Enclosure 2

MFN 08-771

Transmittal of Non-Proprietary Version of the Initial Licensing Topical Report (LTR) NEDO-33279P, "*ESBWR Containment Fission Product Removal Evaluation Model, October 2006*"

Non-Proprietary Version

GE Energy
Nuclear
NEDO-33279
DRF 0000-0054-9445
Class I
October 2006

Licensing Topical Report

ESBWR Containment Fission Product Removal Evaluation Model

Copyright 2006 General Electric Company

C. Pratt
E. Kirstein
W. Marquino

Approved:

D. Hinds, ESBWR Engineering Manager

IMPORTANT NOTICE REGARDING CONTENTS OF THIS REPORT
PLEASE READ CAREFULLY

The information contained in this document is furnished for the purpose of obtaining NRC approval of the ESBWR DCD and associated analyses. The only undertakings of General Electric Company with respect to information in this document are contained in contracts between General Electric Company and participating utilities, and nothing contained in this document shall be construed as changing those contracts. The use of this information by anyone other than that for which it is intended is not authorized; and with respect to **any unauthorized use**, General Electric Company makes no representation or warranty, and assumes no liability as to the completeness, accuracy, or usefulness of the information contained in this document.

NON-PROPRIETARY INFORMATION NOTICE

This document is the non-proprietary version of NEDE-33279P, and thus, has the proprietary information removed. Portions of this document that have been removed are indicated by open and closed double brackets, as shown here [[ ]].

ACRONYMS AND ABBREVIATIONS

ABWR	Advanced Boiling Water Reactor
AIDA	Aerosol Impaction and Deposition Analysis
AOO	Anticipated Operational Occurrence
AST	Alternative Source Term
ASME	American Society of Mechanical Engineers
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
EAB	Exclusion Area Boundary
EBAS	Emergency Breathing Air System
ECCS	Emergency Core Cooling System
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
FW	Feedwater
GDCS	Gravity Driven Cooling System
GE	General Electric Company
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
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)

CONTENTS

		Page
1.0	**Introduction**	**1-1**
1.1	Background	1-1
1.2	Summary	1-1
1.3	Accident Scenarios Evaluated	1-2
2.0	**Licensing Requirements**	**2-1**
2.1	10 CFR 50, Appendix A, General Design Criterion 19	2-1
2.2	10 CFR 50.34	2-1
2.3	Standard Review Plan Guidelines (NUREG-0800)	2-1
2.4	Regulatory Guide 1.183	2-2
2.5	Regulatory Treatment of Non-Safety Systems	2-3
3.0	**Analytical Techniques and Computer Codes**	**3-1**
3.1	MELCOR	3-1
3.2	RADTRAD	3-1
3.3	ChemSheet	3-2
4.0	**Source Terms and Removal Mechanisms**	**4-1**
4.1	Source Term Assumptions	4-1
	4.1.1 Iodine Chemical For Distribution	4-1
	4.1.2 Pool pH Evaluation	4-1
	4.1.3 Release Timing	4-10
4.2	PCCS as a Fission Product Removal Mechanism	4-11
	4.2.1 Initial PCCS Testing for the SBWR	4-11
	4.2.2 MELCOR Modeling: Heat and Mass Transfer	4-11
	4.2.3 MELCOR Modeling: Particle deposition	4-13
4.3	Containment Plateout	4-16
	4.3.1 Elemental Iodine Plateout	4-16
	4.3.2 Aerosol Iodine	4-17
4.4	Main Steam Isolation Valve Leakage	4-21
4.5	Containment and Reactor Building Leakage Paths	4-24
5.0	**Offsite Dose Calculations**	**5-1**
6.0	**Conclusions**	**6-1**
7.0	**References**	**7-1**

Appendix A - LOCA Dose Assumptions

Appendix B – Reactor Building Mixing Assumptions

List of Tables

Table 1 - Generation of Hydrochloric Acid
Table 2 - Generation of Nitric Acid
Table 3 - RADTRAD Removal Coefficients for Accident Scenario 1
Table 4 - Drywell and RPV Upper Plenum Concentration Comparison
Table 5 - Off-Site χ/Q Values
Table 6 - Control Room X/Q Values (sec./m^3)
Table 7 - Dose Consequences for Accident Scenario 1 (REM TEDE)

List of Figures

Figure 1. Typical Titration Curve
Figure 2. Pool pH, Accident Scenario 1, ESBWR RADTRAD Doses, Start pH=5.3
Figure 3. MELCOR Core Release for AS-1
Figure 4. Airborne CsI Mass for AS-1
Figure 5. Containment Removal Coefficients for AS-1
Figure 6. MSIV Adjustment Factor – DCD Nominal Case
Figure 7. MSIV Adjustment Factor – DCD Bounding Case
Figure 8. Control Room Envelope Model

ABSTRACT

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 Design Basis Accidents (DBA). The passive systems are different from those used in current generation BWRs, thus many of the regulations and methodologies used in previous analyses are not directly applicable to the ESBWR design. Additional research and evaluation was performed to develop a basis for revised methodologies to be used in evaluating the ESBWR.

This report summarizes the methodology used by GE to evaluate the potential dose consequences due to a design basis Loss of Coolant Accident (LOCA). The methodologies documented in this report differ in some areas from those presented in Revision 1 of the ESBWR Design Control Document. In many areas this information is based on analyses performed after the DCD analysis was prepared. Other changes were made to address areas of potential regulatory concerns. This Licensing Topical Report (LTR) analysis demonstrates that the ESBWR systems, in conjunction with natural processes, are adequate to ensure that the dose consequences due to 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.

1.0 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. This research often led to a release of fisson products that was significantly less than that assumed in older off-site and control room dose consequence calculations based on TID-14844. 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 many of the regulations 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 General Electrical 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 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 GE to evaluate the potential dose consequences due to a design basis Loss of Coolant Accident (LOCA). The methodologies documented in this report differ in some areas from that presented in Revision 1 of the ESBWR Design Control Document. This report is intended to provide the technical basis of the LOCA dose calculation for

the ESBWR. In many areas this information is based on analyses performed after the DCD analysis was prepared. Other changes were made to address areas of potential regulatory concerns.

An analysis was performed to determine the dose consequences based on the methodologies documented in this report. This analysis utilized the technical information obtained since the initial DCD analysis was prepared. 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 credited 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:** Bottom drain line break, with ADS and degraded low pressure makeup, restoration of adequate core cooling after 2 hrs.
- **Accident Scenario 2:** Bottom drain line break, with degraded high pressure makeup and ADS failure, restoration of adequate core cooling after 2 hrs.

- **Accident Scenario 3:** Loss of Preferred Power/Loss of FW with ADS, and degraded low pressure makeup, restoration of adequate core cooling after 2 hrs.

The MELCOR results for Accident Scenarios 2 and 3 were not available at the time Revision 0 of this report was prepared, therefore the results were not included. The results of those scenarios will be transmitted to the NRC once they are completed.

2.0 LICENSING REQUIREMENTS

2.1 10 CFR 50, Appendix A, General Design Criterion 19

This regulation requires that a control room be provided from which actions can be taken to operate the nuclear power unit safely under normal conditions and to maintain it in a safe condition under accident conditions, including loss-of-coolant accidents. Adequate radiation protection is required to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 0.05 Sv (5 REM) Total Effective Dose Equivalent (TEDE) for the duration of the accident.

2.2 10 CFR 50.34

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

(i) An individual located at any point on the boundary of the exclusion area for any 2-hour period following the onset of the postulated fission product release, would not receive a radiation dose in excess of 0.25 Sv (25 rem) total effective dose equivalent (TEDE).

(ii) 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]. Draft Revision 3 was issued in April 1996, however Revision 3 was never issued as final. 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]. Draft Revision 3 was issued in April 1996, however Revision 3 was never issued as final. 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 AST analyses, 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 15.6.5, 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]. Draft Revision 3 was issued in April 1996, however Revision 3 was never issued as final. 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. However in some cases detailed information is not available. Sensitivity studies are performed to determine the relative importance of several parameters as documented in Section 5 of this 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 conservative for use 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 containent 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 (CO_2) 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 CO_2. NUREG/CR-5950, Section 2.2.3 states that pure water will attain a pH approaching 5.65 due to absorption of CO_2 from air and the subsequent formation of Carbonic Acid. As such, the effects of carbon dioxide are considered and bounded by evaluations assuming the minimum pool pH allowed by specifications. No detailed calculations explicitly accounting for CO_2 were performed.

Cesium Hydroxide:

Cesium Hydroxide (CsOH) is a strong base introduced into the primary containment and subsequently to the containment pools with the release of cesium post accident. The production of this base is considered within this assessment. The 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 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 572°F, pyrolysis is not considered within this assessment.

The production of HCl by irradiating cables is estimated to be 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. [[

]]

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

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

Semi-infinite Cloud

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

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:

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

Accounting for the GF the containment dose rates then become

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

Dose rates and doses were then determined, which in turn was used to determine the HCl released as a result of radiolysis. The results of which are presented in Table 1. [[

]]

Table 1 – Generation of Hydrochloric Acid

[[

]]

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/lb/ Mrad.

The dose rates presented previously in Table 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. The HCl production was calculated using the same formula as in previous cases by using the amount of water from MELCOR. The calculated radiation doses and production rates are presented in Table 2.

Table 2 - Generation of Nitric Acid

[[

]]

[[

]]

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 for the Standby Liquid Control (SLC)system. SLC system would be used as an injection source following a line break with a loss of the gravity driven coolant system (GDCS), because it a high pressure water source. Buffering by SLC system is considered in this evaluation.

[[ ]] 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, $\gamma_\pm$:

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

In Pitzer formalism the mean activity coefficient is expressed by:

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

where

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

μ_{MXN} ternary interaction parameter

Pitzer's equation for the aqueous phase is a virial coefficient expansion of Debye–Hückel's theory and is capable of describing the ionic activities of aqueous species in concentrated solutions usually up to 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 1.

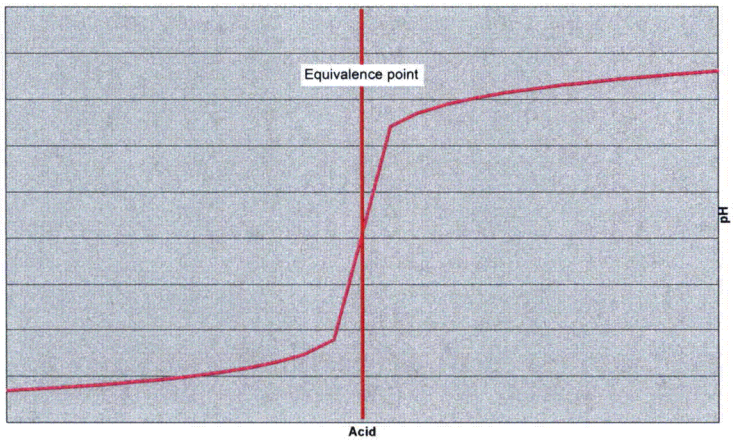


Figure 1. Typical Titration Curve

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.

[[

]]

[[

]]

4.1.3 Release Timing

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. Because this report was developed to closely correspond to the ESBWR MELCOR containment model, the EIV release initiation will be based on the results of that analysis. Because the majority of fission products are released in the EIV phase, release timing was chosen based on the start of the EIV release phase. The release fractions themselves will continue to be based on Regulatory Guide 1.183, Table 2.

[[

]]

[[

]]

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. [[

]]

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],

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

Here L is the latent heat [J/kg], which is calculated at T_l. It was assumed that the water film temperature profile is linear. T_l is the average liquid temperature [K] and T the average gas

temperature [K]. P is the perimeter of the heat exchanger tube [m], c_{pn} the nitrogen gas heat capacity [J/kg K], c_{pw} the water vapor heat capacity [J/kg K] and c_{pl} the liquid water heat capacity [J/kg K], respectively. The mass fluxes for the water vapor, nitrogen gas and liquid water are $\dot{m}_w$, $\dot{m}_n$ and $\dot{m}_l$ [kg/s]. The heat fluxes q^T and q^{Ts} [W/m^2] are calculated from:

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

and

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

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

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

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

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

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

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

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

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

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

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

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

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

$$\dot{m}_w = \rho_w U A,$$

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:

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

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

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

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

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

4.3 Containment Plateout

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

4.3.1 Elemental Iodine Plateout

The elemental iodine coefficient is based on guidance found in SRP 6.5.2 [Ref. 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,

$\lambda_n =$ 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:

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

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,

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

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

The removal rate constant will be taken as 0.137 cm/sec (16.18 ft/hr) based on NUREG/CR-0009, Page 17 [Ref. 32].

$$\lambda_n = 16.18(ft/hr)\left(\frac{1.34E4\,ft^2}{2.36E5\,ft^3}\right) = 0.92hr^{-1}.$$

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, suppression pool scrubbing, 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.

[[

]]

The mass values were obtained from the MELCOR results. Two coefficients were calculated: (1) and "instantaneous" value was determined using the reduction for each time step, and (2) an average value was calculated using the reduction over the previous 30 minutes. The coefficients chosen for input into RADTRAD for Accident Scenario 1 are presented in Table 3 (and are also included on Figure 5).

[[

]]

Table 3 – RADTRAD Removal Coefficients for Accident Scenario 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 GE and the BWR Owner's Group (BWROG). This methodology is documented in NEDO-31858, "BWROG Report for Increasing MSIV Leakage Rate Limits and Elimination of Leakage Control Systems" [Ref. 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. The dose consequences are not overly significant to assumptions related to plateout in the main steam lines and main steam lines' drain lines, therefore no credit is taken for deposition or plateout in the lines themselves.

Because no credit is taken for plateout in the main steam lines or main steam lines' drain lines, leakage though the MSIVs is assumed to be released directly into the condenser. The condenser is of robust design, and it is designed to the Safe Shutdown Earthquake (SSE) requirements of the BWROG methodology. As such, it will serve as a holdup volume for decay for leakage past the MSIVs following a LOCA. Credit for deposition in the main condenser is credited based on the BWROG methodology.

The DCD, Revision 1 LOCA dose calculation assumed a total MSIV leakage rate of 46 scfh at containment pressure and temperature. This value is somewhat lower than those found for most US BWRs. The dose analysis prepared in support of this report assumed a revised MSIV maximum leakage rate of 100 scfh per valve, with the total from all four MSIVs not to exceed 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 1, 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 1, 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 6 and 7.

Figure 6 - MSIV Adjustment Factor - DCD Nominal Case

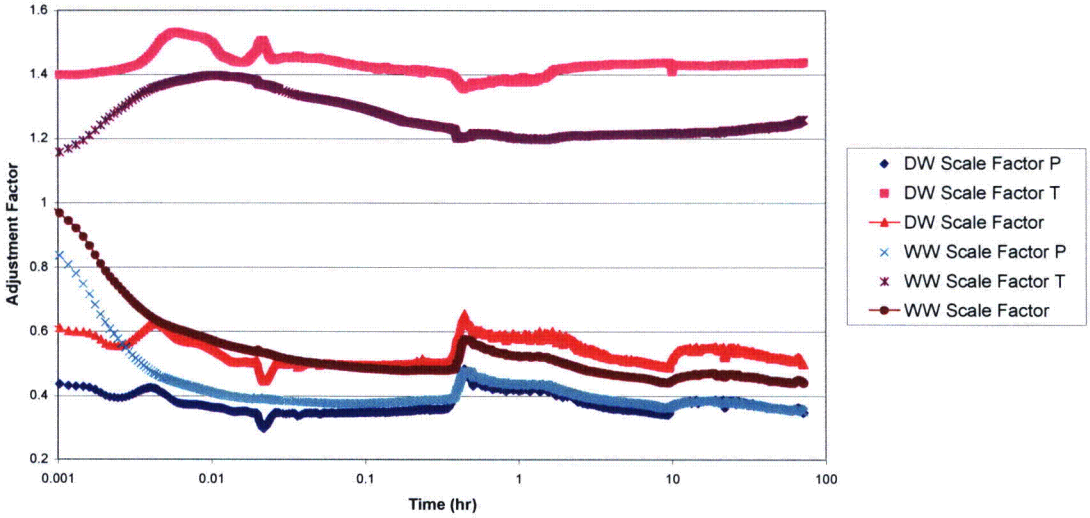
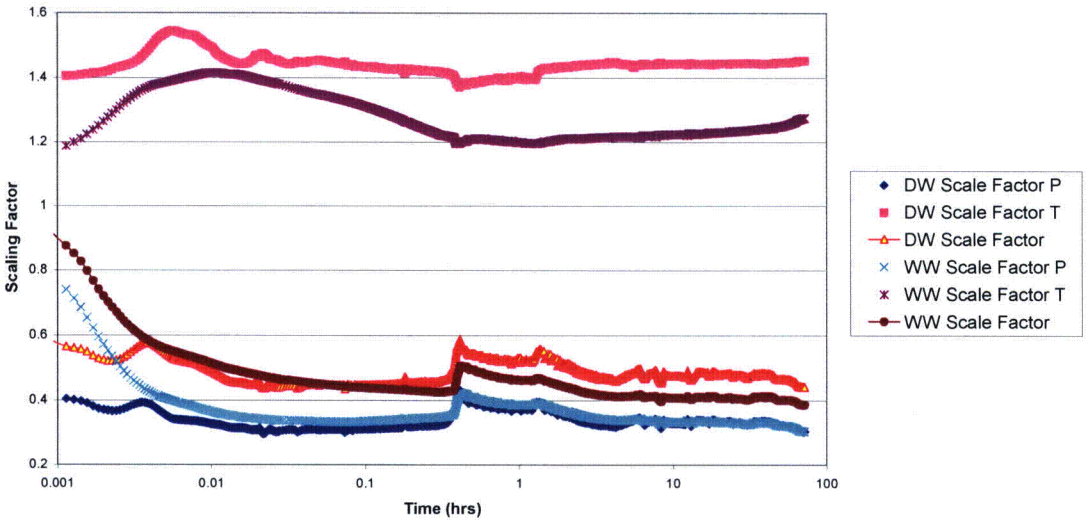


Figure 7 - MSIV Adjustment Factors - DCD Bounding Analysis



[[

]]. The bounding MSIV adjustment factor was calculated to be 0.66 at 0.44 hours as shown in Figure 6. 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:

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

[[

]]

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 in Section 4.4 above);
- 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 is 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 1 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. The TRACG computer code was used to evaluate many of the design basis accident scenarios evaluated in DCD Chapters 6 and 15. The TRACG output can include the boiloff rate for the PCCS and IC pools. The TRACG rates can then be converted to a volumetric flow rate by accounting for the density of the steam. High quality steam at saturated conditions for atmospheric pressure is assumed.

$$VFR_{\max} = \dot{M}\left({}^{kg}/_{s}\right) \times v\left({}^{m^3}/_{kg}\right)$$

[[

]] It is assumed that 2% of the overall containment leakage is released through this pathway. Plant Technical Specifications will be updated to reflect this limit.

Note the calculated flow rate is extremely conservative in that it assumes a steam quality of 1.0. In reality there would be a significant fraction of water that would condense in the Reactor Building. Also the discharge from this area has a moisture separator to help retain PCC and IC pool inventory. The condensation and removal of water would likely lower the building flow rate significantly. Finally, the flow rate of 17.5 m^3/s is based on the maximum boil-off rate from the PCCS pools. The actual flow rate would be significantly lower for most of the event.

5.0 OFFSITE DOSE CALCULATIONS

As discussed previously, while this analysis was primarily based on the DCD analysis, several modifications were made to address more recent technical information or potential regulatory concerns. Each change is discussed briefly. A detailed comparison of the base analysis to the DCD assumptions is presented in Appendix A.

The DCD, Revision 1 LOCA dose analysis assumed that the control room was operated with the Control Room Emergency Breathing Air System (EBAS) in opration for the duration of the event. EBAS is a Safety-related system. Design changes are ongoing to ensure that EBAS will have the ability to supply the control room with 7 days of air. After 7 days it is inherently assumed that replenishment air is obtained from off-site resources. EBAS is designed for 3 banks of bottled air, each of which is capable of supplying 2.83 m^3/min. (100 CFM) of clean air following an accident. Further, EBAS shall be designed to ensure that no single active failure (SAF) will impact air flow to the control room for the first 72 hours. The final system design will also ensure that no SAF will result in the loss of inventory. As such the volume of air in the "failed" tank will be available, however, addition manual operator actions may be required (after 72 hours). Plant Technical Specifications shall be revised to reflect the 7 day capacity of air for EBAS.

This study will assume 0.113 m^3/min. (4.0 cfm) unfiltered inleakage while the control room is operated under EBAS. No credit is taken for non-safety-related charcoal filter trains. The discharge flow from the control room is adjusted proportionally to account for the additional inleakage. Because the analysis will assume unfiltered inleakage the purging effect of this additional 4 cfm will also be taken into account. [[

]]

[[

]]

[[

]]

The Reactor Building volume and leakage rate assumptions are also adjusted for this analysis. As discussed in Section 4.5, a maximum of 50% of a building volume may be credited for holdup and decay of radioactive materials for a secondary containment. While the DCD Revision 1 analysis credited 100% of the building volume, this base analysis credits only 40% based on the ESBWR configuration (Appendix B). Also, the DCD Revision 1 analysis utilized a simplifying assumption that the Reactor Building release rate was 100 volume % per day, whereas this analysis will assume a value of 50% per day. Also, 98% of the containment leakage term is released into the Reactor Building (0.49 containment volume % per day). The remaining 2% is released quickly to the environment via the PCCS/IC pool airspace (0.01 containment volume % per day). Plant Technical Specifications will be revised to reflect this assumed distribution.

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 consequence analysis for this report was developed to closely correspond to the MELCOR containment model, and as such the release timing. The releases times were set such that the beginning of the EIV phase began when significant amounts of fission products were being released from the fuel. The gap release is assumed to occur for the 30 minutes prior to the EIV release. As discussed previously in Section 4.1.3, the gap release is assumed to occur from 20 minutes to 50 minutes for AS-1, and the EIV phase is assumed to occur

from 50 minutes to 2 hours 20 minutes. 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 3 for Accident Scenario 1. It should be noted that in the DCD analysis credit was also taken for deposition of elemental iodine in the PCCS condensers. 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.

Finally, this analysis revised the atmospheric dispersion factors (χ/Q) used to calculate both off-site and control room doses. The off-site values were "back calculated" to determine the bounding values which would result in doses just under the Regulatory Limit of 25 REM TEDE. The off-site dispersion factors are presented in Table 5.

Table 5 – Off-Site χ/Q Values

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

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

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. The χ/Q values assumed for control room doses are presented in Table 6. The values were generic yet bounding in nature. They are chosen based on the ESBWR design and the worst alignment for the assumed plant layout.

- **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 release 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 release is assumed to be a point release.

- **MSIV Leakage:** MSIV leakage is assumed to be released from the main condenser. This release is assumed to be a diffuse release.

Table 6 – Control Room X/Q Values (sec./m^3)

Time	Reactor Building Leakage	PCCS Leakage (RB Roof)	MSIV Leakage (Condenser)
0 – 2 hrs	2.10E-03	4.00E-03	1.80E-03
2 – 8 hrs	1.70E-03	3.30E-03	1.60E-03
8 – 24 hrs	5.80E-04	1.40E-03	6.20E-04
1 – 4 days	5.70E-04	1.20E-03	5.70E-04
4 – 30 days	5.20E-04	1.00E-03	4.20E-04

The dose consequences for AS-1 are presented in Table 7. The dose from each dose contributor, as well as the total dose is presented.

Table 7 – Dose Consequences for Accident Scenario 1 (REM TEDE)

[[

]]

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.

7.0 REFERENCES

[1] 10 CFR 50.34, "Contents of Applications; Technical Information"

[2] Not Used.

[3] Regulatory Guide 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," Revision 0.

[4] TID-14844, Calculation of Distance Factors for Power and Test Reactor Sites.

[5] SRP Section 6.2.1, "Containment Functional Design," Revision 2.

[6] SRP Section 6.2.3, "Secondary Containment Functional Design," Revision 2.

[7] SRP Section 6.5.5, "Pressure Suppression Pool as a Fission Product Cleanup System," Revision 0.

[8] SRP Section 15.0.1, "Radiological Consequence Analyses Using Alternative Source Terms," Revision 0.

[9] SRP Section 15.6.5, "Loss-of-Coolant Accidents Resulting from Spectrum of Postulated Piping Breaks within the Reactor Coolant Pressure Boundary," Revision 2.

[10] NRC memorandum to the docket file dated July 24, 1995, "Consolidation of SECY-94-084 and SECY-95-132, addressing the Regulatory Treatment of Non-Safety Systems (RTNSS).

[11] Not Used.

[12] NUREG-1465, "Accident Source Terms for Light Water Nuclear Power Plants."

[13] Not Used.

[14] NUREG/CR-5732, "Iodine Chemical Forms in LWR Severe Accidents."

[15] NUREG/CR-6119, "MELCOR Computer Code Manual."

[16] NUREG/CR-6189, "A Simplified Model of Aerosol Removal by Natural Processes in Reactor Containments."

[17] NUREG/CR-6604, "RADTRAD: a Simplified Model for Radionuclide Transport and Removal and Dose Estimation," April 1998 (Including Supplements 1 and 2).

[18] Werkstoffwoche '96, Symposium & Simulation, Modellierung, Informationssysteme, (1997), 47 (ISBN 3-88355-236-4, published by DGM Informationsgesellschaft mbH, Hamburger Allee 26, D-60486 Frankfurt, Germany).

[19] 22A6642AH, *ESBWR Design Control Document*, Revision 1, General Electric Nuclear Energy.

[20] NEDO-31858, "BWROG Report for Increasing MSIV Leakage Rate Limits and Elimination of Leakage Control Systems," Revision 1.

[21] Not Used.

[22] Not Used.

[23] Not Used.

[24] Not Used.

[25] ENE53/46/2000, "Investigation on Aerosol Deposition in a Heat Exchanger Tube," Jouni Hokkinen et al., August 1, 2001.

[26] SRP Section 6.5.2, "Containment Spray as a Fission Product Cleanup System," Revision 3, December 2005.

[27] Waldmann, L., Schmitt, K.H., Thermophoresis and diffusiophoresis of particles in a heated boundary layer, In: Davies, C.N. (Ed.), Aerosol Science, Academic Press, London, 1966.

[28] Talbot, L., Cheng, R.K., Schefer, R.W., Willis, D.R., Thermophoresis of particles in a heated boundary layer, J. Fluid Mech. 101, (1980) 737-758.

[29] Hinds, W.C., Aerosol Technology, second ed., Wiley-Interscience, 1999.

[30] Papavergos, P.G., Hedley, A.B., Particle deposition behaviour from turbulent flows, Chem. Eng. Res. Des. 62 (1984) 275-295.

[31] VTT-R-04413-06, "Estimation and Modeling of Effective Fission Product Decontamination Factor for ESBWR Containment," A. Auvinen et al.

[32] NUREG/CR-0009, "Technological Bases for Models of Spray Washout of Airborne Contaminants in Containment Vessels," USNRC, October 1978.

[33] Pitzer, K.S., J. Phys. Chem. 1973, Volume 77, pgs. 268-277.

[34] Pitzer, K.S., J. Phys. Chem. 1973, Volume 77, pgs. 2300-2308.

[35] NUREG/CR-5950, *Iodine Evolution and pH Control*, December 1992.

[36] Regulatory Guide 1.3, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident for Boiling Water Reactors," Revision 2 (ML003739601).

[37] NEDO-33201, *Licensing Topical Report – ESBWR Certification, Probabilistic Risk Assessment*, Revision 1.

Appendix A - LOCA Dose Assumptions

Parameter	DCD, Rev. 1 Analysis [22A6642 BP, Rev.1]	LTR Base Deck Value	Recommended Verification Method [Note 6]
I. Data and Assumptions Used to Estimate Source Terms			
A. Power Level, MWt	4590	4590	TS 1.1 (102% of RTP)
B. Fraction of Core Inventory Released	RG 1.183, Table 1	RG 1.183, Table 1	NR
C. Iodine Chemical Species			
Elemental, %	4.85	4.85	NR
Particulate, %	95	95	NR
Organic, %	0.15	0.15	NR
D. PCCS			
Decontamination Factors			
Noble Gas	1	1	NR
Elemental	10	[Note 1]	
Particulate	10		
Organic	1		
Flow Rate to/from Containment			
0 – 2 hrs, m^3/min	4.82E+02	[Note 1]	NR
2 – 4 hrs, m^3/min	4.34E+02		
4 – 8 hrs, m^3/min	3.86E+02		
8 – 12 hrs, m^3/min	3.62E+02		
12 – 16 hrs, m^3/min	3.14E+02		
16 – 24 hrs, m^3/min	2.89E+02		
1 – 30 days, m^3/min	5.66E+04		

Parameter	DCD, Rev. 1 Analysis [22A6642 BP, Rev.1]	LTR Base Deck Value	Recommended Verification Method [Note 6]
II. Data and Assumptions Used to Estimate Activity Released			
A. Primary Containment			
Leak Rate			
0 – 72 hrs, %/day	0.50	0.50	TS 5.5.9.b
72 hrs – 30 days, %/day	0.00	0.50	TS 5.5.9.b
Volume, m^3	7.2E+03	7.2E+03	ITAAC
Elemental iodine removal rate constant, hr^{-1}	0.92	0.92	NR
Particulate Deposition Model	Power's (10)	[Note 1]	NR
Primary Containment Leakage Rate Distribution			
Fraction to Reactor Building Atmosphere, %	N/A	98%	NR
Fraction through PCCS, %	N/A	2%	TS [Note 5]
B. Reactor Building			
Leak Rate, %/day	100	50	TS 3.6.3.1
Mixing Efficiency, %	100	40	NR
Total Volume, m^3	1.53E+06	6.05E+04	NR
Mixing Volume, m^3	1.53E+06	2.42E+04	ITAAC [Note 7]
C. Condenser Data			
Free Air Volume, m^3	6.23E+03	6.23E+03	NR
Mixing Fraction, %	20	20	NR
Mixing Volume, m^3	1.25E+03	1.25E+03	ITAAC
Iodine Removal Factors			
Particulate, %	99.5	99.5	COL
Elemental, %	99.5	99.5	COL
Organic, %	0.0	0.0	NR

Parameter	DCD, Rev. 1 Analysis [22A6642 BP, Rev.1]	LTR Base Deck Value	Recommended Verification Method
D. MSIV Data			
MSIV Leakage per Valve, m^3/min (post-LOCA containment conditions)	4.07E-03	3.12E-02	TS 3.6.1.3
Total MSIV Leakage, m^3/min (post-LOCA containment conditions)	1.63E-02	6.23E-02	TS 3.6.1.3
Main Steam Line Length, m	16.8	[Note 2]	NR
Main Steam Line Volume, m^3	5.32	[Note 2]	NR
Drain Line Length, m	50.6	[Note 2]	NR
Drain Line Volume, m^3	3.77E-01	[Note 2]	NR
Main Steam Line OD/Thickness, mm	711/37	[Note 2]	NR
Drain Line OD/Thickness, mm	114/9	[Note 2]	NR
Main Steam Line Insulation Thickness, mm	114	[Note 2]	NR
Drain Line Insulation Thickness, mm	89	[Note 2]	NR
Plateout Factors	Ref. 20	[Note 2]	NR
III. Control Room Parameters			
A. Control Room Volume, m^3	2.62E+03	2.46E+03	ITAAC
B. EBAS Flow			
0 – 30 days, m^3/min	0 [Note 3]	2.83	TS 3.7.1
C. Unfiltered Inleakage			
EBAS, m^3/min	0	0.113	COL
Filtered Intake/Recirculation, m^3/min	0	NR	NR
D. Ventilation Parameters (>72 hrs)			
Filtered Inleakage from Atmosphere, m^3/min	14.2	N/A	NR
Recirculation Rate, m^3/min	7.1	N/A	NR
Intake/Recirculation Filter Efficiency, %	99	N/A	NR
IV. Pool Parameters and pH Calculations			
A. Pool Volumes			
GDCS Pool Volume (m^3)	N/A	1.86E+03	ITAAC
Suppression Pool Volume (m^3)	N/A	3.80E+03	ITAAC
B. Exposed (Chlorine Bearing) Cable Insulation Mass (kg)	N/A	3400	ITAAC

Notes:

1. Based on the MELCOR analysis [Ref. 31] the PCCS DFs and flow rates are no longer explicitly modeled. See Section 4.3 of this report for additional discussion.
2. The values for the Main Steam Lines and Main Steam Lines' Drain Lines data were used to calculate the amount of deposition in the piping. Because deposition in these lines is no longer credited, the actual values for these parameters are not needed to perform this analysis.
3. The DCD analysis assumed operation of EBAS, however no flow was explicitly assumed in that analysis, hence the 0 m^3/min value listed.
4. Filtered recirculation and intake is assumed after 72 hours to ensure conservative doses are calculated, hence the flow listed is 0 cfm. However, EBAS has the capability to be replenished via a safety-related connection if necessary following an event.
5. The maximum permissible leakage rate though the PCCS will be added to plant Technical Specifications, however the exact location and format has not yet been determined.
6. The "recommended" verification methods are defined as follows:

   ITAAC - (Inspections Tests Analyses and Acceptance Criteria) - This item will be verified by GE once design is completed (as appropriate)

   COL - (Combined Operating License) - This item will be verified by the utility/customer via the COL process

   TS - (Technical Specifications) - This item has a TS Surveillance Requirement, therefore no additional actions/verifications are required (Note that the actual TS section is identified)

   NR - (Not Required) - The basis for this assumptions is well defined via regulations or by analysis, therefore no additional verification/confirmation is required.
7. 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, therefore the steam supply lines penetrate containment.
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 08-771

Affidavit – David H. Hinds

GE-Hitachi Nuclear Energy Americas LLC

AFFIDAVIT

I, David H. Hinds, state as follows:

(1) I am General 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 to be discussed and sought to be withheld is delineated in the letter from Mr. Richard E. Kingston to U.S. Nuclear Regulatory Commission, entitled "*Transmittal of Non-Proprietary Versions of the Initial Licensing Topical Report (LTR) NEDE-33279P, ESBWR Containment Fission Product Removal Evaluation Model, October 2006*," dated October 8, 2008. The information to be withheld is "MFN 08-771 *Transmittal of Non-Proprietary Versions of the Initial Licensing Topical Report (LTR) NEDE-33279P, ESBWR Containment Fission Product Removal Evaluation Model, October 2006*," – *GEH Proprietary Information*" contains proprietary information, and is identified by [[dotted underline inside double square brackets{3}]]. Figures and other large objects are identified with double square brackets before and after the object. In each case, the superscript notation {3} refers to Paragraph (3) of this affidavit, which provides the basis for the proprietary determination.

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

Executed on this 8th day of October 2008.

David H. Hinds
GE-Hitachi Nuclear Energy Americas LLC