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Proprietary Notice
This letter forwards proprietary information in accordance with 10CFR2.390. Upon the removal of Enclosure 3, the balance of this letter may be considered non-proprietary.

MFN 08-676

Docket No. 52-010

September 10, 2008

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. 168 Related to the ESBWR Design Certification – Licensing Topical Report NEDO-33337 ESBWR Initial Core Transient Analysis – RAI Numbers 15.2-16, 15.2-18, 15.2-24, 15.2-28 through 15.2-30 and 15.2-38**

The purpose of this letter is to submit the GE Hitachi Nuclear Energy (GEH) responses to the U.S. Nuclear Regulatory Commission (NRC) Request for Additional Information (RAI) sent by NRC letter dated March 13, 2008. GEH responses to RAI Numbers 15.2-16, 15.2-18, 15.2-24, 15.2-28 through 15.2-30 and 15.2-38 are addressed in Enclosure 1. Enclosure 2 contains the subject LTR markups that will be reflected in Revision 1 to the Licensing Topical Report (LTR).

Enclosure 3 contains GEH proprietary information as defined by 10 CFR 2.390. GEH customarily maintains this information in confidence and withholds it from public disclosure. The 2-D Channel Flow and Core Pressure Drop for BOC, MOC and EOC Figures provided in Enclosure 3 are entirely proprietary and as a result, no public version is available.

The affidavit contained in Enclosure 4 identifies that the information contained in Enclosure 3 has been handled and classified as proprietary to GEH. GEH hereby requests that the information of Enclosure 3 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 08-247, Letter from U.S. Nuclear Regulatory Commission to Robert E. Brown, GEH, *Request For Additional Information Letter No. 168 Related To ESBWR Design Certification Application*, dated March 13, 2008

Enclosures:

1. Response to Portion of NRC Request for Additional Information Letter No. 168 Related to ESBWR Design Certification Application – Licensing Topical Report NEDO-33337 ESBWR Initial Core Transient Analysis – RAI Numbers 15.2-16, 15.2-18, 15.2-24, 15.2-28 through 15.2-30 and 15.2-38
2. MFN 08-676 Licensing Topical Report NEDO-33337 Markups
3. MFN 08-676 RAI 15.2-18 – 2-D Channel Flow and Core Pressure Drop for BOC, MOC and EOC Figures – GEH Proprietary Information
4. Affidavit – Larry J. Tucker – Executed September 10, 2008

cc:	AE Cubbage	USNRC (with enclosure)
	RE Brown	GEH/Wilmington (with enclosure)
	eDRFs	0000-0089-9752 – RAI 15.2-16
		0000-0090-0951 – RAI 15.2-18
		0000-0089-2291 – RAI 15.2-24
		0000-0089-6458 – RAI 15.2-28
		0000-0087-2925 – RAIs 15.2-29 and 15.2-30
		0000-0088-2642 – RAI 15.2-38

Enclosure 1

MFN 08-676

Response to Portion of NRC Request for Additional Information Letter No. 168 Related to ESBWR Design Certification Application

Licensing Topical Report NEDO-33337 ESBWR Initial Core Transient Analysis

RAI Numbers 15.2-16, 15.2-18, 15.2-24, 15.2-28 through 15.2-30 and 15.2-38

NRC RAI 15.2-16:

Throughout NEDO-33337, it is stated that the DCD analyses are "applicable" rather than "bounding." For example, in Section 1.2, for Analysis of Design Basis Accidents, it is stated that: "----The analysis documented in the DCD is applicable to both equilibrium and initial core -----." Also in Sections 2.4.7 to 2.4.12, it is stated that "Analysis is in the DCD applies to the initial core." There are total of 4 sets of analyses from GEH: DCD equilibrium analyses on operation point SP0, initial core transient analyses on operation point SP0, and initial core transient analyses on operation points SP1 and SP2 (FW Temp Operating Domain Transient and Accident Analyses LTR). Identify the bounding analysis for each event identified in the DCD Chapter 15 from these 4 sets of analyses. Justify why transient analyses for all events on operation points SP1 and SP2 with equilibrium core are not needed to identify the most limiting transient.

GEH Response:

Rather than only listing bounding events of the 4 sets of events in the DCD (Reference (a)), all events are either directly included or referenced in the DCD. The equilibrium core results at SP0 conditions are included directly, and the initial core results at SP0, SP2, and SP1 are included by reference. NEDO-33337 and NEDO-33338 (References (b) and (c) respectively) are referenced in:

- DCD Subsection 15.2.6 for Anticipated Operational Occurrences,
- DCD Subsection 15.3 for Infrequent Events,
- DCD Subsection 15.5.1.2 for the Overpressurization analysis,
- DCD Subsection 15.5.4.4 for ATWS,
- DCD Subsection 15.5.5.3 for Station Blackout,
- DCD Subsection 4.4.3.2 for Steady State, and
- DCD Subsection 4D.1.6 for Stability.

The bounding events of the 4 sets of events are shown by comparison in Section 3 of Reference (b) and Section 4 of Reference (c). Also, in response to RAI 15.0-31 (Reference (d)), an Appendix 15D was added to the DCD to summarize the conclusions of the analyses from NEDO-33337 and NEDO-33338 performed with the initial core.

Some analyses in the DCD are applicable to the same events in NEDO-33337, and no re-analyses are needed. For example, the LOCA analyses documented in the DCD (Reference (a)) are applicable to both equilibrium and initial cores. The reasons for this are discussed in the last bullet of Section 1.2 of NEDO-33337. In summary, for LOCA events not reanalyzed, the results with an initial core will not be different from those performed with the equilibrium core because the power, pressure, and temperature are the same; the core neutronics feedback is not important due to a scram, if present; and the hydraulic design is the same between the two cores. Also, the decay heats for both initial core and equilibrium core at 100% are very similar.

The analyses of the events in the DCD applies to the initial core for the events in NEDO-33337 Sections 2.4.7 through 2.4.12, 2.4.14, and 2.4.16 as discussed below:

2.4.7. Control Rod Withdrawal Error During Refueling – The following statement from DCD subsection 15.3.7.3, "The withdrawal of a control rod (or highest worth pair of control rods associated with the same HCU) does not result in criticality" is applicable to both equilibrium and initial cores.

2.3.3.1 (New) & 2.4.8 Control Rod Withdrawal Error During Startup – The analysis in the DCD is performed with an initial core as stated in DCD Subsection 15.3.8.3.1.

2.3.3.2 (New) & 2.4.9 Control Rod Withdrawal Error During Power Operation – Automated Thermal Limit Monitor (ATLM) stops rod movements and prevents violation of thermal operating limits for both equilibrium and initial cores. In the unlikely event that ATLM fails also, the Multi-Channel Rod Block Monitor (MRBM) prevents more than 1000 rod fuel failure.

2.4.10 Fuel Assembly Loading Error, Mislocated Bundle – The worst result discussed in the DCD is true for the initial core. That is, as stated in DCD subsection 15.3.10.4, "the potential exists that if the fuel bundle operates above the thermal-mechanical limit, one or more fuel rods may experience cladding failure. The detection of the fuel leak and power suppression of the damaged rod is the same regardless of the fuel.

2.4.11 Fuel Assembly Loading Error, Misoriented Bundle – The potential exists in this scenario that one or more fuel rods could experience cladding failure. The detection of the fuel leak and power suppression of the damaged rod is the same regardless of the fuel.

2.4.12 Inadvertent SDC Function Operation – The following statement in the DCD Subsection 15.3.12.3 is valid for both initial and equilibrium cores, "The increased subcooling caused by misoperation of the RWCU/SDC shutdown cooling mode could result in a slow power increase due to the reactivity insertion. During power operation, the reactor settles in a new steady state. During startup, if the power rises such that the neutron flux setpoint is reached, the power rise is terminated by a flux scram before approaching fuel thermal limits."

2.4.14 Inadvertent Opening of a Depressurization Valve – The following statement and corresponding conclusions from DCD subsection 15.3.14.3 apply to an initial core analysis, "The SB&PC system senses the nuclear system pressure decrease and within a few seconds closes the TCVs sufficiently to stabilize the reactor vessel pressure at a slightly lower value with the reactor returning to nearly the initial power level. The plant automatically scrams on high drywell pressure. After scram, depressurization of the RPV will resume." No fuel damage results from the event, and containment pressure does not reach containment design pressure limits.

2.4.16 Liquid Containing Tank Failure – As discussed in DCD 15.3.16, the analysis is only analyzed for radiological consequences of the spill and is independent of the fuel.

As stated in Section 4.2 of Reference (c), limiting transients will be reanalyzed for each reload including events at SP1 (or SP1M) and SP2 conditions, and the limiting event among the statepoints can be identified.

NEDO-33337 is updated to summarize this discussion for each of these events.

References:

(a) GE-Hitachi Nuclear Energy, "ESBWR Design Control Document," Tier 2, 26A6642, Revision 5, May 2008.

(b) GE-Hitachi Nuclear Energy, "ESBWR Initial Core Transient Analyses", NEDO-33337, Class I, Revision 0, October 2007.

(c) GE-Hitachi Nuclear Energy, "ESBWR Feedwater Temperature Operating Domain Transient and Accident Analysis", NEDO-33338 Class I, Revision 0, October 2007.

(d) GE Hitachi Nuclear Energy Letter # MFN 08-323 dated April 21, 2008, "Response to NRC Request for Additional Information Letter No. 157 Related to the ESBWR Design Certification – Safety Analyses - RAI Number 15.0-31."

LTR Impact:

LTR NEDO-33337 Revision 1 will reflect the attached markup.

NRC RAI 15.2-18:

For each exposure point analyzed (BOC, MOC, and EOC) in Section 2.2 "Stability Evaluation" of NEDO-33337, provide data similar to Figures 4A.1a through 4A.1e in the DCD Tier 2, Chapter 4,Appendix A. In addition, provide the 2-D channel flow distribution in the core and the core pressure drop.

GEH Response:

NEDC-33326P (Reference (a)) contains the figures for the initial core that are similar to Figures 4A.1a through 4A.1e that were included in DCD Tier 2 for the equilibrium core. In Reference (a), various state points were included in the figures and tables, but only 3 state points are analyzed for stability in NEDO-33337 (Reference (b)):

- 0 MWd/ST is used for "BOC",
- 3431 MWd/ST is used for "MOC", and
- 10665 MWd/ST is used for "EOC".

The following table maps the figures in the DCD to the similar figures in Reference (a):

DCD Figures	NEDC-33326P **0** MWd/ST Figure	NEDC-33326P **3431** MWd/ST Figure	NEDC-33326P **10665** MWd/ST Figure
4A-#a	3-4	3-5	3-8
4A-#b	3-9	3-10	3-13
4A-#c	3-14	3-15	3-18
4A-#d	3-21 & 3-22	3-23 & 3-24	3-29 & 3-30
4A-#e	--	--	3-20

All bundle average exposures (corresponding to DCD 4A-#e Figures) at BOC are 0. The bundle average exposure at MOC is provided in the attached figure.

The attached figures (see Enclosure 3 of GEH letter MFN 08-676) also contain the 2-D channel flow and core pressure drop for the 3 state points. The figures contain entirely proprietary information.

LTR Impact:

No changes to the subject LTR will be made in response to this RAI.

References:

(a) NEDC-33326P, Revision 0, "GE14E For ESBWR Initial Core Nuclear Design Report" dated July 2007.

(b) NEDO-33337, Revision 0, "ESBWR Initial Core Transient Analyses" dated October 2007

NRC RAI 15.2-24:

NEDO-33337, Table 2.3-1 is incomplete. Provide the following variables used or calculated in TRACG for AOO and Infrequent event analyses:

* *Core Flow*
* *Steam Flow*
* *Feedwater Flow Rate Analysis Value*
* *Vessel Core Pressure*
* *Core Leakage Flow %*
* *Safety Relief Valve Capacity*
* *Safety Function Delay*
* *Safety Function Opening Time*

NEDO-33337, Section 2.3.2.1.2, Closure of One Turbine Control Valve, states that Table 2.3-2 will provide the Closure time assumed in the slow closure analysis. But the closure time is not provided in the Table 2.3-1. Include the slow closure time in the table. Add the unit "seconds" in the parameter column of the Table.

GEH Response:

Table 2.3-1 of NEDO-33337 is updated to include the following parameters with their corresponding values as shown on the attached markup.

* *Core Flow*
* *Steam Flow*
* *Feedwater Flow Rate Analysis Value*
* *Core Leakage Flow %*
* *Safety Relief Valve Capacity*

"Vessel Core Pressure" was deleted from DCD Table 15.2-1 because the value is not an input to Anticipated Operational Occurrences (AOO) or Infrequent Event analyses. Also, the similar parameter, "vessel dome pressure" is already included. For consistency with the DCD and because of redundancy with vessel dome pressure, "Vessel Core Pressure" is not added to the LTR.

There are 2 AOO/Infrequent Events that have SRV/SV lifts: Inadvertent Opening of a Safety-Relief Valve (IOSRV) and Stuck-Open Safety Relief Valve (SRVSO). The Safety Function Delay is not applicable for either event, and for this reason, the "Safety Function Delay" is not added to the table.

For the SRVSO, the relief valves are already open at the initiation of the transient, and Safety Function Opening Time is not applicable. For the IOSRV event, fast opening time is conservative and assumed in the analysis. Subsection 2.4.13.3 is updated to document an instantaneous opening of the Safety-Relief Valve; therefore, "The Safety Function Opening Time" is not listed in the table.

Section 2.3.2.1.2 correctly references Table 2.3-1 rather than Table 2.3-2. Table 2.3-1 includes the TCV closure time shown in the 8th row (or 10th row in updated table). The unit "s" for seconds is correctly identified as the units for the closing time.

Other changes were made to the table as noted in the attached markup to add clarity, include units (or dual units), or provide consistency with DCD, Tier 2, Table 15.2-1.

LTR Impact:

LTR NEDO-33337 Revision 1 will reflect the attached markups.

NRC RAI 15.2-28:

SCRRI/SRI power increase Section 2.3.2.4.3 of NEDO-33337 states SCRRI/SRI will activate to reduce the power to 60% in the Turbine Trip with Turbine Bypass analysis. In Figure 15.2-6a of DCD Tier 2, SCRRI/SRI does reduce power to 60%, where power then appears to stabilize. In Figure 2.3-6a of NEDO-33337, power is shown to reduce to 60% but then increase to approximately 75%. This is additionally shown in the results of the Generator Load Rejection with Turbine Bypass analysis in Figure 2.3-4a, where power approaches 80%. Explain this power increase and describe if it continues to rise after the 400 seconds.

GEH Response:

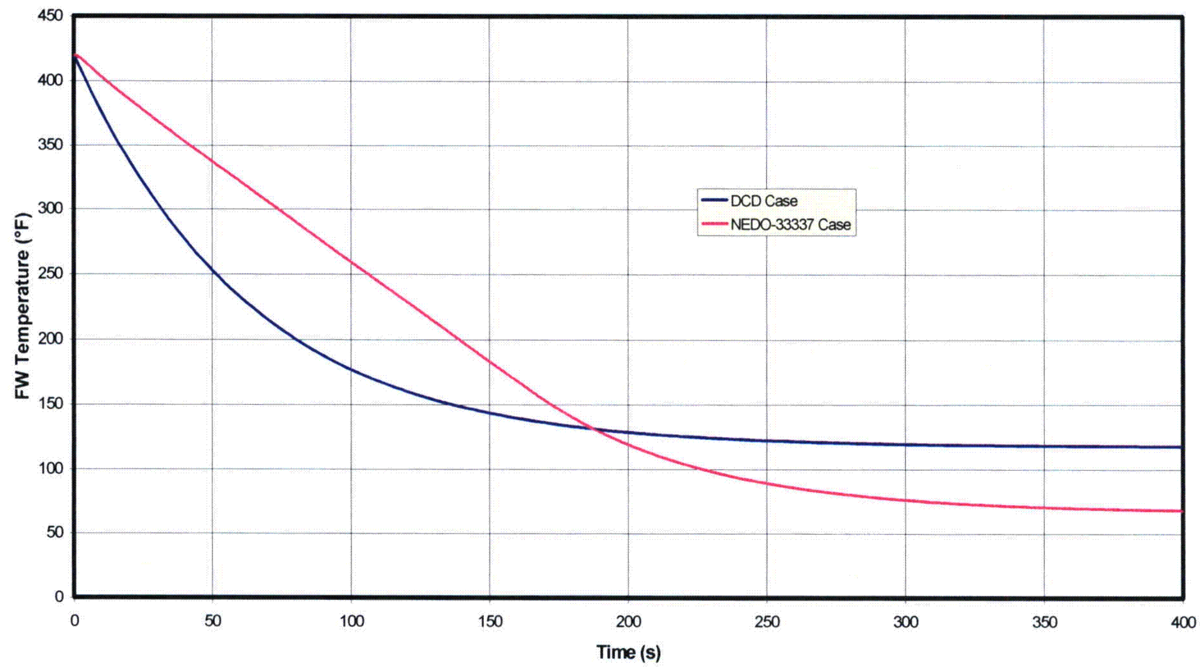
In the Generator Load Rejection and Turbine Trip with Turbine Bypass (LR/TTWBP) event evaluated in LTR NEDO-33337, the assumptions for feedwater (FW) temperature response following a LR/TTWBP were adjusted in TRACG to more accurately model the expected response based on the ESBWR balance of plant design (See DCD Tier 2 Table 10.1-1 and Figure 10.1-2). The FW temperature reduction following the event was adjusted to simulate the effect of the No. 4 heater (open FW heater). An unintended result of the change to the TRACG simulation was to lower the resulting, post event, steady state FW temperature to a temperature lower than is expected, ~20 °C (68°F). This lower steady state FW temperature is responsible for the increase in power observed at the end of the LR/TTWBP.

The FW temperature comparison for the TTWBP case in DCD Tier 2, Figure 15.2-6 and NEDO-33337, Figure 2.3-6 is shown in Figure 15.2-28-1. Note that the FW temperature in the NEDO-33337 case is very close to being steady at 400 seconds; however, the core power may increase a few more percent as the reactor comes to steady state. The FW temperature following a LR/TTWBP is expected to be closer to 49°C (120°F) (as shown in the rated heat balance DCD Tier 2, Figure 10.1-2) or higher, which would result in a steady state power level closer to, or lower than the DCD LR/TTWBP evaluation. The reduced steady state FW temperature produces conservative ΔCPR/ICPR results. Note that the limiting ΔCPR/ICPR comes from channel number 2420 (See NEDO-33337 Figure 2.6-6g). The minimum CPR for channel 2420 is approximately 1.40 at about 36 seconds, which is well below the value at 400 seconds (1.49); therefore, if the power increases slightly and CPR decreases slightly as the event comes to steady state, the channel number 2420 CPR will remain above the minimum during the transient. Because the results in NEDO-33337 are acceptable, no changes are made to the LTR.

LTR Impact:

No changes to the subject LTR will be made in response to this RAI.

Figure 15.2-28-1 – FWT Response Comparison



NRC RAI 15.2-29:

NEDO-33337, Section 2.3.2.2.1 states that: "No SCRRI rods are assigned." Confirm whether SRI rods are assigned. In the same paragraph it is stated: "The SCRRI/SRI function reduces the core power and limits the change in MCPR after a generator load rejection with turbine bypass." If no SCRRI rods are selected, but SRI rods are only selected explain how the SRI function alone reduces the core power and limits the change in the MCPR. For the analyses shown in the DCD, credit is taken for SCRRI and SRI. Why is deviation taken for the initial core analyses compared with the DCD analysis?

GEH Response:

Select Rod Insertion (SRI) rods are assigned.. SRI rods perform the same function as Selected Control Rod Run-In (SCRRI) rods, to reduce the core power and limit the change to Minimum Critical Power Ratio (MCPR). However, SRI rods are scrammed into the core by their associated Hydraulic Control Units (HCUs) where SCRRI rods insert via the Fine Motion Control Rod Drive (FMCRD) motors.

In the initial core analysis for the Generator Load Rejection With Turbine Bypass, acceptable Critical Power Ratio (CPR) performance was demonstrated with the SRI rod groups alone; as a result, SCRRI rods were not assigned in the analysis. SCRRI rods may be assigned, as needed, to reduce power further. This event is analyzed each cycle, as discussed in DCD Tier 2, Revision 5, Subsection 15.2.2.2.3, to evaluate a SCRRI/SRI rod pattern.

See GEH's response to RAI 4.6-28 Supplement 1, provided in GEH letter MFN 08-415 "Response to Portion of NRC Request for Additional Information Letter No. 157 - Related to ESBWR Design Certification Application - RAI Number 4.6-28 Supplement 1", dated April 24, 2008, for further SCRRI/SRI discussion.

LTR Impact:

LTR NEDO-33337, Revision 1 will reflect the attached markup.

NRC RAI 15.2-30:

NEDO-33337, Section 2.3.4.1. Confirm whether SRI rods are assigned. In the same paragraph it is stated: "The SCRRI/SRI function reduces the core power and limits the change in MCPR after a generator load rejection with turbine bypass." If no SCRRI rods are selected, but SRI rods are only selected explain how the SRI function reduces the core power and limits the change in the MCPR. For the analyses shown in the DCD, credit is taken for SCRRI and SRI. Why deviation is taken for the initial core analyses compared with the DCD analysis?

GEH Response:

GEH assumes this question is for Section 2.3.2.4.1, Turbine Trip With Turbine Bypass.

Select Rod Insertion (SRI) rods are assigned. SRI rods perform the same function as Selected Control Rod Run-In (SCRRI) rods, to reduce the core power and limit the change to Minimum Critical Power Ratio (MCPR). However, SRI rods are scrammed into the core by their associated Hydraulic Control Units (HCUs) where SCRRI rods insert via the FMCRD motors.

In the initial core analysis for the Turbine Trip With Turbine Bypass, acceptable Critical Power Ratio (CPR) performance was demonstrated with the SRI rod groups alone so SCRRI rods were not assigned in the analysis. SCRRI rods may be assigned, as needed, to reduce power further. The SCRRI/SRI rod pattern used in the Turbine Trip with Turbine Bypass is the same rod pattern used in Generator Load Reject with Turbine Bypass.

See GEH's response to RAI 4.6-28 Supplement 1, provided in GEH letter MFN 08-415 "Response to Portion of NRC Request for Additional Information Letter No. 157 - Related to ESBWR Design Certification Application - RAI Number 4.6-28 Supplement 1", dated April 24, 2008, for further SCRRI/SRI discussion.

LTR Impact:

LTR NEDO-33337, Revision 1 will reflect the attached markup.

NRC RAI 15.2-38:

NEDO-33337, Sections 2.4.13.3 and 2.4.15.2 state that open SRV capacity is assumed to be 1.1 times the capacity of the SV in DCD Tier 2, Table 5.2-2, and that this increased capacity, 154.2 Kg/s, is assumed to observe bounding depressurization results. What is the basis for assuming 154.2 kg/s, when the design capacity of the SRV is 138 kg/s?

GEH Response:

The Safety Relief Valves (SRVs) provide two main protection functions: overpressure relief function and depressurization operation. The SRVs are not required or expected to open during an Anticipated Operation Occurrence (AOO), but if an SRV is opened, the steam is discharged to the suppression pool, and there is a possibility that it can get stuck in the open position. As a result, the suppression pool temperature increases, reaching the scram set-point and finally scramming the reactor. In the "Inadvertent Opening of a Safety-Relief Valve" (IORV) event and "Stuck Open Safety-Relief Valve" (SORV) event, the sudden increase in the rate of steam flow leaving the reactor vessel causes a mild depressurization transient. Nevertheless, it is necessary to evaluate the effect of the depressurization.

Table 5.2-2 from DCD Tier 2 provides the SRV and Safety Valve (SV) ASME rated capacity. It is a minimum rated capacity based on two main protection functions: overpressure relief and depressurization. In the IORV and SORV events, the effect of unintended operation is evaluated. In this case (to evaluate the effect of the depressurization), the minimum capacity is not conservative; therefore, an arbitrary increase in the minimum capacity of an SV (because SV minimum capacity is slightly larger than the SRV) is assumed in the analysis.

The SV capacity, 140.2 kg/s, is increased by 10% as it is discussed in NEDO-33337 Sections 2.4.13 and 2.4.15. The value of 152.4 kg/s is a conservative assumption and it is not a prediction of actual valve performance.

LTR Impact:

No changes to the subject LTR will be made in response to this RAI.

Enclosure 2

MFN 08-676

Licensing Topical Report NEDO-33337 Markups

Table 2.3-4

Results Summary of Anticipated Operational Occurrence Events

Sub-section I.D.	**Description**	**Max. Neutron Flux,** % NBR	**Max. Dome Pressure,** MPaG (psig)	**Max. Vessel Bottom Pressure,** MPaG (psig)	**Max. Steamline Pressure,** MPaG (psig)	**Max. Core Average Surface Heat Flux,** % of Initial	**ΔCPR/ICPR or Minimum Water Level** (m over TAF)
2.3.2.1	Loss of Feedwater Heating	100	7.08 (1026)	7.21 (1046)	7.04 (1021)	100	<0.01
2.3.2.1	Closure of One Turbine Control Valve. FAST/SLOW	125 110	7.20 (1043) 7.20 (1043)	7.33 (1063) 7.33 (1063)	7.16 (1038) 7.16 (1038)	102 102	0.04 0.03
2.3.2.2	Generator Load Rejection with Turbine Bypass	128	7.15 (1037)	7.29 (1057)	7.28 (1056)	101	0.07*
2.3.2.3	Generator Load Rejection with a Single Failure in the Turbine Bypass System	151	7.37 (1070)	7.50 (1088)	7.37 (1069)	102	0.02
2.3.2.4	Turbine Trip with Turbine Bypass	116	7.12 (1033)	7.26 (1053)	7.20 (1043)	101	0.07*
2.3.2.5	Turbine Trip with a Single Failure in the Turbine Bypass System	131	7.34 (1065)	7.48 (1085)	7.34 (1065)	101	0.01
2.3.2.6	Closure of One MSIV	114	7.16 (1038)	7.30 (1059)	7.13 (1033)	102	0.03
2.3.2.7	Closure of All MSIV	102	7.67 (1112)	7.80 (1131)	7.67 (1112)	100	≤0.01
2.3.2.8	Loss of Condenser Vacuum	107	7.12 (1032)	7.26 (1053)	7.20 (1044)	100	≤0.01
2.3.4.1	Inadvertent Isolation Condenser Initiation	111	7.08 (1027)	7.22 (1047)	7.04 (1021)	109	0.09
2.3.4.2	Runout of One Feedwater Pump	103	7.08 (1027)	7.22 (1047)	7.04 (1021)	101	≤0.01
2.3.5.1	Opening of One Turbine Control or Bypass Valve	101	7.08 (1027)	7.21 (1046)	7.04 (1021)	100	≤0.01
2.3.5.2	Loss of Non-Emergency AC Power to Station Auxiliaries	139	7.13 (1035)	7.28 (1056)	7.28 (1056)	102	5.37m
2.3.5.3	Loss of Feedwater Flow	100	7.08 (1027)	7.21 (1046)	7.04 (1021)	100	5.28m

* A more detailed channel grouping, as utilized in the Loss of Feedwater Heating and Inadvertent Isolation Condenser Initiation events, would yield lower ΔCPR/ICPR values.

Table 3.1-2

Comparison of Results Summary of Anticipated Operational Occurrence Events Between Equilibrium and Initial Core Cases

Description	Equilib. Core (Equil.) or Initial Core (IC)	Max. Neutron Flux, % NBR	Max. Dome Pressure, MPaG (psig)	Max. Vessel Bottom Pressure, MPaG (psig)	Max. Steamline Pressure, MPaG (psig)	Max. Core Average Surface Heat Flux, % of Initial	ΔCPR/ICPR or Minimum Water Level (m over TAF)
Loss of Feedwater Heating*	Equil	100.2	7.08 (1027)	7.21 (1046)	7.04 (1024)	100	0.04
	IC	116	7.11 (1031)	7.24 (1050)	7.06 (1024)	119	0.09
Closure of One Turbine Control Valve. FAST/SLOW	Equil	124	7.20 (1043)	7.33 (1063)	7.16 (1038)	102	0.04
		112	7.20 (1043)	7.33 (1063)	7.16 (1038)	102	0.03
	IC	125	7.20 (1043)	7.33 (1063)	7.16 (1038)	102	0.04
		110	7.20 (1043)	7.33 (1063)	7.16 (1038)	102	0.03
Generator Load Rejection with Turbine Bypass**	Equil	135	7.15 (1037)	7.29 (1057)	7.28 (1056)	102	0.03
	IC	128	7.15 (1037)	7.29 (1057)	7.28 (1056)	101	0.07***
Generator Load Rejection with a Single Failure in the Turbine Bypass System	Equil	168	7.39 (1072)	7.53 (1091)	7.39 (1070)	103	0.03
	IC	151	7.37 (1070)	7.50 (1088)	7.37 (1069)	102	0.02
Turbine Trip with Turbine Bypass**	Equil	120	7.12 (1033)	7.26 (1053)	7.20 (1043)	101	0.02
	IC	116	7.12 (1033)	7.26 (1053)	7.20 (1043)	101	0.07***
Turbine Trip with a Single Failure in the Turbine Bypass System	Equil	146	7.37 (1069)	7.50 (1088)	7.37 (1067)	102	0.02
	IC	131	7.34 (1065)	7.48 (1085)	7.34 (1065)	101	0.01
Closure of One MSIV	Equil	114	7.16 (1038)	7.30 (1059)	7.13 (1033)	101	0.02
	IC	114	7.16 (1038)	7.30 (1059)	7.13 (1033)	102	0.03

Table 3.1-2 (cont) Comparison of Results Summary of Anticipated Operational Occurrence Events Between Equilibrium and Initial Core Cases							
Description	**Equilib. Core (Equil.) or Initial Core (IC)**	**Max. Neutron Flux,** % NBR	**Max. Dome Pressure,** MPaG (psig)	**Max. Vessel Bottom Pressure,** MPaG (psig)	**Max. Steamline Pressure,** MPaG (psig)	**Max. Core Average Surface Heat Flux,** % of Initial	**ΔCPR/ICPR or Minimum Water Level** (m over TAF)
Closure of All MSIV	Equil	103	7.76 (1126)	7.89 (1143)	7.76 (1126)	100	≤ 0.01
	IC	102	7.67 (1112)	7.80 (1131)	7.67 (1112)	100	≤ 0.01
Loss of Condenser Vacuum	Equil	110	7.11 (1031)	7.26 (1053)	7.20 (1044)	100	≤ 0.01
	IC	107	7.12 (1032)	7.26 (1053)	7.20 (1044)	100	≤ 0.01
Inadvertent Isolation Condenser Initiation	Equil	113	7.08 (1027)	7.22 (1047)	7.04 (1021)	109	0.08
	IC	111	7.08 (1027)	7.22 (1047)	7.04 (1021)	109	0.09
Runout of One Feedwater Pump	Equil	103	7.08 (1027)	7.22 (1047)	7.05 (1023)	100	<0.01>
	IC	103	7.08 (1027)	7.22 (1047)	7.04 (1021)	101	≤ 0.01
Opening of One Turbine Control or Bypass Valve	Equil	102	7.08 (1027)	7.21 (1046)	7.04 (1021)	100	≤ 0.01
	IC	101	7.08 (1027)	7.21 (1046)	7.04 (1021)	100	≤ 0.01
Loss of Non-Emergency AC Power to Station Auxiliaries	Equil	136	7.13 (1035)	7.28 (1056)	7.28 (10546)	102	5.40 m
	IC	139	7.13 (1035)	7.28 (1056)	7.28 (1056)	102	5.37m
Loss of Feedwater Flow	Equil	100	7.08 (1027)	7.21 (1046)	7.04 (1021)	100	5.28 m
	IC	100	7.08 (1027)	7.21 (1046)	7.04 (1021)	100	5.28m

* Results are very different because the SRI/SCRRI is conservatively not credited for the case with the initial core.

** Differences in the ΔCPR/ICPR is due to a change in the SCRRI/SRI rod pattern

*** A more detailed channel grouping, as utilized in the Loss of Feedwater Heating and Inadvertent Isolation Condenser Initiation events, would yield lower ΔCPR/ICPR values.

Table 2.3-1

Input Parameters And Initial Conditions and Assumptions Used In AOO and Infrequent Event Analyses

Parameter	Value
Thermal Power Level, MWt	4500
Core Flow kg/s (Mlbm/hr) [1]	10070 (79.9)
Steam Flow, kg/s (Mlbm/hr) [1] Analysis Value	2434 (19.3)
Feedwater Flow Rate, kg/s (Mlbm/hr) [1] Analysis Value Total Flow For All Pumps Runout, % of rated at 7.34MPaG (1065 psig) (At rated dome pressure, 7.07 MPaG [1025 psig]). The condensate and feedwater system in conjunction with the feedwater control system provide inventory equivalent to 240 s of rated feedwater flow after MSIV isolation. The condensate and feedwater system in combination with feedwater control system limit the maximum feedwater flow for a single pump to 75% of rated flow following a single active component failure or operator error.	 2428 (19.3) 155 (164)
Feedwater Temperature~~, °C (°F)~~ Rated, °C (°F) FW Heating Temperature Loss, Δ°C (Δ°F) Loss of FW Heating Setpoint (SCRRI/SRI Initiation), Δ°C (Δ°F)	 216 (420) 55.6 (100) 16.67 (30)
Vessel Dome Pressure, MPaG (psig)	7.07 (1025)
Turbine Bypass Capacity, % of rated	110
Total Delay Time from TSV or TCV motion to the start of BPV Main Disc Motion, s	0.02

Table 2.3-1

Input Parameters And Initial Conditions and Assumptions Used In AOO and Infrequent Event Analyses

Parameter	Value
Total Delay Time from TSV or TCV motion to 80% of Total Bypass Valve Capacity, s	0.17
TCV Closure Times, s~~econds~~ Fast Closure Analysis Value (Bounding) Assumed Minimum Servo (Slow) Closure Analysis Value	 0.08 2.5
TSV Closure Times, s~~econds~~	0.100
% of Rated Steam Flow ~~T~~that ~~C~~can ~~P~~pass ~~T~~through 3 Turbine Control Valves	85 (Partial Arc)
Minimum Steamline Pressure Difference Between the Vessel Dome Pressure and the Turbine Throttle Pressure at rated conditions, MPa (psi)(2)~~Turbine Inlet Pressure, MPaG (psig)~~	~~6.57~~0.179 (26)
Fuel Lattice	N
Core Leakage Flow, % (1)	9.4
Control Rod Drive Position versus Time	Table 2.3-2 & 3
Core Design used in TRACG Simulations Exposure:	Reference 2.3-2 Middle of Cycle and End of Cycle
Safety Relief Valve (SRV) and Safety Valve (SV) capacity At design pressure, MPaG (psig) Number of SRVs Number of SVs	See Subsections 2.4.13.3 & 2.4.15.2 8.618 (1250) 10 8
Analysis values for SRV and SV setpoints ~~Low~~SRV Setpoint, MPaG (psig) ~~High~~ SV Setpoint, MPaG (psig)	 8.618 (1250) 8.756 (1270)

Table 2.3-1

Input Parameters And Initial Conditions and Assumptions Used In AOO and Infrequent Event Analyses

Parameter	Value
Closure Scram Position of 2 or More MSIVs, % open Maximum delay time, s	85 0.06
MSIV Minimum Closure Time, s MSIV Maximum Closure Time, s	3.0 5.0
MSIV Closure Profile used to Bound Minimum Closure Time, s 100% open 100% open 1% open 0% open	 0.0 0.6 1.7 3.0
High Flux Trip, % NBR, Sensor Time Constant	125.0 0.03
TSV Closure Scram Position of 2 or more TSV, % open Trip Time delay, s	85 0.06
TCV Fast Closure Scram Trip, s	0.08
High Pressure Scram, MPaG (psig). Maximum scram delay, s	7.619 (1105) 0.7
High Suppression Pool Temperature Scram trip, °C (°F), Maximum Delay Time, s	48.9(120) 1.05
High Suppression Pool Temperature FAPCS actuation, °C (°F)	43.3 (110)
Vessel level Trips (above bottom vessel) Level 9—(L9), m (in) Level 8—(L8), m (in) Level 7 – (L7), m (in), high level alarm Normal Water Level, m (in) Level 4—(L4), m (in), low level alarm Level 3—(L3), m (in) Level 2—(L2), m (in) Level 1—(L1), m (in) Level 0.5 – (L0.5) m (in)	 22.39 (881.5) 21.89 (861.8) 20.83 (820.3) 20.72 (815.7) 20.60 (811.2) 19.78 (778.7) 16.05 (631.9) 11.50 (452.8) 8.45 (332.7)

Table 2.3-1

Input Parameters And Initial Conditions and Assumptions Used In AOO and Infrequent Event Analyses

Parameter	Value
Maximum APRM Simulated Thermal Power Trip Scram, % NBR Time Constant, s	 115 7
Simulated Thermal Power setpoint as a linear function of feedwater temperature for feedwater temperatures above 222.2°C (432°F)[4] Rate of Change limit on simulated thermal power setpoint[5]	115% at 222.2°C (432°F) 101% at 252.2°C (486°F) 26% / hour
~~Total~~ Minimum Steamline Volume, (total of all lines, including header): Vessel to TSV, m^3 (ft^3)[2]	~~135 (4767)~~103.3 (3648)
Minimum Steamline Length (average of all lines): Flow Path from Vessel to TSV, m (ft)[2]	65.26 (214.1)
CRD Hydraulic System minimum capacity, m^3/hr (gpm), Capacity in kg/s (Mlbm/hr) for 990 kg/m^3 density (61.8 lbm/ft^3)	235.1 (1035) 64.6 (0.513)
Maximum time delay from Initiating Signal (Pump 1 & 2), s If offsite power is not available, s	10 & 25 145
Isolation Condensers Max Initial Temperature, °C (°F) Minimum Initial Temperature, °C (°F) Time To injection valve full open (Max), s[~~1~~3] Heat Removal Capacity for 4 Isolation Condensers~~ICs~~, MW (% Rated Power) Isolation Condensers volume, 4 Units, from steam box to discharge at vessel m^3 (ft^3)	 40 (104) 10 (50) 31 ~~(1)~~ 135 (3%) 56.1 (1981)

Table 2.3-1

Input Parameters And Initial Conditions and Assumptions Used In AOO and Infrequent Event Analyses

Parameter	Value

(1) These are calculated steady state values not inputs or assumptions, and may change for different initial condition assumptions.

(2) These values are used in potentially limiting pressurization transients and bound the turbine throttle pressure in DCD, Tier 2, Table 10.1-1. Events that use the bounding steamline inputs use a different fuel bundle CPR R-factor than used in other AOO and infrequent event analyses. This changes the CPR values shown on the plots but does not significantly affect the ΔCPR/ICPR result. Historical values for turbine throttle pressure, 6.57 MPaG, (953 psig) and steamline volume, 135 m^3 (4767 ft^3) and larger steamline length (consistent with volume) are used in non-limiting events and non-pressurization events.

(3) ~~(1)~~ In the analysis, after 1 s logic delay, the isolation condenser~~IC~~ opening valve curve began to open at 15 s for a total opening time of 30 s. For inadvertent isolation condenser operation, ~~IICI~~ the valve begins to open at 15 s with an opening time of 7.5 s.

(4) As the reactor power changes with changes in the feedwater temperature (DCD Tier 2, Figure 4.4-1), the simulated thermal power trip setpoint also changes with feedwater temperature.

(5) Rate of change of the simulated thermal power setpoint is established to ensure that the simulated thermal power trip setpoint does not rapidly change with unexpected changes in the feedwater temperature. This simulated thermal power trip rate of change limit is based on the maximum planned rate in the feedwater temperature setpoint discussed in DCD, Tier 2, Subsection 7.7.3.2.3

2.4.13.3 **Core and System Performance**

Table 2.4-1 provides a summary of the results. The opening of one SRV allows steam to be discharged into the suppression pool. The valve is assumed to open instantaneously, and t~~The~~ sudden increase in the rate of steam flow leaving the reactor vessel causes a mild depressurization transient. In this event, the open SRV capacity is assumed to be 1.1 times the capacity of the SV in DCD, Tier 2 Table 5.2-2 (Reference 2.4-3). This increased capacity, 154.2 kg/s, (340 lbm/s) is assumed to observe bounding depressurization results.

The SB&PC system senses the nuclear system pressure decrease and within a few seconds closes the TCVs far enough to stabilize the reactor vessel pressure at a slightly lower value and the reactor settles at nearly the initial power level. Eventually, the plant automatically scrams on high suppression pool temperature.

Thermal margins decrease only slightly through the transient and no fuel damage results from the event.

2.3.3 Reactivity and Power Distribution Anomalies

~~This discussion in the DCD applies to the initial core.~~

2.3.3.1 **Control Rod Withdrawal Error During Startup**

Analysis in the DCD is performed with the initial core.

2.3.3.2 **Control Rod Withdrawal Error During Power Operation**

Analysis in the DCD applies to the initial core. ATLM stops rod movements and prevents violation of thermal operating limits

2.3.4 Increase in Reactor Coolant Inventory

2.4.7 Control Rod Withdrawal Error During Refueling

Analysis in the DCD applies to the initial core. The withdrawal of a control rod (or highest worth pair of control rods associated with the same HCU) does not result in criticality.

2.4.8 Control Rod Withdrawal Error During Startup with Failure of Control Rod Block

Analysis in the DCD is performed with~~applies to~~ the initial core.

2.4.9 Control Rod Withdrawal Error During Power Operation with ATLM Failure

Analysis in the DCD applies to the initial core. The MRBM prevents more than 1000 rod fuel failure.

2.4.10 Fuel Assembly Loading Error, Mislocated Bundle

Analysis in the DCD applies to the initial core. The potential exists in this scenario that one or more fuel rods could experience cladding failure. The detection of the fuel leak is followed by power suppression of the damaged rod.

2.4.11 Fuel Assembly Loading Error, Misoriented Bundle

Analysis in the DCD applies to the initial core. The potential exists in this scenario that one or more fuel rods could experience cladding failure. The detection of the fuel leak is followed by power suppression of the damaged rod.

2.4.12 Inadvertent SDC Function Operation

Analysis in the DCD applies to the initial core. The increased subcooling caused by misoperation of the RWCU/SDC shutdown cooling mode could result in a slow power increase due to the reactivity insertion. During power operation, the reactor settles in a new steady state. During startup, if the power rises such that the neutron flux setpoint is reached, the power rise is terminated by a flux scram before approaching fuel thermal limits.

2.4.13

2.4.14 Inadvertent Opening of a Depressurization Valve

Analysis in the DCD applies to the initial core. No fuel damage results from the event, and containment pressure does not reach containment design pressure limits.

2.4.15

2.4.16 Liquid Containing Tank Failure

Analysis in the DCD applies to the initial core. The analysis is only analyzed for radiological consequences of the spill and is independent of the fuel.

Enclosure 4

MFN 08-676

Affidavit – Larry J. Tucker

Executed – September 10, 2008

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 Portion of NRC Request for Additional Information Letter No. 168 Related to the ESBWR Design Certification – Licensing Topical Report NEDO-33337 ESBWR Initial Core Transient Analysis – RAI Numbers 15.2-16, 15.2-18, 15.2-24, 15.2-28 through 15.2-30 and 15.2-38," dated September 10, 2008. The information in Enclosure 3, which is entitled *MFN 08-676 RAI 15.2-18 – 2-D Channel Flow and Core Pressure Drop for BOC, MOC and EOC Figures – GEH Proprietary Information,* contains proprietary information, and is identified by [[dotted underline inside double square brackets{3}]]. Figures and other large objects are identified with double square brackets before and after the object. In each case, the superscript notation {3} refers to Paragraph (3) of this affidavit, which provides the basis for the proprietary determination.

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

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

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

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

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

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

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

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

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

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

(8) The information identified in paragraph (2) above is classified as proprietary because it contains containment pressure analysis results developed by GEH for analyzed accident scenarios. Development of these containment pressure response curves was achieved at a significant cost to GEH, and is considered 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 10th day of September 2008.

Larry J. Tucker

Larry J. Tucker
GE-Hitachi Nuclear Energy Americas LLC