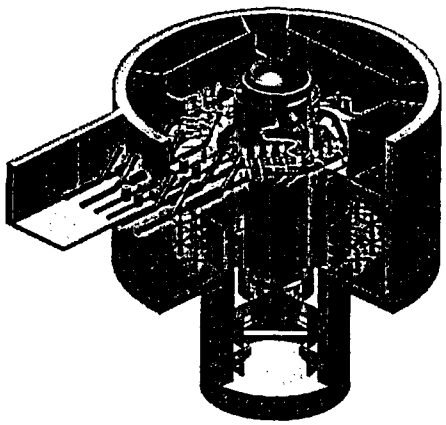
Chapter 6 LOCA Analysis
Sequence of Events and Credited Systems



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Outline

Background

Summary of Systems

ECCS/LOCA Analyses

Containment/LOCA Analyses

Summary

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Background

ECCS/LOCA Performance Evaluation

- Acceptance Criteria (extracted from 10 CFR 50.46)
  - PCT ≤ 2200 °F
  - Max. Cladding Oxidation ≤ 17%
  - Max. Metal-water Reaction ≤ 1%
  - Long-term core cooling shall be maintained
- ESBWR ECCS/LOCA Evaluation
  - Conformance to criteria is assured by water level inside shroud
  - Core never uncovers during a DBA event
    - Assuring no core heatup in short- and long-term

Background (Cont.)

Containment/LOCA Performance Evaluation

- General Design Criteria
  - Containment is designed with sufficient margin (GDC 16 and 50)
- ESBWR Containment/LOCA Evaluation
  - Calculated peak containment pressure is below the design value by sufficient margin
  - Long-term core coverage

Key measures in LOCA Event:
Water Level inside shroud and Peak DW Pressure

Summary of Safety Systems
(Credited in the 72-hrs ESBWR LOCA Analyses)

System, Structure or Component (SSC)	Function	Impact on Plant Responses
RPS	Detect signals	Initiate scram; Shutdown reactor
CRDS	Insert control rods	Shutdown reactor
ICS	Condense RPV steam	Depressurize RPV; Transfer decay heat from RPV to IC/PCC pools
ADS – SRVs	Discharge steam from RPV to suppression pool, via SRVs	Depressurize RPV
ADS – DPVs	Discharge steam from RPV to DW, via DPVs	Depressurize RPV
GDCS – Injection valves	Allow GDCS water to flow into RPV via GDCS lines	Increase RPV inventory; Increase water level and increase margin to core uncovery
GDCS – Equalization valves	Allow suppression pool water to flow into RPV via equalization lines	Increase RPV inventory; Provide long-term post-LOCA core cooling
MSIV	Isolate RPV	Minimize radiological consequences
SLCS	Inject SLCS water into RPV	Increase RPV inventory; Increase water level and increase margin to core uncovery
PCCS	Condense DW steam	Limit containment pressure; Transfer decay heat from RPV to IC/PCC pools
Float valves to spillover pipes	Allow DW annulus water to flow into suppression pool via spillover pipes	Assure long-term RPV water level recovery; Complete the water flow paths among RPV, DW annulus and suppression pool
Connection valves - (Dryer/Separator storage pool to PCC pool)	Allow water in Dryer/Separator storage pool) to flow into the IC/PCC expansion pools	Increase IC/PCC pool water inventory for 72-hour decay heat removal capability
Vacuum Breakers	Allow the flow of gas from WW to DW to equalize the DW and WW pressure	Minimize differential pressure (WW-DW)

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Summary of Non-Safety Systems
(None Credited in the 72-hrs ESBWR LOCA Analyses)

System, Structure or Component (SSC)	Function	Impact on Plant Responses (When SSC is available)
HP_CRD	Provide high-pressure makeup water to the RPV	Increase RPV inventory; Increase water level and increase margin to core uncovery
RWCU/SDC	Provide additional decay heat removal capability	Cooldown RPV to atmospheric conditions; Increase margin to containment design pressure
FAPCS – Suppression pool cooling and cleanup	Provide suppression pool cooling for accident recovery	Decrease WW and DW pressure; Increase margin to containment design pressure
FAPCS – Drywell Spray	Spray cooled water to DW air space	Reduce DW pressure; Increase margin to containment design pressure
FAPCS - LPCI	Inject cooled water from suppression pool into RPV via FW loop A	Increase RPV inventory; Increase water level and increase margin to core uncovery
Fire Protection System	Provide emergency backup source of makeup water to IC/PCC pools	Increase IC/PCC pool level and decay heat removal rate; Increase margin to containment design pressure

Water Level and Pressure margins increase with availability of one or more non-safety systems

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ECCS/LOCA Analyses

Key measures

- Water level inside shroud
- No core uncovery, no core heatup

Feedwater Line Break

- Nominal conditions
- Failure of 1 GDCS Injection Valve
- Event time ~ 0 to 2000 seconds

Table 6.3-7

Operational Sequence of ECCS For A Feedwater Line Break with Failure of One GDCS Injection Valve (Nominal Calculation)

Time (sec)	Events
0	Guillotine break of feedwater line inside containment; normal auxiliary power assumed to be lost; feedwater is lost; Scram assumed.
2	Loss of normal auxiliary power confirmed; reactor scram activated; IC initiated.
10	Level 2 is reached; Reactor isolation timer initiated.
14	Level 1.5 is reached.
24	Level 1.5 signal confirmed; ADS/GDCS/SLC timer initiated; ADS-SRV actuation begins.
75	First DPV opens; SLC flow starts.
79	Peak DW pressure of 307 kPa (44.6 psia) is reached.
104	Minimum chimney water level (8.95 m) is reached.
174	GDCS timer timed out. GDCS injection valves open.
280	Vessel pressure decreases below maximum injection pressure of GDCS. GDCS flow into the vessel begins. Chimney and downcomer water levels start to rise.
374	SLCS flow depleted.
from 100 to 2000	RPV water level remains higher than Level 0.5. Therefore equalizing line valves are not expected to open for this event.

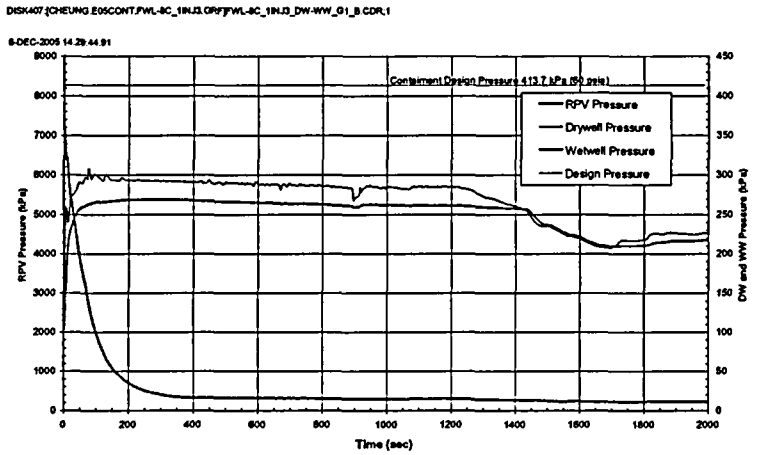

Figure 6.3-10 System Pressures

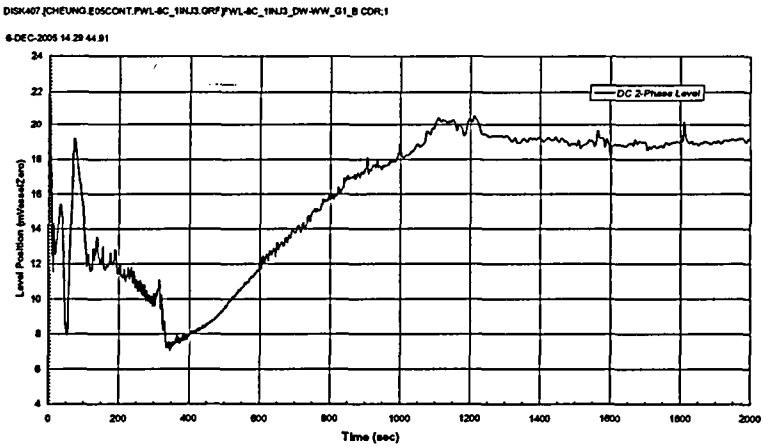

Figure 6.3-9 Downcomer Water Level

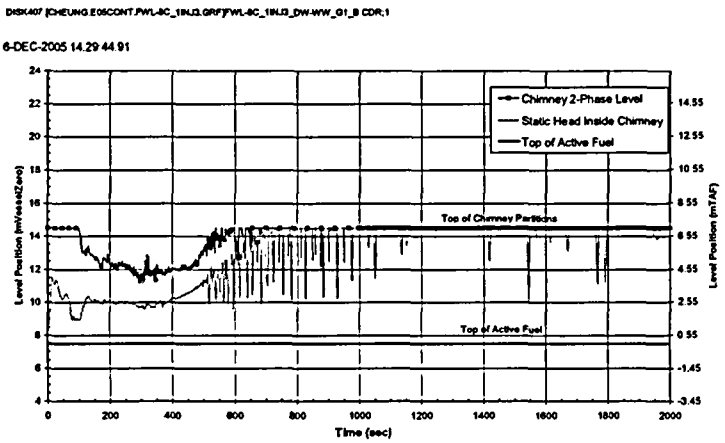

Figure 6.3-8 Chimney Water Level

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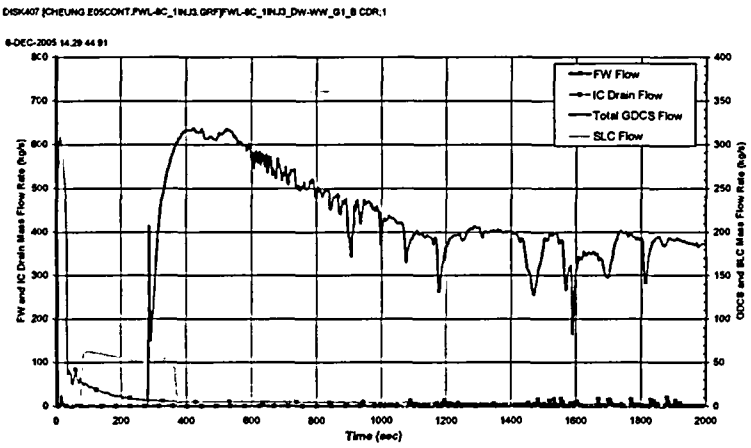

Figure 6.3-13 Flow Into Vessel

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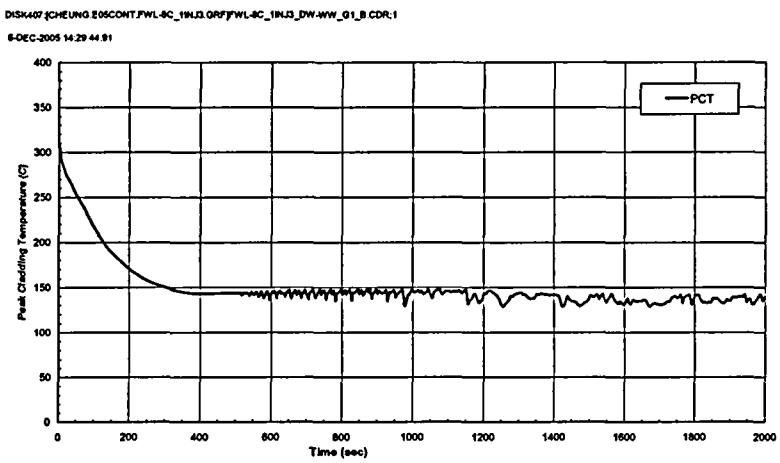

Figure 6.3-14 Peak Cladding Temp.

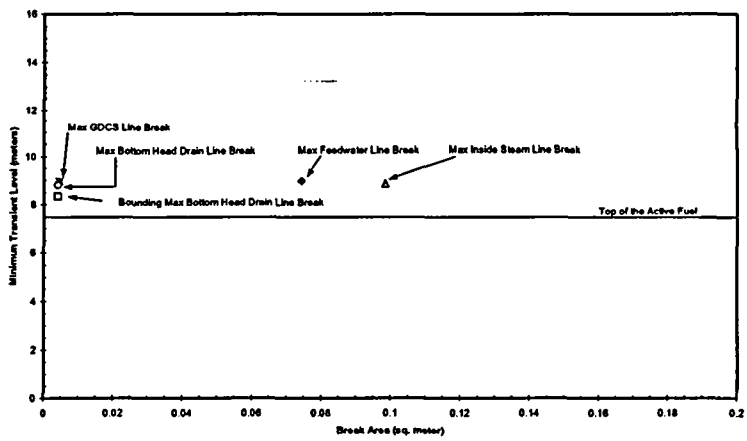

Figure 6.3-6 Min. Chimney Water Level vs. Break Area

Containment/LOCA Analyses

Key measures

- Sufficient margin to containment design pressure
- Long-term water level inside shroud
- Calculated containment temperature < 171 °C (design value)

Feedwater Line Break

- Nominal conditions
- Failure of 1 SRV Valve
- Event time ~ 0 to 72 hours

Table 6.2-7 Operational Sequence of ECCS For A Feedwater Line Break *with Failure of One SRV (Nominal Case)*

Time (sec)	Events
0	Guillotine break of feedwater line inside containment; normal auxiliary power assumed to be lost; feedwater is lost; Scram assumed.
2	Loss of normal auxiliary power confirmed; reactor scram activated; IC initiated.
5	Level 3 is reached; Level 3 scram initiated.
10	Level 2 is reached; Reactor isolation timer initiated.
14	Level 1.5 is reached.
24	Level 1.5 signal confirmed; ADS/GDCS/SLCS timer initiated; ADS-SRV actuation begins.
74	First DPV opens; SLC flow starts.
78	Maximum DW pressure of 341 kPa (49.5 psia) is reached.
174	GDCS timer timed out. GDCS injection valves open.
296	Vessel pressure decreases below maximum injection pressure of GDCS. GDCS flow into the vessel begins. Chimney and downcomer water levels start to rise.
376	SLCS flow depleted.
From ~ 290 to 259000 (72 hrs)	RPV water level remains higher than Level 0.5. Therefore equalizing line valves are not expected to open for this event.
~ 146000 (40.6 hrs)	PCC pool level drops below the elevation of 29.6 m; top ¼ portion of the PCC tube length becomes uncovered; connection valves open to allow the water from the Dryer/Separator storage pool to flow into the PCC pools.
~ 259000 (72 hrs)	Long-term DW pressure of 292 kPa (42.4 psia) is reached.

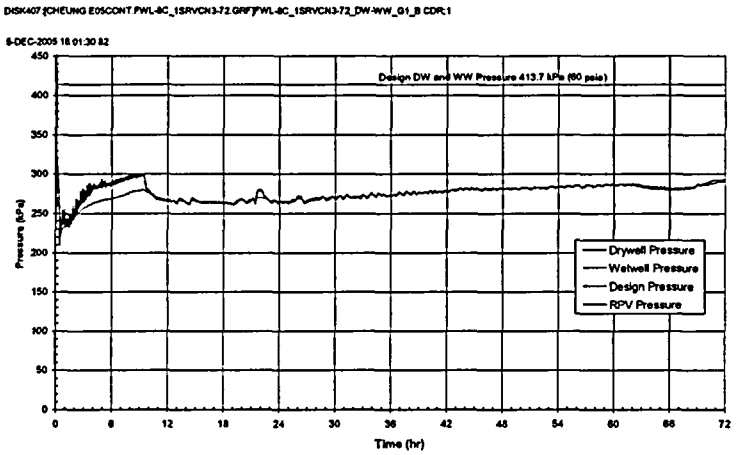

Figure 6.2-9 Containment Pressures

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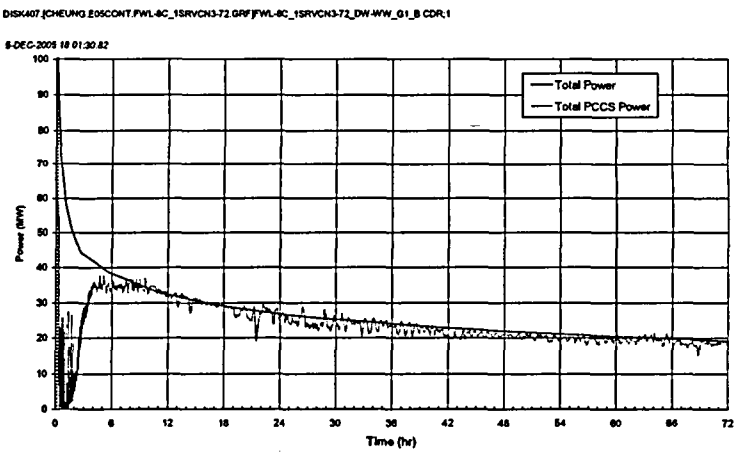

Figure 6.2-11 PCCS Heat Removal versus Decay Heat

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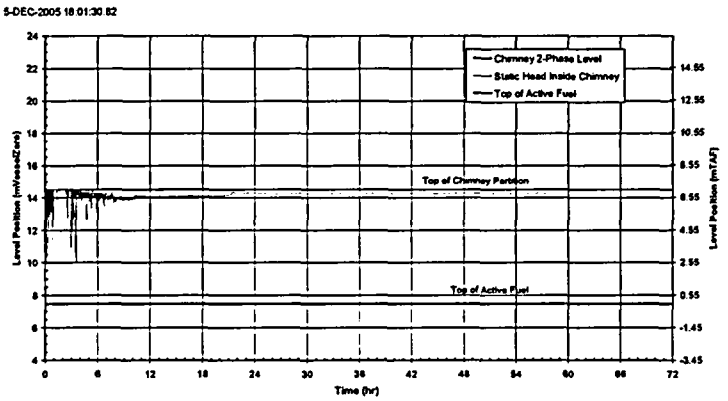
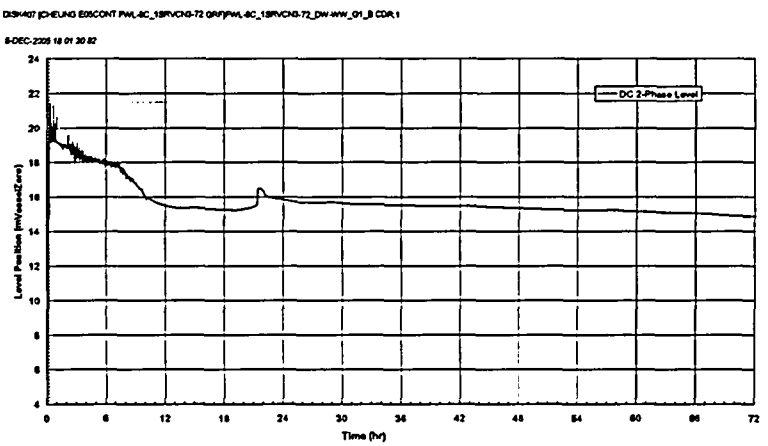
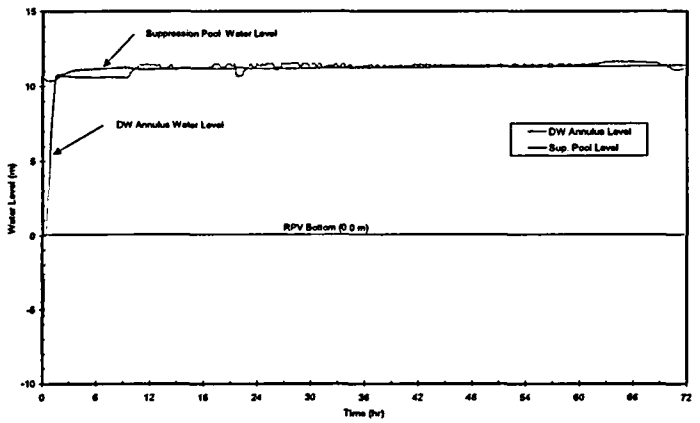


Long-term Chimney Water Level



Long-term Downcomer Water Level



Water Levels – DW Annulus and Suppression Pool

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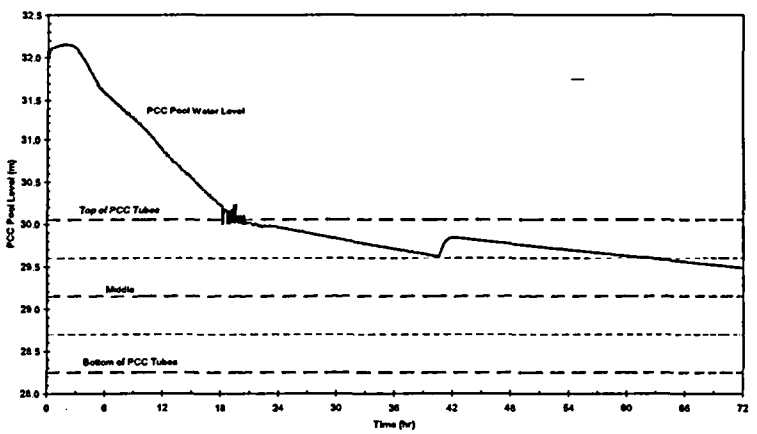

Water Levels in PCC Pool

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Table 6.2-5

Summary of Calculated Results for A Feedwater Line Break with Failure of One SRV

	Peak DW Pressure (a)	Peak DW Pressure (g)	Margin to Design Pressure of 45 psig (%)	Design Pressure (g)
Nominal Case	341 kPa (49.5 psia)	240 kPa (gauge) (34.8 psig)	23	310 kPa (gauge) (45 psig)
Bounding Case	344 kPa (49.9 psia)	243 kPa (gauge) (35.2 psig)	22	310 kPa (gauge) (45 psig)

	Short-term DW Temperature (°C)	Long-term DW Temperature (°C)	Long-term WW Temperature (°C)	Containment Design Temp. (°C)
Nominal Case	161	141	90	171
Bounding Case	165	145	94	171

Summary

ESBWR ECCS and Containment Evaluations

- Based **only** on safety systems
  - Conformance to acceptance criteria is assured by <u>water level inside shroud</u>
  - Core never uncovers during a DBA event
    - Assuring no core heatup in short- and long-term
    - Satisfying Acceptance criteria
  - Peak DW pressure below the design value; margin > 20%
    - Calculated containment temperatures below design value
- With availability of non-safety systems
  - Taking credits of one or more non-safety systems can result in additional water level and pressure margins

ESBWR Chapter 15
Safety Analysis
Discussion with NRC Staff

Wayne Marquino
April 19, 2006

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NEDE-33175, "Classification of ESBWR Abnormal Events and Determination of Their Safety Analysis Acceptance Criteria"

- Anticipated Operational Occurrences and Infrequent Events categorized on a probability basis
  - Event frequency calculations were made to justify those events that are categorized as "Infrequent Events".
  - Calculation considers failures in safety and non-safety systems in determining the event frequency.
  - Reliability improvements in ESBWR result in less severe AOO's,
  - Some legacy AOOs became Infrequent Events
- Design Basis Accidents & Special Events
  - Evaluated regardless of probability

Non-safety-related systems or components are assumed to be operational in the Chapter 15 analyses in the following situations:

(a) When the function of a non-safety-related system results in a more limiting transient

(b) When a detectable and non-consequential random, independent failure must occur in order to disable the system

(c) When non-safety-related components are used as backup protection (ie not primary success path)

AOOs, Infrequent Events and Special Events

- Organized by the effect of the events on reactor parameters
- Demonstrate how we classify events into AOO, Infrequent and Special Events per NEDE 33175
- Focus on Thermal Hydraulic analysis
  - Systems credited in Radiological Analyses have been discussed with NRC staff in previous meetings

Moderator Temperature Decrease:

- AOO, Loss of FW Heating, SCRRI run in
- Infrequent, Loss of FW Heating, SCRRI failed

Pressure Increase:

- AOOs, Load Rejection, with and w/o single failure in bypass system
- Infrequent, Load Rejection with multiple failures/no bypass
- Special Event, MSIV Closure ATWS

Moderator Temperature Decrease

- AOO (Anticipated Operational Occurrence)
  - **Loss of Feedwater Heating**

Time(s)	Event
0	Initiate a 55.6°C (100°F) temperature reduction in the FW system.
22 (est.)	RC&IS initiates Selected Control Rod Run-In
25 (est.)	Initial effect of unheated FW starts to raise core power level.
23.0 57.0 89.0	SCRRI group start insertion
123.0 157.0 189.0	SCRRI group finish insertion
300.0 (est.)	Reactor variables settle into new steady state.

Table 15.2-4 Sequence of Events for Loss of Feedwater Heating

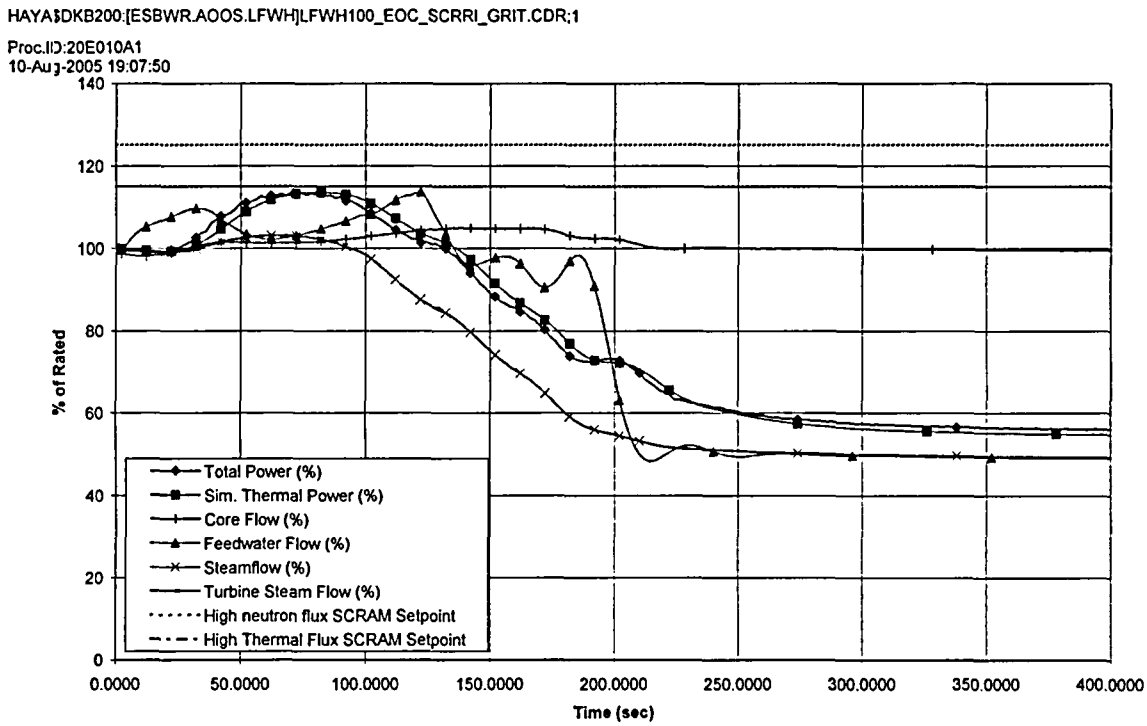

Figure 15.2-1a. Loss of Feedwater Heating

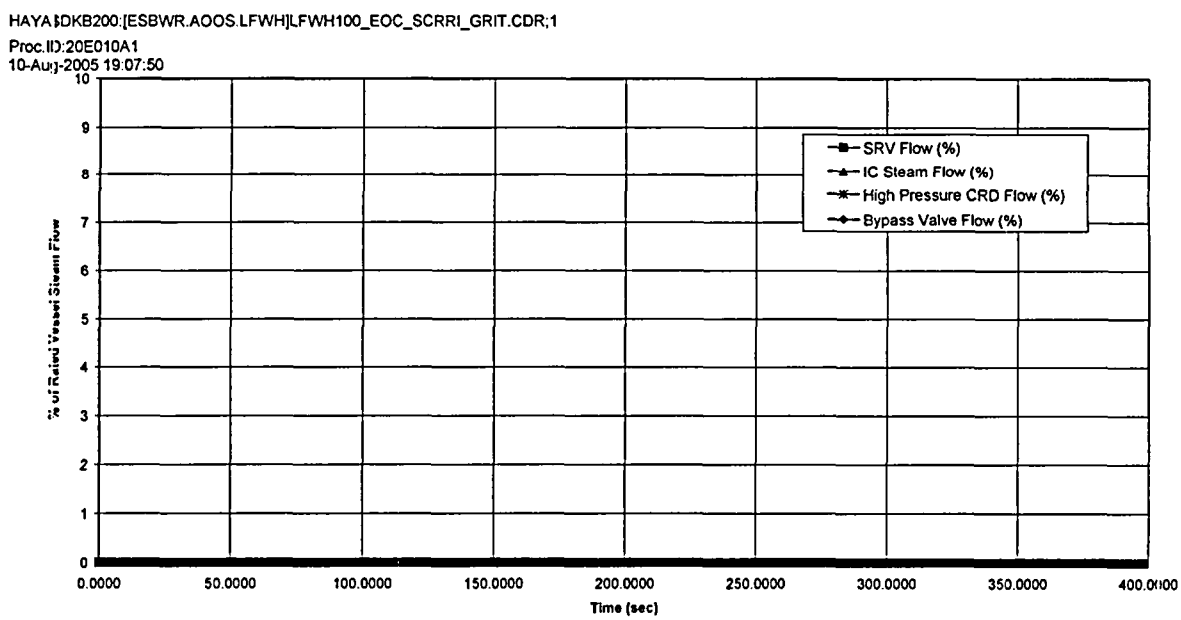

Figure 15.2-1b. Loss of Feedwater Heating

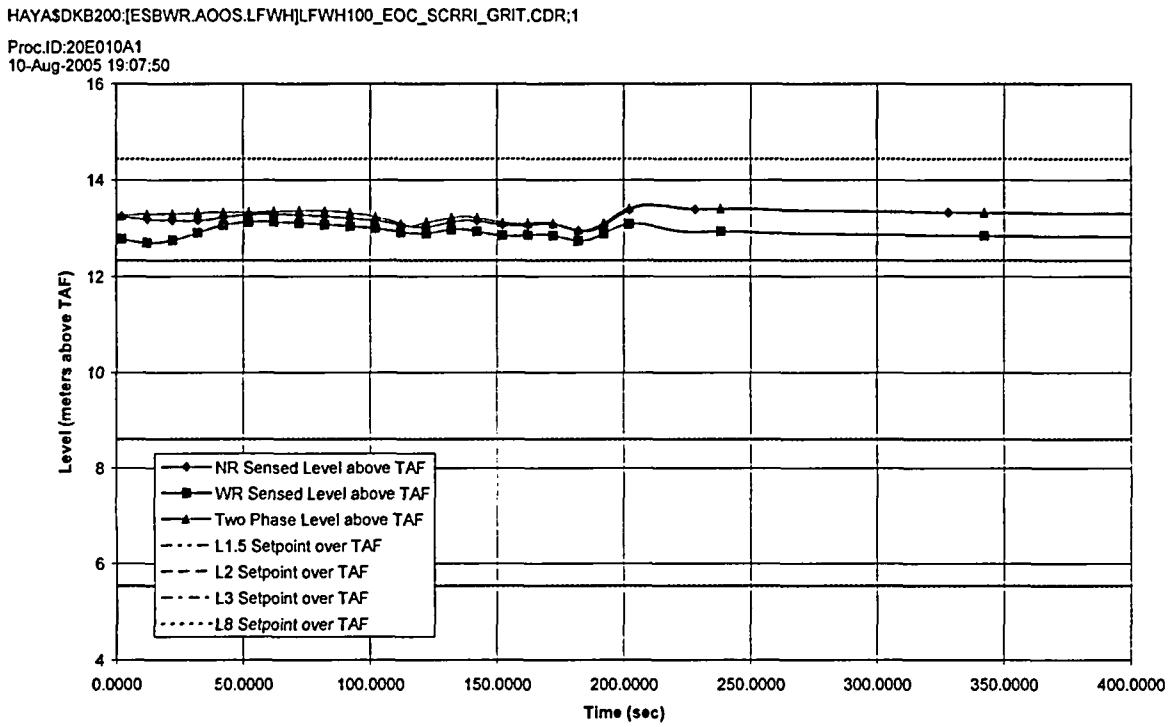

Figure 15.2-1c. Loss of Feedwater Heating

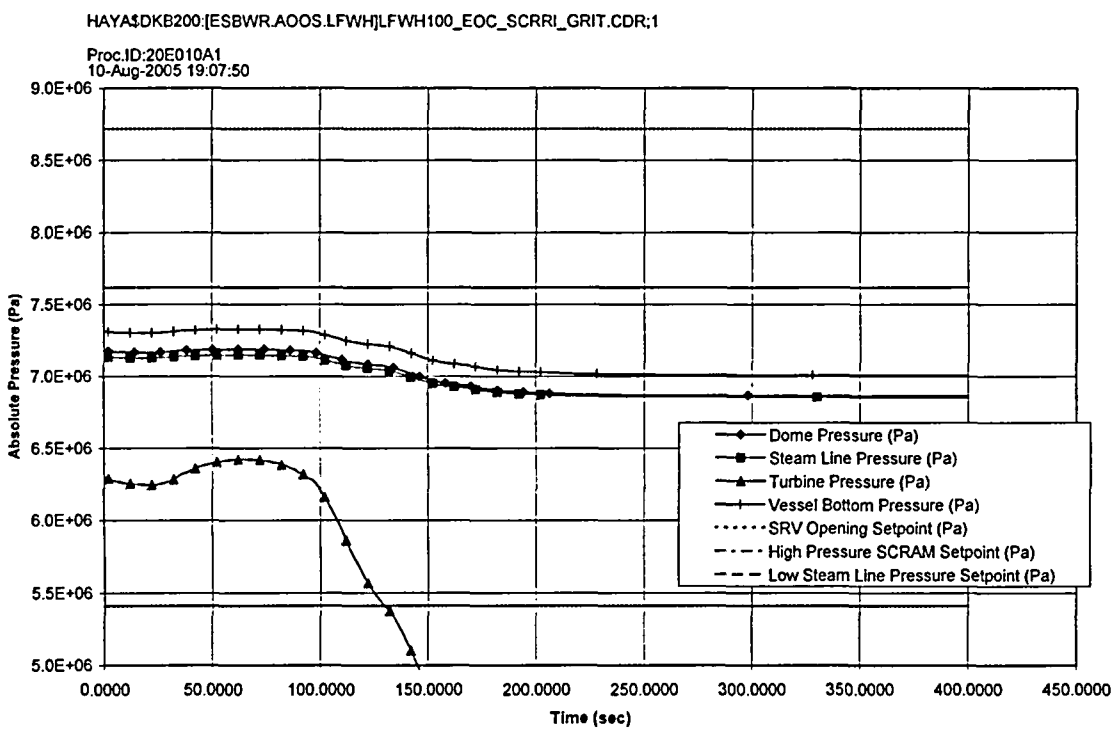

Figure 15.2-1d. Loss of Feedwater Heating

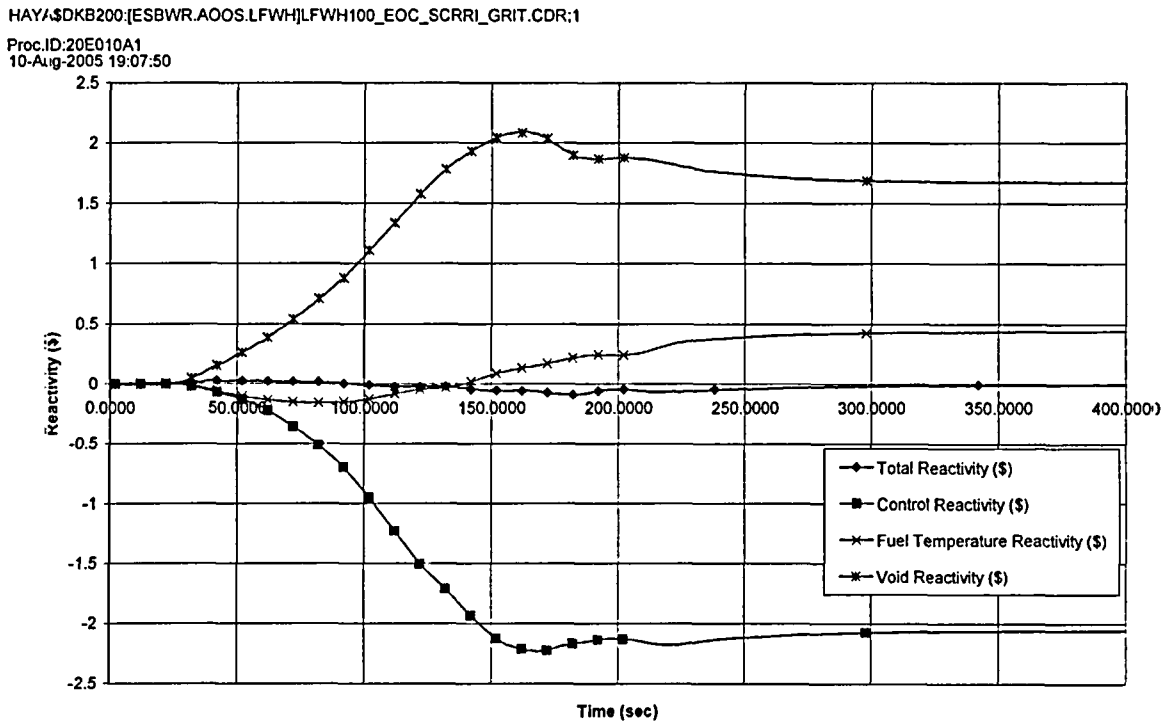

Figure 15.2-1e. Loss of Feedwater Heating

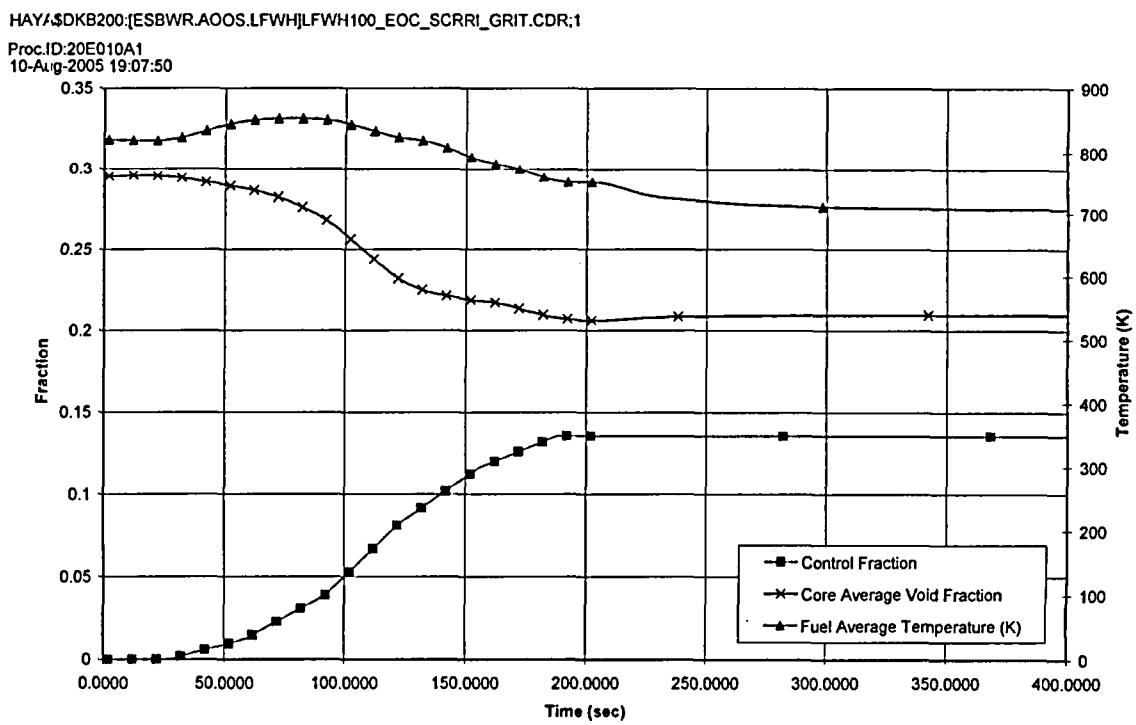

Figure 15.2-1f. Loss of Feedwater Heating

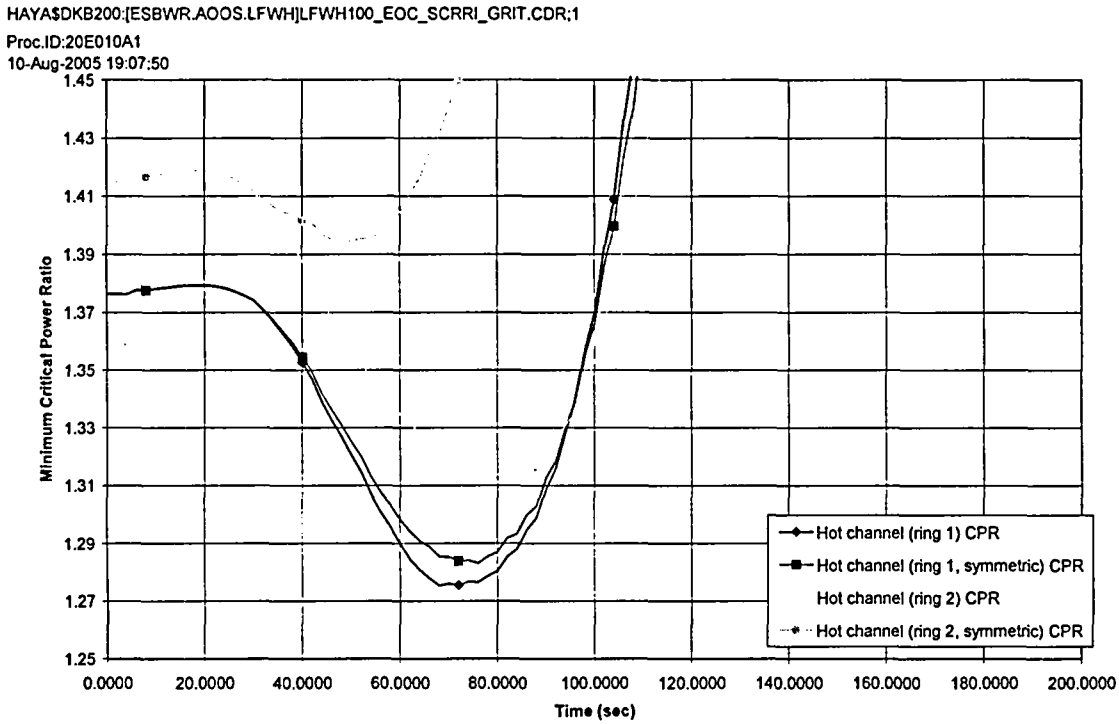

Figure 15.2-1g. Loss of Feedwater Heating

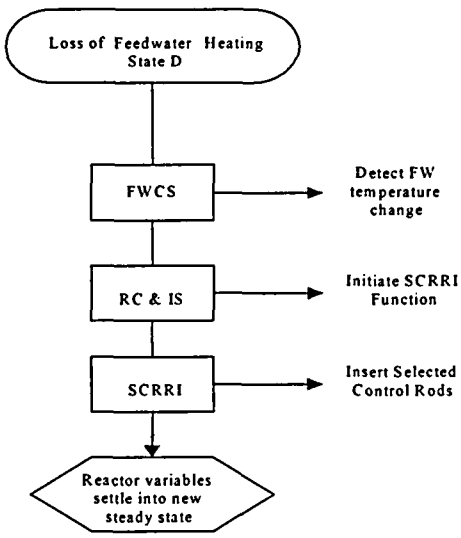

Figure 15.1-2. Event Diagram – Loss of Feedwater Heating

- Infrequent Event
  - Loss of Feedwater Heating with Failure of Selected Control Rod Run-In

Time (s)	Event*
0	Initiate a 55.6°C (100°F) temperature reduction in the FW system.
25 (est.)	Initial effect of unheated FW starts to raise core power level.
80	High thermal simulated Scram is reached but it is not credited.
300 (est.)	New Steady State Reached.

Table 15.3-2 Sequence of Events for Loss of Feedwater Heating With Failure of Selected Control Rod Run-In

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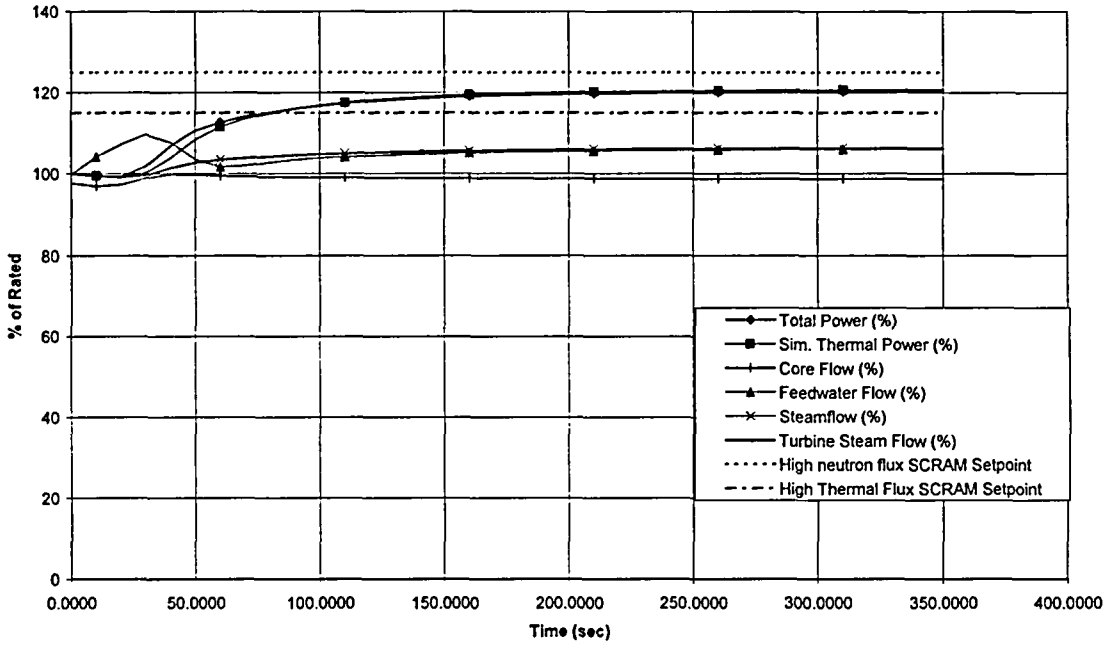

Figure 15.3-1a. Loss of Feedwater Heating with SCRRI Failure

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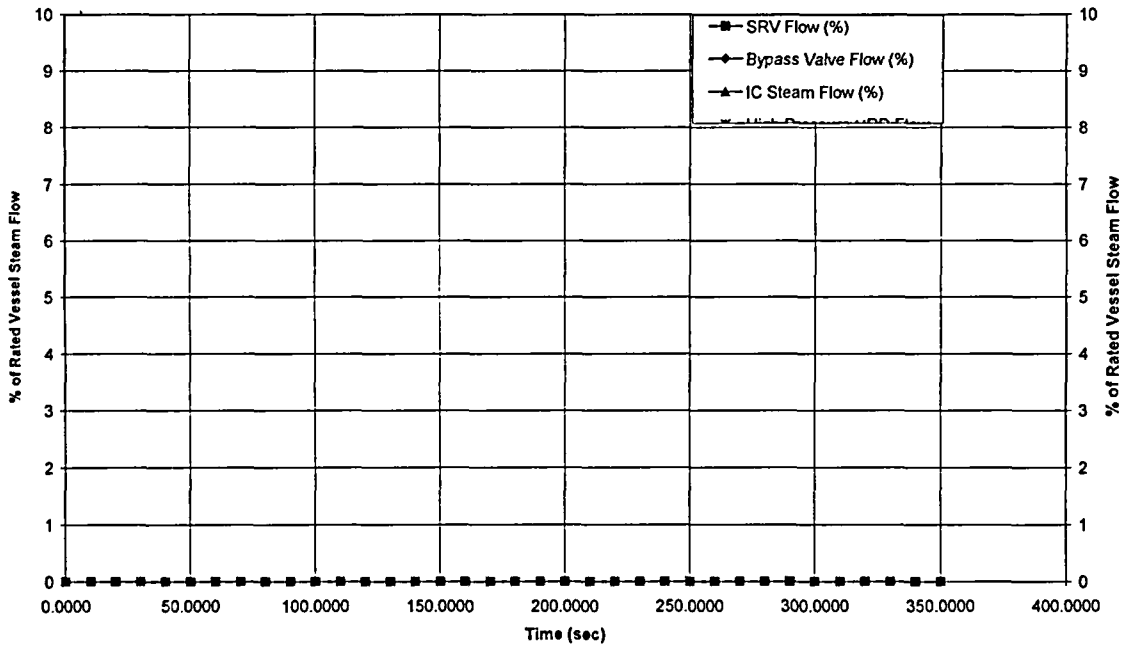

Figure 15.3-1b. Loss of Feedwater Heating with SCRRI Failure

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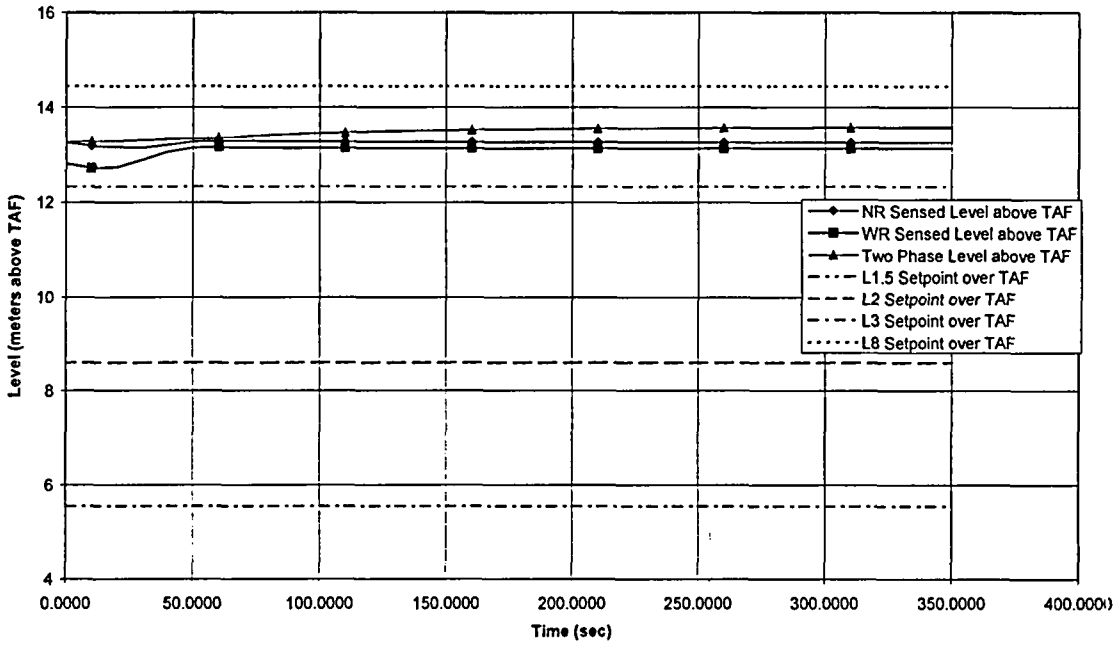

Figure 15.3-1c. Loss of Feedwater Heating with SCRRI Failure

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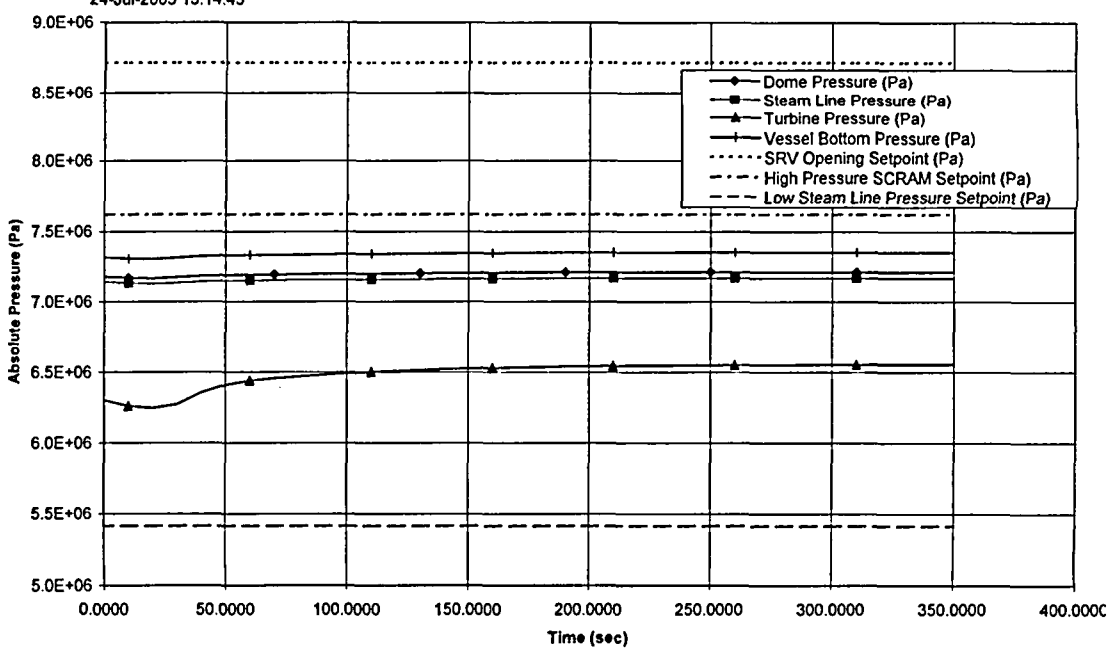

Figure 15.3-1d. Loss of Feedwater Heating with SCRRI Failure

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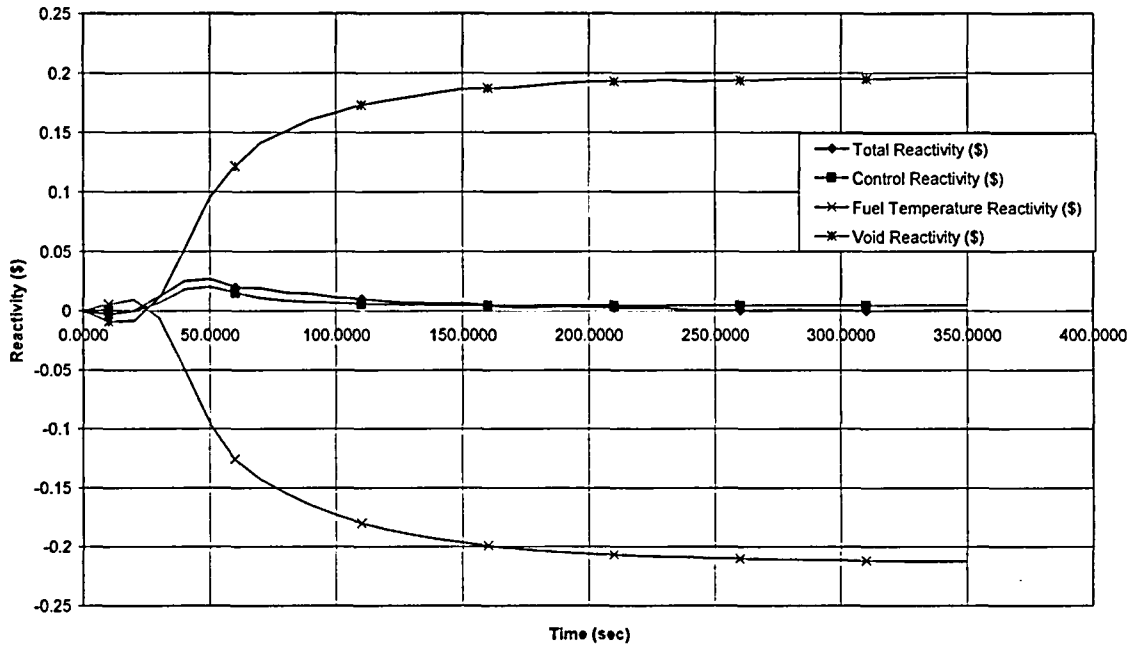

Figure 15.3-1e. Loss of Feedwater Heating with SCRRI Failure

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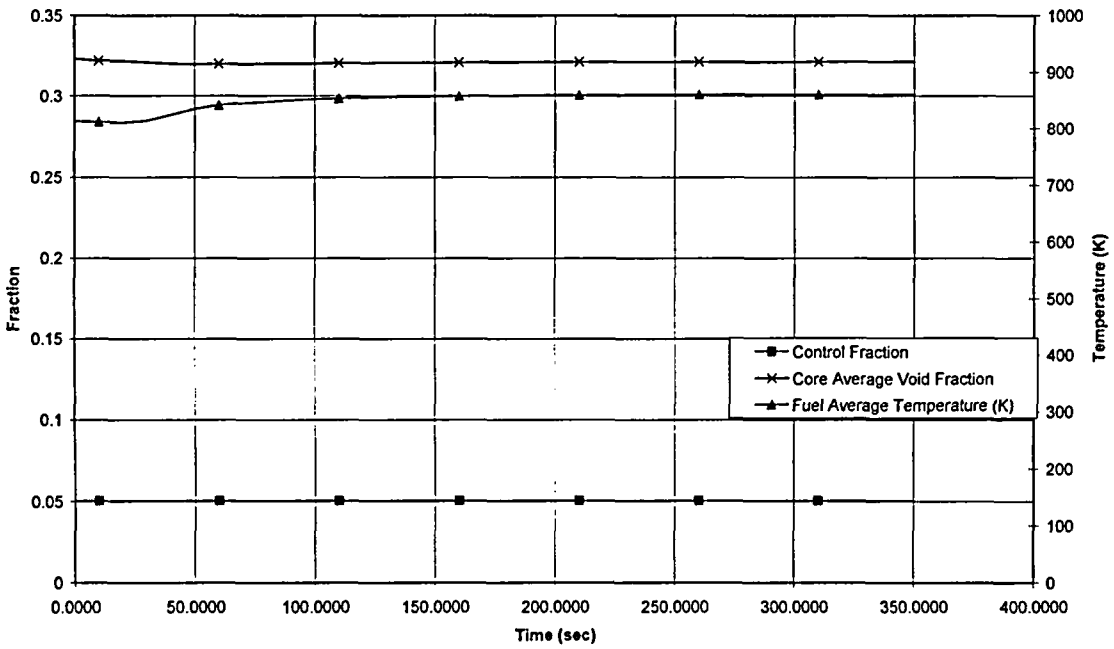

Figure 15.3-1f. Loss of Feedwater Heating with SCRRI Failure

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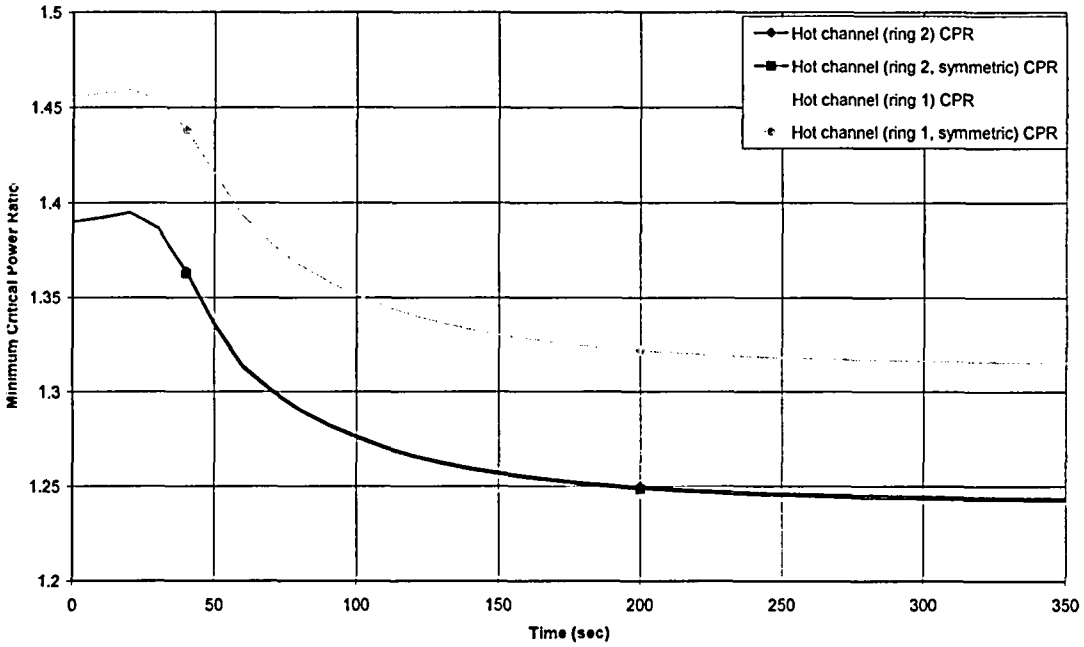


Figure 15.3-1g. Loss of Feedwater Heating with SCRRI Failure

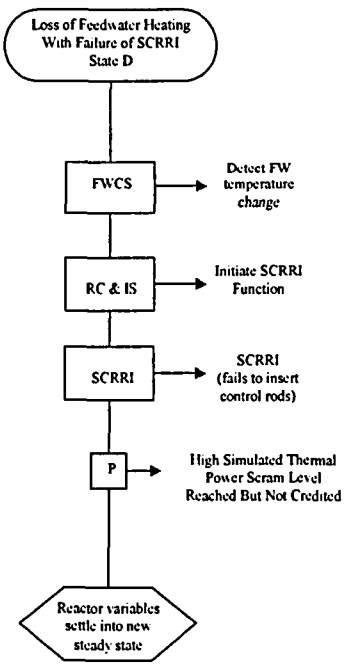


Figure 15.1-17. Event Diagram – Loss of Feedwater Heating With Failure of Selected Control Rod Run-In

Pressure Increase

- AOO
  - Generator Load Rejection with Turbine Bypass

Time (sec)	Event*
-0.015	Turbine-generator detection of loss of electrical load.
0.0	Turbine-generator load rejection sensing devices trip to initiate TCVs fast closure and main turbine bypass system operation.
0.02	Turbine bypass valves start to open.
0.10	Turbine control valves closed (Realistic closure timing).
0.22	SCRRI activated
1. 35. 68	SCRRI groups start insertion
60	FW temperature is decreasing because of loss of turbine extraction steam to FW heaters
101. 135. 168.	SCRRI groups finish insertion
300	New steady state is established

Table 15.2-8 Sequence of Events for Generator Load Rejection with Turbine Bypass

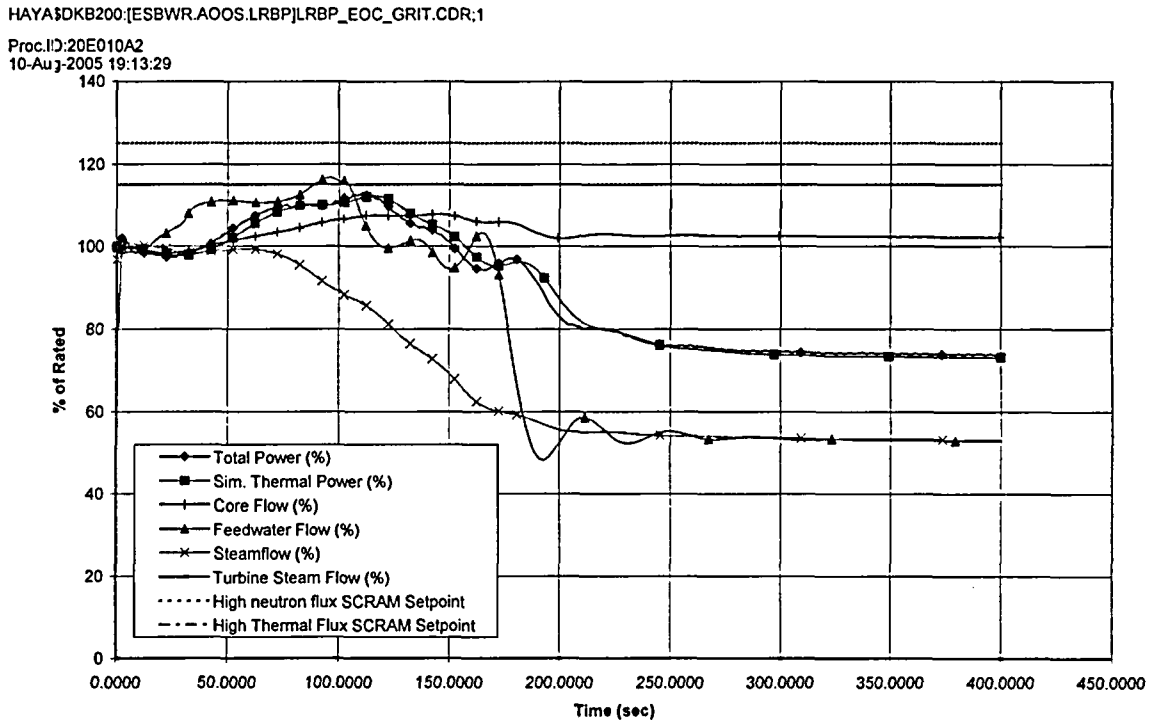


Figure 15.2-4a. Generator Load Rejection with Turbine Bypass

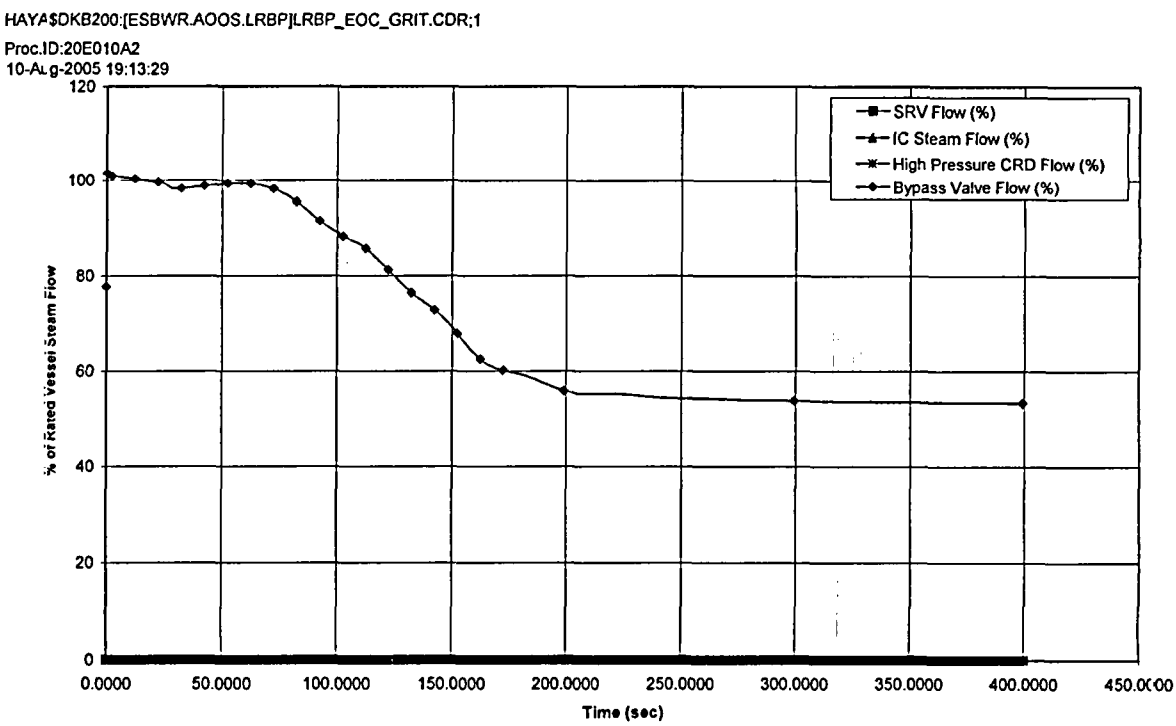


Figure 15.2-4b. Generator Load Rejection with Turbine Bypass

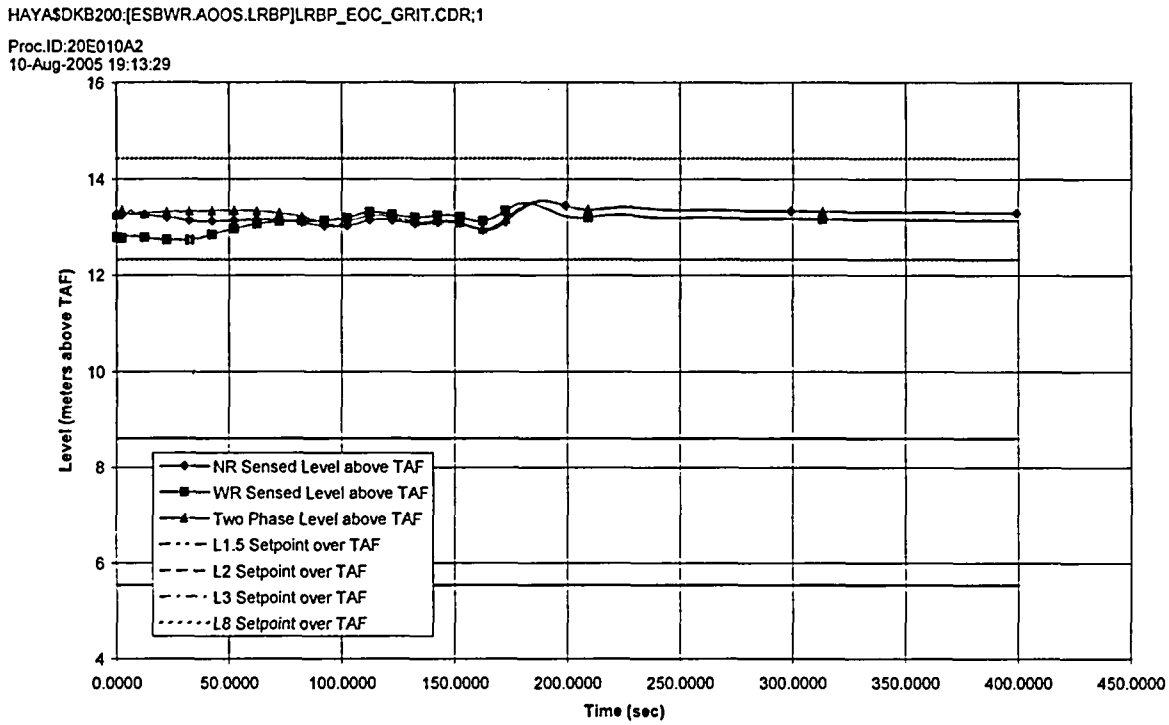


Figure 15.2-4c. Generator Load Rejection with Turbine Bypass

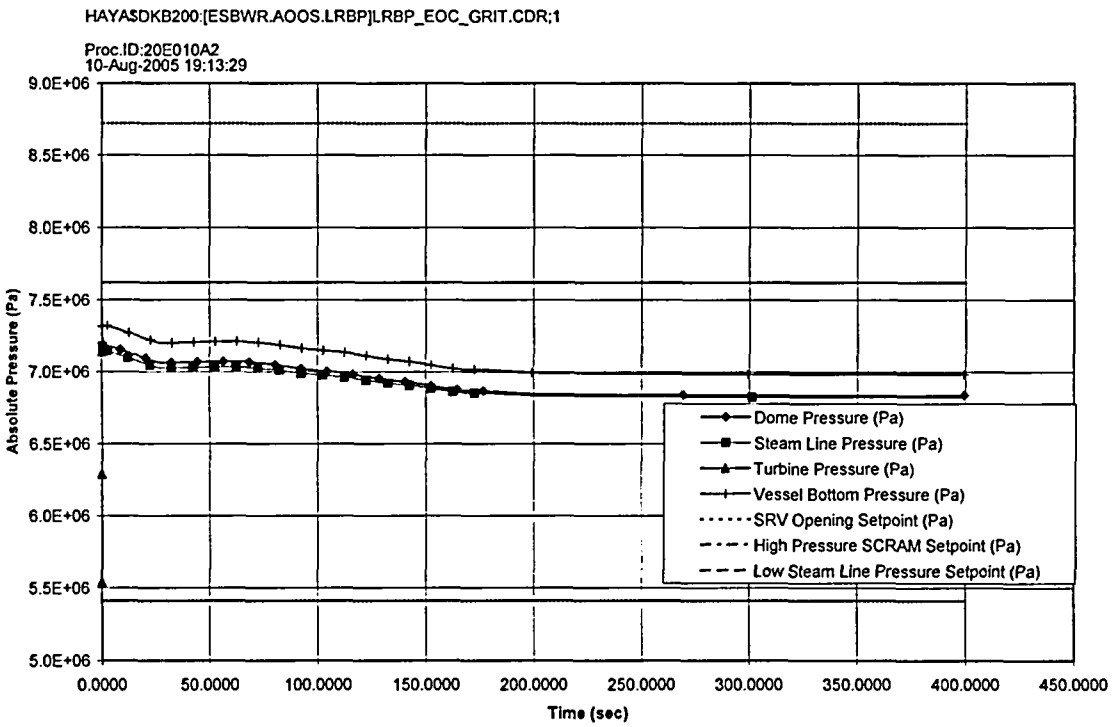


Figure 15.2-4d. Generator Load Rejection with Turbine Bypass

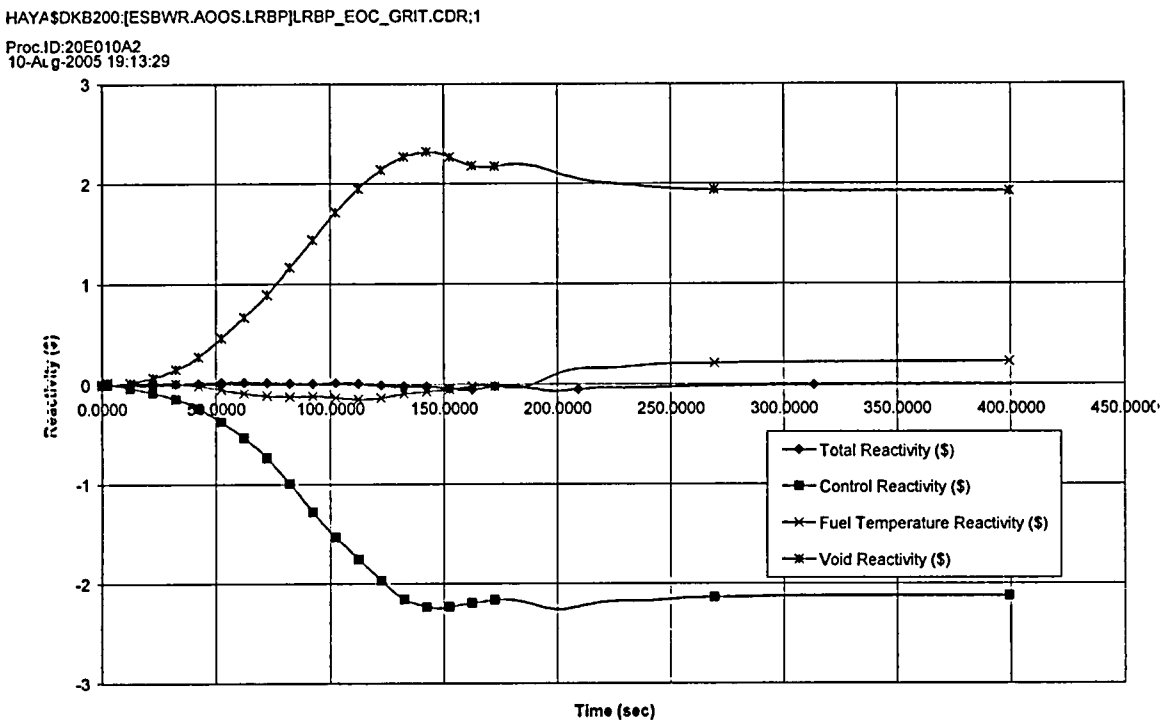


Figure 15.2-4e. Generator Load Rejection with Turbine Bypass

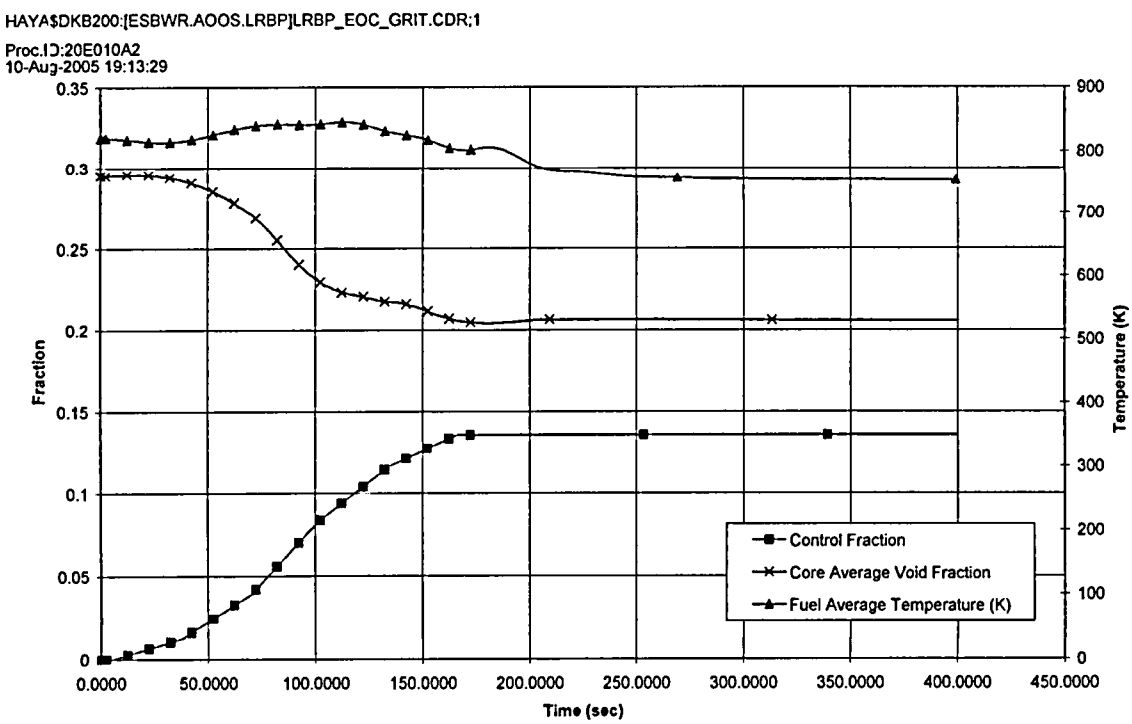


Figure 15.2-4f. Generator Load Rejection with Turbine Bypass

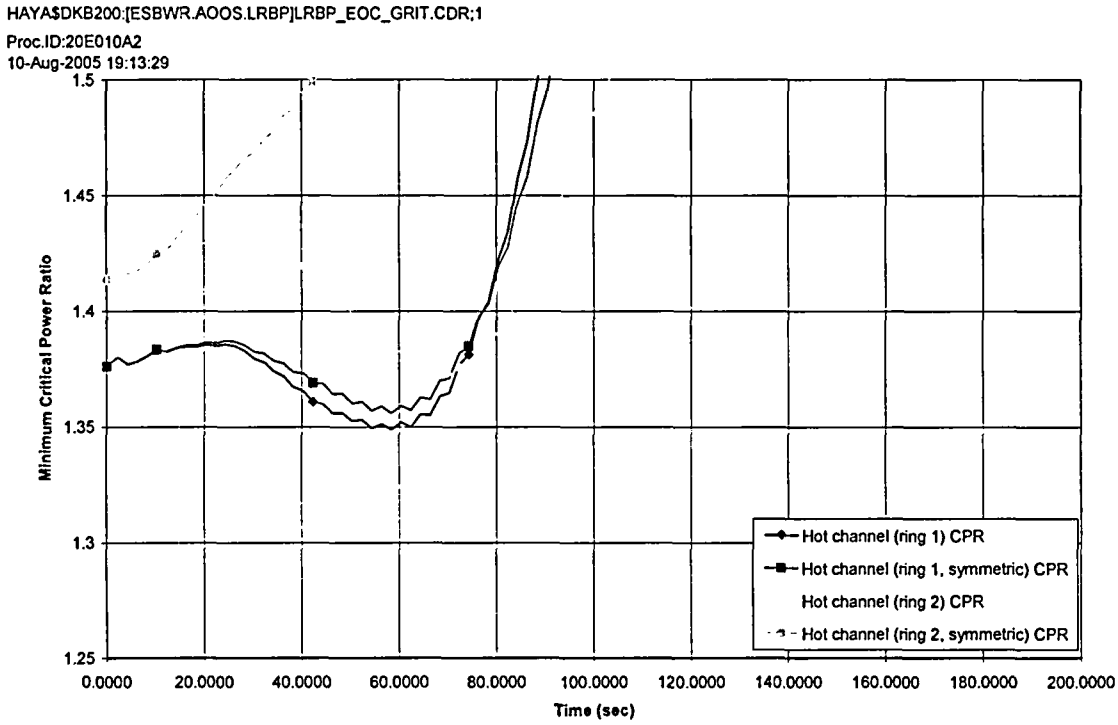


Figure 15.2-4g. Generator Load Rejection with Turbine Bypass

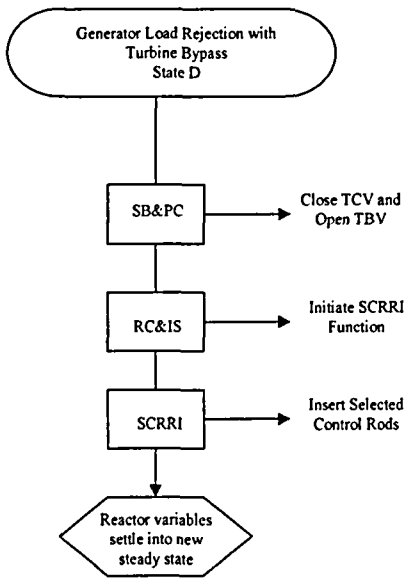


Figure 15.1-4. Event Diagram Generator Load Rejection with Turbine Bypass

AOO

- Generator Load Rejection with a Single Failure in the Turbine Bypass System

Time (sec)	Event*
-0.015	Turbine-generator detection of loss of electrical load.
0.0	Turbine-generator load rejection sensing devices trip to initiate TCVs fast closure and main turbine bypass system operation.
0.02	Turbine bypass valves start to open (Half fail to open).
0.08	Turbine control valves closed.
0.22	Not enough turbine bypass availability is detected and the plant is scrammed
0.40	Control Rods begin to enter in the core.
Long term	L2 is reached and HP_CRD is activated to recover the level

Table 15.2-9 Sequence of Events for Generator Load Rejection with a Single Failure in the Turbine Bypass System

HAYA$DKB200:[ESBWR.AOOS.LRHBP]LRHBP_EOC_GRIT.CDR;1

Proc.ID:20E00D1A
21-Jul-2005 14:56:49

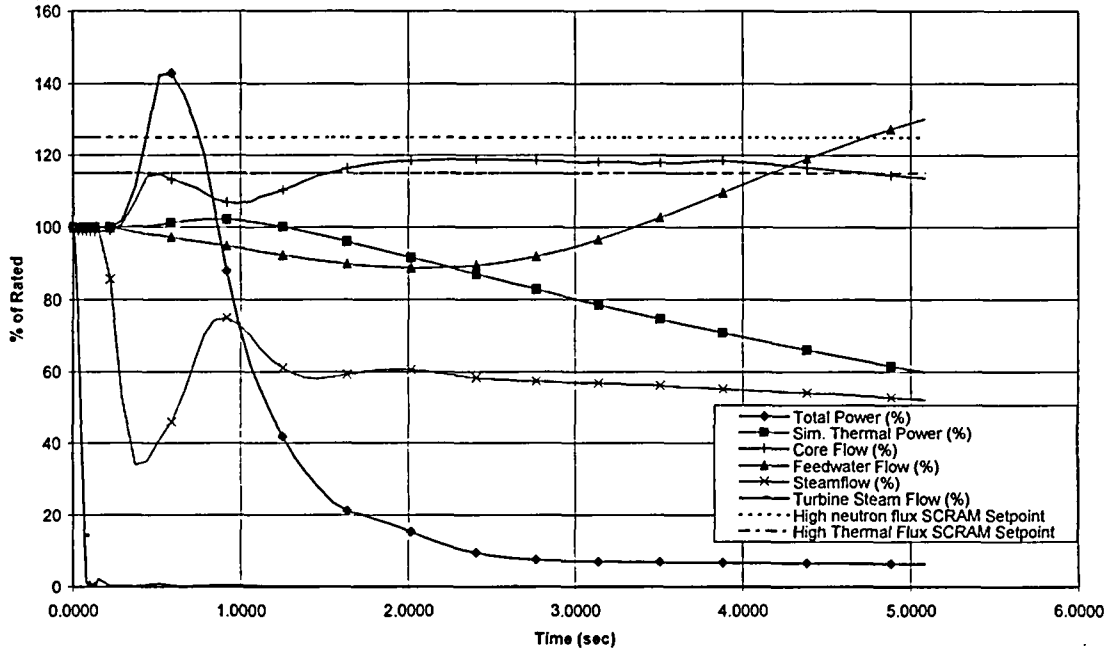


Figure 15.2-5a. Generator Load Rejection with a Single Failure in the Turbine Bypass System

HAYA$DKB200:[ESBWR.AOOS.LRHBP]LRHBP_EOC_GRIT.CDR;1

Proc.ID:20E00D1A
21-Jul-2005 14:56:49

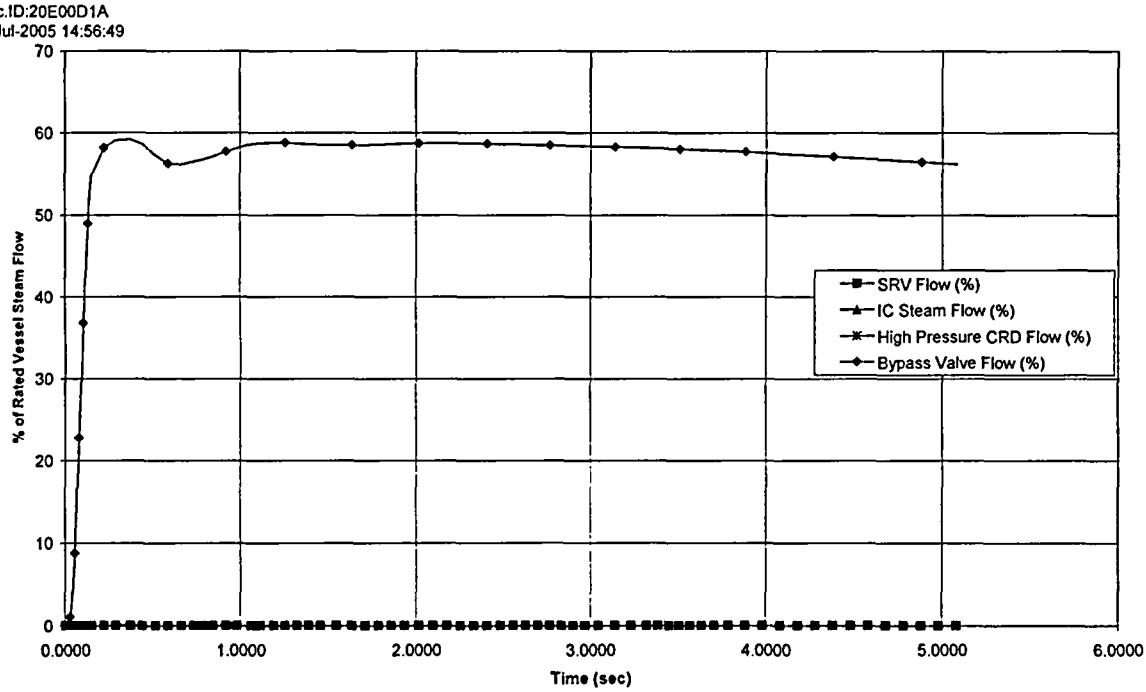


Figure 15.2-5b. Generator Load Rejection with a Single Failure in the Turbine Bypass System

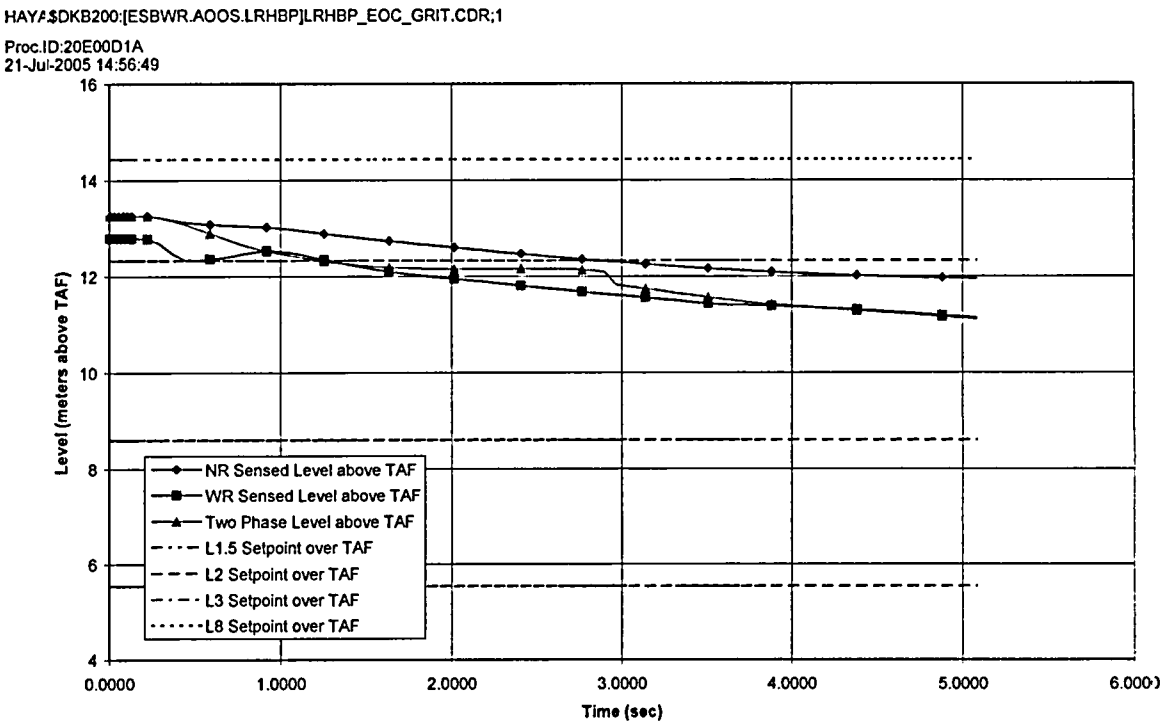


Figure 15.2-5c. Generator Load Rejection with a Single Failure in the Turbine Bypass System

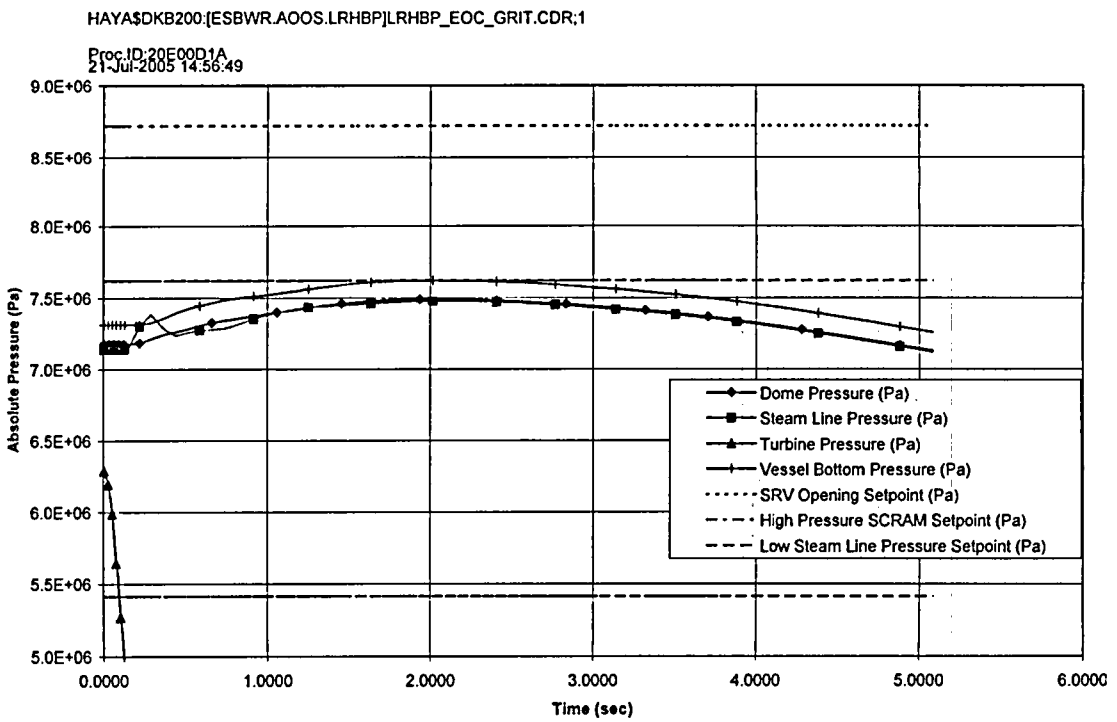


Figure 15.2-5d. Generator Load Rejection with a Single Failure in the Turbine Bypass System

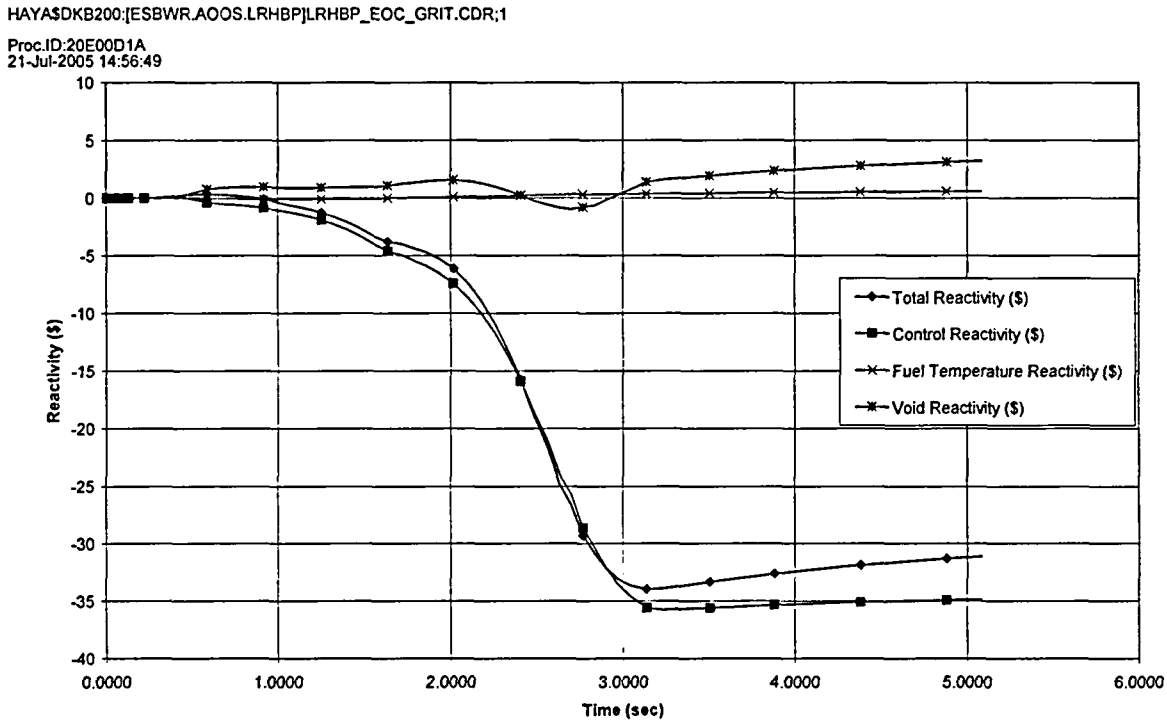


Figure 15.2-5e. Generator Load Rejection with a Single Failure in the Turbine Bypass System

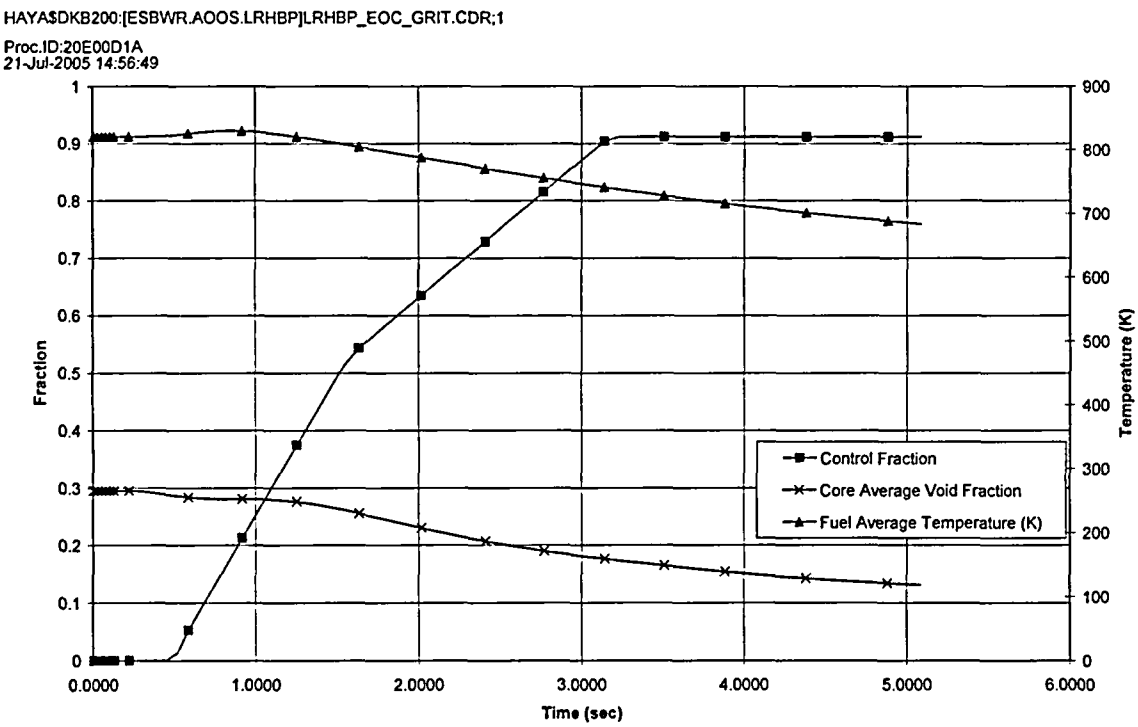


Figure 15.2-5f. Generator Load Rejection with a Single Failure in the Turbine Bypass System

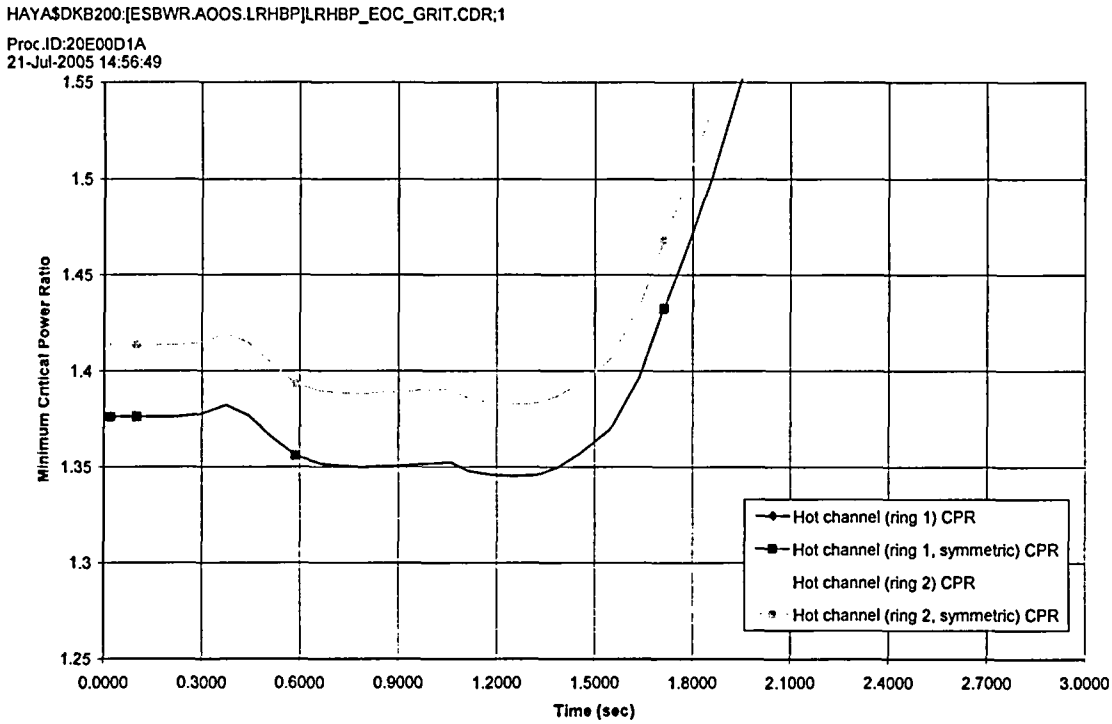


Figure 15.2-5g. Generator Load Rejection with a Single Failure in the Turbine Bypass System

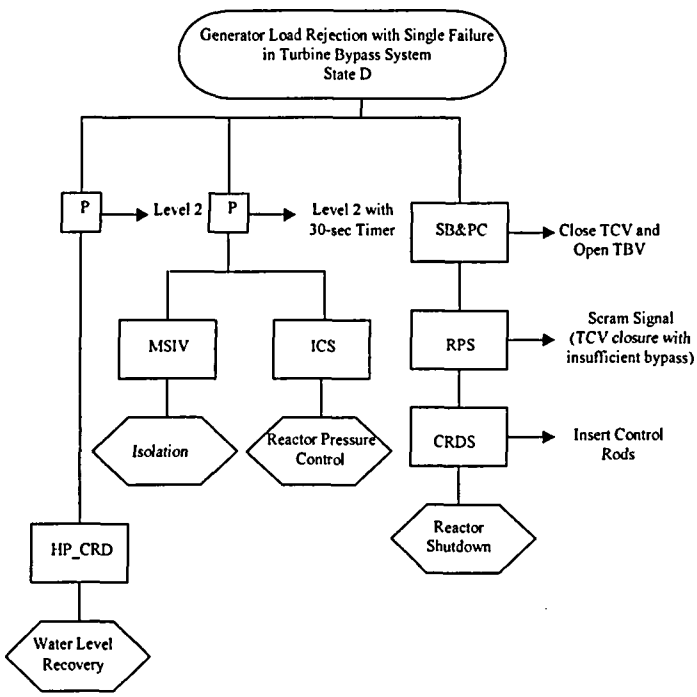


Figure 15.1-5. Event Diagram
Generator Load Rejection with a Single Failure in the Turbine Bypass System

- Infrequent Event
  - Generator Load Rejection With Total Turbine Bypass Failure

Time (sec)	Event*
(-)0.015	Turbine-generator detection of loss of electrical load.
0.0	Turbine-generator load rejection sensing devices trip to initiate turbine control valves fast closure.
0.0	Turbine bypass valves fail to operate.
0.08	Turbine control valves closed.
0.15	After detection of not enough bypass availability the RPS initiates a reactor scram.
0.40	The rods begin to enter inside the core
Long term	HP_CRD is activated on L2 to recover the level

Table 15.3-6
Sequence of Events for Generator Load Rejection With Total Turbine Bypass Failure

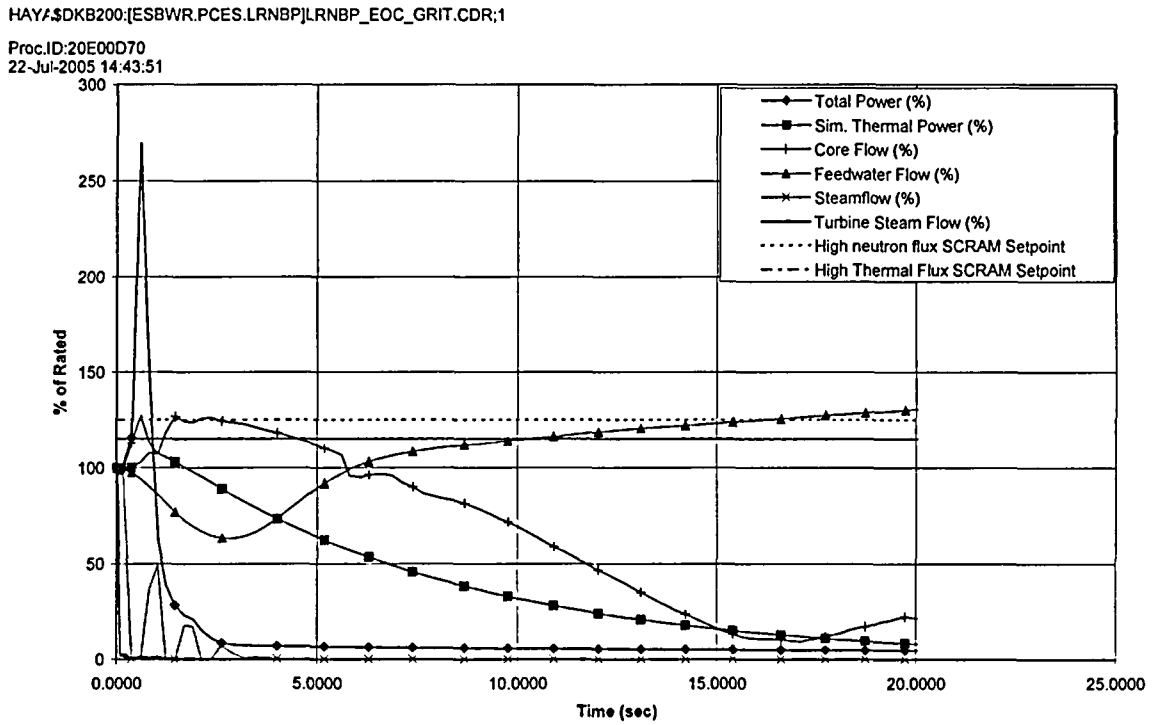


Figure 15.3-5a. Generator Load Rejection With Total Turbine Bypass Failure

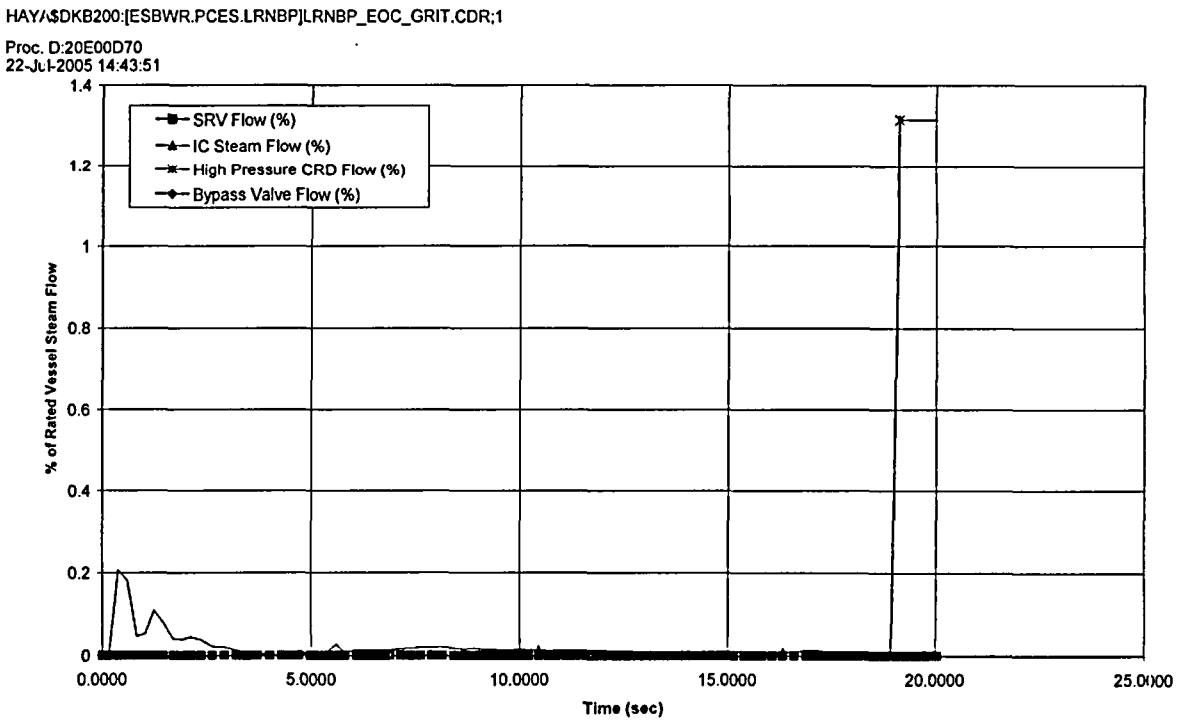


Figure 15.3-5b. Generator Load Rejection With Total Turbine Bypass Failure

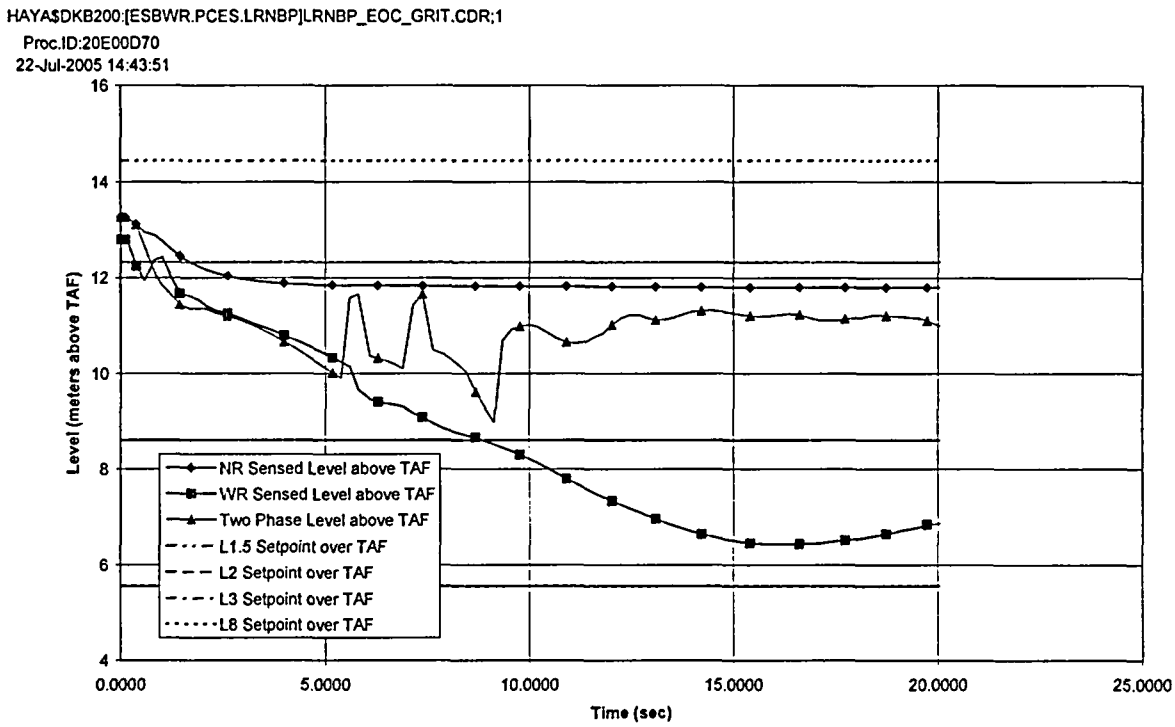


Figure 15.3-5c. Generator Load Rejection With Total Turbine Bypass Failure

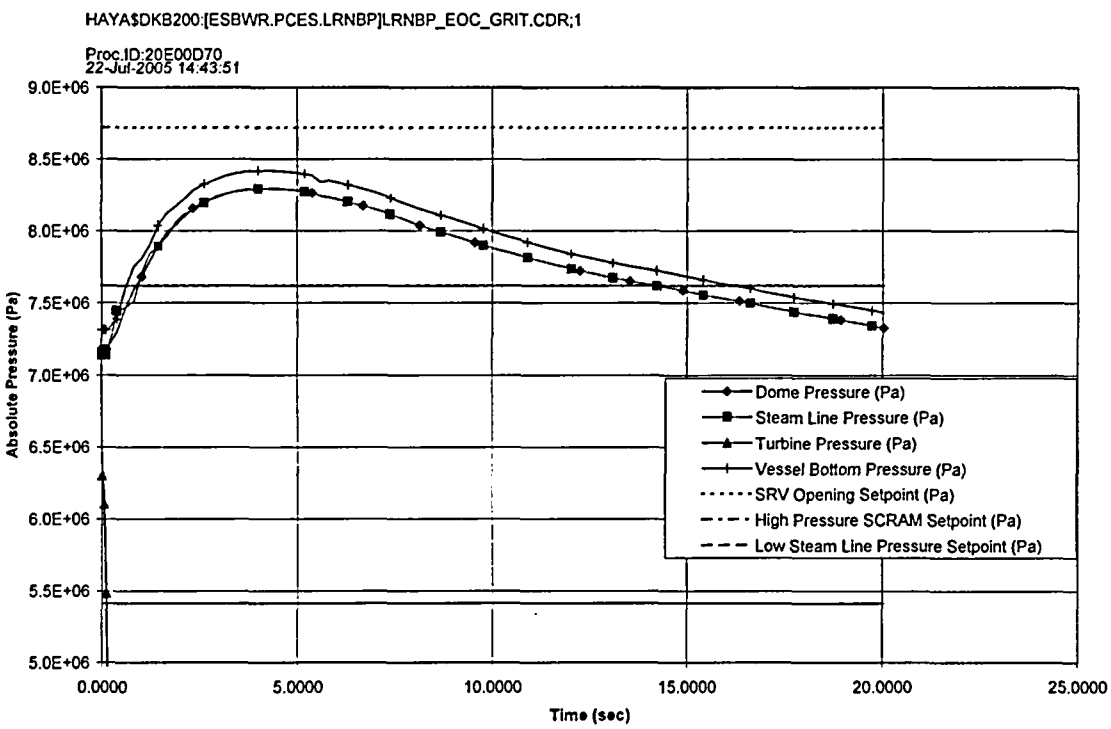


Figure 15.3-5d. Generator Load Rejection With Total Turbine Bypass Failure

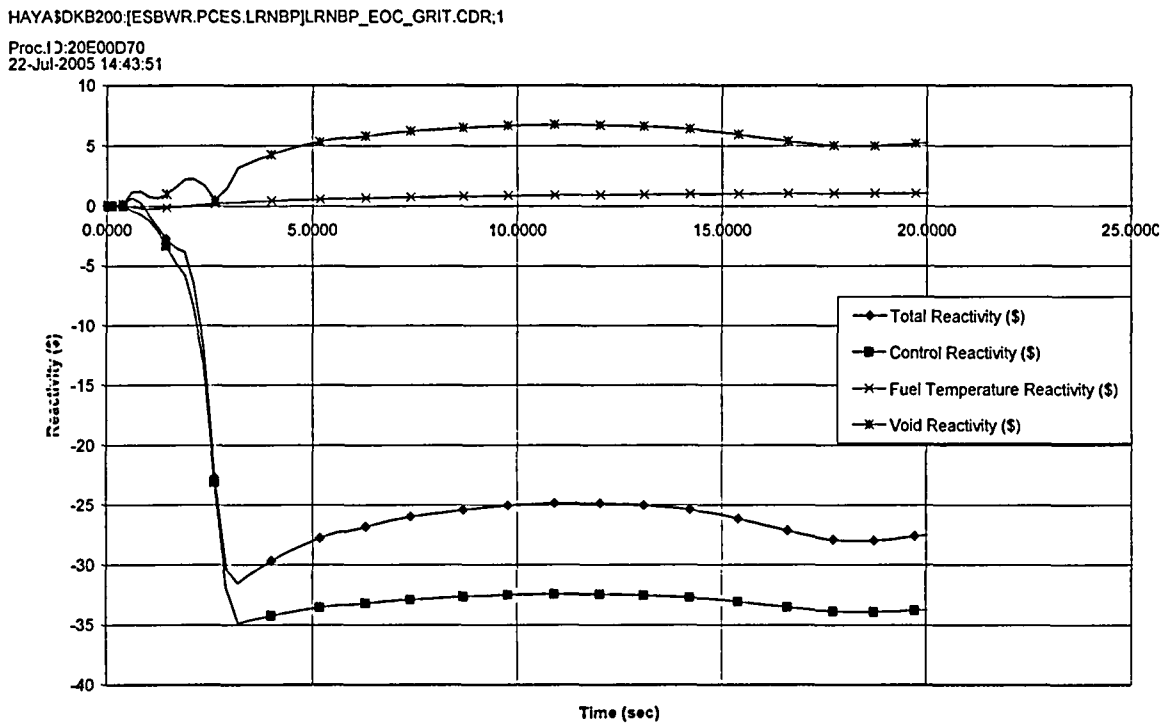


Figure 15.3-5e. Generator Load Rejection With Total Turbine Bypass Failure

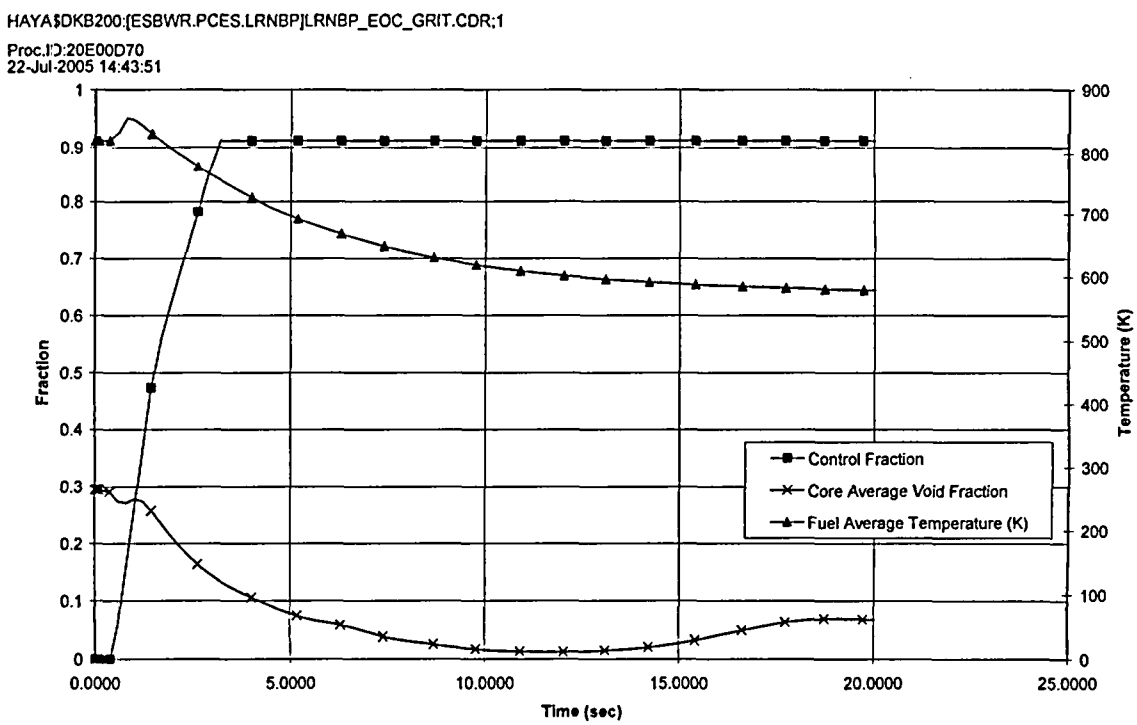


Figure 15.3-5f. Generator Load Rejection With Total Turbine Bypass Failure

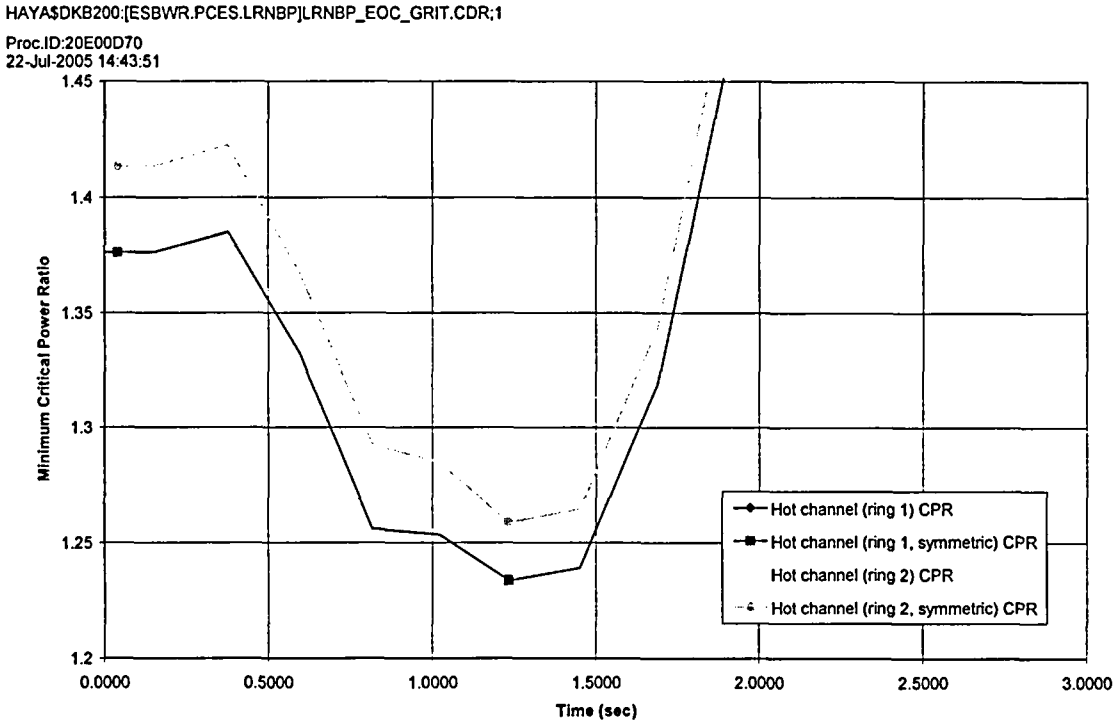


Figure 15.3-5g. Generator Load Rejection With Total Turbine Bypass Failure

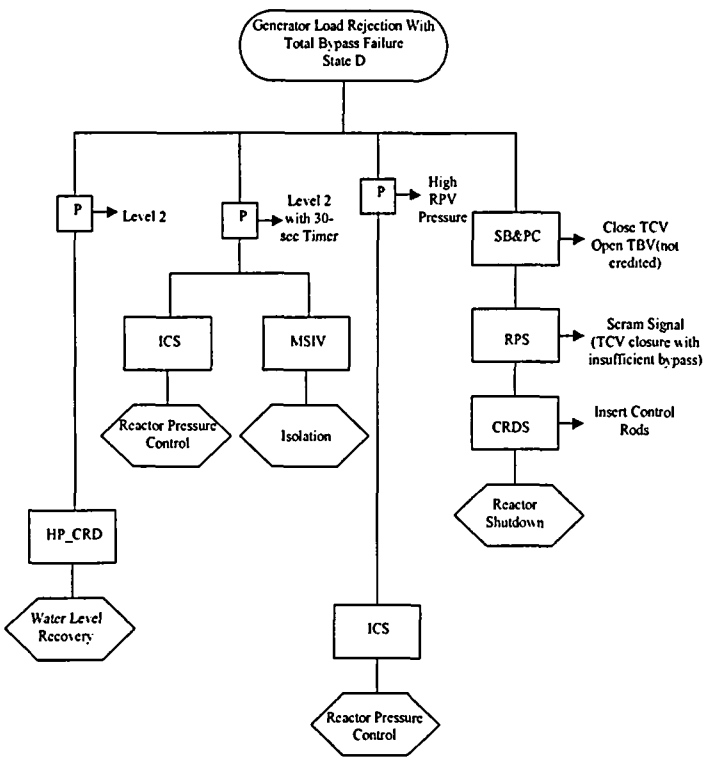


Figure 15.1-21. Event Diagram – Generator Load Rejection with Total Bypass Failure

Special Event

--MSIV Closure ATWS

Parameter	Value	Time
Maximum Neutron Flux, %	228	3s
Maximum Vessel Bottom Pressure, MPaG (psig)	9.67 (1402.8)	20s
Maximum Bulk Suppression Pool Temperature, °C (°F)	54.0 (129.1)	37s
Associated Containment Pressure, kPaG (psig)	147.0 (21.26)	37s
Peak Cladding Temperature, °C (°F)	940.6 (1725)	25s

Table 15.5-4a
ATWS MSIV Closure Summary - ARI Case

DISK211 JCBOTT ESBWR ATWS CAT2JARI_R11_DCD CDR.1
Proc ID 20205E24
18-AUG-2005 14 55 22 05

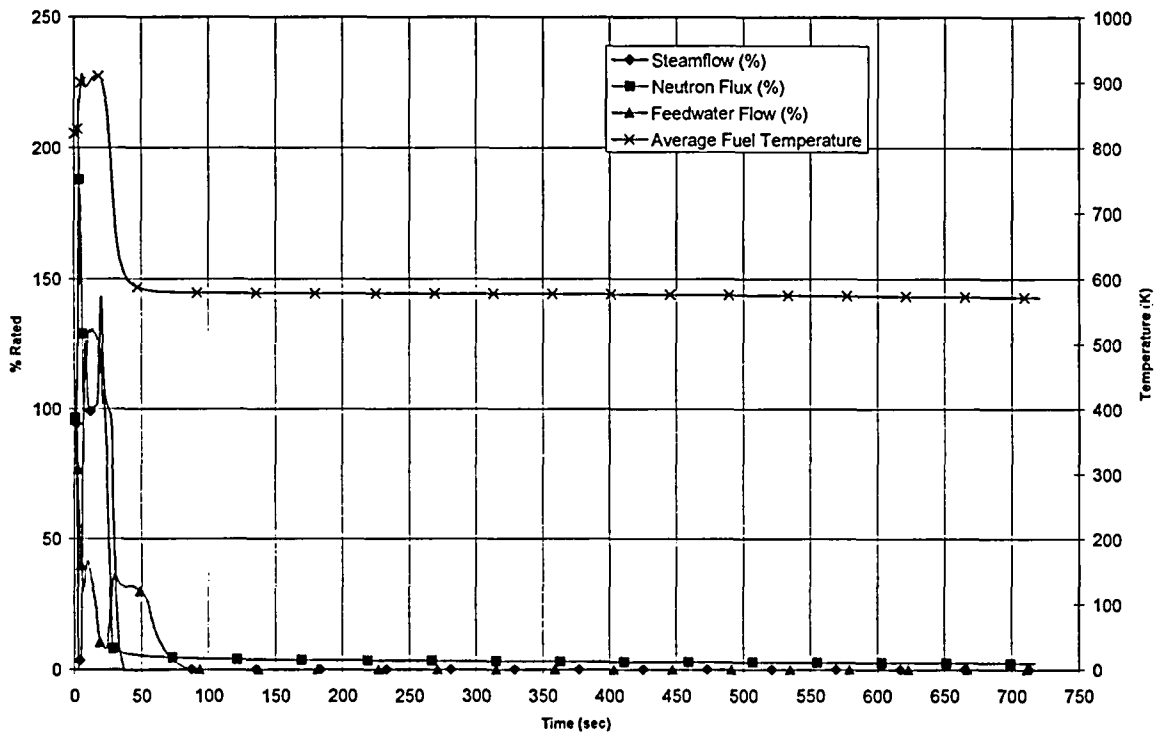


Figure 15.5-1a. MSIV Closure with ARI

DISK211 JCBOTT ESBWR ATWS CAT2JARI_R11_DCD CDR.1
Proc ID 20205E24
18-AUG-2005 14 55 22 05

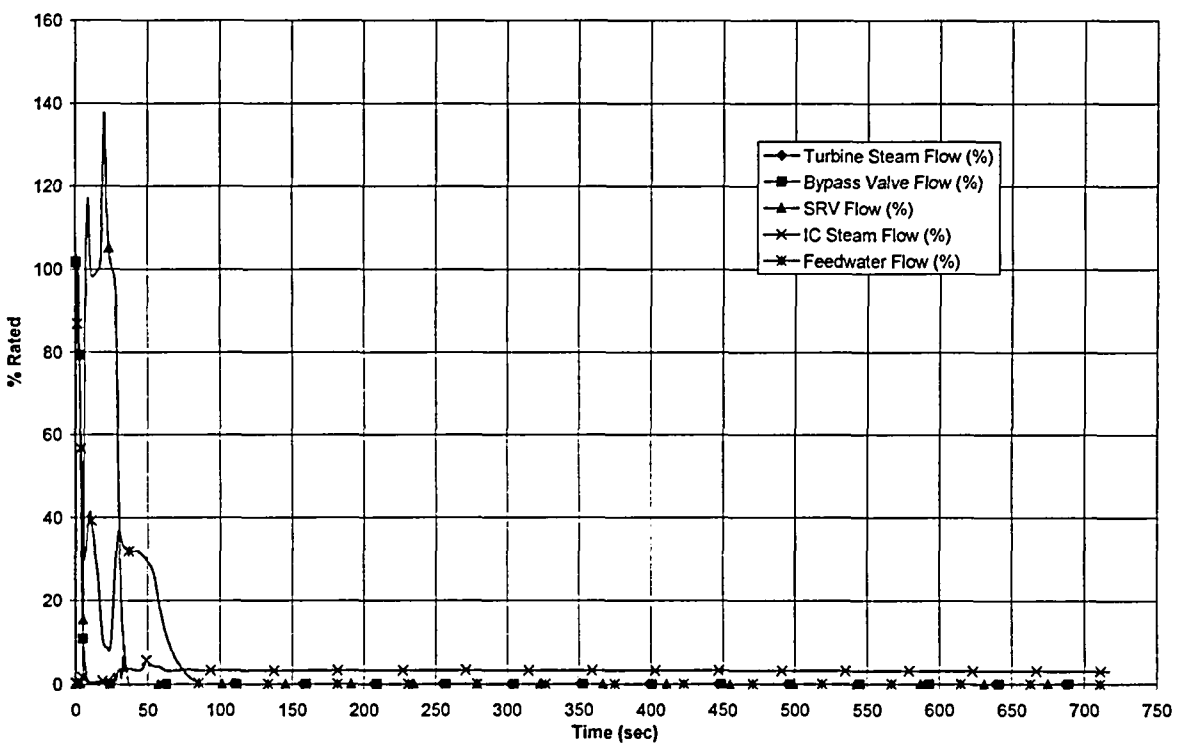


Figure 15.5-1a. MSIV Closure with ARI

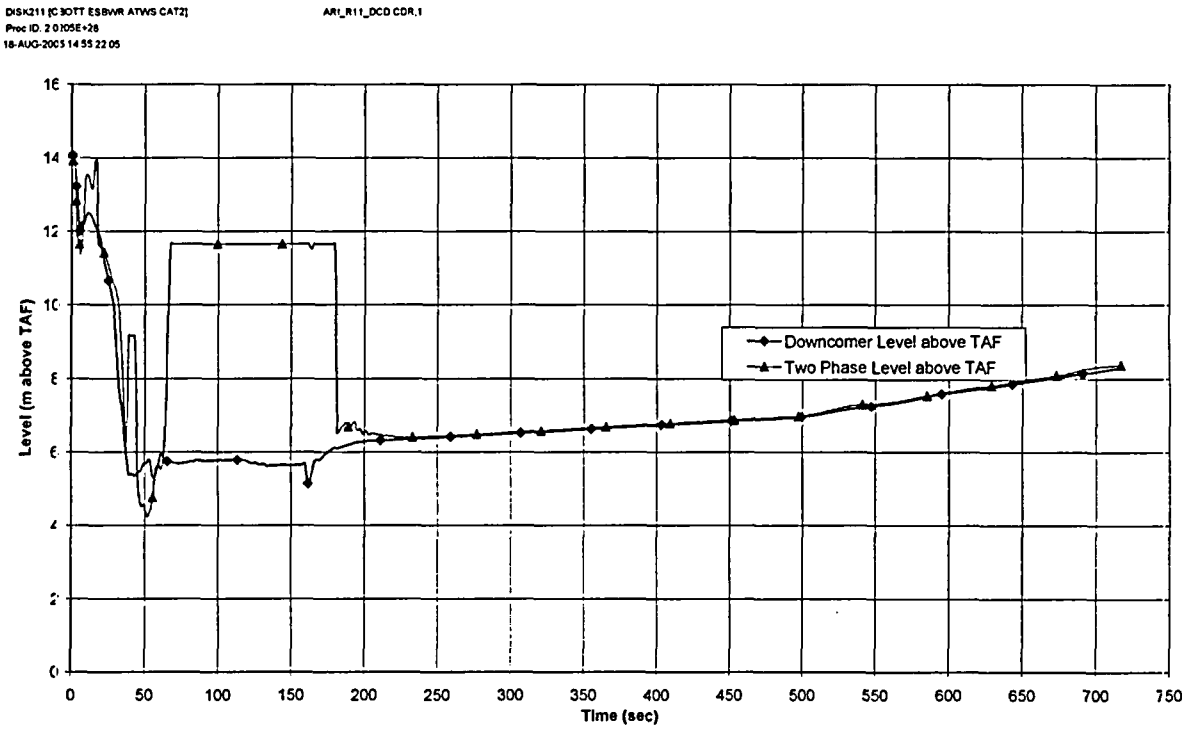


Figure 15.5-1b. MSIV Closure with ARI

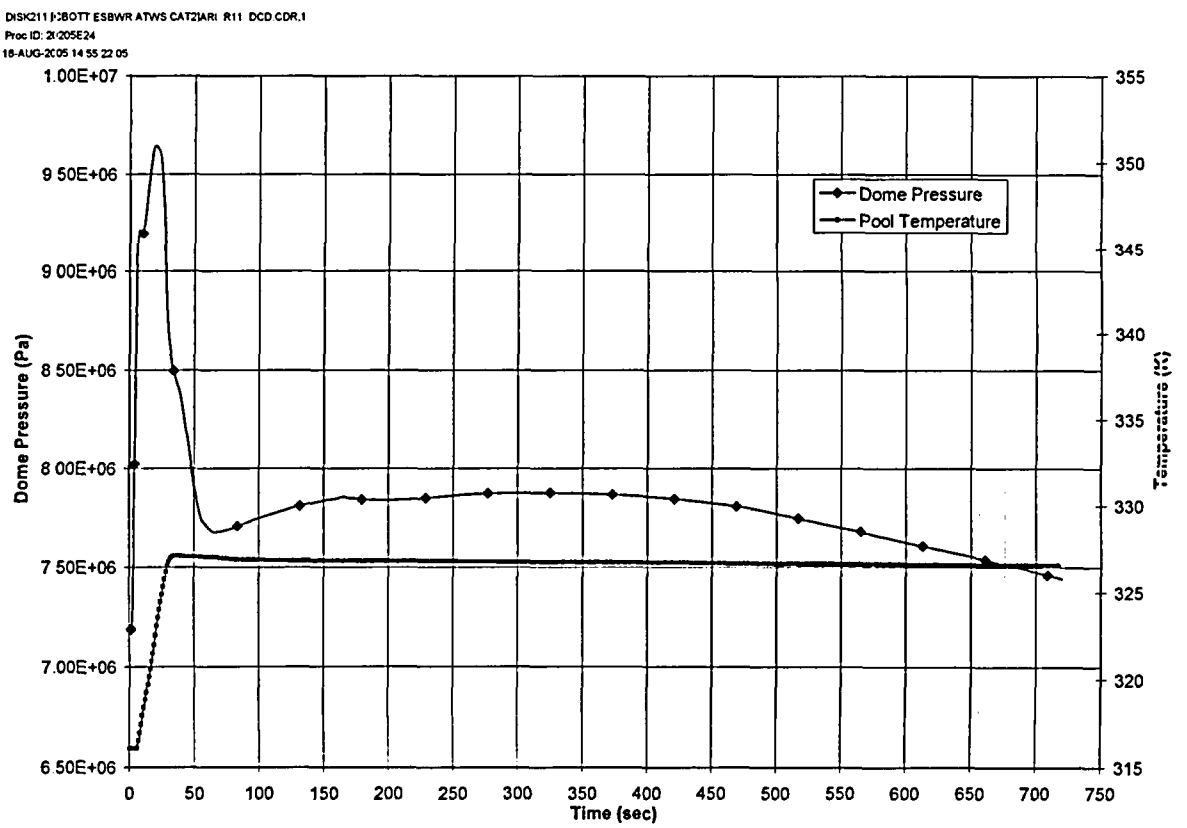


Figure 15.5-1b. MSIV Closure with ARI

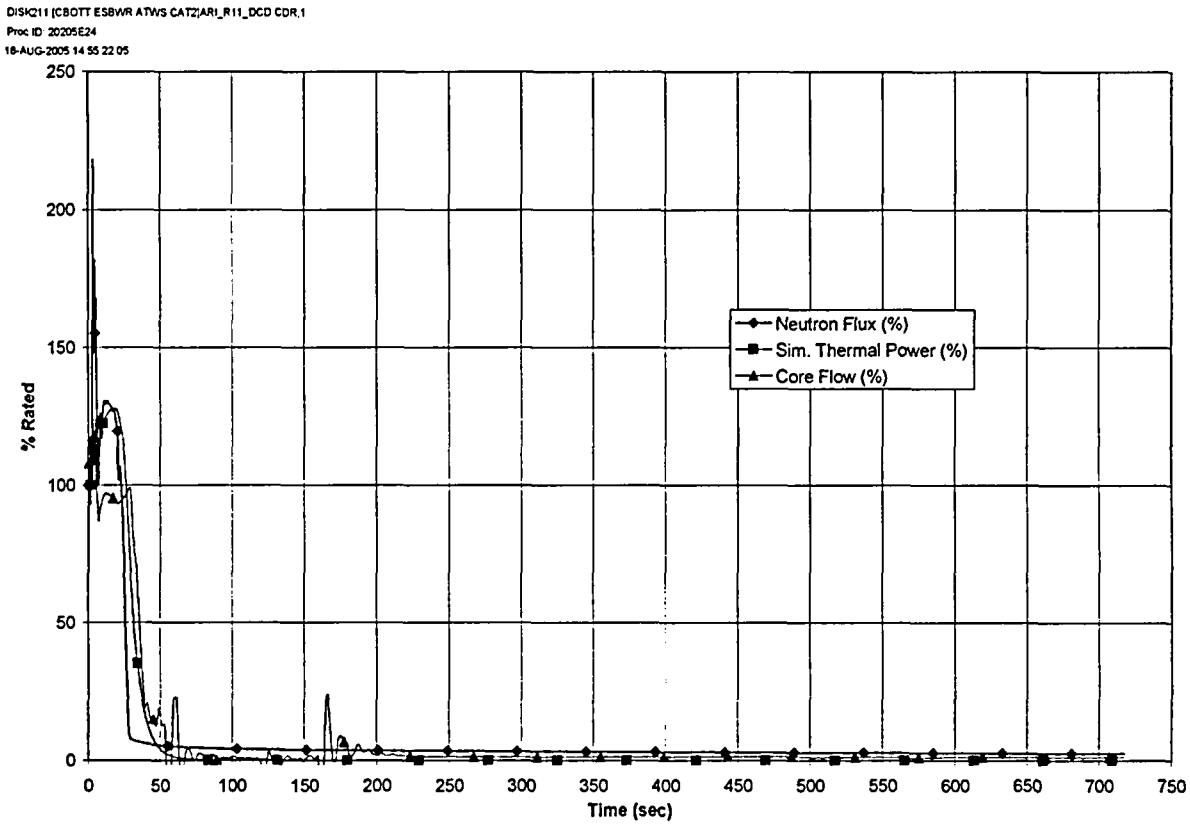


Figure 15.5-1c. MSIV Closure with ARI

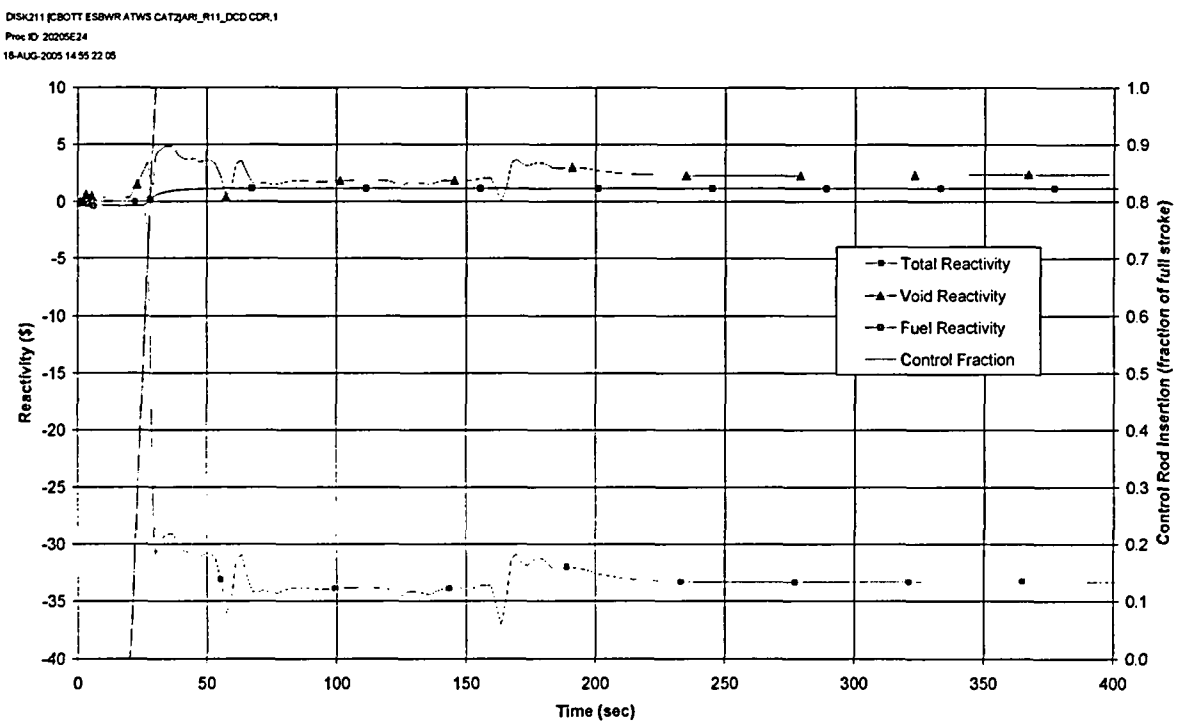


Figure 15.5-1c. MSIV Closure with ARI

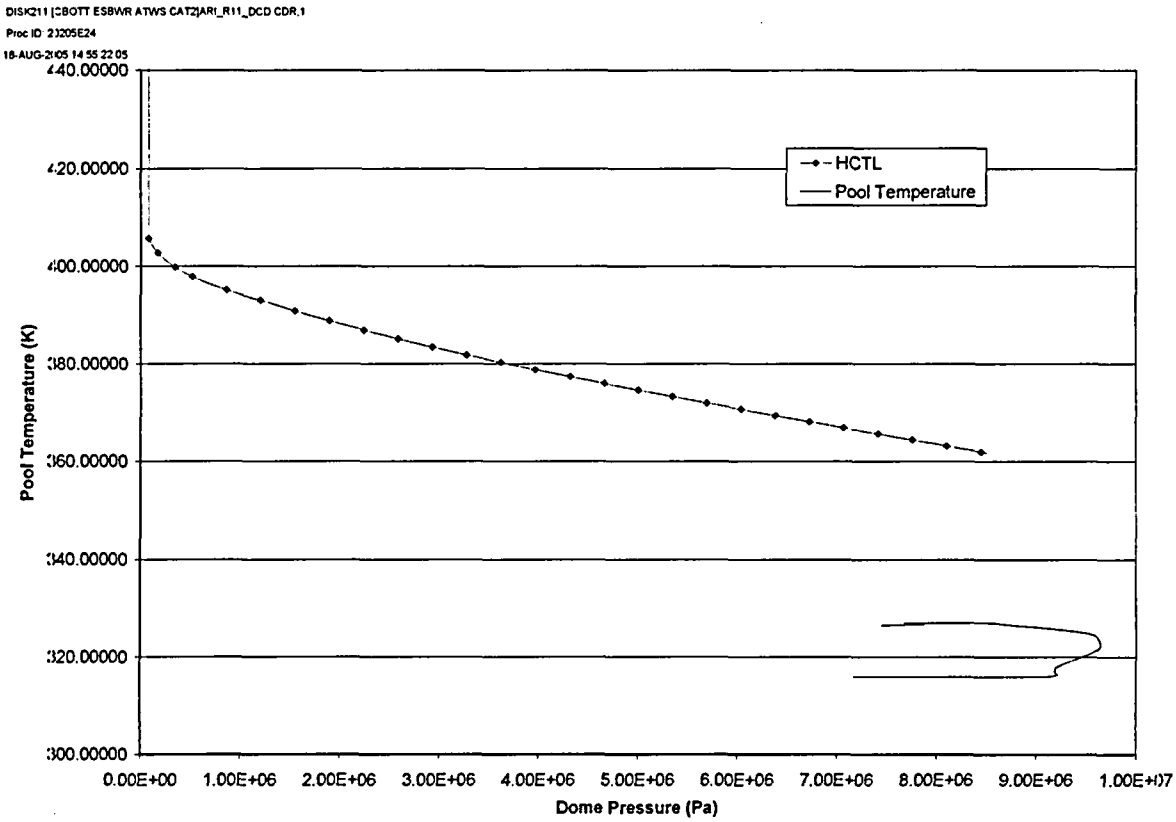


Figure 15.5-1d. MSIV Closure with ARI

Parameter	Value	Time
Maximum Neutron Flux, %	228	3s
Maximum Vessel Bottom Pressure, MPaG (psig)	9.83 (1426.1)	31s
Maximum Bulk Suppression Pool Temperature, °C (°F)	68.8 (155.8)	320s
Associated Containment Pressure, kPaG (psig)	172.0 (24.90)	320s
Peak Cladding Temperature, °C (°F)	952.8 (1747.0)	31s

Table 15.5-4b ATWS MSIV Closure Summary - FMCRD Case

DISK211 [CBOTT ESBWR ATWS CAT2]FMCRD_R11_DCD CDR,1
Proc ID 20206A20
18-AUG-2005 14 53 45 78

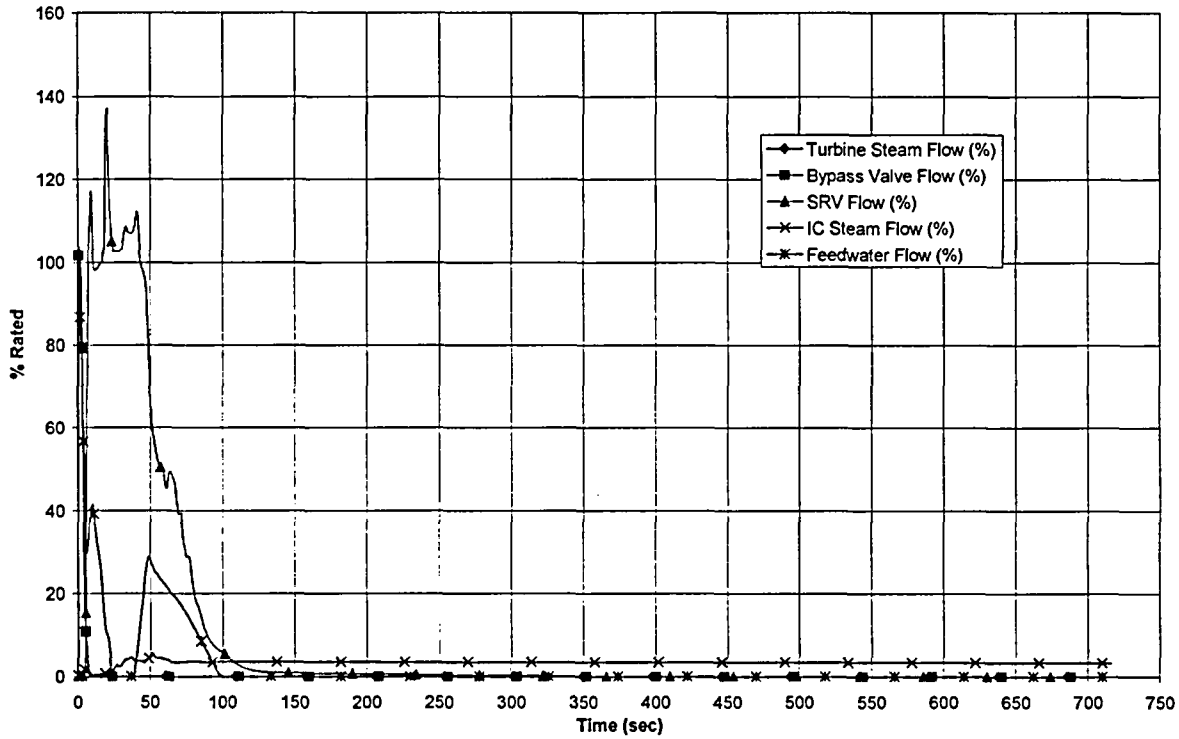


Figure 15.5-2a. MSIV Closure with FMCRD Run-in

DISK211 [CBOTT ESBWR ATWS CAT2] FMCRD_R11_DCD CDR,1
Proc ID 20206A20
18-AUG-2005 14 53 45 78

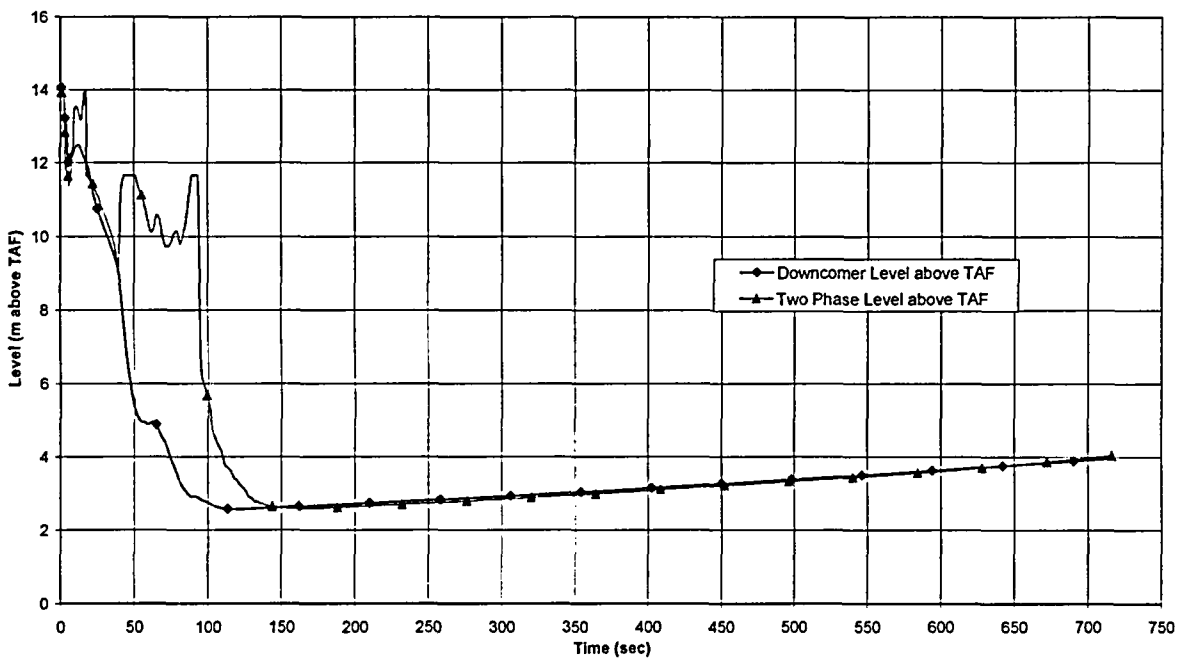


Figure 15.5-2b. MSIV Closure with FMCRD Run-in

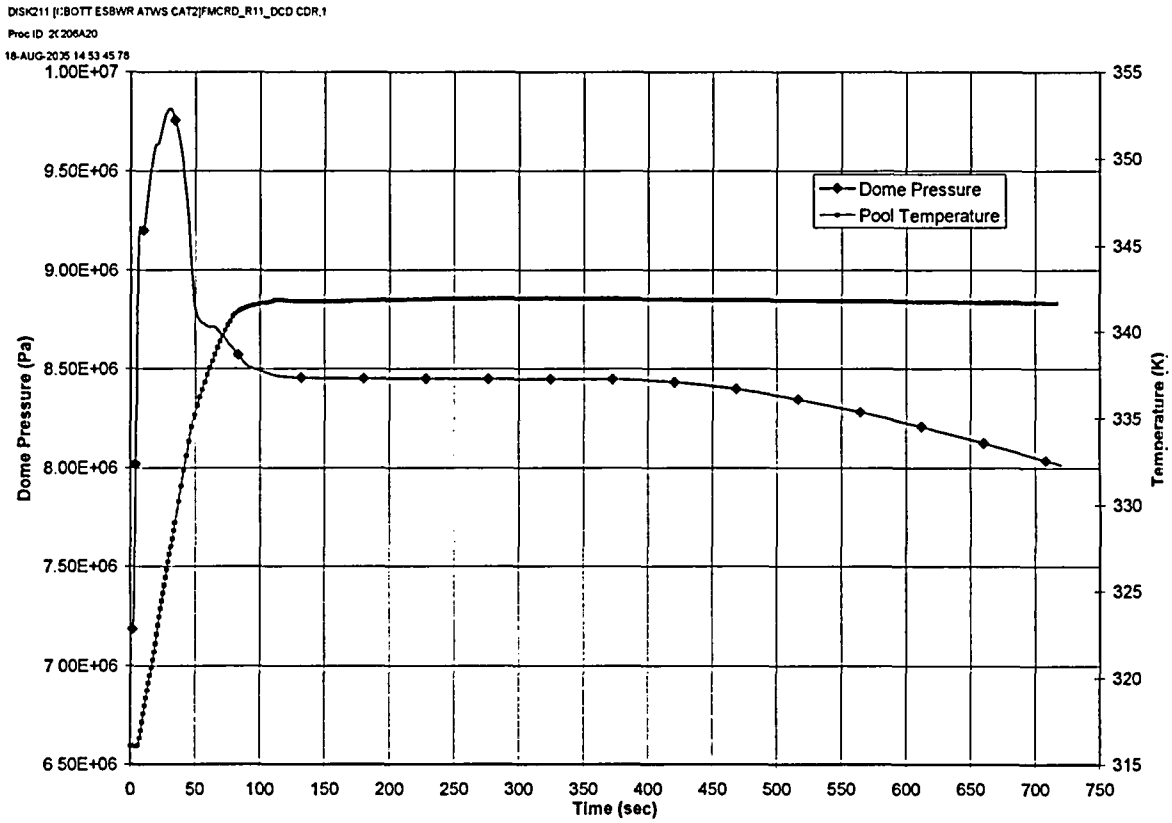


Figure 15.5-2b. MSIV Closure with FMCRD Run-in

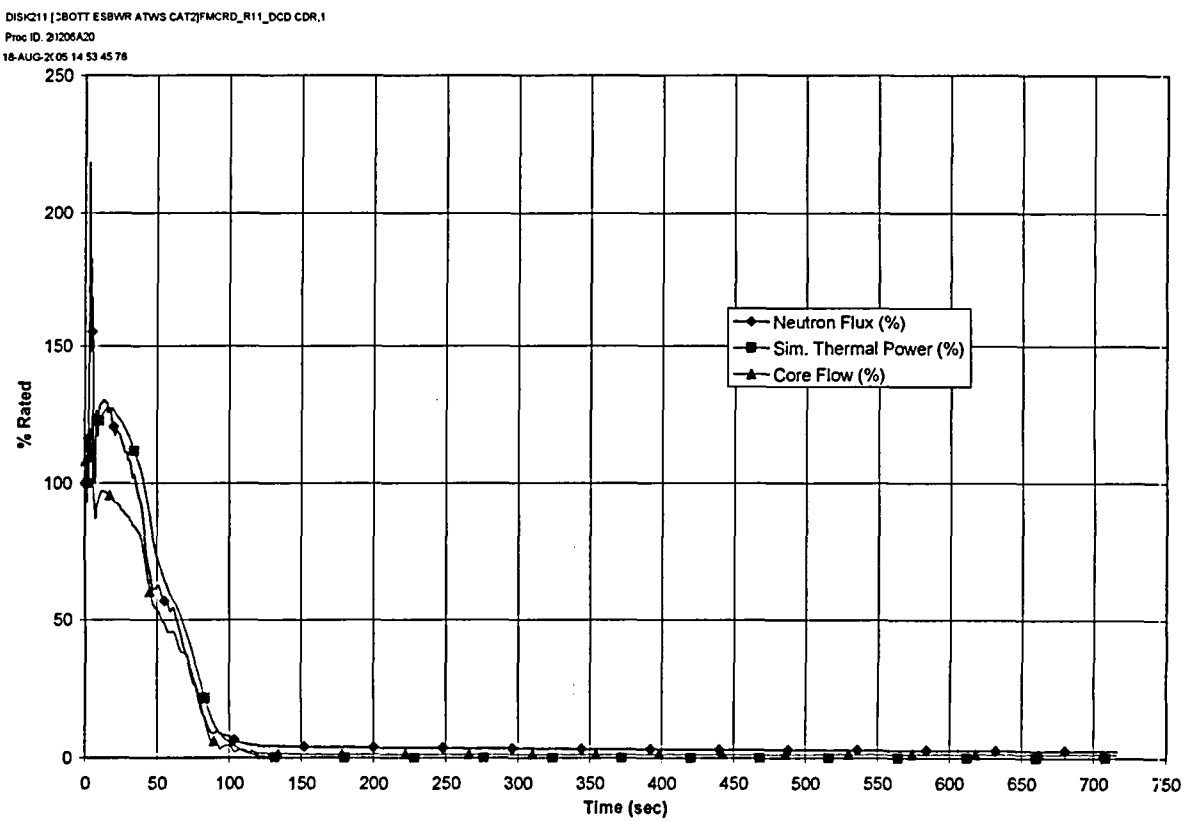


Figure 15.5-2c. MSIV Closure with FMCRD Run-in

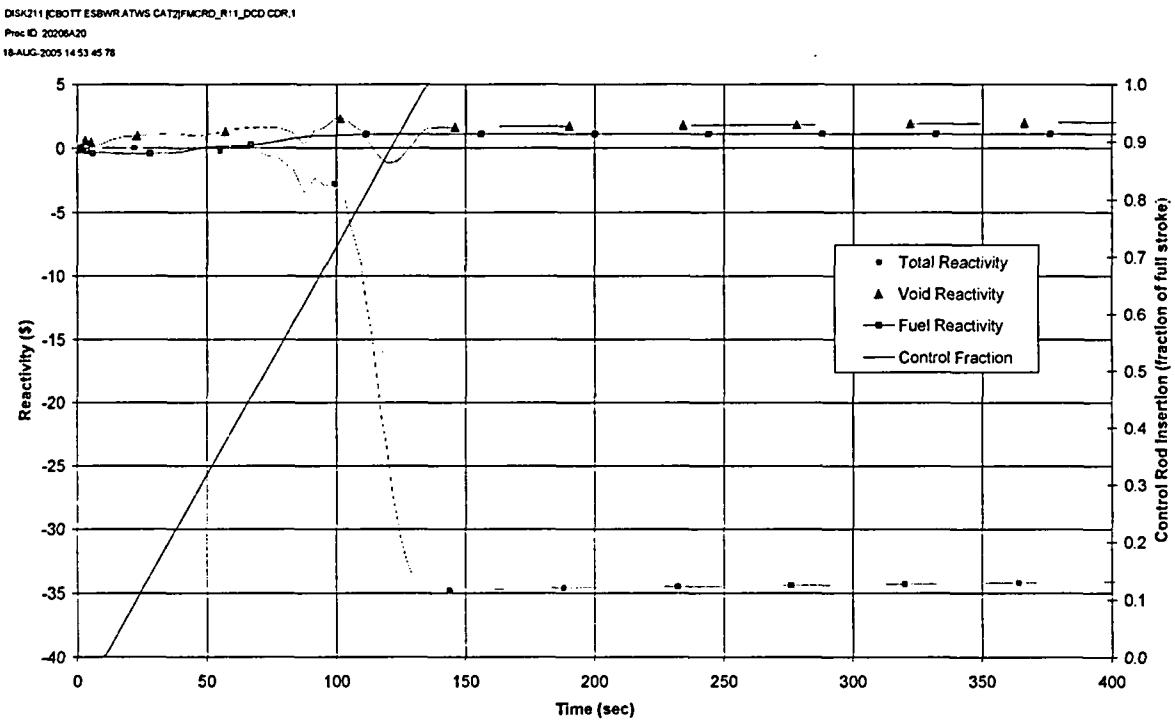


Figure 15.5-2c. MSIV Closure with FMCRD Run-in

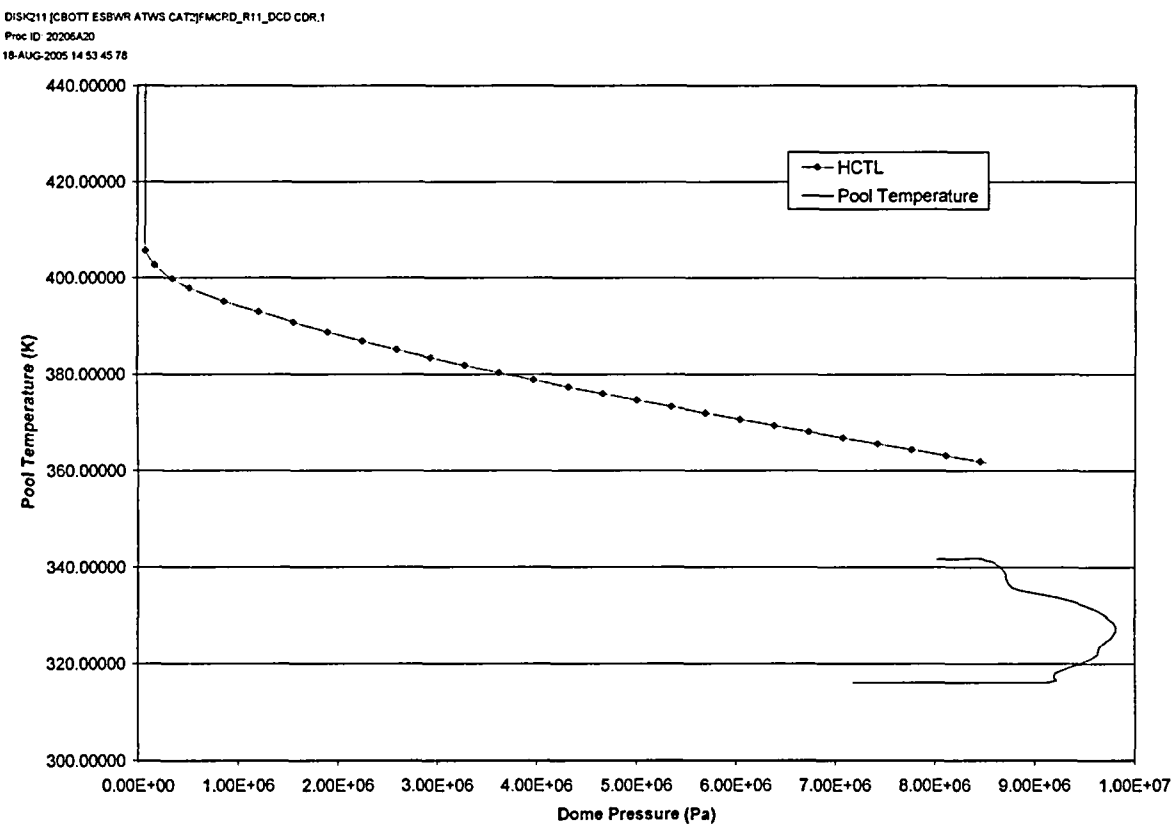


Figure 15.5-2d. MSIV Closure with FMCRD Run-in

Parameter	Value	Time
Maximum Neutron Flux, %	228	3s
Maximum Vessel Bottom Pressure, MPaG (psig)	9.81 (1422)	29s
Maximum Bulk Suppression Pool Temperature, °C (°F)	77.6(172)	254s
Associated Containment Pressure, kPaG (psig)	191.0 (27.77)	254s
Peak Cladding Temperature, °C (°F)	916.2(1681.1)	25s

Table 15.5-4c
ATWS MSIV Closure Summary - SLCS Case

DISK403 [SS ESBWR ATWS MSIV SCOPING2]ATWS-MSIV-EOC-BOUND-R1_DCD CDR,1
Proc ID: 30205050
16-AUG-2005 16 58 09 41

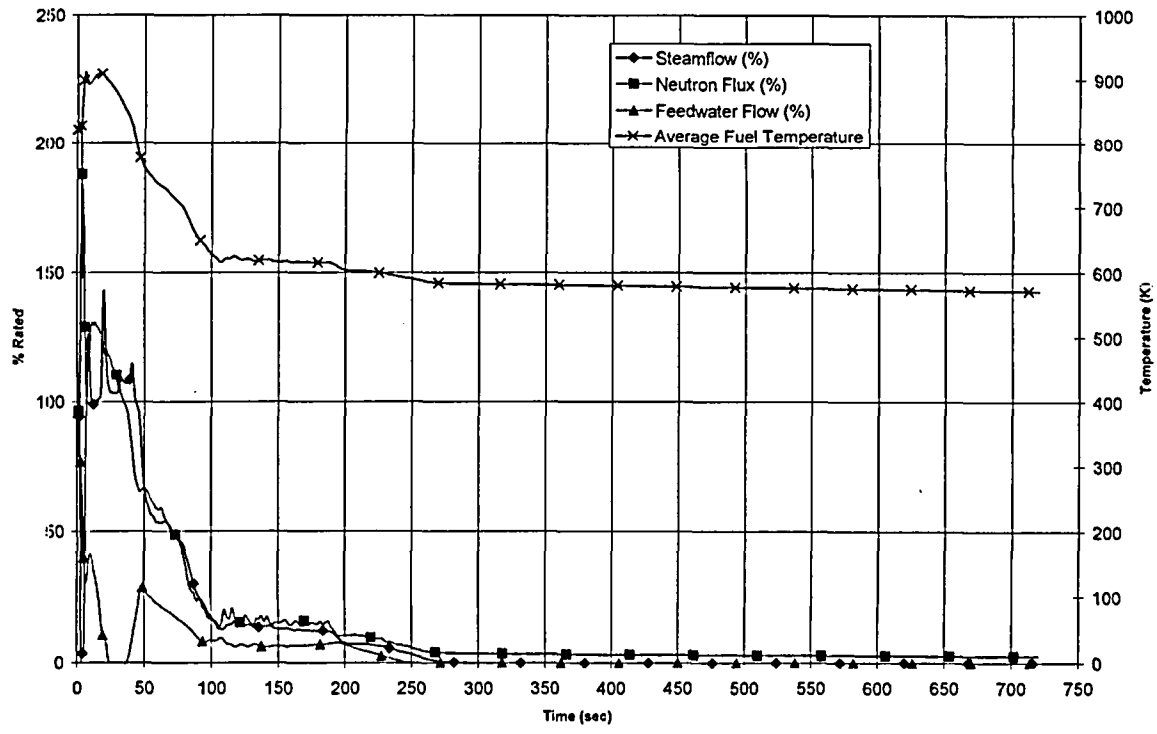


Figure 15.5-3a. MSIV Closure with Boron Injection

DISK403 [SS ESBWR ATWS MSIV SCOPING2]ATWS-MSIV-EOC-BOUND-R1_DCD CDR.1
Proc ID 20205050
16-AUG-2005 16 58 09 41

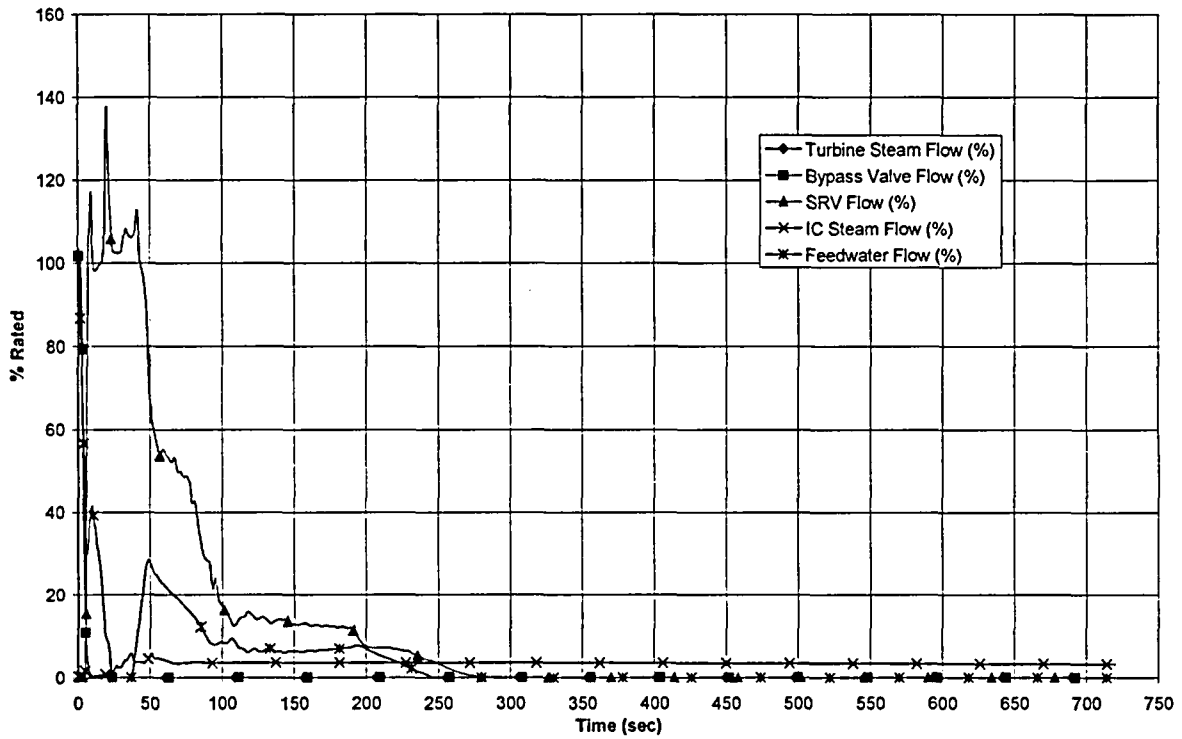


Figure 15.5-3a. MSIV Closure with Boron Injection

DISK403 [SS ESBWR ATWS MSIV SCOPING2]ATWS-MSIV-EOC-BOUND-R1_DCD CDR.1
Proc ID 20205050
16-AUG-2005 16 58 09 41

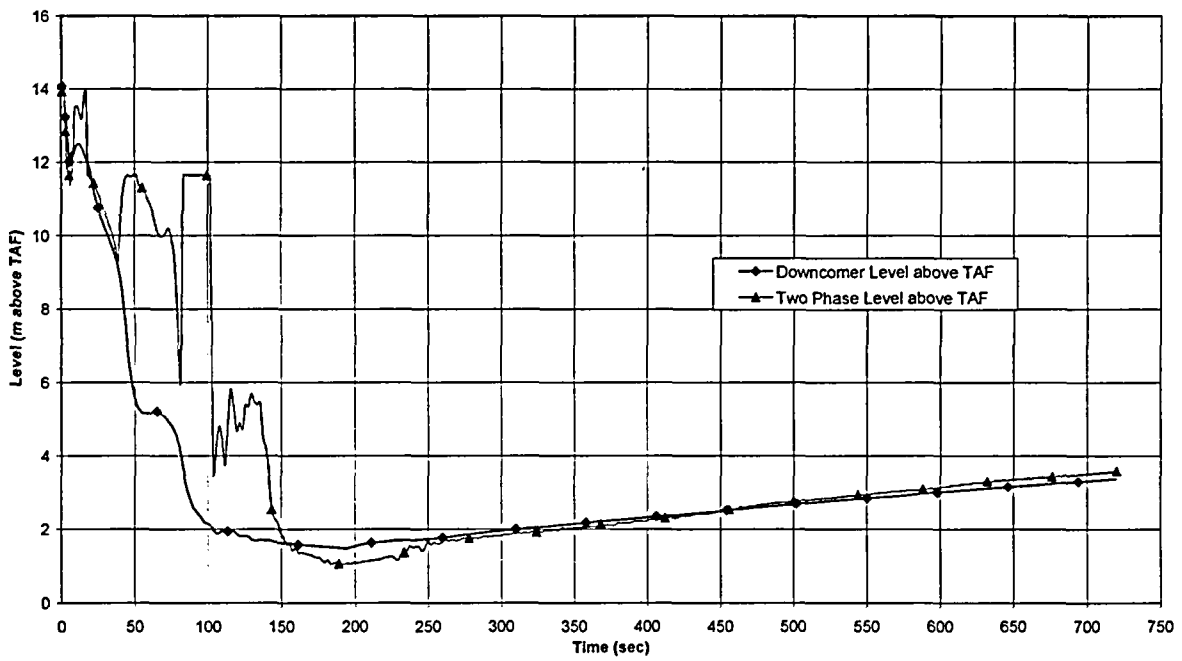
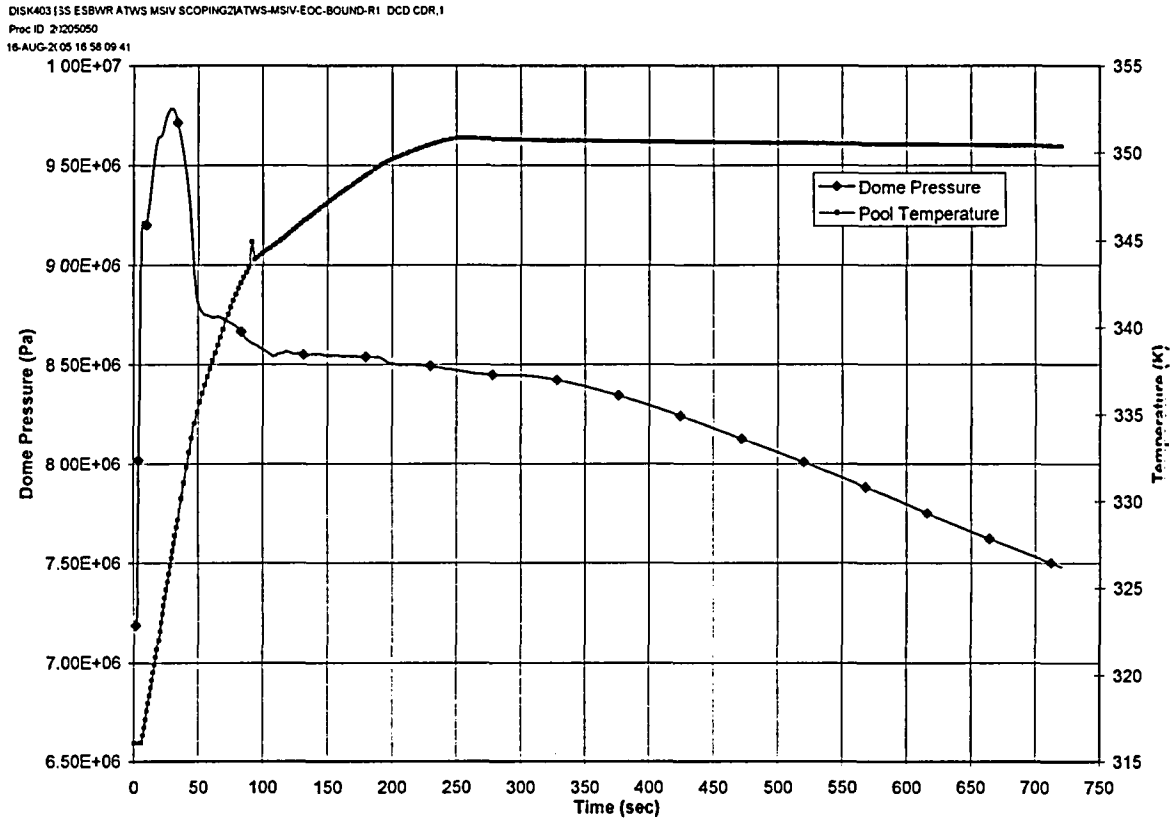
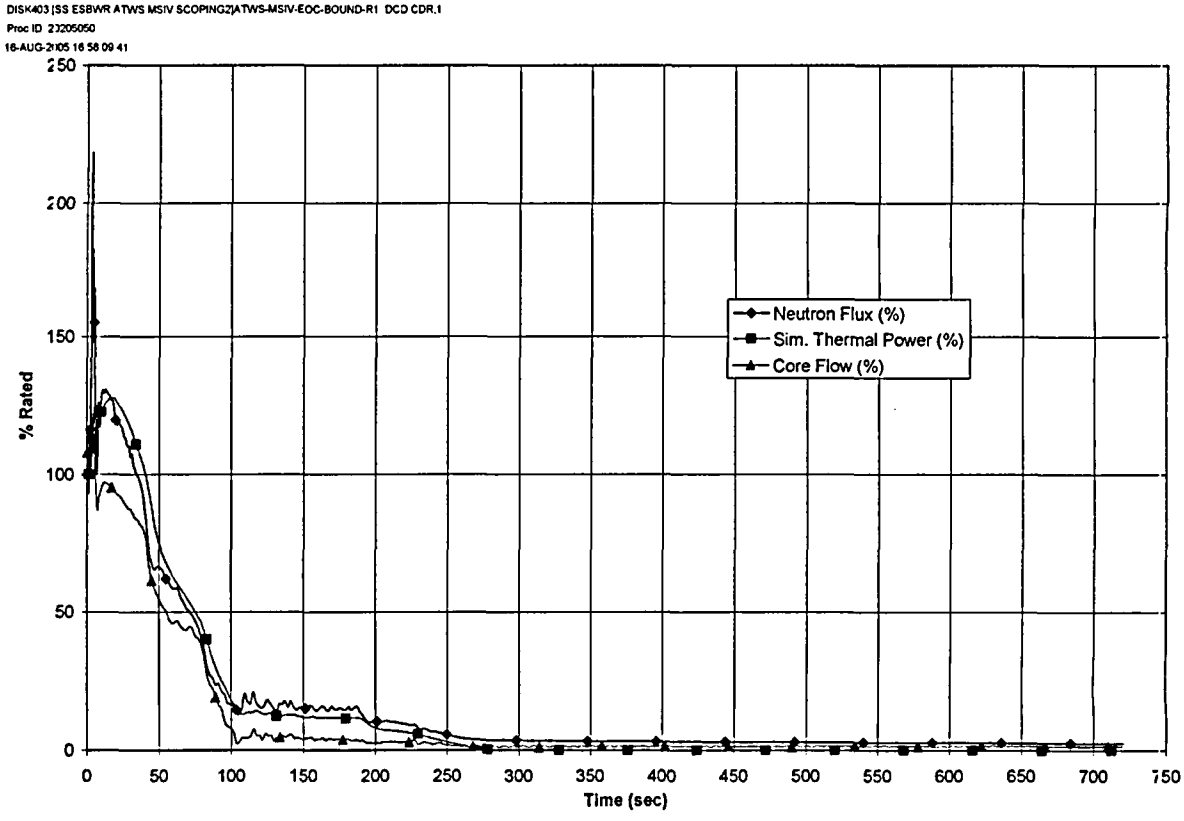


Figure 15.5-3b. MSIV Closure with Boron Injection



77 WM 19Apr2006

Figure 15.5-3b. MSIV Closure with Boron Injection



78 WM 19Apr2006

Figure 15.5-3c. MSIV Closure with Boron Injection

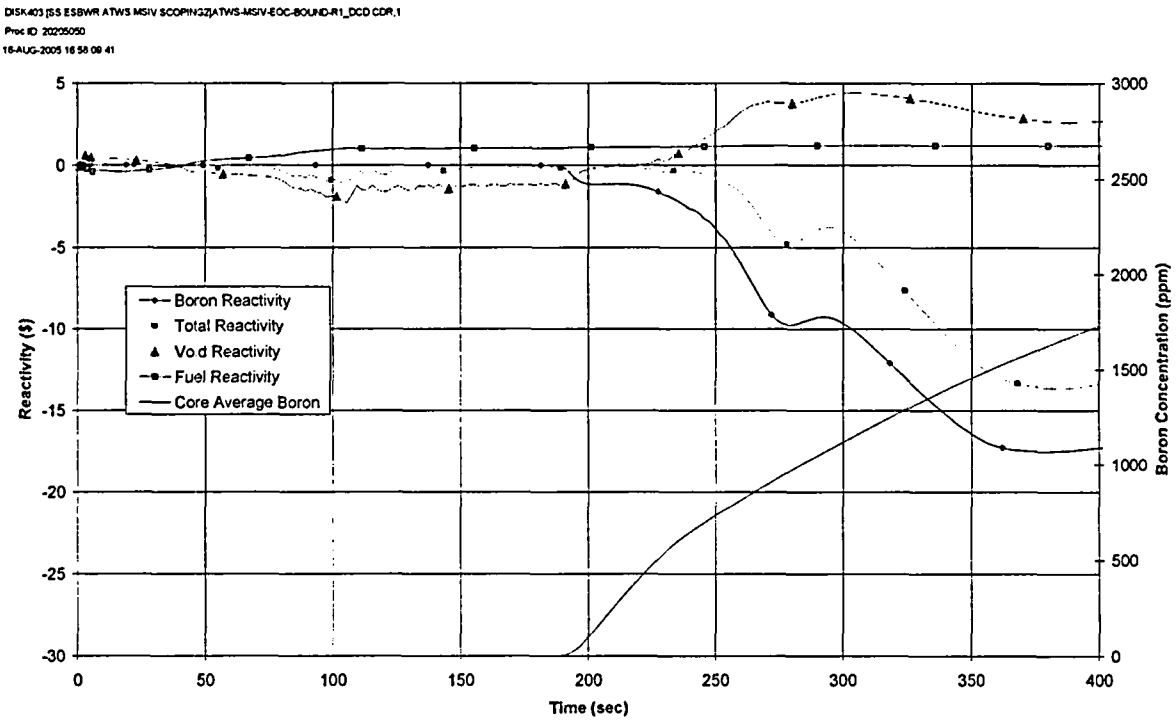


Figure 15.5-3c. MSIV Closure with Boron Injection

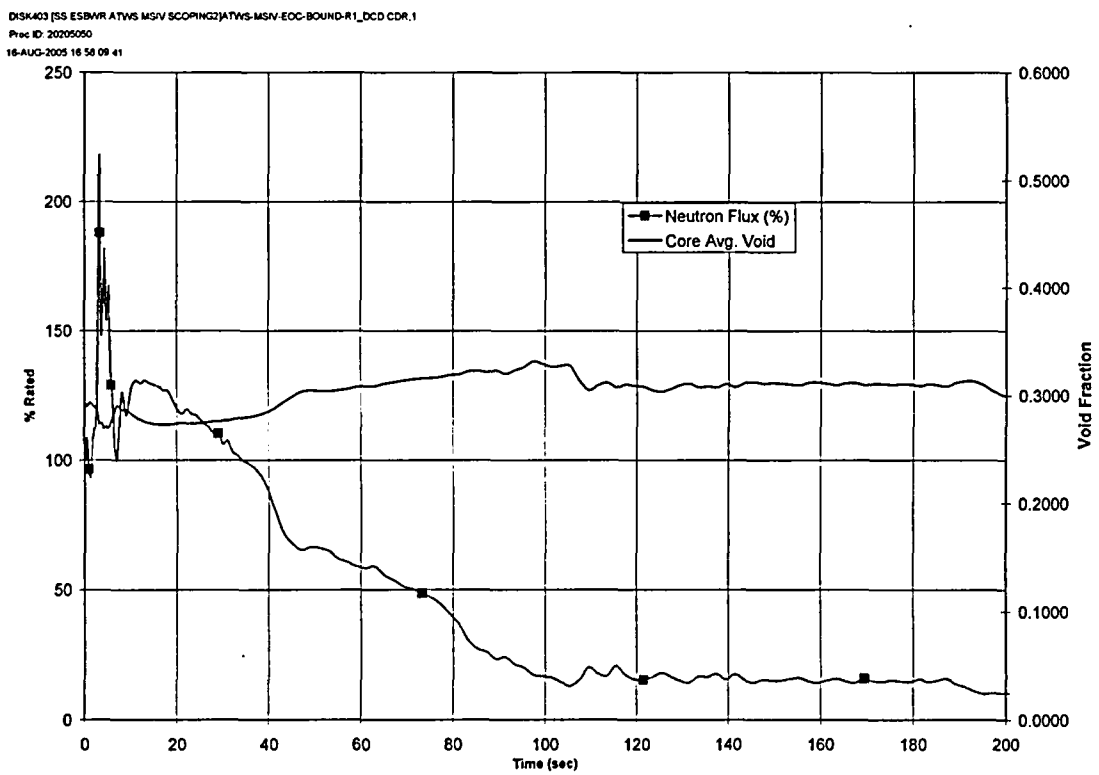


Figure 15.5-3d. MSIV Closure with Boron Injection

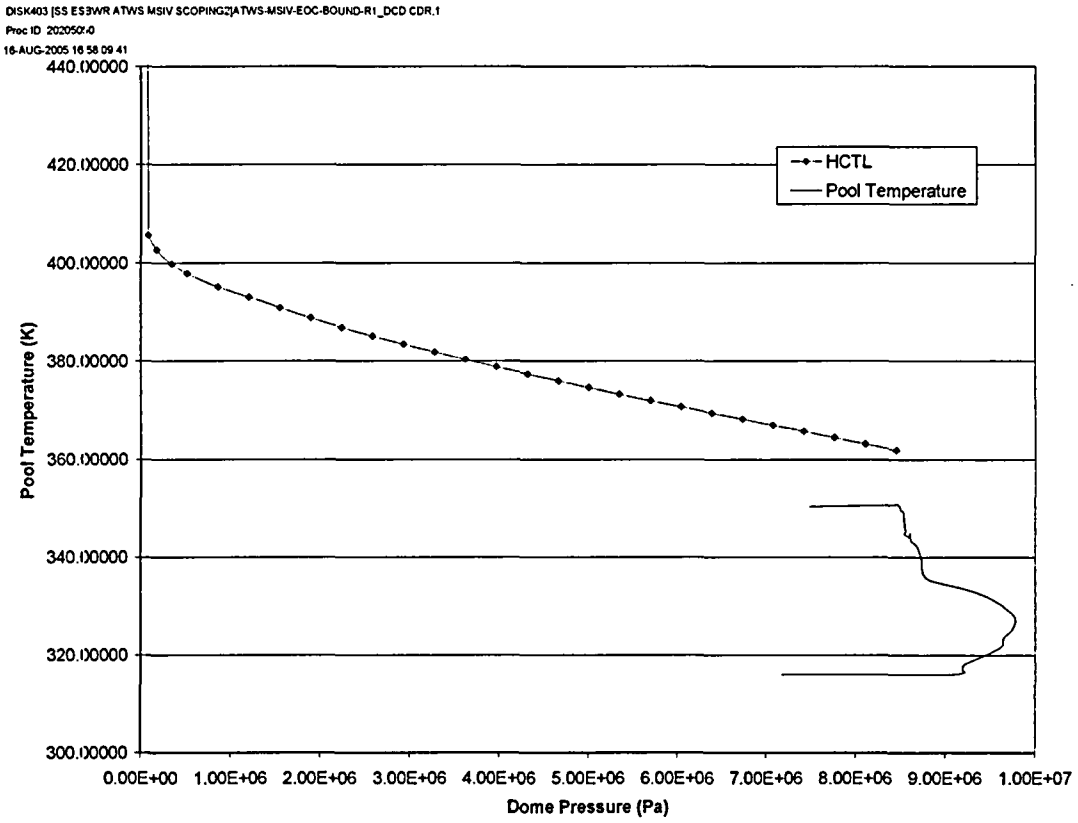


Figure 15.5-3d. MSIV Closure with Boron Injection

Time (s)			Event
ARI	FMCRD	SLCS	
0	0	0	MSIV Closure starts
2	2	2	IC initiates
4	4	4	ATWS trip set at high pressure
5	5	5	SRVs open
33	41	42	Level drops below L2 set point
43	51	52	HP_CRD flow starts
-	-	189	SLCS injection starts*
-	-	710	High pressure design volume of borated solution injected into bypass
20	5	-	Start of Rod Motion
30	-	-	Alternate Rod Insertion complete**
-	135	-	FMCRD Run-in complete***

Table 15.5-4d ATWS MSIVC Sequence of Events

* SLCS is automatically initiated if ARI and FMCRD functions fail
** ARI constitutes the first course of action for ATWS mitigation
*** FMCRD run-in occurs if ARI fails

Decrease in Reactor Coolant Inventory

- AOO

Section 8B.3 Analysis Assumptions

(1) SBO starts with loss of all alternating current (AC) power, which occurs at time zero.

(2) Loss of AC power trips reactor feedwater pumps at event initiation.

(3) The reactor scrams occurs at 2.0 seconds due to loss of power supply to feedwater pumps. It is considered that when feedwater flow is lost, there is a scram signal with a conservative delay time of 2.0 seconds.

(4) Feedwater is ramped down linearly to zero in 5.0 seconds after event initiation.

(5) Closure of all Main Steam Isolation Valves (MSIVs) is automatically initiated when the reactor water level reaches Level 2 with a 30.0 second delay, and the valves are fully closed at 5.0 seconds.

(6) A single failure is assumed. The systems available for initial vessel inventory and pressure control, containment pressure/temperature control and suppression pool temperature control are:

a. Isolation Condensers (ICs);
b. Control Rod Drive (CRD) pumps;
c. Safety Relief Valves (SRVs);
d. Depressurization Valves (DPVs);
e. Gravity-Driven Cooling System (GDCS) squib valves;
f. GDCS loops; and
g. Passive Containment Cooling System (PCCS) loops.

9) 3 ICs (assuming single failure of one IC) are automatically initiated upon the closure of MSIVs to stabilize the plant. Operators control ICs to assure the maximum cool down rate not exceeding 100°F/hr,
if necessary.

10) 2 CRD pumps are automatically initiated to provide vessel inventory makeup. The maximum delayed time is 145 seconds upon restoring AC power through nonsafety-related standby diesel generators (DGs). CRD pumps shall keep the water level above Level 0.5 to avoid any Automatic Depressurization System (ADS) initiation to blow down the vessel.

11) ICs and CRD flow stabilize the plant. SRVs, DPVs, PCCS and GDCS can be utilized if IC does not stabilize the plant, which is most unlikely.

12) The event is terminated after an 8-hour coping period.

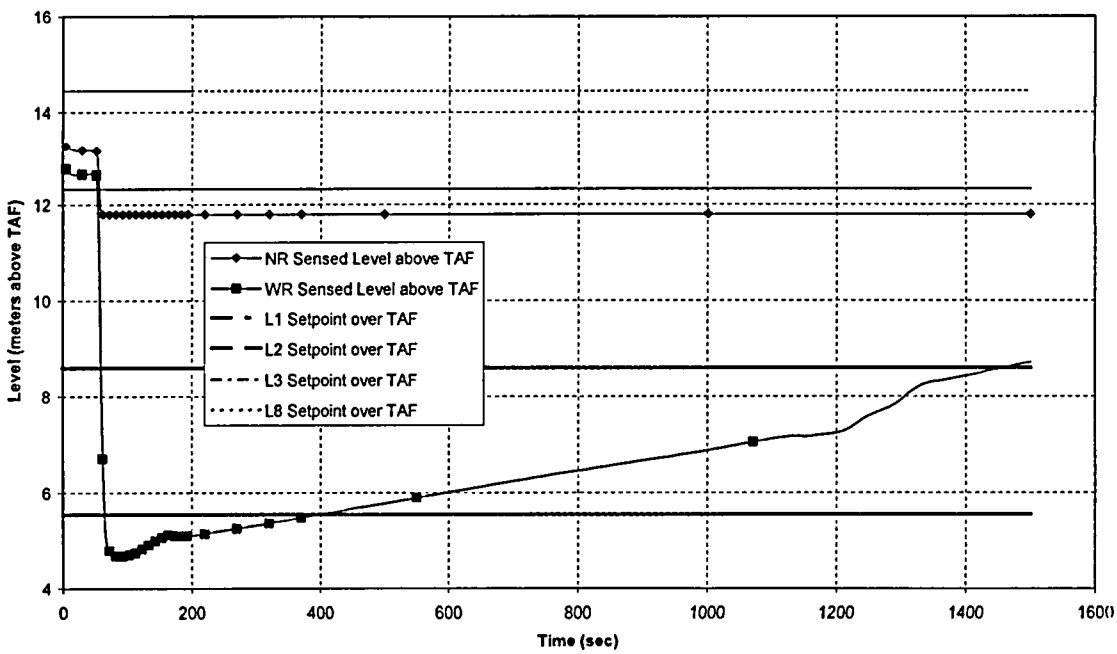


Figure 8B-1 Water Level Response for SBO

• Special Event

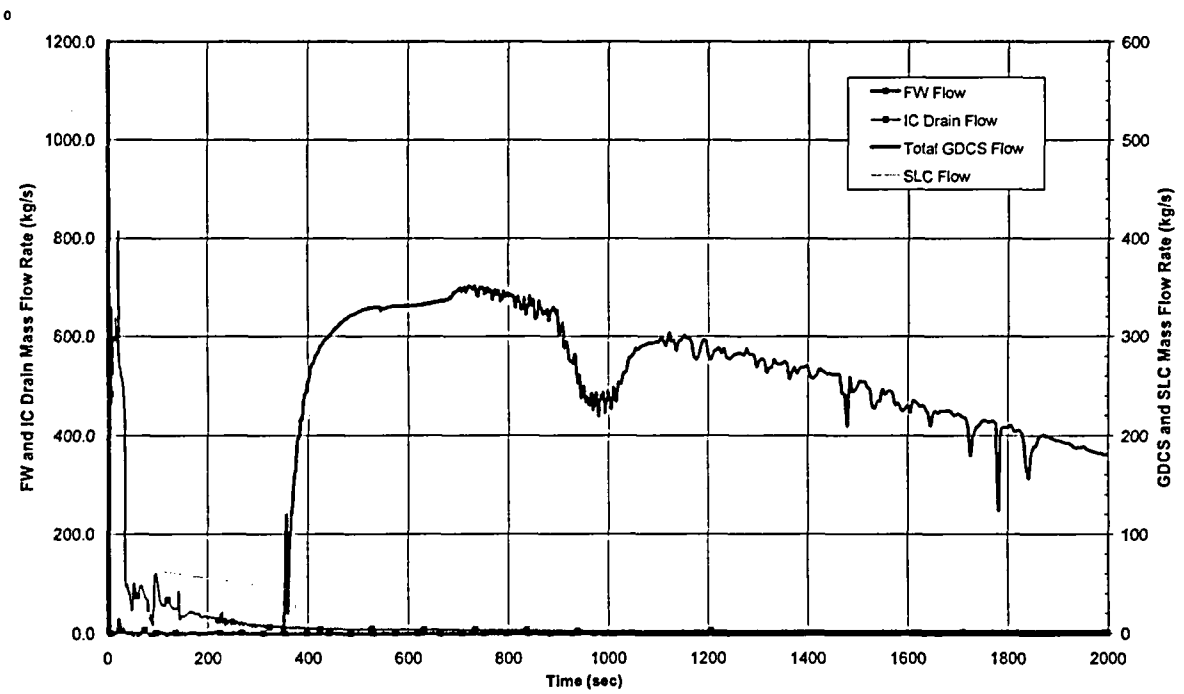


Figure 15.5-10. Vessel Inventory Makeup Flow Response for SBO

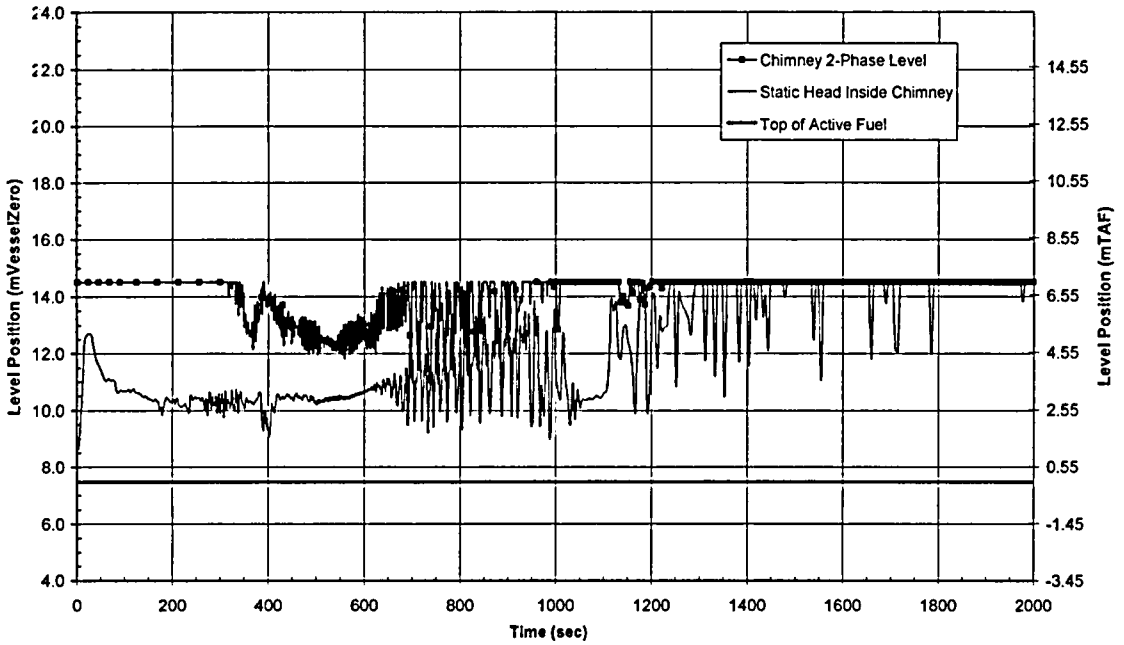


Figure 15.5-11. Water Level Response for SBO

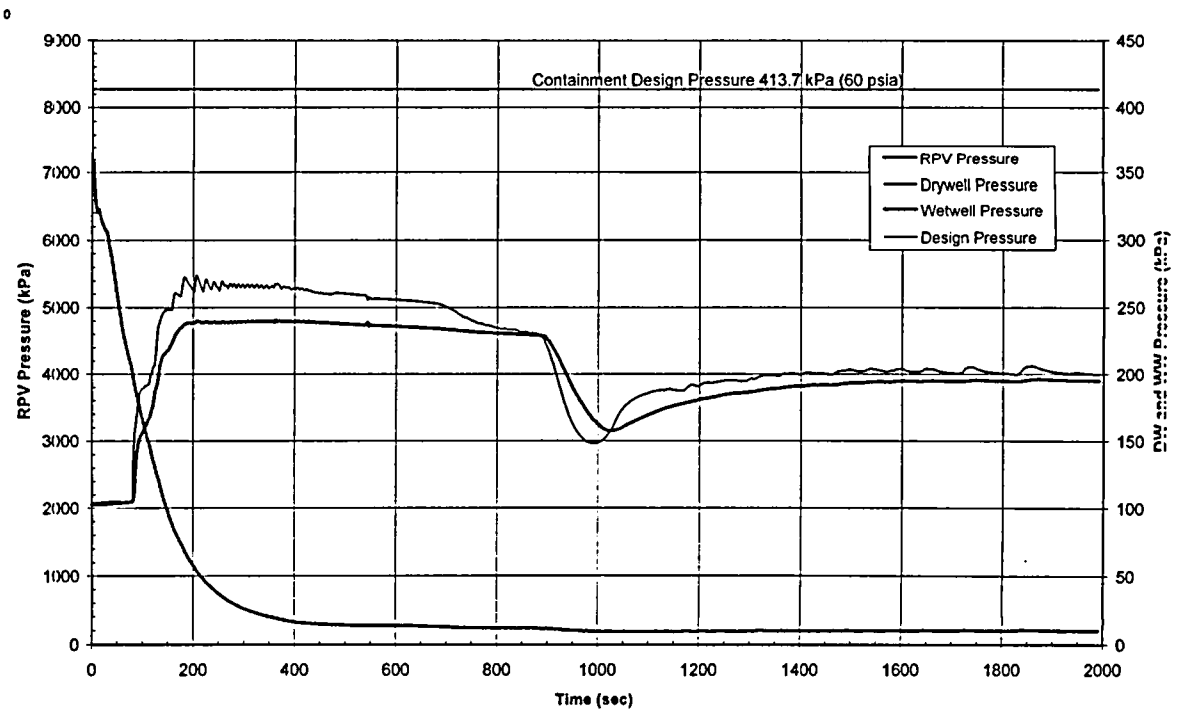
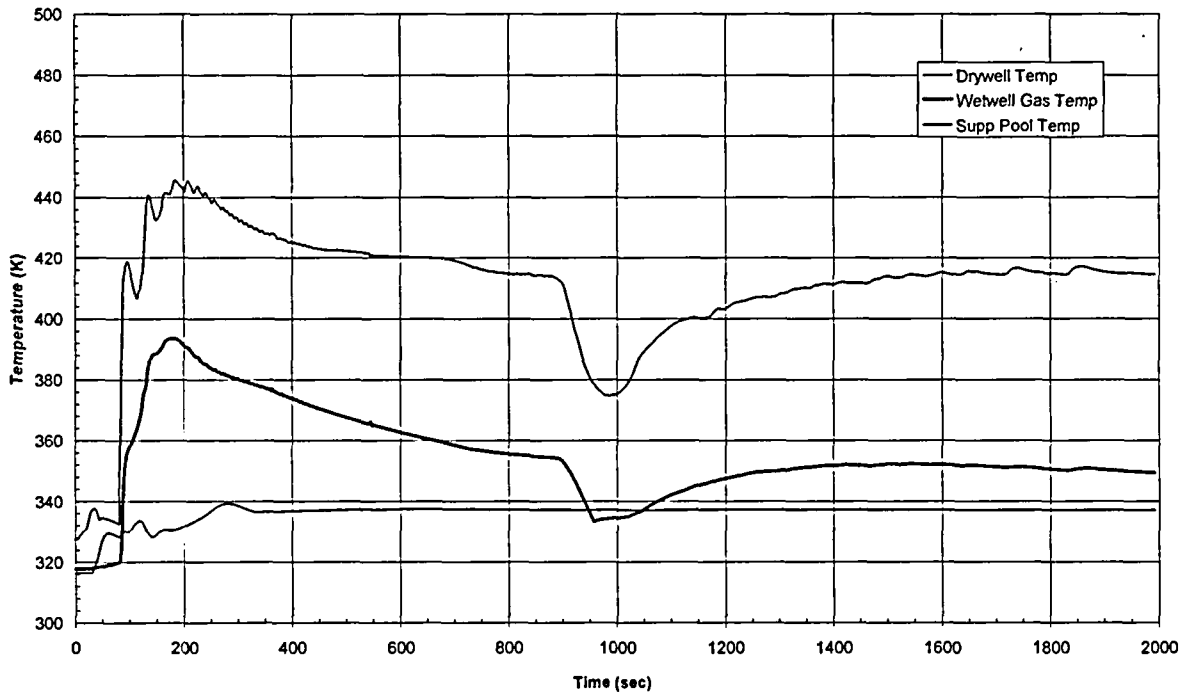
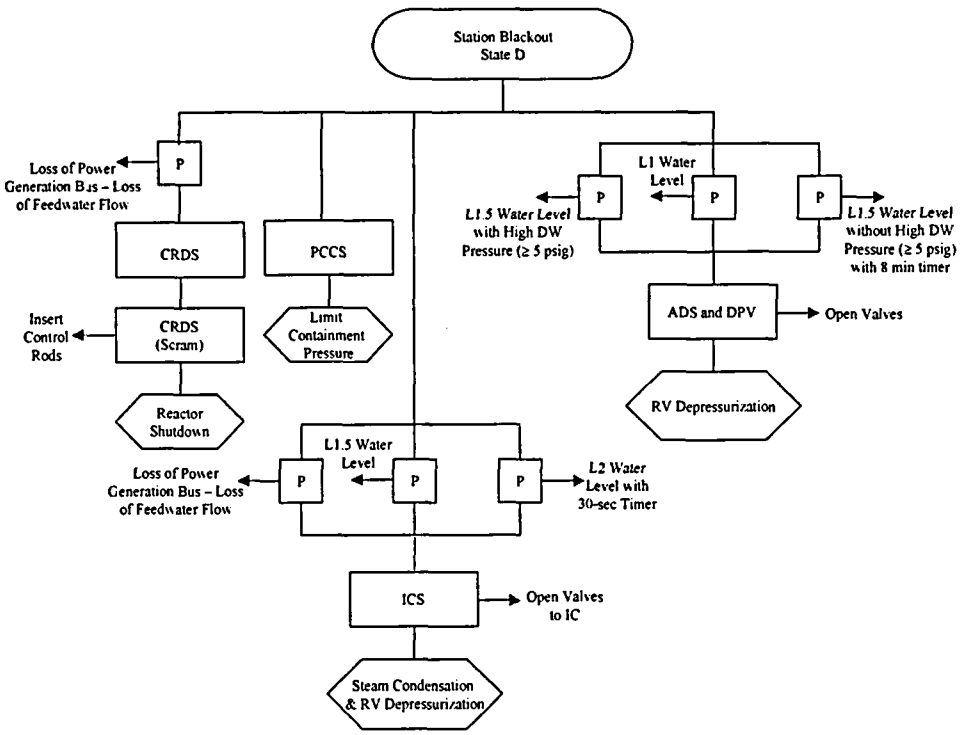


Figure 15.5-12 Vessel and Containment Pressure Response for SBO



91 WM 19Apr2006 **Figure 15.5-13. Containment and Suppression Pool Temperature Response for SBO**



92 WM 19Apr2006 **Figure 15.1-45a. Event Diagram – Station Blackout**

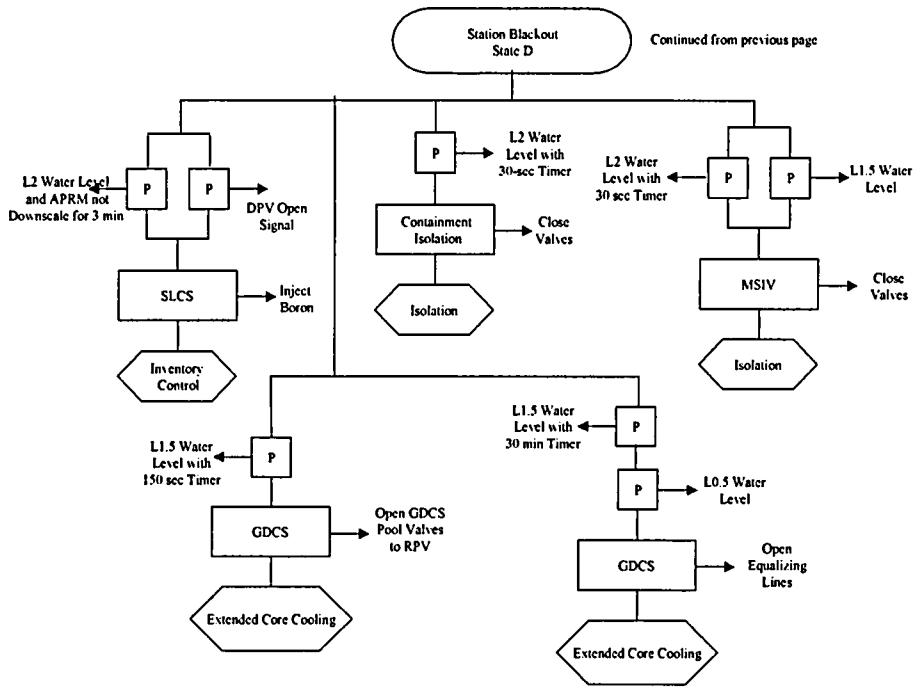


Figure 15.1-45b. Event Diagram – Station Blackout

Determination of Systems Credited in Safety Analyses

- Nuclear Safety Operational Analysis (NSOA) DCD Tables and Figures in 15.1-1 were developed from Chapter 6 & 15 Seq.of Events and text.
- Listed Systems, Structures, and Components (SSCs) using:
  - Nuclear Safety Operational Analysis (NSOA) Event Diagrams
    - DCD Figures 15.1-1 through 15.1-47
  - Sequence of Events Diagrams
    - DCD Tables in 15.2, 15.3, 15.4, 6.2, and 6.3
  - Results and Descriptions in 15.2, 15.3, 15.4, 6.2, & 6.3
- Includes Primary Success Path and Backup SSCs
- Verification in process
  - Includes check by originator of the analysis

SSCs considered in Chapters 6 & 15

- Preliminary unverified lists

Justification of SSC's credited in Safety Analysis with no related Tech Spec

- CRHAHVS
- FAPCS
- FBHVS
- Float valves to spillover pipes
- HP_CRD
- PRMS
- RBHVS

Preliminary unverified

5 of the above are backup systems - Associated primary success paths are:

HP_CRD - ADS/GDCS/PCC

FAPCS - ADS/GDCS/PCC

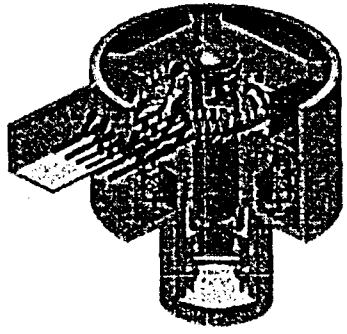
CRHAHVS - backup to EBAS in the CR dose calcs, CRHAVS credited when it produces higher dose, or needed post 72 h.

FBHVS & RBHVS - No specific credit in DCD or Fuel Handling accident, need to clarify in DCD

Need to Disposition 2 others:

- Consider controls for:
  Float valves to spillover pipes
  PRMS: high radiation trip which initiates off gas isolation

Technical Specifications for
Chapter 6 and 15 Credited Systems



April 19, 2006

"Primary Success Path"

10 CFR 50.36(c)(2)(ii)

- *Criterion 3:* A structure, system, or component that is part of the primary success path and which functions or actuates to mitigate a design basis accident or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

ITS Screening (Split) Report (NEDO-31466, Nov. 1987)

- The primary success path is ... the combinations and sequences of equipment needed to operate, which results in acceptable plant accident or transient response...

NRC Report on NEDO-31466 (May 1988)

- "Only primary equipment ... does not include backup and diverse equipment or instrumentation"

AOO Primary Success Path

DCD Table 15.0-3
Safety Analysis Acceptance Criteria for AOOs

- Pressures in the reactor coolant and main steam systems shall be maintained below 110% of their design values (i.e., not exceed ASME Code Service Level B), and the reactor steam dome pressure shall be maintained less than or equal to the Reactor Coolant System Pressure Safety Limit in the Technical Specifications
- Fuel-cladding integrity should be maintained by ensuring that the reactor core is designed with appropriate margin during any conditions of normal operation, including the effects of AOOs. The minimum value of the critical power ratio (CPR) reached during the AOO should be such that 99.9 percent of the fuel rods in the core would not be expected to experience boiling transition during core-wide transients. (This criterion corresponds to the greater than 99.9% of the fuel rods in the core would be expected to avoid boiling transition related safety limit in the Technical Specifications.)
- Uniform cladding strain ≤ 1%
- No fuel centerline melt (core-wide AOOs only)
- Energy generation is < 170 cal/g (RWE during startup only)
- Containment and suppression pool pressures and temperatures shall be maintained below their design values
- An AOO should not generate a more serious plant condition unless other faults occur independently
- There is no loss of function of any fission product barrier (Safety/Relief Valve or Depressurization Valve discharge does not apply)

AOO w-SAF Primary Success Path

DCD Table 15.0-4

Safety Analysis Acceptance Criteria for AOOs In Combination With An Additional Single Active Component Failure or Single Operator Error

- Reactor water level shall be maintained above the top of the core (i.e., active fuel).
- Pressures in the reactor coolant and main steam systems shall be maintained below 110% of their design values (i.e, not exceed ASME Code Service Level B), and the reactor steam dome pressure shall be maintained less than or equal to the Reactor Coolant System Pressure Safety Limit in the Technical Specifications
- If containment isolation is involved, containment and suppression pool pressures and temperatures shall be maintained below their design values
- Except for fuel cladding, there shall be no loss of function of any fission product barrier.
- Fuel cladding failures shall be limited such that the radiological consequence shall be ≤ 0.05 rem TEDE.

Infrequent Events Primary Success Path

DCD Table 15.0-5
Safety Analysis Acceptance Criteria for Infrequent Events

- Reactor water level shall be maintained above the top of the core (i.e., active fuel).
- Pressures in the reactor coolant and main steam systems shall be maintained below the ASME Service Level C limit, which corresponds to 120% of design pressure.
- Radiological consequence shall be ≤ 2.5 rem TEDE. However, if the applicable SRP section specifies an accident-specific (i.e., different or additional) radiological acceptance criterion, then the accicent-specific SRP acceptance criterion/criteria is/are applied.
- Containment and suppression pool pressures and temperatures shall be maintained below their design values.
- Control room personnel shall not receive a radiation exposure in excess of 5 rem TEDE for the duration of the event.

Accident Primary Success Path

DCD Table 15.0-6
Safety Analysis Acceptance Criteria for Accidents

- Pressures in the reactor coolant and main steam systems shall be maintained below the ASME Service Level C limit, which corresponds to 120% of design pressure
- Radiological consequence shall be ≤ 2.5 rem TEDE, 6.3 rem TEDE, or 25 rem TEDE, depending on the accident-specific acceptance criterion in NUREG-0800, SRP 15.0.1
- The calculated maximum fuel element cladding temperature shall not exceed 1204°C (2200°F)
- The calculated total oxidation of the cladding shall nowhere exceed 0.17 times the total cladding thickness before oxidation
- The calculated total amount of hydrogen generated from the chemical reaction of the cladding with water or steam shall not exceed 0.01 times the hypothetical amount that would be generated if all of the metal in the cladding cylinders surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react
- Calculated changes in core geometry shall be such that the core remains amenable to cooling
- After any calculated successful initial operation of the ECCS, the calculated core temperature shall be maintained at an acceptably low value and decay heat shall be removed for the extended period of time required by the long-lived radioactivity remaining in the core
- Containment and suppression pool pressures and temperatures shall be maintained below their design values
- Control room personnel shall not receive a radiation exposure in excess of 5 rem TEDE for the duration of the accident.

Summary of Safety Systems — Preliminary / Unverified Draft

(Credited in the ESBWR Chapter 6 LOCA Analyses)

SSC	Technical Specification SSC	Technical Specification Support/Supported SSC
RPS	3.3.1.x	
CRDS	3.1.3	3.1.4; 3.1.5
ICS	3.5.4	3.3.5.3; 3.3.5.4; 3.8.1; 3.8.5; 3.8.7
ADS – SRV	3.5.1	3.3.5.1; 3.3.5.2; 3.8.1; 3.8.5; 3.8.7
ADS – DPV	3.5.1	3.3.5.1; 3.3.5.2; 3.8.1; 3.8.5; 3.8.7
GDCS – Injection Valves	3.5.2	3.3.5.1; 3.3.5.2; 3.8.1; 3.8.5; 3.8.7
GDCS – Equalizing Valves	3.5.2	3.3.5.1; 3.3.5.2; 3.8.1; 3.8.5; 3.8.7
MSIV	3.6.1.3	3.3.6.1; 3.3.6.2
SLCS	3.1.7	3.3.5.1; 3.3.5.2; 3.8.1; 3.8.5; 3.8.7
PCCS	3.6.1.7	3.7.5
Float valves to spillover pipes	[*T.B.Drafted*]	
Connection valves - (Dryer/Separator storage pool to PCC pool)	SR 3.6.1.7.2 SR 3.7.5.4 & 5	
Vacuum Breakers	3.6.1.6	SR 3.6.1.1.2

Summary of Safety Systems — Preliminary / Unverified Draft

(Credited in the Chapter 15 Analyses)

SSC	Technical Specification SSC	Technical Specification Support/Supported SSC
FWCS, RC&IS, and SCRRI	3.2.2 (MCPR)	(COLR; possible Availability Controls)
~~SCRRI (Generator Load Reject)~~	~~-- NA --~~	~~Investment Protection Only~~
Turbine Bypass Valve System	3.7.3	(3.2.2, MCPR)
TCV (closure)	3.3.1.1, Fn 11	
HP_CRD	-- NA --	Serves as Backup to ADS/GDCS/PCCS
MSIV	3.6.1.3	
ICS	3.5.4	
RPS / CRDS (Scram)	3.1.3; 3.1.4; 3.1.5 & 3.3.1.x	