Enclosure 2

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Non-Proprietary 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)
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
EBAS	Emergency Breathing Air System
ECCS	Emergency Core Cooling System
EFU	Emergency Filter Unit
EIV	Early In-vessel Release Phase for AST
ESBWR	Economic Simplified Boiling Water Reactor
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
HELB	High Energy Line Break

HVAC	Heating, Ventilation, and Cooling System
IC	Isolation Condenser
IFTS	Inclined Fuel Transfer System
LOCA	Loss of Coolant Accident
LPZ	Low Population Zone
LTR	Licensing Topical Report
MELCOR	NRC Code to Evaluate Severe Accidents
MSIV	Main Steam Isolation Valve
MSLDL	Main Steam Lines Drain Lines
MSL	Main Steam Lines
MWth	Mega-Watt Thermal
NRC	United States Nuclear Regulatory Commission
PCCS	Passive Containment Cooling System
PCT	Peak Cladding Temperature
PWR	Pressurized Water Reactor
RADTRAD	NRC Code used to Evaluate Off-Site and Control Room Dose Consequences
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
TRACG	GE version of the Transient Reactor Analysis Code
VFR	Volumetric Flow Rate
χ/Q or X/Q	Atmospheric Dispersion Factor (Chi over Q)

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. 4]. Following the Three Mile Island accident, the US 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. 12]. The NRC issued Regulatory Guide 1.183, Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors [Ref. 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 Release 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 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 50.34(a)(1) and 10 CFR 50, Appendix A, General Design Criterion (GDC) 19.

1.3 Accident Scenarios Evaluated

There are numerous "Loss of Coolant Accident" (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 design basis accident, such as a LOCA. No active systems, such as containment sprays or Standby Gas Treatment Systems (SGTS), 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, Top Ten (Probabilistic Risk Assessment [PRA]) Level 1 Accident Scenarios of NEDO-33201, Revision 1 [Ref. 37] was reviewed. Based on the review a Loss of Preferred Power with failure of IC's and high pressure makeup, 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 FW vs. Loss of Preferred power) contribute to about 90% of the 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 3 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, restoration of adequate core cooling after 2 hrs. The isolation condensers (IC) are not credited in any of the three scenarios. Injection by the Standby Liquid Control (SLC) system is not credited to mitigate fuel damage for any of the three scenarios, however it is credited in the determination of post accident pool pH levels. Injection of the Gravity Drain Cooling System (GDCS) is inhibited until approximately 2 hours following the onset of fuel damage. The Reactor Pressure Vessel (RPV) Automatic Depressurization System (ADS) is assumed to operate as designed. A detailed sequence of events is provided in Table 1.1.
- **Accident Scenario 2 (AS-2):** Bottom drain line break, with degraded high pressure makeup and ADS failure, restoration of adequate core cooling after 2 hrs. 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. A detailed sequence of events is provided in Table 1.2.

- **Accident Scenario 3 (AS-3):** Loss of Preferred Power/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. A detailed sequence of events is provided in Table 1.3.

Table 1.1 – Accident Scenario 1 Sequence of Events

Event	Timing (s)
Reactor SCRAM	0
Level 2 Signal	2
Level 1 Signal	8
Safety Relief Valve (SRV) #1 Open	19
SRV #2 Open	64
Depressurization valve (DPV) # 1 Open	108
DPV #2 Open	154
Level Top of Active Fuel (TAF) – Core Uncovered	327
Fuel Oxidation Starts	1486
Fission Product Gap Release Begins	1983
Core Fully Uncovered	2125
Onset of Core Support Plate Failure	5372
GDCS injection line open*	7400
Equalization line opened	7400
Core fully recovered	8220

* Two separate runs are documented in the VTT reports for AS-1 due to complications with MELCOR in the initial development for the ESBWR model. One run assumed GDCS injection began 6083 seconds after the break, and the second assumed GDCS injection at 7400 seconds. The removal coefficients determined in Section 4 are based on the 7400 second run.

Table 1.2 – Accident Scenario 2 Sequence of Events

Event	Timing (s)
Feedwater off	0
Bottom Drain Line Break	0
Level 2 signal	2
Level 1 signal	7
MSIV closed	23.2
SRV Pressure Relief Operation Begins	116
Level TAF – Core Uncovered	938
Level Bottom of Active Fuel (BAF)	1361
Cladding oxidation begins	1421
FP release begins	1539
RPV lower head empty of water	2436
Core support structure begins to Fail	4181
GDCS injection started	5140
DPV#1 open	5140
Equalization Line opened	5140
ADS/SRV#1 started	5150
DPV#2 open	5150
ADS/SRV#1 started	5195

Table 1.3 – Accident Scenario 3 Sequence of Events

Event	Timing [s]
Feedwater off – Loss of A/C Power	0
MSIV closed	0
Level 2 signal	2
Level 1 signal	3
SRV pressure relief operation begins	96
Level TAF – Core Uncovered	4916
Cladding oxidation begins	6222
FP release begins	6578
Water level in the core reaches BAF	8505
Equalization Line Opened	13780
DPV open	13780
GDCS injection started	13780
ADS/SRV#1 open	13818

2.0 LICENSING REQUIREMENTS

2.1 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 50.34

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) total effective dose equivalent (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) total effective dose equivalent (TEDE).

2.3 Standard Review Plan Guidelines (NUREG-0800)

SRP Section 6.2.1, Revision 2, "Containment Functional Design," was issued in July 1981 [Ref. 5]. This SRP discusses the requirements to ensure that primary containment for reactors meets GDC 16, 50, 52, 53, and 54 through 57. 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, Revision 2, "Secondary Containment Functional Design," was issued in July 1981 [Ref. 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" per se, however the Reactor Building 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, Revision 3, "Containment Spray as a Fission Product Cleanup System" was issued in December 2005 [Ref. 26]. The ESBWR does not credit containment sprays to remove airborne radioiodine following a LOCA. However, SRP Section 6.5.2 also contains information on methodology acceptable to quantify removal of elemental iodine through deposition on containment surfaces.

SRP Section 6.5.5, Revision 0, "Pressure Suppression Pool as a Fission Product Cleanup System," was issued in December 1988 [Ref. 7]. The SRP provides guidance to licensees concerning the amount of radioactivity that may be removed via suppression pool scrubbing. 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, Revision 0, "Radiological Consequence Analyses Using Alternative Source Terms," was issued in July 2000 [Ref. 8]. This SRP section contains information concerning the requirements for licensees which voluntarily adapt the AST dose methodologies, including the results for the LOCA and other design basis events (Main Steam Line Break Outside Containment, Fuel Handling Accident, etc.). 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," was issued in March 2007 [Ref. 38]. This SRP section 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. 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 50.34(a)(1), 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 Design Basis Accidents (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, Revision 2, "Loss-of-Coolant Accidents Resulting from Spectrum of Postulated Piping Breaks within the Reactor Coolant Pressure Boundary," was issued in July 1981 [Ref. 9]. Information concerning acceptable assumptions with respect to containment releases for dose consequence analyses is provided in several Appendices. Appendix A addresses assumptions concerning most LOCA dose calculation assumptions, 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 Subsection 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, Revision 0, "Alternative Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," was issued in July 2000. This Regulatory Guide 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 to the docket file dated July 24, 1995 addressing the Regulatory Treatment of Non-Safety Systems (RTNSS) for advanced passive reactor designs [Ref. 10]. One of the criteria the memo is intended to apply to is "SSC 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. 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 Reactor Building;
- 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 determine models that may be used to quantify various fission product removal mechanisms. Also, 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 US 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, *Accident Source Terms for Light Water Nuclear Power Plants* [Ref. 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. 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 4 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 3.0.3 was used for the dose consequence calculations documented in this Licensing Topical Report.

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.

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. 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 6 of the 7 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 sequence a total of 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 of 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 fraction 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 an accident. The methodology used is consistent with NUREG/CR-5950, Iodine Evolution and pH Control [Ref. 35]. NUREG/CR-5950 discusses a number of chemicals that would potentially affect the post-accident in the containment pools. Each of the contributors is discussed below.

Carbon Dioxide:

Carbon Dioxide (CO2) depresses the pH of pure water by absorption. Carbonic acid is a weak acid and is insignificant compared to the other acids produced in the primary containment during an accident. However, the initial pool pH may be depressed below 7.0 during normal operations by the absorption of CO2. NUREG/CR-5950, Section 2.2.3 states that pure water will attain a pH approaching 5.65 due to absorption of CO2 from air and the subsequent formation of Carbonic Acid. The initial pH is assumed to be 5.3. 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 CO2 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. For the main analysis the cesium that is not in the chemical form of CsI is assumed to exit the RCS in the form of cesium hydroxide (CsOH). [[

]

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 the all iodine was released from the core during core heatup phase as CsI. Further, the calculated mole fraction of HI in the gas phase in the containment was calculated to be negligible. [[

]]

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.

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. 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. 36] were then used to determine submersion doses.

Infinite Cloud (Cloud centered)

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

Semi-infinite Cloud (Surface body)

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

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. 17] provides such a factor for main control room dose calculations:

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

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

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

Dose rates and time-integrated dose (TID) were then determined, which in turn was used to determine the HCl released as a result of radiolysis. The doses shown in Table 4.1 were calculated based on the containment airborne activity for AS-1. The doses were conservatively increased by 10% for determining HCl for AS-2 and AS-3. [[

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Table 4.1 – Containment Airborne Dose Rates and Doses for AS-1				
Time (hr)	Beta Dose Rate (R/hr)	Gamma Dose Rate (R/hr)	Beta TID (Rad)	Gamma TID (Rad)
0.83	1.74E+06	5.40E+05	7.12E+05	2.22E+05
1.23	6.88E+06	1.70E+06	2.44E+06	6.69E+05
1.83	1.20E+07	2.65E+06	8.10E+06	1.97E+06
2.33	1.52E+07	3.15E+06	1.49E+07	3.42E+06
3.00	1.26E+07	2.35E+06	2.42E+07	5.26E+06
6.00	8.11E+06	1.10E+06	5.52E+07	1.04E+07
8.33	6.84E+06	7.52E+05	7.27E+07	1.26E+07
12.00	5.73E+06	4.75E+05	9.57E+07	1.48E+07
24.33	4.18E+06	2.22E+05	1.57E+08	1.91E+07
720.33	1.21E+05	3.85E+03	1.65E+09	9.79E+07

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 x 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. [[

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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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] The buffer is mixed into the RPV water inventory and transported according to BDL break flow to the lower drywell.

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 with 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 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, 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 $H^0_{i,298}$, standard entropy $S^0_{i,298}$, and heat capacity polynomial $C_{pi}(T)$ (fitted from measured data).

The thermodynamic software that ChemSheet uses (ChemApp) for equilibrium calculation 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 as assumed to be ideal so 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 realistic calculation of solution equilibrium necessitate 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. 33], which is widely used, was also applied in this work (and is included to 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, $\gamma_\pm$:

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

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 strength of 20 m [Ref. 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 base looks like the curve in Figure 4.1.

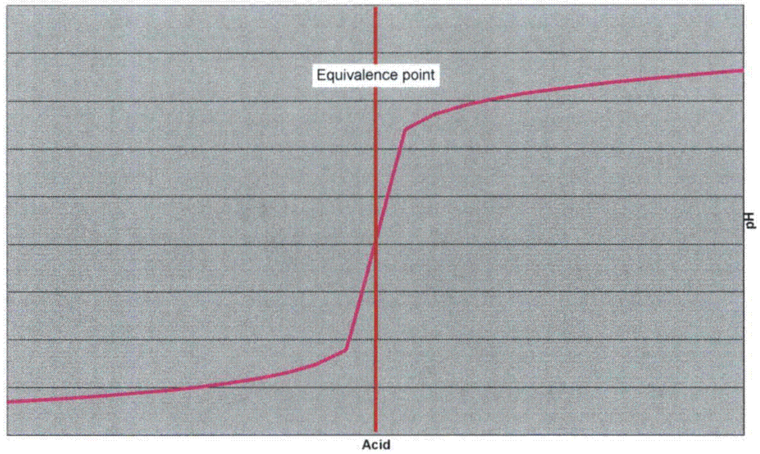

Figure 4.1: Typical Titration Curve

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 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 more striking effect to 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. 31]. The scenario most applicable to the ESBWR (Case A) calculates the impact of both HCl and HNO_3.

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4.1.3 Release Timing

NUREG-1465 states that for BWRs the gap release is assumed to begin 30 seconds into the event and last for 30 minutes, and the Early In-Vessel (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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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. 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. 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],

$$\frac{dT_l}{dx} = -\frac{P\left[-q^{Ts} + q^{T} + Lq''' + c_{pw}(T - T_l)\right]}{\dot{m}_l c_{pl}}$$

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/m2] 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. 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:

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

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

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

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

$$q_w''' = Sh \cdot D \cdot \frac{\rho_w - \rho_{ws}}{d_h},$$

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. 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 the 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 however 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. 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. 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. 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 as 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. 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:

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. 26]. Specifically, the iodine removal rate constant for a particular compartment "n" will be based on the following formula:

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

where,

λ_n = removal rate constant 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 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:

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

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

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

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

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

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

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

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. 17] and NUREG/CR-6189 [Ref. 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. 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 mass values were obtained from the MELCOR results. Each accident scenario was reviewed independently, as shown in Figures 4.9, 4.10, and 4.11 for AS-1, AS-2, and AS-3, respectively. The computer code RADTRAD only allows up to 10 removal coefficients to be input for dose calculations. The values assumed in the RADTRAD dose calculations are also shown in the figures. The RADTRAD values summarized in Table 4.2 are adjusted to correspond to the dose calculation

assumption that fuel damage begins from the onset of the event, as discussed previously in Section 4.1.3.

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Table 4.2 – RADTRAD Removal Coefficients

Time Step	Accident Scenario 1		Accident Scenario 2		Accident Scenario 3	
	Start Time (hr)	λ (hr^{-1})	Start Time (hr)	λ (hr^{-1})	Start Time (hr)	λ (hr^{-1})
Time Correction*	0.333	-	0.333	-	1.917	-
1	0.000	4.5	0.000	0.00	0.00	0.40
2	0.500	2.5	0.834	2.00	0.83	3.00
3	2.167	0.8	2.000	0.80	2.58	0.90
4	2.667	0.5	4.167	0.40	2.83	0.60
5	3.917	1.0	7.167	0.25	3.58	0.80
6	4.917	0.8	8.667	0.20	5.08	0.40
7	5.417	0.4	12.000	0.00	7.08	0.20
8	7.167	0.2	-	-	7.41	0.15
9	8.917	0.1	-	-	8.58	0.20
10	12.000	0.0	-	-	12.00	0.00

* The value listed represents the delay for fuel damage based on the MELCOR results. The dose calculation conservative neglects this time period as discussed in Section 4.1.

4.4 Main Steam Isolation Valve Leakage

Leakage past Main Steam Isolation Valves (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 GEHand 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. 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. 4]. As such the methodology may not be accurate for use with AST assumptions.

Plateout in the main steam lines or main steam lines' drain lines is not explicitly modeled; therefore, leakage though the MSIVs is assumed to be released directly into the condenser. The condenser is used as a mitigative volume based upon the determination that such components, designed to standard engineering practice, are sufficiently strong to withstand SSE conditions. As such, it will serve as aholdup volume for decay for leakage past the MSIVs following a LOCA. Figure 4.14 sketches the piping and valves for the ESBWR relevant to implementing the BWROG methodology.

Credit for deposition in the main condenser is credited based on the BWROG methodology. The value assumed in the dose consequence analysis is based on the Advanced Boiling Water Reactor (ABWR) design. Table 15.6-8 of the ABWR DCD [Ref. 41] lists a plateout fraction of 0.993 for the ABWR condenser. This value was determined for the ABWR using the older TID dose methodology [Ref. 4]. Additional analyses were performed to ensure that the value assumed in the dose consequence analysis bounds the ESBWR. The main steam lines, main steam line drain lines, and the condenser were evaluated in Part 3 of the VTT MELCOR analyses [Ref. 31]. All three accident scenarios were reviewed. The analyses calculated plateout fractions of 99.45% for AS-1, 99.65% for AS-2, and 99.83% for AS-3. As such, the value of 99.3% assumed based on the ABWR is conservative for the ESBWR dose analyses.

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, Revision 3, Table 6.2-1 states that the containment design pressure is 414 kPa (60 psia) 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, Revision 3, Figures 6.2-9 and 6.2-10 show the pressure and temperature following the "nominal" Feedwater Line Break (FWLB). DCD Figures 6.2-12 and 6.2-13 show the pressure and temperature for the "bounding case" FWLB, respectively. Adjustments factors as a function of time were calculated for both DCD cases and are presented in Figures 4.12 and 4.13.

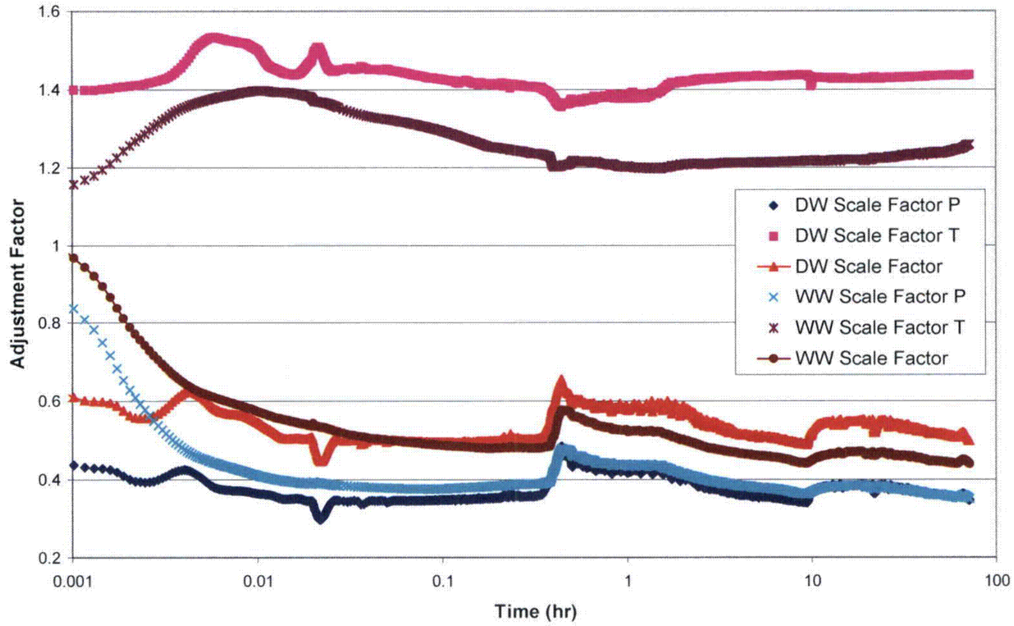

Figure 4.12: MSIV Adjustment Factor – DCD Nominal Case

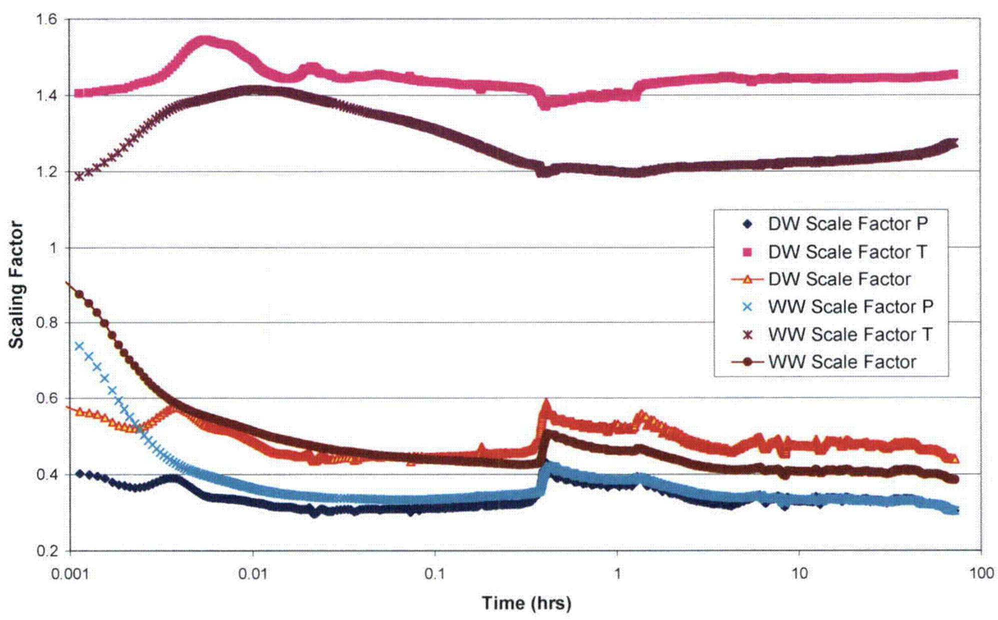

Figure 4.13: MSIV Adjustment Factors DCD Bounding Analysis

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]]. The bounding MSIV adjustment factor was calculated to be 0.66 at 0.44 hours as shown in Figure 4.12. The calculated containment pressure at 0.44 hour is 210.5 kPa, and the calculated temperature is 391°K per DCD Tier 2 Revision 1, Figures 6.2-9 and 6.2-10. The adjustment factor based on this temperature and pressure was conservatively applied to adjust MSIV leakage for the duration of the event, therefore the MSIV leakage rate is

$$\dot{V}_{MSIV}(cfh) = (200scfh)(0.66) = 132cfh = 2.2cfm.$$

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

$$\dot{V}_{Cond}(cfh) = (200scfh)(1.0) = 200cfh = 3.33cfm$$

[[

]] Detailed tables were not provided in the VTT reports for AS-2 or AS-3. However, there is significant margin based on AS-1 such that no concerns exist with the remaining accident scenarios.

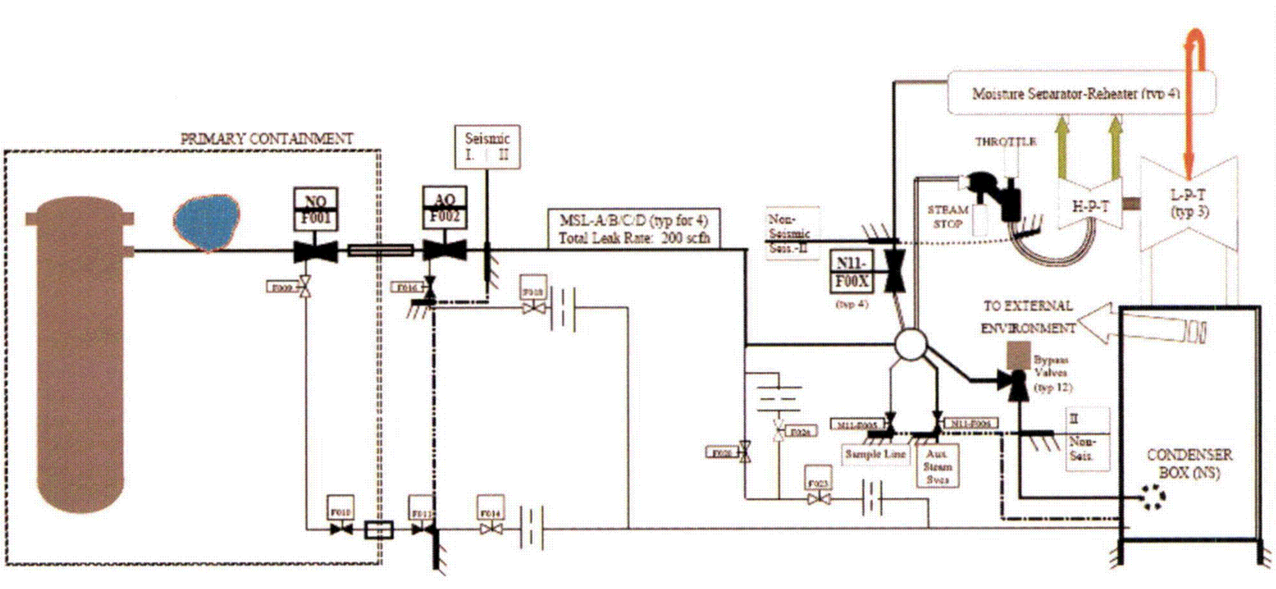

Figure 4.14: ESBWR Main Steam Line and Condenser Piping Sketch for Dose Calculations

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 Gravity Driven Cooling System (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 PCC condensers contain a steam/air/water mixture. Any leakage from the PCC 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. Similarly, the Isolation Condensers 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 Reactor Building (RB) is discussed in depth in Subsection 6.2.3 of Tier 2 of the ESBWR Design Control Document [Ref. 19]. The building is of a robust design and is 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 Reactor Building.

SRP Section 6.2.3, Revision 2, "Secondary Containment Functional Design," was issued in July 1981 [Ref. 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 Reactor Building is not held to the required vacuum of –0.25" w.g., however the Reactor Building is credited for the holdup of fission products prior to the release to the atmosphere. Regulatory Guide 1.183 [Ref. 3] allows a maximum of 50% of the secondary containment volume to be credited for holdup and decay. The Reactor Building is credited in the design basis LOCA dose consequence analysis for the holdup and decay of fission products. A review of the containment penetration locations was performed. This review is documented in Attachment B. The review determined that 40% of the Reactor Building volume would be available for mixing for leakage through the containment penetrations.

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 Reactor Building 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 available to monitor plant releases and appropriate measures would be taken to mitigate the consequences of the accident. Therefore, engineering judgment dictates that the bounding scenario is with no A/C power. Technical Specifications (TS) Surveillance Requirement (SR) 3.6.3.1.4, located in Tier 2, Revision 3 of the DCD, requires verification that the Reactor Building exfiltration rates are within limits. This analysis assumed an overall Reactor Building leakage rate of 50% per day.

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

There are some potential containment leakage paths that may not readily mix with the Reactor Building volume. Of specific concern are the PCCS condensers. Although leakage past the condensers and associated piping would be released into the Reactor Building, 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 Reactor Building 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.

4.6 Safety Relief Valve (SRV) Flow and Suppression Pool Scrubbing

Regulatory Guide 1.183, Appendix A, Section 3.5 states

> "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

> "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 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. 31]. The MELCOR calculations determined very high DFs during the time periods corresponding to the gap and EIP release phases (DF ranges from ~100 to ~1E12). As such, applying a DF of 10 is clearly reasonable and conservative. Note that the MELCOR analysis did not explicitly calculate DFs for elemental iodine. Engineering judgement dictates that the results would be similar, and use of a DF of 10 is reasonable for elemental iodine as well.

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

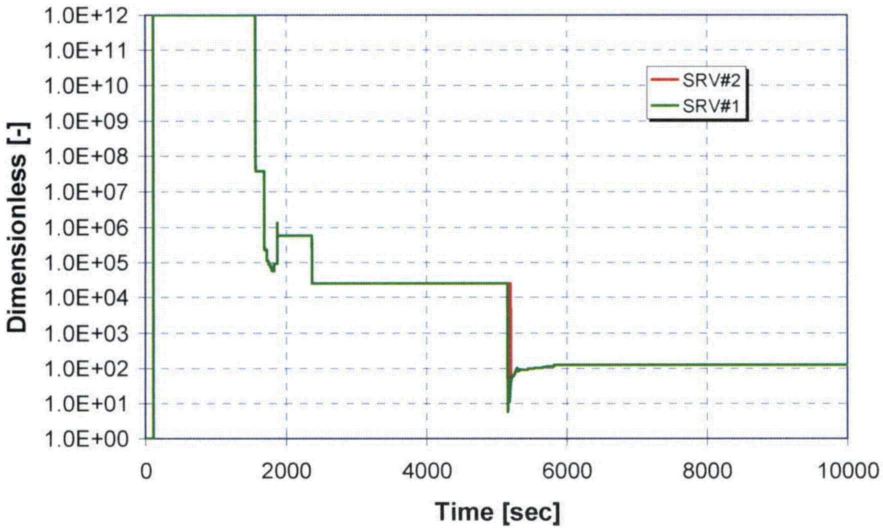

Figure 4.15: AS-2 Instantaneous CsI Aerosol DF for SRVs Suppression Pool Discharge

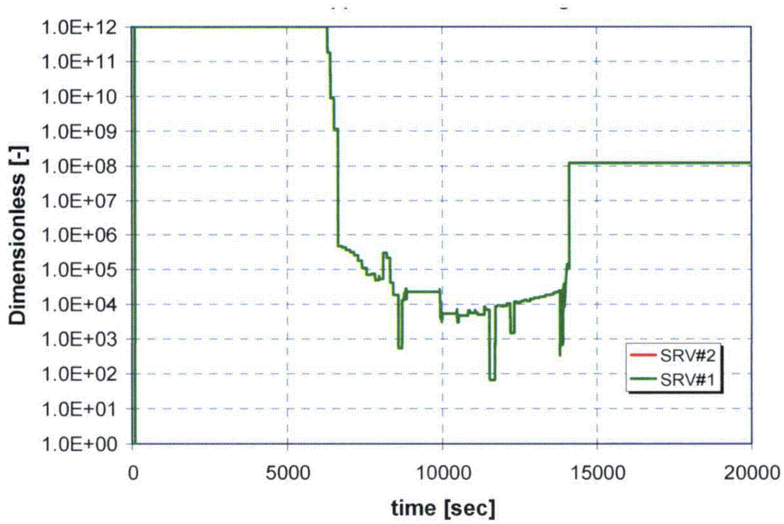

Figure 4.16: AS-3 Instantaneous CsI Aerosol DF for SRVs Suppression Pool Discharge

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. Figure 21 of Part 2 of the VTT report shows the SRV and DPV flow for AS-2, and Figure 22 shows the flow through the break itself. For AS-3 the flow through the SRVs and DPVs are show in Figure 2 of Part 2 of the VTT report. 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 4.4.

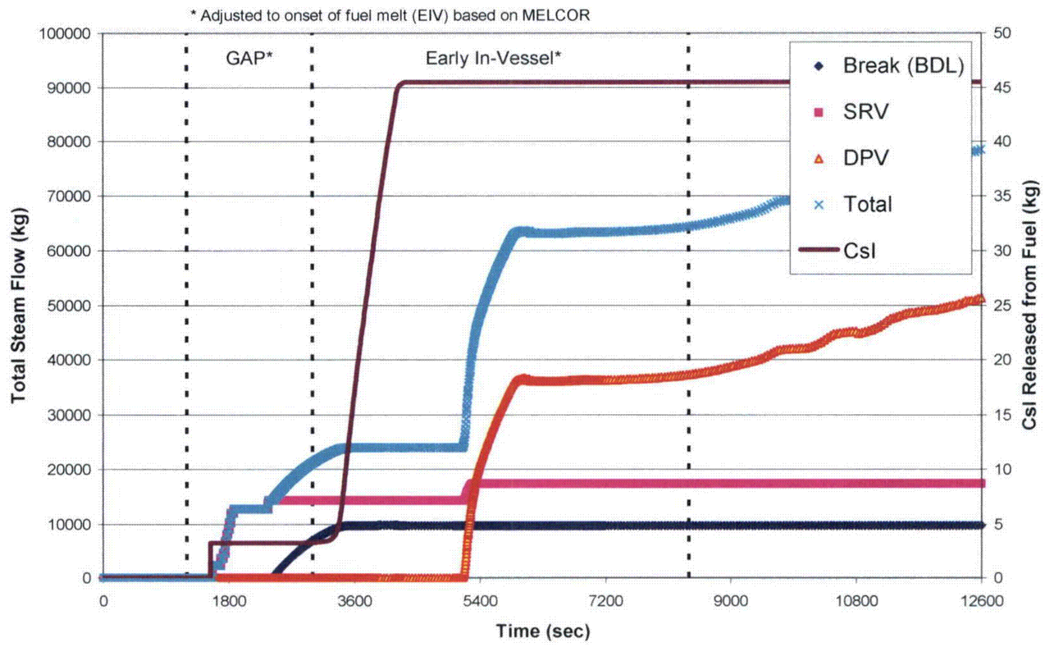

Figure 4.17: Integral Steam Flow After Fuel Damage for AS-2

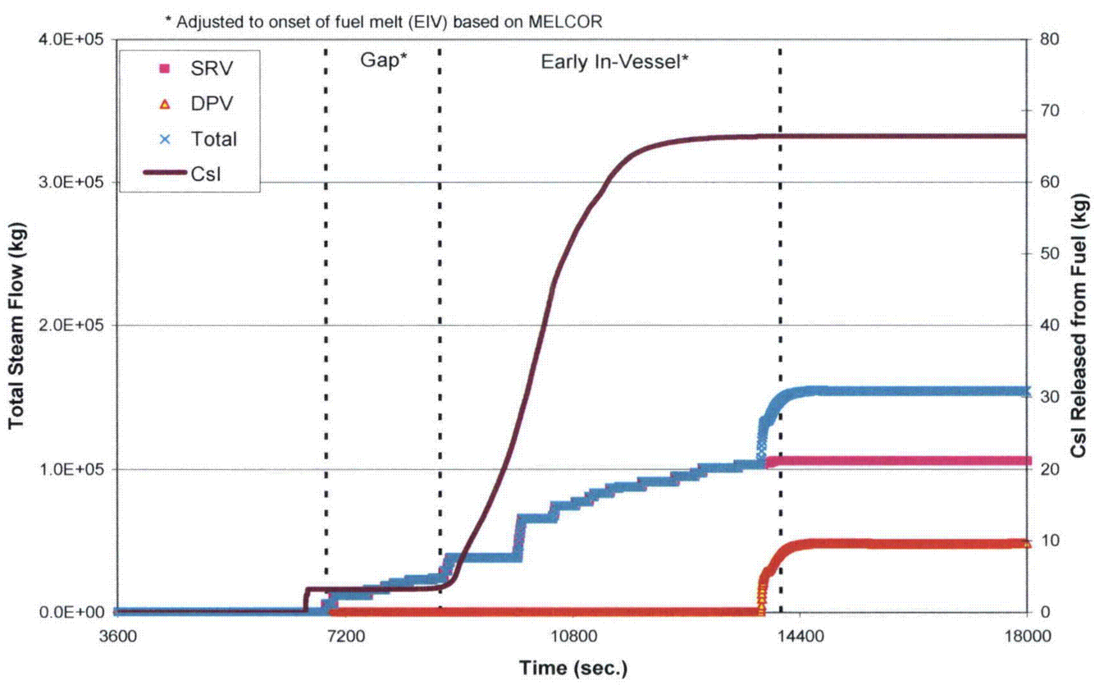

Figure 4.18: Integral Steam Flow After Fuel Damage for AS-3

Table 4.4: SRV Flow Rate Distribution

	AS-2		AS-3	
	SRVs	**DPV/Break**	**SRVs**	**DPV***
Flow % - Gap Actual	61.4%	38.6%	100.0%	0.0%
Flow % - Gap Assumed	60.0%	40.0%	100.0%	0.0%
Flow % - EIV Actual	7.4%	92.6%	63.5%	36.5%
Flow - EIV Assumed	7.0%	93.0%	60.0%	40.0%

* 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 Reactor Building 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. This analysis credits only 40% of the Reactor Building volume based on the ESBWR configuration (Appendix B). This analysis will assume a building release rate of 50% per day.

The containment leakage rate is assumed to be 0.4 wt. % per day[1]. The majority of the containment leakage term is released into the Reactor Building (0.39 containment wt. % per day). The remaining containment leakage is released directly to the environment via the PCCS/IC pool airspace.

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 30 seconds into the event and last for 30 minutes, and the Early In-Vessel (EIV) release is assumed to begin at ~30 minutes and last for 1.5 hours. The dose calculation conservatively neglects the 20 minute decay time for AS-1 and AS-2, as well as the 1.9 hour delay before the release in AS-3. 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 Table 4.2. 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. 39] requirements and are tested in accordance with ASME AG-1 [Ref. 40]. The EFU 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 200 l/s (424 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

[1] Technical Specification Bases for 3.6, and TS 5.5.9 will be updated in DCD, Tier 2, Revision 5 as appropriate to reflect the revised containment leakage rate.

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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For control room doses, separate dispersion factors were chosen for the three dose contributors: (1) containment leakage via the Reactor Building, (2) containment leakage via PCCS leakage, and (3) MSIV leakage. There are also numerous receptor locations: (1) the control building louvers (the assumed location for Control Room unfiltered inleakage), (2) The normal air intake, (3) the "north" emergency air intake, and (4) the "south" emergency air intake. For convenience one set of χ/Q values is used to bound all 3 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 – Reactor Building:** The containment leakage released to the Reactor Building is assumed to be released on the "east" side of the generic ESBWR plant layout. The location is assumed to be a diffuse source.

- **Containment Leakage – PCCS :** The PCCS leakage is assumed to be ducted to the top of the Reactor Building. This location is assumed to be a point release.
- **MSIV Leakage:** MSIV leakage is assumed to be released from the main condenser. This location is assumed to be a diffuse release.

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:

$$Flow_{adj} = \left(200\,{}^{l}\!/\!_{s}\right)\left(\frac{\chi/Q_{\text{intake}}}{\chi/Q_{\text{louvers}}}\right)$$

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

Note * 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 eorth 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/m³
LPZ • 0 - 8 hours • 8 - 24 hours • 1 - 4 days • 4 - 30 days	1.90E-04 sec/m³ 1.40E-04 sec/m³ 7.50E-05 sec/m³ 3.00E-05 sec/m³

Note*: 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 calculated doses for the low pressure scenario consisted of five volumes: (1) the Containment/drywell, (2) the Reactor Building, (3) the Main Condenser, (4) the Environment, and (5) the Control Room. As discussed previously, RADTRAD only allows input of one set of X/Q values for each receptor location. As such, the model shown in Figure 5.2 was divided into three separate input decks. The results are presented in Table 5.4.

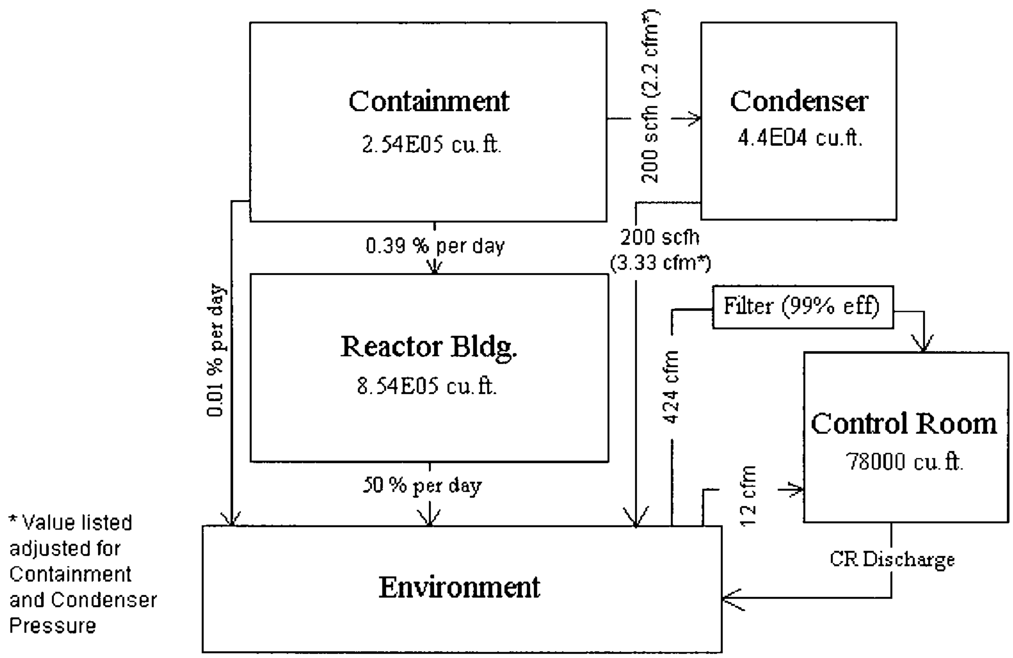

Figure 5.2: AS-1 RADTRAD Model

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 60% of the flow is assumed through the SRVs, and for the EIV release phase only 7% 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 60% of the flow is released via the SRVs, with the remaining 40% released via the DPV. The model used for AS-3 is shown in Figure 5.3. The results are presented in Table 5.4.

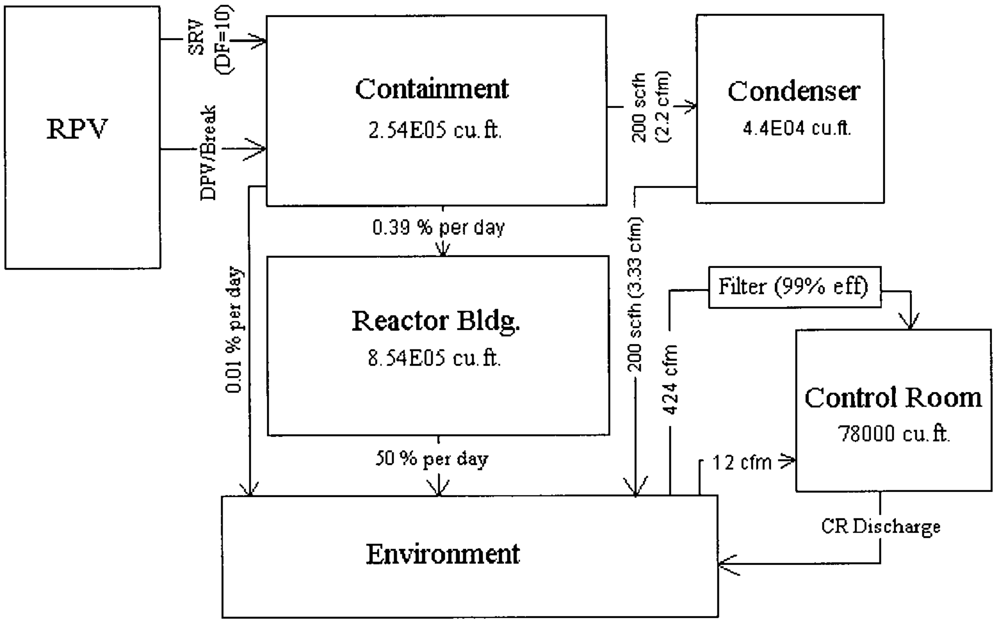

Figure 5.3: AS-2 and AS-3 RADTRAD Model

5.5 Dose Calculation Results

Table 5.4 shows the dose calculation results for all three Accident Scenarios. The results show that AS-2 is bounding for both off-site and control room doses.

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	7.88E+00	1.19E+01	2.85E+00	1.01E+01	1.57E+01	3.86E+00	3.21E+00	4.79E+00	1.01E+00
MSIV	7.16E-01	3.55E+00	5.67E-01	7.25E-01	3.60E+00	5.78E-01	6.99E-01	3.45E+00	5.47E-01
PCCS	3.77E+00	8.08E-01	3.93E-01	4.79E+00	1.05E+00	5.20E-01	1.76E+00	3.85E-01	1.68E-01
Total	***12.37***	***16.23***	***3.81***	***15.59***	***20.37***	***4.96***	***5.66***	***8.63***	***1.72***

6.0 CONCLUSIONS

The ESBWR systems are redundant and diverse. The ESBWR DCD explains that for Design Basis Loss of Coolant Accident 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 design basis accidents and other plant events. The base analysis prepared in support of this report uses reasonable, yet conservative assumptions to evaluate that 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 50.34(a)(1) and 10 CFR 50, Appendix A, GDC 19.

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7-41 "Advanced Boiling Water Reactor Design Control Document," Revision 0.

APPENDIX A 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	0.40	TS 5.5.9
Containment Release Rate to Reactor Building Atmosphere, %/day[Note 1]	0.39	TS 5.5.9
Containment Leakage Rate through PCCS, %/day[Note 1]	0.01	TS 5.5.9
Free Air Volume, m^3 (ft^3)	7.2E+03 (2.54E+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	10	NR
Noble Gases	1	NR
B. Reactor Building		
Leak Rate, %/day	50	TS 3.6.3.1
Mixing Efficiency, %	40	NR
Total Volume, m^3 (ft^3)	6.05E+04 (2.13E+06)	NR
Mixing Volume, m^3 (ft^3)	2.42E+04 (8.54E+05)	ITAAC [Note 3]
C. Condenser Data		

Parameter	Value	Verification Method [Note 2]
Free Air Volume, m^3 (ft^3)	6.23E+03 (2.20E+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)	200 (424)	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 (0.4 wt. % per day) is assumed to leak through two separate pathways. The majority of the leakage (0.39 wt. % per day) is assumed to leak to the Reactor Building, 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.
3. The volume credited in the dose analysis will be verified against plant drawings. For the Reactor Building the final volume should be determined using methodology similar to that presented in Attachment B of this report.

APPENDIX B Reactor Building Mixing Assumptions

This Appendix presents the methodology used to determine the mixing fraction assumed for the Reactor Building. Plant general area diagrams show that there are piping penetrations located on the –6400 mm elevation for the fine motion control rod drive (FMCRD) hydraulic control units (HCU). This elevation also has a personnel airlock (PAL) and equipment airlock (EAL). There are electrical penetrations located on the -1000 mm elevation for the FMCRD. There are also electrical penetrations shown on elevation 13570 mm. The 13570 mm elevation also contains a wetwell access hatch. The main steam and Feedwater lines' penetrations are located on the 17500 mm elevation. These lines penetrate containment and lead to the main steam tunnel. This elevation also contains both electrical and piping penetrations. The electrical and piping penetrations are located inside of individual rooms interior to the building. The elevation also contains a personnel airlock and an equipment hatch.

The PCCS and IC heat exchangers and pools are located on the 27000 mm elevation of the Reactor Building. This elevation, and those above it (34000 mm), is above the primary containment. The floor cross-section is 47m x 51m. Of the 47 m, 27.75 m is attributed to the PCCS and IC equipment and pools, and the remaining 27.25 m includes the reactor well, buffer pool, and the dryer/separator storage pool.

RB Elevation (mm)	Containment Penetrations	Comments
-11500	Piping (HCW and LCW)	RWCU equipment located on this elevation, as well as HCUs and FACPS (Fuel Building). See 26A6407.
-6400	Piping Personnel Airlock Equipment Airlock	CRD equipment.
-1000	Electrical	
4650	None	Suppression Pool on containment side
9060	None	Wetwell on containment side
13570	Electrical	Wetwell on containment side
17500	Electrical Piping (incl. FW and MSL) Personnel Airlock Equipment Hatch	GDCS pools also located on this elevation
27000	Piping	This elevation is above containment, however the PCCS and IC heat exchangers are located at this elevation.
34000	None	Refueling floor

With the exception of the main steam and feedwater lines, and the airlocks, the containment penetrations are roughly equally spaced with the same number in each quadrant. Also, many of

the containment penetrations are shown to be within individual rooms, which would significantly increase holdup with no ventilation (due to the lack of A/C power). The calculated volumes for individual Reactor Building nodes are presented below. The total Reactor Building volume was calculated to be ~60500 m^3. There are no penetrations above the 34000 mm elevation, or on the 9060 and 4650 mm elevations. If those volumes are neglected, the total volume for mixing was determined to be ~25000 m^3, or roughly 40% of the total Reactor Building volume (see following table).

Elevation (mm)	**Reactor Building Volume (m^3)**	**Containment Penetrations**	**Mixing Volume Credited (m^3)**
34000	30179.9	N	0.0
27000	672.0	Y	672.0
17500	5253.0	Y	5253.0
13570	2139.8	Y	2139.8
9060	2650.0	N	0.0
4650	2608.6	N	0.0
-1000	5868.1	Y	5868.1
-6400	5404.5	Y	5404.5
-11500	5726.5	Y	5726.5
Total	**60502.4**		**25063.9**
Mix Fraction	***41.43%***		

Enclosure 3

MFN 06-205, Supplement 2

Affidavit for GE-Hitachi Nuclear Energy Americas LLC Proprietary Information for the NRC

Executed by David H. Hinds, August 30, 2007

GE-Hitachi Nuclear Energy Americas LLC

AFFIDAVIT

I, David H. Hinds, state as follows:

(1) I am the Manager, New Units 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 sought to be withheld is contained in Enclosure 1 of MFN 06-205, Supplement 2, Mr. James C. Kinsey to U.S. Nuclear Regulatory Commission, *MFN 06-205, Supplement 2 – Submittal of Licensing Topical Report (LTR) NEDE-33279P, Revision 1, ESBWR Containment Fission Product Removal Evaluation Model* dated August 30, 2007. The information in Enclosure 1, which is entitled *MFN 06-205, Supplement 2 – Submittal of Licensing Topical Report (LTR) NEDE-33279P, Revision 1, ESBWR Containment Fission Product Removal Evaluation Model,"* contains GEH Proprietary Information. The proprietary information in Enclosure 2, which is entitled "*MFN 06-205, Supplement 2 – Submittal of Licensing Topical Report (LTR) NEDE-33279P, Revision 1, ESBWR Containment Fission Product Removal Evaluation Model*", is delineated by a [[dotted underline inside double square brackets.{3}]]. Figures and large equation 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 identifies detailed GEH ESBWR calculations related to source terms and removal mechanisms and associated offsite and control room dose calculations for three accident scenarios. Development of these source terms and removal mechanisms and associated offsite and control room dose calculations for three accident scenarios was achieved at a significant cost to GEH, on the order of four hundred thousand dollars and would result in a significant economic and competitive advantage to a competitor, and constitutes a major GEH asset.

(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 August, 2007.

David H. Hinds
GE-Hitachi Nuclear Energy Americas LLC

Enclosure 4

MFN 06-205, Supplement 2

Licensing Topical Report (LTR) NEDE-33279P, Revision 1
ESBWR Containment Fission Product Removal
Evaluation Model, August 2007

Summary of Revisions

Summary of Revisions

Revision 0: Initial Issue

Revision 1: Revision 1 is a complete revision to the report. This revision includes the following items:

- Total containment leakage rate reduced from 0.5 wt. % per day to 0.4 wt. % per day.
- Revision to the Control Room model. Specifically, the Emergency Breathing Air System (EBAS) has been replaced with Safety Related Control Room Emergency Filter Units (EFUs).
- Minor changes to the removal coefficients for Accident Scenario 1 due to minor changes in recovery time for emergency core cooling.
- Addition of dose and pH calculations for Accident Scenario 2, high pressure bottom line break, and Accident Scenario 3, loss of feedwater/loss of A/C power.
- Addresses numerous NRC RAIs and other concerns, including RAI 15.4-6, RAI 15.4-7, RAI 15.4-8, RAI 15.4-16, RAI 15.4-17, RAI 15.4-22, RAI 15.4-24, RAI 15.4-25, and comments from the NRC audit in May 2007. Note that there were multiple impacts for many RAIs; therefore, the revision to the LTR may only include a portion of the overall resolution. The RAI responses contain the complete resolutions.
- Addition of information concerning Suppression Pool scrubbing for Safety Relief Valve flow.
- Condenser removal coefficient revised to 99.3% consistent with the ABWR value. Additional information added for ESBWR specific analyses that demonstrate that the ABWR value is more conservative than the results from the ESBWR for all three accident scenarios.
- Release timing for dose calculations changed to coincide with Regulatory Guide 1.183 timing (Revision 0 of LTR credited 20 minute decay time until fuel damage).
- Revised dose consequence model with respect to modeling PCCS leakage. PCCS leakage is now assumed to be released directly to the environment with no credit for holdup in PCCS pool airspace (Revision 0 credited a small amount of holdup in the airspace).

Detailed Revision Description

Chapter 1.0

Section 1.1 – Minor change deleting superfluous information.

Section 1.2 – Editorial changes deleting comparisons to DCD, Revision 1 LOCA analysis.

Section 1.3 – Minor editorial changes. Also, added information for Accident Scenarios 2 and 3 and detailed sequence event tables for all three Accident Scenarios.

Chapter 2.0

Section 2.1 – No changes.

Section 2.2 – No changes.

Section 2.3 – Updated information to reflect new SRP Section revisions from March 2007.

Section 2.4 – No changes.

Section 2.5 – No changes.

Chapter 3.0

Section 3.1 – No changes.

Section 3.2 – Updated to clarify RADTRAD Version 3.0.3 was used in the dose analyses.

Section 3.3 – No changes.

Chapter 4.0

Note there were some editorial changes throughout the chapter. They are not discussed in detail unless considered significant.

Section 4.1 – Significant revision to this section.

- pH information updated for Accident Scenario 1.
- Information for pH analyses added for Accident Scenarios 2 and 3 (both inputs and results).
- Subsection 4.1.3 updated to clarify that dose consequence analyses now assume release timing in accordance with Regulatory Guide 1.183. Decay time of 20 minutes for AS-1 and AS-2, and 1.92 hours for AS-3 conservatively neglected. However, containment removal coefficients still use MELCOR timing to ensure removal coefficients correspond to when actual fission products may be present.

Section 4.2 – No changes.

Section 4.3 – Significant revision to this section.

- Revised elemental plateout coefficient. Specifically, drywell volume used in LTR Revision 0 for elemental natural deposition coefficient was different than that used in dose calculations. This fact was corrected.
- Information added for determination of particulate removal coefficients AS-2 and AS-3 added.
- Minor changes to AS-1 particulate removal coefficients due to minor changes in recovery of emergency core cooling.

Section 4.4 – Information added to address several NRC RAIs. Condenser plateout coefficient reduced from 99.5% to 99.3% (consistent with ABWR value). Additional information provided to document ESBWR specific MELCOR models demonstrating the assumed value is conservative.

Section 4.5 – Containment leakage rate reduced from 0.5 weight % per day to 0.4 weight % per day. Also, the release model for leakage through the PCCS heat exchangers was revised. The LTR Revision 1 model now assumes all releases through PCCS are released directly to the environment with no credit for holdup or decay.

Section 4.6 – New section added to discuss modeling of SRV flow and suppression pool scrubbing for AS-2 and AS-3.

Chapter 5.0

Significant revision to Chapter 5.

- Revised control room model discussed. Specifically, the ESBWR design change that added the Control Room safety-related Emergency Filter Units (EFU).

- Revised model for AS-1 doses discussed. Dose consequences updated for AS-1.

- Model and results added for dose consequences for AS-2 and AS-3.

Chapter 6.0

No changes.

Appendix A

Table updated to reflect current assumptions, Technical Specifications, and DCD Tier 1,

Appendix B

No changes.