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Proprietary Notice
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 08-630, Supplement 2

Docket No. 52-010

May 11, 2009

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

Subject: **Response to Portion of NRC Request for Additional Information Letter No. 281 Related to the ESBWR Design Certification – Reactor Building Mixing and Leakage Requirements – RAI Number 6.2-165 S01**

The purpose of this letter is to submit the GE Hitachi Nuclear Energy (GEH) response to the U.S. Nuclear Regulatory Commission (NRC) Request for Additional Information (RAI) sent by NRC letter dated December 4, 2008 (Reference 1). GEH submitted a plan for responding to portions of this RAI via GEH letter MFN 08-630, Supplement 1 dated March 10, 2009. This supplemental letter provides the final RAI 6.2-165 S01 response to all the RAI items and is addressed in Enclosure 1. Enclosure 3 contains the subject DCD Tier 2 markups that will be reflected in Revision 6.

Enclosure 1 contains GEH proprietary information as defined by 10 CFR 2.390. GEH customarily maintains this information in confidence and withholds it from public disclosure. Enclosure 2 is a non-proprietary version that is suitable for public disclosure.

The affidavit contained in Enclosure 4 identifies that the information contained in Enclosure 1 has been handled and classified as proprietary to GEH. GEH 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.

If you have any questions or require additional information, please contact me.

D068
NRO

Sincerely,

Richard E. Kingston

Richard E. Kingston
Vice President, ESBWR Licensing

Reference:

1. MFN 08-392, Letter from U.S. Nuclear Regulatory Commission to Robert E. Brown, GEH, *Request For Additional Information Letter No. 281 Related To ESBWR Design Certification Application*, dated December 4, 2008

Enclosures:

1. MFN 08-630, Supplement 2 Response to Portion of NRC Request for Additional Information Letter No. 281 Related to ESBWR Design Certification Application – Reactor Building Mixing and Leakage Requirements – RAI Number 6.2-165 S01 – GEH Proprietary Information
2. MFN 08-630, Supplement 2 Response to Portion of NRC Request for Additional Information Letter No. 281 Related to ESBWR Design Certification Application – Reactor Building Mixing and Leakage Requirements – RAI Number 6.2-165 S01 – Public Version
3. MFN 08-630, Supplement 2 Response to Portion of NRC Request for Additional Information Letter No. 281 Related to ESBWR Design Certification Application – Reactor Building Mixing and Leakage Requirements – RAI Number 6.2-165 S01 – DCD Tier 2 Markups
4. Affidavit – Larry J. Tucker – May 11, 2009

cc:	AE Cubbage	USNRC (with enclosures)
	JG Head	GEH/Wilmington (with enclosures)
	DH Hinds	GEH/Wilmington (with enclosures)
	eDRFs	0000-0097-7387 Revision 1

Enclosure 3

MFN 08-630, Supplement 2

Response to Portion of NRC Request for Additional Information Letter No. 281

Related to ESBWR Design Certification Application

Reactor Building Mixing and Leakage Requirements

RAI Number 6.2-165 S01
Addendum

DCD Tier 2 Markups

Table 1.3-3

Comparison of Containment Design Characteristics

Component (1)	Units	ESBWR	BWR/1 Dodewaard	ABWR
Design temperature of drywell	°C (°F)	171 (340)	150 (302)	171 (340)
Leakage rate	% weight in free volume / day	0.435	0.5	0.5

Note for Table 1.3-3:

(1) Where applicable, containment parameters are based on rated power.

The configuration of the pressure suppression containment with the suppression pool maximizes the scrubbing action of fission products by the suppression pool. The particulate and halogen content of the containment atmosphere following an accident is thereby substantially reduced compared to the Reg. Guide 1.183 source terms. The Passive Containment Cooling System (PCCS) condensing function contributes to reduce many of the airborne fission products.

Containment leakage is limited to less than 0.435% of the weight in the containment free volume per day.

1B.3.2 Post-Accident Access of Areas and Systems

This section addresses any area that may require occupancy to permit an operator to aid in the mitigation of or recovery from an accident. Areas that must be accessible after an accident are the control room and technical support center.

Areas requiring post-accident access also include consideration (in accordance with NUREG-0737, II.B.2) of the containment isolation reset control area, manual ECCS alignment area, motor control centers and radwaste control panels. However, the ESBWR design does not require a containment isolation reset control area or a manual ECCS alignment area, as these functions are available from the control room or are not applicable for the passive ECC systems. Areas requiring post-accident access that are normally areas of mild environment allowing unlimited access are not reviewed for access.

Systems specific to the ESBWR that may require post-accident access are those for long-term core cooling, fission product control and combustible gas monitoring, as well as the auxiliary systems necessary for their operation (i.e., instrumentation, control and monitoring, power, cooling water, and air cooling).

1B.3.3 Post-Accident Operation

Post-accident operations are those necessary to (1) maintain the reactor in a safe shutdown condition, (2) maintain adequate core cooling, (3) assure containment integrity, and (4) control radioactive releases within 10 CFR ~~50.34(a)~~52.47(a)(2)(iv) ~~guidelines~~limits.

Safety-related systems are required for scram and to achieve a safe shutdown condition. However, they are not necessarily needed to maintain safe shutdown. The systems identified in Section 1B.5 are the systems used to maintain the plant in a safe shutdown condition.

For purposes of this review, the plant is assumed to remain in the safe shutdown condition.

The basis for this position is that the foundation of plant safety is the provision of sufficient redundancy of systems and logic to assure that the plant is shut down and that adequate core cooling is maintained. Necessary shutdown and post-accident operations are performed from the control room, except for the manual external connections for the IC/PCCS and fuel pools makeup.

- Houses safety-related systems.

The RB consists of rooms/compartments, which are served by one of the three ventilation subsystems; Contaminated Area Ventilation Subsystem (CONAVS), Refuel and Pool Area Ventilation Subsystem (REPAVS), and Clean Area Ventilation Subsystem (CLAVS). None of these compartmentalized areas communicate with each other.

Under accident conditions, the RB (CONAVS and REPAVS areas) automatically isolate on high radiation to provide a hold up volume for fission products. When isolated, the RB (CONAVS and REPAVS areas) can be serviced by the RB HVAC Purge Exhaust Filter units (Subsection 9.4.6). No credit is taken for the filters in dose consequence analyses (Subsection 15.4.4). With low leakage and stagnant conditions, the basic mitigating function is the hold up of fission products in the RB CONAVS area itself. The ESBWR design does not include a secondary containment; however credit is taken for the existence of the RB CONAVS area surrounding the primary containment vessel in radiological analyses. CONAVS areas envelope all containment penetrations except penetration for main steam and feedwater lines located in the main steam tunnel. The radiological dose consequences for LOCAs, based on an assumed containment leak rate of 0.354% per day and RB CONAVS area leakage rate of 141.6 l/s (300 cfm), show that offsite and control room doses after an accident are less than allowable limits, as discussed in Chapter 15.

During normal plant operation, potentially contaminated areas within the RB are kept at a negative pressure with respect to the environment while clean areas are maintained at positive pressure. The ESBWR does not need, and thus has no filter system that performs a safety-related function following a design basis accident, as discussed in Subsection 6.5.2.3. Therefore the design criterion of GDC 43 is not applicable.

The affect of RB leakage less than the maximum leak rate used in the accident dose calculations has the potential to increase the radiation dose inside the RB following a design basis accident. The evaluation of the increased radiation levels to equipment is addressed through the environmental qualification program and any increased hazards during post-accident RB re-entry are addressed by the emergency planning program through emergency operating procedures.

Personnel and equipment entrances to the RB consist of vestibules with interlocked doors and hatches. Large equipment access is by means of a dedicated, external access tower that provides the necessary interlocks.

6.2.3.1 Design Bases

The RB is designed to meet the following safety design bases:

- The RB maintains its integrity during the environmental conditions postulated for a DBA.
- The Reactor Building HVAC System (RBVS) subsystems (CONAVS and REPAVS) automatically isolates upon detection of high radiation levels in their respective ventilation exhaust system.
- Openings through the RB boundary, such as personnel and equipment doors, are closed during normal operation and after a DBA by interlocks or administrative control. These doors are provided with position indicators and alarms that are monitored in the control room.

Table 6.2-1

Containment Design Parameters

Design Conditions:	
Upper and Lower Drywell	
Design Pressure	310 kPaG [45 psig]
Design Temperature	171°C (340°F)
Internal minus External Differential Pressure	-20.7 kPaD [-3.0 psid]
DW minus WW Differential Pressure	241 kPaD [35 psid] / -20.7 kPaD [-3.0 psid]
Inerting Gas	Nitrogen (with ≤ 3% Oxygen by Volume)
Wetwell	
Design Pressure	310 kPaG [45 psig]
Design Temperature	121°C (250°F)
Inerting Gas	Nitrogen (with ≤ 3% Oxygen by Volume)
Horizontal Vent System	
Design Pressure	310 kPaG [45 psig]
Design Temperature	171°C (340°F)
Containment Leak Rates	
Maximum Containment Leakage Excluding MSIV Leakage	~~0.4~~0.35% of Weight of Containment Free Volume per 24 hours at Pressure 310 kPaG [45 psig] and Standard Temperature 20°C (68°F)
Vacuum Breakers Between Drywell and Wetwell	
Number of Vacuum Breakers	Three (3)
Vacuum Breaker Opening Differential Pressure (WW Pressure minus DW Pressure)	3.07 kPaD [0.445 psid]
Vacuum Breaker Closing Differential Pressure (WW Pressure minus DW Pressure)	2.21 kPaD [0.320 psid]

15.4.4.5.1.2 Core Inventory

The core inventory assumed is discussed in Appendix 15B.

15.4.4.5.1.3 Reactor Power

The rated core thermal power of the ESBWR is 4500 MWt. Adding an additional 2% to account for instrument uncertainty yields a core thermal power for this analysis of 4590 MWth.

15.4.4.5.1.4 Iodine Chemical Distribution

RG 1.183, Appendix A, Section 2 states: "If the sump or suppression pool pH is controlled at values of 7 or greater, the chemical form of radioiodine released to the containment should be assumed to be 95% cesium iodine (CsI), 4.85% elemental iodine, and 0.15% organic iodine." Based on the application of the systems identified in Subsection 15.4.4.5.2.2 to control pH , this chemical distribution for pH-controlled pools is assumed in the analysis.

15.4.4.5.1.5 Radiation Decay and Daughter Products

The computer code RADTRAD allows tracking of radiation decay for the duration of the event. It also has an option to account for the buildup of daughter products. Both options are used in this analysis.

15.4.4.5.2 Radionuclide Releases and Pathways

The removal mechanisms for the ESBWR primary containment are passive in nature. They depend on the thermal hydraulic condition of the containment building. The MELCOR computer code is used to determine the amount of radionuclides removed from containment by passive means. Early in the event, high PCCS flow is due primarily to the high drywell pressure. The PCCS is a primary removal mechanism for airborne particulates. Therefore, assuming the fission products are released at the onset of the event (for determining containment removal coefficients) would be non-conservative. Instead, the removal coefficients are determined based on the onset of the bulk release of fission products (i.e., fuel melt), or the onset of the early in-vessel release phase. However, for the dose calculation itself, the release timing is based on NUREG-1465 and Regulatory Guide 1.183 (Subsection 15.4.4.5).

The dose consequence analysis considers leakage from the primary containment building and leakage through the Main Steam Isolation Valves. Leakage through the MSIVs is not included in the containment leakage summation, as discussed in Subsection 6.2.6.3. The primary containment leakage pathway is assumed to be no greater than an equivalent release of 0.354% wt per day from the containment. The majority of the primary containment leakage is released into the Reactor Building (RB). As an allowance, a small portion of the primary containment leakage is conservatively assumed to bypass the reactor building. This bypass leakage could occur through the feedwater (FW) isolation valves, or the PCCS condensers, and is released directly to the environment (see Table 15.4-5). MSIV leakage is directed to the Turbine Building condenser. This pathway is discussed separately below.

The RB is discussed in depth in Subsection 6.2.3. 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

could become acidic roughly 9 hours following the event; however, at that point the pool is essentially depleted and contains minimal fission products. The pH in the RPV could drop below 7 late in the event (~28 days); however, due to hold up and plateout in the containment, reactor building, and condenser (consistent with the main analysis assumptions), doses are insignificant after this pH transition, and any doses as a result of the re-evolution of elemental iodine are within the conservative assumptions of the dose analysis. Similarly, doses from re-evolution of iodine in the lower drywell is not of concern due to either release timing or the small amount of CsI present in the pool. Re-evolution is not included in the doses presented in Table 15.4-9.

15.4.4.5.2.3 Reactor Building Mixing Analysis

The ESBWR RB provides a holdup volume and delays transport of radioactivity from the containment to the environment. The RB credited mixing volume presented in Table 15.4-5 is the mixing volume that is assumed in the LOCA dose analysis. The LOCA dose analysis model produces uniform mixing within that volume. The GOTHIC computer code (Reference 15.4-19) is used for a detailed analysis of the RB and confirms that the mixing volume presented in Table 15.4-5 is a conservative characterization of the RB holdup and transport delay. The GOTHIC model assumes the same containment leakage rate and RB exfiltration rate as the LOCA dose analysis.

Several sub-volumes of the RB are modeled in GOTHIC. They include the CONAVS and CLAVS areas (Subsection 9.4.6), and stairwells. The CONAVS ventilation area envelopes all the containment penetrations, except those in the steam tunnel. Leakage from the steam tunnel penetrations is separately treated in the LOCA dose analysis. In some cases, the CLAVS areas are barriers between the CONAVS areas and the environment. The stairwells act as a transport path from the CONAVS areas to the environment. All the interior doors connecting the different rooms in the building, as well as the doors that connect to other buildings or to the environment, are modeled. Additionally, the HVAC ductwork connecting the appropriate volumes is also modeled in GOTHIC. Selected rooms within the CONAVS area are subdivided in the GOTHIC analysis.

A comparison of the GOTHIC and LOCA dose analysis results confirms that the credited mixing volume (Table 15.4-5) is conservative relative to the radiological releases traversing through the highly compartmentalized ESBWR RB. The comparison is a ratio of exfiltration to the environment over leakage into the RB. The GOTHIC analysis shows that hypothetical release from multiple penetrations into multiple RB sub-volumes provides significant holdup. The hypothetical release has to traverse through multiple volumes, ductwork, door gaps, and stairwells. GOTHIC demonstrates that under design basis accident conditions for a LOCA concurrent with LOOP and fuel damage, the mixing volume assumed in the LOCA dose analysis is conservative. Additional detail of the GOTHIC analysis ~~will be~~is presented in Reference 15.4-13, Appendix B.

The ESBWR Reactor Building design has flow resistances in the CONAVS area that provide hold up of radioactive releases from containment. A detailed GOTHIC analysis has been performed to model the amount of hold up in the ESBWR design. The GOTHIC analysis confirms that adequate resistances exist in the ESBWR design when the CONAVS area boundary flows are equal to or less than the flow assumed in the dose analysis.

Additionally, the ESBWR design has different containment leakage and safety envelope (CONAVS area) limits than the passive plant limits presented in NUREG-1242, Chapter 1B, Annex C, Issue 5 (Reference 15.4-20). The Issue 5 guidance of NUREG-1242 is utilized when meeting the dose acceptance criteria in the ESBWR design, although the containment leakage rate limit is lower and the safety envelope leakage limit is higher.

15.4.4.5.2.4 Main Steamline Modeling

The second potential release pathway is via the main steamline through leakage in the main steamline isolation valves. It is assumed that a pathway exists which permits the containment atmosphere, or in the non-break case, pressure vessel air-space direct access to the main steamlines. The main steamline isolation valves leakage is provided in the Technical Specification limit. Furthermore, it is assumed that the most critical main steamline isolation valve fails in the open position. Therefore, the total leakage through the steamlines contributes to the total Technical Specification limit.

The main steamlines are classified (see Table 3.2-1) as Seismic Category I from the pressure vessel interface to the outboard seismic restraint outboard of the downstream MSIV, thereby providing a qualified safety-related mitigation system for fission product leakage. The primary purpose of this system is to stop any potential flow through the main steamlines. Downstream of the seismic restraint referred to above, the steamlines pass through the Reactor Building - Turbine Building interface into the Turbine Building steam tunnel. The Turbine Building steam tunnel is a heavily shielded reinforced concrete structure designed to shield workers from main steamline radiation shine. The steamlines and their associated branch lines outboard of the last Reactor Building seismic restraint are Quality Group B structures. In addition, these lines and structures are required to be dynamically analyzed to SSE conditions (Table 3.2-1) that determine the flexibility and structural capabilities of the lines under hypothetical SSE conditions.

The analysis of leakage from the containment through the main steamlines involves the determination of

- Probable and alternate flow pathways;
- Physical conditions in the pathways; and
- Physical phenomena that affect the flow and concentration of radionuclides in the pathways.

The most probable pathway for radionuclide transport from the main steamlines is from the outboard MSIVs into the drain lines coming off the outboard MSIV and then into the Turbine Building to the main condenser. A secondary path is found along the main steamlines into the turbine though flow through this pathway, as described below is a minor fraction of the flow through the drain lines.

Consideration of the main steamlines and drain line complex downstream of the Reactor Building as a mitigating factor in the analysis of LOCA leakage is based upon the following determination:

- The main steamlines and drain lines are high quality lines inspected on a regular schedule.

15.4-12 NUREG/CR-6119, "MELCOR Computer Code Manuals," USNRC, September 2005.

15.4-13 NEDE-33279P, "ESBWR Containment Fission Product Removal Evaluation Model," Revision 1, August 2007.

15.4-14 U.S. Nuclear Regulatory Commission Regulatory Guide 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," July 2000.

15.4-15 NUREG-1465, "Accident Source Terms for Light-Water Nuclear Power Plants," USNRC, February 1995.

15.4-16 10CFR 52.47, "Contents of Applications; technical information."

15.4-17 10CFR 50, Appendix A, General Design Criterion 19, "Control Room."

15.4-18 Standard Review Plan 15.0.3, "Design Basis Accident Radiological Consequences of Analyses for Advanced Light Water Reactors," March 2007.

15.4-19 GOTHIC Containment Analysis Program, Version 7.2a(QA), EPRI, Palo Alto, CA.

15.4-20 NUREG-1242 Vol. 3 Pt. 1, "NRC Review of Electric Power Research Institute's Advanced Light Water Reactor Utility Requirements Document," USNRC, August 1994.

Table 15.4-5

Loss-of-Coolant Accident Dose Consequence Analysis Parameters

I. Data and Assumptions used to estimate source terms.	
A. Power Level, MWt	4590
B. Fraction of Core Inventory Released	RG 1.183, Table 1
C. Iodine Chemical Species	
Elemental, %	4.85
Particulate, %	95
Organic, %	0.15
D. Decay time*	Not Credited
E. Core Source Term	Table 15B-1
II. Data and Assumptions used to estimate activity released	
A. Primary Containment	
Total Leak rate, % per day	0.354
Reactor Building Bypass Leakage Rates###	
Release to PCCS Airspace, % per day	0.01
FW Isolation Valve Leakage (total all lines), standard m^3/min (standard ft^3/min)	7.00E-04 (2.47E-02)
FW Isolation Valve Leakage rate from containment (total all lines), adjusted for post-LOCA containment pressure and temperature, m^3/min (ft^3/min)#	
0 – 24 hr	5.60E-04 (1.98E-02)
> 24 hr	4.20E-04 (1.48E-02)
Volume, m^3 (ft^3)	1.260E+04 (4.447E+05)
Elemental iodine removal rate constant, hr^{-1}	
0 – 6.5 hrs+	0.86
> 6.5 hrs	0.0

Table 15.4-5

Loss-of-Coolant Accident Dose Consequence Analysis Parameters

Aerosol removal rate constants, hr^{-1}	
0 hr	6.500
0.5 hr	1.850
2.0 hrs	0.600
3.5 hr	0.950
4.75 hr	0.550
6.5 hr	0.300
8.0 hr	0.150
11.0 hr	0.100
12.5 hr	0.055
>24 hr^{++}	0.000
B. Reactor Building	
Leak rate, l/s (cfm)	141.6 (300)
(Deleted)	
Total volume, m^3 (ft^3)	6.05E+04 (2.14E+06)
Mixing Volume Credited, m^3 (ft^3)	1.16~~0~~E+04 (4.11~~5.65~~E+05)
C. Condenser Data	
Total free air volume, m^3 (ft^3)	5.93E+03 (2.09E+05)
Mixing fraction, %	20
Iodine removal factors	
Particulate, %	99.3
Elemental, %	99.3
Organic, %	0

5.5 Programs and Manuals

5.5.9 Containment Leakage Rate Testing Program (continued)

2. The visual examination of the steel liner plate inside containment intended to fulfill the requirements of 10 CFR 50, Appendix J, Option B testing shall be performed in accordance with the requirements of and frequency specified by ASME Code Section XI, Subsection IWE, except where relief has been authorized by the NRC.

COL 16.0-1-A
5.5.9-1

[3. . .]

b. The calculated peak containment internal pressure for the design basis loss of coolant accident, P_a, is ~~282.9~~310 kPaG (~~41.1~~45 psig). The containment design pressure is 310 kPaG (45 psig).

c. The maximum allowable containment leakage rate, L_a, at P_a, shall be ~~0.4~~0.35% of containment air weight per day.

d. Leakage rate acceptance criteria are:

1. Containment leakage rate acceptance criterion is ≤ 1.0 L_a for leakage from Containment . During the first unit startup following testing in accordance with this program, the leakage rate acceptance criteria are < 0.60 L_a for the Type B and C tests and ≤ 0.75 L_a for Type A tests.

2. Air lock testing acceptance criteria are:

a) Overall air lock leakage rate is ≤ 0.05 L_a when tested at ≥ P_a.

b) For each door, leakage rate is ≤ 0.01 L_a when pressurized to ≥ 10 psig.

3. Passive Containment Cooling System (PCCS) leakage rate acceptance criterion is ≤ ~~0.025 L_a~~0.01% of containment air weight per day.

e. The provisions of SR 3.0.3 are applicable to the Containment Leakage Rate Testing Program.

f. Nothing in these Technical Specifications shall be construed to modify the testing Frequencies required by 10 CFR 50, Appendix J.

BASES

APPLICABLE SAFETY ANALYSES

The safety design basis for the containment is that it must withstand the pressures and temperatures of the limiting DBA without exceeding the design leakage rate such that the postulated release of fission-product radioactivity subsequent to a DBA will not result in doses in excess of the values given in the licensing basis.

The DBA that results in a release of radioactive material within containment is a LOCA. In the analysis of this accident, it is assumed that containment is OPERABLE at event initiation such that release of fission products to the environment is controlled by the rate of containment leakage.

Analytical methods and assumptions involving the containment are presented in References 4 and 5. The safety analyses assume a non-mechanistic fission-product release following a DBA that forms the basis for determination of off-site doses. The fission-product release is in turn based on an assumed leakage rate from the containment. OPERABILITY of the containment ensures that the leakage rate assumed in the safety analyses is not exceeded, and that the site boundary radiation dose will not exceed the limits of 10 CFR 52.47(a)(2)(iv) and Regulatory Guide 1.183 (Refs. 1 and 2, respectively) even if the non-mechanistic release were to occur.

The maximum allowable leakage rate for the containment (L_a) is ~~0.4~~0.35% by weight of the containment air per 24 hours at the maximum calculated containment pressure (Ref. 4), excluding MSIV leakage. The bulk of the containment leakage ~~(0.975 L_a)~~ is released into the Reactor building. The remaining portion of primary leakage ~~(0.025 L_a)~~ is assumed to leak through the Passive Containment Cooling System (PCCS) into the airspace directly above the PCCS and Isolation Condenser pools and is quickly vented directly to the atmosphere.

Containment satisfies Criterion 3 of 10 CFR 50.36(d)(2)(ii).

BASES

BACKGROUND (continued)

to 10 CFR 50, Appendix J, Option B (Ref. 2), as modified by approved exemptions described in the Containment Leakage Rate Testing Program.

APPLICABLE SAFETY ANALYSES

The DBA that postulates the maximum release of radioactive material within containment is a LOCA. In the analysis of this accident, it is assumed that containment is OPERABLE, such that release of fission products to the environment is controlled by the rate of containment leakage. The containment is designed with an allowable leakage rate of ~~0.4~~0.35% by weight of the containment per 24 hours at the calculated maximum containment pressure (Ref. 3), excluding MSIV leakage. This allowable leakage rate forms the basis for the acceptance criteria imposed on the SRs associated with the air lock.

The containment air lock satisfies Criterion 3 of 10 CFR 50.36(d)(2)(ii).

LCO

As part of the containment pressure boundary, the air lock's safety function is related to control of containment leakage rates following a DBA. Thus, the air lock's structural integrity and leak tightness are essential to the successful mitigation of such an event.

Two containment air locks are required to be OPERABLE. For the air lock to be considered OPERABLE, both air lock doors must be OPERABLE, the air lock interlock mechanism must be OPERABLE, and the air lock must be in compliance with the Type B air lock leakage testing requirements as described in the Containment Leakage Rate Testing Program.

The closure of either the inner or outer door in each air lock is sufficient to provide a leak tight barrier following postulated events. However, both doors are kept closed when the air lock is not being used for normal entry into or exit from containment.

The air lock interlock mechanism allows only one air lock door to be opened at one time. This provision ensures that a gross breach of containment does not exist when the containment is required to be OPERABLE.

B 3.6 CONTAINMENT SYSTEMS

B 3.6.3.1 Reactor Building (Contaminated Area Ventilation Subsystem (CONAVS) Area)

BASES

BACKGROUND

The Reactor Building (RB) is a reinforced concrete structure that completely surrounds the containment (except the basemat). The RB provides an added barrier to fission product release from the containment during an accident; contains, dilutes, and holds up any leakage from the containment; and, houses safety-related systems.

The ESBWR design does not include a secondary containment; however credit is taken for the existence of the RB Contaminated Area Ventilation Subsystem (CONAVS) areas surrounding the primary containment vessel in radiological analyses. RB HVAC system performs no safety-related function, other than the building ventilation isolation function, but credit is taken for hold up in the RB CONAVS area volume as discussed in Reference 1. The radiological dose consequences for LOCAs are based on an assumed containment leak rate of ~~0.4~~0.35 weight percent per day. The bulk of the containment leakage ~~(0.975 L_a)~~ is released into the RB (CONAVS area) and the RB (CONAVS area) leaks to the environment at a maximum rate of 211 scfm (Ref. 2). The remaining portion of primary leakage ~~(0.025 L_a)~~ is assumed to leak through the Passive Containment Cooling System (PCCS) into the airspace directly above the PCCS and Isolation Condenser pools and is quickly vented directly to the atmosphere.

The RB (CONAVS area) envelops all penetrations through the containment (except penetrations for MSIV and feedwater lines located in the main steam tunnel and IC/PCC pools). Under accident conditions, the CONAVS area of the RB is isolated or passively sealed (e.g., water loop seals) to provide a hold up barrier. Therefore, containment isolation valve leakage as well as penetration leakage collects in the RB (CONAVS area). With low leakage and stagnant conditions, the RB (CONAVS area) provides a significant volume for hold up to enhance the basic mitigating functions provided by containment.

Automatic RB (CONAVS area) isolation dampers (other than MSIVs) are actuated by the Safety System Logic and Control/Engineered Safety Features (SSLC/ESF) portion of LD&IS as described in Bases for LCO 3.3.6.3, "Isolation Instrumentation," and LCO 3.3.6.4, "Isolation Actuation." The automatic RB (CONAVS area) isolation function of the

BASES

APPLICABLE SAFETY ANALYSES

The radiological dose consequences for LOCAs are based on an assumed containment leak rate of ~~0.4~~0.35 weight percent per day. The bulk of the primary containment leakage (~~97.5%, or 0.39~~0.34% per day) is released into the RB (CONAVS area) and leaks to the environment at a rate of 211 scfm (Ref. 2). Therefore, some credit is taken for hold up in the RB (CONAVS area) because the building is sealed during isolation.

Reactor Building (CONAVS Area) satisfies Criteria 3 of 10 CFR 50.36(d)(2)(ii).

LCO

This LCO requires that RB (CONAVS area) OPERABILITY is maintained by keeping all RB (CONAVS area) equipment hatches closed, keeping RB (CONAVS area) access doors closed, except for entry and exit, and ensuring RB CONAVS ventilation dampers actuate when required. RB (CONAVS area) OPERABILITY also requires RB (CONAVS area) leakage to be within limits.

For each RB CONAVS isolation damper, the LCO requires OPERABILITY of the required safety-related ~~actuator~~initiators associated with DC and Uninterruptible AC Electrical Power Distribution Divisions required by LCO 3.8.6, "Distribution Systems - Operating."

APPLICABILITY

The RB (CONAVS area) is required to be OPERABLE in MODES 1, 2, 3, and 4 because a DBA could cause a release of radioactive material to containment and the RB (CONAVS area) provides an added barrier to fission product release from the containment during an accident.

In MODES 5 and 6, the probability and consequences of these events are reduced due to the pressure and temperature limitations of these MODES. Therefore, the RB (CONAVS area) is not required to be OPERABLE in MODES 5 and 6.

ACTIONS

The ACTIONS are modified by two Notes. The first Note allows The RB (CONAVS area) boundary to be unisolated intermittently under administrative controls. This Note only applies to openings in the RB (CONAVS area) boundary that can be rapidly restored to the design condition, such as doors, hatches, floor plugs, and access panels. For entry and exit through doors, the administrative control of the opening is performed by the person(s) entering or exiting the area. For other

Enclosure 4

MFN 08-630, Supplement 2

Affidavit

Larry J. Tucker

May 11, 2009

GE-Hitachi Nuclear Energy Americas LLC

AFFIDAVIT

I, Larry J. Tucker, state as follows:

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

(2) The information to be discussed and sought to be withheld is delineated in the letter from Mr. Richard E. Kingston to U.S. Nuclear Regulatory Commission, entitled "*Response to NRC Request for Additional Information Letter No. 281 Related to the ESBWR Design Certification Application – Reactor Building Mixing and Leakage Requirements – RAI Number 6.2-165 S01,*" dated May 11, 2009. The information in Enclosure 1, which is entitled "*Response to NRC Request for Additional Information Letter No. 281 Related to the ESBWR Design Certification Application – Reactor Building Mixing and Leakage Requirements – RAI Number 6.2-165 S01*" – *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 identifies detailed GEH ESBWR calculations and analyses assumptions and inputs related to Reactor Building Mixing and Leakage GOTHIC analyses application. Development of this GOTHIC analysis was achieved at a significant cost to GEH, and results in a significant economic and competitive advantage to a competitor, and constitutes a major GEH asset.

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

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

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

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

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

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

Executed on this 11th day of May 2009.

Larry J. Tucker

Larry J. Tucker
GE-Hitachi Nuclear Energy Americas LLC