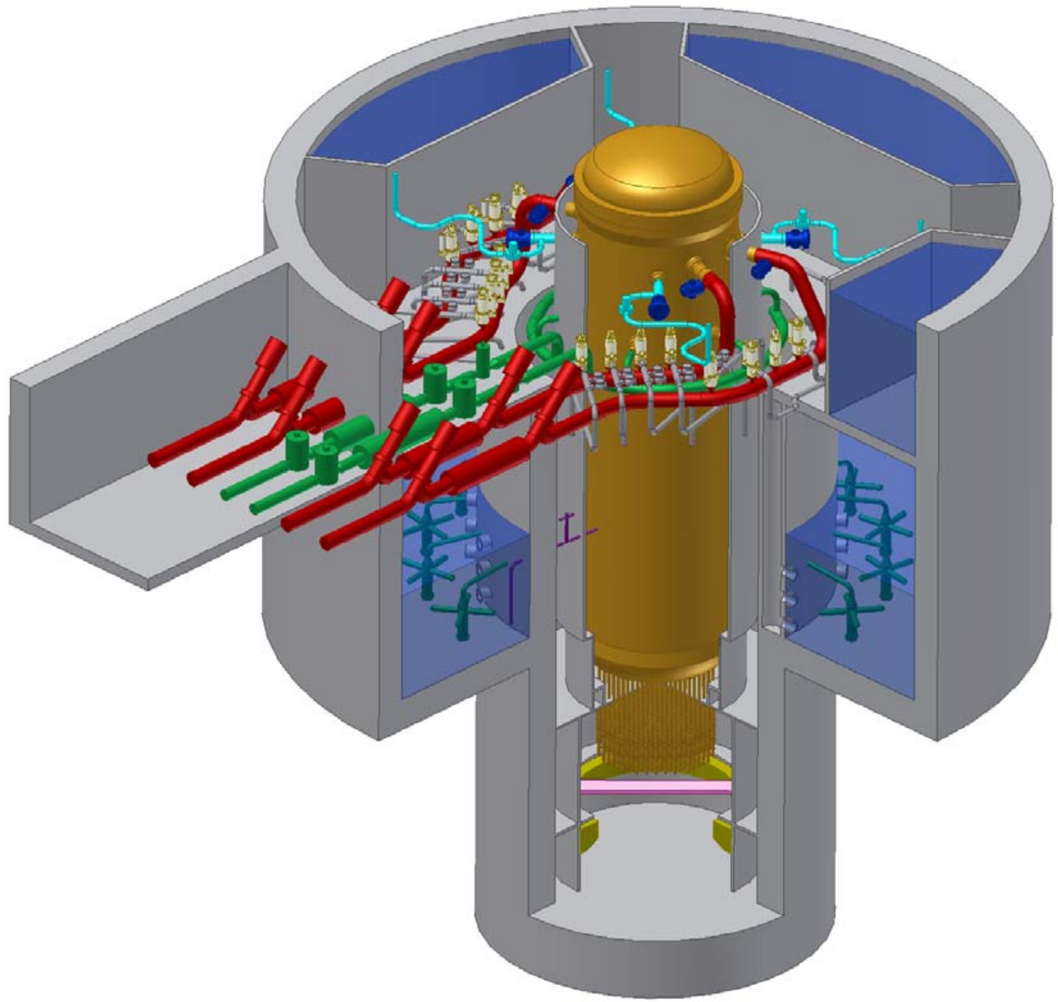
Chapter 15 Safety Analyses



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Chapter 15 Safety Analyses

Evaluation of:

- > Anticipated Operational Occurrences
- > Infrequent Events
- > Accidents
- > Special Events
- > Offsite dose/Radiological assessment
- > FMEA
- > Event Frequency determination of Infrequent Events

Comparison of results to acceptance criteria

Present Highlights of Chapter 15

Summarize the results of selected events

Illustrate response of ESBWR to events

Examples of how fault tolerant control system, and improved BOP reliability changes event categories

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TRACG Applied to all Thermal Hydraulic evaluations of ESBWR

Follows process described in ESBWR Licensing topical reports:

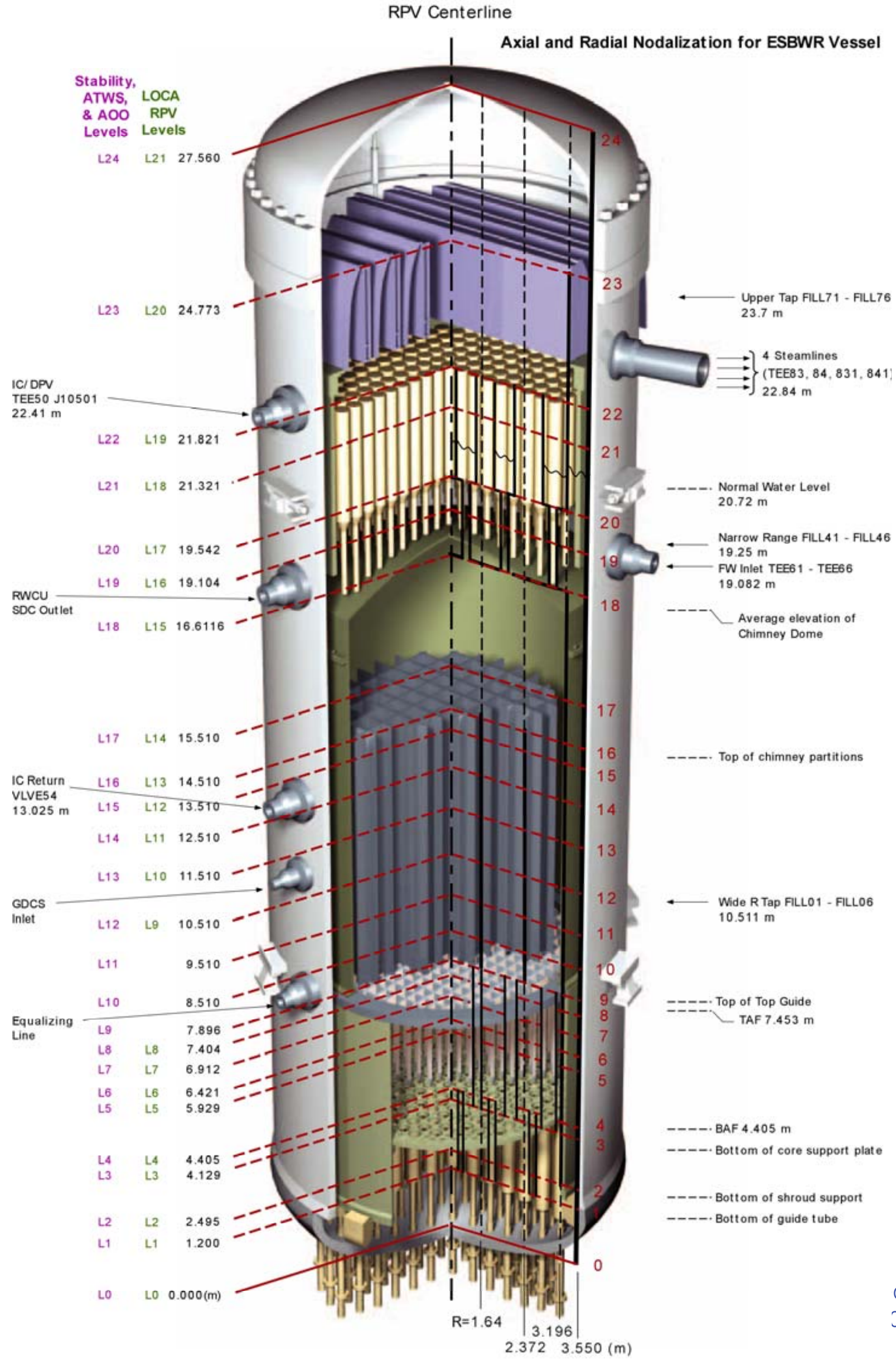
NEDE-33083 (LOCA and AOO's)

NEDE-33083 Supplement 1 (Stability)

TRACG model of ESBWR reactor vessel

2D Reactor vessel for LOCA

3D Reactor vessel for AOO, ATWS and Stability



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TRACG model of ESBWR reactor vessel

2D Reactor vessel for LOCA

3D Reactor vessel for AOO, ATWS and Stability

Sequence of Event for Loss of FW 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.

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Figure 15.2-1e. Loss of Feedwater Heating

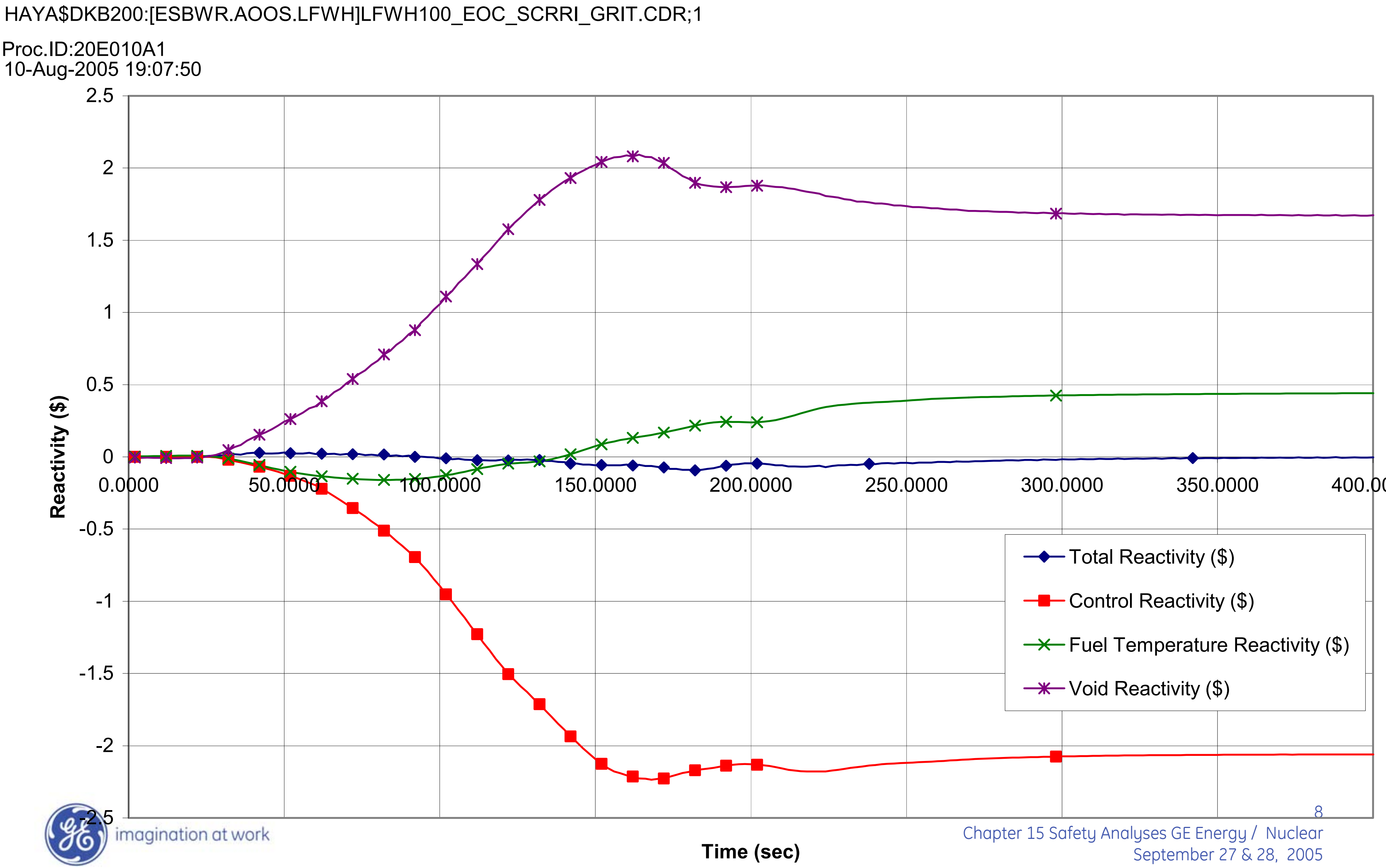

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Figure 15.2-1f. Loss of Feedwater Heating

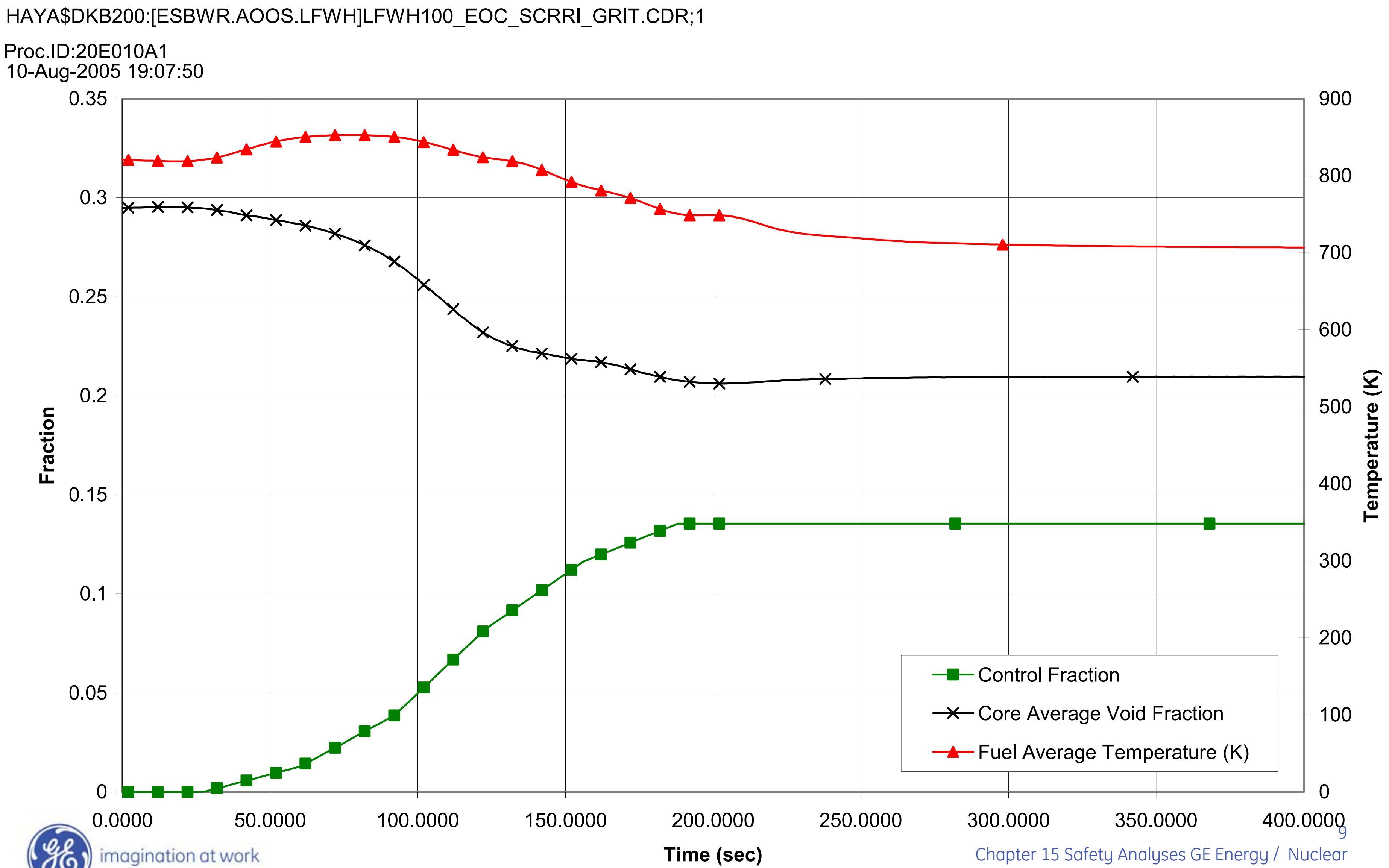

Figure 15.2-1a. Loss of Feedwater Heating

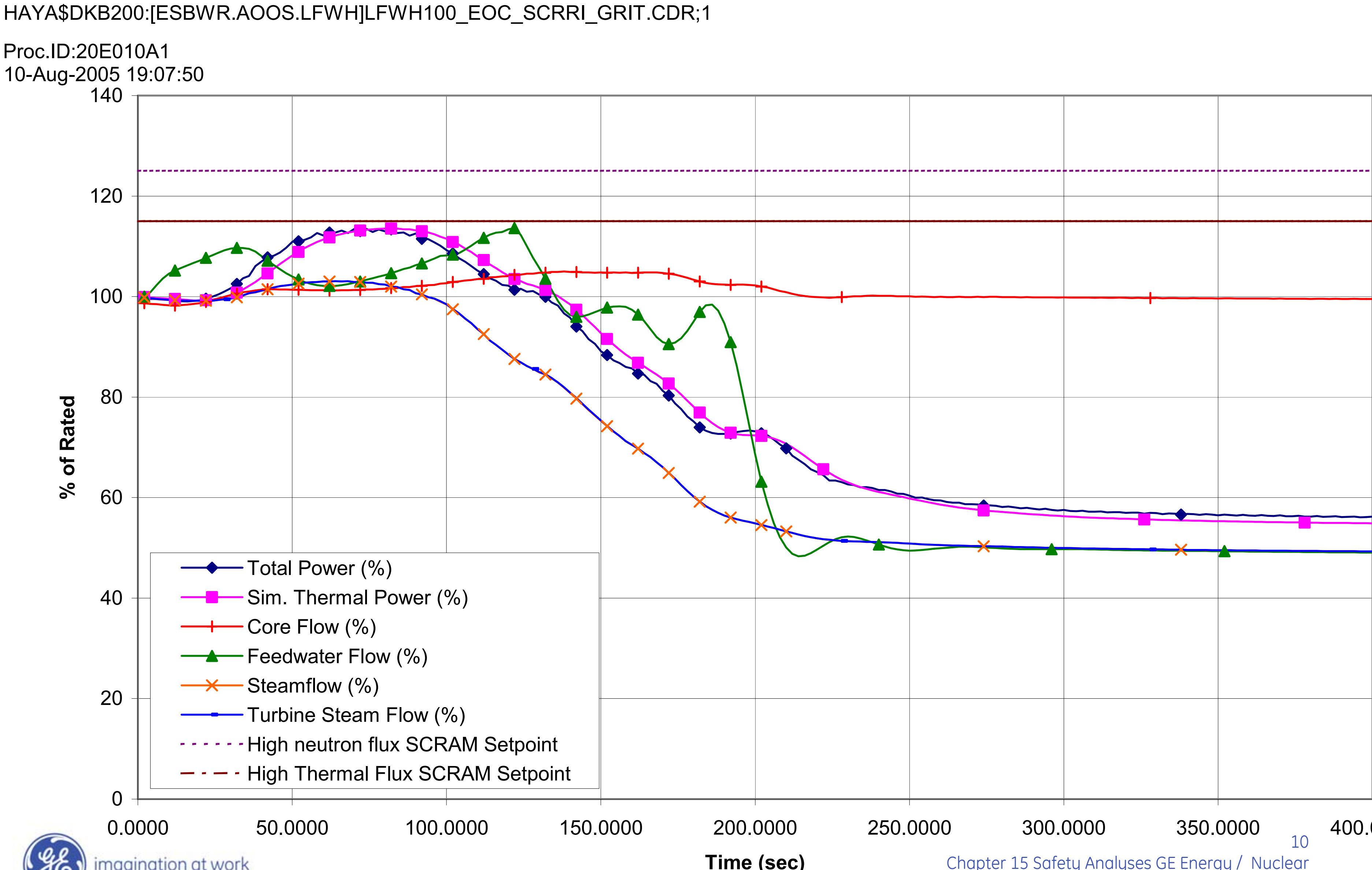

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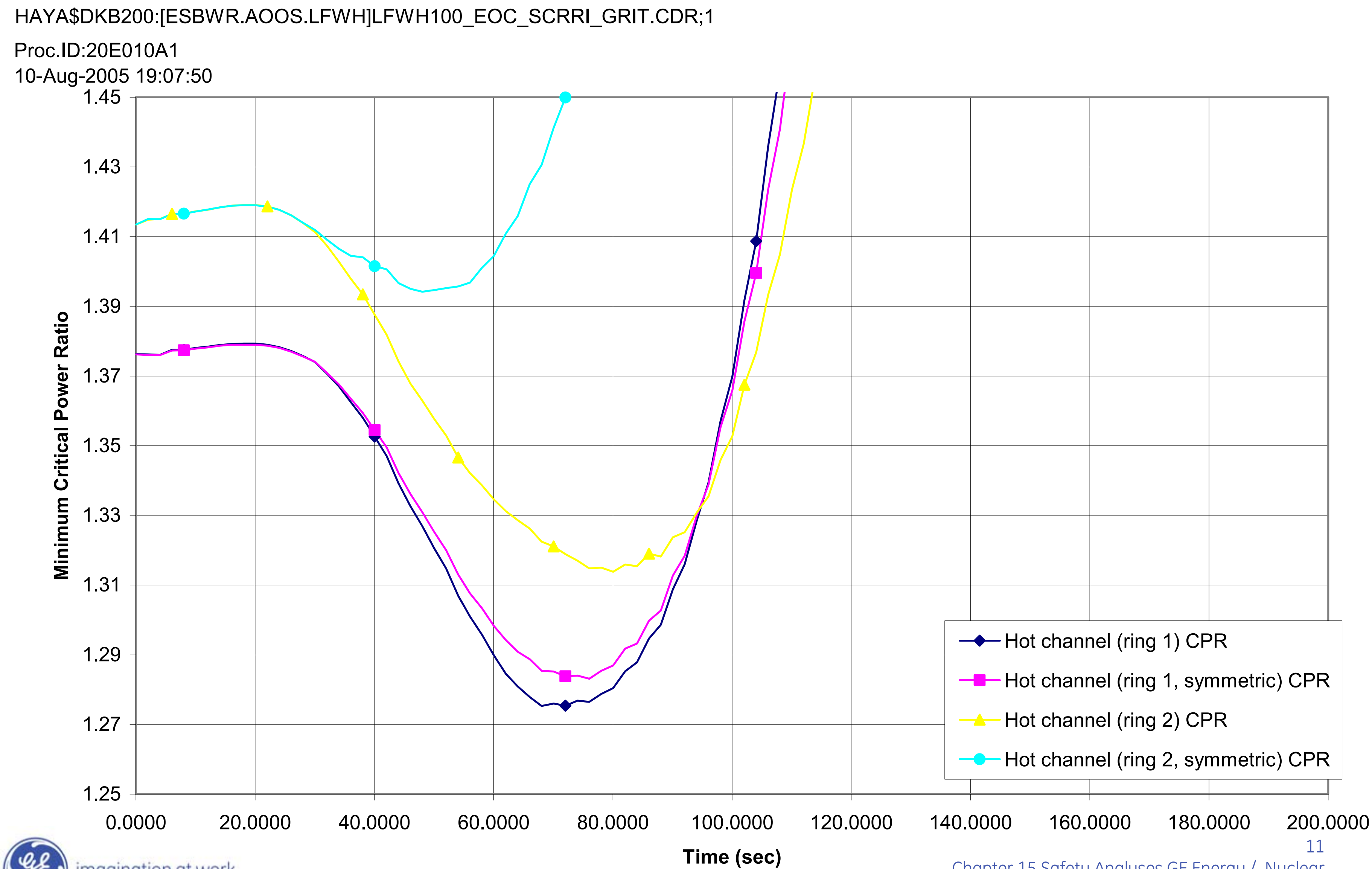
Figure 15.2-1g. Loss of Feedwater Heating



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Sequence of Events for Generator Load Rejection

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.
0.08	Turbine control valves closed.
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

Figure 15.2-4a. Generator Load Rejection

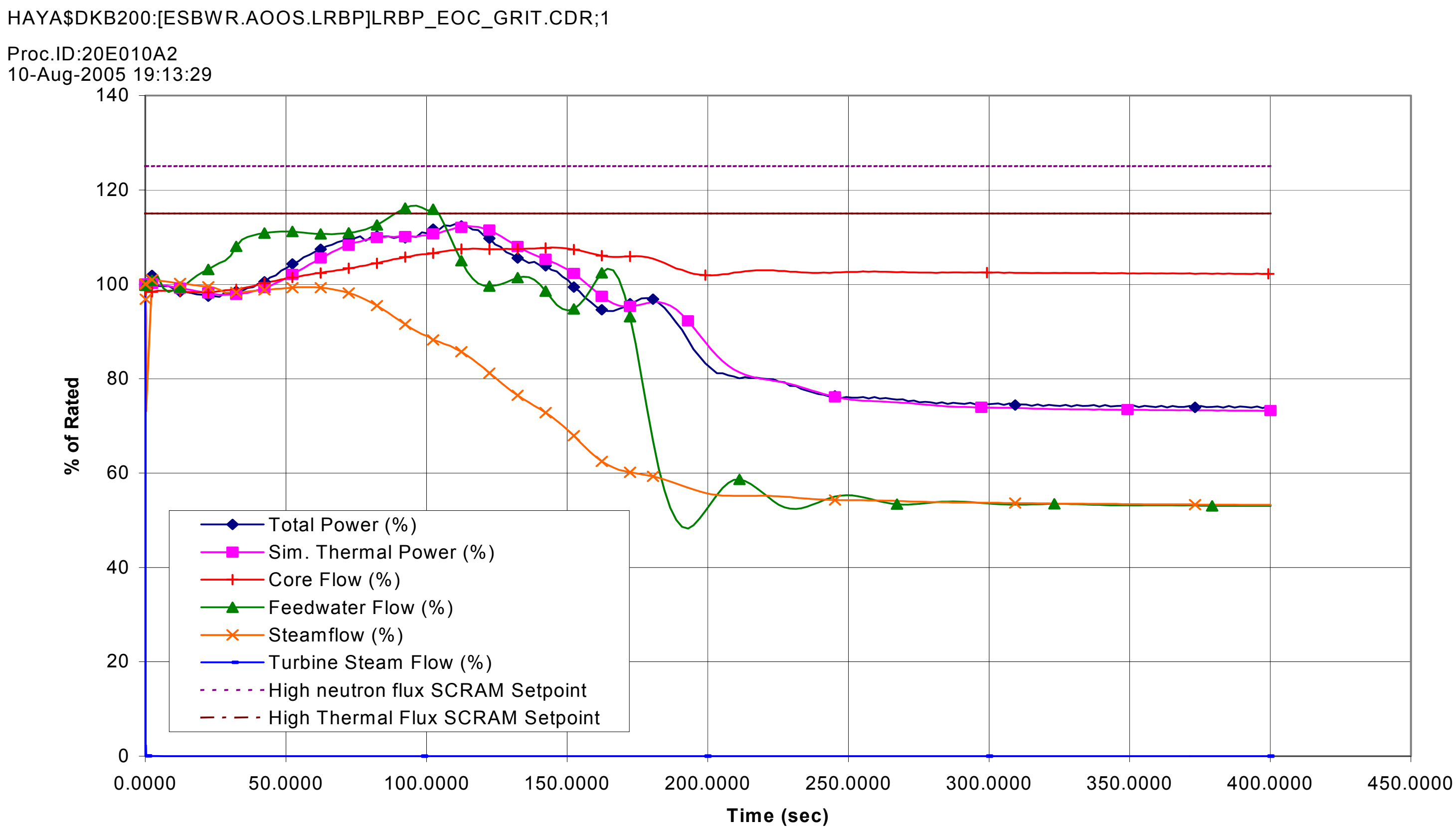

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Figure 15.2-4b. Generator Load Rejection

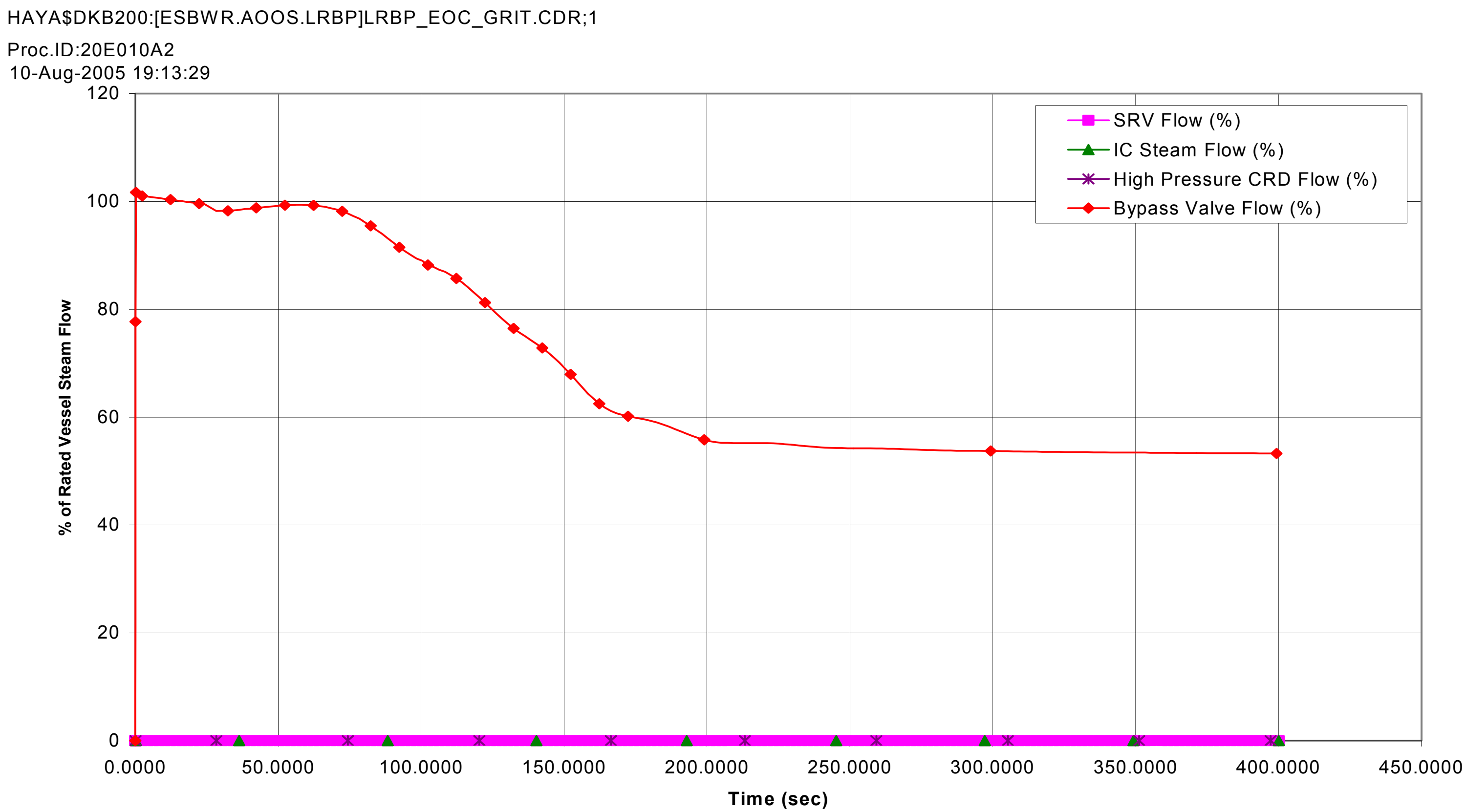

Figure 15.2-4d. Generator Load Rejection

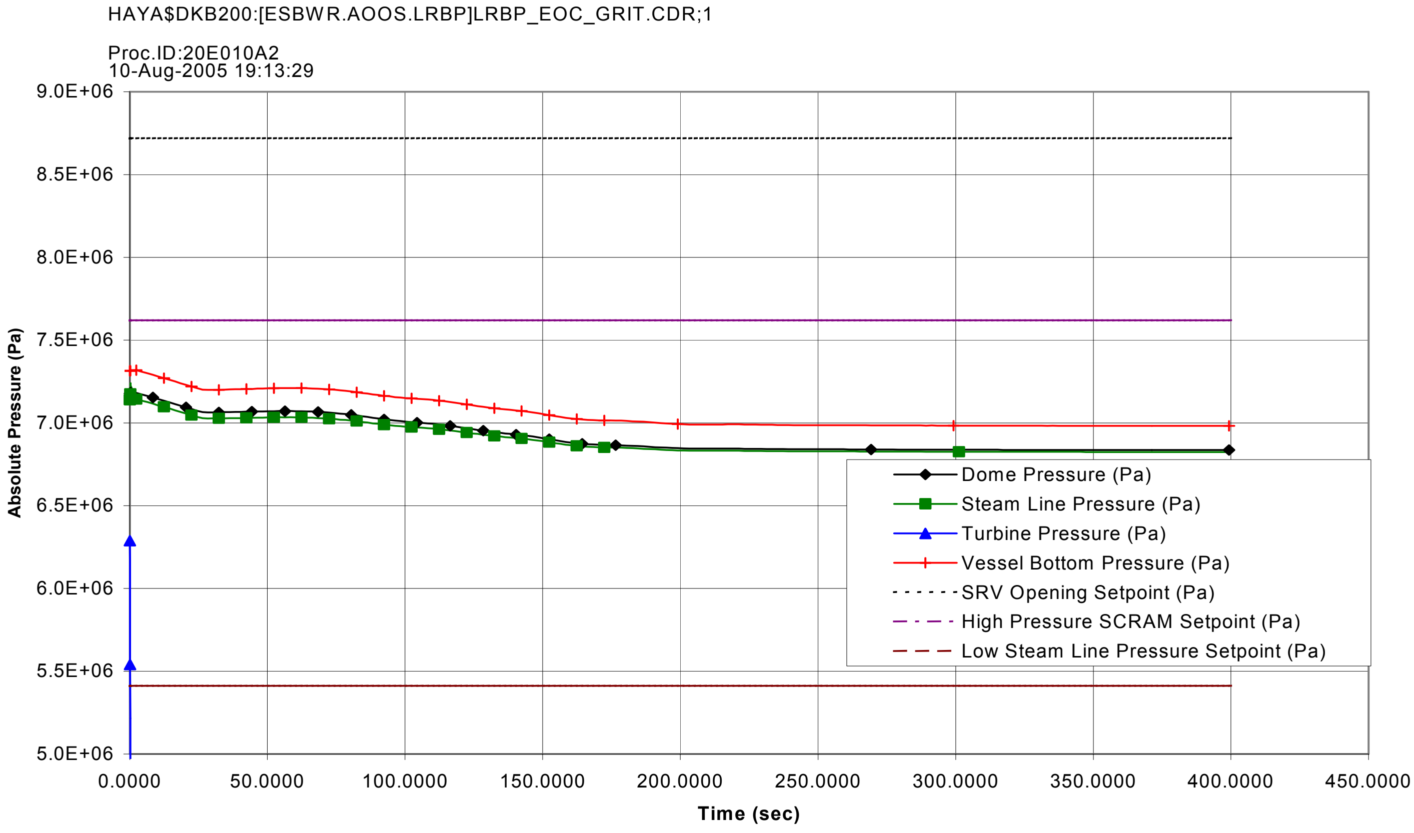

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Figure 15.2-4e. Generator Load Rejection

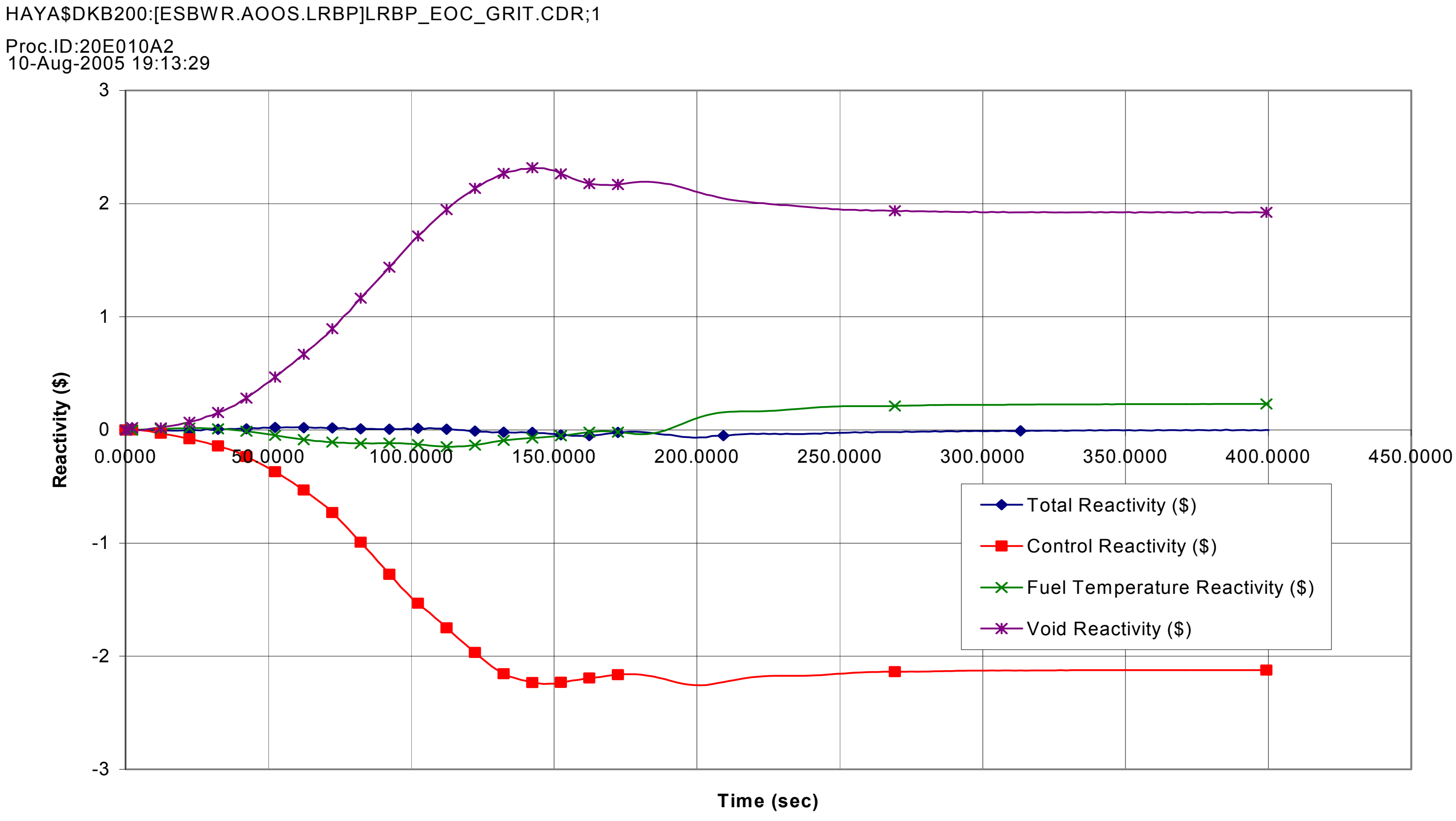

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Figure 15.2-4f. Generator Load Rejection

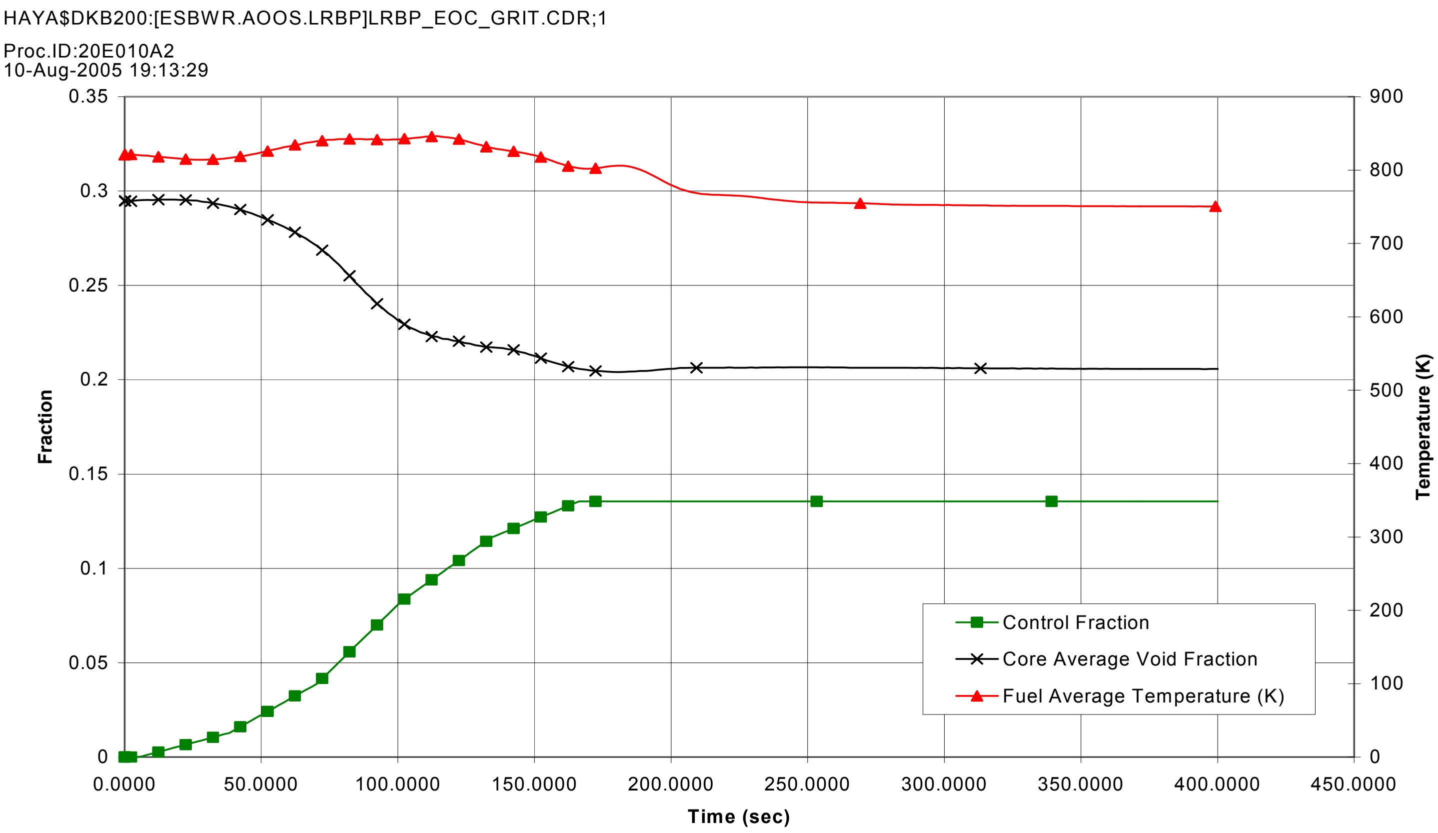
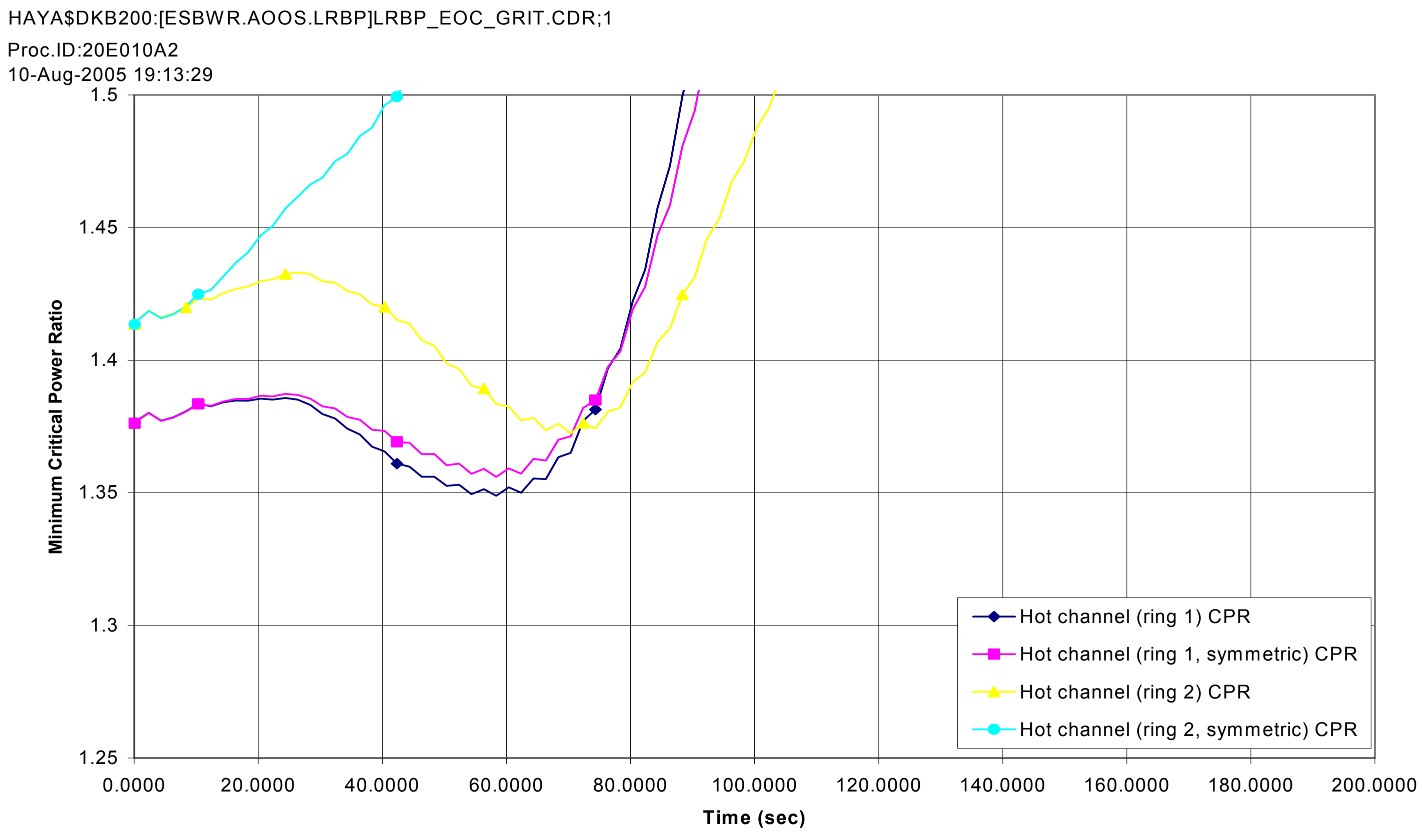


Figure 15.2-4g. Generator Load Rejection



Sequence of Events for 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 HPCRD is activated to recover the level

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

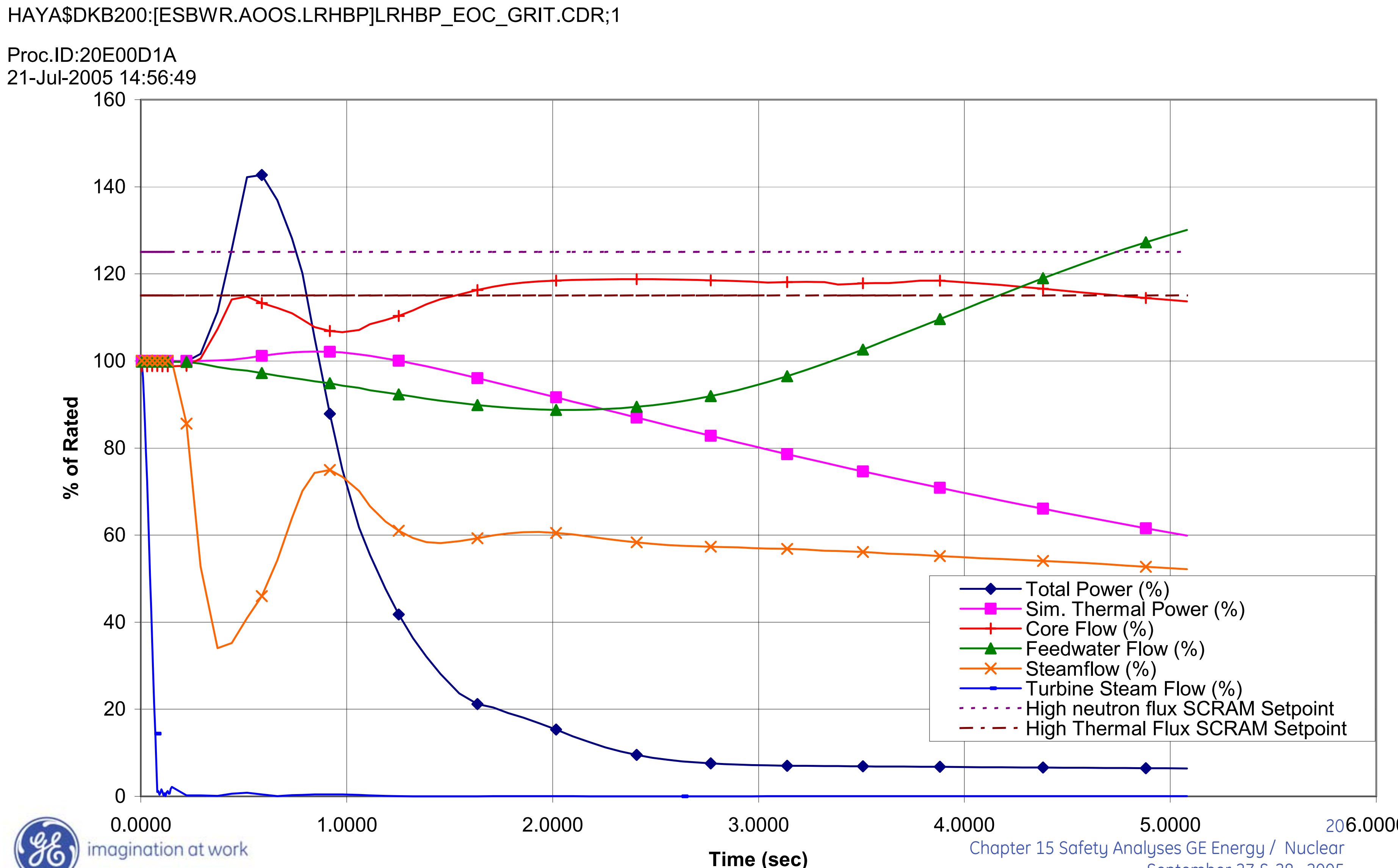


Figure 15.2-5b. Generator Load Rejection with a Single Failure in the Turbine Bypass System

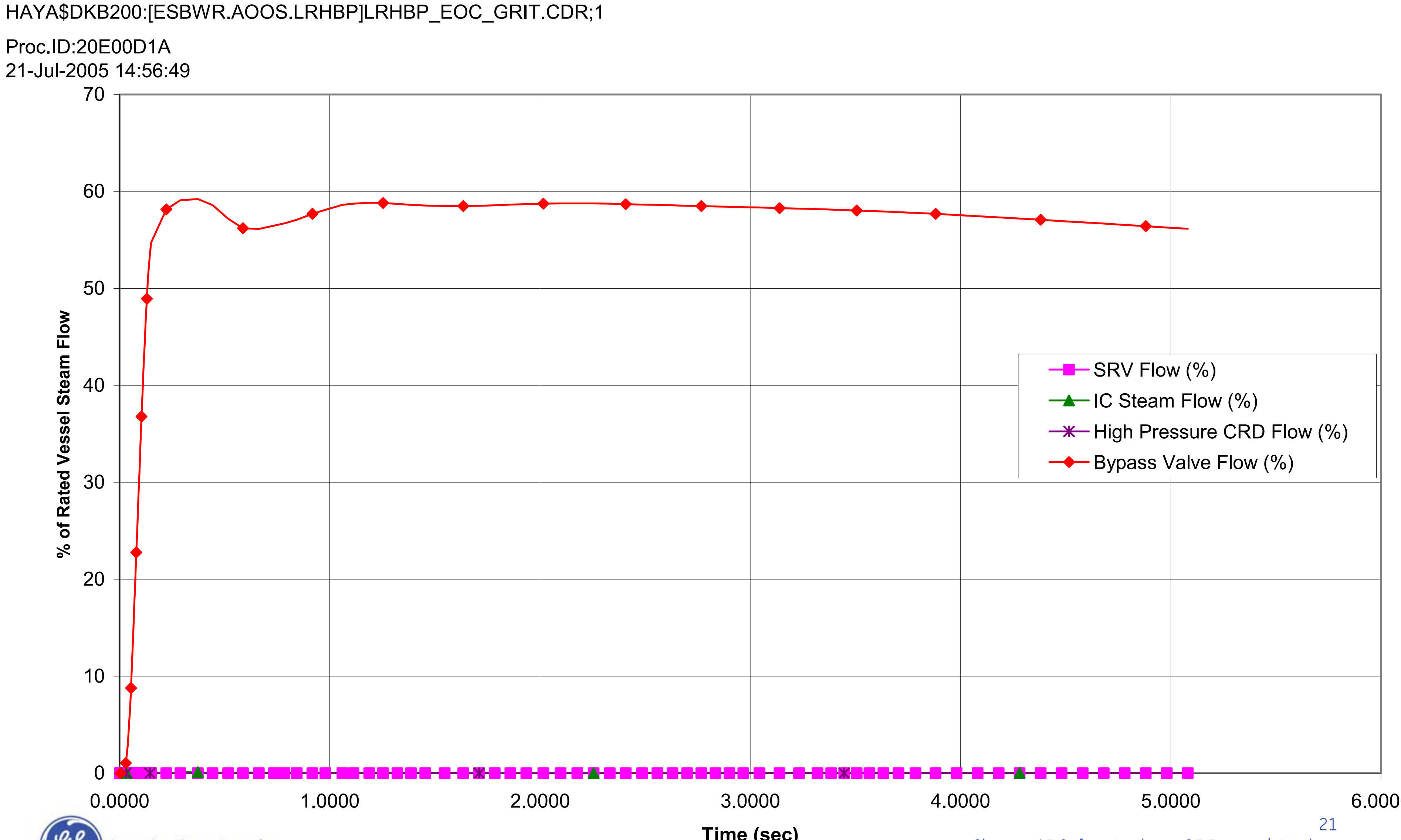


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Figure 15.2-5c Generator Load Rejection with a Single Failure in the Turbine Bypass System

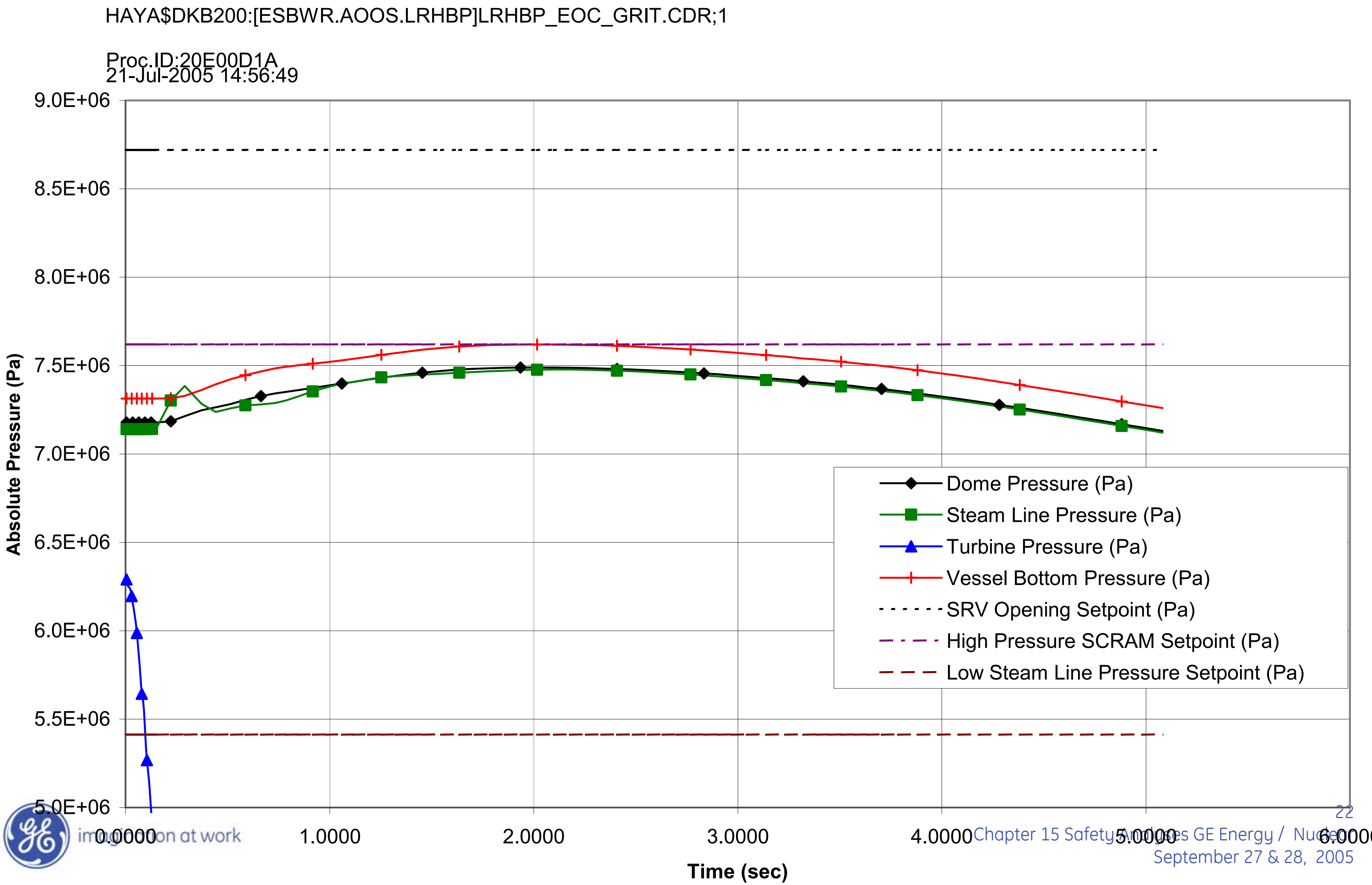

Figure 15.2-5e. Generator Load Rejection with a Single Failure in the Turbine Bypass System

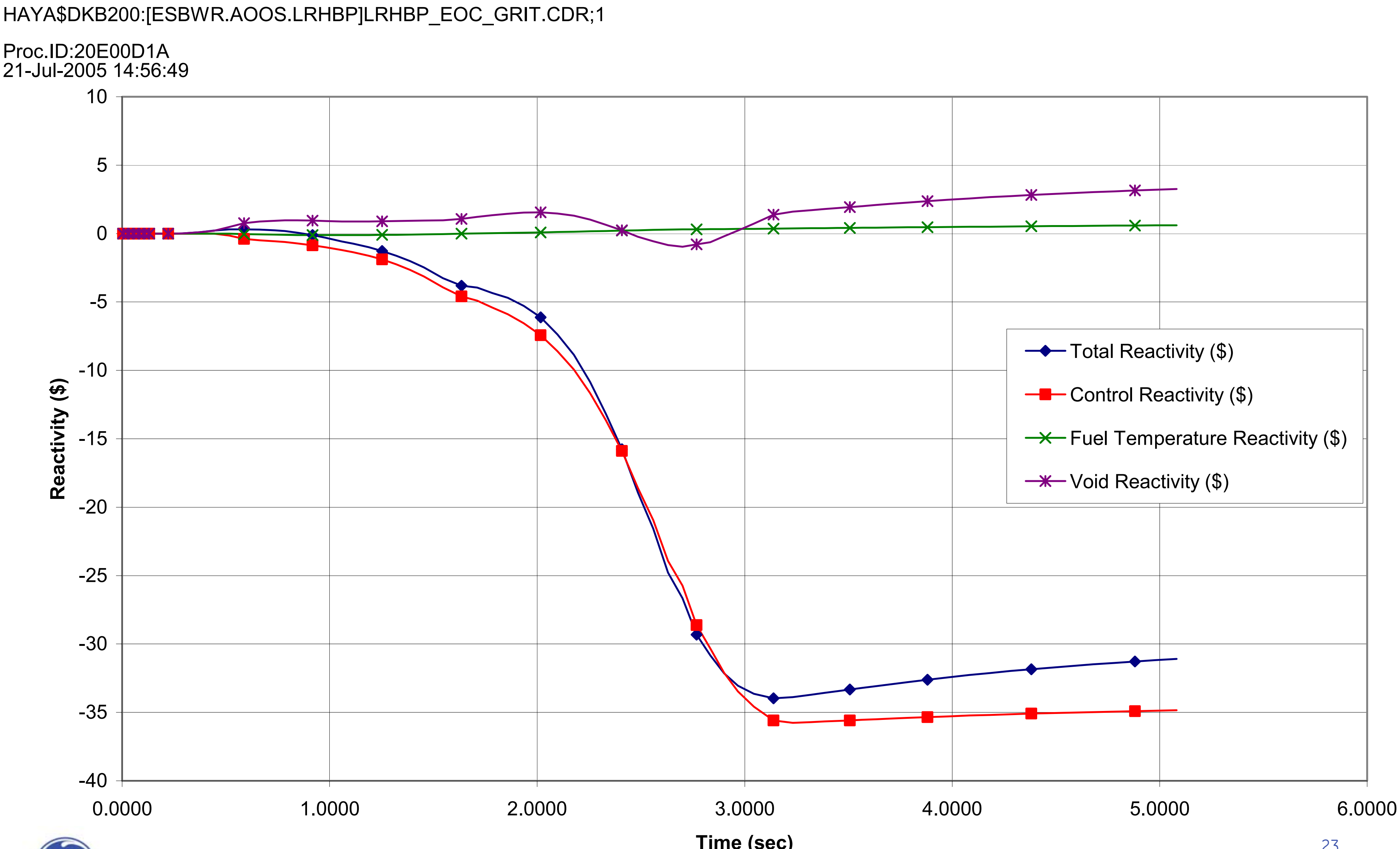

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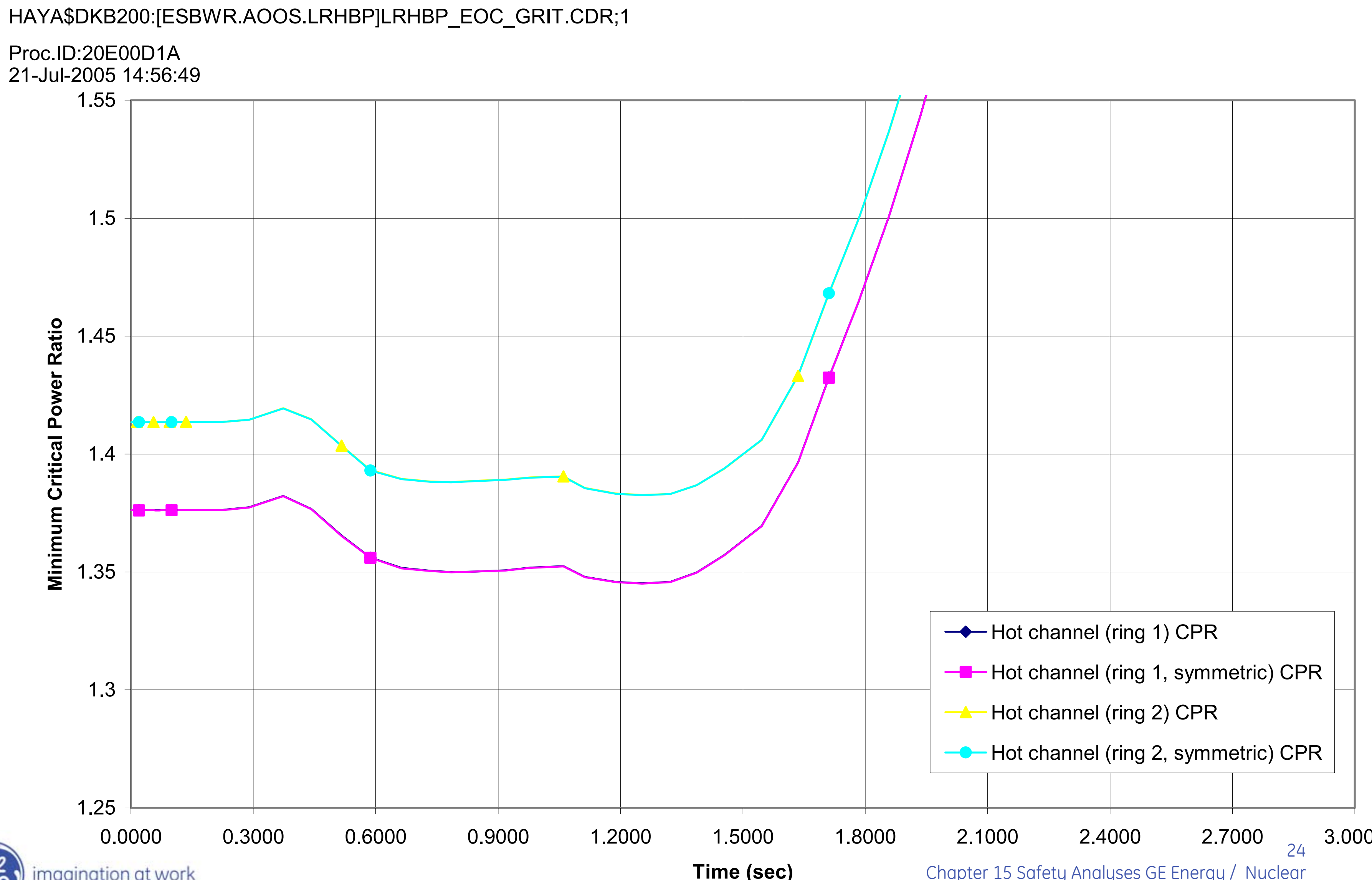
Figure 15.2-5g. Generator Load Rejection with a Single Failure in the Turbine Bypass System



Sequence of Events for Loss of Non-Emergency AC Power to Station Auxiliaries

Time (sec)	**Event ***
0.0	Loss of AC power to station auxiliaries, which initiates a generator trip.
0.0	Additional Failure assumed in transfer to "Island mode", Feedwater, condensate and circulating water pumps are tripped.
0.0	Turbine control valve fast closure is initiated.
0.0	Turbine control valve fast closure initiates main turbine bypass system operation.
0.0	Feedwater and condenser pumps are tripped.
0.02	Turbine bypass valves start to open.
0.08	Turbine control valves closed.
2.0	Loss of power on the four power generation busses is detected and initiates a reactor scram and activation of ICs.
5.0	Feedwater flow decay to 0.
6.0	Low condenser vacuum setpoint is detected and initiates turbine bypass closure.
6.0	Loss of condenser Vacuum rate is reduced due to bypass valve closure
6.61	Vessel water level reaches Level 3
7.0	ICs drops cold water inside the vessel
9.9	Vessel water level reaches Level 2.
14.0	Low-Low condenser vacuum signal closes the MSI valves .
33.0	ICs are at rated flow
145.0	CRD high pressure injection mode is initiated.
100	The level recovers above 13m
850	The level recovers above 15m

Figure 15.2-15a. Loss of Non-Emergency AC Power to Station Auxiliaries

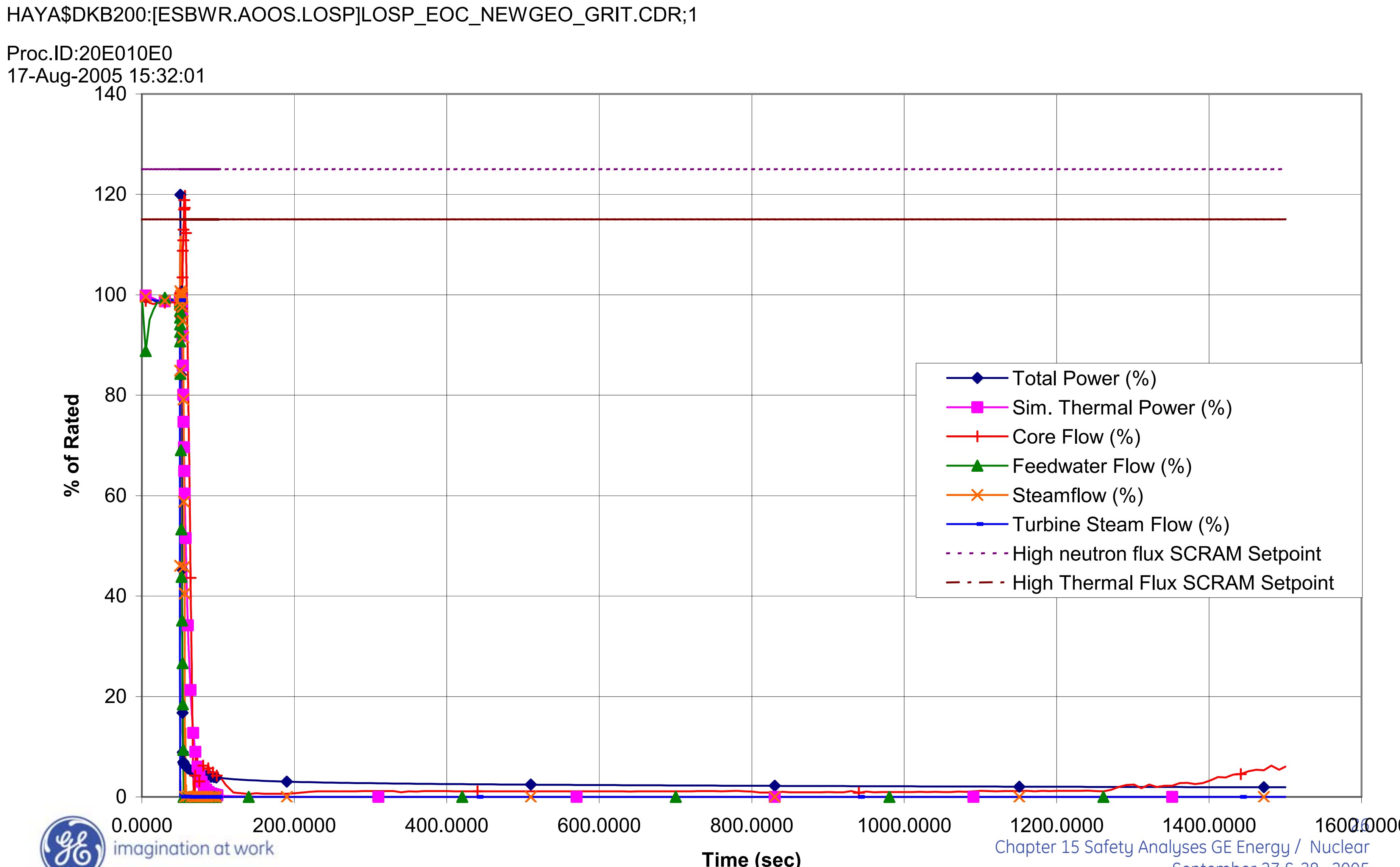


Figure 15.2-15b. Loss of Non-Emergency AC Power to Station Auxiliaries

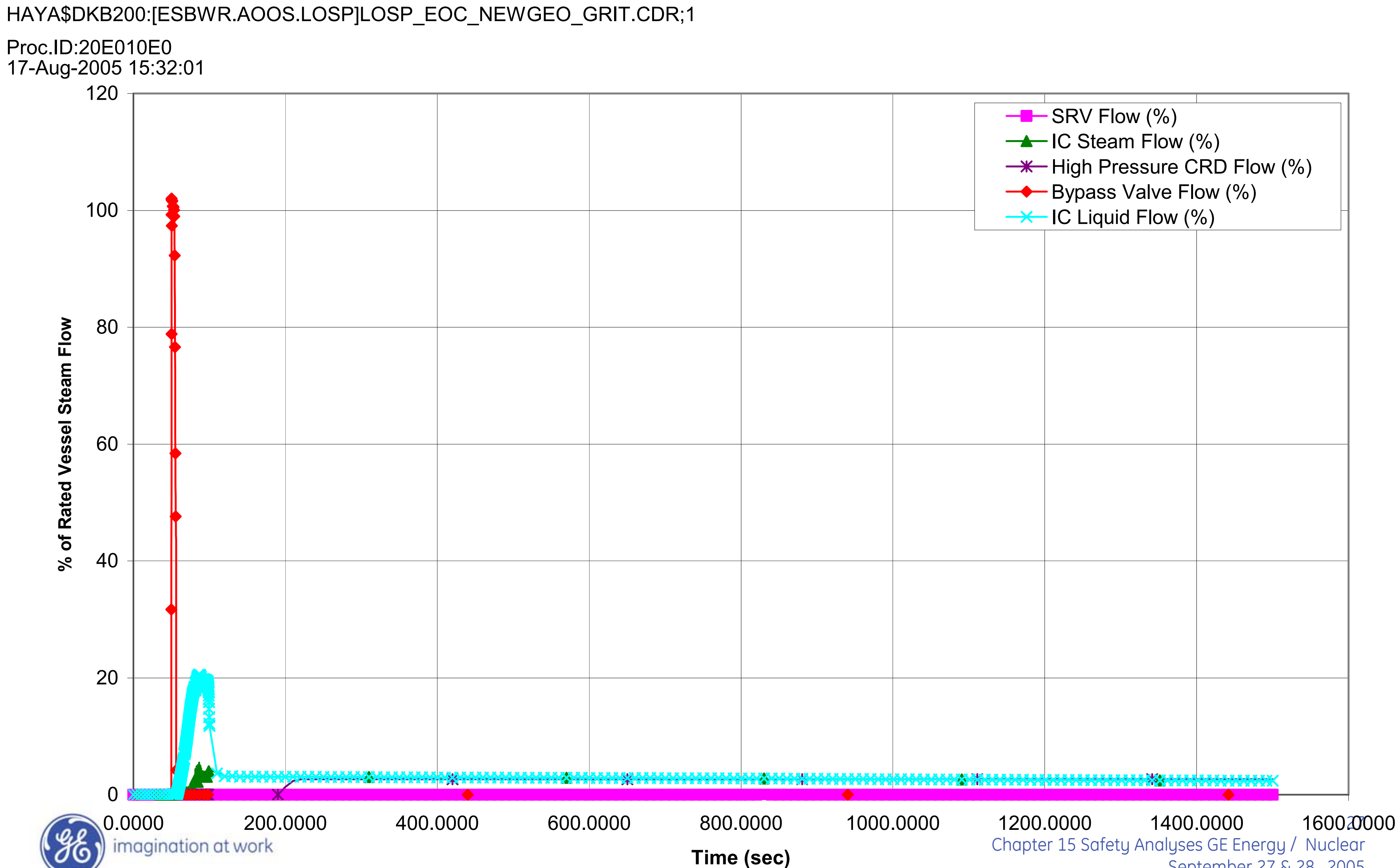


Figure 15.2-15c. Loss of Non-Emergency AC Power to Station Auxiliaries

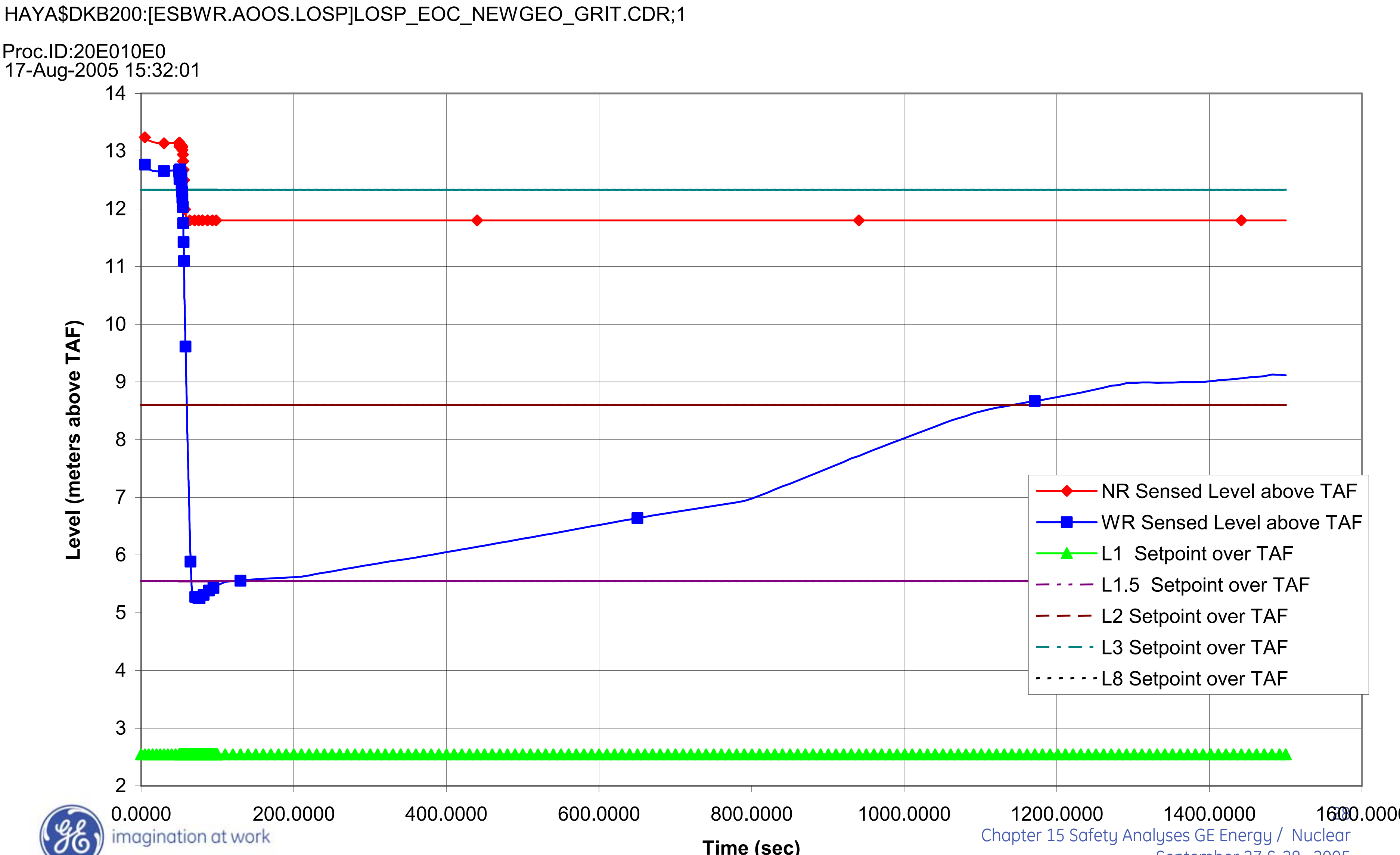


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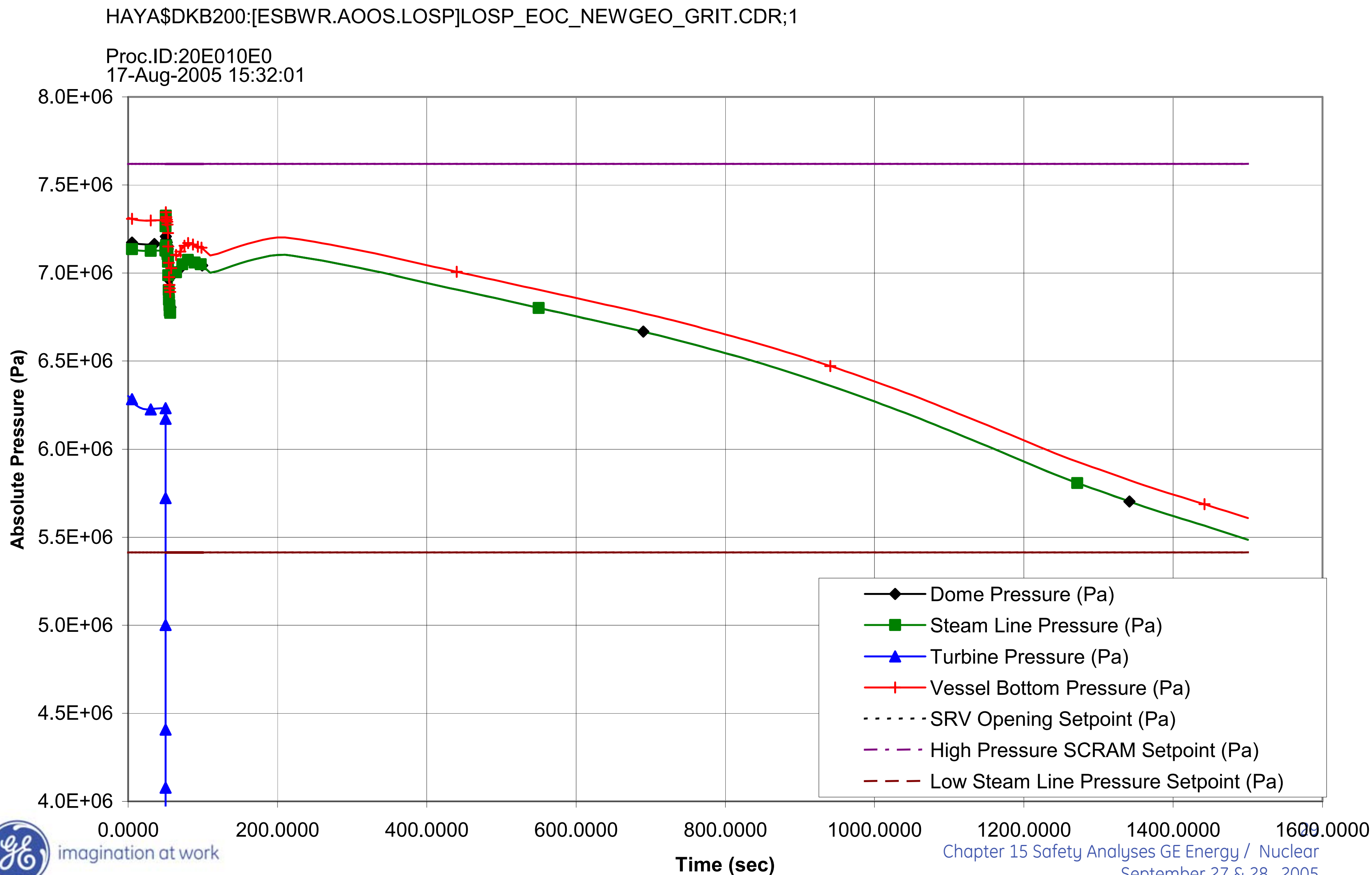
Figure 15.2-15d. Loss of Non-Emergency AC Power to Station Auxiliaries



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ble 15.3-6 Sequence of Events for Generator Load Rejection th 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

Loss FW Heating w/ SCRRI failure

Time (s)	Event
0	Initiate a 55.6°C (100°C) 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

Time (s)	Event
0	Initiate a 55.6°C (100°C) 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.

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

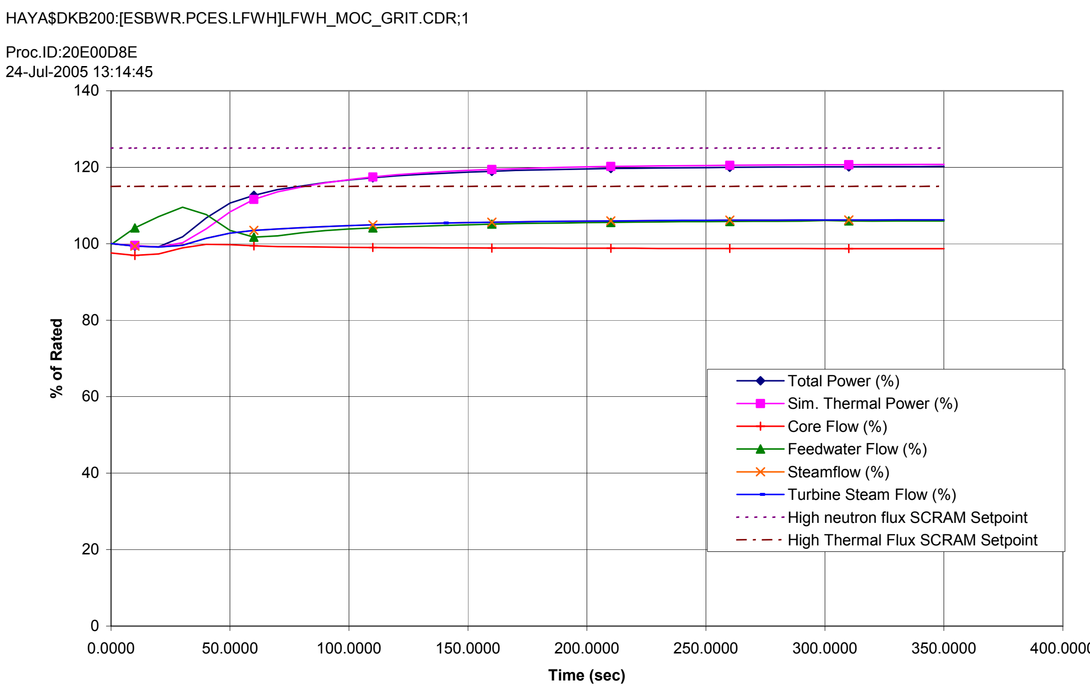

Figure 15.3-1b. Loss of Feedwater Heating with SCRRI Failure

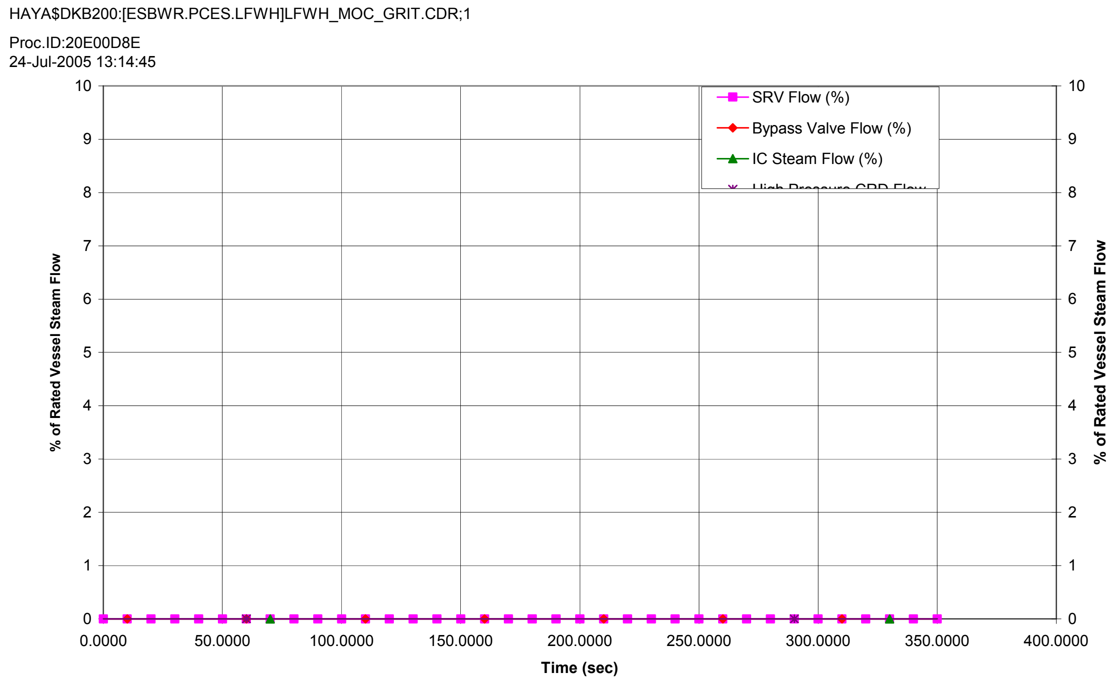

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Figure 15.3-1c. Loss of Feedwater Heating with SCRRI Failure

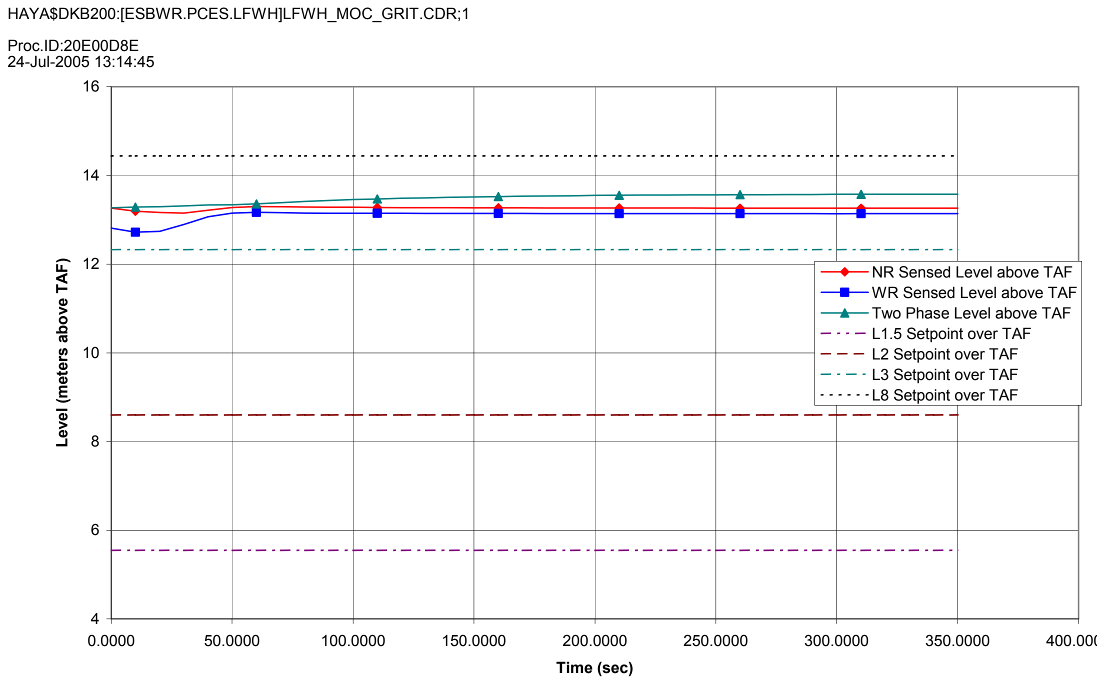

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Figure 15.3-1d. Loss of Feedwater Heating with SCRRI Failure

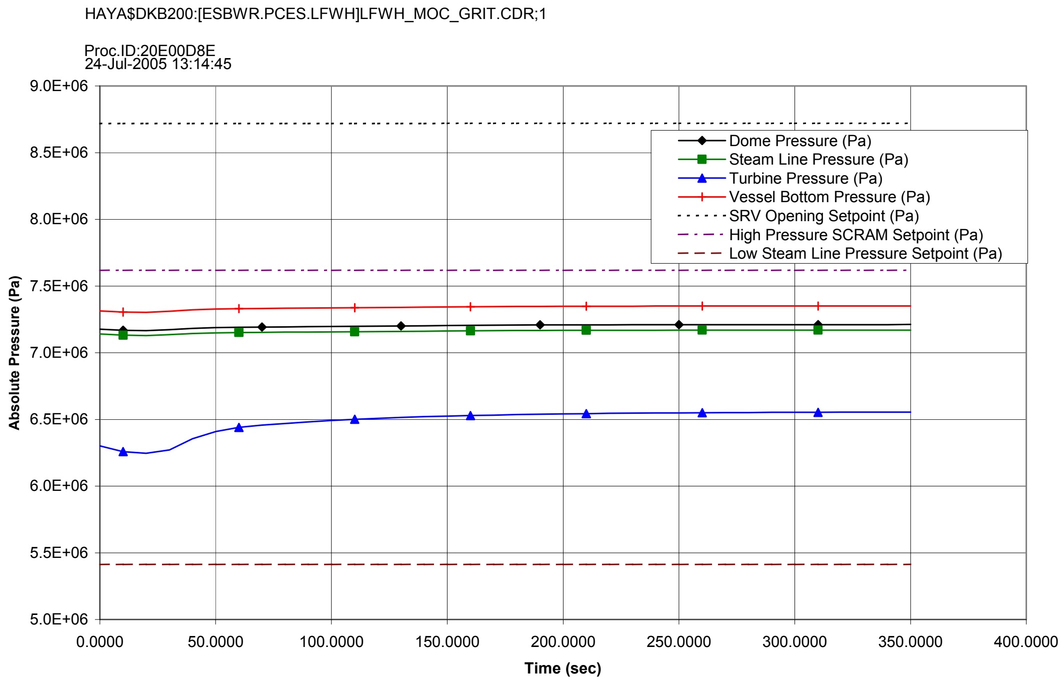

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Figure 15.3-1e. Loss of Feedwater Heating with SCRRI Failure

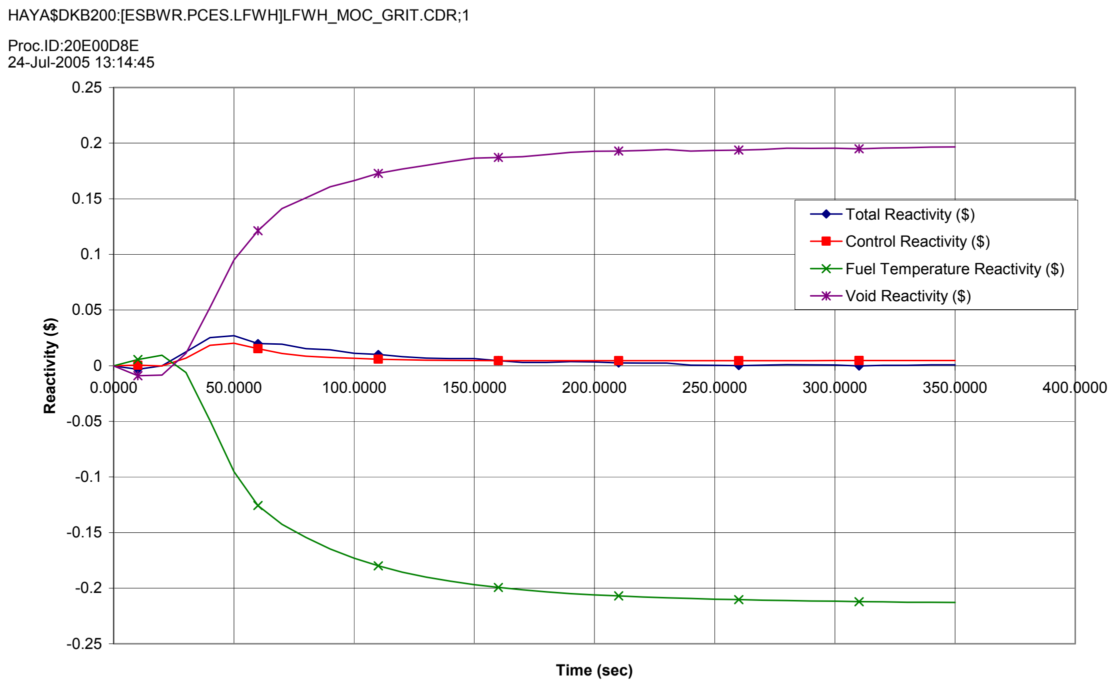

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Figure 15.3-1f. Loss of Feedwater Heating with SCRRI Failure

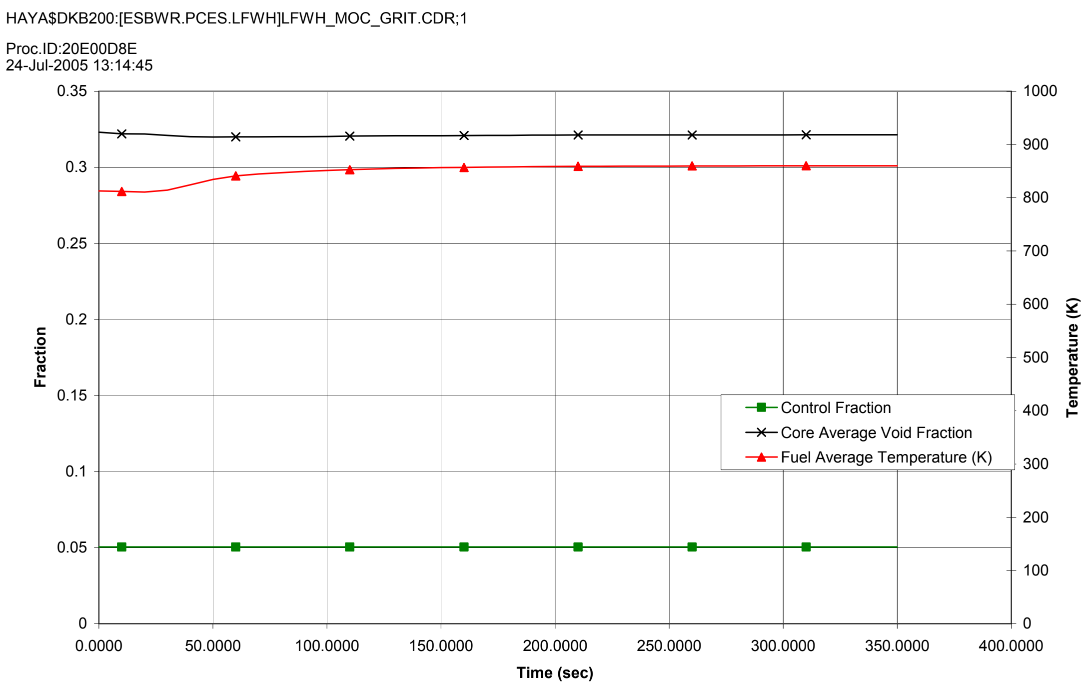

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Figure 15.3-1g. Loss of Feedwater Heating with SCRRI Failure

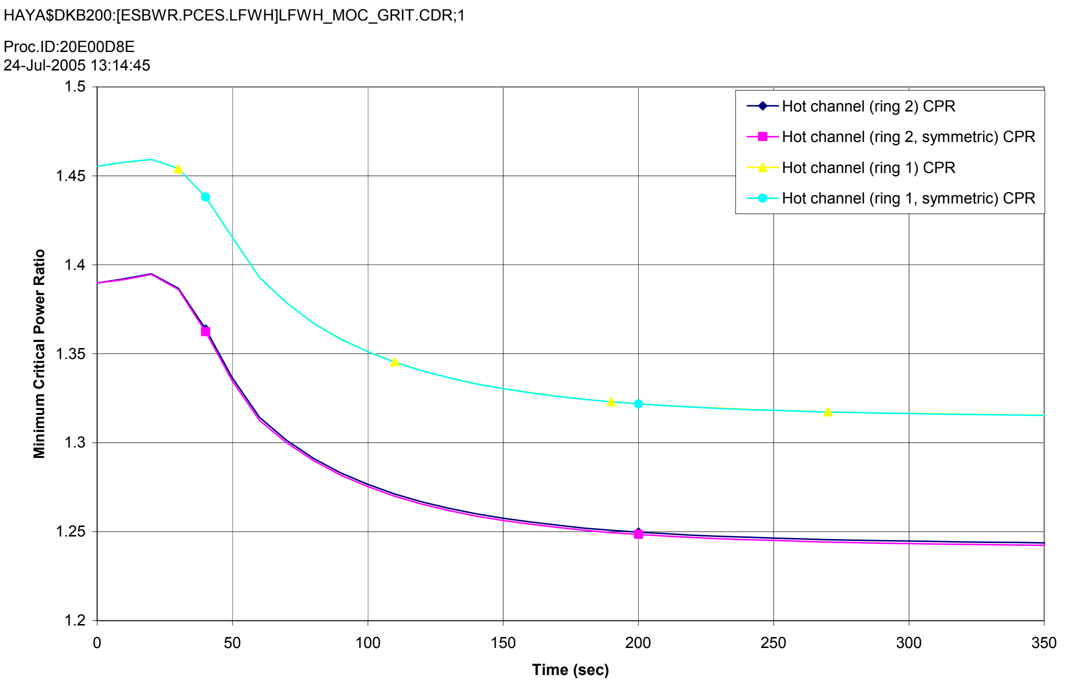

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Table 15.3-6. Sequence of Events for Generator Load Rejection With Total Turbine Bypass Failure

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

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

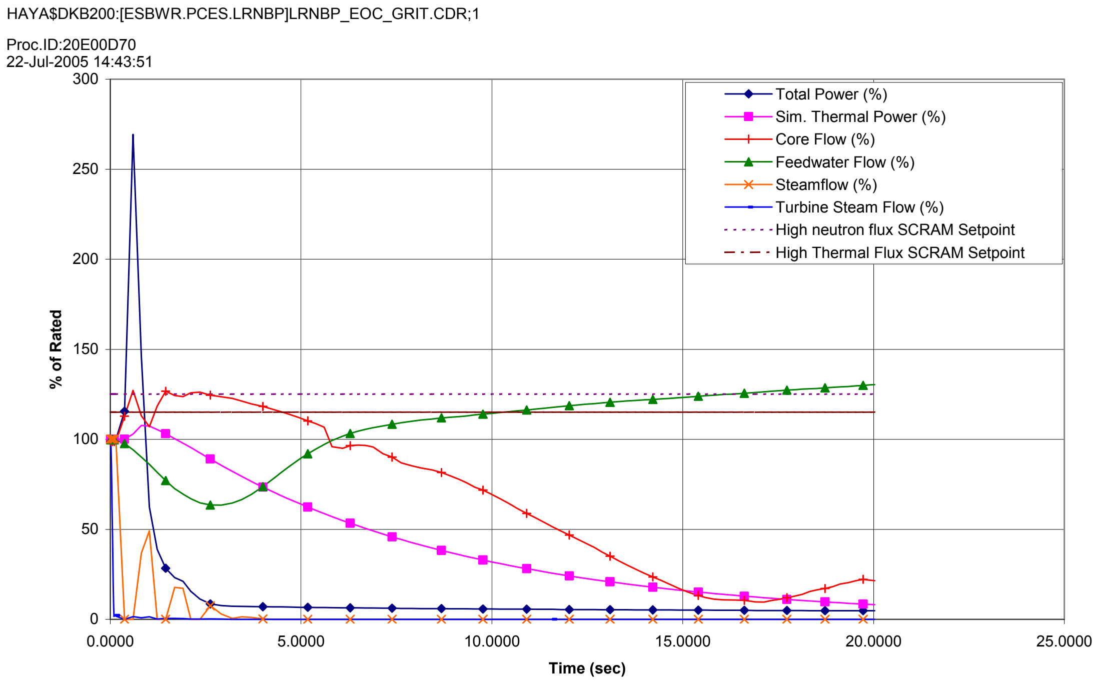

Figure 15.3-5b. Generator Load Rejection With Total Turbine Bypass Failure

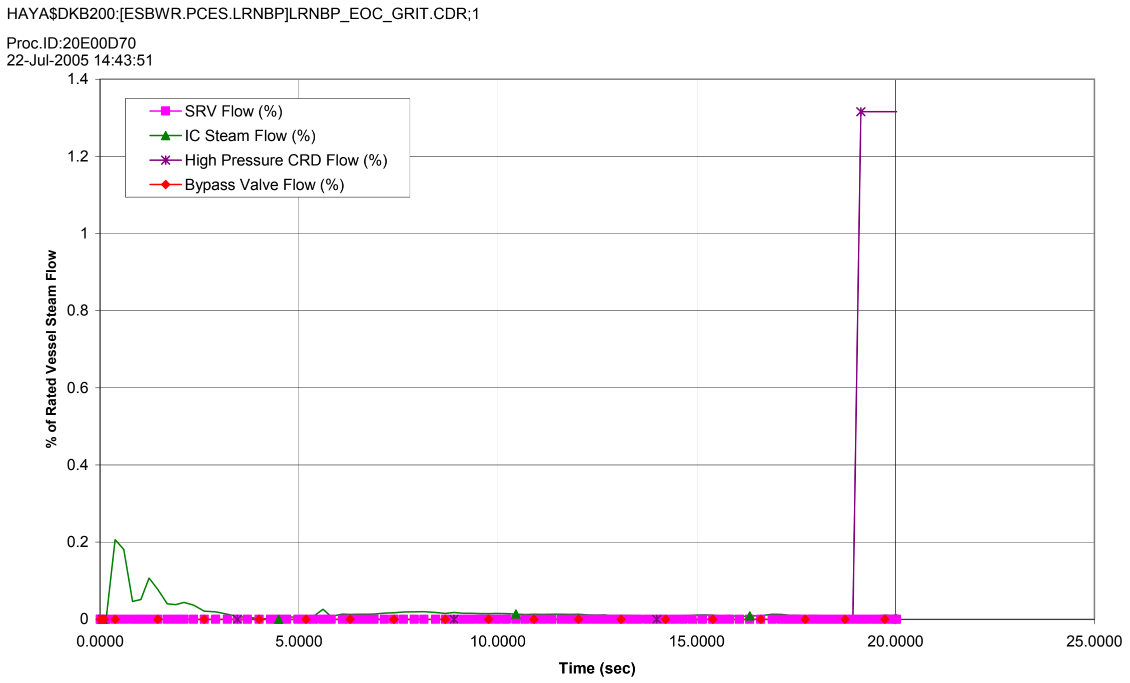

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Figure 15.3-5c. Generator Load Rejection With Total Turbine Bypass Failure

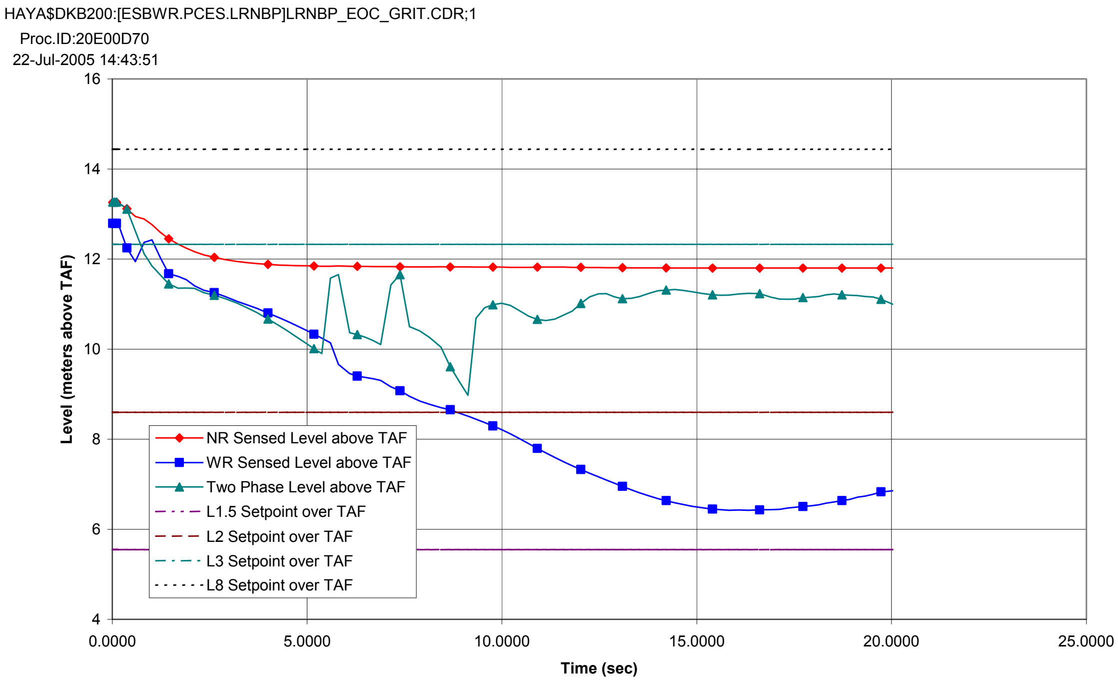

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Figure 15.3-5d. Generator Load Rejection With Total Turbine Bypass Failure

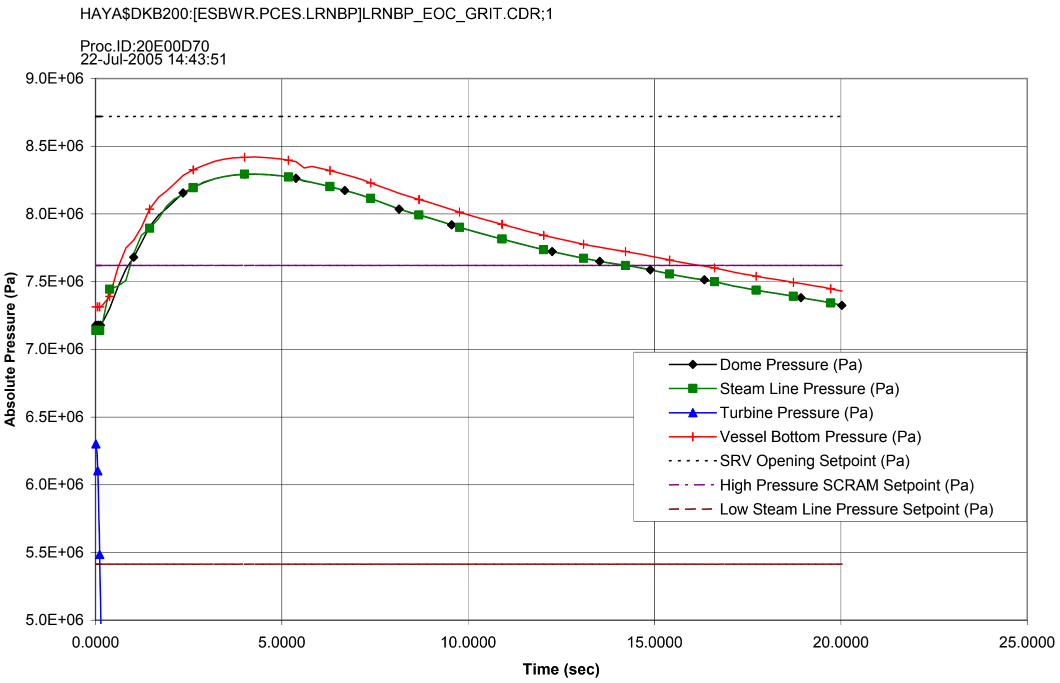

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Figure 15.3-5e. Generator Load Rejection With Total Turbine Bypass Failure

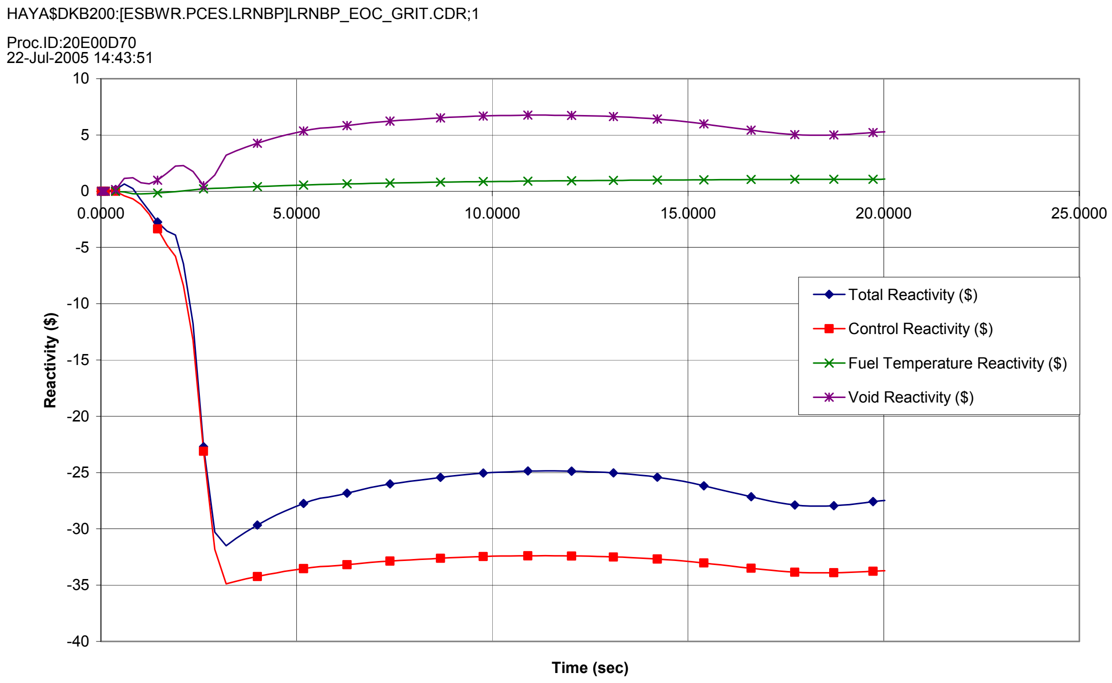

Figure 15.3-5f. Generator Load Rejection With Total Turbine Bypass Failure

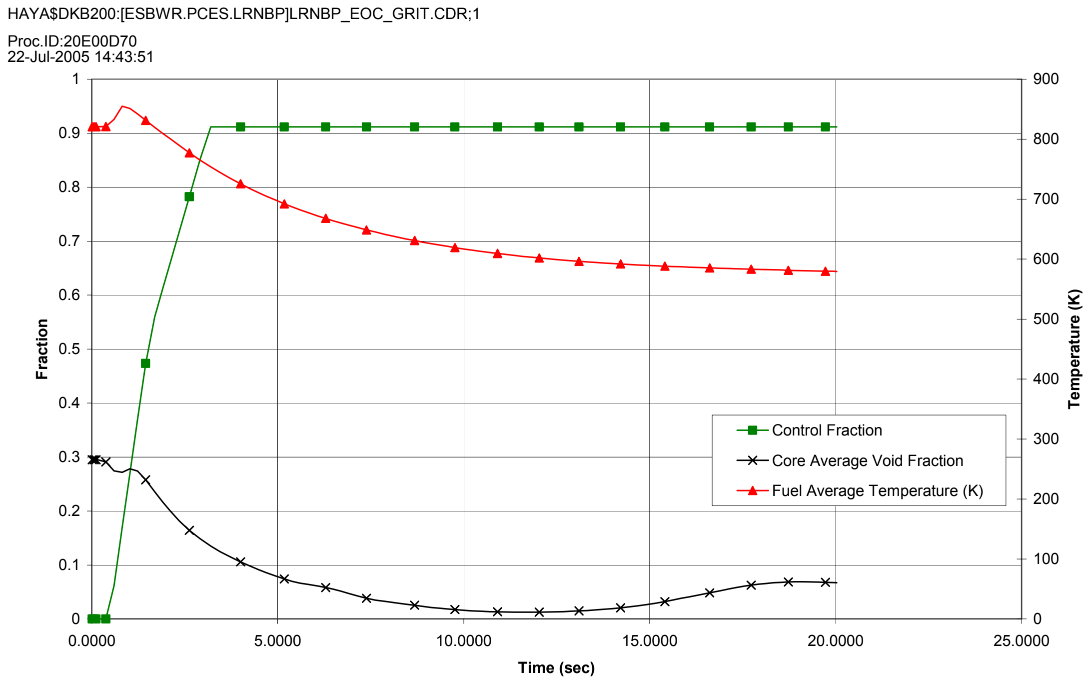

Figure 15.3-5g. Generator Load Rejection With Total Turbine Bypass Failure

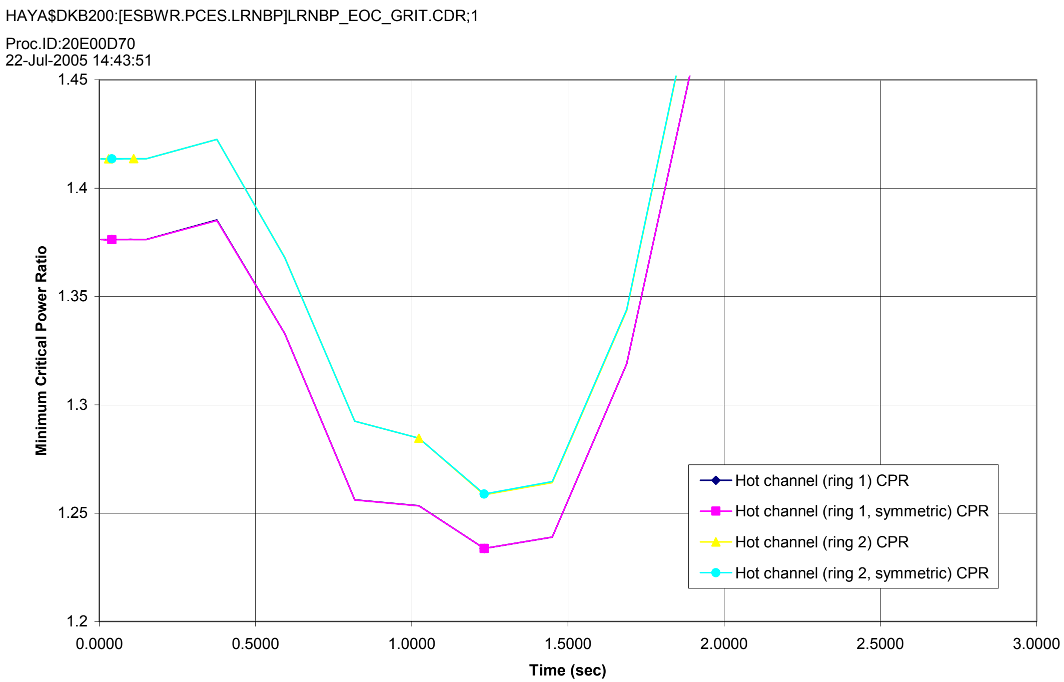

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Table 15.3-8. Sequence of Events for Continuous Control Rod Withdrawal Error During Reactor Startup

Time (sec)	Events
0	Operator withdraws a gang of rods (or a single rod) continuously; or a gang of rods (or single rod) is withdrawn continuously due to a malfunction of the Automated Rod Movement Control System
~6	Neutron flux increases rapidly due to the continuous reactivity addition, with a very short period
~14	The SRNM Period-Based Rod Block Trip initiates rod block due to short period (less than the 20-second setpoint)
~25	The SRNM Period-Based Scram Trip initiates reactor scram due to short period (less than the 10-second setpoint)
~27	Reactor is scrammed (all rods inserted) and the event is terminated

Table 15.3-9 Sequence of Events for the Mislocated Bundle

(1)	During the core loading operation, a bundle is loaded into the wrong core location.
(2)	Subsequently, the bundle designated for this location is incorrectly loaded into the location of the previous bundle.
(3)	During the core verification procedure, the two errors are not observed.
(4)	The plant is brought to full power operation without detecting misplaced bundles.
(5)	The plant continues to operate throughout the cycle.

Table 15.3-10 Sequence of Events for the Misoriented Bundle

(1)	During the core loading operation, a bundle is rotated and loaded with incorrect orientation.
(2)	During the core verification procedure, the orientation error is not observed.
(3)	The plant is brought to full power operation without detecting the misoriented bundle.
(4)	The plant continues to operate throughout the cycle.

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Misloaded Fuel Bundle

ESBWR follows GESTAR Amendment 28 licensing under review by NRC

Measures to assure Misloaded fuel bundle remains an Infrequent event

Periodic confirmatory actions by licensee

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Rod Withdrawal Error at Full Power

ESBWR follows ABWR certified design – ATLM (Automated Thermal Limits Monitor) monitors approach to the OL MCPR (Operating Limit Min.Critical Power Ratio).

Blocks rod prior to reaching OLMCPR, SAFDL is not approached.

Rod Withdrawal Error During Startup

Short period rod block and scram

Backed up by 15% power scram setpoint during startup

Fuel enthalpy maintained below clad failiure threshold.

Control Rod Drop

ESBWR design same as ABWR Control Rod Drive design, in that a load cell detects the weight of the CR blade

Withdrawal block occurs if the blade separates from the drive and is hung up in the core.

Because the drive can not be withdrawn out from under a disconnected blade, the blade can not drop.

Drive design makes this an incredible event.

DCD Chapter 15 Radiological Analyses

Accidents/Events Considered with Radiological Consequences

LOCA Inside Containment

Fuel Handling Accident

Main Steam Line Break Accident Outside Containment

Feedwater Line Break Outside Containment

Reactor Water Cleanup / Shutdown Cooling System Line Failure Outside Containment

Failure of Small Line Carrying Primary Coolant Outside Containment

Liquid Containing Tank Failure

Spent Fuel Cask Drop Accident

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

Waste Gas System Failure (DCD Chapter 11)

DCD Chapter 15 Radiological Analyses (cont.)

Conformance to Regulatory Documents

Regulatory Guide 1.183 – “Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Plants”

SRP 15.7.5 – “Spent Fuel Cask Drop Accidents”

SRP 15.6.2 – “Radiological Consequences of the Failure of Small Lines Carrying Primary Coolant Outside Containment”

Various other SRPs

Computer Codes Used in Analyses

RADTRAD 3.03

CONAC04A (GE radiological code for design basis accidents)

Table 15.4-5 Loss-of-Coolant Accident Parameters

I. Data and Assumptions used to estimate source terms.	
A. Power Level, MWt	4590
B. Fraction of Core Inventory Released	RG 1.183
C. Iodine Chemical Species	
Elemental, %	4.85
Particulate, %	95
Organic, %	0.15
D. PCCS Decontamination Factors	
Noble Gas	1
Elemental iodine	10
Particulates	10
Organic iodines	1
II. Data and Assumptions used to estimate activity released	
A. Primary Containment Leakage	
Leak rate, %/day	0.5
MSIV leakage (total all lines), cfm	0.58
B. Reactor Building Leakage	
Leak rate, %/day	100
C. Condenser Data	
Free air volume, ft^3	2.20E+05
Fraction of volume involved, %	20
Iodine removal factors	
Particulate, %	99.5
Elemental, %	99.5
Organic, %	0

Table 15.4-5 Loss-of-Coolant Accident Parameters (Cont.)

III. Control Room Parameters	
A. Control Room Volume, ft^3	9.27E+04
B. Recirculation Rates	
0-72 hours	
Unfiltered inflow, ft^3/min	0
> 72 hours	
Filtered in leakage, ft^3/min	500
Recirculation rate, ft^3/min	250
Filter efficiency, %	99
Unfiltered inflow, ft^3/min	0
IV. Dispersion and Dose Data	
A. Meteorology	Table 15.4-9
B. Method of Dose Calculation	RG 1.183
C. Dose Conversion Assumptions	RG 1.183
D. Activity / Inventory Releases	Tables 15.4-6, 7, 8
E. Dose Evaluations	Table 15.4-9

Table 15.4-9. LOCA Inside Containment Analysis Results

Exposure Location	Meteorology (s/m^3)	Maximum Calculated TEDE (rem)	10 CFR 50.67 Acceptance Criterion TEDE (rem)
Exclusion Area Boundary (EAB)	1.00E-03	5.0	25
Outer boundary of Low Population Zone (LPZ)	1.35E-04 (0-8 h) 1.00E-04 (8-24 h) 5.40E-05 (1-4 d) 2.20E-05 (4-30 d)	5.6	25
Control Room	1.00E-03	0.2	5

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Table 15.5-1
ATWS Performance Requirements

Case	**RPV Peak Pressure** MPa (psia)	**Maximum Pool Temperature** °C (°F)	**Fuel Integrity**	**Maximum Containment Pressure** kPa (psia)
ARI	10.34 (1500)	121(250)	Coolable Geometry	414(60)
FMCRD Run-in	10.34 (1500)	121(250)	Coolable Geometry	414(60)
Boron Injection	10.34 (1500)	121(250)	Coolable Geometry	414(60)

Table 15.5-2. ATWS Initial Operating Conditions

Parameters	Value
Dome Pressure, MPaG (psig)	7.07(1025)
Natural Circulation Core Flow, Mkg/hr (Mlb/hr)	36.9(81.4)
Vessel Diameter, m (ft)	7.1(23.3)
Numbers of Fuel Bundles	1132
Power, MWt/% NBR	4500/100
Steam/Feed Flow, kg/sec (Mlbm/hr)	2433(19.31)
Feedwater Temperature, °C (°F)	215.6(420)
Nuclear Condition	EOC
Suppression Pool Volume, m^3 (ft^3)	3610 (127,500)
Initial Suppression Pool Temperature, °C (°F)	43.3(110)
SLCS accumulator driven initial flow, m^3/s (gpm)	0.03 (475)

Table 15.5-3
ATWS Equipment Performance Characteristics

Parameters	Value
MSIV Closure Time[SS1], sec	≥3.0
Delay before start of Electro-Hydraulic Rod Insertion[SS2], sec	≤1
Electro-Hydraulic Control Rod Insertion Time[SS3], sec	≤130
Maximum time for start of motion of ARI rods[SS4], sec	15
Maximum time for all ARI rods to be fully inserted[SS5], sec	25
Safety/Relief Valve (SRV) System Capacity, % NBR Steam Flow/No. of Valves[SS6]	≥89.5/18
SRV Setpoint Range[SS7], MPaG (psig)	8.618 to 8.756 (1250-1270)
SRV Opening Time[SS8], sec	<1.7
Pressure Drop Below Setpoint for SRV Closure [nominal/analysis assumption],% nameplate[SS9]	≤96
CRD (high Pressure Make-Up Function) Low Water Level Initiation Setpoint[SS10], cm (in)	1605.0(631.9)
CRD (High Pressure Make-Up Function) Flow Rate[SS11], m^3/sec (gal/min)	0.065(1035)
ATWS Dome Pressure Sensor Time Constant[SS12], sec	≤0.5
ATWS Logic Time Delay[SS13], sec	≤1
Pool Cooling Capacity[SS14], MW	5.985
Low Water Level For Closure of MSIVs[SS15], cm (in)	1605.0(631.9)
Low Steamline Pressure For Closure of MSIVs[SS16], MPaG (psig)	5.412 (785)
Temperature For Automatic Pool Cooling[SS17], ºC (ºF)	48.9 (120)

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

Parameter	Value	Time
Maximum Neutron Flux, %	228	3s
Maximum Vessel Bottom Pressure, MPaG (psig)	9.83 (1426.1)	31s
Maximum Bulk Suppression Pool Temperature, ºC (ºF)	68.8 (155.8)	320s
Associated Containment Pressure, MPaG (psig)	0.197 (28.61)	320s
Peak Cladding Temperature, ºC (ºF)	952.8 (1747.0)	31s

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

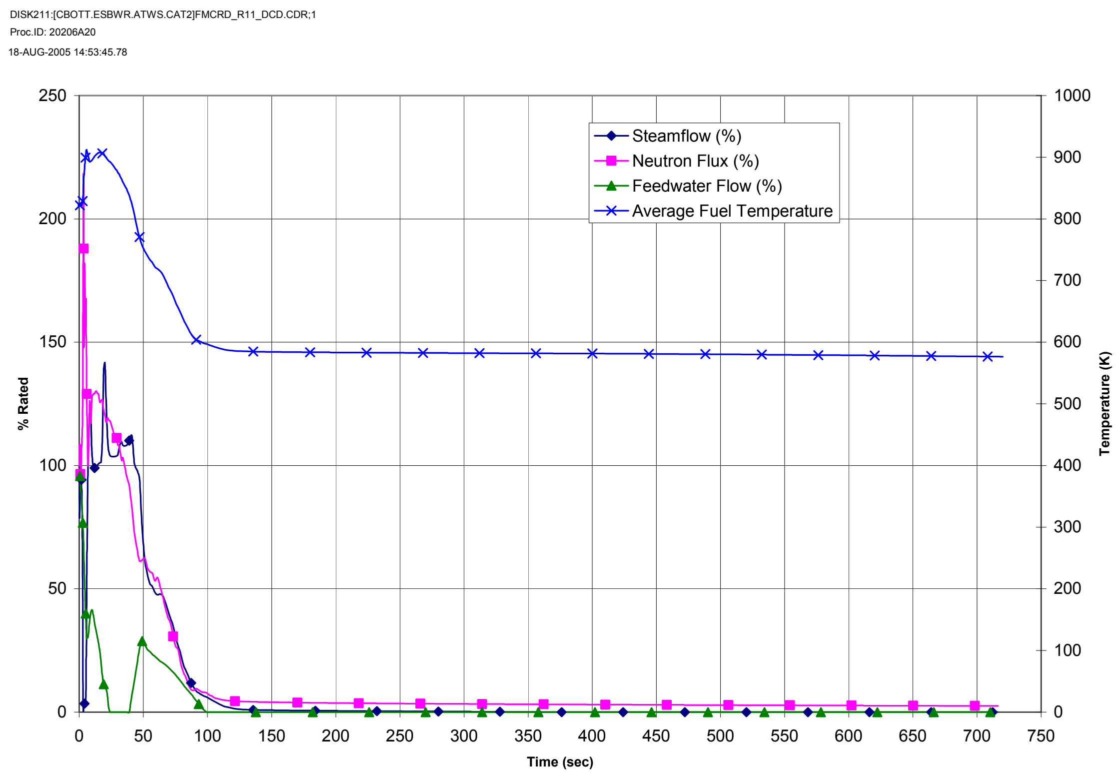

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Figure 15.5-2a. MSIV Closure with FMCRD Run-in (Cont.)

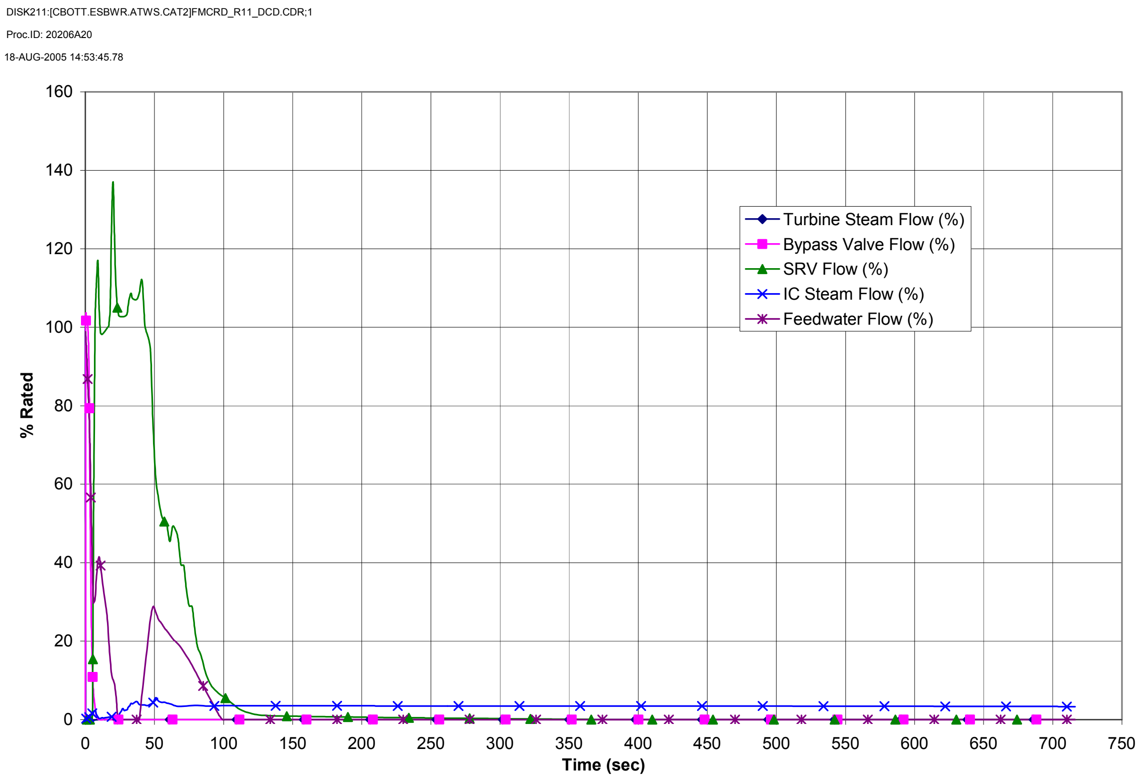

Figure 15.5-2b. MSIV Closure with FMCRD Run-in

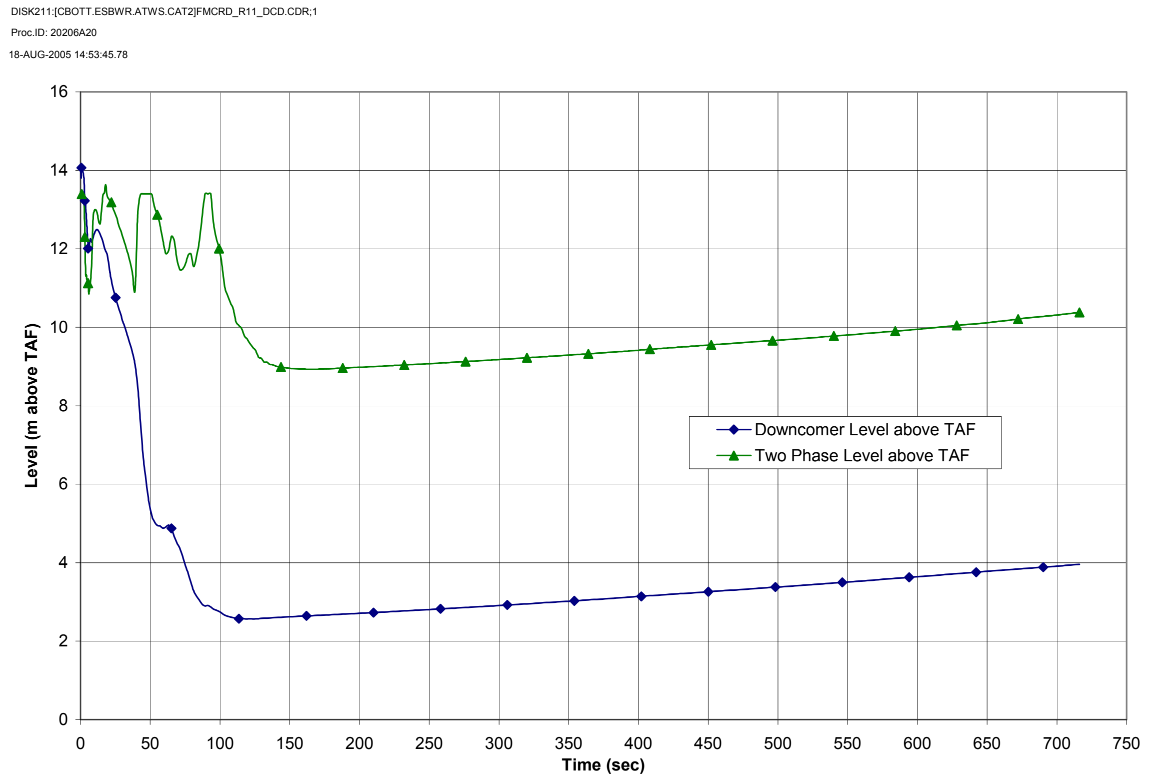

Figure 15.5-2b. MSIV Closure with FMCRD Run-in (Cont.)

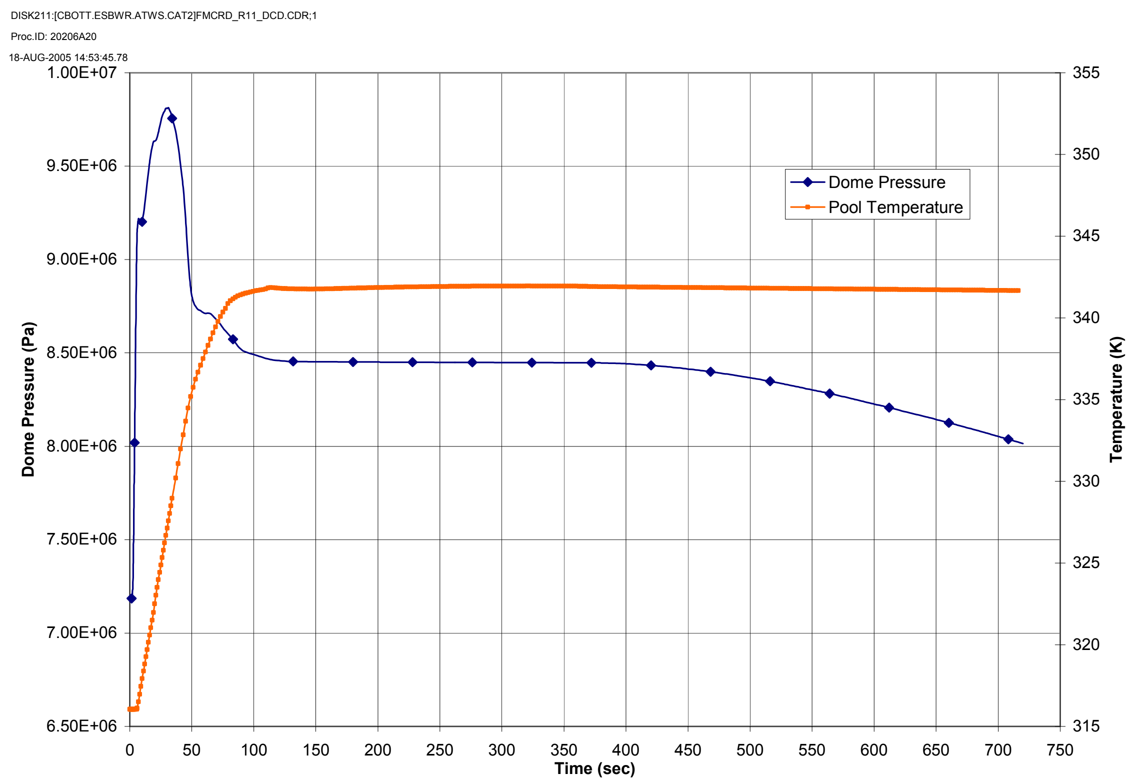

Figure 15.5-2c. MSIV Closure with FMCRD Run-in

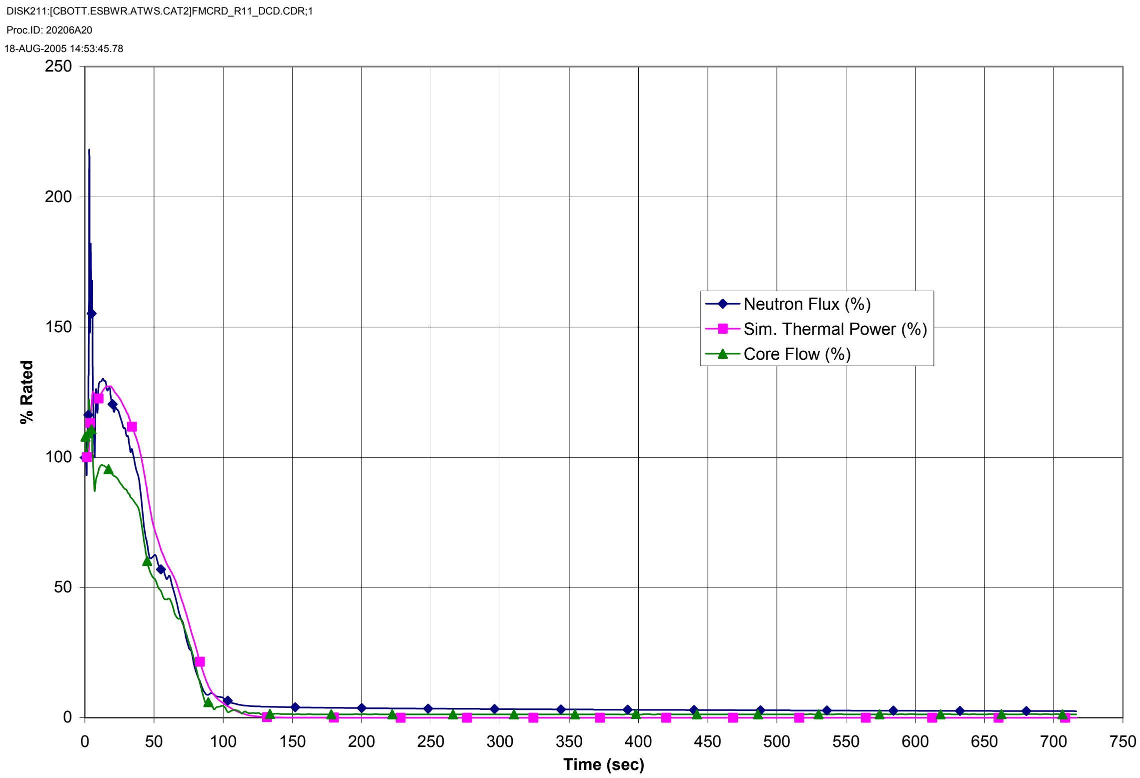

Figure 15.5-2c. MSIV Closure with FMCRD Run-in (Cont.)

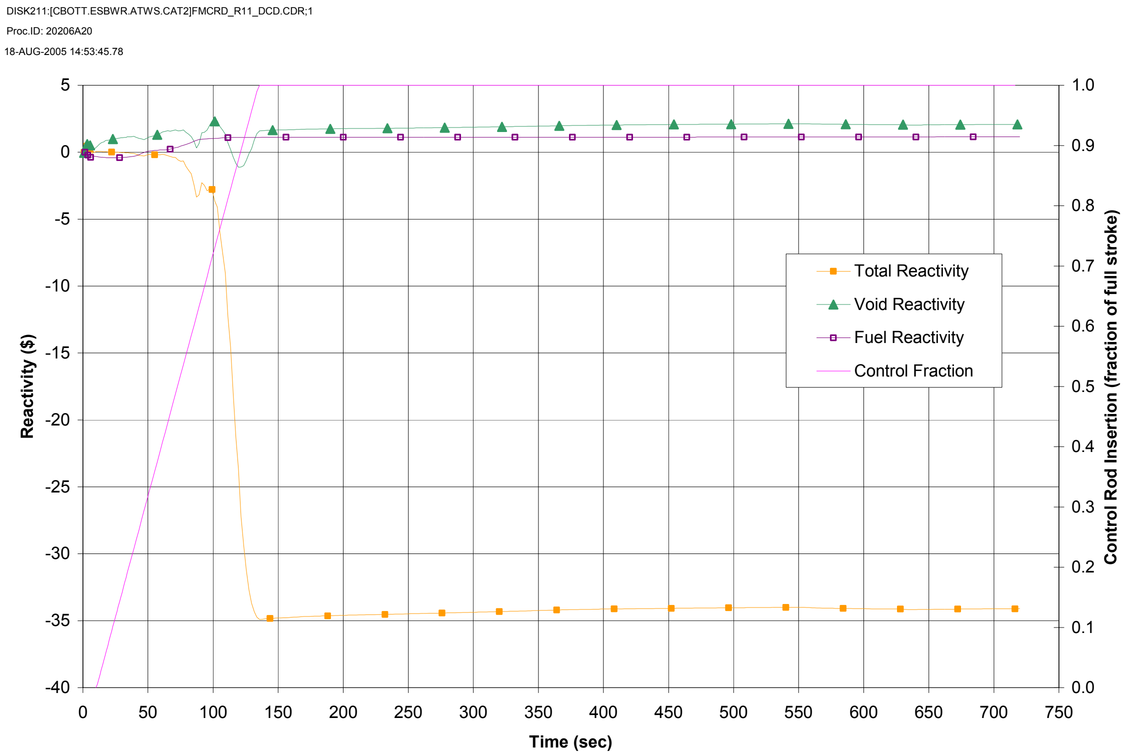

Figure 15.5-2d. MSIV Closure with FMCRD Run-in

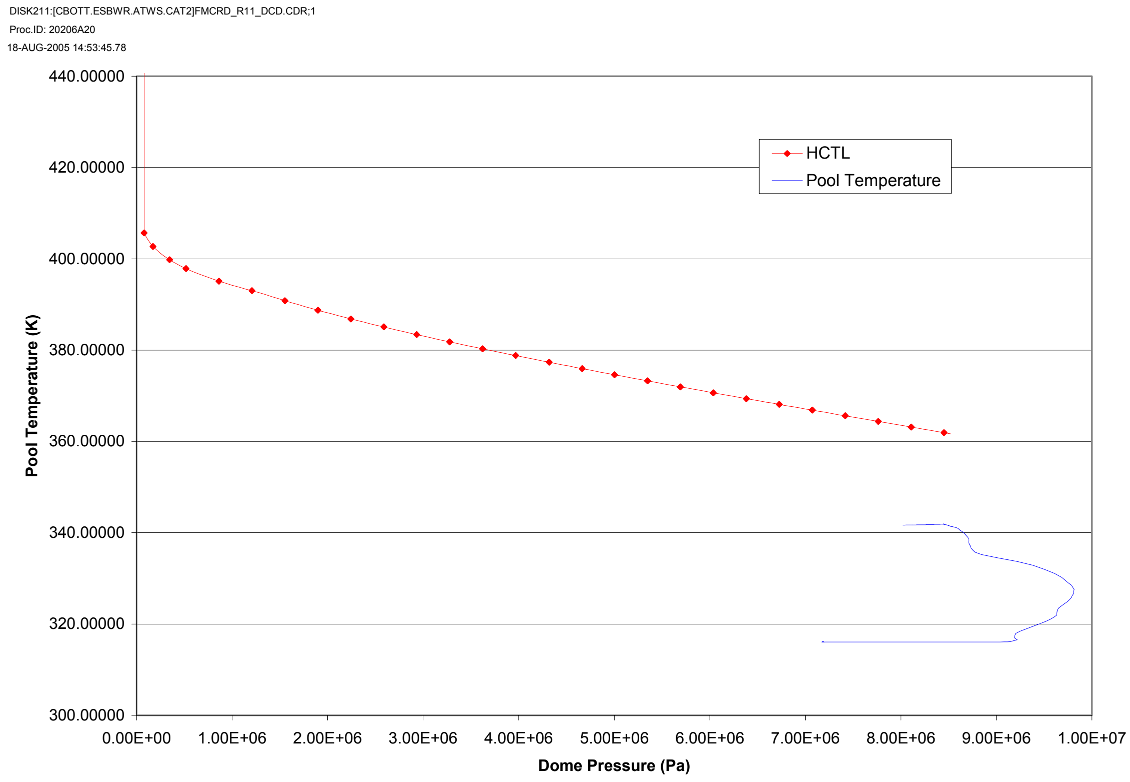

Table 15.5-4c. ATWS MSIV Closure Summary - SLCS Case

Parameter	Value	Time
Maximum Neutron Flux, %	228	3s
Maximum Vessel Bottom Pressure, MPaG (psig)	9.81 (1422)	28s
Maximum Bulk Suppression Pool Temperature, ºC (ºF)	77.6(172)	254s
Associated Containment Pressure, MPaG (psig)	0.218(31.57)	254s
Peak Cladding Temperature, ºC (ºF)	916.2(1681.1)	25s

Figure 15.5-3a. MSIV Closure with Boron Injection

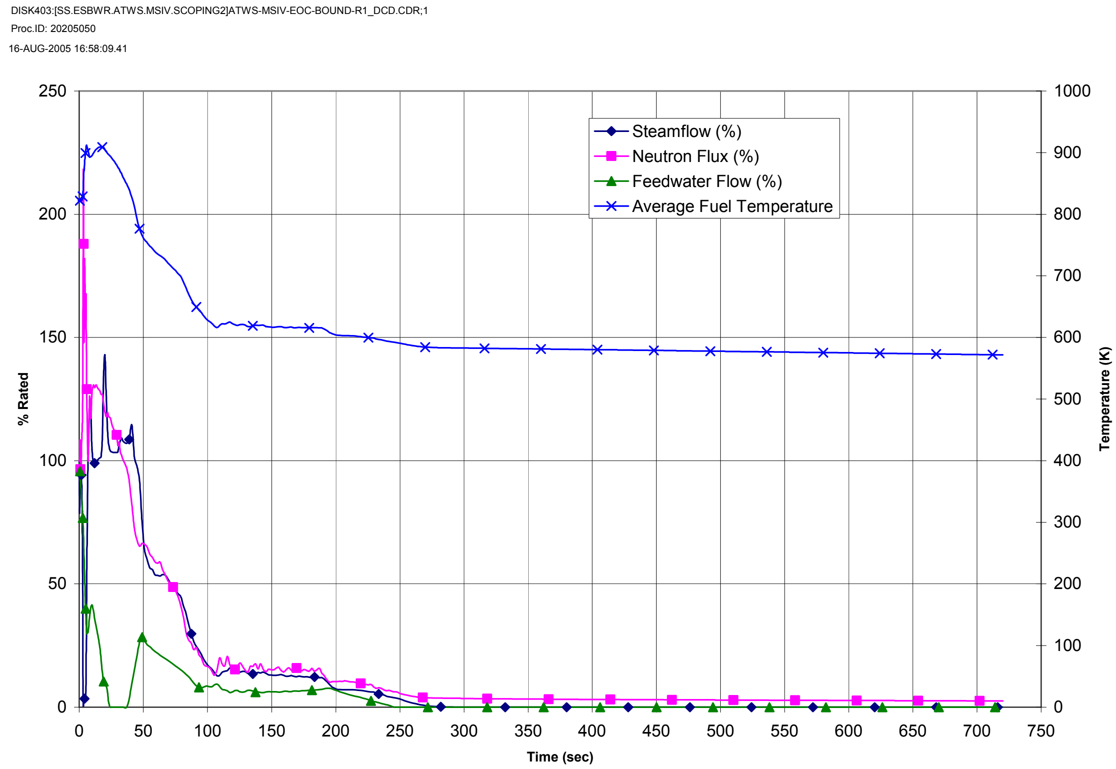

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Figure 15.5-3a. MSIV Closure with Boron Injection (Cont.)

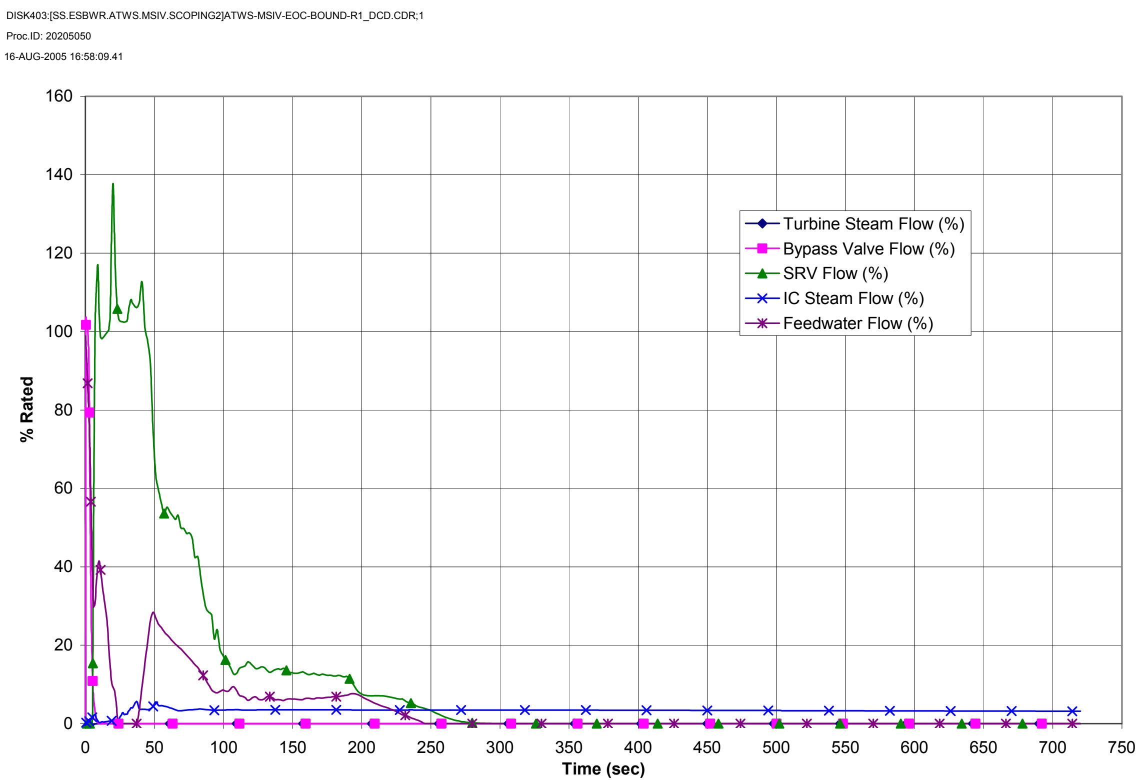

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Figure 15.5-3b. MSIV Closure with Boron Injection

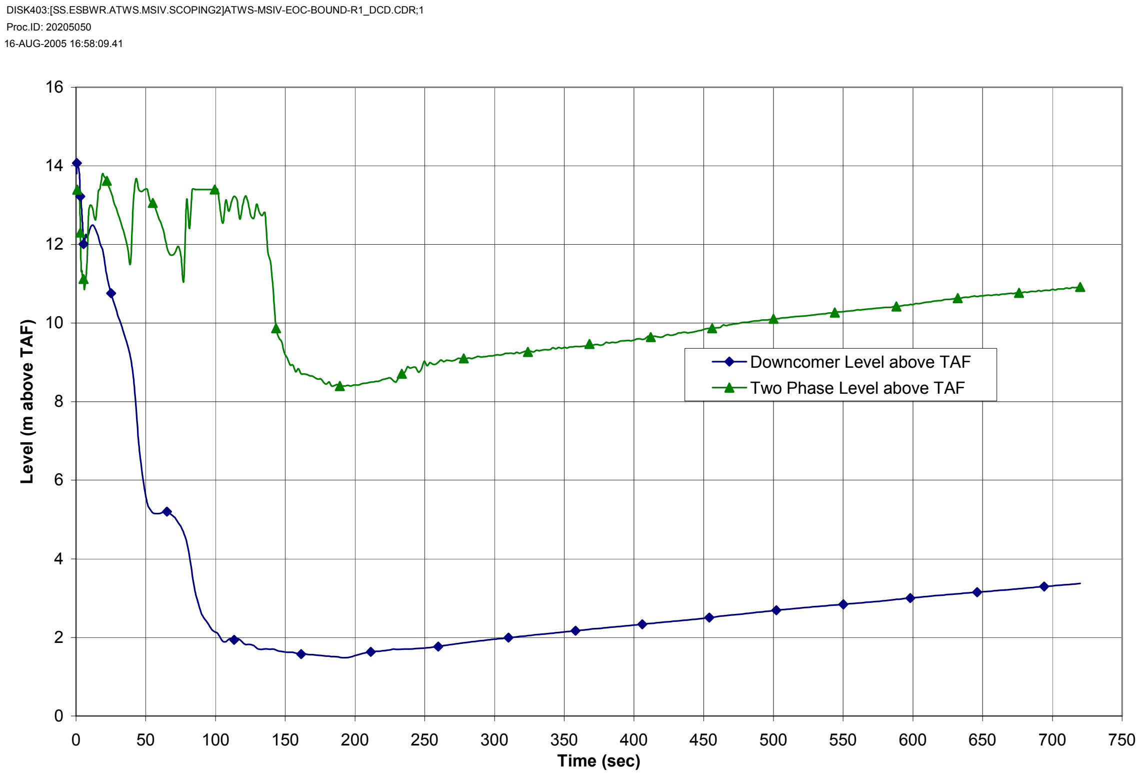

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Figure 15.5-3b. MSIV Closure with Boron Injection (Cont.)

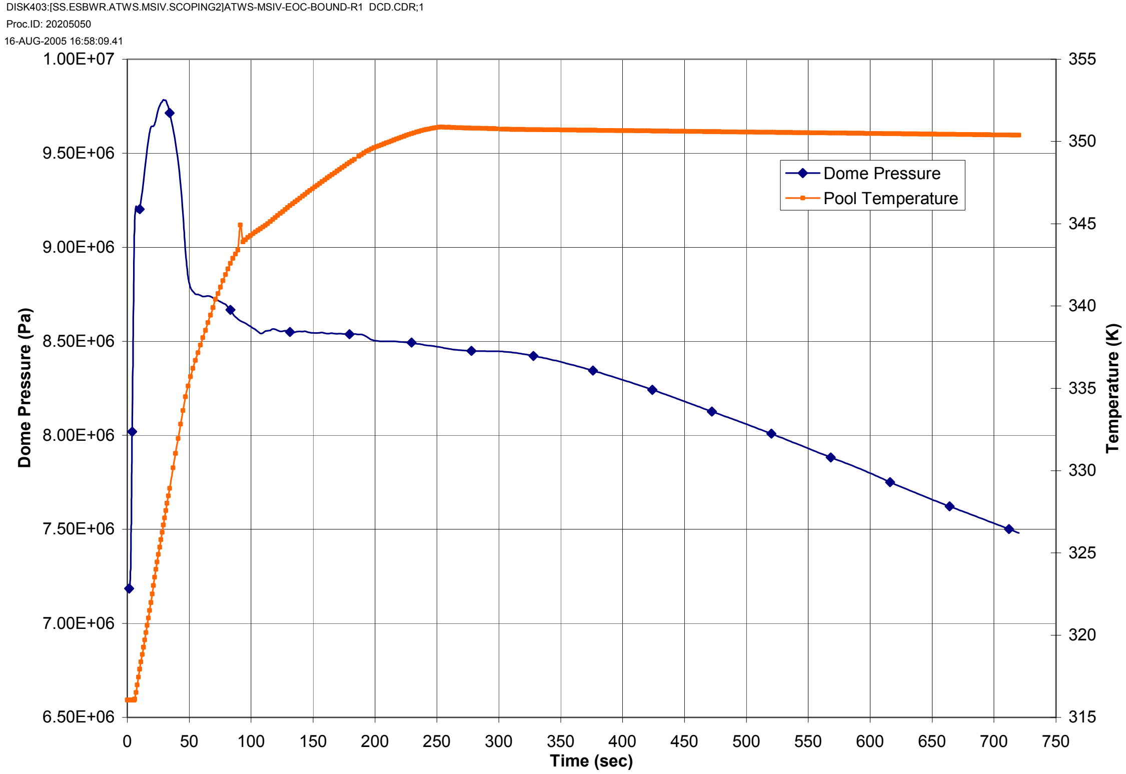

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Figure 15.5-3c. MSIV Closure with Boron Injection

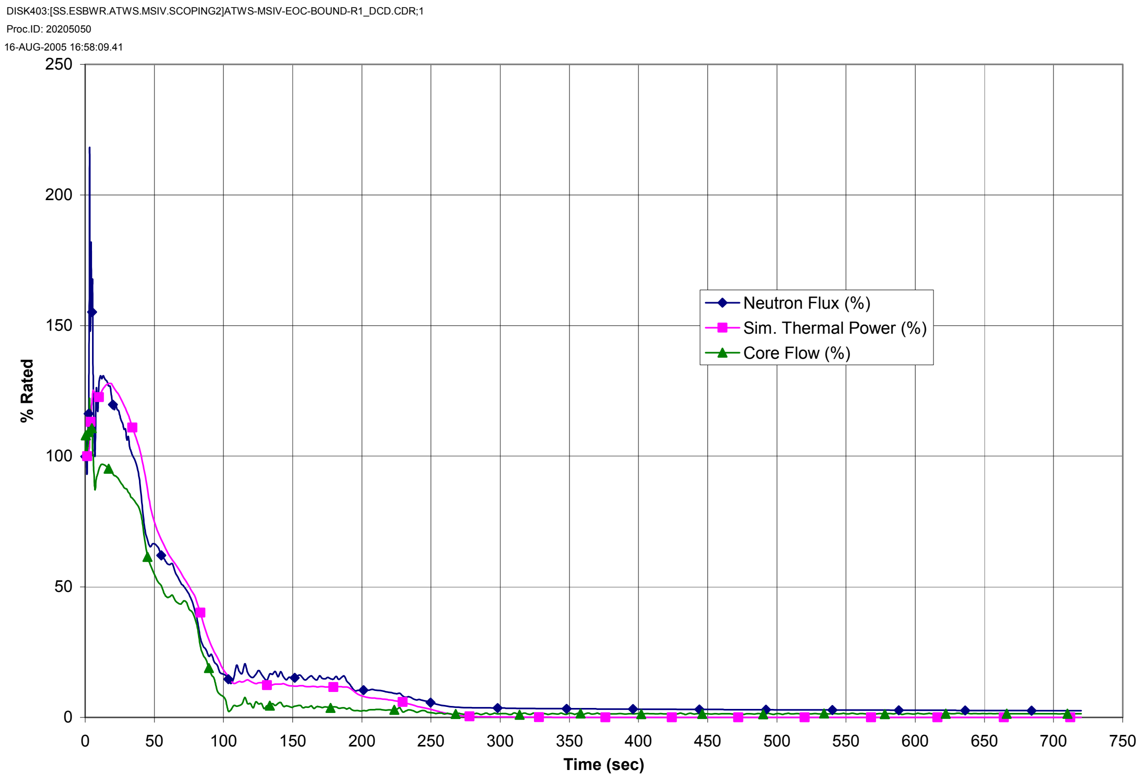

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Figure 15.5-3c. MSIV Closure with Boron Injection (Cont.)

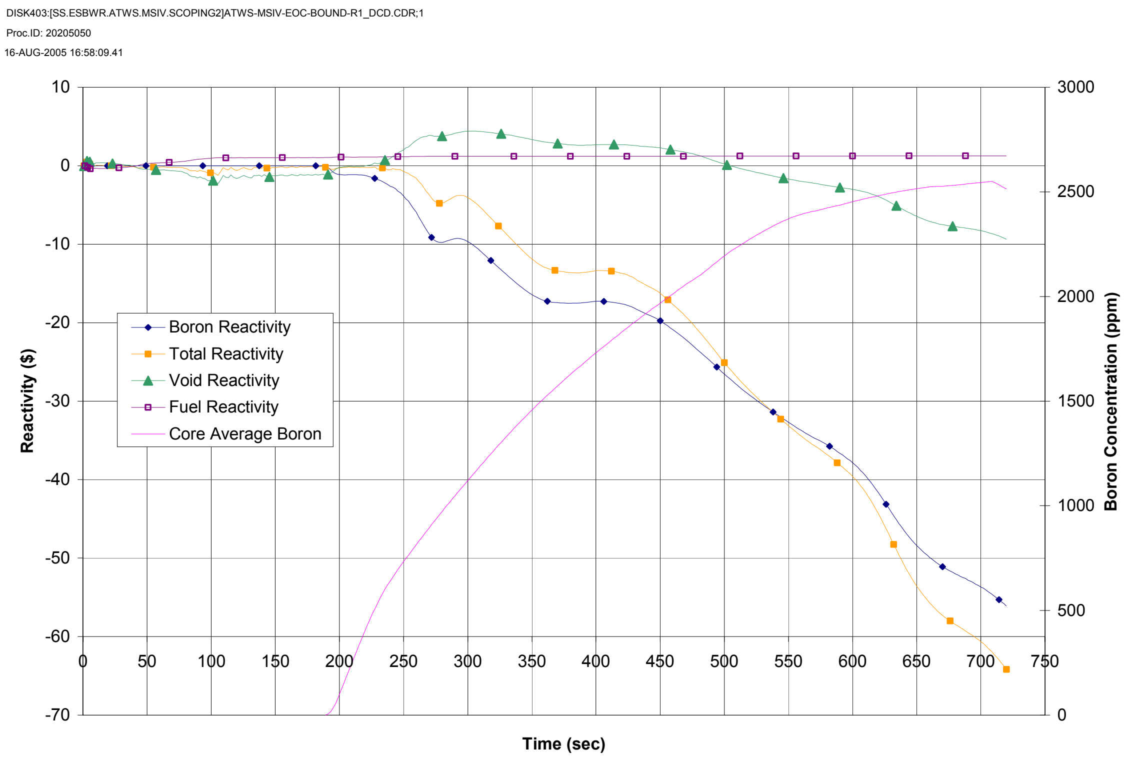


Figure 15.5-3d. MSIV Closure with Boron Injection

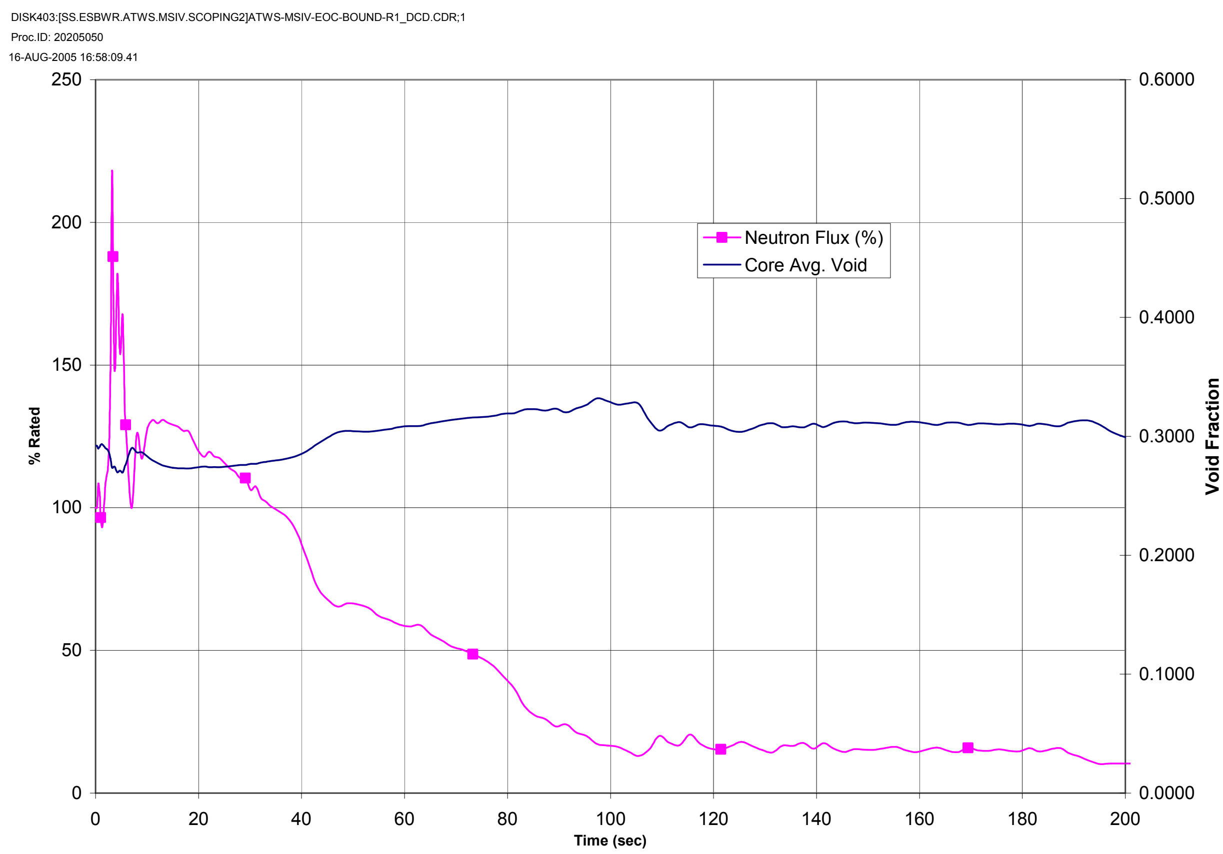
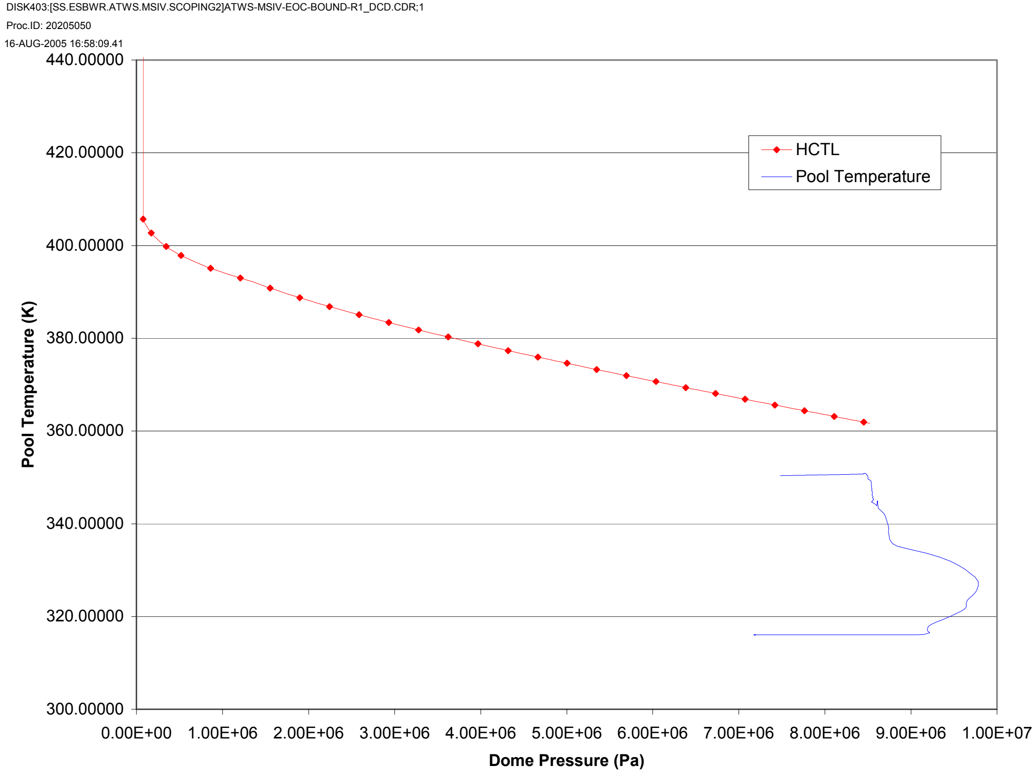


Figure 15.5-3d. MSIV Closure with Boron Injection (Cont.)



ESBWR Abnormal Event Classifications

Event Classification Process

Regulatory criteria for event classification were reviewed to determine the appropriate abnormal event classifications and their associated safety analysis acceptance criteria.

Generated LTR NEDO-33175, based on NRC Documents.

Priority:

1. 10 CFR regulations
2. USNRC Standard Review Plan (SRP) Section 15
3. RG 1.70 Chapter 15
4. NUREG-1503, FSER for the ABWR Design, Chapter 15

SECY-94-084

Event Classification Terms

Final ESBWR Abnormal Event Classifications:

Anticipated Operational Occurrences
(DCD Tier 2 Section 15.2)

Infrequent Events
(DCD Tier 2 Section 15.3)

Accidents
(DCD Tier 2 Section 15.4)

Special Events
(DCD Tier 2 Section 15.5)

Event Classification Terms

Event Definitions

An ***anticipated operational occurrence*** (AOO) is any abnormal event with a probability of occurrence of $\geq 1/100$ per year.

Event Classification Terms

Event Definitions (cont'd)

An **infrequent event** is defined as a DBE (with or without assuming a single active component failure or single operator error) with probability of occurrence of < 1/100 per year, and a radiological consequence less than an accident.

Event Classification Terms

Event Definitions (cont'd)

An **accident** is defined as a postulated DBE that is not expected to occur during the lifetime of a plant, which equates to either an ASME Code Service Level C or D incident, and results in radioactive material releases with calculated doses comparable to (but not to exceed) the 10 CFR 50.34(a) exposures.

Event Classification Terms

Event Definitions (cont'd)

A ***special event*** is not included as a design basis event in 10 CFR 50.49, and

i. is postulated in the 10 CFR regulations to demonstrate some specified prevention, coping or mitigation capabilities, without specifically requiring a radiological evaluation, and/or

ii. includes a common mode equipment failure or additional failure(s) beyond the Single Failure Criterion.

Note: Does not include severe accidents and other events that are only evaluated as part of the plant PRA.

Event Classification

AOOs

Loss of Feedwater Heating

Closure of One Turbine Control Valve

Generator Load Rejection with Turbine Bypass

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

Turbine Trip with Turbine Bypass

Turbine Trip with a Single Failure in the Turbine Bypass System

Closure of One Main Steam Isolation Valve

Closure of All Main Steam Isolation Valves

Event Classification

AOOs (cont'd)

Loss of Condenser Vacuum

Loss of Shutdown Cooling Function of RWCU/SDC

Inadvertent Isolation Condenser Initiation

Runout of One Feedwater Pump

Opening of One Turbine Control or Bypass Valve

Loss of Unit Auxiliary Transformer

Loss of Grid Connection

Loss of All Feedwater Flow

Event Classification

Infrequent Events

Event	Annual Probability
Loss of Feedwater Heating With Failure of Selected Control Rod Run-In	1.5E-4
Feedwater Controller Failure – Maximum Demand	<< 1E-2
Pressure Regulator Failure - Opening of All Turbine Control and Bypass Valves	6.9E-6
Pressure Regulator Failure – Closure of All Turbine Control and Bypass Valves	6.9E-6
Generator Load Rejection with Total Turbine Bypass Failure	<< 4.7E-3
Turbine Trip with Total Turbine Bypass Failure	2.8E-3
Control Rod Withdrawal Error During Refueling	<< 1.0E-5
Control Rod Withdrawal Error During Startup	<< 1.0E-5

Event Classification

Infrequent Events (cont'd)

Event	Annual Probability
Control Rod Withdrawal Error During Power Operation	<< 1.0E-5
Fuel Assembly Loading Error, Mislocated Bundle	<< 1.0E-2
Fuel Assembly Loading Error, Misoriented Bundle	<< 1.0E-2
Inadvertent SDC Function Operation	<< 1.0E-2
Inadvertent Opening of a Safety/Relief Valve	7.5E-3
Inadvertent Opening of a Depressurization Valve	1.0E-2*
Stuck Open Safety/Relief Valve	7E-3
Liquid-Containing Tank Failure (COL applicant scope)	<< 1.0E-2

* Additional review required.

Event Classification

Accidents

Fuel Handling Accident

LOCA Inside Containment

Main Steamline Break Outside Containment

Control Rod Drop Accident

Feedwater Line Break Outside Containment

Failure of Small Line Carrying Primary Coolant Outside Containment

RWCU/SDC System Line Failure Outside Containment

Spent Fuel Cask Drop Accident

Event Classification

Special Events

MSIV Closure With Flux Scram (Overpressure Protection)

Shutdown Without Control Rods(i.e., SLC system shutdown capability)

Shutdown from Outside Main Control Room

Anticipated Transients Without Scram

Station Blackout

Safe Shutdown Fire

Waste Gas System Leak or Failure

NUCLEAR SAFETY OPERATIONAL ANALYSIS (NSOA)

APPROACH

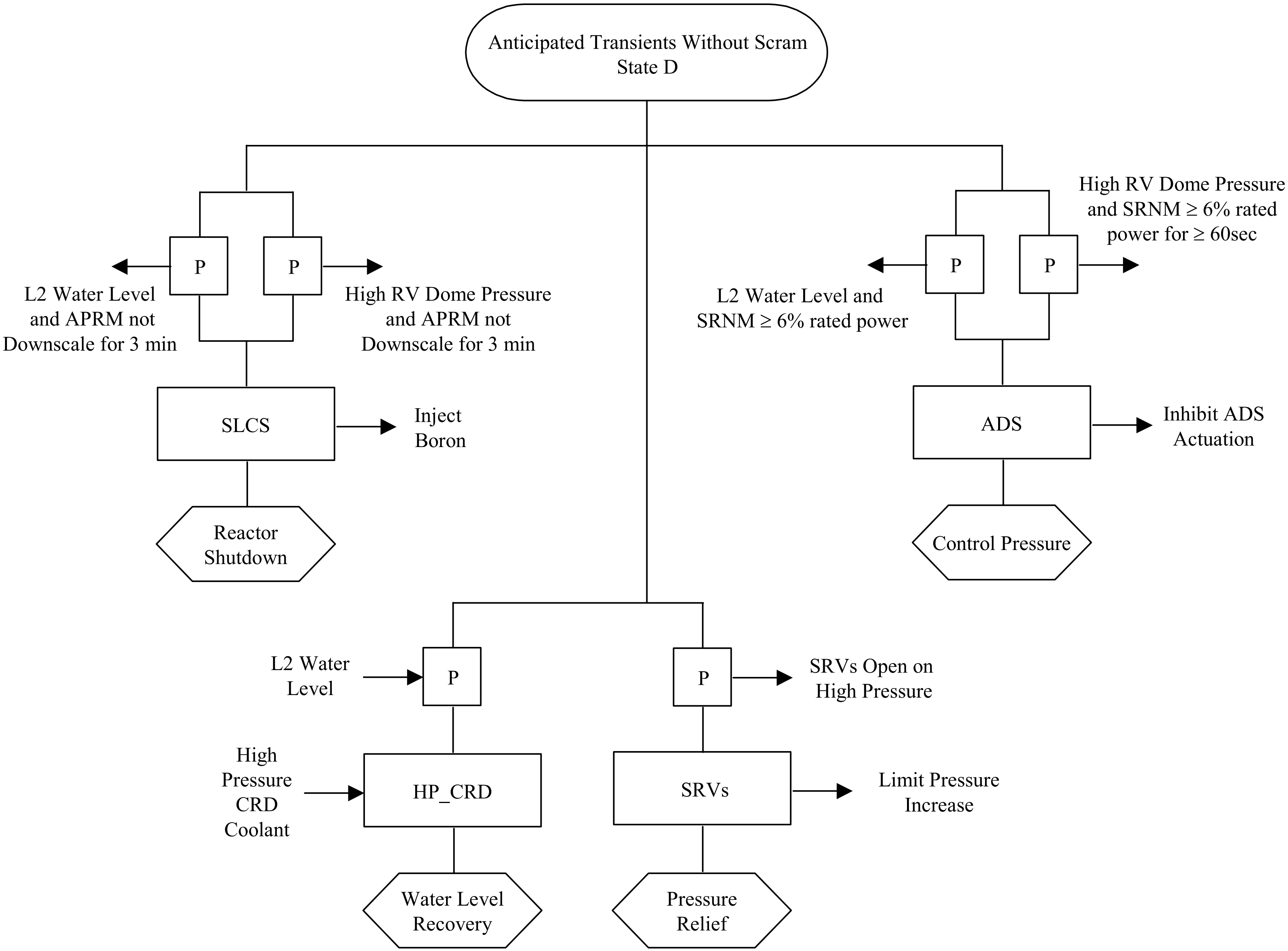
> Address all Chapter 15 Safety Analysis Events

> Identify system level requirements that ensure plant can be brought to a stable safe condition

RESULTS

> Identify systems and actions for each Chapter 15 event
-Event Diagrams
-Summary Matrices

imagination at work



Anticipated Transients Without Scram State D
P
P
L2 Water Level and APRM not Downscale for 3 min
High RV Dome Pressure and APRM not Downscale for 3 min
SLCS
Inject Boron
Reactor Shutdown
High RV Dome Pressure and SRNM ≥ 6% rated power for ≥ 60sec
P
P
L2 Water Level and SRNM ≥ 6% rated power
ADS
Inhibit ADS Actuation
Control Pressure
L2 Water Level
P
High Pressure CRD Coolant
HP_CRD
Water Level Recovery
P
SRVs Open on High Pressure
SRVs
Limit Pressure Increase
Pressure Relief

Parking lot

15. SAFETY ANALYSES
15.0 Analytical Approach
15.0.1 Classification and Selection of Events
15.0.2 Abnormal Events To Be Evaluated
15.0.3 Determination of Safety Analysis Acceptance Criteria
15.0.4 Event Analysis Format
15.0.5 Single Failure Criterion
15.1 Nuclear Safety Operational Analysis
15.1.1 Analytical Approach
15.1.2 Method of Analysis
15.1.3 NSOA Results
15.1.4 Event Evaluations
15.2 Analysis of Anticipated Operational Occurrences
15.2.1 Decrease In Core Coolant Temperature
15.2.1.1 Loss Of Feedwater Heating
15.2.2 Increase In Reactor Pressure
15.2.2.1 Closure of One Turbine Control Valve
15.2.2.2 Generator Load Rejection With Turbine Bypass
15.2.2.3 Generator Load Rejection With a Single Failure in the Turbine Bypass System
15.2.2.4 Turbine Trip With Turbine Bypass
15.2.2.5 Turbine Trip With a Single Failure in the Turbine Bypass System
15.2.2.6 Closure of One Main Steamline Isolation Valve
15.2.2.7 Closure of All Main Steamline Isolation Valves
15.2.2.8 Loss of Condenser Vacuum
15.2.2.9 Loss of Shutdown Cooling Function of RWCU/SDC
15.2.3 Reactivity and Power Distribution Anomalies
15.2.4 Increase in Reactor Coolant Inventory
15.2.4.1 Inadvertent Isolation Condenser Initiation
15.2.4.2 Runout of One Feedwater Pump
15.2.5 Decrease in Reactor Coolant Inventory
15.2.5.1 Opening of One Turbine Control or Bypass Valve.
15.2.5.2 Loss of Non-Emergency AC Power to Station Auxiliaries
15.2.5.3 Loss of All Feedwater Flow
15.3 Analysis Of Infrequent Events
15.3.1 Loss Of Feedwater Heating With Failure of Selected Control Rod Run-In
15.3.2 Feedwater Controller Failure – Maximum Demand
15.3.3 Pressure Regulator Failure – Opening of All Turbine Control and Bypass Valves
15.3.4 Pressure Regulator Failure–Closure of All Turbine Control and Bypass Valves
15.3.5 Generator Load Rejection With Total Turbine Bypass Failure
15.3.6 Turbine Trip With Total Turbine Bypass Failure
15.3.7 Control Rod Withdrawal Error During Refueling
15.3.8 Control Rod Withdrawal Error During Startup
15.3.9 Control Rod Withdrawal Error During Power Operation
15.3.10 Fuel Assembly Loading Error, Mislocated Bundle
15.3.11 Fuel Assembly Loading Error, Misoriented Bundle