HITACHI

GE Hitachi Nuclear Energy

Richard E. Kingston
Vice President, ESBWR Licensing

P.O. Box 780
3901 Castle Hayne Road, M/C A-55
Wilmington, NC 28402 USA

T 910.675.6192
F 910.362.6192
rick.kingston@ge.com

MFN 08-618

Docket No. 52-010

August 11, 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 Nos. 152 and 168 Related to the ESBWR Design Certification – Licensing Topical Reports (LTR) NEDO-33337 "Initial Core Transient Analysis" and NEDO-33338 "ESBWR Feedwater Temperature Operating Domain Transient and Accident Analysis" – RAI Numbers 15.2-14, 15.2-17, 15.2-19, 15.2-21, 15.2-25, 15.2-26, 15.2-31, 15.2-32, 15.2-35 and 15.2-37**

This letter submits GE Hitachi Nuclear Energy (GEH) responses to the U.S. Nuclear Regulatory Commission (NRC) Request for Additional Information (RAI) sent by NRC letters dated February 11, 2008 and March 13, 2008, respectively. GEH responses to RAI Numbers 15.2-14, 15.2-17, 15.2-19, 15.2-21, 15.2-25, 15.2-26, 15.2-31, 15.2-32, 15.2-35 and 15.2-37 are addressed in Enclosure 1.

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

Sincerely,

Richard E. Kingston

Richard E. Kingston
Vice President, ESBWR Licensing

DO68
NRO

References:

1. MFN 08-121, Letter from U.S. Nuclear Regulatory Commission to Robert E. Brown, GEH, *Request For Additional Information Letter No. 152 Related To ESBWR Design Certification Application*, dated February 11, 2008
2. 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 Nos. 152 and 168 Related to ESBWR Design Certification Application – Licensing Topical Reports (LTR) NEDO-33337 "Initial Core Transient Analysis" and NEDO-33338 "ESBWR Feedwater Temperature Operating Domain Transient and Accident Analysis" – RAI Numbers 15.2-14, 15.2-17, 15.2-19, 15.2-21, 15.2-25, 15.2-26, 15.2-31, 15.2-32, 15.2-35 and 15.2-37
2. LTR NEDO-33337 and NEDO-33338 Markups

cc:	AE Cubbage	USNRC (with enclosure)
	RE Brown	GEH/Wilmington (with enclosure)
	eDRFs	0000-0088-0199 – RAI 15.2-14
		0000-0087-2922 – RAI 15.2-17
		0000-0087-3151 – RAI 15.2-19
		0000-0087-2923 – RAI 15.2-21
		0000-0087-4184 – RAIs 15.2-25 and 15.2-26
		0000-0087-2920 – RAI 15.2-31
		0000-0087-3861 – RAI 15.2-32
		0000-0087-2927 – RAIs 15.2-35 and 15.2-37

Enclosure 1

MFN 08-618

Response to NRC Request for Additional Information Letter Nos. 152 and 168 Related to ESBWR Design Certification Application

Licensing Topical Reports (LTR) NEDO-33337 "Initial Core Transient Analysis" and NEDO-33338 "ESBWR Feedwater Temperature Operating Domain Transient and Accident Analysis"

RAI Numbers 15.2-14, 15.2-17, 15.2-19, 15.2-21, 15.2-25 15.2-26, 15.2-31, 15.2-32, 15.2-35 and 15.2-37

NRC RAI 15.2-14:

Verify that the transient analyses include the kinetics effects of the FWT controller responding to the transient conditions and maintaining the requested FWT.

GEH Response:

The Feedwater Control System (FWCS) is designed to maintain a specific feedwater temperature. The analyses of the Anticipated Operational Occurrences, Infrequent Events, and Special Events in NEDO-33338 do not model the FWCS because the kinetic effects are either bounded by operation without the feedwater temperature (FWT) control, or there is no appreciable effect.

The following limiting events are analyzed in NEDO-33338. The effect of the FW controller is discussed for each event:

- Loss of Feedwater Heating – The FWT decreases. If modeled, the FWCS will mitigate the severity of the transient by either increasing the feedwater temperature or slowing the rate of feedwater temperature decrease.
- Fast Closure of One Turbine Control Valve – There is no significant change in the FW temperature during the first 5 seconds when the critical power ratio (CPR) is the most important. FW controller has little to no effect.
- Generator Load Rejection with Turbine Bypass – FWT decreases during this event. The FWC would tend to increase the FW temperature and mitigate the severity of the event.
- Generator Load Rejection with a Single Failure in the Turbine Bypass System – Reactor scrams quickly and CPR immediately increases. Any changes in the FWT afterwards and corresponding FWCS response have no effect on the minimum CPR result.
- Inadvertent Isolation Condenser Initiation – The FWT changes very little. FW controller has little to no effect.
- Loss of Feedwater Heating with Selected Control Rod Run-In (SCRRI)/Select Rod Insertion (SRI) Failure – The FWT decreases. If modeled, the FWCS will mitigate the severity of the transient by either increasing the feedwater temperature or slowing the rate of feedwater temperature decrease unless the FWC failed. In this instance, the FW controller has no effect, and there is no change in the transient results.
- Generator Load Rejection with Total Turbine Bypass Failure – Reactor scrams quickly and CPR immediately increases. Any changes in the FWT afterwards and corresponding FWCS response have no effect on the minimum CPR result.
- Stuck-Open Safety Relief Valve (SRV) – Reactor scrams quickly and CPR immediately increases. Any changes in the FWT afterwards and corresponding FWCS response have no effect on the minimum CPR result.

- Overpressure protection (Main Steam Isolation Valve (MSIV) Closure with Flux Scram) – The feedwater system is conservatively assumed to trip at the initiation of the event. If assumed to operate, the feedwater will spray into the reactor vessel, condense the steam and terminate the pressurization.
- Anticipated Transients Without Scram (ATWS) – The FWT at the inlets to the reactor vessel do not change prior to approximately 25 seconds. After approximately 25 seconds, there is no feedwater flow. The FWC has no effect on the transient.
- Station Blackout – Reactor scrams quickly and CPR immediately increases. Any changes in the FWT afterwards and corresponding FWCS response have no effect on the minimum CPR result.

<u>LTR Impact</u>:

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

NRC RAI 15.2-17:

NEDO-33337, Table 1.2.1 - Add the new AOO, Control Rod Withdrawal Error During Power Operation, as shown in the DCD Table 15.0-2 mark up sent in letter dated December 12, 2007 (MFN-07-655). Incorporate this new AOO into the LTR.

GEH Response:

The new "Control Rod Withdrawal Error During Power Operation" and "Control Rod Withdrawal Error During Startup" events are added to the LTR.

Similar to the discussion in the DCD Tier 2 Subsection 15.3.9, for the Control Rod Withdrawal Error During Power Operation, the automated thermal limit monitor (ATLM) will block rod movement prior to exceeding minimum critical power ratio (MCPR) or maximum linear heat generation rate (MLHGR) operating limits for the initial core. Therefore, the analysis in the DCD is applicable.

The anticipated operational occurrence (AOO), Control Rod Withdrawal Error During Startup, was also added to the DCD Tier 2 Subsection 15.2.3.1. The AOO discussion refers to the bounding analysis in DCD Tier 2 Subsection 15.3.8, which is performed with an initial core, and therefore is applicable. The LTR will be updated to reflect this.

LTR Impact:

LTR NEDO-33337, Rev 1 will be revised as noted on the attached markup.

NRC RAI 15.2-19:

In the figures of DCD Tier 2, Chapter 4, Appendix A, the core flow appears to be a constant value of 78.508 Mlb/hr. In Table 2.2-1 of NEDO-33337, the core flow as function of exposure appears to be calculated based on power distributions, subcooling, and downcomer water level. What code was used to calculate the data in Table 2.2-1 of NEDO-33337? Are the conditions presented in the figures in the DCD Tier 2, Chapter 4, Appendix A representative of an ESBWR equilibrium core?

GEH Response:

DCD Tier 2, Chapter 4, Appendix A provides the control rod depletion results for the ESBWR equilibrium core design as determined by PANAC11. For that evaluation, a constant core flow of 78.508 Mlb/hr is used. For licensing basis rodded depletion evaluations, a best estimate constant core flow is assumed. This assumption is adequate in establishing appropriate cycle exposure dependent core conditions (e.g., power distribution, void history, isotopics, etc). The conditions and results presented in Chapter 4, Appendix A of the DCD are representative of an ESBWR equilibrium core loading GE14E fuel.

Please note that NEDO-33337 provides the transient analysis results for the ESBWR initial core design. The ESBWR initial core design is described in Reference 15.2-19-1. This reference provides the control rod depletion results based on the constant core flow provided in LTR NEDC-33326P, Section 3.1 table, "GE14E for ESBWR Initial Core Analysis Report" submitted via GNF letter FLN-2007-025 dated July 18, 2007.

Table 2.2-1 of NEDO-33337 provides the TRACG04 calculated core flow values at three exposure state points for the initial core design loading GE14E fuel. Some variation in this TRACG calculated exposure dependent core flow is expected relative to the constant core flow value used in the PANAC11 evaluation. If we compare the constant core flow value used in the PANAC11 evaluation to an average core flow value from Table 2.2-1, the variation is less the 0.5%. Such a difference is insignificant in establishing the cycle exposure dependent core conditions as mentioned above.

References

15.2-19-1 "GE14E For ESBWR Initial Core Design Report," NEDO-33326, Rev. 0, July 2007.

LTR Impact:

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

NRC RAI 15.2-21:

Figure 2.2-3 of NEDO-33337 and Figure 4D-3 of DCD Tier 2, Rev 4 appear to show conflicting trends of axial power shape with respect to exposure. In the DCD figure, the MOC condition has higher power on the top of the core and lower power in the bottom of the core than the BOC condition. In the NEDO-33337 figure, the shifts in power appear to be reversed (lower power at the core top in MOC than in BOC). In addition, the axial peaking factors appear to have changed significantly (~1.7 in node 4 for NEDO-33337, and ~1.4 in node 3 for DCD). Please explain the differences.

GEH Response:

The request is broken down into distinct items for clarity.

1. Clarify the difference observed in the axial power shape shifting with respect to exposure between Figure 2.2-3 of NEDO-33337 and Figure 4D-3 of DCD Tier 2, Rev 4.
2. Explain the difference in the axial peaking factors in aforementioned figures.

Response Part 1

Axial Power Shapes (APS) are affected by a number of parameters, control rod pattern, bundle design, core loading pattern, previous cycle operating history, etc., during a BWR operating cycle. Figure 4D-3 of DCD Tier 2, Rev 4 presents the core average APS at three cycle points (BOC, MOC and EOC) for the ESBWR equilibrium core design. Figure 2.2-3 of NEDO-33337, Rev 0 APS for initial core gives three cycle points. The differences observed in APS throughout the cycle for the initial core and equilibrium core are expected and are a result of two different core designs where the differences in cycle energy, cycle operation strategy, bundle design and operating limits all contribute to the behavior of the power shape during the cycle. The reported APS for each cycle is representative for their respective application with both cycles maintain appropriate limits.

Please note that in Figure 4D-3 of DCD Tier 2, Rev 5, the MOC condition is updated as a result of searching for the limiting exposure level in terms of stability performance. The updated MOC condition is established at the Peak Hot Excess (PHE) reactivity point. Therefore, Figure 4D-3 is updated in DCD Tier 2, Rev 5 so that MOC condition at PHE has the highest axial peaking comparing with BOC and EOC conditions.

Response Part 2

In Figure 4D-3 of DCD Tier 2, Rev 5, the highest axial peaking factor shows ~1.4 at the 0.18 relative elevation while the highest axial peaking factor in Figure 2.2-3 of NEDO-33337 gives ~1.6 at about the same relative elevation. The difference in the magnitude of axial peaking factors is expected for two different cycles. As mentioned in Part 1 of the response, the APS is affected by a number of parameters. Initial core cycle in NEDO-33337 is a rather unique cycle relative to an equilibrium cycle in the DCD since all fuel is fresh and APS is governed by rod pattern and axial varying gadolinia. There are the same affects in the equilibrium cycle but in addition, only 40% of the fuel in the EC is fresh; therefore, the APS is also affected by the exposure distribution from the remaining 60% of the fuel exposed in previous cycles. Variations in axial power are to be expected for these two different cycles. However, there are basic similarities between the APS for these two cycles because of general BWR operating strategy; i.e., bottom peaked at BOC and top peaked at EOC, the classic spectral shift behavior that is applied in BWRs to maximize exposure capability.

LTR Impact:

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

NRC RAI 15.2-25:

NEDO-33337, Section 2.3.1.1.1 states: "Under circumstances in which no SCRRI/SRI pattern is defined, the power would rise to the simulated thermal power trip scram setpoint." This implies that there is no credit taken for SCRRI for the initial core analysis. What is the difference between this event and Loss of Feedwater Heating without SCRRI?

GEH Response:

The Loss of Feedwater Heating (LOFWH) event in the LTR will be updated to match the same event found in Chapter 15.2 of DCD Tier 2, Revision 5. The attached markup also reflects changes from DCD Revision 5. This analysis uses Select Rod Insertion (SRI) rods but no Selected Control Rod Run-In (SCRRI) rods. SRI rods perform a similar function to SCRRI rods, to reduce the core power and limit the change to Minimum Critical Power Ration (MCPR). SRI rods are used when the slower moving SCRRI rods would not adequately affect the top peaked power profile. SCRRI rods may be added to reduce power further (as needed), but are not required to demonstrate acceptable CPR performance.

See GEH response to RAI 4.6-28 Supplement 1, contained in GEH Letter MFN 08-415 dated April 24, 2008 for further SCRRI/SRI discussion.

LTR Impact:

LTR NEDO-33337, Rev 0 will be revised as noted on the attached markup.

NRC RAI 15.2-26:

NEDO-33337, Section 2.3.1.1.1 states that no single operator error or equipment failure shall cause a loss of more than 55.6 °C (100 °F) FW heating. However, Table 2.3-5 shows the analysis assumes a change of 39 °C (70 °F). Why is 70 °F assumed instead of 100 °F?

GEH Response:

A feedwater (FW) temperature reduction of 39°C (70°F) is assumed because it increases the power to the simulated thermal power trip setpoint (115%) without initiating the scram. Any additional FW temperature reduction results in a scram, and terminates the event. The LOFWH event in the LTR will be updated to match the event found in Chapter 15.2 of the DCD Tier 2, Revision 5, including credit for Selected Control Rod Run-In (SCRRI)/Select Rod Insertion (SRI) function and a FW temperature reduction of 55.6°C (100°F). The attached markup also reflects changes from DCD Revision 5.

LTR Impact:

LTR NEDO-33337, Rev 0 will be revised as noted on the attached markup.

NRC RAI 15.2-31:

NEDO-33337, Section 2.3.2.4.2, Systems Operation - In the DCD Section 15.2.2.4.2, credit is taken for RPS. Why is credit not taken for RPS in the initial core transient analyses?

GEH Response:

The ESBWR design includes full turbine bypass capability. Following a generator load rejection or turbine trip, Reactor Protection System (RPS) senses successful turbine bypass valve operation and inhibits the reactor scram (DCD Tier 2 Subsection 7.2.1.2.4.2). The RPS scram is inhibited in the turbine trip with turbine bypass event, (both in the DCD and in NEDO-33337) due to successful operation of the turbine bypass. Because there is no scram in the event, there is no need to state that RPS is credited; therefore, the statement "Credit is taken for successful operation of the RPS" has been deleted from DCD Tier 2 Subsection 15.2.2.4.2, Revision 5. There is no difference, with respect to the RPS assumption, between the DCD and the LTR turbine trip with turbine bypass analysis.

LTR Impact:

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

NRC RAI 15.2-32:

NEDO-33337, Section 2.3.6 states, "The Potentially limiting events that establish the CPR operating limit are identified below." However, there are no events listed. What are the potentially limiting events used to establish the CPR operating limit for the initial core?

GEH Response:

The potentially limiting anticipated operational occurrences (AOO) events that establish the critical power ratio (CPR) operating limit are given in Subsection 15.2.6 of DCD Tier 2, Rev. 5, are:

- Loss of Feedwater Heating;
- Closure of One Turbine Control Valve;
- Generator Load Rejection with Turbine Bypass;
- Generator Load Rejection with a Single Failure in the Turbine Bypass System; and
- Inadvertent Isolation Condenser Initiation.

LTR Impact:

Changes to the first paragraph of Section 2.3.6 of the LTR (NEDO-33337) will be made in response to this RAI as follows:

2.3.6 AOO Analysis Summary

The results of the system response analyses are presented in Table 2.3-4. Based on these results, the limiting AOO events have been identified. The potentially limiting events that establish the CPR operating limit are identified in Subsection 15.2.6 of DCD Tier 2, Chapter 15. System response analyses bounding operation in the feedwater temperature operating domain are documented in Reference 2.3-4.

NRC RAI 15.2-35:

NEDO-33337, Figure 2.3-9b, Confirm whether the IC flow shown is only for one IC or the total IC flow.

GEH Response:

The isolation condenser (IC) flow shown on Figure 2.3-9b of NEDO-33337 is for the operation of 3 out of 4 ICs. The GEH analysis is based upon one of the ICs inoperable. The IC flow shown in Figure 2.3-9b is the combined flow of the 3 operating ICs.

LTR Impact:

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

NRC RAI 15.2-37:

NEDO-33337, Section 2.4.4.5 states: "Because the delta CPR/ICPR for this event is bounded by the limiting AOO value, no fuel failures are expected; therefore, no radiological analysis is required." Why is the AOO value applied for an Infrequent Event? DCD Tier 2, Section 15.3.4.5 states: "A radiological analysis was performed for an event where 1000 fuel rods fail as a result of entering boiling transition." Why are the results different? Please clarify.

GEH Response:

The anticipated operational occurrence (AOO) value is applied for an infrequent event because the AOO value bounds the infrequent event value. An AOO and an infrequent event are abnormal events where an AOO is an abnormal event that has a higher event probability than an infrequent event.

DCD Tier 2 Subsection 15.3.4.5 no longer states: "A radiological analysis was performed for an event where 1000 fuel rods fail as a result of entering boiling transition." Subsection 15.3.4.5 currently states: "Because this event does not result in any fuel failures or any release of primary coolant to the environment, there is no radiological consequences associated with this event."

The results are similar with the only difference being that the DCD analysis considers an equilibrium core while the LTR analysis considers an initial core.

LTR Impact:

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

Enclosure 2

MFN 08-618

LTRs NEDO-33337 and NEDO-33338 Markups

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 applies to the initial core.

2.3.3.2 Control Rod Withdrawal Error During Power Operation

Analysis in the DCD applies to the initial core.

2.3.4 Increase in Reactor Coolant Inventory

Table 2.3-4a

Results Summary of Anticipated Operational Occurrence Events (1)

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. Simulated Thermal Power ~~Core Average Surface Heat Flux~~, % of Initial	ΔCPR/ICPR or Minimum Water Level (m [ft] over TAF)
2.3.2.1	Loss of Feedwater Heating	116	7.11 (1031)	7.24 (1050)	7.06 (1024)	119	0.09
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 (17.6 ft)
2.3.5.3	Loss of Feedwater Flow	100	7.08 (1027)	7.21 (1046)	7.04 (1021)	100	5.28m (17.3 ft)

(1) This table summarizes the events calculated with the TRACG code. Table 15.2-4b contains the summary of the remaining AOO Events.

Table 2.3-4b

Results Summary of Anticipated Operational Occurrence Events

Data in the DCD applies to the initial core.

2.3.1.1 **Loss Of Feedwater Heating**

2.3.1.1.1 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 shall cause 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 Reference 2.3-5. This prevents the reactor from violating any thermal limits. These functions are also collectively referred to as SCRRI/SRI.~~The Feedwater Control System (FWCS) includes logic to mitigate the effects of a loss of FW heating capability. The system is constantly monitoring the actual FW temperature and comparing it with a reference temperature. When a loss of FW heating is detected [i.e., when the difference between the actual and reference temperatures exceeds a ΔT setpoint], the FWCS sends an alarm to the operator and sends a signal to the Non-Safety Related Distributed Control and Information System (N-DCIS) to initiate the Selected Rods Insertion (SRI) and selected control rods run-in (SCRRI) function to automatically reduce the reactor power. 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 SRI/SCRRI is able to suppress the neutron power increase and ensure 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. ~~Under circumstances in which no SCRRI/SRI rod pattern is defined the power would rise to the to the Simulated Thermal Power Trip (STPT) scram setpoint. This scenario is analyzed below.~~ 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 no SCRRI rods were assigned for this rod pattern. SCRRI rods were not defined in this rod pattern; they were not required to show acceptable CPR results.

2.3.1.1.2 ***Sequence of Events and Systems Operation***

Sequence of Events

Table 2.3-5 lists the sequence of events for Figure 2.3-1.

~~For conservatism, the scram is not credited in this analysis. As soon as possible, the operator should verify that no operating limits are being exceeded. Also, the operator should determine the cause of failure prior to returning the system to normal.~~There is no scram during this event. There is no operator action required to mitigate the event.

Systems Operation

In establishing the expected sequence of events and simulating the plant performance~~, it was assumed that normal functioning occurred in~~ the plant instrumentation and controls, plant protection and reactor protection systems are assumed to function normally. A failure of a single Hydraulic Control Unit (HCU) is assumed.

2.3.1.1.3 ***Core and System Performance***

Input Parameters and Initial Conditions

The event is simulated by ~~initiating a reduction~~programming a change in ~~feedwater~~FW ~~temperature that results in power reaching the STPT, but does not initiate a scram. This bounds the case in which the reduction in feedwater temperature is larger and results in an STPT scram.~~enthalpy to the assumed loss of feedwater heating, shown in Table 2.3-5.

Results

Because the power increase during this event is controlled by the SCRRI/SRI insertion, the reduction of the MCPR is very small and is turned around when the SCRRI/SRI function takes effect. The results are summarized in Table 2.3-4.

No scram is assumed in this analysis. Nuclear system pressure does not significantly change, and consequently, the RCPB is not threatened.

2.3.1.1.4 ***Barrier Performance***

As noted previously, the effects of this event do not result in any temperature or pressure transient in excess of the criteria for which the fuel ~~(as stated in the Section 8.3 of Reference 2.3-1 regarding the centerline melt protection discussion with the TRACG methodology),~~ pressure vessel or containment are designed. Therefore, these barriers maintain their integrity and function as designed.

2.3.1.1.5 ***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.

Table 2.3-4

Results Summary of Anticipated Operational Occurrence Events

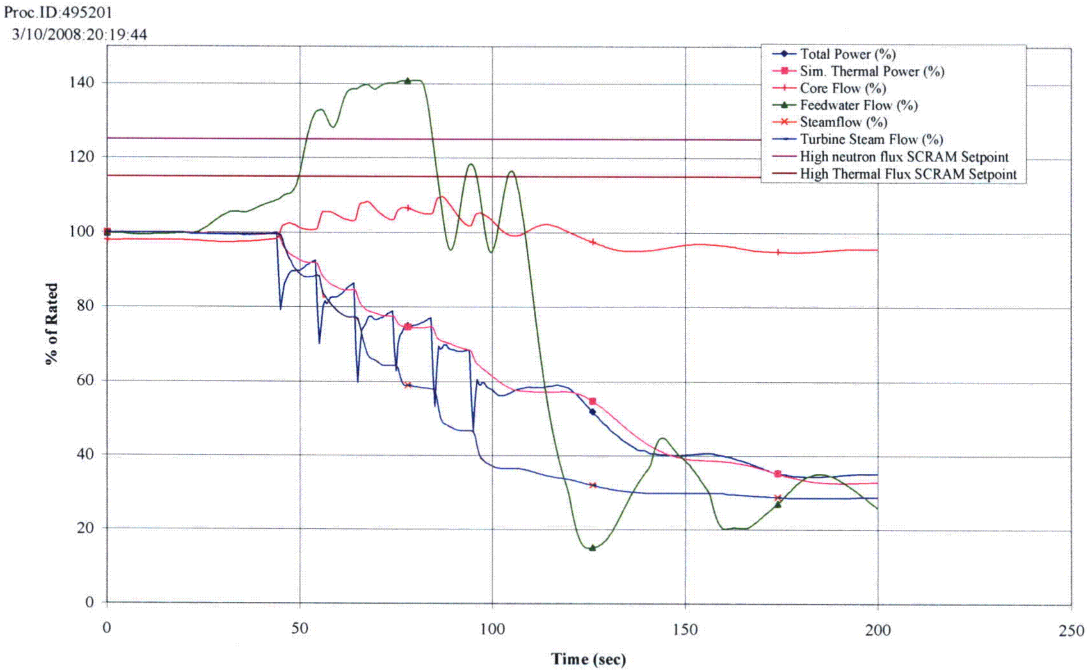
Sub-section I.D.	Description	Max. Neutron Flux, % NBR	Max. Dome Pressure, MPaG (psig)	Max. Vessel Bottom Pressure, MPaG (psig)	Max. Steamline Pressure, MPaG (psig)	Max. Core Average Surface Heat Flux, % of Initial	ΔCPR/ICPR or Minimum Water Level (m over TAF)
2.3.2.1	Loss of Feedwater Heating	~~116~~100	~~7.11~~7.08 (~~1031~~1026)	~~7.24~~7.21 (~~1050~~1046)	7.04~~6~~ (~~1024~~1021)	~~119~~100	~~0.09~~<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

Table 2.3-5

Sequence of Events for Loss of Feedwater Heating

Time (s)	Event
0	Initiate a ~~39~~55.6°C (~~70~~100°F) temperature reduction in the FW system.
23~~22~~ (est)	RC&IS initiates Selected Rod Insertion plus Selected Control Rod Run-In (SCRRI/SRI). No rods are assigned for this exposure point.
25 (est.)	Initial effect of unheated FW starts to raise core power level.
~~≈150.0~~	~~The STPT setpoint (115%) is reached, the activation of Scram is not credited~~
24	First SRI group inserts (one HCU, 2 control rods, fails to actuate) and SCRRI starts
34	Second SRI group inserts
44	Third SRI group inserts
54	Fourth SRI group inserts
57	Steam flow below 60% rated
64	Fifth SRI group inserts
74	Sixth SRI group inserts
82	Power below 60% rated
300 (est.)	Reactor variables settle into new steady state.

* See Figure 2.3-1. This figure has 20 s of steady state, a time of 0 s on the table corresponds to 20 s on the figure.



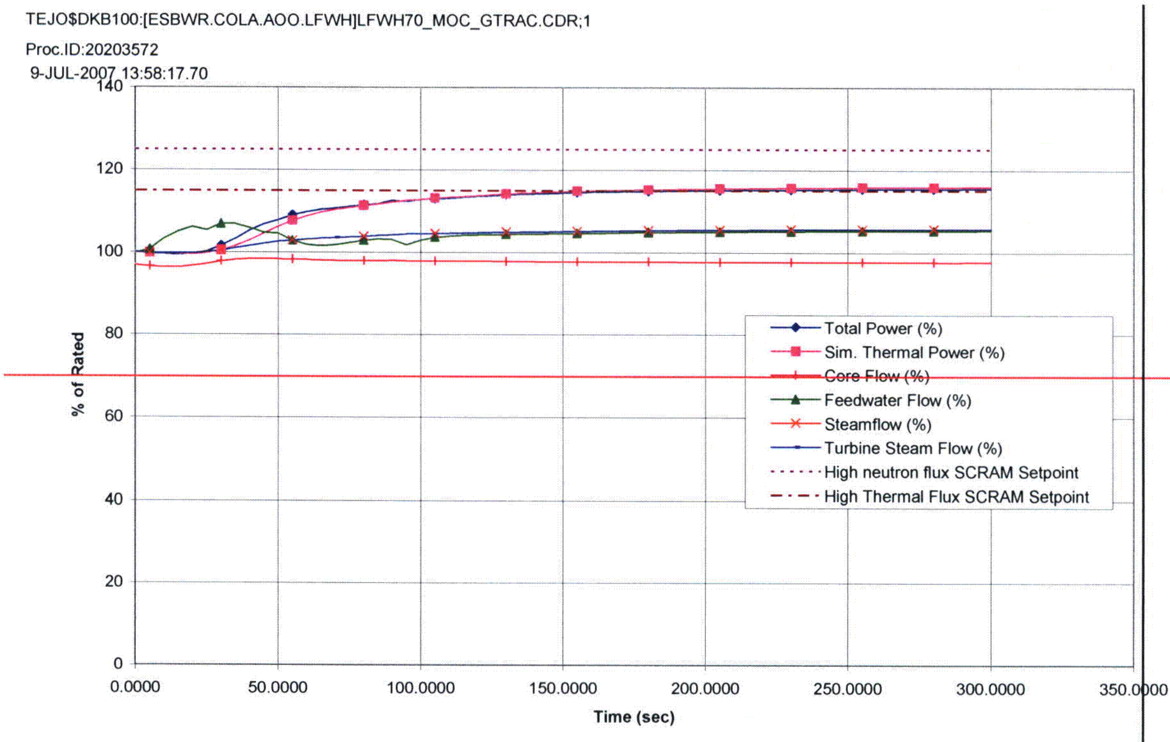
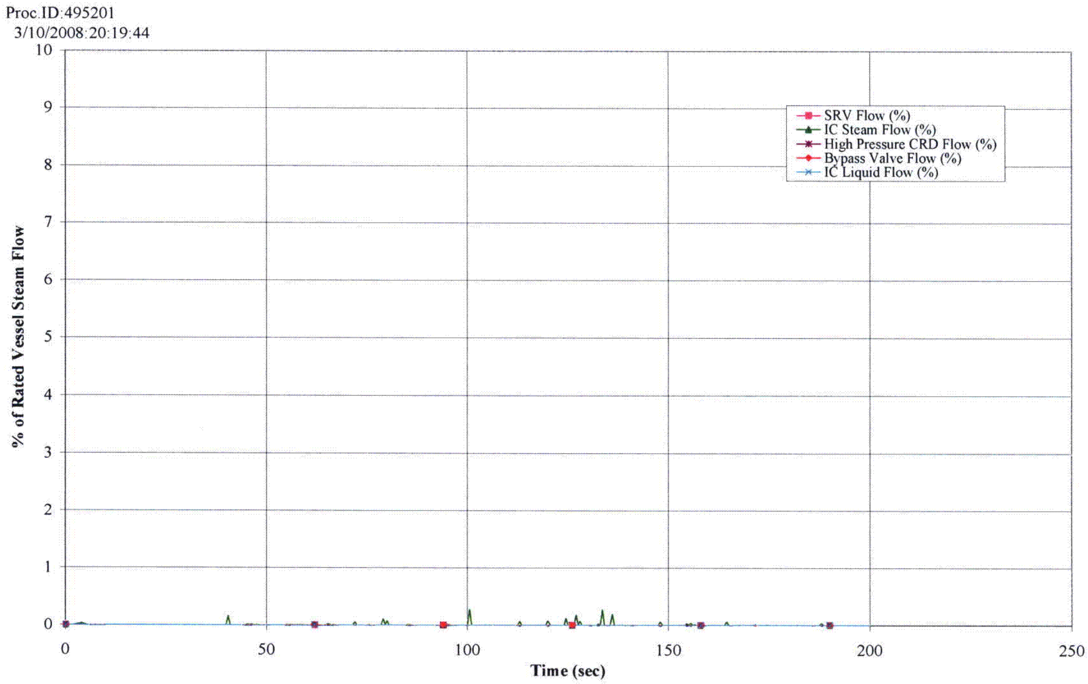


Figure 2.3-1a. Loss of Feedwater Heating



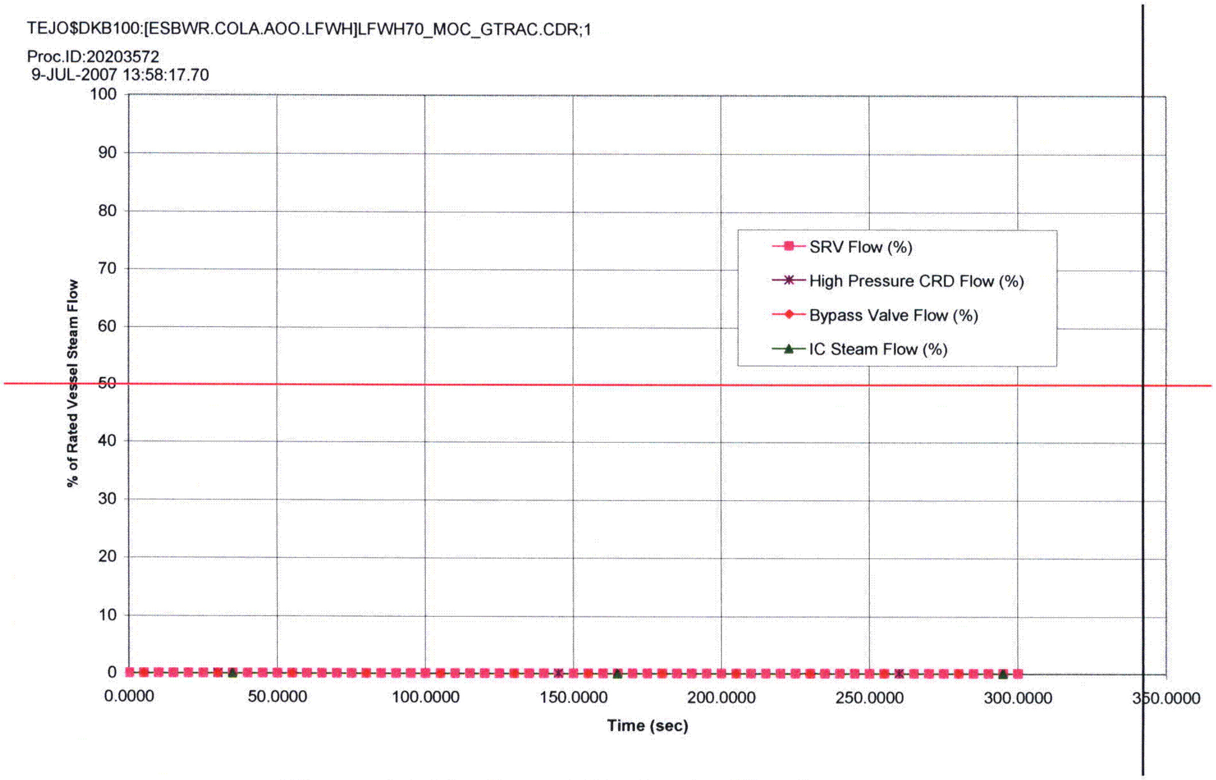
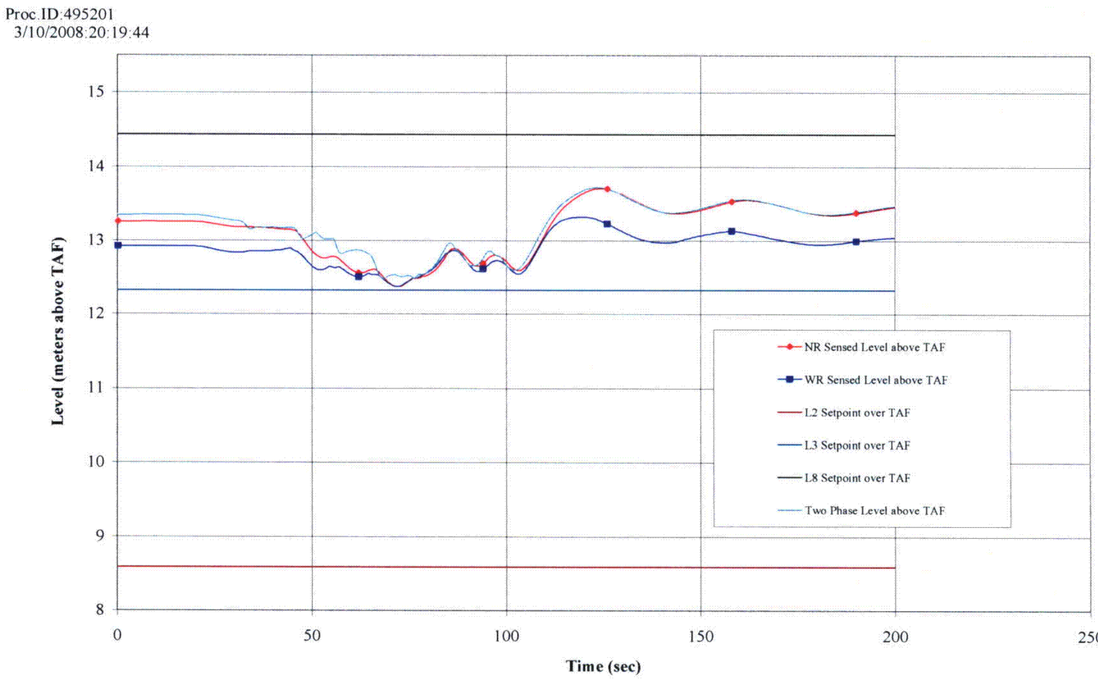


Figure 2.3-1b. Loss of Feedwater Heating



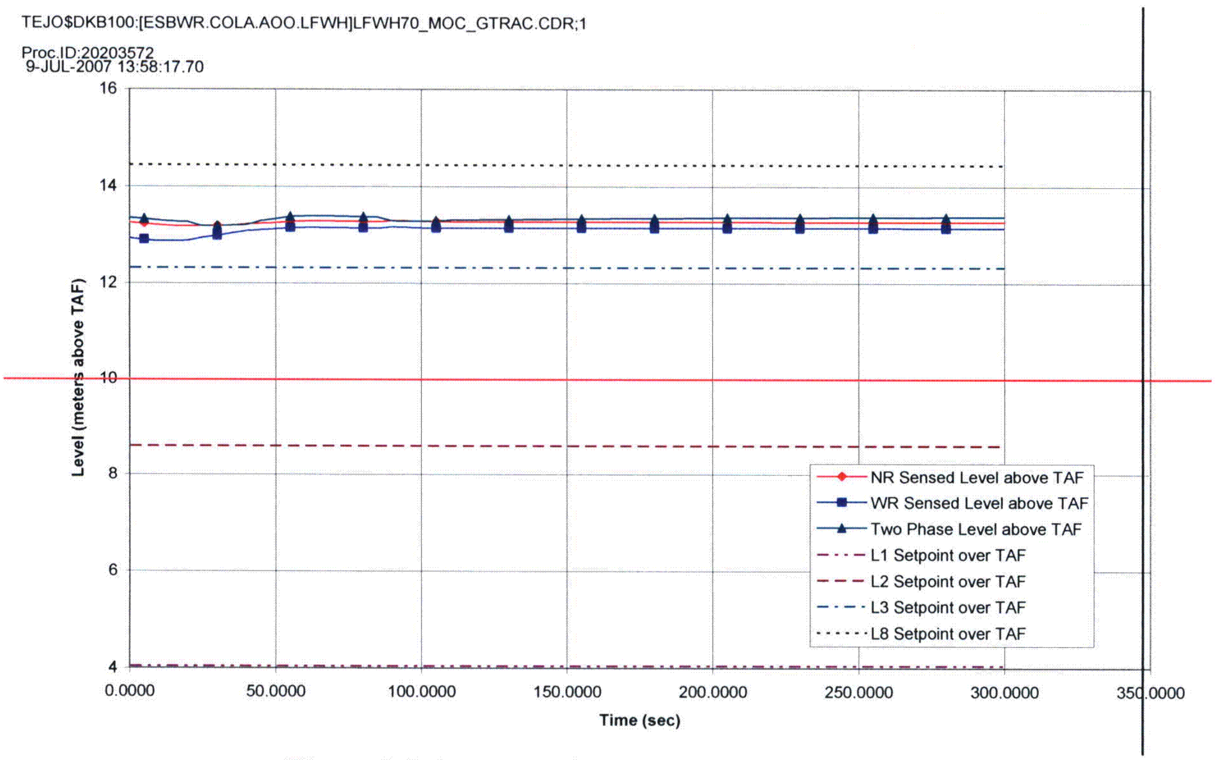
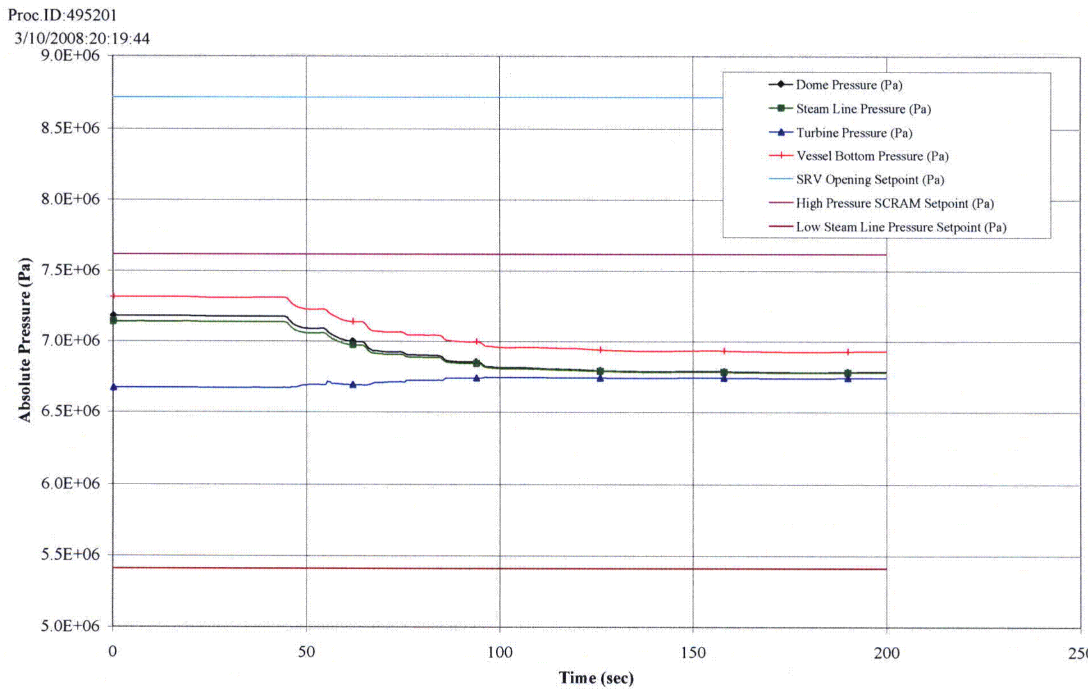


Figure 2.3-1c. Loss of Feedwater Heating



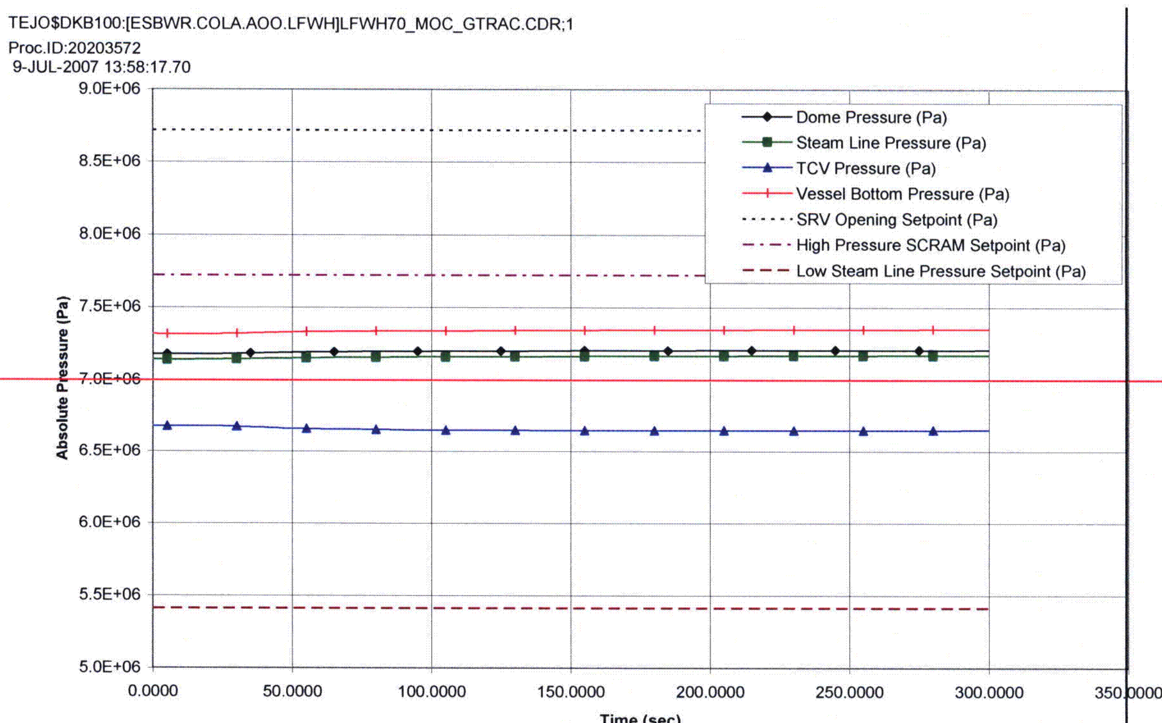
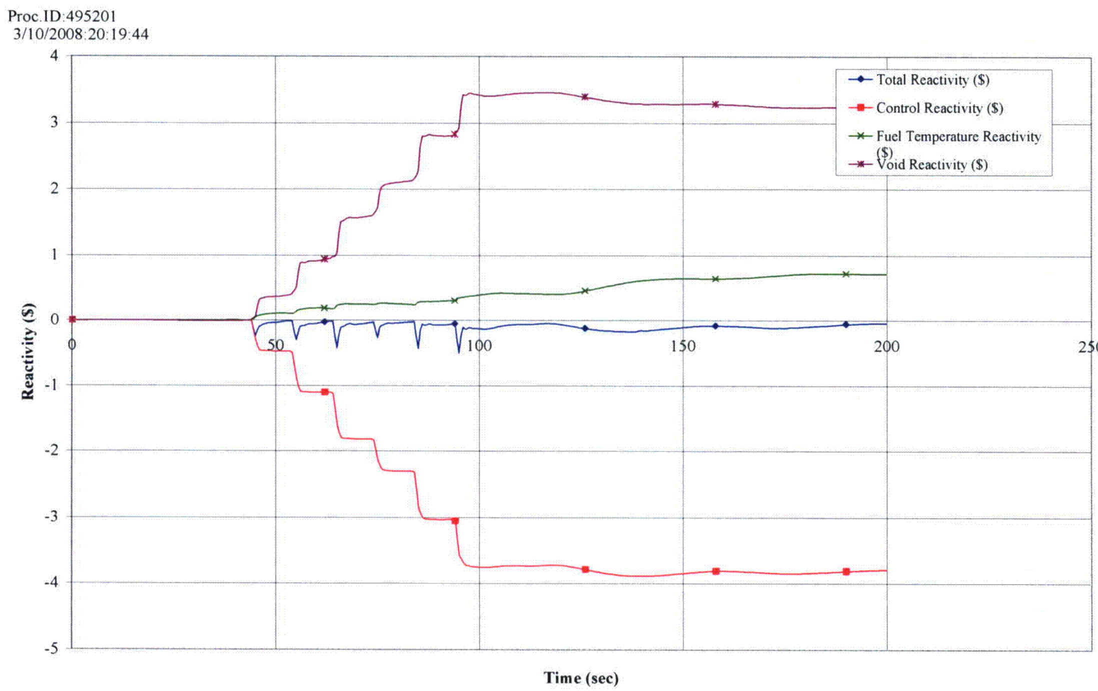


Figure 2.3-1d. Loss of Feedwater Heating



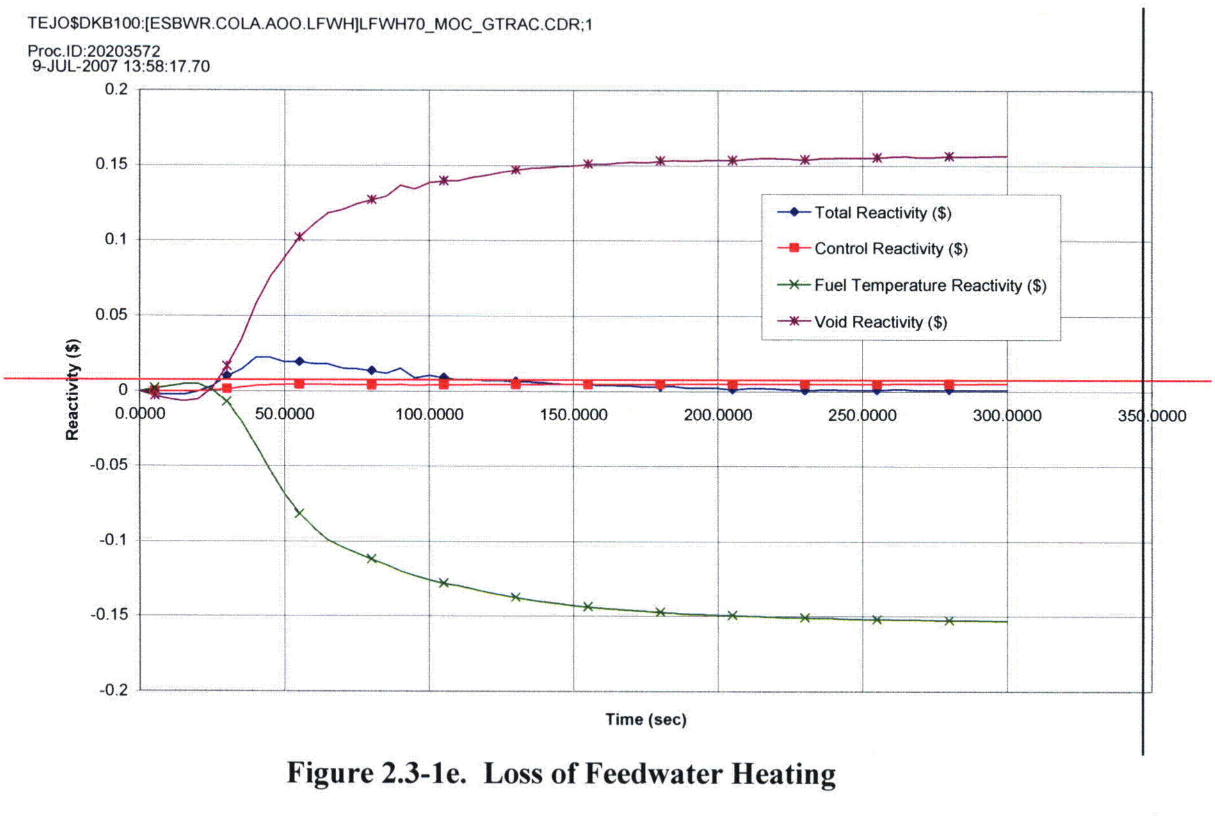
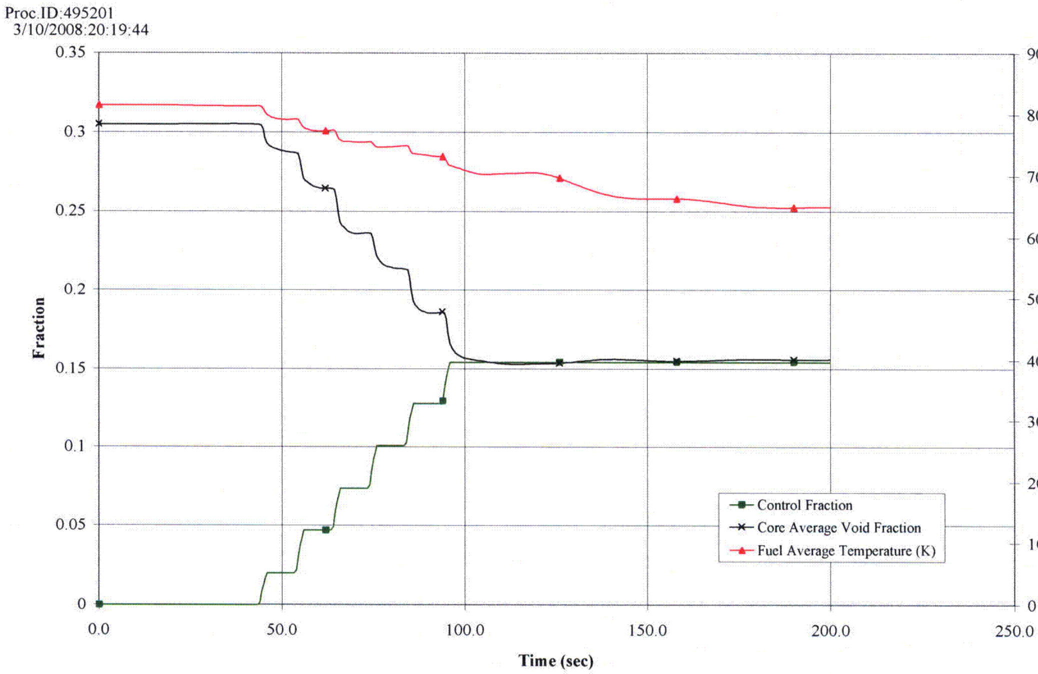


Figure 2.3-1e. Loss of Feedwater Heating



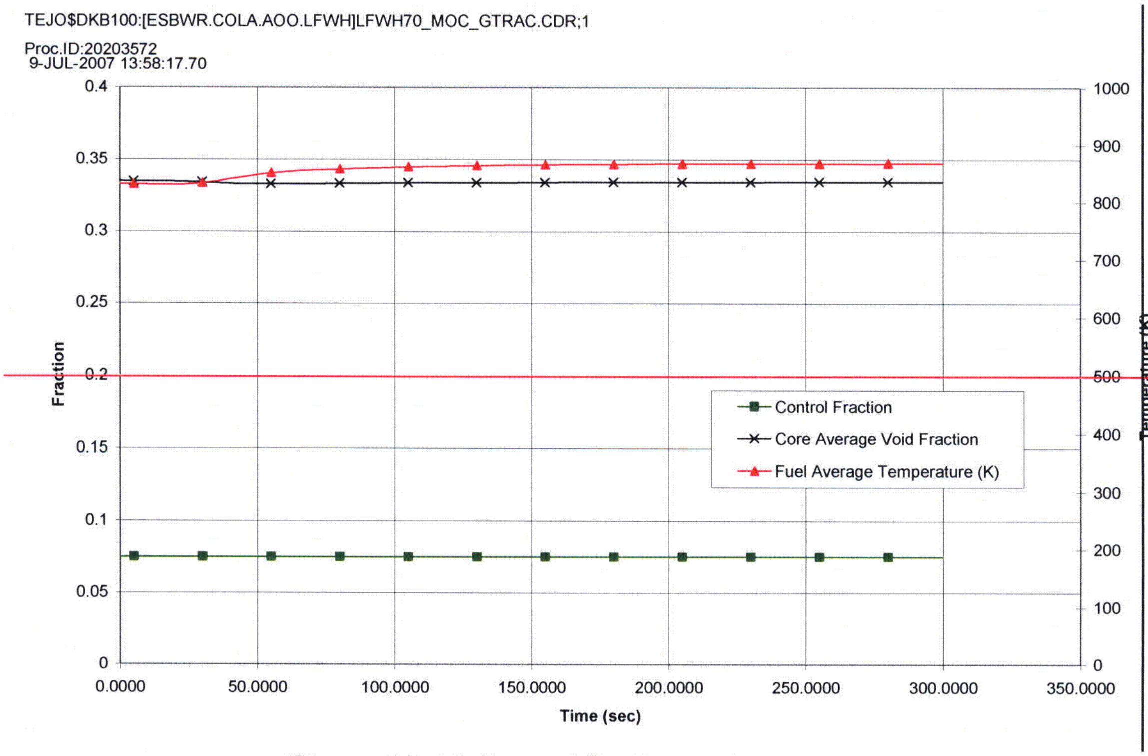


Figure 2.3-1f. Loss of Feedwater Heating

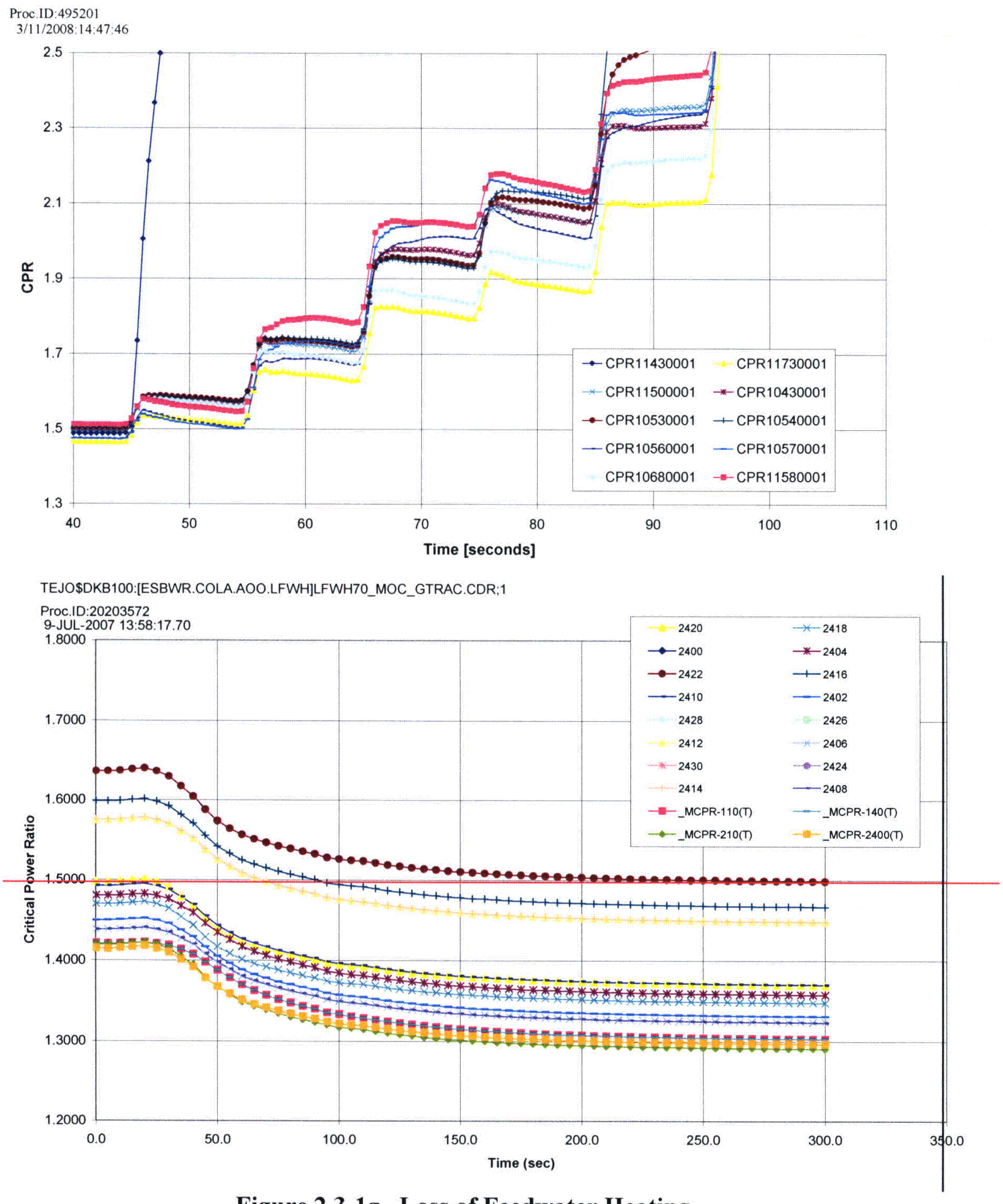


Figure 2.3-1g. Loss of Feedwater Heating

2.3-7 References

2.3-1 GE Nuclear Energy, "TRACG Application for Anticipated Operational Occurrences Transient Analysis" NEDE-32906P-A, Revision 1, April 2003.

2.3-2 Deleted

2.3-3 Global Nuclear Fuel, "ESBWR Initial Core Nuclear Design Report", NEDC-33326-P, Class III (Proprietary), Revision 0, July 2007, NEDO-33326, Class I (Non-proprietary), Revision 0, July 2007.

2.3-4 GE-Hitachi Nuclear Energy, "ESBWR Feedwater Temperature Operating Domain Transient and Accident Analysis", NEDO-33338 Class I, Revision 0, Scheduled October 2007.

2.3-5 GE Nuclear Energy, "ESBWR Design Control Document" 26A6642, Revision 5~~4~~, ~~September 2007~~May 2008.

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~~<0.01
	IC	100~~116~~	7.08 (1026)~~7.11 (1031)~~	7.21 (1046)~~7.24 (1050)~~	7.04 (1021)~~7.06 (1024)~~	100~~119~~	<0.01~~0.09~~