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Enclosure 2

Licensing Topical Report
NEDO-33279, Revision 3
"ESBWR Fission Product Removal Evaluation Model"

Public Version

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Licensing Topical Report

ESBWR CONTAINMENT FISSION PRODUCT REMOVAL EVALUATION MODEL

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ACRONYMS AND ABBREVIATIONS

ABWR	Advanced Boiling Water Reactor
ADS	Automatic Depressurization System
AIDA	Aerosol Impaction and Deposition Analysis
AOO	Anticipated Operational Occurrence
AS-*n*	Accident Scenario *n*
AST	Alternative Source Term
ASME	American Society of Mechanical Engineers
BAF	Bottom of Active Fuel
BWR	Boiling Water Reactor
BWROG	Boiling Water Reactor Owners' Group
BWR/*n*	GE BWR product line *n* (*n* can be 2, 3, 4, 5, or 6)
CDF	Core Damage Frequency
CRDS	Control Rod Drive System
DBA	Design Basis Accident
DCD	Design Control Document (Reference 17 for the ESBWR)
DF	Decontamination Factor
DPV	Depressurization Valve
EAB	Exclusion Area Boundary
ECCS	Emergency Core Cooling System
EFU	Emergency Filter Unit
EIV	Early In-vessel Release Phase for AST
ESF	Engineered Safety Feature
FAPCS	Fuel and Auxiliary Pool Cooling System
FP	Fission Product
FW	Feedwater
GDCS	Gravity Driven Cooling System
GE	General Electric Company
GEH	GE-Hitachi Nuclear Energy Americas, LLC
GESTAR	GE Standard Application for Reactor Fuel
GF	Geometry Factor
HELB	High Energy Line Break
HVAC	Heating, Ventilation, and Cooling System
IC	Isolation Condenser

IFTS	Inclined Fuel Transfer System
LDW	Lower Drywell
LOCA	Loss of Coolant Accident
LOPP	Loss of Preferred Power
LOOP	Loss of Off-site Power
LPZ	Low Population Zone
LTR	Licensing Topical Report
MELCOR	NRC Code to Evaluate Severe Accidents
MSIV	Main Steam Isolation Valve
MSL	Main Steam Lines
MWth	Mega-Watt Thermal
NRC	United States Nuclear Regulatory Commission
PCCS	Passive Containment Cooling System
PCT	Peak Cladding Temperature
PRA	Probabilistic Risk Assessment
PWR	Pressurized Water Reactor
RADTRAD	NRC Code used to Evaluate Off-Site and Control Room Dose Consequences
RB	Reactor Building
RCPB	Reactor Coolant Pressure Boundary
RHR	Residual Heat Removal
RPV	Reactor Pressure Vessel
RTNSS	Regulatory Treatment of Non-Safety Systems
RWCU/SDC	Reactor Water Cleanup/Shutdown Cooling System
SA	Severe Accident
SAF	Single Active Failure
SBWR	Simplified Boiling Water Reactor
SER	Safety Evaluation Report
SLC	Standby Liquid Control
SRP	Standard Review Plan
SRV	Safety Relief Valve
TAF	Top of Active Fuel
TEOM	Tapered Element Oscillating Microbalance
TID	Total Integrated Dose
TRACG	GE version of the Transient Reactor Analysis Code
TS	Technical Specification(s)

VFR	Volumetric Flow Rate
SR	Surveillance Requirement
UDW	Upper Drywell
WW	Wetwell (same as Suppression Pool in pH discussions)
χ/Q or X/Q	Atmospheric Dispersion Factor (Chi over Q)

SUMMARY OF CHANGES (FROM REV 2 TO REV 3)

Most changes from Revision 2 to Revision 3 of this document are made to accommodate the following:

a) Editorial changes to correct grammar, spelling, punctuation, and acronym usage.

b) RAI 6.2-165 S1.

c) Consistency with ESBWR DCD, as needed.

Further details of the change are listed below.

Item	Location	Description of Change
1.	Cover page	Updated.
2.	Proprietary Statement	Updated as needed.
3.	Acknowledgments	Removed for Revision 3 to be consistent with GEH LTR Class III template.
4.	Acronyms	Updated to be consistent with ESBWR DCD list of acronyms.
5.	Entire Document	Made editorial changes throughout the document to correct grammar, spelling, punctuation, etc. Used only correct acronyms that conform to the DCD acronyms list. Updated all references for revision numbers and date of publication. Also, updated the text as per DCD changes from Rev 5 to Rev 6.
6.	Section 4.2.3.4	Two paragraphs and two equations were added to the section to replace text that had been mistakenly deleted in Revision 1 of the LTR.
7.	Table 4.7	Values to populate the column "AS-2 Assumed" were added to the table.
8.	Section 4.4	Design pressure updated to be consistent with DCD Revision 6 Table 6.2-1.
9.	Section 4.5	The RB mixing volume updated to be consistent with response to RAI 6.2-165 S01.
10.	Section 5.1	• The primary containment total leak rate, containment leakage rate into the RB, and the RB mixing volume updated to be consistent with response to RAI 6.2-165 S01 and the LOCA Dose analysis supporting DCD Revision 6 Section 15.4. • The Control Room EFU design flow rate updated to be consistent with DCD Revision 6 design change in the text of the section and also in the X/Q adjustment equation.

Item	Location	Description of Change
11.	Figure 5.1	The Control Room EFU design flow rate updated to be consistent with DCD Revision 6 design change.
12.	Section 5.2, Figure 5.2 , Figure 5.3, and Table 5.4	Updated to be consistent with the LOCA Dose analysis supporting DCD Revision 6 Section 15.4.
13.	Table A-1	• The containment leakage rate and mixing volume credited in dose analysis updated to be consistent with LOCA Dose analysis supporting DCD Revision 6 Section 15.4. • The Control Room EFU design flow rate updated to be consistent with DCD Revision 6 design change in the text of the section and also in the X/Q adjustment equation.
14.	References	Reference 7-1 was replaced with "Not Used". In addition, the revisions to References 7-7, 7-9, 7-20, and 7-26 were updated to be consistent with DCD Table 1.9-20.
15.	B.4.2 and B.5	Reworded for clarity.
16.	Figure B-1	Figure updated to include the LOCA results supporting DCD Revision 6 Subsection 15.4.4.
17.	Appendix C, Figure C-1, and Table C-2	Updated to be consistent with supporting analysis.

1.0 INTRODUCTION

1.1 BACKGROUND

Early plant Design Basis Accident (DBA) dose consequence evaluations were performed using source terms derived from TID-14844, "Calculation of Distance Factors for Power and Test Reactor Sites" [Ref. 7-4]. Following the Three Mile Island accident, the United States Nuclear Regulatory Commission (NRC) and other entities performed a significant amount of research into plant responses to Severe Accident (SA) scenarios at nuclear power plants. Many of the insights obtained by the significant amount of work done by the NRC and others are summarized in NUREG-1465, "Accident Source Terms for Light Water Nuclear Power Plants" [Ref. 7-12]. The NRC issued Regulatory Guide 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors" [Ref. 7-3], in July 2000.

The ESBWR is a passive design nuclear reactor. The passive design is intended to increase reliability and eliminate reliance on active systems to mitigate the consequences of postulated DBAs. The passive systems are radically different from those used in current generation BWRs, thus certain regulations (source terms) and methodologies used in previous analyses are not directly applicable to the ESBWR design. As such, additional research and evaluation was performed to develop a basis for revised methodologies to be used in evaluating the ESBWR. The purpose of this Licensing Topical Report (LTR) is to document the assumptions and methodology GEH will use in evaluating the dose consequences of DBAs. The specific items addressed in this report are

- The methodology used in modeling the Passive Containment Cooling System (PCCS) as a fission product removal source;
- The model to be used to credit the natural deposition of aerosol fission products and elemental iodine in the ESBWR primary containment;
- The impact of suppression pool scrubbing on releases through the Safety Relief Valves (SRVs);
- The model used to calculate holdup and removal of fission product leakage through the Main Steam Isolation Valves (MSIV);
- The revised model used to calculate doses to control room operators; and
- Use of the Reactor Building (RB) for holdup and decay of fission products prior to release to the environment.

1.2 SUMMARY

This report summarizes the methodology used by GEH to evaluate the potential dose consequences due to a design basis Loss of Coolant Accident (LOCA). This report is intended to provide the technical basis of the LOCA dose calculation for the ESBWR.

An analysis was performed to determine the dose consequences based on the methodologies documented in this report. The analysis demonstrates that the ESBWR systems, in conjunction with natural processes, are adequate to ensure that the dose consequences resulting from a design

basis LOCA would meet the criteria set forth in 10 CFR 52.47 and 10 CFR 50, Appendix A, General Design Criterion (GDC) 19.

1.3 ACCIDENT SCENARIOS EVALUATED

There are numerous LOCA scenarios that are considered in the design and licensing phases for nuclear power plants. Most of the current generation power plants have active systems that operate within specific design parameters despite which LOCA scenario is being evaluated. For example, a containment spray pump is rated for a certain flow rate under accident conditions, and the operation of the system is identical whether the pipe break is a "small" line or a "large" line. Thus, off-site and control room dose consequences are typically calculated for only the bounding scenario which is the scenario that results in the largest amount of fuel damage.

The ESBWR design concept relies primarily on the passive systems during a DBA, such as a LOCA. No active systems, such as containment sprays or Standby Gas Treatment Systems, are provided to limit the release of radioactivity to the environment following a postulated accident. Therefore, removal of fission products is dependent on natural processes such as plating out on containment surfaces, entrainment in containment pools, etc.

Three accident scenarios were chosen to envelope the spectrum of potential breaks that would constitute a LOCA. Table 7.2-5 of "Top Ten (Probabilistic Risk Assessment [PRA]) Level 1 Accident Scenarios" of NEDO-33201, Revision 1 [Ref. 7-37] was reviewed. Based on the review, a Loss of Preferred Power (LOPP) with failure of Isolation Condensers (ICs) and high-pressure makeup, Automatic Depressurization System (ADS), and failure of low-pressure makeup will be added to the LOCA scenarios considered for fission product removal. This scenario, and a similar one that differs only in the initiating event (Loss of Feedwater (FW) vs. LOPP), contribute to about 90% of the core damage frequency (CDF). The other scenarios in the top ten are not included because of one or more of the following:

- They contribute less than 1% to CDF; and
- They are similar to one of the other three scenarios.

Based on the above considerations, the following three scenarios were chosen:

- **Accident Scenario 1 (AS-1):** Bottom drain line break, with ADS and degraded low-pressure makeup. All three scenarios assume a loss of off-site power (LOOP) with a Safe Shutdown Earthquake (SSE). The ICs are not credited in any of the three scenarios. The PCCS is assumed to function as designed for all three accident scenarios. Injection of the Gravity Drain Cooling System (GDCS) is inhibited until approximately 2 hours. GDCS is initially inhibited to ensure fuel melt occurs ("Early In-Vessel" (EIV) release from NUREG-1465), and restoration is required to prevent the melted core from penetrating the vessel ("Ex-Vessel" release per NUREG-1465). Regulatory Guide 1.183 states that DBAs should terminate at the end of the EIV release phase; thus, restoration of cooling is required to prevent the ex-vessel release and preserve Regulatory Guide 1.183 requirements. The Reactor Pressure Vessel (RPV) ADS is assumed to operate as designed.
- **Accident Scenario 2 (AS-2):** Bottom drain line break, with degraded high-pressure makeup and initial failure of ADS. Restoration of adequate core cooling and ADS is credited after approximately 2 hrs (to meet Regulatory Guide 1.183 assumptions). The

assumptions for AS-2 are similar to AS-1 with the exception that operation of ADS is not assumed until fuel damage is complete.

- **Accident Scenario 3 (AS-3):** LOPP/Loss of FW with ADS, and degraded low-pressure makeup, restoration of adequate core cooling approximately 2 hrs after fuel damage. No break is assumed for AS-3. Emergency injection is not credited until just prior to RPV failure. The ADS is inhibited until just prior to RPV failure as well.

Two cases were evaluated for each of the accident scenarios discussed above. One case assumed all MSIVs operated as designed and isolated because of the LOCA. The second case assumed the single-failure of one MSIV. Removal coefficients were then determined based on each case. The removal coefficients in containment were chosen to ensure that both cases were bounded for each Accident Scenario.

2.0 LICENSING REQUIREMENTS

2.1 10 CFR 50, APPENDIX A, GENERAL DESIGN CRITERION 19

This regulation requires that a control room be provided from which actions can be taken to operate the nuclear power unit safely under normal conditions and to maintain it in a safe condition under accident conditions, including loss-of-coolant accidents. Adequate radiation protection is required to permit access 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.

2.2 10 CFR 52.47

This regulation requires that licensees evaluate the dose consequences due to DBAs to ensure they meet the following criteria:

(1) 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) TEDE.

(2) 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) TEDE.

2.3 STANDARD REVIEW PLAN GUIDELINES (NUREG-0800)

SRP Section 6.2.1, "Containment Functional Design", discusses the requirements to ensure that primary containment for reactors meets GDC 16, 50, 52, 53, and 54 through 57 [Ref. 7-5]. Acceptable assumptions with respect to containment leakage and dose calculations are discussed elsewhere in the SRP (primarily Section 15.6.5).

SRP Section 6.2.3, "Secondary Containment Functional Design", provides information concerning crediting of secondary containment structures for holdup, decay, and treatment of fission products by Engineered Safety Feature (ESF) charcoal filter trains [Ref. 7-6]. The ESBWR does not have a "secondary containment" per se; however, the RB is credited for the holdup of fission products prior to the release to the atmosphere. One requirement in SRP 6.2.3 is that secondary containment is maintained at a negative pressure (<-0.25" w.g.) with respect to the atmosphere.

SRP Section 6.5.2, "Containment Spray as a Fission Product Cleanup System", contains information on methodology acceptable to quantify removal of elemental iodine through deposition on containment surfaces [Ref. 7-26]. The ESBWR does not credit containment sprays to remove airborne radioiodine following a LOCA.

SRP Section 6.5.3, "Fission Product Control Systems and Structures", provides information concerning the systems and structures that are reasonable to credit for the mitigation of dose consequences follow a DBA [Ref. 7-42]. Information on leakage rates, potential bypass paths, and systems and structure requirements is provided.

SRP Section 6.5.5, "Pressure Suppression Pool as a Fission Product Cleanup System", provides guidance to licensees concerning the amount of radioactivity that may be removed via

suppression pool scrubbing [Ref. 7-7]. The SRP states "If the time integrated DF values claimed by the applicant for removal of particulate and elemental iodine are 10 or less for a Mark II or III, or are 5 or less for a Mark I containment, the applicant's values may be accepted without any need to perform calculations."

SRP Section 15.0.1, "Radiological Consequence Analyses Using Alternative Source Terms", contains information concerning the requirements for licensees who voluntarily adapt the alternate source term (AST) dose methodologies, including the results for the LOCA and other design basis events (Main Steam Line Break Outside Containment, Fuel Handling Accident, etc.) [Ref. 7-8]. The SRP states: "This SRP section and the Referenced RG-1.183 may contain information that contradicts that provided in other SRP sections. In these cases, the most recent applicable information should be used." The SRP Section does not contain very detailed information concerning assumptions. In most areas, it defers to the guidance provided in Regulatory Guide 1.183.

SRP Section 15.0.3, "Design Basis Accident Radiological Consequence Analysis for Advanced Light Water Reactors", provides guidance for licensees pursuing new reactor licenses using advanced designs, such as the ESBWR. Section I.2 states "Standard reactor designs are certified with a postulated set of short-term atmospheric relative concentration (χ/Q) values at an EAB and LPZ in lieu of site-specific meteorological data and actual distances to the EAB and LPZ [Ref. 7-38]. The NRC has determined, for purposes of the ESP review, that the certified standard reactor designs meet the radiological consequence evaluation factors identified in 10 CFR 52.47, provided that the site parameters fall within those postulated in the design certification." Table 1 of the SRP provides the dose acceptance limits for the various DBA dose consequence analyses. These acceptance criteria are consistent with those provided in Regulatory Guide 1.183 (Section 2.4) and SRP 15.0.1 (discussed previously).

SRP Section 15.6.5, "Loss-of-Coolant Accidents Resulting from Spectrum of Postulated Piping Breaks within the Reactor Coolant Pressure Boundary", provides guidance concerning acceptable assumptions with respect to containment releases for dose consequence analyses is provided in several Appendices [Ref. 7-9]. Appendix A addresses assumptions concerning most LOCA dose calculations, including leakage from the primary and secondary containment. Appendix B addresses the dose consequences of liquid leakage from ESF injection systems outside of containment, and Appendix D addresses leakage through Main Steam Isolation Valves. Note that Appendix C was deleted. Many of the assumptions with respect to dose consequences analyses documented in Section 15.6.5, including the appendices, were affected significantly by AST, and the updated assumptions and methodologies are documented in Regulatory Guide 1.183.

2.4 REGULATORY GUIDE 1.183

Regulatory Guide 1.183, "Alternative Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors", documents the assumptions and methodology acceptable to the NRC in evaluating the dose consequences of postulated DBAs utilizing the AST dose methodology. Appendix A to the Regulatory Guide documents the assumptions for evaluating the radiological consequences of a LOCA. The information contained in the Regulatory Guide often contradicts information in the older (~1981) revisions of the SRP. However, SRP Section 15.0.1 explicitly

states that the most recent applicable information should be used, which is that contained in Regulatory Guide 1.183.

The Regulatory Guide contains useful information for current generation nuclear power plants; however, not all of the guidance can be directly translated to "next generation" plants that use passive systems, such as the ESBWR. For example, the Regulatory Guide discusses assumptions applicable to the Mark I, Mark II and Mark III containments; however, because the ESBWR containment design differs significantly to each of those designs much of the information in the Regulatory Guide is not directly applicable.

2.5 REGULATORY TREATMENT OF NON-SAFETY SYSTEMS

The NRC issued a memorandum addressing the Regulatory Treatment of Non-Safety Systems (RTNSS) for advanced passive reactor designs [Ref. 7-10]. One of the criteria in the memo is intended to apply is "SSC [structure, system, and component] functions [that are] relied upon to resolve long term-safety (beyond 72 hours)." Dose consequence evaluations are intended to be performed for the "duration of the event," which is typically taken to be 30 days. The NRC memo also addresses control room habitability with respect to RTNSS ventilation systems.

3.0 ANALYTICAL TECHNIQUES AND COMPUTER CODES

3.1 MELCOR

The computer code MELCOR is a fully integrated, engineering level computer code that is used to model the progression of various accident scenarios for light water nuclear power plants. The code is discussed in detail in NUREG/CR-6119, "MELCOR Computer Code Manual" [Ref. 7-15]. MELCOR models major plant systems and their coupled reactions. Reactor plant systems and their response to off-normal or accident conditions include:

- Thermal-hydraulic response of the primary reactor coolant system, the reactor cavity, the containment, and the RB;
- Core uncovery (loss of coolant), fuel heatup, cladding oxidation, fuel degradation (loss of rod geometry), and core material melting and relocation;
- Heatup of the reactor vessel lower head from relocated fuel materials and the thermal and mechanical loading and failure of the vessel lower head, and transfer of core materials to the reactor vessel cavity;
- Core-concrete attack and ensuing aerosol generation;
- In-vessel and ex-vessel hydrogen production, transport, and combustion;
- Fission product release (aerosol and vapor), transport, and deposition;
- Behavior of radioactive aerosols in the reactor containment building, including scrubbing in water pools, and aerosol mechanisms in the containment atmosphere such as particle agglomeration and gravitational settling; and
- Impact of engineered safety features on thermal hydraulic and radionuclide behavior.

The primary use of MELCOR in this analysis is to quantify various fission product removal mechanisms. In addition, the thermal hydraulic conditions for containment may be based on information obtained from the MELCOR code. The information will then be formatted such that it may be used in off-site and control room dose consequence analyses.

A detailed methodology for modeling of the various removal mechanisms for MELCOR is presented in Section 4 of this report.

3.2 RADTRAD

Following the Three Mile Island accident, the NRC and other entities performed a significant amount of research into plant responses to SA scenarios at nuclear power plants. The research often concluded that releases of fission products were significantly less than those assumed in older off-site and control room dose consequence calculations. Many of the insights obtained by the significant amount of work done by the NRC and others are summarized in NUREG-1465 [Ref. 7-12].

The RADTRAD computer code is discussed in detail in NUREG/CR-6604, "RADTRAD: A Simplified Model for Radionuclide Transport and Removal and Dose Estimation" [Ref. 7-17]. The code was developed for the NRC to estimate the transport and removal of radionuclides, and ultimately determine the dose consequences at selected receptor locations. The code was developed in support of the NRC's research into SAs as well as in the development of AST. As

such, it is integral to the AST dose consequence methodology discussed in NUREG-1465 and Regulatory Guide 1.183.

RADTRAD is a nodal transport code. It allows up to 10 nodes (compartments) including the environment and the control room, and allows up to 25 pathways. The code allows users to account for numerous radionuclide removal mechanisms such as natural deposition in the containment, scrubbing by suppression pools, deposition in piping, etc. Material can flow between buildings, to the environment, or into the control room. An accounting of the amount of radioactive materials retained due to these tortuous pathways is maintained. Decay and in-growth of daughters can be calculated over time as the material is transported. The code allows up to four release durations, and the source term may be distributed over multiple nodes as needed.

The RADTRAD model uses information obtained from the results of MELCOR to model the various removal mechanisms for radioisotopes in containment. Version 3.03 was used for the dose consequence calculations documented in this LTR.

3.3 CHEMSHEET

ChemSheet combines the flexibility and practicality of spreadsheet applications with the thermodynamic and simulation capabilities of Gibbs Energy minimization. ChemSheet applies the ChemApp thermodynamic programming library, which handles repetitive complex equilibrium calculations for a diverse range of chemical and thermodynamic applications. ChemApp can be used to calculate both the composition and the thermodynamic properties of a multi-phase, multi-component system at given conditions.

ChemApp is derived from the ChemSage family of thermochemical calculation programs (which in turn are based on SOLGAS/SOLGASMIX programs). These are widely used in universities, corporate and government laboratories.

ChemApp consists of a library of subroutines for data handling and phase equilibrium calculation purposes. The same comprehensive library of models for non-ideal solution phases available in ChemSage is also built into ChemApp. Thus, the wide range of existing thermochemical data for ChemSage is also available for ChemApp. ChemApp also uses the same thermochemical data-file format as ChemSage.

The primary use of ChemSheet in this analysis is to determine pH in the containment pools. The input parameters for the pH calculation are obtained from MELCOR simulation results.

3.4 GOTHIC

The computer code GOTHIC (Generation of Thermal-Hydraulic Information for Containments) is an integrated, general purpose thermal-hydraulics software package for design, licensing, safety and operating analysis of nuclear power plant containments and other confinement buildings. The code is discussed in detail in the GOTHIC code manual [Ref. B-1]. GOTHIC is used to analyze thermo-hydraulic transients of multi-phase systems in complex geometries. The GOTHIC calculation program solves equations for conservation of mass, momentum, and energy for multi-phase flow. Complex systems and geometries can be modeled using a multi-block mesh in 3-D as well as lumped parameters. GOTHIC models major plant systems and their coupled reactions.

The primary use of GOTHIC in this analysis is to justify the mixing assumptions in the RADTRAD dose analysis. The model is used to determine the transport of radionuclides through the RB. The results are compared to the assumptions made in RADTRAD in order to show that RADTRAD results are conservative.

A detailed methodology for modeling of the RB in GOTHIC is presented in Appendix B of this LTR.

4.0 SOURCE TERMS AND REMOVAL MECHANISMS

4.1 SOURCE TERM ASSUMPTIONS

4.1.1 Iodine Chemical for Distribution

The chemical form of iodine documented in NUREG-1465 is based on work documented in NUREG/CR-5732, "Iodine Chemical Forms in LWR Severe Accidents" [Ref. 7-14]. NUREG/CR-5732 documents seven accident scenarios that were evaluated for four plants: Grand Gulf (BWR with a Mark III containment), Peach Bottom (BWR with a Mark I containment), Sequoyah (PWR with an ice condenser), and Surry (PWR with a large containment). For six of the seven scenarios the amount of iodine entering the containment was almost entirely in the form of CsI, with less than 0.1% of the total iodine being HI or I. For the remaining scenarios, 3.2% was I and HI (2.8% and 0.4%, respectively). As a result, NUREG-1465 states that 95% of the iodine released should be in the form of CsI, 0.15% should be assumed to be organic iodine (3% of the remaining 5%), and the remaining 4.85% is assumed to be elemental iodine. This iodine chemical distribution is recommended in Regulatory Guide 1.183 as well.

The failure mechanisms for fuel in the ESBWR are similar to those in previous BWRs. Fuel failure is not expected for any DBA scenario, as the core remains covered, however; this analysis assumes fuel damage and release durations consistent with Regulatory Guide 1.183 and NUREG-1465 guidance. Both NUREG-1465 and NUREG/CR-5732 document the fact that the organic and elemental iodine assumptions are conservative. Therefore, the iodine chemical distribution recommended by Regulatory Guide 1.183 is used in the ESBWR LOCA dose consequence analyses.

4.1.2 Pool pH Evaluation

4.1.2.1 NUREG/CR-5950 Assumptions and Methodology

The iodine chemical distribution recommended by Regulatory Guide 1.183 and NUREG-1465 is assumed to be predominately aerosol iodine. Regulatory Guide 1.183 states that the iodine chemical distribution is applicable if sump or suppression pool pH is maintained above 7. The general concern is that iodine could change chemical forms and re-evolve to the containment atmosphere if pool pH is not maintained.

The ESBWR has several separate pool volumes that could potentially contain fission products following a LOCA. A detailed chemistry analysis was performed to determine the pH in the various containment pools following a LOCA. The methodology used is consistent with NUREG/CR-5950, "Iodine Evolution and pH Control" [Ref. 7-35]. NUREG/CR-5950 discusses a number of chemicals that would potentially affect the post-accident chemistry in the containment pools. Each of the contributors is discussed below. The assumptions used to calculate the chemical makeup of fission products available for pH analyses are identical to those used to calculate the core average source term documented in DCD, Tier 2, Appendix 15B.

Carbon Dioxide:

Carbon dioxide (CO_2) depresses the pH of pure water by absorption. Carbonic acid (H_2CO_3) is a weak acid and is insignificant compared to the other acids produced in the primary containment

following a LOCA. However, the pool pH may be depressed below 7.0 during normal operations by the absorption of CO_2. NUREG/CR-5950, Subsection 2.2.3 states that pure water will attain a pH approaching 5.65 due to absorption of CO_2 from air and the subsequent formation of carbonic acid. The initial pH is assumed to be 5.7. As such, the effects of carbon dioxide are considered and bounded by evaluations assuming the minimum pool pH allowed by specifications. No detailed calculations explicitly accounting for CO_2 were performed.

Cesium Hydroxide:

Cesium hydroxide (CsOH) is a strong base introduced into the primary containment and subsequently to the containment pools with the release of cesium post accident. The production of this base is considered within this assessment. The pH analyses scavenge the cesium necessary to form a CsI molecule for every ion released; however, there is significantly more cesium than iodine (on a molar basis). The excess cesium would most likely react with water to produce hydrogen gas and CsOH, like all the metal water reactions. For the main analysis, 50% of the cesium that is not in the chemical form of CsI is assumed to exit the RCS in the form of cesium hydroxide (CsOH). [[

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Hydriodic Acid:

Hydriodic acid (HI) is a strong acid introduced into the primary containment with the release of post accident iodine. Per Section 4.5 of NUREG-1465 and Subsection 2.2.2 of NUREG/CR-5950, no more than 0.15% of the core iodine inventory is released from the RCS in this chemical form. As such, the production of this acid is considered within this assessment. In the performed analyses, HI was included in the database of Chemsheet compounds, but all iodine was released from the core during the core heatup phase as CsI. Further, the calculated mole fraction of HI in the gas phase in the containment was calculated to be negligible. [[

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Hydrochloric Acid:

Hydrochloric acid (HCl) is also a strong acid which is produced by the radiolysis of chloride-bearing cable insulation during accidents. The production of this acid is considered within this assessment. Pyrolysis of chloride-bearing cable insulation produces HCl as well; however, only at temperatures near 570°K (572°F) per Subsection 2.2.5.3 of NUREG/CR-5950. Because the RB primary containment temperature is evaluated to be significantly less than 570°K (572°F), pyrolysis is not considered within this assessment. [[

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The production of HCl by irradiating cables is estimated to be 1.0×10^{-3} mol per kg (4.6×10^{-4} mol per lb) of insulation per Mrad [Ref. 7-31]. This estimate is based on the model description of electrical cable and a radiation G value of 2.1. [[

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Dose rates and doses were determined for the containment using a simple RADTRAD model with the 60 isotopes used for off-site doses. RADTRAD was used to determine the radioactivity that remains airborne in the containment volume, and the airborne concentration was determined. The dose rate formulas from Regulatory Guide 1.3 [Ref. 7-36] were then used to determine submersion doses.

Infinite Cloud (Cloud centered)

$$ {}_{\beta}D_{\infty}'\left(\text{R}/\text{s}\right) = 0.457\overline{E}_{\beta}\left(MeV\right)\chi\left(\text{Ci}/\text{m}^3\right) $$
$$ {}_{\gamma}D_{\infty}'\left(\text{R}/\text{s}\right) = 0.507\overline{E}_{\gamma}\left(MeV\right)\chi\left(\text{Ci}/\text{m}^3\right) $$

Semi-infinite Cloud (Surface body)

$$ {}_{\beta}D_{\infty}'\left(\text{R}/\text{s}\right) = 0.23\overline{E}_{\beta}\left(MeV\right)\chi\left(\text{Ci}/\text{m}^3\right) $$
$$ {}_{\gamma}D_{\infty}'\left(\text{R}/\text{s}\right) = 0.25\overline{E}_{\gamma}\left(MeV\right)\chi\left(\text{Ci}/\text{m}^3\right) $$

Because this application is for cables, the cables themselves would provide self-shielding for beta radiation; therefore, the "semi-infinite cloud" model was used for beta dose rates. Due to the penetrating nature of gamma radiation, self-shielding of gamma is negligible. However, this penetrating ability also makes the "infinite cloud" model overly conservative. To account for the finite volume of the containment, a finite model geometry factor (GF) was applied. NUREG/CR-6604 [Ref. 7-17] provides such a factor for main control room dose calculations:

$$ GF = \frac{1173}{V^{0.338}} $$

with the volume (V) in the units of cubic feet. Accounting for the GF, the containment dose rates then become

$$ {}_{\gamma}D_{f}'\left(\text{R}/\text{s}\right) = \frac{0.507\overline{E}_{\gamma}\left(MeV\right)\chi\left(\text{Ci}/\text{m}^3\right)}{GF}. $$

Dose rates and total-integrated doses (TID) were then determined, which in turn were used to determine the HCl released as a result of radiolysis. The values provided in Table 4.1 account for the 25% safety factor applied for the revised removal coefficients and additional Accident Scenarios (as discussed in Section 6 of Reference 7-31c).

Table 4.1
Containment Airborne Dose Rates and Doses

Time (hr)	Beta Dose Rate (rad/hr)	Gamma Dose Rate (rad/hr)	Beta TID (rad)	Gamma TID (rad)
0.83	2.18E+06	6.75E+05	8.90E+05	2.78E+05
1.23	8.60E+06	2.13E+06	3.05E+06	8.36E+05
1.83	1.50E+07	3.31E+06	1.01E+07	2.46E+06
2.33	1.90E+07	3.94E+06	1.86E+07	4.28E+06
3.00	1.58E+07	2.94E+06	3.03E+07	6.58E+06
6.00	1.01E+07	1.38E+06	6.90E+07	1.30E+07
8.33	8.55E+06	9.40E+05	9.09E+07	1.58E+07
12.00	7.16E+06	5.94E+05	1.20E+08	1.85E+07
24.33	5.23E+06	2.78E+05	1.96E+08	2.39E+07
720.33	1.51E+05	4.81E+03	2.06E+09	1.22E+08

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Nitric Acid

Nitric acid (HNO_3) is also a strong acid that is introduced into the primary containment with the release of post accident source terms. This acid is produced by irradiation of air and water. According to the NUREG/CR-5950 report, the radiation G value for nitric acid production is 0.007 molecules/100 eV and this value corresponds to 7.3×10^{-6}mol HNO_3/L/Mrad. The dose rates and doses presented previously in Table 4.1 were also used to evaluate the HNO_3 production. The decrease of the activities was estimated to be linear between given times. The total dose is taken as a sum of β- and γ-doses.

Nitric acid is also produced within the pools themselves. As the event progresses, the LDW becomes submerged from steam that condenses on containment structures, and also from fluid through the break (AS-1 and AS-2). The element masses, as a function of time in the pools of WW, LDW, GDCS and RPV, were obtained for each Accident Scenario from the applicable MELCOR results. The reactor core radioactive inventory was calculated via the ORIGEN computer code (See DCD, Tier 2, Appendix 15B). The detailed ORIGEN calculations contain inventories for several time periods, including 0 seconds, 1 hour, 1 day and 30 days. Radioactive decay calculations were performed using lognormal interpolation to solve for the inventory at other desired time steps.

For dose rates in water, the most significant elements are Cs, I, Rb, Te, Se, Ba and Sr. Each element's specific mass fraction was obtained by dividing the MELCOR mass at each time under consideration with the ORIGEN mass at time 0 seconds. Multiplying this mass fraction value by the isotope specific inventory value of ORIGEN results in the corresponding MELCOR activity value. Finally, the dose rate is determined by multiplying the MELCOR activity by the isotope specific dose factor and dividing by the water volume.

The MELCOR activity values are calculated using the following equation:

$$A(t)_{isotope}^{MELCOR} = A(t)_{isotope}^{ORIGEN} \cdot \frac{m(t)_{class}^{MELCOR}}{m(0)_{class}^{ORIGEN}}$$

Where:

$A(t)_{isotope}^{MELCOR}$ is the activity of the isotope in MELCOR [Bq],

$A(t)_{isotope}^{ORIGEN}$ is the activity of the isotope in ORIGEN [Bq],

$m(t)_{class}^{MELCOR}$ is the mass of the class in MELCOR [kg], and

$m(0)_{class}^{ORIGEN}$ is the mass of the class in ORIGEN [kg].

The dose rate in the pool is calculated as follows:

$$D(t)_{isotope} = \frac{A(t)_{isotope}^{MELCOR} \cdot DF_{isotope}}{V}$$

Where:

$D(t)_{isotope}$ is the dose rate of the isotope [Gy/h],

$DF_{isotope}$ is the dose conversion factor for water immersion [Gy/Bq-h-m^{-3}],

V is the liquid volume in the pool [m^3].

Dose rates were determined assuming an infinite water pool. A simplistic approach was adopted and the Federal Guidance Report (FRG) 12 Table III.2 [7-43] effective dose conversion factors were used. This is reasonable due to small differences in the photon mass energy-absorption coefficients between water and human tissues. For example, the preceding coefficient for a 1 MeV photon is 0.031 in water and 0.0308 in muscle. The results of the water immersion dose rates are presented in Tables 4.2, 4.3, and 4.4 for AS-1, AS-2, and AS-3, respectively [Ref. 7-31c].

Table 4.2
Pool Dose Rates for AS-1 (rad/hr)

Time (s	WW	GDCS	DW	RPV
[[				
				]]

Table 4.3 Pool Dose Rates for AS-2 (rad/hr)				
Time [s	**WW**	**GDCS**	**DW**	**RPV**
[[				
				]]

Table 4.4
Pool Dose Rates for AS-3 (rad/hr)

Time [s	WW	GDCS	DW	RPV
[[				
				]]

The total HNO_3 generated in the containment atmosphere and the containment pools in 30 days, using this methodology, is presented in Table 4.5.

Table 4.5 HNO_3 Generation (mol in 30 days)			
Location/Contributor	**AS-1**	**AS-2**	**AS-3**
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			]]

Sodium Pentaborate

Sodium pentaborate ($Na_2O*5B_2O_3*10H_2O$) is a buffering solution primarily utilized as a backup means of criticality control within a post accident reactor vessel. Sodium pentaborate is supplied by the Standby Liquid Control (SLC) system. The SLC system would be used as an injection source following confirmation of a LOCA. Buffering by the SLC system is considered in this evaluation.

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4.1.2.2 Pool pH Determination

Pool pH is calculated using the computer code ChemSheet. Chemical reactions taking place in multiphase systems are calculated by the Gibbs Energy minimization method. As a result of minimization, the equilibrium composition of the system is obtained. The method requires that temperature, pressure and initial composition (initial amounts of species like $H_2O(l)$, HCl(g), NaOH(s)) are known and given as input parameters.

The Gibbs energy minimization method is a general method; therefore, knowledge of the exact reaction paths between the chemical species is not required. Chemical species are linked together by their elemental composition, i.e., the elements (like C, H, O) that they are composed of. The equilibrium composition is the composition that gives the minimum Gibbs Energy without violating the elementary mass balances (mole number of each element in equilibrium composition must be same as in initial composition). As such, the equilibrium calculation corresponds to the mathematical problem of finding the global minimum of a constrained function.

In many cases, the real systems are not at global equilibrium. There can be many physical mechanisms like mass transfer between the phases that constrain the reactions. In the case of large water containers and slow (relative to volume) flows between them, it can be assumed that the time scale is long enough for the system to be close to or at equilibrium (vapour/liquid equilibrium).

The Gibbs Energy is a function of temperature, pressure and composition. Gibbs Energy for a multiphase system can be given as:

$$G = \sum_p \sum_i n_i^p \mu_i^p$$

Where: n_i^p is amount of species i in phase p and μ_i^p is its chemical potential. The chemical potential can be separated into ideal and non-ideal terms:

$$\mu_i = \mu_i^0 + RT \ln(\gamma_i x_i)$$

Where:

μ_i^0 standard chemical potential of species i

R gas constant

T temperature

γ_i activity coefficient of species i

x_i mole fraction of species i

The activity α_i of a species is the product of activity coefficient and mole fraction: $\alpha_i = \gamma_i x_i$.

Standard chemical potential of a species is a function of temperature (and pressure) and is typically given as a polynomial where the coefficients of the polynomial are fitted from measured data:

$$G_i(T) = A + BT + CT\ln(T) + DT^2 + ET^3 + \frac{F}{T}$$

Where: T is the temperature and A through F are the coefficients. Coefficients are tabulated and listed in handbooks or stored to thermodynamic database programs from which they can be retrieved.

Standard chemical potential of a species can also be calculated from measured formation enthalpy $H_{i,298}^0$, standard entropy $S_{i,298}^0$, and heat capacity polynomial $C_{pi}(T)$ (fitted from measured data).

The thermodynamic software that ChemSheet uses (ChemApp) for equilibrium calculations enables both ways for entering the needed thermodynamic data.

Activity coefficient of a species is typically a function of temperature and phase composition. In this study, the gas phase is assumed to be ideal; therefore, the activity of a gas phase species corresponds to its partial pressure. The aqueous phase on the other hand contains relatively concentrated aqueous solutions, which can be strongly non-ideal and the realistic calculation of solution equilibrium necessitates the modelling of excess thermodynamic properties of the system process as a function of solution composition within the temperature range of operation. The Pitzer model [Ref. 7-33], which is widely used, was also applied in this work (and is included in ChemApp as a selectable solution phase model).

The chemical potential for the electrolytic dissociation of a salt, e.g., NaOH in polar solute can be described as follows:

$$\mu_{NaOH} = \mu^0_{NaOH} + RT\ln(\alpha_{NaOH}) = \mu^0_{Cs^+} + \mu^0_{OH^-} + RT\ln(\alpha_{NaOH})$$

The activity of dissociating salts in polar solutions is expressed as the product of the concentration (molarity or molality, m) of ionic species and their mean activity coefficient, $\gamma_{\pm}$:

$$\alpha_{NaOH} = \gamma^2_{\pm} m_{Na^+} m_{OH^-}$$

In Pitzer formalism, the mean activity coefficient is expressed by:

$$\ln(\gamma_{\pm}) = \frac{G_E}{RT} = n_w f(I) + \left(\frac{1}{n_w}\right)\sum_M\sum_X \lambda_{MX} n_M n_X + \left(\frac{1}{n_w}\right)^2 \sum_M\sum_X\sum_N \mu_{MXN} n_M n_X n_N$$

Where:

G_E	excess Gibbs energy
$f(I)$	Pitzer function, dependent only from ionic strength
n_w	mol number for water
n_M	mol number for species M
n_X	mol number for species X
n_N	mol number for species N
λ_{MX}	binary interaction parameter
μ_{MXN}	ternary interaction parameter

Pitzer's equation for the aqueous phase is a virial coefficient expansion of Debye–Hückel's theory and is capable of describing the ionic activities of aqueous species in concentrated solutions usually up to a strength of 20 m [Ref. 7-34]. The use of Pitzer's equation is restricted to the amount of existing data on the solutions.

In the model, the following ion pairs had binary or tertiary Pitzer interaction parameters:

Cl(-a)	H(+a)	Na(+a)
Cl(-a)	H(+a)	
OH(-a)	H(+a)	Na(+a)
OH(-a)	H(+a)	
OH(-a)	Na(+a)	
Cl(-a)	OH(-a)	
Cs(+a)	I(-a)	
Cs(+a)	OH(-a)	
Cs(+a)	H(+a)	

H(+a) I(-a)

After calculating the equilibrium composition, the pH of an aqueous solution can be calculated from H^+ ion activity:

$$pH = -\log_{10} \alpha_{H^+}$$

A typical curve for a solution initially containing acid and then titrated with a base looks like the curve in Figure 4.1.

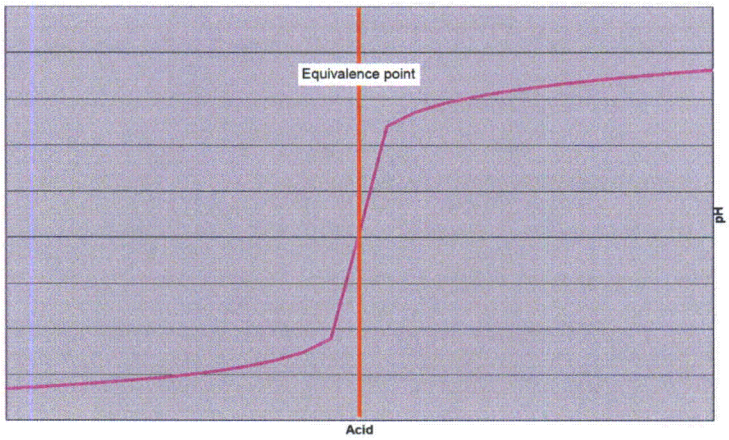

Figure 4.1. Typical Titration Curve

The pH scale is logarithmic which means that, in order to change the pH by one, the concentration of H^+ ion must change by 10 times. The equivalence point is the pH where the added base fully neutralizes the acid initially in the solution. When the pH increases and gets closer to the equivalent point, the number of free H^+ ions is also decreased. This means that as the concentration of free H^+ ion gets smaller, then the same added base amount has a more striking effect on the H^+ ion concentration and pH. Typically, the pH changes very rapidly around the equivalence point.

4.1.2.3 pH Evaluation Results

A number of pH scenarios were reviewed as documented in VTT-R-04413-06 [Ref. 7-31a]. The scenario most applicable to the ESBWR (Case A of Reference 7-31a) calculates the impact of both HCl and HNO_3.

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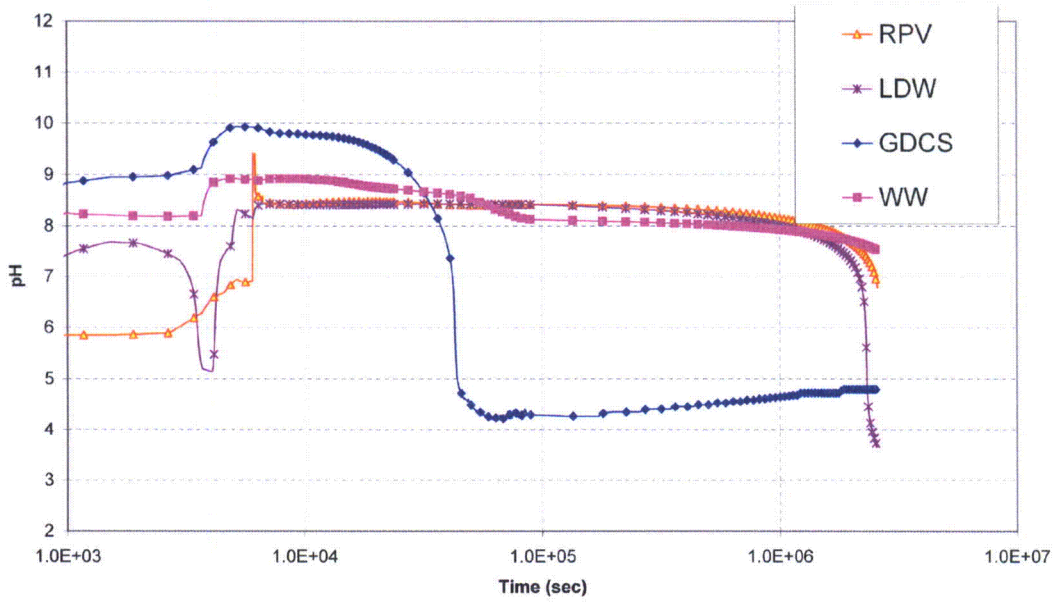

Figure 4.2. Pool pH Calculation Results for Accident Scenario 1

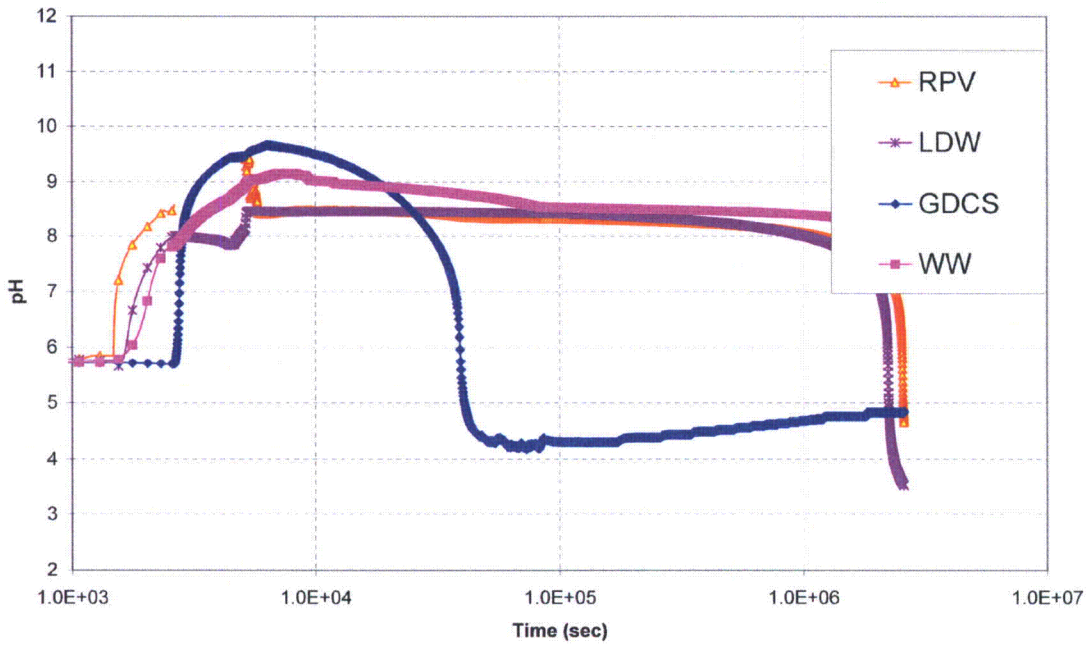

Figure 4.3. Pool pH Calculation Results for Accident Scenario 2

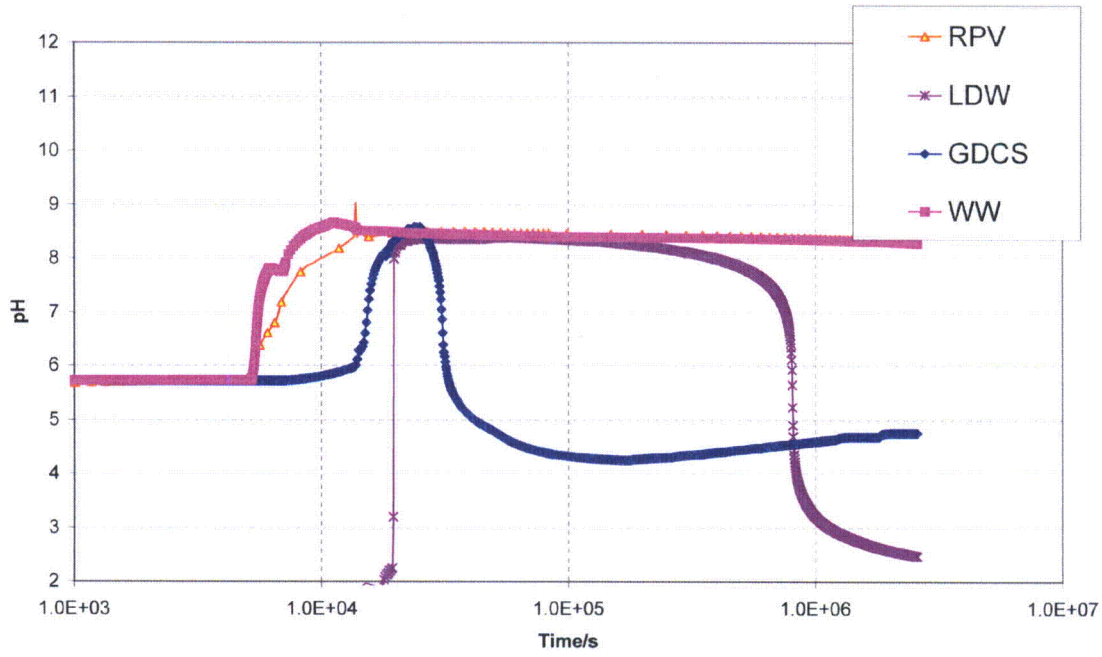

Figure 4.4. Pool pH Calculation Results for Accident Scenario 3

4.1.3 Release Timing

Regulatory Guide 1.183 states that for BWRs the gap release is assumed to begin 2 minutes into the event and last for 30 minutes, and the EIV release is assumed to begin at ~30 minutes and last for 1.5 hours. The dose calculations conservatively neglect the coolant release phase and assume that the gap release phase begins at the onset of the event. The release fractions assumed for each chemical group are based on Regulatory Guide 1.183, Table 2.

Early in the event, the drywell pressure is high due to the initial blowdown of the RPV. Since the PCCS flow is dependent on the drywell pressure, PCCS flow would be very high early in the event. If removal coefficients were determined for this time period, they would likely be over-conservative. Therefore, the removal coefficients are determined to correspond to the onset of the EIV release phase.

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Figure 4.5. MELCOR CsI Released for AS-1

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Figure 4.6. MELCOR CsI Released from Fuel for AS-2

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Figure 4.7. MELCOR CsI Released for AS-3

4.2 PCCS AS A FISSION PRODUCT REMOVAL MECHANISM

4.2.1 Initial PCCS Testing for the SBWR

Early in the design phase for the PCCS condensers and the SBWR, concerns arose with respect to the deposition of aerosols on condenser tubing and the potential impact to the heat removal capabilities of the PCCS. Several tests were performed to quantify the aerosol deposition rates and the detrimental impact to the heat removal capabilities of the condenser. The tests confirmed that the heat exchangers are able to perform as required even with deposition of aerosols. They also confirmed that the heat exchangers are effective at removing aerosols as well.

Testing to determine the impact of aerosol deposition in PCCS condenser tubes was performed as documented in ENE53/46/2000, "Investigation on Aerosol Deposition in a Heat Exchanger Tube" [Ref. 7-25]. VTT Energy in Finland performed the testing. [[

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An independent MELCOR analysis was performed to benchmark the ability of MELCOR to predict deposition in the PCCS tubes against the VTT test discussed above. This benchmark analysis was performed independent of the MELCOR analysis for the overall removal from containment.

4.2.2 MELCOR Modeling: Heat and Mass Transfer

The heat and mass fluxes of the system were estimated using a simple plug flow balance model with appropriate Nusselt (*Nu*) and Sherwood (*Sh*) numbers [Ref. 7-18]. For the gas temperature T [°K], if assumed that the latent heat associated with steam condensation is not conducted to the gas:

$$\frac{dT}{dx} = -\frac{Pq^T}{\dot{m}_w c_{pw} + \dot{m}_n c_{pn}}$$

Analogously, for the water film temperature T_l [°K],

$$\frac{dT_l}{dx} = -\frac{P\left[-q^{Ts} + q^T + Lq^m + c_{pw}(T - T_l)\right]}{\dot{m}_l c_{pl}}$$

Here L is the latent heat [J/kg], which is calculated at T_l. It was assumed that the water film temperature profile is linear. T_l is the average liquid temperature [°K] and T the average gas temperature [°K]. P is the perimeter of the heat exchanger tube [m], c_{pn} the nitrogen gas heat capacity [J/kg-°K], c_{pw} the water vapor heat capacity [J/kg-°K] and c_{pl} the liquid water heat capacity [J/kg-°K], respectively. The mass fluxes for the water vapor, nitrogen gas and liquid water are $\dot{m}_w$, $\dot{m}_n$ and $\dot{m}_l$ [kg/s]. The heat fluxes q^T and q^{Ts} [W/m^2] are calculated from:

$$q^T = Nu \cdot k_g \frac{T - T_{ls}}{d_h}$$

and

$$q^{Ts} = k_l \frac{T_{ls} - T_s}{\delta}$$

Where: k_g and k_l are the thermal conductivities of the gas-vapor mixture and liquid water [W/m-°K], respectively. d_h is the hydraulic diameter of the heat exchanger tube [m], T_{ls} the temperature at the liquid film surface [°K] and T_s the temperature at the tube surface [°K]. The assumption of the linear temperature profile across the liquid film satisfies:

$$T_l = \frac{T_{ls} + T_s}{2}$$

The liquid film thickness δ [m] can be approximated by [Ref. 7-18]:

$$\delta = \left(\frac{3\mu\dot{m}_l}{\rho_l^2 g d_h}\right)^{1/3}$$

Where: μ_l is the liquid viscosity [N/s-m^2], ρ_l the liquid density [kg/m^3]and g the gravitational acceleration [m/s^2].

Besides the energy balance equations, the mass balances are also formulated for solving a solution of the system simultaneously. For the nitrogen $\dot{m}_n$, water vapor $\dot{m}_w$ and liquid water $\dot{m}_l$ mass fluxes [kg/s] we obtain:

$$\frac{d\dot{m}_n}{dx} = 0$$

$$\frac{d\dot{m}_w}{dx} = -Pq_w^m$$

$$\frac{d\dot{m}_l}{dx} = Pq_w^m$$

The water vapor condensation mass flux q_w^m [kg/s-m^2] is calculated from:

$$q_w^m = Sh \cdot D \cdot \frac{\rho_w - \rho_{ws}}{d_h}$$

Where: D is the diffusion coefficient of water vapor in nitrogen [m^2/s], ρ_w the mass concentration of water vapor in the gas [kg/m^3] and ρ_{ws} the equilibrium vapor mass concentration at the film surface temperature T_{ls} [kg/m^3]. The mass concentration and mass flux are related to the following:

$$\dot{m}_w = \rho_w UA$$

Where: U is the gas velocity [m/s] and A the cross-sectional flow area [m^2].

For the laminar and turbulent flow regimes, different correlations for the Nusselt and Sherwood numbers [Ref. 7-18] were chosen:

For the laminar flow regime,

$$Nu = 3.66$$

$$Sh = 3.66$$

For the turbulent flow regime, the Dittus-Boelter correlations for were used:

$$Nu = 0.023 \cdot \mathrm{Re}^{0.8} \cdot \mathrm{Pr}^{0.3}$$

$$Sh = 0.023 \cdot \mathrm{Re}^{0.8} \cdot Sc^{0.3}$$

Where: Pr, Re and Sc are the Prandtl, Reynolds and Schmidt number, respectively.

4.2.3 MELCOR Modeling: Particle deposition

In addition to steam condensation, the model includes the particle deposition onto the heat exchanger tube wall. The deposition mechanisms to be considered are: diffusiophoresis, thermophoresis, gravitational settling and turbulent eddy impaction.

4.2.3.1 Diffusiophoresis

Diffusiophoresis is flow of aerosol particles down a concentration gradient of gas or vapor due to bombardment of particles by the gas or vapor molecules as they diffuse down the same gradient. To maintain a constant total pressure near a condensing surface, the concentration gradient of vapor is balanced by an equal and opposite concentration gradient of non-condensable gas. The effect of gas molecules diffusing away from the surface on the transport of aerosol particles is cancelled out by an aerodynamic flow of gas towards the surface (Stefan flow). Therefore, the diffusiophoretic deposition velocity of particles onto the walls, u_p^{DPH} [m/s], is directly proportional to the water vapor condensation rate q_w^m [kg/m^2-s] [Ref. 7-27]:

$$u_p^{DPH} = \frac{x_w\sqrt{M_w}}{x_w\sqrt{M_w} + x_n\sqrt{M_n}} \frac{q_w^m}{\rho_w}$$

Where: x_w and x_n are the mole fractions and M_w and M_n the molecular weights of water and nitrogen [g/mol], respectively and ρ_w is the mass concentrations of water [kg/m^3] in the gas flow. Diffusiophoresis is approximately independent of particle size.

4.2.3.2 Thermophoresis

Thermophoresis is the result of the temperature gradients. On the hotter side, gas molecules colliding with particles carry, on average, a higher momentum than on the colder side, thus causing a net transport in the direction of colder temperature. The thermophoretic deposition velocity is calculated using a generally accepted formula over a wide range of particle diameters [Ref. 7-28]:

$$u_p^{TPH} = -K\frac{\nu}{T}\nabla T$$

Where:

$$K = 2C_s \frac{(\alpha + C_t Kn)Cn}{(1+3C_m Kn)(1+2\alpha+2C_t Kn)}.$$

Here C_s=1.147, C_t=2.20, C_m=1.146, Cn is the Cunningham slip correction factor, ν the kinematic viscosity [m^2/s], T temperature [K], $\alpha = \lambda_g/\lambda_p$ is the ratio of gas to particle thermal conductivities, and Kn the Knudsen number. The Knudsen number $Kn = l_g/r_p$ is the ratio of the gas mean free path to the particle radius. In above equations, the thermophoretic velocity in the free molecular regime is interpolated with the corresponding expression in the continuum regime. Because thermophoresis is proportional to the temperature gradient, it is closely related to heat transfer. The actual value for the temperature gradient at the surface, which is required for calculating the thermophoretic deposition velocity u_p^{TPH}, can be obtained using the heat transfer correlations for the Nusselt number Nu, which is the dimensionless temperature gradient at the surface. Consequently, we obtain the following simple equation:

$$u_p^{TPH} = -K\nu Nu\frac{T - T_{ls}}{Td_h}$$

4.2.3.3 Gravitational settling

Gravitational settling is caused by the effects of gravity on the particles. Settling affects particle transport in the PCC only if the tubes are not vertical. For spherical particles of density ρ_{den_p} [kg/m^3] and diameter d_p [m] in the range of 1-100 μm, the gravitational deposition velocity can be calculated from [Ref. 7-29]:

$$u_p^G = \frac{\rho_{den\_p} d_p^{\,2} \boldsymbol{g}}{18\mu} \cdot \boldsymbol{n}$$

Where: $\boldsymbol{g}$ is the gravitational acceleration [m/s^2] and $\boldsymbol{n}$ the unit vector normal to the tube wall. For submicron particles, gravitational deposition can be considered negligible.

4.2.3.4 Turbulent impaction

Turbulent impaction is an important deposition mechanism for large particles, when the boundary layer between the surface and the host flow is turbulent. Inside the turbulent boundary layer turbulent eddies have a velocity component, which is normal to the main flow. Eddies may give enough momentum for particles to cross the laminar sublayer and finally to deposit on the wall.

At present, there is no generally accepted mechanistic model available for turbulent deposition. Rough predictions can be made by using experimental correlations. The experimental deposition rate is usually given in such a way that the dimensionless deposition velocity u^+ is plotted as a function of the dimensionless stopping distance τ^+. The dimensionless stopping distance τ^+ characterizes the ability of the particles to react to sudden changes of the fluid. In constant conditions, it depends on particle size and other flow variables in the following way:

$$\tau^+ = \frac{1}{36} \frac{\rho_{den\_p}}{\rho_{den\_g}} \left(\frac{d_p}{d_h} \right)^2 \mathrm{Re}^2 \, f(\mathrm{Re})$$

Where: f is the Fanning friction factor. The deposition velocity u^+ is the actual velocity, with which the particles deposit, normalized with "wall variables" [Ref. 7-30]:

$$u^+ = \frac{u_p^{TUR}}{U\frac{f}{2}}$$

Submicron range particles ($\tau^+ < 0.2$) tend to follow the streamlines of fluid motion. This means that in the absence of thermophoresis, Brownian motion is the mechanism mainly responsible for deposition. Therefore, it is assumed that u^+ is independent of τ^+ and is a function of Schmidt number only:

$$u^+ = 0.086 Sc^{-0.7}$$

($Sc = \nu / D$, where: ν is the kinematic viscosity of the fluid [m^2/s] and D the Brownian diffusivity [m^2/s])

However, when τ^+ is greater than 0.2, the deposition velocity becomes independent of *Sc*. Particles in this range diffuse towards the wall due to radial velocity fluctuations (turbulent diffusion) and then deposit onto the wall by a free-flight mechanism through the viscous sublayer. This is caused by the inability of the particles to follow the turbulent eddies in the vicinity of the wall. This inability can be conveniently described by the concept of a stopping distance. In this range, the experimental deposition data can be roughly correlated using the following equation:

$$u^+ = 3.5 \cdot 10^{-4} \tau^{+2}$$

Neither of the correlations above are applicable for dimensionless stopping distances that are very small, however as the particle stopping distance increases beyond $\tau^+ > 30$, the particles are too large to respond to the fluid fluctuations, and the $u^+(\tau^+)$ curve levels off to an approximately constant value 0.17 (see Reference 7-30 for details). This is also approximately the point, where gravitation starts to play an increasingly important role in particle depositions dynamics.

The reduction in the particle mass flux $\dot{m}_p$ [kg/s], due to deposition can be obtained from:

$$\frac{d\dot{m}_p}{dx} = -\left\{P\rho_p\left(u_p^{DPH} + u_p^{TPH} + u_p^{G} + u_p^{TUR}\right)\right\}$$

where ρ_p is the particle mass concentration [kg/m^3] in the gas flow.

4.3 CONTAINMENT PLATEOUT

The LOCA dose consequence calculation credited the natural deposition of particulate and elemental iodine on containment surfaces.

4.3.1 Elemental Iodine Plateout

The elemental iodine coefficient is based on guidance found in SRP 6.5.2 [Ref. 7-26]. Specifically, the iodine removal rate constant for a particular compartment "n" will be based on the following formula:

$$\lambda_{elem} = k_g\left(\frac{A}{V}\right)$$

Where:

λ_{elem} = removal rate constant for elemental iodine due to surface deposition,

k_g = average mass transfer coefficient,

A = surface area for deposition, and

V = volume of the contained gas.

The area used in the analysis is the wall surface area of the building and the floor area for elevation 17500. Other surfaces, such as the bioshield wall for the drywell (above Elevation 17500), will conservatively be neglected. The inside diameter of the drywell below elevation 17500 is 9292 mm:

$$A_{DW,<17500} = \pi DH = 803.5\ m^2$$

Only 50% of the floor area will be credited (to account for the GDCS Pools, the RPV, etc.). The diameter of the drywell is 33.5 m, therefore,

$$A_{DW,17500} = 50\% * \pi r^2 = 440.7\ m^2$$

$$A_{tot} = 803.5\ m^2 + 440.7\ m^2 = 1244.2\ m^2 = 13392.5\ ft^2$$

The removal rate constant will be taken as 0.137 cm/sec (16.18 ft/hr) based on NUREG/CR-0009, Page 17 [Ref. 7-32]. The upper drywell free air volume is 6016 m^3 and the LDW net airspace volume is 1190 m^3, for a total assumed drywell volume of 7206 m^3 (2.54E+05 ft^3).

$$\lambda_n = 16.18(ft/hr)\left(\frac{1.34E4\,ft^2}{2.54E5\,ft^3}\right) = 0.86hr^{-1}$$

This value is assumed to be independent of the Accident Scenario under consideration.

SRP 6.5.2 states that the maximum credit for removal from natural deposition should be limited to a factor of 200. The elemental iodine activity in containment, neglecting removal mechanisms other than natural deposition, can be expressed with the following generic formula:

$$A(t) = A(0)e^{-\lambda t} + S\left(\frac{1-e^{-\lambda t}}{\lambda}\right)$$

Where:

A = Airborne activity in containment at time "t,"

λ_{elem} = Containment removal coefficient (0.86 hr^{-1} for elemental iodine),

S = Normalized activity source in containment.

Since Regulatory Guide 1.183 release timing is assumed, "S" varies as a function of time depending on the release phase. Therefore, the activity in containment is:

$$A(t \le 0.5hr) = S_{gap}\left(\frac{1-e^{-\lambda t}}{\lambda}\right)$$

$$A(0.5 < t \le 2.0hr) = A(0.5hr)e^{-\lambda(t-0.5hr)} + S_{EIV}\left(\frac{1-e^{-\lambda(t-0.5hr)}}{\lambda}\right)$$

$$A(t \ge 2.0hr) = A(2.0hr)e^{-\lambda(t-2.0hr)}$$

The time that corresponds to a DF of 200 is ~7.35 hours; however, the dose calculations conservatively terminate deposition of elemental iodine at 6.5 hours to correspond to aerosol removal coefficients as discussed in Subsection 4.3.2.

4.3.2 Aerosol Iodine

The computer code RADTRAD has an internal option to use the Powers natural deposition model described in detail in NUREG/CR-6604 [Ref. 7-17] and NUREG/CR-6189 [Ref. 7-16]. The Powers model is comprised of simplified formulae that were developed for estimating the aerosol decontamination that can be achieved by natural processes in the containment of light

water reactors. The simplified formulae were derived by the correlation of the results of uncertainty analyses using Monte Carlo uncertainty analyses of detailed models of aerosol behavior under accident conditions. The DCD, Revision 1 LOCA dose analyses utilized the Powers model for natural deposition of particulate iodine in the drywell of the ESBWR; however, this analysis assumes deposition coefficients specific to the ESBWR.

This report, and its supporting analyses, utilized a slightly different approach in modeling the amount of radioactivity that is removed from the containment atmosphere as a result of natural deposition. The MELCOR analysis models removal of airborne aerosols by passive means (plateout, etc.) using processes similar to that discussed previously in Section 4.2. By modeling the various radioiodine removal mechanisms independently (natural deposition, removal via PCCS, etc.), this report utilized the MELCOR results to determine an integral removal coefficient. This was modeled via the "natural deposition" model in the RADTRAD computer code, utilizing the "user-defined coefficients" input option for the drywell compartment.

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The removal coefficients for the case with "no MSIV failure" were generally lower than those with the case for the assumed single failure of an MSIV. This assumption introduces significant conservatism into the dose analysis. Leakage through the MSIVs is assumed to occur in the dose calculation irregardless of the removal coefficient applied. For example, the particulate removal coefficient assumed from 2 to 3.5 hours for AS-1 is 0.6 hr^{-1}, which corresponds to the case with no MSIV failures. If the more accurate value of [[ ]]hr^{-1} were assumed, the calculated dose consequences would be somewhat lower. This approach of treating the variables as though they are independent is not physically possible; however, it simplifies the calculation by preventing multiple evaluations.

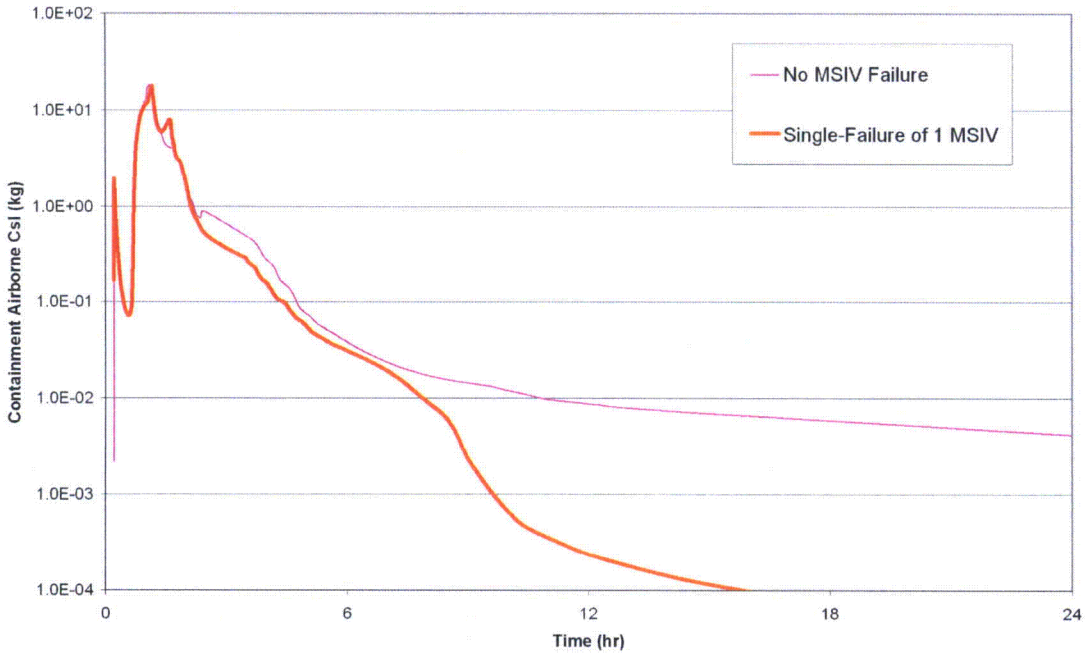

Figure 4.8. MELCOR Airborne CsI in Containment for AS-1

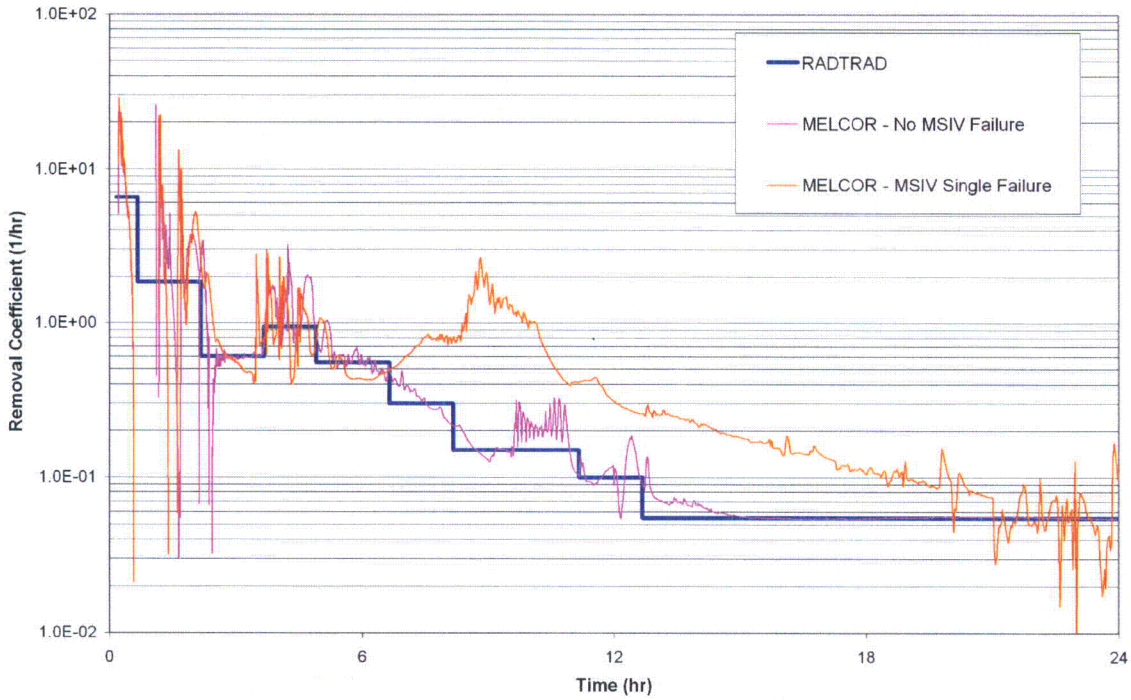

Figure 4.9. Containment Removal Coefficients for AS-1

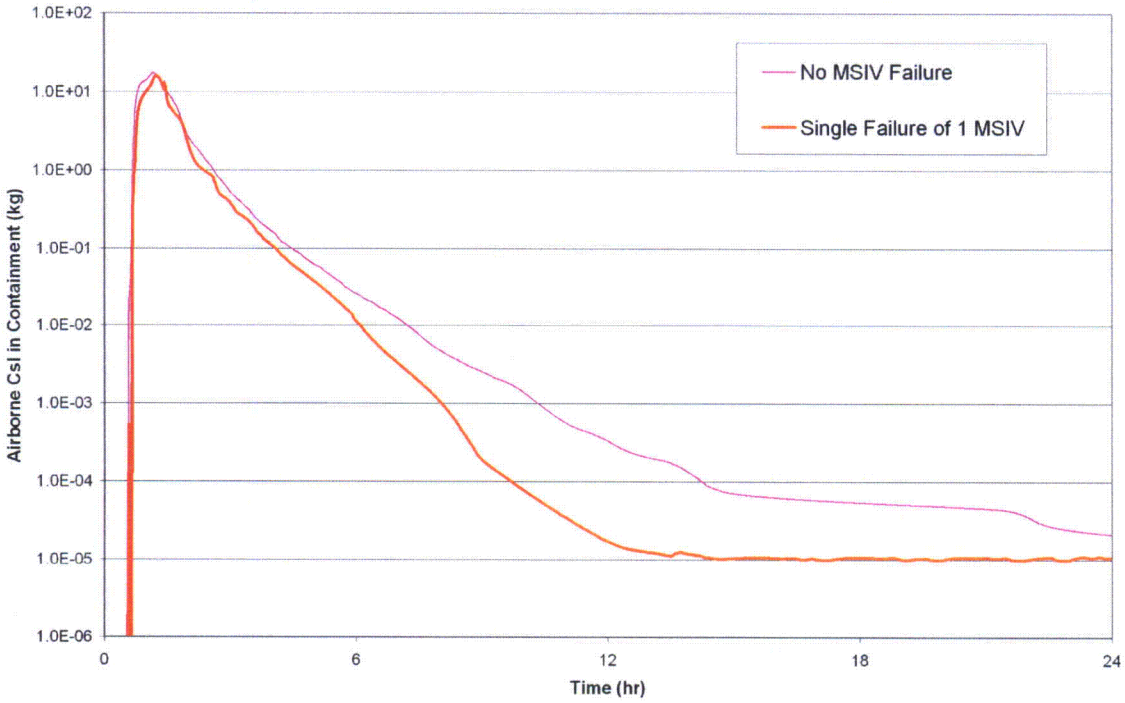

Figure 4.10. MELCOR Airborne CsI in Containment for AS-2

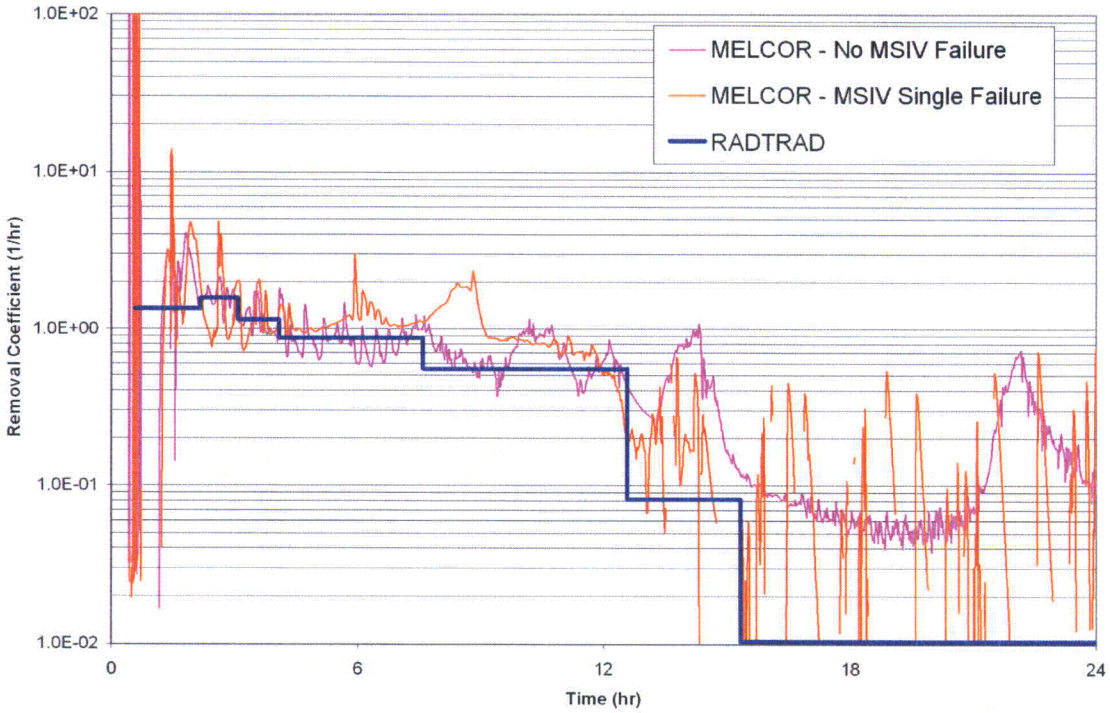

Figure 4.11. Containment Removal Coefficients for AS-2

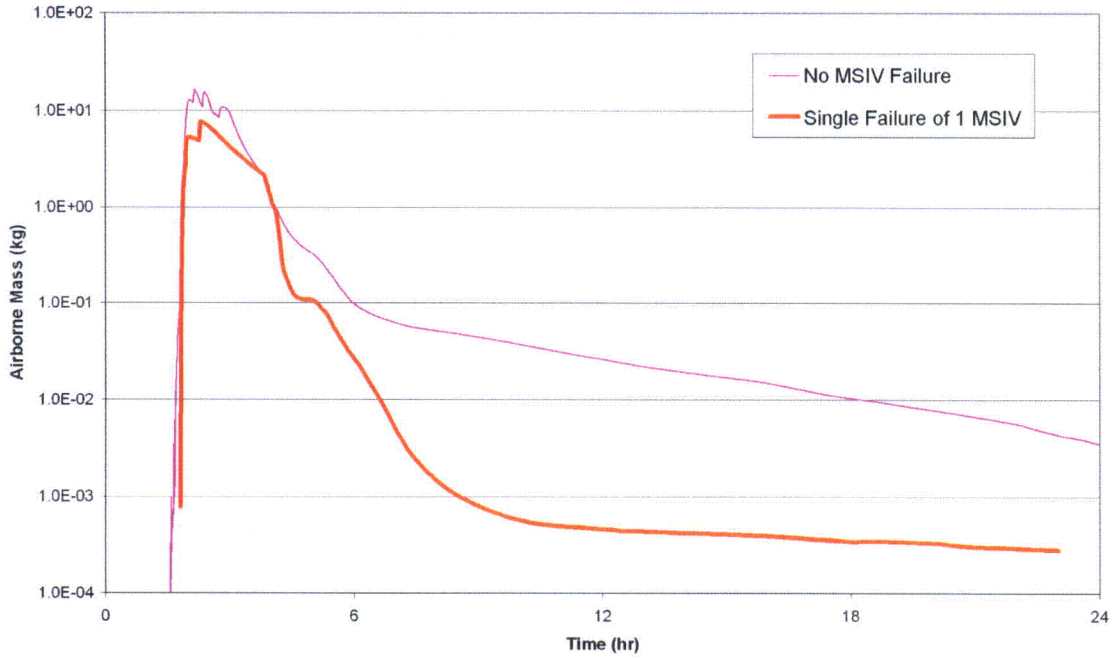

Figure 4.12. MELCOR Airborne Csl in Containment for AS-3

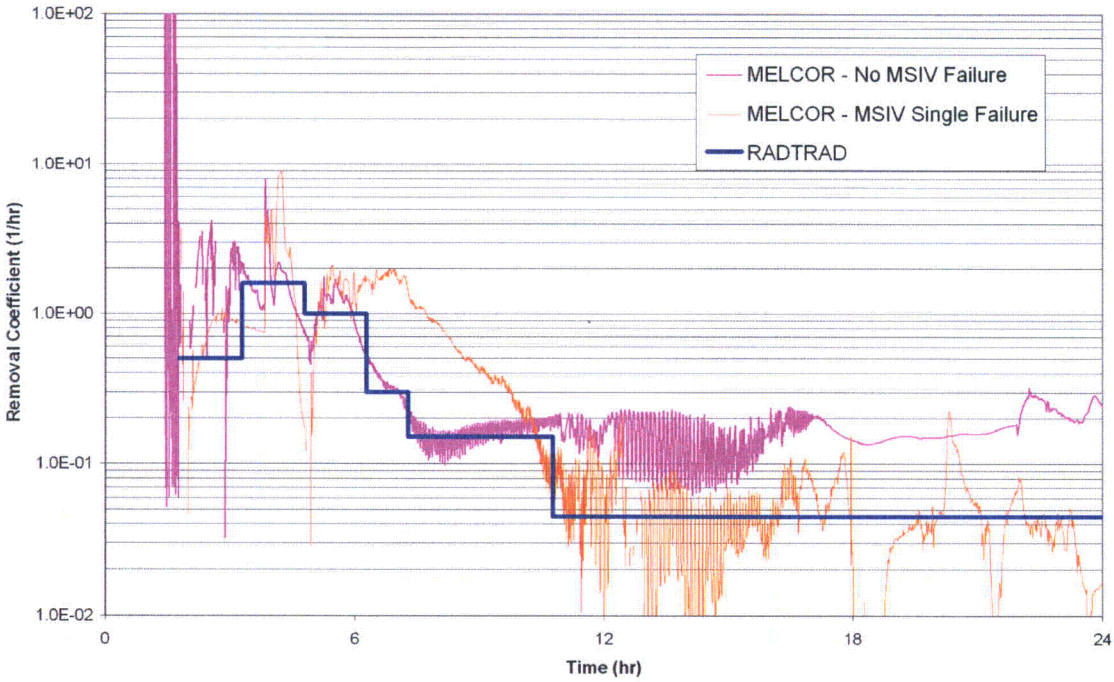

Figure 4.13. Containment Removal Coefficients for AS-3

Table 4.6
RADTRAD Removal Coefficients for AS-1

Time* [RADTRAD (hr)	No MSIV Failure			With MSIV Failure			Minimum λ (hr^{-1})	AS-1 Assumed λ (hr^{-1})
	Time [MELCOR] (hr)	Cont. Airborne Mass (kg)	Calculated λ (hr^{-1})	Time [MELCOR] (hr)	Cont. Airborne Mass (kg)	Calculated λ (hr^{-1})		
0.00	[[							6.500
0.50								1.850
2.00								0.600
3.50								0.950
4.75								0.550
6.50								0.300
8.01								0.150
11.01								0.100
12.51								0.055
24.00							]]	0.000

* RADTRAD time is adjusted to the onset of fuel damage (10 min. for AS-1).

** Airborne masses not relevant since alternate methodology used to calculate removal coefficients (presented previously).

Table 4.7
RADTRAD Removal Coefficients for AS-2

Time* [RADTRAD] (hr)	No MSIV Failure			With MSIV Failure			Minimum λ (hr^{-1})	AS-2 Assumed λ (hr^{-1})
	Time [MELCOR] (hr)	Cont. Airborne Mass (kg)	Calculated λ (hr^{-1})	Time [MELCOR] (hr)	Cont. Airborne Mass (kg)	Calculated λ (hr^{-1})		
0.00	[[							0.000
0.50								1.348
2.00								1.567
3.00								1.149
4.00								0.879
7.50+								0.739
11.00+								0.552
12.50								0.081
15.26								0.010
24.00							]]	0.000

* RADTRAD time is adjusted to the onset of fuel damage (5 min. for AS-2).

** Airborne masses not relevant since alternate methodology used to calculate removal coefficients (presented previously).

\+ The removal coefficient for 11 hours was conservatively assumed to begin at 7.5 hours due to the RADTRAD limitation of only 10 removal coefficients (one time-step was required to reduce the elemental removal coefficient at 6.5 hours as discussed in Subsection 4.3.1.

Table 4.8
RADTRAD Removal Coefficients for AS-3

Time* [RADTRAD] (hr)	No MSIV Failure			With MSIV Failure			Minimum λ (hr-1)	AS-3 Assumed λ (hr-1)
	Time [MELCOR] (hr)	Cont. Airborne Mass (kg)	Calculated λ (hr-1)	Time [MELCOR] (hr)	Cont. Airborne Mass (kg)	Calculated λ (hr-1)		
0.00	[[							0.000
0.50								0.500
2.00								1.600
3.50								1.000
5.00								0.300
6.00								0.150
9.50								0.045
22.72+							]]	0.000

* RADTRAD time is adjusted to the onset of fuel damage (5 min. for AS-3).

** Airborne masses not relevant since alternate methodology used to calculate removal coefficients (presented previously).

\+ Data is only available for 24 hours. Since AS-3 has a longer coolant release time (1.3 hours vs. ~5-10 minutes), the difference in time may not be negligible. As such, the particulate removal coefficient will be set to 0 at 22.72 hours in RADTRAD for AS-3.

4.4 MAIN STEAM ISOLATION VALVE LEAKAGE

Leakage past MSIVs typically bypasses secondary containment for BWRs; and therefore can be released untreated to the environment. To minimize the dose consequences from MSIV leakage many plants utilize a methodology developed by GE and the BWR Owner's Group (BWROG). This methodology is documented in NEDO-31858, "BWROG Report for Increasing MSIV Leakage Rate Limits and Elimination of Leakage Control Systems" [Ref. 7-20].

The BWROG methodology quantified the amount of deposition in both the main steam lines and the main steam lines' drain lines. The BWROG methodology was developed using the older dose consequence methodology based on TID-14844 [Ref. 7-4]. As such, the methodology may not be accurate for use with AST assumptions.

A detailed model was developed using the MELCOR computer code to determine the amount of deposition in the main steam lines, the main steam line drain lines, and the main condenser. The MELCOR model is discussed in Reference 7-31c. The report evaluated all three Accident Scenarios. A fourth Scenario was evaluated for the purposes of plateout in the main steam lines, main steam line drain lines, and the main condenser. Specifically, a main steam line break inside of containment was also reviewed because the amount of plateout in the main steam lines could conceivably affect that Scenario (identified as AS-4 for the purposes of the MSIV leakage discussion only). The amount of plateout in five volumes was calculated: (1) the main steam

lines between the RPV and the inboard MSIVs, (2) the volume between the MSIVs, (3) the volume down stream of the outboard MSIVs, (4) the main steam line drain lines, and (5) the main condenser. The amount of plateout calculated in each volume is presented in Table 4.9. Leakage was modeled directly from the containment to the main condenser. The main steam lines and main steam line drain lines were not explicitly modeled in the dose calculation. As such, only the overall reduction factor needs to be considered. The limiting Scenario was determined to be AS-2, which yielded a total plateout fraction of 99.7%. This analysis conservatively assumes a plate-out fraction of 99.3% (consistent with the ABWR design [Ref. 7-41]).

Table 4.9
Plateout Fractions for MSIV Leakage

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** Total retention is calculated as follows: $(1\text{-}F_{MSL\text{-}MSIV1})$ $(1\text{-}F_{between\ MSIV})$ $(1\text{-}F_{MSIV2\text{-}MSLDL})$ $(1\text{-}F_{MSLDL})$ $(1\text{-}F_{Condenser})$, where: F is the retention fraction in the referenced volume. As such, the fraction released to the environment would be: $1 - F_{Total}$

The dose analysis prepared in support of this report assumed a total release from all four MSIVs not to exceed 1.57E-03 standard m^3/sec (200 scfh). As such, the release rate assumed in the dose analysis was 200 scfh.

Because the assumed leakage rate is in "standard" units, the total MSIV leakage rate must be adjusted (based on the ideal gas law) to account for post-accident containment pressures and temperatures as follows:

$$\dot{V}(cfh) = \dot{V}(scfh)\left[\frac{T_{cont}}{T_{STD}}\right]\left[\frac{P_{STD}}{P_{cont}}\right]$$

where: temperature is in Kelvin and pressure is in pascals (absolute units).

DCD Table 6.2-1 states that the containment design pressure is 310 kPaG (45 psig) and the design temperature is 171°C (340°F). If these values are used, the adjustment factor for MSIV leakage is calculated to be 0.377.

DCD Figures 6.2-9 and 6.2-10 show the pressure and temperature following a LOCA assuming "nominal" conditions. DCD Figures 6.2-12 and 6.2-13 show the pressure and temperature for

LOCAs under "bounding"conditions. The Bottom Drain Line Break, Feedwater Line Break, and Main Steam Line Break LOCA-Containment analyses were reviewed. Adjustments factors as a function of time were calculated for both nominal and bounding conditions.

[[

]]. Regulatory Guide 1.183, Appendix A, Section 6.2 states that the MSIVs should be assumed to leak at the maximum value allowed by Technical Specifications (TS); however, it also states that the leakage may be reduced by up to 50% after 24 hours if that reduction can be supported by plant analyses. As a result, adjustment factors have been determined for the first 24 hours (technically from 5 minutes until 24 hours), and from 24 hours to 30 days. The results are presented in Table 4.10. The maximum adjustment factor for the first 24 hours was calculated for the MSLB – Nominal evaluation, with a result of 0.754. The maximum value after 24 hours occurred in the Bottom Drain Line Break scenario with an AF of 0.506. These results will be conservatively rounded to 0.8 for the first 24 hours, and 0.6 after 24 hours. The resultant MSIV leakage flow rates from the drywell can then be determined:

$$\dot{V}_{DW,0-24hr}(cfm) = (0.8)(200scfh) = 160cfh = 2.667cfm$$
$$\dot{V}_{DW,\geq 24hr}(cfm) = (0.6)(200scfh) = 120cfh = 2.0cfm$$

Since the condenser is relatively close to standard conditions, no adjustment is made for leakage from that volume:

$$\dot{V}_{condenser}(cfm) = (1.0)(200scfh) = 200cfh = 3.333.$$

Table 4.10
Conversion Factors for MSIV Leakage

Description	Maximum Adjustment Factor for DW (> 5 min)	Maximum Adjustment Factor for DW (> 24 hrs)
BDL	0.699	0.506
FW Nominal	0.636	0.481
FW Bounding	0.584	0.448
MSLB Nominal	0.754	0.451
MSLB Bounding	0.689	0.419
Maximum Calculated Value	***0.754***	***0.506***
Bounding Event	MSLB Nominal	BDL
Corresponding Time (hr)	0.618	24.007
Corresponding Drywell Pressure (kPa)	191.3	276.7
Corresponding Drywell Temperature (°C)	144.3	132.0

[[

Table 4.11
Drywell and RPV Upper Plenum Concentration Comparison

]] Detailed tables were not provided in References 7-31b and 7-31c. However, there is significant margin based on AS-1 such that no concerns exist with the remaining accident scenarios.

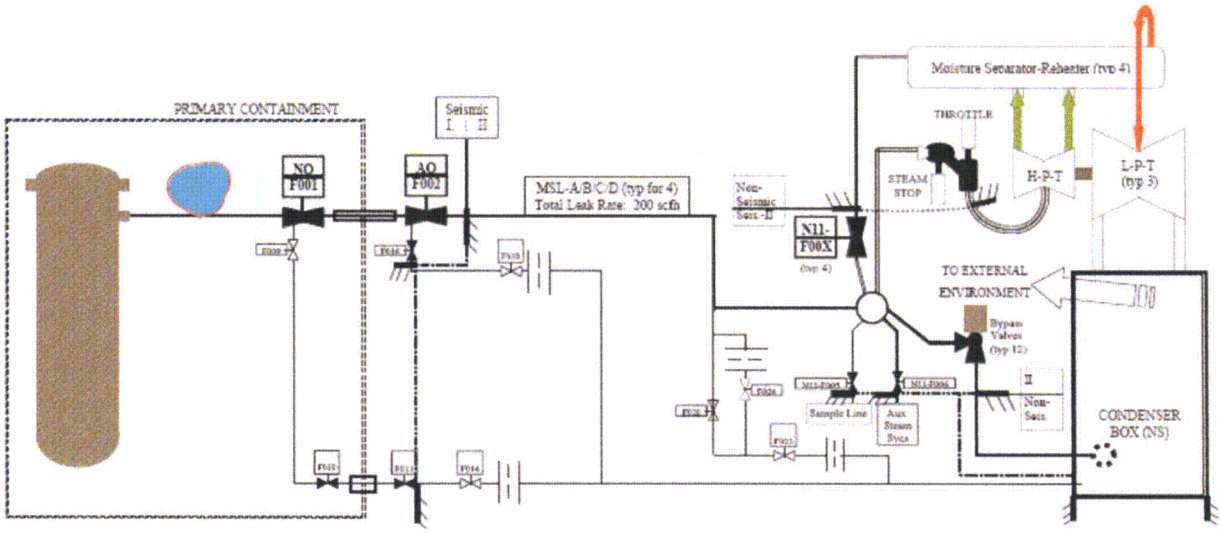

Figure 4.14. ESBWR Main Steam Line and Condenser Piping Sketch for Dose Calculations

4.5 CONTAINMENT AND REACTOR BUILDING LEAKAGE PATHS

Regulatory Guide 1.183 also requires that the dose consequences due to potential liquid leakage from ESF injection systems be evaluated if portions of the system are located outside of the primary containment building. The GDCS is contained entirely in the primary containment. The PCCS is also contained entirely in containment building with the exception of the condensers and the piping to/from the condensers. The condensers are completely submerged except for a relatively short time. Specifically, the pool level drops below the top of the PCCS condenser from 18 to 72 hours. The PCCS condensers contain a steam/air/water mixture. Any leakage from the PCCS condensers will be included in the overall containment leakage term.

Liquid leakage from the PCC condensers and associated piping is not considered credible as the PCCS pools would simply dilute it, and the dose contribution would be negligible. In any case, leakage from PCCS is conservatively treated as air leakage, which yields larger dose consequences than an equivalent amount of liquid leakage. Similarly, the ICs also contain a steam/air/water mixture and the dose contribution would be considered negligible for the same reasons. Because no credible source for ESF liquid leakage outside of containment exists, no ESF liquid leakage term will be evaluated.

Containment leakage can occur through numerous containment penetrations. Piping penetrations include:

- Main Steam (discussed previously in Section 4.4);
- Feedwater;
- Isolation Condenser System;
- Control Rod Drive System (CRDS);
- Standby Liquid Control (SLC) system;
- Decay Heat Removal Systems (Fuel and Auxiliary Pools Cooling System [FAPCS], Reactor Water Cleanup/Shutdown Cooling System [RWCU/SDC]);
- Station Auxiliary Systems (Makeup Water, Chilled Water, Nitrogen Supply);
- Containment and Environmental Control Systems (PCCS, Containment Inerting System, Containment Monitoring Systems); and
- Equipment and Floor drains.

In addition to the piping penetrations, there are also instrumentation and electrical penetrations.

The RB is discussed in depth in Subsection 6.2.3 of Tier 2 of the ESBWR Design Control Document [Ref. 7-19]. The RB is a robust structure designed to Seismic Category I criteria. All openings through the RB boundary, such as personnel and equipment doors, are closed during normal operation and after a DBA by interlocks or administrative controls. The doors are provided with position indicators and alarms that are monitored in the control room. The compartments in the RB are designed to withstand the maximum pressure due to a high-energy line break (HELB) in the RB.

SRP Section 6.2.3, Revision 2, "Secondary Containment Functional Design," was issued in July 1981 [Ref. 7-6]. The SRP provides information concerning crediting of secondary containment structures for holdup, decay, and treatment of fission products by Engineered Safety Feature (ESF) charcoal filter trains. The ESBWR does not have a secondary containment as the RB is not held to the required vacuum of 62 Pa (–0.25" w.g.); however, the RB is credited for the holdup of fission products prior to the release to the atmosphere, in part because the RB is robust design and is designed to Seismic Category I criteria. Regulatory Guide 1.183 [Ref. 7-3] allows a maximum of 50% of the secondary containment volume to be credited for holdup and decay. The RB CONAVS area is credited in the design basis LOCA dose consequence analysis for the holdup and decay of fission products. A detailed analysis was developed using the GOTHIC computer code to determine how radioactivity released from containment mixes within the RB prior to release to the environment. The GOTHIC analysis is summarized in Appendix B. The GOTHIC results confirm that the RB mixing volume credited of 11,600 m^3 (411,000 ft^3) is conservative for use in the dose calculation.

Because there are no safety-related emergency diesel generators for the ESBWR, there is no on-site A/C electrical power assumed to be available immediately following a LOCA. As such, there are no significant heat loads in the RB following a DBA LOCA. If A/C power were available immediately following a LOCA, then additional injection systems would be available, which would minimize fuel damage. Also, radiation monitors would be available to monitor plant releases and appropriate measures would be taken to mitigate the consequences of the accident. Therefore, the bounding scenario is with no A/C power.

The RB is subdivided into three distinct volumes, each serviced by a separate HVAC subsystem. The CLAVS subsystem services areas that are "clean." No radioactivity is expected in the CLAVS volume during normal operation. The CLAVS area includes equipment such as batteries and switchgear. The CONAVS services areas where the potential for airborne contamination could exist. The CONAVS treats areas that include equipment such as RWCU/SDC heat exchangers, the RB filter units, containment airlocks, and containment penetrations. Finally, the REPAVS services the refuel area of the RB. The results of the GOTHIC analysis discussed in Appendix B confirm that essentially all containment leakage is within the CONAVS volume. Therefore, any leakage from this volume is conservatively released directly to the environment to maximize the dose consequences.

TS Surveillance Requirement (SR) 3.6.3.1.5, located in Tier 2 of the DCD, requires verification that the RB exfiltration rates are within limits. This analysis assumed an overall CONAVS leakage rate of 141.6 l/s (300 cfm) at accident conditions.

The majority of containment piping penetrations is for systems that terminate in the CONAVS mixing volume (or the fuel building for FAPCS); therefore, leakage through these penetrations is assumed to mix with the RB atmosphere as discussed previously. Because they are interior to the building, it is also assumed that leakage through electrical penetrations mixes with the CONAVS atmosphere.

There are some potential containment leakage paths that may not readily mix with the CONAVS volume. Of specific concern are the PCCS condensers and leakage through the feedwater lines. Although leakage past the ICs and associated piping would be released into the RB, the airspace above the pools is relatively small and it is vented directly to the environment (through moisture separators); therefore, it does not mix with the remainder of the RB volume. Leakage past the IC

containment isolation valves could fall in this category as well. For this area, the PCCS and IC pools would be boiling, thus providing the driving force for this leakage. This leakage is conservatively assumed to be released directly to the environment, with no holdup credited in the PCCS/IC pool airspace. Leakage through the PCCS heat exchangers is assumed to be 0.01 wt. % per day of the containment atmosphere.

The feedwater lines are not seismically supported in the Turbine Building; therefore, the integrity of the lines is not assured following a safe-shutdown earthquake. Leakage past the feedwater isolation valves is conservatively assumed to be released directly to the environment with no credit taken for holdup in the Turbine Building itself. Leakage through the feedwater lines is assumed to be 7.0E-04 standard m^3 per minute (2.47E-02 scfm). This value will be tested via TS SR 3.6.1.3.10. Since the acceptance limit is in "standard" units, the release rate is adjusted (based on the ideal gas law) to the post-accident containment pressure and temperature as described for MSIV leakage:

$$\dot{V}_{FW,0-24hr}(cfm) = (0.8)(700sccm) = 560ccm = 0.0198cfm$$
$$\dot{V}_{FW,\geq 24hr}(cfm) = (0.6)(700sccm) = 420ccm = 0.0148cfm$$

4.6 SAFETY RELIEF VALVE (SRV) FLOW AND SUPPRESSION POOL SCRUBBING

Regulatory Guide 1.183, Appendix A, Section 3.5 states the following.

> "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. 7). Analyses should consider iodine re-evolution if the suppression pool liquid pH is not maintained greater than 7."

The guidance provided by current regulatory documents (Reg. Guides, SRP, etc.) is intended to address blowdown of the drywell through the suppression pool vents, not necessarily flow through the SRVs. NUREG-1465 states the following.

> "It is emphasized that the release fractions for the source terms presented in this report are intended to be representative or typical, rather than conservative or bounding values, of those associated with a low pressure core melt accident, except for the initial appearance of fission products from failed fuel, which was chosen conservatively."

Although NUREG-1465 used low-pressure[1] scenarios, the Accident Scenarios chosen to determine the removal coefficients for the ESBWR primary containment include both low-pressure (AS-1) and high-pressure (AS-2 and AS-3) events. The flow through the SRVs is negligible for low pressure events as confirmed by the MELCOR results for AS-1. The radioactivity is released through either the depressurization valves (DPVs), once they are assumed to operate, or the break itself. However, high-pressure events result in an appreciable flow through the SRVs. In AS-2 and AS-3 the DPVs are assumed not to operate until just before

[1] "Low pressure" refers to the pressure in the RPV.

RPV failure, hence the "high pressure" in the RPV. MELCOR calculations confirm that the RPV pressure remains sufficiently high to cause SRVs to lift during both the gap and EIV release phases for the high-pressure events.

The SRVs which lift during the events evaluated in this report are discharged through spargers in the Suppression Pool. SRP 6.5.5 states "If the time integrated (decontamination factor) DF values claimed by the applicant for removal of particulate and elemental iodine are 10 or less for a Mark II or III ... the applicant's values may be accepted without any need to perform calculations." The DF values in the SRP apply to the suppression pool vents rather than the SRV spargers. The vents allow "slug flow" to pass, whereas the spargers are designed to maximize quenching of the steam released from the RPV.

The MELCOR computer code was used to demonstrate that the DFs provided in the SRP are reasonable and conservative for releases through the SRVs. As discussed previously, the flow through the SRVs is negligible for AS-1 due to the fact that the DPVs lower RPV pressure to significantly below the SRV lift pressures. Figures 4.15 and 4.16 show the DF calculated by MELCOR for AS-2 and AS-3, respectively [Ref. 7-31]. The MELCOR calculations determined very high DFs during the time periods corresponding to the gap and EIV release phases (DF ranges from ~100 to ~1.0E+12). As such, applying a DF of 50 is clearly reasonable and conservative. Note that the MELCOR analysis did not explicitly calculate DFs for elemental iodine. Engineering judgment dictates that the results would be similar, and use of a DF of 10 is reasonable for elemental iodine as well.

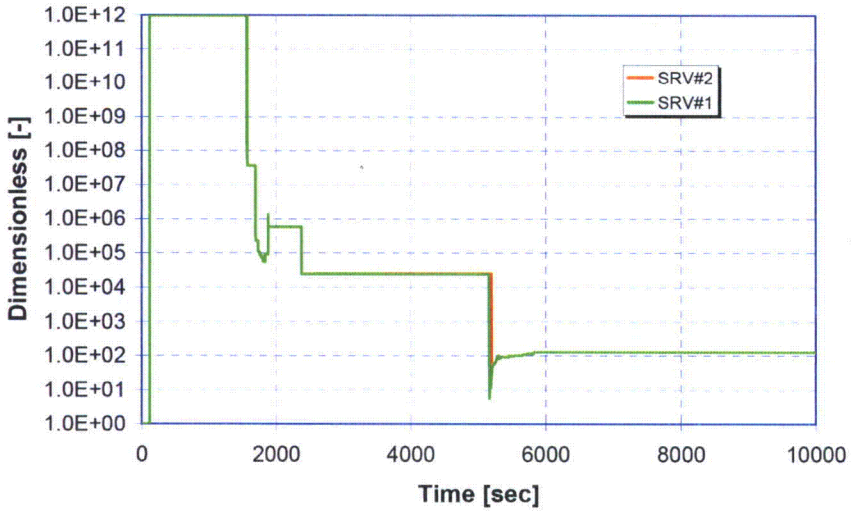

Figure 4.15. AS-2 Instantaneous Csl Aerosol DF for SRVs Suppression Pool Discharge

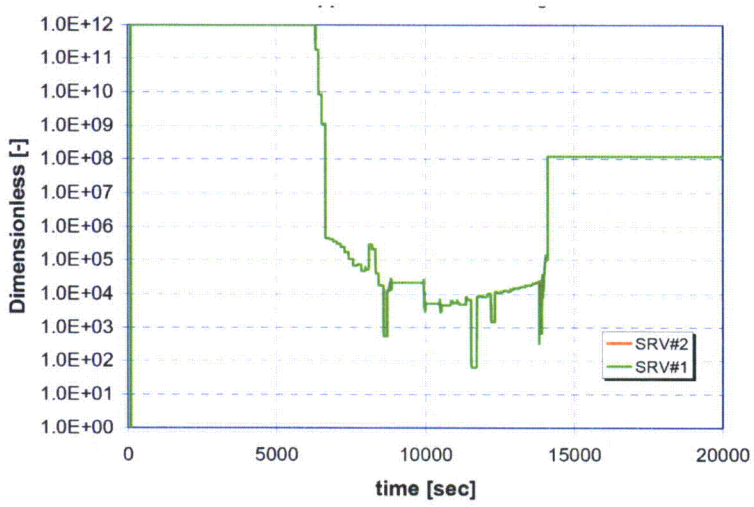

Figure 4.16. AS-3 Instantaneous Csl Aerosol DF for SRVs Suppression Pool Discharge

Not all of the release is assumed to occur through the SRVs. Releases through the DPVs or the break itself would not be scrubbed by the suppression pool. The fraction of flow through the SRVs is determined using the MELCOR analyses. Figures G-3 and G-4 of Reference 7-31c shows the SRV and DPV flow for AS-2, and Figure G-5 shows the flow through the break itself. For AS-3 the flow through the SRVs and DPVs are show in Figures H-2 and H-3 of Reference 7-31c. The integral flow over each release phase was adjusted to the onset of the gap phase based on the MELCOR results as shown in Figures 4.17 and 4.18 for AS-2 and AS-3, respectively. The fraction of flow through the SRVs was then determined. The results are presented in Table 4.12.

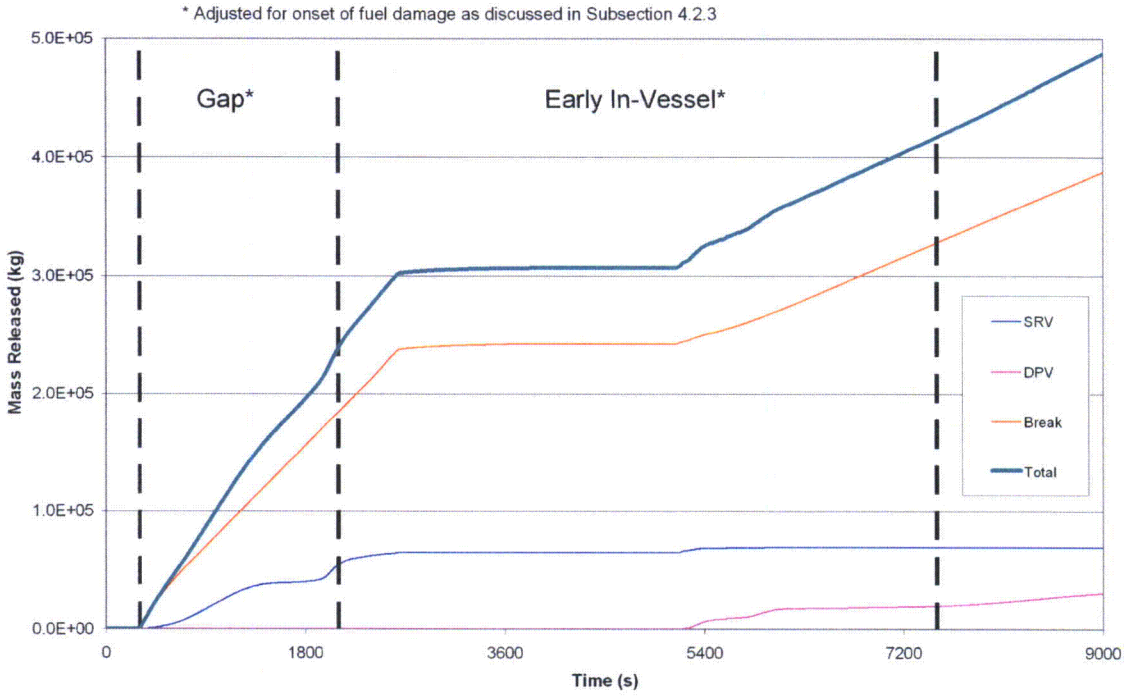

Figure 4.17. Integral Steam Flow After Fuel Damage for AS-2

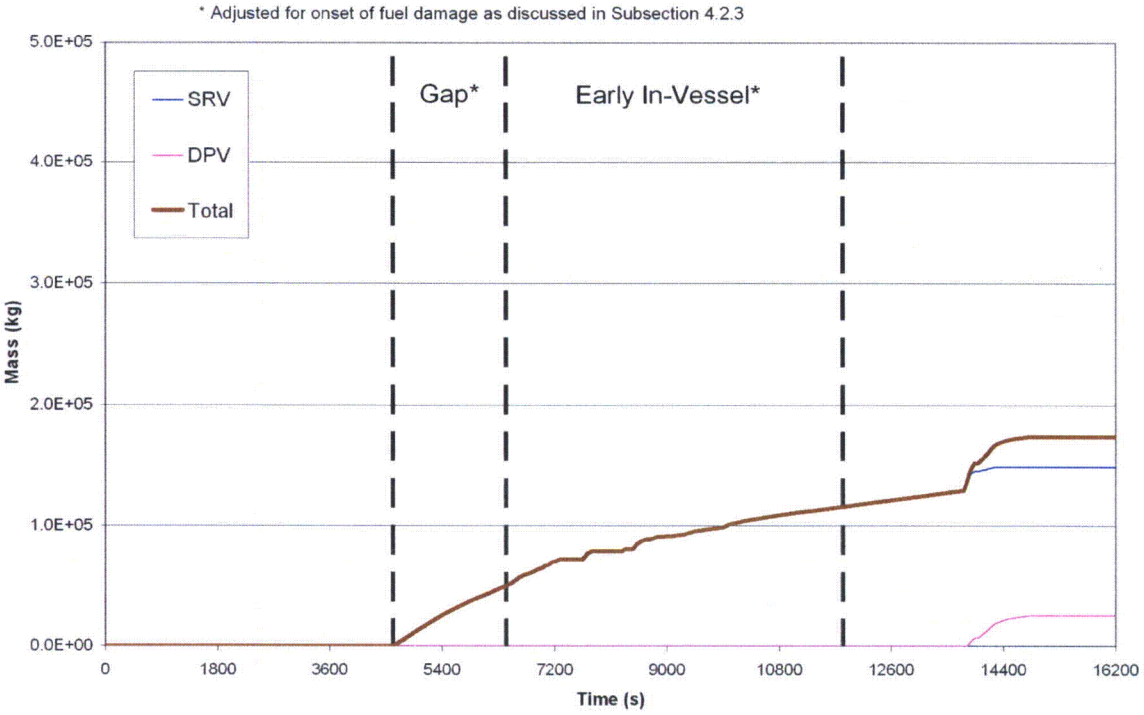

Figure 4.18. Integral Steam Flow After Fuel Damage for AS-3

Table 4.12
SRV Flow Rate Distribution

	AS-2		AS-3	
	SRVs	**DPV/Break**	**SRVs**	**DPV***
Flow % - Gap	22.7%	77.3%	100.0%	0.0%
Flow % - Gap Assumed	22.7%	77.3%	50.0%	50.0%
Flow % - EIV Actual	8.4%	91.6%	100.0%	0.0%
Flow - EIV Assumed	8.4%	91.6%	50.0%	50.0%

* No break in AS-3.

5.0 OFFSITE AND CONTROL ROOM DOSE CALCULATIONS

5.1 GENERIC MODEL INFORMATION

As discussed in Section 4.5, the RB is credited for hold-up and decay of fission products following a LOCA. Regulatory Guide 1.183 allows a maximum of 50% of a building volume may be credited for holdup and decay of radioactive materials for a secondary containment. Although the RB volume is 60500 m^3 (2140000 ft^3), this analysis conservatively credits only 11600 m^3 (411000 ft^3) of the CONAVS volume based on the GOTHIC analysis discussed in Appendix B. This analysis will assume a building release rate of 300 cfm from the RB.

The containment leakage rate is assumed to be 0.35 wt. % per day. The majority of the containment leakage term is released into the RB (0.34 containment wt. % per day). A value of 0.01 (0.025 L_a) is released directly to the environment via the PCCS/IC pool airspace. Finally, an additional term of 700 sccm is released through the feedwater isolation valves directly to the environment (via the Turbine Building).

Several parameters are updated based on the results of the MELCOR analyses performed for the ESBWR. Regulatory Guide 1.183 states that for BWRs, the gap release is assumed to begin 2 minutes into the event and last for 30 minutes, and the EIV release is assumed to begin immediately following the gap release and last for 1.5 hours. The dose consequence calculation conservatively neglects radioactive decay during the "coolant" release phase. The release is conservatively assumed to occur at the onset of the event. However, the removal coefficients for containment are adjusted to ensure that the containment thermal hydraulic conditions correspond to when activity would actually be released (as discussed previously in Section 4.3). The release fractions were based on Regulatory Guide 1.183, Table 2.

The methodology used to model natural deposition in the containment and removal of fission products by the PCCS condenser was modified to reflect the results of the MELCOR analysis as discussed previously in Section 4.3. Rather than model each removal mechanism separately, the MELCOR results were used to determine the amount of radioiodine that remained airborne in the containment building, thus available for release to the environment. The removal coefficients used were presented previously in Tables 4.6, 4.7, and 4.8. Because MELCOR did not explicitly model elemental iodine, no credit was taken for removal of elemental iodine by the PCCS condenser in this analysis. Only natural deposition is credited for the removal of elemental iodine as discussed in Section 4.3.

The ESBWR Control Room ventilation system includes safety-related Emergency Filter Units (EFU). The Control Room ventilation system is discussed in detail in DCD, Tier 2, Sections 6.4 and 9.4. The EFUs meet Regulatory Guide 1.52 [Ref. 7-39] requirements and are tested in accordance with ASME AG-1 [Ref. 7-40]. The EFUs are automatically initiated when high radioactivity is detected in the normal air supply duct, or upon an extended loss of AC power.

The Control Room EFU supplies air with a design flow rate of 220 l/s (466 cfm), and it is designed to maintain the control room envelope at a positive pressure with respect to adjacent compartments. An intake filter efficiency of 99% is assumed for particulate, elemental, and organic iodine species. The system does not include filtered recirculation. Although the control room is maintained at a positive pressure, this study will assume 5.66 l/s (12.0 cfm) unfiltered inleakage. The discharge flow from the control room is adjusted proportionally to account for

the additional inleakage. The Control Room model used for this study is shown in Figure 5.1. The total control room volume was determined to be ~2464 m^3 (87000 ft^3). This value will be reduced by ~10% to account for equipment, structures, etc. Therefore, a volume of 2209 m^3 (78000 ft^3) is assumed in the actual dose consequence evaluations.

[[

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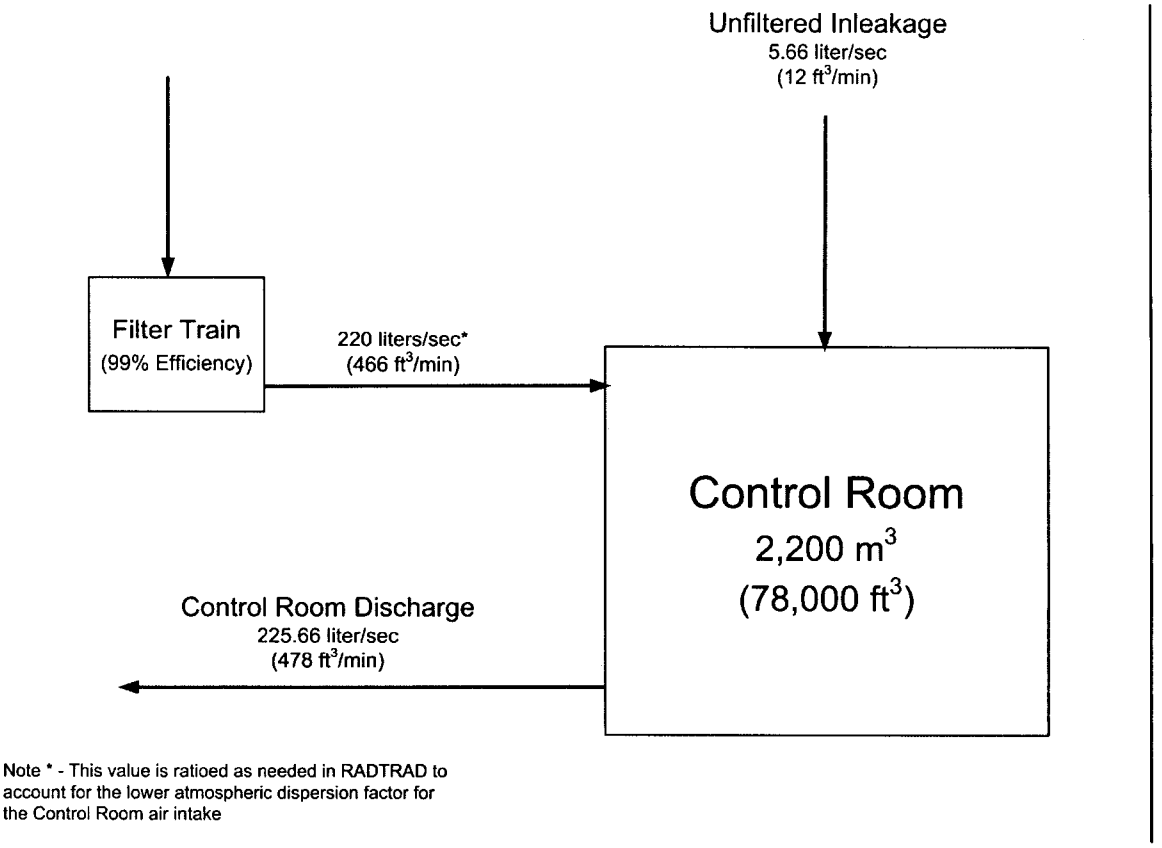

Figure 5.1. Control Room Envelope Model

For doses, separate dispersion factors were chosen for the three release locations: (1) containment leakage via the RB, (2) containment leakage via PCCS leakage, and (3) the Turbine Building (MSIV and feedwater leakage). There are also numerous receptor locations: (1) the control building louvers (the assumed location for Control Room unfiltered inleakage), (2) the "north" emergency air intake, and (3) the "south" emergency air intake. For convenience, one set of χ/Q values is used to bound both emergency air intake locations. The two sets of χ/Q

values assumed for control room doses are presented in Table 5.1. These values must be verified by COL applicants as documented in DCD Chapters 2 and 15.

- **Containment Leakage – RB:** The containment leakage released to the RB is assumed to be released on the "east" side of the generic ESBWR plant layout.
- **Containment Leakage – PCCS:** The PCCS leakage is assumed to be ducted to the top of the RB. This location is assumed to be a point release.
- **Containment Leakage – Feedwater Isolation Valves:** The small amount of leakage through the feedwater isolation valves is released to the environment through the Turbine Building.
- **MSIV Leakage:** MSIV leakage is assumed to be released from the main condenser through the Turbine Building.

The computer code RADTRAD is limited to only one set of χ/Q values. For all release locations, the unfiltered inleakage X/Q is greater or equal to the filtered intake χ/Q. To account for the lower χ/Q, the filtered intake flow is adjusted for each time step and each release location as follows:

$$Flow_{adj} = (220\,l/s)\left(\frac{\chi/Q_{\text{intake}}}{\chi/Q_{\text{louvers}}}\right)$$

The revised flow rates are presented in Table 5.2.

Table 5.1
Control Room X/Q Values (sec./m^3)

Time	Reactor Building Leakage		PCCS Leakage (RB Roof)		MSIV Leakage* (Condenser)	
	Louvers	Emergency Intake	Louvers	Emergency Intake	Louvers	Emergency Intake
0 – 2 hrs	1.90E-03	1.50E-03	3.40E-03	3.00E-03	1.20E-03	1.20E-03
2 – 8 hrs	1.30E-03	1.10E-03	2.70E-03	2.50E-03	9.80E-04	9.80E-04
8 – 24 hrs	5.90E-04	5.00E-04	1.40E-03	1.20E-03	3.90E-04	3.90E-04
1 – 4 days	5.00E-04	4.20E-04	1.10E-03	9.00E-04	3.80E-04	3.80E-04
4 – 30 days	4.40E-04	3.80E-04	7.90E-04	7.00E-04	3.20E-04	3.20E-04

* The values for the Turbine Building were intentionally assumed to be identical due to the fact that the distance between the Turbine Building and the Control Building louver is very close to the distance between the Turbine Building and the emergency air intake.

Table 5.2
Control Room Adjusted Flow Rates

Time	Reactor Building Leakage		PCCS Leakage (RB Roof)		MSIV Leakage (Condenser)	
	$\frac{\chi/Q_{Intake}}{\chi/Q_{Louvers}}$	Adjusted CR Intake Flow [l/s]	$\frac{\chi/Q_{Intake}}{\chi/Q_{Louvers}}$	Adjusted CR Intake Flow [l/s]	$\frac{\chi/Q_{Intake}}{\chi/Q_{Louvers}}$	Adjusted CR Intake Flow [l/s]
0 – 2 hrs	78.9%	157.9	88.2%	176.5	100.0%	200.0
2 – 8 hrs	84.6%	169.2	92.6%	185.2	100.0%	200.0
8 – 24 hrs	84.7%	169.5	85.7%	171.4	100.0%	200.0
1 – 4 days	84.0%	168.0	81.8%	163.6	100.0%	200.0
4 – 30 days	86.4%	172.7	88.6%	177.2	100.0%	200.0

The off-site dispersion factors are presented in Table 5.3. These values are applicable to all potential release points for the ESBWR.

Table 5.3
Off-Site χ/Q Values

Location	χ/Q Value
EAB	
0 - 2* hours	2.00E-03 sec/m^3
LPZ	
0 - 8 hours	1.90E-04 sec/m^3
8 - 24 hours	1.40E-04 sec/m^3
1 - 4 days	7.50E-05 sec/m^3
4 - 30 days	3.00E-05 sec/m^3

* The value listed corresponds to the 0 – 2 hour value. However, because AST calculations are required to determine the "worst 2-hour" dose, this value is applied to the entire 30 days.

5.2 AS-1 DOSE CALCULATION

The model to calculate doses for the low pressure scenario consisted of five volumes: (1) the containment/drywell, (2) the RB, (3) the main condenser, (4) the environment, and (5) the control room. RADTRAD v3.03 only allows inputting one set of X/Q values for each receptor location (EAB, LPZ, and control room). As such, the model shown in Figure 5.2 was divided into four separate input decks. One input deck was used to evaluate the containment leakage released through the RB. A second input deck was used to evaluate containment leakage through the PCCS ventilation stack. A third input deck was used to evaluate feedwater line leakage past

the feedwater isolation valves. The final RADTRAD input deck was used to model MSIV leakage through the main steam lines, main steam line drain lines, and the main condenser.

The final doses are then determined by summing the doses from the four contributors. The results for all four contributors are presented in Table 5.4.

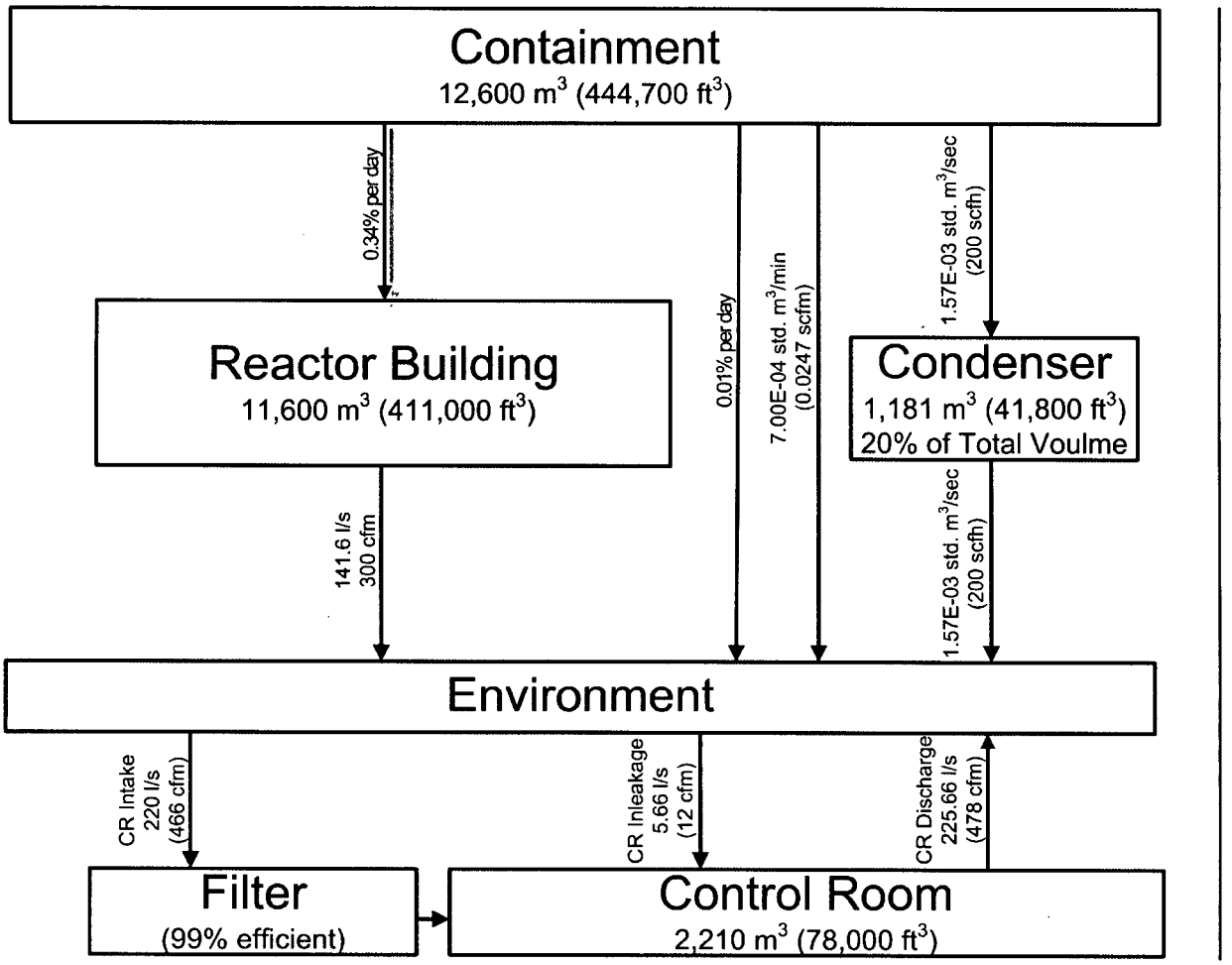

Figure 5.2. AS-1 RADTRAD Model

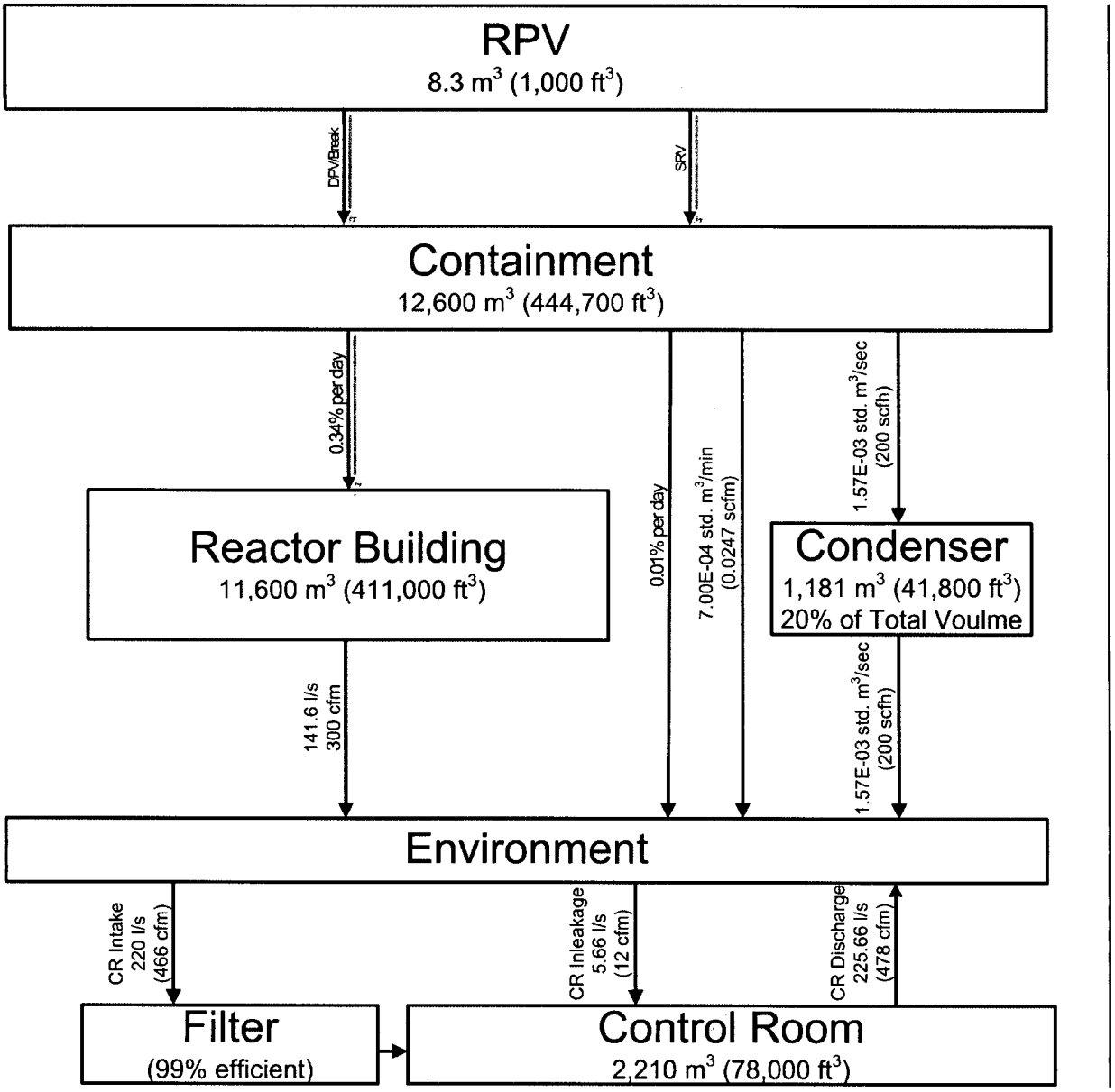

Figure 5.3. AS-2 and AS-3 RADTRAD Model

5.3 AS-2 DOSE CALCULATION

The model used to evaluate the high pressure scenarios (AS-2 and AS-3) is similar to that used previously; however, an additional compartment is required to model the RPV. No credit is taken for any removal in the RPV. An arbitrary volume of 28.3 m^3 (1000 ft^3) was assumed. Two pathways were used to model this flow. The first is flow from the SRVs through the suppression pool spargers, which is scrubbed by the suppression pool. The second pathway was flow through the DPV and the break itself, which is not scrubbed. A very high flow rate (472 m^3/sec., 1.0E6 cfm) was assumed from the RPV to the drywell. The flow through each pathway is scaled based on the MELCOR flow rates as discussed in Section 4.6. For the gap release phase, 22.7% of the flow is assumed through the SRVs, and for the EIV release phase, only 8.4%

is assumed through the SRVs for AS-2. The model used for AS-2 is shown in Figure 5.3. The results are presented in Table 5.4.

5.4 AS-3 DOSE CALCULATION

The RADTRAD model used for AS-3 is essentially identical to the model for AS-2. The only significant modifications are updating the drywell removal coefficients and updating the RPV to containment flow rates for the SRV and the DPVs. Since there is no break, all of the activity released from the RPV to the drywell is released through the SRVs during the gap release phase. During the EIV phase, all of the flow is also released via the SRVs; however, the dose analysis conservatively assumed only 50% of the flow occurred through the SRVs for both the gap and EIV release phases. The model used for AS-3 is shown in Figure 5.3. The results are presented in Table 5.4.

5.5 DOSE CALCULATION RESULTS

Table 5.4 shows the dose calculation results for all three Accident Scenarios. The results show that AS-1 is bounding for both off-site and control room doses. The EAB results for the four separate RADTRAD runs indicate that the "worst 2-hour" period for each contributor does not coincide with the worst 2-hour period for the other contributors. Therefore, the peak EAB dose interval was 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 as described in Section 4.1.5 of RG 1.183. Specifically, a sliding two-hour sum was calculated for each of the four RADTRAD runs, and results were summed to obtain a total dose for each time increment considered. The "maximum two-hour TEDE" peak dose interval was then extracted from the results. The worst 2-hour period occurs beginning at 2.3 hours for AS-1, 1.6 hours for AS-2, and 1.6 hours for AS-3. The EAB doses presented in Table 5.4 correspond to these times.

Table 5.4
Dose Calculation Results (rem TEDE)

Contributor	AS-1			AS-2			AS-3		
	EAB	**LPZ**	**CR**	**EAB**	**LPZ**	**CR**	**EAB**	**LPZ**	**CR**
RB	1.72+01	1.69E+01	3.74E+00	1.50E+01	1.23E+01	2.59E+00	1.27E+01	1.28E+01	2.79E+00
MSIV	3.67E-01	2.15E+00	3.31E-01	2.78E-01	2.11E+00	3.31E-01	2.73E-01	2.12E+00	3.26E-01
PCCS	2.85E+00	1.05E+00	4.94E-01	3.83E+00	8.50E-01	4.00E-01	3.53E+00	8.15E-01	3.81E-01
FW	1.94E+00	6.84E-01	1.19E-01	2.60E+00	5.65E-01	1.04E-01	2.40E+00	5.29E-01	9.21E-02
Re-evolution*	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	1.71E-01	5.11E-02
Total	***22.4***	***20.7***	***4.69***	***21.7***	***15.8***	***3.42***	***16.3***	***16.4***	***3.64***

* Doses from re-evolution of iodine in the LDW pool for AS-3 are calculated in Attachment C.

6.0 CONCLUSIONS

The ESBWR systems are redundant and diverse. The ESBWR DCD explains that for DBA LOCA scenarios with a loss of offsite power and the most limiting single active failure, the core would remain covered for the duration of the event and fuel damage is not expected to occur. The MELCOR analysis utilized to determine the timing of fuel damage, as well as the associated plant thermal-hydraulic parameters, assumed no injection into the RPV until just prior to a breach of the RPV. This scenario would take multiple failures, which is well beyond "design basis" requirements. However, the assumptions used to estimate the fuel damage are similar to those used to determine the initial source term assumptions documented in NUREG-1465. Because the failure mechanisms are similar, the release fractions from Regulatory Guide 1.183 were applied to the ESBWR.

The ESBWR utilizes passive systems to respond to potential DBAs and other plant events. The base analysis prepared in support of this report uses reasonable, yet conservative assumptions to evaluate the dose consequences due to a design basis LOCA. Thus, the ESBWR systems, in conjunction with natural removal processes are sufficient to ensure that the dose consequences meet the criteria set forth in 10 CFR 52.47 and 10 CFR 50, Appendix A, GDC 19.

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b. VTT-R-04413-06, Part 2, December 2006 (Accident Scenarios 1, 2, and 3 removal coefficients and pH calculations)

c. VTT-R-06771-07, Part 3, May 2008 (pH sensitivity studies for Accident Scenarios 1, 2, and 3, and MSIV leakage pathway calculations)

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APPENDIX A LOCA Dose Assumptions

Table A-1
LOCA Dose Assumptions

Parameter	Value	Verification Method [Note 2]
I. Data and Assumptions Used to Estimate Source Terms		
A. Power Level, MWt	4590	TS 1.1 (102% of RTP)
B. Fraction of Core Inventory Released	RG 1.183, Table 1	NR
C. Iodine Chemical Species		
Elemental, %	4.85	NR
Particulate, %	95.00	NR
Organic, %	0.15	NR
II. Data and Assumptions Used to Estimate Activity Released		
A. Containment		
Total Containment Leak Rate, wt. %/day	0.35	TS 5.5.9
(Deleted)		
Containment Leakage Rate through PCCS, %/day[Note 1]	0.01	TS 5.5.9
Containment Leakage Rate through Feedwater lines, standard m^3/min (standard ft^3/min)	7.00E-04 (2.47E-02)	TS 3.6.1.3
Free Air Volume, m^3 (ft^3)	1.260E+04 (4.447E+05)	ITAAC
Elemental Iodine Removal Rate Constant, hr^{-1}	0.86	NR
Particulate Iodine Removal Rate Constant, hr^{-1}	Table 4.2	NR
SRV Flow Rate	Table 4.4	NR
Suppression Pool Decontamination Factors		
Elemental Iodine	10	NR
Organic Iodine	1	NR
Particulates	50	NR
Noble Gases	1	NR
B. Reactor Building		
Leak Rate, l/s (cfm)	141.6 (300)	TS 3.6.3.1
(Deleted)		
Total Volume, m^3 (ft^3)	6.05E+04 (2.14E+06)	NR
Mixing Volume, m^3 (ft^3)	1.16E+04 (4.11E+05)	ITAAC

Table A-1
LOCA Dose Assumptions

Parameter	Value	Verification Method [Note 2]
C. Condenser Data		
Free Air Volume, m^3 (ft^3)	5.93E+03 (2.09E+05)	NR
Mixing Fraction, %	20	NR
Mixing Volume, m^3 (ft^3)	1.25E+03 (4.40E+04)	ITAAC
Iodine Removal Factors		
Particulate, %	99.3	NR
Elemental, %	99.3	NR
Organic, %	0.0	NR
D. MSIV Data		
Total MSIV Leakage, standard m^3/sec (scfh)	1.57E-03 (200)	TS 3.6.1.3
Total MSIV Leakage, m^3/sec (cfm) [Adjusted for post-LOCA containment Pressure and Temperature]	1.04E-03 (2.2)	NR
Total MSIV Leakage from Condenser, m^3/sec (cfm) [Adjusted for post-LOCA condenser pressure]	1.57E-03 (3.33)	NR
III. Control Room Parameters		
A. Control Room Volume Credited, m^3 (ft^3)	2.2E+03 (7.8E+04)	ITAAC
B. Control Room Emergency Filter Unit (EFU) Intake Flow, l/s (cfm)	220 (466)	TS 5.5.13
C. Unfiltered Inleakage	5.66 (12)	COL
D. EFU Filter Efficiency, %	99	TS 5.5.13
IV. Pool Parameters and pH Calculations		
A. Pool Volumes		
GDCS Pool Volume, m^3 (ft^3)	1.86E+03 (6.57E+04)	ITAAC
Suppression Pool Volume, m^3 (ft^3)	3.80E+03 (1.34E+05)	ITAAC
B. Exposed (Chlorine Bearing) Cable Insulation Mass, kg (lb)	3400 (7480)	ITAAC

Notes:

1. Containment leakage (0.35 wt. % per day) is assumed to leak through two separate pathways. The majority of the leakage (0.34 wt. % per day) is assumed to leak to the RB, with the remainder (0.01 wt. % per day) released directly to the environment via the PCCS airspace stack.
2. The verification methods are defined as follows:
   a. **ITAAC -** (Inspections Tests Analyses and Acceptance Criteria) - This item will be verified as documented in DCD Tier 1.
   b. **COL -** (Combined Operating License) - This item will be verified by the utility/customer via the COL process
   c. **TS -** (Technical Specifications) - This item is a TS requirement, therefore no additional actions/verifications are required (Note that the actual TS section is identified)
   d. **NR -** (Not Required) - The basis for this assumptions is well defined via regulations or analysis; therefore no additional verification/confirmation is required.

APPENDIX B Reactor Building Mixing Assumptions

B.1 SCOPE

This Appendix provides an analysis to show that the mixing volume assumed in Appendix A is a conservative characterization of the RB that provides holdup and transport delay of radionuclides. This appendix discusses the methodology for a detailed analysis using the GOTHIC 7.2a computer code [Ref. B-1] and its comparison to RADTRAD to justify that the mixing volume assumed in RADTRAD is conservative. The RADTRAD dose analysis is discussed in Section 5.0. Additional description of the GOTHIC analysis is provided in Reference B-2.

B.2 INTRODUCTION

The ESBWR RB provides a holdup volume and delays transport of radionuclides from the containment to the environment. The RADTRAD dose analysis models the RB as a single volume that has uniform mixing. The mixing volume assumed in RADTRAD is presented in Appendix A. The GOTHIC computer code is used for a detailed analysis of the RB and confirms that the mixing volume presented in Appendix A is a conservative characterization of the RB that provides holdup and transport delay of radionuclides. The GOTHIC model assumes the same containment leakage rate and the same RB release rate with respect to the RADTRAD dose analysis.

B.3 GOTHIC ANALYSIS

B.3.1 Assumptions

B.3.1.1 Accident Conditions

- LOCA with fuel damage concurrent with LOOP.
- Containment leakage into the RB contaminated areas.
- Normal HVAC is unavailable during the accident that includes the clean, contaminated and refueling floor ventilation systems (these systems are discussed in DCD Tier 2 Subsection 9.4.6).
- Communication between volumes is through different door types and HVAC ductwork.
- Communication between contaminated areas, clean areas, adjacent buildings, and the refueling floor is through doors that have minimal door gaps.
- All volumes are initially at atmospheric pressure.
- Ventilation ductwork connects the contaminated room volumes.
- All doors and hatches in the RB boundary are assumed to be closed.
- Wind is causing an overpressure on one side of the RB and a sub-pressure on the opposite side which encourages mixing.
- The RB has a differential pressure of approximately 62 Pa (1/4" w.g.) with respect to the atmosphere.

- Exfiltration rate to the environment from the RB is approximately 300 cfm.
- The room temperature profiles are consistent with the heat up analysis presented in ESBWR DCD Tier 2 Appendix 3H.
- The RB room free volumes are assumed to be 90% of the room total volume to account for occupied space.

B.3.1.2 Containment Releases

Most of the mass inside containment during the accident will be nitrogen and steam, which carry the radioactive releases. The ESBWR containment is inerted with nitrogen and steam comes from the pipe break. Radioactive releases reach the environment via nitrogen or steam. Comparing the representative containment releases and RB exfiltration rates allows for a comparison of the GOTHIC and RADTRAD models. The GOTHIC containment and RB exfiltration rates are measured by the migration of the inert gas nitrogen, and the RADTRAD containment and RB exfiltration rates are measured by the migration of krypton-85 (Kr-85).

B.3.1.2.1 Nitrogen

In the GOTHIC analysis, the radiological leakage from containment is modeled with nitrogen, the representative inert gas. Nitrogen has been chosen because the ESBWR containment is inerted with nitrogen and it does not condense. It is assumed that the nitrogen in the containment is under high pressure and low temperature so that a maximum amount of nitrogen is assumed in the containment.

B.3.1.2.2 Krypton-85

The calculated results for Kr-85 from the RADTRAD analysis is used to calculate the comparable containment and RB exfiltration rates. Krypton-85 is a long-lived isotope whose parent isotope decays quickly. Effectively, Kr-85 is constant throughout the period of interest.

B.3.1.3 GOTHIC Door Flow Paths

The RB door locations are based on the ESBWR DCD Tier 2 Nuclear Island General Arrangement drawings. The door gaps are based on existing plant data. Doors gaps for all doors have been increased to account for degradation. Doors that have insignificant leakage, based on the door gaps, have been modeled for volumes that connect contaminated areas to clean areas, adjacent buildings or the environment. Doors within the contaminated areas are modeled as fire rated doors, which have a relativity large amount of leakage in comparison to the other doors, so that flow is not restricted in these volumes. Door gaps on the contaminated stairwells have been modeled as having a relatively large gap to allow for the wind to pressurize the RB and to allow for an exfiltration rate out of the RB similar to that assumed in the RADTRAD analysis.

B.3.1.4 GOTHIC HVAC Ductwork Flow Paths

The contaminated area ventilation ductwork allows for communication between the contaminated volumes in the RB. Only the contaminated area ductwork is considered in the analysis.

B.3.1.5 Containment Leakage

The containment leakage rate in RADTRAD is presented in Appendix A. RADTRAD assumes that the containment leakage is released into a single volume. The GOTHIC model assumes the same containment leakage rate as the RADTRAD analysis. In the GOTHIC analysis, the leakage from the containment to the RB is divided amongst the ESBWR containment penetrations. The amount of leakage through a particular penetration is proportional to the penetration circumference. The penetration room locations assumed in the GOTHIC analysis as well as the percentage of the total leakage is presented in Table B-1.

B.3.2 GOTHIC Methodology

Several sub-volumes of the RB are modeled in GOTHIC. They include the CONAVS and CLAVS areas, and stairwells. The CONAVS ventilation area envelopes all the containment penetrations except those in the steam tunnel. In some cases, the CLAVS areas are barriers between the CONAVS and the environment. The stairwells act as a transport path from the CONAVS to the environment. All the interior doors connecting the different rooms in the building as well as the doors that connect to other buildings or to the environment are modeled. Additionally, the HVAC ductwork connecting the appropriate volumes is also modeled in GOTHIC.

CONAVS exfiltration is tested by TS SR 3.6.3.1.5. The portion of the CONAVS exfiltration that enters CLAVS is not separately determined by surveillance, and therefore in this analysis the leakage is maintained at a low value. If the CONAVS to CLAVS leakage were higher, CLAVS would provide an additional holdup volume for radiological release to the environment. The flow from CONAVS to the environment is maximized for comparison to the RADTRAD mixing volume, by reducing the flow loss for that pathway.

Each RB room is assigned a specific geometry determined by its volume, height, and elevation. More specific geometric characteristics are assigned to the rooms important to transport based on the room size, location and significance of flow obstructions. These selected rooms within the CONAVS areas are subdivided in the GOTHIC model.

Nitrogen is modeled as the radionuclide transport gas from the containment leakage, because it, along with steam, is the most abundant fluid in the containment and it does not condense. GOTHIC differentiates the nitrogen that leaks from the containment from the nitrogen that is part of the mixture of gases that make up air in the RB by treating the air as a gas rather than a mixture of gases.

The temperature of the nitrogen entering the RB from the containment is at the same temperature of the room it enters. If the nitrogen were discharged into the rooms at LOCA temperatures stratification could occur. Thermal stratification in the top of the rooms would cause a lower exfiltration to the environment, considering that the top of the doors are lower than the ceilings and that most of the containment leakage is at elevations above ground.

The nitrogen traverses through the RB due to the pressurization from wind loading. The wind direction is assumed to originate from the side of the RB that is opposite the control building, which causes the RB release to be close to the control building. The wind speed is assumed to be approximately 10 m/s (22 mph). The corresponding differential pressure in the RB is 62 Pa (1/4" w.g.) and the RB exfiltration rate is consistent with the RADTRAD dose analysis. Higher

wind speed would produce higher differential pressure, but also better atmospheric dispersion in the environment. The wind speed/differential pressure chosen is conservative relative to the stable air conditions assumed in calculating atmospheric dispersion. The nitrogen migrates to the contaminated stairwell, which is the closest release point to the control room. The nitrogen then migrates from the contaminated areas downward through the stairwell and is released to the environment.

B.4 RESULTS

B.4.1 GOTHIC Mixing Model

The nitrogen released from the containment to the RB slowly traverses through the contaminated areas. The nitrogen flows through the door and ductwork flow paths, eventually reaching the environment. An insignificant amount of nitrogen reaches the clean areas of the RB via the door flow paths. After 72 hours, the amount of nitrogen that reaches the environment over the amount released into the RB is approximately 48%.

B.4.2 GOTHIC vs. RADTRAD Comparison

A comparison of the results for the detailed RB GOTHIC model and the RADTRAD model, which assumes a credited mixing volume, is achieved via an integrated leak rate percentage to the environment of the total release into the RB. The comparison is a ratio of leakage to the environment from the RB over containment leakage into the RB. The GOTHIC percentage is calculated by the ratio of nitrogen released to the environment over the nitrogen released into the RB from the containment. The RADTRAD percentage is calculated by the ratio of the Kr-85 activity leaked out of the RB to the environment over the Kr-85 activity leaked into the RB from primary containment, which is recorded at various times during the LOCA from the RADTRAD output of the Kr-85 in the environment and the RB.

GOTHIC $$F_{GOTHIC} = \frac{\int (\dot{M}_{env}) dt}{\int (\dot{M}_{RB}) dt}$$ Eqn. 1

RADTRAD $$F_{RADTRAD} = \frac{\int (\dot{A}_{env,Kr-85}) dt}{\int (\dot{A}_{RB,Kr-85}) dt}$$ Eqn. 2

Where:

$\dot{M}_{env}$ = Mass release rate from the RB to the environment,

$\dot{M}_{RB}$ = Mass release rate from the containment to the RB,

$\dot{A}_{env,Kr-85}$ = Activity release rate from the RB to the environment, and

$\dot{A}_{RB,Kr-85}$ = Activity release rate from the containment to the RB.

The comparison is shown in Figure B-1 which includes the plots for the LOCA analyses supporting DCD Revision 5 and DCD Revision 6. It should be noted that the difference in the DCD Revision 5 and DCD Revision 6 curves is a result of reductions in the assumptions for (1) the credited mixing volume and (2) the leakage rate from containment into the RB. The results calculated in the GOTHIC model remain below the RADTRAD calculation during the accident.

An additional comparison was also performed to compare GOTHIC to RADTRAD. Specifically, the assumed RB volume was increased to determine an "equivalent volume" for RADTRAD that results in releases that resemble the release predicted by GOTHIC. The RADTRAD equivalent volume was determined to be 21950 m^3 (775000 ft^3). The results of the revised RADTRAD run are presented in Figure B-1.

B.5 CONCLUSION

The comparison of the GOTHIC and RADTRAD integrated leak rate percentage to the environment of the total release into the RB confirm that the mixing volume assumed in the RADTRAD LOCA model presented in Table A-1 is conservative for modeling the radiological release traversing the highly compartmentalized ESBWR RB. The comparison is a ratio of leakage to the environment from the RB over containment leakage into the RB. The GOTHIC analysis shows that hypothetical release from multiple penetrations into multiple RB sub-volumes provides significant holdup. The hypothetical release has to traverse through multiple volumes, ductwork, door gaps and stairwells. GOTHIC demonstrates that under DBA conditions for a LOCA concurrent with LOOP and fuel damage the mixing volume assumed in the LOCA dose analysis is conservative. The equivalent volume calculated is significantly greater than the value assumed in the LOCA dose calculation (See Appendix A).

B.6 REFERENCE

B-1 GOTHIC Containment Analysis Program, Version 7.2a(QA), EPRI, Palo Alto, CA.

B-2 EA Document 092-134-F-M-06000, "Reactor Building GOTHIC Mixing Model Calculation Report", Issued May 2008.

Table B-1 GOTHIC Penetration Location and Containment Leakage Distribution

Penetration Location Room Number [1,2]	% of Total Leakage
# 1742 Mechanical Penetration Room D	7.917 %
# 1732 Mechanical Penetration Room C	17.386 %
# 1722 Mechanical Penetration Room B	6.675 %
# 1712 Mechanical Penetration Room A	14.281 %
# 1600 Wetwell Access/Fan Room	30.458 %
# 1342 Division 4 Electrical Penetration Room	4.036 %
# 1332 Division 3 Electrical Penetration Room	4.502 %
# 1322 Division 2 Electrical Penetration Room	4.502 %
# 1312 Division 1 Electrical Penetration Room	10.245 %

(1) Approximately 77% of the leakage from containment is located in elevations 17500 and 13570, which are above grade and have the closest path for a direct release to the environment.

(2) Room numbers are identified in DCD, Tier 2, Figures 1.2-3, 1.2-6 and 1.2-7.

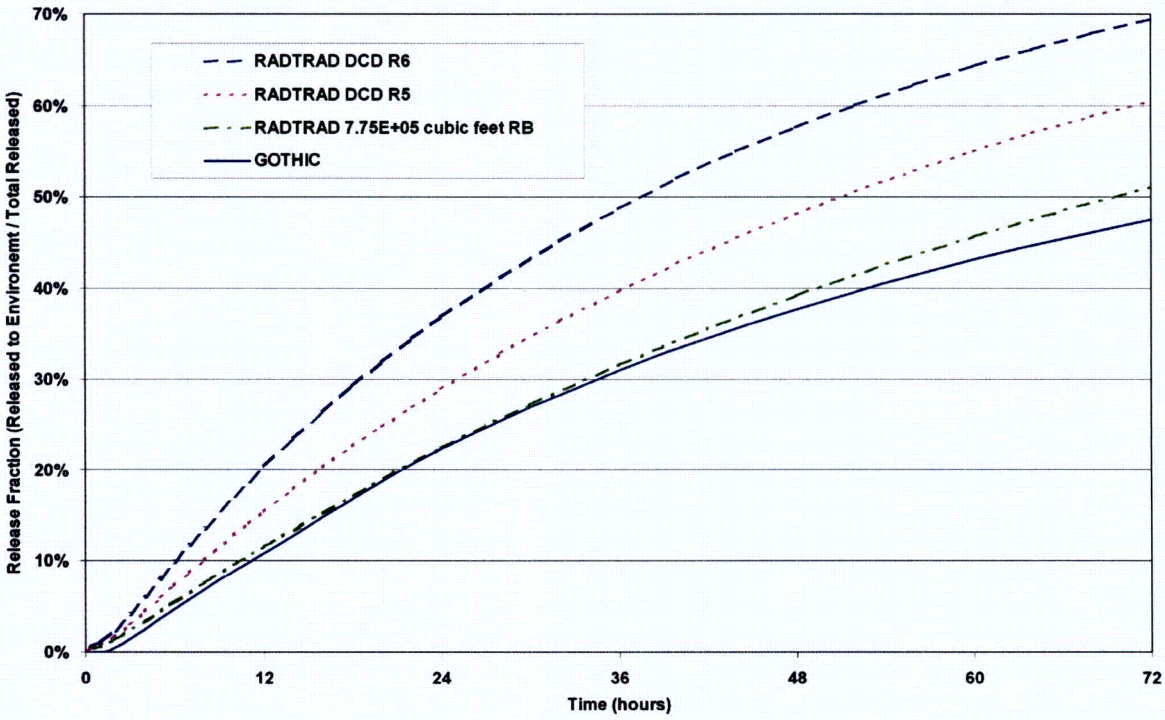

Figure B-1. GOTHIC vs. RADTRAD Released to Environment from RB/Total Release to RB

APPENDIX C Iodine Re-Evolution for the Lower Drywell

C.1 PURPOSE

The LDW could become acidic at 25.15, 24.26, and 8.78 days for AS-1, AS-2, and AS-3, respectively, as discussed in Reference 7-31c. A decay time of 24 to 25 days provides a significant amount of time for radioactive decay of the iodine isotopes. The bulk of the elemental iodine from re-evolution from these pools would be held up in containment since containment integrity is assured for 30 days. Only a small amount would leak through the various release pathways (Containment leakage to the RB, PCCS, and feedwater, as well as MSIV leakage). Also, a significant amount of the re-evolved elemental iodine would plate-out on containment structures. Another consideration is the release duration. Appreciable amounts of iodine do not begin to re-evolve until pH drops below ~4. Although the change in pH appears to occur suddenly in Figure 4.2, the duration between a pH of 7 and a pH of 4 is on the order of several days (due to the log-log" plot). Finally, parameters used to determine the dose consequences, such as control room occupancy factors and atmospheric dispersion factors, are relatively low this late into the event. As such, the dose consequences from re-evolution for AS-1 and AS-2 are negligible as a result of the parameters just discussed.

However, the pH drops below 7 for AS-3 significantly sooner that the other two Scenarios (~8 days vs. ~25 days). Since radioactive decay is a major consideration in the argument posed for AS-1 and AS-2, additional research into the dose consequences is warranted. As such, this Appendix models the re-evolution of iodine from the LDW for AS-3.

C.2 METHODOLOGY

The methodology used for this analysis is similar to that used in the design basis LOCA dose calculation discussed in Sections 4 and 5 of this report. The calculation evaluated four separate pathways to the environment in determining the overall dose consequences:

- Containment Leakage – Released through the RB;
- Containment Leakage – Released via the PCCS ventilation stack;
- Containment Leakage – Released through the Feedwater Lines (Turbine Building); and
- MSIV Leakage – Released from the Main Condenser (Turbine Building).

The RB, PCCS, FW and MSIV releases were evaluated using the NRC computer code RADTRAD v3.03, as described in Section 5.2.

This analysis utilizes the RADTRAD v3.03 code to evaluate the results from the re-evolution of iodine from the LDW for AS-3.

Release timing of the elemental iodine is based on the pH calculations documented in Reference 7-31c. The results of this analysis are included in the final dose results for AS-3 as presented in Section 5.5.

C.3 RADTRAD MODEL CHANGES

As discussed previously, the RADTRAD model used in this analysis is very similar to that used in the main LOCA dose calculation performed in Section 5 of this report. The changes are limited to:

- Source Term Assumptions – source term modeling will be based on the re-evolution of elemental iodine from the LDW pool (release timing, fractions, and composition).
- Plateout Modeling – Elemental iodine is modeled consistent with the methodology discussed in Section 4.3.1 of this report; however, timing is adjusted to ensure SRP 6.5.2 requirements are met (i.e., the maximum DF for iodine does not exceed 200).

The remaining assumptions remain consistent with the LOCA analysis documented in Section 5 of this report. The details for each of the changes are discussed in detail in the Subsections that follow. The model is represented in Figure C-1.

C.3.1 Source Term and Release Timing

Figure 27 of Reference 7-31C shows that long term, there is ~3.8 kg of CsI in the LDW pool, and Table I of that report states that the pH drops below 7 at 210.83 hrs. The total CsI in the core is 76.3 kg. This evaluation conservatively assumes that the entire 3.8 kg is released for a total release fraction of 0.0498. Figure 31 of Reference 7-31C shows the pH as a function of time in the LDW for AS-3. The results of the pH analysis are presented in Table C-1 (See Figure 4.4). Figure 3.1 of NUREG/CR-5950 shows the fraction of Iodine in solution for various pH levels for various concentrations. For AS-3, the LDW pool has a water mass of 515385 to 839184 kg per Figure 23 of Reference 7-31C. If it is conservatively assumed that the density of water is 1 g/cc, this equates to 515385 to 839184 liters. Accounting for the atomic weight of Cs and I, there is approximately 1.86 kg of I in the pool. This equates to a concentration of <3E-05 g-atoms/liter. Figure 3.1 of NUREG-CR-5950 shows that the fraction of iodine in the elemental form for a pH of 3 and an iodine concentration of 1E-5 g-atoms/liter is ~0.4. This analysis will conservatively re-evolve all of the iodine in the LDW (i.e., release fraction of 1.0). The release timing should be based on the time when pH = 3; however, this value will conservatively be reduced by a factor of ~2 for an assumed release duration of 54 hours. The release begins when the LDW pH drops below 7 (210.83 hrs).

Table C-1
Lower Drywell pH Results Crediting 50% CsOH

pH	Time (hr)	Time since pH < 7 (hr)
7	210.8	-
6	223.5	12.7
5	225.5	14.7
4	232.8	22.0
3	317.5	106.7

C.3.2 Plateout Modeling

The RADTRAD model used to evaluate re-evolution is very similar to that used to evaluate the overall dose consequences. The primary exception is the removal coefficient timing. Particulate removal in primary containment is irrelevant since the source term is assumed entirely to be elemental iodine. Removal of elemental iodine is credited. The timing is adjusted to correspond to the re-evolution of iodine:

$$A(t') = A(0)e^{-\lambda t'} + S\left(\frac{1-e^{-\lambda t'}}{\lambda}\right)$$

$$A(54hr) = S\left(\frac{1-e^{-\lambda t'}}{\lambda}\right) = \left(\frac{0.0498}{54hrs}\right)\left(\frac{1-e^{-(0.86\,1/hr)(54hr)}}{0.86\,1/hr}\right) = 1.072E-03$$

$$A(t'>54) = A(54)e^{-\lambda(t'-54hr)}$$

Where: t' = 0 signifies the beginning of the re-evolution source term (210.83 hours after the onset of the LOCA). The solution to the above equation necessary to meet SRP requirements is 2.49E-04 (=0.0498/200). The time that corresponds to the solution is ~55.7 hours (A=2.4917E-04); therefore, plateout of elemental iodine for this analysis will be terminated at

210.8 hr + 55.7 hr = 266.5 hrs.

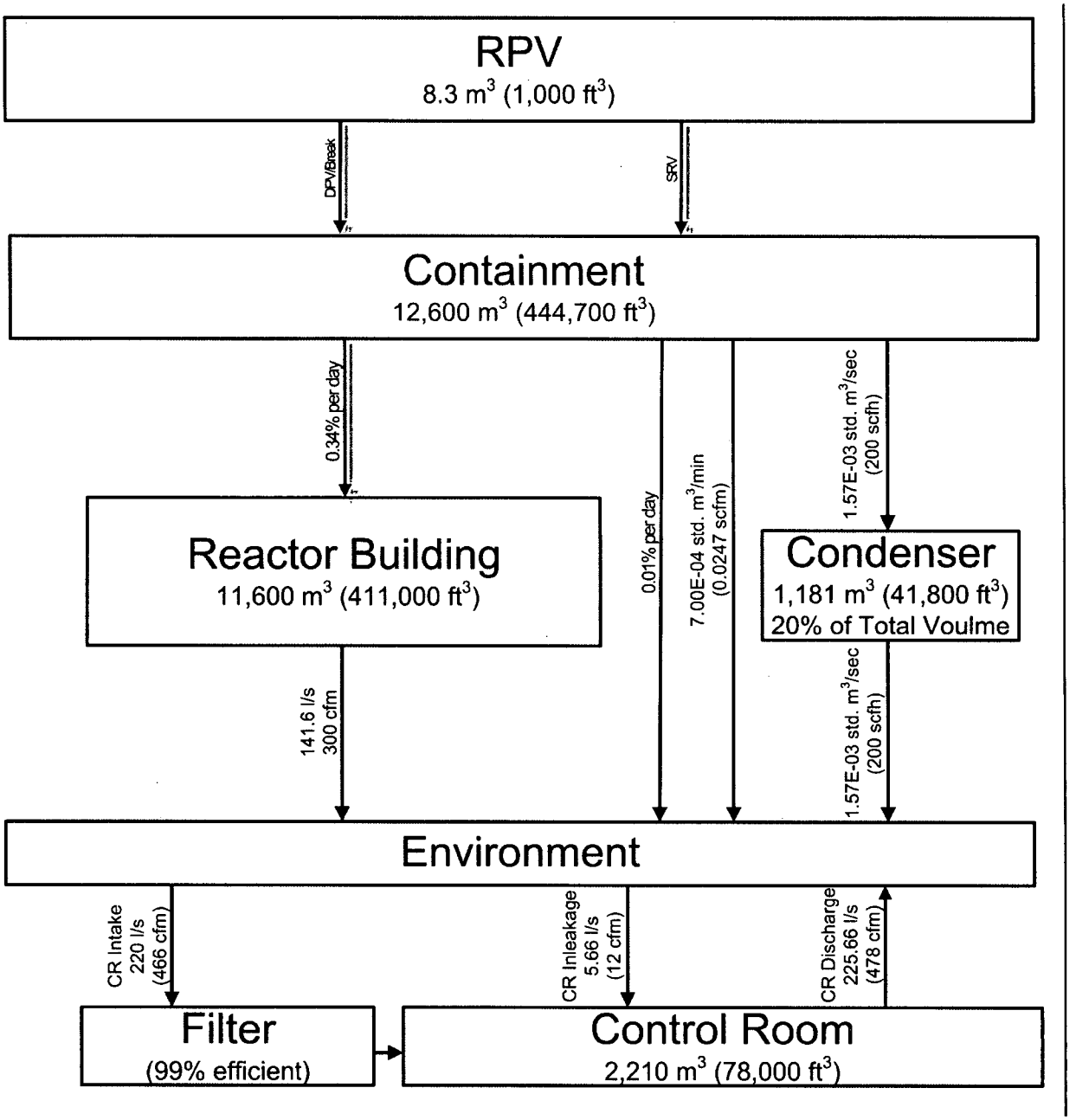

Figure C-1. Lower Drywell Re-Evolution Model (AS-3)

C.4 RESULTS AND DISCUSSION

The results are presented in Table C-2. The results indicate that AS-3 does not become the bounding scenario. The statement in the DCD that AS-1 provides the bounding calculated doses remains valid even if re-evolution of iodine from the LDW is conservatively modeled and accounted for in the final AS-3 doses.

Table C-2
TEDE Dose Calculation Results for LDW Re-evolution

Dose*	LPZ Sv (rem)	CR Sv (rem)
RB	1.63E-03 (1.63E-01)	4.77E-04 (4.77E-02)
PCCS	5.27E-05 (5.27E-03)	2.77E-05 (2.77E-03)
MSIV	7.56E-06 (7.56E-04)	1.68E-06 (1.68E-04)
FW	2.53E-05 (2.53E-03)	5.55E-06 (5.55E-04)
Total	***1.71E-03 (0.17)***	***5.11E-04 (0.05)***
AS-3 Results w/o Re-evolution)	*1.63E-01 (16.3)*	*3.59E-02 (3.59)*
AS-3 Total w Re-evolution	***1.65E-01 (16.5)***	***3.64E-02 (3.64)***
AS-1 Results	*2.07E-01(20.7)*	*4.69E-02 (4.69)*

* EAB doses are calculated for the worst 2-hour period. The base LOCA analysis documented in Section 5 determined that this occurs beginning at 1.6 hours into the event for AS-3; therefore, EAB doses are not increased as a result of re-evolution of iodine (since it occurs >8 days into the event)

MFN 09-437

Enclosure 3

Licensing Topical Report
NEDE-33279P, Revision 3
"ESBWR Fission Product Removal Evaluation Model"

Affidavit

June 30, 2009

GE-Hitachi Nuclear Energy Americas LLC

AFFIDAVIT

I, Larry J. Tucker, state as follows:

(1) I am Manager, ESBWR Engineering, GE-Hitachi Nuclear Energy Americas LLC ("GEH"), have been delegated the function of reviewing the information described in paragraph (2) which is sought to be withheld, and have been authorized to apply for its withholding.

(2) The information to be discussed and sought to be withheld is delineated in the letter from Mr. Richard E. Kingston to U.S. Nuclear Regulatory Commission, entitled "Transmittal of Licensing Topical Reports (LTR) NEDE-33279P and NEDO-33279, Revision 3 "ESBWR Fission Product Removal Evaluation Model" dated June 30, 2009. The information in Enclosure 1, which is entitled *Licensing Topical Report NEDE-33279P, Revision 3 "ESBWR Fission Product Removal Evaluation Model" – GEH Proprietary Information* contains proprietary information, and is identified by [[dotted underline inside double square brackets$^{\{3\}}$]]. Figures and other large objects are identified with double square brackets before and after the object. In each case, the superscript notation $^{\{3\}}$ refers to Paragraph (3) of this affidavit, which provides the basis for the proprietary determination.

(3) In making this application for withholding of proprietary information of which it is the owner or licensee, GEH relies upon the exemption from disclosure set forth in the Freedom of Information Act ("FOIA"), 5 USC Sec. 552(b)(4), and the Trade Secrets Act, 18 USC Sec. 1905, and NRC regulations 10 CFR 9.17(a)(4), and 2.390(a)(4) for "trade secrets" (Exemption 4). The material for which exemption from disclosure is here sought also qualify under the narrower definition of "trade secret", within the meanings assigned to those terms for purposes of FOIA Exemption 4 in, respectively, Critical Mass Energy Project v. Nuclear Regulatory Commission, 975F2d871 (DC Cir. 1992), and Public Citizen Health Research Group v. FDA, 704F2d1280 (DC Cir. 1983).

(4) Some examples of categories of information which fit into the definition of proprietary information are:

a. Information that discloses a process, method, or apparatus, including supporting data and analyses, where prevention of its use by GEH's competitors without license from GEH constitutes a competitive economic advantage over other companies;

b. Information which, if used by a competitor, would reduce his expenditure of resources or improve his competitive position in the design, manufacture, shipment, installation, assurance of quality, or licensing of a similar product;

c. Information which reveals aspects of past, present, or future GEH customer-funded development plans and programs, resulting in potential products to GEH;

d. Information which discloses patentable subject matter for which it may be desirable to obtain patent protection.

The information sought to be withheld is considered to be proprietary for the reasons set forth in paragraphs (4)a. and (4)b. above.

(5) To address 10 CFR 2.390(b)(4), the information sought to be withheld is being submitted to NRC in confidence. The information is of a sort customarily held in confidence by GEH, and is in fact so held. The information sought to be withheld has, to the best of my knowledge and belief, consistently been held in confidence by GEH, no public disclosure has been made, and it is not available in public sources. All disclosures to third parties, including any required transmittals to NRC, have been made, or must be made, pursuant to regulatory provisions or proprietary agreements which provide for maintenance of the information in confidence. Its initial designation as proprietary information, and the subsequent steps taken to prevent its unauthorized disclosure, are as set forth in paragraphs (6) and (7) following.

(6) Initial approval of proprietary treatment of a document is made by the manager of the originating component, the person most likely to be acquainted with the value and sensitivity of the information in relation to industry knowledge, or subject to the terms under which it was licensed to GEH. Access to such documents within GEH is limited on a "need to know" basis.

(7) The procedure for approval of external release of such a document typically requires review by the staff manager, project manager, principal scientist, or other equivalent authority for technical content, competitive effect, and determination of the accuracy of the proprietary designation. Disclosures outside GEH are limited to regulatory bodies, customers, and potential customers, and their agents, suppliers, and licensees, and others with a legitimate need for the information, and then only in accordance with appropriate regulatory provisions or proprietary agreements.

(8) The information identified in paragraph (2) above is classified as proprietary because it contains modeling of the containment fission product removal mechanisms developed by GEH. Development of this containment fission product removal model was achieved at a significant cost to GEH.

(9) Public disclosure of the information sought to be withheld is likely to cause substantial harm to GEH's competitive position and foreclose or reduce the availability of profit-making opportunities. The information is part of GEH's comprehensive BWR safety and technology base, and its commercial value extends beyond the original development cost. The value of the technology base goes beyond the extensive physical database and analytical methodology and includes development of the expertise to determine and apply the appropriate

evaluation process. In addition, the technology base includes the value derived from providing analyses done with NRC-approved methods.

The research, development, engineering, analytical and NRC review costs comprise a substantial investment of time and money by GEH.

The precise value of the expertise to devise an evaluation process and apply the correct analytical methodology is difficult to quantify, but it clearly is substantial.

GEH's competitive advantage will be lost if its competitors are able to use the results of the GEH experience to normalize or verify their own process or if they are able to claim an equivalent understanding by demonstrating that they can arrive at the same or similar conclusions.

The value of this information to GEH would be lost if the information were disclosed to the public. Making such information available to competitors without their having been required to undertake a similar expenditure of resources would unfairly provide competitors with a windfall, and deprive GEH of the opportunity to exercise its competitive advantage to seek an adequate return on its large investment in developing and obtaining these very valuable analytical tools.

I declare under penalty of perjury that the foregoing affidavit and the matters stated therein are true and correct to the best of my knowledge, information, and belief.

Executed on this 30th day of June 2009.

Larry J. Tucker
GE-Hitachi Nuclear Energy Americas LLC