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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 08-690

Docket No. 52-010

October 17, 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 NEDO-33337 ESBWR Initial Core Transient Analysis – RAI Numbers 15.2-20, 15.2-27, 15.2-33, 15.2-34 through 15.2-36**

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 15.2-20, 15.2-27, 15.2-33, 15.2-34 and 15.2-36 are addressed in Enclosure 1. Enclosure 3 contains the subject LTR markups that will be reflected in Revision 1 to the Licensing Topical Report (LTR), and 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 08-247, Letter from U.S. Nuclear Regulatory Commission to Robert E. Brown, GEH, *Request For Additional Information Letter No. 168 Related To NEDO-33337 ESBWR Initial Core Transient Analysis*, dated March 13, 2008

Enclosures:

1. Response to Portion of NRC Request for Additional Information Letter No. 168 Related to NEDO-33337 ESBWR Initial Core Transient Analysis – RAI Numbers 15.2-20, 15.2-27, 15.2-33, 15.2-34 and 15.2-36 – GEH Proprietary Information
2. Response to Portion of NRC Request for Additional Information Letter No. 168 Related to NEDO-33337 ESBWR Initial Core Transient Analysis – RAI Numbers 15.2-20, 15.2-27, 15.2-33, 15.2-34 and 15.2-36 – Public Version
3. MFN 08-690 DCD Tier 2 and Licensing Topical Report NEDO-33337 Markups
4. Affidavit – Larry J. Tucker

cc:	AE Cubbage	USNRC (with enclosures)
	RE Brown	GEH/Wilmington (with enclosures)
	DH Hinds	GEH/Wilmington (with enclosures)
	eDRFs	0000-0088-2032 – RAI 15.2-20
		0000-0088-5557 – RAI 15.2-27
		0000-0087-2926 – RAIs 15.2-33, 34, 36

Enclosure 2

MFN 08-690

Response to Portion of NRC Request for Additional Information Letter No. 168 Related to NEDO-33337 – ESBWR Initial Core Transient and Accident Analysis

RAI Numbers 15.2-20, 15.2-27, 15.2-33, 15.2-34 and 15.2-36

Public Version

NRC RAI 15.2-20:

Table 2.2-2 of NEDO-33337 does not show a calculated value for the regional mode decay ratio at EOC conditions, even though the regional mode appears to dominate the ESBWR response and it increases from BOC to MOC. Has GEH performed an EOC regional mode calculation at EOC for any core configuration? Provide a justification or additional data why the MOC exposure point results in the limiting stability condition for the cycle.

GEH Response:

In Table 2.2-2 of NEDO-33337, Rev 0, the regional mode decay ratio at end of cycle (EOC) condition is not given mainly because EOC condition is determined not limiting for regional mode oscillation. As shown in the Figure 2.2-3 of NEDO-33337, Rev 0, comparing with middle of cycle (MOC) condition the axial power shape at EOC is much less bottom peaked so that the boiling boundary shifts upward, which results in less two-phase pressure drop in the core; therefore, the channels are expected to be more stable. In addition, the subcriticality for higher order harmonic flux at EOC condition is evaluated to be higher than at MOC, which indicates that EOC condition is less susceptible to regional out-of-phase oscillations. Furthermore, the channel decay ratio as given in Table 2.2-2 of NEDO-33337, Rev 0 has the value of approximately 0.0, which also indicates that the regional decay ratio at EOC is bounded by MOC.

However, to complete the variation of regional mode decay ratio over the cycle and also to confirm the conclusion that MOC condition is limiting, GEH performs the regional mode stability analysis at EOC condition of initial cycle with the method documented in Licensing Topical Report (LTR) NEDE-33083P-A, Supplement 1 and Revision 1 (MFN-08-016, 01/28/2008). The regional decay ratio is evaluated to be 0.20 at EOC, which clearly demonstrates that MOC exposure point results in the limiting stability condition for the initial cycle.

DCD or LTR Impact:

No DCD changes will be made in response to this RAI.

LTR NEDO-33337, Revision 1, Subsection 2.2.1.3.1, Table 2.2-2 and Table 3.1-1 will be revised as noted in the Enclosure 3 markup.

NRC RAI 15.2-27:

Sections 2.3.2.2.1 and 2.3.2.4.1 of NEDO-33337 state "A typical rod pattern is analyzed in this event" for both turbine trip with turbine bypass and generator load rejection with turbine bypass analyses. How is a typical rod pattern defined? Demonstrate that this is the most conservative rod pattern for both events. If it is not the most conservative, demonstrate how the ESBWR avoids violating local thermal limits with the most conservative pattern.

GEH Response:

A) How is a typical rod pattern defined for the ESBWR?

The 'typical rod pattern' refers to the Selected Control Rod Run-In/Select Rod Insertion (SCRRI/SRI) rod pattern. The wording 'typical rod pattern' will be removed from NEDO-33337 [2] and NEDO-33338 [3], which document the initial core analyses.

The SCRRI/SRI rod patterns analyzed for the ESBWR initial core (NEDO-33337 Rev. 1, [2] and NEDO-33338 rev.1 [3]) and the equilibrium core (DCD Tier 2 Rev. 5 [1]) are defined according to Table 1:

Table 1: SCRRI/SRI Transient Analyses Assumptions

Event	Equilibrium Core (DCD [1])	Initial Core (Reference 2 and 3)	Core Channel Grouping
LOFWH	6 SRI/ No SCCRI (Subsection 15.2.1.1)	6 SRI/ No SCRRI (Subsection 2.3.1.1/3.3.1.1 [3])	Detailed
LR/TTWBP	4 SRI/ 1 SCRRI (Subsection 15.2.2.2 and 15.2.2.4)	6 SRI/ No SCRRI (Subsection 2.3.2.2/3.3.2.2 [3] and 2.3.2.4/3.3.2.4 [3])	Coarse

The Loss of Feedwater Heating (LOFWH) is run with a different set of SRI rods and a more detailed core channel grouping than the Generator Load Rejection and Turbine Trip with Turbine Bypass (LR/TTWBP) transient analyses. This demonstrates that the LOFWH ΔCPR/ICPR is very low (< 0.01) with the more detailed core channel grouping (equilibrium core and initial core). The feedwater temperature (FWT) reduction in the LR/TTWBP occurs more slowly than the instantaneous 55.6°C (100°F) FWT reduction assumed in the LOFWH. Therefore, ΔCPR/ICPR is bounded by the LOFWH ΔCPR/ICPR given the same SCRRI/SRI rod pattern. However, even with the very conservative coarse channel grouping and lack of SCRRI rods, the LR/TTWBP results are acceptable.

Note that Table 1 description of the DCD analysis is correct, and that there is an error in the DCD Rev. 5 SCRRI/SRI group discussion in Subsection 15.2.1.1. The corrected pages of the DCD Rev. 5 Subsection 15.2.1.1 are attached to this response.

The credited SCRRI/SRI rod pattern for the fuel cycle design will be specified in the Core Operating Limits Report (COLR). As stated in DCD Rev. 5 [1] Subsection 15.2.2.4.1, the SCRRI/SRI pattern for the LR and TTWBP will be identical. The DCD states that different rod patterns can be defined for different events (i.e. LOFWH vs. TT/LRWBP).

LRWBP and TTWBP are very similar events, and both are potentially limiting for Operating Limit Minimum Critical Power Ratio (OLMCPR), because of the effect that cycle-to-cycle changes to the SCRRI/SRI rod pattern have on ΔCPR/ICPR. The load rejection event results in slightly higher ΔCPR/ICPR values. Therefore the LRWBP event and the SCRRI/SRI rod pattern will be reevaluated for each fuel cycle.

B) Demonstrate that this is the most conservative rod pattern for both events.

The SCRRI/SRI rod pattern will be defined in the COLR per Technical Specification 5.6.3 and LCO 3.7.6. The pattern defined is analyzed at the most limiting of the BOC, MOC, and EOC conditions. The results of that analysis are used to establish core operating limits prior to each reload cycle, or prior to any remaining portion of a reload cycle, in accordance with Technical Specification 5.6.3.

C) If it is not the most conservative, demonstrate how the ESBWR avoids violating local thermal limits with the most conservative pattern

Any defined SCRRI/SRI control rod pattern that differs from the one specified in B above must be evaluated to prove that the new pattern ensures thermal limits are not violated for the limiting transient events specified in the DCD Tier 2 [1] and in LTRs NEDO-33337 [2] and NEDO-33338 [3]. Any changes to the defined control rods assumed to insert and the final control rod pattern achieved to accomplish the SCRRI /SRI functions have to be analyzed for each fuel cycle and documented in the COLR per Technical Specification 3.7.6. This analysis is also performed for and is documented in the COLR per Technical Specification 5.6.3 and LCO 3.7.6.

References:

[1] GE-Hitachi Nuclear Energy, "ESBWR Design Control Document," Tier 2, 26A6642, Revision 5, May 2008.

[2] GE-Hitachi Nuclear Energy, "ESBWR Initial Core Transient Analyses", NEDO-33337, Class I, Revision 0, October 2007.

[3] GE-Hitachi Nuclear Energy, "ESBWR Feedwater Temperature Operating Domain Transient and Accident Analysis", NEDO-33338, Class I, Revision 0, October 2007

DCD or LTR Impact:

DCD Subsection 15.2.1.1 changes will be made in response to this RAI to correctly state SCRRI/SRI assumptions. Please see attached Enclosure 3 markup.

LTR NEDO-33337, Revision 0 and LTR NEDO-33338 Revision 0 will be revised to remove the statement: "A typical rod pattern is analyzed in this event" for the initial core analysis as noted on the attached Enclosure 3 markups. This affects Sections 2.3.2.2.1 and 2.3.2.4.1 of LTR NEDO-33337, and Sections 2.3.3.2 and 3.3.3.2 of LTR NEDO-33338.

NRC RAI 15.2-33:

NEDO-33337, Table 2.3-20 Sequence of Events for Opening of one TCV - In the analysis shown in the DCD, Turbine Control Valve (TCV) opens at 0.1 second, but in the initial core analysis, TCV opens only at 3 to 7 seconds. Explain the difference in the sequence between the DCD analysis and the initial core analysis.

GEH Response:

The event is evaluated by simulating opening of one bypass valve at time zero as stated in the sequence of events table. The Turbine Control Valve (TCV) does not open during the transient. The TCV does close slightly to control pressure in response to the open bypass valve. It is the timing of the closing of the TCV that is inconsistent between the sequence of events table in DCD Tier 2, Table 15.2-20 and in NEDO-33337, Table 2.3-20. The response of the TCV between the DCD case and the NEDO-33337 case is very similar. The inconsistent time is due to different interpretation of the event description. A detailed review of the case shows that the valve begins to close between 0.3 and 0.4 seconds. The DCD valve closure begins at about the same time, which is very close to the approximate time given in the DCD. The event description in both the DCD and the LTR will be changed to be clear that the event is the beginning of valve closure. The exact time the valve begins to close is not important in this event due to the large margins in the anticipated operational occurrence (AOO) acceptance criteria. Therefore, the time the TCV begins to close listed in both DCD and LTR tables will be changed to "less than 1 second."

DCD or LTR Impact:

DCD Rev 5 Chapter 15, Table 15.2-20 Sequence of Events for Opening of one Turbine Control or Bypass Valve, will be changed to be the same described on LTR NEDO-33337 Revision 1, Table 2.3-20 Sequence of Events for Opening of one Turbine Control or Bypass Valve.

LTR NEDO-33337, Table 2.3-20 Sequence of Events for Opening of one Turbine Control or Bypass Valve, will be revised in Revision 1 as shown on the attached Enclosure 3 markups.

NRC RAI 15.2-34:

NEDO-33337, Figure 2.3-1g-Nodes 2404 to 2430 are identified in the figure. Submit a diagram to show the relative positions of the nodes in the core model.

GEH Response:

A core diagram is included as Figure 15.2-34-1. The numbers in the figure represent channel groups in TRACG. The number in the figure corresponds to the number in the legend of Figure 2.3-1g. For example: the channel group 2404 in the figure is in a channel group in TRACG that represents one channel. The CPR value for this channel is shown in NEDO-33337, Figure 2.3-1g and labeled as "2404" in the legend. Also, Table 15.2-34-1 shows the initial power for each channel corresponding to nodes (channels) 2404 to 2430 in NEDO-33337, Figure 2.3-1g. Channels 2400 and 2402 are included as well.

DCD or LTR Impact:

No DCD changes will be made in response to this RAI.

No Changes to LTR (NEDO-33337) will be made in response to this RAI.

Table 15.2-34-1Tabulated Values of Channels Average Power at Steady State Included in Figure 15.2-34-1

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Public Version

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Figure 15.2-34-1 "Loss of Feedwater Heating – Channel Grouping

NRC RAI 15.2-36:

NEDO-33337, Figure 2.3-16g includes MCPR for nodes 0110, 0140,0210, and 0240. Confirm whether the single line shown is applicable for all nodes?

GEH Response:

Yes, the single line on Figure 2.3-16g is applicable for nodes 0110, 0140, 0210 and 0240. Values between those nodes are very close one to the each other and they appear to be only one single line in the plot.

DCD or LTR Impact:

No DCD changes will be made in response to this RAI.

No changes to LTR (NEDO-33337) will be made in response to this RAI.

Enclosure 3

MFN 08-690

DCD Tier 2 and Licensing Topical Report
NEDO-33337 Markups

Core stability is evaluated at BOC, MOC and EOC conditions. The calculations are made with the 3-D kinetics model interacting with the thermal hydraulics parameters. The response to a pressure perturbation in the steam line is analyzed to obtain the decay ratio.

Regional Stability

The 'nominal' decay ratio for out-of-phase regional oscillations was calculated by perturbing the core in the out-of-phase mode about the line of symmetry for the azimuthal harmonic mode.

The initial conditions were the same as for the channel and core stability cases at nominal conditions. The decay ratio calculations were made at ~~both~~ BOC, ~~and~~ MOC and EOC conditions ~~because of the lowest value of the sub-criticality and highest bottom peaking at these conditions.~~ The channel decay ratios are the highest at BOC and MOC because of the bottom peaked axial flux shape. The decay ratio and oscillation frequency were extracted from the responses for the individual channel groups.

Results

The results for channel, super bundle, core and regional stability are tabulated in Table 2.2-2. The channel decay ratio was the highest at MOC because of the bottom peaked axial power shape. The channel decay ratios meet the design goal of approximately 0.4. The oscillation time period is approximately twice the transit time for the void propagation through the channel. The transit time through the chimney does not contribute to the oscillation time period. There is pressure equalization at the top of the bypass region, which reduces the importance of the chimney. Moreover, there are insignificant frictional losses in the chimney, and the static head does not affect the stability performance.

The super bundle decay ratio was lower than that for the single high power bundle, because of the lower average power for the group of 16 bundles.

The core decay ratio was the highest at MOC conditions due to the combination of axial power shape and void coefficient. The oscillation time period corresponds to twice the vapor transit time through the core region. The core decay ratios meet the design goal of approximately 0.4.

The decay ratio and oscillation frequency for regional stability were extracted from the responses for the individual channel groups. The results for the limiting channel group are tabulated in Table 2.2-2. Several other channel groups were within 0.02 of the highest group. The regional decay ratio is near to the design goal of approximately 0.4 and meets the acceptance criterion of 0.8.

2.2.1.4 Statistical Analysis of ESBWR Stability

2.2.1.4.1 Channel Decay Ratio Statistical Analysis

A Monte Carlo analysis of channel stability was performed and documented in Reference 2.2-1 for the equilibrium core at rated power and flow and at the conditions that were determined to be limiting. A total of 59 trials were made. In each trial, random draws are made for each of the parameters determined to be important for stability. Some of these parameters are not important for channel stability per se, but the same set of parameters was perturbed for both channel and core stability. These parameters and their individual probability distributions are listed in Reference 2.2-1. The value for each of these parameters is drawn from the individual probability distribution for that parameter. A TRACG calculation is made with this perturbed set of

Table 2.2-2

Baseline Stability Analysis Results

Mode	BOC		MOC		EOC	
	Decay Ratio	Frequency (Hz)	Decay Ratio	Frequency (Hz)	Decay Ratio	Frequency (Hz)
Channel	0.14	0.85	0.16	0.85	~0.0	-
Superbundle			~0.0	-		
Core	0.18	0.67	0.28	0.75	0.22	0.63
Regional	0.32	0.84	0.43	0.88	0.20	0.66

Table 3.1-1

Comparison of Baseline Stability Analysis Results Between Equilibrium and Initial Core Cases

Mode	Equilib. Core (Equil.) or Initial Core (IC)	BOC		MOC		EOC	
		Decay Ratio	Frequency (Hz)	Decay Ratio	Frequency (Hz)	Decay Ratio	Frequency (Hz)
Channel	Equil	0.23	0.80	0.09	~0.75	0.05	~0.7
Channel	IC	0.14	0.85	0.16	0.85	~0.0	-
Superbundle	Equil	0.14	0.74				
Superbundle	IC			~0.0	-		
Core	Equil	0.26	0.74	0.33	0.74	0.29	0.66
Core	IC	0.18	0.67	0.28	0.75	0.22	0.63
Regional	Equil	0.40	0.82				
Regional	IC	0.32	0.84	0.43	0.88	0.20	0.66

15.2.1.1.1 Identification of Causes

A feedwater (FW) heater can be lost in at least two ways:

- Steam extraction line to heater is closed; and/or
- FW is bypassed around heater.

The first case produces a gradual cooling of the FW. In the second case, the FW bypasses the heater and no heating of the FW occurs. In either case, the reactor vessel receives colder FW. The maximum number of FW heaters that can be tripped or bypassed by a single event represents the most severe event for analysis considerations.

The ESBWR is designed such that no single operator error or equipment failure causes a loss of more than 55.6°C (100°F) FW heating.

The loss of FW heating causes an increase in core inlet subcooling. This increases core power due to the negative void reactivity coefficient. However, the power increase is slow.

A LOFWH that results in a significant decrease in feedwater temperature is independently detected by the ATLMs and by the Diverse Protection System (DPS), either of which mitigates the event by initiating SCRRI and SRI functions as discussed in Subsections 7.7.2.2.7.7, 7.7.3.3 and 7.8.1.1.3. This prevents the reactor from violating any thermal limits. These functions are also collectively referred to as SCRRI/SRI.

Control rod insertion is conservatively assumed to start only when the temperature difference setpoint is reached in the FW nozzle. The SCRRI/SRI is able to suppress the neutron power increase and ensure that the MCPR reduction is small.

The SCRRI/SRI function reduces the core power and limits the change in MCPR after a Loss of Feedwater Heating. The SCRRI/SRI rod pattern depends on the fuel cycle exposure and initiating event. A typical rod pattern is analyzed in this event. ~~The rod pattern analyzed is divided in five control rod groups. Four SRI groups, with scattered insertion times (a separation of 10 seconds between each subgroup) and a SCRRI group with a total insertion time of 110 seconds, activates simultaneously with the first SRI group.~~The rod pattern analyzed is divided in seven control rod groups, six SRI groups, with scattered insertion times (a separation of 10 seconds between each subgroup) and one SCRRI group. SCRRI rods are not defined in this rod pattern; they are not required to show acceptable CPR results.

Events may exist where the SCRRI/SRI is not activated because the loss of feedwater temperature is less than 16.67°C (30°F). These events have a power increase of up to approximately 106% starting at rated conditions. The resulting ΔCPR/Initial Critical Power Ratio (ICPR) is approximately 0.04 and is bounded by the inadvertent isolation condenser initiation (Table 15.2-4a). Therefore, these events do not need to be reanalyzed for any specific core configuration.

Table 15.2-5

Sequence of Events for Loss of Feedwater Heating

Time (s)	Event
0	Initiate a 55.6°C (100°F) temperature reduction in the FW system
22.7 (est)	Selected Control Rod Run-In and Select Rod Insert (SCRRI/SRI) is initiated
24 (est)	Initial effect of unheated FW starts to raise core power level
24.0	First SRI group inserts (one HCU, 2 control rods, fails to actuate) and SCRRI start insertion
34.0	Second SRI group inserts
44.0	Third SRI group inserts
54.0	Fourth SRI group inserts
64	Fifth SRI group inserts
~~90.0~~67	Steam flow below 60% ~~of~~ rated
~~134.0~~74	~~SCRRI/SRI groups totally inserted~~Sixth SRI group inserts
~~140.0~~114	Power below 60% ~~of~~ rated
250.0 (est.)	Reactor variables settle into new steady state

* See Figure 15.2-1. This Figure has 20 s of steady state, a time of 0 s on the table corresponds to 20 s on the figure.

2.3.2.1.3 Core and System Performance

A simulated fast closure of one TCV (using the bounding steamline inputs in Table 2.3-1) is presented in Figure 2.3-2. Neutron flux increases, because of the void reduction caused by the pressure increase. ~~However,~~ Using bounding steamline inputs, the ~~sensed~~ calculated neutron flux ~~does not~~ reaches the high neutron flux scram setpoint; however, the scram is not credited because the peak is close to the high flux scram analytical limit. When the sensed reactor pressure increases, the pressure regulator opens the bypass valves, keeping the reactor pressure at a constant level. The calculated peak simulated thermal ~~flux~~power is provided (results provided for the bounding steamline and historical steamline inputs from Table 2.3-1) in Table 2.3-4. The number of rods in boiling transition, for this transient, remains within the acceptance criterion for AOOs. Therefore, the design basis is satisfied.

A slow closure of one TCV is also analyzed as shown in Figure 2.3-3. ~~As in the fast closure case, t~~The neutron flux increase does not reach the high neutron flux scram setpoint. ~~Also, a reactor scram on high reactor pressure may also be generated.~~ The results of this event are very similar to the fast closure event discussed above. During the transient, the number of rods in boiling transition remains within the acceptance criterion for AOOs. Therefore, the design basis is satisfied.

2.3.2.1.4 Barrier Performance

Peak pressure at the SRVs is below the SRV setpoints. Therefore, there is no steam discharged to the suppression pool. The peak vessel bottom pressure is below the upset pressure limit.

2.3.2.1.5 Radiological Consequences

Because this event does not result in any fuel failures or any release of primary coolant to the environment, there is no radiological consequence associated with this event.

2.3.2.2 Generator Load Rejection With Turbine Bypass

2.3.2.2.1 Identification of Causes

Fast closure of the TCVs is initiated whenever electrical grid disturbances occur which result in significant loss of electrical load on the generator. The TCVs are required to close as rapidly as possible to prevent excessive over-speed of the turbine-generator (TG) rotor. Closure of the TCVs causes a sudden reduction in steam flow. To prevent an increase in system pressure, sufficient bypass capacity is provided to pass steam flow diverted from the turbine.

After sensing a significant loss of electrical load on the generator, the TCVs are commanded to close rapidly. At the same time, the turbine bypass valves are signaled to open in the "fast" opening mode by the SB&PC system, which uses a triplicated digital controller. As presented in Subsection 2.3.~~4.2~~5.1.1, no single failure can cause all turbine bypass valves to fail to open on demand.

Assuming no single failure the plant will have the full steam bypass capability available, the Reactor Protection System (RPS) will verify that the bypass valves are open. The fast closure of the TCVs will produce a pressure increase that is negligible, because all the steam flow is bypassed through the steam bypass valves. The reactor power decreases when the SCRRI/SRI actuates.

The SCRRI/SRI function reduces the core power and limits the change in MCPR after a generator load rejection with turbine bypass. The SCRRI/SRI rod pattern depends on the fuel cycle exposure and initiation event. ~~A typical rod pattern is analyzed in this event.~~ The rod pattern analyzed is divided in six SRI control rod groups with scattered insertion times. The insertion times are listed in Table 2.3-8. No SCRRI rods are assigned.

2.3.2.2.2 Sequence of Events and System Operation

Sequence of Events

A loss of generator electrical load from high power conditions produces the sequence of events listed in Table 2.3-8.

Identification of Operator Actions

There is no scram during this event. There is no operator action required to mitigate the event.~~Relatively small changes in plant conditions are experienced. The operator should, after checking that the SCRRI/SRI system has been activated, check reactor water level, reactor pressure and MSIV status. If conditions are normal, no further operator actions are needed.~~

System Operation

To properly simulate the expected sequence of events, the analysis of this event assumes normal functioning of plant instrumentation and controls, plant protection and reactor protection systems unless stated otherwise.

All plant control systems maintain normal operation unless specifically designated to the contrary.

2.3.2.2.3 Core and System Performance

Input Parameters and Initial Conditions

The Turbine Generator Control System (TGCS)~~turbine electro-hydraulic control system (EHC)~~ detects load rejection before a measurable turbine speed change takes place.

The closure characteristics of the TCVs are assumed such that the valves operate in the full arc (FA) mode. For this event, Table 2.3-1 provides the worst case full stroke closure time (from fully open to fully closed) for the TCVs, which is assumed in the analysis.

The bypass valve opening characteristics are simulated using the specified delay together with the specified opening characteristic required for bypass system operation.

Results

Figure 2.3-4 shows the results of the generator trip from the 100% rated power conditions and with the turbine bypass system operating normally. Although the peak neutron flux and average simulated thermal ~~heat flux~~ power increase, the number of rods expected in boiling transition remains within the acceptance criterion for AOOs.

2.3.2.2.4 Barrier Performance

Peak pressure at the SRVs is below the SRV setpoints. Therefore, there is no steam discharged to the suppression pool. The peak vessel bottom pressure remains below the upset pressure limit.

The pressurization and/or the reactor scram may compress the water level to the low level trip setpoint (Level 2) and initiate the CRD high pressure makeup function, and if the low level signal remains for 30 seconds, MSIV closure, and isolation condenser (IC) operation. Should this occur, it would follow sometime after the primary concerns of fuel thermal margin and overpressure effects have occurred.

Results

Figure 2.3-5 shows the results of the generator trip from the 100% rated power conditions (using the bounding steamline inputs in Table 2.3-1) assuming only 50% of the total turbine bypass system capacity. Table 2.3-4 shows the results for this event for bounding steamline and historical steamline inputs from Table 2.3-1. Although the peak neutron flux and average simulated thermal ~~heat flux~~power increase, the number of rods in boiling transition remains within the acceptance criterion for AOOs in combination with an additional single active component failure or operator error.

2.3.2.3.4 Barrier Performance

Peak pressure at the SRVs is below the SRV setpoints. Therefore, there is no steam discharged to the suppression pool. The peak vessel bottom pressure remains below the upset pressure limit.

2.3.2.3.5 Radiological Consequences

Because this event does not result in any fuel failures or any release of primary coolant to the environment, there is no radiological consequence associated with this event.

2.3.2.4 Turbine Trip With Turbine Bypass

2.3.2.4.1 Identification of Causes

A variety of turbine or nuclear system malfunctions can initiate a turbine trip. Some examples are high velocity separator drain tank high levels, large vibrations, operator lockout, loss of control fluid pressure, low condenser vacuum and reactor high water level.

After the main turbine is tripped, turbine bypass valves are opened in their fast opening mode by the SB&PC system. The reactor power decreases when the SCRRI/SRI actuates.

The SCRRI/SRI function reduces the core power and limits the change in MCPR after a turbine trip with turbine bypass. ~~A typical rod pattern is analyzed in this event.~~ The SCRRI/SRI rod pattern used in the turbine trip with turbine bypass is the same as the one used in the generator load rejection with turbine bypass discussed in Subsection 2.3.2.2.1

2.3.2.4.2 Sequence of Events and Systems Operation

Sequence of Events

Turbine trip at high power produces the sequence of events listed in Table 2.3-10.

Identification of Operator Actions

There is no scram during the event. There is no operator action required to mitigate the event.~~Relatively small changes in plant conditions are experienced. The operator should, after checking that the SCRRI/SRI system has been activated, check reactor water level, reactor pressure and MSIV status. If conditions are normal, no further operator actions are needed.~~

Table 2.3-20

Sequence of Events for Opening of one Turbine Control or Bypass Valve

Time (sec)	**Events** *
0	One Turbine Bypass opens
~~3.0~~ ~~7.0~~≤1	TCV begins to closes slightly to control pressure
30.0	New steady state is established

* See Figure 2.3-14.

Table 15.2-19

Sequence of Events for Runout of One Feedwater Pump

Time (s)	**Events** *
0	Initiate simulated increase in speed of one FW pump. The maximum pump flow is 75% at rated conditions.
~0.1	Feedwater controller starts to reduce the FW flow from the FW pumps.
6.0	Vessel water level reaches its peak value and starts to return to its normal value.
21.0 (est.)	Vessel water level returns to its normal value.

* See Figure 15.2-13.

Table 15.2-20

Sequence of Events for Opening of one Turbine Control or Bypass Valve

Time (s)	**Events** *
0	One Turbine Bypass opens
≤1 ~~0.1~~	TCV begins to closes slightly to control pressure
30.0	New steady state is established

* See Figure 15.2-14.

Enclosure 4

MFN 08-690

Affidavit

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 NEDO-33337 ESBWR Initial Core Transient Analysis – RAI Numbers 15.2-20, 15.2-27, 15.2-33, 15.2-34 and 15.2-36*," dated October 17, 2008. The information in Enclosure 1, which is entitled "*Response to Portion of NRC Request for Additional Information Letter No. 168 Related to NEDO-33337 ESBWR Initial Core Transient Analysis – RAI Numbers 15.2-20, 15.2-27, 15.2-33, 15.2-34 and 15.2-36*" – *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 channel average power results for Loss of Feedwater Event results developed by GEH for analyzed transients. Development of these channel average power results 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 17th day of October 2008.

Larry J. Tucker

Larry J. Tucker
GE-Hitachi Nuclear Energy Americas LLC