to determine a value of k_{eff} for the licensing basis using fuel design and reactor operating parameter values that appropriately encompass the range of design and operating conditions for the proposed contents. Verify that the applicant performed the calculation of the k_{eff} value using models and analysis assumptions that allow accurate representation of the physics in the storage container, as discussed in Section 7A.4 of Appendix 7A to this chapter. Pay attention to the need to do the following:

- Account for and effectively model the axial and horizontal variation of the burnup within a spent fuel assembly (e.g., the selection of the axial burnup profiles, number of axial material zones).

- Consider the potential for increased reactivity because of the presence of burnable absorbers or control rods (fully or partially inserted) during irradiation.

- Account for the irradiation environment factors to which the proposed assembly contents were exposed, including fuel temperature, moderator temperature and density, soluble boron concentration, specific power, and operating history.

YAEC-1937, "Axial Burnup Profile Database for Pressurized Water Reactors," issued May 1997, provides a source of representative data that can be used for establishing profiles to use in the licensing-basis safety analysis. However, exercise care when reviewing profiles intended to bound the range of potential k_{eff} values for the proposed contents for each burnup range, particularly near the upper end of the licensing basis parameter ranges stated in this guidance. NUREG/CR-6801, "Recommendations for Addressing Axial Burnup in PWR Burnup Credit Analyses," issued March 2003, provides additional guidance on selecting axial profiles.

A licensing-basis modeling assumption, where the assemblies are exposed during irradiation to the maximum (neutron absorber) loading of BPRs for the maximum burnup, encompasses all assemblies that may or may not have been exposed to BPRs. Such an assumption in the licensing-basis safety analysis should also encompass the impact of exposure to fully inserted or partially inserted control rods in typical domestic PWR operations. Assemblies exposed to atypical insertions of control rods (e.g., full insertion for one full cycle of reactor operation) should not be loaded unless the safety analysis explicitly considers such operational conditions. If the assumption on BPR exposure is less than the maximum for which burnup credit is requested, confirm that the applicant has provided a justification commensurate with the selected value. For example, the lower the exposure, the greater the need to (1) support the assumption with available data, (2) indicate how administrative controls would prevent a misload of an assembly exposed beyond the assumed value, and (3) address such misloads in a misload analysis.

For assemblies exposed to integral burnable absorbers, the appropriate analysis assumption for absorber exposure varies depending upon burnup and absorber material. The appropriate assumption may be to neglect the absorber while maintaining the other assembly parameters (e.g., enrichment) the same for some absorber materials or for exposures up to moderate burnup levels (typically 20–30 GWd/MTU). Thus, a safety analysis including assemblies with integral burnable absorbers should include justification of the absorber exposure assumptions used in the analysis. For assemblies exposed to flux suppressors (e.g., hafnium suppressor inserts) or combinations of integral absorbers and BPRs or control rods, the safety analysis should use assumptions that provide a bounding safety basis, in terms of the effect on storage container k_{eff}, for those assemblies.

Confirm that the applicant's licensing-basis evaluation includes analyses that use irradiation conditions that produce bounding values for k_{eff}, as discussed in Section 7A.4 of Appendix 7A to this chapter. The bounding conditions may differ for actinide-only burnup credit versus actinide-plus-fission product burnup credit and may depend on the population of fuel intended to be loaded in the storage container (e.g., all PWR assemblies versus a site-specific population). Loading limitations tied to the actual operating conditions may be needed unless the operating condition values used in the licensing-basis evaluation can be justified as those that produce the maximum k_{eff} values for the anticipated SNF inventory.

7.5.5.3 *Code Validation—Isotopic Depletion*

Confirm that the applicant validated the computer codes used to calculate isotopic depletion. A depletion computer code is used to determine the concentrations of the isotopes important to burnup credit. To ensure accurate criticality calculation results, the selected code should be validated and the bias and bias uncertainty of the code should be determined at a 95-percent probability, 95-percent confidence level. Ensure that the application reflects the following considerations in the selection of the code and code validation approach for the fuel depletion analysis.

The selected depletion code and cross section library should be capable of accurately modeling the fuel geometry and the neutronic characteristics of the environment in which the fuel was irradiated. Two-dimensional depletion codes have been effectively used in burnup credit analyses. Although one-dimensional codes have been used in some applications and suffice for making assembly average isotopic predictions for fuel burnup, they are limited in their ability to model increasingly complex fuel assembly designs and generally produce larger bias and bias uncertainty values because of the approximations necessary in the models. Section 7A.4 of Appendix 7A to this chapter provides detailed discussions of the modeling considerations for the code validation analyses.

The destructive radiochemical assay (RCA) data selected for code validation should include detailed information about the SNF samples. This information should include the pin location in the assembly, axial location of the sample in the pin, any exposure to strong absorbers (control rods, BPRs), the boron letdown, moderator temperature, specific power, and any other cycle-specific data for the cycles in which the sample was irradiated. Note that some RCA data are not suitable for depletion code validation because the depletion histories or environments of these samples are either difficult to accurately define in the code benchmark models or are unknown. NUREG/CR-7108, "An Approach for Validating Actinide and Fission Product Burnup Credit Criticality Safety Analyses—Isotopic Composition Predictions," issued April 2012, provides a recommended set of RCA data suitable for depletion code validation.

The selected code validation approach should be adequate for determining the bias and bias uncertainty of the code for the specific application. The burnup credit analysis results should be adjusted using the bias and bias uncertainty determined for the fuel depletion code, accounting for any trends of significance with respect to different control parameters such as burnup-to-enrichment ratio or ratio of uranium-235 to plutonium-239. NUREG/CR-6811, "Strategies for

Application of Isotopic Uncertainties in Burnup Credit," issued June 2003, provides several methodologies the NRC finds acceptable for isotopic depletion validation, including the isotopic correction factor, direct-difference, and Monte Carlo uncertainty sampling methods. Section 7A.4 of Appendix 7A to this chapter provides detailed discussions of the advantages and disadvantages of these methods. In general, the isotopic correction factor method is considered to be the most conservative because individual nuclide composition uncertainties are represented as worst case. The direct-difference method provides a realistic "best estimate" of the depletion code bias and bias uncertainty, in terms of difference in k_{eff} (Δk_{eff}). The Monte Carlo uncertainty sampling method is more complex and computationally intensive than the other methods, but it provides a way to make use of limited measurement data sets for some nuclides. NUREG/CR-7108 provides detailed descriptions of the direct-difference and Monte Carlo uncertainty sampling methods.

In lieu of an explicit benchmarking analysis, the applicant may use the bias (β_i) and bias uncertainty (Δk_i) values estimated in NUREG/CR-7108 using the Monte Carlo uncertainty sampling method, as shown in Tables 7-3 and 7-4. These values may be used directly, provided that all of the following is true:

- The applicant uses the same depletion code and cross section library as were used in NUREG/CR-7108 (SCALE/TRITON and the ENDF/B-V or ENDF/B-VII cross section library).
- The applicant can justify that its storage container design is similar to the hypothetical 32-PWR-assembly-capacity, generic burnup credit cask (GBC-32) system design (NUREG/CR-6747, "Computational Benchmark for Estimation of Reactivity Margin from Fission Products and Minor Actinides in PWR Burnup Credit," issued October 2001) and used as the basis for the NUREG/CR-7108 isotopic depletion validation.
- Credit is limited to the specific nuclides listed in Table 7-2.

Section 7A.5 of Appendix 7A to this chapter provides detailed discussions of the technical basis for the restrictions on directly applying the bias and bias uncertainty values. Bias values should be added to the calculated storage container k_{eff}, while bias uncertainty values may be statistically combined with other independent uncertainties. Table 7-5 summarizes the recommendations related to isotopic depletion code validation.

Table 7-3 Isotopic k_{eff} Bias Uncertainty (Δk_i) for the Representative PWR SNF System Model Using ENDF/B VII Data (β_i = 0) as a Function of Assembly Average Burnup

Burnup (BU) Range (GWd/MTU)	Actinides Only Δk_i	Actinides and Fission Products Δk_i
0≤BU<5	0.0145	0.0150
5≤BU<10	0.0143	0.0148
10≤BU<18	0.0150	0.0157
18≤BU<25	0.0150	0.0154
25≤BU<30	0.0154	0.0161
30≤BU<40	0.0170	0.0163
40≤BU<45	0.0192	0.0205
45≤BU<50	0.0192	0.0219
50≤BU≤60	0.0260	0.0300

Table 7-4 Isotopic k_{eff} Bias (β_i) and Bias Uncertainty (Δk_i) for the Representative PWR SNF System Model Using ENDF/B-V Data as a Function of Assembly Average Burnup

Burnup (BU) Range (GWd/MTU)[a]	β_i for Actinides and Fission Products	Δk_i for Actinides and Fission Products
0≤BU<10	0.0001	0.0135
10≤BU<25	0.0029	0.0139
25≤BU≤40	0.0040	0.0165

a. Bias and bias uncertainties associated with ENDF/B-V data were calculated for a maximum of 40 GWd/MTU. For burnups higher than this, confirm that the applicant provided an explicit depletion code validation analysis, using one of the methods described in Appendix 7A to this chapter, along with appropriate RCA data.

Table 7-5 Summary of Code Validation Recommendations for Isotopic Depletion

Applicant's Approach	Recommendation
Applicant uses SCALE/TRITON and the ENDF/B-V or ENDF/B-VII cross section library, and demonstrates that the design application is similar to GBC-32.	Use code bias and bias uncertainty values from Tables 7-3 and 7-4 of this SRP.
- or -	
Applicant uses other code or cross section library, or both, or design application is not similar to GBC-32.	Use either isotopic correction factor or direct-difference method to determine code bias and bias uncertainty.

7.5.5.4 *Code Validation—k_{eff} Determination*

7.5.5.4.1 Actinide-Only Credit

Actinide credit should be limited to the specific nuclides listed in Table 7-2. Criticality validation for these actinides should be based on the critical experiments available in NUREG/CR-6979, "Evaluation of the French Haut Taux de Combustion (HTC) Critical Experiment Data," issued September 2008, also known as the HTC data, supplemented by MOX critical experiments as appropriate. NUREG/CR-7109, "An Approach for Validating Actinide and Fission Product Burnup Credit Criticality Safety Analyses—Criticality (k_{eff}) Predictions," issued April 2012, contains a detailed discussion of available sets of criticality validation data for actinide isotopes, and the relative acceptability of these sets. Note that NUREG/CR-7109 demonstrates that fresh UO_2 experiments are not applicable to burned fuel compositions.

Verify that the applicant's determination of the bias and bias uncertainty associated with actinide-only burnup credit was performed according to the guidance in NUREG/CR-6361. This guidance includes criteria for the selection of appropriate benchmark data sets, as well as statistics and trending analysis for the determination of criticality code bias and bias uncertainty. Section 6 of NUREG/CR-7109 provides an example of bias and bias uncertainty determination for actinide-only burnup credit.

7.5.5.4.2 Fission Product and Minor Actinide Credit

Confirm that the applicant has determined an adequate and conservative bias and bias uncertainty associated with fission product and minor actinide credit. The applicant may credit the minor actinide and fission product nuclides listed in Table 7-2, provided the bias and bias uncertainty associated with the major actinides is determined as described above. The bias from these minor actinides and fission products is conservatively covered by 1.5 percent of their worth. Because of the conservatism in this value, no additional uncertainty in the bias needs to be applied. This estimate is appropriate if the applicant does the following:

- uses the SCALE code system with the ENDF/B-V, ENDF/B-VI, or ENDF/B-VII cross section libraries, or MCNP5 or MCNP6 with the ENDF/B-V, ENDF/B-VI, ENDF/B-VII, or ENDF/B-VII.1 cross section libraries.

- can justify that its storage container design is similar to the hypothetical GBC-32 system design (NUREG/CR-6747) used as the basis for the NUREG/CR-7109 criticality validation

- demonstrates that the credited minor actinide and fission product worth is no greater than 0.1 in k_{eff}

For well-qualified industry standard code systems other than SCALE or MCNP, the applicant may use a conservative estimate for the bias associated with minor actinide and fission product nuclides of 3.0 percent of their worth. If the applicant uses a minor actinide and fission product bias less than 3.0 percent, ensure that the application includes additional justification that the lower value is an appropriate estimate of the bias associated with that code system (e.g., a minor actinide and fission product worth comparison to SCALE results or an analysis similar to that described in NUREG/CR-7109 or NUREG/CR-7205, “Bias Estimates Used in Lieu of Validation of Fission Products and Minor Actinides in MCNP K_{eff} Calculations for PWR Burnup Credit Casks”). Table 7-6 summarizes the recommendations related to minor actinide and fission product code validation for k_{eff} determination. For actinide criticality validation in all cases, the applicant should perform criticality code validation analyses to determine bias and bias uncertainty associated with actinides using HTC critical experiments, supplemented by applicable MOX critical experiments. Ensure that the applicant performed the validation analyses correctly and adequately.

Table 7-6 Summary of Minor Actinide and Fission Product Code Validation Recommendations for k*eff* Determination

Applicant’s Approach	Recommendation
Applicant uses SCALE code system with ENDF/B-V, ENDF/B-VI, or ENDF/B-VII cross section libraries, or MCNP5 or MCNP6 with the ENDF/B-V, ENDF/B-VI, ENDF/B-VII, or ENDF/B-VII.1 cross section libraries; design application is similar to GBC-32; and credited minor actinide and fission product is worth <0.1 in $k_{eff.}$	Use bias equal to 1.5 percent of minor actinide and fission product worth.
- or -	
Applicant uses other code with ENDF/B-V, ENDF/B-VI, or ENDF/B-VII cross section libraries; design application is similar to GBC-32; and credited minor actinide and fission product is worth <0.1 in $k_{eff.}$	Use bias equal to 3.0 percent of minor actinide and fission product worth, or provide justification for lower number.
- or -	
Applicant uses cross section library other than ENDF/B-V, ENDF/B-VI, or ENDF/B-VII; design application is not similar to GBC-32; or credited minor actinide and fission product is worth >0.1 in $k_{eff.}$	Perform explicit criticality code validation for minor actinide and fission product nuclides.

7.5.5.5 *Loading Curve and Burnup Verification*

Confirm that the applicant provided burnup credit loading curves to determine which fuel assemblies may be loaded in a storage container. Confirm that the burnup credit evaluations include loading curves that specify the minimum required assembly average burnup as a function of initial enrichment for the purpose of loading SNF storage containers. Confirm that separate loading curves are established for each set of applicable licensing conditions. For example, a separate loading curve should be provided for each minimum cooling time to be considered in the container loading. In addition, confirm that the SAR includes a justification of the applicability of the loading curve to bound various fuel types or burnable absorber loadings.

Ensure that the criticality analysis and operations description chapters in the SAR include performance of burnup verification to ensure that a storage container evaluated using burnup credit is not loaded with an assembly more reactive than those included in the loading criteria. Verification should include a measurement that confirms the reactor record for each assembly. Confirmation of reactor records using measurement of a sample of fuel assemblies will be considered if the sampling method can be justified in comparison to measuring every assembly.

The assembly burnup value to be used for loading acceptance (termed the assigned burnup loading value) should be the confirmed reactor record value as adjusted by reducing the record value by a combination of the uncertainties in the record value and the measurement. NUREG/CR-6998, “Review of Information for Spent Nuclear Fuel Burnup Confirmation,” issued December 2009, contains bounding estimates of reactor record burnup uncertainty.

Measurements should be correlated to reactor record burnup, enrichment, and cooling time values. Measurement techniques should account for any measurement uncertainty (typically within a 95-percent confidence interval) in confirming reactor burnup records. They should also include a database of measured data (if measuring a sampling of fuel assemblies) to justify the adequacy of the procedure in comparison to procedures that measure each assembly.

7.5.5.5.1 Misload Analyses

Misload analyses may be performed in lieu of a burnup measurement. A misload analysis should address potential events involving the placement of assemblies into a SNF storage container that do not meet the proposed loading criteria. Confirm that the applicant has demonstrated that the container remains subcritical for misload conditions, including calculation biases, uncertainties, and an appropriate administrative margin that is not less than 0.02 Δk. If any administrative margin less than the normal 0.05 Δk is used, verify that the SAR provides an adequate justification that includes the level of conservatism in the depletion and criticality calculations, sensitivity of the container to further upset conditions, and the level of rigor in the code validation methods.

If used, ensure that the misload analysis considers (1) misloading of a single, severely underburned assembly and (2) misloading of multiple, moderately underburned assemblies.

The severely underburned assembly for the single misload analysis should be chosen such that the misloaded assembly’s reactivity bounds 95 percent of the discharged PWR fuel population considered unacceptable for loading in a particular storage container with 95-percent confidence. The moderately underburned assemblies for the multiple-misload analysis should be assumed to make up at least 50 percent of the container payload and should be chosen such that the misloaded assemblies’ reactivity bounds 90 percent of the total discharged PWR fuel population.

The NRC finds the results of the most recent Energy Information Administration nuclear fuel data survey, RW-859, “Nuclear Fuel Data Survey,” or later similar fuel data sources, acceptable to estimate the discharged fuel population characteristics.

Also ensure that the misload analysis considers the effects of placing the underburned assemblies in the most reactive positions within the loaded container (e.g., middle of the fuel basket). If removable nonfuel absorbers were credited as part of a criticality safety analysis (e.g., poison rods added to guide tubes), ensure that the misload analysis considers misloading of these absorbers. Additionally, ensure that the misload analysis considers assemblies with greater burnable absorber or control rod exposure than assumed in the criticality analysis if the assumed exposure is not bounding. NUREG/CR-6955, “Criticality Analysis of Assembly Misload in a PWR Burnup Credit Cask,” issued January 2008, illustrates the magnitude of k_{eff} changes that can be expected as a result of various misloads in a theoretical GBC-32 SNF storage system.

7.5.5.5.2 Administrative Procedures

Confirm that the applicant has included administrative loading procedures that will protect against misloads. Ensure that the misload analysis is coupled with additional administrative procedures to ensure that the SNF storage container will be loaded with fuel that is within the specifications of the approved contents. Procedures the applicant may consider to protect against misloads in storage containers that rely on burnup credit for criticality safety include the following:

- verification of the location of high-reactivity fuel (i.e., fresh or severely underburned fuel) in the spent fuel pool, both before and after loading
- qualitative verification that the assembly to be loaded is burned (visual or gross measurement)
- quantitative measurement of any fuel assemblies without visible identification numbers
- independent, third-party verification of the loading process, including the fuel selection process and generation of the fuel move instructions
- minimum soluble boron concentration in pool water, to offset the misloads described above, during loading and unloading

Table 7-7 summarizes the recommendations for burnup verification.

Table 7-7 Summary of Burnup Verification Recommendations

Applicant’s Approach	Recommendation
Applicant takes burnup verification measurement.	Perform measurement for each assembly to be loaded or for a statistically significant sample of assemblies.
- or -	
Applicant conducts misload analysis and provides additional administrative procedures.	Analyze misload of fuel assembly that bounds reactivity of 95 percent of underburned fuel population with 95-percent confidence.
	Analyze misload of 50 percent of system capacity with fuel assemblies with reactivity that bounds 90 percent of total fuel population.
	Include additional administrative procedures as part of storage container loading.

7.5.6 Reactor-Related Greater-Than-Class-C Waste and HLW (SL)

(SL) The specifications for materials stored at a DSF should include the ranges of properties of concern for criticality analysis, which may include HLW and reactor-related GTCC waste characteristics if these wastes are to be stored at the DSF and they contain fissile material. Chapter 3, "Principal Design Criteria Evaluation," of this SRP provides guidance on the data that should be provided.

(SL) For these wastes, characteristics of concern for the various criticality analyses include those listed below. Verify that the SAR states these characteristics for the radioactive materials in these wastes for which criticality analysis is appropriate. Ensure that the SAR identifies radioactive materials that, because of their atomic or physical properties, are not of criticality concern, and includes this as the rationale for not including criticality analyses. Verify that the applicant has provided the data identified below, regardless of whether or not they are included in the applicant's analytical approach, as they may be needed for confirmatory and independent analyses by the NRC staff:

- the isotopes present and their densities
- means by which the fissile and fissionable isotope densities are limited
- geometric data on the configuration (e.g., racks, basket) holding the materials, including tolerances and uncertainties, and neutron-absorption material integral to the configuration
- characteristics (materials, densities, geometries, tolerances, uncertainties) of any encapsulation used to provide confinement and structural support during handling and when within the storage container

(SL) Verify that the applicant has demonstrated that HLW and reactor-related GTCC wastes containing fissile material will remain subcritical. In general, reactor-related GTCC waste containers are not expected to contain significant amounts of fissile material. The most likely types of reactor-related GTCC waste that may contain fissile material are fission chambers, some neutron sources, filters, and ion-exchange resins. Verify that the applicant has addressed these potential sources of fissile material (if present) and has demonstrated that their quantity is insignificant. For those HLW and reactor-related GTCC waste forms for which criticality is a concern, verify that the applicant has demonstrated that the most reactive configurations of the wastes have been analyzed and that their k_{eff} values remain below 0.95. Also verify that the analysis includes adequate benchmarking, consistent with the guidance in Section 7.5.4.3 of this SRP but appropriately applied for these wastes. In general, for reactor-related GTCC wastes, it is not necessary to perform independent confirmatory analyses.

(SL) Also verify that the applicant has demonstrated that storage of GTCC waste will not adversely affect the safe storage of SNF and HLW at the DSF. In general, containers of GTCC waste located with SNF and HLW storage containers at an ISFSI or MRS are not expected to increase the reactivity of the SNF and HLW storage containers.

7.5.7 Supplemental Information

Ensure that the SAR includes all supportive information or documentation. This may include, but not be limited to, justification of assumptions or analytical procedures, test results, photographs, computer program descriptions, input/output, and applicable pages from referenced documents. In addition, confirm that the SAR includes a list of fuel designs with the acceptable parametric limits and the maximum enrichments for which the criticality analysis is valid. Request any additional information needed to complete the review.

7.6 Evaluation Findings

The NRC reviewer should prepare evaluation findings upon satisfaction of the applicable regulatory requirements in Section 7.4 of this SRP. If the documentation submitted with the application fully supports positive findings for each of the regulatory requirements, the statements of findings should be similar to the following:

F7.1 The applicant has described the SSCs important to criticality safety in sufficient detail in Chapters ______ of the SAR to enable an evaluation of their effectiveness in accordance with [for SL use: 10 CFR 72.24(b) and 10 CFR 72.24(c); for CoC use: 10 CFR 72.236(b)].

F7.2 **(CoC)** The applicant has designed the ____ DSS, including its transfer cask for canister-based systems, to be subcritical under all credible conditions in accordance with 10 CFR 72.124(a) and 10 CFR 72.236(c).

(SL) The applicant has designed the _______ DSF's SSCs involved in the loading, unloading, packaging, handling, transfer, and storage of the SNF at the DSF to be subcritical under all credible conditions in accordance with 10 CFR 72.124(a).

F7.3 The applicant based the criticality design of the [DSS or DSF SSCs] on favorable geometry, fixed neutron poisons, and soluble poisons[1] [as applicable]. The applicant's evaluation of the fixed neutron poisons in the storage container has shown that the fixed neutron poisons will remain effective for the storage term requested in the [CoC or specific license] application and there is no credible way for the fixed neutron poisons to significantly degrade during the requested storage term in the [CoC or specific license] application. Therefore, there is no need to provide a positive means to verify their continued efficacy as required in 10 CFR 72.124(b).

[For specific license applications for a DSF, the design and operations of which include a pool or other SSCs that use fixed neutron poisons, use the following finding for the applicant's evaluation of these fixed poisons: The applicant has provided an adequate means to verify, during the licensed storage term, the continued efficacy of the fixed neutron poisons in the [list applicable DSF SSCs] as required in 10 CFR 72.124(b).]

F7.4 The applicant's analysis and evaluation of the criticality design and performance of the [DSS or DSF SSCs] have demonstrated that the [DSS

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[1] Soluble poisons may be relied upon for wet loading or unloading. For DSSs and for DSFs that are co-located at 10 CFR Part 50- or 10 CFR Part 52-licensed reactor facilities and share the pool, this would be soluble poisons in the 10 CFR Part 50- or 10 CFR Part 52-licensed facility's SNF pool. For DSF designs and operations that include a pool as part of the specific license application, this would be soluble poisons in the DSF's pool.

or DSF] will enable the storage of SNF for the term requested in the [CoC or specific license] application (for SL: 10 CFR 72.24(c); for CoC: 10 CFR 72.236(g)).

F7.5 **(SL)** The design and operations of the proposed DSF and the characteristics of the materials to be stored at the proposed DSF provide reasonable assurance that the activities authorized by the specific license can be conducted without endangering the health and safety of the public, in compliance with 10 CFR 72.40(a)(13). This includes the use of necessary criticality monitoring systems as required in 10 CFR 72.124(c), and the necessary design and operations parameters to ensure HLW or reactor-related GTCC waste to be stored at the DSF and that contains fissile materials remains subcritical under all credible conditions.

F7.6 The design and proposed [use of the DSS/operations of the DSF], including SSCs involved in the handling, packaging, transfer, and storage of the radioactive materials to be stored, acceptably ensure that the materials will remain subcritical and that, before a nuclear criticality accident is possible, at least two unlikely, independent, and concurrent or sequential changes must occur in the conditions essential to nuclear criticality safety. The applicant's analyses in the SAR and confirmatory analysis by the NRC adequately show that acceptable margins of safety will be maintained in the nuclear criticality parameters commensurate with uncertainties in the data and methods used in calculations, and demonstrate safety for the handling, packaging, transfer, and storage conditions and in the nature of the immediate environment under accident conditions in compliance with 10 CFR 72.124(a) [and (for a CoC) 10 CFR 72.236(c)].

F7.7 The proposed [CoC or license] conditions, including the technical specifications, include those items necessary to ensure nuclear criticality safety in the design, fabrication, construction, and operation of the [DSS or DSF] SSCs [(*for CoC*) consistent with what is considered necessary to ensure compliance with 10 CFR 72.236(a), 72.236(b), and 72.236(c); (*for SL*) in accordance with the requirements in 10 CFR 72.24(g) and 10 CFR 72.44(c)].

F7.8 The SAR provides specifications of the [(*for CoC*) spent fuel contents to be stored in the [DSS designation]; (*for SL*) the materials to be stored at the [DSF designation]] in sufficient detail that adequately defines the allowed [contents/materials] and allows evaluation of the [DSS or DSF designation] nuclear criticality safety design for the proposed [contents/materials]. The SAR includes analyses that are adequately bounding for the proposed [contents'/materials'] specifications. (CoC: 10 CFR 72.236(a); SL: 10 CFR 72.24(c))

F7.9 **(CoC)** The applicant has designed the ____ DSS, including its transfer cask for canister-based systems, for criticality safety purposes, to be compatible with wet and dry loading and unloading facilities and, to the extent practicable, removal of the stored spent fuel from the site and transportation in accordance with 10 CFR 72.236(h) and 10 CFR 72.236(m).

The reviewer should provide a summary statement similar to the following:

> The staff concludes that the criticality design features for the [DSS or DSF designation] are in compliance with 10 CFR Part 72, as exempted [if applicable], and that the applicable design and acceptance criteria have been satisfied. The evaluation of the criticality design provides reasonable assurance that the [DSS or DSF designation] will allow safe storage of SNF [and HLW and reactor-related GTCC waste, as applicable for the DSF]. This finding is reached on the basis of a review that considered the regulation itself, appropriate regulatory guides, applicable codes and standards, and accepted engineering practices.

7.7 <u>References</u>

10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities."

10 CFR Part 52, "Licenses, Certifications, and Approvals for Nuclear Power Plants."

10 CFR Part 72, "Licensing Requirements for the Independent Storage of Spent Nuclear Fuel, High-Level Radioactive Waste, and Reactor-Related Greater than Class C Waste."

American National Standards Institute (ANSI)/American Nuclear Society (ANS) 8.1-1998 (Reaffirmed 2007), "Nuclear Criticality Safety in Operations with Fissionable Materials Outside Reactors," American Nuclear Society, La Grange Park, Illinois.

Regulatory Guide 3.71, "Nuclear Criticality Safety Standards for Fuels and Materials Facilities."

Information Notice No. 91-26, "Potential Nonconservative Errors in the Working Format Hansen-Roach Cross-Section Set provided with the KENO and Scale Codes," U.S. Nuclear Regulatory Commission, April 2, 1991.

"International Handbook of Evaluated Criticality Safety Benchmark Experiments," Nuclear Science Committee, Nuclear Energy Agency, updated and published annually, https://www.oecd-nea.org/science/wpncs/icsbep/handbook.html.

MCNP5, "MCNP – A General Monte Carlo N-Particle Transport Code, Version 5; Volume II: User's Guide," LA-CP-03-0245, Los Alamos National Laboratory, April 2003.

NUREG-1745, "Standard Format and Content for Technical Specifications for 10 CFR Part 72 Cask Certificates of Compliance," June 2001 (Agencywide Documents Access and Management System Accession No. ML011940387).

NUREG/CR-6328, "Adequacy of the 123-Group Cross-Section Library for Criticality Analyses of Water-Moderated Uranium Systems," ORNL/TM-12970, Oak Ridge National Laboratory, June 1995.

NUREG/CR-6361, “Criticality Benchmark Guide for Light-Water-Reactor Fuel in Transportation and Storage Packages,” ORNL/TM-13211, Oak Ridge National Laboratory, March 1997.

NUREG/CR-6747, “Computational Benchmark for Estimation of Reactivity Margin from Fission Products and Minor Actinides in PWR Burnup Credit,” ORNL/TM-2000/306, Oak Ridge National Laboratory, October 2001.

NUREG/CR-6801, “Recommendations for Addressing Axial Burnup in PWR Burnup Credit Analyses,” ORNL/TM-2001/273, Oak Ridge National Laboratory, March 2003.

NUREG/CR-6811, “Strategies for Application of Isotopic Uncertainties in Burnup Credit,” ORNL/TM-2001/257, Oak Ridge National Laboratory, June 2003.

NUREG/CR-6979, “Evaluation of the French Haut Taux de Combustion (HTC) Critical Experiment Data,” ORNL/TM-2007/083, Oak Ridge National Laboratory, September 2008.

NUREG/CR-6955, “Criticality Analysis of Assembly Misload in a PWR Burnup Credit Cask,” ORNL/TM-2004/52, Oak Ridge National Laboratory, January 2008.

NUREG/CR-6998, “Review of Information for Spent Nuclear Fuel Burnup Confirmation,” ORNL/TM-2007/229, Oak Ridge National Laboratory, December 2009.

NUREG/CR-7108, “An Approach for Validating Actinide and Fission Product Burnup Credit Criticality Safety Analyses—Isotopic Composition Predictions,” ORNL/TM-2011/509, Oak Ridge National Laboratory, April 2012.

NUREG/CR-7109, “An Approach for Validating Actinide and Fission Product Burnup Credit Criticality Safety Analyses—Criticality (k_{eff}) Predictions,” ORNL/TM-2011/514, Oak Ridge National Laboratory, April 2012.

NUREG/CR-7194, “Technical Basis for Peak Reactivity Burnup Credit for BWR Spent Nuclear Fuel in Storage and Transportation Systems,” ORNL/TM-2014/240, Oak Ridge National Laboratory, April 2015.

NUREG/CR-7205, “Bias Estimates Used in Lieu of Validation of Fission Products and Minor Actinides in MCNP K_{eff} Calculations for PWR Burnup Credit Casks,” ORNL/TM-2012/544, Oak Ridge National Laboratory, September 2015.

Oak Ridge National Laboratory, “SCALE: A Comprehensive Modeling and Simulation Suite for Nuclear Safety Analysis and Design,” ORNL/TM-2005/39, Version 6.1, June 2011. Available as CCC-785 from the Radiation Safety Information Computational Center at Oak Ridge National Laboratory, https://rsicc.ornl.gov/Catalog.aspx?c=CCC.

RW-859, “Nuclear Fuel Data Survey,” Energy Information Administration, https://www.eia.gov/nuclear/spent_fuel/.

UCID-21830, “Determination and Application of Bias Values in the Criticality Evaluation of Storage Cask Designs,” W.R. Lloyd, Lawrence Livermore National Laboratory, January 1990.

YAEC-1937, “Axial Burnup Profile Database for Pressurized-Water Reactors,” Yankee Atomic Electric Company, May 1997. Available as Data Package DLC-201, PWR-AXBUPRO-SNL, from the Radiation Safety Information Computational Center at Oak Ridge National Laboratory, https://rsicc.ornl.gov/Catalog.aspx?c=DLC.

APPENDIX 7A TECHNICAL RECOMMENDATIONS FOR THE CRITICALITY SAFETY REVIEW OF PRESSURIZED-WATER REACTOR TRANSPORTATION PACKAGES AND STORAGE CASKS THAT USE BURNUP CREDIT

7A.1 Introduction

The overall reactivity decrease of nuclear fuel irradiated in light water reactors is from the combined effect of the net reduction of fissile nuclides and the production of parasitic neutron absorbing nuclides (non-fissile actinides and fission products). Burnup credit refers to accounting for partial or full reduction of SNF reactivity caused by irradiation. Section 7.5.5 of this standard review plan (SRP) provides guidance to the U.S. Nuclear Regulatory Commission (NRC) staff for its use in the review of SNF container designs that seek burnup credit. This appendix provides the technical bases for the burnup credit recommendations for dry storage provided in the SRP and for transportation; thus, the appendix discusses both storage and transportation. As noted in Section 7.5.5, these recommendations and their technical bases are based on dry storage system (DSS)-type storage container designs, which are commonly referred to as casks or storage systems in this appendix. Application of the recommendations to other SNF storage container designs (in specific license applications for dry storage facilities (DSFs)) should involve consideration of the differences between container designs that are like DSSs and those that are not and the applicability of the recommendations' technical bases to non-DSS-like containers. This is also true for application of the recommendations, in specific license applications, to criticality analyses for SNF in other relevant DSF SSCs (e.g., a pool that is part of the DSF design and operations).

Historically, criticality safety analyses for transportation and dry cask storage of SNF assumed the fuel contents to be unirradiated (i.e., "fresh" fuel). In 2002, the NRC Spent Fuel Project Office (SFPO) issued Interim Staff Guidance-8 (ISG-8), "Burnup Credit in the Criticality Safety Analyses of PWR Spent Fuel in Transport and Storage Casks," Revision 2 to provide recommendations for the use of actinide-only burnup credit (i.e., burnup credit using only major actinide nuclides) in storage and transport of pressurized-water reactor (PWR) SNF. Based on the data available for burnup credit depletion and criticality computer code validation at the time ISG-8, Revision 2, was published, SFPO staff recommended actinide-only credit. Additionally, the staff recommended that a measurement be performed to confirm the reactor record burnup value for SNF assemblies to be stored or transported in cask or package designs that credit burnup in the criticality analysis.

Since ISG-8, Revision 2, was published, significant progress has been made in research on the technical and implementation aspects of burnup credit, with the support of the NRC Division of Spent Fuel Storage and Transportation (SFST, formerly SFPO) by the NRC Office of Nuclear Regulatory Research (RES) and its contractors at Oak Ridge National Laboratory (ORNL). This appendix summarizes the findings of a number of reports and papers published as part of the research program directed by RES over the last several years. It is recommended that the staff read the referenced reports and papers to understand the detailed evaluation of specific burnup credit parameters discussed in this appendix. A comprehensive bibliography of burnup credit-related technical reports and papers is provided at http://www.ornl.gov/sci/nsed/rnsd/pubs_burnup.shtml.

7A.2 General Approach in Safety Analysis

Criticality safety analyses of SNF storage or transportation systems involve a great deal of complexity in both the computer modeling of the system, as well as the necessary fuel information. The assumption of unirradiated fuel at maximum initial enrichment provides a straightforward approach for the criticality safety analysis of a SNF dry storage or transportation system. This approach is conservative in terms of criticality safety and limits the system capacity. In comparison to the fresh fuel assumption, performing criticality safety analyses for SNF systems that credit burnup require the following:

- additional information and assumptions for input to the analysis
- additional analyses to obtain the SNF compositions
- additional validation efforts for the depletion and decay software
- enhanced validation to address the additional nuclides in the criticality analyses
- verification that the fuel assembly to be loaded meets the minimum burnup requirements made before loading the system

The use of burnup credit for SNF storage casks and transportation packages provides for increased fuel capacities and higher limits on allowable initial enrichments for such systems. Applications for PWR SNF storage cask and transportation package certificates of compliance (CoCs) have generally shifted to high-capacity designs (i.e., 32 fuel assemblies or greater) in the past 15 years. In order to fit this many assemblies in a similarly sized SNF system, applicants have removed flux traps present in lower-capacity designs (i.e., 24 fuel assemblies or less), and replaced them with single neutron absorber plates between assemblies. Flux traps consist of two neutron absorber plates separated by a water region, with the water serving to slow neutrons down for more effective absorption. Single neutron absorber plates are less effective absorbers than flux trap designs, and result in a system that cannot be shown to be subcritical in unborated water without the use of some level of burnup credit.

An important outcome from a burnup credit criticality safety analysis is a SNF loading curve, showing the minimum burnup required for loading as a function of initial enrichment and cooling time. For a given system loading of SNF, the effective neutron multiplication factor (k_{eff}) will increase with higher initial enrichments, decrease with increases in burnup, and decrease with cooling time from 1 year to approximately 100 years. Information that should be considered in specifying the technical limits for fuel acceptable for loading includes fuel design, initial enrichment, burnup, cooling time, and reactor conditions under which the fuel is irradiated. Thus, depending on the assumptions and approach used in the safety analysis and the limiting k_{eff} criterion, a loading curve or set of loading curves can be generated to define the boundaries between acceptable and unacceptable SNF specifications for system loading.

The recommendations in Section 7.5.5 of this SRP include the following:

- general information on limits for the licensing basis
- recommended assumptions regarding reactor operating conditions
- guidance on code validation with respect to the isotopic depletion evaluation

- guidance on code validation with respect to the k_{eff} evaluation
- guidance on preparation of loading curves and the process for assigning a burnup loading value to an assembly

A criticality safety analysis that uses burnup credit should consider each of these five areas.

The five recommendations listed above were developed with intact fuel as the basis. An extension to fuel that is not intact may be warranted if the applicant can demonstrate that any additional uncertainties associated with the irradiation history and structural integrity (both during and subsequent to irradiation) of the fuel assembly have been addressed. In particular, a model that bounds the uncertainties associated with the allowed fuel inventory and fuel configuration in the system should be applied. Such a model should include the selection of appropriate burnup distributions and any potential rearrangement of fuel that is not intact during normal and accident conditions. The applicant should also apply each of the recommendations provided in this review guidance and justify any exceptions taken because of the nature of the fuel (e.g., the use of an axial profile that is not consistent with the recommendation). Section 8.5.15.1 of this SRP provides guidance for classifying the condition of the fuel (e.g., damaged, intact) for SNF storage and transportation.

The validation methodologies presented in Sections 7A.5 and 7A.6 of this appendix were performed for a representative cask model, known as the generic burnup credit cask (GBC)-32, described in NUREG/CR-6747, "Computational Benchmark for Estimation of Reactivity Margin from Fission Products and Minor Actinides in PWR Burnup Credit." As will be discussed later in this appendix, in order to directly use bias and bias uncertainty numbers developed in NUREG/CR-7108, "An Approach for Validating Actinide and Fission Product Burnup Credit Criticality Safety Analyses—Isotopic Composition Predictions," and NUREG/CR-7109, "An Approach for Validating Actinide and Fission Product Burnup Credit Criticality Safety Analyses—Criticality (k_{eff}) Predictions," applicants must use the same isotopic depletion and criticality code and nuclear data as were used in the isotopic depletion and criticality validation performed in those reports. Additionally, applicants must demonstrate that their SNF storage or transportation system design is similar to the GBC-32 used to develop the validation methodologies in NUREG/CR-7108 and NUREG/CR-7109. This demonstration should consist of a comparison of system materials and geometry, including neutron absorber material and dimensions, assembly spacing, and reflector materials and dimensions. This demonstration should also include a comparison of neutronic characteristics such as hydrogen-to-fissile atom ratios (H/X), energy of average neutron lethargy causing fission (EALF), neutron spectra, and neutron reaction rates. Applicability of the validation methodology to systems with characteristics that deviate substantially from those for the GBC-32 should be justified. Sensitivity and uncertainty analysis tools, such as those provided in the SCALE code system, can provide a quantitative comparison of the GBC-32 to the application of interest.

The recommendations provided in this review guidance were developed with PWR fuel as the basis. Boiling-water reactor (BWR) burnup credit has not typically been sought by dry storage and transportation applicants because of the complexity of the fuel and irradiation parameters, the lack of code validation data to support burnup credit, and a general lack of need for such credit in existing designs. The NRC has initiated a research project to obtain the technical basis for BWR burnup credit. BWR fuel assemblies typically have neutron absorbing material, typically gadolinium oxide, mixed in with the uranium oxide of the fuel pellets in some rods. This neutron absorber depletes more rapidly than the fuel during the initial parts of its irradiation, which causes the fuel assembly reactivity to increase and reach a maximum value at an assembly average burnup typically less than 20 gigawatt days per metric ton of uranium (GWd/MTU). Then reactivity decreases for the remainder of fuel assembly irradiation. Criticality analyses of BWR spent fuel pools typically employ what are known as "peak reactivity" methods to account for this behavior. NUREG/CR-7194, "Technical Basis for Peak Reactivity Burnup Credit for BWR Spent Nuclear Fuel in Storage and Transportation Systems," reviews several existing peak reactivity methods, and demonstrates that a conservative set of analysis conditions can be identified and implemented to allow criticality safety analysis of BWR spent fuel assemblies at peak reactivity in storage or transportation systems. Consult NUREG/CR-7194 if the applicant uses peak reactivity BWR burnup credit methods in its criticality analysis.

Credit for BWR burnup beyond peak reactivity is not addressed in this SRP, and is currently being evaluated by an NRC research program to investigate methods for conservatively including such credit in a BWR criticality analysis for SNF storage systems. The NRC does not recommend burnup credit beyond peak reactivity at this time. Conservative analyses of BWR burnup credit beyond peak reactivity should be considered on a case-by-case basis, consulting the latest research results in this area (i.e., NRC letter reports, NUREG/CRs).

The remainder of this appendix discusses recommendations in each of the five burnup credit areas and provides technical information and references that should be considered in the review of the safety analysis report (SAR).

7A.3 Limits for Licensing Basis (Chapter 7, Section 7.5.5.1 of the SRP)

Available validation data support actinide-only and actinide and fission product burnup credit for uranium dioxide (UO_2) fuel enriched up to 5.0 weight percent uranium-235, that is irradiated in a PWR to an assembly-average burnup value up to 60 GWd/MTU and cooled out-of-reactor between 1 and 40 years.

7A.3.1 Nuclides of Importance

Several studies have been performed to identify the nuclides that have the most significant effect on the calculated value of k_{eff} as a function of burnup and cooling time. These results are summarized in NUREG/CR-6665, "Review and Prioritization of Technical Issues Related to Burnup Credit for LWR Fuel." This report concludes that the actinides and fission products listed in Tables 7A-1 and 7A-2 are candidates for inclusion in burnup credit analyses for storage and transportation systems, based on their relative reactivity worth at the cooling times of interest. The relative reactivity worth of the nuclides will vary somewhat with fuel design, initial enrichment, and cooling time, but the important nuclides (fissile nuclides and select non-fissile absorbers) remain the same and have been substantiated by numerous independent studies. These nuclides have the largest impact on k_{eff}, and there is a sufficient quantity of applicable experimental data available for validation of the analysis methods, as Sections 7A.5 and 7A.6 of this appendix discuss. Accurate prediction of the concentrations for the nuclides in Tables 7A-1 and 7A-2 requires that the depletion and decay calculations include nuclides beyond those listed in the tables. Additional actinides and fission products are needed to assure the transmutation chains and decay chains are accurately handled. Methods are also needed to accurately simulate the influence of the fission product compositions on the neutron spectrum, which in turn impacts the burnup-dependent cross sections. To accurately predict the reactivity effect of fission products, explicit representation of the important fission product transmutation and decay chains is needed to obtain the individual fission product compositions.

Table 7A-1 Recommended Set of Nuclides for Actinide Only Burnup Credit

^{234}U	^{235}U	^{238}U
^{238}Pu	^{239}Pu	^{240}Pu
^{241}Pu	^{242}Pu	^{241}Am

Table 7A-2 Recommended Set of Additional Nuclides for Actinide and Fission Product Burnup Credit

^{95}Mo	^{99}Tc	^{101}Ru	^{103}Rh
^{109}Ag	^{133}Cs	^{147}Sm	^{149}Sm
^{150}Sm	^{151}Sm	^{152}Sm	^{143}Nd
^{145}Nd	^{151}Eu	^{153}Eu	^{155}Gd
^{236}U	^{243}Am	^{237}Np	

Applicants attempting to credit neutron-absorbing isotopes other than those listed in these tables should ensure that such isotopes are nonvolatile, nongaseous, and relatively stable, and applicants should provide analyses to determine the additional depletion and criticality code bias and bias uncertainty associated with these isotopes. These analyses should be accompanied by additional relevant critical experiment and radiochemical assay (RCA) data, to the extent practicable, or include sufficient penalties to account for the lack of such data.

7A.3.2 Burnup and Enrichment Limits

NUREG/CR-7108 demonstrates that the range of existing RCA data that are readily available for validation extends up to 60 GWd/MTU and 4.657 weight percent uranium-235 initial enrichment. Though limited RCA data are available above 50 GWd/MTU, it is the staff's judgment that credit can reasonably be extended up to 60 GWd/MTU. Credit should not be extended to assembly-average burnups beyond this level, though local burnups can be higher. Fuel with an assembly average burnup greater than 60 GWd/MTU can be loaded into a burnup credit system, but credit should only be taken for the reactivity reduction up to 60 GWd/MTU. Additionally, while the enrichment range covered by the available assay data has increased, it has not increased enough to warrant a change with regard to the maximum initial enrichment that can be considered in a burnup credit analysis; thus, the initial enrichment limit for the licensing basis remains at 5.0 weight percent uranium-235.

7A.3.3 Cooling Time

Figure 7A-1 illustrates the expected reactivity behavior for SNF in a hypothetical GBC-32 system for an analysis using major actinide concentrations and various actinide and fission product concentrations in the calculation of k_{eff}. The fact that reactivity begins to rise around 100 years after discharge means the timeframe for interim SNF storage should be considered in the evaluation of acceptable cooling times. The curve indicates that the reactivity of the fuel at 40 years is about the same as that of fuel cooled to 200 years. The Commission has recently instructed staff to review the regulatory programs for SNF storage and transportation, considering extended storage beyond 120 years (NRC 2010). In light of the increasingly likely scenario of extended dry storage of SNF, the CoC for a SNF transportation package or the CoC or license for dry storage may require an additional condition with regard to the applicability of the credited burnup of the SNF contents. The condition would be dependent upon the type of credit taken and the post irradiation decay time credited in the analysis. For example, crediting of 40 years would result in a CoC or license condition limiting the applicability of the credited burnup to 160 years after fuel discharge. Note that approval of a cooling time longer than 5 years for burnup credit in dry storage or transportation systems does not automatically guarantee acceptance for disposal without repackaging. NUREG/CR-6781, "Recommendations on the Credit for Cooling Time in PWR Burnup Credit Analyses," provides a comprehensive study of the effect of cooling time on burnup credit for various cask designs and SNF compositions.

7A.3.4 Summary

The acceptance criteria for burnup credit are based on the characteristics of SNF discharged to date, the parameter ranges considered in the majority of technical investigations, and the experimental data available to support development of a calculational bias and bias uncertainty. As indicated, a safety analysis that uses parameter values outside those recommended by the SRP should (1) demonstrate that the measurement or experimental data necessary for proper code validation have been included, and (2) provide adequate justification that the analysis assumptions or the associated bias and bias uncertainty have been established in such a fashion as to bound the potential impacts of limited measurement or experimental data. Even within the recommended range of parameter values, the reviewer should exercise care in assessing whether the analytic methods and assumptions used are appropriate, especially near the ends of the range.

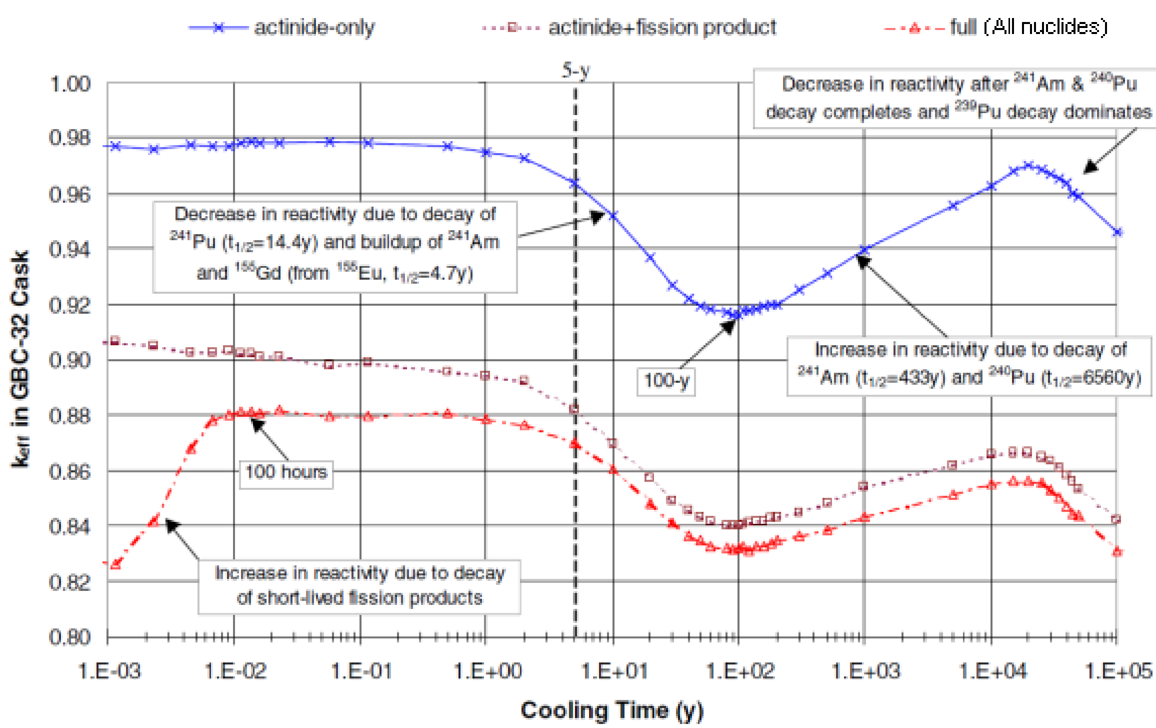

Figure 7A-1 Reactivity Behavior in The GBC 32 Cask as a Function of Cooling Time for Fuel with 4.0 Weight Percent Uranium-235 Initial Enrichment and 40 Gwd/MTU Burnup (Source: NRC 2010)

7A.4 Licensing-Basis Model Assumptions (Chapter 7, Section 7.5.5.2 of the SRP)

The actinide and fission product compositions used to determine a value of k_{eff} for the licensing basis should be calculated using fuel design and reactor operating parameter values that encompass the range of design and operating conditions for the proposed contents. Note that the proposed contents may consist of the entire population of discharged PWR fuel assemblies, a specific design of PWR fuel assembly (e.g., W17 × 17 optimized fuel assembly (OFA)), or a smaller, specific population from a particular site. The calculation of the k_{eff} value should be performed using cask models, analysis assumptions, and code inputs that allow accurate representation of the physics in the system. The following provides a discussion of important parameters that should be addressed in depletion analyses and k_{eff} calculations in a burnup credit evaluation.

7A.4.1 Reactor Operating History and Parameter Values

Section 4.2 of NUREG/CR-6665 discusses the impacts of fuel temperature, moderator temperature and density, soluble boron concentration, specific power and operating history, and burnable absorbers on the k_{eff} of SNF in a cask.

As the assumed fuel temperature used in the depletion model increases, the k_{eff} for the SNF in the cask will increase. The k_{eff} will also increase with increases in either moderator temperature (lower density) or the soluble boron concentration. Analyses for both actinide-only and actinide-plus-fission product evaluations exhibit these trends in k_{eff}. Figures 7A-2 to 7A-4 provide examples of the Δk impact seen from differences in fuel temperature, moderator temperature, and soluble boron concentration. The system modeled to determine these results was an infinite array of storage cells, but similar results have been confirmed for finite, reflected systems. All of these increases are because of the parameter increase causing increased production of fissile plutonium nuclides and decreased uranium-235 utilization.

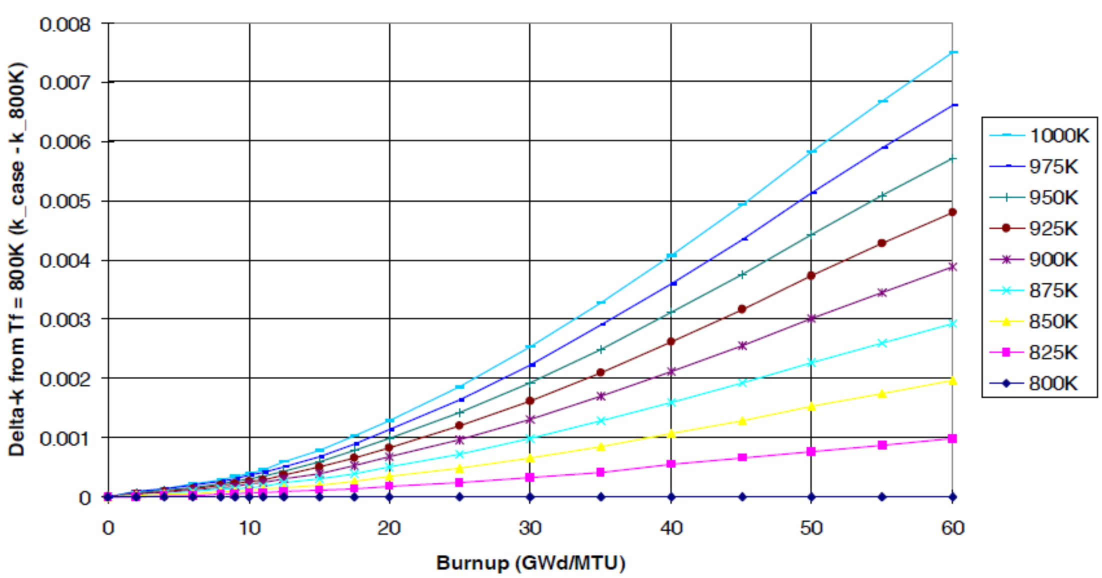

Figure 7A-2 Reactivity Effect of Fuel Temperature During Depletion on K_{inf} in an Array of Poisoned Storage Cells; Results Correspond to Fuel with 5.0 Weight Percent Initial Uranium-235 Enrichment (Source: Withee 2002)

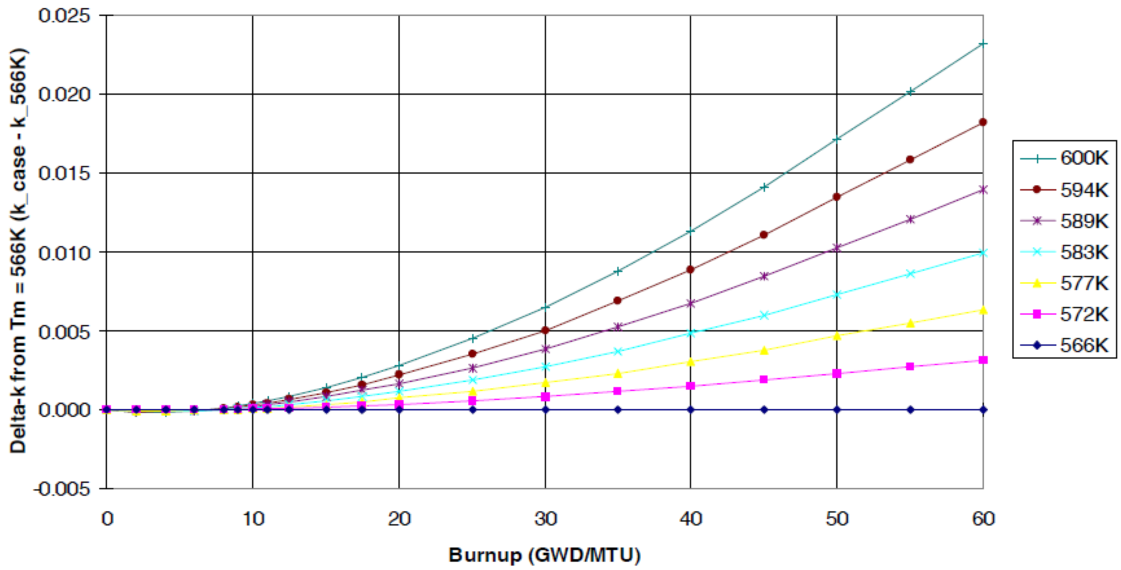

Figure 7A-3 Reactivity Effect of Moderator Temperature During Depletion on K_{inf} in an Array of Poisoned Storage Cells; Results Correspond to Fuel with 5.0 Weight Percent Initial Uranium-235 Enrichment (Source: Withee 2002)

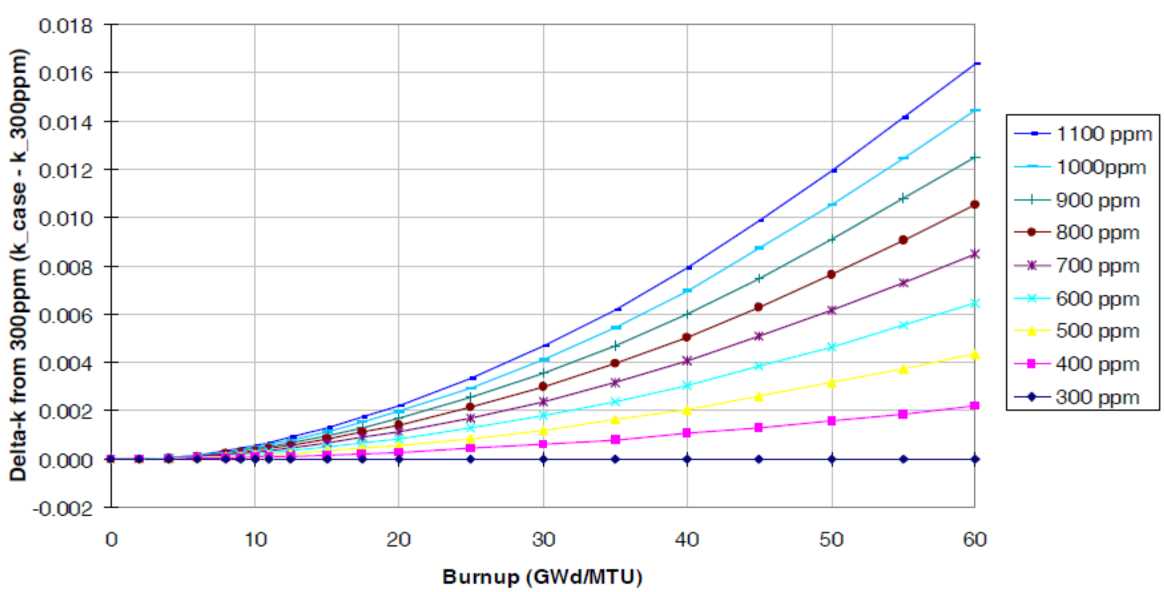

Figure 7A-4 Reactivity Effect of Soluble Boron Concentration During Depletion on K_{inf} in an Array of Poisoned Storage Cells; Results Correspond to Fuel with 5.0 Weight Percent Initial Uranium-235 Enrichment (Source: Withee 2002)

The impact of specific power and operating history is much more complex but has a very small impact on the cask k_{eff} value. Figures 7A-5 and 7A-6 show the variation of k_{inf} with specific power for various initial enrichment and burnup combinations, for actinide-only and actinide-plus-fission product burnup credit, respectively. Irradiation at higher specific power results in a slightly higher k_{eff} for actinide-only burnup credit, but the reverse is true for burnup credit that includes actinides and fission products (see Section 3.4.2.3 of DeHart 1996). Although the specific power at the end of irradiation is most important, the assumption of constant full-power is more straightforward and acceptable while having minimal impact on the k_{eff} value relative to other assumptions.

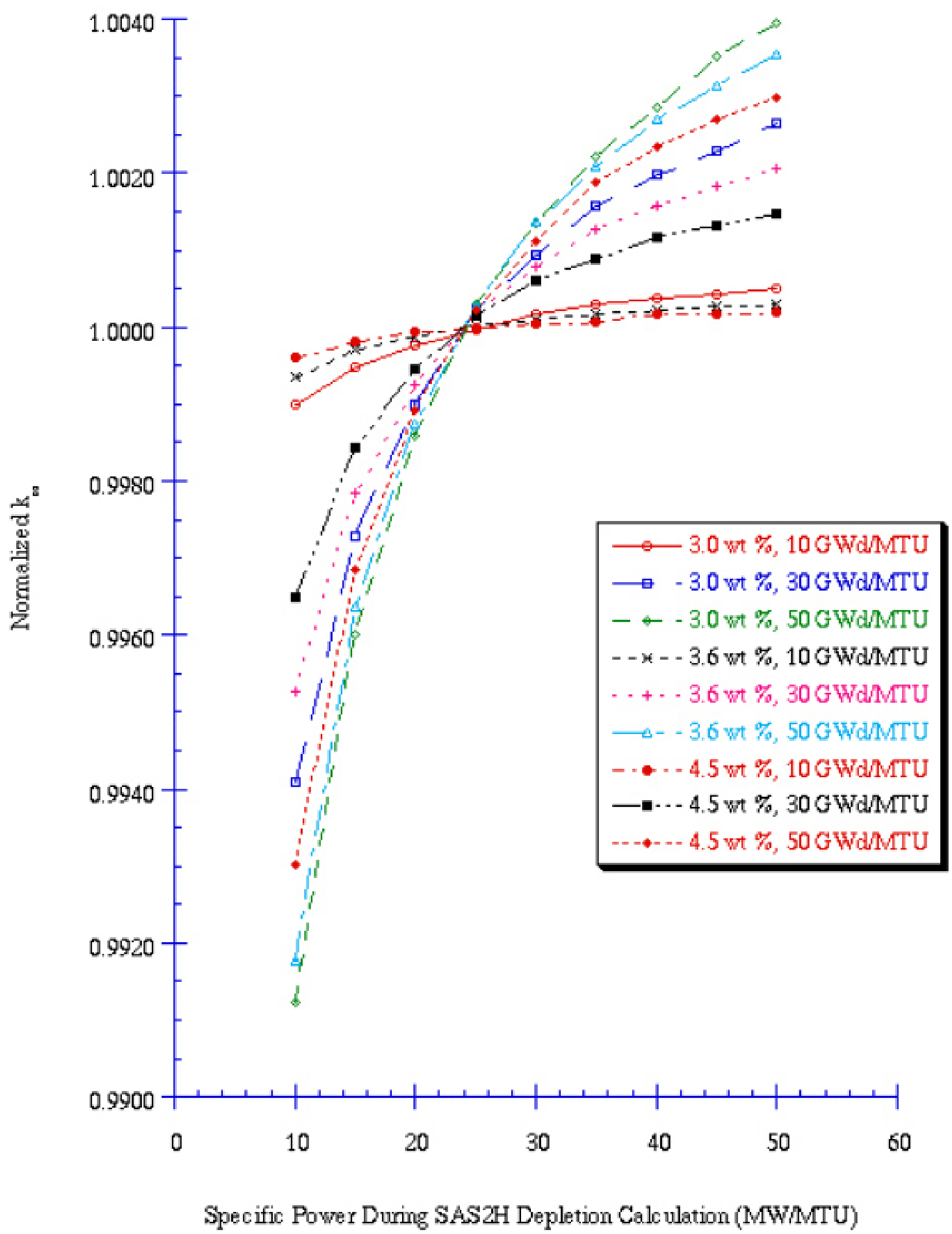

Figure 7A-5 Reactivity Effect of Specific Power During Depletion on k_{inf} in an Array of Fuel Pins (Actinides Only) (Source: DeHart 1996)

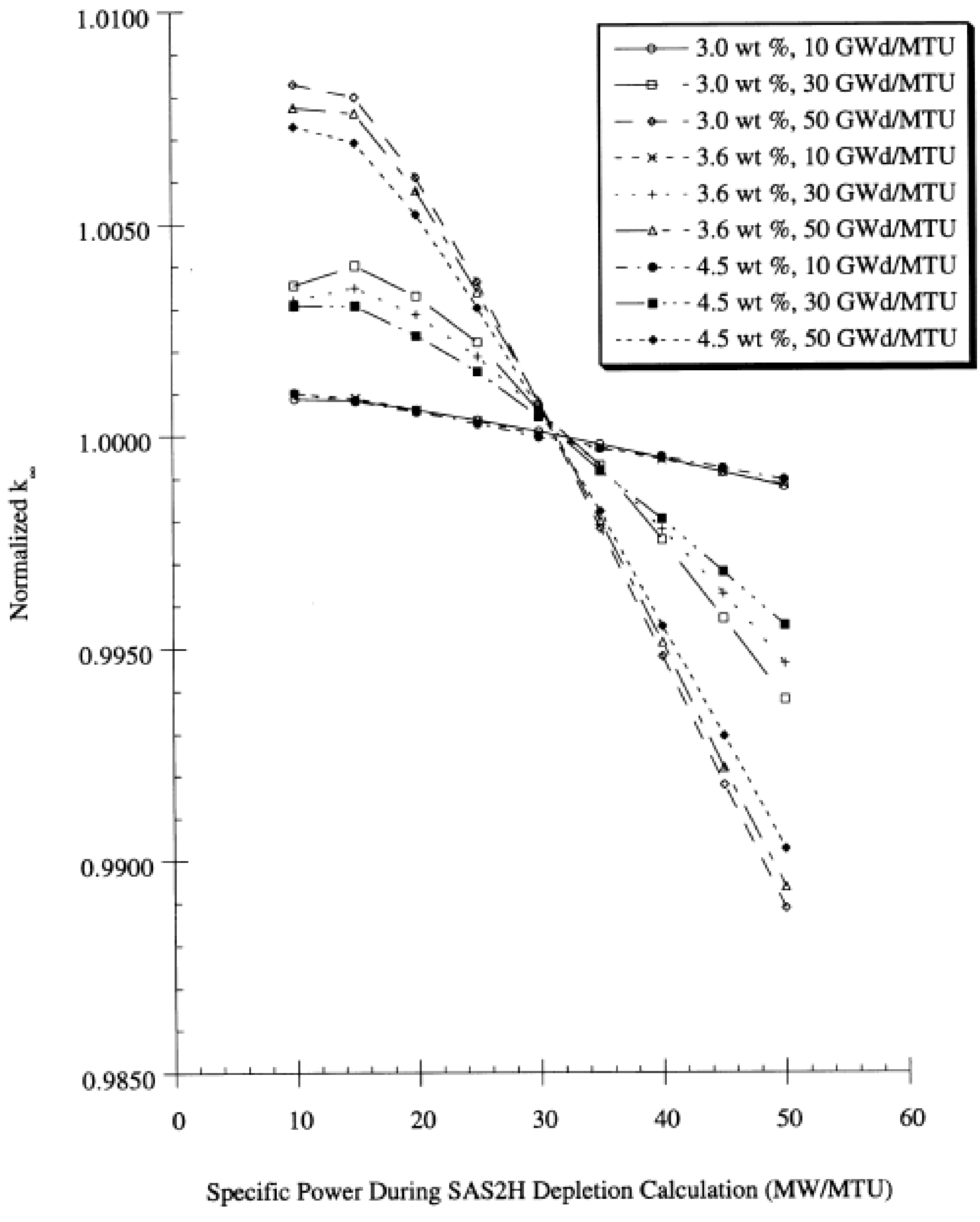

Figure 7A-6 Reactivity Effect of Specific Power During Depletion on K_{inf} in an Array of Fuel Pins (Actinides and Fission Products) (Source: DeHart 1996)

More detailed information on the impact of each parameter or phenomenon that should be assumed in the depletion model is provided in NUREG/CR-6665 and DeHart (1996). Each of the trends and impacts has been substantiated by independent studies. However, to model the irradiation of the fuel to produce bounding values for k_{eff} consistent with realistic reactor operating conditions, information is needed on the range of actual reactor conditions for the proposed SNF to be loaded in a cask. Loading limitations tied to the actual operating conditions will be needed unless the operating condition values assumed in the model can be justified as those that produce the maximum k_{eff} values for the anticipated SNF inventory. As illustrated by the case of specific power and operating history, the bounding conditions and appropriate limitations may differ for actinide-only burnup credit versus actinide-plus-fission product burnup credit, since the parameter impact may trend differently for these two types of burnup credit. Note that the sensitivity to variations in the depletion parameter assumptions differs for the two types of burnup credit, with actinide-plus-fission product burnup credit analyses exhibiting greater sensitivity for some parameters (see NUREG/CR-6800, "Assessment of Reactivity Margins and Loading Curves for PWR Burnup-Credit Cask Designs").

Also, the most reactive fuel design prior to irradiation will not necessarily have the highest reactivity after discharge from the reactor, and the most reactive fuel design may differ at various burnup levels. Thus, if various fuel designs are to be allowed in a particular cask design, parametric studies should be performed to demonstrate the most reactive SNF design for the range of burnup and enrichments considered in the safety analysis. Another option is to provide loading curves for each fuel assembly design and allow only one assembly type in each cask loading.

7A.4.2 Horizontal Burnup Profiles

Consideration of pin-by-pin burnups (and associated variations in SNF composition) does not appear to be necessary for analysis of the integral k_{eff} value in a SNF cask. To date, PWR cores have been managed such that the vast majority of assemblies experience a generally uniform burnup horizontally across the assembly during an operating cycle. However, assemblies on the periphery of the core may have a significant variation in horizontal burnup after a cycle of operation (see DOE/RW-0496, "Horizontal Burnup Gradient Datafile for PWR Assemblies"). In large storage or rail casks, the probability that underburned quadrants of multiple fuel assemblies will be oriented in such a way as to have a substantial impact on k_{eff} is not expected to be significant. However, for smaller systems, the effect can be significant. The safety evaluation should address the impact of horizontal burnup gradients such as found in DOE/RW-0496 on their system design or demonstrate that the assemblies to be loaded in the system will be verified to not have such gradients. One acceptable approach would be to determine the difference in k_{eff} for a system loaded with fuel having a horizontal burnup gradient and a system loaded with the same fuel having a uniform horizontal burnup (i.e., no gradient). The fuel with the gradient would be arranged so as to maximize the reactivity effect of the gradient. The reactivity difference between the two cases could then be applied to the remaining analyses.

7A.4.3 Axial Burnup Profiles

Considerable attention should be paid to the axial burnup profile(s) selected for use in the safety evaluation. A uniform axial profile is generally bounding at low burnups but is increasingly nonconservative at higher burnups because of the increasing relative worth of the fuel ends, as demonstrated in NUREG/CR-6801, "Recommendations for Addressing Axial Burnup in PWR Burnup Credit Analyses". Figure 7A-7 illustrates an example of this phenomenon for an actinide-only burnup credit analysis. As the figure shows, a uniform axial profile was conservative for that analysis at burnups less than about 20 GWd/MTU, but nonconservative at higher burnups. The burnup range at which this transition occurs will vary with fuel design and the type of burnup credit.

Section 7.5.5.2 of this SRP and this appendix indicate that any analysis should provide "an accurate representation of the physics" in the system. Thus, the applicant should select and model the axial burnup profile(s) in the analyses (including an appropriate number of axial material zones) that encompass the proposed contents and their range of potential k_{eff} values. The applicant should account for the fact that the axial effect will vary with burnup, cooling time, SNF nuclides used in the prediction of k_{eff}, and cask design. The reviewer should consider the range of profiles anticipated for the fuel to be loaded in the system.

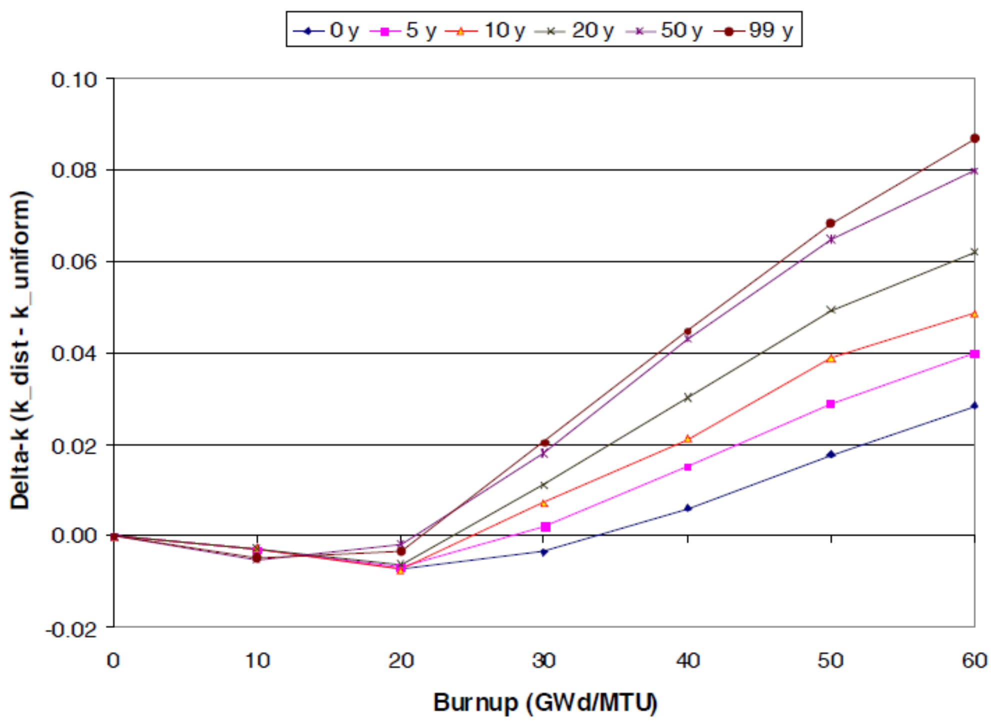

Figure 7A-7 Effect of Axial Burnup Distribution on K_{eff} in the GBC-32 Cask for Actinide-Only Burnup Credit and Various Cooling Times for Fuel with 4.0 Weight Percent Initial Enrichment (Source: Withee 2002)

The publicly available database of axial profiles in YAEC-1937, "Axial Burnup Profile Database for Pressurized Water Reactors," is recommended as an appropriate source for selecting axial burnup profiles that will encompass the SNF anticipated for loading in a burnup credit cask. While the database represents only 4 percent of the assemblies discharged through 1994, NUREG/CR-6801 indicates that it provides a representative sampling of discharged assemblies. This conclusion is reached on the basis of fuel vendor/ reactor design, types of operation (i.e., first cycles, out-in fuel management and low-leakage fuel management), burnup and enrichment ranges, use of burnable absorbers (including different absorber types), and exposure to control rods (CRs) (including axial power shaping rods (APSRs)). NUREG/CR-6801 also indicates that while the database has limited data for burnup values greater than 40 GWd/MTU and initial enrichments greater than 4.0 weight percent uranium-235, there is a high probability that the profiles resulting in the highest reactivity at intermediate burnup values will yield the highest reactivity at higher burnups. Thus, the existing database should be adequate for burnups beyond 40 GWd/MTU and initial enrichments above 4.0 weight percent uranium-235 if profiles are selected that include a margin for the potential added uncertainty in moving to the higher burnups and initial enrichments allowed per Section 7.5.5.1 of this SRP and Section 7A.3 of this appendix. Given the limited nature of the database, NUREG/CR-6801 includes an evaluation of the database's limiting profiles and the impacts of loading significantly more reactive assemblies in the place of assemblies with limiting profiles. NUREG/CR-6801 concludes that, based on the low consequence of the more reactive profiles, the nature of the database's limiting profiles, and their application to all assemblies in a cask, the database is adequate for obtaining bounding profiles for use in burnup credit analyses.

While the preceding discussion indicates that the database is an appropriate source of axial burnup profiles, the reviewer should ensure that profiles taken from the database are applied correctly. The application of the profiles in the database may not be appropriate for all assembly designs. This would include assemblies of different lengths than those evaluated in the database. While the database included some assemblies with axial blankets (natural or low enriched), these assemblies were not irradiated in a fully blanketed core (i.e., they were test assemblies). Thus, application of the database profiles to assemblies with axial blankets may also be inappropriate, as the impact of axial blankets has not been fully explored. However, it is generally conservative to assume fuel is not blanketed, using the enrichment of the non-blanketed axial zone and the limiting axial profile.

Other sources of axial burnup profiles may be appropriate to replace or supplement the database of YAEC-1937. The reviewer should assure that a description and evaluation of these other burnup profile sources similar to that demonstrated for the YAEC-1937 database in NUREG/CR-6801 has been performed. The reviewer should assure that the process used to obtain axial profiles included in the safety analysis has been described and that the profiles are justified as encompassing the realistic profiles for the entire burnup range over which they are applied. The process of selecting and justifying the appropriate bounding axial profile may be simplified and/or conservatism may be reduced if a measurement of the axial burnup profile is performed before or during the cask loading operation. The measurement should demonstrate that the actual assembly profile is equally or less reactive than that assumed in the safety evaluation.

7A.4.4 Burnable Absorbers

Assemblies exposed to fixed neutron absorbers (also referred to as integral burnable absorbers (IBAs)) and removable neutron absorbers (also referred to as burnable poison rod assemblies (BPRs)) can have higher k_{eff} values than assemblies that are not exposed. This is due to the hardening of the neutron spectrum and will lead to increased fissile plutonium nuclide production and reduced uranium-235 depletion. In addition, when removable neutron absorbers are inserted, the spectrum is further hardened because of the displacement of the moderator. NUREG/CR-6761, "Parametric Study of the Effect of Burnable Poison Rods for PWR Burnup Credit," and NUREG/CR-6760, "Study of the Effect of Integral Burnable Absorbers on PWR Burnup Credit," provide characterizations of the effects of burnable absorbers on SNF. The results of these studies indicate that a depletion analysis with a maximum realistic loading of BPRs (i.e., maximum neutron poison loading) and maximum realistic burnup for the exposure should provide an adequate bounding safety basis for fuel with or without BPRs. An evaluation relying on exposures to less than the maximum BPR loading or for less than the maximum burnup (for which credit is requested), or both, needs adequate justification for the selected values (e.g., provision of available data to support the value selection and/or indication of how administrative controls will prevent a misload of an assembly with higher exposure).

For IBAs, the results of these studies indicate that the impact on k_{eff} depends on the material type and the burnup level. Exposure to the maximum absorber loading was seen to be bounding for zirconium diboride-type IBAs (known as integral fuel burnable absorbers) at burnups above about 30 GWd/MTU. At lower burnups, neglecting the presence of the absorber was seen to be bounding. Neglecting the absorber in the case of IBAs that use erbia, gadolinia, and alumina-boron carbide was also bounding for all burnups investigated for these IBAs. Exposures to absorber types or materials not considered in the references supporting this appendix, whether fixed, removable, or a combination of the two, should be evaluated on a case-by-case basis.

7A.4.5 Control Rods

As with BPRs, CRs fully or partially inserted during reactor operation can harden the spectrum in the vicinity of the insertion and lead to increased production of fissile plutonium nuclides. In addition, CRs can alter the axial burnup profile. In either case the CR would have to be inserted for a significant fraction of the total irradiation time for these effects to be seen in terms of a positive Δk on the SNF cask. Domestic PWRs typically do not operate with CRs inserted, although the tips of the rods may rest right at the fuel ends. However, some older domestic reactors and certain foreign reactors may have used CRs in a more extensive fashion, such that the impact of CR insertion would be significant.

Based on the results of NUREG/CR-6759, "Parametric Study of the Effect of Control Rods for PWR Burnup Credit, U.S. Nuclear Regulatory Commission," and the fact that BPRs and CRs cannot be inserted in an assembly at the same time, the inclusion of BPRs in the assembly irradiation model should adequately account for the potential increase in k_{eff} that may occur for typical SNF exposures to CRs during irradiation. However, exposures to atypical CR insertions (e.g., full insertion for one full reactor operation cycle) may not be fully accounted for by inclusion of BPRs in the irradiation model, and assemblies irradiated under such operational conditions should be explicitly evaluated. Also, since the previously discussed axial burnup profile database (NUREG/CR-6800) includes a representative sampling of assemblies exposed to CRs and APSRs, the appropriate selection of a limiting axial profile(s) from that database would be expected to adequately encompass the potential impact for axial profile distortion caused by CRs and APSRs.

Exposures to CR or APSR insertions or materials not considered in the references supporting the guidance in Section 7.5.5 of this SRP and this appendix should be explicitly evaluated. This would also apply to exposures to flux suppressors (e.g., hafnium suppressor inserts) or similar hardware which affect reactivity. Safety analyses for exposures to these items should use assumptions (e.g., duration of exposure, cycle(s) of exposure) that provide an adequate bounding safety basis and include appropriate justification for those assumptions. Additionally, the axial burnup and power distributions in assemblies exposed to these devices may be unusual; thus, it may be necessary to use actual axial burnup shapes for those assemblies.

7A.4.6 Depletion Analysis Computational Model

For depletion analyses, computer codes that can track a large number of nuclides should be used in order to obtain an accurate estimate of the SNF nuclide concentration. Although certain nuclides that are typically tracked may not directly impact the concentrations of the nuclides in Tables 7A-1 and 7A-2, they can indirectly impact the production and depletion via their effect on the neutron spectrum. Tracking of a sufficiently large number of nuclides, the use of accurate nuclear data, and the prediction of burnup-dependent cross sections representative of the spatial region of interest are necessary for an accurate depletion analysis model.

Two-dimensional codes are routinely used together with axial segmentation of the fuel assembly in the criticality model to approximate axial variation in depletion. The two-dimensional flux calculations can capture the planar neutron flux distribution in each axial segment of a fuel assembly. The two-dimensional model is built to calculate the isotopic composition of the assembly at a series of burnup values, derived from the chosen axial burnup profile and the assembly average burnup. This approach is acceptable because it accounts for both the planar and axial flux variation to achieve a relatively accurate depletion simulation. Ideally, three-dimensional computer codes would be useful for fuel assembly depletion analyses to accurately simulate this phenomenon. However, three-dimensional depletion analysis codes are not recommended at the present time because of their current limitations.

Several two-dimensional neutron transport theory based codes are available, such as CASMO, HELIOS, and the SCALE TRITON sequence (DeHart 2009). The reviewer should be aware of the limitations of a particular code and version, such as those designed to use lumped cross sections for multiple nuclides. Such limitations may require additional justification of the code's utility for burnup credit criticality analyses. Review of depletion analyses should focus on the suitability and accuracy of the code and modeling of the fuel assembly depletion history.

Previously, because of the limited availability of accurate two-dimensional computer codes, most burnup credit calculations used one-dimensional depletion codes to determine SNF isotopic concentrations averaged over the assembly. With appropriate code benchmarking against assay measurements and appropriate treatment of the fuel assembly spatial heterogeneity (e.g., Dancoff factor correction, disadvantage factor correction (Duderstadt and Hamilton 1976)), one-dimensional physics models of PWR assembly designs can produce sufficiently accurate assembly average SNF compositions. However, in order to use a one-dimensional model, a cylindrical flux-weighted and geometry-equivalent supercell depletion model needs to be constructed to preserve the effective fuel assembly neutronics characteristics. Burnup-dependent cross sections are then generated using the flux-weighted and geometry-modified point-depletion model. This approach is sensitive to the accurate construction of the supercell materials and the approximation of the assembly geometry.

It is essential that the burnup-dependent cross sections are updated with sufficient frequency in the depletion analysis model and that the physics model used to update the cross sections is one that is representative of the assembly design and reactor operating history. As with analyses used to determine k_{eff}, the depletion analysis should be appropriately validated. The application analysis should use the same code and cross section library and the same, or similar, modeling options as were used in the depletion validation analysis. Issues associated with isotopic depletion code validation are discussed in greater detail in Section 7A.5 of this appendix.

7A.4.7 Models for Prediction of k_{eff}

The expectations regarding the codes and modeling assumptions to be used to determine k_{eff} of a dry storage cask are documented in this SRP as well as the following documents:

- NUREG/CR-5661, "Recommendations for Preparing the Criticality Safety Evaluation of Transportation Packages"
- NUREG/CR-6361, Criticality Benchmark Guide for Light-Water-Reactor Fuel in Transportation and Storage Packages"

Monte Carlo codes capable of three-dimensional solutions of the neutron transport equation are typically required for such applications. A loading of SNF, including specific combinations of assembly-average burnup, initial enrichment, and cooling time, should be used for each cask analysis. However, unlike unirradiated fuel, the variability of the burnup (and thus the isotopic concentrations) along the axial length is an important input assumption.

In particular, the burnup gradient will be large at the ends of the fuel regions. Thus, the cask model should include several fuel zones, each with isotopic concentrations representative of the average burnup across the zone. Burnup profile information from reactor operations is typically limited to 18–24 uniform axial regions. NUREG/CR-6801 has shown that subdividing the zones beyond that provided in the profile information (assuming at least 18 uniform axial zones) yields insignificant changes in the k_{eff} value for a cask.

In reality, the end regions of the fuel have the lowest burnup and provide the largest contribution to the reactivity of the system. Thus, the model boundary condition at the ends of the fuel will potentially be of greater importance than for uniform or fresh-fuel cases where the reactivity in the center of the fuel dominates reactivity. The end fitting regions above and below the fuel contain steel hardware with a significant quantity of void space (typically 50 percent or more) for potential water inleakage. The analyses in Appendix A to NUREG/CR-6801 demonstrate that both modeling the end regions as either 100 percent steel or full-density water provides a higher value of k_{eff} than a combination (homogenized mixture 50 percent water and 50 percent steel assumed) of the two. For the cask that was studied, the all-steel reflector provided a k_{eff} change of nearly 1 percent over that of full-density water. Although use of 100 percent steel is an extreme boundary condition (since water will always be present to some degree), the results indicate that the applicant should be attentive to the selection of a conservative boundary condition for the end regions of the fuel.

The large source of fissions distributed nonuniformly, because of the axial burnup profile, over a large source volume in a SNF cask, can cause difficulty in properly converging the analysis to the correct k_{eff} value. Problems performed in an international code comparison study (Blomquist et al. 2006) have demonstrated that results can vary based on user selection of input parameters crucial to proper convergence. Strategies that may be used in the calculations to accelerate the source convergence (e.g., starting particles preferentially at the more reactive end regions) should be justified and demonstrated to be effective.

An important issue in burnup credit criticality modeling is the need to verify that the correct SNF composition associated with the depletion and decay analysis is inserted in the correct spatial zone in the cask model. The data processing method to select and extract the desired nuclide concentrations from the depletion and decay analyses, and input them correctly to the various spatial zones of the criticality analysis, is not a trivial process that has the potential for error. The reviewer should verify the interface process, the computer code used to automate the data handling, or both. As with fresh fuel criticality analyses, the reviewer should verify that the criticality analyses for burnup credit are appropriately validated. In other words, the application analysis should use the same code and cross section library and the same, or similar, modeling options as were used in the criticality code validation. Issues associated with criticality code validation are discussed in greater detail in Section 7A.6 of this appendix.

7A.5 Code Validation—Isotopic Depletion (Chapter 7, Section 7.5.5.3 of the SRP)

An isotopic depletion code typically consists of three parts:

1. a library of nuclear reaction cross sections

2. a geometric and material representation of the fuel assembly as well as the reactor core configuration

3. an algorithm to predict the isotopic transmutation over time as the fuel assembly is irradiated in the reactor and decays after discharge

To assure the accuracy of the code and identify the biases and uncertainties associated with the algorithm, nuclear data, and modeling capability, the depletion code should be validated against measured data from RCA measurements of SNF samples.

Validation of the depletion analysis code serves two purposes. The first is to determine if the code is capable of accurately modeling the depletion environment of fuel assemblies for which burnup credit is taken. The second is to quantify the bias and bias uncertainty of the depletion code against the depletion parameters, fuel assembly design characteristics, initial enrichment, and cooling time.

In general, validation of the depletion code consists of the following steps:

1. select RCA sample data sets that are suitable for validation of the depletion code

2. build and run depletion models for SNF samples that are selected for depletion code validation

3. apply the bias and bias uncertainty of the depletion calculation to the criticality analysis code implicitly through the use of adjusted isotopic concentrations of the depletion model, or determine the bias and bias uncertainties associated with the fuel depletion analysis code in terms of Δk_{eff}, as discussed in NUREG/CR-7108

7A.5.1 Selection of Validation Data

Validation data consist of measurements of isotopic concentrations from destructive RCA samples of SNF. Reliable depletion code validation results require a sufficient number of data sets that include all isotopes for which burnup credit is taken. The applicant, therefore, should provide justification of the sample size for each nuclide. For example, the applicant should demonstrate that isotopic uncertainty is appropriately increased to account for uncertainty associated with a small number of available measurement data or for uncertainty associated with non-normal isotopic validation data. The analyses in NUREG/CR-7108 use appropriate methods to account for these uncertainties.

Sample data necessary for depletion code validation includes initial enrichment and burnup, depletion history, assembly design characteristics, and physical location within the assembly. Over the past several decades, various RCA measurements of SNF samples have been performed at different laboratories. Detailed descriptions and analyses of the RCA measurements available for use in isotopic depletion validation have been published by the NRC and ORNL in the following references:

- NUREG/CR-7012, "Uncertainties in Predicted Isotopic Compositions for High Burnup PWR Spent Nuclear Fuel"

- NUREG/CR-7013, "Analysis of Experimental Data for High-Burnup PWR Spent Fuel Isotopic Validation—Vandellós II Reactor"

- NUREG/CR-6968," Analysis of Experimental Data for High Burnup PWR Spent Fuel Isotopic Validation—Calvert Cliffs, Takahama, and Three Mile Island Reactors"

- NUREG/CR-6969, "Analysis of Experimental Data for High Burnup PWR Spent Fuel Isotopic Validation-ARIANE and REBUS Programs (UO_2 Fuel)"

NUREG/CR-7108 analyzes the available data sets and identified 100 fuel samples suitable for depletion code validation for SNF storage and transportation systems. The reviewer should examine the sample data and depletion models to ensure that these sample data are used in the application to determine the bias and bias uncertainty associated with the chosen isotopic depletion methodology. If different RCA data are used for the isotopic depletion validation, the applicant should provide all relevant information associated with that data (e.g., burnup, enrichment, cool time, local irradiation environment) and justify that this data is appropriate for the intended purpose. RCA data from samples with incomplete or unknown physical and irradiation history data should be avoided. Note that the burnup values associated with the RCA measurements are the actual sample burnup, rather than fuel assembly average burnup, which is typically used in burnup credit calculations. Reviewers should ensure that the benchmark models constructed by the applicant for depletion code validation use the appropriate burnup value.

Because of differences in the techniques used in RCA measurement programs, the results may vary significantly between different measurements of the same nuclide, in some cases. These variations may result in a large uncertainty in the calculated concentration for a particular nuclide, and reviewers should expect to see such large uncertainties for certain nuclides until a better database of measurements is available.

7A.5.2 Radiochemical Assay Modeling

The depletion validation analysis should use the time-dependent irradiation environment and decay time for each individual RCA sample. Accurate sample depletion parameters should be used in the depletion code validation analysis models. A sample should not be used if its depletion history and environment are not well known. Note that some samples were taken from specific locations in the fuel assembly, while other samples have been taken on an assembly average basis. The latter type is typically found in earlier RCA data.

A depletion model should be built for each set of measurement data that were obtained from a RCA sample. To validate the computer code and obtain the bias and bias uncertainty, the depletion model should be able to accurately represent the environment in which each SNF sample was irradiated. For example, a sample from a fuel rod near a water hole will have a different neutron flux spectrum than a sample in a location where it is surrounded by fuel rods. Similarly, a fuel assembly with BPR insertion will have a different neutron spectrum in comparison to one without BPR exposure. Furthermore, a sample taken from the end of a fuel rod would have different specific power, fuel temperature, moderator temperature, and moderator density in comparison with that of a sample taken from the middle of a fuel assembly. Finally, time-dependent, three-dimensional effects, such as CR insertion, BPR insertions, and partial rod or gray rod insertions during part of the depletion processes should also be captured. These local effects are averaged in a one-dimensional depletion code, and the reviewer should expect to see relatively large uncertainties associated with one-dimensional depletion code calculations of individual RCA sample nuclide concentrations.

7A.5.3 Depletion Code Validation Methods

One of the objectives of code validation is to determine the bias and bias uncertainty associated with the isotopic concentration calculations. NUREG/CR-6811, "Strategies for Application of Isotopic Uncertainties in Burnup Credit," discusses several approaches to treat the bias and bias uncertainty associated with isotopic concentration calculations. NUREG/CR-7108 expands on two of these approaches in greater detail, and provides reference results for representative SNF storage and transportation systems. These approaches are discussed in the following paragraphs.

1. Isotopic Correction Factor Method

This approach uses a set of correction factors for isotopes that are included in burnup credit analyses. Correction factors are derived by statistical analysis of the ratios of the calculated-to-measured isotopic concentrations of the RCA samples for each isotope. The mean value, plus or minus the standard deviation multiplied by a tolerance factor appropriate to yield a 95/95 confidence level, is determined as the correction factor for a specific isotope. For the fissile isotopes, the correction factor is the mean value plus the modified standard deviation. For non-fissile absorber isotopes, the correction factor is the mean value minus the modified standard deviation. Fissile isotope correction factors that are below 1.0 are conservatively set to 1.0, and absorber isotope correction factors that are above 1.0 are conservatively set to 1.0. Since this method includes all the uncertainties associated with the measurements, computer algorithm, data library, and modeling, and since the correction factors are only modified in a manner that will increase k_{eff}, the result is considered bounding.

2. Direct-Difference Method

The direct-difference method directly computes the k_{eff} bias and bias uncertainty associated with the depletion code for the same set of isotopes by using the measured and calculated isotopic concentrations in the criticality analysis models separately. Two k_{eff} values are obtained in each pair of calculations, and a Δk_{eff} is calculated for each set of measured data. A statistical analysis is performed to calculate the mean value and the uncertainty associated with the mean value of the Δk_{eff}. Regression analysis is performed to determine the bias of the mean Δk_{eff} value as a function of various system parameters (e.g., burnup, initial enrichment).

Note that the direct-difference method requires a full set of measured data for all isotopes for which this method is used to determine the bias and bias uncertainty of the isotopic depletion analysis code. However, many isotopes in Tables 7A-1 and 7A-2, particularly the fission products, do not have sufficient numbers of measured data for performing significant statistical analysis. In these cases, surrogate data have been used, as described in NUREG/CR-7108. This surrogate data set was generated using the available measured data for an isotope as the basis to populate the missing data in the measured data sets. A surrogate data value was determined by multiplying the calculated nuclide concentration by the mean value of the measured-to-calculated concentration ratio values obtained from samples with measured data. The fundamental assumption of this approach is that the limited available measured data are representative of the entire population of isotopic concentration values. When the number of available measured data for a specific isotope is low or covers a small burnup range, the applicant should ensure that this assumption is still valid, as was done for molybdenum-95, ruthenium-101, rhodium-103, and cesium-133 in Section 6.2 of NUREG/CR-7108.

Based on the studies published in NUREG/CR-7108, decay time correction is an important factor when using the direct-difference method. In cases where there are differences between the cooling times of the samples used in code validation and the design-basis fuel cooling time, the error in the isotopic calculations can be large. NUREG/CR-7108 provides a discussion of the method to correct decay times for the samples that were selected for code validation. This method uses the Bateman Equation (Benedict et al. 1981) to adjust the measured isotopic concentration of the nuclide of interest to the design basis cooling time of the application. For a general case of nuclide B with a decay precursor A and a daughter product C (i.e., A → B → C), the content of nuclide B at a reference cooling time can be obtained by solving the Bateman Equation. The time-adjusted isotopic concentration should be used in the validation rather than the measurement data. In the case where only a fraction of the decay leads to the production of nuclide B, the fraction of decay of nuclide A leading to nuclide B should also be included. For a nuclide without a significant precursor, the contribution from decay of precursors should be set to zero, and only the decay of nuclide B needs to be considered.

3. Monte Carlo Uncertainty Sampling Method

The Monte Carlo uncertainty sampling method generates a depletion code k_{eff} bias (β_i) and bias uncertainty, Δk_i for the group of nuclides for which burnup credit is taken. It determines the bias and bias uncertainty using a statistical method that adjusts the isotopic concentrations of the SNF in the criticality analysis model by a factor randomly sampled within the uncertainty band of measured-to-calculated isotopic concentration ratios of each nuclide. NUREG/CR-7108 provides a more detailed discussion of this approach. Research results published in NUREG/CR-7108 indicate that this method, although statistically complex and computationally intensive, can be used to determine a more realistic bias and bias uncertainty of the depletion code.

Using the Monte Carlo uncertainty sampling method, ORNL has developed reference bias and bias uncertainty values for the hypothetical GBC-32 storage and transportation system. The NRC finds it acceptable for the applicant to use the bias and bias uncertainty values from Tables 7A-3 and 7A-4 directly, in lieu of an explicit depletion validation analysis, provided the following conditions are met:

- the applicant uses the same depletion code and cross section library as was used in NUREG/CR-7108 (SCALE/TRITON and the ENDF/B-V or ENDF/B-VII cross section library)

- the applicant can justify that its design is similar to the hypothetical GBC-32 system design used as the basis for the NUREG/CR-7108 isotopic depletion validation

- credit is limited to the specific nuclides listed in Tables 7A-1 and 7A-2

Bias values should be added to the calculated system k_{eff}, while bias uncertainty values may be statistically combined with other independent uncertainties, consistent with standard criticality safety practice. Demonstration of system similarity to the GBC-32 should consist of a comparison of materials and geometry, as well as neutronic characteristics such as H/X ratio, EALF, neutron spectra, and neutron reaction rates. If any of the above conditions are not met, the applicant should use the direct-difference or isotopic correction factor methods discussed previously.

Table 7A-3 Isotopic k_{eff} Bias Uncertainty (Δk_i) for the Representative PWR SNF System Model using ENDF/B VII data (β_i = 0) as a Function of Assembly Average Burnup

Burnup (BU) Range (GWd/MTU)	Actinides Only Δk_i	Actinides and Fission Products Δk_i
0≤BU<5	0.0145	0.0150
5≤BU<10	0.0143	0.0148
10≤BU<18	0.0150	0.0157
18≤BU<25	0.0150	0.0154
25≤BU<30	0.0154	0.0161
30≤BU<40	0.0170	0.0163
40≤BU<45	0.0192	0.0205
45≤BU<50	0.0192	0.0219
50≤BU≤60	0.0260	0.0300

Table 7A-4 Isotopic k_{eff} Bias (β_i) and Bias Uncertainty (Δk_i) for the Representative PWR SNF System Model using ENDF/B V Data as a Function of Assembly Average Burnup

Burnup (BU) Range (GWd/MTU)[a]	β_i for Actinides and Fission Products	Δk_i for Actinides and Fission Products
0≤BU<10	0.0001	0.0135
10≤BU<25	0.0029	0.0139
25≤BU≤40	0.0040	0.0165

a Bias and bias uncertainties associated with ENDF/B-V data were calculated for a maximum of 40 GWd/MTU. For burnups higher than this, applicants should provide an explicit depletion code validation analysis using one of the methods described in this appendix, along with appropriate RCA data.

7A.6 Code Validation—K_{eff} Determination (Chapter 7, Section 7.5.5.4 of the SRP)

For the k_{eff} component of burnup credit criticality calculations, validation is the process by which a criticality code system user demonstrates that the code and associated data predict actual system k_{eff} accurately. The criticality code validation process should include an estimate of the bias and bias uncertainty associated with using the codes and data for a particular application.

As stated in American National Standards Institute (ANSI)/American Nuclear Society (ANS) 8.1, "Nuclear Criticality Safety in Operations with Fissionable Materials Outside Reactors":

> Bias shall be established by correlating the results of critical and exponential experiments with results obtained for these same systems by the calculational method being validated.

The previous technical basis for burnup credit in ISG-8, Revision 2, limited credit to the major actinides, since there were not adequate critical experiments at the time for estimating the bias and bias uncertainty relative to modeling SNF in a cask environment. This technical basis considered the fact that no critical experiments existed which included the fission product isotopes important to burnup credit. Additionally, critical experiments available for actinide validation were limited to only (1) fresh low-enriched UO_2 systems and (2) fresh mixed uranium and plutonium oxide systems. These systems are not entirely representative of SNF in a transportation package, as fresh UO_2 systems contain no plutonium, and the MOX experiments generally do not have plutonium isotopic ratios consistent with that of burned fuel.

While there were no representative critical experiments for SNF transportation or storage criticality validation, there were considered to be adequate RCA data for validating actinide isotopic depletion calculations for major actinide absorbers. For this reason, as well as the criticality validation limitations discussed above, the NRC staff deemed that it was appropriate to recommend "actinide-only" credit for SNF transportation and storage criticality safety evaluations. This approach represented the bulk of the reduction in k_{eff} due to depletion of the fuel (see Table 7A-5) and excluded the fission products that served as additional margin to cover uncertainties due to modeling actinide depletion k_{eff} effects.

Table 7A-5 FP Reactivity Worth for "Typical" Burnup in Generic Burnup Credit Cask (GBC-32) with 4 Weight Percent Uranium-235 Westinghouse 17 X 17 OFA, Burned to 40 GWd/MTU

Credited Nuclides	k_{eff}	Δk	%Δk[a]
Fresh Fuel	1.13653		
8 Major Actinides[b]	0.94507	0.19146	71.9
All Actinides	0.93486	0.01021	3.8
Key 6 Fission Products[c]	0.88499	0.04987	18.7
All Remaining Fission Products	0.87010	0.01489	5.6
Totals		0.26643	100

a. This is the percent of total Δk for the burnup attributable to the portion of the total nuclide population in the first column.
b. Eight major actinides include uranium-235, uranium-238, plutonium-238, plutonium-239, plutonium-240, plutonium-241, plutonium-242, and americium-241.
c. Six key fission products include rhodium-103, cesium-133, samarium-149, samarium-151, 143Nd, and 155Gd.

Although there continue to be insufficient critical experiments for a traditional validation of the code-predicted reduction in k_{eff} due to fission products and minor actinides in SNF, a group of critical experiments designed for validating SNF k_{eff} reduction due to major actinides has become available since ISG-8, Revision 2, was published. This actinide criticality validation data is described in detail in NUREG/CR-6979, "Evaluation of the French Haut Taux de Combustion (HTC) Critical Experiment Data," and is available to applicants from ORNL, subject to execution of a non-disclosure agreement. These experiments are more appropriate for validating the code-predicted reduction in k_{eff} resulting from actinide depletion than fresh UO_2 or other MOX critical experiments. The HTC experiments consisted of fuel pins fabricated from mixed uranium and plutonium oxide, with the uranium and plutonium isotopic ratios designed to approximate what would be expected from UO_2 fuel burned in a PWR to 37.5 GWd/MTU. While these experiments were designed to correspond to a single burnup rather than the range of burnups that would be ideal for criticality validation, this data set represents a significant improvement to the criticality validation data available for actinide isotopes.

The improvement to the actinide criticality validation data set allows applicants for burnup credit SNF transportation packages and storage casks to perform a traditional validation for the actinide component of the reduction in k_{eff} resulting from burnup, per the recommendations of NUREG/CR-6361. ORNL has performed a representative actinide criticality validation for the GBC-32 transportation package provided in NUREG/CR-7109 using the best available validation data.

Although the contribution from fission products to the reduction in k_{eff} resulting from burnup is relatively small (see Table 7A-5), applicants for SNF transportation packages have requested the additional credit represented by these absorbers. The apparent need for fission product credit result from the significant increase in the percentage of discharged PWR fuel assemblies capable of being stored or shipped in a high-capacity (e.g., 32-assembly) system. Figure 7A-8 represents a typical discharged PWR fuel population in terms of initial enrichment and burnup. Two representative loading curves, one for actinide-only burnup credit and another for actinide and fission product burnup credit, are overlain on this figure, showing the relative amounts of the PWR fuel population that would be transportable in a hypothetical package. Although the loading curve does not move significantly from actinide-only credit to actinide and fission product credit, the curve moves across the bulk of the discharged fuel population, making a greater percentage of this population transportable. If a greater number of transportation packages can have this high capacity, then the total number of eventual SNF shipments could be reduced.

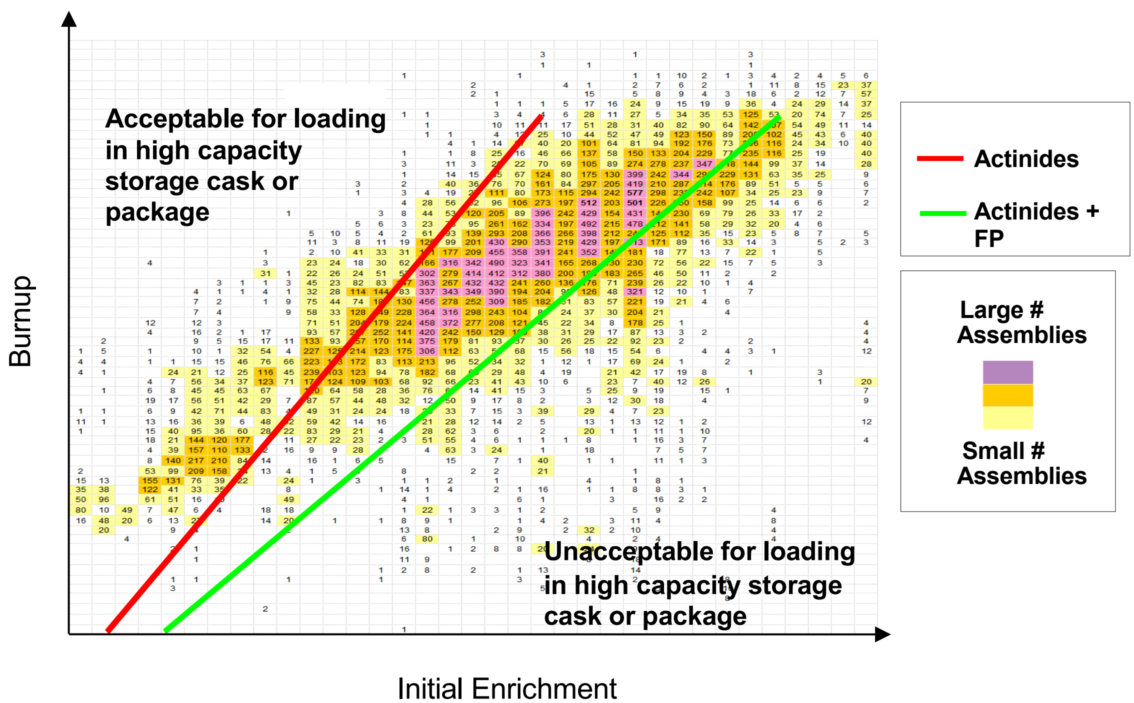

Figure 7A-8 Representative Loading Curves and Discharged PWR Population

The ability to properly validate criticality codes for actinide burnup credit is a crucial step toward recommending fission product credit, as the actinides represent the bulk of the reduction in k_{eff} resulting from burnup. However, it is still necessary to be able to estimate the bias and bias uncertainty due to modeling fission products in SNF, and critical experiments that include fission product absorbers continue to be exceedingly rare. As of this writing, there are only a handful of such publicly available critical experiments: one set involving samarium-149 (LEU-COMP-THERM-050), another involving rhodium-103 (LEU-COMP-THERM-079), and a third involving elemental samarium, cesium, rhodium, and europium (LEU-MISC-THERM-005). The preferred method for further fission product criticality validation would be the development of numerous and varied critical experiments involving both actinide and fission product absorbers in concentrations representative of SNF of various initial enrichments and burnups. Given the cost and practical difficulties associated with such a critical experiment program (e.g., obtaining specific absorber isotopes as opposed to natural distributions of isotopes), the NRC staff does not expect to see such experiments carried out within a reasonable timeframe. In the absence of such important criticality validation data, the NRC staff and contractors at ORNL sought alternative methodologies for estimating fission product bias and bias uncertainty.

In order to achieve an appropriate estimate of the k_{eff} bias and bias uncertainty due to fission products, ORNL developed a methodology based on the SCALE Tools for Sensitivity and Uncertainty Methodology Implementation (TSUNAMI) code (Rearden 2009), developed as part of the SCALE code system. This methodology uses the nuclear data uncertainty estimated for each fission product cross section known as the cross section covariance data. These data are provided with the ENDF/B-VII cross section library. The TSUNAMI code is used to propagate the cross section uncertainties represented by the covariance data into k_{eff} uncertainties for each fission product isotope used in a particular application. The theoretical basis of this validation technique is that computational biases are primarily caused by errors in the cross section data, which are quantified and bounded, with a 1σ confidence, by the cross section covariance data. NUREG/CR-7109 discusses the validity of this theoretical basis in greater detail.

This methodology has been benchmarked against a large number of low enrichment uranium critical experiments, high enrichment uranium critical experiments, plutonium critical experiments, and mixed uranium and plutonium critical experiments to demonstrate that the k_{eff} uncertainty estimates generated by the method are consistent with the calculated biases for these systems. The k_{eff} uncertainty results for specific fission products were also compared to fission product bias estimates obtained from the limited number of critical experiments that include fission products. NUREG/CR-7109 describes the uncertainty analysis method and provides details of the comparisons. The results demonstrate that, for a generic SNF transportation package evaluated with the SCALE code system and the ENDF/B-V, ENDF/B-VI, or ENDF/B-VII cross section libraries, the total fission product nuclear data uncertainty (1σ) does not exceed 1.5 percent of the total minor actinide and fission product worth for the 19 nuclides (Table 7A-2) considered over the burnup range of interest (i.e., 5 to 60 GWd/MTU). Since the uncertainty in k_{eff} resulting from the uncertainty in the cross section data is an indication of how large the actual code bias could be, the 1.5-percent value should be used as a bias (i.e., added directly to the calculated k_{eff}). Because of the conservatism in this value, no additional uncertainty in the bias needs to be applied.

In order to use the 1.5-percent value directly as a bias, applicants must demonstrate that they have used the code in a manner consistent with the modeling options and initial assumptions used in NUREG/CR-7109. Applicants must also demonstrate that their SNF storage or transportation system design is similar to the GBC-32 used to develop the bias estimate. This demonstration should consist of a comparison of materials and geometry, as well as neutronic characteristics such as H/X ratio and EALF. Since improved actinide validation with the HTC experiments discussed previously represents a considerable part of the technical basis for crediting fission product absorbers, applicants should validate the actinide portion of the k_{eff} evaluation against this data set.

Applicants may also use a different criticality code, provided that the code uses ENDF/B-V, ENDF/B-VI, or ENDF/B-VII cross section data. In this case, the combined minor actinide and fission product bias and bias uncertainty should be increased to 3.0 percent. NUREG/CR-7109 shows that the bias and bias uncertainty are based largely on the uncertainty in the nuclear data. However, there are differences in how different codes handle the same cross section data, potentially affecting bias and bias uncertainty. Since validation studies similar to that performed in

NUREG/CR-7109 have not been performed for other codes, the staff finds that an additional k_{eff} penalty should be applied to cover any additional uncertainties, and that doubling the 1.5 percent determined for the SCALE code system is conservative. ORNL performed additional analyses with MCNP5 and MCNP6, with ENDF/B-V, ENDF/B-VI, ENDF/B-VII, and ENDF/B-VII.1 cross section data. These analyses, documented in NUREG/CR-7205, “Bias Estimates Used in Lieu of Validation of Fission Products and Minor Actinides in MCNP K_{eff} Calculations for PWR Burnup Credit Casks,” demonstrate that the 1.5-percent value is also acceptable for use with these codes and cross section libraries.

The reviewer should consider applicant requests to use the 1.5-percent value for other well-qualified industry standard code systems, provided the application includes additional justification that this value is appropriate for that specific code system (e.g., a minor actinide and fission product worth comparison to SCALE results). For applications where the applicant uses cross section libraries other than ENDF/B-V, ENDF/B-VI, or ENDF/B–VII, where the application system cannot be demonstrated to be similar to the GBC-32, or where the credited minor actinide and fission product worth is significantly greater than 0.1 in k_{eff}, an explicit validation analysis should be performed to determine the bias and bias uncertainty associated with minor actinides and fission products.

7A.6.1 Integral Validation

ANSI/ANS 8.27-2008, “Burnup Credit for LWR Fuel,” provides a burnup credit criticality validation option consisting of analysis of applicable critical systems consisting of irradiated fuel with a known irradiation history. This is known as integral, or “combined,” validation, since the bias and bias uncertainty associated with the depletion calculation method is inseparable from that associated with the criticality calculation method. The most common publicly available sources of integral validation data are commercial reactor critical (CRC) state points. These CRC state points consist of either a hot zero-power critical condition attained after sufficient cooling time to allow the fission product xenon inventory to decay or an at-power equilibrium critical condition where xenon worth has reached a fairly stable value.

CRC state points have been shown to be similar to cask-like environments, with respect to neutron behavior, in NUREG/CR-6951, “Sensitivity and Uncertainty Analysis of Commercial Reactor Criticals for Burnup Credit.” With integral validation, however, the biases and uncertainties for the depletion approach cannot be separated from those associated with the criticality calculation, and only the net biases and uncertainties from the entire procedure are obtained. This approach allows for compensating errors between the depletion methodology and the criticality methodology (e.g., under prediction of a given nuclide’s concentration coupled with simultaneous over prediction of this nuclide’s effect on k_{eff}). It is desirable to understand the sources of uncertainty associated with the depletion methodology separately from those for the criticality methodology in order to ensure that the overall bias and bias uncertainty are determined correctly for the cask system for the entire range of system parameters.

Additionally, concerns remain regarding the physical differences between CRC state points and cask systems, such as borated water in a reactor versus fresh water in a cask, high worth absorber plates in a cask versus none in a reactor, low moderator density in a reactor versus full density in a cask, and high temperature in a reactor versus low temperature in a cask. CRC state points also consist of calculated isotopic concentrations, as opposed to the measured concentrations one would expect in a typical laboratory critical experiment. Furthermore, CRC state points are inherently complicated to model, given the large number of assemblies and axial zones with different initial enrichments and burnups necessary to accurately model the reactor core. All of these concerns introduce additional uncertainties into a validation approach that attempts to make use of CRC state points.

For the reasons stated above, using integral validation approaches is not recommended, with CRC state points or any other available integral validation data, for burnup credit criticality validation. However, if integral validation is used, the applicant should account for additional uncertainties, such as those identified above, and consider the use of a k_{eff} penalty to offset those uncertainties.

7A.7 Loading Curve and Burnup Verification (Chapter 7, Section 7.5.5.5 of the SRP)

As part of storage and transportation operations, loading curves are used to display acceptable combinations of assembly average burnup and initial enrichment for loading fuel assemblies. Assemblies with insufficient burnup, in comparison with the loading curve, are not acceptable for loading, as shown in Figure 7A-8. Misloads have occurred in both dry storage casks and spent fuel pools, in which fuel did not satisfy allowable parameters (e.g., burnup, cooling time, and enrichment). Misloads occur because of misidentification, mischaracterization, or misplacement of fuel assemblies. This has resulted in unanalyzed loading configurations during storage of SNF in some cases. To date, the known dry storage cask misload events have not had significant implications on criticality safety.

For efficiency and economic purposes in power plant operations, it is desirable to ensure that the maximum power output is extracted from a fuel assembly before discharging it from the reactor. However, some fuel assemblies have been removed from the reactor before achieving their desired burnup because of fabrication or performance issues. Once discharged from the reactor, these fuel assemblies are stored in the spent fuel pool. Because the spent fuel pool may contain assemblies with varying burnups, enrichments, and cooling times, the potential for a more reactive assembly to be misloaded exists. A misload can occur as a result of several factors, including assemblies with fabrication issues, errors in reactor records, or operator actions which impact fuel handling activities.

ISG-8, Revision 2, specified that certain administrative procedures should be established to ensure that fuel designated for a particular storage or transportation system is within the specifications for approved contents. Burnup measurement was recommended in the guidance as a way to protect against misloads by identifying potential errors in reactor records or misidentification of assemblies being loaded into the system. As part of the overall initiative to revise staff burnup credit criticality review recommendations, the potential effects of misloaded assemblies on system reactivity were investigated.

Misloading of unirradiated fuel assemblies is unlikely for several reasons. First, storage and transportation system loading typically occurs when unirradiated fuel is not present in the spent fuel pool. Second, SNF is noticeably different than unirradiated fuel (color, deformation) and visually identifiable. Finally, there is an economic incentive involved with new fuel assemblies that would make permanent misloads of unirradiated fuel assemblies in dry storage casks or transportation packages unlikely.

Although misloading of unirradiated fuel assemblies is considered to be unlikely, it is conceivable that an assembly that has been irradiated to less than the target burnup value (i.e., underburned) could be misloaded into an SNF system. Misloading of one or more underburned fuel assemblies can cause an increase in the overall system reactivity. The amount of reactivity increase depends on several factors, including the degree of burnup in comparison to the loading curve, the cooling time, and the location of the assembly within the system.

A number of events involving misloads occurring within spent fuel pools and dry storage casks have been reported to the NRC. The majority of these misloads occurred as a result of inadequate fuel selection procedures or inaccurate parameter data (i.e., burnup, enrichment, cooling time). Using available misload data, the RES report, “Estimating the Probability of Misload in a Spent Fuel Cask,” (NRC 2011) evaluated the likelihood of misloading fuel assemblies within a SNF transportation package. This report determined the probability of single and multiple assembly misloads for ranges of burnup values dependent on the available spent fuel pool inventory. RES determined that the overall probability of misloading a fuel assembly that does not meet the burnup credit loading curve is in the 10^{-2} to 10^{-3} range, which is considered credible.

NUREG/CR-6955, “Criticality Analysis of Assembly Misload in a PWR Burnup Credit Cask,” evaluated the effects of single and multiple misloaded assemblies on the reactivity in a storage or transportation system. This evaluation covered the misloading of unirradiated and underburned PWR fuel assemblies in a GBC-32 high-capacity storage and transportation system. The scope of this report included varying the degree to which misloaded assemblies were underburned to determine the change in reactivity when including actinide-only and actinide and fission product burnup credit. This was done over a range of enrichments up to 5.0 weight percent uranium-235, while placing between 1 and 4 misloaded assemblies into the most reactive positions within the system. All assemblies within the system were assumed to undergo a cooling period of 5 years. The misloaded assemblies were evaluated at 90, 80, 50, 25, 10, and 0 percent (unirradiated) of the minimum assembly average burnup value required by the loading curve.

The evaluation in NUREG/CR-6955 concluded that for the particular system design and fuel assembly parameters used, a reactivity increase between 2.0 and 5.5 percent in k_{eff} could be expected for various misloaded systems. Given the operational history and the accuracy of the reactor records, this information can be used along with the misload probability to determine an appropriate method of addressing assembly misloads as part of the criticality evaluation. Applicants may perform a misload analysis in lieu of a confirmatory burnup measurement.

7A.7.1 Misload Evaluation

The applicant’s misload evaluation should be based on a reliable and relatively recent estimate of the discharged PWR fuel population, and should reflect the segment of that population that is intended to be stored or transported in the cask or package design. Note that this population may consist of the entire population of discharged PWR fuel assemblies, a specific design of PWR fuel assembly (e.g., W17 x 17 OFA), or a smaller, specific population from a particular site. An acceptable source of discharged fuel data as of this writing is the 2002 Energy Information Administration (EIA) RW-859, “Nuclear Fuel Data Survey” (EIA 2004), although more recent data may be available.

An applicant’s misload analysis should evaluate both a single, severely underburned misload and a misload of multiple moderately underburned assemblies in a single SNF system. The single severely underburned assembly should be chosen such that any assembly average burnup and initial enrichment along an equal reactivity curve bound 95 percent of the discharged fuel population considered unacceptable for loading in a particular storage or transportation system with 95-percent confidence. Applicants should provide a statistical analysis of the underburned fuel population to support the selection of severely underburned assemblies.

The 95/95 criterion for evaluations of single high-reactivity misloads, along with the administrative procedures for misload prevention (see Administrative Procedures below), is reasonably bounding as more reactive misloads are unlikely. The assembly average burnup and initial enrichment that match this 95/95 criterion are dependent upon the loading curve for the storage or transportation system. Applicants are likely to seek a level of burnup credit that results in qualification of the greatest possible amount of the fuel population for storage or shipment in the system. Therefore, assemblies matching the 95/95 criterion will be those of relatively high enrichment and low burnup (e.g., 5 wt. percent uranium-235 and 15 GWd/MTU). Based on the data available in the 2002 EIA RW-859, the number of discharged assemblies of greater reactivity is very small, even for cases where all discharged assemblies of a given burnup and initial enrichment are located in a single spent fuel pool.

For the evaluation of the application system with multiple moderately underburned assemblies, misloaded SNF should be assumed to make up at least 50 percent of the system payload, and should be chosen such that the assembly average burnups and initial enrichments along the equal reactivity curve bound 90 percent of the total discharged fuel population. Such an evaluation is reasonably bounding for cases of multiple misloads in a single SNF system based upon the considerations in the following paragraph.

The 90-percent criterion is based on the total discharged fuel population and not the specific loading curve for the system design. The distribution of discharged fuel peaks within a relatively narrow band of burnup for each initial enrichment value. The curve that represents a reactivity that bounds 90 percent of the discharged population is expected to pass through burnup and enrichment combinations that are below this peak. However, the population along this curve is still large enough to represent possible misload scenarios involving multiple assemblies. Below the 90-percent criterion curve, with few exceptions, the numbers of assemblies for each burnup and enrichment combination drop significantly. Thus, it is reasonable to expect that misloading of multiple assemblies of the remaining 10 percent of the discharged population would be less likely. Although there are larger numbers of low burnup assemblies for specific initial enrichments, facilities that have a significant number of these assemblies can reduce the likelihood of misloading multiples of these assemblies in the same system with proper administrative controls.

The recommendation for misloading at least 50 percent of the system is based on consideration of the history of misloads in dry SNF storage operations as well as the fact that systematic errors can result in misloading of multiple assemblies. Misloads that have occurred in dry SNF storage operations have typically involved multiple assemblies. The most significant of these incidents resulted in less than 25 percent of the cask capacity being misloaded. While the probability of a multiple-misload scenario decreases with increasing number of assemblies involved, systematic errors can increase the likelihood of such misloads. Considering these factors, there is reasonable assurance that a scenario that involves misloading at least 50 percent of the cask capacity would bound the extent of likely multiple-misload conditions. The implementation of the administrative procedures recommended in Section 7.5.5.5 of this SRP and this appendix for preventing misloads provides additional assurance against more extensive misload situations.

It is possible that SNF systems designed for specific parts of the fuel population (e.g., particular sites or fuel types) will have loading curves that already bound 90 percent of the discharged fuel population. In these cases, the misload analysis for multiple assemblies does not need to be performed.

A SNF storage or transportation system should be designed to have a limited sensitivity to misloads, such that increases in k_{eff} when considering misloads are minimized. In any case, the applicant should demonstrate that the system remains subcritical under misload conditions including biases, uncertainties, and an administrative margin. The analysis should use the design parameters and specifications that maximize system reactivity as is done for nominal loading analyses. The administrative margin is normally 0.05. However, for the purposes of the misload evaluations, a different administrative margin may be used given two conditions. First, the administrative margin should not be less than 0.02. Second, any use of an administrative margin less than 0.05 should be adequately justified. An adequate justification should consider the level of conservatism in the depletion and criticality calculations, sensitivity of the system to further upset conditions, and the level of rigor in the code validation methods.

An administrative margin is used with criticality evaluations to ensure that a system that is calculated to be subcritical is actually subcritical. This margin is used to insure against unknown errors or uncertainties in the method of calculating k_{eff} as well as impacts of system design and operating conditions not explicitly considered in the analysis. Allowance for using different administrative margins is given in criticality safety practices in other regulated areas. Experience with identified code errors and an understanding of uncertainties in cross section data and their impacts on reactivity indicates that an administrative margin of at least 0.02 is necessary for analyses to show subcriticality with misloads.

Taking credit for burnup reduces the margin in the analyses and makes them more realistic. Additionally, reducing the administrative margin for misload analyses further reduces the margin for subcriticality. This reduction in overall criticality safety margin necessitates a greater justification for a lower administrative margin. This justification should demonstrate a greater level of assurance that the various sources of bias and bias uncertainty have been taken into account and that the bias and bias uncertainty are known with a high degree of accuracy. The principles and concepts discussed in FCSS ISG-10, “Justification for Minimum Margin of Subcriticality for Safety” (NRC 2000) are useful in understanding the kinds of evaluations and evaluation rigor that should be considered for justification of a lower administrative margin. These concepts include assurances of the consistent presence and amount of conservatism in the evaluations which may be relied upon, the quality and number of benchmark experiments as they relate to the application and the misload cases, and evaluation of the sensitivity of k_{eff} to other system parameter changes.

7A.7.2 Administrative Procedures

Along with the misload analysis, administrative procedures should be established, in addition to those typically performed for non-burnup credit systems, to ensure that the system will be loaded with fuel that is within approved technical specifications. Procedures considered to protect against misloads in storage and transportation systems that rely on burnup credit for criticality safety may include the following:

- verification of the location of high reactivity fuel (i.e., fresh or severely underburned fuel) in the spent fuel pool both before and after loading
- qualitative verification that the assembly to be loaded is burned (visual or gross measurement)
- verification, under a 10 CFR Part 71 quality assurance program, of the system inventory and loading records before shipment for previously loaded systems
- quantitative measurement of any fuel assemblies without visible identification numbers

- independent, third-party verification of the loading process, including the fuel selection process and fuel move instructions

- use and confirmation of minimum soluble boron concentration in pool water, to offset the misloads described above, during loading and unloading of storage containers under 10 CFR Part 72

The majority of these recommendations are intended to ensure that high reactivity fuel is not present in the pool during loading, or is otherwise accounted for and determined not to have been loaded into a SNF storage or transportation system. The verification of the system inventory and loading records is intended to ensure that the contents of previously loaded systems are as expected before shipment. This verification should be performed under an approved 10 CFR Part 71 quality assurance program. Quantitative measurement of SNF without visible identification is recommended since there is no other apparent way to demonstrate that such assemblies are tied to a specific burnup value. Independent, third-party verification of the fuel selection process means verification of correct application of fuel acceptability standards and the fuel move instructions.

Soluble boron is recommended as an unloading condition, to ensure that misloads are protected against when future unloading operations occur, since the conditions of such operations are currently unknown and may inadvertently introduce unborated water into the system. Soluble boron is typically present during PWR SNF loading operations for dry storage or transportation systems. An appropriate soluble boron concentration during loading and unloading would be that required to maintain system k_{eff} below 0.95 with the more limiting (in terms of k_{eff}) of the single, severely underburned or multiple moderately underburned misloads described previously. Consistent with requirements such as those in 10 CFR 71.55(b), transportation package analyses cannot credit the soluble boron present during PWR SNF loading into or unloading from the package. Therefore, the discussion regarding use of a minimum soluble boron concentration during loading and unloading (and credit for this soluble boron in analyses) applies only to loading and unloading for dry storage under 10 CFR Part 72.

Misload analyses are included in this revision of the criticality safety review guidance for burnup credit in this SRP as an alternative to burnup confirmation using measurement techniques. A number of misloads have occurred within spent fuel pools and casks as a result of human errors or inaccurate assembly data. Efforts have been made to evaluate the criticality effects of misloading assemblies into a SNF transportation package. Using credible bounding assumptions, a misload analysis could be generated to account for potential events involving loading, while maintaining an appropriate safety margin.

7A.8 References

10 CFR Part 71, “Packaging and Transportation of Radioactive Materials.”

10 CFR Part 72, “Licensing Requirements for the Independent Storage of Spent Nuclear Fuel, High-Level Radioactive Waste, and Reactor-Related Greater Than Class C Waste.”

American National Standards Institute (ANSI)/American Nuclear Society (ANS) 8.1-1998 (R2007), “Nuclear Criticality Safety in Operations with Fissionable Materials Outside Reactors,” American Nuclear Society, La Grange Park, Illinois.

ANSI/ANS 8.27-2008, "Burnup Credit for LWR Fuel," American Nuclear Society, La Grange Park, Illinois.

Benedict, M., T.H. Pigford, and H.W. Levi, Nuclear Chemical Engineering, Second Edition, McGraw Hill, 1981.

DeHart, M.D., "Sensitivity and Parametric Evaluations of Significant Aspects of Burnup Credit for PWR Spent Fuel Packages," ORNL/TM-12973, Lockheed Martin Energy Research Corp., Oak Ridge National Laboratory, May 1996.

DeHart, M.D., "Triton: A Two-Dimensional Transport and Depletion Module for Characterization of Spent Nuclear Fuel," ORNL/TM-2005/39, Version 6, Vol. I, Sect. T1, January 2009.

Duderstadt, J.J. and L.J. Hamilton, *Nuclear Reactor Analysis*, John Wiley & Sons Inc., 1976.

Blomquist, Roger N., Malcolm Armishaw, David Hanlon, Nigel Smith, Yoshitaka Naito, Jinan Yang, Yoshinori Mioshi, Toshihiro Yamamoto, Olivier Jacquet, and Joachim Miss, "Source Convergence in Criticality Safety Analyses, Phase I: Results for Four Test Problems," NEA No. 5431, NEA OECD, 2006.

DOE/RW-0496, "Horizontal Burnup Gradient Datafile for PWR Assemblies," U.S. Department of Energy, Office of Civilian Radioactive Waste Management, May 1997.

MCNP5, "MCNP – A General Monte Carlo N-Particle Transport Code, Version 5; Volume II: User's Guide," LA-CP-03-0245, Los Alamos National Laboratory, April 2003.

NUREG/CR-5661 "Recommendations for Preparing the Criticality Safety Evaluation of Transportation Packages," ORNL/TM-11936, Oak Ridge National Laboratory, March 1997.

NUREG/CR-6361, "Criticality Benchmark Guide for Light-Water-Reactor Fuel in Transportation and Storage Packages," ORNL/TM-13211, Oak Ridge National Laboratory, March 1997.

NUREG/CR-6665, "Review and Prioritization of Technical Issues Related to Burnup Credit for LWR Fuel," ORNL/TM-1999/303, Oak Ridge National Laboratory, February 2000.

NUREG/CR-6747, "Computational Benchmark for Estimation of Reactivity Margin from Fission Products and Minor Actinides in PWR Burnup Credit," ORNL/TM-2000/306, Oak Ridge National Laboratory, October 2001.

NUREG/CR-6759, "Parametric Study of the Effect of Control Rods for PWR Burnup Credit, U.S. Nuclear Regulatory Commission," ORNL/TM 2001/69, Oak Ridge National Laboratory, February 2002.

NUREG/CR-6760, "Study of the Effect of Integral Burnable Absorbers on PWR Burnup Credit," ORNL/TM-2000/321, Oak Ridge National Laboratory, March 2002.

NUREG/CR-6761, "Parametric Study of the Effect of Burnable Poison Rods for PWR Burnup Credit," ORNL/TM-2000/373, Oak Ridge National Laboratory, March 2002.

NUREG/CR-6781, "Recommendations on the Credit for Cooling Time in PWR Burnup Credit Analyses," ORNL/TM-2001/272, Oak Ridge National Laboratory, January 2003.

NUREG/CR-6800, “Assessment of Reactivity Margins and Loading Curves for PWR Burnup-Credit Cask Designs,” ORNL/TM-2002/6, Oak Ridge National Laboratory, March 2003.

NUREG/CR-6801, “Recommendations for Addressing Axial Burnup in PWR Burnup Credit Analyses,” ORNL/TM-2001/273, Oak Ridge National Laboratory, March 2003.

NUREG/CR-6811, “Strategies for Application of Isotopic Uncertainties in Burnup Credit,” ORNL/TM-2001/257, Oak Ridge National Laboratory, June 2003.

NUREG/CR-6951, “Sensitivity and Uncertainty Analysis of Commercial Reactor Criticals for Burnup Credit,” ORNL/TM-2006/87, Oak Ridge National Laboratory, September 2008.

NUREG/CR-6955, “Criticality Analysis of Assembly Misload in a PWR Burnup Credit Cask,” ORNL/TM-2004/52, Oak Ridge National Laboratory, January 2008.

NUREG/CR-6968, “Analysis of Experimental Data for High Burnup PWR Spent Fuel Isotopic Validation-Calvert Cliffs, Takahama, and Three Mile Island Reactors,” ORNL/TM-2008/071, Oak Ridge National Laboratory, February 2010.

NUREG/CR-6969, “Analysis of Experimental Data for High Burnup PWR Spent Fuel Isotopic Validation-ARIANE and REBUS Programs (UO_2 Fuel),” ORNL/TM-2008/072, Oak Ridge National Laboratory, February 2010.

NUREG/CR-6979, “Evaluation of the French Haut Taux de Combustion (HTC) Critical Experiment Data,” ORNL/TM-2007/083, Oak Ridge National Laboratory, September 2008.

NUREG/CR-7012, “Uncertainties in Predicted Isotopic Compositions for High Burnup PWR Spent Nuclear Fuel,” ORNL/TM-2010/41, Oak Ridge National Laboratory, January 2011.

NUREG/CR-7013, “Analysis of Experimental Data for High-Burnup PWR Spent Fuel Isotopic Validation—Vandellós II Reactor,” ORNL/TM-2009/321, Oak Ridge National Laboratory, January 2011.

NUREG/CR-7108, “An Approach for Validating Actinide and Fission Product Burnup Credit Criticality Safety Analyses – Isotopic Composition Predictions,” ORNL/TM-2011/509, Oak Ridge National Laboratory, April 2012.

NUREG/CR-7109, “An Approach for Validating Actinide and Fission Product Burnup Credit Criticality Safety Analyses – Criticality (k_{eff}) Predictions,” ORNL/TM-2011/514, Oak Ridge National Laboratory, April 2012.

NUREG/CR-7194, “Technical Basis for Peak Reactivity Burnup Credit for BWR Spent Nuclear Fuel in Storage and Transportation Systems,” ORNL/TM-2014/240, Oak Ridge National Laboratory, April 2015.

NUREG/CR-7205, “Bias Estimates Used in Lieu of Validation of Fission Products and Minor Actinides in MCNP K_{eff} Calculations for PWR Burnup Credit Casks,” ORNL/TM-2012/544, Oak Ridge National Laboratory, September 2015.

Oak Ridge National Laboratory, “SCALE: A Comprehensive Modeling and Simulation Suite for Nuclear Safety Analysis and Design,” ORNL/TM-2005/39, Version 6.1, June 2011. Available as

CCC-785 from the Radiation Safety Information Computational Center at Oak Ridge National Laboratory, https://rsicc.ornl.gov/Catalog.aspx?c=CCC.

Rearden, B.T., "TSUNAMI-3D: Control Module for Three-Dimensional Cross-Section Sensitivity and Uncertainty Analysis for Criticality," ORNL/TM-2005/39, Version 6, Vol. I, Section C9, January 2009.

RW-859 "Nuclear Fuel Data Survey, Energy Information Administration, https://www.eia.gov/nuclear/spent_fuel/.

U.S. Nuclear Regulatory Commission (NRC) Interim Staff Guidance (ISG)-8, Revision 2, "Burnup Credit in the Criticality Safety Analyses of PWR Spent Fuel in Transport and Storage Casks," Spent Fuel Project Office, U.S. Nuclear Regulatory Commission, September 27, 2002.

NRC, FCSS ISG-10, Revision 0, "Justification for Minimum Margin of Subcriticality for Safety," U.S. Nuclear Regulatory Commission, July 2000 (Agencywide Documents Access and Management System Accession No. ML061650370).

NRC, "Revisiting the Paradigm for Spent Fuel Storage and Transportation Regulatory Programs," Staff Requirements Memorandum COMDEK-09-0001, U.S. Nuclear Regulatory Commission, February 2010 (ADAMS Accession No. ML092160033).

NRC, "Estimating the Probability of Misload in a Spent Fuel Cask," Office of Nuclear Regulatory Research, U.S. Nuclear Regulatory Commission, June 2011 (ADAMS Accession No. ML113191144).

Withee, C.J., Memorandum to M. Wayne Hodges, "ISG-8, Rev. 2, Supporting Document," U.S. Nuclear Regulatory Commission, September 27, 2002 (ADAMS Accession No. ML022700395).

YAEC-1937, "Axial Burnup Profile Database for Pressurized-Water Reactors," Yankee Atomic Electric Company, May 1997. Available as Data Package DLC-201, PWR-AXBUPRO-SNL, from the Radiation Safety Information Computational Center at Oak Ridge National Laboratory, https://rsicc.ornl.gov/Catalog.aspx?c=DLC.

8 MATERIALS EVALUATION

8.1 Review Objective

The objective of the U.S. Nuclear Regulatory Commission's (NRC's) materials review is to ensure adequate materials performance of structures, systems, and components (SSCs) to ensure compliance with Title 10 of the *Code of Federal Regulations* (10 CFR) Part 72, "Licensing Requirements for the Independent Storage of Spent Nuclear Fuel, High-Level Radioactive Waste, and Reactor-Related Greater than Class C Waste," for a dry storage system (DSS) or dry storage facility (DSF), which includes independent spent fuel storage installations (ISFSIs) and monitored retrievable storage installations (MRS) involving handing, packaging, transfer, and storage. Materials must meet applicable codes, standards, and specifications and support intended functions of SSCs under all credible loads and environments for normal, off-normal, and accident conditions. The review also includes the evaluation of operations that ensure adequate materials performance, including material qualification, welding, spent nuclear fuel (SNF) drying, inerting of the confinement system, and the management of materials degradation.

8.2 Applicability

This chapter applies to the review of applications for specific licenses for an ISFSI or MRS and certificates of compliance (CoCs) of a DSS for use at a general license facility. Differences between the review of a specific license (SL) and a CoC are noted; in particular, specific licenses may involve SSCs associated with the storage of reactor-related greater-than-Class-C (GTCC) waste and high level radioactive waste (HLW) and facilities associated with SNF and waste handling, packaging, transfer, and storage.

8.3 Areas of Review

This chapter addresses the following areas of review:

- System and Facility Design
  - Drawings
  - Codes and Standards
  - Weld Design, Inspection, and Testing
- Material Properties
  - Mechanical Properties of Metals
  - Thermal Properties
  - Radiation Shielding Materials
  - Criticality Control Materials
  - Concrete and Reinforcing Steel
  - Bolt Applications
  - Seals
- Environmental Degradation; Corrosion and Other Reactions
  - Corrosion Resistance
  - Protective Coatings
  - Content Reactions
  - Management of Aging Degradation
- Fuel Cladding Integrity
  - Fuel Classification

- Uncanned Spent Fuel
- Canned Spent Fuel

8.4 Regulatory Requirements and Acceptance Criteria

This section summarizes those parts of 10 CFR Part 72 that are relevant to the review areas addressed by this chapter. Tables 8-1a and 8-1b match the relevant regulatory requirements to the areas of review covered in this chapter for applications for an ISFSI site license and CoC, respectively. The reviewer should refer to the language in the regulations and verify the association of regulatory requirements with the areas of review presented in these tables to ensure that no requirements are overlooked as a result of unique applicant design features.

Table 8-1a Relationship of Regulations and Areas of Review for a DSF (SL)

Areas of Review	10 CFR Part 72 Regulations				
	72.24	**72.120**	**72.122**	**72.124**	**72.128**
Design Criteria	(c)(3)	(a)			(a)
Code Use and Quality Standards	(c)(4)		(a)		
Material Properties	(d)			(b)	
Environmental Degradation; Chemical and Other Reactions		(d)	(b)(1), (c)	(b)	
Fuel Cladding Integrity and Retrievability			(h)(1), (h)(5), (l)		

Table 8-1b Relationship of Regulations and Areas of Review for a DSS (CoC)

Areas of Review	10 CFR Part 72 Regulations			
	72.122[A]	**72.124**	**72.234**	**72.236**
Design Criteria				(b)
Code Use and Quality Standards	(a)		(b)	
Material Properties		(b)		(g)
Environmental Degradation; Chemical and Other Reactions	(b)(1), (c)	(b)		(h)
Fuel Cladding Integrity and Retrievability	(h)(1), (h)(5), (l)			(a), (m)

[A] While not directly applicable to CoCs, DSS design should facilitate general licensee compliance with these requirements.

The materials evaluation seeks to ensure that materials will perform in a manner that supports the functions of the SSCs of storage systems and site facilities by fulfilling the following principal acceptance criteria that reflect the above regulations and areas of review:

- The applicant must provide information on materials of construction, including their fabrication, testing, and general arrangement, with sufficient detail to support a safety finding.

- Materials and special processes must conform to all applicable codes and standards. Non-code materials must have adequate controls for their qualification and fabrication.

- Material properties should have an adequate technical basis and must demonstrate the ability to support the performance of the intended functions of SSCs under credible loads in normal, off-normal, and accident conditions.

- Materials must not undergo adverse environmental degradation, chemical reactions, or other reactions that could challenge the ability of SSCs to safety handle, package, transfer, and store SNF, reactor-related GTCC waste, or HLW.

- The applicant must ensure that the SNF cladding is protected against gross ruptures or otherwise be confined and that the SNF, HLW, and reactor-related GTCC waste are always retrievable.

8.5 <u>Review Procedures</u>

Figure 8-1 shows the interrelationship between the materials evaluation and the other areas of review described in this standard review plan (SRP). The materials reviewer should survey the safety analysis report (SAR) and design drawings to identify the materials issues that are associated with the specific design proposal in the application. Examine the chapters of the SAR on criticality, shielding, confinement, structural, and thermal to identify cross-cutting issues that should be coordinated among the technical disciplines.

8.5.1 Drawings

Licensing drawings usually appear in SAR Chapters 1 or 2. Although developed for the review of transportation packages, the staff considers the guidance in NUREG/CR-5502, "Engineering Drawings for 10 CFR Part 71 Package Approvals," appropriate for the recommended content of storage drawings. Examine the drawings for material specifications, alternatives, and fabrication instructions including welding and nondestructive examination (NDE) requirements. Ensure that the applicant adequately specified any materials substitutes, either on the drawing or in the SAR. Ensure welding codes are clearly identified.

Standard welding and NDE symbols may be found in AWS A2.4, "Symbols for Welding, Brazing, and Nondestructive Testing," to aid interpretation of drawings. Section 8.5.3, "Welding," of this SRP provides additional guidance for the expected level of detail for weld filler metal and welding processes.

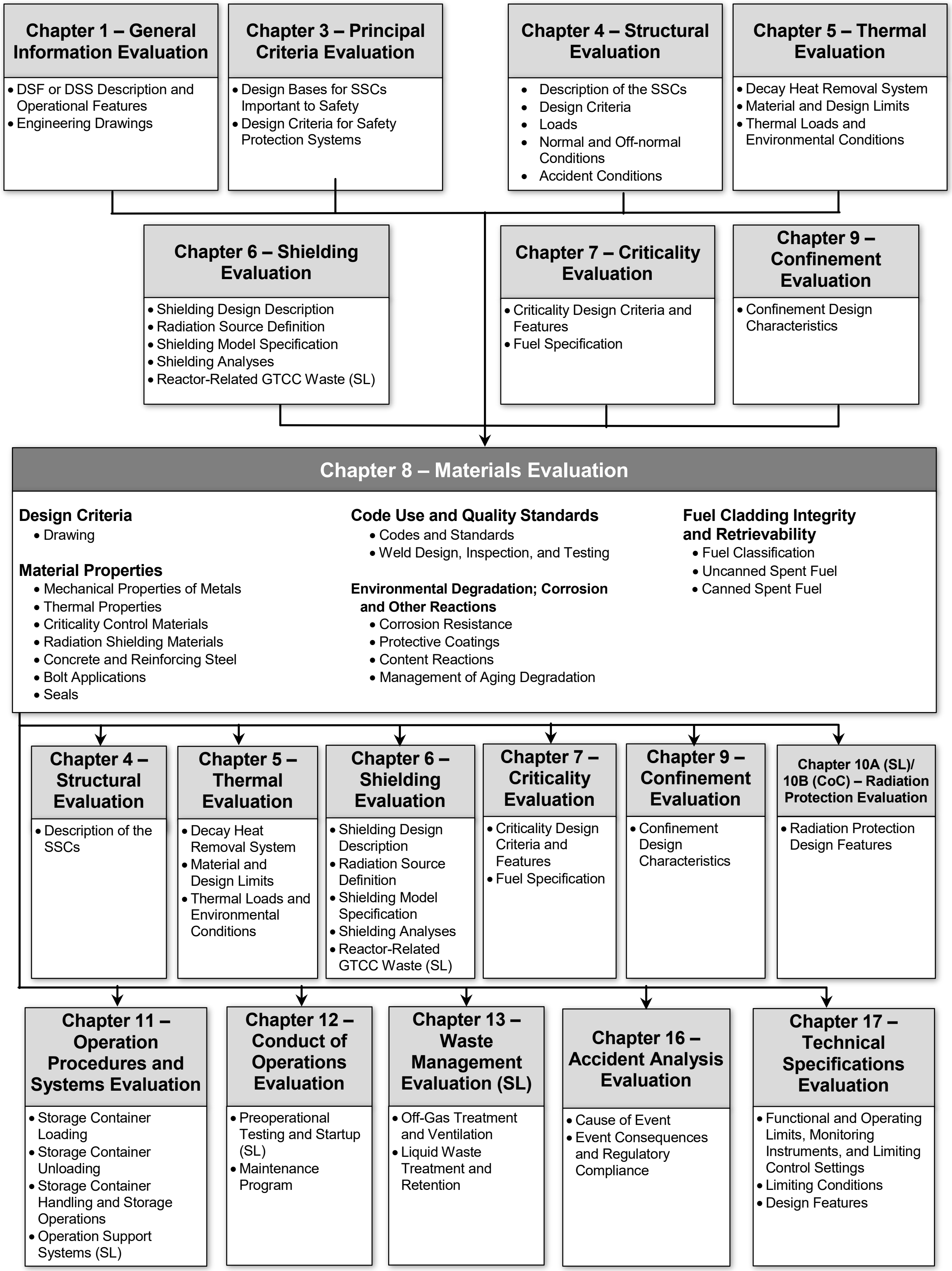

Figure 8-1 Overview of Materials Evaluation

Design drawings often do not identify a year or revision for codes and standards for materials specifications because the latest revision is widely considered to be appropriate for use. For example, metal producers routinely supply plates and forgings only to the latest revision of the ASTM standards, and thus it is expected that the DSS or DSF fabricator will necessarily procure material to the latest revision. Consequently, when a specific revision of a standard is not provided, base the materials review on the latest revision. An exception to this guidance is when this SRP or other NRC guidance recommends a particular version (or elements of an earlier revision) as a basis for the staff review. In that case, either (1) verify that key elements of the recommended earlier revision of the code or standard are still maintained in the latest version, or (2) consider whether the drawings should be revised to specifically cite the recommended earlier revision.

Other technical review disciplines may recommend that drawings include specific revisions of a code or standards associated with their review areas (e.g., SRP Section 4.5.1.1, “Structures, Systems, and Components Important to Safety,” for the structural design code). In that case, ensure that materials specifications are appropriate for the specific code version cited in the drawings.

8.5.2 Codes and Standards

The following guidance describes the materials codes and standards that the NRC finds acceptable for the construction of DSSs and DSFs. In several cases, the NRC staff recommends exceptions or additions to the codes and standards to address unique aspects of DSS and DSF designs.

8.5.2.1 *Usage and Endorsement*

For SSCs important to safety, ensure that the applicant specifies U.S. industry consensus codes and standards, such as the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (B&PV Code), American Welding Society (AWS) Code, American National Standards Institute (ANSI) standards, American Concrete Institute (ACI) Code, and ASTM International (ASTM) standards. Foreign codes and standards generally are not acceptable for SSCs or materials important to safety and would only be approved on a case-by-case basis. If used, ensure that foreign codes cross reference the appropriate ASME B&PV Code.

Approved storage containers are those that have been designed in accordance with the ASME B&PV Code. The NRC has accepted the design of confinement SSCs fabricated in accordance with ASME B&PV Code, Section III, “Rules for Construction of Nuclear Facility Components,” Subsection NB, “Class 1,” criteria; of fuel basket structures fabricated in accordance with ASME B&PV Code Section III, Subsection NG, “Core Supports”; and of other safety structures fabricated in accordance with ASME B&PV Code Section III, Subsection NF, “Supports.” For SSCs not associated with the confinement boundary or fuel basket, the NRC has accepted alternatives to the ASME B&PV Codes cited above. For example, the NRC has accepted the design of transfer casks to ASME B&PV Code Section III, Subsection NC, “Class 2,” criteria, and of other steel structures to the American Institute of Steel Construction (AISC) “Manual of Steel Construction.” Finally, as discussed in detail in Section 8.5.8, “Concrete and Reinforcing Steel,” of this SRP, NRC-accepted concrete structure designs have used ACI Codes.

The reviewer should ensure that the materials and their fabrication are consistent with the construction code or standard. Although written for the design of shipping containers, NUREG/CR-3854, “Fabrication Criteria for Shipping Containers,” may be used to identify where materials and fabrication criteria (e.g., heat treatment, examination, testing) are defined in the ASME B&PV Code sections. SSCs important to safety that are constructed in accordance with ASME B&PV Code Section III are normally fabricated from ASME Section II materials. Important-to-safety attachments to the confinement boundary, as well as structural components of the overpack, may be ASME or ASTM materials, depending on the code of record for the component. For non-ASME SSCs important to safety, ASTM materials may be used.

Codes and standards frequently reference one another, and the reviewer should note these relationships when verifying their proper use by the applicant. For example, all ASME materials are a subset of AWS and ASTM materials. However, not all ASTM materials are endorsed for use by ASME or other codes that may be used in storage system designs.

The applicant should describe proprietary materials important to safety (specifically neutron poisons and polymeric neutron shields) adequately for the staff to make a safety finding. The reviewer should ensure that the technical specifications incorporate by reference the governing quality assurance and quality control documents, key manufacturing procedures, and key testing protocols for proprietary materials. The use of proprietary materials should be reviewed by NRC on a case-by-case basis.

The applicant may specify non-important-to-safety items by generic names such as “stainless steel,” “aluminum,” or “carbon steel,” provided that the reviewer has sufficient information to evaluate potential impacts that components that are not important to safety may have on components of packaging that are important to safety (e.g., galvanic corrosion).

8.5.2.2 *Code Case Use and Acceptability*

The reviewer should assess any referenced ASME B&PV Code cases against Regulatory Guide (RG) 1.193, “ASME Code Cases Not Approved for Use.” Note that the NRC has found Code Case N-595 (any revision) unacceptable. The reviewer should also review any referenced ASME B&PV Code cases against RG 1.84, “Design, Fabrication, and Materials Code Case Acceptability, ASME Section III.” Table 1 of RG 1.84 provides lists of cases acceptable to the NRC, while Table 2 of RG 1.84 provides a list of conditionally approved cases. The reviewer should verify that all of the supplemental requirements are met in order to provide an acceptable level of quality and safety. Also examine Tables 3, 4, and 5 of RG 1.84 to ensure that they do not reference annulled or superseded codes cases.

8.5.3 Welding

The ASME B&PV Code defines required welding criteria, including welding processes, filler metal, qualification procedures, heat treatment, and examination and testing. Review the relevant portions of the ASME B&PV Code to ensure that the SAR and drawings for the storage confinement boundary and fuel baskets are consistent with the code-required welding criteria. This review should include the relevant articles in ASME B&PV Code, Section III, Subsection NB-4000 and, in particular, Article NB-4330, “General Requirements for Welding Procedure Qualification Tests.” Although written for the welding of shipping containers, NUREG/CR-3019, “Recommended Welding Criteria for Use in the Fabrication of Shipping Containers for Radioactive Materials,” is a relevant resource for identifying the locations in the ASME B&PV Code of the welding criteria for storage containers.

The welding of SSCs not associated with the confinement boundary or fuel baskets are frequently governed by the ASME B&PV Code (transfer casks constructed per Section III Subsection NC) or

AISC standards (canister support structures), which, in turn, may reference AWS Codes. Similar to the ASME B&PV Code, AWS D1.1, “Structural Welding Code-Steel,” and AWS D1.6, “Structural Welding Code-Stainless Steel,” provide detailed welding criteria and weld procedure qualification requirements.

If the DSS or DSF design is consistent with the ASME or AWS codes, and the SAR and design drawings clearly define the applicability of the code, there is no need to review and verify the presence of specific welding criteria, such as the filler metal and weld process, in the drawings. The staff considers the ASME and AWS codes to have been proven to be effective in controlling qualification methodology, materials, heat treating, inspection, and testing. Note that this guidance is only applicable if the materials of construction comply with the ASME or AWS codes.

If materials and welding processes are not fully consistent with the ASME or AWS codes, verify that the application provides a technical basis for the integrity of the non-code welds and that the SAR and drawings sufficiently describe the welding criteria. The technical basis for non-code welds should demonstrate that the alternative material or welding process has been qualified in a manner similar to that described in accepted codes. The specified weld metal strength should equal or exceed the specified base metal strength. In addition, filler metals and the welding parameters should be selected in consideration of the potential for microstructural phase instabilities in the weld and the weld heat affected zone. These microstructural changes may include the formation of secondary or intermetallic phases that reduce ductility or fracture toughness and/or increase the susceptibility of the weld or the weld heat affected zone to environmental degradation such as corrosion or stress corrosion cracking. For example, welding of austenitic stainless steels that are not low carbon grades can result in sensitization of the weld heat affected zones, which can increase susceptibility to intergranular corrosion and stress corrosion cracking in corrosive environments. Secondary phase formation in duplex stainless steels as a result of slow cooling during welding can significantly reduce fracture toughness.

Detailed guidance is provided below for welds associated with the confinement boundary. Confinement boundary welds provide both structural integrity and confinement leak tightness. Ensure that the applicant provided sufficient detail to demonstrate that the welds are capable of fulfilling these functions. The guidance for the design, inspection, and testing of confinement boundary welds follows the ASME B&PV Code as practicable; however, exceptions are allowed to accommodate the unique application of the codes to DSSs.

8.5.3.1 *Confinement Weld Design*

The preferred construction code for the storage confinement boundary is the ASME B&PV Code, Section III, Division 1, Subsection NB for Class 1 nuclear facility components. ASME B&PV Code Section III is supplemented by supporting code sections that detail how special processes such as welding and NDE are to be qualified and executed. ASME B&PV Code Section IX, “Welding, Brazing, and Fusing Qualifications,” details the requirements for specifying and qualifying a welding procedure and for testing and qualifying welders. ASME B&PV Code Section V, “Nondestructive Examination,” describes the required qualifications for NDE examiners and the requirements and methods for performing NDE.

Review the relevant articles in ASME B&PV Code Subsection NB-2000 to verify that the applicant specified the appropriate testing requirements for materials of construction of the confinement boundary. In addition, verify that the confinement boundary welds are full-penetration welds, constructed in accordance with ASME B&PV Code Subsection NB-4240 requirements, with the following exception: Because of the difficulty with the fabrication of full-penetration welds for some joint geometries, canister top closure welds may be partial-penetration welds. These excepted welds include the shell-to-top cover welds and the welds associated with siphon and vent port covers.

8.5.3.2 *Confinement Weld Inspection*

Inspections are performed to verify the structural integrity of the welded joints. ASME B&PV Code Subsection NB-5200 requires welds to be inspected by both volumetric and surface techniques. Volumetric techniques may include either radiographic (RT) or ultrasonic (UT) testing. Surface techniques may include either liquid penetrant (PT) or magnetic particle testing (MT). Note that magnetic particle testing is applicable only to ferromagnetic materials such as carbon and low-alloy steels. The applicant should examine austenitic and duplex stainless steel canisters by the liquid penetrant method.

For certain welds, progressive surface examinations may be performed during the buildup of the weld in lieu of the post-weld volumetric examination. This exception is permitted when the geometry of the joint or the material prevent effective volumetric examinations. For example, there currently are no approved techniques for the volumetric examination of fillet welds associated with the austenitic stainless steel canister shell-to-lid joint. A progressive surface examination is defined as performing an examination of weld deposit layers at pre-calculated intervals in addition to the surface examination of the root and final weld layers.

8.5.3.2.1 Austenitic Stainless Steel Closure Lid Welds

The progressive, or multipass, surface examinations of austenitic stainless steel structural welds may be used in lieu of the volumetric examination provided that the following conditions are met:

- Structural calculations apply a stress-reduction factor of 0.8 to the allowable design stress to account for imperfections or flaws that may be missed by progressive surface examinations.

- The interval between surface examinations during the buildup of the weld are calculated as follows:

    1. Calculate the critical flaw size (depth) assuming a buried flaw. Postulate a full circumferential (360-degree) flaw. Use the requirements in ASME Section XI, “Rules for Inservice Inspection of Nuclear Power Plant Components,” Division 1, IWB 3600, for alternative flaw acceptance criteria. Use of J-integral or net section stress is acceptable. Verify that the analysis is consistent with the expected failure mode. The approach used in ASME B&PV Code Section XI Nonmandatory Appendix C, “Evaluation of Flaws and Piping,” may be reviewed for guidance.

    2. Establish the maximum allowable surface examination interval by using the critical flaw depth calculated in Step 1.

    3. PT the root layer, every intermediate layer established in Step 2, and the final weld layer. It is assumed that the root layer is single pass. If the root layer is multipass, calculate the critical flaw depth (Step 1) to establish the maximum allowable intermediate weld deposit depth inspection interval. Assume a surface connected flaw when calculating the critical flaw depth for a multi-pass root layer.

Regarding criterion (3), verify that, if the applicant desires to use a thicker root pass, the applicant should limit the amount of weld deposit to the ratio of the fracture toughness K values (or J values) for the different flaw types (buried K divided by surface K) multiplied by the maximum depth. This will limit the depth of the root pass to the critical flaw size for a surface connected flaw. Thus, if an applicant desires to use a thicker weld deposit for the root pass, then a limiting flaw size analysis establishes a structural basis.

The staff recognizes that, for stainless steel, K, or even J, is not entirely correct for evaluating failure in austenitic stainless steel due to the large capacity for plastic deformation. Generally the result is failure due to net section stress, not fracture. However, the stress intensity ratio suggested above is acceptable for this purpose.

Evaluate the applicant's analysis of the critical flaw size using the above methodology based on service temperature, dynamic fracture toughness, and critical design stress parameters as specified in ASME Section XI, Division 1.

8.5.3.2.2 Duplex Stainless Steel Closure Lid Welds

The progressive, or multipass, surface examinations of duplex stainless steel structural welds may be used in lieu of the volumetric examination provided that the following conditions are met:

- Structural calculations apply a stress-reduction factor of 0.8 to the allowable design stress to account for imperfections or flaws that may be missed by progressive surface examinations.

- The interval between surface examinations during the buildup of the weld are calculated using the critical flaw size as described in Section 8.5.3.2.1 above.

Verify that the applicant included specific qualification testing and acceptance criteria for duplex stainless steel welds that are consistent with the assessment of the critical flaw size. For example ASTM A923-14, "Standard Test Methods for Detecting Detrimental Intermetallic Phase in Duplex Austenitic/Ferritic Stainless Steels," may be used to define acceptance criteria for impact toughness testing of base metal, welds, and weld heat affected zones.

8.5.3.2.3 Carbon and Low-Alloy Steel Closure Lid Welds

Verify that UT examination of the structural lid weld is in accordance with the ASME Section III, Division 1, Subsection NB-5000 requirements and acceptance criteria.

Note that the NRC can approve progressive surface examinations utilizing a PT or MT on a case-by-case basis only if unusual design and loading conditions exist. For progressive PT or MT without a volumetric NDE of the closure lid welds, a stress-reduction factor of 0.8 is imposed on the weld strength of the closure joint to account for imperfections or flaws that may have been missed by progressive surface examinations. Verify that the applicant has determined an allowable interval between surface examinations during the buildup of the weld using an assessment of the critical flaw size, as discussed in Section 8.5.3.2.1.

In addition, also verify that the applicant has considered all the closure lid weld material and technique improvements that accrued from previous DSS design and fabrication experience. For example, refer to the technical evaluation in NRC Confirmatory Action Letter 97-7-001, where instances of cracking of ASTM SA-516 Grade 70 steel welds led to improvements, such as the use of low-hydrogen electrodes, low-carbon equivalent materials, and maintenance of proper preheat and postheat treatments.

8.5.3.3 *Confinement Weld Testing*

The entire confinement boundary should be pressure tested by either hydrostatic or pneumatic methods to the requirements of ASME B&PV Code Section II, Division 1, Subsections NB-6220 or 6300, respectively.

Following the application of the test pressure for the required time, all joints, connections, and regions of high stress, such as regions around openings and thickness transition sections, should be visually examined for leakage. This visual examination shall be performed in accordance with ASME Code requirements and shall be performed at a pressure equal to or greater than the design pressure or three-fourths of the test pressure. This pressure test and visual examination applies to both the canister body constructed at a fabrication facility and the lid-to-shell welds fabricated and closed in the field.

8.5.3.3.1 Pressure Testing

The entire confinement boundary should be pressure tested by either hydrostatic or pneumatic methods to the requirements of ASME B&PV Code Section III, Division 1, Subsections NB-6220 or 6300, respectively.

Following the application of the test pressure for the required time, all joints, connections, and regions of high stress, such as regions around openings and thickness transition sections, should be visually examined for leakage. This visual examination shall be performed in accordance with ASME Code requirements and shall be performed at a pressure equal to or greater than the design pressure or three-fourths of the test pressure. This pressure test and visual examination applies to both the canister body constructed at a fabrication facility and the lid-to-shell welds fabricated and closed in the field.

8.5.3.3.2 Helium Leakage Testing

The applicant should conduct a helium leakage test of the confinement boundary in accordance with ANSI N14.5, "American National Standard for Leakage Tests on Packages for Shipment of Radioactive Materials," with an allowed exception discussed below. The leakage test provides reasonable assurance that the confinement body is free of defects that could lead to a leakage rate greater than the allowable design-basis leakage rate specified in the confinement analyses. This ensures that the following conditions are met:

- The helium inerting gas will remain in the canister in sufficient quantity over the license period to protect the fuel assembles and cask or canister internals from the deleterious oxidizing effects of moisture.

- The helium gas heat transfer medium will remain in sufficient quantity over the licensing period to assure that fuel cladding temperatures are controlled at safe levels.

The applicant should test the confinement boundary at the fabrication shop to the extent practicable. Leakage testing of lid-to-shell welds and welds associated with the siphon and vent ports may be tested in the field by the cask user.

The large lid-to-shell confinement boundary field welds of austenitic stainless steel canisters with redundant confinement closures may be excepted from the leakage testing, provided that the following conditions are met:

- The weld is multipass with at least three distinct weld layers. Each layer should be complete across the width of the weld joint and may be composed of one or more adjacent weld beads.
- If only three weld layers comprise the full thickness of the weld, each layer is PT examined.
- For more than three weld layers, not all weld layers need to be PT examined. The maximum weld deposit depth allowed before a PT examination is necessary is based upon flaw-tolerance calculations described in the volumetric examination exception discussion in Section 8.5.3.2.1. Regardless, at least three different weld layers should be examined (e.g., the root pass, a mid-layer, and the cover pass).
- The weld cannot have been executed under conditions where the root pass might have been subjected to pressurization from the helium fill in the canister itself.

The above exception to the leakage testing requirement is not applicable to the siphon and vent port covers. It is assumed that mechanical closure devices (e.g., a valve or quick-disconnect) permit helium leaks. Consequently, welds potentially subjected to helium pressure by way of leakage through a mechanical closure device should be subsequently helium leak tested.

8.5.3.3.3 Leakage Testing Review Examples

The redundant weld requirement for the confinement system closure creates two closure boundaries. Verify that at least one of the redundant boundaries is helium leakage tested, or that some closure welds are leakage tested and the remaining closure welds of the same boundary designed so that the above leakage test exception criteria are met. Only a boundary that is testable or excluded from testing per this guidance should be considered the confinement boundary of the redundant closures. The application of these criteria to two currently approved designs is provided here.

Leakage Testing of a Single Lid with Cover Plate Design (Figure 8-2)

In Figure 8-2, the dotted line marked (1) defines one closure boundary. Starting on the left side of the sketch, the closure boundary can be traced from the canister shell, through the large, multipass weld joining the canister shell to the combined shield and structural lid. The boundary continues through the lid to the small weld joining the lid to the vent-and-drain-port closure plate, and back to the lid. For all cases, the remainder of the boundary (and sketch) is assumed to be symmetrical with or similar to the half-sketch portion that is shown.

This boundary demonstrates confinement integrity by means of the large multipass weld leakage exception criteria for the canister shell-to-lid weld and by helium leakage testing of the small vent-and-drain-port closure plate weld. The large, canister shell-to-lid weld is exempted from the helium leak test because it is a multipass weld meeting the flaw tolerance and other appropriate portions of this guidance. Note that this weld is executed before filling the canister with helium (excluding purging and welding gas, as applicable).

Before the remaining welds of this first closure boundary are executed, the canister is drained, dried, purged, and filled with helium to the design operating pressure. The helium line connection is closed off and the vent or drain port closure plate is welded into place. Since the vent or drain port closure weld may have been pressurized from the helium fill gas because of assumed leakage from the closure valve, it should be helium leakage tested in accordance with the methods described in ANSI N14.5.

This completes the first closure boundary. Here again, one weld was exempted from the helium leak test by the design criteria, and the other weld was leak tested. This closure boundary demonstrates compliance with regulatory requirements and is consistent with the staff guidance by ensuring at least one of the two redundant closure boundaries is leak tested or exempted from leak testing by conformance with the multipass weld exception guidance.

The second boundary, delineated by line (2), can be traced from the canister shell on the left side of the sketch up through the fillet weld joining the canister shell to the structural lid cover plate. The boundary continues through the cover plate to the fillet weld joining the cover plate to the canister lid. The welds joining the cover plate to the canister shell and lid cannot be helium leak tested since there is no feasible means to do so. However, since the first closure boundary, delineated by line (1), was tested (or exempted through design), the need to helium leak test at least one of the closure boundaries has been satisfied. Since this second boundary does not meet all the criteria for a confinement boundary, it may not be designated as the confinement boundary. The first closure is thereby the confinement boundary in this design, as it meets all the applicable criteria for a confinement boundary.

<u>*Leakage Testing a Dual Lid Design (Figure 8-3)*</u>

In Figure 8-3, the dotted line marked (1) defines one of the redundant closure boundaries. It may be traced from the canister shell on the left side of the sketch. The boundary proceeds through the partial penetration weld joining the canister wall to the shield lid and into the shield lid. It continues through the small fillet weld joining the vent or drain port cover plate and back through the same fillet weld to the shield lid.

This closure boundary may satisfy the leak test guidance by several methods, depending on details of the weld design. The canister shell-to-shield-lid weld may be designed in several ways. The weld may be a small seal weld, which would necessitate subsequent helium leak testing. Conversely, it could be a large, multipass weld consistent with the leakage test exception guidance described in this chapter. Either way, note that this weld (canister-to-shield-lid weld) is executed before filling and pressurizing the canister with helium (use of purge or backing gas for welding operations is not considered filling or pressurizing).

Next, the canister is drained, dried, purged, and filled with helium to the design operating pressure. The helium line connection is closed off. The vent or drain port cover plate is welded into place. Since this weld may potentially be pressurized from the helium fill gas because of assumed leakage through the closure valve, it should be helium leakage tested.

This completes the first closure boundary. Note that one weld was either helium leakage tested or excepted from the leak test by the design criteria. The other weld was leak tested. Thus, this closure boundary demonstrates compliance with regulatory requirements and is consistent with staff guidance by ensuring at least one of the two redundant closures is leak tested or excepted by conformance to this guidance. This closure may therefore be designated as the confinement boundary.

The secondary boundary, delineated by line (2), can be traced from the canister shell on the left side of the sketch up through the canister shell-to-structural-lid weld and into the structural lid. The weld joining the canister shell and structural lid cannot be helium leakage tested because helium is not present. Note, however, that this weld may comply with the leakage testing exception criteria described in this chapter. In this case, the second closure also qualifies for designation as the confinement boundary.

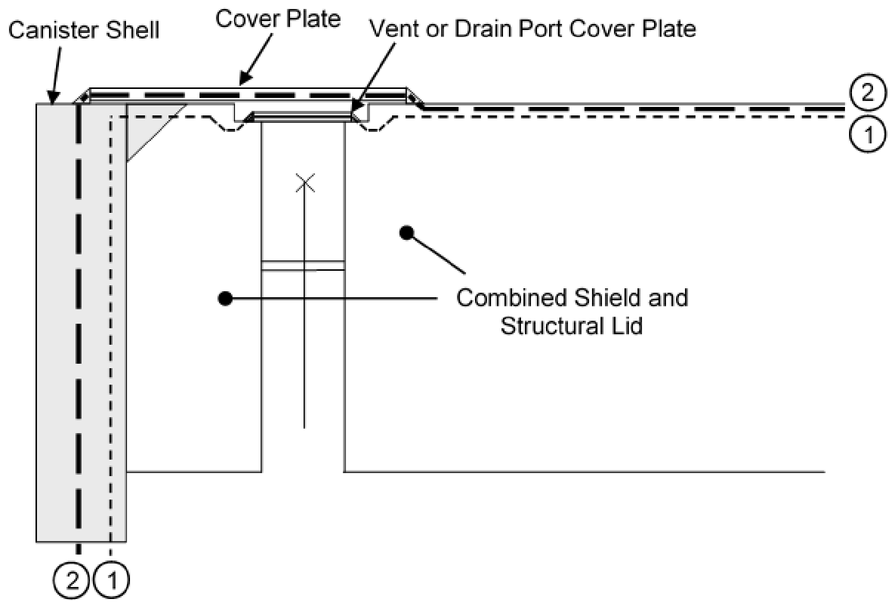

Figure 8-2 Single Lid with Cover Plate Design

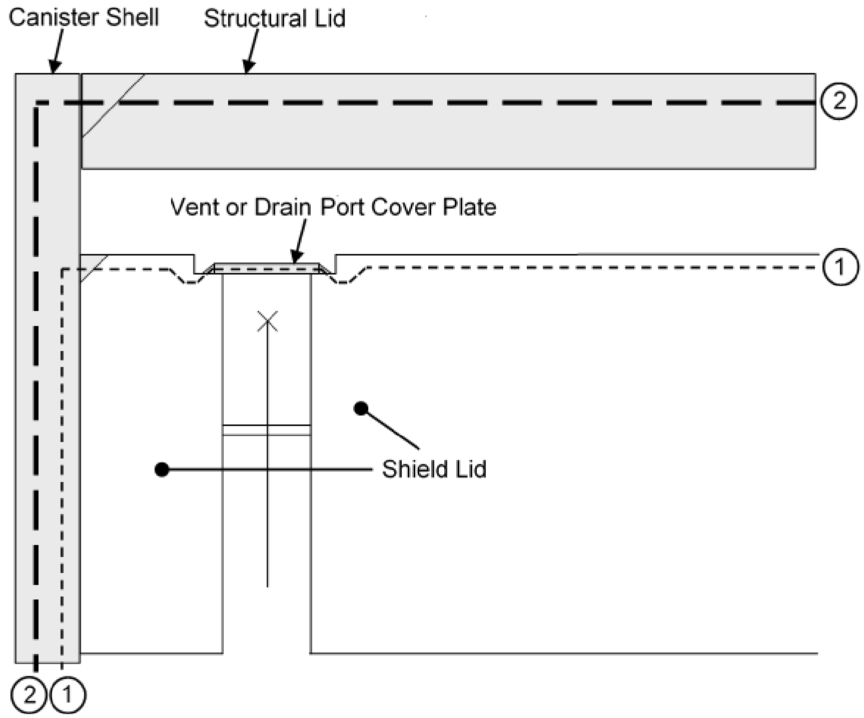

Figure 8-3 Dual Lid Design

For this design in Figure 8-3, the designer therefore has the freedom to designate either of the redundant closures as the confinement boundary. Only one of the two closures is designated as the confinement boundary.

8.5.4 Mechanical Properties of Metals

Assess the acceptability of all material mechanical properties for SSC subcomponents that have a structural role, that is, are relied on for maintaining the analyzed configuration for the stored SNF, HLW or reactor-related GTCC waste (e.g., canister, cask, basket, overpack), excluding the SNF subcomponents. Ensure that the mechanical properties used in the structural evaluation are adequate upon consideration of the environmental and operating conditions during the requested license or storage term (e.g., 40 years), including loading, transfer, storage, and retrieval operations.

8.5.4.1 *Tensile Properties*

Verify that the SAR clearly references acceptable sources of all material properties. Acceptable material properties, allowable stresses, temperature limits, and other requirements include those provided in ASME B&PV Code Section II, Part A, “Ferrous Metals;” Part B, “Nonferrous Metals;” Part C, “Welding Rods, Electrodes, and Filler Metals;” and Part D, “Properties.” The use of certified material test reports for defining mechanical properties is generally not permissible. These property values may be nonconservative because samples may be taken at a portion of the ingot, billet, or forging that have optimum materials properties during certification. Coordinate with the structural reviewer (SRP Chapter 4, “Structural Evaluation”) if the applicant selected inadequate material properties.

Confirm that the SAR and SSC drawings identify the design criteria (codes, standards, specifications) for SSC subcomponents providing structural support. Verify that material mechanical properties used in the structural evaluation are consistent with the design criteria. For example, if an SSC subcomponent is designed to a particular subsection of ASME B&PV Code Section III, the material properties and requirements for the given SSC should be consistent with those allowed by that subsection.

The application may contain a tabulated list of all materials used for SSC subcomponents providing structural support and the proposed service conditions during loading, transfer, storage, retrieval, and waste management operations. The tables may list the subcomponent name, safety classification, intended safety function, fabrication specification (i.e., grade, type and class of material), and material property values (e.g., elastic modulus, yield strength, tensile strength) assumed in the structural evaluation. This information may also be found in the design drawings and multiple tables, as applicable, across the SAR. Evaluate the assumed property values upon consideration of the thermal, radiation, or other applicable environments that may impact structural performance.

8.5.4.2 *Fracture Resistance*

The reviewer should be familiar with ASME Section III NB-2300, "Fracture Toughness Requirements for Material," when evaluating a new DSS or new material for an SSC. Metals having a face-centered, cubic-crystal structure, such as austenitic stainless steels, remain tough and ductile to very low temperatures and are not a concern in this regard. Note that ASME Section III NB-2311(a)(7) includes nonferrous material as material for which impact testing is not required. This notation only applies to nonferrous materials that are included in ASME Section II, Tables 2A and 2B. For some DSS designs, SSCs not part of the confinement boundary use materials that are not included in ASME Section II Tables 2A and 2B. In these cases, determine if fracture toughness testing of these materials is necessary. Review the materials that provide a structural function to determine adequate resistance to fracture.

The reviewer should verify that calculated values of fracture toughness using correlation equations based on impact toughness data such as Charpy V-notch toughness are appropriate for the materials considered. Numerous correlations have been developed for pressure vessel steels and other specific alloys (Roberts and Newton 1981). Ensure that the applicant justified the use of a correlation equation that was not developed for the alloy system used in a DSS SSC that has a structural function.

Because embrittlement of metals may occur under exposure to neutron radiation, the NRC staff calculated the maximum potential accumulated neutron fluence on DSS components, considering components most directly exposed to the radiation source (middle of the fuel basket) and assuming fuel is loaded immediately after it is removed from the reactor vessel and stored for 100 years. To further provide a bounding estimate, the staff assumed a cask design that uses 40 Westinghouse 17 x 17 pressurized-water reactor (PWR) fuel assembles with an average burnup of 70 gigawatt days per metric ton of uranium (GWd/MTU) and 4.0 fuel enrichment. The staff calculated the neutron source term for neutrons with energy at or greater than 1 million (mega) electron volts (MeV) using the Origen/Arp computer code of the SCALE 6.1 computer code system. At this location, the total accumulated neutron fluence after 100 years of storage was calculated to be 2.63×10^{16} neutrons per square centimeter (1.70×10^{17} neutrons per square inch). This value is several orders of magnitude lower than fluence levels known to affect the mechanical properties of steel (Nikolaev et al. 2002; Odette and Lucas 2001), stainless steel

(Gamble 2006; Caskey et al. 1990), and aluminum (Farrell and King 1973; Alexander 1999). As a result, there is no need to consider the effects of irradiation on the fracture resistance of these metals.

8.5.4.2.1 Ferritic Steels

Several types of ferritic steels may become brittle at low service temperatures. ASME B&PV Code, Section III, contains requirements for material fracture toughness; however, these requirements were developed for reactor components and do not address hypothetical accident conditions for storage systems (e.g., impacts at low temperatures). Therefore, in the evaluation of ferritic steels, refer to the guidance for fracture toughness criteria and test methods described in RG 7.11, "Fracture Toughness Criteria of Base Material for Ferritic Steel Shipping Cask Containment Vessels with a Maximum Wall Thickness of 4 Inches," and RG 7.12, "Fracture Toughness Criteria of Base Material for Ferritic Steel Shipping Cask Containment Vessels with a Wall Thickness Greater Than 4 Inches, But Not Exceeding 12 Inches."

RG 7.11 and RG 7.12 specify the types of tests and data needed to qualify a material for designs that specify ferritic steels other than those listed. Those tests and data include dynamic fracture toughness and nil-ductility or fracture appearance transition temperature test data. Toughness testing (e.g., Charpy impact) of welds is governed by the ASME B&PV Code, Section III, as supported by Section IX.

8.5.4.2.2 Duplex Stainless Steels

Duplex stainless steel has both ferritic and austenitic phases and are susceptible to phase instability that may affect fracture toughness. Verify that the applicant included specific qualification testing and acceptance criteria for duplex stainless steel welds that are consistent with the assessment of the critical flaw size. For example, ASTM A923-14, "Standard Test Methods for Detecting Detrimental Intermetallic Phase in Duplex Austenitic/Ferritic Stainless Steels," may be used to define acceptance criteria for impact toughness testing of base metal, welds, and weld heat affected zones.

8.5.4.3 *Performance of Aluminum Components*

Storage container basket assemblies use aluminum alloys, aluminum-based metal matrix composites (MMCs), and laminates consisting of aluminum and boron carbides (e.g., Boral^{TM}) and are particularly susceptible to property changes at elevated temperatures. Thus, use the detailed guidance below to verify that the DSS or DSF design uses appropriate aluminum properties, considering all service temperatures.

8.5.4.3.1 Tensile Properties of Aluminum

The reviewer should verify that the applicant considered appropriate tensile properties for storage container basket components with a structural function and manufactured from aluminum alloys. There are a variety of aluminum alloys that have been used in basket construction. Some aluminum alloys, such as Al 6061, are precipitation-hardened and are commercially available in several tempers with significantly different yield and tensile strengths and ductility values. Al 6061 has magnesium and silicon as its major alloying elements and is available in pre-tempered grades such as annealed 6061-O and tempered grades such as 6061-T6 and 6061-T651. Good combinations of strength and ductility are obtained in Al 6061 by heat treating it to induce a finely distributed precipitate of magnesium silicide phase. The T6 condition consists of annealing at

532 degrees Celsius (°C) (990 degrees Fahrenheit (°F)) for 1 hour, quenching in water to room temperature, then aging (tempering) at 160 °C (320 °F) for 18 hours to precipitate the magnesium silicide phase.

Elevated temperatures can affect the properties of Al 6061. Temperature affects the allowable stress for all tempers including T4, T451, T6, and T651, but especially for the T6 and the T651 tempers. Aging at higher temperature or holding at higher temperature after aging at 160 °C (320 °F) will coarsen the magnesium silicide precipitates and correspondingly reduce the strength of the alloy (Farrell 1995). Verify that the mechanical properties used for precipitation-hardened aluminum alloys for structural components exposed to elevated temperatures account for the microstructural changes that affect yield and tensile strength. Note that ASME Section II, Part D, Table 1B, requires the use of time-dependent properties for precipitation-hardened Al 6061 at temperatures at or above 177 °C (350 °F).

8.5.4.3.2 Fracture Resistance of Aluminum

The fracture toughness of traditional aluminum alloys varies widely and is dependent on composition and alloy condition for heat-treated or precipitation-hardened aluminum alloys. Compare the applicant's reported value of fracture toughness to tabulated values in materials handbooks and peer-reviewed publications as necessary (e.g., ASM Metals Handbook Desk Edition; Kaufman et al. 1971).

The fracture toughness of aluminum MMCs depend on many factors including (1) particle composition, (2) particle size, (3) particle loading, (4) particle distribution or clustering, (5) alloy composition, and (6) alloy condition for aluminum alloys that can be age hardened. The fracture toughness of aluminum MMC has been found to range from 8 to 30 ksi·in$^{1/2}$ (Flom et al. 1989; Flom and Arsenault 1989; Lewandowski 2000; Miserez 2003; Rabiei et al. 2008). Verify that the applicant has assessed the fracture resistance of aluminum MMCs for structural applications using valid fracture toughness data.

8.5.4.3.3 Creep Behavior of Aluminum

More recent storage system designs have specified ever higher design temperatures for the fuel basket components in order to accommodate higher loading densities and higher burnup fuel. This trend has pushed the various aluminum components well into creep regime operating temperatures.

Review the design maximum temperatures and stresses for aluminum components and verify that the applicant has performed a creep analysis if any structural load-bearing aluminum components operate at a design temperature above approximately 93 °C (200 °F). In the event temperatures exceed the ASME B&PV Code, Section II, nominal 204 °C (400 °F) temperature limit for aluminum, other sources for creep data should be examined. One previously cited reference for this information is Wilson et al. (1969); however, the reviewer should recognize that designs evaluation through the time of this writing have had design stresses (on the order of tens of pounds per square inch) that were substantially below the creep-rupture stresses provided in the referenced report. Nevertheless, ensure that the design calculations include an assessment of creep deformation over the storage period.

Borated aluminum neutron poison materials should be considered on a case-by-case basis if they are subjected to structural load bearing beyond their own dead-weight loads. These materials have inherently low ductility and generally unknown creep properties.

8.5.5 Thermal Properties

Coordinate with the thermal reviewer (SRP Chapter 5, "Thermal Evaluation") to determine the properties of the materials important to the thermal analysis. Verify the material compositions and thermal properties, such as thermal conductivity, thermal expansion, specific heat, and heat capacity, as a function of the temperature over the range in which the components are to operate. Verify that the applicant has evaluated the potential change in these material properties due to material degradation over their service life. Temperature and anisotropic dependencies of thermal properties should be considered.

8.5.6 Radiation Shielding Materials

8.5.6.1 *Neutron Shielding Materials*

Boron-filled polymers are often used for neutron shielding materials. Dose limits are calculated at the site or controlled area boundary, as applicable, and not the canister surface; therefore, these materials are considered important to safety.

Ensure that the SAR describes the composition and geometries of shielding materials. Ensure that the SAR includes references for all materials used, including nonstandard materials (e.g., proprietary neutron shield material), for the source of the material composition and density data along with validation of the data.

In-service performance monitoring of these materials typically is conducted during periodic radiation surveys. Should a decline in the shielding effectiveness be detected, the staff expects that there will be enough time and opportunity for engineering evaluation and corrective action. Therefore, the qualification and acceptance testing of neutron shielding materials are not expected to be included in the technical specifications.

8.5.6.2 *Assessing Previously Unreviewed (New) Neutron Shielding Materials*

Confirm that temperature-sensitive (e.g., polymeric) neutron shielding materials will not be subject to temperatures at or above their design limits during normal conditions. Determine whether the applicant properly examined the potential for shielding material to experience changes in material densities at temperature extremes. For example, elevated temperatures may reduce hydrogen content through loss of water in concrete or other hydrogenous shielding materials.

With respect to polymeric neutron shields, verify that the SAR describes the following:

- the test(s) demonstrating the neutron-absorbing ability of the shield material
- the testing program, data, and evaluations that demonstrate the thermal stability of the resin over its design life while at the upper end of the design temperature range
- the nature of any temperature-induced degradation and its effect(s) on neutron shield performance
- what provisions exist in the neutron shield design to assure that excessive neutron streaming will not occur as a result of shrinkage under conditions of extreme cold; this description is required because polymers generally have a relatively large coefficient of thermal expansion when compared to metals

- any changes or substitutions made to the shield material formulation; for such changes, describes how they were tested and how that data correlated with the original test data regarding neutron absorption, thermal stability, and handling properties during mixing and pouring or casting

- the acceptance tests conducted to verify that any filled channels used on production storage containers did not have significant voids or defects that could lead to greater-than-calculated dose rates (see SRP Section 12.5.2.4, "Shielding Tests")

- the material's ability to withstand the effects of heat and irradiation (e.g., the possibility of heat and radiation altering polymer structures to reduce ductility and fracture toughness, and also creating gaseous products such as hydrogen)

Confirm that the SAR describes the potential for shielding materials to experience changes in material properties at temperature extremes and accumulated radiation exposure.

8.5.6.3 *Gamma Shielding Materials*

Concrete, steel, cast iron, uranium, and lead typically serve as gamma radiation shields. Collaborate with the shielding reviewer (SRP Chapter 4) to ensure that the material compositions and densities used in the shielding models are consistent with the design features described in the SAR. The shielding properties should account for manufacturing tolerances and expected degradation from corrosion reactions, elevated temperature, and accumulated radiation exposure.

Confirm that the SAR describes the physical dimensions of shielding materials, including seams, penetrations, or voids. For example, lead shielding may be put into place as stacked bricks or plates, and lead wool is occasionally used to fill gaps. Ensure that manufacturing controls are in place to address any potential paths for gamma streaming.

8.5.7 Criticality Control Materials

Various materials containing boron are used in the nuclear industry as neutron absorbers for criticality control. Neutron absorbers can consist of alloys of boron compounds with aluminum or steel in the form of sheets, plates, rods, liners, and pellets. Likewise, neutron absorbers can consist of a core containing mixed aluminum and boron carbide particles, clad on both sides with aluminum (a composite).

8.5.7.1 *Neutron Absorbing (Poison) Material Specification*

The neutron absorber material must be demonstrated to be adequately durable for the service conditions of the application (10 CFR 72.124(b)). The materials should have excellent physical and chemical stability, including a high resistance to radiation and corrosion. Further, these materials should experience no reduction in effectiveness under normal, off-normal, and accident conditions. These assurances are usually obtained during qualification testing of the material. In addition, acceptance tests (SRP Chapter 12, "Conduct of Operations Evaluation") are performed on samples from each production run of the material. This procedure will ensure that the properties for the plates or other shapes produced are in compliance with the specifications and requirements of the application. The uniformity of the distribution of boron-10 may be addressed in both the qualification and the acceptance tests.

For all boron-containing absorber materials, verify that the SAR and its supporting documentation describe the absorber material's chemical composition, physical and mechanical properties, fabrication process, and minimum poison content. If the applicant intends to use an absorber material with a specific trade name, verify that the manufacturer's data sheet is submitted to supplement the above information. In the case of absorber plates or sheets, the SAR should specify the minimum poison content as an areal density (e.g., milligrams of boron-10 per cm^2).

Qualification testing of neutron absorber (poison) materials is conducted to ensure the following:

- The material used will have sufficient durability for the application for which it has been designed.

- The physical characteristics of the components of the absorber materials will meet the design requirements, and the uniformity of the distribution of boron-10 is sufficient to meet the requirements of the applications for which the absorber materials will be used. Materials that have passed the qualification tests should be acceptance tested (see SRP Chapter 12) for use in systems to be employed in the storage or transport of nuclear fuel.

ASTM C1671-15, "Standard Practice for Qualification and Acceptance of Boron Based Metallic Neutron Absorbers for Nuclear Criticality Control for Dry Cask Storage Systems and Transportation Packaging," with some exceptions, additions, and clarifications, is considered appropriate for staff use in its review activities. Appendix 8A, "Clarifications, Guidance, and Exceptions to ASTM Standard Practice C1671-15," to this SRP provides the exceptions, additions, and clarifications to this standard. The use of ASTM C1671 is not a regulatory requirement; alternative approaches are acceptable if technically supported.

8.5.7.2 *Computation of Percent Credit for Boron-Based Neutron Absorbers*

This section illustrates one method used by the materials reviewer to compute the level of credit to be allowed for 1/v neutron absorber materials[1] in the criticality safety analysis of packages for storing fissile materials, including fresh nuclear fuel and SNF. The allowed level of credit uses the results of neutron attenuation measurements performed on samples of the absorber material placed in a beam of thermal neutrons.

The staff has accepted an upper limit of 90-percent credit to be applied to boron-based solid absorbers, meaning that the material is computationally modeled as containing only 90 percent of the boron-10 shown to be present. The staff has concluded that limiting the poison credit to 90 percent adequately accounts for the uncertainties arising in extrapolating the validation for boron-based absorber materials.

Neutron channeling has been shown to occur in a commercial product that uses coarse particles of B_4C dispersed in an aluminum matrix. The nonuniformities and channeling effects for heterogeneous absorber materials further limit the poison credit to levels below 90 percent. For

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1 Involves that region at the low end of the neutron energy spectrum where neutron absorption is inversely proportional to particle velocity.

heterogeneous absorber materials, the reviewer should verify the applicant's value for poison credit using the following definitions and equations:

A_a = manufacturer's acceptance value of neutron absorber density based on neutron attenuation measurements,

T = lower tolerance limit of neutron absorber density as calculated in ASTM C1671-15.

The value of A_a should be based on a qualified homogeneous absorber standard such as zirconium diboride, or a heterogeneous calibration standard that is traceable to nationally recognized standards, or calibrated with a monoenergetic neutron beam to the known cross section of boron-10. Calibration standards should be evaluated at 111 percent (i.e., 1/0.90) of the poison density assumed in the criticality computational model.

Thus, in addition to the 90-percent limit on poison credit that is used to offset validation uncertainties for all absorbers, the additional penalty for heterogeneous absorbers should be calculated as follows:

If $T \geq A_a$, then 90-percent credit is given.

If $T < A_a$, then compute the fractional credit from 0.75 to 0.90 as follows:

Fractional Credit = $0.30 + 0.6(T / A_a)$.

If the fractional credit is less than 0.75, the absorber is regarded as unsuitable and should be given no credit.

Other remedies beyond the scope of this guidance may be necessary in addressing the potentially more complex neutron-spectral effects and validation uncertainties encountered with materials based on non-1/v-absorbers such as cadmium or gadolinium. The current guidance applies only to 1/v absorbers such as boron or lithium.

8.5.7.3 *Qualifying the Neutron Absorber Material Fabrication Process*

For the qualification of properties not associated with neutron attenuation, in past reviews the staff has accepted the following qualification testing:

1. Mechanical testing to ensure that the neutron poison material is structurally sound, even if the absorber is not used for structural purposes.

   In the past, the staff has accepted ASTM B557, "Standard Test Methods for Tension Testing Wrought and Cast Aluminum- and Magnesium-Alloy Products," for the tensile testing of samples that demonstrated the following:

   – 0.2-percent offset yield strength no less than 1.5 thousand pounds per square inch (ksi)

   – ultimate strength no less than 5.0 ksi

   – elongation no less than 1 percent

Alternatively, the staff has accepted bend tests under ASTM E290, "Standard Test Methods for Bend Testing of Material for Ductility," with a 90-degree bend without failure as the passing criteria.

2. Porosity measurements to ensure that the corrosion resistance (which is directly linked to hydrogen generation in the spent fuel pool) of the neutron poison material is maintained, and that the general structural characteristics of the material are controlled.

   The methodology for porosity is up to the discretion of the applicant. The technical specifications should explicitly state limits on both the total porosity of the material and the "open" or "interconnected" porosity of the material. Excluding Boral™, the total open porosity of the neutron poison material should be limited to 0.5 volume percent or less.

   In general, the conditions of SNF loading, unloading, and storage do not require qualification testing to demonstrate resistance to thermal-, radiation-, or corrosion-induced degradation if the neutron absorber is only made of boron carbide and an aluminum alloy meeting ASTM chemical requirements for the 1000 or 6000 series of aluminum. Other aluminum alloys (particularly those that are not heat-treatable) may also be acceptable to the staff without qualification testing. However, porosity measurements on the neutron poison material should not be waived, regardless of the aluminum alloy used in the neutron absorber.

3. A sufficient number of samples should be used to measure the thermal conductivity of the neutron poison material at room and elevated temperature. Note that clad neutron poison materials are thermally anisotropic.

4. For clad materials, the qualifying tests should include a test demonstrating resistance to blistering during the drying process. In the past, the staff has accepted testing where samples of clad materials are soaked in either pure or borated water for 24 hours and then inserted into a preheated oven at approximately 440 °C (825 °F) for a minimum of 24 hours. The samples are then visually inspected for blistering and delamination before undergoing qualifying mechanical testing.

Additional qualifying tests should be conducted for structural neutron poisons. Mechanical and thermal tests should include tensile testing, impact testing (or K_{IC} measurements), creep testing, and (if applicable) mechanical testing of weldments.

Samples of neutron poison material (i.e., the use of transmission electron microscopy or scanning electron microscopy) should be examined for the following changes:

- redistribution or loss of boron
- dimensional changes (material instability)
- cracking, spalling, or debonding of the matrix from the boron-containing particles
- weight changes caused by leaching, dissolution, corrosion, wear, or off-gassing
- embrittlement
- chemical changes such as oxidation or hydriding
- molecular decomposition of the material as a result of radiation (radiolysis)

Coupons should be taken so as to be representative of the neutron poison material. To the extent practical, test locations on coupons should be stratified to minimize errors because of location or position within the coupon. Locations should include the ends, corners, centers, and irregular locations. These locations represent the most likely areas to contain variances in thickness. Adequate numbers of samples should be taken from components (e.g., plate, rod) produced from a lot to obtain a good representation. A lot is defined as all plates from a single billet. Overall, the coupons should be a representative sample of the material.

For containers that will be loaded or unloaded in a SNF pool or similar environment, verify that the applicant has evaluated or tested absorber material for environmental and galvanic interactions and the generation of hydrogen in the pool environment. If environmental testing is employed, the test conditions (time, temperature) should equal or exceed those expected for loading, unloading, and transfer operations. For environmental tests, the absorber materials should be coupled to dissimilar metals, as may be appropriate to the application. The environment may be borated or deionized water, as appropriate. The evaluation should also consider the effects of any residual pool water remaining in the container after removal from the pool. Generally, for common engineering materials, an evaluation based upon consultation of a corrosion reference (galvanic series) should suffice for pool loading and unloading situations.

The applicant should take appropriate measures to assess the strength or ductility of the material, depending on the structural requirements of the application.

Acceptance testing of the fabricated materials is discussed in Chapter 12 of this SRP.

8.5.8 Concrete and Reinforcing Steel

8.5.8.1 *Embedment Materials*

The reviewer should evaluate the material to be used for embedments, inserts, conduits, pipes, or other items embedded in the concrete. Embedments should satisfy the requirements of the code used in designing the reinforced concrete structure in which they are embedded (e.g., ACI 359, “Code for Concrete Reactor Vessels and Containments,” ACI 349, “Code Requirements for Nuclear Safety-Related Concrete Structures and Commentary,” or ACI 318, “Building Code Requirements for Structural Plain Concrete and Commentary”). Zinc, zinc-rich coatings, zinc-clad materials, and aluminum should not be used for any embedded objects in structures designed to ACI 349 or ACI 359 that will be in contact with wet concrete because of the potential for concrete degradation from an adverse chemical reaction. Embedments and attachments are considered to include components cast or grouted into the reinforced concrete structure, inserts, embedded pipes, conduits, or lightning protection and grounding systems.

Unless otherwise specified in this SRP, steel structural attachments should comply with the appropriate requirements in ACI 349.

8.5.8.2 *Concrete Design and Temperature Limits*

The NRC accepts the use of ACI 318 for the design and material specifications for reinforced concrete structures, although such structures typically are not important to safety. If ACI 349 is used for the design of such structures, the NRC accepts the use of ACI 318 for construction. The NRC also accepts the following criteria as an alternative to the temperature requirements of ACI 349, but only for the specified use and temperature ranges:

1. If concrete temperatures in general or local areas are a maximum of 93 °C (200 °F) in normal conditions, off-normal conditions, or occurrences, no tests are needed to prove capability for elevated temperatures or reduced concrete strength.

2. If concrete temperatures in general or local areas exceed 93 °C (200 °F) but are less than 149 °C (300 °F), no tests are required to prove capability for elevated temperatures or reduced concrete strength if Type II cement is used and temperature-appropriate aggregates are used. The following criteria for fine and coarse aggregates are acceptable:

   – Satisfy the requirements in ASTM C33, "Standard Specification for Concrete Aggregates," and requirements references in ACI 349 for aggregates.

   – Have a demonstrated coefficient of thermal expansion (tangent in temperature range of 20–38 °C (70–100 °F) no greater than 11×10^{-6} millimeter (mm)/mm/°C (6×10^{-6} inches (in.)/in./°F), or be one of the following materials: limestone, dolomite, marble, basalt, granite, gabbro, or rhyolite.

3. If concrete temperatures in general or local areas under normal or off-normal conditions do not exceed 107 °C (225 °F), the criteria 1 and 2 (above) apply to the coarse aggregate. Fine aggregate that meets 1 (above) and is also composed of quartz sands or sandstone sands may be used in place of 2 (above) and satisfy the criteria.

The strength and modulus of elasticity of concrete increase as it ages for about the first 10 years after fabrication (Washa et al. 1989; NRC 1996b). For example, for a normal weight concrete typical of that used in the construction of storage pads, strength has been shown to increase by 67 percent relative to the recorded 28-day strength. For drop and tipover events, such increases in concrete pad hardness can result in more severe accelerations on storage system components. The reviewer should ensure that changes in concrete properties with time are considered in structural calculations that evaluate the capability of SSCs to withstand design-basis accidents.

8.5.8.3 *Omission of Reinforcement*

Frequently, designers specify the omission of reinforcing steel ("rebar") in concrete aboveground structures that have the purpose of gamma shielding only. This is acceptable since it is to avoid the inadvertent formation of voids in the concrete because of the presence of the rebar, which can act to block the aggregate in the concrete from filling all intended areas.

Concrete applied around buried steel structures should be reinforced to alleviate shrinkage crack propagation. Concrete alleviates soil corrosion by creating a beneficial chemical buffering effect (high pH) around the steel, except in environments with high chloride concentrations. Cracks allow ground water plus electrolyte intrusion, which reduces the effectiveness of the concrete protective barrier.

8.5.8.4 *Radiation Damage*

Radiation effects on concrete properties depend on the gamma and neutron radiation doses, temperature, and exposure period. Gamma radiation can decompose and evaporate water in concrete (Bouniol and Aspart 1998). Because most of the water is contained in the cement paste, the effect of gamma radiation on cement paste is more significant than on the aggregates. Gamma radiation can also decompose the silicon monoxide bond within calcium silicate hydrate (Kontani et al. 2010). Neutron radiation deteriorates concrete by reducing stiffness, forming cracks by swelling, and changing the microstructure of the aggregates. This, consequently, reduces concrete strength (Kontani et al. 2010). The changes in aggregate microstructure also can lead to higher reactivity of aggregates to certain aggressive chemicals.

NUREG/CR-7171, “A Review of the Effects of Radiation on Microstructure and Properties of Concretes Used in Nuclear Power Plants,” provides a comprehensive review of the effects of gamma and neutron radiation on the microstructure and properties of concrete used in nuclear power plants. Concrete structures have been regarded as being sound as long as the cumulative radiation does not exceed critical levels over the life of the structure. In general, the critical radiation levels to reduce concrete strength and elastic modulus are considered to be approximately $1x10^{19}$ n/cm^2 ($6.5x10^{19}$ n/in^2) for fast neutrons (neutron energy greater than 1 MeV) and $1^{-2}x10^{10}$ rad ($1^{-2}x10^{8}$ grays) for gamma rays (Hilsdorf et al. 1978; EPRI 2012; IAEA 1998; ASME B&PV Code).

As discussed in Section 8.5.4.2 above, the maximum accumulated neutron fluence for any storage system SSC was estimated by the staff to be $2.63x10^{16}$ n/cm^2 ($1.70x10^{17}$ n/in^2) after 100 years of storage, which is three orders of magnitude below the level that would lead to a reduction of concrete strength and elastic modulus. The gamma dose is also expected to be several orders of magnitude less than the limits defined in the above references, per the specific DSS design bases.

Review the radiation damage analyses for concrete structures to determine that the critical radiation levels discussed above will not be exceeded during dry storage operations.

8.5.9 Bolt Applications

If threaded fasteners are employed for SSCs important to safety, verify that the bolt material(s) have adequate resistance to corrosion and brittle fracture and a coefficient of thermal expansion similar to the materials being bolted together. Also, verify that the fasteners have adequate creep resistance under expected service conditions.

For pressure-retaining and confinement boundary bolts, verify that the applicant has followed the requirements of ASME B&PV Code, Section III, NB-3230, “Stress Limits For Bolts,” and has used the mechanical properties, temperature limits and design stress intensity limits listed in ASME B&PV Code, Section II, Part D, Table 4, “Section III, Classes 1, TC, and SC; and Section VIII, Division 2, Design Stress Intensity Values S_m For Bolting Materials.” Generic guidance on closure bolts for transportation canisters is available in NUREG/CR-6007, “Stress Analysis of Closure Bolts for Shipping Casks”; however, ASME B&PV Code Section III, NB-3230 is preferred for bolt materials included in ASME B&PV Code Section II, Part D, Table 4.

Coordinate with the structural reviewer (SRP Chapter 4), who has the responsibility to verify that closure bolt stresses are within allowable limits.

8.5.10 Seals

Applicants for SNF storage canisters with metallic seals generally rely on data from the seal manufacturer to determine the maximum service temperatures for seals. Seals that may potentially be exposed to high temperature may not have been tested by independent laboratories (such as the National Institute of Standards and Technology and Factory Mutual). Because of the importance of seal integrity, ensure that the SAR includes laboratory test results using qualified procedures or data sheets that reference such test results.

8.5.10.1 *Metallic Seals*

Bolted lid canisters employ redundant metallic seals as part of the confinement boundary. These seals are SSCs important to safety. The primary materials issue is the temperature resistance of the seal spring material. Generally, this is a nickel-base alloy with excellent temperature and creep resistance. Verify that the metallic seal spring is constructed of a material that will not creep to an extent that may degrade its sealing performance. The seal cover material may be soft aluminum or silver. Aluminum-faced seals have failed in service because of corrosion from inadvertent rainwater intrusion (see NRC Information Notice 2013-07, "Premature Degradation of Spent Fuel Storage Cask Structures and Components from Environmental Moisture," dated April 16, 2013). Substitution of silver alloy-faced seals appears to have alleviated the susceptibility of mechanical seals to this corrosion-induced failure mechanism. If the applicant uses aluminum-faced seals, verify that the design includes provisions to prevent corrosion, such as the use of weather covers.

8.5.10.2 *Elastomeric Seals*

Bolted lid canister designs may also employ a weather cover to preclude rainwater from the confinement boundary seals. These weather covers may be sealed against the weather with an elastomeric seal such as Viton. As such, these seals may be susceptible to thermal- and radiation-induced aging (hardening). Consequently, a replacement program may be warranted if the heat or radiation exposure is sufficient. The seal manufacturer can generally provide guidance as to radiation or thermal resistance. Elastomeric seals have never been SSCs important to safety in storage canisters.

Radiation generally causes polymerization of elastomers to an extent that would adversely affect the performance when the dose reaches 10^5 grays (10^7 rads). For higher-dose rate environments, elastomer O-rings should not be specified. The use of fluorocarbons, which are known to be particularly susceptible to radiation damage, should be restricted if the dose is expected to exceed 100 grays (10^4 rads).

The reviewer should verify that O-ring seals do not reach their maximum operating temperature limit during normal and off-normal conditions of storage. Ensure that the SAR includes the O-ring manufacturer's data sheets specifying temperature and radiation tolerances. The applicant's evaluation should demonstrate that the minimum normal operating temperature (usually -40 °C (-40 °F)) will neither fail the O-ring seal by brittle fracture nor stiffen the O-ring (lose elasticity) to an extent that prevents the seal from meeting its service requirements.

Verify that, under the environmental conditions expected in storage service, O-ring seals will not chemically react or decompose in a manner that would significantly affect other components of the DSS.

8.5.11 Corrosion Resistance

The corrosion rates of engineering alloys are dependent on a number of factors including humidity, time of wetness, atmospheric contaminants, and oxidizing species (Fontana 1986). Consider the range of environmental conditions that are encountered for the DSS and DSF SSCs. For example, storage containers may be exposed to a variety of environments associated with fuel loading, canister closure, fuel drying, container transfer, and storage.

The following sections address specific considerations for commonly used engineering alloys for SSCs important to safety that may be exposed to environments where the effects of corrosion should be considered. In addition to material selection, other corrosion-control measures may be employed, provided adequate documentation is supplied to demonstrate efficacy. For example, coatings may be specified to alleviate the coastal atmospheric corrosion issue. However, unless supporting data are available to demonstrate the predicted coating life, the coating should be periodically inspected and maintained. Verify that any coating that is relied upon for corrosion resistance is screened in as important to safety. See Section 8.5.12 below for additional guidance on coatings.

8.5.11.1 *Environments*

Materials within the SNF container interior will be in an environment that contains very little water and is backfilled with helium to provide heat transfer and maintain a nonoxidizing environment for the fuel cladding and canister internals. Contaminants, such as chloride and sulfur species, can significantly accelerate general corrosion rates of engineering alloys. However, these species are strictly controlled in operating reactor coolant (EPRI 2000, 2007) and thus are not expected to be present in any residual moisture remaining inside the storage container after drying. Evacuating the canister or cask under vacuum and backfilling with an inert gas such as helium will significantly reduce the water content and humidity inside the canister and also reduce the oxidizing potential of the environment, both of which will significantly decrease the uniform corrosion rate of carbon steel and the potential for localized corrosion of passive alloys such as aluminum alloys and stainless steels.

While operational experience has shown only a few cases of atmospheric degradation of the external surfaces of DSS or DSF SSCs, it should be recognized that inspections have been limited. Generally, the DSS or DSF SSCs are subjected (long term) to a mild atmospheric environment. The range of environmental conditions may be limited and well defined for a site-specific application. For a CoC application, assume that the DSS SSCs may be exposed to a range of atmospheric conditions, including exposures to chloride-containing environments such as marine atmospheres, roadway deicing salt, and cooling-tower effluents. The presence and accumulation of chloride-containing salts can accelerate atmospheric corrosion rates. In addition, consider the effects of temperature fluctuations at the range of possible ISFSI sites when evaluating DSS designs and material selection. Corrosion rates for engineering alloys, including carbon and low-alloy steels, stainless steels, and aluminum alloys in a range of natural and industrial environments, may be found in corrosion references such as “Corrosion Engineering” (Fontana 1986), “Corrosion Data Survey by the National Association of Corrosion Engineers,” (Graver 1985), “Corrosion and Corrosion Control,” (Revie and Uhlig 2008), “Uhlig’s Corrosion Handbook” (Revie 2000), and the ASM Handbook Volume 13, “Corrosion.” Additional information on alloys and materials in specific environments is available in specialized publications such as the ASTM Special Technical Publications series. The National Aeronautics and Space Administration’s Kennedy Space Center Corrosion Technology Laboratory has also issued numerous reports on corrosion of alloys exposed to marine environments as well as testing of coatings to prevent corrosion.

8.5.11.2 *Carbon and Low-Alloy Steels*

For carbon and low-alloy steels that are not in an inert environment or embedded environment such as concrete, verify that the corrosion allowance specified is adequate for the applied term of the license or certificate. Corrosion rates for carbon steel in air may be found in the corrosion references discussed above in Section 8.5.11.1.

In environments such as locations that are in marine atmospheres, near roadways where deicing salts are used, or exposed to cooling-tower effluents, the presence and accumulation of heavy chloride-containing salts can significantly accelerate the normally slight atmospheric corrosion rates to unacceptable values for some storage canister or cask module designs, such as those that employ carbon steel structural elements inside a storage overpack.

To address the increased atmospheric corrosion rates found at coastal marine (salt water) sites, some applicants have specified the use of “weathering steels,” such as Cor-Ten, that form a protective layer of corrosion products that reduce additional loss of material. Weathering steels usually contain a minimum of 0.20 percent copper, but they also typically contain small additions of nickel, chromium, and phosphorous (Murata 2000). The Kennedy Space Flight Center has collected data that have demonstrated the benefit of copper-bearing and weathering steels for significantly reducing corrosion at coastal marine sites. Therefore, for coastal marine sites, the use of copper-bearing steels (containing a minimum of 0.20 percent copper) or weathering steels may be necessary. Such steels are covered by ASTM A242, “Standard Specification for High-Strength Low-Alloy Structural Steel,” and ASTM A588, “Standard Specification for High-Strength Low-Alloy Structural Steel, up to 50 ksi [345 MPa] Minimum Yield Point, with Atmospheric Corrosion Resistance,” supplemental requirements to ASTM A36, “Standard Specification for Carbon Structural Steel,” and other specifications.

8.5.11.3 *Austenitic Stainless Steels*

When stainless steel is used for storage containers, the primary concern generally is not corrosion but rather various types of localized corrosion such as pitting or crevice corrosion and stress-corrosion cracking. These corrosion mechanisms are possible in environments that contain chlorides. In the case of dry storage canisters, these corrosion mechanisms may be initiated in environments where airborne chlorides can be transported to the canister surfaces and deliquescence of the deposited chloride-containing salts results in an aqueous film containing chloride ions. Localized corrosion and chloride-induced stress-corrosion cracking (CISCC) of stainless steel components exposed to marine environments have been observed at operating reactors, as documented in the NRC Information Notice 2012-20, “Chloride-Induced Stress Corrosion Cracking of Austenitic Stainless Steel and Maintenance of Dry Cask Storage System Canisters,” dated November 14, 2012. However, no occurrence of localized corrosion or CISCC has been observed in the limited inspections of DSSs conducted to date.

NUREG/CR-7170, “Assessment of Stress Corrosion Cracking Susceptibility for Austenitic Stainless Steels Exposed to Atmospheric Chloride and Non-Chloride Salts,” describes laboratory tests for CISCC of austenitic stainless steels and the evaluation of the temperature and relative humidity conditions needed for deliquescence of chloride-containing salts. These tests show that all austenitic stainless steels used for DSS confinement boundaries are susceptible to CISCC, with lower alloy grades, such as 304, more susceptible than the low-carbon, molybdenum-containing 316L. Tests also showed that sensitized material was more susceptible to CISCC than nonsensitized material. The Electrical Power Research Institute (EPRI) conducted a review of the conditions under which CISCC has been observed, evaluated the effects of CISCC, and developed susceptibility assessment criteria for ISFSI locations and welded austenitic stainless steel canisters (EPRI 2013, 2014a, 2014b, 2015).

Based on testing and reviews of operational experience, degradation of austenitic stainless steels as a result of CISCC is expected to be limited to welded structures with tensile residual stresses in environments with elevated airborne chloride concentrations. In addition, CISCC can only occur when the combination of atmospheric conditions and the temperatures of SSCs allow the formation of a chloride ion containing aqueous phase. In most environments, the development of these conditions would take years or decades to develop on the surfaces of DSS or DSF SSCs. Further, the rates of CISCC propagation are limited by a number of factors, including atmospheric conditions and residual stresses. Consequently, CISCC is not a degradation mode that is expected to affect SSCs important to safety constructed from welded austenitic stainless steels in the initial storage period. According to NUREG-1927, "Standard Review Plan for Renewal of Specific Licenses and Certificates of Compliance for Dry Storage of Spent Nuclear Fuel," aging management programs may be needed to address CISCC in periods of extended operation.

8.5.11.4 *Duplex Stainless Steels*

In aggressive environments, where CISCC is more likely to occur, an applicant may specify more corrosion-resistant materials. For the confinement boundary, verify that the materials specified are approved for ASME B&PV Code, Section III, Class 1 construction. Duplex stainless steel UNS S31803 has been approved in ASME B&PV Code Case N-635-1 (for construction of Class 1 components, and the NRC has accepted this code case in RG 1.84. Stainless steel S31803 is a 22-percent chromium, 5-percent nickel stainless steel that has both ferritic and austenitic phases. Duplex S31803 has greater corrosion resistance to pitting, crevice corrosion, and CISCC and has been used in offshore oil production applications where harsh environmental conditions are expected. Note that ASME B&PV Code Case N-635-1 is specific to S31803. A similar duplex stainless steel, S32205, was introduced subsequent to S31803. Duplex S32205 has tighter compositional specification ranges for chromium, molybdenum, and nitrogen. Dual-certified material (i.e., material that meets the requirements of S32205 and S31803) has been produced.

Note that 22-percent chromium, 5-percent nickel duplex stainless steel such as S31803 and S32205 are susceptible to microstructural alteration during welding that can have a significant effect on corrosion resistance (Leonard 2003). Liou et al. (2002) showed that cooling rate and nitrogen content had a marked effect on the austenite to ferrite content. Chen et al. (2002) showed significant decreases in impact energy for S32205 exposed to temperatures in the range of 800 to 950 °C (1,472 to 1,742 °F) for periods of 10 minutes or less, corresponding to 5 percent σ (sigma) phase. Sieurin and Sandstrom (2007) compared time-temperature-transformation curves and critical cooling temperature curves for S32205 duplex stainless steels and concluded that, in order to avoid sigma precipitation and at the same time obtain a sufficient ferrite–austenite phase balance, the cooling rate should be approximately in the range 0.25–50 Kelvin (K)/second. In addition, Sieurin and Sandstrom (2007) stated that, in order to avoid more than 1 percent σ (sigma) phase, the cooling rate from the solution treatment temperature should exceed 0.23K/second, and the aging time must not exceed 134 seconds at the most critical temperature 865 °C (1,590 °F).

As a result of the operational experience with welded duplex stainless steels, the American Petroleum Institute (API) published Technical Report 938-C, "Use of Duplex Stainless Steels in the Oil Refining Industry," which provides guidance for the acceptance and welding of duplex stainless steels. The API guidance references ASTM A923, "Standard Test Methods for Detecting Detrimental Intermetallic Phase in Duplex Austenitic/Ferritic Stainless Steels," which includes specific screening tests, microstructural evaluation methods for detecting detrimental microstructural phases, and impact toughness requirements for base metals and welded duplex stainless steels. Verify that DSS designs that specify duplex stainless steels for the confinement boundary have (1) adequately addressed the unique microstructural considerations associated with these alloys and (2) included specific testing and acceptance criteria to ensure that fabrication and welding of the duplex stainless steel do not result in detrimental microstructural alterations that negatively impact the corrosion resistance or toughness of the alloy.

8.5.12 Protective Coatings

Coatings in DSSs are used primarily as corrosion barriers or to facilitate decontamination. They may have additional roles, such as improving the heat rejection capability by increasing the emissivity of internal components. Coatings typically are not SSCs important to safety. The SSCs to which the coatings are applied are generally important to safety. No coating should be credited for protecting the substrate material or extending the useful life of the substrate material unless a periodic coating inspection and maintenance program is required.

Coatings generally have low safety significance with the exception of coating issues that may result in adverse chemical or galvanic reactions. Typically, the information the applicant provides on coatings is not generally subject to further confirmation as part of the review. However, the applicant may specify unique or innovative coatings to perform a specific function unique to the storage system. In these instances, use discretion in implementing the review guidance in this section.

8.5.12.1 *Review Guidance*

Determine the appropriateness of the coating(s) for the intended application by reviewing the coating specification for each coating that is applied to an SSC important to safety. A specification that describes the scope of the work, required materials, the coating's purpose, and key coating procedures should ensure that appropriate and compatible coatings have been selected by the DSS designers.

8.5.12.2 *Scope of Coating Application*

Ensure that the SAR describes the function of the coating, a list of the components to be coated, and a description of the expected environmental conditions (e.g., expected conditions during loading, unloading, and dry storage).

8.5.12.3 *Coating Selection*

Verify that the coating specification identifies the manufacturer's name and the type of primers and topcoat(s) comprising the coating system. Because of the unique nature of coating properties and coating application techniques, the manufacturer's literature may be the only source of information on the particular coating.

Verify that the coating selected for the storage container components is capable of withstanding the intended service conditions over the design service life. Verify that the coatings will not react with the container internal components and contents and will remain adherent and inert when exposed to the various service environments. The most prevalent, potentially degrading environments include the immersion in borated SNF pool water during loading and unloading operations, and high-temperature and high-radiation environments encountered during vacuum drying and long-term storage. Failures can be prevented by ensuring that the selection and the application of the coating are controlled by adhering to the coating manufacturer's recommendations.

8.5.12.4 *Coating Qualification Testing*

Ensure that the coatings (including paints or plating) used inside a DSS have been tested to demonstrate the coatings performance under all conditions of loading and storage. The conditions evaluated should include exposure to radiation, high temperature during vacuum drying and storage, and immersion during loading, unloading, and transfer operations. The applicant should demonstrate that the coating will remain intact and inert for the full duration of the DSS design life.

There are a number of standardized ASTM tests for coatings performance. In reviewing ASTM (or other) tests used to qualify coatings for service in storage containers, consider the applicability of a test to the service conditions.

Ensure that a qualified coatings engineer (e.g., certified by the National Association of Corrosion Engineers) performed the planning, execution, and interpretation of coating qualification tests. Ensure that the applicant has employed appropriate, qualified expertise for any coatings qualification program. In addition, unless supporting data are available to demonstrate the predicted coating life, the coating should be periodically inspected and maintained.

8.5.13 Content Reactions

Verify that the contents of SNF, reactor-related GTCC waste, and HLW are stable and that there will be no adverse reactions with the container or internal baskets or supports over the storage period (10 CFR 72.120(d) and 10 CFR 72.236(h)).

Verify that the applicant has provided an adequate description of the contents so that the reviewer can fully evaluate its stability and compatibility with the container. Key parameters of the applicant's description include the physical and chemical form (e.g., activated metal, process waste), the geometric form (e.g., particulates, bulk solid), the maximum quantity of waste to be stored, and the radionuclide inventory.

8.5.13.1 *Flammable and Explosive Reactions*

Verify that the applicant has demonstrated that the contents will not lead to potentially flammable or explosive conditions.

Metallic contents may be subject to pyrophoricity, or auto-ignition, when the content surface area is sufficiently large (e.g., fine particulates) and oxygen or humidity, or both, are present at elevated temperatures. If metallic contents could potentially support pyrophoricity, the applicant should demonstrate that measures are taken to remove moisture or oxygen from the container, such as through vacuum or inerting. The applicant also should consider the potential for content materials, such as polymers, to decompose when exposed to heat and radiation, which may generate the moisture to support pyrophoricity.

In addition, hydrogen or other flammable gases may be generated during wet loading and unloading operations. For example, aluminum used in basket components can react with moisture to generate hydrogen. Efforts to passivate the aluminum components have proven inadequate to eliminate the generation of hydrogen. The use of zinc, zinc-rich coatings, or zinc-clad materials (e.g., galvanized steel) in particular is known to generate potentially large quantities of hydrogen gas during wet loading in SNF pools. In addition, flammable gas may be generated from waste radiolysis, biodegradation, and chemical reaction. Verify that the operating procedures contain measures for detecting the presence of hydrogen and preventing the ignition of combustible gases during cask loading and unloading operations. The technical specifications (SRP Chapter 17) should incorporate these procedures by reference.

Refer to NRC Bulletin 96-04, "Chemical, Galvanic, or Other Reactions in Spent Fuel Storage and Transportation Casks," issued July 5, 1996, for information about operational issues associated with hydrogen generation. This bulletin describes a case where a zinc coating on a canister interior reacted with borated SNF pool water to generate hydrogen, which ignited during the canister closure welding. Confirm that the applicant has demonstrated that no such adverse reactions will occur between the canister content materials, fuel payload, and the operating environments (10 CFR 72.120(d) and 10 CFR 72.236(h)).

8.5.13.2 *Corrosion*

Corrosive reactions between the contents and the internal environment, as well as reactions between the contents and the confinement container, may degrade structural integrity and confinement, and also may adversely impact retrievably of the SNF. Ensure that the SAR demonstrates that corrosion wastage will not lead to a loss of intended functions.

Refer to Section 8.5.15.2.3 of this SRP for guidance on the review of spent fuel cladding oxidation. For noncladding hardware components, the staff has previously reviewed a number of hardware components and materials for compliance with 10 CFR 72.120(d) to ensure that there are no significant chemical, galvanic, or other reactions. These stainless steel and zirconium alloy components are various neutron source assemblies, burnable poison rod assemblies, thimble plug devices, and other types of control elements. The staff has found the following materials to be acceptable for storage when the canister is constructed of stainless steel with stainless steel and aluminum basket components:

- Neutron source materials composed of stainless steel or zirconium alloy cladding containing antimony-beryllium, americium-beryllium, plutonium-beryllium, polonium-beryllium, and californium—The NRC assessed the exposure of these various contents to the wet loading and dry storage environment and determined that corrosion would not lead to a loss of intended functions.

- Control elements composed of zircaloy or stainless steel cladding containing boron carbide, borosilicate glass, silver-indium-cadmium alloy, or thorium oxide—The NRC assessed the exposure of these various contents to the wet loading and dry storage environment and determined that corrosion would not lead to a loss of intended functions.

8.5.14 Management of Aging Degradation

8.5.14.1 *Initial Storage Term*

In some cases, materials degradation may challenge the ability of a component to fulfill its intended function for the duration of the storage term. If an applicant cannot demonstrate adequate materials performance, then the SAR should describe maintenance programs (e.g., monitoring, inspections) to address issues associated with materials aging degradation. Some examples of such maintenance activities from previous reviews include the following:

- transfer cask maintenance programs that inspect for corrosion, wear, and loose or damaged fasteners

- coatings inspections, in cases where coatings are credited for preventing corrosion, enhancing heat transfer, or where coating debris could interfere with ventilation pathways

- concrete inspections to identify deterioration and basemat settling

- radiation surveys to monitor neutron shield effectiveness

Ensure that proposed maintenance activities provide for timely identification of materials degradation such that corrective actions can be implemented before a loss of component intended functions. Monitoring and inspection activities should take the following measures:

- use methods that are demonstrated to be capable of evaluating the degradation mechanism

- be performed at a frequency that is sufficient to identify degradation before a loss of component function

- include clear, actionable acceptance criteria

Consider appropriate codes and standards, such as ACI 349.3R for the evaluation of concrete and the ASTM standards on coatings assessment that are endorsed in RG 1.54, "Service Level I, II, and III Protective Coatings Applied to Nuclear Power Plants."

Coordinate with the operating procedures reviewers (SRP Chapter 11 and Chapter 12, "Conduct of Operations Evaluation").

8.5.14.2 *Amendment Applications Submitted During a Renewal Review or after a Renewal is Issued*

The NRC may renew a specific license or a CoC for a term not to exceed 40 years, in accordance with 10 CFR 72.42(a) or 10 CFR 72.240(a), respectively. Renewal applications must address aging mechanisms and aging effects that could affect SSCs relied upon for the safe storage of SNF.

NUREG-1927, Revision 1, provides detailed staff guidance for reviewing amendments that are submitted (1) concurrently with a renewal application or (2) after a renewal has been issued. Verify that the following information is included in either the amendment application or the renewal application:

- a scoping evaluation that identifies any new SSCs (and associated subcomponents) included in the amendment request and discusses whether the SSCs are included or excluded from the scope of renewal, following the guidance in Chapter 2 of NUREG-1927, Revision 1

- an aging management review that identifies any applicable aging mechanisms and effects for the new SSCs (and associated subcomponents) within the scope of renewal

- changes to the final SAR, which should include the following:

    - scoping results and identification of any new in-scope SSCs

- revised table of analysis model report results

- identification of previously approved time-limited aging analyses that address the new in-scope SSCs, or identification and a summary of any revised or new time-limited aging analyses that support the amendment

- identification of previously approved aging management programs that encompass the new in-scope SSCs, or a summary of any revised or new aging management programs that will apply to the new in-scope SSCs

For concurrent amendment and renewal applications, if there are different materials reviewers for the renewal review and the amendment review, coordinate across the reviews to ensure that renewal aspects are covered for the amendment.

8.5.15 Spent Fuel

The materials review ensures that the mechanical properties of the cladding materials are adequate to ensure that the SNF remains in the configuration analyzed in the SAR.

The review guidance in this section addresses dry storage of all SNF of burnups the NRC currently licenses for commercial power plant operations. SARs with burnup levels exceeding those licensed by the NRC Office of Nuclear Reactor Regulation (NRR), or for cladding materials not licensed by NRR, may require additional justifications by the applicant.

8.5.15.1 *Spent Fuel Classification*

Verify that the SAR (and, where appropriate, the license or CoC) identifies the allowable SNF contents and condition of the assembly or rods consistent with the definitions in this SRP for intact, undamaged, and damaged fuel (see the Glossary of this SRP).

The reviewer should analyze damaged fuel in terms of the characteristics needed to perform functions to assure compliance with fuel-specific and system-related regulations. A fuel-specific regulation defines a characteristic or performance requirement of the SNF assembly. Examples of such regulations include 10 CFR 72.122(h)(1) and 10 CFR 72.122(l). A system-related regulation defines a performance requirement placed on the fuel so that the DSS can meet its regulatory requirements. Examples of such regulations include 10 CFR 72.122(h)(5) and 10 CFR 72.124(a).

Verify that the applicant considered whether the material properties, and possibly the configuration, of the spent fuel assemblies (SFAs) can be altered during extended irradiation or dry storage. If this alteration is significant enough to prevent the fuel or assembly from performing its intended functions during dry storage, then the SFA should be classified as damaged.

Ensure that the SAR discusses the following to support that the SNF (rods, assembly) to be loaded are either intact or undamaged:

1. the acceptable physical characteristics of the SNF (i.e., acceptable assembly defects and cladding breaches)

2. the intended functions the applicant has imposed on the SNF for demonstrating compliance with fuel-specific and system-related regulatory requirements

3. the alteration and degradation mechanisms during dry storage that could credibly compromise the ability of the fuel to meet its fuel-specific or system-related functions

4. discussions or analyses demonstrating that the mechanisms in 3 (above) will not reasonably affect the physical characteristics of the SNF (as defined in 1 above) or result in reconfiguration beyond the safety analyses in the SAR

Recognize that SFAs with any of the following characteristics, as identified during the fuel selection process, are expected to be classified as damaged unless an adequate justification is provided for otherwise:

- There is visible deformation of the rods in the SFA. This is not referring to the uniform bowing that occurs in the reactor; instead, this refers to bowing that significantly opens up the lattice spacing.

- Individual fuel rods are missing from the assembly. The assembly may be classified as intact or undamaged if the missing rod(s) do not adversely affect the structural performance of the assembly, radiological, and criticality safety (e.g., no significant changes to rod pitch). Alternatively, the assembly may be classified as intact or undamaged if a dummy rod that displaces a volume equal to, or greater than, the original fuel rod is placed in the empty rod location.

- The SFA has missing, displaced, or damaged structural components such that any one of the following conditions occur:

    - Radiological or criticality safety is adversely affected (e.g., significantly changed rod pitch).

    - The structural performance of the assembly may be compromised during normal, off-normal, and accident conditions of storage.

    - The assembly cannot be handled by normal means (i.e., crane and grapple), if the design bases relies on ready retrieval of individual fuel assemblies.

- Reactor operating records or fuel classification records indicate that the SFA contains fuel rods with gross breaches. (See NRC Information Notice 2018-01, “Noble Fission Gas Releases During Spent Fuel Cask Loading Operations”)

- The SFA is no longer in the form of an intact fuel bundle (e.g., consists of, or contains, debris such as loose fuel pellets or rod segments).

Recognize that defects such as dents in rods, bent or missing structural members, small cracks in structural members, and missing rods do not necessarily render an assembly as damaged, if the applicant can show that the intended functions of the assembly are maintained; that is, the performance of the assembly does not compromise the ability to meet fuel-specific and system-related regulations.

The staff considers a gross cladding breach as any cladding breach that could lead to the release of fuel particulate greater than the average size fuel fragment. A pellet is approximately 1.1 cm (0.4 in.) in diameter in 15 x 15 PWR assemblies. Pellets from a boiling-water reactor (BWR) are somewhat larger, and those from 17 x 17 PWR assemblies are somewhat smaller. The pellet's length is slightly longer than its diameter. During the first cycle of irradiation in-reactor, the pellet fragments into 25–35 smaller interlocked pieces, plus a small amount of finer powder, because of pellet-to-pellet abrasion. When the rod breaches, about 0.1 gram (3.5 ounces) of this fine powder may be carried out of the fuel rod at the breach site (see NUREG/CR-1773, "Fission Product Release from BWR Fuel Under LOCA Conditions," issued July 1981). Modeling the fragments as either spherical- or pie-shaped pieces indicates that a cladding-crack width of at least 2–3 mm (0.08–0.12 in.) would be required to release a fragment. Hence, gross breaches should be considered to be any cladding breach greater than 1 mm (0.04 in.).

8.5.15.2 *Uncanned Spent Fuel*

The review procedures in this section apply to undamaged or intact SNF that is not placed inside a separate fuel can in the DSS confinement; that is, the safety analyses rely on the integrity of the fuel cladding for maintaining the analyzed configuration.

8.5.15.2.1 Cladding Alloys

Identify the specific cladding alloys (e.g., Zircaloy-2, Zircaloy-4, ZIRLO™, M5®) and maximum burnup of the SNF to be stored. The staff considers the peak rod average burnup as an appropriate measure of maximum fuel burnup in the materials evaluation. Ensure that the SAR indicates that the fuel and cladding alloy contents are consistent with the technical bases in the structural evaluation.

Determine whether the SNF to be stored includes boron-based integral fuel burnable absorbers. Consider the fact that these rods have the potential to increase the fuel rod internal pressure from decay gas generation (helium) when evaluating aging mechanisms during dry storage, particularly for periods beyond 20 years (see Section 8.5.15.1). Decay gases are not generated in rods with gadolinium-based integral fuel burnable absorbers, which will not result in increased rod pressures beyond those generated by the fuel fission products.

8.5.15.2.2 Cladding Mechanical Properties

Ensure that the structural evaluation is bounding to all cladding alloys in the allowable contents (i.e., Zircaloy-2, Zircaloy-4, ZIRLO™, M5®). Ensure that the SAR provides a justification that the cladding mechanical properties are bounding upon consideration of alloy type and fabrication process (cold work stress relieved annealed, recrystallized annealed), hydrogen content, neutron fluence (burnup), oxide thickness, and cladding temperature.

The reviewer should recognize that the applicant may use mechanical properties of as-irradiated/in-reactor or pre-hydrided/irradiated cladding (i.e., not accounting for the potential reorientation of hydrides during DSS loading and storage operations) in the structural evaluation of the SNF assembly.

Alternatively, the applicant may use mechanical properties of cladding accounting for reoriented hydrides in the structural evaluation of the SNF assembly. However, to date, the database for these properties is very limited. Preferred sources of cladding materials data include manufacturer's test data obtained under an approved quality assurance program, NRC-approved topical reports, staff-accepted technical reports, as well as peer-reviewed articles, research reports, and texts. Ensure that the SAR includes adequate justification of the applicability and acceptability of any source of information.

The NRC deems the mechanical property models from PNL-17700 (Geelhood et al. 2008) acceptable for previous licensing and certification actions. However, the determination of acceptability should consider the limitations of these models based on the data used for model validation (refer to Chapter 5 of PNL-17700 for additional details). Note that the models in PNL-17700 were validated with experimental measurements on Zircaloy-4, Zircaloy-2, and ZIRLO™ cladding. Therefore, confirm that the applicant used other references for defining bounding mechanical properties for M5® cladding. Limited nonproprietary data are available for M5® cladding, that is, publicly available data from the French Competent Authority (Institut de Radioprotection et de Sûreté Nucléaire). The SAR should justify that the limited temperature-dependent M5® cladding property data are reasonably bounding upon consideration of hydrogen content, neutron fluence (burnup), oxide thickness, and cladding temperature. Coordinate with the structural reviewer (SRP Chapter 4) to ensure that there is sufficient safety margin in the respective vibration and drop analyses to ensure that the assumed properties are adequate. The reviewer may also rely on engineering judgment, which should be informed by the staff's findings on previous NRC-approved topical reports.

Ensure that the SAR justifies that the assumed hydrogen content and neutron fluence is adequately bounding to the maximum burnup of the cladding contents (refer to Chapter 5 of PNNL-17700 for additional details). In addition, ensure that the SAR justifies the assumed temperature for the cladding mechanical properties. For example, the applicant may choose to use cladding mechanical properties corresponding to the maximum fuel assembly temperature at the location of the peak stress identified in the dynamic analyses.

The models in PNL-17700 only account for mechanical properties of cladding with circumferential hydrides. The staff recognizes that the public database of mechanical properties of materials with both circumferential and radial hydrides is very limited (e.g., Kim et al. 2015a, 2015b). However, based on static bend testing of cladding with a high density of radial hydrides (see NUREG/CR-7198, "Mechanical Fatigue Testing of High-Burnup Fuel for Transportation Applications," issued in 2017), the staff considers that these mechanical properties are adequate for the design-bases drop accidents during short-term loading operations (postulated accidents under 10 CFR 71.122(b)) or nonmechanistic DSS cask tipover accidents.

8.5.15.2.3 Effective Cladding Thickness

<u>Cladding Oxidation</u>

The structural evaluation should account for the reduced effective thickness of the cladding due to waterside corrosion (i.e., oxidation) during reactor service. The cladding oxide should not be considered load-bearing in the structural evaluation. The extent of oxidation and cladding wall thinning depends on the composition of the cladding (type of alloy) and burnup of the fuel. Note that the oxide will differ for the various cladding alloys and will not be of a uniform thickness along the axial length of the fuel rods. Ensure that the SAR defines an effective cladding thickness that is reduced by a bounding oxide layer to the specific cladding contents to be stored. Verify that the applicant has used a value of cladding oxide thickness that is justified by experimental oxide thickness measurements, computer codes validated using experimentally measured oxide thickness data, or other means that the staff finds appropriate. The NRC has determined that waterside corrosion models in the computer code FRAPCON 3.5 are acceptable for calculating oxide thickness values for Zircaloy-2, Zircaloy-4, ZIRLO™ and M5® cladding (see NUREG/CR-7022, "FRAPCON-3.5: A Computer Code for the Calculation of Steady-State, Thermal-Mechanical Behavior of Oxide Fuel Rods for High Burnup," issued October 2014).

Hydride Rim

During irradiation, some of the hydrogen generated due to water-side corrosion of the cladding will diffuse into the cladding. This results in the precipitation of hydrides in the circumferential-axial direction of the cladding when the amount of hydrogen generated exceeds the solubility limit in the cladding. The circumferential orientation of the hydrides is from the texture of manufactured cladding. The number density of these circumferential hydrides varies across the cladding wall because of the temperature drop from the fuel side (hotter) to the coolant side (cooler) of the cladding during reactor operation. Further, migration and precipitation of dissolved hydrogen to the coolant side of the cladding results in a rather dense hydride rim just below the corrosion (oxide) layer. The hydride number density and thickness of the rim depend on reactor operating conditions. For example, fuel rods operated at high linear heat rating to high burnup generally have a very dense hydride rim that is less than 10 percent of the cladding wall thickness. Conversely, fuel rods operated at low linear heat ratings to high burnup have a more diffuse hydride distribution that could extend as far as 50 percent across the cladding wall.

The applicant may have conservatively considered the cladding's outer hydride rim as wastage when determining the effective cladding thickness for the structural evaluation. However, there is no reliable predictive tool available to calculate this rim thickness, which varies along the fuel-rod length, around the circumference at any given axial location, from fuel rod to fuel rod within an assembly, and from assembly to assembly. Further, recent ring compression test results from the Argonne National Laboratory indicate that for the range of gas pressures anticipated during drying and storage, the hydride rim remains intact following slow cooling under conditions of decreasing pressure (Billone et al. 2013, 2014, 2015). These results indicate that the hydride rim is load bearing and can be accounted for in the effective cladding thickness calculation if mechanical test data referenced in the structural evaluation has adequately accounted for its presence. Historically, this has been the case during the review of DSSs, as applicants have provided mechanical property data generated from tests with irradiated cladding samples with an intact hydride rim. This includes test data derived from axial tensile tests or pressurized tube tests of samples that do not have a machined gauge section. For example, the mechanical property models used in PNL-17700 have been validated with experimental data from axial tensile tests on full cladding tubes and ring tests with no machined gauge section taken on irradiated recrystallized annealed Zircaloy-2 and Zircaloy-4 and stress-relief annealed ZIRLO™ cladding. As such, the staff considers any prior consideration to treat the rim as wastage to be unnecessary when calculating the effective cladding of the thickness, as the hydride rim has been properly accounted in the mechanical property models.

Drying Adequacy

Evaluate the descriptions related to draining and drying of the DSS confinement cavity during fuel loading operations, as discussed in the operating procedures chapter of the SAR. More specifically, ensure that the SAR clearly describes the procedures for removing water vapor and oxidizing material to an acceptable level, and that those procedures are appropriate.

The NRC staff has accepted vacuum-drying methods comparable to those recommended in PNL-6365, "Evaluation of Cover Gas Impurities and Their Effects on the Dry Storage of LWR Spent Fuel" (Knoll and Gilbert 1987). This report evaluates the effects of oxidizing impurities on the dry storage of light-water reactor (LWR) fuel and recommends limiting the maximum quantity of oxidizing gasses (e.g., oxygen, carbon dioxide, and carbon monoxide) to a total of 1 gram-mole per cask. This corresponds to a concentration of 0.25 volume percent of the total gases for a 7.0-cubic-meter (about a 247-cubic-foot) cask gas volume at a pressure of about 0.15 MPa

(1.5 atm) at 300 °K (80.3 °F). This 1 gram-mole limit reduces the amount of oxidants to below levels where cladding degradation is expected. Moisture removal is inherent in the vacuum drying process, and levels at or below those evaluated in PNL-6365 (about 0.43 gram-mole water) are expected if adequate vacuum drying is performed.

If alternative methods other than vacuum drying are used (such as forced helium recirculation), ensure that the applicant provides additional analyses or tests to sufficiently justify that moisture and impurity levels of the fuel cover gas will prevent unacceptable cladding degradation.

The following examples illustrate the accepted methods for cask draining and drying in accordance with the recommendations of PNL-6365 (Knoll and Gilbert 1987):

- The DSS confinement cavity should be drained of as much water as practicable and evacuated to less than or equal to 4.0×10^{-4} MPa (4 millibar, 3.0 mm Hg or Torr). After evacuation, adequate moisture removal should be verified by maintaining a constant pressure over a period of about 30 minutes without vacuum pump operation (or the vacuum pump is running but it is isolated from the cask with its suction vented to atmosphere). The DSS confinement cavity is then backfilled with an inert gas (e.g., helium) for applicable pressure and leak testing. Care should be taken to preserve the purity of the cover gas and, after backfilling, cover gas purity should be verified by sampling.

- The procedures should reflect the potential for blockage of the evacuation system or masking of defects in the cladding of non-intact rods, as a result of icing during evacuation. Icing can occur from the cooling effects of water vaporization and system depressurization during evacuation. Icing is more likely to occur in the evacuation system lines than in the DSS confinement cavity because of decay heat from the fuel. A staged draw down or other means of preventing ice blockage of the cask evacuation path may be used (e.g., measurement of cask pressure not involving the line through which the cask is evacuated).

- The procedures should specify a suitable inert cover gas (such as helium) with a quality specification that ensures a known maximum percentage of impurities to minimize the source of potentially oxidizing impurity gases and vapors and adequately remove contaminants from the cask.

- The process should provide for repetition of the evacuation and repressurization cycles if the DSS confinement cavity is opened to an oxidizing atmosphere following the evacuation and repressurization cycles (as may occur in conjunction with remedial welding, seal repairs). Refer to Appendix 8B, “Fuel Cladding Creep,” to this SRP chapter for additional considerations on cladding oxidation and splitting.

Ensure that the drying specifications are consistent with the proposed operating controls and limits described in the technical specifications chapter of the SAR. In addition, assess the need for any additional technical specifications.

8.5.15.2.4 Maximum (Peak) Cladding Temperature

The acceptance criteria below and review procedures are designed to provide reasonable assurance that the spent fuel is maintained in the configuration analyzed in the SAR. The criteria below are applicable to all commercial spent fuel burnup levels and cladding materials. In order to assure integrity of the cladding material, the following criteria should be met:

1. For all fuel burnups (low and high), the maximum calculated fuel cladding temperature should not exceed 400°C (752°F) for normal conditions of storage and short-term loading operations (e.g., drying, backfilling with inert gas, and transfer of the cask to the storage pad). However, for low burnup fuel, a higher short-term temperature limit may be used, if the applicant can show by calculation that the best estimate cladding hoop stress is equal to or less than 90 MPa (13,053 psi) for the temperature limit proposed

2. For off-normal and accident conditions, the maximum cladding temperature should not exceed 570°C (1058°F).

Coordinate with the thermal reviewer (SRP Chapter 5) to verify that the calculated maximum cladding temperature is based upon the peak rod temperature, not the average rod temperature. By employing the peak rod temperature, the safety analyses are conservatively bounding to all fuel rods in the content. Also confirm that the thermal models (and associated uncertainties) the applicant used for calculating cladding temperatures are acceptable to the thermal reviewer.

8.5.15.2.5 Thermal Cycling during Drying Operations

The reviewer should review fuel loading procedures to assure that any repeated thermal cycling (repeated heatup or cooldown cycles) during loading operations is limited to less than 10 cycles, where cladding temperature variations during each cycle do not exceed 65 °C (117 °F). The intent of the thermal cycling acceptance criteria is to limit precipitation of radial hydrides during loading operations. Evaluate the technical bases provided in support of any thermal cycling inconsistent with this criterion on a case-by-case basis. Further, reflooding of the previously dried high burnup fuel is not allowable unless the technical basis has adequately addressed the consequences of this operation on the performance of the cladding.

The applicant may use mechanical properties of cladding accounting for reoriented hydrides in the structural evaluation of the SFA. However, the database for these properties is very limited. For such applications, the loading procedures do not need to describe any thermal cycling limits if the applicant has adequately justified that the mechanical properties are reasonably bounding to reorientation expected for the design-bases heatup and cooldown cycles.

8.5.15.2.6 Cover Gas

Verify that the application defines the composition of the cover gas for the fuel during dry storage. Once the fuel rods are placed inside of the DSS confinement cavity and water is removed to a level that exposes any part of the rods to a gaseous atmosphere, the applicant must demonstrate that the SNF cladding will be protected against splitting from fuel pellet oxidation (10 CFR 72.122(h)(1)). If that atmosphere is oxidizing, then the fuel pellet may oxidize and expand, placing stress on the cladding. The expansion may eventually cause a gross rupture in the cladding, resulting in SNF that must be classified as damaged since it is not able to meet the requirement in 10 CFR 72.122(h)(1). The configuration of the fuel should remain bounded by the reviewed safety analyses. Further, the release of fuel fines, or grain-sized powder, from ruptured fuel into the confinement cavity may be a condition outside the design-bases for the DSS design. Three possible options exist to address the potential for and consequences of fuel oxidation:

1. Maintain the fuel rods in an inerted environment such as argon, nitrogen gas, or helium to prevent oxidation.

2. Ensure that there are not any cladding breaches (including hairline cracks and pinhole leaks) in the fuel pin sections that will be exposed to an oxidizing atmosphere

3. Determine the time-at-temperature profile of the rods while they are exposed to an oxidizing atmosphere and calculate the expected oxidation to determine if a gross breach would occur. The analysis should indicate that the time required to incubate the splitting process will not be exceeded. Such an analysis would have to address expected differences in characteristics between the fuel to be loaded and the fuel tested in the referenced data. The design-bases maximum allowable cladding temperature should be limited to the temperature at which calculations show that cladding splitting is not expected to occur. Such evaluations should address uncertainties in the referenced database.

If option 3 is chosen, coordinate with the thermal reviewer (SRP Chapter 5) to determine that the operating procedures (SRP Chapter 11, “Operation Procedures and Systems Evaluation”) and the technical specifications (SRP Chapter 17, “Technical Specification Evaluation”) of the license or CoC, as submitted by the applicant, provide an adequate analysis of the potential for cladding splitting should fuel rods be exposed to an oxidizing gaseous atmosphere.

Fuel oxidation and cladding splitting follow Arrhenius time-at-temperature behavior. For fuel burnups not exceeding 45 GWd/MTU and Zircaloy cladding, the time-at-temperature curves for uranium-based fuel developed to date (e.g., Einziger and Strain 1986) can be used to determine the allowable exposure duration on an oxidizing atmosphere for a given design-bases fuel cladding temperature. For example, using Figure 3-9 of Einziger and Strain (1986), at 360 °C (680 °F) one would expect to incur splitting at between 2 and 10 hours. On the other hand, if one expected the cladding temperature to stay at temperature for 100 hours, then the fuel temperature should be kept below 290 °C (554 °F). Refer to Appendix 8C, “Fuel Oxidation and Cladding Splitting,” to this SRP for additional information on cladding splitting.

8.5.15.2.7 High Burnup Fuel Monitoring and Assessment (dry storage periods beyond 20 years)

Under the regulations in 10 CFR 72.42, “Duration of license; renewal,” and 10 CFR 72.238, “Issuance of an NRC Certificate of Compliance,” an applicant may request an initial license or CoC storage period, respectively, that does not exceed 40 years. Experimental confirmatory data, as described in NUREG/CR-6745, “Dry Cask Storage Characterization Project—Phase 1; CASTOR V/21 Cask Opening and Examination,” and NUREG/CR-6831, “Examination of Spent PWR Fuel Rods after 15 Years in Dry Storage,” has shown that the integrity of low burnup fuel (less than or equal to 45 GWd/MTU) in dry storage is not expected to be impacted for periods up to 40 years.

For high burnup fuel (i.e., fuel with burnups generally exceeding 45 GWd/MTU), dry storage has been allowed for periods up to 20 years without the need to provide confirmatory data that the SNF configuration will remain as analyzed. However, for a license or storage term exceeding 20 years, verify that the applicant provided a maintenance plan to obtain such confirmatory data. Refer to NUREG-1927, Revision 1, when evaluating proposed maintenance activities for providing confirmatory data. These maintenance activities should be consistent with aging management activities (e.g., aging management program) during a renewed license or CoC storage period; that is, periods between 20 and 60 years. Refer to the discussions on Chapters 2 and 3, and Appendices D and B to NUREG-1927, Revision 1, for the review of acceptable maintenance activities.

8.5.15.2.8 Release Fractions

The materials reviewer should coordinate with the confinement reviewer to ensure that the SAR has provided adequate release fractions for the proposed fuel contents if the DSS confinement is non-leaktight. The technical basis may include an adequate description of the supporting experimental data, including a description of the burnups of the test specimens, number of tests, and test specimen pressure at the time of fracture. Further, the collection method used for quantification of the release fractions should be sophisticated enough to gather respirable release fractions.

The materials reviewer should recognize that high burnup fuel has different characteristics than low burnup fuel with respect to CRUD thickness, cladding oxide thickness, hydride content, radionuclide inventory and distribution, heat load, fuel pellet grain size, fuel pellet fragmentation, fuel pellet expansion and fission gas release to the rod plenum (see Appendix C.5 to NUREG/CR-7203, "A Quantitative Impact Assessment of Hypothetical Spent Fuel Reconfiguration in Spent Fuel Storage Casks and Transportation Packages," issued September 2015 (NRC 2015) for a description of high burnup fuel). Differences in these characteristics affect the mechanisms by which the fuel can breach and the amount of fuel that can be released from failed fuel rods. Hence, the SAR may provide different release fractions (CRUD, fission gases, volatiles, and fuel fines) for low and high burnup fuel in non-leaktight confinement.

8.5.15.3 *Canned Spent Fuel*

SNF that has been classified as damaged for storage must be confined in a can designed for damaged fuel or in an acceptable alternative (10 CFR 72.122(h)(1)). The purpose of a can designed for damaged fuel is to (1) confine gross fuel particles, debris, or damaged assemblies to a known volume within the cask; (2) demonstrate compliance with the criticality, shielding, thermal, and structural requirements; and (3) permit normal handling and retrieval from the storage container (if ready retrieval of the can is required per the design-bases). The can designed for damaged fuel may need to contain neutron-absorbing materials if results of the criticality safety analysis depend on the neutron absorber to meet the requirements in 10 CFR 72.124(a).

The configuration of the fuel inside the fuel can is generally not restricted; therefore, the applicant should perform bounding safety analyses assuming full reconfiguration of the fuel inside the fuel can. Ensure that the assumed mechanical properties of the fuel can are adequate for the calculated temperatures in the reconfiguration analyses. Ensure that the mechanical properties of the fuel can are also adequate for demonstrating adequate structural performance to ensure that the fuel remains confined to the can during normal, off-normal, and design-bases accident conditions.

8.6 Evaluation Findings

The NRC reviewer should prepare evaluation findings upon satisfaction of the regulatory requirements in Section 8.4 of this SRP. If the documentation submitted with the application fully supports positive findings for each of the regulatory requirements, the statements of finding should be similar to the following:

Specific License

F8.1 The applicant has met the requirements in 10 CFR 72.24(c)(3) and 10 CFR 72.120(a). The applicant described the materials used for SSCs important to safety in sufficient detail to support a safety finding.

F8.2 The applicant has met the requirements in 10 CFR 72.24(d) and 10 CFR 72.128(a). The properties of the materials in the storage facility design have been demonstrated to support the safe storage and handling of SNF, HLW, and reactor-related GTCC waste for the storage term under normal, off-normal, and accident conditions.

F8.3 The applicant has met the requirements in 10 CFR 72.124(b). Neutron absorbing materials are demonstrated to effectively control criticality without significant degradation over the storage life.

F8.4 The applicant has met the requirements in 10 CFR 72.120(d), 10 CFR 72.122(b)(1), and 10 CFR 72.124(b). Materials and storage contents are compatible with their operating environment such that there will be no adverse degradation or significant chemical or other reactions.

F8.5 The applicant has met the requirements in 10 CFR 72.122(c). Operating procedures contain measures for detecting the presence of hydrogen and preventing the ignition of combustible gases during cask loading and unloading operations.

F8.6 The applicant has met the requirements in 10 CFR 72.122(h)(1). The SNF cladding has been demonstrated to be adequately protected against gross ruptures, or the fuel has been demonstrated to be otherwise confined.

F8.7 The applicant has met the requirements in 10 CFR 72.122(h)(5) and 10 CFR 72.122(l). The packaging of HLW and reactor-related GTCC waste ensures that handling and retrievably is adequately maintained. The storage system is designed to allow ready retrieval of SNF, HLW, and reactor-related GTCC waste.

F8.8 The applicant has met the requirements in 10 CFR 72.24(c)(4) and 10 CFR 72.122(a). The use of codes and standards, quality assurance programs, and control of special processes are demonstrated to be adequate to ensure that the design, testing, fabrication, and maintenance of materials support SSC intended functions.

Certificate of Compliance

F8.9 The applicant has met the requirements in 10 CFR 72.236(b). The applicant described the materials design criteria for SSCs important to safety in sufficient detail to support a safety finding.

F8.10 The applicant has met the requirements in 10 CFR 72.124(b). Neutron-absorbing materials are demonstrated to effectively control criticality without significant degradation over the storage life.

F8.11 The applicant has met the requirements in 10 CFR 72.236(g). The properties of the materials in the storage system design have been demonstrated to support the safe storage of SNF.

F8.12 The applicant has met the requirements in 10 CFR 72.236(h). The materials of the SNF storage container are compatible with their operating environment such that there are no adverse degradation or significant chemical or other reactions.

F8.13 The applicant has met the requirements in 10 CFR 72.236(a) and 10 CFR 72.236(m). SNF specifications have been provided and adequate consideration has been given to compatibility with retrieval of stored fuel for ultimate disposal.

F8.14 The applicant has met the requirements in 10 CFR 72.234(b). Quality assurance programs and control of special processes are demonstrated to be adequate to ensure that the design, testing, fabrication, and maintenance of materials support SSC intended functions.

The reviewer should provide a summary statement similar to the following:

> The staff concludes that the [DSS or DSF designation] design adequately considers material properties, environmental degradation and other reactions, fuel clad integrity, content retrievability, and material quality controls such that the design is in compliance with 10 CFR Part 72. The evaluation of these materials considerations provides reasonable assurance the [DSS or DSF designation] will allow safe storage of [SNF/HLW/GTCC waste content designation] for a licensed (certified) life of [X] years. This finding is reached on the basis of a review that considered the regulation, itself, appropriate regulatory guides, applicable codes and standards, and accepted engineering practices.

8.7 References

Alexander, D.J. "Effects of Irradiation on the Mechanical Properties of 6061-T651 Aluminum Base Metal and Weldments." ASTM Special Technical Publication. Vol. 1325. pp. 1,027–1,044, 1999.

ACI 318, "Building Code Requirements for Structural Plain Concrete and Commentary," Farmington Hills, MI: American Concrete Institute.

ACI 349, "Code Requirements for Nuclear Safety-Related Concrete Structures and Commentary," Farmington Hills, MI: American Concrete Institute.

ACI 359, "Code for Concrete Reactor Vessels and Containments," Farmington Hills, MI: American Concrete Institute.

AISC, Manual of Steel Construction, 9th Edition, American Institute of Steel Construction, 1989.

ANSI, Standard N14.33-2005, “Characterizing Damaged Spent Nuclear Fuel for the Purpose of Storage and Transport,” American National Standards Institute, 2005.

ANSI, Standard N14.5, “American National Standard for Leakage Tests on Packages of Shipment of Radioactive Materials,” American National Standards Institute, 2014.

ANSI/American Society of Mechanical Engineers (ASME) Standard NQA-1, “Quality Assurance Program for Nuclear Facility Applications,” American National Standards Institute.

ANSI/ASME Standard N45.2.9, “Requirements for Collection, Storage, and Maintenance of Quality Assurance Records for Nuclear Power Plants,” American National Standards Institute.

API, “Use of Duplex Stainless Steels in the Oil Refining Industry,” API Technical Report 938-C Second Edition, American Petroleum Institute, April 2011.

ASM International, “ASM Metals Handbook Desk Edition,” (p. 54) 2nd Edition, J.R. Davis Editor, Materials Park, OH: ASM International, 1998.

ASM International, “ASM Handbook - Volume 13 Corrosion,” Materials Park, OH: ASM International, 2000.

ASME Boiler and Pressure Code (ASME B&PV Code), 2007 – Addenda 2008.
  Section I, “Power Boilers.”
  Section II, “Materials.”
  Section III, “Rules for Construction of Nuclear Facility Components.”
    Division 1, “Metallic Components”; Subsection NB through NH, and Appendices
  Section V, “Nondestructive Examination.”
  Section VIII, “Rules for Construction of Pressure Vessels.”
  Section IX, “Welding, Brazing, and Fusing Qualifications.”
  Section XI, “Rules for Inservice Inspection of Nuclear Power Plant Components,”
    Division 1, IWB 3600.

ASME, Code Case N-635-1, “Use of 22Cr 5Ni 3Mo N (Alloy UNS 531803) Forgings, Plate, Bar, Welded and Seamless Pipe, and/or Tube, Fittings, and Fusion Welded Pipe with Addition of Filler Metal, Classes 1, 2, and 3,” Section III, Division 1, 2003.

ASTM, Standard A36, “Standard Specification for Carbon Structural Steel,” West Conshohocken, PA: ASTM International.

ASTM, Standard A242, “Standard Specification for High-Strength Low-Alloy Structural Steel,” West Conshohocken, PA: ASTM International.

ASTM, Standard A588, “Standard Specification for High-Strength Low-Alloy Structural Steel, up to 50 ksi [345 MPa] Minimum Yield Point, with Atmospheric Corrosion Resistance,” West Conshohocken, PA: ASTM International.

ASTM, Standard B557, “Standard Test Methods for Tension Testing Wrought and Cast Aluminum- and Magnesium-Alloy Products,” West Conshohocken, PA: ASTM International, 2015.

ASTM, Standard C33, “Standard Specification for Concrete Aggregates,” West Conshohocken, PA: ASTM International.

ASTM, Standard C1671-15 “Standard Practice for Qualification and Acceptance of Boron Based Metallic Neutron Absorbers for Nuclear Criticality Control for Dry Cask Storage Systems and Transportation Packaging,” West Conshohocken, PA: ASTM International, 2015.

ASTM, Standard E290, “Standard Test Methods for Bend Testing of Material for Ductility,” West Conshohocken, PA: ASTM International, 2014.

ASTM, Standard A923-14, “Standard Test Methods for Detecting Detrimental Intermetallic Phase in Duplex Austenitic/Ferritic Stainless Steels”, West Conshohocken, PA: ASTM International, 2014.

American Welding Society (AWS), Standard A2.4, “Standard Symbols for Welding, Brazing, and Nondestructive Examination,” American Welding Society.

AWS, Standard D1.1, “Structural Welding Code-Steel,” Miami, FL: American Welding Society.

AWS, Standard D1.6, “Structural Welding Code-Stainless Steel,” Miami, FL: American Welding Society.

Billone, M.C., T.A. Burtseva, Z. Han, and Y.Y. Liu. “Embrittlement and DBTT of High-Burnup PWR Fuel Cladding Alloys,” FCRD-UFD-2013-000401, ANL-13/16, Argonne National Laboratory, Lemont, Illinois, 2013.

Billone, M.C., T.A. Burtseva, Z. Han, and Y.Y. Liu, “Effects of Multiple Drying Cycles on High-Burnup PWR Cladding Alloys,” DOE Used Fuel Disposition Report FCRD-UFD-2014-000052, ANL Report ANL-12/11, September 26, 2014.

Billone, M.C., T.A. Burtseva, and Y.Y. Liu. “Characterization and Effects of Hydrides in High-Burnup PWR Cladding Alloys.” *Proceedings of the International High-Level Radioactive Waste Management Conference*, Charleston, South Carolina. Paper No. 12617. American Nuclear Society. April 12–16, 2015.

Bouniol, P. and A. Aspart. “Disappearance of Oxygen in Concrete Under Irradiation: The Role of Peroxides in Radiolysis.” *Cement and Concrete Research*. Vol. 28. pp. 1,669–1,681, 1998.

Caskey, G.R., R.S. Ondrejcin, P. Aldred, R.B. Davis, and S.A. Wilson. “Effects of Irradiation on Intergranular Stress Corrosion Cracking of Type 304 Stainless Steel.” Proceedings of 45th NACE Annual Conference, April 23–27, 1990, Las Vegas, Nevada, 1990.

Chen, T.H., K.L. Weng, J.R. Yang, “The effect of high-temperature exposure on the microstructural stability and toughness property in a 2205 duplex stainless steel,” *Materials Science and Engineering A*, Vol. 338, pp. 259-270, 2002.

Einziger, R.E., and R.V. Strain, “Oxidation of Spent Fuel at Between 250° and 360°C,” Electric Power Research Institute Report NP-4524, 1986.

Electrical Power Research Institute (EPRI), “BWR Water Chemistry Guidelines -2000 Revision,” Nonproprietary version: BWRVIP-79 TR-103515-R2 Final Report, Palo Alto, CA: EPRI, March 2000.

EPRI, “Pressurized Water Reactor Primary Water Chemistry Guidelines,” Volume 1, Revision 6. EPRI-1014986, Palo Alto, CA: Electric Power Research Institute, 2007.

EPRI. “Effect of Radiation on Concrete–A Literature Survey and Path Forward.” Final Report 1025584. Palo Alto, California: Electric Power Research Institute, 2012.

EPRI, “Failure Modes and Effects Analysis (FMEA) of Welded Stainless Steel Canisters for Dry Cask Storage Systems,” EPRI-3002000815, Palo Alto, CA: Electric Power Research Institute, December 2013.

EPRI, “Literature Review of Environmental Conditions and Chloride-Induced Degradation Relevant to Stainless Steel Canisters in Dry Cask Storage Systems,” EPRI-3002002528, Palo Alto, CA: Electric Power Research Institute, May 2014a.

EPRI, “Flaw Growth and Flaw Tolerance Assessment for Dry Cask Storage Canisters,” EPRI-3002002785, Palo Alto, CA: Electric Power Research Institute, October 2014b.

EPRI, “Susceptibility Assessment Criteria for Chloride-Induced Stress Corrosion Cracking (CISCC) of Welded Stainless Steel Canisters for Dry Cask Storage Systems,” EPRI-3002005371, Palo Alto, CA: Electric Power Research Institute, September 2015.

Farrell, K., “Assessment of Aluminum Structural Materials for Service within the ANS Reflector Vessel,” ORNL/TM-13049, August 1995.

Farrell, K. and R.T. King. “Radiation-Induced Strengthening and Embrittlement in Aluminum.” Metallurgical Transactions A. Physical Metallurgy and Materials Science. Vol. 4, Issue 5. pp. 1,223–1,231, 1973.

Flom, Y., B.H. Parker and H.P. Chu, “Fracture Toughness of SiC/Al Metal Matrix Composite,” NASA Technical Memorandum 100745, August 1989.

Flom, Y., and R. J. Arsenault, “Effect of Particle Size on Fracture Toughness of SiC/Al Composite Material,” Acta Metall. Vol. 37, No. 9, pp. 2413-2423, 1989.

Fontana, M.G., *Corrosion Engineering*, Third Edition, McGraw Hill, 1986.

Gamble, R. “BWRVIP-100-A: BWR Vessel and Internal Project, Updated Assessment of the Fracture Toughness of Irradiated Stainless Steel for BWR Core Shrouds.” EPRI-1013396. Palo Alto, California: Electric Power Research Institute, 2006.

Graver, D.L, ”Corrosion Data Survey – Metals Section, 6th Edition Houston, TX: National Association of Corrosion Engineers, 1985.

Geelhood, K.J., CE Beyer, WG Luscher. “PNNL Stress/Strain Correlation for Zircaloy,” Pacific Northwest National Laboratory, PNNL-17700, July 2008.

Hilsdorf, H.R., J. Kroop, and H.J. Koch. “The Effects of Nuclear Radiation on the Mechanical Properties of Concrete.” *Douglas McHenry International Symposium on Concrete and Concrete Structures*, American Concrete Institute Publication SP-55, 1978.

IAEA. “Assessment and Management of Ageing of Major Nuclear Power Plant Components Important to Safety: Concrete Containment Buildings.” IAEA–TECDOC–1025, Vienna, Austria: International Atomic Energy Agency, 1998.

Kaufman, J. G., R. L. Moore and P. E. Schilling, “Fracture Toughness of Structural Aluminum Alloys,” *Engineering And Fracture Mechanics* Vol. 2. Pp. 197-210. 1971.

Kim, J.-S., Y.-J. Kim, D.-H. Kook, Y.-S. Kim. “A Study on Hydride Reorientation of Zircaloy-4 Cladding Tube Under Stress,” *Journal of Nuclear Materials*, Vol. 456, pp. 246–252, 2015a.

Kim, J.-S., T.-H. Kim, D.-H. Kook, Y.-S. Kim. “Effects of Hydride Morphology on the Embrittlement of Zircaloy-4 Cladding,” *Journal of Nuclear Materials*, Vol. 456, pp. 235–245, 2015b.

Knoll, R.W. and E.R. Gilbert, “Evaluation of Cover Gas Impurities and Their Effects on the Dry Storage of LWR Spent Fuel,” PNL-6365, Pacific Northwest Laboratory, Richland, WA, 1987.

Kontani, O., Y. Ichikawa, A. Ishizawa, M. Takizawa, and O. Sato. “Irradiation Effects on Concrete Structures.” Proceedings of International Symposium on the Ageing Management & Maintenance of Nuclear Power Plants. pp. 173–182. 2010.

Leonard, A. “Review of external stress corrosion cracking of 22%Cr duplex stainless steel, Phase 1 – Operational data acquisition,” HSE RR 129, Her Majesty's Stationery Office, Norwich, UK. 2003.

Lewandowski, J.J., “Fracture and Fatigue of Particulate MMCs, in Comprehensive Composite Materials,” Volume 3: Metal Matrix Composites, T.W. Clyne, Editor, Oxford UK: Pergamon. pp. 151-187, 2000.

Liou, H-Y., R-I. Hsieh, W-T. Tsai, “Microstructure and stress corrosion cracking in simulated heat-affected zones of duplex stainless steels,” *Corrosion Science*, Vol 44, pp. 2841–2856, 2002.

Miserez, A.G.T., “Fracture and Toughening of High Volume Fraction Ceramic Particle Reinforced Metals,” PhD Thesis, École Polytechnique Fédérale de Lausanne, 2003.

Murata, T., “Weathering Steel,” in Uhlig’s Corrosion Handbook, 2nd Edition, R.W. Revie, Ed., New York, NY: John Wiley and Sons, Inc., pp. 569-580, 2000.

National Association of Corrosion Engineers, *Corrosion Data Survey*, 1985.

Nikolaev, Yu, A.V. Nikolaeva, and Ya.I. Shtrombakh. “Radiation Embrittlement of Low-Alloy Steels.” *International Journal of Pressure Vessels and Piping*. Vol. 79. pp. 619–636. 2002.

U.S. Nuclear Regulatory Commission (NRC), Information Notice 2013-07, “Premature Degradation of Spent Fuel Storage Cask Structures and Components from Environmental

Moisture,” 2013 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML12320A697).

NRC, Bulletin 96-04, “Chemical, Galvanic, or Other Reactions in Spent Fuel Storage and Transportation Casks,” July 5, 1996.

NRC, Confirmatory Action Letter 97-7-001, Certificate Holder: Sierra Nuclear Corporation, 1998 (ADAMS Accession No. ML060620420).

NRC, Information Notice 2012-20, “Potential Chloride-Induced Stress Corrosion Cracking of Austenitic Stainless Steel and Maintenance of Dry Cask Storage System Canisters,” 2012 (ADAMS Accession No. ML12319A440).

NRC, NUREG/CR-6424, “Report on Aging of Nuclear Power Plant Reinforced Concrete Structures,” ORNL/TM-13148, Oak Ridge National Laboratory, March 1996b.

NUREG-1927, “Standard Review Plan for Renewal of Specific Licenses and Certificates of Compliance for Dry Storage of Spent Nuclear Fuel," Revision 1.”

NUREG/CR-1773, “Fission Product Release from BWR Fuel Under LOCA Conditions,” Oak Ridge National Laboratory, July 1981.

NUREG/CR-3019, “Recommended Welding Criteria for Use in the Fabrication of Shipping Containers for Radioactive Materials,” UCRL-53044, Lawrence Livermore National Laboratory, March 1985.

NUREG/CR-3854, “Fabrication Criteria for Shipping Containers,” UCRL-53544, Lawrence Livermore National Laboratory, March 1985.

NUREG/CR-5502, “Engineering Drawings for 10 CFR Part 71 Package Approvals,” UCRL-ID-130438, Lawrence Livermore National Laboratory, May 1998.

NUREG/CR-6007, “Stress Analysis of Closure Bolts for Shipping Casks,” UCRL-ID-II0637, Lawrence Livermore National Laboratory, April 1992.

NUREG/CR-6407, “Classification of Transportation Packaging and Dry Spent Fuel Storage System Components According to Importance to Safety,” INEL-95/0551, Idaho National Engineering Laboratory, February 1996.

NUREG/CR-6745, “Dry Cask Storage Characterization Project—Phase 1; CASTOR V/21 Cask Opening and Examination,” INEEL/EXT-01-00183, Idaho National Engineering and Environmental Laboratory, September 2001.

NUREG/CR-6831, “Examination of Spent Fuel Rods After 15 Years in Dry Storage,” ANL-03/17, Argonne National Laboratory, September 2003.

NUREG/CR-7022, “FRAPCON-3.5: A Computer Code for the Calculation of Steady-State, Thermal-Mechanical Behavior of Oxide Fuel Rods for High Burnup,” PNNL-19418, Vol. 1, Rev. 1, Pacific Northwest National Laboratory, October 2014 (ADAMS Accession No. ML14295A539).

NUREG/CR-7171, “A Review of the Effects of Radiation on Microstructure and Properties of Concretes Used in Nuclear Power Plants,” ORNL/TM-2013/263, Oak Ridge National Laboratory, November 2013.

NUREG/CR-7170, “Assessment of Stress Corrosion Cracking Susceptibility for Austenitic Stainless Steels Exposed to Atmospheric Chloride and Non-Chloride Salts,” Center for Nuclear Waste Regulatory Analyses and U.S. Nuclear Regulatory Commission, February 2014.

NUREG/CR-7198, “Mechanical Fatigue Testing of High-Burnup Fuel for Transportation Applications,” ORNL/TM-0216/689, Oak Ridge National Laboratory, 2017.

NUREG/CR-7203, “A Quantitative Impact Assessment of Hypothetical Spent Fuel Reconfiguration in Spent Fuel Storage Casks and Transportation Packages,” ORNL/TM-2013/92, Oak Ridge National Laboratory, September 2015.

Odette, G.R. and G.E. Lucas. “Embrittlement of Nuclear Reactor Pressure Vessels.” *Journal of Metals*. Vol. 53, Issue 7, pp.18-22. 2001.

Regulatory Guide 1.54, “Service Level I, II, and III Protective Coatings Applied to Nuclear Power Plants.”

Regulatory Guide 1.84, “Design, Fabrication, and Materials Code Case Acceptability, ASME Section III.”

Regulatory Guide 1.193, “ASME Code Cases Not Approved for Use.”

Regulatory Guide 7.11, “Fracture Toughness Criteria of Base Material for Ferritic Steel Shipping Cask Containment Vessels with a Maximum Wall Thickness of 4 Inches.”

Regulatory Guide 7.12, “Fracture Toughness Criteria of Base Material for Ferritic Steel Shipping Cask Containment Vessels with a Wall Thickness Greater Than 4 Inches, But Not Exceeding 12 Inches.”

Revie, R.W, *Uhlig’s Corrosion Handbook*, Second Edition New York, NY: John Wiley and Sons, 2000.

Revie, R.W. and H.H. Uhlig, *Corrosion and Corrosion Control*, Fourth Edition, Hoboken, NJ: John Wiley & Sons, Inc., 2008

Rabiei, A., L. Vendra, and T. Kishi, “Fracture behavior of particle reinforced metal matrix composites,” *Composites*, Part A vol. 39 pp. 294–300, 2008.

Roberts, R. and C. Newton, “Interpretive Report on Small Scale Test Correlations with K_{IC} data,” *Welding Research Council Bulletin* 265, February 1981.

Sieurin and Sandstrom, “Sigma phase precipitation in duplex stainless steel 2205,” *Materials Science and Engineering A*, Vol. 444, pp. 271–276, 2007.

Uhlig, H.S., *Corrosion and Corrosion Control*, Wiley & Sons, Inc., 1985.

Washa, G.W., J.C. Saemann, and S.M. Cramer, “ Fifty-Year Properties of Concrete Made in 1937,” ACI Materials Journal, Vol. 86, Issue 4, pp. 367-371, 1989.

Wilson, D.W., J.W. Freeman, and H.R. Voorhees, *Creep-Rupture Testing of Aluminum Alloys to 100,000 Hours, First Progress Report*, prepared for the Metal Properties Council, New York, November 1969.

APPENDIX 8A CLARIFICATIONS, GUIDANCE, AND EXCEPTIONS TO ASTM STANDARD PRACTICE C1671-15

The American Standard for Testing and Materials (ASTM) standard practice C1671-15, "Standard Practice for Qualification and Acceptance of Boron Based Metallic Neutron Absorbers for Nuclear Criticality Control for Dry Cask Storage Systems and Transportation Packaging," with some exceptions, additions, and clarifications, is appropriate for staff use in their review activities. This appendix provides guidance to the staff that supplements guidance provided in Chapter 8 of this standard review plan (SRP). Alternative approaches are acceptable if technically supportable.

8A.1 Specific Clarifications, Exceptions, and Guidance

8A.1.1 Use of ASTM C1671-15

The staff considers the terminology and statements within ASTM Standard Practice C1671-15 as acceptable guidance with some additions, clarifications, and exceptions delineated below, for reviewing spent nuclear fuel (SNF) storage cask and transportation packages.

8A.1.2 Clarification Regarding Use of Section 5.2.1.3 of ASTM C1671-15

If the supplier has shown that process changes do not cause changes in the density, open porosity, composition, surface finish, or cladding (if applicable) of the neutron absorber material, the supplier should not need to requalify the material with regard to thermal properties or resistance to degradation by corrosion and elevated temperatures.

8A.1.3 Additional Guidance Regarding Use of Section 5.2.5.3 of ASTM C1671-15

The following additional guidance applies to Section 5.2.5.3: Neutron-absorbing materials should undergo testing to simulate submersion and subsequent cask drying conditions, as part of a qualifying test program. Clad aluminum/boron carbide neutron absorbers with open porosities between 1 and 3 percent have exhibited blistering after canister drying. This blistering was from flash steaming of water that was trapped in pores. The staff is concerned that such blistering could have an adverse impact on fuel retrievability and the ability of the absorber to perform its criticality safety function.

Unclad aluminum/boron carbide neutron absorbing materials with open porosities less than 0.5 volume percent may not be required to undergo simulated submersion and drying tests.

8A.1.4 Clarification Regarding Use of Section 5.2.6.2 of ASTM C1671-15

If a coupon contiguous to every plate of neutron-absorbing material is not examined during acceptance testing, the neutron attenuation program should be done with a sufficient number of samples to ensure that the neutron-absorbing properties of the materials meet the minimum required areal density of the neutron absorber, as defined in the technical specifications. In the past, the staff has accepted the following:

- for a neutron-absorbing material with a significant qualification program and non-statistically derived minimum guaranteed properties, wet chemistry analysis of mixed powder batches followed by additional neutron attenuation testing of a minimum of 10-percent of the neutron poison plates

- sampling plans where at least one neutron transmission measurement is taken for every 2,000 square inches of neutron poison plate material in each lot

- a sampling plan that requires that each of the first 50 sheets of neutron absorber material from a lot, or a coupon taken there from, be tested (by neutron attenuation). Thereafter, coupons shall be taken from 10 randomly selected sheets from each set of 50 sheets. This 1-in-5 sampling plan shall continue until there is a change in lot or batch of constituent materials of the sheet (i.e., boron carbide powder or aluminum powder) or a process change. A measured value less than the required minimum areal density of boron-10 during the reduced inspection is defined as nonconforming, along with other contiguous sheets, and mandates a return to 100-percent inspection for the next 50 sheets

8A.1.5 Additional Guidance Regarding Use of Section 5.2.6.2 and 5.3.4.1 of ASTM C1671-15

The following additional guidance applies to Section 5.2.6.2: The minimum areal density of boron-10 present in each type of neutron-absorbing material used in the calculation of the effective neutron multiplication factor, k_{eff}, should be clearly stated in the materials information of a 10 CFR Part 71, “Packaging and Transportation of Radioactive Material,” application, and the proposed technical specifications in a 10 CFR Part 72, ““Licensing Requirements for the Independent Storage of Spent Nuclear Fuel, High-Level Radioactive Waste, and Reactor-Related Greater Than Class C Waste” application.

It has been the staff’s practice to limit the credit for neutron-absorber materials to only 75 percent of the minimum amount of boron-10 confirmed by acceptance tests. The staff has accepted up to 90-percent credit in certain cases where the absorber materials are shown by neutron attenuation testing of production lots to be effectively homogeneous.

If 90-percent credit is taken for the efficacy of the neutron absorber, methods other than neutron attenuation should be used only as verification or partial substitution for attenuation tests. Benchmarking of other methods against neutron attenuation testing should be done periodically throughout acceptance testing, under appropriate attenuation conditions and with proper sample sizes. This should be done to confirm the adequacy of the proposed methods, as the staff considers direct measurement of neutron attenuation to be the most reliable method of measuring the expected neutron absorbing behavior of the poison plates.

Direct neutron attenuation measurements are only expected for the qualification of alternative characterization methods (e.g., wet chemistry analyses) when only 75-percent credit is taken for the boron-10 areal density of the neutron absorbing material. Once qualified and benchmarked, neutron attenuation is no longer expected for acceptance testing as the alternative method is considered properly validated by neutron attenuation.

Applicants should be encouraged to provide statistically significant data showing the correspondence between neutron attenuation testing and wet chemistry data and the precision of both methods. Such data may permit the partial substitution of neutron attenuation measurements with chemical methods for materials receiving 90-percent credit.

8A.1.6 Additional Guidance Regarding Use of Section 5.2.6.2(2) of ASTM C1671-15

The following additional guidance applies to Section 5.2.6.2(2): The size of the collimated neutron beam should be specified for attenuation testing, and limited to 2.54-cm diameter, with a tolerance of 10 percent. In the past, staff has had concerns that attenuation measurements conducted with neutron beams greater than 1-cm diameter may lack the resolution to detect localized regions of the neutron absorbing material which have a low concentration of boron-10. The staff conducted an independent criticality study using a SNF transportation package to determine if neutron attenuation measurements using beam sizes in excess of 1 cm were unable to detect localized regions in the neutron-absorbing material deficient in neutron absorber. In the study, it was assumed that the neutron absorber boron-10 arranged itself into a “checkerboard” fashion of alternating boron-rich and boron-deficient regions where the boron concentration was 50-percent greater than and 50-percent less than the average amount of boron in a homogenous plate of boron and aluminum. The staff considers this hypothetical configuration bounding of any possible “real-life” defects that might occur in actual manufacturing. In the simulations, two models were considered. One model permitted a non-constant density, where boron was removed from boron-deficient regions and directly added to adjacent regions. In the second model, the quantity of aluminum and carbon were adjusted in each of the regions so that the overall mass density of the plate remained uniform. The sizes of the boron-rich and boron-deficient regions were then gradually increased, and changes in k_{eff} were observed. This is plotted in Figure 8A-1.

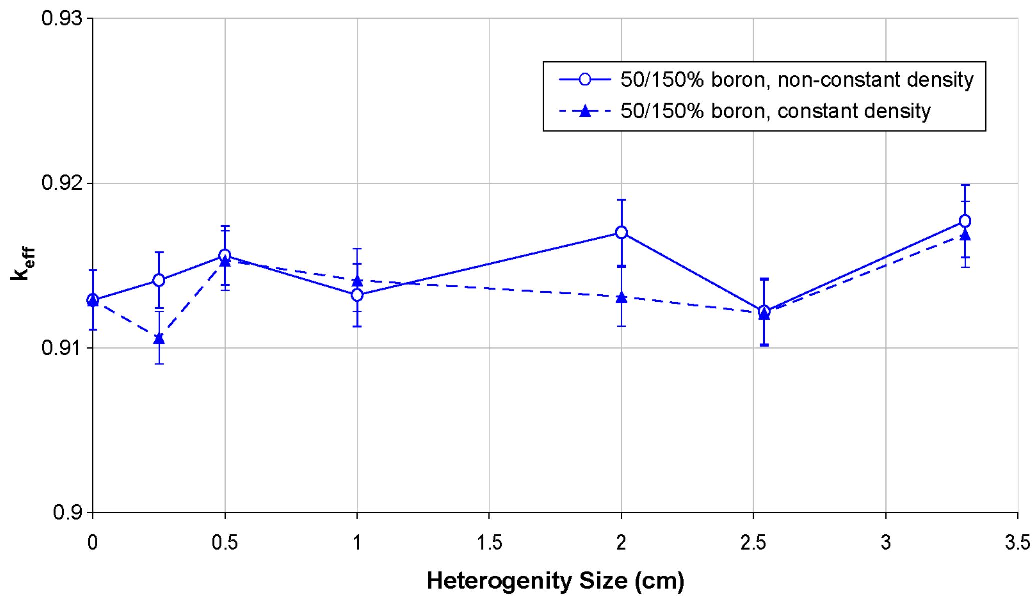

Figure 8A-1 Plot of the Effective Neutron Multiplication Factor, K_{eff}, as a Function of Heterogeneity Size

The results of the study showed no significant difference in k_{eff} when the size of the heterogeneities (the length of each boron deficit or rich region) increased from 1 cm to 2.54 cm. It should be noted that this study was conducted on a single transportation package design. The staff considers the heterogeneities introduced in the neutron-absorbing materials sufficiently exaggerated such that this study may be used to make a general determination.

As such, the staff regards collimated neutron beams with nominal diameters between 1 cm and 2.54 cm, with tolerances of 10 percent, as sufficiently capable of detecting defects within the neutron-absorbing material, and should be considered acceptable for the purposes of qualification and acceptance testing of neutron-absorbing materials.

8A.1.7 Additional Guidance Regarding Use of Section 5.2.6.3 of ASTM C1671-15

The following additional guidance applies to Section 5.2.6.3: The maximum permissible thickness deviation of the neutron-absorbing material should be specified, as should actions to be taken if the thickness is outside the permissible limits.

During the production of neutron-absorbing materials, minor deviations from the specified physical dimensions are expected. These deviations, and, in particular, variations of the neutron-absorbing material thickness should be discussed in the application, in a way that is referenced in the certificate of compliance (CoC). The applicant should specify the maximum permissible thickness deviation (for both over and under tolerances), and the actions taken if the thickness is outside the permissible limits. This is done to ensure adequate performance of the neutron absorbing materials. In the past, the staff has allowed acceptance testing where a minimum plate thickness is specified, which permitted local depressions, so long as the depressions were no more than 0.5 percent of the area on any given plate, and the thickness at their location was not less than 90 percent of the minimum design thickness.

8A.1.8 Additional Guidance Regarding Use of Section 5.2.6.4 of ASTM C1671-15

The following additional guidance applies to Section 5.2.6.4: A visual inspection procedure that describes the nominal inspection criteria should be specified in the applicant's acceptance tests. Visual inspection should be conducted on all neutron-absorbing materials intended for service.

As part of the visual inspection of the neutron-absorbing material, it is important to ensure that there are no defects that might lead to problems in service, such as delaminations or cracks that could appear on clad neutron-absorbing materials. The concern is that gross defects on the plate or plate edge may lead to separations, especially from vibrations during transportation; this could lead to a lack of absorber capability over the missing or misplaced region within a plate material.

8A.1.9 Clarification Regarding Use of Sections 5.2.7 and 5.3 of ASTM C1671-15

When implementing Sections 5.2.7 and 5.3, a description of the key processes, major operations process controls, and the acceptance testing steps of neutron-absorbing materials should be included in the acceptance tests of the safety analysis report and the proposed technical specifications for a 10 CFR Part 72 application.

8A.1.10 Additional Guidance Regarding Use of Section 5.2.7.1 of ASTM C1671-15

In addition to the guidance provided in Section 5.2.7.1, a change of the matrix alloy, or a change in the material's heat treatment that may cause an undesirable reaction to occur within the matrix itself or between the matrix and a secondary phase should also be considered key processes.

8A.1.11 Additional Guidance Regarding Use of Section 5.4 of ASTM C1671-15

The following additional guidance applies to Section 5.4: Neutron-absorbing materials intended for criticality control should have a safety classification of "A" under the guidance of NUREG/CR-6407, "Classification of Transportation Packaging and Dry Spent Fuel Storage System Components According to Importance to Safety."

8A.2 <u>References</u>

10 CFR Part 71, "Packaging and Transportation of Radioactive Material."

10 CFR Part 72, "Licensing Requirements for the Independent Storage of Spent Nuclear Fuel, High-Level Radioactive Waste, and Reactor-Related Greater Than Class C Waste."

American Standard for Testing and Materials, Standard C1671-15 "Standard Practice for Qualification and Acceptance of Boron Based Metallic Neutron Absorbers for Nuclear Criticality Control for Dry Cask Storage Systems and Transportation Packaging," 2015.

NUREG/CR-6407, "Classification of Transportation Packaging and Dry Spent Fuel Storage System Components According to Importance to Safety," INEL-95/0551, Idaho National Engineering Laboratory, February 1996.

APPENDIX 8B FUEL CLADDING CREEP

Creep is the dominant mechanism for cladding deformation under normal conditions of storage. The relatively high temperatures, differential pressures, and corresponding hoop stress on the cladding will result in permanent creep deformation of the cladding over time. Several laboratory programs have demonstrated that spent nuclear fuel (SNF) has significant creep capacity even after 15 years of dry storage. NUREG/CR-6831, ““Examination of Spent Fuel Rods After 15 Years in Dry Storage,” issued September 2003, reported that irradiated Surry-2 pressurized-water reactor (PWR) fuel rods (35.7 gigawatt days per metric ton of uranium (GWd/MTU)) that were stored for 15 years at an initial temperature of 350 degrees Celsius (°C) (662 degrees Fahrenheit (°F)) (with temperatures reaching as high as 415 °C (779 °F) for up to 72 hours) experienced thermal creep, which was estimated to be less than 0.1 percent. Post-storage creep tests were conducted to assess the residual creep capacity of the Surry-2 fuel rods. One rod segment experienced a creep strain of 0.92 percent without rupture at 380 °C (716 °F) and 220 megapascals (MPa) in 1,820 hours (75.8 days). A different rod segment was tested at 400 °C (752 °F) and 190 MPa for 1,873 hours (78 days), followed by 693 hours (28.9 days) at 400 °C and 250 MPa, and experienced a creep strain of more than 5 percent without failure (Tsai et al. 2006). Profilometry measurements on that fuel rod indicated that the creep deformation was uniform around the circumference of the cladding with no signs of localized bulging, which can be a precursor for rupture. A report of the literature (Beyer 2001) also indicates that some SNF cladding can accommodate creep strains of 2.8—7.5 percent at temperatures between 390 and 420 °C and hoop stresses between 225 and 390 MPa. Other significant contributions to the understanding of the effects of creep on SNF cladding can be found in several references (Einziger et al. 1982; Rashid et al. 2000; Hendricks 2001; Rashid and Dunham 2001; Machiels 2002). In general, these data and analyses support the conclusions that (1) deformation caused by creep will proceed slowly over time and will decrease the rod pressure, (2) the decreasing cladding temperature also decreases the hoop stress, and this too will slow the creep rate so that during later stages of dry storage, further creep deformation will become exceedingly small, and (3) in the unlikely event that a breach of the cladding from creep occurs, it is believed that this will not result in gross rupture.

Based on these conclusions, the staff has reasonable assurance that creep under normal conditions of storage will not cause gross rupture of the cladding and that the geometric configuration of the SNF will be preserved, provided that the maximum cladding temperature does not exceed 400 °C (752 °F).

References

Beyer, C.E., Letter from C. E. Beyer, Pacific Northwest National Laboratory, to K. Gruss, NRC. Subject: Transmittal of “Update of CSFM Methodology for Determining Temperature Limits for Spent Fuel Dry Storage in Inert Gas.” November 27, 2001.

Einziger, R.E., S. D. Atkin, D. E. Stellbrecht, and V. Pasupathi, “High Temperature Postirradiation Materials Performance of Spent Pressurized Water Reactor Fuel Rods Under Dry Storage Conditions.” *Nuclear Technology*, Vol. 57, p. 65. 1982.

Hendricks, L., Letter from L. Hendricks, NEI, to M. W. Hodges, NRC. Subject: Transmittal of Responses to the NRC Request for Additional Information on storage of high burnup fuel, August 16, 2001.

Machiels, A., “Regulatory Applications Lessons Learned -- Industry Perspective.” NEI Dry Storage Information Forum. Naples, FL. May 15-16, 2002.

NUREG/CR-6831, “Examination of Spent Fuel Rods After 15 Years in Dry Storage,” ANL-03/17, Argonne National Laboratory, September 2003.

Rashid, Y.R., D. J. Sunderland, and R. O. Montgomery, “Creep as the Limiting Mechanism for Spent Fuel Dry Storage - Progress Report.” EPRI TR-1001207. 2000.

Rashid, Y.R., R. S. Dunham, “Creep Modeling and Analysis Methodology for Spent Fuel in Dry Storage.” EPRI TR-1003135. 2001.

Tsai, H. and M.C. Billone, “Thermal Creep of Irradiated Zircaloy Cladding,” *Journal of ASTM International*, Vol. 3, No. 1, pp. 1–16, 2006.

APPENDIX 8C FUEL OXIDATION AND CLADDING SPLITTING

Irradiated uranium dioxide (UO_2) exposed to an oxidizing atmosphere will eventually oxidize to triuranium octoxide (U_3O_8). The time it takes to oxidize is a function of burnup and temperature. At temperatures during dry storage system (DSS) fuel loading operations, this reaction can occur within a matter of hours.

The grain boundaries of irradiated fuel are highly populated with voids and gas bubbles. Initially, the grain boundaries are oxidized to U_4O_9, resulting in a slight matrix shrinkage and further opening of the pellet structure. Oxidation then proceeds into the grain until there is complete transformation of the grains to U_4O_9 (Einziger et al. 1992). The grains remain in this phase for a temperature-dependent duration until the fuel resumes oxidizing to the U_3O_8 state. The transformation to U_3O_8 occurs with about 33-percent lattice expansion that breaks the ceramic fragment structure into grain-sized particles. At higher temperatures, these two transformations occur so rapidly that they are difficult to distinguish. The mechanism of oxidation in irradiated fuel appears to be different than in unirradiated fuel where U_3O_7 is formed and oxidation proceeds from the fragment surface and not down the grain boundaries. This mechanistic change occurs between about 10 and 30 gigawatt days per metric ton of uranium (GWd/MTU).

When the UO_2 is in the form of a fuel rod, the expansion of the fuel when it transforms to U_3O_8 induces a circumferential stress in the cladding. Because of the swelling of the fuel, the process is usually initially localized to the original cladding crack site. The cladding strains because of this stress range from 2 to 6 percent before the initial crack starts to propagate along the rod. The incubation time to initiate the propagation and the rate of propagation have an Arrhenius temperature dependence. Axial propagation, spiral propagation, and a combination of the modes that result in splitting have been observed in pressurized-water reactor (PWR) rods (Einziger and Strain 1986).

The database for oxidation was developed mostly in the 1980s in the United States, Canada, England, and Germany. The data usually appear in four forms: (1) O/M ratio (ratio of oxygen to metal content of the oxide) versus time, (2) time to the $UO_{2.4}$ plateau versus time, (3) cladding splitting incubation versus time, and (4) cladding splitting rate versus time. Japanese researchers performed some later work on the effects of oxygen depletion (Nakamura 1995). French researchers are also working on similar questions (Ferry et al. 2005). Work on cladding splitting was done in the early 1980s by researchers in the United States (Einziger and Cook 1984; Einziger and Strain 1986; Johnson et al. 1984) and Canada (Novak and Hastings 1984; Boase and Vandergraaf 1977) and is limited. The Department of Energy (DOE) (Bechtel 2005) has issued an analysis of the oxidation issue in relationship to the handling of potentially breached fuel in a proposed handling facility at a repository. This analysis depends on variables such as the gap between the fuel and the cladding, and burnup in a manner that is currently under technical review. In total, this research has shown that there are a number of variables that can affect the rates at which the fuel oxidizes and the cladding splits: burnup, moisture content of the air, cladding material, and type of initial defect.

The DOE developed a model for fuel oxidation and cladding splitting (Bechtel 2005) for use during long durations at a disposal facility that tries to account for the fuel-to-cladding gap and burnup of the fuel. The gap is the as-measured cold gap and does not account for the closing of the gap as a result of differential thermal expansion of the cladding and fuel material, which could be calculated. There are inadequate data to verify the correctness of the DOE model. Plots in Einziger and Strain (1986) present actual data and comparisons with the data taken by other researchers at 30 GWd/MTU. The measurements of splitting implicitly account for the gap closure. However, no burnup effects can be inferred from these data.

No oxidation or cladding splitting studies have been conducted on fuel with burnup greater than 45 GWd/MTU. Data between 30 and 45 GWd/MTU show a decrease in the oxidation rate as a result of the presence of certain actinides and fission products that are burned into the fuel. There is no reason that this should not continue at higher burnups, but the strength of the effect may change with burnup. Higher burnup fuel (greater than 55 GWd/MTU) forms an external rim on the pellets that consists of very fine grains (1 micron). As indicated earlier, the oxidation process is a grain boundary effect. The fuel pellet should be divided into two regions for the purpose of oxidation analysis: the center of the pellet where the grains have grown slightly, and the rim. While the rate of the oxidation may decrease with burnup, the total amount of fuel that is oxidized may increase because of a much greater intergranular surface area in the rim region. The DOE model (Bechtel 2005) uses a linear decrease in oxidation with burnup, but this has not been substantiated as of yet. A burnup effect is supported by Hanson's analysis (Hanson 1998) of Einziger and Cook's data (1984) from the NRC whole-rod tests, in which defect propagation was observed to occur earlier at the defects at the lower end of the rod where the burnup was lower.

Studies using a low partial pressure of water vapor in air have not shown any dependence of the oxidation rate on the moisture content of the air (Ferry et al. 2005). On the other hand, some studies have shown a large increase in the oxidation rate when the moisture content is above 50 percent of the dew point. Oxidation in a 100-percent steam atmosphere is a different process. Studies also indicate that the oxidation rate will decrease if the oxygen content in the atmosphere drops into the range of a few torr or less (Nakamura 1995). It does not appear that there is an effect of oxygen content at higher oxygen levels, but the data are sparse.

With few exceptions, oxidation studies on fuel have been conducted on light-water reactor fuel (Einziger and Strain 1986; Johnson et al. 1984). However, the UO_2 matrix is essentially the same in both PWR and boiling-water reactor (BWR) fuel. At the higher burnups, oxidation behavior may vary slightly as the actinide and fission product burn-in varies. The effect of the process on the splitting of the cladding may vary considerably because of the difference in gap size between the cladding types, and the thicker cladding in BWR rods.

Limited cladding splitting studies have been conducted on Zircaloy-clad PWR (Einziger and Cook 1984; Einziger and Strain 1986; Johnson et al. 1984) and Canada Deuterium Uranium (CANDU) fuel. Defects were put in the fuel either by a stress-corrosion cracking process producing small, sharp holes, more typical of those found in reactor-initiated stress-corrosion cracking, and by drilling, which produced a larger, duller hole. Most of the defects used in the studies were of the latter type. No measurements were made in cladding above 30 GWd/MTU. Very few data points were measured to determine the splitting rate; therefore, the time to start splitting has to be determined by interpolation. As a result, there is large uncertainty in both measurements. Further, the splitting of other alloy types (e.g., ZIRLO™, M5®) or at higher burnups should be assessed per the design-bases fuel contents. Fuel oxidation would introduce uncertainties for fuel performance and fuel retrievability.

References

Bechtel, "Commercial Spent Nuclear Fuel Handling in Air Study," 000-30R-MGR0-00700-000000, March 2005.

Boase, D.G. and T.T. Vandergraaf, "The Canadian Spent Fuel Storage Canister: Some Materials Aspects," *Nuclear Technology*, Vol. 32, pp. 60–71, 1977.

Einziger, R.E., and J.A. Cook, "LWR Spent Fuel Dry Storage Behavior at 229 °C," HEDLTME 84-17, NUREG/CR-3708, Hanford Engineering Development Laboratory, August 1984.

Einziger, R.E., and R.V. Strain, "Oxidation of Spent Fuel at Between 250° and 360°C," Electric Power Research Institute Report NP-4524, 1986.

Einziger, R.E., L.E. Thomas, H.V. Buchanan, and R.B. Stout, "Oxidation of Spent Fuel in Air at 175 to 195 °C," *J. Nucl. Mater.*, Vol. 190, p. 53, 1992.

Einziger, R.E., S. D. Atkin, D. E. Stellbrecht, and V. Pasupathi, "High Temperature Postirradiation Materials Performance of Spent Pressurized Water Reactor Fuel Rods Under Dry Storage Conditions." *Nuclear Technology*, Vol. 57, p. 65. 1982.

Ferry, C, C. Poinssot, P. Lovera, A. Poulesquen, V. Broudic, C. Cappelaere, L. Desgranges, P. Garcia, C. Jegou, D. Roudil, P. Marimbeau, J. Gras, and P. Bouffioux, "Synthesis on the Spent Fuel Long Term Evolution," Rapport CEA-R6084, 2005.

Hanson, B.D., "The Burnup Dependence of Light Water Reactor Spent Fuel Oxidation," PNNL-11929, Richland, Washington, Pacific Northwest National Laboratory, TIC: 238459, 1998.

Johnson, A.B., E.R. Gilbert, D. Stahl, V. Pasupathi, and R. Kohli, "Exposure of Breached BWR Fuel Rods at 325 °C to Air and Argon," *Proc. NRC Workshop on Spent Fuel/Cladding Reaction During Dry Storage*, Gaithersburg, Maryland, August 1983, NUREG/CR-0049, D. Reisenweaver, Ed., U.S. Nuclear Regulatory Commission, 1984.

Nakamura, J., T. Otomo, T. Kikuchi, and S. Kawasaki, "Oxidation of Fuel Rods under Dry Storage Condition," *J Nuc. Sci. Tech.*, Vol. 32, No. 4, p. 321, April 1995.

Novak, J., and I.J. Hastings, "Post-Irradiation Behavior of Defected UO2 in Air at 220–250 °C," *Proc. NRC Workshop on Spent Fuel/Cladding Reaction During Dry Storage*, Gaithersburg, Maryland, August 1983, NUREG/CR-0049, D. Reisenweaver, Ed., U.S. Nuclear Regulatory Commission, 1984

9 CONFINEMENT EVALUATION

9.1 Review Objective

The objectives of the U.S. Nuclear Regulatory Commission's (NRC's) confinement review of the dry storage system (DSS) and dry storage facility (DSF) with regard to the confinement features and capabilities of the proposed storage container system is to ensure that radiological releases to the environment would be within the limits established by the regulations and that the stored spent fuel cladding and spent fuel assemblies will be sufficiently protected against degradation that might otherwise lead to gross ruptures. In addition, the review evaluates any proposed confinement-related monitoring systems.

9.2 Applicability

This chapter applies to the review of applications for specific licenses for an independent spent fuel storage installation (ISFSI) or a monitored retrievable storage installation (MRS), categorized as a DSF. This chapter also applies to the review of applications for a certificate of compliance (CoC) of a DSS for use at a general license facility. Sections, tables, or paragraphs of this chapter that apply only to a DSF-specific license application (for an ISFSI and MRS) are identified with "**(SL)**". Sections, tables, or paragraphs that apply only to DSS CoC applications have "**(CoC)**". A subsection without an identifier applies to both types of applications.

9.3 Areas of Review

This chapter provides guidance for use in evaluating the design and analysis of the proposed storage container confinement system for normal, off-normal, and accident conditions. This evaluation includes a more detailed assessment of the confinement-related design features and criteria initially presented in the chapters of the applicant's safety analysis report (SAR) on general information and principal design criteria, as well as the proposed confinement monitoring capability, as applicable. In addition, the NRC staff reviews the applicant's analyses that assess the potential releases of radionuclides associated with spent nuclear fuel (SNF) and that estimate their potential leakage to the environment and subsequent impact on a hypothetical individual located at or beyond the controlled area boundary.

This chapter addresses the following areas of review:

- confinement design characteristics
  - design criteria
  - design features
- confinement monitoring capability
- nuclides with potential for release
- confinement analyses
  - normal conditions
  - off-normal conditions (anticipated occurrences)
  - design-basis accident conditions (including natural phenomenon events)
  - identification of release events **(SL)**
  - evaluation of release estimates for SNF and high-level radioactive waste (HLW) **(SL)**
  - evaluation of release estimates for greater-than-Class-C (GTCC) waste **(SL)**
- supplemental information

9.4 Regulatory Requirements and Acceptance Criteria

This section summarizes those parts of Title 10 of the *Code of Federal Regulations* (10 CFR) Part 72, “Licensing Requirements for the Independent Storage of Spent Nuclear Fuel and High-Level Radioactive Waste, and Reactor-Related Greater than Class C Waste,” that are relevant to the review areas this chapter addresses. The NRC staff reviewer should refer to the exact language in the regulations. Tables 9-1a and 9-1b match the relevant regulatory requirements to the areas of review covered in this chapter. The NRC staff reviewer should verify the association of regulatory requirements with the areas of review presented in the matrix to ensure that no requirements are overlooked as a result of unique design features.

Table 9-1a Relationship of Regulations and Areas of Review for a DSF (SL)

Areas of Review	10 CFR Part 72 Regulations							
	72.24	72.44	72.104	72.106	72.120	72.122	72.126	72.128
Confinement Design Characteristics	●				(a)(d)	(a)(b)		(a)
Confinement Monitoring Capability	●	(c)	(a)	(b)		(h)(i)	(d)	(a)
Nuclides with Potential for Release			(a)	(b)				
Confinement Analyses	●	(c)	(a)	(b)			(d)	(a)

Table 9-1b Relationship of Regulations and Areas of Review for a DSS (CoC)

Areas of Review	10 CFR Part 72 Regulations		
	72.230	72.234	72.236
Confinement Design Characteristics	●	●	(a)(b)(d)(e)(g) (j)(l)
Confinement Monitoring Capability	●	●	(d)(e)(g)(l)
Nuclides with Potential for Release	●	●	(a)(d)
Confinement Analyses	●	●	(a)(d)

As prescribed in 10 CFR Part 72, the regulatory requirements for doses at and beyond the controlled area boundary include both the direct dose (i.e., from shielding review) and that from an estimated release of radionuclides to the atmosphere (based on the leak test of the confinement). Thus, an overall assessment of the compliance of the proposed DSS with these regulatory limits is presented in Chapter 10, “Radiation Protection Evaluation,” of this standard review plan (SRP). In addition, the performance of the storage container confinement system under accident conditions, as evaluated in this chapter, may also be addressed in the overall accident analyses presented in Chapter 16, “Accident Analysis Evaluation,” of this SRP.

In general, the DSS or DSF confinement evaluation seeks to ensure that the proposed design fulfills the following acceptance criteria, which the NRC staff considers to be minimally acceptable, to meet the confinement requirements in 10 CFR Part 72.

9.4.1 Confinement Design Characteristics

The design should provide redundant sealing of the confinement boundary (10 CFR 72.236(e)). Typically, this means that field closures of the confinement boundary should either have two seal welds or two metallic O-ring seals.

The confinement design should be consistent with the regulatory requirements as well as the applicant's general design criteria, reviewed in accordance with Chapter 3, "Principal Design Criteria Evaluation," of this SRP. The NRC staff has previously accepted construction of the primary confinement barrier in conformance with the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (B&PV) Code, Section III, "Rules for Construction of Nuclear Facility Components," Division 1, Subsections NB or NC. The B&PV Code defines the standards for all aspects of construction, including materials, design, fabrication, examination, testing, inspection, and certification, required in the manufacture and installation of components. In such instances, the staff has relied upon Section III to define the minimum acceptable margin of safety. Therefore, the applicant must fully document and completely justify any deviations from the specifications of Section III. In some cases, after careful and deliberate consideration, the staff has made exceptions to this requirement. In addition, ASME published in 2005 Division 3 to Section III, which is written specifically for containments for the transportation and storage of SNF, but the NRC has not yet endorsed it.

The design must provide a nonreactive environment to protect fuel assemblies against fuel matrix degradation and fuel cladding degradation, which might otherwise lead to gross rupture (Knoll and Gilbert 1987). Measures for providing a nonreactive environment within the confinement storage container typically include drying and backfilling with a nonreactive cover gas (such as helium). To reduce the potential for fuel oxidation and subsequent cladding failure, an inert atmosphere (e.g., helium cover gas) has been used for storing uranium dioxide (UO_2) SNF in a dry environment. Chapter 11, "Operation Procedures and Systems Evaluation," of this SRP provides more detailed information on the cover gas filling process. Note that other fuel types, such as graphite fuels for the high-temperature, gas-cooled reactors, may not exhibit the same oxidation reactions as UO_2 fuels and, therefore, may not require an inert atmosphere; however, the application should discuss the prevention of fuel and cladding degradation.

(SL) The SAR must describe the confinement system for SNF, HLW, and waste management facilities. Chapter 13, "Waste Management Evaluation," of this SRP discusses the review of waste management facilities.

(SL) If appropriate, the SAR must also describe the confinement features or system implemented for reactor-related GTCC waste. The applicant should provide assurance that the reactor-related GTCC waste will be adequately contained and shielded under normal, off-normal, and accident conditions in accordance with the 10 CFR Part 72 dose limits.

9.4.2 Confinement Monitoring Capability

(SL) Confinement monitoring for an ISFSI and MRS has two aspects. The first is monitoring storage confinement closure seals or overall closure effectiveness. The second is providing a system to measure radionuclides released to the environment under normal, off-normal, and accident conditions. This second aspect includes all areas where there is the potential for significant releases to the environment and may include storage containers, pool facilities, and waste management facilities; Chapter 13 discusses the review of releases other than from storage containers, such as pool facilities and waste management facilities. The SAR should present a discussion of the extent of monitoring required consistent with 10 CFR Part 72 requirements for both of these aspects of confinement monitoring.

The application should describe the proposed monitoring capability and surveillance plans for mechanical closure seals. In instances involving welded closures, the staff has accepted that no closure monitoring system is required. This practice is consistent with the fact that other welded joints in the confinement system are not monitored because the initial staff review considers the integrity of the confinement boundary for the licensing period. For welded closures, typical surveillances include checking for blockage of the air vents or temperature monitoring.

To show compliance with the requirement for continuous monitoring, 10 CFR 72.122(h)(4), storage container vendors have proposed, and the staff has accepted, routine surveillance programs and active instrumentation to meet the continuous monitoring requirements.

(SL) For reactor-related GTCC waste**,** the SAR should describe the programs and procedures in place to maintain confinement of the GTCC waste and prevent degradation of the waste form and containers. In general, the SAR should describe programs that give full consideration to maximum anticipated storage time for any projected corrosion to ensure that the dose limits established in 10 CFR Part 72 are not exceeded.

9.4.3 Nuclides with Potential for Release

Verify that the applicant estimated the maximum credible quantity of radionuclides with the potential for release to the environment. The radionuclides potentially available for release to the environment would be based on or derived from the same calculation as the radiological source term presented in Chapter 6, “Shielding Evaluation,” of this SRP.

9.4.4 Confinement Analyses

The application should specify the maximum allowed leakage rates for the total primary confinement boundary and redundant seals. The maximum allowed leakage rate is based on the “as-tested” leak rate measured by the leak test performed on the entire confinement boundary. Generally, as discussed below in the review procedures, the applicant evaluates the allowable leakage rate for its radiological consequences and its effect on maintaining an inert atmosphere within the storage container. However, the analyses discussed below are unnecessary[1] for a storage container, including its closure lid, that is designed and tested to be “leaktight” as defined in American National Standards Institute (ANSI) N14.5, “American National Standard for Leakage Tests on Packages for Shipment of Radioactive Materials.” Additional items to consider include the following:

- The analysis of potential releases should be consistent with the methods described in ANSI N14.5.

[1] The guidance provided in Sections 9.5.3 and 9.5.4 is not applicable for casks that are demonstrated to be “leaktight,” as defined in ANSI N14.5, recognizing that confinement boundary failure under design-basis normal, off-normal, and accident conditions is not acceptable and that the confinement boundary is to remain “leaktight” under all conditions.

- During normal operations and anticipated occurrences, verify that dose calculations based on the allowable leakage rate demonstrate that the annual dose equivalent to any "real individual" who is located beyond the controlled area does not exceed the limits given in 10 CFR 72.104(a).

- For any design-basis accident, verify that dose calculations based on the allowable leakage rate demonstrate that an individual at the boundary or beyond the nearest boundary of the controlled area does not receive a dose that exceeds the limits given in 10 CFR 72.106(b) (discussed further in Chapter 16 of this SRP).

- Verify that the analysis of potential releases demonstrates that an inert atmosphere will be maintained within the storage container during the licensed storage lifetime.

- For storage containers that employ a pressurized inert gas to facilitate internal natural convection heat transfer, verify that the analysis of potential releases demonstrates that the pressurized atmosphere will be maintained within the storage container and keep temperatures below allowable limits during the licensed storage lifetime.

9.4.5 Supplemental Information

The application should include all supportive information or documentation that justifies assumptions or analytical procedures.

9.5 Review Procedures

Figures 9-1a and 9-1b show the interrelationship between the confinement evaluation and the other areas of review described in this SRP for specific licenses and CoC applications, respectively. The text within those chapters and sections that are related to confinement will help the confinement review.

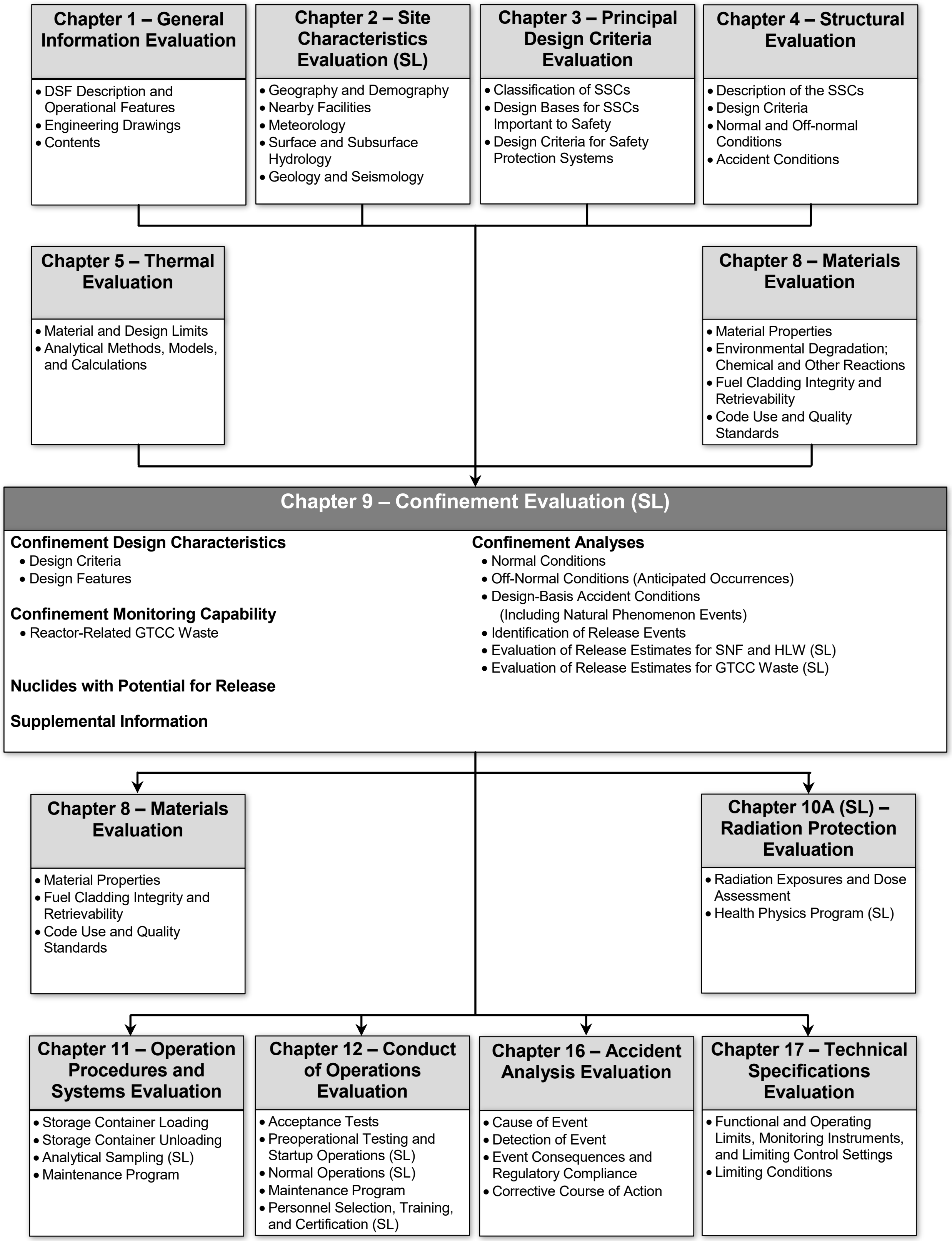

Figure 9-1a Overview of Confinement Evaluation of Specific License Applications for a DSF (SL)

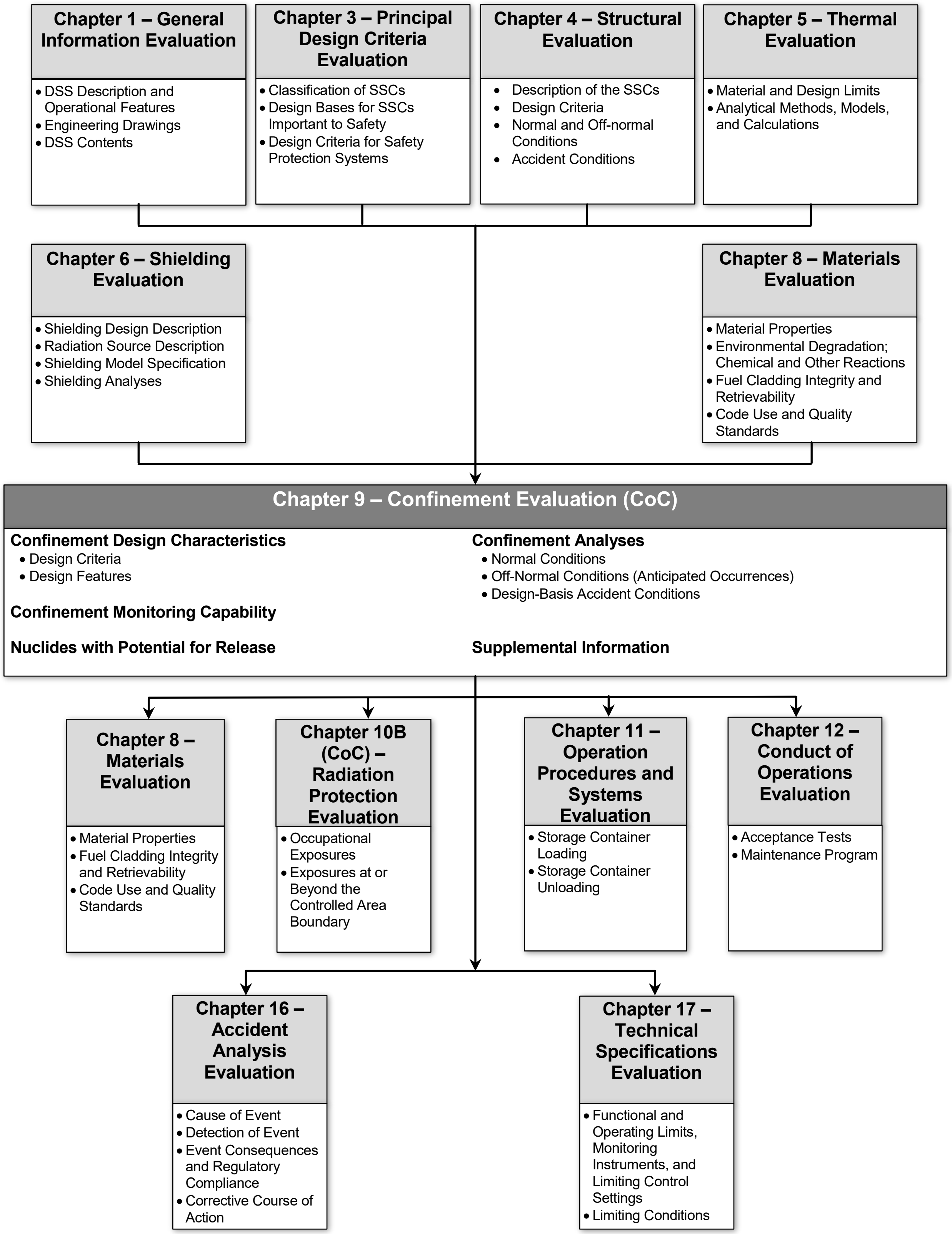

Figure 9-1b Overview of Confinement Evaluation of Applications for a DSS (CoC)

9.5.1 Confinement Design Characteristics

9.5.1.1 *Design Criteria*

Review the principal design criteria presented in the SAR, as well as any additional detail provided in the chapter of the SAR on confinement.

9.5.1.2 *Design Features*

Review the general description of the storage container presented in the SAR, as well as any additional information provided in the chapter of the SAR on confinement. Verify that all drawings, figures, and tables describing confinement features provide sufficient detail to support in-depth staff evaluation.

Verify that the applicant has clearly identified the confinement boundaries. This identification should include the confinement vessel, its penetrations, valves, seals, welds, and closure devices and corresponding information concerning the redundant sealing. Details of the closure confinement boundary are found in Section 8.5.3, especially Figures 8-2 and 8-3, of this SRP.

Verify that the design and procedures provide for drying and evacuation of the storage container interior as part of the loading operations. Also, verify that the confinement design is acceptable for the pressures that may be experienced during normal, off-normal, and accident conditions.

Verify that, on completion of storage container loading and reaching thermal equilibrium, the gas fill of the storage container interior is at a positive pressure level that is expected to maintain a nonreactive environment and heat transfer capabilities of the storage container interior under both normal and off-normal conditions and events for the license period. Verification can include pressure testing, leak testing, seal monitoring, and maintenance for storage containers with seals that are not welded if these are included as described in Chapter 17, “Technical Specifications Evaluation,” of this SRP as conditions of use. Acceptance tests for pressure testing and leak testing are described in Sections 12.5.2.1, “Structure and Pressure Tests,” and 12.5.2.2, “Leak Tests,” of this SRP. Testing and writing the helium leak test procedures should be performed by qualified personnel. NRC Information Notice 2016-04, “ANSI N14.5-2014 Revision and Leakage Rate Testing Considerations,” contains additional relevant information on leak testing and should be reviewed. In addition, review details of leak testing are found in Section 8.5.3.3.2, “Helium Leakage Testing,” of this SRP.

Coordinate with the structural and materials disciplines conducting reviews under Chapter 4, “Structural Evaluation,” and Chapter 8, “Materials Evaluation,” of this SRP, respectively, to ensure that the applicant has provided proper specifications for all welds and, if applicable, that the bolt torque for closure devices is adequate and properly specified. If applicable, verify the capability of the seal to maintain long-term closure. Because of the performance requirements over the license period (e.g., 20 years, 40 years), evaluate the potential for seal deterioration associated with bolted closures (NRC Information Notice 2013-07). The NRC staff has accepted only metallic seals for the primary confinement. Details on seals are found in Section 8.5.10, “Seals,” of this SRP. Coordinate this review with the thermal discipline to ensure that the operational temperature range for the seals (specified by the manufacturer) will not be exceeded. For specific licenses in which originally offsite canisters are to be stored, ensure that the integrity of the confinement boundary and the content discussed in the SAR reflects their condition after arrival and being loaded at the site.

Welded canisters can be used as a confinement system, provided the following design and qualification guidance, as appropriate, is met:

- The canister is constructed from austenitic stainless steel.
- The confinement welds meet the guidance of Chapter 8 of this SRP.
- The canister maintains its confinement integrity during normal conditions, anticipated occurrences, and credible accidents and natural phenomena as required in 10 CFR Part 72.
- The canister shell has been helium leak tested before its loading as required by 10 CFR 72.236(j). This test demonstrates that the canister is free of defects that could lead to a leakage rate greater than the design-basis leakage rate, which could result in doses at the control area boundary in excess of the regulatory limits.
- Activities related to inspection, evaluation, documentation of fabrication, and closure welding of canisters are to be performed in accordance with an NRC-approved quality assurance program as required in 10 CFR Part 72, Subpart G, “Quality Assurance.”

(SL) For reactor-related GTCC waste, review the general description of the reactor-related GTCC waste confinement systems presented in the SAR. Verify that the programs and procedures in place concerning the confinement system for reactor-related GTCC waste are clearly identified in relation to the form of the GTCC waste. Acceptable program descriptions specify the maximum leakage rate from each reactor-related GTCC container or the maximum leakage rate permitted from the total reactor-related GTCC inventory at the ISFSI or MRS.

9.5.2 Confinement Monitoring Capability

The NRC staff has found that storage containers closed entirely by welding do not require seal monitoring. However, for storage containers with bolted closures, the staff has found that a seal monitoring system is required to adequately demonstrate that seals can function to limit releases and maintain an inert atmosphere in the storage container. A seal monitoring system, combined with periodic surveillance, enables a determination as to when to take corrective action to maintain safe storage conditions.

Although the details of the monitoring system may vary, the general design approach has been to pressurize the region between the redundant seals with a nonreactive gas to a pressure greater than that of the storage container cavity and the atmosphere. The storage container lid design should prevent exposure of the outer seal to the atmosphere and potential resulting deterioration. The monitoring system is leak tested to the same leak rate as the confinement boundary. Installed instrumentation is routinely checked per surveillance requirements. A decrease in pressure between these seals indicates that the nonreactive gas is leaking either into the storage container cavity or out to the atmosphere. For normal operations, radioactive material should not be able to leak to the atmosphere; hence, this design allows for detecting a faulty seal without radiological consequence. Note that the volume between the redundant seals should be pressurized using a nonreactive gas, thereby preventing contamination of the interior cover gas.

If the region between redundant, confinement boundary, mechanical seals is maintained at a pressure greater than that in the storage container cavity, the monitoring system boundaries are tested to a leakage equal to the confinement boundary, the pressure is routinely checked, and the instrumentation is verified to be operable in accordance with a technical specification surveillance requirement, the NRC staff has accepted that no discernible leakage is credible for the pressure monitoring system and, therefore, the pressure monitoring system does not have to be included in the confinement dose calculations at the controlled area boundary from atmospheric releases during normal conditions.

The staff has accepted the classification of monitoring systems as not important to safety for those systems designed such that failure of the monitoring system alone would not result in a gross release of radioactive material. This is because, although its function is to monitor confinement seal integrity, the failure of the monitoring system alone would not result in a gross release of radioactive material. It is classified as not important to safety because most of the associated hardware has not met the program controls important to safety, such as design or procurement. Consequently, the monitoring system for bolted closures need not be designed to the same requirements as the confinement boundary (i.e., ASME B&PV Code, Section III). Additional review details associated with monitoring are described in Section 3.5.3.2, “Other Safety Protection Systems,” of this SRP.

Depending on the monitoring system design, there could be a lag time before the monitoring system indicates a postulated degraded seal leakage condition. Degraded seal leakage is leakage greater than the tested rate that is not identified within a few monitoring system surveillance cycles. The occurrence of a degraded seal without detection is considered a “latent” condition and should be presumed to exist concurrently with other off-normal and design-basis events (see Section 3.5.2.4, “External Conditions,” of this SRP). Verify that once the degraded seal condition is detected, the storage container user will initiate corrective actions.

For the “latent” condition, the monitoring system boundary would remain intact, and this condition would be bounded by the off-normal analysis. If the monitoring system would not maintain integrity under design-basis accident conditions, additional safety analysis may be necessary. The staff recognizes that the possibility of a degraded seal condition is small and that the possibility of a degraded seal condition concurrent with a design-basis event that breaches the monitoring system pressure boundary is very remote. However, these probabilities have not been quantified. To address this concern, the staff has accepted a demonstration that the dose consequences of this event are within the limits of 10 CFR 72.106(b).

Verify that the specified pressure of the gas in the monitored region is higher than both the storage container cavity and the atmosphere. Coordinate with the structural and thermal reviewers (Chapter 4 and Chapter 5, respectively) of this SRP to verify the pressure in the storage container cavity.

Verify that the SAR indicates the total volume of gas in the cavity is such that normal seal leakage will not cause all of the gas to escape over the lifetime of the storage container. Confirm that the proposed maximum leakage rate is based on the confinement evaluation described in Sections 9.5.3 and 9.5.4 below. Verify that the maximum allowable leakage rate is specified as a minimum acceptance test criterion in the chapters of the SAR on acceptance tests and the maintenance program and on technical specifications and operating controls and limits, even though the actual leakage rate of the seals is expected to be significantly lower.

For redundant welded closures, ensure that the applicant has provided adequate justification that the welds have been sufficiently designed, fabricated, tested, and examined to ensure that the weld will behave similarly to the adjacent parent material of the storage container.

Verify that any leakage test, monitoring, or surveillance conditions are appropriately and consistently specified in the chapters of the SAR on acceptance tests and the maintenance program, accident analysis, and technical specifications and operational controls and limits and in the CoC, as applicable. Discussion of acceptance tests is in Section 12.5.2, “Acceptance Tests,” of this SRP.

9.5.3 Nuclides with Potential for Release

For determination of the radionuclide inventory available for release, the NRC staff has accepted, as a minimum for the analysis, the activity from the cobalt-60 in the crud, the activity from iodine, fission products that contribute greater than 0.1 percent of design-basis fuel activity, and actinide activity that contributes greater than 0.01 percent of the design-basis activity. In some cases, the applicant may have to consider additional radioactive nuclides, depending on the specific analysis. The total activity of the design-basis fuel should be based on the storage container design loading that yields the bounding radionuclide inventory (considering initial enrichment, burnup, and cool time). If necessary, the output of the depletion codes used by the shielding reviewer can provide nuclide quantities and can be used as an independent confirmation of the values described in the SAR confinement chapter.

The staff has determined that, as a minimum, the fractions of radioactive materials available for release from SNF, provided in Table 9-2 for pressurized-water reactor (PWR) fuel and boiling-water reactor (BWR) fuel for normal, anticipated occurrences (off-normal), and accident conditions, should be used in the confinement analysis to demonstrate compliance with 10 CFR Part 72. These fractions account for radionuclides trapped in the fuel matrix and radionuclides that exist in a chemical or physical form that is not releasable to the environment under credible normal, off-normal, and accident conditions. Other release fractions may be used in the analysis, provided the applicant properly justifies the basis for their usage. For example, the staff has accepted, with adequate justification, reduction of the mass fraction of fuel fines that can be released from the storage container. Also, when an applicant uses the release fractions in Table 9-2, ensure that the condition of the fuel described in the SAR is bounded by the experimental data presented in NUREG/CR-6487, “Containment Analysis for Type B Packages Used to Transport Various Contents,” issued November 1996. Specifically, these experimental data are based on low burnup fuel and the release from a single breach of one fuel rod; these data should not be used for SNF described as damaged. The reviewer may consider other release fractions for conditions other than those described in NUREG/CR-6487 if the applicant has provided adequate justification.

For fuel rods that are classified as damaged, verify that the applicant has established release fractions (particulates, gases, cred, volatiles) for normal/accident conditions based on applicable physical data and other analyses that account for the specific type of fuel, estimated number of damaged fuel rods, presence of a damaged fuel can, impacts of accidents, and damaged condition of the DSS following an impact.

Fuel rods that are damaged because of a preloading cladding breach may not have a driving force for the release of particulate from the rod under normal or off-normal conditions, providing the canister is not pressurized. However, under an impact accident, damaged fuel rods might release additional fuel fines from the fracture of the fuel, especially the rim region in high burnup fuel. In addition, some canisters may be pressurized to several atmospheres and storage container blowdown could also affect releases. Alternatively, a leak-tight confinement boundary may be specified to preclude the release analyses of damaged fuel.

Table 9-2 Fractions of Radioactive Materials Available for Release from Spent Fuel[a]

Variable	Fractions Available for Release[b]	
	PWR and BWR Fuel	
	Normal and Off- normal Conditions	Design-Basis Accident Conditions
Fraction of Fuel Rods Assumed to Fail	0.01 (normal) 0.10 (off-normal)	1
Fraction of Gases Released from a Cladding Breach, f_G[c]	0.3	0.3
Fraction of Volatiles Released from a Cladding Breach, f_V[c]	$2x10^{-4}$	$2x10^{-4}$
Mass Fraction of Fuel Released as Fines from a Cladding Breach, f_F	$3x10^{-5}$	$3x10^{-5}$
Fraction of Crud that Spalls Off Cladding, f_C	0.15[d]	1.0[d]

a Values in this table are taken from NUREG/CR-6487.

b Except for cobalt-60, only failed fuel rods contribute significantly to the release. Total fraction of radionuclides available for release should be multiplied by the fraction of fuel rods assumed to have failed.

c In accordance with NUREG/CR-6487, gases species include hydrogen-3, iodine-129, krypton-81, krypton-85, and xenon-127; volatile species include cesium-134, cesium-135, cesium-137, ruthenium-103, ruthenium-106, strontium-89, and strontium-90.

d The source of radioactivity in crud is cobalt-60 on fuel rods. At the time of discharge from the reactor, the specific activity, S_c, is estimated to be 140 microcuries per square centimeter (μCi/cm^2) for PWRs and 1,254 μCi/cm^2 for BWRs. Total cobalt-60 activity is this estimate times the total surface area of all rods in the storage container (Sandoval et al. 1991). Decay of cobalt-60 to determine activity at the minimum time before loading is acceptable.

The staff has accepted the rod breakage fractions in Section 5.5.4.6, “Pressure Analysis,” of this SRP for the confinement evaluations. It is important to recognize that confinement boundary failure under design-basis normal, off-normal, or accident conditions is not acceptable. Confinement boundary structural integrity during design-basis conditions is confirmed by the structural analysis. The confinement analyses demonstrate that, at the measured leakage rates and assumed relevant nominal meteorological conditions, the requirements of 10 CFR 72.104(a) and 10 CFR 72.106(b) can be met. Each DSS or DSF is also required to have a site-specific confinement analysis and dose assessment to demonstrate environmental compliance with these regulations for SNF, HLW, and reactor-related GTCC waste containers.

9.5.4 Confinement Analyses

In general, the NRC evaluates analyses for normal, off-normal, and accident conditions. The reviewer should note that the dose limits differ between 10 CFR 72.104(a) (annual limits for normal plus off-normal conditions) and 10 CFR 72.106(b) (limit per event for accident conditions). For 10 CFR 72.104(a), the limits are for whole body dose and doses to the thyroid and any other critical organ. These limits are based on the methodology in the International Commission on Radiological Protection (ICRP) Publication 2, “Report of Committee II on Permissible Dose for Internal Radiation,” issued in 1959. For 10 CFR 72.106(b), the limits are for total effective dose equivalent (TEDE), the sum of the deep dose equivalent (DDE) and the committed dose equivalent (CDE) for any individual organ or tissue, lens dose equivalent (LDE), and shallow dose equivalent (SDE) to skin or any extremity. These limits are based on the methodology in ICRP Report 26, “Recommendations of the International Commission on Radiological Protection,” issued in 1977. As noted in SRP Chapter 10A, “Radiation Protection Evaluation for Dry Storage

Facilities," and later in this chapter, the NRC has accepted the use of other dose quantities as surrogates for whole body dose, that is, TEDE and effective dose equivalent from external exposures (EDEX).

Review the applicant's confinement analysis and the resulting doses for the normal, off-normal, and accident conditions at the controlled area boundary. The analysis typically includes the following common elements:

- calculation of the specific activity (curies per cubic centimeter) for each radioactive isotope in the storage container cavity based on rod breakage fractions, release fractions, isotopic inventory, and cavity free volume

- using the tested leak rate and conditions during testing as input parameters, calculation of the adjusted maximum confinement boundary leakage rates (cubic centimeters per second) under normal, off-normal, and accident conditions (e.g., temperatures and pressures)

- calculation of isotope specific leak rates Q_i (curies per second) by multiplying the isotope specific activity by the maximum confinement boundary leakage rates for normal, off-normal, and accident conditions

- determination of doses for which limits are specified in 10 CFR 72.104(a) and 10 CFR 72.106(b) from inhalation and immersion exposures at the controlled area boundary (considering atmospheric dispersion factors -χ/Q; (second per cubic meters), as described in Regulatory Guide (RG) 1.145, "Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants")

Verify that the application specifies maximum allowable "as tested" confinement boundary leakage rates as a technical specification, as discussed in SRP Chapter 17. Guidance on the calculations of the specific activity for each isotope in the storage container and the maximum allowable helium confinement boundary leakage rates for normal, off-normal, and accident conditions can be found in NUREG/CR-6487 and ANSI N14.5. The minimum distance between the storage containers and the distance to the controlled area boundary is generally also a design criterion; however, 10 CFR 72.106(b) requires this distance to be at least 100 meters (328 feet) from the DSS or DSF.

For dose calculations, the NRC staff has accepted the use of either an adult breathing rate (BR) of 2.5×10^{-4} cubic meters per second (m^3/s) (8.8×10^{-3} cubic feet per second (ft^3/s)), as specified in RG 1.109, "Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I," or a worker breathing rate of 3.3×10^{-4} m^3/s (1.2×10^{-2} ft^3/s), as specified in the Environmental Protection Agency (EPA) Federal Guidance Report No. 11, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion," issued September 1988. Ensure that the calculation uses the dose conversion factors (DCFs) in EPA Federal Guidance Report No. 11 for committed effective dose equivalent (CEDE—the total dose to the body from internal exposures) and the CDE (the dose to an organ from internal exposures) for the thyroid and other organs from inhalation. Confirm that the SAR reflects a bounding DCF from EPA Federal Guidance Report No. 11 for each isotope unless the applicant justifies an alternate value. The staff does not accept weighting or normalization of the DCFs. For each isotope (i), $CEDE_i$ or CDE_i is calculated as follows:

$$CEDE_i \text{ or } CDE_i = Q_i * DCF_i * \chi/Q * BR * \text{Duration} * \text{conversion factor}$$

The conversion factor, if required, converts the input units into the desired form (e.g., Sv, rem, mrem, mSv). The duration term is 1 year for normal conditions and an appropriate duration for each individual off-normal and accident condition. Thus, the results should be in terms of mrem. However, the dose for an off-normal condition is summed with the annual normal condition dose to give a total annual dose (mrem in a year) to evaluate compliance with 10 CFR 72.104(a) limits. This also applies to the calculations of doses described in the equations below.

For the contributions to the EDEX (total dose to the body from external exposures) and the dose equivalent (DE_{ext}) to the thyroid, other organs, and the skin from air submersion (external) exposure, ensure that the SAR reflects the DCFs in EPA Federal Guidance Report No. 12, "External Exposure to Radionuclides in Air, Water, and Soil," issued September 1993. Again, the NRC staff does not accept weighting or normalization of the DCFs.

The $EDEX_i$, the $DE_{ext,i}$ for each organ, and the SDE_i are calculated as follows:

$$EDEX_i,\ DE_{ext,I}, \text{ or } SDE_i = Q_i * DCF_i * \chi/Q * BR * \text{Duration} * \text{conversion factor}$$

The description above for the duration and conversion-factor terms apply in this equation as well. Summing the calculated doses over all isotopes (i) results in the total effluent contributions for the CEDE, EDEX, CDE, DE_{ext}, and SDE. For compliance with the limits in 10 CFR 72.104(a) and 10 CFR 72.106(b) that include internal and external dose contributions, ensure that the SAR uses the following equations:

TEDE = CEDE + EDEX

For a given organ or tissue, the total dose to the organ or tissue = CDE + DE_{ext}

10 CFR 72.106(b) organ doses = EDEX + CDE

As already described, the actual dose limits in 10 CFR 72.104(a) include a limit for whole body dose. EPA Federal Guidance Report Nos. 11 and 12 do not give DCFs for whole body dose because of the differences in dose methodology compared to the regulatory limit. However, as noted in Chapter 10A of this SRP, the NRC has accepted the use of TEDE as a surrogate for whole body dose. Based on information in NRC's regulatory guidance, the EDEX may also be an appropriate surrogate for whole body dose when doses are calculated for uniform body exposures associated with semi-infinite cloud dose modeling (see RG 1.195, "Methods and Assumptions for Evaluating Radiological Consequences of Design Basis Accidents at Light-Water Nuclear Power Reactors," Section 4.1.4). This calculation approach is consistent with the analysis assumptions that are the basis of the confinement evaluation.

The limits in 10 CFR 72.104(a) include limits for critical organs. EPA Federal Guidance Report Nos. 11 and 12 give DCFs for some of the critical organs for the radionuclides (critical organs vary from one radionuclide to another) considered in the confinement analysis. Because the doses from effluents have been very small compared to the 10 CFR 72.104(a) dose limits and compared to the direct radiation doses for the analyzed organs, the NRC expects that the doses to the other critical organs for the analyzed radionuclides, for which DCFs are not provided, would also be similarly very small. However, in cases where analyzed organ doses from effluents are relatively significant and analyzed doses are close to the limits, calculations for the other critical organs using appropriate methods may be necessary.

Note that the actual organ dose limits in 10 CFR 72.106(b) are stated to be the summation of the CDE for the organ or tissue and the DDE. However, EPA Federal Guidance Report No. 12 does not include DCFs for the DDE. A true calculation of DDE may likely require the use of computer codes that are capable of analyses for external doses from effluent plumes and that include DDE as an analytical result. The DCFs in Report No. 12 calculate the EDEX. Based on information in NRC's regulatory guidance, the EDEX is nominally equivalent to the DDE if the whole body is irradiated uniformly for submergence (in a semi-infinite cloud) exposure situations (see RG 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," Section 4.1.4). The assumptions that are the basis of the confinement evaluation are consistent with these conditions. Hence the equation above for the 10 CFR 72.106(b) organ doses is written using the EDEX instead of the DDE.

The limits in 10 CFR 72.106(b) also include limits for LDE. EPA Federal Guidance Report No. 12 does not include DCFs for LDE. While not the same as LDE, the DDE and the SDE may be acceptable surrogates for estimating the LDE based on the following. Various National Council on Radiation Protection (NCRP) and ICRP reports (e.g., NCRP Report Number 122, "Use of Personal Monitors to Estimate Effective Dose Equivalent and Effective Dose to Workers For External Exposure to Low-LET Radiation," issued in 1995, and ICRP Publication 103, "The 2007 Recommendations of the International Commission on Radiological Protection") indicate that SDE and DDE may be used for LDE under certain conditions, including the following, which are consistent with the analysis approach for the confinement evaluation. First, the analyses assume uniform external exposure of the body from an effluent plume. Second, the effluent contribution to dose is minor compared to the contribution from direct radiation, or the total dose is significantly less than the regulatory limits. Additionally, Bordy (2015) indicates that the SDE and DDE can be bounding for LDE over most gamma energies of interest. That SDE and DDE do not bound LDE for all gamma energies of interest would be acceptable given the second condition described above for using SDE and DDE to estimate LDE. For instances where the second condition is not met, an appropriately justified factor should be applied to the DDE or SDE to account for gamma energies where they would under predict LDE.

9.5.4.1 *Normal Conditions*

For normal conditions, a bounding exposure duration assumes that an individual is present at the controlled area boundary for 1 full year (8,760 hours). The NRC staff may consider an alternative exposure duration if the applicant provides justification.

Because any potential release resulting from confinement boundary leakage would typically occur over a substantial period of time, the staff has accepted calculation of the atmospheric dispersion factors (χ/Q) according to RG 1.145, assuming D-stability diffusion and a wind speed of 5 meters per second (m/s) (16 feet per second (ft/s)).

(SL) For a DSF, the number of storage containers will be known based on the (proposed) license condition that limits the amount of SNF, HLW, and reactor-related GTCC waste that can be stored at the facility. Thus, the analyses for normal conditions should be for the planned facility storage container array(s) and the number of storage containers that will be used at the DSF.

(CoC) As noted above, a DSF will have multiple storage containers. When reviewing a DSS, therefore, confirm that the resulting doses from a single storage container will be a small fraction of the limits prescribed in 10 CFR 72.104(a) to accommodate an array of storage containers and the external direct dose.

9.5.4.2 *Off-Normal Conditions (anticipated occurrences)*

Off-normal conditions can affect confinement in a variety of ways (e.g., temperature and pressure within the storage container, larger release pathway); Section 9.5.2 above and Chapter 3 of this SRP provide further discussion on off-normal considerations. For off-normal conditions, the bounding exposure duration and atmospheric dispersion factors (χ/Q) are the same as those discussed above for normal conditions.

To demonstrate compliance with 10 CFR 72.104(a), the staff has accepted dose calculations for releases from a single storage container undergoing off-normal conditions. However, the dose contribution from storage container leakage should also be a fraction of the limits specified in 10 CFR 72.104(a) because the doses from normal conditions and doses from other radiation sources are added to this contribution. Coordinate this review with the SRP Chapters 6 and 10A/10B reviewers.

9.5.4.3 *Design-Basis Accident Conditions (including natural phenomenon events)*

For accident-level conditions, the duration of the release is assumed to be 30 days (720 hours). A bounding exposure duration assumes that an individual is also present at the controlled area boundary for 30 days. This time period is the same as that used to demonstrate compliance for reactor facilities licensed in accordance with 10 CFR Part 50, “Domestic Licensing of Production and Utilization Facilities,” and provides good defense in depth because recovery actions to limit releases are not expected to exceed 30 days.

For accident conditions, the staff has accepted calculation of the atmospheric dispersion factors (χ/Q) of RG 1.145 on the basis of F-stability diffusion and a wind speed of 1 m/s (3.3 ft/s). (Note: RG 1.23, “Meteorological Monitoring Programs for Nuclear Power Plants (Safety Guide 23),” and RG 1.111, “Methods for Estimating Atmospheric Transport and Dispersion of Gaseous Effluents in Routine Releases from Light-Water-Cooled Reactors,” provide background information that describes atmospheric dispersion and deposition parameters.)

To demonstrate compliance with 10 CFR 72.106(b), the staff has accepted dose calculations for releases of radionuclides from a single storage container.

9.5.4.4 *Identification of Release Events* ***(SL)***

Discuss the proposed site operations with other reviewers (e.g., structural, operations, site characteristics) to determine the spectrum of events that should be considered for the specific site. Focus on the physical condition of the confinement system for normal operations and off-normal operations, and for design-basis accidents. Use these discussions to understand (1) the physical condition of the equipment that might serve to contain radionuclides, and (2) the forces (e.g., physical displacement, pressure differences, temperatures) that could move radionuclides into the accessible environment if the confinement system fails. Categorize the selected events as either (1) normal operations and off-normal operations or (2) design-basis accidents.

9.5.4.5 *Evaluation of Release Estimates for Spent Nuclear Fuel and High-Level Radioactive Waste **(SL)***

Refer to Sections 9.5.3 and 9.5.4 (through 9.5.4.4) of this chapter.

9.5.4.6 *Evaluation of Release Estimates for Reactor-Related Greater than Class C Waste **(SL)***

The issues considered for an evaluation of release estimates for reactor-related GTCC waste are similar to those for SNF; however, the activity and release associated with reactor-related GTCC may be less than that for SNF. For reactor-related GTCC waste, verify that the SAR, at a minimum, presents a clear description of the operating limits regarding the confinement features of the reactor-related GTCC storage design or system. Verify that the application identifies the quantity of radionuclides that would be released to the environment from the ISFSI or MRS during normal operations, off-normal operations, and design-basis accidents. The estimates should be based on an evaluation of the reactor-related GTCC waste form and the physical process that will move radionuclides into the environment or retain them in the confinement system.

Verify that the confinement system, analyses, and procedures demonstrate, with reasonable assurance, that for the package contents and assumed nominal meteorological conditions, the requirements of 10 CFR 72.104(a) and 10 CFR 72.106(b) can be met.

Analysis methods that determine the dose limits are not exceeded may include those used for SNF evaluations. Verify that the reactor-related GTCC dose calculations use the assumptions used for SNF (e.g., meteorological conditions, DCFs, breathing rates, distance of the real individual) unless the applicant can justify alternative assumptions. The applicant must adequately justify the value of the release fractions based on the form of reactor-related GTCC waste and the design of the container.

Verify that each ISFSI or MRS has a site-specific confinement analysis and dose assessment to demonstrate regulatory compliance. Meteorological conditions similar to those used to perform the confinement analyses for SNF or HLW should be used in the analysis. For DCFs, the NRC has accepted the use of EPA Federal Guidance Report Nos. 11 and 12.

9.5.5 Supplemental Information

Ensure that all supportive information or documentation has been provided or is readily available. This includes, but is not limited to, justification of assumptions or analytical procedures, test results, photographs, computer program descriptions, input and output, and applicable pages from referenced documents. Request any additional information needed to complete the review. Consider relevant generic communications (e.g., NRC information notices) as part of the review.

9.6 Evaluation Findings

The NRC reviewer should prepare evaluation findings upon satisfaction of the regulatory requirements in Section 9.4. If the documentation submitted with the application fully supports positive findings for each of the regulatory requirements, the statements of findings should be similar to the following:

Certificate of Compliance

F9.1 Chapter(s) _____ of the SAR describe(s) SSCs important to safety that are relied on for confinement in sufficient detail to permit evaluation of their effectiveness, in accordance with 10 CFR 72.230(a), 10 CFR 72.230(b), and 10 CFR 72.236.

F9.2 The design of the [DSS designation] adequately protects the SNF cladding against degradation that might otherwise lead to gross ruptures, in accordance with 10 CFR 72.236(g). The chapter of the safety evaluation report (SER) on thermal evaluation discusses the relevant temperature considerations.

F9.3 The design of the [DSS designation] provides redundant sealing of the confinement system closure joints, in accordance with 10 CFR 72.236(e), by _______.

F9.4 The confinement system will be monitored with a _________ monitoring system as discussed above [if applicable] to demonstrate compliance with 10 CFR 72.236(d)(e)(g) and (l). No instrumentation is required to remain operational under accident conditions.

F9.5 The quantity of radioactive nuclides postulated to be released to the environment has been assessed to evaluate compliance with 10 CFR 72.236(d). The SER chapter on radiation protection shows that the dose from these releases will be added to the direct dose to show that the (DSS designation) satisfies the regulatory requirements of 10 CFR 72.104(a) and 10 CFR 72.106(b).

F9.6 The storage container confinement system will be inspected to ascertain that there are no cracks, pinholes, uncontrolled voids, or other defects that could significantly reduce its confinement effectiveness, in accordance with 10 CFR 72.236(j).

F9.7 The storage container confinement system has been evaluated (by appropriate tests or by other means acceptable to the NRC) to demonstrate that it will reasonably maintain confinement of radioactive material under normal, off-normal, and credible accident conditions, in accordance with 10 CFR 72.236(l).

Specific License

F9.8 Chapter(s) _____ of the SAR describe(s) structures, systems, and components (SSCs) important to safety that are relied on for confinement in sufficient detail to permit evaluation of their effectiveness, in accordance with 10 CFR 72.128(a).

F9.9 The quantity of radionuclides postulated to be released to the environment has been assessed as discussed above, in accordance with 10 CFR 72.104(a) and 10 CFR 72.106(b). The SER chapter on radiation protection shows that the dose from these releases will be added to the direct dose to show that the [DSF designation] satisfies the regulatory requirements of 10 CFR 72.104(a) and 10 CFR 72.106(b).

F9.10 If the confinement system is provided by an unsealed system, the following would be applicable: The [DSF designation] includes the following confinement systems that are important to safety and that require monitoring over anticipated ranges for normal and off-normal operations: ________ [identify]. The following monitoring systems must remain operational under accident conditions: ________ [identify]. The SAR acceptably describes instrumentation and control systems that should provide these capabilities, in compliance with 10 CFR 72.122(i) and 10 CFR 72.128(a).

F9.11 The proposed operations of the [DSF designation] provides adequate measures for protecting the SNF cladding against degradation that might otherwise lead to gross ruptures of the material to be stored, in compliance with 10 CFR 72.122(h)(1).

(SL) In the case of the evaluation of releases from confinement for specific licenses, the acceptability of releases can be determined only after reviewing the results of the dose assessment, which is addressed in Chapters 10 and 16 of this SRP.

The reviewer should provide a summary statement similar to the following:

> The staff concludes that the design of the confinement system of the [storage container designation] is in compliance with 10 CFR Part 72 and that the applicable design and acceptance criteria have been satisfied. The evaluation of the confinement system design provides reasonable assurance that the [storage container designation] will allow for the safe storage of SNF. This finding is reached on the basis of a review that considered the regulation itself, appropriate regulatory guides, applicable codes and standards, the applicant’s analysis, and accepted engineering practices.

9.7 <u>References</u>

10 CFR Part 20, “Standards for Protection Against Radiation.”

10 CFR Part 50, “Domestic Licensing of Production and Utilization Facilities.”

10 CFR Part 72, “Licensing Requirements for the Independent Storage of Spent Nuclear Fuel, High-Level Radioactive Waste, and Reactor-Related Greater than Class C Waste.”

American National Standards Institute (ANSI) N14.5, “Radioactive Materials—Leakage Tests on Packages for Shipment, 2014.

American Society of Mechanical Engineers (ASME) Boiler and Pressure (B&PV) Code, 2007—Addenda 2008.
Section III, “Rules for Construction of Nuclear Facility Components.”
Division 1, “Metallic Components”; Subsections NB and NC
Division 3, “Containments for Transportation & Storage of Spent Nuclear Fuel and High Level Radioactive Material & Waste” (no NRC position on this has been established)

ASME NQA-1-2008, “Quality Assurance Requirements for Nuclear Facility Applications,” American Society of Mechanical Engineers, New York, NY.

ASME NQA-1A-2009 Addenda to ASME NQA-1-2008, “Quality Assurance Requirements for Nuclear Facility Applications,” American Society of Mechanical Engineers, New York, NY.

Bordy, J.M. 2015, “Monitoring of eye lens doses in radiation protection,” *Radioprotection* **50**(3), 177-185.

International Commission on Radiological Protection (ICRP) Publication 2, “Report of Committee II on Permissible Dose for Internal Radiation,” Pergamon Press, 1959.

ICRP Publication 26, “Recommendations of the International Commission on Radiological Protection,” *Annals of the ICRP*, Pergamon Press, 1977.

ICRP Publication 103, “The 2007 Recommendations of the International Commission on Radiological Protection,” *Annals of the ICRP*, Elsevier, 2007.

Knoll, R.W. and E.R. Gilbert, “Evaluation of Cover Gas Impurities and Their Effects on the Dry Storage of LWR Spent Fuel,” PNL-6365, DE88 003983, Pacific Northwest National Laboratory, November, 1987.

National Council on Radiation Protection and Measurements Report No. 122, “Use of Personal Monitors to Estimate Effective Dose Equivalent and Effective Dose to Workers for External Exposure to Low-LET Radiation,” 1995.

NRC Information Notice 2013-07, “Premature Degradation of Spent Fuel Storage Cask Structures and Components from Environmental Moisture,” dated April 16, 2013.

NRC Information Notice 2016-04, “ANSI N14.5-2014 Revision and Leakage Rate Testing Considerations,” dated March 28, 2016 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML16063A287).

NUREG/CR-6487, “Containment Analysis for Type B Packages Used to Transport Various Contents,” UCRL-ID-124822, Lawrence Livermore National Laboratory, November 1996.

Regulatory Guide 1.23, “Meteorological Monitoring Programs for Nuclear Power Plants (Safety Guide 23).”

Regulatory Guide 1.109, “Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I.”

Regulatory Guide 1.111, “Methods for Estimating Atmospheric Transport and Dispersion of Gaseous Effluents in Routine Releases from Light-Water-Cooled Reactors.”

Regulatory Guide 1.145, “Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants.”

Regulatory Guide 1.183, “Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors.”

Regulatory Guide 1.195, “Methods and Assumptions for Evaluating Radiological Consequences of Design Basis Accidents at Light-Water Nuclear Power Reactors.”

Sandoval, R.P., R.E. Einziger, H. Jordan, A.P. Malinauskas, and W.J. Mings, “Estimate of CRUD Contribution to Shipping Cask Containment Requirements,” SAND88-1358, TTC-0811, UC-71, Sandia National Laboratories, January 1991.

U.S. Environmental Protection Agency (EPA) Federal Guidance Report No. 11, “Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion,” September 1988.

EPA Federal Guidance Report No. 12, “External Exposure to Radionuclides in Air, Water, and Soil,” September 1993.

10A RADIATION PROTECTION EVALUATION FOR DRY STORAGE FACILITIES (SL)

10A.1 Review Objective

The objective of the U.S. Nuclear Regulatory Commission's (NRC's) radiation protection evaluation is to determine the following:

- The applicant has proposed a functional radiation protection program that will effectively manage, monitor, and control radiation exposures and doses to facility workers and members of the public from a dry storage facility (DSF) that is either an independent spent fuel storage installation (ISFSI) or a monitored retrievable storage installation (MRS) in compliance with NRC regulations and acceptance criteria.

- The proposed DSF radiation protection features meet the NRC design criteria for direct radiation and effluent controls.

- The applicant has proposed engineering features and operating procedures for the DSF that will ensure that occupational exposures will remain as low as reasonably achievable (ALARA).

- Occupational radiation doses will not exceed the limits specified in the NRC's radiation protection standards.

- Radiation doses to the public will meet regulatory standards during both normal conditions and anticipated occurrences and will meet the regulatory dose limits for accident conditions.

- Radiation exposures and radioactive effluent releases will be maintained at levels that meet ALARA objectives and comply with the NRC limits.

For the purposes of this standard review plan (SRP) chapter, radiation protection refers to organizational, design, and operational elements that are relied upon to limit radiation exposures from normal operations, anticipated occurrences (that is, off-normal conditions), and accidents and natural phenomenon events (collectively referred to as accident conditions or design-basis accidents). This includes those design and other elements that may have a different primary function but are nonetheless credited or considered in the applicant's radiation protection evaluation.

10A.2 Applicability

This chapter applies to the review of applications for specific licenses for ISFSIs and MRSs, referred to as DSFs. Thus, the chapter title is denoted with **(SL)**.

10A.3 Areas of Review

The areas of review include means and methods used to protect workers and members of the public, facility design features, dose assessments and dose assessment methods, radiation monitoring instrumentation, sampling and analytical equipment, and operational elements and procedures.

This chapter addresses the following areas of review:

- ALARA objectives
  - policies and programs
  - design considerations
  - operational considerations
- radiation protection design features
  - installation design features
  - access control
  - radiation shielding
  - confinement and ventilation
  - area radiation and effluent monitoring and instrumentation
  - radiological environmental monitoring program
- radiation exposures and dose assessment
  - basis and assumptions of dose assessment
  - onsite dose
  - offsite dose
- health physics program
  - organization and staffing
  - equipment, instrumentation, and facilities
  - policies and procedures

10A.4 Requirements and Acceptance Criteria

This section summarizes those parts of Title 10 of the *Code of Federal Regulations* (10 CFR) Part 72, "Licensing Requirements for the Independent Storage of Spent Nuclear Fuel, High-Level Radioactive Waste and Reactor-Related Greater Than Class C Waste," and 10 CFR Part 20, "Standards for Protection Against Radiation," that are relevant to the review areas this chapter addresses. The reviewer should refer to the exact language in the regulations.

This section describes the acceptance criteria used to guide the review of radiation protection features and programs. The safety analysis report (SAR) should address these acceptance criteria. The acceptance criteria are organized according to the areas of review specified in Section 10A.3 above. The reviewer should consider the applicability and implementation of NRC and industry guidance against that presented in the SAR.

The radiation protection review also requires coordination with other reviews under this SRP related to site characteristics (Chapter 2), principal design criteria (Chapter 3), shielding (Chapter 6), confinement (Chapter 9), operation procedures and systems (Chapter 11), conduct of operations (Chapter 12), waste management (Chapter 13), accident analysis (Chapter 16), and technical specifications (Chapter 17). A complete evaluation of the facility's radiation protection program, as outlined in this chapter, is also dependent on accurate and adequate evaluations of these other aspects of the facility's design and operation.

This guidance recognizes that applicants have various options on how to demonstrate compliance with NRC regulations and NRC guidance (e.g., rely only on NRC guidance or use alternative methods). In general, the acceptance criteria listed in the SAR should adopt, by reference, appropriate NRC guidance or, alternatively, cite relevant and appropriate industry codes and standards. The SAR should identify and justify alternative approaches used to demonstrate compliance with applicable NRC guidance and industry codes and standards. Use of a code or standard in lieu of NRC guidance may require the applicant to discuss the applicability of the code or standard and the basis for its selection and use. Section 10A.5, "Review Procedures," of this SRP provides more specific guidance on the conduct of reviews whenever the SAR cites industry codes and standards.

With respect to the implementation of NRC guidance, the SAR should identify whether the applicant has adopted the NRC guidance in whole or in part. The SAR should identify any differences between this SRP chapter and design features, analytical techniques, exposure and dose assessment codes, and procedural measures proposed for the facility and discuss how the proposed alternatives to this SRP acceptance criteria provide acceptable methods of complying with regulations. In any case, the SAR should provide sufficient information and data for the staff to conduct an independent evaluation in confirming compliance with regulatory requirements and SRP acceptance criteria. The reviewer will confirm that the applicant has adequately addressed these things in the SAR.

If there are multiple versions of a guidance document, such as a regulatory guide or an industry standard, the SAR should describe which version of the guidance document the applicant used, whether it is the most current revision and the basis for using the selected version. In the case of an industry standard, the applicant should consider what, if any, staff position exists with respect to the acceptability of the standard and its different revisions as part of that selection. An applicant may propose to use a particular revision because the proposed DSF is co-located with the applicant's facility licensed under 10 CFR Part 50 "Domestic Licensing of Production and Utilization Facilities," or 10 CFR Part 52, "Licenses, Certifications, and Approvals for Nuclear Power Plants," which used that revision of the guidance as part of its approved licensing basis. As a result, the reviewer will identify the guidance documents the applicant used and assess whether the version of each document the applicant adopted is adequate for demonstrating compliance with NRC requirements.

Table 10A-1 matches the relevant regulatory requirements to the areas of review covered in this chapter. While Table 10A-1 includes specific 10 CFR Part 20 requirements, additional requirements in 10 CFR Part 20 may also apply. Accordingly, the reviewer should consult 10 CFR Part 20 to identify relevant requirements and ensure that the SAR addresses them. Moreover, the applicant and reviewer should be aware of and consider the relevant requirements in 10 CFR Part 19, "Notices, Instructions and Reports to Workers: Inspection and Investigations," including the requirements in 10 CFR 19.11, "Posting of Notices to Workers," 10 CFR 19.12, "Instruction to Workers," and 10 CFR 19.13, Notifications and Reports to Individuals."

The reviewer should also be aware that the Environmental Protection Agency (EPA) has established annual dose limits, which apply to DSFs, in 40 CFR Part 191, "Environmental Radiation Protection Standards for Management and Disposal of Spent Nuclear Fuel, High-Level and Transuranic Radioactive Wastes," particularly in 40 CFR 191.03(a). These limits are the same as the limits in 10 CFR 72.104(a). Thus, compliance with the limits in 10 CFR 72.104(a) ensures compliance with the EPA's limits.

As the guidance in this chapter indicates, evaluations and information concerning 10 CFR Part 20 include demonstrations that the facility design and operations are adequate to ensure compliance with the 10 CFR Part 20 requirements, including dose limits. These evaluations include dose assessments. These dose assessments provide an indication as to whether regulatory dose limits will be met and whether the facility design and operations have adequately considered radiation protection, including the dose limits and as low as is reasonably achievable (ALARA) principles. For locations at or beyond the 10 CFR Part 72 controlled area boundary, the dose estimates should show the limits will not be exceeded. For locations within that controlled area boundary, dose estimates should be fairly low versus limits. For instances where that is not the case, the applicant should provide additional information, such as any administrative controls or measures or physical design features to be used, to show how the applicant as a licensee will ensure limits are met and ALARA principles are followed.

Table 10A-1 Relationship of Regulations and Areas of Review

Areas of Review	10 CFR Part 20 Regulations									
	20.1101	20.1201 (a)	20.1301	20.1302	20.1406 (a)(c)	20.1501	20.1601 (a(b)(c)(d)(e)	20.1602	20.1701	20.1702
ALARA Objectives	(a)(b)(c)(d)		(d)		•	(a)(1)				•
Radiation Protection Design Features	(a)(b)(d)		(e)	(a)	•	(a)(1), (c)(d)	•	•	•	•
Radiation Exposures and Dose Assessment	(a)(b)(d)	•	(a)(b)(d) (e)(f)	(a)(b)						
Health Physics Program	(a)(b)(c)(d)	•	(b)(d)	(a)	•	(a)(1), (c)(d)				•

Areas of Review	10 CFR Part 72 Regulations									
	72.24	72.40	72.44 (c)(d)	72.100	72.104	72.106	72.120 (a)(b)(c)	72.122 (b)(4), (e)(h), (3), (4), (5)	72.126	72.128 (a)(2), (3)
ALARA Objectives	(b)(c)(d)(e)(l)	(a)(1), (5)(13)	•		(b)		•		(a) (d)	
Radiation Protection Design Features	(b)(c)(d)(e)(l)	(a)(1), (2)(5) (13)	•		(a)(b)(c)	(a)(b)(c)	•	•	(a)(b)(c) (d)	•
Radiation Exposures and Dose Assessment	(d)(e)(m)(l)	(a)(1), (2)(5), (13)		•	(a)(c)	(a)(b)		•	(d)	
Health Physics Program	(e)(l)	(a)(1), (5)(13)	•						(a) (d)	

10A.4.1 ALARA Objectives

In evaluating the elements of the ALARA program, the applicant should describe a functional program (including a management policy and organizational structure), proposed engineering design features, activities conducted by individuals having responsibility for radiation protection, and operating procedures that will ensure that occupational exposures and doses to members of the public will be maintained ALARA objectives and meet regulatory standards during normal conditions and anticipated occurrences. The applicant should demonstrate that releases of radioactive materials in liquid and gaseous effluents will be ALARA and describe how the applicant will ensure that releases will be maintained at levels that are ALARA and comply with NRC regulations.

10A.4.1.1 *Policies and Programs*

As a minimum, the policy, program, and activities for ensuring that radiation exposures will be ALARA should include the elements described below in this section. Regulatory Guide (RG) 8.8, "Information Relevant to Ensuring that Occupational Radiation Exposures at Nuclear Power Stations Will Be As Low As Is Reasonably Achievable," and RG 8.10, "Operating Philosophy for Maintaining Occupational Radiation Exposures as Low as Is Reasonably Achievable," provide acceptable guidance on the development and implementation of an ALARA program. Additionally, International Commission of Radiological Protection (ICRP) Publication 27, "Problems Involved in Developing an Index of Harm," issued in 1997, and Publication 55, "Optimization and Decision-Making in Radiological Protection," issued in 1990, and National Council on Radiation Protection and Measurements (NCRP) Report No. 116, "Limitation of Exposure to Ionizing Radiation," issued in 1993, provide useful information for developing an ALARA policy and program.

Policy Statement

The SAR should include a written policy that states management's commitment to maintain exposures to workers and the public at ALARA levels and addresses both facility design and operations. The policy should include the following provisions:

- No practice involving radiation exposure will be undertaken unless evaluation of the practice demonstrates that its use will produce a net benefit to society.
- All exposures will be kept ALARA, with technological, economic, and social factors considered.
- Individual dose limits will be established that are appropriate for practices involving radiation exposure, and exposures to individuals will not exceed these limits.
- Supervisors will integrate appropriate radiation protection controls into all work activities.
- Workers will be appropriately instructed in the objectives and implementation of the ALARA program, with this information included in training modules.
- There will be strict compliance with all regulatory requirements and license conditions regarding procedures, radiation exposures, and releases of radioactive materials.

- A comprehensive program will be maintained, and periodically evaluated, to ensure that both individual and collective doses meet ALARA objectives and do not exceed acceptable levels.

Program Organization

This element should include an organizational structure of the ALARA program, describe the functional responsibilities at all staff levels with respect to its implementation, provide adequate staffing, and include the duties of the personnel directly responsible for the implementation of the ALARA program and policies. The health physics manager and facility health physics staff should have the authority to supervise, monitor, and halt any facility operations and procedures that could result in unnecessary radiation exposures to workers and members of the public or lead to doses in excess of administrative limits and NRC regulations.

Program Elements

The SAR should document how the implementation of the ALARA program will ensure that ALARA objectives are achieved for both onsite and offsite radiation exposures and for monitoring and controlling effluent releases. ALARA program elements should include the use of the following:

- procedures and engineering controls to minimize doses to site personnel, radiation workers, and members of the public allowed access into controlled areas

- tracking of individual doses to identify trends and causes and use of such data in developing alternative procedures that would yield lower doses

- periodic training and exercises for management, radiation workers, health physics staff, and other site workers in radiation protection, ALARA, operating procedures, and emergency response, and then periodic evaluation of the effectiveness of such training and exercises

- periodic reviews and evaluations of the other program elements to ensure their continued effectiveness, making improvements where beneficial, such as revising training programs, drills, and exercises to keep up to date with current radiation protection practices, operations practices, and emergency response plans and procedures

- procedures and controls to monitor, process, and treat radioactive effluents before release into the environment to minimize discharges of radioactive materials and minimize doses to members of the public

- administrative controls and radiation monitoring equipment to prevent unmonitored and uncontrolled releases of radioactive materials

10A.4.1.2 *Design Considerations*

The applicant’s discussion of the facility’s design, including design of facility features and structures, systems, and components (SSCs), and the facility’s layout, including overall layout and layout within facility structures, should demonstrate consideration of ALARA principles and operational knowledge. The design criteria for the facility’s features and SSCs, described in the

SAR's principle design criteria chapter, should include ALARA criteria, and the SAR should identify choices between otherwise comparable alternatives affected by ALARA considerations and the basis for the selected alternative(s). Applicants should use RG 8.8 for ALARA design guidance, although they may use specific alternative approaches if clearly indicated in the SAR. Examples of ALARA design considerations include the following:

- engineered design features that minimize radiation levels and the total amount of time that maintenance, health physics, or inspection personnel must stay in restricted areas while performing their duties

- engineered design features that minimize the need for maintenance

- provisions for the use of remotely operated or robotic equipment, such as automated welders, wrenches, and remote radiation monitors

- use of closed-circuit television to monitor for possible blockage of air cooling passages, to perform inspections and other activities

- provisions for remote placement and use of temporary shielding

- incorporation of materials, design features, and operational practices that minimize the potential for accumulation of radioactive materials or surface contamination, and facilitate decontamination and decommissioning of facilities and equipment

- incorporation of design experience from other ISFSIs, MRSs, or waste management facilities using ALARA design alternatives that are similar to or are improvements of those used at these other facilities

- use of relevant operations experience from other ISFSIs, MRSs, or waste management facilities

- placement of occupiable areas (e.g., office, security stations, access and egress control points, or health physics and laboratory facilities) away from sources of radiation and radioactivity

- ALARA provisions built into health physics training facilities and equipment

10A.4.1.3 *Operational Considerations*

Operational procedures, methods of operation, and methods to develop detailed plans and procedures should incorporate ALARA principles and objectives to ensure personnel exposures and contamination levels are ALARA. The SAR description of these methods and procedures should include the criteria or conditions under which various procedures or techniques are implemented to ensure personnel exposures and residual contamination levels for all facility SSCs that handle radioactive materials are ALARA. The associated operational requirements should be reflected in facility design, as described in Sections 6.4.1 and 6.5.1 of this SRP as well as this chapter. Detailed plans and procedures should be developed in accordance with RG 1.33, "Quality Assurance Program Requirements (Operation)," RG 8.8, and RG 8.10, and should consider the following to the extent practical:

- tradeoffs between requirements for increased monitoring or more frequent maintenance activities (and the resulting increases in radiation exposures) and potential hazards (e.g., premature failures or reduced effectiveness of SSCs) associated with reduced frequency of these activities

- performance of storage container (e.g., cask) preparation efforts (for loading) away from the spent nuclear fuel (SNF) pool or dry transfer facility

- sequencing the placement of SNF, reactor-related greater-than-Class-C waste (GTCC), or high-level radioactive waste (HLW), as appropriate, in a manner that maximizes the shielding effectiveness of storage containers and structures

- conducting of dry runs to develop proficiency in procedures involving radiation exposures; determination of exposures likely to be associated with specific procedures; identification of conditions likely to be associated with specific operational evolutions leading to potentially higher exposures; and consideration, development, and implementation of more efficient alternative procedures in order to control and minimize exposures and doses

- consideration and inclusion of tested and proven contingency plans and procedures in responding to potential anticipated occurrences

- consideration and incorporation of ALARA operational alternatives based on related industry experience at other ISFSIs, MRSs, or similar types of waste management facilities

- research, evaluation, and development of improved operational procedures, types of tools and instruments, and use of personal protective equipment to minimize radiation exposures, releases of radioactive materials, and duration of exposures and reduce risks associated with exposures

10A.4.2 Radiation Protection Design Features

This element addresses the adequacy of the incorporation of radiation protection considerations into the facility design, including meeting regulatory requirements and ALARA objectives. For this element, the SAR should provide information on facility design features, access control, provisions for and effective use of shielding, confinement and ventilation, and means and methods in monitoring external radiation exposure rates and airborne radioactivity concentrations. RG 8.8 includes guidance that, where applicable, may be useful for ensuring adequate incorporation of radiation protection considerations into the facility design.

The SAR descriptions should include facility features and SSCs used for facility operations, including package receipt; package decontamination and unloading; package loading and preparation; waste (SNF, reactor-related GTCC waste, HLW) transfer between package and storage container; storage container preparation, loading, movement, and use; storage container array(s); and site-generated waste treatment packaging, storage, and shipment. The SAR should also describe the provisions made for personnel protective measures, particularly for areas where radioactive materials may become airborne. This information may be referenced from other sections of the SAR as appropriate. The SAR should include scaled layout and arrangement drawings for the facility. These drawings should include locations where SNF, reactor-related GTCC waste, HLW waste, and site-generated wastes will be stored. The SAR should also include information on definition of work areas, designation of radiologically controlled areas and their boundaries (e.g., radiation areas, restricted areas, controlled area), shield wall thicknesses, individual and equipment decontamination areas, contamination control areas and types of controls, personnel and vehicular traffic patterns, health physics facility locations, area radiation monitoring and airborne radioactivity monitoring locations, locations of onsite analytical laboratories (for chemical and radioactive sample analyses) and counting room facilities, and other pertinent facility features and SSCs relevant for radiation protection.

10A.4.2.1 *Installation Design Features*

Installation design features for radiation protection can minimize either offsite or onsite exposures. Features that specifically minimize offsite exposures include the following:

- Siting Considerations—The facility is located away from population centers to the extent feasible, consistent with other factors.

- Controlled Area or Perimeter Distance—The DSF controlled area is located to maintain sufficient distances to the perimeter of the site and locations of public occupancy.

- Transfer Route—Transfer routes for DSF containers are located to maintain sufficient distances from the site perimeter.

- Effluent Discharges and Impacts—Natural and manmade contours, existing or planned rerouting of natural surface water, and points at which surface water exits the site relative to residences and public use areas are considered and incorporated. Cutoffs, drains, well points, or other means are used to control surface water flow into uncontrolled areas.

- Engineered Features—Berms, shield walls, or other engineered features are used as needed to reduce direct radiation exposures and levels beyond the DSF storage area(s).

Features that minimize onsite exposures include the following:

- Transfer Route—Transfer routes for DSF containers to or from the storage area and the handling areas (intermodal transfer points, or wet or dry transfer facility) are located to minimize the route between the handling and storage facilities, minimize other traffic on the route, remain within controlled areas, and maintain appropriate distances between radioactive materials and other site functions and work stations.

- Multiple Restricted Areas—The controlled area contains multiple restricted areas to limit access to areas with elevated radiation levels that would pose unacceptable risks or exposures to workers within those areas.

- Controlled Area and Perimeter Distance—Radioactive material-handling and storage functions are separated from other functions on the site. Distances are maximized, to the extent practical, between radioactive material and both the boundary of the controlled area and the adjacent onsite work stations outside the restricted area.

10A.4.2.2 *Access Control*

Access to controlled and restricted areas is controlled for the purposes of radiation protection as well as safeguards and security. This section addresses the control of access for purposes of limiting exposure to external radiation and radiological contamination hazards.

In consideration of the provisions of 10 CFR 73.21(b) on information to be protected, the description of the DSF design should include the following access control elements:

- site layout to scale showing the DSF controlled area and its boundary (given 10 CFR 72.106, “Controlled area of an ISFSI or MRS,” criteria) and any traversing right(s) of way

- description of the barrier(s) used to preclude ready access to the controlled area

- location and summary description of individual and vehicular access gates and security overlook stations

The SAR should identify the criteria used to designate restricted areas (or zones within restricted areas). It should describe all protective features designed to limit access to restricted areas, including physical barriers, locked entryways, and audible or visible alarm signals. The SAR should also describe continuous direct or electronic surveillance used to prevent unauthorized entry.

Restricted areas may require further designation as high or very high radiation areas (per the definitions in 10 CFR 20.1003, “Definitions”) and be controlled according to 10 CFR 20.1601, “Control of access to high radiation areas,” and 10 CFR 20.1602, “Control of access to very high radiation areas,” respectively, and the requisite postings in accordance with requirements in 10 CFR Part 20, Subpart J, “Precautionary Procedures.” RG 8.38, “Control of Access to High and Very High Radiation Areas of Nuclear Power Plants,” provides guidance on access control features applicable to these areas.

Restricted areas may be further divided to identify areas where the potential for contamination exists. The SAR should identify criteria used to designate contamination control areas (including airborne radioactivity areas). Such criteria, and facility features and operational considerations used to meet them, should be developed, designed, and implemented in compliance with the requirements in 10 CFR 20.1406 “Minimization of contamination.” Access control features applicable to contamination control areas may include the following:

- incorporation of access control features and equipment into the designs of the facility’s buildings or provisions to use temporary or mobile-type access control features and equipment immediately adjacent to the confinement barrier of the potentially contaminated area

- gender-designated change rooms, including lavatories and showers; provisions for personal protective equipment; stations for detecting and monitoring hands, feet, and whole body for contamination; and locations of designated stepoff pads or threshold stations used for the removal of personal protective equipment upon leaving controlled areas

- shower and lavatory water collection and storage, and provisions for routing of potentially contaminated water to treatment, storage, and monitoring systems

Useful information to consider for assessing compliance with 10 CFR 20.1406 in minimizing contamination appears in RG 4.21, “Minimization of Contamination and Radioactive Waste Generation: Life-Cycle Planning,” and NUREG-0800, “Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition,” Section 12.3–12.4, “Radiation Protection Design Features.”

The SAR should document that appropriate measures are provided for the collection of possibly contaminated wash water and that leakage of possibly contaminated liquid onto or into the ground is precluded. The SAR should explain in detail the systems or design features (including their functions) included in the facility design to fulfill these measures, with drawings showing the locations of these systems or features in the design. Wash water may include liquids temporarily stored pending sampling and sample analysis before being released to the sanitary sewer (in accordance with 10 CFR 20.2003, “Disposal by release into sanitary sewerage”); collected, treated, monitored, and held as radioactive waste in designated tanks; or treated, monitored, and released in surface bodies under the provisions of 10 CFR 20.1301, “Dose limits for individual members of the public,” 10 CFR 20.1302, “Compliance with dose limits for individual members of the public,” Table 2, “Effluent Concentration,” Column 2, “Water,” of Appendix B, “Annual Limits on Intake (ALIs) and Derived Air Concentrations (DACs) of Radionuclides for Occupational Exposure; Effluent Concentrations; Concentrations for Release to Sewerage,” to 10 CFR Part 20, and in Footnote 4 of Appendix B to 10 CFR Part 20 in applying the sum of the ratios for radionuclide mixtures.

10A.4.2.3 *Radiation Shielding*

The DSF design should incorporate, and the SAR should describe, provisions for effective shielding as an integral part of the ALARA and radiation protection programs to protect the public and workers against direct radiation. The SAR descriptions should include special protective features that use shielding, geometric arrangement (including separation), or remote handling to ensure exposures will meet ALARA objectives. The SAR should describe the materials of construction and penetrations of facility SSCs and features relied upon for shielding. The SAR should include descriptions of the use of portable shielding, berms, or special buildings at the site that are used for shielding, if applicable.

SRP Chapter 6, “Shielding Evaluation,” provides guidance for conducting detailed engineering evaluations aimed at determining the performance and effectiveness of the proposed shield design. However, Section 10A.4.1 above provides criteria for determining whether the proposed shielding and installation designs satisfy dose and ALARA requirements. The radiation protection review uses dose rate estimates from the shielding review in combination with estimates of radionuclide release rates or doses from effluents (from Chapter 9, “Confinement Evaluation,” and Chapter 13, “Waste Management Evaluation” of this SRP) to ensure that combined doses (i.e., from all sources and exposure pathways) meet the acceptance criteria, as described in Sections 10A.4.3.2 and 10A.4.3.3 below.

10A.4.2.4 *Confinement and Ventilation*

Confinement refers to the ability of the DSF to prevent the release of radioactive materials from controlled areas (e.g., fuel handling, loading, and unloading areas) and SSCs (e.g., containers), in which these materials are contained, into other areas of the facility and the surrounding environment. Confinement barrier systems may be sealed, as in the case of the facility's storage containers, or vented with off-gas treatment systems, as in the case of the facility's waste management systems. For the latter, intake and exhaust filters and dampers, as well as portions of ducts and stacks of ventilation systems, function as elements of the confinement system. Together, confinement and ventilation function to protect personnel and the public against radiation exposures associated with releases of radioactive materials under normal conditions, anticipated occurrences, and accidents.

Chapters 9 and 13 of this SRP address the evaluation of the confinement and ventilation systems' performance and effectiveness and resulting radionuclide release rates and doses from effluents. These considerations are included in the evaluation of compliance with regulatory dose requirements, including maintaining exposures and releases ALARA.

Area Monitoring and Effluent Monitoring Instrumentation

The SAR should describe the locations, types, capabilities, and operational parameters of fixed-area radiation monitors and equipment, such as continuous airborne monitoring instrumentation, used to control and monitor releases of radioactive materials in liquid and gaseous effluents. The SAR descriptions should include appropriate details in the drawings and specifications defining the DSF design. The operational parameter descriptions should include the range, sensitivity, reliability, accuracy, performance testing, energy dependence, calibration methods and frequency, alarms and alarm setpoints (including criteria and methods for determining those setpoints), limits for action, readouts, release paths to be monitored, sampling frequency, and locations for sampling line pumps and obtaining samples from effluent monitors. The SAR should describe the operational personnel's intended responses to alarms and emergency conditions.

For a DSF, the NRC accepts, to the extent applicable, the criteria and guidance for such equipment and monitoring that are described in the following documents:

- American National Standards Institute (ANSI)/Health Physics Society (HPS) N13.1, "Sampling and Monitoring Releases of Airborne Radioactive Substances from the Stacks and Ducts of Nuclear Facilities," as it relates to principles for obtaining valid samples of airborne radioactive materials and acceptable methods and materials for gas and particle sampling

- ANSI/America Nuclear Society (ANS)/HPS Standards Committee 6.8.1-1981, "Location and Design Criteria for Area Radiation Monitoring Systems for Light Water Nuclear Reactors," as it relates to the criteria for locating fixed continuous area gamma radiation monitors and for design features and ranges of measurement

- NUREG-0800, Section 11.5, "Process and Effluent Radiological Monitoring Instrumentation and Sampling Systems"

- RG 1.13, "Spent Fuel Storage Facility Design Basis"

- RG 1.21, "Measuring, Evaluating, and Reporting Radioactive Material in Liquid and Gaseous Effluents and Solid Waste"

- RG 4.1, "Radiological Environmental Monitoring for Nuclear Power Plants"

- RG 8.25, "Air Sampling in the Workplace," as it relates to use of fixed and portable air samplers in the workplace

The following documents contain criteria and guidance that also may be useful in relation to monitoring and monitoring equipment for a DSF:

- NCRP Report No. 57, "Instrumentation and Monitoring Methods for Radiation Protection"

- NCRP Report No. 112, "Calibration of Survey Instruments Used in Radiation Protection for the Assessment of Ionizing Radiation Fields and Radioactive Surface Contamination"

- NCRP Report No. 169, "Design of Effective Radiological Effluent Monitoring and Environmental Surveillance Programs"

- NUREG-0800, Section 11.2, "Liquid Waste Management System"

- NUREG-0800, Section 11.3, "Gaseous Waste Management System"

- RG 4.15, "Quality Assurance for Radiological Monitoring Programs (Inception through Normal Operations to License Termination)—Effluent Streams and the Environment"

Classification of auxiliary power sources for monitoring instrumentation as "emergency" (for SSCs important to safety) or "standby" (for SSCs not important to safety) should correspond to the classification of the instrumentation itself. The following describes discriminators in classifying instrumentation and auxiliary power sources as important to safety:

- If data provided by the monitoring system can have an immediate and determining effect on personnel actions and operations to maintain compliance with established safety criteria and limits, including prevention of unacceptable doses to workers, then monitoring instrumentation should be classified as emergency.

- If any of the following is true, then the instrumentation and its auxiliary power source may not be important to safety:

  – Instrumentation data are not provided in real time to a central control room or, if provided, do not trigger an alarm that results in actions that should preclude or mitigate unacceptable consequences.

  – Instrumentation does not trigger an alarm necessary to avoid unacceptable worker exposures at its location when a setpoint threshold is reached.

  – Data are collected only periodically.

  – No normal, off-normal, or accident events or conditions can result in changes in the monitored phenomena that can jeopardize satisfaction of safety criteria and limits.

10A.4.2.5 *Radiological Environmental Monitoring Program*

The SAR should describe the radiological environmental monitoring program for the facility. A licensee uses the program to verify compliance with the 10 CFR 72.104(a) dose limits during DSF operations. The program employs a combination of methods, as appropriate, including direct radiation measurements (such as thermoluminescent or optically stimulated luminescent dosimeters) and sampling and analyses of gaseous and liquid effluents and environmental samples.

The SAR description of the radiological environmental monitoring program should include information regarding the exposure pathways that will be monitored. The monitored pathways should include the pathways that lead to the highest potential external and internal radiation exposures of individuals that result from DSF operations. The programs should be designed to provide data on exposures and radionuclide concentration levels for those exposure pathways. The SAR should identify the sample types (e.g., water, soil, vegetation), number of samples, sample locations, collection frequency, and sample analysis to be performed along with its frequency. The SAR should include a map of suitable scale that identifies the sampling locations to show distance and direction of monitoring stations, with release points and relevant boundaries (e.g., controlled area boundary, site boundary) also indicated on the map. The SAR description should include the program for continuing meteorological data collection and evaluation to supplement the estimates of individuals' external and internal radiation exposures developed in accordance with Section 10A.4.3.3 below. Additionally, the SAR description of the radiological environmental monitoring program should also include the approach for determining background levels and the contribution of the facility's incremental releases to background levels. The SAR should include the results of the background level determination.

10A.4.3 Radiation Exposures and Dose Assessment

The SAR should provide dose estimates and describe the methods and means, including all assumptions and bases, used to derive dose estimates for occupational workers, members of the public located at or beyond the controlled area boundary, members of the public using public access facilities (e.g., highways, railways, waterways) that traverse the controlled area, and nonradiation worker facility personnel and others that may access the site (e.g., carriers involved in shipments of materials to/from the facility, construction workers brought onsite for building additional storage pads). The dose estimates should include individual and collective doses from direct radiation exposures and effluent releases.

It should be noted that there is considerable overlap in the information presented in the SAR between this section and the section describing radiation protection design features. The overlap offers a dual purpose and benefits. From the shielding evaluation (SRP Chapter 6), estimated dose rates for direct radiation should be provided for representative points within the controlled area (as defined in 10 CFR Part 72) and any restricted areas (as defined in 10 CFR Part 20), as well as on and beyond the boundary of the controlled area. Additionally, the confinement evaluation (SRP Chapter 9) and site-generated waste management evaluation (SRP Chapter 13) should have produced estimates of radioactive materials (radionuclide concentrations) present in effluents and dose (rate) estimates from effluents. Accordingly, the radiation protection evaluation includes a dose assessment that incorporates results of each of these evaluations, as applicable. The major elements of the dose assessment and the applicable acceptance criteria are described below.

10A.4.3.1 *Basis and Assumptions of Dose Assessment*

The applicant should provide sufficient information describing and justifying the bases, models, and assumptions applied in estimating all doses. This description should identify all exposure pathways, locations and occupancy (or residence) times with their bases, essential parameters and their selected values, sources of the data for these values (site specific, default from the NRC, or industry guidance), computer codes and software version, and dose results. For any codes used, the SAR should provide information to demonstrate the validation of the codes in a manner similar to what is described in Sections 6.4.4.1 and 6.5.4.1 of this SRP. The discussion of dose results should address the degree of conservatism applied in all assumptions and parameters, whether any part of the dose assessment was modified in light of the results of separate sensitivity analyses, and conclusions in demonstrating compliance with the NRC regulations and acceptance criteria. If results for individuals located at or beyond the 10 CFR Part 72 controlled area boundary are marginally close but still in compliance with the NRC dose criteria in 10 CFR Part 20 and 10 CFR Part 72, the SAR should describe the direction and magnitude of underlying uncertainties, given all assumptions, in providing reasonable assurance that such doses represent conservative bounding estimates. If results approach or exceed the appropriate limits for individuals onsite (within the 10 CFR Part 72 controlled area boundary), the SAR should describe the controls, conditions or other means by which the licensee will ensure individual doses will not exceed the appropriate limits. Chapter 9 of this SRP contains additional guidance regarding the information the SAR should contain related to analyses of doses from effluents or releases from the storage containers. That guidance may also be useful for identifying the information that the SAR should contain related to analyses of doses from effluents or releases from the site-generated waste management systems (discussed in Chapter 13 of this SRP).

10A.4.3.2 *Onsite Dose*

The SAR should provide the objectives and criteria for design dose rates for the various areas of the facility. Individual and collective doses should be calculated for all onsite areas at which workers will be exposed to elevated radiation levels (e.g., greater than 2 millirem per hour (mrem/hr) (0.02 millisieverts per hour (mSv/hr)) or airborne radioactivity concentrations during normal operation and anticipated occurrences. The dose estimates should be based on direct exposure and inhalation of airborne radioactivity and should be derived for workers performing specific DSF functions, including routine, contingency, maintenance, or repair procedures or other activities that can occur in areas with elevated dose rates. Individual and collective doses should also be determined for onsite functions outside the DSF restricted areas associated with package receipt and with package preparation and transfer to conveyance for shipment of the radioactive materials to be stored at the facility.

The SAR should include estimates of occupancy times for personnel involved in these functions, including the maximum expected total hours per year for any individual and total person-hours per year for all personnel. The annual collective doses associated with each major function and each radiation area should be estimated. Individual doses to workers should be well below the dose limits specified in 10 CFR 20.1201, “Occupational dose limits for adults.”

Collective doses should be consistent with the objectives contained in the applicant’s ALARA program. The information provided by the applicant should allow for the determination of compliance with these criteria. In general, the following information will allow for such a determination:

- The SAR should identify and list collective and individual doses associated with all operations involved with placing one full storage container in the storage position according to the associated function.

- The SAR should provide estimates of the annual collective and individual doses by multiplying the single-storage container dose by the maximum annual placement rate of containers into storage. This estimation assumes that the same personnel will be involved in the same operations for each container. If the doses exceed those allowed by 10 CFR 20.1201(a), the planned conduct of operations (SRP Chapter 12) should include conditions (e.g., staffing plan, monitoring) that will ensure that 10 CFR 20.1201(a) dose limits are not exceeded.

- The SAR should provide estimates of annual doses for operation of the DSF for material in storage and material in wet holding or wet storage for comparison with maximum allowable doses given in 10 CFR 20.1201.

- The SAR should include a discussion of sensitivity of dose results to assumptions and uncertainties, including the use of conservative parameters.

Depending on the applicant's proposed conduct of operations (see SRP Chapter 12), not all facility personnel may necessarily be radiation workers. This may include administrative staff among others. In addition, carrier personnel involved in the shipments of materials to or from the site may have access to the controlled area but not be radiation workers. For these individuals, the 10 CFR Part 20 dose limits for members of the public apply, and the onsite dose evaluations should address the ability to meet those limits, which are given in 10 CFR 20.1301. The applicant should describe and justify the bases and any assumptions used in the evaluation. The SAR should include a description of any administrative controls the applicant will use to ensure that the bases of the evaluation and assumptions remain valid during facility operation.

10A.4.3.3 *Offsite Dose*

Dose rates and doses should be controlled so that doses in any unrestricted areas, which include areas beyond the controlled area boundary, do not exceed the 10 CFR 20.1301(a)(2) limit of 2 mrem (0.02 mSv) in any single hour from external sources from all licensed activities at the site and the 10 CFR 20.1101(d) constraint on airborne radioactive material emissions of 10 mrem (0.1 mSv) total effective dose equivalent (TEDE) per year.

For normal operations and anticipated occurrences, the estimated dose to any real individual located at or beyond the controlled area boundary may not exceed the limits of 10 CFR 72.104(a). Note that the 10 CFR 72.104(a) dose limits are expressed as annual dose equivalent to the whole body, the thyroid, and any other critical organ.

Calculated doses must include both direct radiation and associated exposures to airborne radioactivity, such as from planned discharges of radioactive materials, if applicable (see 10 CFR 72.104(a)). The doses must also include the radiation (direct and effluent) from other activities (e.g., reactor, enrichment facility radioactive waste storage facility) in the region (see 10 CFR 72.104(a)). Assessments of doses should consider all sources of radiation and radioactivity (including effluents) and exposure pathways (external and internal) as potential contributors to doses to members of the public from all onsite facilities. Since anticipated occurrences are expected to occur at a frequency of once per year, the sum of the doses from normal operations and the bounding anticipated occurrence (that is, off-normal condition) must comply with the limits in 10 CFR 72.104(a).

Applicants may demonstrate compliance with 10 CFR 72.104(a) in one of two ways.

1. Show that an individual's dose at the controlled area boundary with full-time occupancy will not exceed the regulatory dose limits.

– OR –

2. Identify individuals within the geographical location of the DSF and estimate their maximum radiological exposures. Use this information to identify a maximally exposed real individual. Calculations may involve site-specific information, such as the number of storage containers; the container array configuration(s); the characteristics of the actual SNF, HLW, or reactor-related GTCC waste (or any combination of the three) to be stored at the facility; the site characteristics; and the surrounding topography features. Alternatively, the calculations may involve bounding parameters for each of these items. This approach should consider the current as well as potential changes in population and water and land use based upon projections of these aspects described as part of the site evaluation (SRP Chapter 2). Calculations may estimate the amount of time that a real individual spends near the facility, the distance the real individual is from the facility, and other factors that may mitigate radiological exposure to the real individual.

If the second approach is taken, then the applicant should establish measures in the radiological protection program, environmental monitoring program, and operating procedures, as applicable, to identify and periodically reevaluate potential increases in exposure to the real individual during the term of the license.

For exposures occurring under accident conditions, including design-basis accidents and natural phenomenon events, the estimated doses to any individual located on or beyond the nearest boundary of the controlled area may not exceed the limits specified in 10 CFR 72.106(b). The estimated doses should include the contributions from direct radiation and any releases that occur as a result of the accident.

If radioactive effluents from the DSF are anticipated, the applicant should provide the estimated annual collective dose (in person-rem or person-Sievert) related to the DSF. The SAR should present details on estimated radioactive effluents and models and equations used to determine doses. The applicant should also provide estimated collective doses resulting from releases under accident conditions. Doses should be based on all important exposure pathways (e.g., airborne releases) and modes of exposure (e.g., external exposure, inhalation) and should be specified as whole-body, or effective dose equivalent. In addition, the SAR should identify the organs, including critical organs, receiving the highest doses and provide their doses.

The applicant should apply a methodology that the NRC accepts, as described in applicable NRC guidance. If an application uses alternative methods and assumptions in deriving doses, the SAR should contain sufficient information for the staff to independently confirm the results presented in the SAR. This information is used to evaluate the facility's impacts in accordance with 10 CFR 72.100(a). The SAR should include appropriate justification for why these estimated collective doses are ALARA. For these analyses, the applicant should consider current, and potential changes in, population and land and water use.

The following considerations also apply to the offsite dose assessments:

- The applicant should calculate dose rates from direct radiation on the basis of the maximum quantity or inventory of radioactive materials permitted by the DSF license.

- The dose assessment should assume that radioactive materials are distributed in such a manner as to produce the highest perimeter dose rate, unless such arrangements are specifically precluded by operational considerations, license conditions, or technical specifications.

- RG 4.20, "Constraint on Releases of Airborne Radioactive Materials to the Environment for Licensees other than Power Reactors," provides guidance on methods the NRC considers acceptable for meeting the airborne emissions constraint in 10 CFR 20.1101(d).

- (Proposed) license conditions or technical specifications regarding facility design or operations that affect offsite doses (e.g., stored materials quantities and specifications, dose rate limits, contamination limits).

Additional engineering features, such as berms or shield walls, may be used to mitigate doses to real individuals near the site. However, if these features are relied upon to comply with the dose limits in 10 CFR 72.104(a) or in 10 CFR 72.106(b) for any individual, the applicant should adequately describe and analyze such features in the SAR and classify them as important to safety (at the appropriate category).

10A.4.4 Health Physics Program

The SAR should include a description of the health physics program for the proposed facility. The program's scope should be sufficiently broad to support all expected operational events, including normal operations, anticipated occurrences, and accident conditions, and demonstrate compliance with the applicable requirements of 10 CFR Parts 19, 20, and 72. Table 10A-2 lists major program elements, along with the parameters and applicable regulatory criteria and guidance documents, for each element that the applicant should describe in the SAR.

This section addresses the health physics program's organization, staffing, lines of authority, facilities (including equipment and instrumentation), and administrative policies and procedures used in implementing radiation protection functions.

The management and functions of the health physics program should be commensurate with expected radiological conditions and ranges of radiation exposure rates and doses. The DSF should have the facilities, equipment, and instrumentation necessary to ensure that the health physics program can be properly carried out and the health physics staff can adequately discharge its functions and responsibilities. In part, the evaluations described in this SRP chapter and results of evaluations described in other SRP chapters (e.g., Chapters 6, 9, 13, and 16) provide supporting information in bracketing the range of expected radiological conditions.

Table 10A-2 Program Elements of the Health Physics Program

Item	Description	Criteria
Radiation surveys	Method, frequency, and plans for conducting radiation surveys, records of surveys	10 CFR 20.1501(a) and 10 CFR 20.2103
ALARA plans	Plans developed to ensure occupational exposures will be ALARA	10 CFR 20.1101(b) RG 8.8 and RG 8.10
Access control and postings	Physical and administrative functions and measures (e.g., personnel monitoring) for controlling access to and limiting stay times in restricted and controlled areas	10 CFR 20.1601, 10 CFR 20.1602, 10 CFR 20.1702, and 10 CFR 20.1902; 10 CFR 72.126(a)(3) and 72.126(b) RG 8.38
External exposure monitoring	Monitoring criteria, types of dosimeters, collection frequency, processing, review of results (including how results are used for operational planning)	10 CFR 20.1502 RG 8.2, “Administrative Practices in Radiation Surveys and Monitoring” RG 8.4, “Personnel Monitoring Device—Direct-Reading Pocket Dosimeters” RG 8.28, “Audible-Alarm Dosimeters” RG 8.34, “Monitoring Criteria and Methods to Calculate Occupational Radiation Doses”
Internal exposure monitoring	Types of monitoring (e.g., whole-body counts, lung counts, urinalysis), monitoring criteria, procedures for estimating dose from bioassay results, and review of results	10 CFR 20.1204, 10 CFR 20.1502, 10 CFR 20.1703(c)(2), (c)(4)(i), and 20.1703(i) RG 8.9, “Acceptable Concepts, Models, Equations, and Assumptions for a Bioassay Program” RG 8.26, “Applications of Bioassay for Fission and Activation Products” RG 8.34
Air sampling and analysis	Methods and procedures for air sampling and analysis, evaluation and control of airborne radioactivity, requirements and procedures for special air sampling	10 CFR 20.1204(a)(1), 10 CFR 20.1501(a)(2)(ii), 10 CFR 20.1502, 10 CFR 20.1701, 10 CFR 20.1702(a), and 10 CFR 20.1703(c)(1), (c)(4)(i) RG 8.25
Effluent releases and monitoring	Means, methods, procedures and equipment to sample, analyze, monitor, and control airborne and liquid effluents from facility systems and buildings	10 CFR Part 72.126(c) and (d)
Minimization of contamination and waste generation	Methods and procedures to monitor, control, and reduce contamination levels in facilities (including personnel, equipment, and surfaces) and waste generation	10 CFR 20.1406, 10 CFR 72.24(f), and 10 CFR 72.126(a)(1),(2),(4) RG 4.21
Respiratory protection program	Policy statement on respirator usage; respirator certification, fit-testing, and usage; medical surveillance of respirator users	10 CFR 20.1702 and 10 CFR 20.1703 RG 8.15, “Acceptable Programs for Respiratory Protection”

Item	Description	Criteria
Radiation protection	Requirements for initial and refresher training, contents (topics), health physics-related qualification of workers	RG 1.8, “Qualification and Training of Personnel for Nuclear Power Plants” RG 8.2 RG 8.27, “Radiation Protection Training for Personnel at Light-Water-Cooled Nuclear Power Plants” RG 8.29, “Instruction Concerning Risks from Occupational Radiation Exposure”
Pregnant worker protection	Provisions to inform female workers of fetal protection requirements, to monitor fetal dose, and to provide alternatives to minimize fetal dose	10 CFR 20.1208 RG 8.13, “Instruction Concerning Prenatal Radiation Exposure”
Instrument QA	Requirements and procedures for calibration, maintenance, and care of radiation detection, monitoring, and dosimetry instruments and records	10 CFR 20.1501(b), 10 CFR 20.1501(c),(d) and 10 CFR 20.2103
Recordkeeping and reports	Preparing of reports and records for health physics program contents and audits, surveys, calibrations, personnel monitoring results	10 CFR Part 20, Subpart L, “Records,” and Subpart M, “Reports”

10A.4.4.1 *Organization and Staffing*

RG 8.8, RG 8.10, and NUREG-0800, Section 12.5, “Operational Radiation Protection Program,” include guidance applicable to the organization and planning for health physics (radiation protection) activities at a DSF. The DSF management organization should identify an individual with clearly designated responsibilities for health physics. To avoid the potential for conflict of interest, this individual’s reporting line should not include facility managers responsible for the operation of the DSF. The health physics manager and facility health physics staff should have the authority to supervise, monitor, and halt facility operations and procedures that could result in unnecessary radiation exposures to workers and members of the public or lead to doses in excess of administrative limits and NRC regulations. The health physics manager position (or its equivalent title) should be maintained for the operational life of the facility, including all decontamination and decommissioning operations. The health physics organization should include adequate staffing with appropriate experience, training, and qualifications. RG 8.2 and RG 8.8 describe acceptable programs and methods for complying with NRC requirements. RG 1.8 provides guidance for reactors that may also be useful for DSFs.

10A.4.4.2 *Equipment, Instrumentation, and Facilities*

The SAR should describe health physics program equipment, instrumentation, and facilities. The need for specific health physics equipment and facilities depends on the nature of the installation and its operations, such as whether specific laboratory functions are performed at offsite facilities.

In all cases, the program should include adequate means to properly monitor all expected operational evolutions and associated radiological conditions. The equipment should include portable and laboratory equipment, such as the following:

- personal radiation monitoring devices for external dosimetry, including provisions for dosimeter processing by a dosimetry service accredited by the National Voluntary Laboratory Accreditation Program

- an appropriate number of handheld and portable radiation survey meters and detectors for performing radiation and contamination surveys for each type of survey to be performed (e.g., Geiger-Mueller survey instruments for contamination surveys and personnel “frisking,” ionization chambers for exposure rate surveys, neutron detectors for conducting neutron flux or dose rate surveys)

- methods and equipment, including radioactive sources and standards (National Institute of Standards and Technology-traceable primary and secondary), used to check the operation and to calibrate fixed and portable radiation monitoring survey equipment and laboratory radioanalytical equipment

- methods and equipment used to calibrate flow rates of air sampling equipment, including ambient air portable and fixed sampling stations and airborne effluent release points (e.g., facility stacks or building vents)

- portable air sampling equipment and airborne radioactivity monitors

- facilities for internal radiation monitoring, including whole-body counters, thyroid counters, bioassay sample collection and analytical equipment

- personal protective equipment (including anticontamination clothing and respirators certified by the National Institute for Occupational Safety and Health, Mine Safety and Health Administration)

- designated areas and facilities to inspect, maintain, clean, and store equipment and the means to test personnel for respiratory qualification and fitness

- decontamination equipment and facilities, including spill control materials, shower, eyewash, changing facilities

- area radiation monitoring equipment

- laboratory facilities and equipment for radioactive materials and sample analyses

- contamination control and monitoring equipment and areas

The SAR should describe the types of radiation detectors and monitors, numbers, locations, operational sensitivity and range, and frequency and methods of calibration for all of the equipment and instrumentation identified above.

Health physics facilities can be set up in permanent structures, temporary buildings, or trailers. Facilities should be located outside restricted areas and, if practicable, away from areas with elevated external dose rates and potential sources of airborne radioactivity. Exceptions can include facilities for storing items that need to be readily available within restricted or elevated dose rate areas, as well as personnel decontamination, shower, and changing facilities. The site plot drawings of the installation should identify and describe the health physics facilities to sufficiently demonstrate the applicant's understanding of the associated requirements and operational functions.

The following regulatory guides and industry standards provide information, recommendations, and guidance on various aspects of health physics equipment, instrumentation, and facilities. The NRC considers these sources as acceptable guidance for describing the basis for implementing activities to comply with applicable regulatory requirements:

- ANSI/HPS N13.1
- RG 8.2
- RG 8.4
- RG 8.25
- RG 8.28
- NUREG-0800, Sections 11.5 and 12.5
- NCRP Report No. 57
- NCRP Report No. 112

10A.4.4.3 *Policies and Procedures*

Under 10 CFR 20.1101, "Radiation protection programs," licensees are required to "develop, document, and implement a radiation protection program commensurate with the scope and extent of licensed activities." The SAR should describe the radiation protection program, including details of all health physics-related policies and procedures to be implemented at the DSF, including an annual review of the program content and implementation.

In addition to the regulatory guides identified in Table 10A-2, the following documents contain applicable guidance and criteria for health physics procedures relevant to DSF operations:

- ANSI/HPS N13.6, "Practice for Occupational Radiation Exposure Record Systems"

- American Society for Testing and Materials (ASTM) E1167, "Standard Guide for Radiation Protection Program for Decommissioning Operations"

- ASTM E1168, "Standard Guide for Radiological Protection Training for Nuclear Facility Workers"

- ANSI/HPS N13.30, "Performance Criteria for Radiobioassay"

- ANSI/HPS N13.32, "Performance Testing of Extremity Dosimeters"

- ANSI/HPS N13.41, "Criteria for Performing Multiple Dosimetry"

- ANSI/HPS N13.42, "Internal Dosimetry for Mixed Fission and Activation Products"

- NCRP Report No. 87, "Use of Bioassay Procedures for Assessment of Internal Radionuclide Deposition," issued 1987

- NCRP Report No. 112, “Calibration of Survey Instruments Used in Radiation Protection for the Assessment of Ionizing Radiation Fields and Radioactive Surface Contamination,” issued 1991

- NCRP Report No. 116, “Limitation of Exposure to Ionizing Radiation,” issued 1993

- NCRP Report No. 127, “Operational Radiation Safety Program,” issued June 1998

- NCRP Report No. 134, “Operational Radiation Safety Training,” issued 2000

- National Safety Council, “Accident Prevention Manual: Engineering and Technology,” 14th edition, 2015.

- NUREG-0800, Section 12.5

10A.5 Review Procedures

This section describes review procedures used to evaluate (1) compliance of facility design and operations with regulatory requirements for radiation protection, (2) implementation of design and operations features and programs to ensure that exposures (public and personnel) will be ALARA and comply with regulatory dose limits, and (3) the adequacy of the applicant’s radiation protection and health physics programs for the proposed DSF. Figure 10A-1 shows the interrelationship between the radiation protection evaluation and the other areas of review described in this SRP.

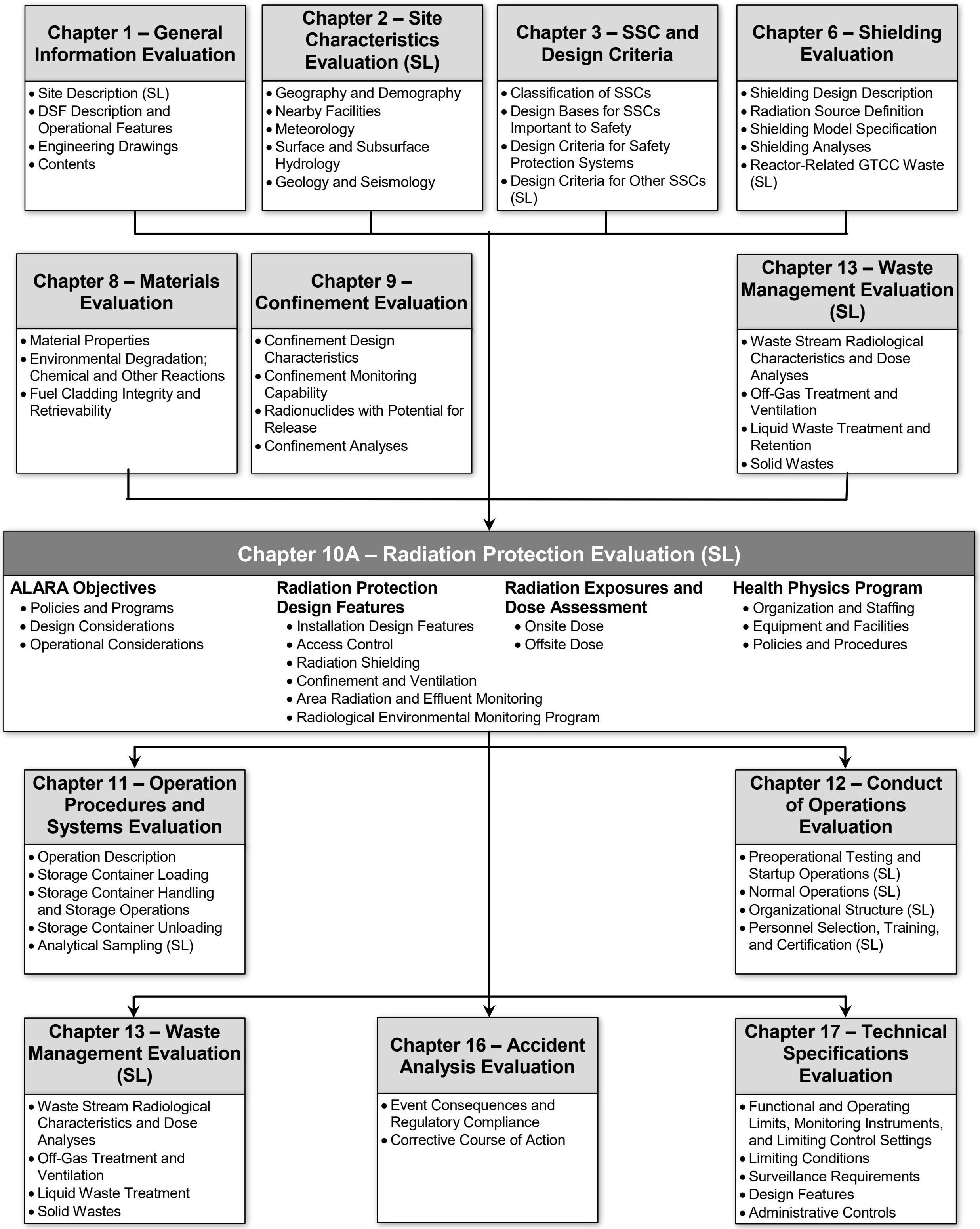

Figure 10A-1 Overview of Radiation Protection Evaluation

The radiation protection review includes evaluation of compliance with all regulatory requirements and acceptance criteria given in this SRP and other applicable NRC documents and accepted codes and standards. Always assume that such a comprehensive scope of the review applies, even though it is not further detailed or repeated in this section. Coordinate with the conduct of operations (SRP Chapter 12) reviewer to ensure that preoperational testing includes testing of design features and procedures that are significant to radiation protection and that ensure doses are ALARA.

10A.5.1 ALARA Objectives

This section provides procedures for reviewing the scope and objectives of the ALARA program in protecting workers and members of the public. The review of the ALARA program addresses policies, procedures, and facility design features that reduce radiation exposures, dose rates and doses and minimize release of radioactive materials in the environment. The review includes evaluations of compliance with all regulatory requirements and acceptance criteria given in this SRP and other applicable NRC documents and accepted codes and standards. Section 10A.5.2 also provides review guidance related to ALARA because of the significant overlap between that section and this section.

10A.5.1.1 *Policies and Programs*

Determine that an effective ALARA program and objectives will be applied to most functions associated with construction, operation, and eventual decontamination and decommissioning of the DSF. Verify that ALARA philosophies and program goals are evident throughout the SAR in the description of equipment, facility designs, and operational procedures. In addition, through discussions with reviewers of other topics, verify that other topic areas of the SAR appropriately reflect the ALARA policies (e.g., facility design and operations descriptions).

Ensure that the applicant’s ALARA policy and program includes a written policy statement that expresses management’s commitment to maintain exposures to workers and the public ALARA and addresses both facility design and operations. Review the proposed ALARA program organization and ensure that it identifies the organizational structure, including descriptions of responsibilities and activities of ALARA personnel. Review the ALARA policy and program content and ensure that the policy includes the elements identified in Section 10A.4.1.1 above and that the program content includes provisions for those items described in Section 10A.4.1, including the program organization and programmatic elements listed in Section 10A.4.1.1. In addition, consider the guidance of NUREG-0800, Section 12.1, “Assuring that Occupational Radiation Exposures Are As Low As Is Reasonably Achievable,” and RG 8.8, as they provide guidance that may be applicable to the review of an applicant’s proposed ALARA program for a DSF.

10A.5.1.2 *Design Considerations*

Ensure that the facility design and layout demonstrate consideration of ALARA principles. Ensure that the design criteria (see SRP Chapter 3, “Principal Design Criteria Evaluation”) also incorporate ALARA criteria in facility features. Determine whether the SAR identifies choices between otherwise comparable alternatives affected by ALARA considerations and provides sufficient bases for the selected option(s) as the most appropriate. Evaluate the design and layout for consideration of the factors identified in Sections 10A.4.1.2 and 10A.4.2 above.

10A.5.1.3 *Operational Considerations*

Determine that the descriptions of proposed operations adequately demonstrate that the applicant has incorporated ALARA principles into operational procedures. Ensure that the applicant has developed plans, methods of operation, and procedures in accordance with applicable guidance and that these items adequately address considerations detailed in Section 10A.4.1.3 above.

10A.5.2 Radiation Protection Design Features

This section addresses review procedures that apply to installation design, access control, shielding, confinement and ventilation, area radiation and effluent monitoring (including the instrumentation), and the radiological environmental monitoring program. In support of this process, NUREG-0800, Sections 12.3–12.4, also provide guidance that the NRC finds acceptable to use to review DSF radiation protection design features.

In reviewing the DSF design features and dose analyses, as described in Section 10A.5.3 below, consider whether the license (in the technical specifications) should include dose rate limits for some of the facility SSCs and features, such as the SNF, reactor-related GTCC waste, or HLW storage containers. In determining the need for such limits, consider factors such as the dose rates for different operational configurations, the nature of the DSF design, potential dose impacts of design changes, and the need for such limits to ensure continued compliance with 10 CFR Part 72 and 10 CFR Part 20 dose limits. Ensure that any dose rate limits are derived from the applicant's dose rate and dose analyses for normal (and off-normal) conditions. The limits should be developed for appropriate configurations of appropriate facility SSCs and features and aspects of these SSCs and features that are important for personnel or public doses. The limits should be compared against the maximum measured dose rates. Ensure that the license (technical specification) condition that specifies dose rate limits also specifies an appropriate number of measurements at appropriate locations on facility SSC or feature surfaces. The specified measurements (numbers, locations, SSC, or feature surfaces) should be sufficient to ensure compliance with the dose rate limits. Also consider whether the license technical specifications should include any limits and measurement requirements for (removable) contamination for appropriate facility SSCs (e.g., SNF, reactor-related GTCC waste, or HLW storage containers). Considerations should include impacts to dose estimates. Appropriate technical specifications should result in contamination levels that contribute negligibly to doses and dose rates at or beyond the controlled area boundary (that is, off site).

10A.5.2.1 *Installation Design Features*

Review the SAR installation design features and ensure that the site and facility drawings and diagrams clearly identify facility features that affect the radiation protection analyses. Ensure that the radiation protection analyses are consistent with, or are bounding for, the design of the facility and the site as described in the site and facility drawings and diagrams. The facility should be constructed in accordance with the design drawings and diagrams. Ensure that the drawings and diagrams also clearly identify any public access facilities that traverse the controlled area (e.g., as allowed by 10 CFR 72.106(c)).

For systems used to treat liquid and gaseous effluents, coordinate with the waste management (SRP Chapter 13) reviewer to review piping and instrumentation diagrams and system process flow diagrams and verify that the applicant has adequately characterized and included in its analyses the aspects of these systems relevant to the DSF radiation protection design and analyses. These aspects include all sources and volumes of liquid process and effluent streams;

points of collection of liquid wastes; flow-paths of process streams through each system, including potential bypasses; the treatment provided and expected decontamination factors or removal efficiencies for radionuclides and holdup or decay time; and points of release of liquid and gaseous effluents to the environment. With respect to potential bypasses, ensure that the applicant adequately considered improper connection to nonradioactive systems and the possibility of uncontrolled and unmonitored liquid and gaseous effluent releases.

10A.5.2.2 *Access Control*

Review the SAR description and provisions for access control and verify that (1) the facility and operational planning incorporate the necessary and desirable personnel protective measures, (2) the facility's design provisions reflect a radiological and engineering appreciation of potential dose rates and contamination levels in the transfer facilities (dry or wet (e.g., a pool)) and waste management facilities, (3) the descriptions of ALARA and other radiological protection features as well as the planning for implementation of physical protection incorporate provisions for access control, and (4) the facility design and operations incorporate the necessary means and methods (e.g., barriers, arrangements with appropriate enforcement agencies) for controlling access to controlled areas in order to ensure public health and safety.

10A.5.2.3 *Radiation Shielding*

Examine the applicant's evaluation of the facility shielding design; coordinate this review with the shielding (SRP Chapter 6) reviewer. Confirm that the applicant has identified facility design and site features that have a bearing on occupational and public doses and dose rates. These features include aspects of the facility and site that result in increased dose rates (e.g., streaming paths) as well as those that help to reduce dose rates (e.g., shield walls). Confirm that the applicant's evaluation treats these features in a manner that is consistent with, or bounding for, the facility design and site features descriptions, including the drawings and diagrams, in the chapters of the SAR that provide general information, site characteristics, and principal design criteria. Confirm that the applicant's evaluations for the different design-basis conditions (i.e., normal conditions, anticipated occurrences, and accident conditions) account for the effects of those conditions on the facility design and site features.

Also ensure that the applicant's evaluations adequately address the different configurations of the facility's features and SSCs consistent with the variety of facility operations, including those that may only be temporary. This includes, for example, construction work to expand a storage array that removes or exposes materials relied on for shielding that are not otherwise removed or exposed during normal operations. Depending upon the applicant's analyses, consider the need for any license or technical specification conditions regarding these configurations and operations. ANSI/ANS 6.4.2, "Specification for Radiation Shielding Materials," includes information that may be useful to consider as part of this review. Also confirm that the applicant's evaluations account for facility layout and the maximum quantities of SNF, reactor-related GTCC waste, and HLW that will be stored at the facility.

Examine the dose rates the applicant derived from its shielding analysis, coordinating with the shielding reviewer (SRP Chapter 6). Confirm that the evaluations produce dose rates for a sufficient number of locations to support the evaluation of the occupational doses and public doses. These locations should include surfaces of facility features and SSCs that are used to handle or store SNF, reactor-related GTCC waste, or HLW; locations of personnel conducting operations (e.g., during storage container loading, transfer, surveillance, maintenance activities); other locations on site that will be occupied by facility personnel in restricted areas and outside of restricted areas, including both radiation worker and nonradiation worker personnel (e.g., administrative staff); locations on site of public access facilities (e.g., roads and waterways); and locations at the controlled area boundary and beyond the controlled area boundary that are needed to determine doses to real individuals around the facility. The applicant's analysis should provide sufficient dose rate information to support the evaluations for demonstrating the facility design and operations meet, or will meet, the 10 CFR Part 20 dose limits for facility personnel and the public and the limits in 10 CFR 72.104(a) and 10 CFR 72.106(b).

The shielding (SRP Chapter 6) reviewer is responsible for reviewing the applicant's shielding analysis, including the computer codes and models and calculation of dose rates, and the performance of any confirmatory shielding calculations. However, since the radiation protection review is based, in part, on the outcome of the shielding analysis, coordinate the review of this SRP chapter with the shielding reviewer to determine the adequacy and acceptability of the applicant's shielding analysis. This coordination includes confirming with the shielding reviewer that the applicant's analyses, including model parameters and assumptions, are appropriate and that the calculated dose rates are reasonable. This coordination may also include identification of the need for confirmatory calculations and determining the level of effort that should be expended in performing calculations. In addition to the considerations described in Section 6.5.4.4 of this SRP, determination of the level of effort should include consideration of the margins in estimated doses to dose limits, such as the limits in 10 CFR 72.104(a) and 10 CFR 72.106(b).

Evaluate whether the proposed shielding use is consistent with the applicant's design objectives relative to keeping radiation doses ALARA. As part of this evaluation, consider the applicant's descriptions, if any, in the SAR of the use of temporary or portable shielding, remote handling, or other protective features. Ensure that use of these features or other actions is included in appropriate sections of the SAR, such as in the descriptions of facility operations for the handling, receipt, transfer, and storage of materials. Descriptions of the facility may include placement of barriers between occupied areas and radioactive materials; use of these barriers may be for ALARA purposes.

Consider scenarios and designs where the use of additional features or SSCs or certain kinds of remote operations may not be in keeping with ALARA objectives. Such scenarios and designs include those where significant extra shielding or the use of remote operations (e.g., container movements directed from a separate area using lasers and cameras) is needed to ensure adequate protection of personnel. Designs with such features or modes of operations introduce the possibility of scenarios leading to potential significant personnel doses in the event they need to perform specific actions to recover from anticipated occurrences (e.g., crane, laser, or camera malfunction) occurring when containers are not located within the extra shielding. For such designs, ensure that the SAR includes additional information to justify that the facility design and operations are consistent with ALARA objectives. This information may include descriptions of actions taken to minimize the likelihood of such occurrences (e.g., equipment failure) or other kinds of features or operations that the applicant will use to minimize doses in such instances. Ensure that the design and operations adequately follow ALARA principles. Also consider whether conditions regarding the design or operations may be needed in the license technical specifications to ensure adequate protection, compliance with regulatory requirements (including dose limits), and adequate consideration of ALARA. These technical specifications may include verifications of correct operations, monitoring the condition of SSCs when the extra shielding is used, specifications (e.g., thicknesses) of the extra shielding features and remote operations equipment, requirements for use of these features and equipment, requirements for recovery actions for off-normal events, preoperational testing of remote operations and equipment, and limits on the duration of high dose-rate configurations.[1]

10A.5.2.4 *Confinement and Ventilation*

The confinement evaluation (SRP Chapter 9) includes an assessment of the applicant's estimates of radionuclide releases to the environment from the SNF, HLW, and reactor-related GTCC waste containers. The waste management evaluation (SRP Chapter 13) addresses radionuclide releases from site-generated wastes. Those analyses include confinement and ventilation aspects applicable to sealed storage containers (for which releases are usually minimal) and to systems and components that are not designed to be sealed.

The radiation protection review of confinement and ventilation has two components. The first is to evaluate information from the confinement and waste management evaluations and to determine if estimates of radionuclide release rates and other site-specific information or estimates of release/effluent doses or dose rates are adequate for estimating onsite and offsite doses, as described in Section 10A.4.3 above. The second is to evaluate the protection features of the waste management facility's and other DSF facility's (e.g., pool) ventilation systems. This part of the review should identify how the confinement and ventilation system components and controls function to do the following:

- Maintain all radiation exposures and doses ALARA.
- Prevent the spread of radioactive materials and contamination between and among areas, including the possibility of unmonitored and uncontrolled releases.
- Limit the spread of radioactive materials within ventilation system(s) beyond installed filtration components (e.g., high-efficiency particulate air and charcoal filters).
- Handle process off-gases (e.g., waste treatment, venting of storage containers, venting of liquid waste collection tanks).
- Monitor off-gases, including through sample collection and analysis, in complying with effluent release limits in unrestricted areas.

Ensure that confinement and ventilation systems conform to the applicable guidance of NUREG-0800. Section 11.3, RG 1.140, "Design, Inspection, and Testing Criteria for Air Filtration and Adsorption Units of Normal Atmosphere Cleanup Systems in Light-Water-Cooled Nuclear Power Plants," and RG 1.143, "Design Guidance for Radioactive Waste Management Systems, Structures, and Components Installed in Light-Water-Cooled Nuclear Power Plants."

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[1] The shielding design features are important for ensuring compliance with regulatory dose limits, including the limits in 10 CFR 72.104(a) and 10 CFR 72.106(b). These limits apply to all dry storage operations, including loading and unloading operations as well as storage at the ISFSI pad. Thus, for canister-based storage container designs, the limits apply to operations with the transfer cask as well. This would be true even for a DSF that is co-located with a 10 CFR Part 50 or 10 CFR Part 52 facility and loading and unloading operations occur in that Part 50 or Part 52 facility's spent fuel building. This position is consistent with the November 16, 2006, rulemaking's definition between 10 CFR Part 50 and 10 CFR Part 72 for criticality safety (see Volume 71 of the Federal Register, page 66648).

10A.5.2.5 Area Monitoring and Effluent Monitoring Instrumentation

Evaluate the applicant's description of fixed area radiation monitoring instrumentation and continuous airborne and liquid (as applicable) monitoring instrumentation, placement of such monitors, and whether the equipment includes automatic control features (such as terminating or diverting effluent releases as warranted by safety classification). Review the criteria and methods used for determining alarm setpoints. Review the information provided on the auxiliary and emergency power supply. Evaluate information and specifications on instrument range, sensitivity, accuracy, energy dependence, calibration methods and frequency, recording devices, readouts, and alarms. NUREG-0800, Section 11.5, describes acceptable guidance for conducting reviews of DSF area radiation and effluent monitoring instrumentation. NUREG-0800 Sections 11.2, 11.3, and 12.3–12.4 also include guidance that may be useful for these reviews.

The documents referenced here and in Section 10A.4.2.5 above include criteria and guidance that the NRC accepts to the extent it is applicable to the equipment and monitoring for a DSF. Confirm that the SAR demonstrates that the equipment and its placement are adequate to ensure that the DSF design and operations will meet regulatory requirements, including those in 10 CFR 72.104(b), 72.104(c), 72.126(b), 72.126(c), 72.126(d), and 10 CFR 20.1406, as well as any technical specifications regarding monitoring and effluents. The equipment should have sufficient sensitivity and response capabilities to detect the expected area dose rates and nuclide concentrations in effluents as well as changes to these parameters that would indicate a problem and require personnel actions. The equipment should be adequate for the functions for which monitoring is to be performed, detecting the types and spectra of radiation to be monitored, detecting radionuclides (considering the chemical and physical properties of the nuclides in the effluents), and performing under all required conditions. The equipment should also be adequate to ensure prompt detection of a problem (includes appropriate alarm setpoints) and enable prompt personnel response to address the problem and avoid unmonitored or uncontrolled releases and spreading of contamination to non-radiation systems and areas. In addition, the reviewer should ensure that the SAR describes the intended personnel responses to alarms and emergency conditions. Coordinate, as required, with other reviewers (e.g., conduct of operations, operating procedures, and accident analysis). Ensure that the personnel responses are reasonable to enable or ensure that the DSF design and operations will meet the relevant regulatory requirements and limits.

10A.5.2.6 Radiological Environmental Monitoring Program

Review the description and scope of the effluent and environmental monitoring program. Ensure that it considers all potential exposure pathways and provides the necessary data to identify and assess those pathways that would lead to the highest potential external and internal exposures of the offsite population (both collectively and for the maximally exposed real individual(s)). Also confirm that the program is designed to yield information and results that can be used to (1) estimate collective doses with reasonable accuracy, (2) estimate doses to offsite individuals (to ensure compliance with 10 CFR 72.104(a) and other applicable regulatory limits), and (3) assess the effectiveness of radiological controls applied to minimize effluent releases and maintain releases and offsite doses ALARA. Confirm that the program is also capable of verifying that the assumptions and bases used in the SAR dose assessments are valid and maintained during facility operations. NUREG-0800, Section 11.5, and RG 4.1 present useful guidance on the development and implementation of a radiological environmental monitoring program for DSFs. Finally, ensure that the license technical specifications include appropriate program information in accordance with 10 CFR 72.44(d) and that the program, as described in the SAR and technical specifications, is adequate to fulfill the purposes identified in 10 CFR 72.44(d).

10A.5.3 Radiation Exposures and Dose Assessment

This section addresses the review of dose assessment methods and results presented for evaluating doses to individuals and collective doses on site (i.e., within the controlled area) and off site (i.e., at or beyond the controlled area boundary) for compliance with applicable regulatory criteria. For onsite dose evaluations, ensure that the results are adequate to support evaluations for facility personnel that are occupational workers and facility personnel that are non-radiation workers (e.g., administrative staff) as well as evaluations for members of the public for facilities that include public access areas within the controlled area boundary (e.g., as allowed, in accordance with 10 CFR 72.106(c)).

Coordinate with the shielding (SRP Chapter 6), confinement (SRP Chapter 9), and waste management (SRP Chapter 13) reviewers to understand the bases for estimates of doses and dose rates and radionuclide concentrations in effluents. Coordinate with these reviewers to also ensure that the analyses adequately and appropriately consider the effects of the facility's design features and SSCs and facility operations as well as site characteristics, including layout and features, as described in the SAR. Ensure that the analyses address the effects of potential configuration changes of the storage container contents (e.g., reconfiguration of damaged fuel) under different conditions.

Ensure that the applicant considered design and operations effects that may result in configurations and conditions that only exist for limited durations, and not for the life of the facility, and are not traditionally considered or evaluated in normal facility configurations. Such configurations include scenarios where construction at a facility to expand the storage array removes materials relied on for shielding or exposes those materials to the impacts of normal, off-normal, and accident conditions that may occur during that period of time. Such configurations may also necessitate consideration and evaluation of off-normal and accident conditions that are not typically considered in DSF SARs.

Consider whether the facility SSC designs and operations could result in significant dose impacts to personnel or members of the public for anticipated occurrences and ensure that the applicant's analyses adequately account for those effects, including during recovery from the anticipated occurrences. For example, the design of an SSC may necessitate that operations be conducted remotely under normal conditions (due to significantly high dose rates), but recovery for an anticipated occurrence may require that personnel perform recovery actions near the SSC. The extended time that this configuration exists, compared with the duration under normal operations, may also impact doses to members of the public, including those evaluated for 10 CFR 72.104(a).[2]

The results of the confinement and waste management evaluations include doses and dose rates for effluents and releases from the storage containers and the facility's waste management systems, respectively. Thus, evaluation of those analyses is the purview of those reviewers. Coordinate with those reviewers to ensure that the results are sufficient to support the dose assessments described in this section.

Ensure that the dose analyses include contributions from direct radiation, effluents, and, as appropriate, surface contamination (at the levels allowed by the technical specifications). Since the evaluation of doses from surface contamination would be similar to that for effluents,

[2] See Footnote 1 on page 10A-30 regarding applicability of regulatory dose limits to all dry storage operations.

coordinate as needed with the confinement reviewer to evaluate any contamination contributions, which technical specifications limits should make negligible for offsite doses. The guidance below regarding effluent doses and dose rates should also be understood to include, where appropriate, contributions from surface contamination.

10A.5.3.1 *Basis and Assumptions of Dose Assessment*

Review the calculation of dose rates and doses associated with radioactive releases or effluents, as needed, including applications that use computer codes to calculate these dose rates and doses, cases where the confinement or waste management reviewers need assistance, and when there's a need to evaluate effluent release rates or dose rates and doses at locations in addition to those considered in the confinement or waste management evaluations. The NRC recognizes that various computer codes are available for analyzing radiological impacts associated with releases of radioactive materials. The considerations and guidance described in Chapter 6 of this SRP for computer codes used for shielding analyses (assessment of direct radiation dose) apply to the use of these computer codes and review of their use.

As part of this review, the staff should carefully evaluate the applicability of the codes described in the application and the reasonableness of all assumptions and parameters forming the basis of each set of dose results. Examples of these codes include the following:

- MARC-1—a suite of linked computer codes used for calculating the radiological effects of releases of radionuclides to the environment developed by the United Kingdom's National Radiological Protection Board

- LINGAP and HMARC—modules of MARC-1 used to calculate the effects of an atmospheric release

- NRCDose—a code that implements the method described in RG 1.109, "Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I." This code expresses doses as whole body and critical organ doses

In reviewing the results of the dose assessment, confirm that the applicant has provided sufficient information describing the bases and assumptions of the dose assessment in demonstrating compliance with the NRC dose limits. As part this review, confirm the appropriateness of all selected exposure pathways, applied values for all essential parameters, sources of the data supporting the use of these values (site specific, default from NRC guidance, or from industry guidance), and computer codes and software versions. For any codes used, ensure that the SAR demonstrates the code has been properly validated for its use, in a manner similar to that described for shielding codes (see SRP Sections 6.4.4.1 and 6.5.4.1). Also, review a sample input file to verify proper entry of facility information into the code and that the applicant used proper input parameters and code features. Consider the levels of conservatism applied in all assumptions and selection of parameters, and the extent to which some of the analyses were modified in light of the results of separate sensitivity analyses, as identified by the applicant. If dose results for individuals located at or beyond the 10 CFR Part 72 controlled area boundary are marginally close but in compliance with the NRC dose criteria in 10 CFR Part 20 and 10 CFR Part 72, independently assess the direction and magnitude of underlying uncertainties to confirm that dose results represent conservative upper bounding estimates and still comply with the NRC limits and criteria when the impacts of uncertainties are taken into account.

10A.5.3.2 *Onsite Dose*

Use all relevant information to estimate total individual and collective doses DSF workers receive and determine whether applicable dose and ALARA criteria have been met. This onsite dose evaluation includes the following steps:

- Review the estimated annual occupancy times, including the maximum expected total number of hours per year for any individual and total person-hours per year for all personnel for each radiation area, including storage areas during normal operations and anticipated occurrences and ensure these times are reasonable.

- Ensure that estimates of annual doses are based on the maximum number of storage containers placed into storage in 1 year and include both direct radiation and inhalation of airborne radioactive materials, as warranted by operations.

- Ensure that descriptions of procedures that involve exposures to workers are compatible with the occupancy times and proximities assumed in the bases of dose estimates.

- Ensure that estimates of individual and collective doses are based on reasonable assumptions regarding presumption of skill levels and training, extent of care taken in managing and conducting facility operations (including nuclear safety related), presence of proper supervision and quality control, and other factors that might tend to increase doses.

- Ensure that dose calculation methods are appropriate and correctly implemented, and confirm that there is sufficient information in the SAR for the staff to conduct an independent evaluation of dose results.

Perform independent estimates of onsite collective doses. The level of effort for these estimates may depend upon various factors, such as indications that the SAR estimates may not be bounding, reasonableness of assumptions and parameters used in the analyses, and applicable considerations discussed in SRP Chapter 6, Section 6.5.4.4, “Confirmatory Analysis,” regarding level of effort for confirmatory shielding analyses. Clearly identify assumptions or models that differ from those in the SAR, and discuss whether the staff’s assessment of collective dose estimates support the applicant’s considerations related to maintaining occupational exposures ALARA.

Compare the estimated annual individual occupational doses with the dose limits in 10 CFR 20.1201(a). If the estimated doses approach or exceed these limits, confirm that the planned conduct of operations (SRP Chapter 12) includes conditions (e.g., staffing plan, monitoring) that ensure that individual doses will be controlled and that all dose limits will not be exceeded.

Consider all relevant information presented in the applicant’s evaluation for meeting the 10 CFR 20.1301 limits for members of the public, which apply to personnel that are not radiation workers (i.e., are not receiving an occupational dose as defined in 10 CFR 20.1003) and to members of the public when access to controlled areas is allowed (e.g., facilities such as those in 10 CFR 72.106(c) that traverse a controlled area). Consider the assumptions and bases of the applicant’s evaluation, including actions to be taken or administrative controls to be instituted by the applicant to ensure that doses meet regulatory limits.

10A.5.3.3 *Offsite Dose*

For offsite doses, evaluate the following four principal sets of doses and the calculation methods against the relevant acceptance criteria: (1) annual collective (person-rem) dose to the surrounding population if effluents are anticipated from the facility, (2) annual dose to the maximally exposed real individual, (3) maximum hourly dose in unrestricted areas, and (4) maximum dose from any design-basis accident to any individual located on or beyond the controlled area boundary. If effluent releases are anticipated from accidents, then ensure that the applicant calculated the collective dose to the surrounding population for accidents as well for evaluations for 10 CFR 72.100(a). For each of these determinations, ensure that the applicant's analysis considers all potential exposure pathways; identifies the pathways the applicant determined to lead to the highest external and internal doses; describes the methods and data applied in assessing doses (e.g., estimated radionuclide concentrations, atmospheric dispersion and deposition parameters (both long and short term)); and provides the bases for all selected data, methods, and exposure pathways assessment. Ensure that dose contributions from other activities in the surrounding area (i.e., within the surrounding region) are also addressed in analyses of compliance with 10 CFR 72.104(a) limits, where applicable.

Consult with the confinement (SRP Chapter 9) and waste management (SRP Chapter 13) reviewers to obtain dose or dose rate estimates for effluents or releases from the storage containers and the facility's waste management systems. Obtain doses or dose rates from effluents or releases for normal, off-normal, and accident conditions (from the confinement and waste management reviewers). Ensure that the total doses from both direct radiation and effluents or releases do not exceed the relevant acceptance criteria. For the annual dose to the maximally exposed real individual, the total of the annual doses from normal conditions and bounding doses from anticipated occurrences, together with doses from other facilities in the region, should not exceed the limits in 10 CFR 72.104(a). If the confinement or waste management analyses only provide effluent dose results at 100 meters (328 feet) or for only a single storage container (confinement only), coordinate with these reviewers to (1) evaluate how effluent releases may contribute to doses at additional distances and for the full array(s) of storage containers to be allowed by the proposed license and (2) determine what additional analyses may be needed in the SAR. For analyses where the applicant chooses to demonstrate compliance with the limits in 10 CFR 20.1301, using the option described in 10 CFR 20.1302(b)(2), coordinate review of the analysis results with the confinement and waste management reviewers to ensure all criteria for that option are met.

<u>Collective Dose to Surrounding Population</u>

In reviewing annual collective doses attributable to direct radiation and facility effluents, ensure that the models, assumptions, and parameters that were used to estimate doses have duly considered the site's and surrounding region's characteristics (SRP Chapter 2) and facility shielding, confinement, and waste management design features (SRP Chapters 6, 9, and 13, respectively). These characteristics and features include the following:

- site layout and location of all onsite facilities and sources of radiation exposures and radioactive effluents
- land and water use, topography, and population data, both current and projected distributions, in each sector and radial distances from the site

- direct radiation exposure and dose rates as a function of sector and radial distances from the site and dose receptor locations

- meteorological data for the site and its surroundings in each sector and radial distances from the site

- radioactive material release rates, downwind dispersion, and deposition in site surroundings and at locations of identified offsite dose receptors

- engineered design features such as berms and shield walls and their configurations

Ensure that the applicant has determined a collective dose for the surrounding population and that the dose considers all important exposure pathways (e.g., direct radiation, airborne releases) and modes of exposure (e.g., external exposure, inhalation). Assess the increment by which the collective dose would be increased by the presence of any other (existing or projected) activities (e.g., fuel cycle facility) within the surrounding area or region of the proposed DSF. Ensure that the computational models or equations and assumptions used are acceptable and consistent with NRC guidance. Ensure that the data used in computer models or equations are appropriate and accurate.

Confirm that there is sufficient information in the SAR for the NRC staff to conduct an independent, confirmatory evaluation of collective doses. In performing an independent evaluation of doses, the level of effort may vary depending on several factors (e.g., large uncertainties in results, analyses use methods that are not consistent with those described in the SRP). Determination of the necessary level of effort may involve coordination with the shielding (SRP Chapter 6), confinement (SRP Chapter 9), and waste management (SRP Chapter 13) reviewers. If the SAR methods and assumptions are deemed acceptable, perform an appropriate number of confirmatory or spotcheck calculations. For all independent, confirmatory calculations, clearly identify any assumptions or models that differ from those in the SAR.

Evaluate collective dose estimates for accidents in a similar manner as that for annual collective dose estimates for normal operations, applying appropriate considerations regarding the nature of such events. A primary consideration is the limited duration of an accident event and recovery operations. These considerations should influence selection of modeling parameters and values for site characteristics used in the analysis (e.g., meteorological conditions that result in bounding doses and impacted sectors).

Determine whether the annual collective dose estimates support the applicant’s considerations and conclusions in maintaining radioactive effluent releases and offsite doses ALARA for normal and off-normal conditions. Determine whether these annual collective dose estimates and the accident collective dose estimates sufficiently characterize the radiological impacts of the facility on populations in the surrounding region, in compliance with 10 CFR 72.100(a).

Dose to Maximally Exposed Real Individual

Determine whether the highest offsite dose received by a real individual is less than the limits specified in 10 CFR 72.104(a). Many of the same factors considered in the collective dose assessment are applicable to this review. Refer to the two approaches discussed in Section 10A.4.3.3 above to demonstrate compliance with dose limits and assess the implications of the approach the applicant used for its site. Ensure that the methods, including any computational models or equations and assumptions, used are acceptable and appropriate for this analysis. Confirm that there is sufficient information in the SAR for the staff to conduct an independent, confirmatory evaluation of doses.

Evaluate the applicant's assessment of direct dose rates and radioactive material concentrations in effluents or dose rates from the effluents at locations beyond the controlled area boundary for normal operations and anticipated occurrences. Identify the location of offsite individual(s) likely to receive the highest dose from direct radiation and doses associated with releases of facility effluents. Confirm that the applicant's dose estimates include the contributions from the relevant exposure pathways for the individual(s). Confirm that the applicant has adequately and correctly identified the location(s) and individual(s) that are likely to receive the highest doses.

Assess the annual whole-body dose equivalent, as well as the dose equivalent to the thyroid and any other critical organs (other than the thyroid). The NRC has accepted the use of TEDE as a surrogate for whole-body dose equivalent. Confirm that the applicant has described the methods used for these dose calculations.

As in the collective dose assessment, perform independent, confirmatory calculations, as necessary, to verify the applicant's results and adequacy of assumptions. Clearly identify any assumptions or models that differ from those in the SAR and confirm that such assumptions and associated parameters are adequate and conservatively bounding.

Assess the increment by which the whole-body dose would be increased by the presence of other (existing or projected) activities (e.g., a radioactive waste facility) within the area or region surrounding the proposed DSF. Ensure that the combined annual dose equivalent from the DSF and the other activities does not exceed 25 mrem to the whole body, 75 mrem to the thyroid, and 25 mrem to any other critical organ.

Assess the TEDE resulting only from airborne effluent releases, and ensure that this dose does not exceed the 10-mrem (0.1-mSv) per year ALARA constraint level in 10 CFR 20.1101(d) for an individual member of the public likely to receive the highest dose from effluents. Confirm that all supporting assumptions and associated parameters are adequate and conservatively bounding.

Review the applicant's determination that the maximum dose in any unrestricted areas resulting from external sources does not exceed 0.002 rem (0.02 mSv) in any single hour (10 CFR 20.1301(a)(2)), and determine whether the distance to the nearest boundary of the controlled area is sufficient to ensure compliance with this dose standard. Confirm that all supporting assumptions and associated parameters are adequate and conservatively bounding.

Determine whether the highest offsite dose from accident conditions is less than the limits in 10 CFR 72.106(b). Confirm that the applicant has calculated the dose consequences for each accident condition. Also confirm that the applicant calculated the doses for locations on the nearest boundary of the controlled area. Ensure that the doses include the contributions from direct radiation and releases resulting from the impacts of the accident conditions on the affected facility SSCs and features (e.g., SNF, reactor-related GTCC waste, or HLW waste containers, and waste management systems). Finally, ensure that the doses for each accident condition are based on a reasonably bounding or conservative time duration that includes recovery from the accident condition's impacts (e.g., moving SNF from a damaged container to a configuration consistent with normal operations). The NRC has accepted 30 days as sufficient for previous applications. Ensure that the applicant's selected timeframe is appropriate based on considerations unique to the facility design and operations.

10A.5.4 Health Physics Program

This section addresses procedures to review the scope, functions, and capabilities of the health physics program. NUREG-0800, Sections 11.5, 12.1, 12.5, 13.2.2, “Non-Licensed Plant Staff Training,” and 13.4, “Operational Programs,” provide guidance that may be useful and applicable in reviewing DSF health physics programs. Moreover, NUREG-1736, “Consolidated Guidance: 10 CFR Part 20 – Standards for Protection Against Radiation,” provides additional NRC guidance on the implementation of a health physics program and radiation protection.

10A.5.4.1 *Organization and Staffing*

Evaluate the administrative organization of the applicant’s health physics program. As part of the review, confirm that the program describes the authority, responsibility, experience, and qualifications of the personnel responsible for the health physics program and that the program is sufficiently staffed to conduct all program operations. Confirm that the health physics manager and health physics staff have the authority to supervise, monitor, and halt any facility operations and procedures that could result in unnecessary radiation exposures to workers and members of the public or lead to doses in excess of administrative limits and NRC regulations. Ensure that the organization and staffing description satisfactorily addresses the other criteria provided in Section 10A.4.4.1 above. Some of this information may be described in the SAR chapter on conduct of operations; thus, coordinate with the conduct of operations reviewer (SRP Chapter 12, particularly Sections 12.4.1.1–2, 12.4.6.1–2, 12.5.1.1–2, and 12.5.6).

10A.5.4.2 *Equipment Instrumentation and Facilities*

Review the applicant’s description of the portable, fixed, and laboratory equipment and instrumentation for performing radiation and contamination surveys, sampling airborne radioactive materials in ambient facility areas and release points (e.g., building vents or liquid discharges into onsite or offsite surface water bodies), monitoring area radiation, and monitoring personnel exposures during normal operations, anticipated occurrences, and accident conditions.

With respect to operational descriptions and functions of radiation monitoring equipment, confirm the types and locations of annunciations and alarms and actions each type of instrumentation initiates. Confirm that once tripped by an alarm setpoint, the instrumentation system properly initiates and completes the expected action, such as providing local and remote audio and visual warnings, and, if so equipped, terminating or diverting a release or process stream to appropriate systems.

Confirm that the SAR indicates that an appropriate number of survey instruments will be available for all facility radiation monitoring functions and types of radiation surveys to be performed (e.g., Geiger-Mueller survey instruments for contamination surveys, release of equipment and tools from controlled areas, personnel “frisking;” ionization chambers used in external radiation exposure rate surveys; neutron detectors used to determine neutron flux or dose rates).

In supporting the implementation of surveys requiring sample collection, confirm that sampling and analytical equipment will be provided to collect and analyze the spectrum of expected samples, including gases, water vapors, water, wastes (e.g., dry, solid, and wet), wipes or smears, filters and absorption media, bioassays, and environmental media (e.g., soil, sediment, air, water, and biota, as described in the radiological environmental monitoring program). RG 4.1 provides supporting details for assessing compliance.

The guidance in Section 10A.5.2.5 above lists criteria to consider in this evaluation, as applicable, in addition to the criteria presented here and in Section 10A.4.4.2.

10A.5.4.3 *Policies and Procedures*

Review the applicant's plans and procedures to ensure that provisions have been made for the following:

- controlling, storing, securing, and moving radioactive materials on site, including radioactive wastes, contaminated equipment and tools, and calibration sources and standards the health physics program uses

- physical and administrative measures aimed at ensuring that occupational doses are ALARA and are within administrative limits and NRC requirements and criteria

- radiation monitoring equipment calibration and maintenance, including systems used at fixed monitoring locations, portable radiation survey equipment, fixed and portable air sampling equipment, liquid and gaseous effluent monitoring (process and release points), analytical laboratory equipment (operational samples and bioassays), and personal dosimetry devices, including documentation of National Voluntary Laboratory Accreditation Program certification when using third-party commercial dosimetry services

- personal protective equipment maintenance, inspection, and issuance and qualification and testing of fitness for the use of respiratory equipment

- records of waste management activities, including compilation of inventories of radioactive materials by physical and chemical forms, quantities (volumes or weight), radioactivity levels (according to radionuclide distributions and concentrations present in such wastes, materials, or calibration sources), and disposition (onsite or offsite storage, processing by waste brokers, shipped for offsite disposal, equipment sent out for refurbishment, or returned to manufacturer for disposition)

- retention of records for personnel dosimetry results, bioassays, radiation surveys, personnel qualification and training, personal qualification of respiratory fitness, data on radioactive sources and standards (National Institute of Standards and Technology traceable primary and secondary) used in implementation of health physics program, instrument and sampling equipment calibration methods and results, and data on radiological events that would support the planning and decommissioning of the facility, whenever initiated

<u>*Measurement Methods and Analyses*</u>

Review the applicant's methods to convert raw instrumentation readings into meaningful radiological results to use in assessing radioactivity levels, concentrations, exposure rates, and doses to confirm compliance with the criteria identified in this chapter and in the regulatory requirements. These methods may include reliance on the use of easy-to-detect surrogate radionuclides to identify the presence and determine the concentrations of hard-to-detect radionuclides. The methods may also include radiological determinations using gross beta-gamma or alpha concentrations to infer the concentrations of specific radionuclides. Ensure that the selected methods are appropriate for the analyses for which they will be used and that they are based on sound principles.

Equipment Calibration and Maintenance

Review the program descriptions for calibrating and maintaining survey equipment, area radiation monitors, continuous airborne monitors, effluent monitors, and laboratory equipment. Consider descriptions of instrumentation calibration methods and procedures in confirming instrumentation response characteristics, sensitivity levels and detection limits, and detection ranges for facility-derived radionuclides expected during normal operation, anticipated occurrences, and accident conditions. Compare the types, levels, energy spectra, and, for radioactive materials, concentrations described as the design basis of the facility to the methods described for specifying the types and ranges of radiation monitoring instrumentation. When two or more radiation-monitoring systems are used for routine operations or accident monitoring in a single system (e.g., area radiation monitoring or an effluent release point), ensure that the SAR describes the differences in instrumentation response characteristics over their overlapping operational ranges and expected radionuclide distributions and concentrations. Confirm that the calibration and maintenance methods and program are adequate and appropriate to ensure that monitoring and laboratory equipment will perform properly for the characteristics of the radiation and radioactive materials they are used to detect, measure, and analyze.

10A.6 Evaluation Findings

The NRC reviewer should prepare evaluation findings upon satisfaction of compliance with the regulatory requirements in Section 10A.4 of this SRP. Such a review includes coordination with other reviewers to make determinations on aspects such as radiation exposure rates, doses, and releases of airborne radioactive materials. If the documentation submitted with the application fully supports positive findings for each of the regulatory requirements, the statements of finding should be similar to the following, as applicable:

F10A.1 The SAR includes adequately detailed descriptions of the [DSF designation] SSCs' design and operation characteristics, including design criteria and design bases for the radiation protection evaluation and the radioactive materials to be stored at the facility, in compliance with 10 CFR 72.24(b), 10 CFR 72.24(c), 10 CFR 72.24(l), 10 CFR 72.120(a), 10 CFR 72.120(b), and 10 CFR 72.120(c). The SAR also includes evaluations of the performance of the facility's SSCs important to safety with respect to radiation protection, in compliance with 10 CFR 72.24(d).

F10A.2 The SAR includes descriptions that establish the owner controlled area and the controlled area boundary for the [DSF designation] in accordance with 10 CFR 72.106(a). The descriptions show the boundary meets the minimum distance requirements in 10 CFR 72.106(b). The SAR also describes effective and appropriate arrangements to adequately protect public health and safety and adequately control traffic on public access facilities (e.g., highways, railroads, or waterways) that traverse the controlled area in compliance with 10 CFR 72.106(c).

F10A.3 The design and operating procedures of the [DSF designation] provide acceptable means for controlling and limiting occupational radiation exposures within the limits given in 10 CFR Part 20 and for meeting the objective of maintaining exposures to meet ALARA objectives, in compliance with 10 CFR 72.24(e).

F10A.4 The SAR provides reasonable assurance that the activities authorized by the license can be conducted without endangering the health and safety of the public and that the operations procedures are adequate to protect health and minimize danger to life or property in compliance with 10 CFR 72.40(a)(5) and 10 CFR 72.40(a)(13).

F10A.5 [If appropriate] The proposed [DSF designation] is to [be on the same site as/near other, specify] nuclear facilities, [identify]. The cumulative effects of the combined operations of these facilities will not constitute an unreasonable risk to the health and safety of the public, in compliance with 10 CFR 72.122(e).

F10A.6 The SAR provides analyses showing that the cumulative effects of the combined operations of these facilities will be within the dose limits given in 10 CFR 72.104(a). These analyses include both direct radiation and effluent releases from the [DSF designation] to the general environment during normal operations and anticipated occurrences. The SAR also includes appropriate and adequate operational restrictions and limits to meet the limits in 10 CFR 72.104(a) and ALARA objectives in compliance with 10 CFR 72.104(b) and 10 CFR 72.104(c).

F10A.7 The SAR provides analyses of the doses from accident conditions at the facility in accordance with 10 CFR 72.24(m), and these analyses show these doses will not exceed the limits in 10 CFR 72.106(b).

F10A.8 The SAR provides analyses that show that the doses to members of the public will not exceed the limits in 10 CFR Part 20.

F10A.9 The SAR provides adequate evaluations that show the effects of the proposed site and facility, including effects due to operation and releases under normal and accident conditions on the regional environment and populations in the region in accordance with 10 CFR 72.100.

F10A.10 The SAR describes adequate measures that will preclude transport of radioactive materials to the environment through an aquifer over which the facility is located that serves as a major water resource in accordance with 10 CFR 72.122(b)(4).

F10A.11 The design of the [DSF designation] provides suitable shielding for radiation protection and confinement of radioactive materials under normal, off-normal (that is, anticipated occurrences), and accident conditions, in compliance with 10 CFR 72.128(a)(2) and 10 CFR 72.128(3). This includes ventilation systems and off-gas systems, continuous monitoring capability for the storage confinement systems, and HLW and reactor-related GTCC waste packaging that allows handling and retrievability without releases or exposures in excess of regulatory limits in accordance with 10 CFR 72.122(h)(3~5).

F10A.12 The facility design and operations include adequate means for controlling personnel exposures and for controlling and monitoring effluents and direct radiation, in compliance with 10 CFR 72.126.

F10A.13 The facility operations include programs, such as the health physics program, environmental monitoring program, and other pertinent programs that are needed to ensure compliance with the requirements in 10 CFR Part 20 and 10 CFR Part 72. These programs include the necessary elements to perform their intended functions, including the policies for and management commitments to the programs and their objectives.

F10A.14 The proposed license technical specifications include those items necessary to ensure adequate radiation protection in the design, fabrication, construction, and operation of the DSF SSCs in accordance with the requirements in 10 CFR 72.44(c) and to meet the requirements in 10 CFR 72.44(d).

F10A.15 The facility design and operations will, to the extent practicable, minimize contamination of the facility and the environment and generation of radioactive wastes in accordance with 10 CFR 20.1406(a) and 10 CFR 20.1406(c).

The reviewer should provide a summary statement similar to the following:

> The staff finds, with reasonable assurance, that the radiation protection design and program for the [DSF designation] meet the requirements in 10 CFR Part 20 and 10 CFR Part 72 and that the applicable design and acceptance criteria have been satisfied. The staff also finds, with reasonable assurance, that the facility design, operations, and programs are adequate to ensure compliance with the regulatory dose limits and ALARA requirements in 10 CFR Part 20 and 10 CFR Part 72 for personnel and the public. The evaluation of the radiation protection program, facility design features, ALARA objectives, and health physics program provide reasonable assurance that the [DSF designation] will allow safe storage of SNF, HLW [applies to MRS only], and reactor-related GTCC waste. The staff reached this finding based on a review that considered applicable NRC regulations and regulatory guides, codes and standards, accepted health physics practices, the statements and representations contained in the SAR, and the staff's confirmatory analyses.

10A.7 References

10 CFR Part 19, "Notices, Instructions and Reports to Workers: Inspection and Investigations."

10 CFR Part 20, "Standards for Protection Against Radiation."

10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities."

10 CFR Part 52, "Licenses, Certifications, and Approvals for Nuclear Power Plants."

10 CFR Part 72, "Licensing Requirements for the Independent Storage of Spent Nuclear Fuel, High-Level Radioactive Waste, and Reactor-Related Greater Than Class C Waste."

40 CFR Part 191, “Environmental Radiation Protection Standards for Management and Disposal of Spent Nuclear Fuel, High-Level and Transuranic Radioactive Wastes.”
Subpart A - Environmental Standards for Management and Storage

American National Standards Institute (ANSI)/American Nuclear Society (ANS) 6.4.2, “Specification for Radiation Shielding Materials.”

ANSI/ANS-Health Physics Society Standards Committee-6.8.1, “Location and Design Criteria for Area Radiation Monitoring Systems for Light Water Nuclear Reactors.”

ANSI/Health Physics Society (HPS) N13.1, “Sampling and Monitoring Releases of Airborne Radioactive Substances from the Stacks and Ducts of Nuclear Facilities.”

ANSI/HPS N13.6, “Practice for Occupational Radiation Exposure Record Systems.”

ANSI/HPS N13.30, “Performance Criteria for Radiobioassay.”

ANSI/HPS N13.32, “Performance Testing of Extremity Dosimeters.”

ANSI/HPS N13.41, “Criteria for Performing Multiple Dosimetry.”

ANSI/HPS N13.42, “Internal Dosimetry for Mixed Fission and Activation Products.”

American Society for Testing and Materials (ASTM) E1167, “Standard Guide for Radiation Protection Program for Decommissioning Operations.”

ASTM E1168, “Standard Guide for Radiological Protection Training for Nuclear Facility Workers.”

International Commission on Radiological Protection (ICRP) Publication 27, “Problems Involved in Developing an Index of Harm,” *Annals of the ICRP*, Vol. 1, Issue 4, 1977.

ICRP Publication 55, “Optimization and Decision-Making in Radiological Protection,” *Annals of the ICRP*, Vol. 20, Issue 1, 1990.

National Radiological Protection Board, “MARC-1,” December 1981. Modules of MARC-1 used to calculate the effects of an atmospheric release:

> Hill M.D., J.R. Simmonds, and J.A. Jones, “NRPB methodology for assessing the radiological consequences of accidental releases of radionuclides to atmosphere - MARC-1,” Chilton, NRPB-R224 (1988) (London HMSO)
>
> Jones J.A. and D. Charles, “AD-MARC: The atmospheric dispersion module in the methodology for assessing the radiological consequences of accidental releases,” Chilton, NRPB-M72, (1982)
>
> Charles D., M.J. Crick, T.P. Fell, and J.R. Greenhalgh, “DOSE-MARC: The dosimetric module in the methodology for assessing the radiological consequences of accidental releases,” Chilton NRPB-M74 (1982).
>
> Clarke, R.H. and G.N. Kelly, “MARC, The NRPB methodology for assessing radiological consequences of accidental releases of activity,” NRPB-R127.

Hemming, C.R., D. Charles, D.J. Alpert, R.M. Ostmeyer, “Comparison of the MARC and CRAC 2 Programs for Assessing the Radiological Consequences of Accidental Releases of Radioactive Material,” NRPB-R149 (1983).

Jones, J.A. and D. Charles, “AD-MARC: The Atmospheric Dispersion Model in the Methodology for Assessing the Radiological Consequences in Accidental Releases,” NRPB-M72. National Radiological Protection Board, Chilton, U.K

National Council on Radiation Protection and Measurements (NCRP) Report No. 57, “Instrumentation and Monitoring Methods for Radiation Protection,” 1978.

NCRP Report No. 59, “Operational Radiation Safety -Training,” 1978.

NCRP Report No. 71, “Operational Radiation Safety Training,” 1983.

NCRP Report No. 87, “Use of Bioassay Procedures for Assessment of Internal Radionuclide Deposition,” 1987.

NCRP Report No. 112, “Calibration of Survey Instruments Used in Radiation Protection for the Assessment of Ionizing Radiation Fields and Radioactive Surface Contamination,” 1991.

NCRP Report No. 116, “Limitation of Exposure to Ionizing Radiation,” 1993.

NCRP Report No. 127, “Operational Radiation Safety Program,” June 1998.

NCRP Report No. 134, “Operational Radiation Safety Training,” 2000.

NCRP Report No. 169, “Design of Effective Radiological Effluent Monitoring and Environmental Surveillance Programs,” 2010.

National Safety Council, “Accident Prevention Manual: Engineering and Technology,” 14th edition, 2015.

NRCDose, “Code System for Evaluating Routine Radioactive Effluents from Nuclear Power Plants with Windows Interface,” Version 2.3.20, Tape list C00684, PC586 14, Radiation Safety Information Computational Center, U.S. Department of Energy, Oak Ridge National Laboratory.

NUREG-0800, “Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition.”

NUREG-1736, “Consolidated Guidance: 10 CFR Part 20—Standards for Protection Against Radiation.”

Regulatory Guide 1.8, “Qualification and Training of Personnel for Nuclear Power Plants.”

Regulatory Guide 1.13, “Spent Fuel Storage Facility Design Basis.”

Regulatory Guide 1.21, “Measuring, Evaluating, and Reporting Radioactive Material in Liquid and Gaseous Effluents and Solid Waste.”

Regulatory Guide 1.33, “Quality Assurance Program Requirements (Operations).”

Regulatory Guide 1.109, “Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I.”

Regulatory Guide 1.140, “Design, Inspection, and Testing Criteria for Air Filtration and Adsorption Units of Normal Atmosphere Cleanup Systems in Light-Water-Cooled Nuclear Power Plants.”

Regulatory Guide 1.143, “Design Guidance for Radioactive Waste Management Systems, Structures, and Components Installed in Light-Water-Cooled Nuclear Power Plants.”

Regulatory Guide 4.1, “Radiological Environmental Monitoring for Nuclear Power Plants.”

Regulatory Guide 4.15, “Quality Assurance for Radiological Monitoring Programs (Inception through Normal Operations to License Termination)—Effluent Streams and the Environment.”

Regulatory Guide 4.20, “Constraint on Releases of Airborne Radioactive Materials to the Environment for Licensees other than Power Reactors.”

Regulatory Guide 4.21, “Minimization of Contamination and Radioactive Waste Generation: Life-Cycle Planning.”

Regulatory Guide 8.2, “Administrative Practices in Radiation Surveys and Monitoring.”

Regulatory Guide 8.4, “Personnel Monitoring Device—Direct-Reading Pocket Dosimeters.”

Regulatory Guide 8.8, “Information Relevant to Ensuring that Occupational Radiation Exposures at Nuclear Power Stations Will Be as Low Is Reasonably Achievable.”

Regulatory Guide 8.9, “Acceptable Concepts, Models, Equations, and Assumptions for a Bioassay Program.”

Regulatory Guide 8.10, “Operating Philosophy for Maintaining Occupational Radiation Exposures as Low as Reasonably Achievable.”

Regulatory Guide 8.13, “Instruction Concerning Prenatal Radiation Exposure.”

Regulatory Guide 8.15, “Acceptable Programs for Respiratory Protection.”

Regulatory Guide 8.25, “Air Sampling in the Workplace.”

Regulatory Guide 8.26, “Applications of Bioassay for Fission and Activation Products.”

Regulatory Guide 8.27, “Radiation Protection Training for Personnel at Light-Water-Cooled Nuclear Power Plants.”

Regulatory Guide 8.28, “Audible-Alarm Dosimeters.”

Regulatory Guide 8.29, “Instruction Concerning Risks from Occupational Radiation Exposure.”

Regulatory Guide 8.34, “Monitoring Criteria and Methods to Calculate Occupational Radiation Doses.”

Regulatory Guide 8.38, “Control of Access to High and Very High Radiation Areas of Nuclear Power Plants.”

U.S. Nuclear Regulatory Commission, “Criticality Control of Fuel Within Dry Storage Casks or Transportation Packages in a Spent Fuel Pool,” *Federal Register*, Vol. 71, No. 221, November 16, 2006, pp 66648–66657.

10B RADIATION PROTECTION EVALUATION FOR DRY STORAGE SYSTEMS (CoC)

10B.1 Review Objective

The objective of the U.S. Nuclear Regulatory Commission's (NRC's) radiation protection evaluation is to (1) determine that the proposed spent nuclear fuel (SNF) dry storage system (DSS) complies with the applicable regulatory requirements for radiation protection and (2) ensure that the DSS design and operations include reasonable consideration of, and facilitate licensees' compliance with, the requirements that licensees who use the DSS must meet.

For the purposes of this standard review plan (SRP) chapter, radiation protection refers to design and operational elements that are relied upon to limit radiation exposures from normal operations, anticipated occurrences (that is, off-normal conditions), and accidents and natural phenomenon events (collectively referred to as accident conditions or design-basis accidents (DBAs)). This includes those design features that may have a different primary function but are nonetheless credited or considered in the applicant's radiation protection evaluation.

10B.2 Applicability

This chapter applies to the review of applications for certificates of compliance (CoCs) for DSSs. As such, the chapter title is denoted with **(CoC)** to signify that the scope of this chapter applies only to DSSs.

10B.3 Areas of Review

The areas of review include means and methods used to protect workers and members of the public, DSS design features, DSS storage configurations, dose assessments and dose assessment methods, and operational elements and procedures.

This chapter addresses the following areas of review:

- radiation protection design features
- occupational exposures
- exposures at or beyond the controlled area boundary
  - normal operations and anticipated occurrences
  - accidents and natural phenomenon events
- as low as is reasonably achievable (ALARA) design
  - design considerations
  - procedures and engineering controls

10B.4 Regulatory Requirements and Acceptance Criteria

This section summarizes those parts of Title 10 of the *Code of Federal Regulations* (10 CFR) Part 72, "Licensing Requirements for the Independent Storage of Spent Nuclear Fuel, High-Level Radioactive Waste and Reactor-Related Greater Than Class C Waste," that are applicable to the review areas this chapter addresses. This section also includes specific 10 CFR Part 72 and 10 CFR Part 20, "Standards for Protection Against Radiation," requirements that, while not applicable to CoC applicants (i.e., they only apply to licensees), the reviewer should consider in the review. This is because the radiation protection review may include elements needed to assist the general licensee in meeting these regulatory requirements and to ensure the DSS design and operations include reasonable consideration of and facilitate the licensee's compliance with these requirements. The NRC reviewer should refer to the exact language in the applicable regulations. Table 10B-1 matches these regulatory requirements to the areas of review covered in this chapter. However, Table 10B-1 does not represent an exhaustive listing of regulations that may need consideration. Thus, the reviewer should confirm that all applicable regulations are identified and appropriately addressed in the safety analysis report (SAR).

In general, the radiation protection evaluation seeks to ensure that the proposed design fulfills the following acceptance criteria:

- The SAR demonstrates that the DSS includes shielding and confinement that are sufficient to meet the requirements in 10 CFR 72.104, "Criteria for radioactive materials in effluents and direct radiation from an ISFSI or MRS," and 10 CFR 72.106, "Controlled area of an ISFSI or MRS," including the dose limits, in compliance with 10 CFR 72.236(d).

- Dose rates, design features, and operations for the DSS are consistent with and demonstrate appropriate consideration for ALARA principles and objectives.

- The DSS design includes features that facilitate decontamination to the extent practicable in meeting the requirements in 10 CFR 72.236(i) in minimizing radioactive contamination.

The SAR should address these acceptance criteria. The acceptance criteria are organized according to the areas of review specified in Section 10B.3 above. The reviewer should consider the applicability and implementation of NRC and industry guidance against that presented in the SAR. The radiation protection review also requires coordination with the shielding (SRP Chapter 6, "Shielding Evaluation"), confinement (SRP Chapter 9, "Confinement Evaluation"), operating procedures (SRP Chapter 11, "Operation Procedures and Systems Evaluation"), accident analysis (SRP Chapter 16, "Accident Analysis Evaluation"), and technical specifications (SRP Chapter 17, "Technical Specifications Evaluation") reviews.

In general, the acceptance criteria listed in the SAR should adopt by reference appropriate NRC guidance or alternatively cite relevant and appropriate industry codes and standards. The SAR should identify and justify alternative approaches used to demonstrate compliance with applicable NRC guidance and industry codes and standards.

This guidance recognizes that applicants have options on how to demonstrate compliance with the NRC regulations and NRC guidance (e.g., rely only on NRC guidance or use alternative methods). With respect to the implementation of NRC guidance, the SAR should identify whether the NRC guidance has been adopted in whole or in part. The SAR should identify any differences between this SRP chapter and design features, analytical techniques, exposure and dose assessment codes, and procedural measures proposed for the DSS and discuss how the proposed alternatives provide acceptable methods of complying with regulations. In any case, the SAR should provide sufficient information and data for the staff to conduct an independent evaluation in confirming compliance with regulatory requirements and SRP acceptance criteria. The reviewer will confirm that the applicant has adequately addressed these considerations in the SAR.

If there are multiple versions of a guidance document, such as a regulatory guide or an industry standard, the applicant should note which version of the guidance document has been adopted in the SAR, whether it is the most current revision, and the basis for using the selected version. In the case of an industry standard, the applicant should consider what, if any, staff position exists with respect to acceptability of the standard and different revisions of the standard as part of that selection. As a result, the reviewer will identify the guidance documents the applicant used and assess whether the version of each document the applicant adopted is adequate for demonstrating compliance with NRC requirements.

Table 10B-1 Relationship of Regulations and Areas of Review

Areas of Review	10 CFR Part 72 Regulations			
	72.104[A]	72.106(b)[A]	72.126[B]	72.236
Radiation Protection Design Features	(a)(b)(c)	●	(a)(1)(2)(4)(5)(6), (d)	(b)(d)(g)(i)
Occupational Exposures			(a)(1)(2)(4)(5)(6)	(b)(g)(i)
Exposures at or Beyond the Controlled Area Boundary	(a)(c)	●	(d)	(b)(d)(g)(i)
ALARA	(b)		(a)(1)(2)(4)(5)(6), (d)	(b)(d)(i)

Areas of Review	10 CFR Part 20 Regulations[B]		
	20.1101	20.1201(a)	20.1301(a)(b)
Radiation Protection Design Features	(b)(d)		
Occupational Exposures	(b)	●	
Exposures at or Beyond the Controlled Area Boundary	(b)(d)		●
ALARA	(b)(d)		

A This requirement applies to CoCs and CoC applications through the requirement in 10 CFR 72.236(d).
B While not directly applicable to CoCs, DSS design should facilitate general licensee compliance with these requirements.

10B.4.1 Radiation Protection Design Features

The SAR should describe the DSS design features relied on for shielding and confinement to meet radiation protection criteria and requirements. The descriptions of these features should be in the respective SAR chapters and address the information described in Chapters 6 and 9 of this SRP.

The radiation protection chapter of the SAR should include any additional information that is needed beyond what is in the shielding and confinement chapters to demonstrate that together these features are sufficient to ensure, or enable, compliance with regulatory requirements for radiation protection and ALARA objectives. This information should include an evaluation of the use of DSS features during operations. The descriptions should include any additional or supplemental shielding that is needed for radiation protection whether for occupational workers or the public. This would include features such as shield berms relied on in the dose analyses and shielding used in the DSS preparation area and with the transfer equipment to enable personnel to perform operations safely around the DSS. The SAR should also include descriptions of how the DSS design features facilitate decontamination (see 10 CFR 72.236(i)) as well as inspection and servicing in consideration of regulations such as 10 CFR 72.126(a)(5).

The SAR should also describe any operational controls and limits that are necessary to use the DSS and ensure compliance with regulatory requirements and ALARA objectives. While establishing operational controls and limits for ensuring compliance with requirements such as 10 CFR 72.104(b), 10 CFR 72.104(c), 10 CFR 72.126(a), and 10 CFR 72.126(d) is mainly the responsibility of the general licensee using the DSS, it may be necessary or appropriate for some controls and limits to be included as part of the DSS design. The inclusion of these controls and limits may be needed to support DSS evaluations for 10 CFR 72.236(d) as well as to ensure that the design facilitates the licensee's compliance with other requirements. These controls and limits may include surface dose rate limits and measurement requirements for prominent DSS design features that are important for doses to personnel or the public. These controls and limits may also include limits and measurement requirements for (removable) contamination. The SAR descriptions should also include the bases for the proposed controls and limits. Some of these may be included as conditions of the CoC, in the technical specifications (see Chapter 17 of this SRP). Note that controls and limits include appropriate specifications of the allowable contents for storage in the DSS, which specifications are adequately supported by and used in the analysis in the SAR and defined in the technical specifications. SAR descriptions of operating procedures should also include or reflect implementation of these controls and limits as well as efforts to minimize contamination and ensure doses are ALARA.

10B.4.2 Occupational Exposures

The SAR should include estimates of doses to workers as a result of DSS operations. These estimates should include individual and collective doses. Separate estimates may be provided for different activities or operational sequences. For example, the applicant should provide dose estimates for the sequence beginning with DSS loading in the SNF pool and ending with placement of the DSS at the ISFSI pad, the reverse sequence of operations, and for the conduct of maintenance and surveillance activities. Since a general licensee may load multiple DSSs in a year and licensee personnel are likely to perform other functions at the general licensee's 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities," or 10 CFR Part 52, "Licenses, Certifications, and Approvals for Nuclear Power Plants," facility, in addition to dry storage operations, the applicant's dose estimates should indicate that individual doses to workers will be well below the dose limits specified in 10 CFR 20.1201(a). Collective doses should be consistent with the objectives of a well-structured ALARA program. Additional justification of acceptability of the DSS design may be necessary for systems with high occupational dose estimates.

Typically, the applicant only needs to estimate doses for normal DSS operations. However, for DSSs for which operational conditions and dose rates to which personnel may be exposed in responding to anticipated occurrences may differ significantly from normal operations, the SAR should also include dose estimates for actions performed to recover from the anticipated occurrences. The SAR operating procedures chapter should also include a description of such recovery actions for such kinds of anticipated occurrences.

The SAR should include information sufficient to support evaluation of compliance with these criteria. This information should include dose rates for representative points on and near the surfaces of the DSS structures, systems, and components (SSCs) for the different DSS configurations encountered during DSS operations. The information should also include dose rates at appropriate distances from the DSS and from a DSS array, as appropriate, for personnel performing surveillance and maintenance activities. The SAR should also describe the number of personnel involved in operations and the duration of the operations. Dose rate locations should be consistent with the locations of all personnel involved in the DSS operations, with dose rates for these locations being derived in the SAR's shielding evaluation chapter. Dose estimates should be broken down by work tasks, or appropriate groupings of tasks (e.g., a group of tasks involve personnel at the same locations at and around the DSS surfaces for the same DSS configuration). The SAR should describe the bases for all assumptions used in the analysis and the reasonableness of these assumptions.

10B.4.3 Exposures At or Beyond the Controlled Area Boundary

The SAR should include analyses of radiation exposures and doses to individuals at or beyond the controlled area boundary for normal operations, anticipated occurrences, and DBAs.

10B.4.3.1 Normal Operations and Anticipated Occurrences

The doses during normal operations and anticipated occurrences to any "real individual" located beyond the ISFSI controlled area may not exceed the values specified in 10 CFR 72.104(a). A real individual is defined as a person who lives, works, or engages in recreation or other activities close to the dry storage facility for a significant portion of the year. For the purposes of the 10 CFR 72.104(a) limits, the analysis excludes the occupational doses radiation workers receive while working.

For DSS CoC applications, responsibility for determining compliance with these limits ultimately rests with the general licensee because demonstration of compliance considers factors that are specific to the licensee's site (e.g., geometric arrangement of DSS arrays, distances to the controlled area boundary, information about public areas around the site, and maximum SNF quantity to be stored at the site). However, the CoC applicant is responsible for and must demonstrate that the DSS design complies with the requirements in 10 CFR 72.236 in accordance to 10 CFR 72.234(a). These requirements include that the DSS's shielding and confinement features are sufficient to meet 10 CFR 72.104, including dose limits (see 10 CFR 72.236(d)).[1] Section 10B.5.3.1 of this SRP describes acceptable ways to address this requirement. The analysis should address the contributions from direct radiation and any effluent releases from the DSS when the DSS has a specified, analyzed leak rate. Though, based on design requirements, radioactive materials are not expected to be released from DSSs, a DSS will have a specified, analyzed leak rate and effluent doses when it is not designed and tested to be

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[1] Note that the requirements in 10 CFR 72.236, "Specific requirements for spent fuel storage cask approval and fabrication," are the responsibility of the CoC applicant (Volume 64 of the Federal Register, page 56114 (64 FR 56114), October 15, 1999). Thus, the regulations require the DSS to be designed to meet 10 CFR 72.104 and 10 CFR 72.106 (according to 10 CFR 72.236(d)) and place that responsibility with the DSS designer (CoC applicant). This responsibility cannot be passed to the general licensee through 10 CFR 72.212, "Conditions of general license issued under § 72.210," or a 10 CFR Part 50 or 10 CFR Part 52 program. For canister-based systems, this applies to the transfer cask as well. It also applies regardless of the DSS's location; the requirements do not distinguish between a loaded DSS in a 10 CFR Part 50 or 10 CFR Part 52 SNF building or on the co-located ISFSI storage pad. This is consistent with the November 16, 2006, rulemaking's definition of the regulatory boundary between 10 CFR Part 72 and 10 CFR Part 50 for criticality safety (71 FR 66648).

either leak-tight or to meet the noncredible leakage criterion. The materials and confinement evaluation chapters of this SRP should include the details regarding the design criteria and testing related to the DSS leak rate. Also, the analysis should address doses from anticipated occurrences since the dose limits apply to the annual doses from both normal operations and anticipated occurrences.

10B.4.3.2 Accidents and Natural Phenomenon Events

The doses to any individual located on or beyond the nearest boundary of the controlled area from any DBA may not exceed the limits specified in 10 CFR 72.106(b). As is described in Section 10B.4.3.1 above, responsibility for determining compliance with the 10 CFR 72.106(b) limits ultimately rests with the general licensee. However, the applicant is responsible for and must demonstrate that the DSS design complies with the requirements in 10 CFR 72.236 in accordance to 10 CFR 72.234(a). These requirements include that the DSS's shielding and confinement features are sufficient to meet 10 CFR 72.106, including the dose limits (see 10 CFR 72.236(d)). Section 10B.5.3.2 of this SRP describes acceptable ways to address this requirement. The analysis should address the contributions from direct radiation and, when the DSS has a specified, analyzed leak rate for DBA conditions, any effluents from the DSS.

10B.4.4 As Low As Is Reasonably Achievable Design

The SAR should describe how the applicant has incorporated ALARA principles into the DSS design and operations to enable a general licensee using the DSS to ensure doses to workers and the public will be ALARA.

10B.4.4.1 Design Considerations

The applicant should demonstrate that ALARA principles have been incorporated into the DSS design to the extent practical. As part of this demonstration, the SAR should describe the bases for the selection and design of DSS features, including geometric and materials aspects, and include appropriate radiation protection, technological, and economic considerations, as applicable. The SAR should show that the applicant considered ALARA principles as part of the following design elements:

- geometric design (e.g., physical sizes of design features, surface features and shapes that minimize accumulation of contamination, features that minimize or simplify needed maintenance activities, labyrinthine inlet and outlet vents to reduce radiation streaming)

- arrangement of design features (e.g., placement of vent paths with respect to the SNF contents)

- materials design (e.g., type and density of concrete selected to minimize dose rates, application of corrosion- and abrasion-resistant coatings to prevent accumulation of contamination in surface pores of SSCs)

In these and any other appropriate aspects of the design, the applicant should consider how general licensees will need to operate the DSS. Such considerations include means to minimize necessary decontamination efforts, minimize generation of radioactive wastes, and minimize the time required for personnel to perform necessary operations (e.g., provide sufficient space to easily perform all expected operations). Considerations should also include actions necessary to recover from anticipated occurrences. For DSS designs where personnel performing recovery actions may be exposed to significantly higher dose rates as compared to normal operations, the applicant may need to provide further justification that such designs are adequate from an ALARA perspective as well as from a general radiation protection perspective. In the case of design changes (e.g., in an amendment), the applicant should describe how the design changes maintain or improve the DSS's effective implementation of ALARA principles. The SAR should, as appropriate and applicable, describe how the applicant has used its experience with past DSS designs to develop the proposed DSS and improve implementation of ALARA principles.

10B.4.4.2 Procedures and Engineering Controls

The SAR should describe plans and procedures that have been developed in accordance with applicable guidance of SRP Chapter 11. These plans and procedures should adequately demonstrate the implementation of ALARA principles. This includes describing, in the SAR's operating procedures chapter, the use of appropriate engineering controls or equipment that licensees should employ to maintain doses ALARA for DSS operations. The appropriate procedures should also include cautions and warnings regarding streaming paths or other potential radiological hazards (e.g., higher dose rates from radioactive material such as CRUD that is entrained in water being drained from the DSS) for operations that may involve such hazards. The sequencing of procedures should also reflect consideration of ALARA principles as well. The engineering controls and procedures described in the SAR should be founded upon sound engineering design criteria and radiation protection principles.

10B.5 <u>Review Procedures</u>

This section describes review procedures for evaluating DSS designs and descriptions of operations with regard to radiation protection requirements, doses to workers and to members of public, and implementation of ALARA principles in the designs and operations of DSSs. The radiation protection review includes evaluation of compliance with all regulatory requirements and acceptance criteria given in this SRP and other applicable NRC documents and accepted codes and standards. The reviewer should always assume that such a comprehensive scope of the review applies, even though it is not further detailed or repeated in this section. Figure 10B-1 shows the interrelationship between the radiation protection evaluation and the other areas of review described in this SRP. Based on its review, as described in the following sections, the reviewer should work with the technical specifications (SRP Chapter 17) reviewer to ensure that any CoC condition regarding preoperational testing includes testing of design features and procedures that are significant to radiation protection, as appropriate.

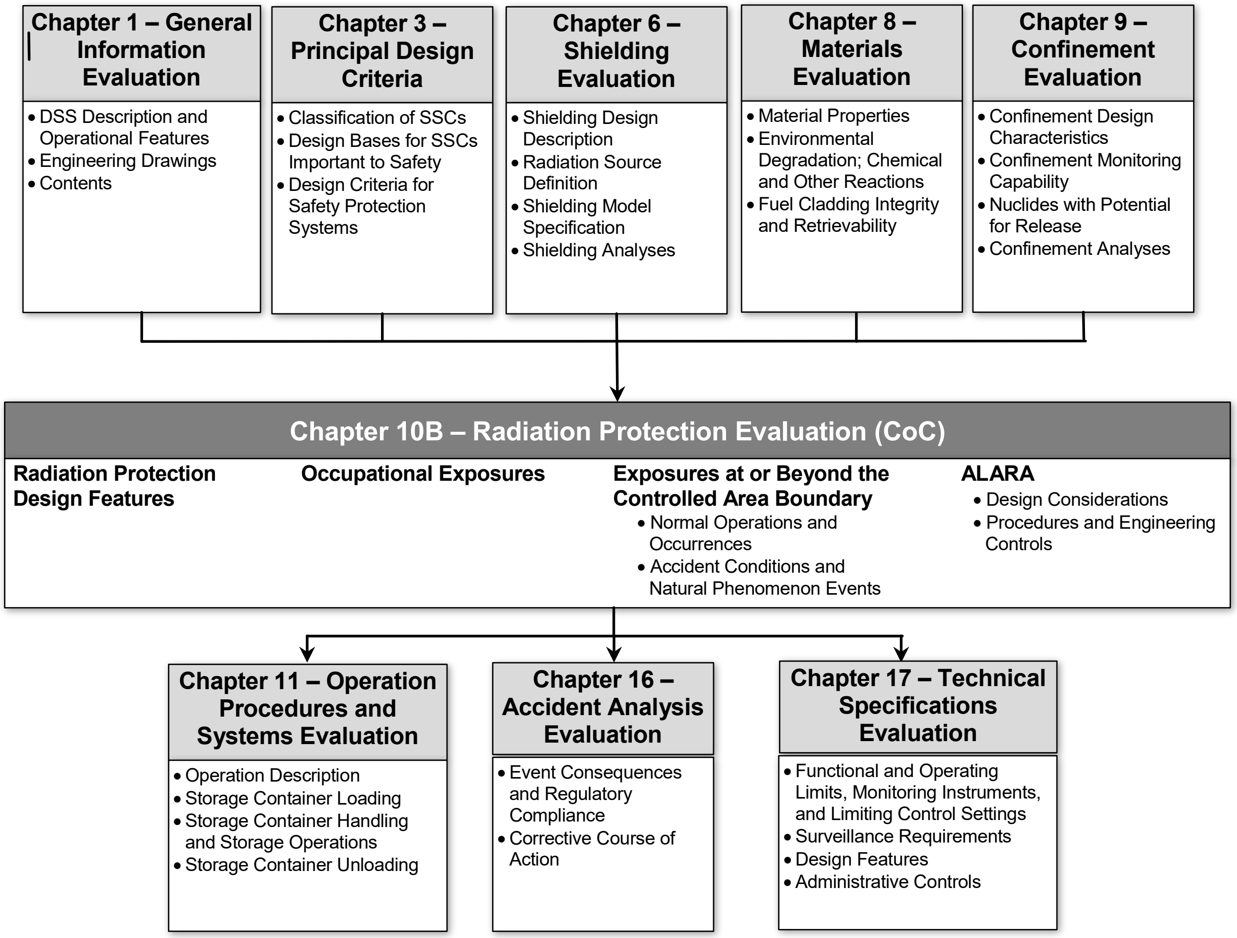

Figure 10B-1 Overview of Radiation Protection Evaluation

10B.5.1 Radiation Protection Design Features

Carefully review the general description and functional features of the DSS and the technical drawings presented in the SAR. In addition, review information on SSCs and design criteria as well as any additional details regarding radiation protection provided in the SAR. Based on this review, the staff should identify those DSS SSCs and features that are relevant to radiation protection. Since some of this information may be in the shielding and confinement chapters of the SAR; coordinate this review with the reviewers of the SAR shielding and confinement chapters to (1) obtain a sufficient understanding of the relevant features and the evaluations of those features, (2) ensure the features are described and analyzed adequately to evaluate their overall effectiveness for radiation protection purposes for all applicable configurations and conditions, and (3) ensure that any identified inadequacies or inconsistencies are appropriately addressed.

Verify that the SAR demonstrates and confirms that the DSS design adequately meets the following criteria:

- The DSS design includes shielding and confinement features that are sufficient to meet the requirements in 10 CFR 72.104 and 10 CFR 72.106 (10 CFR 72.236(d)).

- The DSS features are designed to facilitate decontamination to the extent practicable in accordance with 10 CFR 72.236(i). This includes minimizing contamination or preventing accumulation of contamination.

- The DSS design includes adequate consideration of ALARA principles and is sufficient to facilitate compliance (by the general licensee) with applicable public and occupational exposure requirements of 10 CFR Part 20 and ALARA objectives.

Evaluation of the adequacy of radiation protection features necessarily includes consideration of dose rates and doses, in terms of both the public and workers. Sections 10B.5.2 and 10B.5.3 below address the evaluation of the doses. For purposes of this section, consider factors such as comparisons between the proposed DSS and DSSs that the NRC has already certified. While some allowance should be given for differences in the SNF contents and the design features, similarities in DSS dose rates, dose estimates for personnel, distances to meet 10 CFR 72.104(a) limits for sample DSS array sizes, and estimates of doses at the controlled area boundary for accident conditions can provide a good indicator about the adequacy of the DSS design in terms of radiation protection. If the values of the preceding items are significantly larger than those of currently certified systems, or if they seem to be large considering the proposed SNF contents compared with those of currently certified systems, consider whether the proposed design is sufficiently protective and seek further justification of the design's adequacy to protect personnel and the public.

For a DSS design that necessitates operations methods that are unusually different or are significant departures from those methods and descriptions that are common for certified systems in order to maintain personnel or public doses at reasonable levels, which are similar to those of the certified systems, consider seeking further justification regarding the design's adequacy. In some cases, design changes may be necessary so that the design will adequately protect personnel and the public.

Consider the regulatory limits and requirements in 10 CFR Part 20 and 10 CFR Part 72 beyond those directly applicable to CoCs to inform these kinds of evaluations. For these scenarios, consider the need for conditions in the CoC, in the technical specifications, regarding the DSS's design features or operational controls and methods, and coordinate the review with the technical specification reviewer (SRP Chapter 17). These conditions and technical specifications may include items such as clear descriptions of any extra shielding items as part of the DSS design and specifications (e.g., thicknesses), specifications of any remote operations equipment, requirements regarding use of these extra shield features and remote operations equipment, monitoring of operations and SSCs conditions, requirements for recovery actions for off-normal events, preoperational testing of any remote operations and equipment, limits on the duration of high dose-rate configurations, and added considerations for general licensees to address in their evaluations for using the DSS.[2] Also review the SAR chapter for operating procedures and coordinate with that reviewer (SRP Chapter 11) to evaluate the relevant aspects of the DSS design and operations that are covered in that chapter.

Evaluate the adequacy of ALARA considerations in the DSS design and operations. The regulatory guides cited in Section 10B.5.4 provide information that may be useful for this evaluation. Consider the physical design features together with the descriptions of DSS

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[2] The shielding design features are important for ensuring compliance with regulatory dose limits, including the limits in 10 CFR 72.104(a) and 10 CFR 72.106(b). See also Footnote 1 on page 10B-5.

operations in the SAR. For example, designs that require unique methods of operation in order to protect personnel and the public may not have adequately incorporated ALARA. While doses during normal operations may be acceptable, the need to implement these unique operational methods and the equipment to facilitate them could introduce scenarios, including off-normal events, that typically do not need to be evaluated or considered but that could, for these designs, have potentially significant dose consequences to personnel and members of the public. Such a situation may not be consistent with ALARA principles and should be carefully evaluated. Flag and be attentive to whenever the SAR identifies the need for methods of operations or use of specialized equipment that is significantly or unusually atypical compared with operations that are common for existing, certified DSSs.

Coordinate the review with the technical specification reviewer (SRP Chapter 17) to ensure that the CoC, including the technical specifications, adequately describes the DSS, including any important features that the analysis relies on for demonstrating compliance with regulatory limits. In addition to what is commonly considered part of the DSS, the CoC should include the description of (1) shield berms that support the results of dose analyses described in Section 10B.5.3; (2) significant shielding components required for personnel protection to enable personnel to perform operations on or around the DSS; and (3) parameters for ensuring that the shielding remains adequate for normal, off-normal, and accident conditions where normal activities (e.g., ISFSI expansion) could otherwise remove materials relied on for shielding.

Evaluate the applicant's proposed operational controls to ensure that exposures and doses to workers and members of the public are controlled, within NRC dose limits, and consistent with ALARA objectives. Ensure that the technical specifications include any necessary operational controls and limits as discussed in Section 10B.4.1 above. Also ensure that the SAR descriptions of operating procedures include implementation of needed operational controls and limits. Coordinate this effort with the operating procedures reviewer (SRP Chapter 11). Also coordinate with the shielding, confinement, and conduct of operations reviewers to ensure that the SAR includes acceptance tests and maintenance programs that are sufficient to ensure that the DSS will perform as designed in terms of radiation protection for the duration of its certified life and use.

10B.5.2 Occupational Exposures

Ensure that the SAR includes occupational dose estimates for DSS operations as well as descriptions of the methods and parameters (e.g., inputs, assumptions) used to develop those estimates. Verify that the estimates and descriptions adequately address the operations sequences and considerations identified in Section 10B.4.2.

Review the SAR chapters that describe systems operations. Coordinate with the reviewers of these chapters to ensure that all descriptions are consistent with and adequately detailed to support the occupational dose estimates and the bases for those estimates. These descriptions should include the necessary actions and cautions to ensure operations are conducted in a manner that is consistent with the bases of the occupational dose estimates. In addition, ensure that dose estimates for periodic or routine maintenance as well as surveillance activities include reasonable assumptions regarding dose contributions from adjacent DSSs or the DSS array depending on the storage configuration and the expected personnel actions and positions the applicant described.

Coordinate with the shielding reviewer (SRP Chapter 6) to ensure that the SAR includes dose rates at adequate locations and numbers of locations around the DSS for all of the different configurations that arise during normal operations and anticipated occurrences for systems where such evaluations are needed (see Section 10B.4.2). Ensure that the SAR includes dose rates on and near DSS surfaces where personnel will be performing operations on or close to the DSS. Verify that the SAR also includes dose rates at appropriate distances from the DSS for operations that involve personnel positioned at such distances from the DSS.

Verify that the applicant presents sufficient information in describing the methods, bases, and assumptions used for the dose assessment. This information should include the rationale used to justify the bases for various exposure times, personnel locations relative to the DSSs (including hot spots), number of personnel required for each operation, and appropriate gamma and neutron dose rates at all assumed locations. Verify that calculated doses and applied assumptions are consistent with these estimates and SAR descriptions of operating procedures. Also verify the reasonableness of these assumptions. Comparisons with other NRC-certified systems may provide useful insights for this evaluation. Confirm that the SAR provides dose estimates that consider all configurations that will occur during operations. The dose estimates should be refined adequately enough to appropriately capture the differences in personnel positions (e.g., personnel positioned at the canister lid vs. standing near the DSS base) and changes in DSS configurations (e.g., water present in the DSS canister vs. drained out of the canister). Regarding method, it may simply involve multiplying the dose rates (calculated in the shielding analysis) for different locations for each operation by the number of individuals and the time duration associated with that operation and summing the totals for each operation over each operation sequence. If a more complex method is chosen that involves computer codes (beyond that used for the dose rates in the shielding analysis), consult Chapter 6, Sections 6.4.4.1, “Computer Codes,” and 6.5.4.1, “Computer Codes,” of this SRP for applicable review guidance.

Determine the reasonableness of the estimated doses for the different operations. To do this, consider the doses estimated for other NRC-certified systems in line with the considerations described above in Section 10B.5.1 as well as consideration of implications for a licensee’s ability to meet 10 CFR Part 20 exposure and dose limits and requirements when using the DSS. In evaluating the estimated doses, keep in mind that a general licensee using the DSS will conduct DSS operations under the licensee's radiation protection program, which includes personnel dose monitoring to ensure compliance with 10 CFR Part 20 limits and any licensee administrative limits. Regulatory Guide 8.34, “Monitoring Criteria and Methods To Calculate Occupational Radiation Doses,” which was developed to implement revisions to 10 CFR Part 20, contains information to consider in evaluating the acceptability of the applicant’s occupational exposure evaluation and monitoring recommendations.

10B5.3 Exposures at or Beyond the Controlled Area Boundary

As described in Section 10B.4.3, demonstration of compliance with the requirements in 10 CFR 72.104 and 10 CFR 72.106 is ultimately the responsibility of the general licensee that uses the DSS because that demonstration considers factors that are specific to the licensee’s site. However, 10 CFR 72.234(a) requires the CoC holder and applicant for a certificate to ensure the DSS design complies with 10 CFR 72.236, which includes sufficient shielding and confinement features to meet the requirements in 10 CFR 72.104 and 10 CFR 72.106 (see 10 CFR 72.236(d)). Confirm that the applicant has provided analyses that are adequate to demonstrate that the DSS is sufficiently designed to meet these requirements in accordance with 10 CFR 72.236(d).[3] These analyses also facilitate the general licensee’s evaluations for its site’s compliance with 10 CFR 72.104 and 10 CFR 72.106.

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[3] See Footnote 1 on page 10B-5.

Ensure that the SAR adequately describes the methods, assumptions, and bases used in the analyses and that these methods, assumptions, and bases are appropriate for the analyses and the conditions being evaluated. The analyses should include the contributions both from direct radiation and from any effluents for the DSS, including, as appropriate, surface contamination (at the levels allowed by the technical specifications). Since the evaluation for surface contamination would be similar to that for effluents, coordinate with the confinement reviewer to evaluate any contamination contributions, which technical specifications limits should make negligible for offsite doses. The analyses should also account for all appropriate exposure pathways for effluents. The SAR should include dose calculations for a single DSS and a theoretical array of DSSs, assuming design-basis source terms and full-time occupancy. Other aspects of the analyses are described in the sections that follow for the different conditions of operation. It should be noted that, because of the design requirements for DSSs, direct radiation is expected to be the major contributor to exposures and doses. Also, because of these design requirements, radioactive materials are not expected to be released from DSSs during normal, off-normal, or accident conditions. However, as noted elsewhere, the analyses will include effluent dose contributions for DSSs with specified, analyzed leak rates (i.e., the DSS is not designed and tested to be leak-tight or meet the noncredible leakage criteria).

Coordinate with the shielding, confinement, and accident evaluation reviewers to obtain the dose and dose rate results from those evaluations and to ensure that they are sufficient to support the evaluations in this part of the review. Also consider the results of staff's confirmatory analyses for those reviews in evaluating compliance with requirements in this review, particularly if those confirmatory analyses indicate significant differences in comparison with the applicant's analyses. If the confinement analysis only provides effluent dose results at 100 meters (328 feet) or only for a single DSS, coordinate with the confinement reviewer to evaluate how effluents may contribute to doses at additional distances and for DSS arrays in order to determine if additional analyses are needed in the SAR to address these scenarios. Also coordinate with other reviewers, such as the structural and shielding reviewers, to understand the impacts of different events (anticipated occurrences and accident conditions) to ensure the SAR analysis adequately addresses the dose impacts for all relevant events for all relevant DSS operating configurations.

10B.5.3.1 *Normal Conditions and Anticipated Occurrences*

Ensure that the applicant's evaluations for these conditions include analyses for a single DSS and for a sample array of DSSs on an ISFSI pad. These analyses have typically only considered the DSS in its storage configuration on the ISFSI pad. The NRC has accepted this practice for most systems because the other operation configurations (e.g., loading and transfer) are of very short duration so that dose contributions beyond the controlled area boundary are expected to be very small to negligible. Also, the limits in 10 CFR 72.104(a) include doses from both normal conditions and anticipated occurrences. For DSSs in their storage configuration, anticipated occurrences typically do not affect the DSSs. So, the dose rates and doses are not affected by anticipated occurrences. Anticipated occurrences have also not typically affected DSS dose rates for other operation configurations, and dose rates, though high in some cases, have not necessitated consideration of anticipated occurrences for those operations either.

The guidance in this section is generally based on these practices. However, be aware of instances where the impacts of anticipated occurrences or these other operation configurations should be considered in the analysis. These instances include DSSs where design features, dose rates, or operations methods for these other operations configurations are significantly different from those that are typical of certified DSSs. An example would be a DSS with significantly higher dose rates in a particular operation configuration that, if the DSS remained in this configuration for a reasonable duration (resulting from either normal conditions or an anticipated occurrence), could have a nonnegligible effect on doses beyond the controlled area. In such cases, ensure that the SAR adequately considers the impacts of these operations in demonstrating the adequacy of the DSS's shielding for meeting the limits in 10 CFR 72.104(a).[4] Also ensure that dose analyses for normal conditions and anticipated occurrences adequately consider variations in the storage configuration(s) that may occur for DSS designs where normal, though likely infrequent, actions may alter the DSS's shielding in its storage configuration. Such actions include construction activities associated with expansion of an operating ISFSI that removes material relied on for DSS shielding or otherwise exposes this material when it would not otherwise be exposed. In cases when their consideration is necessary, the applicant's analysis should include the dose impacts from the bounding anticipated occurrence, assuming a reasonable event duration that includes the necessary time to recover from the event. Coordinate with the shielding, structural, and other relevant reviewers, as appropriate, in evaluating these scenarios.

Ensure that the SAR includes analyses for a single DSS and for a hypothetical array of DSSs. The hypothetical array should consist of at least 20 DSSs in a 2 x 10 array configuration. The SAR analyses should include dose or dose rate versus distance curves to facilitate site-specific evaluations for general licensees. The NRC has accepted the use of dose (rate) versus distance curves for a single DSS and a DSS array as a means to demonstrate the DSS design is sufficient to meet the 10 CFR 72.104(a) limits.

Ensure that the applicant's analyses assume appropriately bounding conditions. Such conditions include design-basis source terms, no intervening shielding between the DSS or DSS array and location of the dose receptor, and full-time yearly occupancy at each analyzed distance. Ensure the distances for which doses are provided include the doses at 100 meters (328 feet) from the single DSS and the DSS array since 100 meters is the minimum distance to the nearest ISFSI controlled area boundary, as noted in 10 CFR 72.106(b). Analyses that only include distances that are larger than 100 meters may be acceptable if the longer distance is made a condition of use in the CoC. In addition, ensure that the SAR determines the degree to which dose rates under normal conditions could change for other identified operating conditions and anticipated occurrences. For the array analyses, the applicant may account for shielding among DSSs, but should provide sufficient details regarding how that is done. Ensure that the analyses appropriately address this inter-DSS shielding when credited. If the analyses credited some engineered feature (e.g., a shield wall or berm), then ensure the CoC includes this feature as part of the DSS design and that the SAR includes appropriate descriptions and technical drawings for this feature.

Identify the minimum distance that the applicant's analysis indicates is required to meet the dose regulation in 10 CFR 72.104(a) for both the single DSS and the array of DSSs. Past applications have shown this distance to be typically within 200 meters (656 feet) for a single DSS. Consider the minimum distance for the single DSS and the DSS array and evaluate whether the distance indicates that the DSS includes shielding and confinement features that are sufficient to meet the dose regulation in 10 CFR 72.104. Compare how the distances for this DSS compare with those of certified DSSs, accounting for the relevant considerations identified in Section 10B.5.1. Also consider, to the extent practical, typical general licensee site features. These may include typical distances to owner-controlled area boundaries, typical distances to locations of the public around the licensee's site (e.g., residences, recreation areas), and residency times. Consider that a general licensee may store as much SNF as it generates at its 10 CFR Part 50 or 10 CFR Part 52 reactor facility. For DSSs for which significant distances are needed to meet the

[4] See Footnote 1 on page 10B-5.

10 CFR 72.104(a) limits, accounting for the considerations listed above, the SAR may need to include additional information to justify how the DSS design is sufficient to enable a general licensee to reasonably meet those limits. Typically, the DSS design and the SAR analyses do not include engineered features such as shield berms. However, the general licensee may choose to use such features at its ISFSI, which is permissible, to mitigate doses to individuals near the site. Thus, verify that the CoC includes a condition to ensure that a general licensee that chooses to use these features will adequately manage these features. This condition should be similar to the following:

> Supplemental Shielding: In cases where engineered features (e.g., earthen berms, shield walls) are used to ensure that the requirements of 10 CFR 72.104(a) are met, such features are to be considered important to safety, evaluated to determine the applicable quality assurance category, and appropriately evaluated under 10 CFR 72.212(b).

Be aware that the general licensee that uses the DSS must perform a written evaluation to demonstrate that the requirements in 10 CFR 72.104 are met, as required in 10 CFR 72.212(b)(5)(iii), for any real individual (as defined in Section 10B.4.3.1) located beyond the controlled area of the licensee's site. The licensee may use information provided in the DSS SAR as well as site-specific information in performing this evaluation. Although evaluations the general licensee performs are not submitted to the NRC for approval, they are subject to NRC inspection and should be recorded and maintained by the general licensee.

The CoC should include a condition that ensures that a general licensee using the DSS will implement the necessary monitoring to ensure compliance with the limits in 10 CFR 72.104(a) during the operations of its ISFSI. By virtue of being a general licensee, the licensee already has programs including its radiological protection program and environmental monitoring program to meet its 10 CFR Part 50 or 10 CFR Part 52 license requirements, which may also be applied toward monitoring for compliance with 10 CFR 72.104. Thus, a CoC condition that directs the general licensee to update its radiological protection and environmental monitoring programs to incorporate its SNF operations and to address compliance with 10 CFR 72.104(a) limits may be sufficient. Consult the CoC conditions (likely in the CoC technical specifications) of currently certified DSSs in developing an appropriate monitoring condition for the DSS being reviewed. The monitoring program should address both direct radiation and effluents, as appropriate, as well as have operating procedures to identify and reevaluate potential increases in exposures to individuals located beyond the site's controlled area.

As noted in Section 10B.5.1, the reviewer should consider the need to include operational controls and limits in the CoC conditions (in the technical specifications). As noted in Section 10B.4.1, controls and limits to be considered include dose rate limits and associated measurements. The determination for the need for these limits is discussed in those sections and is contingent upon a variety of factors. These factors include, but are not limited to, DSS dose rates for the different operations and configurations, the nature of the DSS design, potential dose impacts of changes to that design, and the need for such limits to ensure continued compliance with 10 CFR 72.236(d). Such dose rate limits should be derived from the applicant's dose rate and dose analyses for normal and off-normal conditions. The limits should be developed for appropriate DSS configurations and should be compared against the maximum measured dose rates. The dose rate limit condition should include an appropriate number of measurements at appropriate DSS surface locations to adequately ensure compliance with the limits.

10B.5.3.2 *Accident Conditions and Natural Phenomenon Events*

Ensure that doses are calculated for all relevant accident conditions for all relevant DSS configurations. Thus, the SAR should include accident condition doses for DSS configurations in addition to the final storage configuration, such as the loaded transfer cask for canister-based DSSs. Refer to the accident analysis evaluation chapter (SRP Chapter 16), which includes a list of accidents that are typically analyzed. Consider whether the DSS design may be susceptible to other types of accidents for which doses should be analyzed. Also consider whether the design introduces the possibility of other, atypical, configurations for which such accidents should be analyzed. For example, accident dose analyses should include doses for accidents that occur when material relied on for shielding may be removed or exposed under some normal, though temporary, operating conditions when that material otherwise would not be removed or exposed. Typically, accident condition doses are analyzed for only a single DSS; however, consider whether there may be scenarios for the DSS design when an accident could affect the entire array. Ensure that the applicant analyzed doses for a DSS array in such a case.

Ensure that the applicant's analyses assume appropriate bounding conditions. These conditions include assumptions such as no intervening shielding between the DSS and the individual at the analyzed distance(s), full-time occupancy at the analyzed distance(s), and a reasonably bounding duration of the event. The event duration should include the time to recover from the event and its impacts. A typically assumed duration is 30 days. The sum of the doses from each applicable contributing factor (direct radiation, effluents, surface contamination) should not exceed the limits in 10 CFR 72.106(b).

Verify that the applicant calculated doses at 100 meters (328 feet) from the DSS, the minimum distance allowed in regulations from the ISFSI storage and handling facilities to the nearest boundary of the controlled area. Applicants may calculate doses or dose rates at a discrete distance(s) or may develop a curve that shows dose versus distance. Ensure that the analysis shows that the DSS will not exceed the 10 CFR 72.106(b) dose limits at 100 meters from the DSS. For those DSSs for which a greater distance or engineered design features, such as berms, are needed to meet these dose limits, ensure that the CoC includes this distance or the engineered feature(s) as a condition of DSS use. Ensure also that the CoC includes the engineered feature(s) in the description of the DSS and that the SAR includes adequate descriptions of the engineered feature(s), including technical drawings.

10B.5.4 As Low As Is Reasonably Achievable Design

Evaluate the DSS with regard to implementation of ALARA, both in the physical design features and descriptions of operating procedures. To perform this evaluation, consider the DSS's design features and the operating procedures described in the SAR. Ensure that the DSS design and operations address ALARA for both occupational and public exposures. Also consider how ALARA is incorporated into other NRC-approved DSSs, as appropriate, and the state of technology to inform the evaluation of the proposed DSS. Consider consulting available regulatory guides (e.g., Regulatory Guide 8.8, "Information Relevant to Ensuring that Occupational Radiation Exposures at Nuclear Power Stations Will Be As Low As Is Reasonably Achievable") that contain information regarding ALARA that may also be useful to inform the review. Reference to these regulatory guides may be useful to inform the evaluation of the DSS's adequacy for meeting and for facilitating licensees' compliance with regulatory requirements and ALARA objectives.

Consider actions that general licensees would be reasonably expected to take to implement ALARA during DSS operations. Among others, these actions include the use of lead blankets, other types of temporary shielding, preoperation planning, prestaging of equipment, and preassembly of equipment and components. The DSS design and operations should ensure that these and other reasonable actions will be sufficient to ensure licensee compliance with ALARA requirements.

10B.5.4.1 *Design Considerations*

Coordinate with the shielding and confinement reviewers to ensure that the DSS design features adequately incorporate ALARA principles to the extent practical as described in Section 10B.4.4.1 of this SRP. These principles should also be reflected in any design criteria the applicant described in the SAR to support the materials, geometric, and dimensional aspects of the DSS design. Ensure that the applicant has adequately justified that the proposed DSS design incorporates ALARA to the extent practical and necessary, or reasonable. Credit for incorporation of ALARA should be limited to features that are part of the DSS design that are adequately described in the SAR, including the technical drawings and schematics. Designs that necessitate operations that are atypical of approved DSSs in order to maintain reasonable personnel doses for normal operations or that could result in potentially significant exposures to personnel involved in actions to recover from an off-normal condition may not meet ALARA objectives. There may also be implications for public doses and ALARA considerations for those doses. In such cases, seek further justification from the applicant regarding the adequacy of ALARA incorporation into the DSS design and consider whether any CoC condition is needed in this regard.

10B.5.4.2 *Procedures and Engineering Controls*

Confirm that the descriptions of proposed DSS operations adequately demonstrate implementation of ALARA principles into operating procedures as described in Section 10B.4.4.2 above. Confirm that the description of operating procedures includes necessary controls and actions to minimize dose and minimize contamination. Identify operations where elevated dose rates may occur, and ensure that operations descriptions include proper cautions and warnings and, where appropriate, personnel actions. Examples of these operations include those that necessitate personnel to perform actions near streaming paths or where radioactive particulates may be entrained in water draining from SSCs of the DSS. Some of the actions may include recommendations to use temporary, portable shielding such as lead blankets, recommendations on positioning of personnel involved in the procedures, or wetting the DSS surfaces exposed to SNF pool water to minimize adherence of radioactive particles (contamination control). Ensure that the proposed procedures and controls include those that are necessary for the DSS to meet 10 CFR 72.236(d) and support licensee compliance with 10 CFR 72.104(b), 10 CFR 72.104(c), and 10 CFR 72.126(a) as well as relevant 10 CFR Part 20 requirements.

10B.6 Evaluation Findings

The NRC reviewer should prepare evaluation findings upon satisfaction of compliance with the regulatory requirements in Section 10B.4, as determined through a review conducted in accordance with the information in this SRP chapter. Such a review includes coordination with other reviewers as described in the guidance in this chapter. If the documentation submitted with the application fully supports positive findings for each of the regulatory requirements, the statements of finding should be similar to the following:

F10B.1 The [DSS designation, *specify*] provides radiation shielding and confinement features that are sufficient to meet the requirements of 10 CFR 72.104 and 10 CFR 72.106, in accordance with 10 CFR 72.236(d).

F10B.2 The design and operating procedures of the [DSS designation, *specify*] provide acceptable means for controlling and limiting occupational radiation exposures within the limits given in 10 CFR Part 20 and for meeting the ALARA objective with respect to exposures, consistent with 10 CFR 20.1101(b).

F10B.3 The [DSS designation, *specify*] is adequately designed to facilitate decontamination in accordance with 10 CFR 72.236(i) and includes, to the extent practical and appropriate, adequate features, operating procedures, and controls that are designed to assist a general licensee to meet the radiological protection criteria in 10 CFR 72.126(a) and 10 CFR 72.126(d).

The reviewer should provide a summary statement similar to the following:

> The staff finds, with reasonable assurance, that the radiation protection design of the [DSS designation, *specify*] is in compliance with 10 CFR Part 72 and that the applicable design and acceptance criteria have been satisfied. The evaluation of the radiation protection design provides reasonable assurance that the [DSS designation, *specify*] will allow safe storage of SNF. The staff reached this finding based on a review that considered applicable NRC regulations and regulatory guides, codes and standards, accepted health physics practices, statements and representations contained in the SAR, and the staff's confirmatory analyses.

10B.7 References

10 CFR Part 20, "Standards for Protection Against Radiation."

10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities."

10 CFR Part 52, "Licenses, Certifications, and Approvals for Nuclear Power Plants."

10 CFR Part 72, "Licensing Requirements for the Independent Storage of Spent Nuclear Fuel, High-Level Radioactive Waste, and Reactor-Related Greater Than Class C Waste."

Regulatory Guide 8.8, "Information Relevant to Ensuring that Occupational Radiation Exposures at Nuclear Power Stations Will Be as Low as Is Reasonably Achievable."

Regulatory Guide 8.34, "Monitoring Criteria and Methods To Calculate Occupational Radiation Doses."

U.S. Nuclear Regulatory Commission (NRC), “Expand Applicability of Part 72 to Holders of, and Applicants for, Certificates of Compliance,” *Federal Register*, Vol. 64, No. 199, October 15, 1999, pp. 56114–56128.

NRC, “Criticality Control of Fuel Within Dry Storage Casks or Transportation Packages in a Spent Fuel Pool,” *Federal Register*, Vol. 71, No. 221, November 16, 2006, pp 66648–66657.

11 OPERATION PROCEDURES AND SYSTEMS EVALUATION

11.1 Review Objective

The objective of the U.S. Nuclear Regulatory Commission's (NRC's) review of operating procedures and operations systems for a dry storage system (DSS) or dry storage facility (DSF) is to evaluate associated applications for clarity and completeness to verify the following:

- The description of the applications provides sufficient detail to ensure that reviewers can understand the operations and their effects on the design evaluations.
- The DSS or DSF operations are consistent with the design bases for which the DSS or DSF was designed and analyzed in the other chapters of the safety analysis report (SAR) and this standard review plan (SRP).
- The DSS or DSF operations incorporate and are consistent with the conditions of the certificate of compliance (CoC) or a specific license, including the proposed technical specifications.

11.2 Applicability

This chapter applies to the review of applications for specific licenses for an independent spent fuel storage installation (ISFSI) or a monitored retrievable storage installation (MRS), categorized as a DSF. This chapter also applies to the review of applications for a CoC of a DSS for use at a general license facility. Sections or tables of this chapter that apply only to a DSF specific license application (for an ISFSI and MRS) are identified with "**(SL)**" in the heading. Sections or tables that apply only to DSS CoC applications have "**(CoC)**" in the heading. A subsection without an identifier applies to both types of application.

11.3 Areas of Review

This chapter addresses the following areas of review:

- operation description
- storage container loading
- storage container handling and storage operations
- storage container unloading
- repair and maintenance **(SL)**
- other operating systems **(SL)**
- operation support systems **(SL)**
- control room and control area **(SL)**
- analytical sampling **(SL)**
- fire and explosion protection **(SL)**

11.4 Regulatory Requirements and Acceptance Criteria

This section summarizes those parts of Title 10 of the *Code of Federal Regulations* (10 CFR) Part 72, "Licensing Requirements for the Independent Storage of Spent Nuclear Fuel, High-Level Radioactive Waste, and Reactor-Related Greater than Class C Waste," that are relevant to the review areas this chapter addresses. The NRC staff reviewer should refer to the exact language in the regulations. Tables 11-1a and 11-1b match the relevant regulatory requirements to the areas of review covered in this chapter for specific license and CoC reviews, respectively. In addition, requirements in 10 CFR Part 20, “Standards for Protection Against Radiation,” also apply to reviews for specific license applications. The reviewer should coordinate with the radiation protection reviewer (Chapter 10A of this SRP) to determine the applicable 10 CFR Part 20 requirements.

Table 11-1a Relationship of Regulations and Areas of Review for a DSF (SL)

Area of Review	10 CFR Part 72 Regulations									
	72.24 (b)(e)(f)(l)	72.40 (a)(5)(13)	72.44 (c)(1)(2)(3)(5), (d)(1)(2)	72.104	72.106(b)	72.122 (f)(h)(i)(j)(k)(l)	72.124	72.126 (a)(2)(3)(4), (b)(c)(d)	72.128 (a)(1)(2)	72.150
Operation Description	•	•	•	•	•	•	•	•	•	•
Storage Container Loading	•			•	•		•	•	•	•
Storage Container Handling and Storage Operations	•			•	•		•	•	•	•
Storage Container Unloading	•			•	•		•	•	•	•
Repair and Maintenance								•	•	
Other Operating Systems	•		•	•	•	•	•	•	•	•
Operation Support Systems	•		•			•		•		
Control Room and Control Area						•				
Analytical Sampling	•		•	•		•		•		
Fire and Explosion Protection						•				

Table 11-1b Relationship of Regulations and Areas of Review for a DSS (CoC)

Area of Review	10 CFR Part 72 Regulations				
	72.104(b)	72.106(b)	72.124	72.234(f)	72.236(c)(d)(f)(g)(h)(i)(l)(m)
Operation Description	•	•	•	•	•
Storage Container Loading	•	•		•	•
Storage Container Handling and Storage Operations	•	•		•	•
Storage Container Unloading	•	•		•	•

The following sections describe acceptance criteria, which are designed to ensure that the applicant fully describes the information on systems and significant operating sequences and actions in the SAR chapters. A sufficient level of detail is needed for the reviewer to conclude that the DSS or DSF operations are consistent with the design bases, will adequately protect health and minimize danger to life or property, protect the fuel from significant damage or degradation, and provide for the safe performance of tasks and operations. The applicant should provide an adequate description of the functional systems operations and identify the proper functioning of each system in a manner that adequately supports the purposes described above for the operations procedures descriptions and the evaluations in the other chapters of the SAR.

11.4.1 Operation Description

Operation description relates to the overall storage functions and operation of the DSS or DSF. The description should identify and describe the sequences of operations, actions, and controls that are important to safety for spent nuclear fuel (SNF), high-level radioactive waste (HLW), and reactor-related greater-than-Class-C (GTCC) waste handling and storage, including loading and unloading operations, as applicable. Sufficient detail should be included to enable the reviewer to evaluate engineering and operational controls. The operation description also should include the principal design features, procedures, and special techniques associated with criticality prevention, chemical safety, operation shutdown modes, instrumentation, radiation protection, protection of radioactive contents from significant damage or degradation, and maintenance techniques. The description should be sufficiently detailed to provide for the safe performance of tasks and operations.

Major operating procedures should exist for the principal activities expected to occur during loading, storage preparation, dry storage, and unloading. Section 11.3 above describes the areas of review for the SAR operating procedure descriptions, as does Chapter 8 of Regulatory Guide (RG) 3.61, “Standard Format and Content for a Topical Safety Analysis Report for a Spent Fuel Dry Storage Cask,” relevant sections of RG 3.48, "Standard Format and Content for the Safety Analysis Report for an Independent Spent Fuel Storage Installation or Monitored Retrievable Storage Installation (Dry Storage), " and RG 3.62, “Standard Format and Content for the Safety Analysis Report for Onsite Storage of Spent Fuel Storage Casks.” The applicant should submit operating procedure descriptions as part of the application to address the DSS or DSF design features and operations.

- Operating procedure descriptions should identify measures to control processes and mitigate potential hazards that may be present during planned, normal operations. Section 11.5 of this chapter discusses the review of previously identified processes and potential hazards.

- Operating procedure descriptions should ensure conformance with the applicable operating controls and limits described in the DSS CoC or DSF license conditions and technical specifications provided in the SAR chapter on technical specifications and operating controls and limits.

- Operating procedure descriptions should reflect planning to ensure that operations will fulfill the following acceptance criteria:

  - Occupational radiation exposures will be maintained as low as is reasonably achievable (ALARA) and within the limits of 10 CFR Part 20.

- Effective measures will be taken to preclude potential unplanned and uncontrolled releases of radioactive materials and otherwise minimize potential releases under normal operations conditions.

- Doses for members of the public will be maintained within the limits of 10 CFR Part 20 and 10 CFR 72.104, “Criteria for radioactive materials in effluents and direct radiation from an ISFSI or MRS,” for normal operations, and 10 CFR 72.106, “Controlled area of an ISFSI or MRS,” for accident conditions.

In addition, the operating procedure descriptions should support and be consistent with the bases used to estimate radiation exposures and total doses as defined in the radiation protection review guidance in this SRP that applies to the particular application (Chapter 10A for specific license applications and Chapter 10B for CoC applications).

Operating procedure descriptions should include provisions for the following activities:

- testing, surveillance, and monitoring of the stored material and storage containers during storage and loading and unloading operations

- contingency actions triggered by inspections, checks, observations, instrument readings, and so forth; some of these may involve off-normal and accident conditions addressed in the chapter of the SAR on accident analyses

RG 3.61, RG 3.62, and RG 3.48 provide further detail on operating procedure descriptions.

11.4.2 Storage Container Loading

In addition to the acceptance criteria specified above for the operation description, there are additional acceptance criteria for storage container loading, as follows:

- The operating procedures descriptions should include provisions for loading of SNF, reactor-related GTCC waste, and HLW storage containers, as applicable.

- The operating procedure descriptions should facilitate reducing the amount of water vapor and oxidizing material within the storage container to an acceptable level in order to protect the SNF cladding against degradation that might otherwise lead to gross ruptures.

- Operating procedures should specify methods for placing damaged fuel in a damaged-fuel can before loading into a SNF storage container, as applicable.

11.4.3 Storage Container Handling and Storage Operations

The regulatory requirements in 10 CFR 72.24, “Contents of application: Technical information,” **(SL)**, 10 CFR 72.124, “Criteria for nuclear criticality safety,” 10 CFR 72.128, “Criteria for spent fuel, high-level radioactive waste, and other radioactive waste storage and handling,” **(SL)**, and 10 CFR 72.236, “Specific requirements for spent fuel storage cask approval and fabrication,” **(CoC)** address the information to be included in a SAR for handling storage containers loaded with SNF, HLW, and reactor-related GTCC waste, as applicable to review of CoC and specific license applications. The SAR should include information as described in RG 3.61,RG 3.62, and RG 3.48 on handling systems for SNF and reactor-related GTCC waste (and HLW if for a MRS), as applicable. The descriptions of the SNF, HLW, or reactor-related GTCC waste handling systems and operations should be clear. The applicant should address the functions of transfer from transportation vehicles, receipt inspection, and initial decontamination. The applicant should include a statement indicating whether the NRC reviewed these operations or the systems used to perform these operations, as applicable, as part of a licensing action under 10 CFR Part 50, “Domestic Licensing of Production and Utilization Facilities,” or 10 CFR Part 52, “Licenses, Certifications, and Approvals for Nuclear Power Plants.” The SAR should include a description of the transfer facility and its use, including its use during the stages of operation of the DSS or DSF. The descriptions should include consideration of potential off-gassing (including hydrogen generation).

11.4.4 Storage Container Unloading

In addition to the acceptance criteria specified above for the operation description, the descriptions should include provisions for unloading SNF, reactor-related GTCC waste, and HLW, as applicable. The operating procedures should facilitate ready retrieval of the contents stored in the DSS or DSF storage containers.

11.4.5 Repair and Maintenance (SL)

The SAR should contain a description of the repair and maintenance facilities and describe the operation of these facilities, including provision for contamination control and occupational exposure minimization. Chapter 12, “Conduct of Operations Evaluation,” of this SRP provides useful guidance for the evaluation of maintenance operations. Note that the maintenance and use of a transportation package for shipping radioactive material is governed only by the requirements in 10 CFR Part 71, “Packaging and Transportation of Radioactive Material.” Thus, any repair and maintenance operations involving transport packaging must be done in accordance with those requirements (see 10 CFR 71.17, “General License: NRC-Approved Package,” and 10 CFR Part 71, Subpart H, “Quality Assurance”), including requirements in the transport CoC for the package.

11.4.6 Other Operating Systems (SL)

The scope of the review of this section includes all operating systems important to safety that are not covered in the preceding sections, except for the acceptance criteria for instrumentation and control (I&C), which are in Section 11.4.7, “Operation Support Systems,” and acceptance criteria for analytical sampling, which are in Section 11.4.9, “Analytical Sampling,” of this SRP. The applicant should prepare the SAR sections on auxiliary systems and other operating systems that are important to safety as described in RG 3.62 and RG 3.48 and noted in the narrative descriptions or flowcharts describing the operation of the ISFSI or MRS. The regulations in 10 CFR 72.24 require that the SAR include clear descriptions of the systems and system equipment and controls used to assure safety. These items should be consistent with other parts of the SAR. Examples of other operating systems that may be classified as important to safety include ventilation and off-gas systems, electrical systems, air supply systems, steam supply and distribution systems, water supply systems, fire protection systems, air sampling systems, decontamination systems, and systems related to chemical hazards. This information should include an analysis or other acceptable basis for determining that operation support systems important to safety remain operational under off-normal and accident conditions.

11.4.7 Operation Support Systems (SL)

The SAR should include information on operation support systems, primarily I&C systems and component spares or alternative equipment, as required by 10 CFR 72.122, "Overall requirements". These items should be as described in RG 3.62 and RG 3.48. This information should include an analysis or other acceptable basis for determination that operation support systems important to safety remain operational under normal, off-normal, and accident conditions. The SAR should include clear descriptions of the operation support systems and descriptions of equipment and controls used to assure safety and that are consistent with other parts of the SAR.

11.4.8 Control Room and Control Area (SL)

The SAR should include a discussion of how a control room and control areas permit the installation to operate safely under normal, off-normal, and accident conditions (10 CFR 72.122(j) The SAR should include clear descriptions of the control room and control area. In addition, 10 CFR 72.122(j) requires that a control room or control area, if appropriate for the DSF design, must be designed to permit occupancy and actions to be taken to monitor the safety of the DSF under normal conditions and to provide safe control of the DSF under off-normal or accident conditions.

The NRC has accepted omission of a control room for ISFSI or MRS operations that have not involved use of a powered cooling system for material in storage.

11.4.9 Analytical Sampling (SL)

The SAR should include a discussion of the provisions for obtaining samples for analyses necessary to ensure that the ISFSI or MRS is operating within prescribed limits. The SAR should include a description of the facilities and equipment available to perform the required tests.

11.4.10 Fire and Explosion Protection (SL)

The regulations in 10 CFR 72.122(c) require the DSS or DSF structures, systems, and components (SSCs) important to safety and their contents to have adequate protection against fires and explosions to ensure the SSCs continue to effectively perform their safety functions under credible or design-basis fire and explosion conditions.

The regulations in 10 CFR 72.122(c) require the applicant to take measures for fire prevention, fire detection, fire suppression, and fire containment for the protection of the DSS or DSF SSCs important to safety and their contents. The SAR should include a discussion of these capabilities.

11.5 Review Procedures

The focus of this review is twofold: (1) the operations descriptions of the DSS or DSF and (2) the functions needed for operability and the compatibility of proposed systems with performance of those functions. The NRC does not review and approve detailed procedures (e.g., standard operating procedures). However, the NRC does review, and the SAR should include, operations procedure descriptions that are sufficient in detail to illustrate the important actions and processes to be done and demonstrate that the operations will be conducted in a manner that (1) is consistent with the CoC or license conditions, as appropriate, and technical specifications, (2) ensures that the DSS or DSF operations will be consistent with the design bases and fulfill safety functions, and (3) includes adequate consideration of radiation protection and ALARA for the public and personnel. Also, the review of the descriptions of functions of the proposed systems constitutes another principal basis for assessing that the DSS or DSF will be operated in the manner described above. Reviews in other SRP sections determine quantitative functional performance for functional and structural performances.

Figure 11-1 shows the interrelationship between the operating procedures evaluation and the other areas of review described in this SRP.

An applicant's operating procedures are, in a significant way, how the applicant's conduct of operations is implemented. Therefore, the review should be coordinated with the conduct of operations review (SRP Chapter 12) to ensure that there are no inconsistencies.

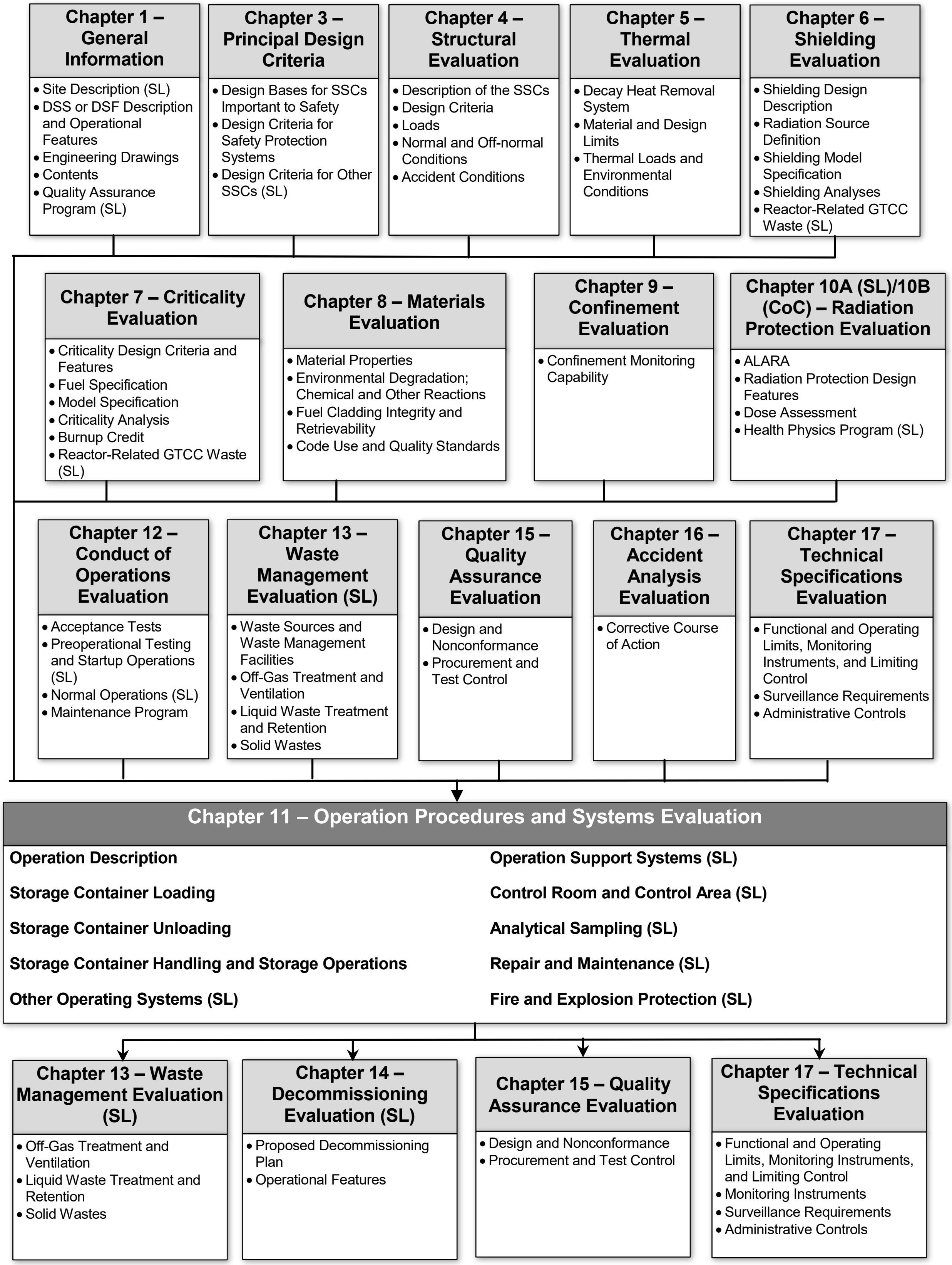

Figure 11-1 Overview of Operation Procedures and System Evaluation

11.5.1 Operation Description

Review the description of operation systems' functions for completeness. Compare the functions with descriptions included in other licensing documentation to confirm acceptability. For a specific license application, if a previously certified DSS design is used, check the functions described in the DSF SAR under review for compatibility with those functions that were included in the SAR for the certified DSS.

Review flowcharts and narrative descriptions of steps as provided on general operating functions. Ensure that the applicant has adequately described the appropriate operations, equipment involved, and personnel requirements.

Review the operating procedure sequences described in the SAR. Use the direct dose rate information in the chapter of the SAR on shielding to assess compliance with radiation protection requirements. Coordinate the evaluation of the operating procedure sequences with the shielding and radiation protection evaluations covered in Chapter 6, "Shielding Evaluation," and Chapter 10A (for DSFs) or 10B (for DSSs) of this SRP.

American National Standards Institute (ANSI)/American Nuclear Society (ANS) 57.9, "Design Criteria for an Independent Spent Fuel Storage Installation (Dry Storage Type)," applies to dry storage operating procedures. NUREG/CR-4775, "Guide for Preparing Operating Procedures for Shipping Packages," issued December 1988, provides guidance on preparing operating procedures for shipping packages. Although NUREG/CR-4775 specifically addresses 10 CFR Part 71, most of the guidance can be adapted for storage casks that are governed by 10 CFR Part 72. Therefore, become familiar with this information before initiating the operating procedures review.

The DSF applicant will, as the DSF licensee, develop detailed written procedures (e.g., standard operating procedures) that should be based on the operations descriptions in the SAR. For DSSs, the licensee that will use the DSS will develop the detailed procedures. These detailed procedures should be based on the operations descriptions in the DSS SAR operations chapter. The reviewer should ensure that the CoC contains a condition that makes this a requirement. In general, perform the following actions in the process of evaluating all of the operating procedure descriptions and operational sequences provided in the SAR:

- Verify that the proposed operating procedure descriptions incorporate and are compatible with the applicable operating limits and controls in the chapter of the SAR on technical specifications and operational controls and limits. Coordinate with the operating controls and limits review, as described in Chapter 17, "Technical Specifications Evaluation," of this SRP.

- Ensure that the proposed operating procedure descriptions properly consider the prevention of hydrogen gas generation from any cause. Prevention of hydrogen generation or adequate purging of hydrogen is essential during loading and unloading operations that involve seal welding, seal cutting, grinding, or other forms of hot work.

- Determine whether the descriptions include appropriate precautions to minimize occupational radiation exposures in accordance with ALARA principles and the limits given in 10 CFR Part 20, as required in 10 CFR 72.24(e), and consistent with the requirements in 10 CFR 72.126(a). Provisions may include the use of remotely controlled equipment, monitoring, and the use of portable shielding.

- Verify that the operating procedure descriptions include a general listing of the major tools and equipment needed to support loading, preparation for storage, storage, and unloading operations. Confirm that the descriptions address installation, use, and removal of the storage container and its contents, tools, and equipment. In addition, ensure that the descriptions address any specialized tools and equipment, such as lifting yokes, transporter equipment, welding and cutting equipment, and vacuum drying equipment, in sufficient detail to provide a clear understanding of their function(s). The use of any such equipment is subject to approval as part of the application review if that equipment is either classified as being important to safety or, though not important to safety, per the design bases, the equipment's failure could negatively impact fulfillment of a function that is important to safety. Ensure that the SAR identifies and describes such equipment in detail, identifies the performance characteristics of the equipment, and contains an evaluation the equipment's design.

11.5.2 Storage Container Loading

The operating procedure descriptions in the SAR should present the activities sequentially in the anticipated order of performance. Review the generic procedures in the SAR to ensure that they include appropriate key prerequisite, preparation, and receipt inspection activities to be accomplished before storage container loading. Verify that the SAR specifies the tests, inspections, verifications, and cleaning procedures required in preparation for storage container loading. In addition, where applicable, verify that the procedure descriptions include actions needed to ensure that any fluids such as shield water and primary coolants fill their respective cavities according to design specifications. In addition, verify that the procedure descriptions incorporate the applicable operating controls and limits described in the chapter of the SAR on technical specifications and operating controls and limits. These controls and limits include any dose rate and contamination measurements necessary to confirm compliance with the respective limits in the technical specifications.

11.5.2.1 *Specifications for Spent Nuclear Fuel, Reactor-Related Greater-Than-Class-C Waste, and High-Level Radioactive Waste*

Verify that the loading procedure description appropriately addresses the SNF specifications (e.g., burnup, cooling period, source terms, heat generation, cladding damage, associated nonfuel hardware) in the chapters of the SAR on principal design criteria and technical specifications and operation controls and limits. For storage containers relying on burnup credit, ensure that the loading procedure description includes verification that assemblies selected for loading meet the specifications for assembly operational history and the burnup credit loading curve. In addition, ensure that the loading procedure description includes performance of measurements to confirm assembly burnup values. For general license facilities and for specific license DSFs used to store SNF from a co-located 10 CFR Part 50 or 10 CFR Part 52 reactor facility's SNF pool, depending on the types and specifications of fuel assemblies stored in the reactor SNF pool, detailed site-specific procedures may be necessary to ensure that all fuel loaded in the storage container meets the fuel specifications for the storage container design. These detailed procedures can be evaluated only on a site-specific basis and will generally be evaluated through inspections rather than during the licensing review. However, check that the SAR indicates that such procedures may be necessary and describes the essential elements of the procedures.

(SL) For specific license DSFs that will also store reactor-related GTCC waste or MRS's that will store HLW, verify that the loading procedure description appropriately addresses the waste specifications and the acceptance criteria for storage at the facility that are described in the SAR's principle design criteria and technical specification and operation controls and limits chapters. For DSFs that receive SNF, reactor-related GTCC waste, or HLW from other locations (besides the 10 CFR Part 50 and 10 CFR Part 52 waste with which the DSF may be co-located), ensure that the operations descriptions include how the licensee will ensure the items received at the DSF meet the license specifications for storage at the DSF.

11.5.2.2 *Damaged Fuel*

Verify that the SAR includes appropriate measures for the loading of damaged fuel, if damaged fuel is included in the proposed storage container contents. Chapter 3, "Principal Design Criteria Evaluation," and Chapter 8, "Materials Evaluation," of this SRP provide criteria for the storage of damaged fuel. Use information in Section 8.5.15.1, "Spent Fuel Classification," of this SRP to identify the conditions that determine when SNF is to be classified as damaged fuel. Review Sections 8.5.15.1 and 8.5.15.3 of this SRP to determine the classification, documentation, and special confinement requirements for damaged fuel, and determine whether operating procedures address these requirements.

11.5.2.3 *Subcriticality Features*

Where applicable, verify that the procedure descriptions include the use of temporary or removable features important to criticality safety that may require installation during loading operations. Such items include fuel spacers and items (e.g., blocks) used to prevent loading of contents in selected SNF basket locations. The procedure descriptions should include installation, or verification of the installation, of these items before loading for storage containers that rely upon these features in the criticality analysis. Additionally, ensure that the procedure descriptions include verification, in accordance with technical specification requirements, of the minimum soluble boron level necessary for SNF loading into storage containers that require soluble boron to ensure subcriticality.

11.5.2.4 *ALARA*

Verify that the procedure descriptions incorporate ALARA principles and practices. These may include provisions to perform radiological surveys, establish exposure and contamination control measures, and use or install temporary shielding and inclusion of caution statements related to actions that could change radiological conditions.

11.5.2.5 *Offsite Release*

Where applicable, verify that the SAR describes methods to minimize offsite releases. Examples of these methods include, but are not limited to, decontamination of the storage containers, means for minimizing contamination of DSS or DSF SSCs, controls for processing of liquids and gases removed from the storage container during the draining and drying process, filtered ventilation, and temporary containments (tents). Ensure that the procedure descriptions also provide for minimizing the generation of radioactive waste.

11.5.2.6 *Draining and Drying*

Evaluate the descriptions related to methods for use in draining and drying the storage container for wet loading operations and, if applicable, HLW and reactor-related GTCC waste containers. Ensure that the SAR clearly describes the procedures for removing water vapor and oxidizing material to an acceptable level. Assess whether those procedures are appropriate.

The NRC staff has accepted vacuum drying methods comparable to those recommended in PNL-6365, "Evaluation of Cover Gas Impurities and Their Effects on the Dry Storage of LWR Spent Fuel," issued November 1987 (Knoll and Gilbert 1987). This report evaluates the effects of oxidizing impurities on the dry storage of light-water reactor (LWR) fuel and recommends limiting the maximum quantity of oxidizing gasses (such as oxygen, carbon dioxide,[1] and carbon monoxide) to a total of 1 gram-mole per cask. This corresponds to a concentration of 0.25 volume percent of the total gases for a 7.0-cubic-meter (about 247-cubic-foot) container gas volume at a pressure of about 0.15 megapascals (MPa) (1.5 atmosphere) at 300 Kelvin (K) (80.3 degrees Fahrenheit (°F)). This 1-gram-mole limit reduces the amount of oxidants below levels where any cladding degradation is expected. Moisture removal is inherent in the vacuum drying process, and levels at or below those evaluated in PNL-6365 (about 0.43 gram-mole water) are expected if adequate vacuum drying is performed.

If alternative methods other than vacuum drying are used (such as forced helium recirculation), ensure that the applicant provides additional analyses or tests to sufficiently justify that cover gas moisture and impurity levels as specified in the chapter of the SAR on operating procedures are met and will not result in unacceptable cladding degradation.

The following examples illustrate the accepted methods for container draining and drying in accordance with the recommendations of PNL-6365:

- The container should be drained of as much water as practicable and evacuated to less than or equal to 4.0×10^{-4} MPa (4 millibar, 3.0 millimeters of mercury or Torr). After evacuation, adequate moisture removal should be verified by maintaining a constant pressure over a period of about 30 minutes without vacuum pump operation (or the vacuum pump is running but is isolated from the container with its suction vented to atmosphere). The container is then backfilled with an inert gas (e.g., helium) for applicable pressure and leak testing, with care being taken to preserve the purity of the cover gas. After backfilling, cover gas purity should be verified by sampling.

- The procedures should reflect the potential for blockage of the evacuation system or masking of defects in the cladding of nonintact rods for SNF storage containers as a result of icing during evacuation. Icing can occur from the cooling effects of water vaporization and system depressurization during evacuation. Icing is more likely to occur in the evacuation system lines than in the container because of decay heat from the fuel. A staged drawdown or other means of preventing ice blockage of the container evacuation path may be used (e.g., measurement of container pressure not involving the line through which the container is evacuated).

- The procedures should specify a suitable inert cover gas (such as helium) with a quality specification that ensures a known maximum percentage of impurities to minimize the source of potentially oxidizing impurity gases and vapors and adequately remove contaminants from the container.

- The process should provide for repetition of the evacuation and repressurization cycles if the container interior is opened to an oxidizing atmosphere following the evacuation and repressurization cycles (as may occur in conjunction with remedial welding, seal repairs).

[1] Can be broken down by radiolysis.

Ensure that the drying specifications are consistent with the proposed operating controls and limits described in the technical specifications provided in the SAR. In addition, assess the need for any additional technical specifications.

11.5.2.7 *Welding and Sealing*

Coordinate with the structural and materials reviewers' evaluation of welded lids as described in Section 8.5.9, "Bolt Applications," of this SRP for applying the proper weld joint, welding procedures, and nondestructive examination methods to ensure that the appropriate operating procedures are in place and acceptable. Verify that the procedures are acceptable for nondestructive examination and welding of the closure welds. Confirm that the SAR also ensures that ALARA principles are followed and includes acceptable provisions for correcting weld defects and any additional drying and purging that may be necessary.

Verify that provisions for placing and tightening any closure bolts, such as those associated with concrete overpacks, are consistent with information presented in SAR chapters that address applicable design criteria, structural evaluation, and the acceptance tests and maintenance program. The materials discipline should ensure that the closure bolts satisfy the conditions given in Section 8.5.9, "Bolt Applications," of this SRP. Ensure that the SAR specifies the torque required to properly seal the closure lid. The inner seal should be tested using a helium leak test with the interior of the cask pressurized as previously described. The outer seal should also be tested using a helium leak test with the between-seal volume pressurized as required by the respective subsection of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code, Section III, "Rules for Construction of Nuclear Facility Components."

11.5.2.8 *Filling and Pressurization*

Verify that the procedure recommendations address steps to fill and pressurize the confinement with inert gas such as helium with a known maximum percentage of impurities. The operating procedures should state that the filling and pressurization (or evacuation and backfill) process be repeated if the cavity is exposed to the atmosphere. Ensure that the procedure recommendations include the requirements in the chapter of the SAR on technical specifications and operation controls and limits.

Ensure that the SAR specifies the leak rate criteria (e.g., total leakage, leakage per closure, sensitivities of tests). Verify that these criteria are consistent with those presented in chapters of the SAR on principal design criteria, operating procedures, and technical specifications and operational controls and limits. In addition, assess the general methods of leak testing (e.g., pressure rise, mass spectrometry) as they apply to the leak rate being tested. Pay particular attention to the possible use of quick-disconnect fittings for draining and filling operations. Although no credit is taken for these devices as part of the confinement boundary, if leaking, their presence can cause leak paths through adjacent welds and affect the results of the leak test; the SAR should provide guidance regarding their use. In addition, the guidelines presented in the SAR should note that leak testing is in accordance with ANSI N14.5, "Radioactive Materials—Leakage Tests on Packages for Shipment."

Ensure that the SAR presents applicable pressure testing criteria (e.g., test pressure, hold periods, inspections).

11.5.3 Storage Container Handling and Storage Operations

For sites that propose to have SNF, HLW, or reactor-related GTCC waste handling facilities, the SAR should include operations descriptions for some or all of the following functions:

- receipt and inspection of loaded transportation packages
- SNF, HLW, or reactor-related GTCC waste transfer and examination
- fuel reconstitution
- SNF, HLW, or reactor-related GTCC waste container short-term storage
- storage container decontamination
- storage container loading and storage preparation
- storage container transfer to storage
- SNF, HLW, or reactor-related GTCC waste container removal from the storage pad and transfer into a transport package
- damaged fuel element containerization

Ensure that the applicant has adequately described the appropriate procedures, equipment involved, and personnel requirements. Ensure that the receipt, handling, and transfer descriptions include a functional description of the associated systems and descriptions of all features, systems, or special handling techniques that provide for safe operations under both normal and off-normal conditions. Follow the same procedures to review operations for handling HLW or reactor-related GTCC waste, as applicable, as used for SNF. Because the SNF, HLW, and reactor-related GTCC waste handling systems have many interfaces with other systems of the facility (e.g., SNF pool), verify that the applicant addressed these interfaces and that continuity of operations can occur under all operational conditions.

Pay particular attention to ensure that all accident events applicable to transfers are bounded by the design events analyzed in the chapters of the SAR on principal design criteria, structural evaluation, and accident analyses. This includes procedures to be specified in the SAR for use after a design-basis accident for testing the effectiveness of the shielding. The structural and thermal disciplines should coordinate their review to ensure that all conditions for lifting and handling methods are bounded by the evaluations in their respective chapters of the SAR. Coordinate as needed with the review of Chapter 17 of this SRP regarding technical specifications associated with cask transfer operations, such as restricting lift heights and environmental conditions (e.g., high and low temperatures).

Verify that the necessary operations descriptions include inspection, surveillance, and maintenance activities that are applicable during storage. Ensure that the appropriate surveillance and monitoring requirements are also include in the chapter of the SAR on technical specifications and operational controls and limits, and that maintenance is included in the chapter of the SAR on conduct of operations. Note that if the confinement vessel closure is bolted, the

NRC generally requires that the successful operation of the seals be demonstrated with an initial leak test and a monitoring system or a surveillance program, or both, as discussed in Chapter 12 of this SRP.

The shielding and radiation protection reviewers should verify that proposed operations descriptions give due consideration to maintaining ALARA with respect to doses during storage container handling and storage operations.

11.5.4 Storage Container Unloading

Verify that the SAR adequately describes the necessary unloading operations. The unloading procedure descriptions should present the activities sequentially in the anticipated order of performance, including those key prerequisite and preparation tasks that should be accomplished before storage container unloading. Where applicable, verify that the procedure guidance in the SAR ensures that any fluids, such as shield or borated water, fill their respective cavities according to design specifications. Additionally, for storage containers that require borated water to maintain subcriticality, ensure that the procedure guidance in the SAR includes verification that the water to be used for container reflood meets the minimum soluble boron content required by the technical specifications. Verify that the operations descriptions in the SAR incorporate the applicable operating controls and limits described in the chapter of the SAR on technical specifications and operation controls and limits.

11.5.4.1 *Damaged Fuel*

Ensure that the SAR includes appropriate additional measures for the potential presence of damaged fuel. Procedures should be designed to maximize worker protection from unanticipated radiation exposures or contaminants from damaged fuel in accordance with ALARA principles and, to the maximum extent possible, to prevent any uncontrolled releases to the environment. The following points outline the relevant safety concerns and one acceptable approach to address damaged fuel contingencies in unloading:

- The procedure descriptions should provide for fuel unloading under normal conditions.

- The unloading process should ensure that the fuel can be safely unloaded with regard to structural, criticality, thermal, and radiation protection considerations. This includes the provision for safe maintenance of the fuel and storage container while any additional measures needed to address suspected damaged fuel are planned and implemented.

- The unloading process should reflect the potential for damaged fuel and changing radiological conditions.

- The process should include measures to check for and detect damaged fuel conditions (such as cask (or canister) atmosphere samples) before opening the storage container. (Note that fuel oxidation resulting from exposure to air at temperatures typical for dry storage is a known form of fuel degradation. Therefore, the presence of air in a storage container designed to maintain an inert atmosphere indicates that the fuel may be degraded. The detection of fission gases is another indicator that the fuel may be degraded.)

- The process may establish sample result thresholds above which damaged fuel is suspected. Other technically sound methods may be used to check for potential air leakage paths. Such methods may include designs that monitor storage container internal pressure or seal integrity and alert the licensee to a problem before oxidation could occur. However, this method may not address detection of potential fuel degradation resulting from other mechanisms (such as a storage container drop accident).

- If the sample indicates normal conditions, the normal unloading process should be followed.

- If damaged fuel is suspected or found, the procedure description should stipulate that additional measures, appropriate for the specific conditions that include the canning of the damaged fuel, are to be planned, reviewed, and approved by the designated approval authority and implemented to minimize exposures to workers and radiological releases to the environment. These additional measures may include provision of filters, respiratory protection, and other methods to control releases and exposures in accordance with ALARA.

11.5.4.2 *Cooling, Venting, and Reflooding*

Verify that the SAR describes applicable operational measures to control storage container cooling, venting, and reflooding (when appropriate). Verify that these measures are consistent with the results of the structural, materials, and thermal evaluations in the SAR, respectively. Storage container cooling, venting, and reflooding should not result in damage to the fuel. Operational measures may include external cooling of the storage container for initial temperature reduction, restricting reflood flow rates to control and limit internal pressure from steam formation, and limiting cooldown rates.

Devote special attention to reviews in this area since analyses of existing designs have predicted fuel temperatures during storage and transfer in excess of 260 °C (500 °F) for design-basis heat loads. Operational controls may be required to address the following potential effects during a cooldown and reflood evolution:

- Storage container pressurization may occur as a result of steam formation as reflood water contacts hot surfaces.

- Excessive cooling rates may cause fuel cladding and fuel rod component damage and release of radioactive material as a result of stress (e.g., thermal, internal pressure) beyond material strengths (see Sections 8.5.15.2.3, “Drying Adequacy,” and 8.5.15.2.4, “Maximum (Peak) Cladding Temperature,” of this SRP).

- Excessive cooling rates may induce thermal stress that causes gross deformation of the fuel assembly components and subsequent binding with the basket.

- Storage container supply and vent line failures from inadequate design for pressure and temperature could result in radiological exposures and personnel hazards (e.g., steam burns).

11.5.4.3 *Fuel Crud*

Verify that the procedure descriptions in the SAR include contingencies for protection from fuel crud particulate material. Appendix E to ANSI/ANS 57.9 provides a short discussion of crud with respect to dry transfer systems. Verify that the unloading procedures include an alert to operations personnel to wait until any loose particles have settled and to slowly move the fuel assemblies to minimize crud dispersion in the SNF pool. Experience with wet unloading of boiling-water reactor fuel after transport has involved handling significant amounts of crud. This fine crud, which includes cobalt-60 and iron-55, will remain suspended in water or air for extended periods. The reflood process, during unloading of boiling-water reactor fuel, has the potential to disperse crud into the fuel transfer pool and the pool area atmosphere, thereby creating airborne exposure and personnel contamination hazards. By contrast, no significant crud dispersal problems have been observed in handling pressurized-water reactor fuel because of differences in the characteristics of crud on this type of fuel.

11.5.4.4 *ALARA*

Verify that the procedure descriptions in the SAR incorporate ALARA principles and practices. These may include provisions to perform radiological surveys, implement exposure and contamination control measures, or use temporary shielding and inclusion of caution statements related to specific actions that could change radiological conditions.

11.5.4.5 *Offsite Release*

Where applicable, verify that the SAR describes methods to minimize offsite releases. These methods may include filtered ventilation, decontamination of the storage containers, temporary containments, and the methods described in Section 11.5.2.5 above. The procedures should also provide for minimizing generation of radioactive waste.

11.5.5 Repair and Maintenance (SL)

A concern for review of any storage container repair capability incorporated into the DSF is that the applicant recognizes the need for inspection of loaded containers and for container decontamination. This need would apply to the storage containers used at the site as well as any loaded transportation packages received at the site. If the licensee will provide a repair capability on site for the repair of storage containers and related SSCs (e.g., overpacks and onsite transfer casks) and transport packages, verify that the SAR describes the skills and equipment necessary for performing such repairs. Section 12.5.4, “Maintenance Program,” of this SRP provides guidance useful for the evaluation of maintenance and repair operations.

11.5.6 Other Operating Systems (SL)

For other systems that are also considered important to safety, review the description of the locations of the various systems in relationship to their functional objectives. Verify that the applicant has described provisions for coping with unscheduled occurrences so that a single failure within one of the auxiliary systems will not result in a release of radioactive material or unanalyzed conditions that may affect any safety functions, such as nuclear criticality safety, of the DSS or DSF SSCs. Evaluate the systems to ensure that the design includes performance under normal operating loads, off-normal operating loads, loading situations resulting from primary failure and/or accident conditions, and loading situations required for the safety of a shutdown operation. If a system requires a technical specification, verify that the SAR includes the required technical specification, and ensure that it is part of the license.

11.5.7 Operation Support Systems (SL)

Review the descriptions of the I&C systems in the SAR and determine whether the applicant's definition of their function is adequate. Ensure that, for SSCs important to safety, the SAR describes all major components, operating characteristics, locations of sensors and alarms, threshold levels for I&C that produce alarms, automatic and manual control actions to be triggered, and safety criteria.

Consider the projected accident and off-normal events (addressed in SRP Chapter 16, "Accident Analysis Evaluation") and the roles that the I&C systems have in avoiding or mitigating significant radiological consequences of those events. Verify that the applicant has considered the redundancy required to ensure safe operation or safe curtailment of operations under accident conditions. Verify that the SAR reflects that spare or alternative instrumentation, if provided, has been designed to ensure safe functioning.

Ensure that the applicant has proposed technical specifications that include reliance on an I&C system performance as outlined in RG 3.62 and RG 3.48.

11.5.8 Control Room and Control Area (SL)

Review the control room and control area functions, equipment, I&C links, and staffing for consistency and appropriateness for the intended functional control and safety roles. Information on these different aspects of the control room or control area, as applicable, may be at various locations within the SAR.

Ensure that the SAR includes an explanation for an omission of a control room, monitoring room, control area, or monitoring area, as applicable. Explanations might include, but not be limited to, a description of functions and procedures (flowcharts and narrative descriptions) that provide for performance without the need for a centralized control room, the acceptability of accident and off-normal event and condition analyses that show acceptable levels of maximum response and safety without use of a control room, and the desire that damage avoidance and mitigation be based on passive measures to the extent feasible.

11.5.9 Analytical Sampling (SL)

Verify that the types of samples and rates of sampling are appropriate for the conditions being monitored. Ensure that the SAR includes provisions for obtaining samples during off-normal conditions to ensure that prescribed limits have not been exceeded. Confirm that the SAR describes the facilities and equipment that will be available to perform the analyses. Ensure the SAR also describes disposition of laboratory wastes.

Compare the proposed analytical sampling operations with those of existing similar facilities as documented in the final SARs for licensed DSFs. Determine whether the proposed analytical sampling operations are reasonable and the descriptions of the operations, facilities, and equipment are adequate given this comparison.

11.5.10 Fire and Explosion Protection (SL)

11.5.10.1 *General Consideration* ***(SL)***

Depending on the design, magnitude, scope, and fire hazards of a proposed DSF, the applicant may have to institute a fire protection program (FPP) to satisfy the requirements of 10 CFR 72.122(c). Ensure that the applicant performed a fire and explosives hazards analysis of the facility and, if warranted, instituted an FPP. The applicant may use the following guidance:

- RG 1.189, “Fire Protection for Nuclear Power Plants,” as it relates to the design provisions given to implement the FPP

- RG 1.78, “Evaluating the Habitability of a Nuclear Power Plant Control Room During a Postulated Hazardous Chemical Release,” as it relates to habitable areas, such as the control room and to the use of specific fire extinguishing agents

- the NRC technical position on fire protection for fuel cycle facilities

11.5.10.2 *Spent Nuclear Fuel Storage Containers* ***(SL)***

The DSF may use DSSs approved under Subpart L, “Approval of Spent Fuel Storage Casks,” of 10 CFR Part 72, provided, in part, that the applicant satisfies the fire requirements identified in the CoC, if any, and 10 CFR 72.122(c).

Verify that the SAR indicates that the DSS materials, such as protective coatings, are compatible with water used in the DSS cavity so as to preclude or minimize the potential for combustible gas generation. For background, refer to NRC Bulletin 96-04, “Chemical, Galvanic, or Other Reactions in Spent Fuel Storage and Transportation Casks,” dated July 5, 1996.

11.5.10.3 *Guidance for a Fire Protection Program* ***(SL)***

Verify that waste confinement systems important to safety have adequate fire and explosive protection. Specifically, verify that an FPP provides assurance that a fire will not impact the ability of SSCs important to safety to continue to effectively perform their safety and design functions in accordance with the general design criteria in 10 CFR 72.122(c). This includes adverse effects from both the operation and the failure of the fire suppression system. A defense-in-depth approach should achieve balance among prevention, detection, containment, and suppression of fires. Confirm that the SAR indicates that there is a fire protection policy for the protection of SSCs important to safety at each facility and for the procedures, equipment, and personnel required to implement the program at the site. Ensure that the FPP consists of fire detection and extinguishing systems and equipment, administrative controls and procedures, and trained personnel.

Portions of the review procedures of Section 9.5.1.1, “Fire Protection Program,” of NUREG-0800, “Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition,” and the guidelines of Chapter 7 “Fire Safety” of NUREG-1520, “Standard Review Plan for Fuel Cycle Facilities License Applications,” Revision 2, may apply to the MRS or ISFSI contingent on the design of the installation and associated fire hazards. Many of the national codes and standards cited in these NRC guidance documents, in particular, the codes and standards of the National Fire Protection Association, could apply to the ISFSI or MRS.

Review the SAR to determine that the appropriate levels of management and trained, experienced personnel are responsible for the design and implementation of the FPP in accordance with RG 1.189.

Review the SAR's analysis of the fire potential in facility areas important to safety and the hazard of fires to these areas to determine that the proposed FPP is able to ensure that DSF SSCs important to safety will continue to effectively perform their safety and design functions in the event of a fire.

Review the elevated temperatures that may be of concern because of their effects on strength, heat treatment, durability, other properties, or change of state. A small amount of exterior concrete spalling may result from a fire or other high-temperature condition or application of fire, water, or rain on heated surfaces. Spalling from temperature gradients typically is considered to have minor (at most) structural significance, but such condition could partially block ventilation passages, depending on the design.

Evaluate the FPP piping and instrumentation diagrams (P&IDs) and facility layout drawings to verify that facility arrangement, buildings, and structural and compartment features that affect the methods used for fire protection, fire control, and control of hazards are acceptable for the protection of safety-related equipment.

Determine that design criteria and bases for the detection and suppression systems for smoke, heat, and flame control are in accordance with the fire protection guidance in NUREG-0800, Section 9.5.1.1, "Fire Protection Program" and NUREG-1520, Chapter 7, "Fire Safety," and provide adequate protection for SSCs important to safety. Determine whether fire protection support systems, such as emergency lighting and communication systems, floor drain systems, and ventilation and exhaust systems, are designed to operate, consistent with this objective. Verify the results of an FPP failure modes and effect analysis to assure that the entire fire protection system for one safety-related area cannot be impaired by a single failure.

Verify that the applicant's technical specifications for fire protection, specifically the limiting conditions for operation and surveillance requirements of the technical specifications, are in agreement with the requirements developed as a result of the staff review. RG 1.189 provides guidance for fire detection and suppression as well as the fire protection water system.

Confirm that the control room or control area ventilation system P&IDs show monitors located in the system intakes that are capable of detecting radiation, smoke, and toxic chemicals. Ensure that the monitors actuate alarms in the control room. Confirm that the P&IDs show provisions for isolation of the control room upon smoke detection at the air intakes. Although the isolation may be actuated manually for most cases, special cases may require automatic isolation, such as for fires resulting from aircraft crashes. Consult RG 1.189 for additional guidance.

Verify that miscellaneous areas, such as shops, warehouses, auxiliary boiler rooms, fuel oil tanks, and flammable and combustible liquid storage tanks, are located and protected so that a fire or effects of a fire, including smoke, will not adversely affect any SSCs important to safety.

Confirm that acetylene-oxygen gas cylinder storage locations are not in areas that contain or expose equipment important to safety, or the fire protection systems that serve those areas important to safety, exposing these locations to explosive hazards. The applicant should propose a permit system to use this equipment in areas of the facility that are important to safety (also see

RG 1.189). Verify that unused ion exchange resins and hazardous chemicals are not to be stored in areas that contain or expose equipment important to safety.

Verify that materials that collect and contain radioactivity, such as spent ion exchange resins, charcoal filters, and high-efficiency particulate air filters, are stored in closed metal tanks or containers that are located in areas free from ignition sources or combustibles. These materials should also be protected from exposure to fires in adjacent areas. Consideration should be given to requirements for the removal of decay heat from the radioactive materials.

11.6 Evaluation Findings

The NRC reviewer should prepare evaluation findings upon satisfaction of the regulatory requirements in Section 11.4 of this SRP. If the documentation submitted with the application fully supports positive findings for each of the regulatory requirements, the statements of findings should be similar to the following:

F11.1 **(SL)** [If applicable] The DSF is to be located on the same site as another facility licensed by the NRC. Potential interactions between these facilities and the DSF have been evaluated, in accordance with 10 CFR 72.24(a) and have been determined to be acceptable and pose no undue risk to any of the facilities.

F11.2 The SAR includes acceptable descriptions and discussions of the DSS or DSF operations, operating characteristics and safety considerations, in compliance with 10 CFR 72.24(b) or 10 CFR 72.234(f).

F11.3 **(CoC)** The [DSS designation] is compatible with [wet/dry] loading and unloading in compliance with 10 CFR 72.236(h). General procedure descriptions for these operations are summarized in Chapter(s)___of the applicant's SAR. Detailed procedures will need to be developed and evaluated on a site-specific basis.

F11.4 The DSS or DSF storage container design allows for ready retrieval of the SNF and, as applicable for a DSF, reactor-related GTCC waste, and HLW for further processing or disposal as required. The descriptions of the proposed [DSS or DSF] functions and operating systems with regard to retrieval of stored radioactive material from storage, in normal and off-normal conditions, are acceptable and comply with 10 CFR 72.122(l) and with 10 CFR 72.236(m).

F11.5 The smooth surface [or other feature] of the DSS or DSF SSCs is designed to facilitate decontamination in compliance with 10 CFR 72.126(a)(2) and 10 CFR 72.236(i). Only routine decontamination will be necessary after the storage container is removed from the SNF pool.

F11.6 (SL) Radioactive waste expected to be generated during operations associated with the DSF will be minimized in compliance with 10 CFR 72.24(f). [Note that contaminated water from the SNF pool will be governed by the 10 CFR Part 50 or 10 CFR Part 52 license conditions for DSFs co-located with and using those facilities.]

F11.7 No significant radioactive effluents are expected to be produced during storage. [Note that any radioactive effluents generated during the storage container loading will be governed by the 10 CFR Part 50 or 10 CFR Part 52 license conditions for DSSs and for DSFs co-located with a 10 CFR Part 50 or 10 CFR Part 52 licensed facility.]

F11.8 The content of the operations descriptions in the SAR is adequate to protect health and minimize damage to life and property that is in compliance with 10 CFR 72.24(h) for a DSF or 10 CFR 72.234(f) for a DSS.

F11.9 The radiation protection chapter of this SER evaluates the operations descriptions and systems, including implementation of operational limits and restrictions to meet the applicable regulatory requirements in 10 CFR Part 20 and in 10 CFR Part 72 (i.e., 10 CFR 72.104 and 10 CFR 72.126) for a DSF or, for a DSS, to facilitate compliance with these requirements by licensees using the DSS and to meet 10 CFR 72.236(d). For a DSS, a licensee using the DSS may also establish additional restrictions for use of the DSS its site.

F11.10 **(SL)** [One of the following, as appropriate]

The design of the [DSF designation] provides for an acceptable [control room/control area] as part of the facilities to be built, in compliance with 10 CFR 72.122(j).

– OR –

The operating procedures and schedule of operations for the [DSF designation] acceptably provide for control during storage operations to be accomplished from the security, monitoring, or surveillance office facility, as appropriate, and for control during loading, transfer, and unloading operations from temporary control facilities, and the design includes acceptable provisions for such facilities. This is considered to comply with 10 CFR 72.122(j).

– OR –

The [DSF designation] is to be located on a site with existing facilities suitable and available for control of [DSF designation] operations under off-normal or accident conditions, and their use will not interfere with other operations on the site important to safety, in compliance with 10 CFR 72.40(a)(3) and 10 CFR 72.122(j).

F11.11 **(SL)** The proposed [DSF designation] facilities include the following utility service systems: [identify]. [If appropriate] The following utility service systems are important to safety: [identify]. The [DSF designation] design provides for redundant systems to the extent necessary to maintain, with adequate capacity, the ability to perform safety functions, assuming a single failure, in compliance with 10 CFR 72.122(k)(1).

F11.12 **(SL)** The proposed design of the [DSF designation] emergency utility services acceptably permits testing of the functional operability and capacity of each system and permits operation of associated safety systems, in compliance with 10 CFR 72.122(k)(2).

F11.13 **(SL)** The proposed design of the [DSF designation] includes the following systems and subsystems that require continuous electric power to permit continued functioning of all systems essential to safe storage: [identify]. The design of the [DSF designation] acceptably provides for timely emergency power for these systems and subsystems, in compliance with 10 CFR 72.122(k)(3).

F11.14 The design and procedures for the DSF provide acceptable capability to test and monitor components important to safety, in compliance with 10 CFR 72.128(a)(1), for DSFs, and 10 CFR 72.234(f), for CoCs.

For a DSF only, if the design of the SNF storage system to be used at the DSF has been previously certified under 10 CFR Part 72, Subpart L, the following evaluation finding statement would also be appropriate:

The proposed DSF uses a SNF storage system that has been previously certified by the NRC.

F11.15 **(SL)** The staff concludes that the site-specific fire and explosion hazards are acceptable and that the fire protection program meets the requirements in 10 CFR 72.122(c). This conclusion is based on the applicant meeting the guidelines in RG 1.189, "Fire Protection for Nuclear Power Plants," as well as the applicable industry standards. In meeting these guidelines, the applicant has provided an acceptable basis for the [ISFSI/MRS] design and location of safety-related structures and systems to minimize the probability and effect of fires and explosions; has used noncombustible and heat-resistant materials whenever practical; and has provided fire detection and firefighting systems of appropriate capacity and capability to minimize adverse effects of fire on SSCs important to safety.

The reviewer should provide a summary statement similar to the following:

The staff concludes that the operations descriptions, including procedures and guidance, for the operation of the [DSS or DSF] are in compliance with 10 CFR Part 72 and that the applicable acceptance criteria have been satisfied. The evaluation of the operations descriptions provided in the SAR offers reasonable assurance that the DSS or DSF will enable the safe storage of SNF and, as applicable for DSFs, reactor-related GTCC waste and HLW. This finding is based on a review that considered the regulations, appropriate regulatory guides, applicable codes and standards, and accepted practices.

11.7 References

10 CFR Part 20, "Standards for Protection Against Radiation."

10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities."

10 CFR Part 52, "Licenses, Certifications, and Approvals for Nuclear Power Plants."

10 CFR Part 71, "Packaging and Transportation of Radioactive Material."

10 CFR Part 72, "Licensing Requirements for the Independent Storage of Spent Nuclear Fuel, High-Level Radioactive Waste, and Reactor-Related Greater than Class C Waste."

American National Standards Institute (ANSI) N14.5, "Radioactive Materials—Leakage Tests on Packages for Shipment."

ANSI/American Nuclear Society (ANS) 57.9, "Design Criteria for an Independent Spent Fuel Storage Installation (Dry Storage Type)."

ANSI/ANS 3.2, "Managerial, Administrative, and Quality Assurance Controls for the Operational Phase of Nuclear Power Plants."

American Society of Mechanical Engineers, Boiler and Pressure Vessel Code
Section III, "Rules for Construction of Nuclear Facility Components"

Knoll, R.W. and E.R. Gilbert, "Evaluation of Cover Gas Impurities and Their Effects on the Dry Storage of LWR Spent Fuel," PNL-6365, DE88 003983, Pacific Northwest National Laboratory, November 1987.

NRC Bulletin 96-04, "Chemical, Galvanic, or Other Reactions in Spent Fuel Storage and Transportation Casks," July 5, 1996.

NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition."

NUREG-1520, "Standard Review Plan for Fuel Cycle Facilities License Applications," Revision 2, June 2015.

NUREG/CR-4775, "Guide for Preparing Operating Procedures for Shipping Packages," UCID-20820, Lawrence Livermore National Laboratory, December 1988.

Regulatory Guide 1.33, "Quality Assurance Program Requirements (Operation)."

Regulatory Guide 1.78, "Evaluating the Habitability of a Nuclear Power Plant Control Room During a Postulated Hazardous Chemical Release."

Regulatory Guide 1.189, "Fire Protection for Nuclear Power Plants."

Regulatory Guide 3.48, "Standard Format and Content for the Safety Analysis Report for an Independent Spent Fuel Storage Installation or Monitored Retrievable Storage Installation (Dry Storage)."

Regulatory Guide 3.61, “Standard Format and Content for a Topical Safety Analysis Report for a Spent Fuel Dry Storage Cask.”

Regulatory Guide 3.62, “Standard Format and Content for the Safety Analysis Report for Onsite Storage of Spent Fuel Storage Casks.”

12 CONDUCT OF OPERATIONS EVALUATION

12.1 Review Objective

The objective of the U.S. Nuclear Regulatory Commission's (NRC's) conduct of operations review is to ensure that the applicant has (1) described an appropriate infrastructure to manage, test, operate, and maintain the facility, including provisions for effective training, emergency planning, and physical security programs for a dry storage facility (DSF), and (2) developed appropriate acceptance tests and maintenance programs to ensure that its dry cask storage system (DSS) or DSF structures, systems, and components (SSCs) are fabricated and maintained in accordance with the design described in the safety analysis report (SAR).

12.2 Applicability

This chapter applies to the review of an applicant's SAR with respect to licenses for an independent spent fuel storage installation (ISFSI) or a monitored retrievable storage installation (MRS), categorized as a DSF. It also applies to the review of an applicant's SAR for a certificate of compliance (CoC) of a DSS for use at a general license ISFSI. Sections and tables of this chapter that apply only to a specific license for a DSF application have "**(SL)**" in the heading. A section, paragraph, or table without an identifier applies to both types of applications.

Some of the review procedures in this chapter relate to the conduct of operations associated with spent nuclear fuel (SNF) pools. Carefully review all of the review procedures in this SRP for applicability to SNF pools.

12.3 Areas of Review

This chapter addresses the following areas of review, which can have an impact on SSCs important to safety:

- organizational structure (SL)
- acceptance tests
- preoperational testing and startup operations (SL)
- maintenance program
- normal operations (SL)
- personnel selection, training, and certification (SL)
- emergency planning (SL)
- physical security and safeguards contingency plans (SL)

12.4 Regulatory Requirements and Acceptance Criteria

This section summarizes those parts of Title 10 of the *Code of Federal Regulations* (10 CFR) Part 72, "Licensing Requirements for the Independent Storage of Spent Nuclear Fuel, High-Level Radioactive Waste, and Reactor-Related Greater than Class C Waste," that are relevant to the review areas this chapter addresses. The NRC staff reviewer should refer to the exact language in the regulations. Tables 12-1a and 12-1b match the relevant regulatory requirements to the areas of review covered in this chapter.

12.4.1 Organizational Structure (SL)

The SAR should describe the organizational structure and administrative control system that will be used for the proposed ISFSI or MRS (i.e., through construction, preoperational testing and initial operations, normal operations, and decommissioning) as required in 10 CFR 72.24(h). Chapter 10A, "Radiation Protection Evaluation for Dry Storage Facilities," of this SRP, particularly Sections 10A.4.4, "Health Physics Program," and 10A.4.4.1, "Organization and Staffing," for regulatory requirements, and 10A.5.4, "Health Physics Program," and 10A.5.4.1, "Organization and Staffing," for review procedures, provides additional guidance on the information the SAR should include as related to the radiation protection and health physics aspects of the organizational structure and staffing.

12.4.1.1 *Corporate Organization* ***(SL)***

The SAR must describe the corporate organization responsible for the ISFSI or MRS (10 CFR 72.24(h)), which should include organization charts and position descriptions. If the corporation is made up from two or more corporate identities, the SAR should describe the relationship and responsibilities between each entity.

The applicant must demonstrate the financial capabilities of the corporation to construct, operate, and decommission the installation, as required in 10 CFR 72.22(e). The scope of this SRP does not include specific guidance for reviewing the financial qualifications required in 10 CFR 72.22(e)(1) and 10 CFR 72.22(e)(2); this information is part of the application but separate from the SAR. However, Chapter 14, "Decommissioning Evaluation," of this SRP does provide guidance for reviewing financial qualifications regarding decommissioning, as required in 10 CFR 72.22(e)(3) and 10 CFR 72.30, "Financial assurance and recordkeeping for decommissioning," as part of the SAR. The Department of Energy (DOE) is exempt from financial assurance requirements per 10 CFR 72.22(e), 10 CFR 72.40(a)(6), and 10 CFR 72.40(a)(10).

Financial reviews should be coordinated with the NRC Office of Nuclear Reactor Regulation. The NRC project manager should ensure that the application contains financial data, in accordance with 10 CFR 72.22(e), that shows that the licensee can carry out the activities being sought for the requested duration. Information should state where the activity will be performed, the general plan for carrying out the activity, and the period of time for which the license is requested.

Table 12-1a Relationship of Regulations and Areas of Review for a DSF (SL)

Areas of Review	10 CFR Part 72 Regulations								
	72.22(e)	72.24	72.28	72.30	72.32	72.40(a)	72.82(d)	72.122 (a)(f)	72.124(b)
Organizational Structure	●	(h)	●	●		(13)			
Acceptance Tests		(h)(i)					●	(a)	●
Preoperational Testing and Startup Operations		(h)(i)(p)				(13)			
Maintenance Program						(13)	●	(f)	
Normal Operations		(h)	●			(5)(13)			
Personnel Selection, Training, and Certification		(h)(j)	●			(4)(9)			
Emergency Planning		(k)	●		●	(11)			
Physical Security and Safeguards Contingency Plans		(o)				(8)(14)			

Areas of Review	10 CFR Part 72 Regulations (cont.)							
	72.156	72.162	72.174[a]	72.180	72.184	72.190	72.192	72.194
Organizational Structure **(SL)**								
Acceptance Tests		●						
Preoperational Testing and Startup Operations **(SL)**								
Maintenance Program								
Normal Operations **(SL)**	●		●					
Personnel Selection, Training, and Certification **(SL)**						●	●	●
Physical Security and Safeguards Contingency Plans **(SL)**				●	●			

a This requirement specifies the retention of quality assurance records. Other requirements related to the retention of records include 10 CFR 72.70, "Safety Analysis Report Updating," 72.72(a), 72.44(b)(4), 72.74, "Reports of Accidental Criticality or Loss of Special Nuclear Material," 72.76, "Material Status Report," 72.78, "Nuclear Material Transaction Reports," and 73.21, "Protection of Safeguards Information: Performance Requirements."

Table 12-1b Relationship of Regulations and Areas of Review for a DSS (CoC)

Areas of Review	10 CFR Part 72 Regulations			
	72.124(b)	72.162	72.234(a)	72.236
Acceptance Tests	●	●	●	(j)(l)
Maintenance Program			●	(g)

The SAR should describe the corporate functions, responsibilities, and authorities related to each aspect of the installation (e.g., design, engineering, construction, quality assurance (QA), testing). The SAR must describe the inhouse organization and technical staff (e.g., numbers of personnel, qualifications, educational and experience backgrounds), as required in 10 CFR 72.28. The SAR should describe the relationship between the applicant's inhouse organization and outside contractors and suppliers, including the extent of dependence on those sources for design, construction, QA, and other functions.

The applicant should also describe the relationship between the corporate and onsite organizations and explain the nature of interaction between corporate management and the site related to health and safety, including any role in policy/procedure development, audits, inspections, and investigations.

12.4.1.2 *Onsite Organization* ***(SL)***

The SAR should describe the onsite organization, including organizational charts and position descriptions with emphasis on positions that perform functions important to safety. Such positions include, but are not limited to, those with responsibilities in health physics, nuclear criticality safety, training and certification, emergency planning and response, operations, maintenance, engineering, and QA.

The discussion of positions and responsibility should illustrate how these functions or aspects of these functions are performed, including the degree of separation between the facility operations organization and other parts of the onsite organization that perform functions important to safety. The SAR should also identify alternates who are authorized to act in the absence of individuals assigned to key positions and identify which positions have shut-down or stop-work authority for health or safety reasons.

The SAR should identify minimum staffing levels for major entities within the onsite organization.

The SAR should identify whether the onsite organization includes a safety committee (or committees) and describe the membership, duties, responsibilities, operating characteristics, and reporting function of proposed safety committees.

12.4.1.3 *Identification of Agents and Contractors* ***(SL)***

The SAR should identify the prime agents or contractors for the design, construction, and operation of the installation. All principal consultants and outside service organizations, including those providing QA services, should be identified. The SAR should clearly define the division and assignment of responsibilities among these parties.

12.4.1.4 *Management and Administrative Controls* ***(SL)***

The SAR should describe the proposed management and administrative control system (10 CFR 72.24(h)), including provisions for the following:

- administrative and general plant procedures including implementation of good radiation protection practices and objectives to ensure occupational exposures will remain as low as is reasonably achievable (ALARA)
- a program of surveillance, testing, and inspections of items and activities important to safety
- periodic independent audits
- change control
- employee training and certification programs
- records preparation and maintenance

Administrative procedures address planning, administrative controls, and document issuance. The procedures provide rules and instructions on personnel conduct, preparation and retention of plant documents, and interfaces among plant organizations. General facility procedures prescribe the actions required to achieve safe operation and provide necessary instruction for the operation and maintenance of facility systems and equipment, including implementation of good radiation practices and ALARA objectives. The SAR should describe the program for preparation, review, change, and approval of procedures. The applicant should also identify the onsite organizations that use procedures and the activities or operations that are covered by such procedures. Sections 12.4.5.1 and 12.5.5.1 below provide guidance on evaluating procedures for normal plant operation.

The applicant should describe the program of surveillance, testing, and inspection to ensure satisfactory inservice performance of items and activities important to safety. The description should address the development and use of procedures that set forth the steps to be taken and identify the standards or criteria to be applied. The program should include provisions for the following:

- preoperational testing (see Sections 12.4.3 and 12.5.3 below) to demonstrate facility operability and identify conditions adverse to safety
- operational testing and surveillance to verify and record characteristics of facility equipment and components
- surveillance, testing, and inspection after modification or when corrective actions have been completed

The management control system description should also include requirements for planned and scheduled internal and external audits to evaluate the application and effectiveness of management controls, facility procedures, and other activities affecting safety. The audit program should describe audit frequency, methods for documenting and communicating audit findings, resolution of issues, and implementation of corrective actions.

The applicant should also describe the system for change control, including how change control is integrated into the management control system. The SAR should describe the coordination of change between and among potentially affected organizations (e.g., engineering, operations, maintenance, training). The SAR should describe how operations are shut down to effect changes and how all facility equipment and procedural changes are completed. The training of staff before resumption of operations should also be addressed.

The management system description should also include the system for maintaining records of facility operation (as addressed in Sections 12.4.5.2 and 12.5.5.2 below).

12.4.2 Acceptance Tests

The acceptance tests demonstrate that the DSS or DSF SSCs and features have been fabricated in accordance with the design criteria and that the initial operation of the DSS or DSF SSCs and features complies with regulatory requirements. A comprehensive evaluation should encompass, but may not be limited to, the following acceptance tests:

- structural/pressure tests
- leak tests
- visual and nondestructive examination (NDE) inspections
- shielding tests
- neutron absorber tests
- thermal tests
- storage container identification

In general, the acceptance tests outlined in the SAR should cite appropriate authoritative codes and standards. Table 12-2 lists the standards and codes the NRC has previously accepted as the regulatory basis for the design, fabrication, inspection, and testing of SNF storage system and container components. The SAR should clearly identify any exceptions to the listed codes and standards (see SRP Chapter 17, “Technical Specifications Evaluation”).

Table 12-2 Acceptable Regulatory Basis for the Design, Fabrication, Inspection, and Testing of DSS or DSF Components

System/Component	Acceptable Regulatory Basis
Confinement System	American Society of Mechanical Engineers (ASME), Boiler and Pressure Vessel (B&PV) Code, Section III, “Rules for Construction of Nuclear Facility Components,” Division 1 American National Standards Institute (ANSI) N14.5, “Radioactive Materials—Leakage Tests on Packages for Shipment”
Confinement Internals (e.g., basket)	ASME B&PV Code, Section III, Subsection NG
Metal Cask Overpack	ASME B&PV Code, Section VIII, “Rules for Construction of Pressure Vessels”
Concrete Cask Overpack	American Concrete Institute (ACI) 318, “Building Code Requirements for Structural Concrete and Commentary”; ACI 349, “Code Requirements for Nuclear Safety-Related Concrete,” as appropriate
Other Metal Structures	ASME B&PV Code, Section III, Subsection NF American Institute of Steel Construction 360, “Specification for Structural Steel Buildings.”

12.4.3 Preoperational Testing and Startup Operations (SL)

The SAR must describe the plans for preoperational testing and initial facility (startup) operations (10 CFR 72.24(g)). Regulatory Guide (RG) 3.62, “Standard Format and Content for the Safety Analysis Report for Onsite Storage of Spent Fuel Storage Casks,” and RG 3.48, “Standard Format and Content for the Safety Analysis Report for an Independent Spent Fuel Storage Installation or Monitored Retrievable Storage Installation (Dry Storage),” provide guidance on the information to be included in the SAR related to preoperational testing and startup operations.

The preoperational testing and operating startup activities are determined by the type of radioactive material to be stored.

The SAR should describe the administrative procedures used for conducting the testing and startup activities. This description should include the system to be used for preparing, approving, and executing the test procedures and for evaluating, documenting, and approving test results. Provisions should be made for incorporating changes to the system or individual procedures on the basis of inadequacies in test procedures or unexpected test results. The organizational responsibilities for administering the system should be identified, and the qualifications of involved personnel should be described.

12.4.3.1 *Preoperational Testing Plan* ***(SL)***

The preoperational testing plan description should identify the testing objectives and the general methods to meet those objectives. The SAR should identify each item (facility, component, piece of equipment, operation) to be tested. For each physical or operational item, the applicant should provide the following information:

- the type of test to be performed
- the expected response

- the acceptable margin of difference from the expected response
- the method of validation (if applicable)
- appropriate corrective action for unexpected or unacceptable results

If the proposed ISFSI or MRS contains any SSCs important to safety for which functional adequacy or reliability has not been demonstrated or otherwise validated, the preoperational test plan must include a description and schedule showing how these safety questions will be resolved before the initial receipt of the radioactive materials to be stored (10 CFR 72.24(i)).

12.4.3.2 *Startup Plan **(SL)***

The operating startup plan should identify those specific operations involving the initial handling of radioactive material to be placed into storage. Although the operating startup plan does not necessarily include the facility procedures to be used for normal operations or during steady-state conditions, the plan should evaluate the effectiveness of those procedures. For considerations related to the ALARA principle, the applicant should perform as many of the operating startup actions as feasible during preoperational testing (i.e., before sources of exposure are present).

The operating startup plan should include the following elements:

- tests and confirmation of procedures and exposure times involving actual radioactive sources (e.g., radiation monitoring)

- direct radiation monitoring of storage containers (and other SSCs used to handle or contain radioactive materials) and shielding for radiation dose rates, streaming, and surface “hot spots” and containment surveys

- verification of effectiveness of heat removal features

- documentation of results of tests and evaluations

12.4.4 Maintenance Program

The maintenance program describes actions that the licensee needs to implement during the storage period to ensure that the DSS or DSF SSCs and features perform their intended functions. A comprehensive evaluation should identify and describe the necessary maintenance programs and address the following for each of the identified maintenance programs:

- inspection
- tests
- repair, replacement, and maintenance

In general, the maintenance programs outlined in the SAR should cite appropriate authoritative codes and standards. The NRC has previously accepted the codes and standards listed in Table 12-2 as the regulatory basis for the design, fabrication, inspection, and testing of SNF storage system components.

12.4.5 Normal Operations (SL)

12.4.5.1 *Procedures **(SL)***

The SAR should describe or state that the licensee (i.e., applicant as the licensee) will conduct all facility operations that are important to safety according to detailed written procedures that are based on and consistent with the operations descriptions in the operating procedures chapter of the SAR and the acceptance tests and maintenance program descriptions in the conduct of operations chapter of the SAR. The SAR should also state that proposed procedures and revisions will be reviewed and approved by the health, safety, and QA organizations that are independent from the operating management function.

The identification of proposed written procedures should include all routine and projected contingency operations. The applicant should also describe the review, change, and approval practices for all operating, maintenance, and testing procedures. This description may refer to the appropriate management controls addressed in Section 12.4.1.4 above.

The listing of operations requiring written procedures should include the following, as applicable to the ISFSI or MRS:

- all operations identified in the proposed technical specifications
- all operating, maintenance, testing, and surveillance functions important to safety

The procedures listed should clearly indicate, by title or subject, their purpose and applicability. The applicant should identify any standards used for the preparation of these procedures.

12.4.5.2 *Records **(SL)***

The SAR should describe the management system for maintaining records. This description may refer to the appropriate management controls addressed in Section 12.4.1.4 above. Although all records need not be maintained centrally, the management system should ensure that cognizance is being maintained of all records, the responsible staff, and locations.

Records stored in electronic media will generally be acceptable if the capability is maintained to produce legible, accurate, and complete records over the required retention period. The record format should include all pertinent information, such as stamps, initials, and signatures. The SAR should specify the retention period for each type of record because it varies depending on applicable regulatory requirements. The management system should also provide for adequate safeguards against tampering and loss of records over the retention period.

The SAR should identify, by type, the records to be maintained. Records maintained should include the following:

- construction records, as specified in applicable construction codes (e.g., ACI 349) and including as-built drawings and specifications, material certifications, and audit trail to the applicable SSCs, inspection records, test reports, and certifications (10 CFR 72.30(f)(2), 10 CFR 72.156, “Identification and control of materials, parts, and components,” and 10 CFR 72.174, “Quality assurance records”)

- as required in 10 CFR 72.30(f)(3), a list of the following, contained in a single document and updated no less than every 2 years:

  – all areas designated and formerly designated as restricted areas as defined under 10 CFR 20.1003, "Definitions"

  – all areas outside of restricted areas that require documentation under 10 CFR 72.30(f)(1) (see next entry)

- records of spills or other abnormal occurrences involving the spread of radiation in and around the facility, equipment, or site (10 CFR 72.30(f)(1))

- records of the cost estimate performed for the decommissioning funding plan or of the amount certified for decommissioning and records of the funding method used for ensuring funds, if either funding plan or certifications are used (10 CFR 72.30(f)(4)) (i.e., record copy of proposed decommissioning plan filed with license application, attached decommissioning funding plan, any modifications to these plans, and final decommissioning plan when prepared)

- receipt, inventory, disposal, acquisition, and transfer of all SNF, high-level radioactive waste (HLW) and reactor-related greater-than-Class-C (GTCC) waste in storage, as required in 10 CFR 72.72(a) (including provisions for duplicate records storage at different locations, in accordance with 10 CFR 72.72(d))

- records of physical inventories and current material control and accounting procedures (10 CFR 72.72(b) and 10 CFR 72.72(c))

- operating records, including principal maintenance, alternations, or additions made (10 CFR 72.70(b)(1) and 10 CFR 72.70(c)(4)(ii))

- records of off-normal occurrences and events associated with radioactive releases (10 CFR 72.44(d)(3))

- records of employee certification (10 CFR 72.44(b)(4))

- QA records (10 CFR 72.174)

- environmental survey records and environmental reports, including those related to the radiological environmental monitoring program (see SRP Sections 10A.4.2.5 and 10A.5.2.5 for a description of this program)

- radiation monitor readings or records (e.g., stripcharts or electronic results)

- radiation protection program records (per Subpart L, "Records," of 10 CFR Part 20, "Standards for Protection Against Radiation"), including those related to the following:

  – program contents, audits, and reviews
  – radiation surveys
  – determination of prior occupational dose
  – planned special exposures
  – individual (worker) monitoring results

  - dose to individual members of the public
  - radioactive waste disposal
  - tests of entry control devices for very high radiation areas

- records of changes to the physical security plan (10 CFR 72.44(e) and 10 CFR 72.186, “Change to Physical Security and Safeguards Contingency Plans”), and other physical security records (10 CFR 73.21 and 10 CFR 73.70, “Records”)

- records of occurrence and severity of natural phenomena (10 CFR 72.92, “Design Basis External Natural Events”)

- record copies of the following:

  - SAR, SAR updates, final SAR (10 CFR 72.70)

  - reports of accidental criticality or loss of special nuclear material (10 CFR 72.74 and 10 CFR 73.71, “Reporting of safeguards events”)

  - material status reports (10 CFR 72.76)

  - nuclear material transfer reports (10 CFR 72.78)

  - physical security plan (10 CFR 72.180, “Physical protection plan”)

  - “other” records and reports (10 CFR 72.82, “Inspections and tests”)

- report of preoperational test acceptance criteria and test results

- written procedures

The radiation protection records required by 10 CFR Part 20, Subpart L should incorporate the units of curie, rad, and rem, as applicable, including multiples or subdivisions of those units (e.g., megacurie, millicurie, millirem). Where dose is part of a record, the dose quantity used on the record (e.g., total effective dose equivalent, committed effective dose equivalent, shallow dose equivalent) should be clearly indicated. Chapter 10A of this SRP, particularly Sections 10A.4.4 and 10A.5.4, includes additional guidance regarding records as related to the licensee’s radiation protection and health physics programs and operations.

12.4.6 Personnel Selection, Training, and Certification (SL)

The SAR should describe the organization responsible for personnel selection, training, and certification. The SAR should also describe the program that will be established and implemented to ensure that personnel whose responsibilities include functions that are important to safety will be appropriately qualified and trained. The process of selecting and training security guards should be described. Chapter 10A of this SRP, particularly Sections 10A.4.4 and 10A.5.4, includes additional guidance regarding personnel selection, training, and certification that relate to the radiation protection and health physics organization personnel and radiation safety training for all licensee personnel.

12.4.6.1 *Personnel Organization **(SL)***

The SAR should include a discussion of the organization and management of the training component and should identify the personnel responsible for the development of training programs, conducting training and retraining of employees (including new employee orientations), and maintaining up-to-date records on the status of trained personnel.

12.4.6.2 *Selection and Training of Operating Personnel* ***(SL)***

The applicant should identify the functions that are important to safety and describe the qualifications for personnel performing those functions. These personnel qualifications should include the following:

- minimum qualification requirements for operating, technical, and maintenance supervisory personnel, including any qualification requirements identified in the evaluations throughout the SAR (e.g., certification requirements for individuals writing procedures for and performing leakage testing identified in the operating procedures and conduct of operations chapters of the SAR)

- qualifications, in resume form, of persons who will be assigned to managerial and technical positions.

The program description should identify the scope of operational and safety training. Operational training should include topics such as installation design and operations, instrumentation and control, methods of dealing with operating functions, decontamination procedures, and emergency procedures. Radiation safety training should include topics such as the nature and sources of radiation, methods of controlling exposure and contamination, radiation monitoring, shielding, dosimetry, biological effects, and criticality hazards control.

The SAR should list the type and level of training to be provided for each job description (personnel classification), including specific training provided to specific job descriptions. Alternatively, the SAR may describe the basis used to identify the type and level of training by job description.

The SAR should clearly identify the requirements for the certification of personnel who will operate equipment and controls that are important to safety. The requirements must address the physical condition and general health of personnel to be certified in accordance with 10 CFR 72.194, “Physical requirements.”

The SAR should describe methods of testing to determine the effectiveness of the training program. The applicant should evaluate the effectiveness of the training program against established objectives and criteria, identifying any standards used for development and implementation of the training program.

The SAR should describe the frequency of retraining, and the nature and duration of the retention of training and testing records. Retraining should be periodic and not less than every 2 years. Training records should be kept up to date and retained for a minimum of 3 years.

The SAR should describe implementation of the training program before conduct of operations involving radioactive material (i.e., preoperational training). The applicant should commit to a substantial completion of staff training and certification before the receipt of the radioactive material for storage.

The applicant should identify any standards used for the selection, training, and certification of personnel.

12.4.6.3 *Selection and Training of Security Guards **(SL)***

The SAR must describe the process by which security guards (including watchmen, armed response persons) are selected and qualified (10 CFR 73.55(c)(4)). This information may be submitted as part of the applicant's physical security plan.

The criteria used must conform to the general criteria for security personnel contained in Appendix B, "General Criteria for Security Personnel," to 10 CFR Part 73, "Physical Protection of Plants and Materials." RG 5.20, "Training, Equipping, and Qualifying of Guards and Watchmen," provides guidance in this area.

12.4.7 Emergency Planning (SL)

The purpose of the review of the applicant's emergency plan (EP) is to ensure that the plan (1) complies with regulatory requirements, (2) is based on the proposed ISFSI or MRS, and (3) provides acceptable hazards analysis. The emergency planning regulations in 10 CFR 72.32 have additional requirements for a MRS and an ISFSI that is licensed to process and/or repackage spent fuel. For the ease of clarification, the evaluation criteria for these additional requirements will be identified as "MRS" or "MRS only."

12.4.7.1 *Description of Facility and Site **(SL)***

The applicant should provide a concise description of all site features affecting emergency response, including communications and assessment centers, assembly and relocation areas, and emergency equipment storage areas. The EP should identify any additional site features related to the safety of site operations. Most of this information will be presented in the SAR's discussion of site characterization. However, supplemental information may be presented with the information on emergency planning.

The applicant may provide a detailed map of the site. An enlarged duplicate of the drawing suitable for use as a wall map may also be provided. The detailed map may be drawn to scale and show the following:

- ISFSI storage areas or storage structures, and any holding areas for loaded transportation packages
- MRS storage areas or storage structures, pool, dry transfer facilities, intermodal transfer stations, and any holding areas for loaded transportation packages
- onsite structures and adjacent structures with descriptive labels (and building numbers, if applicable)
- other major site features, such as administrative and public access areas
- bar scale in both meters and feet
- compass indicating north
- onsite roads and parking lots
- onsite routes for transferring material to and from storage

- site, controlled area, and restricted area boundaries, including locations of gates
- liquid retention tanks and ponds (include note if tanks or ponds are potentially contaminated
- roads, railroads, and navigable water in close proximity to the site
- rivers, lakes, streams, wetlands, or other ground water sources on site and adjacent to the site

12.4.7.2 *Description of the Area Near the Site* ***(SL)***

The EP should describe the principal characteristics of the area near the site (out to approximately 1.6 kilometers (1 mile)) that may impact emergency planning, such as impediments to emergency response to the site (e.g., drawbridges, rivers), or facilities that may pose a threat to the site (e.g., chemical plants, petroleum gas terminals). The applicant should provide a general map (an approximately 16-kilometer (10-mile) radius) and a U.S. Geological Survey topographical map. Although most of this information will be presented in the SAR's discussion of site characterization, supplemental information may be presented with the information on emergency planning.

The EP should include a map of the area surrounding the site (out to approximately 1.6 kilometers (1 mile)) that provides the following information:

- locations of population concentrations (such as towns, cities, office buildings, factories, arenas, stadiums, hospitals, nursing homes, and recreational areas)
- locations of facilities (such as schools, arenas, stadiums, nursing homes, hospitals, prisons)
- identification of primary routes for access of emergency equipment or for evacuation, as well as potential impediments to traffic flow (such as rivers, drawbridges, railroad grade crossings)
- locations of fire and police stations, hospitals, and other offsite emergency support organizations (specify whether offsite emergency support organizations received training to handle exposure to radioactive materials)

12.4.7.3 *Types of Accidents* ***(SL)***

The EP should identify and describe each type of accident for which actions may be needed to prevent or minimize exposure from radiation, radioactive materials, or both, to onsite personnel for the ISFSI and MRS. The accidents should be described in terms of the process and physical location where they could occur, how the accidents could occur (e.g., equipment malfunction, instrument failure, human error), possible contributing or complicating factors, and possible onsite consequences. The accident descriptions should include any radiological material releases that could impact emergency response efforts. Chapter 16, "Accident Analysis Evaluation," of this SRP describes the evaluation of this information.

12.4.7.4 *Classification of Accidents **(SL)***

NUREG-1140, “A Regulatory Analysis on Emergency Preparedness for Fuel Cycle and Other Radioactive Material Licensees,” issued January 1988, describes incidents involving radioactive material for an ISFSI. NUREG-1092, “Environmental Assessment for 10 CFR Part 72 ‘Licensing Requirements for the Independent Storage of Spent Fuel and High-Level Radioactive Waste,’” issued August 1984, describes postulated accidents for surface cask storage of canistered fuel for an MRS.

Regulations for ISFSIs located away from a reactor site require only one level of emergency classification: Alert.

Regulations for ISFSIs and MRSs authorized to process and repackage SNF have two classes of accidents: Alert or Site Area Emergency (SAE).

The EP should include the emergency action levels at which an Alert or SAE will be declared.

12.4.7.4.1 Alerts

An Alert is defined as an incident that has led to or could lead to a release of radioactive or other hazardous material to the environment, but the release is not expected to require a response by an offsite response organization to protect offsite individuals. The EP should identify events that could lead to initiation of an alert, such as the following:

- severe natural phenomena (e.g., beyond-design-basis earthquake, hurricanes) or other incidents (e.g., fire, release of flammable gas) that have the potential to affect the confinement barrier
- indications of severe loss of control (e.g., radiation or contamination levels at the facility that are a factor of 100 over normal levels)
- a security compromise lasting more than 15 minutes
- accidental release of radioactivity due to failure of the confinement barrier
- other conditions that warrant precautionary activation of the licensee’s emergency response organization

The plan should include a description of the applicant’s emergency response organization mobilization, steps taken to mitigate consequences of the emergency, and steps to be taken to escalate the classification, if necessary.

12.4.7.4.2 Site Area Emergency

An SAE is defined as an incident that has led to or could lead to a significant release of radioactive or hazardous material and that could require a response by an offsite organization to protect offsite personnel. The EP should identify the events that could initiate an SAE, such as the following:

- a compromise to systems or SSCs important to safety or a compromise to the integrity of SNF, HLW or reactor-related GTCC because of severe natural phenomena (e.g., earthquake, flood, tsunamis) or severe incidents (e.g., aircraft crash into the facility, explosion, fire)

- imminent or actual loss of physical control of the facility

- rupture of the storage container confinement barrier and release of radioactivity outside of outer confinement barrier (e.g., loading facility building, SNF building)

The EP should include a description of the applicant's emergency response organization mobilization, steps taken to mitigate consequences of the emergency, and procedures to notify offsite response organizations (fire, medical, police).

12.4.7.5 *Detection of Accidents* ***(SL)***

The EP should describe the means of detecting each type of accident identified in the plan (e.g., visual observation, monitors, detectors, process alarms).

The EP should also describe the means to notify the operating staff of any abnormal operating condition or of any other danger to safe operation (e.g., a severe weather warning).

12.4.7.6 *Mitigation of Consequences* ***(SL)***

For the events identified in Section 12.4.7.3 above, the EP should briefly describe the means and equipment provided for mitigating the consequences of each type of accident. The plan should include the mitigation of consequences to workers on site. Mitigating actions could include steps to reduce or stop any releases and steps to protect personnel and environment (e.g., evacuation, shelter, decontamination).

12.4.7.6.1 Limiting Actions

The EP must describe the means and equipment provided for limiting the consequences of each type of accident identified in the plan (e.g.; fire detection and suppression systems, automatic shutoff of process or ventilation flow). The plan should address the actions and systems in place to reduce the magnitude or the effect of a radioactive or hazardous material release that has occurred (e.g., filtration or holdup systems, use of water sprays on airborne releases). The plan should include actions to be taken to limit and mitigate the consequences to public and workers. Based upon the type of emergency, the plan should describe the criteria for the shutdown of systems or the facility and the steps to be taken to ensure a safe, orderly shutdown and the approximate time required for a safe shutdown.

12.4.7.6.2 Onsite Protective Actions

The EP should describe the nature of onsite protective actions, criteria for implementing those actions, the areas involved, and the procedures to notify potentially affected persons. The plan should allow for the timely relocation of onsite personnel, the effective use of protective equipment and supplies, and the use of appropriate contamination control measures.

The EP should describe the means for controlling and/or minimizing radiological exposures for emergency response workers. The onsite exposure guidelines should be consistent with the Environmental Protection Agency's (EPA's) "PAG Manual: Protective Action Guides and Planning Guidance for Radiological Incidents," issued January 2017, Section 3.1, "Controlling Occupational

Exposure and Doses to Emergency Workers," to be used in actions to control fires, stop releases, or protect the facilities.

The plan should include methods for onsite non-essential personnel evacuation and accountability, such as the following:

- criteria for ordering a site evacuation at the site area emergency classification
- means and timely notification of onsite persons impacted
- search and rescue
- locations of onsite and offsite assembly areas
- evacuation routes and means for transporting onsite personnel (e.g., privately owned vehicles, buses, company vehicles)
- monitoring of evacuees for contamination and control measures if contamination is found
- criteria for command center and assembly area evacuation and re-establishment at an alternate location
- means for evacuating and treating onsite injured personnel, including potentially contaminated personnel
- provisions for determining and maintaining accountability of assembled and evacuated personnel, and for identifying and determining the locations of personnel that were not evacuated

The EP should describe provisions for preventing further spread of radioactive materials and for minimizing personnel exposures from radioactive materials.

The EP should describe provisions for determining the doses and dose commitments from external radiation exposure and internally deposited radioactive material received by emergency response personnel, including personnel from offsite emergency response organizations (fire, medical, police).

The EP should describe arrangements made for hospital and medical services and their capabilities to evaluate and treat contaminated, injured persons and injuries involving radiation, and radioactive materials. The medical facility description should include capabilities to control any contamination that may be associated with the physical injuries. The EP should specify how injured personnel who are potentially contaminated will be transported to offsite medical facilities. The commitment to provide ambulance and hospital personnel with health physics support should be included.

Emergency Response Equipment and Facilities

The EP should describe the onsite equipment and facilities designated for use during emergencies. The plan should describe an emergency facility from which evaluation and coordination of all licensee activities related to an emergency is to be carried out, and from which the licensee shall provide information to Federal, State and local authorities responding to emergencies. The notification and coordination must be planned so that unavailability of some personnel, parts of the facility, and some equipment will not prevent the notification and coordination.

The plan should identify locations from which licensee emergency workers would be dispatched to perform radiation surveys, damage assessment, emergency repair, or other mitigating tasks.

The EP should describe the protective equipment and supplies available to emergency-response personnel. Types of equipment and supplies may include the following:

- individual respiratory equipment
- protective clothing
- firefighting equipment and gear
- supplemental lighting
- medical supplies
- contamination control and decontamination equipment
- communications equipment
- radiation detection equipment (e.g., radiation meters, air samplers, dosimeters)
-

The EP should describe locations of emergency equipment and supplies, and the means for distributing these items. The plan should also include inventory lists indicating the emergency equipment and supplies provided at specified locations.

The EP should describe the primary and alternate onsite and offsite communication systems that would be used to transmit and receive information throughout the emergency. The plan should state the planned frequency of operational tests. A backup means of offsite communication to a commercial telephone should be provided for the notification of emergencies and requests for assistance. The frequency of operability checks should be stated.

12.4.7.7 *Assessment of Releases (SL)*

The EP should discuss the actions to be taken to determine the extent of the problem and to decide what corrective actions may be required for each class of emergency. This should include the types and methods of onsite and offsite sampling and monitoring in case of a release of radioactive or other hazardous material. The EP should describe the provisions for projection of offsite radiation exposures.

12.4.7.8 *Responsibilities* ***(SL)***

The EP should describe the emergency organization to be activated on site for possible events and offsite for augmentation and support. The plan should delineate the authorities and responsibilities of key positions and groups and identify the communication chain for notifying and mobilizing personnel during normal and nonworking hours. Personnel with the responsibility for promptly notifying offsite response organizations and the NRC should be identified.

The EP should identify by position those with responsibility to declare an emergency and to initiate the appropriate response. The EP should include provisions for an annual review and audit of the emergency preparedness program to ensure that the program remains adequate. Elements of the audit should include a review of the following:

- EP and associated procedures
- Radiological emergency response training activities (MRS only)
- records of emergency facilities, equipment, and supplies
- records associated with offsite response agencies interface (such as radiological emergency response training (MRS only) and letters of agreement)
- exercises, drills, communications, and inventory checks
- activation of the EP since the last audit

12.4.7.8.1 Onsite Emergency Response Organization

The EP should identify the onsite emergency response organization for the facility, including during periods such as holidays, weekends, and extended periods when normal operations are not being conducted. If the organization is activated in phases, the plan should describe the basic organization and each additional component that may be activated to augment the organization. The plan should clearly state the minimum level of staffing needed to effectively implement the plan for each period or phase described.

12.4.7.8.2 Direction and Coordination

The EP should designate the position of the person, and alternate(s), with the principal responsibility for implementing and directing the emergency response. This person's duties and authorities would include the following:

- control of the situation
- escalation or termination of the emergency condition
- coordination of the staff and offsite personnel who augment the staff
- communication with parties requesting information regarding the event
- request of support from offsite response organizations

The plan should also describe this person's authority to delegate responsibilities and the individuals who may be delegated certain emergency responsibilities.

12.4.7.8.3 Onsite Staff Emergency Assignments

The EP should specify the organizational group or groups assigned to the functional areas of emergency activity listed below. The plan should also describe strategies for staffing these positions if the emergency lasts longer than one working shift. The duties, authorities, and interface with other groups and offsite assistance should be described. The organizational groups should provide support in the following areas:

- facility systems operations
- personnel evacuation and accountability
- search and rescue operations
- first aid
- communications
- radiological survey and assessment (both on site and off site)

- personnel and facility decontamination
- facility repair and damage control
- post event assessment
- recordkeeping
- media contact

12.4.7.8.4 Emergency Response Records

The EP should describe the assignment of responsibility for reporting and recording incidents of abnormal operation, equipment failure, and accidents that led to a facility emergency. The EP records to be maintained should include the following information:

- cause of the incident
- personnel and equipment involved
- extent of injury and damage (on site and off site) as a result of the incident
- locations of contamination with the final decontamination survey results
- corrective actions taken to terminate the emergency
- actions taken or planned to prevent a recurrence of the incident
- onsite and offsite assistance requested and received
- any program changes as a resulting from a critique of emergency response activities

The records associated with emergency planning that will be kept should also be described. These should include the following:

- training and retraining (including lesson plans and test questions)

- drills, exercises, and related critiques

- inventory and locations of emergency equipment and supplies

- maintenance, surveillance, calibration, and testing of emergency equipment and supplies

- letters of agreement with offsite support organizations

- reviews and updates of the EP

- notification of onsite personnel and offsite response organizations affected by an update of the plan or its implementing procedures

12.4.7.8.5 Responsibilities at Site of Government Agencies

The EP should identify the principal State agency and other government (local, county, State, and Federal) agencies or organizations with authority for emergency preparedness and response. The plan should list the location and specific response capabilities, in terms of personnel and resources, of these agencies and organizations.

12.4.7.9 *Notification and Coordination* ***(SL)***

The EP should describe the means used to activate the emergency response organization for each class of emergency during both regular and nonregular hours. The plan should describe the means provided to detect and notify the licensee's operating staff of any abnormal operating conditions or of any danger to safe operations (e.g., a severe weather warning). The means to promptly notify offsite response organizations and the NRC should be described.

The EP should describe the ability to request offsite assistance, including medical assistance for the treatment of contaminated injured onsite workers. The plan should include the commitment to notify the NRC response center immediately after notification of local authorities but no later than 1 hour after an emergency is declared.

12.4.7.10 *Information to be Communicated* ***(SL)***

The EP should describe the type of information to be communicated to offsite response organizations and the NRC. The types of information to be communicated should include the status of the facility, if a release of radioactive material is occurring or could occur, and recommendations for protective actions that may be implemented by the offsite response organization responsible for implementing protective actions. The recommended approach is to have estimated a range of potential source terms for each accident type in the planning, and then decide in the planning what recommendations would be made to offsite response organizations for each accident type. The plan should include a standard reporting checklist to facilitate timely notification for each postulated accident.

12.4.7.11 *Training* ***(SL)***

The EP should include a description of the training provided to licensee staff on how to respond to an emergency. The plan should also include special instructions and orientations provided to offsite emergency response organizations. For an MRS only, the plan should describe emergency radiological response training provided for offsite response organizations that may be called to assist in an emergency onsite. The plan should include a description of training requirements for each position in the emergency organization, frequency of retraining, and training of onsite personnel who are not members of the emergency response staff.

12.4.7.12 *Safe Condition* ***(SL)***

The EP should generally describe procedures for restoring the facility to a safe status after an accident and recovery plans. The plan should describe the position/title, authority and responsibilities of individuals who will fill key positions in the facility recovery organization. This organization shall include technical personnel with responsibilities to develop, evaluate and direct recovery and reentry operations. The plan should describe requirements for returning emergency equipment and supplies used during an accident to a state of readiness.

12.4.7.13 *Exercises* ***(SL)***

The EP should describe the provisions for periodic drills and exercises. Communications checks with offsite agencies and radiological/health physics, medical, and fire drills should be performed at the interval established in 10 CFR 72.32(a) or 10 CFR 72.32(b).

The biennial onsite exercise required by 10 CFR 72.32, “Emergency Plan,” should test the effectiveness of the personnel, plan, procedures, and readiness of facilities, equipment, supplies, and instrumentation.

The applicant should invite offsite response organizations to participate in the periodic drills and exercises although recommended, their participation is not required. The EP should describe who has authority to develop the exercises, requirements for nonparticipating observers to evaluate the effectiveness of the exercise, the need for a critique of the exercise, and, if deficiencies are found, how they will be corrected.

12.4.7.14 *Hazardous Chemicals* ***(SL)***

The EP must certify compliance with the Emergency Planning and Community Right-to-Know Act of 1986, with respect to any hazardous materials processed at the facility (10 CFR 72.32(a)(13) and 10 CFR 72.32(b)(13)).

12.4.7.15 *Comments on the Emergency Plan* ***(SL)***

The EP should contain requirements for obtaining comments from offsite response organizations on the initial plan before submittal to the NRC with the license application. The licensee should communicate changes to the EP to the affected offsite response organizations. Letters of agreement with offsite response organizations should be reviewed annually and renewed on a periodic basis. Letters of agreement may be included in the EP or maintained separately.

12.4.7.16 *Offsite Assistance* ***(SL)***

The EP should describe provisions and arrangements for assistance from offsite response organizations during and after an emergency. The licensee should clearly communicate exposure guidelines to offsite emergency response personnel. The plan should identify the services to be performed, means of communication and notification, and types of agreements that are in place for the following:

- medical treatment facilities
- first aid personnel and ambulance service, as needed
- fire fighters
- law enforcement assistance

The EP should describe the measures that will be taken to ensure that offsite response organizations maintain an awareness of their respective roles in an emergency and have the necessary equipment, supplies, and periodic radiological emergency response training (MRS only) to carry out their emergency response functions. The plan should describe any provisions to suspend security or safeguards measures for site access during an emergency.

The licensee should offer to meet at least annually with each offsite response organization to review items of mutual interest, including relevant changes to the EP. The licensee should discuss the emergency action level scheme, notification procedures, and overall response coordination process during these meetings.

12.4.7.17 *Public Information (SL)*

The EP should describe how information to the news media and the public would be disseminated during an emergency.

12.4.8 Physical Security and Safeguards Contingency Plans (SL)

The SAR must contain a physical protection plan as required by 10 CFR 72.180 and a safeguards contingency plan as required by 10 CFR 72.184, "Safeguards contingency plan." Security plans are Safeguards Information and must describe how the applicant will comply with the applicable requirements in 10 CFR Part 73 and the requirements imposed by NRC orders for additional security measures. The EP should provide for the physical security of materials during transport to and from the ISFSI or MRS, as well as during the storage period. The plan must establish a security organization and include the following:

- physical protection design features
- safeguard contingency plan
- guard training plan
- tests, inspections, audits, and other means to demonstrate compliance

If the application is from the DOE, the SAR must include (1) a description of the physical security plan for protection against radiological sabotage (as required by Subpart H, "Physical Protection," of 10 CFR Part 72), and (2) a certification that the plan will provide safeguards at the ISFSI or MRS that meet the requirements for comparable surface DOE facilities (required by 10 CFR 72.24(o)).

The safeguards contingency plan must comply with the format and content requirements of Appendix C, "Licensee Safeguards Contingency Plans," to 10 CFR Part 73. An acceptable plan must contain (1) a predetermined set of decisions and actions to satisfy stated objectives; (2) an identification of the data, criteria, procedures, and mechanisms necessary to efficiently implement the decisions; and (3) a stipulation of the individual, group, or organizational entity responsible for each decision and action.

RG 5.55, "Standard Format and Content of Safeguards Contingency Plans for Fuel Cycle Facilities," provides guidance on safeguards contingency plans that are specifically applicable to DSF facilities.

12.5 Review Procedures

To begin the conduct of operations review, determine whether the applicant has submitted the respective elements described in RG 3.61, "Standard Format and Content for a Topical Safety Analysis Report for a Spent Fuel Dry Storage Cask," RG 3.62, and RG 3.48. The review guidance provided in the following sections is predicated on the reviewer having access to the required products for the review and is based on lessons learned that should be applied in evaluating the submitted documentation.

Figure 12-1 Overview of Conduct of Operations evaluation shows the interrelationship between the conduct of operations evaluation and the other areas of review described in this SRP.

An applicant's conduct of operation is, in a significant way, implemented by the applicant's procedures. Therefore, the reviewer of this chapter should coordinate with the reviewer of the operating procedures (SRP Chapter 11, "Operation Procedures and Systems Evaluation") to ensure that there are no inconsistencies.

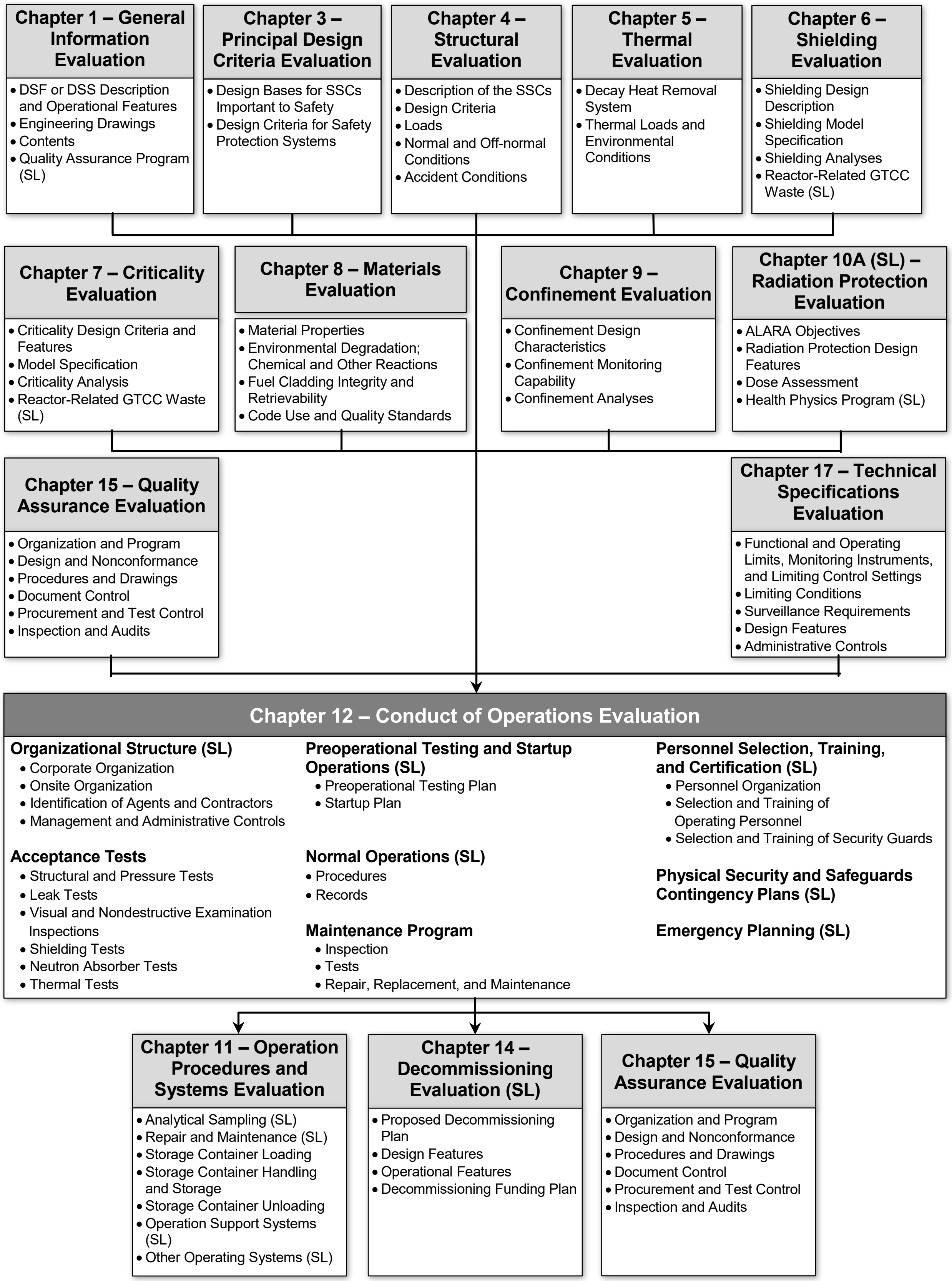

Figure 12-2 Overview of Conduct of Operations Evaluation

12.5.1 Organizational Structure (SL)

In addition to the guidance here, see also Chapter 10A of this SRP, specifically Sections 10A.5.4 and 10A.5.4.1, for additional guidance regarding the radiation protection and health physics aspects of the organizational structure and staffing.

12.5.1.1 *Corporate Organization* ***(SL)***

Ensure that the relationship between the corporate organizations and the site organizations is clearly defined. Review the submitted documentation to gain an understanding of the delineation of authority and responsibilities regarding site activities. Ensure that the SAR specifies the frequency and scope of any audits or inspections conducted by the corporate organizations.

12.5.1.2 *Onsite Organization* ***(SL)***

Review the material to gain a clear understanding of the distribution of responsibility to specific parts of the site organization and ensure that the site organization and the distribution of responsibilities for functions important to safety are clearly evident. Verify that the functions of radiation protection, nuclear criticality safety, and other safety entities are organizationally separate from the entity responsible for facility operations.

Determine whether the onsite organization includes a safety committee (or equivalent function) with appropriate representation and responsibilities. In making this determination, consider whether membership includes representatives from operating and safety support organizations. Ensure that the safety committee has appropriate review and approval authority and procedures for the systematic review of proposed operations and changes. Confirm that the committee reports directly to the facility manager or other senior management.

In reviewing the proposed staffing levels and descriptions, consider the extent of expected operations. For example, in cases where the full spectrum of radioactive materials (e.g., full range of fuel types, HLW, or reactor-related GTCC waste) to be stored and the potential storage configurations and the kinds of handling operations is limited in scope or the applicant's evaluations significantly bound the contents spectrum and storage configurations and configurations of handling operations, the level of onsite technical support (e.g., in areas such as nuclear criticality safety or structural design analysis) can be lower than in cases where the spectrum of contents and the potential storage configurations are not limited in scope.

12.5.1.3 *Identification of Agents and Contractors* ***(SL)***

Verify that the SAR identifies the prime agents or contractors for the design, construction, and operation of the installation. Verify that the SAR identifies all principal consultants and outside service organizations, including those providing QA services. Confirm that the SAR clearly defines the division and assignments of responsibilities among those parties.

12.5.1.4 *Management and Administrative Controls* ***(SL)***

Ensure that the applicant paid adequate attention to a proposed system of management and administrative controls. Verify that the SAR addresses each of the system elements identified in the acceptance criteria for management and administrative controls (Section 12.4.1.4 above). Pay particular attention to the proposed system for procedures, including provisions for initial preparation, review, change, and approval.

12.5.2 Acceptance Tests

The review procedures in this SRP chapter are focused on the testing of the storage containers that are loaded with the proposed radioactive materials contents. Additional tests may be needed for specific license applications for DSF SSCs or features that perform important functions (e.g., shielding, subcriticality, and confinement of radioactive materials, including wastes generated from DSF operations) at the site to ensure that the DSF design and operations meet regulatory requirements.

The review procedures described in this section are presented in a format intended to facilitate a single, independent review. Although one or more individual(s) may be tasked with preparing the corresponding section of the safety evaluation report (SER) related to the proposed acceptance tests, all review team members should examine the related information presented in the SAR. Information in the SAR related to the acceptance tests may be located in the chapters related to specific disciplines (e.g., those related to the thermal evaluation) or in the chapter of the SAR on conduct of operations evaluation, or elsewhere. Devote special attention to those tests (or the lack of tests) that affect the respective functional area of review. If the descriptions included in the SAR are not sufficiently detailed to allow a complete evaluation concerning fulfillment of the acceptance criteria, request additional information from the applicant.

In general, applicants state that they will design, construct, and test the DSS or DSF under review to the codes and standards identified in the chapter of the SAR on principal design criteria. The NRC does not generally review detailed test procedures as part of the licensing process; however, the applicant is expected to describe (in the SAR) the essential elements of the proposed test programs. The staff may inspect selected portions of test procedures as part of its onsite activities.

The following subsections provide representative examples of acceptance tests that should be described in the SAR. Review the description of each test to ensure that the applicant has identified the purpose of the test, explained the proposed test method (including any applicable standard to which the test will be performed), defined the acceptance criteria and bases for the test, and described the actions to be taken if the acceptance criteria are not satisfied.

The following guidance is presented on the basis of tests the NRC deems acceptable. The guidance is based on operational experience and the knowledge from past reviews. Alternative tests and criteria may be used if the SAR provides appropriate explanation and adequate justification. Additional tests and criteria may be needed, depending on the operational experience and uniqueness of the proposed DSS or DSF design.

12.5.2.1 *Structural and Pressure Tests*

Lifting trunnions should be fabricated and tested in accordance with ANSI N14.6, "Radioactive Materials—Special Lifting Devices for Shipping Containers Weighing 10,000 pounds (4,500 kilograms) or More." For a DSS or DSF where operations involve movement of SSCs into or out of a pool (e.g., SNF pool), site-specific details of the pool and lifting procedures may enable the DSS or DSF storage container to be considered a noncritical load, as defined by this standard. Generally, however, the DSS or DSF storage container is considered a critical load during its handling in the pool. Consequently, trunnion testing should be performed at a minimum of 150 percent of the maximum service load if redundant lifting is employed or at a minimum of 300 percent of the service load if nonredundant lifting applies. These load tests should be performed to ensure that the trunnions and DSS or DSF storage container are conservatively constructed and provide an adequate margin of safety when filled with the proposed radioactive material contents (e.g., SNF). Trunnion load testing should also be performed annually for the transfer cask, for DSS or DSF designs that use them, and at least 1 year before use for the storage container. Load testing of integral trunnions is not required once the loaded storage container has been placed on the pad. Ensure that the SAR chapter on technical specifications and operating controls and limits includes any restrictions on storage container lifting resulting from these tests. Ensure that the SAR explicitly states the testing values. Periodic NDE, in lieu of annual load tests, is acceptable for the trunnions provided that other conditions, as specified in ANSI N14.6, are also met.

The entire storage container confinement boundary should be pressure tested hydrostatically or pneumatically to 125 or 110 percent of the design pressure, respectively. The pressure test should be performed in accordance with the governing code associated with the confinement boundary, which typically has been ASME B&PV Code, Section III, Division 1, Subsection NB or NC for DSSs. The test pressure should be maintained for a minimum of 10 minutes, after which a visual inspection should be performed to detect any leakage. Ensure that the sections in the SAR describing the acceptance tests and maintenance programs clearly specify the hydrostatic and pneumatic test pressures. The helium leakage test, per ANSI N14.5, is not considered as a substitute for the ASME B&PV Code-required pressure test, and, conversely, the ASME B&PV Code-required pressure test is not a substitute for the helium leakage test.

Some storage containers (or DSS or DSF SSCs) include a neutron shielding material that may off-gas at higher temperatures. Such material is usually contained inside a thin steel shell to prevent loss of mass and provide protection from minor accidents and natural phenomenon events. Rupture disks or relief valves are generally provided to prevent catastrophic failure of this shell. The shell should be tested to 125 percent of the rupture disk burst pressure, which is usually equivalent to 125 percent of the shell design pressure. Verify that the SAR clearly specifies the burst pressure for the rupture disk, along with its coincident burst temperature and tolerance on burst pressure.

Some storage container designs use ferritic steels that are subject to brittle fracture failures at low temperature. ASME B&PV Code, Section II, “Materials,” Part A, “Ferrous Materials Specifications,” contains procedures for testing ferritic steel used in low-temperature applications. NUREG/CR-1815, “Recommendations for Protecting Against Failure by Brittle Fracture in Ferritic Steel Shipping Containers Up to Four Inches Thick,” issued June 1981, provides staff guidance concerning materials and thickness ranges subject to brittle fracture testing. On the basis of guidance in NUREG/CR-1815, Section 5.1.1, the NRC has established two methods for identifying suitable materials:

1. The nil-ductility transition temperature should be determined by either direct measurement (American Society for Testing and Materials (ASTM) E208, “Standard Test Method for Conducting Drop-Weight Test to Determine Nil-Ductility Transition Temperature of Ferritic Steels”) or indirect measurement (ASTM E604, “Standard Test Method for Dynamic Tear Testing of Metallic Materials”), and the minimum operating temperature of the steel should be specified as 28 degrees Celsius (50 degrees Fahrenheit) higher than the nil-ductility transition.

2. The NRC staff accepts ASME Charpy testing procedures for verification of the material’s minimum absorbed energy. Acceptable energy absorption values and test temperatures of Charpy V-notch impact tests are listed in the ASME B&PV Code, Section II, SA-20, “Specifications for General Requirements for Steel Plates for Pressure Vessels,”

Table A1.15. Coordinate with the thermal reviewer (SRP Chapter 5, “Thermal Evaluation”) to ensure that the applicant selected the correct temperatures for the tests and that the SAR specifies the method of testing. For storage containers (or DSS or DSF SSCs) with ferritic steel walls thicker than 102 millimeters (4 inches), follow the guidance provided in NUREG/CR-3826, “Recommendations for Protecting Against Failure by Brittle Fracture in Ferritic Steel Shipping Containers Greater than Four Inches Thick,” issued July 1984.

12.5.2.2 *Leak Tests*

Confirm that the applicant has described the leak tests to be performed on all confinement boundaries except as excluded in Section 8.5.3.3 of this SRP, which only applies to the closure welds typically made in the field. Leak testing should show that the inner closure weld of the storage container lid and primary welds of the vent and drain port cover plates meet the leakage limit. For all-welded confinement boundaries, the NRC has, with adequate justification, considered it acceptable for licensees and CoC holders to omit leak testing of the second (i.e., redundant) welds associated with the lid and its corresponding vent and drain port cover plates (see Figures 8-2 and 8-3 of this SRP). As shown in the figures, the redundant welds are not pressurized (or potentially pressurized because of closure valves, as described in Section 8.5.3.3.2) at the time of welding. A fabrication leak test should be performed on every storage container in the shop to ensure that the tested leakage rate meets the appropriate design leakage rate criteria (and regulatory criteria). Leak tests of the confinement boundary should be performed during the fabrication process such that subsequent fabrication procedures do not adversely affect the integrity of the confinement boundary.

Leakage criteria in units of Pascal cubic meter per second or that reference cubic centimeters per second should be at least as restrictive as those specified in the principal design criteria provided in the SAR. The SAR should also indicate the general testing methods (e.g., pressure increase, mass spectrometer) and required sensitivities. If storage container closure depends on more than one seal (e.g., lid, vent port, drain port), the leakage criteria should ensure that the total leakage is within the design requirements. Leak testing should be conducted in accordance with ANSI N14.5.

12.5.2.3 *Visual and Nondestructive Examination Inspections*

Verify that the applicant will fabricate and examine storage container components in accordance with an accepted design standard such as ASME B&PV Code, Section III or VIII. These sections define the examination requirements mentioned in Section II; Section V, “Nondestructive Examination”; and Section IX, “Welding and Brazing Qualification.” The following guidance assumes that the ASME B&PV Code is applicable to the storage container being reviewed.

Confirm that the NDE of weldments is well characterized on drawings, using standard NDE symbols and notations (see American Welding Society (AWS) A2.4, “Standard Symbols for Welding, Brazing, and Nondestructive Examination”). Verify that each fabricator is required to establish and document a detailed, written weld inspection plan in accordance with an approved QA program that complies with Subpart G, “Quality Assurance,” of 10 CFR Part 72. Verify that the inspection plan includes visual, liquid (dye) penetrant (PT), magnetic particle (MT), ultrasonic (UT), and radiographic (RT) testing, as applicable. Confirm that the inspection plan identifies welds to be examined, the examination sequence, type of examination, and the appropriate acceptance criteria as defined by either the ASME B&PV Code or an alternative approach proposed and justified by the applicant. Inspection personnel should be qualified, in accordance with the current revision of the American Society for Nondestructive Testing (ASNT)

Recommended Practice No. SNT-TC-1A, “Personnel Qualification and Certification in Nondestructive Testing,” as specified by the ASME B&PV Code. All weld-related NDE should be performed in accordance with written and approved procedures. Fabrication controls and specifications should be in place and field tested to prevent postwelding operations (such as grinding) from compromising the design requirements (such as wall thickness).

Verify that confinement boundary nonclosure welds meet the requirements of ASME B&PV Code, Section III, Division 1, Subsections NB or NC, Article NB/NC-5200, “Required Examination of Welds for Fabrication and Preservice Baseline.” This section requires volumetric examination and either PT or MT for all Category A and most Category B or Category C welded joints in vessels, and longitudinal or full-penetration welded joints in other components. The ASME-approved specifications for RT, UT, PT, and MT are detailed in ASME B&PV Code, Section V, Articles 2, 4, 6, and 7, respectively.

Confirm that the acceptance standards for nondestructive testing are in accordance with ASME B&PV Code, Section III, Division 1, Subsection NB or NC-5300, “Acceptance Standards.” Testers should reject unacceptable imperfections (such as a crack, a zone of incomplete fusion or penetration, elongated indications with lengths greater than specified limits, and rounded indications in excess of the limits in ASME B&PV Code, Section III, Division 1, Appendix VI). Repaired welds should be reexamined in accordance with the original examination method and associated acceptance criteria.

For confinement welds that cannot be volumetrically examined using RT, the licensee may use 100 percent UT. The ASME-approved UT specifications are detailed in ASME B&PV Code, Section V, Article 4. Ensure that acceptance criteria are defined in accordance with ASME B&PV Code, Section III, Division 1, Subsection NB or NC-5330, “Ultrasonic Acceptance Standards.” Cracks, lack of fusion, or incomplete penetration are unacceptable, regardless of length.

The NRC has accepted multiple surface examinations of welds, combined with helium leak tests for inspecting the final redundant seal welded closures.

For storage container internals, confirm that the licensee will perform all NDE testing in accordance with ASME B&PV Code, Section III, Division 1, Subsection NG.

Verify that nonconfinement welds will meet the requirements of ASME B&PV Code, Section III, Subsection NF, or Section VIII, Division 1, as applicable. Welds on internal components (e.g., baskets) should meet the requirements of ASME B&PV Code, Section III, Subsection NG. The required volumetric examination of welds is either RT or UT, as discussed in ASME B&PV Code, Section III, NF-5200, “Required Examination of Welds,” and Section VIII, UW-11. The appropriate specifications from ASME B&PV Code, Section V, are invoked in Article 2 for RT and in Article 5 for UT. Acceptance standards for RT are detailed in ASME B&PV Code, Section III, Subsection NF, NF-5320, “Radiographic Acceptance Standards,” and for UT in NF-5330, “Ultrasonic Acceptance Standards.” For Section VIII weldments, ensure that the RT acceptance criteria are in accordance with ASME B&PV Code, Section VIII, Division 1, UW-51, and the repair of unacceptable defects is in accordance with UW-38. Repaired welds should be reexamined in accordance with the original acceptance criteria.

Nonconfinement welds that cannot be examined using RT should undergo UT in accordance with ASME B&PV Code, Section V, Article 4. Ensure that acceptance criteria are in accordance with ASME B&PV Code, Section VIII, Division 1, UW-53 and Appendix 12, and the repair of unacceptable defects is in accordance with UW-38. Repaired welds should be reexamined in accordance with the original examination methods and associated acceptance criteria. If applicable, the SAR should also justify the rationale for not requiring RT examination of these welds.

Verify that nonconfinement welds for storage container components that are designed and fabricated in accordance with ASME B&PV Code, Section III, that cannot be examined using RT or UT undergo PT or MT examination in accordance with ASME B&PV Code, Section V, Articles 6 and 7, respectively. Ensure that acceptance criteria are in accordance with Articles NF-5350, "Liquid Penetrant Acceptance Standards," and NF-5340, "Magnetic Particle Acceptance Standards," respectively. Repaired welds should be reexamined in accordance with the original acceptance criteria. If applicable, the SAR should also justify the rationale for not requiring volumetric inspection techniques (RT or UT) for these welds.

Nonconfinement welds may also be welded, repaired, and examined in accordance with AWS D1.1, "Structural Welding Code—Steel"; D1.3, "Structural Welding Code—Sheet Steel"; and D1.6, "Structural Welding Code—Stainless Steel." Confirm that the design drawings call out the use of these standards.

Finished surfaces of the storage container should be visually examined in accordance with the ASME B&PV Code Section V, Article 9. For welds examined using visual testing, ensure that the acceptance criteria are in accordance with ASME B&PV Code, Section VIII, Division 1, UW-35 and UW-36, or NF-5360, "Acceptance Standards for Visual Examination of Welds." Note that O-ring seating, such as for a bolted lid cask design, may have surface finish acceptance criteria defined by the O-ring manufacturer.

Verify that the acceptance tests include the use of PT to detect discontinuities (such as cracks, seams, laps, laminations, and porosity) that open to the surface of nonporous metals. PT should be performed in accordance with ASME B&PV Code, Section V, Article 6. Ensure also that acceptance criteria for PT examination of confinement welds are in accordance with ASME B&PV Code, Section III, Subsection NB/NC, Article NB/NC-5350. Ensure that repair procedures are in accordance with ASME B&PV Code, Section III, Article NB/NC-4450, "Repair of Weld Metal Defects." Ensure that acceptance criteria for PT examination of nonconfinement welds are in accordance with ASME B&PV Code, Section VIII, Division 1, Appendix 8, or NF-5350. Ensure that repair procedures are in accordance with ASME B&PV Code, Section III or NF-2500, "Examination and Repair of Material," and NF-4450, "Repair of Weld Material Defects."

12.5.2.4 *Shielding Tests*

The materials that comprise the DSS or DSF SSCs should sufficiently maintain their physical and mechanical properties during all conditions of operations. DSS or DSF gamma shielding materials (e.g., lead, steel, and concrete) should not experience cracks, pinholes, uncontrollable voids, slumping, or loss of shielding effectiveness to an extent that compromises safety. The shield should perform its intended function throughout the licensed or certified period of storage operations.

DSS or DSF materials used for neutron shielding should be designed to perform their safety function without significant degradation, gas release, or physical alteration for the full term of the licensed or certified period of storage operations. Tests are required to ensure these conditions are met.

Tests of the effectiveness of both the gamma and neutron shielding may be required if, for example, the DSS or DSF design includes materials such as poured lead for gamma shielding or a special (polymer-based) neutron absorbing material. In such instances, verify that the SAR describes any scanning or probing with an auxiliary source for the purpose of characterizing the shielding effectiveness. This shield testing should be done for every DSS or DSF SSC that uses these kinds of shielding materials to demonstrate proper fabrication in accordance with the design drawings. Even in instances were these shields may be installed in the DSS or DSF SSCs in prefabricated pieces, verify that the SAR includes SSC fabrication descriptions and tests to ensure fit-up of the prefabricated shielding materials with the SSCs. Such descriptions and tests should ensure that the prefabricated materials perform as designed, have the necessary dimensional and material properties, and that fit-up precludes unanalyzed streaming paths in the SSCs. For materials such as polymer-based neutron shields, tests may need to include qualifications testing of the fabrication process to ensure proper material specifications and uniformity of these specifications and material composition throughout the material.

Verify that shielding effectiveness tests include dose rate scans over the extent of the SSC surfaces where the shielding materials are present. Ensure that the tests use appropriate acceptance criteria that are based on the design specifications of the SSCs and shielding materials, including any dimensional and material tolerances. The criteria may be dose rates that are calculated using a computer code or are measured using a mock-up of the SSC, with either method using the same radiation source (properties), source-SSC-detector geometry, and the design specifications of the SSC (including material and dimensional tolerances). Any SSC dose rates that exceed the dose rate criteria indicate the SSC shielding is not acceptable. Any areas of an SSC that are affected by efforts to fix any shielding problems should be re-tested to the same criteria.

Alternatively, the applicant may propose an alternate testing program(s) with appropriate justification. For example, the applicant may use dose rate measurements of loaded DSS or DSF storage containers, in lieu of an auxiliary source, to verify shielding effectiveness with appropriate scanning of the shield and an appropriate testing program that uses the actual source strength, configuration, and other appropriate characteristics of the loaded contents for determining the acceptance criteria of the test.

12.5.2.5 *Neutron Absorber Tests*

Neutron absorber materials require both qualification and acceptance testing to provide assurance that the control of criticality by absorbing thermal neutrons will be met in systems designed for nuclear fuel storage, transportation, or both. Both qualification and acceptance testing are generally described in ASTM C1671, “Standard Practice for Qualification and Acceptance of Boron Based Metallic Neutron Absorbers for Nuclear Criticality Control for Dry Cask Storage Systems and Transportation Packaging,” with exceptions, additions, and clarifications provided in Chapter 8, “Materials Evaluation,” of this SRP. Section 8.5.7, “Criticality Control Materials,” of this SRP provides detailed guidance on qualification testing.

Acceptance tests are used to ensure that material properties for plates and other shapes produced in a given production run are in compliance with the materials requirements of the application. In one sense, acceptance tests verify that the material of a given production run has yielded products that have been shown to be like the products that were used in the qualification testing. Acceptance tests are used to ensure that the production process is operating in a satisfactory manner and use statistical data for selected measurable parameters. For all boron-containing absorber materials, acceptance tests should (1) verify boron-10 content and uniformity, (2) require visual examinations to establish that only acceptable levels of defects are present from cracks, porosity, blisters, or foreign inclusions, and (3) make dimensional determinations (e.g., plate thickness which is important to the areal density).

Neutron attenuation tests are calibrated using appropriate standards such as those based on (coated with) zirconium diboride plates to ensure the accuracy of the measured values. As described in Appendix 8A, “Clarifications, Guidance, and Exceptions to ASTM Standard Practice C1671-15,” to Chapter 8 of this SRP, approved substitutes may be used for the attenuation tests for material for which 75-percent credit is taken for boron content. These include tests such as chemical analysis, provided that (1) both the neutron attenuation tests and the alternative tests have at least the sensitivity of tests specified in ASTM C1671 and (2) the alternate form of testing is regularly benchmarked against calibrated neutron attenuation tests. Chemical analyses should also include spectrochemical analysis for material impurity levels and boron-10 content. Uniformity is assessed using statistical sampling techniques that ensure that the entire plate of material and all plates in a lot meet a 95/95 criterion. This means that a test result has a 95-percent likelihood of containing the minimum required amount of boron-10, and that this is known at the 95-percent confidence level.

Confirm that the calculation of minimum poison content (e.g., poison areal density) conservatively accounts for tolerance limits on material density, poison concentration, and component dimensions. Thickness tolerances on rolled plates, sheets, or other shapes are typically on the order of ±10 percent. The acceptance testing should provide a representative sampling of coupons for plates or sheets from a given lot. Statistical sampling can be used to the extent practical, using test locations on a coupon that will account for local variations or anomalies within the coupon and hence within the plates represented by the coupon. Confirm that the applicant has taken the adequate numbers of samples to ensure the confidence level required for the application.

12.5.2.5.1 Acceptance Testing of Fabricated Materials for 75-Percent Boron Credit

For multiphase absorber materials analyzed with 75-percent poison credit (or less), confirm that acceptance testing is consistent with the following:

- The effective boron-10 content should be verified from plate coupons by either (1) neutron attenuation testing or (2) chemical assay to determine boron content with mass spectrometric analysis for isotopic composition (see conditions in Appendix 8A to Chapter 8 of this SRP).

- Sufficient coupons should be taken for acceptance testing to justify the level of credit given. Rejection of a coupon should result in rejection of the plate from which it is taken. Sampling may be reduced to lesser percentages of the coupons taken (e.g., to 50 percent of all coupons) after acceptance of all coupons in the first 25 percent of the lot. A rejection during reduced inspection should invoke a 100-percent inspection for coupons from that lot.

- A visual examination of all plates for defects should be conducted.

12.5.2.5.2 Acceptance Testing for Greater Than 75-Percent Boron Credit

For acceptance testing of borated absorbers at levels of poison credit beyond 75 percent, the extent of the acceptance testing and inspection is enhanced. Some of the data helpful in meeting the guidance in ASTM C1671, Section 5.3.4, are as follows:

- The effective boron-10 content is verified by neutron attenuation testing of coupons. An adequate number of coupons should be acceptance tested for each lot of materials to statistically demonstrate that the 95/95 criterion is satisfied for the minimum required boron-10 content. The minimum areal density is specified in the SAR.

- Sufficient coupons should be taken to satisfy the 95/95 criterion. For example, coupons are taken from at least every other plate unless justification for fewer is given. Measurements are made on samples taken from 100 percent of all coupons. Rejection of a coupon should result in rejection of the plate. Sampling may be reduced to 50 percent of all coupons after acceptance of all coupons in the first 25 percent of the lot. A rejection during reduced inspection should invoke a return to 100-percent inspection for that lot.

- The applicant should perform a statistical analysis of the neutron attenuation results for all plates in a lot. This analysis should show that the lot meets the 95/95 criterion. That is, using a one-sided tolerance limit factor for a normal distribution with at least 95-percent probability, the areal density is greater than or equal to the specified minimum value with 95-percent confidence level. Failure to meet this acceptance criterion of this statistical analysis should result in rejection of the entire lot for use at 100-percent (90-percent credit in k_{eff} calculations). Applicants may choose to convert all areal densities determined by neutron attenuation to a volume density by dividing by the thickness of the coupon. The one-side tolerance limit of volume density with 95-percent probability and 95-percent confidence may then be determined. The minimum specified value of the areal density may be divided by the 95/95 lower tolerance limit of boron-10 volume density to arrive at the minimum plate thickness. Hence, all plates that have any locations thinner than this minimum should be rejected, and those equal to or thicker may be accepted.

- A visual examination of all plates for defects should be conducted.

Refer to Section 8.5.7.2, “Computation of Percent Credit for Boron-Based Neutron Absorbers,” of this SRP regarding how to compute the level of credit.

12.5.2.6 *Thermal Tests*

Depending on the details of the design and operational aspects of the DSS or DSF SSCs, testing may be required to verify adequate thermal performance. Adequate thermal performance would be established based on the thermal analysis results and applicable technical specifications (limiting conditions for operation and surveillance requirements). Confirm that the applicant has established acceptance criteria on the basis of the conditions of the test (e.g., test heat loading, ambient conditions, temperatures, pressures).

12.5.3 Preoperational Testing and Startup Operations (SL)

Review the preoperational testing plan to determine that it includes all of the necessary tests and provides for proper evaluation, approval, and use of the test results. Determine that the testing descriptions, responses expected, and contingent corrective actions are appropriate for the item being tested. In performing these assessments, seek the assistance of NRC staff with expertise in the specific topical areas covered by the tests.

In determining whether the preoperational testing plan is comprehensive, consider the inclusion of the following types of testing and evaluation, as applicable:

- tests associated with construction (or reference to submitted construction specifications)
- preoperational testing specified in technical specifications
- calibration and testing of all equipment and instruments, monitors, and systems with a safety or security function
- tests of supplier-owned equipment to be used in functional operations (e.g., storage container haul trailer and positioning equipment) and in testing
- load tests of rigging, spreaders, and lift points
- evaluations of the effectiveness of procedures and consideration of potentially improved alternatives
- tests of physical and programmed limits on travel of lifting and transfer equipment (e.g., travel over a pool, lift heights, positioning force)

12.5.4 Maintenance Program

In general, applicants should design, construct, and periodically test the DSS or DSF under review to the codes and standards identified in the principal design criteria chapter of the SAR. The NRC does not generally review detailed periodic test and maintenance procedures as part of the certification or licensing process; however, the applicant is expected to describe important or essential elements of the maintenance programs in the SAR.

The following subsections describe (some of) the maintenance program elements that are subject to NRC review. Review each program element for each maintenance program included in the SAR to ensure that the applicant has identified the purpose of the periodic test, explained the proposed test method (including any applicable standard to which the test will be performed), defined the acceptance criteria and bases for the test, and described the actions to be taken if the acceptance criteria are not satisfied. Confirm that the SAR describes the accessibility of SSCs important to safety for inspection, maintenance, and testing, in accordance with 10 CFR 72.122(f) for a specific license or 10 CFR 72.236(g) for a CoC.

DSSs or DSF storage containers are typically designed as passive units requiring minimal maintenance. Ensure that the SAR addresses the areas described in the subsections below, as applicable.

12.5.4.1 *Inspection*

Usually, the DSS or DSF has at least one monitoring system (e.g., pressure, temperature, dosimetry). Confirm that the SAR discusses how such systems will be used to provide information regarding possible off-normal events and what surveillance actions may be necessary to ensure that these systems function properly. The licensee at the site will develop and implement detailed procedures.

Confirm that the SAR describes routine, periodic visual surface and weld inspections, which should be limited to the readily accessible surfaces (e.g., the exterior surface of the DSS or DSF storage container and all surfaces of empty transfer casks). In addition, the SAR should discuss inspection of lifting and rotating trunnion load-bearing surfaces. The SAR should discuss any other appropriate inspections for other DSF SSCs.

12.5.4.2 *Tests*

Verify that the SAR describes any periodic tests of DSS or DSF SSCs and features or calibration of monitoring instrumentation, as well as periodic tests to verify shielding, thermal, and confinement capabilities. Confirm that the applicant has otherwise justified that aging and degradation of materials related to the shielding, confinement, and thermal designs are not credible during the certified storage or licensed period of the DSS or DSF. Verify that the SAR also describes procedures for any applicable periodic testing of neutron poison effectiveness. As an alternative to the periodic testing of neutron poison effectiveness, the applicant may show continued poison effectiveness in the manner described in Chapter 7, "Criticality Evaluation," of this SRP. The qualification tests of the poison material, discussed in SRP Chapter 8 may also be useful in showing the material's continued effectiveness.

In addition, verify that the SAR discusses any routine testing of support systems (e.g., vacuum drying, helium backfill, and leak testing equipment). Ensure that the SAR discusses any other appropriate tests for other DSF SSCs.

12.5.4.3 *Repair, Replacement, and Maintenance*

Verify that the SAR discusses the repair and replacement of DSS or DSF SSCs and features, as may be required during the lifetime of the DSS or DSF. This discussion should include methods of repair or replacement, testing procedures, and acceptance criteria. Confirm that the SAR also describes procedures for routine maintenance (such as lubrication and reapplication of corrosion inhibiting materials in the event of scratches) through the expiration of the service life of the equipment. Such information is also often included in the chapter of the SAR on accident analysis, which describes actions to be taken following an off-normal event or accident condition. Ensure that the SAR describes any other appropriate repair, replacement, and maintenance activities for other DSF SSCs.

12.5.5 Normal Operations (SL)

12.5.5.1 *Procedures* ***(SL)***

Ensure that the SAR states that the applicant, as the licensee, will conduct all operations that are important to safety according to detailed written procedures and that these procedures will be based on and consistent with the operations, acceptance tests, and maintenance programs descriptions in the SAR. Determine whether the identified subjects for written procedures include all routine and projected contingency operations and correlate with the descriptions of operations at the ISFSI or MRS.

12.5.5.2 *Records* ***(SL)***

Determine whether the records identified for retention include all those required by regulations (refer to the listing and guidance in the acceptance criteria in Section 12.4.5.2 above).

12.5.6 Personnel Selection, Training, and Certification (SL)

Review proposed training for inclusion of regulatory requirements relating to personnel selection, training, certification, exercises, and training records. Determine acceptability based on satisfaction of regulatory requirements, guidance in RG 3.62 and RG 3.48, and evidence of experience in planning and conducting training programs.

Review the minimum qualifications for operating, technical, maintenance, and supervisory personnel and compare proposed requirements with those of other approved license applications. If there are no standard minimum qualifications, this evaluation will rely on the reviewer's judgment. However, the minimum qualifications for these personnel generally include a bachelor's degree and several years of experience in a related technical area that is commensurate with the level of assigned responsibility. Higher level managers typically have the same experience requirements plus previous supervisory or management experience. Discussion of leak testing qualifications can be found in Information Notice 16-04, "ANSI N14.5-2014 Revision and Leakage Rate Testing Considerations," dated March 28, 2016.

Ensure that the SAR adequately addresses any qualification requirements identified in evaluations throughout the SAR (e.g., qualifications for writing leak test procedures and performing leak tests identified in the confinement evaluation). Chapter 10A of this SRP, particularly Sections 10A.4.4, 10A.4.4.1, 10A.5.4, and 10A.5.4.1, includes added guidance regarding personnel selection, training and certification for radiation protection and health physics personnel, and radiation safety training for all licensee personnel. Coordinate with the Chapter 10A reviewer to ensure that the SAR adequately addresses this guidance.

Ensure that the SAR adequately addresses the implementation of the training program before the initiation of operations with SNF, HLW, or reactor-related GTCC waste, including a statement that most of the staff training and certification will be completed before receipt of the radioactive material to be stored.

Review the selection and qualification process for security personnel. Determine whether the process will ensure that security personnel will meet the requirements in 10 CFR 73.55, "Requirements for Physical Protection of Licensed Activities in Nuclear Power Reactors Against Radiological Sabotage," and will be qualified to perform each assigned security job duty in accordance with Appendix B to 10 CFR Part 73 or the requirements imposed by NRC orders for additional security measures.

The following references provide additional guidance on training criteria and training program content:

- ANSI/ANS 8.20, “Nuclear Criticality Safety Training”
- ANSI/ANS 3.1, “Selection, Qualification, and Training of Personnel for Nuclear Power Plants”
- ASNT Recommended Practice No. SNT-TC-1A
- ANSI/ASNT CP-189, “American National Standard ASNT Standard for Qualification and Certification of Nondestructive Testing Personnel.”
- ASTM E1168, “Standard Guide for Radiological Protection Training for Nuclear Facility Workers”
- RG 1.8, “Qualification and Training of Personnel for Nuclear Power Plants”
- RG 1.134, “Medical Assessment of Licensed Operators or Applicants for Operator Licenses at Nuclear Power Plants”
- RG 8.27, “Radiation Protection Training for Personnel at Light-Water-Cooled Nuclear Power Plants”
- RG 8.29, “Instruction Concerning Risks from Occupational Radiation Exposure”
- NUREG-0800, “Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition,” Section 13.2.2, “Non-Licensed Plant Staff Training”

12.5.7 Emergency Planning (SL)

The NRC staff should review the license application SAR and other applicable documents because they contain information that may be relevant to the EP.

12.5.7.1 *Description of Facility and Site* ***(SL)***

Review the description of the facility and the site to ensure that the applicant adequately described both the site and the adjacent area. Review the maps submitted as part of the EP to determine whether the site, ISFSI or MRS cask storage areas, other onsite structures, and major site features are sufficiently detailed.

12.5.7.2 *Description of Area Near Site* ***(SL)***

Review the EP to determine whether the applicant has adequately described the principal characteristics of the area near the site. Review the maps provided to ensure that locations with emergency planning significance have been identified.

12.5.7.3 *Types of Accidents* ***(SL)***

Review the EP to determine whether the applicant has adequately identified and described the types of radioactive material accidents. Based on submittals from other licensees and other available information, determine whether the EP addresses all postulated accidents.

12.5.7.4 *Classification of Accidents* ***(SL)***

Review the emergency action levels at which an Alert or SAE will be declared. Review the procedures available to the NRC staff for classifying accidents.

12.5.7.4.1 Alert

Review the EP to determine whether the definition of an Alert is consistent with the NRC's definition and whether initiating events are realistic and comprehensive. Review the mobilization efforts at the Alert level to determine whether the workers will be adequately protected.

12.5.7.4.2 Site Area Emergency

Review the EP to determine whether the definition of SAE is consistent with the NRC's definition and whether initiating events are realistic and comprehensive. Review the procedure for facility mobilization if an SAE is declared. Review the steps taken to notify offsite response organizations that an SAE has been declared.

12.5.7.5 *Detection of Accidents* ***(SL)***

Review the means used at the facility for detecting accidents. Review the location of radiation monitors, smoke or heat detectors, process alarms, and criticality alarms.

12.5.7.6 *Mitigation of Consequences* ***(SL)***

12.5.7.6.1 Limiting Actions

Review the processes and equipment available to mitigate the consequences of accidents identified in the EP. Review whether sprinkler systems, other fire suppression systems, fire detection systems, and filtration or holdup systems are identified.

12.5.7.6.2 Onsite Protective Actions

Review the EP to determine whether it describes onsite protective actions to be taken, criteria for implementing the actions, and notification procedures for potentially affected personnel. Review exposure guidelines to determine whether the guidelines are consistent with the EPA's "PAG Manual: Protective Action Guides and Planning Guidance for Radiological Incidents." Review the evacuation and relocation procedures to determine whether they are adequate. Review arrangements with offsite medical facilities to determine whether provisions to transport injured site personnel are adequate.

12.5.7.6.3 Emergency Response Equipment and Facilities

Review the EP to ensure that emergency response equipment and facilities are adequately described. Ensure that the EP specifies the types of equipment necessary and the locations of the equipment. Review the provisions to inventory emergency response equipment.

12.5.7.6.4 Offsite Protective Actions

Review the conditions that would require offsite protective actions. The recommended approach is to estimate a range of potential source terms for each accident type in the planning and then decide in the planning what recommendations would be made to offsite response organizations for each accident type.

12.5.7.7 *Assessment of Releases* ***(SL)***

Review the EP to determine how the licensee will assess radioactive releases to the environment. Review the description of the types of sampling and monitoring equipment to determine adequacy. This does not mean real-time assessment. It means measurements made after the release has occurred to determine how much material was released. The recommended approach is to estimate a range of potential source terms for each accident type in the planning and then decide in the planning what recommendations would be made to offsite response organizations for each accident type.

12.5.7.8 *Responsibilities* ***(SL)***

Review the description of the onsite emergency organization to determine its adequacy to properly assess the situation. Review authorities and responsibilities of key positions and groups.

12.5.7.8.1 Normal Facility Operation

Review the description of the normal operating facility organization. Verify that it identifies the positions with responsibility to declare an emergency and to initiate the appropriate response, as well as the personnel with the responsibility for maintaining the EP and implementing procedures.

12.5.7.8.2 Onsite Emergency Response Organization

Review the onsite emergency response organization to determine whether there is sufficient staff to manage the emergency situation. Review the method of activating the emergency response organization. Determine whether the EP includes the minimum level of staffing.

12.5.7.8.3 Direction and Coordination

Review the EP to determine whether it designates the position of the person, and his or her alternates, who has the principal responsibility for implementing and directing the emergency response. Determine whether the EP contains authorization for delegating responsibilities.

12.5.7.8.4 Onsite Emergency Assignments

Ensure that the EP specifies which personnel and organizational groups are to provide support in the event of an emergency. Review the strategies for staffing the facility if the emergency is of long duration.

12.5.7.8.5 Emergency Response Records

Review the EP to determine if it describes how records will be retained and the length of retention, as required in 10 CFR 72.80(c).

12.5.7.9 *Notification and Coordination* ***(SL)***

Review the means used to activate the emergency response organization for each class of accident. Ensure that the EP describes how the communication with licensee personnel during both regular and nonregular hours is performed. Review the method the licensee has in place to notify local, State, and Federal authorities if an accident occurs.

12.5.7.10 *Information to be Communicated* ***(SL)***

Review the EP and implementing procedures to determine whether the licensee has developed a clear, concise statement to be communicated to offsite response organizations and the NRC. Review the standard reporting checklist to determine whether the licensee has notified all responsible agencies during an emergency.

12.5.7.11 *Training* ***(SL)***

Review the emergency response training program to determine whether licensee personnel will be adequately trained.

12.5.7.12 *Safe Condition* ***(SL)***

Review the EP for the general plans of restoring the facility to safe operation after an accident. Review the requirements for ensuring that emergency response equipment is restored to a state of readiness.

12.5.7.13 *Exercises* ***(SL)***

Review the provisions for conducting periodic drills and exercises.

12.5.7.14 *Hazardous Chemicals* ***(SL)***

Ensure that the licensee has certified compliance with the Emergency Planning and Community Right-to-Know Act of 1986, with respect to any hazardous materials processed at the facility.

12.5.7.15 *Comments of the Plan* ***(SL)***

Review the EP's requirements for obtaining comments from offsite response organizations. Review any comments received from the offsite organizations and the resolution of the comments.

12.5.7.16 *Offsite Assistance* ***(SL)***

Review provisions for requesting assistance from offsite response agencies during and after an emergency. Review radiological emergency response training provided to offsite emergency responders to an MRS only.

12.5.8 Physical Security and Safeguards Contingency Plans (SL)

Review the physical security plan against the applicable requirements in 10 CFR Part 73 and the applicable NRC orders for additional security measures and ensure that the plan adequately provides for each of the required elements. If the application is from the DOE, verify that it includes a description of the physical security plan for protection against radiological sabotage (as required in 10 CFR Part 72, Subpart H) and a certification that it will provide safeguards at the ISFSI or MRS that meet the requirements for comparable surface DOE facilities.

Ensure that the safeguards contingency plan complies with the format and content requirements of Appendix C to 10 CFR Part 73, including (1) a predetermined set of decisions and actions to satisfy stated objectives, (2) an identification of the data, criteria, procedures, and mechanisms necessary to efficiently implement the decisions, and (3) a stipulation of the individual, group, or organizational entity responsible for each decision and action. Consult RG 5.55 for guidance on acceptable contents and format of safeguards contingency plans applicable to ISFSI or MRS installations. Although the applicant currently is not required to submit the written procedures that will implement the safeguards contingency plan (although the procedures are subject to NRC inspection on a periodic basis), review these procedures as needed to verify that the safeguards contingency plan meets the requirements of Appendix C to 10 CFR Part 73.

12.6 Evaluation Findings

The NRC reviewer should prepare evaluation findings upon satisfaction of the regulatory requirements in Section 12.4 of this SRP. If the documentation submitted with the application fully supports positive findings for each of the regulatory requirements, the statements of findings should be similar to the following:

F12.1 **(SL)** The SAR includes an acceptable description of the applicant's organization to demonstrate the financial capabilities to construct, operate, and decommission the installation, as required by 10 CFR 72.22(e).

F12.2 **(SL)** The SAR includes an acceptable description of the program covering preoperational testing and initial operations, in compliance with 10 CFR 72.24(p).

F12.3 **(SL)** The SAR includes an adequate, acceptable description of the applicant's operating organization, delegations of responsibility and authority, and the minimum skills and experience qualifications relevant to the various levels of responsibility and authority, in compliance with 10 CFR 72.28(c).

F12.4 **(SL)** The SAR provides acceptable assurance with regard to the management, organization, and planning for preoperational testing and initial operations that the activities authorized by the license can be conducted without endangering the health and safety of the public, in compliance with 10 CFR 72.40(a)(13).

F12.5 SSCs important to safety will be designed, fabricated, erected, tested, and maintained to quality standards commensurate with the importance to safety of the function(s) they are intended to perform. Chapter _____ of the SAR identifies the safety importance of SSCs, and Chapter(s) ______ present(s) the applicable standards for their design, fabrication, and testing in accordance with 10 CFR 72.82(d), 10 CFR 72.122(a), 10 CFR 72.122(f), 10 CFR 72.124(b), 10 CFR 72.162, 10 CFR 72.234(b) and 10 CFR 72.236(b), (g), (j) and (l).

F12.6 The applicant or licensee, as appropriate, will examine and test, as needed, the [DSS or DSF designation] SSCs and features to ensure they do not exhibit any defects that could significantly reduce their confinement effectiveness. Chapter(s) ______ of the SAR describe(s) this inspection and testing, in compliance with 10 CFR [72.162/72.236(l)] or 10 CFR 72.122(a).

F12.7 **(SL)** The SAR includes an acceptable plan for the conduct of operations, in compliance with 10 CFR 72.24(h), that provides reasonable assurance that operations important to safety will be performed in accordance with detailed written procedures, that the operating procedures are adequate in accordance with 10 CFR 72.40(a)(5), and that describes a records management system that will provide retention for all those required by regulation.

F12.8 **(SL)** The applicant has provided acceptable technical qualifications, including training and experience, for personnel who will be engaged in the proposed activities, in compliance with 10 CFR 72.24(j) and 10 CFR 72.28(a).

F12.9 **(SL)** The SAR includes an acceptable description of a personnel training program to comply with 10 CFR 72.24(j), 10 CFR 72.28(b), 10 CFR 72.40(a)(9), and 10 CFR Part 72, Subpart I.

F12.10 **(SL)** The SAR includes information that ensures that the applicant will have and maintain an adequate complement of trained and certified installation personnel before receipt of SNF, HLW, or reactor-related GTCC waste for storage, in compliance with 10 CFR 72.24(j) and 10 CFR 72.28(d).

F12.11 **(SL)** The SAR provides acceptable assurance that the applicant is qualified by reason of training and experience to conduct the operations covered by the regulations in compliance with 10 CFR 72.40(a)(4).

F12.12 **(SL)** The SAR includes an acceptable description of the emergency planning program, in compliance with 10 CFR 72.24(k), 10 CFR 72.32, and 10 CFR 72.40(a)(11).

F12.13 **(SL)** The SAR provides an acceptable description of the physical security and safeguards contingency plans, in compliance with 10 CFR 72.24(o), 10 CFR 72.40(a)(8), 10 CFR 72.40(a)(14), 10 CFR 72.180 and 10 CFR 72.184.

F12.14 **(SL)** [If appropriate] The design of the DSF includes ____________ [specify the SSCs], the functional adequacy or reliability of which has not been demonstrated by previous use for the same purpose. The SAR describes acceptable planned tests and demonstration of capability in the areas of uncertainty before use, in compliance with 10 CFR 72.24(i).

The reviewer should provide a summary statement similar to the following:

> The staff concludes that the conduct of operations program is [or for a DSS, in place of "conduct of operations program is," can read as: acceptance tests and maintenance programs are] in compliance with 10 CFR Part 72 and that the applicable acceptance criteria have been satisfied. The evaluation of the conduct of operations program provides reasonable assurance that the [DSS or DSF] will allow for the safe storage of SNF and, as applicable for a DSF, reactor-related GTCC waste and HLW throughout its licensed or certified period of storage. This finding is reached on the basis of a review that considered the regulation itself, appropriate regulatory guides, applicable codes and standards, and accepted practices.

12.7 References

10 CFR Part 20, "Standards for Protection Against Radiation."

10 CFR Part 72, "Licensing Requirements for the Independent Storage of Spent Nuclear Fuel, High-Level Radioactive Waste, and Reactor-Related Greater than Class C Waste."

10 CFR Part 73, "Physical Protection of Plants and Materials."

American Concrete Institute (ACI) 318, "Building Code Requirements for Structural Concrete and Commentary."

ACI 349, "Code Requirements for Nuclear Safety-Related Concrete Structures and Commentary."

American Institute of Steel Construction 303-10, "Code of Standard Practice for Steel Buildings and Bridges," included in the *Steel Construction Manual*.

American National Standards Institute (ANSI) N14.5, "Radioactive Materials—Leakage Tests on Packages for Shipment," 2014

ANSI N14.6, "Radioactive Materials—Special Lifting Devices for Shipping Containers Weighing 10,000 Pounds (4500 Kilograms) or More."

ANSI/American Nuclear Society (ANS) 3.1-1993, "Selection, Qualification, and Training of Personnel for Nuclear Power Plants," 1993.

ANSI/ANS 8.20, "Nuclear Criticality Safety Training," 1991, reaffirmed 2015.

American Society of Mechanical Engineers, Boiler and Pressure Vessel (B&PV) Code, 2007—Addenda 2008.
- Section II, "Materials," Part A, "Ferrous Materials Specifications," SA-20
- Section III, "Rules for Construction of Nuclear Facility Components"
  - Division 1, "Metallic Components," Subsections NB, NC, NF, and NG
- Section V, "Nondestructive Examination," Articles 2, 4, 5, 6, 7, 9
- Section VIII, "Rules for Construction of Pressure Vessels" Division 1
- Section IX, "Welding and Brazing Qualifications"

American Society for Nondestructive Testing (ASNT) Recommended Practice No.SNT-TC-1A, "Personnel Qualification and Certification in Nondestructive Testing."

ANSI/ASNT CP-189, “American National Standard ASNT Standard for Qualification and Certification of Nondestructive Testing Personnel.”

American Society for Testing and Materials (ASTM) C1671, “Standard Practice for Qualification and Acceptance of Boron Based Metallic Neutron Absorbers for Nuclear Criticality Control for Dry Cask Storage Systems and Transportation Packaging.”

ASTM E208, “Standard Test Method for Conducting Drop-Weight Test to Determine Nil-Ductility Transition Temperature of Ferritic Steels.”

ASTM E604, “Standard Test Method for Dynamic Tear Testing of Metallic Materials.”

ASTM E1168, “Standard Guide for Radiological Protection Training for Nuclear Facility Workers.”

American Welding Society (AWS) A2.4, “Standard Symbols for Welding, Brazing, and Nondestructive Examination.”

AWS D1.1, “Structural Welding Code—Steel.”

AWS D1.3, “Structural Welding Code—Sheet Steel.”

AWS D1.6, “Structural Welding Code—Stainless Steel.”

American Society for Nondestructive Testing Recommended Practice No. SNT-TC-1A, “Personnel Qualification and Certification in Nondestructive Testing.”

Emergency Planning and Community Right-to-Know Act of 1986, 100 Stat. 1613, Public Law 99-499.

NRC, Information Notice 16-04, “ANSI N14.5-2014 Revision and Leakage Rate Testing Considerations,” March 28, 2016.

NUREG-0800, “Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition.”

NUREG-1140, “A Regulatory Analysis on Emergency Preparedness for Fuel Cycle and Other Radioactive Material Licensees,” January 1988.

NUREG/CR-1815, “Recommendations for Protecting Against Failure by Brittle Fracture in Ferritic Steel Shipping Containers Up to Four Inches Thick,” Lawrence Livermore National Laboratory, June 1981.

NUREG/CR-3826, “Recommendations for Protecting Against Failure by Brittle Fracture in Ferritic Steel Shipping Containers Greater than Four Inches Thick,” Lawrence Livermore National Laboratory, July 1984.

Regulatory Guide 1.8, “Qualification and Training of Personnel for Nuclear Power Plants.”

Regulatory Guide 1.134, “Medical Assessment of Licensed Operators or Applicants for Operator Licenses at Nuclear Power Plants.”

Regulatory Guide 3.48, "Standard Format and Content for the Safety Analysis Report for an Independent Spent Fuel Storage Installation or Monitored Retrievable Storage Installation (Dry Storage)."

Regulatory Guide 3.61, "Standard Format and Content for a Topical Safety Analysis Report for a Spent Fuel Dry Storage Cask."

Regulatory Guide 3.62, "Standard Format and Content for the Safety Analysis Report for Onsite Storage of Spent Fuel Storage Casks."

Regulatory Guide 5.20, "Training, Equipping, and Qualifying of Guards and Watchmen."

Regulatory Guide 5.55, "Standard Format and Content of Safeguards Contingency Plans for Fuel Cycle Facilities."

Regulatory Guide 8.27, "Radiation Protection Training for Personnel at Light-Water-Cooled Nuclear Power Plants."

Regulatory Guide 8.29, "Instruction Concerning Risks from Occupational Radiation Exposure."

U.S. Environmental Protection Agency (EPA) EPA-400/R-17/001, "PAG Manual: Protective Action."

13 WASTE MANAGEMENT EVALUATION (SL)

13.1 Review Objective

The objective of the U.S. Nuclear Regulatory Commission's (NRC's) waste management review is to ensure that the design and proposed operation of the independent spent fuel storage installation (ISFSI) or monitored retrievable storage installation (MRS) provide for the safe confinement and management of any radioactive waste generated as a result of facility operations. This review includes an evaluation of the applicant's analysis to ensure that the dose contribution from radioactive wastes generated by the handling and storage of spent nuclear fuel (SNF), reactor-related greater-than-Class-C (GTCC) waste or, for an MRS, high-level radioactive waste (HLW) at the dry storage facility (DSF), when added to other dose contributors at the site will meet, the requirements in Title 10 of the *Code of Federal Regulations* (10 CFR) Part 20, "Standards for Protection Against Radiation," and 10 CFR Part 72, "Licensing Requirements for the Independent Storage of Spent Nuclear Fuel, High-Level Radioactive Waste, and Reactor-Related Greater Than Class C Waste."

13.2 Applicability

This chapter applies to the review of applications for specific licenses "**(SL)**" for an ISFSI or an MRS, categorized as a dry storage facility (DSF).

13.3 Areas of Review

This chapter addresses the following areas of review:

- waste sources and waste management facilities
- off-gas treatment and ventilation
- liquid waste treatment and retention
- solid wastes
- waste stream radiological characteristics and dose analyses

13.4 Regulatory Requirements and Acceptance Criteria

This section summarizes those sections of 10 CFR Part 72 and 10 CFR Part 20 that are relevant to the review areas this chapter addresses. The NRC staff reviewer should refer to the exact language in the regulations. Table 13-1 matches the relevant regulatory requirements to the areas of review covered in this chapter.

Table 13-1 Relationship of Regulations and Areas of Review

Areas of Review	10 CFR Part 20 Regulations[a]				
	20.1101 (b)(d)	20.1301 (a)(b)(d)(e)	20.1302(a)(b)	20.2001 (a)	20.2003 (a)
Waste Sources and Waste Management Facilities					
Off-Gas Treatment and Ventilation				•	
Liquid Waste Treatment and Retention				•	•
Solid Wastes					
Waste Stream Radiological Characteristics and Dose Analyses	•	•	•		

Areas of Review	**10 CFR Part 72 Regulations (cont.)**						
	72.24	72.40 (a)(13)	72.104	72.106 (b)	72.122 (b)(1) (2)(4), (e)[A] (h)(3)	72.126	72.128
Waste Sources and Waste Management Facilities	(d)(e)(f) (l)		•	•	•	(a)(b)(c) (d)	•
Off-Gas Treatment and Ventilation	(d)(e)(f) (l)		•	•	•	(a)(b)(c) (d)	•
Liquid Wastes, Treatment and Retention	(d)(e)(f) (l)		•	•	•	(a)(b)(c) (d)	•
Solid Wastes and Treatment	(d)(e)(f)		•	•	•	(a)(b)(c) (d)	•
Waste Stream Radiological Characteristics and Dose Analyses	(e)(l)(m)	•	•	•	(b)(4), (h)(3)	(b)(c) (d)	

a Note that the regulations in 10 CFR Part 20, as specified in the scope in 10 CFR 20.1002, “Method for Obtaining Approval of Proposed Disposal Procedures,” “…apply to persons licensed by the Commission to receive, possess, use, transfer, or dispose of byproduct, source, or special nuclear material or to operate a production or utilization facility under parts 30 through 36, 39, 40, 50, 52, 60, 61, 63, 70, or 72 of this chapter.”

The following are principal acceptance criteria that apply to confinement and management of site-generated waste:

- establishment of operational restrictions and limits that ensure effluents and direct radiation levels from the DSF waste management system(s), in addition to other site dose contributors, will meet the as low as reasonable achievable (ALARA) objectives and not exceed the limits in 10 CFR 72.104(a), in accordance with 10 CFR 72.104(b) and 10 CFR 72.104(c)

- demonstration that the DSF waste storage and management system(s) is (are) designed to limit radioactive materials releases to ALARA conditions under normal and off-normal operation conditions, control releases under accident conditions, and analyses of doses from the system(s) to support evaluation of DSF compliance with regulatory limits for the applicable conditions, in accordance with 10 CFR 72.126(d)

- analyses and identification of maximum doses and concentrations of radioactive materials in effluents from the waste storage and management system(s) to support evaluation of DSF compliance with 10 CFR 20.1301, “Dose Limits for Individual Members of the Public”

- design and operation of the DSF waste storage and management system(s) in such a way to ensure occupational and public doses will meet ALARA objectives for the DSF and DSF air emissions will not exceed the constraint in 10 CFR 20.1101(d), in accordance with 10 CFR 20.1101, “Radiation Protection Programs.”

- safe confinement of all radioactive waste materials generated as a result of DSF operations until disposal

- implementation of the waste confinement objectives; equipment; structures systems and components (SSCs); and program necessary for the protection against radiation (as described in Chapter 10A, “Radiation Protection Evaluation For Dry Storage Facilities,” of this SRP)

The results of the dose and radioactive materials concentrations in this chapter are integrated into the radiation protection evaluations (see Chapter 10A of this SRP), by which facility compliance with the dose, ALARA, and other radiation protection criteria is demonstrated and determined.

Additional acceptance criteria apply to the descriptions in the safety analysis report (SAR) of waste sources and management systems, waste characteristics, operations, and monitoring. The SAR must describe the design bases for systems and equipment that maintain control over radioactive material in gaseous and liquid effluents and identify the equipment and facilities important to safety (10 CFR 72.24(l)). The SAR must also include the design objectives and the means to be employed to maintain ALARA with respect to the levels of radioactivity in effluents and to minimize the generation of waste (10 CFR 72.24(l)). The SAR should describe waste operations, from generation and collection to final disposal off site, including narrative text and flowcharts.

The following sections address specific requirements related to waste sources, off-gas treatment and ventilation, liquid waste treatment and retention, and solid wastes. Reviewers can also use NUREG-0800, “Standard Review Plan for the Review of Safety Analysis Reports for Nuclear

Power Plants: LWR Edition" (primarily Chapter 11, "Radioactive Waste Management"), to identify acceptance criteria for these categories.

13.4.1 Waste Sources and Waste Management Facilities

Radioactive wastes that result from a DSF can be separated into two main categories:

- Effluents—radioactive materials that are discharged to the environment in gaseous or liquid form. The activity content of these effluents must comply with appropriate regulatory limits and ALARA criteria (e.g., 10 CFR 72.24(l) and 10 CFR 20.1101(d)).

- Wastes—radioactive materials that are of sufficient hazard or regulatory concern that they require special care before final disposal. The generation of such wastes must be reduced to the extent practicable (10 CFR 72.24(f) and 10 CFR 72.128(a)(5)).

The SAR should identify all actual and potential sources of site-generated radioactive waste. Waste sources described should include activities that give rise to potentially radioactive wastes that would require treatment or special handling. The identification of sources should be comprehensive.

Anticipated radioactive wastes should be described and classified with respect to source; physical, chemical, and radiological characteristics; and method and design for treatment, handling, and storage mode before disposal. Chapters 6, "Shielding Evaluation," and 9, "Confinement Evaluation," of this SRP include guidance regarding source characterization that may also be applicable to the sources for this waste management evaluation. These sources may include items such as crud that is removed during SNF handling (in a pool). The SAR descriptions should include sources of radioactive materials that may become airborne in areas accessible to or normally occupied by operating personnel. Descriptions should include gaseous and particulate materials and calculated nuclide concentrations during normal, off-normal, and accident conditions, as well as calculation models and parameters. The SAR should also identify and describe sources of nonradioactive waste, such as combustion products and chemical wastes, to the extent necessary to enable or support determination as to whether site activities can result in radioactive materials being added to such sources.

The SAR should include an estimate of the total volume of liquid waste discharged to the environment to provide the bases for determining concentrations and activities of radionuclides in liquid effluents. An estimate of total sanitary sewer flow may be needed to determine concentrations of radionuclides in waste disposed to the sanitary sewer.

The SAR facility description should include descriptions of the waste management facilities and systems. These would include systems and SSCs used to store, handle, and treat the radioactive wastes generated as a result of DSF operations. The descriptions should identify facilities and systems in scaled layout and include arrangement drawings of the DSF as well as layout and arrangement drawings of the waste management facilities and systems themselves. These drawings should show the locations of all sources identified and described in the waste management evaluation, including storage locations of wastes. The descriptions should be adequate to enable a clear understanding of movements of wastes within the DSF waste management facilities and systems. The descriptions should also be adequate to enable a clear understanding of the connections between systems that could allow movement of radioactive materials from radioactive waste systems to nonradioactive waste systems and areas and the methods and features used to preclude such movement. The descriptions should include the design criteria for the SSCs for the facilities and systems, the adequacy of those criteria for ensuring safety and regulatory compliance, and demonstrations that these SSCs meet the design criteria.

13.4.2 Off-Gas Treatment and Ventilation

Off-gas treatment and ventilation systems typically are provided for removing radioactive and nonradioactive hazardous materials from the atmosphere within a confinement barrier before releasing to the environment. The SAR should describe the DSF's ventilation and off-gas treatment systems. The descriptions should include the systems' functions and performance objectives and the physical areas of the facility serviced by each unit system, each unit system's design, and interfaces between systems and with process off-gas systems and equipment (e.g., waste treatment, storage container venting). The SAR descriptions should include drawings, flowsheet, and narrative descriptions. The SAR should also describe actual operations of ventilation and off-gas treatment equipment and the minimum expected performance. The SAR should identify design criteria and limits for operations, safety margins, and performance limits that need to be met for safety. General design criteria should be based on site conditions, including normal, off-normal, and accident condition analyses, design function and performance objectives, and projected volumes of gaseous (or airborne) waste.

The SAR should provide design parameters such as those associated with facility stacks and building ventilation exhaust vents, as they relate to their onsite locations, release heights, exhaust flow, velocity rates, and flow temperatures in determining the types of releases and atmospheric dispersion (X/Q) and deposition parameters (D/Q).

The SAR should also indicate those radioactive wastes that will be produced as a result of off-gas treatment. The applicant should show that system capacity is consistent with the confinement system requirements during normal and off-normal conditions.

The SAR should describe the program for evaluating the performance and efficiency of filters and other treatment devices and the criteria for replacing or regenerating them. The descriptions should also address the replacement and disposal or regeneration of items such as filters and scrubber solutions, including the treatment (with any transfers to other waste systems), handling, and disposal of these wastes. The SAR should describe how these activities are to be done and any possible radiological effects of these activities, including potential personnel exposures and contamination that can result from handling operations. The descriptions should also address how conduct of these activities will meet ALARA objectives.

The SAR should also describe the systems and equipment to monitor gas effluents. This description should include the system and equipment features, locations, and release paths to be monitored. The SAR should also describe the expected reliability and sensitivity of each system and justify the selection of each system and instrument. The SAR should also describe the frequency of sampling, limits for action, and plans to be used to maintain continued integrity of analyses. Such systems would include continuous monitoring systems to detect effluent radioactivity and to alarm on effluent activity levels that exceed operations limits. The SAR would describe the bases for these limits. Chapter 10A of this SRP addresses monitoring and monitoring instrumentation.

13.4.3 Liquid Waste Treatment and Retention

The SAR should identify the sources of all liquid wastes generated and their flow into and out of the liquid treatment systems. These wastes may include laboratory wastes, cask washdown, liquid spills, and decontamination and cleanup solutions. The SAR should also describe the expected inventory levels and characteristics of these wastes and identify the waste streams that will be processed to achieve volume reduction or solidification.

The SAR should describe the systems and equipment for the handling, treatment, and retention of liquid wastes generated from DSF operations. The descriptions should include drawings, flowcharts, and narrative descriptions to enable a clear understanding of the system's design and operations, including design criteria and objectives, function, capabilities, and interfaces with other DSF systems and SSCs.

13.4.3.1 *Design Objectives*

Basic liquid waste treatment concepts include volume reduction, immobilization of radioactive elements, change of composition, and removal of radioactive elements from the waste stream. The description of the facility liquid waste treatment and retention systems should identify the design objectives and demonstrate that the system can handle the expected volume of potentially radioactive and nonradioactive hazardous wastes generated during normal and off-normal operations. The SAR should also describe criteria that incorporate backup and special features to ensure safe confinement of wastes and to minimize personnel doses.

In general, engineered features should be emphasized over procedures to meet protective requirements.

13.4.3.2 *Equipment and System Description*

The SAR should describe the equipment and systems to be installed, including, as applicable, backup and special features that may be relied upon as needed. The SAR should describe the features, systems, and special handling techniques that are important to safety and included in the systems to provide for safe operation. Associated drawings should include the location of equipment, flowpaths, piping, valves, instrumentation, and other physical features. Seismic and quality group classification should conform to the guidelines of Regulatory Guide (RG) 1.143, "Design Guidance for Radioactive Waste Management Systems, Structures, and Components Installed in Light-Water-Cooled Nuclear Power Plants." Where feasible, the system should use gravity flow to reduce pressure and to avoid or minimize contamination of pumping and pressure system equipment. The SAR should also reflect measurement capability to determine the volume, concentration, and radioactivity of wastes fed into collection tanks.

Each waste stream fed to the central collection system should use individual lines, where necessary, to prevent chemical reactions or introduction of contaminants such as complexing agents that can interfere with waste decontamination. Individual lines outside confinement (and liquid containment) barriers should be designed not to rupture in the event of frost heave, earth or structure settlement, or earth-structure motions during design-basis earthquakes or other accident or natural phenomenon events. A separate confinement barrier (e.g., drained outer pipe or drained tunnel) should be provided for these lines.

A suitable secondary confinement structure (e.g., secondary vessel, elevated threshold, or dike) should collect or retain spills, overflows, or leakage from storage vessels. The SAR should indicate the capability to transfer liquid from the secondary confinement to a suitable storage location. All transfer lines should have individual identification.

The piping should be designed to minimize entrapment and buildup of solids in the system. The design should not have any bypasses that route waste streams around collection tanks. Provisions should be made for clean out or decontamination of liquid waste piping, as necessary, to clear potential blockages, perform maintenance or repair, or maintain occupational doses to meet ALARA objectives.

Volume reduction or solidification methods may be used to process liquid wastes. Redundancy and other special features may be incorporated to safely confine the wastes. Adequate shielding should be provided for radioactive liquid waste system components, as necessary.

The SAR should describe how influents to radioactive liquid waste systems are controlled (as necessary, depending on the sources) to prevent introduction of material that may adversely affect system performance. Such materials include, but are not limited to, oils, other organics, insoluble solids, solvents, and hazardous wastes.

The SAR should also describe the liquid monitoring systems, including liquid effluent monitoring systems. The descriptions should include information similar to that described for gas effluent monitoring in Section 13.4.2 above. In addition to release paths, the SAR should describe any other parameters or items that the liquid monitoring systems will use to monitor.

13.4.3.3 *Operating Procedures*

The flow sheets and narrative descriptions of operations should describe the design features and procedures that minimize generation of liquid waste and the possibility of spills, and they should provide for control and containment of spills. The procedures described in the SAR should include performance tests, action levels, actions to be taken under normal and off-normal conditions, and methods for testing to ensure functional operation. The SAR should also describe monitoring and controlling of wastes to the system or facility limits.

13.4.3.4 *Characteristics, Concentrations, and Volumes of Solidified Wastes*

The SAR should describe the physical, chemical, and thermal characteristics of solidified (extracted or residue of liquid) wastes and should also provide estimates of waste volumes and radionuclide concentrations, including the bases for the estimates. Those characteristics and estimates should be compatible with the design criteria and capacity of the liquid waste treatment and retention systems.

13.4.3.5 *Packaging for Onsite Storage*

The SAR shall describe the containers for storing solidified liquid wastes (10 CFR 72.24(l)(3)). The container information should show the materials of construction and include welding design information on the critical boundary welds in regard to the minimum allowable weld joint size configuration. It should also show the maximum temperatures for the waste and container at the highest design heat loads, the homogeneity of the waste contents, the corrosive interactions of the waste on materials of construction, the means for preventing over-pressurization of the container, and the confinement provided by the container under normal, off-normal and accident conditions. The description should also address the container’s ability to perform functions in addition to waste confinement (e.g., shielding), as applicable. The applicant should also demonstrate suitability of packaging for holding and storage of wastes on site at the designated location.

The SAR should also describe aspects of the operating quality assurance program that specifically apply to solidified waste container.

13.4.3.6 *Storage Facilities*

The SAR should describe the storage facilities and operations for site-generated liquid or solidified waste. The descriptions should evaluate the damage to containers (e.g., accidental puncture) from off-normal and accident conditions. The descriptions should also, as applicable, address external corrosion of the container from the environment at the site and within the waste storage facility. The SAR should describe the movement of containers into and out of storage and the expected monitoring. Equipment, waste routing, and spare storage volume should be installed and available to transfer the contents of one tank to another. The minimum spare volume should exceed the maximum liquid content of any one tank.

Provisions should be made so that liquids can be analyzed before transfer. The storage vessels should have agitators, when necessary, to promote mixing of the waste to ensure uniform decay heat distribution, minimize settling, or provide representative waste samples.

If liquid wastes are to be held until site decommissioning or for radioactive decay, the SAR should demonstrate (by analyses or relevant operating experience) that the storage capability is appropriate for the duration of the life of the DSF, or for the projected decay holding time, and the chemistry of the contents. The SAR should also show how the wastes will be handled at the time the installation is permanently decommissioned.

13.4.4 Solid Wastes

The SAR should describe the solid wastes produced during DSF operations, identifying the sources of all generated solid wastes and their flows into and out of the solid waste treatment systems. The SAR should list and characterize the wastes (see Section 13.4.4.4 below), and describe the systems used to treat, package, and contain these wastes. The descriptions should include appropriate drawings, flowcharts, and narrative descriptions to enable a clear understanding of the systems' design and operations, including design criteria and objectives, function, capabilities, and interfaces with other DSF systems and SSCs. The descriptions should include waste radionuclide content, container size, and generation rate.

13.4.4.1 *Design Objectives*

The SAR should identify the design objectives for the systems, including methods and equipment, and demonstrate that the systems can handle the expected volume of potentially radioactive solid wastes generated during normal and off-normal operations. The design objectives should reflect waste minimization as well as safe management. If the design basis includes regulatory limits, the SAR should identify these limits.

In general, engineered features should be emphasized over procedures to meet protective requirements.

13.4.4.2 *Equipment and System Description*

The SAR should describe the equipment and systems to be installed. The SAR should describe the features, systems, and special handling techniques that are important to safety and included in the systems to provide for safe operation. Drawings should identify the locations of equipment and associated features that will be used for volume reduction, confinement, packaging, storage, and disposal.

System and equipment descriptions should address the types of waste treatment methods to be used at the DSF. Fundamental solid waste treatment concepts include volume reduction, immobilization of radioactive material, change of composition, and removal of radioactive material from the waste stream. Solid waste management systems should include provisions for shielding, confinement, handling, and decontamination, as necessary, to ensure that occupational doses are maintained to meet ALARA objectives and to minimize doses to the public from these wastes.

The SAR should also describe the solid waste monitoring systems. The descriptions should, as applicable, include information similar to that described for liquid waste monitoring in Section 13.4.3.2 above. The SAR should describe the procedures, equipment, and instrumentation to be used as well as parameters or items that the monitoring systems will monitor (e.g., integrity of waste container confinement).

13.4.4.3 *Operating Procedures*

The SAR should describe the procedures associated with solid waste system or equipment operations. The procedures should identify performance or functional testing, process limits, action levels, and actions to be taken under normal and off-normal conditions. The SAR should also describe the means for monitoring and controlling to the identified process limits.

13.4.4.4 *Characteristics, Concentrations, and Volumes of Solid Wastes*

The SAR should describe the physical, chemical, radiological, and thermal characteristics of the solid wastes and provide estimates of the waste volumes generated. These characteristics include the radionuclides and their estimated concentrations. The SAR should also provide the bases for the estimates. These characteristics and estimates should be compatible with the design criteria and capacity of the solid waste treatment and retention systems.

13.4.4.5 *Packaging for Onsite Storage*

The SAR shall describe the containers for solid wastes (as for solidified liquid waste described in Section 13.4.3.5 above) (10 CFR 72.24(l)(3)). The SAR should also describe aspects of the operating quality assurance program that specifically apply to solid waste containers.

If a laundry is to be used (e.g., to minimize solid-waste generation), the SAR should describe the containers for transferring the used items. If the laundry is off site, it should be identified and should be licensed to possess radioactive material of the type and quantity to be generated at the DSF. (Note: An offsite laundry is not licensed under 10 CFR Part 72, but an onsite laundry capability to support the DSF should be included in the installation license.)

13.4.4.6 *Storage Facilities*

The SAR should describe the solid waste storage facilities and operations and the movement of containers into and out of storage as well as expected monitoring. The SAR should also address the corrosive aspects of the wastes and monitoring of the containers' confinement barrier. The SAR should appropriately address impacts of other conditions, including off-normal and accident conditions.

The SAR should describe planned disposal of the wastes. If solid wastes are to be held until site decommissioning or for radioactive decay, the SAR should demonstrate (by analyses or relevant operating experience) that the storage containers or confinement, as applicable, are appropriate for the duration of the life of the DSF or for the projected decay holding time. The SAR should also show how the wastes will be handled at the time the installation is permanently decommissioned.

13.4.5 Waste Stream Radiological Characteristics and Dose Analyses

The SAR should provide a summary of the radiological impacts of wastes generated during normal site operations, include the following:

- a summary identifying each effluent and waste type
- the amount of each waste type generated per metric ton of SNF, reactor-related GTCC waste, or HLW handled and stored per unit of time (e.g., per year)
- the quantity and concentration of each radionuclide in each waste stream
- identification of locations, both on site and off site, where individuals may be that are potentially affected by radioactive materials in effluents; these locations include those for personnel who would receive an occupational dose and those individuals receiving a dose as members of the public. Considerations of locations should include the different areas associated with the site (e.g., restricted areas, the controlled area, beyond the controlled area) as defined in 10 CFR Part 20 and 10 CFR Part 72 and the regulatory limitations for who may access these areas
- the estimated concentrations of radionuclides, dose rates, and doses, including, as appropriate, collective (person-rem) doses, at the identified locations for normal, off-normal and accident conditions; the dose and dose rate results should identify the contribution of each radionuclide
- sample calculations and a discussion of the reliability of the concentration and dose estimates
- for each effluent, a summary of the constraints imposed on process systems and equipment to ensure safe operation.

The results of the analyses performed for this waste management evaluation should be sufficient to support the evaluation of compliance with the radiological requirements, including dose limits, for occupational personnel and members of the public. Chapter 10A of this SRP describes the evaluation of the SAR with respect to compliance with the radiological requirements. The results of these analyses should include both the direct radiation and effluent contributions to doses from the wastes. The shielding and confinement chapters of this SRP (Chapters 6 and 9) provide guidance that is useful for the calculation of direct radiation dose and effluent dose, respectively, and the information the SAR should include regarding those calculations. The direct radiation dose analyses should identify and include the contributions from each waste stream and locations of that waste (e.g., solid and liquid wastes in containers and tanks). The SAR should describe the methods used to determine radionuclide concentrations and doses and dose rates, including any computer codes, equations, models, assumptions, and input data used. The SAR should justify the appropriateness and adequacy of the methods and the results. Chapter 10A of this SRP also describes information in the SAR for evaluating effluents and dose analyses applicable to the evaluations in this chapter.

13.5 Review Procedures

This section describes review procedures used to evaluate the wastes generated as a result of DSF operations; the waste management systems used to treat, handle, and store these wastes; and the radiological analyses of these wastes and the management systems. The reviewer should also refer to Chapter 11 of NUREG-0800, which contains guidance that may also be useful for this review.

Figure 13-1 shows the interrelationship between the waste management evaluation and the other areas of review described in this SRP.

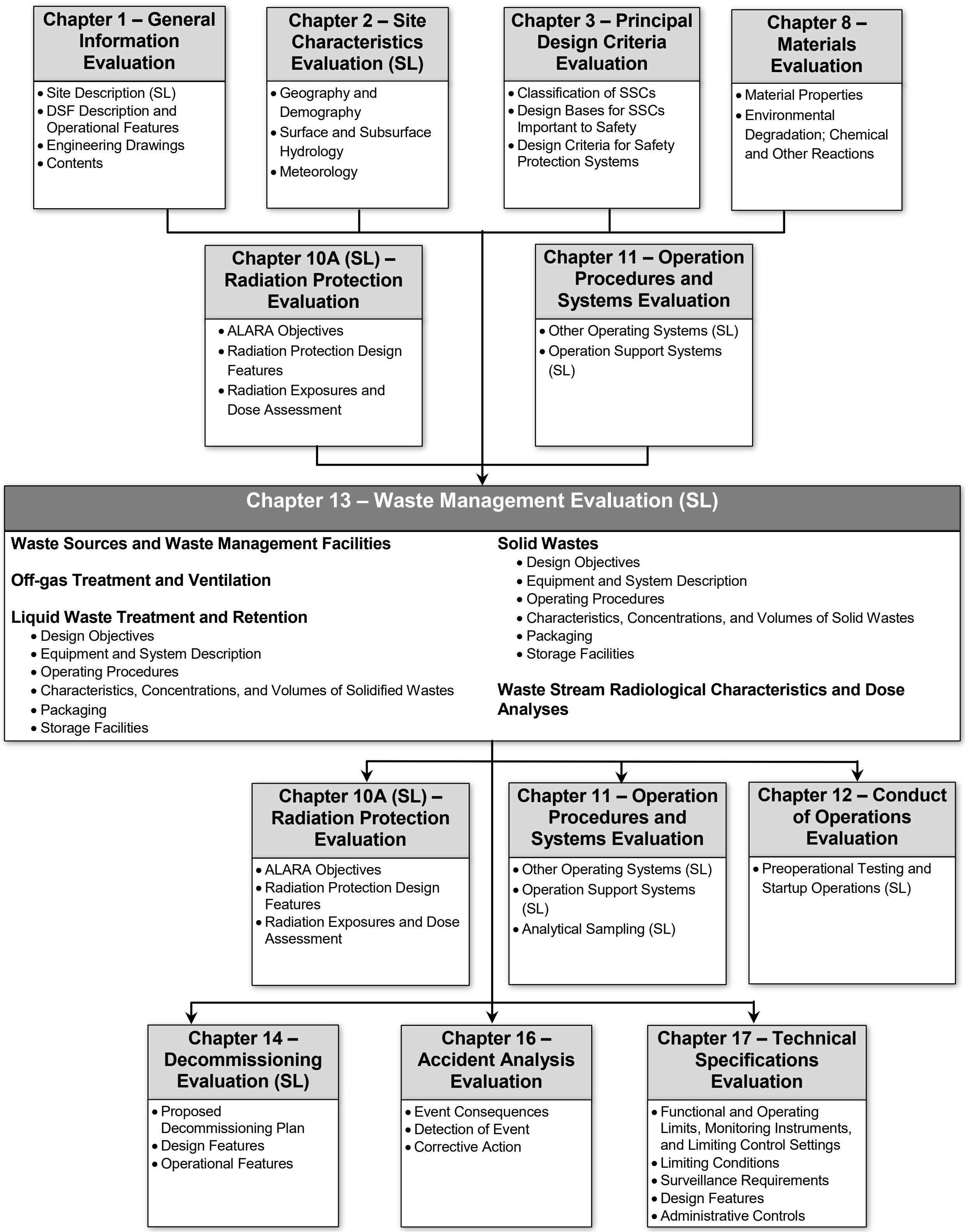

Figure 13-1 Overview of Waste Management Evaluation

13.5.1 Waste Sources and Waste Management Facilities

Determine that the SAR demonstrates that all waste materials generated as a result of facility operations will be safely contained until disposal.

Review the general description and operating features of the facility discussed in the general information chapter of the SAR. Verify that the descriptions include the DSF's waste management facilities and systems and provide the information identified in Section 13.4.1 above. Verify that the features of the facility design and operations reduce, to the extent practicable, the quantity of radioactive waste generated at the installation. Confirm that the waste management systems are adequately designed to handle, treat, and store the wastes generated at the DSF, including having an adequate capacity for the wastes to be generated over the facility lifetime. If applicable, compare flowcharts and facility drawings and diagrams to ensure that the waste confinement and management systems are designed to minimize the quantity of radioactive wastes generated. Verify that the types of waste generated are identified and characterized and that the identification and characterization are consistent with the DSF design and operations. Also verify that the method and design for treatment, handling, and the mode of storage of the wastes before disposal are sufficiently described and are generally accurate and acceptable.

Ensure that the SAR identifies all sources of waste, including on drawings or sketches. Consider the following list (not meant to be all-inclusive) of sources that can exist at a DSF, depending on the facility design and operations:

- wastes associated with normal operations
- filters and membranes (for liquids and from the heating, ventilation, and air conditioning systems of the waste management facilities)
- wipes
- heating, ventilation, and air conditioning duct flushing fluid
- laboratory samples
- decontamination station effluent
- disposable (one-time use) and reusable personal protective clothing and equipment
- laundry effluent (e.g., from washing personal protective clothing, clothing bags)
- contaminated equipment and tools
- radioactive waste containers and bags
- wastes associated with off-normal events and conditions and that, therefore, may be radioactive or handled as possibly radioactive
- confinement area sprinkler runoff
- earth contaminated by spills or from other causes

- pool-related wastes for a DSF with a pool (filters, membranes, materials skimmed or separated from pool water by cleanup systems, piping flushing fluids, coolant seepage and minor leaks, condensate on facility interior surfaces, pool coolant)

Review the waste analysis and check for potential interactions between nonradioactive chemical wastes or combustion products and radioactive materials. If applicable, review the method and design for the treatment, handling, and disposal of chemical wastes or combustion products.

Verify that the SAR describes the means by which waste management facilities and systems and operations will prevent degradation of the wastes and waste systems, including containers and tanks, and will confine the waste materials. Depending upon the design of the waste management SSCs (e.g., types and confinement capabilities of seals), releases to the environment under normal, off-normal, and accident conditions may be possible. Verify that the SAR presents estimates of radionuclides released to the environment for normal conditions, off-normal operations, and accident conditions. The estimates should be based on evaluation of the design and the physical processes of the actual waste management SSCs that will move radionuclides into the environment (e.g., vapor pressure in conjunction with convective flow) or retain them in the storage or holding systems. Have a clear understanding of the components that are designed to reduce the flow of radionuclides into the environment and their performance capabilities (e.g., filtration systems and their nuclide removal efficiencies). Ensure that the release estimates include the contributions from each component of the systems from which releases can occur. These releases should be added to the effluents normally intended to be discharged, or expected to be released, from the facility, if any.

Verify that the SAR estimates waste management facility emissions resulting from off-normal conditions, including possible emissions of radioactive gases from sealed fuel containers that may fail. Verify also that the SAR determines any waste management facility emissions that may result from accident conditions. The NRC accepts that other sources on the site can be assumed to be at normal conditions during such accident conditions unless the same initiating event affects these other sources.

13.5.2 Off-Gas Treatment and Ventilation

Review the design drawings, flowcharts, and narrative descriptions of off-gas treatment and ventilation systems design and operations. Confirm that the information in the SAR is sufficient to understand the design and operations of the systems, identify interfaces between individual systems, and evaluate the performance of the systems. In addition, coordinate with the radiation protection reviewer to evaluate the systems in accordance with the guidance in Section 10A.5.2.1, “Installation Design Features,” of this SRP. Confirm that the operations descriptions are consistent with the design and selection of equipment and facilities, general design criteria, and regulatory limits. Ensure that the description of the facility off-gas, waste treatment, and ventilation systems identifies the relevant regulatory requirements, design and performance objectives, and function and general design criteria, including safety margins. The design and design criteria for each unit system should adequately address site conditions and be based on, or include, reasonable estimates of airborne waste generation rates for normal, off-normal, and accident conditions. Ensure design criteria, descriptions, and analyses address the components (e.g., sealed waste containers, ductwork, filters) for all the systems.

Confirm that the systems service the appropriate portions of the facility based upon DSF design and operations descriptions. Verify that the descriptions of the systems design and operations demonstrate that the systems have sufficient capabilities, including capacity, to confine radioactive materials during projected operations conditions, including normal and off-normal conditions. Ensure that the design of the systems includes sufficient margins such that a single component failure will not result in an uncontrolled release of materials. Ensure that the SAR demonstrates that unit ventilation systems alone and in conjunction with other ventilation systems will be operable. Verify that the design includes satisfactory features for interfacing with other effluent and ventilation systems and with process off-gas equipment. Also ensure that the SAR descriptions demonstrate that the systems design and operations will effectively prevent or limit the spread of radioactive materials, including within the ventilation systems, and control the spread of contamination. In that regard, ensure that the applicant adequately considered potential bypasses, such as improper connections between radioactive systems and nonradioactive systems, and the possibility of uncontrolled and unmonitored effluent releases. In evaluating the proposed design and operations of these systems, consider the design and operations of systems for similar facilities that the NRC has reviewed and approved.

Verify that the design and operations descriptions include provisions to adequately monitor off-gas treatment and ventilation system performance, including such parameters as filter and other treatment device efficiency. Ensure that the design addresses replacement and disposal or regeneration of items such as filters and scrubber solution, including the treatment (with any transfers to other waste systems), handling, and disposal of these wastes. Verify that the design addresses potential personnel exposure and contamination that could result from handling operations.

Ensure that the design and operations of the ventilation and off-gas systems and equipment incorporate adequate consideration of ALARA principles and represent a reasonable effort to minimize releases and exposures (workers and public) to radioactive materials. This includes verifying that the design and operations descriptions of the systems demonstrate that radioactive releases during normal operations and radiation exposure levels will meet ALARA objectives.

Coordinate with the radiation protection and confinement reviewers (Chapters 10A and 9 of this SRP) to evaluate the ventilation and off-gas monitoring systems. Ensure that the SAR addresses the information on monitoring described in Section 13.4.2 above. Ensure that the selected equipment and parameters, locations, and release paths are adequate to ensure that the design criteria and regulatory requirements will be met. Ensure that monitoring processes and equipment are appropriate and reasonable for the expected release paths and materials expected in releases or that should otherwise be monitored for. Ensure that the equipment has adequate detection and alarm capabilities. Section 10A.5.2.5, “Area Monitoring and Effluent Monitoring Instrumentation,” of this SRP provides useful guidance and criteria for evaluating effluent monitoring.

13.5.3 Liquid Waste Treatment and Retention

Review the SAR descriptions, including drawings, flow sheets, and narrative descriptions, of liquid waste system design and operations. Confirm the information in the SAR is sufficient to understand the system design and operations, to identify interfaces with other DSF systems, and to evaluate the system’s performance. In addition, coordinate with the radiation protection reviewer to evaluate the system in accordance with the guidance in Section 10A.5.2.1 of this SRP. Confirm that the operations descriptions are consistent with the design and selection of equipment and facilities, general design criteria, and regulatory limits. Also ensure, based on the DSF design and operations descriptions, that the SAR adequately identifies and characterizes all liquid wastes, including their sources and expected generation rates and volumes, that may be generated as a result of DSF operations. Determine the reasonableness of the expected inventory levels and that handling, treatment and storage provisions (including any volume reduction and solidification processes) are sufficient to handle the projected wastes and inventory levels, with some level of margin as appropriate. Ensure that equipment and processes are adequate for the radiation levels of the various wastes.

Verify that the design includes satisfactory features for interfacing with DSF systems, including waste or effluent systems and equipment. Also ensure that the SAR descriptions demonstrate that the systems design and operations will effectively prevent, or limit, the spread of radioactive materials and control the spread of contamination. In that regard, ensure that the applicant adequately considered potential bypasses, such as improper connections between radioactive systems and nonradioactive systems, and the possibility of uncontrolled and unmonitored effluent releases. In evaluating the proposed design and operations of these systems, consider the design and operations of systems for similar facilities that the NRC has reviewed and approved.

13.5.3.1 *Design Objectives*

Review the design objectives and verify that the system can handle the expected volume of potentially radioactive liquid wastes generated during normal and off-normal operations, safely confine the wastes, and minimize personnel doses. Ensure that the design objectives clearly identify which waste streams will be processed to achieve volume reduction or solidification. Verify that all sources of liquid waste have been identified. Assess the applicant's estimates of expected inventories for each stream and determine whether they are reasonable for design purposes.

13.5.3.2 *Equipment and System Description*

Verify that the SAR describes the features, systems, and special handling techniques that are important to safety. Verify that pressure vessels, tanks, and piping systems important to safety will be constructed in accordance with the appropriate quality standard(s). Ensure that the SAR describes any backup or special features that will be used as necessary to ensure design objectives and regulatory requirements are met and adequately justify the selection of these features.

Review the design to ensure that (1) adequate measurement is provided (to determine liquid waste volume and radioactivity concentration and to monitor system performance), (2) the system is not vulnerable to contamination buildup, (3) liquid wastes entering the liquid waste systems do not include materials (e.g., oils, insoluble solids, solvents, hazardous wastes) that may adversely affect system performance, (4) secondary confinement is provided for waste lines outside of the confinement barriers, and (5) provisions are made, as necessary, for component shielding and cleanout or decontamination of piping.

Coordinate with the radiation protection reviewer to ensure that the SAR addresses the information on monitoring described in Section 13.4.3.2 above. Ensure that the selected equipment and parameters, locations, and release paths are adequate to ensure that the design criteria and regulatory requirements will be met. Ensure that the monitoring and equipment are appropriate and reasonable for all release paths and materials. Ensure that the equipment has adequate detection and alarm capabilities. Section 10A.5.2.5 of this SRP provides useful guidance and criteria for evaluating effluent monitoring.

13.5.3.3 *Operating Procedures*

Review the flow sheets and narrative descriptions of operations to verify that proposed design features and procedures will minimize liquid waste generation and the possibility of spills and provide for control and containment of spills. Verify that appropriate provisions are made for ensuring functional operation, including testing procedures, action levels, and associated actions for normal and off-normal conditions as well as means for monitoring and controlling limits.

13.5.3.4 *Characteristics, Concentrations, and Volumes of Solidified Wastes*

Review the applicant's description of the physical, chemical, and thermal characteristics of the solidified wastes and confirm that they are consistent with the applicant's estimates of liquid waste radionuclide concentrations and waste volumes generated. Verify that the solidified wastes are compatible with the design criteria and capacity of the liquid waste treatment and retention systems.

13.5.3.5 *Packaging for Onsite Storage*

Review the descriptions of solidified liquid waste containers and verify that the container specifications are compatible with the forms of waste for which the containers will be used. In making this determination, consider materials of construction (including welding design information on the critical boundary welds in regard to the minimum allowable weld joint size configuration, if appropriate), heat load, potential corrosive interactions of the waste and container materials, prevention of overpressurization, and confinement provided by the container under normal, off-normal, and accident conditions.

13.5.3.6 *Storage Facilities*

Review the description of storage facilities and operations and determine whether the storage capacity is consistent with the estimates of liquid and solidified waste volumes to be generated and stored over the life of the facility or the projected decay holding time (if not held for the entire life of the facility). Review proposed operations to ensure that the movement of containers into and out of storage, monitoring, equipment, waste routing, and spare storage volume (for liquid transfers) have been taken into account, as necessary. Ensure that provisions exist for spills, overflows, or leakage. Confirm that the SAR evaluates and describes damage from off-normal and accident conditions and evaluates the container integrity against corrosion from the environment within the waste storage facility. Verify that long-term storage options are reasonable in light of ultimate disposal plans and availability.

13.5.4 Solid Wastes

Review the SAR descriptions, including drawings, process flow diagrams, and narrative descriptions, of the solid waste system design and operations. Confirm that the information in the SAR is sufficient to understand the system design and operations, identify interfaces with other DSF systems, and evaluate the system's performance. Confirm that the operations descriptions are consistent with the design and selection of equipment and facilities, general design criteria, and regulatory limits. Also ensure, based on the DSF design and operations descriptions, that the SAR adequately identifies and characterizes all solid wastes, including their sources and expected generation rates and volumes, that may be generated as a result of DSF operations. Verify that the design includes satisfactory features for interfacing with DSF systems, including waste or effluent systems and equipment. Also ensure that the SAR descriptions demonstrate that the systems design and operations will effectively prevent, or limit, the spread of radioactive materials and control the spread of contamination. In evaluating the proposed design and operations of these systems, consider the design and operations of systems for similar facilities that the NRC has reviewed and approved.

13.5.4.1 *Design Objectives*

Verify that the system is capable of handling, treating, and storing the projected wastes (including potentially radioactive and nonradioactive wastes) and waste volumes generated during normal and off-normal operations. Specifically, review waste generated from the use of personal protective clothing and equipment that is to be treated as potentially contaminated because these items typically constitute a large portion of the total volume of waste.

13.5.4.2 *Equipment and System Description*

Review the descriptions of equipment and systems, including drawings, for solid waste collection and treatment to ensure that (1) features, systems, and special handling techniques that are important to safety have been identified; (2) locations of equipment and associated features that are used for volume reduction, confinement, or packaging, storage, and disposal are identified; and (3) provisions exist for shielding, confinement, handling, and decontamination, as necessary, to ensure that occupational doses are maintained to meet ALARA objectives and to minimize doses to the public from these wastes.

Coordinate with the radiation protection reviewer to ensure that the SAR addresses the information on monitoring described in Section 13.4.4.2 above. Ensure that the selected equipment, parameters, and locations are adequate to ensure that the design criteria and regulatory requirements will be met. Ensure that the monitoring and equipment are appropriate and reasonable for the purposes for which they are to be used, including monitoring of system performance. The equipment should have adequate detection and alarm capabilities. Section 10A.5.2.5 of this SRP provides useful guidance and criteria for evaluating monitoring and monitoring instrumentation.

13.5.4.3 *Operating Procedures*

Review the procedures associated with solid waste system, equipment operations, and use of instrumentation and verify that the SAR properly addresses performance testing, process limits, and means for monitoring and controlling identified process limits. Ensure that the SAR provides action levels and associated response actions for normal and off-normal conditions.

13.5.4.4 *Characteristics, Concentrations, and Volumes of Solid Wastes*

Review the applicant's description of the physical, chemical, radiological, and thermal characteristics of the solid wastes and the estimates of waste volumes generated. Verify that the solid wastes are compatible with the design criteria and capacity of the solid waste treatment and retention systems.

13.5.4.5 *Packaging for Onsite Storage*

Review the descriptions of solid waste containers and verify that the container specifications are appropriate for the forms of waste for which the containers will be used. As with solidified wastes (but to a lesser extent), consider materials of construction, heat load, potential corrosive interactions of the waste and container materials, and confinement provided by the container under off-normal and accident conditions.

If an onsite laundry is to be used, verify that the SAR adequately describes the containers and procedures for the transfer of potentially contaminated items and that such containers and procedures are reasonable. If the laundry is off site, ensure that the SAR describes the appropriate procedures for shipping the containers to the offsite laundry facility and addresses the applicable NRC and Department of Transportation regulatory requirements for offsite transportation.

13.5.4.6 *Storage Facilities*

If solid wastes are to be held until site decommissioning or for radioactive decay, review the description of the storage facilities and operations and determine whether the storage capacity is consistent with the estimates of solid and solidified waste volumes to be generated and stored over the life of the facility or the projected decay holding time (if not held for the entire life of the facility). Also ensure, in coordination with the confinement reviewer, that facility design and operations include appropriate confinement features and monitoring and maintain container integrity against conditions such as corrosion from the environment within the waste storage facility. Review the proposed operations to ensure that the SAR adequately addresses, as necessary, the movement of containers into and out of storage, monitoring, equipment, and impacts of operation conditions (normal, off-normal, and accident conditions). Verify that long-term storage options are reasonable in light of ultimate disposal plans and availability.

13.5.5 Waste Stream Radiological Characteristics and Dose Analyses

Verify that the SAR includes the information, analyses, and results identified in Section 13.4.5 of this SRP. Confirm the completeness and accuracy of the information and analyses based on the DSF design and operations and the characteristics of the proposed DSF site. Verify that the analyses account for each system process and facility activity which could result in the generation of wastes or effluent releases during routine operations and off-normal conditions. Ensure that the results include radionuclide concentrations, doses, and dose rates for each waste stream and waste locations (e.g., storage containers, tanks, and discharge or leak points) and are provided for normal, off-normal, and accident conditions.

Coordinate with the shielding, confinement, and radiation protection reviewers to evaluate the appropriateness and adequacy of the methods used to determine dose rates, doses, and nuclide concentrations as well as the results themselves. Verify that the doses and dose rates include both direct radiation and effluent contributions. For airborne effluent releases, evaluate the proposed short- and long-term atmospheric dispersion (X/Q) and deposition (D/Q) parameters the applicant used and confirm that they are appropriate for calculating gaseous effluent concentrations and doses based on the meteorology information presented in the site characteristics evaluation chapter of the SAR (see SRP Sections 2.4.3, “Meteorology,” and 2.5.3, “Meteorology”). Section 13.4.2 above summarizes facility design parameters that should also be accounted for in the review. RG 1.23, “Meteorological Monitoring Programs for Nuclear Power Plants (Safety Guide 23),” and RG 1.111, “Methods for Estimating Atmospheric Transport and Dispersion of Gaseous Effluents in Routine Releases from Light-Water-Cooled Reactors,” describe acceptable methods to develop the associated atmospheric dispersion and deposition parameters in evaluating the consequences associated with postulated releases of radioactive materials during routine operations and anticipated occurrences, as well as accident conditions.

Confirm that the analyses for airborne effluent releases account for all of the types of system processes or facility activities that could result in routine airborne effluent releases or releases from anticipated occurrences and adequately characterize their radioactive source terms. Ensure that the analyses properly, or conservatively, account for (1) processes by which radioactive materials can, or are assumed to, be released in the environment from the systems; (2) system design features for which credit is applied to mitigate radioactive material releases; (3) the duration of such releases (e.g., continuous or periodic); atmospheric dispersion; and (4) deposition. Confirm that the SAR provides appropriate bases for the selection of downwind sector(s).

Evaluate the analyses of any liquid effluents to confirm that the analyses appropriately and adequately characterize the nuclide concentrations, doses, and dose rates resulting from these effluents. Ensure that the analyses adequately account for the mechanisms for movement of these effluents within the environment, including addressing similar considerations as described above for airborne effluents, as applicable. Section 2.5.4.9, "Environmental Assessment of Effluents," of this SRP provides useful guidance for evaluating the analyses of liquid effluents.

Coordinate with the radiation protection reviewer to ensure that the information and results from the waste management evaluation are sufficient to support the radiation protection evaluation (see Chapter 10A of this SRP). This includes ensuring that the doses and dose rates, and, as applicable, the radionuclide concentrations in effluents are calculated for all locations relevant to the radiation protection evaluation for doses to personnel and the public. The radiation protection reviewer will use the results of the waste management evaluation in combination with the results of the shielding and confinement evaluations to evaluate doses to DSF workers and the public and to evaluate compliance with the radiological requirements in the regulations. Depending on the approach for some analyses, this may include evaluations of radionuclide concentrations in the gaseous and liquid effluents.

13.6 Evaluation Findings

The NRC reviewer should prepare evaluation findings on satisfaction of the regulatory requirements in Section 13.4 of this SRP. If the documentation submitted with the application fully supports positive findings for each of the regulatory requirements, the statements of finding should be similar to the following:

F13.1 The SAR adequately describes acceptable features of the [DSF designation] design and operating modes that reduce, to the extent practicable, the radioactive waste volumes generated by the installation, in compliance with 10 CFR 72.24(f) and 10 CFR 72.128(a)(5).

F13.2 The SAR adequately describes acceptable equipment to be installed to maintain control over radioactive materials in gaseous and liquid effluents produced during normal operations and expected operational occurrences; estimated radionuclide releases; and provisions for packaging, storage, and disposal of solid wastes containing radioactive materials resulting from treatment of gaseous and liquid effluents and from other sources, in compliance with 10 CFR 72.24(l).

F13.3 The SAR provides evaluations of the waste confinement and management activities that are sufficient to demonstrate that the activities to be authorized by the license will not endanger public health and safety, in compliance with 10 CFR 72.40(a)(13).

F13.4 [If the DSF is located over an aquifer that is a major water resource (which may be interpreted as over any ground water)]: The [DSF designation] design and operations provide acceptable measures to preclude the transport of radioactive materials from the waste management facilities to the environment through the aquifer, in compliance with 10 CFR 72.122(b)(4).

F13.5 [If appropriate] The SAR evaluations of the waste management activities are sufficient to demonstrate that the effects of operation of the proposed [DSF designation] combined with those of other nuclear facilities at the site will not constitute an unreasonable risk to the health and safety of the public, in compliance with 10 CFR 72.122(e).

F13.6 [If appropriate] The design of the [DSF designation] provides acceptable ventilation and off-gas systems to ensure the adequate confinement of airborne radioactive particulate materials during normal or off-normal conditions, in compliance with 10 CFR 72.122(h)(3).

F13.7 [If appropriate] The design of the [DSF designation] provides [an] acceptable effluent system[s], which include[s] means for measuring the amount of radionuclides in effluents during normal operations and under accident conditions, and for measuring the flow of the diluting medium, in compliance with 10 CFR 72.126(c).

F13.8 The design of the [DSF designation] acceptably provides means to limit the release of radioactive materials in effluents during normal operation and to control the release of radioactive materials under accident conditions, in compliance with 10 CFR 72.126(d).

F13.9 The design of the [DSF designation] includes radioactive waste treatment facilities that include a capability for packing site-generated, low-level wastes in a form suitable for storage on site while awaiting transfer to disposal sites, in compliance with 10 CFR 72.128(b).

F13.10 The SAR provides reasonable and appropriate information, including dose rates, to enable performance of the evaluations required in 10 CFR 72.24(e) and 10 CFR 72.24(m) and to allow evaluation of the DSF's compliance with the radiation protection requirements for members of the public in 10 CFR 72.104, 10 CFR 72.106, and 10 CFR Part 20. These evaluations are described in the radiation protection review (SRP Chapter 10A).

The reviewer should provide a summary statement similar to the following:

> The proposed waste management system designs and operations provide reasonable assurance that wastes generated as a result of DSF operations will be managed in a manner that supports safe storage of SNF, reactor-related GTCC waste, or HLW at the DSF. This finding is reached on the basis of a review that considered the regulation itself, appropriate regulatory guides, accepted practices, the statements and representations in the application, and the staff's independent, confirmatory evaluations.

13.7 References

10 CFR Part 20, "Standards for Protection against Radiation."

10 CFR Part 72, "Licensing Requirements for the Independent Storage of Spent Nuclear Fuel, High-Level Radioactive Waste, and Reactor-Related Greater than Class C Waste."

NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition."

Regulatory Guide 1.23, "Meteorological Monitoring Programs for Nuclear Power Plants (Safety Guide 23)."

Regulatory Guide 1.111, "Methods for Estimating Atmospheric Transport and Dispersion of Gaseous Effluents in Routine Releases from Light-Water-Cooled Reactors."

Regulatory Guide 1.143, "Design Guidance for Radioactive Waste Management Systems, Structures, and Components Installed in Light-Water-Cooled Nuclear Power Plants."

14 DECOMMISSIONING EVALUATION (SL)

14.1 Review Objective

The objective of the U.S. Nuclear Regulatory Commission's (NRC's) review with regard to decommissioning is to ensure that the applicant's provisions for eventual decontamination and decommissioning of the independent spent fuel storage installation (ISFSI) or monitored retrievable storage installation (MRS) provide reasonable assurance that (1) the proposed provisions for eventual decontamination and decommissioning of the ISFSI or MRS will provide adequate protection of public health and safety, (2) funds will be available to decommission the ISFSI or MRS, and (3) the design and operational features of the ISFSI or MRS facilitate eventual decontamination and decommissioning.

14.2 Applicability

This chapter applies to the review of applications for specific licenses **(SL)** for an ISFSI or an MRS facility, categorized as a dry storage facility (DSF). For sites with a general license in accordance with Title 10 of the *Code of Federal Regulations* (10 CFR) 72.210, "General license issued," the decommissioning review is conducted as part of the licensing review under 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities," or 10 CFR Part 52, "Licenses, Certifications, and Approvals for Nuclear Power Plants."

14.3 Areas of Review

This chapter addresses the following areas of review:

- proposed decommissioning plan
- decommissioning funding plan
- design features
- operational features

14.4 Regulatory Requirements and Acceptance Criteria

This section summarizes those parts of 10 CFR Part 72, "Licensing Requirements for the Independent Storage of Spent Nuclear Fuel, High-Level Radioactive Waste, and Reactor-Related Greater Than Class C Waste," that are relevant to the review areas addressed by this chapter. The NRC staff reviewer should refer to the exact language in the regulations. Table 14-1 matches the relevant regulatory requirements to the areas of review covered in this chapter.

Table 14-1 Relationship of Regulations and Areas of Review

Areas of Review	10 CFR Part 72 Regulations			
	72.22(e)(3)	72.24	72.30	72.130
Proposed Decommissioning Plan		(q)●	●	
Decommissioning Funding Plan	●		●	
Design Features		(f)(g)●		●
Operational Features		(f)(g)●		

Decommissioning of a DSF must meet the requirements for license termination in Subpart E, "Radiological Criteria for License Termination," to 10 CFR Part 20, "Standards for Protection Against Radiation." The requirements related to eventual decommissioning of the DSF applicable at the time of initial licensing are satisfied if the applicant adequately addresses the following acceptance criteria for the proposed decommissioning plan (DP), decommissioning funding plan, design features, and operational features.

14.4.1 Proposed Decommissioning Plan

The NRC's regulations in 10 CFR 72.24(q) and 10 CFR 72.30, "Financial assurance and recordkeeping for decommissioning," require an applicant to submit a proposed DP with the license application. The proposed DP must include a decommissioning funding plan (10 CFR 72.30(b)) containing information on how reasonable assurance will be provided that funds will be available to decommission the DSF (see Section 14.4.2 below). As required in 10 CFR 72.30(a), the proposed DP must also discuss the design features of the DSF that will facilitate decontamination and decommissioning (see Section 14.4.3 below). DOE is exempt from financial assurance requirements per 10 CFR 72.22(e), 10 CFR 72.40(a)(6) and 10 CFR 72.40(a)(10).

The proposed DP must contain information on proposed practices and procedures for the decontamination of the site and facilities and for the disposal of residual radioactive materials after all spent nuclear fuel (SNF), high-level radioactive waste (HLW), and reactor-related greater-than-Class-C (GTCC) waste have been removed from the facility (10 CFR 72.30(a)). This information must be sufficient to provide reasonable assurance that the licensee can conduct eventual decontamination and decommissioning of the DSF in a manner that adequately protects the health and safety of workers and the public (10 CFR 72.30(a)).

In accordance with 10 CFR 72.54, "Expiration and termination of licenses and decommissioning of sites and separate buildings or outdoor areas," a licensee is not required to submit a "final" DP for NRC review and approval until one of the criteria in 10 CFR 72.54(d) is triggered. The proposed DP submitted at the time of license application is expected to be conceptual in nature. It should identify and discuss the anticipated types of contamination and waste generated and the anticipated practices and procedures for decontamination, decommissioning, and disposal of residual radioactive materials. It should be in sufficient detail to support the decommissioning cost estimate required by 10 CFR 72.30(b). The proposed DP should include a commitment to submit a timely final DP for NRC review and approval before initiating decommissioning activities, in accordance with 10 CFR 72.54. The proposed DP should include a commitment to decommission the facility for unrestricted use in accordance with the radiological criteria for license termination in 10 CFR Part 20, Subpart E.

14.4.2 Decommissioning Funding Plan

The NRC's regulations in 10 CFR 72.22(e)(3) and 10 CFR 72.30 require that the application include a decommissioning funding plan. The decommissioning funding plan must demonstrate reasonable assurance that the applicant will have sufficient funds such that decommissioning will be completed after the removal of SNF, HLW, and reactor-related GTCC waste, as appropriate, from the site (10 CFR 72.30(b)(1)). The decommissioning funding plan must contain a detailed cost estimate for an independent contractor to remediate the site to unrestricted release criteria and an adequate contingency factor (10 CFR 72.30(b)(2)). It must also include means for adjusting cost estimates and associated funding levels periodically over the life of the facility (10 CFR 72.30(b)(4)).

Chapter 4, "Financial Assurance Overview," and Appendix A, "Standard Format and Content of Financial Assurance Mechanisms for Decommissioning," to NUREG-1757, Volume 3, "Consolidated NMSS Decommissioning Guidance: Financial Assurance, Recordkeeping, and Timeliness," provide guidance on decommissioning financial assurance and submittal of the decommissioning funding plan. DOE is exempt from financial assurance requirements per 10 CFR 72.22(e), 10 CFR 72.40(a)(6) and 10 CFR 72.40(a)(10).

14.4.3 Design Features

The NRC's regulations in 10 CFR 72.24(f), 10 CFR 72.30, and 10 CFR 72.130, "Criteria for decommissioning," require that the application identify and discuss the features included in the design of the DSF that will facilitate decommissioning. This includes decontamination of structures and equipment, minimizing the quantity of radioactive wastes and contaminated equipment, and facilitating the removal of radioactive wastes and contaminated materials at the time the DSF is permanently decommissioned.

Design features include surfaces that are less susceptible to contamination (or activation) and are readily decontaminated, as well as shielding to minimize any occupational exposure associated with decontamination. Design features also include equipment to facilitate the decontamination and removal of air circulation and filtration systems, and components of waste treatment and packaging systems.

14.4.4 Operational Features

The NRC's regulations in 10 CFR 72.24(f) and 10 CFR 72.130 require that the application identify the operational features or provisions that will facilitate eventual decontamination and decommissioning of the DSF and reduce radioactive waste volumes generated at the facility. Such features include minimizing contamination buildup on components and maintaining records of information important to decommissioning, such as records of spills or other unusual occurrences involving the spread of contamination and accurate as-built drawings. This information may be in the safety analysis report (SAR) or the proposed DP.

14.5 Review Procedures

Figure 14-1 shows the interrelationship between the decommissioning evaluation and the other areas of review described in this standard review plan (SRP).

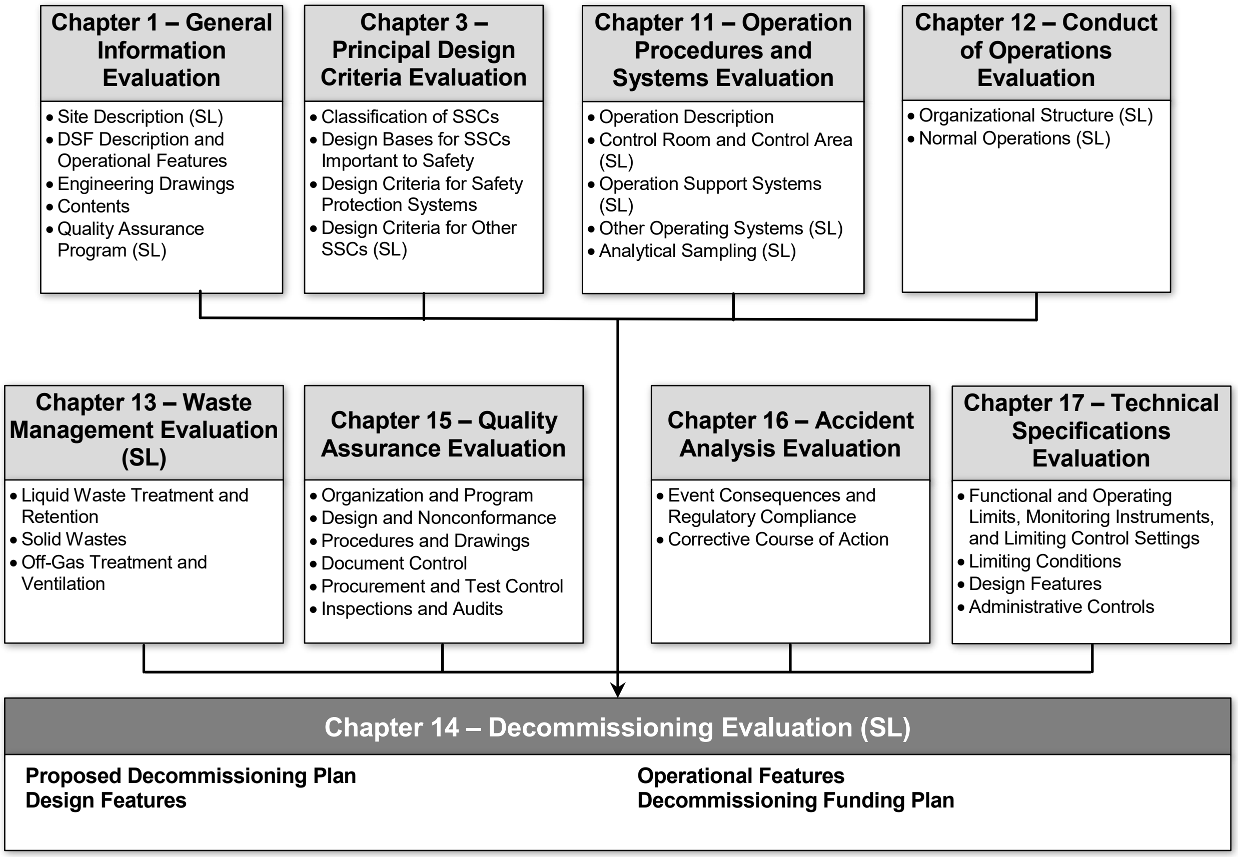

Figure 14-1 Overview of Decommissioning Evaluation

14.5.1 Proposed Decommissioning Plan

Evaluate the applicant's proposed DP and verify that it addresses the areas described below to ensure that it provides reasonable assurance that the licensee can conduct eventual decontamination and decommissioning in a manner that adequately protects the health and safety of workers and the public.

Verify that the applicant has identified the anticipated types of waste to be generated, the anticipated types of contamination, and the anticipated practices and procedures for decontamination, decommissioning, and disposal of residual radioactive materials after all SNF, HLW, and reactor-related GTCC waste have been removed from the site.

Verify that the applicant has committed to submit a timely final DP for NRC review and approval before initiating decommissioning activities, in accordance with 10 CFR 72.54. Verify that the applicant has committed to decommission the facility in accordance with the radiological criteria for unrestricted use for license termination in 10 CFR Part 20, Subpart E.

Although the proposed DP specifically applies to activities licensed under 10 CFR Part 72, there may be interrelationships with other licensed activities, including collocated 10 CFR Part 50 or 10 CFR Part 52 facilities. Evaluate any proposed provisions intended to accommodate conditions associated with the collocated facilities.

14.5.2 Decommissioning Funding Plan

Coordinate the review of the decommissioning funding plan with the NRC Office of Nuclear Reactor Regulation or the Office of Nuclear Material Safety and Safeguards, Division of Decommissioning, Uranium Recovery, and Waste Programs.

Specific guidance for reviewing decommissioning financial assurance appears in NUREG-1757, Volume 3 (Chapter 4 and Appendix A).

Ensure the application includes a decommissioning funding plan containing information on how the applicant provides reasonable assurance that funds will be available to decommission the ISFSI or MRS. Ensure that the decommissioning funding plan includes the following:

- a detailed cost estimate for decommissioning, in an amount that reflects the cost of an independent contractor to perform all decommissioning activities, the cost of meeting the unrestricted use criteria in 10 CFR 20.1402, "Radiological criteria for unrestricted use," and an adequate contingency factor

- identification of and justification for using the key assumptions contained in the decommissioning cost estimate

- a description of the method of assuring funds for decommissioning (using one of the permissible methods of assuring funds listed in 10 CFR 72.30(e)), including means for adjusting cost estimates and associated funding levels periodically over the life of the facility

- the potential volume of onsite subsurface material containing residual radioactivity that will require remediation to meet the criteria for license termination

- a certification that financial assurance for decommissioning has been provided in the amount of the cost estimate for decommissioning

Ensure that the applicant has a plan to update and resubmit the decommissioning funding plan, as specified in 10 CFR 72.30(c) and 10 CFR 72.30(d), and maintain records of the cost estimate performed for the decommissioning funding plan and records of the funding method used for assuring funds, as provided in 10 CFR 72.30(f)(4). The applicant should also ensure that the funds will only be used for decommissioning activities and to monitor and replenish decommissioning funds and report to the NRC, as provided in 10 CFR 72.30(g).

14.5.3 Design Features

Ensure that the SAR or DP identifies the design features of the ISFSI or MRS that will facilitate decontamination and decommissioning, as required in 10 CFR 72.24(f), 10 CFR 72.30, and 10 CFR 72.130. Examine the SAR and the proposed DP for this information. The design can be considered to meet these requirements if (1) the applicant incorporated provisions that are feasible and economic and (2) the applicant has selected design choices over competing alternatives that support decommissioning, or an acceptable rationale for not adopting the most favorable alternatives is provided. The NRC has accepted the priority of important to safety features and capabilities over decommissioning considerations when such tradeoffs arise.

In determining whether the design facilitates decommissioning, consider the extent to which the applicant has selected design features that have characteristics favorable to decommissioning, such as the following:

- Select materials and processes to minimize waste production.
- Minimize mass of shielding materials subject to activation.
- Facilitate future demolition and removal by use of modular design and inclusion of lifting points (with anticipation of the size of containers that may be used for transportation and permanent disposal).
- Select materials compatible with projected decommissioning and waste processing.
- Use finishes with minimum surface roughness on structures, systems, and components.
- Use selected coatings to preclude penetration of radioactive gas, condensate, or deposited aerosols (if present) into porous materials to permit future decontamination by surface treatment.
- Incorporate features to contain leaks and spills.
- Consider current industry technology for the minimization of waste production.
- Conduct a radiation survey of the proposed site of the ISFSI or MRS before construction to facilitate eventual demonstration of compliance with decommissioning criteria.

In performing these design reviews, ensure that the design features have adequately considered health and safety, including provisions to maintain occupational and public radiation exposures to as low as reasonably achievable during decommissioning.

Refer to Regulatory Guide (RG) 4.21, “Minimization of Contamination and Radioactive Waste Generation: Life-Cycle Planning,” which provides guidance for the implementation of 10 CFR 20.1406, “Minimization on Contamination.”

14.5.4 Operational Features

Review the SAR and DP for operational features or provisions that facilitate eventual decommissioning and reduce radioactive waste volume. Verify with the reviewers of the conduct of operations (SRP Chapter 12, “Conduct of Operations Evaluation”) and waste management (SRP Chapter 13, “Waste Management Evaluation”) that the applicant has procedures, processes, or programs to identify and minimize the spread of contamination. Verify that the applicant has committed to a plan to keep records of information important to decommissioning as provided in 10 CFR 72.30(f), such as spills or other unusual occurrences, until the license is terminated. Records should include information on contamination that may have spread to inaccessible areas, as in the case of seepage into porous materials such as concrete. Records should include any known information on the identification of nuclides, quantities, forms, and concentrations. Verify that the applicant has a plan to maintain records of as-built drawings and modifications (or suitable substitute records if drawings are not available) of structures and equipment in areas where radioactive materials are used, handled, transferred, or stored and of locations of possible inaccessible contamination.

Consult with the reviewer for waste management to determine whether proposed operations of waste management systems have adequately addressed the facilitation of decommissioning. Consult with the radiation protection reviewer (SRP Chapter 10A) to determine whether the proposed health physics surveys and recordkeeping will facilitate decommissioning.

Refer to RG 4.22, “Decommissioning Planning During Operations,” which provides guidance for the implementation of 10 CFR 20.1406 on the minimization of contamination.

14.6 Evaluation Findings

The NRC reviewer should prepare evaluation findings on satisfaction of the regulatory requirements in Section 14.4 of this SRP. If the documentation submitted with the application fully supports positive findings for each of the regulatory requirements, the statements of findings should be similar to the following:

F14.1 The staff has reviewed the proposed decommissioning plan the applicant submitted for the [DSF designation] and the description of the plan in the SAR. The staff has determined that the proposed decommissioning plan the applicant submitted contains sufficient information on the proposed practices and procedures for decontamination of the site and disposal of residual radioactive materials to provide reasonable assurance that the eventual decommissioning of the [DSF designation] at the end of its useful life will provide adequate protection of the health and safety of the public. The staff, therefore, concludes that the proposed decommissioning plan complies with 10 CFR 72.24(q) and 10 CFR 72.30(a).

F14.2 The staff has reviewed the decommissioning funding plan the applicant submitted for the [DSF designation]. The staff has determined that (a) the decommissioning funding plan the applicant submitted is sufficient to provide reasonable assurance that costs related to decommissioning as characterized by the proposed decommissioning plan have been adequately estimated, (b) the financial assurance method the applicant described is sufficient to provide reasonable assurance that adequate funds will be available to decommission the facility at the end of its useful life, and (c) the applicant has provisions for adjusting cost estimates and associated funding levels periodically over the life of the [DSF designation] and plans to maintain records of the cost estimate performed for the decommissioning funding plan and records of the funding method used for assuring funds. The staff, therefore, concludes that the decommissioning funding provisions comply with 10 CFR 72.22(e)(3), 10 CFR 72.30(b), 10 CFR 72.30(e), and 10 CFR 72.30(f)(4).

F14.3 The staff has reviewed the application for the [DSF designation]. The staff has determined that the applicant has identified and discussed those design features of the [DSF designation] that facilitate decontamination and decommissioning, minimize the quantity of radioactive wastes and contaminated equipment, and facilitate removal of radioactive wastes and contaminated materials at the time the [DSF designation] is permanently decommissioned. The staff, therefore, concludes that the application complies with 10 CFR 72.24(f), 10 CFR 72.30(a), and 10 CFR 72.130.

F14.4 The staff has reviewed the application for the [DSF designation]. The staff has determined that the applicant identifies and discusses those [DSF designation] operating modes that reduce radioactive waste volumes generated at the installation and facilitate decontamination and decommissioning, and includes plans to maintain records of information important to decommissioning. The staff, therefore, concludes that the application complies with 10 CFR 72.24(f) and 10 CFR 72.30(f).

The reviewer should provide a summary statement similar to the following:

> The staff concludes that the decommissioning plan and decommissioning funding plan of the [DSF designation] are in compliance with 10 CFR Part 72 and that the applicable acceptance criteria have been satisfied. The evaluation of the decommissioning plan provided in the SAR offers reasonable assurance that the [DSF designation] will, at the conclusion of the safe storage of SNF, HLW, and reactor-related GTCC waste (as applicable), enable remediation of the site and termination of the license in accordance with 10 CFR Part 20, Subpart E. This finding is based on a review that considered the regulations, appropriate regulatory guides, applicable codes and standards, and accepted practices.

14.7 References

10 CFR Part 20, "Standards for Protection against Radiation."

10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities."

10 CFR Part 52, "Licenses, Certifications, and Approvals for Nuclear Power Plants."

10 CFR Part 72, "Licensing Requirements for the Independent Storage of Spent Nuclear Fuel, High-Level Radioactive Waste, and Reactor-Related Greater than Class C Waste."

NUREG-1757, Volume 3, "Consolidated NMSS Decommissioning Guidance: Financial Assurance, Recordkeeping, and Timeliness."

Regulatory Guide 4.21, "Minimization of Contamination and Radioactive Waste Generation: Life-Cycle Planning."

15 QUALITY ASSURANCE EVALUATION

15.1 Review Objective

The objective of the U.S. Nuclear Regulatory Commission's (NRC's) quality assurance (QA) review is to determine whether the applicant for a dry storage system (DSS) certificate or a specific license for a dry storage facility (DSF) has submitted a quality assurance program description (QAPD) in the applicant's safety analysis report (SAR). The QAPD must demonstrate that the applicant's QA program complies with the requirements of Subpart G, "Quality Assurance," of Title 10 of the *Code of Federal Regulations* (10 CFR) Part 72, "Licensing Requirements for the Independent Storage of Spent Nuclear Fuel, High-Level Radioactive Waste, and Reactor-Related Greater than Class C Waste."

The basis for that determination is developed from an evaluation of the applicant's high-level QAPD against the 18 criteria provided in Section 15.5, "Review Procedures," of this standard review plan (SRP) chapter, 10 CFR Part 72, and any associated information found in the *Federal Register* since the last rulemaking has been completed, as applicable. (Note: The scope of review does not include actual procedures and instructions that implement the QA program, although they may be described in the QAPD.)

The determination that the applicant's QA program is in compliance occurs during the NRC inspection activities that evaluate implementation of the QA plan. (Note: The scope of an inspection does include the actual procedures and instructions that implement the QA program.)

15.2 Applicability

This chapter applies to the review of applications for specific licenses for an independent spent fuel storage installation (ISFSI) or a monitored retrievable storage installation (MRS), categorized as a DSF. It also applies to the review of applications for a certificate of compliance (CoC) of a DSS for use at a general license facility.

15.3 Areas of Review

This chapter addresses the following areas of review:

- QA organization
- QA program
- design control
- procurement document control
- instructions, procedures, and drawings
- document control
- control of purchased material, equipment, and services
- identification and control of materials, parts, and components
- control of special processes
- licensee and certificate holder inspection
- test control
- control of measuring and test equipment
- handling, storage, and shipping control
- inspection, test, and operation status
- nonconforming materials, parts, or components

- corrective action
- quality assurance records
- audits

15.4 Regulatory Requirements and Acceptance Criteria

The NRC staff reviewer should refer to the exact language in 10 CFR Part 72, Subpart G.

The acceptance criteria in Section 15.5, below reflect the 18 quality criteria in 10 CFR Part 72, Subpart G, and describe the information to be included in the applicant's QAPD. Examples of measures are provided for each criterion to assist the reviewer in determining whether the QAPD meets the applicable criterion. For each of the activities and items identified as important to safety, the applicant should identify the applicable QA programmatic elements and include, as applicable, provisions for meeting each of the quality criteria listed in Section 15.5.

15.5 Review Procedures

The purpose of the QA review is to obtain reasonable assurance that the applicant has developed and described a QA program for design, fabrication, construction, testing, operations, modification, and decommissioning activities associated with structures, systems, and components (SSCs) important to safety.

It is important that the applicant's QAPD and associated portions of the SAR provide sufficient detail to enable the reviewer to assess whether the applicant has committed to comply with the program and that the QA program complies with the applicable requirements in 10 CFR Part 72, Subpart G. Section 15.6, "Evaluation Findings," of this SRP describes the course of action if the reviewer determines that sufficient detail does not exist in the QAPD. If the QAPD indicates a commitment to follow certain standards or codes, then the reviewer should consider the commitments as an integral part of the QA program.

An application for QA program approval may be included as a section of the applicant's SAR or it may be separate from the SAR. Because some aspects of the QA program may be described in different portions of the application (the SAR or a submittal separate from the SAR), consider the entire description when evaluating the program against the acceptance criteria. Therefore, if possible, coordinate the QAPD review with other aspects of the review, as shown in Figure 15-1. Such coordination will allow reviewers to derive a more accurate and complete assessment of the applicant's level of commitment to the overall QA program, the selection of quality criteria and quality levels, and the proposed implementation methods.

The applicant's QA program may be structured to apply QA measures and controls to all activities and items in proportion to their importance to safety, commonly referred to as a graded approach. The QAPD should address the use of a graded approach for the application of QA by adequately assigning appropriate grading classifications and providing an associated justification. However, an applicant may instead choose to apply the highest level of QA and control to all activities and items. The QA program should identify the items and attributes that are important to safety and the degree or category, as applicable, of their importance. For application of a graded approach, the highly important-to-safety activities and items must have a high level of quality control, whereas those less important may have a lower level of quality control. If the QA program is graded, the staff should be able to conclude that the structure of the graded program is acceptable and that the highest levels of QA are applied to those SSCs that are most important to safety. In making determinations about the application of QA to those SSCs important to safety, coordinate with the appropriate NRC project manager and associated technical staff to possibly evaluate other chapters or portions of the applicant's submittal. In evaluating the QA program, the QA reviewer may also use NUREG/CR-6314, "Quality Assurance Inspections for Shipping and Storage Containers," as an additional source of information in determining the program's compliance with regulatory requirements.

If the reviewer finds the QAPD submitted as part of a SAR to be acceptable, this should be documented in the safety evaluation report (SER). If the applicant's QAPD was submitted before submittal of the SAR, the acceptance of the QAPD should be documented in a letter to the applicant and, if possible, appended to the SER at a later time. In either case, the documentation of the review should include the basis for acceptance as noted in Section 15.6 of this SRP. Section 15.6 also describes the process for making any recommendations (requests for additional information process) for modifications to the application that are required before the application can be accepted.

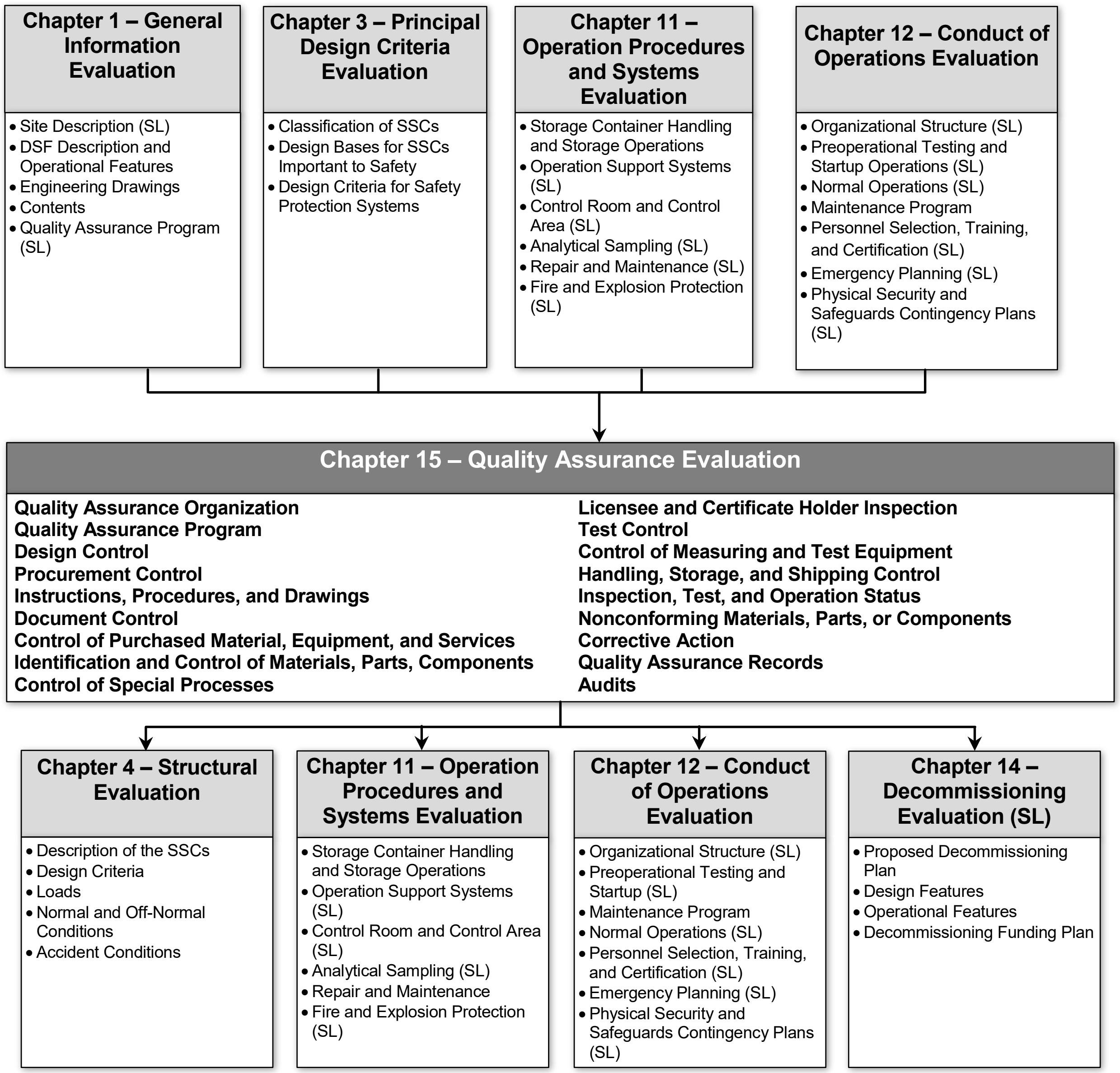

Figure 15-1 Overview of QA Evaluation

15.5.1 Quality Assurance Organization

Ensure that the QAPD describes the structure, interrelationships, and areas of functional responsibility and authority for all organizational elements that will perform activities related to quality and safety. The following are examples of areas and items that may be addressed to support implementation of the quality criteria:

- measures to retain and exercise responsibility for the QA program; the assignment of responsibility for the overall QA program in no degree relieves line management of its responsibility for the achievement of quality

- measures to identify and describe the QA functions performed by the applicant's QA organization or delegated to other organizations that will provide controls to ensure implementation of the applicable elements of the QA criteria

- measures to provide clear management controls and effective lines of communication between the applicant's QA organizations and suppliers to ensure proper direction of the QA program and resolution of QA-related problems

- measures to identify onsite and offsite organizational elements that will function under the purview of the QA program and the lines of responsibility

- measures to designate a position that retains overall authority and responsibility for the QA program (e.g., manager or director of QA) and independently reports to at least the same organizational level authority as the highest line manager directly responsible for performing activities affecting quality

- measures to ensure that high-level management is responsible for documenting and promulgating the applicant's QA policies, goals, and objectives, and that this management level maintains a continuing involvement in QA matters; the application should also describe the lines of communication between intermediate levels of management and between high-level management and the manager (or director) of QA

- measures to provide authority and independence of the individual responsible for managing the QA program such that he or she can direct and control the organization's QA program, effectively ensure conformance to quality requirements, and remain sufficiently independent of undue influences and responsibilities of schedules and costs

- measures for individuals or groups responsible for defining and controlling the content of the QA program and related manuals to have appropriate organizational position and authority, as should the management level responsible for final review and approval

- measures describing the qualification requirements for the principal QA management positions so as to demonstrate management and technical competence commensurate with the responsibilities of these positions

- measures to ensure that conformance to established requirements will be verified by individuals or groups who do not have direct responsibility for performing the work being verified; the quality control function may be part of the line organization, provided the QA organization performs periodic surveillance to confirm sufficient independence from the individuals who performed the activities

- measures to ensure that persons and organizations performing QA functions have direct access to management levels that will ensure accomplishment of quality-affecting activities; these individuals should have sufficient authority and organizational freedom to perform their QA functions effectively and without reservation and should be able to identify quality problems; initiate, recommend, or provide solutions through designated channels; and verify implementation of solutions

- measures to ensure that designated QA individuals or organizations have the responsibility and authority, delineated in writing, to stop unsatisfactory work and control further processing, delivery, or installation of nonconforming material; the application should describe how stop-work requests will be initiated and completed

- measures to determine the extent of QA controls to be identified by the QA staff in combination with the line staff and to depend on the specific activity or item complexity and level of importance to safety

15.5.2 Quality Assurance Program

Ensure that the QAPD provides acceptable evidence that the applicant's proposed QA program will be well documented, planned, implemented, and maintained to provide the appropriate level of control over activities and SSCs consistent with their relative importance to safety. The following are examples of areas and items that may be addressed to support implementation of the quality criteria:

- measures used to ensure that the QA program meets applicable acceptance criteria

- measures for management to regularly assess the effectiveness of the QA program; measures for management (above and beyond the QA organization) to regularly assess the scope, status, adequacy, and compliance of the QA program to the requirements of 10 CFR Part 72; measures to provide for management's frequent appraisal of program status through reports, meetings, and audits as well as performance of a periodic assessment that is planned and documented with corrective actions identified and tracked

- measures to ensure that activities important to safety are accomplished using appropriate production and test equipment, suitable environmental conditions, applicable codes and standards, and proper work instructions

- measures used to ensure that trained, qualified personnel within the organization will be assigned to determine that functions delegated to contractors are properly accomplished

- summaries of the corporate QA policies, goals, and objectives and establishment of a meaningful channel for transmittal of these policies, goals, and objectives down through the levels of management

- measures to designate responsibilities for implementing the major activities addressed in the QA manuals

- measures to control the distribution of the QA manuals and revisions

- measures for communicating to all responsible organizations and individuals that policies, QA manuals, and procedures are mandatory requirements

- measures to provide a comprehensive listing of QA procedures, as well as a matrix of these procedures cross-referenced to each of the QA criteria, to demonstrate that the QA program will be fully implemented by documented procedures

- identification of SSCs, items, and attributes important to safety and how the QA program will control them

- measures for the applicant to review supplier documents for agreement with QA program provisions and ensure implementation of a program meeting the QA criteria

- measures for the resolution of disputes involving quality arising from a difference of opinion between QA personnel and personnel from other departments (e.g., engineering, procurement, manufacturing)

- measures for indoctrination, training, and qualification programs that fulfill the following criteria:

  – instruction of personnel responsible for performing activities affecting quality as to the purpose, scope, and implementation of the quality-related manuals, instructions, and procedures

  – training and qualification in the principles and techniques of the activities being performed for personnel performing activities affecting quality

  – maintenance of the proficiency of personnel performing quality-affecting activities by retraining, reexamining, and recertifying

  – preparation and maintenance of documentation of completed training and qualification

  – qualification of personnel in accordance with accepted codes and standards

15.5.3 Design Control

Ensure that the QAPD describes the approach the applicant will use to define, control, and verify the design and development of the DSS or DSF. The following are examples of areas and items that may be addressed to support implementation of the quality criteria:

- measures to carry out design activities in a planned, controlled, and orderly manner

- measures to correctly translate the applicable regulatory requirements and design bases into specifications, drawings, written procedures, and instructions

- measures to describe how the applicant will specify quality standards in the design documents and control deviations and changes from these quality standards

- measures to describe how the applicant will review designs to ensure that design characteristics can be controlled, inspected, and tested and that inspection and test criteria are identified

- measures to describe how the applicant will establish both internal and external design interface controls; these controls should include review, approval, release, distribution, and revision of documents involving design interfaces with participating design organizations

- measures to describe how the applicant will properly select and perform design verification processes such as design reviews, alternative calculations, or qualification testing; when a test program is to be used to verify the adequacy of a design, measures to describe how the applicant will use a qualification test of a prototype unit under adverse design conditions

- measures to ensure that design verifications (i.e., confirmation that the design of the SSC is suitable for its intended purpose) are completed by an individual with a level of skill at least equal to that of the original designer; measures to ensure design checking is also performed, recognizing design checking can be performed by a less experienced person (as an example, confirmation that the correct computer code has been used is part of design verification. Design checking includes confirmation of the numerical accuracy of computations and the accuracy of data input to computer codes); measures to describe how design verification will be performed by persons other than those performing design checking; measures to include how individuals or groups responsible for design verification will not include the original designer and normally not include the designer's immediate supervisor

- measures to ensure that design and specification changes are subject to the same design controls and the same or equivalent approvals that were applicable to the original design

- measures to ensure the documentation of all errors and deficiencies in the design or the design process that could adversely affect SSCs, items, and attributes important to safety; measures for adequate corrective action, including root cause evaluation of significant errors and deficiencies, to preclude repetition

- measures to review the suitability of any materials, parts, and equipment for the intended application before selecting such items that are standard, commercial (off-the-shelf), or have been previously approved for a different application

- measures to provide written procedures to identify and control the authority and responsibilities of all individuals or groups responsible for design reviews and other design verification activities

- measures that include the use of valid industry standards and specifications for the selection of suitable materials, parts, equipment, and processes for SSCs important to safety

15.5.4 Procurement Document Control

Ensure that documents used to procure SSCs or services include or reference applicable design bases and other requirements necessary to ensure adequate quality. The following are examples of areas and items that may be addressed to support implementation of the quality criteria:

- measures to establish procedures that clearly delineate the sequence of actions to be accomplished in the preparation, review, approval, and control of procurement documents

- measures to ensure that qualified personnel review and concur with the adequacy of quality requirements stated in procurement documents and ensure that the quality requirements are correctly stated, inspectable, and controllable; there are adequate acceptance and rejection criteria; and the procurement document has been prepared, reviewed, and approved in accordance with QA program requirements

- measures to document the review and approval of procurement documents before they are released, with the documentation available for verification

- measures to ensure that procurement documents identify the applicable QA requirements that should be compiled and described in the supplier's QA program and to ensure that the applicant reviews and concurs with the supplier's QA program; if subtier suppliers are also used, measures to ensure that the supplier's QA program applies to the subtier suppliers

- measures to ensure that procurement documents contain or reference the regulatory requirements, design bases, and other technical requirements

- measures to ensure that procurement documents identify the documentation (e.g., drawings, specifications, procedures, inspection and fabrication plans, inspection and test records, personnel and procedure qualifications, and chemical and physical test results of material) to be prepared, maintained, and submitted to the purchaser for review and approval

- measures to ensure that procurement documents identify records to be retained, controlled, and maintained by the supplier and those records to be delivered to the purchaser before use or installation of the hardware

- measures to ensure that procurement documents specify the procuring agency's right of access to the supplier's facilities and records for source inspection and audit

- measures to ensure that changes and revisions to procurement documents are subject to the same or equivalent review and approval as the original documents

15.5.5 Instructions, Procedures, and Drawings

Ensure that the QAPD defines the applicant's proposed procedures for ensuring that activities affecting quality will be prescribed by, and performed in accordance with, documented instructions, procedures, or drawings of a type appropriate for the circumstances. The following are examples of areas and items that may be addressed to support implementation of the quality criteria:

- measures to ensure that activities affecting quality are prescribed and accomplished in accordance with documented instructions, procedures, or drawings

- measures to establish provisions that clearly delineate the sequence of actions to be accomplished in the preparation, review, approval, and control of instructions, procedures, and drawings

- measures to ensure that instructions, procedures, and drawings specify the methods for complying with each of the applicable QA criteria

- measures to ensure that instructions, procedures, and drawings include quantitative acceptance criteria (such as dimensions, tolerances, and operating limits) as well as qualitative acceptance criteria (such as workmanship samples) as verification that activities important to safety have been satisfactorily accomplished

- measures to ensure that the QA organization reviews and concurs with the procedures, drawings, and specifications related to inspection plans, tests, calibrations, and special processes, as well as any subsequent changes to these documents

15.5.6 Document Control

Ensure that the QAPD defines the applicant's proposed procedures for preparing, issuing, and revising documents that specify quality requirements or prescribe activities affecting quality. The following are examples of areas and items that may be addressed to support implementation of the quality criteria:

- identification of all documents to be controlled under this subsection, including, as a minimum, design specifications; design and fabrication drawings; procurement documents; QA manuals; design criteria documents; fabrication, inspection, and testing instructions; and test procedures

- measures to ensure the establishment of procedures to control the review, approval, and issuance of documents, and any subsequent changes, before release to ensure that the documents are adequate and applicable quality requirements are stated

- measures to ensure the establishment of provisions to identify individuals or groups responsible for reviewing, approving, and issuing documents and subsequent revisions to the documents

- measures to ensure that document revisions receive review and approval by the same organizations that performed the original review and approval or by other qualified responsible organizations designated by the applicant

- measures to ensure that approved changes are included in instructions, procedures, drawings, and other documents before the change is implemented

- measures to ensure the control of obsolete or superseded documents to prevent inadvertent use

- measures to ensure that documents are available at the location where the activity is performed

- measures to ensure the establishment of a master list (or equivalent) to identify the current revision number of instructions, procedures, specifications, drawings, and procurement documents; measures to ensure the updating and distribution of the list to predetermined, responsible personnel to avoid the use of superseded documents

15.5.7 Control of Purchased Material, Equipment, and Services

Ensure that the QAPD defines the applicant's proposed procedures for controlling purchased material, equipment, and services to ensure conformance with specified requirements. The following are examples of areas and items that may be addressed to support implementation of the quality criteria:

- measures to ensure that qualified personnel evaluate the supplier's capability to provide services and products of acceptable quality before the award of the procurement order or contract; measures to ensure that QA and engineering groups participate in the evaluation of those suppliers providing critical items and services important to safety, including a definition of the responsibilities for each participating group

- measures to ensure the evaluation of suppliers should consider establishing the following provisions (if applicable)

  – the supplier's capability to comply with the elements of the QA criteria that are applicable to the type of material, equipment, or service being procured

  – review of previous records and performance of suppliers that have provided similar articles or services of the type being procured

  – a survey of the supplier's facilities and QA program to assess the capability to supply a product that meets applicable design, manufacturing, and quality requirements

- measures to ensure the documentation and filing of the results of supplier evaluations

- measures to ensure the planning and performance of adequate surveillance of suppliers during fabrication, inspection, testing, and shipment of materials, equipment, and components in accordance with written procedures to ensure conformance to the purchase order requirements; the measures should ensure that the procedures provide the following information:

  – instructions that specify the characteristics or processes to be witnessed, inspected or verified, and accepted; the method of surveillance and the extent of documentation required; and those responsible for implementing these instructions

  – procedures for audits and surveillance to ensure that the supplier complies with the quality requirements (surveillance should be performed for SSCs for which verification of procurement requirements cannot be determined upon receipt)

- measures to ensure that the supplier furnishes the following records to the purchaser:

  – documentation that identifies the purchased material or equipment and the specific procurement requirements (e.g., codes, standards, and specifications) met by the items

  – documentation that identifies any procurement requirements that have not been met and a description of any nonconformances designated "accept as is" or "repair"

- measures to describe the proposed procedures for reviewing and accepting these documents and, as a minimum, to ensure that this review and acceptance will be undertaken by a responsible QA individual

- measures to ensure the performance of periodic audits, independent inspections, or tests to ensure the validity of the suppliers' certificates of conformance

- measures to ensure the performance of a receiving inspection of supplier-furnished material, equipment, and services to ensure fulfillment of the following criteria:

  - proper identification of the material, component, or equipment in a manner that corresponds with the identification on the purchasing and receiving documentation

  - inspection of material, components, equipment, and acceptance records and judgment of their acceptability in accordance with predetermined inspection instructions before installation or use

  - availability of inspection records or certificates of conformance attesting to the acceptance of material, components, and equipment before installation or use

  - identification of the inspection status for accepted items and ensuring associated markings are attached before the accepted items are forwarded to a controlled storage area or released for installation or further work

- measures to assess the effectiveness of suppliers' quality controls at intervals consistent with the importance to safety, complexity, and quantity of the SSCs procured

15.5.8 Identification and Control of Materials, Parts, and Components

Ensure that the QAPD defines the applicant's proposed provisions for identifying and controlling materials, parts, and components to ensure that incorrect or defective SSCs are not used. The following are examples of areas and items that may be addressed to support implementation of the quality criteria:

- measures to establish procedures to identify and control materials, parts, and components (including partially fabricated subassemblies)

- measures to determine identification requirements during the generation of specifications and design drawings

- measures to ensure that identification will be maintained either on the item or on records traceable to the item to preclude the use of incorrect or defective items

- measures to ensure that the identification of materials and parts for items important to safety is traceable to the appropriate documentation (such as drawings, specifications, purchase orders, manufacturing and inspection documents, deviation reports, and physical and chemical mill test reports)

- measures to ensure that the location and method of identification do not affect the fit, function, or quality of the item being identified

- measures to verify and document the correct identification of all materials, parts, and components before releasing them for fabrication, assembly, shipping, and installation

15.5.9 Control of Special Processes

Ensure that the QAPD describes the controls the applicant will establish to ensure the acceptability of special processes (such as welding, heat treatment, nondestructive testing, and chemical cleaning) and that the proposed controls are performed by qualified personnel using qualified procedures and equipment. The following are examples of areas and items that may be addressed to support implementation of the quality criteria:

- measures to establish procedures to control special processes (such as welding, heat treating, nondestructive testing, and cleaning) for which direct inspection is generally impossible or disadvantageous, as well as a providing listing of these special processes

- measures to qualify procedures, equipment, and personnel connected with special processes in accordance with applicable codes, standards, and specifications

- measures to ensure that qualified personnel perform special processes in accordance with written process sheets (or the equivalent) with recorded evidence of verification

- measures to establish, file, and keep current qualification records of procedures, equipment, and personnel associated with special processes

15.5.10 Licensee and Certificate Holder Inspection

Ensure that the QAPD defines the applicant's proposed provisions for the inspection of activities affecting quality to verify conformance with instructions, procedures, and drawings. The following are examples of areas and items that may be addressed to support implementation of the quality criteria:

- measures to establish, document, and conduct an inspection program that effectively verifies the conformance of quality-affecting activities with requirements in accordance with written, controlled procedures

- measures to ensure that inspection personnel are sufficiently independent from the individuals performing the activities being inspected

- measures to ensure that inspection procedures, instructions, and checklists provide the following details:

  - identification of characteristics and activities to be inspected

  - identification of the individuals or groups responsible for performing the inspection operation

  - acceptance and rejection criteria

  - a description of the method of inspection

  - procedures for recording evidence of completing and verifying a manufacturing, inspection, or test operation

  - identification of the recording inspector or data recorder and the results of the inspection operation

- measures to ensure the use of inspection procedures or instructions with the necessary drawings and specifications when performing inspection operations

- measures to qualify inspectors in accordance with applicable codes, standards, and company training programs and to keep inspector qualifications and certifications current

- measures to inspect modifications, repairs, and replacements in accordance with the original design and inspection requirements or acceptable alternatives

- measures to establish provisions that identify mandatory inspection hold points for witnessing by a designated inspector

- measures to identify the individuals or groups who will perform receiving and process verification inspections, demonstrating that these individuals or groups have sufficient independence and qualifications

- measures to establish provisions for indirect control by monitoring processing methods, equipment, and personnel if direct inspection is not possible

15.5.11 Test Control

Ensure that the QAPD defines the applicant's proposed provisions for tests to verify that SSCs conform to specified requirements and will perform satisfactorily in service. The following are examples of areas and items that may be addressed to support implementation of the quality criteria:

- measures to establish, document, and conduct a test program to demonstrate that the item will perform satisfactorily in service in accordance with written, controlled procedures

- measures to ensure that written test procedures incorporate or reference the following information:

  - requirements and acceptance limits contained in applicable design and procurement documents

  - instructions for performing the test

  - test prerequisites

  - mandatory inspection hold points

  - acceptance and rejection criteria

  - methods of documenting or recording test data results

- measures to ensure a qualified, responsible individual or group documents test results and evaluates their acceptability; when practicable, the measures should ensure that testing of the SSC occurs under conditions that will be present during normal and anticipated off-normal operations

15.5.12 Control of Measuring and Test Equipment

Ensure that the QAPD defines the applicant's proposed provisions to ensure that tools, gauges, instruments, and other measuring and testing devices are properly identified, controlled, calibrated, and adjusted at specified intervals. The following are examples of areas and items that may be addressed to support implementation of the quality criteria:

- measures to ensure that documented procedures describe the calibration technique and frequency, maintenance, and control of all measuring and test equipment (instruments, tools, gauges, fixtures, reference and transfer standards, and nondestructive test equipment) that will be used in the measurement, inspection, and monitoring of SSCs important to safety

- measures to ensure that measuring and test equipment are identified and traceable to the calibration test data

- measures to ensure the use of labels, tags, or documents for measuring and test equipment to indicate the date of the next scheduled calibration and to provide traceability to calibration test data

- measures to calibrate measuring and test instruments at specified intervals on the basis of the required accuracy, precision, purpose, degree of usage, stability characteristics, and other conditions that could affect the accuracy of the measurements

- measures to assess the validity of previous inspections when measuring and test equipment is found to be out of calibration, and measures to document the assessment and to take control of the equipment that is out of calibration

- measures to document and maintain the complete status of all items under the calibration system

- measures to ensure that reference and transfer standards are traceable to nationally recognized standards, or to document the basis for calibration where national standards do not exist

15.5.13 Handling, Storage, and Shipping Control

Ensure that the QAPD defines the applicant's proposed provisions to control the handling, storage, shipping, cleaning, and preservation of SSCs in accordance with work and inspection instructions to prevent damage, loss, and deterioration. The following are examples of areas and items that may be addressed to support implementation of the quality criteria:

- measures to establish and accomplish special handling, preservation, storage, cleaning, packaging, and shipping requirements in accordance with predetermined work and inspection instructions

- measures to control the cleaning, handling, storage, packaging, shipping, and preservation of materials, components, and systems in accordance with design and specification requirements to preclude damage, loss, or deterioration by environmental conditions (such as temperature or humidity)

15.5.14 Inspection, Test, and Operating Status

Ensure that the QAPD defines the applicant's proposed provisions to control the inspection, test, and operating status of SSCs to prevent the inadvertent use of SSCs or bypassing of inspections and tests. The following are examples of areas and items that may be addressed to support implementation of the quality criteria:

- measures to know the inspection and test status of items throughout fabrication

- measures to establish procedures to control the application and removal of inspection and welding stamps and operating status indicators (such as tags, markings, labels, and stamps)

- measures to ensure that procedures under the cognizance of the QA organization control the bypassing of required inspections, tests, and other critical operations

- measures to specify the organization responsible for documenting the status of nonconforming, inoperative, or malfunctioning SSCs and for identifying the item to prevent inadvertent use

15.5.15 Nonconforming Materials, Parts, or Components

Ensure that the QAPD defines the applicant's proposed provisions to control the use or disposition of nonconforming materials, parts, or components. The following are examples of areas and items that may be addressed to support implementation of the quality criteria:

- measures to establish procedures to control the identification, documentation, tracking, segregation, review, disposition, and notification of affected organizations regarding nonconforming materials, parts, components, services, or activities

- measures to provide for adequate documentation to identify nonconforming items and describe the nonconformance, its disposition, and the related inspection requirements; such measures should also provide for adequate documentation and include signature approval of the disposition

- measures to establish provisions to identify those individuals or groups with the responsibility and authority for the disposition and closeout of nonconformance

- measures to ensure that nonconforming items are segregated from acceptable items and identified as discrepant until properly dispositioned and closed out

- measures to verify the acceptability of reworked or repaired materials, parts, and SSCs by reinspecting and retesting the item as originally inspected and tested or by using a method that is at least equal to the original inspection and testing method; the measures should provide for documentation of the relevant inspection, testing, rework, and repair procedures

- measures to ensure that nonconformance reports designated “accept as is” or “repair” are made part of the inspection records and forwarded with the hardware to the customer for review and assessment

- measures to periodically analyze nonconformance reports to show quality trends and help identify root causes of nonconformance. Significant results should be reported to responsible management for review and assessment

15.5.16 Corrective Action

Ensure that the QAPD defines the applicant’s proposed provisions to ensure that conditions adverse to quality are promptly identified and corrected, and that measures are taken to preclude recurrence. The following are examples of areas and items that may be addressed to support implementation of the quality criteria:

- measures to evaluate conditions adverse to quality (such as nonconformance, failures, malfunctions, deficiencies, deviations, and defective material and equipment) in accordance with established procedures to assess the need for corrective action

- measures to initiate corrective action to preclude the recurrence of a condition identified as adverse to quality

- measures to conduct follow-up activities to verify proper implementation of corrective actions and close out the corrective action documentation in a timely manner

- measures to document significant conditions adverse to quality, as well as the root causes of the conditions, and the corrective actions taken to remedy and preclude recurrence of the conditions; this information should be reported to cognizant levels of management for review and assessment

15.5.17 Quality Assurance Records

Ensure that the QAPD in the SAR defines the applicant’s proposed provisions for identifying, retaining, retrieving, and maintaining records that document evidence of the control of quality for activities and SSCs important to safety. The following are examples of areas and items that may be addressed to support implementation of the quality criteria:

- measures to define the scope of the records program such that sufficient records will be maintained to provide documentary evidence of the quality of items and activities affecting quality; to minimize the retention of unnecessary records, the records program should list records to be retained by type of data rather than by record title

- measures to ensure that QA records include operating logs; results of reviews, inspections, tests, audits, and material analyses; monitoring of work performance; qualification of personnel, procedures, and equipment; and other documentation such as drawings, specifications, procurement documents, calibration procedures and reports, design review and peer review reports, nonconformance reports, and corrective action reports

- measures to ensure that records are identified and retrievable

- Measures to ensure that requirements and responsibilities for record creation, transmittal, retention (such as duration, location, fire protection, and assigned responsibilities), and maintenance subsequent to completion of work are consistent with applicable codes, standards, and procurement documents

- measures to ensure that inspection and test records contain the following information, where applicable:

  – a description of the type of observation
  – the date and results of the inspection or test
  – information related to conditions adverse to quality
  – identification of the inspector or data recorder
  – evidence as to the acceptability of the results
  – action taken to resolve any noted discrepancies

- Measures to ensure that record storage facilities are constructed, located, and secured to prevent destruction of the records by fire, flood, theft, and deterioration by environmental conditions (such as temperature or humidity); measures to ensure that the facilities are maintained by, or under the control of, the licensee throughout the life of the DSS or DSF or the individual product

15.5.18 Audits

Ensure that the QAPD defines the applicant's proposed provisions for planning and scheduling audits to verify compliance with all aspects of the QA program and to determine the effectiveness of the overall program. The following are examples of areas and items that may be addressed to support implementation of the quality criteria:

- measures to perform audits in accordance with written procedures or checklists such that qualified personnel tasked with performing these audits do not have direct responsibility for the achievement of quality in the areas being audited

- measures to ensure that audit results are documented and reviewed by management with responsibility in the area audited

- measures to establish provisions for responsible management to undertake appropriate corrective action as a follow up to audit reports; the measures should ensure that auditing organizations schedule and conduct appropriate follow up to ensure that the corrective action is effectively accomplished

- measures to perform both technical and QA programmatic audits to achieve the following objectives:

  – comprehensive, independent verification and evaluation of procedures and activities affecting quality

  – verification and evaluation of the suppliers' QA programs, procedures, and activities

- measures to ensure that audits are led by appropriately qualified and certified audit personnel from the QA organization; measures to ensure that the audit team membership includes personnel (not necessarily QA organization personnel) with technical expertise in the areas being audited

- measures to schedule regular audits on the basis of the status and importance to safety of the activities being audited; measures to provide that audits are initiated early enough to ensure effective QA during design, procurement, and contracting activities

- measures to analyze and trend audit deficiency data as well as ensure that the resulting reports, indicating quality trends and the effectiveness of the QA program, are given to management for review, assessment, corrective action, and follow up

- measures to ensure that audits objectively assess the effectiveness and proper implementation of the QA program and address the technical adequacy of the activities being conducted

- measures to establish provisions requiring the performance of audits in all areas to which the requirements of the QA program apply

15.6 Evaluation Findings

The NRC reviewer should prepare evaluation findings upon satisfaction of the regulatory requirements in Section 15.4 of this SRP. If the reviewer determines that the applicant's QAPD does not adequately address the requirements in 10 CFR Part 72, a request for additional information must be prepared and submitted to the NRC project manager to be forwarded to the applicant for resolution and response to the NRC. If the reviewer concludes that information provided with the application, along with additional information provided in response to the NRC's request for additional information, shows that the QAPD meets the requirements, statements of finding similar to the following should be included in the staff's SER or in a letter to the applicant, if the applicant's QAPD was submitted separately from the SAR:

F15.1 The applicant's description of the QA program indicates that the requirements, procedures, and controls that, when properly implemented, should comply with the requirements of 10 CFR Part 72, Subpart G.

F15.2 The applicant's description of the QA program covers activities affecting SSCs, items, and attributes important to safety, as identified in the SAR.

F15.3 The applicant's description of the QA program covers activities affecting other SSCs, items, and attributes with consideration to their relative importance to safety, as identified in the SAR.

F15.4 The applicant's description of the QA program describes organizations and persons performing QA functions, indicating that sufficient independence and authority should exist to perform their functions without undue influence from those directly responsible for costs and schedules.

F15.5 The applicant's description of the QA program is in compliance with applicable NRC regulations and industry standards, and the acceptance of the QA program description by NRC allows implementation of the associated QA program for the [specify: design, fabrication and construction, operation, decommissioning] phase[s] of the installation's life cycle.

The reviewer should provide a summary statement similar to the following:

> The staff finds, with reasonable assurance, that the QA program for the [DSS/DSF] installation meets the requirements in 10 CFR Part 72 and addresses all 18 criteria as required in Subpart G to 10 CFR Part 72. The staff also finds, with reasonable assurance, that the QA program encompasses facility design controls, materials and services procurement controls, records and document controls, fabrication controls, nonconformance and corrective actions controls, an audit program, and operations or programs controls, as appropriate, adequate to ensure that the [DSS/DSF] installation will allow safe storage of spent nuclear fuel, high-level radioactive waste [applies to MRS only], and reactor-related greater than Class C waste. The staff reached this finding based on a review that considered applicable NRC regulations and regulatory guides and the statements and representations contained in the SAR.

15.7 References

10 CFR Part 72, "Licensing Requirements for the Independent Storage of Spent Nuclear Fuel, High-Level Radioactive Waste, and Reactor-Related Greater Than Class C Waste."

NUREG/CR-6314, "Quality Assurance Inspections for Shipping and Storage Containers," INEL95-0061, Idaho National Engineering Laboratory, April 1996.

16 ACCIDENT ANALYSIS EVALUATION

16.1 Review Objective

The objective of the U.S. Nuclear Regulatory Commission's (NRC's) accident analysis review is to conduct a systematic evaluation of the applicant's identification and analysis of hazards for both off-normal and accident conditions involving structures, systems, and components (SSCs) important to safety, and other SSCs that may affect SSCs important to safety. This chapter provides guidance for a minimum set of events the applicant should consider in its safety analysis report (SAR). Depending on the design of the dry storage system (DSS) or dry storage facility (DSF) or the DSF location, the applicant may need to consider additional events or additional DSS or DSF SSC configurations against which the events should be evaluated.

The accident analysis review ensures that the applicant has conducted thorough accident analyses as reflected by completing the following:

- identified all relevant off-normal conditions for the DSF
- identified all credible accidents for the DSF
- identified the envelop or bounding set of off-normal conditions and accident conditions that are relevant to the DSS design and operations and for which the DSS is analyzed to ensure performance of its design functions
- provided complete information in the SAR
- analyzed the safety performance of the DSF or DSS in each review area
- fulfilled all applicable regulatory requirements

16.2 Applicability

This chapter applies to the review of applications for specific licenses for an independent spent fuel storage installation (ISFSI) or a monitored retrievable storage installation (MRS), categorized as a DSF. It also applies to the review of applications for a certificate of compliance of a DSS for use at a general license facility. Sections or paragraphs of this chapter that apply only to specific license applications are identified with "**(SL)**."

16.3 Areas of Review

The accident analysis evaluation covers the applicant's identification and analysis of hazards, as well as the summary analysis of system responses. It places particular emphasis on the safety performance of the storage container under off-normal events and conditions and accident or design-basis events.

This chapter addresses the following areas of review that may encompass a comprehensive accident analysis evaluation:

- cause of the event
- definition of operating environment and physical parameters
- detection of the event
- summary of event consequences and regulatory compliance
- corrective course of action

The review for each off-normal and each accident condition, as presented in the SAR, should address each of these five areas.

16.4 Regulatory Requirements and Acceptance Criteria

This section summarizes those parts of Title 10 of the *Code of Federal Regulations* (10 CFR) Part 72, “Licensing Requirements for the Independent Storage of Spent Nuclear Fuel, High-Level Radioactive Waste, and Reactor-Related Greater than Class C Waste,” that are relevant to the review areas this chapter addresses. The reviewer should refer to the exact language in the regulations. Tables 16-1a and 16-1b match the relevant regulatory requirements to the areas of review this chapter covers. Note that regulatory requirements in 10 CFR Part 20, “Standards for Protection Against Radiation,” (see SRP Chapters 10A and 10B, “Radiation Protection Evaluation,” for a DSF and a DSS, respectively) also apply for off-normal events, and the reviewer must consider those regulations in evaluations of these events.

Accidents and natural phenomena events may share common regulatory and design limits. Consequently, this chapter sometimes refer to these scenarios collectively as accident conditions.

By contrast, off-normal conditions (anticipated occurrences) are distinguished, in part, from accidents or natural phenomena by the appropriate regulatory guidance and design criteria. For example, the radiation dose from an off-normal event, in combination with doses from normal operations, must not exceed the limits specified in 10 CFR Part 20 and 10 CFR 72.104(a), whereas the radiation dose from an accident or natural phenomenon must not exceed the limits specified in 10 CFR 72.106(b). Accident conditions may also have different allowable structural and thermal criteria compared to off-normal conditions.

Table 16-1a Relationship of Regulations and Areas of Review for a DSF (SL)

Areas of Review	10 CFR Part 72 Regulations					
	72.24(a)(d)(1)(2), (m)	72.90 (c)	72.92	72.94	72.100 (a)	72.104 (a)(c)
Cause of the Event	●	●	●	●		
Detection of the Event						
Summary of Event Consequences and Regulatory Compliance	●			●	●	●
Corrective Course of Action	●				●	●

Areas of Review	10 CFR Part 72 Regulations (cont.)				
	72.106 (b)	72.122(b)(c)(d)(e)(g)(h)(i)(j)(k)(1)(3)(4), (l)	72.124	72.126 (b)(c)(d)	72.128 (a)(1)(2)(3)(4)
Cause of the Event		●			
Detection of the Event		●	●	●	●
Summary of Event Consequences and Regulatory Compliance	●	●	●	●	●
Corrective Course of Action	●	●			●

Table 16-1b Relationship of Regulations and Areas of Review for a DSS (CoC)

Areas of Review	10 CFR Part 72 Regulations					
	72.104 (a)(c)	72.106 (b)	72.122(b)(c)(d)(g)(h)(i)(j)(k)(4), (l)[A]	72.124	72.128 (a)(1)(2)(3)(4)[a]	72.236 (c)(d)(l)
Cause of the Event			●			
Detection of the Event			●	●	●	
Summary of Event Consequences and Regulatory Compliance	●	●	●	●	●	●
Corrective Course of Action	●	●	●		●	

A Note that while 10 CFR 72.122, "Overall Requirements," and 10 CFR 72.128, "Criteria for Spent Fuel, High-Level Radioactive Waste, and Other Radioactive Waste Storage and Handling," are not applicable to an application for a CoC, the CoC applicant should describe how the DSS design facilitates the ability of the licensee to meet these requirements.

In general, the accident analysis evaluation seeks to ensure that the design and the applicant's hazard identification and analyses of related DSS or DSF responses fulfill the relevant design and regulatory criteria (including the criteria discussed in Sections 16.4.1–6 below) for the following types of events or conditions. The hazard identification and analyses should include appropriate consideration of the different operation configurations that may occur, or are likely to occur, for the DSS or DSF design, including temporary configurations. Such configurations may include construction activities to expand an operating array of storage containers that removes or exposes shielding materials.

- Off-Normal Events and Conditions—The following is a minimum list of the off-normal events that the applicant must consider in the SAR:

  - partial vent blockage (if applicable)
  - operational events resulting in radioactive release
  - off-normal ambient temperatures
  - off-normal events associated with the facility **(SL)**

- Accident Events and Conditions—The following is a minimum list of the accident conditions that the applicant must consider in the SAR:

  - storage container tipover
  - storage container drop
  - flood
  - fire and explosion
  - lightning
  - earthquake
  - loss of shielding
  - adiabatic heatup
  - tornadoes and missiles generated by natural phenomena
  - accidents at nearby sites **(SL)**
  - building structural failure onto SSCs **(SL)**

- Other Off-Normal and Accident Events and Conditions—In addition to all of the accidents and off-normal events listed above, the applicant must list and evaluate other off-normal and accident events that are specific to the applicant's design. These events include those that might have negligible consequences for most designs, but characteristics of the proposed design may result in nonnegligible consequences for the same events (e.g., crane malfunction). If these other off-normal and accident events have results that are enveloped by the events previously considered, the applicant must provide the basis for this evaluation, and no further consideration is required. It is expected that the required off-normal events and accidents listed in this section may envelope events such as human errors, operational errors, and material aging.

16.4.1 Dose Limits for Off-Normal Events

During normal operations and off-normal conditions (that is, anticipated occurrences), the DSF must meet the annual dose limits in 10 CFR 72.104(a). The DSF applicant must also demonstrate that the DSF will meet the requirements specified in 10 CFR Part 20.

16.4.2 Dose Limit for Accidents

The dose from any accident to any individual located on or beyond the nearest boundary of the controlled area may not exceed the limits specified in 10 CFR 72.106(b).

16.4.3 Criticality

In accordance with 10 CFR 72.124(a) and, for DSSs, 10 CFR 72.236(c), the licensee must maintain the SNF in a subcritical condition under credible conditions (i.e., effective neutron multiplication factor (k_{eff}), including all biases and uncertainties, equal to or less than 0.95). DSS or DSF SSCs must be designed so that at least two unlikely, independent, and concurrent or sequential changes in the conditions essential to nuclear criticality safety must occur before a nuclear criticality accident is possible (double contingency). Similar criteria should be applied, as appropriate, to other radioactive materials to be stored at a DSF (e.g., HLW at a MRS).

16.4.4 Confinement

The regulation in 10 CFR 72.128(a) states that DSF systems must be designed with confinement structures and systems and 10 CFR 72.236(d) requires DSS' confinement (and shielding) systems be sufficient to meet the requirements in §§ 72.104 and 72.106. The applicant must evaluate the DSS or DSF SSCs and features important to safety using appropriate tests or by other means acceptable to the NRC to demonstrate that the SSCs will reasonably maintain confinement of radioactive material under accident conditions, consistent with 10 CFR 72.122(b), 10 CFR 72.122(c), and 10 CFR 72.122(h) for DSFs and as specified in 10 CFR 72.236(l) for DSSs. The applicant should show that a breach of a confinement barrier does not occur as a result of any off-normal or accident event. A confinement system is defined in 10 CFR 72.3, "Definitions," as a system, including ventilation, which acts as a barrier between areas containing radioactive substances and the environment.

16.4.5 Recovery and Retrievability

Recovery is the capability of returning the stored radioactive materials from an accident to a safe condition without endangering public health and safety or causing significant or unnecessary exposure to workers. Any potential release of radioactive materials during recovery operations must not exceed the radioactive exposure limits in 10 CFR Part 20.

Retrievability is applicable only during normal and off-normal conditions and does not apply to accident conditions. Retrievability is specified in 10 CFR 72.122(l), which states that "storage systems must be designed to allow ready retrieval of spent fuel, high level radioactive waste, and reactor-related greater than class C waste for further processing or disposal." Ready retrieval is defined as the ability to safely remove the SNF, HLW, or reactor-related GTCC from storage for further processing or disposal. A storage system must be designed to allow for ready retrieval in the initial design, amendments to the design, and in licenses and CoCs, as applicable, through the licensing period(s) of the design, including through renewals. The retrievability requirement applies to all ISFSIs operating under a general license or a specific license. The requirements in 10 CFR 72.236(m) for CoC holders states that "[t]o the extent practicable in the design of spent fuel storage casks, consideration should be given to compatibility with removal of the stored spent fuel from a reactor site, transportation, and ultimate disposition by the Department of Energy." Any potential release of radioactive materials during retrieval operations must not exceed the radioactive exposure limits in 10 CFR Part 20.

In order to demonstrate the ability for ready retrieval of SNF, a licensee should demonstrate it has the ability to perform any of the three options shown below. Note that these options may be utilized individually or in any combination or sequence, as appropriate:

- Remove individual or canned spent fuel assemblies from wet or dry storage.
- Remove a canister loaded with spent fuel assemblies from a storage cask or overpack, as applicable.
- Remove a cask or DSF storage container, as applicable, loaded with spent fuel assemblies from the storage location.

16.4.6 Instrumentation

For DSFs, the SAR must identify all instruments and control systems that must remain operational under normal, off-normal and accident conditions as required by 10 CFR 72.122(i). For DSSs, the SAR should show how the DSS design facilitates the general licensee's ability to meet 10 CFR 72.122(i).

16.5 Review Procedures

This section provides review guidance for each off-normal and accident event evaluation. The review guidance varies in complexity for each evaluation. In general, the staff's review includes the operating environment, the physical parameters, the methodology used, and the actual analysis the applicant performed as part of its review.

Items of unique or special safety significance should receive special emphasis. Refer to Chapter 3, "Principal Design Criteria Evaluation," of this SRP for a discussion of the SSCs important to safety.

The effects of various off-normal events and accidents may be interrelated, and some degree of overlap is expected to occur during the accident analyses review process. An example of such overlap would be a tornado missile accident leading to a loss of shielding as described in Section 16.5.2.7 of this chapter, or an accident reviewed according to Section 16.5.2.9. If two or more off-normal events and accidents are interrelated, assess the combined occurrence and effects of the interrelated off-normal and accident conditions.

Ensure that the applicant identifies and evaluates all relevant, credible off-normal and accident conditions, including any that are unique to the design and, for DFSs, the site. Ensure that the applicant's evaluations include the occurrence and effects of these events for all relevant and likely, or credible, operating configurations, including temporary normal conditions. Ensure that the applicant's evaluations address all relevant criteria. Coordinate the review with the other SAR reviewers to evaluate the design and site characteristics to determine whether all relevant off-normal and accident conditions have been identified and evaluated. The detailed evaluations of the conditions may be done in the accident chapter of the SAR or in the respective SAR chapters for each technical discipline, with the accident chapter merely summarizing and referencing the evaluations in those other chapters. In either case, coordinate with the reviewers of those chapters to ensure the evaluations are adequate and to verify that the design and regulatory criteria are met. Also identify those parameters that may need to be included in the technical specifications based on the accident evaluation analyses. For example, DSS applications are not for a particular site and so must make assumptions in its analyses regarding conditions, such as natural phenomena, that may occur at sites that may use the DSS. Some of the assumptions may need to be translated into one or more appropriate conditions in the technical specifications for the DSS.

Figure 16-1 shows the interrelationship between the accident analysis evaluation and the other areas of review described in this SRP.

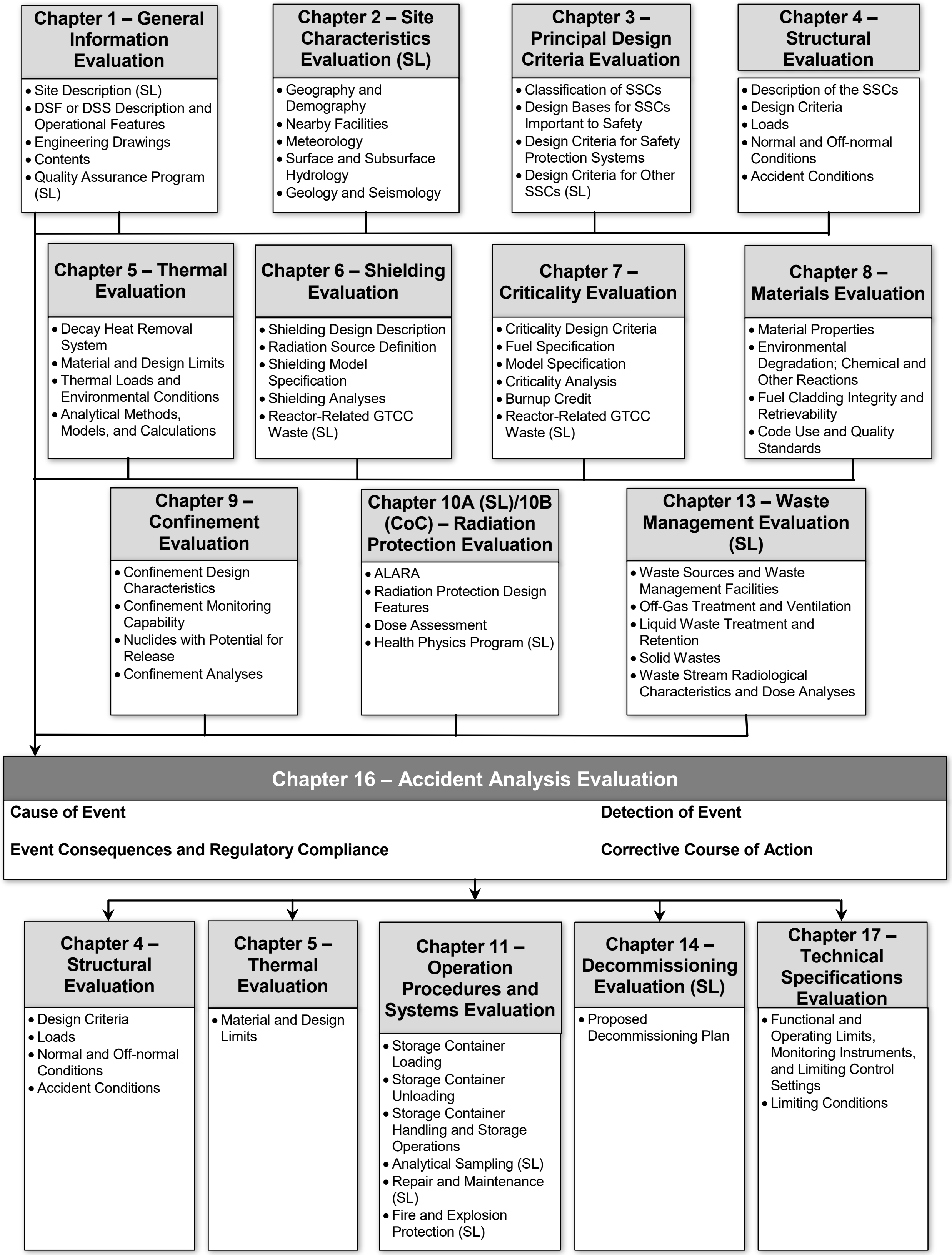

Figure 16-1 Overview of Accident Analysis Evaluation

For each off-normal and accident event described in these review procedures, verify that the applicant has addressed the following areas of review: (1) the cause of the event, (2) means of detecting the event, (3) a summary of the analysis of the event, including estimated consequences and comparison to regulatory limits, and (4) a corrective course of action.

- Cause of the Event—The applicant should describe the cause of the off-normal or accident condition. The description should include the chain of events that leads to the credible off-normal or accident condition and any bounding conditions.

- Definition of Operating Environment and Physical Parameters—The applicant should describe the conditions and environment that the DSF or DSS SSCs experience for off-normal and accident conditions. This includes parameters and information items such as the configuration and physical location (as applicable) of the DSS or DSF SSCs, ambient conditions, extent of degradation (e.g., fraction of vent blockage), surface contamination levels, properties of impact objects or surfaces, and sources of hazards (e.g., flood water source).

- Detection of the Event—The applicant may detect an event through surveillance programs or monitoring instrumentation and alarms. Surveillance programs and monitoring instrumentation and alarms should have reasonable flexibility to allow for the identification of an accident condition or noncompliance situation that has not been previously considered in the SAR. The method of detection will be intuitively obvious for some events, whereas other events (e.g., fuel rod rupture) may remain undetected for a significant period of time.

- Summary of Event Consequences and Regulatory Compliance—The applicant should address event consequences in each functional area corresponding to earlier chapters of the SAR (i.e., structural, thermal, shielding, criticality, confinement, materials, and radiation protection). This area of review includes evaluation of (1) the analysis method and (2) the event analysis. The SAR should describe the analysis method(s) the applicant used, including the tools and techniques. The SAR should present the analysis of the event, including the design criteria and design codes and standards, as applicable. This discussion should refer back to each SAR chapter in which the individual consequences are evaluated in detail. The applicant should provide a summary of the accident dose calculations and show that the consequences comply with the applicable regulatory criteria. For off-normal conditions, the applicant should include the resulting doses with the doses from normal operations in the evaluations for demonstrating the facility design and operations meet, or will meet, the requirements, including dose limits, demonstrate compliance within 10 CFR Part 20 as well as 10 CFR Part 72. As applicable and appropriate, the consequence analyses should address occupational doses as well as doses to members of the public.

- Corrective Course of Action—The applicant should identify what action(s), if any, would be necessary to recover from the event. If various courses of action are possible, the applicant should present a discussion concerning the selection of the most appropriate action. Because the SNF, HLW, or reactor-related GTCC, as applicable, must be readily retrievable after an off-normal event and after returning to storage after an accident, reloading the SNF, HLW, or reactor-related GTCC, as applicable, into a new storage container is a viable option. If corrective courses of action are to be included in operating procedures or administrative programs, then the applicable sections of the SAR that cover operating procedures and administrative programs should be referenced.

16.5.1 Off-Normal Events

This section discusses the review of off-normal conditions that may include malfunctions of systems, minor leakage, limited loss of external power, and operator error. The consequences of these events should not have a significant effect beyond the facility operation areas (e.g., handling, loading, storage areas).

Verify that the SAR also defines the analysis and design criteria and design codes and standards (as applicable) for each off-normal event as related to HLW or reactor-related GTCC waste storage and handling systems.

16.5.1.1 *Partial Vent Blockage (if applicable)*

For confinement systems, such as natural convection cooling systems that are subject to a temperature rise from a partial vent blockage, verify that the applicant has made an evaluation of the event. The purpose of the evaluation is not to establish a surveillance frequency, as in the case of the adiabatic heatup accident, but rather to establish that no critical temperature limits will be reached for an extended time period.

16.5.1.1.1 Define the Operating Environment and Physical Parameters

Verify that the SAR identifies the operating environment of the off-normal event, including the following:

- the operational configuration of the confinement system
- the fraction of vent blockage
- the ambient temperature
- the design-basis decay heat load

16.5.1.1.2 Review the Analysis Methodology

Verify that the SAR defines the analysis methodology used in the evaluation, including assumptions and calculational models or experimental testing.

16.5.1.1.3 Off-Normal Event Analysis

Verify the identification of the vent flow area and revised vent flow loss coefficients associated with any blockage of the normal air inlet vent flow area.

Verify the air outlet temperature and the unit internal material maximum temperatures for all key DSS or DSF SSCs. Use the flow areas and flow loss coefficients assuming normal ambient air temperature (as defined in Chapter 3 of this SRP). Also use the maximum design-basis decay heat and the identical thermal models and computer codes that were used in the normal conditions thermal analysis of the DSS or DSF SSCs.

Compare the calculated maximum material temperatures with their respective off-normal temperature limits, and verify that no critical temperature limits will be reached for the time period.

Coordinate with the structural integrity reviewer to ensure that these temperatures are used to determine the appropriate allowable stress-intensity levels.

Verify that the applicant evaluated the worker dose required to clear debris that is blocking air inlet(s) using the design-basis calculated dose rate at the air inlets and an appropriate estimate of the time necessary to clear the vents. Ensure that the doses are below the worker dose limits in 10 CFR 20.1201, “Occupational Dose Limits for Adults.”

16.5.1.2 *Operational Events Resulting in Radioactive Release*

This subsection shows the process for evaluating a typical off-normal condition and any resulting radiological consequences.

16.5.1.2.1 Define the Operating Environment and Physical Parameters

Verify that the SAR describes the maximum allowable container surface contamination, based on applicable technical specifications or health physics procedures, or both. This contamination is usually expressed in terms of counts per minute, counts per unit surface area, or microcuries per square centimeter, and different values are provided for alpha contamination and beta or gamma contamination.

Verify the calculation of the total surface area of the SNF, HLW, or reactor-related GTCC waste container.

Verify the calculation of the total container surface contamination by multiplying the values of surface contamination (in terms of curies of activity per unit surface area) and surface area.

16.5.1.2.2 Review the Analysis Methodology

Verify that the SAR contains the 95-percent probability value for the atmospheric dispersion factor from the SNF storage facility container to members of the public at or beyond the controlled area boundary. The technical basis and applicability of the atmospheric dispersion value should be included. Regulatory Guide (RG) 1.145, “Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants,” provides detailed directions on acceptable methods for calculations of values of dispersion parameters. The NRC has previously accepted RG 1.4, “Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident for Pressurized Water Reactors,” or RG 1.25, “Assumptions Used for Evaluating the Potential Radiological Consequences of a Fuel Handling Accident in the Fuel Handling and Storage Facility for Boiling and Pressurized Water Reactors,” for conservative generic values of atmospheric dispersion factors in the absence of site-specific meteorological data.

Use the guidance in Chapters 6, “Shielding Evaluation,” 9, “Confinement Evaluation,” and 10A and 10B to evaluate the applicant’s analyses of event consequences for releases. Verify that the applicant uses an appropriate method for evaluating radiological consequences to operations personnel and members of the public on the site.

16.5.1.2.3 Off-Normal Event Analysis

Use the guidance in Chapters 6, 9, 10A, and 10B to evaluate the applicant's analyses of event consequences for releases. Verify that the applicant has determined the dose consequences to individuals on site for purposes of demonstrating compliance with 10 CFR 20.1101(d), 10 CFR 20.1201(a), and 10 CFR 20.1301(b).

16.5.1.3 *Off-Normal Ambient Temperatures*

Off-normal ambient temperatures are expected to occur during the operational life of the DSS or DSF. The numerical values of off-normal ambient temperatures are expected to be greater than the normal ambient temperature but less than the accident ambient temperature. The higher probability of occurrence of off-normal ambient temperatures, compared to the accident temperatures, requires that calculated material temperatures as a result of off-normal ambient temperatures meet the normal operational material temperature limits.

16.5.1.3.1 Define the Operating Environment and Physical Parameters

Verify that the SAR specifies appropriate maximum and minimum off-normal ambient temperatures. Examples of previously accepted conditions include maximum and minimum ambient temperature values of 52 degrees Celsius (°C) (125 degrees Fahrenheit (°F)) and -40 °C (-40 °F). For previously licensed or certified DSF or DSS, a typical annual average ambient temperature has been 24 °C (75 °F). The maximum and minimum ambient temperature values should equal the 99-percent values in Table 1, "Climatic Conditions for the United States," in the American Society of Heating, Refrigeration and Air-Conditioning Engineers' publication, "ASHRAE Handbook—Fundamentals." If the DSF or DSS does not correspond with a location cited in this reference, verify that the applicant has supplied technical justification for using the same climatic data as shown in the ASHRAE Handbook.

Similarly, verify the site-specific or generic value of solar insolation or heat flux for the DSF or DSS. This value should be used in conjunction with the normal and off-normal maximum ambient temperature, but a value of zero solar heat flux should be used with the minimum ambient air temperature scenario.

16.5.1.3.2 Review the Analysis Methodology and Off-Normal Event Analysis

Verify that the applicant calculated the steady-state temperature distribution within the DSS or DSF SSCs using the same methodology and computer codes that were used for the normal ambient air temperature scenario.

Evaluate the calculated temperature distribution in terms of material temperature limits (e.g., fuel cladding, concrete, and proprietary neutron shielding materials) and thermal stresses. The material temperature limits should be consistent with the acceptable limits identified in the thermal analysis evaluation.

16.5.1.4 *Other Off-Normal Events Associated with the Facility*

16.5.1.4.1 Define the Off-Normal Events

The following off-normal events are estimated to occur with a frequency of approximately once per year of storage operation and should be evaluated regardless of which American National

Standards Institute (ANSI) standard the SAR cites. The list is intended to be representative and not all inclusive.

- failure of any single active component to perform its intended function on demand
- spurious operation of certain active components such as a relief valve or a control valve
- loss of external power supply for a limited duration (e.g., less than 8 hours) that could cause loss of cooling
- single-operator error followed by proper corrective action
- minor leakage from component

If the SAR cites ANSI/American Nuclear Society (ANS) 57.2, "Design Requirements for Light Water Reactor Spent Fuel Storage Facilities at Nuclear Power Plants," then the applicant should consider a single failure in the electrical or control system in addition to the above events.

16.5.1.4.2 Define the Operating Environment

Verify that the SAR identifies the operating environment and conditions of the off-normal events.

16.5.1.4.3 Define the Physical Parameters

Verify that the SAR defines the physical parameters associated with the off-normal events, possibly including the following:

- level or temperature of water at the time of failure or spurious operation of active components
- any protective devices designed to mitigate the consequences of the off-normal events
- alarms and response times for corrective action

16.5.1.4.4 Review the Analysis Methodology

Verify that the SAR defines the analysis methodology for evaluating the consequences of the off-normal events, including assumptions used as a part of the off-normal event.

16.5.1.4.5 Off-Normal Event Analysis

Verify that the SAR presents the analysis, design criteria, and design codes and standards for each of the off-normal events that the SAR defines. The following codes and standards are the primary design and construction codes acceptable to the NRC; consult ANSI/ANS 57.2 or ANSI/ANS 57.7, "Design Criteria for an Independent Spent Fuel Storage Installation (Water Pool Type)," for a more detailed listing of design codes and standards.

- SNF storage racks
    - American Society of Mechanical Engineers (ASME) Boiler and Pressure (B&PV) Code, Section III, "Rules for Construction of Nuclear Facility Components."

- SNF storage container and HLW or reactor-related GTCC handling systems

  - Crane Manufacturers Association of America, Specification No. 70, "Specifications for Top Running Bridge and Gantry Type Multiple Girder Electric Overhead Traveling Cranes"

  - ANSI N14.6, "Radioactive Materials—Special Lifting Devices for Shipping Containers Weighing 10,000 Pounds (4500 kg) or More," for special lifting devices for shipping containers weighing more than 10,000 pounds

  - ASME B30.2, "Overhead and Gantry Cranes (Top Running Bridge, Single or Multiple Girder, Top Running Trolley Hoist)," for overhead and gantry cranes for ANSI/ANS 57.2 designs

- SNF or waste form handling systems

  - Institute of Electrical and Electronics Engineers (IEEE) C2, "National Electrical Safety Code"

  - IEEE 835, "Standard Power Cable Ampacity Tables."

  - National Fire Protection Association (NFPA) 70, "National Electrical Code"

  - ASME B30.16, "Overhead Hoists (Underhung)"

- heating, ventilation, and air-conditioning systems

  - ASHRAE Handbook
  - Air Movement and Control Association standards and application guides
  - ASME N509, "Nuclear Power Plant Air-Cleaning Units and Components"
  - International Code Council, "International Building Code"
  - International Code Council, "International Mechanical Code"

- buildings

  - ANSI/American Concrete Institute (ACI) 349, "Code Requirements for Nuclear Safety-Related Concrete Structures and Commentary," for reinforced concrete for ANSI/ANS 57.2 designs and ANSI/ACI 318, "Building Code Requirements for Structural Concrete and Commentary," for ANSI/ANS 57.7 designs, and as appropriate for ANSI/ANS 57.9 designs

  - NFPA 780, "Standard for the Installation of Lightning Protection Systems"

  - American Iron and Steel Institute, "Steel Products Manual"

- radioactive waste treatment

  - 10 CFR Part 71, "Packaging and Transportation of Radioactive Material"
  - 10 CFR Part 20 for radiation protection

Verify that the applicant has identified any radiological consequences related to these other off-normal events associated with the facility and has calculated dose rates and doses.

16.5.2 Accidents

Verify that the SAR includes a rigorous discussion of potential accidents, both external natural events and man-induced events. The accident analysis review focuses on the effects of the natural phenomena and man-induced events on SSCs important to safety. Ensure that the SAR presents analytical techniques, uncertainties, and assumptions.

For those SNF storage facility license applications that propose to use a certified DSS listed in 10 CFR 72.214, "List of Approved Spent Fuel Storage Casks," the SAR may reference rather than repeat the evaluation of the impacts of accidents to the DSS that have been previously evaluated as part of that DSS's certification. However, verify that the SAR shows that the analyses for the DSS (or the conditions used for those analyses) bound the relevant conditions of the proposed facility. Also, ensure that the SAR addresses any effects to the facility and facility equipment from the event (e.g., pool lining damage).

For each accident condition, the SAR discussion of each event should include (a) a discussion of the cause of the event, (b) the means of detection of the event, (c) an analysis of the consequences (particularly any radiological consequences) and the protection provided by devices or systems designed to limit the extent of the consequences, and (d) any actions required of the operator. For each accident the applicant should provide and discuss the results of dose calculations.

16.5.2.1 *Storage Container Tipover*

Confirm that the SAR evaluates the container tipover accident. For this analysis, the NRC will accept container tipover about a lower corner onto a receiving surface from a position of balance with no initial velocity. Other analyses of tipover accidents may also be accepted; for example, the NRC has also accepted analysis of container drops with the longitudinal axis horizontal that, together with the longitudinal axis vertical, could bound a nonmechanistic tipover analysis.

16.5.2.1.1 Define the Operating Environment

Verify that the SAR identifies the operating environment of the accident, including the following:

- the operational configuration of the storage container (e.g., a storage container on the pad, a canister inside a transfer cask suspended from a cable on a crane or hoist, with or without impact limiters)

- the physical location of the tipover accident

16.5.2.1.2 Define the Physical Parameters

Verify that the SAR defines the physical parameters necessary to evaluate the accident, including the following:

- the receiving surface upon which the storage container slaps down (e., the storage pad materials, dimensions, and properties and the foundation properties; the surface must be defined to quantify the maximum deceleration levels)

- the design of the storage container and associated SSCs (i.e., material properties, dimensions, and weights)

16.5.2.1.3 Review the Analysis Methodology

Verify that the SAR defines the analysis methodology used in the evaluation, such as the following:

- reference to horizontal and vertical analyses if the tipover can be shown to be bounded by these two accidents

- specific analysis modeling tools such as closed-form manual techniques or computer codes

16.5.2.1.4 Accident Analysis

Verify that the SAR presents the accident analysis, design criteria, and design codes and standards, such as the following:

- deceleration level

- design code for evaluation—the NRC accepts the ASME B&PV Code, Section III, Service Level D

- specification if elastic or elastic-plastic analysis is used and appropriate citation of design code

- evaluation of calculated stress-intensity level against the allowable stress-intensity level at the design temperature and pressure for each component in the storage container; the evaluation should also consider components associated with confinement O-rings or seals and relevant pressure-monitoring systems for bolted lids

- evaluation of buckling stability for each component member of the storage container subject to compressive loading

- evaluation of deformation of container internal members that contribute to the spacing geometry for criticality safety

- evaluation of deformation of, or damage to, the SNF or HLW (MRS only) contents of the storage container

- evaluation of damage or deformation of the reactor-related GTCC waste storage container **(SL)**

- evaluation of impacts to other facility systems or features **(SL)**

- calculation of dose consequences

16.5.2.2 *Storage Container Drop*

The drop of the storage container is one of the hypothetical accident scenarios that the applicant must evaluate. The following steps provide an outline of the methodology that the applicant should provide in the SAR. The steps are representative of a typical SNF storage container but are not intended to cover every aspect of every possible container design.

16.5.2.2.1 Define the Operating Environment

Verify that the SAR identifies the operating environment of the accident, including the following:

- the operational configuration of the storage container (e.g., a storage container with no other SSCs, a canister inside the transfer cask or transportation package, with or without impact limiters)

- the storage container orientation at the moment of impact (i.e., end drop on top or bottom, side drop at various azimuths, and corner drop at various azimuths and inclinations)

- the physical location of the drop accident (i.e., outside the SNF pool building or inside the SNF pool building or other DSF structures or buildings where the materials stored at the site or the storage containers are handled)

16.5.2.2.2 Define the Physical Parameters

Verify that the SAR defines the physical parameters associated with the accident, including the following:

- the receiving surface upon which the storage container impacts (i.e., the storage pad materials, dimensions, and properties and the foundation properties, or dimensions and properties of the SNF pool or building floor materials or floor materials of other DSF buildings where operations occur); the surface should be sufficiently characterized to quantify the maximum deceleration levels

- the design of the storage container and associated SSCs (i.e., material properties, dimensions, and weights)

- the drop height of the storage container onto the receiving surface for each orientation; the analysis should use the maximum height above the impact surface to which the container could be lifted

16.5.2.2.3 Review the Analysis Methodology

Verify that the SAR defines the analysis methodology used in the evaluation, such as the following:

- static equivalent deceleration with appropriate dynamic load factors
- dynamic modeling with appropriate test data to benchmark deceleration
- specific analysis modeling tools such as manual techniques or computer codes

16.5.2.2.4 Accident Analysis

Verify that the SAR presents the accident analysis, design criteria, and design codes and standards, such as the following:

- deceleration level for each case considered
- design code for evaluation—the NRC accepts the ASME B&PV Code, Section III, Service Level D
- specification if elastic or elastic-plastic analysis is used and appropriate citation of the design code
- evaluation of calculated stress-intensity level against the allowable stress-intensity level at the design temperature and pressure for each component member of the storage container; the evaluation should also consider components associated with confinement O-rings or seals and relevant pressure-monitoring systems for bolted lids
- evaluation of the buckling stability for each component member of the storage container subjected to compressive loading
- evaluation of the deformation of container internal members that contribute to spacing geometry of the SNF assemblies or HLW materials that are subject to criticality safety as given in Chapter 7, “Criticality Evaluation,” of this SRP
- evaluation of deformation of, or damage to, the SNF or HLW (MRS only) contents of the storage container
- evaluation of damage or deformation of the reactor-related GTCC waste storage container **(SL)**
- evaluation of impacts to other facility systems or features **(SL)**
- calculation of accidental dose consequences

16.5.2.3 *Flood*

The flood accident is one of the accidents that the applicant must evaluate, in accordance with 10 CFR 72.122(b)(2)(i). Coordinate the review of the flood evaluation in the SAR with that of the site characteristics for DSF specific license applications. For DSS applications, ensure that the SAR defines a set of flood parameters, the effects of which the DSS must withstand, and the basis for the selection of those parameters, including the evaluation of any entrained debris. The following steps provide an outline of the methodology that the applicant should provide in the SAR.

16.5.2.3.1 Define the Operating Environment

Verify that the SAR identifies the operating environment of the accident, including the following:

- the operational configuration of the storage container or other SSCs important to safety (e.g., a storage container on a storage pad, a storage container in a shielding structure)

- the physical location of the SSCs important to safety at the time of the hypothetical flood **(SL)**

- the source of the flood water based on historical data for the site as well as current and projected site characteristics (e.g., nearby dams and reservoirs) **(SL)**

- objects that may pose a flood-borne hazard

16.5.2.3.2 Define the Physical Parameters

Verify that the SAR defines the physical parameters associated with the flood condition, including the following:

- the quantity of flood water (i.e., the static head of water and the maximum flow velocity)

- any protection devices placed at the site to prevent containers from tipping over or sliding

- any protections against flood-borne objects **(SL)**

16.5.2.3.3 Review the Analysis Methodology

Verify that the SAR defines the analysis methodology used in the evaluation, such as the following:

- sliding and overturning
- evaluation of external pressure stress intensity

16.5.2.3.4 Accident Analysis

Verify that the SAR presents the accident analysis, design criteria, and design codes and standards, such as the following:

- the design-basis flood conditions

- the determination of the maximum drag force acting on the confinement container or other SSCs important to safety

- the determination of the pressure loading acting on the SSCs

- the determination of the external pressure stress intensity and comparison with the allowable stress as found in the ASME B&PV Code, Section III, Service Level C

- determination that there is no sliding and overturning of the SSCs, or other damage to SSCs

- determination of the consequences of impacts from flood-borne objects and hazards

- compliance with RG 1.59, "Design Basis Floods for Nuclear Power Plants," and RG 1.102, "Flood Protection for Nuclear Power Plants," where applicable

- calculation of dose consequences

16.5.2.4 *Fire and Explosions*

The applicant must evaluate fire and explosion accidents, in accordance with 10 CFR 72.122(c). Coordinate the evaluation of these accidents with that for the site characteristics, as defined in the SAR and reviewed using Chapter 2, "Site Characteristics Evaluation for Dry Storage Facilities," of this SRP for DSF specific license applications. For DSS applications, ensure that the SAR defines a set of fire and explosion parameters, the effects of which the DSS must withstand, and the basis for the selection of those parameters. The following steps provide with an outline of the methodology for evaluating the fire and explosion accidents.

16.5.2.4.1 Define the Operating Environment

Verify that site characteristics chapter (for SLs), thermal chapter, and the materials chapter of the SAR identify the operating environment for a fire or explosion accident, including the following:

- the presence of materials that could accidentally burn or explode in the vicinity of the SNF storage facility or along the route of transfer at the site for DSFs; for DSSs, the presence of materials close to the DSS that could burn or explode (e.g., fuel tank of transporter moving the DSS to the storage pad) and other conditions that are reasonable to anticipate for sites that may use the DSS

- operational conditions that could accidentally initiate combustion or explosion

16.5.2.4.2 Define the Physical Parameters

Verify that the SAR defines the physical parameters associated with the accidents, including the following:

- the quantity of combustible fuel and materials present at the site for DSFs; for DSSs, the quantity of such materials assumed present and the basis for the assumptions

- the barriers in place to protect the SSCs from damage by heat or explosive overpressure

- the presence of a fire protection program **(SL)**

16.5.2.4.3 Review the Analysis Methodology

Verify that the SAR defines the methodology by which the fire or explosion hazards are to be evaluated, including the following:

- modeling techniques for calculating the temperature rise of SSCs

- assumptions and modeling techniques for predicting the structural response of SSCs to external or internal pressure

16.5.2.4.4 Accident Analysis

Verify that the SAR presents the accident analysis and design criteria and standards to do the following:

- Establish design criteria for temperature limits for temperature-sensitive materials and SSCs such as concrete, fuel cladding, shielding materials, and confinement boundary components subject to internal pressure rise or external pressure rise.

- Show that the maximum temperature resulting from the accidental fire does not reach the design limit and that the effect on the SSCs has been evaluated in the structural evaluation chapter.

- Show that the maximum internal pressure for a storage container is properly evaluated and verify that the maximum internal pressure of the storage container remains within its design pressures for accident conditions (assuming 100-percent fuel rod rupture with 100 percent of the initial fill gas and 30 percent of the fission product gas generated within the fuel rods during operation).

- Show that the maximum external pressure does not cause a breach of the confinement boundary and that the stress-intensity level is below the stress limit (i.e., ASME B&PV Code, Section III, Service Level D). Also consider the effect of confinement O-rings or seals and relevant pressure monitoring systems of bolted lid designs.

- Verify that a fire protection program provides assurance that a fire will not significantly increase the risk of radioactive releases to the environment; ensure that the fire protection program consists of fire detection and extinguishing systems and equipment, administrative controls and procedures, and trained personnel. **(SL)**

- Confirm that control room or control area ventilation system piping and instrumentation drawings show monitors located in the system intakes that can detect radiation, smoke, and toxic chemicals, if applicable. **(SL)**

- Confirm that monitors actuate alarms in the control room or other suitable locations, if applicable; consult RG 1.189, “Fire Protection for Nuclear Power Plants,” for detailed guidance. **(SL)**

- Verify that areas storing flammable, combustible, and hazardous materials are located and protected so that a fire, explosion, or release of hazardous materials will not adversely affect any SSCs important to safety. **(SL)**

- Verify that materials that collect and contain radioactive materials, such as spent ion exchange resins, charcoal filters, and high-efficiency particulate air filters, are stored in closed metal tanks located away from ignition sources and combustible material.

- Confirm that any accidental release together with direct radiation results in doses that do not exceed the limits in 10 CFR 72.106(b).

16.5.2.5 *Lightning*

Lightning is an event that the applicant must evaluate, in compliance with 10 CFR 72.122(b)(2)(i) and 10 CFR 72.236(b). The following steps provide an outline of the methodology for evaluating the lightning accident.

16.5.2.5.1 Define the Operating Environment and Physical Parameters

Verify that the SAR identifies the operating environment condition for a lightning strike, including the following:

- storage container SSCs that are exposed to possible lightning strikes
- other storage facility SSCs that are exposed to lightning strikes **(SL)**
- lightning protective devices included as a part of the design

16.5.2.5.2 Review the Analysis Methodology and Accident Analysis

Verify that the SAR presents an analysis or discussion of the effects of lightning strikes on all SSCs important to safety and, for DSFs, facility buildings, including the following:

- a discussion of structural materials or components, including monitoring or surveillance instrumentation and equipment, that might be damaged by heat or mechanical forces generated by passing current to ground

- any radiological consequences associated with the lightning strike

16.5.2.6 *Earthquake*

The earthquake accident is one of the accidents that the applicant must evaluate, in accordance with 10 CFR 72.122(b)(2)(i) and 10 CFR 72.236(b). Coordinate the evaluation of this topic in the SAR with the site characteristics evaluation under Chapter 2 of this SRP for DSF specific license applications. For DSS applications, ensure that the SAR defines a set of earthquake or ground-motion parameters, the effects of which the DSS must withstand, and the basis for the selection of those parameters. The following steps provide an outline of the methodology that the applicant should provide in the SAR.

16.5.2.6.1 Define the Operating Environment

Determine the design ground motion according to the SAR. For SLs, refer to Chapter 2 of this SRP, which discusses this parameter, including the evaluation of the rationale for its selection.

Verify that the SAR has defined the configuration of the SSCs at the time of the seismic event (e.g., the container on the storage pad, the loaded transfer cask during transfer operations, the loaded transfer cask or the container suspended from a crane); the applicant should consider multiple configurations in the evaluation of seismic events and their impacts, including temporary expected configurations (e.g., construction activities to expand an operating array of storage containers that removes or exposes materials relied on for shielding by the operating storage containers).

16.5.2.6.2 Define the Physical Parameters

Determine which components of the DSS or DSF must be designed to withstand the effects of the design earthquake. General Design Criterion 2, "Design Bases for Protection Against Natural Phenomena," of Appendix A, "General Design Criteria for Nuclear Power Plants," to 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities," requires that nuclear power plant SSCs be designed to withstand the effects of earthquakes without loss of capability to perform their safety functions. RG 1.29, "Seismic Design Classification," describes a method for identifying those features of a light-water reactor that should be designed to withstand the effects of the safe-shutdown earthquake. The staff has interpreted this regulatory guide to mean that those SSCs identified as important to safety, and other SSCs that could affect SSCs important to safety, should be designed for the design-basis earthquake. Refer to Chapter 3 of this SRP for an evaluation of the identification of these components. Confirm that the applicant has identified protection devices to mitigate effects of the event, such as a seismic sensor to trip power to overhead cranes or extra seismic supports to be installed during transfer operations.

16.5.2.6.3 Review the Analysis Methodology

If the applicant uses an equivalent static load method, verify that the method produces conservative results and that the SSCs can be realistically represented by a simple model.

If the applicant uses a response spectrum analysis technique, verify that the response spectra meet the requirements in RG 1.60, "Design Response Spectra for Seismic Design of Nuclear Power Plants," and that damping ratios are in accordance with RG 1.61, "Damping Values for Seismic Design of Nuclear Power Plants."

If the applicant has performed a time-history analysis, verify that the time-history of acceleration is in compliance with American Society of Civil Engineers (ASCE) 4-98, "Seismic Analysis of Safety-Related Nuclear Structures."

16.5.2.6.4 Accident Analysis

Verify that the analysis has used the three components of earthquake motion and has combined them in accordance with NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition," Section 3.7.2, "Seismic System Analysis," Subsection 6 and RG 1.92, "Combining Modal Responses and Spatial Components in Seismic Response Analysis."

In accordance with NUREG-0800, Section 3.7.2, Subsection 14, verify that the applicant has considered a determination of Category I structure overturning moments. To be acceptable, the determination of the design overturning moment should incorporate three components of input motion and conservative consideration of vertical and lateral seismic forces. Verify that the structure neither overturns nor slides because of the design earthquake.

Verify that the applicant has provided a summary of natural frequencies of the SSCs important to safety. If the applicant has used the direct integration method of analysis, verify that total responses of the SSCs have been calculated.

Verify that the applicant has identified any radiological consequences associated with the seismic event and calculated dose rates and doses. Although SSCs are not required to survive accident condition earthquakes without permanent deformation, verify that the stress intensities are less than the stress allowables (i.e., ASME B&PV Code, Section III, Service Level D).

16.5.2.7 *Loss of Shielding*

The applicant must evaluate the loss of shielding of any SSCs identified as important to safety to determine the dose to workers and the public. Loss of shielding can occur because of a variety events, such as a penetration of the concrete shielding induced by the impact of a tornado missile, the reduction in hydrogen content of neutron shielding by high-temperature exposure, loss of water or lowering of the water level by leakage from shields that are composed of water, or structural failure or melting of shielding by fire or explosion, and others. The following steps provide an outline of the methodology that the applicant should provide in the SAR.

16.5.2.7.1 Define the Operating Environment and Physical Parameters

Verify that the SAR identifies the operating environment and the physical parameters of the accident, including the following:

- the operational configuration of the SCCs such as a container design that uses a liquid shielding material

- the design threshold for safety pressure-relief valves or rupture discs for liquid shield tanks

- relevant material specifications for shield materials (e.g., melting temperature, temperature of decomposition, mechanical strength)

16.5.2.7.2 Review the Analysis Methodology and Accident Analysis

Verify that the applicant has appropriately determined the maximum reduction of the radiation shielding thickness, material shielding effectiveness, or loss of temporary shielding in DSS or DSF SSCs and features as a result of postulated accidents such as tornado missiles, explosions, fires, liquid shield tank leaks, and container drop. Confirm that the applicant evaluated all possible shielding areas.

Verify that the applicant has performed a revised neutron and gamma dose rate shielding analysis with the accident-induced reduction or loss of shielding. The analysis should use computer codes and methodologies, as applicable, identical to those of the design shielding calculations for the DSS or DSF SSCs and features.

16.5.2.8 *Adiabatic Heatup*

Adiabatic heatup is a key assumption for an evaluated accident because it ensures that the applicant has evaluated the most conservative thermal transient response of the DSS or DSF SSCs. The transient temperature response of internal container components, including the contents, is solely a result of the decay heat of the contents and the individual container material heat capacity (i.e., mass and specific heat). The following steps provide an outline of the methodology that the applicant should provide in the SAR.

16.5.2.8.1 Define the Operating Environment

Verify that the SAR defines the ambient temperature, including insolation, used in the accident analysis. Verify that the applicant has defined the configuration of the SSCs (e.g., all inlets and outlets blocked for casks). Evaluate the highest design-basis decay heat load of the design, which should be stated in the principal design criteria chapter of the SAR.

16.5.2.8.2 Define the Physical Parameters

Verify the minimum mass of each material that constitutes a component of the DSS or DSF SSCs and features and the stored radioactive materials. Such materials are typically uranium dioxide, zircaloy, stainless steel, inconel, carbon steel, neutron absorber plates (e.g., boral, borated aluminum), (borated) resin, (borated) polyethylene, and concrete. In general, the mass can be calculated by determining the volume of the material and using a value for density of the material that is obtained from an established reference of material properties. The density should be appropriate for the anticipated temperature range for this calculation.

Determine the specific heat of each material from established references for the expected range of temperatures.

Determine the maximum short-term accident temperature limit of each material comprising DSS or DSF SSCs and features from established references.

16.5.2.8.3 Review the Analysis Methodology and Accident Analysis

Ensure that all containers that rely on natural air convection through internal labyrinthine passages assume that all air inlet and outlet passages are completely blocked. The thermal response must be calculated by assuming that no heat loss to the environment occurs. For example, for SNF casks having multiple air inlets and outlets; the staff has previously found it unacceptable to assume that one air outlet would become an air inlet while the other air outlets would continue to function as outlets. The staff has rejected this assumption because it has not been verified by experimental test data.

Calculate the sum of the product of mass and specific heat for each material. This is denoted as the heat capacity of the DSS or DSF SSCs.

Calculate the adiabatic heatup rate of the SSCs by dividing the total DSS or DSF storage container maximum decay heat load by the total SSC heat capacity.

Using the highest calculated temperature for each material at normal operating ambient temperatures, the maximum short-term accident temperature limit for each material, and the DSS or DSF SSC adiabatic heatup rate that was calculated in accordance with the above paragraph, determine the earliest time that a specific material temperature limit will be exceeded after the onset of an adiabatic heatup scenario.

Report, as the key result, the minimum time to reach the first material temperature limit during an adiabatic heatup event. The technical specifications must include a surveillance frequency. Ensure that the applicant provided a technical specification for any material that might exceed its temperature limit during an adiabatic heatup. See Chapter 5, “Thermal Evaluation,” of this SRP for more details.

Verify that the applicant has identified any radiological consequences associated with the adiabatic heatup and has calculated dose rates and doses.

16.5.2.9 *Tornadoes and Missiles Generated by Natural Phenomena*

The applicant must evaluate tornado and tornado-generated missile accidents, in accordance with 10 CFR 72.122(b)(2). Coordinate the evaluation of this material in the SAR with the site characteristics review based on Chapter 2 of this SRP for DSF specific license applications. For DSS applications, ensure that the SAR defines a set of tornado and missile parameters, the effects of which the DSS must withstand, and the basis for the selection of those parameters. The following steps provide an outline of the methodology that the applicant should provide in the SAR.

16.5.2.9.1 Define the Operating Environment and Physical Parameters

Review the SAR to determine the design wind and tornado wind velocities. Verify that the applicant analyzed design-basis tornado characteristics given in RG 1.76, "Design Basis Tornado and Tornado Missiles for Nuclear Power Plants."

Verify that the applicant used design-basis tornado missile spectra and maximum horizontal speeds from RG 1.76 in the analysis of missile impacts.

The NRC considers the missiles described in RG 1.76 capable of striking in all directions with horizontal velocities of V_{Mh}^{max} and vertical velocities equal to 67 percent of V_{Mh}^{max}. Barrier design should be evaluated assuming a normal impact to the surface for the Schedule 40 pipe and automobile missiles. The automobile missile is considered to impact at all altitudes less than 9.14 meters (30 feet) above all grade levels within 0.8 kilometer (0.5 mile) of the plant structures. Table 2 of RG 1.76 includes a different size and weight automobile for Region III than for Regions I and II (as defined in defined in RG 1.76). The heavier automobile used in the calculations for Regions I and II will have a lower kinetic energy in Region III. This effect is a consequence of the low maximum horizontal speed V_{Mh}^{max} of the heavier automobile in the Region III tornado wind field.

16.5.2.9.2 Review the Analysis Methodology and Accident Analysis

Verify the transformation of wind velocity into pressure. The NRC staff accepts the procedures used to transform the wind velocity into an effective pressure to be applied to structures and parts and portions of structures found in ASCE/Structural Engineering Institute (SEI) 7, "Minimum Design Loads for Buildings and Other Structures." These procedures specify that the maximum velocity pressure, *p* (in pounds per square foot), should be obtained from the formula, $p = 0.00256\ V^2$, where *V* is in miles per hour; the velocity pressure should be assumed constant with height; and the maximum pressure applies at the radius of the tornado funnel at which the maximum velocity occurs. ASCE Paper No. 3269, "Wind Forces on Structures," issued in 1961, may be used to obtain the effective wind pressures for cases that ASCE/SEI 7 does not cover.

Verify that the applicant has analyzed all SSCs important to safety for damage from missiles that the design-basis tornado might generate (note: the design-basis tornado can vary depending on the location of the DSS or DSF). Also review the applicant's analysis of missile impact on SSCs important to safety. In previous submittals, the NRC has accepted the use of "A Review of Procedures for the Analysis and Design of Concrete Structures to Resist Missile Impact Effects"

(Kennedy 1975); “Design of Structures for Missile Impact” (Linderman et al. 1974); and “U.S. Reactor Containment Technology” (Cottrell and Savolainen 1965).

Verify that the applicant has calculated the most adverse combination of tornado wind, differential pressure, and missile load and used it in combination with other loads. To obtain the most adverse combination, the combinations should include wind alone, differential pressure alone, missile alone, wind plus half of the differential pressure, wind plus missile, and wind plus missile plus half of the differential pressure.

Verify that the applicant has identified any radiological consequences associated with the tornado and tornado-generated missiles and calculated the dose rates and doses.

16.5.2.10 *Accidents at Nearby Sites* ***(SL)***

Verify that the applicant has considered potential accidents at nearby sites and transportation routes. Reviews conducted under other sections of this SRP will have covered the procedures for reviewing these accidents (e.g., a natural gas explosion at a nearby site may result in an explosive overpressure and the effects of a fire at a nearby site). Verify that the effects of nearby site accidents have been encompassed by the effects of other accidents identified and evaluated in the SAR and reviewed as part of this SRP chapter.

Confirm that the SAR defines the analysis, design criteria, and design codes and standards (as applicable) for each off-normal and accident event as related to SNF, HLW, or reactor-related GTCC waste storage and handling systems.

16.5.2.11 *Structural Failures Resulting from Fire and Their Potential Impacts* ***(SL)***

Buildings must be designed to withstand collapse from the effects of flood, fire and explosion, lightning, earthquake, tornado and tornado-generated missiles, and accidents at nearby sites in accordance to their importance to safety or the potential impacts of their failures on SSCs important to safety. Other parts of Section 16.5 of this chapter present the review procedures for these events for SSCs important to safety. Verify that the applicant has analyzed the building structure to meet the applicable portions of these procedures. The applicant’s analysis should provide evidence that, although equipment or structures may be damaged, the surviving equipment and structures will continue to protect the SNF, HLW, and reactor-related GTCC waste and that the radiological consequences are within acceptable levels.

16.5.3 Other Non-Specified Off-Normal Events and Accidents

Evaluate other off-normal and accident scenarios included in the SAR but not identified in the previous subsections of this SRP. Coordinate the accident analysis review with the reviewers of all technical chapters of this SRP to verify that design and operations characteristics of the DSS or DSF do not pose potential off-normal events or accidents that the applicants have not identified or evaluated.

16.6 Evaluation Findings

The NRC reviewer should prepare evaluation findings on satisfaction of the regulatory requirements in Section 16.4 of this SRP. If the documentation submitted with the application fully supports positive findings for each of the regulatory requirements, the statements of findings should be similar to the following:

F16.1 The SAR includes acceptable analyses of the design and performance of confinement and SSCs important to safety, and other SSCs that affect SSCs important to safety, under off-normal and accident scenarios to meet the requirements in 10 CFR 72.24 for a DSF or 10 CFR 72.236(c), (d), and (l) for a DSS. Applicable off-normal events analyzed in the SAR include [reviewer to select from the following:] partial vent blockage, operational events resulting in radioactive release [reviewer to list], off-normal ambient temperature scenarios, and [other off-normal events identified by the applicant or as part of the review]. Applicable accident events analyzed in the SAR include [reviewer to select from the following:] container tipover, container drop, flood, fire and explosion, lightning, earthquake, loss of shielding [if applicable], adiabatic heatup of the container, tornadoes and missiles generated by natural phenomena, accidents at nearby sites, building structural failure onto SSCs, and [other scenarios identified by the applicant or as part of the review].

F16.2 **(SL)** The analyses of off-normal and accident events and conditions and reasonable combinations of these and normal conditions show that the design of the DSF will acceptably meet the applicable regulatory requirements without endangering the public health and safety, in compliance with the overall requirements in 10 CFR 72.122.

F16.3 **(CoC)** The analyses of off-normal and accident events and conditions and reasonable combinations of these and normal conditions show that the design of the DSS will facilitate meeting the applicable regulatory requirements without endangering the public health and safety, in compliance with the overall requirements in 10 CFR 72.122.

F16.4 The analyses of off-normal and accident events and conditions and reasonable combinations of these and normal conditions show that the design of the DSS or DSF will acceptably meet the requirements of 10 CFR 72.124, “Criteria for Nuclear Criticality Safety,” and, for DSSs, 10 CFR 72.236(c) regarding the maintenance of the SNF or HLW, or both, in a subcritical condition.

F16.5 The analyses of off-normal and accident events and conditions and reasonable combinations of these and normal conditions show that the design of the DSS or DSF will acceptably meet the requirements in (10 CFR 72.126, “Criteria for Radiological Protection,” (for DSFs) or 10 CFR 72.236(d) (for DSSs)) regarding criteria for radiological protection.

F16.6 **(SL)** The analyses of off-normal and accident events and conditions and reasonable combinations of these and normal conditions show that the design of the DSF will acceptably meet the requirements of 10 CFR 72.128 regarding handling and storage of the SNF and other radioactive material and confinement.

F16.7 **(CoC)** The analyses of off-normal and accident events and conditions and reasonable combinations of these and normal conditions show that the design of the DSS will facilitate meeting the requirements of

10 CFR 72.128 regarding handling and storage of the SNF and other radioactive material and confinement.

F16.8 No instruments or control systems are required to remain operational under accident conditions [as applicable] under 10 CFR 72.122(i).

The reviewer should provide a summary statement similar to the following:

> The staff concludes that the accident design criteria for the [DSS or DSF designation] are in compliance with 10 CFR Part 72, and the accident design and acceptance criteria have been satisfied. The applicant's accident evaluation of the DSS or DSF adequately demonstrates that it will provide for the safe storage of the stored radioactive materials during off-normal and accident conditions (for DSFs) or the accident conditions for which the DSS was designed (for DSSs) and during off-normal conditions (for which it was designed (for DSSs)). This finding is reached on the basis of a review that considered independent confirmatory calculations, the regulation itself, appropriate regulatory guides, applicable codes and standards, and accepted engineering practices.

16.7 <u>References</u>

10 CFR Part 20, "Standards for Protection Against Radiation."

10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities."

10 CFR Part 71, "Packaging and Transportation of Radioactive Material."

10 CFR Part 72, "Licensing Requirements for the Independent Storage of Spent Nuclear Fuel, High-Level Radioactive Waste, and Reactor-Related Greater than Class C Waste."

10 CFR Part 73, "Physical Protection of Plants and Materials."

Air Movement and Control Association, Standards and Application Guides.

American Concrete Institute (ACI) 318, "Building Code Requirements for Structural Concrete and Commentary."

ACI 349, "Code Requirements for Nuclear Safety Related Concrete Structures and Commentary."

American Iron and Steel Institute, "Steel Products Manual"

American National Standards Institute (ANSI) N14.6, "Radioactive Materials—Special Lifting Devices for Shipping Containers Weighing 10,000 Pounds (4500 kg) or More," Institute for Nuclear Materials Management.

ANSI/American Nuclear Society (ANS) 57.2, "Design Requirements for Light Water Reactor Spent Fuel Storage Facilities at Nuclear Power Plants."

ANSI/ANS 57.7, "Design Criteria for an Independent Spent Fuel Storage Installation (Water Pool Type)."

ANSI/ANS 57.9-1992, “Design Criteria for an Independent Spent Fuel Storage Installation (Dry Storage Type),” 1992 (reaffirmed 2000).

American Society of Civil Engineers (ASCE) 4-98, “Seismic Analysis of Safety-Related Nuclear Structures.”

ASCE/Structural Engineering Institute 7, “Minimum Design Loads for Buildings and Other Structures.”

ASCE Paper No. 3269, “Wind Forces on Structures,” *Transactions*, 126(Part II), pp. 1124–1198, 1961.

American Society of Heating, Refrigeration and Air-Conditioning Engineers, “ASHRAE Handbook—Fundamentals.”

American Society of Mechanical Engineers (ASME) Boiler and Pressure (B&PV) Code.
Section III, “Rules for Construction of Nuclear Facility Components.”

ASME B30.2, “Overhead and Gantry Cranes (Top Running Bridge, Single or Multiple Girder, Top Running Trolley Hoist).”

ASME B30.16, “Overhead Hoists (Underhung).”

ASME N509, “Nuclear Power Plant Air-Cleaning Units and Components.”

Cottrell, W.B. and A.W. Savolainen, “U.S. Reactor Containment Technology,” in ORNL-NSIC-5, Volume 1, Chapter 6, Oak Ridge National Laboratory, August 1965.

Crane Manufacturers Association of America Specification No. 70, “Specifications for Top Running Bridge and Gantry Type Multiple Girder Electric Overhead Traveling Cranes.”

Institute of Electrical and Electronics Engineers (IEEE) C2, “National Electrical Safety Code.”

IEEE 835, “Standard Power Cable Ampacity Tables.”

International Code Council (ICC), “International Building Code.”

ICC, “International Mechanical Code.”

Kennedy, R.P., “A Review of Procedures for the Analysis and Design of Concrete Structures to Resist Missile Impact Effects,” in ORNL-NSIC-5, Volume 1, Chapter 6, Holmes and Narver, Inc., September 1975.

Linderman, R.B., J.V. Rotz, and G.C.K. Yeh, “Design of Structures for Missile Impact,” Topical Report BC-TOP-9-A, Revision 2, Bechtel Power Corporation, September 1974.

National Fire Protection Association (NFPA) 70, “National Electrical Code.”

NFPA 780, “Standard for the Installation of Lightning Protection Systems.”

NUREG-0800, “Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition.”

Regulatory Guide 1.4, “Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident for Pressurized Water Reactors.”

Regulatory Guide 1.25, “Assumptions Used for Evaluating the Potential Radiological Consequences of a Fuel Handling Accident in the Fuel Handling and Storage Facility for Boiling and Pressurized Water Reactors.”

Regulatory Guide 1.29, “Seismic Design Classification.”

Regulatory Guide 1.59, “Design Basis Floods for Nuclear Power Plants.”

Regulatory Guide 1.60, “Design Response Spectra for Seismic Design of Nuclear Power Plants.”

Regulatory Guide 1.61, “Damping Values for Seismic Design of Nuclear Power Plants.”

Regulatory Guide 1.76, “Design Basis Tornado and Tornado Missiles for Nuclear Power Plants,” Revision 1, issued March 2007.

Regulatory Guide 1.92, “Combining Modal Responses and Spatial Components in Seismic Response Analysis.”

Regulatory Guide 1.102, “Flood Protection for Nuclear Power Plants.”

Regulatory Guide 1.109, “Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I.”

Regulatory Guide 1.145, “Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants.”

Regulatory Guide 1.189, “Fire Protection for Nuclear Power Plants.”

17 TECHNICAL SPECIFICATIONS EVALUATION

17.1 Review Objective

The objective of the U.S. Nuclear Regulatory Commission's (NRC's) review with regard to technical specifications is to ensure the conditions and technical specifications of the dry storage facility (DSF) license or dry storage system (DSS) certificate of compliance (CoC) are sufficient and include those conditions and limits that are necessary to ensure that the design and operations of the DSF or DSS will meet the requirements in Title 10 of the *Code of Federal Regulations* (10 CFR) Part 72, "Licensing Requirements for the Independent Storage of Spent Nuclear Fuel, High-Level Radioactive Waste and Reactor-Related Greater Than Class C Waste."

17.2 Applicability

This chapter applies to the review of applications for specific licenses to an independent spent fuel storage installation (ISFSI) or a monitored retrievable storage installation (MRS), categorized as a DSF. It also applies to the review of applications for a DSS CoC.

17.3 Areas of Review

This chapter applies to the evaluation of the proposed technical specifications that the applicant deems necessary for correct fabrication and safe operation of the proposed DSS or DSF. A comprehensive review of the proposed technical specifications will assess the applicant's compliance with the regulations to provide a level of control commensurate with the applicable regulations specified in Section 17.4 below. This chapter addresses the following areas of review:

- functional and operating limits, monitoring instruments, and limiting control settings
- limiting conditions
- surveillance requirements
- design features
- administrative controls

17.4 Regulatory Requirements and Acceptance Criteria

This section summarizes those parts of 10 CFR Part 72 that are relevant to the review areas addressed in this chapter. The NRC staff reviewer should refer to the exact language in the regulations. Tables 17-1a and 17-1b match the relevant regulatory requirements to the areas of review covered in this chapter.

Table 17-1a Relationship of Regulations and Areas of Review for a DSF (SL)

Areas of Review	10 CFR Part 72 Regulations		
	72.24(g)	72.26	72.44(c)(d)
Functional and Operating Limits, Monitoring Instruments, and Limiting Control Settings	●	●	●
Limiting Conditions	●	●	●
Surveillance Requirements	●	●	●
Design Features	●	●	●
Administrative Controls	●	●	●

Table 17-1b Relationship of Regulations and Areas of Review for a DSS (CoC)

Areas of Review	10 CFR Part 72 Regulations								
	72.236								
	(a)	(b)	(c)	(d)	(e)(f)(h)	(g)	(i)	(j)	(l)
Functional and Operating Limits, Monitoring Instruments, and Limiting Control Settings	●		●	●	●	●	●		●
Limiting Conditions	●		●	●		●			●
Surveillance Requirements			●	●		●		●	
Design Features		●	●	●	●	●	●		●
Administrative Controls	●		●	●			●		●

The applicant should identify, as needed, proposed license or CoC conditions, including technical specifications, that are necessary to maintain subcriticality, confinement, shielding, heat removal, and structural integrity under normal, off-normal, and accident conditions. In addition, the applicant should identify the basis for each of the proposed technical specifications by reference to the analysis in the safety analysis report (SAR).

While the regulations in 10 CFR 72.26, "Contents of application: Technical specifications," and 10 CFR 72.44, "License conditions," do not specifically require technical specifications for CoCs like they do for specific licenses, the regulations do allow for certificate conditions. For consistency with specific licenses, the staff has used technical specifications as the process for including conditions in CoCs. Examples of this include references in the regulations to "terms, conditions, and specifications" for a CoC that a general licensee would need to meet (see 10 CFR 72.212(b)(3), 10 CFR 72.212(b)(5)(i), 10CFR 72.212(b)(11), and 10 CFR 72.48(c)(1)(ii)(B)). Thus, proposed technical specifications should be provided in CoC applications. The proposed technical specifications should be derived with consideration of what is needed to ensure compliance with the requirements in 10 CFR 72.236, "Specific requirements for spent fuel storage cask approval and fabrication," as Table 17-1b identifies.

For a DSF, the staff should refer to Regulatory Guide (RG) 3.62, “Standard Format and Content for the Safety Analysis Report for Onsite Storage of Spent Fuel Storage Casks,” RG 3.61, “Standard Format and Content for a Topical Safety Analysis Report for a Spent Fuel Dry Storage Cask,” and RG 3.48, “Standard Format and Content for the Safety Analysis Report for an Independent Spent Fuel Storage Installation or Monitored Retrievable Storage Installation (Dry Storage).”

For a DSS, the NRC staff can use NUREG-1745, “Standard Format and Content for Technical Specifications for 10 CFR Part 72 Cask Certificates of Compliance” as an appropriate template in the review of the technical specifications. However, the staff may impose alternative or additional technical specifications to NUREG-1745 guidance based on operational experience and uniqueness of the DSS design and operations and based on the NRC Office of the General Counsel legal interpretations that have been made since issuance of NUREG-1745. For example, interpretations made since the issuance of NUREG-1745 have found certain aspects of that guidance to be legally unacceptable. These aspects include the guidance in Section 2.2 of NUREG-1745 regarding an option for controlling and obtaining NRC approval of changes to some spent nuclear fuel (SNF) parameters outside of the technical specifications.

Additionally, NUREG-1745 indicates some items that are usually included as limiting conditions for operation (see Section 17.4.2, “Limiting Conditions,” below) that may be dealt with in the administrative controls section (see Section 17.4.5, “Administrative Controls,” below) of the technical specifications. In order for this option to be used, the administrative controls section would need to include appropriate programs, including program elements, and methods to ensure the conditions will be maintained for which a limiting condition for operation would otherwise have been specified. The applicant would then need to include descriptions of operations that implement the administrative controls’ programs and methods in the operations description chapter of the SAR. The reviewer would need to coordinate review of these programs and methods with the reviewer of Chapter 11, “Operation Procedures and Systems Evaluation,” of this standard review plan (SRP) to ensure that the SAR operations descriptions include the necessary operations to effectively and adequately implement the proposed programs and methods.

This chapter focuses on the technical specifications for a license or CoC, as appropriate; however, licenses and CoCs should include terms and conditions in addition to technical specifications that are necessary to ensure compliance with the regulations. NUREG-1745 includes descriptions and examples of standard CoC conditions that may also be applicable to a license. The staff should review approved licenses and CoCs similar to the DSF or DSS under review to garner information on the kinds of terms and conditions that should be included in licenses and CoCs, respectively.

17.4.1 Functional and Operating Limits, Monitoring Instruments, and Limiting Control Settings

Functional and operating limits, monitoring instruments, and limiting control settings should include limits placed on fuel, waste handling, and storage conditions to protect the integrity of the SNF and container, to protect the employees against occupational exposures, to ensure doses to the public will not exceed limits, to ensure subcriticality, and to guard against the uncontrolled release of radioactive materials.

17.4.2 Limiting Conditions

Limiting conditions identify the lowest functional capability or performance level of structures, systems, and components (SSCs) required for safe operation. Limiting conditions should include limits placed on fuel, waste handling, and storage conditions to protect the integrity of the contents and SSCs important to safety, and to ensure protection of employees against occupational exposures, to ensure doses to the public will not exceed limits, to ensure subcriticality, and to guard against the uncontrolled release of radioactive materials.

17.4.3 Surveillance Requirements

Acceptance criteria for establishing surveillance requirements include the frequency and scope of surveillance requirements to verify the performance and availability of SSCs important to safety, and, as needed, to verify that the bases for the proposed limiting conditions are maintained. Acceptance criteria also include verifying that the surveillance requirements are sufficient to verify that the limiting conditions, operating limits, functional limits, and limiting control settings are met and that monitoring instruments are performing as designed and needed.

17.4.4 Design Features

Design features should include the specific codes and standards to which DSS or DSF SSCs and design features will be fabricated, constructed, and tested and include other necessary design-specific specifications for SSCs (e.g., minimum flux trap sizes, minimum neutron absorber boron-10 content). The condition or technical specifications should also describe a process to address necessary deviations from the applicable codes. In such cases, the applicant should request authorization to use an alternative to the requirements of the applicable code. If the staff finds that the deviation does not adversely impact safety, it may authorize the requested alternative in writing.

Currently, there is an existing code for the design and construction of metallic SNF storage casks. This code is Subsection WC of Division 3 of Section III of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (B&PV) Code. Subsection WC was first issued as the 2005 addenda to the 2004 ASME B&PV Code. The NRC staff has not taken a position regarding the acceptability of this subsection. In the past, the NRC staff has used Division 1 of the ASME B&PV Code and allowed alternatives to some provisions of that document judged to not be applicable to SNF storage casks. The NRC issued early SNF dry storage licenses and CoCs without documenting which specific alternatives to ASME B&PV Code, Section III that the staff had approved. Poor quality assurance practices during design and fabrication sometimes led to significant deviations from the ASME B&PV Code without appropriate certificate holder design review or NRC review and approval. Therefore, the applicant should document that fabrication, construction, and testing will be done in accordance with ASME B&PV Code, Section III, with proposed alternatives in the application.

Likewise, the NRC should document this information in the technical specifications along with its approval of the proposed alternatives in the safety evaluation report (SER). The NRC should include a statement (in the technical specifications in the SER) that refers the reader to the SAR and applicable SERs for any alternatives to the codes if not already included in the technical specifications. In addition, the applicant should identify other codes and standards applied to SSCs important to safety in the SAR and should include the same in the technical specifications. Figure 17-1 presents an example of a technical specification provision for allowing alternatives to applicable codes.

#.#.# Codes and Standards
The ASME B&PV Code, Section III, is the governing code for the storage system.

#.#.#.# Design Alternatives to Codes, Standards, and Criteria
SAR Table #-# lists all approved alternatives for the design of the DSS or DSF.

#.#.#.# Construction and Fabrication Alternatives to Codes, Standards, and Criteria
Proposed alternatives to the ASME B&PV Code, Section III, including alternatives referenced in Section [XXX], may be used when authorized by the Director of the Office of Nuclear Material Safety and Safeguards or designee.

The proposal to the NRC must demonstrate that the alternatives would provide an acceptable level of quality and safety, or that compliance with the specified requirements of the ASME B&PV Code, Section III, would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety.

Figure 17-1 Example of a Provision for Allowing Alternatives to Applicable Codes

In addition, acceptance criteria for design features include specifications important to criticality safety. The applicant should ensure that the assemblies' active fuel length remains within the storage container region when required for criticality analyses. One common method is the installation of fuel spacers, upper or lower spacers, as needed, to maintain the assemblies' position under all credible conditions. The minimum boron-10 content of the solid neutron absorbers is another important design feature specification, together with the qualification and acceptance-testing method for ensuring that the neutron absorbers meet the required minimum boron-10 content throughout the absorber material. The proximity of fuel assemblies to each other also affects the storage container's reactivity, generally with reactivity increasing as the assemblies are brought closer together. Therefore, the applicant may specify a minimum dimension(s) between adjacent assembly locations. This dimension may be a minimum flux trap width or a minimum fuel cell pitch. The applicant should also include these design parameters and requirements in the technical specifications.

Additional DSS or DSF design features specifications that may need to be included in the technical specifications include items such as the following:

- important time and other limits associated with draining and drying of the storage container

- systems or features used for corrosion protection of the storage containers

- parameters of features needed for container cooling or combustible gas monitoring

- parameters and controls for features and SSCs related to shielding or radiation protection (e.g., use of shield berms or walls for compliance with 10 CFR 72.104, “Criteria for radioactive materials in effluents and direct radiation from an ISFSI or MRS,” or 10 CFR 72.106, “Controlled area of an ISFSI or MRS,” limits, and requirement for maintenance, including categorization as important to safety)

- site feature parameters to ensure adequate performance of shielding and other functions during different operations configurations for normal, off-normal, and accident conditions (e.g., minimum distances between loaded storage containers and adjacent construction operations (to expand the storage container array) that removes or exposes materials relied on for shielding, use, characteristics)

- unique features and operations characteristics and actions needed for those features to ensure adequate shielding and radiation protection of the public or personnel for different SSCs and operations (e.g., significant supplemental shielding components that are necessary to ensure adequate shielding of personnel during storage container loading operations and restrictions on personnel when such supplemental shielding is not in place), and any needed evaluations for such features and operations for possible operations configurations under normal, off-normal, and accident conditions

- other site parameters and features related to limits of the use of a DSS such as seismic and environmental characteristics

17.4.5 Administrative Controls

Administrative controls should include the organizational and management procedures, recordkeeping, review and audit systems, and reporting necessary to ensure that the DSS or DSF is managed and operated in a safe and reliable manner. Administrative action that must be taken in the event of noncompliance with a limit or condition should be specified.

Administrative controls also should include programs that are needed to address the following items:

- the requirements in 10 CFR 72.44(d) for a DSF

- safe DSS or DSF operations and handling of storage containers (e.g., storage container transport program, lifting (height) or handling parameter limits, as appropriate)

- radiological environmental monitoring program requirements and effluent control program requirements

- envelope of site characteristics for which the DSS has been evaluated, as needed (e.g., design-basis earthquake)

- operating limits (e.g., temperatures restrictions for handling or transport operations, as applicable)

- radiation protection program requirements (e.g., dose rate limits, evaluations, dose rate measurement procedures for verifying limit compliance

17.5 Review Procedures

The technical specifications define the conditions that are deemed necessary for safe DSS or DSF fabrication and operation. Specifically, they define operating limits and controls, monitoring instruments and control settings, surveillance requirements, design features, and administrative controls and programs that ensure safe operation of the DSS or DSF. As such, the DSF license or DSS CoC, as appropriate, includes technical specifications. Ensure that each specification is clearly documented and justified in the technical evaluation sections of the SAR and the associated SER, as necessary, and adequate to ensure safe DSS or DSF operation. With respect to a DSF, the scope includes the whole ISFSI or MRS.

If a reviewer determines that a design feature, content specification, analytical assumption, operating assumption, limiting condition of operation, or other SAR item is important and should not be changed without NRC staff approval, then that item should be further evaluated and considered as a potential technical specification. For example, the reviewer should consider safety margins, operational experience design novelty, and other issues that are unique to each proposed design. The reviewer should also implement the guidance in this chapter for establishing such conditions and technical specifications in the facility license or CoC. Figure 17-2 presents an overview of the evaluation process and can be used as a guide to assist in coordinating among review disciplines.

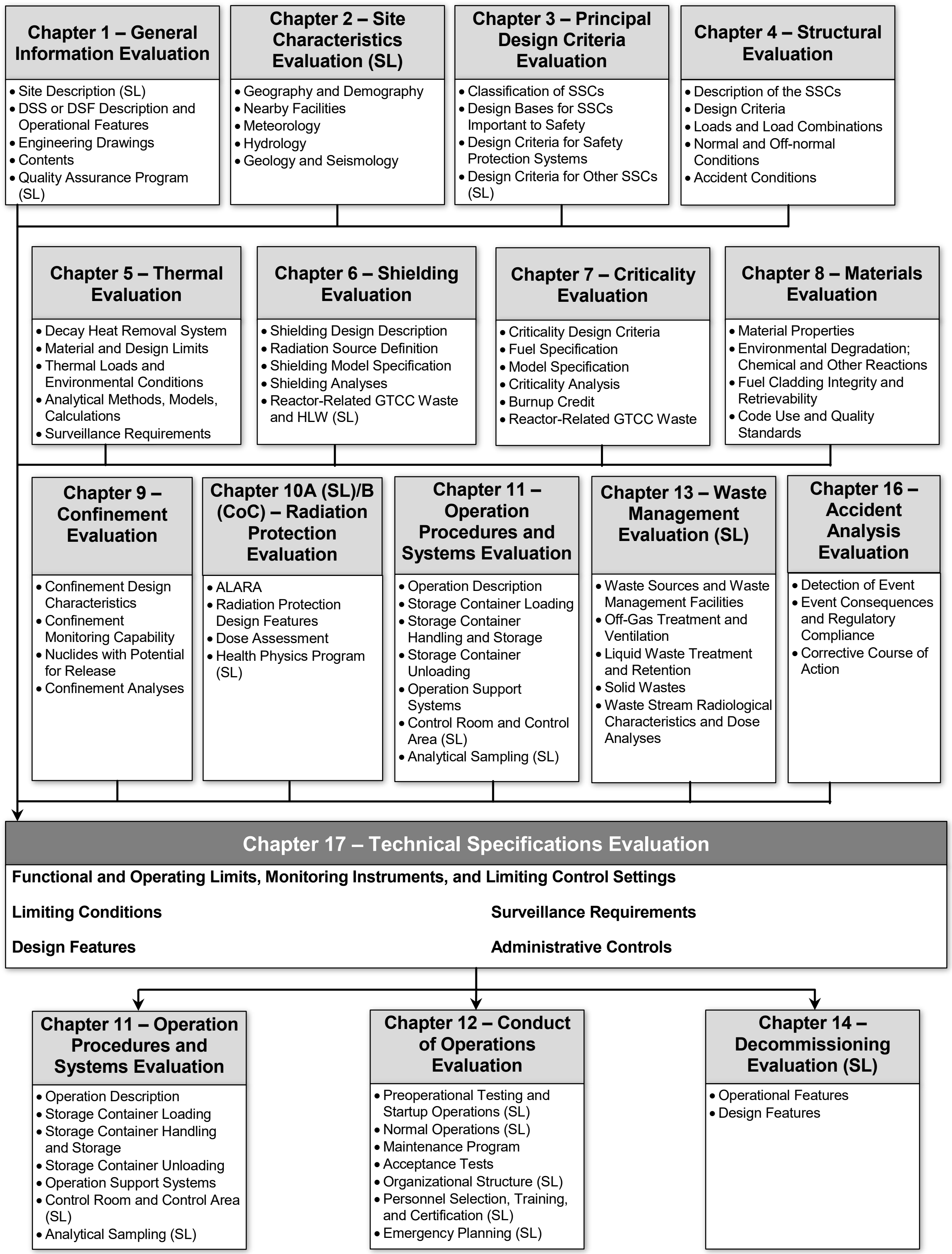

Figure 17-2 Overview of Technical Specifications Evaluation

The NRC staff should evaluate each chapter of the SAR with a goal of establishing the technical specifications or identifying those things that may need technical specifications. The variability of designs and operations makes it impossible to define each instance for which a technical specification is necessary. For this reason, it is important to conduct a coordinated, detailed, and thorough evaluation of each technical section of the SAR. Note all instances in which the SAR either makes an assumption or imposes a condition that should be identified as a technical specification. Note any instances in which the SAR requests alternatives or other conditions that are identified as an operational limit or condition. Such code alternatives should be clearly identified and documented in the SAR chapter on technical specifications.

The various technical disciplines should review the results of their specific evaluations and compare their list of technical specifications to those the applicant identified. The NRC staff should ensure that the conditions for use, as evaluated and approved by the technical reviewers, complement one another and are not contradictory. In addition, the staff will coordinate the resolution of any disputed condition, limit, or specification and is responsible for identifying any unique specifications (e.g., administrative) that may not be covered in the technical sections, although input may be solicited from the technical reviewers regarding any topic.

Become familiar with the technical specifications of similar DSS or DSF designs the NRC staff has previously approved. Note that for a specific license the maximum quantity of SNF, HLW and reactor related GTCC waste is a condition of the license and not included in the technical specifications. For example, the staff has previously approved DSS designs and issued technical specifications regarding a variety of items including, but not limited to, the following examples:

- general requirements and conditions regarding site-specific parameters, operating procedures, quality assurance, heavy loads, training

- preoperational training exercises and demonstrations of most operations, including loading, sealing, and drying (using mockups as appropriate); placement of a storage container on the storage pad; and return of fuel to the SNF pool

- specifications for the SNF to be stored, including, but not limited to, the type of SNF (i.e., boiling-water reactor, pressurized-water reactor, or both), the minimum and maximum allowable enrichments of the fuel before irradiation, burnup (i.e., megawatt days per metric ton uranium), the minimum acceptable cooling time of the SNF before storage, the maximum heat designed to be dissipated, the maximum SNF loading limit, condition of the SNF (i.e., intact assembly or consolidated fuel rods, allowable cladding condition), associated nonfuel hardware, and physical parameters (e.g., length, width, depth, weight). The reviewer should be aware that the technical specifications that rely on burnup credit will need to include additional SNF specifications regarding operational history parameters (e.g., minimum burnup vs enrichment, moderator temperature, in-core soluble boron concentrations, and operations under control rod banks or with control rod insertion)

- as applicable for a specific license, appropriate specifications of the reactor-related greater-than-Class-C (GTCC) waste and high-level radioactive waste (HLW) to be stored, such as waste chemical and physical form, radionuclide characteristics, and heat generation rates (some of this information may be included in either the technical specifications or a separate license condition)

- criticality controls, such as storage container water boron concentrations, minimum flux trap and fuel cell pitch, use of fuel spacers, minimum neutron absorber boron-10 loading, and neutron absorber acceptance tests and qualification program

- the inerting atmosphere requirements during vacuum drying and helium backfill parameters

- handling restrictions, such as lift height limits and operational temperature limit (high-low) conditions

- storage container confinement barrier requirements, such as helium leak rate limits

- thermal performance parameters, such as maximum temperatures or delta-temperatures

- radiological controls such as operational (SSC surface) radiation dose rate and contamination limits and conditions regarding design parameters, operations, and programmatic controls that affect offsite doses

- storage array and spacing limits, as appropriate, for thermal performance and radiological considerations

- definition of damaged fuel

- fabrication and design codes and alternatives to specific code requirements

- specifications or requirements for alternative materials for important-to-safety SSCs

- manufacture and testing of neutron poison material(s) for criticality control

- hydrogen monitoring and mitigation, as appropriate, during wet loading and unloading

- maintaining inert atmosphere during and after storage container draining or flooding to prevent oxidation.

- use of copper-bearing or weathering steel for structural steel components at coastal marine DSF sites or for DSSs (or other corrosion mitigation measures)

- operational controls to maintain cladding temperature limits

- low-temperature ductility of ferritic steels

- testing of design features and procedures that are significant to radiation protection and environmental releases

- minimum distance between loaded storage containers and construction activities that would disturb (remove, or expose) materials relied on for shielding for the loaded container

- requirements for active systems that may be used to ensure safety performance of the storage container (e.g., active corrosion protection system for the storage configuration, active supplemental cooling system during transfer operations)

All disciplines should coordinate their review of the proposed technical specifications to ensure the operational limitations are measurable and inspectable. Other topics may include the following:

- frequency and scope proposed for the surveillance requirements
- administrative controls that include organization and administrative systems and procedures, recordkeeping, review, and audit systems required to ensure that the DSS or DSF is managed in a safe and reliable manner, not already required by regulation
- action(s) that must be taken in the event of noncompliance with a limit or condition

Identify any additional technical specifications deemed necessary using the recommended format from RG 3.62 and RG 3.48.

For a DSS, NUREG-1745 provides a recommended format for applicants to present proposed technical specifications and certificate conditions. However, this format may not be applicable to all technical specifications. Since the basis for a technical specification may be extensively discussed in earlier chapters of the SAR, the applicant may use an abbreviated format of the basis discussion in the technical specifications chapter of the SAR.

Ensure that all necessary technical specifications are explicitly delineated in the SER chapter on the technical specifications and in the technical specifications accompanying the DSF facility license or DSS CoC. These delineations typically restate the technical specifications defined in the SAR but may be modified or supplemented, as the staff deems appropriate. Ensure that the SER and technical specifications clearly identify and document the code alternatives the applicant requested. The staff may prepare a separate table or appendix to the SER, as needed, to explicitly designate the technical specifications that are applicable to the DSS or DSF.

This evaluation is based on information that the applicant presented in the SAR chapter on technical specifications; accepted practices; and the applicant's analyses, design, and operations descriptions discussed in the SAR or in correspondence with the NRC subsequent to submission of the application. Describe in the SER any additional operating controls and limits that are deemed necessary and add, as appropriate, to the DSS's or DSF's CoC or license conditions or accompanying technical specifications.

17.6 Evaluation Findings

The NRC reviewer should prepare evaluation findings on satisfaction of the regulatory requirements in Section 17.4 of this SRP. This section also lists evaluation findings developed or included in all SER sections relating to technical specifications. With respect to a DSF, the findings should cover the whole facility, including specifications related to the proposed storage of any HLW (MRS only) and reactor-related GTCC waste at the facility. In addition, the findings should include a listing of any additional technical specifications that the NRC staff identified as necessary (beyond those identified by the applicant). If the documentation submitted with the application fully supports positive findings for each of the regulatory requirements, the statements of finding should be similar to the following:

F17.1 The staff concludes that the conditions for [DSS/DSF name] identify necessary technical specifications to satisfy 10 CFR Part 72 and that the applicable acceptance criteria have been satisfied.

F17.2 [if applicable] In addition to the applicant’s proposed technical specification(s), the staff finds that the technical specification(s) added by the NRC is/are required for safe operation.

The reviewer should provide a summary statement similar to the following:

The applicant’s proposed technical specifications and the technical specifications specified by the NRC provide reasonable assurance that the DSS or DSF will allow for the safe storage of spent fuel, and (as applicable for the (list site specific license)) reactor-related GTCC waste and HLW. This finding is reached on the basis of a review that considered the regulation itself, appropriate regulatory guides, applicable codes and standards, accepted practices, and the statements and representations in the application.

17.7 References

10 CFR Part 72, “Licensing Requirements for the Independent Storage of Spent Nuclear Fuel, High-Level Radioactive Waste, and Reactor-Related Greater Than Class C Waste.”

American Society of Mechanical Engineers (ASME), Boiler and Pressure (B&PV) Code, 2007 –Addenda 2008.
Section III, “Rules for Construction of Nuclear Facility Components.”

NUREG-1745, “Standard Format and Content for Technical Specifications for 10 CFR Part 72 Cask Certificates of Compliance,” issued June 2001 (Agencywide Documents Access and Management System Accession No. ML011940387).

Regulatory Guide 3.48, "Standard Format and Content for the Safety Analysis Report for an Independent Spent Fuel Storage Installation or Monitored Retrievable Storage Installation (Dry Storage)."

Regulatory Guide 3.61, “Standard Format and Content for a Topical Safety Analysis Report for a Spent Fuel Dry Storage Cask.”

Regulatory Guide 3.62, “Standard Format and Content for the Safety Analysis Report for Onsite Storage of Spent Fuel Storage Casks.”

APPENDIX A INTERIM STAFF GUIDANCE (ISG) INCORPORATED INTO NUREG-2215

ISG # & Rev.	Title	NUREG-2215 Status
ISG 1 Rev. 2	Damaged Fuel	Added
ISG 2 Rev. 2	Fuel Retrievability	Added
ISG 3	Post Accident Recovery and Compliance with 10 CFR 72.122(l)	Added
ISG 4 Rev. 1	Cask Closure Weld Inspections	Superseded by ISGs 15 and 18
ISG 5 Rev. 1	Confinement Evaluation	Added
ISG 6	Establishing Minimum Initial Enrichment for the Bounding Design Basis Fuel Assembly(s)	Added
ISG 7	Potential Generic Issue Concerning Cask Heat Transfer in a Transportation Accident	Added
ISG 8 Rev. 3	Burnup Credit in the Criticality Safety Analyses of PWR Spent Fuel in Transport and Storage Casks	Added
ISG 9 Rev. 1	Storage of Components Associated with Fuel Assemblies	Added
ISG 10 Rev. 1	Alternatives to the ASME Code	Added
ISG 11 Rev. 3	Cladding Considerations for the Transportation and Storage of Spent Fuel	Added
ISG 12 Rev. 1	Buckling of Irradiated Fuel Under Bottom End Drop Conditions	Added
ISG 13	Real Individual	Added
ISG 14	Supplemental Shielding	Added
ISG 15	Materials Evaluation	Added
ISG 16	Emergency Planning	Added
ISG 17	Interim Storage of Greater Than Class C Waste	Added
ISG 18 Rev. 1	The Design & Testing of Lid Welds on Austenitic Stainless Steel Canisters as Confinement Boundary for Spent Fuel Storage	Added
ISG 19	Moderator Exclusion Under Hypothetical Accident Conditions and Demonstrating Subcriticality of Spent Fuel Under the Requirements of 10 CFR 71.55(e)	NA
ISG 20	Transportation Package Design Changes Authorized Under 10 CFR Part 71 Without Prior NRC Approval	NA
ISG 21	Use of Computational Modeling Software	Added

ISG # & Rev.	Title	NUREG-2215 Status
ISG 22	Potential Rod Splitting Due to Exposure to an Oxidizing Atmosphere During Short-Term Cask Loading Operations in LWR or Other Uranium Oxide Based Fuel	Added
ISG 23	Application of ASTM Standard Practice C1671-07 When Performing Technical Reviews of Spent Fuel Storage and Transportation Packaging Licensing Actions	Added
ISG 24	Reserved	N/A
ISG 25	Pressure and Helium Leakage Testing of the Confinement Boundary of Spent Fuel Storage Casks	Added
ISG 26 (Draft)	Reserved	N/A

NRC FORM 335 (12-2010) NRCMD 3.7 U.S. NUCLEAR REGULATORY COMMISSION **BIBLIOGRAPHIC DATA SHEET** *(See instructions on the reverse)*	1. REPORT NUMBER (Assigned by NRC, Add Vol., Supp., Rev., and Addendum Numbers, if any.) **NUREG-2215**
2. TITLE AND SUBTITLE **Standard Review Plan for Spent Fuel Dry Storage Systems and Facilities** Final Report	3. DATE REPORT PUBLISHED MONTH: **April** YEAR: **2020** 4. FIN OR GRANT NUMBER
5. AUTHOR(S)	6. TYPE OF REPORT Technical 7. PERIOD COVERED (Inclusive Dates)

8. PERFORMING ORGANIZATION - NAME AND ADDRESS (If NRC, provide Division, Office or Region, U. S. Nuclear Regulatory Commission, and mailing address; if contractor, provide name and mailing address.)

9. SPONSORING ORGANIZATION - NAME AND ADDRESS (If NRC, type "Same as above", if contractor, provide NRC Division, Office or Region, U. S. Nuclear Regulatory Commission, and mailing address.)

Office of Nuclear Material Safety and Safeguards
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555-0001

10. SUPPLEMENTARY NOTES

11. ABSTRACT (200 words or less)

Abstract

This Standard Review Plan (SRP) provides guidance to the U.S. Nuclear Regulatory Commission (NRC) staff for reviewing safety analysis reports (SARs) for (1) a Certificate of Compliance (CoC) for a dry storage system for use at a general license facility and (2) a specific license for a dry storage facility that is either an independent spent fuel storage installation (ISFSI) or a monitored retrievable storage installation (MRS). This SRP does not apply to wet storage ISFSIs or MRSs (e.g., GE-Morris).

The objectives of this SRP are to assist the NRC staff in its reviews by doing the following:

- promoting a consistent regulatory review of a SAR for an ISFSI or MRS license, or for a CoC
- promoting quality and uniformity of these reviews across each technical discipline
- presenting a basis for the review's scope
- identifying acceptable approaches to meeting regulatory requirements
- suggesting possible evaluation findings that can be used in the safety evaluation report

This SRP may be revised and updated as the need arises on a chapter-by-chapter basis to clarify the content, correct errors, or incorporate modifications approved by the Director of the Division of Spent Fuel Management. Comments, suggestions for improvement, and notices of errors or omissions should be sent to and will be considered by the Director, Division of Spent Fuel Management, Office of Nuclear Material Safety and Safeguards, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001.

12. KEY WORDS/DESCRIPTORS (List words or phrases that will assist researchers in locating the report.)	13. AVAILABILITY STATEMENT
List of keywords or phrases that will assist researchers in locating this NUREG. standard review plan, spent nuclear fuel, spent fuel storage, independent spent fuel storage installation, specific license application, certificate of compliance application, monitored retrievable storage installation, dry storage system	unlimited 14. SECURITY CLASSIFICATION *(This Page)* unclassified *(This Report)* unclassified 15. NUMBER OF PAGES 16. PRICE

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, DC 20555-0001

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NUREG-2215
Final

Standard Review Plan for Spent Fuel Dry Storage Systems and Facilities

April 2020