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DRAFT REGULATORY GUIDE

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(Proposed Revision 1 of Regulatory Guide 1.183, dated July 2000)

ALTERNATIVE RADIOLOGICAL SOURCE TERMS FOR EVALUATING DESIGN BASIS ACCIDENTS AT NUCLEAR POWER REACTORS

A. INTRODUCTION

This regulatory guide describes a method that the staff of the U.S. Nuclear Regulatory Commission (NRC) considers acceptable in complying with alternative source term (AST) regulations for design basis accident (DBA) dose consequence analysis. This guidance for light-water reactor (LWR) designs includes the scope, nature, and documentation of associated analyses, evaluations; consideration of impacts on analyzed risk; and content of submittals. This guide establishes the AST based on NUREG-1465, "Accident Source Terms for Light-Water Nuclear Power Plants" (Ref. 1), and identifies significant attributes of other accident source terms that may be acceptable. This guide also identifies acceptable radiological analysis assumptions for use in conjunction with the AST. In some cases, unusual site characteristics, plant design features, or other factors may require different assumptions which will be considered by the staff on an individual case basis.

As required by Title 10 of the *Code of Federal Regulations*, Section 50.34, "Contents of Applications; Technical Information" (10 CFR 50.34), each applicant for a construction permit or operating license must provide an analysis and evaluation of the design and performance of structures, systems, and components of the facility with the objective of assessing the risk to public health and safety

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This regulatory guide is being issued in draft form to involve the public in the early stages of the development of a regulatory position in this area. It has not received final staff review or approval and does not represent an official NRC final staff position.

Public comments are being solicited on this draft guide (including any implementation schedule) and its associated regulatory analysis or value/impact statement. Comments should be accompanied by appropriate supporting data. Written comments may be submitted to the Rulemaking, Directives, and Editing Branch, Office of Administration, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001; e-mailed to nrcrep.resource@nrc.gov; submitted through the NRC's interactive rulemaking Web page at http://www.nrc.gov; or faxed to (301) 492-3446. Copies of comments received may be examined at the NRC's Public Document Room, 11555 Rockville Pike, Rockville, MD. Comments will be most helpful if received by January 13, 2010.

Electronic copies of this draft regulatory guide are available through the NRC's interactive rulemaking Web page (see above); the NRC's public Web site under Draft Regulatory Guides in the Regulatory Guides document collection of the NRC's Electronic Reading Room at http://www.nrc.gov/reading-rm/doc-collections/; and the NRC's Agencywide Documents Access and Management System (ADAMS) at http://www.nrc.gov/reading-rm/adams.html, under Accession No. ML090960464.

resulting from operation of the facility. Applicants are also required by 10 CFR 50.34 and 10 CFR Part 52, “Licenses, Certifications, and Approvals for Nuclear Power Plants,” to provide an analysis of the proposed site. Sections 52.47 and 52.79, “Contents of applications; technical information in final safety analysis report,” of 10 CFR Part 52 also require standard design certification and combined license applicants to provide a similar analysis and evaluation.

For stationary power reactor applications before January 10, 1997, the criteria for evaluating the radiological aspects of the proposed site appear in 10 CFR 100.11,[1] “Determination of Exclusion Area, Low Population Zone, and Population Center Distance.” A footnote to 10 CFR 100.11 states that the fission product release assumed in these evaluations should be based upon a major accident involving substantial meltdown of the core with subsequent release of appreciable quantities of fission products.

Technical Information Document (TID) 14844, “Calculation of Distance Factors for Power and Test Reactor Sites” (Ref. 2), is cited in 10 CFR 100.11, “Determination of Exclusion Area, Low Population Zone, and Population Center Distance,” as a source of further guidance on these analyses. Although initially used only for siting evaluations, the TID-14844 source term has been used in other design basis applications, such as environmental qualification (EQ) of equipment under 10 CFR 50.49, “Environmental Qualification of Electric Equipment Important to Safety for Nuclear Power Plants,” and in some requirements stated in NUREG-0737, “Clarification of TMI Action Plan Requirements” (Ref. 3).

The facility final safety analysis report (FSAR) documents the analyses and evaluations required by 10 CFR 50.34 and 10 CFR Part 52. Fundamental assumptions that are design inputs, including the source term, are to be included in the FSAR and become part of the facility design basis.[2]

Since the publication of TID-14844, significant advances have been made in understanding the timing, magnitude, and chemical form of fission product releases from severe nuclear power plant accidents. A holder of an operating license issued prior to January 10, 1997, or a holder of a renewed license under 10 CFR Part 54, “Requirements for Renewal of Operating Licenses for Nuclear Power Plants,” whose initial operating license was issued prior to January 10, 1997, can, in accordance with 10 CFR 50.67, “Accident Source Term,” voluntarily revise the accident source term used in design basis radiological consequence analyses.

In general, information provided by regulatory guides is reflected in NUREG-0800, “Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants” (hereafter referred to as the SRP) (Ref. 4). The NRC staff uses the SRP to review applications to construct and operate nuclear power plants. This regulatory guide applies to Chapter 15.0.1 of the SRP for operating reactors and Chapter 15.0.3 of the SRP for advanced LWRs.

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1 Per 10 CFR 100.21, the NRC requires applicants for a construction permit or an operating license who applied on or after January 10, 1997, to meet radiological criteria provided in 10 CFR 50.34. The NRC requires applicants for an early site permit, standard design certification, combined license, standard design approval or manufacturing license under 10 CFR 52 to meet radiological criteria provided in the applicable section of Part 52.

2 As defined in 10 CFR 50.2, “Definitions,” “design bases” means information that identifies the specific functions to be performed by a structure, system, or component of a facility and the specific values or ranges of values chosen for controlling parameters as reference bounds for design. These values may be (1) restraints derived from generally accepted “state-of-the-art” practices for achieving functional goals, or (2) requirements derived from analysis (based on calculation or experiments or both) of the effects of a postulated accident for which a structure, system, or component must meet its functional goals. The NRC considers the accident source term to be an integral part of the design basis because it sets forth specific values (or a range of values) for controlling parameters that constitute reference bounds for design.

The NRC issues regulatory guides to describe to the public methods that the staff considers acceptable for use in implementing specific parts of the agency's regulations, to explain techniques that the staff uses in evaluating specific problems or postulated accidents, and to provide guidance to applicants. Regulatory guides are not substitutes for regulations and compliance with them is not required.

This regulatory guide contains information collection requirements covered by 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities," that the Office of Management and Budget (OMB) approved under OMB control number 3150-0011. The NRC may neither conduct nor sponsor, and a person is not required to respond to, an information collection request or requirement unless the requesting document displays a currently valid OMB control number.

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B. DISCUSSION

An accident source term is intended to be representative of a major accident involving significant core damage, not exceeded by that from any accident considered credible, and is typically postulated to occur in conjunction with a large loss-of-coolant accident (LOCA). Although the LOCA is typically the maximum credible accident, the NRC staff experience in reviewing license applications has indicated the need to consider other accident sequences of lesser consequence but higher probability of occurrence. Facility-analyzed DBAs are not intended to be actual event sequences; rather, they are intended to be surrogates to enable deterministic evaluation of the response of engineered safety features (ESFs). These accident analyses are intentionally conservative to compensate for known uncertainties in accident progression, fission product transport, and atmospheric dispersion.

Probabilistic risk assessments (PRAs) can provide useful insights into system performance and suggest changes in how the desired defense in depth is achieved. However, defense in depth continues to be an effective way to account for uncertainties in equipment and human performance. The NRC's policy statement on the use of PRA methods (Ref. 5) calls for the use of PRA technology in all regulatory matters in a manner that complements the NRC's deterministic approach and supports the traditional defense-in-depth philosophy.

In 1995, the NRC published NUREG-1465 (Ref. 1) which provides estimates of the accident source term that are more physically based and that could be applied to the design of advanced LWRs. NUREG-1465 presents a representative accident source term for a boiling-water reactor (BWR) and for a pressurized-water reactor (PWR). These source terms are characterized by the composition and magnitude of the radioactive material, the chemical and physical properties of the material, and the timing of the release to the containment. The NRC staff considered the applicability of the revised source terms to operating reactors and determined that the current analytical approach based on the TID-14844 source term would continue to be adequate to protect public health and safety. Operating reactors licensed under that approach would not be required to re-analyze accidents using the revised source terms. However, the NRC staff determined that some operating reactor licensees might request to use an AST in analyses to support cost-beneficial licensing actions.

The NRC staff, therefore, initiated several actions to provide a regulatory basis for operating reactors to use an AST[3] in design basis analyses. These initiatives resulted in the development and issuance of 10 CFR 50.67 and this regulatory guide.

A series of regulatory guides and SRP chapters describe the NRC's traditional methods for calculating the radiological consequences of DBAs. The staff developed that guidance to be consistent with the TID-14844 source term and the whole body and thyroid dose guidelines stated in 10 CFR 100.11. Many of those analysis assumptions and methods are inconsistent with the AST and with the total effective dose equivalent (TEDE) criteria provided in 10 CFR 50.34, 10 CFR Part 52, and 10 CFR 50.67. This guide provides assumptions and methods that are acceptable to the NRC staff for performing design basis radiological analyses using an AST. This guidance supersedes corresponding radiological analysis assumptions provided in other regulatory guides and SRP chapters when used in conjunction with an approved AST and the TEDE criteria provided in 10 CFR 50.34, 10 CFR Part 52, and 10 CFR 50.67. The affected guides will not be withdrawn because the guidance still applies when an AST is not used. Specifically, the affected regulatory guides include the following:

3. The NUREG-1465 source terms have often been referred to as the "revised source terms." In recognition that additional source terms may be identified in the future, 10 CFR 50.67 addresses "alternative source terms." This regulatory guide endorses a source term derived from NUREG-1465 and provides guidance on the acceptable attributes of other ASTs.

- Regulatory Guide 1.3, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident for Boiling Water Reactors" (Ref. 6)

- Regulatory Guide 1.4, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident for Pressurized Water Reactors" (Ref. 7)

- Regulatory Guide 1.5, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Steam Line Break Accident for Boiling Water Reactors" (Ref. 8)

- Regulatory Guide 1.25, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Fuel Handling Accident in the Fuel Handling and Storage Facility for Boiling and Pressurized Water Reactors" (Ref. 9)

- Regulatory Guide 1.77, "Assumptions Used for Evaluating a Control Rod Ejection Accident for Pressurized Water Reactors" (Ref. 10)

For plants licensed using the TID-14844 source term that have not implemented an AST for EQ, the guidance in Regulatory Guide 1.89, Revision 1, "Environmental Qualification of Certain Electric Equipment Important to Safety for Nuclear Power Plant, Revision 1" (Ref. 11) remains valid for the determination of integrated doses for EQ purposes.

This guide primarily addresses DBAs, such as those addressed typically in Chapter 15 of FSARs. This guide does not address all areas of potentially significant risk. Although this guide addresses fuel handling accidents, other events that could occur during shutdown operations are not currently addressed. The NRC staff has several ongoing initiatives involving risks of shutdown operations, extended burnup fuels, and risk-informing current regulations. The information in this guide may be revised in the future as NRC staff evaluations are completed and regulatory decisions on these issues are made.

C. REGULATORY POSITION

1. Implementation of Accident Source Term

1.1 Generic Considerations

As used in this guide, the AST is an accident source term that is derived principally from NUREG-1465 and differs from the TID-14844 source term used in the original design and licensing of operating reactor facilities. The AST has been approved for use in advanced LWRs under 10 CFR Part 52 and for operating reactors under 10 CFR 50.34 and 10 CFR 50.67. This guide identifies an AST that is acceptable to the NRC staff and identifies significant characteristics of other source terms that may be found acceptable. While the NRC staff recognizes several potential uses of an AST, it is not possible to foresee all possible uses. The NRC staff will allow licensees to pursue technically justifiable uses of the ASTs in the most flexible manner so long as they are compatible with maintaining a clear, logical, and consistent design basis. The NRC staff will approve these license amendment requests if the facility, as modified, will continue to provide sufficient safety margins with adequate defense in depth to address unanticipated events and to compensate for uncertainties in accident progression and analysis assumptions and parameter inputs.

1.1.1 Safety Margins

Licensees should evaluate the proposed uses of this guide and the associated proposed facility modifications and changes to procedures to determine whether the proposed changes are consistent with the principle that sufficient safety margins are maintained, including a margin to account for analysis uncertainties. The safety margins are products of specific values and limits contained in the technical specifications (which cannot be changed without NRC approval) and other values, such as assumed accident or transient initial conditions or assumed safety system response times. Changes, or the net effect of multiple changes, that result in a reduction in safety margins may require prior NRC approval. Once the staff has approved the initial AST implementation and it has become part of the facility design basis, licensees may use 10 CFR 50.59, "Changes, Tests and Experiments," and its supporting guidance to assess facility modifications and changes to procedures that are described in the updated FSAR.

1.1.2 Defense in Depth

Licensees should evaluate the proposed uses of an AST and the associated proposed facility modifications and changes to procedures to determine whether the proposed changes are consistent with the principle that adequate defense in depth is maintained to compensate for uncertainties in accident progression and analysis data. Consistency with the defense-in-depth philosophy is maintained if system redundancy, independence, and diversity are preserved commensurate with the expected frequency, consequences of challenges to the system, and uncertainties. For facilities to which the general design criteria apply, compliance with these criteria (see Appendix A, "General Design Criteria for Nuclear Power Plants," to 10 CFR Part 50) is essential. Modifications proposed for the facility generally should not create a need for compensatory programmatic activities (e.g., reliance on manual operator actions, use of potassium iodide as a prophylactic drug) or self-contained breathing apparatus.

Licensees should evaluate proposed modifications that seek to downgrade or remove required engineered safeguards equipment to confirm that the modification does not invalidate assumptions made in facility PRAs and does not adversely impact the facility's severe accident management program.

1.1.3 Integrity of Facility Design Basis

The DBA source term used for dose consequence analyses is a fundamental assumption upon which a significant portion of the facility design is based. Additionally, many aspects of an operating reactor facility are derived from the radiological design analyses that incorporated the TID-14844 accident source term. Although a complete reassessment of all facility radiological analyses would be desirable, the NRC staff determined that recalculation of all design analyses for operating reactors would generally not be necessary. Regulatory Position 1.3 provides guidance on which analyses should be updated as part of the AST implementation submittal and which may need to be updated in the future as additional modifications are performed.

This approach for operating reactors creates two tiers of analyses-one based on the previous TID-14844 source term and one based on an AST. The radiological acceptance criteria would also differ from some analyses based on whole body and thyroid criteria and some based on TEDE criteria. Full implementation of the AST revises the plant licensing basis to specify the AST in place of the previous TID-14844 accident source term and establishes the TEDE dose as the new acceptance criteria. Selective implementation of the AST also revises the plant licensing basis and may establish the TEDE dose as the new acceptance criteria. Selective implementation differs from full implementation only in the scope of the change. In either case, the facility design bases should clearly indicate that the source term assumptions and radiological criteria in these affected analyses have been superseded and that future revisions of these analyses, if any, will use the updated approved assumptions and criteria.

Radiological analyses generally should be based on assumptions and inputs that are consistent with corresponding data used in other design basis safety analyses unless these data would result in nonconservative results or otherwise conflict with regulatory guidance.

1.1.4 Emergency Preparedness Applications

The regulations in 10 CFR 50.47, "Emergency Plans," include the requirements for emergency preparedness at nuclear power plants. Additional requirements are set forth in Appendix E, "Emergency Planning and Preparedness for Production and Utilization Facilities," to 10 CFR Part 50. NUREG-0396, "Planning Basis for the Development of State and Local Government Radiological Emergency Response Plans in Support of Light Water Nuclear Power Plants"[4] (Ref. 12), includes the planning basis for many of these requirements. This joint effort by the U.S. Environmental Protection Agency (EPA) and the NRC considered the principal characteristics (such as nuclides released and distances) likely to be involved for a spectrum of design basis and severe (core melt) accidents. No single accident scenario is the basis of the required preparedness. The objective of the planning is to provide public protection that would encompass a wide spectrum of possible events with a sufficient basis for extension of response efforts for unanticipated events. The NRC and EPA issued these requirements after a long period of involvement by numerous stakeholders, including the Federal Emergency Management Agency, other Federal agencies, local and State governments (and in some cases, foreign governments), private citizens, utilities, and industry groups.

Although the NRC based the AST provided in this guide on a limited spectrum of severe accidents, the particular characteristics are tailored specifically for DBA analysis use. The AST is not representative of the wide spectrum of possible events that make up the planning basis of emergency preparedness. Therefore, the AST is insufficient by itself as a basis for requesting relief from the emergency preparedness requirements of 10 CFR 50.47 and Appendix E to 10 CFR Part 50.

This guidance does not, however, preclude the appropriate use of the insights of the AST in establishing emergency response procedures such as those associated with emergency dose projections, protective measures, and severe accident management guides.

1.1.5 Applicability to 10 CFR Part 52

The NRC originally created Regulatory Guide 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," for use by existing nuclear power reactors to satisfy regulations under 10 CFR 50.34 and 10 CFR 50.67. Draft Regulatory Guide DG-1199, the proposed revision of Regulatory Guide 1.183, extends the applicability of the proposed regulatory guide for use in satisfying the radiological dose analysis requirements contained in 10 CFR Part 52 for advanced light-water reactor design and siting. For applicants and licensees that voluntarily use Regulatory Guide 1.183 to meet the requirements of 10 CFR 52, "Licenses, Certifications, and Approvals for Nuclear Power Plants," regarding radiological consequences analyses, the staff will use, where applicable, the methodology and assumptions stated in this draft revision to Regulatory Guide 1.183.

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4. NUREG-0654, Revision 1, "Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants," issued November 1980 (Ref. 13), also addresses this planning basis.

1.2 Scope of Implementation

The AST described in this guide is characterized by radionuclide composition and magnitude, chemical and physical form of the radionuclides, and the timing of the release of these radionuclides. The accident source term is a fundamental assumption upon which a large portion of the facility design is based.

For operating reactors for which 10 CFR 50.67 is applicable, a complete implementation of an AST would upgrade all existing radiological analyses and would consider the impact of all five characteristics of a source term as defined in 10 CFR 50.2, "Definitions." However, the NRC staff has determined that there could be implementations for which this level of reanalysis may not be necessary. For holders of operating licenses, as defined in the applicability section of 10 CFR 50.67, two categories of AST implementation are defined: full and selective. These are described in Regulatory Positions 1.2.1 and 1.2.2 below.

For new reactors applicants (e.g. 10 CFR Part 52, 10 CFR 100.21) implementation of an AST should consider all characteristics of a source term as defined in 10 CFR 50.2 and detailed in Regulatory Position 3. Full and selective implementations, as used in the regulatory positions that follow, are not applicable to new reactor applicants.

1.2.1 Full Implementation

Full implementation is a modification of the facility design basis that addresses all characteristics of the AST, specifically, composition and magnitude of the radioactive material, its chemical and physical form, and the timing of its release. Full implementation revises the plant licensing basis to specify the AST in place of the previous accident source term and establishes the TEDE dose as the new acceptance criteria. This applies not only to the analyses performed in the application (which may only include a subset of the plant analyses), but also to all future design basis analyses. At a minimum, for full implementations the DBA LOCA must be reanalyzed using the guidance in Appendix A to this guide. In performing this analysis, licensees should evaluate the spectrum of DBA LOCAs in order to ensure the bounding LOCA is identified and evaluated from a dose consequences perspective. Regulatory Position 1.3 of this guide provides additional guidance on the analysis. Since the AST and TEDE criteria would become part of the facility design basis, new applications of the AST would not require prior NRC approval unless stipulated by 10 CFR 50.59 or unless the new application involved a change to a technical specification. However, a change from an approved AST to a different AST that is not approved for use at that facility would require a license amendment under 10 CFR 50.67.

1.2.2 Selective Implementation

Selective implementation is a modification of the facility design basis that (1) is based on one or more of the characteristics of the AST, or (2) entails reevaluation of a limited subset of the design basis radiological analyses. The NRC staff will allow licensees to have flexibility in adopting technically justified selective implementations, provided a clear, logical, and consistent design basis is maintained. An example of an application of selective implementation would be one in which a licensee desires to use the release timing insights of the AST to increase the required closure time for a containment isolation valve by a small amount. Another example would be a request to remove the charcoal filter media from the spent fuel building ventilation exhaust. In the latter example, the licensee may only need to reanalyze DBAs that credited the iodine removal by the charcoal media. Regulatory Position 1.3 of this guide provides additional analysis guidance. NRC approval for the AST (and the TEDE dose criterion) will be limited to the particular selective implementation proposed by the licensee. The licensee would be able to make subsequent modifications to the facility and changes to procedures based on the selected AST characteristics incorporated into the design basis under the provisions of 10 CFR 50.59. However, use of other characteristics of an AST or use of TEDE criteria that are not part of the approved design basis, and changes to previously approved AST characteristics, would require prior staff approval under 10 CFR 50.67. As an example, a licensee with an implementation involving only timing, such as relaxed closure time on isolation valves, could not use 10 CFR 50.59 as a mechanism to implement a modification involving a reanalysis of the DBA LOCA. However, the licensee could extend use of the timing characteristic to adjust the closure time on isolation valves not included in the original approval.

1.3 Scope of Required Analyses

1.3.1 Design basis Radiological Analyses

There are several regulatory requirements for which compliance is demonstrated, in part, by the evaluation of the radiological consequences of DBAs. A plant's licensing bases may include, but are not limited to, the following:

- EQ of equipment (10 CFR 50.49)
- control room habitability (General Design Criterion (GDC) 19, "Control Room," of Appendix A to 10 CFR Part 50)
- emergency response facility habitability (Paragraph IV.E.8 of Appendix E to 10 CFR Part 50)
- alternative source term (10 CFR 50.67)
- environmental reports (10 CFR Part 51, "Environmental Protection Regulations for Domestic Licensing and Related Regulatory Functions")
- facility siting (10 CFR 100.11)[5]
- early site permits, standard design certifications, combined licenses (10 CFR Part 52)

There may be other areas in which the technical specification bases and various licensee commitments refer to evaluations that use an AST. A plant's licensing bases may include, but are not limited to, the following sections of NUREG-0737 (Ref. 3):

- postaccident access shielding (II.B.2)
- postaccident sampling capability (II.B.3)
- accident monitoring instrumentation (II.F.1)
- leakage control (III.D.1.1)
- emergency response facilities (III.A.1.2)
- control room habitability (III.D.3.4)

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5 For licensees that have implemented an AST, the dose guidelines of 10 CFR 50.67 supersede those of 10 CFR 100.11.

1.3.2 Reanalysis Guidance

Any full or selective implementation of an AST, and any associated facility modification should be supported by evaluations of all significant radiological and nonradiological impacts of the proposed actions. This evaluation should consider the impact of the proposed changes on the facility's compliance with the regulations and commitments listed above as well as any other facility-specific requirements. These impacts may be caused by (1) the associated facility modifications, or (2) the differences in the AST characteristics. The scope and extent of the reevaluation will necessarily be a function of the specific proposed facility modification[6] and whether a full or selective implementation is being pursued. The NRC staff does not expect a complete recalculation of all facility radiological analyses, but does expect licensees to evaluate all impacts of the proposed changes and to update the affected analyses and design bases appropriately. The NRC considers an analysis to be affected if the proposed modification changes one or more assumptions or inputs used in that analysis such that the results, or the conclusions drawn on those results, are no longer valid. The licensees may use NRC-approved generic analyses, such as those performed by owner groups or vendor topical reports, provided that the licensee justifies the applicability of the generic conclusions to the specific facility and implementation. Sensitivity analyses, discussed below, may also be an option. If affected design basis analyses are to be recalculated, the licensee should update all affected assumptions and inputs and address all selected characteristics of the AST and the TEDE criteria. Any license amendment request should describe the licensee's reanalysis effort and provide statements regarding the acceptability of the proposed implementation, including modifications, against each of the applicable analysis requirements and commitments identified in Regulatory Position 1.3.1 of this guide.

The NRC staff has evaluated the impact of the AST on three representative operating reactors (Ref. 14). This evaluation determined that radiological analysis results based on the TID-14844 source term assumptions (Ref. 2) and the whole body and thyroid methodology generally bound the results from analyses based on the AST and TEDE methodology. Licensees may use the applicable conclusions of this evaluation in addressing the impact of the AST on design basis radiological analyses. However, this does not exempt the licensee from evaluating the remaining radiological and nonradiological impacts of the AST implementation and the impacts of the associated plant modifications. For example, a selective implementation based on the timing insights of the AST may change the required isolation time for the containment purge dampers from 2.5 seconds to 5.0 seconds. This application might be acceptable without dose calculations. However, the licensee may need to evaluate the ability of the damper to close against increased containment pressure or the ability of ductwork downstream of the dampers to withstand increased stresses.

For full implementation, the licensee should perform a complete DBA LOCA analysis, as described in Appendix A to this guide, at a minimum. The licensee should update other design basis analyses in accordance with the guidance in this section.

A selective implementation of an AST and any associated facility modification based on the AST should evaluate all of the radiological and nonradiological impacts of the proposed actions as they apply to the particular implementation. The licensee should update design basis analyses in accordance with the guidance in this section. There is no minimum requirement that a DBA LOCA analysis be performed. The analyses performed need to address all impacts of the proposed modification, the selected

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[6] For example, a proposed modification to change the timing of a containment isolation valve from 2.5 seconds to 5.0 seconds might be acceptable without any dose calculations. However, a proposed modification that would delay containment spray actuation could involve recalculation of DBA LOCA doses, reassessment of the containment pressure and temperature transient, recalculation of sump pH, reassessment of the emergency diesel generator loading sequence, integrated doses to equipment in the containment, and more.

characteristics of the AST, and, if dose calculations are performed, the TEDE criteria. For selective implementations based on the timing characteristic of the AST (e.g., change in the closure timing of a containment isolation valve), reanalysis of radiological calculations may not be necessary if the modified elapsed time remains a fraction (e.g., 25 percent) of the time between accident initiation and the onset of the gap release phase. Longer time delays may be considered on an individual basis. For longer time delays, evaluation of the radiological consequences and other impacts of the delay, such as blockage by debris in sump water, may be necessary. If affected design basis analyses are to be recalculated, all affected assumptions and inputs should be updated and all selected characteristics of the AST and the TEDE criteria should be addressed.

1.3.3 Use of Sensitivity or Scoping Analyses

It may be possible to demonstrate by sensitivity or scoping evaluations that existing analyses have sufficient margin and need not be recalculated. As used in this guide, a sensitivity analysis is an evaluation that considers how the overall results vary as an input parameter (in this case, AST characteristics) is varied for a given set of assumptions. A scoping analysis is a brief evaluation that uses conservative, simple methods to show that the results of the analysis bound those obtainable from a more complete treatment. Sensitivity analyses are particularly applicable to suites of calculations that address diverse components or plant areas but are otherwise largely based on generic assumptions and inputs. Such cases might include postaccident vital area access dose calculations, shielding calculations, and equipment EQ (integrated dose). It may be possible to identify a bounding case, reanalyze that case, and use the results to draw conclusions regarding the remainder of the analyses. It may also be possible to show that for some analyses the whole body and thyroid doses determined with the previous source term would bound the TEDE obtained using the AST. Where present, arbitrary "designer margins" may be adequate to bound any impact of the AST and TEDE criteria. If sensitivity or scoping analyses are used, the license amendment request should include a discussion of the analyses performed and the conclusions drawn. Scoping or sensitivity analyses should not constitute a significant part of the evaluations for the design basis exclusion area boundary (EAB), low population zone (LPZ), or control room dose unless a clear and defensible basis exists for doing so.

1.3.4 Updating Analyses Following Implementation

Full implementation of the AST replaces the previous accident source term with the approved AST and the TEDE criteria for all design basis radiological analyses. The implementation may have been supported in part by sensitivity or scoping analyses that concluded that many of the design basis radiological analyses would remain bounding for the AST and the TEDE criteria and would not require updating. After the implementation is complete, there may be a subsequent need (e.g., a planned facility modification) to revise these analyses or to perform new analyses. For these recalculations, the NRC staff expects that all characteristics of the AST and the TEDE criteria incorporated into the design basis will be addressed in all affected analyses on an individual as-needed basis. Reevaluation using the previously approved source term may not be appropriate. Since the AST and the TEDE criteria are part of the approved design basis for the facility, use of the AST and TEDE criteria in new applications at the facility does not constitute a change in analysis methodology that would require NRC approval.[7]

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7 In performing screenings and evaluations pursuant to 10 CFR 50.59, it may be necessary to compare dose results expressed in terms of whole body and thyroid with new results expressed in terms of TEDE. In these cases, the previous thyroid dose should be multiplied by 0.03 and the product added to the whole body dose. The result is then compared to the TEDE result in the screenings and evaluations. This change in dose methodology is not considered a change in the method of evaluation if the licensee was previously authorized to use an AST and the TEDE criteria under 10 CFR 50.67.

This guidance is also applicable to selective implementations to the extent that the affected analyses are within the scope of the approved implementation as described in the facility design basis. In these cases, the updated analyses should consider the characteristics of the AST and TEDE criteria identified in the facility design basis. Use of other characteristics of the AST or TEDE criteria that are not part of the approved design basis, and changes to previously approved AST characteristics, requires prior NRC staff approval under 10 CFR 50.67.

1.3.5 Equipment Environmental Qualification

Current EQ analyses may be impacted by a proposed plant modification associated with the AST implementation. The licensee should update EQ analyses that have assumptions or inputs affected by the plant modification to address these impacts.

For new facilities that are proposing to implement an AST and have EQ analyses impacted by a proposed plant modification associated with the AST implementation, the guidance that is being developed in a draft guide, Draft Regulatory Guide DG-1239, Environmental Qualification of Certain Electric Equipment Important to Safety for Nuclear Power Plant," which will be published soon, should be used.

1.4 Risk Implications

This guide provides regulatory assumptions that licensees should use in their calculation of the radiological consequences of DBAs. These assumptions have no direct influence on the probability of the design basis initiator. These analysis assumptions cannot increase the core damage frequency (CDF) or the large early release frequency (LERF). However, facility modifications made possible by the AST could have an impact on risk. If the proposed implementation of the AST involves changes to the facility design that would invalidate assumptions made in the facility's PRA, the licensee should evaluate the impact on the existing PRAs.

Consideration should be given to the risk impact of proposed implementations that seek to remove or downgrade the performance of previously required engineered safeguards equipment on the basis of the reduced postulated doses. The NRC staff may request risk information if there is a reason to question adequate protection of public health and safety.

The licensee may elect to use risk insights in support of proposed changes to the design basis that are not addressed in currently approved NRC staff positions. For guidance, refer to Regulatory Guide 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis" (Ref. 15).

1.5 Submittal Requirements

According to 10 CFR 50.90, "Application for Amendment of License, Construction Permit, or Early Site Permit," an application for an amendment must fully describe the changes desired and should follow, as far as applicable, the form prescribed for original applications. Regulatory Guide 1.70, "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants (LWR Edition)" (Ref. 16), provides additional guidance. The NRC staff's finding as to whether an amendment is to be approved or rejected is partially based on the licensee's analyses, since it is these analyses that will become part of the design and licensing basis of the facility. The NRC staff accomplishes these reviews by evaluating the information submitted in the amendment request against the current plant design as documented in the FSAR, staff safety evaluation reports, regulatory guidance, other licensee commitments, and staff experience gained in approving similar requests for other plants. The NRC staff's assessment may include performance of independent analyses to confirm the licensee's conclusion. Licensees should expect an NRC staff effort to resolve critical differences in analysis assumptions, inputs, and methods used by the licensee and those deemed acceptable to the NRC staff.[8]

The amendment request should describe the licensee's analyses of the radiological and nonradiological impacts of the proposed modification in sufficient detail to support review by the NRC staff. Licensees should ensure that adequate information, including analysis assumptions, inputs, and methods, are presented in the submittal to support the staff's assessment. Consistent with 10 CFR 50.90, "Application for Amendment of License, Construction Permit or Early Site Permit," the licensee shall, as far as applicable, follow the form prescribed for original applications. Typically, original applications included FSAR pages and technical specifications. Licensees should submit affected FSAR pages and technical specifications annotated with changes that reflect the revised analyses. Additionally, the NRC staff recommends that licensees submit the actual calculation documentation. In lieu of submitting marked up FSAR pages, licensees should include a detailed listing, preferably in tabular format, of all changes and associated justification being proposed between the current facility licensing basis and the requested license amendment.

If the licensee has used a current approved version of an NRC-sponsored computer code, the NRC staff review can be made more efficient if the licensee identifies the code used and submits the inputs that the licensee used in the calculations made with that code. In many cases, this will reduce the need for NRC staff confirmatory analyses. This recommendation does not constitute a requirement that the licensee use NRC-sponsored computer codes.

Applications for licenses, certifications and approvals under Part 52 have similar requirements as stated above for license amendment submittals. Regulatory Guide 1.206, "Combined License Applications for Nuclear Power Plants (LWR Edition)" (Ref 17), provides additional guidance on combined license applications.

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8 The analyses required by 10 CFR 50.67 are important, and 10 CFR 50.34, "Contents of Construction Permit and Operating License Applications; Technical Information," requires reanalyses of the design basis safety analyses and evaluations; they are considered to be a significant input to the evaluations required by 10 CFR 50.92, "Issuance of Amendment," or 10 CFR 50.59.

1.6 Final Safety Analysis Report Requirements

The regulations in 10 CFR 50.71, "Maintenance of Records, Making of Reports," include the requirements for updating the facility's FSAR. Specifically, 10 CFR 50.71(e) requires that the FSAR be updated to include all changes made in the facility or procedures described in the FSAR and all safety analyses and evaluations performed by the licensee in support of approved requests for license amendments or in support of conclusions that changes did not require a license amendment in accordance with 10 CFR 50.59. The analyses required by 10 CFR 50.67 are subject to this requirement. The licensee should update the affected radiological analysis descriptions in the FSAR to reflect the design basis changes to the methodology and input. The analysis descriptions should contain sufficient detail to identify the methodologies used, significant assumptions and inputs, and numeric results. Regulatory Guide 1.70 (Ref. 16) provides additional guidance. The licensee should remove the descriptions of superseded analyses from the FSAR in the interest of maintaining a clear design basis.

2. Attributes of an Acceptable Accident Source Term

The NRC did not set forth an acceptable AST in 10 CFR 50.67. Regulatory Position 3 of this guide identifies an AST that is acceptable to the NRC staff for use in new power reactor applications and operating power reactors. The NRC, its contractors, various national laboratories, peer reviewers, and others expended substantial effort in performing severe accident research and in developing the source terms provided in NUREG-1465 (Ref. 1). However, future research may identify opportunities for changes in these source terms. The NRC staff will consider applications for an AST different from that identified in this guide. However, the NRC staff does not expect to approve any source term that is not of the same level of quality as the source terms in NUREG-1465. To be considered acceptable, an AST must have the following attributes:

a. The AST must be based on major accidents hypothesized for the purposes of design analyses or consideration of possible accidental events that could result in hazards not exceeded by those from other accidents considered credible. The AST must address events that involve a substantial meltdown of the core with the subsequent release of appreciable quantities of fission products.

b. The AST must be expressed in terms of times and rates of appearance of radioactive fission products released into containment, the types and quantities of the radioactive species released, and the chemical forms of iodine released.

c. The AST must not be based upon a single accident scenario but instead must represent a spectrum of credible severe accident events. Risk insights may be used, not to select a single risk-significant accident, but rather to establish the range of events to be considered. However, risk insights alone are not an acceptable basis for excluding a particular event. Relevant insights from applicable severe accident research on the phenomenology of fission product release and transport behavior may be considered.

d. The AST must have a defensible technical basis supported by sufficient experimental and empirical data, be verified and validated, and be documented in a scrutable form that facilitates public review and discourse.

e. The AST must be peer-reviewed by appropriately qualified subject matter experts. The peer-review comments and their resolution should be part of the documentation supporting the AST.

3. Accident Source Term

This regulatory position provides an AST that is acceptable to the NRC staff. It provides guidance on the fission product inventory, release fractions, timing of the release phases, radionuclide composition, chemical form, and the fuel damage for LOCA and non-LOCA DBAs. The data in Regulatory Positions 3.1 through 3.5 are fundamental to the definition of an AST. Once approved, the AST assumptions or parameters specified in these positions become part of the facility's design basis. The NRC will evaluate deviations from this guidance against Regulatory Position 2. After the NRC staff has approved an implementation of an AST, subsequent changes to the AST will require NRC staff review under 10 CFR 50.67.

3.1 Fission Product Inventory

The inventory of fission products in the reactor core and available for release to the containment should be based on the maximum full-power operation of the core with, as a minimum, current licensed values for fuel enrichment, fuel burnup, and an assumed core power equal to the currently licensed rated thermal power times the emergency core cooling system (ECCS) evaluation uncertainty.[9] These parameters should be examined to maximize fission product inventory. The period of irradiation should be of sufficient duration to allow the activity of dose-significant radionuclides to reach equilibrium or to reach maximum values.[10] The core inventory should be determined using an appropriate isotope generation and depletion computer code. Core inventory factors (curies per megawatt thermal (Ci/MWt)) provided in TID-14844 (Ref. 2) and used in some analysis computer codes were derived for low burnup, low enrichment fuel and should not be used with higher burnup and higher enrichment fuels. The code should model the fuel geometries, material composition and burnup and the cross-section libraries used should be applicable to the projected fuel burnup.

For the DBA LOCA, all fuel assemblies in the core are assumed to be affected and the analysis should use the core average inventory. For DBA events that do not involve the entire core, the fission product inventory of each of the damaged fuel rods is determined by dividing the total core inventory by the number of fuel rods in the core. To account for differences in power level across the core, the analysis should apply the radial peaking factors from the facility's core operating limits report (COLR) or technical specifications in determining the inventory of the damaged rods.

The licensee should make no adjustment to the fission product inventory for events postulated to occur during power operations at less than full-rated power or those postulated to occur at the beginning of core life. For events postulated to occur while the facility is shut down (e.g., a fuel handling accident), the licensee may model radioactive decay from the time of shutdown.

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9 The uncertainty factor used in determining the core inventory should be that value provided in Appendix K, "ECCS Evaluation Models," to 10 CFR Part 50, which is typically 1.02. A value lower than 1.02, but not less than 1.00 (correlates to the licensed power level), may be used provided the proposed alternative value has been demonstrated to account for uncertainties caused by power level instrumentation error.

10 Note that for some radionuclides, such as cesium-137, equilibrium will not be reached before fuel offload. Thus, the maximum inventory at the end of life should be used.

3.2 Release Fractions[11]

Table 1 (for BWRs) and Table 2 (for PWRs) list the core inventory release fractions, by radionuclide groups, for the gap release and early in-vessel damage phases for DBA LOCAs and non-LOCA DBAs where the fuel is melted and the cladding is breached. These fractions are applied to the equilibrium core inventory described in Regulatory Position 3.1.

For non-LOCA DBAs, where only the cladding is postulated to be breached, Table 3 gives the fractions of the core inventory for the various radionuclides assumed to be in the gap for a fuel rod. The release fractions from Table 3 are used in conjunction with the calculated fission product inventory calculated with the maximum core radial peaking factor. The licensing basis of some facilities may include non-LOCA events that assume the release of the gap activity from the entire core (e.g., heavy load drop accident). For events involving the entire core, the core-average gap fractions of Tables 1 and 2 may be used and the radial peaking factor may be omitted.

For reactivity initiated accidents (RIAs) such as BWR control rod drop accident and PWR control rod ejection accident, the total fraction of fission products available for release equals the steady-state fission product gap inventory in Table 3 for a fuel rod plus the transient fission product release resulting from the rapid power excursion. Table 4 list the combined fission product inventory, by radionuclide groups, available for release for a fuel rod during a RIA. The transient fission product release component is presented as a function of increase in radial average fuel enthalpy (ΔH, cal/g). This component of the overall fission product inventory may be calculated separately for each axial node which experiences the RIA power pulse and then combined to yield the total transient fission product release for a particular fuel rod. The sum total of combined fission product inventories from each fuel rod predicted to experience cladding failure (all failure modes) should be used in the dose assessment.

The applicability of Table 3 non-LOCA fission product gap fractions is limited to fuel assemblies with peak rod power histories below the nodal power envelope depicted in Figure 1. Reference 18 documents the methods used to calculate the Table 3 and Table 4 fission product inventories, including application of modeling uncertainties.

The RIA combined release fractions provided in Table 3 and 4 of this guide are not applicable to fuel rods which experience fuel melting. The total fission product inventory for at-power RIA scenarios experiencing limited centerline fuel melting may be considered on a case-by-case basis.

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11 The NRC has determined the release fractions listed here to be acceptable for use with currently approved LWR fuel with a peak rod average burnup up to 62,000 megawatt days per metric ton of uranium (MWD/MTU) (PWR) and a peak pellet burnup up to 70,000 MWD/MTU (BWR). The data in this section are not applicable to cores containing mixed oxide (MOX) fuel.

Table 1 BWR Core Inventory Fraction Released into Containment Atmosphere

Group	Gap Release Phase	Early In-Vessel Phase	Total
Noble Gases	0.05	0.95	1.0
Halogens	0.05	0.25	0.3
Alkali Metals	0.05	0.20	0.25
Tellurium Metals	0.00	0.05	0.05
Ba, Sr	0.00	0.02	0.02
Noble Metals	0.00	0.0025	0.0025
Cerium Group	0.00	0.0005	0.0005
Lanthanides	0.00	0.0002	0.0002

Table 2 PWR Core Inventory Fraction Released into Containment Atmosphere

Group	Gap Release Phase	Early In-Vessel Phase	Total
Noble Gases	0.05	0.95	1.0
Halogens	0.05	0.35	0.4
Alkali Metals	0.05	0.25	0.3
Tellurium Metals	0.00	0.05	0.05
Ba, Sr	0.00	0.02	0.02
Noble Metals	0.00	0.0025	0.0025
Cerium Group	0.00	0.0005	0.0005
Lanthanides	0.00	0.0002	0.0002

Table 3 Non-LOCA Fraction of Fission Product Inventory in Gap

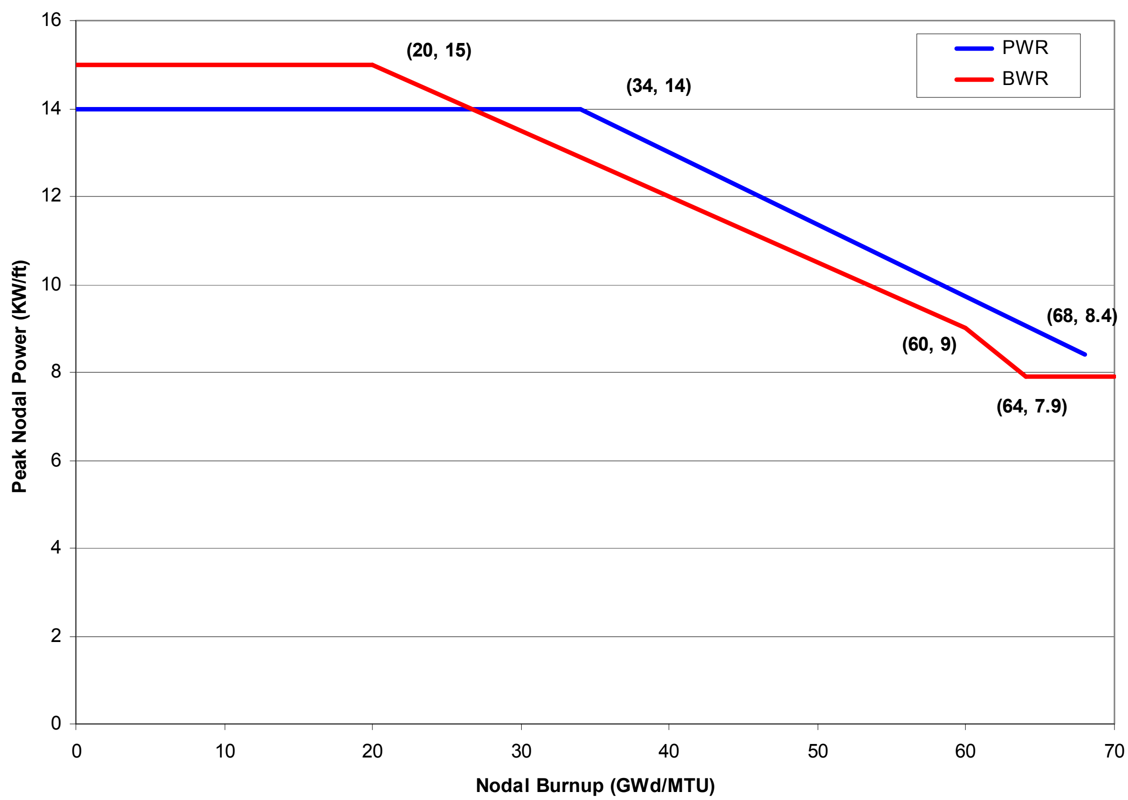
Group	Fraction
I-131	0.08
I-132	0.23
Kr-85	0.35
Other Noble Gases	0.04
Other Halogens	0.05
Alkali Metals	0.46

Table 4 Fraction of Fission Product Inventory Available for Release from

Reactivity Initiated Accidents

Group	Combined Release Fraction[12,13]
I-131	((0.08) + (0.00073 * ΔH))
I-132	((0.23) + (0.00073 * ΔH))
Kr-85	((0.35) + (0.0022 * ΔH))
Other Noble Gases	((0.04) + (0.00073 * ΔH))
Other Halogens	((0.05) + (0.00073 * ΔH))
Alkali Metals	((0.46) + (0.0031 * ΔH))

Figure 1 Maximum Allowable Power Operating Envelope for Non-LOCA Gap Fractions



3.3 Timing of Release Phases

Table 5 provides the onset and duration of each sequential release phase for LOCA DBAs. The specified onset is the time following the initiation of the accident (i.e., time = 0). The early in-vessel release phase immediately follows the gap release phase. The activity released from the core during each

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12 ΔH = increase in radial average fuel enthalpy, cal/g

13 This table is not applicable to fuel rods predicted to experience fuel melting.

release phase should be modeled as increasing in a linear fashion over the duration of the phase.[14] For non-LOCA DBAs in which fuel damage is projected, the release from the fuel gap and the fuel pellet should be assumed to occur instantaneously with the onset of the projected damage.

Table 5 LOCA Release Phases

	PWRs		BWRs	
Phase	**Onset**	**Duration**	**Onset**	**Duration**
Gap Release	0.5 minutes	0.5 hours	2 minutes	0.5 hours
Early In-Vessel	30.5 minutes	1.3 hours	32 minutes	1.5 hours

The early in-vessel release phase begins immediately following the gap release phase. For facilities licensed with leak-before-break methodology, the licensee may assume the onset of the gap release phase to be 10 minutes. A licensee may propose an alternative time for the onset of the gap release phase based on facility-specific calculations using suitable analysis codes or on an accepted topical report shown to be applicable to the specific facility. In the absence of approved alternatives, the licensee should use the gap release phase onsets in Table 5. Regardless of delays in the onset, the duration of the gap release phase is 0.5 hours.

3.4 Radionuclide Composition

Table 6 lists the elements in each radionuclide group that should be considered in design basis analyses.

Table 6 Radionuclide Groups

Group	**Elements**
Noble Gases	Xe, Kr
Halogens	I, Br
Alkali Metals	Cs, Rb
Tellurium Group	Te, Sb, Se
Barium, Strontium	Ba, Sr
Noble Metals	Ru, Rh, Pd, Mo, Tc, Co
Lanthanides	La, Zr, Nd, Eu, Nb, Pm, Pr, Sm, Y, Cm, Am
Cerium	Ce, Pu, Np

3.5 Chemical Form

Of the radioiodine released from the reactor coolant system (RCS) to the containment in a postulated accident, 95 percent of the iodine released should be assumed to be cesium iodide (CsI), 4.85 percent elemental iodine, and 0.15 percent organic iodide. This includes releases from the gap and the fuel pellets. With the exception of elemental and organic iodine and noble gases, fission products should be assumed to be in particulate form. The transport of these iodine species following release from the

[14] In lieu of treating the release in a linear ramp manner, the activity for each phase can be modeled as being released instantaneously at the start of that release phase (i.e., in step increases).

fuel may affect these assumed fractions. The accident-specific appendices to this regulatory guide provide additional details.

3.6 Fuel Damage in Non-LOCA Design Basis Accidents

The amount of fuel damage caused by non-LOCA design basis events should be analyzed to determine, for the case resulting in the highest radioactivity release, the fraction of the fuel that reaches or exceeds the initiation temperature of fuel melt and the fraction of fuel elements for which the fuel cladding is breached. Cladding failure mechanisms include high temperature failure modes (e.g., critical heat flux, local oxidation, and ballooning) and pellet-to-cladding mechanical interaction.

For the postulated main steamline break, steam generator tube rupture, and locked rotor accidents, the licensee should evaluate the amount of fuel damage assuming that the highest worth control rod is stuck at its fully withdrawn position.

Appendix B to this guide addresses the amount of fuel damage caused by a fuel handling accident.

4. Dose Calculational Methodology

The NRC staff has determined that there is an implied synergy between the ASTs and TEDE criteria and between the TID-14844 source terms and the whole body and thyroid dose criteria. Therefore, the staff does not expect to allow the TEDE criteria to be used with TID-14844 calculated results. The guidance provided in this regulatory position applies to all dose calculations performed with an AST pursuant to 10 CFR 50.67 and 10 CFR Part 52. It also provides guidance for the determination of control room and offsite doses and the control room and offsite dose acceptance criteria. Certain selective implementations may not require dose calculations, as described in Regulatory Position 1.3 of this guide.

4.1 Offsite Dose Consequences

The licensee should use the following assumptions in determining the TEDE for persons located at or beyond the EAB:

1. The dose calculations should determine the TEDE. TEDE is the sum of the committed effective dose equivalent (CEDE) from inhalation and the effective dose equivalent (EDE) from external exposure. The calculation of these two components of the TEDE should consider all radionuclides, including progeny from the decay of parent radionuclides that are significant with regard to dose consequences and the released radioactivity.[15]

2. The exposure-to-CEDE factors for inhalation of radioactive material should be derived from the data provided in ICRP Publication 30, "Limits for Intakes of Radionuclides by Workers" (Ref. 19). Table 2.1 of Federal Guidance Report 11, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion" (Ref. 20), provides tables of conversion factors acceptable to the NRC staff. The factors in the column headed "effective," yield doses corresponding to the CEDE.

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15 The prior practice of basing inhalation exposure on only radioiodine and not including radioiodine in external exposure calculations is not consistent with the definition of TEDE and the characteristics of the revised source term.

3. Table III.1 of Federal Guidance Report 12, “External Exposure to Radionuclides in Air, Water, and Soil” (Ref. 21), provides external EDE conversion factors acceptable to the NRC staff. The factors in the column headed “effective,” yield doses corresponding to the EDE.

4. No correction should be made for depletion of the effluent plume by deposition on the ground.

5. The TEDE should be determined for an individual at the most limiting EAB location. The maximum EAB TEDE for any 2-hour period following the start of the radioactivity release should be determined and used in determining compliance with the dose criteria in 10 CFR 50.67[16] and 10 CFR Part 52. The maximum 2-hour TEDE should be determined by calculating the postulated dose for a series of small time increments and performing a “sliding” sum over the increments for successive 2-hour periods. The maximum TEDE obtained is taken as the analysis results. The time increments should appropriately reflect the progression of the accident to capture the peak dose interval between the start of the event and the end of radioactivity release (see analysis release duration in Table 7). The analysis should assume that the most limiting 2-hour EAB χ/Q value occurs simultaneously with the limiting release to the environment (see also Regulatory Position 5.3 of this guide). In calculations, the maximum 2-hour EAB χ/Q should be used for the entire duration of the release to the environment to ensure that the limiting case is identified.

   If multiple release paths are analyzed separately, additional processing is needed to identify the maximum 2-hour TEDE that is the sum of all paths, since the maximum periods may not be the same for each path. In these cases, it will be necessary to assess each release using the maximum 2-hour EAB χ/Q, sum the doses for each pathway for each time increment, and then identify the maximum 2-hour EAB TEDE. As a conservative alternative, the maximum 2-hour TEDE for each path could be summed to determine the value for the accident.

   For the duration of the event, the breathing rate of this individual should be assumed to be $3.5\text{x}10^{-4}$ cubic meters per second.

6. TEDE should be determined for the most limiting receptor at the outer boundary of the LPZ for the duration of the accident and should be used in determining compliance with the dose criteria in 10 CFR 50.67 and 10 CFR Part 52.

   For the first 8 hours, the breathing rate of persons off site should be assumed to be $3.5\text{x}10^{-4}$ cubic meters per second. From 8 to 24 hours following the accident, the breathing rate should be assumed to be $1.8\text{x}10^{-4}$ cubic meters per second. After that and until the end of the accident, the rate should be assumed to be $2.3\text{x}10^{-4}$ cubic meters per second.

4.2 Control Room Dose Consequences

The following guidance should be used in determining the TEDE for persons located in the control room:

16 With regard to the EAB TEDE, the maximum 2-hour value is the basis for screening and evaluation under 10 CFR 50.59. Changes to doses outside of the 2-hour window are only considered in the context of their impact on the maximum 2-hour EAB TEDE.

4.2.1 Sources of Radiation

The TEDE analysis should consider all sources of radiation that will cause exposure to control room personnel. The applicable sources will vary from facility to facility, but typically will include the following:

(1) contamination of the control room atmosphere by the intake or infiltration of the radioactive material contained in the radioactive plume released from the facility,

(2) contamination of the control room atmosphere by the intake or infiltration of airborne radioactive material from areas and structures adjacent to the control room envelope,

(3) radiation shine from the external radioactive plume released from the facility,

(4) radiation shine from radioactive material in the reactor containment, and

(5) radiation shine from radioactive material in systems and components inside or external to the control room envelope (e.g., radioactive material buildup in recirculation filters).

4.2.2 Materials Releases and Radiation Levels

The radioactive material releases and radiation levels used in the control room dose analysis should be determined using the same source term, in-plant transport, and release assumptions used for determining the EAB and the LPZ TEDE values, unless these assumptions would result in non-conservative results for the control room.

4.2.3 Transport Models

The models used to transport radioactive material into and through the control room,[17] and the shielding models[18] used to determine radiation dose rates from external sources, should be structured to provide suitably conservative estimates of the exposure to control room personnel.

4.2.4 Engineered Safety Features

The licensee may assume credit for ESFs that mitigate airborne radioactive material within the control room. Such features may include control room isolation or pressurization or intake or recirculation filtration. Refer to Section 6.5.1, "ESF Atmospheric Cleanup System," of the SRP (Ref. 4) and Regulatory Guide 1.52, Revision 3, "Design, Inspection, and Testing Criteria for Air Filtration and Adsorption Units of Post-Accident Engineered-Safety-Feature Atmosphere Cleanup Systems in Light-

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[17] The iodine protection factor (IPF) methodology of Reference 22 may not be adequately conservative for all DBAs and control room arrangements because it models a steady-state control room condition. Since many analysis parameters change over the duration of the event, the IPF methodology should only be used with caution. The NRC computer codes HABIT (Ref. 23) and RADTRAD (Ref. 24) incorporate suitable methodologies.

[18] The nuclides used for modeling dose from airborne radioactivity inside the control room may not be conservative for determining the dose from radioactivity outside the control room.

Water-Cooled Nuclear Power Plants" (Ref. 25), for guidance. The control room design is often optimized for the DBA LOCA and the protection afforded for other accident sequences may not be as advantageous. In most designs, control room isolation is actuated by ESF signals or radiation monitors (RMs). In some cases, the ESF signal is effective only for selected accidents, placing reliance on the RMs for the remaining accidents. Several aspects of RMs can delay the control room isolation, including the delay for activity to build up to concentrations equivalent to the alarm setpoint and the effects of different radionuclide accident isotopic mixes on monitor response.

4.2.5 Personal Protective Equipment

The licensee should generally not take credit for the use of personal protective equipment or prophylactic drugs such as potassium iodide. The NRC may consider deviations on a case-by-case basis.

4.2.6 Dose Receptor

The dose receptor for these analyses is the hypothetical maximum exposed individual who is present in the control room for 100 percent of the time during the first 24 hours after the event, 60 percent of the time between 1 and 4 days, and 40 percent of the time from 4 days to 30 days.[19] For the duration of the event, the licensee should assume the breathing rate of this individual to be 3.5×10^{-4} cubic meters per second (Ref. 27).

4.2.7 Dose Conversion Factor

The licensee should calculate control room doses using the dose conversion factors identified in Regulatory Position 4.1 for use in offsite dose analyses. The calculation should consider all radionuclides, including progeny from the decay of parent radionuclides that are significant with regard to dose consequences and the released radioactivity. The EDE from photons may be corrected for the difference between finite cloud geometry in the control room and the semi-infinite cloud assumption used in calculating the dose conversion factors. The following expression may be used to correct the semi-infinite cloud dose, EDE_{∞}, to a finite cloud dose, EDE_{finite}, where the control room is modeled as a hemisphere that has a volume, V, in cubic feet, equivalent to that of the control room (Ref. 22).

Equation 1: $$EDE_{finite} = \frac{EDE_{\infty} V^{0.338}}{1173}$$

4.3 <u>Other Dose Consequences</u>

The licensee should use the guidance provided in Regulatory Positions 4.1 and 4.2, as applicable, to reassess the radiological analyses identified in Regulatory Position 1.3.1, such as those in NUREG-0737 (Ref. 3). The licensee should update design envelope source terms provided in

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19 These occupancy factors are already included in the determination of the χ/Q values using the Murphy and Campe methodology (Ref. 22) and should not be credited twice. The ARCON96 Code (Ref. 26) does not incorporate these occupancy factors in the determination of the χ/Q values. Therefore, when using ARCON96 χ/Q values, dose calculations should include the occupancy factors.

NUREG-0737 for consistency with the AST. In general, radiation exposures to plant personnel identified in Regulatory Position 1.3.1 should be expressed in terms of TEDE.

4.4 Acceptance Criteria

The accident dose radiological criteria for the EAB, the outer boundary of the LPZ, and for the control room are in 10 CFR 50.34, 10 CFR Part 52, 10 CFR 50.67, and GDC 19. These criteria are stated for evaluating reactor accidents of exceedingly low probability of occurrence and low risk of public exposure to radiation, (e.g., a large-break LOCA). For events with a higher probability of occurrence, postulated EAB and LPZ doses should not exceed the criteria tabulated in Table 7 (e.g., a fuel handling accident). The accident dose for the EAB must not exceed the acceptance criteria for any 2-hour period following the onset of the fission product release. The accident dose for the LPZ must not exceed the acceptance criteria during the entire period of the passage of the fission product release.

The acceptance criteria for the various NUREG-0737 (Ref. 3) items generally reference GDC 19 in Appendix A to 10 CFR Part 50 or specify criteria derived from GDC 19. These criteria are generally specified in terms of whole body dose or its equivalent to any body organ. For facilities applying for, or having received, approval for the use of an AST, licensees should update the applicable criteria for consistency with the TEDE criterion in 10 CFR 50.67(b)(2)(iii).

For new reactor applicants, the technical support center habitability acceptance criterion is based on the requirement of Paragraph IV.E.8 of Appendix E to 10 CFR Part 50 to provide an onsite TSC from which effective direction can be given and effective control can be exercised during an emergency. The radiation protection design of the TSC is acceptable if the total calculated radiological consequences for the postulated fission product release fall within the exposure acceptance criteria specified for the control room of 5 rem TEDE for the duration of the accident.

Table 7[20] Accident Dose Criteria for EAB, LPZ, and Control Room Locations

Accident or Case	EAB and LPZ Dose Criteria (TEDE)	Control Room Dose Criteria[21] (TEDE)	Analysis Release Duration
LOCA	0.25 sievert (Sv) (25 rem)	0.05 Sv (5.0 rem)	30 days for containment, ECCS, and MSIV (BWR) leakage
BWR Main Steamline Break			Instantaneous puff
Fuel Damage or Preincident Spike	0.25 Sv (25 rem)	0.05 Sv (5.0 rem)	
Equilibrium Iodine Activity	0.025 Sv (2.5 rem)	0.05 Sv (5.0 rem)	
BWR Rod Drop Accident	0.063 Sv (6.3 rem)	0.05 Sv (5.0 rem)	24 hours
PWR Steam Generator (SG) Tube Rupture			<u>Affected SG</u>: time to isolate[22]; <u>Unaffected SG(s)</u>: until cold shutdown is established
Fuel Damage or Preincident Spike	0.25 Sv (25 rem)	0.05 Sv (5.0 rem)	
Coincident Iodine Spike	0.025 Sv (2.5 rem)	0.05 Sv (5.0 rem)	
PWR Main Steamline Break			Until cold shutdown is established
Fuel Damage or Preaccident Spike	0.25 Sv (25 rem)	0.05 Sv (5.0 rem)	
Coincident Iodine Spike	0.025 Sv (2.5 rem)	0.05 Sv (5.0 rem)	
PWR Locked Rotor Accident	0.025 Sv (2.5 rem)	0.05 Sv (5.0 rem)	Until cold shutdown is established
PWR Control Rod Ejection Accident	0.063 Sv (6.3 rem)	0.05 Sv (5.0 rem)	30 days for containment pathway; until cold shutdown is established for secondary pathway
Fuel handling Accident	0.063 Sv (6.3 rem)	0.05 Sv (5.0 rem)	2 hours

The column labeled "Analysis Release Duration" summarizes the assumed radioactivity release durations identified in the individual appendices to this guide. Refer to these appendices for complete descriptions of the release pathways and durations.

20 For PWRs with steam generator (SG) alternative repair criteria, different dose criteria may apply to SG tube rupture and main steamline break analyses.

21 The control room exposure period is 30 days for all accidents.

22 Tube rupture in the affected SG may result in the need to control SG water level using steam dumps. These releases may extend the duration of the release from the affected SG beyond the initial isolation.

5. Analysis Assumptions and Methodology

5.1 General Considerations

5.1.1 Analysis Quality

The analyses discussed in this guide are reanalyses of the design basis safety analyses required by 10 CFR 50.67 and/or evaluations required by 10 CFR 50.34, 10 CFR Part 52, and GDC 19. These analyses are considered to be a significant input to the evaluations required by 10 CFR 50.92 or 10 CFR 50.59 and 10 CFR 52. The licensee should prepare, review, and maintain these analyses in accordance with quality assurance programs that comply with Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," to 10 CFR Part 50.

These design basis analyses were structured to provide a conservative set of assumptions to test the performance of one or more aspects of the facility design. Many physical processes and phenomena are represented by conservative bounding assumptions rather than being modeled directly. The staff has selected assumptions and models that provide an appropriate and prudent safety margin against unpredicted events in the course of an accident and compensate for large uncertainties in facility parameters, accident progression, radioactive material transport, and atmospheric dispersion. Licensees should exercise caution in proposing deviations based upon data from a specific accident sequence since the DBAs were never intended to represent any specific accident sequence; the proposed deviation may not be conservative for other accident sequences.

5.1.2 Credit for Engineered Safeguard Features

The licensee may take credit for accident mitigation features that are classified as safety related, are required to be operable by technical specifications, are powered by emergency power sources, and are either automatically actuated or, in limited cases, have actuation requirements explicitly addressed in emergency operating procedures. However, the licensee should not take credit for engineered safeguards features that would affect the generation of the source term described in Tables 1 and 2. For example, licensees should not credit emergency core cooling system operation during the first two hours of the DBA in order to reduce or mitigate the source term generation within the core. Additionally, the licensee should assume the single active component failure that results in the most limiting radiological consequences. Assumptions regarding the occurrence and timing of a loss of offsite power should be selected with the objective of maximizing the postulated radiological consequences. The licensee should consider design basis delays in actuation of these features, especially for features that rely on manual intervention.

5.1.3 Assignment of Numeric Input Values

The licensee should select the numeric values to be used as inputs to the dose analyses with the objective of determining a conservative postulated dose. In some instances, a particular parameter may be conservative in one portion of an analysis but may be nonconservative in another portion of the same analysis. For example an assumption of minimum containment system spray flow is usually conservative for estimating iodine scrubbing, but in many cases may be nonconservative when determining sump pH. Sensitivity analyses may be needed to determine the appropriate value to use. As a conservative alternative, the limiting value applicable to each portion of the analysis may be used in the evaluation of that portion. A single value may not be applicable for a parameter for the duration of the event, particularly for parameters affected by changes in density. For parameters addressed by technical specifications, the value used in the analysis should be that specified in the technical specifications.[23] If a range of values or a tolerance band is specified, the value that would result in a conservative postulated dose should be used. If the parameter is based on the results of less frequent surveillance testing (e.g., steam generator nondestructive testing), consideration should be given to the degradation that may occur between periodic tests in establishing the analysis value.

5.1.4 Applicability of Prior Licensing Basis

The NRC staff considers the implementation of an AST to be a significant change to the design basis of the facility that is voluntarily initiated by the licensee. To issue a license amendment authorizing the use of an AST and the TEDE dose criteria, the NRC staff must make a current finding of compliance with regulations applicable to the amendment. The characteristics of the ASTs and the revised dose calculational methodology may be incompatible with many of the analysis assumptions and methods currently reflected in the facility's design basis analyses. The NRC staff may find that new or unreviewed issues are created by a particular site-specific implementation of the AST, warranting review of staff positions approved subsequent to the initial issuance of the license. This is not considered a backfit as defined by 10 CFR 50.109, "Backfitting." However, prior design bases that are unrelated to the use of the AST, or are unaffected by the AST, may continue as the facility's design basis. Licensees should ensure that analysis assumptions and methods are compatible with the ASTs and the TEDE criteria.

5.2 Accident-Specific Assumptions

The appendices to this regulatory guide provide accident-specific assumptions that are acceptable to the staff for performing site specific analyses as required by 10 CFR 50.34, 10 CFR Part 52, 10 CFR 50.67, and GDC 19. Licensees should review their license basis documents for guidance pertaining to the analysis of radiological DBAs other than those provided in this guide. The DBAs addressed in these attachments were selected from accidents that may involve damage to irradiated fuel. This guide does not address DBAs with radiological consequences based on technical specification reactor or secondary coolant-specific activities only. The inclusion or exclusion of a particular DBA in this guide should not be interpreted as indicating that an analysis of that DBA is required or not required. Licensees should analyze the DBAs that are affected by the specific proposed applications of an AST and changes to the facility or to the radiological analyses.

The NRC staff has determined that the analysis assumptions in the appendices to this guide provide an integrated approach to performing the individual analyses and generally expects licensees to address each assumption or to propose acceptable alternatives. Such alternatives may be justifiable on the basis of plant-specific considerations, updated technical analyses, or, in some cases, a previously approved licensing basis consideration. The assumptions in the appendices are deemed consistent with the AST identified in Regulatory Position 3 and internally consistent with each other. Although licensees are free to propose alternatives to these assumptions for consideration by the NRC staff, licensees should avoid use of previously approved staff positions that would adversely affect this consistency.

The NRC is committed to using PRA insights in its regulatory activities and will consider licensee proposals for changes in analysis assumptions based upon risk insights. The staff will not approve proposals that would reduce the defense in depth deemed necessary to provide adequate

[23] Note that for some parameters, the technical specification value may be adjusted for analysis purposes by factors provided in other regulatory guidance. For example, ESF filter efficiencies are based on the guidance in Regulatory Guide 1.52 (Ref. 25), rather than the surveillance test criteria in the technical specifications. Generally, these adjustments address possible changes in the parameter between scheduled surveillance tests.

protection for public health and safety. In some cases, this defense in depth compensates for uncertainties in the PRA analyses and addresses accident considerations not adequately addressed by the CDF and LERF surrogate indicators of overall risk.

5.3 Meteorology Assumptions

Atmospheric dispersion factors (χ/Q values) for the EAB, the LPZ, and the control room that the staff approved during initial facility licensing or in subsequent licensing proceedings may be used in performing the radiological analyses identified by this guide, provided such values remain relevant to the particular accident, its release points, and receptor locations. If the previously approved values are based on a misapplication of a methodology or calculational errors are identified in the values, the NRC staff will pursue necessary corrections with the applicant or licensee. Regulatory Guides 1.3, 1.4, and 1.145, "Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants," and the paper by Murphy-Campe entitled, "Nuclear Power Plant Control Room Ventilation System Design for Meeting General Criterion 19" (Refs. 6, 7, 22, and 28), document methodologies that have been used in the past for determining χ/Q values.

Regulatory Guides 1.145 (Ref. 28) and 1.194, "Atmospheric Relative Concentrations for Control Room Radiological Habitability Assessments at Nuclear Power Plants" (Ref. 31), should be used if the FSAR χ/Q values are to be revised or if values are to be determined for new release points or receptor distances. EAB χ/Q values are determined for the limiting 2-hour period within a 30-day period following the start of the radioactivity release. Control room χ/Q values are generally determined for initial averaging periods of 0–2 hours and 2–8 hours and the LPZ χ/Q value for a 0–8 hour averaging period. The control room and LPZ χ/Q values are also generally determined for averaging periods of 8–24 hours, 24–96 hours, and 96–720 hours. The period of the most adverse release of radioactive materials to the environment should be assumed to occur coincident with the period of most unfavorable atmospheric dispersion. One acceptable methodology for calculating the control room and LPZ χ/Q values is as follows. If the 0–2 hour χ/Q value is calculated, this value should be used coincident with the limiting portion of the release to the environment. The 2–8 hour χ/Q value is used for the remaining 6 hours of the first 8-hour time period. Part of this 6-hour interval may occur before and/or after the limiting 2-hour period. The 8–24, 24–96, and 96–720 hour χ/Q values should similarly be used for the remainder of the release duration.

D. IMPLEMENTATION

The purpose of this section is to provide information to applicants and licensees regarding the NRC's plans for using this draft regulatory guide. The NRC does not intend or approve any imposition or backfit in connection with its issuance.

The NRC has issued this draft guide to encourage public participation in its development. The NRC will consider all public comments received in development of the final guidance document. In some cases, applicants or licensees may propose an alternative or use a previously established acceptable alternative method for complying with specified portions of the NRC's regulations. Otherwise, the methods described in this guide will be used in evaluating compliance with the applicable regulations for license applications, license amendment applications, and amendment requests.

REGULATORY ANALYSIS

Statement of Problem

The NRC staff is proposing to develop and issue a revision to Regulatory Guide 1.183. The NRC is proposing in this revision incorporation of guidance for the radiological source term for new reactor licensing and improvement of the current guidance for ASTs for operating reactors. The staff proposes to issue a draft guide for public review and comment, and upon resolution of public comments, to finalize and implement the revised regulatory guide.

In the early 1970s, the NRC staff issued regulatory guides for evaluating radiological consequences using the radiological source term described in TID-14844. Since the publication of TID 14844 (Ref. 2), significant advances have been made in understanding the timing, magnitude, and chemical form of fission product releases from severe nuclear power plant accidents. In 1995, the NRC published NUREG-1465 (Ref. 1), which uses updated research to provide more realistic estimates of the accident source term that were physically based and that could be applied to the design of future light-water power reactors. In addition, the NRC determined that the analytical approach based on the TID-14844 source term would continue to be adequate to protect public health and safety for the current licensed power reactors. The NRC staff also determined that some current licensees may wish to use the NUREG-1465 source term, referred to as the AST, in analyses to support cost-beneficial licensing actions. The NRC staff, therefore, initiated several actions to provide a regulatory basis for operating reactors to use an AST in design basis analyses. These initiatives resulted in the development and issuance of 10 CFR 50.67 and Regulatory Guide 1.183. Issuance of RG 1.183 provided the first comprehensive accident source term guidance for performing radiological consequence analyses using the AST.

Since the initial issuance of Regulatory Guide 1.183, the NRC staff and the commercial nuclear industry have both gained substantial experience with the implementation of an AST, in whole or part, for current licensed facilities. Based on this experience and on specific feedback and comments from licensees, as well as the anticipation of licensing advanced LWRs, the NRC needs to update this regulatory guide for performing evaluations of fission product releases and radiological consequences of postulated LWR DBAs.

Existing Regulatory Framework

According to 10 CFR 50.34, each applicant for a stationary power reactor construction permit on or after January 10, 1997 (new reactors), shall comply with the requirements of 10 CFR 50.34(a)(1)(ii). In particular, 10 CFR 50.34(a)(1)(ii)(D)(1) states, “An individual located at any point on the boundary of the exclusion area for any 2 hour period following the onset of the postulated fission product release, would not receive a radiation dose in excess of 25 rem total effective dose equivalent (TEDE).” Furthermore, 10 CFR 50.34(a)(1)(ii)(D)(2) states, “An individual located at any point on the outer boundary of the low population zone, who is exposed to the radioactive cloud resulting from the postulated fission product release (during the entire period of its passage) would not receive a radiation dose in excess of 25 rem total effective dose equivalent (TEDE).”

Appendix A to 10 CFR Part 50 establishes minimum requirements for the design criteria for water-cooled nuclear power plants. GDC 19, as it applies to new reactors[1] or holders of operating licenses using an AST under 10 CFR 50.67, states, "adequate radiation protection shall be provided to ensure that radiation exposures shall not exceed 0.05 Sv (5 rem) total effective dose equivalent (TEDE) as defined in 10 CFR 50.2 for the duration of the accident."

A holder of an operating license issued before January 10, 1997, or a holder of a renewed license under 10 CFR Part 54 whose initial operating license was issued before January 10, 1997 (operating reactors), is allowed by 10 CFR 50.67 to voluntarily revise their current accident source term used in design basis radiological consequence analyses.

A licensee who seeks to revise its current accident source term in design basis radiological consequence analyses shall apply for a license amendment under 10 CFR 50.90. The application shall contain an evaluation of the consequences of applicable DBAs previously analyzed in the safety analysis report.

As stated in 10 CFR 50.67(2)(i), "An individual located at any point on the boundary of the exclusion area for any 2-hour period following the onset of the postulated fission product release, would not receive a radiation dose in excess of 0.25 Sv (25 rem) total effective dose equivalent (TEDE)." Furthermore, 10 CFR 50.67(2)(ii) states, "An individual located at any point on the outer boundary of the low population zone, who is exposed to the radioactive cloud resulting from the postulated fission product release (during the entire period of its passage), would not receive a radiation dose in excess of 0.25 Sv (25 rem) total effective dose equivalent (TEDE)." In addition, 10 CFR 50.67(2)(iii) restates the control room habitability criteria of GDC 19 for use of an AST. Specifically, this section states, "Adequate radiation protection is provided to permit access to and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 0.05 Sv (5 rem) total effective dose equivalent (TEDE) for the duration of the accident."

Applicants for new reactors are required by 10 CFR Part 52, "Licenses, Certifications, and Approvals for Nuclear Power Plants," to provide an analysis of the proposed site. Sections 52.47 and 52.79 of 10 CFR Part 52 also require standard design certification and combined license applicants to provide a similar analysis and evaluation.

Objective of the Regulatory Action

The objective of the proposed revision to Regulatory Guide 1.183 is to provide more useful and up-to-date guidance for complying with the regulations described above in the Existing Regulatory Framework section. Specifically, Regulatory Guide 1.183 provides methods and assumptions for performing evaluations of fission product releases and radiological consequences of several postulated LWR DBAs. The NRC is updating this guide to describe assumptions and methods that are acceptable to the NRC staff for performing design basis radiological analyses using an AST. The revised guide will describe the source and the scope, nature, and documentation of associated analyses and evaluations. It will also describe the content of submittals acceptable to the NRC staff.

The staff has determined that holders of operating licenses may continue to use methods and assumptions previously approved by the NRC unless they are subject to the requirements of 10 CFR 50.109, "Backfitting." The NRC staff expects that licensees could use the information in the

[1] For the purpose of this paragraph, "new reactors" are defined as "Applicants and holders of construction permits and operating licenses who apply on or after January 10, 1997, applicants for design approvals or certifications under part 52 of this chapter who apply on or after January 10, 1997, applicants for and holders of combined licenses or manufacturing licenses under part 52 of this chapter who do not reference a standard design approval or certification."

guide if they voluntarily decide to replace previous approved methods and assumptions with those specified in this guide.

Alternative Approaches

The NRC considered the following alternative approaches:

Do not revise Regulatory Guide 1.183.
Revise Regulatory Guide 1.183.

Alternative 1: Do Not Revise Regulatory Guide 1.183

Under this alternative, the NRC staff would not issue the proposed revised guidance and the current guidance would be retained. If the NRC does not take action, there would not be any changes in the cost or benefit to the public, licensees, or the NRC. However, the "no-action" option would not address the unnecessary burden for industry as well as for the NRC staff. This burden would be in the preparation and response to NRC staff's requests for additional information (RAIs), reanalyses, and supplementation of licensee applications or license amendment requests.

Alternative 2 Revise Regulatory Guide 1.183

In this alternative, the staff would revise Regulatory Guide 1.183 to include applications for new reactors and to update the regulatory guide based on operating reactor experience with applying an AST to design basis dose consequence analysis. Issuing the proposed revised guidance would maintain public safety by ensuring that safety analyses use appropriate analysis assumptions and methods, reduce unnecessary regulatory burden by providing clear AST methods and assumptions for dose consequence analysis, improve efficiency and effectiveness, as the revised guidance would provide licensees with the staff positions, thereby minimizing RAIs and resubmittals, and maintain public confidence by providing guidance that ensures that safety analyses are adequate to ensure that regulatory requirements are met.

The impact to NRC would be the costs associated with preparing and issuing the revision. The impact to the public would be the voluntary costs associated with reviewing and providing comments to NRC during the public comment period.

The NRC staff has determined that this alternative—issuing a revised Regulatory Guide 1.183—is the most advantageous approach to addressing the need of updated regulatory guidance.

Evaluation of Values and Impacts

New reactor license applicants are required to evaluate the radiological consequences for select DBAs for site evaluations and control room habitability. A license applicant or licensee may propose alternative approaches to demonstrate compliance with the NRC's regulations. Existing licensees of operating reactors would revise their current methods and assumptions for evaluating radiological consequences only if they perceive it to be in their interest to do so or if they are subject to the requirements of 10 CFR 50.109. The following qualitative advantages of revising Regulatory Guide 1.83 also apply:

- Completion of the proposed action is estimated to require from 0.2 to 0.5 full-time equivalent staff members. Associated costs include publication costs. The NRC would revise Regulatory Guide 1.183 internally.

- Regulatory Guide 1.183 has improved regulatory efficiency by providing an acceptable approach and by encouraging consistency in the assessment of control room habitability and offsite accident consequences. The revised Regulatory Guide 1.183 would provide enhanced guidance for new reactor applicants and existing licensees by updating analysis guidelines, clarify NRC regulatory positions, and correct minor typographical and content errors. The revised guide would reduce the likelihood for followup questions and possible revisions in licensees' analyses and plant modifications. The proposed regulatory guide would simplify NRC reviews because license applications and amendments should be more predictable and analytically consistent.

- The revised regulatory guide would result in cost savings to both the NRC and industry. The NRC will incur one-time incremental costs to revise the regulatory guide, submit it for public comment, and publish the final revision. However, the NRC should also realize cost savings associated with more efficient review of new reactor applicants and existing reactor licensee submittals. The staff believes that the continuous and ongoing cost savings associated with these reviews should offset the one-time development costs.

- The industry would also realize a net savings, as the one-time incremental cost to review and comment on a revised regulatory guide would be compensated for by the efficiencies to be gained in minimizing followup questions and revisions associated with each licensee application or amendment submittal.

- With the possible exception of applicant agencies, such as Tennessee Valley Authority or municipal licensees, no other governmental agencies would be affected by the proposed Regulatory Guide 1.183 revision.

Conclusion

Based on this regulatory analysis, the staff recommends that the NRC revise Regulatory Guide 1.183. Experience with license amendment reviews under Regulatory Guide 1.183 since its publication has demonstrated the need for up-to-date and revised guidance for performing radiological dose calculations for new reactors. Currently licensed plants may elect to use the updated guidance on a voluntary basis. Based on this regulatory analysis, the staff recommends that the NRC prepare a revised Regulatory Guide 1.183 for calculating the radiological consequences of DBAs and issue the revision as a draft regulatory guide for public comment and, upon resolution of public comments, finalize the regulatory guide.

BACKFIT ANALYSIS

The proposed regulatory guide revision does not require a backfit analysis as described in 10 CFR 50.109(c) because it does not impose a new or amended provision in the NRC's regulations. It does not impose a regulatory staff position that interprets the NRC's regulations differently than a previously applicable staff position. Regulatory guides are not substitutes for regulations, and compliance with them is not required. Methods and solutions different from those set out in the regulatory guide will be acceptable if they provide a basis for the regulatory findings needed to support issuance or continuance of a permit or license by the Commission. A licensee can select a preferred method of achieving compliance with a license condition, the rules, or orders of the Commission as described in 10 CFR 50.109(a)(7).

This regulatory guide revision provides licensee with an opportunity to use an updated method for determining control room and offsite radiological assessments, if that is the method the licensee prefers.

The NRC staff will use this guide to evaluate licensee-initiated changes if there is a clear nexus between the proposed change and the guidance contained in the guide. The staff will also use it to review changes when the licensees have committed to using this guide.

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[1] Publicly available NRC published documents such as Regulations, Regulatory Guides, NUREGs, and Generic Letters listed herein are available electronically through the Electronic Reading room on the NRC's public Web site at: http://www.nrc.gov/reading-rm/doc-collections/. Copies are also available for inspection or copying for a fee from the NRC's Public Document Room (PDR) at 11555 Rockville Pike, Rockville, MD; the mailing address is USNRC PDR, Washington, DC 20555; telephone 301-415-4737 or (800) 397-4209; fax (301) 415-3548; and e-mail PDR.Resource@nrc.gov.

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APPENDIX A

ASSUMPTIONS FOR EVALUATING THE RADIOLOGICAL CONSEQUENCES OF LIGHT-WATER REACTOR LOSS-OF-COOLANT ACCIDENTS

The assumptions in this appendix are acceptable to the staff of the U.S. Nuclear Regulatory Commission (NRC) for evaluating the radiological consequences of loss-of-coolant accidents (LOCAs) at light-water reactors (LWRs). These assumptions supplement the guidance provided in the main body of this guide.

Appendix A, "General Design Criteria for Nuclear Power Plants," to 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities," defines LOCAs as those postulated accidents that result from a loss-of-coolant inventory at rates that exceed the capability of the reactor coolant makeup system. Leaks up to a double-ended rupture of the largest pipe of the reactor coolant system (RCS) are included. The LOCA, as with all design basis accidents (DBAs), is a conservative surrogate accident that is intended to challenge selective aspects of the facility design. Analyses are performed using a spectrum of break sizes to evaluate fuel and emergency core cooling system (ECCS) performance. With regard to radiological consequences, a large-break LOCA is assumed as the design basis case for evaluating the performance of release mitigation systems and the containment and for evaluating the proposed siting of a facility. As such, the licensee should analyze the spectrum of large-break LOCAs credible for its facility. The analysis should determine the limiting large-break LOCA, assuming substantial core damage, from the perspective of dose consequences to the public and control room workers.

A-1. Source Term

Regulatory Position 3 of this guide provides acceptable assumptions regarding core inventory and the release of radionuclides from the fuel.

A-1.1 If the sump or suppression pool pH is controlled at values of 7 or greater, the chemical form of radioiodine released to the containment should be assumed to be 95 percent cesium iodide (CsI), 4.85 percent elemental iodine, and 0.15 percent organic iodide. Iodine species, including those from iodine reevolution, for sump or suppression pool pH values less than 7 will be evaluated on a case-by-case basis. Evaluations of pH should consider the effect of acids and bases created during the LOCA event (e.g., radiolysis products). With the exception of elemental and organic iodine and noble gases, fission products should be assumed to be in particulate form.

A-2. Transport in Primary Containment

Acceptable assumptions related to the transport, reduction, and release of radioactive material in and from the primary containment in pressurized-water reactors (PWRs) or the drywell in boiling-water reactors (BWRs) are as follows:

A-2.1 The radioactivity released from the fuel should be assumed to mix instantaneously and homogeneously throughout the free air volume of the primary containment in PWRs or the drywell in BWRs as it is released. This distribution should be adjusted if there are internal compartments that have limited ventilation exchange. The suppression pool free air volume may be included provided there is a mechanism to ensure mixing between the drywell to the wetwell.

The release into the containment or drywell should be assumed to terminate at the end of the early in-vessel release phase.

A-2.2 Reduction in airborne radioactivity in the containment by natural deposition within the containment may be credited. Chapter 6.5.2, "Containment Spray as a Fission Product Cleanup System," of NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants" (hereafter referred to as the SRP) (Ref. A-1), and in NUREG/CR-6189, "A Simplified Model of Aerosol Removal by Natural Processes in Reactor Containments" (Ref. A-2), describe acceptable models for removal of iodine and aerosols (DBA analyses should use the 10th percentile values). The analysis code RADTRAD (Ref. A-3) incorporates the latter model. The NRC staff no longer accepts the prior practice of deterministically assuming that a 50-percent plateout of iodine is released from the fuel because it is inconsistent with the characteristics of the revised source terms. Some licensees may consider specific containment design features to evaluate aerosol fission product removal. The amount of removal will be evaluated on an individual case basis.

A-2.3 Reduction in airborne radioactivity in the containment by containment spray systems that have been designed and are maintained in accordance with Chapter 6.5.2 of the SRP (Ref. A-1) may be credited. Chapter 6.5.2 of the SRP and NUREG/CR-5966, "A Simplified Model of Aerosol Removal by Containment Sprays" (Ref. A-4), describe acceptable models for the removal of iodine and aerosols (DBA analyses should use the 10th percentile values). The analysis code RADTRAD (Ref. A-3) incorporates this simplified model.

The evaluation of the containment sprays should address areas within the primary containment that are not covered by the spray drops. The mixing rate attributed to natural convection between sprayed and unsprayed regions of the containment building, provided that adequate flow exists between these regions, is assumed to be two turnovers of the unsprayed region volume per hour, unless other rates are justified. On a case-by-case basis, the licensee may consider containment mixing rates determined by the cooldown rate in the sprayed region and the buoyancy-driven flow that results. The containment building atmosphere may be considered a single, well-mixed volume if the spray covers at least 90 percent of the containment building space and an engineered safety feature (ESF) ventilation system is available for adequate mixing of the unsprayed compartments.

As provided in the SRP, the maximum decontamination factor (DF) for elemental iodine is based on the maximum iodine activity in the primary containment atmosphere when the sprays actuate divided by the activity of iodine remaining at some time after decontamination. The SRP also states that the particulate iodine removal rate should be reduced by a factor of 10 when a DF of 50 is reached. The reduction in the removal rate is not required if the removal rate is based on the calculated time-dependent airborne aerosol mass. There is no specified maximum DF for aerosol removal by sprays. The maximum activity to be used in determining the DF is defined as the iodine activity in the columns labeled "Total" in Tables 1 and 2 of this guide multiplied by 0.05 for elemental iodine and by 0.95 for particulate iodine (i.e., aerosol treated as particulate in SRP methodology).

A-2.4 Reduction in airborne radioactivity in the containment by in-containment recirculation filter systems may be credited if these systems meet the guidance of Regulatory Guide 1.52, Revision 3, "Design, Inspection, and Testing Criteria for Air Filtration and Adsorption Units of Post-Accident Engineered-Safety-Feature Atmospheric Cleanup Systems in Light-Water-Cooled Nuclear Power Plants" (Ref. A-5). The filter media loading caused by the increased aerosol release associated with the revised source term should be addressed.

A-2.5 Reduction in airborne radioactivity in the containment by suppression pool scrubbing in BWRs should generally not be credited. However, the staff may consider such reduction on an individual case basis. The evaluation should consider the relative timing of the blowdown and the fission product release from the fuel, the force driving the release through the pool, and the potential for any bypass of the suppression pool (Ref. A-6). For suppression pool solutions having a pH less than 7, elemental iodine vapor should be conservatively assumed to evolve into the containment atmosphere.

A-2.6 Reduction in airborne radioactivity in the containment by retention in ice condensers, or other ESFs not addressed above, should be evaluated on an individual case basis. See Section 6.5.4 of the SRP (Ref. A-1).

A-2.7 The evaluation should assume that the primary containment (i.e., drywell and wetwell for Mark I and II containment designs) will leak at the peak pressure technical specification leak rate for the first 24 hours. For PWRs, the leak rate may be reduced after the first 24 hours to 50 percent of the technical specification leak rate. For BWRs, leakage may be reduced after the first 24 hours, if supported by plant configuration and analyses, to a value not less than 50 percent of the technical specification leak rate. Leakage from subatmospheric containments is assumed to terminate when the containment is brought to and maintained at a subatmospheric condition as defined by technical specifications.

For BWRs with Mark III containments, the leakage from the drywell into the primary containment should be based on the steaming rate of the heated reactor core, with no credit for core debris relocation. This leakage should be assumed during the 2-hour period between the initial blowdown and termination of the fuel radioactivity release (gap and early in-vessel release phases). After 2 hours, the radioactivity is assumed to be uniformly distributed throughout the drywell and the primary containment.

A-2.8 If the primary containment is routinely purged during power operations, the licensee should analyze releases via the purge system before containment isolation and should sum the resulting doses with the postulated doses from other release paths. The purge release evaluation should assume that 100 percent of the radionuclide inventory in the RCS liquid is released to the containment at the initiation of the LOCA. This inventory should be based on the technical specification RCS equilibrium activity. Iodine spikes need not be considered. If the purge system is not isolated before the onset of the gap release phase, the licensee should consider release fractions associated with the gap release and early in-vessel release phases as applicable.

A-3. Dual Containments

For facilities with dual containment systems, the acceptable assumptions related to the transport, reduction, and release of radioactive material in and from the secondary containment or enclosure buildings are as follows:

A-3.1 Leakage from the primary containment should be considered to be collected, processed by ESF filters, if any, and released to the environment via the secondary containment exhaust system during periods in which the secondary containment has a negative pressure as defined in technical specifications. Credit for an elevated release should be assumed only if the point of physical release is more than 2.5 times the height of any adjacent structure.

A-3.2 Leakage from the primary containment is assumed to be released directly to the environment as a ground-level release during any period in which the secondary containment does not have a negative pressure as defined in technical specifications.

A-3.3 The effect of high wind speeds on the ability of the secondary containment to maintain a negative pressure should be evaluated on a case-by-case basis. The wind speed to be assumed is the 1-hour average value that is exceeded only 5 percent of the total number of hours in the dataset. Ambient temperatures used in these assessments should be the 1-hour average value that is exceeded either 5% or 95% of the total numbers of hours in the data set, whichever is conservative for the intended use (e.g., if high temperatures are limiting, use those exceeded only 5 % of the time) (Ref . A-7).

A-3.4 Credit for dilution in the secondary containment may be allowed when adequate means to cause mixing can be demonstrated. Otherwise, the leakage from the primary containment should be assumed to be transported directly to exhaust systems without mixing. Credit for mixing, if found to be appropriate, should generally be limited to 50 percent. This evaluation should consider the magnitude of the containment leakage in relation to contiguous building volume or exhaust rate, the location of exhaust plenums relative to projected release locations, the recirculation ventilation systems, and internal walls and floors that impede stream flow between the release and the exhaust.

A-3.5 Primary containment leakage that bypasses the secondary containment should be evaluated at the bypass leak rate incorporated in the technical specifications. If the bypass leakage is through water (e.g., via a filled piping run that is maintained full), credit for retention of iodine and aerosols may be considered on a case-by-case basis. Similarly, deposition of aerosol radioactivity in gas-filled lines may be considered on a case-by-case basis.

A-3.6 Reduction in the amount of radioactive material released from the secondary containment because of ESF filter systems may be taken into account provided that these systems meet the guidance of Regulatory Guide 1.52 (Ref. A-5).

A-4. Assumptions on Engineered Safety Feature System Leakage

ESF systems that recirculate sump water outside of the primary containment are assumed to leak during their intended operation. This release source includes leakage through valve packing glands, pump shaft seals, flanged connections, and other similar components. This release source may also include leakage through valves isolating interfacing systems (Ref. A-6). The licensee should analyze the radiological consequences from the postulated leakage and combine them with consequences postulated for other fission product release paths to determine the total calculated radiological consequences from the LOCA. The following assumptions are acceptable for evaluating the consequences of leakage from ESF components outside the primary containment for BWRs and PWRs:

A-4.1 With the exception of noble gases, all fission products released from the fuel to the containment (as defined in Tables 1 and 2 of this guide) should be assumed to instantaneously and homogeneously mix in the primary containment sump water (in PWRs) or suppression pool (in BWRs) at the time of release from the core. In lieu of this deterministic approach, suitably conservative mechanistic models for the transport of airborne activity in containment to the sump water may be used. Note that many of the parameters that make spray and deposition models conservative with regard to containment airborne leakage are nonconservative with regard to the build up of sump activity.

A-4.2 The leakage should be taken as two times[1] the sum of the simultaneous leakage from all components in the ESF recirculation systems above which the technical specifications, or licensee commitments to item III.D.1.1 of NUREG-0737, "Clarification of TMI Action Plan Requirements" (Ref. A-8), would require declaring such systems inoperable. Design leakage from any systems not included in technical specifications that transport primary coolant sources outside of containment should be added to the total leakage. The applicant should justify the design leakage used. The leakage should be assumed to start at the earliest time the recirculation flow occurs in these systems and end at the latest time the releases from these systems are terminated and should account for the ESF leakage at accident conditions. Consideration should also be given to design leakage through valves isolating ESF recirculation systems from tanks vented to the atmosphere (e.g., ECCS pump miniflow return to the refueling water storage tank).

A-4.3 With the exception of iodine, all radioactive materials in the recirculating liquid should be assumed to be retained in the liquid phase.

A-4.4 If the temperature of the leakage exceeds 212 degrees Fahrenheit (F), the fraction of total iodine (i.e., aerosol, elemental, and organic) in the liquid that becomes airborne should be assumed to equal the fraction of the leakage that flashes to vapor. This flash fraction, FF, should be determined using a constant enthalpy, h, process, based on the maximum time-dependent temperature of the sump water circulating outside the containment using the following formula:

$$FF = \frac{h_{f_1} - h_{f_2}}{h_{fg}}$$

Where: h_{f1} is the enthalpy of liquid at system design temperature and pressure; h_{f2} is the enthalpy of liquid at saturation conditions (14.7 pounds per square inch absolute, 212 degrees F); and h_{fg} is the heat of vaporization at 212 degrees F.

A-4.5 If the temperature of the leakage is less than 212 degrees F or the calculated FF is less than 10 percent, the amount of iodine that becomes airborne should be assumed to be 10 percent of the total iodine activity in the leaked fluid, unless a smaller amount can be substantiated. The justification of such values should consider the sump pH history; changes to the leakage pH caused by pooling on concrete surfaces, leaching through piping insulation, evaporation to dryness, and mixing with other liquids in drainage sumps; area ventilation rates and temperatures; and subsequent reevolution of iodine.

A-4.6 The radioiodine that is postulated to be available for release to the environment is assumed to be 97 percent elemental and 3 percent organic.[2] Reduction in release activity by dilution or holdup within buildings, or by ESF ventilation filtration systems, may be credited where applicable. Filter systems used in these applications should be evaluated against the guidance of Regulatory Guide 1.52 (Ref. A-5).

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1 The multiplier of 2 is used to account for increased leakage in these systems over the duration of the accident and between surveillances or leakage checks.

2 The 97-percent elemental, 3-percent organic speciation is a conservative deterministic assumption based on the hypothesis that most of the iodine released to the environment will be in elemental form with a small percentage converted to organic as supported in Section 3.5 of NUREG-1465, "Accident Source Terms for Light-Water Nuclear Power Plants—Final Report," issued February 1995 (Ref. A-9).

A-5. Main Steam Isolation Valve Leakage in Boiling-Water Reactors

For BWRs, the main steam isolation valves (MSIVs) have design leakage that may result in a radioactivity release. The licensee should analyze and combine the radiological consequences from postulated MSIV leakage with consequences postulated for other fission product release paths to determine the total calculated radiological consequences from the LOCA. The following assumptions are acceptable for evaluating the consequences of MSIV leakage:

A-5.1 The source of the MSIV leakage is assumed to be the activity concentration in the reactor vessel steam dome. At the end of the early in-vessel release phase, the activity concentration in the vessel dome should be assumed to equal the containment (or drywell) activity concentration.

The radioactivity released from the fuel to the MSIV source volume should be assumed to mix instantaneously and homogeneously throughout the free air volume of the MSIV source volume. No credit should be assumed for activity reduction by the steam separators or by iodine partitioning in the reactor vessel.

For Mark I, II and III containment designs, Section 5.2 of the report entitled, "Analysis of Main Steam Isolation Valve Leakage in Design Basis Accidents Using MELCOR 1.8.6 and RADTRAD" (Ref. A-10), describes an acceptable model for estimating the radioactivity available for release via MSIV leakage. This method uses the containment source term given in Regulatory Position 3 using Table 5-3 of Reference A-10 to provide a MSIV source concentration. Table 5-3 values for a Mark II containment designs may be obtained by adjusting the values in Table 5-1 of Reference A-10 as described in Section 5.2 of Reference A-10.[3]

For BWR designs other than those discussed above, other models of MSIV source concentration will be considered on a case-by-case basis.

A-5.2 The chemical form of radioiodine released to the reactor vessel steam dome and drywell should be assumed to be 95 percent cesium iodide (CsI), 4.85 percent elemental iodine, and 0.15 percent organic iodide. With the exception of elemental and organic iodine and noble gases, fission products should be assumed to be in particulate form.

A-5.3 Natural deposition in drywell may be credited. An acceptable model for removal of iodine and aerosols is in NUREG/CR-6189 (Ref. A-2). The analysis code RADTRAD (Ref. A-3) incorporates this model (DBA analyses should use the 10th percentile values).

A-5.4 Reduction in drywell radioactivity due to operable containment spray systems that have been designed and are maintained in accordance with Chapter 6.5.2 of the SRP (Ref. A-1) may be credited on a case-by-case basis.

A-5.5 All the MSIVs should be assumed to leak at the maximum leak rate above which the technical specifications would require declaring the MSIVs inoperable. The leakage should be assumed to continue for the duration of the accident as specified in Table 7 of this guide and should be assigned to steamlines so that the accident dose is maximized. Postulated leakage may be reduced after the first 24 hours, if supported by site-specific analyses, to a value not less than 50

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[3] The Table 5-3 values for a Mark II containment are calculated as follows: 0.0–0.5 hours—9.6E-5 * V_{sd}, 0.5–1.0 hours—4.2E-5 * V_{sd}, and 1.0–2.0 hours— 6.3E-6 * V_{sd} where V_{sd} is the free volume of the Mark II steam dome in cubic feet.

percent of the maximum leak rate. Section 5.4 of Reference A-10 describes an acceptable model for estimating the volumetric flow rate in the steamline.

A-5.6 A reduction in MSIV releases that is caused by holdup and deposition in main steam piping and main condenser, including the treatment of air ejector effluent by offgas systems, may be credited if the components and piping systems used in the release path are capable of performing their safety function during and following a safe-shutdown earthquake and are powered by emergency power sources. These reductions are allowed for safety grade steam system piping segments that are enclosed by physical barriers, such as closed valves. The piping segments and physical barriers are to be designed, constructed, and maintained to Quality Group A and Seismic Category 1 of ASME Section III requirements (A-11) or have been evaluated to be rugged as described in Regulatory Position A-5.7. The amount of reduction allowed will be evaluated on an individual case basis.

A-5.7 Licensees who have already evaluated the seismic ruggedness of the steamlines, alternate drain paths, and the main condensers, and who have obtained prior staff approval, may credit the piping addressed in that approval. Also, licensees that have not previously applied for such approval may do so in accordance with the guidance in Reference A-12 for establishing a seismically rugged alternative drain path.

A-5.8 Section 6.3 of Reference A-10 describes an acceptable model for estimating the aerosol deposition in horizontal piping. From the start of the accident to the termination of the early in-vessel release phase, the amount of reduction in the steamline is determined by the removal coefficients in Table 6-2 of Reference A-10. After the early in-vessel release phase ends, the removal coefficients are given by the values in Table 6-1 of Reference A-10.[4]

For BWR designs other than plants with Mark I, II, or III containment design, other models of aerosol deposition in piping will be considered on a case-by-case basis.

A-5.9 Reduction of the amount of released elemental iodine by plateout deposition on steam system piping may be credited, but the amount of reduction in concentration allowed will be evaluated on an individual case basis. The model should be based on the assumption of well-mixed volumes. Reference A-13 provides guidance on an acceptable model.

A-5.10 Reduction of the amount of released organic iodine (e.g., Brockman-Bixler model in RADTRAD (Ref. A-3)) should not be credited.

A-5.11 In the absence of collection and treatment of releases by ESFs such as the MSIV leakage control system, or as described in Regulatory Position A-5.6, above, the MSIV leakage should be assumed to be released to the environment as an unprocessed, ground-level release. Holdup and dilution in the turbine building should not be assumed.

A-6. Containment Purging

The licensee should analyze the radiological consequences from post-LOCA primary containment purging as a combustible gas or pressure control measure. If the installed containment purging capabilities are maintained for purposes of severe accident management and are not credited in any design basis analysis, radiological consequences need not be evaluated. If the primary containment purging is

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4 A removal coefficient of 0.0 hr^{-1} should be used for the removal coefficient for the in-board piping as described in the footnotes for Tables 6-1 and 6-2 of Ref. A-10.

required within 30 days of the LOCA, the results of this analysis should be combined with consequences postulated for other fission product release paths to determine the total calculated radiological consequences from the LOCA. The licensee may take into account the reduction in the amount of radioactive material released via ESF filter systems provided that these systems meet the guidance in Regulatory Guide 1.52 (Ref. A-5).

APPENDIX A

REFERENCES

A-1. U.S. Nuclear Regulatory Commission, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants," NUREG-0800.

A-2. D.A. Powers et al, "A Simplified Model of Aerosol Removal by Natural Processes in Reactor Containments," NUREG/CR-6189, U.S. Nuclear Regulatory Commission, July 1996.

A-3. S.L. Humphreys et al., "RADTRAD: A Simplified Model for Radionuclide Transport and Removal and Dose Estimation," NUREG/CR-6604, U.S. Nuclear Regulatory Commission, April 1998, and Supplement 1, SAND 98-0272/1, June 1999 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML081850607).

A-4. D.A. Powers and S.B. Burson, "A Simplified Model of Aerosol Removal by Containment Sprays," NUREG/CR-5966, U.S. Nuclear Regulatory Commission, June 1993.

A-5. U.S. Nuclear Regulatory Commission, "Design, Inspection, and Testing Criteria for Air Filtration and Adsorption Units of Post-Accident Engineered-Safety-Feature Atmospheric Cleanup Systems in Light-Water-Cooled Nuclear Power Plants," Regulatory Guide 1.52, Revision 3, June 2001.

A-6. U.S. Nuclear Regulatory Commission, "Potential Radioactive Leakage to Tank Vented to Atmosphere," Information Notice 91-56, September 19, 1991.

A-7. U.S. Nuclear Regulatory Commission, "Recent Discovery of a Phenomenon Not Previously Considered in the Design of Secondary Containment Pressure Control," Information Notice 88-76, September 19, 1988.

A-8 U.S. Nuclear Regulatory Commission, "Clarification of TMI Action Plan Requirements," NUREG-0737, November 1980.

A-9. L. Soffer et al., "Accident Source Terms for Light-Water Nuclear Power Plants," NUREG-1465, U.S. Nuclear Regulatory Commission, February 1995.

A-10. "Analysis of Main Steam Isolation Valve Leakage in Design Basis Accidents Using MELCOR 1.8.6 and RADTRAD," Sandia Letter Report, SAND2008-6601, October 2008 (ADAMS Accession No. ML083180196).

A-11. ASME Boiler & Pressure Vessel Code, Section III, "Nuclear Power Plant Components," American Society of Mechanical Engineers, New York, New York.

A-12. U.S. Nuclear Regulatory Commission, "Safety Evaluation of GE Topical Report, NEDC-31858P (Proprietary GE report), Revision 2, BWROG Report for Increasing MSIV Leakage Limits and Elimination of Leakage Control Systems, September 1993," letter dated March 3, 1999, (ADAMS Accession No. ML010640286, NUDOCS Accession No. 9903110303).

A-13. J.E. Cline, "MSIV Leakage Iodine Transport Analysis," Letter Report dated March 26, 1991, (ADAMS Accession No. ML003683718).

APPENDIX B

ASSUMPTIONS FOR EVALUATING THE RADIOLOGICAL CONSEQUENCES OF A FUEL HANDLING ACCIDENT

This appendix provides assumptions acceptable to the staff of the U.S. Nuclear Regulatory Commission (NRC) for evaluating the radiological consequences of a fuel handling accident at light-water reactors. These assumptions supplement the guidance provided in the main body of this guide.[1]

B-1. Source Term

Regulatory Position 3 of this guide provides acceptable assumptions regarding core inventory and the release of radionuclides from the fuel. The following assumptions also apply:

B-1.1 The number of fuel rods damaged during the accident should be based on a conservative analysis that considers the most limiting case. This analysis should consider parameters such as the weight of the dropped heavy load or the weight of a dropped fuel assembly (plus any attached handling grapples), the height of the drop, and the compression, torsion, and shear stresses on the irradiated fuel rods. The analysis should also consider damage to adjacent fuel assemblies, if applicable (e.g., events over the reactor vessel).

B-1.2 The fission product release from the breached fuel is based on Regulatory Position 3.2 of this guide and the estimate of the number of fuel rods breached. All the gap activity in the damaged rods is assumed to be instantaneously released. Radionuclides that should be considered include xenons, kryptons, halogens, cesiums, and rubidiums.

B-1.3 The chemical form of radioiodine released from the fuel to the spent fuel pool should be assumed to be 95 percent cesium iodide (CsI), 4.85 percent elemental iodine, and 0.15 percent organic iodide. The CsI released from the fuel is assumed to completely dissociate in the pool water. Because of the low pH of the pool water, the iodine reevolves as elemental iodine. This is assumed to occur instantaneously. The NRC staff will consider, on a case-by-case basis, justifiable mechanistic treatment of the iodine release from the pool.

B-1.4 The radioactive material available for release is assumed to be from the assemblies with the peak inventory. The fission product inventory for the peak assembly represents an upper limit value. The inventory should be calculated assuming the maximum achievable operational power history and burnup. These parameters should be examined to maximize fission product inventory. This inventory calculation should include appropriate assembly peaking factors.

2. Water Depth

If the depth of water above the damaged fuel is 23 feet or greater, an overall effective decontamination factor (DF) of 200 (i.e., 99.5 percent of the total iodine released from the damaged rods

[1] These assumptions may also be used in assessing the radiological consequences of a heavy load drop over fuel accident. If the event is postulated to damage all of the rods in the core, the release activity may be based on the core-average gap fractions of Tables 1 and 2, and the radial peaking factor may be omitted.

is retained by the water) may be assumed. The difference in DFs for elemental (99.85 percent) and organic (0.15 percent) iodine species results in the iodine above the water that is composed of 70 percent elemental and 30 percent organic species. If the depth of water is not at least 23 feet, the DF will have to be determined on a case-by-case method (Ref. B-1). Proposed increases in the pool DF above 200 will need to address reevolution of the scrubbed iodine species over the accident duration and should be supported by empirical data. For release pressures greater than 1,200 pounds per square inch gauge, the iodine DFs will be less than those assumed in this guide and must be calculated on a case-by-case basis using assumptions comparable in conservatism to those of this guide.

B-3. Noble Gases and Particulates

The retention of noble gases in the water in the fuel pool or reactor cavity is negligible (i.e., DF of 1). Particulate radionuclides are assumed to be retained by the water in the fuel pool or reactor cavity (i.e., infinite DF).

B-4. Fuel handling Accidents within the Fuel Building

For fuel handling accidents postulated to occur within the fuel building, the following assumptions are acceptable to the NRC staff:

B-4.1 The radioactive material that escapes from the fuel pool to the fuel building is assumed to be released to the environment over a 2-hour time period. The release rate is generally assumed to be a linear or exponential function over this time period.

B-4.2 A reduction in the amount of radioactive material released from the fuel pool by engineered safety feature (ESF) filter systems may be taken into account, provided these systems meet the guidance of Regulatory Guide 1.52, Revision 3, "Design, Inspection, and Testing Criteria for Air Filtration and Adsorption Units of Post-Accident Engineered-Safety-Feature Atmosphere Cleanup Systems in Light-Water-Cooled Nuclear Power Plants" (Ref. B-2). The radioactivity release analyses should determine and account for delays in radiation detection, actuation of the ESF filtration system, or diversion of ventilation flow to the ESF filtration system.[2]

B-4.3 The radioactivity release from the fuel pool should be assumed to be drawn into the ESF filtration system without mixing or dilution in the fuel building. If mixing can be demonstrated, credit for mixing and dilution may be considered on a case-by-case basis. This evaluation should consider the magnitude of the building volume and exhaust rate, the potential for bypass to the environment, the location of exhaust plenums relative to the surface of the pool, recirculation ventilation systems, and internal walls and floors that impede stream flow between the surface of the pool and the exhaust plenums.

B-5. Fuel handling Accidents within Containment

For fuel handling accidents postulated to occur within the containment, the following assumptions are acceptable to the NRC staff:

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[2] These analyses should consider the time for the radioactivity concentration to reach levels corresponding to the monitor setpoint, instrument line sampling time, detector response time, diversion damper alignment time, and filter system actuation, as applicable.

B-5.1 If the containment is isolated[3] during fuel handling operations, no radiological consequences need to be analyzed.

B-5.2 If the containment is open during fuel handling operations, but designed to automatically isolate in the event of a fuel handling accident, the release duration should be based on delays in radiation detection and completion of containment isolation. If it can be shown that containment isolation occurs before radioactivity is released to the environment,[2] no radiological consequences need to be analyzed for the isolated pathway.

B-5.3 If the containment is open during fuel handling operations (e.g., personnel air lock or equipment hatch is open),[4] the radioactive material that escapes from the reactor cavity pool to the containment is released to the environment over a 2-hour time period. The release rate is generally assumed to be a linear or exponential function over this time period.

B-5.4 A reduction in the amount of radioactive material released from the containment by ESF filter systems may be taken into account provided that these systems meet the guidance of Regulatory Guide 1.52 (Ref. B-2). The radioactivity release analyses should determine and account for delays in radiation detection, actuation of the ESF filtration system, or diversion of ventilation flow to the ESF filtration system.[2]

B-5.5 Credit for dilution or mixing of the activity released from the reactor cavity by natural or forced convection inside the containment may be considered on a case-by-case basis. Such credit is generally limited to 50 percent of the containment free volume. This evaluation should consider the magnitude of the containment volume and exhaust rate, the potential for bypass to the environment, the location of exhaust plenums relative to the surface of the reactor cavity, recirculation ventilation systems, and internal walls and floors that impede streamflow between the surface of the reactor cavity and the exhaust plenums.

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3 Containment isolation does not imply containment integrity as defined by technical specifications for nonshutdown modes. The term isolation is used here collectively to encompass both containment integrity and containment closure, typically in place during shutdown periods. To be credited in the analysis, the technical specifications should address the appropriate form of isolation.

4 Technical specifications that allow such operations usually include administrative controls to close the airlock, hatch, or open penetrations within 30 minutes. Such administrative controls generally require that a dedicated individual be present, with necessary equipment available, to restore containment closure should a fuel handling accident occur. Radiological analyses should generally not credit this manual isolation.

APPENDIX B

REFERENCES

B-1. G. Burley, "Evaluation of Fission Product Release and Transport," Staff Technical Paper, 1971. (NUDOCS Accession No. 8402080322)

B-2. U.S. Nuclear Regulatory Commission, "Design, Inspection, and Testing Criteria for Air Filtration and Adsorption Units of Post-Accident Engineered-Safety-Feature Atmospheric Cleanup Systems in Light-Water-Cooled Nuclear Power Plants," Regulatory Guide 1.52, Revision 3, June 2001.

APPENDIX C

ASSUMPTIONS FOR EVALUATING THE RADIOLOGICAL CONSEQUENCES OF A BOILING-WATER REACTOR ROD DROP ACCIDENT

This appendix provides assumptions acceptable to the staff of the U.S. Nuclear Regulatory Commission (NRC) for evaluating the radiological consequences of a rod drop accident at boiling-water reactors. These assumptions supplement the guidance provided in the main body of this guide.

C-1. Regulatory Position 3 of this guide provides assumptions acceptable to the NRC staff regarding core inventory. The fission product release from the breached fuel to the coolant is based on Regulatory Position 3.2 of this guide and the estimate of the number of fuel rods breached.

C-2. If no or minimal[1] fuel breach is postulated for the limiting event, the released activity should be the maximum coolant activity (typically a preaccident spike of 4 microcuries/gram (μCi/gm) dose equivalent iodine-131 (DE I-131)) allowed by the technical specifications.

C-3. The assumptions acceptable to the NRC staff related to the transport, reduction, and release of radioactive material from the fuel and the reactor coolant are as follows:

C-3.1 The activity released from the fuel from either the gap and/or from fuel pellets is assumed to be instantaneously mixed in the reactor coolant within the pressure vessel.

C-3.2 Credit should not be assumed for partitioning in the pressure vessel or for removal by the steam separators.

C-3.3 Of the activity released from the reactor coolant within the pressure vessel, 100 percent of the noble gases, 10 percent of the iodine, and 1 percent of the remaining radionuclides are assumed to reach the turbine and condensers.

C-3.4 Of the activity that reaches the turbine and condenser, 100 percent of the noble gases, 10 percent of the iodine, and 1 percent of the particulate radionuclides are available for release to the environment. The turbine and condensers leak to the environment as a ground-level release at a rate of 1 percent per day[2] for a period of 24 hours, at which time the leakage is assumed to terminate. No credit should be assumed for dilution or holdup within the turbine building. Radioactive decay during holdup in the turbine and condenser may be assumed.

C-3.5 In lieu of the transport assumptions provided in Regulatory Positions C-3.2 through C-3.4 above, a more mechanistic analysis may be used on a case-by-case basis. Such analyses account for the

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1 Minimal fuel breach is defined as an amount of damage that will yield reactor coolant system activity concentration levels less than the maximum technical specification limits. The activity assumed in the analysis should be based on the activity associated with the projected fuel breach or the maximum technical specification values, whichever maximizes the radiological consequences. In determining the dose equivalent iodine-131 (DE I-131), only the radioiodine associated with normal operations or iodine spikes should be included. Activity from projected fuel damage should not be included.

2 If there are forced flowpaths from the turbine or condenser, such as unisolated motor vacuum pumps or unprocessed air ejectors, the leakage rate should be assumed to be the flow rate associated with the most limiting of these paths. Credit for collection and processing of releases, such as by off gas or standby gas treatment, will be considered on a case-by-case basis.

quantity of contaminated steam carried from the pressure vessel to the turbine and condensers based on a review of the minimum transport time from the pressure vessel to the first main steam isolation valve (MSIV) and MSIV closure time.

C-3.6 The iodine species released from the reactor coolant within the pressure vessel should be assumed to be 95 percent cesium iodide as an aerosol, 4.85 percent elemental iodine, and 0.15 percent organic iodide. The release from the turbine and condenser should be assumed to be 97 percent elemental and 3 percent organic.

APPENDIX D

ASSUMPTIONS FOR EVALUATING THE RADIOLOGICAL CONSEQUENCES OF A BOILING-WATER REACTOR MAIN STEAMLINE BREAK ACCIDENT

This appendix provides assumptions acceptable to the staff of the U.S. Nuclear Regulatory Commission (NRC) for evaluating the radiological consequences of a main steamline accident at boiling-water reactor (BWR) light-water reactors. These assumptions supplement the guidance provided in the main body of this guide.

Source Term

D-1. Regulatory Position 3 of this guide provides assumptions acceptable to the NRC staff regarding core inventory and the release of radionuclides from the fuel. The release from the breached fuel is based on Regulatory Position 3.2 of this guide and the estimate of the number of fuel rods breached.

D-2. If no or minimal[1] fuel damage is postulated for the limiting event, the released activity should be the maximum coolant activity allowed by technical specification. The iodine concentration in the primary coolant is assumed to correspond to the following two cases in the nuclear steam supply system vendor's standard technical specifications:

D-2.1 The concentration that is the maximum value (typically 4.0 microcuries per gram (μCi/gm) dose equivalent iodine-131 (DE I-131)) permitted and corresponds to the conditions of an assumed pre-accident spike, and

D-2.2 The concentration that is the maximum equilibrium value (typically 0.2 μCi/gm DE I-131) permitted for continued full power operation.

D-3. The activity released from the fuel should be assumed to mix instantaneously and homogeneously in the reactor coolant. Noble gases should be assumed to enter the steam phase instantaneously. The release from the breached fuel is based on Regulatory Position 3.2 of this guide and the estimate of the number of fuel rods breached. Noble gases should be assumed to enter the steam phase instantaneously.

Transport

D-4. Assumptions acceptable to the NRC staff related to the transport, reduction, and release of radioactive material to the environment are as follows:

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[1] Minimal fuel breach is defined as an amount of damage that will yield reactor coolant system activity concentration levels less than the maximum technical specification limits. The activity assumed in the analysis should be based on the activity associated with the projected fuel damage or the maximum technical specification values, whichever maximizes the radiological consequences. In determining dose equivalent I-131 (DE I-131), only the radioiodine associated with normal operations or iodine spikes should be included. Activity from projected fuel damage should not be included.

D-4.1 The main steamline isolation valves should be assumed to close in the maximum time allowed by technical specifications.

D-4.2 The total mass of coolant released should be assumed to be that amount in the steamline and connecting lines at the time of the break plus the amount that passes through the valves before closure.

D-4.3 All radioactivity in the released coolant should be assumed to be released to the environment instantaneously as a ground-level release. No credit should be assumed for plateout, holdup, or dilution within facility buildings.

D-4.4 The iodine species released from the main steamline should be assumed to be 95 percent cesium iodide as an aerosol, 4.85 percent elemental iodine, and 0.15 percent organic iodide.

APPENDIX E

ASSUMPTIONS FOR EVALUATING THE RADIOLOGICAL CONSEQUENCES OF A PRESSURIZED-WATER REACTOR STEAM GENERATOR TUBE RUPTURE ACCIDENT

This appendix provides assumptions acceptable to the staff of the U.S. Nuclear Regulatory Commission (NRC) for evaluating the radiological consequences of a steam generator tube rupture accident at pressurized-water reactors. These assumptions supplement the guidance provided in the main body of this guide.[1]

Source Term

E-1. Regulatory Position 3 of this guide provides assumptions acceptable to the NRC staff regarding core inventory and the release of radionuclides from the fuel.

E-2. If no or minimal[2] fuel damage is postulated for the limiting event, the activity released should be the maximum coolant activity allowed by technical specification. Two cases of iodine spiking should be assumed:

E-2.1 A reactor transient has occurred before the postulated steam generator tube rupture (SGTR) and has raised the primary coolant iodine concentration to the maximum value (typically 60 microcuries per gram (μCi/gm) dose equivalent iodine-131 (DE I-131)) permitted at full-power operations by the technical specifications (i.e., a preaccident iodine spike case).

E-2.2 The primary system transient associated with the SGTR causes an iodine spike in the primary system. The increase in primary coolant iodine concentration is estimated using a spiking model that assumes that the iodine release rate from the fuel rods to the primary coolant (expressed in curies per unit time) increases to a value 335 times greater than the release rate corresponding to the iodine concentration at the equilibrium value (typically 1.0 μCi/gm DE I-131) specified in technical specifications (i.e., concurrent iodine spike case). A concurrent iodine spike need not be considered if fuel damage is postulated. The assumed iodine spike duration should be 8 hours. Shorter spike durations may be considered on a case-by-case basis if it can be shown that the activity released by the 8-hour spike exceeds that available for release from the fuel pins assumed to have defects.

E-3. The activity released from the fuel, if any, should be assumed to be released instantaneously and homogeneously through the primary coolant. The release from the breached fuel is based on Regulatory Position 3.2 of this guide and the estimate of the number of fuel rods breached.

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1 Facilities licensed with, or applying for, alternative repair criteria should use this section in conjunction with the guidance that is being developed in Draft Regulatory Guide DG-1074, "Steam Generator Tube Integrity," issued December 1998, for acceptable assumptions and methodologies for performing radiological analyses.

2 Minimal fuel breach is defined as an amount of damage that will yield reactor coolant system activity concentration levels less than the maximum technical specification limits. The activity assumed in the analysis should be based on the activity associated with the projected fuel damage or the maximum technical specification values, whichever maximizes the radiological consequences. In determining dose equivalent I-131 (DE I-131), only the radioiodine associated with normal operations or iodine spikes should be included. Activity from projected fuel damage should not be included.

E-4. The specific activity in the steam generator liquid at the onset of the SGTR is at the maximum value permitted by secondary activity technical specifications (typically 0.1 μCi/gm).

E-5. Iodine releases from the steam generators to the environment should be assumed to be 97 percent elemental iodine and 3 percent organic iodide.

Transport

E-6. Assumptions acceptable to the NRC staff related to the transport, reduction, and release of radioactive material to the environment are as follows:

E-6.1 The primary-to-secondary leak rate in the steam generators should be assumed to be the leak rate limiting condition for operation specified in the technical specifications. The leakage should be apportioned between affected and unaffected steam generators in such a manner that the calculated dose is maximized.

E-6.2 The density used in converting volumetric leak rates (e.g., gallons per minute) to mass leak rates (e.g., pound mass per hour) should be consistent with the basis of surveillance tests used to show compliance with leak rate technical specifications. These tests are typically based on cool liquid. Facility instrumentation used to determine leakage is typically located on lines containing cool liquids. In most cases, the density should be assumed to be 1.0 gram per cubic centimeter (62.4 pounds mass per cubic foot).

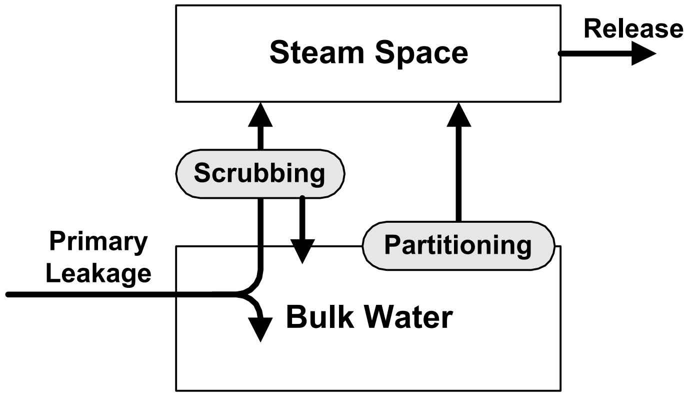
E-6.3 The primary-to-secondary leakage should be assumed to continue until the primary system pressure is less than the secondary system pressure, or until the temperature of the leakage is less than 100 degrees Celsius (212 degrees Fahrenheit). The release of radioactivity from the unaffected steam generators should be assumed to continue until shutdown cooling is in operation and releases from the steam generators have been terminated. The release of radioactivity from the affected steam generator should be assumed to continue until shutdown cooling is operating and releases from the steam generator have been terminated, or the steam generator is isolated from the environment such that no release is possible, whichever occurs first.

E-6.4 The release of fission products from the secondary system should be evaluated with the assumption of a coincident loss of offsite power.

E-6.5 All noble gas radionuclides released from the primary system should be assumed to be released to the environment without reduction or mitigation.

E-6.6 The transport model described in this section should be utilized for iodine and particulate releases from the steam generators. Figure E-1 illustrates this model which is summarized below:

Figure E-1
Transport Model



E-6.6.1 A portion of the primary-to-secondary leakage will flash to vapor, based on the thermodynamic conditions in the reactor and secondary coolant.

- During periods of steam generator dryout, all of the primary-to-secondary leakage is assumed to flash to vapor and be released to the environment with no mitigation.

- With regard to the unaffected steam generators used for plant cooldown, the primary-to-secondary leakage can be assumed to mix with the secondary water without flashing during periods of total tube submergence. During periods of uncovery, a flash fraction should be determined.

E-6.6.2 The leakage that immediately flashes to vapor will rise through the bulk water of the steam generator and enter the steam space. Credit may be taken for scrubbing in the generator, using the models in NUREG-0409, “Iodine Behavior in a PWR Cooling System Following a Postulated Steam Generator Tube Rupture Accident” (Ref. E-1), during periods of total submergence of the tubes.

E-6.6.3 The leakage that does not immediately flash is assumed to mix with the bulk water.

E-6.6.4 The radioactivity in the bulk water is assumed to become vapor at a rate that is the function of the steaming rate and the partition coefficient.[3] A partition coefficient for iodine of 100 may be assumed. The retention of noniodine particulate radionuclides in the steam generators is limited by the moisture carryover from the steam generators.

E-6.7 Operating experience and analyses have shown that for some steam generator designs, tube uncovery may occur for a short period following any reactor trip (Ref. E-2). The potential impact of tube uncovery on the transport model parameters (e.g., flash fraction, scrubbing credit) should be considered. The impact of emergency operating procedure restoration strategies on steam generator water levels should be evaluated.

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[3] “Partition coefficient” is defined as follows:

$$PC = \frac{mass\ \ of\, I_2\ per\ unit\ mass\ of\ liquid}{mass\ \ of\, I_2\ per\ unit\ mass\ of\ gas}$$

APPENDIX E

REFERENCES

E-1. U.S. Nuclear Regulatory Commission, "Iodine Behavior in a PWR Cooling System Following a Postulated Steam Generator Tube Rupture Accident," NUREG-0409, January 1978.

E-2. U.S. Nuclear Regulatory Commission, "Steam Generator Tube Rupture Analysis Deficiency," Information Notice 88-31, May 25, 1988.

APPENDIX F

ASSUMPTIONS FOR EVALUATING THE RADIOLOGICAL CONSEQUENCES OF A PRESSURIZED-WATER REACTOR MAIN STEAMLINE BREAK ACCIDENT

This appendix provides assumptions acceptable to the staff of the U.S. Nuclear Regulatory Commission (NRC) for evaluating the radiological consequences of a main steamline break accident at pressurized-water reactors. These assumptions supplement the guidance provided in the main body of this guide.[1]

Source Term

F-1. Regulatory Position 3 of this regulatory guide provides assumptions acceptable to the NRC staff regarding core inventory and the release of radionuclides from the fuel.

F-2. If no or minimal[2] fuel damage is postulated for the limiting event, the activity released should be the maximum coolant activity allowed by the technical specifications. Two cases of iodine spiking should be assumed:

F-2.1 A reactor transient has occurred before the postulated mainsteam line break (MSLB) and has raised the primary coolant iodine concentration to the maximum value (typically 60 microcuries per gram (µCi/gm) dose equivalent iodine-131 (DE I-131)) permitted by the technical specifications (i.e., a preaccident iodine spike case).

F-2.2 The primary system transient associated with the MSLB causes an iodine spike in the primary system. The increase in primary coolant iodine concentration is estimated using a spiking model that assumes that the iodine release rate from the fuel rods to the primary coolant (expressed in curies per unit time) increases to a value 500 times greater than the release rate corresponding to the iodine concentration at the equilibrium value (typically 1.0 µCi/gm DE I-131) specified in technical specifications (i.e., concurrent iodine spike case). A concurrent iodine spike need not be considered if fuel damage is postulated. The assumed iodine spike duration should be 8 hours. Shorter spike durations may be considered on a case-by-case basis if it can be shown that the activity released by the 8-hour spike exceeds that available for release from the fuel gap assumed to have defects.

F-3. The activity released from the fuel should be assumed to be released instantaneously and homogeneously through the primary coolant. The release from the breached fuel is based on Regulatory Position 3.2 of this guide and the estimate of the number of fuel rods breached.

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1 Facilities licensed with, or applying for, alternative repair criteria (ARC) should use this section in conjunction with the guidance that is being developed in Draft Regulatory Guide DG-1074, "Steam Generator Tube Integrity," issued December 1998 (Ref. F-1), for acceptable assumptions and methodologies for performing radiological analyses.

2 Minimal fuel breach is defined as an amount of damage that will yield reactor coolant system activity concentration levels less than the maximum technical specification limits. The activity assumed in the analysis should be based on the activity associated with the projected fuel damage or the maximum technical specification values, whichever maximizes the radiological consequences. In determining dose equivalent I-131 (DE I-131), only the radioiodine associated with normal operations or iodine spikes should be included. Activity from projected fuel damage should not be included.

F-4. The specific activity in the steam generator liquid at the onset of the MSLB should be assumed to be at the maximum value permitted by secondary activity technical specifications (typically 0.1 µCi/gm DE I-131).

F-5. Iodine releases from the steam generators to the environment should be assumed to be 97 percent elemental iodine and 3 percent organic iodide. These fractions apply to iodine released as a result of fuel damage and to iodine released during normal operations, including iodine spiking.

Transport

F-6. Assumptions acceptable to the NRC staff related to the transport, reduction, and release of radioactive material to the environment are as follows:

F-6.1 The bulk water in the faulted[3] steam generator is assumed to rapidly blow down to the environment. The duration of the blowdown is obtained from thermal-hydraulic analysis codes. The activity in the faulted steam generator bulk water is assumed released to the environment without mitigation.

F-6.2 For facilities that have not implemented alternative repair criteria (ARC) (see Ref. F-1), the primary-to-secondary leak rate in the steam generators should be assumed to be the leak rate limiting condition for operation specified in the technical specifications. For facilities with traditional steam generator specifications (both per generator and total of all generators), the leakage should be apportioned between faulted and unaffected steam generators in such a manner that the calculated dose is maximized. For example, for a four-loop facility with a limiting condition for operation of 500 gallons per day (gpd) for any one generator not to exceed 1 gallon per minute (gpm) from all generators, it would be appropriate to assign 500 gpd to the faulted generator and 313 gpd to each of the unaffected generators.

For facilities that have implemented ARC, the primary-to-secondary leak rate in the faulted steam generator should be assumed to be the maximum accident-induced leakage derived from the repair criteria and burst correlations. For the unaffected steam generators, the leak rate limiting condition for operation specified in the technical specifications is equally apportioned between the unaffected steam generators.

F-6.3 The density used in converting volumetric leak rates (e.g., gpm) to mass leak rates (e.g., pound mass per hour) should be consistent with the basis of the parameter being converted. The ARC leak rate correlations are generally based on the collection of cooled liquid. Surveillance tests and facility instrumentation used to show compliance with leak rate technical specifications are typically based on cooled liquid. In most cases, the density should be assumed to be 1.0 gram per cubic centimeter (62.4 pounds mass per cubic foot).

F-6.4 The primary-to-secondary leakage should be assumed to continue until the primary system pressure is less than the secondary system pressure, or until the temperature of the leakage is less than 100 degrees Celsius (212 degrees Fahrenheit). The release of radioactivity from unaffected

[3] "Faulted" refers to the state of the steam generator in which the secondary side has been depressurized by a MSLB such that protective system response (main steam line isolation, reactor trip, safety injection, etc.) has occurred.

steam generators should be assumed to continue until shutdown cooling is in operation and releases from the steam generators have been terminated.

F-6.5 All noble gas radionuclides released from the primary system are assumed to be released to the environment without reduction or mitigation.

F-6.6 The transport model described in this section should be utilized for iodine and particulate releases from the steam generators.

F-6.6.1 The primary-to-secondary leakage to the faulted steam generator is assumed to flash to vapor and be released to the environment with no mitigation.

F-6.6.2 With regard to the unaffected steam generators used for plant cooldown, the primary-to-secondary leakage can be assumed to mix with the secondary water without flashing during periods of total tube submergence. If the tubes are uncovered, a portion of the primary-to-secondary leakage will flash to vapor, based on the thermodynamic conditions in the reactor and secondary coolant.

- The leakage that immediately flashes to vapor will rise through the bulk water of the steam generator and enter the steam space. Credit may be taken for scrubbing in unaffected generators, using the models in NUREG-0409, "Iodine Behavior in a PWR Cooling System Following a Postulated Steam Generator Tube Rupture Accident" issued May 1985 (Ref. F-2), during periods of total submergence of the tubes.

- The leakage to the unaffected generators that does not immediately flash is assumed to mix with the bulk water.

- The radioactivity in the bulk water of the unaffected generators is assumed to become vapor at a rate that is the function of the steaming rate and the partition coefficient. A partition coefficient[4] for iodine of 100 may be assumed. The retention of particulate radionuclides in the steam generators is limited by the moisture carryover from the steam generators.

F-6.7 Operating experience and analyses have shown that for some steam generator designs, tube uncovery may occur for a short period following any reactor trip (Ref. F-3). The potential impact of tube uncovery on the transport model parameters (e.g., flash fraction, scrubbing credit) needs to be considered. The impact of emergency operating procedure restoration strategies on steam generator water levels should be evaluated.

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4 "Partition coefficient" is defined as follows:

$$PC = \frac{mass\ \ of\, I_2\ per\ unit\ mass\ of\ liquid}{mass\ \ of\, I_2\ per\ unit\ mass\ of\ gas}$$

APPENDIX F

REFERENCES

F-1. U.S. Nuclear Regulatory Commission, “Steam Generator Tube Integrity,” Draft Regulatory Guide DG-1074, December 1998.

F-2. U.S. Nuclear Regulatory Commission, “Iodine Behavior in a PWR Cooling System Following a Postulated Steam Generator Tube Rupture Accident,” NUREG-0409, May 1985.

F-3. U.S. Nuclear Regulatory Commission, “Steam Generator Tube Rupture Analysis Deficiency,” Information Notice 88-31, May 25, 1988.

APPENDIX G

ASSUMPTIONS FOR EVALUATING THE RADIOLOGICAL CONSEQUENCES OF A PRESSURIZED-WATER REACTOR LOCKED ROTOR ACCIDENT

This appendix provides assumptions acceptable to the staff of the U.S. Nuclear Regulatory Commission (NRC) for evaluating the radiological consequences of a locked rotor accident at pressurized-water reactors.[1] These assumptions supplement the guidance provided in the main body of this guide.

Source Term

G-1. Regulatory Position 3 of this regulatory guide provides assumptions acceptable to the NRC staff regarding core inventory and the release of radionuclides from the fuel. The release from the breached fuel is based on Regulatory Position 3.2 of this guide and the estimate of the number of fuel rods breached.

G-2. If no fuel damage is postulated for the limiting event, a radiological analysis is not required as the consequences of this event are bounded by the consequences projected for the main steamline break outside containment.

G-3. The activity released from the fuel should be assumed to be released instantaneously and homogeneously through the primary coolant.

G-4. The chemical form of radioiodine released from the fuel should be assumed to be 95 percent cesium iodide (CsI), 4.85 percent elemental iodine, and 0.15 percent organic iodide. Iodine releases from the steam generators to the environment should be assumed to be 97 percent elemental and 3 percent organic. These fractions apply to iodine released as a result of fuel damage and to iodine released during normal operations, including iodine spiking.

Release Transport

G-5. Assumptions acceptable to the NRC staff related to the transport, reduction, and release of radioactive material to the environment are as follows:

G-5.1 The primary-to-secondary leak rate in the steam generators should be assumed to be the leak-rate-limiting condition for operation specified in the technical specifications. The leakage should be apportioned between the steam generators in such a manner that the calculated dose is maximized.

G-5.2 The density used in converting volumetric leak rates (e.g., gallons per minute) to mass leak rates (e.g., pound mass per hour) should be consistent with the basis of surveillance tests used to show compliance with leak rate technical specifications. These tests are typically based on cool liquid. Facility instrumentation used to determine leakage is typically located on lines containing cool

[1] Facilities licensed with, or applying for, alternative repair criteria (ARC) should use this section in conjunction with the guidance that is being developed in Draft Regulatory Guide DG-1074, "Steam Generator Tube Integrity," issued December 1998, for acceptable assumptions and methodologies for performing radiological analyses.

liquids. In most cases, the density should be assumed to be 1.0 gram per cubic centimeter (62.4 pounds mass per cubic foot).

G-5.3 The primary-to-secondary leakage should be assumed to continue until the primary system pressure is less than the secondary system pressure, or until the temperature of the leakage is less than 100 degrees Celsius (212 degrees Fahrenheit). The release of radioactivity should be assumed to continue until shutdown cooling is in operation and releases from the steam generators have been terminated.

G-5.4 The release of fission products from the secondary system should be evaluated with the assumption of a coincident loss of offsite power.

G-5.5 All noble gas radionuclides released from the primary system are assumed to be released to the environment without reduction or mitigation.

G-5.6 The transport model described in Regulatory Position E-6.6 and E-6.7 of Appendix E to this guide should be utilized for iodine and particulates.

APPENDIX H

ASSUMPTIONS FOR EVALUATING THE RADIOLOGICAL CONSEQUENCES OF A PRESSURIZED-WATER REACTOR CONTROL ROD EJECTION ACCIDENT

This appendix provides assumptions acceptable to the staff of the U.S Nuclear Regulatory Commission (NRC) for evaluating the radiological consequences of a control rod ejection accident at pressurized-water reactors.[1] These assumptions supplement the guidance provided in the main body of this guide. Two release paths are considered: (1) release via containment leakage and (2) release via the secondary plant. Each release path is evaluated independently as if it were the only pathway available. The consequences of this event are acceptable if the dose from each path considered separately is less than the acceptance criterion in Table 7.

Source Term

H-1. Regulatory Position 3 of this guide provides assumptions acceptable to the NRC staff regarding core inventory. The fission product release from the breached fuel to the coolant is based on Regulatory Position 3.2 of this guide and the estimate of the number of fuel rods breached.

H-2. If no fuel breach is postulated for the limiting event, a radiological analysis is not required as the consequences of this event are bounded by the consequences projected for the loss-of-coolant accident, main steamline break, and steam generator tube rupture.

H-3. In the first release case, 100 percent of the activity released from the fuel should be assumed to be released instantaneously and homogeneously through the containment atmosphere. In the second case, 100 percent of the activity released from the fuel should be assumed to be completely dissolved in the primary coolant and available for release to the secondary system.

H-4. The chemical form of radioiodine released to the containment atmosphere should be assumed to be 95 percent cesium iodide (CsI), 4.85 percent elemental iodine, and 0.15 percent organic iodide. If containment sprays do not actuate or are terminated before accumulating sump water, or if the containment sump pH is not controlled at values of 7 or greater, the iodine species should be evaluated on an individual case basis. Evaluations of pH should consider the effect of acids created during the control rod ejection accident event (e.g., pyrolysis and radiolysis products). With the exception of elemental and organic iodine and noble gases, fission products should be assumed to be in particulate form.

H-5. Iodine releases from the steam generators to the environment should be assumed to be 97 percent elemental iodine and 3 percent organic iodide.

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[1] Facilities licensed with, or applying for, alternative repair criteria should use this section in conjunction with the guidance that is being developed in Draft Regulatory Guide DG-1074, "Steam Generator Tube Integrity," issued December 1998, for acceptable assumptions and methodologies for performing radiological analyses.

Transport from Containment

H-6. Assumptions acceptable to the NRC staff related to the transport, reduction, and release of radioactive material in and from the containment are as follows:

H-6.1 A reduction in the amount of radioactive material available for leakage from the containment that is due to natural deposition, containment sprays, recirculating filter systems, dual containments, or other engineered safety features may be taken into account. Refer to Appendix A to this guide for guidance on acceptable methods and assumptions for evaluating these mechanisms.

H-6.2 The containment should be assumed to leak at the leak rate incorporated in the technical specifications at peak accident pressure for the first 24 hours, and at 50 percent of this leak rate for the remaining duration of the accident. Peak accident pressure is the maximum pressure defined in the technical specifications for containment leak testing. Leakage from subatmospheric containments is assumed to be terminated when the containment is brought to a subatmospheric condition as defined in technical specifications.

Transport from Secondary System

H-7. Assumptions acceptable to the NRC staff related to the transport, reduction, and release of radioactive material in and from the secondary system are as follows:

H-7.1 A leak rate equivalent to the primary-to-secondary leak rate limiting condition for operation specified in the technical specifications should be assumed to exist until shutdown cooling is in operation and releases from the steam generators have been terminated.

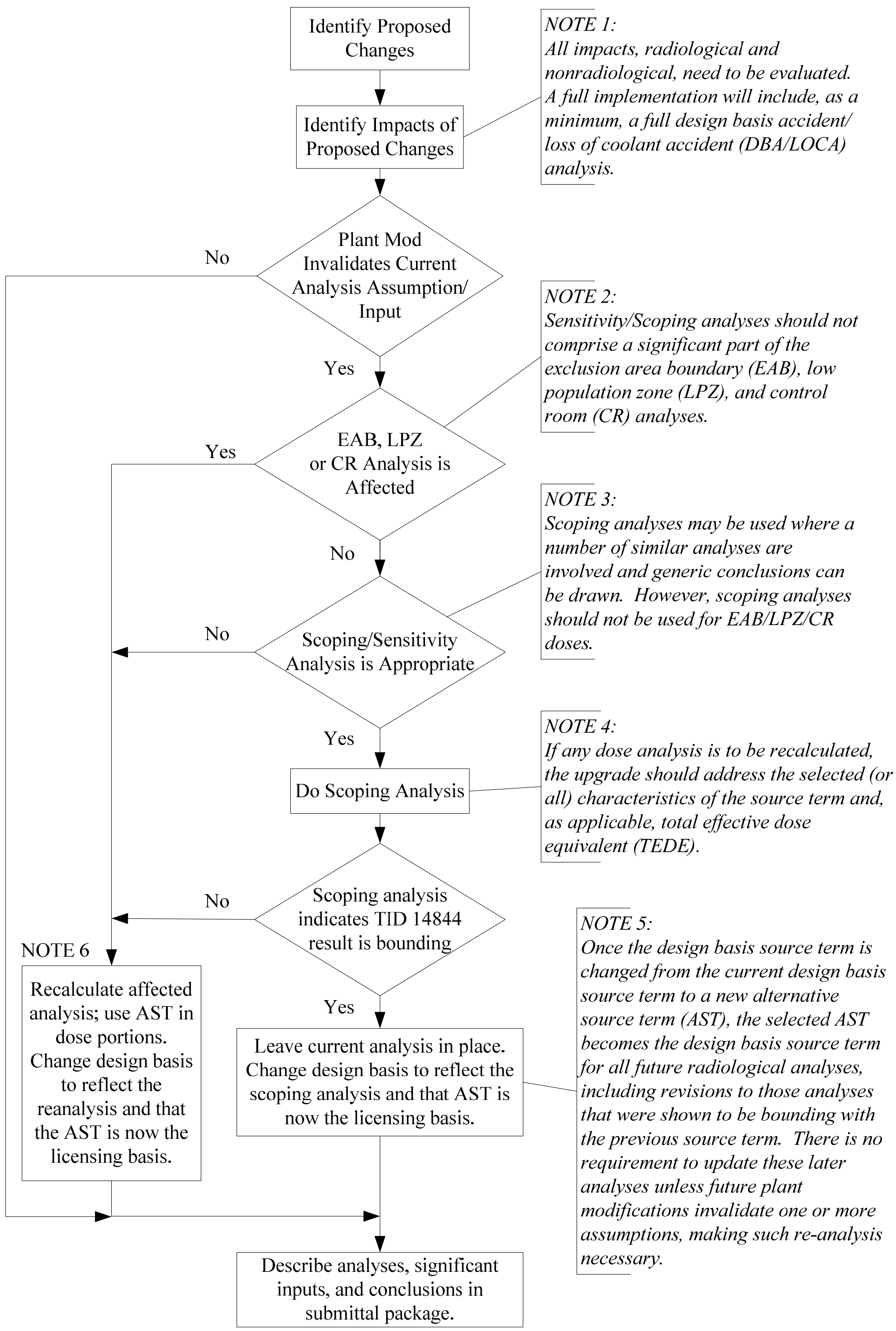
H-7.2 The density used in converting volumetric leak rates (e.g., gallons per minute) to mass leak rates (e.g., pound mass per hour) should be consistent with the basis of surveillance tests used to show compliance with leak rate technical specifications. These tests typically are based on cooled liquid. The facility's instrumentation used to determine leakage typically is located on lines containing cool liquids. In most cases, the density should be assumed to be 1.0 gram per cubic centimeter (62.4 pounds mass per cubic foot).

H-7.3 All noble gas radionuclides released to the secondary system are assumed to be released to the environment without reduction or mitigation.

H-7.4 The transport model described in Regulatory Position E-6.6 of Appendix E to this guide should be utilized for iodine and particulates.

APPENDIX I

ANALYSIS DECISION FLOWCHART



APPENDIX J

ACRONYMS

ADAMS	Agencywide Documents Access and Management System
AST	alternative source term
ARC	alternative repair criteria
BWR	boiling-water reactor
C	Celsius
CDF	core damage frequency
CFR	*Code of Federal Regulations*
CEDE	committed effective dose equivalent
Ci/MWt	curies per megawatt thermal
COLR	core operating limits report
CsI	cesium iodide
DBA	design basis accident
DE	dose equivalent
DF	decontamination factor
EAB	exclusion area boundary
ECCS	emergency core cooling system
EDE	effective dose equivalent
EPA	Environmental Protection Agency
EQ	environmental qualification
ESF	engineered safety feature
F	Fahrenheit
FF	flash fraction
FSAR	final safety analysis report
GDC	general design criterion/criteria
GWd/MTU	gigawatt day per metric ton
gpm	gallon per minute
gpd	gallon per day
IPF	iodine protection factor
LERF	large early release fraction
LOCA	loss-of-coolant accident
LPZ	low-population zone

LWR	light-water reactor
μCi/gm	microcuries per gram
MOX	mixed oxide
MSIV	main steam isolation valve
MSLB	main steamline break
MWD/MTU	megawatt day per metric ton of uranium
NRC	Nuclear Regulatory Commission
PRA	probabilistic risk assessment
PWR	pressurized-water reactor
RAI	request for additional information
RCS	reactor cooling system
RIA	reactivity-initiated accident
RM	radiation monitor
SGTR	steam generator tube rupture
SRP	Standard Review Plan
TEDE	total effective dose equivalent
TID	technical information document