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

Regulatory guides are issued to describe and make available to the public such information as methods acceptable to the NRC staff for implementing specific parts of the NRC's regulations, techniques used by the staff in evaluating specific problems or postulated accidents, and data needed by the NRC staff in its review of applications for permits and licenses. Regulatory guides are not substitutes for regulations, and compliance with them is not required. Methods and solutions different from those set out in the guides will be acceptable if they provide a basis for the findings requisite to the issuance or continuance of a permit or license by the Commission.

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TABLE OF CONTENTS

A.	INTRODUCTION	1
B.	DISCUSSION	2
C.	REGULATORY POSITION	4
1.	IMPLEMENTATION OF AST	4
1.1	Generic Considerations	4
1.2	Scope of Implementation	6
1.3	Scope of Required Analyses	7
1.4	Risk Implications	10
1.5	Submittal Requirements	10
1.6	FSAR Requirements	11
2.	ATTRIBUTES OF AN ACCEPTABLE AST	11
3.	ACCIDENT SOURCE TERM	12
3.1	Fission Product Inventory	12
3.2	Release Fractions	13
3.3	Timing of Release Phases	14
3.4	Radionuclide Composition	15
3.5	Chemical Form	15
3.6	Fuel Damage in Non-LOCA DBAs	16
4.	DOSE CALCULATIONAL METHODOLOGY	16
4.1	Offsite Dose Consequences	16
4.2	Control Room Dose Consequences	17
4.3	Other Dose Consequences	19
4.4	Acceptance Criteria	19
5.	ANALYSIS ASSUMPTIONS AND METHODOLOGY	20
5.1	General Considerations	20
5.2	Accident-Specific Assumptions	22
5.3	Meteorology Assumptions	22
6.	ASSUMPTIONS FOR EVALUATING THE RADIATION DOSES FOR EQUIPMENT QUALIFICATION	23
D.	IMPLEMENTATION	23
	REFERENCES	24

APPENDICES

A.	Assumptions for Evaluating the Radiological Consequences of a LWR Loss-of-Coolant Accident	A-1
B.	Assumptions for Evaluating the Radiological Consequences of a Fuel Fuel Handling Accident	B-1
C.	Assumptions for Evaluating the Radiological Consequences of a BWR Rod Drop Accident	C-1
D.	Assumptions for Evaluating the Radiological Consequences of a BWR Main Steam Line Break Accident	D-1
E.	Assumptions for Evaluating the Radiological Consequences of a PWR Main Steam Line Break Accident	E-1
F.	Assumptions for Evaluating the Radiological Consequences of a PWR Main Steam Generator Tube Rupture Accident	F-1
G.	Assumptions for Evaluating the Radiological Consequences of a PWR Locked Rotor Accident	G-1
H.	Assumptions for Evaluating the Radiological Consequences of a PWR Rod Ejection Accident	H-1
I.	Assumptions for Evaluating Radiation Doses for Equipment Qualification	I-1
J.	Analysis Decision Chart	J-1
K.	Acronyms	K-1

A. INTRODUCTION

This guide provides guidance to licensees of operating power reactors on acceptable applications of alternative source terms; the scope, nature, and documentation of associated analyses and evaluations; consideration of impacts on analyzed risk; and content of submittals. This guide establishes an acceptable alternative source term (AST) and identifies the significant attributes of other ASTs that may be found acceptable by the NRC staff. This guide also identifies acceptable radiological analysis assumptions for use in conjunction with the accepted AST.

In 10 CFR Part 50, “Domestic Licensing of Production and Utilization Facilities,” Section 50.34, “Contents of Applications; Technical Information,” requires that each applicant for a construction permit or operating license 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 resulting from operation of the facility. Applicants are also required by 10 CFR 50.34 to provide an analysis of the proposed site. In 10 CFR Part 100, “Reactor Site Criteria,” Section 100.11,¹ “Determination of Exclusion Area, Low Population Zone, and Population Center Distance,” provides criteria for evaluating the radiological aspects of the proposed site. 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. 1), is cited in 10 CFR Part 100 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 of equipment under 10 CFR 50.49, “Environmental Qualification of Electric Equipment Important to Safety for Nuclear Power Plants,” and in some requirements related to Three Mile Island (TMI) as stated in NUREG-0737, “Clarification of TMI Action Plan Requirements” (Ref. 2). The analyses and evaluations required by 10 CFR 50.34 for an operating license are documented in the facility final safety analysis report (FSAR). 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 whose initial operating license was issued prior to January

¹ Applicants for a construction permit, a design certification, or a combined license that do not reference a standard design certification who applied after January 10, 1997, are required by regulation to meet radiological criteria provided in 10 CFR 50.34.

² As defined in 10 CFR 50.2, *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.

10, 1997, is allowed by 10 CFR 50.67, "Accident Source Term," to 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, the Standard Review Plan (SRP) (Ref 3). 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.

The information collections contained in this regulatory guide are covered by the requirements of 10 CFR Part 50, which were approved by the Office of Management and Budget (OMB), approval number 3150-0011. The NRC may not conduct or sponsor, and a person is not required to respond to, a collection of information unless it displays a currently valid OMB control number.

B. DISCUSSION

An accident source term is intended to be representative of a major accident involving significant core damage 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, NRC staff experience in reviewing license applications has indicated the need to consider other accident sequences of lesser consequence but higher probability of occurrence. The design basis accidents (DBAs) were not intended to be actual event sequences, but rather, were intended to be surrogates to enable deterministic evaluation of the response of a facility's engineered safety features. These accident analyses are intentionally conservative in order to compensate for known uncertainties in accident progression, fission product transport, and atmospheric dispersion. Although probabilistic risk assessments (PRAs) can provide useful insights into system performance and suggest changes in how the desired depth is achieved, 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. 4) 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.

Since the publication of TID-14844 (Ref. 1), 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, "Accident Source Terms for Light-Water Nuclear Power Plants" (Ref. 5). NUREG-1465 used this research to provide estimates of the accident source term that were more physically based and that could be applied to the design of future light-water power reactors. 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. The NRC staff also determined that some licensees might wish 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 AST³ in design basis analyses. These initiatives resulted in the development and issuance of 10 CFR 50.67 and this regulatory guide.

The NRC's traditional methods for calculating the radiological consequences of design basis accidents are described in a series of regulatory guides and SRP chapters. That guidance was developed 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 ASTs and with the total effective dose equivalent (TEDE) criteria provided in 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 50.67. The affected guides will not be withdrawn as their guidance still applies when an AST is not used. Specifically, the affected regulatory guides are:

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)

The guidance in Regulatory Guide 1.89, "Environmental Qualification of Certain Electric Equipment Important to Safety for Nuclear Power Plant." (Ref. 11), regarding the radiological source term used in the determination of integrated doses for environmental qualification purposes is superseded by the corresponding guidance in this regulatory guide for those facilities that are proposing to, or have already, implemented an AST. All other guidance in Regulatory Guide 1.89 remains effective.

This guide primarily addresses design basis accidents, such as those addressed in Chapter 15 of typical final safety analysis reports (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

³ The NUREG-1465 source terms have often been referred to as the "revised source terms." In recognition that there may be additional source terms 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 alternative source terms.

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 AST

1.1 Generic Considerations

As used in this guide, an AST is an accident source term that is different from the accident source term used in the original design and licensing of the facility and that has been approved for use under 10 CFR 50.67. This guide identifies an AST that is acceptable to the NRC staff and identifies significant characteristics of other ASTs 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 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

The proposed uses of an AST and the associated proposed facility modifications and changes to procedures should be evaluated 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 effects of multiple changes, that result in a reduction in safety margins may require prior NRC approval. Once the initial AST implementation has been approved by the staff and has become part of the facility design basis, the licensee may use 10 CFR 50.59 and its supporting guidance in assessing safety margins related to subsequent facility modifications and changes to procedures.

1.1.2 Defense in Depth

The proposed uses of an AST and the associated proposed facility modifications and changes to procedures should be evaluated 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. In all cases, compliance with the General Design Criteria in Appendix A to 10 CFR Part 50 is essential. Modifications proposed for the facility generally should not create a need for compensatory programmatic activities, such as reliance on manual operator actions.

Proposed modifications that seek to downgrade or remove required engineered safeguards equipment should be evaluated to be sure 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 design basis accident source term is a fundamental assumption upon which a significant portion of the facility design is based. Additionally, many aspects of facility operation derive from the design analyses that incorporated the earlier accident source term. Although a complete re-assessment of all facility radiological analyses would be desirable, the NRC staff determined that recalculation of all design analyses would generally not be necessary. Regulatory Position 1.3 of this guide provides guidance on which analyses need updating as part of the AST implementation submittal and which may need updating in the future as additional modifications are performed.

This approach would create two tiers of analyses, those based on the previous source term and those based on an AST. The radiological acceptance criteria would also be different with 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 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, radiological and nonradiological, unless these data would result in nonconservative results or otherwise conflict with the guidance in this guide.

1.1.4 Emergency Preparedness Applications

Requirements for emergency preparedness at nuclear power plants are set forth in 10 CFR 50.47, "Emergency Plans." Additional requirements are set forth in Appendix E, "Emergency Planning and Preparedness for Production and Utilization Facilities," to 10 CFR Part 50. The planning basis for many of these requirements was published in 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). This joint effort by the 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. These requirements were issued 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 AST provided in this guide was based on a limited spectrum of severe accidents, the particular characteristics have been tailored specifically for DBA analysis use. The AST is not

⁴ This planning basis is also addressed in NUREG-0654, "Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants" (Ref. 13).

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.

1.2 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. Additionally, many aspects of facility operation derive from the design analyses that incorporated the earlier accident source term. A complete implementation of an AST would upgrade all existing radiological analyses and would consider the impact of all five characteristics of the AST as defined in 10 CFR 50.2. However, the NRC staff has determined that there could be implementations for which this level of re-analysis may not be necessary. Two categories are defined: Full and selective implementations.

1.2.1 Full Implementation

Full implementation is a modification of the facility design basis that addresses all characteristics of the AST, that is, 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 re-analyzed using the guidance in Appendix A of this guide. Additional guidance on analysis is provided in Regulatory Position 1.3 of this guide. 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, "Changes, Tests, and Experiments," 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 re-evaluation of a limited subset of the design basis radiological analyses. The NRC staff will allow licensees flexibility in 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. For the latter, the licensee may only need to re-analyze DBAs that credited the iodine removal by the charcoal media. Additional analysis guidance is provided in Regulatory Position 1.3 of this guide. 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, this 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 design basis accidents. These requirements include, but are not limited to, the following.

- Environmental Qualification of Equipment (10 CFR 50.49)
- Control Room Habitability (GDC-19 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)
- Facility Siting (10 CFR 100.11)⁵

There may be additional applications of the accident source term identified in the technical specification bases and in various licensee commitments. These include, but are not limited to, the following from Reference 2, NUREG-0737.

- Post-Accident Access Shielding (NUREG-0737, II.B.2)
- Post-Accident Sampling Capability (NUREG-0737, II.B.3)
- Accident Monitoring Instrumentation (NUREG-0737, II.F.1)
- Leakage Control (NUREG-0737, III.D.1.1)
- Emergency Response Facilities (NUREG-0737, III.A.1.2)
- Control Room Habitability (NUREG-0737, III.D.3.4)

1.3.2 Re-Analysis Guidance

Any implementation of an AST, full or selective, 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 due to (1) the associated facility modifications or (2) the differences in the AST characteristics. The scope and extent of the re-

⁵ Dose guidelines of 10 CFR 100.11 are superseded by 10 CFR 50.67 for licensees that have implemented an AST.

evaluation will necessarily be a function of the specific proposed facility modification⁶ 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 the design bases appropriately. An analysis is considered 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. Generic analyses, such as those performed by owner groups or vendor topical reports, may be used provided 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 re-calculated, all affected assumptions and inputs should be updated and all selected characteristics of the AST and the TEDE criteria should be addressed. The license amendment request should describe the licensee's re-analysis 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 performed an evaluation of 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. 1) 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, evaluations may need to be performed regarding 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, a complete DBA LOCA analysis as described in Appendix A of this guide should be performed, as a minimum. Other design basis analyses are updated 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 the radiological and nonradiological impacts of the proposed actions as they apply to the particular implementation. Design basis analyses are updated 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 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, re-analysis of radiological calculations may not be

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, re-assessment of the containment pressure and temperature transient, recalculation of sump pH, re-assessment of the emergency diesel generator loading sequence, integrated doses to equipment in the containment, and more.

necessary if the modified elapsed time remains a fraction (e.g., 25%) 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 re-calculated, 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. 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 environmental qualification (integrated dose). It may be possible to identify a bounding case, re-analyze 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.

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 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. Re-evaluation 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 do not constitute a change in analysis methodology that would require NRC approval.⁷

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 characteristics of the AST and TEDE criteria identified in the facility design basis need to be considered in updating the analyses. 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 environmental qualification (EQ) analyses may be impacted by a proposed plant modification associated with the AST implementation. The EQ analyses that have assumptions or inputs affected by the plant modification should be updated to address these impacts. The NRC staff is assessing the effect of increased cesium releases on EQ doses to determine whether licensee action is warranted. Until such time as this generic issue is resolved, licensees may use either the AST or the TID14844 assumptions for performing the required EQ analyses. However, no plant modifications are required to address the impact of the difference in source term characteristics (i.e., AST vs TID14844) on EQ doses pending the outcome of the evaluation of the generic issue. The EQ dose estimates should be calculated using the design basis survivability period.

1.4 Risk Implications

The use of an AST changes only the regulatory assumptions regarding the analytical treatment of the design basis accidents. The AST has no direct effect on the probability of the accident. Use of an AST alone 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 impact on the existing PRAs should be evaluated.

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, 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 that the amendment may be approved must be based on the licensee's analyses,

since it is these analyses that will become part of the design basis of the facility. 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. The staff recommends that licensees submit affected FSAR pages annotated with changes that reflect the revised analyses or submit the actual calculation documentation.

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.

1.6 FSAR Requirements

Requirements for updating the facility's final safety analysis report (FSAR) are in 10 CFR 50.71, "Maintenance of Records, Making of Reports." The regulations in 10 CFR 50.71(e) require that the FSAR be updated to include all changes made in the facility or procedures described in the FSAR and all safety evaluations performed by the licensee in support of requests for license amendments or in support of conclusions that changes did not involve unreviewed safety questions. The analyses required by 10 CFR 50.67 are subject to this requirement. The affected radiological analysis descriptions in the FSAR should be updated to reflect the replacement of the design basis source term by the AST. 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 descriptions of superseded analyses should be removed from the FSAR in the interest of maintaining a clear design basis.

2. ATTRIBUTES OF AN ACCEPTABLE AST

An acceptable AST is not set forth in 10 CFR 50.67. Regulatory Position 3 of this guide identifies an AST that is acceptable to the NRC staff for use at operating power reactors. A substantial effort was expended by the NRC, its contractors, various national laboratories, peer reviewers, and others in performing severe accident research and in developing the source terms provided in NUREG-1465 (Ref. 5). 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:

- 2.1** 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.

- 2.2 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.
- 2.3 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. Relevant insights from applicable severe accident research on the phenomenology of fission product release and transport behavior may be considered.
- 2.4 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.
- 2.5 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 section provides an AST that is acceptable to the NRC staff. The data in Regulatory Positions 3.2 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. Deviations from this guidance must be evaluated 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 current licensed rated thermal power times the ECCS evaluation uncertainty.⁸ 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.⁹ The core inventory should be determined using an appropriate isotope generation and depletion computer code such as ORIGEN 2 (Ref. 17) or ORIGEN-ARP (Ref. 18). Core inventory factors (Ci/MWt) provided in TID14844 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.

⁸ The uncertainty factor used in determining the core inventory should be that value provided in Appendix K to 10 CFR Part 50, typically 1.02.

⁹ Note that for some radionuclides, such as Cs-137, equilibrium will not be reached prior to fuel offload. Thus, the maximum inventory at the end of life should be used.

For the DBA LOCA, all fuel assemblies in the core are assumed to be affected and the core average inventory should be used. 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, radial peaking factors from the facility's core operating limits report (COLR) or technical specifications should be applied in determining the inventory of the damaged rods.

No adjustment to the fission product inventory should be made 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 shutdown, e.g., a fuel handling accident, radioactive decay from the time of shutdown may be modeled.

3.2 Release Fractions¹⁰

The core inventory release fractions, by radionuclide groups, for the gap release and early in-vessel damage phases for DBA LOCAs are listed in Table 1 for BWRs and Table 2 for PWRs. These fractions are applied to the equilibrium core inventory described in Regulatory Position 3.1.

For non-LOCA events, the fractions of the core inventory assumed to be in the gap for the various radionuclides are given in Table 3. The release fractions from Table 3 are used in conjunction with the fission product inventory calculated with the maximum core radial peaking factor.

Table 1
BWR Core Inventory Fraction
Released Into Containment
Gap Release Early In-vessel
Group Phase Phase Total

Group	Gap Release	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

¹⁰ The release fractions listed here have been determined to be acceptable for use with currently approved LWR fuel with a peak burnup up to 62,000 MWD/MTU. The data in this section may not be applicable to cores containing mixed oxide (MOX) fuel.

Table 2
PWR Core Inventory Fraction
Released Into Containment

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 3¹¹
Non-LOCA Fraction of Fission Product Inventory in Gap

Group	Fraction
I-131	0.08
Kr-85	0.10
Other Noble Gases	0.05
Other Halogens	0.05
Alkali Metals	0.12

3.3 Timing of Release Phases

Table 4 tabulates the onset and duration of each sequential release phase for DBA LOCAs at PWRs and BWRs. The specified onset is the time following the initiation of the accident (i.e., time = 0). The early in-vessel phase immediately follows the gap release phase. The activity released from the core during each release phase should be modeled as increasing in a linear fashion over the duration of the phase.¹² 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.

¹¹ The release fractions listed here have been determined to be acceptable for use with currently approved LWR fuel with a peak burnup up to 62,000 MWD/MTU provided that the maximum linear heat generation rate does not exceed 6.3 kw/ft peak rod average power for burnups exceeding 54 GWD/MTU. As an alternative, fission gas release calculations performed using NRC-approved methodologies may be considered on a case-by-case basis. To be acceptable, these calculations must use a projected power history that will bound the limiting projected plant-specific power history for the specific fuel load. For the BWR rod drop accident and the PWR rod ejection accident, the gap fractions are assumed to be 10% for iodines and noble gases.

¹² 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.

Table 4
LOCA Release Phases

Phase	PWRs		BWRs	
	Onset	Duration	Onset	Duration
Gap Release	30 sec	0.5 hr	2 min	0.5 hr
Early In-Vessel	0.5 hr	1.3 hr	0.5 hr	1.5 hr

For facilities licensed with leak-before-break methodology, the onset of the gap release phase may be assumed 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 gap release phase onsets in Table 4 should be used.

3.4 Radionuclide Composition

Table 5 lists the elements in each radionuclide group that should be considered in design basis analyses.

Table 5
Radionuclide Groups

Group	Elements
Noble Gases	Xe, Kr
Halogens	I, Br
Alkali Metals	Cs, Rb
Tellurium Group	Te, Sb, Se, 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 same chemical form is assumed in releases from fuel pins in FHAs and from releases from the fuel pins through the RCS in DBAs other than FHAs or LOCAs. However, the transport of these iodine species following release from the fuel may affect these assumed fractions. The accident-specific appendices to this regulatory guide provide additional details.

3.6 Fuel Damage in Non-LOCA DBAs

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 clad is breached. Although the NRC staff has traditionally relied upon the departure from nucleate boiling ratio (DNBR) as a fuel damage criterion, licensees may propose other methods to the NRC staff, such as those based upon enthalpy deposition, for estimating fuel damage for the purpose of establishing radioactivity releases.

The amount of fuel damage caused by a FHA is addressed in Appendix B of this guide.

4. DOSE CALCULATIONAL METHODOLOGY

The NRC staff has determined that there is an implied synergy between the ASTs and total effective dose equivalent (TEDE) criteria, and between the TID-14844 source terms and the whole body and thyroid dose criteria, and therefore, they do not expect to allow the TEDE criteria to be used with TID-14844 calculated results. The guidance of this section applies to all dose calculations performed with an AST pursuant to 10 CFR 50.67. Certain selective implementations may not require dose calculations as described in Regulatory Position 1.3 of this guide.

4.1 Offsite Dose Consequences

The following assumptions should be used in determining the TEDE for persons located at or beyond the boundary of the exclusion area (EAB):

4.1.1 The dose calculations should determine the TEDE. TEDE is the sum of the committed effective dose equivalent (CEDE) from inhalation and the deep dose equivalent (DDE) 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.¹³

4.1.2 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.

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

¹³ 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.

4.1.4 The DDE should be calculated assuming submergence in semi-infinite cloud assumptions with appropriate credit for attenuation by body tissue. The DDE is nominally equivalent to the effective dose equivalent (EDE) from external exposure if the whole body is irradiated uniformly. Since this is a reasonable assumption for submergence exposure situations, EDE may be used in lieu of DDE in determining the contribution of external dose to the TEDE. 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.

4.1.5 The TEDE should be determined for the most limiting person at the EAB. The maximum EAB TEDE for any two-hour period following the start of the radioactivity release should be determined and used in determining compliance with the dose criteria in 10 CFR 50.67.¹⁴ The maximum two-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 two-hour periods. The maximum TEDE obtained is submitted. 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 also Table 6).

4.1.6 TEDE should be determined for the most limiting receptor at the outer boundary of the low population zone (LPZ) and should be used in determining compliance with the dose criteria in 10 CFR 50.67.

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

4.2 Control Room Dose Consequences

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

4.2.1 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:

- Contamination of the control room atmosphere by the intake or infiltration of the radioactive material contained in the radioactive plume released from the facility,
- 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,
- Radiation shine from the external radioactive plume released from the facility,

¹⁴ With regard to the EAB TEDE, the maximum two-hour value is the basis for screening and evaluation under 10 CFR 50.59. Changes to doses outside of the two-hour window are only considered in the context of their impact on the maximum two-hour EAB TEDE.

- Radiation shine from radioactive material in the reactor containment,
- 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 The radioactive material releases and radiation levels used in the control room dose analysis should be determined using the same source term, 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 The models used to transport radioactive material into and through the control room,¹⁵ and the shielding models 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 Credit for engineered safety features that mitigate airborne radioactive material within the control room may be assumed. 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. 3) and Regulatory Guide 1.52, “Design, Testing, and Maintenance Criteria for Postaccident Engineered-Safety-Feature Atmosphere Cleanup System Air Filtration and Adsorption Units of Light-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 engineered safeguards feature (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 Credit should generally not be taken for the use of personal protective equipment or prophylactic drugs. Deviations may be considered on a case-by-case basis.

4.2.6 The dose receptor for these analyses is the hypothetical maximum exposed individual who is present in the control room for 100% of the time during the first 24 hours after the event, 60% of the time between 1 and 4 days, and 40% of the time from 4 days to 30 days.¹⁶ For the duration of the event, the breathing rate of this individual should be assumed to be 3.5×10^4 cubic meters per second.

¹⁵ The iodine protection factor (IPF) methodology of Reference 22 may not be adequately conservative for all DBAs and control room arrangements since 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.

¹⁶ This occupancy is modeled in the χ/Q values determined in Reference 22 and should not be credited twice. The ARCON96 Code (Ref. 26) does not incorporate these occupancy assumptions, making it necessary to apply this correction in the dose calculations.

4.2.7 Control room doses should be calculated using dose conversion factors identified in Regulatory Position 4.1 above for use in offsite dose analyses. The DDE 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, DDE_{∞} , to a finite cloud dose, DDE_{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).

$$DDE_{finite} = \frac{DDE_{\infty} V^{0.338}}{1173} \quad \text{Equation 1}$$

4.3 Other Dose Consequences

The guidance provided in Regulatory Positions 4.1 and 4.2 should be used, as applicable, in re-assessing the radiological analyses identified in Regulatory Position 1.3.1, such as those in NUREG-0737 (Ref. 2). Design envelope source terms provided in NUREG-0737 should be updated 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. Integrated radiation exposure of plant equipment should be determined using the guidance of Appendix I of this guide.

4.4 Acceptance Criteria

The radiological criteria for the EAB, the outer boundary of the LPZ, and for the control room are in 10 CFR 50.67. 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. The control room criterion applies to all accidents. For events with a higher probability of occurrence, postulated EAB and LPZ doses should not exceed the criteria tabulated in Table 6.

The acceptance criteria for the various NUREG-0737 (Ref. 2) items generally reference General Design Criteria 19 (GDC 19) from 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, the applicable criteria should be updated for consistency with the TEDE criterion in 10 CFR 50.67(b)(2)(iii).

Table 6¹⁷
Accident Dose Criteria

Accident or Case	EAB and LPZ Dose Criteria	Analysis Release Duration
LOCA	25 rem TEDE	30 days for containment, ECCS, and MSIV (BWR) leakage
BWR Main Steam Line Break		Instantaneous puff
Fuel Damage or Pre-incident Spike	25 rem TEDE	
Equilibrium Iodine Activity	2.5 rem TEDE	
BWR Rod Drop Accident	6.3 rem TEDE	24 hours
PWR Steam Generator Tube Rupture		Affected SG: time to isolate; Unaffected SG(s): until cold shutdown is established
Fuel Damage or Pre-incident Spike	25 rem TEDE	
Coincident Iodine Spike	2.5 rem TEDE	
PWR Main Steam Line Break		Until cold shutdown is established
Fuel Damage or Pre-incident Spike	25 rem TEDE	
Coincident Iodine Spike	2.5 rem TEDE	
PWR Locked Rotor Accident	2.5 rem TEDE	Until cold shutdown is established
PWR Rod Ejection Accident	6.3 rem TEDE	30 days for containment pathway; until cold shutdown is established for secondary pathway
Fuel Handling Accident	6.3 rem TEDE	2 hours

The column labeled “Analysis Release Duration” is a summary of 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.

5. ANALYSIS ASSUMPTIONS AND METHODOLOGY

5.1 General Considerations

5.1.1 Analysis Quality

The evaluations required by 10 CFR 50.67 are re-analyses of the design basis safety analyses and evaluations required by 10 CFR 50.34; they are considered to be a significant input to the evaluations required by 10 CFR 50.92 or 10 CFR 50.59. These analyses should be prepared, reviewed, and maintained 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

¹⁷ For PWRs with steam generator alternative repair criteria, different dose criteria may apply to steam generator tube rupture and main steam line break analyses.

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

Credit may be taken 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. The single active component failure that results in the most limiting radiological consequences should be assumed. Assumptions regarding the occurrence and timing of a loss of offsite power should be selected with the objective of maximizing the postulated radiological consequences.

5.1.3 Assignment of Numeric Input Values

The numeric values that are chosen as inputs to the analyses required by 10 CFR 50.67 should be selected 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 be nonconservative in another portion of the same analysis. For example, assuming 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.¹⁸ 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 (NDT), 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. In order 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

¹⁸ 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) and in Generic Letter 99-02 (Ref. 27) rather than the surveillance test criteria in the technical specifications. Generally, these adjustments address potential changes in the parameter between scheduled surveillance tests.

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 analyses that are required by 10 CFR 50.67. 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.

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 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 probabilistic risk analysis (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 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 core damage frequency (CDF) and large early release frequency (LERF) surrogate indicators of overall risk.

5.3 Meteorology Assumptions

Atmospheric dispersion values (χ/Q) for the EAB, the LPZ, and the control room that were approved by the staff during initial facility licensing or in subsequent licensing proceedings may be used in performing the radiological analyses identified by this guide. Methodologies that have been used for determining χ/Q values are documented in Regulatory Guides 1.3 and 1.4, Regulatory Guide 1.145, “Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants,” and the paper, “Nuclear Power Plant Control Room Ventilation System Design for Meeting General Criterion 19” (Refs. 6, 7, 22, and 28).

References 22 and 28 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. Fumigation should be considered where applicable for the EAB and LPZ. For the EAB, the assumed fumigation period should be timed to be included in the worst 2-hour exposure period. The NRC computer code PAVAN (Ref. 29) implements Regulatory Guide 1.145 (Ref. 28) and its use is acceptable to the NRC staff. The methodology of the NRC computer code ARCON96¹⁹ (Ref. 26) is generally acceptable to the NRC staff for use in determining control room χ/Q values. Meteorological data collected in accordance with the site-specific meteorological measurements program described in the facility FSAR should be used in generating accident χ/Q values. Additional guidance is provided in Regulatory Guide 1.23, "Onsite Meteorological Programs" (Ref. 30). All changes in χ/Q analysis methodology should be reviewed by the NRC staff.

6. ASSUMPTIONS FOR EVALUATING THE RADIATION DOSES FOR EQUIPMENT QUALIFICATION

The assumptions in Appendix I to this guide are acceptable to the NRC staff for performing radiological assessments associated with equipment qualification. The assumptions in Appendix I will supersede Regulatory Positions 2.c(1) and 2.c(2) and Appendix D of Revision 1 of Regulatory Guide 1.89, "Environmental Qualification of Certain Electric Equipment Important to Safety for Nuclear Power Plants" (Ref. 11), for operating reactors that have amended their licensing basis to use an alternative source term. Except as stated in Appendix I, all other assumptions, methods, and provisions of Revision 1 of Regulatory Guide 1.89 remain effective.

The NRC staff is assessing the effect of increased cesium releases on EQ doses to determine whether licensee action is warranted. Until such time as this generic issue is resolved, licensees may use either the AST or the TID14844 assumptions for performing the required EQ analyses. However, no plant modifications are required to address the impact of the difference in source term characteristics (i.e., AST vs TID14844) on EQ doses pending the outcome of the evaluation of the generic issue.

D. IMPLEMENTATION

The purpose of this section is to provide information to applicants and licensees regarding the NRC staff's plans for using this regulatory guide.

Except in those cases in which an applicant or licensee proposes an acceptable alternative method for complying with the specified portions of the NRC's regulations, the methods described in this guide will be used in the evaluation of submittals related to the use of ASTs in radiological consequence analyses at operating power reactors.

¹⁹ The ARCON96 computer code contains processing options that may yield χ/Q values that are not sufficiently conservative for use in accident consequence assessments or may be incompatible with release point and ventilation intake configurations at particular sites. The applicability of these options and associated input parameters should be evaluated on a case-by-case basis. The assumptions made in the examples in the ARCON96 documentation are illustrative only and do not imply NRC staff acceptance of the methods or data used in the example.

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{See the inside front cover of this guide for information on obtaining NRC documents.}

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Appendix A

ASSUMPTIONS FOR EVALUATING THE RADIOLOGICAL CONSEQUENCES OF A LWR LOSS-OF-COOLANT ACCIDENT

The assumptions in this appendix are acceptable to the NRC staff 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 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 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 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.

SOURCE TERM ASSUMPTIONS

1. Acceptable assumptions regarding core inventory and the release of radionuclides from the fuel are provided in Regulatory Position 3 of this guide.
2. 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% cesium iodide (CsI), 4.85 percent elemental iodine, and 0.15 percent organic iodide. Iodine species, including those from iodine re-evolution, 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.

ASSUMPTIONS ON TRANSPORT IN PRIMARY CONTAINMENT

3. Acceptable assumptions related to the transport, reduction, and release of radioactive material in and from the primary containment in PWRs or the drywell in BWRs are as follows:
 - 3.1 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 phase.
 - 3.2 Reduction in airborne radioactivity in the containment by natural deposition within the containment may be credited. Acceptable models for removal of iodine and aerosols are

described in Chapter 6.5.2, “Containment Spray as a Fission Product Cleanup System,” of the Standard Review Plan (SRP), NUREG-0800 (Ref. A-1) and in NUREG/CR-6189, “A Simplified Model of Aerosol Removal by Natural Processes in Reactor Containments” (Ref. A-2). The latter model is incorporated into the analysis code RADTRAD (Ref. A-3). The prior practice of deterministically assuming that a 50% plateout of iodine is released from the fuel is no longer acceptable to the NRC staff as it is inconsistent with the characteristics of the revised source terms.

- 3.3** 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. Acceptable models for the removal of iodine and aerosols are described in Chapter 6.5.2 of the SRP and NUREG/CR-5966, “A Simplified Model of Aerosol Removal by Containment Sprays”¹ (Ref. A-4). This simplified model is incorporated into the analysis code RADTRAD (Refs. A-1 to A-3).

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 regions per hour, unless other rates are justified. The containment building atmosphere may be considered a single, well-mixed volume if the spray covers at least 90% of the volume and if adequate mixing of unsprayed compartments can be shown.

The SRP sets forth a maximum decontamination factor (DF) for elemental iodine 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).

- 3.4** 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 and Generic Letter 99-02 (Refs. A-5 and A-6). The filter media loading caused by the increased aerosol release associated with the revised source term should be addressed.
- 3.5** 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,

¹ This document describes statistical formulations with differing levels of uncertainty. The removal rate constants selected for use in design basis calculations should be those that will maximize the dose consequences. For BWRs, the simplified model should be used only if the release from the core is not directed through the suppression pool. Iodine removal in the suppression pool affects the iodine species assumed by the model to be present initially.

and the potential for any bypass of the suppression pool (Ref. 7). Analyses should consider iodine re-evolution if the suppression pool liquid pH is not maintained greater than 7.

- 3.6** Reduction in airborne radioactivity in the containment by retention in ice condensers, or other engineering safety features not addressed above, should be evaluated on an individual case basis. See Section 6.5.4 of the SRP (Ref. A-1).
- 3.7** The primary containment (i.e., drywell for Mark I and II containment designs) should be assumed to 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% 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% 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 two-hour period between the initial blowdown and termination of the fuel radioactivity release (gap and early in-vessel release phases). After two hours, the radioactivity is assumed to be uniformly distributed throughout the drywell and the primary containment.

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

ASSUMPTIONS ON DUAL CONTAINMENTS

- 4.** 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.

- 4.1** Leakage from the primary containment should be considered to be collected, processed by engineered safety feature (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 two and one-half times the height of any adjacent structure.

- 4.2** 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.
- 4.3** The effect of high wind speeds on the ability of the secondary containment to maintain a negative pressure should be evaluated on an individual case basis. The wind speed to be assumed is the 1-hour average value that is exceeded only 5% of the total number of hours in the data set. Ambient temperatures used in these assessments should be the 1-hour average value that is exceeded only 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%).
- 4.4** 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%. 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.
- 4.5** 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.
- 4.6** 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) and Generic Letter 99-02 (Ref. A-6).

ASSUMPTIONS ON ESF SYSTEM LEAKAGE

5. 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-7). The radiological consequences from the postulated leakage should be analyzed and combined 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.

- 5.1** With the exception of noble gases, all the 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 buildup of sump activity.

- 5.2** The leakage should be taken as two times 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 (Ref. A-8), would require declaring such systems inoperable. 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. Consideration should also be given to design leakage through valves isolating ESF recirculation systems from tanks vented to atmosphere, e.g., emergency core cooling system (ECCS) pump miniflow return to the refueling water storage tank.
- 5.3** With the exception of iodine, all radioactive materials in the recirculating liquid should be assumed to be retained in the liquid phase.
- 5.4** If the temperature of the leakage exceeds 212°F, the fraction of total iodine in the liquid that becomes airborne should be assumed equal to 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:

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

Where: h_{f_1} is the enthalpy of liquid at system design temperature and pressure; h_{f_2} is the enthalpy of liquid at saturation conditions (14.7 psia, 212°F); and h_{fg} is the heat of vaporization at 212°F.

- 5.5** If the temperature of the leakage is less than 212°F or the calculated flash fraction is less than 10%, the amount of iodine that becomes airborne should be assumed to be 10% of the total iodine activity in the leaked fluid, unless a smaller amount can be justified based on the actual sump pH history and area ventilation rates.
- 5.6** The radioiodine that is postulated to be available for release to the environment is assumed to be 97% elemental and 3% organic. 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) and Generic Letter 99-02 (Ref. A-6).

ASSUMPTIONS ON MAIN STEAM ISOLATION VALVE LEAKAGE IN BWRS

- 6.** For BWRs, the main steam isolation valves (MSIVs) have design leakage that may result in a radioactivity release. The radiological consequences from postulated MSIV leakage should be analyzed and combined 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.

- 6.1** For the purpose of this analysis, the activity available for release via MSIV leakage should be assumed to be that activity determined to be in the drywell for evaluating containment leakage (see Regulatory Position 3). No credit should be assumed for activity reduction by the steam separators or by iodine partitioning in the reactor vessel.
- 6.2** 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. Postulated leakage may be reduced after the first 24 hours, if supported by site-specific analyses, to a value not less than 50% of the maximum leak rate.
- 6.3** Reduction of the amount of released radioactivity by deposition and plateout on steam system piping upstream of the outboard MSIVs may be credited, but the amount of reduction in concentration allowed will be evaluated on an individual case basis. Generally, the model should be based on the assumption of well-mixed volumes, but other models such as slug flow may be used if justified.
- 6.4** In the absence of collection and treatment of releases by ESFs such as the MSIV leakage control system, or as described in paragraph 6.5 below, 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.
- 6.5** A reduction in MSIV releases that is due to holdup and deposition in main steam piping downstream of the MSIVs and in the 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 (SSE). The amount of reduction allowed will be evaluated on an individual case basis. References A-9 and A-10 provide guidance on acceptable models.

ASSUMPTION ON CONTAINMENT PURGING

7. The radiological consequences from post-LOCA primary containment purging as a combustible gas or pressure control measure should be analyzed. 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 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. Reduction in the amount of radioactive material released via ESF filter systems may be taken into account provided that these systems meet the guidance in Regulatory Guide 1.52 (Ref. A-5) and Generic Letter 99-02 (Ref. A-6).

Appendix A REFERENCES

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- A-10 USNRC, "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 Number 9903110303.

Appendix B

ASSUMPTIONS FOR EVALUATING THE RADIOLOGICAL CONSEQUENCES OF A FUEL HANDLING ACCIDENT

This appendix provides assumptions acceptable to the staff 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. SOURCE TERM

Acceptable assumptions regarding core inventory and the release of radionuclides from the fuel are provided in Regulatory Position 3 of this guide. The following assumptions also apply.

- 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. Damage to adjacent fuel assemblies, if applicable (e.g., events over the reactor vessel), should be considered.
- 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.
- 1.3 The chemical form of radioiodine released from the fuel to the spent fuel pool should be assumed to be 95% 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 re-evolves 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.

2. WATER DEPTH

If the depth of water above the damaged fuel is 23 feet or greater, the decontamination factors for the elemental and organic species are 500 and 1, respectively, giving an overall effective decontamination factor of 200 (i.e., 99.5% of the total iodine released from the damaged rods is retained by the water). This difference in decontamination factors for elemental (99.85%) and organic iodine (0.15%) species results in the iodine above the water being composed of 57% elemental and 43% organic species. If the depth of water is not 23 feet, the decontamination factor will have to be determined on a case-by-case method (Ref. B-1).

3. NOBLE GASES

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

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.

- 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.
- 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 and Generic Letter 99-02 (Refs. B-2, B-3). Delays in radiation detection, actuation of the ESF filtration system, or diversion of ventilation flow to the ESF filtration system¹ should be determined and accounted for in the radioactivity release analyses.
- 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.

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.

- 5.1** If the containment is isolated² during fuel handling operations, no radiological consequences need to be analyzed.
- 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

¹ 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.

² Containment *isolation* does not imply containment integrity as defined by technical specifications for non-shutdown 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 appropriate form of isolation should be addressed in technical specifications.

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,¹ no radiological consequences need to be analyzed.

- 5.3** If the containment is open during fuel handling operations (e.g., personnel air lock or equipment hatch is open),³ the radioactive material that escapes from the reactor cavity pool to the containment is released to the environment over a 2-hour time period.
- 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 and Generic Letter 99-02 (Refs. B-2 and B-3). Delays in radiation detection, actuation of the ESF filtration system, or diversion of ventilation flow to the ESF filtration system should be determined and accounted for in the radioactivity release analyses.¹
- 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% 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 stream flow between the surface of the reactor cavity and the exhaust plenums.

³ The staff will generally require that technical specifications allowing such operations include administrative controls to close the airlock, hatch, or open penetrations within 30 minutes. Such administrative controls will 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. (NRC Accession number 8402080322 in ADAMS or PARS)
- B-2. USNRC, "Design, Testing, and Maintenance Criteria for Postaccident Engineered-Safety-Feature Atmosphere Cleanup System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants," Regulatory Guide 1.52, Revision 2, March 1978.
- B-3. USNRC, "Laboratory Testing of Nuclear Grade Activated Charcoal," Generic Letter 99-02, June 3, 1999.

Appendix C

ASSUMPTIONS FOR EVALUATING THE RADIOLOGICAL CONSEQUENCES OF A BWR ROD DROP ACCIDENT

This appendix provides assumptions acceptable to the NRC staff for evaluating the radiological consequences of a rod drop accident at BWR light-water reactors. These assumptions supplement the guidance provided in the main body of this guide.

1. Assumptions acceptable to the NRC staff regarding core inventory are provided in Regulatory Position 3 of this guide. For the rod drop accident, the release from the breached fuel is based on the estimate of the number of fuel rods breached and the assumption that 10% of the core inventory of the noble gases and iodines is in the fuel gap. The release attributed to fuel melting is based on the fraction of the fuel that reaches or exceeds the initiation temperature for fuel melting and on the assumption that 100% of the noble gases and 50% of the iodines contained in that fraction are released to the reactor coolant.
2. If no or minimal¹ fuel damage is postulated for the limiting event, the released activity should be the maximum coolant activity (typically 4 $\mu\text{Ci/gm}$ DE I-131) allowed by the technical specifications.
3. The assumptions acceptable to the NRC staff that are related to the transport, reduction, and release of radioactive material from the fuel and the reactor coolant are as follows.
 - 3.1 The activity released from the fuel from either the gap or from fuel pellets is assumed to be instantaneously mixed in the reactor coolant within the pressure vessel.
 - 3.2 Credit should not be assumed for partitioning in the pressure vessel or for removal by the steam separators.
 - 3.3 Of the activity released from the reactor coolant within the pressure vessel, 100% of the noble gases, 10% of the iodine, and 1% of the remaining radionuclides are assumed to reach the turbine and condensers.
 - 3.4 Of the activity that reaches the turbine and condenser, 100% of the noble gases, 10% of the iodine, and 1% of the particulate radionuclides are available for release to the environment. The turbine and condensers leak to the atmosphere as a ground-level release at a rate of 1% per day² 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

¹ 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 the 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.

² If there are forced flow paths 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.

turbine building. Radioactive decay during holdup in the turbine and condenser may be assumed.

- 3.5** In lieu of the transport assumptions provided in paragraphs 3.2 through 3.4 above, a more mechanistic analysis may be used on a case-by-case basis. Such analyses account for the 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 (MSIV) and considers MSIV closure time.
- 3.6** The iodine species released from the reactor coolant within the pressure vessel should be assumed to be 95% CsI as an aerosol, 4.85% elemental, and 0.15% organic. The release from the turbine and condenser should be assumed to be 97% elemental and 3% organic.

Appendix D

ASSUMPTIONS FOR EVALUATING THE RADIOLOGICAL CONSEQUENCES OF A BWR MAIN STEAM LINE BREAK ACCIDENT

This appendix provides assumptions acceptable to the NRC staff for evaluating the radiological consequences of a main steam line accident at BWR light water reactors. These assumptions supplement the guidance provided in the main body of this guide.

SOURCE TERM

1. Assumptions acceptable to the NRC staff regarding core inventory and the release of radionuclides from the fuel are provided in Regulatory Position 3 of this guide. 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.
2. If no or minimal¹ 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.
 - 2.1 The concentration that is the maximum value (typically 4.0 $\mu\text{Ci/gm}$ DE I-131) permitted and corresponds to the conditions of an assumed pre-accident spike, and
 - 2.1 The concentration that is the maximum equilibrium value (typically 0.2 $\mu\text{Ci/gm}$ DE I-131) permitted for continued full power operation.
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.

TRANSPORT

4. Assumptions acceptable to the NRC staff related to the transport, reduction, and release of radioactive material to the environment are as follows.
 - 4.1 The main steam line isolation valves (MSIV) should be assumed to close in the maximum time allowed by technical specifications.
 - 4.2 The total mass of coolant released should be assumed to be that amount in the steam line and connecting lines at the time of the break plus the amount that passes through the valves prior to closure.

¹ 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.

- 4.3** All the radioactivity in the released coolant should be assumed to be released to the atmosphere instantaneously as a ground-level release. No credit should be assumed for plateout, holdup, or dilution within facility buildings.
- 4.4** The iodine species released from the main steam line should be assumed to be 95% CsI as an aerosol, 4.85% elemental, and 0.15% organic.

Appendix E

ASSUMPTIONS FOR EVALUATING THE RADIOLOGICAL CONSEQUENCES OF A PWR MAIN STEAM LINE BREAK ACCIDENT

This appendix provides assumptions acceptable to the NRC staff for evaluating the radiological consequences of a main steam line break accident at PWR light water reactors. These assumptions supplement the guidance provided in the main body of this guide.¹

SOURCE TERMS

1. Assumptions acceptable to the NRC staff regarding core inventory and the release of radionuclides from the fuel are provided in Regulatory Position 3 of this regulatory guide. 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. The fuel damage estimate should assume that the highest worth control rod is stuck at its fully withdrawn position.
2. If no or minimal² 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.
 - 2.1 A reactor transient has occurred prior to the postulated main steam line break (MSLB) and has raised the primary coolant iodine concentration to the maximum value (typically 60 $\mu\text{Ci/gm}$ DE I-131) permitted by the technical specifications (i.e., a preaccident iodine spike case).
 - 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 $\mu\text{Ci/gm}$ 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 of all fuel pins.
3. The activity released from the fuel should be assumed to be released instantaneously and homogeneously through the primary coolant.

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," for acceptable assumptions and methodologies for performing radiological analyses.

² 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.

4. The chemical form of radioiodine released from the fuel should be assumed to be 95% 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% elemental and 3% organic. These fractions apply to iodine released as a result of fuel damage and to iodine released during normal operations, including iodine spiking.

TRANSPORT ³

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

5.1 For facilities that have not implemented alternative repair criteria (see Ref. E-1, DG-1074), 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 generator specifications (both per generator and total of all generators), the leakage should be apportioned between affected and unaffected steam generators in such a manner that the calculated dose is maximized.

5.2 The density used in converting volumetric leak rates (e.g., gpm) to mass leak rates (e.g., lbm/hr) 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 gm/cc (62.4 lbm/ft³).

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°C (212°F). The release of radioactivity from unaffected steam generators should be assumed to continue until shutdown cooling is in operation and releases from the steam generators have been terminated.

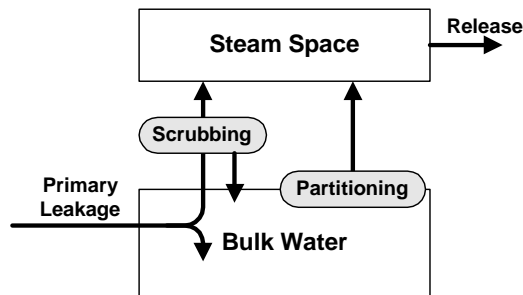
5.4 All noble gas radionuclides released from the primary system are assumed to be released to the environment without reduction or mitigation.

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

³ In this appendix, *ruptured* refers to the state of the steam generator in which primary-to-secondary leakage rate has increased to a value greater than technical specifications. *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. *Partitioning Coefficient* is defined as:

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

**Figure E-1
Transport Model**



5.5.1 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.

5.5.2 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-2), during periods of total submergence of the tubes.

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

5.5.4 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. A partition coefficient 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.

5.6 Operating experience and analyses have shown that for some steam generator designs, tube uncover may occur for a short period following any reactor trip (Ref. E-3). The potential impact of tube uncover 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.

Appendix E REFERENCES

- E-1 USNRC, "Steam Generator Tube Integrity," Draft Regulatory Guide DG-1074, December 1998.
- E-2. USNRC, "Iodine Behavior in a PWR Cooling System Following a Postulated Steam Generator Tube Rupture Accident," NUREG-0409, May 1985.
- E-3 USNRC, "Steam Generator Tube Rupture Analysis Deficiency," Information Notice 88-31, May 25, 1988.

Appendix F

ASSUMPTIONS FOR EVALUATING THE RADIOLOGICAL CONSEQUENCES OF A PWR STEAM GENERATOR TUBE RUPTURE ACCIDENT

This appendix provides assumptions acceptable to the NRC staff for evaluating the radiological consequences of a steam generator tube rupture accident at PWR light-water reactors. These assumptions supplement the guidance provided in the main body of this guide.¹

SOURCE TERM

1. Assumptions acceptable to the NRC staff regarding core inventory and the release of radionuclides from the fuel are in Regulatory Position 3 of this guide. 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.
2. If no or minimal² 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.
 - 2.1 A reactor transient has occurred prior to the postulated steam generator tube rupture (SGTR) and has raised the primary coolant iodine concentration to the maximum value (typically 60 $\mu\text{Ci/gm}$ DE I-131) permitted by the technical specifications (i.e., a preaccident iodine spike case).
 - 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 $\mu\text{Ci/gm}$ 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 of all fuel pins.
3. The activity released from the fuel, if any, should be assumed to be released instantaneously and homogeneously through the primary coolant.

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" (USNRC, December 1998), for acceptable assumptions and methodologies for performing radiological analyses.

² 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.

4. Iodine releases from the steam generators to the environment should be assumed to be 97% elemental and 3% organic.

TRANSPORT ³

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

- 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 affected and unaffected steam generators in such a manner that the calculated dose is maximized.
- 5.2 The density used in converting volumetric leak rates (e.g., gpm) to mass leak rates (e.g., lbm/hr) 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 gm/cc (62.4 lbm/ft³).
- 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° C (212° F). 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.
- 5.4 The release of fission products from the secondary system should be evaluated with the assumption of a coincident loss of offsite power.
- 5.5 All noble gas radionuclides released from the primary system are assumed to be released to the environment without reduction or mitigation.
- 5.6 The transport model described in Regulatory Positions 5.5 and 5.6 of Appendix E should be utilized for iodine and particulates.

³ In this appendix, *ruptured* refers to the state of the steam generator in which primary-to-secondary leakage rate has increased to a value greater than technical specifications.

Appendix G

ASSUMPTIONS FOR EVALUATING THE RADIOLOGICAL CONSEQUENCES OF A PWR LOCKED ROTOR ACCIDENT

This appendix provides assumptions acceptable to the NRC staff for evaluating the radiological consequences of a locked rotor accident at PWR light water reactors.¹ These assumptions supplement the guidance provided in the main body of this guide.

SOURCE TERM

1. Assumptions acceptable to the NRC staff regarding core inventory and the release of radionuclides from the fuel are in Regulatory Position 3 of this regulatory guide. 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.
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 steam line break outside containment.
3. The activity released from the fuel should be assumed to be released instantaneously and homogeneously through the primary coolant.
4. The chemical form of radioiodine released from the fuel should be assumed to be 95% 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% elemental and 3% 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

5. Assumptions acceptable to the NRC staff related to the transport, reduction, and release of radioactive material to the environment are as follows.
 - 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.
 - 5.2 The density used in converting volumetric leak rates (e.g., gpm) to mass leak rates (e.g., lbm/hr) 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.

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" (USNRC, December 1998), for acceptable assumptions and methodologies for performing radiological analyses.

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 gm/cc (62.4 lbm/ft³).

- 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° C (212° F). The release of radioactivity should be assumed to continue until shutdown cooling is in operation and releases from the steam generators have been terminated.
- 5.4** The release of fission products from the secondary system should be evaluated with the assumption of a coincident loss of offsite power.
- 5.5** All noble gas radionuclides released from the primary system are assumed to be released to the environment without reduction or mitigation.
- 5.6** The transport model described in assumptions 5.5 and 5.6 of Appendix E should be utilized for iodine and particulates.

Appendix H

ASSUMPTIONS FOR EVALUATING THE RADIOLOGICAL CONSEQUENCES OF A PWR ROD EJECTION ACCIDENT

This appendix provides assumptions acceptable to the NRC staff for evaluating the radiological consequences of a rod ejection accident at PWR light water reactors.¹ These assumptions supplement the guidance provided in the main body of this guide.

SOURCE TERM

1. Assumptions acceptable to the NRC staff regarding core inventory are in Regulatory Position 3 of this guide. For the rod ejection accident, the release from the breached fuel is based on the estimate of the number of fuel rods breached and the assumption that 10% of the core inventory of the noble gases and iodines is in the fuel gap. The release attributed to fuel melting is based on the fraction of the fuel that reaches or exceeds the initiation temperature for fuel melting and the assumption that 100% of the noble gases and 25% of the iodines contained in that fraction are available for release from containment. For the secondary system release pathway, 100% of the noble gases and 50% of the iodines in that fraction are released to the reactor coolant.
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 loss-of-coolant accident (LOCA), main steam line break, and steam generator tube rupture.
3. Two release cases are to be considered. In the first, 100% of the activity released from the fuel should be assumed to be released instantaneously and homogeneously through the containment atmosphere. In the second, 100% 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.
4. The chemical form of radioiodine released to the containment atmosphere should be assumed to be 95% cesium iodide (CsI), 4.85% elemental iodine, and 0.15% organic iodide. If containment sprays do not actuate or are terminated prior to 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 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.
5. Iodine releases from the steam generators to the environment should be assumed to be 97% elemental and 3% organic.

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" (USNRC, December 1998), for acceptable assumptions and methodologies for performing radiological analyses.

TRANSPORT FROM CONTAINMENT

- 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.
 - 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.
 - 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% 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

- 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.
 - 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.
 - 7.2** The density used in converting volumetric leak rates (e.g., gpm) to mass leak rates (e.g., lbm/hr) 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 gm/cc (62.4 lbm/ft³).
 - 7.3** All noble gas radionuclides released to the secondary system are assumed to be released to the environment without reduction or mitigation.
 - 7.4** The transport model described in assumptions 5.5 and 5.6 of Appendix E should be utilized for iodine and particulates.

Appendix I

ASSUMPTIONS FOR EVALUATING RADIATION DOSES FOR EQUIPMENT QUALIFICATION

This appendix addresses assumptions associated with equipment qualification that are acceptable to the NRC staff for performing radiological assessments. As stated in Regulatory Position 6 of this guide, this appendix supersedes Regulatory Positions 2.c.(1) and 2.c.(2) and Appendix D of Revision 1 of Regulatory Guide 1.89, "Environmental Qualification of Certain Electric Equipment Important to Safety for Nuclear Power Plants" (USNRC, June 1984), for operating reactors that have amended their licensing basis to use an alternative source term. Except as stated in this appendix, other assumptions, methods, and provisions of Revision 1 of Regulatory Guide 1.89 remain effective.

BASIC ASSUMPTIONS

1. Gamma and beta doses and dose rates should be determined for three types of radioactive source distributions: (1) activity suspended in the containment atmosphere, (2) activity plated out on containment surfaces, and (3) activity mixed in the containment sump water. A given piece of equipment may receive a dose contribution from any or all of these sources. The amount of dose contributed by each of these sources is determined by the location of the equipment, the time-dependent and location-dependent distribution of the source, and the effects of shielding. For EQ components located outside of the containment, additional radiation sources may include piping and components in systems that circulate containment sump water outside of containment. Activity deposited in ventilation and process filter media may be a source of post-accident dose.
2. The integrated dose should be determined from estimated dose rates using appropriate integration factors determined for each of the major source terms (e.g., containment sump, containment atmosphere, ECCS, normal operation). The period of exposure should be consistent with the survivability period for the EQ equipment being evaluated. The survivability period is the maximum duration, post-accident, that the particular EQ component is expected to operate and perform its intended safety function. The period of exposure for normal operation dose is generally the duration of the plant license, i.e., 40 years.

FISSION PRODUCT CONCENTRATIONS

3. The radiation environment resulting from normal operations should be based on the conservative source term estimates reported in the facility's Safety Analysis Report or should be consistent with the primary coolant specific activity limits contained in the facility's technical specifications. The use of equilibrium primary coolant concentrations based on 1% fuel cladding failures would be one acceptable method. In estimating the integrated dose from prior normal operations, appropriate historical dose rate data may be used where available.
4. The radioactivity released from the core during a design basis loss-of-coolant accident (LOCA) should be based on the assumptions provided in Regulatory Position 3 and Appendix A of this regulatory guide. Although the design basis LOCA is generally limiting for radiological

environmental qualification (EQ) purposes, there may be components for which another design basis accident may be limiting. In these cases, the assumptions provided in Appendices B through H of this regulatory guide, as applicable, should be used. Applicable features and mechanisms may be assumed in EQ calculations provided that any prerequisites and limitations identified regarding their use are met. There are additional considerations:

- For PWR ice condenser containments, the source should be assumed to be initially released to the lower containment compartment. The distribution of the activity should be based on the forced recirculation fan flow rates and the transfer rates through the ice beds as functions of time.
- For BWR Mark III designs, all the activity should be assumed initially released to the drywell area and the transfer of activity from these regions via containment leakage to the surrounding reactor building volume should be used to predict the qualification levels within the reactor building (secondary containment).

DOSE MODEL FOR CONTAINMENT ATMOSPHERE

5. The beta and gamma dose rates and integrated doses from the airborne activity within the containment atmosphere and from the plateout of aerosols on containment surfaces generally should be calculated for the midpoint in the containment, and this dose rate should be used for all exposed components. Radiation shielding afforded by internal structures may be neglected for modeling simplicity. It is expected that the shielding afforded by these structures would reduce the dose rates by factors of two or more depending on the specific location and geometry. More detailed calculations may be warranted for selected components if acceptable dose rates cannot be achieved using the simpler modeling assumptions.

6. Because of the short range of the betas in air, the airborne beta dose rates should be calculated using an infinite medium model. Other models, such as finite cloud and semi-infinite cloud, may be applicable to selected components with sufficient justification. The applicability of the semi-infinite model would depend on the location of the component, available shielding, and receptor geometry. For example, beta dose rates for equipment located on the containment walls or on large internal structures might be adequately assessed using the semi-infinite model. Use of a finite cloud model will be considered on a case-by-case method.

7. All gamma dose rates should be multiplied by a correction factor of 1.3 to account for the omission of the contribution from the decay chains of the radionuclides. This correction is particularly important for non-gamma-emitting radionuclides having gamma emitting progeny, for example, Cs-137 decay to Ba-137m. This correction may be omitted if the calculational method explicitly accounts for the emissions from buildup and decay of the radioactive progeny.

DOSE MODEL FOR CONTAINMENT SUMP WATER SOURCES

8. With the exception of noble gases, all the activity released from the fuel should be assumed to be transported to the containment sump as it is released. This activity should be assumed to mix instantaneously and uniformly with other liquids that drain to the sump. This

transport can also be modeled mechanistically as the time-dependent washout of airborne aerosols by the action of containment sprays. Radionuclides that do not become airborne on release from the reactor coolant system, e.g., they are entrained in non-flashed reactor coolant, should be assumed to be instantaneously transported to the sump and be uniformly distributed in the sump water.

9. The gamma and beta dose rates and the integrated doses should be calculated for a point located on the surface of the water at the centerline of the large pool of sump water. The effects of buildup should be considered. More detailed modeling with shielding analysis codes may be performed.

DOSE MODEL FOR EQUIPMENT LOCATED OUTSIDE CONTAINMENT

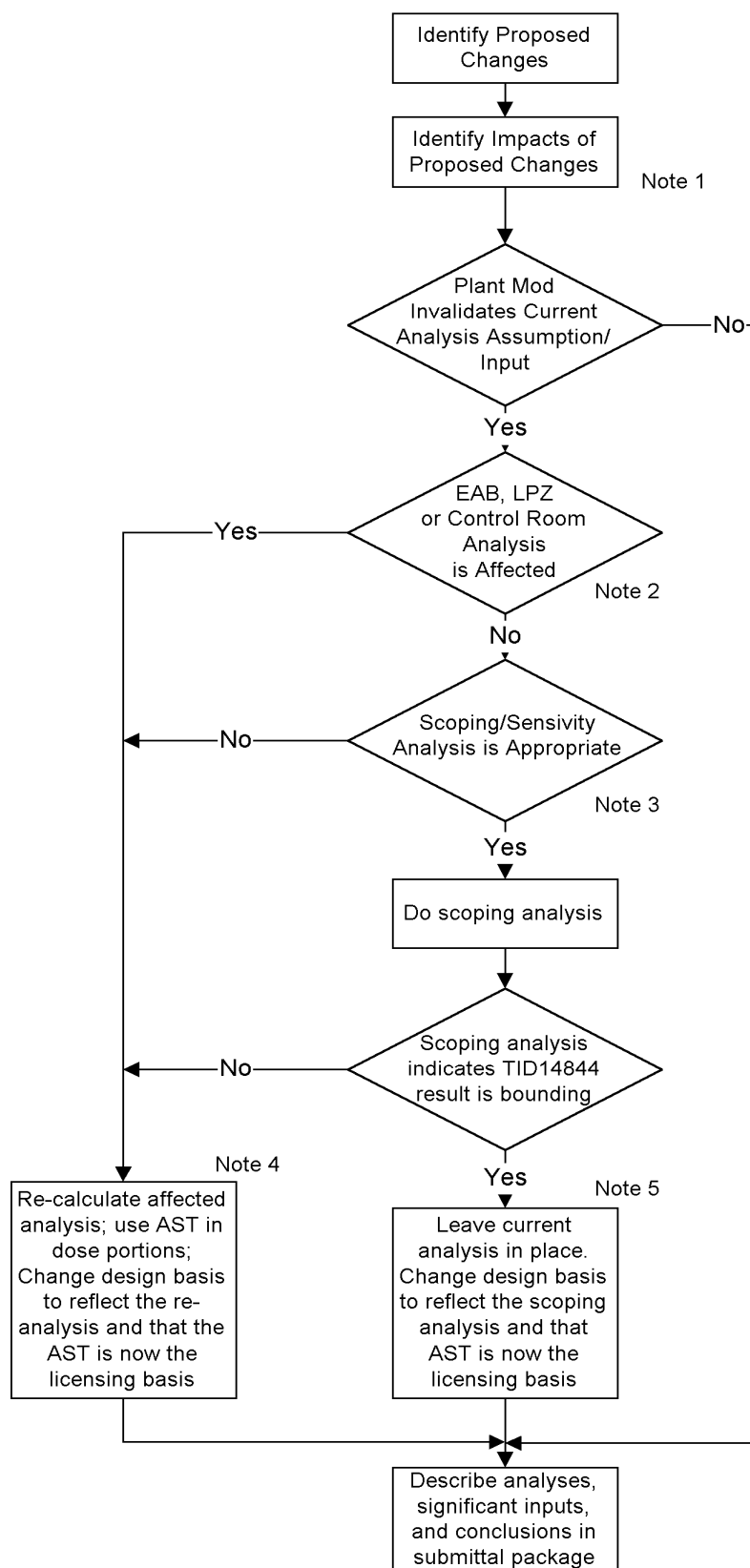
10. EQ equipment located outside of containment may be exposed to (1) radiation from sources within the containment building, (2) radiation from activity contained in piping and components in systems that re-circulate containment sump water outside of containment (e.g., ECCS, RHR, sampling systems), (3) radiation from activity contained in piping and components in systems that process containment atmosphere (e.g., hydrogen recombiners, purge systems), (4) radiation from activity deposited in ventilation and process filter media, and (5) radiation from airborne activity in plant areas outside of the containment (i.e., leakage from recirculation systems). The amount of dose contributed by each of these sources is determined by the location of the equipment, the time-dependent and location-dependent distribution of the source, and the effects of shielding.

11. Because of the large amount of EQ equipment and the complexity of system and component layout in plant buildings, it is generally not reasonable to model each EQ component. A reasonable approach is to determine the limiting dose rate from all sources in a particular plant area (e.g., cubicle, floor, building) to a real or hypothetical receptor and to base the integrated doses for all components in that area on this postulated dose rate. Individual detailed modeling of selected equipment may be performed.

12. The integrated doses from components and piping in systems recirculating sump water should assume a source term based on the time-dependent containment sump source term described above. Similarly, the doses from components that contain air from the containment atmosphere should assume a source term based on the time-dependent containment atmosphere source term described above.

13. Analyses of integrated doses caused by radiation from the buildup of activity on ventilation and process filter media in systems containing containment sump water or atmosphere or both should assume that the ventilation or process flow is at its nominal design value and that the filter media is 100% efficient for iodine and particulates. The duration of flow through the filter media should be consistent with the plant design and operating procedures. Radioactive decay in the filter media should be considered. Shielding by structures and components between the filter and the EQ equipment may be considered.

Appendix J Analysis Decision Chart



Note 1: All impacts, radiological and non-radiological, need to be evaluated. A full implementation will include, as a minimum, a full DBA LOCA analysis.

Note 2: Sensitivity /scoping analyses should not comprise a significant part of the exclusion area boundary (EAB), low population zone (LPZ), and control room analyses.

Note 3: Scoping analyses may be used where a number of similar analyses are involved and generic conclusions can be drawn. However, scoping analyses should not be used for EAB/LPZ/CR doses.

Note 4: If any dose analysis is to be re-calculated, the upgrade should address the selected (or all) characteristics of the source term and, as applicable, TEDE.

Note 5: Once the design basis source term is changed from the current design basis source term to a new AST, the selected AST becomes the design basis source term for all future radiological analyses, including revisions to those analyses that were shown to be bounding with the previous source term. There is no requirement to update these later analyses unless future plant modifications invalidate one or more assumptions, making such re-analysis necessary.

Appendix K

Acronyms

AST	Alternative source term
BWR	Boiling water reactor
CDF	Core damage frequency
CEDE	Committed effective dose equivalent
COLR	Core operating limits report
DBA	Design basis accident
DDE	Deep dose equivalent
DNBR	Departure from nucleate boiling ratio
EAB	Exclusion area boundary
EDE	Effective dose equivalent
EPA	Environmental Protection Agency
EQ	Environmental qualification
ESF	Engineered safety feature
FHA	Fuel handling accident
FSAR	Final safety analysis report
IPF	Iodine protection factor
LERF	Large early release fraction
LOCA	Loss-of-coolant accident
LPZ	Low population zone
MOX	Mixed oxide
MSLB	Main steam line break
NDT	Non-destructive testing
NSSS	Nuclear supply system supplier
PRA	Probabilistic risk assessment
PWR	Pressurized water reactor
RMS	Radiation monitoring system
SG	Steam generator
SGTR	Steam generator tube rupture
TEDE	Total effective dose equivalent
TID	Technical information document
TMI	Three Mile Island

VALUE / IMPACT STATEMENT

A separate value/impact analysis has not been prepared for this Regulatory Guide 1.183. A value/impact analysis was included in the regulatory analysis for the proposed amendments to 10 CFR Parts 21, 50, and 54 published on March 11, 1999 (64 FR 12117). This regulatory analysis was updated as part of the final amendments to 10 CFR Parts 21, 50, and 54, published in December 1999 (64 FR 71998). Copies of both regulatory analyses are available for inspection or copying for a fee in the Commission's Public Document Room at 2120 L Street NW, Washington, DC, under RGIN AG12.

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