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DRAFT REGULATORY GUIDE

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DRAFT REGULATORY GUIDE DG-1081

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

GENERAL INFORMATION

Written comments on this draft regulatory guide may be submitted to the Rules and Directives Branch, Office of Administration, U.S. Nuclear Regulatory Commission, Washington, DC 20555. Comments may be accompanied by relevant information or supporting data. Copies of comments received may be examined at the NRC Public Document Room, 2120 L Street NW., Washington, DC.

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A. INTRODUCTION

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 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.

This guide is being developed to provide guidance to licensees for operating power reactors on acceptable applications of alternative source terms; the scope, nature, and documentation of associated analyses and evaluations; consideration of impacts on

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[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 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.

analyzed risk; and content of submittals. The effective regulatory guide will establish an acceptable alternative source term (AST) and identify the significant attributes of other ASTs that may be found acceptable by the NRC staff. This guide would also identify acceptable radiological analysis assumptions for use in conjunction with the accepted AST.

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 will apply 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 address 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 AST[3] 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 Plants." 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 initiatives involving risks of shutdown operations, extended burnup fuels, and the risk-informing current regulations. The

[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.

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 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.

1.1.2 Defense in Depth

The proposed uses of an AST and the associated proposed facility modifications 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 will authorize technically justifiable partial, or selective, uses of an AST if a clear, logical design basis is maintained. Sensitivity analyses may be able to show that existing analyses are adequately conservative and re-calculation is not warranted. This approach would create two tiers of analyses, those based on the previous source term (and found to be bounding) and those based on an AST. The radiological acceptance criteria would be different with some analyses based upon whole body and thyroid criteria and some based on TEDE criteria. The facility design bases should clearly indicate that the source term assumptions and radiological criteria in these earlier 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 non-radiological, 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] 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 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.

[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).

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. 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

A 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. A full implementation replaces the previous accident source term used in all design basis radiological analyses and establishes the TEDE dose criteria. At a minimum, 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. After a full implementation is approved, all subsequent new or updated analyses should be based on the AST and TEDE criteria. 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

A 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 the maximum 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. In this case, 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. If the licensee desires to use the AST and TEDE criteria in a different application, another license amendment under 10 CFR 50.67 would be required.

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.

- Equipment Environmental Qualification (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, full or selective, of an AST and any associated facility modification is expected to 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-evaluation will necessarily be a function of the specific proposed facility modification[6] and whether a full or selective implementation is being pursued. The NRC staff does not expect a complete recalculation of all facility radiological analyses, but does expect licensees to evaluate all impacts of the proposed changes and to update the affected analyses and 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

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

[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.

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 increase containment pressure or the ability of ductwork downstream of the dampers to withstand increased stresses.

For a 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 is expected to 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. The licensee may not implement other characteristics of the AST or extend the AST to other plant modifications without prior NRC approval under 10 CFR 50.67. 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 necessary if the modified elapsed time remains a small 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 case 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 <u>Use of Sensitivity or Scoping Analyses</u>

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

A 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 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.

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 the approved AST and TEDE criteria in analyses that were not in the scope of the approved implementation 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.

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 non-radiological 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 accident source term is not set forth in 10 CFR 50.67. Regulatory Position 3 of this guide identifies an accident source term 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 alternative source term 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 the into containment, the types and quantities of the radioactive species released, and the chemical forms of iodine.

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 alternative accident source term 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 ASTs 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 Core 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 1.02 times the current licensed rated thermal power.[7] 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.[8] For non-LOCA events, the appropriate radial peaking factor from the facility's core operating limits report (COLR) should be applied. No adjustment to the core 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 Fractions[9]

The core inventory release fractions, by radionuclide groups, for the gap release and early in-vessel damage states 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.

Table 1
BWR Core Inventory Fraction Released Into Containment

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

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[7] The uncertainty factor used in determining the core inventory should be that value provided in Paragraph I.A. of Appendix K to 10 CFR Part 50.

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

[9] 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 NRC has ongoing research activities regarding the impact of extended burnup on fuel releases. These release fractions may be revised in the future pending the outcome of these activities. These data 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**
Noble Gases	0.05	0.95
Halogens	0.05	0.35
Alkali Metals	0.05	0.25
Tellurium Metals	0.00	0.05
Ba, Sr	0.00	0.02
Noble Metals	0.00	0.0025
Cerium Group	0.00	0.0005
Lanthanides	0.00	0.0002

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.[10] These fractions are applied to the equilibrium core inventory described in Regulatory Position 3.1.

Table 3
Non-LOCA
Fraction of Core Inventory in Gap

Group	**Fraction**
I-131	0.12
Kr-85	0.15
Other Noble Gases	0.10
Other Halogens	0.10
Alkali Metals	0.10

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 activity released from the core during each release phase should be modeled as increasing in a linear fashion over the duration of the phase.[11] For non-LOCA DBAs in which fuel damage is projected, the release from the fuel gap and/or the fuel pellet should be assumed to occur instantaneously with the onset of the projected damage.

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[10] The differences in magnitude between the gap releases shown in Tables 1 and 2 and the gap fractions shown in Table 3 compensate for uncertainties in the extrapolation of release data evaluated for LOCA events to the non-LOCA events. The NRC previously assumed 0.1 for all noble gases and halogens, and 30% for Kr-85. The fractions shown in Table 3 are consistent with available data for extended burnup fuel (based on the limiting assembly).

[11] 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

	PWRs		BWRs	
Phase	**Onset**	**Duration**	**Onset**	**Duration**
Gap Release	10-30 sec	0.5 hr	30 sec	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, 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 fuel, 95 percent of the iodine released should be assumed to be cesium iodide (CsI), 4.85 percent as elemental iodine, and 0.15 percent as organic iodide. With the exception of elemental and organic iodine and noble gases, fission products should be assumed to be in particulate form. 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 melting and the fraction of fuel elements for which the fuel clad is breached. Although the NRC staff has traditionally relied upon the departure from nuclear 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.

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.

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 include all radionuclides that are significant with regard to dose consequences and the released radioactivity.[12]

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. 17). 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. 18), provides tables of conversion factors acceptable to the NRC staff. The factors in the column headed “effective” yield the CEDE.

4.1.3 For the first 8 hours, the breathing rate of persons offsite should be assumed to be 3.47×10^{-4} cubic meters per second. From 8 to 24 hours following the accident, the breathing rate should be assumed to be 1.75×10^{-4} cubic meters per second. After that and until the end of the accident, the rate should be assumed to be 2.32×10^{-4} cubic meters per second.

4.1.4 The DDE should be calculated using 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 TEDE. Federal Guidance Report 12, “External Exposure to Radionuclides in Air, Water, and Soil” (Ref. 19), provides external EDE conversion factors acceptable to the NRC staff. The factors in the column headed “effective” yield the EDE.

4.1.5 The TEDE should be determined for the most limiting receptor 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. The maximum two-hour TEDE should be determined by calculating the

[12] 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.

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 reported. The timing of the increments should be consistent with the rate at which analysis parameters change.

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 radioactivity contained in the radioactive plume released from the facility,

- Contamination of the control room atmosphere by the intake or infiltration of airborne radioactivity from areas and structures adjacent to the control room envelope,

- Radiation shine from the external radioactive plume released from the facility,

- Radiation shine from the reactor containment,

- Radiation shine from radioactivity in systems and components inside or external to the control room envelope, e.g., radioactivity buildup in recirculation filters.

4.2.2 The radioactivity 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 exclusion area boundary (EAB) and the low population zone (LPZ) TEDE values, unless these assumptions would result in non-conservative results for the control room.

4.2.3 The models used to transport radioactivity into and through the control room,[13] and the shielding models used to determine radiation shine from external sources, should be structured to provide suitably conservative estimates of the exposure to control room personnel.

[13] The iodine protection factor (IPF) methodology of Reference 20 may not be adequately conservative for all DBAs and control room arrangements and should only be used with caution. The NRC computer codes HABIT (Ref. 21) and RADTRAD (Ref. 22) incorporate suitable methodologies.

4.2.4 Credit for engineered safety features that mitigate airborne activity within the control room may be assumed. Such features may include control room isolation or pressurization, 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. 23), 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. Several aspects of RMs can delay the 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 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, 60% of the time between 1 and 4 days, and 40% of the time from 4 days to 30 days.[14] For the duration of the event, the breathing rate of this individual should be assumed to be 3.47×10^{-4} cubic meters per second.

4.2.7 Control room doses should be calculated using dose conversion factors identified 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 represented by a hemisphere of volume, V, in cubic feet.

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

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. Design envelope source terms provided in NUREG-0737 should be updated for consistency with the AST. In general, all radiation exposures to plant personnel 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

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[14] This occupancy is modeled in the χ/Q values determined in Reference 20 and should not be credited twice. The ARCON96 Code (Ref. 24) does not incorporate these occupancy assumptions, making it necessary to apply this correction in the dose calculations.

In 10 CFR Part 50, § 50.67 establishes the radiological criteria for the EAB, the outer boundary of the LPZ, and for the control room. 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 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 6[15]
Accident Dose Criteria

Accident or Case	EAB and LPZ Dose Criteria
LOCA	25 rem TEDE
BWR Main Steam Line Break	
Fuel Damage or Pre-incident Spike	25 rem TEDE
Coincident Iodine Spike	2.5 rem TEDE
BWR Rod Drop Accident	6.25 rem TEDE
PWR Steam Generator Tube Rupture	
Fuel Damage or Pre-incident Spike	25 rem TEDE
Coincident Iodine Spike	2.5 rem TEDE
PWR Main Steam Line Break	
Fuel Damage or Pre-incident Spike	25 rem TEDE
Coincident Iodine Spike	2.5 rem TEDE
PWR Locked Rotor Accident	2.5 rem TEDE
PWR Rod Ejection Accident	6.25 rem TEDE
Fuel Handling Accident	6.25 rem TEDE

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

[15] For PWRs with steam generator alternative repair criteria, different dose criteria may apply to SGTR and MSLB analyses.

assumptions rather than being modeled directly. The staff has selected assumptions and models that provide an appropriate and prudent safety margin against unpredicted events in the course of an accident and compensate for large uncertainties in facility parameters, accident progression, radioactive material transport, and atmospheric dispersion. Licensees should exercise caution in proposing deviations based upon data from a specific accident sequence since the DBAs were never intended to represent any specific accident sequence -- the proposed deviation may not be conservative for other accident sequences.

5.1.2 Credit for Engineered Safeguard Features

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 maximizing the 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.[16] If a range of values or a tolerance band is specified, the value that would maximize the 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 implementation of the AST,

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[16] Note that for some parameters, the technical specification value may be superseded for analysis purposes by values provided in other regulatory guidance. For example, ESF filter efficiencies are based on the guidance in Regulatory Guide 1.52 (Ref. 23) and in Generic Letter 99-02 (Ref. 25) rather than the surveillance test criteria in the technical specifications.

warranting review to 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. 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 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, 20, and 26).

References 20 and 26 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 and should be timed to be included in the worst 2-hour exposure period. The NRC computer code PAVAN (Ref. 27) implements Regulatory Guide 1.145 (Ref. 26) and its use is acceptable to the NRC staff. The methodology of the

NRC computer code ARCON96[17] (Ref. 24) 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. 28). 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 should be used 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.

D. IMPLEMENTATION

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

This draft guide has been released to encourage public participation in its development. 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 to be described in the active guide incorporating public comments will be used in the evaluation of submittals related to the use of ASTs in radiological consequence analyses at operating power reactors.

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[17] 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.

REFERENCES

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

1. J.J. DiNunno et al., “Calculation of Distance Factors for Power and Test Reactor Sites,” USAEC TID-14844, 1962.

2. USNRC, “Clarification of TMI Action Plan Requirements,” NUREG-0737, November 1980.

3. USNRC, “Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants,” NUREG-0800, September 1981 (or updates of specific sections).

4. USNRC, “Use of Probabilistic Risk Assessment Methods in Nuclear Activities: Final Policy Statement,” *Federal Register*, Volume 60, page 42622 (60 FR 42622) August 16, 1995.

5. L. Soffer et al., “Accident Source Terms for Light-Water Nuclear Power Plants,” NUREG-1465, February 1995.

6. USNRC, “Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident for Boiling Water Reactors.” Regulatory Guide 1.3, Revision 2, June 1974.

7. USNRC, “Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident for Pressurized Water Reactors,” Regulatory Guide 1.4, Revision 2, June 1974.

8. USNRC, “Assumptions Used for Evaluating the Potential Radiological Consequences of a Steam Line Break Accident for Boiling Water Reactors,” Regulatory Guide 1.5, March 1971.

9. USNRC, “Assumptions Used for Evaluating the Potential Radiological Consequences of a Fuel Handling Accident in the Fuel Handling and Storage Facility for Boiling and Pressurized Water Reactors,” Regulatory Guide 1.25, March 1972.

10. USNRC, “Assumptions Used for Evaluating a Control Rod Ejection Accident for Pressurized Water Reactors,” Regulatory Guide 1.77, May 1974.

11. USNRC, “Environmental Qualification of Certain Electric Equipment Important to Safety for Nuclear Power Plants,” Regulatory Guide 1.89, Revision 1, June 1984.

12. USNRC, “Planning Basis for the Development of State and Local Government Radiological Emergency Response Plans in Support of Light Water Nuclear Power Plants,” NUREG-0396, December 1978.

13. USNRC, “Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants,” NUREG-0654, Revision 1 (FEMA-REP-1), November 1980.

14. USNRC, “Results of the Revised (NUREG-1465) Source Term Rebaselining for Operating Reactors,” SECY-98-154, June 30, 1998.

15. USNRC, “An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis,” Regulatory Guide 1.174, July 1998.

16. USNRC, “Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants (LWR Edition),” Regulatory Guide 1.70, Revision 3, November 1978.

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17. ICRP, “Limits for Intakes of Radionuclides by Workers,” ICRP Publication 30, 1979.

18. K.F. Eckerman et al., “Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion,” EPA-520/1-88-020, 1988.

19. K.F. Eckerman and J.C. Ryman, “External Exposure to Radionuclides in Air, Water, and Soil,” Federal Guidance Report 12, EPA-402-R-93-081, 1993.

20. K.G. Murphy and K.W. Campe, “Nuclear Power Plant Control Room Ventilation System Design for Meeting General Criterion 19,” published in *Proceedings of 13th AEC Air Cleaning Conference,* Atomic Energy Commission (now NRC), August 1974.

21. USNRC, “Computer Codes for Evaluation of Control Room Habitability (HABIT V1.1),” Supplement 1 to NUREG/CR-6210, November 1998.

22. S.L. Humphreys et al., “RADTRAD: A Simplified Model for Radionuclide Transport and Removal and Dose Estimation,” NUREG/CR-6604, USNRC, April 1998.

23. 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.

24. J.V. Ramsdell and C.A. Simonen, “Atmospheric Relative Concentrations in Building Wakes, NUREG-6331, Revision 1, USNRC, May 1997.

25. USNRC, “Laboratory Testing of Nuclear-Grade Activated Charcoal,” NRC Generic Letter 99-02, June 3, 1999.

26. USNRC, “Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants,” Regulatory Guide 1.145, Revision 1, November 1982.

27. T.J. Bander, “PAVAN: An Atmospheric Dispersion Program for Evaluating Design Basis Accidental Releases of Radioactive Materials from Nuclear Power Stations,” NUREG-2858, November 1982.

28. USNRC, “Onsite Meteorological Programs,” Regulatory Guide 1.23, February 1972.

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 Draft Regulatory Guide DG-1081.

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 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 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. 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 assure 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 SRP, NUREG-0800 (Ref. 1) and in NUREG/CR-6189, "A Simplified Model of Aerosol Removal by Natural Processes in Reactor Containments" (Ref. 2). The latter model is incorporated into the analysis code RADTRAD (Ref. 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. 1) may be credited. Acceptable models for the removal of iodine and aerosols are described in Chapter 6.5.2 of the SRP[1] and NUREG/CR-5966, "A Simplified Model of Aerosol Removal by Containment Sprays"[2] (Ref. 4). This simplified model is incorporated into the analysis code RADTRAD (Refs. 1-3).

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.

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

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[1] The SRP establishes 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. Since the activity is now assumed to be released continuously over a period of time, the maximum activity needs to be redefined. If the release from the fuel is to be modeled as a linear ramp over the duration of the release phase, the maximum activity should be the activity remaining at the end of the early in-vessel phase. If the release from the fuel is assumed to occur as a step increase at the start of the early in-vessel release phase, maximum activity should be the activity assumed to be released at that time.

[2] 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.

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 Chapter 6.5.4 of the SRP (Ref. 1).

3.7 The primary containment 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, but not less than 50% of the maximum leak rate, after the first 24 hours if the reduced leak rate is supported by plant configuration and analyses. 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.

3.8 For BWRs with Mark III containments, the flow rate 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 flow rate should be assumed during the two-hour period between the initial blowdown and termination of the fuel radioac-tivity release (gap and early in-vessel release phases). After two hours, the radioac-tivity is assumed to be uniformly distributed throughout the drywell and the primary containment.

3.9 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 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 release point 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. 5) and Generic Letter 99-02 (Ref. 6).

Assumptions on ESF System Leakage

5. Engineered safety feature (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. 8). 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 should be used in 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 should be assumed to be instantaneously and homogeneously mixed 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 that transport containment airborne activity 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. 9), 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., ECCS pump miniflow return to the refueling water storage tank.

5.3 In addition to the leakage specified in paragraph 5.2, the evaluation should assume leakage from a gross failure of a passive component at the rate of 50 gallons per minute, starting 24 hours after the accident and lasting for 30 minutes. This evaluation is not required if the facility design provides an ESF ventilation filtration system that exhausts the areas of potential leakage.

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

5.5 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_{f1} is the enthalpy of liquid at system design temperature and pressure; h_{f2} is the enthalpy of liquid at saturation conditions (14.7 psia, 212ºF); and h_{fg} is the heat of vaporization at 212ºF.

5.6 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 activity in the leaked fluid, unless a smaller amount can be justified based on the actual sump pH history and area ventilation rates.

5.7 The radioiodine that is postulated to become airborne should be assumed 97% elemental and 3% organic and should be assumed to be released to the environment. 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. 5) and Generic Letter 99-02 (Ref. 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 should be used 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 from 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, but not less than 50% of the maximum leak rate, after the first 24 hours if the reduced leak rate is supported by site-specific analyses.

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 rather than on slug flow, but other models 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 Reduction in MSIV releases that are 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. Reference 10 provides 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 applicable, 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.

Appendix A REFERENCES

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

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

A-3 S.L. Humphreys et al., “RADTRAD: A Simplified Model for Radionuclide Transport and Removal and Dose Estimation,” NUREG/CR-6604, USNRC, April 1998.

A-4 D.A. Powers and S.B. Burson, “A Simplified Model of Aerosol Removal by Containment Sprays,” NUREG/CR-5966, USNRC, June 1993.

A-5 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.

A-6 USNRC, “Laboratory Testing of Nuclear Grade Activated Charcoal,” Generic Letter 99-02, June 3, 1999.

A-7 D.A. Powers, “A Simplified Model of Decontamination by BWR Steam Suppression Pools,” NUREG/CR-6153, USNRC, May 1996.

A-8 USNRC, “Potential Radioactive Leakage to Tank Vented to Atmosphere,” Information Notice 91-56, September 19, 1991.

A-9 USNRC, “Clarification of TMI Action Plan Requirements,” NUREG-0737, November 1980.

A-10 J.E. Cline, “MSIV Leakage Iodine Transport Analysis,” Letter Report dated March 26, 1991. (ADAMS Accession Number ML993280219)

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

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. Chapter 15.7.4, "Radiological Consequences of Fuel Handling Accidents," of the SRP (Ref. B-1) contains an example of a conservative bounding analysis.

1.2 The gap activity fractions of Table 3 in Regulatory Position 3 of Draft Regulatory Guide DG-1081 should be assumed. 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 Of the radioiodine released from the fuel, 99.75% of the released iodine should be assumed to be in the form of elemental iodine and 0.25% in organic species.

2. Water Depth

If the depth of water above the damaged fuel is 23 feet or greater, the decontamina-tion 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 and organic iodine species results in the iodine above the water being composed of 45% elemental and 55% 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-2).

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 should be made.

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-3, B-4). Delays in radiation detection, actuation of the ESF filtration system, or diversion of ventilation flow to the ESF filtration system[1] 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.

5. Fuel Handling Accidents Within Containment

For fuel handling accidents postulated to occur within the containment, the following assumptions should be made:

5.1 If the containment is isolated[2] 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 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

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[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.

[3] Administrative controls are to be established to close the airlock or hatch in less than 30 minutes. These controls generally state that a dedicated individual must be present at the open airlock or hatch while fuel handling operations are in progress and that this individual must have any necessary equipment to close the airlock or hatch in the required time. Radiological analyses should generally not credit this manual isolation.

guidance of Regulatory Guide 1.52 and Generic Letter 99-02 (Refs. B-3 and B-4). 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.

Appendix B REFERENCES

B-1 USNRC, “Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants.” NUREG-0800.

B-2. G. Burley, “Evaluation of Fission Product Release and Transport,” 1971. (NRC Accession number 8402080322)

B-3. 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-4. 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 and the release of radionuclides from the fuel are provided in Regulatory Position 3 of Draft Regulatory Guide DG-1081. The release from breeched fuel clad should be based on the gap inventory fractions in Table 3 of DG-1081 and the estimate of clad damage. The release from melted fuel should be based on the early in-vessel phase data in Table 1 and the percentage of the fuel affected.

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

2.1 The concentration that is the maximum value (typically 4 µCi/gm DE I-131) permitted and corresponds to the conditions of an assumed pre-accident spike and

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

3. The assumptions 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 melt 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

[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.

ground- level release at a rate of 1% per day[2] for a period of 24 hours, at which time the leakage is assumed to terminate. No credit should be assumed for dilution or holdup within the turbine building. Radioactive decay during holdup in the turbine and condenser may be assumed.

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 accounts 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 release from the reactor coolant within the pressure vessel should be assumed to consist of 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.

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[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.

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 regarding core inventory and the release of radionuclides from the fuel are provided in Regulatory Position 3 of this Draft Regulatory Guide DG-1081. The release from breeched fuel clad should be based on the gap inventory fractions in Table 3 of DG-1081 and the estimate of clad damage. The release from melted fuel should be based on the early in-vessel phase data in Table 1 and the percentage of the fuel affected.

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

2.1 The concentration that is the maximum value (typically 4.0 µCi/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 µCi/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. The assumptions 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.

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[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 within 2 hours as a ground-level release. No credit should be assumed for plateout, holdup, or dilution within facility buildings.

4.4 The iodine release from the main steam line should be assumed to consist of 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 regarding core inventory and the release of radionuclides from the fuel are provided in Regulatory Position 3 of this regulatory guide. The release from breeched fuel clad should be based on the gap inventory fractions in Table 3 of DG-1081 and the estimate of clad damage. The release from melted fuel should be based on the early in-vessel phase data in Table 2 of DG-1081and the percentage of the fuel affected. The fuel damage estimate should assume that the highest worth control rod is stuck at its fully withdrawn position.

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

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 µCi/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 µCi/gm DE I-131) specified in technical specifications (i.e., concurrent iodine spike case). A concurrent iodine spike need not be considered if fuel damage is postulated. The assumed iodine spike duration should be 8 hours.

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

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[1] Facilities licensed with, or applying for, alternative repair criteria (ARC) should use this section in conjunction with the guidance that is being developed in Draft Regulatory Guide DG-1074, "Steam Generator Tube Integrity," 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 via the steam generators 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. The assumptions 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 (ARC) (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/ft^3).

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 All iodine and particulate radionuclides released from the primary system via the faulted steam generators should be assumed to be released to the environment with no mitigation.

5.6 During periods of total submergence of the tubes in the non-faulted steam generators, the primary-to-secondary leakage should be assumed to mix with the

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[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 due to a MSLB such that protective system response (main steam line isolation, reactor trip, safety injection, etc.) has occurred. *Partitioning Coefficient* is defined as:

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

bulk water in the steam generators. This leakage is released to the environment at a rate based on the steam mass flow rate from the steam generators.

5.7 A partitioning coefficient for elemental iodine of 100 should be assumed during periods of total submergence of the tubes in the nonfaulted steam generators. The retention of particulate radionuclides in the steam generators is limited by the moisture carryover from the steam generators.

5.8 Operating experience and analyses have shown that for some steam generator designs, tube uncovery may occur for a short period following any reactor trip (Ref. E-2). Primary-to-secondary leakage that occurs during these periods should be assumed to be released to the environment without mixing in the steam generator bulk water and no credit should be taken for iodine partititioning. The impact of emergency operating procedure restoration strategies on steam generator water level need to be considered.

Appendix E REFERENCES

E-1 USNRC, “Steam Generator Tube Integrity,” Draft Regulatory Guide DG-1074, December 1998.

E-2 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 regarding core inventory and the release of radionuclides from the fuel are in Regulatory Position 3 of this draft regulatory guide. The release from breeched fuel clad should be based on the gap inventory fractions in Table 3 of DG-1081and the estimate of clad damage. The release from melted fuel should be based on the early in-vessel phase data in Table 2 and the percentage of the fuel affected.

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

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 µCi/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 µCi/gm DE I-131) specified in technical specifications (i.e., concurrent iodine spike case). A concurrent iodine spike need not be considered if fuel damage is postulated. The assumed iodine spike duration should be 8 hours.

3. The activity released from the fuel, if any, should be assumed to be released instantaneously and homogeneously through the primary coolant.

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

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[1] Facilities licensed with, or applying for, alternative repair criteria (ARC) should use this section in conjunction with the guidance that is being developed in Draft Regulatory Guide DG-1074, "Steam Generator Tube Integrity" (Ref. F-1), 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.

Transport[3]

5. The assumptions 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/ft^3).

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 During periods of total submergence of the tubes in the ruptured steam generator, the transport model described in this section should be utilized for iodine and particulates. This model is shown in Figure F-1 and summarized below:

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

- The leakage that 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. F-2).

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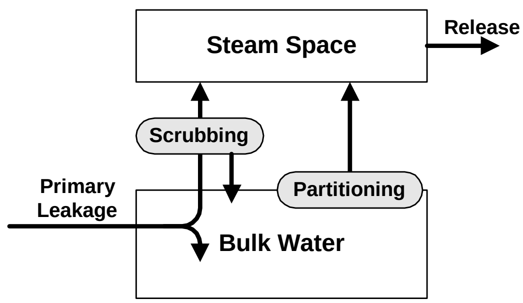
[3] In this appendix, *Ruptured* refers to the state of the steam generator in which primary-to-secondary leakage rate has increased to a value greater than technical specifications. *Partitioning Coefficient* is defined as:

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

- The leakage that does not flash is assumed to mix with the bulk water and will become vapor at a rate that is the function of the steaming rate and the partition coefficient.
- A partitioning coefficient for elemental iodine of 100 may be assumed during periods of total submergence of the tubes. The retention of particulate radionuclides in the steam generators is limited by the moisture carryover from the steam generators.

Figure F-1
Transport Model



5.7 For the non-ruptured steam generators used to perform post-event cooldown, primary coolant leakage should be assumed to mix with the bulk water without flashing. A partitioning coefficient for elemental iodine of 100 may be assumed during periods of total submergence of the tubes. The retention of particulate radionuclides in the steam generators is limited by the moisture carryover from the steam generators, as discussed above.

5.8 Operating experience and analyses have shown that for some steam generator designs, tube uncovery may occur for a short period following any reactor trip (Ref. F-3). Primary-to-secondary leakage that occurs during these periods should be assumed to be released to the environment without mixing in the steam generator bulk water and no credit should be taken for iodine partitioning. The impact of emergency operating procedure restoration strategies on steam generator water level need to be considered.

Appendix F REFERENCES

F-1. USNRC, “Steam Generator Tube Integrity,” Draft Regulatory Guide DG-1074, December 1998.

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

F-3. USNRC, “Steam Generator Tube Rupture Analysis Deficiency,” Information Notice 88-31, May 25, 1988.

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 regarding core inventory and the release of radionuclides from the fuel are in Regulatory Position 3 of this regulatory guide. The release from breeched fuel clad should be based on the gap inventory fractions in Table 3 of this guide and the estimate of clad damage. The release from melted fuel should be based on the early in-vessel phase data in Table 2 and the percentage of the fuel affected.

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 and steam generator tube rupture.

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 via the steam generators 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. The assumptions 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. 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/ft^3).

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

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 During periods of total submergence of the steam generator tubes, the primary-to-secondary leakage should be assumed to mix with the bulk water in the steam generators. This leakage is released to the environment at a rate based on the steam mass flow rate from the steam generators.

5.7 A partitioning coefficient[2] of 100 should be assumed for elemental iodine during periods of total submergence of the tubes. The retention of particulate radionuclides in the steam generators is limited by the moisture carryover from the steam generators.

5.8 Operating experience and analyses have shown that, for some steam generator designs, tube uncovery may occur for a short period following any reactor trip (Ref. G-2). Primary-to-secondary leakage that occurs during these periods should be assumed to be released to the environment without mixing in the steam generator bulk water and no credit should be taken for iodine partitioning. The impact of emergency operating procedure restoration strategies on steam generator water level needs to be considered.

Appendix G REFERENCES

G-1. USNRC, “Steam Generator Tube Integrity,” Draft Regulatory Guide DG-1074, December 1998.

G-2 USNRC, “Steam Generator Tube Rupture Analysis Deficiency,” Information Notice 88-31, May 25, 1988.

[2] Partitioning Coefficient is defined as:

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

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 regarding core inventory and the release of radionuclides from the fuel are in Regulatory Position 3 of this guide. The release from breeched fuel clad should be based on the gap inventory fractions in Table 3 of DG-1081 and the estimate of clad damage. The release from melted fuel should be based on the early in-vessel phase data in Table 2 and the percentage of the fuel affected.

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 LOCA 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 via the steam generators should be assumed to be 97% elemental and 3% organic.

Transport From Containment

6. The assumptions related to the transport, reduction, and release of radioactive material in and from the containment are as follows.

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

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 LOCAs.

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. The assumptions 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/ft^3).

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

7.4 During periods of total submergence of the steam generator tubes, the primary-to-secondary leakage should be assumed to mix with the bulk water in the steam generators. This leakage is released to the environment at a rate based on the steam mass flow rate from the steam generators.

7.5 A partitioning coefficient[2] of 100 should be assumed for elemental iodine during periods of total submergence of the tubes. The retention of particulate radionuclides in the steam generators is limited by the moisture carryover from the steam generators.

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[2] Partitioning Coefficient is defined as:

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

7.6 Operating experience and analyses have shown that, for some steam generator designs, tube uncovery may occur for a short period following any reactor trip (Ref. H-2). Primary-to-secondary leakage that occurs during these periods should be assumed to be released to the environment without mixing in the steam generator bulk water and no credit should be taken for iodine partitioning. The impact of emergency operating procedure restoration strategies on steam generator water level needs to be considered

APPENDIX H REFERENCES

H-1 USNRC, "Steam Generator Tube Integrity," Draft Regulatory Guide DG-1074, December 1998.

H-2 USNRC, "Steam Generator Tube Rupture Analysis Deficiency," Information Notice 88-31, May 25, 1988.

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 DG-1081, 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" (Ref. I-1), 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.

Fission Product Concentrations

2. 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.

3. 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 G of this regulatory guide, as applicable, should be used. The other appendices to this regulatory guide identify facility features and natural phenomena that may be considered in design basis analyses. 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

4. 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 since their inclusion would involve a higher degree of complexity than is warranted. 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 assumptions.

5. 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.

6. 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 isotopes.

Dose Model For Containment Sump Water Sources

7. 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.

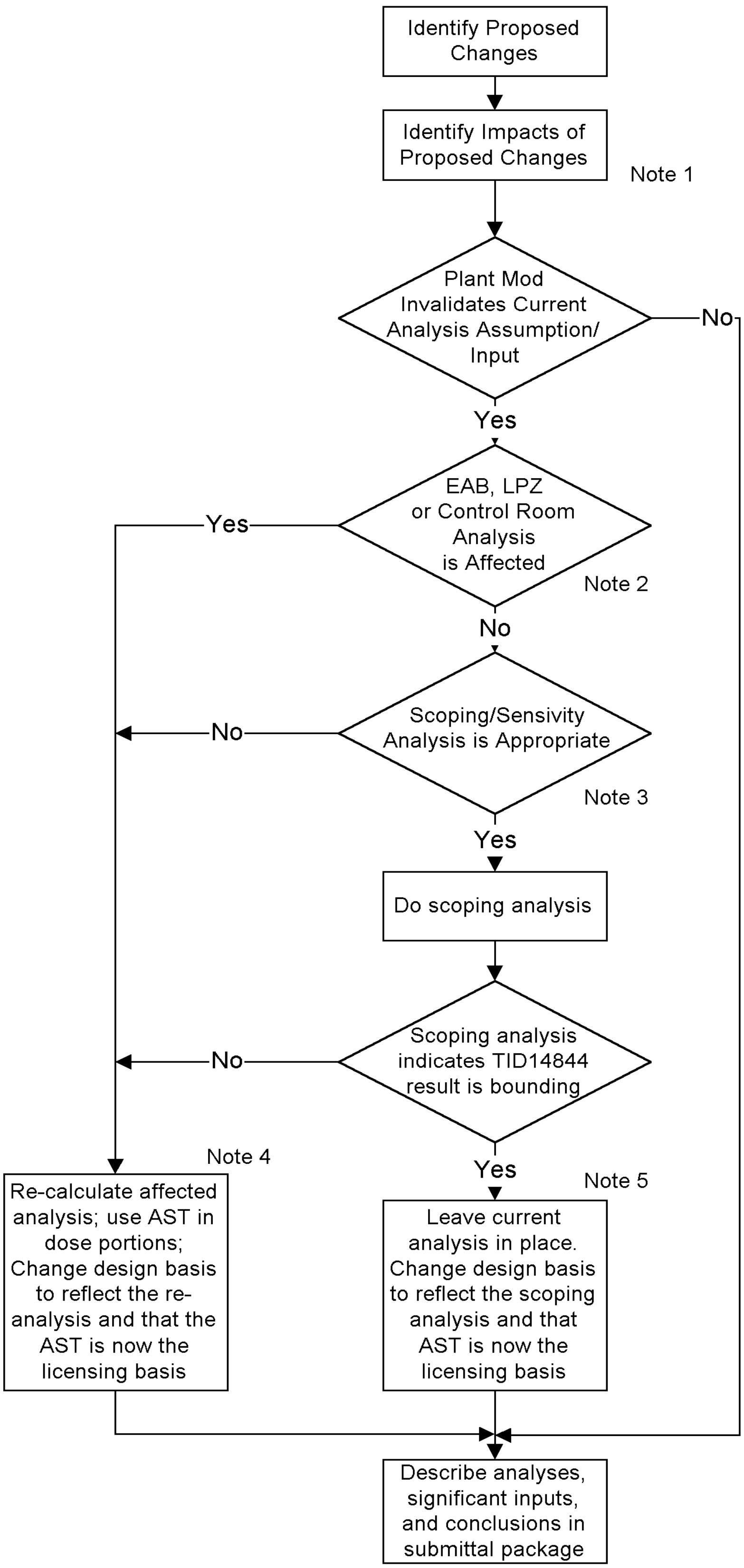
8. The gamma and beta dose rates and the integrated doses should be calculated for a point located on the surface of the water at the centerline of the large pool of sump water. The effects of buildup should be considered.

Appendix I REFERENCE

I-1. USNRC, “Environmental Qualification of Certain Electric Equipment Important to Safety for Nuclear Power Plants,” Regulatory Guide 1.89, June 1984.

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
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
PRA	Probabilistic risk assessment
PWR	Pressurized water reactor
RMS	Radiation monitoring system
NDT	Non-destructive testing
TEDE	Total effective dose equivalent
TID	Technical information document
TMI	Three Mile Island

VALUE / IMPACT STATEMENT

A separate draft value/impact analysis has not been prepared for this draft guide. 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, expected to be published in December 1999. 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.