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Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Plants

Comment On: NRC-2021-0179-0001
Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors

Document: NRC-2021-0179-DRAFT-0011
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Submitter Information

Name: BRIAN MAGNUSON
Address:
Aurora, IL, 60504
Email: magnuson28@msn.com
Phone: 309-948-4274

General Comment

June 20, 2022

RE: Public comments on draft regulatory guide (DG), DG-1389, “Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors”

Dear NRC Staff:

I am opposed to DG-1389.

My prior public comments referenced SAND2008-6601 which determined the BWR MSIV source term methodologies provided in RG 1.183 (Revision 0) are “non-conservative and conceptually inaccurate” in 2008. Additionally, my prior comments expounded on SAND2008-6601 and identified other examples in which RG 1.183 methodologies violate the laws of physics. RG 1.183 allows nuclear power plants (NPPs) to ignore the laws of physics in accident dose calculations that are used to demonstrate compliance with nuclear safety regulations, including General Design Criterion-19 (Appendix A to 10 CFR Part 50). In other words, the errors in RG 1.183 financially benefit nuclear power plants at the expense of public safety.

It appears DG-1389 may correct a few of the technical errors in Revision 0 of RG 1.183; however, any corrections would be negated because it states:
“Revision 0 of RG 1.183 will continue to be available for use by licensees and applicants as a method acceptable to the NRC staff for demonstrating compliance with the regulations.”

RG 1.183 Revision 0 has a broad range of safety ramifications. Until the NRC has reconciled the errors that SAND2008-6601 identified and I reported in prior public comments, it seems imprudent of the NRC to claim it is an acceptable method for demonstrating compliance with regulations. In effect, the errors identified in RG 1.183 Revision 0 provide a means for nuclear power plants to ignore the laws of physics in accident dose calculations in order to feign compliance with federal nuclear safety regulations.

The (Beyond) Design-Basis Accident Contravention
I am opposed to using the DG-1389 term "maximum hypothetical accident (MHA) loss-of-coolant accident (LOCA)." An NRC Regulatory/Draft guide cannot legally be used to redefine "the accident described in the applicable regulations." For example, the applicability of Appendix A to Part 50, General Design Criterion—19 cannot be limited. Nevertheless, the apparent attempt drew attention to the most egregious contravention of RG 1.183 (and DG-1389).

To begin, the NRC acknowledged: "In 1971 Appendix A, "General Design Criteria for Nuclear Power Plants," was added to 10 CFR Part 50. General Design Criterion 19 (GDC-19) specified that adequate protection shall be provided to permit access and occupancy of the control room for the duration of an accident without exceeding a radiation exposure of 5 rem whole body or its equivalent to any part of the body. From its inception, GDC-19 became the limiting dose criteria in almost all radiological dose consequence analyses."

To be clear, GDC-19 was not "limiting" by the late 1970s. By then, the NRC discovered that BWR MSIV leakage was a significant contributor to control room operator doses. Despite this disturbing discovery, the NRC neglected to require nuclear power plants to add this contribution to their accident dose calculations. However, in 2000, the NRC suggested that some nuclear power plants might wish to add MSIV leakage dose contributions to their accident dose calculations if they wanted to reap the "cost-beneficial licensing actions" provided by RG 1.183.

Despite Sandia National Laboratories (SAND2008-6601) and my reports, the NRC continues to allow nuclear power plants to exploit the RG 1.183 errors for "cost-beneficial licensing actions." The NRC allowed nuclear power plants to exploit the errors to (1) increase MSIV technical specification allowable leakage; (2) increase reactor thermal power (electrical generation); (3) increase fuel burnup times and; (4) extend (sometimes twice) the licensing life of old nuclear power plants—that have been violating GDC-19 since its inception.

Based on the timeline of NRC actions since 1971, it appears an underlying purpose of RG 1.183 is to evade the minimum design criteria set forth in Appendix A to Part 50, and purpose of 10 CFR 50.67 is to evade the more limiting requirements 10 CFR 100.11, "Determination of Exclusion Area, Low Population Zone, and Population Center Distance."

Reference 'DG-1389 Magnuson Complete Public Comments" (attached).

Sincerely,
Brian D. Magnuson
magnuson28@msn.com
1020 Station Blvd. #212
Aurora, IL 60504
Lead Emergency Management Specialist—Constellation (formerly Exelon Generation)
Former NRC Licensed Senior Reactor Operator/Operations Shift Manager at QC NPP
—Acting expressly as a member of the public

Attachments

1. 1982.04.16 Information Notice No. 82-23_ Main Steam Isolation Valve (MSIV) Leakage _ NRC.gov

2. 2000.07.00 RG 1.183 Rev. 0

7. 2010.03.10 NRC denies BWROG request to extend comments on DG-1199. ML100500773

6. 2010.01.06 BWROG DG-1199 Comments. 20 times greater .ML100081013

8. 2011.07.26 Technical Basis for DG-1199 . ML111890397

5. 2009.10.00 DG-1199 . ML090960464

14. 1990.02.01 Information Notice No. 90-08_ KR-85 Hazards From Decayed Fuel _ NRC.gov

12. 2020.12.08 Regulatory Guide 1.183 Revision Public Meeting - Magnuson Comments ML20351A321

9. 2018.06.25 Periodic Review of RG 1.183 -REVISE ML18159A069

4. 2009.06.11 Response to a Non-Concurrence on DG-1199 ML091520056

15. 2020.11.08 Mag comment on PRM 50-122 . NRC-2020-0150-0005_attachment_1 (1)

13. 2021.03.05 NRC Power Point - Revision of Regulatory Guide 1.183 . ML21056A058

3. 2008.10.00 SAND2008-6601 . ML083180196

2020.05.31 PRM 50-122 Accident Source Term Methodologies

DG-1389 Magnuson Complete Public Comments

Information Notice No. 82-23: Main Steam Isolation Valve (MSIV) Leakage

SSINS No.: 6835
IN 82-23

UNITED STATES
NUCLEAR REGULATORY COMMISSION
OFFICE OF INSPECTION AND ENFORCEMENT
WASHINGTON, D.C. 20555

July 16, 1982

IE INFORMATION NO. 82-23: MAIN STEAM ISOLATION VALVE (MSIV) LEAKAGE

Addressees:

All boiling water power reactor facilities holding an operating license or a construction permit.

Purpose:

This information notice is provided as notification of events that may have safety significance. It is expected that recipients will review the information for applicability to their facilities; however, no specific action or response is required at this time.

Description of Circumstances:

IE has completed a survey of MSIV performance at BWRs for the years 1979 through 1981. IE found that 19 of 25 operating BWRs had MSIVs which failed to meet, during one or more surveillance tests, the limiting condition for operation (LCO) which specifies the maximum permissible leak rate. The number of MSIV test failures exceeded 151 and occurred with MSIVs supplied by all three MSIV vendors, i.e., Atwood & Morrill, Crane, and Rockwell.

Measured leak rates which exceeded the LCO ranged from greater than 11.5 standard cubic feet per hour (scfh) to 3427 scfh. Twelve stations had 57 MSIV tests with results greater than 11.5 scfh and less than 100 scfh, and five stations (nine units) had 66 MSIV tests with results between 100 and 3500 scfh. Four other licensees had more than 24 test failures but did not measure, estimate, or report the magnitudes of the leak rates. These results are summarized in Attachment (1) and are shown in detail in Attachment (2).

This information indicates that some MSIVs may not adequately limit release of radioactivity to the environment if called upon to do so. NRC is considering the need for improved MSIV maintenance, more frequent MSIV testing or installation of leakage control systems.

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If you have any questions regarding this matter, please contact the Regional Administrator of the appropriate NRC Regional Office, or this office.

Edward L. Jordan, Director
Division of Engineering and
Quality Assurance
Office of Inspection and Enforcement

Technical Contact: R. W. Woodruff
49-24507

Attachments:
1. Distribution of MSIV Test Failures
2. Adverse MSIV Test ResUlts for 1979 - 1981
3. List of Recently Issued IE Information Notices

U.S. NUCLEAR REGULATORY COMMISSION July 2000

REGULATORY GUIDE

OFFICE OF NUCLEAR REGULATORY RESEARCH

REGULATORY GUIDE 1.183

(Draft was issued as DG-1081)

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

AVAILABILITY INFORMATION

Single copies of regulatory guides, both active and draft, and draft NUREG documents may be obtained free of charge by writing the Reproduction and Distribution Services Section, OCIO, USNRC, Washington, DC 20555-0001, or by email to <DISTRIBUTION@NRC.GOV>, or by fax to (301)415-2289. Active guides may also be purchased from the National Technical Information Service on a standing order basis. Details on this service may be obtained by writing NTIS, 5285 Port Royal Road, Springfield, VA 22161.

Many NRC documents are available electronically in our Reference Library on our web site, <WWW.NRC.GOV>, and through our Electronic Reading Room (ADAMS, or PARS, document system) at the same site. Copies of active and draft guides and many other NRC documents are available for inspection or copying for a fee from the NRC Public Document Room at 2120 L Street NW., Washington, DC; the PDR's mailing address is Mail Stop LL-6, Washington, DC 20555; telephone (202)634-3273 or (800)397-4209; fax (202)634-3343; email is <PDR@NRC.GOV>.

Copies of NUREG-series reports are available at current rates from the U.S. Government Printing Office, P.O. Box 37082, Washington, DC 20402-9328 (telephone (202)512-1800); or from the National Technical Information Service by writing NTIS at 5285 Port Royal Road, Springfield, VA 22161; telephone (703)487-4650; or on the internet at <http://www.ntis.gov/ordernow>. Copies are available for inspection or copying for a fee from the NRC Public Document Room at 2120 L Street NW., Washington, DC; the PDR's mailing address is Mail Stop LL-6, Washington, DC 20555; telephone (202)634-3273 or (800)397-4209; fax (202)634-3343; email is <PDR@NRC.GOV>.

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APPENDICES

A. INTRODUCTION

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

In 10 CFR Part 50, “Domestic Licensing of Production and Utilization Facilities,” Section 50.34, “Contents of Applications; Technical Information,” requires that each applicant for a construction permit or operating license provide an analysis and evaluation of the design and performance of structures, systems, and components of the facility with the objective of assessing the risk to public health and safety resulting from operation of the facility. Applicants are also required by 10 CFR 50.34 to provide an analysis of the proposed site. In 10 CFR Part 100, “Reactor Site Criteria,” Section 100.11,[1] “Determination of Exclusion Area, Low Population Zone, and Population Center Distance,” provides criteria for evaluating the radiological aspects of the proposed site. A footnote to 10 CFR 100.11 states that the fission product release assumed in these evaluations should be based upon a major accident involving substantial meltdown of the core with subsequent release of appreciable quantities of fission products.

Technical Information Document (TID) 14844, “Calculation of Distance Factors for Power and Test Reactor Sites” (Ref. 1), is cited in 10 CFR Part 100 as a source of further guidance on these analyses. Although initially used only for siting evaluations, the TID-14844 source term has been used in other design basis applications, such as environmental qualification of equipment under 10 CFR 50.49, “Environmental Qualification of Electric Equipment Important to Safety for Nuclear Power Plants,” and in some requirements related to Three Mile Island (TMI) as stated in NUREG-0737, “Clarification of TMI Action Plan Requirements” (Ref. 2). The analyses and evaluations required by 10 CFR 50.34 for an operating license are documented in the facility final safety analysis report (FSAR). Fundamental assumptions that are design inputs, including the source term, are to be included in the FSAR and become part of the facility design basis.[2]

Since the publication of TID-14844, significant advances have been made in understanding the timing, magnitude, and chemical form of fission product releases from severe nuclear power plant accidents. A holder of an operating license issued prior to January 10, 1997, or a holder of a renewed license under 10 CFR Part 54 whose initial operating license was issued prior to January

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

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

10, 1997, is allowed by 10 CFR 50.67, "Accident Source Term," to voluntarily revise the accident source term used in design basis radiological consequence analyses.

In general, information provided by regulatory guides is reflected in NUREG-0800, the Standard Review Plan (SRP) (Ref 3). The NRC staff uses the SRP to review applications to construct and operate nuclear power plants. This regulatory guide applies to Chapter 15.0.1 of the SRP.

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

B. DISCUSSION

An accident source term is intended to be representative of a major accident involving significant core damage and is typically postulated to occur in conjunction with a large loss-of-coolant accident (LOCA). Although the LOCA is typically the maximum credible accident, NRC staff experience in reviewing license applications has indicated the need to consider other accident sequences of lesser consequence but higher probability of occurrence. The design basis accidents (DBAs) were not intended to be actual event sequences, but rather, were intended to be surrogates to enable deterministic evaluation of the response of a facility's engineered safety features. These accident analyses are intentionally conservative in order to compensate for known uncertainties in accident progression, fission product transport, and atmospheric dispersion. Although probabilistic risk assessments (PRAs) can provide useful insights into system performance and suggest changes in how the desired depth is achieved, defense in depth continues to be an effective way to account for uncertainties in equipment and human performance. The NRC's policy statement on the use of PRA methods (Ref. 4) calls for the use of PRA technology in all regulatory matters in a manner that complements the NRC's deterministic approach and supports the traditional defense-in-depth philosophy.

Since the publication of TID-14844 (Ref. 1), significant advances have been made in understanding the timing, magnitude, and chemical form of fission product releases from severe nuclear power plant accidents. In 1995, the NRC published NUREG-1465, "Accident Source Terms for Light-Water Nuclear Power Plants" (Ref. 5). NUREG-1465 used this research to provide estimates of the accident source term that were more physically based and that could be applied to the design of future light-water power reactors. NUREG-1465 presents a representative accident source term for a boiling-water reactor (BWR) and for a pressurized-water reactor (PWR). These source terms are characterized by the composition and magnitude of the radioactive material, the chemical and physical properties of the material, and the timing of the release to the containment. The NRC staff considered the applicability of the revised source terms to operating reactors and determined that the current analytical approach based on the TID-14844 source term would continue to be adequate to protect public health and safety. Operating reactors licensed under that approach would not be required to re-analyze accidents using the revised source terms. The NRC staff also determined that some licensees might wish to use an AST in analyses to support cost-beneficial licensing actions.

The NRC staff, therefore, initiated several actions to provide a regulatory basis for operating reactors to use an 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 Plant." (Ref. 11), regarding the radiological source term used in the determination of integrated doses for environmental qualification purposes is superseded by the corresponding guidance in this regulatory guide for those facilities that are proposing to, or have already, implemented an AST. All other guidance in Regulatory Guide 1.89 remains effective.

This guide primarily addresses design basis accidents, such as those addressed in Chapter 15 of typical final safety analysis reports (FSARs). This guide does not address all areas of potentially significant risk. Although this guide addresses fuel handling accidents, other events that could occur during shutdown operations are not currently addressed. The NRC staff has several ongoing

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

initiatives involving risks of shutdown operations, extended burnup fuels, and risk-informing current regulations. The information in this guide may be revised in the future as NRC staff evaluations are completed and regulatory decisions on these issues are made.

C. REGULATORY POSITION

1. IMPLEMENTATION OF AST

1.1 Generic Considerations

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

1.1.1 Safety Margins

The proposed uses of an AST and the associated proposed facility modifications and changes to procedures should be evaluated to determine whether the proposed changes are consistent with the principle that sufficient safety margins are maintained, including a margin to account for analysis uncertainties. The safety margins are products of specific values and limits contained in the technical specifications (which cannot be changed without NRC approval) and other values, such as assumed accident or transient initial conditions or assumed safety system response times. Changes, or the net effects of multiple changes, that result in a reduction in safety margins may require prior NRC approval. Once the initial AST implementation has been approved by the staff and has become part of the facility design basis, the licensee may use 10 CFR 50.59 and its supporting guidance in assessing safety margins related to subsequent facility modifications and changes to procedures.

1.1.2 Defense in Depth

The proposed uses of an AST and the associated proposed facility modifications and changes to procedures should be evaluated to determine whether the proposed changes are consistent with the principle that adequate defense in depth is maintained to compensate for uncertainties in accident progression and analysis data. Consistency with the defense-in-depth philosophy is maintained if system redundancy, independence, and diversity are preserved commensurate with the expected frequency, consequences of challenges to the system, and uncertainties. In all cases, compliance with the General Design Criteria in Appendix A to 10 CFR Part 50 is essential. Modifications proposed for the facility generally should not create a need for compensatory programmatic activities, such as reliance on manual operator actions.

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

1.1.3 Integrity of Facility Design Basis

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

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

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

1.1.4 Emergency Preparedness Applications

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

Although the AST provided in this guide was based on a limited spectrum of severe accidents, the particular characteristics have been tailored specifically for DBA analysis use. The AST is not

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

representative of the wide spectrum of possible events that make up the planning basis of emergency preparedness. Therefore, the AST is insufficient *by itself* as a basis for requesting relief from the emergency preparedness requirements of 10 CFR 50.47 and Appendix E to 10 CFR Part 50.

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

1.2 Scope of Implementation

The AST described in this guide is characterized by radionuclide composition and magnitude, chemical and physical form of the radionuclides, and the timing of the release of these radionuclides. The accident source term is a fundamental assumption upon which a large portion of the facility design is based. Additionally, many aspects of facility operation derive from the design analyses that incorporated the earlier accident source term. A complete implementation of an AST would upgrade all existing radiological analyses and would consider the impact of all five characteristics of the AST as defined in 10 CFR 50.2. However, the NRC staff has determined that there could be implementations for which this level of re-analysis may not be necessary. Two categories are defined: Full and selective implementations.

1.2.1 Full Implementation

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

1.2.2 Selective Implementation

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

1.3 Scope of Required Analyses

1.3.1 Design Basis Radiological Analyses

There are several regulatory requirements for which compliance is demonstrated, in part, by the evaluation of the radiological consequences of design basis accidents. These requirements include, but are not limited to, the following.

- Environmental Qualification of Equipment (10 CFR 50.49)
- Control Room Habitability (GDC-19 of Appendix A to 10 CFR Part 50)
- Emergency Response Facility Habitability (Paragraph IV.E.8 of Appendix E to 10 CFR Part 50)
- Alternative Source Term (10 CFR 50.67)
- Environmental Reports (10 CFR Part 51)
- Facility Siting (10 CFR 100.11)[5]

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

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

1.3.2 Re-Analysis Guidance

Any implementation of an AST, full or selective, and any associated facility modification should be supported by evaluations of all significant radiological and nonradiological impacts of the proposed actions. This evaluation should consider the impact of the proposed changes on the facility's compliance with the regulations and commitments listed above as well as any other facility-specific requirements. These impacts may be due to (1) the associated facility modifications or (2) the differences in the AST characteristics. The scope and extent of the re-

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

evaluation will necessarily be a function of the specific proposed facility 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 be updated and all selected characteristics of the AST and the TEDE criteria should be addressed. The license amendment request should describe the licensee's re-analysis effort and provide statements regarding the acceptability of the proposed implementation, including modifications, against each of the applicable analysis requirements and commitments identified in Regulatory Position 1.3.1 of this guide.

The NRC staff has performed an evaluation of the impact of the AST on three representative operating reactors (Ref. 14). This evaluation determined that radiological analysis results based on the TID-14844 source term assumptions (Ref. 1) and the whole body and thyroid methodology generally bound the results from analyses based on the AST and TEDE methodology. Licensees may use the applicable conclusions of this evaluation in addressing the impact of the AST on design basis radiological analyses. However, this does not exempt the licensee from evaluating the remaining radiological and nonradiological impacts of the AST implementation and the impacts of the associated plant modifications. For example, a selective implementation based on the timing insights of the AST may change the required isolation time for the containment purge dampers from 2.5 seconds to 5.0 seconds. This application might be acceptable without dose calculations. However, evaluations may need to be performed regarding the ability of the damper to close against increased containment pressure or the ability of ductwork downstream of the dampers to withstand increased stresses.

For full implementation, a complete DBA LOCA analysis as described in Appendix A of this guide should be performed, as a minimum. Other design basis analyses are updated in accordance with the guidance in this section.

A selective implementation of an AST and any associated facility modification based on the AST should evaluate all the radiological and nonradiological impacts of the proposed actions as they apply to the particular implementation. Design basis analyses are updated in accordance with the guidance in this section. There is no minimum requirement that a DBA LOCA analysis be performed. The analyses performed need to address all impacts of the proposed modification, the selected characteristics of the AST, and if dose calculations are performed, the TEDE criteria. For selective implementations based on the timing characteristic of the AST, e.g., change in the closure timing of a containment isolation valve, re-analysis of radiological calculations may not be

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

necessary if the modified elapsed time remains a fraction (e.g., 25%) of the time between accident initiation and the onset of the gap release phase. Longer time delays may be considered on an individual basis. For longer time delays, evaluation of the radiological consequences and other impacts of the delay, such as blockage by debris in sump water, may be necessary. If affected design basis analyses are to be re-calculated, all affected assumptions and inputs should be updated and all selected characteristics of the AST and the TEDE criteria should be addressed.

1.3.3 Use of Sensitivity or Scoping Analyses

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

1.3.4 Updating Analyses Following Implementation

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

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

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

1.3.5 Equipment Environmental Qualification

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

1.4 Risk Implications

The use of an AST changes only the regulatory assumptions regarding the analytical treatment of the design basis accidents. The AST has no direct effect on the probability of the accident. Use of an AST alone cannot increase the core damage frequency (CDF) or the large early release frequency (LERF). However, facility modifications made possible by the AST could have an impact on risk. If the proposed implementation of the AST involves changes to the facility design that would invalidate assumptions made in the facility's PRA, the impact on the existing PRAs should be evaluated.

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

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

1.5 Submittal Requirements

According to 10 CFR 50.90, an application for an amendment must fully describe the changes desired and should follow, as far as applicable, the form prescribed for original applications. Regulatory Guide 1.70, "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants (LWR Edition)" (Ref 16), provides additional guidance. The NRC staff's finding that the amendment may be approved must be based on the licensee's analyses, since it is these analyses that will become part of the design basis of the facility. The amendment request should describe the licensee's analyses of the radiological and nonradiological impacts of the proposed modification in sufficient detail to support review by the NRC staff. The staff recommends that licensees submit affected FSAR pages annotated with changes that reflect the revised analyses or submit the actual calculation documentation.

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

1.6 FSAR Requirements

Requirements for updating the facility's final safety analysis report (FSAR) are in 10 CFR 50.71, "Maintenance of Records, Making of Reports." The regulations in 10 CFR 50.71(e) require that the FSAR be updated to include all changes made in the facility or procedures described in the FSAR and all safety evaluations performed by the licensee in support of requests for license amendments or in support of conclusions that changes did not involve unreviewed safety questions. The analyses required by 10 CFR 50.67 are subject to this requirement. The affected radiological analysis descriptions in the FSAR should be updated to reflect the replacement of the design basis source term by the AST. The analysis descriptions should contain sufficient detail to identify the methodologies used, significant assumptions and inputs, and numeric results. Regulatory Guide 1.70 (Ref. 16) provides additional guidance. The descriptions of superseded analyses should be removed from the FSAR in the interest of maintaining a clear design basis.

2. ATTRIBUTES OF AN ACCEPTABLE AST

An acceptable AST is not set forth in 10 CFR 50.67. Regulatory Position 3 of this guide identifies an AST that is acceptable to the NRC staff for use at operating power reactors. A substantial effort was expended by the NRC, its contractors, various national laboratories, peer reviewers, and others in performing severe accident research and in developing the source terms provided in NUREG-1465 (Ref. 5). However, future research may identify opportunities for changes in these source terms. The NRC staff will consider applications for an AST different from that identified in this guide. However, the NRC staff does not expect to approve any source term that is not of the same level of quality as the source terms in NUREG-1465. To be considered acceptable, an AST must have the following attributes:

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

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

2.3 The AST must not be based upon a single accident scenario but instead must represent a spectrum of credible severe accident events. Risk insights may be used, not to select a single risk-significant accident, but rather to establish the range of events to be considered. Relevant insights from applicable severe accident research on the phenomenology of fission product release and transport behavior may be considered.

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

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

3. ACCIDENT SOURCE TERM

This section provides an AST that is acceptable to the NRC staff. The data in Regulatory Positions 3.2 through 3.5 are fundamental to the definition of an AST. Once approved, the AST assumptions or parameters specified in these positions become part of the facility's design basis. Deviations from this guidance must be evaluated against Regulatory Position 2. After the NRC staff has approved an implementation of an AST, subsequent changes to the AST will require NRC staff review under 10 CFR 50.67.

3.1 Fission Product Inventory

The inventory of fission products in the reactor core and available for release to the containment should be based on the maximum full power operation of the core with, as a minimum, current licensed values for fuel enrichment, fuel burnup, and an assumed core power equal to the current licensed rated thermal power times the ECCS evaluation uncertainty.[8] The period of irradiation should be of sufficient duration to allow the activity of dose-significant radionuclides to reach equilibrium or to reach maximum values.[9] The core inventory should be determined using an appropriate isotope generation and depletion computer code such as ORIGEN 2 (Ref. 17) or ORIGEN-ARP (Ref. 18). Core inventory factors (Ci/MWt) provided in TID14844 and used in some analysis computer codes were derived for low burnup, low enrichment fuel and should not be used with higher burnup and higher enrichment fuels.

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

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

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

No adjustment to the fission product inventory should be made for events postulated to occur during power operations at less than full rated power or those postulated to occur at the beginning of core life. For events postulated to occur while the facility is shutdown, e.g., a fuel handling accident, radioactive decay from the time of shutdown may be modeled.

3.2 Release Fractions[10]

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

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

Table 1
BWR Core Inventory Fraction
Released Into Containment

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

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

Table 2
PWR Core Inventory Fraction Released Into Containment

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

Table 3[11]
Non-LOCA Fraction of Fission Product Inventory in Gap

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

3.3 Timing of Release Phases

Table 4 tabulates the onset and duration of each sequential release phase for DBA LOCAs at PWRs and BWRs. The specified onset is the time following the initiation of the accident (i.e., time = 0). The early in-vessel phase immediately follows the gap release phase. The activity released from the core during each release phase should be modeled as increasing in a linear fashion over the duration of the phase.[12] For non-LOCA DBAs in which fuel damage is projected, the release from the fuel gap and the fuel pellet should be assumed to occur instantaneously with the onset of the projected damage.

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

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

Table 4
LOCA Release Phases

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

For facilities licensed with leak-before-break methodology, the onset of the gap release phase may be assumed to be 10 minutes. A licensee may propose an alternative time for the onset of the gap release phase, based on facility-specific calculations using suitable analysis codes or on an accepted topical report shown to be applicable to the specific facility. In the absence of approved alternatives, the gap release phase onsets in Table 4 should be used.

3.4 Radionuclide Composition

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

Table 5
Radionuclide Groups

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

3.5 Chemical Form

Of the radioiodine released from the reactor coolant system (RCS) to the containment in a postulated accident, 95 percent of the iodine released should be assumed to be cesium iodide (CsI), 4.85 percent elemental iodine, and 0.15 percent organic iodide. This includes releases from the gap and the fuel pellets. With the exception of elemental and organic iodine and noble gases, fission products should be assumed to be in particulate form. The same chemical form is assumed in releases from fuel pins in FHAs and from releases from the fuel pins through the RCS in DBAs other than FHAs or LOCAs. However, the transport of these iodine species following release from the fuel may affect these assumed fractions. The accident-specific appendices to this regulatory guide provide additional details.

3.6 Fuel Damage in Non-LOCA DBAs

The amount of fuel damage caused by non-LOCA design basis events should be analyzed to determine, for the case resulting in the highest radioactivity release, the fraction of the fuel that reaches or exceeds the initiation temperature of fuel melt and the fraction of fuel elements for which the fuel clad is breached. Although the NRC staff has traditionally relied upon the departure from nucleate boiling ratio (DNBR) as a fuel damage criterion, licensees may propose other methods to the NRC staff, such as those based upon enthalpy deposition, for estimating fuel damage for the purpose of establishing radioactivity releases.

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

4. DOSE CALCULATIONAL METHODOLOGY

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

4.1 Offsite Dose Consequences

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

4.1.1 The dose calculations should determine the TEDE. TEDE is the sum of the committed effective dose equivalent (CEDE) from inhalation and the deep dose equivalent (DDE) from external exposure. The calculation of these two components of the TEDE should consider all radionuclides, including progeny from the decay of parent radionuclides, that are significant with regard to dose consequences and the released radioactivity.[13]

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

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

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

4.1.4 The DDE should be calculated assuming submergence in semi-infinite cloud assumptions with appropriate credit for attenuation by body tissue. The DDE is nominally equivalent to the effective dose equivalent (EDE) from external exposure if the whole body is irradiated uniformly. Since this is a reasonable assumption for submergence exposure situations, EDE may be used in lieu of DDE in determining the contribution of external dose to the TEDE. Table III.1 of Federal Guidance Report 12, "External Exposure to Radionuclides in Air, Water, and Soil" (Ref. 21), provides external EDE conversion factors acceptable to the NRC staff. The factors in the column headed "effective" yield doses corresponding to the EDE.

4.1.5 The TEDE should be determined for the most limiting person at the EAB. The maximum EAB TEDE for any two-hour period following the start of the radioactivity release should be determined and used in determining compliance with the dose criteria in 10 CFR 50.67.[14] The maximum two-hour TEDE should be determined by calculating the postulated dose for a series of small time increments and performing a "sliding" sum over the increments for successive two-hour periods. The maximum TEDE obtained is submitted. The time increments should appropriately reflect the progression of the accident to capture the peak dose interval between the start of the event and the end of radioactivity release (see also Table 6).

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

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

4.2 Control Room Dose Consequences

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

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

- Contamination of the control room atmosphere by the intake or infiltration of the radioactive material contained in the radioactive plume released from the facility,
- Contamination of the control room atmosphere by the intake or infiltration of airborne radioactive material from areas and structures adjacent to the control room envelope,
- Radiation shine from the external radioactive plume released from the facility,

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

- Radiation shine from radioactive material in the reactor containment,

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

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

4.2.3 The models used to transport radioactive material into and through the control room,[15] and the shielding models used to determine radiation dose rates from external sources, should be structured to provide suitably conservative estimates of the exposure to control room personnel.

4.2.4 Credit for engineered safety features that mitigate airborne radioactive material within the control room may be assumed. Such features may include control room isolation or pressurization, or intake or recirculation filtration. Refer to Section 6.5.1, "ESF Atmospheric Cleanup System," of the SRP (Ref. 3) and Regulatory Guide 1.52, "Design, Testing, and Maintenance Criteria for Postaccident Engineered-Safety-Feature Atmosphere Cleanup System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants" (Ref. 25), for guidance. The control room design is often optimized for the DBA LOCA and the protection afforded for other accident sequences may not be as advantageous. In most designs, control room isolation is actuated by engineered safeguards feature (ESF) signals or radiation monitors (RMs). In some cases, the ESF signal is effective only for selected accidents, placing reliance on the RMs for the remaining accidents. Several aspects of RMs can delay the control room isolation, including the delay for activity to build up to concentrations equivalent to the alarm setpoint and the effects of different radionuclide accident isotopic mixes on monitor response.

4.2.5 Credit should generally not be taken for the use of personal protective equipment or prophylactic drugs. Deviations may be considered on a case-by-case basis.

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

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

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

4.2.7 Control room doses should be calculated using dose conversion factors identified in Regulatory Position 4.1 above for use in offsite dose analyses. The DDE from photons may be corrected for the difference between finite cloud geometry in the control room and the semi-infinite cloud assumption used in calculating the dose conversion factors. The following expression may be used to correct the semi-infinite cloud dose, DDE_{∞}, to a finite cloud dose, DDE_{finite}, where the control room is modeled as a hemisphere that has a volume, V, in cubic feet, equivalent to that of the control room (Ref. 22).

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

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 (Ref. 2). Design envelope source terms provided in NUREG-0737 should be updated for consistency with the AST. In general, radiation exposures to plant personnel identified in Regulatory Position 1.3.1 should be expressed in terms of TEDE. Integrated radiation exposure of plant equipment should be determined using the guidance of Appendix I of this guide.

4.4 Acceptance Criteria

The radiological criteria for the EAB, the outer boundary of the LPZ, and for the control room are in 10 CFR 50.67. These criteria are stated for evaluating reactor accidents of exceedingly low probability of occurrence and low risk of public exposure to radiation, e.g., a large-break LOCA. The control room criterion applies to all accidents. For events with a higher probability of occurrence, postulated EAB and LPZ doses should not exceed the criteria tabulated in Table 6.

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

Table 6[17]
Accident Dose Criteria

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

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

5. ANALYSIS ASSUMPTIONS AND METHODOLOGY

5.1 General Considerations

5.1.1 Analysis Quality

The evaluations required by 10 CFR 50.67 are re-analyses of the design basis safety analyses and evaluations required by 10 CFR 50.34; they are considered to be a significant input to the evaluations required by 10 CFR 50.92 or 10 CFR 50.59. These analyses should be prepared, reviewed, and maintained in accordance with quality assurance programs that comply with Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," to 10 CFR Part 50.

These design basis analyses were structured to provide a conservative set of assumptions to test the performance of one or more aspects of the facility design. Many physical processes and phenomena are represented by conservative, bounding assumptions rather than being modeled

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

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

5.1.2 Credit for Engineered Safeguard Features

Credit may be taken for accident mitigation features that are classified as safety-related, are required to be operable by technical specifications, are powered by emergency power sources, and are either automatically actuated or, in limited cases, have actuation requirements explicitly addressed in emergency operating procedures. The single active component failure that results in the most limiting radiological consequences should be assumed. Assumptions regarding the occurrence and timing of a loss of offsite power should be selected with the objective of maximizing the postulated radiological consequences.

5.1.3 Assignment of Numeric Input Values

The numeric values that are chosen as inputs to the analyses required by 10 CFR 50.67 should be selected with the objective of determining a conservative postulated dose. In some instances, a particular parameter may be conservative in one portion of an analysis but be nonconservative in another portion of the same analysis. For example, assuming minimum containment system spray flow is usually conservative for estimating iodine scrubbing, but in many cases may be nonconservative when determining sump pH. Sensitivity analyses may be needed to determine the appropriate value to use. As a conservative alternative, the limiting value applicable to each portion of the analysis may be used in the evaluation of that portion. A single value may not be applicable for a parameter for the duration of the event, particularly for parameters affected by changes in density. For parameters addressed by technical specifications, the value used in the analysis should be that specified in the technical specifications.[18] If a range of values or a tolerance band is specified, the value that would result in a conservative postulated dose should be used. If the parameter is based on the results of less frequent surveillance testing, e.g., steam generator nondestructive testing (NDT), consideration should be given to the degradation that may occur between periodic tests in establishing the analysis value.

5.1.4 Applicability of Prior Licensing Basis

The NRC staff considers the implementation of an AST to be a significant change to the design basis of the facility that is voluntarily initiated by the licensee. In order to issue a license amendment authorizing the use of an AST and the TEDE dose criteria, the NRC staff must make a current finding of compliance with regulations applicable to the amendment. The characteristics of the ASTs and the revised dose calculational methodology may be incompatible with many of the analysis assumptions and methods currently reflected in the facility's design basis analyses. The NRC staff may find that new or unreviewed issues are created by a particular site-specific

[18] Note that for some parameters, the technical specification value may be adjusted for analysis purposes by factors provided in other regulatory guidance. For example, ESF filter efficiencies are based on the guidance in Regulatory Guide 1.52 (Ref. 25) and in Generic Letter 99-02 (Ref. 27) rather than the surveillance test criteria in the technical specifications. Generally, these adjustments address potential changes in the parameter between scheduled surveillance tests.

implementation of the AST, warranting review of staff positions approved subsequent to the initial issuance of the license. This is not considered a backfit as defined by 10 CFR 50.109, “Backfitting.” However, prior design bases that are unrelated to the use of the AST, or are unaffected by the AST, may continue as the facility’s design basis. Licensees should ensure that analysis assumptions and methods are compatible with the ASTs and the TEDE criteria.

5.2 Accident-Specific Assumptions

The appendices to this regulatory guide provide accident-specific assumptions that are acceptable to the staff for performing analyses that are required by 10 CFR 50.67. The DBAs addressed in these attachments were selected from accidents that may involve damage to irradiated fuel. This guide does not address DBAs with radiological consequences based on technical specification reactor or secondary coolant-specific activities only. The inclusion or exclusion of a particular DBA in this guide should not be interpreted as indicating that an analysis of that DBA is required or not required. Licensees should analyze the DBAs that are affected by the specific proposed applications of an AST.

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

The NRC is committed to using probabilistic risk analysis (PRA) insights in its regulatory activities and will consider licensee proposals for changes in analysis assumptions based upon risk insights. The staff will not approve proposals that would reduce the defense in depth deemed necessary to provide adequate protection for public health and safety. In some cases, this defense in depth compensates for uncertainties in the PRA analyses and addresses accident considerations not adequately addressed by the core damage frequency (CDF) and large early release frequency (LERF) surrogate indicators of overall risk.

5.3 Meteorology Assumptions

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

References 22 and 28 should be used if the FSAR χ/Q values are to be revised or if values are to be determined for new release points or receptor distances. Fumigation should be considered where applicable for the EAB and LPZ. For the EAB, the assumed fumigation period should be timed to be included in the worst 2-hour exposure period. The NRC computer code PAVAN (Ref. 29) implements Regulatory Guide 1.145 (Ref. 28) and its use is acceptable to the NRC staff. The methodology of the NRC computer code ARCON96[19] (Ref. 26) is generally acceptable to the NRC staff for use in determining control room χ/Q values. Meteorological data collected in accordance with the site-specific meteorological measurements program described in the facility FSAR should be used in generating accident χ/ Q values. Additional guidance is provided in Regulatory Guide 1.23, "Onsite Meteorological Programs" (Ref. 30). All changes in χ/Q analysis methodology should be reviewed by the NRC staff.

6. ASSUMPTIONS FOR EVALUATING THE RADIATION DOSES FOR EQUIPMENT QUALIFICATION

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

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

D. IMPLEMENTATION

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

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

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

REFERENCES

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

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Appendix A

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

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

Appendix A, "General Design Criteria for Nuclear Power Plants," to 10 CFR Part 50 defines LOCAs as those postulated accidents that result from a loss of coolant inventory at rates that exceed the capability of the reactor coolant makeup system. Leaks up to a double-ended rupture of the largest pipe of the reactor coolant system are included. The LOCA, as with all design basis accidents (DBAs), is a conservative surrogate accident that is intended to challenge selective aspects of the facility design. Analyses are performed using a spectrum of break sizes to evaluate fuel and ECCS performance. With regard to radiological consequences, a large-break LOCA is assumed as the design basis case for evaluating the performance of release mitigation systems and the containment and for evaluating the proposed siting of a facility.

SOURCE TERM ASSUMPTIONS

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

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

ASSUMPTIONS ON TRANSPORT IN PRIMARY CONTAINMENT

3. Acceptable assumptions related to the transport, reduction, and release of radioactive material in and from the primary containment in PWRs or the drywell in BWRs are as follows:

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

3.2 Reduction in airborne radioactivity in the containment by natural deposition within the containment may be credited. Acceptable models for removal of iodine and aerosols are described in Chapter 6.5.2, "Containment Spray as a Fission Product Cleanup System," of the Standard Review Plan (SRP), NUREG-0800 (Ref. A-1) and in NUREG/CR-6189, "A Simplified Model of Aerosol Removal by Natural Processes in Reactor Containments" (Ref. A-2). The latter model is incorporated into the analysis code RADTRAD (Ref. A-3). The prior practice of deterministically assuming that a 50% plateout of iodine is released from the fuel is no longer acceptable to the NRC staff as it is inconsistent with the characteristics of the revised source terms.

3.3 Reduction in airborne radioactivity in the containment by containment spray systems that have been designed and are maintained in accordance with Chapter 6.5.2 of the SRP (Ref. A-1) may be credited. Acceptable models for the removal of iodine and aerosols are described in Chapter 6.5.2 of the SRP and NUREG/CR-5966, "A Simplified Model of Aerosol Removal by Containment Sprays"[1] (Ref. A-4). This simplified model is incorporated into the analysis code RADTRAD (Refs. A-1 to A-3).

The evaluation of the containment sprays should address areas within the primary containment that are not covered by the spray drops. The mixing rate attributed to natural convection between sprayed and unsprayed regions of the containment building, provided that adequate flow exists between these regions, is assumed to be two turnovers of the unsprayed regions per hour, unless other rates are justified. The containment building atmosphere may be considered a single, well-mixed volume if the spray covers at least 90% of the volume and if adequate mixing of unsprayed compartments can be shown.

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

3.4 Reduction in airborne radioactivity in the containment by in-containment recirculation filter systems may be credited if these systems meet the guidance of Regulatory Guide 1.52 and Generic Letter 99-02 (Refs. A-5 and A-6). The filter media loading caused by the increased aerosol release associated with the revised source term should be addressed.

3.5 Reduction in airborne radioactivity in the containment by suppression pool scrubbing in BWRs should generally not be credited. However, the staff may consider such reduction on an individual case basis. The evaluation should consider the relative timing of the blowdown and the fission product release from the fuel, the force driving the release through the pool,

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

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

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

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

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

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

ASSUMPTIONS ON DUAL CONTAINMENTS

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

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

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

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

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

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

4.6 Reduction in the amount of radioactive material released from the secondary containment because of ESF filter systems may be taken into account provided that these systems meet the guidance of Regulatory Guide 1.52 (Ref. A-5) and Generic Letter 99-02 (Ref. A-6).

ASSUMPTIONS ON ESF SYSTEM LEAKAGE

5. ESF systems that recirculate sump water outside of the primary containment are assumed to leak during their intended operation. This release source includes leakage through valve packing glands, pump shaft seals, flanged connections, and other similar components. This release source may also include leakage through valves isolating interfacing systems (Ref. A-7). The radiological consequences from the postulated leakage should be analyzed and combined with consequences postulated for other fission product release paths to determine the total calculated radiological consequences from the LOCA. The following assumptions are acceptable for evaluating the consequences of leakage from ESF components outside the primary containment for BWRs and PWRs.

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

5.2 The leakage should be taken as two times the sum of the simultaneous leakage from all components in the ESF recirculation systems above which the technical specifications, or licensee commitments to item III.D.1.1 of NUREG-0737 (Ref. A-8), would require declaring such systems inoperable. The leakage should be assumed to start at the earliest time the recirculation flow occurs in these systems and end at the latest time the releases from these systems are terminated. Consideration should also be given to design leakage through valves isolating ESF recirculation systems from tanks vented to atmosphere, e.g., emergency core cooling system (ECCS) pump miniflow return to the refueling water storage tank.

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

5.4 If the temperature of the leakage exceeds 212°F, the fraction of total iodine in the liquid that becomes airborne should be assumed equal to the fraction of the leakage that flashes to vapor. This flash fraction, FF, should be determined using a constant enthalpy, h, process, based on the maximum time-dependent temperature of the sump water circulating outside the containment:

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

Where: h_{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.

5.5 If the temperature of the leakage is less than 212°F or the calculated flash fraction is less than 10%, the amount of iodine that becomes airborne should be assumed to be 10% of the total iodine activity in the leaked fluid, unless a smaller amount can be justified based on the actual sump pH history and area ventilation rates.

5.6 The radioiodine that is postulated to be available for release to the environment is assumed to be 97% elemental and 3% organic. Reduction in release activity by dilution or holdup within buildings, or by ESF ventilation filtration systems, may be credited where applicable. Filter systems used in these applications should be evaluated against the guidance of Regulatory Guide 1.52 (Ref. A-5) and Generic Letter 99-02 (Ref. A-6).

ASSUMPTIONS ON MAIN STEAM ISOLATION VALVE LEAKAGE IN BWRS

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

6.1 For the purpose of this analysis, the activity available for release via MSIV leakage should be assumed to be that activity determined to be in the drywell for evaluating containment leakage (see Regulatory Position 3). No credit should be assumed for activity reduction by the steam separators or by iodine partitioning in the reactor vessel.

6.2 All the MSIVs should be assumed to leak at the maximum leak rate above which the technical specifications would require declaring the MSIVs inoperable. The leakage should be assumed to continue for the duration of the accident. Postulated leakage may be reduced after the first 24 hours, if supported by site-specific analyses, to a value not less than 50% of the maximum leak rate.

6.3 Reduction of the amount of released radioactivity by deposition and plateout on steam system piping upstream of the outboard MSIVs may be credited, but the amount of reduction in concentration allowed will be evaluated on an individual case basis. Generally, the model should be based on the assumption of well-mixed volumes, but other models such as slug flow may be used if justified.

6.4 In the absence of collection and treatment of releases by ESFs such as the MSIV leakage control system, or as described in paragraph 6.5 below, the MSIV leakage should be assumed to be released to the environment as an unprocessed, ground- level release. Holdup and dilution in the turbine building should not be assumed.

6.5 A reduction in MSIV releases that is due to holdup and deposition in main steam piping downstream of the MSIVs and in the main condenser, including the treatment of air ejector effluent by offgas systems, may be credited if the components and piping systems used in the release path are capable of performing their safety function during and following a safe shutdown earthquake (SSE). The amount of reduction allowed will be evaluated on an individual case basis. References A-9 and A-10 provide guidance on acceptable models.

ASSUMPTION ON CONTAINMENT PURGING

7. The radiological consequences from post-LOCA primary containment purging as a combustible gas or pressure control measure should be analyzed. If the installed containment purging capabilities are maintained for purposes of severe accident management and are not credited in any design basis analysis, radiological consequences need not be evaluated. If the primary containment purging is required within 30 days of the LOCA, the results of this analysis should be combined with consequences postulated for other fission product release paths to determine the total calculated radiological consequences from the LOCA. Reduction in the amount of radioactive material released via ESF filter systems may be taken into account provided that these systems meet the guidance in Regulatory Guide 1.52 (Ref. A-5) and Generic Letter 99-02 (Ref. A-6).

Appendix A REFERENCES

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A-8 USNRC, "Clarification of TMI Action Plan Requirements," NUREG-0737, November 1980.

A-9 J.E. Cline, "MSIV Leakage Iodine Transport Analysis," Letter Report dated March 26, 1991. (ADAMS Accession Number ML003683718)

A-10 USNRC, "Safety Evaluation of GE Topical Report, NEDC-31858P (Proprietary GE report), Revision 2, *BWROG Report for Increasing MSIV Leakage Limits and Elimination of Leakage Control Systems, September 1993*," letter dated March 3, 1999, ADAMS Accession Number 9903110303.

Appendix B

ASSUMPTIONS FOR EVALUATING THE RADIOLOGICAL CONSEQUENCES OF A FUEL HANDLING ACCIDENT

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

1. SOURCE TERM

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

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

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

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

2. WATER DEPTH

If the depth of water above the damaged fuel is 23 feet or greater, the 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 (99.85%) and organic iodine (0.15%) species results in the iodine above the water being composed of 57% elemental and 43% organic species. If the depth of water is not 23 feet, the decontamination factor will have to be determined on a case-by-case method (Ref. B-1).

3. NOBLE GASES

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

4. FUEL HANDLING ACCIDENTS WITHIN THE FUEL BUILDING

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

4.1 The radioactive material that escapes from the fuel pool to the fuel building is assumed to be released to the environment over a 2-hour time period.

4.2 A reduction in the amount of radioactive material released from the fuel pool by engineered safety feature (ESF) filter systems may be taken into account provided these systems meet the guidance of Regulatory Guide 1.52 and Generic Letter 99-02 (Refs. B-2, B-3). Delays in radiation detection, actuation of the ESF filtration system, or diversion of ventilation flow to the ESF filtration 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. If mixing can be demonstrated, credit for mixing and dilution may be considered on a case-by-case basis. This evaluation should consider the magnitude of the building volume and exhaust rate, the potential for bypass to the environment, the location of exhaust plenums relative to the surface of the pool, recirculation ventilation systems, and internal walls and floors that impede stream flow between the surface of the pool and the exhaust plenums.

5. FUEL HANDLING ACCIDENTS WITHIN CONTAINMENT

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

5.1 If the containment is 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

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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.

delays in radiation detection and completion of containment isolation. If it can be shown that containment isolation occurs before radioactivity is released to the environment,[1] no radiological consequences need to be analyzed.

5.3 If the containment is open during fuel handling operations (e.g., personnel air lock or equipment hatch is open),[3] the radioactive material that escapes from the reactor cavity pool to the containment is released to the environment over a 2-hour time period.

5.4 A reduction in the amount of radioactive material released from the containment by ESF filter systems may be taken into account provided that these systems meet the guidance of Regulatory Guide 1.52 and Generic Letter 99-02 (Refs. B-2 and B-3). Delays in radiation detection, actuation of the ESF filtration system, or diversion of ventilation flow to the ESF filtration system should be determined and accounted for in the radioactivity release analyses.[1]

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

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[3] The staff will generally require that technical specifications allowing such operations include administrative controls to close the airlock, hatch, or open penetrations within 30 minutes. Such adminstrative controls will generally require that a dedicated individual be present, with necessary equipment available, to restore containment closure should a fuel handling accident occur. Radiological analyses should generally not credit this manual isolation.

Appendix B REFERENCES

B-1. G. Burley, “Evaluation of Fission Product Release and Transport,” Staff Technical Paper, 1971. (NRC Accession number 8402080322 in ADAMS or PARS)

B-2. USNRC, “Design, Testing, and Maintenance Criteria for Postaccident Engineered-Safety-Feature Atmosphere Cleanup System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants,” Regulatory Guide 1.52, Revision 2, March 1978.

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

Appendix C

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

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

1. Assumptions acceptable to the NRC staff regarding core inventory are provided in Regulatory Position 3 of this guide. For the rod drop accident, the release from the breached fuel is based on the estimate of the number of fuel rods breached and the assumption that 10% of the core inventory of the noble gases and iodines is in the fuel gap. The release attributed to fuel melting is based on the fraction of the fuel that reaches or exceeds the initiation temperature for fuel melting and on the assumption that 100% of the noble gases and 50% of the iodines contained in that fraction are released to the reactor coolant.

2. If no or minimal[1] fuel damage is postulated for the limiting event, the released activity should be the maximum coolant activity (typically 4 μCi/gm DE I-131) allowed by the technical specifications.

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

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

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

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

3.4 Of the activity that reaches the turbine and condenser, 100% of the noble gases, 10% of the iodine, and 1% of the particulate radionuclides are available for release to the environment. The turbine and condensers leak to the atmosphere as a ground- level release at a rate of 1% per 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

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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 the dose equivalent I-131 (DE I-131), only the radioiodine associated with normal operations or iodine spikes should be included. Activity from projected fuel damage should not be included.

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

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

3.5 In lieu of the transport assumptions provided in paragraphs 3.2 through 3.4 above, a more mechanistic analysis may be used on a case-by-case basis. Such analyses account for the quantity of contaminated steam carried from the pressure vessel to the turbine and condensers based on a review of the minimum transport time from the pressure vessel to the first main steam isolation (MSIV) and considers MSIV closure time.

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

Appendix D

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

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

SOURCE TERM

1. Assumptions acceptable to the NRC staff regarding core inventory and the release of radionuclides from the fuel are provided in Regulatory Position 3 of this guide. The release from the breached fuel is based on Regulatory Position 3.2 of this guide and the estimate of the number of fuel rods breached.

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

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

4.1 The main steam line isolation valves (MSIV) should be assumed to close in the maximum time allowed by technical specifications.

4.2 The total mass of coolant released should be assumed to be that amount in the steam line and connecting lines at the time of the break plus the amount that passes through the valves prior to closure.

[1] The activity assumed in the analysis should be based on the activity associated with the projected fuel damage or the maximum technical specification values, whichever maximizes the radiological consequences. In determining dose equivalent I-131 (DE I-131), only the radioiodine associated with normal operations or iodine spikes should be included. Activity from projected fuel damage should not be included.

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

4.4 The iodine species released from the main steam line should be assumed to be 95% CsI as an aerosol, 4.85% elemental, and 0.15% organic.

Appendix E

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

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

SOURCE TERMS

1. Assumptions acceptable to the NRC staff regarding core inventory and the release of radionuclides from the fuel are provided in Regulatory Position 3 of this regulatory guide. The release from the breached fuel is based on Regulatory Position 3.2 of this guide and the estimate of the number of fuel rods breached. The fuel damage estimate should assume that the highest worth control rod is stuck at its fully withdrawn position.

2. If no or 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. Shorter spike durations may be considered on a case-by-case basis if it can be shown that the activity released by the 8-hour spike exceeds that available for release from the fuel gap of all fuel pins.

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

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

TRANSPORT [3]

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

5.1 For facilities that have not implemented alternative repair criteria (see Ref. E-1, DG-1074), the primary-to-secondary leak rate in the steam generators should be assumed to be the leak rate limiting condition for operation specified in the technical specifications. For facilities with traditional generator specifications (both per generator and total of all generators), the leakage should be apportioned between affected and unaffected steam generators in such a manner that the calculated dose is maximized.

5.2 The density used in converting volumetric leak rates (e.g., gpm) to mass leak rates (e.g., lbm/hr) should be consistent with the basis of the parameter being converted. The ARC leak rate correlations are generally based on the collection of cooled liquid. Surveillance tests and facility instrumentation used to show compliance with leak rate technical specifications are typically based on cooled liquid. In most cases, the density should be assumed to be 1.0 gm/cc (62.4 lbm/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 The transport model described in this section should be utilized for iodine and particulate releases from the steam generators. This model is shown in Figure E-1 and summarized below:

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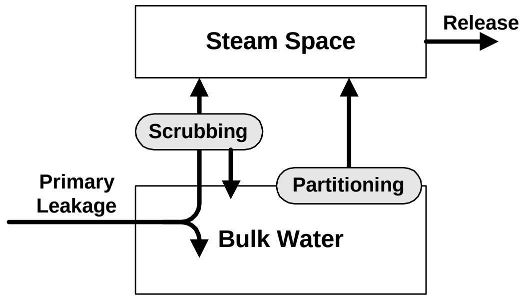
[3] In this appendix, *ruptured* refers to the state of the steam generator in which primary-to-secondary leakage rate has increased to a value greater than technical specifications. *Faulted* refers to the state of the steam generator in which the secondary side has been depressurized by a MSLB such that protective system response (main steam line isolation, reactor trip, safety injection, etc.) has occurred. *Partitioning Coefficient* is defined as:

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

Figure E-1
Transport Model



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

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

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

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

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

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

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

Appendix E REFERENCES

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

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

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

Appendix F

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

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

SOURCE TERM

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

2. If no or 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. Shorter spike durations may be considered on a case-by-case basis if it can be shown that the activity released by the 8-hour spike exceeds that available for release from the fuel gap of all fuel pins.

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

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

[2] The activity assumed in the analysis should be based on the activity associated with the projected fuel damage or the maximum technical specification values, whichever maximizes the radiological consequences. In determining dose equivalent I-131 (DE I-131), only the radioiodine associated with normal operations or iodine spikes should be included. Activity from projected fuel damage should not be included.

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

TRANSPORT [3]

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

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

5.2 The density used in converting volumetric leak rates (e.g., gpm) to mass leak rates (e.g., lbm/hr) should be consistent with the basis of surveillance tests used to show compliance with leak rate technical specifications. These tests are typically based on cool liquid. Facility instrumentation used to determine leakage is typically located on lines containing cool liquids. In most cases, the density should be assumed to be 1.0 gm/cc (62.4 lbm/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 The transport model described in Regulatory Positions 5.5 and 5.6 of Appendix E should be utilized for iodine and particulates.

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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.

Appendix G

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

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

SOURCE TERM

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

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

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

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

RELEASE TRANSPORT

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

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

5.2 The density used in converting volumetric leak rates (e.g., gpm) to mass leak rates (e.g., lbm/hr) should be consistent with the basis of surveillance tests used to show compliance with leak rate technical specifications. These tests are typically based on cool liquid.

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

Facility instrumentation used to determine leakage is typically located on lines containing cool liquids. In most cases, the density should be assumed to be 1.0 gm/cc (62.4 lbm/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 should be assumed to continue until shutdown cooling is in operation and releases from the steam generators have been terminated.

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

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

5.6 The transport model described in assumptions 5.5 and 5.6 of Appendix E should be utilized for iodine and particulates.

Appendix H

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

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

SOURCE TERM

1. Assumptions acceptable to the NRC staff regarding core inventory are in Regulatory Position 3 of this guide. For the rod ejection accident, the release from the breached fuel is based on the estimate of the number of fuel rods breached and the assumption that 10% of the core inventory of the noble gases and iodines is in the fuel gap. The release attributed to fuel melting is based on the fraction of the fuel that reaches or exceeds the initiation temperature for fuel melting and the assumption that 100% of the noble gases and 25% of the iodines contained in that fraction are available for release from containment. For the secondary system release pathway, 100% of the noble gases and 50% of the iodines in that fraction are released to the reactor coolant.

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

3. Two release cases are to be considered. In the first, 100% of the activity released from the fuel should be assumed to be released instantaneously and homogeneously through the containment atmosphere. In the second, 100% of the activity released from the fuel should be assumed to be completely dissolved in the primary coolant and available for release to the secondary system.

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

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

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

TRANSPORT FROM CONTAINMENT

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

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

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

TRANSPORT FROM SECONDARY SYSTEM

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

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

7.2 The density used in converting volumetric leak rates (e.g., gpm) to mass leak rates (e.g., lbm/hr) should be consistent with the basis of surveillance tests used to show compliance with leak rate technical specifications. These tests typically are based on cooled liquid. The facility's instrumentation used to determine leakage typically is located on lines containing cool liquids. In most cases, the density should be assumed to be 1.0 gm/cc (62.4 lbm/ft^3).

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

7.4 The transport model described in assumptions 5.5 and 5.6 of Appendix E should be utilized for iodine and particulates.

Appendix I

ASSUMPTIONS FOR EVALUATING RADIATION DOSES FOR EQUIPMENT QUALIFICATION

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

BASIC ASSUMPTIONS

1. Gamma and beta doses and dose rates should be determined for three types of radioactive source distributions: (1) activity suspended in the containment atmosphere, (2) activity plated out on containment surfaces, and (3) activity mixed in the containment sump water. A given piece of equipment may receive a dose contribution from any or all of these sources. The amount of dose contributed by each of these sources is determined by the location of the equipment, the time-dependent and location-dependent distribution of the source, and the effects of shielding. For EQ components located outside of the containment, additional radiation sources may include piping and components in systems that circulate containment sump water outside of containment. Activity deposited in ventilation and process filter media may be a source of post-accident dose.

2. The integrated dose should be determined from estimated dose rates using appropriate integration factors determined for each of the major source terms (e.g., containment sump, containment atmosphere, ECCS, normal operation). The period of exposure should be consistent with the survivability period for the EQ equipment being evaluated. The survivability period is the maximum duration, post-accident, that the particular EQ component is expected to operate and perform its intended safety function. The period of exposure for normal operation dose is generally the duration of the plant license, i.e., 40 years.

FISSION PRODUCT CONCENTRATIONS

3. The radiation environment resulting from normal operations should be based on the conservative source term estimates reported in the facility's Safety Analysis Report or should be consistent with the primary coolant specific activity limits contained in the facility's technical specifications. The use of equilibrium primary coolant concentrations based on 1% fuel cladding failures would be one acceptable method. In estimating the integrated dose from prior normal operations, appropriate historical dose rate data may be used where available.

4. The radioactivity released from the core during a design basis loss-of-coolant accident (LOCA) should be based on the assumptions provided in Regulatory Position 3 and Appendix A of this regulatory guide. Although the design basis LOCA is generally limiting for radiological environmental qualification (EQ) purposes, there may be components for which another design basis accident may be limiting. In these cases, the assumptions provided in Appendices B through H of this regulatory guide, as applicable, should be used. Applicable features and mechanisms may be assumed in EQ calculations provided that any prerequisites and limitations identified regarding their use are met. There are additional considerations:

- For PWR ice condenser containments, the source should be assumed to be initially released to the lower containment compartment. The distribution of the activity should be based on the forced recirculation fan flow rates and the transfer rates through the ice beds as functions of time.

- For BWR Mark III designs, all the activity should be assumed initially released to the drywell area and the transfer of activity from these regions via containment leakage to the surrounding reactor building volume should be used to predict the qualification levels within the reactor building (secondary containment).

DOSE MODEL FOR CONTAINMENT ATMOSPHERE

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

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

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

DOSE MODEL FOR CONTAINMENT SUMP WATER SOURCES

8. With the exception of noble gases, all the activity released from the fuel should be assumed to be transported to the containment sump as it is released. This activity should be assumed to mix instantaneously and uniformly with other liquids that drain to the sump. This transport can also be modeled mechanistically as the time-dependent washout of airborne aerosols by the action of containment sprays. Radionuclides that do not become airborne on release from the reactor coolant system, e.g., they are entrained in non-flashed reactor coolant, should be assumed to be instantaneously transported to the sump and be uniformly distributed in the sump water.

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

DOSE MODEL FOR EQUIPMENT LOCATED OUTSIDE CONTAINMENT

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

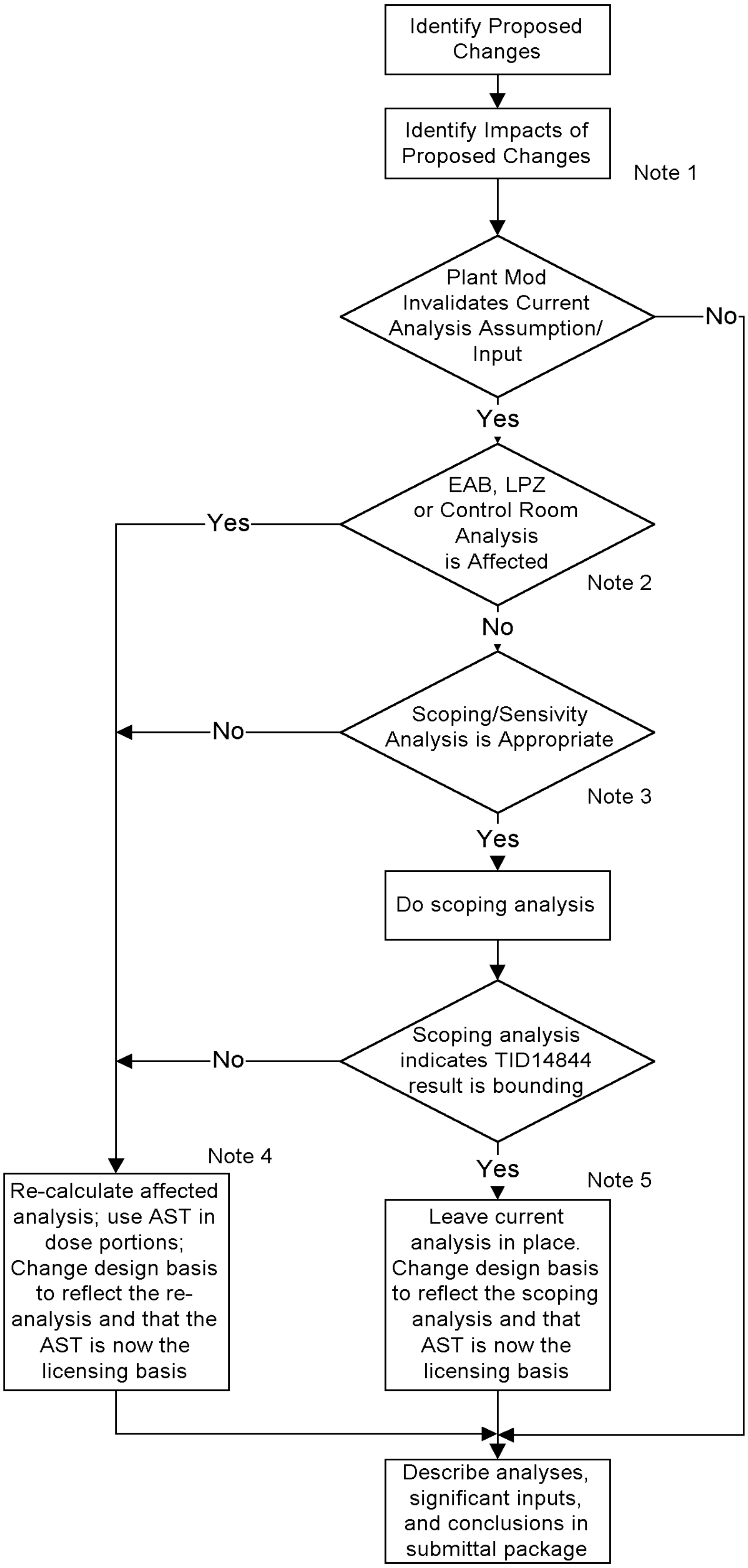
11. Because of the large amount of EQ equipment and the complexity of system and component layout in plant buildings, it is generally not reasonable to model each EQ component. A reasonable approach is to determine the limiting dose rate from all sources in a particular plant area (e.g., cubicle, floor,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

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

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

Appendix J

Analysis Decision Chart



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

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

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

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

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

Appendix K

Acronyms

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

VALUE / IMPACT STATEMENT

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

ADAMS Accession
Number ML003716792

March 22, 2010

Frederick (Ted) Schiffley, BWROG Chairman
Exelon Generation Co., LLC.
Cornerstone II at Cantera
4300 Winfield Road
Warrenville, IL 60555

SUBJECT: RESPONSE TO THE BOILING WATER REACTORS OWNER'S GROUP REQUEST TO EXTEND THE COMMENT PERIOD FOR DRAFT REGULATORY GUIDE – 1199

Dear Mr. Schiffley:

By letter dated January 6, 2010, the Boiling Water Reactor Owner's Group (BWROG) requested an extension of the public comment period for Draft Regulatory Guide – 1199 (DG-1199), "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," Agencywide Documents Access and Management Systems (ADAMS) Accession No. ML090960464, open from October 14, 2009, to January 13, 2010. The extension request stated that, in order to gain an understanding of the implications and potential consequences of the proposed revision, the BWROG will need to perform a detailed review of the Staff's research supporting the proposed changes to modeling of the main steamline isolation valve (MSIV) leakage. The January 6, 2010, letter also included a request for the MELCOR input decks supporting the boiling water reactor (BWR) MSIV leakage analyses.

The Nuclear Regulatory Commission (NRC) staff has reviewed the stated basis for the request to extend the public comment period. Based upon this review, the staff has determined it will not extend the public comment period for the reasons discussed below.

On October 9, 2010, the staff released the technical basis for the proposed DG-1199 MSIV modeling changes to the public in a Sandia National Laboratories Report, SAND2008-6601, "Analysis of Main Steam Isolation Valve Leakage in Design Basis Accidents Using MELCOR 1.8.6 and RADTRAD," ADAMS Accession No. ML083180196. On November 16, 2010, the staff held a full day public workshop that included a presentation on the proposed MSIV modeling changes, including an extensive discussion of the role of the supporting MELCOR work. Based on its review of the request by the BWROG, the staff has determined that no substantive issues with the staff's research were identified as the basis for extending the public comment period. Additionally, the staff believes that an extended period of time has been provided to provide comments on the proposed guidance. Further, the staff anticipates future communications with the industry prior to issuance of the final guidance.

The staff also considered the request to provide the MELCOR decks to the BWROG. Because these decks are based in part on the Peach Bottom and Grand Gulf plants, the staff determined this information cannot be provided directly to the BWROG. The staff recently released the decks directly to Grand Gulf and Peach Bottom as an alternative.

The staff will consider comments received after January 13, 2010, up to the date of issuance of this letter. Please note that the staff is able to ensure consideration only for comments received on or before March 22, 2010. Although a time limit is given, comments and suggested improvements in connection with items for inclusion in guides under development or published guides are encouraged at any time.

Respectfully,

/RA/

Travis L. Tate, Chief
Accident Dose Branch
Division of Risk Assessment
Office of Nuclear Reactor Regulation

Enclosure:
External cc list

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Respectfully,

/RA/

Travis L. Tate, Chief
Accident Dose Branch
Division of Risk Assessment
Office of Nuclear Reactor Regulation

Enclosure:
External cc list

DISTRIBUTION:

Mark Cunningham	Melanie Galloway	Stacey Rosenberg
Michelle Honcharik	Andrea Valentin (RES)	Robert Carpenter (RES)
Richard Lee (RES)	Michael Salay (RES)	

Accession Number: **ML100500773**

OFFICE	NRR/DRA/AADB	NRR/DRA/AADB:BC		
NAME	MBlumberg	TTate		
DATE	2/24/10	3/22/10		

OFFICIAL RECORD COPY

Mr. Oscar Limpias
BWROG, Executive Chairman
Entergy Nuclear
1340 Echelon Parkway.
Jackson, MS 39213-8202
olimpia@entergy.com

Paul J. Davison
BWROG, Executive Vice Chairman
Hope Creek Generating Station
P.O. Box 236
Hancocks Bridge, NJ 08038
paul.davison@pseg.com

Mr. Dennis Madison
Hatch Vice President
Southern Nuclear Operating Co.
11028 Hatch Parkway North
Baxley, GA 31515-2010
drmadiso@southernco.com

Mr. James F. Klapproth
GE-Hitachi Nuclear Energy
M/C A50
3901 Castle Hayne Road
Wilmington, NC 28401
james.klapproth@ge.com

Mr. Frederick Schiffley
BWROG, Chairman
Exelon Generation Co., LLC.
Cornerstone II at Cantera
4300 Winfield Road
Warrenville, IL 60555
frederick.schiffley@exeloncorp.com

Mr. Michael Crowthers
BWROG, Vice Chairman
PPL Susquehanna, LLC
2 North Ninth Street
Allentown, PA 18101-1179
mhcrowthers@pplweb.com

Mr. Douglas Coleman
Chairman, Regulatory Response Group
Energy Northwest
Columbia Generating Station
Mail Drop PE20
P.O. Box 968
Richland, WA 99352-0968
dwcoleman@energy-northwest.com

Mr. Craig J. Nichols
BWROG, Program Manager
GE-Hitachi Nuclear Energy
PO Box 780 M/C F12
3901 Castle Hayne Road
Wilmington, NC 28402
craig.nichols@ge.com

BWR OWNERS' GROUP

Douglas W. Coleman
BWROG Chairman
Tel: (509) 377-4342
Fax: (509) 377-2354
dwcoleman@energy-northwest.com

c/o Energy Northwest, Mail Drop PE04, P.O. Box 968, Richland, WA 99352-0968

BWROG-10003
January 6, 2010

Project No. 691

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2010 JAN -7 AM 9: 34
RULES AND DIRECTIVES BRANCH

Rulemaking, Directives, and Editing Branch
Office of Administration
US Nuclear Regulatory Commission
Washington, DC 20555-0001

SUBJECT: Draft Regulatory Guide, DG-1199 - BWR Owners' Group Request for Supporting Documentation and Comment Period Extension (Docket ID NRC–2009–0453)

REFERENCE: Draft Regulatory Guide, DG-1199, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors". The NRC requesting public comments by January 13, 2010. *Reference 74 Fed. Reg. 52,822, 55,273.*

The BWR Owners' Group has performed a preliminary review of U.S. Nuclear Regulatory Commission (NRC) published Draft Regulatory Guide, DG-1199, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors" released on October 14, 2009. The NRC is requesting public comments by January 13, 2010. *Reference 74 Fed. Reg. 52,822, 55,273.*

We note from our review that substantive changes are being proposed to the modeling of MSIV leakage. Leakage through the steam line pathway currently represents a significant fraction of the postulated LOCA doses in the existing DBA analysis for BWRs, including plants that credit the alternate leakage pathway via the condenser. The proposed changes in DG-1199 would have the effect of increasing the source term concentration entering the steam line by up to 20 times that of the current Regulatory Guide 1.183 methodology and assumptions. In turn, this will significantly impact the LOCA dose analysis.

These increased postulated doses resulting from the proposed changes to the regulatory guide may cause BWR utilities to incur significant additional costs to maintain compliance with 10CFR50.67 in conjunction with future licensing actions. Because of the significant impacts, a thorough review of the basis behind the proposed revision of the Regulatory Guide is warranted.

To gain an understanding of the implications and potential consequences of the revision to the Regulatory Guide and provide substantive comments, the BWROG would need to perform a detailed review of the Staff research supporting the proposed changes to the modeling of MSIV leakage. To facilitate this possible review, the BWROG makes the following two requests:

SUNSI Review Complete
Template = ADM-013

E-RIDS = ADM-03
ADD = R. Carpenter (REC1)
M. Blumberg (WMB1)

1. The BWROG requests that the MELCOR input decks supporting the BWR MSIV leakage analyses in SAND2008-6601 be released to the BWROG. These inputs would expedite the technical review since the BWROG would not need to completely re-create the Staff's analysis.

2. Since this review may result in the generation of important technical comments to the MSIV leakage portion of DG-1199, the BWROG requests an extension of the comment deadline. Any BWROG comments are expected to be available within 90 days of the release of the MELCOR input decks. If the NRC chooses not to release the input decks, any BWROG comments are expected to be available within 180 days after the current comment deadline of January 13, 2010. This additional time is necessary in order to re-create and benchmark the underlying analysis.

The BWR Owners' Group requests that NRC delay final issuance of the Regulatory Guide until after completion of our analysis and NRC receipt of comments.

Respectfully,

Douglas W. Coleman

Douglas W. Coleman, Chairman
BWR Owners' Group

cc: F.P. "Ted" Schiffley, BWROG Vice Chairman
C.J. Nichols, BWROG Program Manager
BWROG Primary Representatives
M. Blumberg, NRC
G. Broadbent, Entergy
T. Mscisz, Exelon
J. DeLaRosa, Exelon

July 26, 2011

MEMORANDUM TO: Travis L. Tate, Chief
Dose Assessment Branch
Division of Risk Assessment
Office of Nuclear Reactor Regulation

FROM: Anthony J. Mendiola, Chief **/RA/**
Nuclear Performance and Code Review Branch
Division of Safety Systems
Office of Nuclear Reactor Regulation

SUBJECT: TECHNICAL BASIS FOR REVISED REGULATORY GUIDE 1.183 (DG-1199) FISSION PRODUCT FUEL-TO-CLADDING GAP INVENTORY

The purpose of this memorandum is to document the technical basis for revised fission product fuel-to-cladding gap inventories within Section 3.2 of Regulatory Guide (RG) 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors." Specifically, the non-Loss-of-Coolant Accident (non-LOCA) fission-product gap inventories listed in Section 3.2 of RG 1.183 and used to assess the radiological consequences for the fuel handling accident, pressurized water reactor (PWR) locked rotor event, PWR sheared shaft event, PWR steam line break event, PWR control rod ejection event, and boiling water reactor (BWR) control blade drop event. Attachment 1 provides proposed revisions to Section 3.2 of RG 1.183. Attachment 2 contains a Pacific Northwest National Laboratory (PNNL) technical report which supports the recommended RG revisions.

In a memorandum dated February 10, 2009 (ML0903602560), the Nuclear Performance and Code Review Branch (SNPB) provided recommended changes to Section 3.2 of RG 1.183. These changes were incorporated and the draft RG was issued as DG-1199. Extensive public comments were received which prompted revision to the RG and supporting technical basis document. Comment resolution tables have been completed and provided informally to AADB. Attachment 1 provides the proposed revision to Section 3.2 of DG-1199 (RG 1.183). enclosure 2 contains the revised PNNL technical report which supports the recommended changes and provides a detailed analytical procedure for calculating gap fractions (Appendix C to PNNL-18212 Revision 1).

CONTACT: Paul Clifford, NRR/DSS
301-415-4043

In summary, the following revisions to DG-1199 are proposed:

- In response to public comment, text was added to Section 3.2 clarifying the applicability of Table 1, 2, 3, and 4 as well as the treatment of fuel melt during reactivity initiated accidents (RIA). Guidance from RG 1.183 returned to Appendices C and H.

- In response to public comment, bounding fuel rod power profiles were extended. This necessitated a re-calculation of the fission product gap inventories. As part of this effort, peak nodal power (Fq) was replaced with rod average power (Fr) which is a better qualifier for fission gas release along the entire fuel stack and overall gap inventory.

    - Table 3 non-LOCA fission product gap inventories updated.
    - Table 4 RIA combined fission product gap inventories updated.

- Analytical method for predicting I-132 inventory revised to more accurately capture precursor Te-132 effects.

- In response to public comment, an acceptable analytical technique for calculating non-LOCA fission product gap inventories based upon specific fuel rod designs or more realistic fuel rod power histories was documented (Appendix C to PNNL-18212 Revision 1).

Since the earlier memo (and end of DG-1199 public comment period), the American Nuclear Society approved the new gas release model used in the development of the revised Table 3 fission-product inventories - ANSI/ANS-5.4-2011 standard, “Method for Calculating the Fractional Release of Volatile Fission Products from Oxide Fuel” (revision of withdrawn standard ANSI/ANS-5.4-1982, approved May 19, 2011).

Enclosure:
As stated

In summary, the following revisions to DG-1199 are proposed:

- In response to public comment, text was added to Section 3.2 clarifying the applicability of Table 1, 2, 3, and 4 as well as the treatment of fuel melt during reactivity initiated accidents (RIA). Guidance from RG 1.183 returned to Appendices C and H.

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  - Table 4 RIA combined fission product gap inventories updated.

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Enclosure:
As stated

DISTRIBUTION: SNPB R/F PClifford AMendiola SBahadur

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OFFICE	NRR/DSS *	NRR/DSS/SNPB
NAME	PClifford	AMendiola
DATE	07/26/11	07/26/11

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Proposed Revisions DG-1199

Revised Section 3.2

Enclosure 1

3.2 Release Fractions[1]

For loss-of-coolant DBAs, Table 1 (for BWRs) and Table 2 (for PWRs) list the core inventory release fractions, by radionuclide groups, for the gap release and early in-vessel damage phases. These fractions are applied to the equilibrium core inventory described in Regulatory Position 3.1.

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

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

The applicability of Table 3 non-LOCA fission product gap fractions is limited to fuel rods with a peak rod average power history below the bounding power envelope depicted in Figure 1. Appendix K provides an acceptable analytical technique for calculating non-LOCA fission product gap inventories based upon specific fuel rod designs or more realistic fuel rod power histories. Reference 18 documents the methods used to calculate the Table 3 and Table 4 fission product inventories, including application of modeling uncertainties.

The non-LOCA fission product gap inventories listed in Table 3 and RIA combined release fractions listed in Table 4 do not include the additional contribution associated with fuel melting. Guidance for adjusting these gap inventories for fuel rods which are predicted to experience limited fuel centerline melting is provided in the event-specific appendices.

Table 1 BWR Core Inventory Fraction Released into Containment Atmosphere

---

1 The NRC has determined the release fractions listed here to be acceptable for use with currently approved LWR fuel. The data in this section are not applicable to cores containing mixed oxide (MOX) fuel.

{ No Change}

Table 2 PWR Core Inventory Fraction Released into Containment Atmosphere

{ No Change}

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

Group	Fraction
I-131	0.08
I-132	0.09
Kr-85	0.38
Other Noble Gases	0.08
Other Halogens	0.05
Alkali Metals	0.50

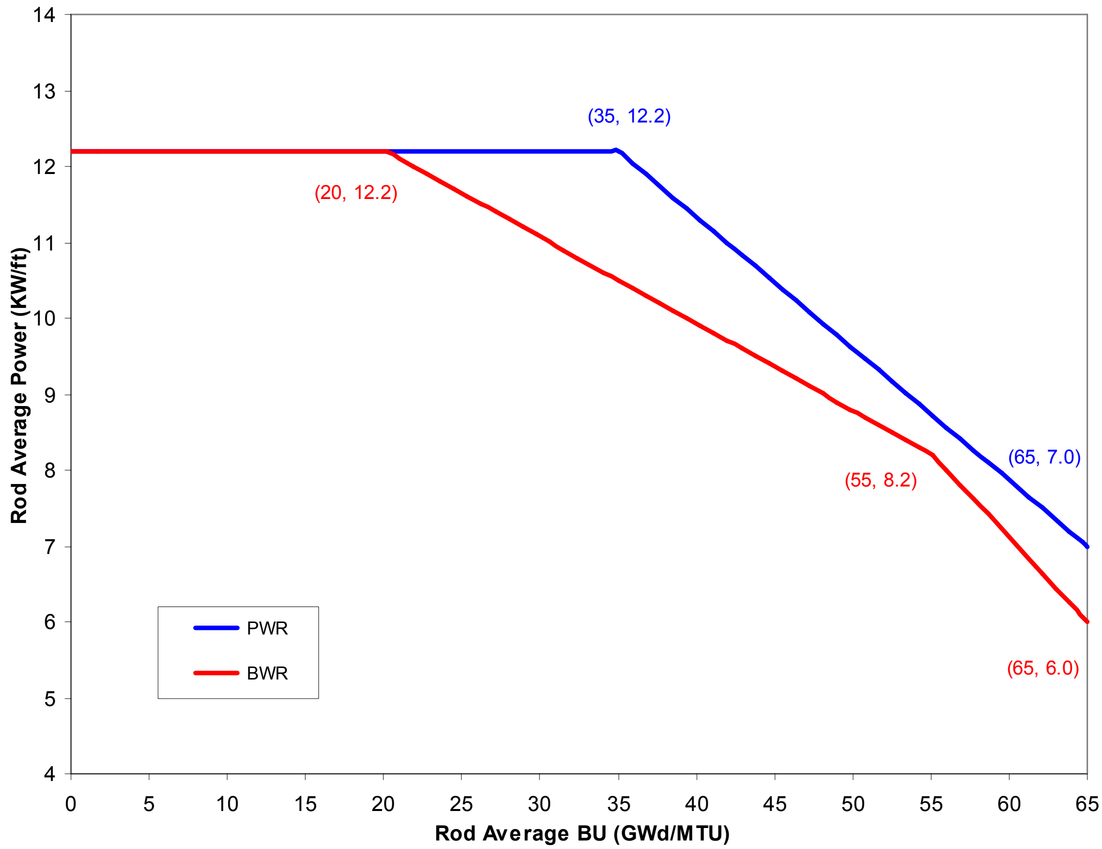
Table 4 Fraction of Fission Product Inventory Available for Release from Reactivity Initiated Accidents

Group	Combined Release Fraction[2,3]
I-131	((0.08) + (0.00073 * ΔH))
I-132	((0.09) + (0.00073 * ΔH))
Kr-85	((0.38) + (0.0022 * ΔH))
Other Noble Gases	((0.08) + (0.00073 * ΔH))
Other Halogens	((0.05) + (0.00073 * ΔH))
Alkali Metals	((0.50) + (0.0031 * ΔH))

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

3 Calculated values of combined release are limited to a value of 1.0.

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



PNNL Technical Report:

Update of Gap Release Fractions for Non-LOCA Events

Utilizing the Revised ANS 5.4 Standard

C. E. Beyer
P. M. Clifford

Revision 01

June 2011

ENCLOSURE 2

U.S. NUCLEAR REGULATORY COMMISSION
OFFICE OF NUCLEAR REGULATORY RESEARCH

October 2009
Division 1

DRAFT REGULATORY GUIDE

Contact: M. Blumberg
(301) 415-1083

DRAFT REGULATORY GUIDE DG-1199

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

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

A. INTRODUCTION

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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B. DISCUSSION

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

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

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

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

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

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

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

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

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

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

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

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

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

C. REGULATORY POSITION

1. Implementation of Accident Source Term

1.1 Generic Considerations

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

1.1.1 Safety Margins

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

1.1.2 Defense in Depth

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

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

1.1.3 Integrity of Facility Design Basis

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

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

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

1.1.4 Emergency Preparedness Applications

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

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

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

1.1.5 Applicability to 10 CFR Part 52

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

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

1.2 Scope of Implementation

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

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

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

1.2.1 Full Implementation

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

1.2.2 Selective Implementation

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

1.3 Scope of Required Analyses

1.3.1 Design basis Radiological Analyses

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

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

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

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

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

1.3.2 Reanalysis Guidance

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

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

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

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

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

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

1.3.3 Use of Sensitivity or Scoping Analyses

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

1.3.4 Updating Analyses Following Implementation

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

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

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

1.3.5 Equipment Environmental Qualification

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

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

1.4 Risk Implications

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

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

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

1.5 Submittal Requirements

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

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

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

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

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

1.6 Final Safety Analysis Report Requirements

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

2. Attributes of an Acceptable Accident Source Term

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

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

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

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

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

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

3. Accident Source Term

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

3.1 Fission Product Inventory

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

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

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

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

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

3.2 Release Fractions[11]

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

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

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

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

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

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

Table 1 BWR Core Inventory Fraction Released into Containment Atmosphere

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

Table 2 PWR Core Inventory Fraction Released into Containment Atmosphere

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

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

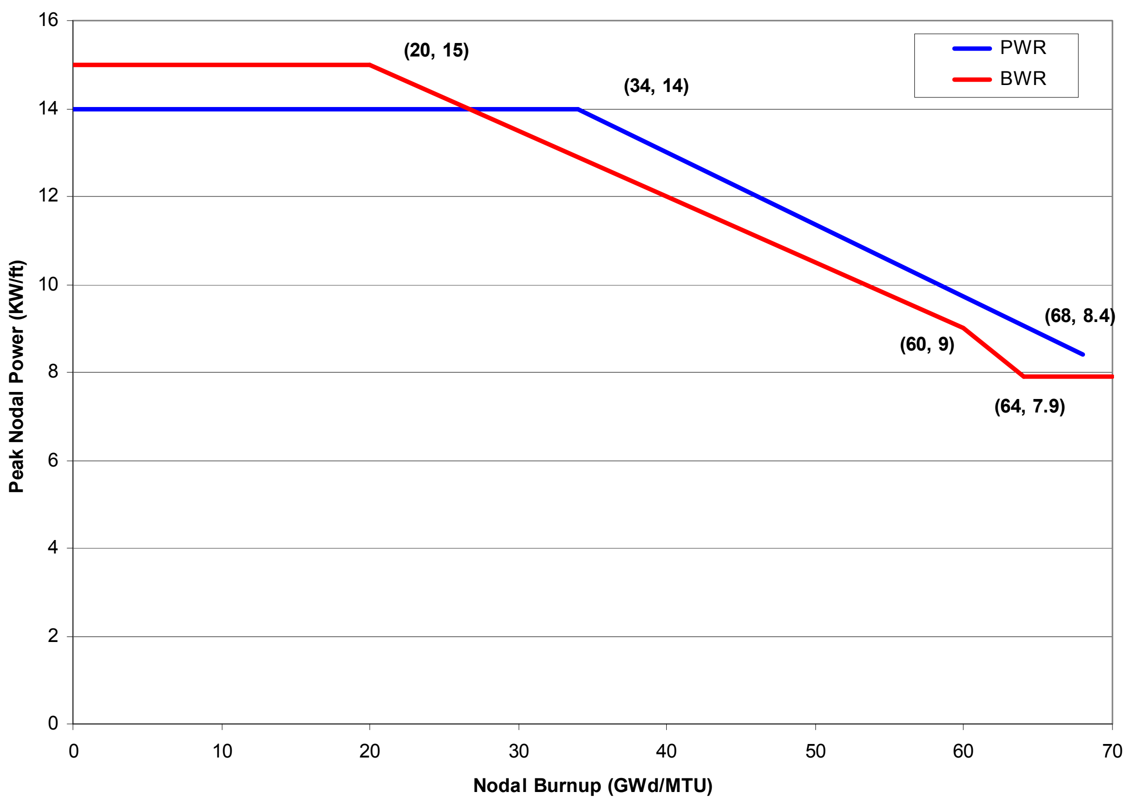
Group	Fraction
I-131	0.08
I-132	0.23
Kr-85	0.35
Other Noble Gases	0.04
Other Halogens	0.05
Alkali Metals	0.46

Table 4 Fraction of Fission Product Inventory Available for Release from

Reactivity Initiated Accidents

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

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



3.3 <u>Timing of Release Phases</u>

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

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

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

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

Table 5 LOCA Release Phases

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

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

3.4 Radionuclide Composition

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

Table 6 Radionuclide Groups

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

3.5 Chemical Form

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

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

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

3.6 Fuel Damage in Non-LOCA Design Basis Accidents

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

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

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

4. Dose Calculational Methodology

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

4.1 Offsite Dose Consequences

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

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

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

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

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

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

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

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

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

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

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

4.2 Control Room Dose Consequences

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

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

4.2.1 *Sources of Radiation*

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

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

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

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

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

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

4.2.2 *Materials Releases and Radiation Levels*

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

4.2.3 *Transport Models*

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

4.2.4 *Engineered Safety Features*

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

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

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

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

4.2.5 Personal Protective Equipment

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

4.2.6 Dose Receptor

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

4.2.7 Dose Conversion Factor

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

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

4.3 Other Dose Consequences

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

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

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

4.4 Acceptance Criteria

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

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

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

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

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

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

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

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

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

5. Analysis Assumptions and Methodology

5.1 <u>General Considerations</u>

5.1.1 Analysis Quality

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

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

5.1.2 Credit for Engineered Safeguard Features

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

5.1.3 Assignment of Numeric Input Values

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

5.1.4 Applicability of Prior Licensing Basis

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

5.2 Accident-Specific Assumptions

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

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

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

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

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

5.3 Meteorology Assumptions

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

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

D. IMPLEMENTATION

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

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

REGULATORY ANALYSIS

Statement of Problem

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

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

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

Existing Regulatory Framework

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

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

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

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

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

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

Objective of the Regulatory Action

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

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

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

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

Alternative Approaches

The NRC considered the following alternative approaches:

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

Alternative 1: Do Not Revise Regulatory Guide 1.183

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

Alternative 2 Revise Regulatory Guide 1.183

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

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

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

Evaluation of Values and Impacts

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

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

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

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

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

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

Conclusion

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

BACKFIT ANALYSIS

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

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

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

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11. U.S. Nuclear Regulatory Commission, "Environmental Qualification of Certain Electric Equipment Important to Safety for Nuclear Power Plants," Regulatory Guide 1.89.

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

12. U.S. Nuclear Regulatory Commission, "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.

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APPENDIX A

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

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

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

A-1. Source Term

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

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

A-2. Transport in Primary Containment

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

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

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

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

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

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

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

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

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

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

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

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

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

A-3. Dual Containments

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

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

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

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

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

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

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

A-4. Assumptions on Engineered Safety Feature System Leakage

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

A-6. Containment Purging

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

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

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

APPENDIX A

REFERENCES

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

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

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

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

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

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

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

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

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

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

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

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

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

APPENDIX B

ASSUMPTIONS FOR EVALUATING THE RADIOLOGICAL CONSEQUENCES OF A FUEL HANDLING ACCIDENT

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

B-1. Source Term

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

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

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

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

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

2. Water Depth

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

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

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

B-3. Noble Gases and Particulates

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

B-4. Fuel handling Accidents within the Fuel Building

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

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

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

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

B-5. Fuel handling Accidents within Containment

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

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

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

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

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

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

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

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

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

APPENDIX B

REFERENCES

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

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

APPENDIX C

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

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

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

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

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

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

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

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

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

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

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

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

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

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

APPENDIX D

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

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

Source Term

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

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

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

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

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

Transport

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

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

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

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

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

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

APPENDIX E

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

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

Source Term

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

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

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

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

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

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

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

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

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

Transport

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

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

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

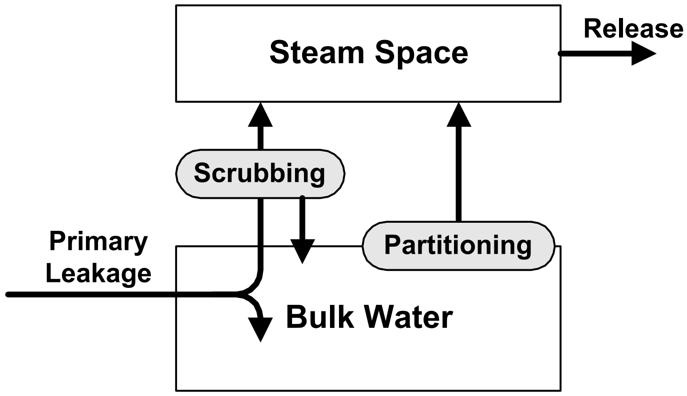
E-6.3 The primary-to-secondary leakage should be assumed to continue until the primary system pressure is less than the secondary system pressure, or until the temperature of the leakage is less than 100 degrees Celsius (212 degrees Fahrenheit). The release of radioactivity from the unaffected steam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

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

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

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

Figure E-1
Transport Model



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

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

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

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

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

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

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

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[3] "Partition coefficient" is defined as follows:

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

APPENDIX E

REFERENCES

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

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

APPENDIX F

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

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

Source Term

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

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

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

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

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

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

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

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

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

Transport

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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[4] "Partition coefficient" is defined as follows:

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

APPENDIX F

REFERENCES

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

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

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

APPENDIX G

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

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

Source Term

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

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

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

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

Release Transport

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

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

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

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

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

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

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

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

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

APPENDIX H

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

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

Source Term

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

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

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

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

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

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

Transport from Containment

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

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

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

Transport from Secondary System

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

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

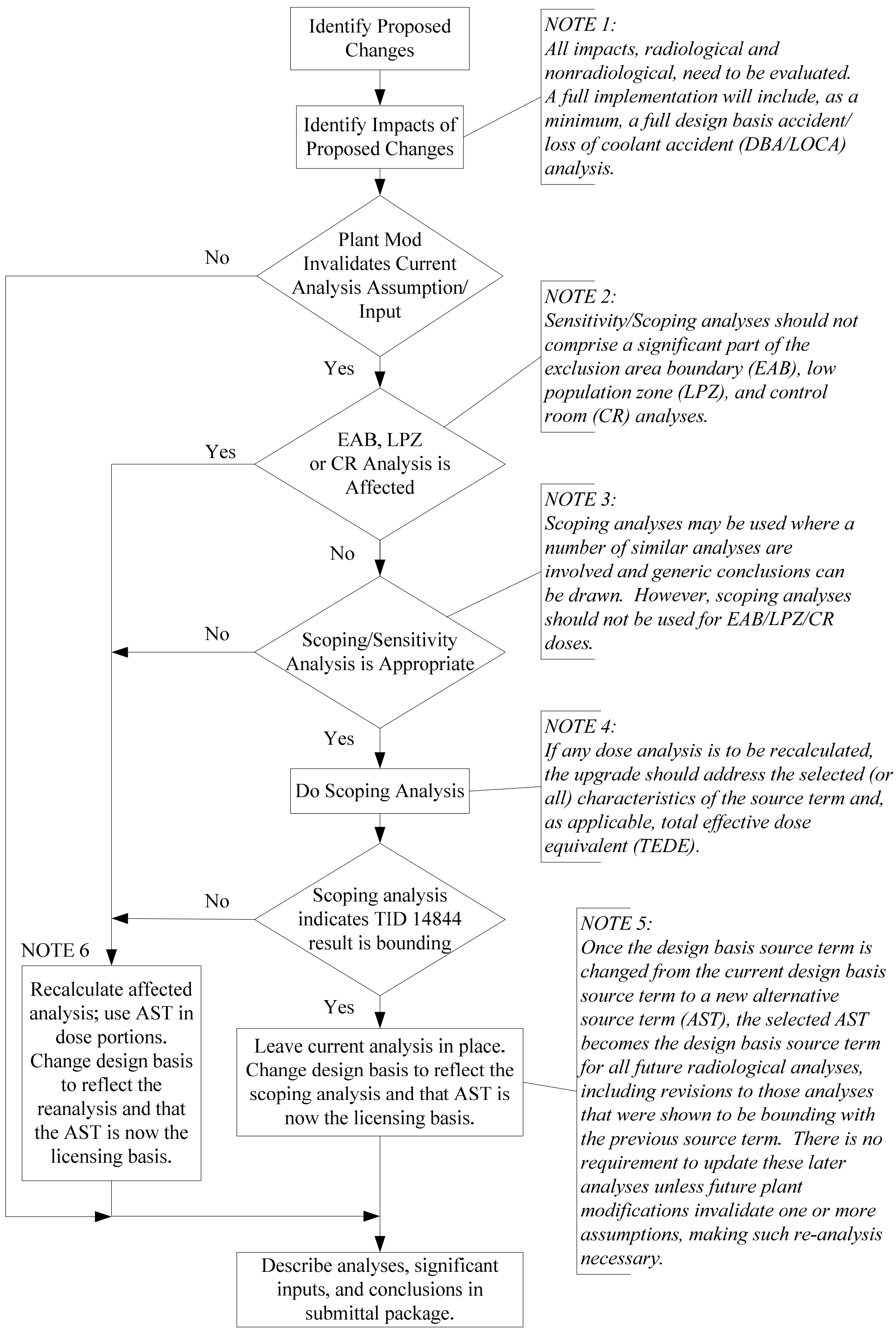
H-7.2 The density used in converting volumetric leak rates (e.g., gallons per minute) to mass leak rates (e.g., pound mass per hour) should be consistent with the basis of surveillance tests used to show compliance with leak rate technical specifications. These tests typically are based on cooled liquid.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

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

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

APPENDIX I

ANALYSIS DECISION FLOWCHART



APPENDIX J

ACRONYMS

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

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

U.S.NRC
United States Nuclear Regulatory Commission
Protecting People and the Environment

Information Notice No. 90-08: KR-85 Hazards From Decayed Fuel

UNITED STATES
NUCLEAR REGULATORY COMMISSION
OFFICE OF NUCLEAR REACTOR REGULATION
WASHINGTON, D.C. 20555

February 1, 1990

Information Notice No. 90-08: KR-85 HAZARDS FROM DECAYED FUEL

Addressees:

All holders of operating licenses or construction permits for nuclear power reactors and holders of licenses for permanently shutdown facilities with fuel on site.

Purpose:

This information notice alerts addressees to potential problems resulting from the accidental release of Kr-85 from decayed fuel. It is expected that recipients will review the information for applicability to their facilities and consider actions, as appropriate, to avoid similar problems. However, suggestions contained in this information notice do not constitute NRC requirements; therefore, no specific action or written response is required.

Description of Circumstances:

During the licensing reviews for the Oconee independent spent fuel storage installation, and in the decommissioning of the La Crosse and Dresden Unit 1 power reactors, the NRC staff analyzed the radiological hazards associated with the gases in decayed spent fuel. The age of the nuclear power industry and the lack of a permanent repository for spent fuel have resulted in the accumulation of decayed spent fuel. Decayed spent fuel is manipulated after long shutdowns of operating reactors, during spent fuel pool re-racking, during movement to alternate reactor sites or independent spent fuel storage installations, and during decommissioning. Analysis of hypothetical accidents involving decayed spent fuel has focused attention on potential difficulties that could be associated with the exposure of onsite personnel to an accidental release of Kr-85. Kr-85 is a noble gas fission product that is present in the gaps between the fuel pellets and the cladding. It has a 10.76-year half-life, and, as a result of the considerably shorter half-lives of virtually all other gaseous fission products (I-129 being the exception, but in low abundance), Kr-85 becomes increasingly the dominant nuclide in the accident source term for gap releases as decay times increase. After 2 weeks of decay, Kr-85 is a significant nuclide in the source term, and after 190 days of decay, it is the predominant gaseous nuclide for a gap release. The unusual decay characteristics of Kr-85 give cause for focusing attention on the onsite consequences of a gap release from decayed fuel.

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.

Discussion:

Kr-85 emits beta radiation with a maximum energy of 0.67 MeV for 99.6 percent of the decays and 0.51 MeV gamma radiation for 0.4 percent of the decays. Consequently, direct exposure to this gas would result in a dose to the skin approximately 100 times the whole-body dose. Analysis of the relative consequences (in terms of radiological doses) of a cask-drop accident as a function of decay time of the fuel is illustrated in Figure 1. In the event of a serious accident involving decayed spent fuel, protective actions would be needed for personnel on site, while offsite doses (assuming an exclusion area radius of 1 mile from the plant site) would be well below the Environmental Protection Agency's Protective Action Guides. Accordingly, it is important to be able to properly survey and monitor for Kr-85, and to assess the skin dose to workers who could be exposed to Kr-85 in the event of an accident with decayed spent fuel.

Licensees may wish to reevaluate whether Emergency Action Levels specified in the emergency plan and procedures governing decayed fuel-handling activities appropriately focus on concern for onsite workers and Kr-85 releases in areas where decayed spent fuel accidents could occur, for example, the spent fuel pool working floor. Furthermore, licensees may wish to determine if emergency plans and corresponding implementing procedures address the means for limiting radiological exposures of onsite personnel who are in other areas of the plant. Among other things, moving onsite personnel away from the plume and shutting off building air intakes downwind from the source may be appropriate.

This information notice requires no specific action or written response. If you have any questions about the information in this notice, please contact one of the technical contacts listed below or the appropriate NRR project manager.

Charles E. Rossi, Director
Division of Operational Events Assessment
Office of Nuclear Reactor Regulation

Technical Contacts: Charles S. Hinson, NRR
(301) 492-3142

Robert A. Meck, RES
(301) 492-3737

Attachments:
1. Figure 1, Dose Consequences of a
   Spent Fuel Drop Accident
2. List of Recently Issued NRC Information Notices

.ENDEND

Page Last Reviewed/Updated Thursday, March 25, 2021

From:	Brian Magnuson
To:	Smith, Micheal
Cc:	Blumberg, Mark; Meighan, Sean
Subject:	[External_Sender] Regulatory Guide 1.183 Revision Public Meeting - Magnuson Comments
Date:	Tuesday, December 08, 2020 10:09:56 PM

December 8, 2020

Micheal:

My RG 1.183 Public Meeting comments are inserted (**bold font**) in the NRC's presentation below.

Please send the ML# when they are placed in ADAMS.

Thank you,
Brian

RG 1.183 Public Meeting November 19, 2020 – Brian Magnuson Comments

The NRC staff has restarted efforts to revise RG 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors."

> **DG-1199 (Draft RG 1.183 Revision 1) was the first effort to revise RG 1.183. It was prompted by SAND2008-6601 and published by the NRC in 2009; however, it was never implemented. After eleven years, what prompted this effort?**

incorporate relevant operating experience as well as recent post-Fukushima seismic risk insights and walkdowns;

> **As important, are the accident source terms insights from Fukushima that were incorporated into RASCAL 4 (NUREG-1430, September 2012) source terms and methodologies. Will these insights be incorporated into RG 1.183-Revision 1?**
>
> **Why is the revision to RG 1.183 lagging behind revisions to RASCAL?**

ensure sufficient guidance is in place for licensing advanced light-water reactors (LWRs), accident tolerant fuel (ATF), high-burnup, and increased
enrichment fuel; and,

> **NUREG-1465 (1995) "Accident Source Terms for Light-Water Nuclear Power Plants":**
>
> > ***"Recent information has indicated that high burnup fuel, that is, fuel irradiated at levels in excess of about 40 GWD/MTU, may be more prone***

to failure during design basis reactivity insertion accidents (RIA) than previously thought. Preliminary indications are that high burnup fuel also may be in a highly fragmented or powdered form, so that failure of the cladding could result in a significant fraction of the fuel itself being released."

The underlying concern identified here, is a cladding failure source term release could exceed that of a fuel melt source term release. What should be considered in RG 1.183-Revision 1, is the radiological consequences of a lessor and more likely accident may be worse than the "maximum credible accident" assumed in licensees' current licensing bases.

Reports and studies (e.g., Resolution of Generic Safety Issues: Issue 170: Fuel Damage Criteria for High Burnup Fuel (Rev. 2)) have evaluated high-burnup fuel and approved higher burn-up levels, but they have neither disputed the fuel disintegration caused by high-burnup nor evaluated the consequences of a powdered fuel source term. Until this NUREG-1465 concern has been eliminated, any revision to RG 1.183 should include a powdered fuel source term.

Limited range of applicability on Non-LOCA release fractions

Notably, DG-1199 significantly increased Non-LOCA nobel gas release fractions (above RG 1.183 Revision 0) and returned them to NUREG-1465 levels.

Excessive MISV leakage rates and realizations from the TMI accident prompted control room habitability studies and modifications to install Control Room Emergency Ventilation/Filter Systems. Subsequently, RG 1.183-Revision 0 required Control Room Operator doses to be evaluated for specific accidents, including the Non-LOCA fuel handling accident (FHA); however, missing from RG 1.183-Revision 0 is a requirement to evaluate doses to those workers/fuel handlers that would be in close proximity to this accident. Given the concerns identified in NRC Information Notice No. 90-08: "*KR-85 Hazards From Decayed Fuel*" and estimations based on FHA doses to control room operators, workers near spent fuel pools during would undoubtedly be overexposed (> 5 Rem TEDE).

Because no amount of water in spent fuel pools will not prevent the release of nobel gas (Kr-85, a pure beta emitter) in a FHA, revisions to RG 1.183 should require the calculation of spent fuel pool doses to ensure workers are aware of the hazards. This calculation could also be used to ensure the viability of FLEX actions to intended to mitigate an extended loss of spent fuel pool cooling.

DG-1199
In October 2009, the NRC issued for public comment DG-1199 as a proposed Rev. 1 of RG 1.183.

Staff received 150 public comments

The reasons for revision of RG 1.183 in DG-1199 were:

Providing additional guidance for modeling BWR MSIV leakage,

> **SAND2008-6601 determined RG 1.183 BWR MSIV leakage source terms and methodologies are "non-conservative and conceptually in error." These conceptual errors (and others) should be corrected in any revision to RG 1.183.**

2019 License Amendment Requests

In 2019, NRC received several AST LARs requesting increased MSIV leakage
As a result, work on DG-1199 was postponed to allow NRC staff to incorporate lessons learned, from evaluation of the LARs, into the revised RG 1.183:

James A. FitzPatrick Amendment No. 338 for AST, July 21, 2020 (ML20140A070)
Quad Cities Nuclear Power Station, Units 1 & 2 – Amendment Nos. 281 and 277 to increase allowable MSIV leakage, June 26, 2020 (ML20150A328)
Nine Mile Point Nuclear Station, Unit 2 – Amendment No. 182 to change allowable MSIV leak rates, October 20, 2020 (ML20241A190)
Dresden Nuclear Power Station, Units 2 & 3 – Amendments Nos. 272 and 265 to increase allowable MSIV leakage, October 23, 2020 (ML20265A240)

> **Does the NRC mean say LARs from last year (2019) cause a 11-year delay? DG-1199 (RG 1.183 Revision 1 Draft) was published by the NRC in 2009. In consideration of "The NRC Approach to Open Government," please explain the 11-year delay.**
>
> **SAND2008-6601 clearly explains/illustrates that RG 1.183 MSIV Leakage source terms and metrologies are "non-conservative and conceptually in error." Given this, why did the NRC approve the use of non-conservative and conceptually inaccurate guidance to increase MSIV leakage?**

The intent of the NRC staff is for RG 1.183 Rev. 0 and Rev. 1 to co- exist

> **With known, fundamental errors in RG 1.183-Revision 0, why would the NRC allow it to co-exist?**
>
> **The NRC's "RESULTS OF PERIODIC REVIEW OF REGULATORY GUIDE 1.183," dated June 25, 2018, states:**
>
>> ***"The known technical and regulatory issues are addressed in a draft revision to RG 1.183 issued for public comment (Draft Guide (DG)-1199, "Alternative Radiological Source Terms for Evaluating Design-Basis***

Accidents at Nuclear Power Reactors," published October 2009 (ADAMS Accession No. ML090960464)). The main technical issues are addressed in Regulatory Position (RP) 3.2, "Release Fractions," RP 5.3, "Meteorology Assumptions," and RP A-5, "Main Steam Isolation Value Leakage in Boiling Water Reactors.""

DG-1199 was prompted by SAND2008-6601, which determined RG 1.183-Revision 0 source terms and methodologies are conceptually inaccurate. The intent of DG-1199 was to correct the fundamental errors in RG 1.183-Revision 0. Is this still the intent of RG 1.183-Revision 1?

RG 1.183 states:

"The design basis accident source term is a fundamental assumption upon which a significant portion of the facility design is based."

Considering the significance of the accident source term, why would the NRC continue to allow licensees to use RG 1.183-Revision 0? Is not negligent to allow licensees to base nuclear power safety (systems) on conceptually inaccurate and non-conservative accident source terms?

Revised Fuel Handling Accident

Revisited the original studies forming the technical basis for the FHA and incorporate updated information.

Model improvements established from the current understanding of reactor fuel pin physics and iodine chemistry under the environmental conditions in which fuel handling operations are taking place.

Concluded that considerable margin exists regarding the scrubbing effects of iodine in the spent fuel or reactor pool and that the current staff DBA FHA fission product transport model can be refined while still maintaining conservatism.

Reference: Memo from RES to NRR, "Closeout to Research Assistance Request for Independent Review of Regulatory and Technical Basis for Revising the Design-basis Accident Fuel Handling Accident," November 23, 2019 (ML19270E335)

Prior to the accident at Three Mile Island (1979) and years afterward, control room operators were not protected by emergency air filtration systems. Operator doses from a DBA FHA (and other DBAs) were not publicly communicated because they exceeded General Design Criterion 19 limits (< 5 Rem whole body). After RG 1.183 was approved, the NRC required control room emergency filtration systems to be installed, and when their dose reduction factors were applied, operator doses were restored to within the

new limits of 10 CFR 50.67 (< 5 Rem TEDE). Even still, today control room operator doses are often the most limiting regulatory dose.

While there may be margin regarding the iodine doses to control room operators, there is no margin regarding the Kr-85 doses in a DBA FHA. No amount of water in spent fuel pools will mitigate or prevent the release of Kr-85 in a FHA, and nobel gasses cannot be filtered. Consideration of "*KR-85 Hazards From Decayed Fuel*" (Information Notice No. 90-08) is conspicuously missing from RG 1.183-Revision 0. Any revision RG 1.183 should address IN 90-08 concerns and require that doses to fuel handlers/workers in the area of a FHA be calculated.

Over the last 10 years no applicant or licensee has adopted the methodology from SAND2008-6601, "Analysis of Main Steam Isolation Valve Leakage in Design Basis Accident Using MELCOR 1.8.6 and RADTRAD."

There have been no communications that applicants or licensees intend to adopt the SAND2008-6601 methodology.

SAND2008-6601 is a scientific study performed by Sandia National Laboratories on behalf of the NRC that clearly explains/illustrates that RG 1.183 BWR MSIV source terms and metrologies are "non-conservative and conceptually in error." It is the technical basis for the "*proposed DG-1199 MSIV modeling changes.*" Nuclear power plant owners (licensees) have not adopted SAND2008-6601 (and have resisted DG-1199) because it is unlikely that they can comply with 10 CFR 50.67 if accurate MSIV leakage models and source terms are used. Please refer to the following January 2010 letters.

January 6, 2010, Draft Regulatory Guide, DG-1199 - BWR Owners' Group Request for Supporting Documentation and Comment Period Extension (Docket ID NRC-2009-0453):

We note from our review that substantive changes are being proposed to the modeling of MSIV leakage. Leakage through the steam line pathway currently represents a significant fraction of the postulated LOCA doses in the existing DBA analysis for BWRs, including plants that credit the alternate leakage pathway via the condenser. The proposed changes in DG-1199 would have the effect of increasing the source term concentration entering the steam line by up to 20 times that of the current Regulatory Guide 1.183 methodology and assumptions. In turn, this will significantly impact the LOCA dose analysis.

January 20, 2010, Nuclear Energy Institute Comments on U.S. Nuclear Regulatory Commission Draft Regulatory Guide DG-1199, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors" *(Federal Register* of October 14, 2009, 74 FR

52822).

"*It is unlikely that BWRs would commit to using it due to extreme penalties with regard to MSIV leakages* (Item 83)."

As stated in NRC's, March 22, 2010, "RESPONSE TO THE BOILING WATER REACTORS OWNER'S GROUP REQUEST TO EXTEND THE COMMENT PERIOD FOR DRAFT REGULATORY GUIDE – 1199":

"By letter dated January 6, 2010, the Boiling Water Reactor Owner's Group (BWROG) requested an extension of the public comment period for Draft Regulatory Guide – 1199 (DG-1199), "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," Agencywide Documents Access and Management Systems (ADAMS) Accession No. ML090960464, open from October 14, 2009, to January 13, 2010. The extension request stated that, in order to gain an understanding of the implications and potential consequences of the proposed revision, the BWROG will need to perform a detailed review of the Staff's research supporting the proposed changes to modeling of the main steam line isolation valve (MSIV) leakage."

"The Nuclear Regulatory Commission (NRC) staff has reviewed the stated basis for the request to extend the public comment period. Based upon this review, the staff has determined it will not extend the public comment period for the reasons discussed below."

"On October 9, 2010* [sic]*, the staff released the technical basis for the proposed DG-1199 MSIV modeling changes to the public in a Sandia National Laboratories Report, SAND2008-6601, "Analysis of Main Steam Isolation Valve Leakage in Design Basis Accidents Using MELCOR 1.8.6 and RADTRAD," ADAMS Accession No. ML083180196. On November 16, 2010* [sic]*, the staff held a full day public workshop that included a presentation on the proposed MSIV modeling changes, including an extensive discussion of the role of the supporting MELCOR work. Based on its review of the request by the BWROG, <u>the staff has determined that no substantive issues with the staff's research were identified</u> as the basis for extending the public comment period. Additionally, the staff believes that an extended period of time has been provided to provide comments on the proposed guidance."

Has the NRC disavowed SAND2008-6601?

If not, why has the NRC allowed licensees to use non-conservative and conceptually inaccurate MSIV leakage models and source terms for the past ten years?

If not, why would the NRC allow RG 1.183-Revision 0 to co-exist with RG 1.183-Revision 1?

The design basis accident source term is a fundamental assumption upon which a significant portion of every nuclear power plant design is based; therefore, RG 1.183-Revision 0 is, essentially, a generic safety issue.

The NRC's failure to act on this fundamental safety issue prompted PRM-50-122—10 CFR Part 2.802 request for rulemaking.

Additional Considerations
Consider revising footnote 7 which provides an incorrect method to convert thyroid dose to TEDE
Implies a back-of-the-envelope calculation appropriately converts between ICRP 2 and ICRP 26/30 dosimetry methodologies.
There is no simple methodology to convert between these two systems of dosimetry.
To correctly calculate the radiological dose consequences for design basis accidents the appropriate dose methodology (and DCFs) must be applied.

During the RG 1.183 public meeting on November 19, 2020, an industry member commented that the incorrect methods, described in RG 1.183, to calculate the radiological dose consequences, were used to assess Operability of structures, systems and components required by plant Technical Specifications.

Again, why would the NRC allow RG 1.183-Revision 0 to co-exist with RG 1.183-Revision 1?

From: Brian Magnuson
Sent: Saturday, December 5, 2020 3:58 PM
To: Smith, Micheal
Subject: RE: RE: Regulatory Guide 1.183 Revision Public Meeting Notice

Micheal:

I apologized for the late response.

The public meeting was informative. -Thank you.

Unfortunately, my attempts to make comments during the meeting failed for some reason. Because of this, I will revise my comments based on what I learned and resubmit them for ADAMS.

Regards,

Brian

From: Smith, Micheal
Sent: Friday, December 4, 2020 10:29 AM
To: Brian Magnuson
Subject: RE: RE: Regulatory Guide 1.183 Revision Public Meeting Notice

Brian,

I have not heard back from you so I did want to make you aware that I intend on placing your email below into ADAMS before the end of next week. I appreciate you taking the time to participate in our public meeting.

Enjoy your weekend!

Micheal Smith

Health Physicist and Assistant Radiation Safety Officer
Radiation Protection and Consequence Branch
Division of Risk Assessment
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
301-415-3763

From: Smith, Micheal
Sent: Thursday, November 19, 2020 4:21 PM
To: Brian Magnuson <magnuson28@msn.com>
Cc: Blumberg, Mark <Mark.Blumberg@nrc.gov>; Meighan, Sean <Sean.Meighan@nrc.gov>
Subject: RE: RE: Regulatory Guide 1.183 Revision Public Meeting Notice

Brian,

Thank you for taking the time to provide us with your questions and comments. As long as you are alright with it I plan on putting your email into ADAMS so that we can make sure we consider your questions and comments as we develop our draft guide. I will provide you the ML# once I have it.

If you have any additional questions just let us know.

Thanks,

Micheal Smith

Health Physicist and Assistant Radiation Safety Officer
Radiation Protection and Consequence Branch
Division of Risk Assessment
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
301-415-3763

From: Brian Magnuson <magnuson28@msn.com>
Sent: Thursday, November 19, 2020 3:25 PM
To: Smith, Micheal <Micheal.Smith@nrc.gov>
Cc: Blumberg, Mark <Mark.Blumberg@nrc.gov>; Meighan, Sean <Sean.Meighan@nrc.gov>
Subject: [External_Sender] RE: Regulatory Guide 1.183 Revision Public Meeting Notice

Micheal:

I have comments and questions.

From: Brian Magnuson
Sent: Thursday, November 19, 2020 11:55 AM
Subject: RE: Regulatory Guide 1.183 Revision Public Meeting Notice

Micheal:

I'm not sure how much time will be available today for comments; therefore, I have included some observations and questions regarding the presentation below.

Please review accordingly and let me know if you have any questions.

Thank you,
Brian

The NRC staff has restarted efforts to revise RG 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors."

DG-1199 (Draft RG 1.183 Revision 1) was approved (but not issued) by the NRC in 2010. After ten years, what prompted this effort?

incorporate relevant operating experience as well as recent post-Fukushima seismic risk insights and walkdowns;

Insights from Fukushima were previously incorporated into RASCAL (NUREG-1430) source terms and methodologies. Will these same insights be incorporated into RG 1.183 Revision 1? Why is the revision to RG 1.183 lagging behind revisions to RASCAL? Also, please explain why RASCAL does not use RG 1.183 source terms and methodologies.

ensure sufficient guidance is in place for licensing advanced light-water reactors (LWRs), accident tolerant fuel (ATF), high-burnup, and increased enrichment fuel; and,

NUREG-1465 (1995) "Accident Source Terms for Light-Water Nuclear Power Plants":

> ***"Recent information has indicated that high burnup fuel, that is, fuel irradiated at levels in excess of about 40 GWD/MTU, may be more prone to failure during design basis reactivity insertion accidents (RIA) than previously thought. Preliminary indications are that high burnup fuel also may be in a highly fragmented or <u>powdered form, so that failure of the cladding could result in a significant fraction of the fuel itself being released</u>."***

The underlying concern identified here is a cladding failure source term release could exceed that of a fuel melt source term release. What should be considered is, the radiological consequences of a lessor and more likely accident may be the new "maximum credible accident."

Reports and studies (e.g., Resolution of Generic Safety Issues: Issue 170: Fuel Damage Criteria for High Burnup Fuel (Rev. 2)) have evaluated high-burnup fuel and approved higher burn-up levels, but they have neither disputed the fuel disintegration caused by high-burnup nor evaluated the consequences of a powdered fuel source term. Until this NUREG-1465 concern has been openly eliminated, any revision to RG 1.183 should include a powdered fuel source term.

Limited range of applicability on Non-LOCA release fractions

Notably, DG-1199 significantly increased Non-LOCA nobel gas release fractions (above RG 1.183 Revision 0) and returned them to NUREG-1465 levels.

Excessive MISV leakage rates and the TMI accident prompted control room habitability studies, regulation and modifications to install Control Room Emergency Ventilation/Filter Systems. Subsequently, RG 1.183 Revision 0 required Control Room Operator) doses to be evaluated for specific accidents, including the Non-LOCA fuel handling accident (FHA); however, missing from

RG 1.183 is a requirement to evaluate doses to those fuel handlers/workers that would be in close proximity to this accident. Given the concerns identified the NRC identified in Information Notice No. 90-08: "*KR-85 Hazards From Decayed Fuel*" and the doses to control room the doses these ground zero workers could exceed federal limits and threaten their health and safety.

Because the water in spent fuel pools will not prevent the release of nobel gas (Kr-85, a pure beta emitter) in a FHA (mechanical damage or overheating), revisions to RG 1.183 should require the analysis of local doses to ensure the safety of workers in the area at the time of the accident. Additionally, the Non-LOCA FHA source term and methodologies should be used to ensure the viability of FLEX actions to intended to mitigate an extended loss of spent fuel pool cooling.

DG-1199
In October 2009, the NRC issued for public comment DG-1199 as a proposed Rev. 1 of RG 1.183.
Staff received 150 public comments

The reasons for revision of RG 1.183 in DG-1199 were:

Providing additional guidance for modeling BWR MSIV leakage,

SAND2008-6601 determined RG 1.183 BWR MSIV leakage source terms and methodologies are "non-conservative and conceptually in error." These conceptual errors (and others) should be corrected in any revision to RG 1.183.

2019 License Amendment Requests

In 2019, NRC received several AST LARs requesting increased MSIV leakage
As a result, work on DG-1199 was postponed to allow NRC staff to incorporate lessons learned, from evaluation of the LARs, into the revised RG 1.183:

James A. FitzPatrick Amendment No. 338 for AST, July 21, 2020 (ML20140A070)
Quad Cities Nuclear Power Station, Units 1 & 2 – Amendment Nos. 281 and 277 to increase allowable MSIV leakage, June 26, 2020 (ML20150A328)
Nine Mile Point Nuclear Station, Unit 2 – Amendment No. 182 to change allowable MSIV leak rates, October 20, 2020 (ML20241A190)
Dresden Nuclear Power Station, Units 2 & 3 – Amendments Nos. 272 and 265 to increase allowable MSIV leakage, October 23, 2020 (ML20265A240)

Does the NRC mean say LARs from last year (2019) cause a 10-year delay? DG-1199 was approved (but not issued) by the NRC in 2010. In consideration of "The NRC Approach to Open Government," please explain the 10-year delay.

Because SAND2008-6601 clearly explains/illustrates that RG 1.183 MSIV Leakage source terms and metrologies are "non-conservative and

conceptually in error," it does not seem that LARs to increase MSIV leakage are in the best interest of public health and safety.

The intent of the NRC staff is for RG 1.183 Rev. 0 and Rev. 1 to co- exist

According to RG 1.183, "T*he design basis accident source term is a fundamental assumption upon which a significant portion of the facility design is based."* Given this and SAND2008-6601, how does the existence (coexistence) and continued use of the non-conservative and conceptual errors in RG 1.183 benefit the health and safety of the public?

Revised Fuel Handling Accident

Revisited the original studies forming the technical basis for the FHA and incorporate updated information.

Model improvements established from the current understanding of reactor fuel pin physics and iodine chemistry under the environmental conditions in which fuel handling operations are taking place.

Concluded that considerable margin exists regarding the scrubbing effects of iodine in the spent fuel or reactor pool and that the current staff DBA FHA fission product transport model can be refined while still maintaining conservatism.

Reference: Memo from RES to NRR, "Closeout to Research Assistance Request for Independent Review of Regulatory and Technical Basis for Revising the Design-basis Accident Fuel Handling Accident," November 23, 2019 (ML19270E335)

While there may be margin regarding the scrubbing effects of iodine, there is no margin regarding the release of Kr-85 in a DBA FHA. Please consider DBA FHA doses to control room operators and extrapolate local area doses. No amount of water in spent fuel pools or the reactor pools, will shield or prevent the release of a nobel gas (Kr-85) in a DBA FHA (or other accidents that cause mechanical or overheating damage in these pools).

Consideration of "*KR-85 Hazards From Decayed Fuel*" (Information Notice No. 90-08) is conspicuously missing from RG 1.183 Revision 0. It should be included in any revision.

Over the last 10 years no applicant or licensee has adopted the methodology from SAND2008-6601, "Analysis of Main Steam Isolation Valve Leakage in Design Basis Accident Using MELCOR 1.8.6 and RADTRAD."

There have been no communications that applicants or licensees intend to adopt the SAND2008-6601 methodology.

SAND2008-6601 clearly explains/illustrates that RG 1.183 BWR MSIV source

terms and metrologies are "non-conservative and conceptually in error." It identifies a safety concern (with a complex array of regulatory implications); however, this concern was not enough to motivate nuclear power plant owners/operators to adopt SAND2008-6601 or otherwise correct the non-conservative errors in RG 1.183—that adversely affect the health and safety of the public. This is the crux of the matter and the reason for PRM-50-122.

From: Brian Magnuson
Sent: Wednesday, November 4, 2020 10:31 PM
To: Smith, Micheal
Cc: Blumberg, Mark; Meighan, Sean
Subject: Re: Regulatory Guide 1.183 Revision Public Meeting Notice

Micheal/Mark:

I appreciate the notification and plan to attend.

Thank you,
Brian

> On Nov 4, 2020, at 10:33, Smith, Micheal <Micheal.Smith@nrc.gov> wrote:
>
> Hello,
>
> My name is Micheal Smith and I am currently the project lead for the revision of Regulatory Guide 1.183. Mark Blumberg (project technical lead) informed me that you might be interested in the revision of RG 1.183 so I am reaching out to inform you that we have a public meeting scheduled for November 19th from 1pm -4pm EST. The link to the public meeting notice is below.
>
> https://www.nrc.gov/pmns/mtg?do=details&Code=20201297
>
> Enjoy the rest of your week!
>
> <image001.jpg>
>
> Micheal Smith

Health Physicist and Assistant Radiation Safety Officer
Radiation Protection and Consequence Branch
Division of Risk Assessment
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
301-415-3763

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

June 25, 2018

MEMORANDUM TO: Brian E. Thomas, Director
Division of Engineering
Office of Nuclear Regulatory Research

FROM: Michael X. Franovich, Director ***/RA/***
Division of Risk Assessment
Office of Nuclear Reactor Regulation

SUBJECT: RESULTS OF PERIODIC REVIEW OF REGULATORY GUIDE 1.183

This memorandum documents the U.S. Nuclear Regulatory Commission's (NRC's) periodic review of Regulatory Guide (RG) 1.183, Revision 0, "Alternative Radiological Source Terms for Evaluating Design-Basis Accidents at Nuclear Power Reactors," published July 2000 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML003716792). RG 1.183 provides guidance to licensees of light water power reactors on acceptable applications of alternative source term (ASTs); the scope, nature, and documentation of associated analyses and evaluations; consideration of impacts on analyzed risk; and content of submittals. This guide also identifies acceptable radiological analysis assumptions for use in conjunction with the accepted AST. As discussed in Management Directive 6.6, "Regulatory Guides," the NRC staff reviews RGs approximately every five years to ensure that the RGs continue to provide useful guidance. Documentation of the Office of Nuclear Reactor Regulation staff review is enclosed.

Based on the results of the periodic review, the NRC staff concludes that a revision to RG 1.183 is warranted. Please see the enclosed periodic review for details.

Enclosure:
Regulatory Guide Periodic Review

CONTACT: W. Mark Blumberg, NRR/DRA
301-415-1083

SUBJECT: RESULTS OF PERIODIC REVIEW OF REGULATORY GUIDE 1.183
DATED: June 25, 2018

DISTRIBUTION:
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MOrr	LPerkins	MBlumberg	CMoulton
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Y020180087
ADAMS Accession No.: **ML18159A069** **NRR-106**

OFFICE	**NRR/DRA/ARCB**	**NRR/DRA/ARCB**	**NRR/DRA**
NAME	MBlumberg	KHsueh	MFranovich
DATE	6/ 14 /2018	6/ 19 /2018	6/ 25 /2018

OFFICIAL RECORD COPY

REGULATORY GUIDE PERIODIC REVIEW

Regulatory Guide Number:	1.183, Revision 0
Title:	Alternative Radiological Source Terms for Evaluating Design-Basis Accidents at Nuclear Power Reactors
Office/division/branch:	NRR/DRA/ARCB
Technical Lead:	W. Mark Blumberg
Staff Action Decided:	Revise

1. What are the known technical or regulatory issues with the current version of the Regulatory Guide?

Regulatory Guide (RG) 1.183, Revision 0, “Alternative Radiological Source Terms for Evaluating Design-Basis Accidents at Nuclear Power Reactors,” published July 2000 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML003716792), provides guidance to licensees of light water power reactors on acceptable applications of alternative source terms (ASTs); the scope, nature, and documentation of associated analyses and evaluations; consideration of impacts on analyzed risk; and content of submittals. This guide establishes an acceptable AST and identifies the significant attributes of other ASTs that may be found acceptable by the U.S. Nuclear Regulatory Commission staff. This guide also identifies acceptable radiological analysis assumptions for use in conjunction with the accepted AST. Licensing applications for new light water reactor designs under Part 52 have also used the guidance in RG 1.183, as applicable to the design, to prepare the design- basis accident (DBA) radiological consequences analyses in the design certification application final safety analysis report (FSAR). In addition, Part 52 combined license application FSAR DBA radiological consequence analyses have referenced design certification analyses which used RG 1.183 as guidance.

The known technical and regulatory issues are addressed in a draft revision to RG 1.183 issued for public comment (Draft Guide (DG)-1199, “Alternative Radiological Source Terms for Evaluating Design-Basis Accidents at Nuclear Power Reactors,” published October 2009 (ADAMS Accession No. ML090960464)). The main technical issues are addressed in Regulatory Position (RP) 3.2, “Release Fractions,” RP 5.3, “Meteorology Assumptions,” and RP A-5, “Main Steam Isolation Value Leakage in Boiling Water Reactors.” In addition the following editorial issues have been identified since the issuance of DG-1199. When Draft Guide-1199 is finalized the revision should include deleting the references to several RGs that have been withdrawn.

2. What is the impact on internal and external stakeholders of <u>not</u> updating the RG for the known issues, in terms of anticipated numbers of licensing and inspection activities over the next several years?

For operating reactors, the staff anticipates approximately 20-25 licensing activities per year (e.g., extended power uprates, AST amendments, etc.) involve the use of RG 1.183, Revision 0. For new reactors, the staff anticipates to be reviewing 1-2 applications in the next several years, which may involve the use of RG 1.183. Therefore, the guidance is being updated for the known issues with RG 1.183. Please see the response to Question 5 for the conceptual plan and the timeframe to address these known issues.

3. What is an estimate of the level of effort needed to address identified issues in terms of full-time equivalent (FTE) and contractor resources?

The FTE to finalize DG-1199 is expected to be 0.30 FTE. This estimate is based upon the assumption that revisions to DG-1199 due to existing public comments do not require the guide to be re-issued for further public review and additional comments.

4. Based on the answers to the questions above, what is the staff action for this guide (Reviewed with no issues identified, Reviewed with issues identified for future consideration, Revise, or Withdraw)?

Revise.

5. Provide a conceptual plan and timeframe to address the issues identified during the review.

The resolution of public comments and issuance of the revision to RG 1.183 is expected prior to second quarter of calendar year 2019.

NOTE: This review was conducted in June 2018 and reflects the staff's plans as of that date. These plans are tentative and subject to change.

June 11, 2009

MEMORANDUM TO: James J. Shea
Senior Reactor Engineer
Engineering Review Branch 2
Division of License Renewal

FROM: Mark A. Cunningham, Director **/RA/**
Division of Risk Assessment

SUBJECT: RESPONSE TO A NON-CONCURRENCE ON DRAFT REGULATORY GUIDE DG-1199, "ALTERNATIVE RADIOLOGICAL SOURCE TERMS FOR EVALUATING DESIGN BASIS ACCIDENTS AT NUCLEAR POWER REACTORS"

On April 14, 2009, you outlined reasons for non-concurring on the subject Draft Guide (DG-1199) that provides methods for modeling the radiological consequences of design basis accidents. I appreciate that you have taken time to provide your concerns and to document your views. It is my understanding that you have previously raised these concerns to your immediate supervisor and that there have been numerous meetings and discussions on these issues. Your experience, views, and efforts to share these views during the development of DG-1199 are important to the NRC's mission to protect the health and safety of the public.

You have chosen Management Directive 10.158 (MD 10.158) entitled, "NRC Non-Concurrence Process," to provide your views and concerns. MD 10.158 directs a formal response to your non-concurrence submittal. Attachment 1 to this letter contains background information on DG-1199. Attachment 2 is a summary of your concerns and a response to these concerns. Attachment 3 is your original non-concurrence submittal. This letter responding to your non-concurrence submittal will be included as part of the concurrence package for DG-1199.

It is the policy of the Nuclear Regulatory Commission (NRC) to maintain a working environment that encourages employees to make known their best professional judgments even though they may differ from the prevailing staff view, disagree with a management decision or policy position, or take issue with a proposed or established agency practice involving technical, legal or policy issues. The NRC management values each staff member's view and encourages staff to express those views.

This response to your non-concurrence submittal concludes efforts to address your concerns through the non-concurrence process. At this time, I have decided not to make any changes to DG-1199 based on your comments. However, I have instructed the Accident Dose Branch Chief to keep a record of your concerns and to reevaluate whether any changes are needed to DG-1199 once public comments are received. If you feel that your concerns were not resolved in an appropriate manner, the NRC's differing professional opinion process is available to pursue your concerns. The differing professional opinion process is documented in Management Directive 10.159, "The NRC Differing Professional Opinions Program."

Enclosures:
As stated

Background on DG-1199

Introduction

In the early 1970s, the Nuclear Regulatory Commission (NRC) staff issued Regulatory Guides (RGs) 1.3 and 1.4 for evaluating the radiological consequences of design basis accidents (DBAs). RGs 1.3 and 1.4 use the radiological source term described in TID-14844, "Calculation of Distance Factors for Power and Test Reactor Sites."

Since the publication of TID-14844 in 1962, significant advances have been made in the understanding of radioactivity released from severe nuclear power plant accidents. In 1995, the NRC published NUREG-1465, "Accident Source Terms for Light-Water Nuclear Power Plants." NUREG-1465 uses updated research from the 1980's that provides a more realistic estimate of the accident source term, including its mix, magnitude, chemical and physical form, and timing of release.

The NRC staff anticipated that some licensees, who used TID-14844 to design their facilities, may wish to update their design bases using the NUREG-1465 source term to take advantage of the more realistic information it provides. The NRC staff, therefore, initiated several actions to provide a regulatory basis for these licensees to use an alternative source term (AST) in design basis analyses. These initiatives resulted in the development and issuance of Title 10 of the Code of Federal Regulation (10 CFR) Section 50.67 (50.67), "Accident source term."

10 CFR 50.67

The NRC, via regulations such as the performance-based 10 CFR 50.67, regulates all U.S. commercial nuclear power plants. 10 CFR 50.67 is an alternative voluntary regulation that allows licensees to revise the accident source term. This source term is used in the radiological analyses for designing their plant. This analysis is often referred to as a "design basis" analysis and the hypothetical or postulated events used to test the facility are known as "design basis accidents" (DBAs).

10 CFR 50.67 provides requirements on the acceptable dose limits from the design basis analyses and the assumption that the fission product release, assumed for these calculations, be based upon a major accident that is historically taken to involve a substantial core melt. The regulatory approach of using design basis accidents and applying performance based regulatory requirements is consistent with the approach provided in other NRC regulations such as 10 CFR 50.46, "Acceptance criteria for emergency core cooling systems for light-water nuclear power reactors," and 10 CFR 50.65 "Requirements for monitoring the effectiveness of maintenance at nuclear power plants."

When 10 CFR 50.67 was codified, the NRC intentionally did not include any reference to NUREG-1465. This is consistent with the NRC regulatory philosophy and the staff's desire to allow changes to the defined source term or the development of other technically sound source term estimates without requiring additional rulemaking. Instead of codifying NUREG-1465 in 10 CFR 50.67, the NRC staff used NUREG-1465 and other technical information to develop RG 1.183, Revision 0, as one methodology acceptable to the staff for complying with 10 CFR 50.67. This has provided the NRC and nuclear industry with the flexibility to consider and incorporate new research and technical advancements without having to conduct rulemaking.

Regulatory Guide 1.183

In July 2000, the NRC staff issued RG 1.183, Revision 0 which provides one method acceptable to the NRC staff for complying with the regulatory requirements contained in 10 CFR 50.67. As with all RGs, RG 1.183 is not a regulatory requirement and, therefore, licensees may propose, and the NRC may approve, alternative methodologies which have a sound technical and scientific basis to demonstrate that the regulatory requirements of 10 CFR 50.67 are satisfied. RG 1.183 simply provides one set of acceptable assumptions and parameters that licensees can use to calculate postulated radiological doses for light-water reactor (LWR) DBAs.

Since the initial issuance of RG 1.183, the NRC staff and the commercial nuclear industry both have gained substantial experience with the implementation of 10 CFR 50.67 and RG 1.183. Based on this experience and on specific feedback and comments from licensees, the anticipation of licensing advanced LWRs, and new research, the NRC is proposing to update RG 1.183. Draft Regulatory Guide 1199 (DG-1199) is the proposed Revision 1 to RG 1.183. In accordance with the NRC's regulatory processes, the staff is proceeding with soliciting internal and external stakeholder feedback through appropriate mechanisms such as Advisory Committee for Reactor Safeguards reviews and a public comment period.

Boiling Water Reactor (BWR) Main Steam Line Leakage Pathway

Research was initiated to determine whether updates to the RG 1.183 BWR main steam isolation valve leakage (MSIV) modeling methodologies were warranted. A discussion of the MSIV pathways follows.

BWRs operate by boiling water in direct contact with the reactor fuel rods and passing this steam directly through the power turbines by means of large main steam lines. Because steam lines could provide a potential direct release pathway from the core to the environment, two quick closing safety-related main steam line isolation valves (MSIVs) were included in the original design. The MSIVs on each steam line isolate the containment boundary from the environment in the event of a core damage accident. This isolation accomplishes a critical safety function of mitigating the release of fission products.

Because MSIVs are not leak tight, acceptable leakage limits were established and incorporated into the plant design and technical specifications in accordance with 10 CFR 50.36, "Technical specifications." Licensees periodically test MSIV leakage in accordance with established surveillance requirements to ensure the leakage remains below these limits. RG 1.183, Revision 0 provides a methodology acceptable to the NRC staff for establishing acceptable MSIV leakage limits. Specifically, it provides guidance on the radioactivity that is assumed released to the steam lines as well as methods for reducing these releases by crediting holdup and deposition in the steam line piping and in the main condenser.

In 1998, the staff performed an assessment to estimate the deposition in steam lines for the first AST application. The methods used by the NRC staff in calculation AEB 98-03, "Assessment of Radiological Consequences for the Perry Pilot Plant Application using the Revised (NUREG-1465) Source Term," (ADAMS ML011230531) have been used by many licensees to model MSIV leakage. However, in 2006, the Office of Nuclear Regulatory Regulation (RES) informed the Office of Nuclear Reactor Regulation (NRR) that the AEB 98-03 report contained some technical errors.

In response, NRR prepared a User Need and coordinated with RES to undertake additional research focused on MSIV leakage modeling. The NRC contracted with Sandia National Laboratories (SNL) to perform a reassessment of the methods in AEB 98-03 using state-of-the-art computer codes and modeling. The results of this reassessment are contained in SNL report SAND2008-6601, “Analysis of Main Steam Isolation Valve Leakage in Design Basis Accidents Using MELCOR 1.8.6 and RADTRAD.” The SNL report provides a state-of-the-art assessment indicating that a revision to the MSIV leakage assumptions in RG 1.183 should be considered.

Main Steam Line Leakage Methodology Update

DG-1199 provides improved methods based upon the SNL report to calculate DBA doses from the MSIV leakage pathway. DG-1199 proposes methods to calculate: 1) the concentration of radioactivity used as the source for the MSIV leakage and 2) deposition of radioactivity in the steam line piping and condenser.

RG 1.183, Revision 0, like its Regulatory Guide 1.3 and 1.4 predecessors, assumed that the concentration of radioactivity used as the source of the MSIV leakage is approximated by the concentration of radioactivity in the containment following a postulated core melt. The radioactivity from the core melt is assumed to mix instantaneously and homogeneously throughout the free air volume of the containment building and reactor coolant system piping. This idealized view presumes that the radioactivity is released from the fuel, transported out of the vessel, and instantaneously and homogenously distributed within the drywell volume. This equilibrated drywell atmosphere is assumed to be the source of the flow through the leaking MSIVs. In reality, the radioactivity in the vessel steam dome is not instantaneously equilibrated with the drywell or containment volume (See Figure 1).

In order to examine more realistically the behavior of radioactivity in the steam dome and containment, SNL used the MELCOR code to make best estimate predictions of the radioactivity released, the transport behavior in the vessel and containment, and the resulting leakage to the environment through leaking MSIVs. SNL used the MELCOR code because it is internationally recognized as the state-of-the-art for modeling nuclear power plant severe accidents.

The SNL report demonstrated that radioactivity transport is very different from the assumptions currently in RG 1.183. The current RG 1.183 methodology underestimates the MSIV leakage source term during the first two hours of the DBA. RG 1.183 assumes that the vessel and containment atmosphere are in equilibrium before reflood and that the containment atmosphere can be used as the source of the MSIV leakage. The technical validity of this assumption was explicitly evaluated in the SNL research. The SNL report documented findings from MELCOR code runs which showed that the concentration of radioactivity in the steam dome may be substantially higher than in containment until reflood occurs. After reflood, the vessel activity is swept into the containment and over time the vessel and containment atmosphere equilibrate. Therefore, consistent with the SNL research, DG-1199 proposes to use the concentration of radioactivity in the steam dome before reflood, and thereafter use the concentration of radioactivity in the containment.

A second proposed update modifies the amount of radioactivity deposited in the steam line. The current methodology used by the NRC is contained in AEB 98-03 discussed above.

The SNL reports showed that the AEB 98-03 methodology non-conservatively overestimated the amount of deposition of radioactivity in the steam line. One factor that contributed to the differences include more realistic steam dome, steamline and condenser modeling in the SNL report.

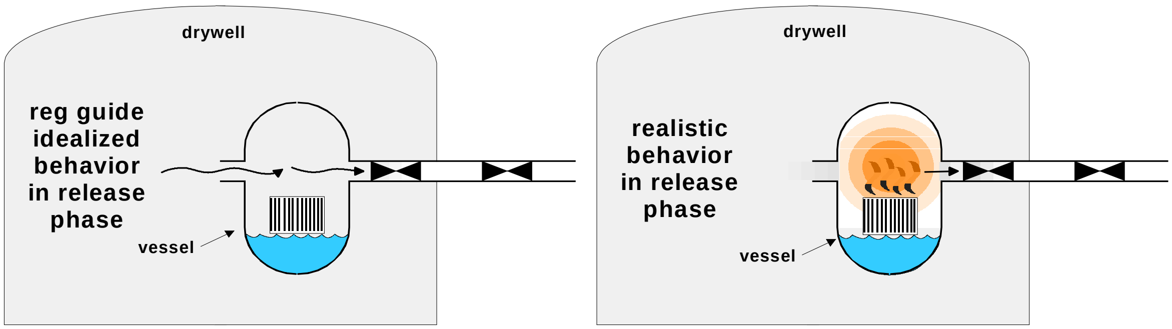

Figure 1 Idealized regulatory model of airborne fission products (left) compared to realistic prediction of airborne radioactivity (right) during release phase of a DBA with core damage. Note, in actuality the source of airborne activity emanates from vessel core (right)

Summary and Resolution of Non-concurrence Issues

Summary

Your non-concurrence submittal was reviewed and is summarized by the following issues:

Scrutinizing the proposed change using an “NRC Rule Making process” is required prior to issuing the draft regulatory guidance because it changes the accident “source term” characteristics and the application of the “source term” in design basis dose consequences analyses.

The assumptions used in the SNL analysis are not appropriate. The proposed change used a beyond design bases analysis by incorporating a MELCOR in-vessel “source term” that maximized radio-aerosol concentration coupled with deterministic assumptions to maximize dose consequence from boiling water reactors (BWR's) main steam lines (MSLs). You cited the specific issues given below:

- The proposed change used a beyond design bases analysis by incorporating a MELCOR in-vessel “source term” that maximized radio-aerosol concentration.

- Experience would show that for some or all of the first hour after a plant transient, flow may actually be into the vessel from residual steam from the turbine stop valves back to the steam dome rather than being biased out to maximize dose to the control room.

- The Sandia report clamped (normalized) the in-vessel concentration after the first hour of the accident instead of taking the analyzed best estimate data that showed the concentration in-vessel substantially declining after the first hour of the accident. This could substantially decrease the dose from the MSL pathway for the duration of the accident and may prove that the AST/TID containment “source term” would, in actuality, be more conservative over the total duration of a DBA LOCA for BWRs.

- In addition to these non-realistic biases used to develop the current guidance, the analysis did not model or assume that the BWR vessel separators and dryers would not reduce dose consequence by deposition. These assumptions were based on a PWR study that is not necessarily applicable to BWR designs.

Response to Non-Concurrence Issues

Non-Concurrence Issue

Scrutinizing the proposed change using an “NRC Rule Making process” is required prior to issuing the draft regulatory guidance because it changes the accident “source term” characteristics and the application of the “source term” in design basis dose consequences analyses.

Response

The proposed DG-1199 changes, related to the treatment of the MSIV leakage pathway source term, were scrutinized to determine if rulemaking would be required. The changes were determined to not conflict with the existing 10 CFR 50.67 rule or the 10 CFR 50.2 definition of source term.

As previously described, 10 CFR 50.67 is a performance-based regulation which allows licensees voluntarily to revise their licensing basis source term used to evaluate the consequences of applicable design basis accidents. 10 CFR 50.67 provides acceptance limits for design purposes and states:

The fission product release assumed for these calculations should be based upon a major accident, hypothesized for purposes of design analyses or postulated from considerations of possible accidental events, that would result in potential hazards not exceeded by those from any accident considered credible. Such accidents have generally been assumed to result in substantial meltdown of the core with subsequent release of appreciable quantities of fission products.

10 CFR 50.2, “Definitions,” defines “source term” as:

Source term refers to the magnitude and mix of the radionuclides released from the fuel, expressed as fractions of the fission product inventory in the fuel, as well as their physical and chemical form, and the timing of their release.

The proposed change does not change the magnitude and mix of the radionuclides in or <u>released from the fuel</u>, expressed as fractions of the fission product inventory in the fuel, or their physical and chemical form or the timing of their release. Instead, the proposed change uses state-of-the-art research to model the realistic transport of fission products generated during a core melt accident instead of solely relying on the assumption that the MSIV leakage source term can be approximated by the concentration of radioactivity in the containment. The SNL MELCOR calculations were performed using methods similar to those used to create the NUREG-1465 source term.

In accordance with the principles of performance-based regulation, 10 CFR 50.2 and 10 CFR 50.67 do not specify any particular transportation pathway from the fuel to the reactor vessel steam dome or the drywell. Therefore, no regulatory conflict exists between the proposed changes in DG-1199 and the existing rule. Discussions with both the technical author and the Office of General Counsel reviewer for the 10 CFR 50.67 rule and Regulatory Guide 1.183 indicate agreement with this assessment.

DG-1199 will be formally scrutinized consistent with Office Instruction ADM-004, Revision 2, “Regulatory Guide Development, Revision, and Withdrawal Process,” which is implemented by the Office of Nuclear Regulatory Research (RES). ADM-004 will guide the reviews and concurrences by 3 program offices (Office of Nuclear Reactor Regulation (NRR), Office of New Reactors (NRO), and RES). Additionally, NRR has specifically requested RES to request reviews of DG-1199 by the Committee to Review Generic Requirements (CRGR) and the Advisory Committee on Reactor Safeguards (ACRS) before publishing the DG-1199 for public comment.[1] Additionally, the Office of General Counsel’s (OGCs) “No Legal Objection” determination will be required prior to issuing the DG for public comment. RES will transmit your reasons for the non-concurrence of DG-1199 and my response to these concerns to all three program offices, as part of the concurrence package for DG-1199.

Non-Concurrence Issue

The Sandia MELCOR analysis inappropriately combined some realistic analyses with deterministic assumptions that result in main steamline models that may not “realistically” or appropriately model the actual dose consequences from a BWR MSL in the event of a DBA LOCA.

Response

The SNL design basis analysis calculations for the main steam line models were structured to inform the staff’s definition of an appropriately conservative set of assumptions to test the performance of the main steam line leakage pathway. The SNL researchers intentionally focused on those processes and phenomenon considered to most accurately predict and model this radionuclide transport pathway. For processes and phenomenon not explicitly modeled, the researchers selected assumptions and models that provide reasonable margin against unpredicted events in the course of an accident and to compensate for large uncertainties in facility parameters, accident progression and radioactive material transport.

It should be noted that the limits contained in 10 CFR 50.67 and RG 1.183 do not constitute acceptable limits for emergency doses to the public under accident conditions. Rather, these analyses calculate reference design doses used in the evaluation of proposed design basis changes to a nuclear power plant. They are not intended to model actual dose consequences and are meant to be intentionally conservative in order to address uncertainties in accident progression, fission product transport and atmospheric dispersion.

[1] Memo transmitting DG-1199 to the Office of Research from Mark Cunningham to Michael Case, entitled “Transmittal of Draft Regulatory Guide DG-1199,” dated March 26, 2009 (ML090050330).

Non-Concurrence Issue

The proposed change used a beyond design bases analysis by incorporating a MELCOR in-vessel “source term” that maximized radio-aerosol concentration coupled with deterministic assumptions to maximize dose consequence from boiling water reactors (BWR's) main steam lines (MSLs).

Response

SNL did not maximize concentrations of radioactivity in the steam dome. SNL used state-of-the-art MELCOR models to calculate the realistic concentrations of radioactivity in the steam dome for several classes of radionuclides. The SNL staff abandoned the approach of using the highest concentration of all radionuclide classes because it significantly overestimated the concentration of radioactivity in the steam dome. Instead, the SNL analysis used the best estimate concentrations of iodine, cesium, and strontium to determine the ratio of radioactivity in the steam dome to the radioactivity in the containment. The state-of-the-art MELCOR code models a realistic sequence driven fission product release with consistent thermal-hydraulics aerosol mechanics and other physics-based models to predict best estimate source terms and transport behavior. These models, therefore, do not reflect a maximized in-vessel source term.

Non-Concurrence Issue

Experience would show that for some or all of the first hour after a plant transient, flow may actually be into the vessel from residual steam from the turbine stop valves back to the steam dome rather than being biased out to maximize dose to the control room.

Response

The MELCOR models actually predict and demonstrate for short periods of time that flow at the inboard MSIV would oscillate between downstream toward the condenser and upstream back into the reactor vessel due to the thermal-hydraulic conditions within the main steam lines. The predicted flow is dependent upon several parameters including the design of non-safety related main steam line piping, thermal contact with the containment, and the amount of insulation present. These parameters can vary between plants and were either not practical to model in the SNL analysis or were not modeled. The phenomenon is also dependent upon the accident scenario.

Given the many uncertainties of parameters that can impact the main steam line flow, flow outward was promoted in the calculations by reducing the heat transfer coefficients of the heat structures of the main steam line piping between the steam dome and the inboard MSIV for one hour. This modeling assumption did not stop the oscillations but used a conservative method to model the flow in light of uncertainties with the heat transfer coefficients, MSIV closure times and differences in BWR designs. This approach is consistent with other DBA models where it is either not possible or impractical to directly model each individual phenomenon or process.

Non-Concurrence Issue

The Sandia report clamped (normalized) the in-vessel concentration after the first hour of the accident instead of taking the analyzed best estimate data that showed the concentration in-vessel substantially declining after the first hour of the accident. This could substantially decrease the dose from the MSL pathway for the duration of the accident and may prove that the AST/TID containment "source term" would, in actuality, be more conservative over the total duration of a DBA LOCA for BWRs.

Response

A significant finding of the SNL work is that during the first two hours of a LOCA the concentration of radioactivity in the steam dome is significantly greater than that in the drywell. Since the steam lines are connected to the steam dome and not the containment, it is the concentration of radioactivity in the steam dome that should be used to determine the design bases for the MSIV leakage pathway.

Consistent with the historical modeling of the LOCA for 50.67, reflood is assumed to occur at two hours. Upon reflood, some of the activity in the vessel will be swept into the containment. Over time, the containment and steam dome are expected to come to an equilibrium concentration of radioactivity.

SNL analyzed MELCOR results to develop the MSIV leakage pathway methodology contained in DG-1199. The MSIV leakage pathway methodology includes a method for modeling the concentration of radioactivity available for release. SNL analyzed the concentrations of radioactivity in the steam dome and containment during the accident. Based upon this information, SNL determined that the concentration of radioactivity in the steam dome was comparable to that of the containment after the first hour and, therefore, it was appropriate to assume it is equal to the concentration in containment after the first hour of the accident. SNL used the following information to justify this assumption.

- The ratio of radioactivity in the steam dome to the radioactivity in the containment varies significantly over time. The steam dome-to-containment concentration ratios for Cesium and Iodine in the Mark-III recirculation line break case are somewhat greater than 1 during the period from 1 to 2 hours (the range is from 100 to 0.5). Despite the ratio of greater than one during the time period from 1 to 2 hours, the SNL methodology assumes the steam dome-to-containment concentration ratio is 1 for times greater than 1 hour. This underestimation of the concentration of radioactivity in the steam dome was done to compensate for times after reflood where the steam dome-to-containment concentration ratio might be less than one and to simplify the analyses licensees would have to perform in order to apply this research.

- Computationally, MELCOR cases that model reflood are very resource intensive and SNL could only one run one case. The one case that modeled core reflood modeled the scenario to 10 hours. Once computational and input issues were resolved, the time to run the case was greater than one week. Therefore, limited data exists for post reflood conditions. For the one SNL reflood case, the steam dome-to-containment concentration ratio varies after reflood, but within 2 hours approaches a value of one (approximately 0.4).

- The steam dome concentrations during the first hour of the accident contribute more to dose than the containment concentrations after reflood. Based upon Figure 2-44 of the SNL report, the impact of the pre-reflood concentration of radioactivity in the steam dome on dose is 200 times greater than the post-reflood concentration. The post-reflood drywell concentration contributes less than 0.5% to dose. Therefore, the most significant impact on dose is due to the concentrations before reflood.

Non-Concurrence Issue

In addition to these non-realistic biases used to develop the current guidance, the (SNL) analysis did not model or assumed that the BWR vessel separators and dryers would not reduce dose consequence by deposition.

These assumptions were based on a PWR study that is not necessarily applicable to BWR designs.

Response

The steam separator and dryer are modeled in the MELCOR deck. MELCOR predicted deposition on these surfaces and a corresponding reduction in the concentration of radioactivity available to be released from the steam dome.

Summary of Responses

In summary, the proposed changes have been thoroughly reviewed and are found to be consistent with existing NRC rules and practices. The NRC's internal processes, reviews and concurrences discussed above will formally confirm this assessment prior to issuing DG-1199 for public comments.

Jim Shea's Non-concurrence

The proposed Alternate Source Term (AST) Regulatory Guide (RG) 1.183 Revision 1 incorporates research that has not been appropriately processed by NRC "Rule Making" given the change to the accident "source term" characteristics and application in design basis accident (DBA) dose consequence analysis. In addition the proposed change used a beyond design bases analysis by incorporating a MELCOR in-vessel "source term" that maximized radio-aerosol concentration coupled with deterministic assumptions to maximize dose consequence from boiling water reactors (BWR's) main steam lines (MSLs). Finally, the MELCOR realistic / deterministic approach for DBA dose consequence in BWRs does not meet the NRC core values of Efficiency, Clarity and Reliability.

<u>Accident Source Term Regulatory Basis</u>

In Title 10 of the Code of Federal Regulations (10 CFR) Part 100 "Siting Criteria" The "source term" has been used as a bounding deterministic (non-realistic) release of radioactivity from the core to containment in order to design and test Engineered Safety Features (containment) for reactor plant siting requirements. For almost 50 years starting with The Technical Information Document (TID) 14844," Calculation of Distance Factors for Power and Test Reactror Sites;" a bounding deterministic containment "source term" has been applied in DBA dose consequence analysis for 10 CFR 100.11, "Determination of exclusion area, low population zone, and population center distance." The TID states on page 12 that, "The objective of estimating the radioactive inventory within the outer containment barrier is to attain a starting point for calculating the potential radiological hazard in the surrounding environs. For people in the proximity of the reactor building, factors such as the physical nature of the material leaking from **the containment vessel**, release height, particle deposition with distance, wind direction, speed and variability, and air temperature gradients become important in determining the extent of these potential hazards. It is from this complexity of interwoven technical parameters that the values for the exclusion area, low population zone and population center distance must be determined."

Since the publication of TID-14844, significant advances have been made in understanding the timing, magnitude, and chemical form of fission product releases. In 1995, the NRC published NUREG-1465, "Accident Source Terms for Light-Water Nuclear Power Plants." NUREG-1465 used updated research to provide more realistic estimates of the containment accident source term that were physically based and that could be applied to the design of future light-water power reactors. The NRC staff also determined that some current licensees may wish to use the NUREG-1465 source term referred to as the AST in analyses to support cost-beneficial licensing actions. The NRC staff, therefore, initiated several actions to provide a regulatory basis for operating reactors to use an AST in design basis analyses. These initiatives resulted in the development and issuance of 10 CFR 50.67, "Accident Source Term" and RG 1.183 (July, 2000)

In the NRC rule making for the AST the fundamental containment source term derived for the maximum credible accident was not changed from the TID concept. The change that provided licensees with some relief to the TID source term is principally due to the change in the timing and chemical make-up of the containment “source term” which in the TID assumed that the release was instantaneous and that 50% of the radioiodine’s were available for release to the environs. The AST did not supplant these licensing bases fundamental accident source term assumptions but rather refined the bounding containment “source term” concept noting that the TID may be more conservative than realistically necessary for licensing purposes.

The licensing bases concepts of a deterministic bounding containment “source term” is imbedded in the foot-notes to 10 CRR 100.11 and 10 CFR 100.67 which states, “The fission product release assumed for these calculations should be based upon a major accident, hypothesized for purposes of design analyses or postulated from considerations of possible accidental events, that would result in potential hazards not exceeded by those from any accident considered credible [maximum credible accident]. Such accidents have generally been assumed to result in substantial meltdown of the core with subsequent release of appreciable quantities of fission products.

In the TID/AST deterministic approach the containment “source term” was conceived and applied in light water reactor licensing to encompass any possible accident by assuming a non-realistic radioactive release from the reactor core to the containment. This was intended to provide a bounding analysis for plant siting purposes.

Purpose for Sandia Research

The research done by Sandia originated from a NRR/DRA/AADB user need dated June 15, 2007 (No response returned as of the time of this writing) to confirm or determine if changes were needed to the NRC staffs radioactive aerosol settling and deposition methodologies used to asses the accident dose contribution from BWR MSL TS leakages. The NRC staff has been using a conservative application of AEB 98-03, “Assessment of Radiological Consequences for the Perry Pilot Plant Application using the Revised (NUREG-1465) Source Term” for determining radio-aerosol deposition on MSL’s for BWRs. In a staff briefing of the results of the Sandia findings based on the MELCOR modeling of the Peach Bottom (PB) plant, it was reported that the MELCOR analysis did confirm that the conservative application of AEB 98-03 was in line with findings from the MELCOR analysis when a containment “source term” is applied for MSL deposition. In addition the Sandia results showed that when the Condenser is credited for hold-up and deposition the dose contributor for the DBA LOCA is practically inconsequential.

User Need Change in Focus

At some point following the Research User Need request it was determined that a containment “source term” from TID/AST was not appropriate for BWR MSL analysis, this conclusion resulted in a new in-vessel “source term” that predicts for the first hour of a LOCA a significantly higher radio-nuclide concentration in-vessel than that found in the containment “source term.”

The application of the new Sandia MELCOR in vessel “source term” analysis combined with the already licensed AST/TID containment “source term” adds additional conservatisms to the licensing bases of BWRs that have not properly been scrutinized in an NRC Rule Making process.

Sandia MELCOR in-vessel "source term"

The MELCOR code could be a valuable tool for a PRA based safety and consequence analysis similar to what has been advocated by the International Atomic Energy Agency (IAEA) Safety Guide on Deterministic Safety Analysis for Nuclear Power Plants (DS 395). The IAEA draft Safety Guide describes the technological shift from a deterministic [AST/TID] safety analyses (past practices) which used rigorous conservative approaches to a more realistic approach (current preferred practice) together with an evaluation of uncertainties or a best estimate analysis.

In the RG update the MELCOR PB model used to develop the proposed dose consequence analysis procedure for BWR MSLs mixed the deterministic bounding core melt parameters with so called realistic core dynamics associated with a DBA LOCA. This modeling of the in–vessel "source term" generated by MELCOR was scaled up to meet the AST concentration values required for a full core melt in NUREG-1465. In addition the MSL flow was biased to ensure flow was always out of the MSL and not into the steam dome in the event of an actual DBA LOCA. Experience would show that for some or all of the first hour after a plant transient flow may actually be in-to the vessel from residual steam from the turbine stop valves back to the steam dome rather than being biased out to maximize dose to the control room. Also the Sandia report clamped (normalized) the in-vessel concentration after the first hour of the accident instead of taking the analyzed best estimate data that showed the concentration in-vessel substantially declining after the first hour of the accident. This could substantially decrease the dose from the MSL pathway for the duration of the accident and may prove that the AST/TID containment "source term" would, in actuality, be more conservative over the total duration of a DBA LOCA for BWRs. In addition to these non-realistic biases used to develop the current guidance, the analysis did not model or assumed that the BWR vessel separators and dryers would not reduce dose consequence by deposition. These assumptions were based on a PWR study that is not necessarily applicable to BWR designs.

It appears that the Sandia MELCOR analysis combined some realistic analysis with additional deterministic assumptions that result in a PB MSL model that may not "realistically" or appropriately model the actual dose consequence from a BWR MSL in the event of a DBA LOCA. During an inquiry of how this would affect the recently approved PB AST using the conservative application of AEB 98-03 for radio-aerosol deposition, I was told that the dose to the CR operator would increase a magnitude of approximately 6 times the currently approved value of close to the allowed limit of 5 rem TEDE.

NRC core values of Efficiency, Clarity and Reliability

The proposed changes to RG 1.183 as presented in revision 1 Appendix A-5 and the associated MELCOR model described in Reference A-10 creates a in-vessel accident "source term" that has no bases in current regulations. In our staff's review of the IAEA draft guidance discussed above AADB concluded that "Best-estimate safety analyses, as described and endorsed by the IAEA draft Safety Guide utilizing up to date computer modeling [such as MELCOR] and PRA techniques to show compliance with reactor siting and control room habitability regulations have not been done by US licensees. Moving from a conservative deterministic analysis in these areas may require revision of the current NRC regulations and regulatory guidance."

In addition to the lack of clear regulatory precedence for this new proposed in-vessel source term, the model presented in RG 1.183 rev 1Appendix A-5 for BWR MSLs used the best estimate code MELCOR as a tool to add un-realistic assumptions and requirements on the leakage path to maximize dose from a BWR MSL that is beyond design bases. In addition the research seemed to bias certain potential realistic parameters that may have provided better insight into the actual predicted dose consequence from this pathway in the event of a DBA LOCA.

The research also confirmed what we had originally asked in the AADB user need. It was shown that the NRC staff practice of a conservative application of the AEB deposition model was appropriate when used with the deterministic AST/TID containment "source term." Additionally the research showed that when crediting the main condenser the dose consequence from BWR MSL's is inconsequential.

Finally the result of the inclusion of a in-vessel "source term" for BWR's would force a licensee who wanted to perform an AST analysis in accordance with 10 CFR 50.67, into a research type evaluation of the dose consequences for the MSL contribution to the DBA LOCA analysis. This analysis was already difficult under AEB 98-03, now a more expensive and laborious process has been created by this RG revision and by definition of its overly conservative beyond current licensing basis requirements. When applied to the recently approve PB AST the results show that most if not all currently approved BWR AST's would fail to meet the CR dose acceptance criteria of 5 Rem TEDE. The proposed RG draft incorporating the Sandia MELCOR in-vessel accident "source term" for BWRs clearly does not meet the agencies goal of efficient, clear and reliable regulation.

Members of the DRA/AADB staff had presented management with an alternative to wholesale incorporation of the new BWR "in-vessel" source term developed by Sandia that was more appropriate to the current licensing structure. Questions were also raised as to why the division of NEW Reactors is not incorporating this research or using this Regulatory Guide for new-reactors as was originally planned. These questions as well as many questions that were sought concerning the details of the Sandia research including the effect on currently approved AST approvals, the affect on the few remaining non-AST BWR's, and Back-fit concerns have not been addressed adequately to justify rushing this regulatory guide through the approval process.

James J. Shea
Senior Reactor Engineer on Detail in DLR

NRC FORM 757
NRC MD 10.158
(3-2009)

U.S. NUCLEAR REGULATORY COMMISSION

NON-CONCURRENCE PROCESS

SECTION A - TO BE COMPLETED BY NON-CONCURRING INDIVIDUAL

TITLE OF DOCUMENT **DRAFT REGULATORY GUIDE DG-1199 (Proposed Revision 1 of RG 1.183)**	ADAMS ACCESSION NO. **ML090570555**
DOCUMENT SPONSOR **Mark ~~Blumberg~~** Cunningham	SPONSOR PHONE NO. **301-415-~~1083~~** 2884
NAME OF NON-CONCURRING INDIVIDUAL **James Shea**	PHONE NO. **301-415-1388**

☑ DOCUMENT AUTHOR ☑ DOCUMENT CONTRIBUTOR ☐ DOCUMENT REVIEWER ☐ ON CONCURRENCE

TITLE **Senior Reactor Engineer**	ORGANIZATION **NRR/DRA/AADB currently on detail in DLR**

REASONS FOR NON-CONCURRENCE

See Attached Word Document

☐ CONTINUED IN SECTION D

SIGNATURE

DATE 4/14/09

SUBMIT FORM TO DOCUMENT SPONSOR AND COPY TO YOUR IMMEDIATE SUPERVISOR AND DIFFERING VIEWS PROGRAM MANAGER

NON CONCURRENCE PROCESS

The proposed Alternate Source Term (AST) Regulatory Guide (RG) 1.183 Revision 1 incorporates research that has not been appropriately processed by NRC "Rule Making" given the change to the accident "source term" characteristics and application in design basis accident (DBA) dose consequence analysis. In addition the proposed change used a beyond design bases analysis by incorporating a MELCOR in-vessel "source term" that maximized radio-aerosol concentration coupled with deterministic assumptions to maximize dose consequence from boiling water reactors (BWR's) main steam lines (MSLs). Finally, the MELCOR realistic / deterministic approach for DBA dose consequence in BWRs does not meet the NRC core values of Efficiency, Clarity and Reliability.

Accident Source Term Regulatory Basis

In Title 10 of the Code of Federal Regulations (10 CFR) Part 100 "Siting Criteria" The "source term" has been used as a bounding deterministic (non-realistic) release of radioactivity from the core to containment in order to design and test Engineered Safety Features (containment) for reactor plant siting requirements. For almost 50 years starting with The Technical Information Document (TID) 14844," Calculation of Distance Factors for Power and Test Reactror Sites;" a bounding deterministic containment "source term" has been applied in DBA dose consequence analysis for 10 CFR 100.11, "Determination of exclusion area, low population zone, and population center distance." The TID states on page 12 that, "The objective of estimating the radioactive inventory within the outer containment barrier is to attain a starting point for calculating the potential radiological hazard in the surrounding environs. For people in the proximity of the reactor building, factors such as the physical nature of the material leaking from **the containment vessel**, release height, particle deposition with distance, wind direction, speed and variability, and air temperature gradients become important in determining the extent of these potential hazards. It is from this complexity of interwoven technical parameters that the values for the exclusion area, low population zone and population center distance must be determined."

Since the publication of TID-14844, significant advances have been made in understanding the timing, magnitude, and chemical form of fission product releases. In 1995, the NRC published NUREG-1465, "Accident Source Terms for Light-Water Nuclear Power Plants." NUREG-1465 used updated research to provide more realistic estimates of the containment accident source term that were physically based and that could be applied to the design of future light-water power reactors. The NRC staff also determined that some current licensees may wish to use the NUREG-1465 source term referred to as the AST in analyses to support cost-beneficial licensing actions. The NRC staff, therefore, initiated several actions to provide a regulatory basis for operating reactors to use an AST in design basis analyses. These initiatives resulted in the development and issuance of 10 CFR 50.67, "Accident Source Term" and RG 1.183 (July, 2000)

In the NRC rule making for the AST the fundamental containment source term derived for the maximum credible accident was not changed from the TID concept. The change that provided licensees with some relief to the TID source term is principally due to the change in the timing and chemical make-up of the containment "source term" which in the TID assumed that the release was instantaneous and that 50% of the radioiodine's were available for release to the environs. The AST did not supplant these licensing bases fundamental accident source term assumptions but rather refined the bounding containment "source term" concept noting that the TID may be more conservative than realistically necessary for licensing purposes.

The licensing bases concepts of a deterministic bounding containment "source term" is imbedded in the foot-notes to 10 CRR 100.11 and 10 CFR 100.67 which states, "The fission product release assumed for these calculations should be based upon a major accident, hypothesized for purposes of design analyses or postulated from considerations of possible accidental events, that would result in potential hazards not exceeded by those from any accident considered credible [maximum credible accident]. Such accidents have generally been assumed to result in substantial meltdown of the core with subsequent release of appreciable quantities of fission products.

In the TID/AST deterministic approach the containment "source term" was conceived and applied in light water reactor licensing to encompass any possible accident by assuming a non-realistic radioactive release from the reactor core to the containment. This was intended to provide a bounding analysis for plant siting purposes.

Purpose for Sandia Research

The research done by Sandia originated from a NRR/DRA/AADB user need dated June 15, 2007 (No response returned as of the time of this writing) to confirm or determine if changes were needed to the NRC staffs radioactive aerosol settling and deposition methodologies used to asses the accident dose contribution from BWR MSL TS leakages. The NRC staff has been using a conservative application of AEB 98-03, "Assessment of Radiological Consequences for the Perry Pilot Plant Application using the Revised (NUREG-1465) Source Term" for determining radio-aerosol deposition on MSL's for BWRs. In a staff briefing of the results of the Sandia findings based on the MELCOR modeling of the Peach Bottom (PB) plant, it was reported that the MELCOR analysis did confirm that the conservative application of AEB 98-03 was in line with findings from the MELCOR analysis when a containment "source term" is applied for MSL deposition. In addition the Sandia results showed that when the Condenser is credited for hold-up and deposition the dose contributor for the DBA LOCA is practically inconsequential.

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James J. Shea
Senior Reactor Engineer on Detail in DLR

NRC FORM 757
NRC MD 10.158
(3-2009)

U.S. NUCLEAR REGULATORY COMMISSION

NON-CONCURRENCE PROCESS

TITLE OF DOCUMENT	ADAMS ACCESSION NO.
Draft Regulatory Guide DG-1199 (Proposed Revision 1 of RG 1.183)	ML090570555

SECTION B - TO BE COMPLETED BY NON-CONCURRING INDIVIDUAL'S SUPERVISOR
(THIS SECTION SHOULD ONLY BE COMPLETED IF SUPERVISOR IS DIFFERENT THAN DOCUMENT SPONSOR.)

NAME
Robert Taylor

TITLE	PHONE NO.
Branch Chief	301-415-3172

ORGANIZATION
NRR/DRA/AADB

COMMENTS FOR THE DOCUMENT SPONSOR TO CONSIDER

- [ ] I HAVE NO COMMENTS
- [x] I HAVE THE FOLLOWING COMMENTS

As Mr. Shea's supervisor and the document sponsor, I greatly appreciate the contributions made by Mr. Shea during the development of DG-1199 and encourage him to continue to voice his concerns so that they may be considered. The concerns identified in his non-concurrence were raised during the development of the DG-1199 and numerous meetings were held to explore them. I solicited input on his technical and regulatory concerns from the subject matter experts in the Offices of Nuclear Regulatory Research, Nuclear Reactor Regulation, and General Counsel. In addition, his technical concerns were provided to Sandia National Laboratories to ensure that they had been appropriately explored and assessed during the development of the research. Mr. Shea was provided responses to these concerns during two distinct internal formal comment periods on the research and draft regulatory guide or provided an explanation during the meetings held on the research and draft guide.

After considering Mr. Shea's concerns, I determined that the appropriate next step was to follow NRC's regulatory processes related to the development of regulatory guides and seek additional stakeholder feedback. As we progress in the development and revision of DG-1199, input will be explicitly sought from the Committee to Review Generic Requirements, the Advisory Committee for Reactor Safeguards, and external stakeholders such as the public and nuclear industry. After receiving this additional stakeholder input, Mr. Shea's concerns will be reevaluated to determine what changes are appropriate to DG-1199.

Mr. Shea's non-concurrence was received after the Director of the Division of Risk Assessment signed out DG-1199. As such, it was not available for review by myself, the document signer and others on concurrence. Upon receipt of his non-concurrence, it was provided to my supervisor, the Director of the Division of Risk Assessment, for consideration. The attached document provides his review of Mr. Shea's non-concurrence.

- [ ] CONTINUED IN SECTION D

SIGNATURE	DATE
	6/11/09

SUBMIT THIS PAGE TO DOCUMENT SPONSOR

NRC FORM 757
NRC MD 10.158
(3-2009)

U.S. NUCLEAR REGULATORY COMMISSION

NON-CONCURRENCE PROCESS

TITLE OF DOCUMENT	ADAMS ACCESSION NO.
Draft Regulatory Guide DG-1199 (Proposed Revision 1 of RG 1.183)	**ML090570555**

SECTION C - TO BE COMPLETED BY DOCUMENT SPONSOR

NAME
Mark Cunningham

TITLE	PHONE NO.
Division Director	**301-415-2884**

ORGANIZATION
NRR/DRA

ACTIONS TAKEN TO ADDRESS NON-CONCURRENCE (This section should be revised, as necessary, to reflect the final outcome of the non-concurrence process, including a complete discussion of how individual concerns were addressed.)

Please see the attached response.

☐ CONTINUED IN SECTION D

SIGNATURE - DOCUMENT SPONSOR	DATE	SIGNATURE - DOCUMENT SIGNER	DATE
[signature]	6/11/09	[signature]	6/11/09

NON-CONCURRING INDIVIDUAL *(To be completed by document sponsor* when process is complete, i.e., after document is signed):

☐ CONCURS
☒ NON-CONCURS
☐ WITHDRAWS NON-CONCURRENCE (i.e., discontinues process)

☒ WANTS NCP FORM PUBLIC
☐ WANTS NCP FORM NON-PUBLIC

PRM-50-122

November 8, 2020

U.S. Nuclear Regulatory Commission

Washington, D.C. 20555-0001

Attention: Rulemakings and Adjudications Staff

As the petitioner, I submit the following observations and insights as a public comment to PRM-50-122, "*Accident Source Term Methodologies and Corresponding Release Fractions*" (NRC-2020-0150).

Further research indicates the errors and omissions of Regulatory Guide 1.183, "*Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors,"* are pervasive and their origins are suspect.

NUREG/CR-5247 (RASCAL) tells us:

> *Predicting* [nuclear accident] *doses or consequences . . . requires several steps: (1) predicting the quantity and timing of the release from the plant (source term), (2) predicting the movement of the plume (transport), and (3) predicting the dose from the plume and predicting the health effects from the dose. Each of these steps requires collection of appropriate data, and data collection and the subsequent computations are subject to uncertainties.*
>
> T*he largest single component of uncertainty is expected in the estimate of the source term. Unanticipated catastrophic containment failure is a case in which the source term could be underestimated by a factor of 1,000,000 if monitor readings are used to estimate the source term.* [Emphasis added]

Because the "*the accident source term is a fundamental assumption upon which a large portion of the facility* [nuclear power plant] *design is based*," its uncertainty is, arguably, the single most consequential factor that affects the confidence level of nuclear safety. While uncertainties cannot reasonably be eliminated in nuclear accident analyses, it is incumbent upon those performing and overseeing these analyses to objectively evaluate the uncertainties, such that the level of confidence given to these analyses appropriately reflect the level of safety they provide. Herein lies the problem with RG 1.183.

The NRC and the Government Accountability Office (GAO) guidelines require that uncertainties be addressed in regulatory analyses for radiological exposure; however, source term uncertainties are not addressed in RG 1.183 or in the regulatory analyses performed by licensees. Furthermore, because of the conceptual errors identified in RG 1.183, the, already, unknown uncertainties in the resulting radiological exposure analyses are multiplied by an unknown factor. Its conceptually inaccurate and non-conservative assumptions have been integrated into a large portion of nuclear power plant design and a wide range of licensing activities. These false assumptions clearly indicate the safety of nuclear power is uncertain—at best.

The intent of this comment is to promote responsible nuclear safety rule-making.

Sincerely,

Brian D. Magnuson

magnuson28@msn.com

1020 Station Blvd. #212

Aurora, IL 60504

Lead Emergency Management Specialist—Exelon Corporation (Warrenville, IL) Former NRC Licensed Senior Reactor Operator/Operations Shift Manager at Quad Cities Nuclear Power Station

—Acting expressly as a member of the public.

Research and Observations:

1. MSIVs are NOT PCIVs

As defined by 10 CFR 50.2, MSIVs are part of the Reactor Coolant System (RCS), as such, they are part of the Reactor Coolant pressure Boundary (RCPB). However, main steam isolation valves (MSIVs) on Boiling Water Reactors (BWR) are commonly considered primary containment isolation valves (PCIVs) in plant Technical Specifications and NRC guidance (e.g., NUREG-1433, Rev. 4). Unfortunately, this characterization is misleading and inaccurate because MSIVs are designed to isolate the *reactor vessel* (reactor coolant pressure boundary)—to control the loss of coolant from the reactor vessel and the release of radioactive materials to the environment in an accident.

This mischaracterization did not always exist. Contrary to plant Technical Specifications and NRC guidance, MSIVs are not Primary Containment Isolation Valves (PCIV); they are, in fact, reactor coolant system (RCS)/reactor coolant pressure boundary isolation valves. The designer of most BWR plants (General Electric) called them "*Reactor Vessel Isolation Valves*." The nuclear industry combined MSIVs with PCIVs and called the group, "*Primary Containment Reactor Vessel Isolation Control System (PCRVICS)*" or just "*CRVICS*"—in plant TS and safety analyses (UFSAR). Sometime later, references to MSIVs as "PCRVICS" or "CRVICS" were systematically removed from plant TS, UFSARs and NRC guidance. Thereafter, MSIVs have been mischaracterized as PCIVs. Nevertheless, stray references to PCRVICS and CRVICS still exist (e.g., Limerick LER 01-001-00, Fitzpatrick LER 87-021, Fermi UFSAR Revision 21).

Because the conceptually inaccurate description of BWR MSIVs has been deeply imbedded in NRC regulations and plant licensing bases, there is a broad range of generic safety issues.

2. MSIV Leakage Technical Specifications are Inadequate

As an apparent consequence of this mischaracterization, BWR plant technical specification surveillance requirements for MSIV leakage are inadequate because MSIVs are incorrectly tested at a pressure corresponding to the peak primary containment accident pressure—instead of the corresponding peak *reactor pressure*. Therefore, measured MSIV leakage rates at BWR plants are non-conservative and grossly inaccurate. This negates the intent of the technical specification limit of MSIV leakage which is presumed to be conservatively set to ensure that offsite dose consequences of design basis accidents are a small fraction of the regulatory limits in 10 CFR Part 100.

Because of the inaccurate MSIV TS SR test pressure, in the event of an accident, the radiological release through (closed) MSIVs will be much greater than that assumed in design basis dose calculations (AST or other). Given the current margins in these calculations, radiation doses to the public and control room operators will likely exceed federal limits.

Because MSIV leakage results in a loss of reactor coolant, inaccurate and nonconservative accident dose analyses (e.g., LOCA, MSLB), the current MSIV TS SR are inimical to the health and safety of the public.

NRC safety evaluations of License Amendment Requests that accept or acknowledge that “the MSIVs are functionally part of the primary containment boundary” are conceptually inaccurate and fail to satisfy 10 CFR 50.92(c) because they create significant hazards. License Amendment Requests approved under this false assumption (1) involve a significant increase in the consequences of an accident previously evaluated; (2) create the possibility different kind of accident from any accident previously evaluated; and (3) involve a significant reduction in a margin of safety.

3. Appendix J to Part 50 is also Conceptually in Error and Misapplied to MSIVs and other Reactor Vessel Isolation Valves

Appendix J to Part 50—Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors:

> *H. "Type C Tests" means tests intended to measure containment isolation valve leakage rates. The containment isolation valves included are those that:*
>
> *1. Provide a direct connection between the inside and outside atmospheres of the primary reactor containment under normal operation, such as purge and ventilation, vacuum relief, and instrument valves;*

2. Are required to close automatically upon receipt of a containment isolation signal in response to controls intended to effect containment isolation;

3. Are required to operate intermittently under post-accident conditions; and

4. Are in main steam and feedwater piping and other systems which penetrate containment of direct-cycle boiling water power reactors.

As previously described, MSIVs are not containment isolation valves. Main steam lines do not provide a direct connection between the inside and outside atmospheres of the primary reactor containment under normal operation. Furthermore, the reactor coolant pressure boundary extends to outermost isolation valves in feedwater, HPCI, RCIC, Reactor Water Cleanup and other reactor coolant system piping. Similar to RG 1.183, Appendix J is conceptually in error and misapplied to these reactor vessel isolation valves (reactor pressure boundary isolation valves).

4. MSIV LLRT Failures are Failures of the Reactor Coolant (System) Pressure Boundary

MSIV Local Leak Rate Test failures are failures of the Reactor Coolant System Pressure Boundary; however, these common failures are not monitored by the Reactor Oversight Process (ROP) as "*Reactor Coolant System Leakage"* or identified in NRC Inspection Manual Chapter 0308, Attachment 1.

MSIV Local Leak Rate Testing failures are violations of plants technical specifications and are often reported under 10 CFR 50.73 (a)(2)(ii) "because an event occurred which resulted in the degradation of one of the plant's principal safety barriers." Regrettably, these License Event Reports do not recognize or acknowledge that these failures are failures of the reactor coolant pressure boundary. As such, the safety consequence of these failure is routinely minimized. These failures of the RCS/RCPB have not been considered "significant conditions adverse to quality"; therefore, measures to assure that corrective actions are taken to preclude repetition have NOT been implemented by the licensees or enforced by the NRC as required by Appendix B to Part 50, XVI. Corrective Action.

Furthermore, contrary to 10 CFR 50.65, "Requirements for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," reported MSIV LLRT failures have not been appropriately categorized as maintenance-preventable functional failures (MPFFs), as such, maintenance that would prevent failures of the reactor coolant pressure boundary is not performed.

Maintenance preventable failures of the reactor coolant pressure boundary are also contrary to Appendix A, "General Design Criteria for Nuclear Power Plants," to Title 10, Part 50, "Domestic Licensing of Production and Utilization Facilities," of the Code of Federal Regulations (10 CFR Part 50). General Design Criterion (GDC) 14, "Reactor Coolant Pressure Boundary," requires that licensees or applicants design, fabricate, erect, and test the reactor coolant pressure boundary (RCPB) so as to ensure an extremely low probability of abnormal leakage.

5. There is Only One Physical Fission Product Barrier

Contrary to NRC and industry publications, there are not necessarily three[1] physical barriers between reactor core fission products (millions of Curies) and the environment.

MSIVs are part of the Reactor Coolant System and pressure-containing components of the Reactor Coolant Pressure Boundary; however, in an accident, closed MSIVs will not prevent the release of fission products to the environment. Therefore, the Reactor Coolant System is not an effective physical barrier. AST LOCA dose consequence analyses indicate MSIV leakage is a significant contributor to operator/main control room doses. Therefore, the failure of only one physical barrier—approximately 0.029 inches of fuel cladding—will result in a significant radiological release to the environment.

Similarly, in any spent fuel pool accident, the fuel cladding provides the only physical barrier between fission products and the environment, and in a design basis Fuel Handling Accident that one barrier is lost. As we have learned from NRC Information Notice No. 90-08*: "KR-85 Hazards From Decayed Fuel*," no amount of water shielding in spent fuel pools will protect workers in the area or prevent the release of nobel gasses to the environment in the event that spent fuel cladding is ruptured from of mechanical damage or overheating.

[1] Because of fuel cladding gap releases and the effects of high burnups which can reduce fuel pellets to powder, the integrity of the fuel pellet form is no longer considered a physical fission product barrier.

6. "Recently" Irradiated Fuel

It should be noted that license amendments that adopted the alternative source term methodology, as prescribed in Title 10 to the Code of Federal Regulations Section 50.67, also revised the technical specification sections associated with the implementation of Technical Specification Task Force (TSTF) - 51 traveler, which provided relaxation of certain requirements during movement of irradiated fuel.

> *"The purpose of the TSTF-51 TS changes is to establish a point where OPERABILITY of ESFs typically used to mitigate the consequences of a FHA are no longer required to meet the SRP guidance on offsite dose limits (i.e., less than 25 percent of the 10 CFR Part 100, Reactor Site Criteria," limits or the limits specified in 10 CFR 50.67). <u>Specifically, the proposal identifies that only "recently" irradiated fuel contains sufficient fission products to require OPERABILITY of the accident mitigation features to meet the accident analysis assumptions</u>. Therefore, the APPLICABILITY requirements for the associated mitigation features (including the electrical support systems) are revised. The requested changes would eliminate TS requirements for ESFs during core alterations by deleting "During CORE ALTERATIONS" from APPLICABILITY. The requested change also adds "recently" to "irradiated fuel" to revise APPLICABILITY to "During movement of recently irradiated fuel." The affected TS Limiting Conditions for Operation (LCO) required ACTION statement to immediately suspend movement of irradiated fuel assemblies in secondary containment, when the LCO is not met, is also revised to require such action only when recently irradiated fuel assemblies are moved."*

TSTF-51 (Revision 2), which was approved by the NRC on October 13,1999. It predates RG 1.183 (July 2000) and conflicts with NRC, "*Information Notice No. 90-08: KR-85 Hazards From Decayed Fuel*" (February 1, 1990) which states:

> *"Analysis of hypothetical accidents involving decayed spent fuel has focused attention on potential difficulties that could be associated with the exposure of onsite personnel to an accidental release of Kr-85. Kr-85 is a noble gas fission product that is present in the gaps between the fuel pellets and the cladding. It has a 10.76-year half-life, and, as a result of the considerably shorter half-lives of virtually all other gaseous fission products (I-129 being the exception, but in low abundance), Kr-85 becomes increasingly the dominant nuclide in the accident source term for gap releases as decay times increase. After 2 weeks of decay, Kr-85 is a significant nuclide in the source term, and after 190 days of decay, it is the predominant gaseous nuclide for a gap release. The unusual decay characteristics of Kr-85 give cause for focusing attention on the onsite consequences of a gap release from decayed fuel."*
>
> *"Kr-85 emits beta radiation with a maximum energy of 0.67 MeV for 99.6 percent of the decays and 0.51 MeV gamma radiation for 0.4 percent of the decays. Consequently, direct exposure to this gas would result in a dose to the skin approximately 100 times the whole-body dose. Analysis of the relative consequences (in terms of radiological doses) of a cask-drop accident as a function of decay time of the fuel is illustrated in Figure 1. In the event of a serious accident involving decayed spent fuel, protective actions would be needed for personnel on site, while offsite doses (assuming an exclusion area radius of 1 mile from the plant site) would be well below the Environmental Protection Agency's Protective Action Guides. Accordingly, it is important to be able to properly survey and monitor for Kr-85, and to assess the skin dose to workers who could be exposed to Kr-85 in the event of an accident with decayed spent fuel."*
>
> *"Licensees may wish to reevaluate whether Emergency Action Levels specified in the emergency plan and procedures governing decayed fuel-handling activities appropriately focus on concern for onsite workers and Kr-85 releases in areas where decayed spent fuel accidents could occur, for example, the spent fuel pool working floor. Furthermore, licensees may wish to determine if emergency plans and corresponding implementing procedures address the means for limiting radiological exposures of onsite personnel who are in other areas of the plant. Among other things, moving onsite personnel away from the plume and shutting off building air intakes downwind from the source may be appropriate."*

The following reference provides additional insights and indicates that TSTF-51 <u>inappropriately</u> allowed licensees to eliminate the technical specification requirements for spent fuel pool area <u>gamma</u> radiation monitors because they would not detect the <u>beta</u> radiation emitted by Kr-85 in a Fuel Handling Accident.

"Comments on Direct Final Rule Change to 10 CFR 50.68" (Serial: RNP-RA/06-0121 USNRC) states:

> *The typical design of a nuclear power plant includes one or more gamma sensitive Area Radiation Monitors (ARMs) located in the area above the SFP. While loading a cask in the SFP, there will be approximately 23 feet of water between the ARM and a potential criticality event in the cask. With this significant amount of intervening shielding, these ARMs will not respond to the direct radiation resulting from a criticality event. The criticality event could result in cladding*

damage and the release of the fuel gap fission products. However, fuel being loaded into dry storage casks will have decayed for at least 3 years, therefore, the only fission product released from the fuel rod gap to the area above the SFP that is of any dose significance would be Kr-85. Kr-85 is essentially a beta emitter (only 1 gamma every 250 disintegrations) and hence the ARMs, which are only sensitive to gamma radiation, would likely not alarm. However, the airborne concentrations of Kr-85 could represent a skin dose hazard to the personnel by the SFP (see NRC Information Notice 90-08). These ARMs cannot meet the sensitivity requirements for criticality monitors as specified in 10 CFR 70.24(a)(1). 10 CFR 72 does not provide similar specific requirements for a criticality monitoring system. If the requirements for criticality monitoring to meet 10 CFR 72.124(c) are more general (e.g., a system that would warn of a radiation hazard to personnel), then the current ARMs would not meet that requirement either due to the Kr-85 impact.

The SFP ARMs cannot be considered criticality monitors because they will not respond to a criticality event. This was the reason nuclear power plants had to apply for exemptions to 10 CFR 70.24 and the reason 10 CFR 50.68 was written. The wording in 10 CFR 50.68 implies that these ARMs are not criticality monitors, as the rule states that in lieu of maintaining a criticality monitoring system, the licensee must meet a number of criteria, one of which is to maintain a radiation monitoring system in the fuel handling area. Licensees have taken credit for the SFP ARMs to meet this requirement. If these ARMs could be considered criticality monitors then 10 CFR 50.68 would not be required. If the interpretation of the requirements of 10 CFR 72.124(c) for underwater monitoring, as provided in the Technical Evaluation, are not corrected, then licensees may have to file exemption requests to 10 CFR 72.124(c).

IN 90-08 appears to have identified a generic safety issue in 1990 that still exists today.

7. MSIVs are Leak Tight *only* in an AST MSLB

RG 1.183, LOCA *ASSUMPTIONS ON MAIN STEAM ISOLATION VALVE LEAKAGE IN BWRs* states:

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

All the MSIVs should be assumed to leak at the maximum leak rate above which the technical specifications would require declaring the MSIVs inoperable. The leakage should be assumed to continue for the duration of the accident. Postulated leakage may be reduced after the first 24 hours, if supported by site-specific analyses, to a value not less than 50% of the maximum leak rate.

MISVs were not designed to leak as RG 1.183 would have you believe. As previously explained, they leak because of (normal wear from high pressure steam flow and ineffective preventative maintenance practices. Reference NRC Staff Evaluation of the NUREG—1285 General Electric Company Nuclear Reactor Study ("Reed Report"):

> *"4.3 Main Steam Isolation Valve Leak Tightness Issue: The issue of leak tightness of main steam isolation valves (MSIVs) was identified in the Reed Report in the section on Mechanical Systems and Equipment, but was not discussed in the GE status report provided in 1978. Main steam isolation valves (MSIVs) have been notorious for leaking at high rates when they are tested during the 18-month leak tightness testing that is generally required by the technical specifications. Most plants have a technical specification leak rate limit of 11.5 standard cubic feet per hour (scfh) per valve. At some plants the as-found leak rate has been as high as 4500 scfh.*
>
> *MSIV leak tightness was a concern in 1975, and it is still a concern that has not been fully resolved. The BWR Owners Group (BWROG) formed a committee to evaluate this same issue independently, with GE giving technical support to the BWROG committee. THIS COMMITTEE GENERALLY FOUND THAT THE HIGH LEAKAGE RATES WERE ATTRIBUTABLE TO VALVE MAINTENANCE PRACTICES.* ***.*** *For those plants that have adopted the BWROG recommendations resulting from their evaluation, the as-found MSIV leak rates have generally been within the plant-specific technical specification limit . . . For example, Peach Bottom 3, had typical as-found leak rates of over 3000 scfh for each of the MSIVs. After following the BWROG recommendations, the next as-found leak rates were found to be less than 11.5 scfh for seven of the eight MSIVs and approximately 14.7 scfh for the eighth MSIV. THIS DEMONSTRATES THAT THE MSIVS CAN BE MAINTAINED WITHIN THEIR RESPECTIVE TECHNICAL SPECIFICATION LEAKAGE LIMITS . . .*" [Emphasis added.]

RG 1.183 reasonably assumes MSIVs will leak for 30 days in a LOCA; however, it tergiversates and illogically assumes the radiological release from a Main Steam Line Break (MSLB) will be terminated when the MSIVs close (approximately 10 seconds). RG 1.183 allows licensees to assume "*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."*

This false assumption results in the under calculation of pre-accident doses that are used to determine the level of protection necessary to protect control room operators and the public from overexposures in the event of a MSLB accident.

Here it is important to recognize that MSIV leakage is the single most significant contributor to operator/main control room doses and the most limiting regulated dose in accident (MSLB) analyses, and consider that Control Room Habitability regulatory guidance that required physical plant modifications to protect control room operators, lagged the identification of MSIV leakage concerns by over twenty years but coincided with the issuance of RG 1.183 / 10 CFR 50.67 that introduced the TEDE dose limits.

Furthermore, contrary to RG 1.196, "*Control Room Habitability at Light-Water Nuclear Power Reactors*, MSLB dose analyses are not reperformed in all licensing activities associated with MSIV leakage requirements. RG 1.196 states:

> *"In determining the limiting condition for potential radiological accidents, it should not be presumed that the LOCA is the limiting accident because it has the largest initial source of activity. Other accidents, e.g., fuel handling accidents, may produce larger control room operator doses because the manner in which the CRHSs respond may provide less protection to the operators."*

8. Post-Accident Dose Analyses Will Not Prevent Accidents or Overexposures

As its name suggests, RG 1.183, “Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors is intended for pre-accident dose analyses; however, as it states, can be applied to post-accident dose assessments as required by 10 CFR 50.47 Emergency Plans and Appendix E to Part 50, but is not. Instead, the nuclear industry continues to use the RASCAL source terms and methodologies, that unlike RG 1.183, have been periodically updated to reflect insights such as those learned from the accident at Fukishima (NUREG- 1940 RASCAL 4).

The RASCAL source terms are more conservative and do not contain the conceptual errors that SAND2008-6601 identified in RG 1.183. This inconsistent application of accident source terms does not appear to be in the best interest of the health and safety of the public.

REFERENCES

10 CFR 50.2: *Reactor coolant pressure boundary means all those pressure-containing components of boiling and pressurized water-cooled nuclear power reactors, such as pressure vessels, piping, pumps, and valves, which are:*
(1) Part of the reactor coolant system
For nuclear power reactors of the direct cycle boiling water type, the reactor coolant system extends to and includes the outermost containment [sic] isolation valve in the main steam [MSIVs] *and feedwater piping.*

APPENDIX A TO PART 50—GENERAL DESIGN CRITERIA FOR NUCLEAR POWER PLANTS

> *GDC 14—Reactor coolant pressure boundary [e.g., MSIVs]. The reactor coolant pressure boundary shall be designed, fabricated, erected, and tested so as to have an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture.*
>
> *GDC 15—Reactor coolant system [e.g., MSIVs] design. The reactor coolant system and associated auxiliary, control, and protection systems shall be designed with sufficient margin to*

assure that the design conditions of the reactor coolant pressure boundary are not exceeded during any condition of normal operation, including anticipated operational occurrences.

GDC 30—Quality of reactor coolant pressure boundary. Components which are part of the reactor coolant pressure boundary [e.g., MSIVs] shall be designed, fabricated, erected, and tested to the highest quality standards practical. Means shall be provided for detecting and, to the extent practical, identifying the location of the source of reactor coolant leakage.

GDC 31—Fracture prevention of reactor coolant pressure boundary. The reactor coolant pressure boundary [e.g., MSIVs] shall be designed with sufficient margin to assure that when stressed under operating, maintenance, testing, and postulated accident conditions (1) the boundary behaves in a nonbrittle manner and (2) the probability of rapidly propagating fracture is minimized. The design shall reflect consideration of service temperatures and other conditions of the boundary material under operating, maintenance, testing, and postulated accident conditions and the uncertainties in determining (1) material properties, (2) the effects of irradiation on material properties, (3) residual, steady state and transient stresses, and (4) size of flaws.

GDC 32—Inspection of reactor coolant pressure boundary. Components which are part of the reactor coolant pressure boundary [e.g., MSIVs] shall be designed to permit (1) periodic inspection and testing of important areas and features to assess their structural and leaktight integrity, and (2) an appropriate material surveillance program for the reactor pressure vessel.

GDC 55: Reactor coolant pressure boundary penetrating containment. Each line that is part of the reactor coolant pressure boundary and that penetrates primary reactor containment [e.g., MSIVs] shall be provided with containment isolation valves . . .

GDC 56: Primary containment isolation. Each line that connects directly to the containment atmosphere [not MSIVs] and penetrates primary reactor containment shall be provided with containment isolation valves . . .

APPENDIX B TO PART 50—QUALITY ASSURANCE CRITERIA FOR NUCLEAR POWER PLANTS AND FUEL REPROCESSING PLANTS

III. Design Control

Measures shall be established to assure that applicable regulatory requirements and the design basis, as defined in § 50.2 and as specified in the license application, for those structures, systems, and components to which this appendix applies are correctly translated into specifications, drawings, procedures, and instructions. These measures shall include provisions to assure that appropriate quality standards are specified and included in design documents and that deviations from such standards are controlled. Measures shall also be established for the selection and review for suitability of application of materials, parts, equipment, and processes that are essential to the safety-related functions of the structures, systems and components.

Measures shall be established for the identification and control of design interfaces and for coordination among participating design organizations. These measures shall include the establishment of procedures among participating design organizations for the review, approval, release, distribution, and revision of documents involving design interfaces.

The design control measures shall provide for verifying or checking the adequacy of design, such as by the performance of design reviews, by the use of alternate or simplified calculational methods, or by the performance of a suitable testing program. The verifying or checking process shall be performed by individuals or groups other than those who performed the original design, but who may be from the same organization. Where a test program is used to verify the adequacy of a specific design feature in lieu of other verifying or checking processes, it shall include suitable qualifications testing of a prototype unit under the most adverse design conditions. Design control measures shall be applied to items such as the following: reactor physics, stress, thermal, hydraulic, and accident analyses; compatibility of materials; accessibility for inservice inspection, maintenance, and repair; and delineation of acceptance criteria for inspections and tests.

V. Instructions, Procedures, and Drawings

Activities affecting quality shall be prescribed by documented instructions, procedures, or drawings, of a type appropriate to the circumstances and shall be accomplished in accordance with these instructions, procedures, or drawings. Instructions, procedures, or drawings shall include appropriate quantitative or qualitative acceptance criteria for determining that important activities have been satisfactorily accomplished.

VI. Document Control

Measures shall be established to control the issuance of documents, such as instructions, procedures, and drawings, including changes thereto, which prescribe all activities affecting quality. These measures shall assure that documents, including changes, are reviewed for adequacy and approved for release by authorized personnel and are distributed to and used at the location where the prescribed activity is performed. Changes to documents shall be reviewed and approved by the same organizations that performed the original review and approval unless the applicant designates another responsible organization.

XI. Test Control

A test program shall be established to assure that all testing required to demonstrate that structures, systems, and components will perform satisfactorily in service is identified and performed in accordance with written test procedures which incorporate the requirements and acceptance limits contained in applicable design documents. The test program shall include, as appropriate, proof tests prior to installation, preoperational tests, and operational tests during nuclear power plant or fuel reprocessing plant operation, of structures, systems, and components. Test procedures shall include provisions for assuring that all prerequisites for the given test have been met, that adequate test instrumentation is available and used, and that the test is performed under suitable environmental conditions. Test results shall be documented and evaluated to assure that test requirements have been satisfied.

XVI. Corrective Action

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Measures shall be established to assure that conditions adverse to quality, such as failures, malfunctions, deficiencies, deviations, defective material and equipment, and nonconformances are promptly identified and corrected. In the case of significant conditions adverse to quality, the measures shall assure that the cause of the condition is determined and corrective action taken to preclude repetition. The identification of the significant condition adverse to quality, the cause of the condition, and the corrective action taken shall be documented and reported to appropriate levels of management.

(1980) NUREG-0737 CLARIFICATION OF TMI ACTION PLAN REQUIREMENTS

The design-basis-accident (DBA) radiation source term should be for the loss-of-coolant accident LOCA containment leakage and engineered safety feature (ESF) leakage contribution outside containment as described in Appendix A and B of Standard Review Plan Chapter 15.6.5. In addition, boiling-water reactor (BWR) facility evaluations should add any leakage from the main steam isolation valves (MSIV) (i. e., valve-stem leakage, valve seat leakage, main steam isolation valve leakage control system release) to the containment leakage and ESF leakage following a LOCA. This should not be construed as altering the staff recommendations in Section D of Regulatory Guide 1.96 (Rev. 2) regarding MSIV leakage-control systems.

Other DBAs should be reviewed to determine whether they might constitute a more-severe control-room hazard than the LOCA. In addition to the accident-analysis results, which should either identify the possible need for control-room modifications or provide assurance that the habitability systems will operate under all postulated conditions to permit the control-room operators to remain in the control room to take appropriate actions required by General Design Criterion 19, the licensee should submit sufficient information needed for an independent evaluation of the adequacy of the habitability systems. Attachment 1 lists the information that should be provided along with the licensee's evaluation.

(1982) Reference INFORMATION NO. 82-23: MAIN STEAM ISOLATION VALVE (MSIV) LEAKAGE

IE [NRC Office of Inspection and Enforcement] has completed a survey of MSIV performance at BWRs for the years 1979 through 1981. IE found that 19 of 25 operating BWRs had MSIVs which failed to meet, during one or more surveillance tests, the limiting condition for operation (LCO) which specifies the maximum permissible leak rate. The number of MSIV test failures exceeded 151 and occurred with MSIVs supplied by all three MSIV vendors, i.e., Atwood & Morrill, Crane, and Rockwell.

Measured leak rates which exceeded the LCO ranged from greater than 11.5 standard cubic feet per hour (scfh) to 3427 scfh. Twelve stations had 57 MSIV tests with results greater than 11.5 scfh and less than 100 scfh, and five stations (nine units) had 66 MSIV tests with results between 100 and 3500 scfh. Four other licensees had more than 24 test failures but did not measure, estimate, or report the magnitudes of the leak rates.

This information indicates that some MSIVs may not adequately limit release of radioactivity to the environment if called upon to do so. NRC is considering the need for improved MSIV maintenance, more frequent MSIV testing or installation of leakage control systems.

(1986) Availability of NUREG-1169, "Technical Findings Related to Generic Issue C-8; Boiling Water Reactor Main Steam Isolation Valve Leakage and Leakage Treatment Methods" (Generic Letter No. 86-17)

(1990) IN 90-08, KR-85 HAZARDS FROM DECAYED FUEL

(1989, 1992) NUREG/CR-5247, RASCAL Version 2.1 User's Guide

> *The Radiological Assessment System for Consequence Analysis (RASCAL) (Athey et al. 1989, 1992) is a set of personal computer-based tools. RASCAL Version 2.1 contains tools to estimate source term, atmospheric transport, and dose from a radiological accident (ST-DOSE), to estimate dose from field measurements of radionuclide concentrations (FM-DOSE), and to compute decay of radionuclides (DECAY). RASCAL was developed for use by U.S. Nuclear Regulatory Commission (NRC) personnel who report to the site of a nuclear accident to conduct an independent assessment of dose projections.*
>
> *Predicting* [nuclear accident] *doses or consequences . . . requires several steps: (1) predicting the quantity and timing of the release from the plant (source term), (2) predicting the movement of the plume (transport), and (3) predicting the dose from the plume and predicting the health effects from the dose. Each of these steps requires collection of appropriate data, and data collection and the subsequent computations are subject to uncertainties.*
>
> <u>*The largest single component of uncertainty is expected in the estimate of the source term*</u>*. Unanticipated catastrophic containment failure is a case in which the source term could be underestimated by a factor of 1,000,000 if monitor readings are used to estimate the source term. For lesser (non-core damage) accidents in which the total release is through a monitored pathway and consists mostly of noble gases, the source term uncertainty can be reduced. However, the transport and dose uncertainties would remain unchanged.* [Emphasis added]

(1994) SECY-94-302, SOURCE TERM-RELATED TECHNICAL AND LICENSING ISSUES PERTAINING TO EVOLUTIONARY AND PASSIVE LIGHT-WATER-REACTOR DESIGNS

(1998) 10 CFR 50.68 CRITICALITY ACCIDENT REQUIREMENTS

> (*6) Radiation monitors are provided in storage and associated handling areas when fuel is present to detect excessive radiation levels and to initiate appropriate safety actions.* [63 FR 63130, <u>Nov. 12, 1998</u>; as amended at 71 FR 66648, Nov. 16, 2006]

(2000) REGULATORY GUIDE 1.183, “ALTERNATIVE RADIOLOGICAL SOURCE TERMS FOR EVALUATING DESIGN BASIS ACCIDENTS AT NUCLEAR POWER REACTORS”

(2001) NUCLEAR ENERGY INSTITUTE, “CONTROL ROOM HABITABILITY ASSESSMENT GUIDANCE,” NEI 99-03, REVISION 0

(2003) REGULATORY GUIDE 1.194, “ATMOSPHERIC RELATIVE CONCENTRATIONS FOR CONTROL ROOM RADIOLOGICAL HABITABILITY ASSESSMENTS AT NUCLEAR POWER PLANTS”

(2003) REGULATORY GUIDE 1.195 USNRC, “METHODS AND ASSUMPTIONS FOR EVALUATING RADIOLOGICAL CONSEQUENCES OF DESIGN BASIS ACCIDENTS AT LIGHT-WATER NUCLEAR POWER REACTORS”

(2003, Revision 0) REGULATORY GUIDE 1.196 CONTROL ROOM HABITABILITY AT LIGHT-WATER NUCLEAR POWER REACTORS

> *The primary design function of CRHSs is to provide a safe environment in which the operator can keep the nuclear reactor and auxiliary systems under control during normal operations and can safely shut down these systems during abnormal situations to protect the health and safety of the public and plant workers. If the control room is not habitable or the response of the operator is impaired during an accident, there could be increased consequences to public health and safety. It is important for the operators to be confident of their safety in the control room to minimize errors of omission and commission. The Regulatory Positions below provide methods acceptable to the NRC staff for ensuring that the public and the control room operators are protected.*
>
> *Over the facility’s lifetime the licensing bases change. The staff may have reviewed and approved the licensing bases of facilities licensed before the issuance of this guide. The original licensing bases may have been submitted as part of the construction permit application. Licensees may have modified them in response to NRC questions. In addition, the licensing bases were part of the application for the OL (FSAR). Depending on the plant vintage, licensees may have modified their licensing bases in response to TMI Action Item III.D.3.4. Amendments to the OL may have resulted in changes to the licensing bases of the CRHSs. Licensees should review the applicable plant changes to their licensing bases to determine the current bases.*
>
> *A group of reactors received their construction permits or OLs before the GDCs were promulgated. During this time, proposed GDCs (sometimes called “Principal Design Criteria”) were published in the Federal Register for comment. These proposed GDCs addressed CRH. Although facilities may have been licensed before the promulgation of the GDCs, licensees may have committed to the form of the GDCs as they existed at the time of licensing. A review of the record associated with the construction permit and OL proceedings should confirm whether licensees made such a commitment. Therefore, licensees that received their construction permits or OLs before the GDCs were promulgated should review their commitments to the draft form of the GDC to understand their CRH licensing bases.*

Over the facility's lifetime the licensing bases change. The staff may have reviewed and approved the licensing bases of facilities licensed before the issuance of this guide. The original licensing bases may have been submitted as part of the construction permit application. Licensees may have modified them in response to NRC questions. In addition, the licensing bases were part of the application for the OL (FSAR). Depending on the plant vintage, licensees may have modified their licensing bases in response to TMI Action Item III.D.3.4. Amendments to the OL may have resulted in changes to the licensing bases of the CRHSs. Licensees should review the applicable plant changes to their licensing bases to determine the current bases. A group of reactors received their construction permits or OLs before the GDCs were promulgated. During this time, proposed GDCs (sometimes called "Principal Design Criteria") were published in the Federal Register for comment. These proposed GDCs addressed CRH. Although facilities may have been licensed before the promulgation of the GDCs, licensees may have committed to the form of the GDCs as they existed at the time of licensing. A review of the record associated with the construction permit and OL proceedings should confirm whether licensees made such a commitment. Therefore, licensees that received their construction permits or OLs before the GDCs were promulgated should review their commitments to the draft form of the GDC to understand their CRH licensing bases.

Consistent with Regulatory Position 2.2.1, licensees should ensure that their assumed control room inleakage input value used in any accident calculations or evaluations (Regulatory Positions 2.4 and 2.5) are validated by the test methods provided in Regulatory Guide 1.197.

Unless a facility relies on a common control room isolation process for all types of radiological accidents, the limiting accident may not be obvious. There are several reasons for this:

• The inleakage characteristics of the envelope may vary with the CRHS's response to an accident.

• The mitigative equipment used to reduce the radioactivity released to the environment may vary with the accident.

• The location of the release points for the various accidents relative to the control room intakes may result in less favorable atmospheric dispersion and higher magnitude intake concentrations.

Licensees should factor all the potential differences in accidents and the CRHS's performance in order to determine the limiting condition.

Licensees should calculate control room operator doses for the methodology and accidents identified in Regulatory Guide 1.195 (Ref. 5) or Regulatory Guide 1.183 (Ref. 6). For CREs under construction, the control room operators' doses should be based on expected CRHS performance values. When the envelope and associated ventilation systems are operational, the inleakage value should be determined using Regulatory Guide 1.197 (Ref. 4).

<u>Some licensees were allowed to leave TMI Action Item III.D.3.4 actions open until the alternative source term rulemaking and regulatory guidance were published. These actions were completed with the issuance of 10 CFR 50.67 and Regulatory Guide 1.183 (Ref. 6).</u> The

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Regulatory Positions in this regulatory guide on control room habitability provide methods acceptable to the NRC staff for closing open TMI Action Item III.D.3.4 actions.

(2003) REGULATORY GUIDE 1.197 "DEMONSTRATING CONTROL ROOM ENVELOPE INTEGRITY AT NUCLEAR POWER REACTORS"

(2007, Revision 1) REGULATORY GUIDE 1.196 CONTROL ROOM HABITABILITY AT LIGHT-WATER NUCLEAR POWER REACTORS

Control Room Acceptance Criteria: The following guidelines may be used in lieu of those provided in SRP 6.4 (Ref. 14) when showing compliance with the dose guidelines in GDC-19 of Appendix A to 10 CFR Part 50. <u>The following guidelines relax the thyroid and skin acceptance criteria from that given in SRP 6.4</u>.

Whole body 5 rem

Thyroid 50 rem

Skin 50 rem[12]

[12]*Credit for the beta radiation shielding afforded by special protective clothing and eye protection is allowed if the applicant commits to their use during severe radiation releases. However, even though protective clothing is used, the calculated unprotected skin dose is not to exceed 75 rem. These limits are design criteria and are not to be interpreted as acceptable occupational doses.*

(2008) SAND2008-6601 ANALYSIS OF MAIN STEAM ISOLATION VALVE LEAKAGE IN DESIGN BASIS ACCIDENTS USING MELCOR 1.8.6 AND RADTRAD

(2008) SECY-08-172, DENIAL OF PETITION FOR RULEMAKING PRM-50-87 CONCERNING CONTROL ROOM HABITABILITY RADIOLOGICAL DOSE REQUIREMENTS AS GOVERNED BY REGULATIONS SPECIFIED IN APPENDIX A TO 10 CFR PART 50 AND IN 10 CFR 50.67

On May 17, 2007, Mr. Crandall submitted a PRM (PRM-50-87) requesting that the U.S. Nuclear Regulatory Commission (NRC) amend Appendix A, "General Design Criteria for Nuclear Power Plants" to Title 10, Part 50, "Domestic Licensing of Production and Utilization Facilities," of the Code of Federal Regulations (10 CFR Part 50) and 10 CFR 50.67, "Accident source term." Specifically, the petitioner requested to delete the 5 rem whole body dose limit specified in General Design Criterion (GDC) 19, "Control Room," of Appendix A to 10 CFR Part 50 and the 0.05 sievert (Sv) (5 rem) total effective dose equivalent (TEDE) limit specified in both GDC 19 and 10 CFR 50.67 (b)(2)(iii). The petitioner stated that the current deterministic radiological dose requirements for control room habitability have resulted in several negative safety consequences including an increased risk to public safety.

The NRC regards the radiological dose standards, 5 rem TEDE in 10 CFR 50.67 and 5 rem whole body in GDC 19, as performance-based regulations. Performance-based regulations do not provide prescriptive requirements and, therefore, do not require licensees to use specific designs or methodologies to comply with the regulations. However, the NRC does provide regulatory guidance to licensees that includes acceptable designs and methodologies for *demonstrating compliance with the regulations. The use of the guidance is optional, and licensees are free to propose alternative means of complying with the NRC's regulations.*

The performance-based control room dose criterion is designed such that an acceptable level of control room habitability will be maintained even under the maximum credible accident scenario. The NRC has determined that providing an acceptable level of control room habitability for design-basis events is necessary to provide reasonable assurance that the control room will continue to be effectively manned and operated to mitigate the effects of the accident and protect public health and safety. By removing the acceptance criteria of 5 rem, a regulatory basis will no longer exist, and would not support the Commission's policy regarding performance-based regulations.

Based upon its review of the petition and the comments submitted, the NRC staff has determined that the conclusions upon which the petitioner relies do not substantiate a basis to eliminate the control room radiological dose acceptance criteria from current regulations as requested. Accordingly, the staff recommends denying the PRM and requests Commission approval to do so and publish the Federal Register notice (Enclosure 1) of the denial.

1. *The petitioner stated that because the primary objective of control room habitability is to ensure continuous occupancy, the primary focus should be on minimizing whole body doses from noble gases. He stated that some common control room designs, such as the filtered air intake pressurization design, focus on compliance with existing dose criteria. He concluded that the current requirements and operational criteria focus on minimizing the thyroid dose at the expense of increasing the whole body dose from noble gases which increases the probability that the control room will require evacuation.*

The NRC reviewed the petitioner's concern regarding the increase in whole body dose from noble gases, which he believes results from the intentional intake of filtered air into the control room under design-basis accident (DBA) conditions. The NRC agrees that a relatively small increase in whole body dose due to noble gases may result from the intake of filtered air into the control room. However, this small increase in dose would not increase the probability of a control room evacuation. Therefore, operators would be able to monitor plant indications and take appropriate accident mitigating actions from the control room, and there would be no increase in risk to public health and safety. The NRC's conclusion is based on a review of several existing DBA control room dose analyses that determined the impact on whole body dose resulting from filtered air intake pressurization to the control room. The NRC performed parametric evaluations and determined that while filtered air intake pressurization may result in a small addition to the control room whole body dose from noble gases, the increase is more than *offset by the reduction in thyroid dose and TEDE from inhalation of radioactive particulates, such as iodine.*

Based upon its analyses, the NRC does not agree with the petitioner's assertion regarding the negative safety impact of providing filtered intake flow into the control room. The NRC's performance-based criterion in GDC 19 requires that an applicant provide a control room habitability design that meets the specified dose criterion. Although NRC regulatory guidance provides examples of acceptable design approaches, the approach used to meet the criterion is largely under the control of an applicant. In order to meet this requirement, many licensees have chosen to incorporate filtered air intake pressurization into their control room emergency ventilation designs to reduce the cumulative dose to operators during a DBA. The purpose of providing filtered air intake pressurization flow is to establish positive pressure in the control room relative to the adjacent areas, thereby reducing the quantity of unfiltered air inleakage. Limiting unfiltered inleakage significantly reduces the thyroid dose from inhalation.

The petitioner based his assertion on the assumption that filterable activity is not likely to be a significant contributor to dose in a reactor accident. As an example, the petitioner used the March 1979 Three Mile Island Unit 2 accident. Since the accident, the NRC has expended considerable resources to better define the expected quantity and distribution of activity that could be released during a major reactor accident. As a result of this research, the NRC promulgated 10 CFR 50.67 on December 23, 1999 (64 FR 72001). Under 10 CFR 50.67, a licensee can apply for a license amendment to adopt an alternative source term (AST) that reflects a more realistic assessment of the timing of the release and the quantity and distribution of activity that could be released during a major accident hypothesized for purposes of design analyses. Many licensees have used this approach to comply with NRC regulations governing control room dose.

<u>In addition, 10 CFR 50.67 revised the control room dose criterion from a 5 rem whole body dose, or its equivalent to any organ, to a 5 rem TEDE. The relatively low thyroid organ weighting factor, as defined in 10 CFR 20.1003, "Definitions," and used in the calculation of TEDE, allows for a significant reduction in the controlling aspects of the thyroid dose, which normally governed compliance with control room dose guidelines</u>.

<u>The NRC has significantly improved the accuracy of the source term and dose methodology used in design-basis dose consequence analyses</u>. The updated source term and dose methodology address the petitioner's concerns regarding the emphasis on thyroid dose in control room habitability analyses.

> *The petitioner noted that the dose from increased iodine concentration can be mitigated by use of potassium iodide (KI) or respiratory protection, but the current regulations do not permit these mitigation measures to be used in design analyses.*

The NRC agrees that KI or Self-Contained Breathing Apparatuses (SCBAs) do have merit as short-term compensatory measures. However, the potential medical complications of KI and the potential adverse impacts to human performance of SCBAs make these measures unsuitable for long-term use. Further, the NRC's policy of ensuring that process or other engineering controls are in place instead of relying on the use of personal protective equipment is clearly set forth in 10 CFR 20.1701, "Use of process or other engineering controls" and 10 CFR 20.1702, *"Use of other controls." This policy is consistent with the recommendations of international and national radiation protection committees as described in Paragraph 167 of the International Commission on Radiological Protection (ICRP) Publication 26.*

Paragraph 167 of ICRP Publication 26 recommends that "[a]s far as is reasonably practicable, the arrangements for restricting occupational exposure should be applied to the source of radiation and to features of the workplace. The use of personal protective equipment should in general be supplementary to these more fundamental provisions. <u>The emphasis should thus be on intrinsic safety in the workplace and only secondarily on protection that depends on the worker's own actions," such as the ingestion of KI or use of respiratory equipment</u>.

As a design criterion, GDC 19 does not supplant the radiation protection standards of 10 CFR Part 20, which treat the radiation exposure of control room operators as occupational exposure.

> *The petitioner recommended that as an alternative to the total removal of dose guidelines from the regulations, most of his concerns could be resolved if the dose criteria were based solely on the whole body dose from noble gases. The NRC does not agree with the proposition that the dose criteria should be based solely on the whole body dose from noble gases. <u>The control room dose criterion of 5 rem whole body or its equivalent to any organ imposes two requirements on licensees: satisfaction of the whole body dose criterion, which is generally dominated by the dose from noble gases; and satisfaction of the organ-specific dose guidelines, which are generally dominated by the thyroid dose from the inhalation of iodine</u>.*

In most cases, demonstrating compliance with thyroid dose guidelines poses a significantly greater challenge to licensees than does compliance with the whole body dose criterion.

The 1999 amendment to 10 CFR 50.67, revised the control room dose limit to allow licensees to show compliance with either the existing limits, using the traditional Technical Information Document (TID)-14844 source term assumptions, or a revised single control room dose criterion of 5 rem TEDE[1], if the licensee adopts the AST. With the ability to reassess a maximum credible radiological release using the AST, many licensees have shown compliance with the § 50.67 single control room dose criterion of 5 rem TEDE. Licensees have accomplished this while achieving an enhanced degree of operational flexibility not realized using the traditional TID-14844 source term with the associated whole body dose criterion and organ dose guidelines. <u>Because compliance with § 50.67 is demonstrated by calculating the TEDE, the relative contribution of the thyroid dose to the demonstration of compliance with the control room criterion has been substantially and appropriately reduced</u>.

In addition, many licensees that continue to use the traditional TID-14844 source term have incorporated the guidance in Regulatory Guide (RG) 1.195, "Methods and Assumptions for Evaluating Radiological Consequences for Design-Basis Accidents at Light-Water Nuclear Power Reactors," to achieve operational flexibility. <u>Following the guidance in RG 1.195, licensees are able to evaluate control room habitability using a 50 rem thyroid dose guideline. This represents a significant relaxation from the 30 rem thyroid dose guideline that was incorporated into previous guidance documents</u>.

(2019) PRM-50-122, RE-SUBMITTAL - 10 CFR 2.802 PETITION FOR RULEMAKING ACCIDENT DOSE CRITERIA

Problem Description:

The U.S. Nuclear Regulatory Commission's (NRC's) design basis accident (DBA) dose criteria and the resulting design of accident mitigation systems could be perceived to emphasize protection of the control room operator over protection of the public. The control room criterion restricts the calculated 30-day accident dose to the annual occupational limit of five rem while the off-site dose criteria allows for a calculated dose of 25 rem in two hours. The off-site dose criteria were derived from the siting practices of the earliest reactors and are not reflective of current health physics knowledge or modern plant construction. As a result, the design of accident mitigation systems may not be optimized in the best interest of NRC's mission of protecting public health and safety. The control room accident dose criterion has proven to be challenging to demonstrate with most plants having very little margin to the regulation.

Proposed Solution:

The proposed voluntary rule would allow licensees to adopt revised accident dose criteria that will; (1) be reflective of modern health physics recommendations and modern plant designs, (2) provide a better balance between protection of the control room operator and protection of the public, and (3) relieve the unnecessary regulatory burden associated with meeting the current control room dose criterion.

The attached petition includes the history of the current dose criteria, proposed changes to § 50.67 Accident source term and General Deign Criterion 19, corresponding revisions to Regulatory Guide 1.183, Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors, as well as other supporting information.

The petitioner has attempted to gain support from NRC staff to initiate rulemaking through internal processes for over ten years without success. The referenced dose criteria are codified in NRC regulations. Since internal processes such as the Non-Concurrence Process and the Differing Profession Opinion process are not applicable to concerns with regulations, the petitioner reluctantly submits the attached § 2.802 Petition for rulemaking as an individual.

PURPOSE:

The U.S. Nuclear Regulatory Commission's (NRC's) design basis accident (DBA) dose criteria and the resulting design of accident mitigation systems could be perceived to emphasize protection of the control room operator over protection of the public. The control room criterion restricts the calculated 30-day accident dose to the annual occupational limit of five rem while the off-site dose criteria allows for a calculated dose of 25 rem in two hours. DBA dose criteria should not be viewed as representing actual doses received by individuals but rather as figures of merit which have a direct impact on the design of structures, systems and components (SSCs) important to safety. The off-site dose criteria were derived from the siting practices of the earliest reactors and are not reflective of current health physics knowledge or modern plant construction. As a result, the design of accident mitigation systems may not be optimized in the best interest of NRC's mission of protecting public health and safety. The control room accident dose criterion has proven to be challenging to demonstrate with many plants having very little margin to the regulation.

The purpose of this petition is to identify concerns with current DBA dose criteria and to recommend a proposed voluntary rule allowing licensees to adopt revised accident dose acceptance criteria that will; (1) be reflective of modern health physics recommendations and modern plant designs, (2) provide a better balance between protection of the control room operator and protection of the public, and (3) relieve the unnecessary regulatory burden associated with meeting the current control room dose criterion.

SUMMARY:

During the 1950s, applicants for reactor construction permits submitted Hazards Summary Reports to the Atomic Energy Commission (AEC) describing the potential dose consequences from what was considered the "maximum credible accident."[1] These evaluations contained wide variations in both the assumed source terms as well as the proposed dose acceptance criteria. In response to the recognition that more definitive siting criteria was needed, the AEC developed a procedural methodology to define reactor siting criteria that was generally consistent with the siting practices in effect at the time. There was a concern within the AEC that it was premature to codify these criteria so early in the development of the nuclear power industry. Notwithstanding this concern, in 1962, the AEC published 10 CFR Part 100, "Reactor Site Criteria", specifying dose acceptance criteria of 25 rem whole body and 300 rem thyroid for a 2 hour period at the Exclusion Area Boundary (EAB) and for the accident duration at the outer boundary of the Low Population Zone (LPZ).

The stated objective of the reactor siting criteria was to avoid serious injury to individuals if an unlikely, but still credible, accident should occur. Both the 25 rem criterion and the concept of an exclusion area addressed the potential for extreme radiological hazards that would exist if a fuel melt source term was released into an unshielded containment[2]. The regulation states that the 25 rem whole body corresponds to the once-in-a-lifetime accidental or emergency dose for radiation workers which according to 1959 national council on radiation protection (NCRP) recommendations may be disregarded in the determination of their radiation exposure status[3]. There is no analogous citation for the 300 rem thyroid dose criterion which was not the dose equivalent to 25 rem whole body. Radiation protection standards at the time would have suggested a 6:1 ratio of thyroid to whole body dose *(resulting in 150 rem) so the 300 rem was somewhat arbitrary. The codification of site criteria fulfilled the need to reduce the subjective nature of judging site suitability while providing a methodology that did not conflict with siting decisions already made by the AEC. The regulation was intended to be an interim measure until the state-of-the-art allowed for more definitive standards to be developed.*

In 1971 Appendix A, "General Design Criteria for Nuclear Power Plants," was added to 10 CFR Part 50. General Design Criterion 19 (GDC-19) specified that adequate protection shall be provided to permit access and occupancy of the control room for the duration of an accident without exceeding a radiation exposure of 5 rem whole body or its equivalent to any part of the body. The originally stated objective for the 5 rem control room accident dose criterion is not readably traceable however the NRC staff believes that the primary objective of the criterion was to provide a safe, comfortable environment that would enable the control room operators to focus attention on accident mitigation. The numerical value chosen fulfilled this objective however the alignment of the control room accident dose criterion with the annual limit for occupational dose has been an ongoing challenge for licensees. The 5 rem control room dose criterion is limiting for many licensees and this raises the question regarding whether a slightly higher value could still satisfy the objective of providing a comfortable environment for the operators while reducing regulatory burden by increasing the small margin many licensees have relative to the current acceptance criterion.

In the late 1970s there were concerns within the NRC that siting practices were not providing enough emphasis on site isolation as an important contributor to defense-in-depth because engineered safety feature (ESF) systems could be designed to make almost any site acceptable from an accident dose calculation point of view. In August 1978, the NRC directed the staff to develop a general policy statement on nuclear power reactor siting which resulted in NUREG- 0625, "Report of the Siting Policy Task Force," recommending that fixed distances should be required for the EAB and the LPZ in lieu of dose consequence analyses. After numerous comments objecting to a proposed rule (57 FR 47802), which was based on NUREG-0625 recommendations, <u>the commission decided to retain source term and dose calculations by relocating a new single dose criterion based on total effective dose equivalent (TEDE) in 10 CFR 50.34 (61 FR 65157 December 11, 1996</u>).

<u>The new TEDE criterion is applicable to all new reactors and existing reactors that choose to adopt the alternative source term (AST) methodology</u>. <u>Depending on the contribution to TEDE dose from iodine in the released source term, the 25 rem TEDE criterion allows for the associated thyroid dose to substantially exceed the previously controlling 300 rem thyroid limitation. Therefore, new reactors are being sited with a less restrictive dose criterion than the earliest reactors</u>.

Modern health physics recommendations suggest that a dose of 25 rem is difficult to justify as adequately fulfilling the objective of not causing serious harm especially when considering the most dose-sensitive members of the public. The same health physics recommendations indicate that the 5 rem control room dose criterion may be overly restrictive.

Therefore, it is recommended that a uniform design basis accident dose criterion of 10 rem TEDE for the control room, EAB, and LPZ boundary be available to licensees on a voluntary basis. Adoption of this voluntary rule would result in a less restrictive control room dose criterion while significantly strengthening the offsite dose criterion. This voluntary change would provide various benefits in that:

(1) it is technically defensible based on modern health physics guidance indicating that an increased cancer risk is not expected for exposures below 10 rem;

(2) it would avoid the poor optics of allowing a higher design basis dose criterion for members of the public (including the most dose-sensitive groups such as children and pregnant women) than for highly trained nuclear professionals occupying the control room;

(3) it would motivate licensees to provide greater emphasis on offsite dose reduction commensurate with NRC's mission to protect public health and safety; and

(4) <u>it would reduce the regulatory burden required to demonstrate the unnecessarily restrictive 5 rem control room dose criterion</u>.

A significant number of plants would be able to meet a uniform 10 rem TEDE dose criterion without making any changes to their dose consequence analyses. <u>Those plants whose existing DBA dose analyses would be challenged by a 10 rem TEDE dose criterion may be able to increase the credit taken for mitigation systems designed to limit releases to the environment while achieving an increased margin in their control room dose analyses. However, no action on the part of any licensees would be required since the proposed rule presented herein would be available for adoption on a voluntary basis.</u>

NUREG-1433 Standard Technical Specifications — General Electric Plants (BWR/4): Specifications (Revision 4, Volumes 1 and 2)

SR 3.6.1.3.13: "Verify leakage rate through each MSIV is ≤ [11.5] scfh when tested at ≥ [28.8] psig."

SR 3.6.1.3.13 The analyses in References 1 and 8 are based on leakage that is less than the specified leakage rate. Leakage through each MSIV must be ≤ [11.5] scfh when tested at ≥ Pt ([28.8] psig). A Note is added to this SR which states that these valves are only required to meet this leakage limit in MODES 1, 2, and 3. In the other conditions, the Reactor Coolant System is not pressurized and specific primary containment leakage limits are not required. This ensures that MSIV leakage is properly accounted for in determining the overall primary containment *leakage rate. The Frequency is required by the Primary Containment Leakage Rate Testing Program.*

INSPECTION MANUAL CHAPTER 0308 ATTACHMENT 1 TECHNICAL BASIS FOR PERFORMANCE INDICATORS Effective Date: 01/01/2021

Performance Indicator: Reactor Coolant System Leakage

Cornerstone: Barrier Integrity

Objective: This indicator monitors the integrity of the RCS pressure boundary, the second of the three barriers to prevent the release of fission products. It measures RCS Identified Leakage as a percentage of the technical specification allowable Identified Leakage to provide an indication of RCS integrity.

Revision of Regulatory Guide 1.183 "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors"

Mark Blumberg, Senior Reactor Engineer (Technical Lead)
NRR/DRA/ARCB
mark.blumberg@nrc.gov

Micheal Smith, Health Physicist (Project Lead)
NRR/DRA/ARCB
micheal.smith@nrc.gov

March 5, 2021

ML21056A058

Agenda

- Key Messages
- Background
- Regulatory Guide (RG) Update Process
- RG 1.183 Guidance Proposed Actions
- Looking Forward
- Feedback/Discussion
- Comments and input from the public

Key Messages

- The NRC staff has restarted efforts to revise RG 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors."

- The objectives of the revision are to:
  - incorporate lessons learned from recent NRC staff reviews of Alternative Source Term (AST) and Main Steam Line Isolation Valve (MSIV) leakage LARs;
  - incorporate relevant operating experience as well as recent post-Fukushima seismic risk insights and walkdowns;
  - respond to change of regulatory environment (e.g., backfit guidance SRM-SECY-18-0049 & NuScale SRM-SECY-19-0036);
  - make the guidance more useful by considering feedback and comments from licensees;
  - ensure sufficient guidance is in place for licensing advanced light-water reactors (LWRs), accident tolerant fuel (ATF), high-burnup, and increased enrichment fuel; and,
  - incorporate insights from new research activities.

Key Messages (Cont'd)

- NRC staff expects for RG 1.183 Rev. 0 and Rev. 1 to co-exist as a result of SRM-SECY-18-0049, "Management Directive and Handbook 8.4, Management of Backfitting, Issue Finality, and Information Collection."
- NRC staff will hold additional public meetings as necessary for external stakeholder engagement on the revision of RG 1.183.
- Publish the draft RG for comment in 4th Quarter CY 2021.
- Final revised RG being issued in 2nd Quarter CY 2022.

Background

Background

- Origin: Footnote to 10 CFR 100.11(a) is a performance-based rule to evaluate the defense-in-depth provided by the containment.
  - TID-14844 Source term provided guidance which assumed the source term is instantaneously available in the containment.

- Radionuclide behavior observed during the TMI accident did not appear at all similar to the TID-14844 source term.
  - NRC initiated research effects in the area of severe accidents which culminate in publication of NUREG-1150.
  - NUREG-1465 source term was derived from the sequences in NUREG-1150.
    - RG 1.183 Rev. 0 adopted the NUREG-1465 early in-vessel fuel melt source term.

Background (cont'd)

- NRC staff developed RG 1.183 Rev. 0 (July 2000) to support implementation of 10 CFR 50.67, "Accident source term"

- RG 1.183 Rev. 0 is applicable to nuclear power reactor applicants and licensees adopting 10 CFR 50.67
  - Limited range of applicability on Non-LOCA release fractions

- RG 1.183 Rev. 0 identified the significant attributes of an acceptable accident AST based on NUREG-1465, "Accident Source Terms for Light-Water Nuclear Power Plants" (1995)

- RG 1.183 Rev. 0 provides assumptions and methods that are acceptable to the NRC staff for performing design basis radiological analyses using an AST

DG-1199

- In October 2009, the NRC issued for public comment DG-1199 as a proposed Rev. 1 of RG 1.183.
- Staff received 150 public comments
- The reasons for revision of RG 1.183 in DG-1199 were:
  – Providing additional guidance for modeling BWR MSIV leakage,
  – Expand applicability of Non-LOCA release fractions to support modern fuel utilization,
  – Extending the applicability of the proposed RG for use in satisfying the radiological dose analysis requirements contained in 10 CFR Part 52 for advanced LWR design and siting,
  – Providing additional meteorological assumption guidance.

Modern Fuel Utilization

- Since DG-1199 was issued for public comment, NRC issued several license amendments to support modern fuel utilization.
  – Oconee Units 1, 2, and 3 (2019)
  – Shearon Harris (2018)
  – H.B. Robinson (2017)
  – Catawba Units 1 and 2, McGuire Units 1 and 2, Oconee Units 1, 2, and 3 (2016)
  – Diablo Canyon Units 1 and 2 (2015)
- Reinforce need for expanded Non-LOCA release fractions

2019 License Amendment Requests

- In 2019, NRC received several AST LARs requesting increased MSIV leakage
- As a result, work on DG-1199 was postponed to allow NRC staff to incorporate lessons learned, from evaluation of the LARs, into the revised RG 1.183:
  - James A. FitzPatrick Amendment No. 338 for AST, July 21, 2020 (ML20140A070)
  - Quad Cities Nuclear Power Station, Units 1 & 2 – Amendment Nos. 281 and 277 to increase allowable MSIV leakage, June 26, 2020 (ML20150A328)
  - Nine Mile Point Nuclear Station, Unit 2 – Amendment No. 182 to change allowable MSIV leak rates, October 20, 2020 (ML20241A190)
  - Dresden Nuclear Power Station, Units 2 & 3 – Amendments Nos. 272 and 265 to increase allowable MSIV leakage, October 23, 2020 (ML20265A240)

Regulatory Guide Update Process

Regulatory Guide Update Process

- Identify which RGs need to be revised based on:
  - Rulemakings
  - Lessons learned
  - Stakeholder feedback
  - Periodic reviews
- Develop draft RG through internal collaboration
- Draft RG available for public comment (4th Quarter CY 2021)
- Internal staff comment resolution
- Finalize RG package for OGC and ACRS review
- Issue final RG (2nd Quarter CY 2022)

RG 1.183 Guidance Proposed Actions

Additional Method for Aerosol Deposition Models

- Staff is considering an additional method for aerosol deposition models
- Staff is addressing issues in RIS 2006-04, "Experience with Implementation of Alternative Source Terms" (considering reconstitution of the AEB-98-03 settling velocity modeling parameters and reviewing the "multigroup method" to address changing settling velocity distributions).
- Regulatory position in Rev. 0 continues to be acceptable. As a result, RG 1.183 Rev. 0 and Rev. 1 are expected to co-exist.
- Over the last 10 years no applicant or licensee has adopted the methodology from SAND2008-6601, "Analysis of Main Steam Isolation Valve Leakage in Design Basis Accident Using MELCOR 1.8.6 and RADTRAD."
- There have been no communications that applicants or licensees intend to adopt the SAND2008-6601 methodology.
- NRC staff plans to consider stakeholder input/feedback to inform the NRC's decision on what methodologies to include in RG 1.183 Rev. 1.

ATF, HBU, Extended Enrichment (LOCA)

- Update RG 1.183 Tables 1, 2, and 4 which hybridizes accident source term tables from SAND2011-0128, "Accident Source Terms for Light Water Nuclear Power Plants Using High-Burnup of MOX Fuel," utilizing the maximum release fractions from the low burnup and high burnup tables.
  - Expanded to encompass near-term ATF design concepts[1] fuel burnup extension to 59 GWD/MTU max assembly-averaged discharge burnup (~68 GWd/MTU peak rod-average) and ^{235}U enrichments up to 8.0 wt%.
  - Staff finds that the extension from 62 GWd/MTU from SAND2011-0128 to 68 GWd/MTU is appropriate[1]. The SAND2011-0128 calculations used MELCOR 1.8.5 for accident progression and ORIGEN for radionuclide and decay heat inventories.
  - Provide conditions and limitations of the report applicability for regulatory purposes.
  - Considering impact of FFRD for the Appendix A assumptions.
  - Not applicable for MOX Fuel and long-term designs concepts (doped UO_2, coated Zirc-cladding, FeCrAl cladding are considered near-term ATF concepts).

1- NRC Memorandum, "Applicability of Source Term for Accident Tolerant Fuel, High Burn Up and Extended Enrichment," dated May 13, 2020, ADAMS Accession Number ML20126G376

ATF, HBU, Extended Enrichment (LOCA) Cont'd

- Initial research efforts are underway to update the SAND2011-0128 accident source term to accommodate higher burnup and increased enrichments for LOCA releases. However, completion of the updated analyses may not be finished before the update to the regulatory guide.
  - **What burnup and enrichment targets are the industry pursuing for PWR and BWR?**
  - **Is there readily available data, studies, and/or analyses which could be useful for NRC review? Note, this is not a request to perform experiments, studies, or analyses.**
- Changes to facility analyses of record must represent those design changes being implemented. For instance, "swapping margin" from atmospheric dispersion data to justify increased radionuclide inventories due to higher reactor core burnups and/or increased fuel enrichments will not be acceptable.
  - **What licensing challenges within the DBA radiological consequence analyses do vendors and licensees foresee before loading these new fuel types (e.g. the design criterion, additional capital improvements)?**

ATF, HBU, Extended Enrichment (LOCA) Cont'd

- **Current draft**: Hybridizes accident source term tables from SAND2011-0128 to utilize most conservative release fractions and timing between the high- and low burnup recommendations.
  - SAND 2011-0128 LBU: 26-38 GWD/MTU discharge burnup, which varied depending on the plant analyzed.
  - SAND 2011-0128 HBU: 59 GWD/MTU max assembly-averaged discharge burnup (~ 62 GWD/MTU peak rod-averaged burnup).
- **Reasoning**: Different radionuclide abundances peak at different burnups throughout the operating cycle. For a facility operating at the 62 GWD/MTU peak rod-averaged burnup envelope, it would therefore be reasonable to select peak abundances which bound potential releases at mid- and end points of the operating cycle.

ATF, HBU, Extended Enrichment (LOCA) Cont'd

Example of a hybridized RG 1.183 Rev. 1, Table 1, BWR Core Inventory Fraction Released into Containment Atmosphere:

Group	Gap	Early In-Vessel Phase	Total
Noble Gases	0.008	0.96	0.968
Halogens	0.003	0.54	0.543
Alkali Metals	0.003	0.14	0.143
Tellurium Metals	0.003	0.39	0.393
Barium, Strontium	0	0.005	0.005
Noble Metals	0	0.0027	0.0027
Cerium Group	0	1.6E-7	1.6E-7
Lanthanides	0	2.0E-7	2.0E-7
Molybdenum	0	0.03	0.03

Example of a hybridized RG 1.183 Rev. 1, Table 4, LOCA Release Phases:

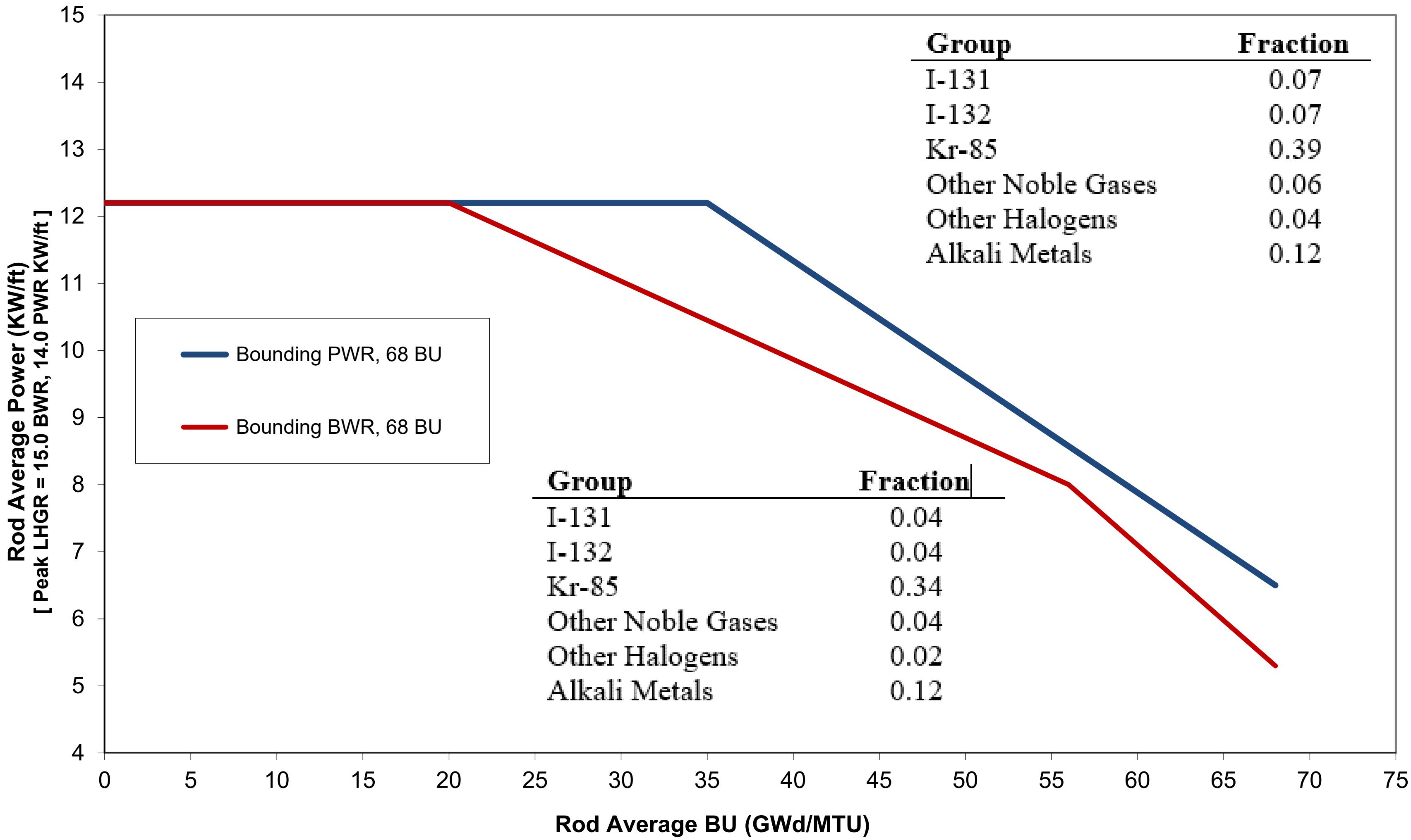
Phase	PWRs		BWRs	
	Onset	Duration	Onset	Duration
Gap Release	30 sec	0.22 hr	2 min	0.16 hr
Early In-Vessel	0.22 hr	4.5 hr	0.16 hr	8.0 hr

These are preliminary examples, not regulatory guidance.

Planned Updates for Non-LOCA Release Fractions

1. Expanded applicability to 68 GWd/MTU rod average burnup
2. Separate BWR and PWR release fractions
3. Burnup-dependent transient FGR correlations for prompt power increase accidents
4. Analytical procedure for calculating revised Non-LOCA release fractions

Draft Non-LOCA Release Fractions

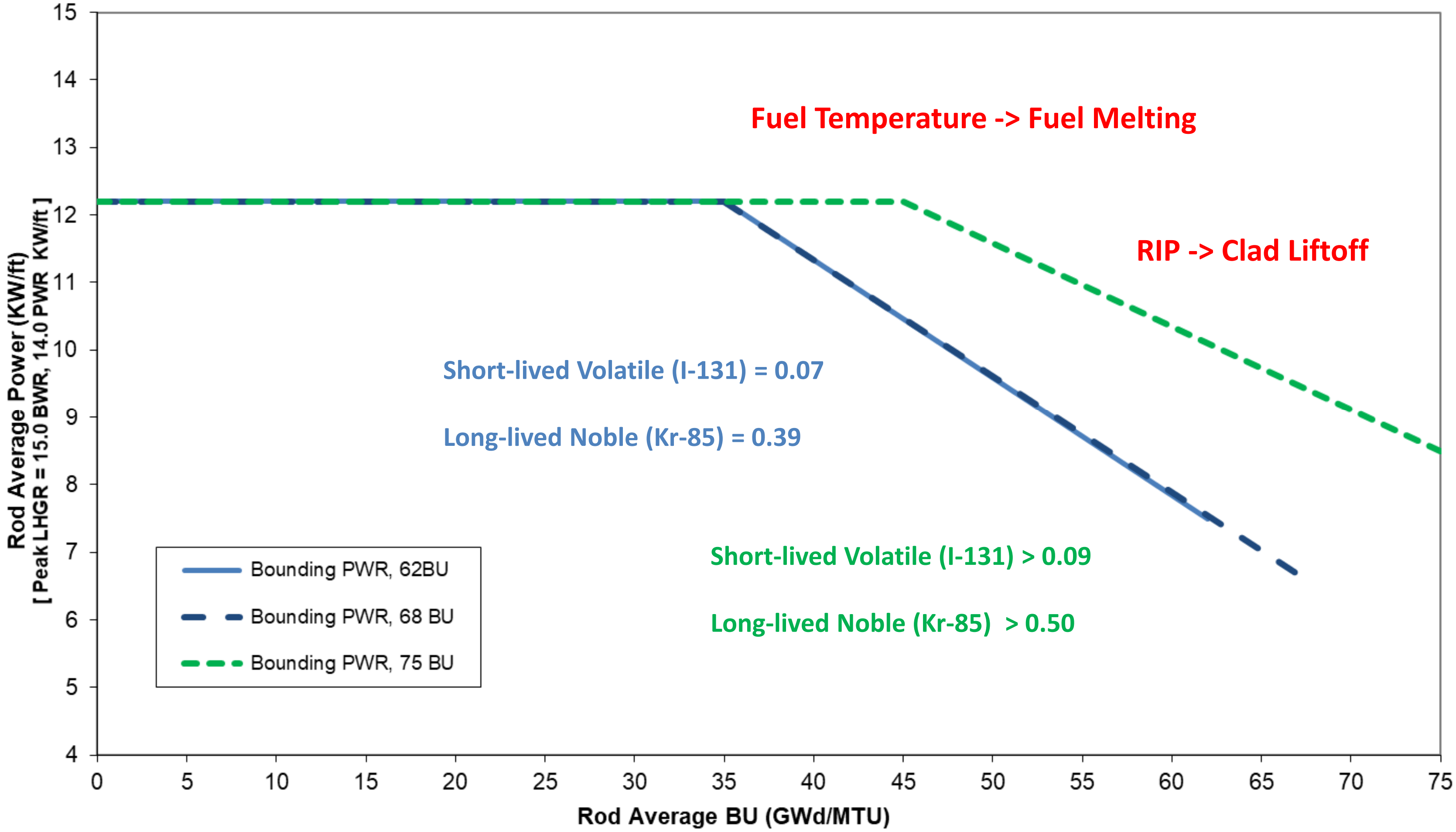


Group	Fraction
I-131	0.07
I-132	0.07
Kr-85	0.39
Other Noble Gases	0.06
Other Halogens	0.04
Alkali Metals	0.12

Group	Fraction
I-131	0.04
I-132	0.04
Kr-85	0.34
Other Noble Gases	0.04
Other Halogens	0.02
Alkali Metals	0.12

Disclaimer: These are preliminary examples, not regulatory guidance.

Impact of Burnup Extension on Non-LOCA Release Fractions



Disclaimer: These are preliminary examples, not regulatory guidance.

Open Items for Non-LOCA Release Fractions

1. Expanded PWR/BWR rod power profiles for 75 GWd/MTU?
2. How to address BWR part-length fuel rods?
   – Treat PLRs as FLRs for dose assessments
3. How to address doped UO_2 fuel pellets?
   – Vendors demonstrate applicability by showing FGR is equivalent or lower than standard UO_2
4. How to address IFBA fuel pellets?
   – Licensees confirm that power setbacks ensure equivalent or lower FGR than standard UO_2

Revised Fuel Handling Accident

- Accidents during refueling operations continue to be creditable and thus need to be evaluated despite their low safety significance.

- Appendix B will be revised to reflect the Revised FHA analysis.[1]
  - Provides regulatory relief and operational flexibility when considering ATF, high burn-up and increased burnup fuels.

- "Mixing models" between Rev. 0 and Rev 1 will not be accepted. Both Rev. 1 iodine transport steps (initial bubble rise and re-evolution) must be modeled.

1 - Memo from RES to NRR, "Closeout to Research Assistance Request for Independent Review of Regulatory and Technical Basis for Revising the Design-basis Accident Fuel Handling Accident," November 23, 2019 (ML19270E335)

Use of Risk and Engineering Insights
Seismic Credit

- Staff is exploring a streamlined approach for quantitative credit for hold-up or retention of MSIV leakage within the power conversion system for BWRs.
- Technical assessment considering 20+ years of operational and seismic risk insights supports seismic ruggedness of power conversion system.
- Extension of leakage aerosol deposition methodologies to steam line downstream of MSIV.
- Additional hold-up or retention may be credited in power conversion system with designated and evaluated pathway (e.g., drain to the main condenser).

Consideration of MSIV Leakage Values

- NRC has approved MSIV leakage of 200 scfh or below per MSIV with a total MSIV leakage of 400 scfh or below. Higher values will be considered on a case-by-case basis with sufficient justification.

- Maintaining MSIV leakage at or below certain values is based on the following considerations:
  - Leakage in excess of 200 scfh per MSIV could be indicative of substantial valve defects
  - These values represent maximum values in existing fleet
  - 400 scfh is on the order of total containment leakage
  - Comparison to original design value of 11.5 scfh per MSIV

Lessons Learned from Licensing Reviews

- Staff are considering the following clarifications:
  - augmenting the expectations for containment spray in BWR drywells/containments as follows: (i.e. Rev. 0 Appendix A Assumption 3.3)

    "In addition, since spray droplets are assumed to be ineffective once they impact a structure, the obstructions present in drywells and containments (particularly in BWR Mark I and Mark II Drywells) should be considered in the determination of decontamination factors and removal coefficients credited for the drywell or containment."

  - augmenting the expectations for performing and using sensitivity analysis as follows: (i.e., Rev. 0, RP 1.3.3)

    "Sensitivity analyses should avoid the inclusion of well-defined parameters such as atmospheric dispersion factors based on site specific data."

Lessons Learned from Licensing Reviews Cont'd

- GDC 19 Control room specifies access and occupancy:
  - The TEDE analysis should consider all sources of radiation that will cause exposure to control room personnel *during access and occupancy*.

- Staff are considering whether to clarify:
  - the expectations for BWR MSIV Leakage LOCA analysis assumptions with respect to pipe breaks

Additional Considerations

- Revising footnote 7 which provides an incorrect method to convert thyroid dose to TEDE to read as follows:

"In performing screenings and evaluations pursuant to 10 CFR 50.59, it may be necessary to compare dose results (figures-of-merit) expressed in terms of whole body and thyroid with results expressed in terms of TEDE. Each figure-of-merit represents different systems of dosimetry (e.g. ICRP 2 and ICRP 26/30) which have recognized dose-conversion-factors specifically designed to compute them. There is no methodology which converts between these systems. When performing 50.59 evaluations, the figure-of-merit of interest must be computed with the appropriate dose-conversion-factors."

Expected General Updates

- The NRC staff expects for RG 1.183 Rev. 0 and Rev. 1 to co-exist.
- In addition to items discussed earlier, NRC plans to include changes proposed in DG-1199 as modified by public comments.
  - Incorporate updates, new or withdrawn regulatory guidance (i.e., RG 1.194 (meteorology)).
  - Guidance for modern fuel utilization (non-LOCA gap fractions).
  - Changes due to Regulatory Information Summaries (i.e., 06-04, 01-19).
  - Lessons learned from license reviews (i.e., clarify DFs and containment isolation as used in the FHA).
  - Clarify TEDE calculation terminology (i.e., EDEX vs. EDE).
  - Remove environmental qualification guidance from RG and refer to RG 1.89.

Looking Forward

- Consider feedback from stakeholders
- Continue development of updated draft RG 1.183 Rev. 1
- Draft RG 1.183 Rev. 1 issued for public comment (4th Quarter CY 2021)
- Hold additional public meetings as necessary prior to the end of public comment period
- Staff review and disposition of public comments
- Update draft RG 1.183 Rev. 1 as necessary
- ACRS and OGC review of final draft (1st Quarter CY 2022)
- Issuance of RG 1.183 Rev. 1 (2nd Quarter CY 2022)

Discussion/Feedback

Questions/Comments?

Mark Blumberg, Senior Reactor Engineer (Technical Lead)
NRR/DRA/ARCB
mark.blumberg@nrc.gov

Micheal Smith, Health Physicist (Project Lead)
NRR/DRA/ARCB
micheal.smith@nrc.gov

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Analysis of Main Steam Isolation Valve Leakage in Design Basis Accidents Using MELCOR 1.8.6 and RADTRAD

Randell O. Gauntt, Tracy Radel, Michael A. Salay, and Donald A. Kalinich

Prepared by
Sandia National Laboratories
Albuquerque, New Mexico 87185 and Livermore, California 94550

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Analysis of Main Steam Isolation Valve Leakage in Design Basis Accidents Using MELCOR 1.8.6 and RADTRAD

Randell O. Gauntt, Tracy Radel, Donald A. Kalinich
Reactor Modeling & Analysis Department
Sandia National Laboratories
P.O. Box 5800
Albuquerque, NM 87185-0156

Tracy Radel
Risk & Reliability Analysis Department
Sandia National Laboratories
P.O. Box 5800
Albuquerque, NM 87185-0156

Michael Salay
Office of Nuclear Regulatory Research
United States Nuclear Regulatory Commission
Washington, D.C. 20555-0001

Abstract

Analyses were performed using MELCOR and RADTRAD to investigate main steam isolation valve (MSIV) leakage behavior under design basis accident (DBA) loss-of-coolant (LOCA) conditions that are presumed to have led to a significant core melt accident. Dose to the control room, site boundary and LPZ are examined using both approaches described in current regulatory guidelines as well as analyses based on best estimate source term and system response. At issue is the current practice of using containment airborne aerosol concentrations as a surrogate for the in-vessel aerosol concentration that exists in the near vicinity of the MSIVs. This study finds current practice using the AST-based containment aerosol concentrations for assessing MSIV leakage is non-conservative and conceptually in error. A methodology is proposed that scales the containment aerosol concentration to the expected vessel concentration in order to preserve the simplified use of the AST in assessing containment performance under assumed DBA conditions. This correction is required during the first two hours of the accident

while the gap and early in-vessel source terms are present. It is general practice to assume that at ~2hrs, recovery actions to reflood the core will have been successful and that further core damage can be avoided. The analyses performed in this study determine that, after two hours, assuming vessel reflooding has taken place, the containment aerosol concentration can then conservatively be used as the effective source to the leaking MSIV's. Recommendations are provided concerning typical aerosol removal coefficients that can be used in the RADTRAD code to predict source attenuation in the steam lines, and on robust methods of predicting MSIV leakage flows based on measured MSIV leakage performance.

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Introduction

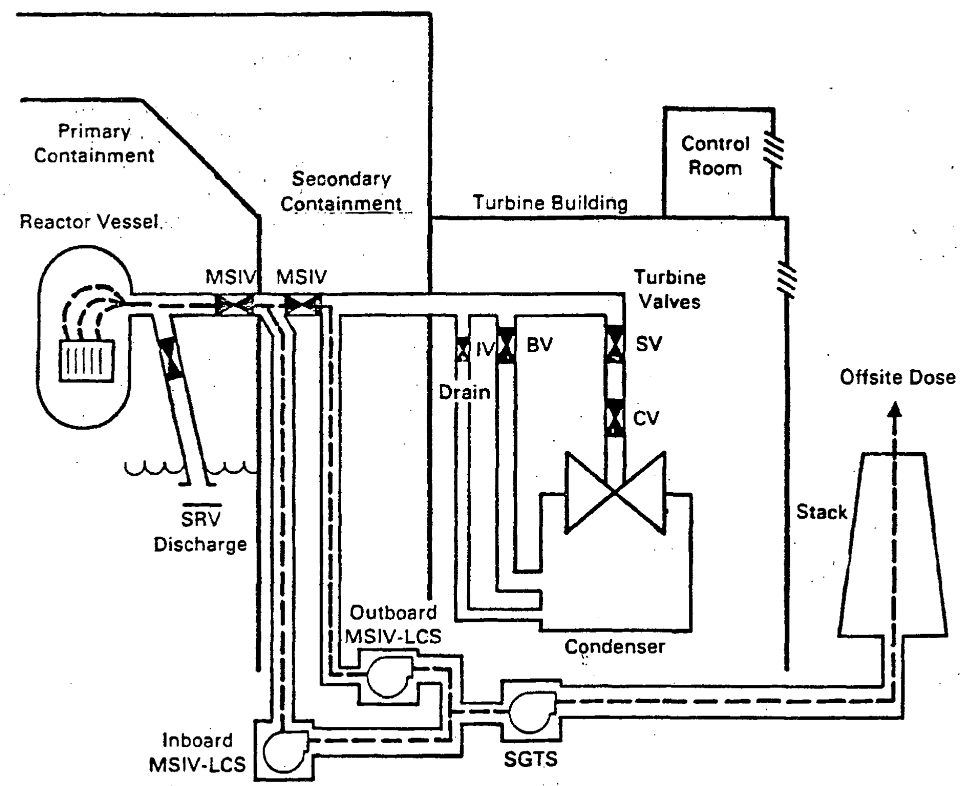
The focus of this work is to evaluate current practices and propose revisions as necessary for the technical basis and regulatory requirements concerning main steam isolation valve (MSIV) performance for boiling water reactors (BWRs) under accident conditions. Current regulatory guidelines for evaluating MSIV performance are described in USNRC Regulatory Guide 1.183 [1] titled “Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors.” This regulatory guide addresses the use of the NUREG-1465 [2] alternative source term (AST) in the evaluation of containment performance under design basis accidents (DBAs)*. This guideline articulates a defense in depth principal by prescribing radiological containment requirements for hypothetical core damage accidents resulting from a design basis event, such as a loss of coolant accident (LOCA). The defense in depth aspect follows from the assumption that core damage with significant release of fission products results from a design basis accident, where, by definition, a design basis accident is an event wherein safety systems are designed to preclude just such a core melt event. The requirements for this safety approach are prescribed in the Code of Federal Regulations under title 10, parts 50 and 100 addressing allowable radiological dose to the reactor control room and to the site boundary. The federal code prescribes that a core damage event involving significant release of fission products from the core must be considered in the design of containment systems. Note that Reg. Guide 1.183 addresses many aspects of containment performance for both pressurized and boiling water reactors, and that MSIV performance in BWR’s is only one aspect of the regulatory guide’s scope.

Motivating this study is a long-standing technical question as to the applicability of the alternative source term prescription, being a stylized source term of radionuclides to the reactor *containment*, to the evaluation of radionuclide releases through MSIV leakage, since the main steam lines are directly connected to the reactor *vessel*, not the containment. Some background leading up to this issue will now be provided.

* It should be pointed out that the regulatory guide only references the portion of NUREG-1465 associated with the gap and early in-vessel release periods and does not employ the entirety of the NUREG-1465 source term.

1 Background

Boiling water reactors operate by boiling water in direct contact with the Zircaloy-clad reactor fuel rods and passing this steam directly through the power turbines by means of large main steam lines as shown in Figure 1-1. Because of this potential direct pathway from the core region to the environment, two main steam line isolation valves, one inboard of the containment boundary and one outboard, are included on each steam line to isolate the containment boundary from the environment in the event of a core damage accident. Anticipating some leakage from these MSIVs, a leakage control system (LCS) is often included to pull off leakage through the valves and route this effluent to the stacks to reduce on-site dose consequences in the control room and dilute releases from the site. Some licensees have removed previously installed leakage control systems having obtained regulatory relief on MSIV leakage requirements. The main steam line isolation valves are quite large and have a documented history of leaking beyond their design specifications and requiring costly maintenance and overhaul to maintain design specification leak rates [3]. In some cases, credit for fission product deposition in the condenser system has been taken whenever the condenser system has been seismically qualified.



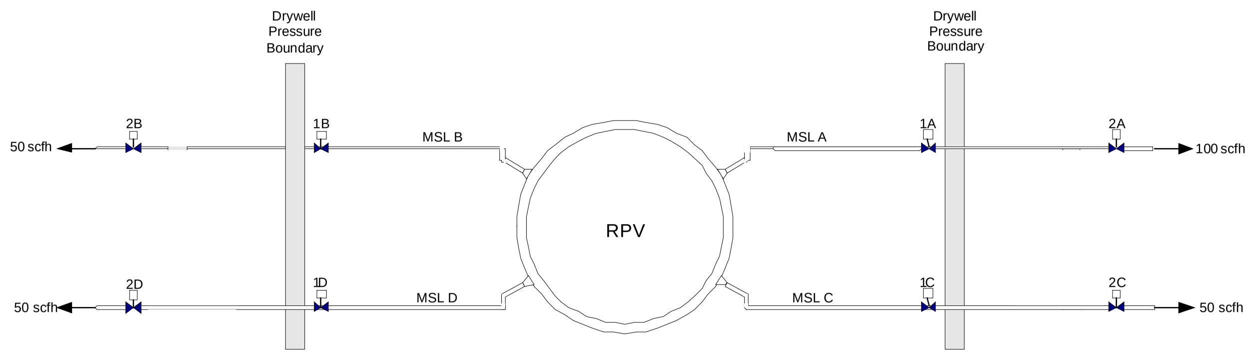


Figure 1-1 Schematic of BWR vessel. steam line and turbine, including location of MSIV's and Leakage Control System (LCS) (the LCS is designed to control leakage from MSIV and vent to stacks).

As seen in the Figure 1-1, the pathway of fission products release from a damaged core to the containment must be either by safety relief valve venting to the wetwell, by venting to the drywell through a break in one of the 4 main steam lines, or more circuitously through a recirculation line or bottom vessel drain line break, depending on the type of design basis accident under consideration. In any event, the source of fission products to the MSIVs is principally from the reactor vessel. Admittedly, if the DBA under consideration is a main steam line break of one steam line between the vessel and the inboard MSIV, then this particular pathway would draw from the drywell volume, while the remaining three intact steam lines would continue to draw from the reactor vessel.

Historically, the leakage of radioactivity through leaking MSIVs has been based on an assumed airborne concentration (Curies/ft^3) of radioactive particles or gas that are available to flow through the valves and a characterized leak rate (ft^3/hr) for the valves. The leak rate of the valves is estimated based on a standard test wherein the space between the inboard and outboard valves (see Figure 1-1) is pressurized to a design specified pressure with air or nitrogen, and a leak rate inferred by the observed gradual depressurization of this intermediate space. The valve design-specified leak rate characterized at standard conditions can then be appropriately scaled to accident conditions to infer a leakage of radioactivity to the environment, providing an appropriate airborne concentration can be determined.

The July 2000 version of Regulatory Guide 1.183 follows previously adopted convention by assuming that the concentration of airborne radioactivity is roughly approximated by the radioactivity released to the containment divided by the containment volume. (Item 6.1 in Appendix A of Reg. Guide 1.183) This idealized view presumes that the radioactivity is released from the fuel, transported out of the vessel, and equilibrated with the drywell volume (and possibly the wetwell volume as well) and that this equilibrated atmosphere subsequently flows through the leaking MSIVs. Additionally, the Regulatory Guide generally permits that natural deposition and aerosol settling processes may be considered in the containment and by implication in regions upstream of outboard MSIVs to reduce in time the airborne activity reaching the MSIVs.

The concept of the drywell space being the source to the MSIVs in the current Regulatory Guidelines can be partially understood in terms of the historical usage of the now-retired TID source term [4], where the release was presumed to be instantaneous. Instantaneously released fission products would be swept by steam advection from the vessel to the drywell where mixing and equilibration with the drywell volume could be expected; however, the major advance introduced by the NUREG-1465 revised source term relative to the TID source term was that the release from fuel was not instantaneous, but instead protracted over time in distinct phases. The NUREG-1465 AST in fact described the time phased release of fission products to the *containment* from the vessel, accounting for the facts that release from the fuel is gradual and occurring over a period of hours, that not all fission products released from the fuel find their way to the containment, some being deposited within the reactor primary system, and that some of these fission products that are retained within the vessel structures can subsequently become re-suspended by revaporization driven by continued decay heating of structures in the vessel.

The NUREG-1465 source characterizes the radioactivity that escapes the fuel and vessel and enters the containment, but does not inform us on the distribution of fission products that have not yet escaped the vessel. In reality, as determined by best estimate analyses, the vessel becomes an ongoing source of airborne radioactivity to the drywell or wetwell (via steam line breaks or SRV venting) as well as the MSIVs on unbroken steam lines as long as release from overheated fuel is taking place and until vessel reflooding and accident recovery takes place. Figure 1-2 illustrates the two conceptual views. After accident recovery and vessel reflooding, the release from the fuel is terminated and the airborne radioactivity in the vessel will be swept into the drywell by the steam generated in the reflooding process, producing vessel airborne concentrations that can be lower than in the drywell region, as illustrated in Figure 1-3.

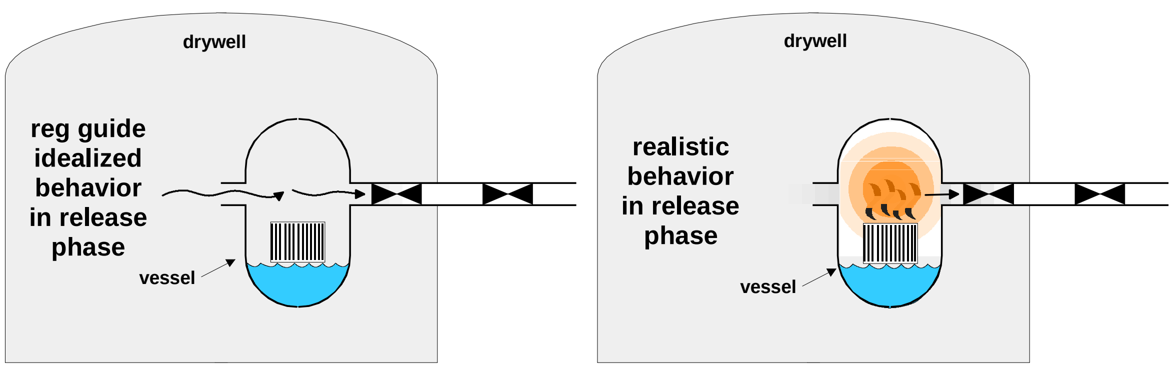


Figure 1-2 Idealized regulatory model of airborne fission products (left) compared to realistic prediction of airborne radioactivity (right) during release phase of a DBA with core damage. Note, source of airborne activity emanates from core (right) more so than the drywell (left).

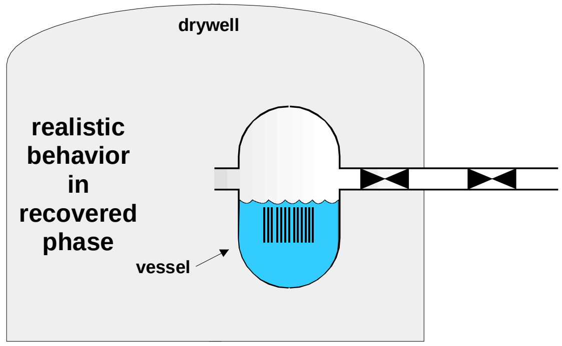


Figure 1-3 Qualitative distribution of airborne activity in post-recovery (reflood) phase of DBA core damage accident.

This misconception or oversimplification in viewing fission product transport from overheated fuel has led to subsequent important conceptual errors in analysis such as proposed use of drywell sprays to reduce airborne radioactivity (as illustrated in Figure 1-4) or equilibrating drywell and wetwell airspace volumes to achieve the same effect, when neither of these processes can directly affect the airborne concentration within the reactor vessel where a continuous source of fission products issues from the overheated fuel. In short, what is needed to evaluate MSIV leakage is a source term to the *reactor vessel steam dome* (which feeds the steam lines) and not a source term to the *containment*.

In order to examine more realistically the behavior of airborne radioactivity in the BWR where design basis accidents with core damage have taken place, the MELCOR code has been applied to make best estimate predictions of fission product release and transport behavior in the vessel and containment systems and the resulting leakage to the environment through leaking MSIVs. Prior to this study two other studies have been performed for Mk-I and Mk-III containment systems (Peach Bottom and Grand Gulf) [5,6]. Using MELCOR 1.8.5, these analyses explored two LOCA DBA scenarios, a recirculation line break and a main steam line break, and compared fractional releases of radionuclides to the environment to those typically produced using the simplified methodology outlined in Regulatory Guide 1.183. Since the MELCOR analyses performed in these studies did not calculate actual dose from released radioactivity, the findings were not conclusive; however, the analyses did indicate that the airborne concentrations in the vessel steam dome could exceed considerably the airborne concentrations in the drywell during the first 2 hours of the accident.

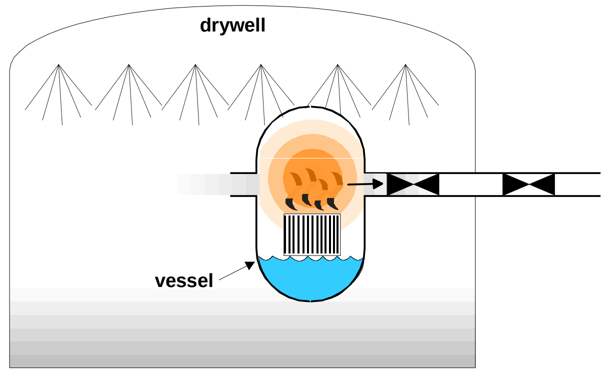


Figure 1-4 Illustration of drywell spray effect on airborne radioactivity in drywell and reactor vessel.

The following sections describe various analyses that have been performed in order to evaluate the conservatism or non-conservatism of the current regulatory guidelines and to establish the technical basis for any recommended revisions of the current regulatory guidelines. The principal new technical information on which these proposed revisions are based are updated full plant MELCOR analyses of two basic design basis accidents. These are described generally in Section 1.1, and in more detail in subsequent sections.

1.1 Full Plant MELCOR Analyses

The present study revisits the analyses reported in references [5,6] using Version 1.8.6 of the MELCOR code with the intent of characterizing releases of radioactivity through leaking MSIVs and then applying these releases in the RADTRAD [7] code in order to calculate the resulting dose to the control room (CR), exclusion area boundary (EAB), and low population zone (LPZ). These calculated doses are then compared to two industry submittals to highlight potential differences by the various methodologies. Analyses are performed using MELCOR 1.8.6 for Mk-I and Mk-III containments for two LOCA DBA scenarios, a recirculation line break and a main steam line break. Both break scenarios are modeled with and without containment spray operation. The effect of including the steam condenser in the release pathway was also investigated with respect to its potential for added radionuclide retention. These analyses are intended to provide a physics-based analysis accident progression, fission product release and transport, and MSIV leakage for the purpose of establishing the technical basis for a simplified regulatory treatment for MSIV performance assessment that corrects for the deficiencies described previously. While these analyses are intended to be essentially “best estimate,” for the purposes of informing regulatory guide recommendations, the analyses are altered in some cases to produce outward flow through the leaking MSIV’s, as described in Section 2.1.1. It is expected that these changes will have a minimal effect on the results. In the remainder of this report, these calculations will be referred to as “full plant analyses” to differentiate them from separate analyses that were performed on the main steam line geometry only. The MELCOR plant models, sequence progression assumptions, and sequence results are described in Section 2.

The MELCOR analyses are characterized as “best estimate” in the sense that physics-based models are used to predict fully integrated and self consistent sequence progression, accounting for primary system thermalhydraulics and thermodynamics associated with LOCA blowdown, core degradation processes such as Zr-steam oxidation and fuel heatup, thermal release of fission products from the fuel, fission product transport and aerosol mechanics. With respect to aerosol mechanics, fission product aerosols are treated using a size-sectional treatment representing the time varying size distribution of the airborne radioactive particles between the limits of 0.1 to 50 micrometers. Treated are agglomeration of smaller particles to form larger particles and a spectrum of deposition processes, including thermophoresis (thermal gradient driven deposition), diffusio-phoresis (steam condensation assisted deposition), and gravitational settling. These phenomena are important in characterizing the transport and deposition behavior of the aerosol particles in terms of so-called "lambda" values (i.e., deposition coefficient) and characterizing these lambda-values for subsequent use in the RADTRAD code are a major focus of this study. The uncertainties in these phenomena are quantified as described in Section 2.1.2 and Section 3.

A principal determination from the full plant analyses is a characterization of the airborne radionuclide concentrations within the vessel upper head region (which feeds the steam lines and MSIVs), and the airborne radionuclide concentration in the drywell (which is used in the current regulatory guidelines as the assumed source to the steam lines and MSIVs). This characterization will be used to scale the drywell airborne concentration determined from the stylized AST (containment source) to a value appropriate for the vessel steam dome. This can be described mathematically as:

$$C_{SD} \approx C_{AST} \times R \qquad (1.1)$$

where

C_{SD} = airborne concentration in the steam dome feeding the MSIV leakage,

C_{AST} = airborne concentration in the containment determined from the NUREG-1465 methodology, and

R = the ratio of the steam dome concentration to the drywell concentration determined by the MELCOR full plant analyses.

A slight complication exists in this approach that must now be described. In the time since NUREG-1465 was issued, current MELCOR best estimate predictions on the timing and magnitude of releases from the core to the containment have changed from that described in the NUREG-1465 prescription. In particular, releases to the containment are now found to be delayed in time and to occur at a lower rate. So, in order to account for this difference between current MELCOR predictions of containment airborne concentrations and those determined by the NUREG-1465 methodology, an additional correction is necessary. This is the ratio of current MELCOR-predicted containment airborne concentrations to the NUREG-1465 predicted containment airborne concentrations. Expressed mathematically,

$$C_{SD} \approx C_{AST} \times R_M \times R^* \qquad (1.2)$$

where,

C_{SD} = airborne concentration in the steam dome feeding the MSIV leakage,

C_{AST} = airborne concentration in the containment determined from the NUREG-1465 methodology,

R_M = the ratio of the steam dome concentration to the drywell concentration determined by the MELCOR full plant analyses, and

R^*= the ratio of NUREG-1465 containment airborne concentrations to MELCOR containment airborne concentrations.

This normalization (R^*) is necessary because NUREG-1465 containment airborne concentrations, when scaled up by MELCOR steam dome to containment concentration ratios, produces an excessively conservative result. This is described in more detail in Section 6, where NUREG-1465 containment concentrations are adjusted in RADTRAD analyses of MSIV leakage to account for the steam dome source effect on dose calculations.

The MELCOR full plant analyses provided the basic sequence and source term progression information for this study; however, it was also desired that uncertainties in aerosol transport and deposition along the pathway through the steam lines and the MSIV's be considered. The transport and deposition behavior of radioactive aerosol, as they move through the steam lines and valves are characterized in terms of depletion rates (removal coefficients, or lambdas) and filtration efficiencies that are used in the RADTRAD code to account for the source term attenuation factors in the dose rate analysis. A quantification of the variability of these attenuation parameters in the steam lines is desired in order that adequate conservatism be reflected in the values used by RADTRAD to calculate doses. To support this objective, an uncertainty characterization of the deposition rates in the steam lines was performed using Monte Carlo methods, as described in the following section.

1.2 MELCOR Main Steam Line Uncertainty Analyses

In addition to the full plant analyses, separate analyses were made on the aerosol behavior in the main steam line portion of the full plant nodalization. These analyses focused on transport and deposition behavior within the main steam lines, accounting for uncertainties in aerosol transport and deposition mechanics. The uncertainty analysis is aimed at determining likely distributions for aerosol removal coefficients, or the so-called "lambda" values used in the RADTRAD code to calculate attenuation of the airborne radionuclide concentrations within each RADTRAD analysis volume of the release pathway. In these analyses, the full plant thermohydraulics and fission product source entering the vessel steam dome region, as calculated in the MELCOR full plant analyses are used as boundary conditions supplied to a reduced nodalization model of the main steam lines. This is done principally for computation efficiency to facilitate Monte Carlo analyses sampling on aerosol mechanics uncertainty. A number of aerosol deposition physics parameters, considered to be uncertain, are considered in these analyses. The principal product of these analyses are distributions for removal coefficients (lambda's) for various segments of the steam lines, including the inboard, intermediate, outboard and steam condenser elements along the release pathway that are modeled by RADTRAD. Because of mixing, convection uncertainties, and re-evolution, the depletion of source aerosol within the steam line segment inboard of the MSIV is neglected – depletion and deposition in this zone is estimated not to reduce the concentration of aerosol reaching the inboard MSIV due to to convective mixing phenomena. These studies are described in Section 3

1.3 RADTRAD Dose Calculations

Finally, RADTRAD analyses were performed using source terms both from the current Regulatory Guide 1.183 AST methodology, and the best estimate MELCOR analyses. In this portion of the study, a proposed correction to the current regulatory guide methodology is investigated that accounts for the fact that the MSIVs draw their radiological source from the vessel steam dome and not the drywell volume. In essence, since the NUREG-1465 AST is a source term to the containment, a correction is proposed where the vessel steam dome concentration is estimated based on a scaling factor, R, applied to the containment (drywell) concentration. This scaling relationship is determined from comparing drywell to vessel steam dome concentrations observed in the full plant MELCOR analyses, as described earlier in Section 1.1.

1.4 Summary of Strategy for Revising Regulatory Guide

The strategy put forth in this study for revising the regulatory guidelines for estimating MSIV leakage is summarized in the following three figures. The current guidelines allow use of a containment source of aerosol determined from the NUREG-1465 AST to evaluate MSIV leakage as illustrated in Figure 1-5. This work points out that it is the source from the vessel that actually determines MSIV leakage, as illustrated in Figure 1-6, where a MELCOR-predicted source from the vessel could in principal be used to evaluate leakage. This study examines the effect of using a MELCOR source term in a RADTRAD analysis in Section 4. Finally, a strategy is proposed that simulates the MELCOR source during the first 2 hours of the accident, followed by a presumed accident recovery through vessel reflooding, after which the source is provided by ingress of the containment activity into the vessel and towards the MSIV's, the vessel and core sourcing having been effectively terminated by the reflooding actions – this is illustrated in Figure 1-7.

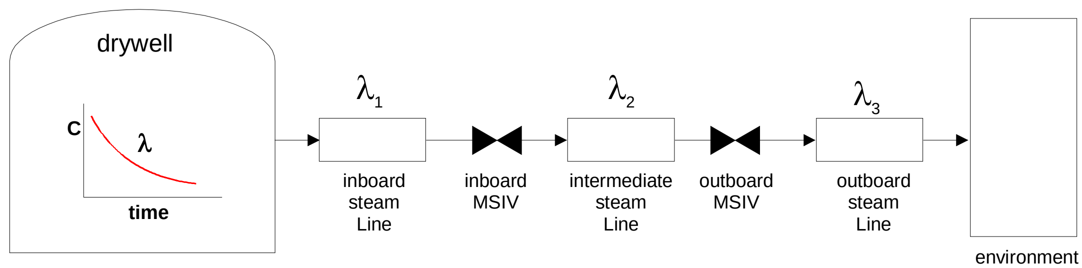


Figure 1-5 Current Reg. Guide 1.183 analysis for dose from leaking MSIV's

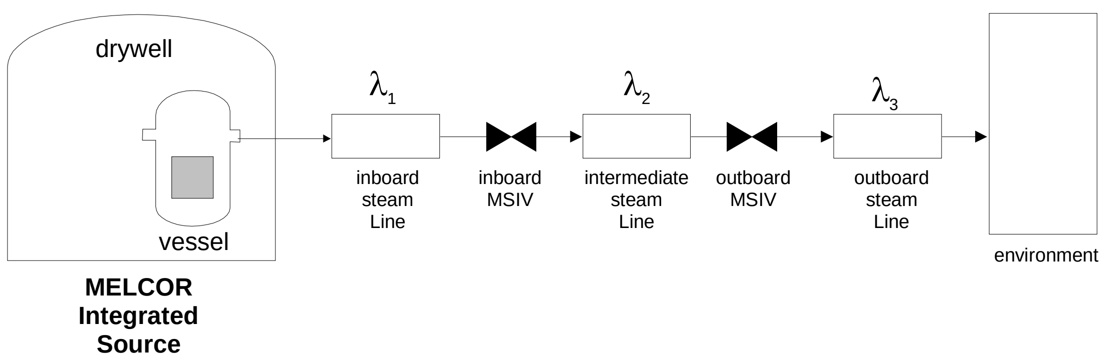


Figure 1-6 RADTRAD analysis using MELCOR source term

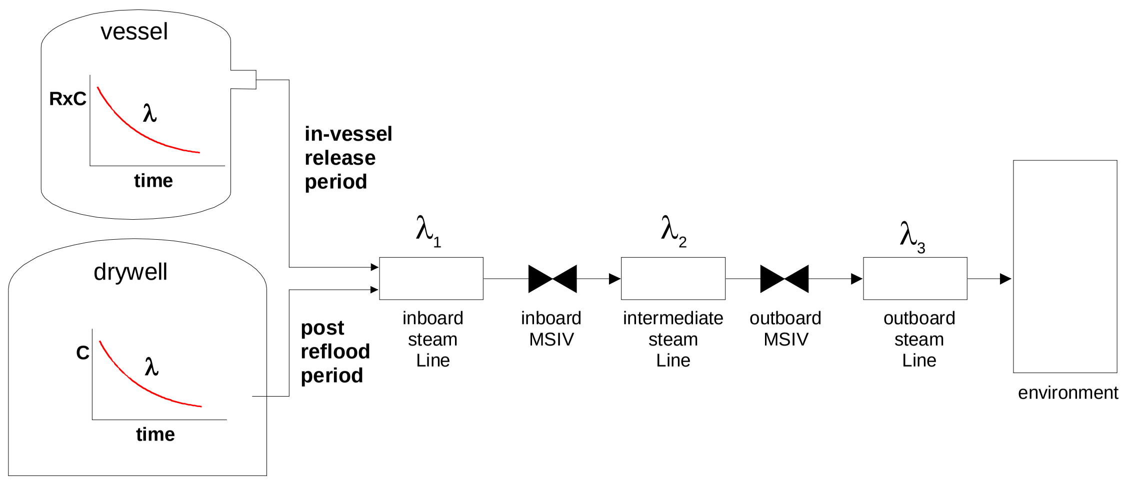


Figure 1-7 Conceptual illustration of modified regulatory methodology to recognize source from vessel steam dome.

1.5 Information Flow Between Models

A variety of different analyses have been performed in this study, sometimes requiring information from one analysis to be used to inform a subsequent analysis. Figure 1-8 illustrates the overall flow of information from each supporting analysis to each dependent subsequent analysis.

One analysis was focused on determining the dose implications of the releases predicted by the full MELCOR sequence analyses, as illustrated by the blue arrows in Figure 1-8. In this analysis, the MELCOR Full Reactor models calculate source term releases (in terms of MELCOR RN classes) to the environment. That information is translated into a form compatible for use by RADTRAD using an Excel worksheet. The RADTRAD code then calculates the doses from this source term using the site specific dispersion (χ/Q) values. This result can be assessed through the time duration for the full plant MELCOR analysis and provide a comparison to doses resulting from a proposed modified regulatory guide methodology.

The MELCOR Full Reactor models also are used to provide the thermo-hydraulic (T-H) information and RN class masses (by aerosol particle size) in the steam dome that are required to run the main steam line (MSL)-only models, as indicated by the red arrows in Figure 1-8. Additionally, the full plant analyses provide information used to scale the drywell airborne radionuclide concentrations appropriately for use in the MSIV leakage analysis. This information is used to establish boundary conditions in the MELCOR MSL-only models.

The MELCOR MSL-only models are run for multiple realizations with various parameters varied (e.g., aerosol physics parameter uncertainty) to ascertain the effects uncertainty on MSL and condenser removal coefficients. Based on the uncertainty analysis results, MSL and condenser removal coefficients are selected.

These removal coefficients are implemented in the RADTRAD MSL-only models. Steam dome-to-drywell radionuclide concentration scaling factors (produced by the MELCOR Full Reactor models) are also implemented in the RADTRAD MSL-only models in order to adjust the containment radioactive airborne concentrations derived from the use of the NUREG-1465 containment source to reflect concentrations in the steam dome that feeds the MSIV's. The RADTRAD MSL-only models are then run to produce doses using BWR core inventory with NUREG-1465 release fractions multiplied by the steam dome-to-drywell radionuclide concentration scaling factors.

Finally, sample analyses for two industry type analyses using RADTRAD and the scaling methodology for use of the AST in MSIV leakage analysis are presented, as indicated by the green arrows in Figure 1-8.

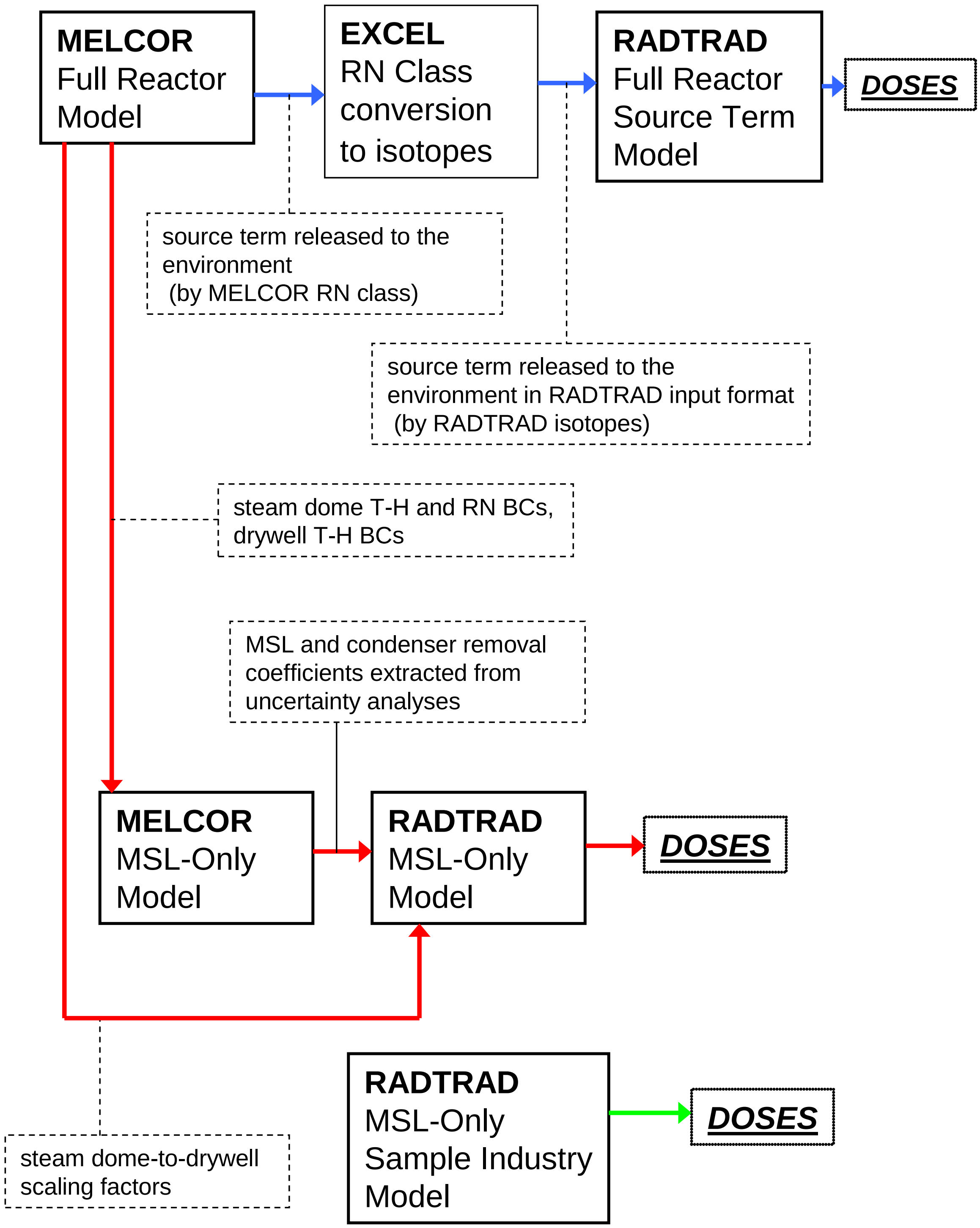


Figure 1-8 Outline of analyses and information flow used in this study.

2 Full Plant MELCOR Analyses

2.1 Description of Full Reactor MELCOR Models

MELCOR calculations to estimate steam dome-to-drywell radionuclide concentration ratios, aerosol removal coefficients in the MSLs, and MSIV leakage source terms were performed using the current state-of-the-art BWR/4 Mk-I and BWR/6 Mk-III MELCOR plant decks developed by Sandia National Laboratories (SNL) for the NRC. MELCOR input data used to develop these models were based on the configuration, geometry and materials of single, representative plants (Peach Bottom and Grand Gulf). Information needed to develop the MELCOR input data were obtained from readily available documents, such as the plant Final Safety Analysis Reports (FSARs) and documentation supporting the Individual Plant Examinations (IPEs). In addition, the models have been configured to conform to current best modeling practices.

The MELCOR nodalization diagrams for the BWR Mk-I and Mk-III models are provided below. Analyses were performed for the following combinations of LOCAs, condenser, and containment sprays.

- Mk-I, main steam line break (MSLB), no sprays,
- Mk-I, MSLB, sprays
- Mk-I, recirculation line break (RLB), no sprays
- Mk-I, RLB, sprays

Mk-III analyses were limited to the RLB and MSLB no containment spray cases.

- Mk-III, MSLB, no spray
- Mk-III, RLB, no spray

The cases were run out to the time of lower head failure. An additional Mk-III RLB case was run in which core sprays were activated 10 minutes prior to the predicted time of lower head failure. This case was run to verify the assumed post-reflood T-H conditions assumed in the MSL-only cases.

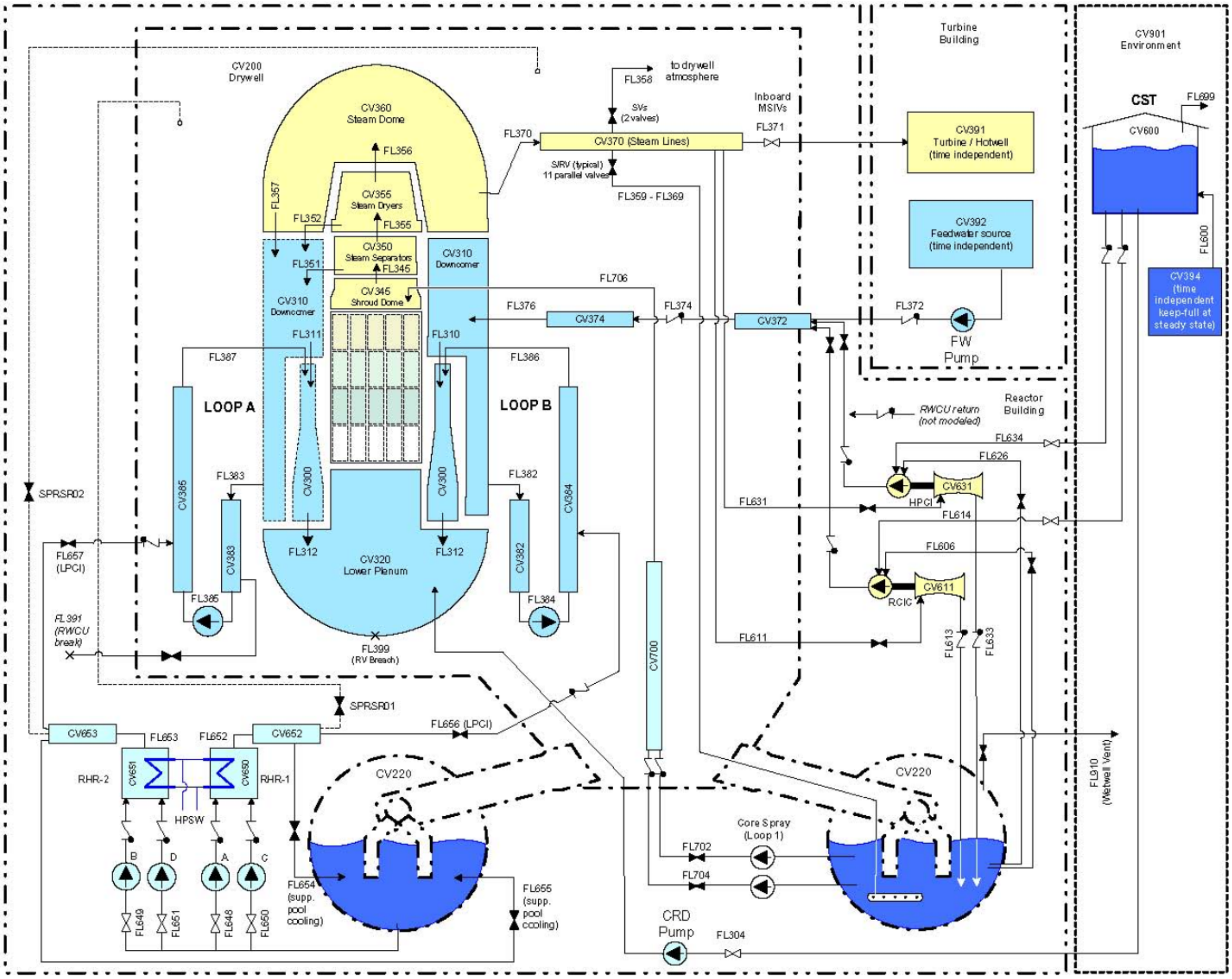


Figure 2-1 BWR Mk-I Reactor Coolant System Nodalization

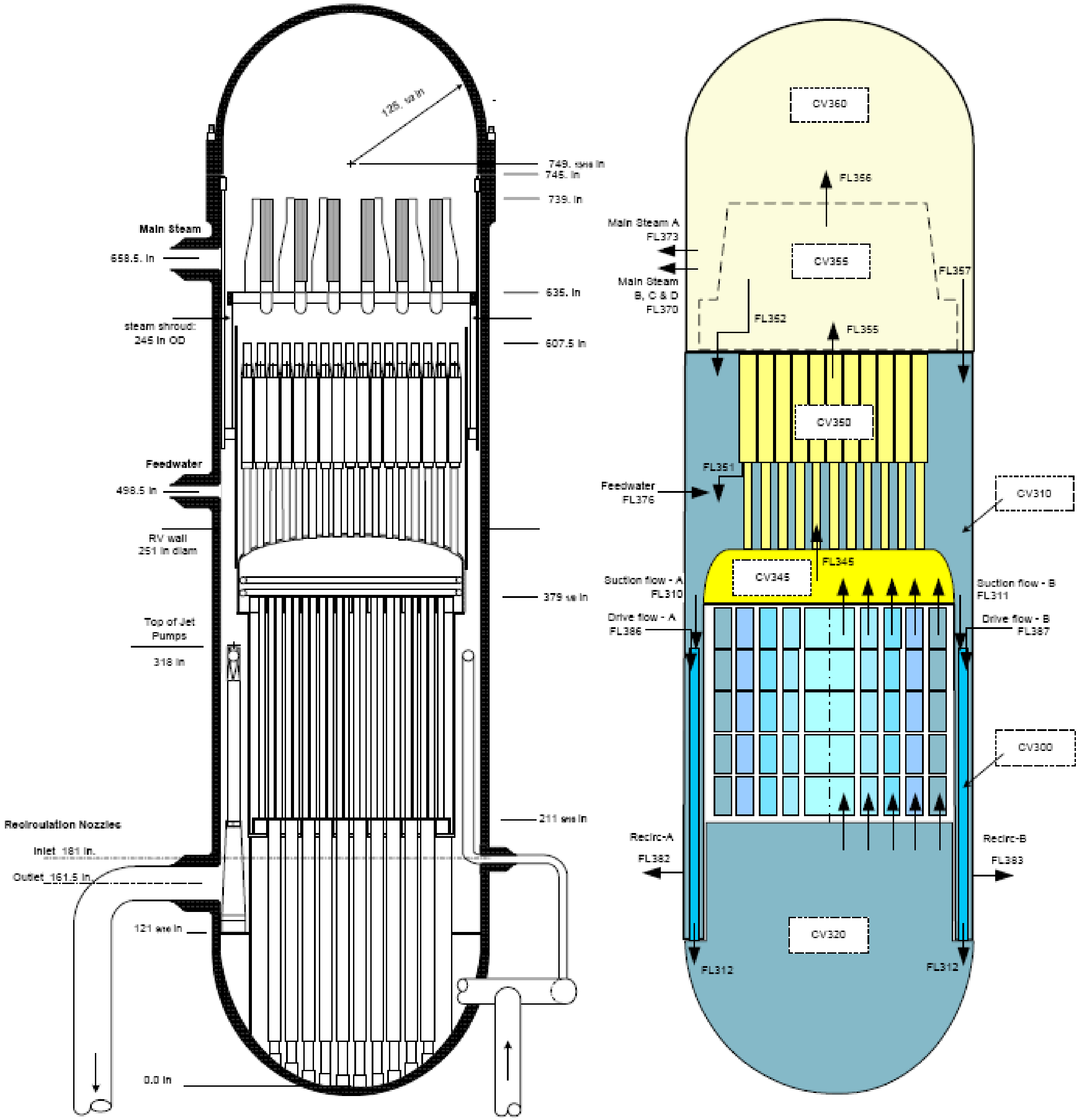


Figure 2-2 BWR Mk-I Reactor Vessel Nodalization Detail

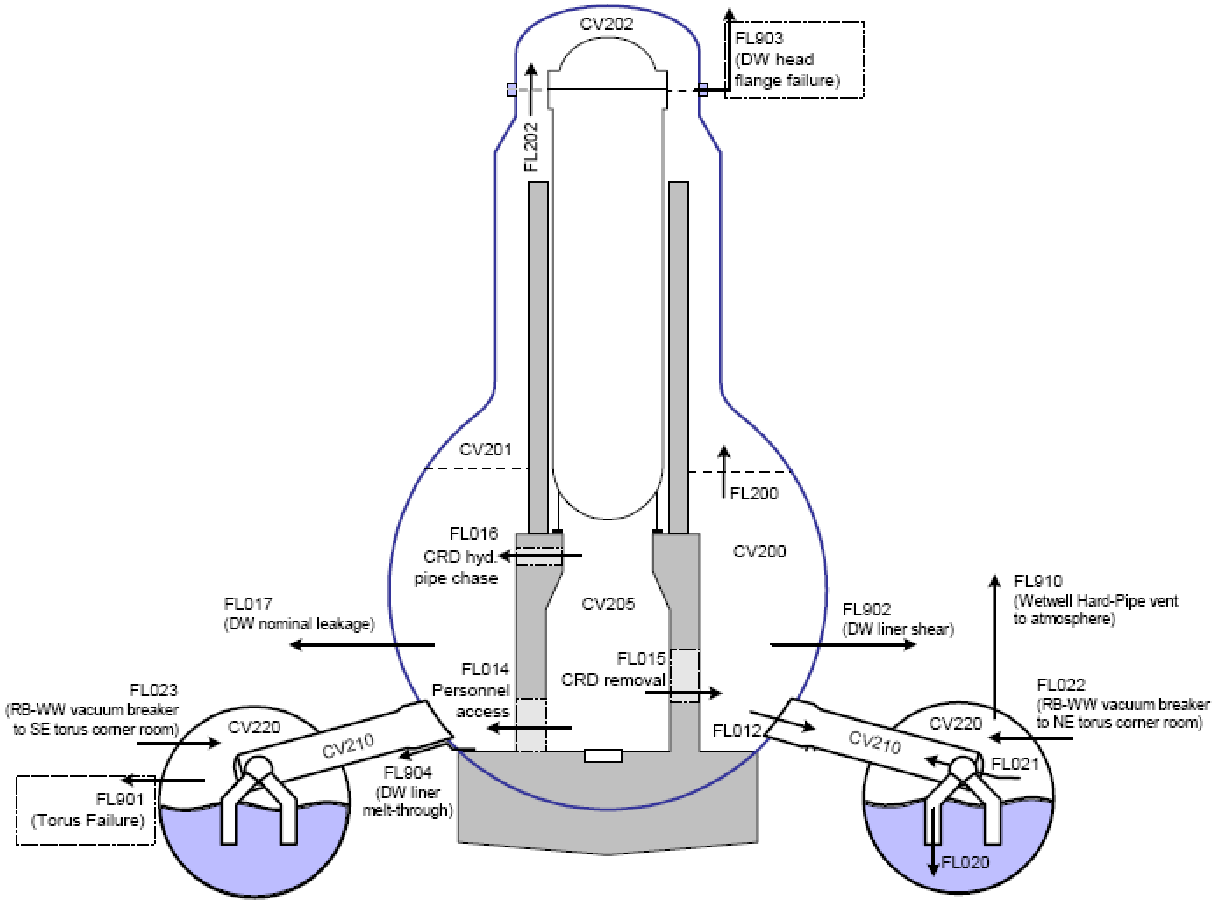


Figure 2-3 MELCOR nodalization used for the Mk-I containment.

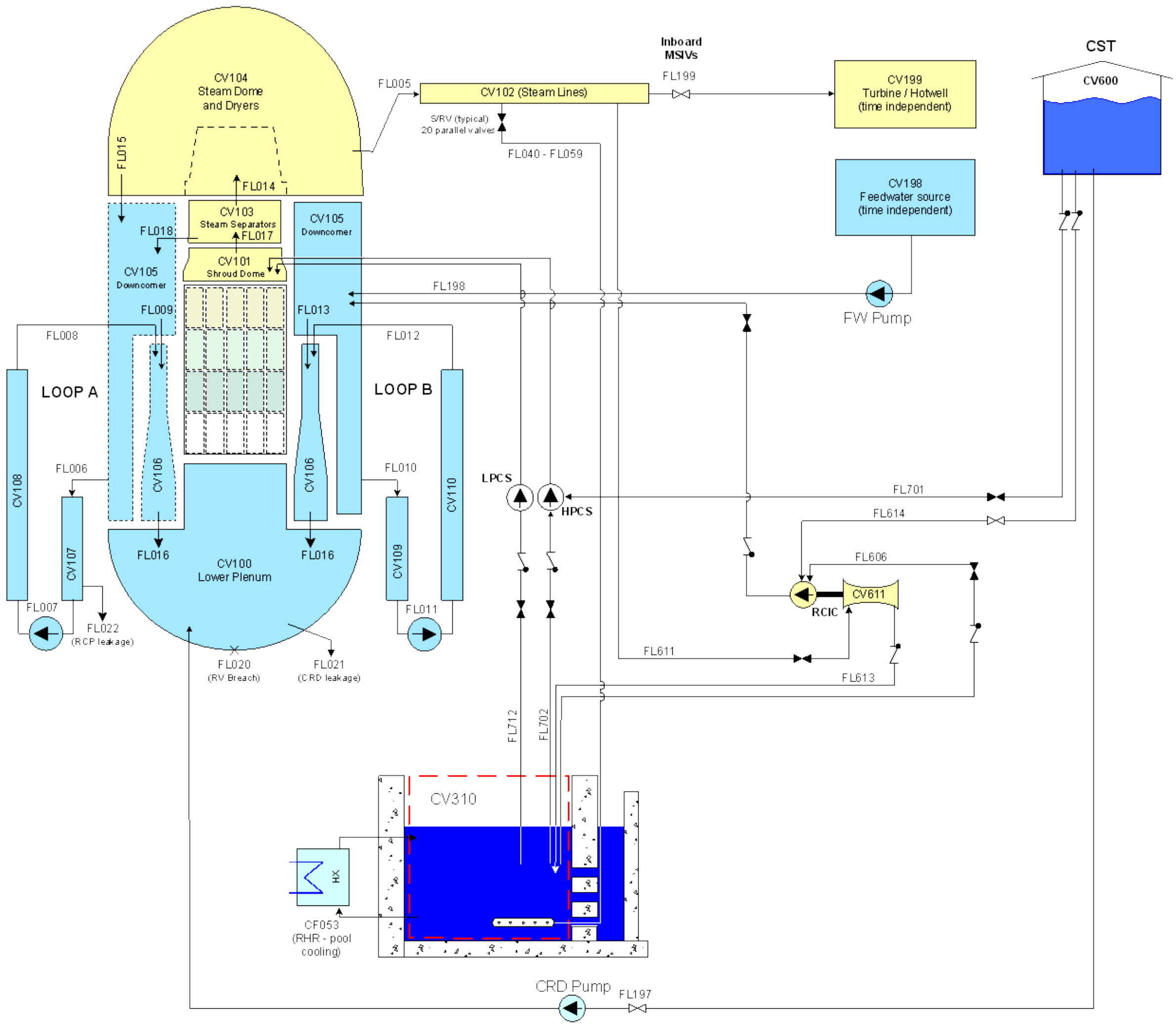


Figure 2-4 BWR Mk-III Reactor Coolant System Nodalization

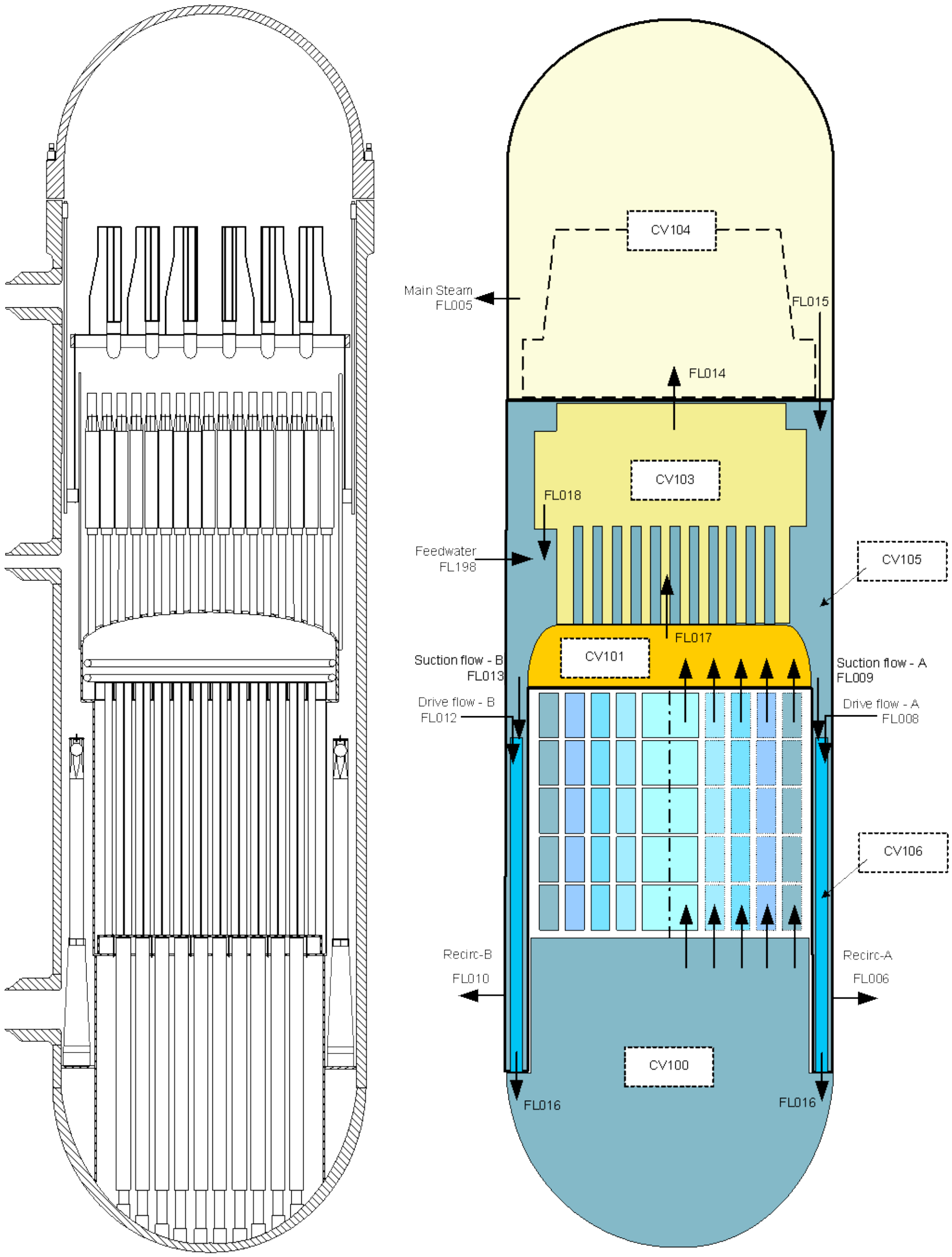


Figure 2-5 BWR Mk-III Reactor Vessel Nodalization Detail

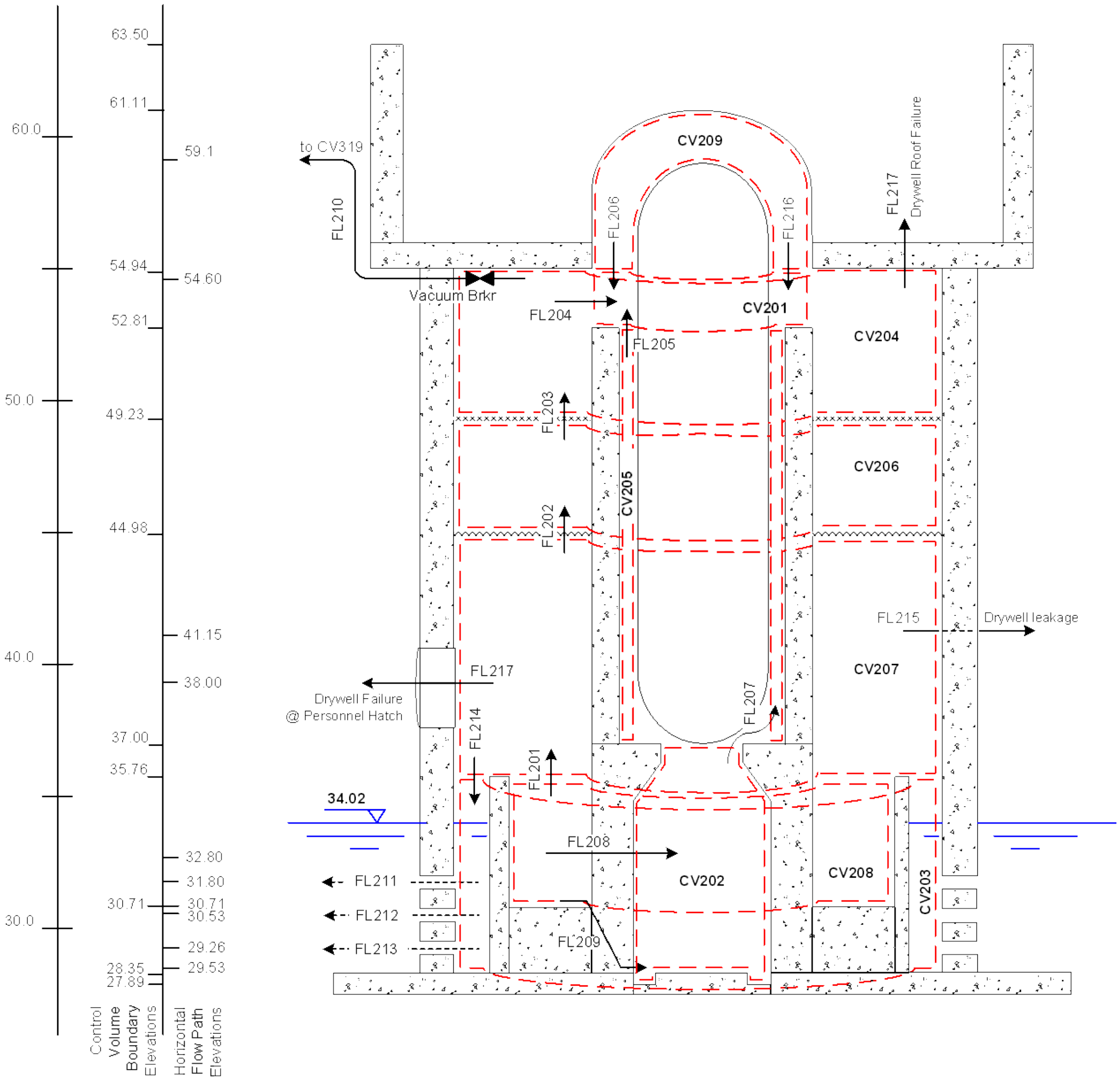


Figure 2-6 Mk-III Drywell Nodalization

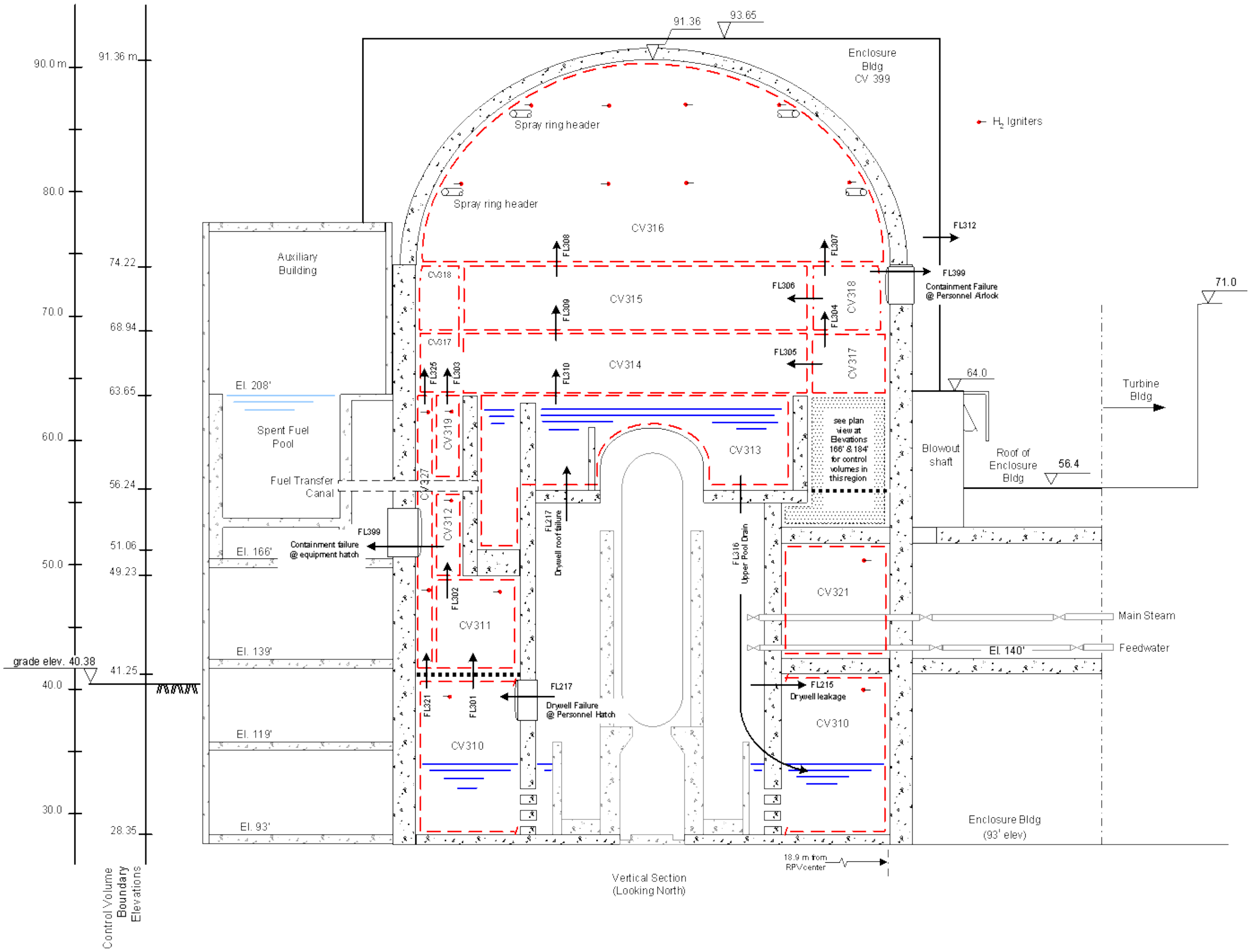


Figure 2-7 BWR Mk-III Containment Nodalization

In this study, each main steam line is modeled separately to allow for different valve leakages on each leg as shown in Figure 2-8 and Figure 2-9. For the cases in which a condenser was included in the model it was added as a control volume connected to the end of the MSLs, modeled as CV860. When used, the condenser was modeled as having 146,996 ft^3 (4162 m^3) at atmospheric pressure and initial temperature of 120F (322K). The deposition horizontal surface area was taken to be the floor area of 252.5 m^2.

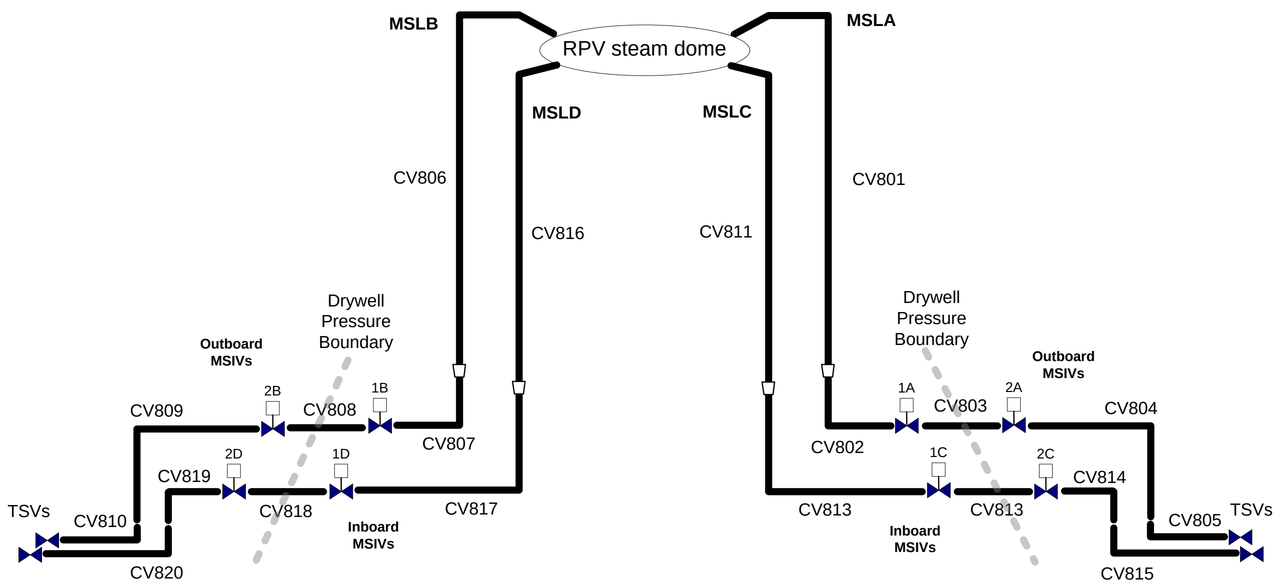


Figure 2-8 BWR Mk-I Main Steam Line Containment Nodalization for MSIV Leakage Calculation

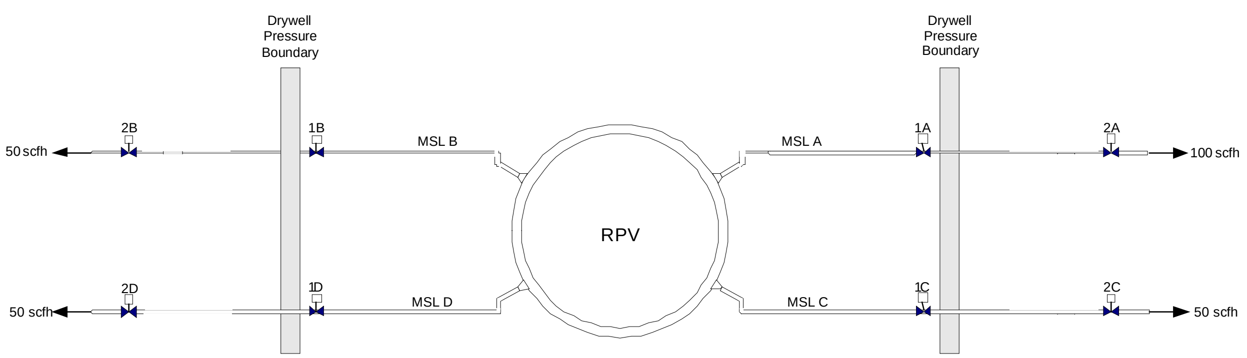


Figure 2-9 BWR Mk-III Main Steam Line Nodalization for MSIV Leakage Calculation

2.1.1 Modifications and Key Assumptions

In the base models the MSIVs close at accident initiation (t = 0 s). This causes the pipe between the MSIVs to be at a higher pressure than the pipe volume between the steam dome and the in-board MSIV. For less than an hour at the beginning of the accident this condition results in back-flow from the in-board MSIV which stops fission products from being passed through the MSLs and on to the environment. The best-estimate base MELCOR model does not account for uncertainties introduced by variations in plant design such as insulation dimensions, MSIV closure times or those from flow recirculation or thermally driven reverse flow, or “chugging [3]”. Because of these uncertainties and to promote MSIV leakage, two modifications were made to the base model: the initiation of closure of the out-board MSIV was delayed by 3 seconds and the heat transfer coefficients of the heat structures of the MSL piping between the steam dome and in-board MSIV were reduced to a low value (0.01 $W/(m^2K)$) for the first hour after accident initiation in order to thermally decouple the intermediate region gas from the hot steam line walls, and thereby suppress the intermediate region pressurization and reverse flow through the inboard MSIV’s. Additionally, anecdotal evidence[†] has indicated that the inboard MSIV tends to unseat when the down stream pressure exceeds the upstream pressure by about 25 psi, which would tend to prevent high pressure trapped in the intermediate steam line from producing reverse flow through the inboard valve. These adjustments to the MELCOR model produce the anticipated valve outflow conditions, and facilitate comparisons with the RADTRAD approach that specifies outflow conditions. For these calculations, the inboard MSIVs start to close at 0 s and completed closing at 3 s. The outboard valves start to close at 3 s and finish at 6 s. For both valves, the closure rate was constant.

The flow path from the condenser to the environment was modeled as having an area equivalent to that of an MSL. An area of this size provides negligible flow resistance, and is hence conservative with respect to fission product release to the environment.

Leakage past the MSIVs in the Mk-I and Mk-III MELCOR models was defined considering a base technical specification of 11.5 scfh ($9.05x10^{-5}$ m^3/sec) maximum when tested at 25 psig, i.e., with upstream pressure at 39.7 psia (0.172 MPa) and downstream pressure at 14.7 psia (0.101 MPa). Associated upstream temperature was taken to be 59 °F (287.8K).

A standalone MELCOR model was constructed to impose the above conditions on a constricted flow path configured to represent MSIV leak geometry as an orifice, i.e., configured to offer negligible frictional losses. Air was specified as the fluid. The model identified, through trial and error execution, that a flow path with an area of 0.2828 m^2 (the full open area of a main steam line in the Mk-I model) and an open fraction of $5.9926x10^{-7}$ would flow $1.1x10^{-4}$ kg/s of air (11.5 scfh taking density at standard temperature and pressure to be 1.2232 kg/m^3).

The base leakage specifications for the MSIVs in the Mk-I models designated for the subject work were:

- 205 scfh (at 25 psig) past the valves in MSL A
- 155 scfh past the valves in MSL B
- Zero leakage past the valves in MSL C
- Zero leakage past the valves in MSL D

The base leakage specifications for the MSIVs in the Mk-III models designated for the subject work were:

[†] Verbal communication with John Ridgely, USNRC.

- 100 scfh (at 25 psig) past the valves in MSL A
- 100 scfh past the valves in MSL B
- 50 leakage past the valves in MSL C
- Zero leakage past the valves in MSL D

Note that these flow areas and open fractions were defined at both the inboard and outboard MSIV locations. As in the standalone MELCOR model, the leakage flow paths in the Mk-I and Mk-III models were configured to offer negligible frictional losses.

The following section presents basic results of the MELCOR full plant analyses, including key accident progression signatures, and airborne concentrations of key radionuclides in the steam dome, drywell and wetwell regions. Ratios of these airborne concentrations are used to provide the scaling parameters to be used in subsequent RADTRAD analyses, as described earlier in Section 1.1.

It should be mentioned that MELCOR does not presently treat wall and surface deposition of elemental iodine from processes other than condensation, this being an area of ongoing research; however, neither does MELCOR introduce any elemental iodine as a result of release from fuel. Rather, it is assumed in MELCOR that all iodine is initially released as CsI. MELCOR can ultimately produce some source of elemental iodine as a result of revaporization of previously deposited CsI onto surfaces where the Cs has been presumed to “chemisorbed,” or chemically adhered, to the surface. In this case, iodine alone is evolved in the form of elemental iodine

The nature of leakage pathways and whether they are best treated as orifices or capillary pathways is often discussed in regulatory applications, especially for purposes of calculating releases from containment leakage. It has been common practice by both the NRC and industry to assume orifice geometry. Standard orifice geometry was assumed for the determination of leakage in this report. Because work on aerosol capture in pathways of various geometries suggest that impaction is not a promising mechanism for crediting additional attenuation, impaction was not considered as a removal mechanism for this study.

2.1.2 Calculation of Removal Coefficients

A key result from the MELCOR analysis of full plant behavior is a characterization of aerosol depletion and deposition behavior that is needed by the RADTRAD code to mimic the depletion behavior predicted by MELCOR. While the mechanics of fission product deposition are complex, its net behavior in the containment and in the steam lines is often characterized by the simplified assumption that the overall deposition rate is proportional to the airborne fission product mass in the containment or MSL vapor space.

$$\dot{m}_{dep} = \lambda m \tag{2.1}$$

where

$\dot{m}_{dep}$ - aerosol mass deposition rate

λ - time-dependent aerosol mass removal coefficient

m - aerosol airborne mass in the volume in question

This term is part of the overall generalized aerosol mass balance

$$\frac{dm}{dt} = \dot{m}_{in} - \dot{m}_{out} - \dot{m}_{dep} + \dot{S} \tag{2.2}$$

where

$\frac{dm}{dt}$ - rate of change of airborne aerosol mass in the volume in question

$\dot{m}_{in}$ - rate of airborne aerosol mass entering the volume from other volumes

$\dot{m}_{in}$ - rate of airborne aerosol mass exiting the volume to other volumes

$\dot{S}$ - rate of aerosol source term injection into the volume

For the purpose of calculating MSL removal coefficients, the MSLs have been conceptually divided into three segments (1) between the steam dome and the in-board MSIV, (2) between the in-board and out-board MSIVs, and (3) between the out-board MSIV and the turbine building (or condenser, if modeled). This division is consistent with the typical MSL nodalization used in RADTRAD MSL models. The removal coefficient for each MSL segment is calculated in the MELCOR analyses at each time-step by dividing the aerosol mass deposition rate by the aerosol mass in the vapor space of that MSL segment ($\lambda \approx \dot{m}_{dep}/m$). Likewise, for cases with a condenser, the removal coefficient in the condenser is calculated in the MELCOR analyses by, at each time-step, dividing the aerosol mass deposition rate in the condenser by the aerosol mass in the vapor space in the condenser. From this, an instantaneous removal rate coefficient can be calculated as part of the MELCOR generated results. This instantaneous removal calculated at each time step exhibits transient fluctuations caused by the variations in the parameters that affect it. Among these are fluctuations associated with volume in-flows and out-flows, aerosol source fluctuations, sudden increases in steam generation, and the associated changes in deposition rates from processes such as diffusiophoresis and revaporization of previously deposited fission products. In order to produce coefficients of the form that are consistent with the idealized approach of exponential decay, the instantaneous removal coefficients are averaged in time providing the effective “lambdas”.

2.2 Results from Full Reactor MELCOR Models

Selected results of the MELCOR full reactor model calculations are presented in the following sections.

2.2.1 Mk-I MSLB, No Sprays

The main steam line break full plant analysis for the Mk-I design with no containment sprays operational is described in this section. Figure 2-10 through Figure 2-12 show the pressure response of the vessel and drywell, temperatures in the steam dome and drywell and the core water level. Note that, while the core water inventory is depleted after 2 hours, lower head failure is not predicted until almost 8 hours. This modern view of the accident progression is in significant variance with the view put forth in NUREG-1465 where head failure was assumed to occur at about 2 hours into such an accident. This observation is evidence that the present NUREG-1465 assumptions about the early in-vessel period in terms of release rate and duration are no longer consistent with current best estimate modeling results.

A significant finding in this work is illustrated by comparing the information shown in Figure 2-13 and Figure 2-14. From these figures it can be seen that while there is significantly more airborne *mass* of CsI in the *drywell* during the first 2 hours of the accident, in contrast, the *concentration* of airborne CsI in the *steam dome* is significantly greater than that of the drywell. Since the steamlines are connected to the steam dome and not the drywell, it is the airborne concentration in the steam dome, not the airborne mass in the drywell, that determines the MSIV leakage behavior. Figure 2-15 provides the ratio of airborne concentrations of select radionuclides in the steam dome relative to the drywell in the MELCOR full plant analyses.

Finally, best estimate removal coefficients (lambda values), characterizing the observed deposition of aerosols by gravity settling, thermophoresis and diffusiophoresis for the main steam lines A and B are shown in Figure 2-16 and Figure 2-17. Here one can see that in the first 2 hours, removal coefficients for the inboard and intermediate lines are about 3 hr^{-1}, decreasing to values between 1 and 2 hr^{-1} after 2 hours. The removal coefficients for the outboard lines are always on the order of 1 hr^{-1} or less, due to the very small aerosol particle sizes reaching these regions. Note that the instantaneous removal coefficients at times can show very transient behavior not consistent with the idealized exponential depletion model, as seen for example in Figure 2-16 where at about 2 hours, the removal coefficient suddenly increases from about 1.8 up to a value of 16 to 18, persisting for about a half hour. This temporary increase in apparent removal rate coefficient is driven by events in the core melt progression where core slumping at about two hours drives vigorous steam generation as evidenced by the rapidly dropping core water level at 2 hrs as seen in Figure 2-12. This produces both significantly increased aerosol concentrations in the steam dome region (Figure 2-14), as well as increased diffusiophoretic deposition rates. Other fine structure in the MELCOR-predicted instantaneous removal coefficients can be observed, again owing to the dynamic behavior of core damage progression and hydrodynamic effects.

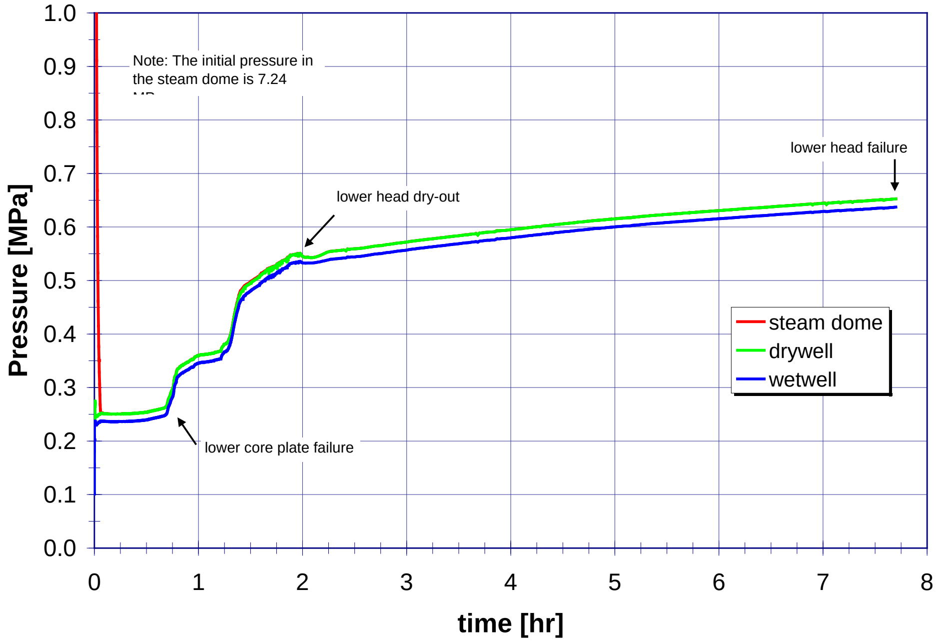


Figure 2-10 BWR Mk-I, MSLB, No Sprays: Steam Dome, Drywell, and Wetwell Pressure

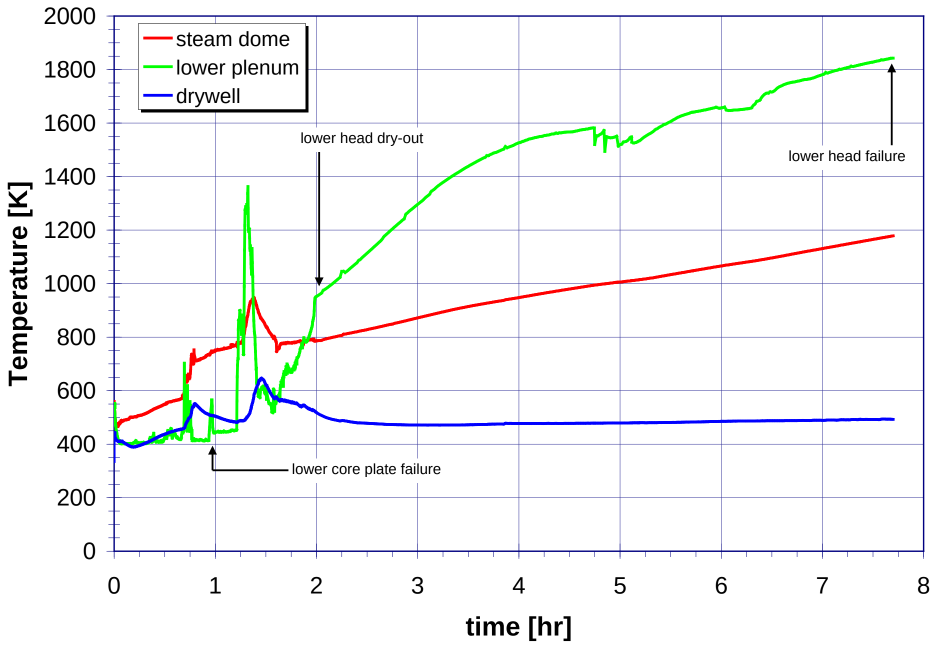


Figure 2-11 BWR Mk-I, MSLB, No Sprays: Steam Dome, Drywell, and Lower Plenum Vapor Temperature

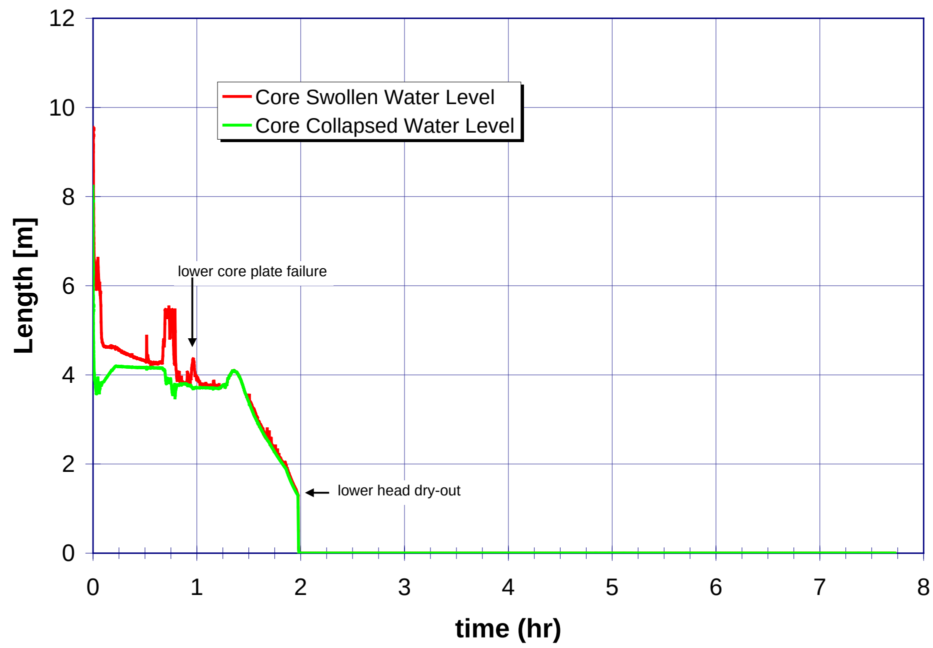


Figure 2-12 BWR Mk-I, MSLB, No Sprays: Core Water Levels

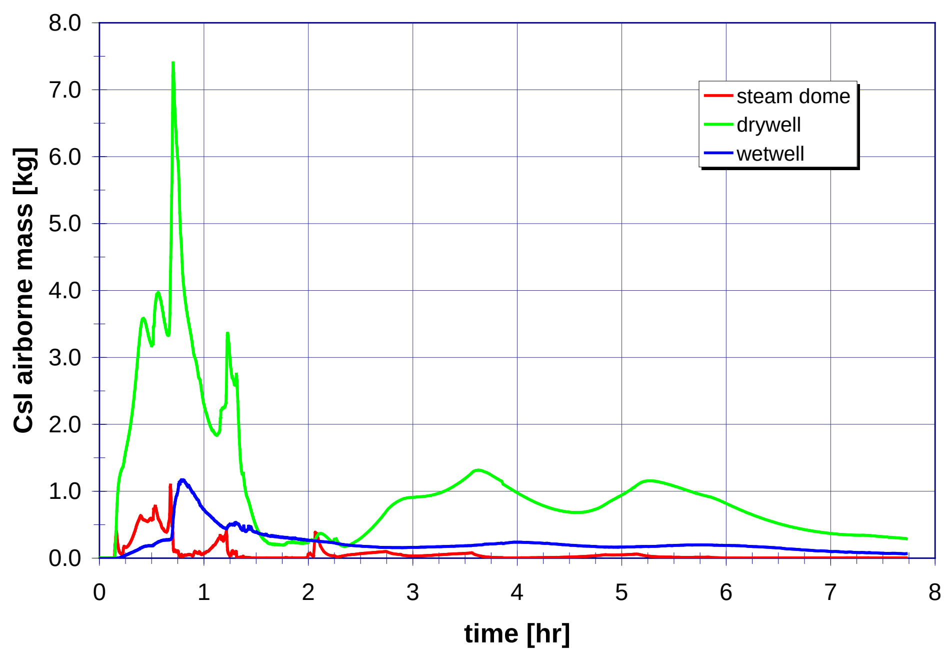


Figure 2-13 BWR Mk-I, MSLB, No Sprays: CsI Mass in the Steam Dome, Drywell, and Wetwell

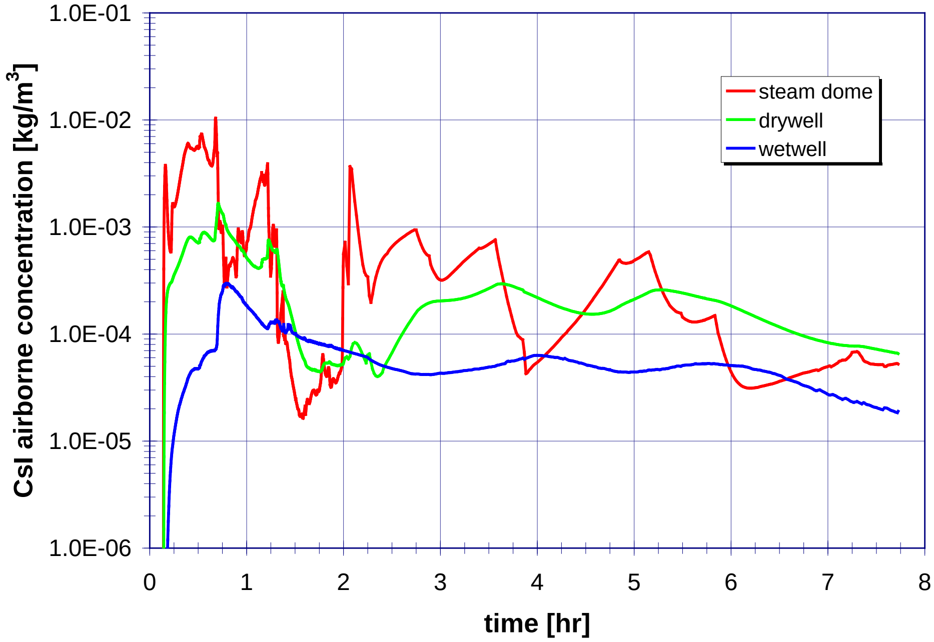


Figure 2-14 BWR Mk-I, MSLB, No Sprays: CsI Concentration in the Steam Dome, Drywell, and Wetwell

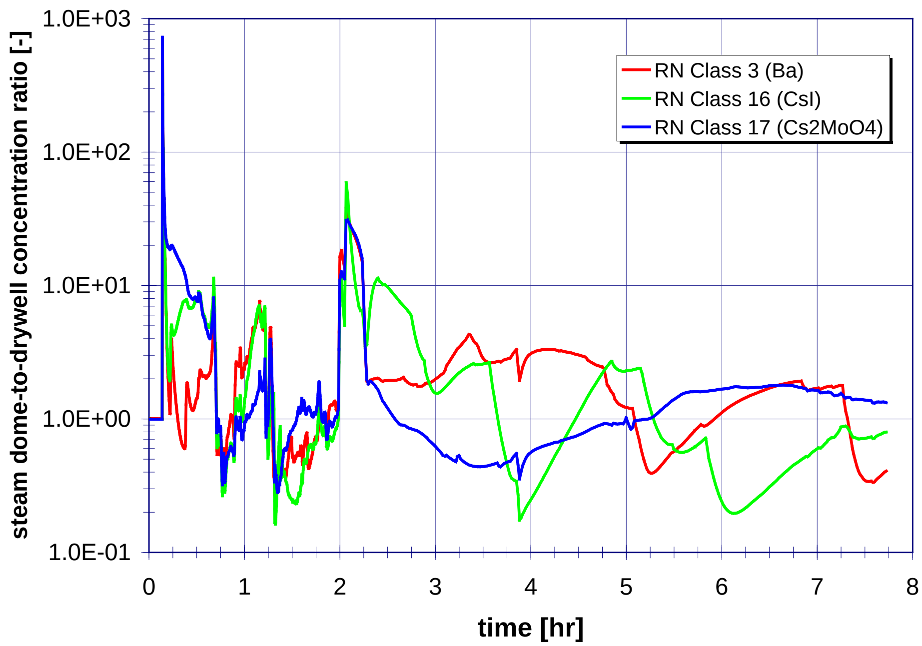


Figure 2-15 BWR Mk-I, MSLB, No Condenser, No Sprays: Steam Dome-to-Drywell Concentration Ratios

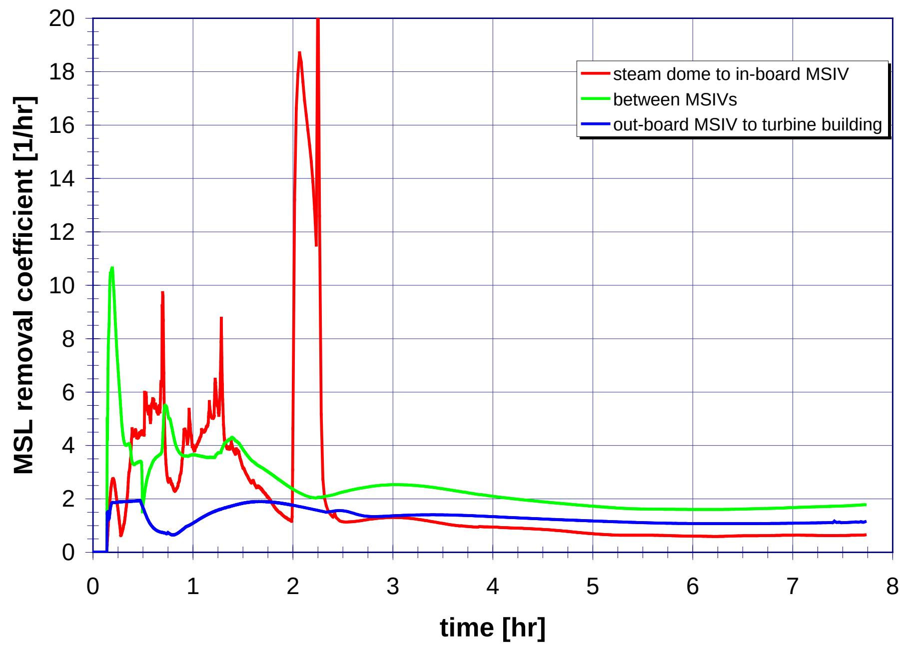


Figure 2-16 BWR Mk-I, MSLB,, No Sprays: MSL-A Removal Coefficients

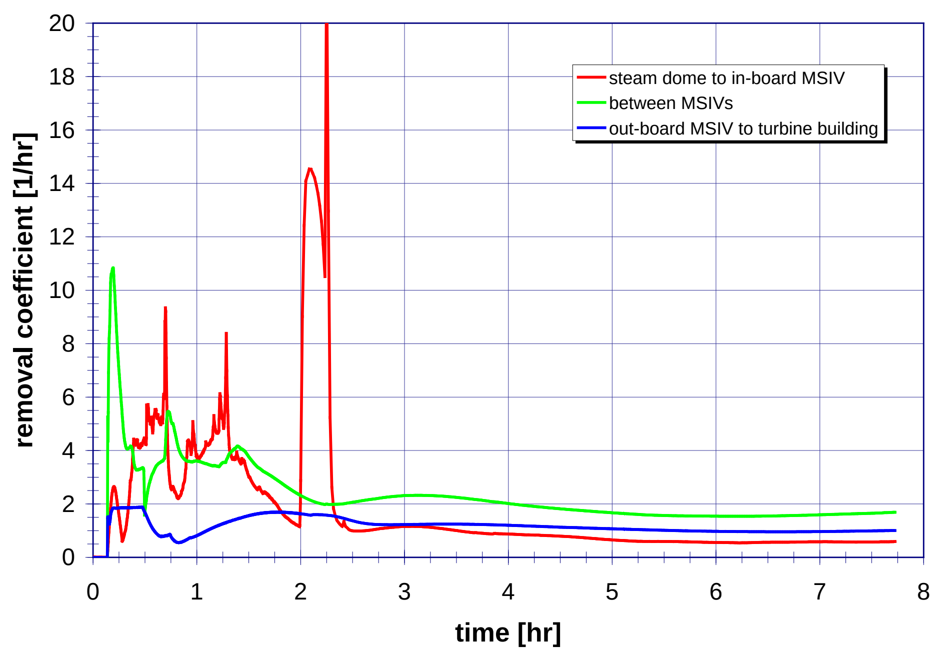


Figure 2-17 BWR Mk-I, MSLB, No Sprays: MSL-B Removal Coefficients

2.2.2 Mk-I MSLB, Sprays

In this section, the automatic actuation of containment sprays at the onset of the accident are examined for the Main Steam Line Break sequence in the Mk-I containment. Results analogous to the previous section are presented here. The purpose of this section is to examine the effect of containment spray actuation on the airborne aerosols in the drywell and in the reactor steam dome, since some license amendment requests have requested credit for aerosol attenuation by sprays in order to reduce the radioactivity escaping through the MSIV's. The basic accident thermal-hydraulic signatures are similar to the cases without containment sprays. These are summarized in Figure 2-18 through Figure 2-20. Again, the drywell airborne total mass exceeds the steam dome mass by a factor of 4 to 5 or more (Figure 2-21), but the steam dome concentration generally exceeds that of the drywell volume by an order of magnitude in the first hour, and a lower factor after 2 hours (Figure 2-22 and Figure 2-23). At times however, steam generation in the vessel can temporarily clear out the steam dome, as occurs between 1.5 and 2 hours. Removal coefficients are summarized for the steam lines in Figure 2-24 and Figure 2-25, again transient behaviors associated with core dynamics and fluctuations in in-vessel steam generation can be observed.

Finally, Figure 2-26 summarizes the effect of containment sprays on the drywell airborne concentrations and on the steam dome aerosol concentrations. From this figure it can be seen that for both the case with sprays and the case without sprays, the steam dome concentrations are essentially the same for the first hour. During this time also there is only a little decrease in the drywell concentration as the aerosol source to the containment from the vessel replenishes the concentrations even as the sprays remove airborne material. After two hours, the drywell sprays significantly reduce the drywell concentrations and would seem also to reduce the steam dome concentrations during this time although the steam dome concentrations still exceed the drywell concentrations. Also after the 2-hour period, the steam dome concentration for the case with no sprays is about an order of magnitude higher than the case with sprays, indicating that after the termination of the early in-vessel release, the containment sprays may encourage outflow from the vessel thereby reducing airborne concentrations in the steam dome.

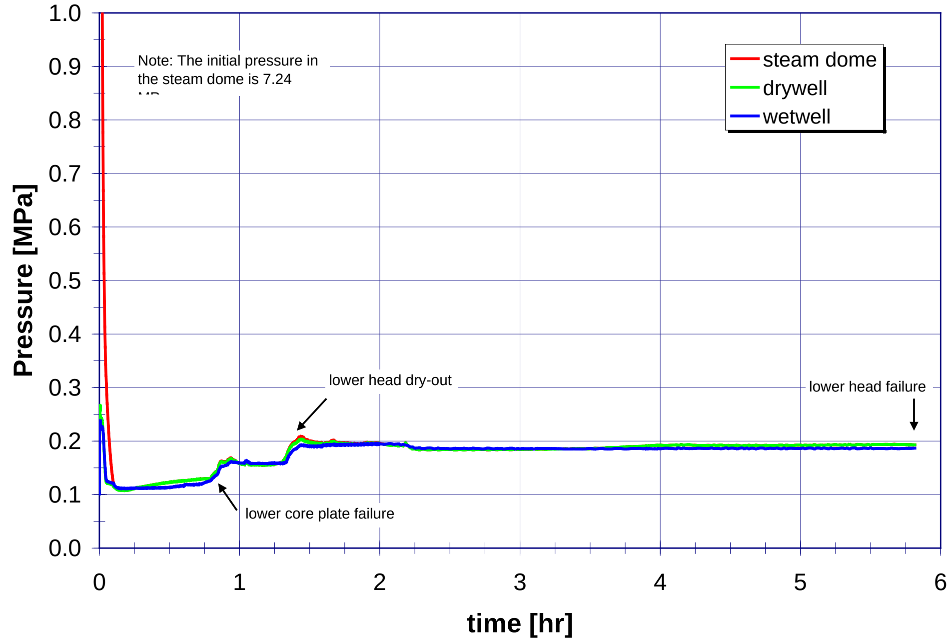


Figure 2-18 BWR Mk-I, MSLB, Sprays: Steam Dome, Drywell, and Wetwell Pressure

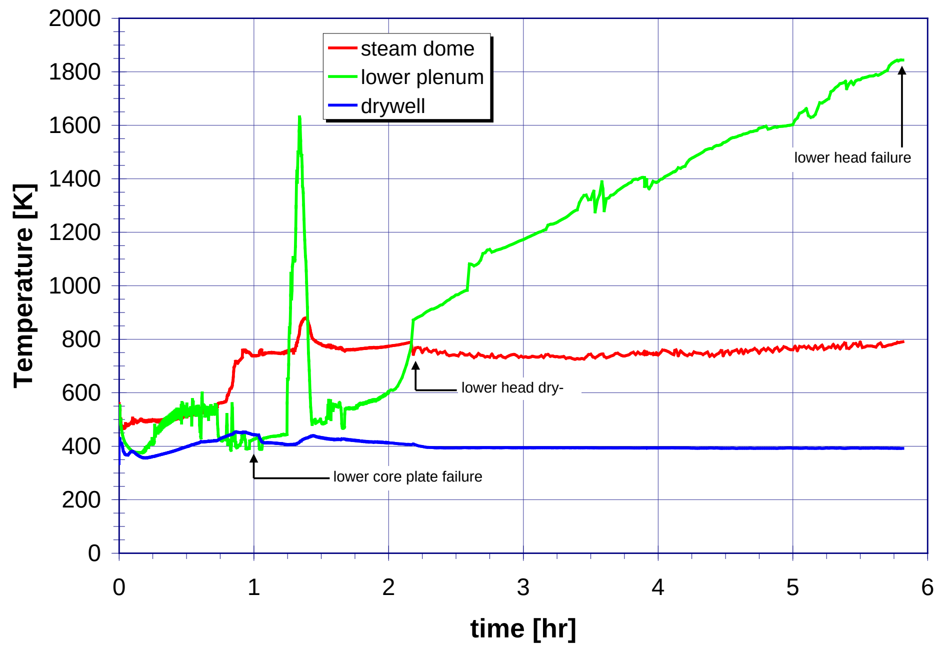


Figure 2-19 BWR Mk-I, MSLB, Sprays: Steam Dome, Drywell, and Lower Plenum Vapor Temperature

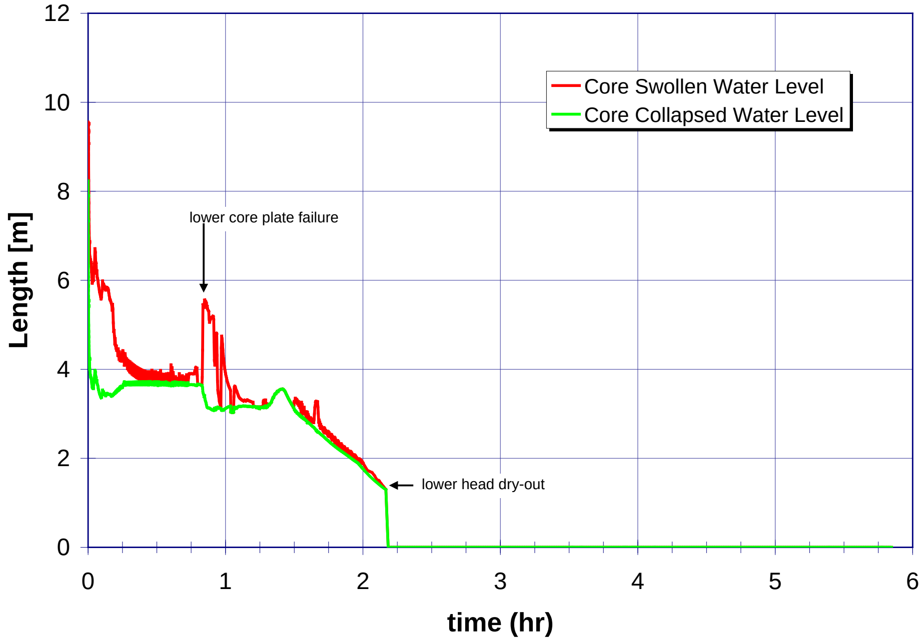


Figure 2-20 BWR Mk-I, MSLB, Sprays: Core Water Levels

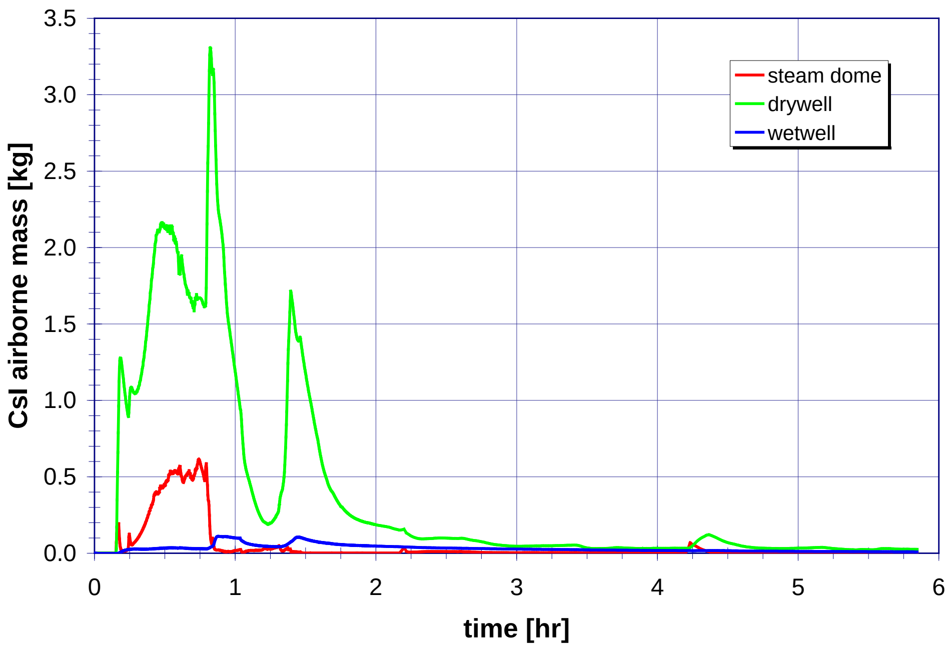


Figure 2-21 BWR Mk-I, MSLB, Sprays: CsI Mass in the Steam Dome, Drywell, and Wetwell

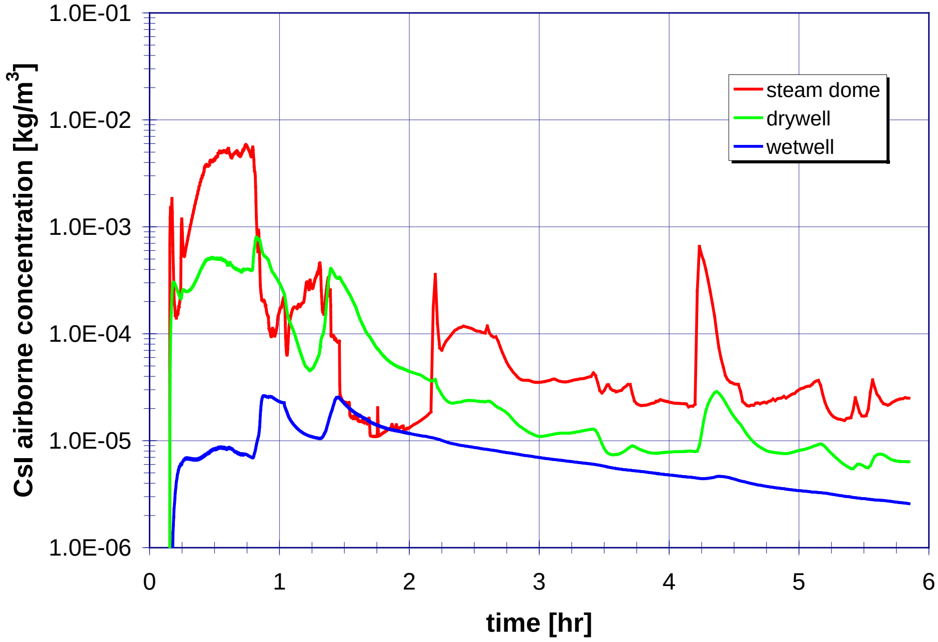


Figure 2-22 BWR Mk-I, MSLB, Sprays: CsI Concentration in the Steam Dome, Drywell, and Wetwell

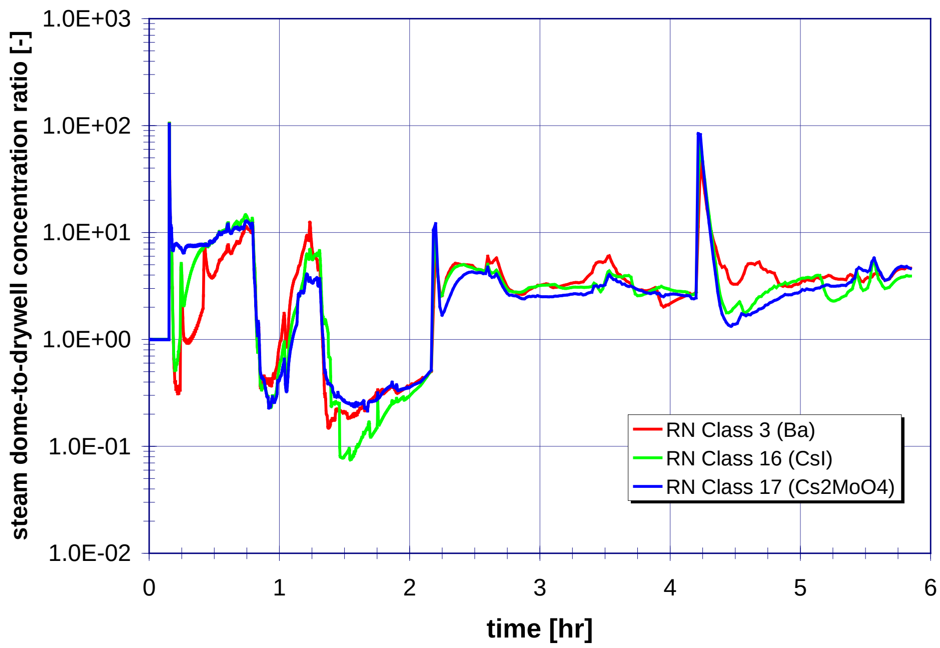


Figure 2-23 BWR Mk-I, MSLB, Sprays: Steam Dome-to-Drywell Concentration Ratios

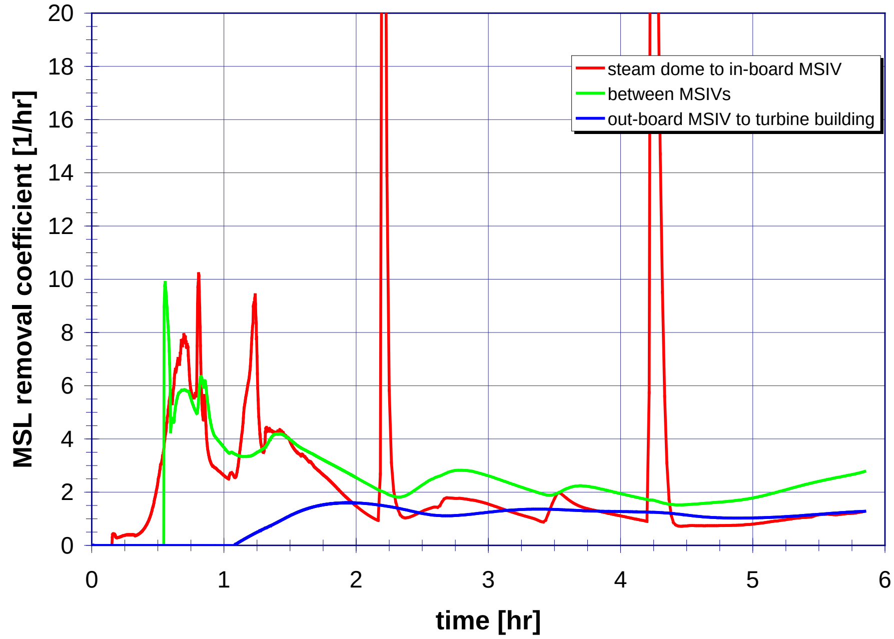


Figure 2-24 BWR Mk-I, MSLB, Sprays: MSL-A Removal Coefficients

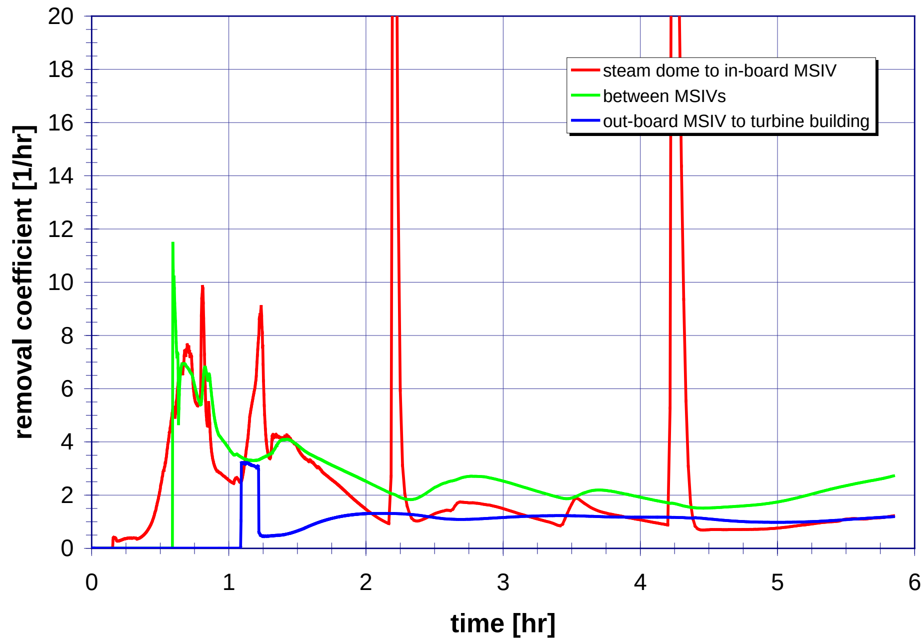


Figure 2-25 BWR Mk-I, MSLB, Sprays: MSL-B Removal Coefficients

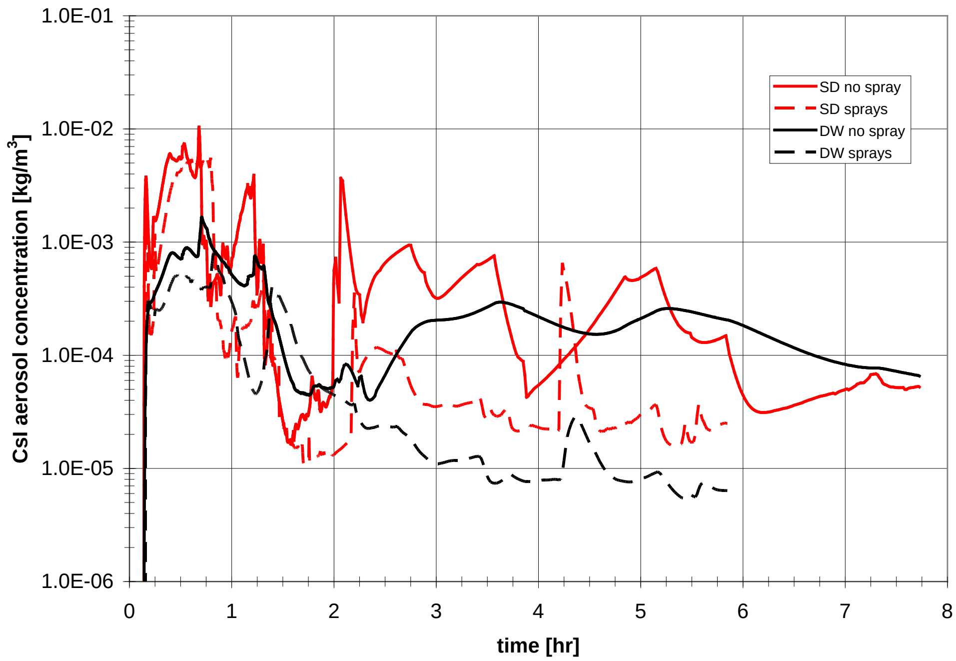


Figure 2-26 BWR Mk-I, MSLB, Comparison of CsI Aerosol Steam Dome and Drywell Concentrations with and without Sprays

2.2.3 Mk-I RLB, Both No Sprays and With Sprays

The following two sections relate the findings for analysis of a recirculation line break in a Mk-I containment, both with and without containment sprays. In these analyses, all steam lines are intact and the breach to the drywell is via the broken recirculation line. The analyses without drywell sprays are presented in Figure 2-27 through Figure 2-34, and the case with drywell sprays are summarized in Figure 2-35 through Figure 2-42. The general characteristics of RLB analyses are similar to the MSLB analyses except for some timing differences in core degradation and vessel failure. A comparison of Figure 2-31 with Figure 2-39 showing CsI concentrations in the steam dome, drywell and wetwell spaces show that steam dome concentrations are again higher than drywell concentrations for both cases with and without drywell sprays, except for brief periods where intense vessel steaming temporarily clears the vessel and steam dome volumes – at these times steam dome and drywell concentrations are similar. Figure 2-43 again summarizes the effect of drywell sprays on the steam dome and drywell aerosol concentrations. Sprays have no appreciable effect on the steam dome concentrations during the first 2 hours of the RLB accident, but after 2 hours, the steam dome concentrations are reduced by about an order of magnitude for the case with sprays, although still an order of magnitude higher than the drywell concentration.

In conclusion, after 2 hours drywell sprays do seem to reduce the steam dome concentrations relative to the no-spray cases; however the steam dome concentrations remain above the drywell concentration in either case.

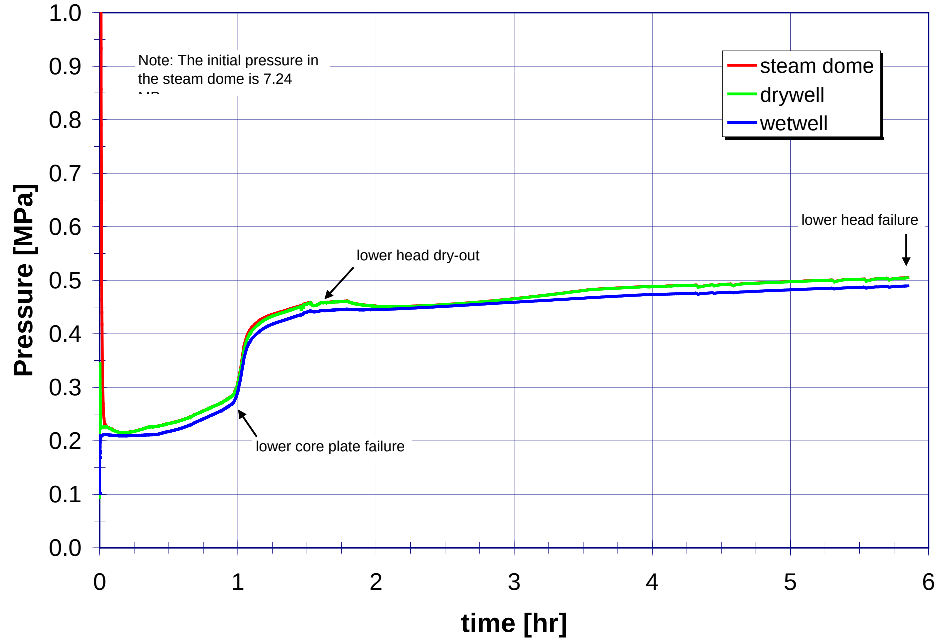


Figure 2-27 BWR Mk-I, RLB, No Sprays: Steam Dome, Drywell, and Wetwell Pressure

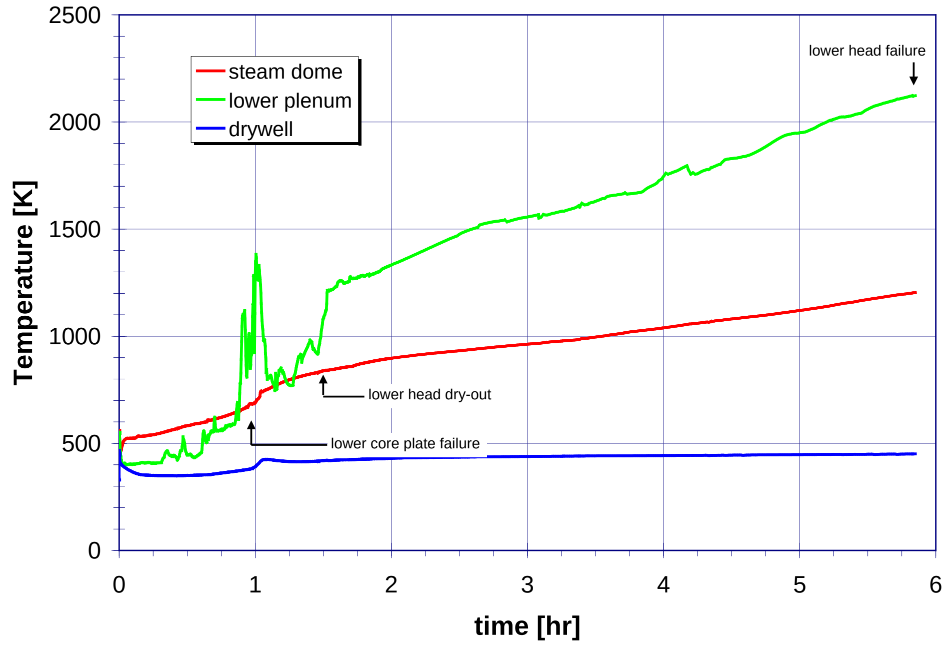


Figure 2-28 BWR Mk-I, RLB, No Sprays: Steam Dome, Drywell, and Lower Plenum Vapor Temperature

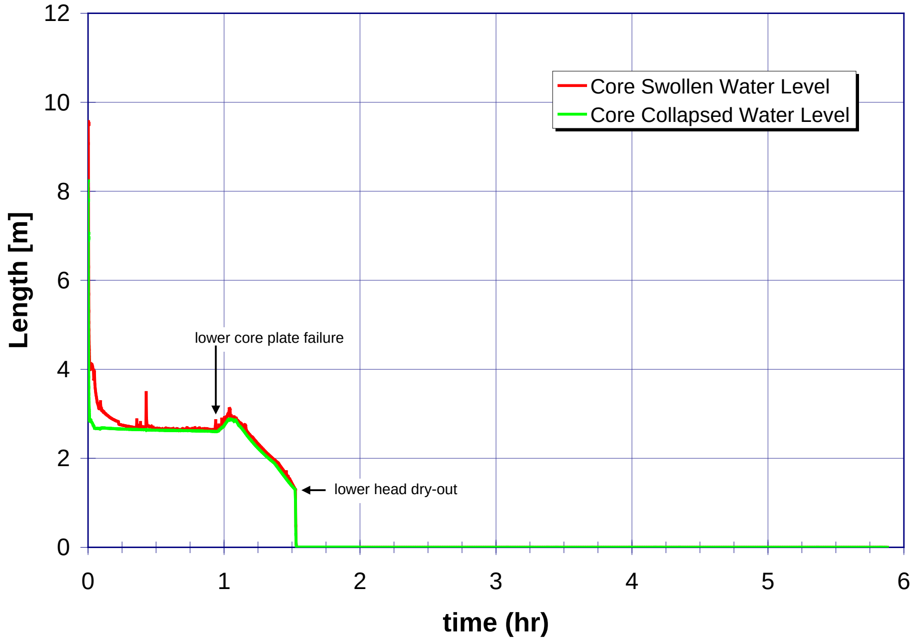


Figure 2-29 BWR Mk-I, RLB, No Sprays: Core Water Levels

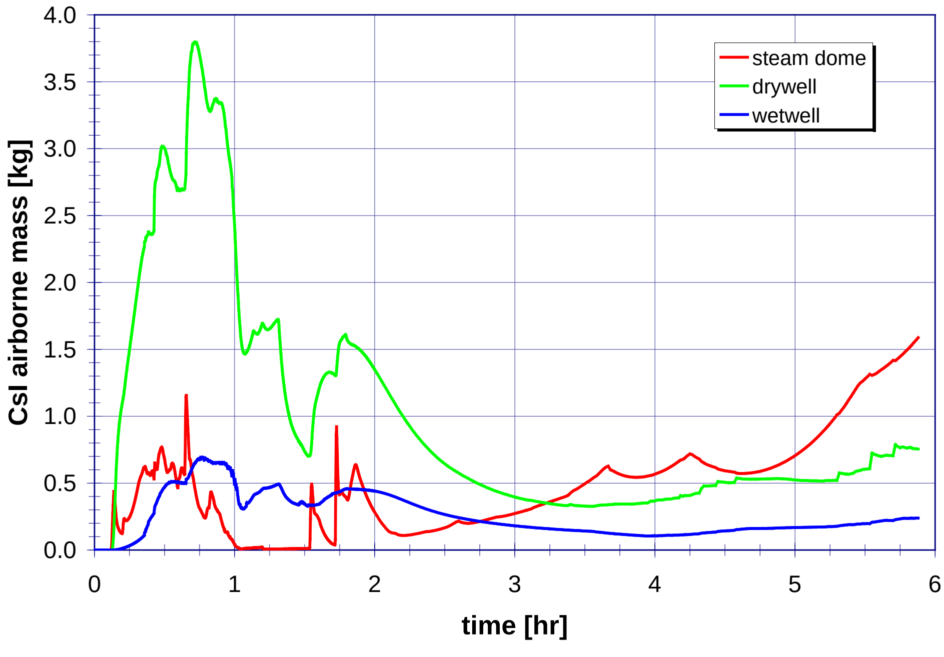


Figure 2-30 BWR Mk-I, RLB, No Sprays: CsI Mass in the Steam Dome, Drywell, and Wetwell

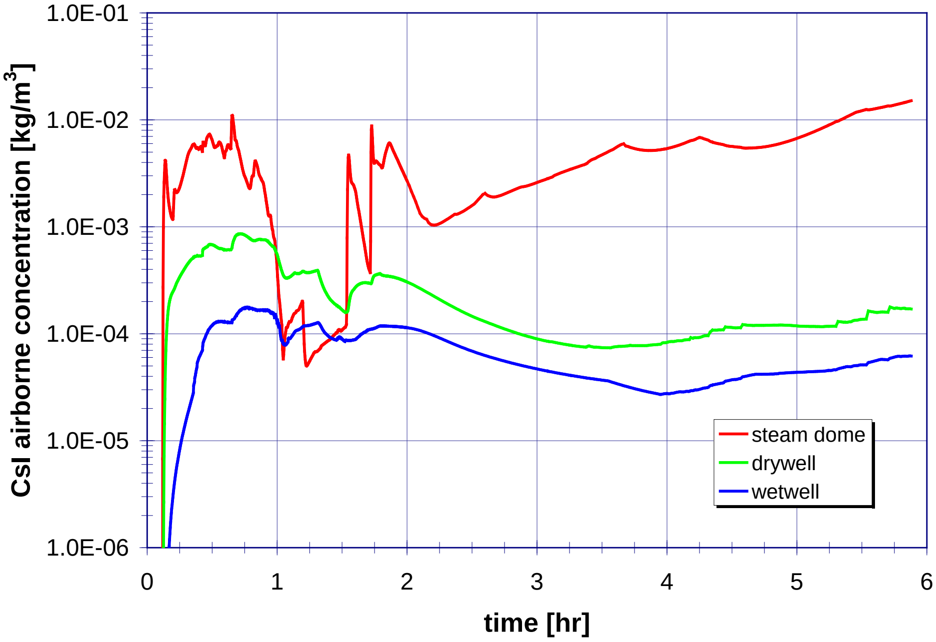


Figure 2-31 BWR Mk-I, RLB, No Sprays: CsI Concentration in the Steam Dome, Drywell, and Wetwell

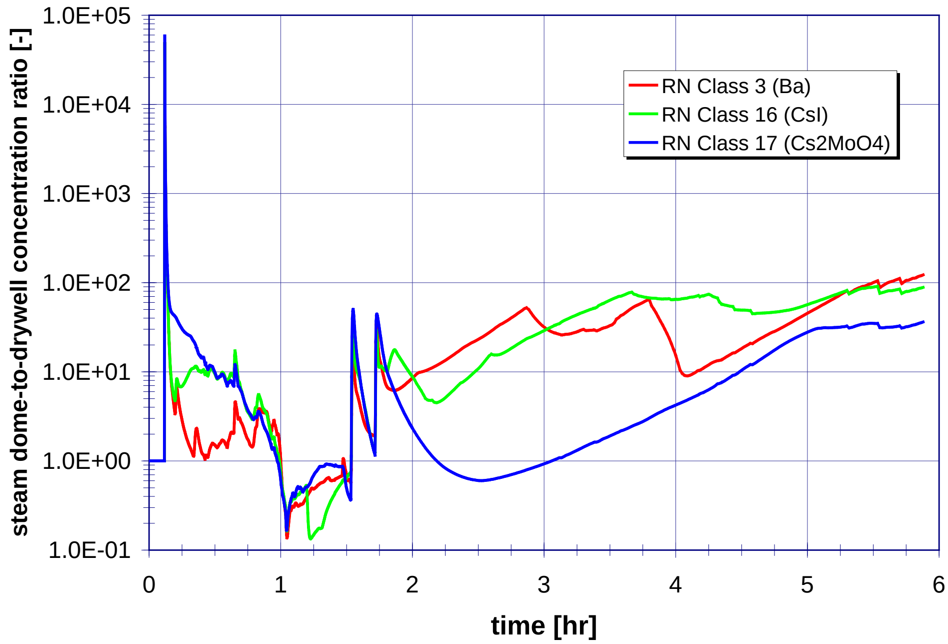


Figure 2-32 BWR Mk-I, RLB, No Sprays: Steam Dome-to-Drywell Concentration Ratios

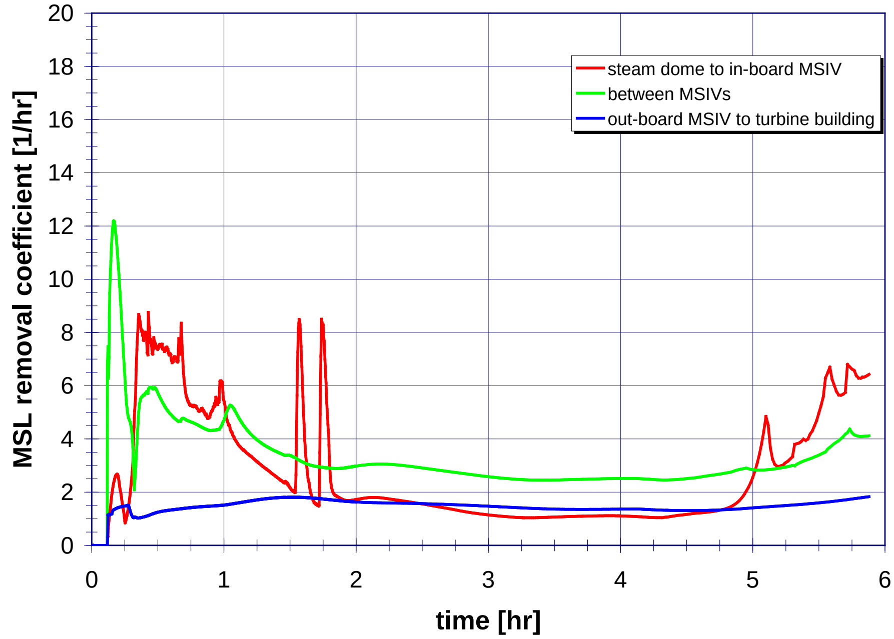


Figure 2-33 BWR Mk-I, RLB,, No Sprays: MSL-A Removal Coefficients

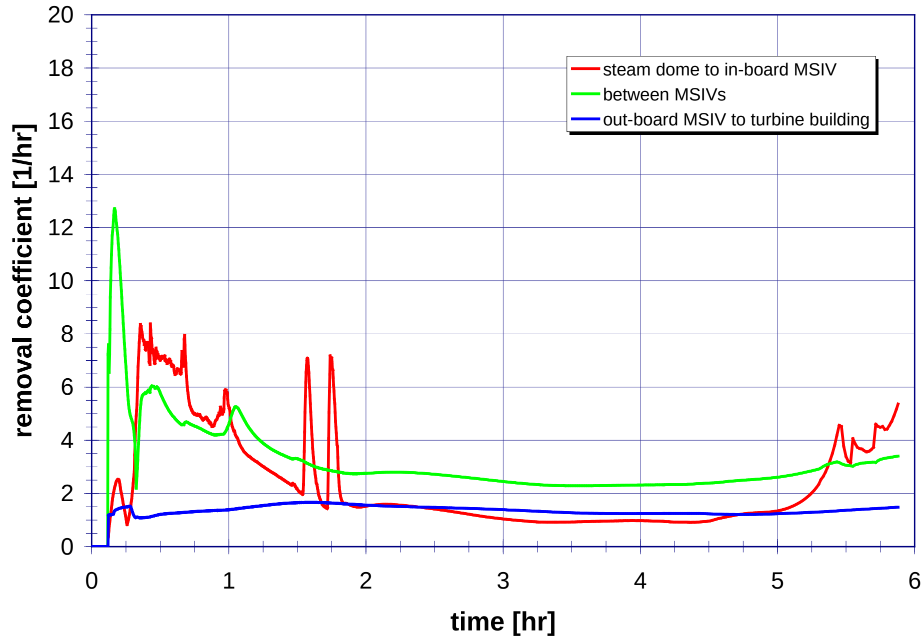


Figure 2-34 BWR Mk-I, RLB, No Sprays: MSL-B Removal Coefficients

2.2.4 Mk-I RLB, Sprays

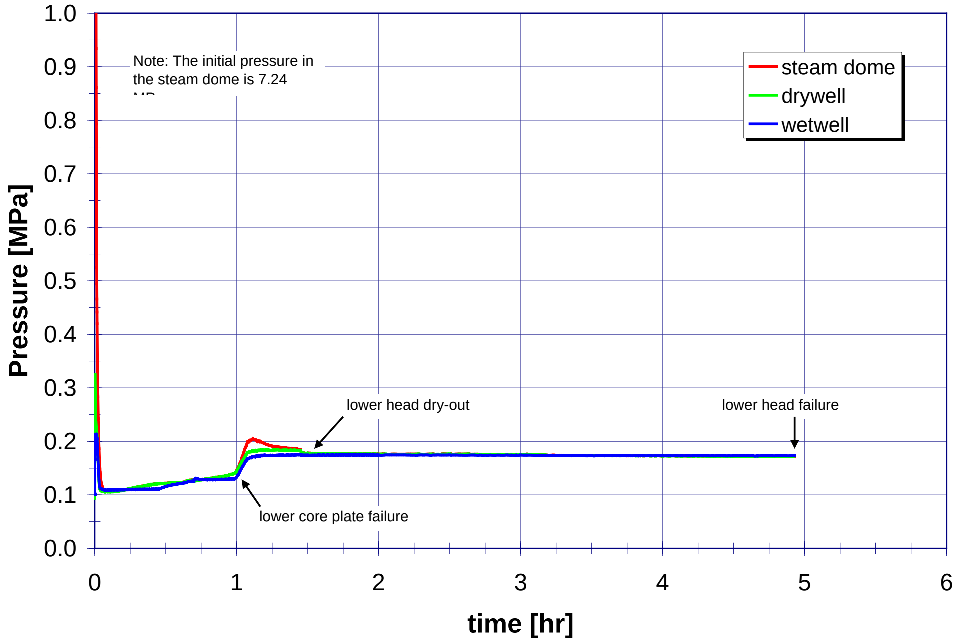


Figure 2-35 BWR Mk-I, RLB, Sprays: Steam Dome, Drywell, and Wetwell Pressure

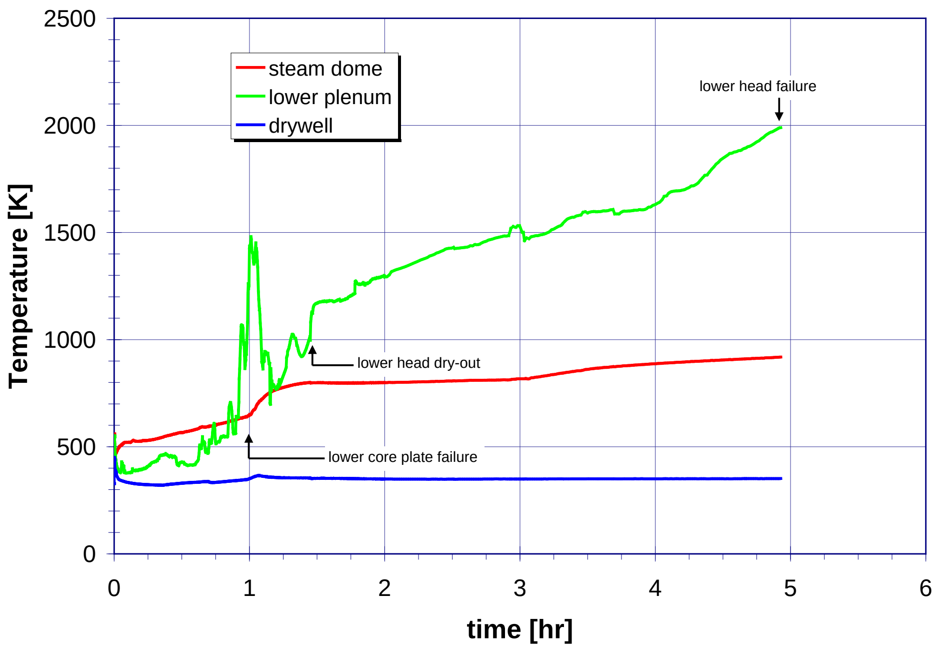


Figure 2-36 BWR Mk-I, RLB, Sprays: Steam Dome, Drywell, and Lower Plenum Vapor Temperature

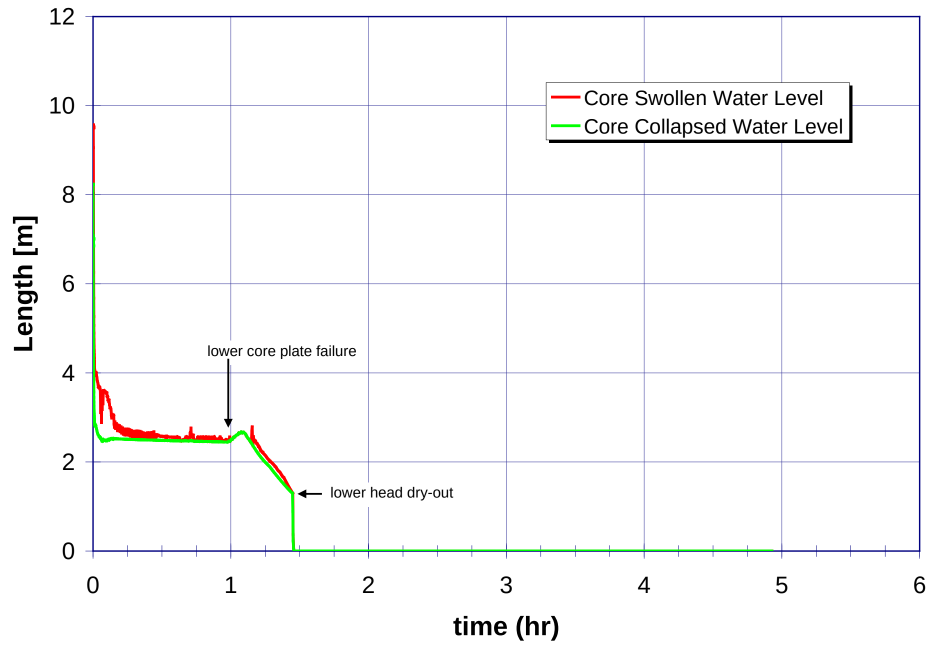


Figure 2-37 BWR Mk-I, RLB, Sprays: Core Water Levels

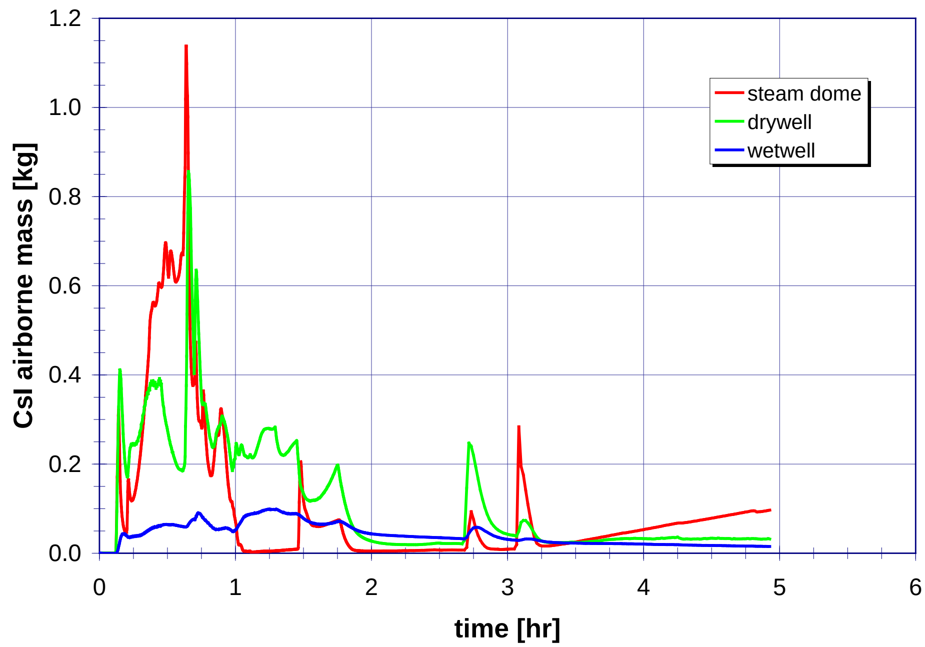


Figure 2-38 BWR Mk-I, RLB, Sprays: CsI Mass in the Steam Dome, Drywell, and Wetwell

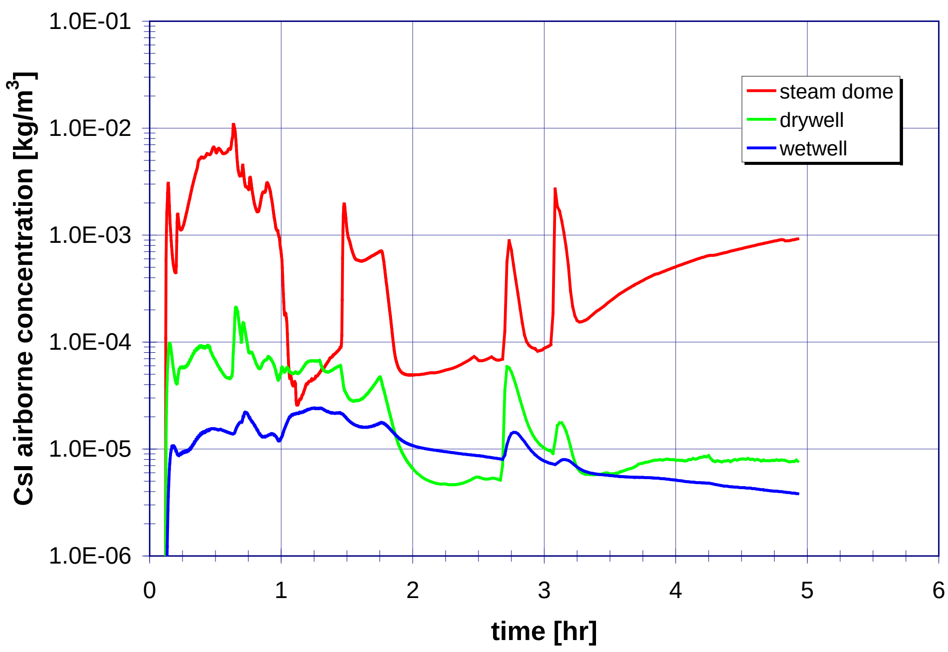


Figure 2-39 BWR Mk-I, RLB, Sprays: CsI Concentration in the Steam Dome, Drywell, and Wetwell

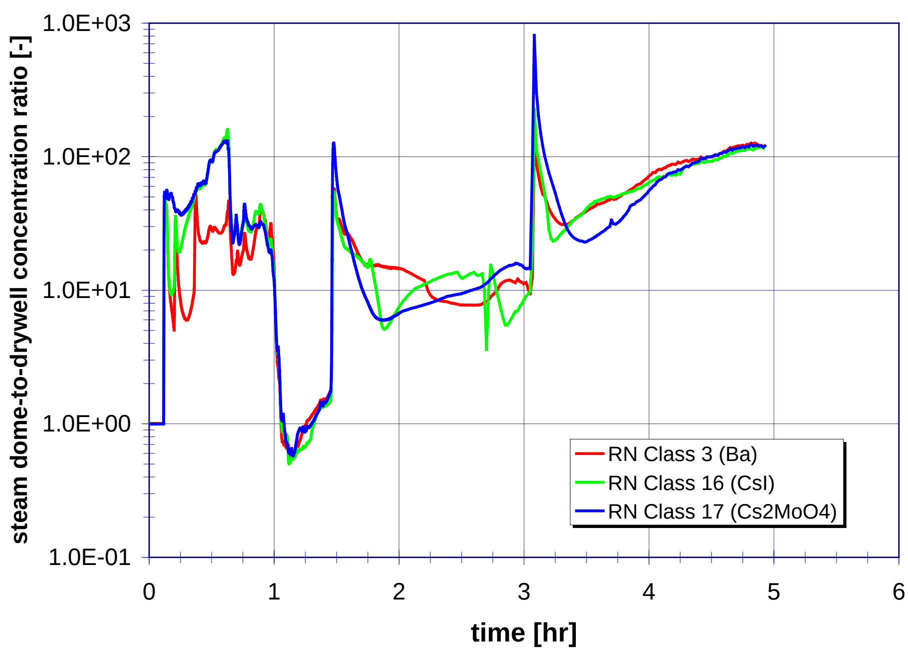


Figure 2-40 BWR Mk-I, RLB, Sprays: Steam Dome-to-Drywell Concentration Ratios

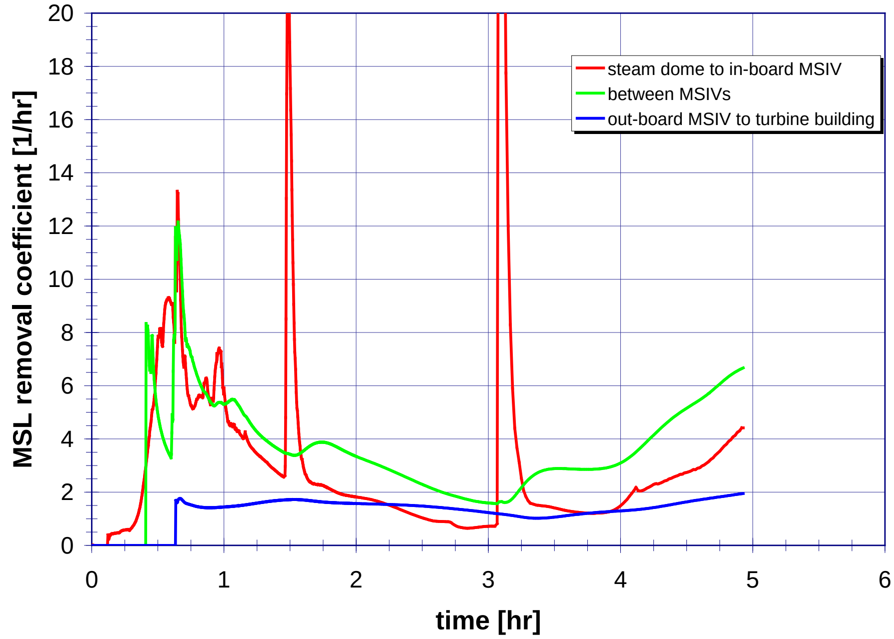


Figure 2-41 BWR Mk-I, RLB, Sprays: MSL-A Removal Coefficients

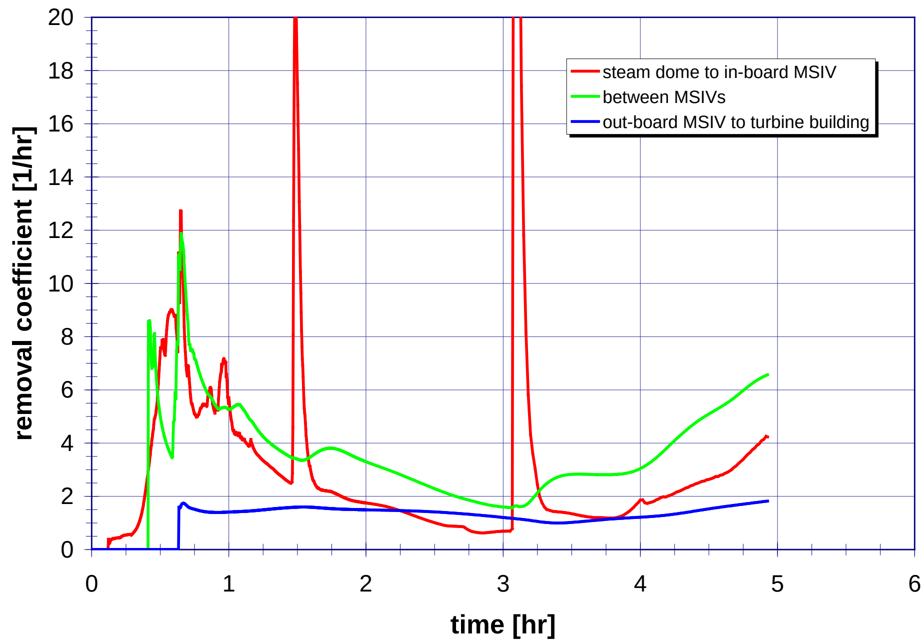


Figure 2-42 BWR Mk-I, RLB, Sprays: MSL-B Removal Coefficients

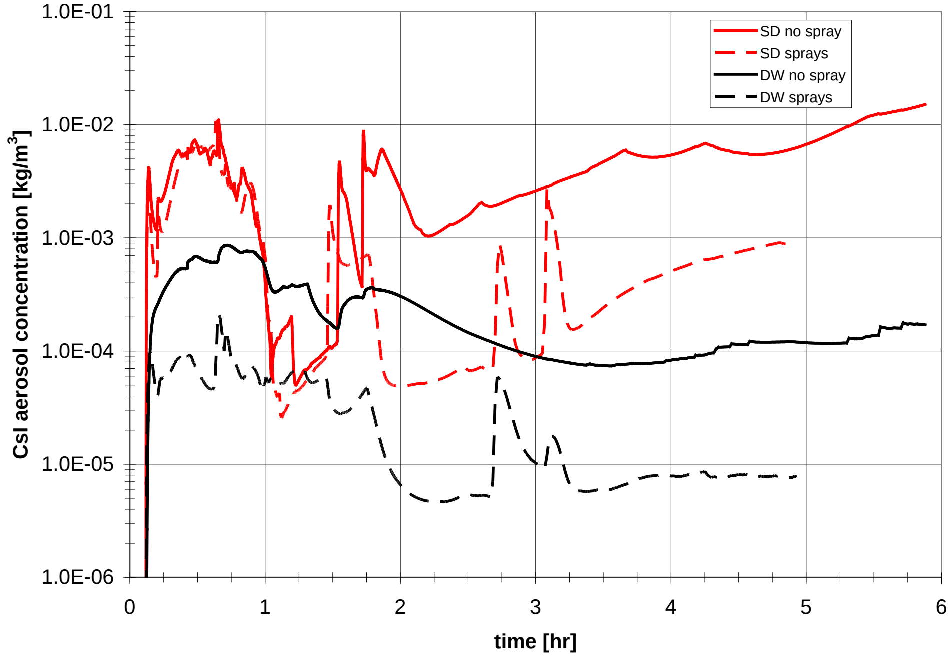


Figure 2-43 BWR Mk-I, RLB, Comparison of CsI Aerosol Steam Dome and Drywell Concentrations with and without Sprays

2.2.5 Evaluation of Post-Reflood Conditions: Effect on Deposition in Steam Lines

One MELCOR analysis for the Mk-III plant was dedicated to including vessel reflood just prior to lower head failure. The purpose of this analysis was to assess deposition processes, in addition to aerosol and thermal hydraulic behavior, after reflood in order to provide guidance for RADTRAD calculations during this time period. Figure 2-44 shows that the airborne mass in the vessel drops by several orders of magnitude following vessel reflooding, while airborne masses in the drywell and wetwell are not significantly altered from their pre-reflood trends. This happens because the core steaming during reflood advects airborne fission products in the reactor vessel into the drywell region and core reflooding terminates any addition fission product source addition from the core. This strongly suggests that following vessel reflooding, that the aerosol concentration in the drywell may be conservatively used as the source to assess MSIV leakage since most airborne activity following reflooding has been transported to the drywell by the escaping steam. At 6639.2 ft^3 the MK-III steam dome volume is approximately 40 times smaller (volume ratio of 0.02459) than the drywell volume. These values can be used to compare the relative fission airborne fission product concentrations provided in this figure.

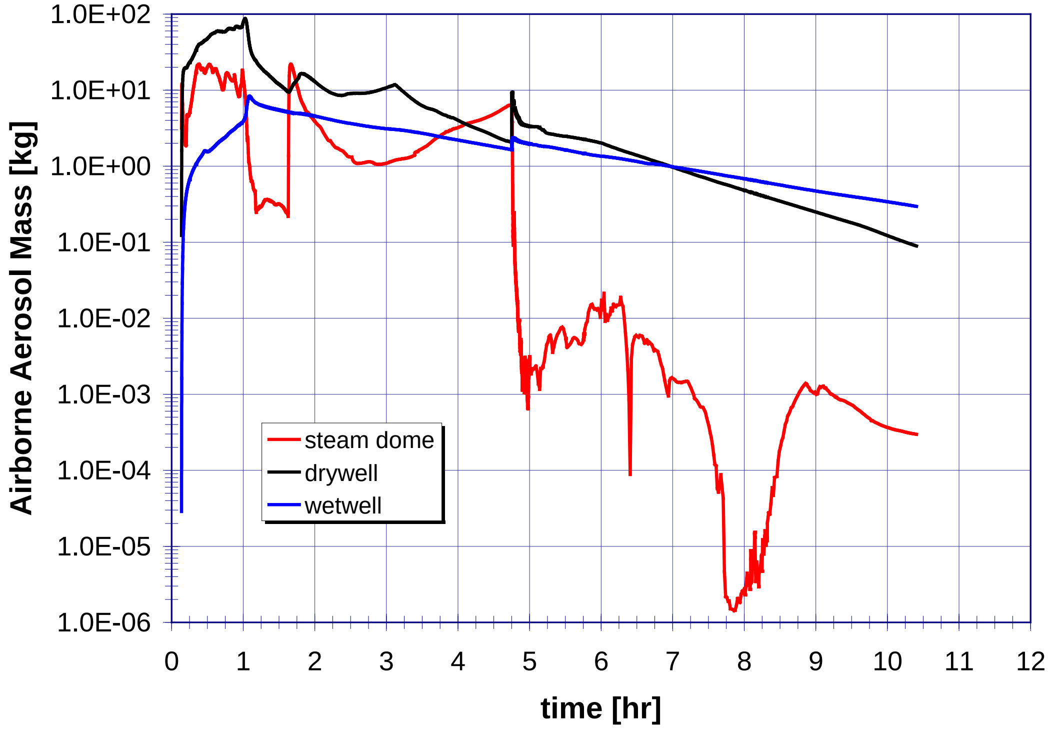


Figure 2-44 BWR Mk-III, RLB, Aerosol airborne mass before and after vessel reflooding.

Figure 2-45 shows the depletion behavior in the first few hours following vessel reflooding. Immediately after reflooding, the removal coefficient (lambda) for the intermediate steam lines is approximately 2.7 hr^{-1}, diminishing to a value of about 1 hr^{-1} after 6 hours. The removal coefficient prior to reflood is quite similar to the earlier analyses presented in this chapter; however, after reflood, the removal coefficient drops owing to the decreasing particle size in the hours following termination of the core source term. Recall that the ongoing pre-reflood core source of aerosols supported a quasi-steady large-diameter component to the vessel aerosol size distribution; after termination of this source, the large particles quickly fall out, leaving only smaller particles and a correspondingly smaller value for the removal coefficient – on the order of 1.0 hr^{-1}.

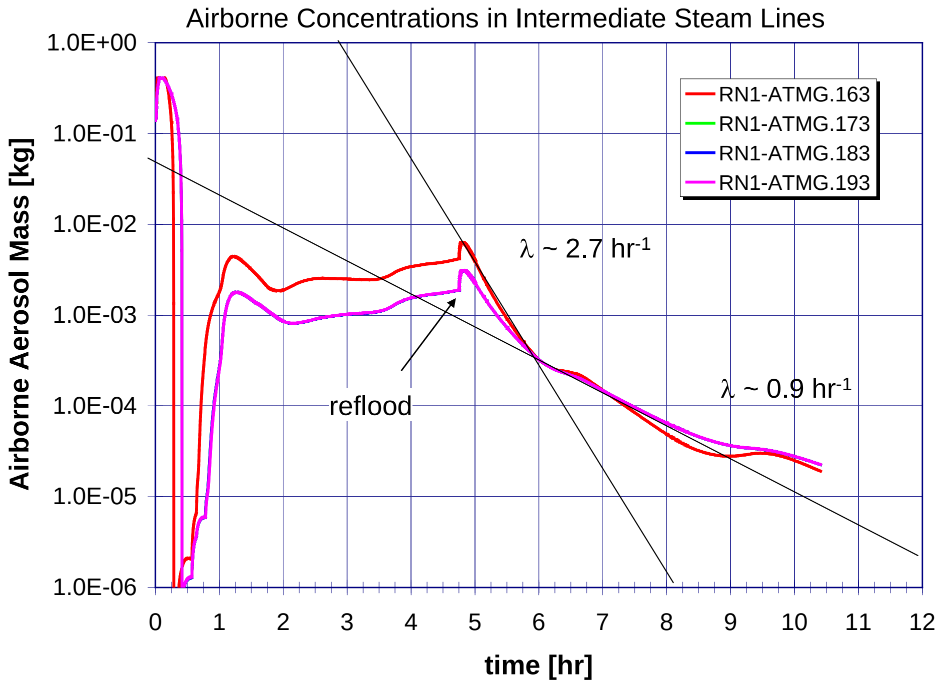


Figure 2-45 Depletion behavior in intermediate steam lines before and after vessel reflooding.

3 MELCOR Main Steam Line Uncertainty Analyses of Removal Coefficients

The results of the MELCOR full plant analyses presented in Section 2 of this report provide information for select DBA sequences for the Mk-I and Mk-III containment designs giving guidance on airborne concentrations in the steam dome and drywell regions as well as their ratio. Other information concerning calculated removal coefficients in the steam dome and steam lines for those analyses were also presented.

A further objective of this study was to explore the potential variability of the removal coefficients as affected by uncertainties in aerosol physics, variability in valve leak area and variances between plant designs in the lengths of steam line piping. In order to estimate these variances, a Monte Carlo method was used to sample over uncertainty distributions estimated for the key uncertain parameters. Since the full fidelity MELCOR model for these plants requires many days of computer execution time, and sample sizes of 150 were desired in order to obtain 95% confidence in the 95th percentile, a faster-running model was developed to explore these uncertainties.

This was done by building a model that consisted of only the reactor steam dome volume, the steam lines, MSIV's and condenser, as shown in Figure 3-1 below. Results from the full reactor MELCOR models (e.g., T-H, airborne radionuclide mass, post-reflood T-H conditions) are used as the boundary conditions to "drive" the MSL-only model such that identical environmental conditions with the full plant MELCOR analysis are maintained in the steam dome volume. It should be pointed out that the MELCOR full plant analyses were calculated out to about 5 hours, whereas the uncertainty analyses performed using the simplified nodalization were calculated out to 24 hours. For the time period beyond 5 hours, the MELCOR thermal hydraulic boundary conditions to the steam dome were held constant, and the aerosol source was terminated to approximate the anticipated effects of reflooding which would effectively cease releases from the core. The residual airborne aerosols in the steam dome after this time are allowed to deplete by gravitational settling.

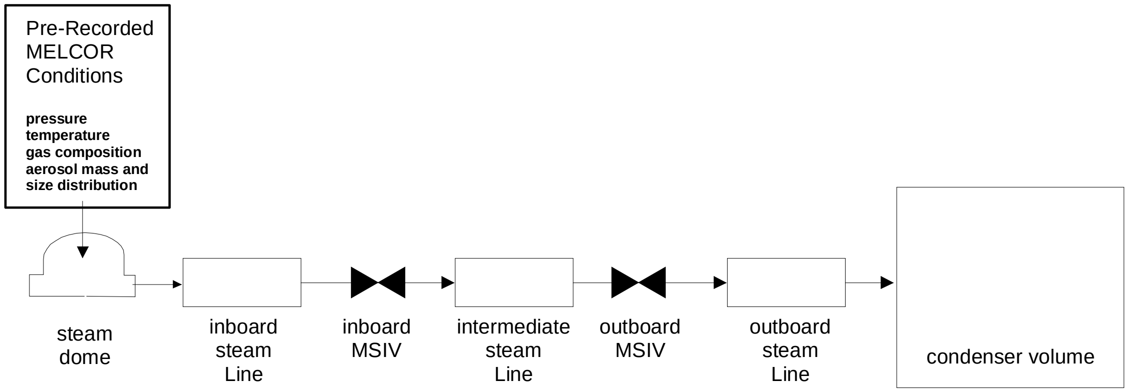


Figure 3-1 Simpified MELCOR model of steam line geometry using full plant model thermal-hydraulic and aerosol sources.

Three separate uncertainty analyses were performed with the Mk-I, MSL model (no sprays, with condenser) using steam dome conditions recorded from the Mk-I RLB scenario described earlier in section 2.2.3. These analyses separately looked at uncertainty in aerosol physics parameters, potential variability in horizontal MSL pipe lengths, and uncertainty in MSIV leakage.

3.1 Aerosol Physics Uncertainties

The following aerosol physics parameters were considered as uncertain in the MSL-only analyses. These are aerosol physics parameters in the MELCOR MAEROS models which affect aerosol settling rate, agglomeration rate and deposition processes. Their distributions were taken from aerosol deposition uncertainty analyses that were previously performed for AP1000 and ESBWR uncertainty analyses. The distributions and their rationale are provided in the following from an earlier report [8] addressing aerosol depletion behavior in the AP-1000 design certification activities. The distributions used to characterize parameter uncertainty were selected based on engineering judgment relative to the degree of certainty held by the authors and, where considerable uncertainty existed, uniform or log-uniform distributions were used.

3.1.1 Chi and Gamma: Aerosol Dynamic and Agglomeration Shape Factors

Both of these parameters were considered to be the same value, and were represented with a non-symmetric Beta distribution as shown in Figure 3-2.

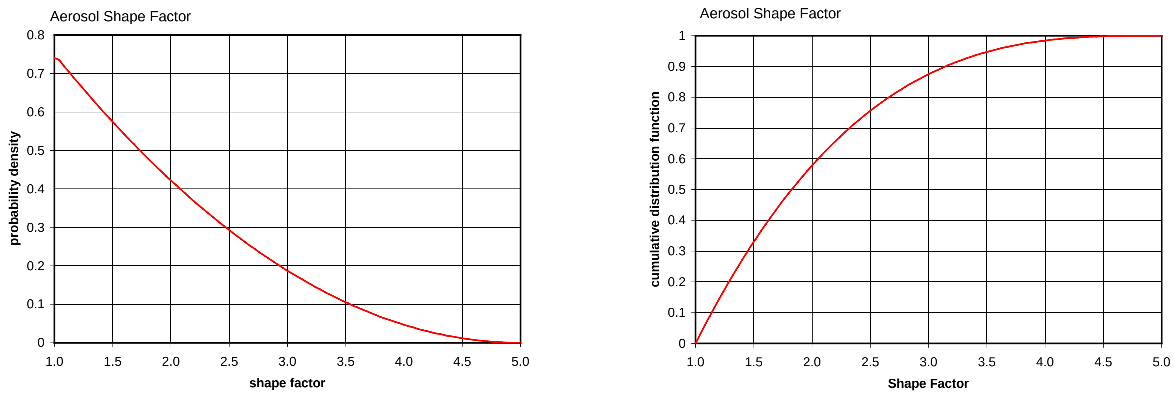


Figure 3-2 Distribution for Chi and Gamma: Dynamic and agglomeration aerosol shape factors.

The particular selection of P and Q (p=1,q=3) produced a distribution that was biased towards 1.0, with diminishing likelihood for Chi and Gamma as the limit of 5 is approached. This specification expresses the belief that the shape factor lies closer to the range of 1 to 3 with diminishing likelihood of having values approaching 5. The lower bound of 1.0 represents perfectly spherical aerosol particles and the upper bound of 5 represents chains of particles. It is rationalized that hygroscopic effects will induce some condensation of moisture on the particles causing the particles to tend towards being spherical and limiting the degree of non-spherical shape.

3.1.2 FSlip: Particle Slip Coefficient in Cunningham Formula

The factor FSlip introduces a correction to the Cunningham slip factor. Its default value in MELCOR is 1.257. Without further justification, since this parameter in MELCOR was specified to 3 places of accuracy, the uncertainty band for this parameter was restricted to lie between 1.2 and 1.3, or roughly a +/- 5% variation, using a Beta distribution as shown in Figure 3-3. The form of the Cunningham factor is shown below.

$$C_m = 1 + \frac{2\lambda}{d_p}\left[F_{slip} + 0.4\exp\left(-1.1 d_p / 2\lambda\right)\right] \tag{3.1}$$

where

λ = mean free path of air at 298 K (~ 0.069 • 10^{-6}m)

F_{slip} = slip factor specified on Input Record RNMS000 (default value of 1.257)

d_p = the particle diameter (m)

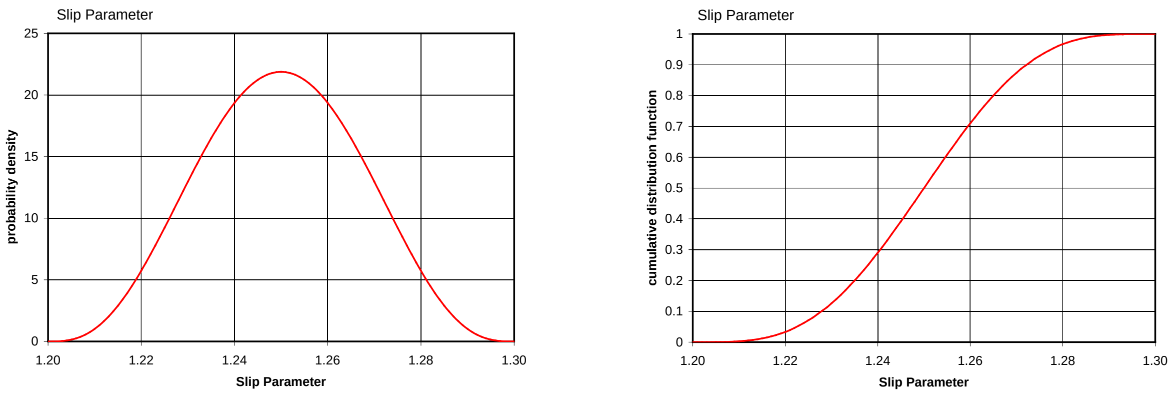


Figure 3-3 FSlip: factor for the Cunningham correction factor.

The Cunningham factor appears in the gravitational settling term,

$$v_{grav} = \frac{d_p^2 \rho_p g C_m}{18 \mu \chi} \tag{3.2}$$

where

v_{grav}	= the downward terminal velocity (m/s)
d_p	= the particle diameter (m)
ρ_p	= the particle density (kg/m^3)
g	= acceleration of gravity = 9.8 m/s^2
C_m	= the particle mobility, or Cunningham slip correction factor, which reduces the Stokes drag force to account for noncontinuum effects.
μ	= viscosity of air at 298 K [~ 1.8 ● 10^5(N ● s/m^2)]
χ	= dynamic shape factor

The factor FSlip also appears in the thermophoretic deposition formulation in MELCOR, discussed later.

3.1.3 *Fstick – Sticking probability for Agglomeration*

The rate of agglomeration is affected by the probability that a collision between two particles results in the two particles actually sticking together. Often this factor is taken as 1.0; however, this may depend on the wetness of the particles and could be influenced by electrostatic phenomena; like-charged particles that might otherwise collide and stick may instead fail to collide as their distance of separation closes. The uncertainty for this parameter was specified as shown in Figure 3-4 using a Beta distribution (p=2.5, q=1) where values nearer to 1.0 are favored. The median value for the sticking probability is 0.88.

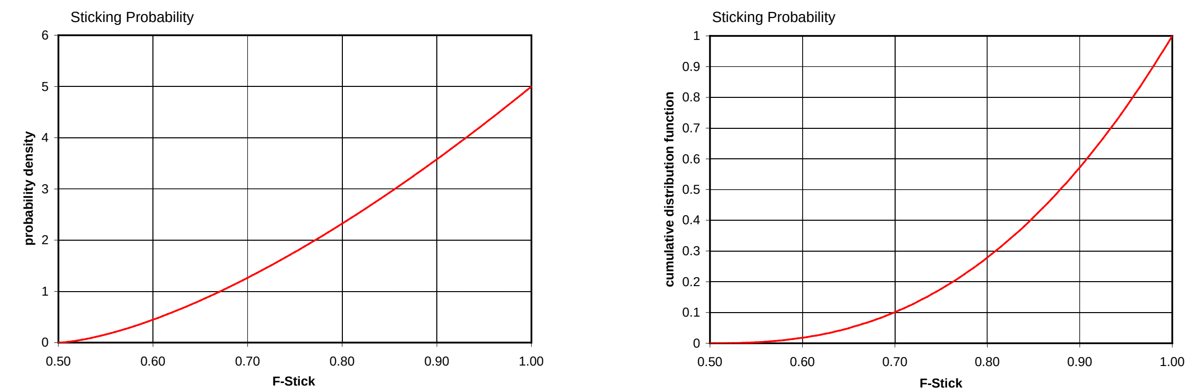


Figure 3-4. Fstick: agglomeration sticking probability.

3.1.4 Boundary layer thickness for Diffusion Deposition:

The boundary layer thickness used in calculating the deposition by diffusion is nominally specified in MELCOR as 10 micrometers as used in the following expression for Brownian deposition velocity:

$$v_{diff} = \frac{k\,T\,C_m}{3\pi\,\mu\,\chi\,d_p\Delta} \tag{3.3}$$

where

v_{diff}	=	diffusion deposition velocity (m/s)
k	=	Boltzmann constant = $1.38 \bullet 10^{-7}$ (J/K)
T	=	atmosphere temperature (K)
μ	=	viscosity ($N \bullet s/m^2$)
χ	=	dynamic shape factor (CHI)
Δ	=	user-specified diffusion boundary layer thickness specified on Input Record RNMS000 (default value of 10^{-5} m)
C_m	=	Cunningham correction factor

In this uncertainty analysis, this value was considered uncertain between the limits of 5 and 20 micrometers, distributed uniformly as shown in Figure 3-5.

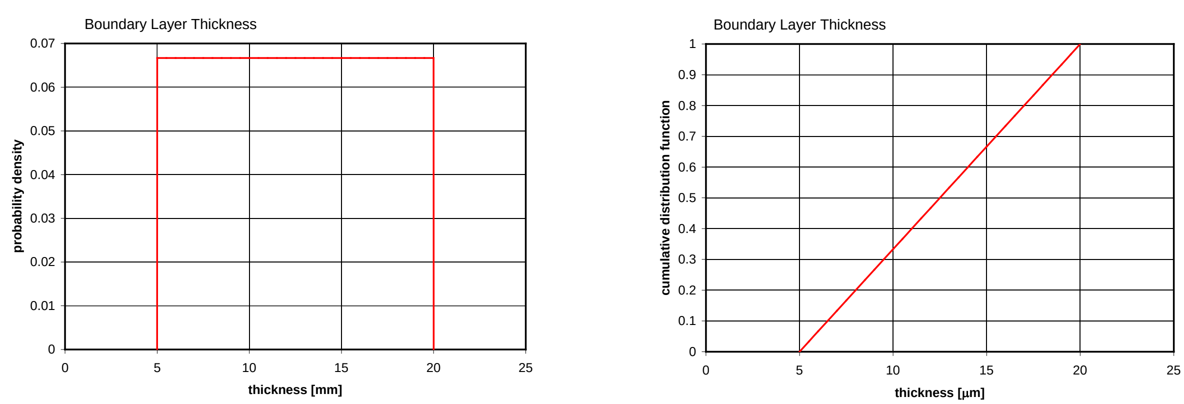


Figure 3-5. DELDF: Diffusion boundary layer thickness.

3.1.5 Thermal Accommodation Coefficient for Thermophoresis

The thermal accommodation coefficient in the expression for thermophoretic deposition velocity is accessible in MELCOR via the factor c_t as shown in the equation below

.

$$v_{therm} = \frac{3\,\mu\, C_m \left(c_t\, Kn + k_{gas}/k_p\right)}{2\;\chi\;\;\rho_{gas}\, T\,\left(1 + 3\, F_{slip} Kn\right)\left(1 + 2\, c_t\;Kn + k_{gas}/k_p\right)}\;\nabla T \qquad (3.4)$$

K_n = $2\lambda/d_p$ (Knudsen number)
k_{gas}/k_p = ratio of thermal conductivity of gas over that for aerosol particle and is user-specified (on Input Record RNMS000) – also uncertain in this study
∇T = structure surface temperature gradient (K/m)
ρ_{gas} = gas density (kg/m^3)
T = wall temperature (K)
F_{slip} = slip factor
c_t = constant associated with the thermal accommodation coefficients (specified on Input Record RNMS000 with default value of 2.25)

This coefficient, c_t, was considered uncertain between the limits of 2.0 and 2.5 using a uniform distribution as shown in Figure 3-6 based on the precision of the default value at two decimal places.

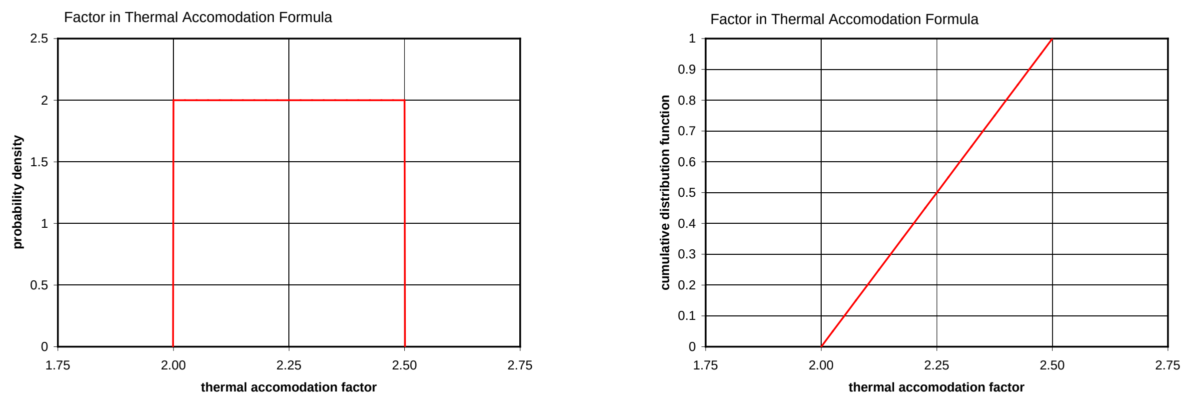


Figure 3-6 Thermal accommodation coefficient in thermophoretic deposition.

3.1.6 Ratio of Thermal Conductivity of particle to gas: TKGOP

Also appearing in the formulation for thermophoretic deposition is the ratio of gas to aerosol particle thermal conductivity, nominally specified in MELCOR as 0.05. This factor is treated as uncertain between the limits of 0.006 and 0.06, based on an inspection of the conductivity of gases (steam, H_2) and aerosol (UO_2, Ag). The range was taken to be distributed log-uniformly owing to the wide range of possible values for this parameter.

Also appearing in the thermophoretic deposition formulation is the factor FSlip, discussed previously in the description of the Cunnningham factor.

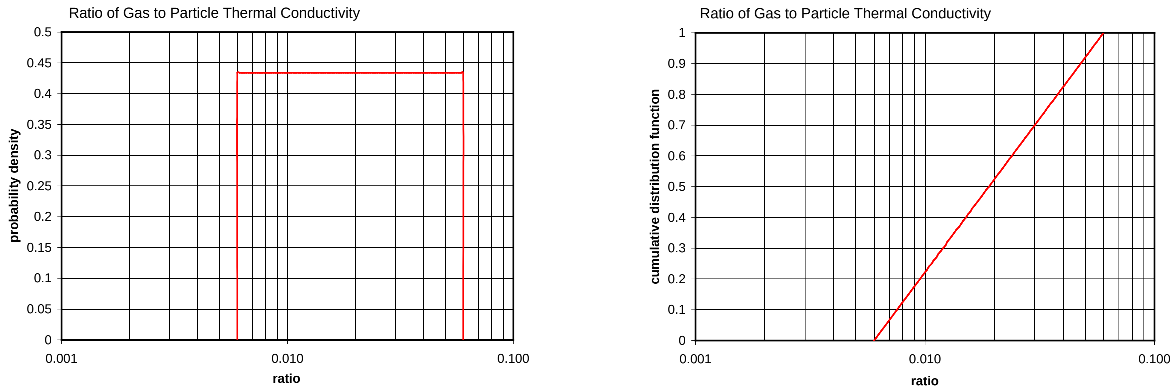


Figure 3-7. TKGOP - Ratio of gas to particle thermal conductivity.

3.1.7 Turbulent Energy Dissipation Factor: TURBDS

The turbulent energy dissipation factor, [default 0.001 m^2/s^3] appears in the agglomeration coefficients in the turbulent shear and turbulent inertial terms [9]. This factor was considered uncertain at +/- 25% as shown in Figure 3-8. A uniform distribution was used with limits between 0.00075 and 0.00125.

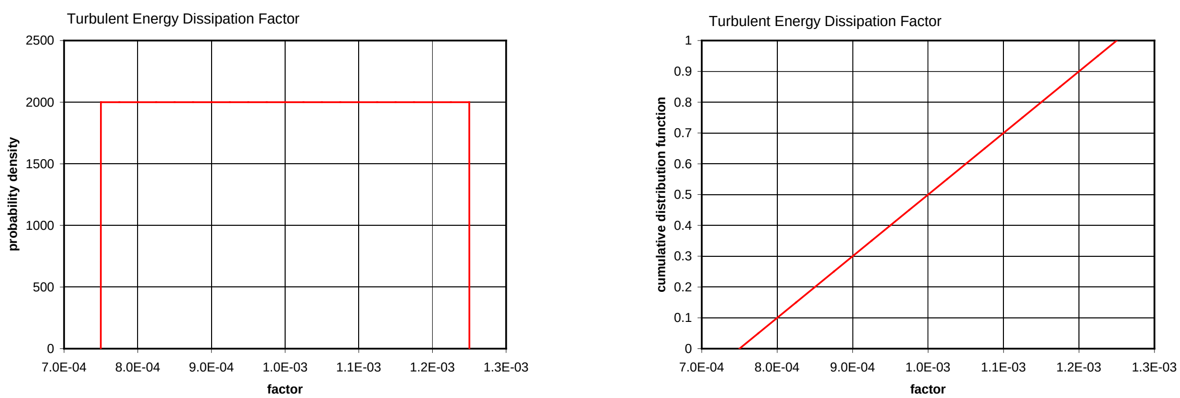


Figure 3-8. Turbulent energy dissipation factor: TURBDS

3.1.8 Multipliers on Heat Transfer and Mass Transfer

The heat and mass transfer coefficients in MELCOR affect the rate of steam condensation, which in turn has a strong effect on the diffusiophoretic deposition calculated for the airborne aerosol in deposition volumes. Since both heat and mass transfer is calculated in MELCOR from the same Nusselt number, we take these two parameters to be correlated. That is, a high heat transfer coefficient should be accompanied by a correspondingly high mass transfer coefficient. These scaling factors for the heat and mass transfer were taken to be roughly 25% above or below the nominal values calculated in MELCOR for the conditions in the containment atmosphere, with diminishing likelihood at the extremes of this range. This belief was represented using a Beta distribution with limits of 0.75 and 1.25, and p and q equal to1.5, as shown in Figure 3-9.

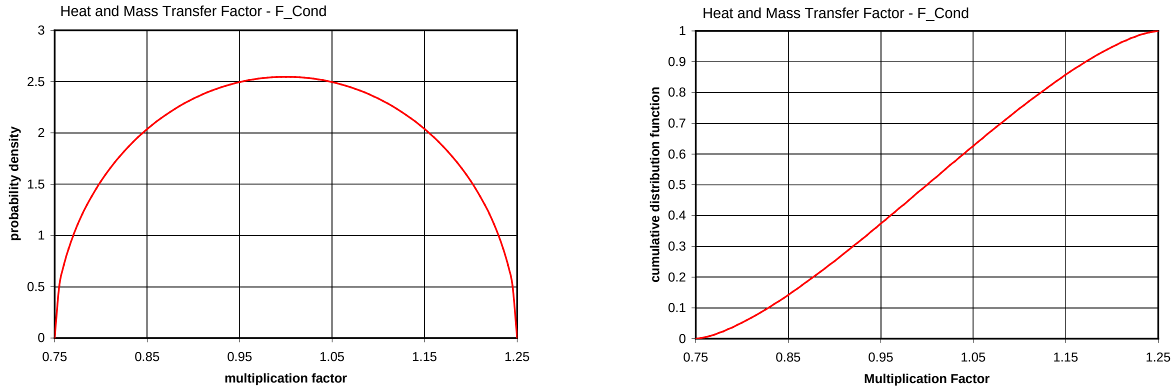


Figure 3-9. Multiplier to heat and mass transfer coefficients for containment shell.

3.1.9 Results of Uncertainty on Aerosol Physics

The aerosol physics parameters treated as uncertain are summarized in the following table. Each of these uncertain parameters was sampled independently (Monte Carlo) to form 150 separate MELCOR analyses. The valve leakage areas for the MSIVs in each loop were set to conform to the assumed leakages as described in 2.1.1. The instantaneous “lambda” values were calculated using the method described in Section 2.1.2 for each section of the MSL illustrated in Figure 3-1, and are displayed in Figure 3-10 through Figure 3-16.

Table 3-1 Aerosol Physics Uncertain Parameters

Aerosol Physics Parameter	Distribution[‡]
aerosol dynamic shape factor/collision factor (-)	Beta: p = 1.0; q = 1.5; min = 1.0; max = 5.0
diffusion boundary layer thickness (m)	Uniform: min = 0.000005; max = 0.0002
slip factor (-)	Beta: p = 4.0; q = 4.0; min = 1.2; max = 1.3
sticking probability (-)	Beta: p = 2.5; q = 1.0; min = 0.5; max = 1.0
thermal accommodation coefficient (-)	Uniform: min = 2.0; max = 2.5
thermal conductivity ratio (-)	log-uniform: min = 0.006; max = 0.06
turbulent dissipation rate	Uniform: min = 0.00075; max = 0.00125
mass transfer coefficient scaling factor (-)	Beta: p = 1.5; q = 1.5; min = 0.75; max = 1.25

[‡] p and q are the two shape parameters that define a beta distribution $f(x,p,q) = \dfrac{x^{p-1}\cdot(1-x)^{q-1}}{\int_0^1 u^{p-1}\cdot(1-u)^{q-1}du}$.

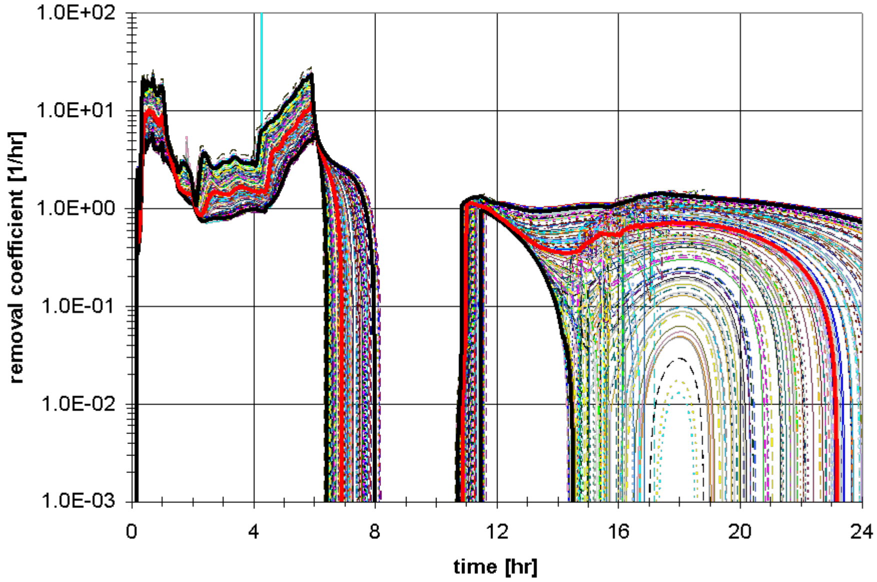


Figure 3-10 Mk-I RLB, Removal Coefficients with Aerosol Uncertainty, MSL-A, In-Board – no sprays, condenser

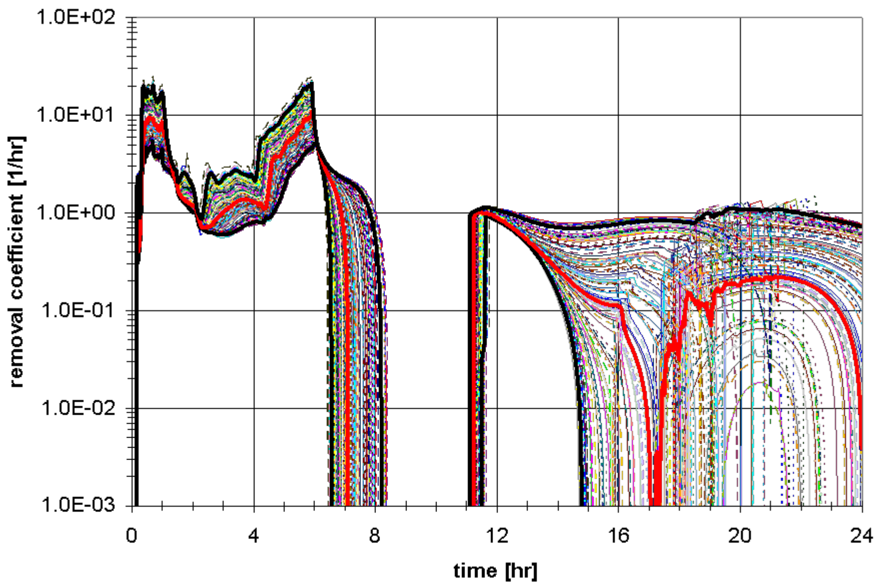


Figure 3-11 Mk-I RLB, Removal Coefficients with Aerosol Uncertainty, MSL-B, In-Board – no sprays, condenser

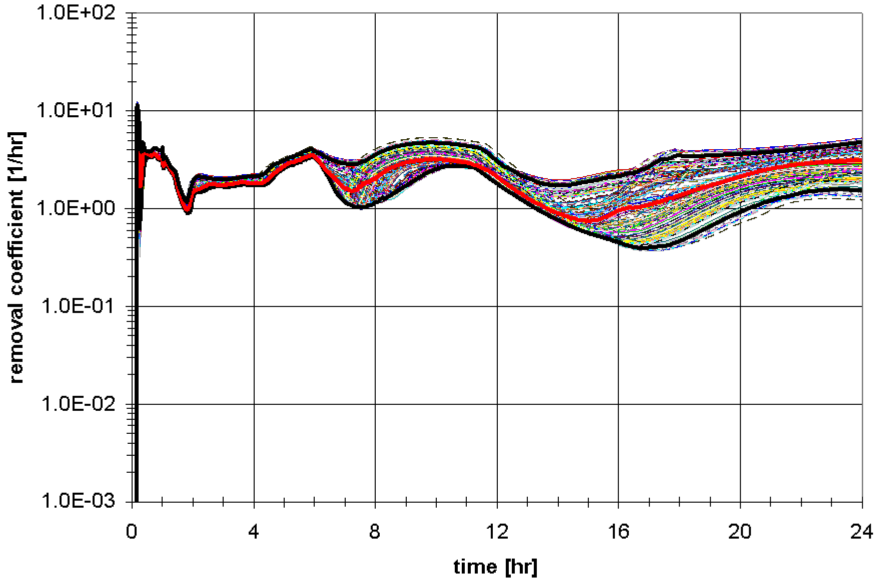


Figure 3-12 Mk-I RLB, Removal Coefficients with Aerosol Uncertainty, MSL-A, Between MSIVs – no sprays, condenser

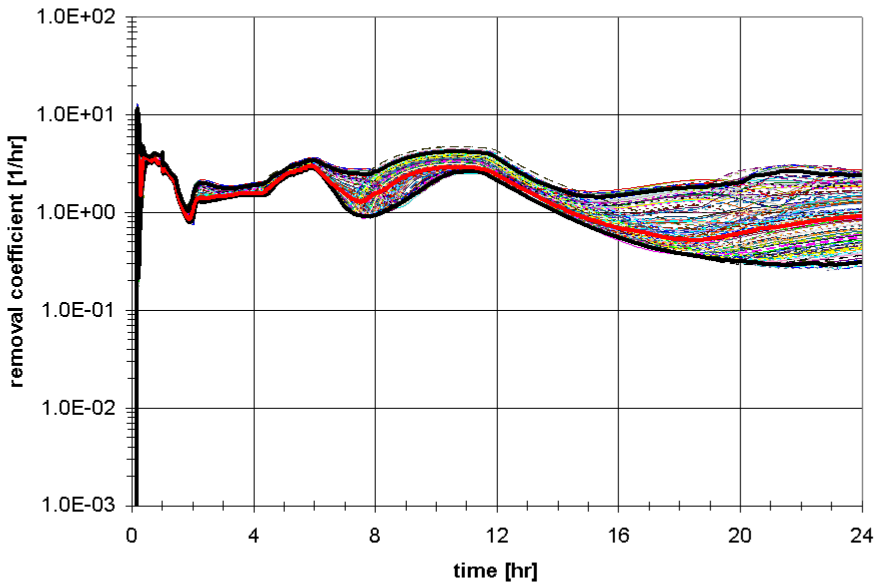


Figure 3-13 Mk-I RLB, Removal Coefficients with Aerosol Uncertainty, MSL-B, Between MSIVs – no sprays, condenser

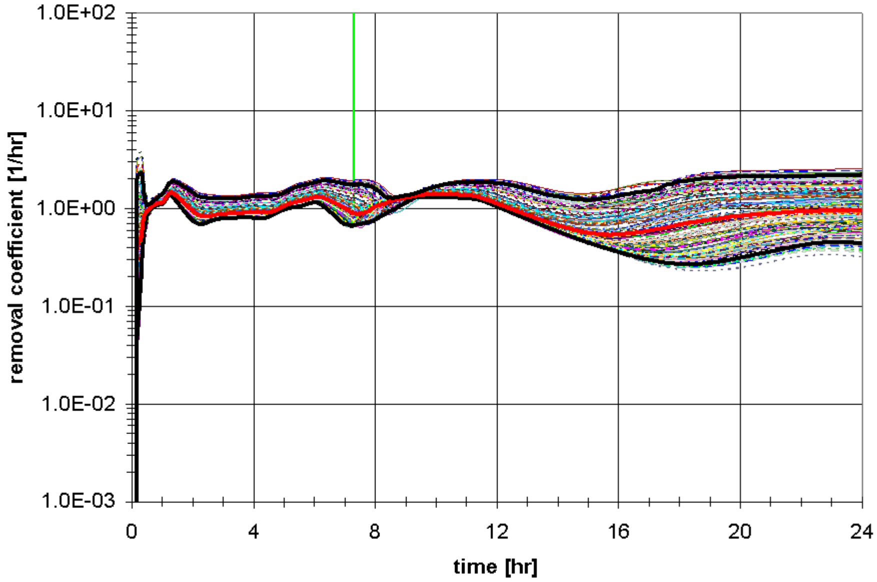


Figure 3-14 Mk-I RLB, Removal Coefficients with Aerosol Uncertainty, MSL-A, Outboard – no sprays, condenser

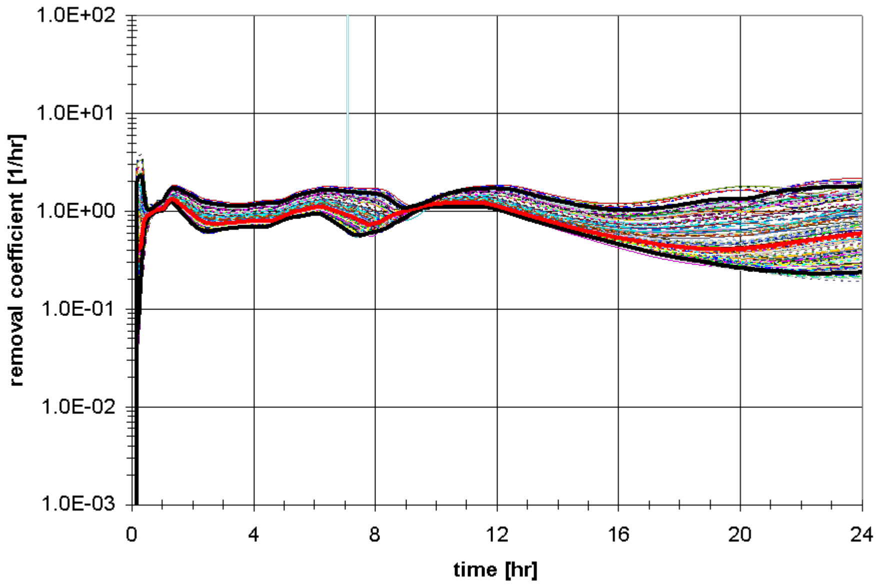


Figure 3-15 Mk-I RLB, Removal Coefficients with Aerosol Uncertainty, MSL-B, Outboard – no sprays, condenser

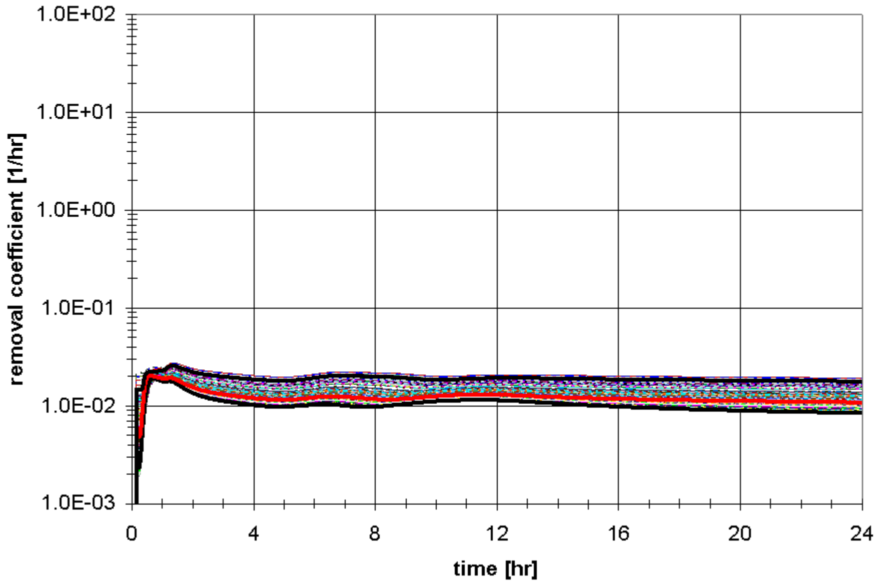


Figure 3-16 Mk-I RLB, Removal Coefficients with Aerosol Uncertainty, Condenser – no sprays, condenser

Table 3-2: Aerosol Uncertainty Analysis MSL and Condenser Removal Coefficients (5th Percentile)

MSL section	0 - 2 (hr)	2 - 12 (hr)	12+ (hr)
MSL-A in-board	2.6	0.85	0.071
MSL-A between MSIVs	2.2	2.0	0.93
MSL-A out-board	0.87	0.99	0.44
MSL-B in-board	2.4	0.77	0.086
MSL-B between MSIVs	2.0	1.8	0.61
MSL-B out-board	0.82	0.84	0.43
condenser	0.014	0.011	0.010

note. removal coefficients are given in 1/hr

Table 3-3: Aerosol Uncertainty Analysis MSL and Condenser Removal Coefficients (50th Percentile)

MSL section	0 - 2 (hr)	2 - 12 (hr)	12+ (hr)
MSL-A in-board	4.4	1.7	0.47
MSL-A between MSIVs	2.5	2.4	1.8
MSL-A out-board	1.0	1.1	0.78
MSL-B in-board	4.1	1.5	0.20
MSL-B between MSIVs	2.4	2.1	0.89
MSL-B out-board	0.94	0.97	0.57
condenser	0.016	0.012	0.012

note. removal coefficients are given in 1/hr

Table 3-4: Aerosol Uncertainty Analysis MSL and Condenser Removal Coefficients (95th Percentile)

MSL section	0 - 2 (hr)	2 - 12 (hr)	12+ (hr)
MSL-A in-board	8.3	3.6	1.1
MSL-A between MSIVs	3.2	3.4	3.1
MSL-A out-board	1.5	1.6	1.8
MSL-B in-board	7.7	3.2	0.88
MSL-B between MSIVs	3.1	3.0	2.1
MSL-B out-board	1.4	1.4	1.4
condenser	0.021	0.019	0.018

note. removal coefficients are given in 1/hr

3.2 Uncertainty in MSIV Leakage Area

The current Mk-I MSIV leakage is calibrated at design leakage conditions. As discussed in Section 3.1.1, this is done by determining the MSIV flowpath open fraction that produces the design leakage at the design pressure. An analysis of uncertainty in MSIV flow was performed by varying the flowpath open fraction between 50% and 150% of the value set by the design leakage conditions. The nominal valve flows for each steam line are as described in Section 3.1.1 (valve A: 205 scfh, valve B 155 scfh). Both valve leak areas were varied in concert so that the minimum flow would correspond to ~75 scfh and the maximum flow would correspond to 410 scfh)

Table 3-5 MSIV Flow Uncertain Parameters

Parameter	Distribution
MSIV flowpath open fraction scaling factor	uniform min = 0.5; max = 1.5

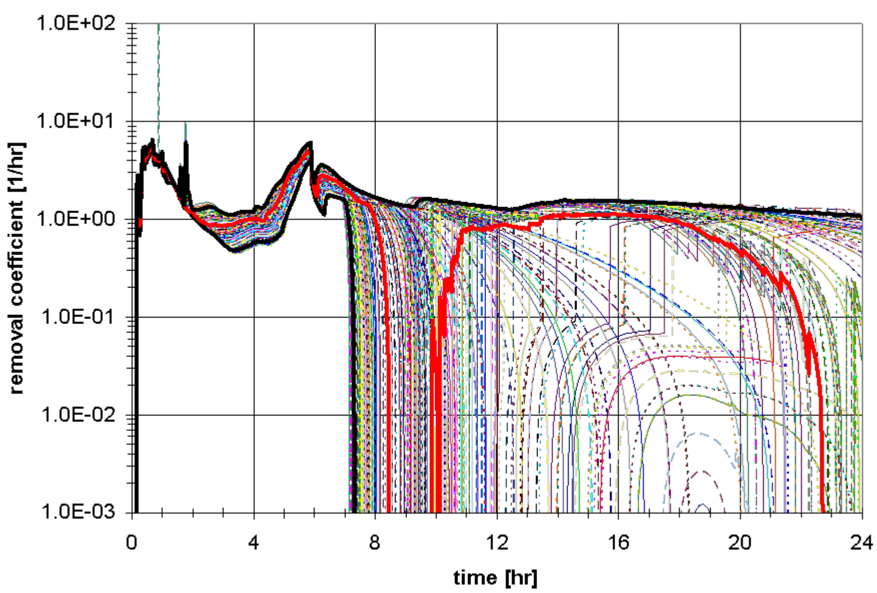


Figure 3-17 Mk-I RLB, Removal Coefficients with Flow Uncertainty, MSL-A, in-board – no sprays, condenser

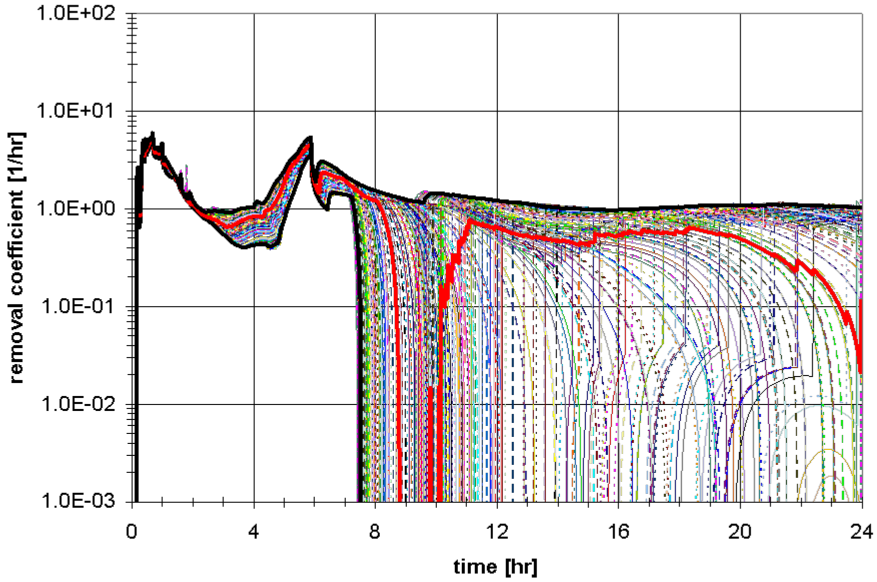


Figure 3-18 Mk-I RLB, Removal Coefficients with Flow Uncertainty, MSL-B, in-board – no sprays, condenser

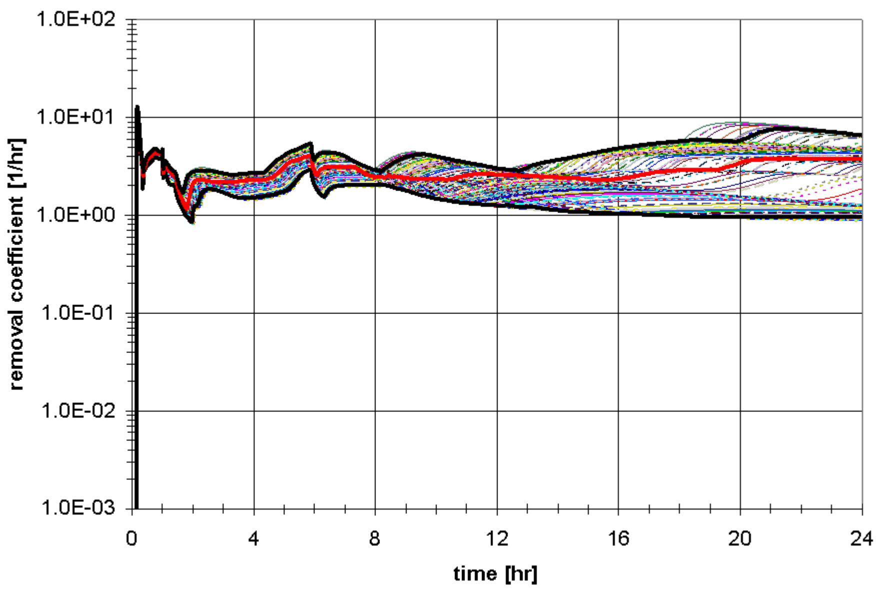


Figure 3-19 Mk-I RLB, Removal Coefficients with Flow Uncertainty, MSL-A, between MSIVs – no sprays, condenser

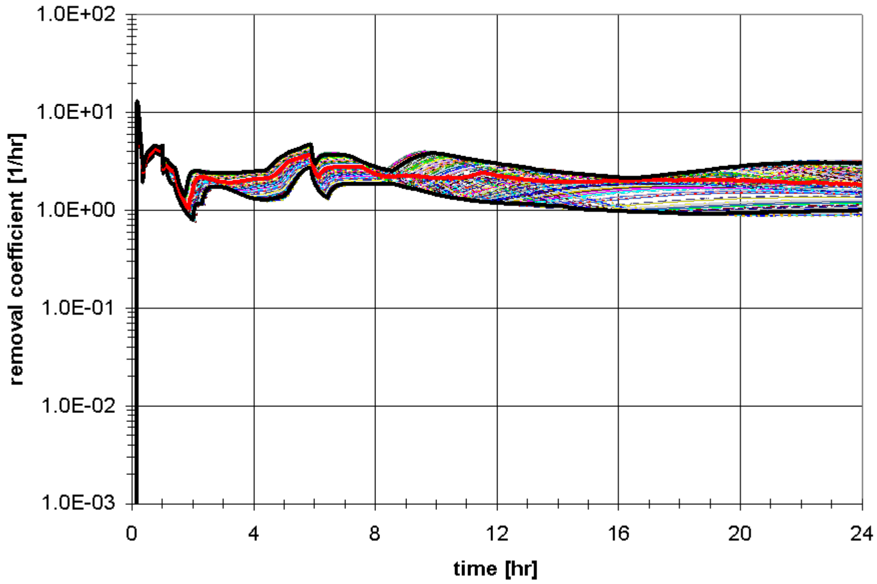


Figure 3-20 Mk-I RLB, Removal Coefficients with Flow Uncertainty, MSL-B, between MSIVs – no sprays, condenser

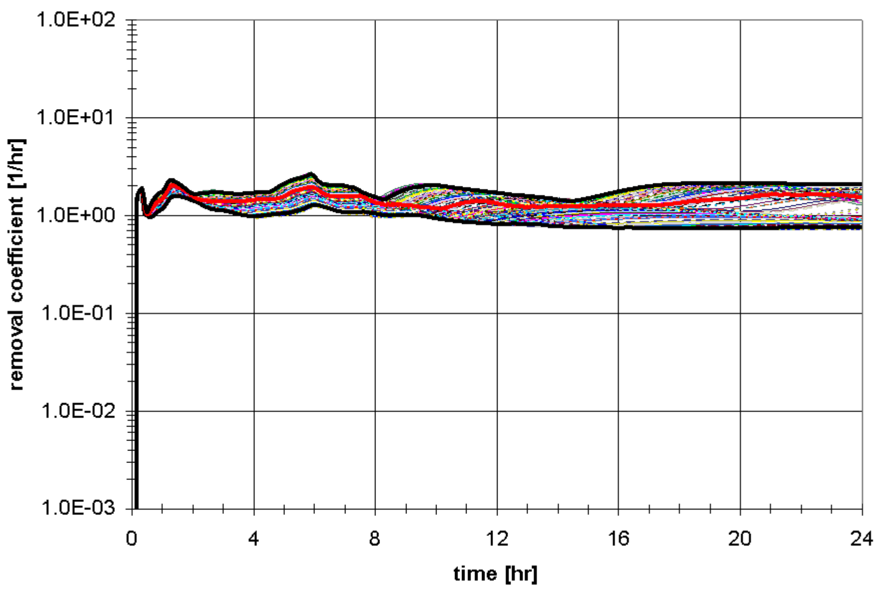


Figure 3-21 Mk-I RLB, Removal Coefficients with Flow Uncertainty, MSL-A, out-board – no sprays, condenser

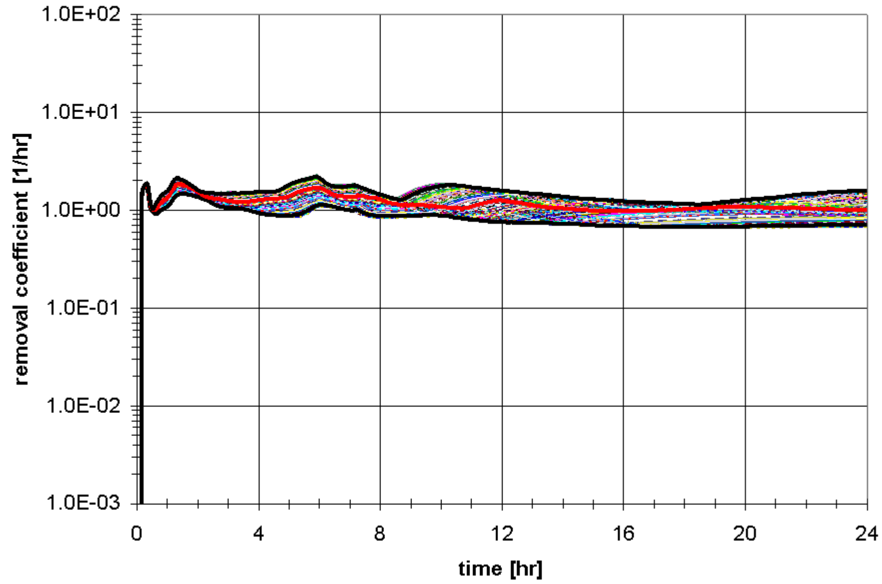


Figure 3-22 Mk-I RLB, Removal Coefficients with Flow Uncertainty, MSL-B, out-board – no sprays, condenser

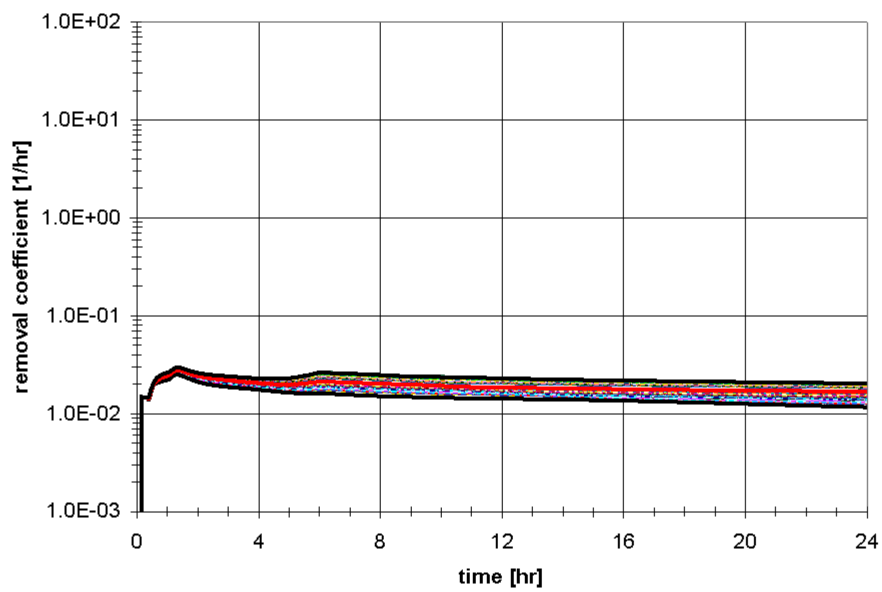


Figure 3-23 Mk-I RLB, Removal Coefficients with Flow Uncertainty, condenser – no sprays, condenser

Table 3-6: Flow Uncertainty Analysis MSL and Condenser Removal Coefficients (5th Percentile)

MSL section	0 - 2 (hr)	2 - 12 (hr)	12+ (hr)
MSL-A in-board	2.5	0.60	0.0
MSL-A between MSIVs	2.8	1.7	1.0
MSL-A out-board	1.3	1.0	0.76
MSL-B in-board	2.4	0.56	0.0
MSL-B between MSIVs	2.7	1.6	1.0
MSL-B out-board	1.2	0.95	0.70
condenser	0.020	0.016	0.013

note. removal coefficients are given in 1/hr

Table 3-7: Flow Uncertainty Analysis MSL and Condenser Removal Coefficients (50th Percentile)

MSL section	0 - 2 (hr)	2 - 12 (hr)	12+ (hr)
MSL-A in-board	2.8	1.2	0.68
MSL-A between MSIVs	3.2	2.6	3.0
MSL-A out-board	1.5	1.4	1.4
MSL-B in-board	2.6	1.1	0.44
MSL-B between MSIVs	3.1	2.4	2.0
MSL-B out-board	1.4	1.3	1.0
condenser	0.022	0.020	0.018

note. removal coefficients are given in 1/hr

Table 3-8: Flow Uncertainty Analysis MSL and Condenser Removal Coefficients (95th Percentile)

MSL section	0 - 2 (hr)	2 - 12 (hr)	12+ (hr)
MSL-A in-board	3.2	1.9	1.4
MSL-A between MSIVs	3.6	3.5	5.5
MSL-A out-board	1.7	1.9	1.9
MSL-B in-board	2.8	1.7	1.1
MSL-B between MSIVs	3.4	3.2	2.6
MSL-B out-board	1.6	1.6	1.3
condenser	0.023	0.024	0.022

note. removal coefficients are given in 1/hr

3.3 Uncertainty in Horizontal Piping Segments

MSL piping lengths vary between the various BWR reactors in the US fleet. To account for this variation the horizontal MSL piping lengths were varied. The horizontal lengths were deemed to be more important than the vertical MSL pipe lengths as aerosol deposition primarily occurs on horizontal, rather than vertical, surfaces. Rather than attempt to build MSL horizontal pipe-length distributions from plant data, scaling factors were defined with sufficiently broad lower and upper bounds. The scaling factors are then used to vary the MSL horizontal pipe lengths. For example, if the in-board scaling factor is equal to 0.5, all of the in-board MSL horizontal pipe lengths are multiplied by 0.5 (i.e., reduced by 50%).

Table 3-9 Horizontal MSL Pipe Length Uncertain Parameters

Parameter	Distribution	Piping Length Distribution [m]	Nominal Piping Length [m]
in-board MSL horizontal length scaling factor	uniform min = 0.1 max = 2.0	uniform min = 1.28 max = 25.6634	12.8
between MSIVs MSL horizontal length scaling factor	uniform min = 0.1 max = 2.0	uniform min = 0.754 max = 15.08	7.54
out-board MSL horizontal length scaling factor	uniform min = 0.1 max = 2.0	uniform min = 8.41 max = 176.8	88.4

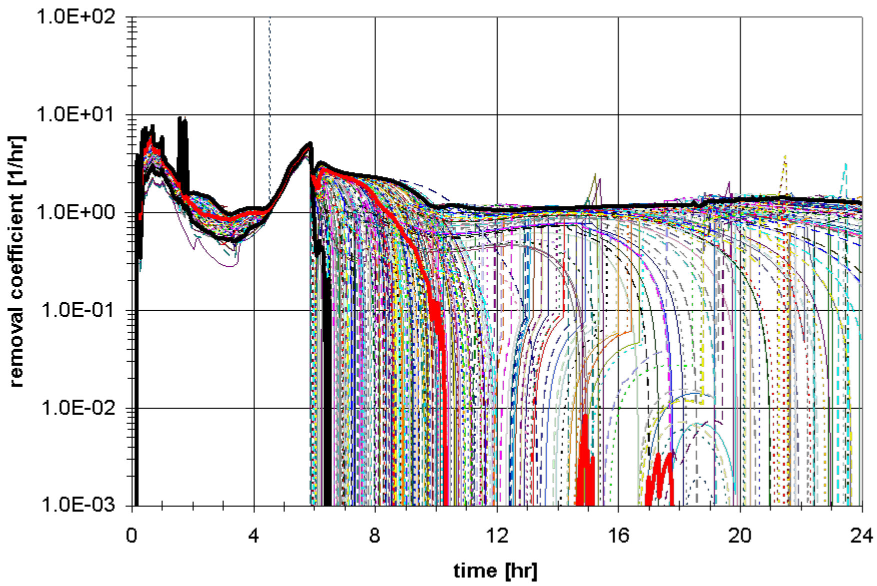


Figure 3-24 Mk-I RLB, Removal Coefficients with Geometric Variability, MSL-A, In-Board – no sprays, condenser

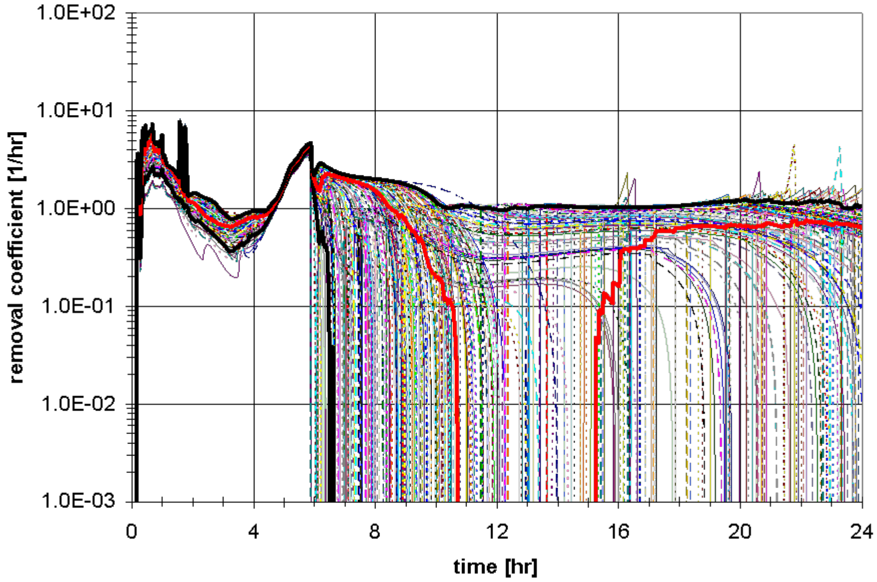


Figure 3-25 Mk-I RLB, Removal Coefficients with Geometric Variability, MSL-B, In-Board – no sprays, condenser

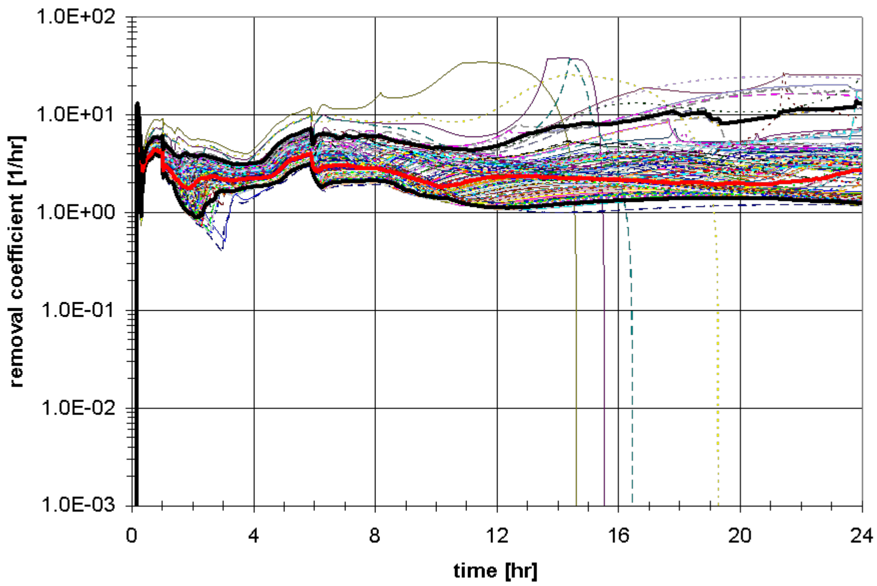


Figure 3-26 Mk-I RLB, Removal Coefficients with Geometric Variability, MSL-A, between MSIVs – no sprays, condenser

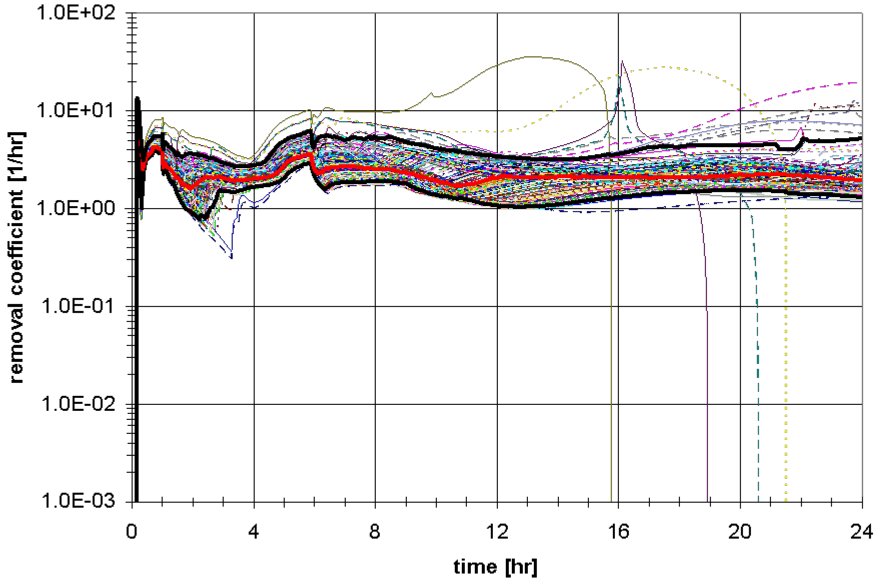


Figure 3-27 Mk-I RLB, Removal Coefficients with Geometric Variability, MSL-B, between MSIVs – no sprays, condenser

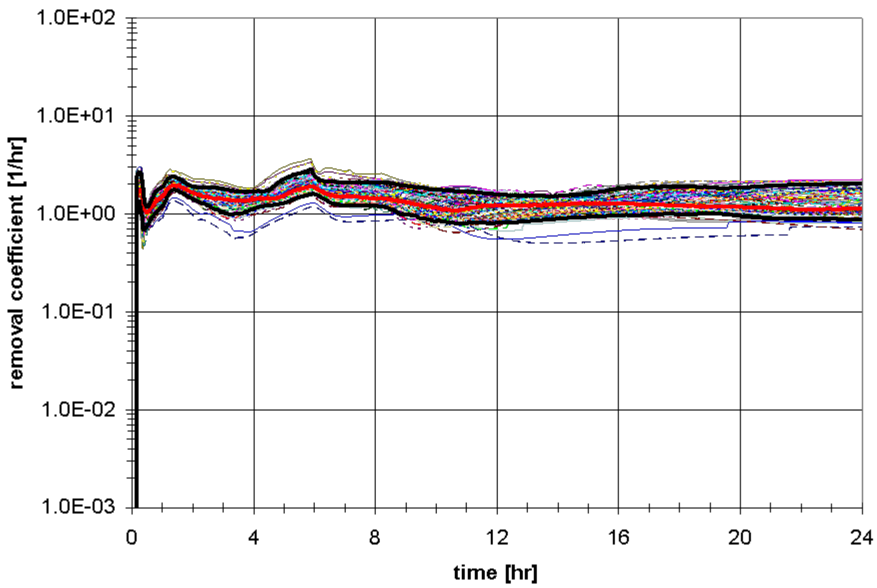


Figure 3-28 Mk-I RLB, Removal Coefficients with Geometric Variability, MSL-A, out-board – no sprays, condenser

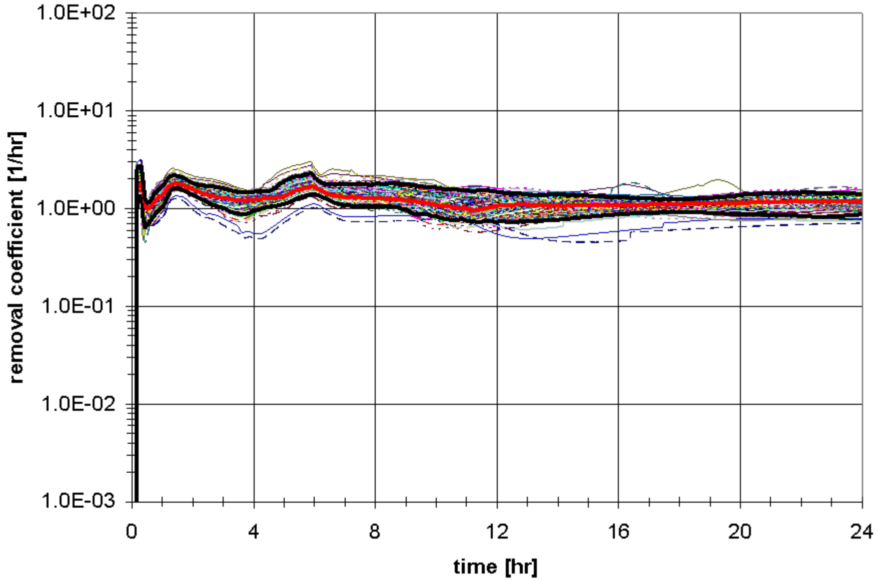


Figure 3-29 Mk-I RLB, Removal Coefficients with Geometric Variability, MSL-B, out-board – no sprays, condenser

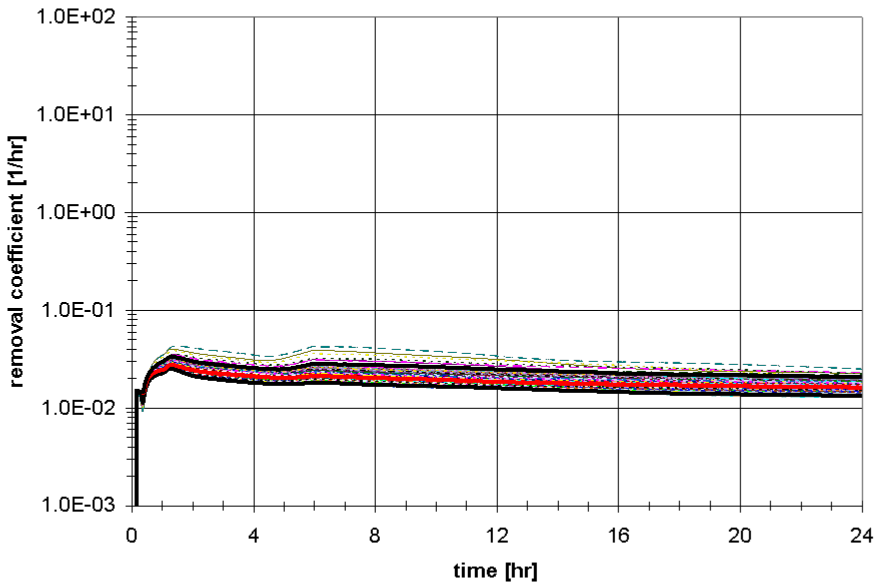


Figure 3-30 Mk-I RLB, Removal Coefficients with Geometric Variability, condenser – no sprays, condenser

Table 3-10: Geometric Variability Analysis MSL and Condenser Removal Coefficients (5th Percentile)

MSL section	0 - 2 (hr)	2 - 12 (hr)	12+ (hr)
MSL-A in-board	1.6	0.55	0.0
MSL-A between MSIVs	2.6	1.8	1.3
MSL-A out-board	1.2	1.1	0.93
MSL-B in-board	1.5	0.50	0.0
MSL-B between MSIVs	2.6	1.6	1.4
MSL-B out-board	1.1	1.0	0.85
condenser	0.020	0.018	0.014

note. removal coefficients are given in 1/hr

Table 3-11: Geometric Variability Analysis MSL and Condenser Removal Coefficients (50th Percentile)

MSL section	0 - 2 (hr)	2 - 12 (hr)	12+ (hr)
MSL-A in-board	2.9	1.2	0.00028
MSL-A between MSIVs	3.2	2.5	2.2
MSL-A out-board	1.5	1.4	1.2
MSL-B in-board	2.7	1.2	0.43
MSL-B between MSIVs	3.2	2.3	2.1
MSL-B out-board	1.4	1.3	1.1
condenser	0.022	0.020	0.017

note. removal coefficients are given in 1/hr

Table 3-12: Geometric Variability Analysis MSL and Condenser Removal Coefficients (95th Percentile)

MSL section	0 - 2 (hr)	2 - 12 (hr)	12+ (hr)
MSL-A in-board	3.8	1.8	1.2
MSL-A between MSIVs	4.7	4.9	9.2
MSL-A out-board	1.9	2.0	1.8
MSL-B in-board	3.5	1.7	1.1
MSL-B between MSIVs	4.5	4.3	4.1
MSL-B out-board	1.8	1.7	1.4
condenser	0.026	0.027	0.022

note. removal coefficients are given in hr^{-1}

3.4 Results from Main Steam Line Uncertainty Analyses

Three uncertainty cases were run using the MELCOR MSL-only model; these included:

- RLB, no sprays, condenser, aerosol physics parameter uncertainty
- RLB, no sprays, condenser, geometric variability
- RLB, no sprays, condenser, MSIV flow uncertainty.

The removal coefficients plots from the cases are given below (see Figure 3-10 through Figure 3-30.

In general, the further away the MSL piping is from the steam dome the lower its removal coefficient. This is due the faster deposition rate of large aerosol particles, which causes them to deposit in the MSL piping closer to the steam dome. The asymptotic drops in the in-board MSL removal coefficients occur at times where the pipe wall temperature is high enough such that deposited fission products are vaporized, causing the deposition rate, and hence the removal coefficient, to become negative. The initial occurrences are driven in part by the elevated gas temperatures associated with the heat-up of the lower plenum prior to lower head failure. The later occurrences are caused by the decay heat from the deposited fission product aerosols.

The 5th, 50th, and 95th values for removal coefficients based on the MSL-only model uncertainty analysis results are provided in Tables 4-13, 4-14, and 4-15. The tabulated values were derived by calculating the integrated average removal coefficient of the percentile of interest (e.g., 5th, 50th, 95th) over the time period of interest (e.g., 0-2 hrs, 2-12 hrs, 12-24 hrs) for each MSL segment (e.g., in-board, between MSIVs, out-board) for both MSLs (e.g., MSL-A, MSL-B) and the condenser from each of the three uncertainty analyses. A simple average was then taken of the results of the three uncertainty analyses for each time period. The MSL-A and MSL-B results were averaged to calculate a single in-board, between MSIVs, and out-board removal coefficient result.

Removal coefficients were calculated for the in-board segments of the MSLs, however it is recommended that no credit (i.e., a removal coefficient of 0.01/hr) be taken for aerosol deposition in this portion of the MSLs. The basis for this recommendation is that at times in the simulation the temperature of portions of the in-board MSL piping are predicted to be high enough to vaporize fission products that had been previously deposited and because of the potential for thermophoretic repulsion: see page 4-15 of reference [3] . This secondary source cannot be easily incorporated into a RADTRAD model, and if the initial deposition (via a non-zero removal coefficient) is credited, the omission of this secondary source would result in an under-prediction of fission products released downstream, and ultimately to the environment.

Decay heat from fission products deposited in the in-board piping can cause natural convection-driven bi-directional flow between the steam dome and the in-board MSLs. While the well-mixed nature of the MSL control volumes does in some fashion capture the enhanced mixing that such flow would cause, it does not address the potential for enhanced bulk transport of fission products from the steam dome into the in-board MSLs. Note that this issue has been previously cited as a basis for not taking credit for aerosol deposition in the in-board MSLs: see page 4-15 of reference [3].

Table 3-13: Recommended MSL and Condenser Removal Coefficients (5th Percentile)

MSL section	0 - 2 (hr)	2 - 12 (hr)	12+ (hr)
in-board	2.2*	0.0*	0.0*
between MSIVs	2.5	1.8	1.0
out-board	1.1	1.0	0.7
condenser	0.018	0.015	0.012

note. removal coefficients are given in hr^{-1}

* calculated values are shown, but a value of 0.0 hr^{-1} is recommended

Table 3-14: Recommended MSL and Condenser Removal Coefficients (50th Percentile)

MSL section	0 - 2 (hr)	2 - 12 (hr)	12+ (hr)
in-board	3.2*	1.3*	0.4*
between MSIVs	2.9	2.4	2.0
out-board	1.3	1.3	1.0
condenser	0.020	0.018	0.015

note. removal coefficients are given in hr^{-1}

* calculated values are shown, but a value of 0.0 hr^{-1} is recommended

Table 3-15: Recommended MSL and Condenser Removal Coefficients (95th Percentile)

MSL section	0 - 2 (hr)	2 - 12 (hr)	12+ (hr)
in-board	4.9*	2.3*	1.1*
between MSIVs	3.8	3.7	4.4
out-board	1.6	1.7	1.6
condenser	0.023	0.023	0.021

note. removal coefficients are given in hr^{-1}

* calculated values are shown, but a value of 0.0 hr^{-1} is recommended

4 RADTRAD Dose Calculations for Full Reactor Models[§]

In this section, the MELCOR source term is converted into estimated doses at the Exclusion Area Boundary (EAB), Low Population Zone (LPZ), and the Control Room (CR). The regulatory limits (set forth in 10 CFR 50.67) to which the RADTRAD dose results are compared are

- 25 rem total effective dose equivalent (TEDE) for the worst two-hour dose at the EAB,
- 25 rem TEDE for the 30 day integrated dose at the LPZ,
- 5 rem TEDE for the 30 day integrated dose in the CR.

The overall purpose of these assessments and comparisons is to evaluate the dose implications of using a source term from a best estimate release analysis. This can be compared to doses estimated using regulatory guidelines methods in order to provide perspective relative to likely degree of conservatism of the simplified regulatory procedures.

4.1 Description of RADTRAD Model for MELCOR Full Plant Decks

A RADTRAD model has been developed to calculate doses from the MSL fission product releases calculated by MELCOR full reactor models. The RADTRAD model consists of a small volume into which the MELCOR-calculated fission product release is input as a source, the environment, and the control room. The source volume is only included because RADTRAD does not allow a source to be placed directly into the environment. The source volume is connected to the environment with a flow path that has a very large volumetric flow rate. This effectively moves the source instantaneously from the source volume to the environment. The model nodalization is shown in Figure 4-1.

[§] MELCOR input data used to develop these models were based on the configuration, geometry and materials of single, representative plants (Grand Gulf and Peach Bottom).

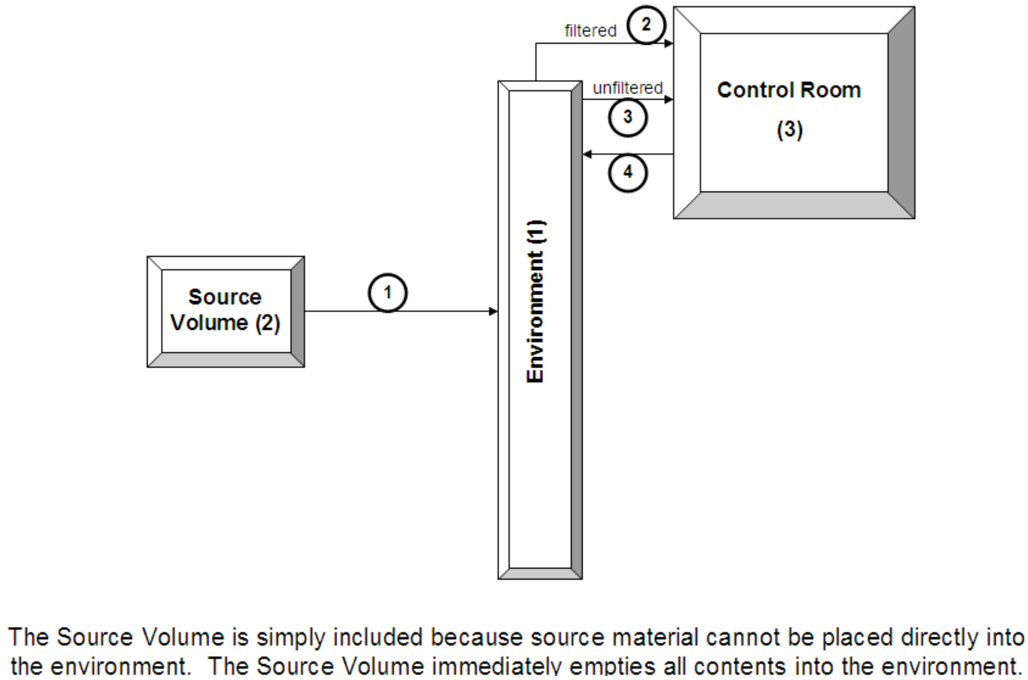


Figure 4-1 Mk-I and Mk-III Full Reactor Source Term Model

The purpose of these full reactor model calculations is to compare early dose results directly from MELCOR to those derived using the proposed methodology. The two sets of results are not expected to match exactly, but they should be comparable and follow similar trends.

4.2 MELCOR to RADTRAD FP Group Conversion

MELCOR and RADTRAD use significantly different accounting schemes to track mass and conservation of radionuclides, with MELCOR using a chemical-family based accounting system and RADTRAD using a radioisotope accounting basis. Importantly, MELCOR accounts for all mass of released materials, both radioactive and stable isotopes as well as the inert mass associated with oxide or hydroxide forms, whereas RADTRAD tracks only selected dose-important isotopes. MELCOR treats all mass in order to account for important aerosol mechanics effects, principally particle agglomeration. In order to map MELCOR predicted fission product source terms into RADTRAD sources, it is necessary to consider the relationship between MELCOR mass inventories and RADTRAD inventories. The following sections describe the mapping methodology used in this study.

To provide context for the description of the fission product release post-processing, a brief discussion MELCOR's and RADTRAD's treatment of fission products is provided.

MELCOR models fission products as a set of radionuclide (RN) classes in which each class is represented by a single chemical compound. For example, while RN class 1 is comprised of a variety of noble gases, it is represented in MELCOR as Xe. Also, certain fission products elements are contained in multiple RN classes (e.g., I is in both the I_2 and CsI RN classes). In contrast, RADTRAD models individual isotopes, but only accounts for 60 fission products that are the most important contributors to dose. Table.4-1 shows the fission product chemical groups used by MELCOR listing the representative element for each group and the fission product elements that correspond to that chemical group. Table.4-2 shows the radionuclides typically considered in RADTRAD, the used inventories of these isotopes for the Mk-I and Mk-III designs, the RADTRAD chemical group to which they belong, and the MELCOR chemical group to which they correspond. The inventories in this table, presented in grams, were converted from the RADTRAD Nuclide Inventory Files (.nif) inventories, in Ci/MW, by using the reactor powers, the decay constants obtained from the half life of each isotope, and the atomic weight of each isotope.

Table.4-1. MELCOR Radionuclide (RN) Classes

RN Class	Name	Representative	Member Elements
1	Noble Gas	Xe	He, Ne, Ar, Kr, Xe, Rn, H, N
2	Alkali Metals	Cs	Li, Na, K, Rb, Cs, Fr, Cu
3	Alkaline Earths	Ba	Be, Mg, Ca, Sr, Ba, Ra, Es, Fm
4	Halogens	I	F, Cl, Br, I, At
5	Chalcogens	Te	O, S, Se, Te, Po
6	Platinoids	Ru	Ru, Rh, Pd, Re, Os, Ir, Pt, Au, Ni
7	Early Transition Elements	Mo	V, Cr, Fe, Co, Mn, Nb, Mo, Tc, Ta, W
8	Tetravalent	Ce	Ti, Zr, Hf, Ce, Th, Pa, Np, Pu, C
9	Trivalents	La	Al, Sc, Y, La, Ac, Pr, Nd, Pm, Sm, Eu, Gd, Tb, Dy, Ho, Er, Tm, Yb, Lu, Am, Cm, Bk, Cf
10	Uranium	U	U
11	More Volatile Main Group	Cd	Cd, Hg, Zn, As, Sb, Pb, Tl, Bi
12	Less Volatile Main Group	Sn	Ga, Ge, In, Sn, Ag
16	Cesium iodide	CsI- – -	CsI
17	Cesium Molybdate	Cs_2MoO_4	Cs_2MoO_4

note: RN classes 13, 14, 18 consist of non-radionuclide aerosol materials

Table.4-2 RADTRAD Nuclide Inventory File Isotopes

Isotope	RADTRAD Chemical Group	MELCOR RN Class	Mk-I Core Inventory (g)	Mk-III Core Inventory (g)
Co-58	7	7	1.701E+01	1.702E+01
Co-60	7	7	5.708E+02	5.708E+02
Kr-85	1	1	3.547E+03	3.487E+03
Kr-85m	1	1	3.562E+00	3.904E+00
Kr-87	1	1	2.040E+00	2.073E+00
Kr-88	1	1	6.479E+00	6.302E+00
Rb-86	3	2	2.826E+00	3.200E+00
Sr-89	5	3	3.395E+03	3.397E+03
Sr-90	5	3	8.223E+04	8.150E+04
Sr-91	5	3	3.694E+01	3.498E+01
Sr-92	5	3	1.128E+01	1.059E+01
Y-90	9	9	2.121E+01	2.106E+01
Y-91	9	9	4.965E+03	5.126E+03
Y-92	9	9	1.477E+01	1.386E+01
Y-93	9	9	4.777E+01	4.539E+01
Zr-95	9	8	7.395E+03	7.677E+03
Zr-97	9	8	8.589E+01	8.466E+01
Nb-95	9	7	4.075E+03	4.227E+03
Mo-99	7	7, 17	3.732E+02	3.778E+02
Tc-99m	7	7	2.982E+01	3.018E+01
Ru-103	7	6	4.599E+03	4.947E+03
Ru-105	7	6	1.525E+01	1.731E+01
Ru-106	7	6	1.823E+04	2.076E+04
Rh-105	7	6	1.149E+02	1.290E+02
Sb-127	4	11	3.833E+01	2.951E+02
Sb-129	4	11	5.416E+00	5.837E+00
Te-127	4	5	3.839E+00	4.303E+00
Te-127m	4	5	1.442E+02	1.608E+02
Te-129	4	5	1.432E+00	1.543E+00
Te-129m	4	5	1.482E+02	2.328E+02
Te-131m	4	5	1.711E+01	1.805E+01
Te-132	4	5	4.444E+02	4.548E+02
I-131	2	4, 16	7.645E+02	7.853E+02
I-132	2	4, 16	1.327E+01	1.361E+01
I-133	2	4, 16	1.730E+02	1.716E+02
I-134	2	4, 16	8.140E+00	8.014E+00
I-135	2	4, 16	5.221E+01	5.179E+01
Xe-133	1	1	1.035E+03	1.023E+03
Xe-135	1	1	3.075E+01	2.968E+01
Cs-134	3	2, 16, 17	1.976E+04	2.223E+04
Cs-136	3	2, 16, 17	9.764E+01	1.156E+02
Cs-137	3	2, 16, 17	1.841E+05	1.704E+05

Isotope	RADTRAD Chemical Group	MELCOR RN Class	Mk-I Core Inventory (g)	Mk-III Core Inventory (g)
Ba-139	6	3	1.094E+01	1.074E+01
Ba-140	6	3	2.358E+03	2.375E+03
La-140	9	9	3.186E+02	3.219E+02
La-141	9	9	2.894E+01	2.894E+01
La-142	9	9	1.116E+01	1.101E+01
Ce-141	8	8	5.558E+03	5.655E+03
Ce-143	8	8	2.354E+02	2.318E+02
Ce-144	8	8	3.980E+04	3.963E+04
Pr-143	9	9	2.248E+03	2.231E+03
Nd-147	9	9	8.012E+02	8.326E+02
Np-239	8	8	8.200E+03	9.982E+03
Pu-238	8	8	3.706E+04	3.921E+04
Pu-239	8	8	6.809E+05	7.774E+05
Pu-240	8	8	1.994E+05	3.205E+05
Pu-241	8	8	2.111E+05	1.895E+05
Am-241	9	9	9.798E+03	7.332E+03
Cm-242	9	9	2.541E+03	2.309E+03
Cm-244	9	9	1.137E+04	2.002E+04

MELCOR does not account for the decay and in-growth of radionuclides[**], while RADTRAD accounts for decay and in-growth of radionuclides once they are released into a volume in the RADTRAD model.

The release rate of a RADTRAD source term is defined by the release timing fraction (.rtf) file. RADTRAD only allows the release rate to be defined as a constant release for three separate time periods. An initial delay in the source term release is also allowed. This delay is used to account for the period between the initiation of the accident (t = 0 hr) and the start of the fission product release to the environment. Conversely, the MELCOR-calculated fission product releases are reported from the model at intervals based on the plot frequency specified in the model (10 s to 180 s).

Also, MELCOR core inventory is a BWR middle-of-cycle (MOC) inventory, which differs both in terms of absolute mass and isotopic composition from the BWR end-of-cycle (EOC) core inventory assumed in RADTRAD.

To account for the different fission product modeling treatments the following post processing steps are applied to a MELCOR-calculated fission product release in order to put it into a RADTRAD-compatible form. This conversion consists of time averaging the MELCOR chemical group releases, distributing the MELCOR chemical group masses among isotopes according to the relative masses of each isotope, scaling to correct for the differences in initial

[**] The MACCS code, developed to calculate consequences from MELCOR-calculated releases, would account for radionuclide decay and in-growth from the time of accident initiation.

core inventories, and decaying the isotopes to account for the reduction of activity at the time of release. Examples from the Mk-I RLB case are used to illustrate the steps.

- Evaluate the MELCOR fission release rate signatures to determine the best fit of the signatures into three release rate periods, and if needed, a delay time before the start of the release.

  For example, based on the release rates shown in Figure 5-2 periods of 0.5 to 2.4 hr, 2.4 to 4.4 hr, and 4.4 to 5.88 hr, with a delay of 0.0 to 0.5 hr are estimated and incorporated in the RADTRAD RTF file.

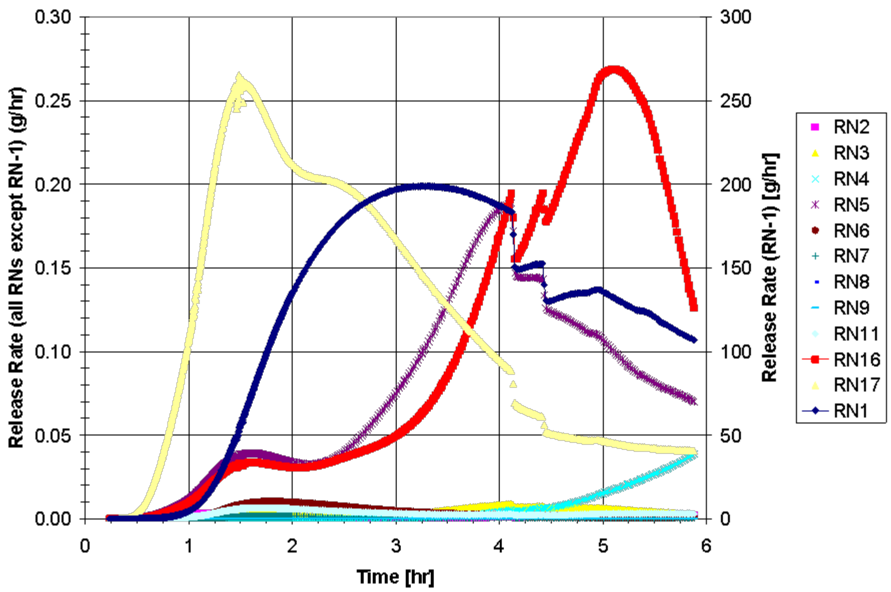


Figure 4-2 Mk-I RLB RN Release Rates

- Calculate the total release of each MELCOR RN class over each time period.

  For example, based on the integrated release output from MELCOR, 124.3 g of RN class 1 were released between 0.5 and 2.4 hr.

- Determine the mass of Cs, I, and Mo contained in MELCOR RN classes 16 (CsI) and 17 (Cs_2MoO_4) and add those masses, respectively, to the RN class 2 (Cs), 4 (I_2), and 7 (Mo) masses.

  For example, between 0.5 and 2.4 hr there are 9.6E-06 g of RN class 4 (I_2) released, and 0.0420155 g of RN class 16 (CsI) released. 49% of the mass of RN class 16 is I (i.e., 0.02061 g). This I mass is added to RN class 4, yielding a total release mass of I_2 over the period of 0.0206069 g.

- Sum the RADTRAD initial core inventory isotope masses by groups that correspond to the MELCOR RN classes. This is done for RN classes 1 through 9, and 11. The other RN classes either are non-radioactive (e.g., RN class 14) or there are no corresponding isotopes in the RADTRAD inventory (e.g., RADTRAD has no U isotopes in its inventory, hence RN class 10 is omitted).

  For example, for MELCOR RN class 1, the sum of the Kr-85, Kr-85m, Kr-87, Kr-88, Xe-133, and Xe-135 in the RADTRAD BWR initial core inventory (for a power level of 3528 MWth) is 4632.9 g.

- Calculate the ratio of the mass of isotopes in each RN class in the RADTRAD initial core inventory to the mass of isotopes in the MELCOR initial core inventory (see Table 4-3).

  For example, the mass of isotopes in the RADTRAD BWR initial core inventory (for a power level of 3528 MWth) that correspond to the MELCOR RN class 1 is 4632.9 g, while there are 531,540.9 g in the MELCOR RN class 1 initial core inventory. Therefore the ratio of the RADTRAD to MELCOR initial core inventory masses for RN class 1 is 4632.9 / 531,540.9 = 0.00870.

- Scale the MELCOR fission product releases, to account for the differences between the RADTRAD and MELCOR initial core inventories, by multiplying the mass of each RN class released in each time period by its respective ratio of RADTRAD to MELCOR initial core inventory masses.

  For example, the ratio of the RADTRAD to MELCOR initial core inventory masses for RN class 1 is 0.00870 and the RN class 1 mass released between 0.5 and 2.4 hr is 124.3 g. Therefore the scaled release mass is 0.00870 x 124.3 g = 1.081 g.

- Calculate the ratio of each RADTRAD isotope to the total mass of all of the isotopes in its equivalent MELCOR RN class. This ratio is used to convert the mass from MELCOR groups to the Isotope masses.

  For example, there are 3548.5 g of Kr-85 in the RADTRAD Mk-I initial core inventory (for a power level of 3528 MWth). Therefore the ratio of Kr-85 to the total mass of the isotopes that correspond to RN class 1 is 3548.6 / 4632.9 = 0.767. The same result can be obtained using the ratio of the corresponding isotope masses provided in Table.4-2 (=$M_{Kr\text{-}85}/(M_{Kr\text{-}85}+M_{Kr\text{-}85m}+M_{Kr\text{-}87}+M_{Kr\text{-}88}+M_{Xe\text{-}133}+M_{Xe\text{-}135})$*) or the ratio of the products of the molecular weights, half lives, and activities (in Ci/Mw) provided in previous Nuclide Inventory Files .*

- Calculate the masses of the individual isotopes of each scaled MELCOR RN class released in each time period. Decay each isotope by half the duration of the time period plus the time previous to the time period (e.g., for a period between 2 and 5 hr a decay time of 3.5 hr would be used).

> *For example, the ratio of Kr-85 to the total mass in RN class 1 is 0.767 and between 0.5 and 2.4 h the scaled release mass is 1.081 g. Therefore the pre-decay release mass of Kr-85 is 0.829 g. Simple decay is accounted for over a time of 0.5 hr + 0.5 x (2.4 – 0.5) hr = 1.45 hr, which yields a scaled, decayed release mass of Kr-85 of 0.829 g.*

The post-processing steps can be described by the following equation

$$m_i^k = m_{cls(i)}^k \times Rc_{cls(i)} \times Ri_i \times \exp\left[-\lambda_i t^k\right] \qquad (4.1)$$

where

m_i^k	- mass of the ith isotope released over the kth release period
$m_{cls(i)}^k$	- mass of the MELCOR RN class containing the ith isotope released over the kth release period, accounting for the mass of Cs, I, and Mo being moved from RN classes 16 and 17 mass to RN classes 2, 4, and 5.
$Rc_{cls(i)}$	- ratio of the RADTRAD to MELCOR initial core inventory masses for RN class containing the ith isotope
Ri_i	- ratio of the ith isotope to the total mass of all isotopes in the RADTRAD initial core inventory that are in the corresponding MELCOR RN class
λ_i	- radioactive decay constant of the ith isotope
t^k	- decay time for the kth release period

Table 4-3 Ratio of RADTRAD to MELCOR Initial Core Inventory

MELCOR RN Class	Mk-I Ratio [-]	Mk-III Ratio [-]
1	0.009	0.005
2	0.727	0.446
3	0.374	0.238
4	0.051	0.031
5	0.015	0.011
6	0.067	0.042
7	0.013	0.009
8	0.765	0.617
9	0.018	0.031
11	0.007	0.021

Note: Accounts for the mass of Cs, I, and Mo being moved from RN classes 16 and 17 to RN classes 2, 4, and 5.

4.3 RADTRAD Results

Selected results of the full reactor model source term RADTRAD predictions are presented here, along with comparisons against results from sample industry RADTRAD models obtained from industry reports submitted to the NRC [12,13.

4.3.1 Mk-I Results

When using the MELCOR-predicted releases to the environment directly as the source to RADTRAD, the resulting predicted doses are limited to the amount of time that the MELCOR simulation was run. This ranges from 4.9 hours to 7.7 hours. Thus, a comparison cannot be made between the LPZ and CR limits and the results from the MELCOR full models. These comparisons will be available with the steam line model discussed later in the report however. Table 4-4 compares the worst 2 hour doses at the EAB for the Mark 1 cases examined.

Table 4-4 Worst 2 Hour TEDE Doses at EAB for Mk 1 Using MELCOR Release to Environment

	EAB worst 2 hr (rem)
10 CFR 50.67 Limit	25
Representative Industry Sample analysis	3.48
RLB	10.66
RLB (sprays)	4.06
RLB (cond)	0.11
RLB (cond and sprays)	0.04
MSLB	2.90
MSLB (sprays)	2.09

The following two figures are the integrated doses at the LPZ and the control room out to 6 hours. Some of the cases only run out close to 5 hours. The dose for the LPZ will flatten out when the simulation has reached the end of the MELCOR source term.

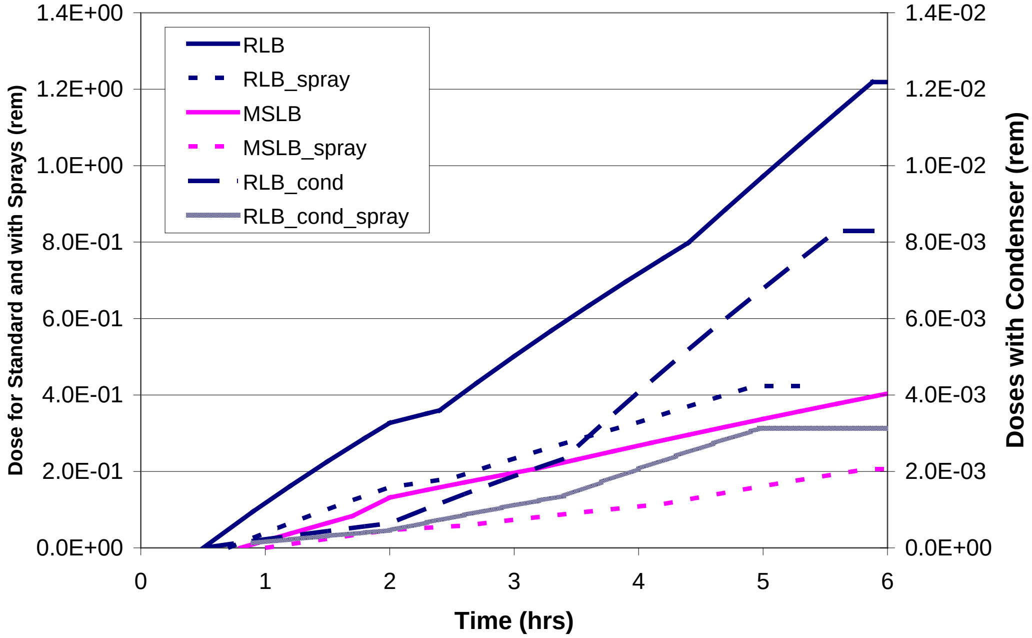


Figure 4-3 LPZ Integrated TEDE for Mk 1 MELCOR Full Model Cases, Notice that the two simulations with condensers use the scale on the right which is lower by a factor of 100.

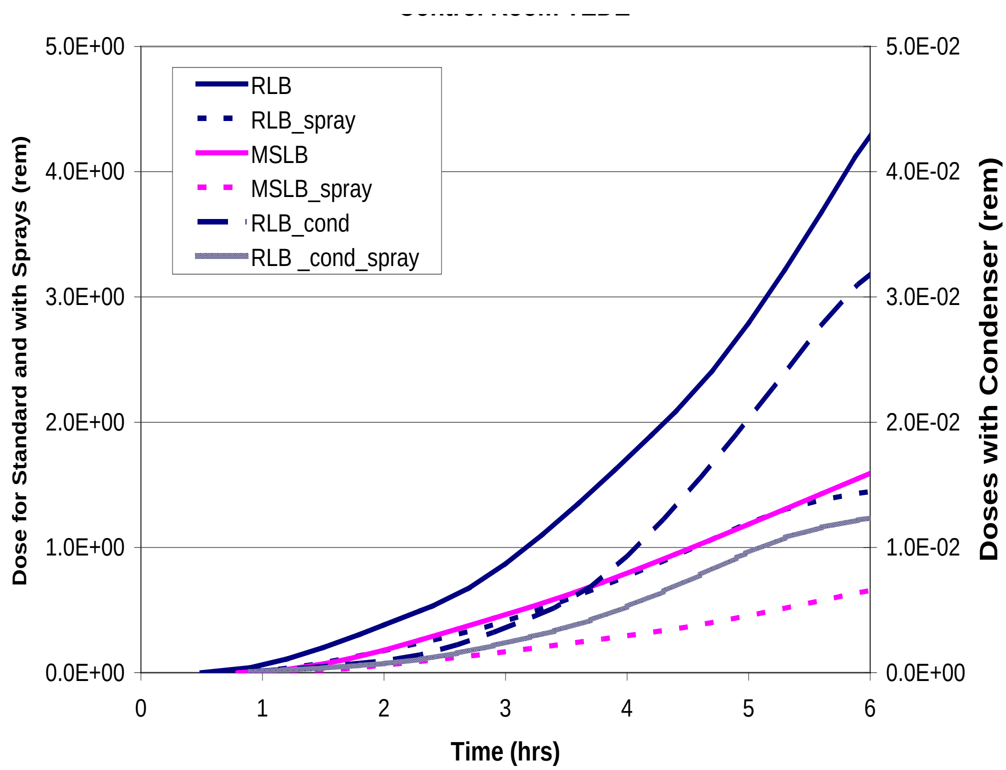


Figure 4-4 Control Room Integrated TEDE for Mk 1 MELCOR Full Model Cases. Notice that the two simulations with condensers use the scale on the right which is lower by a factor of 100

Some insight on the LPZ and CR doses can be gained by comparing them to representative industry calculations without sprays at 5 hours. This comparison is summarized in Table 4-5. The values were taken within 0.1 hours of 5 hours, varying depending on the timesteps within the problem. The RLB analyses produce doses that are larger somewhat than the representative industry value – other cases are closer to the industry value.

Table 4-5 A Comparison of the MELCOR Full Model TEDE Doses to Comparative Industry Calculation without Sprays for LPZ and CR near 5 Hours

RLB(sprays) and RLB(cond) values taken at 4.9 hours, MSLB(sprays) taken at 5.1 hours, the remaining cases were at 5 hours.

	LPZ near 5 hours (rem)	CR near 5 hours (rem)
Representative Industry Sample analysis	0.27	0.99
RLB	0.97	2.79
RLB (sprays)	0.42	1.17
RLB (cond)	0.01	0.02
RLB (cond and sprays)	0.00	0.01
MSLB	0.34	1.19
MSLB (sprays)	0.17	0.48

4.3.2 Mk-III Results

The Mk-III parameters for control room size, flow rates between control room and environment, filter efficiencies, and dose conversion factors were taken from Reference 13. The RADTRAD model labeled "MSIV LEAKAGE RADTRAD RUN" only examines leakage out to 0.3 hours. In order to obtain values out to at least 72 hours, the parameters from the "ESF LIQUID LEAKAGE RAPTOR RUN" and ESF LIQUID LEAKAGE RADTRAD RUN" were used. These parameters seemed to match the initial values in the MSIV leakage RADTRAD run and closely resembled the parameters in the Mk-I studies.

The geometry of the sample Mk-III differs from that of the Mk-I in that there is no piping modeled past the outboard MSIV in MELCOR. This would suggest that the dose will be higher than that of the Mark I, which includes more than 90 meters of piping past the outboard MSIV.

The Mk-III worst 2 hour integrated EAB TEDE results can be found in Table 4-6. The values are approximately 14 rem, still within the 10 CFR 50.67 limit of 25 rem. The LPZ and control room integrated TEDE doses can be seen in Figure 4-5. The control room TEDE has already surpassed the 5 rem TEDE limit after only 4 hours, but the LPZ limit is well below the 25 rem TEDE limit and appears to be flattening out. The result from the Mk-III sample industry model is not considered comparable because it includes a leakage control system. Therefore it is not provided in the table.

Table 4-6: Worst 2 Hour TEDE Doses at EAB for Mk-III Using MELCOR Release to Environment
Notice that the doses are both approaching the 10 CFR limit.

	EAB worst 2 hr (rem TEDE)
10 CFR 50.67 Limit	25
RLB	14.36
MSLB	14.73

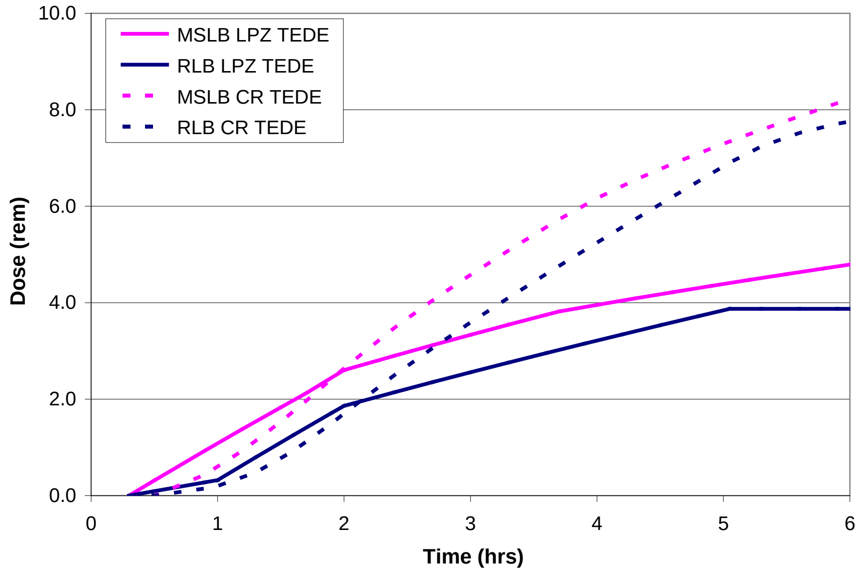


Figure 4-5 Control Room and LPZ Integrated TEDEs for Mark 3 MELCOR Full Model Cases *Notice that the control room doses have exceeded the 5 rem TEDE limit at about 3 hours.*

5 RADTRAD Dose Calculations for Main Steam Line Models

5.1 Description of RADTRAD Main Steam Line Models

The RADTRAD MSL models are examples for how the recommended MSL and condenser removal coefficients and the steam dome-to-drywell concentration ratios would be implemented into RADTRAD. These models are structurally similar to the RADTRAD models currently used by the NRC and licensees to evaluate MSIV leakage consequences and use values for control room size, containment leakage, flow rates between control room and environment, and control room filters from the licensee models. Representative RADTRAD models for Mk-I and Mk-III dose analyses were taken from industry reports submitted to the NRC [12,13]. The RADTRAD input decks used for the MK-I case are provided in Appendix B.

One difference between the Mk-I MSL-only model and the industry sample model is that the MSL-only model uses three steam line volumes rather than two. This is to accurately represent the hold-up in each of these volumes. There are three volumes for both the RLB and the MSLB cases because it was determined that the most conservative approach would be to assume, for the MSLB case, that the break occurs in a non-leaky line, making the geometry the same as that for the RLB case. Therefore, the inboard section is still needed to represent hold-up in the MSLB case.

The reference industry sample model also uses filter efficiencies rather than lambdas. It was discovered during this study that filter efficiencies are not completely equivalent to removal coefficients for all situations, and that their application to volumes that have non-steady state concentrations can result in non-conservative releases. Hence, removal coefficients are used in the MSL models in lieu of filter efficiencies. There were also sprays within the reference model which will not be used in these models.

RADTRAD version 3.03 is limited to modeling no more than ten volumes. As can be seen in Figure 5-1 to Figure 5-4, the RADTRAD MSL models are comprised of more than ten volumes. This was accomplished by breaking the RADTRAD MSL model into two separate submodels, where each submodel contains only one MSL. The submodels are run individually, and the principal of linear superposition was used to add the dose results from the submodels to get the total dose at the EAB, LPZ, and control room.

Another limitation of RADTRAD is that the source partitioning can only be specified once in the input file. So to work around this there are three source volumes modeled. This is to create multiple time-intervals at which to apply the steam dome-to-drywell ratio. Each of the source volumes will only be connected to the model for a specified length of time, with only one source volume connected at any given time, again using a superposition approach.

The nodalization of the models is shown in Figure 5-1-Figure 5-4. The geometry is similar for the Mk-I and Mk-III with a few exceptions. The Mk-III has three lines with MSIV leakage and no outboard line section. This may result in some differences between the Mk-I and Mk-III models, as the outboard line represents nearly 70% of the main steam line length in the Mk-I geometry. The neglected section of pipe in the Mk-III however is shorter than in the Mk-I, so the differences may not be large.

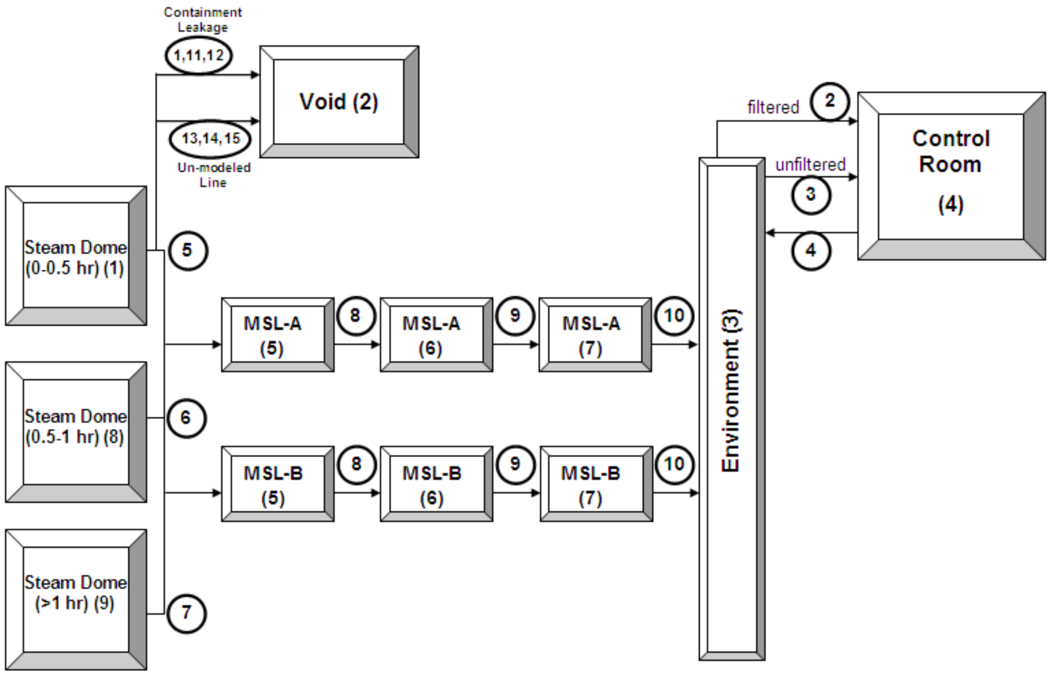


Figure 5-1 Mk-I (RLB and MSLB), No Condenser

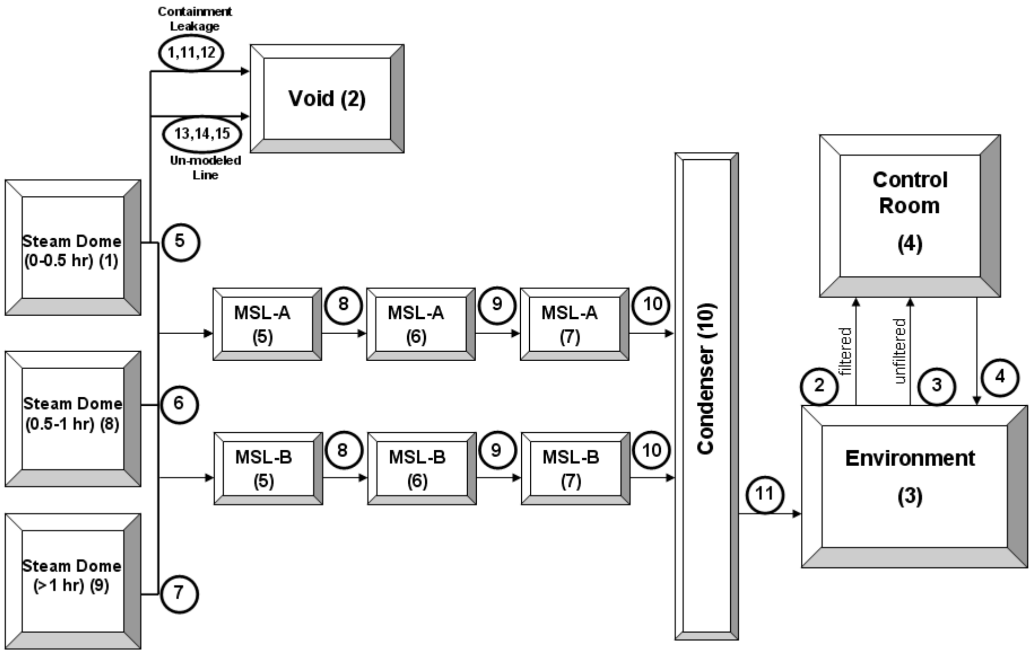


Figure 5-2 Mk-I (RLB and MSLB), With Condenser

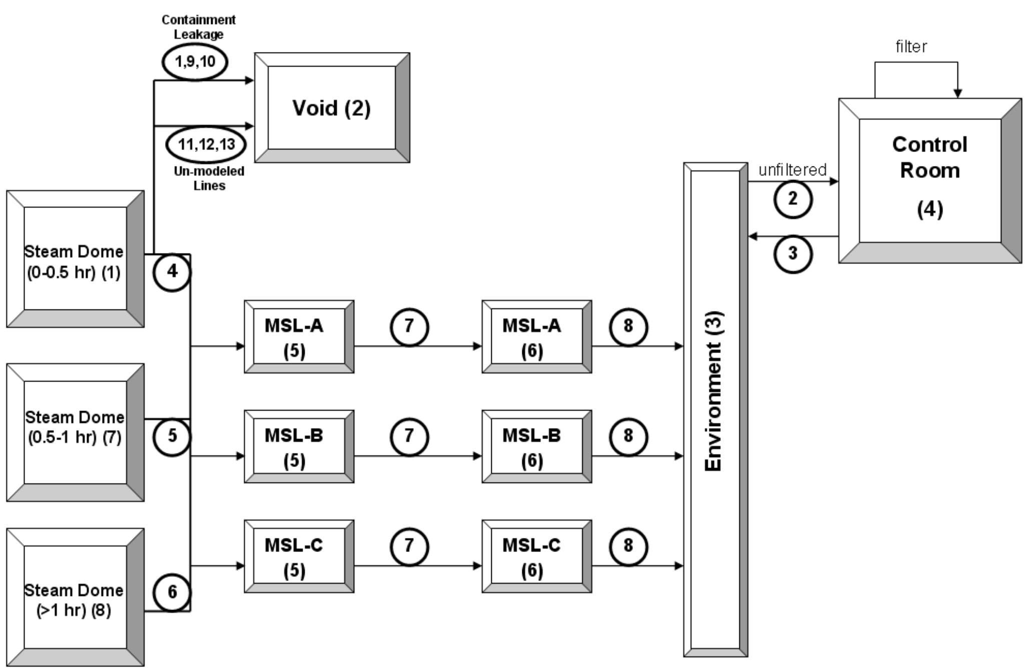


Figure 5-3 Mk-III (RLB and MSLB), No Condenser

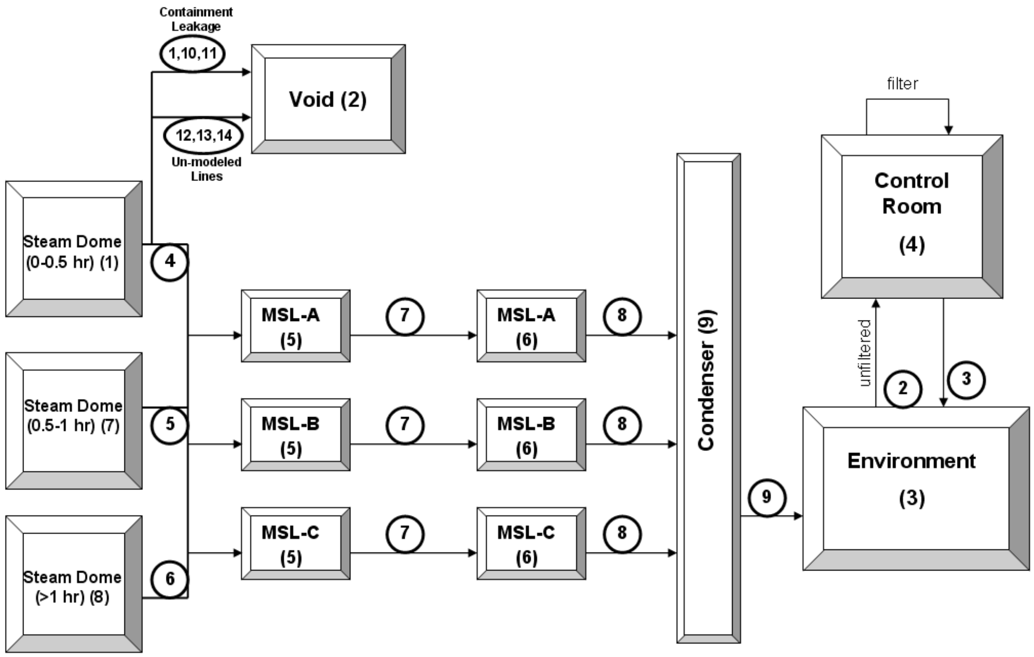


Figure 5-4 Mk-III (RLB and MSLB), With Condenser

5.2 Steam Dome-to-Drywell Ratio

The initial approach for using the steam dome-to-drywell concentration ratio calculated by the MELCOR full reactor models was to use the highest ratios of all of the RN classes. Evaluation of this initial approach found that this significantly overestimated the source term. This was due to the highest ratio of all of the RN classes being very conservative in comparison to the ratios for the RN classes of the isotopes that are most important to dose. Also the drywell concentration in RADTRAD is very different from that in the MELCOR full reactor model.

To address the first issue, the bounding steam dome-to-drywell concentration ratio of the three RN classes that contribute the majority of the dose RN2 (Cs), RN3 (Sr), and RN4 (I) will be used to scale the source term input into RADTRAD.

The drywell concentrations for these three RN classes are different between the MELCOR best estimate calculation and the RADTRAD-AST based calculation because of to the differences in source release rates. Figure 5-5 compares the masses of Cs, Sr, and I in the drywell for both RADTRAD and MELCOR in the Mk-I RLB scenario. Since both analyses model the same drywell volume, this comparison is equivalent to comparing concentrations. Notice that the RADTRAD mass is much higher than that in MELCOR. This difference in airborne concentrations (or mass in this case) illustrates the need for the factor R^*, described earlier in Equation (2-2), to normalize AST-predicted airborne concentrations with MELCOR best estimate predictions. This normalization is necessary in order to avoid excessive conservatism associated with using the AST in this application.

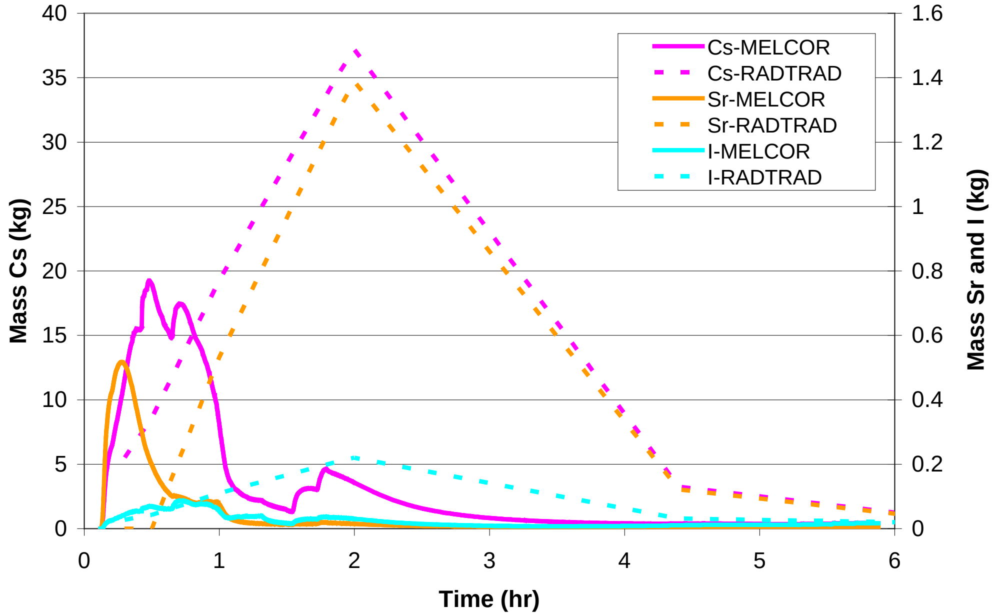


Figure 5-5: Mass of Cs, Sr, and I in RADTRAD and MELCOR Drywells for Mk-I RLB Case

Rather than de-convolve the R_M and R* factors separately, for expediency, they have been derived as their product. That is, the scaling factor is produced as simply the ratio of MELCOR-predicted stream dome concentrations to the RADTRAD-AST predicted drywell concentration. This combined factor is shown in Figure 5-6 for RN class 2, 3, and 4. From the graph, it was determined that an average ratio between 0 and 0.5 hr and 0.5 to 1 hr would be sufficient to characterize the source term. A ratio of 1 after 1 hr was found to be bounding. RN-2 had a limiting ratio of 15.24 for the first 30 minutes, and RN class 4 had a limiting ratio of 6.64 for the following 30 minutes. These ratios will be used in the RADTRAD MSL models for the Mk I to scale the NUREG-1465 derived source term.

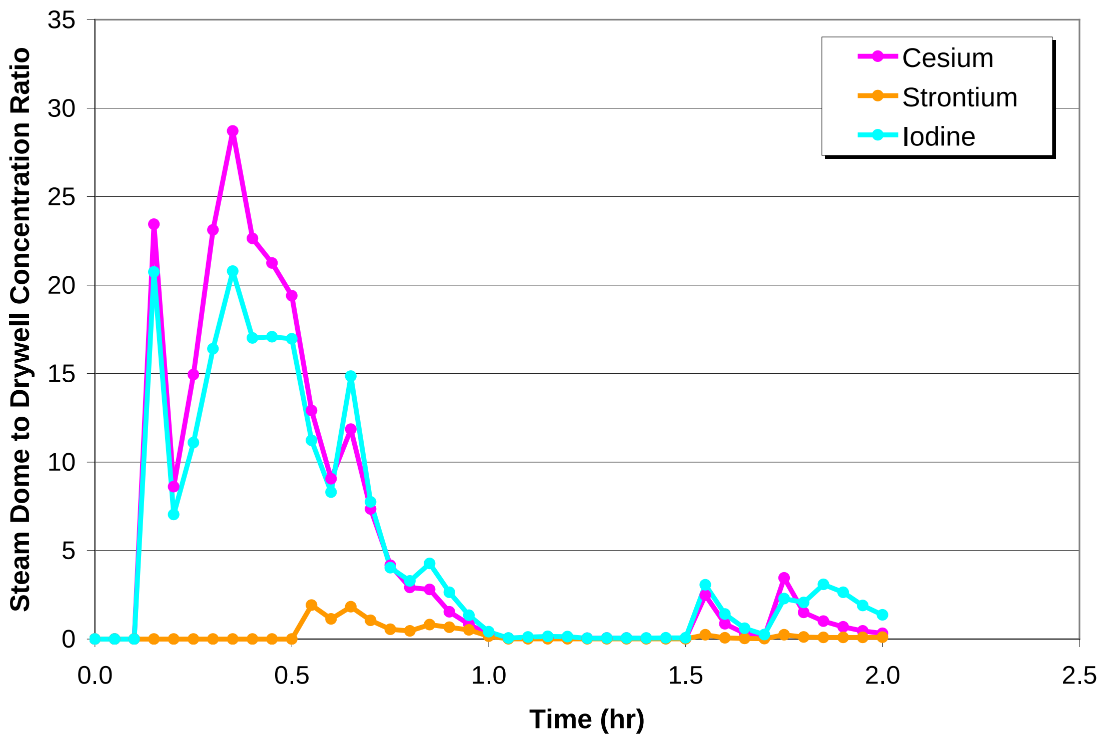


Figure 5-6: Ratio of MELCOR Steam Dome to RADTRAD Drywell for Cs, Sr, and I in Mk-I RLB

Table 5-1: MELCOR Steam Dome to RADTRAD Drywell Ratio for Various Time Intervals for Mk-I RLB

time (hr)	**Cs**	**Sr**	**I**
0.0-0.5	15.24	-----	11.87
0.5-1.0	6.33	0.81	6.64
1.0-2.0	0.59	0.06	0.95

When the steam dome-to-drywell ratio for the Mk-III was examined similarly, it was significantly higher than that of the Mark I. While the steam dome concentrations for the two MELCOR models were similar, the RADTRAD drywell concentration for the Mk-III was much lower than that of the Mk-I. This is a result of the drywell volume for the Mk-III being about 70% larger than that of the Mk-I. The two RADTRAD model core inventories are similar, but when a similar source is placed into a much larger volume the concentration will be lower by a ratio of the volumes. Figure 5-7 shows the ratio of steam dome to drywell for the Mk-III. Notice that the scale on this figure is double that of the Mk-I figure. The average ratios for the same time periods as in Table 5-1 can be found in Table 5-2.

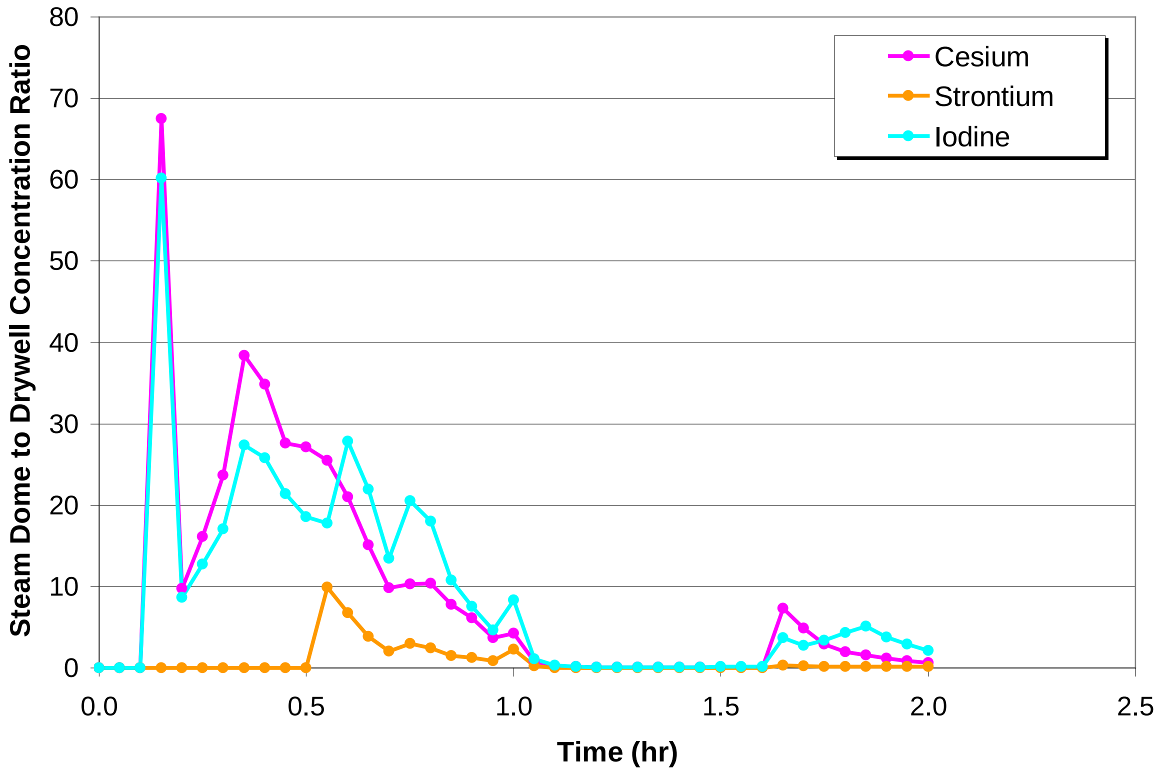


Figure 5-7: Ratio of MELCOR Steam Dome to RADTRAD Drywell for RN Class 2, 3, and 4 in Mk-III RLB

Table 5-2: MELCOR Steam Dome to RADTRAD Drywell Ratio for Various Time Intervals for Mk-III RLB

time (Hr)	Cs	Sr	I
0.0-0.5	21.43	-----	16.90
0.5-1.0	11.61	2.57	14.45
1.0-2.0	1.14	0.14	1.56

These two tables correspond to the Powers 10% aerosol removal by natural processes model for BWR drywells. Values are 20% higher when using the Powers 50% model.

To approximate the effect of a larger drywell increasing the concentration ratio, the ratios determined earlier are multiplied by a scaling factor related to the drywell volume. This scaling factor calculated as the ratio of the drywell volume for the geometry under consideration to the drywell volume that was used to obtain the original ratios (1.59E+05 ft^3 for the Mk-I drywell). For the Mk-III model the scaling factor is calculated to be 1.698. Therefore, the effective steam dome to drywell fission product concentration ratios used for the Mk-III will be:

- 0-0.5 hr: 15.24 * 1.698 = 25.88
- 0.5-1 hr: 6.64 * 1.698 = 11.28

In general, we conclude that this drywell scaling factor should be generally applicable to other BWR containment geometries, such as Mk-II designs.

The steam dome to drywell ratios were implemented in RADTRAD by determining the source concentration using the provided ratios, determining the fraction of AST release that would result in this concentration when placed in the source volume used in RADTRAD, and scaling the AST release to the source volume using these fractions.

In order for the model to represent the physical flow paths of the main steamline connections, the source volume is given the volume of the steam dome for the first two hours (after 2 hrs it is assumed that the vessel has been reflooded with an assumed equilibration of steam dome and drywell FP concentrations). As a result, the radionuclide releases must be scaled according to this volume in order to get the desired concentration within the steam dome. The volumes used for the steam dome, which are taken from the MELCOR models, are 3.7116E+03 ft^3 for the Mk-I and 6.6392E+03 ft^3 for the Mark III. The precise values used for these volumes are not important because the concentrations are scaled to them.

The ratio of steam dome volume to drywell volume for the Mk-I was 0.02334, and the ratio for the Mk-III was 0.02459. Therefore, the final fraction of the source that was placed in each of the modeled steam domes is shown in Table 5-3. Although the steam dome to drywell ratios for Cs and I in the MK-III RLB case are somewhat greater than 1 during the period from 1 to 2 hours, the ratio of steam dome-to-drywell concentration was taken to be 1 for times greater than 1 hour in the calculation of the source fractions (i.e. the source term fraction entering the steam dome after 1 hour is set equal to the steam dome to drywell volume fraction). Because the concentration in the steam dome is expected to drop below that of the drywell upon reflood and remain less than that of the drywell for some period before eventual equilibration with the drywell concentration, a value of 1 is expected to be conservative if the ratio was averaged beyond 2 hours.

Table 5-3: Source Term Fractions to be Placed in Steam Dome Volumes for Mk-I and Mark III

	Mark I	Mark III
Steam Dome 1 (0-0.5 hr)	0.35575	0.63638
Steam Dome 2 (0.5-1 hr)	0.15500	0.27737
Steam Dome 3 (> 1 hr)	0.02334	0.02459

5.3 Removal Coefficients

The removal coefficients (lambdas) used in the steam line only RADTRAD models are those determined by the MELCOR MSL model uncertainty studies. The inboard lambdas are set to zero (see discussion in Section 7.4). The 50th percentile removal coefficients used for the RADTRAD simulations are shown in Table 3-14.

5.4 MSL Flow Rates

The methodology for calculating flow rates for the RADTRAD models is similar to the approach used by Metcalf [10]. The method is based on equations for orifice flow such as described by Bird, Stewart and Lightfoot [11].

The mass flow rate of an ideal gas through a nozzle is given by:

$$w = A \cdot \sqrt{2P_{up}\rho_{up} \cdot \left(\frac{k}{k-1}\right) \cdot \left[\left(\frac{P_{dn}}{P_{up}}\right)^{\frac{2}{k}} - \left(\frac{P_{dn}}{P_{up}}\right)^{\left(\frac{k+1}{k}\right)}\right]} \tag{5.1}$$

where

w = mass flow rate
A = area of nozzle
P_{up} = upstream pressure
P_{dn} = downstream pressure
ρ_{up}= upstream density
ρ_{dn}= upstream density, and
$k = C_p/C_v$ specific heat ratio.

The downstream pressure which results in the maximum flow rate using this equation is referred to as the critical pressure, P_{cr}, and is given by:

$$P_{cr} = P_{up} \cdot \left(\frac{2}{k+1}\right)^{\frac{k}{k-1}}. \tag{5.2}$$

The mass flow rate remains constant with any further reduction of the downstream pressure beyond the critical pressure. Therefore, whenever $P_{dn} < P_{cr}$, the downstream pressure, P_{dn}, in equation 6.1 should be replaced by the critical pressure given in equation 6.2.

These equations can be used to:

1. determine the flow area given the mass flow rate of a given gas and pressures and temperatures on either side of an orifice, and to:
2. determine the mass flow rate of a specified gas given the flow area and pressures and temperatures on either side of an orifice

The flow rate at accident conditions is determined using essentially these two steps.

The first step in the process of determining MSIV leakage under accident conditions is the determination of the flow area that would produce the observed leakage of air during the valve leak testing. Properties for air, the MSIV test conditions, and the limiting mass flow rate are used in the nozzle flow equation to determine a flow area, A. The limiting mass flow rate for this calculation can be obtained by multiplying the limiting volumetric leakage at standard conditions by the air density at the standard conditions. The accident leakage rate is then determined by a subsequent application of the same equation using the calculated leak area, steam properties instead of air properties, and accident pressures and temperatures. The calculated accident mass flow rate can be converted to volumetric flow, which is used as input for RADTRAD, at different locations along the steam lines using the local density.

Although these equations are sufficient to determine the accident leakage rate in the event that a single MSIV closes, further calculations are required for the situation when both MSIVs close in order to account for the fact that both valves do not experience the same pressure difference and for the fact that the gas can be heated by the steam lines, which is especially significant for the volume between the MSIVs due to the potential pressurization of this volume. A detailed explanation of the methods used to determine flow rates for the case when both MSIVs close is provided in the calculation below.

5.4.1 Example calculation

For the determination of the volumetric flows for the pathways connecting the volumes in RADTRAD, Metcalf's approach was followed, including the following major simplifying assumptions:

1. both valves experience the same pressure difference, the drywell/steam dome to atmospheric pressure difference.
2. that the different temperatures upstream of the MSIVs can be used to determine the different volumetric flow rates through the valves

The assumptions and values used in this example are for the purposes of demonstrating the methodology only. Ideally, the temperature and pressure expected upstream of each MSIV should be used in the determination of flow through that valve. The determination of bounding values for accident pressures and temperatures on either side of the MSIVs would require a more thorough heat transfer analysis than was applicable for this study.

Metcalf's analysis essentially assumes both valves see the same pressure drop and that critical flow limits leakage. Downstream of the outboard valve, the volumetric flow is determined by converting the outboard valve volumetric flow at its upstream pressure and temperature to volumetric flow at the temperature and pressure estimated for conditions downstream of the outboard valve using the standard ideal gas relation that PV/T remains constant. This produces an increased volumetric flow since the gas is expanding significantly after exiting the outboard valve under the assumed pressure conditions. Using the Imperial units employed by Metcalf, the method is reiterated as follows:

1. Determine a flow area using the test pressure, air properties, and allowable leakage.
2. Use this area to calculate a flow rate for steam at accident conditions for the inboard portion of the MSL and the inboard MSIV.
3. Calculate the increase in volumetric flow for the outboard MSIV due to an increased temperature in the volume between the MSIVs.
4. Determine the flow rate outboard of the MSIVs at standard atmospheric pressure (P_{atm}).
5. Verify that the assumed conditions indeed resulted in sonic flow.

The critical pressure and a mass flow per unit area (G = *w/A*) were determined using equations 6.2 and 6.3, respectively. Equation 6.3 is simply equation 6.1 rewritten on a per unit area basis with specific volume, $\nu=1/\rho$, used in place of density. Once the allowable leakage has been converted to a mass flow rate under the test conditions using the ideal gas law and specific volume for air (Equation 6.4), this mass flow rate can then be divided by the calculated G to obtain the leakage flow area (Equation 6.5).

$$G=\sqrt{2\times g_c\times\frac{k}{k-1}\times\frac{P}{\nu}\times\left(\left(\frac{P_{cr}}{P}\right)^{\frac{2}{k}}-\left(\frac{P_{cr}}{P}\right)^{\frac{k+1}{k}}\right)} \tag{5.3}$$

$$\text{leak rate limit}\left[\frac{lbm}{s}\right]=\text{leak rate limit}\left[scfh\right]\times\frac{P_{atm}}{P_{test}}\times\frac{1}{\nu\left[\frac{ft^3}{lbm}\right]}\times\frac{1[hr]}{3600[s]} \tag{5.4}$$

$$\mathrm{A}\left[\mathrm{ft}^2\right]=\frac{\text{leak rate limit}\left[\frac{\mathrm{lbm}}{\mathrm{s}}\right]}{\mathrm{G}\left[\frac{\mathrm{lbm}}{\mathrm{s}\cdot\mathrm{ft}^2}\right]} \tag{5.5}$$

Note that when using SI consistent units, the conversion factor g_c is not required.

The mass flux (mass flow rate per unit area) of steam through the inboard MSIV at accident conditions is determined with Equations 6.2 and 6.3 using parameter values that are consistent with those at the accident conditions. The pressure downstream of this valve, however, is assumed to be atmospheric. The pertinent parameters that differ between accident and test conditions are the specific volume, the pressure and temperature on which it depends, and the ratio of specific heats (1.3 for steam versus the 1.4 for air). This newly calculated G can then be multiplied by the specific volume at those conditions and the flow area calculated earlier (Equation 6.6) to determine the volumetric flow through the inboard MSIV.

$$\text{leak rate}\left[\frac{\mathrm{ft}^3}{\mathrm{s}}\right]=\mathrm{G}\left[\frac{\mathrm{lbm}}{\mathrm{s}\cdot\mathrm{ft}^2}\right]\times\mathrm{A}\left[\mathrm{ft}^2\right]\times\nu\left[\frac{\mathrm{ft}^3}{\mathrm{lbm}}\right] \tag{5.6}$$

The pressure between the MSIVs is expected to be closer to the inboard MSL pressure than to ambient. Metcalf assumes that the pressure upstream of the outboard valve is the same as the pressure upstream of the inboard valve. The gas temperature between the MSIVs, however, is assumed to be higher because the steamlines are still near operating temperature. Metcalf has assumed that the gas temperature upstream of the inboard valve is close to the drywell temperature but MELCOR analyses show that this is not the case. With approximately the same pressure as the inboard MSL, the velocity, and also volumetric flow if equal leakage areas are assumed in both MSIVs, is assumed to increase by the ratio of the sonic velocities of the two valves (Equation 6.7) . Therefore, to estimate the volumetric flow rate through the outboard

MSIV, the volumetric flow rate calculated for the inboard MSIV should be multiplied by the ratio calculated using this equation. Since all other parameters in this equation are equal for both volumes (i.e. identical primed and non-primed values) because the same gases are being compared, this sonic velocity ratio becomes the square root of the temperature ratio.

$$\frac{v'_{sonic}}{v_{sonic}} = \frac{\sqrt{(k'-1)\frac{C'_p}{M'}T'}}{\sqrt{(k-1)\frac{C_p}{M}T}} = \sqrt{\frac{(k'-1)C'_p M T'}{(k-1)C_p M' T}} \tag{5.7}$$

The prime in this equation in this calculation indicates conditions upstream of the outboard MSIV (i.e. the volume in between the MSIVs) and the unprimed parameters refer to conditions in the main steam line upstream of the inboard MSIV. M refers to the molecular mass.

The MSL piping downstream of the out-board MSIV will be at atmospheric pressure but the temperature is assumed to be nearly the same as between the MSIVs. The volumetric flow outboard of the MSIVs can be determined using the standard ideal gas relation, PV/T=constant. To determine the volumetric flow rate outboard of both MSIVs, the multiplier calculated by Equation 6.8 should be multiplied by the volumetric flow rate calculated for the outboard MSIV.

$$\left(\frac{P_{inboard}}{P_{atm}}\right)\left(\frac{T_{outboard}}{T_{inboard}}\right) \tag{5.8}$$

In this equation, the *inboard* subscript refers to the gas between the MSIVs and the *outboard* subscript refers to the gas outboard of the outboard MSIV. The volumetric flow rate through the outboard MSIV could alternatively also have been determined using the ideal gas relation.

For the representative Mk-I reactor a test pressure of 25 psig (39.7 psia) was used [12]. The allowable leakage was 205 scfh from one steamline and 155 scfh from a second line. Using these values, flow areas of 3.35E-5 ft^2 and 2.53E-5 ft^2 are calculated for the respective leakage rates. The accident conditions from the drywell are 63.8 psia and a saturation temperature of 296.7° F. The temperature in the volume between the MSIVs was assumed to be 551° F [12]. The pressure in this volume is also expected to be higher than in the drywell for a short period of time but this was neglected for conservatism as was done by Metcalf. Therefore, the pressure between the MSIVs is assumed to be the same as that in the in-board MSL (63.8 psia). The outboard MSL is assumed to be at atmospheric pressure and 551° F. The flow rates calculated for the MK-I design using these conditions and the equations discussed earlier are summarized in Table 5-4.

Table 5-4: Mk-I Main Steamline Flow Rates for RADTRAD Calculations

	Technical Specification Leak Rate Limit (scfh)	Flow Rate for Inboard MSIV and Inboard MSL (cfh)	Outboard MSIV Flow Rate (cfh)	Flow Rate Outboard of MSIVs (cfh)
Line A	205	113.92	131.68	660.58
Line B	155	86.13	99.56	499.46

The representative Mk-III documentation lists a test pressure of 26.2 psia which yields a down stream critical pressure somewhat less than atmospheric. As a result, the standard atmospheric pressure was used in place of critical pressure for Equation 6.4. The allowable leakage for the Mark III, considered to be 100 scfh in the two shortest lines and 50 scfh for the next shortest line, yields flow areas of 3.06E-5 ft^2 and 1.53E-5 ft^2 for the 100 scfh and 50 scfh flows, respectively. The accident conditions are assumed to be at the test pressure of 26.2 psia and a saturated temperature of 242.5° F. Once again, the standard atmospheric pressure, instead of the critical pressure, was used as the downstream pressure in Equation 6.4 for both valves. The vapor temperature in the MSL between MSIVs was assumed to be 500° F with a pressure of 26.2 psia. There is no appreciable amount of outboard piping in the Mk-III steamline, therefore an outboard line is not modeled. Although no outboard section of piping is modeled , the flow outboard of the MSIVs needs to be calculated to determine the volumetric flow from the condenser, which was assumed to be at atmospheric pressure. The flow out of the condenser was assumed to have the same temperature as the area between the MSIVs (500°F) for the purpose of calculating the volumetric flow rate. The flow rates calculated for the MK-III design using these values are summarized in Table 5-5.

Table 5-5: Mk-III Main Steamline Flow Rates for RADTRAD Calculations

	Technical Specification Leak Rate Limit (scfh)	Flow Rate for Inboard MSIV and Inboard MSL (cfh)	Outboard MSIV Flow Rate (cfh)	Flow Rate Outboard of MSIVs (Condenser Out Flow) (cfh)
Line A	100	101.29	118.41	246.69
Line B	100	101.29	118.41	246.69
Line C	50	50.65	59.20	123.35

5.5 Unchanged Model Parameters

Parameters not discussed above were taken from the Peach Bottom [12] and Grand Gulf [13] alternative source term documentation for the Mk-I and Mk-III models respectively. No attempt was made to validate assumptions used in these models. The models were used as presented in the documents with one exception. The Mk-III model for Grand Gulf credited a main steamline leakage control system (MSLCS). Since recent industry submittals typically do not credit leakage control systems, the Grand Gulf model was altered to consider a leakage pathway without the MSLCS.

These parameters include the removal mechanisms in the drywell. For the Mk-I, the 10% Powers model option is used [14]. For the Mk-III , both iodine and aerosol removal coefficients were calculated according to NUREG/CR-0009, reference 27 in the Grand Gulf document [13]. These removal coefficients were compared with using a 10% Powers model and the results were found to be similar, less than 5% difference at 30 days.

Each reference document also sited a time at which the pressure in the drywell has dropped to a point to allow a reduction in the flow rates. For the Mk-I this time was given as 38 hours after the start of the accident, and for the Mk-III it was given as 24 hours.

Leakage from the source term volumes was also included in the models. For the Mk-I this represented 0.7% containment volume per day which was reduced to half after 38 hours. For the Mk-III 3000 cfm representing drywell bypass flow, scaled by the ratio of the steam dome volume to the drywell volume, was used. This value was reduced by half at 24 hours corresponding with the pressure drop in the drywell.

The flows between the environment and the control room were also taken from the reference documents. The Mk-I model has two flows into the control room, one filtered and one not and one unfiltered flow out of the control room. The Mk-III model has one unfiltered flow into the control room and one unfiltered control out of the control room, but unlike the Mk-I it also has recirculating filters on the control room volume.

The other parameters taken from the reference documents were the dose location and simulation parameter values. These include the χ/Q values, occupation factors, breathing rates, and simulation time step sizes.

Iodine removal values were taken from the Peach Bottom reference document. This was a filter of 50% on elemental iodine for each portion of the line that was credited. Consistent with current practice, no organic iodine removal is credited. In the Mk-I models this was two filters on each line. For the Mk-III this was one filter because the inboard portion is not credited and there is not a significant portion of piping modeled downstream of the outboard MSIV. With the treatment of gaseous iodine removal being uncertain, results will be given for the discussed filter configuration as well as no removal of elemental iodine within the steamlines.

Something unique to the Mk-I was a mixing of the drywell and wetwell volumes at 2 hours. The Mk-III documentation discussed mixing the drywell and lower containment volumes, but did not actually use this technique for the MSIV leakage calculation. In the Mk-I simulation, rather than adding a wetwell volume, there was simply a reduction in flow out of the source volume by the ratio of the drywell volume to the total volume of the wetwell and drywell. Through additional RADTRAD simulations, this method was shown to be conservative relative to actually mixing the volumes. The reduction in flow method was maintained in the following calculations because the condenser simulation was already at the 10 volume limit in RADTRAD, preventing the addition of a wetwell volume.

5.6 Main Steam Line Model Results

A summary of the main steam line model results for both the Mk-I and Mk-III models with and without condensers can be found in Table 5-6. For the cases with condensers, none of the doses exceed the 10 CFR limit. However, for the cases without condensers the control room doses far exceed the limit as do the EAB 2 hour integrated doses. The LPZ limit is not exceeded in the case of the Mk-I, but is in the case of the Mk-III. So the Mk-III without a condenser exceeds the dose limits at all three locations.

Table 5-6: Summary of TEDE Dose Results for Mk-I and Mk-III Steam Line Models (Elemental Iodine and Aerosol Removal Included, No Organic Iodine Removal Included)

	EAB, worst 2 hour integrated dose (rem)	**LPZ, integrated dose after 30 days (rem)**	**CR, integrated dose after 30 days (rem)**
10 CFR 50.67 Limit	25	25	5
Mk-I, No Condenser	49.5	8.6	57.7
Mk-I, Condenser	1.0	0.4	3.8
Mk-III, No Condenser	88.1	27.5	71.9
Mk-III, Condenser	0.5	0.4	1.4

The results for the simulations without elemental iodine removal can be seen in Table 5-7. There is not a large difference in the dose, but some increase is apparent. Because the dose results predominantly from the large concentration of aerosols, the difference between the elemental iodine removal case and the no-elemental-iodine-removal case is small.

Table 5-7: Summary of TEDE Dose Results for Mk-I and Mk-III Steam Line Models (No Elemental or Organic Iodine Removal Included, Aerosol Removal Included)

	EAB, worst 2 hour integrated dose (rem)	**LPZ, integrated dose after 30 days (rem)**	**CR, integrated dose after 30 days (rem)**
10 CFR 50.67 Limit	25	25	5
Mk-I, No Condenser	57.7	11.1	86.4
Mk-I, Condenser	1.4	0.62	7.0
Mk-III, No Condenser	142.4	42.6	117.8
Mk-III, Condenser	0.88	0.67	2.7

The LPZ and CR integrated doses out to 30 days and 24 hours are shown in Figure 5-8 through Figure 5-11. The results from all of these plots include iodine removal. It is apparent from the shape of the 30 day figures that the condenser serves as a significant hold-up volume that allows for decay and removal of activity and slowly releases the activity over a longer time period.

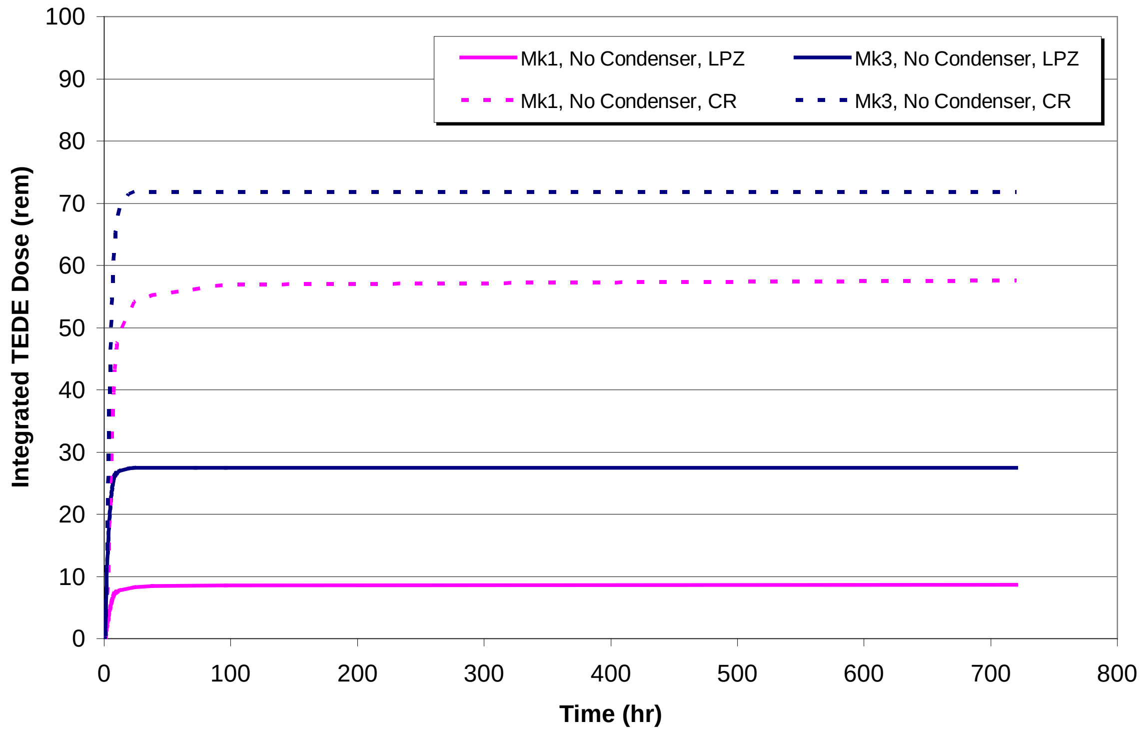


Figure 5-8: Integrated Control Room and LPZ TEDE for No Condenser Cases out to 30 Days

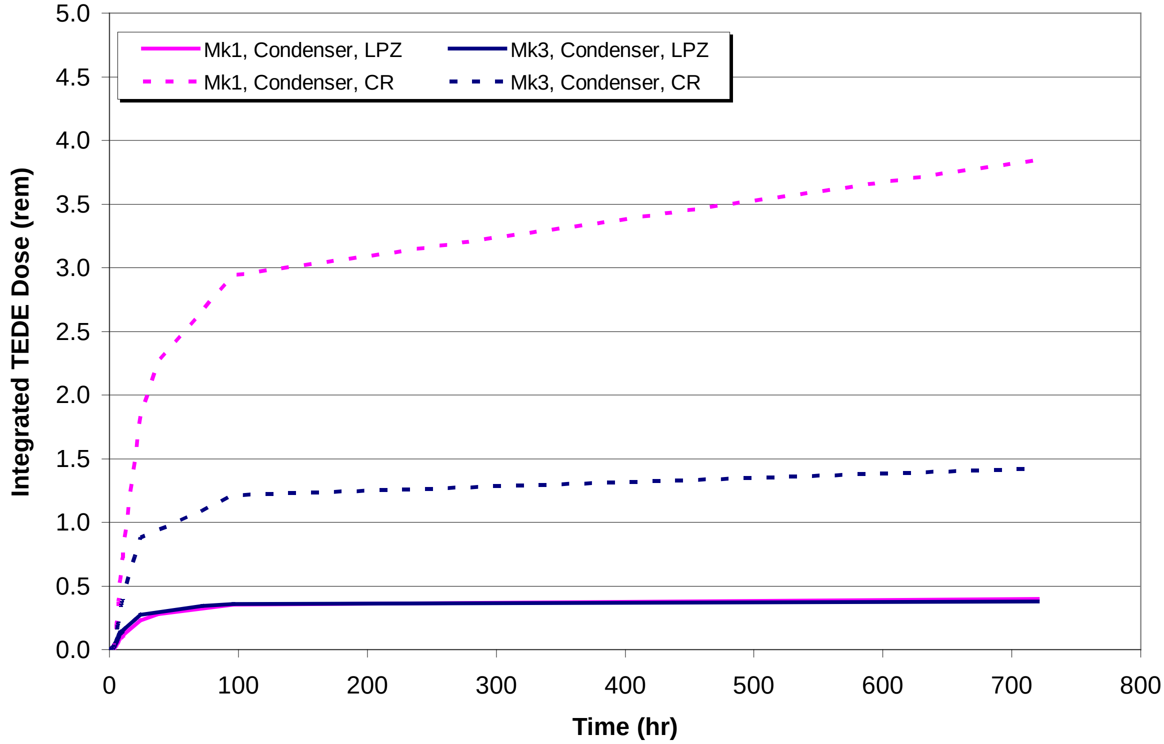


Figure 5-9: Integrated Control Room and LPZ TEDE for Condenser Cases out to 30 Days

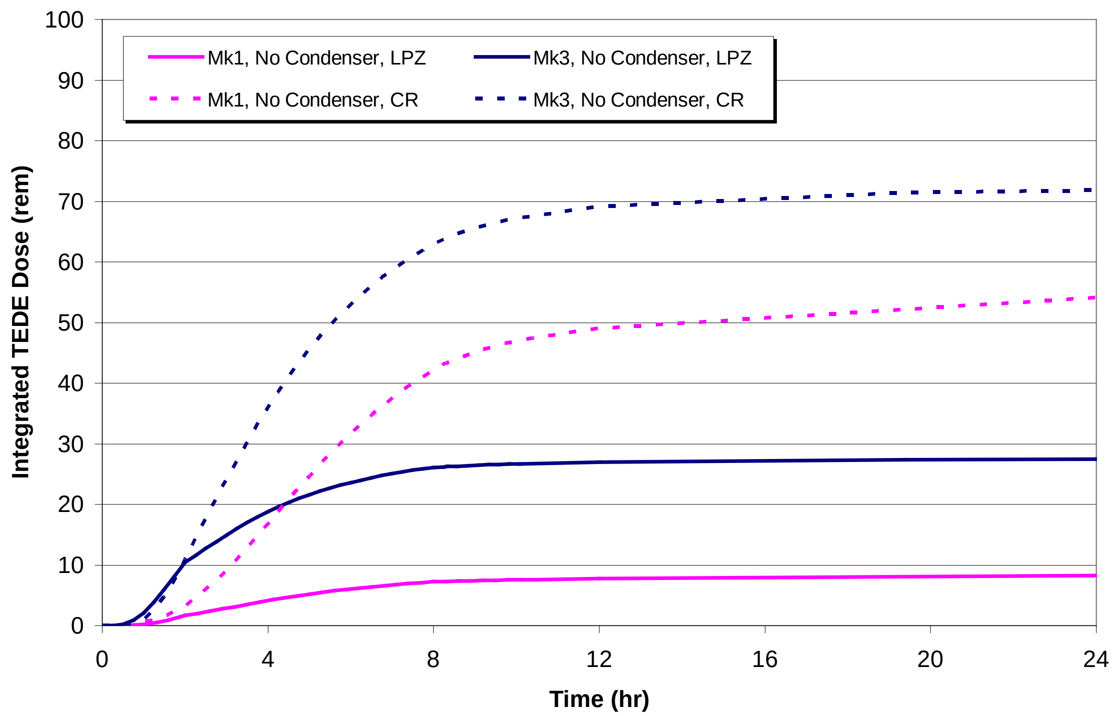


Figure 5-10: Integrated Control Room and LPZ TEDE for No Condenser Cases out to 24 Hours

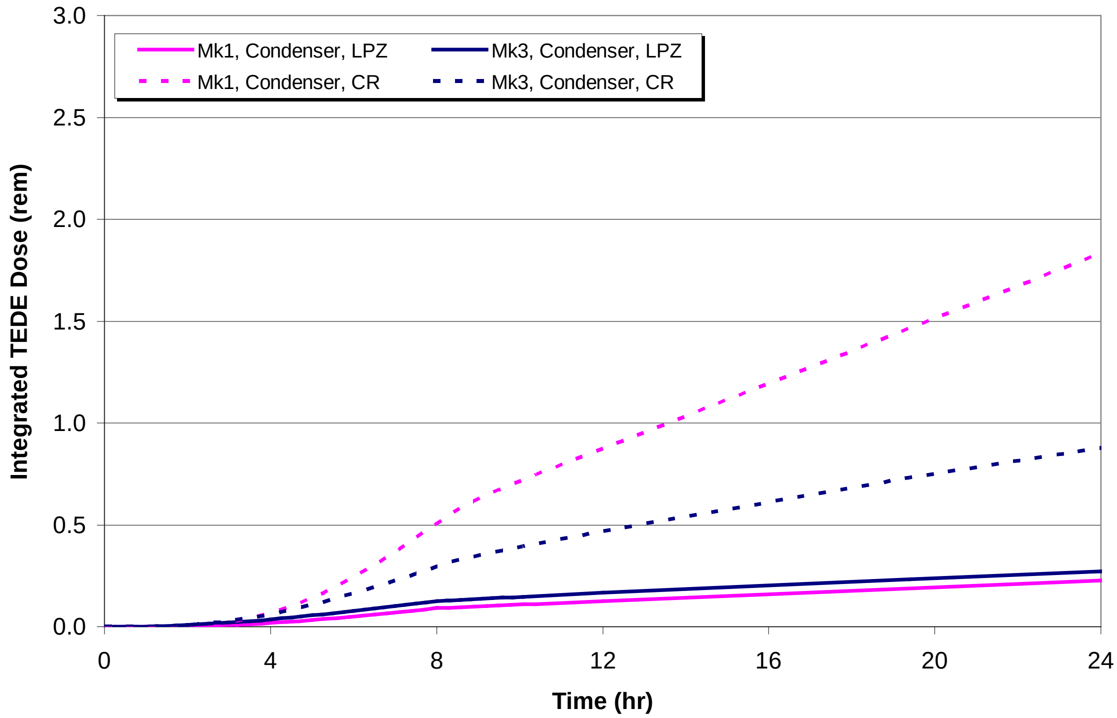


Figure 5-11: Integrated Control Room and LPZ TEDE for Condenser Cases out to 24 Hours

The integrated doses for the LPZ at 2 hours compared with the full model results at two hours are higher by a factor of nearly 5.5. The CR doses at 2 hours are higher by a factor of between 4.9 and 8.7 when compared with the full model doses. It was thought that much of this difference is due to ignoring deposition in the inboard sections of the main steamline. In order to investigate this further, the cases were each run with MELCOR derived lambdas for the inboard portions of the lines.

The results compared with the methodology proposed as well as the MELCOR values can be seen in Figure 5-12 and Figure 5-13. Notice that when the inboard lambdas are included in the simulation there is much better agreement between the Full Model and the MSL-Only Model. At 2 hours this difference is approximately a factor of 2.3 for the EAB and LPZ and 3.3 for the CR. These differences also decrease slightly by the time the simulations reach 5 hours.

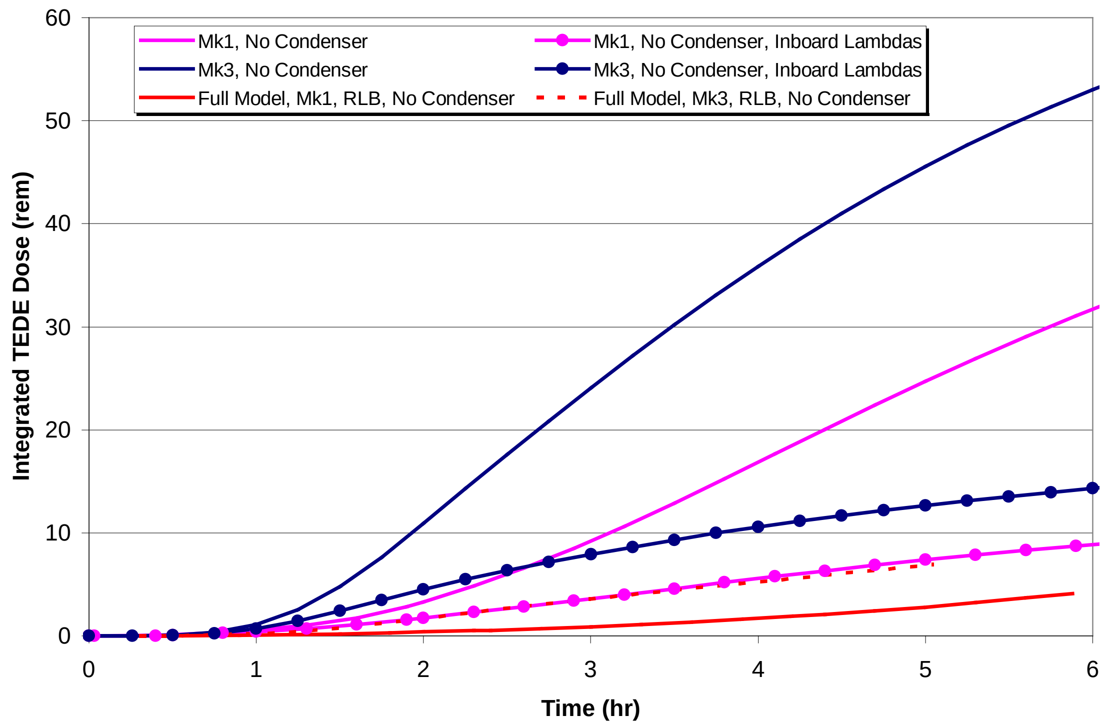


Figure 5-12: Comparison of CR Integrated TEDE with and without Inboard Lambdas to Full Model Results for Mk-I and Mk-III No Condenser Cases

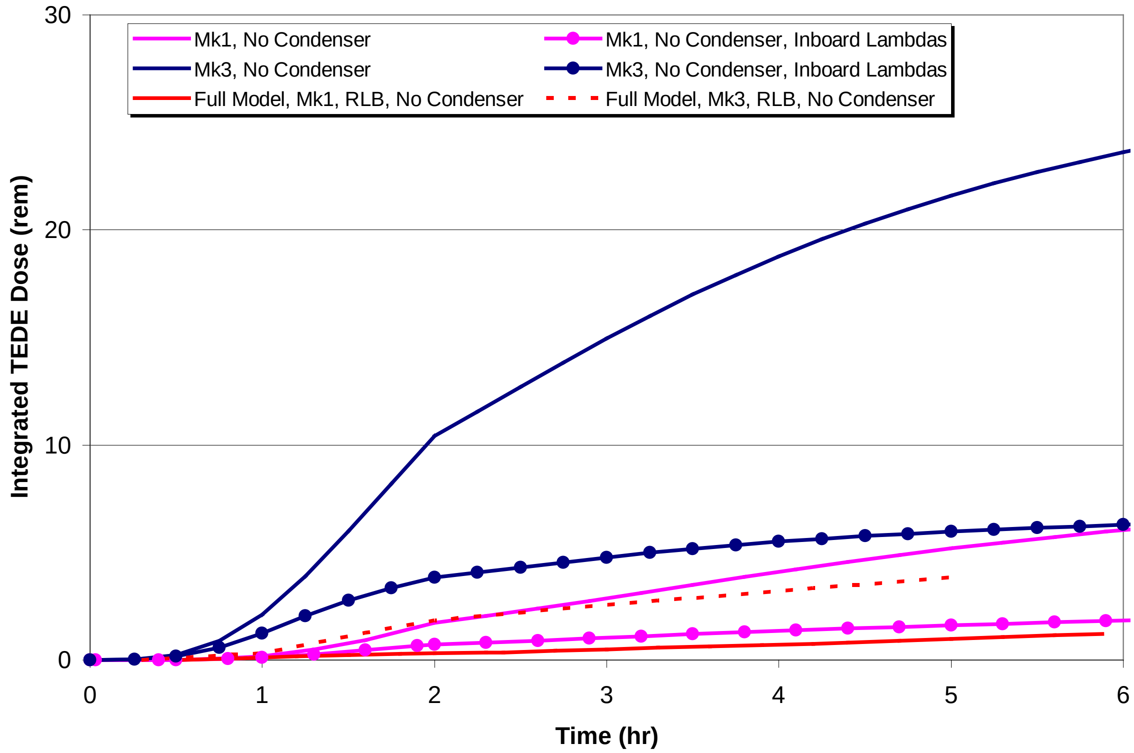


Figure 5-13: Comparison of LPZ Integrated TEDE with and without Inboard Lambdas to Full Model Results for Mk-I and Mk-III No Condenser Cases

5.7 RADTRAD Reference Industry Model

There is no reference model for the Mk-III reactor because in the document used as a source for all representative Mk-III values [13], the MSIV leakage calculation was only carried out to 18 minutes. Since we want to model a "representative" Mk-III, our calculations were carried out to the standard 30-days.

The Mk-I model used for comparison is from the Peach Bottom Analysis Number PM-1077. The title of this report is "Post-LOCA EAB, LPZ, and CR Doses Using Alternative Source Term (AST)." [15] The RADTRAD geometry for this analysis can be found on Page 84 and 85 of the report as Figures 4 and 5. They are shown here as Figure 5-14 and Figure 5-15.

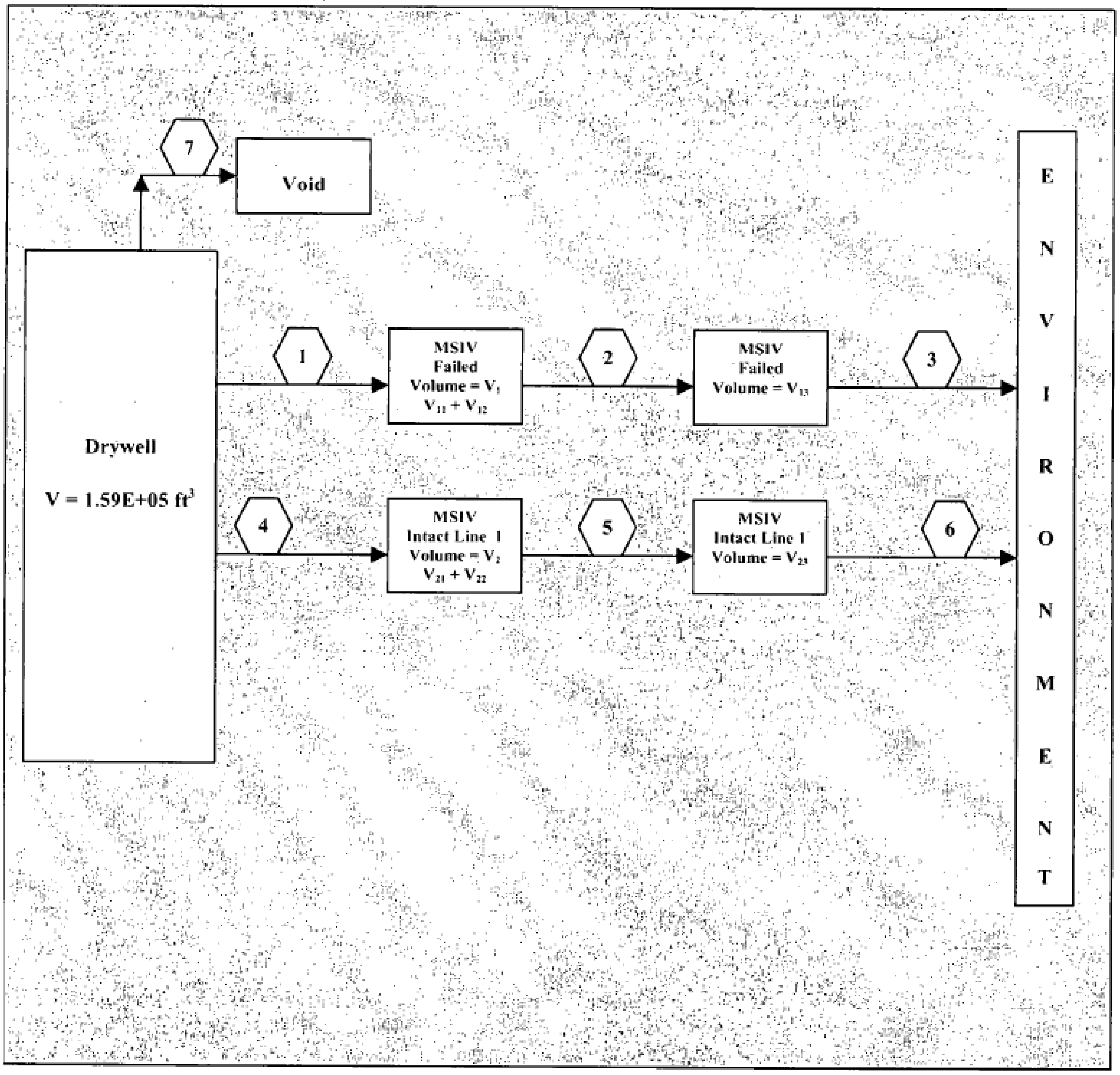


Figure 5-14 Part 1 of Peach Bottom RADTRAD Model, Mark 1 Comparison

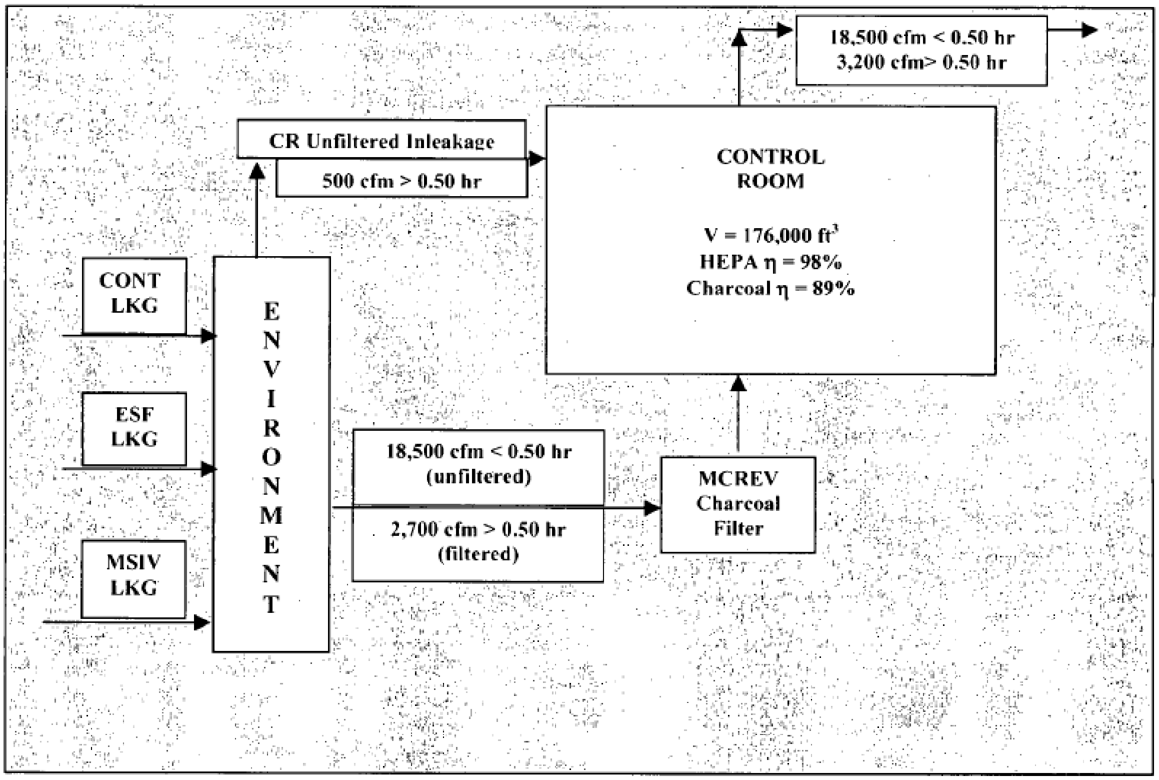


Figure 5-15 Part 2 of Peach Bottom RADTRAD Model, Mark 1 Comparison

The results for the MSIV leakage model start on page 135 of the report. The worst two hour dose for the exclusion area boundary (EAB) is 3.48 rem. The integrated dose out to 30 days for the low population zone (LPZ) is 0.99 rem. The integrated dose out to 30 days for the control room (CR) is 4.36 rem. All are within the regulatory limits.

In examining the input for this case, it was found that the Peach Bottom analysis credited wall deposition of iodine in the drywell using the RADTRAD containment spray model. To have results appropriate for comparison later in this study, a case identical to the Peach Bottom analysis was run in RADTRAD but without wall deposition..

6 Summary and Recommendations

6.1 Recommendations for the Application of Steam Dome-to-Drywell Concentration Ratios to RADTRAD Calculations

MELCOR best estimate analyses of two DBA initiated accidents where significant core melting and fission product release has occurred have been explored to evaluate MSIV leakage behavior for two widely deployed BWR containment designs, Mk-I and Mk-III. These analyses have shown that, during the first two hours of such an accident, the airborne fission product aerosol concentrations in the reactor vessel significantly exceed those in the drywell. Since the atmosphere in the reactor vessel supplies the effluent that ultimately leaks through the MSIV's, not the atmosphere in the drywell volume, these findings conclude that the current regulatory guidelines permitting the use of the fission product concentration in the drywell atmosphere during the first two hours prior to assumed vessel reflood is non-conservative for the purposes of evaluating the dose resulting from MSIV leakage, in addition to being conceptually inaccurate.

This study has investigated means of adapting the current regulatory *containment* source term for application to MSIV leakage analysis by means of scaling factors, accounting for differences in vessel fission product concentration and containment concentrations, and for differences in the NUREG-1465 derived containment concentrations compared to current best estimate derived containment concentrations. The developed scaling methodology preserves the simplified approach currently described in the regulatory guide by maintaining use of the AST; alternatively, detailed physics-based computer codes such as MELCOR, could be used to analyze source term release and transport.

Based on this work, it is recommended that the NUREG-1465 drywell fission product concentrations be scaled based on the time-phased scaling factors presented in Table 5-1 and Table 5-2 for application to Mk-I and Mk-III containments when determining the fission product concentration that is the source for MSIV leakage. These tables correspond to the Powers 10% aerosol removal by natural processes model for BWR drywells. The scaling ratios are approximately 20% higher when using the Powers 50% model. The differences in scaling factors for Mk-I and Mk-III are predominantly a result of the differences in the drywell volumes alone. Therefore, extension of these recommendations to Mk-II or containments with different volumes can be justified by scaling the steam dome to drywell fission product concentration ratios by drywell volume (if the drywell volume is larger, then a larger scaling factor is required to obtain correct vessel concentration). This is a generic recommendation concerning the appropriate airborne concentration for evaluating MSIV leakage. A methodology for accomplishing this recommendation using RADTRAD has been demonstrated, using RADTRAD techniques such as superposition to accommodate the time-phased behavior of the releases; however, other code-based methods could be used to accomplish the same objective.

6.2 Recommendations for Sprays in RADTRAD Calculations

The MELCOR Full Reactor model results show that the activation of drywell sprays reduces the fission product release to the environment by reducing the drywell pressure. This pressure reduction decreases the MSIV leakage flow rates thus reducing fission product release. It also increases the flow rate between the steam dome which moves additional fission products out of the steam dome into the drywell. This increase did not, however, appreciably reduce the fission product concentration observed in the steam dome for approximately 1 hour in the cases that were studied.

While drywell sprays can clearly reduce containment leakage, and indirectly reduce MSIV leakage by lowering drywell pressure, it is recommended that sprays not be credited for any reduction of the airborne concentration in the vessel supplying MSIV leakage during the first two hours, since, as discussed earlier, it is the vessel, not the drywell, that is the source of fission product concentration available for MSIV leakage.

Following a presumed recovery by vessel reflooding at two hours, drywell sprays can be credited for reducing the concentration of containment aerosols flowing back into the vessel and to the MSIV regions. Additionally, it would seem reasonable to allow credit for the pressure reduction resulting from the use of containment sprays, thereby reducing the MSIV pressure difference and the flow rate through the valves, provided adequate engineering analysis of containment response to sprays and valve leakage as a function of pressure drop is performed.

6.3 Recommendations for Removal Coefficients in RADTRAD Calculations

Removal coefficients were calculated for the in-board segments of the MSLs. It is recommended, however, that no credit be taken for aerosol deposition in this portion of the MSLs. The basis for this recommendation is that at times in the simulation the temperature of portions of the in-board MSL piping are predicted to be high enough to vaporize fission products that had been previously deposited. This secondary source cannot easily be incorporated into a RADTRAD model, and if the initial deposition (via a non-zero removal coefficient) is credited, the omission of this secondary source would result in an under-prediction of fission products released downstream, and ultimately to the environment.

There have also been questions raised regarding decay heat from fission products deposited in the in-board piping that could cause natural convection-driven bi-directional flow between the steam dome and the in-board MSLs. While the well-mixed nature of the MSL control volumes does, in some fashion, capture the enhanced mixing that such flow would cause, it does not address the potential for enhanced bulk transport of fission products from the steam dome into the in-board MSLs. Note that this issue has been previously cited as a basis for not taking credit for aerosol deposition in the in-board MSLs – see page 4-15 of reference [3].

Recommended removal coefficients based on the MSL-only model uncertainty results are given in the following tables, repeated for convenience from Section 4. The 50% values shown in

Table 6-2 reflect mean tendencies and could be used to avoid conservatism; alternatively, the 5% values in the previous table reflect a greater degree of conservatism.

Table 6-1: Recommended MSL and Condenser Removal Coefficients (5th Percentile)

MSL section	0 - 2 (hr)	2 - 12 (hr)	12+ (hr)
in-board	2.2*	0.0*	0.0*
between MSIVs	2.5	1.8	1.0
out-board	1.1	1.0	0.7
condenser	0.018	0.015	0.012

note. removal coefficients are given in hr^{-1}
* calculated values are shown, but a value of 0.0 hr^{-1} is recommended

Table 6-2: Recommended MSL and Condenser Removal Coefficients (50th Percentile)

MSL section	0 - 2 (hr)	2 - 12 (hr)	12+ (hr)
in-board	3.2*	1.3*	0.4*
Between MSIVs	2.9	2.4	2.0
out-board	1.3	1.3	1.0
condenser	0.020	0.018	0.015

note. removal coefficients are given in hr^{-1}
* calculated values are shown, but a value of 0.0 hr^{-1} is recommended

Table 6-3: Recommended MSL and Condenser Removal Coefficients (95th Percentile)

MSL section	0 - 2 (hr)	2 - 12 (hr)	12+ (hr)
in-board	4.9*	2.3*	1.1*
Between MSIVs	3.8	3.7	4.4
out-board	1.6	1.7	1.6
condenser	0.023	0.023	0.021

note. removal coefficients are given in hr^{-1}
calculated values are shown, but a value of 0.0 hr^{-1} is recommended

6.4 Recommendations for Post-Reflood Conditions in RADTRAD Calculations

The results from the Mk-III RLB case in which core sprays were activated 10 minutes before lower head failure indicate that the re-introduction of water into the core generates sufficient steam such that the vast majority of fission products in the steam dome and, to a lesser extent, in the drywell are swept into the wetwell, which effectively prevents them from being available for release to the environment via MSIV leakage. This same behavior is also seen in the Mk-I results at the time of lower core support plate failure.

Based on this observation, we recommend that the scaling factors be set to unity following the first two hour period where scaling factors are used to adjust the AST-based containment concentrations to reflect vessel concentrations. This conservatively assumes that the drywell and steam dome environments are well mixed after vessel reflood takes place.

Deposition properties after reflood should be based on the characteristics of the containment aerosol (i.e size effects). In the single analysis that explored post-reflood conditions it was observed that the intermediate pipe deposition lambdas decreased over time following reflood owing to the change in size distribution brought about as the larger particles fall out. These depletion trends were not significantly different from the 5% results reported in Table 6-1 (~1 hr^{-1}). More analyses may be required to summarize the physical effect of decreasing lambda with decreasing particle mean diameter; however, the trends reported in this report serve as a minimal basis for recommending smaller lambdas in the hours and days following reflood.

6.5 Recommendation for the Influence of Flow Rates on MSL and Condenser Removal Coefficients

The results from the MELCOR MSL RLB flow uncertainty case show that there is some relationship between flow rates in the MSLs and the removal coefficient. However, this relationship is also highly dependent on the point in time of the accident progression. This is due to the influence of the aerosol particle size distribution on the removal coefficient. The current results do not support the development of a quantitative relationship between MSL flow and MSL or condenser removal coefficients.

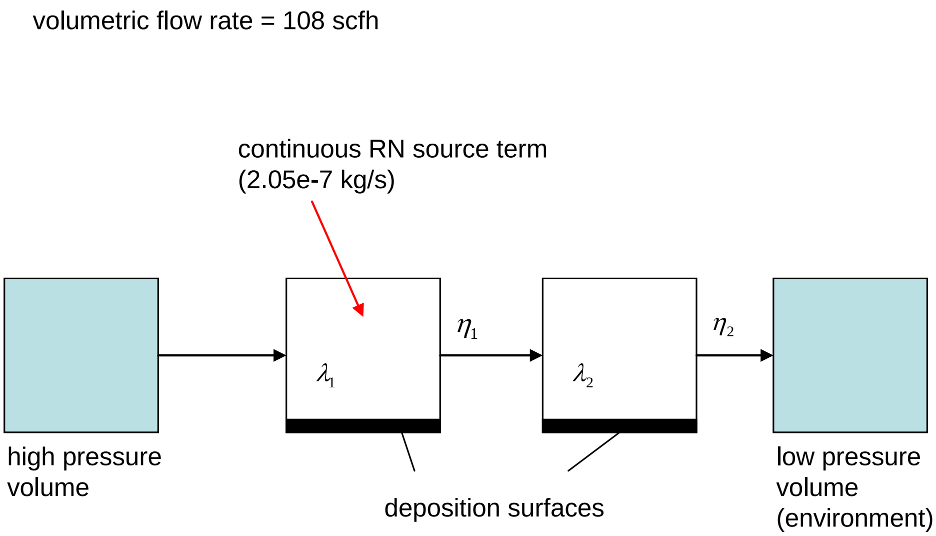
6.6 Recommendation Regarding the Use of Effective Filter Efficiencies for RADTRAD MSL Modeling

In the RADTRAD code the user has the option of specifying either removal coefficients to volumes or filter efficiencies to the flowpaths that connect the volumes. While these two treatments have been deemed equivalent [NRC 1998], this is only true in the specific case where the nuclide storage term is zero (i.e., steady-state conditions). The two options in RADTRAD for treating deposition are both empirical, and each represents very different fundamental views of physics. While a particular filter efficiency can be selected that in the end reflects the degree of holdup and removal of aerosols in a flow path, it cannot mechanistically represent the operative physics. To explore this further a small test problem, with geometry and conditions representative of MSIV leakage flow through steamlines, was investigated as described in the following.

A small MELCOR test problem was developed to evaluate the error in using filter efficiencies rather than removal coefficients under transient conditions. As shown in Figure 7-1, the problem consisted of four volumes: (1) a constant-pressure volume which was set to drive a constant 108 SCFH through down-stream flowpaths and volumes; (2) a volume [high-pressure] in which a single RN aerosol class was introduced at a constant continuous rate (2.05e-7 kg/s), this volume also contains a floor heat structure which provides a deposition location for gravitationally-settled aerosol particles; (3) a volume with floor heat structure; and (4) and a constant-pressure volume [environment] which was set to drive a constant 108 SCFH through up-stream flowpaths and volumes.

The test problem was run out to the time at which a steady-state concentration was reached (i.e., the slope of the curve of integrated RN mass in the environment is constant). Removal coefficients (lambda) and effective filter efficiencies (Feff) were calculated for the problem per the equations in the figure. An identical test problem model was then built for RADTRAD. That model was then run for two cases: (1) using the removal coefficients for the volumes; and (2) using the effective filter efficiencies for the flowpaths. The results were then compared to the MELCOR-predicted RN release to the environment. As shown in Figure 7-2, the case using the filter efficiency under-predicted the RN release (compare the blue curve with the black curve), while the case using the removal coefficients predicted an identical release to the MELCOR test problem result (compare the red dashed curve with the black curve).

Based on these results, it is recommended that deposition in MSL pipes only be modeled in RADTRAD using removal coefficients and that the optional modeling method of using effective filter efficiencies applied to flowpaths not be used unless an actual filtering process is being represented.



$$\lambda = \frac{1}{m}\dot{m}_{dep}$$

$$\eta = \frac{\dot{m}_{dep}}{\dot{m}_{dep} + \dot{m}_{out}}$$

Figure 6-1 MELCOR and RADTRAD Nodalization for Evaluating Effective Filter Efficiencies and Removal Coefficients

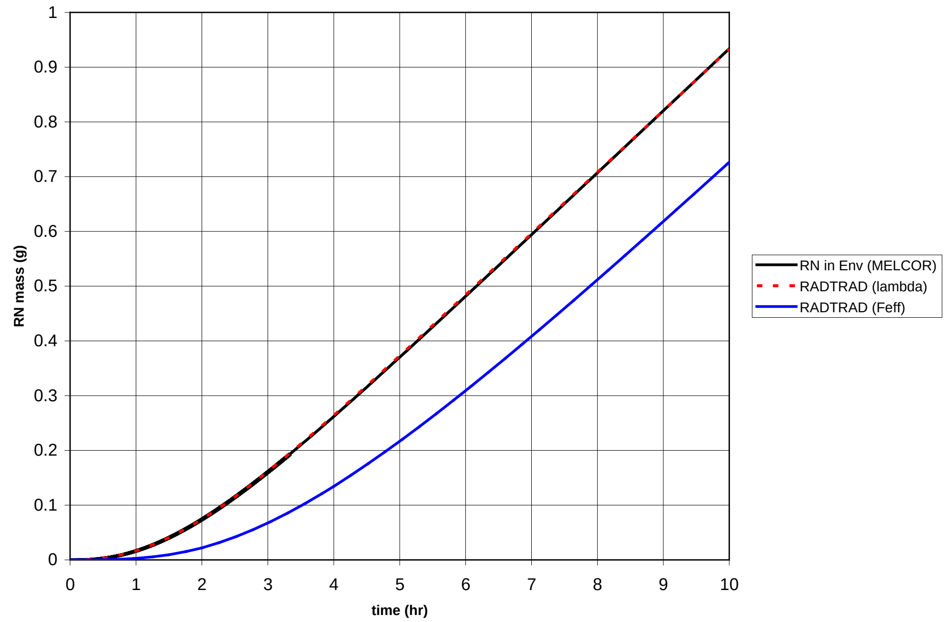


Figure 6-2 Evaluation of Effective Filter Efficiencies and Removal Coefficients --Comparison of MELCOR and RADTRAD RN Mass Releases.

6.7 Recommendation Regarding Calculation of MSIV Leakage Flow

It is recommended that analysis of valve leakage flow be generally based on the flow predicted using nozzle-flow theory as described in many fundamental text books such as Bird, Stewart and Lightfoot [11]. The flow equations described earlier in section 5.4, accommodate critical or sub-sonic flow and serve as a defensible means of analyzing both test leakage behavior and the scaling of these measured flows to the flows that would be expected under accident conditions.

The temperature and pressure expected upstream of each MSIV should be used in the determination of flow through that valve. The determination of bounding (conservative) values for the pressures and temperatures upstream of the MSIVs were not part of this study. Therefore, the temperatures and pressures used in the example are solely for the purposes of demonstrating the methodology and should not be taken as recommended values for MSIV leakage or as representative of the conditions upstream of the MSIVs.

7 Appendix A – Additional Mk-III Containment Analyses

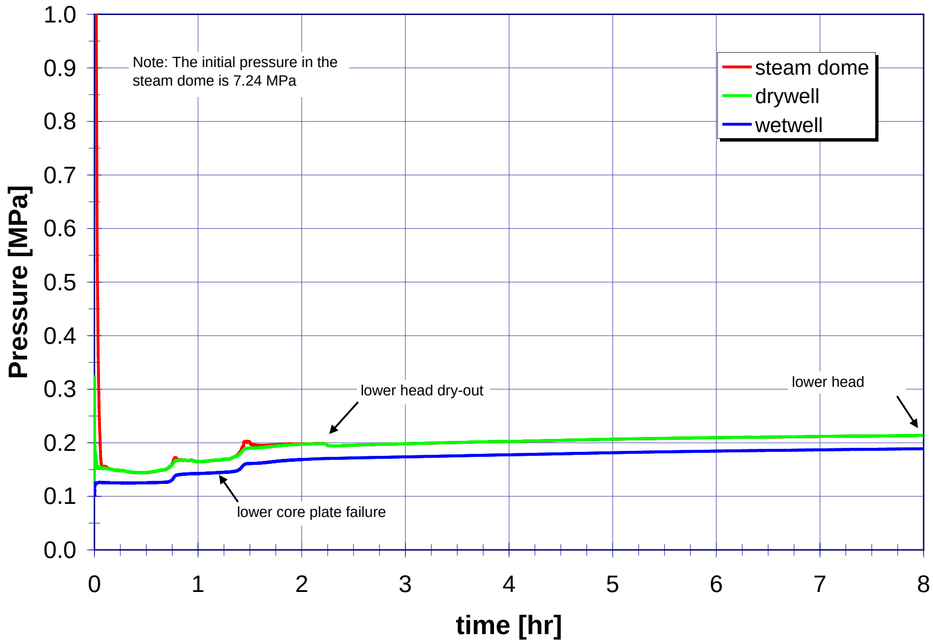


Figure 7-1 BWR Mk-III, MSLB, No Sprays: Steam Dome, Drywell, and Wetwell Pressure

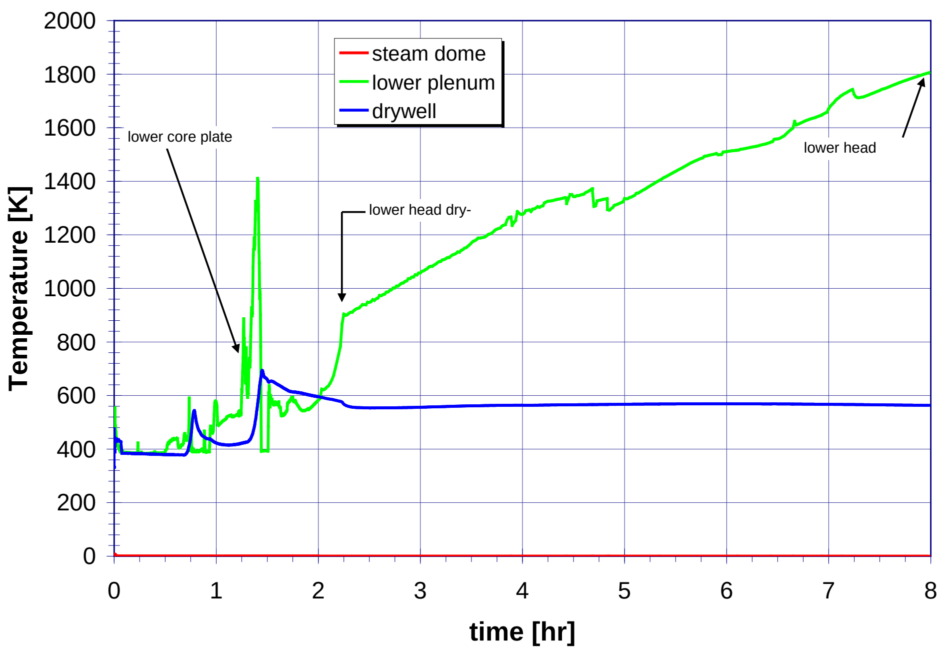


Figure 7-2 BWR Mk-III, MSLB, No Sprays: Steam Dome, Drywell, and Lower Plenum Vapor Temperature

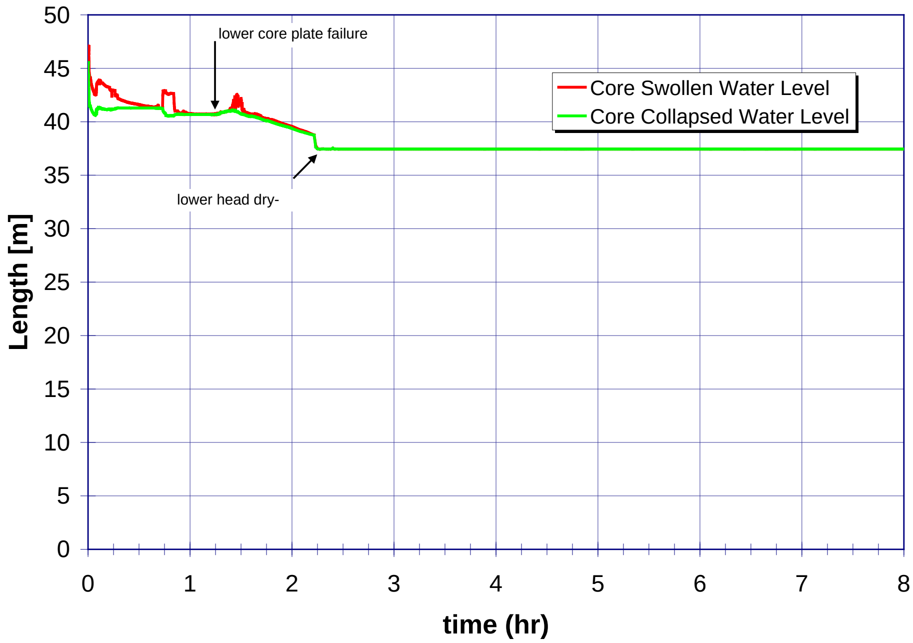


Figure 7-3 BWR Mk-III, MSLB, No Sprays: Core Water Levels

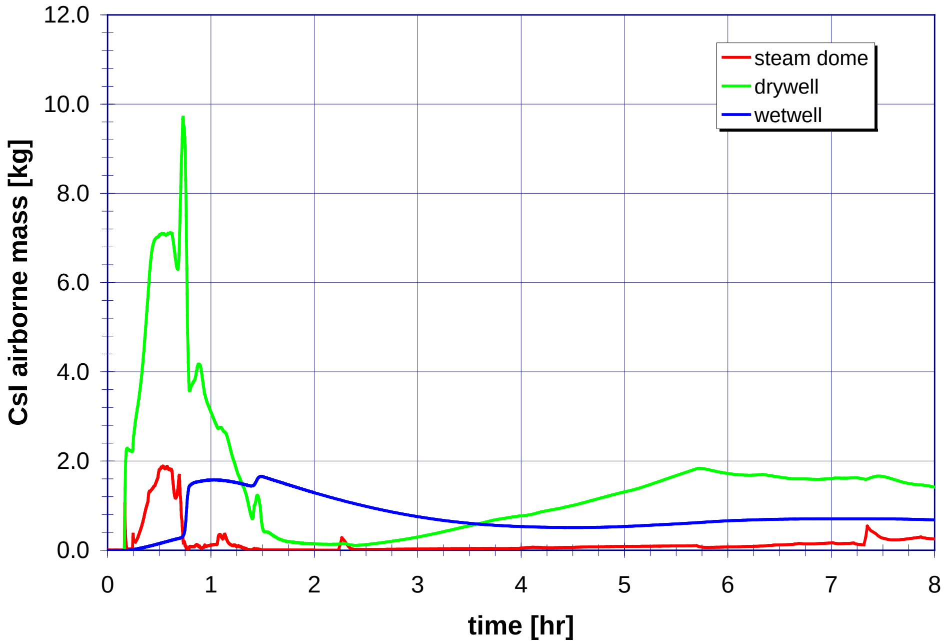


Figure 7-4 BWR Mk-III, MSLB, No Sprays: CsI Mass in the Steam Dome, Drywell, and Wetwell

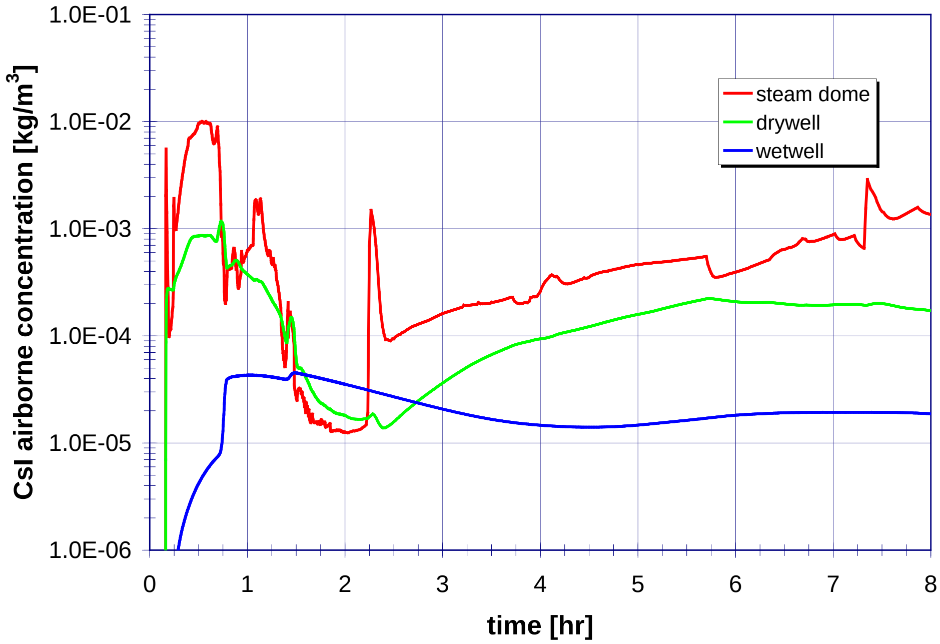


Figure 7-5 BWR Mk-III, MSLB, No Sprays: CsI Concentration in the Steam Dome, Drywell, and Wetwell

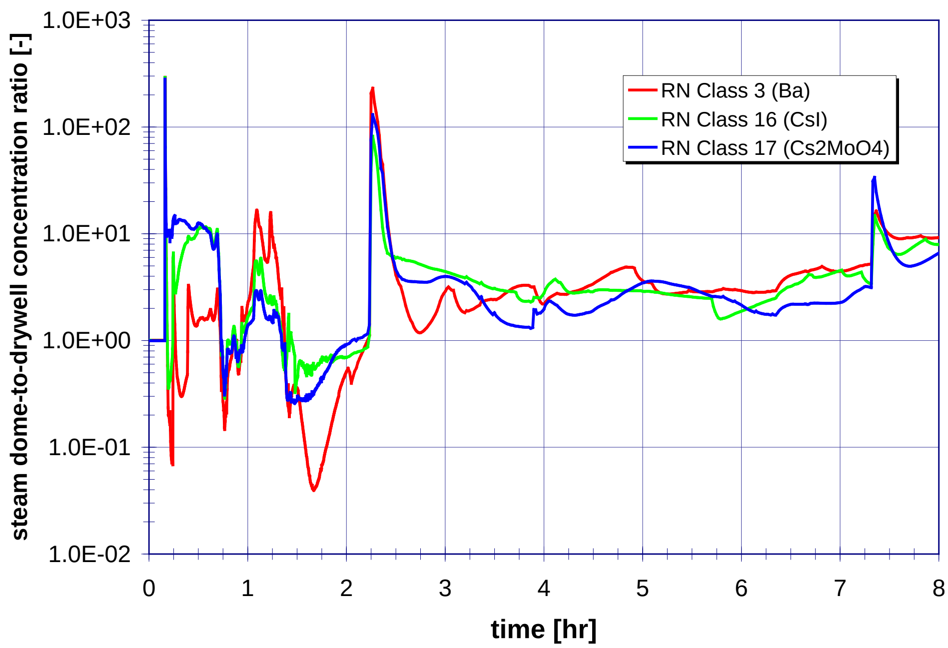


Figure 7-6 BWR Mk-III, MSLB, No Condenser, No Sprays: Steam Dome-to-Drywell Concentration Ratios

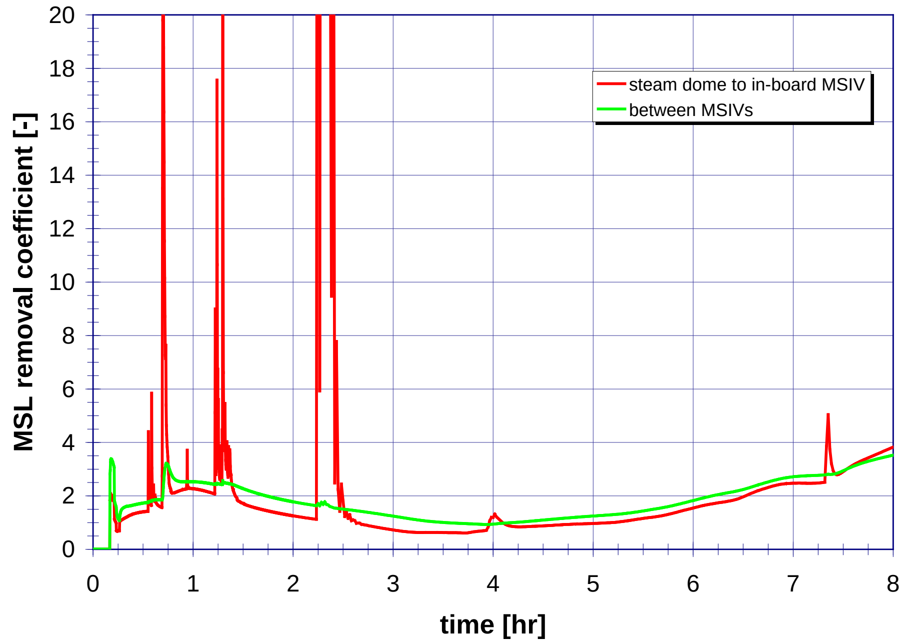


Figure 7-7 BWR Mk-III, MSLB,, No Sprays: MSL-A Removal Coefficients

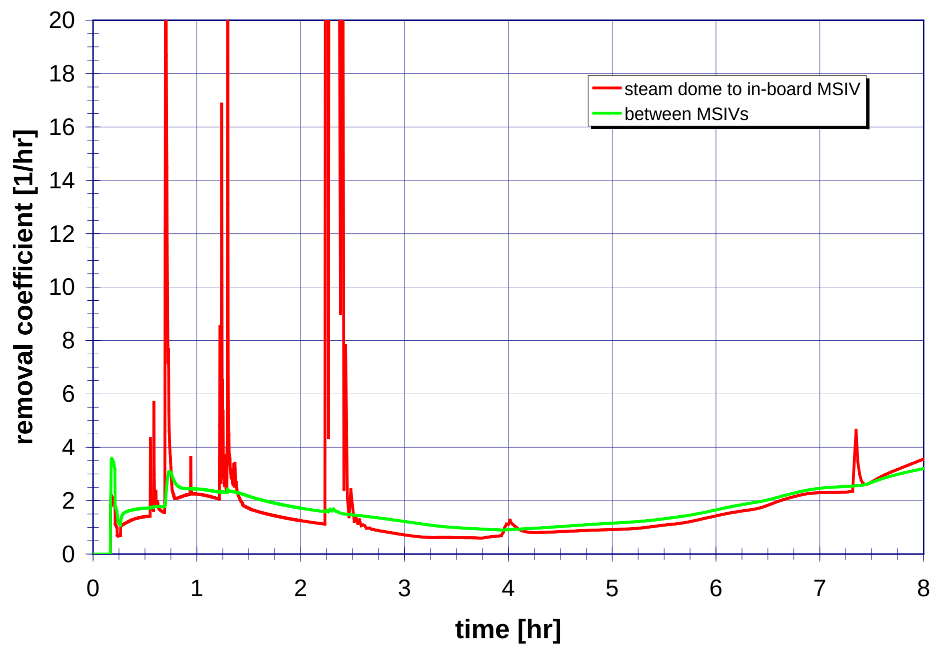


Figure 7-8 BWR Mk-III, MSLB, No Sprays: MSL-B Removal Coefficients

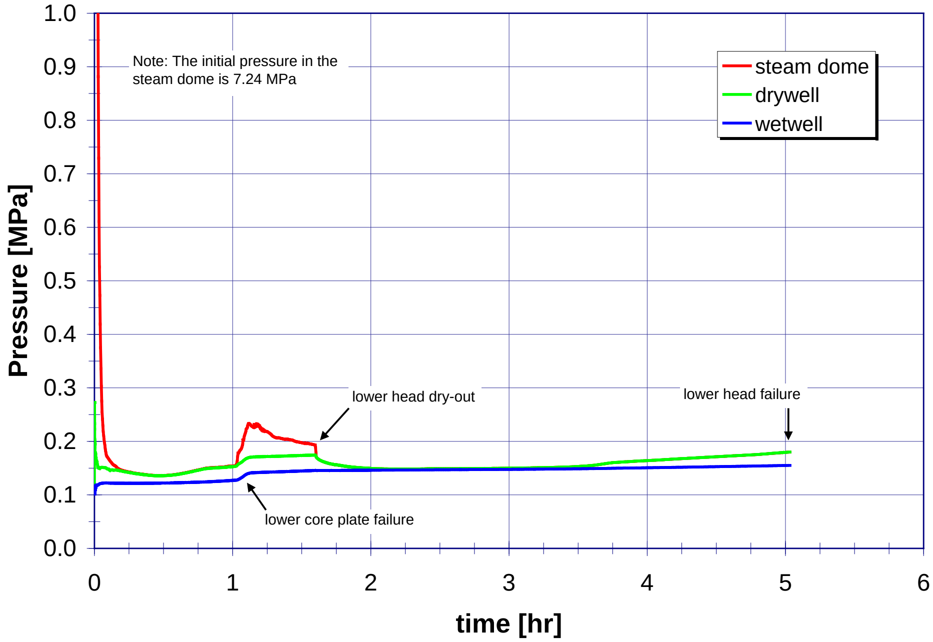


Figure 7-9 BWR Mk-III, RLB, No Sprays: Steam Dome, Drywell, and Wetwell Pressure

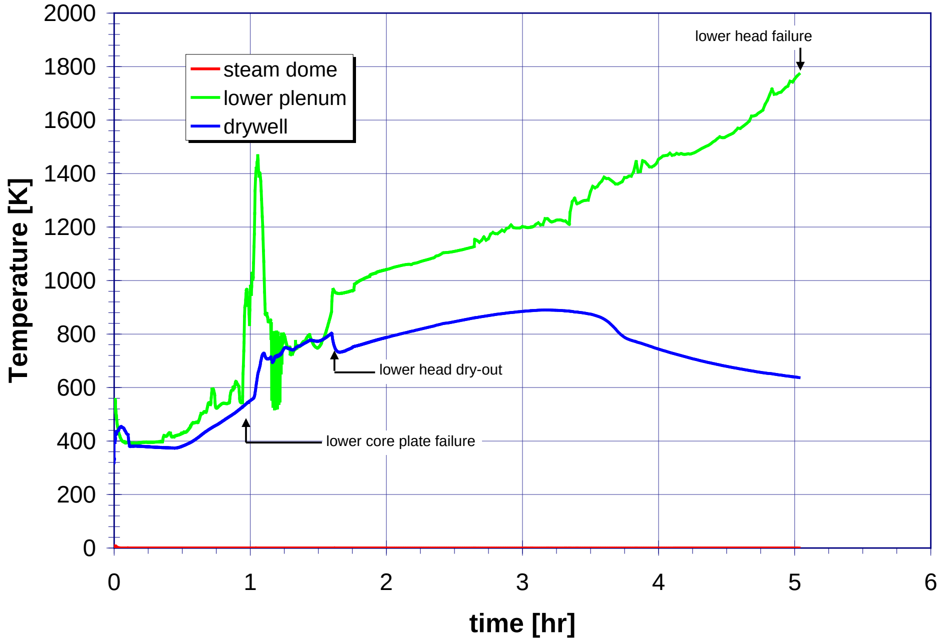


Figure 7-10 BWR Mk-III, RLB, No Sprays: Steam Dome, Drywell, and Lower Plenum Vapor Temperature

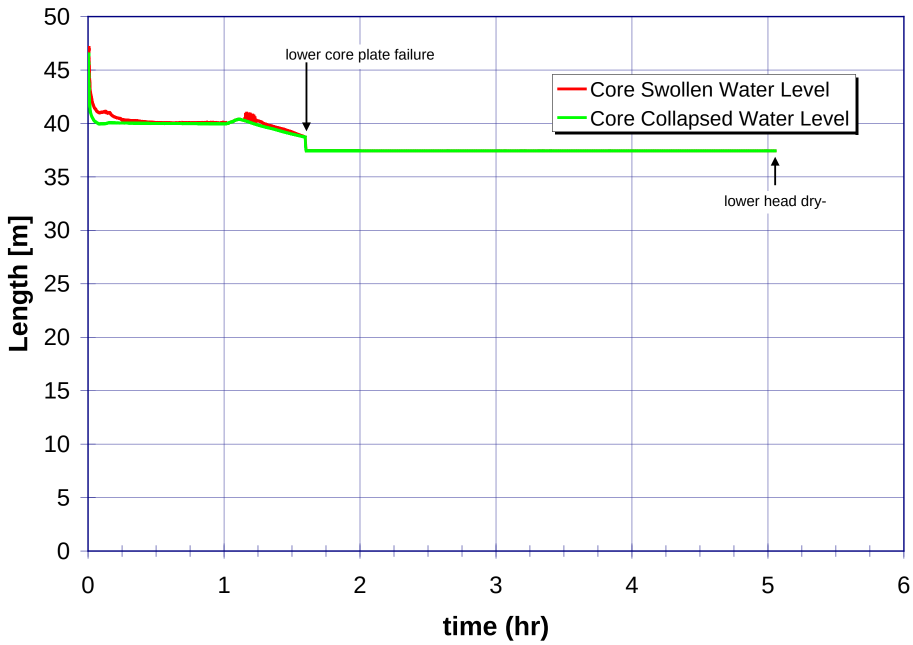


Figure 7-11 BWR Mk-III, RLB, No Sprays: Core Water Levels

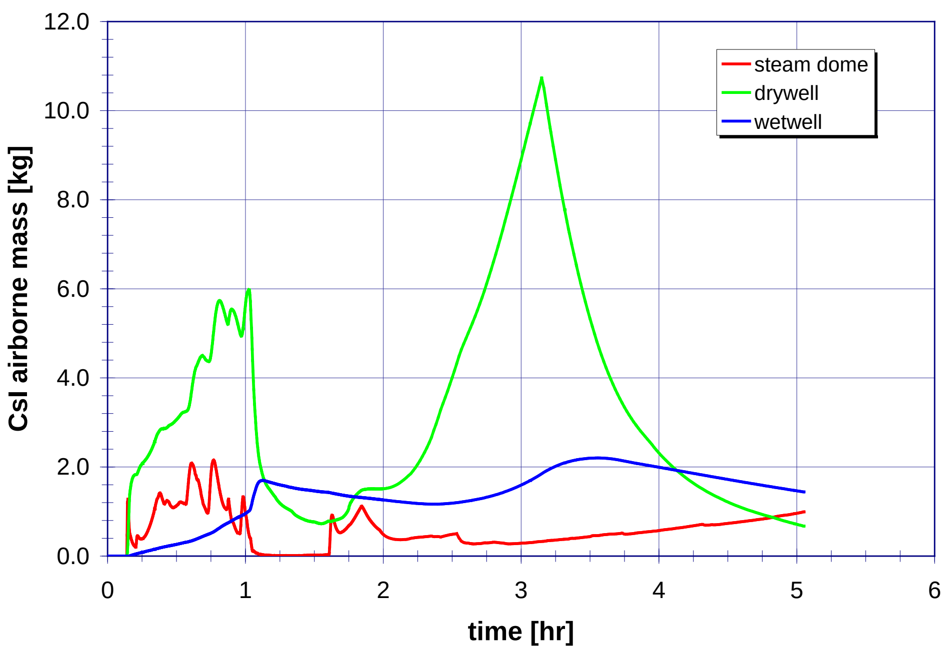


Figure 7-12 BWR Mk-III, RLB, No Sprays: CsI Mass in the Steam Dome, Drywell, and Wetwell

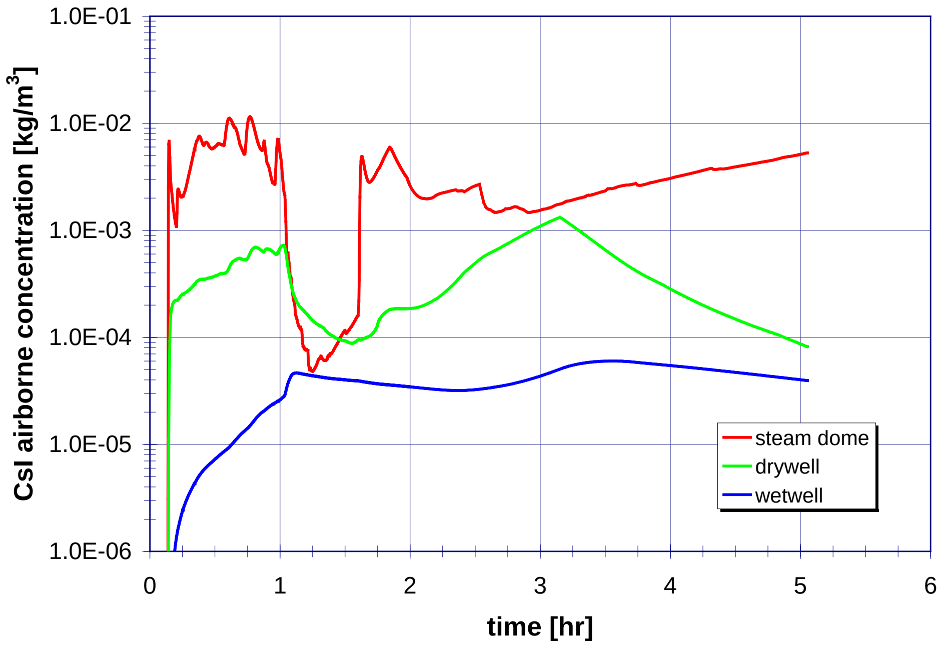


Figure 7-13 BWR Mk-III, RLB, No Sprays: CsI Concentration in the Steam Dome, Drywell, and Wetwell

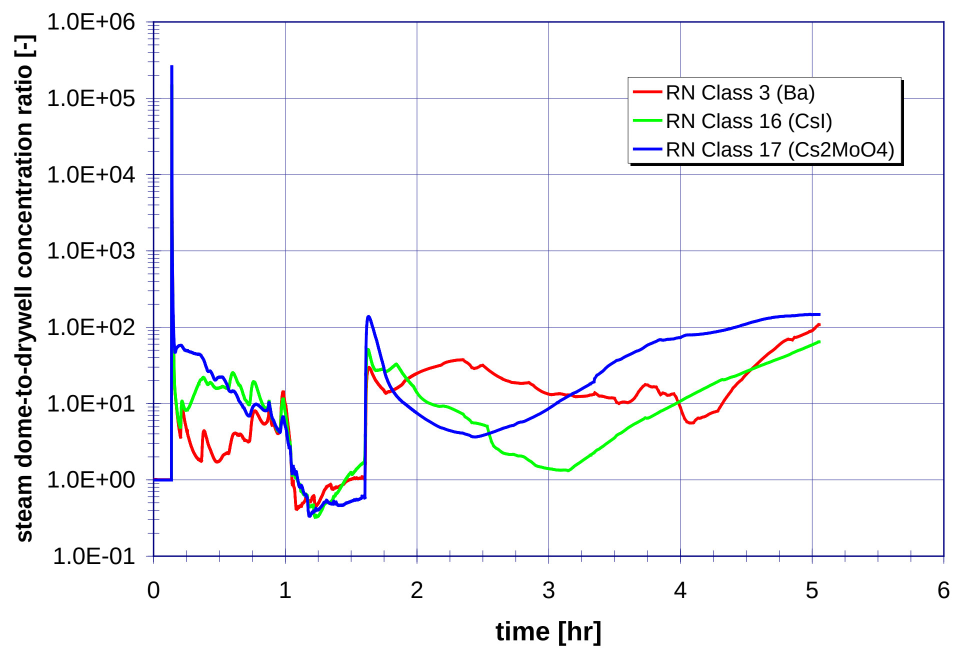


Figure 7-14 BWR Mk-III, RLB, No Condenser, No Sprays: Steam Dome-to-Drywell Concentration Ratios

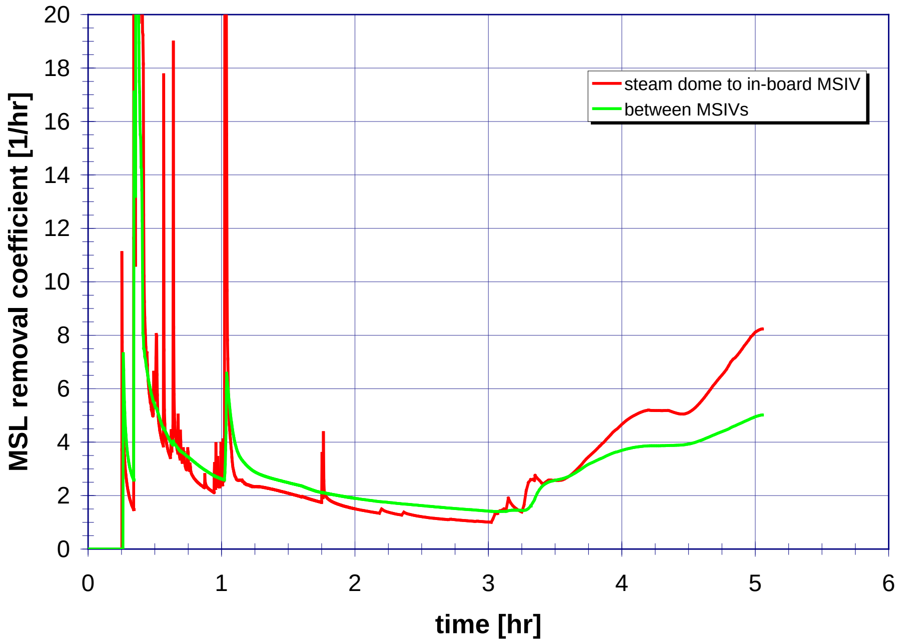


Figure 7-15 BWR Mk-III, RLB,, No Sprays: MSL-A Removal Coefficients

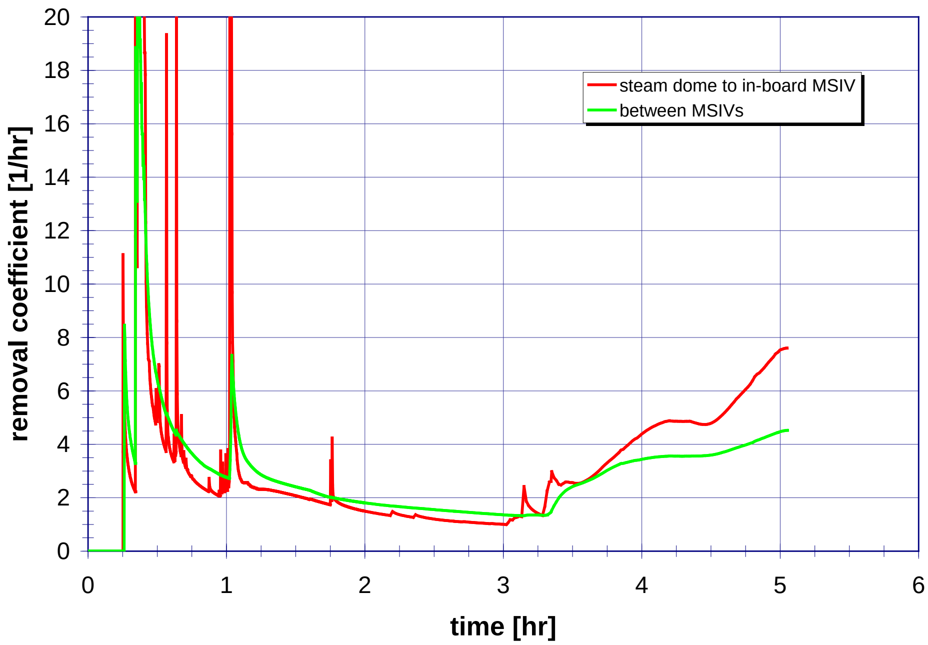


Figure 7-16 BWR Mk-III, RLB, No Sprays: MSL-B Removal Coefficients

8 Appendix B – RADTRAD MK-I Input Files

8.1 RTF file

```
Release Fraction and Timing Name:
 BWR, RG 1.183, Table 1 Section 3.2
 Duration (h):   Design Basis Accident
  0.5000E+00  0.1500E+01  0.0000E+00  0.0000E+00
 Noble Gases:
  0.5000E-01  0.9500E+00  0.0000E+00  0.0000E+00
 Iodine:
  0.5000E-01  0.2500E+00  0.0000E+00  0.0000E+00
 Cesium:
  0.5000E-01  0.2000E+00  0.0000E+00  0.0000E+00
 Tellurium:
  0.0000E+00  0.0500E+00  0.0000E+00  0.0000E+00
 Strontium:
  0.0000E+00  0.2000E-01  0.0000E+00  0.0000E+00
 Barium:
  0.0000E+00  0.2000E-01  0.0000E+00  0.0000E+00
 Ruthenium:
  0.0000E+00  0.2500E-02  0.0000E+00  0.0000E+00
 Cerium:
  0.0000E+00  0.5000E-03  0.0000E+00  0.0000E+00
 Lanthanum:
  0.0000E+00  0.2000E-03  0.0000E+00  0.0000E+00
 Non-Radioactive Aerosols (kg):
  0.0000E+00  0.0000E+00  0.0000E+00  0.0000E+00
 End of Release File
```

8.2 NIF file

```
Nuclide Inventory Name:
 Peach Bottom (PBAPS) AST - in Ci/MW
 Power Level:
  0.1000E+01
 Nuclides:
  60
 Nuclide 001:
Co-58
  7
 0.6117120000E+07
 0.5800E+02
 0.1529E+03
none     0.0000E+00
none     0.0000E+00
none     0.0000E+00
Nuclide 002:
Co-60
  7
 0.1663401096E+09
 0.6000E+02
 0.1830E+03
none     0.0000E+00
none     0.0000E+00
none     0.0000E+00
Nuclide 003:
none     0.0000E+00
none     0.0000E+00
Nuclide 004:
Kr-85m
  1
 0.1612800000E+05
 0.8500E+02
 0.8313E+04
Kr-85    0.2100E+00
none     0.0000E+00
none     0.0000E+00
Nuclide 005:
Kr-87
  1
 0.4578000000E+04
 0.8700E+02
 0.1633E+05
Rb-87    0.1000E+01
none     0.0000E+00
none     0.0000E+00
Nuclide 006:
Kr-88
  1
 0.1022400000E+05
 0.8800E+02
 0.2303E+05
```

```
Rb-88    0.1000E+01
none     0.0000E+00
none     0.0000E+00
Nuclide 007:
Rb-86
  3
 0.1612224000E+07
 0.8600E+02
 0.6518E+02
none     0.0000E+00
none     0.0000E+00
none     0.0000E+00
Nuclide 008:
Sr-89
  5
 0.4363200000E+07
 0.8900E+02
 0.2798E+05
none     0.0000E+00
none     0.0000E+00
none     0.0000E+00
Nuclide 009:
Sr-90
  5
 0.9189573120E+09
 0.9000E+02
 0.3178E+04
Y-90     0.1000E+01
none     0.0000E+00
none     0.0000E+00
Nuclide 010:
Sr-91
  5
 0.3420000000E+05
 0.9100E+02
 0.3801E+05
Y-91m    0.5800E+00
Y-91     0.4200E+00
none     0.0000E+00
Nuclide 011:
Sr-92
  5
 0.9756000000E+04
 0.9200E+02
 0.4017E+05
Y-92     0.1000E+01
none     0.0000E+00
none     0.0000E+00
Nuclide 012:
Y-90
  9
 0.2304000000E+06
 0.9000E+02
 0.3272E+04
none     0.0000E+00
none     0.0000E+00
none     0.0000E+00
Nuclide 013:
Y-91
  9
 0.5055264000E+07
 0.9100E+02
 0.3448E+05
none     0.0000E+00
none     0.0000E+00
none     0.0000E+00
Nuclide 014:
Y-92
  9
 0.1274400000E+05
 0.9200E+02
 0.4029E+05
none     0.0000E+00
none     0.0000E+00
none     0.0000E+00
Nuclide 015:
Y-93
  9
 0.3636000000E+05
 0.9300E+02
 0.4526E+05
Zr-93    0.1000E+01
none     0.0000E+00
none     0.0000E+00
Nuclide 016:
Zr-95
  9
 0.5527872000E+07
 0.9500E+02
 0.4489E+05
Nb-95m   0.7000E-02
Nb-95    0.9900E+00
none     0.0000E+00
Nuclide 017:
Zr-97
  9
```

```
 0.6084000000E+05
 0.9700E+02
 0.4657E+05
Nb-97m   0.9500E+00
Nb-97    0.5300E-01
none     0.0000E+00
Nuclide 018:
Nb-95
  9
 0.3036960000E+07
 0.9500E+02
 0.4512E+05
none     0.0000E+00
none     0.0000E+00
none     0.0000E+00
Nuclide 019:
Mo-99
  7
 0.2376000000E+06
 0.9900E+02
 0.5078E+05
Tc-99m   0.8800E+00
Tc-99    0.1200E+00
none     0.0000E+00
Nuclide 020:
Tc-99m
  7
 0.2167200000E+05
 0.9900E+02
 0.4447E+05
Tc-99    0.1000E+01
none     0.0000E+00
none     0.0000E+00
Nuclide 021:
Ru-103
  7
 0.3393792000E+07
 0.1030E+03
 0.4202E+05
Rh-103m  0.1000E+01
none     0.0000E+00
none     0.0000E+00
Nuclide 022:
Ru-105
  7
 0.1598400000E+05
 0.1050E+03
 0.2908E+05
Rh-105   0.1000E+01
none     0.0000E+00
none     0.0000E+00
Nuclide 023:
Ru-106
  7
 0.3181248000E+08
 0.1060E+03
 0.1730E+05
Rh-106   0.1000E+01
none     0.0000E+00
none     0.0000E+00
Nuclide 024:
Rh-105
  7
 0.1272960000E+06
 0.1050E+03
 0.2752E+05
none     0.0000E+00
none     0.0000E+00
none     0.0000E+00
Nuclide 025:
Sb-127
  4
 0.3326400000E+06
 0.1270E+03
 0.2896E+04
Te-127m  0.1800E+00
Te-127   0.8200E+00
none     0.0000E+00
Nuclide 026:
Sb-129
  4
 0.1555200000E+05
 0.1290E+03
 0.8638E+04
Te-129m  0.2200E+00
Te-129   0.7700E+00
none     0.0000E+00
Nuclide 027:
Te-127
  4
 0.3366000000E+05
 0.1270E+03
 0.2873E+04
none     0.0000E+00
none     0.0000E+00
none     0.0000E+00
```

```
Nuclide 028:
Te-127m
  4
 0.9417600000E+07
 0.1270E+03
 0.3855E+03
Te-127   0.9800E+00
none     0.0000E+00
none     0.0000E+00
Nuclide 029:
Te-129
  4
 0.4176000000E+04
 0.1290E+03
 0.8501E+04
I-129    0.1000E+01
none     0.0000E+00
none     0.0000E+00
Nuclide 030:
Te-129m
  4
 0.2903040000E+07
 0.1290E+03
 0.1267E+04
Te-129   0.6500E+00
I-129    0.3500E+00
none     0.0000E+00
Nuclide 031:
Te-131m
  4
 0.1080000000E+06
 0.1310E+03
 0.3869E+04
Te-131   0.2200E+00
I-131    0.7800E+00
none     0.0000E+00
Nuclide 032:
Te-132
  4
 0.2815200000E+06
 0.1320E+03
 0.3821E+05
I-132    0.1000E+01
none     0.0000E+00
none     0.0000E+00
Nuclide 033:
I-131
  2
 0.6946560000E+06
 0.1310E+03
 0.2687E+05
Xe-131m  0.1100E-01
none     0.0000E+00
none     0.0000E+00
Nuclide 034:
I-132
  2
 0.8280000000E+04
 0.1320E+03
 0.3881E+05
none     0.0000E+00
none     0.0000E+00
none     0.0000E+00
Nuclide 035:
I-133
  2
 0.7488000000E+05
 0.1330E+03
 0.5556E+05
Xe-133m  0.2900E-01
Xe-133   0.9700E+00
none     0.0000E+00
Nuclide 036:
I-134
  2
 0.3156000000E+04
 0.1340E+03
 0.6165E+05
none     0.0000E+00
none     0.0000E+00
none     0.0000E+00
Nuclide 037:
I-135
  2
 0.2379600000E+05
 0.1350E+03
 0.5192E+05
Xe-135m  0.1500E+00
Xe-135   0.8500E+00
none     0.0000E+00
Nuclide 038:
Xe-133
  1
 0.4531680000E+06
 0.1330E+03
 0.5491E+05
```

```
none     0.0000E+00
none     0.0000E+00
none     0.0000E+00
Nuclide 039:
Xe-135
  1
 0.3272400000E+05
 0.1350E+03
 0.2228E+05
Cs-135   0.1000E+01
none     0.0000E+00
none     0.0000E+00
Nuclide 040:
Cs-134
  3
 0.6507177120E+08
 0.1340E+03
 0.7280E+04
none     0.0000E+00
none     0.0000E+00
none     0.0000E+00
Nuclide 041:
Cs-136
  3
 0.1131840000E+07
 0.1360E+03
 0.2027E+04
none     0.0000E+00
none     0.0000E+00
none     0.0000E+00
Nuclide 042:
Cs-137
  3
 0.9467280000E+09
 0.1370E+03
 0.4538E+04
Ba-137m  0.9500E+00
none     0.0000E+00
none     0.0000E+00
Nuclide 043:
Ba-139
  6
 0.4962000000E+04
 0.1390E+03
 0.5084E+05
none     0.0000E+00
none     0.0000E+00
none     0.0000E+00
Nuclide 044:
Ba-140
  6
 0.1100736000E+07
 0.1400E+03
 0.4896E+05
La-140   0.1000E+01
none     0.0000E+00
none     0.0000E+00
Nuclide 045:
La-140
  9
 0.1449792000E+06
 0.1400E+03
 0.5019E+05
none     0.0000E+00
none     0.0000E+00
none     0.0000E+00
Nuclide 046:
La-141
  9
 0.1414800000E+05
 0.1410E+03
 0.4640E+05
Ce-141   0.1000E+01
none     0.0000E+00
none     0.0000E+00
Nuclide 047:
La-142
  9
 0.5550000000E+04
 0.1420E+03
 0.4532E+05
none     0.0000E+00
none     0.0000E+00
none     0.0000E+00
Nuclide 048:
Ce-141
  8
 0.2808086400E+07
 0.1410E+03
 0.4492E+05
none     0.0000E+00
none     0.0000E+00
none     0.0000E+00
Nuclide 049:
Ce-143
  8
```

```
 0.1188000000E+06
 0.1430E+03
 0.4427E+05
Pr-143   0.1000E+01
none     0.0000E+00
none     0.0000E+00
Nuclide 050:
Ce-144
  8
 0.2456352000E+08
 0.1440E+03
 0.3596E+05
Pr-144m  0.1800E-01
Pr-144   0.9800E+00
none     0.0000E+00
Nuclide 051:
Pr-143
  9
 0.1171584000E+07
 0.1430E+03
 0.4293E+05
none     0.0000E+00
none     0.0000E+00
none     0.0000E+00
Nuclide 052:
Nd-147
  9
 0.9486720000E+06
 0.1470E+03
 0.1838E+05
Pm-147   0.1000E+01
none     0.0000E+00
none     0.0000E+00
Nuclide 053:
Np-239
  8
 0.2034720000E+06
 0.2390E+03
 0.5397E+06
Pu-239   0.1000E+01
none     0.0000E+00
none     0.0000E+00
Nuclide 054:
Pu-238
  8
 0.2768863824E+10
 0.2380E+03
 0.1796E+03
U-234    0.1000E+01
none     0.0000E+00
none     0.0000E+00
Nuclide 055:
Pu-239
  8
 0.7594336440E+12
 0.2390E+03
 0.1200E+02
U-235    0.1000E+01
none     0.0000E+00
none     0.0000E+00
Nuclide 056:
Pu-240
  8
 0.2062920312E+12
 0.2400E+03
 0.1288E+02
U-236    0.1000E+01
none     0.0000E+00
none     0.0000E+00
Nuclide 057:
Pu-241
  8
 0.4544294400E+09
 0.2410E+03
 0.6182E+04
U-237    0.2400E-04
Am-241   0.1000E+01
none     0.0000E+00
Nuclide 058:
Am-241
  9
 0.1363919472E+11
 0.2410E+03
 0.9528E+01
Np-237   0.1000E+01
none     0.0000E+00
none     0.0000E+00
Nuclide 059:
Cm-242
  9
 0.1406592000E+08
 0.2420E+03
 0.2388E+04
Pu-238   0.1000E+01
none     0.0000E+00
none     0.0000E+00
```

```
Nuclide 060:
Cm-244
  9
 0.5715081360E+09
 0.2440E+03
 0.2602E+03
Pu-240   0.1000E+01
none     0.0000E+00
none     0.0000E+00
End of Nuclear Inventory File
```

8.3 PSF files

8.3.1 MSL A

```
Radtrad 3.03 4/15/2001
 Mark1, MSIV Leakage Model, RLB, condenser
 Nuclide Inventory File:
 c:\program files\radtrad303\defaults\pbs_def.nif
 Plant Power Level:
  3.5280E+03
 Compartments:
  10
 Compartment 1:
 Steam Dome 1
   3
  3.7116E+03
   0
   0
   0
   1
   0
 Compartment 2:
 Void/Un-modeled Line
   3
  1.0000E+05
   0
   0
   0
   0
   0
 Compartment 3:
 Environment
   2
  0.0000E+00
   0
   0
   0
   0
   0
 Compartment 4:
 Control Room
   1
  1.7600E+05
   0
   0
   0
   0
   0
 Compartment 5:
 MSL-A Volume 1
   3
  3.2800E+02
   0
   0
   0
   1
   0
 Compartment 6:
 MSL-A Volume 2
   3
  7.5000E+01
   0
   0
   0
   1
   0
 Compartment 7:
 MSL-A Volume 3
   3
  9.1700E+02
   0
   0
   0
   1
   0
 Compartment 8:
 Steam Dome 2
   3
  3.7116E+03
   0
```

```
  0
  0
  1
  0
Compartment 9:
Steam Dome 3
  3
 3.7116E+03
  0
  0
  0
  1
  0
Compartment 10:
Condenser
  3
 1.4700E+05
  0
  0
  0
  1
  0
Pathways:
 16
Pathway 1:
Steam Dome 1 to Void (Cont)
  1
  2
  4
Pathway 2:
Filtered Intake to Control Room
  3
  4
  2
Pathway 3:
Unfiltered Inleakage to Control Room
  3
  4
  2
Pathway 4:
Control Room Exhaust to Environment
  4
  3
  2
Pathway 5:
Steam Dome 1 to MSL-A Volume 1
  1
  5
  1
Pathway 6:
Steam Dome 2 to MSL-A Volume 1
  8
  5
  1
Pathway 7:
Steam Dome 3 to MSL-A Volume 1
  9
  5
  1
Pathway 8:
MSL-A Volume 1 to MSL-A Volume 2
  5
  6
  1
Pathway 9:
MSL-A Volume 2 to MSL-A Volume 3
  6
  7
  2
Pathway 10:
MSL-A Volume 3 to Condenser
  7
 10
  2
Pathway 11:
Steam Dome 2 to Void (Cont)
  8
  2
  4
Pathway 12:
Steam Dome 3 to Void (Cont)
  9
  2
  4
Pathway 13:
Steam Dome 1 to Void (Line)
  1
  2
  1
Pathway 14:
Steam Dome 2 to Void (Line)
  8
  2
  1
Pathway 15:
Steam Dome 3 to Void (Line)
```

```
  9
  2
  1
Pathway 16:
Condenser to Environment
 10
  3
  1
End of Plant Model File
Scenario Description Name:

Plant Model Filename:

Source Term:
  3
  1   3.5575E-01
  8   1.5500E-01
  9   2.3343E-02
c:\program files\radtrad303\defaults\fgr11&12.inp
c:\program files\radtrad303\defaults\bwr_dba.rft
  0.0000E+00
  1
  9.5000E-01   4.8500E-02   1.5000E-03   1.0000E+00
Overlying Pool:
  0
  0.0000E+00
  0
  0
  0
  0
Compartments:
 10
Compartment 1:
  1
  1
  0
  0
  0
  0
  0
  3
  3
  1.0000E+01
  1
  1
  0.0000E+00   0.0000E+00
Compartment 2:
  0
  1
  0
  0
  0
  0
  0
  0
  0
Compartment 3:
  1
  1
  0
  0
  0
  0
  0
  0
  0
Compartment 4:
  1
  1
  0
  0
  0
  0
  0
  0
  0
Compartment 5:
  0
  1
  0
  0
  0
  0
  0
  1
  4
  0.0000E+00   0.0000E+00
  2.0000E+00   0.0000E+00
  1.2000E+01   0.0000E+00
  7.2000E+02   0.0000E+00
  0
Compartment 6:
  0
  1
  0
  0
```

```
  0
  0
  0
  1
  4
  0.0000E+00   2.9000E+00
  2.0000E+00   2.4000E+00
  1.2000E+01   2.0000E+00
  7.2000E+02   0.0000E+00
  0
Compartment 7:
  0
  1
  0
  0
  0
  0
  0
  1
  4
  0.0000E+00   1.3000E+00
  2.0000E+00   1.3000E+00
  1.2000E+01   1.0000E+00
  7.2000E+02   0.0000E+00
  0
Compartment 8:
  1
  1
  0
  0
  0
  0
  0
  3
  3
  1.0000E+01
  1
  1
  0.0000E+00   0.0000E+00
Compartment 9:
  1
  1
  0
  0
  0
  0
  0
  3
  3
  1.0000E+01
  1
  1
  0.0000E+00   0.0000E+00
Compartment 10:
  0
  1
  0
  0
  0
  0
  0
  1
  4
  0.0000E+00   2.0000E-02
  2.0000E+00   1.8000E-02
  1.2000E+01   1.5000E-02
  7.2000E+02   0.0000E+00
  0
Pathways:
 16
Pathway 1:
  0
  0
  0
  0
  0
  0
  0
  0
  0
  0
  1
  4
  0.0000E+00   0.0000E+00
  3.3300E-02   7.0000E-01
  3.8000E+01   3.5000E-01
  7.2000E+02   0.0000E+00
  0
Pathway 2:
  0
  0
  0
  0
  0
  1
  4
```

```
   0.0000E+00      1.8500E+04      0.0000E+00      0.0000E+00
0.0000E+00
   3.3300E-02      1.8500E+04      0.0000E+00      0.0000E+00
0.0000E+00
   5.0000E-01      2.7000E+03      9.8000E+01      8.9000E+01
8.9000E+01
   7.2000E+02      0.0000E+00      0.0000E+00      0.0000E+00
0.0000E+00
   0
   0
   0
   0
   0
   0
 Pathway 3:
   0
   0
   0
   0
   0
   1
   4
   0.0000E+00      0.0000E+00      0.0000E+00      0.0000E+00
0.0000E+00
   3.3300E-02      0.0000E+00      0.0000E+00      0.0000E+00
0.0000E+00
   5.0000E-01      5.0000E+02      0.0000E+00      0.0000E+00
0.0000E+00
   7.2000E+02      0.0000E+00      0.0000E+00      0.0000E+00
0.0000E+00
   0
   0
   0
   0
   0
   0
 Pathway 4:
   0
   0
   0
   0
   0
   1
   4
   0.0000E+00      1.8500E+04      0.0000E+00      0.0000E+00
0.0000E+00
   3.3300E-02      1.8500E+04      0.0000E+00      0.0000E+00
0.0000E+00
   5.0000E-01      3.2000E+03      1.0000E+02      1.0000E+02
1.0000E+02
   7.2000E+02      0.0000E+00      0.0000E+00      0.0000E+00
0.0000E+00
   0
   0
   0
   0
   0
   0
 Pathway 5:
   0
   0
   1
   5
   0.0000E+00   1.0000E+00   0.0000E+00
   3.3300E-02   1.0000E+00   1.8987E+00
   5.0000E-01   1.0000E+00   0.0000E+00
   3.8000E+01   1.0000E+00   0.0000E+00
   7.2000E+02   1.0000E+00   0.0000E+00
   1
   5
   0.0000E+00   1.0000E+00   0.0000E+00
   3.3300E-02   1.0000E+00   1.8987E+00
   5.0000E-01   1.0000E+00   0.0000E+00
   3.8000E+01   1.0000E+00   0.0000E+00
   7.2000E+02   1.0000E+00   0.0000E+00
   1
   5
   0.0000E+00   1.0000E+00   0.0000E+00
   3.3300E-02   1.0000E+00   1.8987E+00
   5.0000E-01   1.0000E+00   0.0000E+00
   3.8000E+01   1.0000E+00   0.0000E+00
   7.2000E+02   1.0000E+00   0.0000E+00
   0
   0
   0
   0
   0
   0
   0
 Pathway 6:
   0
   0
   1
   5
   0.0000E+00   1.0000E+00   0.0000E+00
   5.0000E-01   1.0000E+00   1.8987E+00
```

```
  1.0000E+00   1.0000E+00   0.0000E+00
  3.8000E+01   1.0000E+00   0.0000E+00
  7.2000E+02   1.0000E+00   0.0000E+00
  1
  5
  0.0000E+00   1.0000E+00   0.0000E+00
  5.0000E-01   1.0000E+00   1.8987E+00
  1.0000E+00   1.0000E+00   0.0000E+00
  3.8000E+01   1.0000E+00   0.0000E+00
  7.2000E+02   1.0000E+00   0.0000E+00
  1
  5
  0.0000E+00   1.0000E+00   0.0000E+00
  5.0000E-01   1.0000E+00   1.8987E+00
  1.0000E+00   1.0000E+00   0.0000E+00
  3.8000E+01   1.0000E+00   0.0000E+00
  7.2000E+02   1.0000E+00   0.0000E+00
  0
  0
  0
  0
  0
  0
  0
Pathway 7:
  0
  0
  1
  6
  0.0000E+00   1.0000E+00   0.0000E+00
  3.3300E-02   1.0000E+00   0.0000E+00
  1.0000E+00   1.0000E+00   1.8987E+00
  2.0000E+00   1.0000E+00   1.0530E+00
  3.8000E+01   1.0000E+00   5.2648E-01
  7.2000E+02   1.0000E+00   0.0000E+00
  1
  6
  0.0000E+00   1.0000E+00   0.0000E+00
  3.3300E-02   1.0000E+00   0.0000E+00
  1.0000E+00   1.0000E+00   1.8987E+00
  2.0000E+00   1.0000E+00   1.0530E+00
  3.8000E+01   1.0000E+00   5.2648E-01
  7.2000E+02   1.0000E+00   0.0000E+00
  1
  6
  0.0000E+00   1.0000E+00   0.0000E+00
  3.3300E-02   1.0000E+00   0.0000E+00
  1.0000E+00   1.0000E+00   1.8987E+00
  2.0000E+00   1.0000E+00   1.0530E+00
  3.8000E+01   1.0000E+00   5.2648E-01
  7.2000E+02   1.0000E+00   0.0000E+00
  0
  0
  0
  0
  0
  0
  0
Pathway 8:
  0
  0
  1
  5
  0.0000E+00   1.0000E+00   0.0000E+00
  3.3300E-02   1.0000E+00   1.8987E+00
  2.0000E+00   1.0000E+00   1.8987E+00
  3.8000E+01   1.0000E+00   9.4935E-01
  7.2000E+02   1.0000E+00   0.0000E+00
  1
  5
  0.0000E+00   1.0000E+00   0.0000E+00
  3.3300E-02   1.0000E+00   1.8987E+00
  2.0000E+00   1.0000E+00   1.8987E+00
  3.8000E+01   1.0000E+00   9.4935E-01
  7.2000E+02   1.0000E+00   0.0000E+00
  1
  5
  0.0000E+00   1.0000E+00   0.0000E+00
  3.3300E-02   1.0000E+00   1.8987E+00
  2.0000E+00   1.0000E+00   1.8987E+00
  3.8000E+01   1.0000E+00   9.4935E-01
  7.2000E+02   1.0000E+00   0.0000E+00
  0
  0
  0
  0
  0
  0
  0
Pathway 9:
  0
  0
  0
  0
  0
  1
```

```
   6
   0.0000E+00       0.0000E+00       0.0000E+00       0.0000E+00
0.0000E+00
   3.3300E-02       2.1946E+00       0.0000E+00       5.0000E+01
0.0000E+00
   2.0000E+00       2.1946E+00       0.0000E+00       5.0000E+01
0.0000E+00
   3.8000E+01       1.0973E+00       0.0000E+00       5.0000E+01
0.0000E+00
   9.6000E+01       1.0973E+00       0.0000E+00       0.0000E+00
0.0000E+00
   7.2000E+02       0.0000E+00       0.0000E+00       0.0000E+00
0.0000E+00
   0
   0
   0
   0
   0
   0
 Pathway 10:
   0
   0
   0
   0
   0
   1
   6
   0.0000E+00       0.0000E+00       0.0000E+00       0.0000E+00
0.0000E+00
   3.3300E-02       1.1010E+01       0.0000E+00       5.0000E+01
0.0000E+00
   2.0000E+00       1.1010E+01       0.0000E+00       5.0000E+01
0.0000E+00
   3.8000E+01       5.5050E+00       0.0000E+00       5.0000E-01
0.0000E+00
   9.6000E+01       5.5050E+00       0.0000E+00       0.0000E+00
0.0000E+00
   7.2000E+02       0.0000E+00       0.0000E+00       0.0000E+00
0.0000E+00
   0
   0
   0
   0
   0
   0
 Pathway 11:
   0
   0
```

```
   0
   0
   0
   0
   0
   0
   0
   0
   1
   4
   0.0000E+00   0.0000E+00
   3.3300E-02   7.0000E-01
   3.8000E+01   3.5000E-01
   7.2000E+02   0.0000E+00
   0
 Pathway 12:
   0
   0
   0
   0
   0
   0
   0
   0
   0
   0
   1
   4
   0.0000E+00   0.0000E+00
   3.3300E-02   7.0000E-01
   3.8000E+01   3.5000E-01
   7.2000E+02   0.0000E+00
   0
 Pathway 13:
   0
   0
   1
   5
   0.0000E+00   1.0000E+00   0.0000E+00
   3.3300E-02   1.0000E+00   1.4356E+00
   5.0000E-01   1.0000E+00   0.0000E+00
   3.8000E+01   1.0000E+00   0.0000E+00
   7.2000E+02   1.0000E+00   0.0000E+00
   1
   5
   0.0000E+00   1.0000E+00   0.0000E+00
   3.3300E-02   1.0000E+00   1.4356E+00
   5.0000E-01   1.0000E+00   0.0000E+00
```

```
  3.8000E+01   1.0000E+00   0.0000E+00
  7.2000E+02   1.0000E+00   0.0000E+00
  1
  5
  0.0000E+00   1.0000E+00   0.0000E+00
  3.3300E-02   1.0000E+00   1.4356E+00
  5.0000E-01   1.0000E+00   0.0000E+00
  3.8000E+01   1.0000E+00   0.0000E+00
  7.2000E+02   1.0000E+00   0.0000E+00
  0
  0
  0
  0
  0
  0
  0
Pathway 14:
  0
  0
  1
  5
  0.0000E+00   1.0000E+00   0.0000E+00
  5.0000E-01   1.0000E+00   1.4356E+00
  1.0000E+00   1.0000E+00   0.0000E+00
  3.8000E+01   1.0000E+00   0.0000E+00
  7.2000E+02   1.0000E+00   0.0000E+00
  1
  5
  0.0000E+00   1.0000E+00   0.0000E+00
  5.0000E-01   1.0000E+00   1.4356E+00
  1.0000E+00   1.0000E+00   0.0000E+00
  3.8000E+01   1.0000E+00   0.0000E+00
  7.2000E+02   1.0000E+00   0.0000E+00
  1
  5
  0.0000E+00   1.0000E+00   0.0000E+00
  5.0000E-01   1.0000E+00   1.4356E+00
  1.0000E+00   1.0000E+00   0.0000E+00
  3.8000E+01   1.0000E+00   0.0000E+00
  7.2000E+02   1.0000E+00   0.0000E+00
  0
  0
  0
  0
  0
  0
  0
Pathway 15:
  0
  0
  1
  6
  0.0000E+00   1.0000E+00   0.0000E+00
  3.3300E-02   1.0000E+00   0.0000E+00
  1.0000E+00   1.0000E+00   1.4356E+00
  2.0000E+00   1.0000E+00   7.9614E-01
  3.8000E+01   1.0000E+00   3.9807E-01
  7.2000E+02   1.0000E+00   0.0000E+00
  1
  6
  0.0000E+00   1.0000E+00   0.0000E+00
  3.3300E-02   1.0000E+00   0.0000E+00
  1.0000E+00   1.0000E+00   1.4356E+00
  2.0000E+00   1.0000E+00   7.9614E-01
  3.8000E+01   1.0000E+00   3.9807E-01
  7.2000E+02   1.0000E+00   0.0000E+00
  1
  6
  0.0000E+00   1.0000E+00   0.0000E+00
  3.3300E-02   1.0000E+00   0.0000E+00
  1.0000E+00   1.0000E+00   1.4356E+00
  2.0000E+00   1.0000E+00   7.9614E-01
  3.8000E+01   1.0000E+00   3.9807E-01
  7.2000E+02   1.0000E+00   0.0000E+00
  0
  0
  0
  0
  0
  0
  0
Pathway 16:
  0
  0
  1
  5
  0.0000E+00   1.0000E+00   0.0000E+00
  3.3300E-02   1.0000E+00   1.1010E+01
  2.0000E+00   1.0000E+00   1.1010E+01
  3.8000E+01   1.0000E+00   5.5050E+00
  7.2000E+02   1.0000E+00   0.0000E+00
  1
  5
  0.0000E+00   1.0000E+00   0.0000E+00
  3.3300E-02   1.0000E+00   1.1010E+01
  2.0000E+00   1.0000E+00   1.1010E+01
```

```
  3.8000E+01   1.0000E+00   5.5050E+00
  7.2000E+02   1.0000E+00   0.0000E+00
  1
  5
  0.0000E+00   1.0000E+00   0.0000E+00
  3.3300E-02   1.0000E+00   1.1010E+01
  2.0000E+00   1.0000E+00   1.1010E+01
  3.8000E+01   1.0000E+00   5.5050E+00
  7.2000E+02   1.0000E+00   0.0000E+00
  0
  0
  0
  0
  0
  0
  0
Dose Locations:
  3
Location 1:
Exclusion Area Boundary
  3
  1
  2
  0.0000E+00   4.2500E-04
  7.2000E+02   0.0000E+00
  1
  2
  0.0000E+00   3.5000E-04
  7.2000E+02   0.0000E+00
  0
Location 2:
Low Population Zone
  3
  1
  6
  0.0000E+00   4.8100E-05
  2.0000E+00   2.0800E-05
  8.0000E+00   1.3700E-05
  2.4000E+01   5.4900E-06
  9.6000E+01   1.4900E-06
  7.2000E+02   0.0000E+00
  1
  4
  0.0000E+00   3.5000E-04
  8.0000E+00   1.8000E-04
  2.4000E+01   2.3000E-04
  7.2000E+02   0.0000E+00
  0
Location 3:
Control Room
  4
  0
  1
  2
  0.0000E+00   3.5000E-04
  7.2000E+02   0.0000E+00
  1
  4
  0.0000E+00   1.0000E+00
  2.4000E+01   6.0000E-01
  9.6000E+01   4.0000E-01
  7.2000E+02   0.0000E+00
Effective Volume Location:
  1
  6
  0.0000E+00   1.1800E-03
  2.0000E+00   9.0800E-04
  8.0000E+00   4.1400E-04
  2.4000E+01   2.9000E-04
  9.6000E+01   2.2600E-04
  7.2000E+02   0.0000E+00
Simulation Parameters:
  6
  0.0000E+00   1.0000E-01
  2.0000E+00   5.0000E-01
  8.0000E+00   1.0000E+00
  2.4000E+01   2.0000E+00
  9.6000E+01   8.0000E+00
  7.2000E+02   0.0000E+00
Output Filename:
C:\Program Files\radtrad303\MSIV Models\Mk1_Cond_final_a.o0
  1
  1
  1
  0
  0
End of Scenario File
```

8.3.2 MSL B

```
Radtrad 3.03 4/15/2001
 Mark1, MSIV Leakage Model, RLB, condenser
 Nuclide Inventory File:
```

```
c:\program files\radtrad303\defaults\pbs_def.nif
Plant Power Level:
 3.5280E+03
Compartments:
 10
Compartment 1:
Steam Dome 1
  3
 3.7116E+03
  0
  0
  0
  1
  0
Compartment 2:
Void/Un-modeled Line
  3
 1.0000E+05
  0
  0
  0
  0
  0
Compartment 3:
Environment
  2
 0.0000E+00
  0
  0
  0
  0
  0
Compartment 4:
Control Room
  1
 1.7600E+05
  0
  0
  0
  0
  0
Compartment 5:
MSL-B Volume 1
  3
 3.0300E+02
  0
  0
  0
  1
  0
Compartment 6:
MSL-B Volume 2
  3
 6.9000E+01
  0
  0
  0
  1
  0
Compartment 7:
MSL-B Volume 3
  3
 9.3900E+02
  0
  0
  0
  1
  0
Compartment 8:
Steam Dome 2
  3
 3.7116E+03
  0
  0
  0
  1
  0
Compartment 9:
Steam Dome 3
  3
 3.7116E+03
  0
  0
  0
  1
  0
Compartment 10:
Condenser
  3
 1.4700E+05
  0
  0
  0
  1
  0
Pathways:
```

```
 16
Pathway 1:
Steam Dome 1 to Void (Cont)
  1
  2
  4
Pathway 2:
Filtered Intake to Control Room
  3
  4
  2
Pathway 3:
Unfiltered Inleakage to Control Room
  3
  4
  2
Pathway 4:
Control Room Exhaust to Environment
  4
  3
  2
Pathway 5:
Steam Dome 1 to MSL-B Volume 1
  1
  5
  1
Pathway 6:
Steam Dome 2 to MSL-B Volume 1
  8
  5
  1
Pathway 7:
Steam Dome 3 to MSL-B Volume 1
  9
  5
  1
Pathway 8:
MSL-B Volume 1 to MSL-B Volume 2
  5
  6
  1
Pathway 9:
MSL-B Volume 2 to MSL-B Volume 3
  6
  7
  2
Pathway 10:
MSL-B Volume 3 to Condenser
  7
 10
  2
Pathway 11:
Steam Dome 2 to Void (Cont)
  8
  2
  4
Pathway 12:
Steam Dome 3 to Void (Cont)
  9
  2
  4
Pathway 13:
Steam Dome 1 to Void (Line)
  1
  2
  1
Pathway 14:
Steam Dome 2 to Void (Line)
  8
  2
  1
Pathway 15:
Steam Dome 3 to Void (Line)
  9
  2
  1
Pathway 16:
Condenser to Environment
 10
  3
  1
End of Plant Model File
Scenario Description Name:

Plant Model Filename:

Source Term:
  3
  1   3.5575E-01
  8   1.5500E-01
  9   2.3343E-02
c:\program files\radtrad303\defaults\fgr11&12.inp
c:\program files\radtrad303\defaults\bwr_dba.rft
  0.0000E+00
  1
  9.5000E-01   4.8500E-02   1.5000E-03   1.0000E+00
```

```
Overlying Pool:
  0
  0.0000E+00
  0
  0
  0
  0
Compartments:
 10
Compartment 1:
  1
  1
  0
  0
  0
  0
  0
  3
  3
  1.0000E+01
  1
  1
  0.0000E+00   0.0000E+00
Compartment 2:
  0
  1
  0
  0
  0
  0
  0
  0
  0
Compartment 3:
  1
  1
  0
  0
  0
  0
  0
  0
  0
Compartment 4:
  1
  1
  0
  0
  0
  0
  0
  0
  0
Compartment 5:
  0
  1
  0
  0
  0
  0
  0
  1
  4
  0.0000E+00   0.0000E+00
  2.0000E+00   0.0000E+00
  1.2000E+01   0.0000E+00
  7.2000E+02   0.0000E+00
  0
Compartment 6:
  0
  1
  0
  0
  0
  0
  0
  1
  4
  0.0000E+00   2.9000E+00
  2.0000E+00   2.4000E+00
  1.2000E+01   2.0000E+00
  7.2000E+02   0.0000E+00
  0
Compartment 7:
  0
  1
  0
  0
  0
  0
  0
  1
  4
  0.0000E+00   1.3000E+00
  2.0000E+00   1.3000E+00
  1.2000E+01   1.0000E+00
```

```
  7.2000E+02   0.0000E+00
  0
Compartment 8:
  1
  1
  0
  0
  0
  0
  0
  3
  3
  1.0000E+01
  1
  1
  0.0000E+00   0.0000E+00
Compartment 9:
  1
  1
  0
  0
  0
  0
  0
  3
  3
  1.0000E+01
  1
  1
  0.0000E+00   0.0000E+00
Compartment 10:
  0
  1
  0
  0
  0
  0
  0
  1
  4
  0.0000E+00   2.0000E-02
  2.0000E+00   1.8000E-02
  1.2000E+01   1.5000E-02
  7.2000E+02   0.0000E+00
  0
Pathways:
 16
Pathway 1:
  0
  0
  0
  0
  0
  0
  0
  0
  0
  0
  1
  4
  0.0000E+00   0.0000E+00
  3.3300E-02   7.0000E-01
  3.8000E+01   3.5000E-01
  7.2000E+02   0.0000E+00
  0
 Pathway 2:
  0
  0
  0
  0
  0
  1
  4
  0.0000E+00       1.8500E+04       0.0000E+00       0.0000E+00
0.0000E+00
  3.3300E-02       1.8500E+04       0.0000E+00       0.0000E+00
0.0000E+00
  5.0000E-01       2.7000E+03       9.8000E+01       8.9000E+01
8.9000E+01
  7.2000E+02       0.0000E+00       0.0000E+00       0.0000E+00
0.0000E+00
  0
  0
  0
  0
  0
  0
 Pathway 3:
  0
  0
  0
  0
  0
  1
  4
```

```
   0.0000E+00      0.0000E+00      0.0000E+00      0.0000E+00
0.0000E+00
   3.3300E-02      0.0000E+00      0.0000E+00      0.0000E+00
0.0000E+00
   5.0000E-01      5.0000E+02      0.0000E+00      0.0000E+00
0.0000E+00
   7.2000E+02      0.0000E+00      0.0000E+00      0.0000E+00
0.0000E+00
   0
   0
   0
   0
   0
   0
 Pathway 4:
   0
   0
   0
   0
   0
   1
   4
   0.0000E+00      1.8500E+04      0.0000E+00      0.0000E+00
0.0000E+00
   3.3300E-02      1.8500E+04      0.0000E+00      0.0000E+00
0.0000E+00
   5.0000E-01      3.2000E+03      1.0000E+02      1.0000E+02
1.0000E+02
   7.2000E+02      0.0000E+00      0.0000E+00      0.0000E+00
0.0000E+00
   0
   0
   0
   0
   0
   0
 Pathway 5:
   0
   0
   1
   5
   0.0000E+00   1.0000E+00   0.0000E+00
   3.3300E-02   1.0000E+00   1.4356E+00
   5.0000E-01   1.0000E+00   0.0000E+00
   3.8000E+01   1.0000E+00   0.0000E+00
   7.2000E+02   1.0000E+00   0.0000E+00
   1
   5
   0.0000E+00   1.0000E+00   0.0000E+00
   3.3300E-02   1.0000E+00   1.4356E+00
   5.0000E-01   1.0000E+00   0.0000E+00
   3.8000E+01   1.0000E+00   0.0000E+00
   7.2000E+02   1.0000E+00   0.0000E+00
   1
   5
   0.0000E+00   1.0000E+00   0.0000E+00
   3.3300E-02   1.0000E+00   1.4356E+00
   5.0000E-01   1.0000E+00   0.0000E+00
   3.8000E+01   1.0000E+00   0.0000E+00
   7.2000E+02   1.0000E+00   0.0000E+00
   0
   0
   0
   0
   0
   0
   0
 Pathway 6:
   0
   0
   1
   5
   0.0000E+00   1.0000E+00   0.0000E+00
   5.0000E-01   1.0000E+00   1.4356E+00
   1.0000E+00   1.0000E+00   0.0000E+00
   3.8000E+01   1.0000E+00   0.0000E+00
   7.2000E+02   1.0000E+00   0.0000E+00
   1
   5
   0.0000E+00   1.0000E+00   0.0000E+00
   5.0000E-01   1.0000E+00   1.4356E+00
   1.0000E+00   1.0000E+00   0.0000E+00
   3.8000E+01   1.0000E+00   0.0000E+00
   7.2000E+02   1.0000E+00   0.0000E+00
   1
   5
   0.0000E+00   1.0000E+00   0.0000E+00
   5.0000E-01   1.0000E+00   1.4356E+00
   1.0000E+00   1.0000E+00   0.0000E+00
   3.8000E+01   1.0000E+00   0.0000E+00
   7.2000E+02   1.0000E+00   0.0000E+00
   0
   0
   0
   0
   0
```

```
  0
  0
Pathway 7:
  0
  0
  1
  6
  0.0000E+00   1.0000E+00   0.0000E+00
  3.3300E-02   1.0000E+00   0.0000E+00
  1.0000E+00   1.0000E+00   1.4356E+00
  2.0000E+00   1.0000E+00   7.9614E-01
  3.8000E+01   1.0000E+00   3.9807E-01
  7.2000E+02   1.0000E+00   0.0000E+00
  1
  6
  0.0000E+00   1.0000E+00   0.0000E+00
  3.3300E-02   1.0000E+00   0.0000E+00
  1.0000E+00   1.0000E+00   1.4356E+00
  2.0000E+00   1.0000E+00   7.9614E-01
  3.8000E+01   1.0000E+00   3.9807E-01
  7.2000E+02   1.0000E+00   0.0000E+00
  1
  6
  0.0000E+00   1.0000E+00   0.0000E+00
  3.3300E-02   1.0000E+00   0.0000E+00
  1.0000E+00   1.0000E+00   1.4356E+00
  2.0000E+00   1.0000E+00   7.9614E-01
  3.8000E+01   1.0000E+00   3.9807E-01
  7.2000E+02   1.0000E+00   0.0000E+00
  0
  0
  0
  0
  0
  0
  0
Pathway 8:
  0
  0
  1
  5
  0.0000E+00   1.0000E+00   0.0000E+00
  3.3300E-02   1.0000E+00   1.4356E+00
  2.0000E+00   1.0000E+00   1.4356E+00
  3.8000E+01   1.0000E+00   7.1780E-01
  7.2000E+02   1.0000E+00   0.0000E+00
  1
  5
  0.0000E+00   1.0000E+00   0.0000E+00
  3.3300E-02   1.0000E+00   1.4356E+00
  2.0000E+00   1.0000E+00   1.4356E+00
  3.8000E+01   1.0000E+00   7.1780E-01
  7.2000E+02   1.0000E+00   0.0000E+00
  1
  5
  0.0000E+00   1.0000E+00   0.0000E+00
  3.3300E-02   1.0000E+00   1.4356E+00
  2.0000E+00   1.0000E+00   1.4356E+00
  3.8000E+01   1.0000E+00   7.1780E-01
  7.2000E+02   1.0000E+00   0.0000E+00
  0
  0
  0
  0
  0
  0
  0
Pathway 9:
  0
  0
  0
  0
  0
  1
  6
  0.0000E+00       0.0000E+00       0.0000E+00       0.0000E+00
0.0000E+00
  3.3300E-02       1.6593E+00       0.0000E+00       5.0000E+01
0.0000E+00
  2.0000E+00       1.6593E+00       0.0000E+00       5.0000E+01
0.0000E+00
  3.8000E+01       8.2965E-01       0.0000E+00       5.0000E+01
0.0000E+00
  9.6000E+01       8.2965E-01       0.0000E+00       0.0000E+00
0.0000E+00
  7.2000E+02       0.0000E+00       0.0000E+00       0.0000E+00
0.0000E+00
  0
  0
  0
  0
  0
  0
Pathway 10:
  0
  0
```

```
   0
   0
   0
   1
   6
   0.0000E+00       0.0000E+00       0.0000E+00       0.0000E+00
0.0000E+00
   3.3300E-02       8.3244E+00       0.0000E+00       5.0000E+01
0.0000E+00
   2.0000E+00       8.3244E+00       0.0000E+00       5.0000E+01
0.0000E+00
   3.8000E+01       4.1622E+00       0.0000E+00       5.0000E-01
0.0000E+00
   9.6000E+01       4.1622E+00       0.0000E+00       0.0000E+00
0.0000E+00
   7.2000E+02       0.0000E+00       0.0000E+00       0.0000E+00
0.0000E+00
   0
   0
   0
   0
   0
   0
 Pathway 11:
   0
   0
   0
   0
   0
   0
   0
   0
   0
   0
   1
   4
   0.0000E+00   0.0000E+00
   3.3300E-02   7.0000E-01
   3.8000E+01   3.5000E-01
   7.2000E+02   0.0000E+00
   0
 Pathway 12:
   0
   0
   0
   0
   0
   0
   0
   0
   0
   0
   1
   4
   0.0000E+00   0.0000E+00
   3.3300E-02   7.0000E-01
   3.8000E+01   3.5000E-01
   7.2000E+02   0.0000E+00
   0
 Pathway 13:
   0
   0
   1
   5
   0.0000E+00   1.0000E+00   0.0000E+00
   3.3300E-02   1.0000E+00   1.8987E+00
   5.0000E-01   1.0000E+00   0.0000E+00
   3.8000E+01   1.0000E+00   0.0000E+00
   7.2000E+02   1.0000E+00   0.0000E+00
   1
   5
   0.0000E+00   1.0000E+00   0.0000E+00
   3.3300E-02   1.0000E+00   1.8987E+00
   5.0000E-01   1.0000E+00   0.0000E+00
   3.8000E+01   1.0000E+00   0.0000E+00
   7.2000E+02   1.0000E+00   0.0000E+00
   1
   5
   0.0000E+00   1.0000E+00   0.0000E+00
   3.3300E-02   1.0000E+00   1.8987E+00
   5.0000E-01   1.0000E+00   0.0000E+00
   3.8000E+01   1.0000E+00   0.0000E+00
   7.2000E+02   1.0000E+00   0.0000E+00
   0
   0
   0
   0
   0
   0
   0
 Pathway 14:
   0
   0
   1
   5
   0.0000E+00   1.0000E+00   0.0000E+00
```

```
  5.0000E-01   1.0000E+00   1.8987E+00
  1.0000E+00   1.0000E+00   0.0000E+00
  3.8000E+01   1.0000E+00   0.0000E+00
  7.2000E+02   1.0000E+00   0.0000E+00
  1
  5
  0.0000E+00   1.0000E+00   0.0000E+00
  5.0000E-01   1.0000E+00   1.8987E+00
  1.0000E+00   1.0000E+00   0.0000E+00
  3.8000E+01   1.0000E+00   0.0000E+00
  7.2000E+02   1.0000E+00   0.0000E+00
  1
  5
  0.0000E+00   1.0000E+00   0.0000E+00
  5.0000E-01   1.0000E+00   1.8987E+00
  1.0000E+00   1.0000E+00   0.0000E+00
  3.8000E+01   1.0000E+00   0.0000E+00
  7.2000E+02   1.0000E+00   0.0000E+00
  0
  0
  0
  0
  0
  0
  0
Pathway 15:
  0
  0
  1
  6
  0.0000E+00   1.0000E+00   0.0000E+00
  3.3300E-02   1.0000E+00   0.0000E+00
  1.0000E+00   1.0000E+00   1.8987E+00
  2.0000E+00   1.0000E+00   1.0530E+00
  3.8000E+01   1.0000E+00   5.2648E-01
  7.2000E+02   1.0000E+00   0.0000E+00
  1
  6
  0.0000E+00   1.0000E+00   0.0000E+00
  3.3300E-02   1.0000E+00   0.0000E+00
  1.0000E+00   1.0000E+00   1.8987E+00
  2.0000E+00   1.0000E+00   1.0530E+00
  3.8000E+01   1.0000E+00   5.2648E-01
  7.2000E+02   1.0000E+00   0.0000E+00
  1
  6
  0.0000E+00   1.0000E+00   0.0000E+00
  3.3300E-02   1.0000E+00   0.0000E+00
  1.0000E+00   1.0000E+00   1.8987E+00
  2.0000E+00   1.0000E+00   1.0530E+00
  3.8000E+01   1.0000E+00   5.2648E-01
  7.2000E+02   1.0000E+00   0.0000E+00
  0
  0
  0
  0
  0
  0
  0
Pathway 16:
  0
  0
  1
  5
  0.0000E+00   1.0000E+00   0.0000E+00
  3.3300E-02   1.0000E+00   8.3244E+00
  2.0000E+00   1.0000E+00   8.3244E+00
  3.8000E+01   1.0000E+00   4.1622E+00
  7.2000E+02   1.0000E+00   0.0000E+00
  1
  5
  0.0000E+00   1.0000E+00   0.0000E+00
  3.3300E-02   1.0000E+00   8.3244E+00
  2.0000E+00   1.0000E+00   8.3244E+00
  3.8000E+01   1.0000E+00   4.1622E+00
  7.2000E+02   1.0000E+00   0.0000E+00
  1
  5
  0.0000E+00   1.0000E+00   0.0000E+00
  3.3300E-02   1.0000E+00   8.3244E+00
  2.0000E+00   1.0000E+00   8.3244E+00
  3.8000E+01   1.0000E+00   4.1622E+00
  7.2000E+02   1.0000E+00   0.0000E+00
  0
  0
  0
  0
  0
  0
  0
Dose Locations:
  3
Location 1:
Exclusion Area Boundary
  3
  1
```

```
  2
  0.0000E+00   4.2500E-04
  7.2000E+02   0.0000E+00
  1
  2
  0.0000E+00   3.5000E-04
  7.2000E+02   0.0000E+00
  0
Location 2:
Low Population Zone
  3
  1
  6
  0.0000E+00   4.8100E-05
  2.0000E+00   2.0800E-05
  8.0000E+00   1.3700E-05
  2.4000E+01   5.4900E-06
  9.6000E+01   1.4900E-06
  7.2000E+02   0.0000E+00
  1
  4
  0.0000E+00   3.5000E-04
  8.0000E+00   1.8000E-04
  2.4000E+01   2.3000E-04
  7.2000E+02   0.0000E+00
  0
Location 3:
Control Room
  4
  0
  1
  2
  0.0000E+00   3.5000E-04
  7.2000E+02   0.0000E+00
  1
  4
  0.0000E+00   1.0000E+00
  2.4000E+01   6.0000E-01
  9.6000E+01   4.0000E-01
  7.2000E+02   0.0000E+00
Effective Volume Location:
  1
  6
  0.0000E+00   1.1800E-03
  2.0000E+00   9.0800E-04
  8.0000E+00   4.1400E-04
  2.4000E+01   2.9000E-04
  9.6000E+01   2.2600E-04
  7.2000E+02   0.0000E+00
Simulation Parameters:
  6
  0.0000E+00   1.0000E-01
  2.0000E+00   5.0000E-01
  8.0000E+00   1.0000E+00
  2.4000E+01   2.0000E+00
  9.6000E+01   8.0000E+00
  7.2000E+02   0.0000E+00
Output Filename:
C:\Program Files\radtrad303\MSIV Models\Mk1_Cond_final_b.o0
  1
  1
  1
  0
  0
End of Scenario File
```

9 References

1. USNRC Regulatory Guide 1.183: Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors, July 2000.

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

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DISTRIBUTION

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Mail Stop 7 E18
U.S. Nuclear Regulatory Commission
Washington, DC 20555

1 Jay Lee
Mail Stop 7 E18
U.S. Nuclear Regulatory Commission
Washington, DC 20555

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Mail Stop 10 K8
U.S. Nuclear Regulatory Commission
Washington, DC 20555

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Mail Stop 10 C15
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[7590-01-P]

NUCLEAR REGULATORY COMMISSION

10 CFR Part 50

[Docket No. PRM-50-122; NRC-2020-0150]

Accident Source Term Methodologies and Corresponding Release Fractions

AGENCY: Nuclear Regulatory Commission.

ACTION: Petition for rulemaking; notice of docketing and request for comment.

SUMMARY: The U.S. Nuclear Regulatory Commission (NRC) has received a petition for rulemaking from Brian Magnuson dated May 31, 2020, requesting that the NRC revise its regulations to codify the source term methodologies and corresponding release fractions recommended in a report issued by Sandia National Laboratories. The petition was docketed by the NRC on June 18, 2020, and has been assigned Docket No. PRM-50-122. The NRC is examining the issues raised in PRM-50-122 to determine whether they should be considered in rulemaking. The NRC is requesting public comment on this petition at this time.

DATES: Submit comments by **[INSERT DATE 75 DAYS AFTER DATE OF PUBLICATION IN THE *FEDERAL REGISTER*]**. Comments received after this date will be considered if it is practical to do so, but the NRC is able to assure consideration only for comments received on or before this date.

ADDRESSES: You may submit comments by any of the following methods:

- **Federal Rulemaking Web Site:** Go to https://www.regulations.gov and search for Docket ID NRC-2020-0150. Address questions about NRC dockets to Carol Gallagher; telephone: 301-415-3463; e-mail: Carol.Gallagher@nrc.gov. For technical questions contact the individual listed in the FOR FURTHER INFORMATION CONTACT section of this document.

- **E-mail comments to:** Rulemaking.Comments@nrc.gov. If you do not receive an automatic e-mail reply confirming receipt, then contact us at 301-415-1677.

- **Mail comments to:** Secretary, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, ATTN: Rulemakings and Adjudications Staff.

For additional direction on obtaining information and submitting comments, see "Obtaining Information and Submitting Comments" in the SUPPLEMENTARY INFORMATION section of this document.

FOR FURTHER INFORMATION CONTACT: Juan Lopez, Office of Nuclear Material Safety and Safeguards, telephone: 301-415-2338, email: Juan.Lopez@nrc.gov, or Yanely Malave-Velez, Office of Nuclear Material Safety and Safeguards, telephone: 301-415-1519, email: Yanely.Malave-Velez@nrc.gov. Both are staff of the U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001.

SUPPLEMENTARY INFORMATION:

I. Obtaining Information and Submitting Comments

A. Obtaining Information

Please refer to Docket ID NRC-2020-0150 when contacting the NRC about the availability of information for this action. You may obtain publicly-available information related to this action by any of the following methods:

- **Federal Rulemaking Web Site:** Go to https://www.regulations.gov and search for Docket ID NRC-2020-0150.

- **NRC's Agencywide Documents Access and Management System (ADAMS):** You may obtain publicly-available documents online in the ADAMS Public Documents collection at https://www.nrc.gov/reading-rm/pdr.html. To begin the search, select "Begin Web-based ADAMS Search." For problems with ADAMS, please contact the NRC's Public Document Room (PDR) reference staff at 1-800-397-4209, 301-415-4737, or by e-mail to PDR.Resource@nrc.gov.

- The ADAMS accession number for each document referenced (if it is available in ADAMS) is provided the first time that it is mentioned in this document.

- **Attention:** The PDR where you may examine and order copies of public documents, is currently closed. You may submit your request to the PDR via e-mail at PDR.Resource@nrc.gov or call 1-800-397-4209 between 8:00 a.m. and 4:00 p.m. (EST), Monday through Friday, except Federal holidays.

B. Submitting Comments

Please include Docket ID NRC-2020-0150 in your comment submission.

The NRC cautions you not to include identifying or contact information that you do not want to be publicly disclosed in your comment submission. The NRC will post all comment submissions at https://www.regulations.gov as well as enter the comment submissions into ADAMS. The NRC does not routinely edit comment submissions to remove identifying or contact information.

If you are requesting or aggregating comments from other persons for submission to the NRC, then you should inform those persons not to include identifying or contact information that they do not want to be publicly disclosed in their comment submission. Your request should state that the NRC does not routinely edit comment submissions to remove such information before making the comment submissions available to the public or entering the comment into ADAMS.

II. The Petitioner and Petition

The petition for rulemaking (PRM) was filed by Brian Magnuson. The petition requests the NRC revise its regulations in § 50.67 of title 10 of the *Code of Federal Regulations* (10 CFR), “Accident source term,” to codify the source term methodologies and corresponding release fractions recommended in Sandia National Laboratories Report SAND2008-6601, “Analysis of Main Steam Isolation Valve Leakage in Design Basis Accidents Using MELCOR 1.8.6 and RADTRAD,” dated October 2008 (ADAMS Accession No. ML083180196). The petitioner states that the revision would eliminate inconsistences obtained from the use of different source term methodologies and release fractions and would provide the requisite means to ensure compliance with the underlying regulations. The petition may be found in ADAMS under Accession No. ML20170B161.

III. Discussion of the Petition

The petition states that much of the past and present source term methodologies, including release fractions, used by nuclear power plants to perform accident dose calculations are inaccurate and nonconservative. The petition requests that the NRC revise § 50.67 to codify the source term methodologies and recommendations of Sandia National Laboratories report SAND2008-6601 and update and finalize related NRC guidance, Draft Regulatory Guide DG-1199 (Proposed Revision 1 of Regulatory Guide 1.183), “Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors,” dated October 2009 (ADAMS Accession No. ML090960464).

The petition describes the current NRC guidance as "conceptually inaccurate" and "nonconservative" for calculations of radiological release doses, quoting from Sandia Report SAND2008-6601:

> …these findings conclude that the current regulatory guidelines permitting the use of the fission product concentration in the drywell atmosphere during the first two hours prior to assumed vessel reflood is non-conservative for the purposes of evaluating the dose resulting from MSIV leakage, in addition to being conceptually inaccurate.

The petition also states that, despite the NRC acknowledging the safety significance of accident source terms, the NRC has not yet approved Draft Regulatory Guide DG-1199. As a result, the petitioner believes accident doses have been undercalculated for over 25 years. The petition indicates this would account for the uncertainties that high burnup fuel pellets could be reduced to a powder form and dispersed outside of the fuel rod during clad failure accidents (with or without fuel melt), used by the Radiological Assessment System for Consequence Analysis (RASCAL) calculation described in NUREG-1940, "RASCAL 4: Description of Models and Methods," available online at https://www.nrc.gov/reading-rm/doc-collections/nuregs/staff/sr1940/.

IV. Conclusion

The NRC determined that the petition meets the requirements for docketing a petition for rulemaking under § 2.803, "Petition for rulemaking—NRC action." The NRC will examine the merits of the issues raised in PRM-50-122 and any comments received on this document to determine whether these issues should be considered in rulemaking.

Dated: August 7, 2020.

For the Nuclear Regulatory Commission.

/RA/

Annette L. Vietti-Cook,
Secretary of the Commission.

June 20, 2022

RE: Public comments on draft regulatory guide (DG), DG-1389, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors"

Dear NRC Staff:

I am opposed to DG-1389.

My prior public comments referenced SAND2008-6601 which determined the BWR MSIV source term methodologies provided in RG 1.183 (Revision 0) are "*non-conservative and conceptually inaccurate*" in 2008. Additionally, my prior comments expounded on SAND2008-6601 and identified other examples in which RG 1.183 methodologies violate the laws of physics. RG 1.183 allows nuclear power plants (NPPs) to ignore the laws of physics in accident dose calculations that are used to demonstrate compliance with nuclear safety regulations, including General Design Criterion-19 (Appendix A to 10 CFR Part 50). In other words, the errors in RG 1.183 financially benefit nuclear power plants at the expense of public safety.

It appears DG-1389 may correct a few of the technical errors in Revision 0 of RG 1.183; however, any corrections would be negated because it states:

> *"Revision 0 of RG 1.183 will continue to be available for use by licensees and applicants as a method acceptable to the NRC staff for demonstrating compliance with the regulations."*

RG 1.183 Revision 0 has a broad range of safety ramifications. Until the NRC has reconciled the errors that SAND2008-6601 identified and I reported in prior public comments, it seems imprudent of the NRC to claim it is an acceptable method for demonstrating compliance with regulations. In effect, the errors identified in RG 1.183 Revision 0 provide a means for nuclear power plants to ignore the laws of physics in accident dose calculations in order to feign compliance with federal nuclear safety regulations.

The (Beyond) Design-Basis Accident Contravention

I am opposed to using the DG-1389 term "*maximum hypothetical accident (MHA) loss-of-coolant accident (LOCA)."* An NRC Regulatory/Draft guide cannot legally be used to redefine *"the accident described in the applicable regulations."* For example, the applicability of Appendix A to Part 50, General Design Criterion—19 cannot be limited. Nevertheless, the apparent attempt drew attention to the most egregious contravention of RG 1.183 (and DG-1389).

To begin, the NRC acknowledged: "*In 1971 Appendix A, "General Design Criteria for Nuclear Power Plants," was added to 10 CFR Part 50. General Design Criterion 19 (GDC-19) specified that adequate protection shall be provided to permit access and occupancy of the control room for the duration of an accident without exceeding a radiation exposure of 5 rem whole body or its equivalent to any part of the body. <u>From its inception, GDC-19 became the limiting dose criteria in almost all radiological dose consequence analyses</u>."*

To be clear, GDC-19 was not "limiting" by the late 1970s. By then, the NRC discovered that BWR MSIV leakage was a significant contributor to control room operator doses. Despite this disturbing discovery, the NRC neglected to require nuclear power plants to add this contribution to their accident dose calculations. However, in 2000, the NRC suggested that some nuclear power plants might wish to add MSIV leakage dose contributions to their accident dose calculations if they wanted to reap the "*cost-beneficial licensing actions*" provided by RG 1.183.

Despite Sandia National Laboratories (SAND2008-6601) and my reports, the NRC continues to allow nuclear power plants to exploit the RG 1.183 errors for "*cost-beneficial licensing actions.*" The NRC allowed nuclear power plants to exploit the errors to (1) increase MSIV technical specification allowable leakage; (2) increase reactor thermal power (electrical generation); (3) increase fuel burnup times and; (4) extend (sometimes twice) the licensing life of old nuclear power plants—that have been violating GDC-19 since its inception.

Based on the timeline of NRC actions since 1971, it appears an underlying purpose of RG 1.183 is to evade the minimum design criteria set forth in Appendix A to Part 50, and purpose of 10 CFR 50.67 is to evade the more limiting requirements 10 CFR 100.11, *"Determination of Exclusion Area, Low Population Zone, and Population Center Distance."*

Despite the many significant design modifications that have been required in the last 50 years, nuclear power plants cannot be made <u>legally</u> safe. They were not designed based on the "maximum credible accident" as required by TID-14844. Instead, contrary to the admonishments of TID-14844 authors, they were mistakenly designed on its poor example.

The purpose of TID-14844:

> *It is the intent that this document to provide reference information and guidance on procedures and basic assumptions whereby certain factors pertinent to reactor siting as set forth in Title 10 Code of Federal Regulations Part 100 (10 CFR 100)[(1)] can be used to calculate distance requirements for reactor sites which are generally consistent with current siting practices.*
>
> *For any proposed reactor: the performance experience accumulated elsewhere; the engineering safeguards; the inherent stability and safety features; and the quality of*

> *design, materials, construction, management and operation are all important factors that must be included in the evaluation of the suitability of a site.*
>
> *For a particular site; size, topography, meteorology, hydrology, ease of warning and removing people in times of emergency, and thoroughness of plans and arrangements for minimizing injuries and interference with offsite activities, all enter an evaluation.*
>
> *Consideration of these as well as other aspects of hazards evaluation involves so many different situations and such complex technological problems that it would be quite impossible to anticipate and answer all questions that will arise.*
>
> *This technical document sets forth one method of computing distances and exposures, for one general class of reactors. In developing this example conservative assumptions have been intentionally selected.*
>
> *Designers of reactors are expected to examine all significant aspects of the hazards and safety problem they believe are appropriate to the particular situation with which they are dealing. In any case, the designer and/or applicant bears the responsibility for justifying all the assumptions and methods of calculation used in a hazards evaluation. The fact that aspects of the problem are not considered in the example set forth here, does not in any way relieve the designer and/or applicant of the responsibility for carefully examining, in his particular case, every significant facet of the hazards and safety problem.*

Despite the good intentions of the authors, their conservative example was nowhere near conservative. And despite their clear admonishment, nuclear plant designers and owners failed to "*carefully exam . . . every significant facet of the hazards and safety problem*—in part because nuclear technology was still an "infant technology" according to a 1972 AEC Environmental Statement.

As we now know, nuclear technology went through extreme growing pains. The fire at Browns Ferry (1975) and the reactor core meltdown at Three Mile Island (1979) exposed overwhelming design and operational deficiencies (NUREG-0737). The terrorist attack on September 11, 2001 and the accident at Fukishima (2011) exposed even more vulnerabilities that had not been considered in the original safety designs of nuclear plants.

Notwithstanding the lives that were lost and the harm that was caused, these events were costly. Each event exposed major design deficiencies and, each deficiency exposed inadequate regulations. Consequently, each event resulted in a myriad of new safety regulations, requiring extensive plant modifications and operational changes at every nuclear power plant. Despite decades of safety improvements, nuclear power plants cannot be made legally safe. Nuclear power plants cannot comply with 10 CFR 100.11.

Because of the complex ways in which nuclear plants structures, and components can fail during credible nuclear accidents, it is economically infeasible to redesign or retrofit old nuclear power plants to comply with GDC-19. This motivates the industry to circumvent the deterministic requirements of GDC-19 and 10 CFR 100.11.

In 2019, the NRC further acknowledges that "*the control room accident dose criterion has proven to be challenging to demonstrate with most plants having very little margin to the regulation* [Appendix B to Part 50, GDC-19]. Does "*very little margin*" to GDC-19 "*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*" as is apparently required by RG 1.183?

As documented in my previous comments, the NRC clearly knows that the "*Protection by Multiple Fission Product Barriers*" are grossly inadequate. They cannot protect people (and the environment) from severe nuclear accidents as required by 10 CFR 100.11 and 10 CFR 50.67. In fact, the inferior design of these barriers will cause them to overheat, create explosive gases, and catastrophically self-destruct during credible accidents.

Because many of us watched the containment barriers at Fukushima explode, the NRC was compelled to require similar nuclear power plants to install Hardened Containment Vents. In recognition that containment barriers will fail, the NRC now requires nuclear power plant operators to use the Hardened Containment Vents, during credible accidents, to release large quantities of highly radioactive material directly to the environment to prevent their containment barriers from self-destructing and releasing much more radioactive material.

After studying severe accidents for decades—admitting that containment barriers will fail in credible nuclear accidents and watching the containment barriers at Fukushima catastrophically fail, the NRC wrongly allows nuclear power plants to assume that containments will not fail in accident dose calculations using RG 1.183.

Footnote[1] of DG-1389 states:

> "*These evaluations assume containment integrity with* <u>offsite</u> *hazards evaluated based on design basis containment leakage*."

Footnote[2] of DG-1389 states:

> "*The purpose of this approach would be to test the adequacy of the containment and other safety-related systems."*

These footnotes give rise to a circular position. DG-1389 proffers that containment barriers can be adequately tested by using design-basis evaluations that assume they will not fail. This fallacy epitomizes the NRC's design-basis contravention.

NUREG-1777, "*Regulatory Effectiveness Assessment of Option B of Appendix J"* concluded that the voluntary Option B of 10 CFR 50, Appendix J, *Primary Reactor Containment Leakage*

Testing for Water-Cooled Power Reactors, was effective. Notably, while making this conclusion, NUREG-1777 unapologetically explains that the gross ineffectiveness of the mandated *General Design Criteria for Nuclear Power Plants, Criterion—16 Containment Design* was, in part, the basis for establishing the voluntary Option B that relaxed the containment leak testing requirements of Appendix J.

NUREG-1777 states:

> *"Reactor containments constitute one of the principal lines of defense in the defense-in-depth design philosophy embodied in the current generation of light water power reactors."*
>
> *And,*
>
> *"10 CFR 50, Appendix A, General Design Criteria for Nuclear Power Plants, Criterion 16, mandates that the primary containment provide an essentially leak-tight barrier to protect against uncontrolled release of radioactivity to the environment following postulated accidents."*
>
> Regardless,
>
> *"Several mechanisms can cause releases to the environment. These include <u>gross failure of containment</u> due to the pressure forces resulting from an accident, <u>containment base-mat melt-through</u>, <u>failure of containment isolation systems</u>, <u>interfacing system loss-of-coolant accidents</u>, <u>steam generator tube ruptures</u>, and releases as a result of containment leakage."*
>
> And,
>
> *"Containment leakage is a small contributor to overall accident risk. At the lower end of changes in leakage rates, any uncertainties associated with the calculated leakage contribution are minuscule in comparison with other uncertainties, (e.g., <u>prediction of containment failure mode probabilities and magnitudes of fission product source terms</u>.)"*
>
> NUREG-1777 quotes:
>
> - *"The effect of containment leakage is small since risk is dominated by accident sequences that result in <u>failure or bypass of containment</u>."* —NUREG/CR-4330, "Review of Light Water Reactor Regulatory Requirements" (1986)
> - *". . . the overall levels of risk due to containment leakage are less than previous studies because accident <u>risks are dominated by scenarios where the containment fails or is bypassed</u>. -A major finding is that <u>maintaining containment structural integrity post-accident</u> is much more important than containment leak tightness."* —NUREG-1150, "Severe Accident Risk: An Assessment of Five U.S. Nuclear Power Plants" Vols. 1 and 2 (1990)

- "Risk is dominated by containment failure and bypass following severe accidents." –NUREG-1560, Individual Plant Examination Program: Perspectives on Reactor Safety and Plant Performance" (1997)

These, and other, reactor accident studies; the catastrophic containment failures at Fukushima (U.S. designed BWR plants) and; NRC Order EA-13-109, clearly establish that it is physically and economically infeasible to prevent containment failures in credible nuclear accidents. The best that can be done to prevent catastrophic containment failures is to intentionally release the radiation they were inadequately designed to contain. Unfortunately, this only option is not in the best interest of public safety or clean environment.

It is important to recognize that intentionally releasing large quantities of highly radioactive material to the environment is a poor *"defense-in-depth design philosophy."* This current strategy does not defend or protect people or the environment from credible nuclear accidents. Instead, it concedes that intentionally polluting the environment with large quantities of highly radioactive material is better than polluting the environment with larger quantities of highly radioactive material. It concedes that overexposing/sacrificing local communities to radiation is better than overexposing larger populations.

Not only does NRC Order EA-13-109 confirm that GDC-16 was ineffective, it negates GDC-16. It still seems that some form of regulatory assessment would be prudent.

The NRC would not have issued Order EA-13-109 if they knew reactor containments would not fail in credible nuclear accidents. By requiring that containments be vented during credible severe accidents, the NRC essentially created a "*new or different kind of accident from any accident previously evaluated*" that "*involve a significant reduction in a margin to safety*" (§50.92).

If 10 CFR 50.67 is to be effective, it seems that any revision to RG 1.183 must account for the source terms of credible nuclear accidents that result in containment failures or require the use of Hardened Containment Vents.

As was the case of the inadequately designed sea wall at Fukushima, U.S. designed nuclear power plants were/are not designed to protect people and the environment from credible nuclear accidents. They are far to dangerous, and it is impractical to relocate people and business far enough away from nuclear plants that 10 CFR 100.11 could be legitimately satisfied.

The NRC's design-basis contravention attempts to legitimize the safety (and economic viability) of nuclear power plants by using the inferior standards to which they were designed as the basis for complying with regulations, instead of using the legitimate standards to which they should have been designed. It seems the contravention simply and wrongly truncated the AEC's requirement to perform Design Basis Accidents, Transients, and Events analyses.

Whether specious efforts are taken to limit the applicability of regulations to design-basis accidents or explicit claims are made to exclude credible accidents from the applicability of regulations because they are beyond (not) design-basis accidents, the end result is the same; containments and every other fission product barrier will fail in credible nuclear accidents.

Ironically, the design-basis contravention must deviate from design-basis, because even using the contravention, nuclear power plants cannot comply with GDC-19. This is why RG 1.183 and DG-1389 lowers design-basis standards. They credit the use of non-safety related and non-seismically qualified systems and components—that are not credited in design-basis analyses. Proffered inspections of non-safety related equipment (e.g., piping, condensers) do not satisfy legitimate design-basis analyses that can only credit safety-related equipment. RG-1389's "seismically rugged" is simply artifice; legitimate design-basis seismic analyses refer to these components as "seismically unqualified." As such, DG-1389 is contrary to NRC Regulatory Issue Summary 2001-19: *Deficiencies in the Documentation of Design Basis Radiological Analyses Submitted in Conjunction with License Amendment Requests.*

It appears the NRC's RG 1.183 efforts are narrowly focused on obscuring known design deficiencies—providing nuclear power plants with questionable and unscientific methods to perform accident dose calculations so they can feign compliance with regulations; circumvent deterministic regulations; bring them into compliance or; otherwise increase their profit margins.

I am opposed to RG 1.183 and DG-1389 because they provide the means to falsify accident dose calculations and feign compliance with nuclear safety regulations.

Sincerely,
Brian D. Magnuson
magnuson28@msn.com
1020 Station Blvd. #212
Aurora, IL 60504
Lead Emergency Management Specialist—Constellation (formerly Exelon Generation)
Former NRC Licensed Senior Reactor Operator/Operations Shift Manager at QC NPP
—Acting expressly as a member of the public

REFERENCES:

(March 15, 2022) Magnuson initial comments on "*DRAFT REGULATORY GUIDE DG-1389, Proposed Revision 1 to Regulatory Guide 1.183"*

DRAFT REGULATORY GUIDE DG-1389, Proposed Revision 1 to Regulatory Guide 1.183

B. Discussion

This revision of the guide (Revision 1) addresses new issues identified since the guide was originally issued. These include (1) using the term maximum hypothetical accident (MHA)1 loss-of-coolant accident (LOCA) to define the accident described in the applicable regulations in Section A above . . .

Footnote 1: *The maximum hypothetical accident (MHA) (also referred to as the maximum credible accident) is that accident whose consequences, as measured by the radiation exposure of the surrounding public, would not be exceeded by any other accident whose occurrence during the lifetime of the facility would appear to be credible. As used in this guide, the term "LOCA" refers to any accident that causes a loss of core cooling. The MHA LOCA refers to a loss of core cooling resulting in substantial meltdown of the core with subsequent release into containment of appreciable quantities of fission products. These evaluations assume containment integrity with offsite hazards evaluated based on design basis containment leakage.* RG 1.183 (Revision 1) / DG-1389: Magnuson Public Comments

The term "*maximum hypothetical accident (MHA)"* appears misleading. Because Regulatory Guides are not regulations, it seems inappropriate to use an NRC Regulatory Guide to bound applicable regulations to the DG-1389 definition of the MHA. Otherwise stated, the NRC does not have the authority to redefine the applicability of federal nuclear safety regulations (or the current licensing basis (CLB) of nuclear power plants) by using Regulatory Guide 1.183. For example, the requirements of 10 CFR 50, Appendix A, General Design Criterion—19 are not limited to a LOCA or the MHA described in DG-1389.

The term "*maximum credible accident* (MCA)" was developed by the Atomic Energy Commission (TID-14844, *Calculation of Distance Factors for Power and Test Reactor Sites*) to quantify the threat of nuclear power plants in terms of the worst-case possible accident release of radiological material to the environment. In theory, by quantifying the worse-case radiological release, nuclear power plants would be built safe distances from populations. Subsequently, plant designers used the "maximum credible accident" to design safety systems, such that they would prevent or mitigate the quantified threat, thereby, ensuring compliance with ‘source term’ regulations. The AEC authors of TID-14844 developed, what they thought was, a conservative example to describe the “*maximum credible accident,”* but they intentionally did not define it, stating:

> *“Consideration of these as well as other aspects of* [radiological] *hazards evaluation involves so many different situations and such complex technological problems that it would be quite impossible to anticipate and answer all questions that will arise.”*
>
> *“Designers of reactors are expected to examine all significant aspects of the hazards and safety problem they believe are appropriate to the particular situation with which they are dealing. In any case, the designer and/or, applicant bears the responsibility for justifying all the assumptions and methods of calculation used in a* [radiological] *hazards evaluation. The fact that aspects of the problem are not considered in the example set forth here, does not in any way relieve the designer and/or applicant of the responsibility for carefully examining, in his particular case, every significant facet of the hazards and safety problem.”*

As stated in RG 1.183 Revision 0, *“Since the publication of TID-14844* [1962]*, significant advances have been made in understanding the timing, magnitude, and chemical form of fission product releases from severe nuclear power plant accidents.”* These advances were made as the result of severe accident studies conducted by or for the NRC. Conclusions from studies include the following:

> “*As in the DBA-LOCA class, the doses from "melt-through" releases (involving thousands of curies) generally would not exceed even the most restrictive PAG beyond about 10 miles from a power plant. The upper range of the core-melt accidents is categorized by those in which the containment catastrophically fails and releases large quantities of radioactive materials directly to the atmosphere because of over-pressurization or a steam explosion. These accidents have the potential to release very large quantities (hundreds of millions of curies) of radioactive materials. There is a full spectrum of releases between the lower and upper range with all of these releases involving some combination of atmospheric and melt-through accidents. These very severe accidents have the potential for causing serious injuries and deaths.*” [(1978) NUREG-0396 (EPA 520/1-78-016)]
>
> *“The accident at TMI demonstrated the reality of the risk, previously only theoretically assessed, of accidents that result in substantial degradation and melting of the core. This risk arises from the fact that core-degradation accidents can lead to containment failure and the eventual release of large amounts of radioactivity to the environment.”* [(1980) NUREG-0660 (Vol. 1), “NRC Action Plan Developed as a Result of the TMI-2 Accident”]

"In examining the source terms . . . one must assess the probability of escape from the containment under routine use conditions or in any postulated accident situation. [(1983) NUREG/CR-3332, *"A Textbook on Environmental Dose Analysis"*]

"Conclusion 9: Containment performance (survival, failure, or bypass), which is described by input parameters in all current source term codes, is a major factor affecting source terms." [(1986) NUREG-0956*, "Reassessment of the Technical Bases for Estimating Source Terms"*]

"The consequences of severe reactor accidents depend greatly on containment safety features and containment performance in retaining radioactive material. The early failure of the containment structures at the Chernobyl power plant contributed to the size of the environmental release of radioactive material in that accident. Maintaining the integrity of the containment can affect the source term by orders of magnitude. The NRC's 1986 reassessment of source term issues reaffirmed that containment performance "is a major factor affecting source terms."" [(1990) NUREG-1150, "*Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants*"]

"Shortly after the accident at Three Mile Island, the NRC initiated a program to review the adequacy of the methods available for predicting the magnitude of source terms for severe reactor accidents. After considerable effort and extensive peer review, the NRC published a report entitled "Reassessment of the Technical Bases for Estimating Source Terms," NUREG-0956. As expected, the magnitude of the source term varies between different accident progression bins depending on whether or not containment fails, when it fails, and the effectiveness of engineered safety features (e.g., BWR suppression pool) in mitigating the release. However, within an accident progression bin, which represents a specific set of accident progression events, the uncertainty in predicting severe accident phenomena is great." [(1990) NUREG-1150, "*Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants*"]

"With regard to the lack of treatment of some severe accident phenomena and containment failure modes, it would not be appropriate to attempt to identify or analyze all containment failure modes or scenario pathways. It is the intent of the latest sequence selection, reported in NUREG/CR-4624, to analyze at least the most risk significant of these pathways." [(1990) NUREG-1150, "*Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants*"]

These NRC studies conclude that containments will fail in credible nuclear accidents. Regretfully, we no longer need to rely on theoretical studies. We watched the containments at Fukushima Daiichi catastrophically fail.

Despite the efforts of the operators at the Fukushima Daiichi nuclear power plant to maintain control, the reactor cores in Units 1–3 overheated, the nuclear fuel

> *melted and the three containment vessels were breached. Hydrogen was released from the reactor pressure vessels, leading to explosions inside the reactor buildings in Units 1, 3 and 4 that damaged structures and equipment and injured personnel. Radionuclides were released from the plant to the atmosphere and were deposited on land and on the ocean. There were also direct releases into the sea.* [*The Fukushima Daiichi Accident, IAEA Report by the Director General*]
>
> *The common cause failures of multiple safety systems resulted in plant conditions that were not envisaged in the design. Consequently, the means of protection intended to provide the fourth level of defense in depth, that is, Prevention of the progression of severe accidents and mitigation of their consequences, were not available to restore the reactor cooling and to maintain the integrity of the containment.* [*The Fukushima Daiichi Accident, IAEA Report by the Director General*]
>
> *The confinement* [containment] *function was lost as a result of the loss of AC and DC power, which rendered the cooling systems unavailable and made it difficult for the operators to use the containment venting system. Venting of the containment was necessary to relieve pressure and prevent its failure. The operators were able to vent Units 1 and 3 to reduce the pressure in the primary containment vessels. However, this resulted in radioactive releases to the environment. Even though the containment vents for Units 1 and 3 were opened, the primary containment vessels for Units 1 and 3 eventually failed. Containment venting for Unit 2 was not successful, and the containment failed, resulting in radioactive releases.* [*The Fukushima Daiichi Accident, IAEA Report by the Director General*]

Now that we know containments have and will fail in credible nuclear accidents, it is important to understand why they may fail again.

1.1.3 Integrity of Facility Design Basis

The DBA source term used for dose consequence analyses is a fundamental assumption and the basis for much of the facility design. Additionally, many aspects of an operating reactor facility are derived from the radiological design analyses that incorporated the TID-14844 accident source term.

Otherwise stated, nuclear power plants were constructed with physical barriers and safety systems that were specifically designed to prevent and mitigate the TID-14844 "maximum credible accident" source term release example. As we now know that is no fault of TID-14844, its example of the "maximum credible accident" was far from *conservative; it incorrectly assumed that containments would not fail. Unfortunatley, the designers of nuclear power plants did not fulfil their "responsibility for carefully examining, in his [each] particular case, every significant facet of the [radiological] hazards and safety problem." Instead, they designed nuclear plants, and their safety systems, assuming containments would never fail. This is, in part, the fault of the AEC and now the NRC.*

Background

An accident source term is intended to represent a major accident involving significant core damage not exceeded by that from any other 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. Facility-analyzed DBAs are not intended to be actual event sequences; rather, they are intended to be surrogates to enable deterministic evaluation of the response of engineered safety features (ESFs). These accident analyses are intentionally conservative to compensate for known uncertainties in accident progression, fission product transport, and atmospheric dispersion.

After studying containment failures for the past 40 years and watching the containments at Fukushima Daiichi catastrophic fail, the NRC required that similar U.S. designed nuclear power plants install hardened containment vents to reduce the risks posed to the public from credible containment failure accidents.

Now that we know containments will fail, it is important to understand why venting (intentionally releasing) large quantities of highly radioactive material directly to the environment would be required to reduce the risk to the public.

Nuclear power plants were designed and built based on the defense in depth philosophy. They were designed with four (4) physical barriers, such that, each barrier must fail before there would be a significant radiological release to the environment. The first barrier is the uranium-oxide fuel pellet. The second barrier is the fuel pellet cladding. The reactor coolant system provides a third barrier to fission product release. The final and ultimate barrier to fission product release is the reactor containment.

(1) The first barrier proved to be ineffective when the NRC discovered that "*high burnup fuel pellets could fragment, relocate axially and possibly disperse outside of the fuel rod during postulated design-basis accidents including, but not limited to, LOCA*." This means that lesser (clad failure) accidents may have larger radiological release source terms than more severe (core melt) accidents. [SECY-15-0148, *Evaluation of Fuel Fragmentation, Relocation and Dispersal Under Loss-Of-Coolant Accident (LOCA)*

Conditions Relative to The Draft Final Rule on Emergency Core Cooling System Performance During A LOCA (November 30, 2015)]

(2) The second barrier, the fuel cladding, is probably the costliest ‘engineering mistake.’ This design error, not the tsunami, created the high concentrations of hydrogen that exploded, causing the catastrophic containment failures at Fukushima Dai-ichi.

“During accident conditions when the core materials are inadequately cooled, the fuel cladding (zirconium alloy) can overheat which promotes and accelerates a corrosion reaction commonly referred to as the Zirc-Water Reaction. When the core is no longer submerged in water, cladding surface temperature heats up with the uranium fuel being the heat source. At cladding surface temperatures in excess of 2000 $_0$F the reaction rate is significant. At cladding surface temperatures approaching 2500 $_0$F the resultant heat is enough to maintain a high reaction rate (exothermic) regardless of fuel temperature.”

When needed the most, the fuel clad barrier will essentially self-destruct, adding heat to an overheated core while creating large concentrations of explosive gas. The second barrier will contribute to, or directly, cause the failures of barriers three and four (reactor coolant system and containment). This is why the nuclear industry was required to develop Accident Tolerant Fuel.

It is also important to recognize that the fuel cladding is the only physical barrier between the spent fuel and the environment. Spent fuel pool Zirc-fires are also credible accidents that have may have the largest source terms.

(3) During core melt accidents, the third barrier (reactor coolant system) will fail from mechanical damage, core-melt creep rupture or high pressure melt ejection.

(4) According to NRC studies (referenced below), the fourth barrier, containment, is expected to fail “*as a result of direct attack by molten core debris.” “Drywell [containment] rupture due to pedestal failure or rapid over pressurization is also an important contributor to early containment failure.” “Late failure of containment is also most likely to occur in the drywell* [containment] *but in the form of prolonged leakage past the drywell head.*” And, “*early containment failure in station blackout is dominated by hydrogen deflagrations.”*

Because of the “*relatively high probabilities that those* [BWR Mark I and II] *containments would fail should an accident progress to melting the core*,” the NRC issued Order EA-13-109, requiring hardened containment vents to be installed at specified plants. This NRC order, essentially, requires nuclear plant operators to intentionally vent (release) large amounts of highly radiological material to the environment, because the containment barriers were not designed to survive credible accidents, which include hydrogen explosions.

Consideration of International Standards
The International Atomic Energy Agency (IAEA) works with member states and other partners to promote the safe, secure, and peaceful use of nuclear technologies. The IAEA develops Safety Requirements and Safety Guides for protecting people and the environment from harmful effects of ionizing radiation. This system of safety fundamentals, safety requirements, safety guides, and other relevant reports reflects an international perspective on what constitutes a high level of safety. To inform its development of this RG, the NRC considered IAEA Safety

Requirements and Safety Guides pursuant to the Commissions International Policy Statement (Ref. 13) and Management Directive and Handbook 6.6, "Regulatory Guides" (Ref. 14).

The NRC staff did not identify any IAEA Safety Requirements or Guides with enough detailed information relevant for use by Part 50 and 52 licensees and applicants as related to the topic of this RG.

4.1 Offsite Dose Consequences

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

b. The exposure-to-CEDE factors for inhalation of radioactive material should be derived from the data provided in International Commission on Radiological Protection (ICRP) Publication 30, "Limits for Intakes of Radionuclides by Workers," issued in 1979 (Ref. 28).

--Since 1978, the ICRP has twice updated their recommendations. Nevertheless, the NRC continues to resist the adoption of more recent international standards that are "*based on the latest available scientific information of the biology and physics of radiation exposure.*" The NRC evaluated ICRP-103 in 2017: ○ *"The reason for performing this evaluation is that for certain plants, the margin for meeting acceptable dose limits in the control room for certain DBAs is very small. The concern is that the implementation of the new ICRP 103 values could cause these plants to exceed the regulatory dose limits."*

−−*"In design basis radiological consequence analyses* [in which containments do not fail]*, iodine isotopes are a major contributor to the estimated thyroid dose. ICRP Publication 26 assigns a thyroid tissue weighting fact or of 0.03. ICRP Publication 103 assigns a thyroid tissue weighting factor of 0.04. It is seen therefore that the estimated fatal cancer for the thyroid is about 33 percent higher in the ICRP Publication 103 recommendations compared to ICRP Publication 26 recommendations. The evaluation of the application of the ICRP 103 iodine DCFs as compared to the ICRP 26 DCFs results in an increase in control room TEDE of approximately 23 to 25 percent."*

−−The ICRP-103 evaluation confirmed the NRC's overriding concern "*that the implementation of the new ICRP 103 values could cause these plants to exceed the regulatory dose limits."* Therefore, the NRC concluded: *"The NRC staff's decision to discontinuing the rulemaking activities associated with potential changes to the radiation protection and reactor effluents regulations was based on the knowledge that the current NRC regulatory framework continues to provide adequate protection of the health and safety of workers, the public, and the environment."*

The NRC's reason for rejecting ICRP-103 is my overring concern with RG 1.183 (Revision 0) and DG-1389.

(November 2011) NUREG-0654 FEMA-REP-1, Rev. 1 Supplement 3: Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants

Note 1: Rapidly Progressing Severe [Accident] Incident

A rapidly progressing severe incident is a General Emergency (GE) with rapid loss of containment integrity (emergency action levels indicate containment barrier loss) and loss of ability to cool the core. This path is used for scenarios in which containment integrity can be determined as bypassed or immediately lost during a GE with core damage.

(November 2012) NEI 99-01 (Revision 6) Development of Emergency Action Levels for Non-Passive Reactors

For severe core damage events, uncertainties exist in phenomena important to accident progressions leading to containment failure. Because of these uncertainties, predicting the status of containment integrity may be difficult under severe accident conditions. This is why maintaining containment integrity alone following sequences leading to severe core damage is an insufficient basis for not escalating to a General Emergency.

PSAs [probabilistic safety assessments - also known as probabilistic risk assessment, PRA] indicated that leading contributors to latent fatalities were sequences involving a containment bypass, a large Loss of Coolant Accident (LOCA) with early containment failure, a Station Blackout lasting longer than the site-specific coping period, and a reactor coolant pump seal failure. The generic EAL methodology needs to be sufficiently rigorous to address these sequences in a timely fashion.

(November 23, 2019) PRM-50-121, Re-Submittal - 10 CFR 2.802 Petition for rulemaking Accident Dose Criteria

The proposed rule would allow licensees to adopt revised accident dose acceptance criteria as an alternative to the accident dose criteria specified in § 50.67 Accident source term. The revised accident dose criteria would be described in a separate voluntary rule § 50.67(a) specifying a uniform value of 100 milli Sieverts (10 rem) for the off-site locations and for the control room.

Problem Description:

The U.S. Nuclear Regulatory Commission's (NRC's) design basis accident (DBA) dose criteria and the resulting design of accident mitigation systems could be perceived to emphasize protection of the control room operator over protection of the public. The control room criterion restricts the calculated 30-day accident dose to the annual occupational limit of five rem while the off-site dose criteria allows for a calculated dose of 25 rem in two hours. The off-site dose criteria were derived from the siting practices of the earliest reactors and are not reflective of current health physics knowledge or modern plant construction. As a result, the design of accident mitigation systems may not be optimized in the best interest of NRC's mission of protecting public health and safety. The control room accident dose criterion has proven to be challenging to demonstrate with most plants having very little margin to the regulation.

Proposed Solution:

The proposed voluntary rule would allow licensees to adopt revised accident dose criteria that will; (1) be reflective of modern health physics recommendations and modern plant designs, (2) provide a better balance between protection of the control room operator and protection of the public, and (3) relieve the unnecessary regulatory burden associated with meeting the current control room dose criterion.

The attached petition includes the history of the current dose criteria, proposed changes to § 50.67 Accident source term and General Deign Criterion 19, corresponding revisions to Regulatory Guide 1.183, Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors, as well as other supporting information.

SUMMARY:

During the 1950s, applicants for reactor construction permits submitted Hazards Summary Reports to the Atomic Energy Commission (AEC) describing the potential dose consequences from what was considered the “maximum credible accident.”[1] These evaluations contained wide variations in both the assumed source terms as well as the proposed dose acceptance criteria. In response to the recognition that more definitive siting criteria was needed, the AEC developed a procedural methodology to define reactor siting criteria that was generally consistent with the siting practices in effect at the time. There was a concern within the AEC that it was premature to codify these criteria so early in the development of the nuclear power industry. Notwithstanding this concern, in 1962, the AEC published 10 CFR Part 100, “Reactor Site Criteria”, specifying dose acceptance criteria of 25 rem whole body and 300 rem thyroid for a 2 hour period at the Exclusion Area Boundary (EAB) and for the accident duration at the outer boundary of the Low Population Zone (LPZ).

Control Room Dose Criterion: Objectives

In 1971 Appendix A, “General Design Criteria for Nuclear Power Plants,” was added to 10 CFR Part 50. General Design Criterion 19 (GDC-19) specified that adequate protection shall be provided to permit access and occupancy of the control room for the duration of an accident without exceeding a radiation exposure of 5 rem whole body or its equivalent to any part of the body. From its inception, GDC-19 became the limiting dose criteria in almost all radiological dose consequence analyses.

The 5 rem control room dose criterion is limiting for many licensees and this raises the question regarding whether a slightly higher value could still satisfy the objective of providing a comfortable environment for the operators while reducing regulatory burden by increasing the small margin many licensees have relative to the current acceptance criterion.

There are no footnotes or notes in criterion 19 to define the accident condition tobe analyzed as is the case in 10 CFR 100.11[33]. By guidance, licensees are directed to analyze the control room radiological habitability with the same conservative assumptions and MCA source term used in the evaluation of the off-site reference values.

Additional Challenges to Meeting the Requirements of GDC-19

As can be seen by examination of representative MCA results shown in Appendix E[38] of this petition, many licensees' evaluations have a relatively small margin to the control room acceptance value. With the adoption of the TEDE dose criterion many licensees have gained operational flexibility over the previous use of a thyroid dose criterion. The current thyroid dose weighting factor being used in the calculation of TEDE is 0.03 per 10 CFR 20.1003. The International Commission of Radiation Protection (ICRP) Publication 103 has recommended the use of a thyroid weighting factor of 0.04. The NRC's Office of Nuclear Regulatory Research completed a study entitled, "Control Room Dose Evaluation Using ICRP 103 Dose Conversion Factors," letter report (ADAMS Accession No. ML17156A603), which concludes that: "Application of the ICRP 103 DCFs will result in an increase in the range of 23 to 25% in the TEDE doses for the control room." The degree of impact will depend on the amount of credit taken for various iodine removal mechanisms both natural and engineered. However, if the ICRP recommendations are ever incorporated into NRC's regulations and guidance, the incorporation of a thyroid weighting factor of 0.04 will decrease the already small margin many licensees have in their control room dose consequence analysis.

GDC-19 requires that, "*Adequate radiation protection shall be provided to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 5 rem whole body, or its equivalent to any part of the body, for the duration of the accident."* The NRC has not emphasized the issue of control room access in any of the regulatory guides dealing with control room habitability. As such most licensees do not include an evaluation of access dose in their control room dose consequence analysis.

Including access dose in the calculation of the total control room would decrease the already small margin most licensees have in their control room dose consequence analysis.

NUREG-0625 and 10 CFR 50.34

In August 1978, the Nuclear Regulatory Commission directed the staff to develop a general policy statement on nuclear power reactor siting which resulted in NUREG-0625[39], "Report of the Siting Policy Task Force." NUREG-0625 recommended that fixed distances should be required for the EAB and the LPZ.

> ABSTRACT [From NUREG-0625]
>
> In August 1978, the Nuclear Regulatory Commission directed the staff to develop a general policy statement on nuclear power reactor siting. A Task Force was formed for that purpose and has prepared a statement of current NRC policy and practice and has recommended a number of changes to current policy. The recommendations were made to accomplish the following goals:
>
> > 1. To strengthen siting as a factor in defense in depth by establishing requirements for site approval that are independent of plant design consideration. The present policy of permitting plant design features to compensate for unfavorable site characteristics has resulted in improved designs but has tended to deemphasize site isolation.
> >
> > 2. To take into consideration in siting the risk associated with accidents beyond the design basis (Class 9) by establishing population density and

distribution criteria. Plant design improvements have reduced the probability and consequences of design basis accidents but there remains the residual risk from accidents not considered in the design basis. Although this risk cannot be completely reduced to zero, it can be significantly reduced by selective siting.

3. To require that sites selected will minimize the risk from energy generation. The selected sites should be among the best available in the-region where new generating capacity is needed. Siting requirements should be stringent enough to limit the residual risk of reactor operation but not so stringent as to eliminate the nuclear option from large regions of the country. This is because energy generation from any source has its associated risk, with risks from some energy sources being greater than that of the nuclear option.

The concern was that siting practices were not providing enough emphasis on site isolation as an important contributor to defense in depth because ESF systems such as iodine filters, containment sprays, and double containment structures could be designed to make almost any site acceptable from an accident dose calculation point of view.

*

In the late 1970s there were concerns within the NRC that siting practices were not providing enough emphasis on site isolation as an important contributor to defense-in-depth because engineered safety feature (ESF) systems could be designed to make almost any site acceptable from an accident dose calculation point of view. In August 1978, the NRC directed the staff to develop a general policy statement on nuclear power reactor siting which resulted in NUREG- 0625, "Report of the Siting Policy Task Force," recommending that fixed distances should be required for the EAB and the LPZ in lieu of dose consequence analyses. After numerous comments objecting to a proposed rule (57 FR 47802), which was based on NUREG-0625 recommendations, the commission decided to retain source term and dose calculations by relocating a new single dose criterion based on total effective dose equivalent (TEDE) in 10 CFR 50.34 (61 FR 65157 December 11, 1996).

The new TEDE criterion is applicable to all new reactors and existing reactors that choose to adopt the alternative source term (AST) methodology. Depending on the contribution to TEDE dose from iodine in the released source term, <u>the 25 rem TEDE criterion allows for the associated thyroid dose to substantially exceed the previously controlling 300 rem thyroid limitation</u>. Therefore, new reactors are being sited with a less restrictive dose criterion than the earliest reactors.

<u>DISCUSSION:</u>

Hazard Summary Reports issued in the 1950's included the dose consequences from a maximum credible accident (MCA) also referred to as a maximum hypothetical accident (MHA) or a maximum probable accident (MPA). Such evaluations were based on the assumption that the plant experienced a substantial core melt releasing appreciable quantities of fission products into the containment atmosphere. These evaluations assumed <u>containment integrity</u> with offsite hazards evaluated based on design basis containment leakage. Applicants then evaluated the off-site radiological conditions for such an event and proffered various suggestions for dose acceptance criteria. The AEC evaluated these applications on a case by case basis without the benefit of a prescribed set of assumptions regarding the degree of core damage or defined dose acceptance criteria. There was a considerable effort in the AEC and the advisory committee on reactor safeguards (ACRS) during the time from 1958 through 1962 to devise a more systematic method to evaluate the licensee's MCA determinations. These concerns were described in an AEC report to the General Manager[4] by the Director of Licensing and Regulation on Reactor Site Criteria[5] as shown below:

> "The hazards reports as presented by the various applicants have shown a wide variation in estimating the magnitude of the maximum credible accident and in the dose calculational methods and, consequently, in the calculated exposure doses that might result to the offsite public in case of an accident. This situation is due partly to the differences in reactor plant design but even more to the different engineering judgments that can be made in analyzing possible consequences of accidents. AEC and ACRS review has emphasized evaluation of the safety factors that have been included in the plant design and evaluation of the conservatism represented in the analytical procedures as well as the numerical values derived. This subjective manner of arriving at judgment on site suitability has led to requests to have the AEC make more definitive the basis upon which the data are evaluated and to make more specific the safety criteria which govern the AEC's consideration of site suitability."

The promulgation of 10 CFR Part 100 and its basis document TID-14844 served to reduce the amount of subjectivity involved to the evaluation of reactor site suitability by defining the degree of core damage to be assumed in the MCA and by prescribing dose acceptance criteria.

Formally Stated Objectives of 10 CFR Part 100

The AEC first published a Notice of Proposed Rule Making regarding site criteria in 1959 (24 Federal Register Notice (FRN) 4184 1959)[6] announcing that:

> "The Commission is considering the formulation of an amendment to its regulations to state site criteria for the evaluation of proposed sites for nuclear power and test reactors and is publishing for comment safety factors which might be a basis for the development of site criteria."
>
> "In view of the complex nature of the environment, the wide variation in environmental conditions from one location to another and the variations in reactor characteristics and associated protection which can be engineered into a reactor facility, definitive criteria for general application to the siting problems have not been set forth."

The FRN went on to describe in general terms the need to show that, "the occurrence of any credible accident, will not create undue hazard to the health and safety of the public." The FRN described the general concept of an exclusion area under the complete control of the licensee as well as an area of low population density immediately outside the exclusion area.

In 1961, the AEC published 10 CFR Part 100, Reactor Site Criteria, Notice of Proposed Guides, (26 FRN 1224 1961)[7]. These guides were more descriptive and included specific dose criteria as well as an appendix detailing an example calculation of reactor siting distances. This FRN also included a more definitive set of objectives stating that:

> "The basic objectives which it is believed can be achieved under the criteria set forth in the proposed guides, are:
>
> > (a) Serious injury to individuals off-site should be avoided if an unlikely, but still credible, accident should occur;
> >
> > (b) Even if a more serious accident (not normally considered credible) should occur, the number of people killed should not be catastrophic;
> >
> > (c) The exposure of large numbers of people in terms of total population dose should be low. The Commission intends to give further study to this problem in an effort to develop more specific guides on this subject. Meanwhile, in order to give recognition to this concept the population center distances to very large cites may have to be greater than those suggested by these guides."

There were numerous comments[8] received on the proposed Part 100 Site Criteria published for comment on February 11, 1961. There was general agreement that the proposed site criteria represented a distinct improvement over the criteria published on May 23, 1959. There was a concern over the inclusion of the Appendix which was felt to be too descriptive to include in a rule. In addition, there were several comments that objected to the wording of the objectives especially in paragraph (b), "Even if a more serious accident (not normally considered credible) should occur, the number of people killed should not be catastrophic."

The objectives stated in the proposed guides published on February 11, 1961 were not repeated in the final rule which was published on April 13, 1962. The final rule (27 FRN 3509 1962)[9] included the following discussion concerning the objective of the population center distance described in 10 CFR Part 100:

> "One basic objective of the criteria is to assure that the cumulative exposure dose to large numbers of people as a consequence of any nuclear accident should be low in comparison with what might be considered reasonable for total population dose. Further, since accidents of greater potential hazard than those commonly postulated as representing an upper limit are conceivable, although highly improbable, it was considered desirable to provide for protection against excessive exposure doses to people in large centers, where effective protective measures might not be feasible. Neither of these objectives were readily achievable by a single criterion. Hence, the population center distance was added as a site requirement when it was found for several projects evaluated that the specification of such a distance requirement would approximately fulfill the desired objectives and reflect a more accurate guide to current siting practices. In an effort to develop more specific guidance on the total man-dose concept, the Commission intends to give further study to the subject. Meanwhile, in some cases where very large cities are involved, the population center distance may have to be greater than those suggested by

these guides."

Background on the Development of 10 CFR Part 100 – Reactor Site Criteria

The minutes of the ACRS subcommittee held on August 23, 1960[12], contained a draft of site criteria which defined the basis for an Exclusion Area, an Evacuation Area (later termed the Low Population Zone, and a City Distance (later termed Population center distance) as follows:

> Exclusion Area -- An area whose radius is not less than the distance at which total radiation doses received by an individual fully exposed for two hours to the radioactive consequences of the maximum credible accident would be above 25 R (or equivalent). The area should be under the full control of the applicant. Residents subject to ready evacuation are allowed.
>
> Evacuation Area -- An area whose radius is not less than the distance at which total radiation doses received by an individual fully exposed for the entire maximum credible accident would be above 25 R (or equivalent). Total population not to exceed 10,000 people and no more than 2,000 in any 45° sector.
>
> City Distance -- Distance from reactor to nearest fringe of high density population of a substantial city (above 10,000) which must not be less than distance at which total radiation doses received by a person exposed for the entire maximum credible accident would be above 10 R or equivalent. The real basis, however, for this criterion is an uncontained "puff" release" resulting in a LD-50 dose at the city boundary.

This statement by Dr. Beck that, "The real basis, however, for this criterion is an uncontained puff release of radioactivity resulting in an LD-50 [50 percent chance of death without medical intervention] dose at the city boundary," relates to the objective stated in the proposed rule that, "Even if a more serious accident (not normally considered credible) should occur, the number of people killed should not be catastrophic." This statement indicates that the actual criterion in mind was that the distance to the nearest city would be large enough that if the core melted, the **containment failed**, and all the volatile fission products were released with the wind blowing toward the city, the dose at the city boundary would be that which was estimated to kill half the people exposed to its full effect.[13] The severe accident analysis at the time was WASH 740 which predicted 3,400 acute early fatalities for a worst case reactor accident. [14]

In his testimony at the JCAE Hearings, on Radiation Safety and Regulation, June 12-15, 1961, Mr. Robert Loewenstein, Acting Director, AEC Division of Licensing and Regulations specifically discussed the population center distance as follows[15]:

> "If one could be absolutely certain that no accident greater than the "maximum credible accident" would occur, then the 'exclusion area' and 'low population' zone would provide reasonable protection to the public under all circumstances. There does exist, however, a theoretical possibility that substantially larger accidents could occur. It is believed prudent at present, when the practice of nuclear technology does not rest on a solid foundation of extended experience, to provide

<u>protection against the most serious consequences of such theoretically possible accidents</u>. Consideration of a 'population center distance' is therefore prescribed: This is a distance by which the reactor would be so removed from the nearest major concentration of people that lethal exposures would not occur in the population center <u>**even from an accident in which the containment is breached[16].**</u>"

Regulatory Guide 1.3.(**Revision **2) "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident for Boiling"

Section 50.34 of 10 CFR Part 50 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. The design basis loss of coolant accident (LOCA) is one of the postulated accidents used to evaluate the adequacy of these structures, systems, and components with respect to the public health and safety.

After reviewing a number of applications for construction permits and operating licenses for boiling water power reactors, the AEC Regulatory staff has developed a number of <u>appropriately conservative assumptions</u>, based on engineering judgment and on applicable experimental results from safety research programs conducted by the AEC and the nuclear industry, <u>that are used to evaluate calculations of the radiological consequences of various postulated accidents</u>.

This guide lists acceptable assumptions that may be used to evaluate the design basis LOCA of a Boiling Water Reactor (BWR). It should be shown that the offsite dose consequences will be within the guidelines of 10 CFR Part 100. (<u>During the construction permit review, guideline, exposures of 20 rem whole body and 150 rem thyroid should be used rather than the values given in § 100.11 in order to allow for (a) uncertainties in final design details and meteorology or (b) new data and calculational techniques that might influence the final design of engineered safety features or the dose reduction factors allowed for these features.</u>)

C. REGULATORY POSITION

1. The assumptions related to the release of radioactive material from the fuel and containment are as follows:

a. Twenty-five percent of the equilibrium radioactive iodine inventory developed from maximum full power operation of the core should be assumed to be immediately available for leakage from the primary reactor containment. Ninety-one percent of this 25 percent is to be assumed to be in the form of elemental iodine, 5 percent of this 25 percent in the form of particulate iodine, and 4 percent of this 25 percent in the form of organic iodides.

b. One hundred percent of the equilibrium radioactive noble gas inventory developed from maximum full power operation of the core should be assumed to be immediately available for leakage from the reactor containment.

c. The effects of radiological decay during holdup in the containment or other buildings should be taken into account.

d. The reduction in the amount of radioactive material available for leakage to the environment by containment sprays, recirculating filter systems, or other engineered safety features may be taken into account, but the amount of reduction in concentration of radioactive materials should be evaluated on an individual case basis.

e. The primary containment should be assumed to leak at the leak rate incorporated or to be incorporated in the technical specifications for the duration of the accident. The leakage should be assumed to pass directly to the emergency exhaust system without mixing in the surrounding reactor building atmosphere and should then be assumed to be released as an elevated plume for those facilities with stacks.

f. No credit should be given for retention of iodine in the suppression pool.

Bases for Withdrawal -2016

The NRC is withdrawing RG 1.3 because it is outdated. The guidance contained in RG 1.3 has been updated and incorporated into RG 1.183 and RG 1.195. The information in RG 1.183 provides guidance for new and existing LWR plants that have adopted the AST, and RG 1.195 provides guidance for those LWR plants that have not adopted the AST.

(June 11, 2009) RESPONSE TO A NON-CONCURRENCE ON DRAFT REGULATORY GUIDE DG-1199, "ALTERNATIVE RADIOLOGICAL SOURCE TERMS FOR EVALUATING DESIGN BASIS ACCIDENTS AT NUCLEAR POWER REACTORS"

Since the publication of TID-14844 in 1962, significant advances have been made in the understanding of radioactivity released from severe nuclear power plant accidents. In 1995, the NRC published NUREG-1465, "Accident Source Terms for Light-Water Nuclear Power Plants." NUREG-1465 uses updated research from the 1980's that provides a more realistic estimate of the accident source term, including its mix, magnitude, chemical and physical form, and timing of release.

The NRC staff anticipated that some licensees, who used TID-14844 to design their facilities, may wish to update their design bases using the NUREG-1465 source term to take advantage of the more realistic information it provides. The NRC staff, therefore, initiated several actions to provide a regulatory basis for these licensees to use an alternative source term (AST) in design basis analyses. These initiatives resulted in the development and issuance of Title 10 of the Code of Federal Regulation (10 CFR) Section 50.67 (50.67), "Accident source term."

The NRC, via regulations such as the performance-based 10 CFR 50.67, regulates all U.S. commercial nuclear power plants. 10 CFR 50.67 is an alternative voluntary regulation that allows licensees to revise the accident source term. This source term is used in the radiological analyses for designing their plant. This analysis is often referred to as a "design basis" analysis and the hypothetical or postulated events used to test the facility are known as “design basis accidents” (DBAs).

NUREG/CR-7155 - SAND2012-10702P, State-of-the-Art Reactor Consequence Analyses Project - Uncertainty Analysis of the Unmitigated Long-Term Station Blackout of the Peach Bottom Atomic Power Station

The U.S. Nuclear Regulatory Commission (NRC), the nuclear power industry, and the international nuclear energy research community have devoted considerable research over the last several decades to examining severe reactor accident phenomena and offsite consequences. The NRC initiated the State-of-the-Art Reactor Consequence Analyses (SOARCA) project to leverage this research and develop current estimates of the offsite radiological health consequences for potential severe reactor accidents for two pilot plants: the Peach Bottom Atomic Power Station, a boiling-water reactor (BWR) in Pennsylvania and the Surry Power Station, a pressurized-water reactor in Virginia. By applying modern analysis tools and techniques, the SOARCA project developed a body of knowledge regarding the realistic outcomes of select severe nuclear reactor accidents.

This document describes the NRC’s uncertainty analysis of the SOARCA unmitigated long-term station blackout (LTSBO) severe accident scenario for the Peach Bottom Atomic Power Station.

Performing the source term calculations of the Peach Bottom unmitigated LTSBO uncertainty analysis revealed three groupings of similar accident progression sequences within the Peach Bottom unmitigated LTSBO scenario: (1) early stochastic failure of the cycling SRV, which was the deterministic SOARCA scenario in NUREG-1935; (2) thermal failure of the SRV without main steam line (MSL) creep rupture; and (3) thermal failure of the SRV with MSL creep rupture. The three sequence groups exhibited differences in release magnitude, with MSL failure generally leading to the largest environmental releases.

The SOARCA analyses [2] of station blackout accidents in Peach Bottom were performed several years before the accidents at Fukushima occurred and as such, were anticipatory of the real-world events that occurred in the three accidents at Fukushima as evident from comparisons highlighted in the following The Fukushima accidents were all variants of either the long-term or short-term station blackout scenarios identified in the SOARCA Peach Bottom study.

In the SOARCA LTSBO, after returning to full RPV pressure with SRV’s cycling, one SRV is assumed to seize open [RCS BARRIER FAILURE] causing RPV depressurization and concurrent water level loss and core damage.

These comparisons highlight some of the common system responses modeled by the MELCOR code for the Peach Bottom station blackout analyses and consistently observed in the Fukushima real-world events.

Another difference observed between SOARCA Peach Bottom station blackout (SBO) analyses and the Fukushima accidents is with respect to containment failure mode and hydrogen behavior. The SOARCA analyses of Peach Bottom, a significantly larger reactor compared with the Fukushima reactors, consistently predicted drywell liner [CONTAINMENT] failure following vessel lower head failure and release of core material to the drywell cavity, caused by contact between core materials and the steel liner of the containment. This resulted in containment depressurization and release of hydrogen to the torus room at a low elevation in the reactor building.

These comparisons illustrate remarkable consistency in accident sequence progression and overall system response between MELCOR-SOARCA modeling and real-world observations from Fukushima. Differences in the signatures are generally understood and due to differences in operator actions as well as better-than-expected durability of the RCIC turbine driven steam system in the Fukushima accidents. The modeled and observed differences in hydrogen release (i.e., drywell liner [CONTAINMENT] failure versus drywell head flange [CONTAINMENT FAILURE] leakage from over-pressurization) are apparently due to modeled differences in corium behavior in the cavity, perhaps attributable to the comparatively larger Peach Bottom core which may have a higher potential to flow and contact the steel liner [CONTAINMENT]. The real-world observations from Fukushima are consistent with phenomenology and system responses modeled by MELCOR, and give confidence to the overall findings in the SOARCA studies.

The purpose of SOARCA is to evaluate the consequences of postulated severe reactor accident scenarios that might cause a NPP to release radioactive material into the environment.

A detailed uncertainty analysis was performed for a single-accident scenario rather than all seven of the SOARCA scenarios documented in NUREG-1935 [1]. This work does not include uncertainty in the scenario frequency. The SOARCA Peach Bottom BWR Pilot Plant Unmitigated LTSBO scenario [2] is analyzed. While one scenario cannot provide a complete exploration of all possible effects of uncertainties in analyses for the two SOARCA pilot plants, it can be used to provide initial insights into the overall sensitivity of SOARCA results and conclusions to input uncertainty. In addition, since station blackouts (SBOs) are an important class of events for BWRs in general, the phenomenological insights gained on accident progression and radionuclide releases may prove useful for BWRs in general.

An accident sequence begins with the occurrence of an initiating event (e.g., a loss of offsite power, a loss-of-coolant accident (LOCA), or an earthquake) that perturbs the operation of the NPP. The initiating event challenges the plant's control and safety systems, whose failure might cause damage to the reactor fuel and result in the release of radioactive material. Because a NPP has numerous diverse and redundant safety systems, many different accident sequences are possible depending on the type of initiating event that occurs, which equipment subsequently fails, and the nature of the operator actions involved, as described in the SOARCA study [1, 2]. Individual accident sequences can be grouped into accident scenarios that represent functionally similar sequences. The SOARCA project analyzed a handful of important scenarios in detail. The scenario selection process for the SOARCA project is described in NUREG-1935 [1]. Three accident scenarios were chosen for analysis for Peach Bottom (the BWR pilot plant) and four accident scenarios were selected for Surry (the PWR pilot plant) [1].

The process for selecting a SOARCA scenario for this uncertainty analysis considered both the magnitude and timing of the offsite radionuclide release, which have major impacts on both early and latent cancer fatality risks. The examination of candidate scenarios considered both the timing of core damage and the timing of containment failure.

SBOs are an important class of events for NPPs, especially BWRs, which pointed to both Peach Bottom LTSBO and STSBO scenarios as good candidates. Although the uncertainty analysis was already under way by March 2011, the events at the Fukushima Daiichi plant re-confirmed the interest in SBOs for BWRs. <u>The STSBO has a more prompt radiological release and a slightly larger release compared to LTSBO over the same interval of time</u>.

A response to a LTSBO would begin with the onsite emergency response organization and would expand as needed to include utility corporate resources, State and local resources, and resources available from the Federal government, should these be necessary. It is most likely that plant personnel would attempt to mitigate the accident before core melt, but if their efforts were unsuccessful the national level response would provide resources to support mitigation of the [LTSBO] source term [<u>versus the much lower DBA LOCA source term of RG 1.183/DG-1389</u>].

Source term release behavior in terms of the rate and total amount released in-vessel is strongly coupled to in-vessel melt progression behavior owing to the strong temperature dependence of fission product release. The onset of volatile fission product release is set by the time that fuel is heated to a temperature above about 1500 K (about 1227°C), and this is tightly <u>coupled to cladding oxidation rate</u>. Total release of both volatile and less volatile species is affected by the time at which fuel remains at elevated temperatures and the state of the fuel (rods or debris). Therefore, many of the parameters that affect [FUEL] cladding [BARRIER] oxidation and hydrogen generation also affect fission product release.

The parameters selected in the study were considered in terms of both melt progression and fission product release and transport. This includes important phenomena taking place following vessel lower head melt-through such as melt attack of the drywell liner [CONTAINMENT], <u>containment behavior issues</u>, such as uncertainty in onset of drywell head flange leakage CONTAINMENT FAILURE], and uncertainties in radioactive aerosol transport mechanics.

The dominant mechanism of <u>containment failure</u> in accident sequences involving the drywell floor, such as the LTSBO, is thermal failure (melting) of the drywell liner following contact with molten core debris (i.e., drywell liner melt-through). <u>Containment failure</u> by this mechanism occurs after debris is released from the reactor vessel lower head and flows out of the reactor pedestal onto the main drywell floor. If a sufficiently large quantity of debris accumulates in the pedestal, it can flow out of the pedestal through a large doorway in the concrete pedestal wall.

If the debris temperatures remain sufficiently high as it spreads across the drywell floor and contacts the drywell liner, <u>the liner would melt and fail</u>. The precise conditions under which core debris would flow out of the pedestal and across the drywell floor are uncertain. These uncertainties are adequately captured by assuming debris mobility and the potential for liner failure are represented by two key parameters: debris mass (i.e., static head) necessary for lateral flow and debris temperature (which characterizes debris rheological properties and internal energy available to challenge the liner).
If debris flows out of the reactor pedestal and spreads across the drywell floor, as described above, and contacts the outer wall of the drywell, the steel liner [CONTAINMENT] will fail. This failure opens a release pathway to the lower reactor building. Heat transfer between the steel liner and molten core debris is not explicitly calculated in the MELCOR model, due to limitations of the CAV Package, which addresses ex-vessel model debris behavior. The model assumes an opening in the drywell liner [CONTAINMENT FAILURE] occurs 15 minutes after debris first

contacts the drywell wall. This time delay represents an average of estimates for failure time discussed in NUREG/CR-5423 [27] for situations in which the drywell floor is not covered with water.

An ignition source for hydrogen combustion in the reactor building is unclear during a SBO. Since there are no electrically energized components in the reactor building during a SBO, the most likely ignition source will be a hot surface. Default ignition parameters were used in the SOARCA calculations for NUREG/CR-7110 Volume I. However, the accumulation of hydrogen due to an absence of an electrical ignition source is credible. The ignition of hydrogen from a hot surface is caused by local heating of the hydrogen-oxygen mixture to a point where there is a sufficiently large volume of the mixture reaching the auto ignition.

The importance of zircaloy melt breakout temperature (SC1131-2) is explained by the effect this parameter has on oxidation. Larger breakout temperatures lead to greater oxidation. Greater oxidation leads to greater heat generation and earlier MSL rupture. Earlier MSL rupture allows more gaseous iodine to enter the drywell instead of being vented to the wetwell (through the stuck-open SRV) where it would be efficiently scrubbed in the wetwell pool. Once in the drywell, the gaseous iodine is readily available to escape containment through the drywell head flange or a drywell liner melt-through.

When a MSL rupture occurs, containment over pressurizes and leaks past the drywell head flange. This results in an early release.

Whether a surge of water from the wetwell up onto the drywell floor occurs relates to amounts of cesium that deposit in the wetwell pool but fail to be confined there. In a large number of the realizations, a surge of water from the wetwell up onto the drywell floor occurs when the containment depressurizes in response to a breach developing in the drywell liner due to core debris contacting the liner and melting through it. The wetwell pool is saturated at the time and susceptible to flashing given a depressurization. The vacuum breakers between the wetwell and the drywell are overwhelmed and contaminated water from the wetwell surges up onto the drywell floor. Most of the water moves out the liner breach but some of it pools above the core debris on the drywell floor. The pool subsequently evaporates introducing its inventory of fission products to the atmosphere and structures of the drywell where they are available for release to the environment. (Note that the flow path representing the liner breach in the MELCOR model is a 6-cm high horizontal slot with its lowest point 0.41 m off the drywell floor.)

There is a correlation between the uncertainty in the drywell liner breach size and whether a surge of water from the wetwell occurs as evidenced in Figure 6.1-14. Larger sizes cause stronger containment depressurizations and hence larger potentials for water to surge from the wetwell.