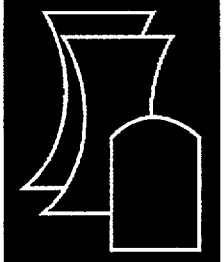

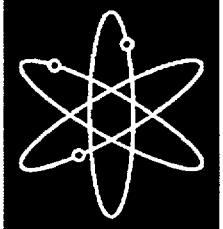

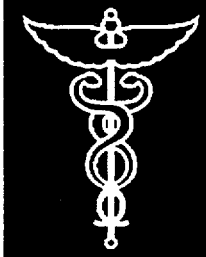

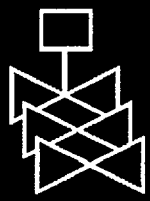

Standard Review Plan for the Review of a License Application for the Tank Waste Remediation System Privatization (TWRS-P) Project

Final Report

U.S. Nuclear Regulatory Commission
Office of Nuclear Material Safety and Safeguards
Washington, DC 20555-0001

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Standard Review Plan for the Review of a License Application for the Tank Waste Remediation System Privatization (TWRS-P) Project

Final Report

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Tank Waste Remediation System Section
Special Projects Branch

Division of Fuel Cycle Safety and Safeguards
Office of Nuclear Material Safety and Safeguards
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

ABSTRACT

This NUREG provides guidance to the NRC staff reviewers in the Office of Nuclear Material Safety and Safeguards for the performance of safety and environmental reviews of a Tank Waste Remediation System (TWRS) Facility under 10 CFR Part 70[1], as revised. The standard review plan (SRP) presented in this NUREG ensures the quality, uniformity, stability, and predictability of staff reviews. It presents a defined basis from which to evaluate proposed changes in the scope and requirements of the staff reviews. The SRP makes information about review acceptance criteria readily available to interested members of the public and the regulated industry. Each SRP section addresses the regulations pertinent to specific technical matters, the acceptance criteria used by the staff, how the review is accomplished, and the conclusions that are appropriate for the Safety Evaluation Report (SER).

[1] Nuclear Regulatory Commission (U.S.) Washington, D.C. "Domestic Licensing of Special Nuclear Material (10 CFR Part 70)." *Federal Register:* Vol. 64, No. 146. pp. 41338--41357. July 30, 1999.

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INTRODUCTION

The "Standard Review Plan for the Review of a License Application for the Tank Waste Remediation System Privatization (TWRS-P) Project]" provides U.S. Nuclear Regulatory Commission (NRC) guidance for the review and evaluation of health, safety, and environmental protection in applications for licenses for remediation of radioactive tank waste at Hanford. The guidance is also applicable to the review and evaluation of proposed amendments and license renewal applications. Specific filing requirements for license applications, and for issuance of such licenses, are in 10 CFR Part 70, "Domestic Licensing of Special Nuclear Material," as revised[1]. Although 10 CFR Part 70, as revised, does not specifically include a TWRS-P facility in § 70.60, "Applicability," the regulation specifies applicable facilities include, "any other activity that the Commission determines could significantly affect public health and safety."

The principal purpose of the Standard Review Plan (SRP) is to ensure the quality and uniformity of staff reviews and to present a well-defined base from which to evaluate proposed changes in the scope, level of detail, and acceptance criteria of reviews. The SRP also should be used as the basis for the review of requests by licensees for changes in their licenses. Thus, the SRP, at any point in time, can provide a basis for the review of proposed new or renewal applications, and amendments to existing licenses, as well as modifications to the SRP resulting from new NRC requirements and licensee initiatives.

Another important purpose of the SRP is to make information about regulatory reviews widely available and to improve communication and understanding of the staff review process. Because the SRP describes the scope, level of detail, and acceptance criteria for reviewers, it can serve as regulatory guidance for applicants who need to determine what information should be presented in a license application.

The responsibility of the staff in the review of a license application, renewal application, or license amendment for a TWRS-P facility is to determine that there is reasonable assurance that the facility can and will be operated in a manner that will not be inimical to the common defense and security, and will provide reasonable protection of the health and safety of workers and the public, and the environment. To carry out this responsibility, the staff evaluates information provided by an applicant and through independent assessments determines that the applicant has demonstrated a reasonable safety program that is in accordance with regulatory requirements. To facilitate carrying out this responsibility, the SRP clearly states and identifies those standards, criteria, and bases that the staff should use in reaching licensing decisions.

The staff believes that a TWRS-P facility is an activity that could significantly affect public health and safety, and therefore plans to invoke the requirements found in Subpart H of 10 CFR Part 70 for this type of facility. As such, 10 CFR Part 70, as revised, require that an applicant submit a complete description of the safety program for the possession and use of

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[1] Nuclear Regulatory Commission (U.S.), Washington, D.C. "Domestic Licensing of Special Nuclear Material (10 CFR Part 70)." *Federal Register:* Vol. 64, No. 146. pp. 41338--41357. July 30, 1999.

SNM to show how compliance with the applicable requirements will be accomplished. The Safety Program Description must be sufficiently detailed to permit the staff to obtain reasonable assurance that the facility is designed and will be operated without undue risk to the health and safety of workers or the public. Prior to submission of the program description, an applicant should have analyzed the facility in sufficient detail to conclude that it is designed and can be operated safely. The Safety Program Description is the principal document with which the applicant provides the information needed by staff to understand the basis for conclusion. When reviewed and approved by the staff, and incorporated in the NRC license by reference, the Safety Program Description, in its entirety and in its parts, is the safety basis on which the license is issued, and may not be changed except through the process defined in 10 CFR 70.72.

The requirements in 10 CFR Part 70 specify, in general terms, the information to be supplied in a Safety Program Description. The specific information to be submitted by an applicant and evaluated by staff is identified in this SRP. Prospective applicants should study the topic areas treated in this document (generally, chapter headings) and the subsections within each topic area, specifically the subsections headed "Areas of Review" and "Acceptance Criteria." A license application should contain a Safety Program Description that addresses all the topics in the Table of Contents of this SRP, in the same order as presented in this document.

In this SRP, information is provided to assist the licensing staff and the applicant in understanding the underlying objective of the regulatory requirements, the relationships among NRC requirements, the licensing process, the major guidance documents NRC staff has prepared for licensing facilities under 10 CFR Part 70, and the details of the staff review process set out in individual SRP sections. Analyses by the staff are intended to provide regulatory confirmation of reasonable assurance of safe design and operation. A staff determination of reasonable assurance leads to a decision to issue or renew a license or to approve an amendment. In the case of a staff determination of inadequate description or commitments, the staff should inform the applicant of what is needed and the basis upon which the determination was made.

The "Acceptance Criteria" delineated in this SRP are intended to communicate the underlying objectives but not to represent the only means of satisfying that objective. An applicant should tailor its safety program to the features of its particular facility. If approaches different from the SRP are chosen, the applicant should identify the portions of its application that differ from the design approaches and acceptance criteria of the SRP and evaluate how the proposed alternatives provide an acceptable method of complying with the Commission's regulations. The staff retains the responsibility to make an independent determination of the adequacy of what is proposed.

The major topics addressed within the Safety Program Description of a facility license application are addressed in separate SRP sections; each of those sections, or chapters, includes subsections described below.

The applicant's integrated safety analysis (ISA) is the central focus for the selection of design and operational safety measures and the management control systems that assure the availability and reliability of those measures. It is the ISA that provides a comprehensive evaluation and presentation, useful to both the applicant and the NRC, of the distribution of

risk among the many activities ongoing at the TWRS-P facility. The NRC expects to be able to use the ISA summary to focus its resources on the dominant risks of facility design and operation and the safety controls and assurances necessary to ensure that those controls remain available and reliable. Accordingly, staff reviewers should conduct a coordinated review of the ISA summary and focus on the portions of the summary that are applicable to each of the technical areas treated in the chapters of the SRP. The acceptance criteria in each of the SRP chapters are the criteria that apply to the dominant risks of operation. The applicant has the opportunity to justify lesser criteria for those design and operational features that can be shown to represent lesser risk than the accident or failure sequences that pose the dominant risks.

While recognizing the fundamental importance of the ISA to understanding the risk at a facility, certain SRP chapters are less dependent on ISA outcomes than others. The chapters concerning radiation safety, environmental protection, emergency management, and decommissioning, for example, contain acceptance criteria that are set primarily by existing regulations and will not be affected by issuing the revision to 10 CFR Part 70. Finally, for new facilities (that have not already been designed, built, licensed and operated), certain baseline design criteria have been specified in 10 CFR Part 70.64, "Requirements for New Facilities or New Processes at Existing Facilities." These criteria identify safety considerations that an applicant must address in its facility design. The ISA for the complete facility design may indicate when reduced levels of assurance may be acceptable. A more detailed description of the application of these criteria is given in the discussion of Section 4, "Acceptance Criteria" below.

Section 1. PURPOSE OF REVIEW

This section is a brief statement of the purpose for and objectives of reviewing the subject areas. It emphasizes the staff's evaluation of the ways the applicant can achieve identified performance objectives and ensures through the review that the applicant has used a multi-disciplinary, systems-oriented approach to establishing designs, controls, and procedures within individual technical areas.

Section 2. RESPONSIBILITY FOR REVIEW

This section identifies the organization and individuals by function, within NRC, responsible for evaluating the subject or functional area covered by the SRP. If reviewers with expertise in other areas are to participate in the evaluation, they are identified by function. In general, the Licensing Project Manager has responsibility for the total review product, a safety evaluation report for an application. However, an identified technical specialist should have primary responsibility for a particular review topic, usually an SRP chapter. One or more specialists may have supporting responsibility. In most situations the review is performed by a team of specialist reviewers including the lead reviewer for the ISA and the project manager. Although they individually perform their review tasks, the reviews are extensively coordinated and integrated to ensure consistency in approach and to ensure risk-informed reviews. The project manager oversees and directs the coordination of the reviewers. The reviewers' immediate line management has the responsibility to ensure that an adequate review is performed by qualified reviewers.

Section 3. AREAS OF REVIEW

This section describes the topics, functions, systems, structures, equipment, and components, analyses, data, or other information that should be reviewed as part of that particular subject area of the license application. Because the section identifies information to be reviewed in evaluating the adequacy of the application, it identifies the acceptable content of an applicant's submittal in the areas discussed. The areas of review identified in this section obviate the need for a separate Standard Format and Content Guide.

The topics identified in this section also set the content of the next two sections of the SRP. Both Section 4, "Acceptance Criteria," and Section 5, "Review Procedures," should address, in the same order, the topics set forth in this section as areas to be reviewed. This section also identifies the information needed or the review expected from other NRC individuals to permit the individual charged with primary review responsibility to complete the review.

Section 4. ACCEPTANCE CRITERIA

This section contains a statement of the applicable NRC criteria based on regulatory requirements, and the bases for determining the acceptability of the applicant's commitments relative to the design, programs, or functions within the scope of the particular SRP section. Technical bases consist of specific criteria such as NRC regulations, regulatory guides, NUREG reports, industry codes and standards, and branch technical positions. To the extent practicable, the acceptance criteria identify, as objectively or quantitatively as is feasible, specific criteria and other technical bases that are to be satisfied. The acceptance criteria (including branch technical positions or other information) present positions and approaches that are acceptable to the staff. They are not considered the only acceptable positions or approaches. Others may be proposed by an applicant.

It is NRC's intent that the SRP present acceptance criteria for each technical function area (e.g., nuclear criticality safety, fire safety, radiation safety), and for the management measures (e.g., quality assurance, maintenance, audits and assessments), that allow an applicant to provide a level of protection commensurate with the accident risk inherent in the process activities proposed. For example, at process stations (or for an entire process or sub-process) for which the inherent risk to workers, the public, or the environment is demonstrably small, the applicant needs to provide only those design and operating controls which assure that small risk. The key element in the regulatory transaction involving presentation by an applicant, and review and approval by the NRC, is an adequate demonstration of acceptable control of risk by the applicant, which then supports a competent and informed review by NRC staff. The starting point for the applicant's demonstration of acceptable control of risk is the ISA.

The applicant's ISA summary (described in and reviewed under Chapter 3 of this SRP) is the primary supporting rationale for the safety level of design and operational features. There are, however, design and operational features and management controls that may be required independent of the ISA results presented by an applicant. This is to meet the requirements of 10 CFR Part 70.64, for new facilities or new processes at existing facilities, or, for all facilities, other NRC requirements such as 10 CFR Parts 20 and 51. The level of detail presented in the ISA summary and in other parts of the application represents the safety basis committed to by the applicant, and it is that basis which is subject to the provisions of 10 CFR Part 70, as

revised, regarding changes that a licensee may make to the facility without prior NRC approval.

NRC should find an application acceptable if an applicant commits to the design features and management measures defined by the acceptance criteria within this SRP. The criteria in this SRP represent the design features or management measures that support an NRC finding of reasonable assurance of adequate protection, independent of any ISA findings or conclusions that could lead to NRC approval of reduced levels of assurance for certain design features or management measures where the associated risk does not warrant the same high level of assurance.

An applicant for license renewal or an amendment for an existing facility responding to the requirements of 10 CFR Part 70, may propose structures, systems, and components (SSC) or management measures that meet less stringent acceptance criteria than described in the SRP based on supporting analyses from the applicant's ISA. The ISA may be used to justify a reduced level of assurance for particular items relied on for safety, that are associated with lesser risk accident sequences, as defined by the applicant's analysis of likelihood and consequences pursuant to 10 CFR Part 70, as revised. The SRP criteria shown in this SRP apply to those SSC and management measures that are involved in the higher risk accident sequences as defined in Part 70, as revised.

For proposed new facilities or amendments for new processes proposed at existing facilities, the acceptance criteria described in the SRP apply for design purposes and should be addressed in the applicant's licensing submittal for all SSC and management measures and that section's requirement to comply with the baseline design criteria (BDCs) of Part 70, as revised. The BDCs are consistent with risk-informed regulation, in that, for new processes or new facilities, NRC recognizes that good engineering practice dictates certain minimum requirements be applied as design and safety considerations, generally independent of the risk-based information ultimately obtained through the ISA. However, the applicant may use the ISA summary to justify reduced criteria for some SSC and management measures consistent with ISA summary for a facility final design. Proposed reductions in the level of assurance should be considered by the NRC staff, and, if accepted, should also constitute compliance with the BDCs.

Section 5. REVIEW PROCEDURES

This section describes how the review should be performed. It describes procedures that the reviewer should follow to achieve an acceptable scope and depth of review and to obtain reasonable assurance that the applicant has provided appropriate commitments to ensure that it will operate the facility safely. This includes identifying licensee commitments to verify and could include directing the reviewer to coordinate with others having review responsibilities for other portions of the application than that assigned to the reviewer. This section should provide whatever procedural guidance is necessary to evaluate the applicant's level of achievement of the acceptance criteria.

Section 6. EVALUATION FINDINGS

This section presents the type of positive conclusion that is sought for the particular review area to support a decision to grant a license or amendment. The review must be adequate to permit the reviewer to support this conclusion. For each section, a conclusion of this type should be included in the staff's Safety Evaluation Report (SER) in which the staff publishes the results of its review. The SER should also contain a description of the review, including aspects of the review that received special emphasis; matters that were modified by the applicant during the review; matters that require additional information or will be resolved in the future; aspects where the plant's design or the applicant's proposals deviate from the criteria in the SRP; and the bases for any deviations from the SRP or proposed exemptions from the regulations. Staff reviews may be documented in the form of draft SERs that identify open issues requiring resolution before the staff can make a positive finding in favor of the license issuance or amendment.

Section 7. REFERENCES

This section lists references that should be consulted in the review process. However, they may not always be relevant to the review, depending on the action and approaches proposed by the applicant.

ACRONYMS

AEGL	Acute Exposure Guideline Level
ALARA	As Low As is Reasonably Achievable
BDC	Baseline Design Criteria
BTP	Branch Technical Position
CM	Configuration Management
DAC	Derived Air Concentration
DFP	Decommissioning Funding Plan
DP	Decommissioning Plan
EA	Environmental Assessment
EALs	Emergency Action Levels
EIS	Environmental Impact Statement
ERPG	Emergency Response Planning Guide
FEMA	Federal Emergency Management Agency :T
FHA	Fire Hazards Analysis
FONSI	Finding of No Significant Impact
HEPA	High-Efficiency Particulate Air
HFE	Human Factors Engineering
H&S	Health and Safety
HS&E	Health Safety and Environmental Protection
HSI	Human-System Interface
I&C	Instrumentation and Control
ISA	Integrated Safety Analysis
MDC	Minium Detectable Concentration
MOU	Memorandum of Understanding

NCS	Nuclear Criticality Safety
NIST	National Institute of Standards and Technology
NEPA	National Environmental Policy Act
OER	Operating Experience Review
OSHA	Occupational Safety and Health Administration
P&ID	Piping and Instrumentation Diagram
PHA	Process Hazard Analysis
PM	Preventive Maintenance
PPE	Personal Protection Equipment
PSI	Process Safety Information
QA	Quality Assurance
QC	Quality Control
RG	Regulatory Guide
RS	Radiation Safety
RSM	Radiation Safety Manager
RWP	Radiation Worker Permit
SER	Safety Evaluation Report
SNM	Special Nuclear Material
SRP	Standard Review Plan
SSCs	Structures, Systems and Components
TEDE	Total Effective Dose Equivalent
TID	Tamper-Indicating Device
UL	Underwriter Laboratories
V&V	Verification and Validations
PSE	Planned Special Exposures :T

GENERAL INFORMATION

1.1 FACILITY AND PROCESS DESCRIPTION

1.1.1 PURPOSE OF REVIEW

The purpose of this review is to establish that the license application (or application for license renewal or amendment) includes an overview of the applicant's facility layout and a summary description of its manufacturing processes. This overview should be of sufficient detail to be used by all reviewers, NRC managers, and the general public to understand the purpose of the facility and its processes. A more detailed description of the facility and its manufacturing processes should be contained in the applicant's integrated safety analysis (ISA) summary.

1.1.2 RESPONSIBILITY FOR REVIEW

Primary: Licensing Project Manager

Secondary: TWRS Site Representative

Supporting: None

1.1.3 AREAS OF REVIEW

The staff should review the general facility description and process descriptions provided in the application. These descriptions should include (1) scaled drawings showing the locations of facility buildings and other major structures, hazardous materials storage areas, on-site roadways, railroad spurs or sidings, and major ingress and egress routes for the site, (2) the proximity of the facility buildings to the site boundary, nearby populations, and within the facility controlled area, (3) a text index with titles that are descriptive of the purpose of each feature, (4) the interrelationships of the features, (5) the relationship of facility features to site features, and (6) a narrative description of the flow of licensed material through the facilities manufacturing process. This information should be consistent with that presented in the applicant's ISA summary and with the applicant's information provided in response to the Environmental Protection and Emergency Management chapters of this SRP.

1.1.4 ACCEPTANCE CRITERIA

1.1.4.1 Regulatory Requirements

The regulations applicable to the areas of review in this SRP are 10 CFR 70.22, "Contents of Applications," § 70.60, "Applicability," § 70.61(f), "Performance Requirements."

1.1.4.2 Regulatory Guidance

There are no regulatory guides that apply to a general facility description for a new facility licensed under 10 CFR Part 70.

1.1.4.3 Regulatory Acceptance Criteria

The applicant's presentation of general information should be considered acceptable if the applicant meets the following Acceptance Criteria:

1. The application presents the facility and process description at a level of detail appropriate to understand the purpose of the facility and processes and the general layout of the facility.

2. The application presents a summary of the facility information contained in the applicant's ISA Summary. This includes descriptions of the overall plant layout on scaled drawings, including site geographical features, and plant structural features such as buildings, towers, and tanks and transportation right of ways. The relationship of specific facility features to the major processes that will be ongoing at the facility is described.

3. The major chemical or mechanical processes involving special nuclear material (SNM) to be licensed are described in summary form and the summary is consistent with information contained in the applicant's ISA Summary. This description should include reference to the building locations of major components of the processes; brief descriptions of the process steps; the chemical forms of SNM in process; the maximum amounts of SNM in process expected in various building locations;, and the types, amounts, and discharge points of waste materials discharged to the environment from the processes.

1.1.5 REVIEW PROCEDURES

1.1.5.1 Acceptance Review

The primary reviewer should evaluate the application to determine whether it addresses the "Areas of Review" discussed in Section 1.1.3. If significant deficiencies are identified, the applicant should be requested to submit additional material before the start of the safety evaluation.

1.1.5.2 Safety Evaluation

After the application is determined to be acceptable for review in accordance with Section 1.1.5.1, the primary reviewer should perform a safety evaluation against the regulatory acceptance criteria described in Section 1.1.4.3. The material to be reviewed is informational in nature, and no technical analysis should be required. The information provided in this section is only used as background for the more detailed descriptions in later sections of the

application. Therefore, the primary reviewer should only confirm that the descriptive information presented is of sufficient detail to adequately understand the applicant's layout and processes and that the information is consistent with the information presented in the applicant's ISA Summary. The primary reviewer should document the evaluation as described in Section 1.1.6.

The TWRS Site Representative should also confirm that the information presented in this section is consistent with the as-built facilities (for existing facilities) and current operational practices. The TWRS Site Representative should report any findings to the primary reviewer.

1.1.6 EVALUATION FINDINGS

After completing the evaluation of material in response to this section, the primary reviewer should write an SER section addressing each topic reviewed under this SRP Section and explain why the NRC staff has reasonable assurance that the facility and process description is acceptable. The SER section should include a summary statement of what was evaluated and the basis for the reviewers' conclusions. License conditions may also be proposed in this SER section to impose requirements where the application is deficient.

The staff can document the evaluation as follows:

> *The staff has reviewed the general facility description for [name of facility] according to the Standard Review Plan Section 1.1. [Insert a summary statement of what was evaluated and why the reviewer finds the submittal acceptable.] The applicant has adequately described (1) the facility and processes so that the staff has an overall understanding of the relationships of the facility features and (2) the function of each feature. The applicant has cross-referenced its general description with the more detailed descriptions elsewhere in the application. The staff concludes that the applicant has complied with the general requirements to provide general information in the application in accordance with 10 CFR 70.22, "Contents of Applications," § 70.60, "Applicability," and § 70.61(f), "Performance Requirements."*

1.1.7 REFERENCES

1. Code of Federal Regulations, *Title 10, Energy*, Part 70, "Domestic Licensing of Special Nuclear Material."

2. Nuclear Regulatory Commission (U.S.), Washington, D.C. "Domestic Licensing of Special Nuclear Material (10 CFR Part 70)." *Federal Register:* Vol. 64, No. 146. pp. 41338–41357. July 30, 1999.

GENERAL INFORMATION
1.2 INSTITUTIONAL INFORMATION AND AUTHORIZED USE

1.2.1 PURPOSE OF REVIEW

The purpose of this review is to establish that the license application includes adequate institutional information concerning the applicant, and information concerning the proposed activities at the applicant's facility.

1.2.2 RESPONSIBILITY FOR REVIEW

Primary: Licensing Project Manager

Secondary: None

Supporting: Office of the General Counsel; Office of Administration/Division of Facilities and Security

1.2.3 AREAS OF REVIEW

Information provided for review should include the identity and address of the applicant's facility and corporate headquarters; corporate information sufficient to show the relationship of the applicant's organization relative to other corporate entities; the existence and extent of foreign ownership or influence; financial information sufficient to indicate the resources available to the applicant to pursue the activities for which the license is sought; the site location as legally described in land records; a description of each proposed licensed activity in the form of requested authorized uses; the type and term of license being applied for; and the type, quantity, and form(s) of material(s) proposed to be used at the licensed facility.

Note: It is expected that most operations of the type reviewed under this SRP will be associated with Department of Energy (DOE) facilities and have financial guarantees for operation and decommissioning form DOE. As such, the applicant's financial viability and guarding of national security information would be expected to be completed before DOE's selection of the applicant as a contractor and therefore not be reviewed by NRC.

1.2.4 ACCEPTANCE CRITERIA

1.2.4.1 Regulatory Requirements

The regulations applicable to the areas of review in this SRP are 10 CFR 70.22, "Contents of Applications," § 70.23, "Requirements for the Approval of Applications," § 70.33, "Renewal of Licenses," and 10 CFR Part 95, "Security Facility Approval and Safeguarding of National Security Information and Restricted Data."

1.2.4.2 Regulatory Guidance

There are no regulatory guides that apply to institutional information for facilities licensed under 10 CFR Part 70.

1.2.4.3 Regulatory Acceptance Criteria

The application should be acceptable if the following criteria are met:

1. Corporate Identity

   The applicant has furnished its full name and address. The address of the facility is provided if it is different from that of the applicant. If the application is for renewal, the applicant identifies the number of the license to be renewed. A full description of the plant site location (State, county, and municipality) is given. The State where the applicant is incorporated or organized and the location of the principal office are indicated. If the applicant is a corporation or other entity, the names and citizenship of its principal officers are provided. The entity to be licensed is clearly described with respect to any higher level related corporate structure. The description clearly identifies and explains any proposed foreign ownership or control of activities. Primary ownership and relationships to other components of the same ownership are explicitly described. The presence and operations of any other company on the site to be licensed are fully described.

2. Financial Qualifications

   As the applicant for this type of facility should be guaranteed by DOE, the applicant only needs to describe the relationship with DOE.

   If the applicant is not a DOE contractor, then a description of financial qualifications demonstrates the applicant's current and continuing access to the financial resources necessary to engage in the proposed activity in accordance with the regulations within 10 CFR Part 70. A reference to the appropriate sections of the applicant's decommissioning funding plan may be acceptable.

3. Type, Quantity, and Form of Licensed Material

   The elemental name, maximum quantity, and specifications, including the chemical and physical form(s), of the special nuclear material the applicant proposes to acquire, deliver, receive, possess, use, transfer or store are identified. The specifications should include the isotopic content and maximum enrichment by weight percent. In addition, fission products and other or transuranics are characterized by identity and maximum concentration.

4. Authorized Uses

   Each activity or process in which special nuclear material is proposed to be acquired, delivered, received, possessed, used, processed, transferred, or stored is described in a general manner. The authorized uses must be consistent with the Atomic Energy Act of 1954, et seq. The description should be verified to be consistent with more detailed process descriptions submitted as part of the ISA summary reviewed under Chapter 3.0 of this SRP.

5. Licensing Term

   The applicant should state the period of time for which the license is requested.

   If the application is for a renewal, the applicant states the period of time for which license renewal is requested, and why the renewal application should be considered timely in accordance with 10 CFR Part 70.

6. Special Exemptions or Special Authorizations

   Specific requests for exemptions or unusual authorizations should be listed in this section and justified in the appropriate technical section of the application.

7. Security of Classified Information

   As a contractor to DOE, the applicant's security programs to protect classified information is expected to be handled by DOE. As such, the applicant should describe the agreements with DOE.

   If this is not the case, the applicant should show that they have received a facility security clearance in accordance with 10 CFR Part 95, as necessary.

1.2.5 REVIEW PROCEDURES

1.2.5.1 Acceptance Review

The primary reviewer should evaluate the application to determine whether it addresses the "Areas of Review" discussed in Section 1.2.3, above. If significant deficiencies are identified, the applicant should be requested to submit additional material before the start of the safety evaluation.

1.2.5.2 Safety Evaluation

After determining that the application is acceptable for review in accordance with Section 1.2.5.1, above, the primary reviewer should perform a safety evaluation against the acceptance criteria described in Section 1.2.4.3. The material to be reviewed is for the most

part informational in nature; a detailed technical analysis is generally not required beyond verifying that information in the acceptance criterion is included in the application. The reviewer should request review assistance, if needed, from the Division of Facilities and Security and the Office of the General Counsel in the review of corporate and financial information. The primary reviewer should document the evaluation as described in Section 1.2.6.

1.2.6 EVALUATION FINDINGS

After completing the evaluation of material in response to this section, the primary reviewer should write an SER section addressing each topic reviewed under this SRP section and explain why the NRC staff has reasonable assurance that the institutional information provided is acceptable. The SER section should include a summary statement of what was evaluated and the basis for the reviewers' conclusions. License conditions may also be proposed in this SER section to impose requirements where the application is deficient.

The staff can document the evaluation as follows:

> *The staff has reviewed the institutional information for [name of facility] according to Standard Review Plan Section 1.2.* [Insert a summary statement of what was evaluated and why the reviewer finds the submittal acceptable.] *Based on the review, the NRC staff has determined that the applicant has adequately described and documented the corporate structure and financial information, and that the applicant is in compliance with those parts of 10 CFR 70.22 and other sections of Part 70 related to other institutional information. In addition, the applicant has adequately described the types, forms, quantities, and proposed authorized uses of licensable materials to be permitted at this facility as follows:*
>
> | *Material* | *Form* | *Quantity* | *Authorized Use(s)* |
> |---|---|---|---|
>
> *The applicant's proposed activities are consistent with the Atomic Energy Act. The applicant has provided all institutional information necessary to understand the ownership, financial qualifications, location, planned activities, and nuclear materials to be handled in connection with the requested license.*

1.2.7 REFERENCES

Code of Federal Regulations, *Title 10, Energy*, Part 70, "Domestic Licensing of Special Nuclear Material."

GENERAL INFORMATION
1.3 SITE DESCRIPTION

1.3.1 PURPOSE OF REVIEW

The purpose of this review is to determine that the information provided by an applicant adequately describes the geographic, demographic, meteorologic, hydrologic, geologic, and seismologic characteristics of the site and the surrounding area. The site description is a summary of the information used by the applicant in preparing the Integrated Safety Analysis (ISA) summary, the Emergency Plan, and the Environmental Report as described in Chapters 3.0, 8.0, and 9.0, respectively, of this Standard Review Plan (SRP).

1.3.2 RESPONSIBILITY FOR REVIEW

Primary: Licensing Project Manager

Secondary: ISA (SRP Chapter 3.0) Reviewer, Environmental Protection (SRP Chapter 9.0) Reviewer, and Emergency Plan (SRP 8.0) Reviewer

Supporting: TWRS Site Representative

1.3.3 AREAS OF REVIEW

The information that the NRC staff should review includes the following:

1. Site Geography

   a. Site location: state, county, municipality, topographic quadrangle (71/2 minute series). Maps should clearly indicate the facility location and boundary, if scale permits.

   b. Major nearby highways.

   c. Nearby bodies of water.

   d. Any other significant geographic feature that may impact accident analysis within one mile of the site (e.g., ridges, valleys, specific geologic structures).

2. Demographics

   a. Latest census results for area of concern.

   b. Description, distance, and direction to nearby population centers.

c. Description, distance, and direction to nearby public facilities (e.g., schools, hospitals, parks).

d. Description, distance, and direction to nearby industrial areas or facilities that may present potential hazards (including other nearby nuclear facilities).

e. Uses of land within one mile of the facility (i.e., residential, industrial, commercial, agricultural).

f. Uses of nearby bodies of water.

3. Meteorology

a. Primary wind directions and average wind speeds.

b. Annual amount and forms of precipitation. The design basis values for accident analysis of maximum snow or ice load, probable maximum precipitation.

c. Type, frequency, and magnitude of severe weather (e.g., lightning, tornado, hurricane). Design basis event descriptions for accident analysis.

4. Hydrology

a. Characteristics of nearby rivers, streams, and bodies of water as appropriate.

b. Depth to the water table; potentiometric surface map.

c. Groundwater flow direction and velocity for the site.

d. Characteristics of the uppermost aquifer.

e. Design basis flood events used for accident analysis.

5. Geology

a. Characteristics of soil types and bedrock.

b. Design basis earthquake magnitudes used for accident analysis.

c. Description of other geologic hazards, e.g., mass wasting.

The above information summarizes and should be consistent with the site information presented in the ISA, the Emergency Plan, and the Environmental Report prepared by the applicant. In contrast to these more detailed descriptions, the summary site description reviewed under this section is expected to be briefer and less detailed.

1.3.4 ACCEPTANCE CRITERIA

1.3.4.1 Regulatory Requirements

The regulation applicable to the areas of review in this SRP section is 10 CFR 70.22, "Contents of Applications."

1.3.4.2 Regulatory Guidance

There are no regulatory guides that apply to the site description for a facility licensed under 10 CFR Part 70.

1.3.4.3 Regulatory Acceptance Criteria

The site description summary should be considered acceptable if it includes the following:

1. A brief description of the site geography, including its location relative to prominent natural and man-made features such as mountains, rivers, airports, population centers, schools, commercial and manufacturing facilities, etc.

2. Population information based on the most current available census data to show population distribution as a function of distance from the facility.

3. Appropriate meteorologic data, including design basis values for accident analysis of maximum snow or ice load, and probable maximum precipitation. The applicant should present appropriate design basis values for lightning, high winds, tornado, hurricane, and other severe weather conditions that are applicable to the site.

4. Appropriate "safe shutdown" hydrology, geology, and seismicity data, including the design basis flood event and the maximum earthquake magnitude and peak ground acceleration (and its expected likelihood, in terms of return period).

The applicant's descriptions should be consistent with the more detailed information presented within the ISA summary information in Chapter 3 of the application, the Environmental Report, and the Emergency Plan, if applicable.

1.3.5 REVIEW PROCEDURES

1.3.5.1 Acceptance Review

The primary reviewer should evaluate the application to determine whether it addresses the "Areas of Review" discussed in Section 1.3.3, above. If significant deficiencies are identified, the applicant should be requested to submit additional material before the start of the safety evaluation.

1.3.5.2 Safety Evaluation

After the application is determined to be acceptable for review in accordance with Section 1.3.5.1, above, the primary reviewer should perform a safety evaluation against the acceptance criteria described in Section 1.3.4.3. The material to be evaluated in this section is informational, summarizing the reports and information that provide the bases for the ISA evaluations. The secondary reviewers should verify that the information accurately portrays and is consistent with the information in the ISA summary, Environmental Report, Emergency Plan and other documents referenced by the applicant. No technical analysis is required, as the primary reference for the information is the ISA. If information being verified is found to be inconsistent from the primary source, the applicant should be requested to submit clarifying information or corrections. This section may also need to be updated by the applicant based upon any information changes made in response to the staff's environmental, emergency management, and ISA reviews. The primary reviewer, with input from the secondary reviewers, should document the evaluation as described in Section 1.3.6.

The TWRS Site Representative should confirm that the information presented in this section is consistent with the site.

1.3.6 EVALUATION FINDINGS

After completing the evaluation of material in response to this section, the primary reviewer should write an SER section that addresses each topic reviewed under this SRP section and that explains why the NRC staff has reasonable assurance that the site description is acceptable. The SER section should include a summary statement of what was evaluated and the basis for the reviewers' conclusions. License conditions may also be proposed in this SER section to impose requirements where the application is deficient.

The staff can document the evaluation as follows:

> *The staff has reviewed the site description for [name of facility] according to the Standard Review Plan Section 1.3.* [Insert a summary statement of what was evaluated and why the reviewer finds the submittal acceptable.] *The applicant has adequately described and summarized general information pertaining to (1) the site geography, including its location relative to prominent natural and man-made features such as mountains, rivers, airports, population centers, schools, and commercial and manufacturing facilities; (2) population information based on the most current available census data to show population distribution as a function of distance from the facility; (3) meteorology, hydrology, and geology for the site; and (4) applicable design basis events. The reviewers have verified the site description is consistent with the information used as a basis for the ISA, the Emergency Plan, and the Environmental Report.*

1.3.7 REFERENCE

1. Code of Federal Regulations, *Title 10, Energy*, Part 70, "Domestic Licensing of Special Nuclear Material."

ORGANIZATION AND ADMINISTRATION

2.1 PURPOSE OF REVIEW

The purpose of the review of the applicant's organization and administration is to verify that the applicant's administrative policies provide reasonable assurance that the licensee plans, implements, and controls site activities in a manner that ensure the safety of workers, the public, and the environment. The review should also confirm that the qualifications for key management positions are adequate.

2.2 RESPONSIBILITY FOR REVIEW

Primary: Licensing Project Manager

Secondary: None

Supporting: Primary reviewers for other SRP Chapters, e.g., technical area chapters and management measures; TWRS Site Representative

2.3 AREAS OF REVIEW

The organizational structure and associated administrative program proposed by the applicant, including administrative policies, procedures, and qualifications of key management positions, along with a description of how these are deemed adequate to provide reasonable assurance that the health, safety, and environmental protection (HS&E) functions will be effective, should be reviewed.

The review should address the integration of authorities and responsibilities among the process designers, the architect-engineering firm, the construction contractor, and the plant operator, as applicable, to obtain reasonable assurance that they will function as needed on the HS&E-related tasks.

The review should address how the administrative policies ensure the establishment and maintenance of design and operations. A description of the relationships among major plant safety functions such as the ISA, configuration management, maintenance, quality assurance (QA), training, radiation safety, nuclear criticality safety, fire safety, chemical safety, environmental monitoring, emergency planning, audits and assessments, and incident investigations should be reviewed. The review should also address the applicant's qualification criteria for education, training, and experience for key management positions.

2.4 ACCEPTANCE CRITERIA

2.4.1 Regulatory Requirements

Administrative policies for the effective implementation of HS&E functions concerning the applicant's corporate organization, qualifications of the staff, and the adequacy of the proposed equipment, facilities, and procedures to provide adequate safety for workers, the public, and the environment are required by 10 CFR 70.22, 70.23, and 70.62(d) of Part 70.

2.4.2 Regulatory Guidance

There are no regulatory guides that apply to the organization and administration description for a facility licensed under 10 CFR Part 70.

2.4.3 Regulatory Acceptance Criteria

The application is acceptable if the following criteria are met. Appropriate commitments relevant to these criteria should be included in the applicant's safety program description.

1. The applicant has identified and functionally described the specific organizational groups responsible for designing, constructing and operating the facility. Organizational charts are included in the application.

2. The qualifications, responsibilities, and authorities of key supervisory and management positions with HS&E managers (or similar positions), are clearly defined in position descriptions that are accessible to affected persons and to the NRC, upon request. Clear, unambiguous management control and communications exist among the organizational units responsible for the design and construction of the facility. A corporate officer is responsible for HS&E activities.

3. The personnel to design, construct, and operate the facility have substantive breadth and level of experience and are appropriately available. The qualifications, responsibilities, and authorities for key supervisory and management positions with HS&E responsibilities, including the plant manager, operations manager, shift supervisor, and HS&E managers (or similar positions), are clearly defined in position descriptions that are accessible to all affected personnel and to the NRC, upon request.

4. The applicant has described specific plans to transition from the design and construction phase to operations.

5. In the organizational hierarchy, the HS&E organization(s) is independent of the operations organization(s), allowing it to provide objective HS&E audit, review, or control activities. "Independent" means that neither organization reports to the other in an administrative sense. Both may report to a common manager. Lines of responsibility and authority are clearly drawn.

6. The individual delegated overall responsibility for the HS&E functions has the authority to shut down operations if they appear to be unsafe, and must in that case approve restart of shutdown operations. Typically, this individual should be at as high a management level as the production or operations manager and have direct line responsibility to the plant manager.

7. The activities essential for effective implementation of the HS&E functions are documented in formally approved, written procedures, prepared in compliance with a formal document control program.

8. The applicant should commit to a simple mechanism for reporting potentially unsafe conditions or activities to the HS&E organization and/or to upper management that is available for use by any person in the plant. Reported concerns are investigated, assessed, and resolved promptly by means of a Corrective Action Program.

9. Effective lines of communication and authority among the organization units involved in the engineering, HS&E, and operations functions of the facility are clearly defined.

10. The applicant has committed to establish formal management measures required to ensure the availability and reliability of items relied on for safety. Management measures and their review are detailed in Chapter 11 of this SRP.

11. Written agreements exist with off-site emergency resources such as fire, police, ambulance/rescue units, and medical services. This is addressed in more detail in Chapter 7, "Fire Safety," and Chapter 8, "Emergency Planning," of this SRP.

Commitments relevant to meeting the acceptance criteria described above are included in the applicant's safety program description.

2.5 REVIEW PROCEDURES

2.5.1 Acceptance Review

The primary reviewer should evaluate the application to determine whether it addresses the "Areas of Review" discussed in Section 2.3, above. If significant deficiencies are identified, the applicant should be requested to submit additional material before the start of the safety evaluation.

2.5.2 Safety Evaluation

After the application is determined to be acceptable for review in accordance with Section 2.5.1, above, the primary reviewer should perform a safety evaluation against the acceptance criteria described in Section 2.4. The objective of the review is to ensure that the corporate-level management and technical support structure, as demonstrated by organizational charts and descriptions of functions and responsibilities, are clear with respect to assignments of

primary responsibility. The primary reviewer consults with the TWRS Site Representative to verify that the applicant's management positions are adequately defined in terms of both numbers of persons and their responsibilities, authorities, and required qualifications.

The supporting reviewers should determine, on the basis of the foregoing, the overall acceptability of the applicant's management system, management qualifications, organizational structure, and administrative procedures. To facilitate the review of the applicant's proposed organization and administration program, the reviewers should examine organization charts, position descriptions, corporate and plant policies, and the descriptions of administrative procedures and guidance documents concerning HS&E. The reviewers should make a determination whether the acceptance criteria of Section 2.4.3 are satisfied and then provide input regarding their findings to the primary reviewer.

The primary reviewer should document the evaluation as described in Section 2.6.

2.6 EVALUATION FINDINGS

After completing the evaluation of material in response to this section, the primary reviewer should write an SER section addressing each topic reviewed under this SRP section and explain why the NRC staff has reasonable assurance that the organization and administration at the facility are acceptable. The SER section should include a summary statement of what was evaluated and the basis for the reviewers' conclusions. License conditions may also be proposed in this SER section to impose requirements where the application is deficient.

The staff can document the evaluation as follows:

> *The staff has reviewed the organization and administration for [name of facility] according to the Standard Review Plan Chapter 2.0.*
>
> *The applicant has described (1) clear responsibilities and associated resources for the design and construction of the facility and (2) its plans for management of the project.* [Insert a summary statement of what was evaluated and why the reviewer finds the submittal acceptable.] *The staff has reviewed these plans and commitments and concludes that they provide reasonable assurance that an acceptable organization, administrative policies, and sufficient competent resources have been established or are committed, to satisfy the applicant's commitments for the design and construction of the facility.*
>
> *In addition, the applicant has described its organization and administrative policies for providing adequate safety management and management measures for the safe operation of the facility.* [Insert a summary statement of what was evaluated and why the reviewer finds the submittal acceptable.] *The staff has reviewed this information and concludes that the applicant has an acceptable organization, administrative policies, and sufficient*

competent resources are established to provide for the safe operation of the facility under both normal and abnormal conditions.

2.7 REFERENCES

Code of Federal Regulations, *Title 10, Part 70*, "Domestic Licensing of Special Nuclear Material."

INTEGRATED SAFETY ANALYSIS

3.1 PURPOSE OF REVIEW

The purpose of this review is to establish that there is reasonable assurance that the applicant has performed an Integrated Safety Assessment (ISA) and submitted an ISA summary as required by 10 CFR 70.65(b) (as revised)[1]. The review should also establish that the facility is designed to meet the performance requirements contained in § 70.61.

3.2 RESPONSIBILITY FOR REVIEW

Primary: Integrated Safety Assessment (ISA) Specialist

Secondary: Licensing Project Manager

Supporting: Technical Area Specific Reviewers (Chemical Safety, Fire Safety, Radiological Protection, etc.)
Site Resident Inspector, if appropriate

3.3 AREAS OF REVIEW

Section 70.62, requires each licensee to perform an ISA to identify the following:

(i) Radiological hazards resulting from possessing or processing licensed material at its facility;

(ii) Chemical hazards of licensed material or hazardous chemicals produced from licensed material resulting from possessing or processing licensed material at its facility;

(iii) Facility hazards (e.g., chemical, fire, electrical and mechanical) which could affect the safety of licensed materials and thus present an increased radiological risk;

(iv) Potential accident sequences caused by process deviations or other events internal to the plant and credible external events, including natural phenomena;

(v) The consequence and the likelihood of occurrence of each potential accident sequence; identified in (iv) above, and the methods used to determine the consequences and likelihoods; and

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[1] Nuclear Regulatory Commission (U.S.), Washington, D.C. "Domestic Licening of Special Nuclear Material (10 CFR 70)". *Federal Register:* Vol. 64, No. 146. pp.41338--41357. July 30, 1999.

(vi) Each item relied on for safety and the characteristics of its preventive, mitigative, or other safety function and the assumptions and conditions under which the item is relied upon to support compliance with the performance requirements of § 70.61..

To assure that this has been done properly and to facilitate the review process, an ISA summary is submitted in accordance with § 70.64(b). The ISA summary should provide the following information for review:

1. Supporting Design Basis Information

   This section should provide enough information to support an evaluation of the completeness and acceptability of the (1) hazard identification task, (2) potential accident sequences task, (3) consequences and likelihood of occurrence of the accidents identified, and (4) items relied on for safety (items (i) through (vi) referenced above).

   a. Process description: This section should include all of the processes necessary to support the ISA summary and should include the intended purpose of the process and its relationship to the rest of the facility and products of the facility.

   b. Site description: This section should address and emphasize those factors that could affect safety, such as geography, meteorology (e.g., high winds, flood potential), seismology, and demography.

   c. Facility description: This section should address and emphasize those features that could affect potential accidents and their consequences. Examples of such features are facility location, facility design information, and the location and arrangement of buildings on the facility site.

   d. Process Theory: This section should consist of a description of the theory of operation of each process analyzed as part of the ISA. Areas include basic process function and theory, major components-their function and operation, and process operating ranges and limits, including expected limits and upset conditions. Schematics and flow diagrams of the process or parts of the process may also need to be included.

   e. Process Design and Equipment: This section should consist of the applicant's references to process safety information (PSI) sufficient to support the process description and process theory sections of the ISA. This should include information on the hazardous materials, technology, and equipment used in each process. The compilation and maintenance of current and accurate PSI should be explained in the applicant's description of its configuration management program.

   f. Drawings and Operating Procedures: This section should contain the applicant's commitment to maintain an accurate reference list of detailed engineering drawings, procedures, schedules, checklists, etc. Information referenced in this section should be supporting information that will also form the basis for facility

configuration management. There is expected to be overlap between this section and the preceding section, with much of the information referenced in the Process Design and Equipment section described above.

2. Process Hazards Analysis (PHA) Summary: This section should contain a brief discussion of the PHA method used for each individual process and the justification for its selection. For purposes of this review, the PHA summary begins with an identification of hazards that are identified in (i) through (iii) described above. Based on a systematic analysis of each plant process and the hazards identified, the ISA identifies a set of individual accident sequences that could result in consequences. The systematic analysis of the individual processes should include any interfaces with other processes and how specific accident sequences can impact those other processes. Information could be drawn from safety specific analyses (e.g., a fire hazard analysis) that look across specific processes. The accidents thus cause the threat of the hazards to become consequences of concern. The section is expected to contain a summary of the following:

   a. A description of the PHA methodology.

   b. Hazard identification.

   c. Accident sequences identification.

3. Safety Analysis Summary: This section should focus on hazard management. Given the PHA, the safety analysis allows for an integrated safety assessment including safety specific disciplines and across disciplines. The results should be compared to the performance requirements of § 70.61 and used to identify the controls relied on for safe operation of the facility. Specifically, this section should contain some form of the following:

   a. A summary of the unmitigated and mitigated consequences of each postulated accident to facility workers or the public.

   b. Comparisons of the consequences of each postulated accident to the performance requirements of § 70.61.

   c. Assignment of accident sequences to likelihood categories and comparison to performance requirements of § 70.61.

   d. Identification of items relied on for safety including engineered and administrative controls involved in each accident sequence.

4. ISA Management Summary: This section should contain information on the ISA team and the ISA process at the facility. Specifically this section should contain the following:

   a. A description of the ISA team.

b. A summary of the procedures for conducting and maintaining the ISA and a reference to the actual detailed procedures.

c. A protocol for informing the NRC of ISA summary updates.

3.4 ACCEPTANCE CRITERIA

3.4.1 Regulatory Requirements

Sections 70.65(b) and 70.61 specifically relate to the requirement to perform an ISA and submit the ISA summary to the NRC.

3.4.2 Regulatory Guidance

Guidance applicable to performing an ISA and documenting the results is given in NUREG-1513, "Integrated Safety Analysis Guidance Document," 1995. Guidance in regard to accident analysis may be found in the "Nuclear Fuel Cycle Facility Accident Analysis Handbook," NUREG/CR-6410, 1998.

3.4.3 Regulatory Acceptance Criteria

The NRC reviewers should find the applicant's submittal regarding the ISA summary provides reasonable assurance that the regulatory review criteria, below, are adequately addressed and satisfied. Some of the information may be referenced from other sections of the application, or incorporated by reference provided that these references are clear and specific.

3.4.3.1 Supporting Design Basis Information

The information provided in this section is acceptable if it allows for the reviewers to evaluate the completeness and acceptability of the ISA summary including (1) hazard identification task, (2) potential accident sequences task, (3) consequences and likelihood of occurrence of the accidents identified, and (4) items relied on for safety (items (i) through (vi) referenced above). If information was incorporated by reference and is needed to support the reviewer's evaluation with respect to the applicant demonstrating the ability to meet the performance criteria, then the reviewer should request through the project manager that the information be submitted.

1. Process Description: The description should be considered acceptable if it contains the following:

   a. A description of all of the processes that have applicability to plant operations that are contained in the ISA.

   b. The purpose of each process and its relationship to the overall facility process.

c. An identification of the components that are integral to plant operations, description, or process. This information should include the general arrangement, function, and operation of these components in the process. It should include process schematics showing the components and instrumentation as well as chemical flow sheets showing the anticipated ranges of compositions of the various process streams. Such information should be provided to the extent necessary to describe the process in regard to performance requirements.

2. Site Description: The description should be considered acceptable if it contains:

   a. A description of the site geography, including its location relative to prominent natural and man-made features such as mountains, rivers, airports, population centers, possibly hazardous commercial and manufacturing facilities, etc., adequate to permit evaluation of the ISA summary.

   b. Population information based on recent census data to show population distribution as a function of distance from the facility adequate to permit evaluation of the ISA summary.

   c. Characterization of natural phenomena (e.g., tornadoes, hurricanes, high winds, and earthquakes) and other external events sufficient to assess their likelihood of occurrence and to assess their impact on plant safety. The discussion identifies the design basis events for the facility and indicates which events are considered incredible and the basis for that determination. The assessment also indicates which events could occur without adversely impacting safety.

   d. Designation of controlled site boundary.

3. Facility Description: The description should be considered acceptable if it contains;

   a. The facility location and the distance from any boundaries established for regulatory compliance, including the distance to the nearest resident and distance to boundaries in the prevailing wind directions. The distances to all publicly accessible locations, if any, within the site boundary shall be included.

   b. A description of all of the buildings that house the processes discussed above.

   c. Design information regarding the ability of the facility to withstand the effects of credible external events, when those failures may impact the performance criteria.

   d. The location and arrangement of buildings on the facility site.

4. Process Theory: The discussion of process theory should be considered acceptable if it includes or references elsewhere in the application:

a. Basic process function and theory, including a general discussion of the basic theory of operation of each described process.

b. Process operating ranges and limits, including the operating ranges and limits for all measured variables (e.g., temperatures, pressures, flows, and compositions) used in engineered or administrative controls to ensure safe operation of the process. A set of postulated abnormal operating conditions, where applicable, should be identified. The process operating limits and ranges are considered acceptable if they provide reasonable assurance of process safety and are consistent with those assumed in the hazards analysis.

c. Schematics indicating safety interrelationships of parts of the process. In particular, either schematics or descriptions indicating the inventory, location, and geometry of special nuclear materials, moderators, and other materials in the process should be sufficient to permit an understanding of the adequacy of controls on mass, geometry, moderation, reflection, and other criticality parameters.

5. Process Design and Equipment: This section of the ISA summary should be considered acceptable if the following process safety information[2] is provided or referenced (external to the application) and that a commitment is provided to maintain the information current and accurate:

a. Hazardous material information including toxicity information, permissible exposure limits, physical data, reactivity data, corrosivity data, and thermal and chemical stability data.

b. Process technology information including block flow diagram or simplified process flow diagram, process chemistry, maximum intended inventory, and safe upper and lower limits for such items as temperatures, pressures, flows, and compositions.

c. Process and safety equipment assurance measures, including codes and standards used for mechanical, civil, chemical, electrical, and instrumentation and control systems.

d. Process and safety equipment information including materials of construction, piping and instrumentation diagrams (P&IDs), electrical classification, material and energy balances, functional logic diagrams, requirement and design specifications, software code, and electrical/electronic schematics.

e. The compilation and maintenance of current and accurate PSI should be explained in the applicant's description of its configuration management program.

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[2] This information is consistent with that of the process safety information contained in 29 CFR 1910.119, "Process Safety Management of Highly Hazardous Chemicals."

6. Drawings and Operating Procedures: This section should be considered acceptable if the final collection of material available at the site as referenced by this section is sufficient to establish the design basis for system configuration management for each system and process discussed under process description. As referenced material is needed in the safety evaluation, then through the licensing project manager, the specific references should be requested to be submitted to the NRC.

3.4.3.2 Process Hazards Analysis Summary

1. The description of the PHA methodology selected should be considered acceptable if it is consistent with the guidance provided in NUREG-1513. For methods used by the applicant but not addressed in NUREG-1513, the applicant should provide justification and references for their use.

   The PHA ordinarily should be considered acceptable if it provides the following:

   a. The PHA summary addresses potential process specific hazards identified in (i) through (iii) in SRP Section 3.3, above. The applicant should identify and justify any hazards eliminated from further consideration.

   b. The PHA summary provides reasonable assurance that the applicant identifies all process specific significant accident sequences (including the controls used to prevent or mitigate the accidents) that could result in radiological and nuclear criticality consequences. Chemical consequences which could result from processing licensed material or adversely affect radiological safety should also be included.

   c. The PHA summary takes into account the interactions of identified hazards and proposed controls, including interactions between systems and processes, to ensure that the overall level of risk at the facility is minimized.

   d. The PHA summary addresses all modes of operation including startup, normal operation, shutdown, and maintenance.

   e. The PHA summary addresses hazards resulting from process deviations (e.g., high temperature, high pressure), initiating events internal to the facility (e.g., fires or explosions), and credible external events (e.g., floods, high winds, earthquakes, and airplane crashes). The PHA summary should address aspects of the entire event sequence. The applicant should provide justification for its determination that certain events are incredible and, therefore, not subject to analysis in the ISA (this could be more categorical in nature rather then for every event).

   f. The PHA summary adequately describes the effects and failures of non-safety systems and components on safety systems and components.

   g. The PHA summary adequately addresses initiation of, or contribution to, accident sequences by human error.

h. The PHA summary adequately addresses common mode failures and system interactions in evaluating systems that are to be protected by double contingency.

2. The summary of the hazard identification results should be considered acceptable if it provides:

   a. A list of materials and chemicals (radioactive, fissile, flammable, and toxic) that could result in hazardous situations affecting safe operation of the facility. The list includes maximum intended inventory amounts and the location of the hazardous materials at the site.

   b. A hazards interaction table showing potential interactions either between materials/chemicals, including radiolysis, that could possibly result in hazardous situations affecting safe operation of the facility.

   c. A list of facility hazards (e.g., chemical, fire, electrical and mechanical) which could affect the safety of licensed materials.

3. The summary of potential accident sequences should be considered acceptable if it includes:

   a. The accident sequences whose unmitigated consequences exceed the performance criteria contained in § 70.61.

   b. The controls or barriers that must fail in order for the accident to occur.

3.4.3.3 Safety Analysis

1. A summary of the unmitigated and mitigated consequences of each postulated accident to facility workers or the public should be acceptable if it contains the following:

   a. Evidence that discipline-specific safety (i.e., radiation, criticality, fire and chemical safety) hazards, accident scenarios, and controls are represented in the summary. The summary has considered all credible cross-discipline interactions that could result in initiation or intensification of an accident such as loss of criticality control caused by water from fire suppression activities.

   b. Comparisons of the consequences of each postulated accident to the performance criteria of § 70.61.

   c. Assignment of accident sequences to likelihood categories and comparison to the performance criteria of § 70.61.

   d. Identification of engineered controls used in the determination of mitigated consequences.

e. A listing of accidents evaluated as incredible events. Adequate justification for their evaluation as incredible should be provided. Reviewers are cautioned against excessive focus on the adequacy of justifications for incredible events that can be qualitatively shown to be so unlikely as to not merit consideration. In addition, events that are unlikely to have adverse impacts on the system need not be considered if similar events that pose greater hazards have already been considered.

2. Evaluation of consequences of accidents should be considered acceptable if:

   a. The narrative demonstrates that valid consequence evaluation methods have been used, as described in the appropriate safety chapters of the license application (e.g., Nuclear Criticality Safety, Chemical Safety);

   b. The narrative contains a description of accidents for which consequences have been evaluated along with the quantitative results in a form that can be directly compared to the performance criteria of § 70.61; and

   c. The summary of accident sequences gives either the calculated consequence values or a traceable reference to the quantitative evaluation that is the basis for the assignment of the accident sequence to the correct consequence category of the performance criteria of § 70.61.

3. To demonstrate sufficiently low likelihood for each accident sequence for compliance with the performance criteria of § 70.61, it is necessary, as a minimum, that the items relied on for safety supported by applicable measures to assure their reliability, meet the following qualitative criteria:

   a. For an accident sequence that results in a nuclear criticality accident, adherence to double contingency should be demonstrated. Adherence to double contingency requires that at least two unlikely, independent, and concurrent changes in process conditions are necessary before a criticality accident can occur. If double contingency is not feasible, then the controls should exhibit sufficient redundancy and diversity to make criticality comparably unlikely.

   b. For an accident sequence that results in "high consequences," other than nuclear criticality, as defined in § 70.61, the likelihood should be comparable to that achieved by double contingency. Normally, multiple independent events are required to achieve such a likelihood. However, in principle, it can be achieved if the sequence requires a single event which is confidently known to be highly unlikely. Alternatively, or in addition, controls may be used to mitigate the consequences of the accident rather than to prevent its occurrence.

   c. For an accident sequence that results in consequences, "intermediate" as defined in § 70.61, at least one single unlikely event must occur before the unmitigated consequences of the accident occur. A mitigative control applied to a sequence must

reduce the consequences below the limits defining the lower bound of the category in order to be credited in determining compliance with § 70.61

4. A list of items relied on for safety required to satisfy the performance criteria of § 70.61 should be considered acceptable if:

   a. It includes all items relied on for safety in the identified accident sequences; and

   b. The description of the items relied on for safety, clearly articulating the specific safety features, their assurance measures, and the associated safety limits and margins are adequate to permit a determination of compliance with § 70.61,

   c. Information concerning the assignment of assurance measures to safety controls is adequate to show compliance with § 70.61. (If a system of graded assurance measures is used, the grade applied to each control should be determinable from information provided.)

3.4.3.4 ISA Management

Management controls should be considered acceptable if the following criteria are met:

1. The ISA team should have a team leader who is knowledgeable in the ISA methodology chosen for the hazard and accident evaluations. In addition, the team leader should be able to demonstrate a basic understanding of all process operations and hazards under evaluation, but should not be the cognizant engineer or expert for that process.

   a. At least one member of the ISA team should have specific and detailed experience in each process under evaluation.

   b. A variety of process operating and engineering design experience should be represented across the team. Radiation safety, nuclear criticality safety, fire protection, and chemical safety disciplines should also be represented.

   c. A manager provides overall administrative and technical direction for the ISA.

2. The description of the facility procedures for conducting and maintaining the ISA should be considered acceptable if it includes:

   a. Management policies.

   b. Organizational responsibilities.

   c. Administrative controls, and procedures governing the performance, review, and approval of the initial ISA and any revisions to the ISA.

d. A commitment to maintain the ISA to reflect changes using a team with similar qualifications to the team that originally prepared the ISA for the system under review.

e. A commitment to maintain the ISA under an adequate configuration management function.

f. Identifies updates to the table on controls necessary to ensure safety, as well as seek prior approval for any changes that raise unreviewed safety questions or increase the level of risk.

g. Administrative controls ensure the independence of reviewing organizations and individual reviewers.

h. Procedures to control records and supporting documentation concerning the ISA.

3. The protocol for informing the NRC of ISA summary updates should be acceptable if it is consistent with the requirements in § 70.72.

3.5 REVIEW PROCEDURES

3.5.1 Acceptance Review

The primary reviewer should evaluate the application to determine whether it addresses the "Areas of Review" discussed in Section 3.3, above. If significant deficiencies are identified, the applicant should be requested to submit additional material before the start of the safety evaluation.

3.5.2 Safety Evaluation

After determining that the application is acceptable for review in accordance with Section 3.5.1, above, the primary reviewer will perform a safety evaluation against the acceptance criteria described in Section 3.4. If during the course of the safety evaluation, the primary reviewer determines the need for additional information, the primary reviewer coordinates a request for additional information with the licensing project manager.

The secondary reviewer (licensing project manager) should assure that the team of supporting reviewers is appropriate for the processes, systems, and events considered. The secondary reviewer should also review the sections of ISA Management.

Because the ISA summary forms the basis for many of the individual discipline specific safety programs (i.e., radiation, criticality, chemical, and fire safety), the supporting reviewers should assure that there is evidence that discipline specific inputs have been incorporated into the Safety Analysis section of the ISA summary. The reviewer should assure that the ISA also addresses events, such as fire or earthquake, that could affect more than one process. The reviewer should also evaluate areas of possible safety conflict, an example being fire

suppression systems and nuclear criticality safety. Furthermore, the supporting reviewers should assure that the identified hazards, accident scenarios, consequences and controls contained in the ISA summary are consistent with the appropriate SRP Sections (i.e., fire, chemical, criticality safety) throughout the application.

When the safety evaluation is complete, the primary staff reviewer, with assistance from the other reviewers, should prepare the Integrated Safety Analysis input for the SER as described in Section 3.6.

3.6 EVALUATION FINDINGS

The primary reviewer should write an SER section addressing each topic reviewed under this SRP chapter and explain why the NRC staff has reasonable assurance that the ISA summary submitted is acceptable. License conditions may be proposed to impose requirements where the application is deficient. The SER should include a summary statement of what was evaluated and the basis for the reviewers' conclusions.

The staff can document the evaluation as follows:

> *The staff has evaluated* ... [Insert a summary statement of what was evaluated and why the reviewer finds the submittal acceptable.] *The applicant has performed an Integrated Safety Analysis (ISA) to identify and evaluate those hazards and potential accidents that could result in unintended exposure of persons to radiation or radioactive materials associated with licensed materials, and to establish safety controls to ensure facility operation within the bounds of the ISA. The NRC staff has reviewed those postulated accidents resulting from the facility hazards that may be anticipated to occur (or are considered unlikely or highly unlikely). To ensure that the performance requirements of § 70. 61 are met, the applicant has established both administrative and engineered safety controls. The staff has reviewed these safety controls and finds them acceptable based on the staff's evaluation of a summary of the applicant's ISA and other supporting information.*
>
> *The staff concludes that the identification and evaluation of the hazards and accidents as part of the ISA and the establishment of controls to maintain safe facility operation from their consequences satisfy the performance requirements of § 70. 61.*

3.7 DEFINITIONS

These definitions have specialized meanings to be applied only in the context of using this SRP chapter.

Accident Sequence

In general, an unintended sequence of events or process failures that would result in adverse consequences. In the context of this SRP, an unintended sequence of events that results in environmental contamination, a radiation exposure, a release of radioactive material, an inadvertent nuclear criticality, or an exposure to hazardous chemicals, provided the chemicals are composed of, or the accident results from the processing of, licensed radioactive material; or if the accident has the potential to jeopardize the safety of regulated activities. The term "accident" may be used interchangeably with accident sequence.

Assurance Measures

An inclusive term for any measures applied to items relied on for safety to ensure their ability to reliably and effectively perform their safety function. Such measures include design procedures, human-system interface analysis, construction procedures, functional testing, inspections, calibration, surveillance monitoring and testing, maintenance, training, configuration management, quality assurance, records management, and audits.

Operating procedures that are relied on for safe operation are considered administrative safety controls, not assurance measures. However, the policy of requiring written operating procedures for the purposes of safety would be one element of an acceptable configuration management program.

Certain assurance measures are of a generic nature in that they apply to the whole system of safety controls, not to any one control in particular. These include incident investigation, safety organization, management independence and authority, and policies or procedures specifying how safety management functions are to be carried out.

Baseline Design Criteria

A set of criteria that identify safety considerations that applicants must address in the design of new facilities or in the design of new processes at existing facilities, in accordance with § 70.64. Applicants are expected to address these baseline design criteria in establishing minimum requirements for all items relied upon for safety.

Consequence

Any result of interest or concern caused by an event or sequence of events. In this context, adverse consequences refers to the adverse health or safety effects on workers or the public. Consequences are specified in § 70.61, in the context of meeting performance requirements.

> Unmitigated Consequences are the consequences that result from an accident sequence when mitigative control fails or does not exist.

Control

A system, device, or procedure intended to regulate a device or process. Controls may be engineered controls or administrative (procedural) controls. Controls may be preventive or mitigative. A process control may not be "an item relied on for safety" if safety controls exist that will perform their function despite frequent or continuous failure of the process control.

Administrative Control: The provisions relating to organization and management, procedures, record keeping, reviews, audits, and reporting necessary to ensure operation of the plant in a safe manner.

Engineered Control: An active or passive structure, system, or component that prevents or mitigates the consequences of accidents from licensed material that could cause significant consequences.

Mitigative Control: A control intended to reduce the consequences of an accident sequence, not to prevent it entirely. When a mitigative control works as intended, the results of the sequence are called the mitigated consequences.

Preventive Control: A control intended to prevent an accident entirely, i.e., to prevent any of the types of radiological or chemical consequences.

Process Control: A control that is not considered a Safety Control.

Safety Control: A system, device, or procedure intended to regulate a device or process so as to maintain a safe state. Effectively synonymous with "item relied on for safety." In the context of this SRP, use of the unmodified term "control" normally means safety control. The function of safety controls is to satisfy the performance requirements contained in § 70.61.

Event

An occurrence; a change of conditions from a prior state.

Credible Event: An initiating (or secondary) event with a likelihood of occurrence greater than one in a million in any year. Any accident sequence identified in the ISA as initiated by a credible event must have its consequences assessed, and controls applied so as to satisfy the performance requirements contained in § 70.61. When determining whether an event (or its likelihood category) is credible, uncertainty in the estimate of likelihood of the event as well as the estimate itself, should be considered. This will help to assure that events or accident sequences are not improperly categorized because of estimation method or choice of data or assumptions.

External Event: An event for which the likelihood cannot be altered by changes to the regulated facility or its operation. This would include all natural phenomena events plus airplane crashes, explosions, toxic releases, fires, etc., occurring near or on the plant site.

Incredible Event: An initiating (or secondary) event that is so unlikely that it alone makes the sequence sufficiently improbable (i.e., likelihood less than or equal to 1 in a million per year) that it need not be addressed further, even for consideration of the maximum credible consequences. For such sequences, there is no need to add controls to prevent occurrence of consequences of concern. In evaluating compliance with § 70.61, using the ISA, justification should be provided that such events are, in fact, of sufficiently low frequency.

Initiating Event: The first event in an accident sequence. In a well-defined accident sequence, an initiating event is normally the first deviation of the system from its intended behavior (a failure), or the occurrence of an abnormal condition beyond the system's design basis. Subsequent events in the sequence are referred to as secondary events.

Internal Event: An event for which changes to the regulated facility or its operation can affect the likelihood of occurrence. This would include all deviations from normal process operating conditions and abnormal events in other plant processes that would, if controls fail, contribute to causing an accident with consequences of concern.

Natural Phenomena Events: Earthquakes, floods, tornadoes, tsunamis, hurricanes, and other events that occur in the natural environment and could adversely affect safety. Natural phenomena events, depending on their likelihood of occurrence, may be credible or incredible.

Items Relied on for Safety

Structures, systems, equipment, components, and activities of personnel that are relied on to prevent or mitigate accidents to satisfy the performance requirements contained in 10 § 70.61. These items include design features and controls, both engineered and administrative, that are relied on to protect the worker, the public, and the environment in all phases of operation, including during normal operation, transients, and accidents in progress (mitigation).

Design features and controls relied on for safety include those that:

1. Confine or contain SNM for safety reasons;

2. Control a process to maintain the chemical form, concentration, geometry, or other property of SNM-bearing material to assure safety;

3. Provide the capability to place or maintain a process containing SNM in a safe shutdown condition;

4. Are operating procedures relied on for safety, or other actions of personnel required for safety;

5. Are items or human actions that, if not functioning properly, could cause the failure of another item relied on for safety;

6. Are items or human actions that, if not functioning properly, could substantially degrade the reliability of another item relied on for safety.

Certain process controls and features may be excluded from being considered items relied on for safety, even though they functionally provide a margin of safety, provided no credit is taken for this safety functionality in assessing the adequacy of the safety performance of the process for compliance with § 70.61.

Uncontrolled Outcome

The sequence of events and consequences that result if no controls or barriers are available to prevent or mitigate an accident sequence. Thus the consequences of an uncontrolled outcome are, by definition, unmitigated. These consequences may also be referred to as uncontrolled consequences.

Unlikely

For the facility unlikely is an implied assessment of a frequency of occurrence (or exceedence) of less than 10^{-2} but greater than 10^{-5} per year. For the facility highly unlikely is an implied assessment of a frequency of occurrence (or exceedence) of less than 10^{-5} per year.

3.8 REFERENCES

1. American Institute of Chemical Engineers (AIChE). "Guidelines for Hazard Evaluation Procedures, Second Edition with Worked Examples." AIChE: New York. September 1992.

2. Nuclear Regulatory Commission (U.S.), Washington, D.C. "Domestic Licensing of Special Nuclear Material (10 CFR Part 70)." *Federal Register:* Vol. 64, No. 146. pp. 41338--41357. July 30, 1999.

3. Nuclear Regulatory Commission (U.S.) (NRC). NUREG-1513, "Integrated Safety Analysis Guidance Document." NRC: Washington, D.C. 1995.

4. Nuclear Regulatory Commission (U.S.) (NRC). NUREG/CR-6410, "Nuclear Fuel Cycle Accident Analysis Handbook.", NRC: Washington, D.C. 1998.

RADIATION SAFETY
4.1 RADIATION SAFETY PROGRAM

4.1.1 PURPOSE OF REVIEW

The purpose of this review is to determine, with reasonable assurance, that the applicant's radiation safety (RS) program is adequate to protect the radiological health and safety of the workers and to comply with the regulatory requirements of 10 CFR Parts 19, 20, and 70.

The applicant's program for protection of members of the public and control of effluent releases is not included in this Chapter but is in SRP Chapter 9.0, "Environmental Protection." While this chapter reviews the applicant's RS *program*, radiation safety design aspects of the facility and the radiation safety aspects of the integrated safety analysis (ISA) are reviewed under SRP Chapter 4.2, "Radiation Safety Design Features."

4.1.2 RESPONSIBILITY FOR REVIEW

Primary: Health Physicist

Secondary: None

Supporting: Licensing Project Manager (as reviewer of SRP Chapters 2.0, 3.0 and Section 11.4)
Environmental Engineer (as reviewer of SRP Chapter. 9.0)
Quality Assurance Specialist (as reviewer of SRP Section 11.1)

4.1.3 AREAS OF REVIEW

An RS program is required to be established and implemented by 10 CFR 20.1101. (As used in this SRP the terms *Radiation Safety Program* and *Radiation Protection Program* are synonymous). The elements of the applicant's proposed RS program that should be reviewed by the staff are identified in the following list.

1. As Low as Is Reasonably Achievable (ALARA) Considerations

   The applicant's management policy should be reviewed with respect to designing and constructing the plant, operating the plant, and the planned organizational structure and how units of that structure interact to maintain occupational doses ALARA. The applicable activities and audits carried on by the individuals in management having responsibility for RS, and commitments to radiological performance goals (ALARA goals) and trend analyses should also be reviewed.

2. Organizational Relationships and Personnel Qualifications

   The applicant's organization of the RS program, the qualification requirements for the RS personnel, and the assignment of specific responsibilities and authorities for key functions should be reviewed.

3. Radiation Safety Procedures and Radiation Work Permits (RWPs)

The applicant's commitments regarding the need for, development and control of, and use of approved written RS procedures and RWPs for activities related to radiological safety should be reviewed.

4. Training

The applicant's proposed RS training for all personnel who have authorized access to restricted areas should be reviewed. The review should include training objectives, management oversight, methodology of training, who receives training, a description and frequency of training and refresher training, and the effectiveness of the training. Further aspects of training are covered in SRP Section 11.4.

5. Air Sampling

The applicant's radiological air sampling objectives and commitments to procedures should be reviewed including the following:

a. The frequency and methods of analysis of airborne concentrations,
b. Sampling methods and frequencies,
c. Counting techniques,
d. Lower limits of detection,
e. Specific calculations for concentrations,
f. Action levels and actions to be taken when they are exceeded, and
g. The locations of continuous air monitors and annunciators and alarms associated with them.

Note that the related area of ventilation systems is reviewed under SRP Section 4.2.

6. Contamination Control

The applicant's control of radiological contamination within the facility including the types and frequency of surveys, administrative contamination threshold levels, the methods and choice of instruments used in the surveys, and the action levels and actions to be taken if exceeded should be reviewed. The design features to control access should also be reviewed, including the following:

a. The technical criteria and levels for defining contamination and high contamination areas,
b. The types and availability of contamination monitoring equipment,
c. Specific limits established for personnel decontamination,
d. Minimum provisions for personnel decontamination,
e. The minimum types of clothing needed to enter contaminated areas,
f. The release criteria for contaminated materials, and
g. The frequency of periodic review of all aspects of access control.

7. External Exposure

The applicant's program for monitoring personnel external radiation dose including the means to measure, assess and record personnel radiation dose should be reviewed. In addition, the types, range, sensitivity, accuracy, and frequency for analyzing personnel dosimetry and the action levels and actions to be taken if action levels or limits are exceeded should be reviewed.

8. Internal Exposure

The applicant's program for monitoring personnel internal radiation doses should be reviewed including the following:

a. The criteria for determining when it is necessary to monitor an individual's internal dose,
b. The methods for determining intake,
c. Frequency of analyses,
d. Minimum detection levels,
e. Action levels and actions to be taken when exceeded.

9. Summing Internal and External Exposure

The applicant's program for summing internal and external exposure, including the procedures used to combine a worker's internal and external dose to demonstrate compliance with NRC regulations, should be reviewed.

10. Respiratory Protection

The applicant's respiratory protection program, including equipment to be used, conditions under which respiratory protection is necessary for routine and non-routine operations, the protection factors to be applied when respirators are being employed, and the locations of respiratory equipment in the plant should be reviewed.

11. Instrumentation

The applicant's provisions for radiological measurement instrumentation, including maintenance and use, ranges, counting modes, sensitivity, alarm set points, planned use, and calibration frequency should be reviewed.

4.1.4 ACCEPTANCE CRITERIA

4.1.4.1 Regulatory Requirements

Regulations applicable to this SRP chapter are listed below [the relevant Acceptance Criteria section is in brackets following the regulatory citation].

Radiation Safety

U.S. Code of Federal Regulations, Title 10, Energy, Part 19, "Notices, Instructions, and Reports to Workers: Inspections and Investigations"

§ 19.12 *Instruction to Workers* **[Sections 4.1.4.3.1, 4.1.4.3.4]**

§ 19.13 *Notifications and Reports to Individuals* **[Sections 4.1.4.3.7, 4.1.4.3.8]**

Code of Federal Regulations, *Title 10, Energy*, Part 20, "Standards for Protection Against Radiation."

§ 20.1101 *Radiation Protection Programs* **[Sections 4.1.4.3.1** (Part 20.1101(b)), **4.4.3.3]**

§ 20.1201 *Occupational Dose Limits For Adults* **[Sections 4.1.4.3.7** (Part 20.1201(a)(1), (a)(2) and (c)), **4.1.4.3.8** (Part 20.1201(a)(1), (d) and (e)), **4.1.4.3.9** (Part 20.1201(a)(1) and (f))]

§ 20.1202 *Compliance with Requirements for Summation of External and Internal Doses* **[Section 4.1.4.3.9]**

§ 20.1203 *Determination of External Dose from Airborne Radioactive Material* **[Section 4.1.4.3.7]**

§ 20.1204 *Determination of Internal Exposure* **[Sections 4.1.4.3.5, 4.1.4.3.8]**

§ 20.1206 *Planned Special Exposures* **[Section 4.1.4.3.7]**

§ 20.1207 *Occupational Dose Limits for Minors* **[Sections 4.1.4.3.7, 4.1.4.3.8, 4.1.4.3.9]**

§ 20.1208 *Dose to Embryo/Fetus* **[Sections 4.1.4.3.7, 4.1.4.3.8, 4.1.4.3.9]**

§ 20.1301 *Dose Limits for Individual Members of the Public* **[Sections 4.1.4.3.7** (Parts 20.1301(a)(1), (a)(2), (b), and (c))]

§ 20.1302 *Compliance with Dose Limits for Individual Members of the Public* **[Sections 4.1.4.3.7** (Parts 20.1302(a), (b)(1) and (b)(2)(ii))]

§ 20.1406 *Minimization of Contamination* **[Section 4.1.4.3.6]**

§ 20.1501 *Surveys and Monitoring - General* **[Sections 4.1.4.3.6** (Parts 20.1501 (a)(2)(ii) and (a)(2)(iii)), **4.1.4.3.7** (Parts 20.1501(a)(2)(i) and (c)), **4.1.4.3.11** (§20.1501(b) and (c)]

§ 20.1502 *Conditions Requiring Individual Monitoring of External and Internal Occupational Doses* **[Sections 4.1.4.3.7** (Part 20.1502(a)), **4.1.4.3.8** (Part 20.1502(b))]

§ 20.1601 *Control of Access to High Radiation Areas* **[Section 4.1.4.3.7]**

§ 20.1602 *Control of Access to Very High Radiation Areas* **[Sections 4.1.4.3.6, 4.1.4.3.7]**

§ 20.1701 *Use of Process or Other Engineering Controls* **[Section 4.1.4.3.10]**

§ 20.1702 *Use of Other Controls* **[Section 4.1.4.3.10]**

§ 20.1703 *Use of Individual Respiratory Protection Equipment* **[Sections 4.1.4.3.5, 4.1.4.3.6** (Part 20.1703(a)(3)(ii)), **4.1.4.3.8** (Parts 20.1703(a)(3)(ii) and (b)), **4.1.4.3.10** (Parts 20.1703(a), (c) and (d))**]**

§ 20.1901 *Caution Signs* **[Sections 4.1.4.3.6, 4.1.4.3.7]**

§ 20.1902 *Posting Requirements* **[Sections 4.1.4.3.5** (Part 20.1902(d)), **4.1.4.3.6** (Part 20.1902(e)), **4.1.4.3.7** (Parts 20.1902(a), (b) and (c)), **4.1.4.3.8** (Part 20.1902(d))**]**

§ 20.1904 *Labeling Containers* **[Section 4.1.4.3.6]**

§ 20.1906 *Procedures for Receiving and Opening Packages* **[Sections 4.1.4.3.6, 4.1.4.3.7]**

§ 20.2101 *Records-General Provisions* **[Sections 4.1.4.3.7, 4.1.4.3.8, 4.1.4.3.9]**

§ 20.2102 *Records of Radiation Protection Programs* **[Section 4.1.4.3.1]**

§ 20.2103 *Records of Surveys* **[Sections 4.1.4.3.5, 4.1.4.3.6, 4.1.4.3.7, 4.1.4.3.8, 4.1.4.3.11]**

§ 20.2104 *Determination of Prior Occupational Dose* **[Section 4.1.4.3.9]**

§ 20.2105 *Records of Planned Special Exposures* **[Sections 4.1.4.3.7, 4.1.4.3.8, 4.1.4.3.9]**

§ 20.2106 *Records of Individual Monitoring Results* **[Sections 4.1.4.3.7, 4.1.4.3.8, 4.1.4.3.9]**

§ 20.2110 *Form of Records* **[Sections 4.1.4.3.1, 4.1.4.3.5, 4.1.4.3.6, 4.1.4.3.7, 4.1.4.3.8, 4.1.4.3.9, 4.1.4.3.10]**

20.2202 *Notification of Incidents* **[Sections 4.1.4.3.7** (Parts 20.2202(a)-(d)), **4.1.4.3.8** (Parts 20.2202(a)-(d)), **4.1.4.3.9** (Parts 20.2202(a)-(d))**]**

§ 20.2203 *Reports of Exposures, Radiation Levels, and Concentrations of Radioactive Materials Exceeding the Limits* **[Sections 4.1.4.3.5** (Parts 20.2203(a)(3)(i)-(ii),

(b), and (d)), **4.1.4.3.6** (Parts 20.2203(a)(3)(i)-(ii) and (b)), **4.1.4.3.7** (Parts 20.2203(a)(2), (a)(3)(i)-(ii), (b) and (d)), **4.1.4.3.8** (Parts 20.2203(a)(2), (b), and (d), **4.1.4.3.9** (Parts 20.2203(a)(2), (b), and (d)]

§ 20.2206 *Reports of Individual Monitoring* **[Sections 4.1.4.3.7, 4.1.4.3.8, 4.1.4.3.9]**

Code of Federal Regulations, *Title 10, Energy*, Part 70, "Domestic Licensing of Special Nuclear Material"

§ 70.22 *Contents of Applications* **[Sections 4.1.4.3.2** (Part 70.22(a)(6)), **4.1.4.3.3** (Part 70.22(a)(8)), **4.1.4.3.4** (Part 70.22(a)(6)), **4.1.4.3.5** (Part 70.22(a)(7))]

§ 70.23 *Requirements for Approval of Applications* **[Sections 4.1.4.3.2, 4.1.4.3.3** (Part 70.23(a)(2))]

4.1.4.2 Regulatory Guidance

Listed in this section are NRC Regulatory Guides (RGs), NUREG reports, Branch Technical Positions (BTPs), and industry standards that, in general, provide a basis that is generally acceptable to the NRC staff for satisfying the regulatory requirements listed in Section 4.1.4.1. The applicable Acceptance Criteria sections, to which a particular guidance document relates, are listed in brackets following each guidance document.

1. NRC Regulatory Guides (RGs)

RG 8.4	Feb. 1973	*Direct and Indirect-Reading Pocket Dosimeters* **[Section 4.1.4.3.7]**
RG 8.7 Rev. 1	June 1992	*Instructions for Recording and Reporting Occupational Radiation Exposure Data* **[Sections 4.1.4.3.7, 4.1.4.3.8, 4.1.4.3.9]**
RG 8.9 Rev. 1	July 1993	*Acceptable Concepts, Models, Equations, and Assumptions for a Bioassay Program* **[Section 4.1.4.3.8]**
RG 8.10 Rev. 1-R	May 1977	*Operating Philosophy for Maintaining Occupational Radiation Exposures As Low as is Reasonably Achievable* **[Section 4.1.4.3.1, 4.1.4.3.2, 4.1.4.3.3, 4.1.4.3.4]**
RG 8.13		*Instructions Concerning Prenatal Radiation Exposures* **[Section 4.1.4.3.8]** *(Draft DG-801 proposed Rev. 3, Oct. 1994).*
RG 8.15	Oct. 1976	*Acceptable Programs for Respiratory Protection* **[Section 4.1.4.3.10]**

RG 8.21 Rev. 1	Oct. 1979	*Health Physics Surveys for Byproduct Material at NRC Licensed Processing and Manufacturing Plants* **[Section 4.1.4.3.6]**
RG 8.25 Rev. 1	June 1992	*Air Sampling in the Workplace* **[Sections 4.1.4.3.5, 4.1.4.3.8]**
RG 8.28	Aug. 1981	*Audible Alarm Dosimeters* **[Sections 4.1.4.3.7, 4.1.4.3.11]**
RG 8.29 Rev. 1	Feb. 1996	*Instructions Concerning the Risks from Occupational Radiation Exposure* **[Section 4.1.4.3.4]**
RG 8.34	July 1992	*Monitoring Criteria and Methods to Calculate Occupational Radiation Doses* **[Sections 4.1.4.3.7, 4.1.4.3.8, 4.1.4.3.9]**
RG 8.35	June 1992	*Planned Special Exposures* **[Sections 4.1.4.3.7, 4.1.4.3.8, 4.1.4.3.9]**
RG 8.36	July 1992	*Radiation Dose to the Embryo/Fetus* **[Section 4.1.4.3.9]**

2. NRC NUREG REPORTS

NUREG-0041	Oct. 1976	*Manual of Respiratory Protection against Airborne Radioactive Materials* **[Sections 4.1.4.3.4, 4.1.4.3.5, 4.1.4.3.10]**
NUREG-1400	Sept. 1993	*Air Sampling in the Workplace* **[Section 4.1.4.3.5]**

3. NRC Branch Technical Positions (BTPs)

April 1993	*License Condition for Leak Testing Sealed Byproduct Material Sources* **[Section 4.1.4.3.6]**
April 1993	*License Condition for Leak Testing Sealed Plutonium Sources* **[Section 4.1.4.3.6]**
April 1993	*License Condition for Plutonium Alpha Sources* **[Section 4.1.4.3.6]**
April 1993	*License Condition for Leak Testing a Sealed Source which Contains Alpha and/or Beta-Gamma Emitters* **[Section 4.1.4.3.6]**
April 1993	*License Condition for Leak Testing Sealed Uranium Sources* **[Section 4.1.4.3.6]**
April 1993	*Guidelines for Decontamination of Facilities and Equipment Prior to Release for Unrestricted Use or Termination of Licenses for Byproduct, Source, or Special Nuclear Material* **[Section 4.1.4.3.6]**

4. Industry Standards: (Although these industry standards represent acceptable practices of the nuclear industry, and have been successfully utilized in past licensing actions, in some cases their use has not been endorsed by NRC through a regulation or RG. Further, inclusion in this SRP is not necessarily an endorsement of a particular standard by NRC. Therefore, their use is encouraged, but alternative, equivalent methods may be proposed in the application with adequate justification.)

ANSI N13.30, 1996	*Performance Criteria for Radiobioassay* **[Section 4.1.4.3.8]**
ANSI N13.4-1971	*Specification for Portable X- or Gamma-Radiation Survey Instruments* **[Section 4.1.4.3.11]**
ANSI N13.6-1966 r.1989	*Practice for Occupational Radiation Exposure Records Systems* **[Section 4.1.4.3.9]**
ANSI N13.11-1983	*Dosimetry-Personnel Dosimetry Performance-Criteria for Testing* **[Section 4.1.4.3.7]**
ANSI N13.15-1985	*Radiation Detectors - Personnel Thermoluminescence Dosimetry Systems - Performance* **[Section 4.1.4.3.7]**
ANSI N13.27-1981	*Performance Requirements for pocket-Sized Alarm Dosimeters and Alarm Ratemeters* **[Section 4.1.4.3.7]**
ANSI N42.12-1980	*Calibration and Usage of Sodium Iodide Detector Systems* **[Section 4.1.4.3.11]**
ANSI N42.15-1980	*Performance Verification of Liquid Scintillation Counting Systems* **[Section 4.1.4.3.11]**
ANSI N42.17A-1989	*Performance Specifications for Health Physics Instrumentation - Portable Instrumentation for Use in Normal Environmental Conditions* **[Section 4.1.4.3.11]**
ANSI N42.17B-1989	*Performance Specifications for Health Physics Instrumentation - Occupational Airborne Radioactivity Monitoring Instrumentation* **[Sections 4.1.4.3.5, 4.1.4.3.8, 4.1.4.3.11]**
ANSI N322-1977	*Inspection and Test Specifications for Direct and Indirect Reading Quartz Fiber Pocket Dosimeters* **[Section 4.1.4.3.7]**
ANSI N323-1978 r.1983	*Radiation Protection Instrumentation Tests and Calibrations* **[Sections 4.1.4.3.6, 4.1.4.3.7, 4.1.4.3.11]**
ANSI N542-1977	*Sealed Radioactive Sources Classification* **[Section 4.1.4.3.6]**

ANSI Z88.2-1992	*Practices for Respiratory Protection* **[Section 4.1.4.3.10]**
ANSI Z88.6-1984	*Physical Qualifications for Respirator Use* **[Section 4.1.4.3.10]**
ASTM C986-1989 r.1995	*Developing Training Programs for the Nuclear Fuel Cycle* **[Section 4.1.4.3.4]**

4.1.4.3 Regulatory Acceptance Criteria

4.1.4.3.1 ALARA (As Low as is Reasonably Achievable)

Acceptability of the application should be based on a finding of reasonable assurance that the applicant would meet those requirements of 10 CFR 19.12 and 10 CFR 20.1101(b) related to ALARA, and the following Acceptance Criteria, or information describing acceptable alternatives:

1. Policy Considerations:

   Acceptability should be based on a clear statement in the application of the applicant's policies and provisions for maintaining individual and collective doses at levels that are ALARA, and the approach toward addressing the regulatory guidance of RG 8.10 with regard to the following:

   a. Ensuring that all plant personnel are aware of management's commitment to ALARA.

   b. Ensuring the performance of periodic reviews to determine if doses can be lowered.

   c. Ensuring the qualifications and appropriate staffing of the RS organization.

   d. Ensuring the appropriate authority and independence of the RS manager.

   e. Ensuring that all workers receive sufficient and appropriate initial and periodic training.

   f. Ensuring that modifications to procedures, facilities, and equipment will be justified.

   g. Ensuring that workers and management will be held accountable for their radiological performance.

   h. Ensuring that plant contamination will be minimized, to the extent practicable.

2. Design Considerations:

   Facility design aspects related to ALARA should be reviewed using SRP Section 4.2.

3. Operational Considerations:

Acceptability of the application's ALARA operational considerations should be based on a comparison with the guidance in RG 8.10 related to vigilance of the radiation safety manager (RSM) and RS staff, including the following:

a. RSM and RS staff will periodically review doses associated with procedures, radiation work permits, and ALARA goals to identify trends (with special audits for unusual exposures).

b. Adequate equipment and supplies will be available to the RS staff to perform all personnel dosimetry, environmental monitoring, and bioassay functions.

c. A system of pre-planning work exists such that progressively higher levels of approval will be required for high-dose activities.

d. A system of operational radiological performance goals (also called ALARA goals) is established.

e. The application should contain a commitment to perform trending analyses during operation of the facility. Examples of trend analysis variables are:

   i. Radiation exposures of plant workers and members of the public,

   ii. Concentrations of airborne radioactivity in plant areas,

   iii. Radioactive contamination in plant areas and on equipment,

   iv. Operation/malfunctions of radiation measurement instrumentation and respiratory protection equipment,

   v. Concentrations of radioactive material in gaseous and liquid effluents, and

   vi. Operation of effluent treatment systems (the last two trending parameters are reviewed in SRP Chapter 9.0, but are included here for completeness)

The system for operational ALARA goals should be acceptable if they are specified in the application, along with their bases and a qualitative description of how they will be achieved (i.e., numerical goals are not expected in the application, but a commitment towards achieving ALARA goals and a methodology for achieving them should be described). Acceptable bases for goals could be collective dose, contamination events of skin or clothing, intakes of radioactive material, contamination areas, radioactive waste generation, and liquid and gaseous releases. Goals are acceptable if: (1) they are measurable, realistic, auditable, and challenging; (2) senior management periodically reviews the goals and progress towards meeting them, and (3) they are evaluated and adjusted accordingly on at least an annual basis.

4. ALARA Committee:

The ALARA committee should be acceptable if it is designated and assigned responsibility and authority for implementing ALARA policy, including the following elements:

a. The ALARA committee is shown to have an organizational structure in which RS personnel will interact, in a timely manner, with production personnel to ensure the methods and techniques for reducing occupational dose are incorporated in facility operation

b. The ALARA committee will perform or receive the results of audits of the RS program at least annually, and reviews the results of the RS organization's internal audits

c. The ALARA committee membership should include a chairman, and management or worker representatives from the RS organization, environmental organization, engineering, safety, and production

d. The ALARA committee will evaluate all major design activities, experiments, or plant modifications, and considers the results of the ISA in determining whether further reduction in occupational radiation doses are reasonable

e. The ALARA committee will evaluate trend analyses and the adequacy and implementation of radiological performance (ALARA) goals

f. The reviews and recommendations of the ALARA committee will be documented and tracked to completion.

4.1.4.3.2 Organizational Relationships and Personnel Qualifications

Acceptability of the application should be based on a finding of reasonable assurance that the applicant would meet those requirements of 10 CFR 70.22(a)(6) and 70.23(a)(2) related to Organizational Relationships and Personnel Qualifications, and the following Acceptance Criteria, or information describing acceptable alternatives:

1. The organizational relationships with respect to RS should be acceptable if the RS functions and responsibilities of the RS staff, operations, support, and engineering organizations are clearly identified; and if each position with RS functions including authorities and responsibilities such as those identified in RG 8.10, §C.1(c) is defined and identified. RS functions include those of the RSM, the RS staff (specialists and technicians), the RS engineering function, the RS training function, RS monitoring and surveillance, dosimetry and counting services, and RS auditing.

2. The application should be acceptable if it provides a description of the organizational relationships that are to exist between the positions identified as responsible for RS functions and other (line) managers, and if the plant manager, or equivalent, has overall responsibility and authority for safety.

3. The responsibilities of the RSM (or equivalent) should be acceptable if it is demonstrated that he/she will have direct responsibility for establishing and implementing the RS program, have input to facility design and operational planning, have assigned organizational emergency duties through the site emergency plan, have stop-work authority, will be independent of operations, and have direct access to the plant manager [See RG 8.10 C.1(e)].

4. The functional organization of the RS staff should be acceptable if RS specialists are shown to have responsibility for specific activities assigned to the RS program (e.g., dosimetry, surveys, audits, bioassay, and calibration) with RS technicians implementing these functions.

5. The minimum staffing of the RS organization should be acceptable if it is based on ensuring that, by shift, all routine RS functions can be performed in a timely manner, and that all RS requirements can be met during routine operations, non-routine operations such as anticipated events, and accidents. For periods of extended absence of the RSM (because of vacations, illness, etc.), a qualified substitute should be available to act on his behalf; this includes qualifications for emergency duties.

6. It is acceptable for certain RS technical support or audit activities (e.g., instrument calibration and dosimetry) to be contracted to qualified off-site corporate or consultant organizations. In these cases, acceptability should be based on a determination that these organizations and their responsibilities are specified in the application, along with a demonstration of how the acceptance criteria of this Section are to be satisfied by the contractor.

7. The RS personnel qualifications should be acceptable if they are based on the following education and experience criteria:

    a. the RSM has a bachelor's degree in science or engineering and at least 5 years experience in applied radiological controls at an operating nuclear facility;

    b. RS specialists have a bachelor's degree in science and engineering and at least 1 year of experience in applied radiological controls at an operating nuclear facility; and

    c. RS technicians have a high school diploma or equivalent, technical training commensurate with their assigned duties (dosimetry, bioassay, etc.), and certification in a technician trainee program.

4.1.4.3.3 Radiation Safety Procedures and Radiation Work Permits (RWPs)

Acceptability of the application should be based on a finding of reasonable assurance that the applicant would meet those requirements of 10 CFR 20.1101 related to Radiation Safety Procedures and Radiation Work Permits (RWPs), and the following Acceptance Criteria, or information describing acceptable alternatives:

1. Activities involving exposure to licensed material should be acceptable if performed in accordance with written, approved RS procedures and/or RWPs.

2. Review, revision, and updating of RS procedures and RWPs should be acceptable if performed periodically, to identify situations for reducing doses] at intervals not exceeding 2 years. Procedures should be reviewed and approved by the RSM, or an individual who has the qualifications of the RSM [RG 8.10 §C.2(b)].

3. Development, maintenance, and use of RS procedures and RWPs should be acceptable if performed under appropriate quality assurance (QA) program requirements, in accordance with the applicant's graded QA program (SRP Section 11.1).

4. A mechanism for providing current copies of RS procedures and RWPs to personnel, and a system for ensuring that RWPs are not used past their expiration date, should be established.

5. A system for receiving and reviewing RS related suggestions from employees should be established, and workers are made knowledgeable of this process [RG 8.10 §C.2(b)].

6. The system for implementing RWPs should be acceptable if the applicant specifies:

   a. How a determination is made to use an RWP,
   b. The levels of approval and positions in the organization authorized to approve and issue RWPs,
   c. The types of information included on an RWP (see acceptance criteria that follows),
   d. Provisions for updating/terminating RWPs, including a system to update RWPs when tasks or environmental changes affect worker safety,
   e. Records to be kept for RWPs and retention times, and
   f. Final disposition of RWPs.

7. The applicant should commit to the use of special reviews and approvals before conducting an activity involving licensed materials with an RWP that is not covered by a written radiation safety procedure.

8. Preparation and approval of RWPs should be acceptable if approval is required from other organizational groups, to ensure that provisions of the RWP address all potential hazards (not just radiological hazards) and operations comply with all applicable regulations.

9. The information on RWPs should be acceptable if it is sufficient to allow independent inspection and reconstruction of the circumstances necessitating the RWP, the factors included, and the results.

10. The applicant should commit to a system that ensures that RWPs are not used past their termination dates. The system should include what types of records are to be kept, the retention times for these records, and the final disposition of the RWP. The record system

should be sufficient to allow independent auditors to reconstruct the circumstances necessitating the RWP, the factors included, and the results.

11. The applicant should commit to using RWPs for specific purposes only and RWPs are reissued when significant changes in the task or changes that affect the safety of the worker are made. The application should state that the RWP will include a list of the safety requirements for work conducted under the authorization and include at least the following, as applicable:

    a. The number of and identification of personnel working on the task;
    b. Expected radiological conditions (radiation, contamination, and airborne levels);
    c. Type and frequency of monitoring and dosimetry (e.g., continuous air monitor [CAM], self alarming dosimetry);
    d. Estimated exposure time and doses for the authorization;
    e. Limiting exposure times and doses for the authorization;
    f. Special instructions or equipment (e.g., mock-up required, special shielding required);
    g. Personnel protective equipment (PPE) requirements;
    h. Authorization signature and date;
    i. Actual doses, time, or other information resulting from the completed work authorization are recorded on the RWP (RG 8.10 §C.2(a)); and
    j. Expiration/termination date of the RWP.

4.1.4.3.4 Radiation Training

Acceptability of the application should be based on a finding of reasonable assurance that the applicant would meet those requirements of 10 CFR 19.12, 70.22(a)(6), and 70.23(a)(2) related to RS training, and the following Acceptance Criteria, or information describing acceptable alternatives:

1. Site access should be acceptable if all personnel and visitors entering restricted areas receive either:

    a. A general indoctrination in site-specific safe practices and emergency situations and escort by an individual who has received RS training, or

    b. RS training.

2. Frequency of RS training should be acceptable if given prior to occupational exposure and periodically thereafter (RG 8.29); for TWRS, refresher RS training should be completed not later than 2 years following the most recent RS training (can be a condensed version of initial training with emphasis on changes in policy, procedures, requirements, and facilities). However, retraining for employees authorized to perform "higher-risk" work (e.g., work on glove boxes, in high contamination areas, high radiation area entry, etc.) should be acceptable if they receive annual requalification (ASTM E1168-1995).

3. The process for developing an RS training program should be acceptable to NRC staff if it follows the process outlined in ASTM C986-89 (reapproved 1995). The acceptability of the RS training program objectives, content, testing, requalifications, recordkeeping, and audits should be based on a comparison with the ASTM E1168-1995 standard and Appendix A of RG 8.29. Equivalence should be demonstrated where these standards are not used.

4. The technical content and extent of RS training should be acceptable if it is commensurate with the radiological risk present in the workplace (RG 8.29 and ASTM C986-1995); and is accomplished by grading the training requirements for general employees, radiation workers (possibly more than one type), RS technicians, and supervisors. In addition, training for all groups, except general employee training, should be acceptable if it includes practical demonstrations, by trainees, of proper equipment use, dosimetry use, PPE use, and incident (e.g., spill) response.

5. The verification of received training should be acceptable if each trainee acknowledges in writing that the RS training has been received and understood (RG 8.29), and records of most recent training and testing are maintained as specified in ASTM E1168-1995.

4.1.4.3.5 Air Sampling

Acceptability of the application should be based on a finding of reasonable assurance that the applicant would meet those requirements of 10 CFR 20.1204; 20.1703; 20.1902(d); 20.2103; 20.2110; 20.2203(a)(3)(i)-(ii), (b), and (d); and 10 CFR 70.22(a)(7) related to air sampling, and the following Acceptance Criteria, or information describing acceptable alternatives:

1. The commitment to provide an air sampling program should be acceptable if a program is evidenced that is consistent with the positions in RG 8.25, including evaluating the need for air sampling, locating samplers, sample representativeness, conditions for adjusting derived air concentrations (DACs), measuring sampled air volume, and evaluating results. NUREG-1400 is a sister document to RG 8.25, and presents examples, methods, and techniques for implementing the recommendations of RG 8.25.
2. The basis for the air sampling program should be acceptable if:

    a. For each work area, a determination that the frequency for analyzing airborne levels of radioactivity, counting techniques, action levels and actions to be taken when action levels are exceeded, and alarm set points are adequate to meet Part 20, and

    b. Calculations and verification of airborne concentrations in various areas are controlled under the applicant's QA program (SRP Section 11.1).

3. The use of and specifications for air sampling instrumentation should be acceptable if consistent with RG 8.25 and ANSI N42.17B-1989. Calibration methods and frequencies for air sampling instruments are acceptable if they ensure proper operation of the instrumentation, including the operation of flow rate meters. The use of CAMs is acceptable if the locations of detectors, readouts, annunciators, and alarms are specified.

(This information can be provided in SRP Section 4.2.4.3.1, under plant and process drawings).

4. The use of action levels for airborne activity should be acceptable if a demonstration that the action levels used are appropriate technical criteria to determine the necessary controls, and if the demonstration includes the minimum detectable concentrations for the radionuclides of interest.

4.1.4.3.6 Contamination Control

Acceptability of the application should be based on a finding of reasonable assurance that the applicant would meet those requirements of 10 CFR 20.1406; 20.1501(a)(1) and (a)(2)(ii)-(iii); 20.1703(a)(3)(ii); 20.1901; 20.1902(e); 20.1904; 20.1906; 20.2103; 20.2110; 20.2203(a)(3)(i)-(ii), and (b);and 10 CFR 70.22(a)(7) related to contamination control, and the following Acceptance Criteria, or information describing acceptable alternatives:

1. Facility operating procedures should include procedures that minimize, to the extent practicable, contamination in the facility pursuant to 10 CFR 20.1406; and a commitment to a contamination survey program.

2. The contamination survey program should be acceptable if it is based on the information provided in RG 8.21 on contamination level limits and types, methods, instruments, and frequencies of surveys. Acceptability should be based on specification, for each area, the types of radiation, the criteria for contamination action levels, for both removable and fixed contamination, and the action levels and actions to be taken if exceeded. Contamination surveys should be acceptable if conducted routinely for the accessible areas of the plant site where contamination is likely, if the types of instruments and methods used in the surveys are adequate to allow assessment of working conditions, and if the instruments are sufficiently sensitive to measure contamination at or below the assigned action levels, and tested and calibrated in accordance with ANSI N323 (or equivalent).

3. Features of the facility that help control contamination should be acceptable if consistent with RG 8.21 and included in the facility descriptions (e.g., fume hoods, step-off pads, personnel monitoring equipment at egress points). (This information can be provided in SRP Section 4.2.4.3.1).

4. The policy for controlling contamination should be acceptable if clearly stated, and if it mandates the use of personnel monitoring equipment, and that personnel perform a whole body survey each time they leave a known contamination area, or a minimum hand and shoe survey each time they leave a potentially contaminated restricted area.

5. Access control and security of stored radioactive material should be acceptable if in accordance with Part 20 and if periodic reviews are performed to verify:

   a. Proper posting, labeling, and operability of access controls;

b. Proper identification of restricted areas to prevent the spread of contamination;

c. Sufficient numbers and appropriate locations step-off pads, change facilities, PPE facilities, and personnel monitoring equipment.

6. Removal of equipment and materials from contaminated areas should be acceptable if a system is established to ensure that equipment and materials removed from contaminated areas are not contaminated above specific release levels. The contamination levels of items (tools, equipment, etc.) given release clearance should be acceptable if in accordance with NRC's BTP, "Guidelines for Decontamination of Facilities and Equipment Prior to Release for Unrestricted Use or Termination of Licenses for Byproduct, Source, or Special Nuclear Material."

7. The use of maximum personnel contamination levels for skin and clothing should be acceptable if established and specified, consistent with RG 8.21; and if means are used to detect contamination in excess of these levels, then decontaminate, investigate, correct and document the source, probable cause, and other pertinent information. The minimum detectable levels should be stated.

8. Contamination surveys, investigations, corrective actions, and reviews should be documented, along with deficiencies. This documentation should be reviewed by the RSM for possible trends and needed corrective actions. Contamination levels and contaminated areas should be tracked as part of the ALARA goals (see Section 4.1.4.3.1).

9. The sealed source leak testing program is acceptable if performed in accordance with written procedures in accordance with the 5 NRC BTPs listed in Section 4.1.4.2, and if procedures include acceptable contamination levels, test frequencies, and actions if limits are exceeded.

4.1.4.3.7 External Exposure

Acceptability of the application should be based on a finding of reasonable assurance that the applicant would meet those requirements of 10 CFR 19.13; 10 CFR 20.1201(a)(1)-(2) and (c); 20.1203; 20.1206; 20.1207; 20.1208; 20.1301(a)(1)-(2), (b) and (c); 20.1302(a), (b)(1), and (b)(2)(ii); 20.1501(a)(1), (a)(2)(i) and (c); 20.1502(a); 20.1601; 20.1602; 20.1901; 20.1902(a), (b) and (c); 20.1906; 20.2101; 20.2103; 20.2106; 20.2110; 20.2202(a)-(d); 20.2203(a)(2), (a)(3)(i)-(ii), (b) and (d); 20.2206; and 10 CFR 70.22(a)(7) related to external exposure, and the following Acceptance Criteria, or information describing acceptable alternatives:

1. Acceptable determinations of who are and are not occupationally exposed individuals, and who is to be monitored for exposure are given in RG 8.34. For non-occupationally exposed workers, the limits for members of the public apply, and acceptability is based on compliance with the surveys required by 10 CFR 20.1302.

2. The type, range, sensitivity, accuracy, and frequency for personnel dosimetry and area dosimetry (including extremity dosimetry), and methods for recording measured dose, are

acceptable if stated and justified based on the types, energy and amount of radiation, and consistent with ANSI N13.11-1983, ANSI N13.15-1985, and ANSI N13.27-1981, ANSI N322-1977, and ANSI N323-r1983.

3. Operational planning systems should be acceptable if dosimetry results are used as a tool, and this process is described and justified in the application. An acceptable program should include use of supplemental dosimetry (e.g., dose and dose rate alarming dosimeters) for work in higher radiation areas, as appropriate, as a means to maintain doses at levels that are ALARA.

4. The use of administrative dose levels, below Part 20 limits, is an acceptable approach for demonstrating that doses are maintained ALARA. The application should be acceptable if the administrative limits are specified, are a fraction (e.g., 20 percent) of Part 20 limits, and actions and approvals necessary to exceed administrative dose limits are identified.

5. Processing and evaluation of personnel dosimetry (except those specified in 10 CFR 20.1501(c)) should be acceptable if processed and evaluated by a dosimetry processor holding accreditation from the National Voluntary Laboratory Accreditation Program (NVLAP), and if the technical bases for ensuring the quality of extremity dosimetry is provided in the application (since these dosimeters do not require NVLAP accreditation).

6. The use of planned special exposures (PSEs) should be acceptable if the requirements of 10 CFR 20.1206, 20.2105, and 20.2206 are satisfied, consistent with RG 8.25.

7. The source identification and control program should be acceptable if:

   a. Sources of external exposure throughout the facility are identified along with controls and responsibilities for restricted, controlled, and unrestricted areas;

   b. Methods are identified for materials inventory, movement, and storage, to prevent releases and limit external exposures; and

   c. Receipt and off-site transfer of radioactive materials will comply with 10 CFR 20.1906, 10 CFR Part 71, and U.S. Department of Transportation requirements (49 CFR 171-178).

4.1.4.3.8 Internal Exposure

Acceptability of the application should be based on a finding of reasonable assurance that the applicant would meet those requirements of 10 CFR 19.13; 10 CFR 20.1201(a)(a), (d) and (e); 20.1204; 20.1207; 20.1208; 20.1501(a)(1); 20.1502(b); 20.1703(a)(3)(ii) and (b); 20.1902(d); 20.2101; 20.2103; 20.2105; 20.2106; 20.2202 (a) and (d); 20.2203(a)(2), (b) and (d); and 20.2206 related to Internal Exposure, and the following Acceptance Criteria, or information describing acceptable alternatives:

1. RG 8.9, RG 8.25, and RG 8.34 provide information, recommendations, and guidance that is acceptable to the NRC staff for establishing and implementing a program to monitor internal doses.

2. The internal dose monitoring program should be acceptable if it specifies:

   a. Criteria for participation;

   b. Frequencies of routine measurements;

   c. Use of confirmatory measurements;

   d. Methods to be used;

   e. Minimum detectable concentrations (MDCs);

   f. The action levels and actions to be taken when exceeded;

   g. The methods for determining worker doses from quantities of radionuclides in the body, in the work area air; and/or combinations of these.

3. When air sampling is used for determining worker intake, the application should be acceptable if it specifies the frequency of sampling and data analyses, the MDC, and the action levels and actions taken when exceeded.

4. When bioassay is used to determine worker intake, the application should be acceptable if it specifies the types of bioassay used, the frequency of data collection for each type, the MDCs, and the action levels and actions taken when exceeded; and if the applicant commits to a continuing QA program on all phases of the bioassay program, including sample collection, qualifications of laboratory personnel, laboratory intercomparisons, computational checks, and use of appropriate blanks and standards.

5. Acceptability should be based on statement of a commitment to use engineering controls to limit the intake of radioactive material, including auxiliary ventilation systems (e.g., portable filtration systems) used to control airborne contaminants (e.g., when servicing primary ventilation or machining contaminated surfaces); and containment structures used to protect personnel working in adjacent areas, when feasible.

4.1.4.3.9 Summing Internal and External Exposure

Acceptability of the application should be based on a finding of reasonable assurance that the applicant would meet those requirements of 10 CFR 20.1201(a)(1) and (f); 20.1202; 20.1207; 20.1208; 20.2101; 20.2103; 20.2104; 20.2105; 20.2106; 20.2110; 20.2202(a)-(d); 20.2203(a)(2), (b), and (d); 20.2206; and 10 CFR 70.22(a)(7) related to summing internal and external dose, and the applicant commits to a policy for combining internal and external dose in accordance with RG 8.7, RG 8.34, and RG 8.36.

4.1.4.3.10 Respiratory Protection

Acceptability of the application should be based on a finding of reasonable assurance that the applicant would meet those requirements of 10 CFR 20.1701; 20.1702; 20.1703(a), (c), and (d); and 20.2110 related to respiratory protection, and the following Acceptance Criteria, or information describing acceptable alternatives:

1. The respiratory protection program should be acceptable if it provides for meeting ANSI Z88.2, with defined responsibilities and requirements in the areas of training, control and use of respiratory protection equipment, mask-fit testing, and breathing air purity. (ANSI Z88.6 provides additional guidance generally acceptable to NRC staff for respiratory protection medical qualification and examinations.)

2. The use of respiratory protection equipment should be acceptable if the application describes the equipment used, the conditions under which respiratory protection is required for routine and non-routine operations (including anticipated events and accidents), the protection factors that are applied when respirators are used, the locations of respiratory protection equipment in the plant; and if adequate numbers and locations of respiratory protection equipment and current training are to be maintained as needed to satisfy emergency response functions.

3. Acceptability should be based on the application adequately specifying the methods to determine internal dose when respiratory protection equipment is used, or when engineering and administrative controls for respiratory protection are used. The methods should be acceptable if engineered controls are preferred over respiratory protection equipment, and if factors in the dose calculation include the time of exposure to airborne radioactive materials, the measurement and variability of airborne concentrations of radioactive material during the exposure, and for respirators, the respirator's protection factor and proper fitting.

4.1.4.3.11 Instrumentation

Acceptability of the application should be based on a finding of reasonable assurance that the applicant would meet those requirements of 10 CFR 20.1501(b) and (c) and 20.2103 related to RS instrumentation and the following Acceptance Criteria, or information describing acceptable alternatives:

1. The policy for the maintenance and use of operating radiation instrumentation should be acceptable if the applicant commits to continuing availability of sufficient numbers and types of instruments for all routine (Part 20) and emergency operations. The number and types of instruments should be shown to be acceptable through a list in the application of the types of instruments that are to be available, including ranges, counting modes, sensitivities, alarm set points, planned uses, and calibration frequencies. Acceptability should be based on comparison with the information on radiation measuring instruments and instrument calibration in ANSI N42.17A, ANSI N42.17B, and ANSI N323.

2. The applicant's criteria for selecting radiation measuring instruments and equipment should be acceptable if it facilitates:

   a. Performing radiation and contamination surveys,

   b. Sampling airborne radioactivity,

   c. Monitoring area radiation,

   d. Monitoring personnel,

   e. Performing radioactive analyses, and

   f. High-range, portable instrumentation, with ranges and a justification for them, as necessary to monitor conditions during and after accidents.

3. The applicant's approach toward instrument calibration should be acceptable if all instruments are to be calibrated at least semi-annually, and recalibrated if the equipment is repaired such that accuracy could be affected.

4. RS procedures should be acceptable (with respect to RS instrument checks) if they establish daily operational checks of continuously operating RS instruments.

5. The facilities related to RS instrumentation should be acceptable if the applicant identifies the locations of, and describes the following:

   a. a radiochemistry laboratory equipped to perform the analyses required by 10 CFR 20.1501;

   b. a low-background counting room equipped to perform routine counting of all plant samples (water, swipes, air); and

   c. instrument storage, calibration, decontamination, and maintenance facilities.

4.1.5 REVIEW PROCEDURES

4.1.5.1 Acceptance Review

The primary reviewer should evaluate the application to determine whether it addresses the "Areas of Review" discussed in Section 4.1.3, above. If significant deficiencies are identified, the applicant should be requested to submit additional material before the start of the safety evaluation.

4.1.5.2 Safety Evaluation

After determining that the application is acceptable for review in accordance with Section 4.1.5.1, above, the primary reviewer should perform a safety evaluation against the acceptance criteria described in Section 4.1.4. If during the course of the safety evaluation, the primary reviewer determines the need for additional information, the primary reviewer should coordinate a request for additional information with the licensing project manager. The primary reviewer of this SRP section should coordinate the efforts of the secondary reviewers identified in Section 4.1.2, as specified below. The final step would be the preparation of the safety evaluation report (SER) input by the primary reviewer, for the licensing project manager, in accordance with Section 4.1.6, "Evaluation Findings."

The following items should be noted regarding the relationships between the primary reviewer and the secondary reviewers for this SRP section, in performing the safety review:

1. The review performed in this section pertains to programmatic aspects of occupational doses during routine operations and anticipated events. Doses from accidents are reviewed under the SRP chapter dealing with the ISA (SRP Chapter 3.0) and the Radiation Safety Design Features Section (SRP Section 4.2). Doses to the public and the environment, including ALARA, are the subject of SRP Chapter 9.0, "Environmental Protection."

2. The plant organization, functional responsibilities, and qualifications of personnel are also reviewed as part of the SRP chapters on Organization and Administration (SRP Chapter 2.0) and Training and Qualifications (SRP Section 11.4). Applicants may choose to provide the information in this section explicitly, or by providing a reference to those chapters. The primary reviewer of this section coordinates with the primary reviewers of the other chapters to verify the completeness and consistency of the information, and that the acceptance criteria are satisfied.

3. The RS training program and the respiratory protection training program could be described by the applicant in the SRP Section on Training and Qualifications (SRP Section 11.4). Applicants may choose to provide the information in this section explicitly, or by providing a reference to that section. The primary reviewer of this section uses the acceptance criteria in this section to evaluate these commitments, regardless of where they appear in the application.

4.1.6 EVALUATION FINDINGS

The primary reviewer should write an SER section that addresses each topic reviewed under this SRP section and explains why the NRC staff has reasonable assurance that the radiation safety program part of the application is acceptable. License conditions may be proposed to impose requirements where the application is deficient. The SER should include a summary statement of what was evaluated and the basis for the reviewers' conclusions.

The staff can document the evaluation as follows:

> *The staff has evaluated .....* [Insert a summary statement of what was evaluated and why the reviewer finds the submittal acceptable.] *The applicant has committed to an acceptable radiation safety program that includes: (1) an effective documented program to ensure that occupational radiological exposures are ALARA; (2) an organization with adequate qualification requirements for the radiation safety personnel; (3) approved written radiation safety procedures or RWPs for radiation safety activities; (4) radiation safety training for all personnel who have access to restricted areas; (5) requirements for radiological air sampling; (7) requirements for control of radiological contamination within the facility; (8) programs for monitoring personnel external and internal radiation exposure; (9) a respiratory protection program; and (10) requirements for radiological measurement instrumentation.*
>
> *The NRC staff concludes, with reasonable assurance, that the applicant's radiation safety program is adequate and that the applicant has the necessary technical staff to administer an effective radiation safety program that meets the requirements of 10 CFR Parts 19, 20, and 70. Conformance to the application and license conditions should ensure safe operation and provide early detection of unfavorable trends to allow prompt corrective action.*

4.1.7 REFERENCES

All referenced documents in the Acceptance Criteria for this review area have been listed in Section 4.1.4.2, and are not repeated here. However, in addition to those documents, the following documents contain information that is specific to nuclear reactors, but which is also relevant to this review area. Applicants may choose to reference portions of these documents in the SAR, with adequate justification.

1. Nuclear Regulatory Commission (U.S.) (NRC). Regulatory Guide 1.33, Rev. 2, "Quality Assurance Program Requirements (Operational)." NRC: Washington, D.C. February 1978.

2. Nuclear Regulatory Commission (U.S.) (NRC). Regulatory Guide 8.8, Rev. 3, "Information Relevant to Ensuring that Occupational Radiation Exposures at Nuclear Power Stations will be as Low as is Reasonably Achievable." NRC: Washington, D.C. June 1978.

3. Nuclear Regulatory Commission (U.S.) (NRC). Regulatory Guide 1.97, Rev. 3, "Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident." NRC: Washington, D.C. May 1983.

RADIATION SAFETY
4.2 RADIATION SAFETY DESIGN FEATURES

4.2.1 PURPOSE OF REVIEW

The purpose of this review is to determine with reasonable assurance that the applicant's design is adequate to protect the radiological health and safety of the workers and to comply with the regulatory requirements of 10 CFR Parts 20 and 70, during routine and non-routine operations including anticipated events. This chapter also facilitates the review of the radiation safety aspects of accidents that are analyzed in the integrated safety analysis (ISA), through an interface with SRP Chapter 3.0.

The protection of members of the public and control of effluent releases is not included in this chapter but is in SRP Chapter 9.0, "Environmental Protection." While this chapter reviews the applicant's radiation safety (RS) *design*, the applicant's RS *program* and administrative controls are reviewed under SRP Chapter 4.1, "Radiation Safety Program."

4.2.2 RESPONSIBILITY FOR REVIEW

Primary: Health Physicist

Secondary: None

Supporting: Licensing Project Manager
Lead reviewer of SRP Chapter 4.1 if different then primary reviewer
Fire Protection Engineer (primary reviewer of SRP Chapter 7.0)
Primary reviewers of Chapter 12

4.2.3 AREAS OF REVIEW

Engineered controls that provide for radiological safety are required to be established and implemented by 10 CFR 20.1101. (As used in this SRP the terms *Radiation Safety* and *Radiation Protection* are synonymous). Six elements of the applicant's proposed RS design features are reviewed by the staff, as identified in the following list.

1. Facility Design Features

   Areas to be reviewed should include the applicant's proposed equipment and facility design features and plant layout as they relate to occupational RS and ALARA concepts. Consistent with maintaining doses at levels that are ALARA, the incorporation of features to minimize contamination and waste production, and facilitate ease of operations, maintenance, replacement, and decommissioning, are also reviewed.

2. Source Identification

Areas to be reviewed should include the applicant's description of the sources of radiation and contamination in the plant during routine and non-routine operations (e.g., maintenance) including anticipated events. The applicant's description of the sources of radiation and contamination that are used in accident analyses in Chapter 3.0, "ISA," should also be reviewed. Areas to be reviewed should include the pertinent information needed for:

a. Input to shielding codes used in the design process;
b. Establishing related facility design features;
c. Plans and procedures development; and
d. Assessment of occupational dose.

The methodology for estimating source magnitudes and locations, at the design stage, after several years of plant operation, and incorporating this information into the design should also be reviewed.

3. ALARA Design Considerations

Areas to be reviewed should include the applicant's organizational relationships and responsibilities with respect to performing radiological design reviews; the application of ALARA into design-stage man-rem estimates, the descriptions and elements of the design review process for RS, and how experience from past designs and from operating plants has been used to develop improved RS design, when ALARA threshold values are exceeded.

4. Ventilation Systems

Areas to be reviewed should include the design and operation of the ventilation systems, as related to radiological safety, including the proposed design objectives, minimum flow velocity at hood openings, the types of filters and the maximum differential pressure across filters, and the frequency and types of tests required to ensure ventilation system performance.

5. Shielding Evaluations

Areas to be reviewed should include the applicant's proposed uses of permanent and temporary radiation shielding as part of the RS program. The information on the shielding design objectives, the types of shielding materials to be used, special analyses of features such as cell penetrations, the determination of requirements in work areas, and the methods (e.g., codes) by which those requirements are satisfied should also be reviewed.

6. Integrated Safety Analysis (ISA)

Areas to be reviewed should include the postulated accidents in the ISA which have RS consequences for the workers, environment, and public. Areas reviewed for the ISA results include all high and a sample of lower risk accident sequences that result in radiation doses of concern. The methodology in assessing the accident consequences, the likelihood, and the risk index associated with each of these accident sequences are also reviewed. In particular, the primary reviewer of this SRP chapter should focus on the ISA source term, transport, and dosimetry analyses. Controls established by the applicant to prevent or mitigate each accident sequence, and the levels of assurance applied to the controls should be reviewed in the context of radiological safety.

4.2.4 ACCEPTANCE CRITERIA

4.2.4.1 Regulatory Requirements

Regulations applicable to this SRP chapter are listed below [followed in brackets by the applicable acceptance criteria sections]:

Code of Federal Regulations, *Title 10, Energy*, Part 20, "Standards for Protection Against Radiation."

§ 20.1101 *Radiation Protection Programs* **[Sections 4.2.4.3.1, 4.2.4.3.3, 4.2.4.3.4, 4.2.4.3.5]**

§ 20.1201 Occupational Dose Limits For Adults **[Sections 4.2.4.3.1, 4.2.4.3.4, 4.2.4.3.5]**

§ 20.1301 *Dose Limits for Individual Members of the Public* **[Sections 4.2.4.3.1, 4.2.4.3.4, 4.2.4.3.5]**

§ 20.1406 *Minimization of Contamination* **[Sections 4.2.4.3.1, 4.2.4.3.3]**

§ 20.1501 *Surveys - General*, Subsection (a) **[Sections 4.2.4.3.3, 4.2.4.3.4, 4.2.4.3.5]**

§ 20.1601 *Control of Access to High Radiation Areas* **[Sections 4.2.4.3.1]**

§ 20.1602 *Control of Access to Very High Radiation Areas* **[Sections 4.2.4.3.1]**

§ 20.1701 *Use of Process or Other Engineering Controls* **[Section 4.2.4.3.4]**

Code of Federal Regulations, *Title 10, Energy,* Part 70, "Domestic Licensing of Special Nuclear Material."

Radiation Safety

§ 70.22 *Contents of Applications*, Subsections (a)(4) and (a)(7) **[Sections 4.2.4.3.1, 4.2.4.3.2, 4.2.4.3.3, 4.2.4.3.4, 4.2.4.3.5]**

§ 70.23 *Requirements for Approval of Applications*, Subsection (a)(3) **[Section 4.2.4.3.1]**

Nuclear Regulatory Commission (U.S.), Washington, D.C. "Domestic Licensing of Special Nuclear Material (10 CFR Part 70)." *Federal Register*: Vol. 64, No. 146. pp. 41338–41357. July 30, 1999.

§ 70.61 *Performance Requirements* **[Section 4.2.4.3.6]**

§ 70.62 *Safety Program and Integrated Safety Analysis* **[Section 4.2.4.3.6]**

§ 70.65 *Additional Content of Applications* **[Section 4.2.4.3.6]**

4.2.4.2 Regulatory Guidance

NRC Regulatory Guides (RGs), NUREG reports, and industry standards that provide a generally acceptable basis to the NRC staff for satisfying the regulatory requirements listed in Section 4.2.4.1 are listed below [followed in brackets by the applicable acceptance criteria sections].

1. NRC Regulatory Guides (RGs)

   RG 8.10, Rev. 1-R Sept 1975 *Operating Philosophy for Maintaining Occupational Radiation Exposures as Low as is Reasonably Achievable* **[Sections 4.2.4.3.2 and 4.2.4.3.3]**

   RG 8.19, Rev. 1 June 1979 *Occupational Radiation Dose Assessment in Light-Water Reactor Power Plants - Design Stage Man-Rem Estimates* **[Sections 4.2.4.3.2 and 4.2.4.3.3]**

2. NRC NUREG Reports

   NUREG-1513 (DRAFT 1998) *Integrated Safety Analysis Guidance Document* **[Section 4.2.4.3.6]**

3. Industry Standards: (Although these industry standards represent acceptable practices of the nuclear industry, and have been successfully utilized in past licensing actions, their use has not been endorsed by NRC through a regulation or RG. Further, inclusion in this SRP is not necessarily an endorsement of a particular standard by NRC. Therefore, alternative but equivalent methods may be proposed in the application with adequate justification.)

   ANSI/ANS-6.1.1-1991 *Neutron and Gamma-Ray Fluence-to-Dose Factors* **[Sections 4.2.4.3.3. and 4.2.4.3.5]**

ANSI/ANS-6.1.2-1991	*Neutron and Gamma-Ray Cross Sections for Nuclear Radiation Protection Calculations for Nuclear Power Plants* **[Section 4.2.4.3.5]**
ANSI/ANS-6.4-1985	*Guidelines on the Nuclear Analyses and Design of Concrete Radiation Shielding for Nuclear Power Plants* **[Section 4.2.4.3.5]**
ANSI/ANS-6.4.2-1985	*Specification of Radiation Shielding Materials* **[Section 4.2.4.3.5]**
ANSI/ASME N510-1980	*Testing of Nuclear Air Cleaning Systems* **[Section 4.2.4.3.4]**
ERDA 76-21	Nuclear Air Cleaning Handbook, C. A. Burchsted, A. B. Fuller, J. E. Kahn **[Section 4.2.4.3.4]**

4.2.4.3 Regulatory Acceptance Criteria

4.2.4.3.1 Facility Design Features

Acceptability of the radiation safety design should be based on a finding of reasonable assurance that the applicant would meet those requirements of 10 CFR 20.1101(b), 20.1201, 20.1301, 20.1406, 20.1601, 20.1602, and 10 CFR 70.22(a)(7) and 70.23(a)(3) related to facility design features for RS, and the following Acceptance Criteria, or information describing acceptable alternatives:

1. The plant and process drawings and descriptions should be acceptable if they identify clearly-readable and scaled RS design features that are:

   a. Relied on to reduce doses to meet Part 20 during routine and non-routine operations (including anticipated events); and/or

   b. Identified by the ISA as items relied on for safety to reduce accident doses.

   The identification of these features should be acceptable if they include:

   a. Locations of detectors and alarm systems;

   b. Locations of permanent shielding (including penetrations, labyrinths, shield doors, etc.);

   c. Provisions for installation/removal of temporary shielding;

   d. Locations and access control points for restricted areas, high radiation, and very high radiation areas;

   e. Change rooms, showers, and locker rooms;

f. The contamination control, decommissioning facilitation, and waste minimization design features required by 10 CFR 20.1406. Shield wall thicknesses for all shielded spaces should be specified on the drawings or provided in separate tables. (Note that this information can be included here or through a reference to information provided for the acceptance criteria in SRP Chapter 3.0.)

2. The predicted radiation doses from licensed activities should be acceptable if they are within the limits of Part 20, including ALARA as required by 10 CFR 20.1101(b), as evidenced in the application by a summary figure or table of predicted annual occupational doses for the types of work functions (e.g., operations, routine maintenance, special maintenance, in-service testing and surveillance, and waste management) provided at the facility.

3. Access controls for high and very high radiations areas should be acceptable if they meet 10 CFR 20.1601 and 20.1602, respectively. For general radiation areas, change rooms are provided for changing into personnel protective equipment (PPE). Change rooms should be adjacent to shower and decontamination facilities and be provided with ventilation systems that filter dispersable radionuclides. Administrative (i.e., programmatic) aspects of access control and storage are reviewed under SRP Section 4.1.5.8, "Contamination Control."

4.2.4.3.2 Source Identification

Acceptability of the application should be based on a finding of reasonable assurance that the applicant would meet those requirements of 10 CFR 70.22(a)(4) and (a)(7), related to specifying the types, form, and amount of licensed material to be used at the facility; and the following Acceptance Criteria, or information describing acceptable alternatives:

1. External Dose Considerations: Acceptability of contained radiation sources descriptions should be based on quantitative descriptions and estimates of contained sources being provided (RG 8.10, Position C.2(a)) and used as the basis for the RS program and for shield design calculations, with consideration of routine and nonroutine operations, including anticipated events and accident conditions. The descriptions are acceptable if they include isotopic composition, locations in the plant, source strength and source geometry, and the basis for the values used in the application.

2. Internal Dose Considerations: Acceptability of contained radiation sources descriptions should be based on quantitative descriptions and estimates of contained sources being provided (RG 8.10, Position C.2(a)) and used as the basis for the internal RS program and for design of the ventilation systems, with consideration of routine and nonroutine operations and accident conditions. The descriptions should be acceptable if they include:

   a. Tabulations of the calculated concentrations of radioactive material, by nuclide, expected during routine and non-routine operations including anticipated events, and

accident conditions identified in the ISA, for equipment cubicles, corridors, and operating areas normally occupied by operating personnel;

b. The models and parameters for the calculations.

3. The contained and airborne radioactivity sources estimated at the design stage should be based on an assumption of several years of facility operation, to account for the buildup of radioactivity and contamination in the plant. These source estimates should also account for the variability of the radioactive properties of the Hanford tank wastes. The application should be acceptable if the specific assumptions, a discussion of uncertainties, and a justification of each assumptions' conservatism are provided.

4.2.4.3.3 ALARA Design Considerations

Acceptability of the application should be based on a finding of reasonable assurance that the applicant would meet those requirements of 10 CFR 20.1101(b), 20.1406, 20.1501, and 10 CFR 70.22(a)(7) related to ALARA design considerations, and the following Acceptance Criteria, or information describing acceptable alternatives:

1. The applicant's design and design activities, with respect to RS, should be acceptable if they are described in the application and are evidenced by provisions to ensure:

   a. The incorporation of measures for reducing the need for time spent in radiation areas;

   b. Measures to improve the accessibility to components requiring periodic maintenance or inservice inspection;

   c. Measures to reduce the distribution and retention of radioactive materials throughout plant systems;

   d. Measures to control (reduce) contamination, facilitate decommissioning, and minimize secondary radioactive waste production in accordance with 10 CFR 20.1406;

   e. Measures instructing designers and engineers in ALARA design objectives;

   f. Measures incorporating experience from operating plants and past designs; and

   g. Commitment to, and description of, continuing RS (ALARA) design reviews for facility or process modifications made during construction and operations.

2. The RS (ALARA) design review process should be acceptable if:

   a. The organizational responsibilities and relationships associated with these reviews and related dose assessments are described;

b. Design reviews and dose assessments are performed by competent personnel including (or with concurrence of) RS staff and RS management;

c. Design reviews include review of previous jobs, designs, operating experience and processes for applicability and improvements;

d. Design reviews include documentation (e.g., ALARA Design Review Checklists) and tracking of recommendations to completion; and

e. Design reviews and approvals required are graded based on the hazard (e.g., are compared to defined ALARA trigger levels). Note that some of this information can be included under SRP Section 4.1.4.3.1.

3. A self-assessment of the submitted plant design, shielding, layout, traffic patterns, expected maintenance, and sources, should be performed and described in the application, and is acceptable if the assessment supports that both collective and individual doses from significant activities will be ALARA for routine and non-routine operations including anticipated events. For purposes of design stage estimates, significant activities could be defined as dose-causing activities conservatively estimated to result in greater than 0.01 person-sievert (1.0 person-rem) per year.

4. The process for seeking RS related design improvements should be acceptable if the application includes a description of how RS related design improvements are sought, considered, and incorporated where practicable (RG 8.10, Position C.1(f)). Acceptability at the design stage should be based on the description of the methods for design stage person-rem estimates and dose assessments; the methods and tables in RG 8.19 are generally acceptable.

4.2.4.3.4 Ventilation

A ventilation system is necessary to provide confinement integrity and to process off-gas before being exhausted to the environment. The review performed in this SRP section concerns those functions of the ventilation and air cleaning system that pertain to occupational RS (specifically, controlling internal dose through limiting airborne radioactivity). Ventilation systems will have many other functions than controlling internal radiation exposure to workers through containment (e.g., off-gas management, prevention of hydrogen gas buildup, heating and air conditioning, accident functions, controlling chemical exposures, reducing effluent releases, etc.). Applicable acceptance criteria for functions other than RS of ventilation and air treatment systems, and construction and performance specifications of ducts, blowers, and filters; are provided in the SRP Chapter 12.0, "Plant Systems."

Acceptability of the application should be based on a finding of reasonable assurance that the applicant would meet those requirements of 10 CFR 20.1101(b), 20.1201, 20.1301, 20.1501, 20.1701; and 10 CFR 70.22(a)(7), related to designing and operating ventilation systems to control internal radiation doses, and the following Acceptance Criteria, or information describing acceptable alternatives:

1. Acceptability should be based on a demonstration that the design and operation of the ventilation system protects workers and public from airborne radioactive material such that limits of 10 CFR Part 20 will not be exceeded during routine and non-routine operations and anticipated events. Recommendations for the design, construction, and testing of nuclear air cleaning systems (e.g., zoning, moisture separation, HEPA filtration, operational/maintenance considerations, etc.) that are generally acceptable to NRC staff are provided in ERDA 76-21.

2. Design objectives for ventilation systems should be acceptable if they are stated and ensure that:

   a. During routine and non-routine operations and anticipated occurrences, airborne concentrations in occupied operating areas are well below the limits of 10 CFR Part 20 Appendix B;

   b. The use of engineering (i.e., design) controls shall be preferred over the use of respirators (10 CFR 20.1701);

   c. Airflow patterns are from areas of lesser contamination potential to areas of greater contamination potential, with periodic checks that ensure that design pressure differentials are maintained; and

   d. Items relied on for safety allow for routine in-place testing of HEPA filtration systems as outlined in ASME N510.

3. The specifications for ventilation system performance should be acceptable with respect to RS, if they include minimum flow velocity at openings of hoods, maximum differential pressure across filters for operability, types of filters to be used, the frequency and types of tests required to measure ventilation system performance, the acceptance criteria, and the actions to be taken if the acceptance criteria are not satisfied.

4. Air monitoring and warning systems associated with the ventilation system, that are required to function during a loss of power, are acceptable if (in addition to performing their specified functions) they are provided with an uninterruptable power supply, unless they can tolerate a temporary loss of function without loss of data, and are provided with a stand-by power supply. Readouts for air monitoring and alarm systems should be acceptable if, in addition to local alarms, central readout and alarm is provided that is accessible during accidents. Certain programmatic aspects of air monitoring and warning systems are reviewed under SRP Section 4.1, "Radiation Safety Program."

4.2.4.3.5 Shielding

The review criteria below for shielding apply only to TWRS unless otherwise noted upon further understanding of the AVLIS design.

Radiation Safety

Acceptability of the application should be based on a finding of reasonable assurance that the applicant would meet those requirements of 10 CFR 20.11o1(b), 20.1201, 20.1301, 20.1501(a), and 10 CFR 70.22(a)(7) related to designing and providing shielding from external radiation sources, and the following Review Criteria, or information describing acceptable alternatives:

1. Facility descriptions (e.g., facility layout diagrams submitted for SRP Section 1.1 or Chapter 3.0) should be acceptable if they describe, in detail, use of and locations where permanent shielding has been included into design to lower dose rates to comply with 10 CFR Part 20 during routine and non-routine operations and anticipated events. Acceptability should also be based on the description of areas that have been provided by design to facilitate installation and removal of temporary shields for non-routine operations. (Where temporary shielding is to be used, local audible and visible alarming radiation monitors should be installed to alert personnel if shielding is not present, consistent with the external radiation hazard).

2. Shielding provided and/or installed to minimize nonpenetrating external radiation doses, including that to the skin, extremities, and lens of the eye (e.g., for glove box operations with significant dose contributions from Sr-90/Y-90 or bremsstrahlung radiation) should be acceptable if the shielding and features such as penetrations meet design goals and are described in sufficient detail to verify results.

3. The derivation of permanent or temporary shielding requirements and specifications should be acceptable if based on design objectives that are identified in the application. Dose or dose-rate design objectives should be acceptable if specified and based on fractions of Part 20 limits and personnel occupancy predictions, for both continually and intermittently occupied areas of the facility. Occupancy accounts for duration and frequency of exposures, and also accounts for the fact that doses in particular areas may either be occupational (radiation worker) or non-occupational (general employee). An objective, for design purposes, of 20 percent of the applicable annual limits in 10 CFR Part 20 (e.g., 1.0 rem/yr for restricted areas), accounting for occupancy estimates, is acceptable to the staff. For continuously occupied areas, this translates to an average dose rate of 0.5 mrem/hr (20 percent of the occupational dose limit of 5 rem in a 2000 hour work-year). (These objectives are comparable to the design limits of 10 CFR 835.1002.) Notwithstanding this design objective, administrative controls would need to supplement the design objective to further reduce doses consistent with ALARA. Another acceptable design objective is that the use of straight-line penetrations of shield walls should be minimized.

4. Adequacy of provided shielding should be acceptable if, for each instance of shielding associated with reducing doses from high or very high radiation areas, the shielding used and features such as penetrations, shield doors, and labyrinths meet design goals and are described in sufficient detail to verify results. Adequate attenuation can be demonstrated by: (a) analyses (calculations), or (b) reference to similar configurations that were previously analyzed or experimentally verified provided that this reference is clear and specific.

5. Where used, analyses for calculating shielding requirements should be acceptable if described and comparable to commonly acceptable shielding calculations, and if realistic assumptions are used regarding source terms, cross sections, shield and source geometries, and transport methods. Codes used should rely on the use of flux-to-dose conversion factors of ANSI/ANS 6.1.1 and cross sections of ANSI/ANS-6.1.2. (recommends ENDF/B library). Generally, only Monte-Carlo calculational methods would be acceptable to NRC staff for analyses of complex geometries (e.g., shield penetrations). Analyses descriptions are acceptable if provided in sufficient detail to allow independent confirmatory calculations.

6. Selection of shielding materials and decisions between permanent or temporary shielding should be acceptable if they consider facilitation of decommissioning and waste minimization, in accordance with §20.1406, as one design consideration. Descriptions of the physical and nuclear properties of shielding materials used for various functions in the plant should be acceptable if consistent with ANSI/ANS-6.4.2.

7. In cases where the confinement barrier or process equipment provides the primary shielding and is relied on for safety as determined by the ISA, the quality assurance program is applied to all aspects of the shielding design, procurement, installation, maintenance, etc. For shielding that is relied on for safety, the design and analyses approaches used by the applicant should be described; for concrete, the methods in ANSI/ANS-6.4-1985 should be acceptable.

8. The applicant should commit to and describe a radiation shielding test program that will verify the efficacy of installed shielding materials in meeting the radiation shielding design goals and the regulatory external dose requirements of Part 20. The objective of this effort should be to verify that sufficient shielding has been provided (particularly with regard to penetrations, labyrinths, shield doors, etc.) for the life of the plant, prior to initiation of operations; and to verify that design models and calculations are representative of actual operating conditions with respect to occupational RS.

4.2.4.3.6 Integrated Safety Analyses (ISA)

Acceptability of the application should be based on a finding of reasonable assurance that the applicant would meet those requirements of 10 CFR 70.61, 70.62 and 70.65; the guidance in NUREG-1513 (DRAFT), and the following Acceptance Criteria, or information describing acceptable alternatives. RS assessments that support the ISA should be acceptable if they:

1. Use appropriate and verified assessment methods, computer codes, and literature values.

2. Consider a complete range of credible accident sequences that could adversely affect radiological exposures and cause the consequences of concern.

3. Reasonably estimate radiological consequences (considering source term, transport, and dosimetry) of accident sequences.

4. Identify effective controls to prevent and mitigate accident sequences and radiological consequences of concern.

5. Describe and commit to appropriate management control systems to ensure the continued availability and reliability of safety controls to prevent and mitigate radiological consequences of concern.

4.2.5 REVIEW PROCEDURES

4.2.5.1 Acceptance Review

The primary reviewer should evaluate the application to determine whether it addresses the "Areas of Review" discussed in Section 4.2.3, above. If significant deficiencies are identified, the applicant should be requested to submit additional material before the start of the safety evaluation.

4.2.5.2 Safety Evaluation

After determining that the application is acceptable for review in accordance with Section 4.2.1, above, the primary reviewer should perform a safety evaluation against the acceptance criteria described in Section 4.2.4. The primary reviewer of this SRP chapter coordinates the efforts of the secondary reviewers identified in Section 4.2.2. If necessary, a request for additional information to the applicant should be coordinated with the licensing project manager. The final step should be the preparation of the safety evaluation report (SER) input by the primary reviewer, for the licensing project manager, in accordance with Section 4.2.6, "Evaluation Findings."

The following items should be noted regarding the relationships between the primary reviewer and the secondary reviewers for this SRP chapter in performing the safety review:

1. While this chapter addresses the applicant's RS *design*, the applicant's RS *program* and administrative controls are reviewed under SRP Chapter 4.1, "Radiation Safety Program." However, certain aspects of the program, such as facility access controls, zoning, and security of stored material, can not be cleanly categorized into either "design" or "program." Review of these areas should be coordinated with the reviewer of SRP Section 4.1, "Radiation Safety Program," since they are partially included in SRP Section 4.2.4.3.1, and in SRP Section 4.1.4.3.6 as part of the review of contamination controls.

2. The information in Section 4.2.4.3.1, regarding the facility and process design drawings and descriptions, could be included by a reference to SRP Chapter 1.1, "Facilities and Process Description," or SRP Chapter 3.0, "Integrated Safety Analyses," (which requires additional process description information through 10 CFR Part 70, as revised). The primary reviewer of this SRP chapter should perform the safety evaluation of this information as it pertains to RS, regardless of where it appears in the license application.

3. The RS aspects of the ventilation and air cleaning systems that are reviewed by the primary reviewer of this SRP chapter, should be coordinated with the primary reviewer of SRP Chapter 12.0, "Plant Systems," for the non-RS related aspects of the ventilation and air cleaning systems, to verify that adequate and consistent information was provided.

4.2.6 EVALUATION FINDINGS

The staff's evaluation should verify that the license application provides sufficient information to satisfy the regulatory requirements of Section 4.2.4.1 and that the regulatory acceptance criteria in Section 4.2.4.3 have been appropriately considered in satisfying the requirements. On the basis of this information, the staff should conclude that this evaluation is complete. The reviewers should write material suitable for inclusion in the SER prepared for the entire application. The SER should include a summary statement of what was evaluated and the basis for the reviewers' conclusions.

The staff can document the evaluation as follows:

The applicant has supplied information on the radiation safety design features and design process for the [insert facility], that demonstrate, with reasonable assurance, that radiation doses will be within the limits of 10 CFR Part 20 and will be as low as is reasonably achievable (ALARA). [Insert a summary statement of what was evaluated and why the reviewer finds the submittal acceptable.] *The applicant has considered contamination control, decommissioning facilitation, and waste minimization, in developing the design features of the facility, as required by 10 CFR 20.1406. Many of the radiation safety design features have been incorporated as a result of the applicant's radiation safety design review and from radiation dose experience gained during the operation of other facilities. [Include examples of design features incorporated to reduce contamination and radiation dose to workers during maintenance operations, reduce radiation sources where operations must be performed, allow quick entry and easy access, provide remote operation capability or reduce the time required for work in radiation fields, and examples of other features that reduce radiation exposure of personnel.]*

The applicant has made estimates of facility radiation sources capable of producing significant radiation levels, and significant airborne radioactivity, based on (include the applicant's basis for radiation and airborne source terms). These estimates demonstrate a conservative approach and are acceptable.

The applicant has described organizational relationships and responsibilities with respect to performing radiological design reviews, that ensure the adequate application of ALARA in design stage activities, and to plant modifications made during construction and operations.

The general shielding design and analysis methodology used by the applicant is consistent with industry practice and is acceptable. The applicant has provided an adequate treatment of features requiring special analyses, such as cell penetrations, and has shown by calculation that doses in work areas meet requirements. The basic radiation transport analysis used for the applicants' shield design is based on (list appropriate shielding computer codes used).

The ventilation system at (plant name) is designed to ensure that plant personnel are not inadvertently exposed to airborne contaminants exceeding those given in 10 CFR Part 20. The applicant intends to maintain personnel exposures as low as is reasonably achievable by: (1) maintaining air flow from areas of potentially low airborne contamination to areas of higher potential concentrations; (2) ensuring negative or positive pressures to prevent exfiltration or infiltration of potential contaminants; and (3) locating ventilation system intakes so that intake of potentially contaminated air from other building exhaust points is minimized.

The NRC staff concludes that there is reasonable assurance that the applicant's radiation safety design process and design features are adequate and, in concert with an effective radiation safety program of SRP Section 4.1, satisfy the requirements of 10 CFR Parts 20 and 70.

4.2.7 REFERENCES

All referenced documents in the Acceptance Criteria for this review area have been listed in Section 4.2.4.2, and are not repeated here. However, in addition to those documents, the following references contain information that is specific to nuclear reactors (or other nuclear facilities), but which is also relevant to this review area. Applicants may choose to reference portions of these documents in the SAR, with adequate justification, and provide that these references are clear and specific.

1. Nuclear Regulatory Commission (U.S.) (NRC). Regulatory Guide 1.33, Rev. 2, "Quality Assurance Program Requirements (Operational)." NRC: Washington, D.C. February 1978.

2. Nuclear Regulatory Commission (U.S.) (NRC). Regulatory Guide 8.8, Rev. 3, "Information Relevant to Ensuring that Occupational Radiation Exposures at Nuclear Power Stations will be as Low as is Reasonably Achievable." NRC: Washington, D.C. June 1978.

3. International Committee on Radiological Protection (ICRP) Publication 55, "Optimization and Decision Making in Radiological Protection." ICRP: Oxford. 1989.

4. American National Standards Institute/American Society of Mechanical Engineers, ANSI/ASME N509-1989, "Nuclear Power Plant Air Cleaning Units and Components." ANS: LaGrange Park, Illinois.

5. Nuclear Regulatory Commission (U.S.) (NRC). Regulatory Guide 1.97, Rev. 3, "Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident." NRC: Washington, D.C. May 1983.

6. American National Standards Institute/American Nuclear Society, ANSI/ANS 6.3.1-1987, "Program for Testing Radiation Shields in Light Water Reactors (LWR)." ANS: LaGrange Park, Illinois.

NUCLEAR CRITICALITY SAFETY (NCS)

5.1 PURPOSE OF REVIEW

The purpose of this review is to determine whether the applicant, in the license application and supported by materials on the docket, has made the appropriate commitments to develop, implement, and maintain an NCS program in support of safe operation of the facility as required generally by Federal Regulations and specifically by 10 CFR 70.24, 70.61, 70.62, 70.64, and 70.65.

5.2 RESPONSIBILITY FOR REVIEW

Primary: Nuclear Process Engineer (NCS Reviewer)

Secondary: None

Supporting: Project Manager, Site Representative, and Fuel Cycle Inspector

5.3 AREAS OF REVIEW

The staff should review the application to determine whether (1) the applicant has identified and committed to the responsibilities and authorities for individuals to develop and implement the NCS program; (2) the facility management measures described in 10 CFR 70.62 have been committed to and will support implementing and maintaining the NCS program; (3) an adequate NCS program is described which includes identifying and committing to the Methodologies and Technical Practices used to ensure the safe operation of the facility as required by 10 CFR 70.24 [Criticality Accident Alarm System (CAAS)], 10 CFR 70.61 [Subcriticality of Operations and Margin of Safety for Subcriticality], 10 CFR 70.64 [Baseline Design Criteria (BDC)], and 10 CFR 70.65 [ISA Summary].

The specific areas for review are as follows:

5.3.1 Organization and Administration

The Primary Reviewer should review the application to determine if the applicant has identified and committed to the responsibilities and authorities for individuals to develop and implement the NCS program. The following areas of the application related to the applicant's Organization and Administration should be reviewed:

1. For familiarity, the general Organization and Administration methods used by the applicant (see Section 2.0).

2. The areas of review listed in Section 2.3.1 (Organization and Administration) as they relate to NCS.

3. Experience and education requirements of NCS management positions.

5.3.2 Management Measures

The Primary Reviewer should review the application to determine whether the facility management measures in 10 CFR 70.62 have been committed to by the applicant and whether they demonstrate the applicant's ability to implement and maintain the NCS program. The following areas of the application related to the applicant's Management Measures should be reviewed:

1. Configuration Management, Procedures, Audits and Assessments, Incident Investigations, and other quality assurance elements used by the applicant (see SRP Sections 11.1 through 11.9).

2. The Training, Procedures, and Audits and Assessments programs specifically related to NCS.

5.3.3 Methodologies and Technical Practices

The Primary Reviewer should review the application to determine whether the applicant has implemented NCS Methodologies and NCS Technical Practices used to make NCS determinations to ensure the safe operation of the facility as required by 10 CFR 70.24 [CAAS], 10 CFR 70.61(d) [Subcriticality of Operations and Margin of Safety for Subcriticality], 10 CFR 70.64(a)(9) [BDC], and 10 CFR 70.65(b) [ISA Summary]. The following areas of the application related to the applicant's NCS Methodologies and NCS Technical Practices should be reviewed:

1. The commitment to use the NCS Methodologies identified by the applicant's NCS program.

2. The commitment to use the NCS Technical Practices identified by the applicant's NCS program.

3. The commitment to fulfill the requirements of 10 CFR 70.24 (CAAS) and to have a CAAS that has been incorporated into the Management Measures.

4. The commitment to detect an inadvertent nuclear criticality and promptly notify personnel to ensure that the radiation exposure to workers shall be minimized.

5. The commitment to the requirements of 10 CFR 70.61 (Subcriticality of Operations and Margin of Subcriticality for Safety).

6. The commitment to the requirements in 10 CFR 70.64 (BDC) as they relate to NCS for new facilities and processes.

7. The areas of review listed in Section 3.3 (ISA Summary) as they relate to NCS.

5.4 ACCEPTANCE CRITERIA

To provide for NCS, the applicant's use of standards should be considered acceptable if the applicant has met the following Acceptance Criteria:

If an applicant intends to conduct activities where a standard applies and the standard has been endorsed by an NRC Regulatory Guide, then a commitment to comply with all of the requirements (i.e., "shalls") and the appropriate recommendations (i.e., "shoulds") of the standard should constitute an acceptable program under the NRC regulations with respect to the safety aspects addressed by the standard. Notwithstanding such a general commitment to a standard, the licensee should clarify broad requirements in the standard by more specific commitments in the application. Any variations from the requirements of the standard should be identified and justified in the application.

Individual commitments to the Acceptance Criteria are expected only when the Acceptance Criteria are relevant to the operations and materials to be licensed.

5.4.1 Regulatory Requirements

The regulatory basis for the review should be the general and additional contents of an application as required by 10 CFR 70.22 and 70.65, respectively. In addition, the NCS review should be conducted to ensure compliance with 10 CFR 70.24, 70.61, and 70.62.

5.4.2 Regulatory Guidance

The NRC Regulatory Guide (RG) 3.71,"Nuclear Criticality Safety Standards for Fuels and Materials Facilities," August 1998, endorses the ANSI/ANS-8 national standards listed below in part or in full.

1. ANSI/ANS-8.1-1983 (Reaffirmed in 1988), "Nuclear Criticality Safety in Operations with Fissionable Materials Outside Reactors."

2. ANSI/ANS-8.3-1997, "Criticality Accident Alarm System."

3. ANSI/ANS-8.5-1996, "Use of Borosilicate-Glass Raschig Rings as a Neutron Absorber in Solutions of Fissile Material."

4. ANSI/ANS-8.6-1983 (Reaffirmed in 1995), "Safety in Conducting Subcritical Neutron-Multiplication Measurements In Situ."

5. ANSI/ANS-8.7-1975 (Reaffirmed in 1987), "Guide for Nuclear Criticality Safety in the Storage of Fissile Materials."

6. ANSI/ANS-8.9-1987 (Reaffirmed in 1995), "Nuclear Criticality Safety Criteria for Steel-Pipe Intersections Containing Aqueous Solutions of Fissile Materials."

7. ANSI/ANS-8.10-1983 (Reaffirmed in 1988), "Criteria for Nuclear Criticality Safety Controls in Operations With Shielding and Confinement."

8. ANSI/ANS-8.12-1987 (Reaffirmed in 1993), "Nuclear Criticality Control and Safety of Plutonium-Uranium Fuel Mixtures Outside Reactors."

9. ANSI/ANS-8.15-1981 (Reaffirmed in 1995), "Nuclear Criticality Control of Special Actinide Elements."

10. ANSI/ANS-8.17-1984 (Reaffirmed in 1997), "Criticality Safety Criteria for the Handling, Storage, and Transportation of LWR Fuel Outside Reactors."

11. ANSI/ANS-8.19-1996, "Administrative Practices for Nuclear Criticality Safety."

12. ANSI/ANS-8.20-1991, "Nuclear Criticality Safety Training."

13. ANSI/ANS-8.21-1995, "Use of Fixed Neutron Absorbers in Nuclear Facilities Outside Reactors."

14. ANSI/ANS-8.22-1997, "Nuclear Criticality Safety Based on Limiting and Controlling Moderators."

15. ANSI/ANS-8.23-1997, "Nuclear Criticality Accident Emergency Planning and Response."

5.4.3 Regulatory Acceptance Criteria

5.4.3.1 Organization and Administration

To provide for NCS, the applicant's Organization and Administration should be considered acceptable if the applicant has met the following Acceptance Criteria or has identified and justified an alternative in the application:

1. The applicant meets the Acceptance Criteria related to NCS in Section 2.4.3 (Organization and Administration).

2. The applicant commits to the requirements in ANSI/ANS-8.1-1983, "Nuclear Criticality Safety in Operations with Fissionable Materials Outside Reactors."

3. The applicant commits to the requirements in ANSI/ANS-8.19-1996, "Administrative Practices for Nuclear Criticality Safety."

4. The applicant commits to the intent of Section 4.11 of ANSI/ANS-8.1-1983, which is: The applicant shall commit to the use of personnel, skilled in the interpretation of data pertinent

to NCS and familiar with the operation of the facility, as a resource in NCS management decisions. These specialists should be independent of operations supervision.

5. The applicant commits to provide NCS postings, as necessary, for areas, operations, work stations, and storage locations to provide operators a reference for ensuring conformance and safe operation.

6. The applicant commits to the policy that: "All personnel shall report defective NCS conditions to the NCS function and shall take no further action, unless specified by approved written procedures, until the NCS function has analyzed the situation."

5.4.3.2 Management Measures

To provide for NCS, the applicant's Management Measures required by 10 CFR 70.62 should be considered acceptable if the applicant has met the following Acceptance Criteria or has identified and justified an alternative in the application:

1. Training:

   a. The applicant commits to the requirements in both ANSI/ANS-8.19-1996, "Administrative Practices for Nuclear Criticality Safety" and ANSI/ANS-8.20-1991, "Nuclear Criticality Safety Training."

   b. The applicant commits to provide instruction in the Training program regarding the use of Process Variables as NCS controls.

   c. The applicant commits to provide instruction in the Training program to all personnel to (1) recognize the CAAS signal and (2) evacuate promptly to a safe area.

   d. The applicant commits to provide instruction in the Training program to all personnel regarding the policy that: "All personnel shall report defective NCS conditions to the NCS function and take no further action, unless specified by approved written procedures, until the NCS function has analyzed the situation."

2. Procedures:

   a. The applicant commits to the requirements in ANSI/ANS-8.19-1996, "Administrative Practices for Nuclear Criticality Safety."

   b. The applicant commits to the policy that: "No single, inadvertent departure from a procedure could cause an inadvertent nuclear criticality."

3. Audits and Assessments:

   a. The applicant commits to the requirements in ANSI/ANS-8.19-1996, "Administrative Practices for Nuclear Criticality Safety."

b. The applicant commits to conducting and documenting Weekly NCS Walkthroughs (e.g., checklists) of all operating SNM process areas such that all operating SNM process areas should be reviewed at least every two weeks. Identified weaknesses should be incorporated into the facility Corrective Actions Program and should be promptly and effectively resolved. A graded approach may be used to justify an alternate plan based on the ISA.

c. The applicant commits to conducting and documenting Quarterly NCS Audits such that all NCS aspects of Management Measures (see Sections 11.1 through 11.9) should be audited at least every 2 years. A graded approach may be used to justify an alternate plan based on the ISA.

5.4.3.3.1 Methodologies

To provide for NCS, the applicant's commitment to NCS Methodologies should be considered acceptable if the applicant has met the following Acceptance Criteria or has identified and justified an alternative in the application:

1. The applicant commits to the requirements in ANSI/ANS-8.1-1983, "Nuclear Criticality Safety in Operations with Fissionable Materials Outside Reactors."

2. The applicant commits to the intent of the requirement in Regulatory Guide 3.71, "Nuclear Criticality Safety Standards for Fuels and Materials Facilities" related to validation reports which is: The applicant should demonstrate: (1) the adequacy of the Margin of Subcriticality for Safety by assuring that the margin is large compared to the uncertainty in the calculated value of k-eff, (2) that the calculation of k-eff is based on a set of variables whose values lie in a range for which the methodology used to determine k-eff has been validated, and (3) that trends in the bias support the extension of the methodology to areas outside the Area(s) of Applicability.

3. The applicant includes a reference to (including date and revision number) and summary description of either a manual or a documented, reviewed, and approved validation report (by NCS and Management) for each methodology which will be used to make an NCS determination (e.g., experimental data, reference books, hand calculations, deterministic computer codes, probabilistic computer codes). The summary description of the reference manual or validation report should have:

   a. A summary of the theory of the methodology in sufficient detail, clarity, and lack of ambiguity that allows understanding of the methodology.

   b. A commitment to apply the methodology only in the Area(s) of Applicability or provide justifications for applying the methodology outside the Area(s) of Applicability.

   c. A commitment to use pertinent computer codes, assumptions, and techniques in the methodology.

d. A commitment to use mathematical relationships only within the context of their fundamental assumptions and limitations.

e. A commitment to use the data consistently with reliable experimental measurements.

f. A commitment to use benchmark experiments that cover the intended ranges of applicability and data derived therefrom that will be used to validate the methodology.

g. A commitment to determine the bias, uncertainty in the bias, uncertainty in the methodology, uncertainty in the data, uncertainty in the benchmark experiments, and Margin of Subcriticality for Safety, when using the methodology.

h. A commitment to use controlled software and hardware when using the methodology.

i. A commitment to use a verification process when using the methodology.

4. The applicant commits to have, at the facility, the reference manual or documented, reviewed, and approved validation report (by NCS and Management) for each methodology used to make an NCS determination. The manual or validation report should have:

   a. A description of the theory of the methodology in sufficient detail, clarity, and lack of ambiguity that allows understanding of the methodology and independent duplication of results.

   b. A description of the Area(s) of Applicability which identifies the range of values for which valid results have been obtained for the parameters used in the methodology. In accordance with the provisions in ANSI/ANS-8.1-1983, "Nuclear Criticality Safety in Operations With Fissionable Material Outside Reactors," any extrapolation beyond the Area(s) of Applicability should be supported by an established mathematical methodology.

   c. A description of the use of pertinent computer codes, assumptions, and techniques in the methodology.

   d. A description of the proper functioning of the mathematical operations in the methodology (e.g., mathematical testing).

   e. A description of the data used in the methodology consistent with reliable experimental measurements.

   f. A description of the benchmark experiments that cover the intended range of applicability and data derived therefrom that were used for validating the methodology.

g. A description of the bias, uncertainty in the bias, uncertainty in the methodology, uncertainty in the data, uncertainty in the benchmark experiments, and Margin of Subcriticality for Safety, as well as the basis for these items, as used in the methodology. If the bias is determined to be advantageous to the applicant, the applicant shall use a bias of 0.0 (e.g., in a critical experiment where the k-eff is known to be 1.0 and the code calculates 1.02, the applicant cannot use a bias of 0.02 to allow calculations to be made above the value of 1.0).

h. A description of the software and hardware that will use the methodology.

i. A description of the verification process and results.

5. The applicant commits to incorporate each reference manual or documented, reviewed, and approved validation report (by NCS and Management) for a methodology as well as assumptions used into the facility Configuration Management program.

6. The applicant commits to performing NCS determinations using specified methods. The applicant should commit to incorporating these methods into the facility Management Measures:

   a. The applicant should commit to assuming credible optimum conditions (i.e., most reactive conditions physically possible or limited by written commitments to regulatory agencies) for each Controlled Parameter unless specified controls are implemented to limit the Controlled Parameter to a certain range of values.

   b. The applicant should commit to set NCS operating and safety limits derived from experimental data, reference books, hand calculations, deterministic computer codes, or probabilistic computer codes which have either a reference manual or a documented, reviewed, and approved validation report (by NCS and Management).

   c. The applicant should commit to consider the variability and uncertainty in a process and the NCS subcritical limit when setting NCS safety limits.

   d. The applicant should commit to consider the variability and uncertainty in a process and the NCS safety limit when setting NCS operating limits.

5.4.3.3.2 Technical Practices

To provide for NCS, the applicant's commitment to NCS Technical Practices should be considered acceptable if the applicant has met the following Acceptance Criteria or has identified and justified an alternative in the application:

1. Although the applicant may use a single NCS control to maintain the values of two or more Controlled Parameters, this use constitutes only one component necessary for Double Contingency Protection.

2. Based on the Performance Requirements in 10 CFR 70.61, the applicant commits to the policy that: "No single credible event or failure could result in a criticality accident."

3. The applicant commits to the preferred use of Passive-Engineered controls to ensure NCS. The applicant should commit to the following preference, in general, for controls to ensure NCS: (1) Passive-Engineered, (2) Active-Engineered, (3) Augmented-Administrative, and (4) Simple-Administrative. When choosing not to use a Passive-Engineered control, the applicant commits to identify and provide justification in the ISA.

4. When evaluating a Controlled Parameter, heterogeneous effects are considered. Heterogeneous effects are particularly relevant for low-enriched uranium processes, where, when all other parameters are equal, heterogeneous systems are more reactive than homogeneous systems.

5. The applicant commits to incorporate Controlled Parameters into the facility Management Measures of 10 CFR 70.62.

6. The applicant commits to perform an evaluation, for all Controlled Parameters, that shows that during both normal and credible abnormal conditions, the Controlled Parameter will be maintained.

7. The applicant commits to describe Controlled Parameters used as NCS control. Examples of Controlled Parameters available for NCS control are: Mass, Geometry, Density, Enrichment, Reflection, Moderation, Concentration, Interaction, Neutron Absorber, and Volume.

8. When Controlled Parameters are controlled for safety reasons by measurement, reliable methods and instruments should be used. It is acceptable if the applicant commits to representative sampling, reliable measurement instruments and methods, and dual independent measurements where there is significant susceptibility to human error.

9. The use of Mass as a Controlled Parameter should be considered acceptable if:

   a. When a given Mass of material has been determined, a percentage factor is used to determine the Mass percentage of SNM in that material.

   b. When fixed geometric devices are used to limit the Mass of SNM, a conservative process density is used.

   c. When physical measurement of the Mass is needed, the measurement is obtained by using instrumentation.

   d. When double batching of SNM is possible, the Mass of SNM is limited to no more than 45 percent of the minimum critical Mass based on spherical geometry.

e. When double batching of SNM is not possible, the Mass of SNM is limited to no more than 75 percent of the critical Mass.

10. The use of Geometry as a Controlled Parameter should be considered acceptable if:

a. Before beginning operations, all dimensions and nuclear properties which use Geometry control are verified. The facility Configuration Management program should be used to maintain these dimensions and nuclear properties.

b. When using large single units, the Margins of Safety are 90 percent of the minimum critical cylinder diameter, 85 percent of the minimum critical slab thickness, and 75 percent of the minimum critical sphere volume.

11. The use of Density as a Controlled Parameter should be considered acceptable if:

a. When Process Variables can affect the Density, the Process Variables are identified as items relied on for safety (IROFS) in the ISA Summary.

b. When physical measurement of the Density is needed, the measurement is obtained by using instrumentation.

12. The use of Enrichment as a Controlled Parameter should be considered acceptable if:

a. When using SNM with differing Enrichment, the SNM is segregated by Enrichment.

b. When physical measurement of the Enrichment is needed, the measurement is obtained by using instrumentation.

13. The use of Reflection as a Controlled Parameter should be considered acceptable if:

a. When investigating an individual unit, the wall thickness of the unit and all reflecting adjacent materials of the unit are considered. The adjacent materials should be farther than one foot away from the unit.

b. After identifying potential reflectors, the controls to prevent the presence of the potential reflectors are identified as IROFS in the ISA Summary.

14. The use of Moderation as a Controlled Parameter should be considered acceptable if:

a. When using Moderation, the applicant commits to the requirements in ANSI/ANS-8.22-1997, "Nuclear Criticality Safety Based on Limiting and Controlling Moderators."

b. When Process Variables can affect the Moderation, the Process Variables are identified as IROFS in the ISA Summary.

c. When physical measurement of the Moderation is needed, the measurement is obtained by using instrumentation.

d. When designing physical structures, the design precludes the ingress of Moderation.

e. When sampling of the Moderation is needed, the sampling program uses dual independent sampling methods.

f. When developing firefighting procedures for use in a Moderation controlled area, restrictions are placed on the use of Moderator material.

g. After evaluating all credible sources of Moderation for the potential for intrusion into a Moderation controlled area, the ingress of Moderation is precluded or controlled.

15. The use of Concentration as a Controlled Parameter should be considered acceptable if:

a. When Process Variables can affect the Concentration, the Process Variables are identified as IROFS in the ISA Summary.

b. High Concentrations of SNM in a process are precluded unless the safety basis for operation at high concentrations is adequate as evaluated in the ISA and reliable controls are instituted.

c. When using a tank containing Concentration controlled solution, the tank is normally closed so that only proper procedural transfers of concentration-controlled solutions are allowed.

d. When sampling of the Concentration is needed, the sampling program uses dual independent sampling methods.

e. After identifying possible precipitating agents, precautions are taken to ensure that such agents will not be inadvertently introduced.

16. The use of Interaction as a Controlled Parameter should be considered acceptable if:

When maintaining a physical separation between units, engineered devices (i.e., spacers) with a minimum spacing are used. The structural integrity of the spacers should be sufficient for normal and credible abnormal conditions. Augmented administrative controls (e.g., visible marks with proper spacing) may be acceptable if adequately justified in the ISA summary.

17. The use of Neutron Absorber as a Controlled Parameter should be considered acceptable if:

a. When using Borosilicate-Glass Raschig Rings, the applicant commits to the requirements in ANSI/ANS-8.5-1996, "Use of Borosilicate-Glass Raschig Rings as a Neutron Absorber in Solutions of Fissile Material."

b. When using Fixed Neutron Absorbers, the applicant commits to the requirements in ANSI/ANS-8.21-1995, "Use of Fixed Neutron Absorbers in Nuclear Facilities Outside Reactors."

c. When evaluating absorber effectiveness, neutron spectra are considered (e.g., cadmium is an effective absorber for thermal neutrons, but ineffective for fast neutrons).

18. The use of Volume as a Controlled Parameter should be considered acceptable if:

a. When using Volume control, geometrical devices are used to restrict the Volume of SNM and engineered devices should limit the accumulation of SNM.

b. When physical measurement of the Volume is needed, the measurement is obtained by using instrumentation.

5.4.3.3.3 Requirements of 10 CFR 70.24 (CAAS)

To provide for NCS, the applicant's commitment to the CAAS requirements in 10 CFR 70.24 should be considered acceptable if the applicant has met the following Acceptance Criteria or has identified and justified an alternative in the application:

1. The applicant has documented that the facility CAAS meets the requirements of 10 CFR 70.24.

2. The applicant commits to the requirements in ANSI/ANS-8.3-1997, "Criticality Accident Alarm System."

3. The applicant commits to the requirements in Regulatory Guide 3.71, "Nuclear Criticality Safety Standards for Fuels and Materials Facilities" which effect the ANSI/ANS-8.3 standard:

a. At or above the 10 CFR 70.24 mass limits, CAAS coverage shall be required in each area in which SNM is handled, stored, or used.

b. 10 CFR 70.24 requires that each area that needs CAAS coverage to be covered by two detectors.

c. 10 CFR 70.24 requires that a CAAS be capable of detecting a nuclear criticality that produces an absorbed dose in soft tissue of 20 rads of combined neutron and gamma radiation at an unshielded distance of 2 meters from the reacting material within 1 minute.

4. The applicant commits to having a CAAS that is uniform throughout the facility for the type of radiation detected, the mode of detection, the alarm signal, and the system dependability.

5. The applicant commits to having a CAAS that is designed to remain operational during credible events such as a seismic shock equivalent to the site-specific design-basis earthquake or the equivalent value specified by the Uniform Building Code.

6. The applicant commits to having a CAAS that is designed to remain operational during credible events such as a fire, an explosion, a corrosive atmosphere, and other credible conditions.

7. The applicant commits to having a CAAS alarm that is clearly audible in areas that must be evacuated or provides alternate notification methods that are documented to be effective in notifying personnel that evacuation is necessary.

8. The applicant commits to rendering operations safe, by shutdown and quarantine if necessary, in any area where CAAS coverage has been lost and not restored within a specified number of hours. The number of hours should be determined on a process by process basis because shutting down certain processes, even to make them safe, may carry a larger risk, than being without a CAAS for a short time. The applicant should commit to compensatory measures (e.g., limit access, halt SNM movement) when the CAAS system is not functioning due to Maintenance.

9. Emergency Management:

    a. The applicant commits to the requirements in ANSI/ANS-8.23-1997, "Nuclear Criticality Accident Emergency Planning and Response."

    b. The applicant either has an Emergency Plan or satisfies the alternate requirements found in 70.22(i)(1)(i).

    c. The applicant commits to provide fixed and personnel accident dosimeters in areas that require a CAAS, as well as a method for prompt onsite dosimeter readouts. These dosimeters should be readily available to personnel responding to an emergency.

    d. The applicant commits to provide emergency power for the CAAS.

5.4.3.3.4 Requirements of 10 CFR 70.61 (Subcriticality of Operations and Margin of Subcriticality for Safety)

To provide for NCS, the applicant's commitment to the Subcriticality of Operations and Margin of Safety for Subcriticality requirements in 10 CFR 70.61 should be considered acceptable if

the applicant has met the following Acceptance Criteria or has identified and justified an alternative in the application:

1. The applicant commits to the use of NCS controls and Controlled Parameters to ensure both Subcriticality of Operations and Margin of Subcriticality for Safety. As required by ANSI/ANS-8.1-1983, "Nuclear Criticality Safety in Operations with Fissionable Materials Outside Reactors," process specifications shall incorporate margins to protect against uncertainties in process variables and against a limit being accidentally exceeded."

2. The applicant commits to the requirements in ANSI/ANS-8.7-1975, "Guide for Nuclear Criticality Safety in the Storage of Fissile Materials."

3. The applicant commits to the requirements in ANSI/ANS-8.9-1987, "Nuclear Criticality Safety Criteria for Steel-Pipe Intersections Containing Aqueous Solutions of Fissile Materials."

4. The applicant commits to the requirements in ANSI/ANS-8.10-1983, "Criteria for Nuclear Criticality Safety Controls in Operations With Shielding and Confinement."

5. The applicant commits to the requirements in ANSI/ANS-8.12-1987, "Nuclear Criticality Control and Safety of Plutonium-Uranium Fuel Mixtures Outside Reactors."

6. The applicant commits to the requirements in ANSI/ANS-8.15-1981, "Nuclear Criticality Control of Special Actinide Elements."

7. The applicant commits to the requirements in ANSI/ANS-8.17-1984, "Criticality Safety Criteria for the Handling, Storage, and Transportation of LWR Fuel Outside Reactors."

8. If the applicant intends to use administrative k-eff margins for normal and credible abnormal conditions, the applicant commits to NRC pre-approval of the administrative margins.

9. The applicant commits to the use of controls or control barriers on IROFS to ensure that an inadvertent nuclear criticality will not occur.

10. The applicant commits to incorporating controls and control barriers into the facility Management Measures of 10 CFR 70.62.

11. The applicant commits to determining subcritical limits for k-eff calculations such that : k-subcritical = 1.0 - bias - margin, where margin includes adequate allowance for uncertainty in the methodology, data, and bias to assure subcriticality.

12. The applicant commits to performing studies to correlate the change in a value of a Controlled Parameter and its k-eff value. The studies should also include changing the value of one Controlled Parameter and determining its effect on another Controlled Parameter and k-eff.

13. The applicant meets the Acceptance Criteria in Section 3.4.1 (ISA Summary) as they relate to Subcriticality of Operations and Margin of Subcriticality for Safety.

Note: This is the Acceptance Criteria to review the High-Risk Accident Sequences and a cross-section of Low-Risk Accident Sequences.

5.4.3.3.5 Requirements of 10 CFR 70.64 (BDC) [for new facilities and processes only]

To provide for NCS, the applicant's commitment to the BDC requirements in 10 CFR 70.64 should be considered acceptable if the applicant has met the following Acceptance Criteria or has identified and justified an alternative in the application:

The applicant commits to the Double Contingency Principle in determining NCS controls in the design of new facilities or new processes at existing facilities.

5.4.3.3.6 Requirements of 10 CFR 70.65 (ISA Summary)

The applicant is required to meet the performance criteria in 10 CFR 70.61(b) and (c) as well as the performance requirements in 70.61(d), which include the requirement to limit the risk of an inadvertent nuclear criticality by assuring that all nuclear processes remain subcritical. The applicant's evaluation of NCS Accident Sequences should be performed in a manner consistent with the applicant's evaluation of non-NCS Accident Sequences used to meet 10 CFR 70.61(b) and (c); however 10 CFR 70.61(d) requires the applicant to use prevention methods as the primary means to meet the performance requirements of 10 CFR 70.61(b) and (c).

To provide for NCS, the applicant's commitment to the ISA requirements in 10 CFR 70.65 should be considered acceptable if the applicant has met the following Acceptance Criteria or has identified and justified an alternative in the application:

1. Accident Sequences:

   a. The applicant meets the Acceptance Criteria in Section 3.4.3 (ISA Summary) related to Accident Sequences for NCS.

   b. The applicant commits to use Appendix A of ANSI/ANS-8.1-1983, "Nuclear Criticality Safety in Operations with Fissionable Materials Outside Reactors" in determining Accident Sequences.

2. Consequences:

   a. The applicant meets the Acceptance Criteria in Section 3.4.3 (ISA Summary) related to Consequences for NCS.

b. The applicant commits to the requirements in ANSI/ANS-8.10-1983, "Criteria for Nuclear Criticality Safety Controls in Operations With Shielding and Confinement." In addition, the applicant should commit to the requirements in RG 3.71, "Nuclear Criticality Safety Standards for Fuels and Materials Facilities" which effect the ANSI/ANS 8.10 standard.

3. Likelihoods:

a. The applicant meets the Acceptance Criteria in Section 3.4.3 (ISA Summary) related to Likelihoods for NCS.

b. The applicant commits to implement an NCS program that ensures Double Contingency Protection when practicable. When evaluating Double Contingency Protection, the term "unlikely" should be used in a manner consistent with ANSI/ANS-8.1-1983.

i. Adherence to Double Contingency Protection: Each process which could have an inadvertent nuclear criticality should have Double Contingency Protection. Double Contingency Protection may be provided by either (a) At Least Two Parameter Control: the control of at least two independent process parameters or (b) Single Parameter Control: a system of multiple independent controls on a single process parameter. The At Least Two Parameter Control method is the preferred approach due to the difficulty of preventing common-mode failure when controlling only one parameter.

ii. As used in Double Contingency Protection, the term "concurrent" means that the effect of the first process change persists until a second change occurs, at which point the process could have an inadvertent nuclear criticality. It does not mean that the two events initiating the change must occur simultaneously. The possibility of an inadvertent nuclear criticality can be markedly reduced if failures of NCS controls are rapidly detected and the processes rendered safe. If not, processes can remain vulnerable to a second failure for extended periods of time.

iii. If the applicant adheres to Double Contingency Protection for an NCS Accident Sequence, then the Likelihood requirements of 10 CFR 70.61(b) should be considered satisfied for that Accident Sequence.

iv. Exceptions to Double Contingency Protection: There may be processes where Double Contingency Protection is not practicable. In those processes, the facility should implement sufficient Redundancy and Diversity in Controlled Parameters such that at least two unlikely and concurrent events, errors, accidents, or equipment malfunctions, are necessary before an inadvertent nuclear criticality is possible. The applicant should commit in the license application to identify and provide justification in the ISA for exceptions to Double Contingency Protection.

4. Risk:

   The applicant meets the Acceptance Criteria in Section 3.4.1 (ISA Summary) related to Risks for NCS.

5. IROFS:

   The applicant meets the Acceptance Criteria in Section 3.4.1 (ISA Summary) related to IROFS for NCS.

5.5 REVIEW PROCEDURES

The reviewer should use the Regulatory Guidance of this chapter; references in this chapter; the applicant's 91-01, 70.50, and 70.74 reports; and 10 CFR Part 70 Appendix A reporting requirements.

5.5.1 Acceptance Review

The Primary Reviewer should review the applicant's NCS information for completeness with respect to the requirements in 10 CFR 70.22, 70.24, 70.61, 70.62, 70.64, and 70.65 and the Acceptance Criteria in Section 5.4. If deficiencies are identified, then either the applicant should be requested to submit additional material prior to the start of the safety evaluation or the application should be denied.

5.5.2 Safety Evaluation

When an acceptable application is received from the applicant, the primary reviewer will conduct a complete review of the application and determine its acceptability, consulting with the supporting reviewers to identify and resolve any issues of concern related to the licensing review. The primary reviewer (acting as a secondary or supporting reviewer) should also coordinate with other reviewers concerning NCS regarding the following:

1. In support of the primary reviewer for Section 2.0, the NCS reviewer should determine whether the Acceptance Criteria in Section 2.0 have been met as they relate to NCS.

2. In support of the primary reviewer for Sections 11.1 through 11.9, the NCS reviewer should determine whether the Acceptance Criteria in Sections 11.1 through 11.9 have been met as they relate to NCS.

3. In support of the primary reviewer for Section 3.0, the NCS reviewer should determine whether the Acceptance Criteria in Chapter 3.0 have been met as they relate to NCS.

4. In support of the primary reviewer for Section 8.0, the NCS reviewer should determine whether the Acceptance Criteria in Section 8.0 have been met as they relate to NCS.

The primary reviewer should determine whether the Acceptance Criteria in Section 5.4 have been met and should prepare the SER NCS chapter in accordance with Section 5.6.

5.6 EVALUATION FINDINGS

If the staff's review verifies that sufficient information has been provided in the safety program description to satisfy the Acceptance Criteria in Section 5.4, the staff should document its review as follows:

The staff has reviewed the Nuclear Criticality Safety (NCS) program for [name of facility] according to Chapter 5.0 of the Standard Review Plan. The staff has reasonable assurance that:

1. The applicant will have in place a staff of managers, supervisors, engineers, process operators, and other support personnel who are qualified to develop, implement, and maintain the NCS program in accordance with the facility Organization, Administration, and Management Measures.

2. The applicant's conduct of operations will be based on NCS Methodologies and NCS Technical Practices which will ensure that the fissile material will be possessed, stored, and used safely according to the requirements in 10 CFR Part 70.

3. The applicant will develop, implement, and maintain a Criticality Accident Alarm System in accordance with the requirements in 10 CFR 70.24 and in accordance with its Emergency Management Program.

4. The applicant will have in place an NCS program in accordance with the Subcriticality of Operations and Margin of Subcriticality for Safety requirements in 10 CFR 70.61 and Baseline Design Criteria requirements in 10 CFR 70.64.

5. Based on this review, the staff concludes that the applicant's NCS program meets the requirements of 10 CFR Part 70 and provides reasonable assurance for the protection of public health and safety, including workers and the environment.

Note: The Evaluation Finding for the ISA Summary requirements for 10 CFR 70.65 should be in SRP Section 3.6.

5.7 REFERENCES

1. Code of Federal Regulations, *Title 10, Energy,* Part 70, "Domestic Licensing of Special Nuclear Material."

2. Paxton, H.C and Pruvost, N.L. *Critical Dimensions of Systems Containing 235U, 239Pu, and 233U.* LA-10860-MS. Los Alamos National Laboratory: Los Alamos, NM, 1987.

3. Pruvost, N.L and Paxton, H.C. *Nuclear Criticality Safety Guide.* LA-12808/UC-714. Los Alamos National Laboratory: Los Alamos, NM, 1996.

4. Clark, H.K. *Maximum Safe Limits for Slightly Enriched Uranium and Uranium Oxide.* DP-1014. Du Pont de Nemours and Co.: Aiken, SC, 1966.

5. Stratton, W.R. (Revised by D. R. Smith). *A Review of Criticality Accidents.* DOE/NCT-04. U.S. Department of Energy. March 1989.

6. Knief, R.A. *Nuclear Criticality Safety -- Theory and Practice.* American Nuclear Society: La Grange Park, IL, 1985.

7. DOE Order 420.1 (Change 2). "Facility Safety." October 24, 1996.

CHEMICAL SAFETY

6.1 PURPOSE OF REVIEW

The purpose of this review is to establish that there is reasonable assurance that the applicant has designed a facility that provides for adequate protection against chemical hazards related to the storage, handling, and processing of radioactive material as required by the Baseline Design Criterion for Chemical Protection, in 10 CFR 70.64 as revised[1].

Safety issues are initially evaluated as part of the applicant's Integrated Safety Analysis (ISA), which identifies potential accidents at the facility (SRP Chapter 3). Chemical safety addresses the consequences of potential accidents due to hazardous chemicals and accidents due to chemicals that create potentially hazardous situations (e.g., an inerting gas incapacitating or suffocating operators or precluding entry to an area of the facility handling licensed radioactive materials), and the controls used to prevent their occurrence or mitigate their consequences. The review should determine that the applicant's facility design and items relied upon for safety provide reasonable assurance of chemical safety at the facility for routine operations, off-normal conditions, and potential accidents.

6.2 RESPONSIBILITY FOR REVIEW

Primary: Chemical Process Specialist

Secondary: Licensing Project Manager

Supporting: Primary Reviewers of SRP Section 1.1, and Chapters 2.0, 3.0, 4.0 and 8.0. Primary Reviewers of Applicable Sections of SRP Chapter 11.0.

6.3 AREAS OF REVIEW

The regulations, 10 CFR 70.61 and 70.62, as revised, require applicants to establish minimum requirements for all items relied on for safety in their process design and description and a safety program. This does not necessarily require the establishment of a separate chemical safety program, but does require that chemical hazards and accident sequences that affect radiological materials be considered and adequately prevented or mitigated.

At NRC-licensed facilities the NRC oversees chemical safety issues related to (i) radiation risk produced by radioactive materials; (ii) chemical risk produced by radioactive material; and (iii) plant conditions which affect or may affect the safety of radioactive materials and thus present an increased radiation risk to workers, the public, and the environment. The NRC does NOT

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[1]Nuclear Regulatory Commission (U.S.), Washington, D.C. "Domestic Licensing of Special Nuclear Material (10 CFR Part 70); Possession of a Critical Mass of Special Nuclear Material." *Federal Register:* Vol. 64, No. 146. pp. 41338-41357. July 30, 1999.

have regulatory responsibility for plant conditions which may result in an occupational risk, but do not affect the safe use of radioactive materials.

The following areas should be reviewed:

1. Chemical Process Description - including process chemistry, flow diagrams, mass/energy balances, inventories, major/significant process steps, and major/significant pieces of equipment.

2. List of Hazardous Chemicals - including potential interactions between chemicals and other materials as determined by the ISA.

3. Chemical Accident Sequences - including unmitigated analyses involving the hazardous chemicals and licensed radioactive materials, as determined by the ISA.

4. Chemical Accident Consequences - including assumptions, bases, and methods used to estimate the consequences of accidents for the worker and the public identified in the ISA Summary that involve hazardous chemicals and licensed radioactive materials.

5. Chemical Safety Controls - including the number and quality of controls used to protect against (reduce frequency and probability of occurrence) or mitigate (reduce consequences) process and chemical accidents involving the release of hazardous chemicals and/or licensed radioactive materials, as determined by the ISA.

6. Chemical Process Safety Interfaces - including a description of how chemical safety interfaces with and is affected by other areas of review, including quality assurance, training, configuration management, maintenance, etc.

This information should be of sufficient quality and detail to allow for an independent review, assessment, and verification by the reviewers. Some of the information may be referenced to other sections of the application, or incorporate by reference, provided that these references are clear, specific, and essentially complete.

6.4 ACCEPTANCE CRITERIA

6.4.1 Regulatory Requirements

The requirements for chemical safety are provided in the following:

> Nuclear Regulatory Commission (U.S.), Washington, D.C. "Domestic Licensing of Special Nuclear Material (10 CFR Part 70)." *Federal Register:* Vol. 64, No. 146. pp. 41338–41357. July 30, 1999.

Specific citations are as follows:

1. § 70.64(a) has a Baseline Design Criterion for "chemical safety" and requirements regarding defense-in-depth practices.

2. § 70.61 contains performance requirements for the facility.

6.4.2 Regulatory Guidance

Listed in this section are the applicable portions of the NRC Inspection Manual and NUREG reports that, in general, provide guidance on satisfying the regulatory requirements cited in SRP Section 6.4.1.

1. Nuclear Regulatory Commission (U.S.) (NRC). Inspection Manual, Chapter-2603, "Inspection of the Nuclear Process Safety Program at Fuel Cycle Facilities," NRC: Washington, D.C. latest revision.

2. -----. NUREG/CR-6410, "Nuclear Fuel Cycle Accident Analysis Handbook," NRC: Washington, D.C. March 1998.

3. -----. NUREG-1513, "Integrated Safety Analysis Document," NRC: Washington, D.C. latest revision.

4. -----. NUREG-1601, "Chemical Process Safety at Fuel Cycle Facilities," NRC: Washington, D.C. 1997.

6.4.3 Regulatory Acceptance Criteria

The NRC reviewers should find the applicant's chemical process safety information acceptable if it provides reasonable assurance that the regulatory review criteria (listed below) are adequately addressed and satisfied. The applicant may elect to incorporate some or all of the requested chemical process information in the ISA (see SRP 3.0) rather than in this section. Either approach is acceptable as long as the information is adequately cross-referenced.

6.4.3.1 Chemical Process Description

The chemical process description should be acceptable if it addresses the Baseline Design Criterion for chemical safety (10 CFR 70.64) and contains the following information:

1. Chemical Process Summary: The chemical process summary should be acceptable if it includes the purpose or objective of the major chemical process steps, including the operations to be performed, and overall mass, energy, radioactivity (curie), and waste balances.

2. Chemical Process Details: The details contained in the chemical process description should be acceptable if they identify chemical reactants and products (input and output) to process steps, rates of reactions, the operating conditions (e.g., temperature, pressure, flow rate, pH), and flow sheets, and identify which chemicals contact radioactive materials

or could significantly impact operations with radioactive materials. The process description should include information sufficient to enable the reviewers to understand the hazards associated with the chemical processes.

3. Process Chemistry: The description of the process chemistry should be acceptable if it provides equations for the chemical reactions and degradation phenomena (including radiolysis) of the chemical species. The process chemistry discussion should address initial startup conditions, normal operation, shutdown, and process testing and qualification.

4. Chemical Process Equipment, Piping, and Instrumentation: The description of the equipment, piping and instrumentation used in chemical processing should be acceptable if it includes descriptions, diagrams, layouts, schematics, and process logic for the major equipment, piping, and controls that may be important to chemical process safety. Specific areas of potential hazards, such as large inventories in vessels or columns, should be mentioned and described. The potential, deleterious effects of the process (e.g., pH, radiation, upset conditions) upon equipment, piping, and instrumentation should be considered and, as appropriate, evaluated in the application.

5. Chemical Process Inventories: The chemical inventory information should be acceptable if it provides the complete chemical and radionuclide inventories within the facility for routine and credible off-normal conditions.

6. Chemical Process Ranges: The description of the range of chemicals should be acceptable if it includes the approximate input, in-process, and output ranges of chemical and radioisotope concentrations, and other properties (e.g., significant enthalpies).

7. Chemical Process Limits: The identification and description of chemical process limits should be acceptable if it identifies and discusses the limits in terms of parameters important to safety, such as chemical concentrations, temperature, pressure, etc., and addresses the consequences of exceeding or operating beyond these limits. The process description should identify those limits that reasonably and conservatively bound potential off-normal and accident conditions and that would be suitable for subsequent consequence analyses.

6.4.3.2 List of Hazardous Chemicals and Potential Interactions

The list of hazardous chemicals and potential interactions should be acceptable if it addresses 10 CFR 70.62 and § 70.64, and the following:

1. Chemicals: The list of hazardous chemicals is acceptable if it includes all of the chemicals introduced into the process and includes the chemical form, physical state(s), locations, and radionuclide content, as appropriate.

2. Chemical Interactions: The list of potential interactions should be acceptable if it considers potential chemical reactions and interactions between materials stored and used

at the facility that have the potential to affect the safe handling of radioactive materials, as determined by the ISA. Items not usually considered as chemicals (e.g., resins and gaskets) should be included if they have the potential to affect process safety and radioactive materials.

3. Unusual and Unexpected: The list of hazardous chemicals and potential interactions should be acceptable if it addresses unusual and unexpected chemical interactions from the different plant conditions (e.g., high levels of radiation) that may affect the safety of radioactive materials, including those that impact controllability and habitability issues, including but not limited to conditions such as inerting, nitrogen oxides, and flammable gas accumulation.

6.4.3.3 Chemical Accident Sequences

The chemical accident sequences should be acceptable if they address 10 CFR 70.61 and 70.62, and the following:

1. Chemical Accident Sequence Bases: The bases and references used in the chemical accident sequences should be acceptable if supported by applicable data and references, and if the applicant includes estimated annual frequencies and probabilities over the plant's operational period. The accident sequences should include the chemical hazard evaluation that identifies the potential interactions between process chemical, radioactive materials, process conditions, plant personnel/operators, and structures, systems, and components.

2. Unmitigated Sequences: The unmitigated chemical accident sequences should be clearly delineated as unmitigated for the purposes of analysis and item categorization for safety.

3. Estimated Concentrations: The estimates of hazardous chemical concentrations should be acceptable if the techniques, assumptions, and models used in the estimates are consistent with industry practice and are verified and/or validated, and follow the guidance on atmospheric and consequence modeling found in NUREG/CR-6410, "Nuclear Fuel Cycle Accident Analysis Handbook."

4. Concentration Limits: The chemical concentration limits should be acceptable if they have a supporting rationale or basis such as AEGL (Acute Exposure Guideline Level) or ERPG (Emergency Response Planning Guide) values or other cited values (e.g., from OSHA, NIOSH). If the applicant does not used a published standard, or if a chemical has an unknown exposure standard, the applicant may propose an alternate standard accompanied by supporting documentation to justify the selection of such an alternative. The performance requirements of §70.61 are based upon acute chemical exposures, and, as such, chemical concentration values such as OSHA permissible exposure limits or other time weighted average values should not be used unless a rational basis is provided in the ISA.

6.4.3.4 Chemical Accident Consequences

Chemical accident consequence reviews should be coordinated with the ISA (SRP Chapter 3.0) and Environmental (SRP 9.0) chapters and meet the requirements for 10 CFR 70.61 and 70.62.

1. Analysis: the accident consequence analysis should be acceptable if the ISA identifies potential accident sequences with hazardous chemicals and radioactive materials, and the consequences are estimated for both workers and members of the public. Dispersion models may be necessary for estimating the concentration and potential impacts of such chemicals at various distances from the point of release. In this case, the applicant should provide information to support the conclusion that the models used are appropriate for the application and physical phenomena occurring, that the models have been validated and verified, and that the assumed data input leads to a conservative estimate of potential consequences. Consequence analyses should follow the guidance found in NUREG/CR-6410, *Nuclear Fuel Cycle Facility Accident Analysis Handbook.*

2. Latent Impacts: the accident consequence analysis should be acceptable if the applicant considers if there are any residual, long-term impacts to worker and public health (i.e., as compared to the previous analyses, which focus primarily on the prompt effects from dispersion modeling via an inhalation pathway) from the chemical hazard effects upon radioactive materials.

3. Uncertainty: the accident consequence analysis should be acceptable if the analysis includes consideration of uncertainty and errors in comparing chemical hazards and radioactive material effects with the performance requirements of § 70.61.

6.4.3.5 Chemical Safety Controls

Chemical safety controls should be acceptable if the identification of the controls used to prevent or mitigate potential accidents is supported by appropriate safety analyses, and if the applicant provides reasonable assurance that these safety controls will be available and reliable upon demand. An adequate application should satisfy the criteria listed below.

1. The application identifies the design basis that provides safety for normal operations. A description could include specified features such as materials of construction, sizing, system fabrication, and process control schemes.

2. The chemical safety control discussion includes a description of the process and engineering design features used to control each process step, including set point ranges and any special administrative or procedural controls. The discussion describes the process safety features that are relied upon for chemical process safety, including the number and quality of controls used to protect against (reduce frequency and probability of occurrence) or mitigate (reduce consequences) accidents involving the release of hazardous chemicals as determined by the ISA.

3. Items relied upon for safety are identified for those accident sequences that contain a chemical/process failure that may lead to radiological consequences that exceed the performance requirements.

4. If the applicant has elected to follow a graded approach to safety in accordance with 70.62(a), the review should establish that the grading of items relied upon for safety is appropriate and sufficient to protect against chemical/process risk, including a consideration of relying upon passive over active systems, defense-in-depth, and fail safe features. For common mode failures, the review should evaluate design features in the application that utilize independent sources of motive force and power for such items as actuators, pumps, and eductors.

5. The application describes the management measures that assure the availability and reliability of items relied upon for safety for chemical and process safety. Management measures may be graded commensurate with risk.

6.4.3.6 Chemical Process Safety Interface

The description of chemical process safety information should be acceptable if the application addresses how the following areas of review interface with aspects of chemical safety at the facility (see the appropriate SRP sections and Chapters as specified in parentheses):

1. Organizational Structure (SRP Section 2.1)
2. Emergency Management (SRP Chapter 8.0)
3. Configuration Management (CM - SRP Section 11.1)
4. Maintenance (SRP Section 11.2)
5. Quality Assurance (QA - SRP Section 11.3)
6. Training and Qualification (SRP Section 11.4)
7. Human Factors (SRP Section 11.5)
8. Audits and Assessments (SRP Section 11.6)
9. Incident Investigations (SRP Section 11.7)
10. Procedures (SRP Section 11.9)

6.5 REVIEW PROCEDURES

6.5.1 Acceptance Review

The primary reviewer evaluates the application to determine whether it addresses the "Areas of Review" discussed in Section 6.3, above. If significant deficiencies are identified, the applicant should be requested to submit additional material before the start of the safety evaluation.

6.5.2 Safety Evaluation

After determining that the application is acceptable for review in accordance with Section 6.5.1, above, the primary reviewer will perform a safety evaluation against the acceptance

criteria described in Section 6.4. If during the course of the safety evaluation, the primary reviewer determines the need for additional information, the primary reviewer coordinates a request for additional information with the licensing project manager.

Because the results of the ISA form the basis for much of the chemical safety of the design and facility, the primary reviewer should also review the ISA (see SRP Chapter 3.0). Chemical safety, as defined in the SAR, should conform to the level of safety deemed necessary by the ISA. The primary reviewer should establish that the applicant's facility design, operations, and chemical safety items provide reasonable assurance that they will function as intended and provide for the safe handling of radioactive materials at the facility. The primary reviewer should identify the mechanisms that will allow the applicant to identify and correct potential problems.

The secondary reviewer should confirm that the chemical safety approach is consistent with other sections of the application. Information provided for chemical safety should be of comparable quality and detail, and should not contradict or adversely impact information contained in other sections of the application.

Supporting reviewers should confirm that provisions made in the application for chemical safety are in accordance and consistent with specified sections of the SRP. For example, the primary reviewer from SRP Chapter 4.0, "Radiation Safety" (usually a health physicist), as a supporting reviewer for chemical safety, should establish that the program described by the applicant provides reasonable assurance for the facility, its operations, and the chemical safety program will not have unacceptably adverse impacts on the radiological safety at the facility.

For an existing facility, the NRC reviewers may wish to visit the site and facility personnel in order to gain a better understanding of the process, its potential hazards, and safety approaches. For a planned facility, the NRC reviewers may wish to meet with the design team in order to gain a better understanding of the process, its potential hazards, and safety approaches.

When the safety evaluation is complete, the primary reviewer, with assistance from the other reviewers, should prepare the chemical safety input for the Safety Evaluation Report (SER), as described in Section 6.6 using the acceptance criteria from Section 6.4. The secondary reviewer should coordinate the chemical safety input with the balance of the reviews and the SER.

6.6 EVALUATION FINDINGS

The primary reviewer writes an SER section addressing each topic reviewed under this SRP Chapter and explains why the NRC staff has reasonable assurance that the chemical safety part of the application is acceptable. License conditions may be proposed to impose requirements where the application is deficient. The SER should include a summary statement of what was evaluated and the basis for the reviewers' conclusions.

The staff can document the evaluation as follows:

The staff has evaluated ..... [Insert a summary statement of what was evaluated and why the reviewer finds the submittal acceptable.] *Based on the review of the license application, the NRC staff has concluded that the applicant has adequately described and assessed accident consequences having potentially significant chemical consequences and effects that could result from the handling, storage, or processing of radioactive materials. A hazard analysis has been conducted that identified and evaluated those chemical process hazards and potential accidents, and established safety controls to ensure safe facility operation. To ensure that the performance requirements in 10 CFR 70.61, as revised, are met, the applicant will ensure that controls are maintained available and reliable. The staff has reviewed these safety controls and the applicant's plan for managing chemical process safety and its potential effects upon radioactive materials, and finds them acceptable.*

The staff concludes that the applicant's plan for managing chemical process safety and the chemical process safety controls meet the requirements of 10 CFR 70.61, 70.62, and 70.64, and provide reasonable assurance that the health and safety of the public, workers, and the environment will be protected.

6.7 REFERENCES

1. Chemical Manufacturers Association, (CMA). "Responsible Care®, Process Safety Code of Management Practices." CMA: Washington, D.C. 1990.

2. Center for Chemical Process Safety, (CCPS). "Guidelines for the Technical Management of Chemical Process Safety." CCPS/American Institute of Chemical Engineers: New York, 1989, Chapter 11, as revised.

3. Nuclear Regulatory Commission (U.S.), Washington, D.C. "Domestic Licensing of Special Nuclear Material (10 CFR Part 70); Possession of a Critical Mass of Special Nuclear Material." *Federal Register:* Vol. 64, No. 146. pp. 41338-41357. July 30, 1999.

4. Code of Federal Regulations, *Title 29, Labor*, Part 1910.119, "Process Safety Management of Highly Hazardous Chemicals," as revised.

5. Nuclear Regulatory Commission, (U.S.) (NRC). NRC Inspection Manual, Chapter 2603, "Inspection of the Nuclear Chemical Process Safety Program at Fuel Cycle Facilities," NRC: Washington, D.C., as revised.

6. ——. Washington, D.C. , "Memorandum of Understanding between the Nuclear Regulatory Commission and the Occupational Safety and Health Administration: Worker Protection at NRC-Licensed Facilities, *Federal Register:* Vol. 53, No. 210. pp. 43950-43951. October 31, 1988.

7. -----. NUREG/CR-6410, "Nuclear Fuel Cycle Accident Analysis Handbook." NRC: Washington, D.C. 1998.

8. -----. NUREG-1601, "Chemical Process Safety at Fuel Cycle Facilities." NRC: Washington, D.C. 1997.

FIRE PROTECTION

7.1 PURPOSE OF REVIEW

This review should establish that there is reasonable assurance that the applicant has designed a facility that provides for "adequate protection against fires and explosions" (§ 70.64(a)(3), as revised)[1] and that is based on defense-in-depth practices (§ 70.64(b)). This review should also establish that radiological consequences from fires are considered in determining how the facility will meet the performance requirements of § 70.61.

7.2 RESPONSIBILITY FOR REVIEW

Primary: Fire Protection Engineer

Secondary: Licensing Project Manager

Supporting: Chemical Engineer
Nuclear Engineer
Quality Assurance Engineer

7.3 AREAS OF REVIEW

To assure that the requirements of 10 CFR Part 70 in regard to fire protection have been met, the following areas should be reviewed:

1. Organization and Conduct of Operations: These issues include organization, staffing, fire prevention, engineering review of design changes, QA, and documentation and record-keeping.

2. Fire Protection Features and Systems: Plant fire protection features and systems include construction features; passive fire-rated barriers; process and operational features; fire detection and alarm systems; fire suppression systems and equipment; design-basis documents; and inspection, maintenance, and testing of fire protection measures.

3. Manual Firefighting Capability: A "baseline needs" assessment should establish the minimum required capabilities of site firefighting forces. This assessment should include minimum staffing, organization and coordination of onsite and offsite firefighting

[1]Nuclear Regulatory Commission (U.S.), Washington, D.C. "Domestic Licensing of Special Nuclear Material (10 CFR 70)." *Federal Register:* Vol. 64, No. 146. pp. 41338–41357. July 30, 1999.

resources, personal protective and firefighting equipment, training, and prefire emergency planning.

4. Fire Hazard Analysis (FHA): The FHA consists of a systematic analysis of the fire hazards, an identification of specific areas and systems important to plant fire safety, the development of design-basis fire scenarios, an evaluation of anticipated consequences, and a determination of the adequacy of plant fire safety. The features and characteristics of an acceptable FHA are listed separately in Appendix A of this SRP.

7.4 ACCEPTANCE CRITERIA

7.4.1 Regulatory Requirements

Section 70.64(a) has a Baseline Design Criterion for "fire protection" and requirements regarding defense-in-depth practices. In addition, § 70.61 contains performance requirements for the facility.

7.4.2 Regulatory Guidance

Regulatory guidance intended for fuel cycle facilities (without specific requirements for an vitrification facilities) was published in the Federal Register as "Guidance on Fire Protection for Fuel Cycle Facilities," FR 57 (No. 154), 35607-35613, August 10, 1992. While providing specific guidance in selected areas of fire safety, the staff's position also references NFPA codes that can provide information on methods of recommended practice that may be applied for TWRS facilities in other areas of fire safety[2]. Guidance in regard to accident analysis may be found in "Nuclear Fuel Cycle Facility Accident Analysis Handbook," NUREG/CR-6410, 1998.

7.4.3 Regulatory Acceptance Criteria

The NRC reviewers should find that the applicant's submittal regarding fire safety provides reasonable assurance that the regulatory review criteria below are adequately addressed and satisfied. Some of the information may be referenced from other sections of the application, or incorporated by reference, provided that these references are clear and specific.

Where specific NFPA or other standards are referenced, it is the intent of the SRP to refer the user to the latest standard, which could have another title or number. For this reason, specific dates are not listed in the reference list. If the applicant references an NFPA or other industry

[2] NFPA Standard 801, Standards for Facilities Handling Radioactive Material, provides additional overall guidance on fire protection for fuel cycle facilities.

standard, it should be dated (as the code of record) so that its criteria can be applied in the review of the applicant's submittal. Specified standards will normally be considered as acceptable means of meeting the review criteria. Alternative means, as well as deviations from specific sections of the standards, will also be considered but may require justification through analysis. Also, depending on the application, standards other than those referenced may be more appropriate for the fire protection required. In addition, hazards may exist or occur at the facility that are not specifically addressed in this SRP section. It is expected that the applicant will select and reference the most applicable standards for all known hazards and fire protection measures at its facility in its license application, beyond those identified in this SRP Chapter.

7.4.3.1 Organization and Conduct of Operations

The organization and conduct of operations should be considered acceptable if the following conditions are met:

1. Organization and Management: The specific responsibilities, required skills, and knowledge of all facility positions involved in plant fire safety functions and activities are clearly identified in a formal, documented plant policy that includes a functional organization chart that shows the position and authority of personnel involved in fire safety in relation to the overall plant organization.

    a. A single senior management plant position is assigned the overall responsibility for plant fire safety. Another position is assigned the responsibility for day-to-day supervision of performance of tasks relating to fire safety. It is not necessary for this position to be a full-time fire safety position. In an organization where neither of these positions includes direct responsibility for manual firefighting activities, there is a provision to establish a formal means of effective liaison and communication to coordinate manual firefighting activities of all groups, both onsite and offsite, as appropriate.

    b. There are provisions to provide sufficient staffing by engineering professionals with expertise in fire protection to assure that proactive elements in the fire protection program, such as FHAs and updated prefire plans, will be accomplished in a timely manner.

    c. There are provisions to establish a fire safety review committee staffed with managers from different engineering disciplines.

2. Training and Qualifications: Qualifications and experience are specified for all positions involved in fire protection functions and activities that affect plant fire safety.

All site personnel should be instructed in the general fire safety program of the plant, specialized fire safety training should be provided for plant personnel involved in operations and maintenance work at the facility, and emergency response team members should be provided specialized fire protection and firefighting training necessary for fire emergency defense.

3. Fire Prevention Program: Administrative procedures for control of combustible materials, including transient combustibles should be provided. These procedures should establish controls for storage, handling, transport, and use of combustible solids, liquids, and gases; including construction materials; materials associated with normal facility processes and operations; and combustibles introduced during maintenance or modification activities. Procedures are established for safe operation of processes and equipment that present fire hazards and for control of ignition sources in areas as identified as important to plant safety.

   a. There are provisions to establish and implement a permit-to-work system to control activities that could: (1) introduce combustible materials, (2) introduce sources of ignition, or (3) degrade fire protection features (active or passive) important to facility fire safety. Impairments to fire protection systems (active or passive) should be governed by a written procedure which tracks the impaired system, identifies personnel to be notified and specifies compensatory fire protection and prevention measures. Such measures should be location specific and supported by analysis in the FHA or ISA.

   b. There are provisions to establish and implement administrative procedures including quality assurance reviews for engineering review of facility and process design and modifications that may impact fire safety.

   c. There are provisions to establish and implement procedures to report and investigate fire incidents.

   d. There are provisions to establish and implement a penetration seal tracking program to record pertinent information regarding the emplacement and modification of fire barrier penetration seals which are identified in the ISA or FHA as relied on for plant safety.

7.4.3.2 Fire Protection Features and Systems

The facility fire protection features and systems should be considered acceptable if the following conditions are met:

1. Buildings containing items relied on for safety are designed to qualify as Type I construction as defined by NFPA Standard 220. This includes structural building

components such as walls, floors, roofs, columns, and beams as well as interior building features. The process layout separates and isolates, as much as practicable, operations presenting fire hazards. This can be accomplished by distance, or compartmentalizing using fire barriers, or both.

2. Electrical wiring for plant facilities determined to be relied on for safety are designed and a commitment is made to maintain such wiring in accordance with the applicable provisions of the National Electric Code (NFPA Standard 70). Cable trays classified as relied on for safety or which may contribute a significant fire load are protected from fire in accordance with IEEE Standard 690. Circuit failure modes which could result in spurious actuations with a potential effect on plant safety should be evaluated in the ISA.

3. Lightning protection for plant buildings determined to be relied on for safety is designed in accordance with the applicable provisions of NFPA Standard 780.

4. The ventilation systems in areas containing items relied on for safety are designed to minimize the spread of fire, smoke, hot gases, and products of combustion from the area of fire origin and in accordance with the applicable provisions of NFPA Standard 90A. Where ventilation systems are designed to prevent the release of radioactive materials, all materials of construction, including high-efficiency particulate air (HEPA) filters, are of the fire-resistant type in accordance with the applicable provisions of Underwriters Laboratories, Inc. (UL), Standard 586. Further fire protection guidance for nuclear filter plenums is contained in Appendix B of this SRP.

5. Building layout provides a safe means of egress for plant personnel in the event of fire in accordance with the applicable provisions of The Life Safety Code (NFPA Standard 101). Emergency lighting for the purpose of personnel egress is in accordance with NFPA Standard 101. The design basis for emergency lighting required to perform any safety related functions during a loss of power should be determined from engineering evaluations and the ISA.

6. The design of openings in passive fire-rated barriers incorporates suitable automatic or fixed closure devices or components, such as fire doors, fire dampers, and fire-rated penetration seals. Fire doors are designed and installed in accordance with the applicable provisions of NFPA Standard 80. Fire dampers are designed and installed in accordance with the applicable provisions of UL Standard 555.

7. Plant areas where a credible risk of large spills of flammable or combustible liquids exist are identified and means of containing, e.g., dikes, and disposing of such spills are provided for in the facility design. The design of containment and drainage systems should consider the rate of water discharge from fixed suppression systems and/or hose lines and be capable of preventing the spread of combustible liquids from pits or confining

areas. Flammable and combustible liquids should be stored, handled, and used in accordance with the applicable provisions in NFPA 30 and/or other industry standards.

8. Plant areas are identified where credible risk of creation of a flammable mixture with hydrogen or other flammable gases exist. Preventive measures in accordance with NFPA 69 and/or other industry standards should be provided.

9. The facility design incorporates a fire-alarm system, designed in accordance with the applicable provisions of NFPA Standard 72, provided throughout areas as determined to be relied on for safety by the ISA. The system should incorporate features such as local and remote annunciation, primary and secondary power supplies, and audible and visual alarm devices.

10. The facility design incorporates an adequate and a reliable water supply system, designed in accordance with NFPA standards for fire protection use. The system should consist of the water source, dedicated storage facilities, fire pumps, a distribution-piping network, sectional isolation valves, and fire hydrants, as applicable to the facility. The design of the fire pumps, where provided, should be in accordance with the applicable provisions of NFPA Standard 20. The design of the distribution piping, valves, and fire hydrants are in accordance with the applicable provisions of NFPA Standard 24. Water supply requirements in terms of stored volume and/or supply rates should be determined in the FHA.

11. Provisions are made to electrically supervise control valves for water-based fire suppression systems or to keep them locked open and monitored under a periodic surveillance program in accordance with NFPA Standard 801.

12. Fire suppression systems and equipment are incorporated in the facility design to protect areas determined to be relied on for safety. Fire suppression systems and equipment may be automatic or manually activated as determined by the ISA or FHA. The design and installation of fire-suppression systems and equipment should be in accordance with the applicable provisions of appropriate NFPA standards. Seismic criteria for portions of the fire water system should be determined by the ISA/FHA. Commonly applied NFPA Standards include NFPA 10, 11, 11A, 12, 13, 14, 15, 16, 16A and 2001.

13. Provisions are made to provide a program of regular inspection, testing and maintenance of fire protection equipment in accordance with the provisions of appropriate NFPA or other industry standards. A commonly applied standard for water-based systems is NFPA Standard 25.

14. Hot cells or canyons in vitrification facilities containing melters are treated as separate fire areas in the facility design. Fire resistance of walls ceilings and floors of such areas should be established by the ISA and should have a minimum fire resistance rating of two

hours and be of non-combustible construction. If significant quantities of combustibles are present inside the enclosure, a fixed suppression system should be provided. Oil contained in windows should be non-combustible or oil with a high flashpoint. Combustible oil should be included in the FHA fire loading survey.

15. Synthetic fire-resistant hydraulic fluids in the master-slave manipulators in hot cells should be used.

16. Provisions are made to construct glove boxes and windows of non-combustible materials. A means of fire detection is to be provided if pyrophoric materials, oxidizers, or organic liquids are handled. Fire suppression or a fixed inerting system should be provided if combustible materials are present, or could be present, in quantities sufficient to cause a breach of integrity.

17. Incinerators, boilers, and furnaces should be installed and maintained in accordance with NFPA 54, 31, 8501 and/or other applicable industry standards.

7.4.3.3 Manual Firefighting Capability

The facility manual firefighting capability should be considered acceptable if the following conditions are met:

1. Plant documentation provides a clear description of the manual firefighting capability proposed. A "baseline needs" assessment should establish the minimum required capabilities of site firefighting forces. Manual firefighting capability may be provided solely by a well-trained and fully equipped onsite fire emergency response team, by qualified offsite resources, or by a coordinated combination of the two approaches, as appropriate for the facility.

2. A specific organizational position is identified to provide coordination and liaison with offsite firefighting resources and to establish a clear line of authority at the fire scene, when any reliance is placed on offsite response.

3. Where reliance for manual firefighting capability is placed on offsite resources (either for a partial or full response), provisions are made to execute a formal agreement that documents the assistance provided by the offsite organization(s). The agreement provides a description of the minimum firefighting manpower and equipment to be provided during fire emergencies and the estimated response time.

4. Where manual firefighting capability is to be provided by an onsite fire emergency response team, the team is identified as established, equipped, and trained to achieve one of the following objectives in accordance with NFPA Standard 600:

a. Incipient-stage firefighting.
b. Advanced exterior firefighting only.
c. Interior structural firefighting only.
d. Both advanced exterior and interior structural firefighting.

5. Provisions are made to develop a prefire plan for each area of the facility determined to be important to plant fire safety, including those areas that present a fire exposure to areas relied upon for safety. (The prefire plan should supplement the information provided in the Emergency Preparedness Plan.) As a minimum, the prefire plan should identify access and egress routes; location of structures, systems, or components determined to be important to plant fire safety; special radiological and toxic hazards; automatic and manually operated fire suppression measures provided in each fire area; specific procedures for fire suppression activities because of nuclear criticality, buildup of explosive gases or other concerns; and location of vital heat-sensitive components or equipment. Responsibilities for specific actions such as shutting down processes may be assigned in the pre-fire plans. The pre-fire plan is to be revised when any of the above listed information changes significantly.

7.4.3.4 Fire Hazard Analysis (FHA)

The FHA should be considered acceptable if it reflects current conditions throughout the facility and it is to be reviewed and updated as necessary at defined, regular intervals to document that fire protection measures are adequate to ensure plant fire safety. In addition, the FHA should be revised to incorporate significant changes and modifications to the facility, processes, or inventories, as needed. (The level of detail provided in the FHA should reflect the complexity of the facility and the anticipated consequences from fire events. A more detailed description of the requirements for an FHA is provided in Appendix A of this SRP.)

7.5 REVIEW PROCEDURES

7.5.1 Acceptance Review

The primary reviewer should evaluate the application to determine whether it addresses the "Areas of Review" discussed in Section 7.3, above. If significant deficiencies are identified, the applicant should be requested to submit additional material before the start of the safety evaluation.

7.5.2 Safety Evaluation

After determining that the application is acceptable for review in accordance with Section 7.5.1, above, the primary reviewer should perform a safety evaluation against the acceptance

criteria described in Section 7.4. If during the course of the safety evaluation, the primary reviewer determines the need for additional information, the primary reviewer should coordinate a request for additional information with the licensing project manager. The safety evaluation forms the basis for staff findings, and supports the reviewer's conclusions (Section 7.6).

The primary reviewer should also review sections of the ISA which address fire safety to insure that those sections are consistent with the fire safety portion of the license application. The primary reviewer should also assure that the requirements for placement and reliability of fire protection measures is consistent with the results of the ISA.

The secondary reviewer should confirm that descriptions in the fire safety section are consistent with descriptions in other sections of the application which may interface with fire safety. The secondary reviewer may also request support from other technical reviewers as required.

Supporting reviewers should confirm that provisions made in the applicant's fire safety section are in accordance with other sections of the SRP within their areas of responsibility. For example, the nuclear engineer, as a supporting reviewer, should establish that the program described by the applicant provides reasonable assurance that the fire safety program will not adversely affect criticality safety.

When the safety evaluation is complete, the primary staff reviewer, with assistance from the other reviewers, should prepare the fire safety input for the Safety Evaluation Report as described in Section 7.6 using the acceptance criteria from Section 7.4.

7.6 EVALUATION FINDINGS

The primary reviewer should write an SER section addressing each topic reviewed under this SRP Chapter and explain why the NRC staff has reasonable assurance that the fire safety part of the application is acceptable. License conditions may be proposed to impose requirements where the application is deficient. The SER should include a summary statement of what was evaluated and the basis for the reviewers' conclusions.

The staff can document the evaluation as follows:

> *The applicant has performed a fire hazards analysis which documents all significant facility fire hazards, fire protection features designed to control those hazards, and the overall adequacy of facility fire safety. In addition to the fire hazards analysis, the applicant also provided the following information in the license application:*
>
> *1. Fire safety organization and conduct of operation,*

2. *Fire protection features and systems, and*
3. *Manual firefighting capability.*

[Insert a summary statement of what was evaluated and why the reviewer finds the submittal acceptable.]

In each of these areas, the staff finds that the applicant's capabilities meet or exceed the guidance provided in SRP Chapter 7.0. The staff concludes that the applicant's proposed equipment, facilities, and procedures provide a reasonable level of assurance that adequate fire protection will be provided and maintained for those items determined to be relied upon for safety to meet the safety performance requirements and the baseline design criteria of 10 CFR Part 70, as revised.

7.7 DEFINITIONS

Combustible: A material, in the form and condition in which it is used, will ignite and burn.

Combustible Liquid[3]**:** A liquid having a flash point at or above 100 of (37.8 Oc.).

Fire Area: A location bounded by fire-rated construction, having a minimum fire resistance rating of 2 hours. The boundaries of exterior fire areas (yard areas) or other locations that represent unique conditions should be as determined by the cognizant fire protection engineer.

Fire Barrier: A continuous membrane such as a wall, floor, or roof that is constructed to limit fire spread and the movement of smoke. Fire barriers have fire resistance ratings and may have protected openings.

Fire Brigade: Facility personnel trained in plant fire-fighting operations.

Fire Door: A fire rated door assembly.

Fire Hazards Analysis (FHA): A comprehensive assessment of potential fires to ensure mitigative features are in place to limit damage from fires to an acceptable level.

Fire Prevention: Measures directed toward avoiding the inception of fires.

Fire Protection: Methods of providing for fire control or fire extinguishment.

[3] Definitions as used in NFPA Fire Protection Handbook and NFPA Standards

Fire Resistance Rating: Time, in minutes or hours, that a material or assembly withstood a fire exposure as specified in NFPA 251, "Standard Methods of Fire Tests of Building Construction and Materials."

Fire Screen: An item of equipment installed ahead of all HEPA filter banks intended to reduce flame propagation and glowing/burning ember products from reaching final HEPA filters.

Flammable Liquid[3]: Liquid with a closed cup flash point below 37.8 Oc. (100 of) and a vapor pressure not exceeding 40 psia at 37.8 Oc. (100 of).

Flammable Gas[3]: A gas that will burn in the normal concentration of oxygen in the air.

Gas[3]: Any substance that in a liquid state exerts a vapor pressure greater than 40 psia at 100° F.

Hydrophoric Materials: Materials that react violently with water or water vapor (such as lithium and lithium hydride).

Limited-Combustible: A building construction material that, in the form in which it is used, has a potential heat value not exceeding 8,141 KJ/kg (3,500 BTU/lb) and has either a structural base of noncombustible material with a surfacing not to exceed 3.2 mm (1/8 in) that has a flame spread rating not greater than 50, or other material having neither a flame spread rating greater than 25 or evidence of continual progressive combustion, even on surfaces exposed by cutting through the material on any plane.

Noncombustible: A material that, in the form in which it is used and under the conditions anticipated, will not ignite, burn, support combustion, or release flammable vapors, when subjected to fire or heat. Materials passing ASTM E136, "Standard Test Method for Behavior of Materials in Vertical Tube Furnace at 750°F," should be considered noncombustible.

Oxidizing Gases: Gases that support combustion.

Pyrophoric Material: A material with an auto-ignition temperature in air at or below 130°F (54.4^{0}C) and 50 percent relative humidity.

Reactive Gases: Gases that will either react with other materials or within themselves by a chemical reaction other than combustion under reasonably anticipated initiating conditions.

7.8 REFERENCES

1. Factory Mutual Research Corporation, *Factory Mutual System Approval Guide-Equipment, Materials, Services, and Conservation of Property.*

2. Institute of Electrical and Electronics Engineers, Inc. (IEEE). Standard 690, "IEEE Standard for the Design and Installation of Cable Systems for Class 1E Circuits in Nuclear Power Generating Stations."

3. National Fire Protection Association, Inc. (NFPA). Standard 10, "Standard for Portable Fire Extinguishers."

4. National Fire Protection Association, Inc. (NFPA). Standard 11, "Standard for Low Expansion Foam."

5. National Fire Protection Association, Inc. (NFPA). Standard 11A, "Standard for Medium- and High-Expansion Foam Systems."

6. National Fire Protection Association, Inc. (NFPA). Standard 12, "Standard on Carbon Dioxide Extinguishing Systems."

7. National Fire Protection Association, Inc. (NFPA). Standard 13, "Standard for the Installation of Sprinkler Systems."

8. National Fire Protection Association, Inc. (NFPA). Standard 14, "Standard for Installation of Standpipes and Hose Systems."

9. National Fire Protection Association, Inc. (NFPA). Standard 15, Standard for Water Spray Fixed Systems for Fire Protection."

10. National Fire Protection Association, Inc. (NFPA). Standard 16, "Standard for the Installation of Deluge Foam-Water Sprinkler and Foam-Water Spray Systems."

11. National Fire Protection Association, Inc. (NFPA). Standard 16A, "Standard for the Installation of Closed-Head Foam Water Sprinkler Systems."

12. National Fire Protection Association, Inc. (NFPA). Standard 20, "Standard for the Installation of Centrifugal Fire Pumps."

13. National Fire Protection Association, Inc. (NFPA). Standard 24, "Standard for the Installation of Private Service Mains and their Appurtenances."

14. National Fire Protection Association, Inc. (NFPA). Standard 25, Standard for the Inspection, Testing, and Maintenance of Water-Based Fire Protection Systems."

15. National Fire Protection Association, Inc. (NFPA). Standard 30, "Flammable and Combustible Liquids Code.".

16. National Fire Protection Association, Inc (NFPA). Standard 31, Standards for Installation of Oil Burning Equipment."

17. National Fire Protection Association, Inc. (NFPA). Standard 54, "National Fuel Gas Code."

18. National Fire Protection Association, Inc. (NFPA). Standard 69, "Standard on Explosion Prevention Systems."

19. National Fire Protection Association, Inc. (NFPA). Standard 70, "National Electric Code."

20. National Fire Protection Association, Inc. (NFPA). Standard 72, "National Fire Alarm Code."

21. National Fire Protection Association, Inc. (NFPA). Standard 80, "Standard for Fire Doors and Fire Windows."

22. National Fire Protection Association, Inc. (NFPA). Standard 90A, "Standard for the Installation of Air Conditioning and Ventilating Systems."

23. National Fire Protection Association, Inc. (NFPA). Standard 92A, "Smoke Control Systems."

24. National Fire Protection Association, Inc. (NFPA). Standard 101, "Life Safety Code."

25. National Fire Protection Association, Inc. (NFPA). Standard 220, "Standard on Types of Building Construction."

26. National Fire Protection Association, Inc. (NFPA). Standard 221, "Standard on Fire Walls and Fire Barrier Walls."

27. National Fire Protection Association, Inc. (NFPA). Standard 241, "Standard on Construction, Alteration, and Demolition Operations."

28. National Fire Protection Association, Inc. (NFPA). Standard 251, "Standard on Fire Tests of Building Construction Materials."

Fire Protection

29. National Fire Protection Association, Inc. (NFPA). Standard 252, "Standard on Fire Tests of Door Assemblies."

30. National Fire Protection Association, Inc. (NFPA). Standard 299, "Standard on Protection of Life and Property from Wildfire."

31. National Fire Protection Association, Inc. (NFPA). Standard 600, Standard on Industrial Fire Brigades."

32. National Fire Protection Association, Inc. (NFPA). Standard 780, "Lightning Protection Code."

33. National Fire Protection Association, Inc. (NFPA). Standard 2001, "Standard on Clean Agent Extinguishing Systems."

34. National Fire Protection Association, Inc. (NFPA). Standard 8501, "Standard for Single Burner Oil Operation." National Fire Protection Association, Inc.

35. Nuclear Regulatory Commission (U.S.), Washington, D.C. "Domestic Licensing of Special Nuclear Material (10 CFR 70)." *Federal Register:* Vol. 64, No. 146. pp. 41338--41357. July 30, 1999.

36. Underwriters Laboratories, Inc. "Underwriters Laboratories Building Materials Directory".

37. Underwriters Laboratories, Inc. "Underwriters Laboratories Fire Protection Equipment Directory".

38. Underwriters Laboratories, Inc. Standard 555, "Standard for Fire Dampers and Ceiling Dampers."

39. Underwriters Laboratories, Inc. Standard 586, "High Efficiency Air Filtration Units."

EMERGENCY MANAGEMENT

8.1 PURPOSE OF REVIEW

The purpose of this review is to establish, with reasonable assurance, that the proposed facility will have adequate emergency management facilities and procedures in place to protect the public, the workers, and the environment during operations. Requirements for emergency management are provided in 10 CFR Part 70, as revised,[1] and address the need for an emergency plan in §70.22(i) and emergency capability as a baseline design criterion (BDC) in §70.64(a)(6), "Emergency capability." An emergency plan is required when an evaluation shows that the maximum dose to a member of the public offsite due to a release of radioactive materials would exceed 1 rem effective dose equivalent. BDC #6, "Emergency capability," is intended to ensure control of licensed material, evacuation of personnel, and availability of emergency facilities.

8.2 RESPONSIBILITY FOR REVIEW

Primary: Emergency Preparedness Specialist

Secondary: Licensing Project Manager
Health Physics Reviewer

Supporting: Regional Emergency Preparedness Inspector
ISA Reviewer
Site Representative

8.3 AREAS OF REVIEW

The review should address the applicant's submittal for an acceptable level of evidence of planning for emergency preparedness directed at situations involving real or potential radiological hazards. The review should address those design features, facilities, functions, and equipment that may affect some aspect of emergency planning or the capability of an applicant to cope with plant emergencies. In addition, the review should address coordination with offsite organizations. The staff should either review the emergency plan made in accordance with 10 CFR 70.22(i)(1)(ii) and with the guidance contained in the acceptance criteria below, or should review the applicant's evaluation that an emergency plan is not needed in accordance with 10 CFR 70.22(i)(1)(i).

The NRC staff reviewer should address the material presented, as described below.

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[1] Nuclear Regulatory Commission (U.S.), Washington, D.C. "Domestic Licensing of Special Nuclear Material (10 CFR Part 70)." *Federal Register:* Vol. 64, No. 146. pp. 41338--41357. July 30, 1999.

8.3.1 Evaluation That No Emergency Plan is Required

If the applicant submits an evaluation to demonstrate that an emergency plan is not required, the staff should review the evaluation against 10 CFR 70.22(i)(1)(i), and NUREG-1140, "A Regulatory Analysis of Emergency Preparedness for Fuel Cycle and Other Radioactive Material Licensees." NUREG/CR-6410, "Nuclear Fuel Cycle Facility Accident Analysis Handbook," also contains useful information. Areas to be evaluated should include the following:

1. A description of the facility,
2. Types of materials used, including both radioactive material and hazardous chemicals,
3. Types of accidents,
4. Detection of accidents,
5. Site specific information used to support the evaluation, and
6. An evaluation of the consequences, both onsite and offsite, of accidents including radioactive and hazardous chemicals. The evaluation shows that the maximum dose to a member of the public offsite due to a release of radioactive materials would not exceed 1 rem effective dose equivalent or an intake of 2 milligrams of soluble uranium in accordance with 10 CFR 70.22(i)(1)(i).
7. The evaluation should address one or more of the factors provided in 10 CFR 70.22(i)(2).

8.3.2 Emergency Plan

If the applicant submits an emergency plan, the staff should evaluate the emergency management program against 10 CFR 70.22(i)(1)(ii) and Regulatory Guide 3.67, "Standard Format and Content for Emergency Plans for Fuel Cycle and Materials Facilities," which provides a standard format and content for an emergency plan. Elements in the emergency plan to be reviewed should include the following:

1. Facility description (including both onsite and offsite emergency facilities),
2. Types of accidents,
3. Classification of accidents,
4. Detection of accidents,
5. Mitigation of consequences (and safe shutdown),
6. Assessment of releases (both radioactive materials and hazards chemicals),
7. Responsibilities of licensee,
8. Notification and coordination,
9. Information to be communicated and parties to be contacted,
10. Training,
11. Safe shutdown (recovery and plant restoration),
12. Exercises (and drills),
13. Hazardous chemicals inventories and locations, and
14. Responsibilities for developing and maintaining the emergency program and its procedures.

8.4 ACCEPTANCE CRITERIA

8.4.1 Regulatory Requirements

The requirements for emergency management are provided in the following:

Nuclear Regulatory Commission (U.S.), Washington, D.C. "Domestic Licensing of Special Nuclear Material (10 CFR Part 70)." *Federal Register:* Vol. 64, No. 146. pp. 41338–41357. July 30, 1999.

Specific citations are as follows:

1. § 70.22(i)(1)(i) specifies when an emergency plan does not have to be submitted to the NRC.

2. § 70.22(i)(3), contains the information that must be included in the emergency plan, if an emergency plan is required to be submitted.

3. § 70.64(6), "Emergency capability," specifies that the applicant or licensee must provide emergency capability to maintain control of licensed material, evacuation of personnel and emergency facilities.

8.4.2 Regulatory Guidance

Regulatory guidance for preparing an emergency plan includes:

1. Nuclear Regulatory Commission (U.S.) (NRC). Regulatory Guide 3.67, "Standard Format and Content for Emergency Plans for Fuel Cycle and Materials Facilities," NRC: Washington, D.C. January 1992.

2. ——. NUREG-1140, "A Regulatory Analysis of Emergency Preparedness for Fuel Cycle and Other Radioactive Materials." NRC: Washington, D.C. January 1988.

3. ——. NUREG/CR-6410, "Nuclear Fuel Cycle Facility Accident Analysis Handbook." NRC: Washington, D.C. March 1998.

8.4.3 Regulatory Acceptance Criteria

8.4.3.1 Evaluation That No Emergency Plan Is Required

The adequacy of the evaluation that no emergency plan is required should be reviewed against the requirements in 10 CFR 70.22(i)(2) and the specific criteria given in the following sections of the SRP. This evaluation should be acceptable if the regulatory requirements and the following criteria are met:

8.4.3.1.1 Facility Description

The evaluation includes a description of the facility and site, the area near the site, and the licensed activities conducted at the facility sufficient to support the evaluation. The description includes the following:

1. A detailed drawing of the site showing (1) onsite and near offsite (within 1 mile) structures with building numbers and labels, (2) roads and parking lots onsite and main roads near the site, (3) site boundaries, showing fences and gates, (4) major site features, (5) water bodies within approximately 1 mile, and (6) the location(s) of nearest residents.

2. The stack heights, typical stack flow rates, and the efficiencies of any emission control devices.

3. A general description of licensed and other major activities conducted at the facility, and the type, form, and quantities of radioactive and other hazardous material normally onsite.

8.4.3.1.2 Types of Accidents

The evaluation describes each type of accident identified by the ISA that has the maximum offsite consequences exceeding the limit of 10 CFR 70.22(i)(1)(i). The description includes:

1. The process and physical location where it could occur,

2. Complicating factors and possible onsite and offsite consequences, including non-radioactive hazardous material released,

3. The accident sequence that has the potential for the greatest radiological and toxic chemical impact.

8.4.3.1.3 Detection of Accidents

The evaluation described for each type of accident identified the following:

1. The means of detecting the accident,

2. The means of detecting any release of radioactive or other hazardous material,

3. The means of alerting the operating staff, and

4. The anticipated response of the operating staff.

8.4.3.1.4 Evaluation of Maximum Public Exposure

In order to demonstrate that no emergency plan is required, an applicant may either (1) request that its total possession limit for radioactive material be reduced below the emergency plan threshold in 10 CFR 70.22(i)(1), or (2) perform a site specific evaluation that demonstrates maximum public exposure is less than the limits in 70.22(i)(1)(i).

The evaluation should include a description of the following information sufficient to allow for independent verification:

1. Type of accident (e.g., fire, exposure, chemical release, nuclear criticality),
2. Location of accident,
3. Maximum source term,
4. Solubility of material,
5. Facility design or engineered safety features in the facility and the proposed release fraction,
6. Location and distance of nearest member of the public to the facility,
7. Dose model used and the process used to verify the reliability of the model and validity of the assumptions,
8. Assumed worst case weather condition, and
9. Maximum calculated dose to a member of the public at the facility boundary.
10. The applicant should provide a discussion of how facility activities have been coordinated with the Hanford Emergency Response Plan, DOE/RL-94-02.

The evaluation should include a list and a description of the factors in 10 CFR 70.22(i)(2) considered in evaluating maximum dose to members of the public. The applicant should demonstrate why the factors used in the evaluation are appropriate when compared to the factors in NUREG-1140. If the factors and evaluation show that the maximum dose to a member of the public offsite due to a release of radioactive materials could not exceed 1 rem effective dose equivalent or the intake of soluble uranium of 2 milligrams, no emergency plan is required in accordance with 10 CFR 70.22(i)(1)(i).

8.4.3.2 Emergency Plan

The adequacy of the proposed emergency plan should be reviewed against the requirements in 10 CFR 70.22(i)(3), and the specific criteria given in the following sections of the SRP. In general the plan should be consistent with the Hanford Emergency Response Plan (DOE/RL 94-02 or replacement, to ensure an integrated response and to eliminate duplication of effort within the planning community. The applicant's emergency plan should be acceptable if the regulatory requirements and the following criteria are met:

8.4.3.2.1 Facility Description

8.4.3.2.1.1 Operational Facilities

Emergency Management

The emergency plan should include a description of the facility and site, the area near the site, and the licensed activities conducted at the facility sufficient to support emergency management activities. The description should include the following:

1. A detailed drawing of the site showing:

    a. onsite and near offsite (within 1 mile) structures with building numbers and labels,
    b. roads and parking lots onsite and main roads near the site,
    c. site boundaries, showing fences and gates,
    d. major site features, and
    e. water bodies within approximately 1 mile.

2. A general area map (approximately 16.09 km [10-mile] radius), a United States Geological Survey topographical quadrangle (7 ½ minute series; including the adjacent quadrangle(s) if site is located less than 1.609 km (1 mile) from the edge of the quadrangle), and a map or aerial photograph indicating onsite structures and near-site structures (about 1.609 km [1-mile] radius). The map should include the location of sensitive facilities near the site such as hospitals, schools, nursing homes, nearest residents, fire department, prisons, and environmental sampling locations, and other structures and facilities important to emergency management.

3. The stack heights, typical stack flow rates, and the efficiencies of any emission control devices.

4. A general description of licensed and other major activities conducted at the facility, and the type, form, and quantities of radioactive and other hazardous materials normally onsite, by location (use and storage) and building, and hazardous characteristics (exposure rates, pH, temperature, and other characteristics) important to emergency management.

5. Certification that the applicant has met responsibilities under Emergency Planning and Right To Know Act of 1986, Title III, Public Law 99-499, in accordance with 10 CFR 70.22(i)(3)(xiii).

8.4.3.2.2 Onsite and Offsite Emergency Facilities

The emergency plan should include a list and description of onsite and offsite facilities that could be relied upon in the event of an emergency. The description should include the following:

1. A list and description of both onsite and offsite emergency facilities by location and purpose of the facility.

2. A description of emergency monitoring equipment which is available for personnel and area monitoring, as well as that for assessing the release of radioactive or hazardous materials to the environment.

3. A description of the onsite and offsite services which support emergency response operations. The following are included:

   a. decontamination facilities,
   b. medical treatment facilities,
   c. first aid personnel,
   d. fire fighters,
   e. law enforcement assistance, and
   f. ambulance services.

4. In addition, the applicant should have emergency facilities, equipment, and resources, which are ready to support emergency response operations, including the following:

   a. Facilities of adequate size and appropriate location that are designated, equipped, and ready for emergency use,

   b. Adequate backup facilities required by the emergency plan and supporting documents that are available and ready for use,

   c. Appropriate equipment and supplies necessary to support emergency response activities that are accessible during accident conditions,

   d. Emergency equipment that is inventoried, tested, and serviced on a periodic basis to ensure accountability and reliability,

   e. Sufficient reliable primary and backup communications channels that are available to accommodate emergency needs,

   f. Offsite emergency resources and services that are identified, and are ready to ensure their timely mobilization and use,

   g. Operational engineering information, such as current as-built drawings and procedures, that are readily available in the emergency facilities,

   h. Sufficient equipment for personnel protection and monitoring, and

   i. Systems in place to alert onsite and offsite personnel in the event of an emergency.

   j. Specific discussion of coordination with Hanford Emergency Response Plan and use of shared or relied upon DOE emergency facilities and systems.

8.4.3.2.3 Types of Accidents

The emergency plan should include a description for each accident identified by the ISA for which protective actions may be needed. The description should include:

1. The process and physical location(s) where the accidents could occur,

2. Complicating factors and possible onsite and offsite consequences, including nonradioactive hazardous material releases that could impact emergency response efforts,

3. The accident sequence that has the potential for the greatest radiological and toxic chemical impact, and

4. Figure(s) projecting dose and toxic substance concentration as a function of distance and time for various meteorological stability classes.

8.4.3.2.4 Classification of Accidents

1. The emergency plan classification system should include the following two classifications:

   - "Alert": Events that may occur, are in progress, or have occurred that could lead to a release of radioactive material or hazardous chemicals incident to the process, but the release is not expected to require a response by an offsite response organization to protect persons offsite[2].

   - "Site area emergency": Events that may occur, are in progress, or have occurred that could lead to a significant release of radioactive material or hazardous chemicals incident to the process that could require a response by offsite emergency response organizations to protect persons offsite.

2. For each accident in the emergency plan, the classification (alert or site area emergency) that is expected for each accident is identified.

3. The emergency plan should specify emergency action levels (EALs) at which an alert or site area emergency will be declared. EALs are specific conditions that require emergency response measures to be performed. The applicant's EALs are consistent with Appendix A of Regulatory Guide 3.67 and are compared with the Environmental Protection Agency's Protective Action Guides (EPA 400-R-92-001, May 1992 Revision). Transportation accidents more than 1 mile from the facility are not classified.

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[2] For facilities located on DOE sites, offsite would start at the applicant's facility boundary and as a minimum be comprised of a portion of the DOE site.

4. The emergency plan should designate the personnel positions and alternates with the responsibility for accident classification during normal and back shift hours.

5. The classification scheme needs to be coordinated with the encompassing Hanford Emergency Response Plan with differences in classifications noted and understood.

6. Dependent on results of the Hazards Analysis portion of the ISA, an additional emergency classification of general emergency may need to be incorporated into the license.

8.4.3.2.5 Detection of Accidents

The emergency plan should describe, for each type of accident identified, the following:

1. The means of detecting the accident,

2. The means of detecting any release of radioactive or other hazardous material,

3. The means of alerting the operating staff, and

4. The anticipated response of the operating staff.

8.4.3.2.6 Mitigation of Consequences

1. The emergency plan should describe for each accident identified, adequate measures and equipment for safe shutdown and for mitigating the consequences to workers onsite and offsite as well as to the public offsite.

2. For impending danger from an accident initiator, the application should describe the following:

    a. The criteria that will be used to determine whether a single process or the entire facility will be shut down,

    b. The steps that will be taken to ensure a safe orderly shutdown of a single process or the entire facility,

    c. The approximate time required to accomplish a safe shutdown of processes, and

    d. The compensatory measures required for safety during the shutdown period following an accident.

8.4.3.2.7 Assessment of Releases

1. The emergency plan should describe the applicant's procedures to be used to promptly and effectively assess the release of radioactive material or hazardous chemicals associated with the processing of radioactive material. The description includes:

a. The procedures for estimating or measuring the release rate or source term,

b. Valid computer codes used to project doses or concentrations to the public or environment and associated assumptions, along with adequate justifications to show the validity of the assumptions,

c. The types, methods, frequencies, implementation time, and other details of onsite and offsite sampling and monitoring that will be performed to assess a release of radioactive material or hazardous chemicals, and

d. Method for assessing collateral damage to the facility, especially safety controls.

2. The emergency plan should describe the applicant's procedure for validating any code used to assess releases of radioactive material or hazardous chemicals.

8.4.3.2.8 Responsibilities

The emergency plan should describe the emergency response organization and administration which ensures effective planning, implementation, and control of emergency preparedness activities and meet the following criteria:

1. The organizational structure and chain of command are clearly defined,

2. Staffing and resources are sufficient to accomplish assigned tasks,

3. Responsibilities and authority for each management, supervisory, and professional position are clearly defined. Responsibility is assigned for the coordination of onsite and offsite radiation/hazardous material emergency response preparedness,

4. Interfaces with supporting groups, both onsite and offsite, are clearly defined,

5. Mutual cooperation agreements exist with local agencies such as fire, police, ambulance/rescue, and medical units, in addition to DOE and its implementation of its Site Emergency Plan,

6. Plant management controls include audit and assessment (SRP Section 11.7) of emergency preparedness to ensure site readiness to handle emergencies and to identify and correct problems,

7. The onsite emergency response organization as described provides reasonable assurance of effective command and control of the site during the assessment, mitigation, and recovery phase of an accident,

8. The emergency public information staff provides advance and ongoing information to the media and public on subjects that would be discussed during an emergency, such as radiation hazards, chemical hazards, site operation, and site emergency plans, and

9. The schedule of emergency preparedness procedure development provides for availability of procedures to support start-up and operation of new processes/facilities onsite.

8.4.3.2.9 Notification and Coordination

1. The emergency plan should provide reasonable assurance that emergency notification procedures will enable the emergency organization to correctly classify emergencies, notify emergency response personnel, and initiate or recommend appropriate actions in a timely manner, based on the following:

   a. Classification of emergency events are based on the current emergency plan.

   b. Notification procedures minimize distractions of shift operating personnel and include concise, preformatted messages. Appropriate follow-up messages to offsite authorities are issued in a timely manner.

   c. Information on the nature and magnitude of the hazards are made available to appropriate emergency response personnel.

   d. Radiological and chemical source term data are available to the command post, technical support center, emergency operation center, and appropriate State personnel, in cooperation with NRC.

   e. When available, offsite field monitoring data are logged, compared with source term data, and used in the protective action recommendation process.

   f. Protective Action Guides are available and used by appropriate personnel in a timely manner.

   g. The emergency public information program ensures timely dissemination of accurate, reliable, and understandable information.

   h. Systems are in place, if required, to alert, notify, and mobilize onsite and offsite response personnel in the event of an emergency.

   i. Notification and coordination with responsible parties when some personnel, equipment, and facility components are not available.

2. The emergency plan should describe how and by whom the following actions will promptly and effectively be taken:

   a. Decision to declare an alert or site area emergency,

b. Activation of onsite emergency response organization during all shifts,

c. Prompt notification of DOE in coordination with its Site Emergency Plan and offsite response authorities that an alert or site area emergency has been declared, including the licensee's initial recommendation for offsite protective actions (normally within 15 minutes),

d. Notification to the NRC Operations Center (as soon as possible and, in any case, no later than one hour after a declared emergency),

e. Decision on what onsite protective actions to initiate,

f. Decision on what offsite protective actions to recommend,

g. Decision to request support from offsite organizations, and

h. Decision to terminate the emergency or enter recovery mode.

8.4.3.2.10 Information To Be Communicated

The emergency plan should describe the information to be communicated during an emergency including the following:

1. A standard reporting checklist to facilitate timely notification,

2. The types of information to be provided concerning facility status, radioactive or hazardous chemical releases, and protective action recommendations,

3. A description of preplanned protective action recommendations to be made to each appropriate offsite organization,

4. The offsite officials to be notified, as a function of the classification of the event,

5. The recommended actions to be implemented by offsite organizations for each accident treated in the emergency plan.

6. The information to be communicated should be coordinated with the prevailing Hanford Emergency Response Plan, to effectively communicate the necessary information.

8.4.3.2.11 Training

The emergency plan should include an adequate training program for onsite and offsite emergency response personnel to ensure knowledge of the emergency plan, the Hanford Emergency Response Plan, assigned duties, and effectively respond to an actual emergency. The description includes:

1. The topics and general content of training programs used for training the onsite and offsite emergency response personnel to satisfy the objectives described above,

2. The administration of the training program, including responsibility for training, the positions to be trained, the schedules for training, the frequency of retraining, use of team training and the estimated number of hours of initial training and retraining,

3. The training to be provided on the use of protective equipment such as respirators, protective clothing, monitoring devices, and other equipment used in emergency response,

4. The training program for onsite personnel who are not members of the emergency response staff, and

5. The instructions and tours that will be provided to fire, police, medical, and other emergency personnel to the extent necessary commensurate with the results of the ISA.

8.4.3.2.12 Safe Shutdown (recovery and plant restoration)

The emergency plan should describe the plans for adequately restoring the facility to a safe status after an accident and recovery after an emergency. The description should include:

1. Appropriate methods and responsibilities for assessing the damage to and the status of the facility's capabilities to safely control radioactive material or hazardous chemicals associated with the process,

2. Procedures for promptly determining the actions necessary to reduce any ongoing releases of radioactive or other hazardous chemicals and to prevent further incidents,

3. Provisions for promptly and effectively accomplishing required restoration action, and

4. Describing the key positions in the recovery organization.

8.4.3.2.13 Exercises and Drills

The emergency plan should commit to conducting exercises and drills in a manner that demonstrates the capability of the organization to plan and perform an effective response to an emergency including effective coordination with DOE. An adequate plan should demonstrate the following:

1. Task-related knowledge is demonstrated through periodic participation by all qualified individuals for each position in the emergency response organization,

2. Drill performance is assessed against specific scenario objectives, using postulated accidents, that adequately test personnel, equipment, and resources, including previously identified weaknesses,

3. Effective player, controller, evaluator, and observer pre-drill briefings are conducted,

4. Scenario data and exercise messages provided by the controllers effectively maintain the time line and do not interfere with the emergency organization's response to exercise scenario events, except where safety considerations are involved,

5. Trained evaluators are used to identify and record participant performance, scenario strengths and deficiencies, and equipment problems,

6. Prestaging of equipment and personnel is minimized to realistically test the activation and staffing of emergency facilities,

7. Critiques are conducted in a timely manner and include a follow-up plan for correcting identified weaknesses and improving training effectiveness,

8. Emergency drills demonstrate that resources are effectively used to control the site, to mitigate further damage, and to control radiological/chemical releases, to perform required onsite activities under simulated radiation/airborne and other emergency conditions, to provide accurate assessments and status during an accident, and to initiate recovery,

9. Emergency drills demonstrate personnel protection measures, including controlling and minimizing hazards to individuals during events such as fires, medical emergencies, mitigation activities, search and rescue, and other similar events,

10. The emergency drill demonstrates that onsite communications effectively support emergency response activities,

11. The emergency drill demonstrates that the emergency public information organization disseminates accurate, reliable, timely, and understandable information,

12. Provisions are made for conducting quarterly communications checks with offsite response organizations, and

13. Offsite organizations are invited to participate in the biennial onsite exercise that tests the major elements of the emergency plan and response organizations.

8.4.3.2.14 Responsibilities for Developing and Maintaining Current the Emergency Program and Its Procedures

The emergency plan should describe the responsibilities for developing and maintaining the emergency program and its procedures. The description should include:

1. The means for ensuring that the revisions to the emergency plan and the procedures which implement the emergency plan are adequately prepared, kept up to date normally

(within 30 days of any changes), and distributed to all affected parties including the NRC, and

2. The provisions for approval of the implementing emergency procedures, making and distributing changes to the procedures, and ensuring that each person responsible for an emergency response function has immediate access to a current copy of emergency procedures. Provisions for approval of changes to the emergency plan and the procedures and those individuals authorized to make these changes are clearly stated.

3. Procedures for allowing offsite response organizations 60 days to comment on the emergency plan before submitting it to the NRC, and to provide NRC any comments received within 60 days along with the plan.

4. Procedures for modifying the emergency plan in accordance with 10 CFR 70.32(i).

5. Procedures to ensure coordination with DOE to ensure the maintenance of the emergency plan is in concert with the Hanford Emergency Response Plan.

8.5 REVIEW PROCEDURES

8.5.1 Acceptance Review

The primary reviewer should evaluate the application to determine whether it addresses the "Areas of Review" discussed in Section 8.3, above. If significant deficiencies are identified, the applicant should be requested to submit additional material before the start of the safety evaluation.

8.5.2 Safety Evaluation

After determining that the application is acceptable for review in accordance with Section 8.5.1, above, the primary reviewer should perform a safety evaluation against the acceptance criteria described in Section 8.4. If during the course of the safety evaluation, the primary reviewer determines the need for additional information, the primary reviewer should coordinate a request for additional information with the licensing project manager.

8.5.2.1 Evaluation That No Emergency Plan Is Required

The primary reviewer should verify that the evaluation is consistent with the potential accident sequences described in the ISA. The ISA reviewer and the primary reviewer should coordinate to assure the resolution of any issues concerning the evaluation relative to ISA information. The final step for the primary reviewer should be to prepare a safety evaluation report (SER) in accordance with Section 8.6 which either agrees with the applicant's conclusion that no emergency plan is required or indicates that the staff does not accept the applicant's evaluation and recommends that an emergency plan be required by the applicant.

8.5.2.2 Emergency Plan

After it is determined that an acceptable application containing an emergency plan has been received from the applicant, the primary reviewer should conduct a complete review and determine its acceptability in accordance with Section 8.4.3.2. The reviewer should verify that emergency planning is consistent with the potential accident sequences described in the ISA. The ISA reviewer and emergency plan reviewer should coordinate to assure the resolution of any issues concerning the emergency plan relative to ISA information.

Although the bulk of this information should be found in the Emergency Management program section of the licensee's submittal, the primary and secondary reviewers should gain familiarity with the site, including the emergency planning zones, demography, land use, plant design and layout, and major accidents postulated by the applicant presented in relevant sections of the SAR. The primary and secondary reviewers should also gain familiarity with proposed radiation protection activities and other operational matters that interface with emergency plans, particularly the programs reviewed against SRP Chapters 4 and 11. Draft and final environmental statements for the proposed facility should be consulted. In addition, for facilities that are located on DOE controlled sites, the respective DOE Emergency Plan should also be consulted. This information may be supplemented by a personal visit to the site by the reviewer and meetings with the applicant. Consultation with FEMA with respect to the relevant state and local government emergency response capabilities may also be necessary.

The final step for the primary reviewer should be to prepare an SER in accordance with Section 8.6, "Evaluation Findings."

8.6 EVALUATION FINDINGS

The primary reviewer writes an SER section addressing each topic reviewed under this SRP Chapter and explains why the NRC staff has reasonable assurance that the emergency management part of the application is acceptable. License conditions may be proposed to impose requirements where the application is deficient. The report includes a summary statement of what was evaluated and why the reviewer finds the submittal acceptable.

The staff can document the evaluation as follows:

> *The staff has evaluated .....* [Insert a summary statement of what was evaluated and why the reviewer finds the submittal acceptable.] *In accordance with 10 CFR 70.22(i), the licensee commits to maintaining and executing an emergency plan for responding to the radiological hazards resulting from a release of radioactive material and to any associated chemical process hazards. The NRC staff reviewed the emergency plan with respect to 10 CFR 70.22(i) and the acceptance criteria in section 8.4.3 of the SRP. NRC staff have determined that the applicant's emergency plan is adequate to demonstrate compliance with 10 CFR 70.22(i), including: (1) the plant is properly configured to limit releases of radioactive materials in the event of an accident, (2) a capability exists for measuring and assessing the significance of accidental releases of radioactive materials, (3) appropriate*

emergency equipment and procedures are provided onsite to protect workers against radiation and other chemical hazards that might be encountered following an accident, (4) a notification system has been established for notifying Federal, State, and local government agencies and recommending appropriate protective actions to protect members of the public, and (5) necessary recovery actions are established for returning the plant to a safe condition following an accident.

8.7 REFERENCES

Code of Federal Regulations, *Title 10, Energy*, Part 70, "Domestic Licensing of Special Nuclear Material."

1. Nuclear Regulatory Commission (U.S.), Washington, D.C. "Domestic Licensing of Special Nuclear Material (10 CFR Part 70)." *Federal Register*: Vol. 64, No. 146. pp. 41338–41357. July 30, 1999.

2. Nuclear Regulatory Commission (U.S.), Washington, D.C. "Part 30 Statements of Consideration and Emergency Preparedness for Fuel Cycle and Other Radioactive Material Licensees." *Federal Register*: Vol. 54, No. 66. pp 14051 - 14059. April 7, 1989.

3. Nuclear Regulatory Commission (U.S.) (NRC), Washington, D.C. NUREG/BR-0150, Vol. 1, Rev. 4, "RTM-96 Response Technical Manual." NRC: Washington, D.C. May 1996.

4. -----. NUREG-0696, "Functional Criteria for Emergency Response Facilities." NRC: Washington, D.C. February 1981.

5. -----. NRC/FCSS Policy and Guidance Directive FC 84-14, Rev. 1, "Radiological Contingency Planning Requirements and License Application Reviews." NRC: Washington, D.C. October 1984.

6. Environmental Protection Agency (EPA) (U.S.). EPA 400-R-92-001, "Manual of Protective Action Guides and Protective Actions for Nuclear Incidents." EPA: Washington, D.C. May 1992.

ENVIRONMENTAL PROTECTION

9.1 PURPOSE OF REVIEW

This purpose of this review is to determine whether there is reasonable assurance that the applicant's proposed environmental protection measures will adequately protect public health and the environment and comply with the regulatory requirements of 10 CFR Parts 20, 51, and 70. In addition to the proposed protection measures, the staff should determine if the applicant needs to submit an Environmental Report for staff use in preparation of either an Environmental Assessment (EA) and Finding of No Significant Impact (FONSI) or an Environmental Impact Statement (EIS) pursuant to 10 CFR Part 51. However, the review of the applicant's Environmental Report is outside the scope of this SRP chapter and is to be conducted separately for implementation of the National Environmental Policy Act (NEPA) requirement. For additional information on Environmental Reports, the reviewer is referred to 10 CFR 51.45(b).

9.2 RESPONSIBILITY FOR REVIEW

Primary: Environmental Engineer/Scientist

Secondary: Licensing Project Manager

Supporting: Primary Reviewer of SRP Chapter 4.0
Primary Reviewer of SRP Chapter 6.0
Primary Reviewer of SRP Chapter 11
TWRS Site Representative

9.3 AREAS OF REVIEW

The review of environmental protection measures should include a review of the applicant's integrated safety analysis (ISA). The following subsections identify the areas of review for each of these components. Greater detail on each component is provided in Section 9.4, which specifies the review acceptance criteria.

The environmental review should focus on that part of the applicant's plant-wide safety program that is established to control and assess the level of radioactive and nonradioactive releases (gaseous, liquid, and solid) to the environment. Therefore, the effluent control portion of the applicant's radiation protection program, as well as effluent and environmental monitoring practices, should be reviewed. In addition, the plant-wide safety program should be reviewed to ensure that the management controls are specified to ensure that these activities meet license objectives.

To receive authorization to possess a critical quantity of special nuclear material, as defined in 10 CFR 70.4, an applicant must also perform an ISA in accordance with 10 CFR Part 70, as

revised[1]. Guidance on the ISA is covered in Chapter 3.0 of this SRP. The environmental safety review of the ISA should include a review of the identified potential accident sequences that result in radiological and nonradiological releases to the environment, as well as the controls specified by the applicant to reduce the risk of these accidents.

The review should examine the date of an application for a license to possess and use special nuclear material for processing and fuel fabrication, scrap recovery, conversion of uranium hexafluoride, or for the conduct of any other activity, which the NRC has determined pursuant to 10 CFR 51 Subpart A will significantly affect the quality of the environment, to verify that the application is submitted at least 9 months before the commencement of construction, as required by 10 CFR Part 70.21(f) and is accompanied by an Environmental Report.

Thus, environmental protection includes four main components: (1) the radiation protection program, (2) effluent and environmental monitoring, (3) the ISA, and (4) provisions for continuing assurance. The areas of review should include the following:

9.3.1 Radiation Protection

- Radiological (i.e., ALARA) goals for effluent control.
- Procedures, engineering controls, and process controls to maintain public doses ALARA
- ALARA reviews and reports to management.
- Waste minimization practices and, for new operations, facility design and operating procedures for waste minimization.

9.3.2 Effluent and Environmental Monitoring

- In-place filter testing procedures for air cleaning systems
- Known or expected concentrations of radionuclides in effluents
- Physical and chemical characteristics of radionuclides in discharges
- Discharge locations
- Environmental media to be monitored and the sample locations
- Sampling collection and analysis procedures, including the minimum detectable concentrations of radionuclides, equipment used, and calibration information
- Action levels and actions to be taken when the levels are exceeded
- Permits, including air discharge and National Pollutant Discharge and Elimination System permits
- Leak detection systems for ponds, lagoons, and tanks
- Pathways analysis methods to estimate public doses
- Recording and reporting procedures, including event notification
- Solid waste handling and disposal programs

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[1] Nuclear Regulatory Commission (U.S.), Washington, D. C. "Domestic Licensing of Special Nuclear Material (10 CFR Part 70)." *Federal Register:* Vol. 64, No. 146. pp. 81338--41357. July 30, 1999.

9.3.3 Integrated Safety Analysis

- Accident sequences (and associated facility processes) which, if unmitigated, result in releases to the environment
- Likelihood and consequences of these accident sequences as they impact the public and the environment
- Controls relied on to reduce the unmitigated risk from "high" risk to an acceptable level
- Availability and reliability of controls

9.3.4 Provisions for Continuing Assurance

The provisions for continuing assurance for environmental protection at the facility include the following areas:

- Organization and Management
- Training and Qualification
- Emergency Plan
- Maintenance and Surveillance
- Audits and Assessments
- Procedures

9.4 ACCEPTANCE CRITERIA

9.4.1 Regulatory Requirements

The applicable and/or relevant requirements for environmental protection are included in the following regulations:

1. 10 CFR Part 20, specifically, radiation protection, the effluent control and treatment measures necessary to meet the dose limits and dose constraints for members of the public specified in Subparts B and D, the minimization of contamination specified in Subpart E, the survey requirements specified in Subpart F, the waste disposal requirements of Subpart K, the records requirements of Subpart L, and the reporting requirements of Subpart M.

2. 10 CFR Part 51, specifically its effluent and environmental monitoring systems that the applicant must establish to provide the information required by 10 CFR 51.60(a).

3. 10 CFR Part 70, specifically an application for a license to possess and use special nuclear material for activities the Commission has determined pursuant to 10 CFR Part 51 will significantly affect the quality of the environment will be filed at least 9 months prior to commencement of construction of the plant or facility and shall be accompanied by an Environmental Report as specified in 10 CFR 70.21(f).

4. 10 CFR Part 70, specifically the proposed facilities and equipment and procedures, including measuring and monitoring instruments and devices and procedures for the disposal of radioactive effluents and wastes that the applicant must demonstrate are adequate to protect public health and the environment as specified 10 CFR 70.22(a)(7) and (8) and 70.23(a)(3) and (4).

9.4.2 Regulatory Guidance

The regulatory guidance for environmental protection is contained in:

1. NRC Regulatory Guide 4.15, "Quality Assurance for Radiological Monitoring Programs (Normal Operations)-Effluent Streams and the Environment."

2. NRC Regulatory Guide 4.16, "Monitoring and Reporting Radioactivity in Releases of Radioactive Materials in Liquid and Gaseous Effluents from Nuclear Fuel Processing and Fabrication Plants and Uranium Hexafluoride Production Plants."

3. NRC Regulatory Guide 4.20, "Constraint on Releases of Airborne Radioactive Materials to the Environment for Licensees Other than Power Reactors."

4. NRC Regulatory Guide 8.37, "ALARA Levels for Effluents from Materials Facilities."

5. NRC Information Notice 94-07, "Solubility Criteria for Liquid Effluent Releases to Sanitary Sewerage Under the Revised 10 CFR Part 20," January 28, 1994.

6. NRC Information Notice 94-23: "Guidance to Hazardous, Radioactive and Mixed Waste Generators on the Elements of a Waste Minimization Program," March 1994.

9.4.3 Regulatory Acceptance Criteria

The applicant's submittal should provide reasonable assurance that the review criteria below are adequately addressed and satisfied for the environmental protection measures. Some of the information may be referenced to other sections of the standard review plan, or incorporated by reference, provided an adequate summary is provided and a single reference essentially contains all of the information.

An applicant's proposed actions for environmental protection should be acceptable if they provide for effluent control as part of the radiation safety program, and effluent and environmental monitoring, in accordance with NRC technical and managerial provisions for continuing assurance.

The acceptance criteria for the radiation protection program, effluent and environmental controls and monitoring, the ISA, and provisions for continuing assurance are given in Sections 9.4.3.1, 9.4.3.2, 9.4.3.3, and 9.4.3.4, respectively.

9.4.3.1 Radiation Protection Program

The proposed radiation protection program should be acceptable from the standpoint of environmental protective measures if it satisfies the following criteria:

1. Radiological (ALARA) Goals for Effluent Control

   ALARA goals are set at a modest fraction (10% to 20%) of the values in Appendix B, Table 2, Columns 1 and 2 and Table 3 and the external dose limit in 10 CFR 20.1302(b)(2)(ii), or the dose limit for members of the public, if the applicant proposes to demonstrate compliance with 10 CFR 20.1301 through a calculation of the TEDE to the individual likely to receive the highest dose. The ALARA program for effluent control should be consistent with guidance found in Regulatory Guide 8.37.

   An applicant's constraint approach should be acceptable if it is consistent with guidance found in Regulatory Guide 4.20 and the applicant's description of the constraint approach provides sufficient detail to demonstrate specific application of the guidance to proposed routine operations and nonroutine operations including anticipated events.

2. Procedures, Engineering Controls, and Process Controls

   The applicant describes and commits to using procedures, engineering controls, and process controls to achieve ALARA goals for effluent minimization. Common control practices include filtration, encapsulation, adsorption, containment, recycling, leakage reduction, and the storage of materials for radioactive decay. Practices for large, diffuse sources such as contaminated soils or surfaces include covers, wetting during routine operations and non-routine operations including anticipated events, and the application of stabilizers. The applicant demonstrates a commitment to reducing unnecessary dose to members of the public and releases to the environment.

3. ALARA Reviews and Reports to Management

   The applicant commits to annual review of the content and implementation of the radiation protection program, which includes the ALARA effluent control program. This review includes analysis of trends in release concentrations, environmental monitoring data, and radionuclide usage, determines whether operational changes are needed to achieve the ALARA effluent goals, and evaluates all designs for system installations or modifications. The applicant also includes a commitment to report the results to senior management along with recommendations for changes in facilities or procedures that are necessary to achieve ALARA goals.

4. Waste Minimization

   The application contains a description of how facility design and procedures for operation will minimize, to the extent practicable, contamination of the facility and the environment,

and minimize, to the extent practicable, the generation of radioactive waste. Waste minimization programs proposed by applicants for both new and existing licenses include:

a. Top management support
b. Identification of responsibilities for waste minimization activities and assessments
c. Methods to characterize waste generation, including types and amounts, and waste management costs, including costs of regulatory compliance, paperwork, transportation, treatment, storage, disposal, etc.
d. Periodic waste minimization assessments to identify waste minimization opportunities and solicit employee or external recommendations
e. Provisions for technology transfer to seek and exchange technical information on waste minimization
f. Provisions to incorporate operational experience
g. Methods for implementation and evaluation of waste minimization recommendations

9.4.3.2 Effluent and Environmental Controls and Monitoring

9.4.3.2.1 Effluent Control and Monitoring

The applicant's effluent monitoring should be acceptable if it meets the following criteria:

1. The known or expected concentrations of radioactive materials in airborne and liquid effluents are below the limits in 10 CFR Part 20, Appendix B, Table 2 or below site specific limits established in accordance with 20.1302(c) and are ALARA.

2. All liquid and airborne effluent discharge locations are identified and monitored. Monitoring locations should be identified, and for those effluent discharge points which have input from two or more contributing sources within the facility, monitoring for each major contributing source should be considered for effective process and effluent control.

   Airborne effluents from all routine operations, and non-routine operations, as well as anticipated events associated with the plant, including effluents from areas not used for processing special nuclear material such as laboratories, experimental areas, storage areas, and fuel element assembly areas, should be continuously sampled. For liquid effluents, representative samples should be taken at each release point for the determination of concentrations and quantities of radionuclides released to an unrestricted area, including discharges to sewage systems. For continuous releases, samples should be continuously collected at each release point. For batch releases, a representative sample of each batch should be collected. If periodic sampling is used in lieu of continual sampling, the applicant shows that the samples are representative of actual releases.

3. Effluents should be sampled unless the applicant has established, by periodic sampling or other means, that radioactivity in the effluent is insignificant and will remain so. In such cases, the effluent should be sampled at least quarterly to confirm that effluents are not significant. For the purposes of this SRP, an effluent is significant if the concentration

averaged over a calendar quarter is equal to 10 percent or more of the appropriate concentration listed in Table 2 of Appendix B to 10 CFR Part 20.

4. Radionuclide specific analyses should be performed on selected composite samples unless (1) the gross alpha and gross beta activities are so low that individual radionuclides could not be present in concentrations greater than 10 percent of the concentrations specified in Table 2 or 3 of Appendix B to 10 CFR Part 20, or (2) the radionuclide composition of the sample is known through operational data, such as the composition of the feed material. Monitoring reports in which estimates of quantities of individual radionuclides are based on methods other than direct measurement should include an explanation and justification of how the results were obtained.

   Examples of cases in which operational data may not be adequate for the determination of radionuclide concentration are (1) plants processing uranium in which extraction, ammonium diuranate precipitation, ion exchange, or other separation processes could result in concentration of thorium isotopes (principally Th-234); (2) plants in which uranium of varying enrichments is processed; and (3) plants processing plutonium in which significant variation in the Pu-238/Pu-239 ratio among batches and the continuous in-growth of Am-241 would preclude the use of feed material data to determine the radionuclide composition of effluents.

   Radionuclide analyses should be performed more frequently than usual under three circumstances: (1) at the beginning of the monitoring program until a predictable and consistent radionuclide composition in effluents is established; (2) whenever there is a significant unexplained increase in gross radioactivity in effluents; or (3) whenever a process change or other circumstance might cause a significant variation in the radionuclide composition.

5. The sample collection and analysis methods and frequencies should be appropriate for the effluent medium and the radionuclide(s) being sampled. Sampling methods ensure that representative samples are obtained by use of appropriate sampling equipment and sample collection and storage procedures. Monitoring instruments should be calibrated at least annually, or more frequently if suggested by the manufacturer.

6. The proposed action levels and actions to be taken if the levels are exceeded are appropriate. The action levels are incremental, such that each increasing action level results in a more aggressive action to assure and control effluents. A slightly higher than normal concentration of a radionuclide in effluent triggers an investigation into the cause of the increase. An action level is specified that will result in the shutdown of an operation if this level is exceeded. These action levels are selected based on the likelihood that a measured increase in concentration could indicate potential violation of the effluent limits.

7. The minimum detectable concentration (MDC) for sample analyses is not more than 5 percent of the concentration limits listed in Table 2 of Appendix B to 10 CFR Part 20. If the actual concentrations of radionuclides in samples are known to be higher than 5 percent of the 10 CFR Part 20 limits, the analysis methods need only be adequate to

measure the actual concentration. However, in such cases, the MDC is low enough to accommodate fluctuations in the concentrations of the effluent and the uncertainty of the MDC.

8. The laboratory quality control (QC) procedures are adequate to support the validity of the analytical results. These QC procedures include the use of established standards such as those provided by the National Institute of Standards and Technology (NIST), as well as standard analytical procedures, such as those established by the National Environmental Laboratory Accreditation Conference.

9. The descriptions of applicable Federal and/or State standards for discharges and any permits issued by local, State, or Federal governments for gaseous and liquid effluents are complete and accurate.

10. If the applicant proposes to adjust the effluent concentrations in Appendix B to 10 CFR 20 in accordance with 20.1302(c) to take into account the actual physical and chemical characteristics of the effluents, the information related to aerosol size distributions, solubility, density, radioactive decay equilibrium, and chemical form is complete and accurate for the radioactive materials to justify the derivation and application of the alternative concentration limits.

11. The systems for the detection of leakage from ponds, lagoons, and tanks are adequate to detect and assure against any unplanned releases to groundwater, surface water, or soil.

12. Releases to sewer systems are controlled and maintained to meet the requirements of 10 CFR 20.2003, including (i) the material is water soluble; (ii) known or expected discharges meet the effluent limits of 10 CFR 20 Appendix B, Table 3; and (iii) the known or expected total quantity of radioactive material released into the sewer system in a year does not exceed 5 Ci (185 GBq) of ^{3}H, 1 Ci (37 GBq) of ^{14}C, and 1 Ci (37 GBq) of all other radioactive materials combined. Solubility is determined in accordance with the procedure described in NRC Information Notice 94-07.

13. Reporting procedures comply with the requirements of 10 CFR 70.59 and the guidance specified in Regulatory Guide 4.16. Reports of the concentrations of principal radionuclides released to unrestricted areas in liquid and gaseous effluents are provided and include the MDC for the analysis and the error for each data point.

14. If the licensee proposes to demonstrate compliance with 10 CFR 20.1301 through a calculation of the TEDE to the individual likely to receive the highest dose in accordance with 20.1302(b)(1), calculation of the TEDE by pathways analyses uses appropriate models and codes and assumptions that accurately represent the facility, the site, and the surrounding area; assumptions are reasonable; input data is accurate; all applicable pathways are considered; and the results are interpreted correctly.

15. The applicant's methods for determining the dose to the maximally exposed individual during normal facility operations and anticipated events should be acceptable if they are

consistent with NCRP Report No. 123, "Screening Models for Releases of Radionuclides to Atmosphere, Surface Water, and Ground," January 1996. The applicant could use computer codes as acceptable tools for pathways analysis if the applicant is able to demonstrate that the code has undergone validation and verification to demonstrate the validity of estimates developed using the code for established input sets. Dose conversion factors used in the pathways analyses should be acceptable if they are based on the methodology described in ICRP 30, "Limits for Intakes of Radionuclides by Workers," as reflected in Federal Guidance Report 11. If the applicant use's alternative methods then these should be considered acceptable with appropriate justification.

16. The applicant's procedures and facilities for solid and liquid waste handling, storage and monitoring result in safe management and timely disposition of the material.

9.4.3.2.2 Environmental Monitoring

The scope of the applicant's environmental monitoring should be acceptable if it is commensurate with the scope of activities at the facility and the expected impacts of routine operations and non-routine operations including anticipated events as identified in the environmental report and meets the following criteria:

1. Background and baseline concentrations of radionuclides in environmental media have been established through sampling and analysis.

2. A preoperational monitoring program is initiated prior to operation. The preoperational program should be of sufficient length to allow a sufficient data base for comparison with operational data.

3. Monitoring includes sampling and analyses for important pathways for the anticipated types of radionuclides released from the facility into the environment from routine and anticipated events during nonroutine operations, including air, surface water, groundwater, soil, sediments, and vegetation, as appropriate. Important environmental media are sampled to estimate radionuclide concentrations in important biota.

4. The description of monitoring identifies adequate and appropriate sampling locations and frequencies for each environmental medium, the frequency of sampling, and the analyses to be performed on each medium. Sampling methods ensure that representative samples are obtained by use of appropriate sampling equipment, sample collection, and sample storage procedures.

5. Monitoring procedures employ acceptable analytical methods and instrumentation to be used, and monitoring procedures and analytical methods are subject to quality controls. The applicant commits to a program of instrument maintenance and calibration appropriate to the instrumentation, as well as participation in round-robin measurement comparisons if the applicant proposes use of its own analytical laboratory for analysis of environmental samples.

6. Appropriate action levels and actions to be taken if the levels are exceeded are specified for each environmental medium and radionuclide.

   Action levels are selected based upon a pathways analysis that demonstrates that below those concentrations, doses to the public will be below the limits in 10 CFR Part 20, Subpart B, and are ALARA. The action levels specify the concentrations at which an investigation would be performed and levels at which process operations would be shut down.

7. MDCs are specified for sample analyses, and are at least as low as those selected for effluent monitoring in air and water. MDCs for sediment, soil, and vegetation are selected based upon the action levels to ensure sampling and analytical methods are sensitive and reliable enough to support application of the action levels.

8. Data analysis methods and criteria to be used for evaluating and reporting the environmental sampling results are appropriate and will indicate when an action level is being approached in time to take corrective actions.

9. The description of the status of all licenses, permits, and other approvals of plant operations required by Federal, State and local authorities is complete and accurate.

10. Environmental monitoring is adequate to assess impacts to the environment from potential radioactive and nonradioactive releases as identified in high and medium risk accident sequences in the ISA.

9.4.3.3 Integrated Safety Analysis

In accordance with 10 CFR Part 70, as revised, TWRS applicants are required to perform an ISA. The applicant's treatment of environmental protection in the ISA should be acceptable if it fulfills the following criteria:

1. The ISA summary should provide a complete list of accident sequences with potential for affecting the environment consistent with the performance requirements contained in 10 CFR Part 70, as revised.

2. The ISA should provide a reasonable estimate for the environmental effects of each accident sequence identified.

3. The ISA should use acceptable methods for estimating environmental effects from accident sequences which result in radiological releases to the environment. Acceptable methods are described in NUREG/CR-6410, "Nuclear Fuel Cycle Facility Accident Analyses Handbook." Models used for consequence analysis should be verified and validated.

9.4.3.4 Provisions for Continuing Assurance

The applicant's provisions for continuing assurance of environmental protection at its facility should be acceptable if the submittal reflects environmental protection in other portions of the application:

1. Organizational Structure (Section 2.1)
2. Emergency Plan (Chapter 8.0)
3. Maintenance and Surveillance (Section 11.2)
4. Training and Qualification (Section11.4)
5. Audits and Assessments (Section 11.3)
6. Procedures (Section 11.9)

9.5 REVIEW PROCEDURES

9.5.1 Acceptance Review

The primary reviewer should evaluate the application to determine whether it addresses the "Areas of Review" discussed in Section 9.3, above. If significant deficiencies are identified, the applicant should be requested to submit additional material before the start of the safety evaluation.

9.5.2 Safety Evaluation

After determining that the application is acceptable for review in accordance with Section 9.5.1, above, the primary reviewer should perform a safety evaluation against the acceptance criteria described in Section 9.4. In addition, the review of renewal or amendment applications should include review of inspection reports and semi-annual effluent reports submitted in accordance with 10 CFR 70.59 to assure licensee performance in environmental protection. The safety evaluation forms the basis for staff findings, and supports the reviewers' conclusions.

The primary reviewer should review the radiation protection program. This review should be coordinated with a supporting reviewer, primary reviewer of Chapter 4.0, and should focus on the applicant's program to maintain public doses ALARA.

The primary reviewer should review the ISA. Evaluation of the ISA should be coordinated with other technical reviewers by the Project Manager for the facility (Secondary Reviewer). All accident sequences identified in the ISA that can have significant consequences due to releases to unrestricted areas, should be reviewed to determine that the list of potential accidents is complete and properly identified. This review should be supported by other reviewers of Chapter 3.0 of this SRP, particularly the primary reviewers of Chapters 4.0 and 6.0 (Supporting Reviewers).

Environmental Protection

For renewal and amendment applications, review of environmental protection by the primary reviewer should include coordination with the TWRS Site Representative responsible for environmental protection (Supporting Reviewer).

Other supporting reviewers should confirm that provisions made in the applicant's submittal are in accordance with specified sections of the SRP. For example, the primary reviewer of Section 11.4, as a supporting reviewer, should establish that the program described by the applicant should provide reasonable assurance that environmental protection staff and management will be appropriately trained.

When the safety evaluation is complete, the primary staff reviewer, with assistance from the other reviewers, should prepare the environmental protection input for the SER as described in Section 9.6 using the acceptance criteria from Section 9.4.

9.6 EVALUATION FINDINGS

The staff reviewers should verify that the information submitted by the applicant is in accordance with 10 CFR Parts 20, 51, and 70, and that this information is consistent with the guidance in this SRP as it applies to environmental protection. In the input to the SER, the primary reviewer should document the bases for determining the adequacy of the application with respect to environmental protection, and should recommend additional license conditions in areas where the license application is not adequate. The primary reviewer should also describe the applicant's approach to ensuring the quality and reliability of the controls required for environmental protection.

Often, environmental protection is reviewed and evaluated in conjunction with the environmental report, and the environmental protection function is summarized in the EA or EIS. However, the EA or EIS does not become part of the license. Issues identified during the review should be discussed briefly in the SER, and any recommended license conditions based on the analysis in the EA or EIS should be added to the license.

If an EA and EIS were prepared for the licensing action, the date the documents were issued should be reported in the environmental safety section of the SER. If the EA resulted in a FONSI, the FONSI's publication date in the Federal Register should be included in the SER. If an EIS is prepared, the SER should include the Federal Register publication date for the Record of Decision. When applicable, the SER should also document the determination that an action meets a categorical exclusion.

The following language would be appropriate for a licensing action that required an EIS in accordance with 10 CFR 51.20.

> *The applicant has committed to adequate environmental protection measures. The NRC staff concludes that there is reasonable assurance that the applicant's environmental protection measures will be adequate to protect public health and the environment and*

comply with the regulatory requirements in 10 CFR Parts 20, 51, and 70. The bases for these conclusions are:

[State the bases for the conclusion, including any recommended license conditions.]

The NRC staff prepared an environmental impact statement (EIS) [publication date] for this licensing action as required by 10 CFR 51.20. Based on the EIS, the NRC stated in its Record of Decision [publication date in the Federal Register] that the preferred option was [state preferred option here].

9.7 REFERENCES

1. American National Standards Institute (ANSI). N13.1-1982, "Guide to Sampling Airborne Radioactive Materials in Nuclear Facilities." ANSI: New York. 1982

2. American National Standards Institute (ANSI). N42.18-1980, "Specification and Performance of On-site Instrumentation for Continuously Monitoring Radioactive Effluents." ANSI: New York. 1980.

3. National Council on Radiation Protection and Measurements (NCRP). Report No. 123 I & II, "Screening Models for Releases of Radionuclides to Atmosphere, Surface Water, and Ground." NCRP: Maryland. January 1996.

4. Nuclear Regulatory Commission (U.S.) (NRC). Information Notice 94-23, "Guidance to Hazardous, Radioactive and Mixed Waste Generators on the Elements of a Waste Minimization Program." NRC: Washington, D.C. March 25, 1994.

5. Nuclear Regulatory Commission (U.S.) (NRC). Information Notice 94-07, "Solubility Criteria for Liquid Effluent Releases to Sanitary Sewers Under the Revised 10 CFR Part 20." NRC: Washington, D.C. January 28, 1994.

6. Nuclear Regulatory Commission (U.S.) (NRC). NMSS/FCSS/Fuel Cycle Licensing Branch, Rev. 6, "Materials Licensing Procedures Manual." NRC: Washington, D.C. April 1998.

7. Nuclear Regulatory Commission (U.S.) (NRC). Regulatory Guide 4.15, Rev. 2, "Quality Assurance for Radiological Monitoring Programs (Normal Operations)–Effluent Streams and the Environment." NRC: Washington, D.C. February 1979.

8. Nuclear Regulatory Commission (U.S.) (NRC). Regulatory Guide 4.16, Rev. 2, "Monitoring and Reporting Radioactivity in Releases of Radioactive Materials in Liquid and Gaseous Effluents from Nuclear Fuel Processing and Fabrication Plants and Uranium Hexafluoride Production Plants." NRC: Washington, D.C. December 1985.

9. Nuclear Regulatory Commission (U.S.) (NRC). Regulatory Guide 4.20, "Constraint on Releases of Airborne Radioactive Materials to the Environment for Licensees other than Power Reactors." NRC: Washington, D.C. December 1996.

10. Nuclear Regulatory Commission (U.S.) (NRC). Regulatory Guide 8.37, "ALARA Levels for Effluents from Materials Facilities." NRC: Washington, D.C. July 1993.

11. Nuclear Regulatory Commission (U.S.) (NRC). NUREG/CR-6410, "Nuclear Fuel Cycle Accident Analysis Handbook." NRC: Washington, D.C. 1998.

12. Nuclear Regulatory Commission (U.S.) (NRC). NUREG-1520 (DRAFT), "Draft Standard Review Plan for the Review of a License Application for a Fuel Cycle Facility." NRC: Washington, D.C. July 1999.

DECOMMISSIONING

Decommissioning and financial assurance review guidance for the TWRS-P project will be addressed on a generic basis in the context of DOE external regulation; therefore, it will not be included in this SRP.

MANAGEMENT MEASURES
11.1 QUALITY ASSURANCE

11.1.1 PURPOSE OF REVIEW

The purpose of this review is to establish that there is reasonable assurance that the applicant has the quality assurance (QA) elements that are needed to provide reasonable assurance that all items relied on for safety[1] will perform their designated safety functions as required by 10 CFR Part 70, "Domestic Licensing of Special Nuclear Material, as revised[2].

11.1.2 RESPONSIBILITY FOR REVIEW

Primary: QA Engineer/Specialist

Secondary: Licensing Project Manager

Supporting: Site Representative/Facility Inspector
Staff Reviewers of applicable SRP Chapters 3 through 15

11.1.3 AREAS OF REVIEW

The regulation, 10 CFR Part 70, as revised, requires that the applicant establish appropriate QA elements to ensure that all items relied on for safety perform their designated safety functions and are continually available and reliable. The following areas should be reviewed:

1. Organization
2. QA Function[3]
3. Design Control
4. Procurement Document Control
5. Instructions, Procedures,[4] and Drawings

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[1] "Items relied on for safety" is defined in the proposed 10 CFR 70, as revised, as "structures, systems, equipment, components, and activities of personnel that are relied on to prevent potential accidents at the facility that could exceed the performance requirements specified in § 70.61 or to mitigate their potential consequences."

[2] Nuclear Regulatory Commission (U.S.), Washington, D.C. "Domestic Licensing of Special Nuclear Material (10 CFR Part 70)." *Federal Register:* Vol. 64, No. 146. pp. 41338-41357. July 30, 1999.

[3] SRP Section 11.4 addresses training and qualification of plant personnel. Section 2 of SRP Appendix C on QA addresses training and qualification of other personnel.

[4] SRP Section 11.5 addresses plant procedures. Section 5 of SRP Appendix C on QA addresses other procedures.

Management Measures

6. Document Control
7. Control of Purchased Items
8. Identification and Control of Items
9. Control of Special Processes
10. Inspection
11. Test Control
12. Control of Measuring and Test Equipment
13. Handling, Storage, and Shipping
14. Inspection, Test, and Operating Status
15. Nonconformances
16. Corrective Action
17. QA Records
18. Audits and Assessments[5]
19. Applicant's Provisions for Continuing QA

11.1.4 ACCEPTANCE CRITERIA

11.1.4.1 Regulatory QA Requirements

The regulatory QA requirements are addressed in the following:

Nuclear Regulatory Commission (U.S.), Washington, D.C. "Domestic Licensing of Special Nuclear Material (10 CFR Part 70)." *Federal Register:* Vol. 64, No. 146. pp. 41338-41357. July 30, 1999.

Specific references are as follows:

1. In § 70.4, "Definitions," the term management measures is defined to "include configuration management, maintenance, training and qualifications, procedures, audits and assessments, incident investigations, records management, and other quality assurance elements."

2. In § 70.62(d), the applicant or licensee is required to establish management measures to provide continuing assurance of compliance with the performance requirements.

3. In § 70.64(a)(1), the design of new facilities or the design of new processes at existing facilities is required to be developed and implemented in accordance with management measures.

4. In § 70.65(a), each application is required to include a description of the management measures.

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[5] Guidance for audits and assessments is given in SRP Section 11.7 as referenced in SRP Appendix C on QA.

11.1.4.2 Regulatory Guidance

Guidance for QA is addressed in the following:

American Society of Mechanical Engineers, "Quality Assurance Requirements for Nuclear Facility Applications." (An American National Standard), NQA-1-1994, New York. 1994.

While this standard has separate sections for "requirements" and "guidance," NRC's regulatory QA requirements exist only in the applicable Commission regulations.

11.1.4.3 Regulatory Acceptance Criteria

The NRC reviewers should find that the applicant's submittal regarding QA provides reasonable assurance that the regulatory acceptance criteria below are adequately addressed and satisfied. Some of the information may be referenced to other sections of the SRP, or incorporated by reference, provided that these references are clear and specific.

The Integrated Safety Analysis (ISA - see SRP Chapter 3) should identify the items and related controls that are required for safety and the degree of their importance. The graded approach for the application of QA should be described unless the applicant chooses to apply the highest level of QA and quality control to all items relied on for safety.

Depending on whether the applicant chooses Option A or Option B noted in SRP Section 11.1.5.2 below, the application should address the criteria specified in that subsection. That is, if Option A is used, the application should (a) include a commitment that the applicant will implement and maintain its QA elements to comply with the applicable "requirements" of NQA-1-1994[6] (that is, the basic and supplemental "requirements" of Parts I and II) or equivalent and should (b) be responsive to the three regulatory acceptance criteria given below. Note that, if Option A is used, only a verification of that commitment and of the response to the regulatory acceptance criteria given below should be performed.

1. Organization - The applicant should describe the organizational structure, functional responsibilities, charts of the lines of responsibilities, interrelationships, and areas of responsibility and authority for all organizations performing activities relied on for safety, including the applicant's organization and, if applicable, the organization of the applicant's principal contractors (architect/engineer, constructor, construction manager, and/or operator). Persons or organizations responsible for ensuring that appropriate QA has been established and verifying that activities affecting quality/safety have been correctly performed should have sufficient authority, access to work areas, and organizational independence to carry out their responsibilities.

[6] This SRP section refers to regulatory QA requirements and NQA-1 "requirements." Regulatory QA requirements are given in the Part 70, as revised. NQA-1 "requirements" are the Basic and Supplementary Requirements given in Parts I and II of ASME NQA-1-1994.

Management Measures

2. QA Function - QA should be well-documented, planned, implemented, and maintained to ensure the availability and reliability of controls relied on for safety. It should be implemented during all phases of the facility's life. It should be functional prior to performing the ISA required by Part 70, as revised.

3. Applicant's Provisions for Continuing QA - The applicant's provisions for continuing QA should address review and updates based on reorganizations, revised activities, lessons learned, changes to applicable regulations, and other QA changes.

If Option B is used, the application should address the checklist items in SRP Appendix C on QA.

In either case, the review of procedures that the applicant uses to meet its QA commitments would be performed during NRC inspections as part of a determination of adequacy of the licensee's QA implementation.

11.1.5 REVIEW PROCEDURES

11.1.5.1 Acceptance Review

The primary reviewer should evaluate the application to determine whether it addresses the "Areas of Review" listed in Section 11.1.3, above, regarding the applicant's (and its principal contractors') QA. If significant deficiencies are identified, the applicant should be requested to submit additional material before the start of the safety evaluation. Note that the applicant's commitment to implement and maintain its QA in conformance with the applicable basic and supplemental "requirements" of Parts I and II of ASME NQA-1-1994 or equivalent should satisfy the acceptance review criteria.

11.1.5.2 Safety Evaluation

After determining that the application is acceptable for review in accordance with Section 11.1.5.1, above, the primary reviewer should review the application to determine whether the applicant, for items relied on for safety has either:

Option A. Addressed the regulatory acceptance criteria given in Subsection 11.1.4.3 above and provided a commitment to implement and maintain its QA in conformance with the applicable "requirements" of Parts I and II of NQA-1-1994 or equivalent.

OR

Option B. Addressed the checklist provided in SRP Appendix C on QA.

In either case, the applicant should also (a) describe how the QA will be graded for items of lesser or no effect on consequences of concern (unless the applicant chooses to apply the highest level of QA and quality control to all items relied on for safety) and (b) list the items

relied on for safety as determined by the applicant's ISA. The primary reviewer should determine whether the applicant and its principal contractors have adequately planned for QA to be accomplished and whether necessary QA policies, procedures, and instructions will be in place before personnel begin activities relied on for safety. Some of the information may be referenced to other sections of the application, or incorporated by reference, provided that these references are clear and specific.

The secondary reviewer should confirm that the applicant and the applicant's principal contractors' QA commitments are consistent with other sections of the submittal. The secondary reviewer should also integrate the QA input into the Safety Evaluation Report (SER).

The supporting reviewer (Site Representative/Fuel Cycle Facility Inspector) should become familiar with the applicant's and principal contractors' QA commitments and determine whether ongoing activities are in agreement with them.

The other supporting reviewers (Staff Reviewers of applicable SRP Chapters 3 through 12) should determine, within their areas of review, whether items relied on for safety are specified to require the appropriate level of QA.

On the basis of its review, the staff may request that the applicant provide additional information or modify the application to meet the acceptance criteria. The staff or applicant may also propose license conditions to ensure the applicant's QA meets the acceptance criteria. The review should result in a determination that there is reasonable assurance that the applicant's and the applicant's principal contractors' QA will provide reasonable assurance that items relied on for safety will perform their safety function in a satisfactory manner.

When the safety evaluation is complete, the primary staff reviewer, with assistance from the other reviewers, should prepare the QA input for the SER as described in Section 11.1.6 using the acceptance criteria from Section 11.1.4.

11.1.6 EVALUATION FINDINGS

The staff's evaluation should verify that the license application provides sufficient information to satisfy the regulatory QA requirements of Part 70, as revised (as listed in SRP Section 11.1.4.1), and that the regulatory acceptance criteria in Section 11.1.4.3 have been appropriately considered in satisfying the regulatory QA requirements. On the basis of this information, the staff should conclude that this evaluation is complete. The reviewers should write material suitable for inclusion in the SER prepared for the entire application. The SER should include a summary statement of what was evaluated and the basis for the reviewers' conclusions.

The staff can document the evaluation as follows:

Management Measures

[Here the primary reviewer provides a summary statement of what was evaluated and why the reviewer finds the submittal acceptable.] Continued with: *Based on its review of the license application, the NRC staff concludes that:*

1. *the applicant has adequately described its QA and*

2. *the applicant's QA meets the regulatory requirements of 10 CFR Part 70 and provides reasonable assurance of protection of public health and safety and of the environment.*

11.1.7 REFERENCES

1. Nuclear Regulatory Commission (U.S.), Washington, D.C. "Domestic Licensing of Special Nuclear Material (10 CFR Part 70)," *Federal Register:* Vol. 64, No. 146. pp. 41338--41357. July 30, 1999.

2. American Society of Mechanical Engineers (ASME), "Quality Assurance Requirements for Nuclear Facility Applications," (An American National Standard). ASME NQA-1-1994, New York. 1994

MANAGEMENT MEASURES
11.2 CONFIGURATION MANAGEMENT

11.2.1 PURPOSE OF REVIEW

The purpose of this review is to establish with reasonable assurance that the applicant has a plan for or has implemented an acceptable configuration management (CM) function. The review should result in a determination that the applicant has described and committed to a CM function over the life cycle of the facility that provides reasonable assurance that design information, safety information, and modifications (both temporary and permanent) that might impact the ability of items relied on for safety to perform their function when needed are maintained in a consistent and up-to-date manner. The review should also result in a determination that the applicant's CM function captures formal documentation governing the design and continued maintenance of those facility structures, systems, and components (SSCs) and supporting management measures, as identified and described in the ISA. The review should assure that the CM function is adequately coordinated and integrated with the other management measures such as maintenance, quality assurance, training and qualifications, procedures, and audits and assessments.

11.2.2 RESPONSIBILITY FOR REVIEW

Primary: Licensing Project Manager

Secondary: Primary ISA Reviewer, Quality Assurance Reviewer, Records Management Reviewer

Supporting: TWRS Site Representative

11.2.3 AREAS OF REVIEW

The applicant's descriptions and commitments for CM should be reviewed with an emphasis on the processes for documenting an established baseline configuration and controlling changes to it to preclude inadvertent degradation of safety. An examination should be conducted of the descriptions of the organizational structure responsible for CM activities and the process, procedures, and documentation required by the applicant for modifying the site, items relied on for safety, and the supporting management measures. The review should focus on the applicant's management level controls that ensure (a) the disciplined documentation of engineering, installation, and operation of modifications; (b) the training and qualification of affected staff, (c) revision and distribution of operating, test, calibration, surveillance, and maintenance procedures and drawings, (d) post-modification testing, and (e) readiness review.

The following topics should be reviewed:

Management Measures

1. CM Policy

   The review should cover the applicant's description of overall CM functions, including at least the following topics: (a) the scope of the SSCs to be included in the CM function (b) objectives of each CM activity, (c) a description of each CM activity, and (d) the organizational structure and staffing interfaces.

   The review should examine the applicant's establishment of a baseline CM policy applicable to all operations, initially independent of ISA results. The review should also examine any reduced level of CM that the applicant may propose for certain SSCs based on the ISA results.

2. Design Requirements

   The review should cover the applicant's demonstration that design requirements and associated design bases have been established and are maintained by an appropriate organizational unit. The applicant's CM controls on the design requirements and the ISA should be evaluated.

3. Document Control

   The review should include the applicant's methods used to establish and control documents within the CM function.

4. Change Control

   The review should examine the applicant's commitments to ensure that the CM function maintains strict consistency among the design requirements, the physical configuration, and the facility documentation. An important component of this review is the applicant's process, within the CM function, for ensuring that the ISA will be systematically reviewed and modified to reflect design or operational changes from an established safety basis, and that all documents outside the ISA that are affected by safety basis changes will be properly modified, authoritatively approved, and made available to personnel.

5. Assessments

   The review should examine the applicant's commitments to conduct initial and periodic assessments of the CM system to determine the function's effectiveness and to correct deficiencies, consistent with the acceptance criteria in SRP Section 11.7, "Audits and Assessments."

11.2.4 ACCEPTANCE CRITERIA

11.2.4.1 Regulatory Requirements

The requirement for CM is addressed in the following:

Nuclear Regulatory Commission (U.S.), Washington, D.C. "Domestic Licensing of Special Nuclear Material (10 CFR Part 70)." *Federal Register:* Vol. 64, No.146. pp.41338--41357. July 30, 1999.

Specific references are as follows:

1. In § 70.4, "Definitions," the term CM is defined.

2. In § 70.4, "Definitions," the term management measures is defined. CM is included as a management measure.

3. In § 70.62(d), the applicant or licensee is required to establish management measures to provide continuing assurance of compliance with the performance requirements.

4. In § 70.64(a)(1), the design of new facilities or the design of new processes at existing facilities is required to be developed and implemented in accordance with management measures.

5. In § 70.65(a), the application is required to include a description of the management measures.

11.2.4.2 Regulatory Guidance

There are no regulatory guides that apply to CM for a new facility licensed under 10 CFR Part 70.

11.2.4.3 Regulatory Acceptance Criteria

The reviewers should determine that an applicant's CM function is acceptable if it satisfies the following criteria.

1. CM Policy

   The applicant's description of overall CM functions describes at least the following topics: (a) the scope of the items relied on for safety (SSCs and management measures) to be included in the CM function (coordinate with the ISA Chapter reviewer for the application), (b) a description of each CM function activity, (c) the objectives of each CM function activity, and (d) the organizational structure and staffing interfaces. The scope of SSCs include all those items relied on for safety as defined by the ISA; furthermore, those items are included in the QA, maintenance, and training and qualifications programs. The functional interfaces with quality assurance (QA), maintenance, and training and qualification are of particular importance and should be addressed individually.

Management Measures

2. Design Requirements

The applicant demonstrates that design requirements and associated design bases have been established and are maintained by an appropriate organizational unit. The applicant demonstrates that the CM system provides for keeping design requirements and the ISA current and that suitable hazard/accident analysis methods, including controlled computer codes, if applicable, are available to evaluate safety margins of proposed changes. Technical management review and approval procedures are described. The specific items relied on for safety included in the CM function are identified within the ISA summary report.

3. Document Control

The applicant describes an acceptable method to establish and control documents within the CM function, including cataloging the document data base, the information content of the document data base, maintenance and distribution of documents, document retention policies, and document retrieval policies. A list of the types of documents controlled is established and includes key documents, such as drawings, procurement specifications, engineering analyses, operating procedures, training/qualification records, and maintenance procedures.

4. Change Control

The applicant demonstrates that the CM function will maintain strict consistency among the design requirements, the physical configuration, and the facility documentation. The applicant describes an acceptable process for identifying and authorizing proposed changes, performing appropriate technical and safety reviews of proposed changes, approving changes, implementing changes, and documenting changes. The applicant describes an acceptable process, within the CM function, for ensuring that the ISA is systematically reviewed and modified to reflect design or operational changes from an established safety basis, and that all documents outside the ISA that are affected by safety basis changes are properly modified, authoritatively approved, and made available to personnel.

5. Assessments

The applicant confirms that assessments, including initial and periodic examinations of the CM system, will be conducted to determine the program's effectiveness and to correct deficiencies. The applicant indicates that such assessments will be systematically planned and conducted in accordance with an overall facility audit and assessment function as described by the applicant and reviewed by NRC in accordance with Section 11.7 of this SRP.

11.2.5 REVIEW PROCEDURES

11.2.5.1 Acceptance Review

The primary reviewer should evaluate the application to determine whether it addresses the "Areas of Review" discussed in Section 11.2.3, above. If significant deficiencies are identified, the applicant should be requested to submit additional material before the start of the safety evaluation.

The reviewer should also determine that the applicant has committed to a formal CM function for establishing design bases and reviewing proposed changes to items, procedures, and processes that may impact SSCs relied on for safety.

11.2.5.2 Safety Evaluation

After determining that the application is acceptable for review in accordance with Section 11.2.1, above, the primary reviewer should perform a safety evaluation against the acceptance criteria described in Section 11.2.4. Review procedures for each criterion are discussed in the following:

1. CM Policy Management

   The primary reviewer should review the CM plan that provides management commitments, and policy directive, and defines key responsibilities, terminology, and equipment scope. The method for initiating immediate corrective actions should be examined. The secondary reviewers should examine the ISA for the identification of dependence on CM of items relied on for safety. Appropriate interfaces both within the CM function and with external organizations and functions should be examined. In particular, the quality assurance specialist should assist in examining the functional interfaces with QA, maintenance, and training (including qualification). The reviewers look for the applicant's identification of required databases and the rules for their maintenance. The reviewers examine implementing procedures for the CM function.

2. Design Requirements

   The primary reviewer should confirm that the design process leading to drawings and other statements of requirements proceeds logically from the design basis. The reviewers should verify that specific personnel are assigned the responsibility for maintaining the design bases and requirements. These may be the same personnel that maintain the ISA and controlled computer codes. The reviewers should verify that the items relied on for safety to be listed under CM are clearly defined in the requirements documents, along with the assignment of any grades or quality levels. The grades or quality levels, if specified, are based on the qualitative risk associated with postulated accident sequences in which the items relied on for safety are required to function. This part of the review should be coordinated with the ISA primary reviewer. The ISA specifies all items relied on for safety, and the applicant should have indicated in the ISA what level of CM attributes are applied

to a particular item. However, in the ISA this indication may only consist of an index or category designation. The definition of the multiple CM levels, if used, should be in the CM Chapter of the application. The primary reviewer for the CM Chapter is responsible to determine if the reduced levels the applicant would apply to safety items for lesser risk accident sequences are adequate.

3. Document Control

The primary reviewer should evaluate the applicant's material showing that the CM system will capture documents that are relevant and important to safety. This includes design requirements, the ISA, as-built drawings, specifications, all safety-important operating procedures, procedures involving training, QA, maintenance, audits and assessments, emergency operating procedures, emergency response plans, system modification documents, assessment reports, and others, as necessary, that the applicant may deem part of the CM function. The primary reviewer should determine whether a controlled document database is used to control documents and track document change status. Rules of storage for originals or master copies of documents within the CM function should follow the guidance of "Records Management" discussed in SRP Section 11.9.

4. Change Control

The primary reviewer should ensure that the description of change control within the CM function commits to acceptable methods in place for: (a) the identification of changes in configurations relied on for safety; (b) technical and management review of changes, and (c) tracking and implementing changes, including placement of documentation in a document control center and dissemination to affected functions such as training, engineering, operations, maintenance, and QA. Post-modification testing of hardware (or procedure drills or walk-throughs) may be done in conjunction with periodic equipment performance monitoring and normal maintenance functions.

5. Assessments

The primary reviewer should ensure that both document assessments and physical assessments (system walkdowns) will be conducted periodically to check the adequacy of the CM function. The primary reviewer should ensure that all assessments and follow-ups are documented. These reports can provide a supporting basis for future changes. The primary reviewer should assure that assessments will include reviews of safety systems from design requirements through implementation.

11.2.6 EVALUATION FINDINGS

The staff's evaluation should verify that the license application provides sufficient information to satisfy the regulatory requirements of Section 11.2.4.1 and that the regulatory acceptance criteria in Section 11.2.4.3 have been appropriately considered in satisfying the requirements. On the basis of this information, the staff should conclude that this evaluation is complete.

The reviewers should write material suitable for inclusion in the SER prepared for the entire application. The SER should include a summary statement of what was evaluated and the basis for the reviewers' conclusions.

The staff can document the evaluation as follows:

The staff has reviewed the Configuration Management (CM) function for (name of facility) according to Section 11.2 of the Standard Review Plan. [Insert a summary statement of what was evaluated and why the reviewer finds the submittal acceptable.]

The applicant has suitably and acceptably described its commitment to a proposed CM system, including the method for managing changes in procedures, facilities, activities, and equipment for systems important to safety. Management level policies and procedures, including an analysis and independent safety review of any proposed activity involving systems important to safety, are described that will ensure that the relationship between design requirements, physical configuration, and facility documentation is maintained as part of a new activity or change in an existing activity involving licensed material. The administrative control will include (or does include) the following elements of CM.

1. *Configuration Management*

The organizational structure, procedures, and responsibilities necessary to implement CM are in place or committed to.

2. *Design Requirements*

The design requirements and bases are documented and supported by analyses and the documentation is maintained current.

3. *Document Control*

Documents, including drawings, are appropriately stored and accessible. Drawings and related documents adequately describe systems important to safety.

4. *Change Control*

Responsibilities and procedures adequately describe how the applicant will achieve and maintain strict consistency among the design requirements, the physical configuration, and the facility documentation. Methods are in place for suitable analysis, review, approval, and implementation of identified changes to systems important to safety. This includes appropriate CM controls to assure configuration verification, functional tests, and accurate documentation for equipment or procedures that have been modified.

Management Measures

5. *Assessments*

Methods or plans are in place to perform initial and periodic examination of the effectiveness of the CM system itself. In the case of existing facilities, assessments and follow-up reports of corrective actions are documented.

In situations where the applicant proposes a graded CM function based on risk significance the following can be added: the applicant has described its approach to applying at least two levels of CM attributes to items relied on for safety, and has identified which safety items involve lower risk and may receive the reduced level of CM requirements. The applicant's proposed reduced CM features are found adequate to contribute to the reliability and availability of the lesser risk items relied on for safety identified in the application.

11.2.7 REFERENCES

1. Nuclear Regulatory Commission (U.S.), Washington, D.C. "Domestic Licensing of Special Nuclear Material (10 CFR Part 70)." *Federal Register:* Vol. 64, No.146. pp. 41338--41357. July 30, 1999.

2. Department of Energy (U.S.) (DOE). DOE-STD-1073-93, "DOE Standard: Guide for Operational Configuration Management Function." Parts I and II, DOE: 1993.

MANAGEMENT MEASURES
11.3 MAINTENANCE

11.3.1 PURPOSE OF REVIEW

The purpose of this review is to establish reasonable assurance, that the facility will have an adequate maintenance program for items relied on for safety–with the exception of personnel activities–to ensure their availability and reliability to perform their intended safety functions when needed. The availability and reliability requirements of the items should be commensurate with risk levels identified in the ISA.

11.3.2 RESPONSIBILITY FOR REVIEW

Primary: Licensing Project Manager

Secondary: Quality assurance, Criticality, chemical, fire, radiation protection and environmental reviewers

Supporting: Site Representative/Facility Inspector

11.3.3 AREAS OF REVIEW

The applicant's description of its maintenance program should be reviewed with emphasis on demonstrating that items relied on for safety with the exception of personnel activities (safety controls) are inspected, calibrated, tested and maintained so as to ensure their ability to perform their safety functions when needed. The safety controls should be identified by the ISA (discussed in Chapter 3.0 of this SRP). Individual components and support systems for the safety controls may have to be individually maintained to ensure the availability and reliability of the control function. The reviewers should review the applicant's description of how each of the following functions is implemented within the site organization.

1. Surveillance/monitoring
2. Corrective maintenance
3. Preventive maintenance
4. Functional testing

11.3.4 ACCEPTANCE CRITERIA

11.3.4.1 Regulatory Requirements

Management Measures

The requirement for maintenance is addressed in the following:

Nuclear Regulatory Commission (U.S.), Washington, D.C. "Domestic Licensing of Special Nuclear Material (10 CFR Part 70)." *Federal Register:* Vol. 64, No. 146. pp. 41338--41357. July 30, 1999.

Specific references are as follows:

1. In § 70.4, "Definitions," the term management measures is defined. Maintenance is included as a management measure.

2. In § 70.62(d), the applicant or licensee is required to establish management measures to provide continuing assurance of compliance with the performance requirements.

3. In § 70.64(a)(1), the design of new facilities or the design of new processes at existing facilities is required to be developed and implemented in accordance with management measures.

4. In § 70.64(a)(8), inspection, testing, and maintenance are required to be addressed as one of the Baseline Design Criteria to provide reasonable assurance that items relied on for safety will be designed to allow them to be adequately inspected, tested and maintained to ensure their availability and reliability to perform their function when needed.

5. In § 70.65(a), the application is required to include a description of the management measures.

11.3.4.2 Regulatory Guidance

There are no regulatory guides that apply to maintenance for a new facility licensed under 10 CFR Part 70.

11.3.4.3 Regulatory Acceptance Criteria

The applicant's submittal should be considered acceptable in the area of maintenance if it adequately addresses the following:

1. Safety Controls Identified in the ISA – An assessment of whether components and support systems need to be individually maintained to ensure the availability and reliability of specific safety controls. The reliability and availability of a particular item should be commensurate with the risk levels identified in the ISA.

2. Essential Components

    a. Surveillance/monitoring – the surveillance/monitoring function, its responsible organization, and the conduct of surveillance at a specified frequency to measure the degree to which safety functions of safety controls meet performance specifications.

This activity is used in setting preventive maintenance frequencies for safety controls and the determination of performance trends for safety controls. How results from incident investigations (described in Section 11.8 of this SRP) and identified root causes are used to modify the affected maintenance function and eliminate or minimize the root cause from recurring should be addressed. For surveillance tests that can only be done while equipment is out of service, proper compensatory measures should be prescribed for the continued normal operation of a process.

b. Corrective maintenance – the documented approach used to perform corrective actions or repairs on safety controls. The maintenance function should provide a planned, systematic, integrated, and controlled approach for the repair and replacement activities associated with identified failures of safety controls.

c. Preventive maintenance – a description of the preventive maintenance function that demonstrates a commitment to conduct preplanned and scheduled periodic refurbishing or partial or complete overhaul for the purpose of ensuring that unplanned outages of selected safety controls do not occur. This activity includes using the results of the surveillance/monitoring component of maintenance. Instrumentation calibration and testing should be addressed as part of this component.

d. Functional testing – a description of the functional testing function that demonstrates a commitment to the functional testing of safety controls after corrective or preventive maintenance or calibration. Functional testing should be conducted using approved procedures that include compensatory measures while the test is being conducted.

3. Work Control Methods – A list of maintenance-related work control methods.

4. Relationship of the Maintenance Elements to Other Management Control Sections Discussed in SRP Chapter 11.0 – A discussion of how the maintenance function utilizes, interfaces with, or is linked to these elements.

11.3.5 REVIEW PROCEDURES

11.3.5.1 Acceptance Review

The primary reviewer should evaluate the application to determine whether it addresses the "Areas of Review" discussed in Section 11.3.3, above. If significant deficiencies are identified, the applicant should be requested to submit additional material before the start of the safety evaluation.

11.3.5.2 Safety Evaluation

After determining that the application is acceptable for review in accordance with Section 11.3.5.1, above, the primary reviewer should perform a safety evaluation against the

acceptance criteria described in Section 11.3.4. The staff review should be based on an assessment of the material presented. The review should determine if the applicant has adequately planned the work to be accomplished and whether necessary policies, procedures, and instructions either are in place or will be in place before work starts. The review should result in a determination that there is reasonable assurance that the applicant's quality assurance, configuration management and maintenance programs, as described in SRP Sections 11.1 through 11.3 are coordinated.

When an applicant's maintenance program references other sections of the application, the primary reviewer should review these other sections of the application to ensure consistency with the applicant's selection of acceptance criteria and the proposed method for implementation.

Secondary staff reviewers should review the maintenance program to ensure there is no contradiction between it and their primary review areas of the application. They should also ensure that the scope of the applicant's maintenance program includes the items relied on for safety that are in their primary review areas of the application. The supporting staff reviewer should become familiar with the applicant's maintenance program and determine whether ongoing activities are in agreement with it.

The final step in the review is the primary staff reviewer's writing of a Safety Evaluation Report (SER) input that should summarize the conduct of the review, identifies what material in the application forms the basis for a finding of reasonable assurance with respect to the acceptance criteria, and presents the bases for license conditions that may be necessary to conclude that reasonable assurance is achieved.

11.3.6 EVALUATION FINDINGS

The staff's evaluation should verify that the license application provides sufficient information to satisfy the regulatory requirements of Section 11.3.4.1 and that the regulatory acceptance criteria in Section 11.3.4.3 have been appropriately considered in satisfying the requirements. On the basis of this information, the staff should conclude that this evaluation is complete. The reviewers should write material suitable for inclusion in the SER prepared for the entire application. The SER should include a summary statement of what was evaluated and the basis for the reviewers' conclusions.

The staff can document the evaluation as follows:

> *The applicant has committed to maintenance of items relied on for safety with the exception of personnel activities (safety controls).* [Insert a summary statement of what was evaluated and why the reviewer finds the submittal acceptable.] *The applicant's maintenance commitments contain the basic elements to ensure availability and reliability: surveillance/monitoring, corrective maintenance, preventive maintenance, and functional testing. The applicant's maintenance function is proactive, using surveillance/monitoring*

and maintenance records to analyze equipment performance and identify the root causes of repetitive failures.

In addition, the surveillance/monitoring activities described in this section of the application provide assurance of the validity of the ISA by examination and calibration and testing of equipment that monitors process safety parameters and acts to prevent or mitigate accident consequences.

The maintenance function: (1) is based on approved procedures; (2) employs work control methods that properly consider personnel safety, awareness of facility operating groups, quality assurance, and the rules of configuration management; (3) links items relied on for safety requiring maintenance to the ISA; (4) justifies the preventive maintenance intervals in the terms of equipment reliability goals; (5) provides for training that emphasizes importance of ISA identified controls, regulations, codes, and personal safety; and (6) creates documentation that includes detailed records of all surveillance, inspections, equipment failures, repairs, and replacements.

The staff concludes that the applicant's maintenance function meets the requirements of 10 CFR Part 70 and provides reasonable assurance that the environment and the health and safety of the public are protected.

11.3.7 REFERENCES

1. Nuclear Regulatory Commission (U.S.), Washington, D.C. "Domestic Licensing of Special Nuclear Material (10 CFR Part 70)." *Federal Register:* Vol. 64, No. 146. pp. 41338--41357. July 30, 1999.

2. Code of Federal Regulations, *Title 10, Energy*, Subpart 50.65, "Requirements for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants."

3. Code of Federal Regulations, *Title 29, Labor*, Subpart 1910.119, "Process Safety Management of Highly Hazardous Chemicals."

4. Code of Federal Regulations, *Title 40, Protection of Environment*, Part 68, "Risk Management Program for Chemical Accidental Release Prevention."

5. Nuclear Regulatory Commission (U.S.), Washington, D.C. "Guidance on Management Controls/Quality Assurance, Requirements for Operation, Chemical Safety, and Fire Protection for Fuel Cycle Facilities." *Federal Register:* Vol. 54, No. 53. pp. 11590-11598. March 21, 1989.

6. Nuclear Regulatory Commission, (U.S.) (NRC). Regulatory Guide 1.160, Rev. 2, "Monitoring the Effectiveness of Maintenance at Nuclear Power Plants." NRC: Washington, D.C. March 1997.

7. Nuclear Regulatory Commission, (U.S.) (NRC). Inspection Procedure 88025, "Maintenance and Surveillance Testing." NRC: Washington, D.C. May 23, 1984.

8. Nuclear Regulatory Commission, (U.S.) (NRC). Inspection Procedure 88062, "Maintenance and Inspection." NRC: Washington, D.C. January 1996.

MANAGEMENT MEASURES
11.4 TRAINING AND QUALIFICATION OF PLANT PERSONNEL

11.4.1 PURPOSE OF REVIEW

The purpose of this review is to establish that there is reasonable assurance that personnel who perform activities relied on for safety at the plant[1] will understand, recognize the importance of, and be qualified to perform these activities as required by 10 CFR Part 70, as revised[2], in a manner that adequately protects the public and worker health and safety and the environment.

11.4.2 RESPONSIBILITY FOR REVIEW

Primary: Training, Quality Assurance or Human Factors Engineer/Specialist

Secondary: Licensing Project Manager

Supporting: Site Representative/Facility Inspector

11.4.3 AREAS OF REVIEW

The regulation, 10 CFR Part 70, as revised, requires that personnel who perform activities relied on for safety be trained, tested, and retested as necessary to ensure that they understand, recognize the importance of, and are qualified to perform these activities in a manner that adequately protects the public and worker health and safety and the environment. Personnel at the plant should have the knowledge and skills necessary to start-up, operate, maintain, modify, and decommission the facility in a safe manner. Therefore, the training, testing, retesting, and qualification of these personnel as described in the license application should be reviewed. This should include the training, testing, retesting, and qualification of managers, supervisors, designers, technical staff, plant operators, technicians, maintenance personnel and other personnel whose level of knowledge is relied on for safety.

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[1]This SRP section provides guidance for the review of information on the training and qualification of plant personnel who perform activities relied on for safety. Section 2 of SRP Appendix C on quality assurance provides review guidance on the subject of training and qualification of other personnel (for example, construction personnel) who perform activities relied on for safety.

[2]Nuclear Regulatory Commission (U.S.), Washington, D.C. "Domestic Licensing of Special Nuclear Material (10 CFR Part 70)." *Federal Register* : Vol. 64, No. 146. pp. 41338-41357. July 30, 1999.

The following areas should be reviewed:

1. Organization and management of training,
2. Trainee selection,
3. Conduct of needs/job analysis and identification of tasks for training,
4. Development of learning objectives as the basis for training,
5. Organization of instruction using lesson plans and other training guides,
6. Evaluation of trainee mastery of learning objectives,
7. Conduct of on-the-job training,
8. Systematic evaluation of training effectiveness,
9. Personnel qualification, and
10. Applicant's provisions for continuing assurance.

11.4.4 ACCEPTANCE CRITERIA

The regulatory requirements, regulatory guidance, and regulatory acceptance criteria applicable to training and qualification of plant personnel are listed in the following sections.

11.4.4.1 Regulatory Requirements

The requirement for training and qualification is addressed in the following:

Nuclear Regulatory Commission (U.S.), Washington, D.C. "Domestic Licensing of Special Nuclear Material (10 CFR Part 70)." *Federal Register*: Vol. 64, No. 146. pp. 41338-41357. July 30, 1999.

Specific references are as follows:

1. In § 70.4, "Definitions," the term management measures is defined. Training and qualification are included as a management measure.

2. In § 70.62(d), the applicant or licensee is required to establish management measures to provide continuing assurance of compliance with the performance requirements

3. In § 70.64(a)(1), the design of new facilities or the design of new processes at existing facilities is required to be developed and implemented in accordance with management measures.

4. In § 70.65(a), the application is required to include a description of the management measures.

An additional requirement for training and qualification is addressed in the following:

Code of Federal Regulations, Title 10, Energy, Part 19, "Notices, Instructions and Reports to Workers: Inspection and Investigations." The specific reference to § 19.12, "Instructions to Workers."

11.4.4.2 Regulatory Guidance

NRC guidance applicable to training and qualification of personnel at nuclear power plants is given in the following:

Nuclear Regulatory Commission (U.S.), Washington, D.C. "Training Review Criteria and Procedures," NUREG-1220, Rev.1, January 1993.

As appropriate, this guidance should be used for training and qualification of plant personnel at other nuclear facilities.

11.4.4.3 Regulatory Acceptance Criteria

The NRC reviewers should find the applicant's submittal regarding training and qualification of plant personnel provides reasonable assurance that the regulatory acceptance criteria below are adequately addressed and satisfied.

In addition to the regulatory acceptance criteria given below, SRP Sections 4.1.5.4 and 4.1.5.6 provide criteria for training and qualification of plant personnel for radiation safety functions.

1. Organization and Management of Training

   The organization and management of training of plant personnel should be acceptable if the start-up, operation, maintenance, modification, and decommissioning of the facility are organized, staffed, and managed to facilitate planning, directing, evaluating, and controlling a systematic training process that fulfills job-related training needs. Formal training should be provided for each position or activity for which the required performance is relied on for safety. The application should state what training will be conducted and which personnel will be provided this training. Training should include retraining of previously trained and qualified personnel based on specified criteria.

   The following commitments should be in the application regarding organization and management of training:

   a. Line management should be responsible for the content and effective conduct of the training.

   b. The job function, responsibility, authority, and accountability of personnel involved in managing, supervising, and implementing training should be clearly defined.

   c. Performance-based training should be used as the primary management tool for analyzing, designing, developing, conducting, and evaluating training.

   d. Procedures should be documented and implemented to ensure that all phases of training are conducted reliably and consistently.

Management Measures

e. Training documents should be linked to the configuration management system to ensure that design changes and plant modifications are accounted for in the training.

f. Exceptions from training may be granted to trainees and incumbents when justified, documented, and approved by management.

g. Auditable training records should be maintained. Training records, both programmatic and individual, should support management information needs and provide required data on each individual's training, job performance, and fitness for intended duty. (Refer to SRP Section 11.9 for detailed guidance on records management.)

2. Trainee Selection

Selection of trainees who will perform activities relied on for safety should be acceptable if minimum requirements for selection of trainees are specified. Trainees should meet entry-level criteria defined for the position including minimum educational, technical, experience, and (if necessary) physical fitness requirements.

3. Conduct of Needs/Job Analysis and Identification of Tasks for Training

The conduct of needs/job analysis and identification of tasks for training should be acceptable if the tasks required for competent and safe job performance are identified, documented, and included in the training.

Operations personnel, training staff, and other subject matter experts, as appropriate, should conduct a needs/job analysis to develop a valid task list for specific jobs. The jobs treated in this manner should include, as a minimum, those responsible for managing, supervising, performing, and verifying the activities specified in the Integrated Safety Analysis (ISA - see SRP Chapter 3) as preventing or mitigating accident sequences. Each task selected for training (initial or continuing) from the facility-specific task list should be matrixed to supporting procedures and training materials. The facility-specific list of tasks selected for training and the comparison to training materials should be reviewed on an established schedule and updated as necessitated by changes in procedures, facility systems/equipment, or job scope.

4. Development of Learning Objectives as the Basis for Training

The development of learning objectives as the basis for training should be acceptable if learning objectives that identify training content and define satisfactory trainee performance are derived from job performance requirements. Learning objectives should state the knowledge, skills, and abilities the trainee should demonstrate, the conditions under which required actions will take place, and the standards of performance the trainee should achieve upon completion of the training activity. Learning objectives should be sequenced based on their relationship to each other.

5. Organization of Instruction Using Lesson Plans and Other Training Guides

   The organization of instruction using lesson plans and other training guides should be acceptable if the plans/guides are based on the required learning objectives derived from specific job performance requirements. Plans/guides should be used for in-class training and on-the-job training and should include standards for evaluating proper trainee performance. Review and approval requirements should be established for all plans/guides and other training materials before their issue and use.

6. Evaluation of Trainee Mastery of Learning Objectives

   The evaluation of trainee mastery of learning objectives should be acceptable if trainees are evaluated periodically during training to determine their progress toward mastery of job performance requirements and at the completion of training to determine their mastery of job performance requirements.

7. Conduct of On-the-Job Training

   The conduct of on-the-job training should be acceptable if on-the-job training used for activities required by the ISA are fully described. On-the-job training should be conducted using well-organized and current performance-based training materials. On-the-job training should be conducted by designated personnel who are competent in the program standards and methods of conducting the training. Completion of on-the-job training should be by actual task performance. When the actual task cannot be performed and is therefore "walked-down," the conditions of task performance, references, tools, and equipment should reflect the actual task to the extent possible.

8. Systematic Evaluation of Training Effectiveness

   A systematic evaluation of training effectiveness and its relation to on-the-job performance should be acceptable if it ensures that the training program conveys all required skills and knowledge and is used to revise the training, where necessary, based on the performance of trained personnel in the job setting. A comprehensive evaluation of individual training programs should be conducted periodically by qualified individuals to identify program strengths and weaknesses. Feedback from trainee performance during training and from former trainees and their supervisors should be used to evaluate and refine the training. Change actions (for example, procedure changes, equipment changes, facility modifications) should be monitored and evaluated for their impact on the development or modification of initial and continuing training and should be incorporated in a timely manner. This should be accomplished through the configuration management system (see SRP Section 11.2). Improvements and changes to initial and continuing training should be systematically initiated, evaluated, tracked, and incorporated to correct training deficiencies and performance problems.

Management Measures

9. Personnel Qualification

   The following commitments should be in the application regarding personnel qualification for managers, supervisors, designers, technical staff, plant operators, technicians, maintenance personnel and other plant staff required to meet NRC regulations:

   a. Managers should have a minimum of a B.S./B.A. or equivalent. Each manager should have either management experience or technical experience in facilities similar to the facility.

   b. Supervisors should have at least the qualifications required of personnel being supervised with either one additional year experience supervising the technical area at a similar facility or should have completed the supervisor training.

   c. Technical staff identified in the ISA whose actions or judgments are relied on for safety to satisfy the performance requirements identified in 10 CFR Part 70, as revised, should have a B.S. in the appropriate technical field and three years experience.

   d. Plant operators, technicians, maintenance personnel, and other staff whose actions are required to comply with NRC regulations should have completed the applicant's training process or have equivalent experience or training.

   e. Candidates for process operators should be required to meet minimum qualifications described in the application. Candidates for job functions other than process operators should also be required to meet minimum qualifications, but these minimum qualifications need not be described in the application.

10. Applicant's Provisions for Continuing Assurance

    The applicant's provisions for continuing assurance of training and qualification of plant personnel should be acceptable if the submittal addresses periodic retesting of personnel as necessary to ensure that they continue to understand, recognize the importance of, and are qualified to perform their activities that are relied on for safety.

11.4.5 REVIEW PROCEDURES

11.4.5.1 Acceptance Review

The primary reviewer should evaluate the application to determine whether it addresses the "Areas of Review" discussed in Section 11.4.3, above. If significant deficiencies are identified, the applicant should be requested to submit additional material before the start of the safety evaluation.

11.4.5.2 Safety Evaluation

After determining that the application is acceptable for review in accordance with Section 11.4.5.1, above, the primary reviewer should perform a safety evaluation against the acceptance criteria described in Section 11.4.4, recognizing that the rigor and formality of a systematic approach to training and the required qualification of plant personnel may be graded to correspond to the hazard potential of the facility and to the complexity of the training needed. The review should determine whether the applicant has adequately planned for the training and qualification of plant personnel to be accomplished and whether necessary policies, procedures, and instructions will be in place and appropriate training and qualification will be accomplished before these personnel begin activities relied on for safety. Some of the information may be referenced to other sections of the application, or incorporated by reference, provided that these references are clear and specific.

The secondary reviewer should confirm that the applicant's commitments regarding the training and qualification of plant personnel are consistent with other sections of the submittal. The secondary reviewer should also integrate the training and qualification of plant personnel input into the Safety Evaluation Report (SER).

The supporting reviewer should become familiar with the applicant's commitments for the training and qualification of plant personnel and determine whether ongoing activities (at an existing facility) are in agreement with them.

On the basis of its review, the staff may request that the applicant provide additional information or modify the application to meet the acceptance criteria. The staff or applicant may also propose license conditions to ensure that the training and qualification of plant personnel meet the acceptance criteria. The review should result in a determination that there is reasonable assurance that the applicant's training and qualification of plant personnel will ensure that only properly trained and qualified personnel will perform activities relied on for safety.

When the safety evaluation is complete, the primary staff reviewer, with assistance from the other reviewers, should prepare the training and qualification of plant personnel input for the SER as described in Section 11.4.6 using the acceptance criteria from Section 11.4.4.

11.4.6 EVALUATION FINDINGS

The staff's evaluation should verify that the license application provides sufficient information to satisfy the regulatory requirements of Part 70, as revised (as listed in SRP Section 11.4.4.1), and that the regulatory acceptance criteria in Section 11.4.4.3 have been appropriately considered in satisfying the requirements. On the basis of this information, the staff should conclude that this evaluation is complete. The reviewers should write material suitable for inclusion in the SER prepared for the entire application. The SER should include a summary statement of what was evaluated and the basis for the reviewers' conclusions.

Management Measures

The staff can document the evaluation as follows:

> [Here the primary reviewer provides a summary statement of what was evaluated and why the reviewer finds the submittal acceptable.] Continued with: *Based on its review of the license application, the NRC staff concludes that:*

1. *the applicant has adequately described its training and qualification of plant personnel, and*

2. *the applicant's training and qualification of plant personnel meet the requirements of 10 CFR Part 70 and provide reasonable assurance of protection of public health and safety and of the environment.*

11.4.7 REFERENCES

1. Nuclear Regulatory Commission (U.S.), Washington, D.C. "Domestic Licensing of Special Nuclear Material, (10 CFR Part 70)." *Federal Register:* Vol. 64, No. 146. pp. 41338-41357. July 30, 1999.

2. Nuclear Regulatory Commission (U.S.), Washington, D.C. "Training Review Criteria and Procedures," NUREG-1220, Rev. 1, January 1993.

MANAGEMENT MEASURES
11.5 PLANT PROCEDURES

11.5.1 PURPOSE OF REVIEW

The purpose of this review is to establish that there is reasonable assurance that the applicant is capable and committed to providing management control of facility operations through the development, review, control, and implementation of written procedures, that will protect the workers, the public and the environment during testing, startup, and full-scale operation of the facility.

11.5.2 RESPONSIBILITY FOR REVIEW

Primary: License Project Manager

Secondary: Primary staff reviewers in all operating areas

Supporting: TWRS Site Representative

11.5.3 AREAS OF REVIEW

The review should address the process the applicant has developed for the production, use and management control of written procedures. This should include the basic elements of identification, development, verification, review and comment resolution, approval, validation, issuance, change control, and periodic review. This should include two general types of procedures:

1. Procedures used to directly control process operations, commonly called "operating procedures". These are procedures for workstation operators and should include directions for normal operations as well as off-normal events caused by human error or failure of equipment. Procedures of this type include required actions to ensure nuclear criticality safety, chemical safety, fire protection, emergency planning, and environmental protection; and,

2. Procedures used to effect activities that support the process operations, that are commonly referred to as "management control procedures". These are procedures used to manage the conduct of activities such as configuration management, radiation safety, maintenance, human-systems interface, quality assurance, design control, test control, startup, training and qualification, audits and assessments, incident investigations, record-keeping and reporting.

11.5.4 ACCEPTANCE CRITERIA

11.5.4.1 Regulatory Requirements

The requirement for Management Measures/Procedures is addressed in the following:

Nuclear Regulatory Commission (U.S.), Washington, D.C. "Domestic Licensing of Special Nuclear Material (10 CFR Part 70). " *Federal Register:* Vol. 64, No. 145. pp. 41338-41357. July 30, 1999.

Specific references are as follows:

1. In § 70.4, "Definitions," the term *management measures* is defined. Procedures are included as a management measure.

2. In § 70.62(d), the applicant or licensee is required to establish management measures to provide continuing assurance of compliance with the performance requirements.

3. In § 70.64(a)(1), the design of new facilities or the design of new processes at existing facilities is required to be developed and implemented in accordance with management measures.

4. In § 70.65(a), the application is required to include a description of the management measures.

5. In § 70.22(a)(8), the application is required to include procedures that protect health and minimize danger to life.

11.5.4.2 Regulatory Guidance

None.

11.5.4.3 Regulatory Acceptance Criteria

The reviewers should determine that the applicant's process for developing and implementing procedures is adequate if the process satisfies the following:

1. Procedures should be written or planned for the conduct of all operations involving controls identified in the ISA as activities relied on for safety and for all management control systems supporting those controls.

2. Operating procedures contain the following elements:

    a. purpose of the activity;
    b. regulations, polices, and guidelines governing the procedure;
    c. type of procedure;

d. steps for each operating process phase;
e. initial startup;
f. normal operations;
g. temporary operations;
h. emergency shutdown;
i. emergency operations;
j. normal shutdown;
k. startup following an emergency or extended downtime;
l. hazards and safety considerations;
m. operating limits;
n. precautions necessary to prevent exposure of hazardous chemicals or licensed special nuclear material;
o. measures to be taken if contact or exposure occurs;
p. safety controls associated with the process and their functions;
q. time frame for which the procedure is valid.

3. Management control procedures contain elements reflecting the important elements of the functions described in the applicable chapters of this SRP. Procedures should exist to manage the following activities:

a. configuration management;
b. radiation safety;
c. maintenance;
d. human-systems interface;
e. quality assurance;
f. training and qualification;
g. audits and assessments;
h. incident investigations;
i. records management;
j. criticality safety;
k. fire safety;
l. chemical process safety;
m. design control;
n. test control;
o. startup;
p. reporting requirements.

4. The applicant's method for identifying the procedures includes using ISA results to identify needed procedures. Process operating procedures should provide specific direction regarding administrative controls to ensure process operational safety.

5. The application should describe the method for identifying, developing, approving, implementing, and controlling procedures. This method should include, as a minimum, that:

a. operating limits and controls are specified in the procedure;
b. procedures include required actions for off-normal conditions of operation as well as normal operations;
c. if needed, safety checkpoints are identified at appropriate steps in the procedure;
d. procedures are validated through field tests;
e. procedures are approved by management personnel responsible and accountable for the operation;
f. a mechanism is specified for revising and reissuing procedures in a controlled manner;
g. the quality assurance and configuration management programs at the plant ensure that current procedures are available and used at all work locations; and
h. the plant training program ensures that the required persons are trained in the use of the latest procedures available.

6. The application should include the following statement regarding procedure adherence: "Activities involving special licensed nuclear material will be conducted in accordance with approved procedures".

7. The application should describe the types of procedures used by the facility. These should typically include management control, operating, maintenance, and emergency procedures. The application should provide information regarding the procedure categories used at the facility. An acceptable identification discussion should clearly state areas for which a procedure is required. The application should provide a listing of the types of activities that are covered by written procedures. This should include the topics of administrative procedures; system procedures that address startup, operation, and shutdown; abnormal operation/alarm response; maintenance activities that address system repair, calibration, inspection and testing; and emergency procedures. Appendix D to this SRP provides an acceptable listing of the items to be included under each topic.

8. The application should indicate that following unusual incidents, such as an accident, unexpected transient, significant operator error, or equipment malfunction, or following any modification to a system, a review of written procedures will take place, as needed.

9. The application should indicate how technical accuracy of procedures will be ensured as written. The discussion should identify who is responsible for verification.

10. The application should indicate how documents will be distributed in accordance with current distribution lists. A process limiting the use of outdated procedures should be addressed.

11. The application should describe how formal requirements governing temporary changes will be developed and implemented.

12. The application should have formal requirements for Design Control for items that are important to safety, and should identify who is responsible for design inputs, processes, outputs, changes, interfaces, and records.

13. A description of the Test Control program should be provided, and should indicate that an effective test program has been established for tests, including commissioning and preoperational tests. Acceptable test control program procedures should provide criteria for determining when a test is required or how and when testing activities are performed.

    a. Tests should be performed under conditions that simulate the most adverse design conditions, as determined by analysis.

    b. Test results should be documented, evaluated, and their acceptability determined by a responsible individual or group.

14. Maintenance procedures involving safety controls should commit to the topics listed below for corrective, preventive, functional testing after maintenance, and surveillance maintenance activities:

    a. Pre-maintenance activity involving reviews of the work to be performed, including procedure reviews for accuracy and completeness.

    b. Steps that require notification of all affected parties (operators and supervisors) prior to performing work and upon completion of maintenance work.

    c. Control of work by comprehensive procedures to be followed by maintenance technicians.

15. The application should contain a commitment to conduct periodic reviews of procedures to ensure their continued accuracy and usefulness and establishes the time frame for reviews of the various types of procedures. At a minimum all procedures should be reviewed every 5 years and emergency procedures should be reviewed every year.

16. The application should describe the use and control of procedures.

17. A pre-operational testing (startup) program should be described. Information pertaining to how, and to what extent, the plant operating, emergency, and surveillance procedures will be user-tested during the test program should be provided.

    Section 5, "Instructions, Procedures, and Drawings," of SRP Appendix C, provides criteria for other procedures.

11.5.5 REVIEW PROCEDURES

11.5.5.1 Acceptance Review

The primary reviewer should evaluate the application to determine whether it addresses the "Areas of Review" discussed in Section 11.5.3, above. If significant deficiencies are identified,

the applicant should be requested to submit additional material before the start of the safety evaluation.

11.5.5.2 Safety Evaluation

After determining that the application is acceptable for review in accordance with Section 11.5.5.1, above, the primary reviewer should perform a safety evaluation against the acceptance criteria described in Section 11.5.4. The safety evaluation forms the basis for staff findings, and supports the reviewers' conclusions that the applicant has committed to:

1. Controls that are identified in the ISA for safety procedures (i.e., procedures that constitute administrative controls for safety).
2. The independent verification and validation of procedures before use.
3. The review and approval by an independent multi-disciplinary safety review team and control by the configuration management function of any change to operating, management control, or maintenance procedure
4. Following approved procedures while processing licensed special nuclear material.
5. Having procedures for the notification of operations personnel before and after maintenance is performed on safety controls.

Secondary staff reviewers should review the management measures/procedures section of the application to ensure that there is no contradiction between it and their primary review areas. They should also ensure that the scope of the applicant's procedures program includes the operating and management control procedures listed in paragraph 11.5.4.3., above.

The supporting staff reviewer should become familiar with the applicant's management control and operating procedures, and determine whether ongoing activities are in agreement with them.

11.5.6 EVALUATION FINDINGS

The staff's evaluation should verify that the license application provides sufficient information to satisfy the regulatory requirements of Section 11.5.4.1 and that the regulatory acceptance criteria in Section 11.5.4.3 have been appropriately considered in satisfying the requirements. On the basis of this information, the staff should conclude that this evaluation is complete. Each reviewers should write material suitable for inclusion in the SER prepared for the entire application. The primary reviewer should have primary responsibility for specific input to the SER. The SER should include a summary statement of what was evaluated and the basis for the reviewers' conclusions.

The staff can document the evaluation as follows:

The application has described suitably detailed processes for the development, approval, and implementation of procedures. [Insert a summary statement of what was evaluated and why the reviewer finds the submittal acceptable.] *Special attention has been paid to items relied on for safety, as well as to systems important to the health of plant workers and the public and to the protection of the environment.*

11.5.7 REFERENCE

U.S. Nuclear Regulatory Commission, (U.S.), Washington, D.C. "Guidance on Management Controls/Quality Assurance, Requirements for Operation, Chemical Safety, and Fire Protection for Fuel Cycle Facilities." *Federal Register:* Vol. 54, No. 53. pp. 11590–11598. March 21, 1989.

MANAGEMENT MEASURES

11.6 HUMAN FACTORS ENGINEERING/PERSONNEL ACTIVITIES

11.6.1 PURPOSE OF REVIEW

The purpose of the review is to establish, with reasonable assurance, that the applicant has applied human factors engineering (HFE) to personnel activities identified as items relied on for safety so they will be available and reliable to perform their intended functions when needed. In addition, the review should verify that HFE practices and guidelines are incorporated into human-system interface (HSI) designs and supporting elements to ensure that the HSIs support safe and reliable personnel activities. This provides assurance that the possibility of human error in the facility operations was addressed during the design of the facility by facilitating correct, and inhibiting wrong decisions by operators and by providing means for detecting, correcting, or compensating for error.

11.6.2 RESPONSIBILITY FOR REVIEW

Primary: Human Factors Specialist

Secondary: Lead reviewer of ISA

Supporting: Site Representative or Fuel Cycle Facility Inspector

11.6.3 AREAS OF REVIEW

The review should address personnel activities contained in the ISA for the protection of the workers, the public, and the environment. The application of HFE on the personnel activities should include HSI design and supporting elements such as staffing, training, and procedures.

This HFE/personnel activities review process can be divided into the following areas of review:

1. HSI Design Review Planning,
2. Identification of Personnel Activities,
3. Operating Experience Review,
4. Function and Task Analysis,
5. HSI Design, Inventory and Characterization,
6. Staffing,
7. Procedure Development,
8. Training Program Development, and
9. Human Factors Verification and Validation.

Judgement regarding the areas of review for a given submittal should be based on evaluation of the information provided with respect to (1) provisions made to address personnel activities consistent with the findings of the ISA, (2) the similarity of the associated HFE issues to those

for similar type plants, and (3) the determination of whether items of special or unique safety significance are involved.

11.6.4 ACCEPTANCE CRITERIA

11.6.4.1 Regulatory Requirements

The requirement for personnel activities is provided in the following:

Nuclear Regulatory Commission (U.S.), Washington, D.C. "Domestic Licensing of Special Nuclear Material (10 CFR Part 70)." *Federal Register*: Vol. 64, No. 146. pp. 41338–41357. July 30, 1999.

Specific citations are as follows:

1. In § 70.4, "Definitions," the term *items relied on for safety* is defined. Personnel activities are included as an item relied on for safety.

2. In § 70.61(e), the states that each item relied on for safety will be available and reliable to perform its intended function when needed.

3. In § 70.62(c)(vi), items relied on for safety are identified through the performance of an integrated safety analysis as part of the safety program.

4. In § 70.65(b)(6), the application is required to a list of items relied on for safety in sufficient detail to understand their functions in relation to the performance requirements.

11.6.4.2 Regulatory Guidance

There are no regulatory guides that apply to HFE/personnel activities for a new facility licensed under 10 CFR Part 70.

11.6.4.3 Regulatory Acceptance Criteria

The applicant's treatment of personnel activities identified as items relied on for safety should be acceptable if the applicant applied HFE practices and criteria to the personnel activities and supporting HSIs that provide reasonable assurance that the personnel activities will take place and satisfy their safety functions when needed. The specific areas of review should include the following:

1. HSI Design Review Planning - Acceptance should be based on confirmation that the applicant has adequately considered the role of HFE and the means by which it is applied during design, construction and operation of the facility to improve reliability personnel activities identified in the ISA. The applicant should address -- commensurate with the results of the ISA--the following functional areas:

a. General HFE Functional Goals and Scope
b. HFE Team and Organization/Individual and Responsibilities
c. HFE Process and Procedures
d. HFE Issues Tracking
e. HFE Functional Description

2. Identification of Personnel Activities - Acceptance should be based on the ability of the applicant to identify the personnel activities as items relied on for safety from the ISA summary. The activities should be described to the extent that the reviewer can understand what the human is to do, which HSIs are involved, and the importance of the action. The personnel activities should include:

   a. Accident sequences in which human errors are causes.

   b. Operator actions that are credited as safeguards.

   c. HSIs intended to support those personnel activities required to prevent, detect, and correct conditions that could be root-causes or contributing factors to accidents.

   d. HSIs intended to support those personnel activities required to mitigate the consequences of accidents.

3. Operating Experience Review (OER) - Acceptance should be based on the verification that the applicant has identified and analyzed for relevance HFE-related problems and issues encountered in previous designs that are similar to the proposed design under review.

4. Functional Allocation Analysis and Task Analysis - Acceptance should be based on verification that (1) the allocation of functions between personnel and plant system elements takes advantage of human strengths and avoids demands that are not compatible with human capabilities, and (2) the task requirements on plant personnel have reasonable performance demands for accomplishing the allocated functions.

5. HSI Design - The HSI design process and the detailed HSI design that is a product of that process should be acceptable by verification that the applicant has appropriately translated function and task requirements to the detailed designs of HSI components (such as alarms, displays, controls, and operator aids) through the systematic application of HFE principles and criteria. In addition, the applicant has appropriately considered environmental conditions that could have an effect on personnel involved in the activity and factored those considerations into the HSI design.

6. Staffing - Staffing should be acceptable if the applicant has reviewed the requirements for the number and qualifications of personnel in a systematic manner that includes a thorough understanding of task requirements and applicable regulatory requirements for the range of applicable plant conditions and personnel activities.

Management Measures

7. Procedure Development - The description of procedure development for personnel activities identified as relied on for safety should satisfy the acceptance criteria in SRP Section 11.5. The procedures should be based on the task analyses and should integrate the personnel activities and the associated HSIs needed to accomplish those activities.

8. Training Program Development - The description of the process for the development of personnel training should satisfy the acceptance criteria in SRP Section 11.4. Training requirements should be based on the task analyses and should focus on the relationship between the personnel activities and the associated HSIs needed to accomplish those activities.

9. Verification and Validation - A description of the verification and validation (V&V) process should be acceptable if confirms that the design conforms to HFE design principles that enables plant personnel to successfully perform personnel activities to achieve plant safety. The scope of V&V should address those personnel activities discussed in item 2 above and HSI design requirements listed in item 5 above. An acceptable V&V process should consist of a combination of the five activities listed below:

    a. HSI task support verification - an evaluation to ensure that HSI components are provided to address personnel activities identified in the ISA. The HSI task support verification should be acceptable by verification that the aspects of the HSI (e.g., alarms, controls, displays, procedures, and data processing) that are required to accomplish personnel activities are available through the HSI. It should also be verified that the HSI minimizes the inclusion of information, displays, controls, and decorative features that inhibit personnel activities.

    b. HFE design verification - an evaluation to determine whether the design of each HSI component reflects HFE principles, standards, and guidelines. The method and the results of the HFE design verification should be acceptable if the HSI has been designed to be appropriate for personnel activities and operational considerations as defined by the HSI design process consistent with accepted HFE guidelines, standards, and principles. Mockup(s), model(s), or other tools can be used by the applicant to perform the HFE design verification.

    c. Integrated system validation - a performance-based evaluation of the integrated design to ensure that the HFE/HSI supports safe operation of the plant. Integrated system validation should be performed after HFE problems identified in earlier review activities have been resolved or corrected because these may negatively affect performance and, therefore, validation results. All critical or risk-significant personnel activities as defined in the task analysis and the ISA should be tested and found to be adequately supported in the design, including the performance of such actions outside the control room.

    d. Human factors issue resolution verification - an evaluation to ensure that the HFE issues identified during the design process have been acceptably addressed and resolved. Issue resolution verification should be acceptable if all issues documented

in the HFE issue tracking system are satisfactorily addressed. Issues that cannot be resolved until the HSI design is constructed, installed, and tested should be specifically identified and incorporated into the final plant HFE/HSI design verification.

e. Final plant HFE/HSI design verification - assurance that the implementation of the final design of the HSI and supporting systems (for example, procedures and training programs) conform to the V&Ved design that resulted from the HFE design process. Final plant HFE/HSI design verification should be performed if the V&V activities, described above, did not fully evaluate the actual installation of the final HSI design in the plant. Final verification should be acceptable if in-plant implementation of the HFE design conforms to the design description that resulted from the HFE design process and V&V activities.

V&V activities should be performed in the order listed above, as necessary. However, iteration of some steps may be necessary to address design corrections and modifications that occur during V&V.

11.6.5 REVIEW PROCEDURES

11.6.5.1 Acceptance Review

The primary reviewer should evaluate the application to determine whether it addresses the "Areas of Review" discussed in Section 11.6.3, above. If significant deficiencies are identified, the applicant should be requested to submit additional material before the start of the safety evaluation.

11.6.5.2 Safety Evaluation

After determining that the application is acceptable for review in accordance with Section 11.6.5.1, above, the primary reviewer should perform a safety evaluation against the acceptance criteria described in Section 11.6.4. If during the course of the safety evaluation, the primary reviewer determines the need for additional information, the primary reviewer should coordinate a request for additional information with the licensing project manager. The staff should use a tiered approach for evaluating the HFE design. The upper tier is at the program description level, with high-level plant mission goals that are divided into the functions necessary to achieve the mission goals. The middle tier encompasses functions that are allocated to human and system resources and that are divided into tasks (personnel activities) for the purposes of specifying the alarms, information, and controls that are designed to accomplish function assignments. The tasks should be arranged into meaningful jobs and the HSI should be designed to best support job task performance. The lower tier is the detailed design (of the HSI, procedures, and training) and how they are incorporated into the facility design. Evaluation of the HFE design should be broad-based and include aspects of normal and emergency operations, testing, maintenance, etc., consistent with findings in the ISA.

Management Measures

The submittal should be reviewed at multiple tiers to ensure personnel activities identified into the ISA are translated into the facility design.

The primary review staff should review the ISA summary to ensure personnel activities have been suitably characterized as part of items relied on for safety that are needed to prevent or mitigate consequences of concern. Information from analyses conducted to address the criteria of SRP Chapter 3 should be incorporated as an input to the HFE design process, including the development of HSI design and test requirements. This input is articulated in acceptance criterion two. The extent HFE elements are applied should be based on the number, type, and complexity of the personnel activities.

The secondary reviewer should ensure that the types of personnel activities relied on for safety are appropriate. Furthermore, the secondary reviewer should ensure there is coordination between HFE and the ISA, and that lessons learned are incorporated into the ISA.

The supporting reviewers should assist in the tiered approach of the review in that they may look at specific examples of human factors engineering in an existing facility.

11.6.6 EVALUATION FINDINGS

The primary reviewer should write an SER section that addresses each topic reviewed under this SRP Section and explains why the NRC staff has reasonable assurance that the personnel activities described in the application are acceptable. License conditions may be proposed to impose requirements where the application is deficient. The SER should include a summary statement of what was evaluated and the basis for the reviewers' conclusions.

The staff can document the evaluation as follows:

The staff has reviewed the human factors activities for the TWRS facility according to Standard Review Plan Section 11.6. [Insert a summary statement of what was evaluated and why the reviewer finds the submittal acceptable.]

The applicant has identified the personnel activities identified in the ISA and demonstrated how human factors engineering (HFE) principles, including function and task analysis, were incorporated into those human-safety interface (HSI) designs to ensure reliability of the activities. The applicant has conducted an operating experience review of applicable facilities and incorporated lessons learned into the design process. In addition, the applicant has verified the adequacy of the HFE principles and HSI through use of validation and verification and has incorporated these principles into identified support functions of training, procedures, and staffing.

Meeting the above requirements provides an acceptable basis for finding that there is reasonable assurance that personnel activities in the context of items relied on for safety will be available and reliable to perform their intended functions when needed.

11.6.7 REFERENCES

1. Nuclear Regulatory Commission (U.S.), Washington, D.C. "Domestic Licensing of Special Nuclear Material (10 CFR Part 70)." *Federal Register*: Vol. 64, No. 146. pp. 41338–41357. July 30, 1999.

2. Nuclear Regulatory Commission (U.S.) (NRC). NUREG-0700, Rev.1, Vol.1, "Human-System Interface Design Review Guideline." NRC: Washington, D.C. June 1996.

3. Nuclear Regulatory Commission (U.S.) (NRC). NUREG-0711, "Human Factors Engineering Program Review Model." NRC: Washington, D.C. July 1994.

4. Department of Defense (DOD). MIL-STD-1472D, "Human Engineering Design Criteria for Military Systems, Equipment and Facilities." DOD: Washington, D.C. March 1989.

5. The Institute of Electrical and Electronics Engineers (IEEE). IEEE Std 1023-1988, "IEEE Guide for the Application of Human Factors Engineering to Systems, Equipment, and Facilities of Nuclear Power Generating Stations." IEEE: New York, NY. May 1989.

MANAGEMENT MEASURES
11.7 AUDITS AND ASSESSMENTS

11.7.1 PURPOSE OF REVIEW

The purpose of this review is to establish that the applicant has developed and adequately described a system of audits and assessments[1] that provides reasonable assurance that an adequate level of protection will be maintained at the facility and to ensure that items will be available and reliable to perform their safety function when needed, as required by 10 CFR Part 70, as revised.[2]

11.7.2 RESPONSIBILITY FOR REVIEW

Primary: Quality Assurance (QA) Engineer/Specialist

Secondary: Licensing Project Manager

Supporting: Site Representative/Facility Inspector

11.7.3 AREAS OF REVIEW

The applicant's system of audits and assessments should consist of two distinct levels of activities:

a. an independent internal and external audit activity to evaluate the scope, status, adequacy, programmatic compliance and implementation effectiveness of QA and other management measures that ensure continued availability and reliability of items relied on for safety.

b. an internal assessment activity to evaluate the scope, status, adequacy, programmatic compliance, and implementation effectiveness of QA and other management measures that ensure continued availability and reliability of items relied on for safety.

The following areas should be reviewed:

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[1]Audits and assessments are evaluations of the scope, status, adequacy, programmatic compliance, and implementation effectiveness of QA and other management measures. Audits are conducted or led by "independent" personnel from the QA organization. Assessments are conducted by or for management above or outside the QA organization.

[2] Nuclear Regulatory Commission (U.S.), Washington, D.C. " Domestic Licensing of Special Nuclear Material (10 CFR Part 70)." *Federal Register*: Vol. 64, No. 146. pp. 41338–41357. July 30, 1999.

Management Measures

1. Audits and assessments - general
2. Audits
3. Assessments
4. Applicant's provisions for continuing assurance

11.7.4 ACCEPTANCE CRITERIA

The regulatory requirements, regulatory guidance, and regulatory acceptance criteria applicable to audits and self assessments are listed in the following sections.

11.7.4.1 Regulatory Requirements

The requirement for audits and assessments is addressed in the following:

Nuclear Regulatory Commission (U.S.), Washington, D.C. " Domestic Licensing of Special Nuclear Material (10 CFR Part 70)." *Federal Register*: Vol. 64, No. 146. pp. 41338–41357. July 30, 1999.

Specific references are as follows:

1. In § 70.4, "Definitions," the term management measures is defined. Audits and assessments are included as a management measure.

2. In § 70.62(d), the applicant or licensee is required to establish management measures to provide continuing assurance of compliance with the performance requirements.

3. In § 70.64(a)(1), the design of new facilities or the design of new processes at existing facilities is required to be developed and implemented in accordance with management measures.

4. In § 70.65(a), each application is required to include a description of the management measures.

11.7.4.2 Regulatory Guidance

There is no regulatory guidance applicable to this area of the SRP.

11.7.4.3 Regulatory Acceptance Criteria

The NRC reviewers should find that the application regarding audits and assessments provides reasonable assurance that the regulatory acceptance criteria below are adequately addressed and satisfied.

1. Audits and Assessments - General: The description of audits and assessments should be acceptable if:

a. The application indicates that internal audits, external audits, and assessments are to be conducted with a graded approach based on the results of the integrated safety analysis (ISA - see SRP Chapter 3). The stated objective of the audits and assessments should be to objectively evaluate the effectiveness and proper implementation of QA and other management measures for items relied on for safety and to address the technical adequacy of items being audited/assessed.

b. The application describes, provides a commitment to, and provides justification for a frequency and scope of audits and assessments of items relied on for safety. A commitment to perform audits and assessments in all areas where the requirements for QA and other management measures are applicable should be provided. The application should indicate that audits and assessments will be regularly scheduled on the basis of the status and the safety significance of the items being audited/assessed and will be initiated early enough to ensure the implementation of effective QA and other management measures.

c. The application describes policy directives that are established for audits and assessments. The application indicates that the policy directives cover schedules, guidance for conducting the audits/assessments, assigned responsibilities, and procedures for recording the audit/assessment results and ensuring that identified deficiencies are corrected in a timely and effective manner for each activity audited/assessed..

d. The application identifies the position title, qualifications, and responsibilities of the manager responsible for the overall success of the audits and assessments. Other organizational responsibilities for audits and assessments should be identified in the application.

e. The application describes the training and qualification requirements for audit and assessment personnel. (SRP Section 11.4 addresses training and qualification requirements in detail.)

f. The application describes the authority each audit and assessment team has to investigate any aspect of the audited/assessed items with access to all relevant information.

g. The application describes how performance indicators are established so that audit and assessment personnel can determine the degree to which selected items relied on for safety are meeting performance requirements.

h. The application indicates that audits and assessments are conducted according to written procedures/checklists.

i. The application indicates that audits and assessments include detailed walk-downs of plant areas, including out-of-the-way and limited-access areas, with provisions for accurate, documented descriptions of any deficiencies.

Management Measures

j. The application describes provisions for on-the-spot corrective actions with appropriate documentation.

k. The application indicates that audit and assessment results are reviewed with management having responsibility in the area audited/assessed.

l. The application indicates that audit and assessment findings and recommendations are documented and distributed to appropriate management for review and response. As described in SRP Section 11.1, a corrective action program is administered to ensure timely and effective corrective action.

m. The application indicates that audit and assessment deficiency data are analyzed and trended and resultant reports, which indicate quality trends and the effectiveness of management measures, are given to appropriate management for review, response, corrective action, and follow-up.

2. Audits: The description of audits should be acceptable if, in addition to addressing the acceptance criteria in Section 11.7.4.3.1 above,

a. The application indicates that audit personnel have no direct responsibility for the items they audit.

b. The application indicates that audits are led by appropriately qualified and certified audit personnel from the QA organization.

c. The application indicates that audit team membership may include personnel (not necessarily from the QA organization) having technical expertise in the areas being audited.

d. The application indicates that both technical and programmatic audits are performed internally (that is, within the applicant's organization) and externally (that is within the organization of suppliers, contractors, and subcontractors) and that these audits provide a comprehensive independent verification and evaluation of procedures and activities affecting the quality of items relied on for safety.

e. The application indicates that auditing organizations schedule and conduct appropriate follow-up to ensure timely and effective corrective action.

f. The application indicates that audit reports are issued to appropriate management on a timely basis.

g. The application indicates that reports on the status of corrective actions for audit-findings are issued periodically to appropriate management.

h. The application indicates that internal audits address compliance with selected operating limits during facility operation.

3. Assessments: The description of assessments should be acceptable if, in addition to addressing the acceptance criteria in Section 11.7.4.3.1 above, the application indicates that responsible management personnel (or that qualified, but not necessarily certified, personnel with no direct responsibility for the items being assessed who are designated by the responsible management) perform the assessments.

4. Applicant's Provisions for Continuing Assurance: The applicant's provisions for continuing audits and assessments should be acceptable if the application indicates that changes to the program of audits and assessments due to reorganizations, revised activities, lessons learned, changes to applicable regulations, and other changes are reviewed and reflected in the program description.

11.7.5 REVIEW PROCEDURES

11.7.5.1 Acceptance Review

The primary reviewer should evaluate the application to determine whether it addresses the "Areas of Review" discussed in Section 11.7.3, above. If significant deficiencies are identified, the applicant should be requested to submit additional material before the start of the safety evaluation.

11.7.5.2 Safety Evaluation

After determining that the application is acceptable for review in accordance with Section 11.7.5.1, above, the primary reviewer should perform a safety evaluation against the acceptance criteria described in Section 11.7.4. The review should determine whether the applicant has adequately planned for audits and assessments to be accomplished and whether necessary policies, personnel, procedures, and instructions will be in place to begin audits and assessments early, that is, during the ISA and the design of items relied on for safety.

Some of the information may be referenced to other sections of the application, or incorporated by reference, provided that these references are clear and specific.

The secondary reviewer should confirm that the applicant's audit and assessment commitments are consistent with other sections of the submittal. The secondary reviewer is also responsible for integrating the audit and assessment input into the Safety Evaluation Report (SER).

The supporting reviewer should become familiar with the applicant's audit and assessment commitments and determine whether the ongoing audits and assessments of the applicant and the applicant's suppliers, contractors, and subcontractors are in agreement with them.

On the basis of its review, the staff may request that the applicant provide additional information or modify the application to meet the acceptance criteria in SRP Section 11.7.4.

Management Measures

The staff or applicant may also propose license conditions to ensure audits and assessments meet the acceptance criteria. The review should result in a determination that there is reasonable assurance that the audits and assessments will provide additional assurance that items relied on for safety will perform satisfactorily in service and that activities relied on for safety will be performed satisfactorily.

When the safety evaluation is complete, the primary staff reviewer, with assistance from the other reviewers, should prepare the audits and assessments input for the SER as described in Section 11.7.6 using the acceptance criteria from Section 11.7.4.

11.7.6 EVALUATION FINDINGS

The staff's evaluation should verify that the license application provides sufficient information to satisfy the regulatory requirements of Part 70, as revised (as listed in SRP Section 11.7.4.1), and that the regulatory acceptance criteria in Section 11.7.4.3 have been appropriately considered in satisfying the requirements. On the basis of this information, the staff should conclude that this evaluation is complete. The reviewers should write material suitable for inclusion in the SER prepared for the entire application. The SER should include a summary statement of what was evaluated and the basis for the reviewers' conclusions.

The staff can document the evaluation as follows:

> [Here the primary reviewer provides a summary statement of what was evaluated and why the reviewer finds the submittal acceptable.] Continued with: *Based on its review of the license application, the NRC staff concludes that:*
>
> 1. *the applicant has adequately described its system of audits and assessments and*
>
> 2. *the applicant's system of audits and assessments meet the requirements of 10 CFR Part 70 and provides reasonable assurance of protection of public health and safety and of the environment.*

11.7.7 REFERENCE

Nuclear Regulatory Commission (U.S.), Washington, D.C. "Domestic Licensing of Special Nuclear Material (10 CFR Part 70)." *Federal Register*: Vol. 64, No. 146. pp. 41338–41357. July 30, 1999.

MANAGEMENT MEASURES
11.8 INCIDENT INVESTIGATIONS

11.8.1 PURPOSE OF REVIEW

The purpose of this review is to establish, with reasonable assurance, that the applicant will have a system in place for the systematic investigation of incidents, assignment and acceptance of corrective actions, and follow-up to ensure completion of the actions. The review should confirm that incidents will be investigated and corrective action taken to prevent (or minimize) their recurrence or their leading to more serious consequences. Furthermore, the review should find that the results of incident investigations will be compared against the integrated safety analysis summary (ISA - see SRP Chapter 3.0) to provide assurance that there is continued compliance with the performance requirements contained in 10 CFR Part 70, as revised.[1]

11.8.2 RESPONSIBILITY FOR REVIEW

Primary: Licensing Project Manager

Secondary: Quality Assurance Engineer/Specialist and ISA Reviewers

Supporting: Site Representative/Facility Inspector

11.8.3 AREAS OF REVIEW

The review should encompass the following areas:

1. The description of the functions, qualifications, and responsibilities of the management person who would lead the investigation team and those of the other team members, the scope of the team's authority and responsibilities, and assurance of cooperation of management.

2. The team's ability to obtain all the information considered necessary and independence from responsibility for or to the functional area involved in the incident under investigation.

3. The maintenance of documentation consistent with Section 11.9, "Records Management."

4. Guidance for the team conducting the investigation on how to apply a reasonable, systematic, structured approach to determine the root cause(s) of the problem.

5. The system for comparing the results of the investigation against the ISA.

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[1]Nuclear Regulatory Commission (U.S.), Washington, D.C. "Domestic Licensing of Special Nuclear Material (10 CFR Part 70)." *Federal Register:* Vol. 64, No. 146. pp. 41338--41357. July 30, 1999.

6. The system for monitoring to ensure completion of any corrective measures specified -- including revisions to the ISA.

11.8.4 ACCEPTANCE CRITERIA

The regulatory requirements, regulatory guidance, and regulatory review criteria applicable to this SRP are listed in the following sections.

11.8.4.1 Regulatory Requirements

Nuclear Regulatory Commission (U.S.), Washington, D.C. "Domestic Licensing of Special Nuclear Material (10 CFR Part 70)." *Federal Register*: Vol. 64, No. 146. pp. 41338–41357. July 30, 1999.

Specific references are as follows:

1. In § 70.4, "Definitions," the term management measures is defined. Incident investigations are included as a management measure.

2. In § 70.62(d), the applicant or licensee is required to establish management measures to provide continuing assurance of compliance with the performance requirements.

3. In § 70.64(a)(1), the design of new facilities or the design of new processes at existing facilities is required to be developed and implemented in accordance with management measures.

4. In § 70.65(a), the application is required to include a description of the management measures.

11.8.4.2 Regulatory Guidance

There is no specific regulatory guidance for the overall conduct of incident investigation. See the References at the end of this section for guidance on specific aspects of incident management such as corrective action and root cause analysis.

11.8.4.3 Regulatory Acceptance Criteria

The NRC reviewers should find the applicant's submittal regarding incident investigations provides reasonable assurance that the regulatory review criteria below are adequately addressed and satisfied. Some of the information may be referenced to other sections of the SRP, or incorporated by reference, provided that these references are clear and specific.

1. Acceptability should be based on commitments for the prompt investigation of incidents that include the following elements:

a. The establishment of teams to investigate incidents that may occur during operation of the facility, to determine the root cause(s) of the event, and to recommend corrective actions. These teams should be independent from the line function(s) involved with the incident under investigation.

b. The monitoring and documenting of corrective actions (including effectiveness) through completion.

c. The maintenance of documentation so that "lessons learned" may be applied to future operations of the facility. Details of the incidents should be compared to incidents already considered in the ISA, and actions should be taken to ensure that the ISA includes the evaluation of the risk associated with incidents of the type actually experienced.

2. Acceptability should be based on the adequacy of the applicant's commitments to establish and use a plan for the investigation of incidents. Acceptability should also be based upon the following acceptance criteria:

a. The licensee has described the overall plan and method for investigating incidents.

b. The functions, responsibilities, and scope of authority of investigation teams are documented in the plan.

c. Qualified internal or external investigators are appointed to serve on investigation teams. Each team should include at least one process expert and one team member trained in root cause analysis.

d. There is a commitment to undertake prompt investigation of any incidents.

e. The investigation process and investigation team are independent of the line management, and participants are assured of no retribution from participating in investigations.

f. A reasonable, systematic, structured approach is used to determine the root cause(s) of incidents. The level of investigation should be based on a graded approach relative to the severity of the incident.

g. Auditable records and documentation related to incidents, investigations, and root cause analysis are maintained. For each incident, the incident report should include a description, contributing factors, root-cause analysis, findings, and recommendations. Relevant findings are reviewed with all affected personnel. These reports should be made available to the NRC on request.

h. Documented corrective actions are taken within a reasonable period to resolve findings from incident investigations.

11.8.5 REVIEW PROCEDURES

11.8.5.1 Acceptance Review

The primary reviewer should evaluate the application to determine whether it addresses the "Areas of Review" discussed in Section 11.8.3, above. If significant deficiencies are identified, the applicant should be requested to submit additional material before the start of the safety evaluation.

11.8.5.2 Safety Evaluation

After determining that the application is acceptable for review in accordance with Section 11.8.5.1, above, the primary reviewer should perform a safety evaluation against the acceptance criteria described in Section 11.8.4. If during the course of the safety evaluation the primary reviewer determines the need for additional information, the primary reviewer should coordinate a request for additional information with the licensing project manager. The review should determine if the applicant and principal contractors have adequately planned for incident investigations to be conducted with resulting corrective actions to be appropriately implemented.

The primary reviewer should review the applicant's plan and procedures for investigating incidents. The review should include the organizational structure, provisions for establishing investigating teams, methods for determining root causes, and procedures for tracking and completing corrective actions and for documenting the process for the purpose of applying the "lessons learned" to other operations as well as validating the ISA. The organizational structure and procedures should be consistent with the relevant sections of other SRP Chapter 11, "Management Measures." This plan should be separate from any required Emergency Plan.

The quality assurance secondary reviewer should review the methods used for determining root causes, the procedures for tracking and implementing the corrective actions, and the process of applying the "lessons learned" to the other operations.

The ISA reviewers should review the procedure that ensures the results of the investigation are compared against the ISA and the necessary follow-up actions occur.

The secondary and supporting reviewers should become familiar with these procedures and determine whether planned future and ongoing activities are consistent with them.

11.8.6 EVALUATION FINDINGS

The primary reviewer should write an SER section that addresses each topic reviewed under this SRP Section and that explains why the NRC staff has reasonable assurance that the incident investigation system is acceptable. License conditions may be proposed to impose

requirements where the application is deficient. The primary reviewer should also describe the applicant's organization, methodology, and support to ensure the quality and reliability of the incident investigation program. The SER should include a summary statement of what was evaluated and the basis for the reviewers' conclusions.

The staff can document the evaluation as follows:

> *Based on its review of the license application,* [Insert a summary statement of what was evaluated and why the reviewer finds the submittal acceptable,] *the NRC staff has concluded that the applicant has performed the following:*
>
> *1. The applicant has committed to and established an organization responsible for performing incident investigations of incidents that may occur during operation of the facility, determining the root cause(s) of the event, and recommending corrective actions for ensuring a safe facility and safe facility operations in accordance with the acceptance criteria of Subsection 11.8.4 of the SRP. As part of the review, the applicant has committed to review the results of the investigation against the ISA.*
>
> *2. The applicant has committed to monitoring and documenting corrective actions through completion.*
>
> *3. The applicant has committed to the maintenance of documentation so that "lessons learned" may be applied to future operations of the facility.*
>
> *Accordingly, the staff concludes that the applicant's description of the incident investigation process complies with applicable NRC regulations and is adequate.*

11.8.7 REFERENCES

1. Nuclear Regulatory Commission (U.S.), Washington, D.C. "Domestic Licensing of Special Nuclear Material (10 CFR Part 70)." *Federal Register*: Vol. 64, No. 146. pp. 41338–41357. July 30, 1999.

2. Department of Energy (U.S.) (DOE). DOE-STD-1010-92, "Guide to Good Practices for Incorporating Operating Experiences." DOE: Washington, D.C. July 1992.

3. Department of Energy (U.S.) (DOE). DOE-NE-STD-1004-92, "Root Cause Analysis Guidance Document." DOE: Washington, D.C. February 1992.

4. Nuclear Regulatory Commission (U.S.) (NRC). NUREG/CR-4616, "Root Causes of Component Failures Program: Methods and Applications." NRC: Washington, D.C. December 1986.

5. Nuclear Regulatory Commission (U.S.) (NRC). NUREG/CR-5665, "A Systematic Approach to Repetitive Failures." NRC: Washington, D.C. February 1991.

6. Nuclear Regulatory Commission (U.S.) (NRC), NRC Information Notice 96-28, "Suggested Guidance Relating to Development and Implementation of Corrective Action." NRC: Washington, D.C. May 1996.

MANAGEMENT MEASURES
11.9 RECORDS MANAGEMENT

11.9.1 PURPOSE OF REVIEW

The purpose of this review is to verify that the applicant has committed to a facility records management system that complies with NRC requirements.

11.9.2 RESPONSIBILITY FOR REVIEW

Primary: Licensing Project Manager

Secondary: None

Supporting: Primary reviewers of SRP Sections 11.1, "Quality Assurance," and 11.2, "Configuration Management"

11.9.3 AREAS OF REVIEW

Areas related to the handling and storing of records generated or needed in the design, construction, operation, and decommissioning phases of the facility, including the following, should be reviewed.

1. The process whereby records, including training, dosimetry, effluents, classified, facility structures, systems, or components having safety-significance are created selected, verified, categorized, indexed, inventoried, protected, stored, maintained, distributed, deleted, or preserved. The process(es) may be linked with or be a part of the facility configuration management (CM) and quality assurance systems.

2. The handling and control of various kinds of records, and the methods of recording media that comprise the records including contaminated and classified records.

3. The physical characteristics of the records storage facilities with respect to the preservation and protection of the records for their designated lifetimes.

11.9.4 ACCEPTANCE CRITERIA

11.9.4.1 Regulatory Requirements

The requirements for records management are addressed in the following:

1. Code of Federal Regulations, *Title 10, Energy*, Part 19, "Notices, Instructions and Reports to Workers: Inspection and Investigations."

2. Code of Federal Regulations, *Title 10, Energy*, Part 20, "Standards for Protection Against Radiation."

3. Code of Federal Regulations, *Title 10, Energy*, Part 21, "Reporting of Defects and Noncompliance."

4. Code of Federal Regulations, *Title 10, Energy*, Part 25, "Access Authorization for Licensee Personnel."

5. Nuclear Regulatory Commission (U.S.), Washington, D.C. "Domestic Licensing of Special Nuclear Material (10 CFR Part 70)." *Federal Register:* Vol. 64, No. 146. pp. 41338-41357. July 30, 1999.

11.9.4.2 Regulatory Guidance[1]

Regulatory guidance applicable to the area of records management is as follows:

U.S. Nuclear Regulatory Commission, (U.S.) (NRC). NUREG-1460, Rev. 1, "Guide to NRC Reporting and Recordkeeping Requirements." NRC: Washington, D.C. July 1994.

11.9.4.3 Regulatory Acceptance Criteria

The reviewer should find the applicant's records management system acceptable if it satisfies the following criteria:

1. Records are specified, prepared, verified, characterized, and maintained.

2. Records are legible, identifiable, and retrievable for their designated lifetimes.

3. Records are protected against tampering, theft, loss, unauthorized access, damage, or deterioration for the time they are in storage.

4. Procedures are established and documented specifying the requirements and responsibilities for record selection, verification, protection, transmittal, distribution, retention, maintenance, and disposition.

5. The organization and procedures are in place to promptly detect and correct any deficiencies in the records management system or its implementation.

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[1] Additional guidance for records is given in SRP Appendix C on quality assurance (Section 17) and in ASME NQA-1-1994 (Basic Requirement 17 and Supplement 17S-1) as referenced in SRP Section 11.1, "Quality Assurance."

Examples of the types of records that could be included in the system, and which contribute to providing reasonable assurance of protection of public health and safety and of the environment, are listed in Appendix E to this SRP. Records should be categorized by relative safety importance to identify record protection and storage needs and to designate the retention period for individual kinds of records. The procedures should assign responsibilities for records management, specify the authority needed for records retention or disposal, specify which records must have controlled access and provide the controls needed, provide for the protection of records from loss, damage, tampering, or theft during an emergency, and specify procedures for ensuring that the records management system remains effective.

For records consisting of computer codes/computerized data relied on for safety, the application should establish and describe procedure(s) for maintaining readability and usability of older codes/data as computing technology changes.

11.9.5 REVIEW PROCEDURES

11.9.5.1 Acceptance Review

The primary reviewer should evaluate the application to determine whether it addresses the "Areas of Review" discussed in Section 11.9.3, above. If significant deficiencies are identified, the applicant should be requested to submit additional material before the start of the safety evaluation.

11.9.5.2 Safety Evaluation

After determining that the application is acceptable for review in accordance with Section 11.9.5.1, above, the primary reviewer should perform a safety evaluation against the acceptance criteria described in Section 11.9.4. If, during the course of the safety evaluation, the primary reviewer determines the need for additional information, the primary reviewer should coordinate a request for additional information with the licensing project manager. The primary reviewer should coordinate this review with the primary reviewers of SRP Sections 11.1, "Quality Assurance, and 11.2, "Configuration Management."

11.9.6 EVALUATION FINDINGS

The primary reviewer should write an SER section that addresses each topic reviewed under this SRP Section and explains why the NRC staff has reasonable assurance that the applicant's commitment to a facility records management system is acceptable. License conditions may be proposed to impose requirements where the application is deficient. The SER should include a summary statement of what was evaluated and the basis for the reviewers' conclusions.

The staff can document the evaluation as follows:

Management Measures

The staff has reviewed the applicant's records management system [Insert a summary statement of what was evaluated] *and has concluded that there is reasonable assurance that the system will (1) be effective in collecting, verifying, protecting, and storing information about the health and safety aspects of the facility and its operations and will be able to retrieve the information in readable form for the designated lifetimes of the records; (2) provide the record storage facilities with the capability to protect and preserve records that are stored there during the mandated periods, including protection of the stored records against loss, theft, or tampering or damage during and after emergencies; and (3) ensure that any deficiencies in the records management system or its implementation will be detected and corrected in a timely manner.*

11.9.7 REFERENCES

1. Nuclear Regulatory Commission (U.S.), Washington, D.C. "Domestic Licensing of Special Nuclear Material, (10 CFR Part 70)." *Federal Register:* Vol. 64, No. 146. pp. 41338-31357. July 30, 1999.

2. U.S. Nuclear Regulatory Commission, (U.S.) (NRC). NUREG-1460, Rev. 1, "Guide to NRC Reporting and Recordkeeping Requirements." NRC: Washington, D.C. July 1994.

PLANT SYSTEMS

12.1 PURPOSE OF REVIEW

The purpose of this review is to establish that there is reasonable assurance that the plant systems will perform their intended safety functions. Examples of plant systems are as follows: (a) a ventilation system necessary to provide certain decontamination factors for normal, offnormal, and accident conditions, (b) a cooling system necessary to provide a heat sink to prevent certain process elements from exceeding temperature limits, or (c) an electrical distribution system necessary to support various systems and components relied on for safety. This section should be used by the secondary and supporting reviewers to ensure that the plant systems as described support and are consistent with their cognitive review areas.

Part 70, as revised[1], contains Baseline Design Criteria (BDCs), §70.64, that provide general design considerations. This SRP section addresses equipment and facilities specifically as plant systems that are either identified as items relied on for safety or that are required to support the items relied on for safety that are identified in the hazard and accident analyses of the integrated safety analysis (ISA) with due consideration given to the BDCs.

12.2 RESPONSIBILITY FOR REVIEW

Primary: Discipline specific engineers

Secondary: Chemical Process Engineer, Health Physicist, Fire Protection Specialist

Supporting: Primary Reviewers of SRP Section 1.1, and Chapters 2.0, 3.0, 4.0 and 8.0. Primary Reviewers of Applicable Sections of SRP Chapter 11.0.

12.3 AREAS OF REVIEW

The review should address plant systems that have safety functions or support safety functions identified in the ISA (e.g., the ISA may state that a particular vessel has a safety limit or heat load that is maintained by the cooling water system; however, in Chapter 12, the physical cooling system is described in sufficient detail to make a determination of its availability when called upon to service the heat load). The review of plant systems should focus on the mechanical, material, electrical, I&C, and structural aspects, as necessary, of the specific systems which are integrated throughout the process flow path and should confirm the plant systems' ability to satisfy its specified performance requirements.

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[1]Nuclear Regulatory Commission (U.S.), Washington, D.C. "Domestic Licensing of Special Nuclear Material (10 CFR Part 70)." *Federal Register*: Vol. 64, No. 146. pp. 41338–41357. July 30, 1999.

12.3.1 Specific Items to be Reviewed for Each System

The specific items to be addressed are as follows:

1. Safety Function -- This provides the link between the ISA to the performance requirements of the plant system.

2. System description and safety analysis -- This is sufficient to provide reasonable assurance that the given system can satisfy its performance requirements

The applicant is encouraged to take advantage of pertinent existing safety analyses and design information (i.e., requirements and their bases) that are immediately available or can be retrieved through reasonable efforts. To facilitate review of this design information, the applicant could provide a roadmap and a brief summary for each such reference that explains its relevance to this chapter.

12.3.2 Typical Plant Systems

A listing of typical plant systems is as follows:

1. Utilities
   a. Electrical
   b. Cooling water
   c. Steam/Condensate
   d. Pneumatic
   e. Lighting systems
2. Ventilation
3. Instrumentation and Controls
4. Load handling systems
5. Chemical control systems
6. Waste management system
   a. Gaseous
   b. Solid
   c. Liquid
   d. Process sampling systems
7. Containment/confinement systems

12.4 ACCEPTANCE CRITERIA

12.4.1 Regulatory Requirements

The requirements for plant systems are addressed in the following:

Code of Federal Regulations, *Title 10, Energy*, Part 70, "Domestic Licensing of Special Nuclear Material (10 CFR Part 70)."

Specific references are as follows:

1. In § 70.22(a)(7), the applicant is required to provide a description of equipment and facilities used by the applicant to protect health and minimize danger to life and property.

2. In § 70.23(a)(3), the Commission is required to make the determination that proposed equipment and facilities are adequate to protect health and minimize danger to life and property.

Nuclear Regulatory Commission (U.S.), Washington, D.C. "Domestic Licensing of Special Nuclear Material (10 CFR Part 70)." *Federal Register*: Vol. 64, No. 146. pp. 41338–41357. July 30, 1999.

Specifically:

> In §70.64, "Requirements for new facilities or new processes at existing facilities," the applicant or licensee is required to address the baseline design criteria in the design of new facilities.

12.4.2 Regulatory Guidance

None, at present, specific to 10 CFR 70 licensees.

12.4.3 Regulatory Acceptance Criteria

The NRC reviewers should find the applicant's submittal regarding plant systems provides reasonable assurance that the regulatory review criteria below are adequately addressed and satisfied. Some of the information may be referenced to other sections of the SRP, or incorporated by reference, provided an adequate summary is provided and a single reference essentially contains all of the information. Accordingly, this review should have considerable interface with other review sections. In Section 12.4.3.1, general acceptance criteria are provided. In Section 12.4.3.2, specific acceptance criteria are provided for certain systems to provide the type of information that the NRC staff reviewer would be looking for as needed to support their evaluations of the respective plant systems ability to meet the performance criteria.

12.4.3.1 General Plant System Acceptance Criteria

1. Safety Function:

   The safety functions of the plant system should be considered acceptable if the submitted information supports the following observations:

   a. The system safety function is consistent with the ISA summary (evaluated for its acceptability in SRP Chapter 3.0, "Integrated Safety Analysis") and is inclusive of systems that support items relied on for safety or are identified as items, themselves,

inclusive of both preventive and mitigative safety features.

b. Functional requirements are clearly stated to allow for the system to be adequately designed to satisfy its intended safety function. Any performance requirements, including environmental operating conditions and system availability, are also identified.

c. Process elements that the plant system supports are clearly stated. Furthermore, if the system is part of a larger process/safety system, the relation of the system to that overall system is described (e.g., I&C systems tend to be part of larger safety systems).

2. System Description:

The plant system description of the plant system should be considered acceptable if the applicant has provided the following:

a. The purpose of the system for both safety and non safety features. Non safety features should be addressed in the context of not preventing the plant system from performing its safety function (see below, Safety Analysis, 3.b).

b. The basic system description and theory, including basic theory of operation, and as applicable, credible environmental conditions, and safe upper and lower limits for such items as temperatures, pressures, flows, and material compositions.

c. As applicable, hazardous material information. This could include reference to the hazardous material information of the process system the plant system supports.

d. Structures and components including pertinent aspects that directly relate to the safety function (e.g., diesel generator load capacity, time to load (if critical)) as opposed to general industrial equipment specifications that fall out from these capabilities (e.g., starting torque, motor insulation, number and type of windings). Such lower tier details should be implicitly included only by reference to the overall specifications.

e. Assurance measures including codes and standards used for mechanical, civil, chemical, electrical, and instrumentation and control systems.

f. Drawings and procedures comprised of process and instrumentation drawings (P&IDs), or a simplified system drawing with reference to P&IDs and operating procedures.

g. I&C and Electrical requirements necessary to accomplish safety functions.

h. Management measures specific to this plant system needed to ensure performance of the system safety functions (i.e., long-term performance, testing, and maintenance

features).

i. System interfaces including a description how the interface could prevent the system from performing its safety function.

3. Safety Analysis:

The safety analysis for the plant system should be considered acceptable if the following are true:

a. The application and referenced material demonstrate how functional requirements are satisfied by system design.

b. The application and referenced material demonstrate how potential failure modes are factored into the system design. As applicable:

i. Communication failures,
ii. Isolation between safety and non-safety-related components,
iii. Potential for common mode failures of redundant systems, and
iv. Inappropriate actions from operators or maintenance personnel that prevent the safety function from being carried out. The application clearly demonstrates how non-safety features do not prevent the system from performing its safety function.

12.4.3.2 Acceptance Criteria Unique to Specific Plant Systems

The following provide acceptance criteria that are unique to a specific plant system. These examples, as applicable, should be discussed in addition to the generic acceptance criteria.

1. **Electrical Systems**:

a. The basic system description and theory should also include relevant elements of off-site ac and dc systems and their interconnections; on-site ac and dc systems; stand-by power systems; system independence; redundancy; sharing of systems. The design features include capacity and capability, protection features, ability to withstand operating and environmental stressors, and considerations for long-term performance, testing and maintenance.

b. Structures and components should including emergency and standby power sources

c. Management measures should include tests and other verification methods that demonstrate how systems requirements have been incorporated into the electrical systems. The electrical systems requirements are traceable through design to the testing phase. Tests and verification methods demonstrate, to a sufficient degree, the ability of the electrical systems to handle credible, unexpected inputs and situations. Also, the electrical system is designed for test, calibration, and in service surveillance

requirements.

d. Safety Analysis: The safety analysis for the electrical systems should be considered acceptable if the application and referenced material demonstrate how functional requirements are satisfied by system design. As appropriate, the application and referenced material addresses how environmental effects including temperature, pressure, humidity, vibration, harmonics, electromagnetic/radio frequency interference are tolerated by the systems.

2. **Cooling Water Systems**

a. The basic system description and theory, including how the cooling water system and its subsystems transfer heat loads from safety-related structures, systems, and components to an appropriate heat sink under both normal operating and accident conditions. The description also addresses the availability of adequate water supply under normal and hazardous conditions, component redundancy and the capability to isolate components, systems or piping for maintaining system safety function under varying system configuration, and the capability of integrated system control.

b. Management measures including tests and other verification methods that demonstrate the structural integrity and system leak tightness, the operability and adequate performance of active system components, and the capability of the system to perform required functions during normal, and accident situations.

c. Descriptions of the system showing the capability for withstanding environmental hazards resulting from pipe line breaks and dynamic effects associated with flow instability and attendant loads such as water hammer, or cavitation, and measures to prevent such dynamic conditions from occurring.

d. Design features which include capacity and capability for detecting leakage of radioactivity, chemical contamination from one system to another, and allowing inservice component inspection, system maintenance, and operational functional testing of the system and its components.

3. **I&C**

a. The basic system description and theory, including real-time parameters(i.e. system response-times, sample rates, and delays), range of input and output values for normal and abnormal conditions, setpoint calculations, accuracy of signal measurements, and other detail requirements governing I&C design and acquisition. A description on how the various I&C components (both hardware and software) carry out the safety function(s). When appropriate, the description addresses how environmental effects (i.e. temperature, humidity, vibration, and electromagnetic/radio frequency interference) are tolerated by the I&C system. In addition, provisions for manual actuation of the I&C safety function(s) are described.

b. Assurance measures including codes and standards for hardware/software development and setpoint methodology.

c. Drawings and procedures comprised of logic diagrams, or a simplified system drawing with reference to drawings and procedures. In addition, references to documents are provided for hardware/software requirements specifications, P&IDs, hardware/software design documents, software code, schematics, test procedures/reports, and supporting manuals.

d. Management measures specific to the plant system needed to ensure performance of the system safety functions including long-term performance, testing, and maintenance features. Tests and other verification/validation methods for hardware/software that demonstrate that all system requirements have been incorporated into the I&C system. I&C system requirements are traceable through design to the testing phase. Tests and verification methods demonstrate, to a sufficient degree, the ability of the I&C system to handle credible, unexpected inputs and situations. Also, the I&C system is designed for easy test and calibration.

e. Description of system interfaces that provide the following information: an evaluation of potential communication failures, isolation between safety and non-safety components(considers both electrical and logic isolation), and inappropriate actions from operators or maintenance personnel that prevent the safety function from being carried out (coordinate with human factors review in Section 11.6). Interfaces with supporting systems such as power supplies, HVAC, etc., are also described.

4. **Containment/Confinement Systems**

a. The basic system description and theory, including how the various structural systems, sub-systems and their interfaces carry out their safety functions. When appropriate, the description addresses how environmental effects including temperature, pressure, humidity, wind, tornadoes, flood, lightning, and earthquakes are tolerated by the systems. The design features include capacity and capability, protection features, withstanding operating and environmental stressors, and long-term performance over time.

b. Management measures specific to the plant system needed to ensure performance of the system safety functions including long-term performance, testing, and maintenance features. Tests and other verification methods that demonstrate all systems requirements have been incorporated into the structural systems. Tests and verification method demonstrate, to a sufficient degree, the ability of the structural system to handle credible, unexpected loads and load combinations.

5. **Ventilation System**

a. The basic system description and theory including system layout and details of fans, ducts, dampers and other components sufficient to describe how the system performs

its safety function. System capacities, pressure differentials between contamination zones, and efficiencies of filters should be included in this description. Alarm set points and interlocks to prevent inadvertent pressurization of an area should also be described.

b. Structures and components including aspects that directly relate to the safety function (e.g. normal operating capacities and maximum capacities of fans and blowers, heating requirements for outside intake units, dispersion capabilities of release stacks and etc.) The capability of the system to withstand severe natural events should also be described.

c. Drawings and procedures comprised of P&IDs, volume flow diagrams or balance reports, or a simplified system drawing with reference to drawings and procedures. References to documents and manuals which describe the ventilation system and its associated support systems.

d. Management measures specific to the plant system needed to ensure performance of the system safety functions including long-term performance, testing, and maintenance features. Tests and other verification methods should demonstrate that all system requirements have been incorporated into the ventilation system. Tests and verification methods should demonstrate, to a sufficient degree, the ability of the ventilation system to handle credible, unexpected concentrations of radioactive or hazardous gases or aerosols including combustion products. Also, as necessary, the ventilation system is designed for testing and in service surveillance requirements.

12.5 REVIEW PROCEDURES

12.5.1 Acceptance Review

The primary reviewer evaluates the application to determine whether it addresses the "Areas of Review" discussed in Section 12.3, above. If significant deficiencies are identified, the applicant should be requested to submit additional material before the start of the safety evaluation.

12.5.2 Safety Evaluation

After determining that the application is acceptable for review in accordance with Section 12.5.1, above, the primary reviewer will perform a safety evaluation against the acceptance criteria described in Section 12.4. If during the course of the safety evaluation, the primary reviewer determines the need for additional information, the primary reviewer coordinates a request for additional information with the licensing project manager.

Because the results of the ISA identify the items relied on for safety that form the safety functions discussed above, the primary reviewers should also review the ISA Summary (see SRP Chapter 3.0). Plant systems, as defined in the SAR, should conform to the level of safety

to support the ISA summary. The primary reviewer should establish that the applicant's facility design and operations provide reasonable assurance that the plant systems satisfy the acceptance criteria in Section 12.4 and will perform their intended safety functions.

The secondary reviewers should confirm that the described plant systems are consistent with other sections of the application. Information provided for plant systems should be of comparable quality and detail, and should not contradict or adversely impact information contained in other sections of the application.

Supporting reviewers should confirm that provisions made in the application for plant systems are in accordance and consistent with specified sections of the SRP. For example, the primary reviewer from SRP Chapter 4.0, "Radiation Safety" (usually a health physicist), as a supporting reviewer for plant systems, should establish that the applicant provides reasonable assurance for the facility and its operations will not have unacceptably adverse impacts on the radiological safety at the facility. For another example, the primary reviewer of SRP Section 11.2, "Configuration Management," as a supporting reviewer for plant systems verifies that these safety systems are properly captured in the applicant's configuration management program.

For an existing facility, the NRC reviewers may wish to visit the site and facility personnel in order to gain a better understanding of the represented plant systems and their intended safety functions. For a planned facility, the NRC reviewers may wish to meet with the design team in order to gain a better understanding of the process, its potential hazards, and safety approaches.

When the safety evaluation is complete, the primary reviewer, with assistance from the other reviewers, should prepare the plant systems input for the Safety Evaluation Report (SER), as described in Section 12.6 using the acceptance criteria from Section 12.4. The secondary reviewer should coordinate the plant systems input with the balance of the reviews and the SER.

12.6 EVALUATION FINDINGS

The primary reviewer writes an SER section addressing each topic reviewed under this SRP Chapter and explains why the NRC staff has reasonable assurance that the chemical safety part of the application is acceptable. License conditions may be proposed to impose requirements where the application is deficient. The SER should include a summary statement of what was evaluated and the basis for the reviewers' conclusions.

The staff can document the evaluation as follows:

> *The staff has evaluated .....* [Insert a summary statement of what was evaluated and why the reviewer finds the submittal acceptable.] *Based on the review of the license application, the NRC staff has concluded that the applicant has adequately described and designed plant systems in order to adequately perform their intended safety functions as identified in the ISA. In doing so the applicant has satisfactorily addressed the baseline*

design criteria contained in 10 CFR Part 70, as revised.

12.7 REFERENCES

1. Nuclear Regulatory Commission (U.S.), Washington, D.C. "Domestic Licensing of Special Nuclear Material (10 CFR Part 70)." *Federal Register*: Vol. 64, No. 146. pp. 41338–41357. July 30, 1999.

2. Nuclear Regulatory Commission (U.S.) (NRC). Draft NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants." Section 3.8.4, "Other Seismic Category I Structures." NRC: Washington, D.C. June 1996.

3. Nuclear Regulatory Commission (U.S.) (NRC). Draft NUREG-0800. Chapter 8, "Electric Power." NRC: Washington, D.C. June 1996.

4. Nuclear Regulatory Commission (U.S.) (NRC). Draft NUREG-0800. Chapter 9, "Auxiliary Systems." NRC: Washington, D.C. June 1996.

5. Nuclear Regulatory Commission (U.S.) (NRC). Draft NUREG-0800. Chapter 14.3, "Inspections, Tests, Analyses, and Acceptance Criteria, Design Certification." NRC: Washington, D.C. June 1996.

6. Nuclear Regulatory Commission (U.S.) (NRC). Draft NUREG-0800. Chapter 7, "Instrumentation and Controls," NRC: Washington, D.C. July 1997.

APPENDIX A
FIRE HAZARDS ANALYSIS PROCEDURES

Most of the guidance in this appendix originated from "The Implementation Guide for use with DOE Orders 420.1 and 440.1 - Fire Safety Program" (G-420.1/B-0, G-440.1/E-0, September 30, 1995). In some cases, the original guidance was modified to reflect specific needs for the Hanford TWRS facilities.

A-1 Purpose: to document specific fire hazards, fire protection features proposed to control those hazards, and the overall adequacy of plant fire safety. The Fire Hazards Analysis (FHA) consists of a systematic analysis of the fire hazards, an identification of specific areas and systems important to plant fire safety, the development of design-basis fire scenarios, an evaluation of anticipated consequences, and a determination of the adequacy of plant fire safety.

A-2 A preliminary FHA should be performed for the Hanford TWRS facilities early in the design phase to ensure incorporation of an acceptable level of protection in the evolving design.

A-3 The FHA should be performed under the direction of a qualified fire protection engineer, with support from chemical, electrical, mechanical, and systems engineers, as well as operations staff as needed.

A-4 The FHA should contain, but not be limited to, a conservative assessment of the following items and safety issues:

Descriptions:
- Construction (Type)
- Fire Hazards
- Fire Protection Features
- Critical process equipment
- Operations

Potential for a toxic or radiation incident from a fire
Impact of natural hazards (earthquake, flood, or wind) on fire safety
Protection of items relied upon for safety
Life safety considerations
Emergency planning
Fire Department/Brigade response
Security and safeguards considerations related to fire protection
Exposure fire potential and the potential for fire spread between two fire areas

A-5 The FHA should assume and evaluate the consequences of a single, worst-case automatic fire protection system malfunction. This could be a detection system that also functions to activate a pre-action type sprinkler system.

A-6 If redundant automatic fire protection systems are provided in the area, only the system that causes the most vulnerable condition is assumed to fail. Passive fire protection features,

such as blank fire-rated walls or continuous fire-rated cable wraps are assumed to remain viable in accordance with their fire endurance rating to the extent that they are properly constructed and maintained.

A-7 The FHA is normally organized by the individual fire areas that comprise the facility. As defined in Section 4.7, a fire area is a location bounded by fire-rated construction, having a minimum fire resistance rating of 2 hours. The FHA through fire modeling (if necessary) and fire loading analysis should document that the fire ratings are appropriate for each fire area boundary. Where a facility is not subdivided by fire rated construction, the fire area should be defined by the exterior walls and roof of the facility.

A-8 The FHA should contain an inventory of items relied on for safety that are susceptible to fire damage within each subarea. Loss of systems such as ventilation, cooling, or electrical power that could cause failures elsewhere in the facility should be evaluated. The improper operation of equipment due to fire damage induced spurious signals should also be considered. In addition, the effects of combustion products, manual firefighting efforts, and the activation of automatic fire suppression systems should be assessed.

A-9 The FHA may need to produce fire related parameters for evaluating fire induced radiation dispersion through the facility air distribution system. The radiological consequences should then be determined as part of the integrated safety assessment.

A-10 The quantity and associated hazards of flammable and combustible material that can be expected to be found within the fire area should be factored into the analyses. Consideration should also be given to the presence of transient combustibles associated with storage and maintenance activities. Average combustible loading, by itself, should not be used to estimate fire area fire severity. As a minimum, for each designated fire area, the following fire hazards should be evaluated for potential fire severity and consequent damage:

a. Fire load from solid combustible materials (both quantity and configuration) including those materials of construction, in-situ materials, and anticipated transient combustible materials. Combustibles are defined as materials which do not meet the definition of noncombustible material as presented in NFPA Standard 220. For the purposes of the fire load survey, combustibles which can be classified as limited-combustible (as per NFPA 220) may be so classified. In performing the fire loading survey, the end uses of the survey in the FHA and/or ISA should be kept in mind. These uses may include, but not be limited to: determining or verifying the proper design basis of the fire suppression system, determining the minimum required fire resistance for barriers, assuring adequate prefire planning, and input to fire propagation or radionuclide transport modeling. Each of these uses may require the data to be presented in different formats or level of detail.

b. Flammable and combustible liquids and gases used in the processes within the fire area (quantities or flow rates).

c. Process chemicals and materials (both quantity and location) that could present a toxic or radiological hazard or that could significantly affect health or the quality of the environment through a release as a result of a fire emergency.

d. Potential ignition sources.

A-11 The FHA should contain an assessment of facility fire water requirements including capacity, pressure, and storage requirements. The assessment should include a list of water based automatic suppression systems and their maximum demands, interior hose stream requirements and exterior hydrant requirements. With this assessment, the facility fire water system layout should also be provided, including the locations and characteristics of pumps, lines, tanks, towers, and sectionalizing valves.

A-12 For each designated fire area determined to be important to plant fire safety, the FHA should provide input to the ISA regarding the postulated accident sequences caused or aggravated by fire. Either quantitative or qualitative methods may be used. Where quantitative analytical methods are used, all input data and assumptions are documented.

A-13 The FHA should define those fire protection systems and procedures that provide reasonable assurance that the defined consequences of an accident sequence will not occur or will be mitigated. The coverage of fire detection and suppression systems should be shown within each fire area. For the identified fire protection measures, the applicant should specify compensatory measures to be implemented on a temporary basis in the event the identified systems are not operable. Both the compensatory measure(s) and the time schedule for implementation should be established.

APPENDIX B
FIRE PROTECTION GUIDANCE FOR NUCLEAR FILTER PLENUMS

B-1 Introduction

Most of the guidance presented here is taken from DOE Standard, "Fire Protection Design Criteria" (DOE-STD-1066-97, March 1997). The items of guidance presented are considered to be pertinent to the filter systems likely to be used at the Hanford TWRS facilities. The items presented also represent the NRC responsibility for fire safety as related to facility nuclear safety rather than property protection. A more comprehensive discussion of Nuclear Filter Plenum Fire Protection can be found in Chapter 14 of the DOE Standard and the references cited in the standard.

B-2 Filter Plenum Construction

All high-efficiency particulate air (HEPA) filters should meet the requirements of ASME AG-1, Section FC and listed as tested in accordance with UL 586. Entrance filters and prefilters located upstream or made part of final HEPA filter exhaust plenums should be listed as Class 1 air filter units as tested in accordance with UL 900. Filter framing systems should be of noncombustible construction.

B-3 Fire Rating Requirements for Plenum Housing, Openings, and Dampers

a. Filter plenum enclosures inside buildings or located less than five feet from an adjacent building must be of 2-hour fire rated construction. For enclosures greater than five feet from an existing building, the fire rating may be either one-hour or as determined by the FHA.

b. Door openings into a two-hour rated filter plenum enclosure should be 1.5-hour minimum fire rated. Door openings into a one-hour rated filter plenum enclosure should be .75-hour minimum fire rated.

c. For ducts not required to function as a nuclear confinement system:

   i. 1.5-hour damper is required where the duct penetrates a two-hour rated barrier.

   ii. A fire damper is not required where the duct penetrates a one-hour barrier provided that automatic sprinkler protection is provided on both sides of the barrier and the duct passes through the wall and extends into the area outside the enclosure. Transfer grills and similar openings without ducting should be provided with an approved damper.

d. Fire dampers should not be utilized when penetrating fire rated construction where ducting is an integral part of the air filter system equipment that is required to continuously function as part of the confinement system. Such duct material may be made part of the fire rated construction by wrapping, spraying, or enclosing the duct with an approved material to provide a minimum 2-hour rating; or be qualified for a 2-

hour fire rated exposure to the duct at the penetration location using the fire damper criteria as specified in UL 555.

e. All mechanical and electrical penetrations made into fire-rated plenum enclosures should be fire stopped by listed materials meeting the requirements of ASTM E-814.

B-4 Materials and Hazards Inside Plenums

a. Filter plenum enclosures should only be used for ventilation control equipment. The storage and accumulation of combustible materials (including spare filters) as well as combustible and flammable liquids should not be permitted.

b. Electrical equipment should comply with NFPA 70 and all electrical wiring inside the enclosure should be in metal conduit.

c. The concentration of flammable vapors inside the final filter plenum should not exceed 25 percent of their lower flammable limit. If flammable and combustible gases are expected as a result of facility processes, fixed combustible gas analyzers should be provided with analyzer alarms set to sound at 25 percent of the lower flammable limit and transmitted to a continuously manned position.

B-5 Fire Screens for Filter Plenums

a. Fire screens should be located upstream from the prefilters and final filter plenums.

b. Fire screens with metal meshes from 8 to 16 openings per inch should be provided and located at least 4 feet upstream from all prefilters and at least 20 feet upstream from all final filter plenum enclosures.

c. Where prefilters are located in final filter enclosures, fire screens should be located at least 20 feet upstream from the prefilters.

B-6 Fire Detection Systems

a. Automatic fire detectors should be rate compensated type heat detectors, approved for the specific use and conform to NFPA 72. The detectors should be of the 190 °F temperature range, unless operations require higher temperature air flows.

b. Heat detectors or pilot sprinkler heads should be provided in the final filter enclosure and in ducting prior to the final filter enclosure. Airflow should be considered when determining detector or pilot head location in ducting.

c. Detector installations should be engineered and installed for testing over the life of the detector. Where contamination levels permit, detectors can be removed and tested externally.

B-7 Deluge Spray Suppression Systems

a. Automatic and manual water deluge spray systems should be provided inside all final filter plenums for protection of the filters where there is a leading filter surface area greater than 16 square feet.

b. Automatic deluge systems should be designed as per the applicable provisions of NFPA 13 and 15 and as follows:

   i. Water spray density should be 0.25 gpm/ft^2 over the entire filter area or 1.0 gpm per 500 cfm air flow, whichever is greater

   ii. Spray heads should be deluge type sprinkler heads

   iii. The spray pattern of the deluge head should be in the form of a downward vertical water curtain approximately 6 inches in front of the filter. Heads should be spaced so that each head does not exceed 4 lineal feet of curtain coverage.

c. Manual spray systems should be designed as per the applicable provisions of NFPA 15 and modified as follows:

   i. Water spray density should be 0.25 gpm/ft^2 over the entire filter area.

   ii. Nozzles should be deluge spray nozzles that form a full circle solid cone discharge.

   iii. Spray nozzles should be horizontally directed at the face of the first series HEPA filters so that all areas of the first stage filters and framing support system are wetted.

d. Automatic and manual water spray system water supplies should be hydraulically calculated and capable of supplying a simultaneous flow of the automatic and manual water spray systems as well as the overhead ceiling automatic fire sprinkler systems for the fire area providing air to the plenum for a minimum period of two hours.

e. Water for the deluge spray system should be provided by two separate water supply connections for reliability. One connection may be a fire department connection.

APPENDIX C
CHECKLIST FOR EVALUATING ACCEPTANCE OF QUALITY ASSURANCE ELEMENTS

1. Organization - The organizational elements responsible for Quality Assurance (QA) are acceptable if:

   a. The responsibility for the overall QA is retained and exercised by the applicant.

   b. The applicant identifies and describes the major delegation of work involved in establishing and implementing its QA program or any part thereof to other organizations.

   c. When major portions of the applicant's QA program are delegated:

      i. The applicant describes how responsibility is exercised for overall QA. The extent of management supervision should be given, including the position location, qualifications, and criteria for determining the number of personnel performing these functions.

      ii. The applicant evaluates the performance of work by the delegated organization (method and frequency - once per year, although a longer cycle is acceptable with other evaluations of individual elements - are stated).

      iii. Qualified individuals or organizational elements are identified by position title within the applicant's organization as responsible for the quality of the delegated work before activities are started.

   d. Clear management controls and effective lines of communication exist for QA activities among the applicant, contractors, and suppliers to ensure direction of QA.

   e. Organizational charts clearly identify all the onsite and offsite organizational elements that function under the purview of QA (such as design, engineering, procurement, manufacturing, construction, inspection, testing, instrumentation, control, operation, and maintenance), the lines of responsibility, and the criteria for determining the size of the QA organization, including the inspection staff.

   f. The applicant describes the QA responsibilities of each of the organizational elements noted on the organization charts.

   g. The applicant identifies a management position that retains overall authority and responsibility for QA. This position may be filled by a person having the title "QA Manager" or other individual performing that function, and this position has the following characteristics:

i. The position resides at least at the same organizational level as the position of the highest line manager directly responsible for performing activities that affect the quality/safety of plant operations (such as engineering, procurement, construction, and operation) and is independent of operational restraints.

ii. The person in the position has effective communication channels with other senior management personnel.

iii. The person in the position has responsibility for approval of QA manuals.

h. Conformance to established requirements (except for designs) is verified by individuals or groups within the QA organization who do not have direct responsibility for performing the work being verified or by individuals or groups trained and qualified in QA concepts and practices who are independent of the organization responsible for performing the task.

i. Persons and organizations performing QA functions have sufficient access to management at a level necessary to ensure the capability to:

i. Identify quality/safety problems;

ii. Initiate, recommend, or provide solutions through designated channels; and

iii. Verify implementation of solutions.

Positions with the above authority are identified by position title and a description of how the above actions are carried out is provided.

j. When work contributes to a situation adverse to safety and has to be stopped, the following provisions apply:

i. Designated QA personnel, sufficiently free from direct pressures resulting from operational concerns, have the responsibility, delineated in writing, to stop work in unsafe situations and to control further operations until the conditions that created the unsafe condition are corrected.

ii. The organizational positions with stop-work authority are identified.

k. Provisions are established for the resolution of disputes involving quality of items relied on for safety arising from a difference of opinion between QA personnel and personnel from other departments (engineering, procurement, manufacturing, etc.).

l. Designated QA individuals are involved in day-to-day activities relied on for safety of plant conditions and operations and QA staff members routinely attend

and participate in status meetings to ensure that they are kept abreast of day-to-day activities and that there is adequate QA coverage of those activities.

m. Policies regarding the implementation of QA are documented and made mandatory. These policies are established at the plant management or at the corporate level.

n. The position description ensures that the individual directly responsible for the definition, direction, and effectiveness of overall QA has sufficient authority to effectively implement responsibilities. This position is to be sufficiently free from operational responsibilities to ensure independence of action. Qualification requirements for this individual are established in a position description that includes the following prerequisites:

   i. Management experience through assignments to responsible positions;

   ii. Knowledge of QA regulations, policies, practices, and standards; and

   iii. Experience in performing QA or QA-related activities in design, construction, or operation in a fuel cycle plant, a power reactor, a low-level waste facility, or in a similar high-technology industry.

o. The person responsible for onsite QA is identified by position and has the appropriate organizational position, responsibilities, and authority to exercise proper control over QA. The duties of this individual are structured such that adequate attention can be given to ensuring that QA at the plant site is being effectively implemented.

Additional guidance for organization is given in SRP Section 2.0, "Organization and Administration."

2. QA Function - The QA function for items relied on for safety is acceptable if:

   a. The scope of QA includes:

      i. A commitment that activities affecting the quality of design, construction, and operation will be subject to the applicable controls of QA and activities covered by QA are identified on QA-defining documents.

      ii. A commitment that any test program for items relied on for safety will be conducted with QA controls and a description of how QA will be applied.

      iii. A commitment that computer programs for functions related to safety will be procured/developed, modified, maintained, and used in accordance with QA controls and a description of how QA will be applied.

iv. A commitment that special items, environmental conditions, skills, or processes will be provided as necessary to ensure the quality of activities having an effect on safety.

b. A brief summary of the applicant's corporate QA policies is given.

c. The following provisions are established to ensure that quality--affecting procedures required to implement QA are consistent with QA commitments and corporate policies and are properly documented, controlled, and made mandatory through a policy statement or equivalent document signed by the responsible official:

i. The QA organization reviews and documents concurrence in the quality--affecting procedures.

ii. The organizational group or individual responsible for the policy statement is identified.

iii. The quality--affecting procedural controls of the principal contractors are provided for the applicant's review with documented agreement of acceptance before the initiation of activities relied on for safety.

d. Provisions are included for notifying the NRC of changes in the implementation of QA from that described in the application.

e. The QA organization and the necessary technical organizations participate early in the QA definition stage to determine and identify QA controls and the extent to which they are to be applied to items as they relate to safety. This effort may involve applying a defined, graded approach to the items in accordance with their importance to safety.

f. A description is provided that emphasizes how the detailed QA will be properly implemented and carried out.

g. A description is provided of how management (above or outside the QA organization) regularly assesses the scope, status, adequacy, and compliance of QA. These measures should include:

i. Frequent appraisals of QA status through reports, meetings, audits and/or self assessments;

ii. Performance of an annual, preplanned, and documented assessment; and

iii. Identification and tracking of corrective actions based on assessment findings.

h. Activities relied on for safety (such as design, procurement, and site investigation) initiated prior to formal NRC acceptance of the QA program are controlled by a QA program in accordance with this SRP section. Approved procedures and appropriately trained personnel should be available to implement the applicable portion of the QA program prior to the initiation of the activity.

i. A summary description is provided on how responsibilities and control of quality-affecting activities are transferred from the principal contractors to the applicant as the design and construction phase is completed.

j. Indoctrination, training, and qualification[1] are established so that:

   i. Personnel responsible for performing and verifying activities affecting quality are instructed as to the purpose, scope, and implementation of the applicable manuals, instructions, and procedures.

   ii. Personnel performing and verifying activities affecting safety and/or quality are trained and qualified in the principles, techniques, and requirements of the activity being performed.

   iii. For formal training and qualification, documentation includes a statement of the training objective and its content, the attendees, and the date of attendance.

   iv. Proficiency tests are given to those personnel performing and verifying activities affecting quality, and acceptance criteria are developed to determine if individuals are properly trained and qualified.

   v. The certificate of qualifications clearly delineates the specific functions personnel are qualified to perform and the criteria used to qualify personnel in each function.

   vi. Proficiency of personnel performing and verifying activities affecting safety/quality is maintained by retraining, reexamining, and/or recertifying, as determined by management or program commitment.

k. The applicant's ISA is developed and maintained under QA controls.

3. Design Control[2] - Control of the design of items relied on for safety is acceptable if:

   a. The scope of design control includes design activities associated with the preparation and review of design documents, including the correct translation of

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[1] Guidance for training and qualification of plant personnel is given in SRP Section 11.4.

[2] Guidance for configuration management is given in SRP Section 11.2.

applicable regulatory safety requirements and associated design bases into design, procurement, and procedural documents.

b. Organizational responsibilities are described for preparing, reviewing, approving, and verifying design documents related to an item or its processes, such as system descriptions, design input and criteria, design drawings, design analyses, computer programs, specifications, and procedures.

c. Organizational responsibilities are described for planning and conducting site characterization, including reviewing, approving, and verifying analyses and conclusions.

d. Errors and deficiencies in approved design documents, including design methods (such as computer codes) that could adversely affect the performance of items and processes are documented, and action is taken to ensure that all errors and deficiencies are corrected.

e. Deviations from specified quality standards are identified, and procedures are established to ensure their control.

f. Internal and external design interface controls, procedures, and lines of communication among participating design organizations and across technical disciplines are established and described for the review, approval, release, distribution, and revision of documents involving design interfaces to ensure that items are compatible geometrically and functionally.

g. Procedures are established and described requiring documented verification of the dimensional accuracy and completeness of design drawings and specifications.

h. Procedures are established and described requiring that design drawings and specifications for items relied on for safety be reviewed by the QA organization to ensure that the documents are prepared, reviewed, and approved in accordance with company procedures and that the documents contain necessary QA requirements, such as inspection and test requirements, acceptance requirements, and those pertaining to the extent of documenting inspection and test results. These reviews are documented.

i. Guidelines or criteria are established and described for determining the method of design verification (design review, alternate calculations, or tests).

j. Procedures are established and described for design verification activities that ensure the following:

    i. The verifier is qualified, and neither the verifier nor the verifier's immediate supervisor is directly responsible for the design. In exceptional circumstances, the designer's immediate supervisor may perform the verification provided:

1. the supervisor is the only technically qualified individual,

2. the need is individually documented and approved in advance by the supervisor's management,

3. QA audits and self assessments cover frequency and effectiveness of the use of supervisors as design verifiers to guard against abuse.

ii. Design verification is completed before release of procurement, manufacturing, or construction to another organization for use in other design activities. When this schedule cannot be met, the design verification may be deferred, provided the justification for this action is documented and the unverified portion of the design output document and all design output documents, based on the unverified data, are appropriately identified and controlled. Construction site activities associated with a design or design change should not proceed without verification past the point where the installation would become irreversible (i.e., require extensive demolition and rework).

iii. Procedural control is established for design documents that reflect the commitments of the Safety Program Description, which includes the ISA; this control differentiates between documents that undergo formal design verification by interdisciplinary or multi-organizational teams and those that can be reviewed by a single individual (a signature and date are acceptable documentation for personnel certification). Design documents that pertain to plant safety and are subject to procedural control include, but are not limited to, specifications, calculations, computer programs, system descriptions, and drawings, including flow diagrams, piping and instrument diagrams, control logic diagrams, electrical single-line diagrams, diagrams of structural systems for major facilities, site arrangements, and equipment locations. Specialized reviews should be used when uniqueness or special design considerations warrant them.

iv. The responsibilities of the verifier, the areas and features to be verified, the pertinent considerations to be verified, and the extent of documentation are identified in procedures.

k. The following provisions are included if the verification method is only by test:

i. Procedures provide criteria that specify when verification should be by test.

ii. Prototype, component, or feature testing is performed as early as possible before installation of plant items or before the installation would become irreversible.

iii. Verification by test is performed under conditions that simulate the most adverse design conditions as determined by analysis.

l. Procedures are established to ensure that verified computer codes are certified for use and that their use is specified.

m. Design and specification changes, including field changes, are subject to the same design controls that were applicable to the original design.

4. Procurement Document Control - Control of procurement documents for the procurement of items relied on for safety is acceptable if:

a. Procedures are established for the review of procurement documents to determine that quality requirements are correctly stated, are inspectable, and are controllable; there are adequate acceptance and rejection criteria; and procurement documents have been prepared, reviewed, and approved in accordance with QA requirements. To the extent necessary, procurement documents should require that contractors and subcontractors provide acceptable QA. The review and documented concurrence of the adequacy of quality requirements stated in procurement documents are performed by independent personnel trained and qualified in QA practices and concepts.

b. Procedures are established to ensure that procurement documents identify applicable regulatory, technical, administrative, and reporting requirements; drawings; specifications; codes and industrial standards; inspection and test requirements; and special process instructions that must be met by suppliers.

c. Organizational responsibilities are described for procurement planning; the preparation, review, approval, and control of procurement documents; supplier selection; bid evaluations; and the review of and concurrence with supplier QA before initiation of activities relied on for safety. The involvement of the QA organization is described.

5. Instructions, Procedures,[3] and Drawings - Activities related to instructions, procedures, and drawings pertaining to items relied on for safety are acceptable if:

a. Organizational responsibilities are described for ensuring that activities affecting the quality of items relied on for safety are prescribed by documented instructions, procedures, and drawings and accomplished through implementation of these documents.

b. Procedures are established to ensure that instructions, procedures, and drawings that could affect safety include quantitative acceptance criteria (such as those pertaining to dimensions, tolerances, and operating limits) for

[3] Guidance for plant procedures is given in SRP Section 11.5.

determining that activities relied on for the safety of plant operations have been satisfactorily performed.

6. Document Control - Control of documents related to items relied on for safety is acceptable if:

   a. The scope of document control is described and the types of controlled documents are identified. As a minimum, controlled documents include:

      i. Design documents (e.g., calculations, drawings, specifications, and analyses), including documents related to computer codes;

      ii. Procurement documents;

      iii. Instructions and procedures for such activities as fabrication, construction, modification, installation, maintenance, testing, and inspection;

      iv. Documents pertaining to as-built conditions;

      v. QA and quality control manuals, procedures, and reports; and

      vi. Technical reports.

   b. Procedures for the review, approval, and issuance of documents and changes thereto are established and described to ensure technical adequacy and inclusion of appropriate safety/quality requirements before implementation. The QA organization, or an individual other than the person who generated the document but who is qualified in QA, reviews and concurs with these documents in regard to QA-related aspects.

   c. Procedures are established to ensure that changes to documents are reviewed and approved by the same organizations as those that performed the initial review and approval or by other qualified, responsible organizations delegated by the applicant.

   d. Procedures are established to ensure that documents are available at the location where the activity will be performed, before commencing work.

   e. Procedures are established and described to ensure that obsolete or superseded documents are removed and replaced by applicable revisions in work areas in a timely manner.

   f. A master list or equivalent document control system is established to identify the current revision of instructions, procedures, specifications, drawings, and procurement documents. When such a list is used, it should be updated and distributed to predetermined responsible personnel.

g. Procedures are established and described to provide for the preparation of drawings pertaining to as-built conditions and related documentation in a timely manner to accurately reflect the actual design.

7. Control of Purchased Items - Control of purchased items relied on for safety is acceptable if:

a. Organizational responsibilities are described for the control of purchased items including interactions between design, procurement, and QA organizations.

b. Verification of suppliers' activities during fabrication, inspection, testing, and shipment of items relied on for safety is planned and performed with QA organization participation in accordance with written procedures to ensure conformance to the purchase order requirements. The procedures, as applicable to the method of procurement, provide for:

i. The specification of the characteristics or processes to be witnessed, inspected or otherwise verified; the method of verification and the required documentation; and the personnel responsible for implementing these procedures; and

ii. Audits, surveillances, or inspections that ensure that the supplier complies with the quality requirements.

c. Procurement of spare or replacement parts for items relied on for safety is subject to QA controls, to codes and standards, and to technical requirements equal to or better than the original technical requirements, or as required to prevent the procurement of defective items.

d. Selection of suppliers is documented and filed.

e. Items are inspected when received to ensure:

i. The item is properly identified and corresponds to the identification on the purchase document and the documentation when the item is received.

ii. The item and acceptance records satisfy the inspection instructions before installation or use of the item.

iii. Specified inspection, test, and other records (such as certificates of conformance attesting that the item conforms to specified requirements) are available at the facility before installation or use of the item.

f. Items accepted and released are identified as to their inspection status before they are forwarded to a controlled storage area or are released for installation or further work.

g. The supplier furnishes the following records to the purchaser:

i. documentation that identifies the purchased item and the specific procurement requirements (e.g., codes, standards, and specifications) met by the item relied on for safety;

ii. documentation that identifies any procurement requirements that have not been met; and

iii. a description of those items that do not conform to the procurement requirements and that are designated "accept as is" or "repair."

The procedure for review and acceptance of these documents is described.

h. For commercial "off-the-shelf" items where specific QA controls cannot be imposed in a practicable manner, special quality verification requirements are established and described to ensure that an acceptable item has been received by the purchaser.

i. Suppliers' certificates of conformance are periodically evaluated by audits, independent inspections, or tests to ensure that they are valid and that the results are documented.

8. Identification and Control of Items - Identification and control of items relied on for safety are acceptable if:

a. Controls are established and described to identify and control items relied on for safety. The description should include organizational responsibilities.

b. Procedures are established that ensure that identification is maintained either on the item relied on for safety or on records traceable to the item, to preclude use of incorrect or defective items.

c. Identification of items relied on for safety can be traced to the appropriate documentation such as drawings, specifications, purchase orders, manufacturing and inspection documents, deviation reports, and physical and chemical test reports.

d. Correct identification of items is verified and documented before they are released for fabrication, assembling, shipping, and installation.

9. Control of Special Processes - Control of special processes related to items relied on for safety is acceptable if:

a. Organizational responsibilities, including those for the QA organization, are described for the qualification of special processes, equipment, and personnel.

b. Procedures are established for recording evidence of an acceptable level of quality for special processes, using qualified procedures, equipment, and personnel.

c. Qualification records of procedures, equipment, and personnel associated with special processes are established, filed, and kept current.

10. Inspection - Inspection of items relied on for plant or process safety is acceptable if:

a. The scope of inspection indicates that an effective inspection program has been established. Procedures provide criteria for determining the accuracy requirements of inspection equipment and criteria for determining when inspections are required or for defining how and when inspections are performed. The QA organization participates in these functions.

b. Organizational responsibilities for inspection are described. Individuals performing inspections are other than those who performed or directly supervised the item/activity being inspected and do not report directly to the immediate supervisors who are responsible for the item/activity being inspected. If the individuals performing inspections are not part of the QA organization, the inspection procedures, personnel qualification criteria, and independence from undue pressure, such as operational needs, should be reviewed and found acceptable by the QA organization before the initiation of the activity.

c. A qualification plan for inspectors is established and documented and the qualifications and certifications of inspectors are kept current.

d. Inspection procedures, instructions, or checklists provide for the following:

i. Identification of characteristics and activities to be inspected;

ii. A description of the method of inspection;

iii. Identification of the individuals or groups responsible for performing the inspection in accordance with the provisions of Item 10.b in this section;

iv. Acceptance and rejection criteria;

v. Identification of required procedures, drawings, and specifications and revisions;

vi. Identification of inspection personnel, measuring and test equipment used (including any data recorders), and the results of the inspection; and

vii. Specification of the necessary measuring and test equipment, including accuracy requirements.

e. Inspection results are documented and evaluated and their acceptability is determined by a responsible individual or group.

11. Test Control - Control of tests of items relied on for safety is acceptable if:

a. The description of the scope of test control indicates that an effective test program has been established for tests, including proof tests before installation and pre-operational tests. Procedures provide criteria for determining the accuracy requirements of test equipment and provide criteria for determining when a test is required or how and when testing activities should be performed.

b. Test procedures or instructions provide, as required, for the following:

i. The requirements and acceptance limits in applicable design and procurement documents;

ii. Instructions for performing the test;

iii. Test prerequisites such as calibrated instrumentation, adequate test equipment and instrumentation, including their accuracy requirements, completeness of items to be tested, suitable and controlled environmental conditions, and provisions for data collection and storage;

iv. Test acceptance and rejection criteria,

v. Mandatory inspection hold points for witness by owner, contractor, or inspector (as applicable);

vi. Methods of documenting or recording test data and results; and

vii. Provisions for ensuring that test prerequisites have been met.

c. Test results are documented and evaluated and their acceptability is determined by a responsible individual or group.

d. A qualification plan is established and documented for those individuals conducting the tests and certifications for those individuals performing the tests are kept current.

12. Control of Measuring and Test Equipment - Control of measuring and test equipment (such as instruments, tools, gauges, fixtures, reference and transfer standards, and nondestructive test equipment) relied on for safety or used to measure tor test other items relied on for safety is acceptable if:

a. The scope for the control of measuring and test equipment is described, along with the types of equipment to be controlled. This information indicates that effective calibrations and adjustments have been established.

b. QA and other organizations' responsibilities are described for establishing, implementing, and ensuring the effectiveness of the calibrations and adjustments.

c. Procedures are established and described for calibration (technique and frequency), maintenance, and control of measuring and test equipment. The review of and documented concurrence with these procedures are described and the organization responsible for these functions is identified.

d. Measuring and test equipment is identified and traceable to the calibration data.

e. Measuring and test equipment is labeled, tagged, or "otherwise controlled" to indicate the due date of the next calibration. The method to "otherwise control" measuring and test equipment should be described.

f. Measuring and test equipment is calibrated at specified intervals on the basis of the required accuracy, purpose, degree of usage, stability characteristics, and other conditions affecting the measurement. The test equipment should have sufficient accuracy to ensure that the equipment being calibrated is within required tolerance, and the basis of acceptance is documented and authorized by responsible management. The management authorized to perform this function is identified.

g. Calibrating standards have greater accuracy than standards being calibrated. Calibrating standards with the same accuracy may be used if they can be shown to be adequate to meet the requirements, and the basis of acceptance is documented and authorized by a responsible member of the management staff. The management staff member authorized to perform this function is documented.

h. Reference and transfer standards are traceable to nationally recognized standards; where national standards do not exist, provisions are established to document the basis for calibration.

i. Measurements are taken and documented to determine the validity of previous inspections and the acceptability of items inspected or tested since the last calibration when measuring and test equipment is found to be out of calibration. Inspections or tests are repeated on items determined to be suspect.

13. Handling, Storage, and Shipping - Handling, storage, and shipping of items relied on for safety are acceptable if:

a. Special handling, preservation, storage, cleaning, packaging, and shipping requirements are established and implemented by suitably trained individuals in accordance with predetermined work and inspection instructions.

b. Procedures are established and described to control the cleaning, handling, storage, packaging, and shipping of items in accordance with design and procedure requirements.

14. Inspection, Test, and Operating Status - Inspection, test, and operating status of items relied on for safety are acceptable if:

a. Procedures are established to indicate the inspection, test, and operating status of items.

b. Procedures are established and described to control the application and removal of inspection and welding stamps and status indicators such as tags, markings, labels, and stamps.

c. Procedures are established and described to control the alteration of the sequence of required tests, inspections, and other operations relied on for safety. Such actions should be subject to the same controls as those for the original review and approval.

d. The status of nonconforming, inoperative, or malfunctioning items and processes is documented and identified to prevent inadvertent use. The organization responsible for this function is identified.

15. Nonconforming Items - Control of nonconforming items relied on for safety is acceptable if:

a. Procedures are established and described for the identification, documentation, segregation, review, disposition, and notification to affected organizations of nonconforming items (including computer codes) if disposition is other than to scrap. The procedures identify authorized individuals responsible for the independent review of nonconforming items, including their disposition and closeout.

b. QA and other organizational responsibilities are described for the definition and implementation of activities related to nonconformance control. This includes identifying those individuals or groups with authority for the disposition of nonconformance.

c. Documentation identifies the nonconforming item; describes the nonconformance, the disposition of the nonconforming item, and the inspection requirements; and includes signature approval of the disposition. Nonconformances are corrected or resolved before the initiation of preoperational testing of the item.

d. Reworked, repaired, and replacement items are inspected and tested in accordance with the original inspection and test requirements or acceptable alternatives.

e. Nonconformance reports are periodically analyzed by the QA organization to show quality trends, and the significant results are reported to upper management for review and assessment.

16. Corrective Action - Corrective actions relied on for safety are acceptable if:

a. Procedures are established and described indicating that effective corrective actions have been established. The QA organization reviews and documents concurrence with the procedures.

b. Corrective action is documented and initiated after the determination of a condition adverse to safety/quality (e.g., nonconformance, failure, malfunction, deficiency, deviation, defective item, a failure to follow operating procedures, or a human error) to preclude recurrence. The QA organization concurrence is required regarding the adequacy of the corrective action.

c. Follow-up action is taken by the QA organization to verify proper implementation of corrective action and to close out the corrective action in a timely manner.

d. Significant conditions adverse to safety, the root cause of the conditions, and the corrective action taken to preclude repetition are documented and reported to immediate management and upper levels of management for review and assessment.

17. QA Records[4] - Control of QA records is acceptable if:

a. QA and other organizations are identified and their responsibilities are described for the definition and implementation of QA records.

b. Inspection and test records contain the following, where applicable:

i. A description of the type of observation,

ii. The date and results of the inspection or test,

iii. Information on conditions adverse to quality,

iv. Identification of the inspector or data recorder,

v. Evidence as to the acceptability of the results, and

vi. Action taken to resolve any discrepancies noted.

c. Suitable facilities for the storage of the records are described.

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[4] Additional guidance for records management is given in SRP Section 11.9.

18. Audits and Assessments - Guidance for audits and assessments is given in SRP Section 11.7.

19. Applicant's Provisions for Continuing QA - The applicant's provisions for continuing QA are acceptable if the submittal addresses reviews and updates of based on reorganizations, revised activities, lessons learned, changes to applicable regulations, and other QA changes that should be reflected in the license application's QA description to keep it current.

APPENDIX D
CHECKLIST FOR PLANT PROCEDURES

All activities listed below should be covered by written procedures. The list is not intended to be all inclusive nor is it intended to imply that procedures be developed with the same titles as those on the list. This listing is divided into four categories and provides guidance on topics to be covered.

1. Management Control Procedures:

    a. Training
    b. Audits and Assessments
    c. Incident Investigation
    d. Records Management
    e. Configuration Management
    f. Quality Assurance
    g. Equipment control (lockout/tagout)
    h. Shift turnover
    i. Work Control
    j. Management control
    k. Procedure management
    l. Nuclear criticality safety
    m. Fire protection
    n. Radiation protection
    o. Radioactive waste management
    p. Maintenance
    q. Environmental protection
    r. Chemical process safety
    s. Operations
    t. Calibration control
    u. Preventive maintenance
    v. Design Control
    w. Test Control
    x. Laser Safety

2. Operating Procedures

    a. System Procedures that Address Startup, Operation, Shutdown Control of Process Operations and Recovery After a Process Upset

        1. Ventilation
        2. Criticality alarms
        3. Shift routines, shift turnover and operating practices
        4. Decontamination operations
        5. Plant Utilities (air, other gases, cooling water, fire water, steam)
        6. Temporary changes in operating procedures

b. Abnormal Operation/Alarm Response:
    1. Loss of cooling water
    2. Loss of instrument air
    3. Loss of electrical power
    4. Loss of criticality alarm system
    5. Fires
    6. Chemical process releases

3. Maintenance Activities that Address System Repair, Calibration, Surveillance, and Functional Testing

    a. Repairs and preventive repairs of items relied on for safety
    b. Testing of criticality alarm units
    c. Calibration of items relied on for safety
    d. HEPA filter maintenance
    e. Functional testing of items relied on for safety
    f. Relief valve replacement/testing
    g. Surveillance/monitoring
    h. Pressure vessel testing
    i. Piping integrity testing
    j. Containment device testing

4. Emergency Procedures:

    a. Response to a criticality
    b. Hazardous process chemical releases

APPENDIX E
RECORDS

The requirements for records management will vary according to the nature of the facility and the hazards and risks posed by it. Examples of the types of records that should be included in the system required by 10 CFR Parts 19, 20, 21, 25 and 70 are presented in the first list below.

In the second list are examples of the types of appropriate records that may be established and maintained to provide reasonable assurance that items relied on for safety will be available and reliable to perform their function when needed, as referenced in 10 CFR 70.64. These listings are organized under the chapter headings of the SRP.

Although both lists provide examples of records, the listings are not intended to be exhaustive or prescriptive in format. Furthermore, the applicant may choose to organize the records in ways other than shown here.

Examples of Records Required by 10 CFR Parts 19, 20, 21, 25, 70

Audits
Access authorization for personnel
Administrative procedures with safety Implications
Air samples
Bioassay data
Change control records for material control and accounting program
Dose to individuals of the public
Exposure history
Individual monitoring data
Individual monitoring results
Individual intakes of radioactive material
Material storage records
Planned special exposures
Radiation protection (and contamination control) records
Radiation training records
Radiation work permits
Records of cumulative occupational radiation dose
Records of receipt, transfer and disposal of radioactive material
Records of waste disposal
Reports of theft/loss of licensed material
Results of surveys/calibrations
Results of measurements used to calculate radioactive effluents
Safety and health compliance records, medical records, personnel exposure records, etc.

Examples of Records that Should Provide Reasonable Assurance that Items Relied on for Safety will be Available and Reliable to Perform their Function

1. General Information

    a. Construction records
    b. Facility and equipment descriptions and drawings
    c. Design criteria, requirements, and bases for safety-related structures, systems, or components, as specified by the facility configuration management system
    d. Records of facility changes and associated integrated safety analyses, as specified by the facility configuration management system
    e. Safety analyses, reports, and assessments
    f. Records of site characterization measurements and data
    g. Records pertaining to onsite disposal of radioactive or mixed wastes in surface landfills
    h. Specifications for safety-related items

2. Organization and Administration

    a. Administrative procedures with safety implications
    b. Change control records for material control and accounting program
    c. Organization charts, position descriptions, and qualifications records
    d. Safety and health compliance records, medical records, personnel exposure records
    e. Quality assurance records
    f. Safety inspections, audits, assessments, and investigations
    g. Safety statistics and trends

3. Integrated Safety Analysis

4. Radiation Safety

    a. Bioassay data
    b. Exposure records
    c. Radiation protection (and contamination control) records
    d. Radiation training records
    e. Radiation work permits

5. Nuclear Criticality Safety

    a. Nuclear criticality control written procedures and statistics
    b. Nuclear criticality safety analyses
    c. Records pertaining to nuclear criticality inspections, audits, investigations, and assessments
    d. Records pertaining to nuclear criticality incidents, unusual occurrences, or accidents
    e. Records pertaining to nuclear criticality safety analyses

6. Chemical Safety

    a. Chemical process safety procedures and plans
    b. Records pertaining to chemical process inspections, audits, investigations, and assessments
    c. Diagrams, charts, and drawings
    d. Records pertaining to chemical process incidents, unusual occurrences, or accidents
    e. Chemical process safety reports and analyses
    f. Chemical process safety training

7. Fire Protection

    a. Fire Hazard Analysis
    b. Fire prevention measures, including hot-work permits and fire-watch records
    c. Records pertaining to inspection, maintenance, and testing of fire protection equipment
    d. Records pertaining to fire protection training and retraining of response teams
    e. Pre-fire emergency plans

8. Emergency Management

    a. Emergency plan(s) and procedures
    b. Comments on emergency plan from outside emergency response organizations
    c. Emergency drill records
    d. Memorandum of understanding with outside emergency response organizations
    e. Records of actual events
    f. Records pertaining to the training and retraining of personnel involved in emergency preparedness functions
    g. Records pertaining to the inspection and maintenance of emergency response equipment and supplies

9. Environmental Protection

    a. Environmental release and monitoring records
    b. Environmental Report and Supplements to the Environmental Report, as applicable

10. Decommissioning

    a. Financial assurance documents
    b. commissioning cost estimates
    c. Site characterization data
    d. Final survey data
    e. Decommissioning procedures

11. Management Measures

11.1 Quality Assurance

Audit records

11.2 Configuration Management

a. Safety analyses, reports, and assessments that support the physical configuration of process designs, and changes to those designs
b. Validation records for computer software used for safety analysis or MC&A
c. ISA documents, including process descriptions, plant drawings and specifications, purchase specifications for items relied on for safety
d. Approved, current operating procedures and emergency operating procedures

11.3 Maintenance

a. Preventive maintenance records, including trending and root cause analysis
b. Calibration and testing data for items relied on for safety
c. Corrective maintenance records

11.4 Training and Qualification of Plant Personnel

a. Personnel training and qualification record
b. Procedures

11.5 Plant Procedures

a. Standard operating procedures
b. Functional test procedures

11.6 Human Factors Engineering/Personnel Activities

Personnel performance trends analyses and human factor improvements

11.7 Audits and Assessments

Audits and assessments of safety and environmental activities

11.8 Incident Investigations

a. Investigation reports
b. Changes recommended by investigation reports, how and when implemented
c. Summary of reportable events for the term of the license
d. Incident investigation policy

11.9 Records Management

a. Policy
b. Material storage records
c. Records of receipt, transfer and disposal of radioactive material

12. Plant Systems

a. Plant systems written procedures and statistics
b. Plant systems safety analyses and management measures
c. Records pertaining to plant systems inspections, audits, investigations, and assessments
d. Records pertaining to a description of equipment and facilities design (electrical systems, structures and components, cooling water systems, containment/confinement systems, ventilation system, etc.)

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Final Report

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10. SUPPLEMENTARY NOTES

11. ABSTRACT *(200 words or less)*

This NUREG provides guidance to the NRC staff reviewers in the Office of Nuclear Material Safety and Safeguards for the performance of safety and environmental reviews of a Tank Waste Remediation System (TWRS) Facility under 10 CFR 70, as revised. The standard review plan (SRP) presented in this NUREG ensures the quality, uniformity, stability, and predictability of staff reviews. It presents a defined basis from which to evaluate proposed changes in the scope and requirements of the staff reviews. The SRP makes information about review acceptance criteria readily available to interested members of the public and the regulated industry. Each SRP section addresses the regulations pertinent to specific technical matters, the acceptance criteria used by the staff, how the review is accomplished, and the conclusions that are appropriate for the Safety Evaluation Report (SER).

12. KEY WORDS/DESCRIPTORS *(List words or phrases that will assist researchers in locating the report.)*

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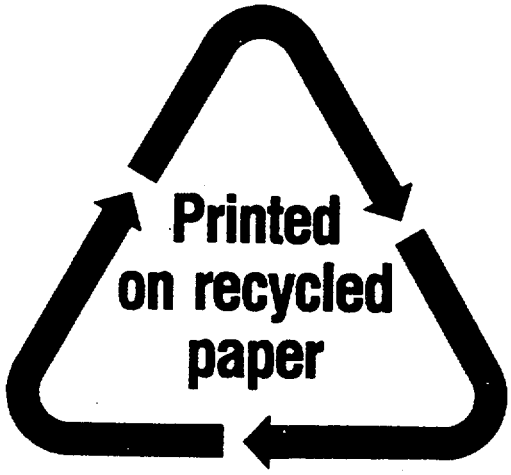

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