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<u>*Proprietary Notice*</u>
This letter forwards proprietary information in accordance with 10CFR2.390. Upon the removal of Enclosure 1, the balance of this letter may be considered non-proprietary.

MFN 06-205

Docket No. 52-010

July 12, 2006

U.S. Nuclear Regulatory Commission
Document Control Desk
Washington, D.C. 20555-0001

Subject: **Submittal of Licensing Topical Report, NEDE-33279P, "ESBWR Containment Fission Product Removal Evaluation Model (Unverified Draft)," June 2006**

The subject Licensing Topical Report (LTR) is hereby submitted for NRC review (Enclosure 1). This report summarizes the methodology used by GE to evaluate the potential dose consequences due to a design basis loss of coolant accident.

GE's commitment to provide this draft evaluation model and draft results was transmitted in the Reference 1 letter.

Enclosure 1 contains GE proprietary information as defined by 10 CFR 2.390. GE customarily maintains this information in confidence and withholds it from public disclosure.

The affidavit contained in Enclosure 3 (and also incorporated in the LTR) identifies that the information contained in Enclosure 1 has been handled and classified as proprietary to GE. GE hereby requests that the information of Enclosure 1 be withheld from public disclosure in accordance with the provisions of 10 CFR 2.390 and 9.17. A non proprietary version of the LTR is provided in Enclosure 2 (NEDO-33279).

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If you have any questions about the information provided here, please let me know.

Sincerely,

Kathy Sedney for

David H. Hinds
Manager, ESBWR

Reference:

1. MFN 05-111, Letter from David H. Hinds to U. S. Nuclear Regulatory Commission, *GE Response to Results of NRC Acceptance Review for ESBWR Design Certification Application – Item 4 (TAC # MC8168)*, October 22, 2005

Enclosures:

1. MFN 06-205 – NEDE-33279P, "ESBWR Containment Fission Product Removal Evaluation Model (Unverified Draft)," June 2006 – GE Proprietary Information
2. MFN 06-205 – NEDO-33279, "ESBWR Containment Fission Product Removal Evaluation Model (Unverified Draft)" June 2006 – Non Proprietary Version
3. Affidavit – Louis M. Quintana – dated June 27, 2006

cc:	WD Beckner	USNRC (w/o enclosures)
	AE Cubbage	USNRC (with enclosures)
	LA Dudes	USNRC (w/o enclosures)
	GB Stramback	GE/San Jose (with enclosures)
	eDRF	0000-0055-6762

ENCLOSURE 2

MFN 06-205

Non Proprietary Version

NEDO-33279, "ESBWR Containment

Fission Product Removal Evaluation Model

(Unverified Draft)," June 2006

GE Energy
Nuclear

NEDO-33279
DRF 0000-0054-9445
Class I
June 2006

Licensing Topical Report

ESBWR Containment Fission Product Removal Evaluation Model (UNVERIFIED DRAFT)

C. Pratt
E. Kirstein
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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 is a non-proprietary version of the document NEDE-33279P which has the proprietary information removed. Portions of the 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
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
SBWR	Simplified Boiling Water Reactor
SER	Safety Evaluation Report
SLCS	Standby Liquid Control System
SRP	Standard Review Plan
TEOM	Tapered Element Oscillating Microbalance
TRACG	GE version of the Transient Reactor Analysis Code
VFR	Volumetric Flow Rate
X/Q	Atmospheric Dispersion Factor (Chi over Q)

CONTENTS

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 is 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). This report is not intended to directly replace the analysis presented in the ESBWR Design Control Document; rather it is intended to provide information for 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. A base analysis was performed to determine the dose consequences based on the best available information. The base analysis demonstrates that the ESBWR systems, in conjunction with natural processes, are adequate to ensure that the dose consequences to a design basis LOCA would meet the criteria set forth in 10 CFR 50.67.

Sensitivity studies were performed to determine the relative importance of several parameters with respect to off-site and control room doses. These studies included evaluating control room model assumptions, reactor building leakage assumptions, containment leakage assumptions, and MSIV leakage assumptions. The studies provide useful information should the DCD LOCA dose analysis ultimately be revised.

1.0 INTRODUCTION

1.1 Background

Early plant Design Basis Accident (DBA) dose consequence evaluations were performed using source terms derived on TID-14844, *Calculation of Distance Factors for Power and Test Reactor Sites* [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 lead to core damage predictions, which were significantly less than those 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* [12]. The NRC issued Regulatory Guide 1.183, *Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors* [3], in July 2000. The NRC also issued 10 CFR 50.67, which allowed licensees to utilize Alternative Source Terms (AST) for DBA dose consequence analyses. AST and 10 CFR 50.67 is also applicable to advanced reactor designs, such as the ESBWR.

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). This report is not intended to directly replace the analysis presented in Chapter 15 of the ESBWR Design Control Document [19]; rather it

is intended to provide information for 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. Sensitivity studies were performed to determine the relative importance of several parameters with respect to off-site and control room doses. The information contained in this report may ultimately be used should the LOCA doses be recalculated for a revision to the DCD, however that effort will be performed independent of this report and the supporting analyses.

A base analysis was performed to determine the dose consequences based on the best available information. This base analysis utilized the technical information obtained since the DCD analysis was prepared. The base 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 50.67.

The base analysis was then used to perform several sensitivity studies. These studies included evaluating control room model assumptions, reactor building leakage assumptions, primary containment leakage assumptions, and MSIV leakage assumptions. [[

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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 50.67

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).

(iii) Adequate radiation protection is provided to permit access to and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 0.05 Sv (5 rem) total effective dose equivalent (TEDE) for the duration of the accident.

2.3 Standard Review Plan Guidelines (NUREG-0800)

SRP Section 6.2.1, Revision 2, "Containment Functional Design," was issued in July 1981 [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 [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 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 [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 [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 [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 [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 Section 15.6.5, including the appendices, were affected significantly by AST, and the updated assumptions and methodologies are documented in Regulatory Guide 1.183.

2.4 Regulatory Guide 1.183

Regulatory Guide 1.183, 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 since 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 [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* [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 TRACG

TRACG is a General Electric (GE) proprietary version of the Transient Reactor Analysis Code (TRAC). TRACG has a multi-dimensional, two-fluid model for the reactor thermal hydraulics and a three-dimensional reactor kinetics model. These features allow for detailed, realistic simulation of the phenomena that are important in evaluating the operation of BWRs. Realistic analyses

performed with TRACG have been used previously to support licensing applications in different areas, including transients otherwise known as Anticipated Operational Occurrences (AOO), and pipe breaks referred to by the acronym ECCS/LOCA (Emergency Core Cooling Systems / Loss of Coolant Accident). The features of TRACG are described in detail in the TRACG Model Description Licensing Topical Report [21].

The application of TRACG for ESBWR analyses is described in TRACG Application for ESBWR [23]. The TRACG model for ESBWR ECCS/LOCA analysis is described in Section 2.7.1.1 of Reference 23. The model includes detailed nodalization for the RPV and ECCS components and a less detailed model for the containment.

TRACG has been approved for use in evaluating Loss of Coolant Accidents [22]. The ECCS/LOCA model can adequately simulate the event if the core oxidation is not excessive, i.e. within 10 CFR 50.46 limits.

3.3 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. This research often led to core damage predictions that 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* [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* [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.

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" [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. 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

As discussed previously, the iodine chemical distribution recommended by Regulatory Guide 1.183 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.

A detailed pH analysis is being prepared to determine any additional design requirements that may be required to ensure that plant pH levels are maintained above 7. The standby liquid control system (SLCS) would be used as an injection source following a line break with a loss of the gravity driven coolant system (GDCS) since it is a high pressure water source. SLCS would provide some buffering. The analysis to determine whether additional buffering is necessary is not yet finalized, however, since the ultimate result is to determine any design requirements necessary to ensure that pool pH is maintained, this analysis utilizes the iodine chemical distribution of Regulatory Guide 1.183.

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. Since this report was developed to closely correspond to the MELCOR containment model, the release timing and duration will be based on the results of that analysis. The release fractions themselves will continue to be based on Regulatory Guide 1.183, Table 2.

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4.2 PCCS as a Fission Product Removal Mechanism

4.2.1 Initial PCCS Testing for the SBWR

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

Testing was performed at the AIDA (Aerosol Impaction and Deposition Analysis) facility of the Paul Scherrer Institute (PSI) as documented in Reference 24. The condenser model was a scaled down version of the SBWR PCC. The original tube length and diameter were preserved, however the total number of tubes was only 8. Tanks were used to simulate the GDCS and wetwell systems. Tests were performed with soluble (CsI) and insoluble (SnO_2) aerosols, as well as a mixture of both types. The aerosol concentrations were on the extreme high end of the expected range following a core melt with core-concrete interaction.

The AIDA tests concluded that the impact of the aerosol load on the PCC performance depends upon the composition of the aerosol. [[

]] Thus, the condenser does act as an effective scrubber.

There was a difference in the deposition behavior between SnO_2 and SnO that was not well understood following the AIDA testing. As a result additional testing was performed as documented in ENE53/46/2000, "Investigation on Aerosol Deposition in a Heat Exchanger Tube" [25]. VTT Energy in Finland performed the testing. [[

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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 [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], respectively. 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 [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], respectively.

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 [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 the 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] [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 on 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 [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 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 [29]:

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

where g is the gravitational acceleration [m/s^2] and 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" [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 [m²/s] and D the Brownian diffusivity [m²/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 [26]. Specifically, the iodine removal rate constant for a particular compartment "n" will be based on the following formula:

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

where,

λ_n = removal rate constant due to surface deposition,

k_g = average mass transfer coefficient,

A = surface area for deposition, and

V = Volume of the contained gas.

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

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

Only 50% of the floor area will be credited (to account for the 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 [32].

$$\lambda_n = 16.18(ft/_{hr})\left(\frac{1.34E4ft^2}{2.36E5ft^3}\right) = 0.92hr^{-1}.$$

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 [17] and NUREG/CR-6189 [16]. The Powers model is 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 LOCA dose analyses utilized the Powers model for natural deposition of particulate iodine in the drywell of 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 iodine in the various containment volumes is calculated in MELCOR, and the resultant airborne masses are determined. [[

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Table 1 - MELCOR CsI Airborne Masses in Containment

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Table 2 – RADTRAD Removal Coefficients

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4.4 Main Steam Isolation Valve Leakage

Leakage past Main Steam Isolation Valves (MSIVs) typically bypasses secondary containment 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" [20]. NEDO-31858 is explicitly referenced in Regulatory Guide 1.183 with respect to guidance on acceptable models.

The BWROG methodology utilizes the rigorous design of the main steam lines (MSL), main steam drain lines (MSDL), and the main condenser. The methodology utilizes a time dependent cooling model based on NUREG/CR-1169 [13]. For an insulated pipe:

$$M_1 C \frac{d(T-T_a)}{dt} = -\left[hA_1\left(T-T_a\right)\right],$$

where:

t =	Time (hr),
M_1 =	Pipe mass per unit length, (lbm)
C =	Pipe heat capacity, (BTU/lbm-°F)
T =	Pipe temperature, (°F)

$T_a =$ Ambient temperature, (°F)

$A_l =$ Heat transfer area per unit length of pipe, (ft^2)

$h =$ Heat transfer coefficient for cooling, (BTU/hr-ft^2-°F), or

$$h = \frac{K}{r_2 \ln\left(r_2/r_1\right)},$$

where,

$K =$ Heat capacity for insulation material, (BTU-hr-ft-°F)

$r_1 =$ Pipe radius, (ft)

$r_2 =$ Insulation radius, (ft)

therefore

$$T - T_a = \left[T - T_a\right]_0 \exp\left[{-hA_l t}/{M_l C}\right].$$

The first step in evaluating potential removal processes is to define the rate of transport to the pipe surface (steel). This is given by

$$D_r = K_g\left[C_b - C_s\right],$$

where

$D_r =$ Deposition rate, (gm/cm^2-sec)

$K_g =$ Mass transfer rate, (cm/sec),

$C_b =$ Iodine concentration in the bulk gas, (gm/cc) and

$C_s =$ Iodine concentration near the surface (gm/cc).

The deposition rate is often given in terms of a deposition velocity, V_d (cm/sec), and is defined as follows:

$$D_r = V_d C_b.$$

The rate of change of airborne activity in the MSL (or the MSLDL) can be expressed as:

$$dC_a/dt = -JC_a.$$

The removal rate can be expressed in terms of the deposition velocity and the surface to volume ratio as follows:

$$J = V_d\left(\frac{A}{V}\right),$$

where,

J =	Removal rate, (1/sec)
C_a =	Iodine airborne concentration, (gm/cc)
A =	Surface Area, (cm^2) and
V =	Volume (cc).

Therefore,

$$dC_a/C_a = -\left[V_d\left(\frac{A}{V}\right)\right]dt$$

After integration

$$C_a = C_{a0}\exp\left[-V_d\left(\frac{A}{V}\right)t\right].$$

A significant factor in evaluating the removal process in the steamline is the residence time, which is a function of the flow velocity through the line. Early estimates of residence time considered plug flow as the limiting mechanism, however later work suggests that the plug flow model is non-conservative [20]. Therefore, the 1988 BWROG study was based on a conservative residence, which is assumed to be 1/8 of that predicted by plug flow.

Deposition velocities are based on "MSIV Leakage Iodine Transport Analysis" prepared by J.E. Cline and Associates for the NRC [11]. That document is also referenced in Appendix A of Regulatory Guide 1.183.

$$\text{Elemental:}\quad V_{de} = \exp\left[2809/T - 12.80(\pm 0.33)\right]$$

$$\text{Aerosol/Particulate:}\quad V_{de} = \exp\left[2809/T - 15.39\right]$$

$$\text{Organic:}\quad V_{de} = \exp\left[2809/T - 19.30\right]$$

Using the information above the fraction of the iodine that remains airborne can be determined, as well as the amount that plates out. Since a "decontamination factor" (DF) is defined as the activity entering a volume divided by that leaving the volume, an effective DF can also be determined.

$$F_{air} = \frac{C_a}{C_{a0}} = \exp\left[-V_d\left(\frac{A}{V}\right)t\right]$$

$$F_{PO} = 1 - F_{air}$$

$$DF = \frac{C_{a0}}{C_a} = \frac{1}{F_{air}}$$

It should be noted that the RADTRAD computer code contains a default plateout model identified as the "Broxman-Bixler" model as documented in Section 2.2.6 of NUREG/CR-6604 [17]. A detailed review confirms that the deposition coefficients for elemental and organic iodines are based on the Cline model.

The Brockman model for aerosol removal accounts for gravitational settling on the horizontally projected lower surface of the flow path during transport. The deposition efficiency, η_g, is modeled as

$$\eta_g = 1 - \exp\left(\frac{-U_g A}{\pi Q}\right)$$

$$U_g = \tau g$$

$$\tau = \frac{d_p^2 \rho_p}{18\mu}$$

where,

U_g =	Gravitational deposition velocity, (m/s)
τ =	Particle relaxation time, (s)
Q =	Gas flow rate, (m^3/s)
A =	Total pipe surface area, (m^3)
μ =	Viscosity of air, (Pa-s)
ρ_P =	Average particle density (kg/m^3).

If flow is turbulent, then turbulence in the central core can propel a particle into the laminar sublayer. If the particle inertia is high, then it will penetrate the sublayer and be collected on the wall.

$$\eta_{turb} = 1 - \exp\left(\frac{-U_t A}{Q}\right)$$

$$U_t = \frac{0.01988 U_{air}}{\mathrm{Re}^{1/8}}$$

$$\mathrm{Re} = \frac{\rho_{air} U_{air} d_H}{\mu}$$

where

$U_t =$ Turbulent inertial deposition velocity, (m/s)

$U_{air} =$ Air velocity, (m/s)

$d_H =$ hydraulic diameter, 4 vol. / A_s, (m)

$\rho_{air} =$ Air density (kg/m^3).

When the flow is slower such that

$$\frac{\tau U_{air}}{d_H} \mathrm{Re}^{3/4} < 326.6,$$

then the turbulent velocity is

$$U_t = 1.861 \times 10^{-7} \left[\frac{\tau U_{air}}{d_H}\right]^2 U_{air} \mathrm{Re}^{13/8}.$$

Small particles can undergo Brownian diffusion and move from areas of high concentration to areas of low concentration. The pipe wall acts as a sink for these small particles, therefore there is a net diffusion from the bulk airflow to the pipe. Brockmann modeled this phenomenon as:

$$\eta_{diff} = 1 - \exp\left(\frac{-U_{diff} A_s}{Q}\right).$$

In laminar flow it is modeled as

$$\eta_{diff} = 1.0 - 2.56\zeta^{2/3} + 1.2\zeta + 0.177\zeta^{4/3}$$

$$\zeta = \frac{A_s D_{iff}}{Q d_H}$$

where

$\zeta =$ dimensionless diffusion parameter and

D_{iff} = Particle diffusion coefficient (m^2/s).

When the flow is slow or the pipe length is large (such that $\zeta > 0.02$), then

$$\eta_{diff} = 0.819\exp(-3.657\zeta) + 0.097\exp(-22.3\zeta) + 0.032\exp(-57\zeta)$$

In turbulent flow, the diffusional efficiency is modeled as

$$\eta_{diff} = 1.0 - \exp\left[-\zeta^{2/3} 0.0187 \mathrm{Re}^{13/24}\left(\frac{L}{d_H}\right)^{1/3}\right]$$

where L is the path length ($=A_s/\pi d_H$) in m.

Modeling for aerosol deposition also accounts for particle inertia when it encounters a change in flow direction, such as a pipe bend. The efficiency of this process is modeled as:

$$\eta_{bend} = \frac{\tau U_{air}\varphi}{d_H}$$

where φ is the sum of the angles the flow has been diverted (in radians). When the flow is turbulent, this efficiency is modeled as

$$\eta_{bend} = 1.0 - \exp\left(\frac{-2.823\tau U_{air}\varphi}{d_H}\right)$$

Thus, the total aerosol deposition efficiency is modeled as:

$$\eta_{total} = 1.0 - \left(1.0 - \eta_g\right)\left(1.0 - \eta_{turb}\right)\left(1.0 - \eta_{diff}\right)\left(1.0 - \eta_{bend}\right).$$

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. The Gravity Driven Cooling System (GDCS) is contained entirely in the primary containment. The PCCS is also contained entirely in containment 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. Since 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. The final penetration list for the ESBWR has not been finalized, however, the preliminary design includes piping penetrations for the following systems:

- Main Steam (discussed in Section 4.4 above),
- Feedwater,
- Isolation Condenser System,
- Control Rod Drive System (CRDS),
- Standby Liquid Control System (SLCS),
- 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 as well.

The Reactor Building is discussed in depth in Section 6.2.3 of the ESBWR Design Control Document [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.

Detailed design for most of the listed piping systems is not finalized. However, the majority of the systems are almost completely housed in the Reactor Building (other than the portions in the containment itself). FAPCS is housed on the bottom elevation of the Fuel Building, adjacent to the Inclined Fuel Transfer System (IFTS) pool and the spent fuel pool. The Fuel Building is located adjacent to the reactor building. The Fuel Building is a Seismic Category I structure (with the exception of the penthouse, which is Category II). The fuel building is not explicitly modeled in the LOCA DCD dose calculation or in any analysis prepared in support of this report.

SRP Section 6.2.3, Revision 2, "Secondary Containment Functional Design," was issued in July 1981 [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 [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.

Since 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 be 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. With no A/C power the dominant heat source would be the PCCS and IC pools. 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.

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

Ten time periods were chosen for input into RADTRAD. The mass flow rates, as well as the corresponding volumetric flow rates are presented in Table 3.[1] Note this approach 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.

As discussed previously containment leakage could occur at numerous locations throughout the containment building. However, the list of containment penetrations has not been finalized. As such, it is difficult to analyze exact leakage paths for containment leakage with any high degree of confidence. One of the worst potential leakage locations is the PCCS condensers themselves since they are located near the top of the reactor building[2]. This location would have less of the reactor building atmosphere to mix with. Also, as discussed previously this is the location that would be the likely source of air flow from boiloff of the pool. However, only leakage from the PCCS condensers would leak into the PCCS pools, except for the short time period where the condensers are not submerged (currently determined to be 18 to 72 hours). Essentially all particulate iodine, as well as a major fraction of the elemental iodine, would likely be retained in the PCCS pool if this leakage were to occur at this location. This analysis will assume a DF of 2 for leakage through the PCCS

[1] The DCD analysis assumed an air flow rate of 100% per day and credited 100% of the building for mixing. [[

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[2] Note that the IC condensers are also located near the top of the reactor building, however, that system is designed to operate at reactor pressure that is significantly higher than the containment design pressure. Therefore, leakage through that system is unlikely and this discussion concentrates on the PCCS condensers. However, leakage though the IC system would behave in a similar way as the PCCS condensers. Also, the ICs can be isolated if a leak is detected.

pools while the condensers are submerged, and a DF of 1 if any portion of the condensers is not submerged.

Table 3 – PCC/IC Volumetric Flow Rates (VFR)

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5.0 OFFSITE DOSE CALCULATIONS AND SENSITIVITY STUDIES

5.1 Background

A "baseline" input deck was created. This baseline deck was based primarily on the analysis presented in the DCD, however a number of minor changes were made in order to address new technical knowledge (i.e., MELCOR results) as well as some regulatory questions and concerns (control room inleakage, etc.). The changes in assumptions include the following:

- Removal of aerosol iodine by the PCCS is updated based on MELCOR results,
- Natural deposition of radioiodine in the containment is updated based on MELCOR results,
- Release timing is updated based on the MELCOR results,
- Unfiltered air inleakage into the control room is assumed to address potential control room habitability issues,
- Reactor Building volume and leakage rate assumptions are updated, and
- Primary Containment leakage rates are updated.

The basis for the revised data was presented previously in Section 4. A comparison of the DCD analysis assumptions to the "baseline" input deck assumptions is presented in Appendix A of this report. The baseline analysis will then be used for the sensitivity studies.

In some areas it is difficult to precisely model the plant due to the fact that the ESBWR detailed design is not finalized. In some areas the dose analyses assumptions will dictate the design requirements (such as control room HVAC design), however in other areas this is not necessarily the case. Therefore, sensitivity studies provide useful insight for the relative importance of some parameters in order to efficiently address design and licensing basis issues.

It should be noted that the baseline analysis is not intended to replace the DCD analysis at this time. Rather the information provided by that analysis, as well as the insight provided by the sensitivity studies documented in this report, will likely be incorporated into a revision to the DCD analysis at a later date.

5.2 RADTRAD Base Analysis

As discussed previously, while the "base 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 LOCA dose analysis assumed that the control room was operated with EBAS for the first 72 hours. After 72 hours it was assumed that the control room utilized filtered intake and

recirculation. The analysis did not account for any potential unfiltered inleakage into the control room. This study will assume 10 cfm unfiltered inleakage while the control room is operated under EBAS in accordance with Regulatory Guide 1.78 assumptions [2]. The analysis will assume 100 cfm when the system is operated under filtered recirculation. The discharge flow will be adjusted proportionally to account for the additional inleakage. Also, when isolated EBAS is designed to supply the control room with 100 cfm for the first 72 hours. Since the analysis will assume unfiltered inleakage the purging effect of this additional 100 cfm will also be taken into account.

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The reactor building volume and leakage rate assumptions are also adjusted for the base analysis. As discussed in Section 4.5, typically only 50% of a building volume is credited for holdup and decay of radioactive materials for a secondary containment. While the DCD analysis credited 100% of the building volume, this base analysis credits only 50% as recommended by current regulations. Also, the DCD analysis utilized a simplifying assumption that the reactor building release rate was 100 volume % per day. This base analysis utilizes volumetric flow rates based on the steaming rates from the PCCS and IC pools as discussed in Section 4.5.

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Figure 2 – Control Room Inleakage Assumptions

The DCD LOCA dose analysis assumed that the containment leaks at the maximum design basis leakage rate of 0.5% per day for the first 72 hours, after which leakage is terminated. Regulatory Guide 1.183, Appendix A allows licensees to reduce the containment flow rate after 24 hours assuming that other plant analyses support these assumptions. For the ESBWR containment pressure remains elevated for the first 72 hours of the event. Therefore, this analysis will

conservatively assume that the containment leakage rate remains 0.5 volume % per day, and 0.25 % per day thereafter.

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, the release timing and duration will be based on that analysis. Note that the release fractions themselves will continue to be 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 2. The comparison of the airborne radioactivity in containment for MELCOR and RADTRAD shown in Figure 1 demonstrates good agreement between the two programs. It should be noted that in the DCD analysis credit was also taken for deposition of elemental iodine in the PCCS condensers. Since 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 as discussed in Section 4.3.

The results of the base analysis are presented in Table 4. The results were somewhat higher than those presented in the DCD. There are several reasons why this is the case. As just discussed, the model used to calculate the amount of removal of airborne particulates in the drywell was revised to reflect the results of VTT's MELCOR analysis. The net result is that less radioiodine is removed; therefore more remains airborne to be released. This analysis only credited 50% of the reactor building volume, whereas the DCD analysis credited 100% of that volume. This fact in conjunction with the release rate assumptions increases the overall release from the reactor building significantly. These phenomenons have a fairly significant impact on both short term (EAB) and long term (LPZ and control room). The containment release assumptions had a significant impact on the long-term doses as well. The DCD analysis assumed no release after 72 hours, whereas this analysis assumed 0.25% per day from containment. The control room doses increased mostly due to the introduction of unfiltered inleakage for the duration of the event.

Table 4 – Base Analysis Results (REM TEDE)

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5.3 Sensitivity Studies

5.3.1 Control Room Assumptions

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5.3.2 Reactor Building Assumptions

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5.3.3 Containment Leakage Assumptions

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5.3.4 MSIV Leakage Assumptions

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6.0 CONCLUSIONS

6.1 General

The ESBWR systems are redundant and diverse. The ESBWR DCD explains that for 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 for 4100 seconds. 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. Since 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 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 50.67.

6.2 Sensitivity Studies

6.2.1 Control Room Assumptions

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6.2.2 Reactor Building Assumptions

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6.2.3 Containment Leakage Assumptions

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6.2.4 MSIV Leakage Assumptions

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6.3 Ongoing Research

Research is continuing in many of the areas discussed in this draft report. As such, additional work may impact the methodology and assumptions presented in this report. A summary of the potential items that can or will impact the analyses documented in this report is provided below.

- **MELCOR Input Deck:** The MELCOR evaluation documented in this draft report utilized an input deck originally prepared for a lower power ESBWR, which in turn was based on the SBWR. Data was scaled as appropriate, but the exact values may change slightly once detailed design values are determined. As such much of the data is subject to minor changes. These changes may have an appreciable impact on the MELCOR analysis, however it is not expected to significantly impact calculated off-site and control room doses.

- **MELCOR and TRACG Comparison:** The timing of fuel damage for the MELCOR evaluation is not consistent with similar evaluations using the TRACG computer code [19]. Additional research shall be performed in an attempt to determine the reason for the discrepancy.

- **Atmospheric Dispersion Factors (X/Q):** The X/Q values used in this report were based on those used in the ESBWR DCD analysis, which were consistent with those used in previous analyses as well (such as the ABWR SSAR evaluation). Many utilities have recently recalculated atmospheric dispersion factors using the ARCON96 and PAVAN computer codes. Additional work may be performed to ensure that the X/Q values used in the ESBWR analysis are reasonable yet conservative.

- **Pool pH Analysis:** The iodine chemical forms assumed in this analysis inherently assumed that the pool pH would remain above 7 for the duration of the event. The pool pH analysis has not been finalized, however, it is not expected to impact the dose consequence analyses.

- **Reactor Building Leakage Rate:** The reactor building leakage rate assumed in the DCD analysis was estimated based on historical information, including the SBWR and current operating plants with Standby Gas Treatment systems. Specifically the SBWR assumed 25% per day, and that value was conservatively increased to 100% per day. This analysis assumed the PCCS and IC pools were the driving force, which yields even more conservative flow rates. Both analytical methodologies and physical testing options are being considered to further optimize reactor building leakage rate. Ultimately the reactor building flow rate will likely be reduced significantly to assist in limiting the dose consequences due to a LOCA or other radiological events.

Note that the above list may not be all-inclusive, however, it represents the best information available at the time this draft report was prepared.

7.0 REFERENCES

[1] 10 CFR 50.67, "Accident Source Terms"

[2] Regulatory Guide 1.78, "Evaluating the Habitability of a Nuclear Power Plant Control Room During a Postulated Hazardous Chemical Release," Revision 0.

[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] "MSIV Leakage Transport Analysis," J.E. Cline & Associates, Inc., March 26, 1991.

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

[13] NUREG/CR-1169, "Resolution to Generic Issue C-8, an Evaluation of Boiling Water Reactor Main Steam Isolation Valve Leakage and the Effectiveness of Leakage Treatment Methods," August 1996.

[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] NEDE-32176P, "TRACG Model Description," Revision 3.

[22] NEDE-32177P, "TRACG Qualification Report," Revision 2.

[23] NEDC-33083P, "TRACG Application for ESBWR."

[24] TM-49-AIDA-1-97, "AIDA Experiments: Aerosol Impact on the Performance of the SBWR Passive Containment Cooling Model," A. Dehbi et. al., October 1997.

[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 and Schmitt, 1966

[28] Talbot *et al.*, 1980

[29] Hinds, 1982

[30] Papavergos and Hedley, 1984

[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.

Appendix A - LOCA Dose Assumptions

Parameter	Current DCD Analysis	Revised Base Value
I. Data and Assumptions Used to Estimate Source Terms		
A. Power Level, MWt	4590	4590
B. Fraction of Core Inventory Released	RG 1.183, Table 1	RG 1.183, Table 1
C. Iodine Chemical Species		
Elemental, %	4.85	4.85
Particulate, %	95	95
Organic, %	0.15	0.15
D. PCCS		
Decontamination Factors		
Noble Gas	1	1
Elemental	10	[Note 1]
Particulate	10	
Organic	1	
Flow Rate to/from Containment		
0 – 2 hrs, m^3/min	4.82E+02	[Note 1]
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	
GDCS Pool Volume (m^3)	5.66E+02	5.66E+02

II. Data and Assumptions Used to Estimate Activity Released		
A. Primary Containment		
Leak Rate		
0 – 72 hrs, %/day	0.50	0.50
72 hrs – 30 days, %/day	0.00	0.25
Volume, m^3	7.206E+03	7.206E+03
Elemental iodine removal rate constant, hr^{-1}	0.92	0.92
Particulate Deposition Model	Power's (1)	[Note 1]
B. Reactor Building		
Leak Rate, %/day	100	120 – 200 [Note 2]
Mixing Efficiency, %	100	50
Volume, m^3	1.53E+06	7.65E+05
C. Condenser Data		
Free Air Volume, m^3	6.23E+03	6.23E+03
Mixing Fraction, %	20	20
Iodine Removal Factors		
Particulate, %	99.5	99.5
Elemental, %	99.5	99.5
Organic, %	0.0	0.0
D. MSIV Data		
MSIV Leakage, m^3/min	1.62E-02	1.62E-02
Main Steam Line Length, m	16.8	16.8
Main Steam Line Volume, m^3	5.32	5.32
Drain Line Length, m	50.6	50.6
Drain Line Volume, m^3	3.77E-01	3.77E-01
Main Steam Line OD/Thickness, mm	711/37	711/37
Drain Line OD/Thickness, mm	114/9	114/9
Main Steam Line Insulation Thickness, mm	114	114
Drain Line Insulation Thickness, mm	89	89
Plateout Factors	Reference [20]	Reference [20]

III. Control Room Parameters		
A. Control Room Volume, m^3	2.62E+03	2.62E+03
B. EBAS Flow		
0 – 72 hrs, m^3/min	0 [Note 3]	100
72 hrs – 30 days, m^3/min	0	0 [Note 4]
C. Unfiltered Inleakage		
EBAS, m^3/min	0	0.283
Filtered Intake/Recirculation, m^3/min	0	2.832
D. Ventilation Parameters (>72 hrs)		
Filtered Inleakage from Atmosphere, m^3/min	14.2	14.2
Recirculation Rate, m^3/min	7.1	7.1
Intake/Recirculation Filter Efficiency, %	99	99

Notes:

1. To adequately model the results of the MELCOR analysis, the PCCS DFs and flow rates are no longer explicitly modeled. See Section 4.3 of this report for additional discussion.
2. Reactor building leakage assumed in this analysis is a volumetric flow rate in cfm. The percentages listed are converted to volume % per day. See Section 4.5 for additional details.
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.

ENCLOSURE 3

MFN 06-205

Affidavit

General Electric Company

AFFIDAVIT

I, Louis M. Quintana, state as follows:

(1) I am Manager, Licensing, General Electric Company ("GE"), have been delegated the function of reviewing the information described in paragraph (2) which is sought to be withheld, and have been authorized to apply for its withholding.

(2) The information sought to be withheld is contained in the GE proprietary report, NEDE-33279P, *ESBWR Containment Fission Product Removal Evaluation Model (Draft)*, Class III (GE Proprietary Information), June 2006. The proprietary information is delineated by a double underline inside double square brackets. Figures and large equation objects are identified with double square brackets before and after the object. In each case, the superscript notation {3} refers to Paragraph (3) of this affidavit, which provides the basis for the proprietary determination.

(3) In making this application for withholding of proprietary information of which it is the owner, GE 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 2.790(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, <u>Critical Mass Energy Project v. Nuclear Regulatory Commission,</u> 975F2d871 (DC Cir. 1992), and <u>Public Citizen Health Research Group v. FDA,</u> 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 General Electric's competitors without license from General Electric 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 General Electric customer-funded development plans and programs, resulting in potential products to General Electric;

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 GE, 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 GE, 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. Access to such documents within GE 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, by the manager of the cognizant marketing function (or his delegate), and by the Legal Operation, for technical content, competitive effect, and determination of the accuracy of the proprietary designation. Disclosures outside GE 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 describes the models and methodologies GE will use in evaluating the dose consequences of design basis accidents (DBAs) for the ESBWR. GE performed significant additional research and evaluation to develop a basis for these revised methodologies to be used in evaluating the ESBWR over a period of several years at a cost of over one million dollars.

(9) Public disclosure of the information sought to be withheld is likely to cause substantial harm to GE's competitive position and foreclose or reduce the availability of profit-making opportunities. The information is part of GE'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 GE.

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.

GE's competitive advantage will be lost if its competitors are able to use the results of the GE 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 GE 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 GE of the opportunity to exercise its competitive advantage to seek an adequate return on its large investment in developing 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 27th day of June 2006

Louis M. Quintana

Louis M. Quintana
General Electric Company