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Subject: **Response to Portion of NRC Request for Additional Information Letter No. 152 - Related to ESBWR Design Certification Application – RAI Number 4.3-13**

The purpose of this letter is to submit the GE Hitachi Nuclear Energy (GEH) response to the U.S. Nuclear Regulatory Commission (NRC) Request for Additional Information (RAI) sent by the Reference 1 NRC letter. GEH response to RAI Number 4.3-13 is addressed in Enclosures 1 and 2.

Verified DCD changes associated with this RAI response are identified in the enclosed DCD markups by enclosing the text within a black box. The marked-up pages may contain unverified changes in addition to the verified changes resulting from this RAI response. Other changes shown in the markups may not be fully developed and approved for inclusion in DCD Revision 5.

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

Sincerely,

James C. Kinsey

James C. Kinsey
Vice President, ESBWR Licensing

D068
NRO

Reference:

1. MFN 08-121, Letter from U.S. Nuclear Regulatory Commission to Robert E. Brown, *Request for Additional Information Letter No. 152 Related to the ESBWR Design Certification Application*, dated February 11, 2008

Enclosures:

1. MFN 08-387 – Response to Portion of NRC Request for Additional Information Letter No. 152 - Related to ESBWR Design Certification Application – RAI Number 4.3-13
2. MFN 08-387 – Response to Portion of NRC Request for Additional Information 7etter No. 152 - Related to ESBWR Design Certification Application – DCD Markups from the Response to RAI Number 4.3-13

cc:	AE Cubbage	USNRC (with enclosure)
	GB Stramback	GEH/San Jose (with enclosure)
	RE Brown	GEH/Wilmington (with enclosure)
	DH Hinds	GEH/Wilmington (with enclosure)
	eDRF	0000-0082-7084

Enclosure 1

MFN 08-387

Response to Portion of NRC Request for

Additional Information Letter No. 152

Related to ESBWR Design Certification Application

RAI Number 4.3-13

NRC RAI 4.3-13

(New RAI from LTR NEDO-33338)

Classify the Feedwater Temperature Controller Failure Event

Classify the failure of the feedwater temperature control system to minimum temperature event (e.g., infrequent event). Justify the classification.

GEH Response

The event "Feedwater Controller Failure – Minimum Temperature Demand" was classified as infrequent. Subsection 15.3.1.1 of DCD Tier 2 Revision 4 is being revised to acknowledge this new event as part of the loss of feedwater heating analysis. This revision is shown in the DCD markups in Enclosure 2.

The classification as "Infrequent Event" of the "Feedwater Controller Failure – Minimum Temperature Demand" is justified by the event's low frequency of occurrence (less than once in 100 years). Subsection 15A.3.5 of DCD Tier 2 Revision 4 is being revised to include the reliability analysis determining the low frequency of the event.

Revision 4 of DCD Tier 2 analyzed the event "Feedwater Controller Failure – Maximum Demand." This name was changed to "Feedwater Controller Failure – Maximum Flow Demand" in order to differentiate from the failure of the new feedwater temperature controller to minimum temperature demand.

DCD Impact

The following DCD Tier 2 sections, tables, and figures will be revised in Revision 5 as shown in Enclosure 2:

Table 15.0-2	Subsection 15.2.1.1.1	Subsection 15A.3.5
Table 15.0-8	Table 15.2-1	Table 15A-3
Table 15.1-3	Table 15.2-4	
Table 15.1-5	Subsection 15.3.1.1	
Table 15.1-6	Subsection 15.3.2	
Table 15.1-7	Table 15.3-1a	
Figure 15.1-18	Table 15.3-3	
	Figures 15.3-2a through 15.3-2g	

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

Enclosure 2

MFN 08-387

Response to Portion of NRC Request for

Additional Information Letter No. 152

Related to ESBWR Design Certification Application

DCD Markups from the Response to RAI Number 4.3-13

Verified DCD changes associated with this RAI response are identified in the enclosed DCD markups by enclosing the text within a black box. The marked-up pages may contain unverified changes in addition to the verified changes resulting from this RAI response. Other changes shown in the markups may not be fully developed and approved for inclusion in DCD Revision 5.

Table 15.0-2

ESBWR Abnormal Event Classifications

Abnormal Event	Event Classification	Relevant SRP(s)
Loss of Feedwater Heating	AOO	15.1.1 - 4
Closure of One Turbine Control Valve	AOO	15.2.1 – 5
Generator Load Rejection with Turbine Bypass	AOO	15.2.1 – 5
Generator Load Rejection with a Single Failure in the Turbine Bypass System	AOO*	15.2.1 – 5
Turbine Trip with Turbine Bypass	AOO	15.2.1 – 5
Turbine Trip with a Single Failure in the Turbine Bypass System	AOO*	15.2.1 – 5
Closure of One Main Steamline Isolation Valve	AOO	15.2.1 – 5
Closure of All Main Steamline Isolation Valves	AOO	15.2.1 – 5
Loss of Condenser Vacuum	AOO	15.2.1 – 5
Loss of Shutdown Cooling Function of RWCU/SDC	AOO	15.2.1 – 5
Control Rod Withdrawal Error During Power Operation	AOO	15.4.2
Control Rod Withdrawal Error During Startup	AOO	15.4.1
Inadvertent Isolation Condenser Initiation	AOO	15.1.1 – 4
Runout of One Feedwater Pump	AOO	15.1.1 – 4
Opening of One Turbine Control or Bypass Valve	AOO	15.1.1 – 4
Loss of Unit Auxiliary Transformer **	AOO	15.2.6
Loss of Grid Connection **	AOO	15.2.6
Loss of All Feedwater Flow	AOO	15.2.7
Loss of Feedwater Heating With Failure of Selected Control Rod Run-In	Infrequent Event	15.1.1 – 4
Feedwater Controller Failure – Maximum Flow Demand	Infrequent Event	15.1.1 – 4
Pressure Regulator Failure - Opening of All Turbine Control and Bypass Valves	Infrequent Event	15.1.1 – 4
Pressure Regulator Failure – Closure of All Turbine Control and Bypass Valves	Infrequent Event	15.1.1 – 4
Generator Load Rejection with Total Turbine Bypass Failure	Infrequent Event	15.2.1-5

Table 15.0-8

ESBWR Safety Analysis Codes

Safety Analysis	Analysis Code
Stability Evaluation (Chapter 4)	TRACG04[1]
Reactor Building Compartment Pressurization Analysis (Chapter 6)	CONTAIN 2.0
Loss of Feedwater Heating	TRACG04[1]
Closure of One Turbine Control Valve	TRACG04[1]
Generator Load Rejection with Turbine Bypass	TRACG04[1]
Generator Load Rejection with a Single Failure in the Turbine Bypass System	TRACG04[1]
Turbine Trip with Turbine Bypass	TRACG04[1]
Turbine Trip with a Single Failure in the Turbine Bypass System	TRACG04[1]
Closure of One Main Steamline Isolation Valve	TRACG04[1]
Closure of All Main Steamline Isolation Valves	TRACG04[1]
Loss of Condenser Vacuum	TRACG04[1]
Loss of Shutdown Cooling Function of RWCU/SDC	N/A
Control Rod Withdrawal Error During Power Operation	N/A
Control Rod Withdrawal Error During Startup	N/A
Inadvertent Isolation Condenser Initiation	TRACG04[1]
Runout of One Feedwater Pump	TRACG04[1]
Opening of One Turbine Control or Bypass Valve	TRACG04[1]
Loss of Non-Emergency AC Power to Station Auxiliaries	TRACG04[1]
Loss of All Feedwater Flow	TRACG04[1]
Loss of Feedwater Heating With Failure of Selected Control Rod Run-In	TRACG04[1] / RADTRAD 3.03
Feedwater Controller Failure – Maximum Flow Demand	TRACG04[1]
Pressure Regulator Failure - Opening of All Turbine Control and Bypass Valves	TRACG04[1]
Pressure Regulator Failure – Closure of All Turbine Control and Bypass Valves	TRACG04[1]
Generator Load Rejection with Total Turbine Bypass Failure	TRACG04[1]
Turbine Trip with Total Turbine Bypass Failure	TRACG04[1]
Control Rod Withdrawal Error During Refueling	N/A
Control Rod Withdrawal Error During Startup With Failure of Control Rod Block	PANAC11

Table 15.1-3

ESBWR Events Associated With Operating Modes

Abnormal Event	Applicable Operating Mode(s)
Loss of Feedwater Heating	1 & 2
Closure of One Turbine Control Valve	1 & 2
Generator Load Rejection with Turbine Bypass	1 & 2
Generator Load Rejection with a Single Failure in the Turbine Bypass System	1 & 2
Turbine Trip with Turbine Bypass	1 & 2
Turbine Trip with a Single Failure in the Turbine Bypass System	1 & 2
Closure of One Main Steamline Isolation Valve	1 - 4
Closure of All Main Steamline Isolation Valves	1 - 4
Loss of Condenser Vacuum	1 - 4
Loss of Shutdown Cooling Function of RWCU/SDC System	2 - 6 & 6S
Control Rod Withdrawal Error During Power Operation	1 & 2
Control Rod Withdrawal Error During Startup	2 - 5 & 6S
Inadvertent Isolation Condenser Initiation	1 - 6 & 6S
Runout of One Feedwater Pump	1 & 2
Opening of One Turbine Control or Bypass Valve	1 - 4
Loss of Non-Emergency AC Power to Station Auxiliaries	1 - 6 & 6S
Loss of All Feedwater Flow	1 & 2
Loss of Feedwater Heating With Failure of Selected Control Rod Run-In	1 & 2
Feedwater Controller Failure – Maximum Flow Demand	1 & 2
Pressure Regulator Failure Opening of All Turbine Control and Bypass Valves	1 - 4
Pressure Regulator Failure – Closure of All Turbine Control and Bypass Valves	1 & 2
Generator Load Rejection with Total Turbine Bypass Failure	1 & 2
Turbine Trip with Total Turbine Bypass Failure	1 & 2

Table 15.1-5

NSOA System Event Matrix

	SRV – Safety Relief Mode	SRV – Power Actuated Mode (ADS)	DPV Actuation	ICS – MSIV Position	ICS – RPV High Dome Pressure (10-second delay)	ICS – RPV Low Water Level (L2 30-sec delay)	ICS – RPV Low Water Level (L1)	TBV Closure – Low-low condenser vacuum[1]	ICS – Loss of Power Generation Bus (Loss of Feedwater Flow)	TBV Initiation – TSV Closure[1]	TBV Initiation – TCV Fast Closure[1]	TSV Closure – RPV High Water Level (L8)[1]	TSV Closure – Low Condenser Vacuum[1]	TCV Fast Closure – Load Rejection[1]	MSIV Closure – RPV Low Water Level (L2 w/30 sec)	MSIV Closure – RPV Low Water Level (L1)	MSIV Closure – Low Turbine Inlet/Main Steamline Pressure	MSIV Closure – Low Main Condenser Vacuum	MSIV Closure - High Steamline Flow	FW Pump Runback – L8	CRD Makeup Water – RPV Low Water Level (L2)[1]	RWCU/SDC Operation	ATWS – Feedwater Flow Runback[1]	ATWS – ADS Inhibition	SLCS – RPV Dome High Pressure - APRM not downscale	SLCS – DPV Open	SLCS – RPV Low Water Level (L2) - APRM not downscale	FAPCS – High Suppression Pool Temperature[1]	SCRRI/SRI[1]	GDCS	GDCS Equilizing Lines	High Radiation MCR EFU Initiation	Passive Containment Cooling PCCS	Feedwater Feedback Level Controller[1]
Loss of All Feedwater Flow									X						X																			
Loss of Feedwater Heating With Failure of SCRRI																																		
Feedwater Controller Failure – Maximum Flow Demand				X	X	X				X		X			X		X																	
Pressure Regulator Failure Opening of All Turbine Control and Bypass Valves				X													X																	
Pressure Regulator Failure – Closure of All Turbine Control and Bypass Valves				X	X	X									X		X																	
Generator Load Rejection with Total Turbine Bypass Failure (at High Power)				X	X	X								X	X		X																	

Table 15.1-6

NSOA Automatic Instrument Trip/Event Matrix

	Scram – APRM High Neutron Flux	Scram – APRM High Simulated Thermal Power	Scram – RPV Low Water Level (L3)	Scram – RPV High Water Level (L8)	Scram – Loss of Power on Four Power Generation Buses	Scram – MSIV Position	Scram – High Suppression Pool Temperature	Scram – TSV Closure (with insufficient bypass available)	Scram – TCV Fast Closure (with insufficient bypass available)	Scram – Low Condenser Vacuum	Scram – (Loss of Power Generation Bus) – Loss of Feedwater Flow	Scram – SRNM Period	Scram – High Drywell Pressure	Rod Block – SRNM Period or ATLM Parameter Exceeded or MRBM Parameter Exceeded[1]
Inadvertent Isolation Condenser Initiation														
Runout of One Feedwater Pump														
Opening of One Turbine Control or Bypass Valve														
Loss of Non-Emergency AC Power to Station Auxiliaries											X			
Loss of All Feedwater Flow											X			
Loss of Feedwater Heating With Failure of SCRRI		No credit												
Feedwater Controller Failure – Maximum Flow Demand				X										

Table 15.1-7

ESBWR NSOA Events

NSOA Event	Subsection Describing Event	Relevant Event Diagram
Loss of Feedwater Heating	15.2.1.1	15.1-2
Closure of One Turbine Control Valve	15.2.2.1	15.1-3
Generator Load Rejection with Turbine Bypass	15.2.2.2	15.1-4
Generator Load Rejection with a Single Failure in the Turbine Bypass System	15.2.2.3	15.1-5
Turbine Trip with Turbine Bypass	15.2.2.4	15.1-6
Turbine Trip with a Single Failure in the Turbine Bypass System	15.2.2.5	15.1-7
Closure of One Main Steamline Isolation Valve	15.2.2.6	15.1-8
Closure of All Main Steamline Isolation Valves	15.2.2.7	15.1-9
Loss of Condenser Vacuum	15.2.2.8	15.1-10
Loss of Shutdown Cooling Function of RWCU/SDC System	15.2.2.9	15.1-11
Inadvertent Isolation Condenser Initiation	15.2.4.1	15.1-12
Runout of One Feedwater Pump	15.2.4.2	15.1-13
Opening of One Turbine Control or Bypass Valve	15.2.5.1	15.1-14
Loss of Non-Emergency AC Power to Station Auxiliaries	15.2.5.2	15.1-15
Loss of All Feedwater Flow	15.2.5.3	15.1-16
Loss of Feedwater Heating With Failure of Selected Control Rod Run-In	15.3.1	15.1-17
Feedwater Controller Failure – Maximum Flow Demand	15.3.2	15.1-18
Pressure Regulator Failure Opening of All Turbine Control and Bypass Valves	15.3.3	15.1-19
Pressure Regulator Failure – Closure of All Turbine Control and Bypass Valves	15.3.4	15.1-20
Generator Load Rejection with Total Turbine Bypass Failure	15.3.5	15.1-21
Turbine Trip with Total Turbine Bypass Failure	15.3.6	15.1-22
Control Rod Withdrawal Error During Refueling	15.3.7	15.1-23
Control Rod Withdrawal Error During Startup	15.2.3.2	15.1-24a

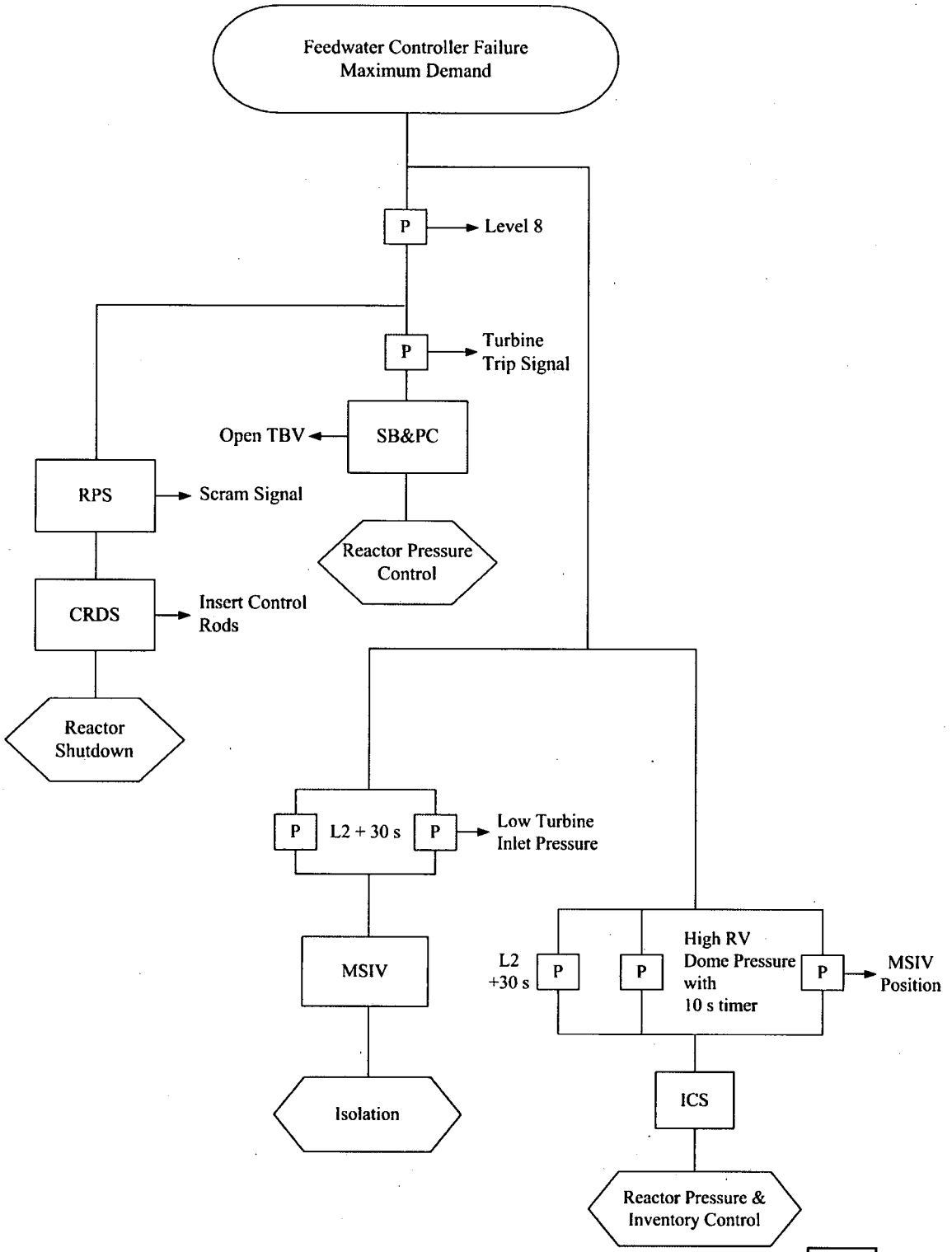


Figure 15.1-18. Event Diagram – Feedwater Controller Failure – Maximum Flow Demand

15.2 ANALYSIS OF ANTICIPATED OPERATIONAL OCCURRENCES

Each of the anticipated operational occurrences (AOOs) addressed in the Section 15.1, "Nuclear Safety Operations Analysis" (NSOA), is evaluated in the following subsections. Appendix 15A provides a determination of event frequency to categorize AOOs as defined in 10 CFR 50 Appendix A. Tables 15.2-1, 15.2-2, and 15.2-3 provide the important input parameters and initial conditions used/assumed in the AOO analyses.

In the analysis of AOOs and Infrequent Events in Section 15.3 nonsafety-related systems or components are considered to be operational in the following situations:

- When assumption of a nonsafety-related system results in a more limiting event;
- When a detectable and nonconsequential random, independent failure must occur in order to disable the system; and
- When nonsafety-related systems or components are used as backup protection (i.e. not the primary success path, included to illustrate the expected plant response to the event).

15.2.0 Assumptions

Assumptions are listed in the event discussions and Table 15.2-1.

15.2.1 Decrease In Core Coolant Temperature

15.2.1.1 Loss Of Feedwater Heating

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 loss of feedwater heating 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 Selected Control Rod Run-In (SCRRI) and Select Rod Insert (SRI) functions as discussed in Subsections 7.7.2.2.7.7, 7.7.3.3 and 7.8.1.1.3. ~~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 Nonsafety-Related Distributed Control and Information System (N-DCIS) to initiate the Selected Control Rod Run-In (SCRRI) and Select Rod Insert (SRI) functions to automatically reduce the reactor power.~~ 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 minimum critical power ration (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.

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 approximately half the magnitude of the loss of feedwater with failure of the SCRRI/SRI evaluated in Subsection 15.3.1. The resulting ΔCPR/ICPR (reduction of the MCPR) ~~similar~~ is also approximately half of the loss of feedwater with failure of the SCRRI/SRI. With the ΔCPR/ICPR of the loss of feedwater with failure of the SCRRI/SRI (see Table 15.3-1) close to the limiting AOO ΔCPR/ICPR, inadvertent IC initiation (See Table 15.2-4) the loss of feedwater temperature of 16.67°C (30°F) is not limiting and does not need to be quantitatively analyzed.~~to the event studied here, however none are limiting.~~

15.2.1.1.2 Sequence of Events and Systems Operation

Sequence of Events

Table 15.2-~~4~~5 lists the sequence of events for Figure 15.2-1

~~Because no scram occurs during this event, no immediate operator action is required. 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 plant performance, the plant instrumentation and controls, plant protection and reactor protection systems are assumed to function normally. A failure of a single HCU is assumed.

15.2.1.1.3 Core and System Performance

Input Parameters and Initial Conditions

The event is simulated by programming a change in FW enthalpy corresponding to the assumed loss in FW heating, shown in Table 15.2-1.

Table 15.2-1

Input Parameters, Initial Conditions and Assumptions used in AOO and Infrequent Event Analyses

Parameter	Value
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)
Maximum APRM Simulated Thermal Power Trip Scram, % NBR Time Constant, s	 115 7

Table 15.2-1

Input Parameters, Initial Conditions and Assumptions used in AOO and Infrequent Event Analyses

Parameter	Value
Simulated Thermal Power setpoint as a linear function of feedwater temperature for feedwater temperatures above 222.2°C (432°F)(5) Rate of Change limit on simulated thermal power setpoint(6)	115% at 222.2°C (432°F) 101% at 252.2°C (486°F) 26% / hour
Total Steamline Volume, m^3 (ft^3)(2)	135 (4767)
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	10 & 25 145
Isolation Condensers Max Initial Temperature, °C (°F) Minimum Initial Temperature, °C (°F) Time To injection valve full open (Max), s(~~2~~4) Heat Removal Capacity for 4ICs, 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)

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

(2) These values are not used when the inputs in Table 15.2-1a are used.

(~~1~~3) The SRV capacity used in the analysis is less than the ASME rated capacity noted in Table 5.2-2.

(~~2~~4) In the analysis, after 1 s logic delay, the IC opening valve curve began to open at 15 s for a total opening time of 30 s. For IICI the valve begins to open at 15 s with a opening time of 7.5 s.

(5) As the reactor power changes with changes in the feedwater temperature (Figure 4.4-1), the simulated thermal power trip setpoint also changes with feedwater temperature. The analytical limit is established to allow for feedwater temperature uncertainty effects on setpoint determination.

(6) 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 Subsection 7.7.3.2.3.

~~Table 15.2-4~~ 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)	~~RC&IS initiates~~ Selected Control Rod Run-In and Select~~ed~~ Rod Insert~~ion~~ (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
90.0	Steam flow below 60% of rated
134.0	SCRRI/SRI groups totally inserted
140.0	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.

15.3 ANALYSIS OF INFREQUENT EVENTS

Appendix 15A provides a determination of event frequency to categorize AOOs as defined in 10 CFR 50 Appendix A, or Infrequent Events. Section 15.0 describes the licensing basis for this categorization.

The input parameters, initial conditions, and assumptions in Tables 15.2-1, 2 and 3 are applied in the TRACG calculations, based on the equilibrium core in Reference 15.3-4, for the Infrequent Events addressed in Subsections 15.3.1 through 15.3.6 and Subsections 15.3.13 and 15.3.15. The summary of the Infrequent Events analyses is given in Tables 15.3-1a and 15.3-1b.

The results of the system response analyses for the initial core design documented in Reference 15.3-5 are provided in Reference 15.3-6. System response analyses bounding operation in the feedwater temperature operating domain is documented in Reference 15.3-7. A summary is provided in Appendix 15D.

15.3.1 Loss Of Feedwater Heating With Failure of Selected Control Rod Run-In

15.3.1.1 Identification of Causes

The loss of a feedwater (FW) heater can occur 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 FW cooling. In the second case, the FW bypasses the heater and no FW heating 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.

This event conservatively assumes the loss of FW heating as shown on Table 15.2-1, causing an increase in core inlet subcooling and core power due to the negative void reactivity coefficient. However, the power increase is slow.

A loss of feedwater heating 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 Selected Control Rod Run-In (SCRRI) and Select Rod Insert (SRI) functions as discussed in Subsections 7.7.2.2.7.7, 7.7.3.3 and 7.8.1.1.3, and scram is avoided.~~The Feedwater Control System (FWCS) logic is provided in Subsection 7.7.3, and includes logic provided to mitigate the effects of a loss of FW heating capability. The FWCS 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 Rod Control and Information System (RC&IS) to initiate the Selected Control Rod Run-In and Select Rod Insertion (SCRRI/SRI) function to automatically reduce the reactor power and avoid a scram.~~ However, for the event analysis, SCRRI/SRI is assumed to fail

and reactor scram on high simulated thermal power is not credited due to uncertainties. Therefore a new steady state is reached.

The frequency of this event is evaluated in Subsection 15A.3.6.

A feedwater temperature controller failure to minimum temperature demand can also result in an inadvertent reduction in feedwater temperature. While operating at normal conditions, a failure in the feedwater controller with minimum temperature demand could result in the opening of the high-pressure feedwater heater bypass valves. The feedwater temperature reduction is bounded by 55.6°C (100°F).

While operating at higher feedwater temperatures within the feedwater temperature increase region of the feedwater temperature operating domain (Figure 4.4-1), the No. 7 feedwater heater steam heating valves are open. A feedwater temperature controller failure to minimum temperature demand results in closure of the No. 7 feedwater heater steam heating valves, and subsequent opening of the high-pressure feedwater heater bypass valves. The resulting decrease in feedwater temperature is potentially greater than 55.6°C (100°F). The frequency of a feedwater temperature controller failure to mimimum temperature demand is evaluated in Subsection 15A.3.5.2. For this event, the loss of feedwater heating is detected by ATLM and DPS, and each sends a signal to initiate the SCRRI/SRI function. This mitigates the power increase caused by the reduction in feedwater temperature.

It is very unlikely that both feedwater temperature controller and SCRRI/SRI fail; however, if both failures occur, the reactor scrams on high simulated thermal power. In this event, the simulated thermal power setpoint is a function of the feedwater temperature as shown in Table 15.2-1. The rate of change limit on the simulated thermal power limit is shown in Table 15.2-1. Crediting the simulated thermal power scram ensures that the event remains bounded by the analysis of the loss of a feedwater heating with failure of SCRRI/SRI starting from the maximum allowable FWT operating point documented in Subsection 2.4.1 of Reference 15.3-7. Because the event is bounded by the loss of feedwater heating event described in Subsection 2.4.1 of Reference 15.3-7, no further discussion of the feedwater controller failure with minimum temperature demand is included in this subsection.

15.3.1.2 Sequence of Events and Systems Operation

Sequence of Events

Table 15.3-2 lists the sequence of events for Figure 15.3-1.

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.

The high simulated thermal power trip (STPT) scram is the primary protection system trip in mitigating the effects of this event. However, credit was not taken for this scram to consider the possibility that, for a similar case with a somewhat lower loss of heating, the scram setpoint might not be reached, while the consequences would only be slightly less severe for this case than the event analyzed here.

15.3.1.3 Core and System Performance

Input Parameters and Initial Conditions

The event is simulated by programming a change in FW enthalpy corresponding to the assumed loss in FW heating, shown in Table 15.2-1.

Results

Reactor scram should be initiated during this event. However, as explained above, credit for STPT scram was not taken. The nuclear system pressure does not significantly change during the event, and consequently, the RCPB is not threatened.

This event is potentially limiting with respect to the number of rods in boiling transition. The OLMCPR is established to the limiting event and documented in the COLR in accordance with Technical Specifications.

15.3.1.4 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 pressure vessel or containment are designed. Therefore, these barriers maintain their integrity and function as designed. In this event, the number of fuel rods that enter transition boiling is bounded by 1000 rods. It was assumed that all rods entering transition boiling fail.

15.3.1.5 Radiological Consequences

A radiological analysis was performed for an event where 1000 fuel rods fail as a result of entering transition boiling.

The scenario considered is for the fission product release paths to the environment consists of the fission products traveling down the main steam lines, eventually reaching the condenser, where they leak from the condenser to the environment. This scenario is modeled after the BWR rod drop accident described in Regulatory Guide 1.183, Appendix C.

The source term for the event is defined in Tables 15.3-13, 15.3-14 and 15.3-15. As can be seen in Table 15.3-16, the off site dose for this event is less than 2.5 REM Total Effective Dose Equivalent (TEDE) assuming the bounding number (1000 rods) of fuel failures.

Assumptions Requiring Confirmation

Site parameter assumptions in the radiological analysis are confirmed in Subsection 2.0.1.

15.3.2 Feedwater Controller Failure – Maximum Flow Demand

15.3.2.1 Identification of Causes

See Subsection 15.2.4.2. This event assumes multiple control system failures, to simultaneously increase the flow in multiple FW pumps to their maximum limit. The frequency of this event is evaluated in Subsection 15A.3.5.1.

Table 15.3-1a

Results Summary of Infrequent Events (1) (2)

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	Maximum Calculated TEDE
15.3.1	Loss of Feedwater Heating with SCRRI failure	122	7.13 (1034)	7.27 (1054)	7.09 (1028)	121	0.11	(Note 4)
15.3.2	FWCF – Maximum Flow Demand	117	7.29 (1057)	7.43 (1078)	7.25 (1052)	109	0.04	--
15.3.3	Pressure Regulator Failure – Opening of all TCVs and BPVs	100	7.08 (1027)	7.21 (1046)	7.04 (1021)	100	0.00	--
15.3.4	Pressure Regulator Failure – Closing of all TCVs and BPVs	137	8.06 (1169)	8.19 (1188)	8.06 (1169)	104	0.05	(Note 4)
15.3.5	Load Rejection with total bypass failure	339	8.14 (1181)	8.27 (1199)	8.15 (1182)	108	0.11	(Note 4)
15.3.6	Turbine Trip with total bypass failure	295	8.13 (1179)	8.26 (1198)	8.13 (1179)	108	0.11	(Note 4)
15.3.13	Inadvertent SRV open	101	7.08 (1027)	7.21 (1046)	6.99 (1014)	101	<0.01	--
15.3.15	Stuck open SRV (2)	100.0	7.08 (1027)	7.21 (1046)	7.04 (1021)	100.0	<0.1	--

(1) The input parameters and initial conditions used to perform the analysis in this table are located in Table 15.2-1.

(2) This table summarizes the events calculated with the TRACG code. Table 15.3-1b contains the summary of the remaining Infrequent Events.

(3) The initiating event can produce some over power, but the Stuck SRV open should not produce any appreciable overpower or MCPR reduction.

(4) The 1000 fuel-rod failure case bounds this event. Results are shown in Table 15.3-16.

Table 15.3-3

Sequence of Events for Feedwater Controller Failure – Maximum Flow Demand

Time (sec)	Event *
0	Initiate simulated runout of all FW pumps (170% at rated vessel pressure).
12.4	Main turbine bypass valves opened to control vessel pressure.
15.4	L8 vessel level setpoint is reached.
16.3	Scram, trip of main turbine and FW pump runback is activated.
16.43	Turbine Bypass fast opening activation limits the pressurization of the vessel.
16.5	The rods begin to enter inside the core.
Later	L2 is reached because no FW availability, activating IC and HP_CRD to recover the level and isolating MSIV's.

* See Figure 15.3-2

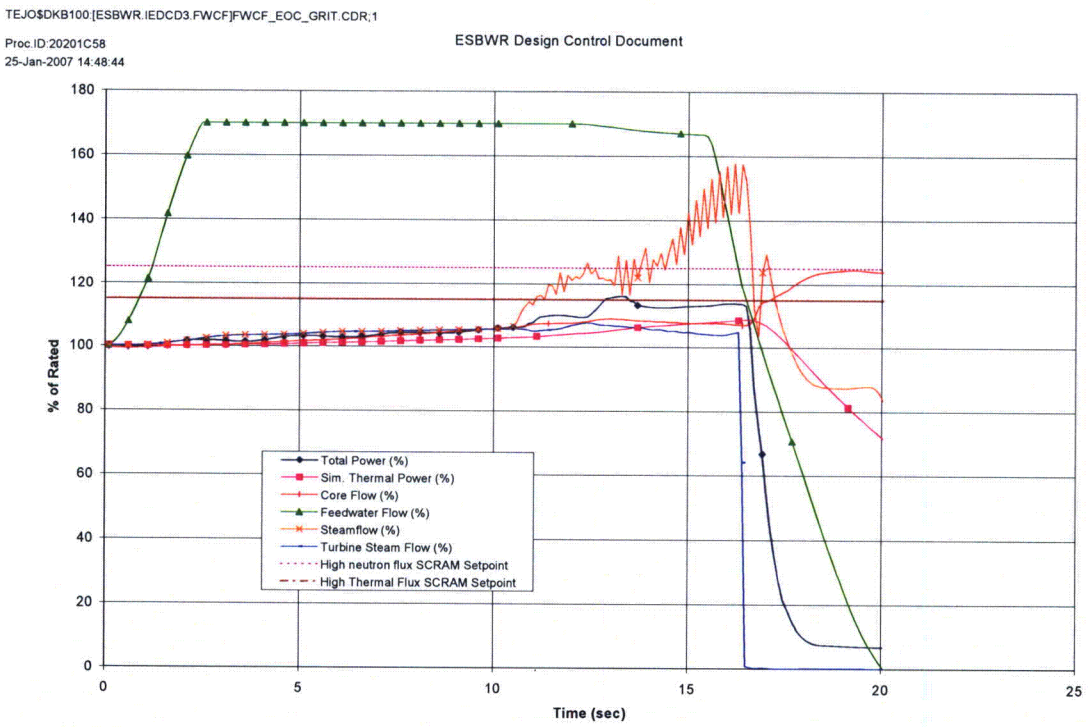
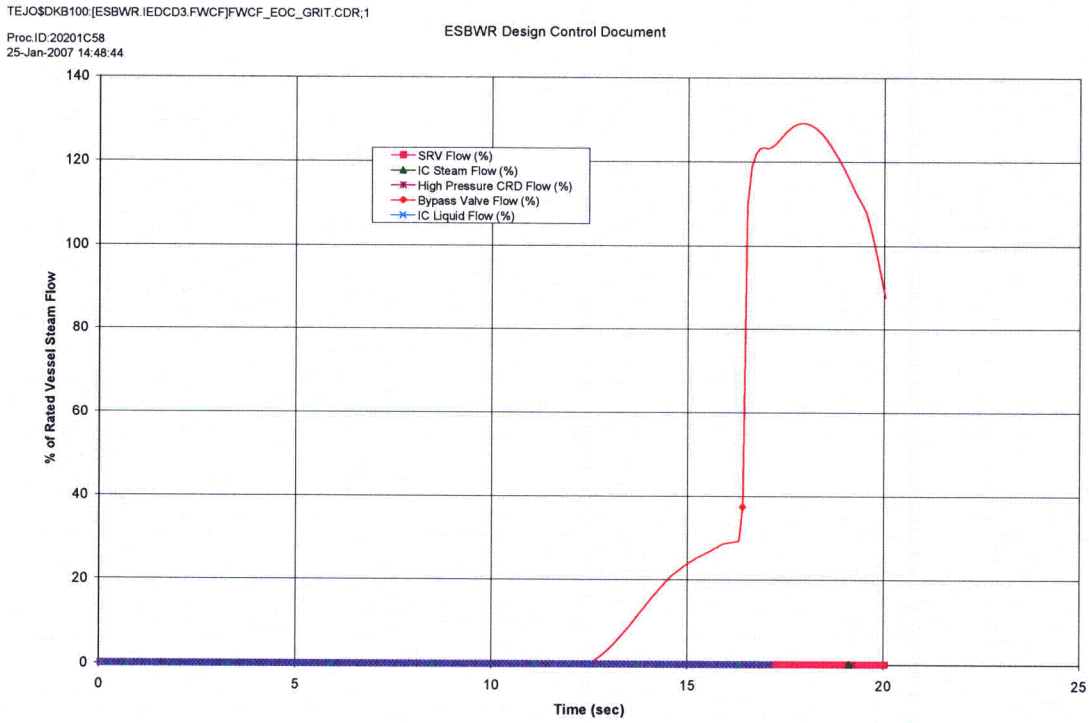


Figure 15.3-2a. Feedwater Controller Failure – Maximum Flow Demand



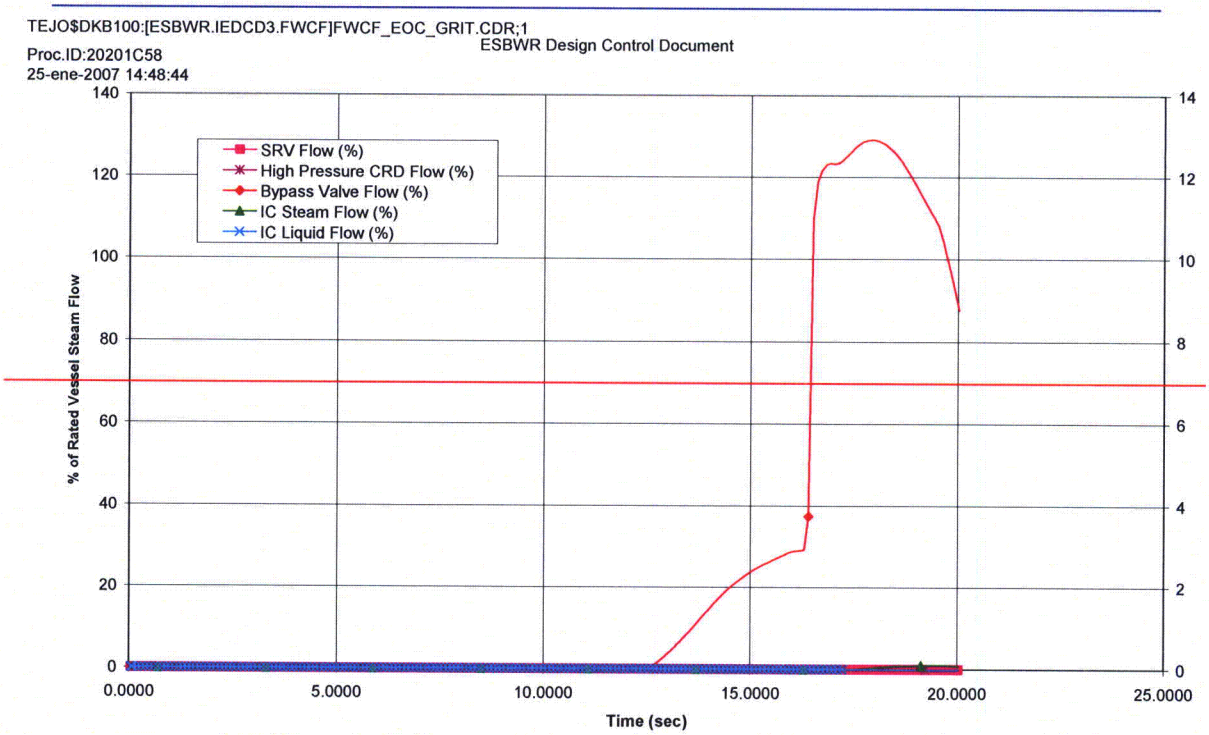


Figure 15.3-2b. Feedwater Controller Failure – Maximum Flow Demand

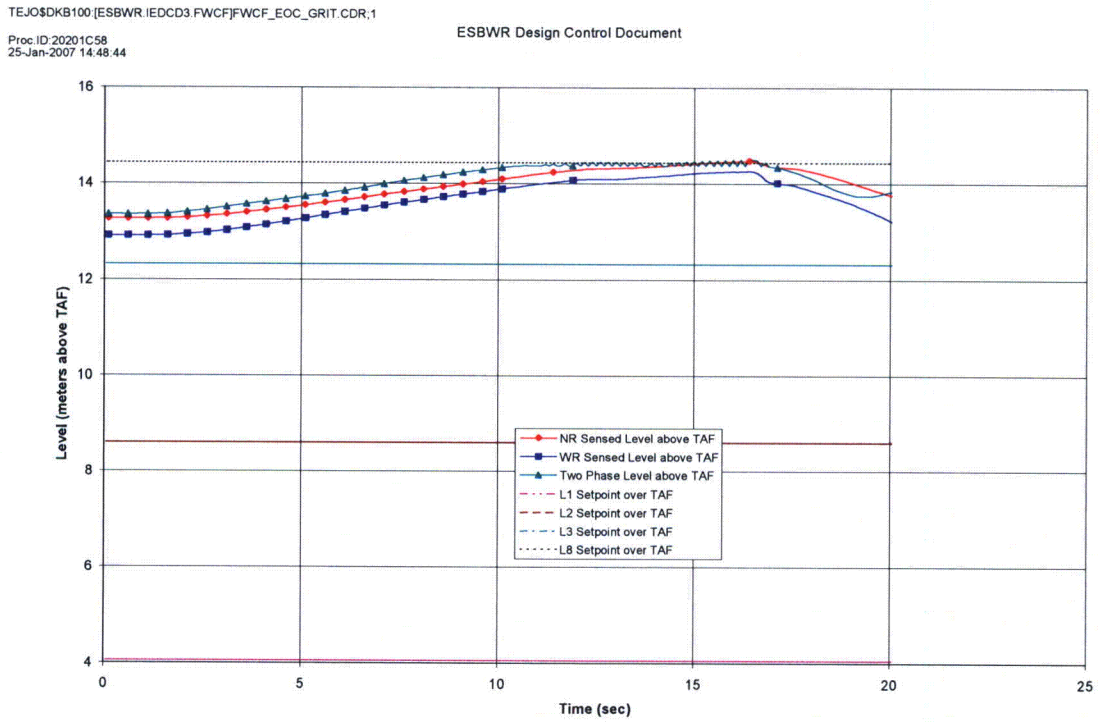


Figure 15.3-2c. Feedwater Controller Failure – Maximum Flow Demand

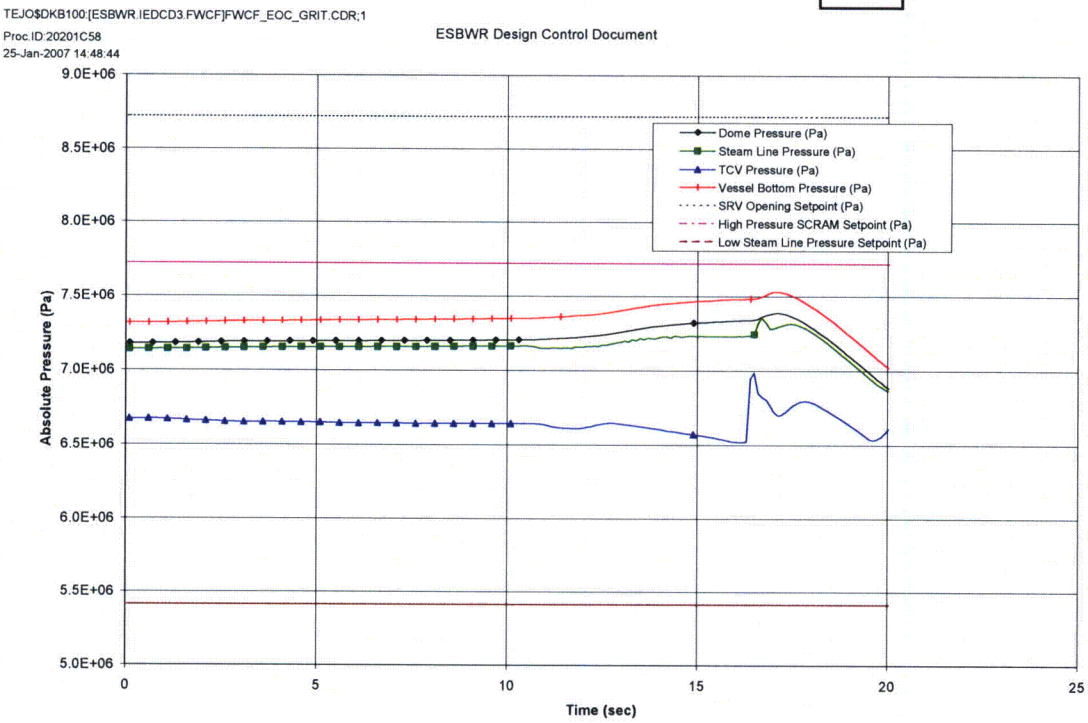


Figure 15.3-2d. Feedwater Controller Failure – Maximum Flow Demand

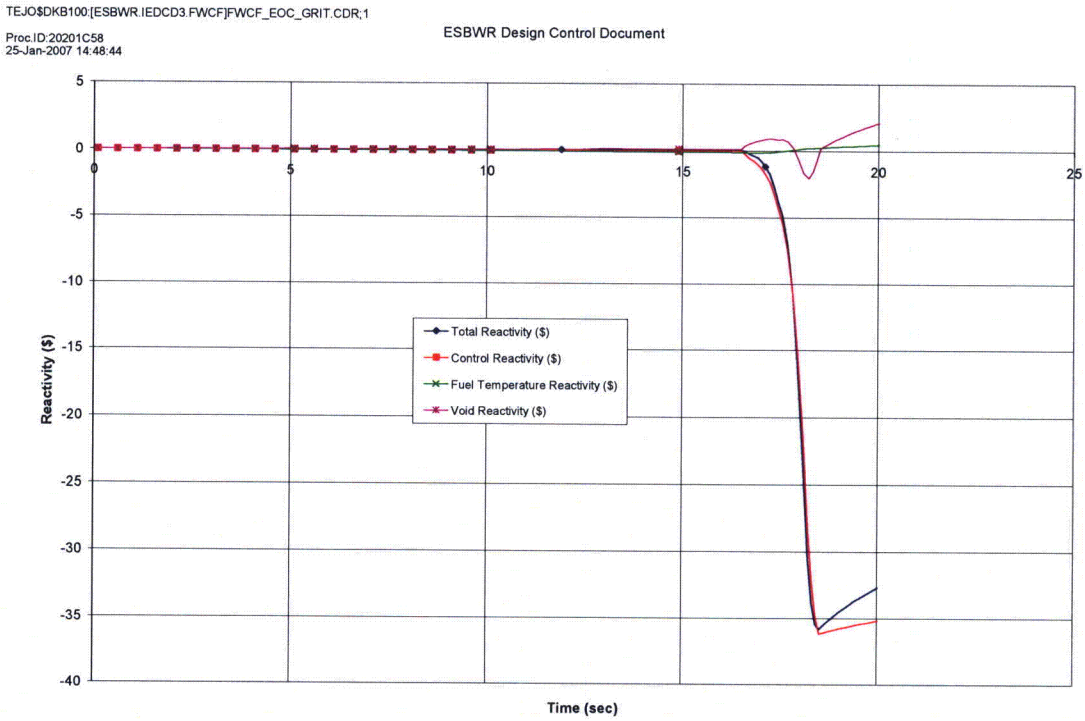


Figure 15.3-2e. Feedwater Controller Failure – Maximum Flow Demand

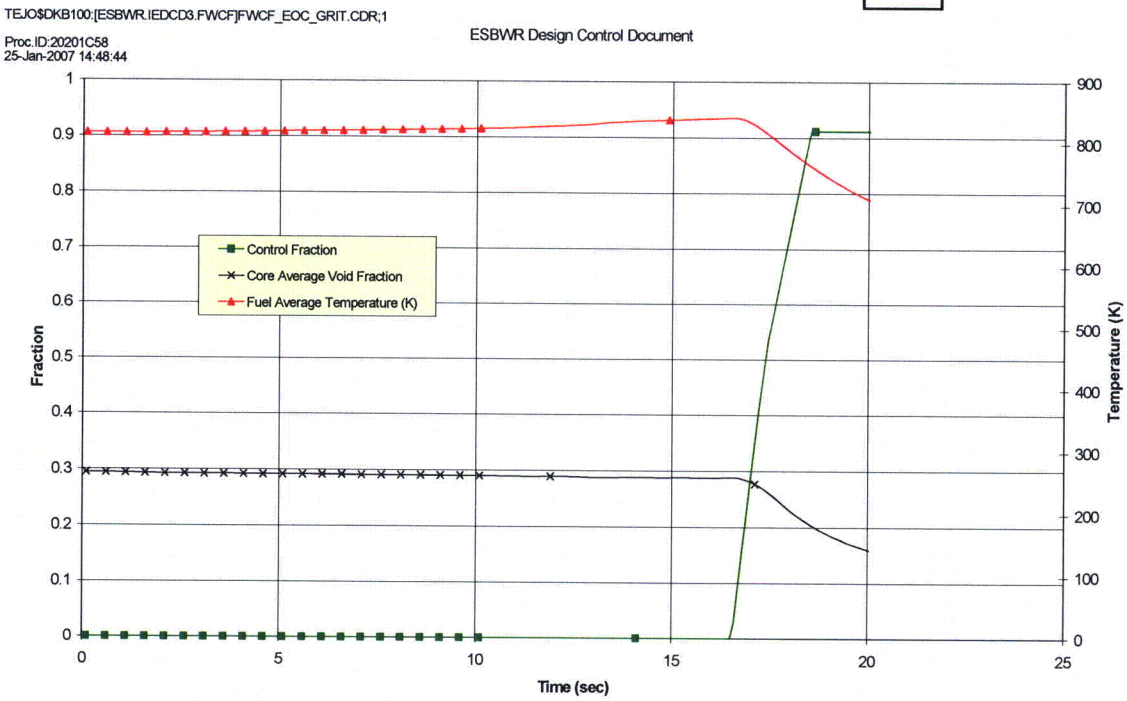
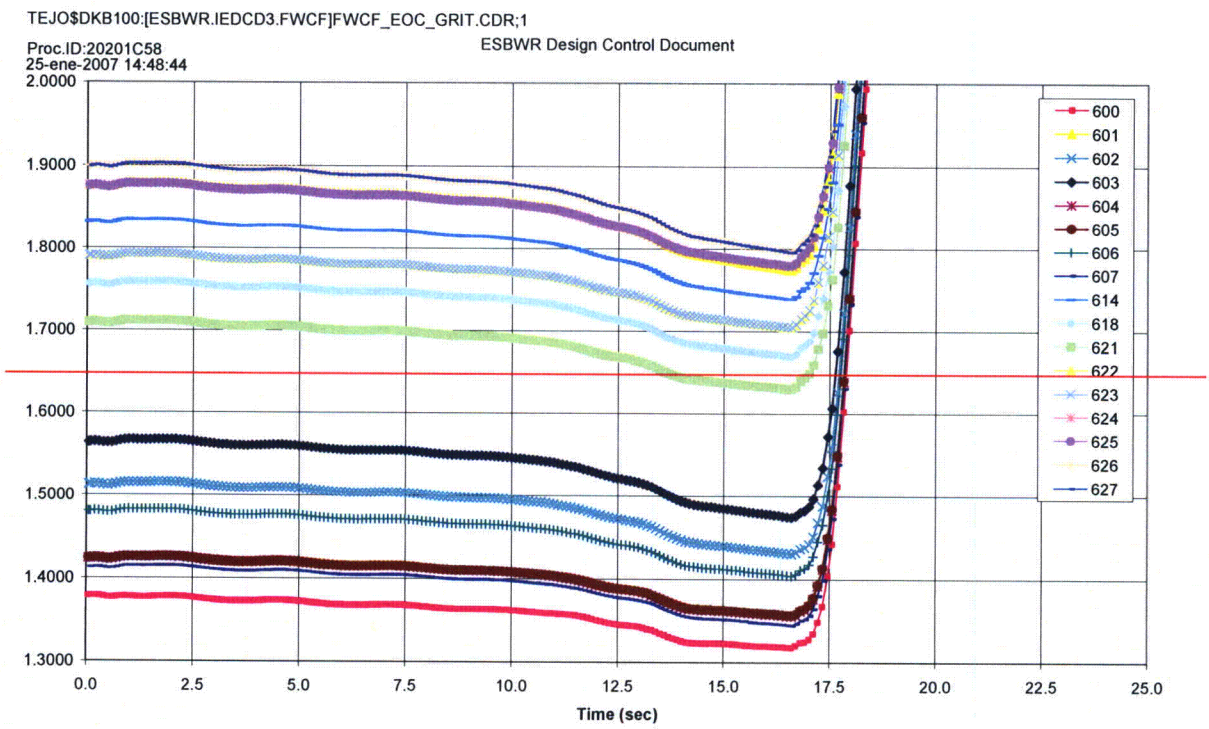


Figure 15.3-2f. Feedwater Controller Failure – Maximum Flow Demand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ESBWR Design Control Document
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1.8000
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1.6000
1.5000
1.4000
1.3000
0.0
2.5
5.0
7.5
10.0
12.5
15.0
17.5
20.0
22.5
25.0
Time (sec)
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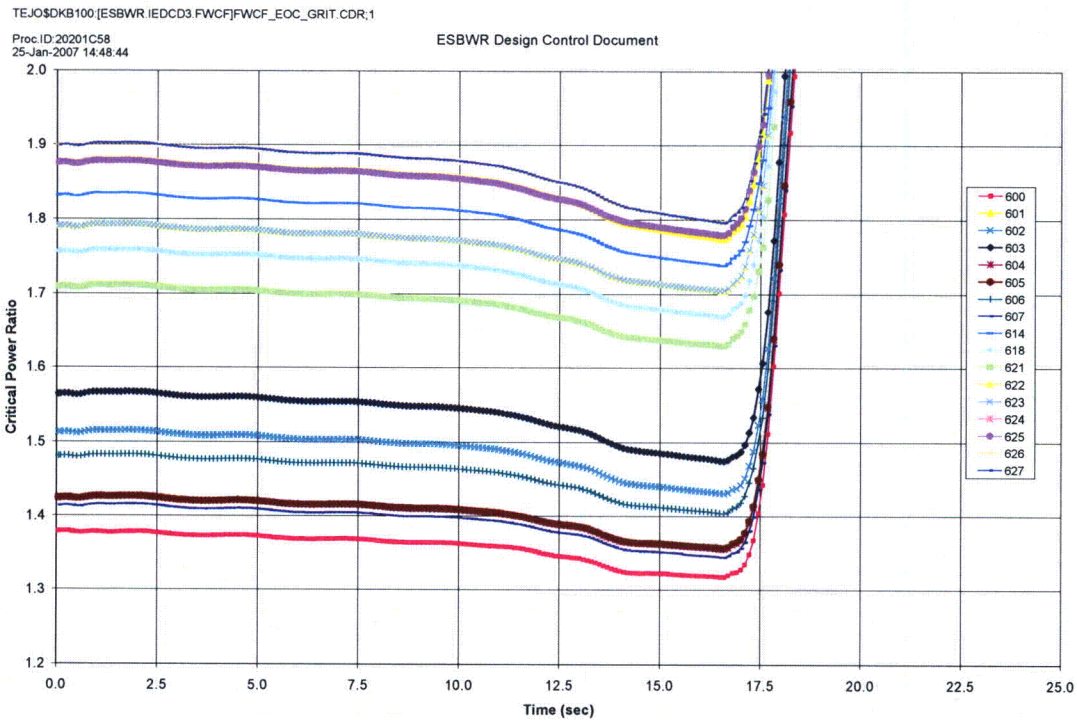

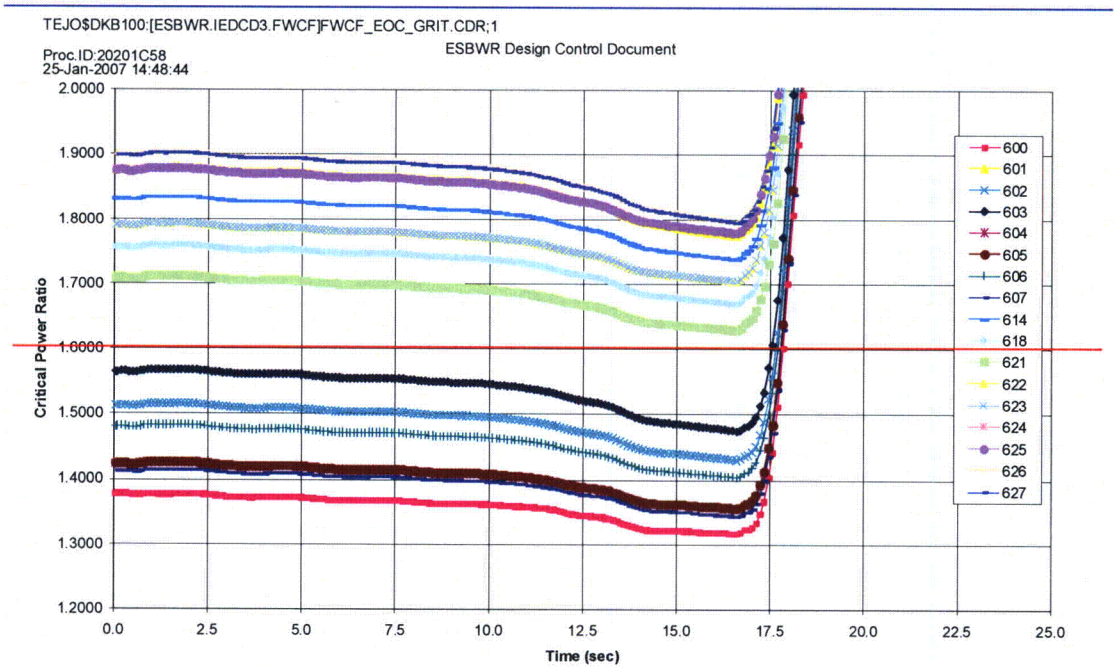

Figure 15.3-2g. Feedwater Controller Failure – Maximum Flow Demand

are required for continued plant operation, there is a high probability that the controllers are available following a turbine trip event. The signal failure probability is judged to be negligible.

Support System Failure: The only relevant support system is the AC power and loss of AC power results in a different category of initiating event. Therefore, the failure of AC power is not considered in this evaluation.

Operator action: Operator action is not needed for TBVs to open and operator action cannot cause a failure of TBVs to open on demand. Therefore, operator error is not considered in this evaluation.

The failure probability of TBVs is 4.4 E-4 per demand, based on the above discussion.

15A.3.4.3 Result

The frequency of generator load rejection with bypass failure is evaluated as a product of the generator load rejection frequency (0.45 per year) and probability of failure of TBVs (4.4E-4 per demand).

The event frequency = (0.45)*(4.4E-4) = 1.98E-4 per year.

This translates to one event in over 5,000 years. Therefore, the event frequency meets the criterion of less than once in 100 years.

15A.3.5 Feedwater Controller Failure ~~– Maximum Demand~~

The Feedwater Control System (FWCS) accomplishes both RPV water level control and feedwater temperature control. RPV water level control is accomplished by manipulating the speed of the feedwater pumps. Feedwater temperature control is accomplished by manipulating the heating steam flow to certain feedwater heaters or directing a portion of the feedwater flow around the high-pressure feedwater heaters. The two functions are performed by two sets of triple-redundant controllers located in separate cabinets, using independent and diverse inputs. The description of the FWCS is provided in Subsection 7.7.3.

There are two events of concern resulting from failures of the FWCS. One event consists of the FWCS erroneously generating a maximum flow demand, and the other one consists of the FWCS erroneously generating a minimum temperature demand. These two events are discussed in the following subsections.

The simultaneous occurrence of a maximum flow demand and a minimum temperature demand is considered incredible due to the independence of the two control schemes. The random probability of the second controller (e.g., temperature) failing while the first controller (e.g., flow) is failed, and before the effects of its failure are mitigated, is insignificant.

15A.3.5.1 Feedwater Controller Failure – Maximum Flow Demand

15A.3.5.1.1 Introduction

One function of the FWCS regulates the flow of feedwater into the RPV to maintain predetermined water level limits during transients and normal plant operating modes~~The Feedwater Control System (FWCS) is a power generation system, which is designed to maintain proper water level in the reactor during operation~~. The event of concern is one that results from

one or more failures in the FWCS that causes multiple FW pumps to go to maximum output. This results in the feedwater pumps delivering a large amount of water, which increases the reactor water level to Level 8, at which time the feedwater pumps are tripped by an independent system, the main turbine is tripped and a reactor scram is initiated. Such an event is called the "Feedwater Controller Failure – Maximum Flow Demand" event. The frequency of this event is evaluated in this subsection.

~~15A.3.5.2~~ 15A.3.5.1.2 Analysis

~~The description of the FWCS is provided in Subsection 7.7.3.~~

The FWCS is designed to maintain proper reactor pressure vessel water level in the operating range from high water level (Level 9) to low water level (Level 2). During normal operation, feedwater flow is delivered to the reactor vessel through three Reactor Feedpumps (RFPs), which operate in parallel. Each RFP is driven by an induction motor that is controlled by an adjustable speed drive (ASD). The fourth RFP is in standby mode and auto-starts if any operating feedpump trips while at power.

To perform the RPV water level control function, ~~T~~the FWCS is equipped with a dedicated triple-redundant, fault-tolerant digital controller (FTDC) including power supplies, and input/output signals. The FTDC consists of three parallel processing channels, each containing the hardware and software for execution of the control algorithms. The FTDC is designed to a high degree of reliability. Based on Subsection 7.7.3, the Mean Time to Failure (MTTF) of the ~~Feedwater System Controller~~FTDC is at least 1,000 years.

The actual reliability of the ~~Feedwater~~ feedwater flow controller is expected to be much higher than the specified minimum MTTF requirement of 1,000 years. It is assumed that the feedwater flow controller can fail high or fail low with equal probability. Therefore, the frequency of the controller failing in a manner to cause maximum demand is less than once in 2,000 years.~~If any one of the three controllers fails either high causing maximum demand (or fails low causing minimum demand), the other two controllers would continue to function and the frequency of two or three controllers failing in a manner to cause maximum demand is once in 2,000 years.~~

~~15A.3.5.3~~ 15A.3.5.1.3 Result

The frequency of the feedwater flow controller failing in a manner to cause maximum demand of feedwater is less than once in 2,000 years and therefore, the event frequency meets the criterion of being less than once in 100 years.

15A.3.5.2 Feedwater Controller Failure – Minimum Temperature Demand

15A.3.5.2.1 Introduction

One function of the FWCS controls feedwater temperature to allow reactor power control without moving control rods. The event of concern is one that results from one or more failures in the FWCS that causes the feedwater heater bypass valves to fully open, and the seventh feedwater heater steam heating valves to fully close. This results in a significant decrease in feedwater temperature, which will be independently detected by the Automated Thermal Limit Monitoring (ATLM) system and by the Diverse Protection System (DPS). Either one of these systems will mitigate the event by initiating SCRRI and SRI functions. Such an event is called

the "Feedwater Controller Failure – Minimum Temperature Demand" event. The frequency of this event is evaluated in this subsection.

15A.3.5.2.2 Analysis

To perform the feedwater temperature control function, the FWCS is equipped with a dedicated triple-redundant FTDC including power supplies, and input/output signals. The FTDC consists of three parallel processing channels, each containing the hardware and software for execution of the control algorithms. The FTDC is designed to a high degree of reliability. Based on Subsection 7.7.3, the Mean Time to Failure (MTTF) of the FTDC is at least 1,000 years.

The actual reliability of the feedwater temperature controller is expected to be much higher than the specified minimum MTTF requirement of 1,000 years. It is assumed that the feedwater temperature controller can fail low or fail high with equal probability. Therefore, the frequency of the controller failing in a manner to cause minimum demand is less than once in 2,000 years.

15A.3.5.2.3 Result

The frequency of the feedwater temperature controller failing in a manner to cause minimum feedwater temperature demand is less than once in 2,000 years and therefore, the event frequency meets the criterion of being less than once in 100 years.

15A.3.6 Loss of Feedwater Heating with Failure of SCRRI and SRI

15A.3.6.1 Introduction

The loss of feedwater heating causes the feed temperature to go down which increases the reactivity level. The ESBWR is designed such that the loss of feedwater results in insertion of selected control rods, so the reactivity level is adjusted appropriately. The failure of feedwater heating followed by the failure of the selected control rods to insert is the event of concern. The assessment of the mitigation capability given a loss of FW heating is estimated based on the failure of either of two functions: SCRRI or SRI. Failure of either method for inserting control rods would fail the mitigation function.

15A.3.6.2 Analysis

The loss of feedwater heating can occur at any given time during normal power range operation (e.g. the feedwater heaters are not operational during low power startup/shutdown conditions when the main turbine is not operational). When this event happens, the feedwater temperature goes down. This is detected by redundant temperature sensors in the feedwater piping lines that lead to the reactor pressure vessel that provide input signals to the Feedwater Control System (FWCS). The description of the FWCS is provided in Subsection 7.7.3. The primary purpose of the FWCS is to maintain proper reactor pressure vessel water level in the operating range. In addition, the FWCS sends signals to the Rod Control and Information System (RC&IS), via the Nonsafety-Related DCIS equipment, to insert selected control rods to mitigate the consequence of the loss of feedwater heating event. The description of the RC&IS is provided in Subsection 7.7.2. The RC&IS equipment in the control room back panel area sends signals to individual control rod logic implemented in the local RC&IS equipment in the reactor building in order to complete the run-in of each of the selected control rods to its associated, pre-defined Selected Control Rod Run-In and Select Rod Insertion (SCRRI/SRI) target position.

Table 15A-3

Summary of Event Frequency Estimates

Section Number	Event	Event Frequency
15A.3.1	Pressure Regulator Failure – Opening of All Turbine Control and Bypass Valves	1 Event in 2,000 Years
15A.3.2	Pressure Regulator Failure – Closure of All Turbine Control and Bypass Valves	1 Event in 2,000 Years
15A.3.3	Turbine Trip with Total Turbine Bypass Failure	1 Event in 1,900 Years
15A.3.4	Generator Load Rejection with Total Turbine Bypass Failure	1 Event in 5,000 Years
15A.3.5.1	Feedwater Controller Failure – Maximum Flow Demand	1 Event in 2,000 Years
15A.3.5.2	Feedwater Controller Failure – Minimum Temperature Demand	1 Event in 2,000 Years
15A.3.6	Loss of Feedwater Heating With Failure of Selected Control Rod Run-In	1 Event in 600 Years
15A.3.7	Inadvertent Shutdown Cooling Function Operation	1 Event in 6,200 Years
15A.3.8	Inadvertent Opening of a Safety Relief Valve	1 Event in 300 Years
15A.3.9	Inadvertent Opening of a Depressurization Valve	1 Event in 1,700 Years
15A.3.10	Stuck Open Safety Relief Valve	1 Event in 4,400 Years
15A.3.11	Control Rod Withdrawal Error During Refueling	1 Event in 1,000 Years
15A.3.12	Control Rod Withdrawal Error During Startup	1 Event in 741,000 Years
15A.3.13	Control Rod Withdrawal Error During Power Operation	1 Event in 40,900 Years
15A.3.14	Fuel Assembly Loading Error, Mislocated Bundle	1 Event in 1,046 Years
15A.3.15	Fuel Assembly Loading Error, Misoriented Bundle	1 Event in 418 Years
15A.3.16	Liquid-Containing Tank Failure	1 Event in 3,000 Years