GE-Hitachi Nuclear Energy Americas LLC

James C. Kinsey
Vice President, ESBWR Licensing

PO Box 780 M/C A-55
Wilmington, NC 28402-0780
USA

T 910 675 5057
F 910 362 5057
jim.kinsey@ge.com

MFN 07-521, Supplement 1

Docket No. 52-010

October 26, 2007

U.S. Nuclear Regulatory Commission
Document Control Desk
Washington, D.C. 20555-0001

Subject: **Transmittal of Licensing Topical Report (LTR) NEDO-33337, "ESBWR Initial Core Transient Analysis"**

GE-Hitachi Energy North Americas (GEH) has changed the Legal Notice in the subject LTR to be consistent with the latest corporate structure, and hereby re-submits the LTR with the changed Legal Notice in Enclosure 1.

If you have any questions or require additional information regarding the information provided here, please contact me.

Sincerely,

Kathy Sedney for

James C. Kinsey
Vice President, ESBWR Licensing

DO68
NRO

Enclosure:

1. Licensing Topical Report NEDO-33337, "ESBWR Initial Core Transient Analysis"

cc:	AE Cubbage	USNRC (with enclosures)
	GB Stramback	GEH /San Jose (with enclosures)
	RE Brown	GEH /Wilmington (with enclosures)
	eDRF	0074-9757

GE-Hitachi Nuclear Energy Americas, LLC
3901 Castle Hayne Road
Wilmington, NC 28401

NEDO-33337
Revision 0
Class I
eDRFSection-0000-0074-9757-00
October 2007

Licensing Topical Report

ESBWR INITIAL CORE TRANSIENT ANALYSES

Author

Craig Goodson

ELECTRONIC COPY
Electronic approvals filed in
eDRFSection-0000-0074-9757-00.

Legal Notice

Acknowledgments

This document is the result of the technical contributions from many individuals and organizations in addition to the author of this document. Significant contributions to the development of this document have been made by the following individuals:

MD Alamgir

Antonio Barrett

Mike Cook

Javier Haces (ENUSA)

Shih-Ping Kao

Wayne Marquino

Jaime Morales

Kellie Norton

Khalid Qureshi

Pradip Saha

Marina Trueba (ENUSA)

Juswald Vedovi

Jun Yang

Yingzhen Yang

Global Abbreviations and Acronyms List

Term	Definition
10 CFR	Title 10, Code of Federal Regulations
ABWR	Advanced Boiling Water Reactor
AOO	Anticipated Operational Occurrence
APRM	Average Power Range Monitor
ARI	Alternate Rod Insertion
ASME	American Society of Mechanical Engineers
ATWS	Anticipated Transients Without Scram
BOC	Beginning of Cycle
BPV	Bypass Valve
BWR	Boiling Water Reactor
CRD	Control Rod Drive
CSAU	Code Scaling Applicability and Uncertainty
DCIS	Distributed Control and Information System
ECCS	Emergency Core Cooling System
EHC	Electro Hydraulic Control
EOC	End of Cycle
ESF	Engineered Safety Feature
FAPCS	Fuel and Auxiliary Pools Cooling System
FWCS	Feedwater Control System
FWRB	Feedwater Runback
GDC	General Design Criteria
GDCS	Gravity-Driven Cooling System
GEH	GE – Hitachi Nuclear Energy
HCTL	Heat Capacity Temperature Limit
HCU	Hydraulic Control Unit
LCV	Loss Condenser Vacuum
LOCA	Loss-of-Coolant-Accident
LOFWH	Loss of Feedwater Heater
LPRM	Local Power Range Monitor
MCPR	Minimum Critical Power Ratio
MSIV	Main Steam Isolation Valve
MSIVF	Main Steam Isolation Closure with Flux Scram
NBR	Nuclear Boiler Rated
NMS	Neutron Monitoring System
NSOA	Nuclear Safety Operations Analysis
OSUTL	One Sided Upper Tolerance Limit
PCCS	Passive Containment Cooling System
RC&IS	Rod Control and Information System
RCPB	Reactor Coolant Pressure Boundary

Term	**Definition**
RPS	Reactor Protection System
RPV	Reactor Pressure Vessel
RWCU/SDC	Reactor Water Cleanup/Shutdown Cooling
SB&PC	Steam Bypass and Pressure Control System
SBO	Station Blackout
SBWR	Simplified Boiling Water Reactor
SCRRI	Selected Control Rod Run-in
SLC	Standby Liquid Control
SRI	Select Rod Insert
SRV	Safety Relief Valve
STPT	Simulated Thermal Power Trip
TAF	Top of Active Fuel
TCV	Turbine Control Valve
TPST	Thermal Power Simulated Trip
TSV	Turbine Stop Valve
USNRC	United States Nuclear Regulatory Commission

Contents

List of Tables

List of Tables

List of Figures

List of Figures

List of Figures

List of Figures

List of Figures

INTENTIONALLY BLANK

1.0 INTRODUCTION

1.1 BACKGROUND

Reference 1.1-1 documents the ESBWR Design Control Document (DCD), Tier 2. The DCD describes the design of the ESBWR plant and includes safety analyses performed to validate the ESBWR design. Steady state and various transient events, i.e., AOOs, Infrequent Events, Special Events, are evaluated against appropriate acceptance criteria described in the DCD. These evaluations were performed with the equilibrium core. Evaluations with the initial core are documented here and are incorporated into the DCD by reference. The initial core loading used in these analyses is documented in Reference 1.1-2.

1.2 EVALUATED EVENTS

The classification of the events for ESBWR, the abnormal event that are evaluated, and the determination of the safety analysis acceptance criteria are outlined in the ESBWR DCD (Reference 1.1-1), and are not repeated here.

The power/temperature statepoint analyzed in this report is 100% power and 420°F feedwater temperature. This condition is noted as SP0. Other conditions are analyzed and are documented in Reference 1.1-3 assuming 100% power and lower feedwater temperature (SP1) and lower power and higher feedwater temperature (SP2).

The selection of analyses reanalyzed for different core design/core loading patterns of the same fuel design is consistent with those stated in the DCD. All analyses identified in the DCD are provided. A listing of the cases analyzed with the initial core is provided herein.

- Steady state reactor heat balance – Similar to the DCD, the steady-state conditions are evaluated at normal operating conditions.
- Stability Analysis
  - Steady State – analysis is performed for channel, core, and regional stability at rated conditions.
  - AOO – The AOO stability for the initial core at nominal conditions is not provided because it is bounded by the stability analysis documented in Reference 1.1-3 that assumes lower feedwater temperature at 100% power.
  - ATWS – The ATWS stability for the initial core at nominal conditions is not provided because it is bounded by the stability analysis documented in Reference 1.1-3 that assumes lower feedwater temperature at 100% power.
  - Start up analysis – the stability performance during startup is analyzed with the initial core.
- Analysis of Anticipated Operational Occurrences (AOOs) – All of the AOOs that are documented in the DCD are repeated with the initial core.
- Analysis of Infrequent Events (IEs) – All of the IEs that are documented in the DCD are repeated with the initial core.
- Analyses of limiting Special Events – three sets of special events are analyzed with the initial core: Station Blackout (SBO), Main Steam Isolation Closure with Flux Scram (MSIVF), and ATWS.
  - The SBO event bounds AOOs with respect to reactor vessel coolant inventory.

  - The MSIVF transient is analyzed to assess the overpressure protection capability.
  - ATWS – The two limiting ATWS cases are analyzed: the bounding MSIV closure and the bounding Loss of Condenser Vacuum (LCV) events.
- Analysis of Design-Basis Accidents – The Loss-Of-Coolant Accident (LOCA) is not reevaluated with the initial core. The analysis documented in the DCD is applicable to both equilibrium and initial cores because:
  - There is no difference in power;
  - There is no difference in pressure; and
  - No difference in temperature.
  - The core neutronic feedback is not important because the reactor is immediately scrammed during a LOCA event.
  - The fuel bundle hydraulic design is unchanged in the initial core.

  Therefore, the LOCA TRACG analysis is not performed with the initial core. LOCA results are in the DCD (Reference 1.1-1). The offsite dose analysis is not performed for the initial core because the source term presented in DCD, Tier 2, Appendix 15B is appropriate for use for both the initial core and the equilibrium core for the ESBWR.

The computer codes used in each event analysis are listed in Table 1.2-1.

Table 1.2-1

ESBWR Safety Analysis Codes

Safety Analysis	Analysis Code
Steady-state evaluation	TRACG04[1]
Stability Evaluation	TRACG04[1]
Startup Analysis	TRACG04[2]
Anticipated Operational Occurrences	
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]
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]
Infrequent Events	
Loss of Feedwater Heating With Failure of Selected Control Rod Run-In	TRACG04[1]
Feedwater Controller Failure – Maximum 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]
Inadvertent Opening of a Safety/Relief Valve	TRACG04[1]
Stuck Open Safety/Relief Valve	TRACG04[1]
Special Events	

Table 1.2-1

ESBWR Safety Analysis Codes

Safety Analysis	Analysis Code
MSIV Closure With Flux Scram (Overpressure Protection)	TRACG04[1]
Station Blackout	TRACG04[1]
Anticipated Transients Without Scram	TRACG04[2]

1. TRACG04A (Alpha VMS version) is used.
2. TRACG04P (PC version) is used.

1.3 SCOPE AND OUTLINE OF THE REPORT

The purpose of this report is to show that the ESBWR initial core design can be safely operated by meeting the relevant acceptance criteria. This report presents the results of the TRACG analyses for the ESBWR limiting events with initial core and compares the results with the equilibrium core documented in the DCD, Tier 2. The analyses are provided in Section 2, the comparison is provided in Section 3.1, and the conclusion is provided in Section 3.2.

1.4 REFERENCES

1.1-1 GE-Hitachi Nuclear Energy, "ESBWR Design Control Document" 26A6642, Revision 4, September 2007.

1.1-2 Global Nuclear Fuel, "ESBWR Initial Core Nuclear Design Report", NEDC-33326-P, Class III (Proprietary), Revision 0, July 2007, NEDO-33326, Class I (Non-Proprietary), Revision 0, July 2007.

1.1-3 GE-Hitachi Nuclear Energy, "ESBWR Feedwater Temperature Operating Domain Transient and Accident Analysis", NEDO-33338, Class I, Revision 0, Scheduled October 2007.

INTENTIONALLY BLANK

2.0 ANALYSES OF SP0 CONDITION

2.1 STEADY STATE REACTOR HEAT BALANCE AND RESULTS

The axial distribution of void fractions for an average channel and a hot channel, as predicted by TRACG, are given in Table 2.1-1a and Table 2.1-1b. The average channel and hot channel exit values are also provided. Similar distributions for steam quality are given in Table 2.1-2a and Table 2.1-2b. The axial power distribution is given in Table 2.1-3a and Table 2.1-3b. The axial void and power distributions as predicted by the core simulator (PANAC), for the channel with the highest exit void fraction for the core reference loading pattern are given in Table 2.1-4.

The expected operating void fraction for the ESBWR is within the qualification basis of the void fraction methods. The void fractions in Table 2.1-1a and 2.1-1b are based on TRACG. The hot channel is in Table 2.1-1b. This hot channel has a maximum void fraction of 0.90. The void fraction qualification database (References 2.1-1 and 2.1-2) contains void fractions in excess of 0.92 and covers the void fraction range expected for normal steady-state operation as well as AOOs. The core simulator maximum exit void fraction, for the steady-state simulation is 0.89 as shown in Table 2.1-4.

During steady-state operations at 100% power, the bypass voiding at the level-D LPRM is less than 5% for the initial core.

The detemination of OLMCPR includes the consideration of uncertainty in the void fraction.

2.1.1 References

2.1-1 GE Nuclear Energy, "Licensing Topical Report TRACG Qualification," NEDE-32177P Revision 2, Class III (proprietary), January 2000.

2.1-2 "TASC-03A, A Computer Program for Transient Analysis of a Single Channel", NEDC-32084P-A, Revision 2, Class III (proprietary), July 2002.

Table 2.1-1a

Void Distribution for MOC Analyzed Core - TRACG Average Channel

Channel Power = 4.534 MW, CPR = 1.65 Active Fuel Length = 3.048 m / 120.00 inches			
Node (m above BAF)	**Average Node Value**	**Node (m above BAF)**	**Average Node Value**
1 (BAF+0.02)	0.00	17 (BAF+0.69)	0.48
2 (BAF+0.06)	0.00	18 (BAF+0.84)	0.58
3 (BAF+0.10)	0.00	19 (BAF+0.99)	0.65
4 (BAF+0.13)	0.00	20 (BAF+1.14)	0.70
5 (BAF+0.17)	0.01	21 (BAF+1.30)	0.73
6 (BAF+0.21)	0.03	22 (BAF+1.45)	0.74
7 (BAF+0.25)	0.05	23 (BAF+1.60)	0.74
8 (BAF+0.29)	0.07	24 (BAF+1.75)	0.76
9 (BAF+0.32)	0.11	25 (BAF+1.91)	0.78
10 (BAF+0.36)	0.14	26 (BAF+2.06)	0.80
11 (BAF+0.40)	0.18	27 (BAF+2.21)	0.81
12 (BAF+0.44)	0.21	28 (BAF+2.36)	0.83
13 (BAF+0.48)	0.25	29 (BAF+2.51)	0.84
14 (BAF+0.51)	0.28	30 (BAF+2.67)	0.84
15 (BAF+0.55)	0.32	31 (BAF+2.82)	0.85
16 (BAF+0.59)	0.35	32 (BAF+2.97)	0.85

Table 2.1-1b

Void Distribution for MOC Analyzed Core - TRACG Hot Channel

Channel Power = 5.276 MW, CPR = 1.41 Active Fuel Length = 3.048 m / 120.00 inches			
Node (m above BAF)	**Average Node Value**	**Node (m above BAF)**	**Average Node Value**
1 (BAF+0.02)	0.00	17 (BAF+0.69)	0.59
2 (BAF+0.06)	0.00	18 (BAF+0.84)	0.68
3 (BAF+0.10)	0.00	19 (BAF+0.99)	0.72
4 (BAF+0.13)	0.00	20 (BAF+1.14)	0.74
5 (BAF+0.17)	0.03	21 (BAF+1.30)	0.75
6 (BAF+0.21)	0.06	22 (BAF+1.45)	0.78
7 (BAF+0.25)	0.09	23 (BAF+1.60)	0.81
8 (BAF+0.29)	0.12	24 (BAF+1.75)	0.83
9 (BAF+0.32)	0.17	25 (BAF+1.91)	0.84
10 (BAF+0.36)	0.21	26 (BAF+2.06)	0.86
11 (BAF+0.40)	0.25	27 (BAF+2.21)	0.87
12 (BAF+0.44)	0.29	28 (BAF+2.36)	0.88
13 (BAF+0.48)	0.33	29 (BAF+2.51)	0.89
14 (BAF+0.51)	0.37	30 (BAF+2.67)	0.90
15 (BAF+0.55)	0.41	31 (BAF+2.82)	0.90
16 (BAF+0.59)	0.45	32 (BAF+2.97)	0.90

Table 2.1-2a

Flow Quality Distribution for MOC Analyzed Core - TRACG Average Channel

Channel Power = 4.534 MW, CPR = 1.65 Active Fuel Length = 3.048 m / 120.00 inches			
Node (m above BAF)	**Average Node Value**	**Node (m above BAF)**	**Average Node Value**
1 (BAF+0.02)	0.00	17 (BAF+0.69)	0.06
2 (BAF+0.06)	0.00	18 (BAF+0.84)	0.09
3 (BAF+0.10)	0.00	19 (BAF+0.99)	0.12
4 (BAF+0.13)	0.00	20 (BAF+1.14)	0.14
5 (BAF+0.17)	0.00	21 (BAF+1.30)	0.17
6 (BAF+0.21)	0.00	22 (BAF+1.45)	0.19
7 (BAF+0.25)	0.00	23 (BAF+1.60)	0.21
8 (BAF+0.29)	0.00	24 (BAF+1.75)	0.23
9 (BAF+0.32)	0.01	25 (BAF+1.91)	0.25
10 (BAF+0.36)	0.01	26 (BAF+2.06)	0.27
11 (BAF+0.40)	0.01	27 (BAF+2.21)	0.28
12 (BAF+0.44)	0.02	28 (BAF+2.36)	0.29
13 (BAF+0.48)	0.02	29 (BAF+2.51)	0.31
14 (BAF+0.51)	0.03	30 (BAF+2.67)	0.32
15 (BAF+0.55)	0.03	31 (BAF+2.82)	0.32
16 (BAF+0.59)	0.04	32 (BAF+2.97)	0.33

Table 2.1-2b

Flow Quality Distribution for MOC Analyzed Core - Hot Channel

Channel Power = 5.276 MW, CPR = 1.41 Active Fuel Length = 3.048 m / 120.00 inches			
Node (m above BAF)	**Average Node Value**	**Node (m above BAF)**	**Average Node Value**
1 (BAF+0.02)	0.00	17 (BAF+0.69)	0.09
2 (BAF+0.06)	0.00	18 (BAF+0.84)	0.13
3 (BAF+0.10)	0.00	19 (BAF+0.99)	0.16
4 (BAF+0.13)	0.00	20 (BAF+1.14)	0.20
5 (BAF+0.17)	0.00	21 (BAF+1.30)	0.23
6 (BAF+0.21)	0.00	22 (BAF+1.45)	0.25
7 (BAF+0.25)	0.00	23 (BAF+1.60)	0.28
8 (BAF+0.29)	0.01	24 (BAF+1.75)	0.30
9 (BAF+0.32)	0.01	25 (BAF+1.91)	0.33
10 (BAF+0.36)	0.02	26 (BAF+2.06)	0.35
11 (BAF+0.40)	0.02	27 (BAF+2.21)	0.36
12 (BAF+0.44)	0.03	28 (BAF+2.36)	0.38
13 (BAF+0.48)	0.03	29 (BAF+2.51)	0.39
14 (BAF+0.51)	0.04	30 (BAF+2.67)	0.41
15 (BAF+0.55)	0.05	31 (BAF+2.82)	0.41
16 (BAF+0.59)	0.06	32 (BAF+2.97)	0.42

Table 2.1-3a

Axial Power Distribution Used to Generate Void and Quality for MOC Analyzed Core

- TRACG Average Channel

Channel Power = 4.534 MW, CPR = 1.65 Active Fuel Length = 3.048 m / 120.00 inches			
Node (m above BAF)	**Average Node Value***	**Node (m above BAF)**	**Average Node Value**
1 (BAF+0.02)	0.10 (0.41)	17 (BAF+0.69)	1.59
2 (BAF+0.06)	0.10	18 (BAF+0.84)	1.52
3 (BAF+0.10)	0.10	19 (BAF+0.99)	1.42
4 (BAF+0.13)	0.10	20 (BAF+1.14)	1.33
5 (BAF+0.17)	0.29 (1.17)	21 (BAF+1.30)	1.24
6 (BAF+0.21)	0.29	22 (BAF+1.45)	1.16
7 (BAF+0.25)	0.29	23 (BAF+1.60)	1.08
8 (BAF+0.29)	0.29	24 (BAF+1.75)	1.02
9 (BAF+0.32)	0.38 (1.52)	25 (BAF+1.91)	0.93
10 (BAF+0.36)	0.38	26 (BAF+2.06)	0.87
11 (BAF+0.40)	0.38	27 (BAF+2.21)	0.82
12 (BAF+0.44)	0.38	28 (BAF+2.36)	0.73
13 (BAF+0.48)	0.41 (1.65)	29 (BAF+2.51)	0.61
14 (BAF+0.51)	0.41	30 (BAF+2.67)	0.48
15 (BAF+0.55)	0.41	31 (BAF+2.82)	0.34
16 (BAF+0.59)	0.41	32 (BAF+2.97)	0.16

NOTE: Normalized to 20.
* In parenthesis the axial power factor considering the four short nodes.

Table 2.1-3b

Axial Power Distribution Used to Generate Void and Quality for MOC Analyzed Core

- TRACG Hot Channel

Channel Power = 5.276 MW, CPR = 1.41 Active Fuel Length = 3.048 m / 120.00 inches			
Node (m above BAF)	**Average Node Value***	**Node (m above BAF)**	**Average Node Value**
1 (BAF+0.02)	0.10 (0.41)	17 (BAF+0.69)	1.59
2 (BAF+0.06)	0.10	18 (BAF+0.84)	1.52
3 (BAF+0.10)	0.10	19 (BAF+0.99)	1.43
4 (BAF+0.13)	0.10	20 (BAF+1.14)	1.35
5 (BAF+0.17)	0.30 (1.21)	21 (BAF+1.30)	1.26
6 (BAF+0.21)	0.30	22 (BAF+1.45)	1.18
7 (BAF+0.25)	0.30	23 (BAF+1.60)	1.10
8 (BAF+0.29)	0.30	24 (BAF+1.75)	1.02
9 (BAF+0.32)	0.39 (1.56)	25 (BAF+1.91)	0.89
10 (BAF+0.36)	0.39	26 (BAF+2.06)	0.82
11 (BAF+0.40)	0.39	27 (BAF+2.21)	0.77
12 (BAF+0.44)	0.39	28 (BAF+2.36)	0.69
13 (BAF+0.48)	0.41 (1.65)	29 (BAF+2.51)	0.59
14 (BAF+0.51)	0.41	30 (BAF+2.67)	0.47
15 (BAF+0.55)	0.41	31 (BAF+2.82)	0.34
16 (BAF+0.59)	0.41	32 (BAF+2.97)	0.15

NOTE: Normalized to 20.

* In parenthesis the axial power factor considering the four short nodes.

Table 2.1-4

Axial Distribution for MOC Core – Core Simulator Hot Channel

Channel Power = 5.275 MW, CPR = 1.42 Active Fuel Length = 3.048 m / 120.00 inches		
Node (m above BAF)	**Axial Power Factor**	**Void Fraction**
1 (BAF+0.08)	0.41	0.00
2 (BAF+0.23)	1.21	0.01
3 (BAF+0.38)	1.56	0.15
4 (BAF+0.53)	1.63	0.35
5 (BAF+0.69)	1.59	0.50
6 (BAF+0.84)	1.52	0.60
7 (BAF+0.99)	1.43	0.67
8 (BAF+1.14)	1.35	0.71
9 (BAF+1.30)	1.26	0.75
10 (BAF+1.45)	1.18	0.77
11 (BAF+1.60)	1.10	0.79
12 (BAF+1.75)	1.02	0.81
13 (BAF+1.91)	0.89	0.83
14 (BAF+2.06)	0.82	0.84
15 (BAF+2.21)	0.77	0.86
16 (BAF+2.36)	0.69	0.87
17 (BAF+2.51)	0.59	0.87
18 (BAF+2.67)	0.47	0.88
19 (BAF+2.82)	0.34	0.88
20 (BAF+2.97)	0.15	0.89

NOTE: Normalized to 20

2.2 STABILITY EVALUATION

The stability licensing criterion for all nuclear power plants is set forth in 10 CFR 50 Appendix A, General Design Criterion 12 (GDC-12). This requires assurance that power oscillations, which can result in conditions exceeding specified acceptable fuel design limits, are either not possible or can be reliably detected and suppressed. Because the most limiting stability condition in the ESBWR normal operating region is at the rated power/flow condition, the ESBWR is designed so that power oscillations are not possible (that is, remains stable) throughout the whole operating region, including plant startup. In addition, the ESBWR is designed to be stable during AOOs. As a backup, the ESBWR implements a Detect and Suppress solution as a defense-in-depth system.

This Section summarizes the stability evaluation of the ESBWR design for the initial core for rated operating conditions.

2.2.1 Stability Performance During Power Operation

2.2.1.1 Stability Criteria

This discussion in the DCD applies to the initial core.

2.2.1.2 Analysis Methods

This discussion in the DCD applies to the initial core.

2.2.1.3 Steady State Stability Performance

2.2.1.3.1 Baseline Analysis

A baseline analysis was performed for the ESBWR at rated conditions, which are the most limiting from the perspective of stability due to the highest power/flow ratio Reference 2.2-1). Analysis was conducted for the GE14E initial core at various points in the cycle: BOC, Middle of Cycle (MOC) near the peak reactivity state, and End of Cycle (EOC). The initial conditions are tabulated in Table 2.2-1. The core average axial power shapes for the three exposure points are shown in Figure 2.2-3.

Channel Stability

Channel stability is evaluated for the highest power channels by perturbing the inlet flow velocity while maintaining constant channel power.

Super Bundle Stability

A super bundle is defined as a group of 16 bundles below a common chimney cell. The hydrodynamic stability of the highest power super bundle was analyzed by perturbing the inlet flow to the group of 16 bundles while maintaining constant power. The calculation was performed at MOC conditions because this is the most limiting for channel hydrodynamic stability for the initial core.

Core wide Stability

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

Regional Stability

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

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

Results

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

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

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

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

2.2.1.4 Statistical Analysis of ESBWR Stability

2.2.1.4.1 Channel Decay Ratio Statistical Analysis

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

Monte Carlo process. A One-Sided Upper Tolerance Limit with 95% content and 95% confidence level (OSUTL95/95) is calculated from the Monte Carlo distribution.

2.2.1.4.2 ***Core Wide Decay Ratio Statistical Analysis***

The Monte Carlo analysis of core stability was performed and documented in Reference 2.2-1 for the equilibrium core at rated power and flow at the conditions that were determined to be limiting. As for channel stability, a total of 59 trials were made. In each trial, random draws are made for each of the parameters determined to be important for stability. A TRACG calculation is made with this perturbed set of parameters to obtain a new steady state. The core decay ratio is then calculated by applying a pressure perturbation in turbine inlet pressure. This constitutes one trial in the Monte Carlo process. An OSUTL95/95 is calculated from the Monte Carlo distribution.

2.2.1.4.3 ***Regional Decay Ratio Statistical Analysis***

The Monte Carlo analysis of regional stability was performed and documented in Reference 2.2-1 for the equilibrium core at rated power and flow at the conditions that were determined to be limiting. As for core-wide stability, a total of 59 trials were made. In each trial, random draws are made for each of the parameters determined to be important for regional stability. A TRACG calculation is made with this perturbed set of parameters to obtain a new steady state. The regional decay ratio is then calculated by applying an instantaneous inlet velocity perturbation. A positive perturbation is applied to all channel groups on one side of the line of symmetry of the harmonic mode; a negative perturbation is applied to the channel groups on the other side. The decay ratio was extracted for the high power channel group from the transient response. This constitutes one trial in the Monte Carlo process. An OSUTL95/95 is calculated from the Monte Carlo distribution.

2.2.1.4.4 ***Comparison with Design Limits***

Figure 2.2-4 shows the stability map with the design criteria. The baseline results for core, channel and regional decay ratios are compared against the design goal. Figure 2.2-4 shows that the design goal is satisfied for the ESBWR core. The demonstration of stability margins has been performed for an initial core based on the GE14 fuel design described in Section 4.2 of Reference 2.2-2. The nominal decay ratio is similar to the equilibrium core loading pattern results documented in Reference 2.2-1. It should be noted that there is no change in the fuel design, only in the core-loading pattern. The Monte Carlo decay ratio adders determined and documented in Reference 2.2-1 are applied to the initial core decay ratio results. The uncertainty results for the initial core may vary. There is sufficient margin to the design criteria to accommodate a variation in the uncertainty.

2.2.1.5 Stability Performance During AOOs

Two limiting AOOs were identified in Reference 2.2-2: Loss-of Feedwater Heater (LOFWH), which results in increased power; and Loss of Feedwater Flow (LOFW), which results in a lower flow.

Based on the results documented in Reference 2.2-1 the LOFWH scenario is more limiting for stability performance than the LOFW event. In addition, the stability analysis at reduced feedwater temperature operation condition (i.e. 100% rated power and reduced feedwater temperature) is considered to bound the analysis at rated conditions (i.e. 100% rated power and rated feedwater temperature). Results of stability analysis for the LOFWH case are documented in Licensing Topical Report, NEDO-33338, "ESBWR Feedwater Temperature Operating Domain Transient and Accident Analysis."

2.2.1.6 Stability Performance During Anticipated Transients Without Scram

This discussion in the DCD applies to the initial core.

2.2.2 Stability Performance During Plant Startup

This discussion in the DCD applies to the initial core.

2.2.2.1 Phenomena Governing Oscillations During Startup

This discussion in the DCD applies to the initial core.

2.2.2.2 TRACG Analysis of Typical Startup Scenarios

This discussion in the DCD applies to the initial core.

2.2.2.2.1 ESBWR Plant Startup

This discussion in the DCD applies to the initial core.

2.2.2.2.2 TRACG Calculation for Simulated Startup Scenarios

This discussion in the DCD applies to the initial core.

2.2.2.2.3 TRACG Calculation of ESBWR Startup with Neutronic Feedback and MSIVs Open

A TRACG simulation of ESBWR startup with the neutronic feedback and MSIVs open is performed using the near limiting heat up rate. This is an example of a startup transient. Results of this simulation demonstrate that at the limiting heat up rate, no difficulties and no large power oscillations were encountered during the startup transient. This TRACG calculation is performed activating the 3D kinetics model. The calculation is initiated at the end of the de-aeration period; see Figure 2.2-6. The water level is maintained near the top of the separators. The MSIVs are kept open to heatup the steam lines simultaneously with the RPV pressurization. Initially all control rods are in fully inserted position.

The 269 control rods in ESBWR are divided into 10 groups and the rod group positions are shown in Figure 2.2-7. Rod Group # 10 represents the control rods for the 25 control cells. The grouping of control rods and the withdrawal sequence during the startup are similar to those used for operating plants. The withdrawal speeds for each of these groups during the transient are specified as TRACG input to simulate the operator actions to maintain the reactor at power during the startup transient. These rod groups are slowly withdrawn to maintain the total reactivity close to 0.0 and the total power level is maintained at around 90 MWt until the reactor is pressurized to the desired value. Subsequently, the bypass valve is opened and the power level is increased in large steps (by means of additional rod withdrawals) to achieve rated pressure.

Figure 2.2-8 shows the withdrawal fraction for all control rods. After Groups 1 through 5 are fully withdrawn, the control rod withdrawal fraction is 0.577, that is, 42.3% of all rods are in fully inserted position. At this time, the reactor is critical. Groups 6 and the next several groups are withdrawn with slower speed to avoid rapid change in total reactivity and reactor power.

Figures 2.2-9 and 2.2-10 show total reactivity and reactor power. It can be seen trajectory of power closely follows trend in reactivity. For the first 1500 seconds, the total reactor power consists mainly of decay heat. After the core becomes critical, there is a step increase in total reactor power. From this time on, the rod groups are slowly withdrawn to maintain the total reactor power at around 90 MWt. The total power is maintained around 90 MWt by the continuous withdrawal of the control rods. No significant core void is calculated until the bypass valve is opened, when the temperature and pressure are near the operating conditions. The heatup rate for this case is little below the maximum allowed rate considering thermal stress of 55°C/hour (~ 100 °F/hour).

Figure 2.2-11 shows the steam dome pressure response for this case. The RPV pressurizes to 6.3 MPa (914 psia) in ~ 3.7 hours and the bypass is opened. With the bypass open, the power is limited by BOP systems not by heatup rate. The control rods are withdrawn further to step up the power and to reach the rated pressure at 4.4 hours. At this time, Rod Groups 1 to 5 and 7 are fully withdrawan, Group 6 is ~ 80% and Group 8 is about 25% withdrawn. Groups 9 and 10 (25 control cell rods) are in fully inserted position.

Figure 2.2-12 shows the core inlet subcooling as a function of time. The local inlet subcooling drops as the system pressurizes to 6.3 MPa (914 psia). The core flow transient response is shown in Figure 2.2-13. There are two periods with small flow noise: around 2000 seconds corresponding to the step increase in power (Figure 2.2-10) and between 7000 and 11000 seconds corresponding to some void initiation at the top of channels (Figure 2.2-15). Steady void fraction is established at the top of the separators after 15500 seconds. There are no significant fluctuations in the neutron flux during these periods. The flow result is similar to the case with MSIVs closed.

Figure 2.2-15 shows core void in the highest power bundles. Vapor generation begins at the top of the high power bundles at pressure of about 1.3 MPa (189 psia). Voids propagate about a quarter of the height into the bundle at 12000 s, by which time the system pressure is above 4.5 MPa (~ 653 psia). The high power bundle flow follows the core average flow response. The peripheral bundles are in upflow throughout the transient.

Margins to thermal limits CPR were calculated for this startup case. Large margins are maintained for all bundles throughout the transient.

Figures 2.2-16 and 2.2-17 show 'Hot Bundle Exit Flow' and 'Hot Bundle CPR' respectively. Corresponding plots for a typical 'Peripheral Bundle' are given in Figures 2.2-18 and 2.2-19. A comparison with the results from the equilibrium-core study documented in the DCD (Reference 2.2-2), wherein MSIVs were kept closed, shows no significant change in the results.

2.2.3 References

2.2-1 GE Nuclear Energy, B.S.Shiralkar, et al, "TRACG Application for ESBWR Stability Analysis," NEDE-33083P, Supplement 1, December 2004.

2.2-2 GE Nuclear Energy, "ESBWR Design Control Document" 26A6642, Revision 4, September 2007.

Table 2.2-1

Initial Conditions for Channel and Core Stability Analysis

Parameter	Value		
	BOC	MOC	EOC
Core Thermal Power (MWt)	4500	4500	4500
Core Flow (kg/s)*	9,930 (78.81 Mlbm/hr)	9,910 (78.65 Mlbm/hr)	10,040 (79.68 Mlbm/hr)
Feedwater temperature (°C)	216 (~420 °F)	216 (~420 °F)	216 (~420 °F)
Narrow range water level (m)	20.7 (67.9 ft)	20.7 (67.9 ft)	20.7 (67.9 ft)
Feedwater flow (kg/s)*	2428 (19.27 Mlbm/hr)	2427 (19.26 Mlbm/hr)	2428 (19.27 Mlbm/hr)
Core inlet subcooling* (°C)	16.4 (29.5 °F)	16.4 (29.5 °F)	16.2 (29.2 °F)
Steam dome pressure (MPa)	7.17 (1040 psia)	7.17 (1040 psia)	7.17 (1040 psia)
ICPR*	1.39	1.41	1.45
Hot Bundle Power (MWt)*	5.25	5.32	5.06
Hot Bundle flow (kg/s)*	8.4 (66.67 klbm/hr)	8.4 (66.67 klbm/hr)	8.8 (69.84 klbm/hr)

*Calculated parameter

Table 2.2-2

Baseline Stability Analysis Results

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

Figure 2.2-1. Two-Dimensional Stability Map for ESBWR
Data in DCD applies to the initial core.

Figure 2.2-2. Three-Dimensional Stability Map for ESBWR
Data in DCD applies to the initial core.

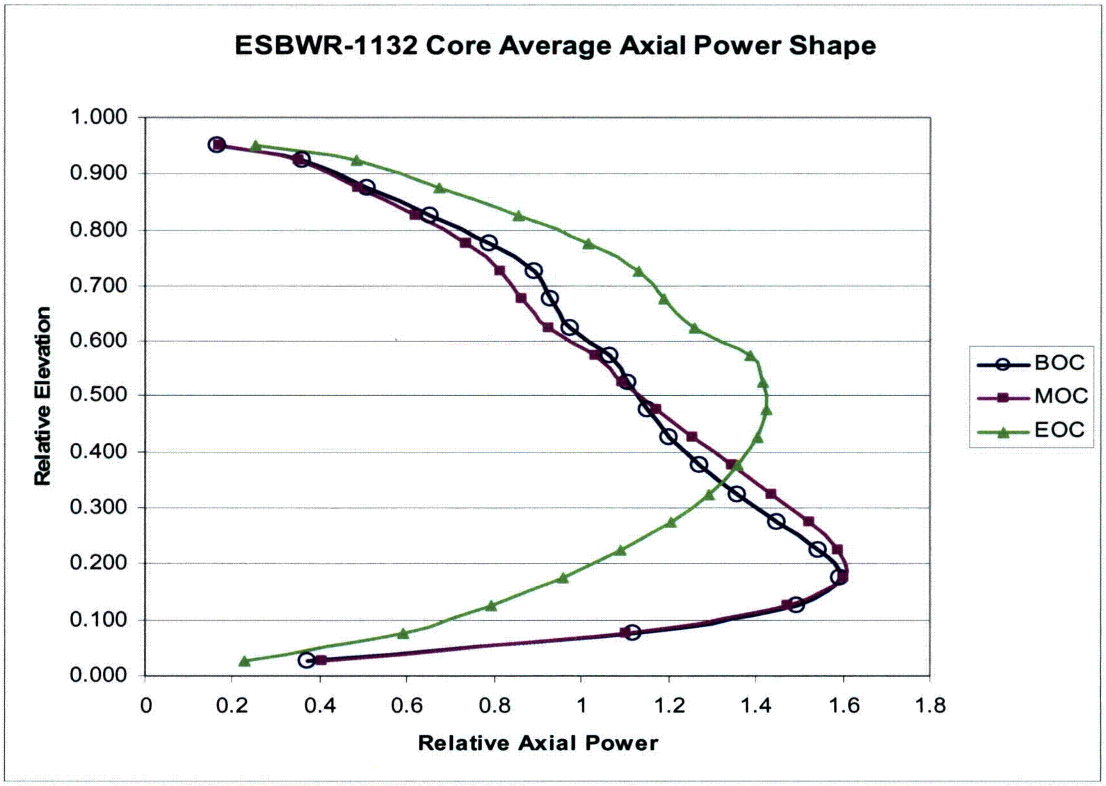


Figure 2.2-3. Core Average Axial Power Shape at Different Exposures

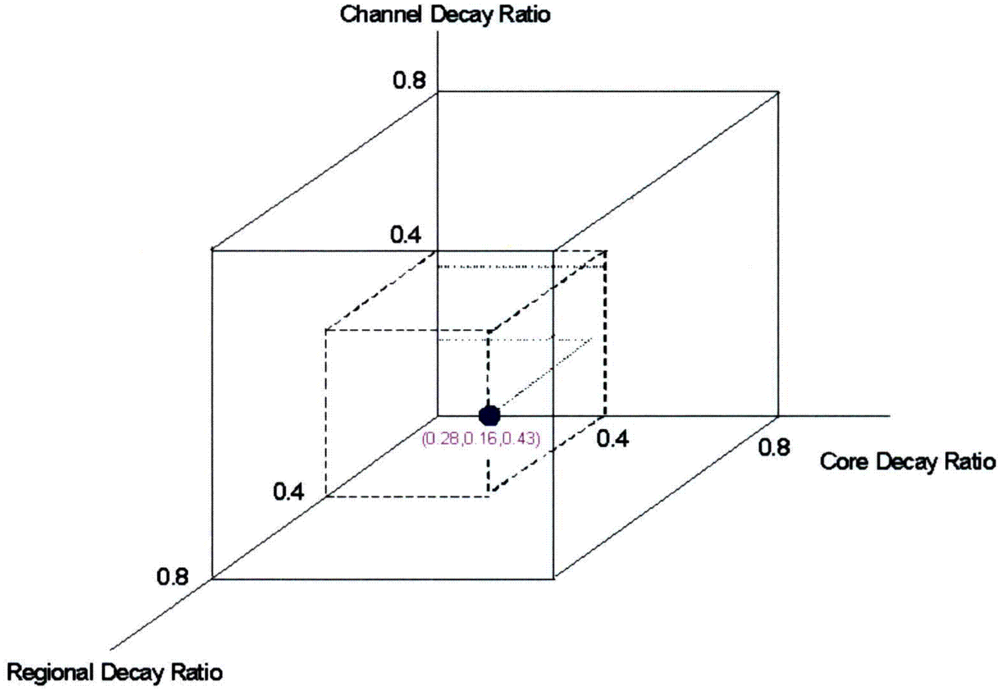


Figure 2.2-4. Decay Ratio Results Compared to Design Criteria

Figure 2.2-5. Stability in Expanded Operating Map

Data in the DCD applies to the initial core.

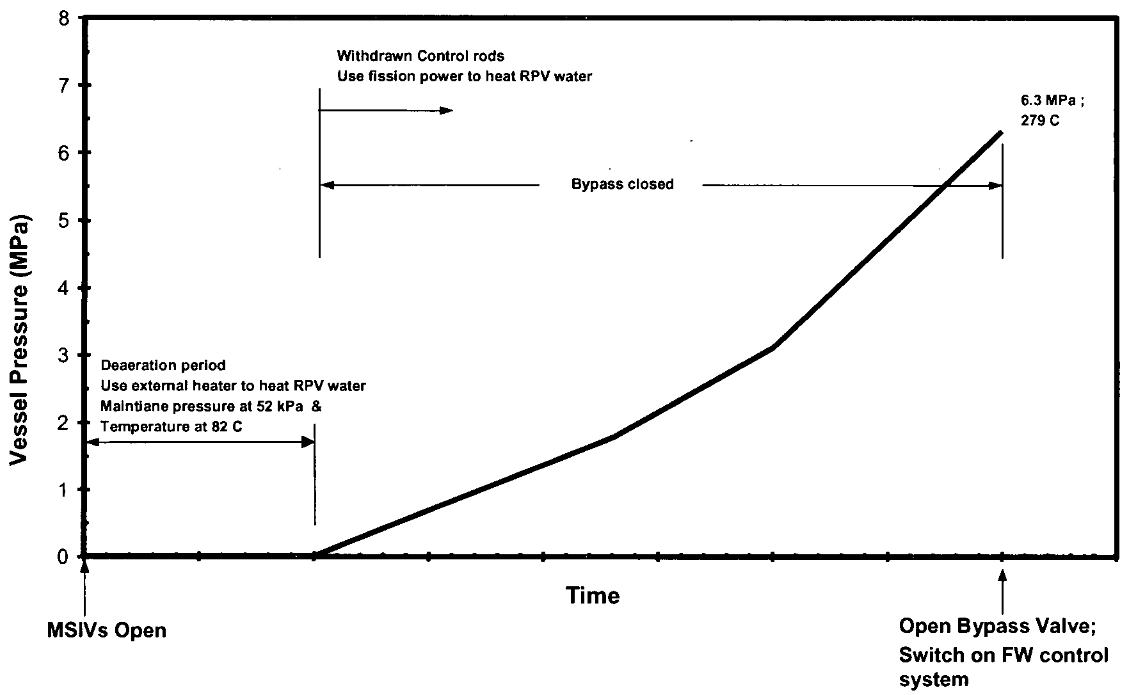


Figure 2.2-6. ESBWR Startup Trajectory with MSIVs Open

Row & Column #	1	3	5	7	9	11	13	15	17	19	21	23	25	27	29	31	33	35	37
1								4	5	3	5	4							
3						5	2	9	1	10E	2	9	1	5					
5					6	4	8	3	8	4	8	3	8	4	6				
7				5	2	9	1	10E	2	9C	1	10E	2	9	1	5			
9			6	4	7	3	8	4	7	3	7	4	8	3	7	4	6		
11		5	2	9	1	10D	2	9B	1	10C	2	9B	1	10D	2	9	1	5	
13		4	8	3	8	4	7	3	8	4	8	3	7	4	8	3	8	4	
15	2	9	1	10E	2	9B	1	10B	2	9A	1	10B	2	9B	1	10E	2	9	1
17	5	3	8	4	7	3	8	4	7	3	7	4	8	3	7	4	8	3	5
19	1	10E	2	9C	1	10C	2	9A	1	10A	2	9A	1	10C	2	9C	1	10E	2
21	5	4	8	3	7	4	8	3	7	4	7	3	8	4	7	3	8	4	5
23	2	9	1	10E	2	9B	1	10B	2	9A	1	10B	2	9B	1	10E	2	9	1
25		3	8	4	8	3	7	4	8	3	8	4	7	3	8	4	8	3	
27		5	2	9	1	10D	2	9B	1	10C	2	9B	1	10D	2	9	1	5	
29			6	3	7	4	8	3	7	4	7	3	8	4	7	3	6		
31				5	2	9	1	10E	2	9C	1	10E	2	9	1	5			
33					6	3	8	4	8	3	8	4	8	3	6				
35						5	2	9	1	10E	2	9	1	5					
37								3	5	4	5	3							

Figure 2.2-7. ESBWR Control Rod Groups for Startup Simulation

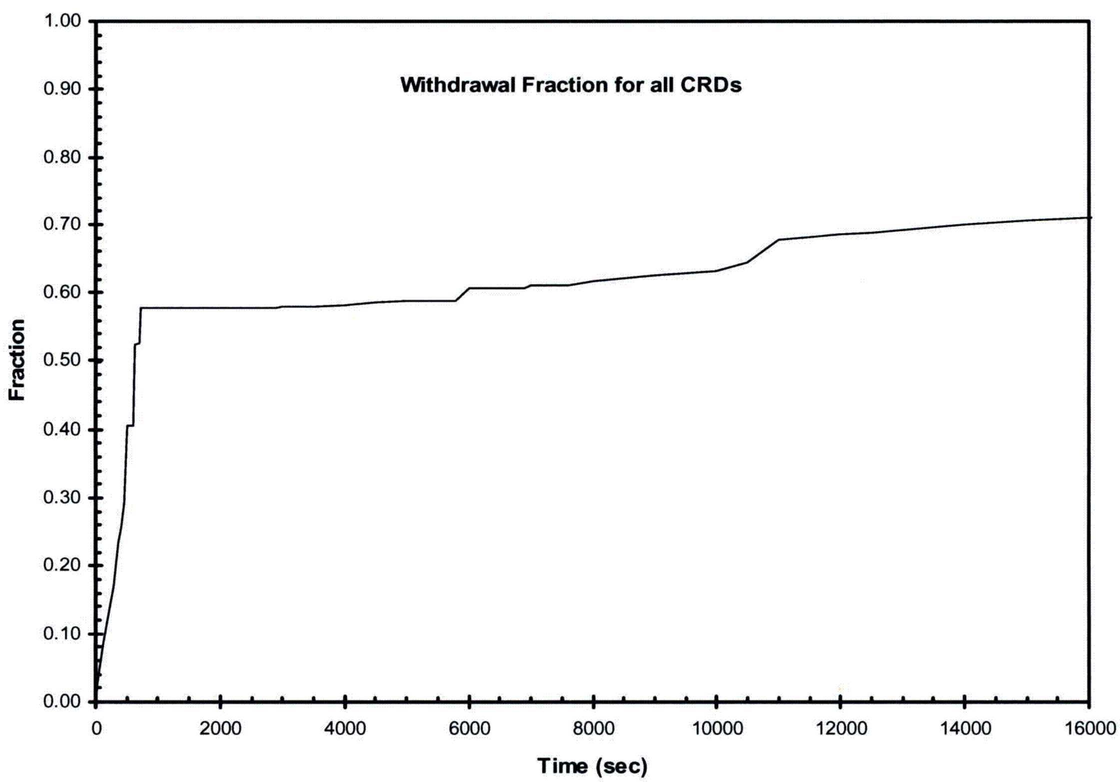


Figure 2.2-8. Withdrawal Fraction for all Control Rods

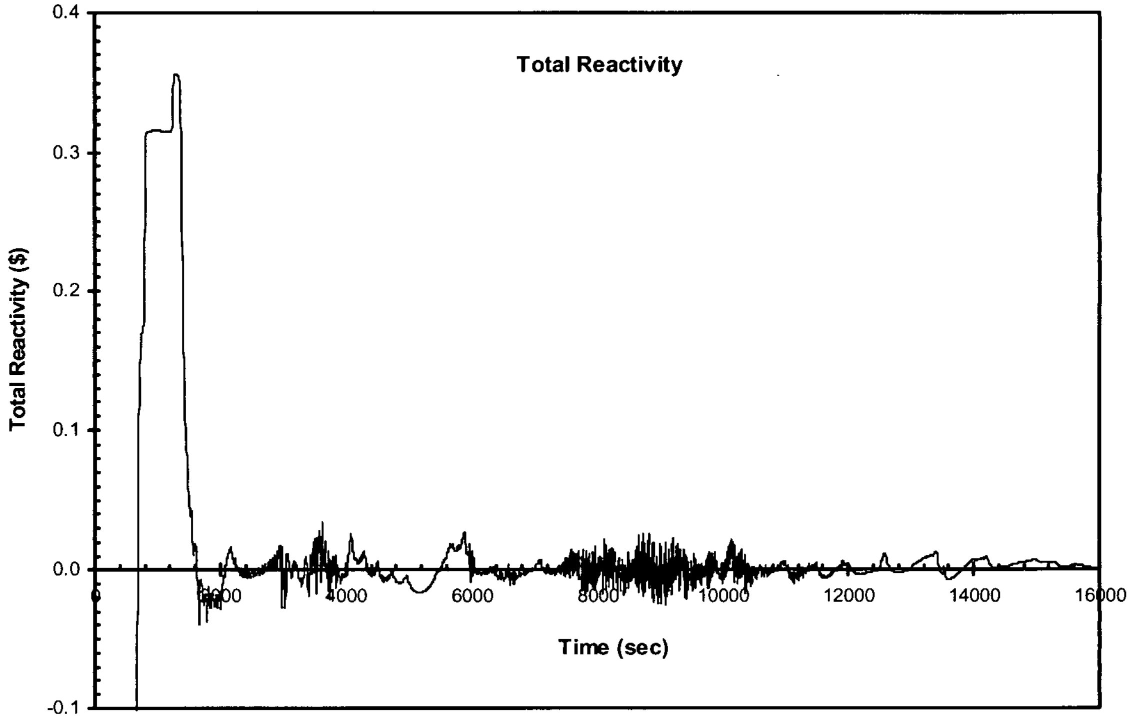


Figure 2.2-9. Total Reactivity

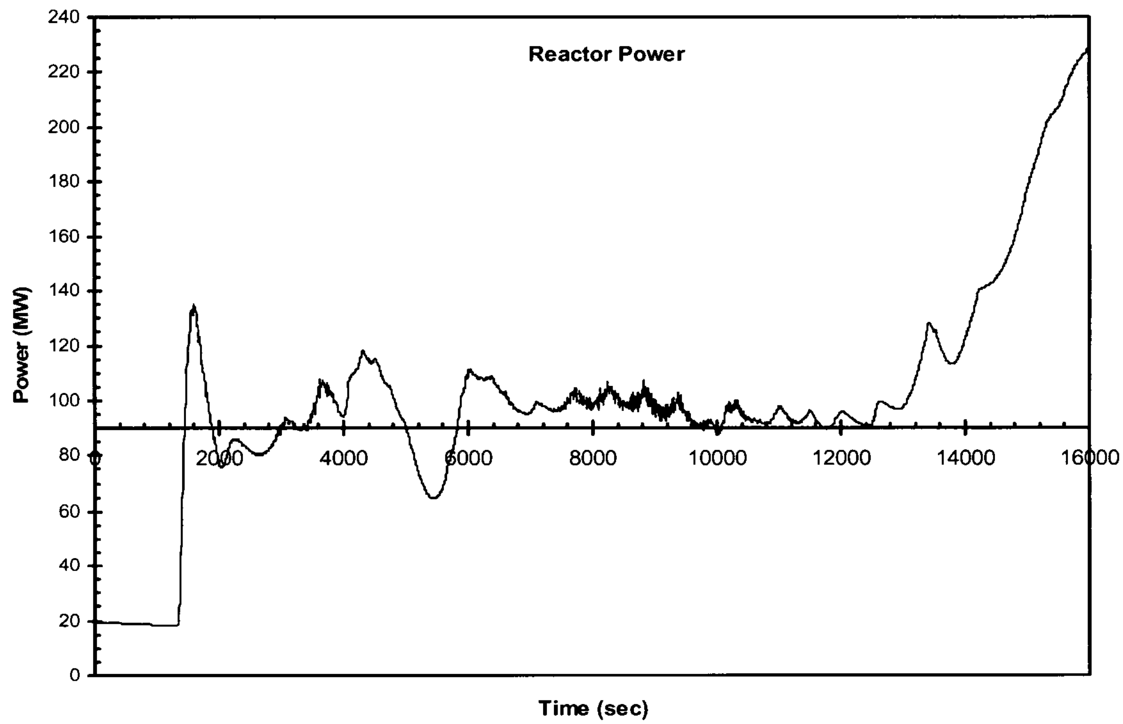


Figure 2.2-10. Reactor Power

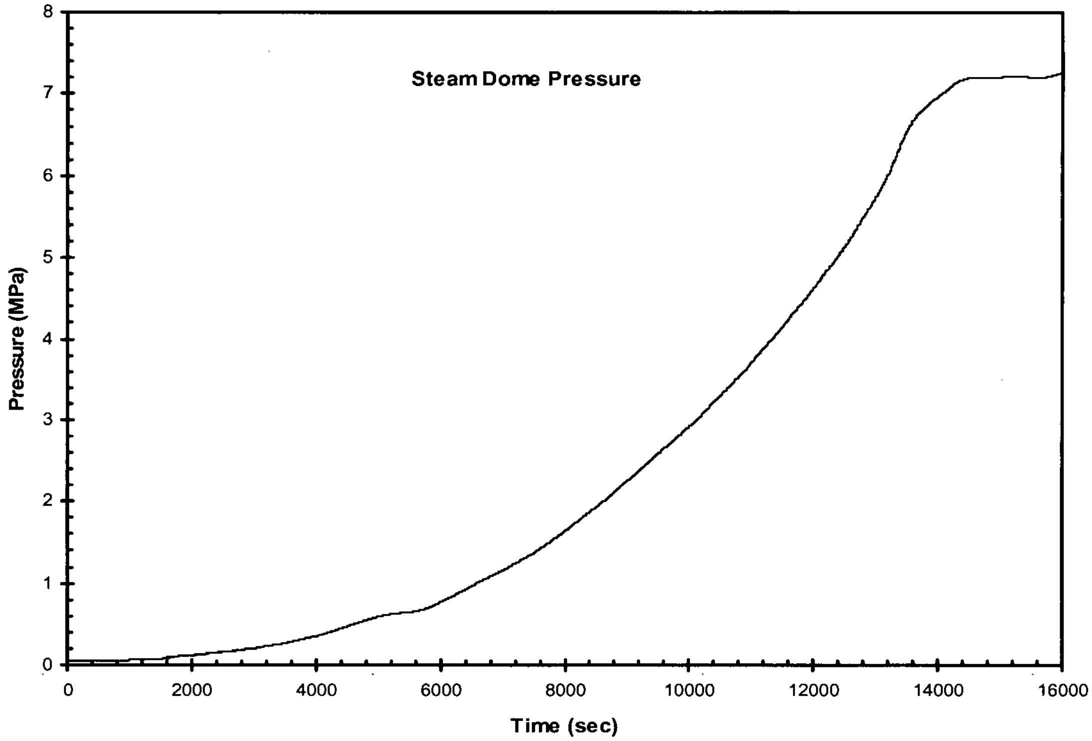


Figure 2.2-11. Steam Dome Pressure

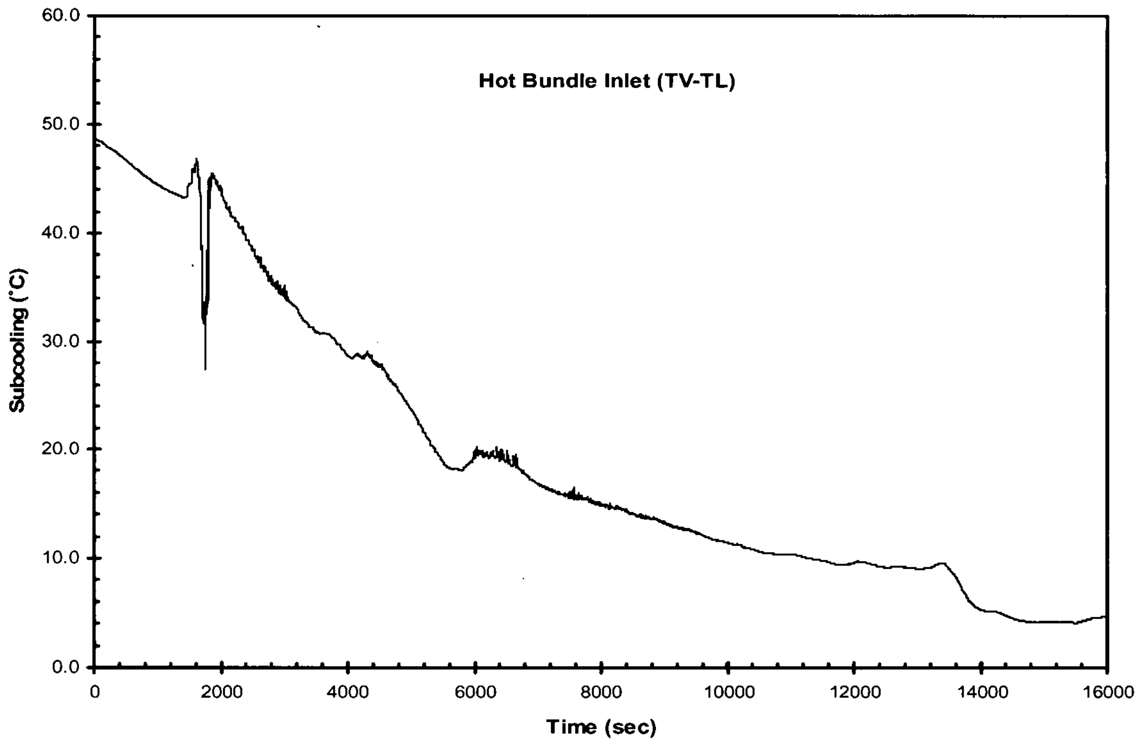


Figure 2.2-12. Core Inlet Subcooling

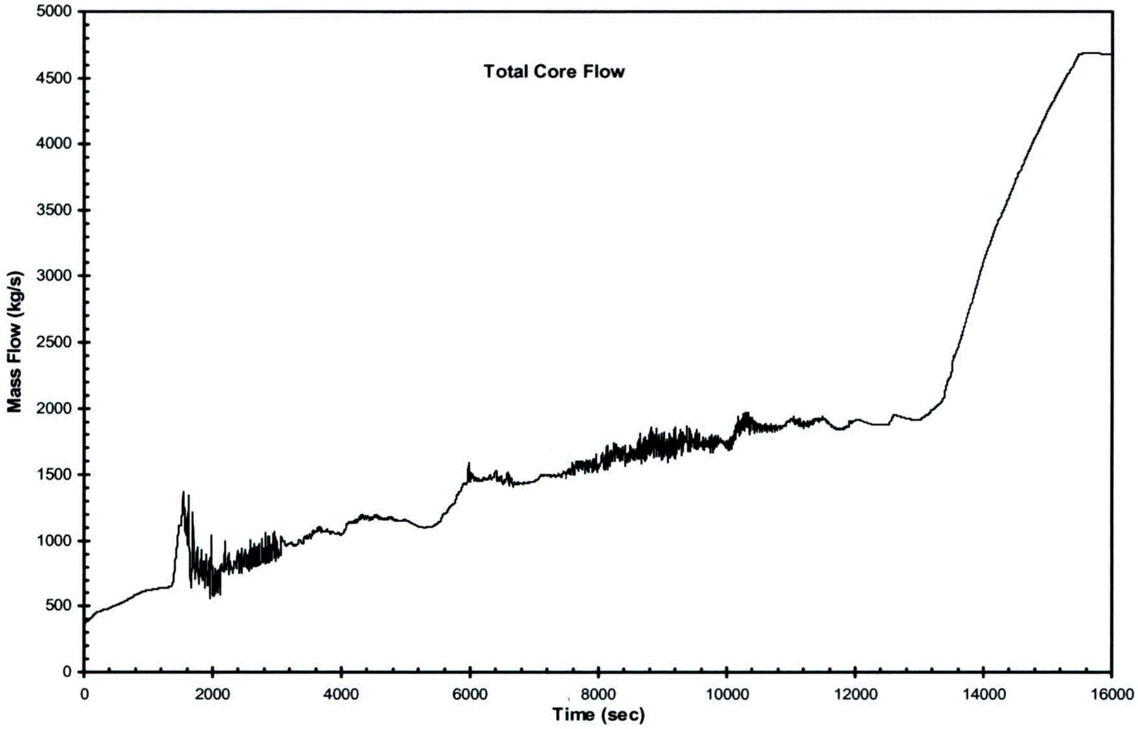


Figure 2.2-13. Core Inlet Flow

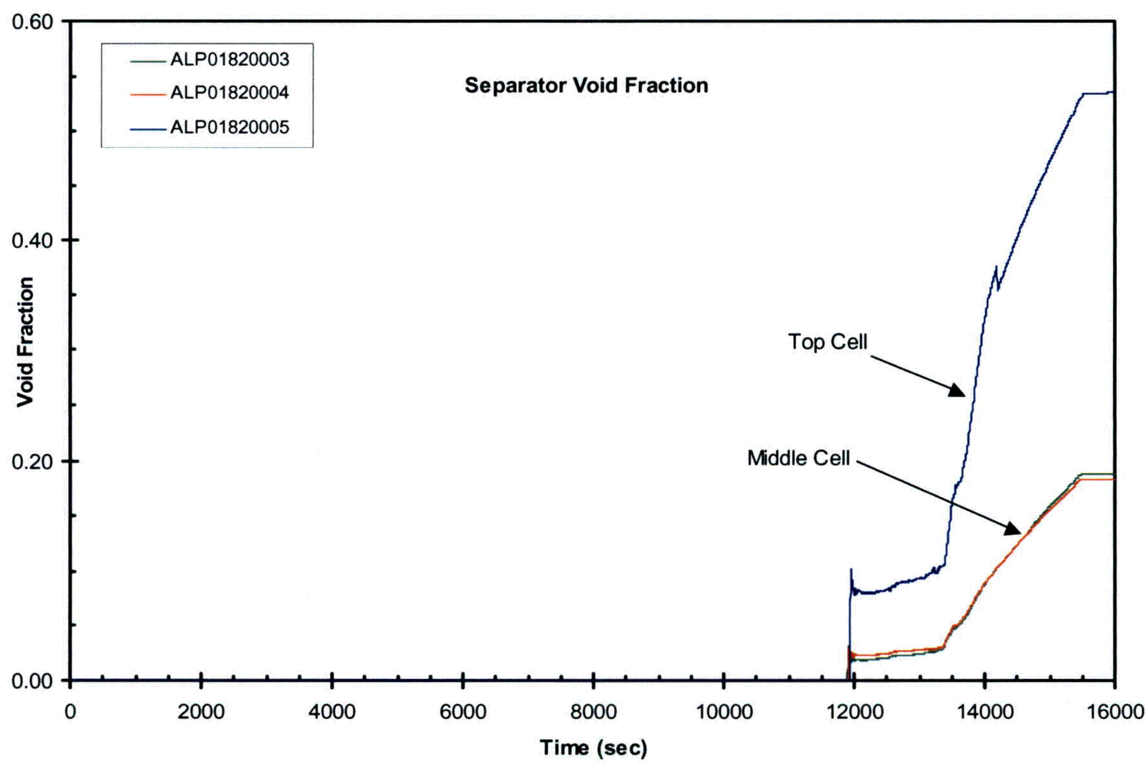


Figure 2.2-14. Separator Void Fraction

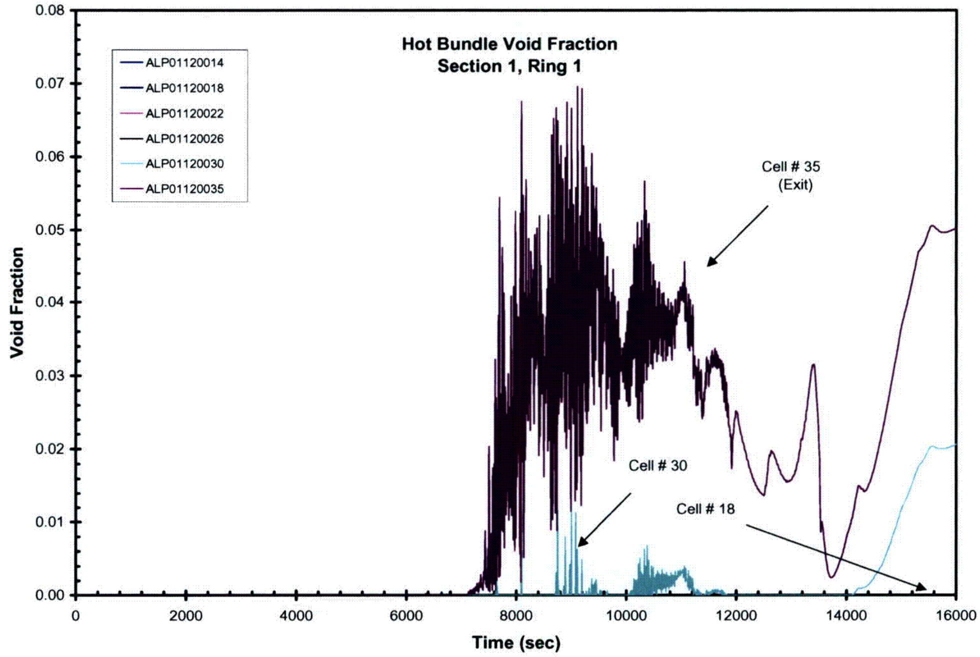


Figure 2.2-15. Hot Bundle Void Fraction

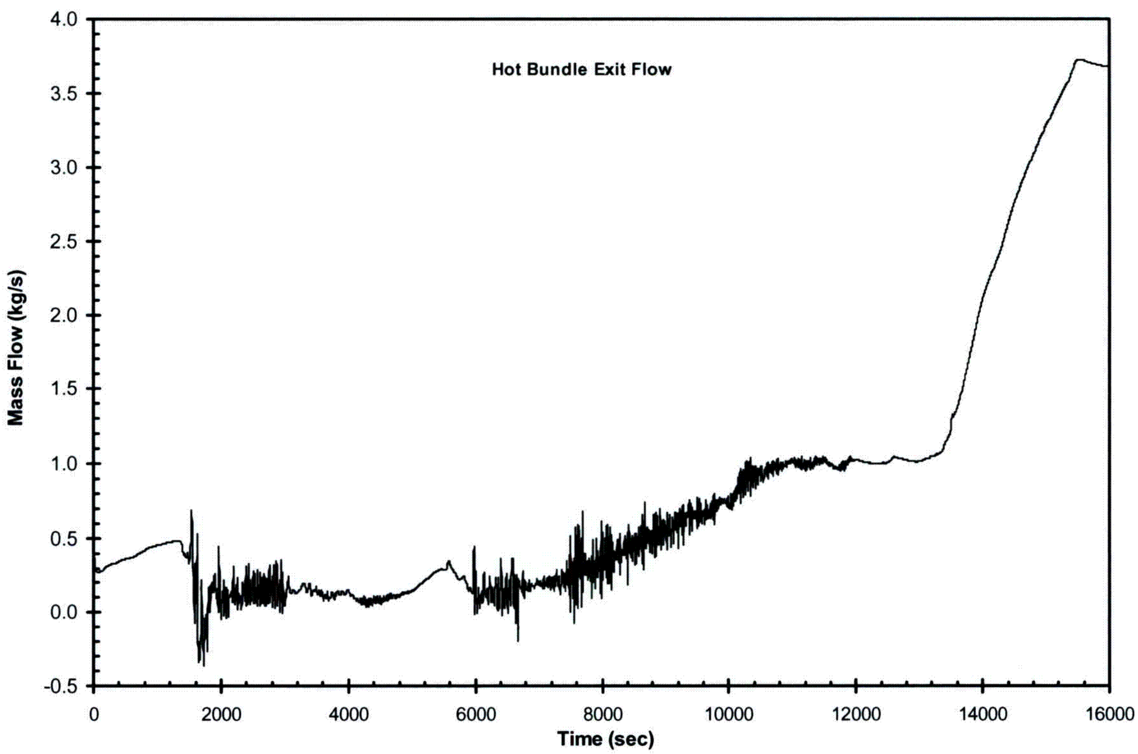


Figure 2.2-16. Hot Bundle Exit Flow

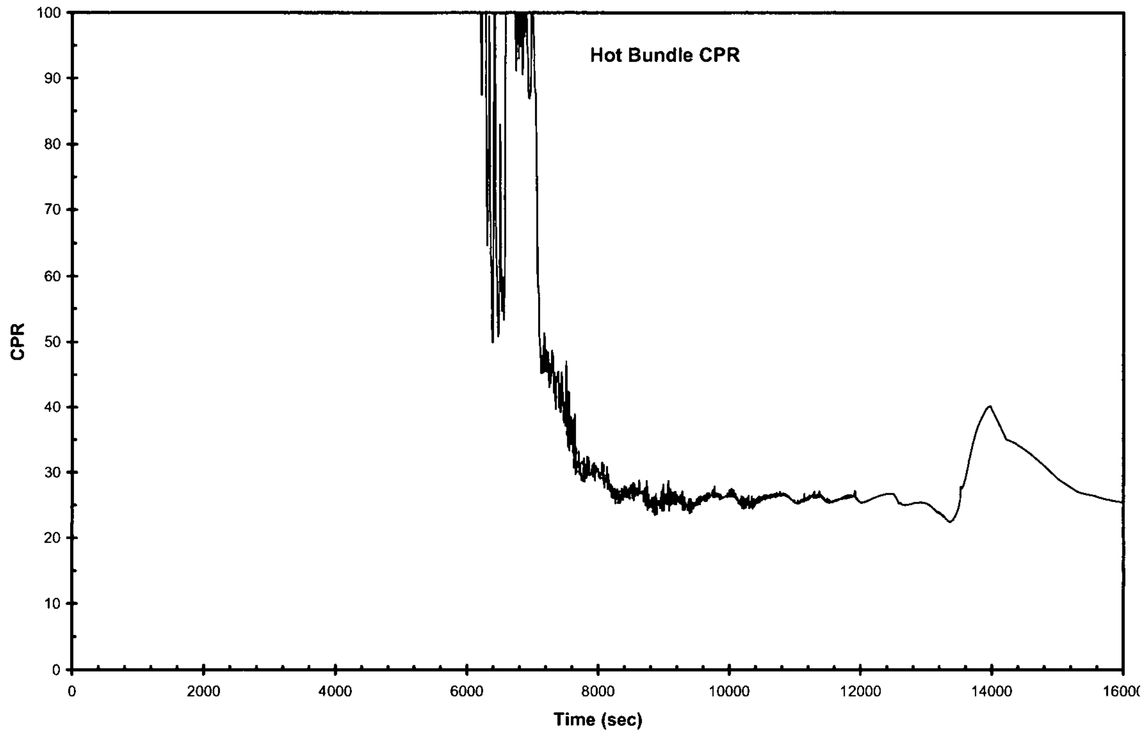


Figure 2.2-17. Hot Bundle CPR

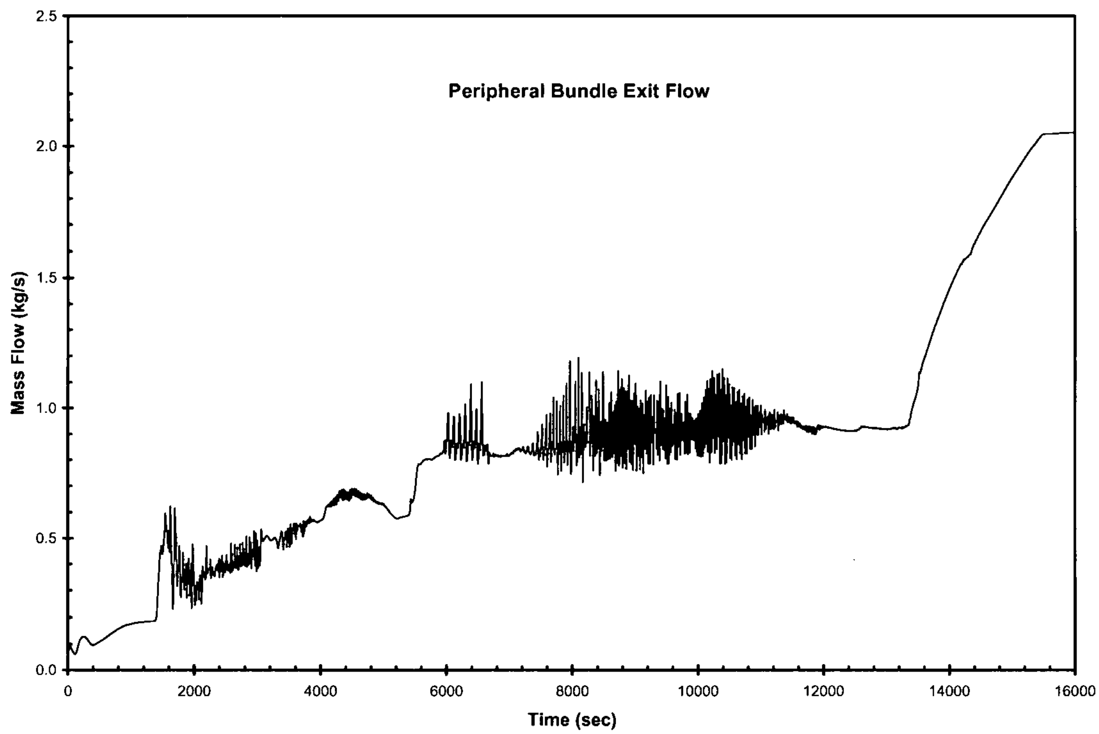


Figure 2.2-18. Peripheral Bundle Exit Flow

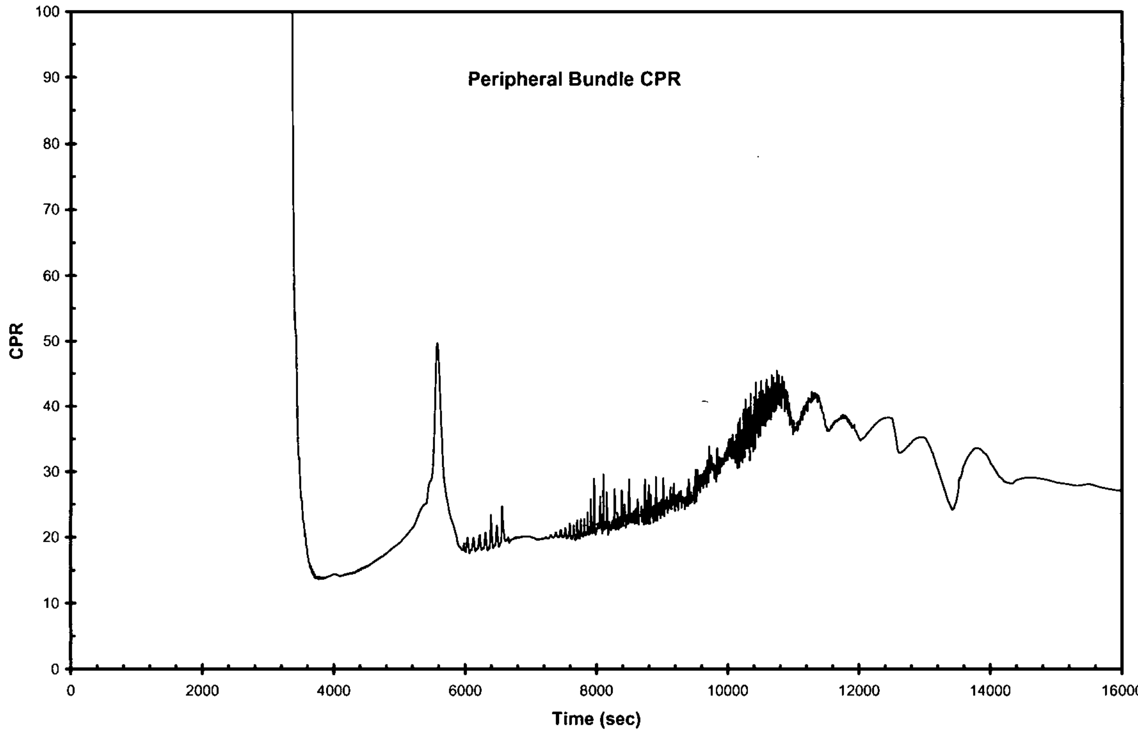


Figure 2.2-19. Peripheral Bundle CPR

2.3 ANALYSIS OF ANTICIPATED OPERATIONAL OCCURRENCES

Each of the anticipated operational occurrences (AOOs) addressed in Section 15.1, "Nuclear Safety Operations Analysis" (NSOA), of Reference 2.3-5, GE Nuclear Energy, "ESBWR Design Control Document" 26A6642BP, Revision 4, September 2007, is evaluated in the following subsections: Appendix 15A of Reference 2.3-5 provides a determination of event frequency to categorize AOOs as defined in 10 CFR 50 Appendix A. Tables 2.3-1, 2.3-2 and 2.3-3 provide the important input parameters and initial conditions used/assumed in the AOO analyses.

In the analysis of AOOs and Infrequent Events in Section 2.4 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-relates 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).

2.3.0 Assumptions

Assumptions are listed in the event discussions and Table 2.3-1.

2.3.1 Decrease In Core Coolant Temperature

2.3.1.1 Loss Of Feedwater Heating

2.3.1.1.1 Identification of Causes

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

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

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

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

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

The Feedwater Control System (FWCS) includes logic to mitigate the effects of a loss of FW heating capability. The system is constantly monitoring the actual FW temperature and comparing it with a reference temperature. When a loss of FW heating is detected [i.e., when the difference between the actual and reference temperatures exceeds a ΔT setpoint], the FWCS sends an alarm to the operator, and sends a signal to the Non-Safety Related Distributed Control and Information System (N-DCIS) to initiate the Selected Rods Insertion (SRI), and selected control rods run-in (SCRRI) function to automatically reduce the reactor power. These functions are also collectively referred to as SCRRI/SRI.

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

The SCRRI/SRI function reduces the core power and limits the change in MCPR after a Loss of Feedwater Heating. The SCRRI/SRI rod pattern depends on the fuel cycle exposure and initiating event. Under circumstances in which no SCRRI/SRI rod pattern is defined the power would rise to the to the Simulated Thermal Power Trip (STPT) scram setpoint. This scenario is analyzed below.

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

Sequence of Events

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

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

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.

2.3.1.1.3 ***Core and System Performance***

Input Parameters and Initial Conditions

The event is simulated by initiating a reduction in feedwater temperature that results in power reaching the STPT, but does not initiate a scram. This bounds the case in which the reduction in feedwater temperature is larger and results in an STPT scram.

Results

The results are summarized in Table 2.3-4.

Nuclear system pressure does not significantly change, and consequently, the RCPB is not threatened.

2.3.1.1.4 ***Barrier Performance***

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

2.3.1.1.5 ***Radiological Consequences***

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

2.3.2 Increase In Reactor Pressure

2.3.2.1 Closure of One Turbine Control Valve

2.3.2.1.1 Identification of Causes

This discussion in the DCD applies to the initial core.

2.3.2.1.2 Sequence of Events and System Operation

Sequence of Events

Postulating an actuator failure of the SB&PC system causes one TCV to close. The pressure increases because the reactor is still generating the initial steam flow. The SB&PC system opens the remaining control valves and some bypass valves. This sequence of events is listed in Table 2.3-6 for Figure 2.3-2, for a fast closure with partial arc, and in Table 2.3-7 for Figure 2.3-3, for a slow closure with partial arc.

Systems Operation

Normal plant instrumentation and control are assumed to function. After a closure of one turbine control valve, the steam flow rate that can be transmitted through the remaining three TCVs depends upon the turbine configuration. For plants with full-arc turbine admission, the steam flow through the remaining three TCVs is at least 95% of rated steam flow. This capacity drops to about 85% of rated steam flow for plants with partial-arc turbine admission. Therefore, this transient is less severe for plants with full-arc turbine admission. In this analysis, the case with partial-arc turbine admission is analyzed to cover all plants.

Table 2.3-1 provides the following data for the TCV:

- Design full stroke closure time, from fully open to fully closed;
- Bounding closure time assumed in the fast closure analysis;
- Closure time assumed in the slow closure analysis; and
- Percent of rated steam flow that can pass through three TCVs.

2.3.2.1.3 Core and System Performance

A simulated fast closure of one TCV is presented in Figure 2.3-2. Neutron flux increases, because of the void reduction caused by the pressure increase. However, the sensed neutron flux does not reach the high neutron flux scram setpoint. When the sensed reactor pressure increases, the pressure regulator opens the bypass valves, keeping the reactor pressure at a constant level. The calculated peak thermal flux is provided in Table 2.3-4. The number of rods in boiling transition, for this transient, remains within the acceptance criterion for AOOs. Therefore, the design basis is satisfied.

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

2.3.2.1.4 Barrier Performance

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

2.3.2.1.5 Radiological Consequences

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

2.3.2.2 Generator Load Rejection With Turbine Bypass

2.3.2.2.1 Identification of Causes

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

After sensing a significant loss of electrical load on the generator, the TCVs are commanded to close rapidly. At the same time, the turbine bypass valves are signaled to open in the "fast" opening mode by the SB&PC system, which uses a triplicated digital controller. As presented in Subsection 2.3.4.2, no single failure can cause all turbine bypass valves to fail to open on demand.

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

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

2.3.2.2.2 Sequence of Events and System Operation

Sequence of Events

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

Identification of Operator Actions

Relatively small changes in plant conditions are experienced. The operator should, after checking that the SCRRI/SRI system has been activated, check reactor water level, reactor pressure and MSIV status. If conditions are normal, no further operator actions are needed.

System Operation

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

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

2.3.2.2.3 *Core and System Performance*

Input Parameters and Initial Conditions

The turbine electro-hydraulic control system (EHC) detects load rejection before a measurable turbine speed change takes place.

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

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

Results

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

2.3.2.2.4 *Barrier Performance*

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

2.3.2.2.5 *Radiological Consequences*

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

2.3.2.3 Generator Load Rejection With a Single Failure in the Turbine Bypass System

2.3.2.3.1 *Identification of Causes*

This discussion in the DCD applies to the initial core.

2.3.2.3.2 *Sequence of Events and System Operation*

Sequence of Events

A loss of generator electrical load with a single failure in the turbine bypass system from high power conditions produces the sequence of events listed in Table 2.3-9.

Identification of Operator Actions

The operator should:

- Verify that all rods are inserted;

- Follow the scram procedure;
- Verify proper bypass valve performance;
- Observe that the FW/level controls have maintained the reactor water level at a satisfactory value;
- Observe that the pressure regulator is controlling reactor pressure at the desired value; and
- Observe reactor peak power and pressure.

System Operation

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

Conservatively, and to cover all possible failures, it is assumed that the system with a single failure only opens to 50% of the total steam bypass capacity.

All plant control systems maintain normal operation unless specifically designated.

2.3.2.3.3 ***Core and System Performance***

Input Parameters and Initial Conditions

The turbine electro-hydraulic control system (EHC) detects load rejection before a measurable turbine speed change takes place.

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

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

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

Results

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

2.3.2.3.4 ***Barrier Performance***

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

2.3.2.3.5 *Radiological Consequences*

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

2.3.2.4 Turbine Trip With Turbine Bypass

2.3.2.4.1 *Identification of Causes*

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

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

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

2.3.2.4.2 *Sequence of Events and Systems Operation*

Sequence of Events

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

Identification of Operator Actions

Relatively small changes in plant conditions are experienced. The operator should, after checking that the SCRRI/SRI system has been activated, check reactor water level, reactor pressure and MSIV status. If conditions are normal, no further operator actions are needed.

Systems Operation

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

2.3.2.4.3 *Core and System Performance*

Input Parameters and Initial Conditions

Table 2.3-1 provides the Turbine Stop Valve (TSV) full stroke closure time design range, and the worst case (bounding) TSV closure time assumed in the analysis.

Results

A turbine trip with the bypass system operating normally is simulated at rated power conditions as shown in Figure 2.3-6. Table 2.3-4 summarizes the analysis results. The neutron flux increases rapidly because of the void reduction caused by the pressure increase. However, the pressure increase is limited by the initiation of the steam bypass operation. Peak simulated thermal heat flux does not significantly exceed (< 1%) of its initial value. After the control system verifies that the bypass capacity is adequate, the system will activate the SCRRI/SRI to reduce the power to 60%. The number of rods in boiling transition during this event remains within the acceptance criterion for AOOs.

2.3.2.4.4 *Barrier Performance*

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

2.3.2.4.5 *Radiological Consequences*

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

2.3.2.5 Turbine Trip With a Single Failure in the Turbine Bypass System

2.3.2.5.1 *Identification of Causes*

This discussion in the DCD applies to the initial core.

2.3.2.5.2 *Sequence of Events and Systems Operation*

Sequence of Events

Turbine trip with a single failure in the turbine bypass system at high power produces the sequence of events listed in Table 2.3-11.

Identification of Operator Actions

The operator should:

- Verify that all rods are inserted;
- Follow the scram procedure;
- Verify that the generator breaker is automatically open to allow electrical buses originally supplied by the generator to be supplied by the incoming power;
- Monitor reactor water level and pressure;
- Check turbine for proper operation of all auxiliaries during coastdown;
- Manually initiate ICs, if necessary, to control reactor pressure;
- Depending on conditions, maintain pressure for restart purposes, or initiate normal operating procedures for cooldown;
- Put the mode switch in the startup position before the reactor pressure decays to below 6 MPa (870 psig);

Systems Operation

All plant control systems maintain normal operation unless specifically designated to the contrary. Credit is taken for successful operation of the Reactor Protection System (RPS).

Conservatively and to cover all possible failures it is assumed that the system with a single failure only opens to 50% of the total steam bypass capacity.

2.3.2.5.3 Core and System Performance

Input Parameters and Initial Conditions

Table 2.3-1 provides the Turbine Stop Valve (TSV) full stroke closure time design range, and the worst case (bounding) TSV closure time assumed in the analysis. A reactor scram occurs due to fast TSV closure, with inadequate availability of turbine bypass.

Results

A turbine trip, assuming only 50% of the total turbine steam bypass capacity available, is simulated at rated power conditions as shown in Figure 2.3-7. Table 2.3-4 summarizes the analysis results. The neutron flux increases rapidly because of the void reduction caused by the pressure increase. However, the flux increase is limited by the partial actuation of the steam bypass system and the initiation of reactor scram. The peak simulated thermal heat flux does not significantly increase (< 10%) above its initial value. The number of rods in boiling transition during this event remains within the acceptance criterion for AOOs in combination with an additional single active component failure or operator error.

2.3.2.5.4 Barrier Performance

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

2.3.2.5.5 Radiological Consequences

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

2.3.2.6 Closure of One Main Steamline Isolation Valve

2.3.2.6.1 Identification of Causes

This discussion in the DCD applies to the initial core.

2.3.2.6.2 Sequence of Events and Systems Operation

When a single MSIV is closed in conformance with normal testing procedures, no reactor scram occurs and the reactor settles into a new steady state operating condition. Closure of a single MSIV at power levels above those of the normal testing procedure may cause closure of all other MSIVs.

Table 2.3-12 lists the sequence of events for Figure 2.3-8

2.3.2.6.3 Core and System Performance

The neutron flux increases slightly while the simulated thermal heat flux shows no increase. The number of rods in boiling transition during this event remains within the acceptance criterion for AOOs. The effects of closure of a single MSIV are considerably milder than the effects of closure of all MSIVs. Therefore, this event does not need to be reanalyzed for any specific core configuration.

Inadvertent closure of one MSIV while the reactor is shut down produces no significant transient. Closures during plant heatup are less severe than closure from maximum power cases.

2.3.2.6.4 Barrier Performance

Peak pressure at the vessel bottom remains below the pressure limits of the reactor coolant pressure boundary. Peak pressure in the main steamline remains below the SRV setpoints. Therefore, there is no steam discharged to the suppression pool.

2.3.2.6.5 Radiological Consequence

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

2.3.2.7 Closure of All Main Steamline Isolation Valves

2.3.2.7.1 Identification of Causes

This discussion in the DCD applies to the initial core.

2.3.2.7.2 Sequence of Events and Systems Operation

Sequence of Events

Table 2.3-13 lists the sequence of events for Figure 2.3-9.

The following is the sequence of operator actions expected during the course of the event, assuming no restart of the reactor. The operator should:

- Verify that all rods are inserted;
- Follow the scram procedure;
- Check that ICs have initiated (i.e., drain valves open);
- Monitor reactor water level and pressure;
- Initiate RWCU/SDC system operation in the shutdown cooling mode appropriate to hot shutdown;
- Determine the cause of valve closure before resetting the MSIV isolation;
- Observe turbine coastdown and break vacuum before the loss of sealing steam (check turbine auxiliaries for proper operation); and
- Check that conditions are satisfactory prior to opening and resetting MSIVs.

Systems Operation

MSIV closure initiates a reactor scram trip via position signals to the RPS. The same signal also initiates the operation of ICs, which prevent the lifting of SRVs.

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

2.3.2.7.3 Core and System Performance

Input Parameters and Initial Conditions

The MSIV design closure time range and the worst case (bounding) closure time assumed in this analysis are provided in Table 2.3-1.

Position switches on the valves initiate a reactor scram, as addressed in Table 2.3-1. Closure of these valves causes the dome pressure to increase.

Results

Figure 2.3-9 shows the changes in important nuclear system variations for the simultaneous isolation of all main steamlines while the reactor is operating at rated power. The neutron flux increases slightly while the simulated thermal heat flux shows no increase. The FW injection and the IC operation terminate the pressure increase. The anticipatory scram prevents any change in the thermal margins. The number of rods in boiling transition during this event remains within the acceptance criterion for AOOs. Therefore, this event does not have to be reanalyzed for any specific core configurations.

Inadvertent closure of all of the MSIVs while the reactor is shut down produces no significant transient. Closures during plant heatup are less severe than the maximum power cases (maximum stored and decay heat) presented.

2.3.2.7.4 ***Barrier Performance***

Peak pressure at the vessel bottom remains below the upset event pressure limit for the reactor coolant pressure boundary (RCPB). Peak pressure in the main steamline remains below the SRV setpoints. Therefore, there is no steam discharged to the suppression pool.

2.3.2.7.5 ***Radiological Consequence***

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

2.3.2.8 **Loss of Condenser Vacuum**

2.3.2.8.1 ***Identification of Causes***

This discussion in the DCD applies to the initial core.

2.3.2.8.2 ***Sequence of Events and Systems Operation***

Sequence of Events

Table 2.3-15 lists the sequence of events for Figure 2.3-10.

The Loss of Condenser Vacuum initially does not affect the vessel, when the turbine trip setpoint is reached it has a simultaneous scram with a bypass valve opening. Six seconds later (see Table 2.3-16), the low vacuum setpoint produces closure of the bypass valve with a small delay the MSIV also closes.

Identification of Operator Actions

The operator should:

- Verify that all rods are inserted;
- Follow the scram procedure;
- Monitor reactor water level and pressure;
- Check turbine for proper operation of all auxiliaries during coastdown;

- Use ICs to control reactor pressure;
- Depending on conditions, maintain pressure for restart purposes, or initiate normal operating procedures for cooldown;
- Put the mode switch in the STARTUP position before the reactor pressure decays below 6 MPa (870 psig).

Systems Operation

In establishing the expected sequence of events and simulating the plant performance, the plant instrumentation and controls, plant protection and reactor protection systems are assumed to normally function.

Tripping functions incurred by sensing main turbine condenser vacuum are presented in Table 2.3-16.

2.3.2.8.3 ***Core and System Performance***

Input Parameters and Initial Conditions

TSV full stroke closure time is as shown in Table 2.3-1.

A reactor scram is initiated on low condenser vacuum at the same time that the turbine trip signal is generated.

The analysis presented here is a hypothetical case with a conservative vacuum decay rate (see Table 2.3-1). Thus, the bypass system is available for several seconds, because the bypass is signaled to close at a vacuum level that is less than the stop valve closure (see Table 2.3-16).

Results

As shown in Table 2.3-15, under the analysis vacuum decay condition, the turbine bypass valves and MSIV closure would follow main turbine trip. This AOO is similar to a normal turbine trip with bypass. The effect of MSIV closure tends to be minimal, because the reactor scram on low condenser vacuum precedes the isolation by several seconds. Figure 2.3-10 shows the transient expected for this event. It is assumed that the plant is initially operating at rated power conditions. Peak neutron flux is shown in Table 2.3-4, and the average simulated thermal heat flux peaks at < 110% of rated. The number of rods in boiling transition during this event remains within the acceptance criterion for AOOs.

2.3.2.8.4 ***Barrier Performance***

Peak nuclear system pressure remains below the ASME code upset limit. Peak pressure in the main steamline remains below the SRV setpoints. Therefore, there is no steam discharged to the suppression pool. A comparison of these values to those for turbine trip at high power shows the similarities between these two transients. The prime difference is the subsequent main steamline isolation.

2.3.2.8.5 ***Radiological Consequences***

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

2.3.2.9 Loss of Shutdown Cooling Function of RWCU/SDC

Analysis in the DCD applies to the initial core.

2.3.3 Reactivity and Power Distribution Anomalies

This discussion in the DCD applies to the initial core.

2.3.4 Increase in Reactor Coolant Inventory

2.3.4.1 Inadvertent Isolation Condenser Initiation

2.3.4.1.1 Identification of Causes

Manual startup of the four individual IC systems is postulated for this analysis (i.e., operator error).

2.3.4.1.2 Sequence of Events and System Operation

Sequence of Events

Table 2.3-17 lists the sequence of events for Inadvertent Isolation Condenser Initiation.

Identification of Operator Actions

Relatively small changes in plant conditions are experienced. The operator should, after hearing the alarm that the IC system has commenced operation, check reactor water level, reactor pressure and MSIV status. If conditions are normal, the operator should shut down the system.

System Operation

To properly simulate the expected sequence of events, the analysis of this event assumes normal functioning of plant instrumentation and controls. Specifically, the pressure regulation and the vessel level control that respond directly to this event.

Required operation of engineered safeguards other than what is described is not expected for this event.

2.3.4.1.3 Core and System Performance

Input Parameter and Initial Conditions

The assumed IC system water temperature and enthalpy, startup time and flow rate are provided in Table 2.3-1.

Inadvertent startup of all loops of the IC system was chosen as the limiting case for analysis because it provides the greatest auxiliary source of cold water into the vessel.

Results

Figure 2.3-11 shows the simulated transient event. It begins with the introduction of cold water into the downcomer region. Full IC loop flow is established. No delays are considered because these are not relevant to the analysis.

Addition of cooler water to the downcomer causes a reduction in inlet enthalpy, which results in a power increase. The flux level settles out slightly above its operating level. The variations in the pressure and thermal conditions are relatively small, and no significant effect is experienced.

The number of rods in boiling transition remains within the acceptance criterion for AOOs, and the fuel thermal margins are maintained.

Consideration of Uncertainties

Important analytical factors, including IC loop condensate water temperature, are assumed to be at the worst conditions so that any deviations in the actual plant parameters would produce a less severe transient. The IC volume is considered to be at the nominal value, however the possibility of a larger volume is compensated by the low initial temperature and the bounding drainage flow rate assumed into the vessel.

2.3.4.1.4 ***Barrier Performance***

Inadvertent Startup of the IC causes only a slight pressure decrease from the initial conditions; therefore, no further RCPB pressure response evaluation is required.

2.3.4.1.5 ***Radiological Consequences***

Because no activity is released during this event, a detailed evaluation is not required.

2.3.4.2 **Runout of One Feedwater Pump**

2.3.4.2.1 ***Identification of Causes***

This discussion in the DCD applies to the initial core.

2.3.4.2.2 ***Sequence of Events and Systems Operation***

Sequence of Events

With momentary increase in FW flow, the water level rises and then settles back to its normal level. Table 2.3-19 lists the sequence of events for Figure 2.3-13.

Identification of Operator Actions

Because no scram occurs for runout of one FW pump, 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.

Systems Operation

Runout of a single FW pump requires no protection system or ESF system operation. This analysis assumes normal functioning of plant instrumentation and controls.

2.3.4.2.3 ***Core and System Performance***

Input Parameters and Initial Conditions

The total FW flow for all pumps runout is provided in Table 2.3-1.

Results

The simulated runout of one FW pump event is presented in Figure 2.3-13. When the increase of FW flow is sensed, the FW controller starts to command the remaining FW pumps to reduce its flow immediately. The vessel water level increases slightly [about 15 cm (6 inch)] and then settles back to its normal level. Vessel pressure increases insignificantly, and the number of rods in boiling transition remains within the acceptance criterion for AOOs.

2.3.4.2.4 Barrier Performance

As previously noted, the effect of this event does not result in any temperature or pressure transient in excess of the design criteria for the fuel, pressure vessel or containment. Therefore, these barriers maintain their integrity and function as designed.

2.3.4.2.5 Radiological Consequences

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

2.3.5 Decrease in Reactor Coolant Inventory

2.3.5.1 Opening of One Turbine Control or Bypass Valve.

2.3.5.1.1 Identification of Causes

This discussion in the DCD applies to the initial core.

2.3.5.1.2 Sequence of Events and Systems Operation

The SB&PC system senses the pressure change and commands the remaining control valves to close, and thereby automatically mitigate the transient and maintain reactor power and pressure.

Table 2.3-20 lists the sequence of events for Figure 2.3-14

2.3.5.1.3 Core and System Performance

Reactor power and pressure is maintained. Reactor scram does not occur.

2.3.5.1.4 Barrier Performance

The effects of this event do not result in any temperature or pressure transient in excess of the design criteria for fuel, pressure vessel or containment. The peak pressure in the bottom of the vessel remains below the ASME code upset limit. Peak steam line pressure near the SRVs remains below the setpoint of the SRVs.

2.3.5.1.5 Radiological Consequences

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

2.3.5.2 Loss of Non-Emergency AC Power to Station Auxiliaries

This event bounds the Loss of Unit Auxiliary Transformer and Loss of Grid Connection events.

2.3.5.2.1 Identification of Causes

This discussion in the DCD applies to the initial core.

2.3.5.2.2 Sequence of Events and Systems Operation

Sequence of Events

For the Loss of Unit Auxiliary Power Transformer, Table 2.3-21 lists the sequence of events for Figure 2.3-15.

Identification of Operator Actions

The operator should maintain the reactor water level by use of the IC system and Control Rod Drive system and control reactor pressure using the ICS and RWCU/SDC system. Verify that the turbine and generator DC oil pumps are operating satisfactorily to prevent turbine bearing damage. Also verify proper switching and loading of the standby diesel generators.

The following is the sequence of operator actions expected during the course of the events when no immediate restart is assumed. The operator should:

- Verify that all rods are inserted;
- Follow the scram procedure;
- Check that diesel generators start and carry their assigned loads;
- Monitor reactor water level and pressure; verify that Control Rod Drive flow is controlling water level;
- Use IC system to control pressure;
- Check turbine for proper operation of all auxiliaries during coastdown;
- Put the mode switch in the STARTUP position before the reactor pressure decays below 6 MPa (870 psig);
- Secure the IC when both reactor pressure and level are under control;

Systems Operation

This event, unless otherwise stated, assumes and takes credit for normal functioning of plant instrumentation and controls, plant protection and reactor protection systems.

The reactor is subjected to a complex sequence of events when the plant loses all auxiliary power. Estimates of the responses of the various reactor systems provide the simulation sequence shown in Table 2.3-21.

2.3.5.2.3 Core and System Performance

Figure 2.3-15 shows graphically the simulated transient. The initial water level is assumed to be at the L4 level. The initial portion of the transient is similar to the load rejection transient. At 2s the loss of the power generation busses signal produces a Scram and activation of the ICs. The load rejection initiation of the SCRRI/SRI function is not credited. At approximately 6 seconds the turbine bypass valves are assumed no longer available to bypass the steam to the main condenser. The MSIV closure is produced at 14s due to low condenser vacuum signal. The CRD high pressure injection is initiated due to low water level (Level 2), but the HP_CRD flow is delayed until diesel power is available (145 seconds). In the case where HP_CRD is unavailable for level control, the system response is similar to the station blackout event described in Subsection 2.5.5 which demonstrates that level can be maintained above the top of active fuel with the ICS as the primary success path. In either case, there is no significant increase in fuel temperature. The number of rods in boiling transition remains within the acceptance criterion for AOOs. Hence, fuel thermal margins are not threatened and the design basis is satisfied. Consequently, this event does not need to be reanalyzed for specific core configurations.

2.3.5.2.4 Barrier Performance

Peak nuclear system pressure at the vessel bottom remains below the ASME code upset limit. Peak pressure in the main steamline remains below the SRV setpoints. Therefore, there is no steam discharged to the suppression pool.

2.3.5.2.5 Radiological Consequences

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

2.3.5.3 Loss of All Feedwater Flow

2.3.5.3.1 Identification of Causes

This discussion in the DCD applies to the initial core.

2.3.5.3.2 Sequence of Events and Systems Operation

Sequence of Events

Table 2.3-22 lists the sequence of events for Figure 2.3-16.

Identification of Operator Actions

The operator should ensure ICS actuation and CRD injection transfer to high pressure injection mode so that water inventory is maintained in the reactor vessel. The operator should also monitor reactor water level, the pressure control, and the TG auxiliaries during shutdown.

The following is the sequence of operator actions expected during the course of the event when no immediate restart is assumed. The operator should:

- Verify that all rods are inserted;
- Follow the scram procedure;
- Monitor reactor water level and pressure; verify that CRD flow is controlling water level;
- Verify IC system initiation; use the IC system to control pressure;
- Monitor turbine coastdown and break vacuum before the loss of steam seals (check turbine auxiliaries for proper operation);
- When desired, the RWCU/SDC system can be put into service.

Systems Operation

Loss of FW flow results in a reduction of vessel inventory, causing the vessel water level to drop. The first corrective action is the loss of the power generation busses scram trip actuation. This scram trip function meets the single-failure criterion.

2.3.5.3.3 Core and System Performance

The results of this transient simulation are presented in Figure 2.3-16. The initial water level is assumed at the L4 level. Feedwater flow terminates, and the loss of the power generation busses scram signal is assumed (with activation of the ICS simultaneously). Subcooling decreases, causing a reduction in core power level and pressure. As the core power level is reduced, the turbine steam flow starts to drop off because of the action of the pressure regulator in attempting to maintain pressure. Water level continues to drop, and the vessel level (Level 3) scram trip setpoint is reached. Note that the reactor has been scrammed previously. The vessel water level continues to drop to Level 2. At that time, CRD high pressure injection and closure of all MSIVs are produced (with 30s delay). In case HP_CRD is unavailable for level control, the system response is similar to the station blackout event described in Subsection 2.5.5 which demonstrates that level can be maintained above the top of active fuel with the ICS as the primary success path. In either case, the number of rods in boiling transition remains within the acceptance criterion for AOOs because increases in the heat flux are not experienced.

2.3.5.3.4 ***Barrier Performance***

The consequences of this event do not result in any temperature or pressure transient in excess of the design criteria for the fuel, pressure vessel or containment. Therefore, these barriers maintain their integrity and function as designed.

2.3.5.3.5 ***Radiological Consequences***

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

2.3.6 AOO Analysis Summary

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

For the core loading in Reference 2.3-3, the calculated OLMCPR is 1.28, using the methodologies listed in Subsections 4.4.3.1.3 and 4.4.2.1.3 and Reference 2.3-1. The calculated OLMCPR is in the range considered in the initial core design (Reference 2.3-3).

2.3.7 References

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

2.3-2 Deleted

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

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

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

Table 2.3-1

Input Parameters And Initial Conditions and Assumptions Used In AOO and Infrequent Event Analyses

Parameter	Value
Thermal Power Level, MWt	4500
Total Flow For All Pumps Runout, % of rated at 1065 psig (At rated dome pressure, 1025 psig). The condensate and feedwater system in conjunction with the feedwater control system provide inventory equivalent to 240 s of rated feedwater flow after MSIV isolation. The condensate and feedwater system in combination with feedwater control system limit the maximum feedwater flow for a single pump to 75% of rated flow following a single active component failure or operator error.	155 (164)
Feedwater Temperature, °C (°F) Rated FW Heating Temperature Loss	 216 (420) 55.6 (100)
Vessel Dome Pressure, MPaG (psig)	7.07 (1025)
Turbine Bypass Capacity, % of rated	110
Total Delay Time from TSV or TCV motion to the start of BPV Main Disc Motion	0.02
Total Delay Time from TSV or TCV motion to 80% of Bypass Capacity	0.17
TCV Closure Times, seconds Fast Closure Analysis Value (Bounding) Assumed Slow Closure Analysis Value	 0.08 2.5
TSV Closure Times, seconds	0.100
% of Rated Steam Flow That Can Pass Through 3 Turbine Control Valves	85 (Partial Arc)

Table 2.3-1

Input Parameters And Initial Conditions and Assumptions Used In AOO and Infrequent Event Analyses

Parameter	Value
Turbine Inlet Pressure, MPaG (psig)	6.57
Fuel Lattice	N
Control Rod Drive Position versus Time	Table 2.3-2 & 3
Core Design used in TRACG Simulations Exposure:	Reference 2.3-2 Middle of Cycle and End of Cycle
Analysis values for SRV setpoints Low Setpoint, MPaG (psig) High Setpoint, MPaG (psig)	 8.618 (1250) 8.756 (1270)
Closure Scram Position of 2 or More MSIVs, % open Maximum delay time	85 0.06
MSIV Minimum Closure Time, s MSIV Maximum Closure Time, s	3.0 5.0
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	0.08
High Pressure Scram, MPaG (psig). Maximum scram delay	7.619 (1105) 0.7
High Suppression Pool Temperature Scram trip, °C (°F), Maximum Delay Time	48.9(120) 1.05
High Suppression Pool Temperature FAPCS actuation, °C (°F)	43.3 (110)

Table 2.3-1

Input Parameters And Initial Conditions and Assumptions Used In AOO and Infrequent Event Analyses

Parameter	Value
Vessel level Trips (above bottom vessel)	
Level 9—(L9), m (in)	22.39 (881.5)
Level 8—(L8), m (in)	21.89 (861.8)
Level 4—(L4), m (in)	20.60 (811.2)
Level 3—(L3), m (in)	19.78 (778.7)
Level 2—(L2), m (in)	16.05 (631.9)
Level 1—(L1), m (in)	11.50 (452.8)
Level 0.5 – (L0.5) m (in)	8.45 (332.7)
APRM Simulated Thermal Power Trip	
Scram, % NBR	115
Time Constant, s	7
Total Steamline Volume, m^3 (ft^3)	135 (4767)
CRD Hydraulic System minimum capacity, m^3/hr (gpm),	235.1 (1035)
Capacity in kg/s (Mlbm/hr) for 990 kg/m^3 density (61.8 lbm/ft^3)	64.6 (0.513)
Maximum time delay from Initiating Signal (Pump 1 & 2), s	10 & 25
If offsite power is not available	145
Isolation Condensers	
Max Initial Temperature, °C (°F)	40 (104)
Minimum Initial Temperature, °C (°F)	10 (50)
Time To injection valve full open (Max), s(1)	31 (1)
Heat Removal Capacity for 4ICs, MW (% Rated Power)	135 (3%)
Isolation Condensers volume, 4 Units, from steam box to discharge at vessel m^3 (ft^3)	56.1

(1) 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.

Table 2.3-2

CRD Scram Times for Vessel Bottom Pressures Below 7.481 MPa gauge (1085 psig)

Data in DCD applies to the initial core.

Table 2.3-3

CRD Scram Times for Bottom Vessel Pressures Between 7.481 MPa gauge (1085 psig) and 8.618 MPa gauge (1250 psig)

Data in DCD applies to the initial core.

Table 2.3-4

Results Summary of Anticipated Operational Occurrence Events

Sub-section I.D.	Description	Max. Neutron Flux, % NBR	Max. Dome Pressure, MPaG (psig)	Max. Vessel Bottom Pressure, MPaG (psig)	Max. Steamline Pressure, MPaG (psig)	Max. Core Average Surface Heat Flux, % of Initial	ΔCPR/ICPR or Minimum Water Level (m over TAF)
2.3.2.1	Loss of Feedwater Heating	116	7.11 (1031)	7.24 (1050)	7.06 (1024)	119	0.09
2.3.2.1	Closure of One Turbine Control Valve. FAST/SLOW	125 110	7.20 (1043) 7.20 (1043)	7.33 (1063) 7.33 (1063)	7.16 (1038) 7.16 (1038)	102 102	0.04 0.03
2.3.2.2	Generator Load Rejection with Turbine Bypass	128	7.15 (1037)	7.29 (1057)	7.28 (1056)	101	0.07
2.3.2.3	Generator Load Rejection with a Single Failure in the Turbine Bypass System	151	7.37 (1070)	7.50 (1088)	7.37 (1069)	102	0.02
2.3.2.4	Turbine Trip with Turbine Bypass	116	7.12 (1033)	7.26 (1053)	7.20 (1043)	101	0.07
2.3.2.5	Turbine Trip with a Single Failure in the Turbine Bypass System	131	7.34 (1065)	7.48 (1085)	7.34 (1065)	101	0.01
2.3.2.6	Closure of One MSIV	114	7.16 (1038)	7.30 (1059)	7.13 (1033)	102	0.03
2.3.2.7	Closure of All MSIV	102	7.67 (1112)	7.80 (1131)	7.67 (1112)	100	≤0.01
2.3.2.8	Loss of Condenser Vacuum	107	7.12 (1032)	7.26 (1053)	7.20 (1044)	100	≤0.01
2.3.4.1	Inadvertent Isolation Condenser Initiation	111	7.08 (1027)	7.22 (1047)	7.04 (1021)	109	0.09
2.3.4.2	Runout of One Feedwater Pump	103	7.08 (1027)	7.22 (1047)	7.04 (1021)	101	≤0.01
2.3.5.1	Opening of One Turbine Control or Bypass Valve	101	7.08 (1027)	7.21 (1046)	7.04 (1021)	100	≤0.01
2.3.5.2	Loss of Non-Emergency AC Power to Station Auxiliaries	139	7.13 (1035)	7.28 (1056)	7.28 (1056)	102	5.37m
2.3.5.3	Loss of Feedwater Flow	100	7.08 (1027)	7.21 (1046)	7.04 (1021)	100	5.28m

Table 2.3-5

Sequence of Events for Loss of Feedwater Heating

Time (s)	Event
0	Initiate a 39°C (70°F) temperature reduction in the FW system
22 (est)	RC&IS initiates Selected Rod Insertion plus Selected Control Rod Run-In (SCRRI/SRI). No rods are assigned for this exposure point
25 (est.)	Initial effect of unheated FW starts to raise core power level
≈150.0	The STPT setpoint (115%) is reached, the activation of Scram is not credited
300 (est.)	Reactor variables settle into new steady state

* See Figure 2.3-1.

Table 2.3-6

Sequence of Events for Fast Closure of One Turbine Control Valve

Time (sec)	**Event***
0	Simulate one main TCV to fast close
0	Failed TCV starts to close
0.08	TCV closed
1.10	Turbine bypass valves start to open
30.0	New steady state is established

* See Figure 2.3-2.

Table 2.3-7

Sequence of Events for Slow Closure of One Turbine Control Valve

Time (sec)	**Event***
0	Simulate one main TCV to slow close
0	Failed TCV starts to close
2.5	TCV closed
2.8	Turbine bypass valves start to open
30.0	New steady state is established

* See Figure 2.3-3.

Table 2.3-8

Sequence of Events for 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
0.08	Turbine control valves closed
0.17	Turbine bypass opened at 80%
0.20	SRI/SCRRI activated (no SCRRI rods assigned)
1.0	First SRI group inserts (one HCU, 2 control rods, fails to actuate)
10.0	Second SRI group inserts
15.0	Third SRI group inserts
25.0	Fourth SRI group inserts
30.0	Fifth SRI group inserts
35.0	Sixth SRI group inserts
≈40.0	Steam flow below 60% of rated
≈70.0	Core power reaches 60%
0.0-400.0	FW temperature is decreasing because of loss of turbine extraction steam to FW heaters
400	New steady state is established

Table 2.3-9

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
0.026	Turbine bypass valves start to open (Half fail to open)
0.08	Turbine control valves closed
0.20	Not enough turbine bypass availability is detected and the plant is scrammed
0.45	Control Rods begin to enter in the core
2.8	L3 level is reached
Long term	L2 is not reached, new steady state

* See Figure 2.3-5.

Table 2.3-10

Sequence of Events for Turbine Trip with Turbine Bypass

Time (sec)	**Event** *
0.0	Turbine trip initiates closure of main stop valves
0.0	Turbine trip initiates bypass operation
0.02	Turbine bypass valves start to open to regulate pressure
0.10	Turbine stop valves closed
0.17	Turbine bypass opened at 80%
0.20	SRI/SCRRI activated (no SCRRI rods assigned)
1.0	First SRI group inserts (one HCU, 2 control rods, fails to actuate)
10.0	Second SRI group inserts
15.0	Third SRI group inserts
25.0	Fourth SRI group inserts
30.0	Fifth SRI group inserts
35.0	Sixth SRI group inserts
≈40.0	Steam flow below 60% of rated
≈70.0	Core power reaches 60%
0.0-400.0	FW temperature is decreasing because of reduced turbine steam flow
400	New steady state is established

Table 2.3-11

Sequence of Events for Turbine Trip with a Single Failure in the Turbine Bypass System

Time (sec)	**Event** *
0.0	Turbine trip initiates closure of main stop valves
0.0	Turbine trip initiates bypass operation
0.02	Turbine bypass valves start to open to regulate pressure (Half fail to open)
0.1	Turbine stop valves closed
0.20	Not enough turbine bypass availability is detected and the plant is scrammed
0.45	Control Rods begin to enter in the core
2.8	L3 level is reached
Long term	L2 is not reached

* See Figure 2.3-7.

Table 2.3-12

Sequence of Events for Closure of one MSIV

Time (sec)	**Event** *
0.0	Closure of one MSIV
2.0	Maximum neutron flux
2.8	Turbine Bypass open
3.0	MSIV is closed
40.0	New steady state is reached

* See Figure 2.3-8.

Table 2.3-13

Sequence of Events for Closure of all MSIV

Time (sec)	**Event** *
0.0	Closure of all MSIVs (MSIV)
0.77	MSIVs reach 85% open
0.84	MSIVs position trip scram initiated
1.81	IC initiated
2.83	L3 is reached
3.0	MSIVs are closed
4.10	Reactor pressure reaches its peak value
18.1	L2 is reached
28.3	HP_CRD is initiated
31.82	The ICs valves are fully open
Long term	The FW is available for a period of time to control the Water Level in vessel

* See Figure 2.3-9.

Table 2.3-14

Typical Rates of Decay for Loss of Condenser Vacuum

Data in DCD applies to the initial core.

Table 2.3-15

Sequence of Events for Loss of Condenser Vacuum

Time (sec)	**Event** *
-3.0	Initiate simulated loss of condenser vacuum trip.
0.0	Low condenser vacuum main turbine trip and Scram actuated
0.02	Turbine bypass valves start to open to regulate pressure
0.1	Turbine stop valves close
0.20	Scram initiated
2.1	Turbine Bypass initiates closure, because of low vessel pressure
6.0	Low condenser vacuum forces main turbine bypass valve closure
6.3	Bypass valve is closed
8.0	Low condenser vacuum initiates MSIV closure
8.8	MSIV closure initiates IC (85% position).
9.8	IC is activated (1 s delay from MSIV closure signal)
14.7	L2 water level is reached.
24.9	HP_CRD is activated
39.8	The ICs valves are fully open
Long term	The FW is available for a period of time to control the Water Level in vessel.

* See Figure 2.3-10.

Table 2.3-16

Trip Signals Associated With Loss of Condenser Vacuum

Data in DCD applies to the initial core.

Table 2.3-17

Sequence of Events for Inadvertent Isolation Condenser Initiation

Time (sec)	**Event** *
10	Simulate IC cold water injection
25	IC drainage valve begins to open
32.5	IC drainage valve is fully open
≈50	Full power established for IC
≈200	MCPR is recovered
≈300	Power increase effect stabilized

* See Figure 2.3-11.

Table 2.3-18

Single Failure Modes for Digital Controls

Data in the DCD applies to the initial core.

Table 2.3-19

Sequence of Events for Runout of One Feedwater Pump

Time (sec)	**Events** *
0	Initiate simulated runout of one FW pump (at system design pressure) the pump runout flow is 75% of rated FW flow)
~0.1	Feedwater controller starts to reduce the FW flow from the FW pumps
6.0	Vessel water level reaches its peak value and starts to return to its normal value
≈21.0 (est.)	Vessel water level returns to its normal value

* See Figure 2.3-13.

Table 2.3-20

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

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

* See Figure 2.3-14.

Table 2.3-21

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 [1]	Loss of power on the four power generation busses is detected and initiates a reactor scram and activation of ICs with 1s delay
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
6.2	Vessel water level reaches Level 3
10.0	Vessel water level reaches Level 2
14.0	Low-Low condenser vacuum signal closes the MSIVs
18.0	ICs begins to drop cold water inside the vessel
33.0	ICs drainage valve is fully open
145.0	HP_CRD injection mode is initiated
≈100	The level recovers above 13m (42.7 ft)
≈600	The level recovers above 15m (49.2 ft)

* See Figure 2.3-15. This Figure has 50s of steady state to change the initial water level to L4, a time of 0 s on the table corresponds to 50 sec on the figure.

[1] Conservatively the insertion of the first SRI previous to the SCRAM is not credited.

Table 2.3-22

Sequence of Events for Loss of All Feedwater Flow

Time (sec)	**Event ***
0	Trip of all FW pumps initiated
2.0	Non FW flow availability initiates reactor scram and initiates IC with 1s delay
5.0	Feedwater flow decays to zero
10.1	Vessel water level reaches Level 2
18.0	ICs begins to drop cold water inside the vessel
20.3	HP_CRD injection mode is initiated
33.0	The ICs drainage valves are fully open
40.2	MSIV closure
≈120	The level recovers above 13m (42.7 ft)
≈580	The level recovers above 15m (49.2 ft)

* See Figure 2.3-16. This Figure has 50 s of steady state to change the initial water level to L4 , a time of 0 s on the table corresponds to 50 s on the figure.

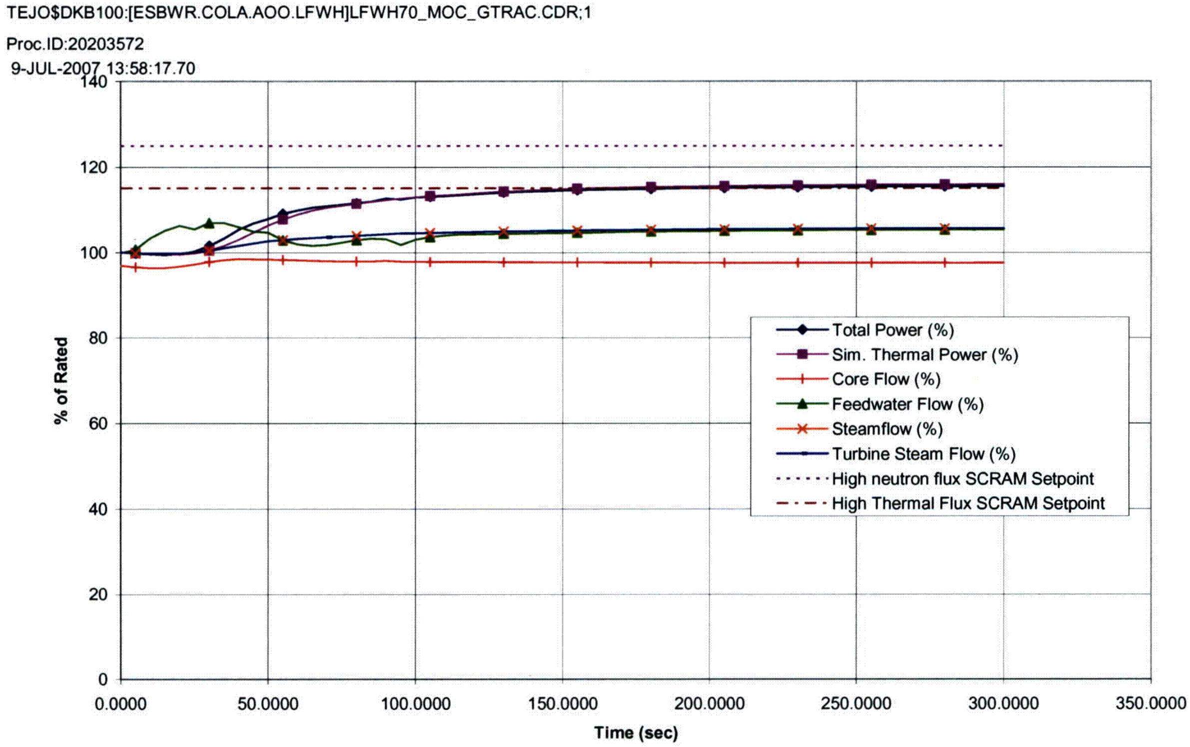


Figure 2.3-1a. Loss of Feedwater Heating

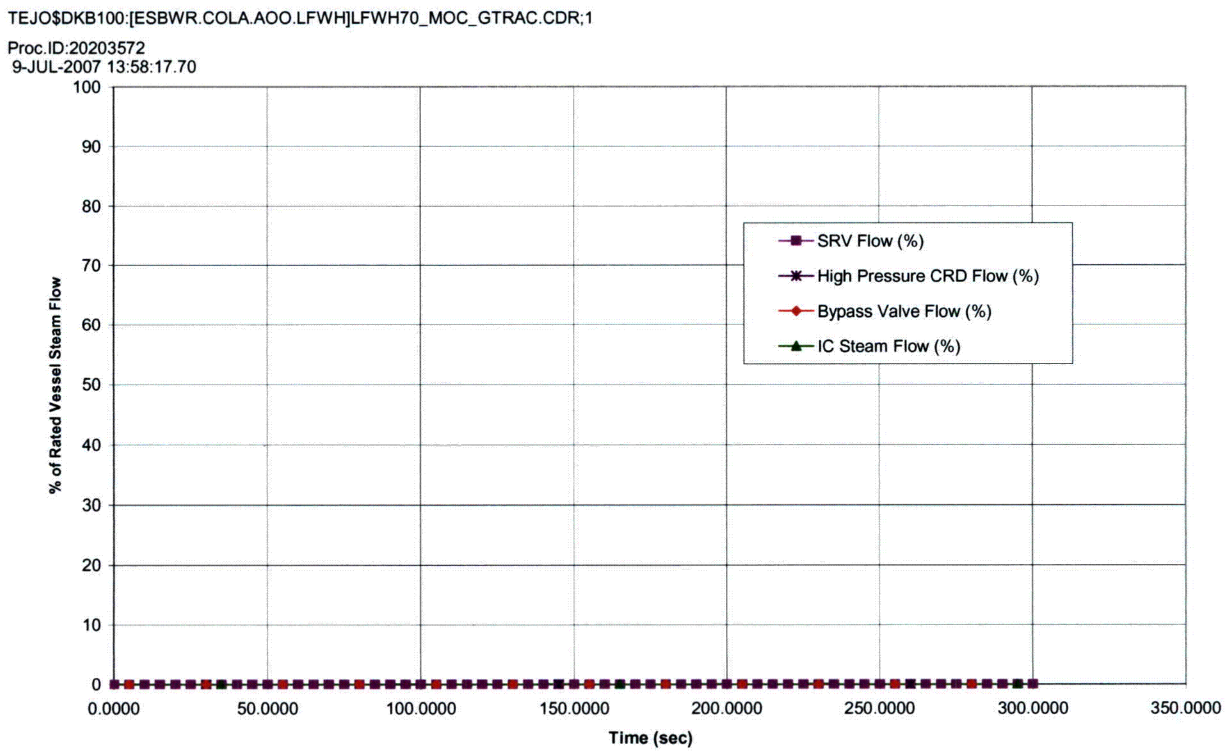


Figure 2.3-1b. Loss of Feedwater Heating

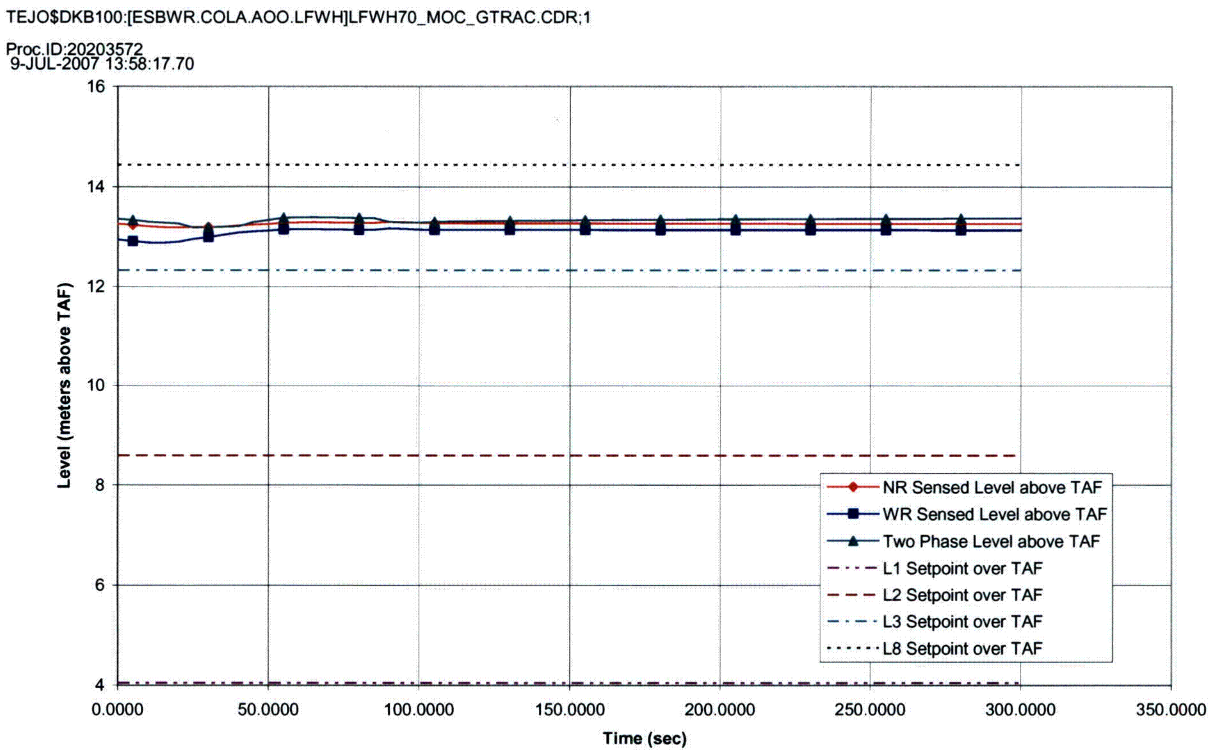


Figure 2.3-1c. Loss of Feedwater Heating

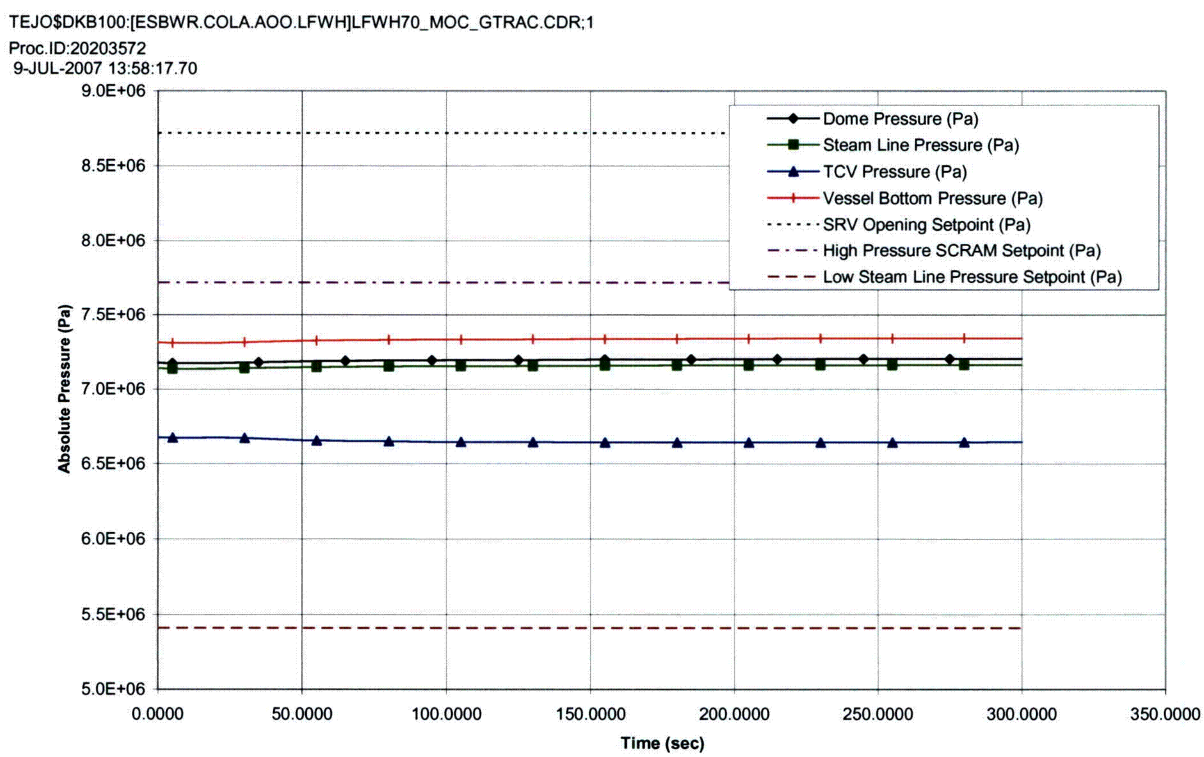


Figure 2.3-1d. Loss of Feedwater Heating

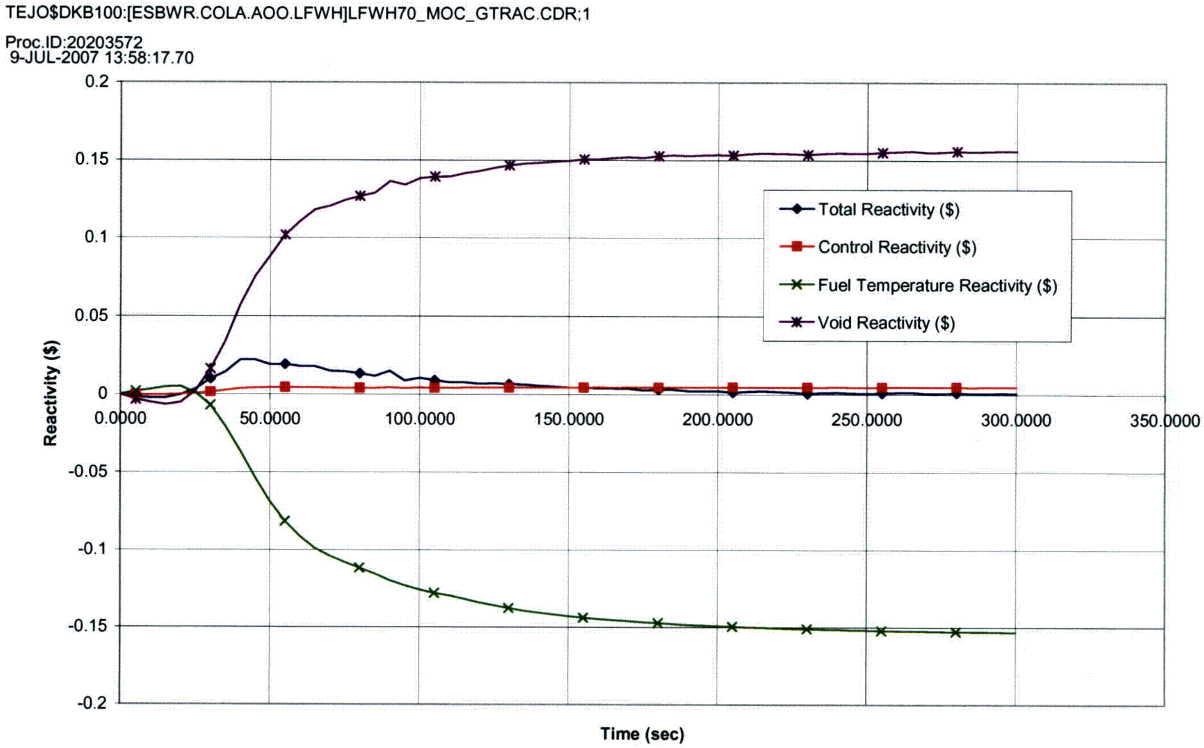


Figure 2.3-1e. Loss of Feedwater Heating

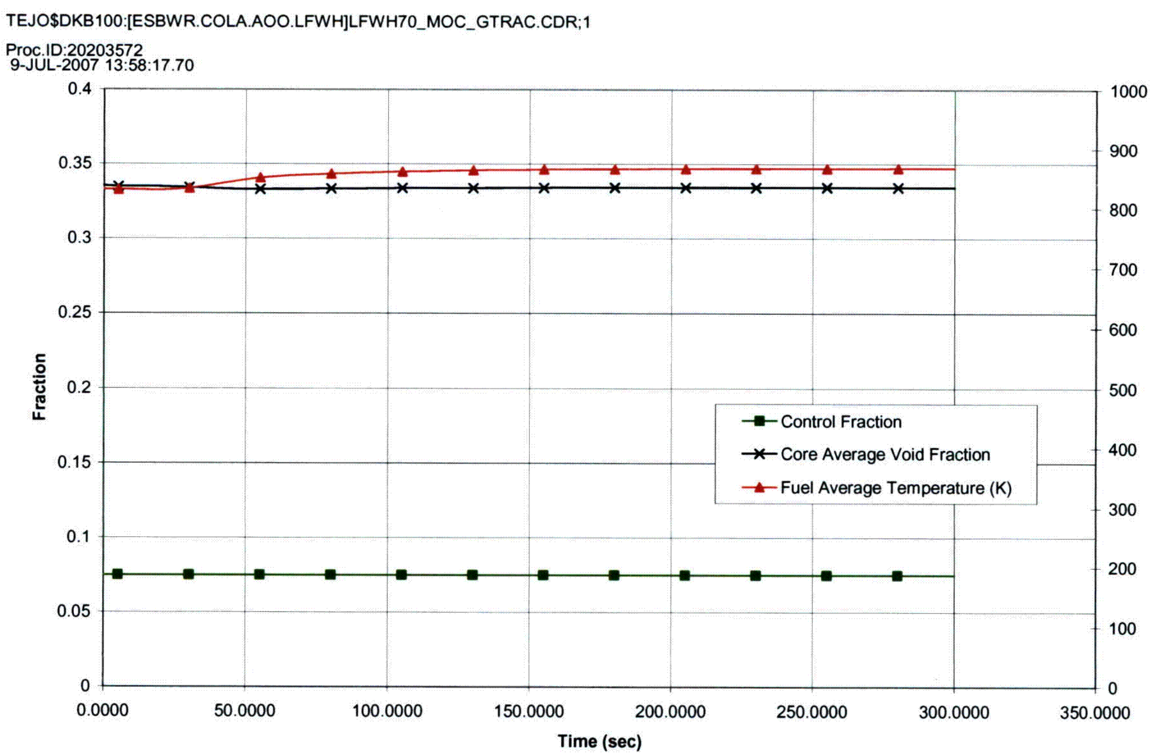


Figure 2.3-1f. Loss of Feedwater Heating

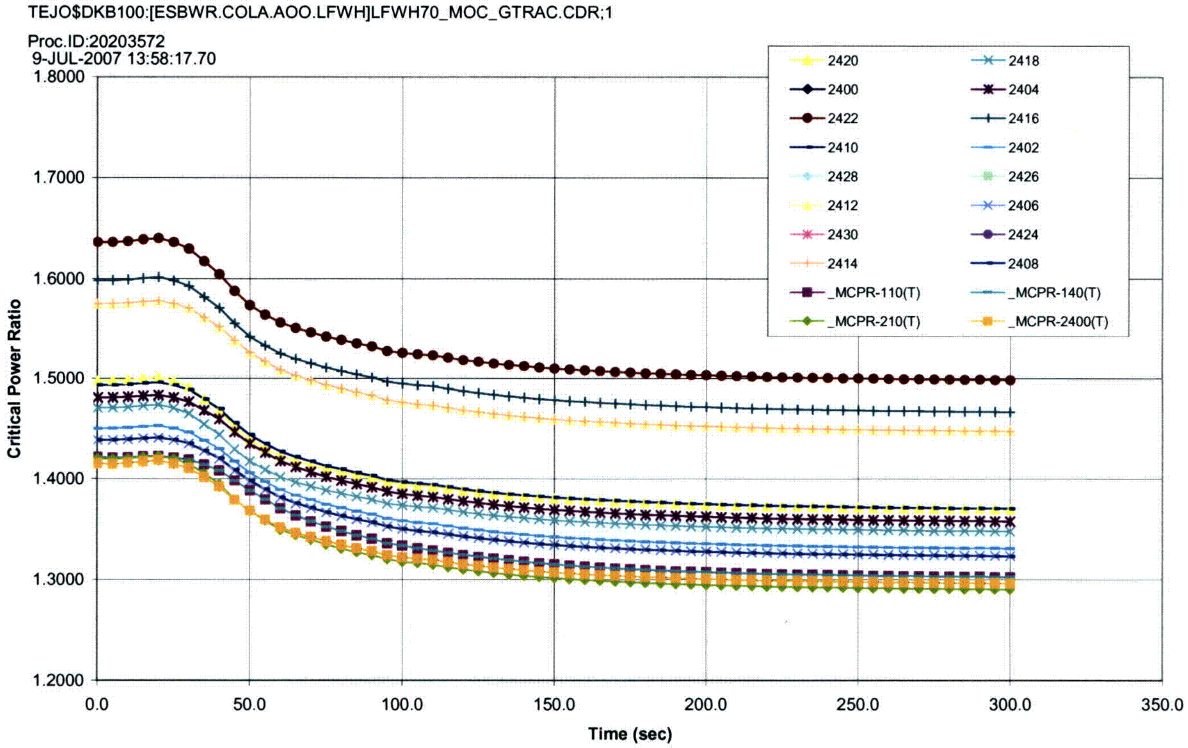


Figure 2.3-1g. Loss of Feedwater Heating

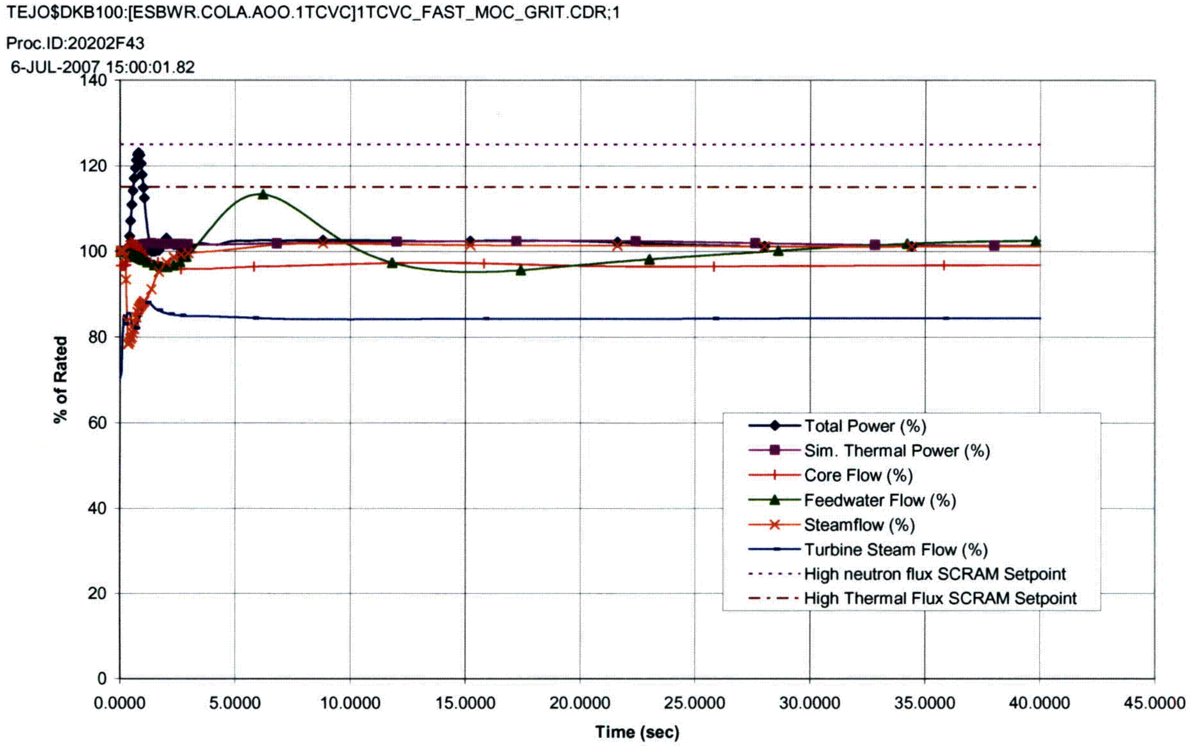


Figure 2.3-2a. Fast Closure of One Turbine Control Valve

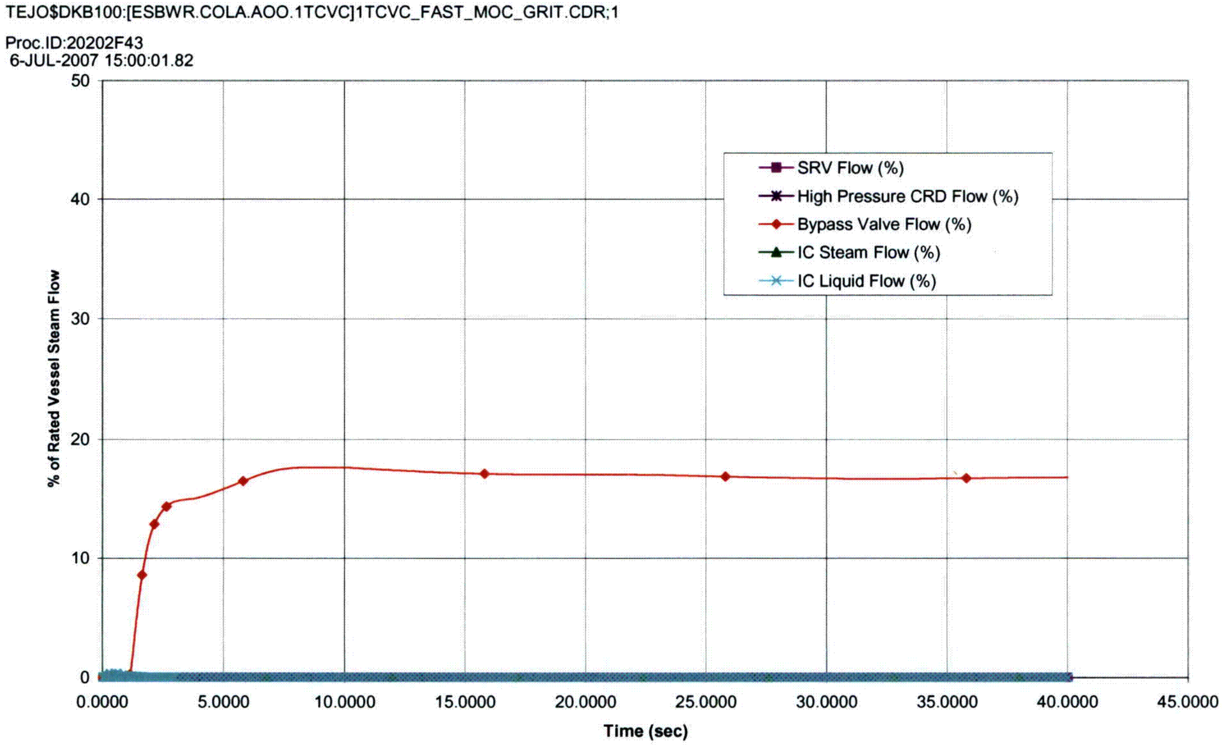


Figure 2.3-2b. Fast Closure of One Turbine Control Valve

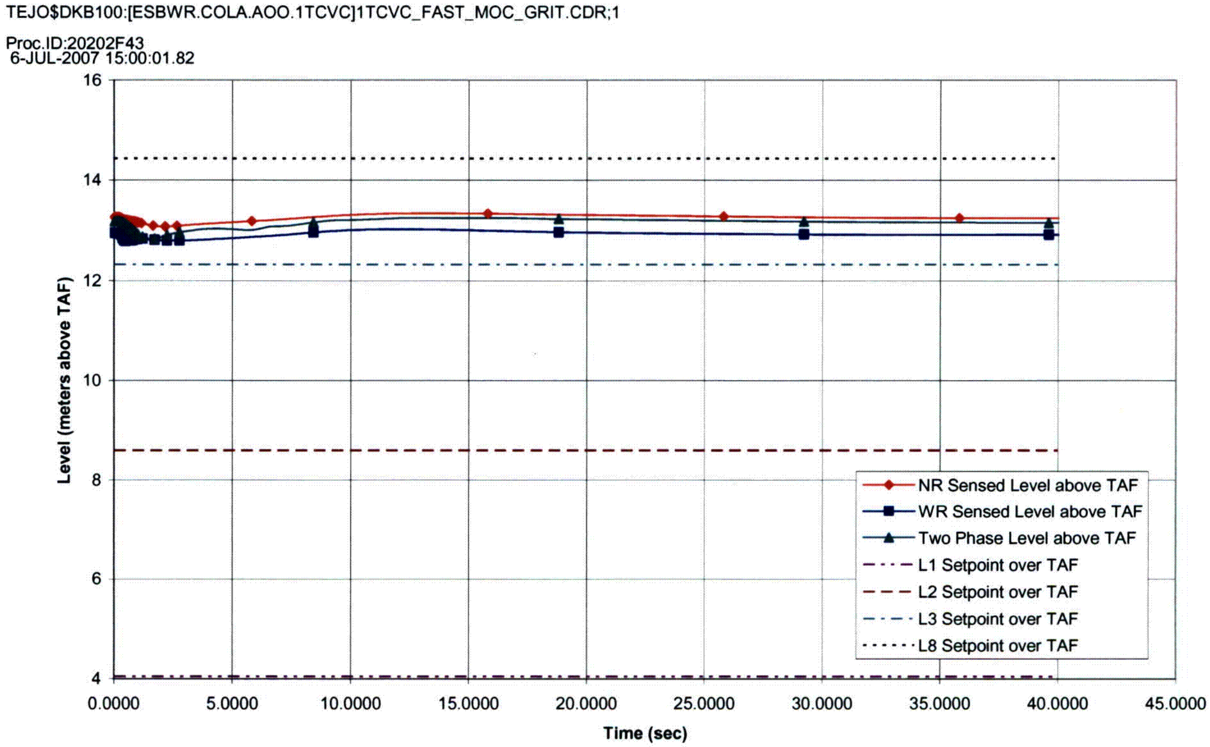


Figure 2.3-2c. Fast Closure of One Turbine Control Valve

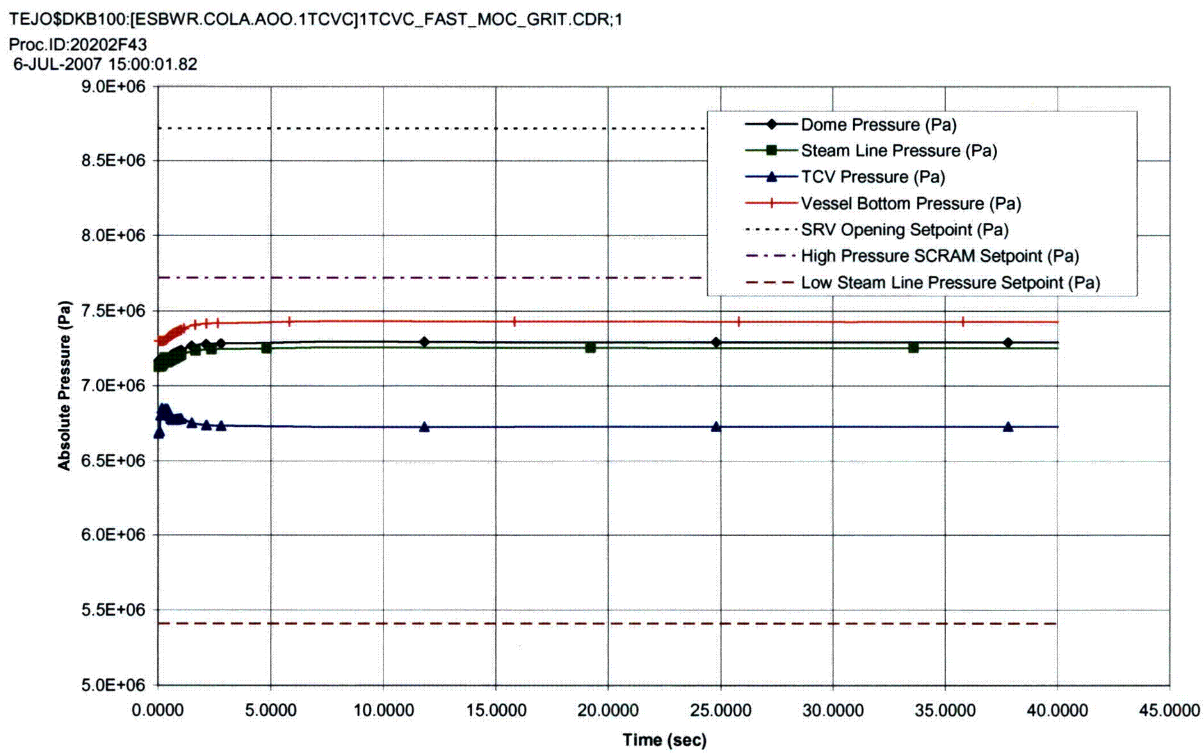


Figure 2.3-2d. Fast Closure of One Turbine Control Valve

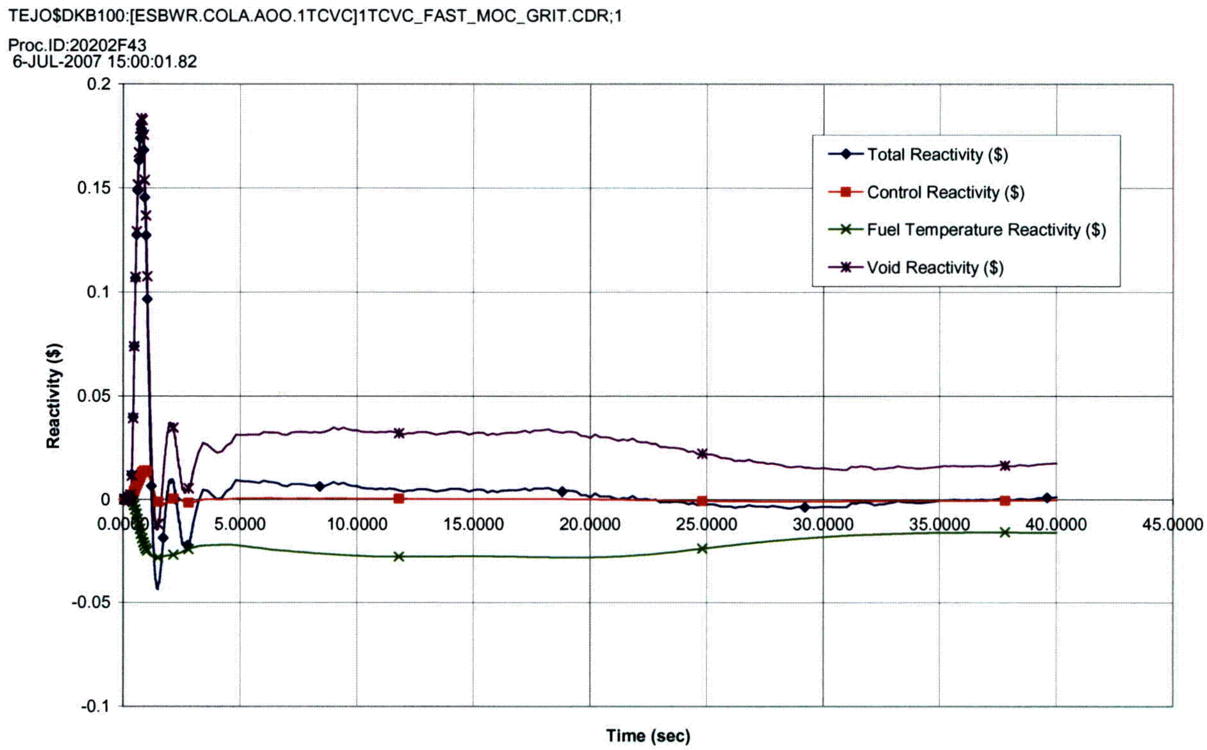


Figure 2.3-2e. Fast Closure of One Turbine Control Valve

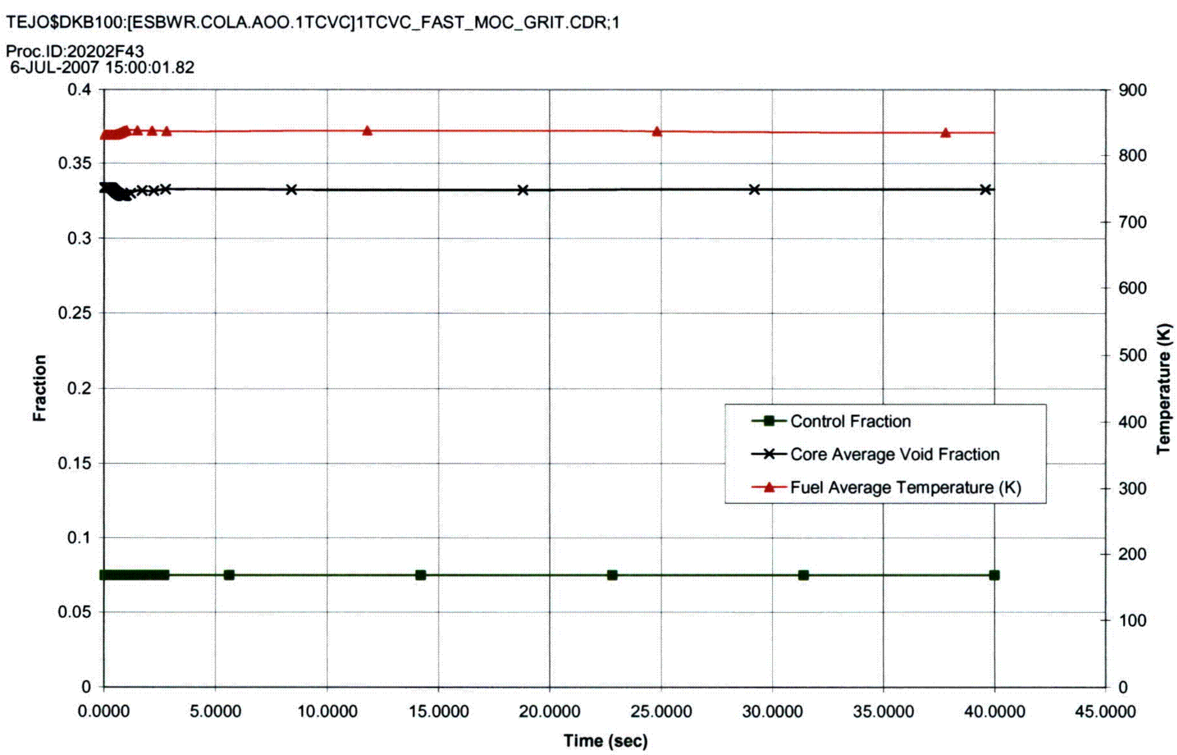


Figure 2.3-2f. Fast Closure of One Turbine Control Valve

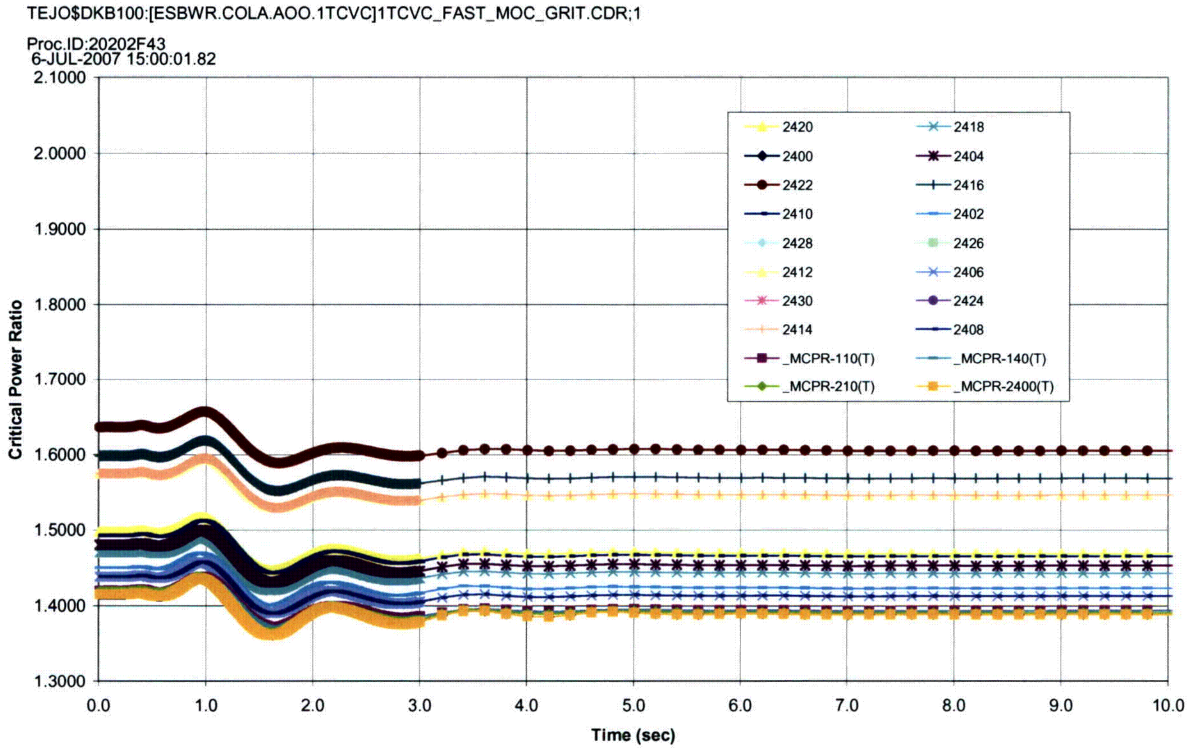


Figure 2.3-2g. Fast Closure of One Turbine Control Valve

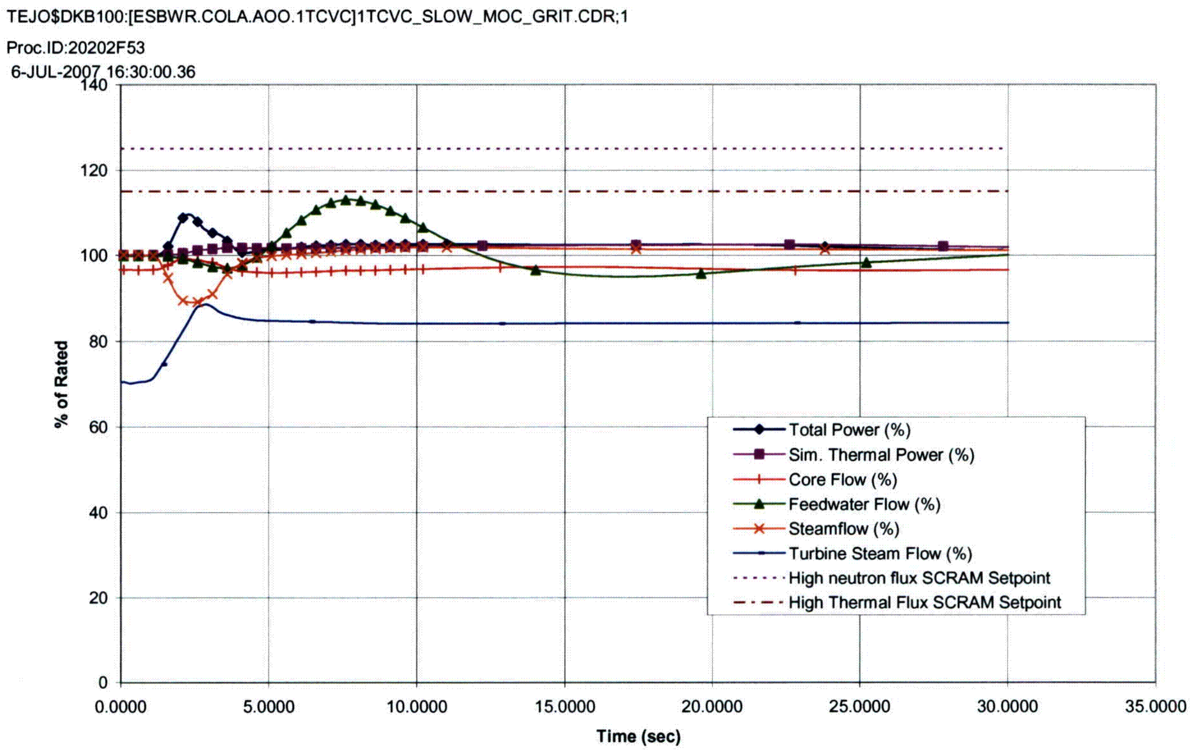


Figure 2.3-3a. Slow Closure of One Turbine Control Valve

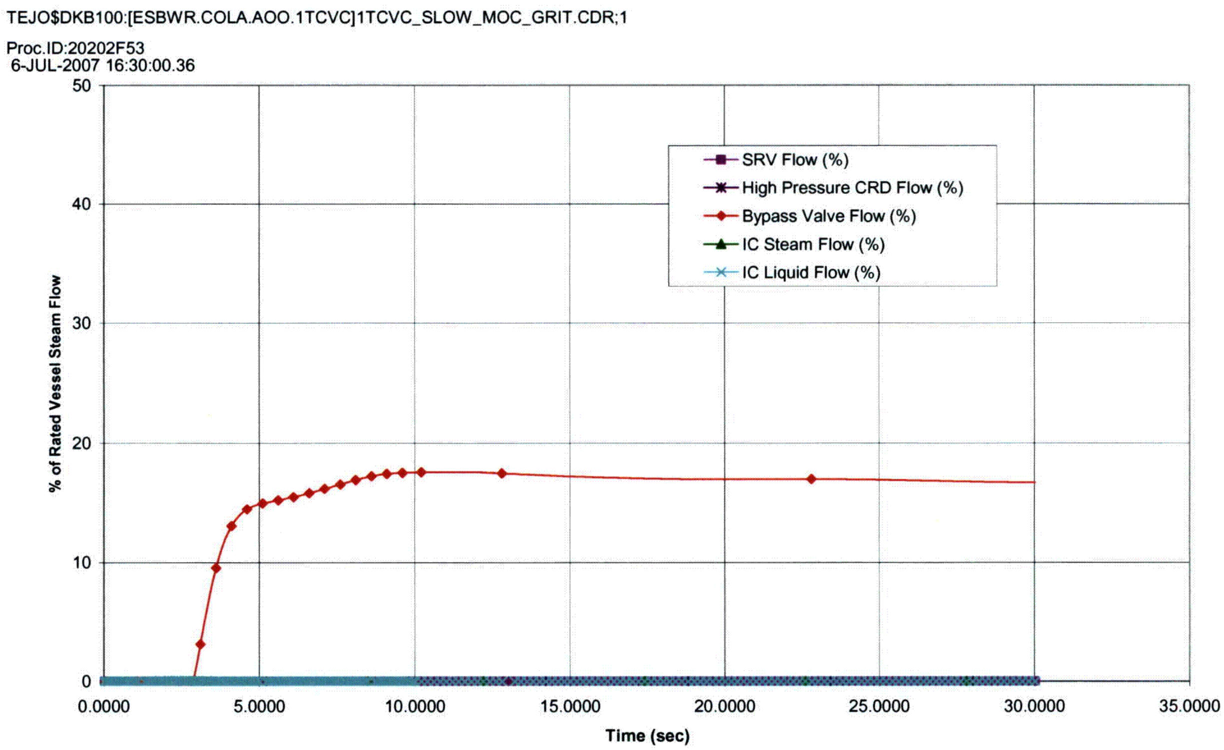


Figure 2.3-3b. Slow Closure of One Turbine Control Valve

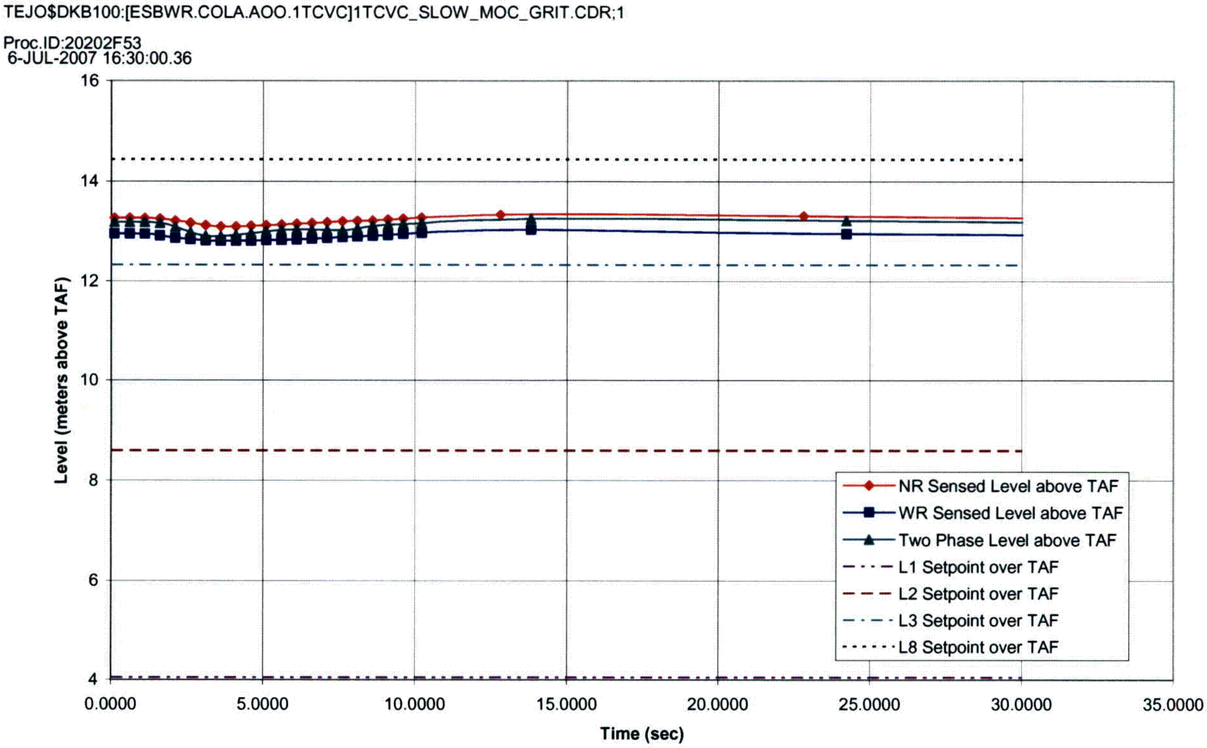


Figure 2.3-3c. Slow Closure of One Turbine Control Valve

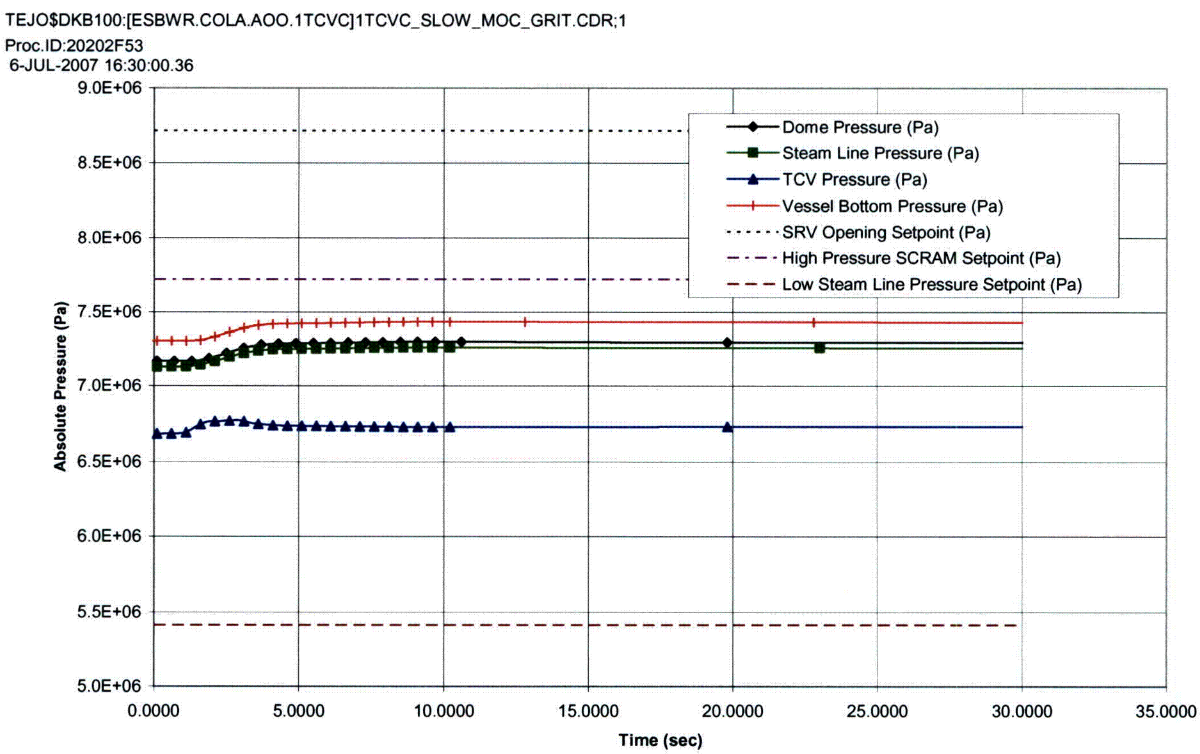


Figure 2.3-3d. Slow Closure of One Turbine Control Valve

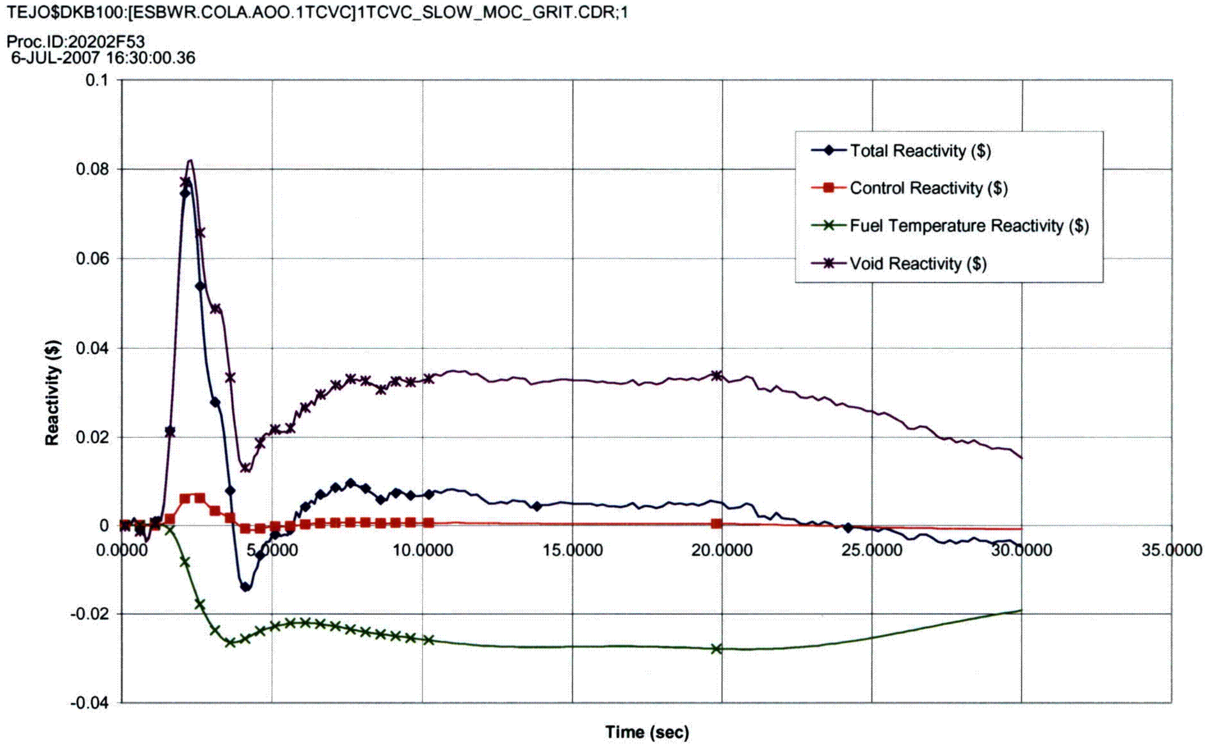


Figure 2.3-3e. Slow Closure of One Turbine Control Valve

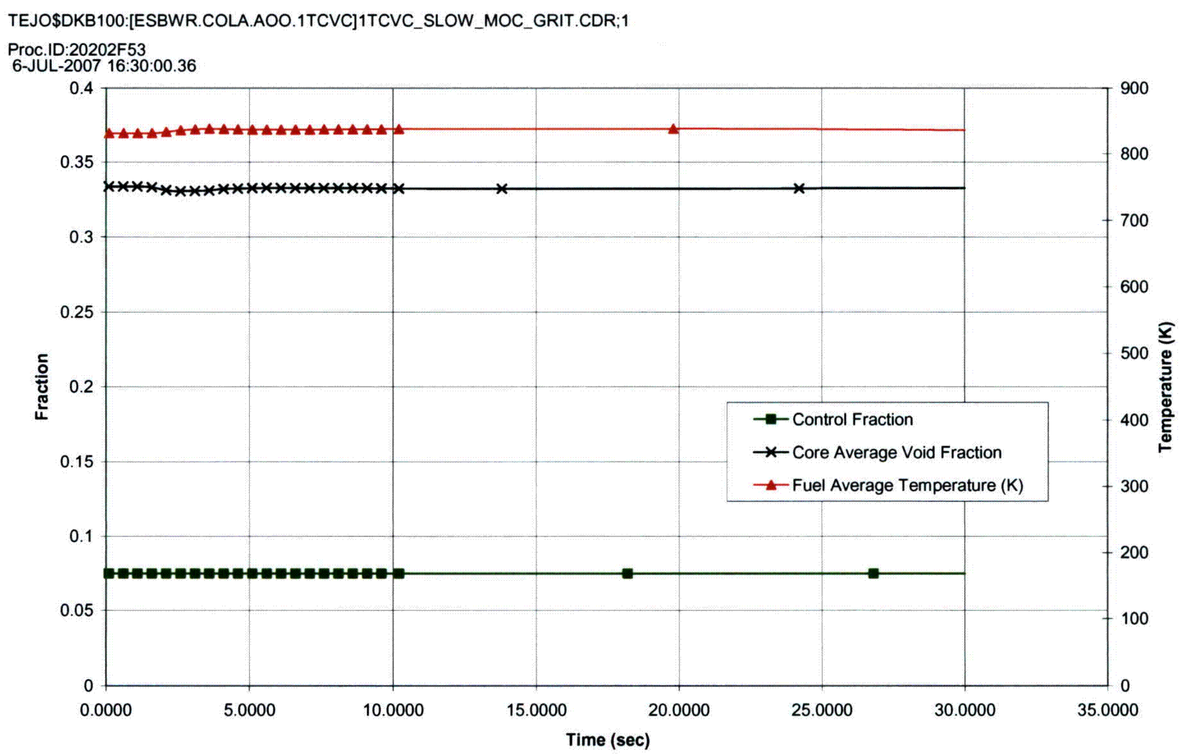


Figure 2.3-3f. Slow Closure of One Turbine Control Valve

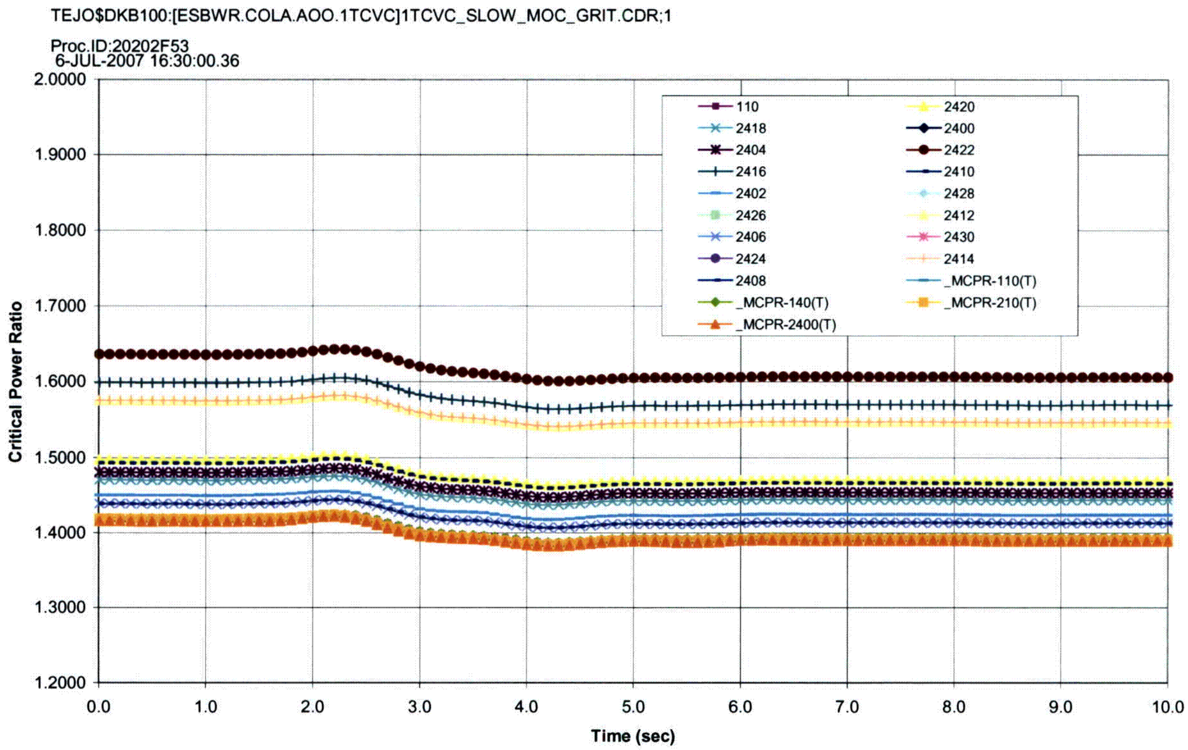


Figure 2.3-3g. Slow Closure of One Turbine Control Valve

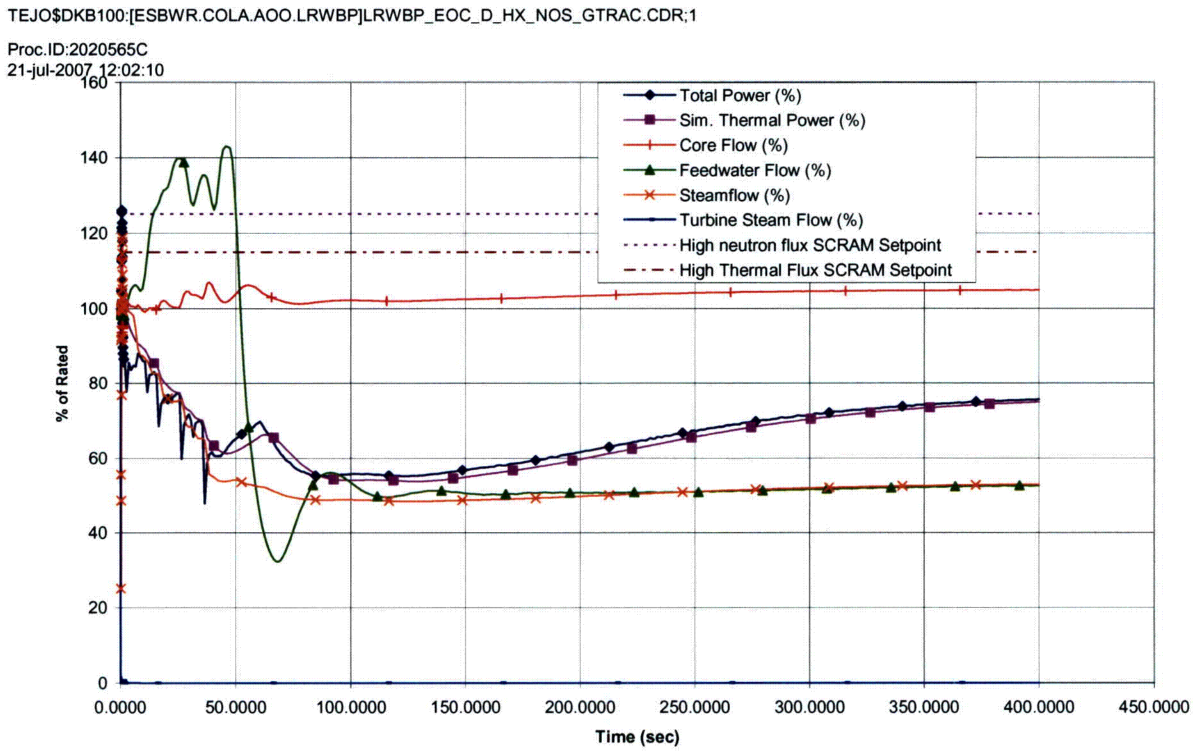


Figure 2.3-4a. Generator Load Rejection with Turbine Bypass

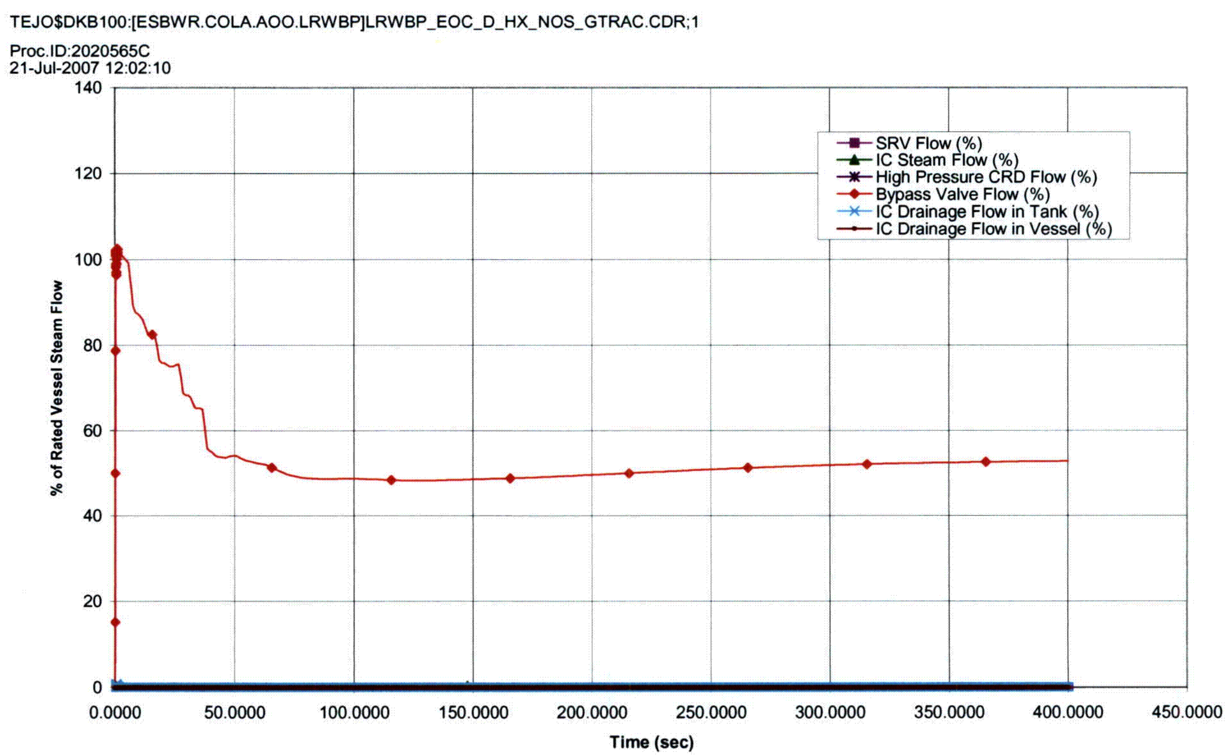


Figure 2.3-4b. Generator Load Rejection with Turbine Bypass

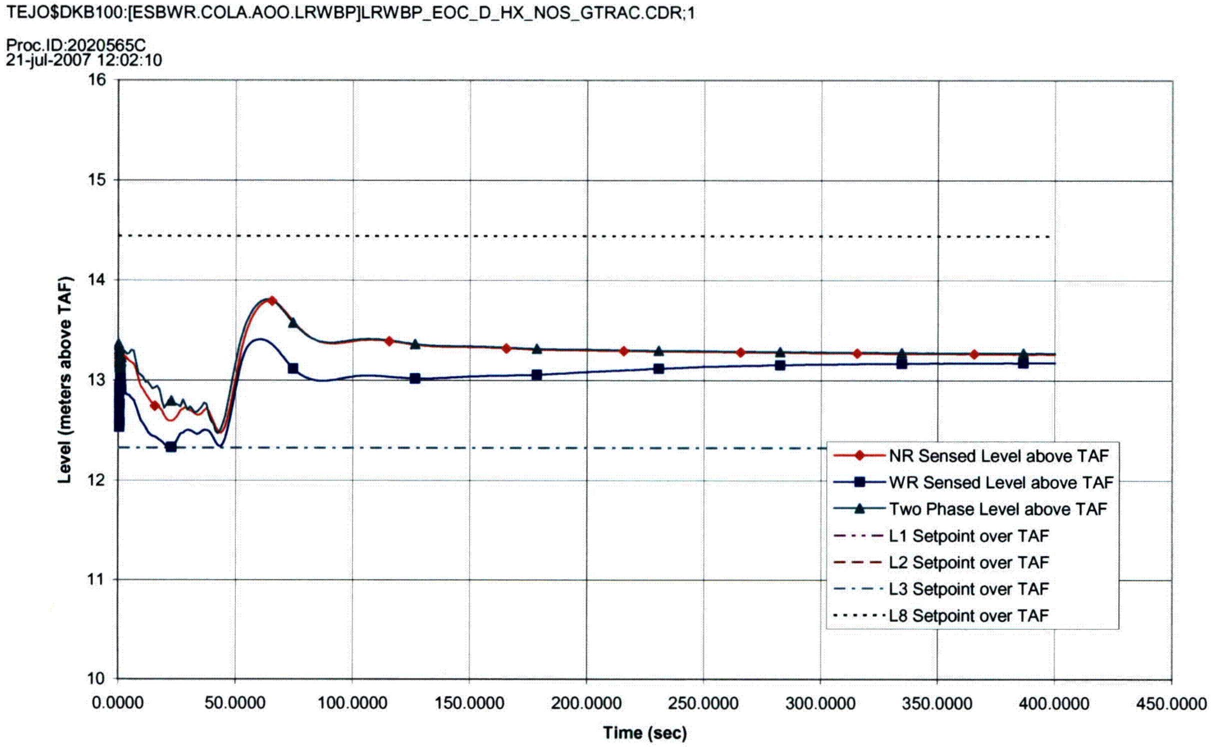


Figure 2.3-4c. Generator Load Rejection with Turbine Bypass

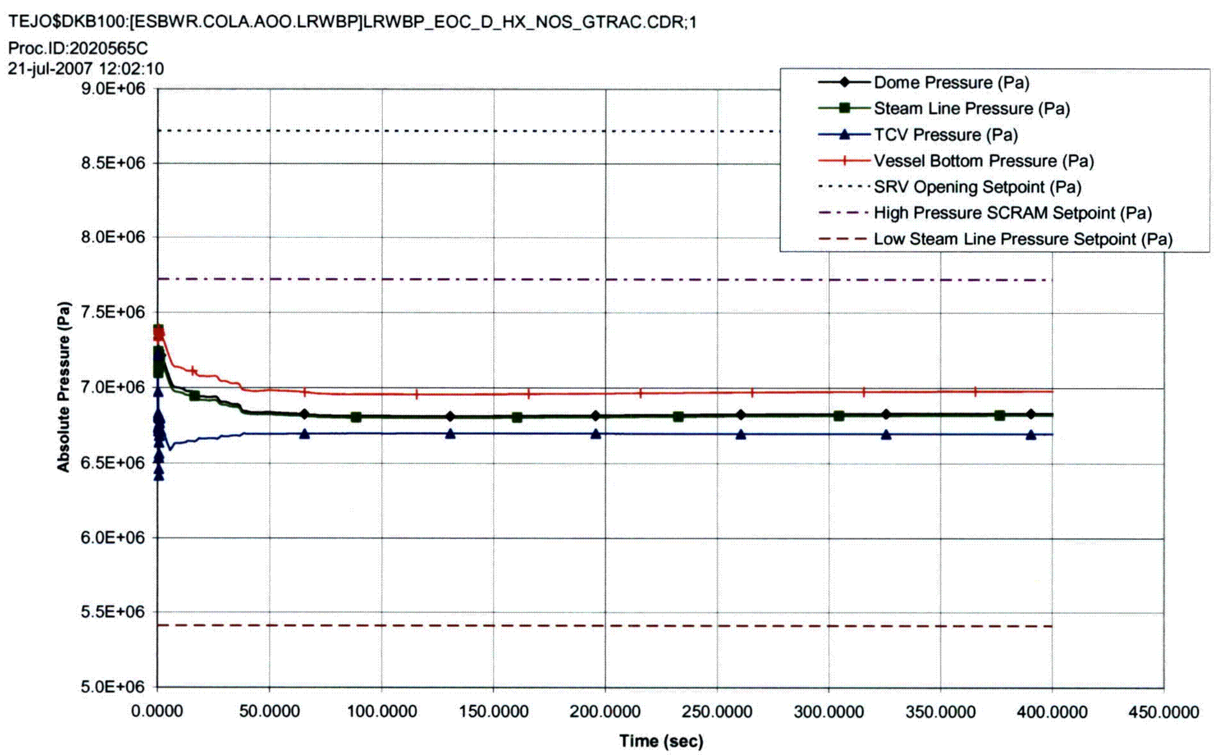


Figure 2.3-4d. Generator Load Rejection with Turbine Bypass

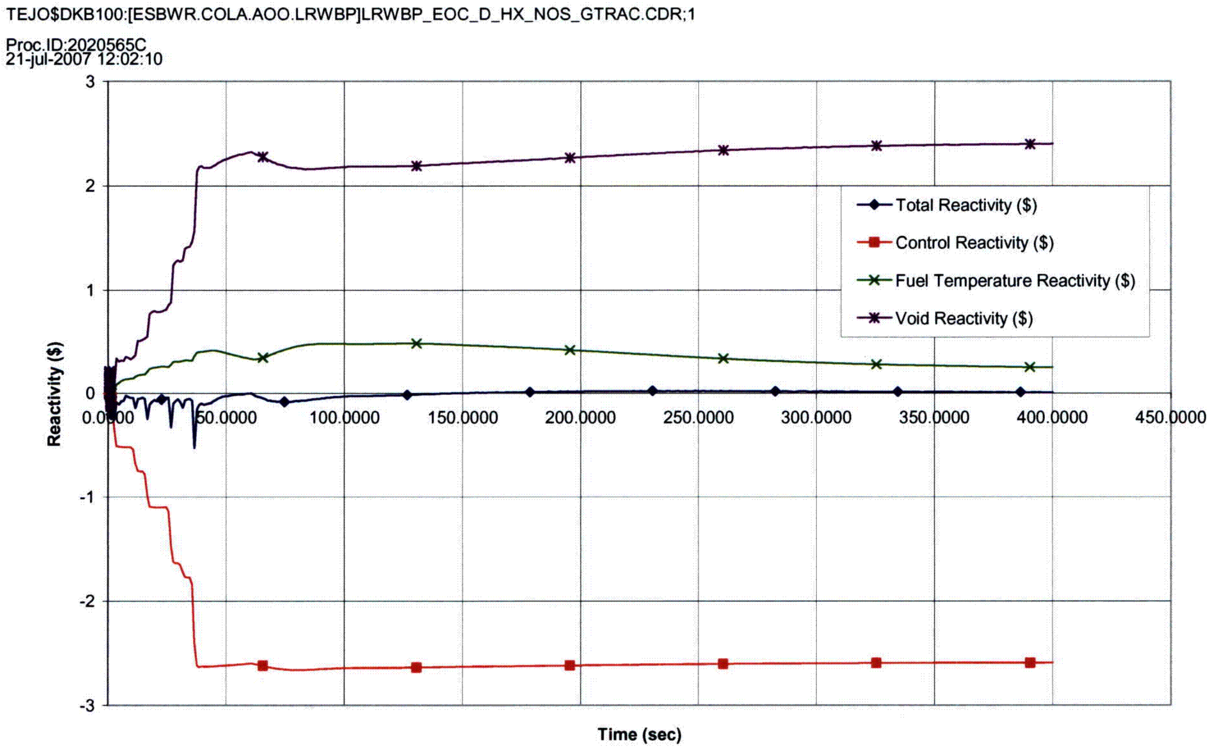


Figure 2.3-4e. Generator Load Rejection with Turbine Bypass

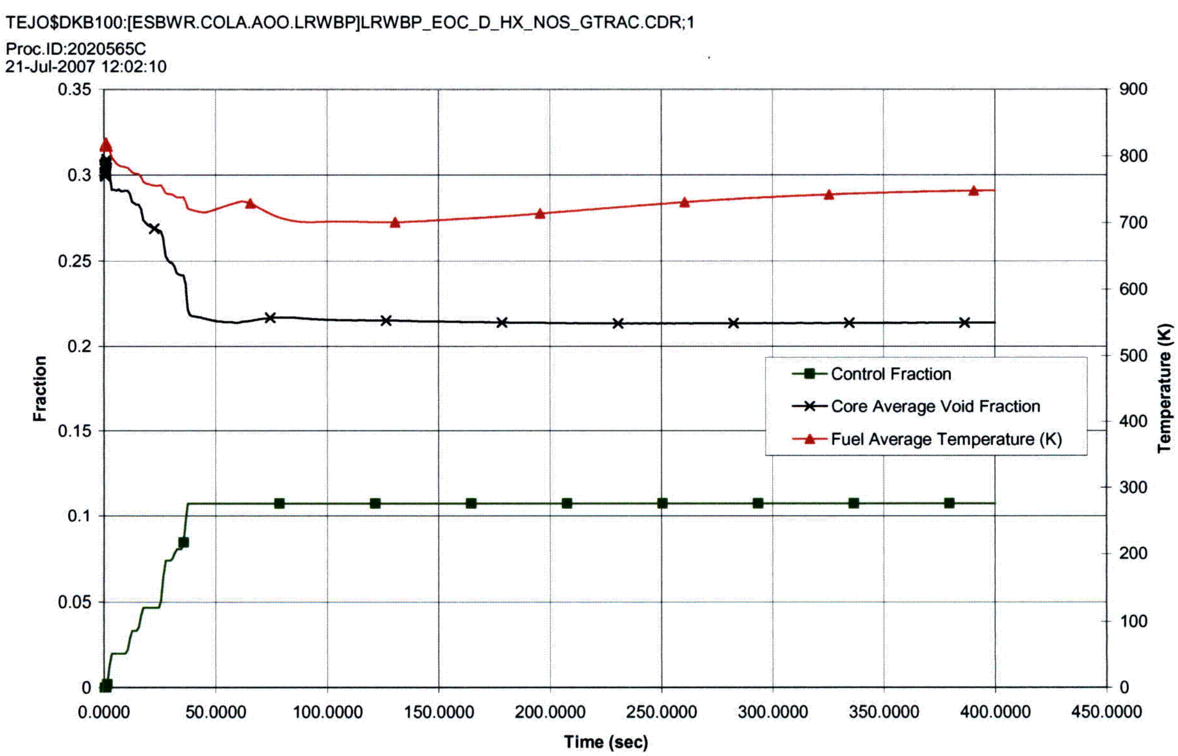


Figure 2.3-4f. Generator Load Rejection with Turbine Bypass

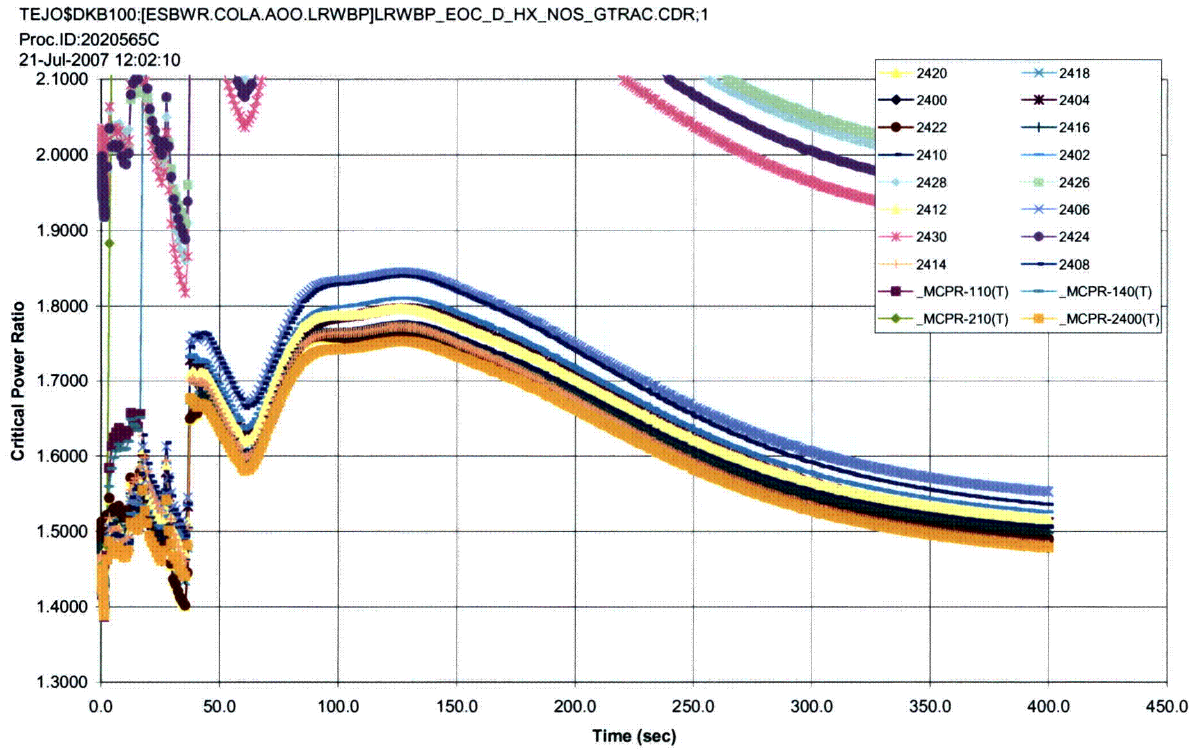


Figure 2.3-4g. Generator Load Rejection with Turbine Bypass

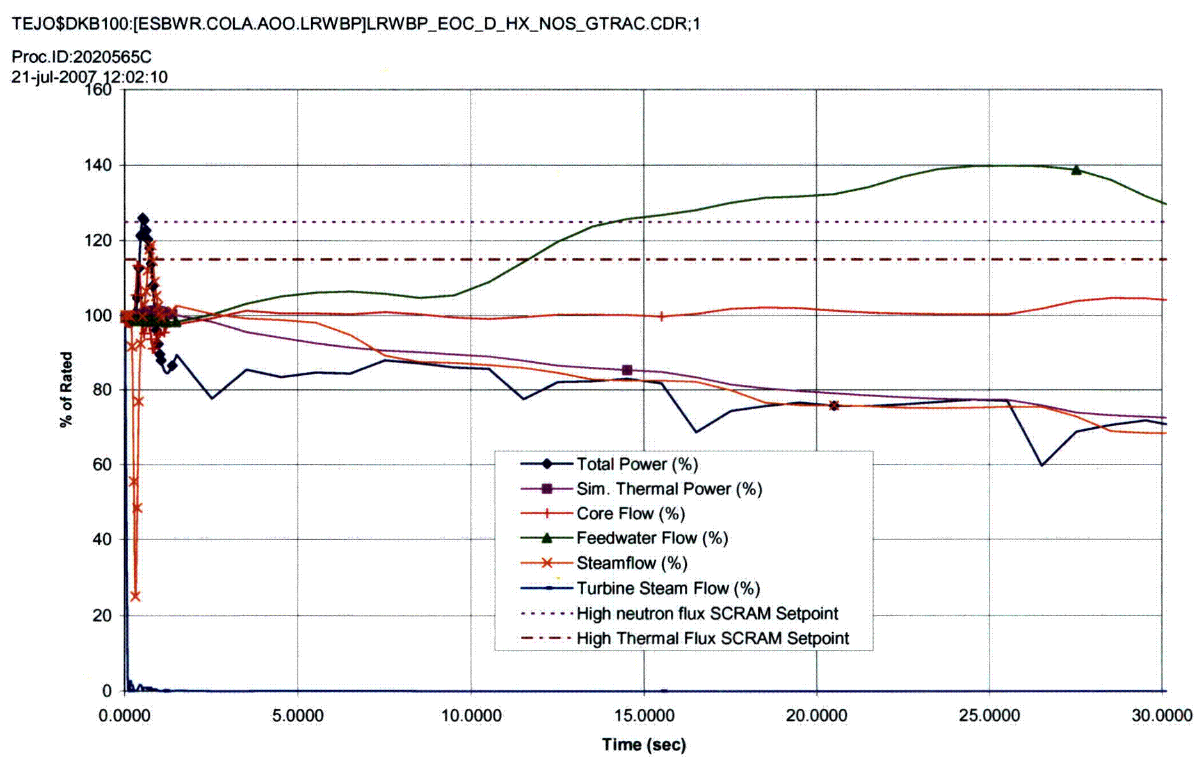


Figure 2.3-4h. Generator Load Rejection with Turbine Bypass (fig a from 0 to 30s)

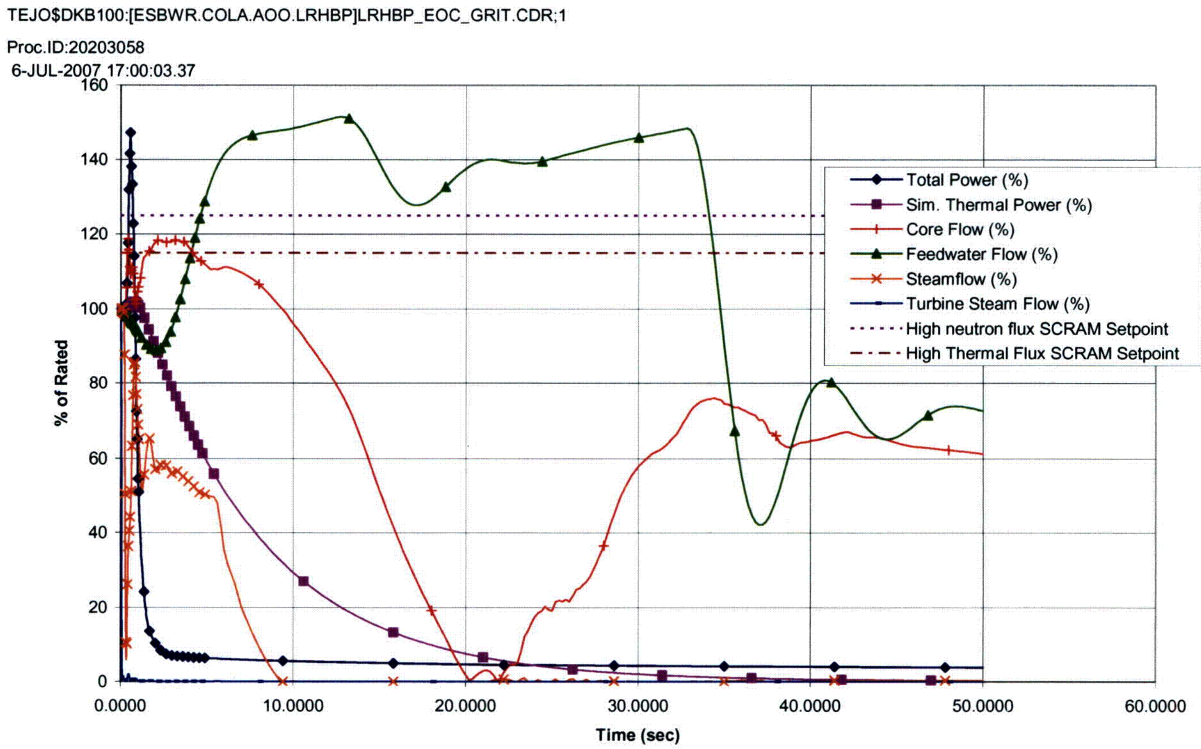


Figure 2.3-5a. Generator Load Rejection with a Single Failure in the Turbine Bypass System

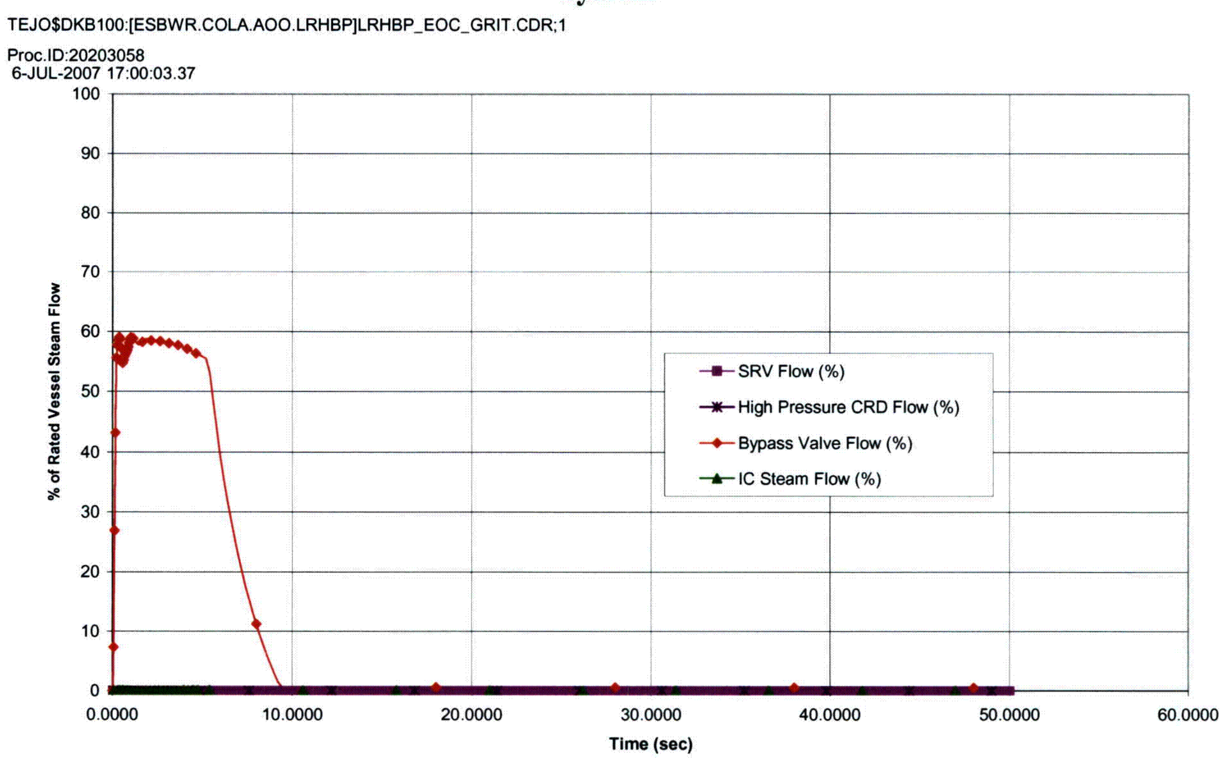


Figure 2.3-5b. Generator Load Rejection with a Single Failure in the Turbine Bypass System

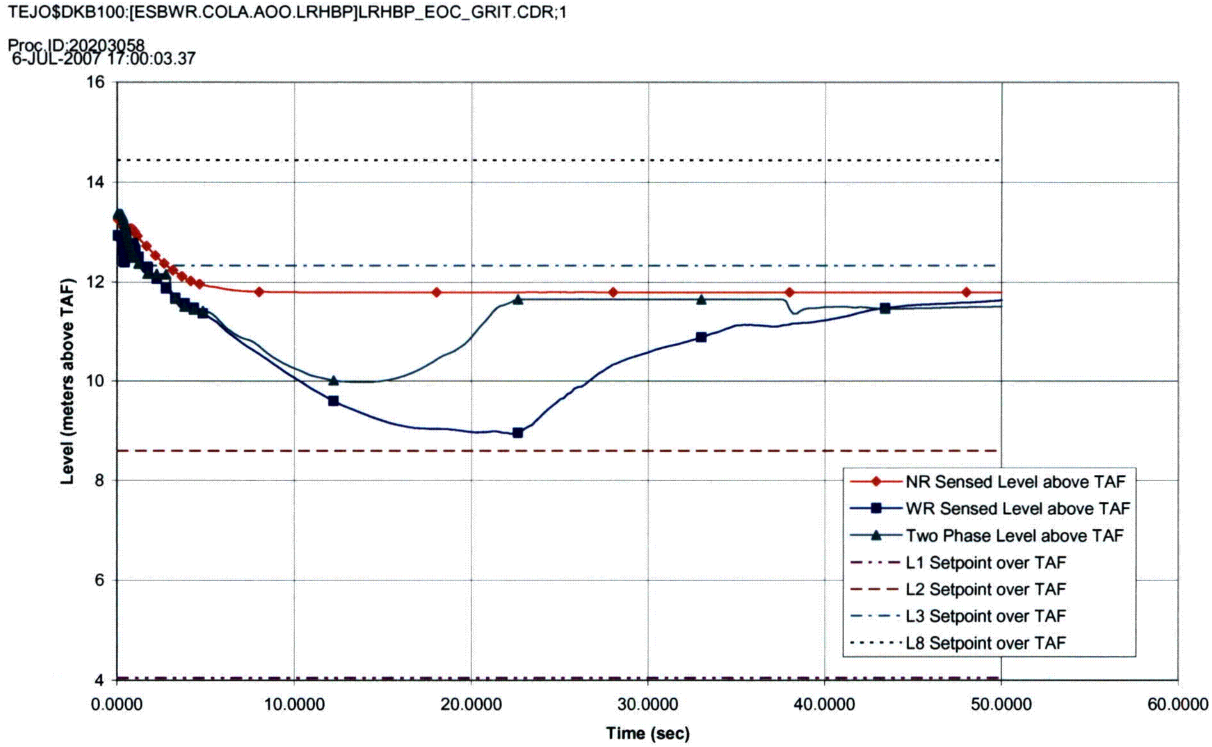


Figure 2.3-5c. Generator Load Rejection with a Single Failure in the Turbine Bypass System

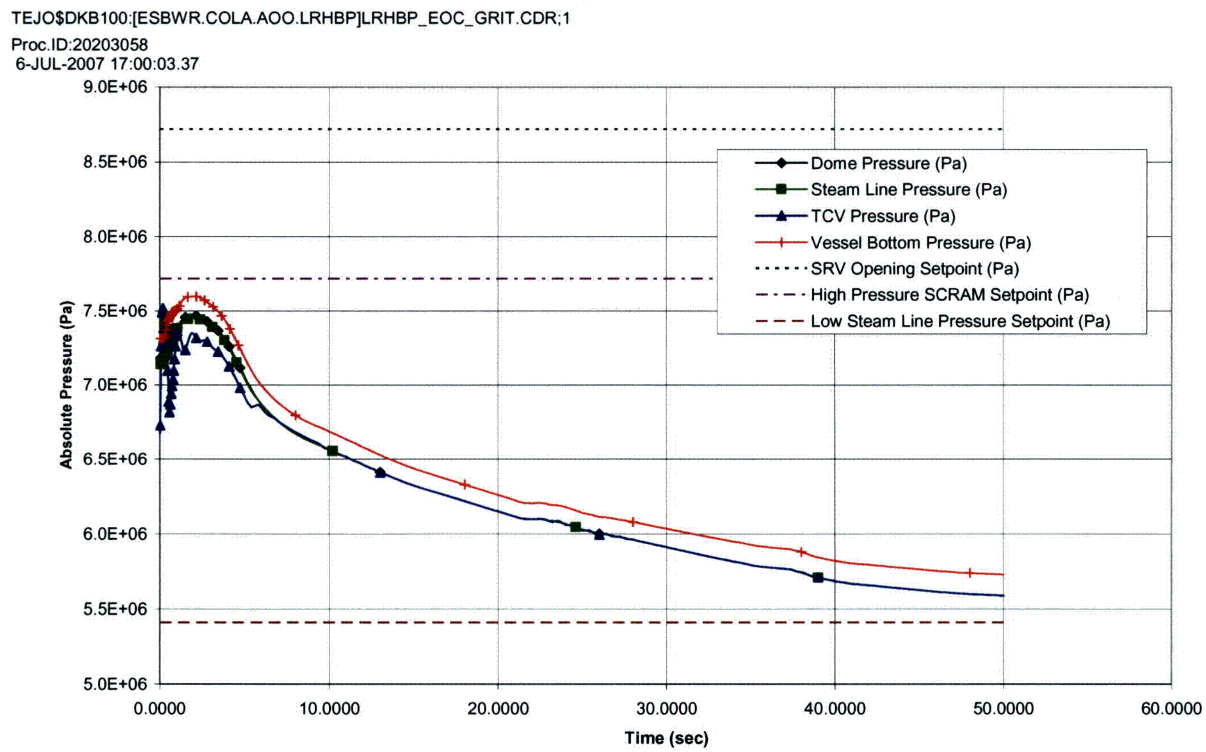


Figure 2.3-5d. Generator Load Rejection with a Single Failure in the Turbine Bypass System

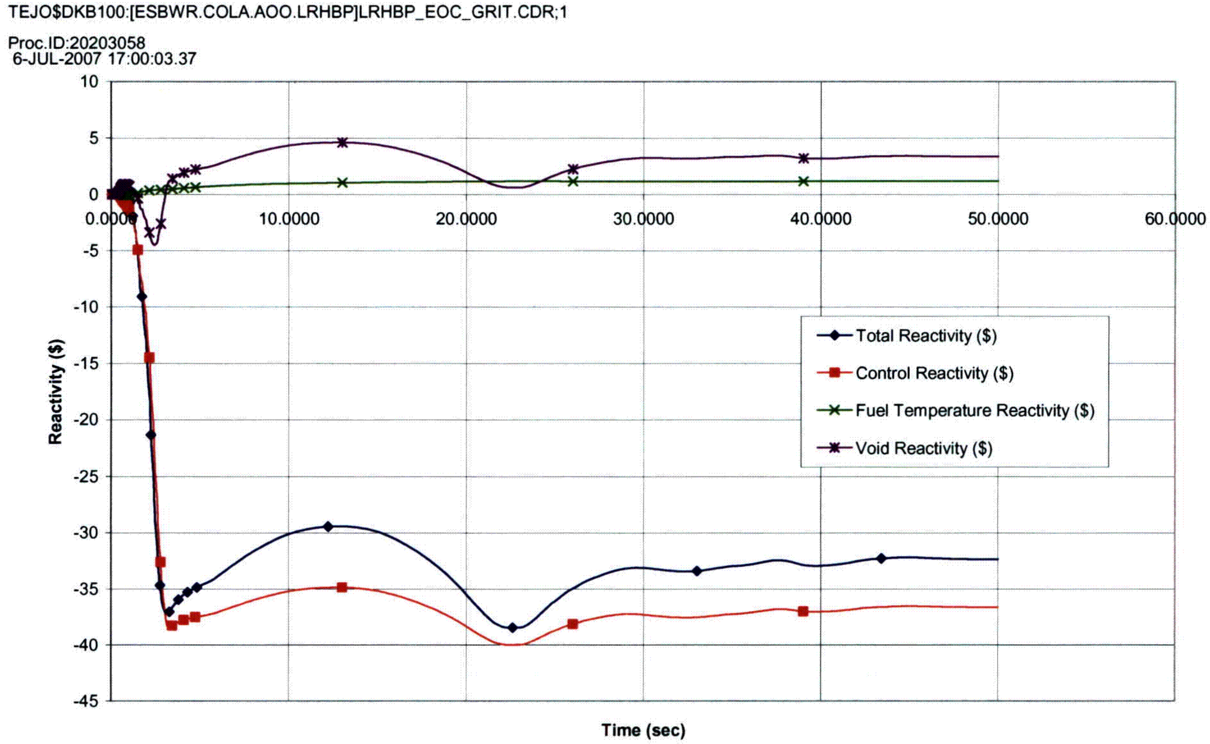


Figure 2.3-5e. Generator Load Rejection with a Single Failure in the Turbine Bypass System

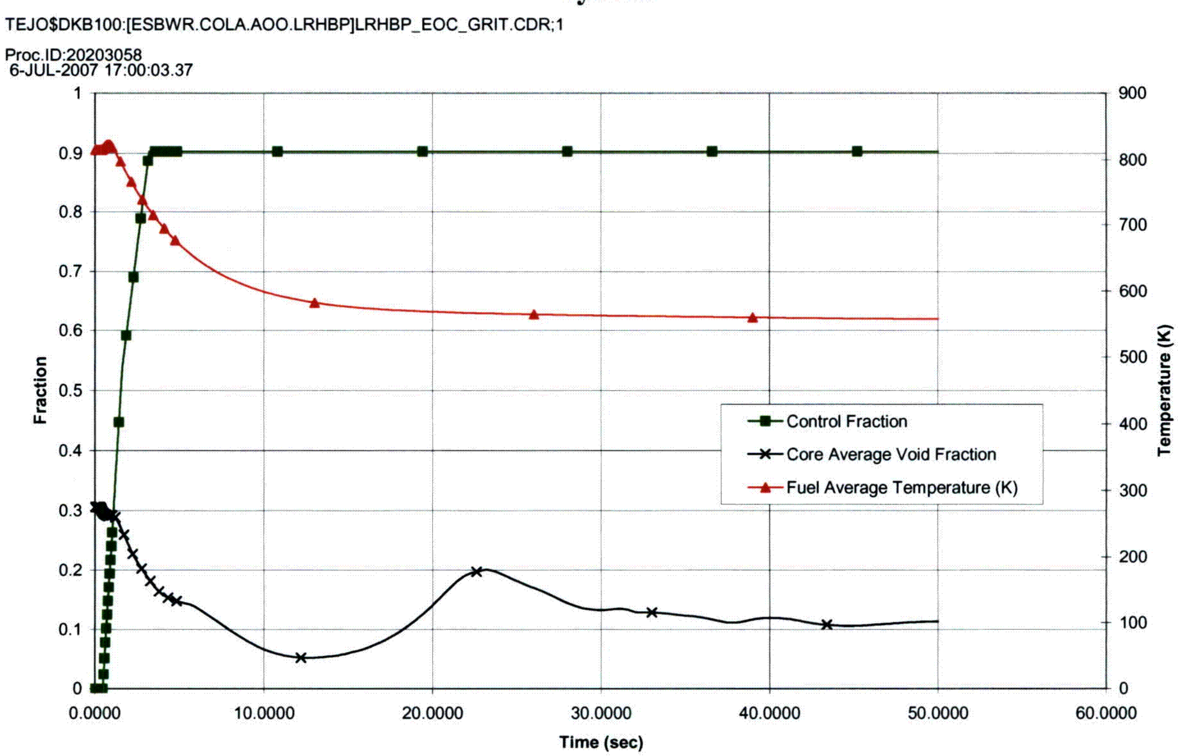


Figure 2.3-5f. Generator Load Rejection with a Single Failure in the Turbine Bypass System

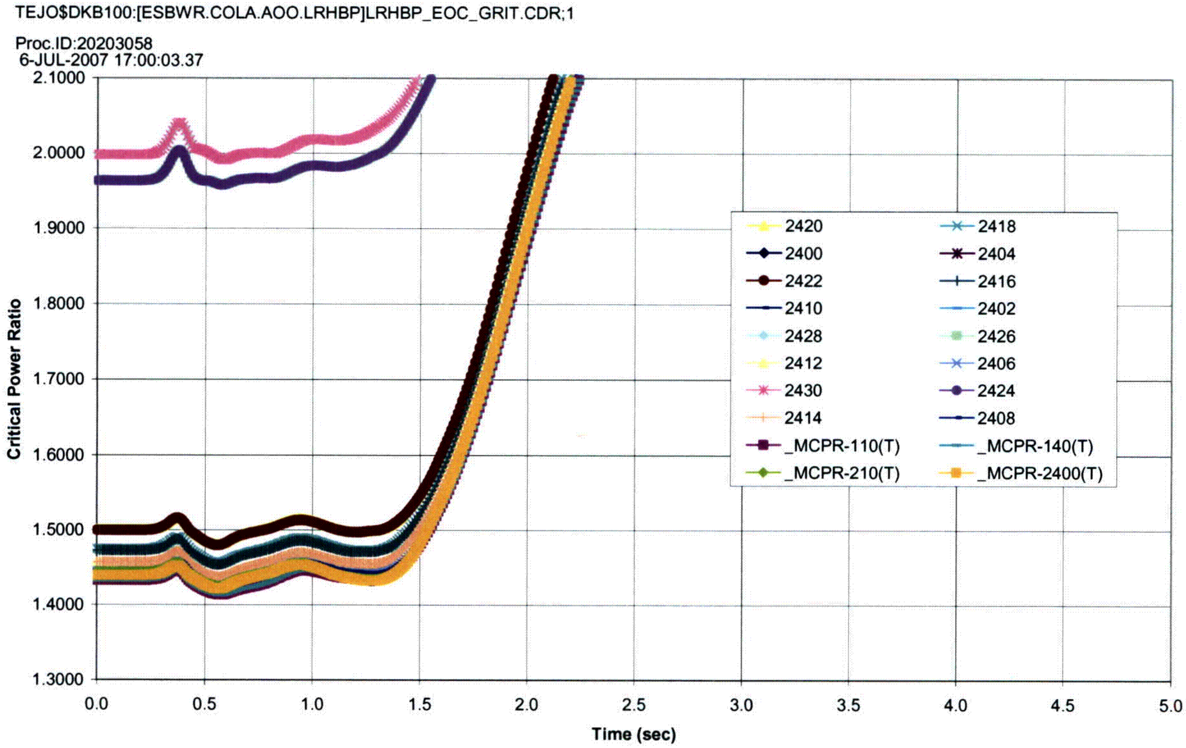


Figure 2.3-5g. Generator Load Rejection with a Single Failure in the Turbine Bypass System

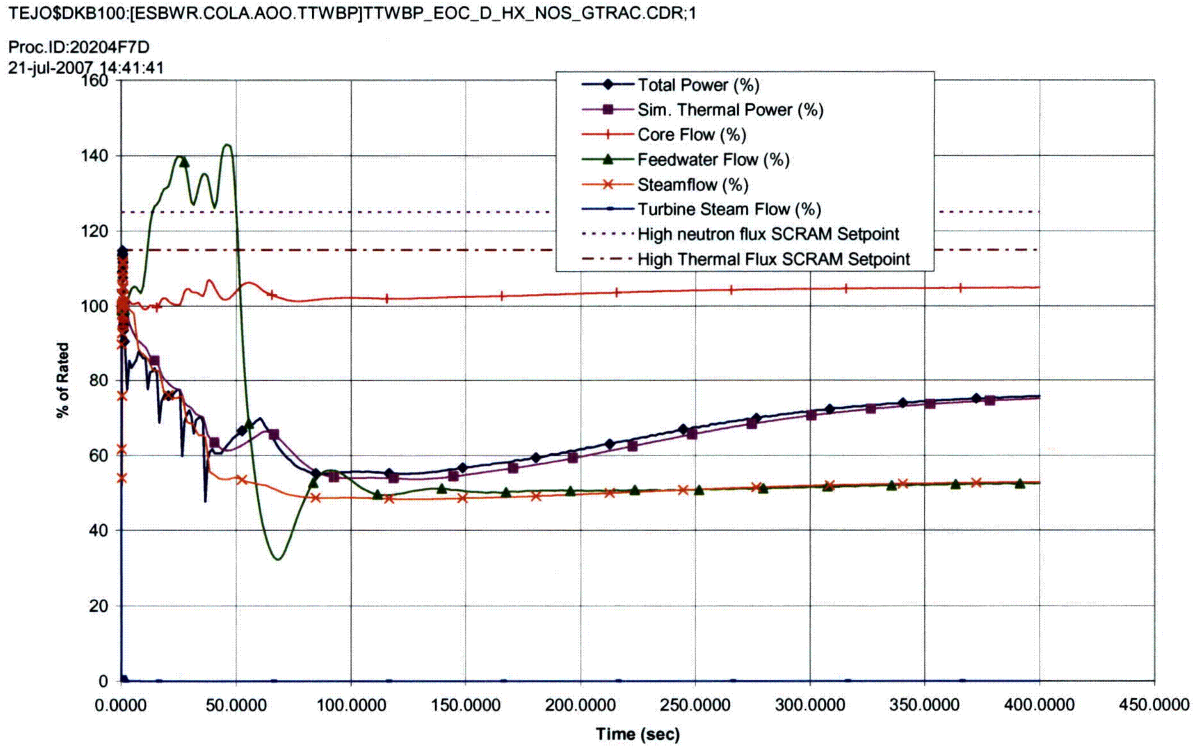


Figure 2.3-6a. Turbine Trip with Turbine Bypass

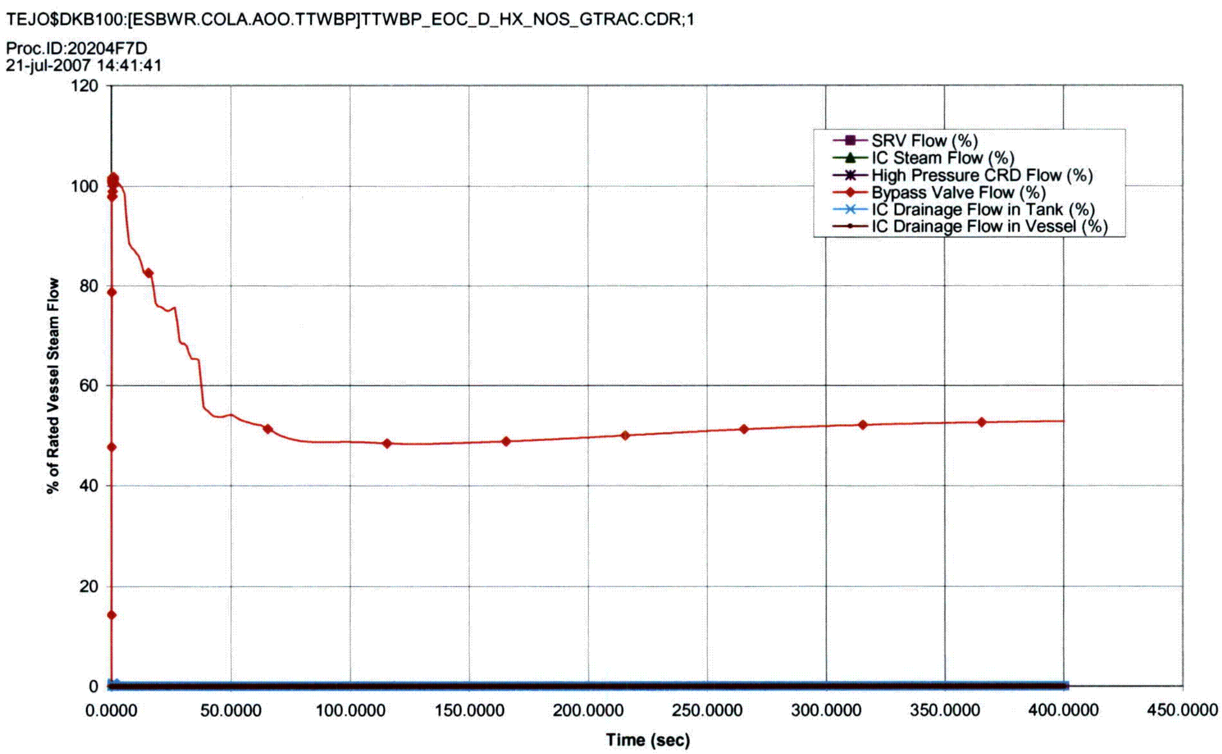


Figure 2.3-6b. Turbine Trip with Turbine Bypass

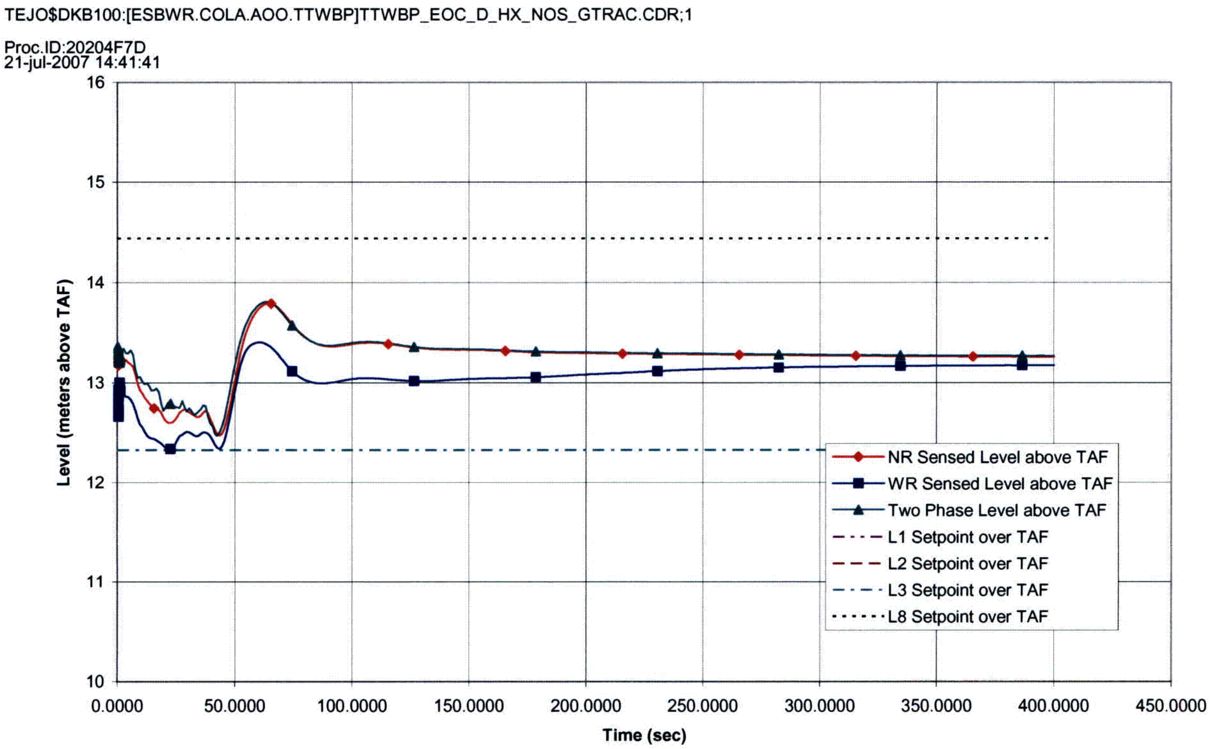


Figure 2.3-6c. Turbine Trip with Turbine Bypass

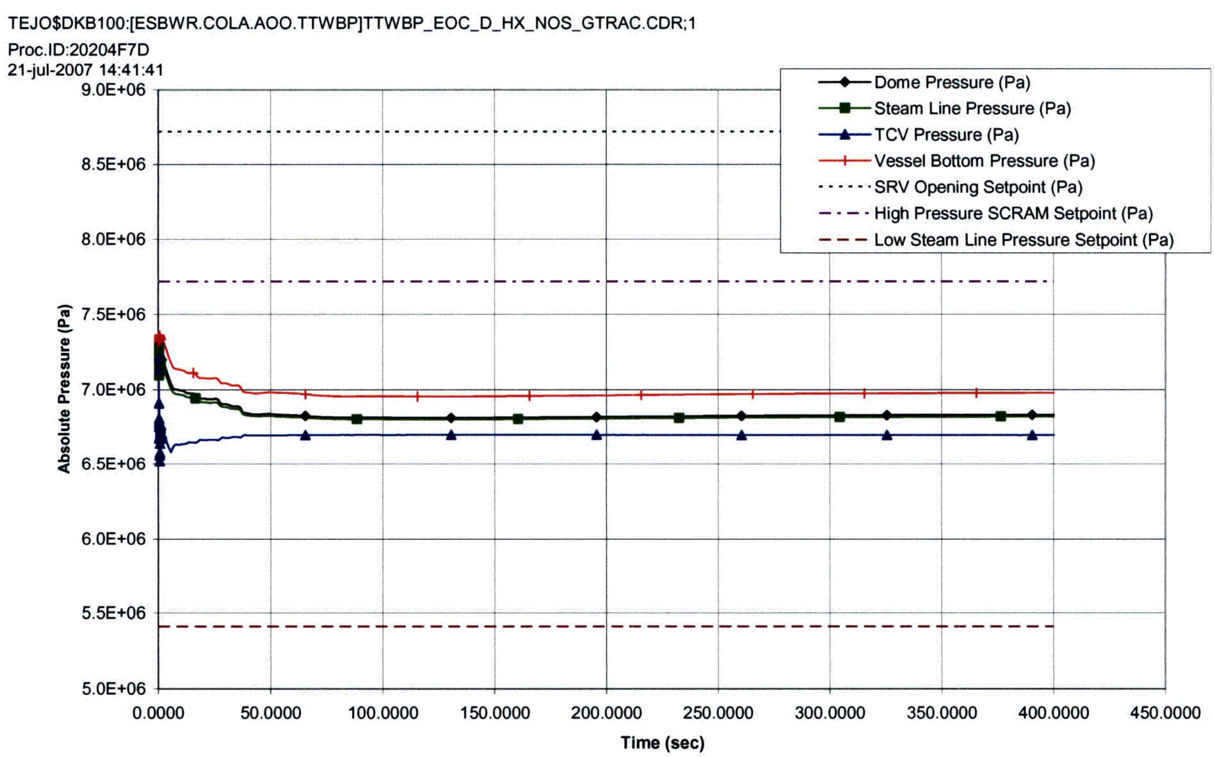


Figure 2.3-6d. Turbine Trip with Turbine Bypass

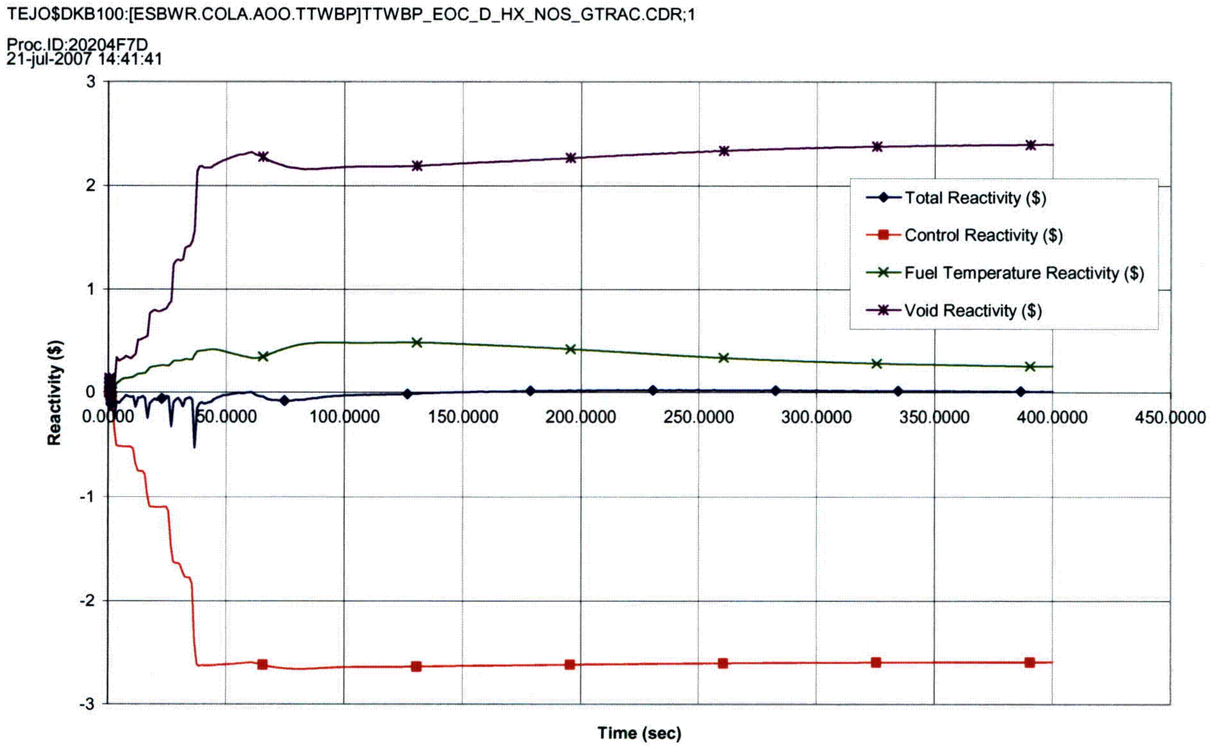


Figure 2.3-6e. Turbine Trip with Turbine Bypass

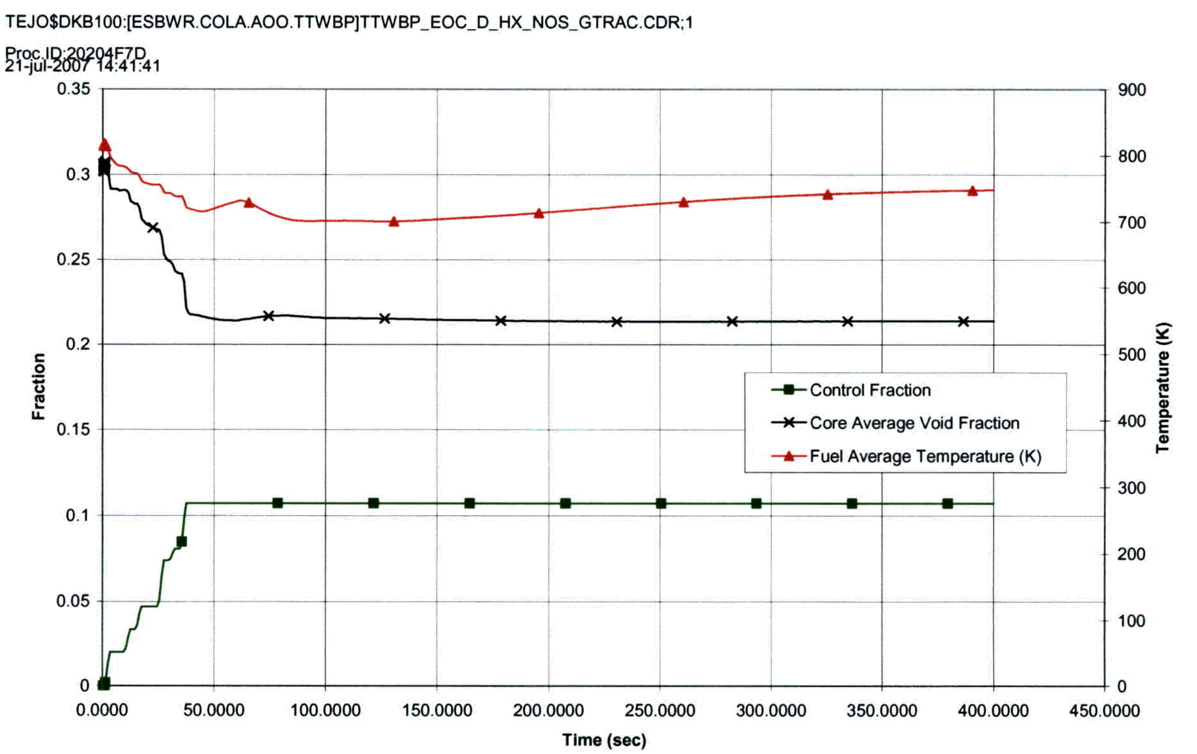


Figure 2.3-6f. Turbine Trip with Turbine Bypass

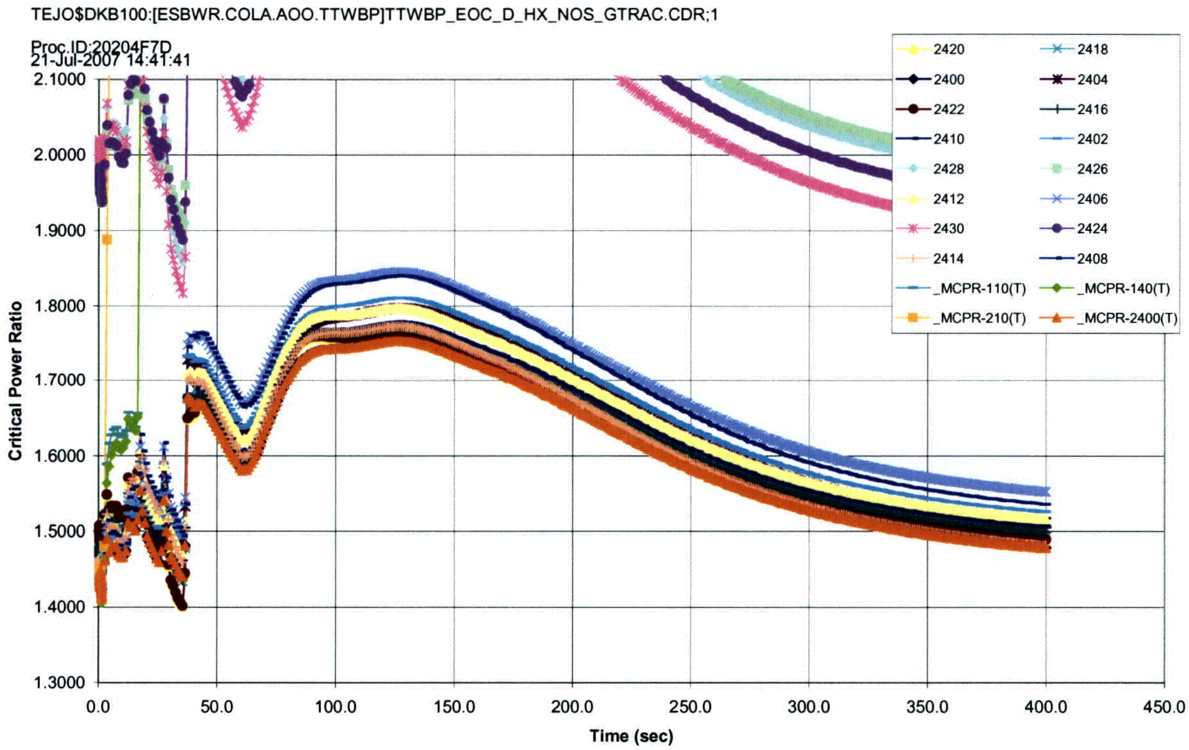


Figure 2.3-6g. Turbine Trip with Turbine Bypass

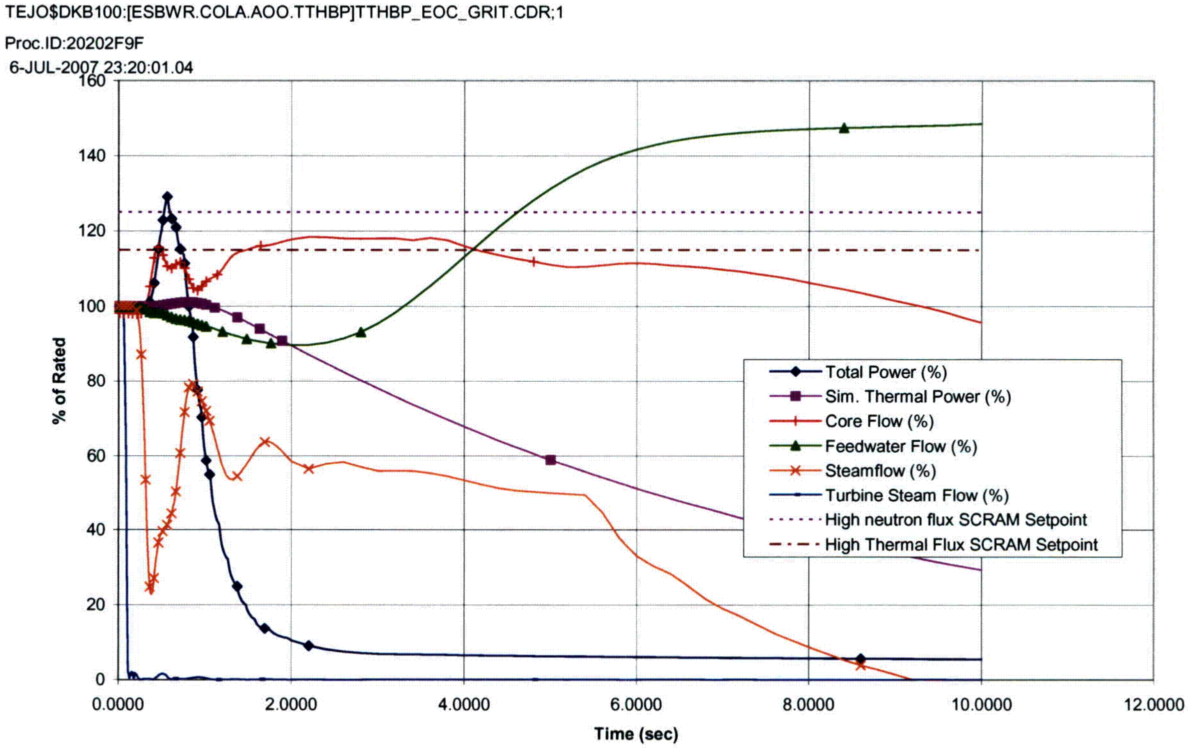


Figure 2.3-7a. Turbine Trip with a Single Failure in the Turbine Bypass System

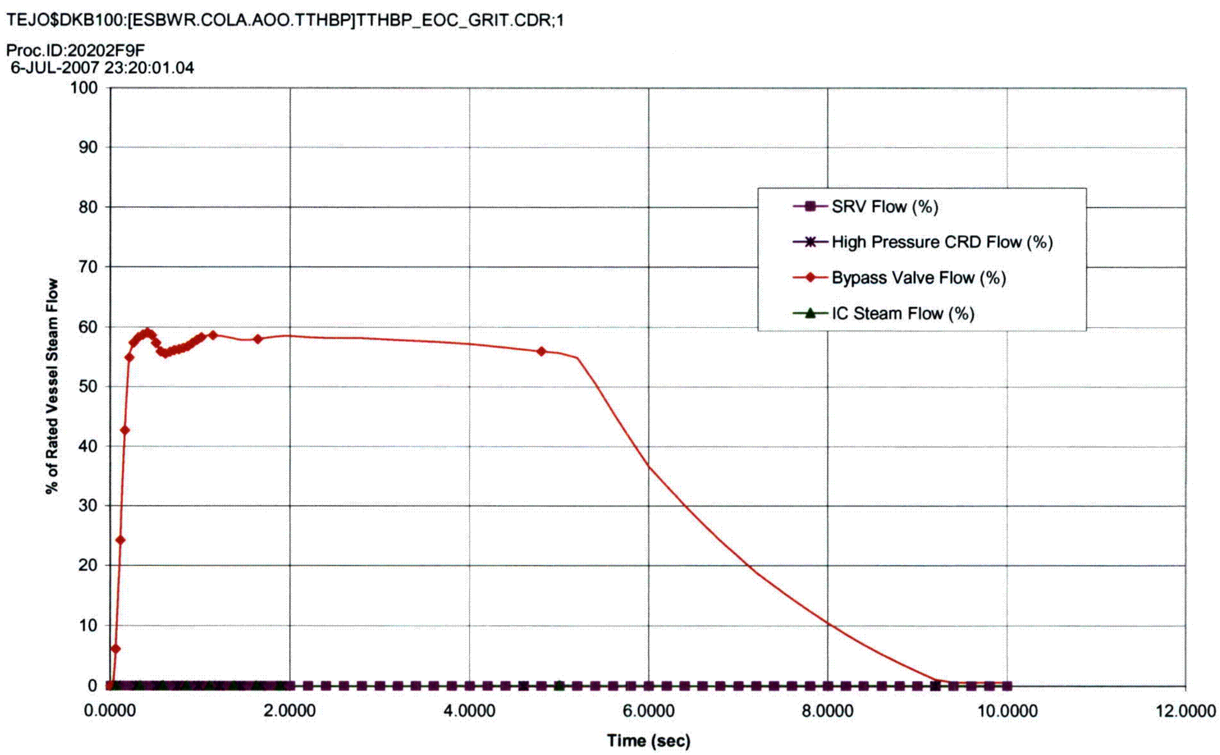


Figure 2.3-7b. Turbine Trip with a Single Failure in the Turbine Bypass System

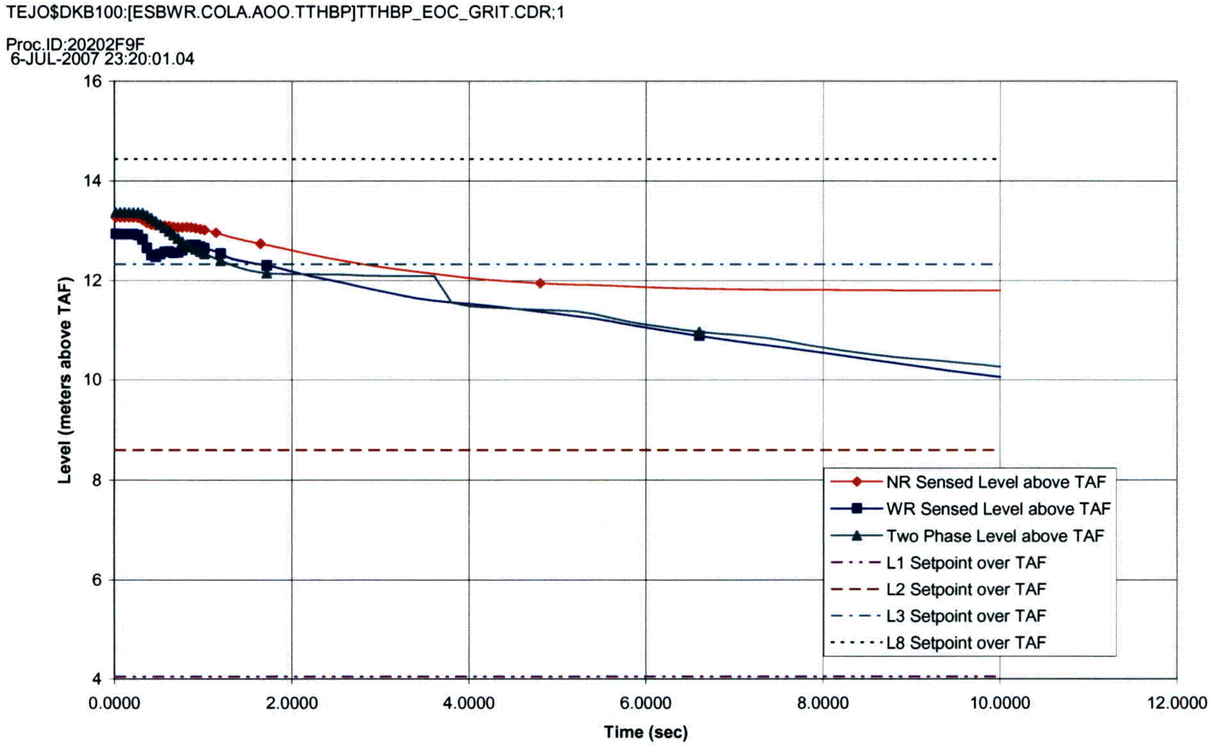


Figure 2.3-7c. Turbine Trip with a Single Failure in the Turbine Bypass System

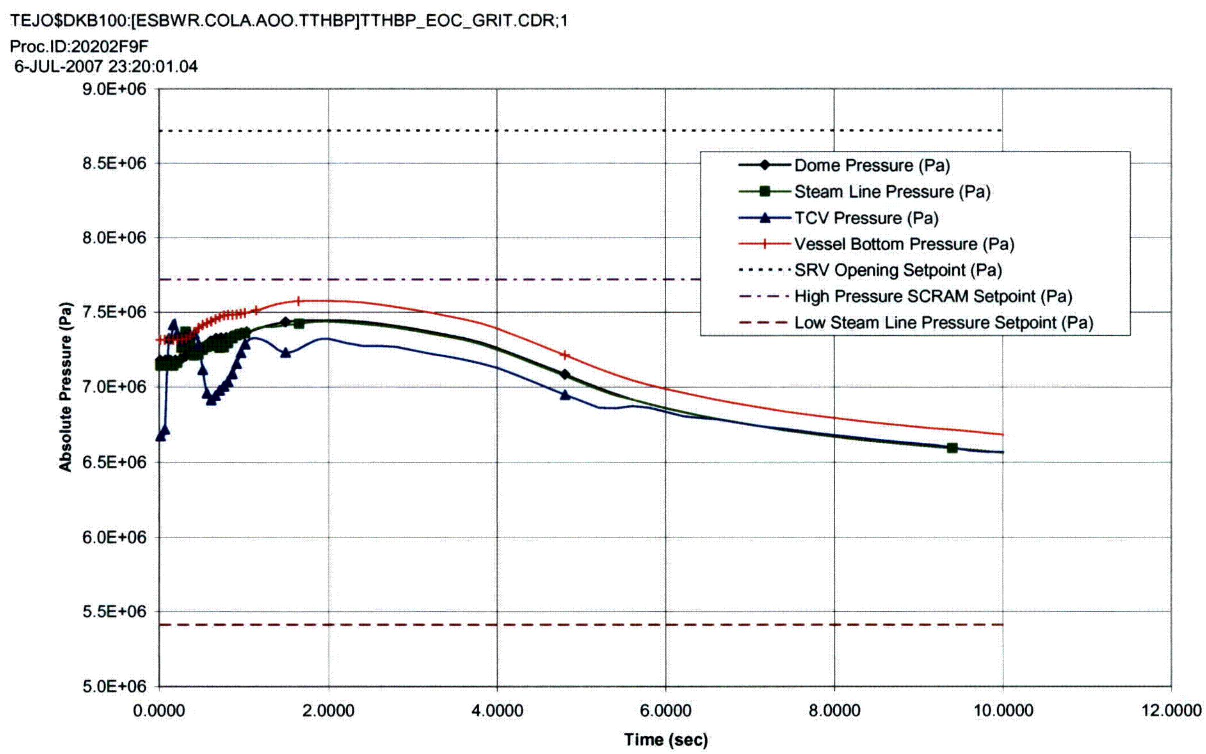


Figure 2.3-7d. Turbine Trip with a Single Failure in the Turbine Bypass System

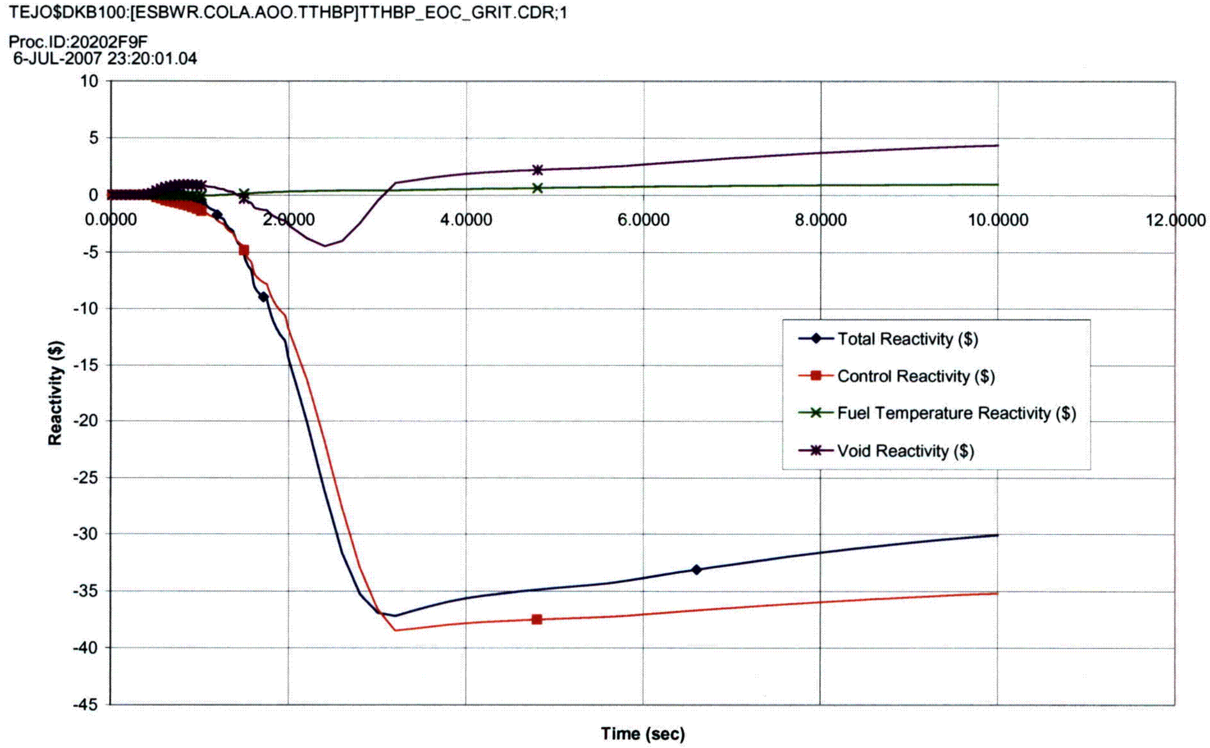


Figure 2.3-7e. Turbine Trip with a Single Failure in the Turbine Bypass System

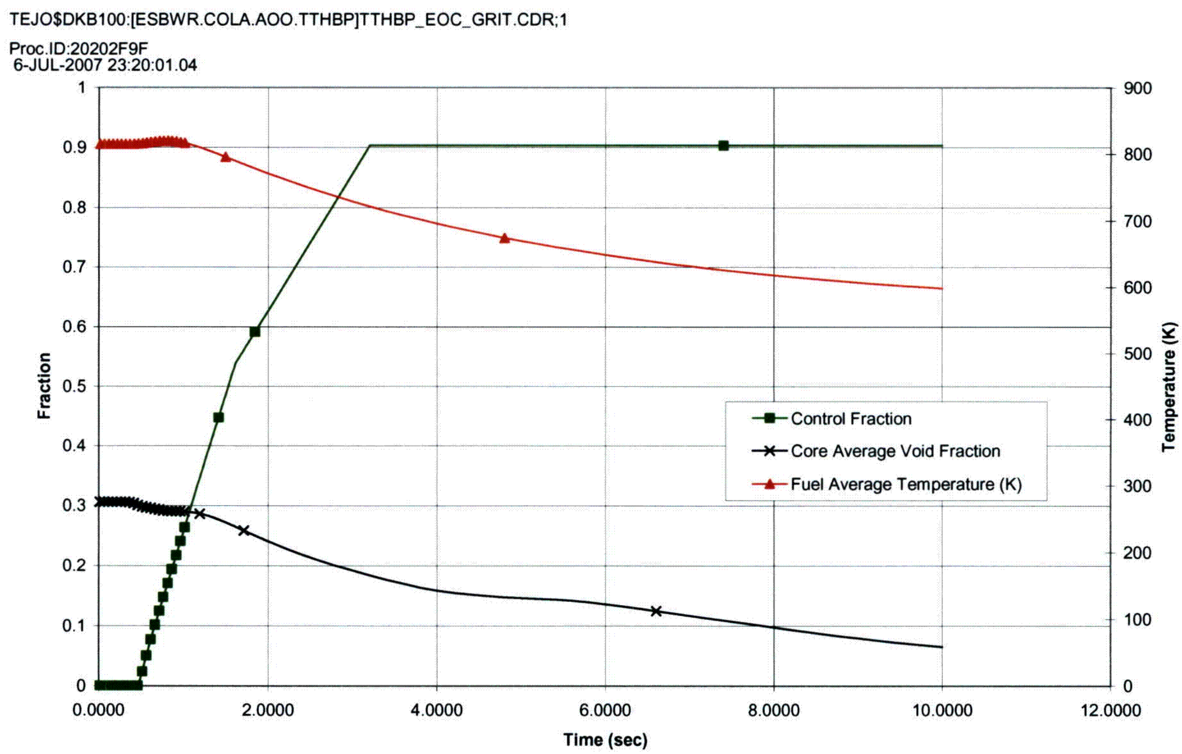


Figure 2.3-7f. Turbine Trip with a Single Failure in the Turbine Bypass System

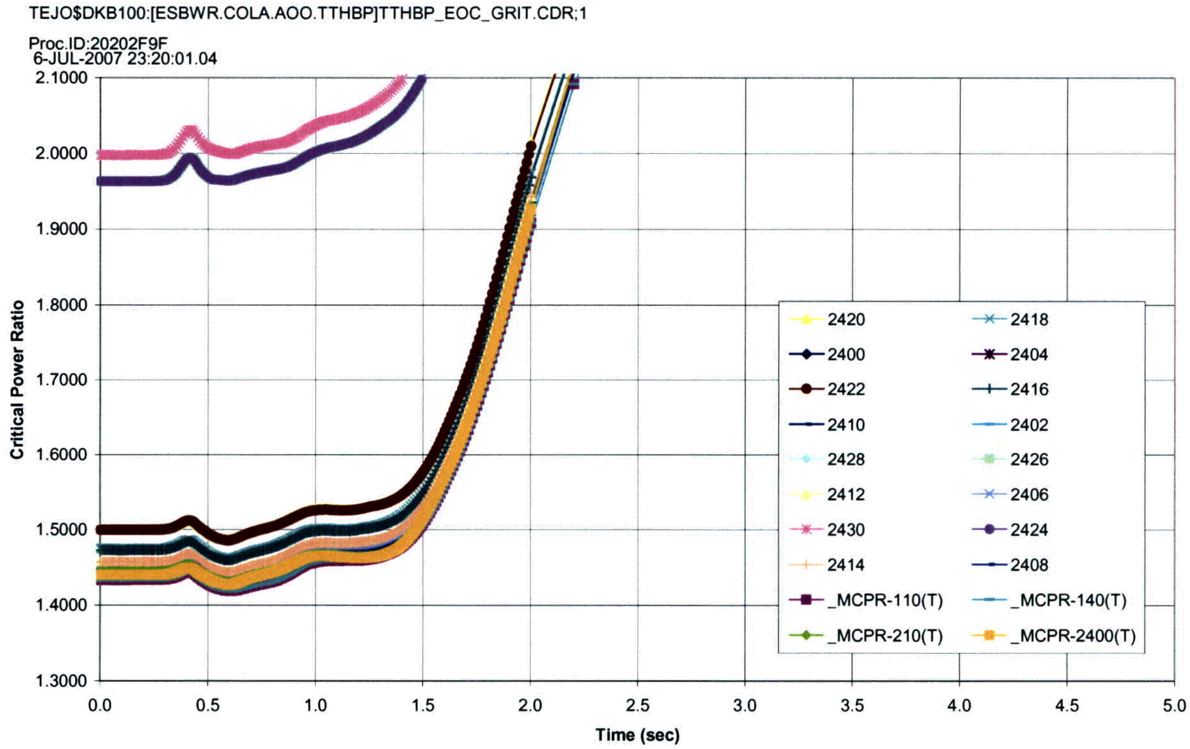


Figure 2.3-7g. Turbine Trip with a Single Failure in the Turbine Bypass System

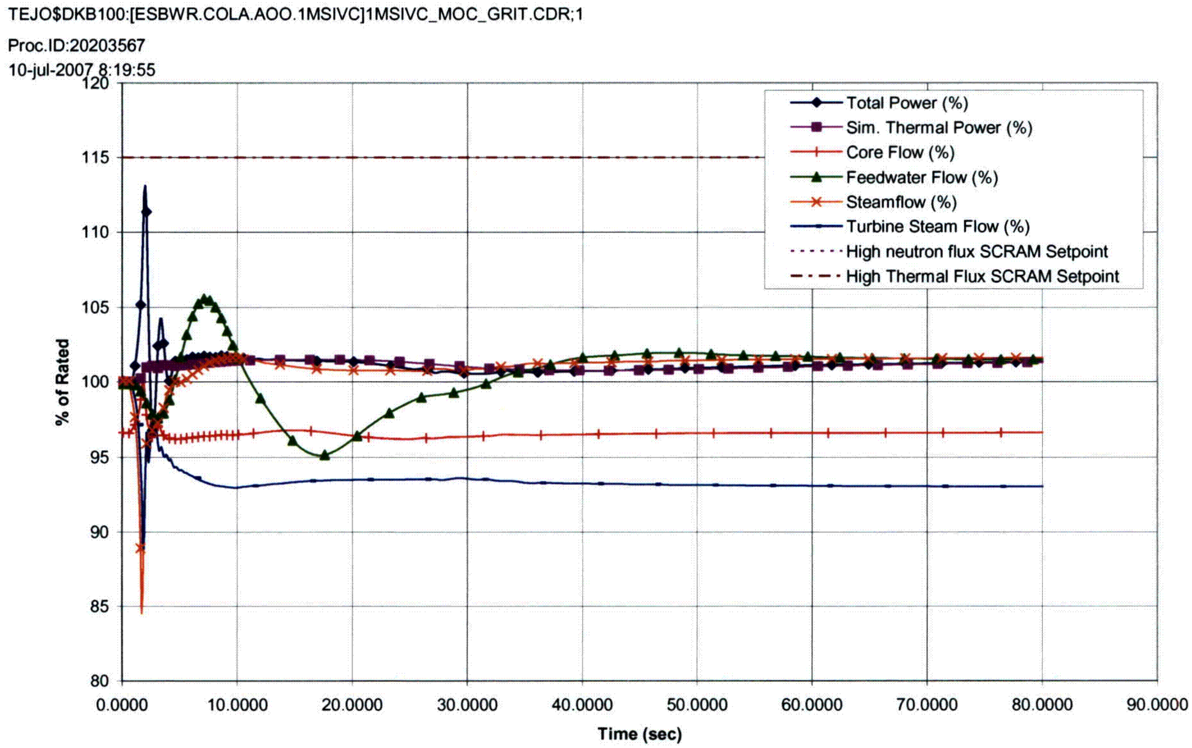


Figure 2.3-8a. One MSIV Closure

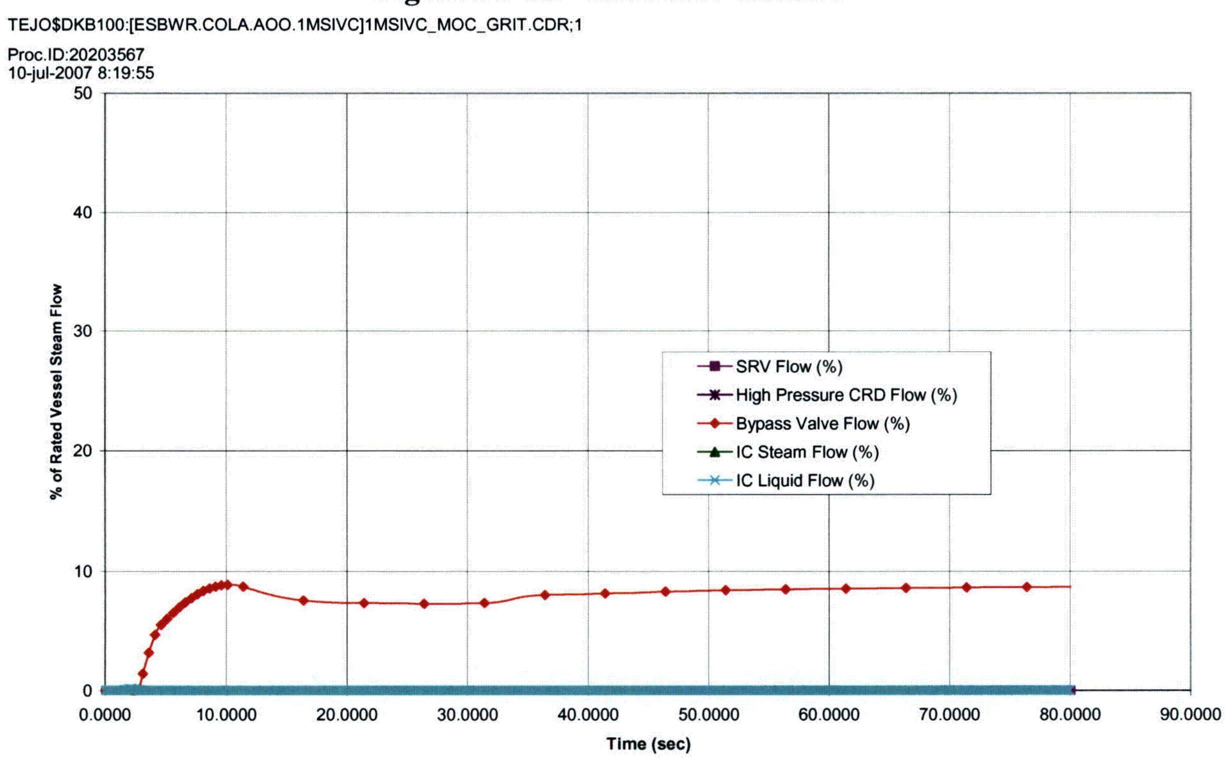


Figure 2.3-8b. One MSIV Closure

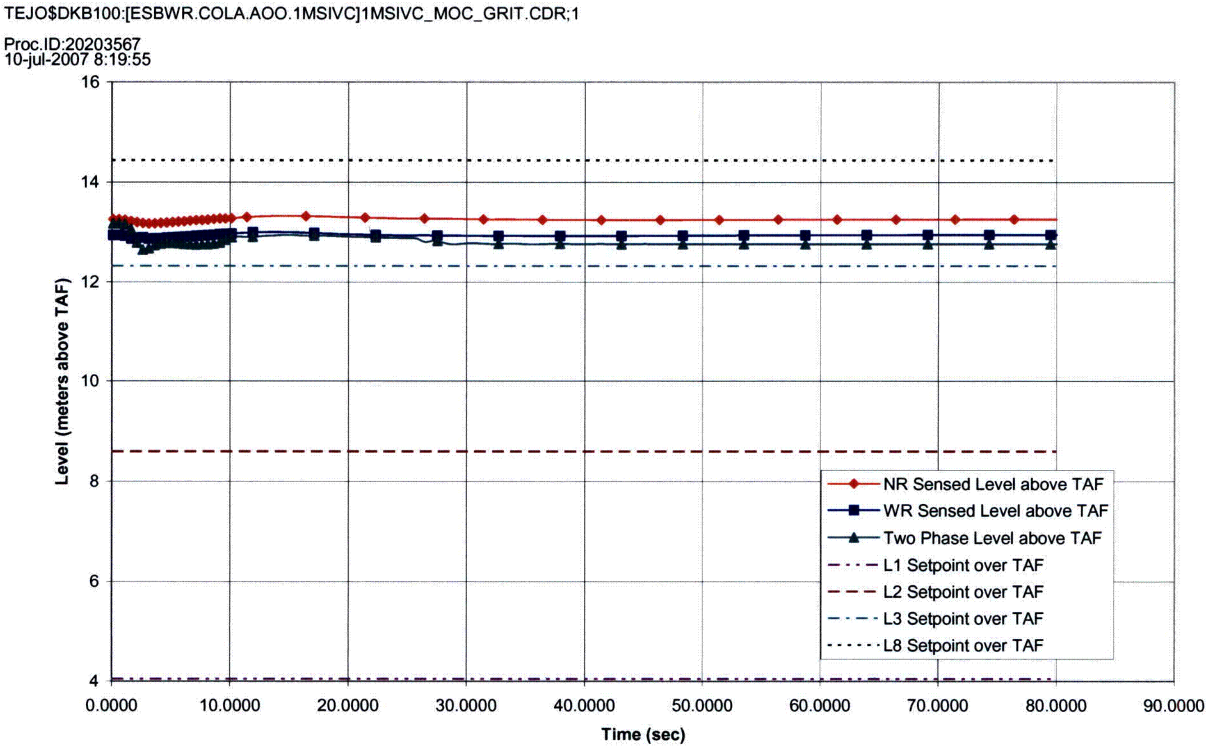


Figure 2.3-8c. One MSIV Closure

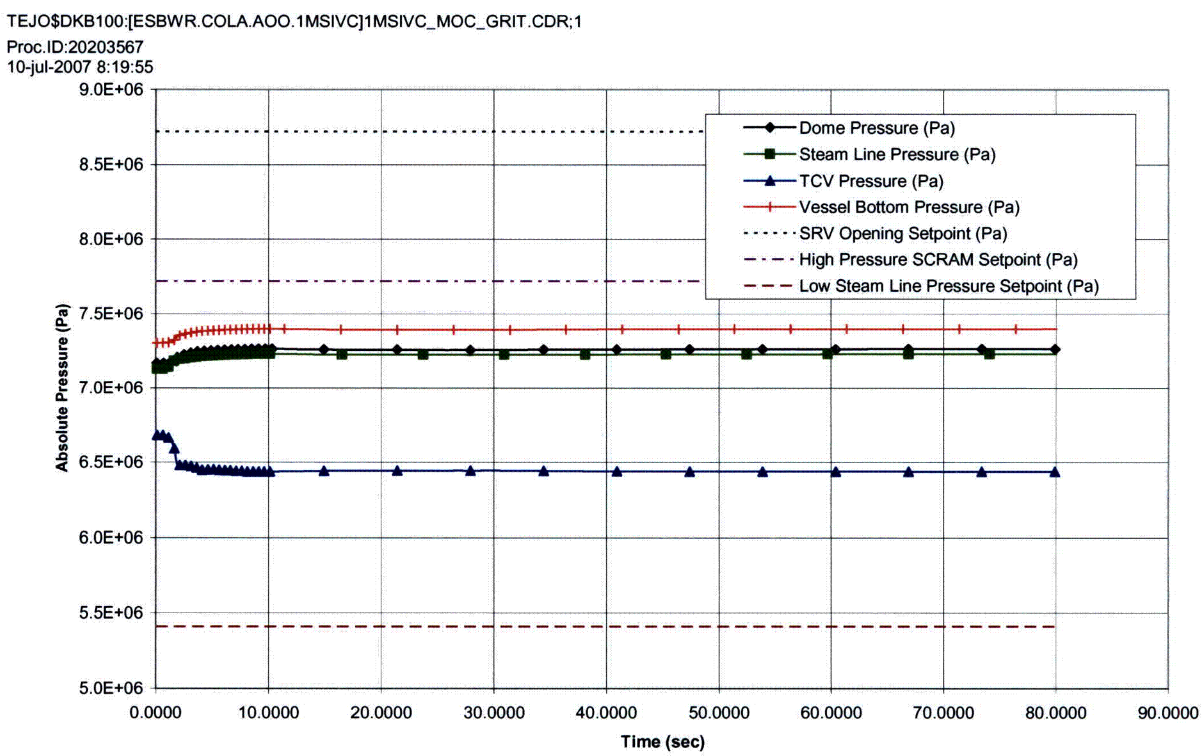


Figure 2.3-8d. One MSIV Closure

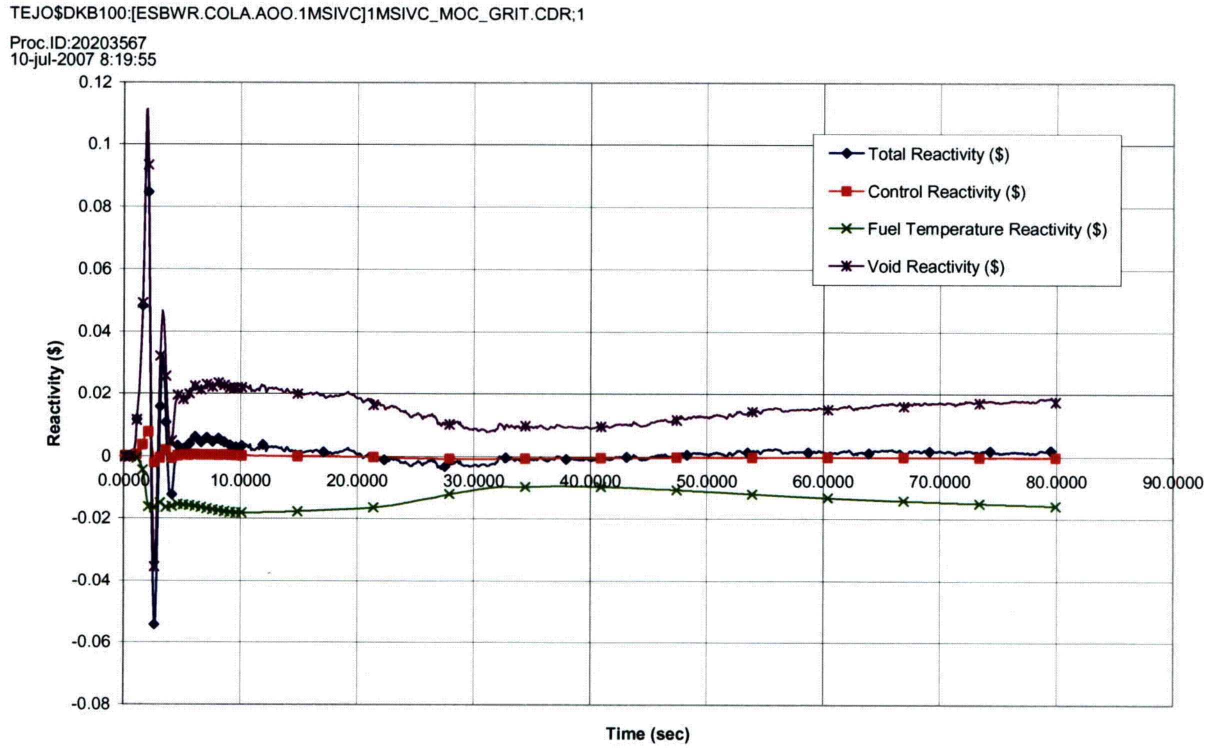


Figure 2.3-8e. One MSIV Closure

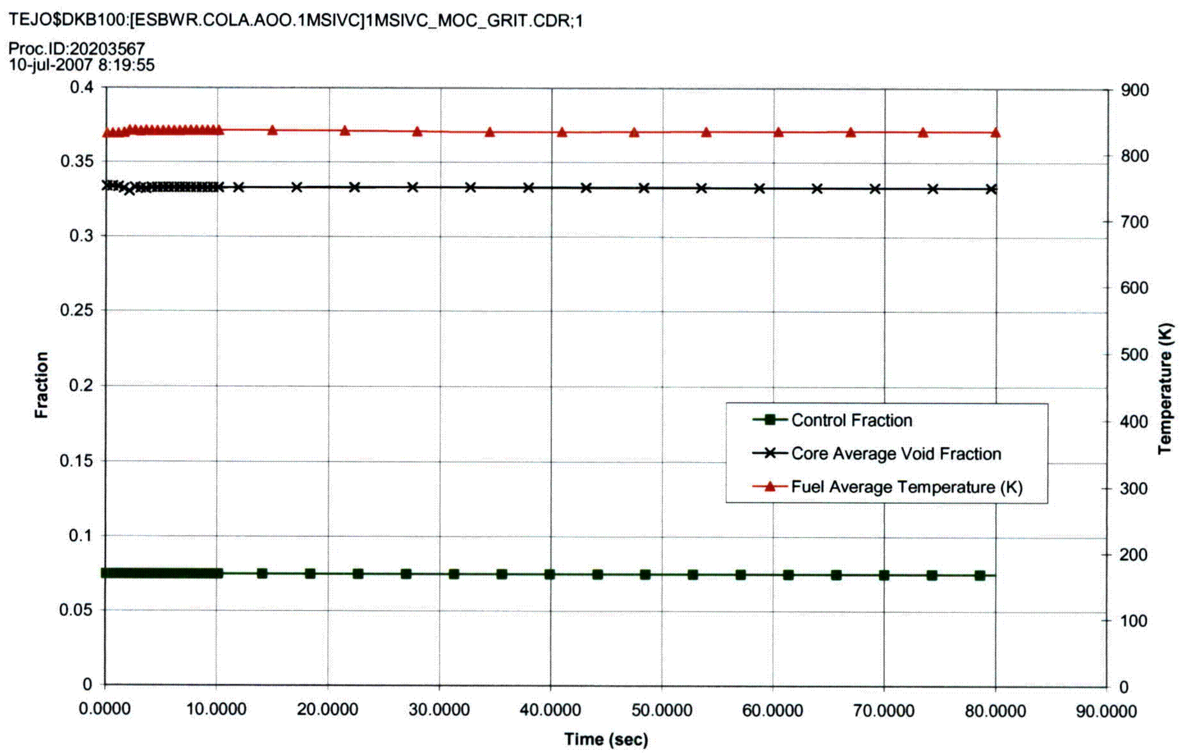


Figure 2.3-8f. One MSIV Closure

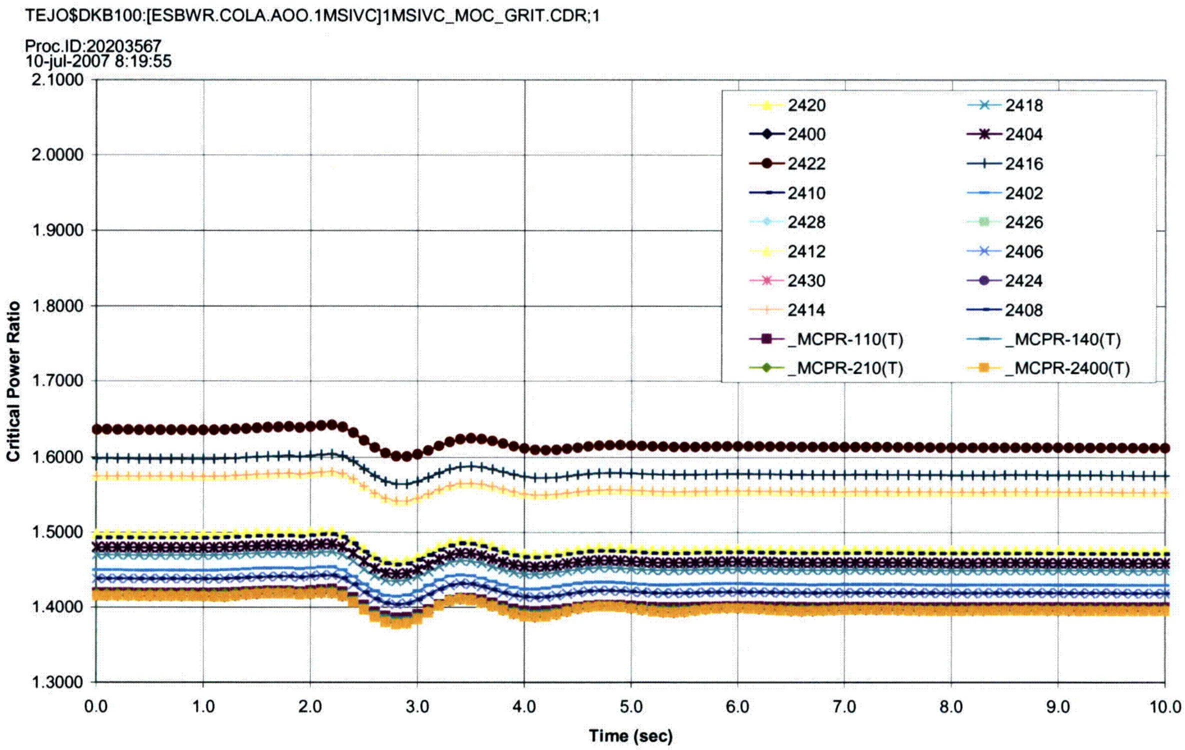


Figure 2.3-8g. One MSIV Closure

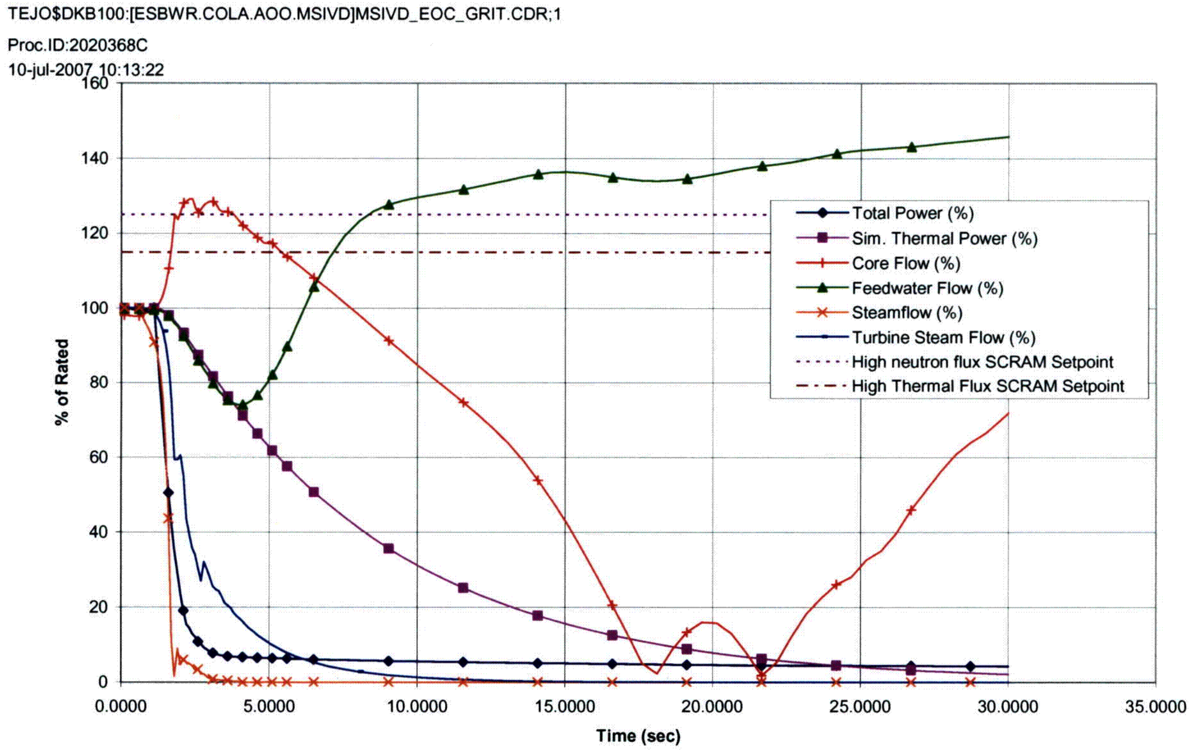


Figure 2.3-9a. MSIV Closure

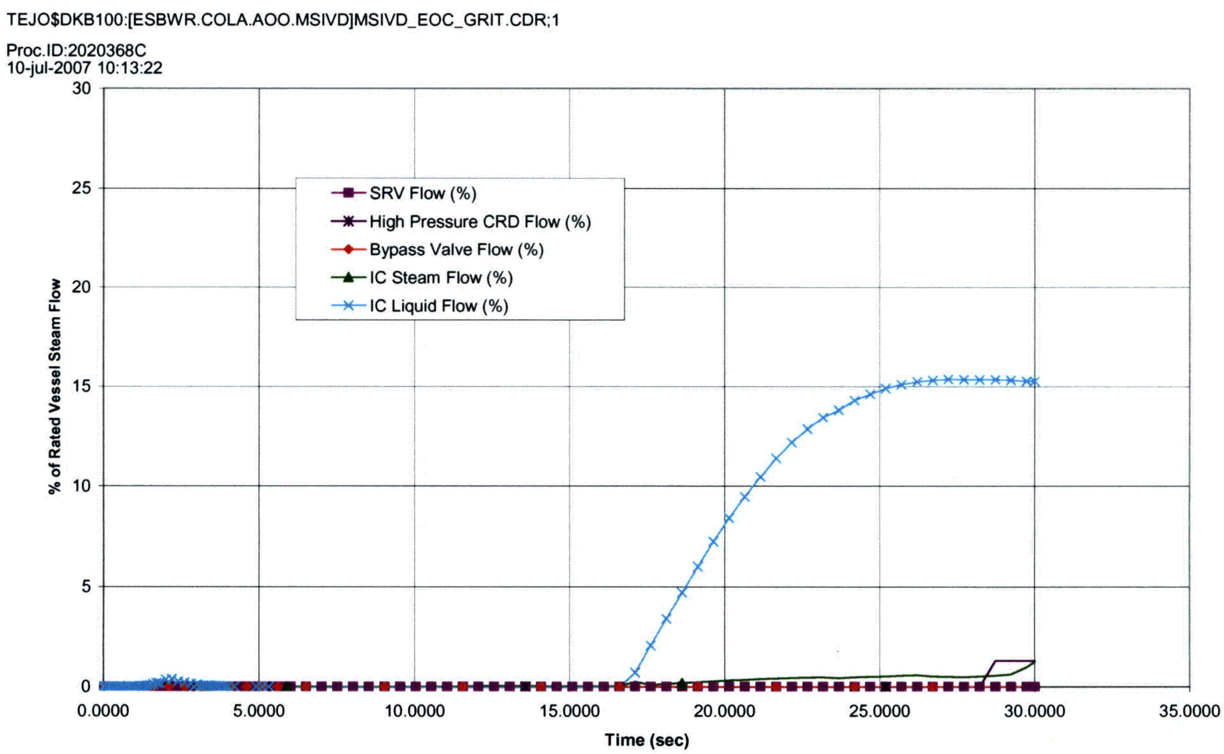


Figure 2.3-9b. MSIV Closure

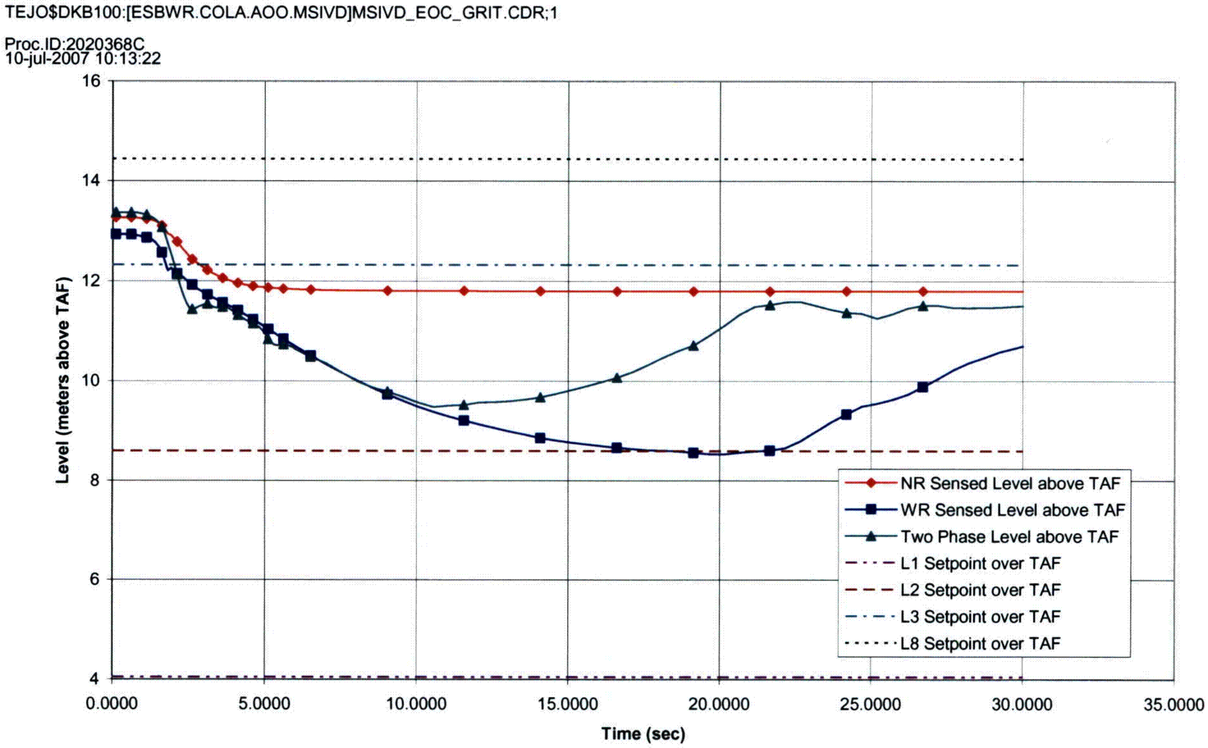


Figure 2.3-9c. MSIV Closure

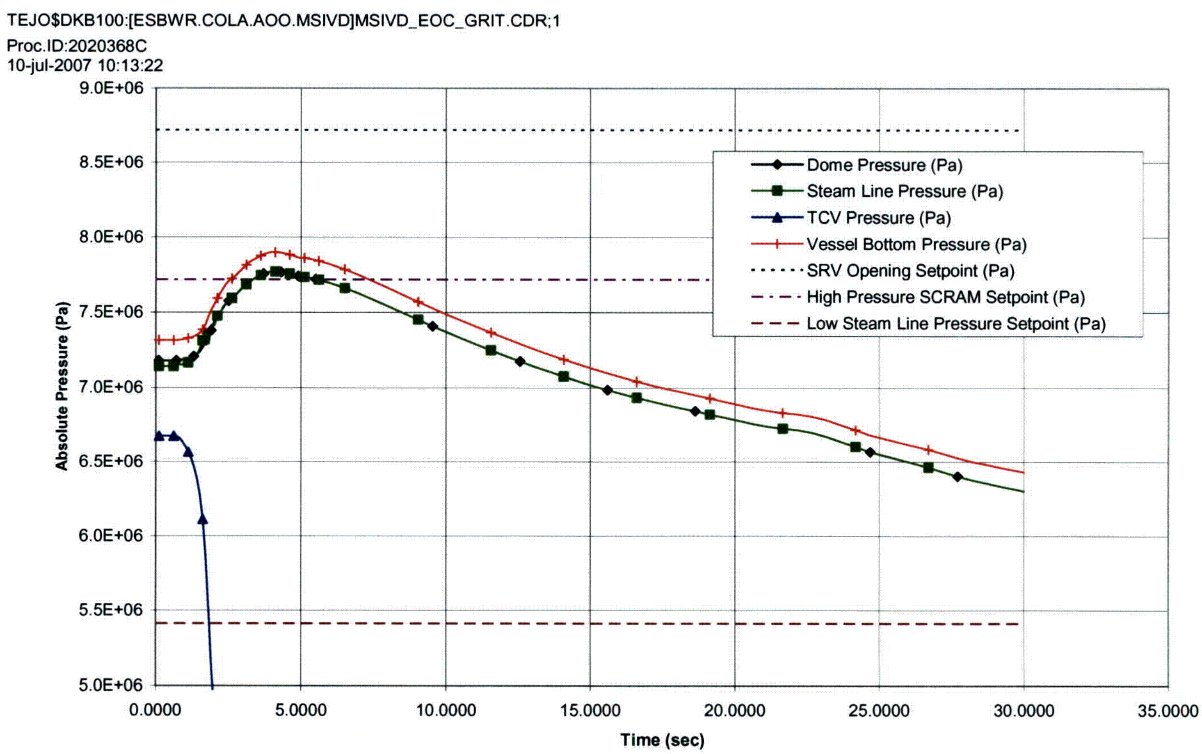


Figure 2.3-9d. MSIV Closure

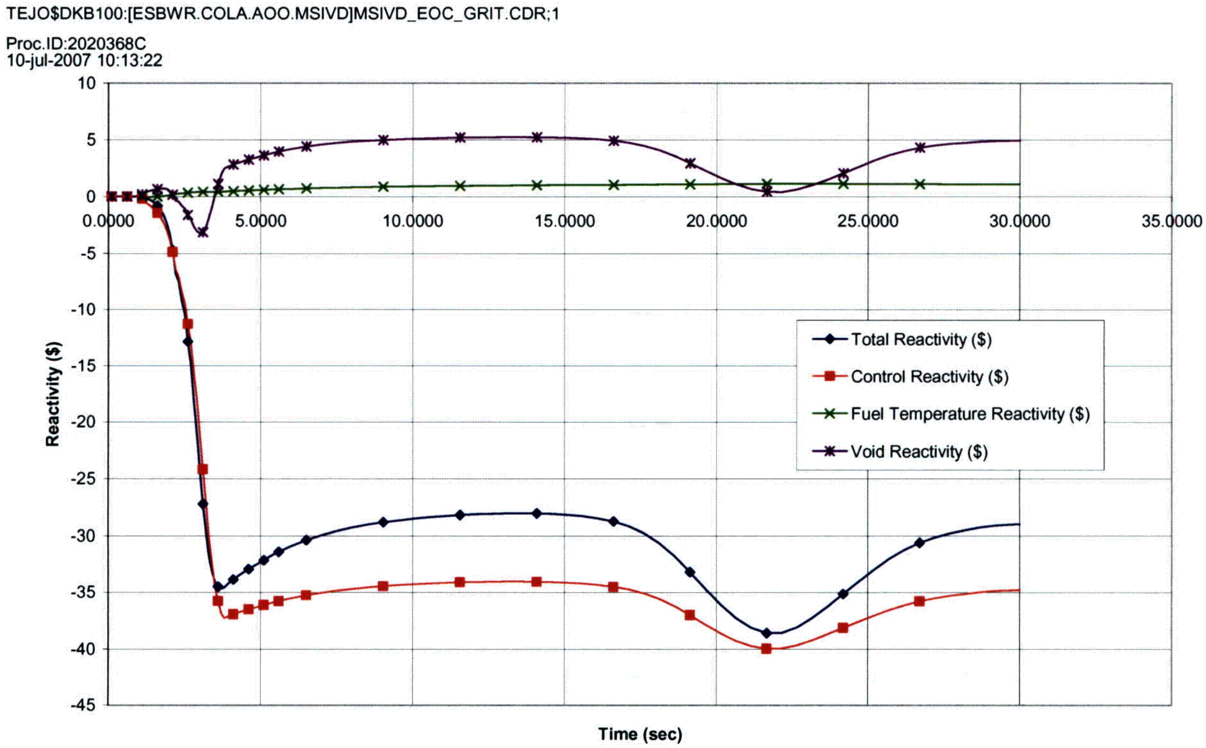


Figure 2.3-9e. MSIV Closure

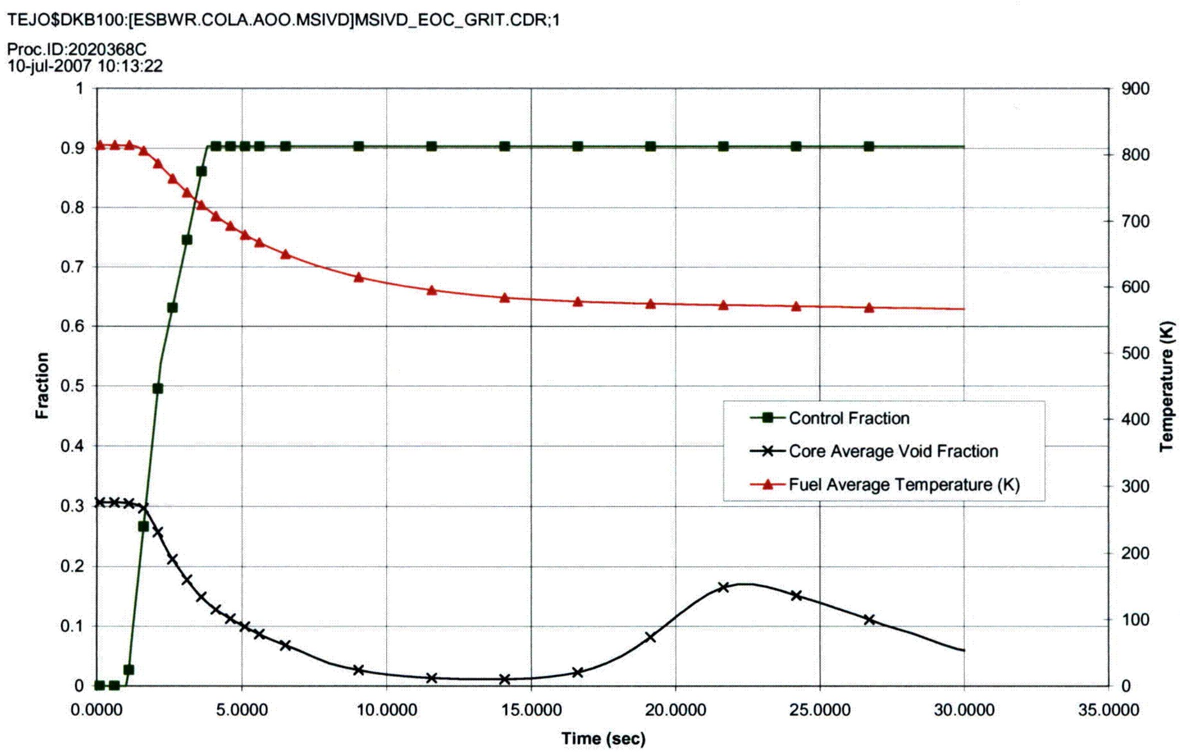


Figure 2.3-9f. MSIV Closure

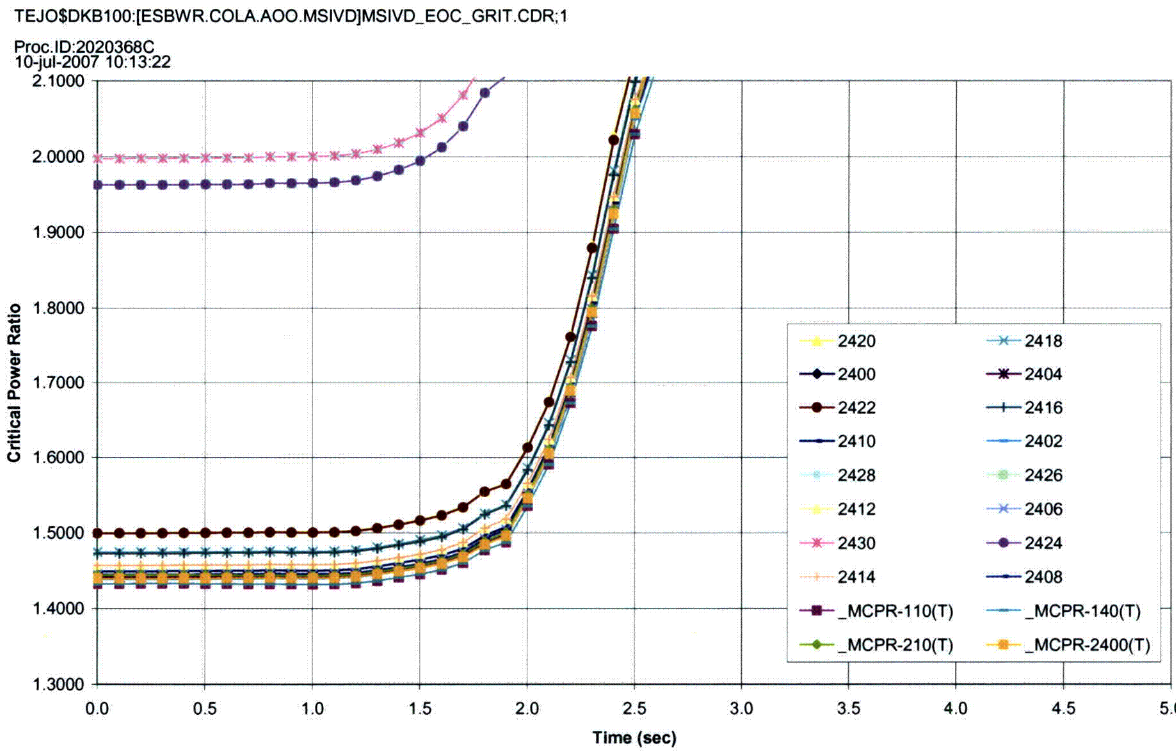


Figure 2.3-9g. MSIV Closure

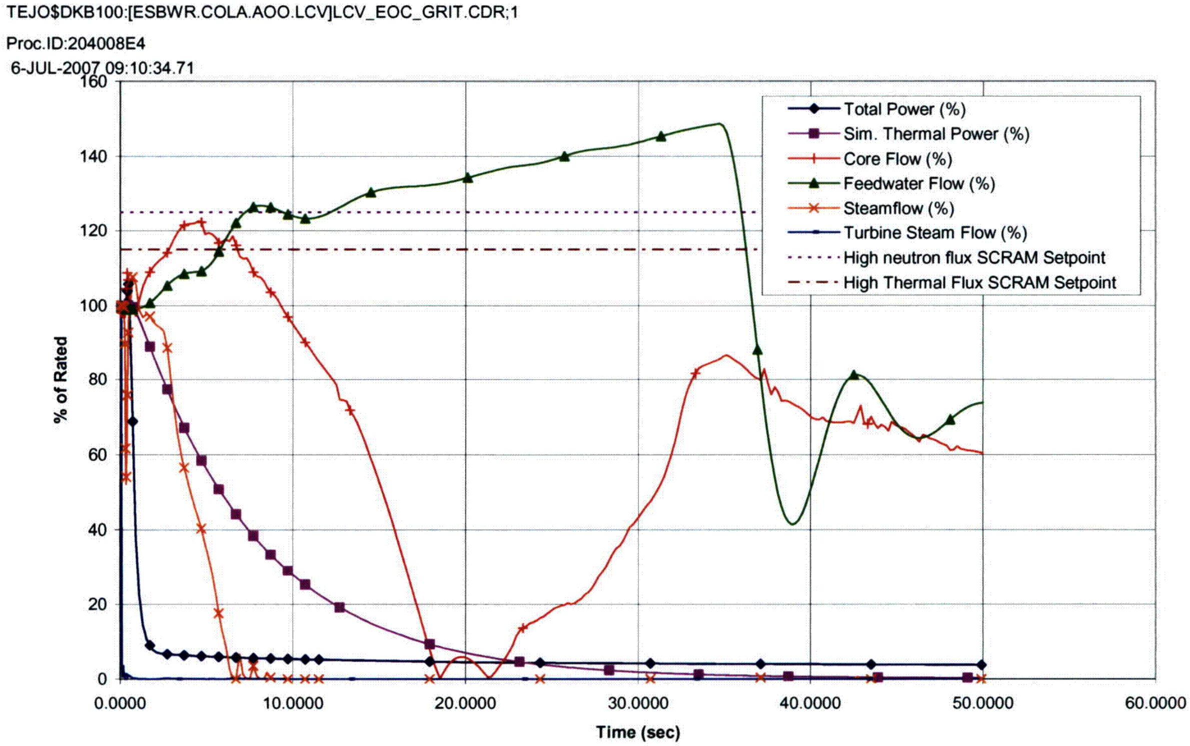


Figure 2.3-10a. Loss of Condenser Vacuum

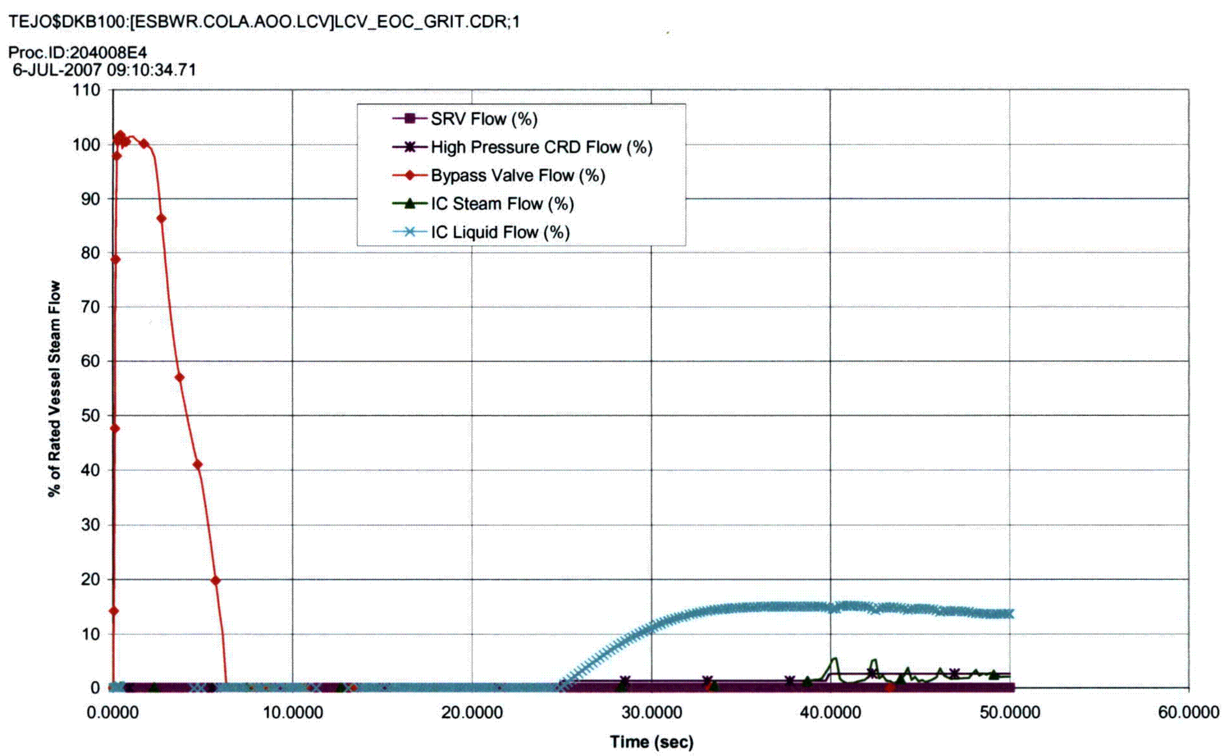


Figure 2.3-10b. Loss of Condenser Vacuum

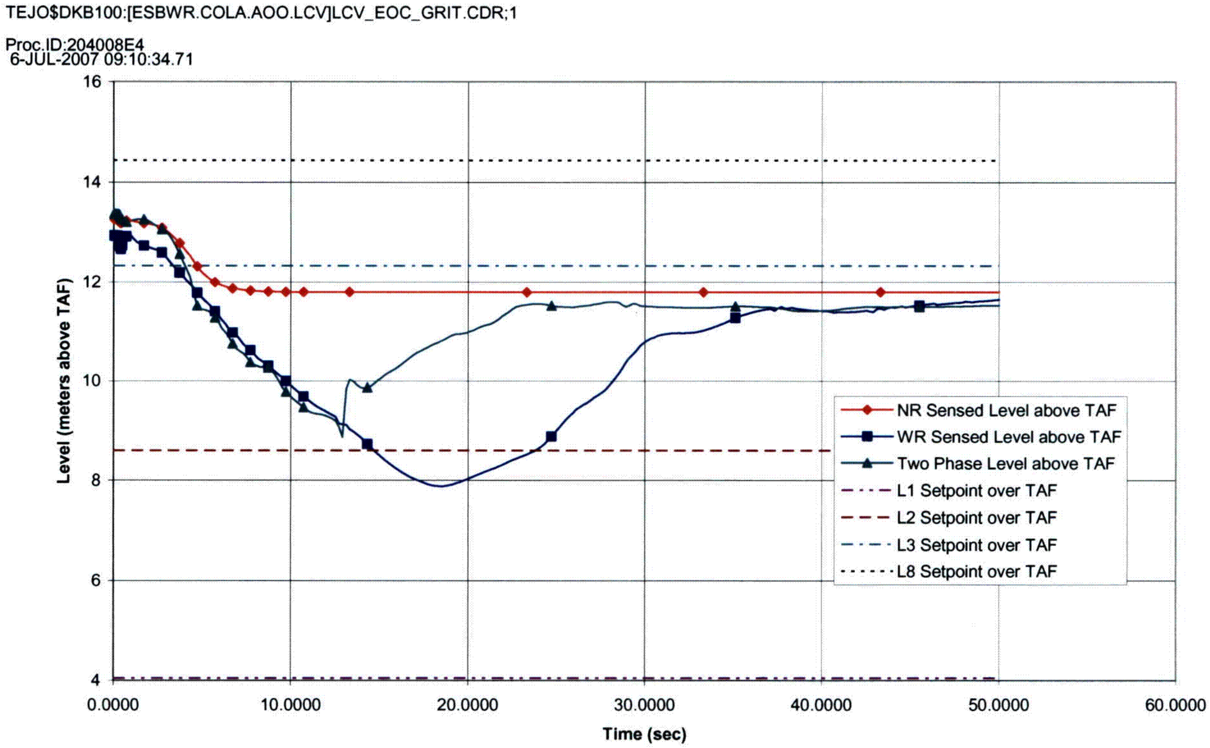


Figure 2.3-10c. Loss of Condenser Vacuum

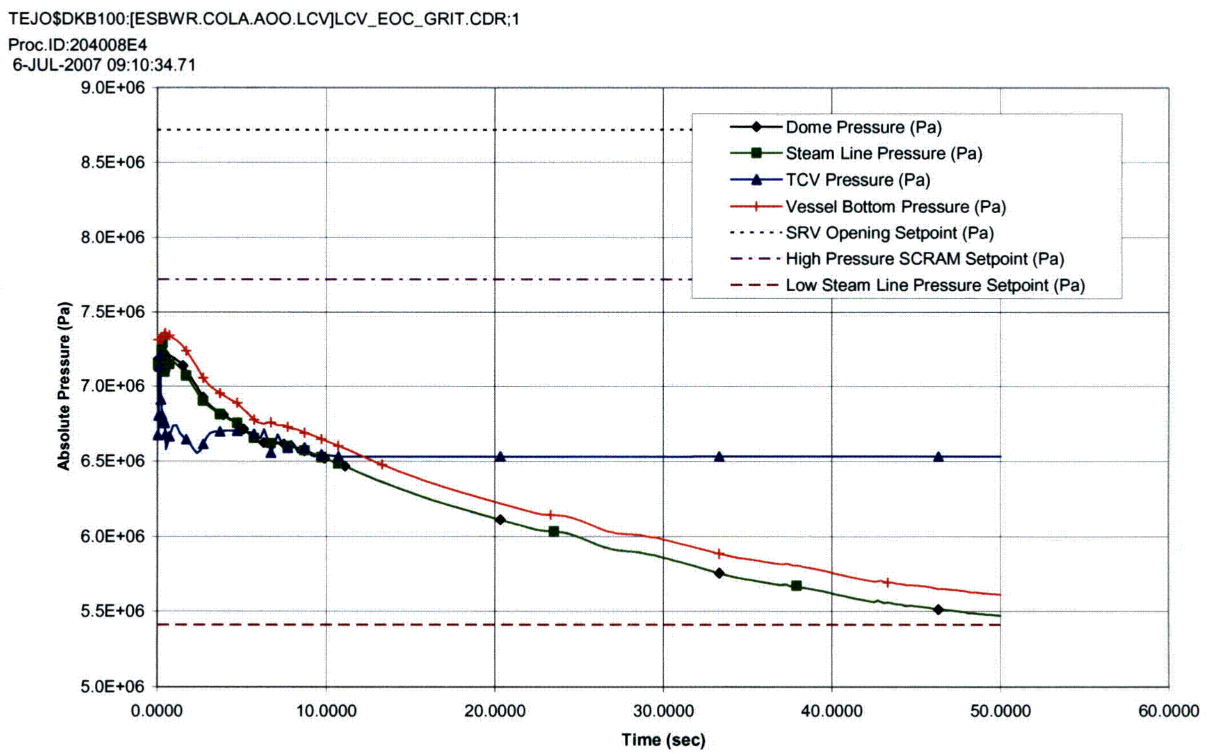


Figure 2.3-10d. Loss of Condenser Vacuum

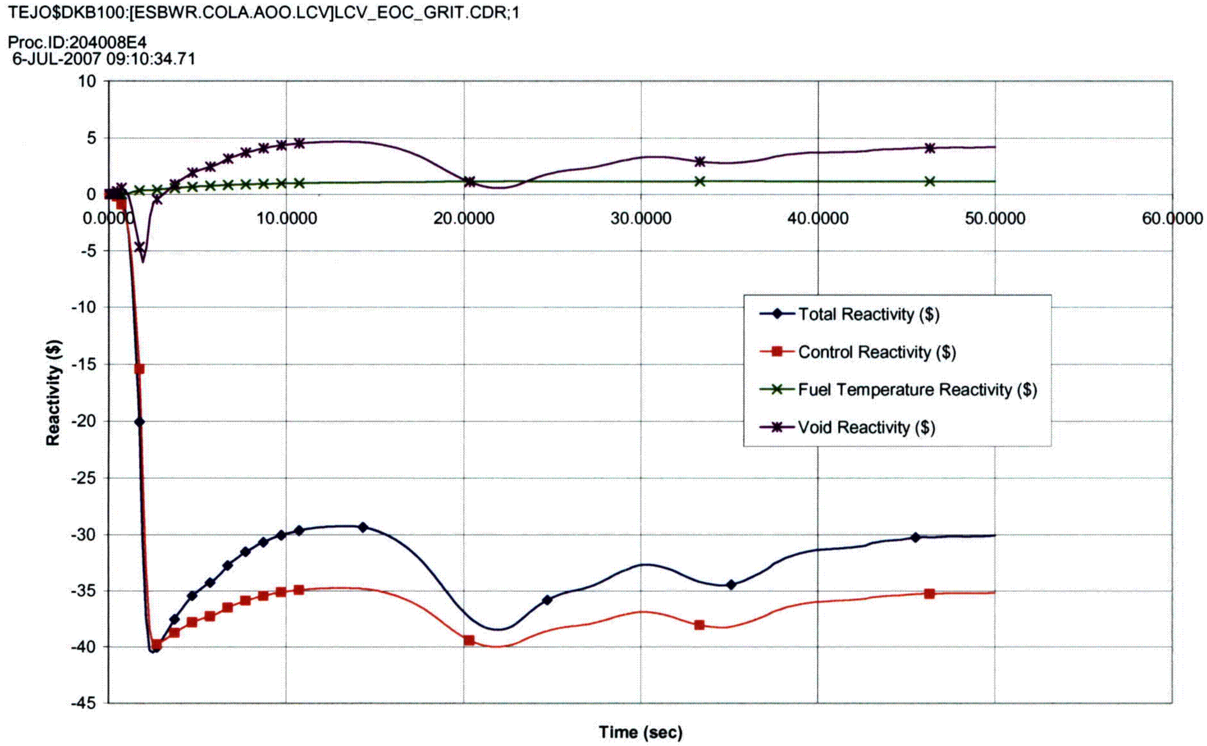


Figure 2.3-10e. Loss of Condenser Vacuum

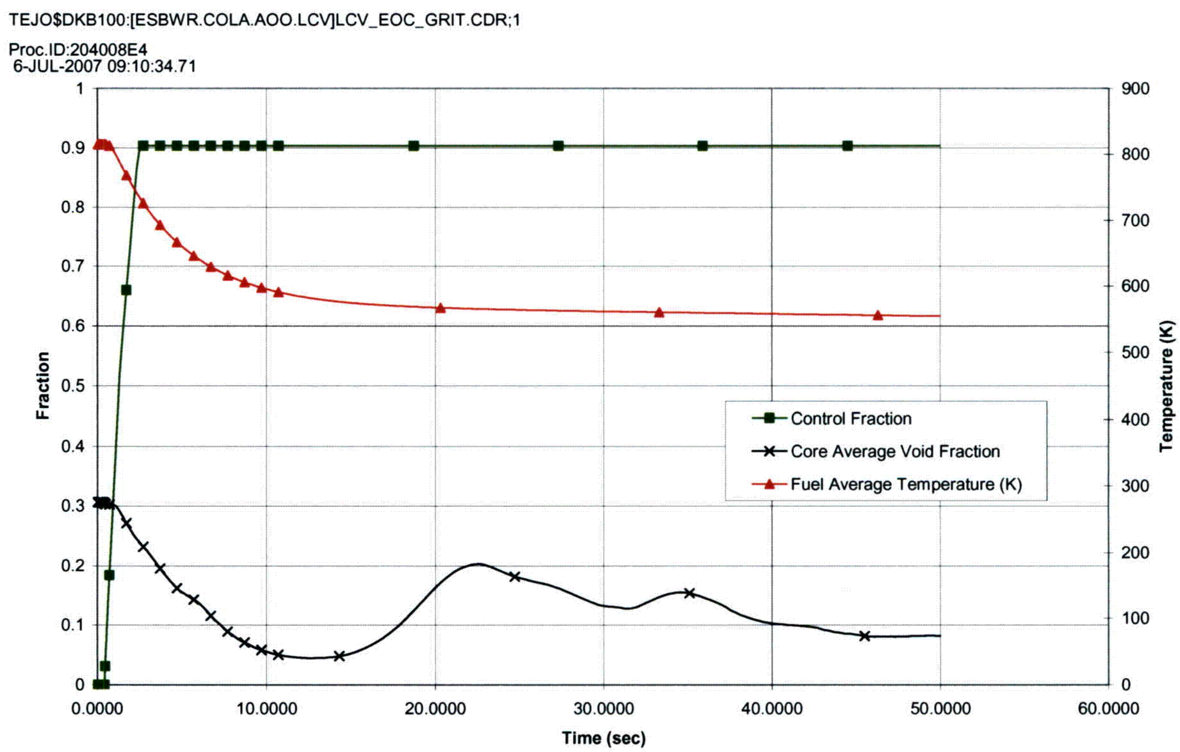


Figure 2.3-10f. Loss of Condenser Vacuum

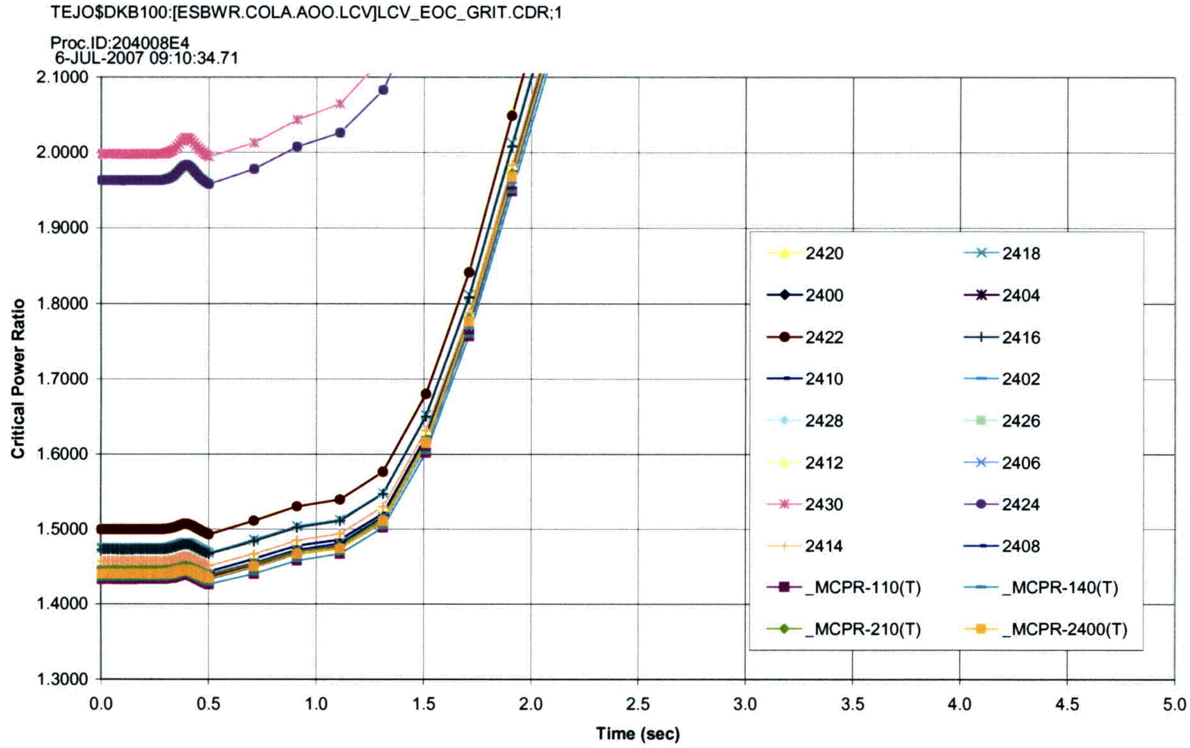


Figure 2.3-10g. Loss of Condenser Vacuum

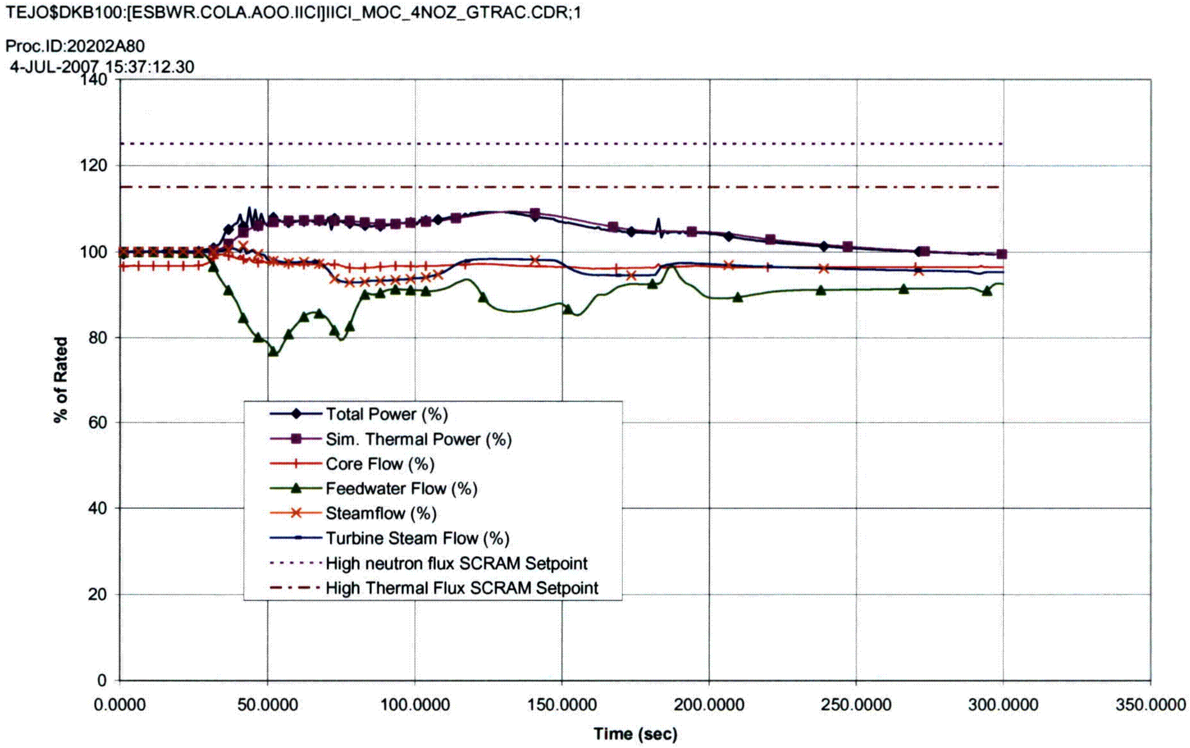


Figure 2.3-11a. Inadvertent Isolation Condenser Initiation

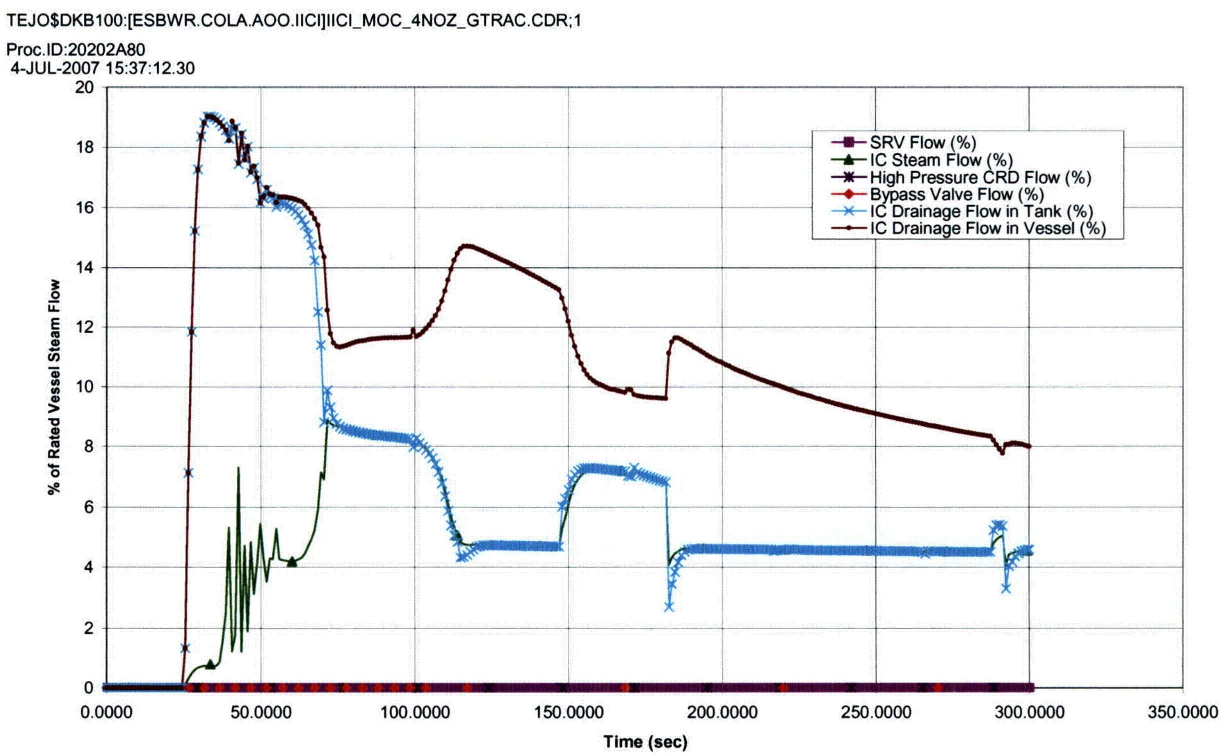


Figure 2.3-11b. Inadvertent Isolation Condenser Initiation

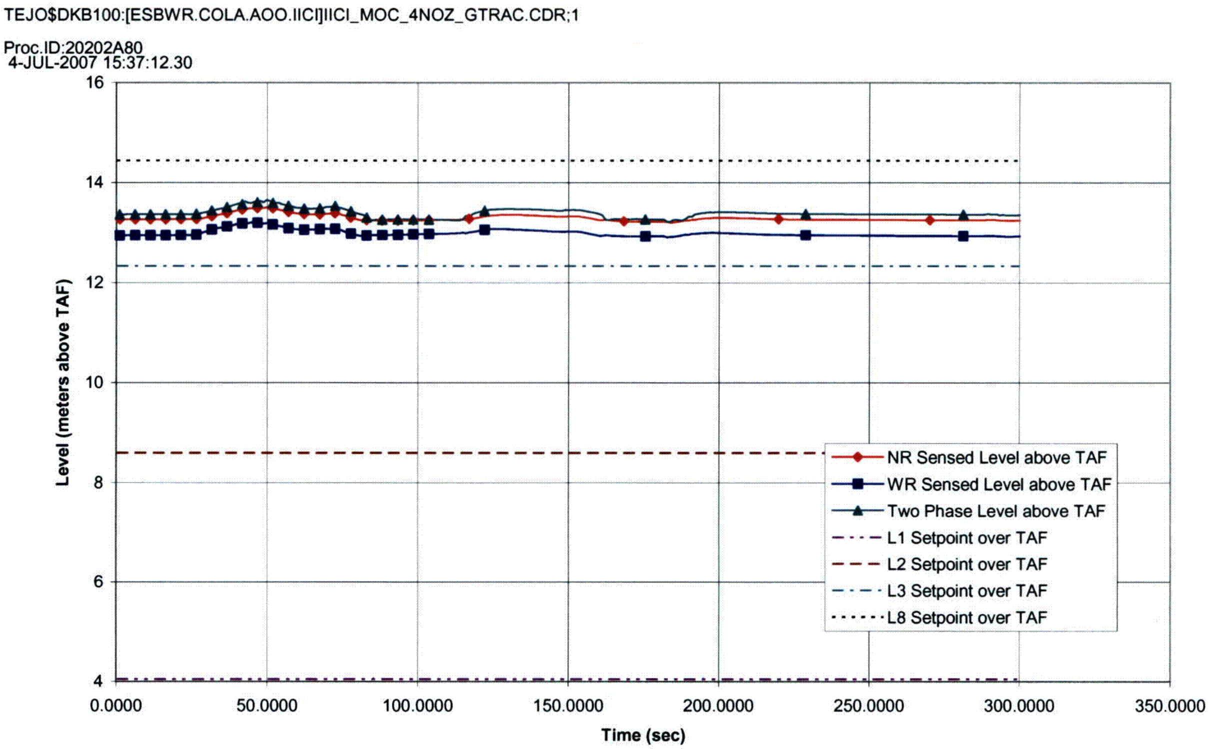


Figure 2.3-11c. Inadvertent Isolation Condenser Initiation

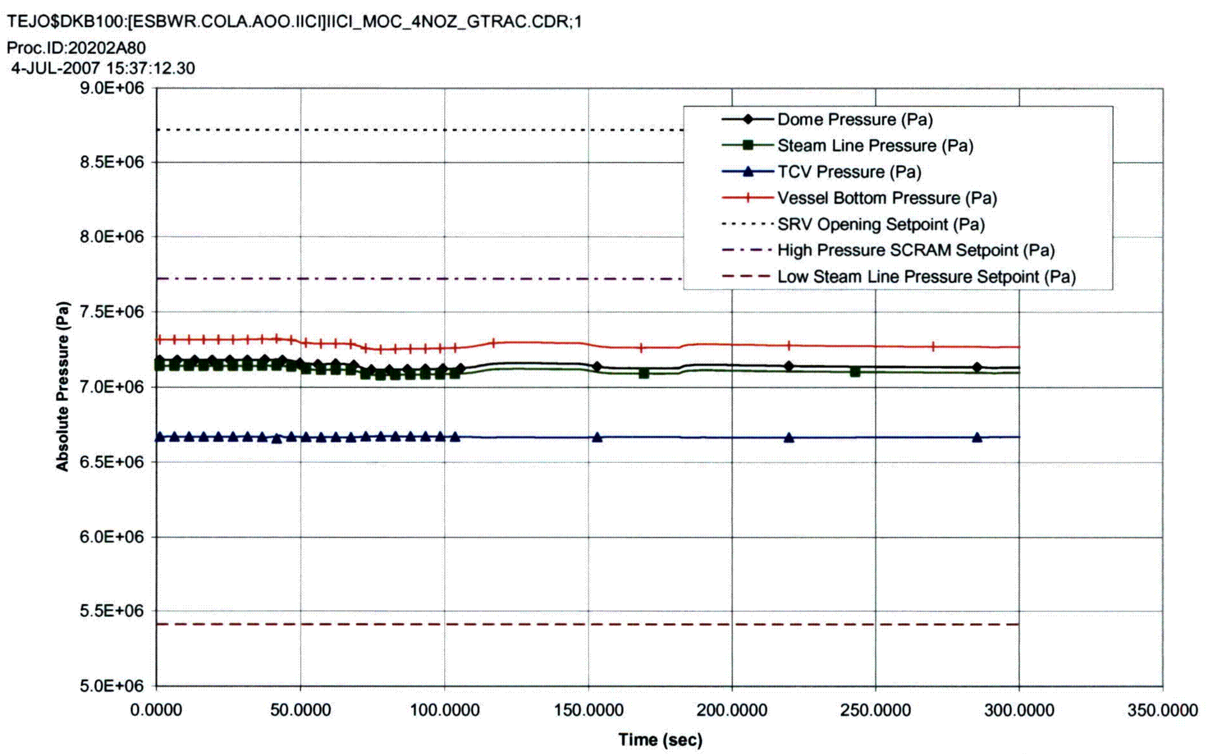


Figure 2.3-11d. Inadvertent Isolation Condenser Initiation

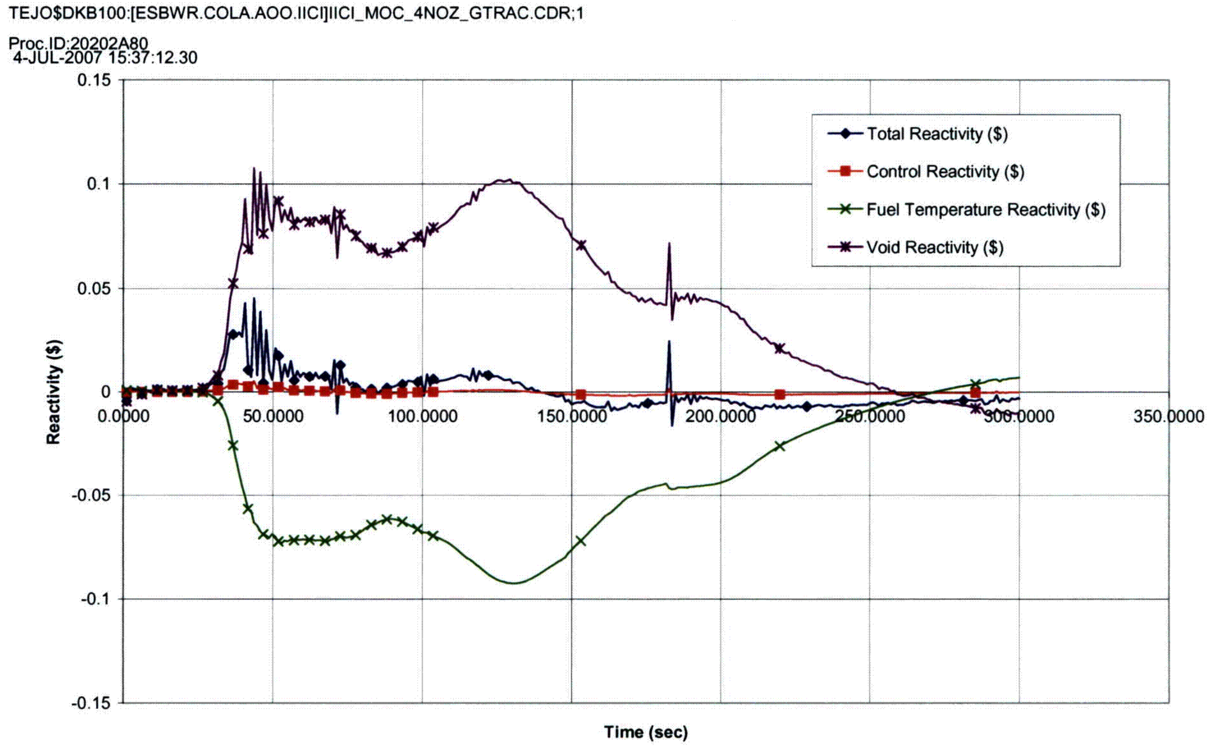


Figure 2.3-11e. Inadvertent Isolation Condenser Initiation

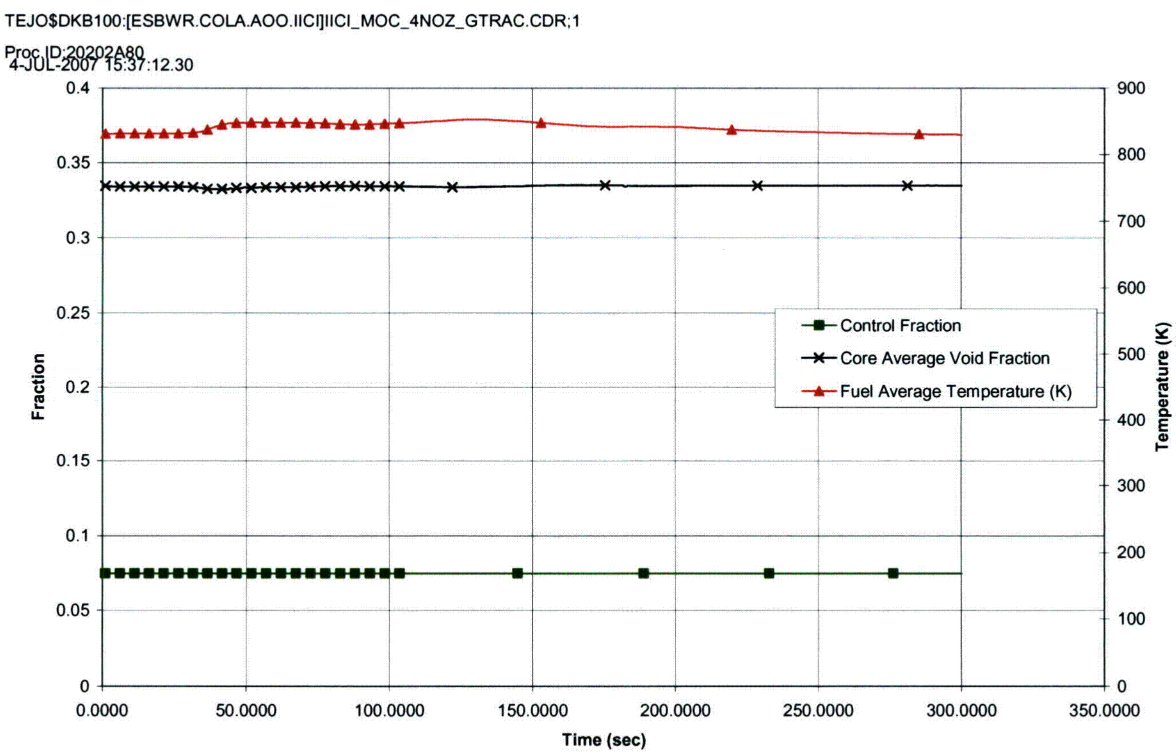


Figure 2.3-11f. Inadvertent Isolation Condenser Initiation

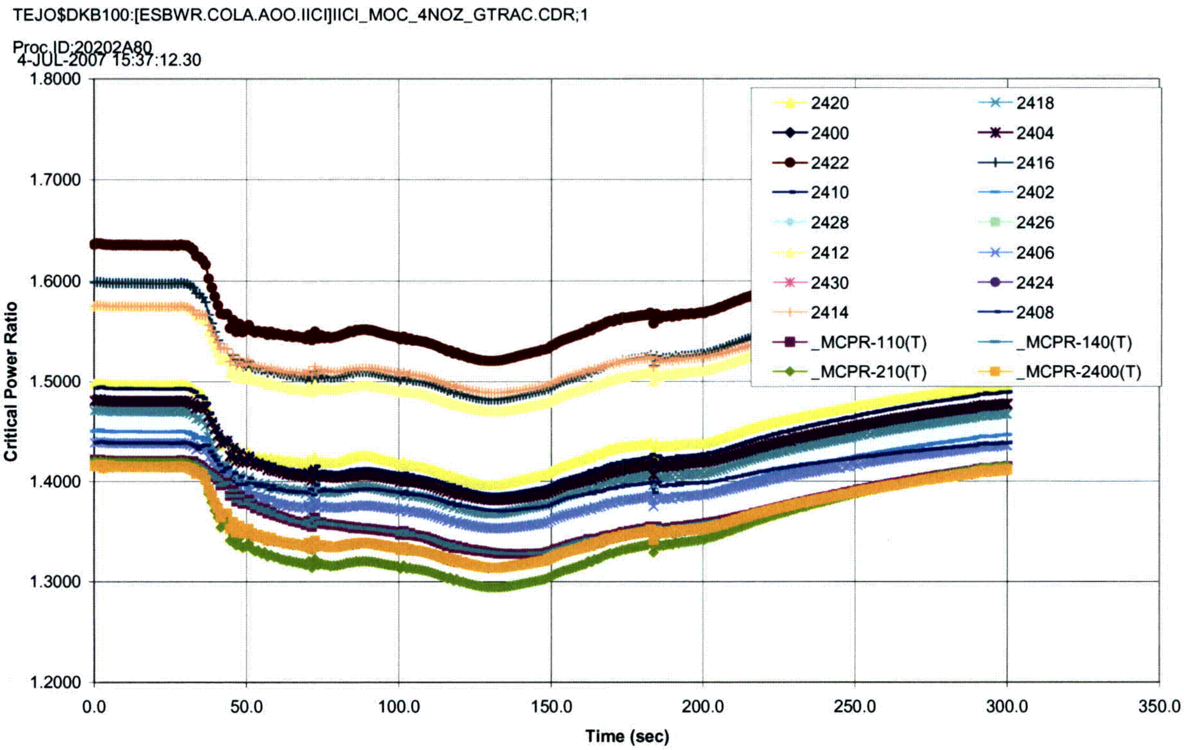


Figure 2.3-11g. Inadvertent Isolation Condenser Initiation

Figure 2.3-12. Simplified Block Diagram of Fault-Tolerant Digital Controller System

Data in DCD applies to the initial core.

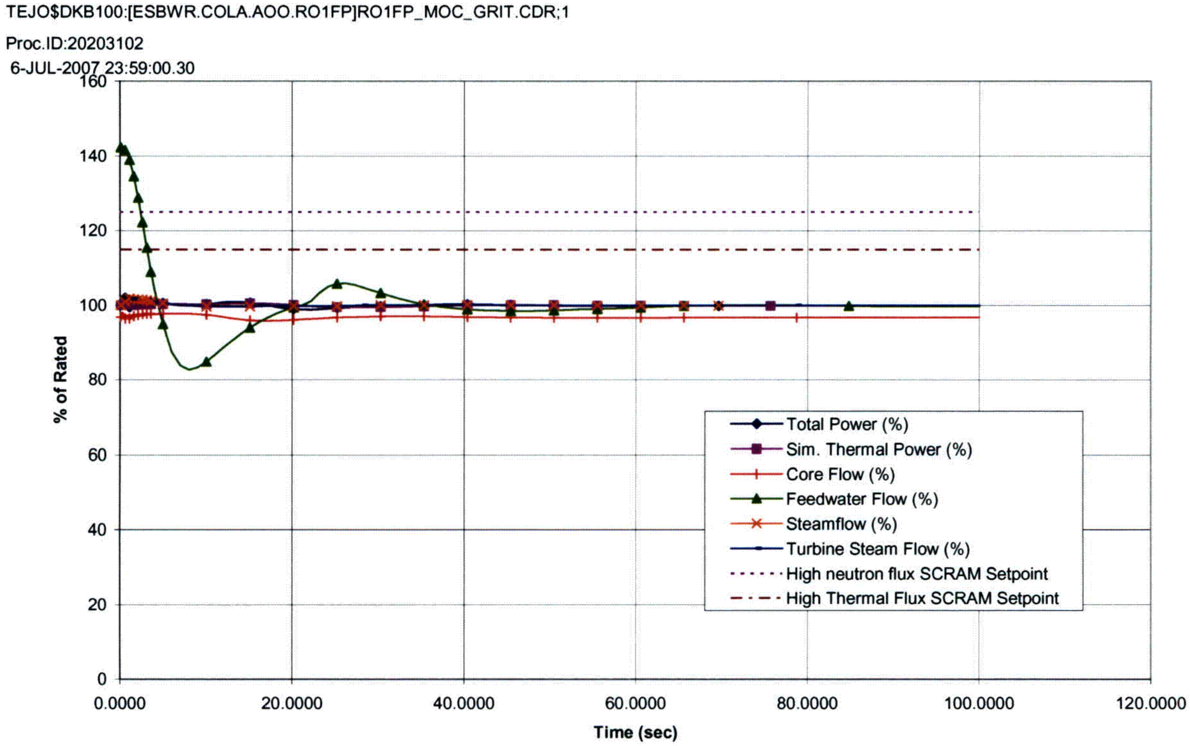


Figure 2.3-13a. Runout of One Feedwater Pump

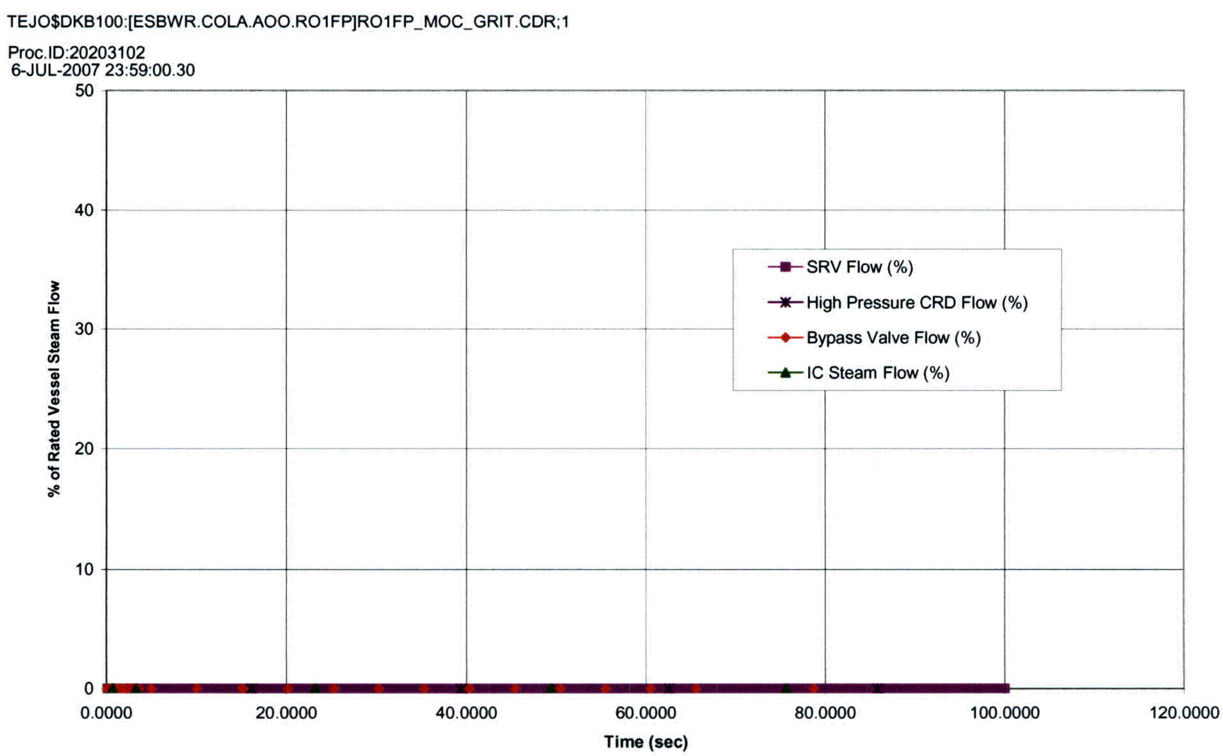


Figure 2.3-13b. Runout of One Feedwater Pump

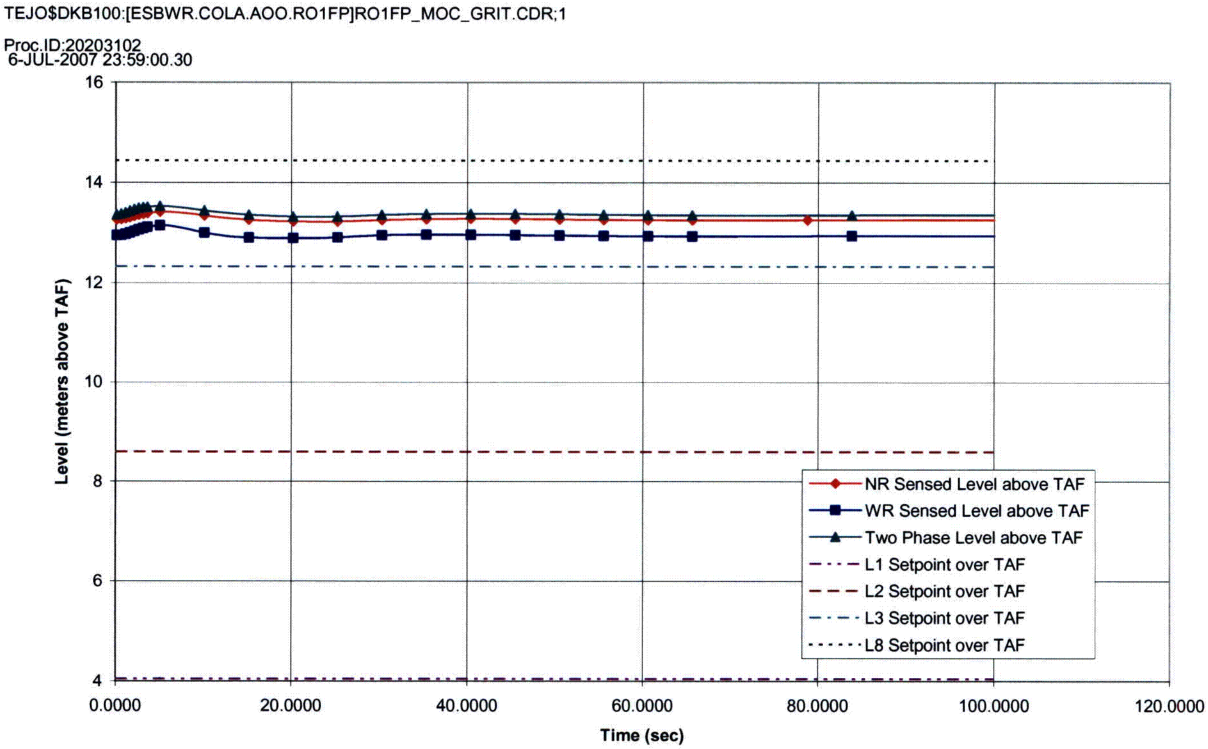


Figure 2.3-13c. Runout of One Feedwater Pump

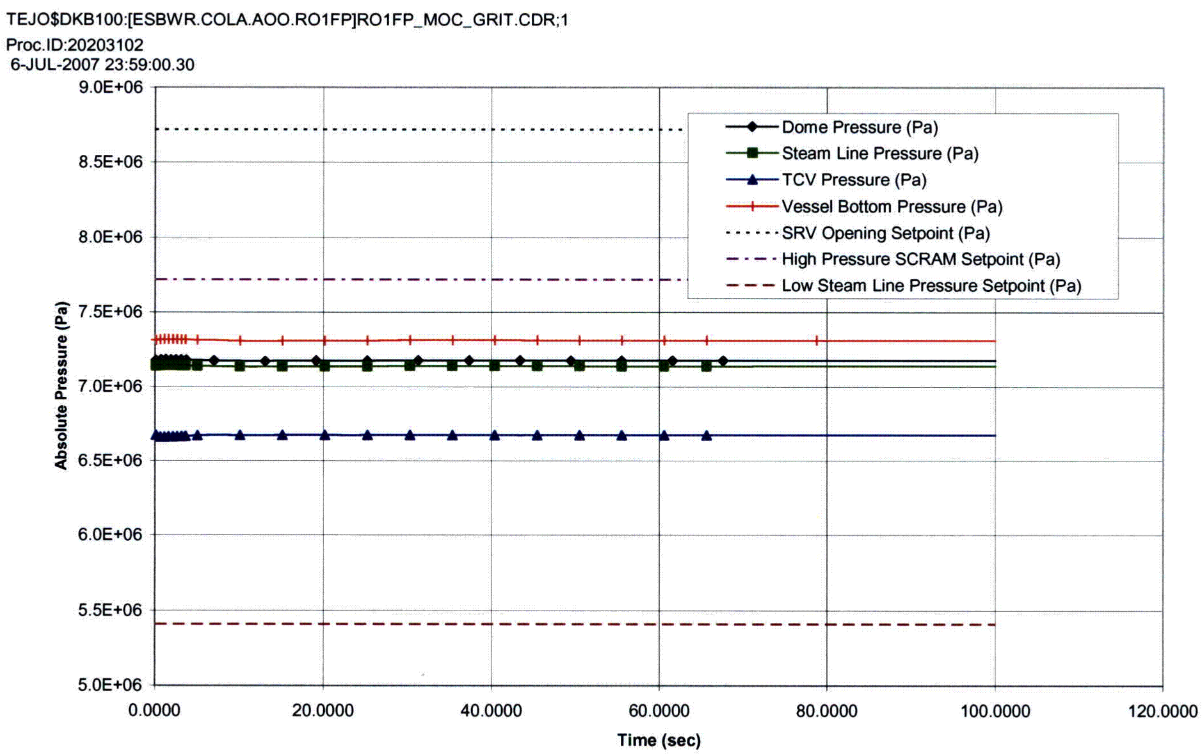


Figure 2.3-13d. Runout of One Feedwater Pump

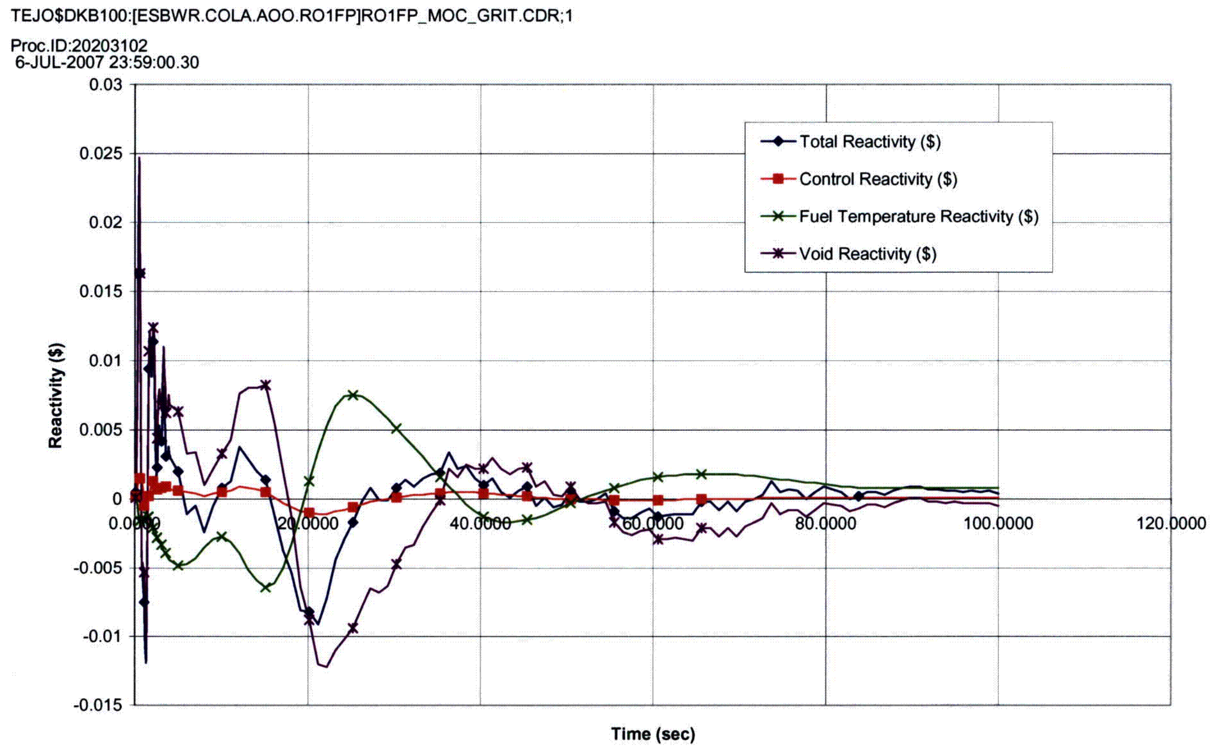


Figure 2.3-13e. Runout of One Feedwater Pump

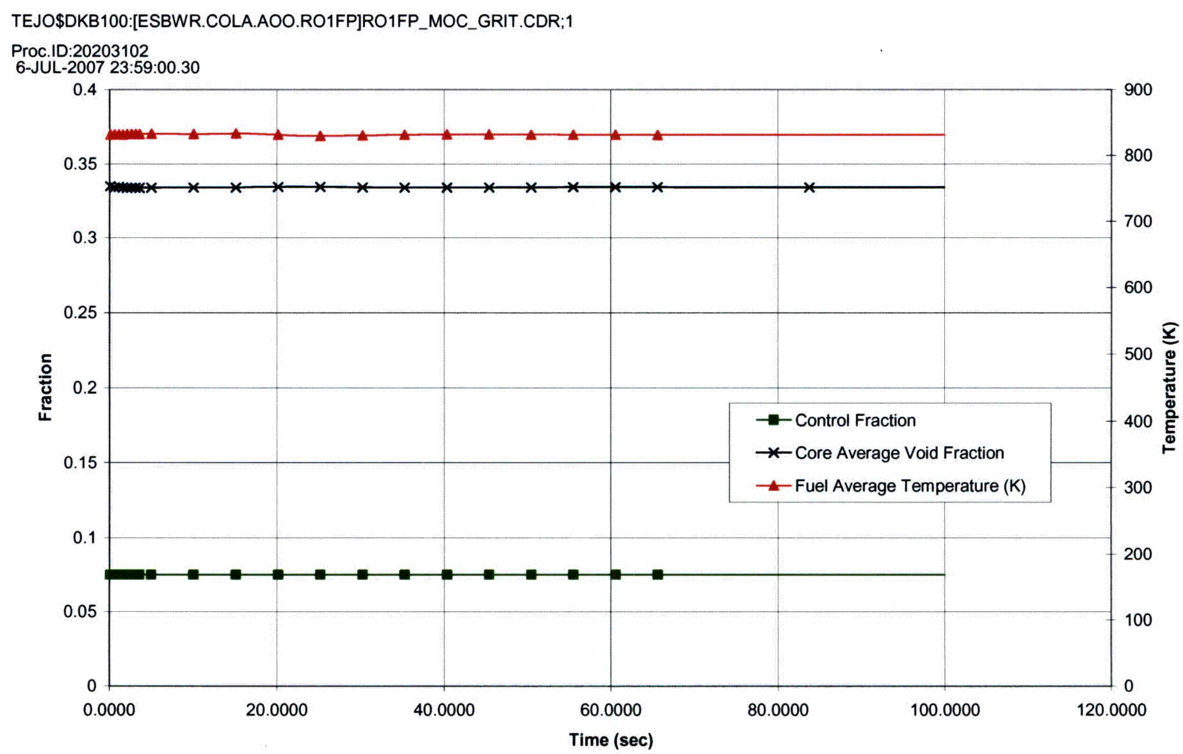


Figure 2.3-13f. Runout of One Feedwater Pump

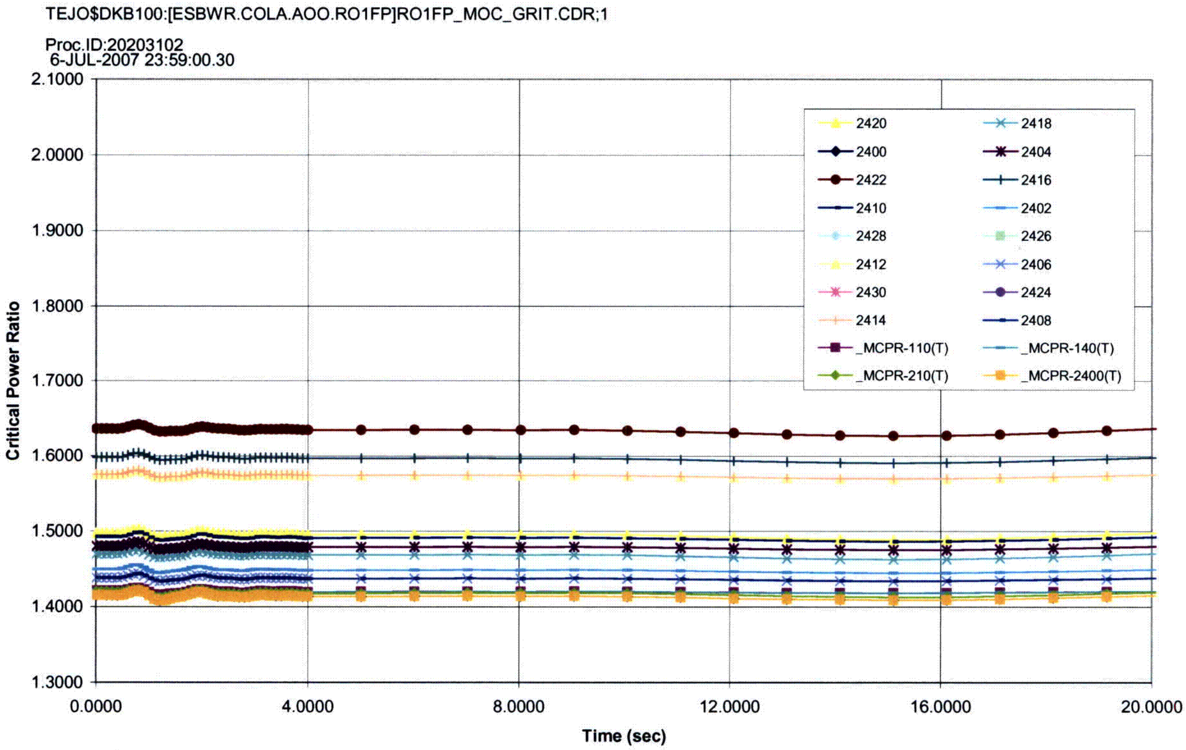


Figure 2.3-13g. Runout of One Feedwater Pump

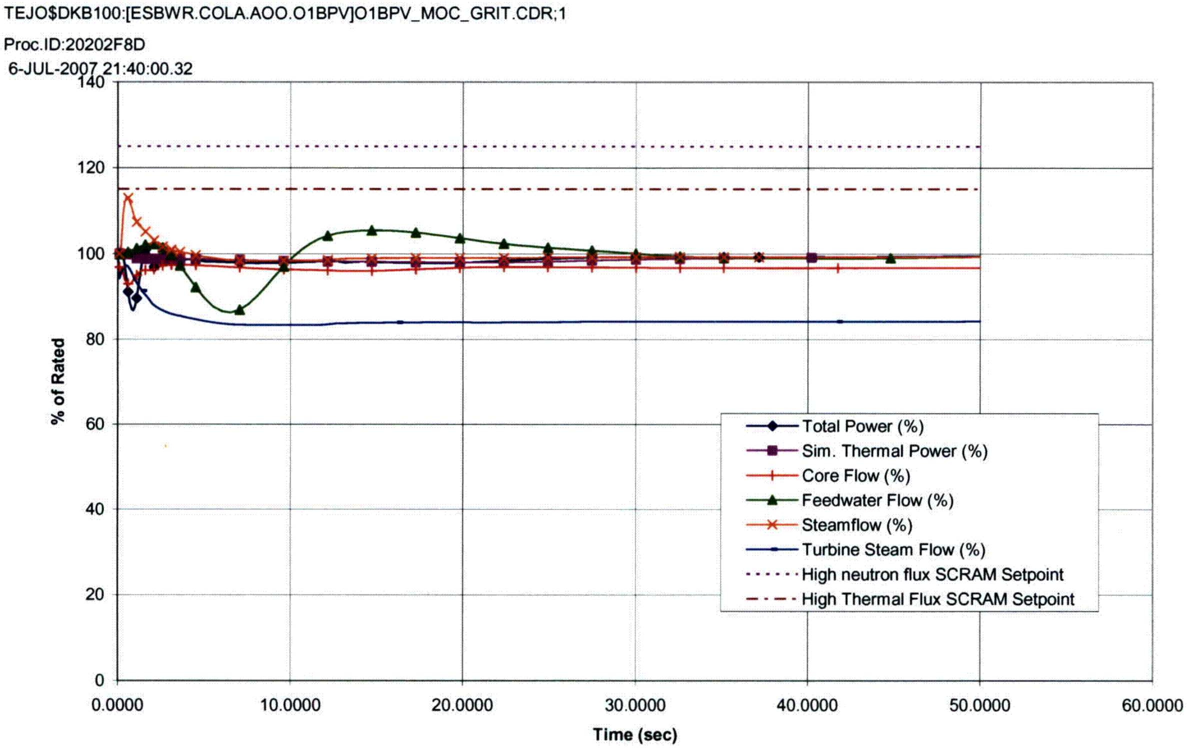


Figure 2.3-14a. Opening of One Turbine Control or Bypass Valve

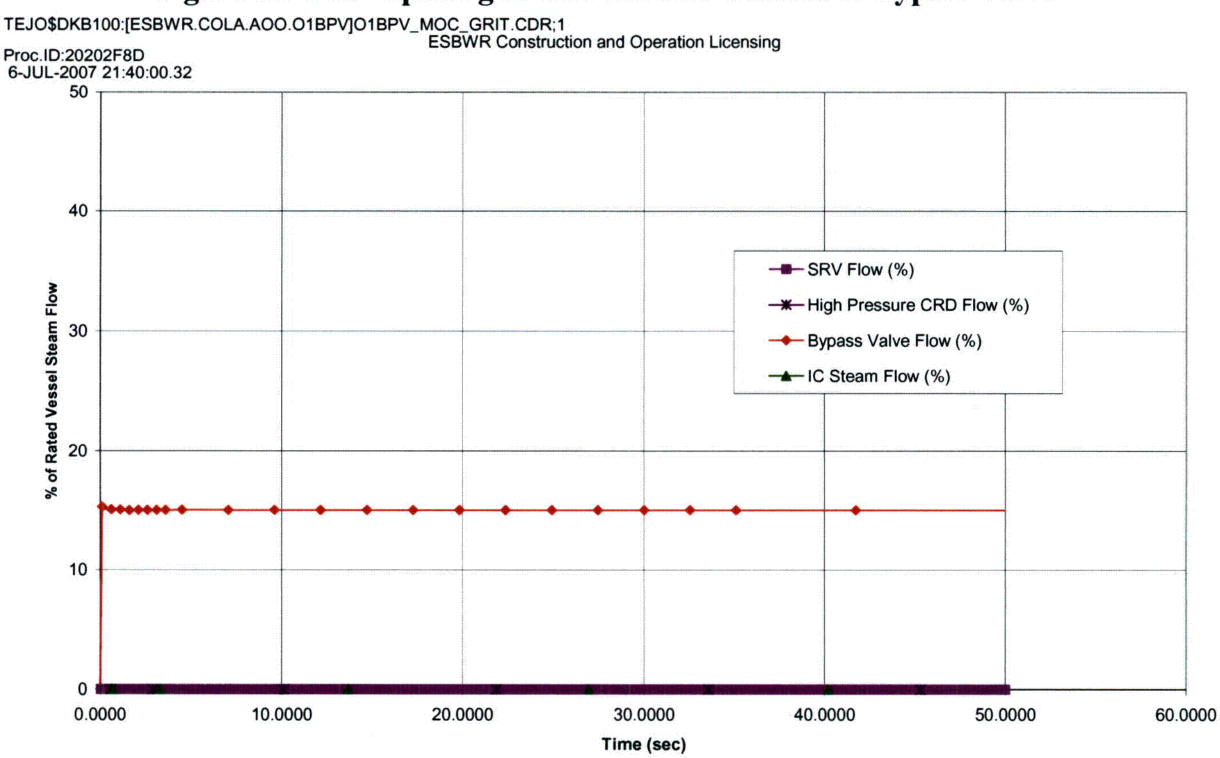


Figure 2.3-14b. Opening of One Turbine Control or Bypass Valve

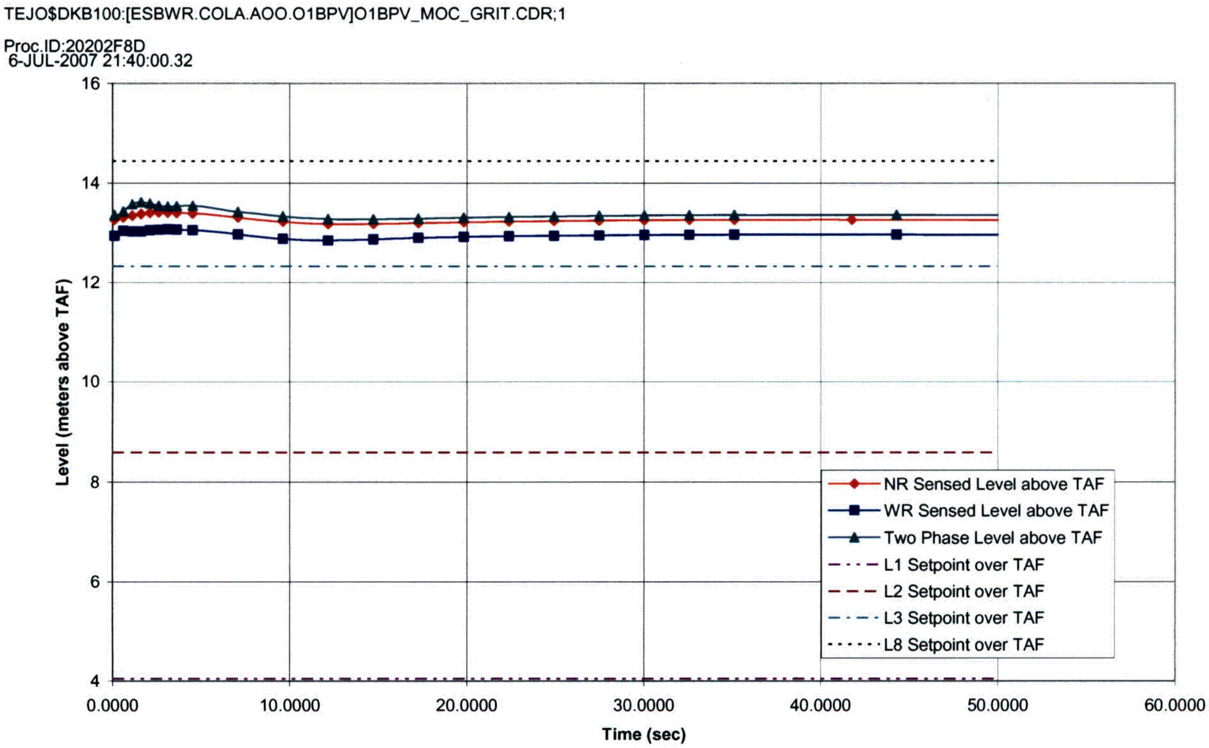


Figure 2.3-14c. Opening of One Turbine Control or Bypass Valve

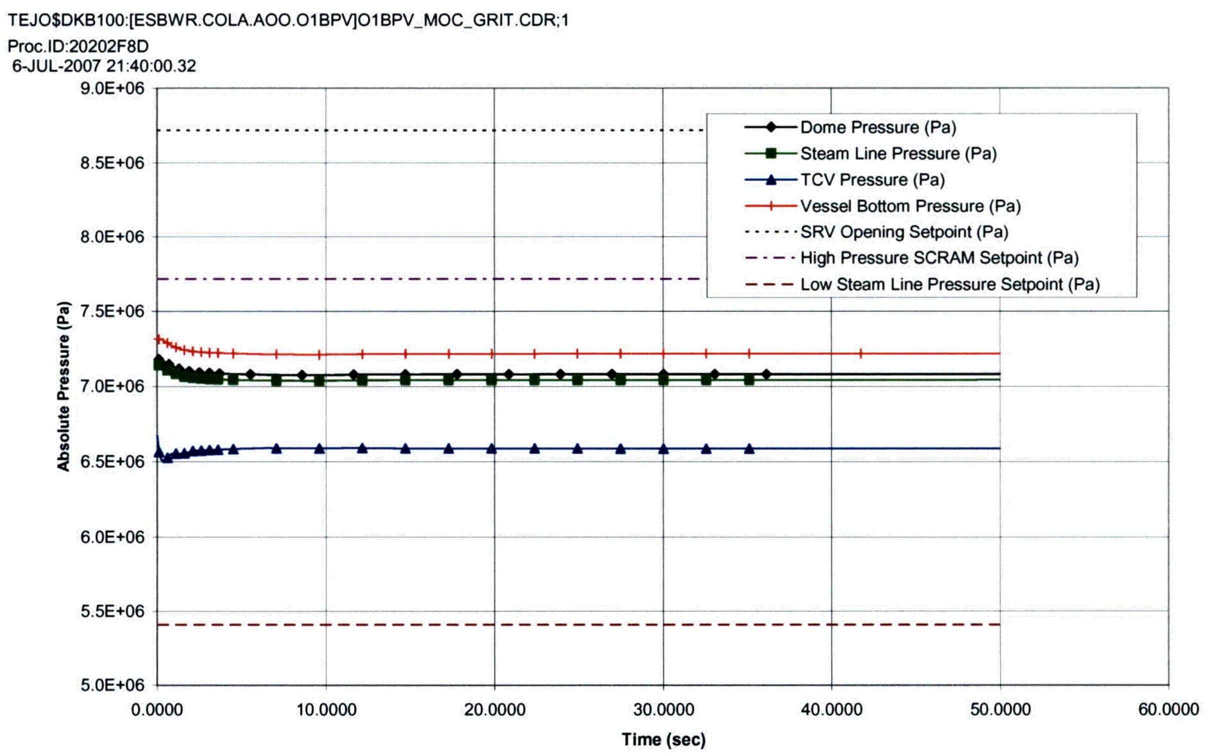


Figure 2.3-14d. Opening of One Turbine Control or Bypass Valve

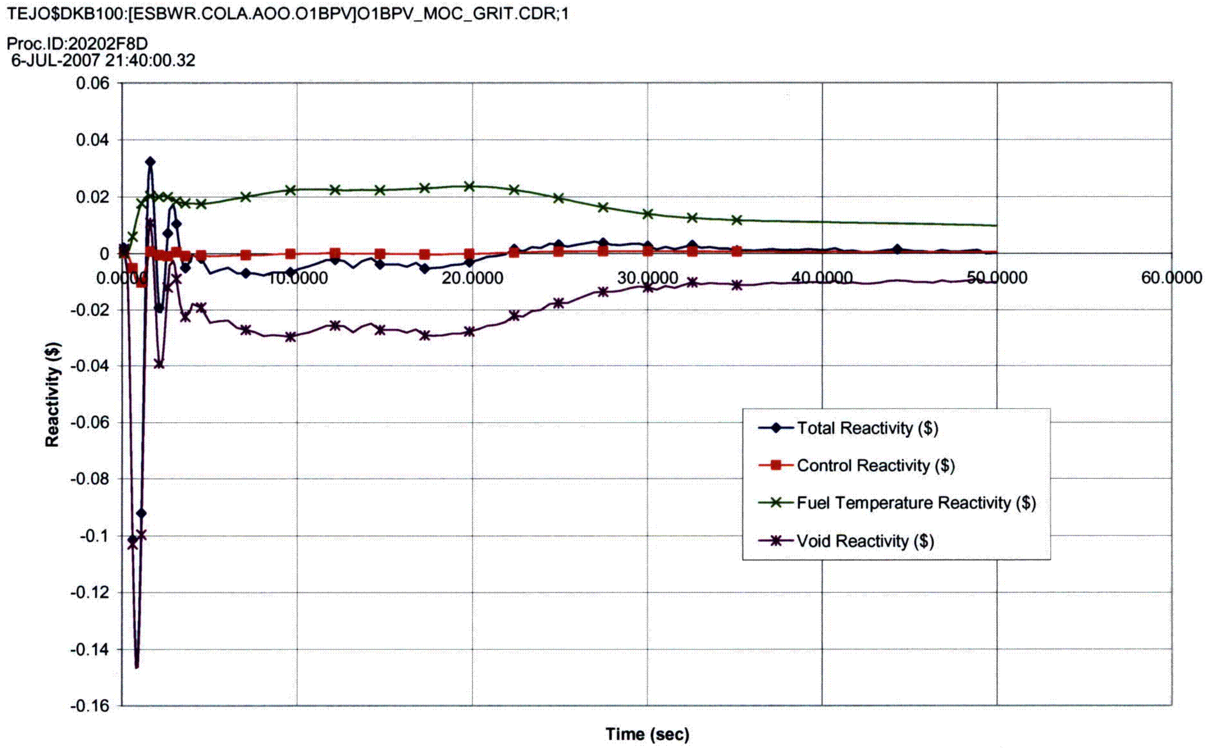


Figure 2.3-14e. Opening of One Turbine Control or Bypass Valve

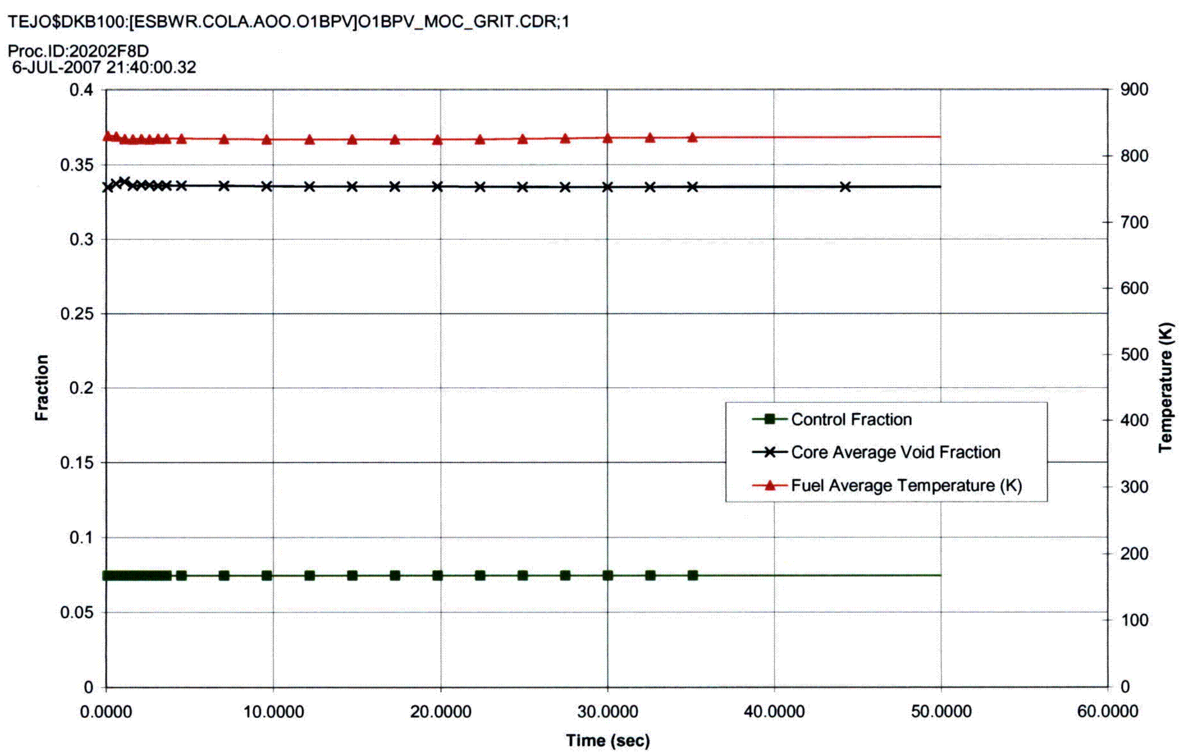


Figure 2.3-14f. Opening of One Turbine Control or Bypass Valve

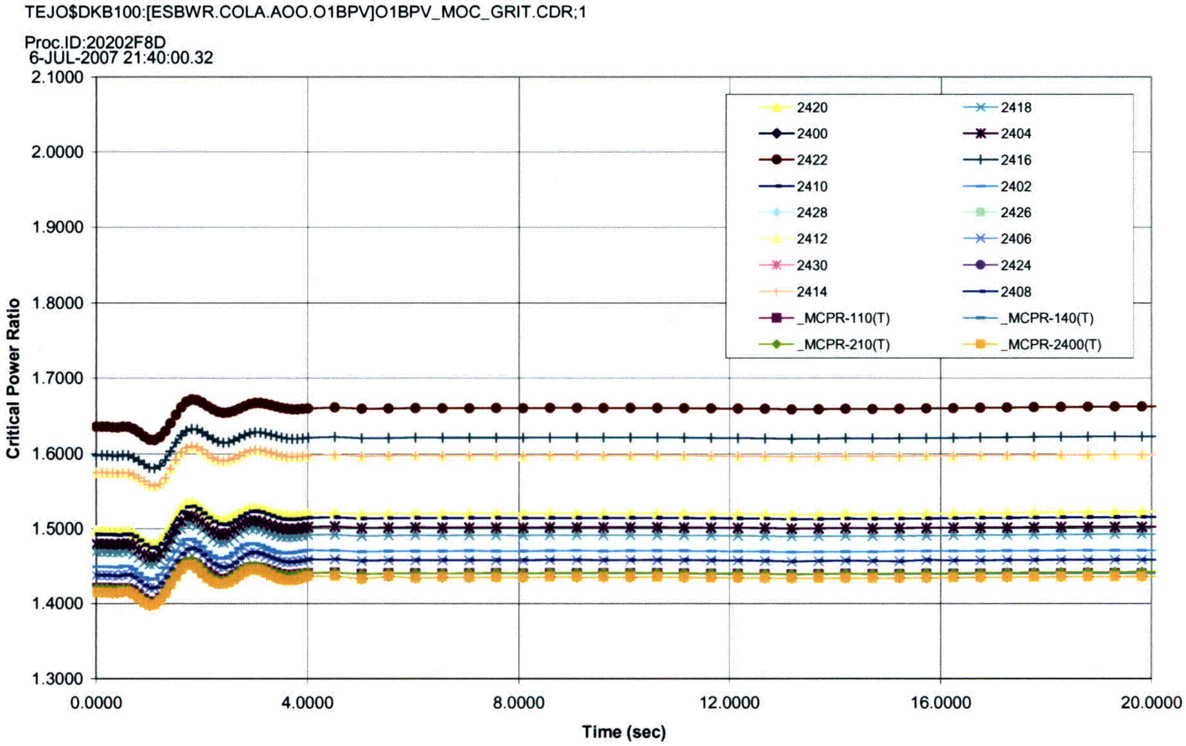


Figure 2.3-14g. Opening of One Turbine Control or Bypass Valve

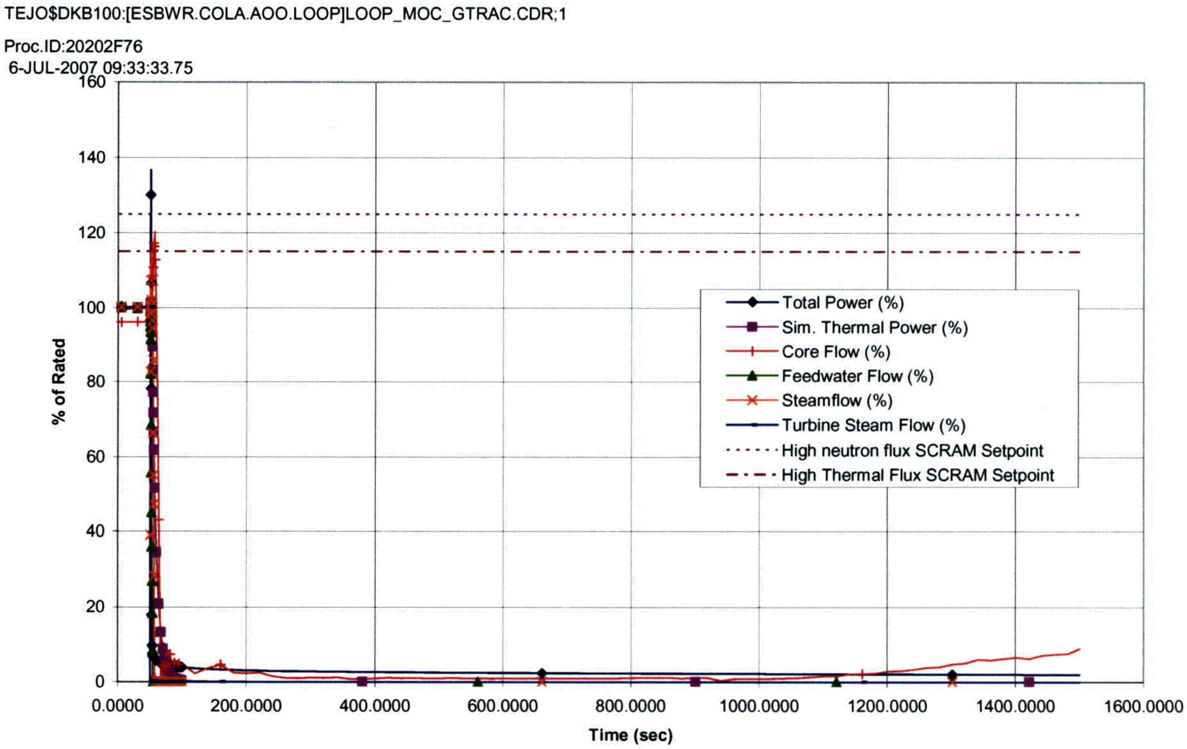


Figure 2.3-15a. Loss of Non-Emergency AC Power to Station Auxiliaries

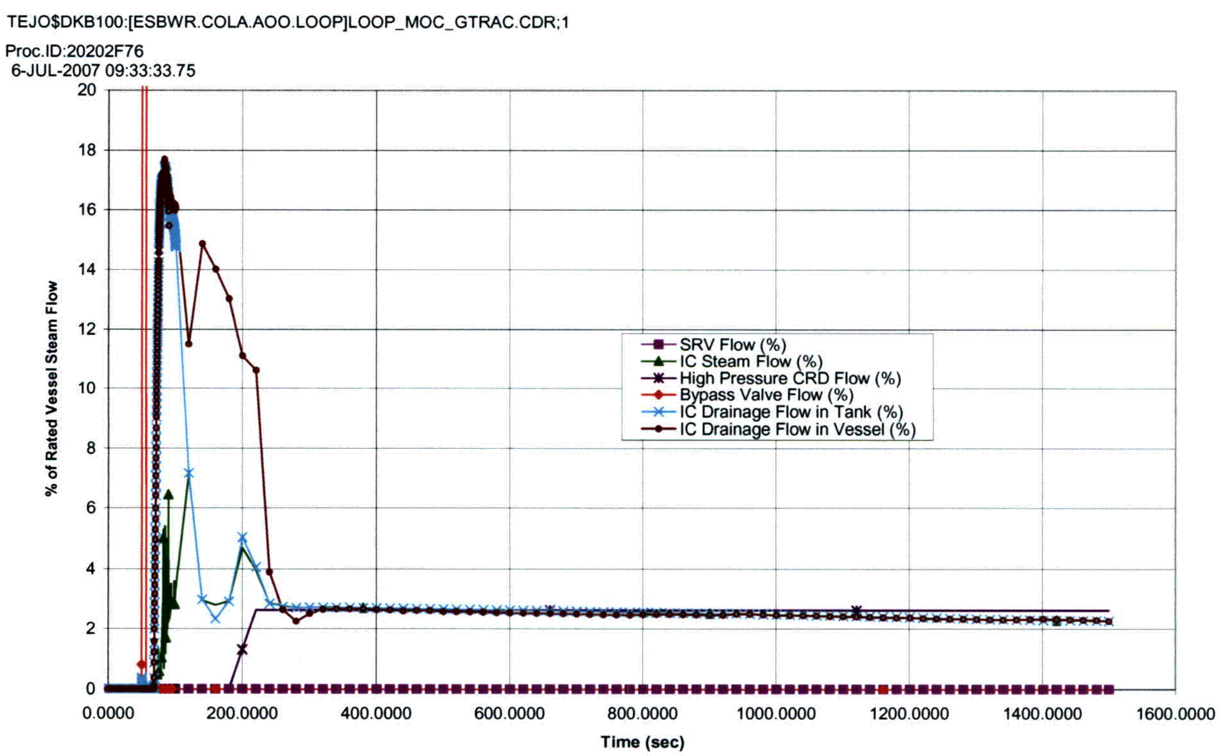


Figure 2.3-15b. Loss of Non-Emergency AC Power to Station Auxiliaries

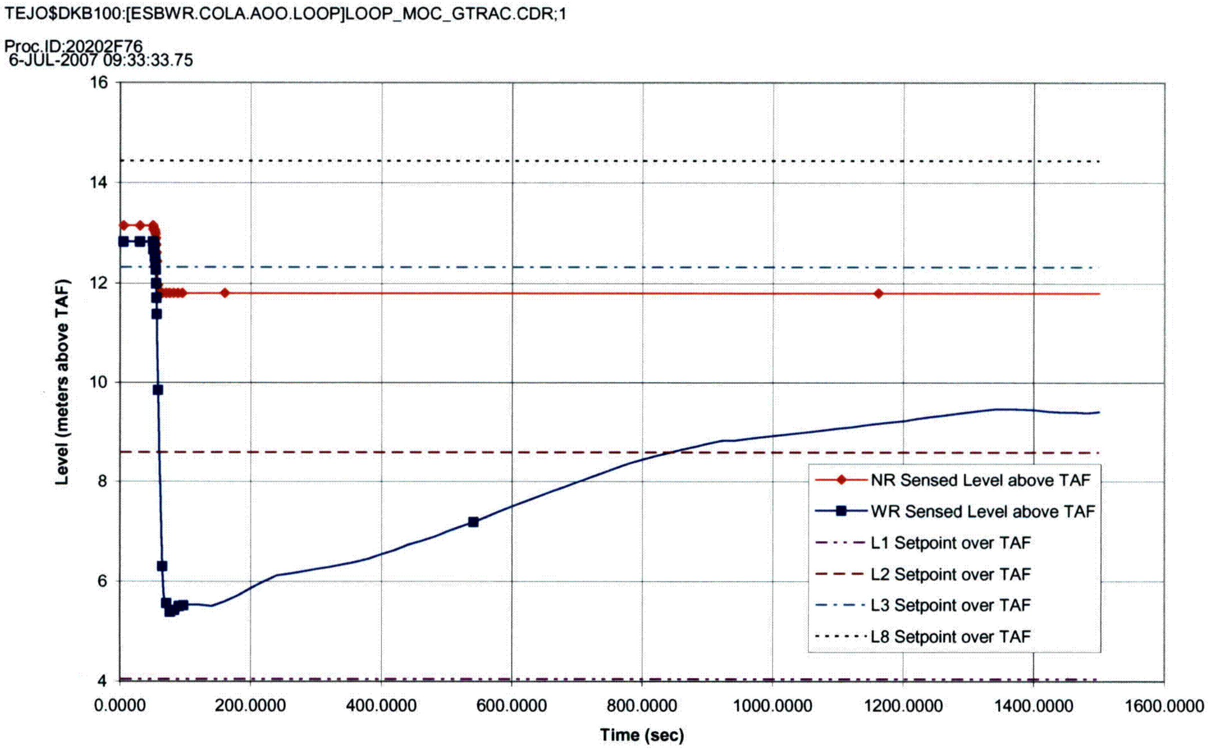


Figure 2.3-15c. Loss of Non-Emergency AC Power to Station Auxiliaries

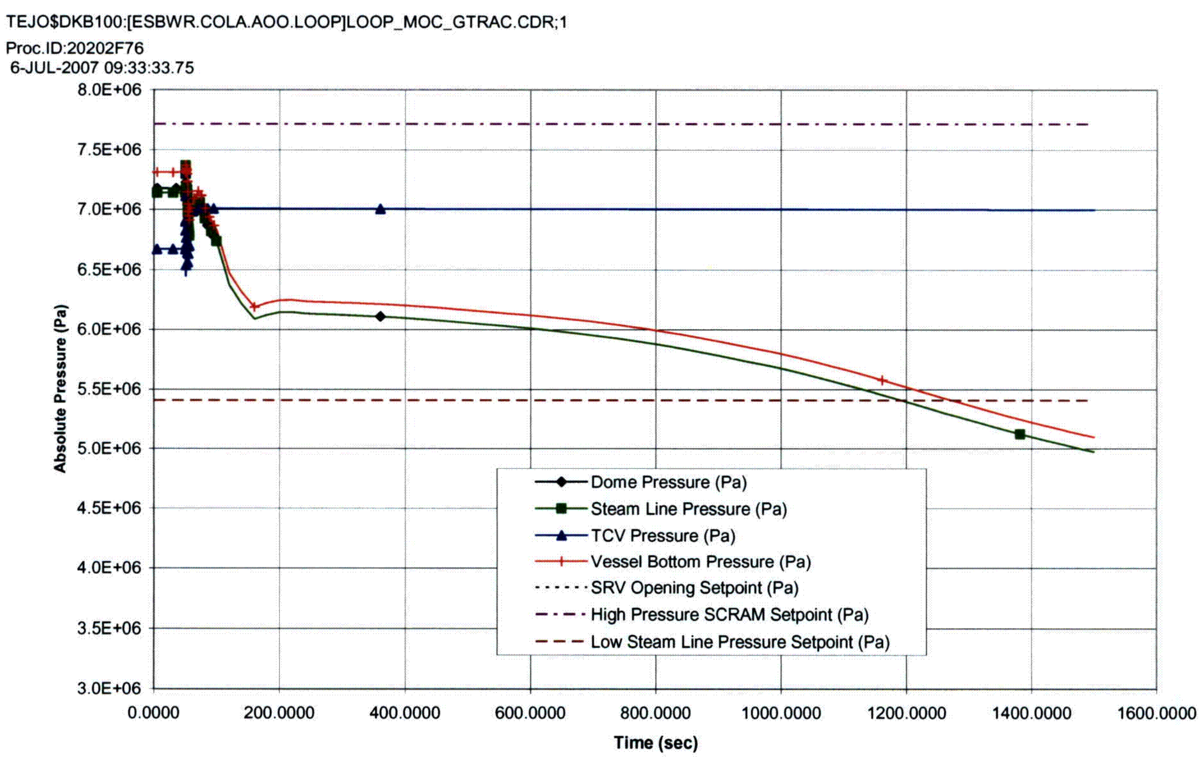


Figure 2.3-15d. Loss of Non-Emergency AC Power to Station Auxiliaries

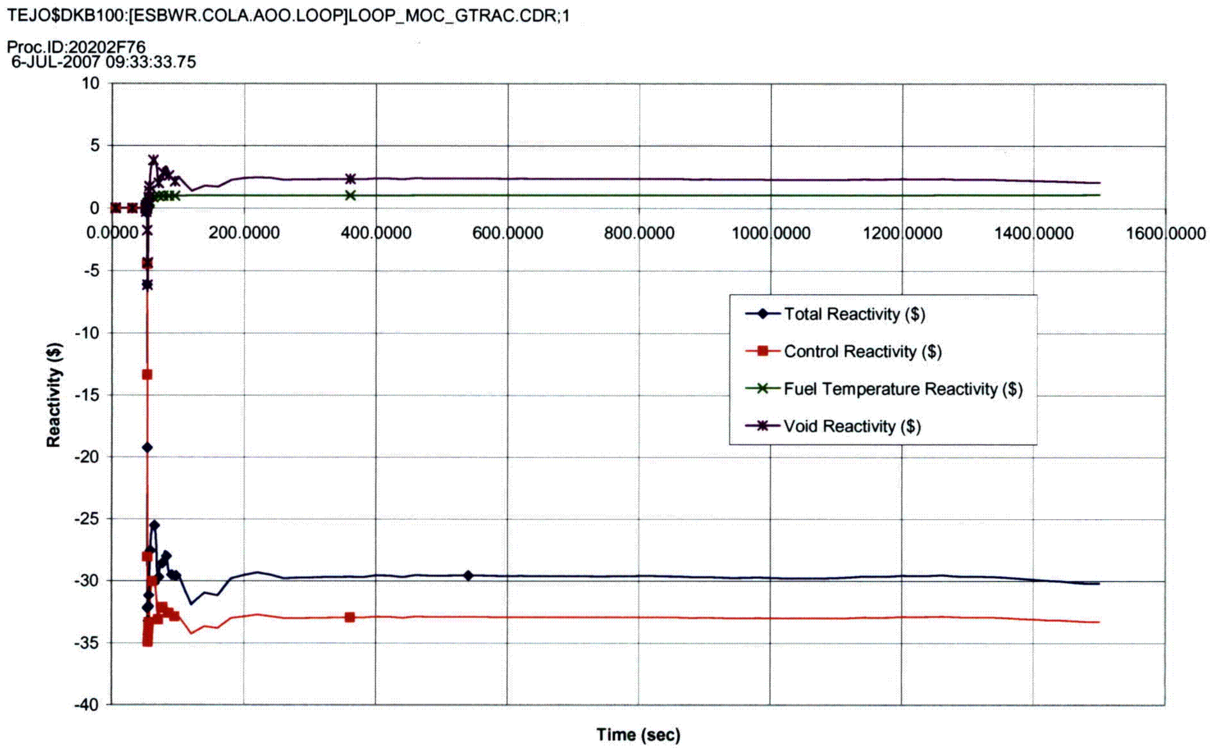


Figure 2.3-15e. Loss of Non-Emergency AC Power to Station Auxiliaries

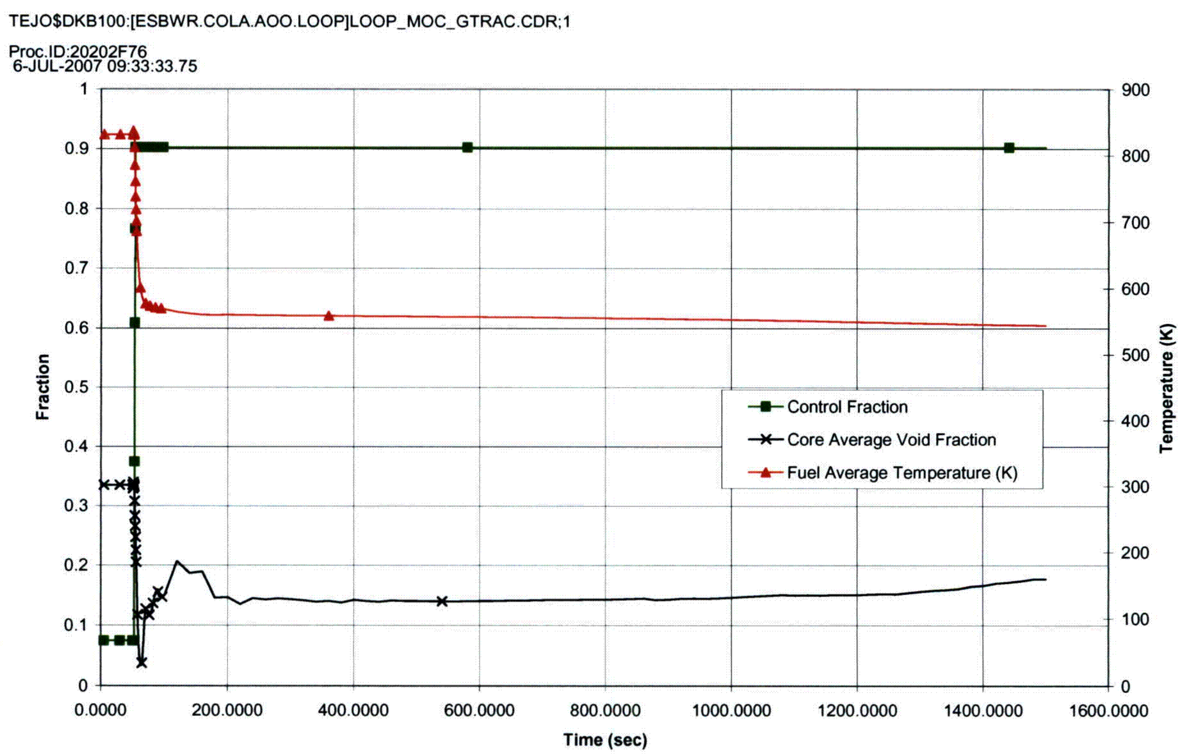


Figure 2.3-15f. Loss of Non-Emergency AC Power to Station Auxiliaries

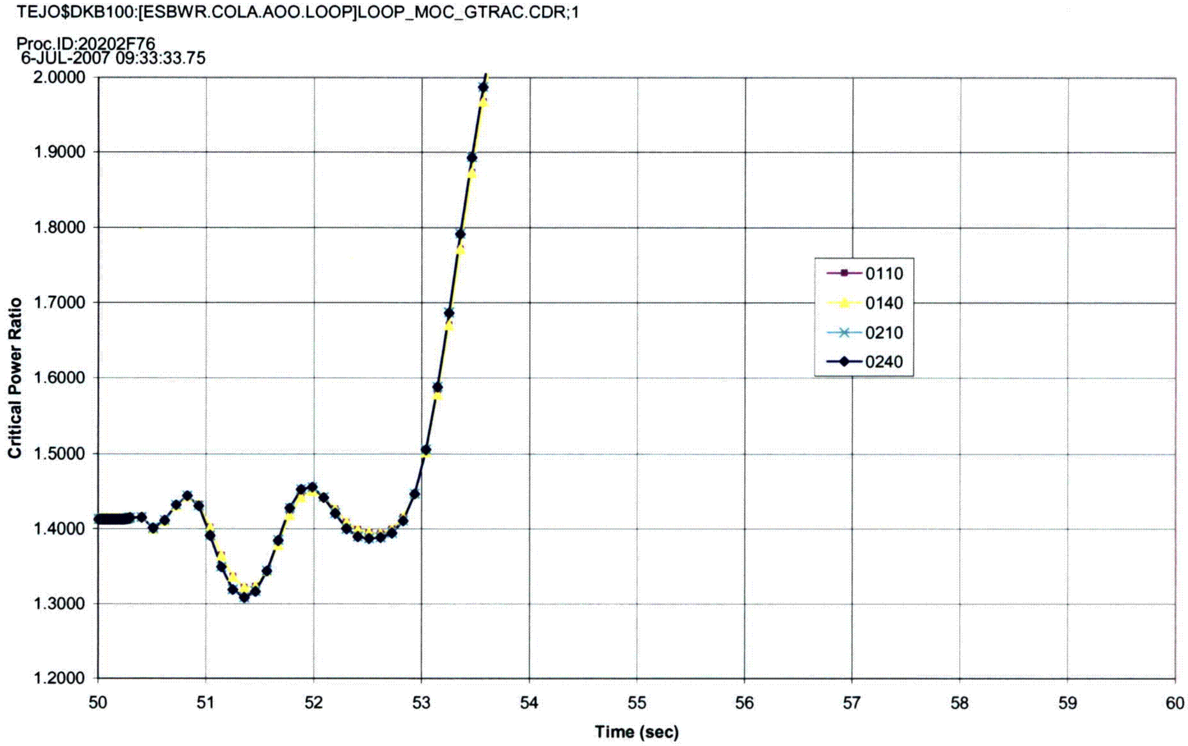


Figure 2.3-15g. Loss of Non-Emergency AC Power to Station Auxiliaries

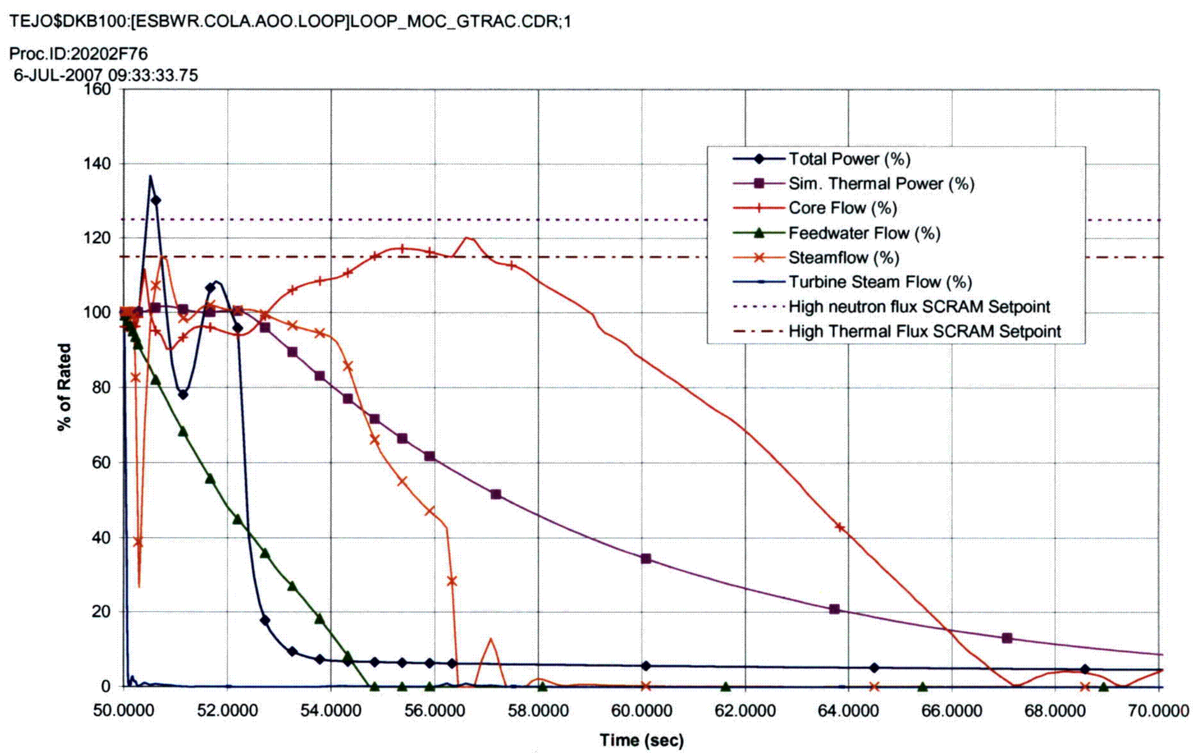


Figure 2.3-15h. Loss of Non-Emergency AC Power to Station Auxiliaries (fig. a from 50 to 70s)

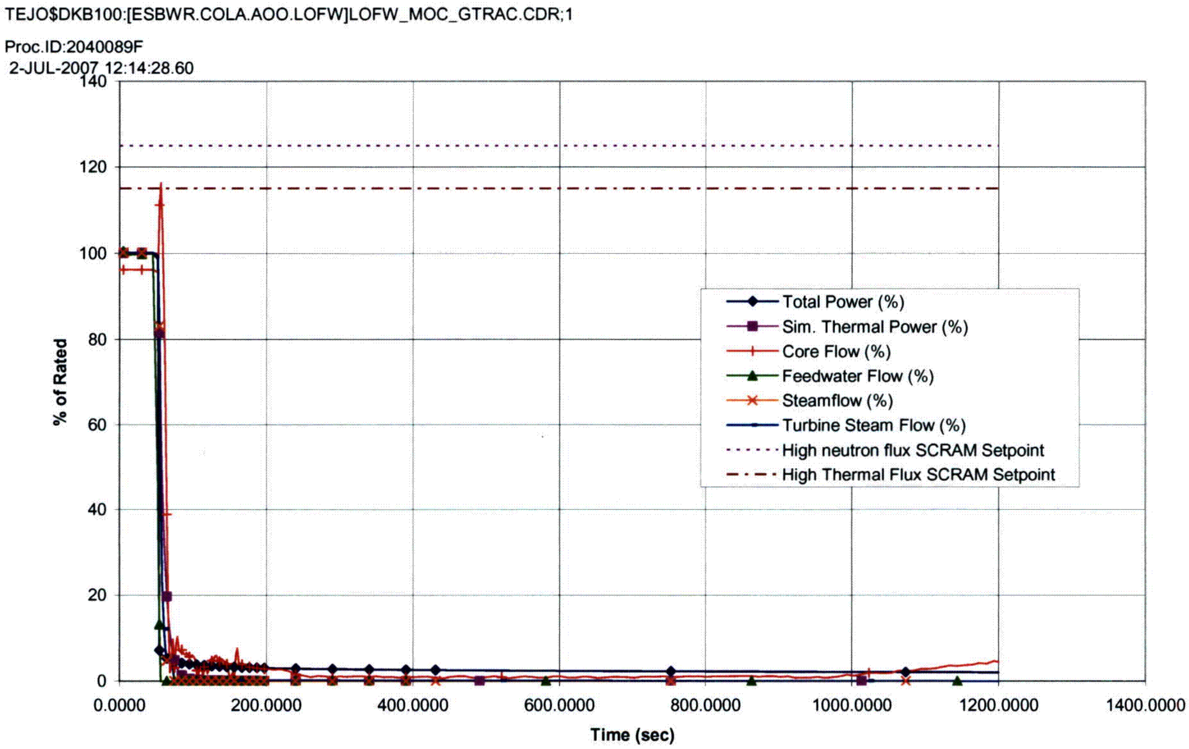


Figure 2.3-16a. Loss of All Feedwater Flow

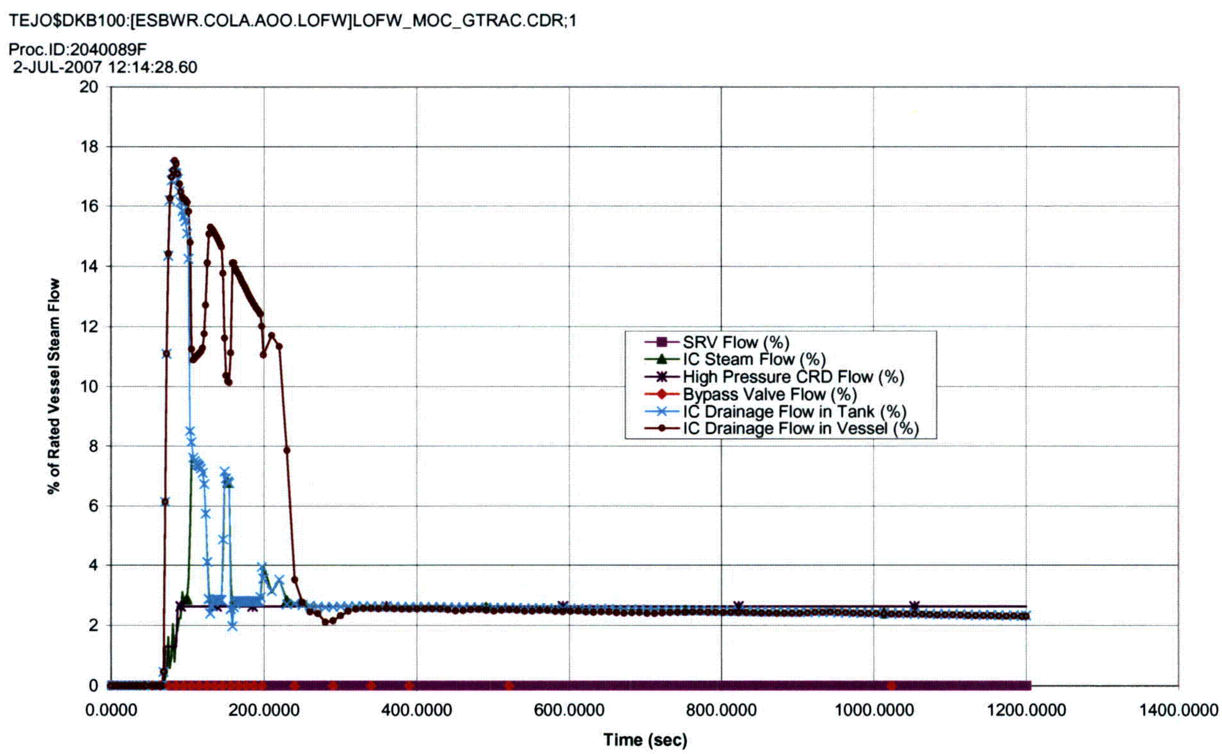


Figure 2.3-16b. Loss of All Feedwater Flow

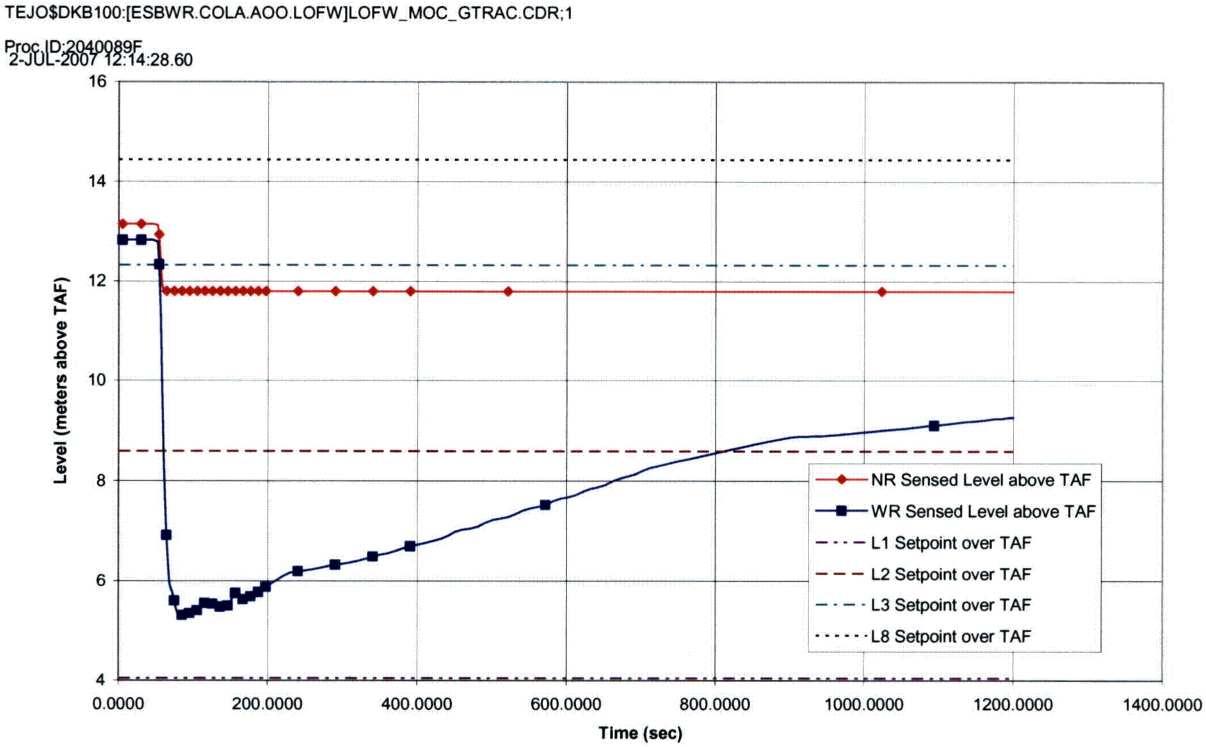


Figure 2.3-16c. Loss of All Feedwater Flow

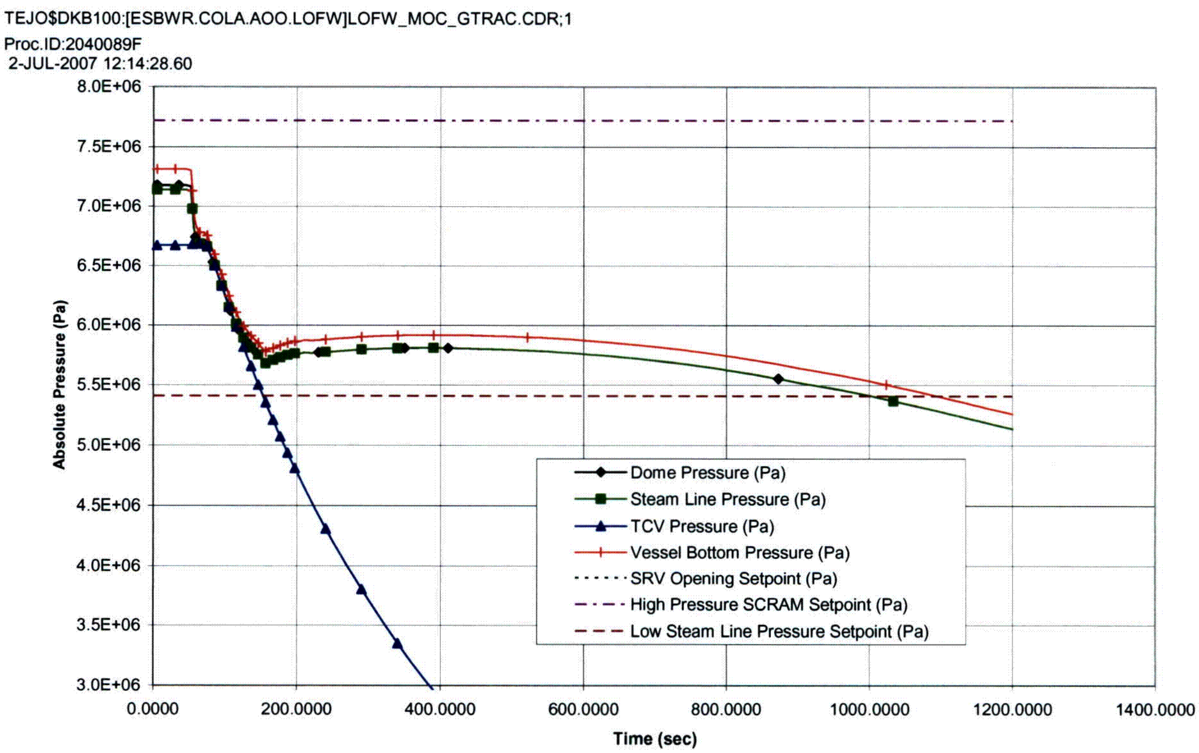


Figure 2.3-16d. Loss of All Feedwater Flow

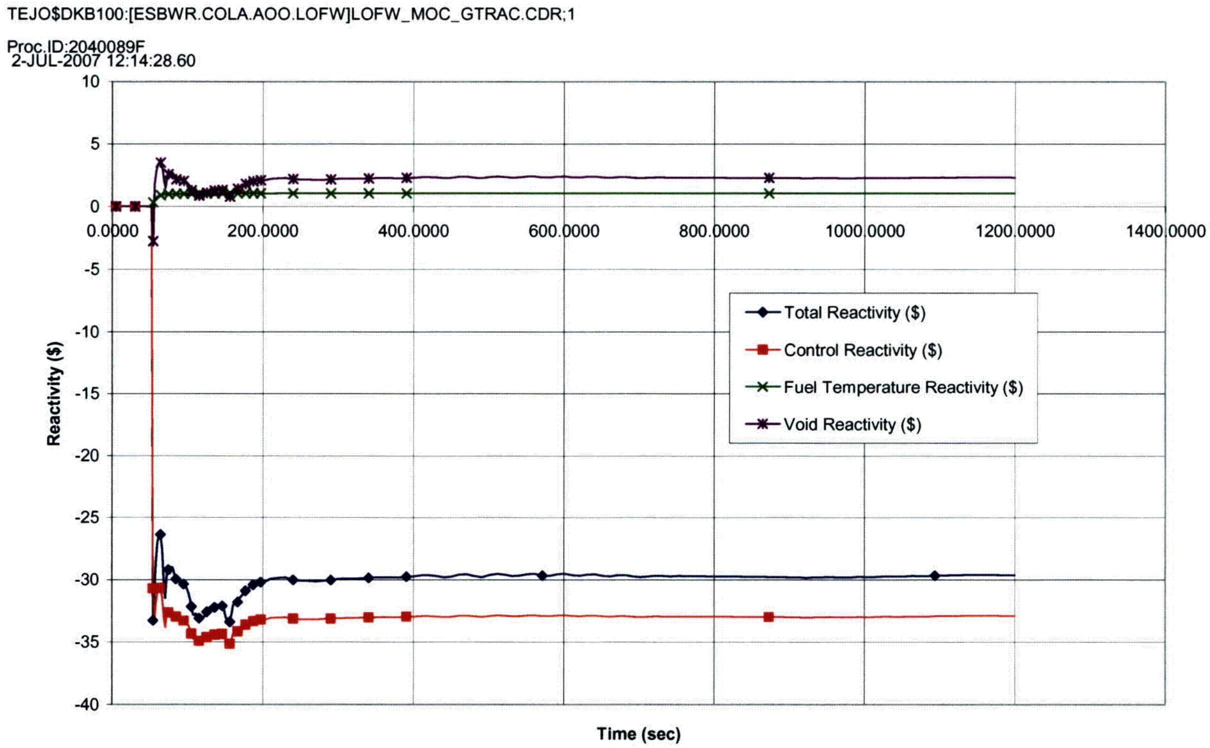


Figure 2.3-16e. Loss of All Feedwater Flow

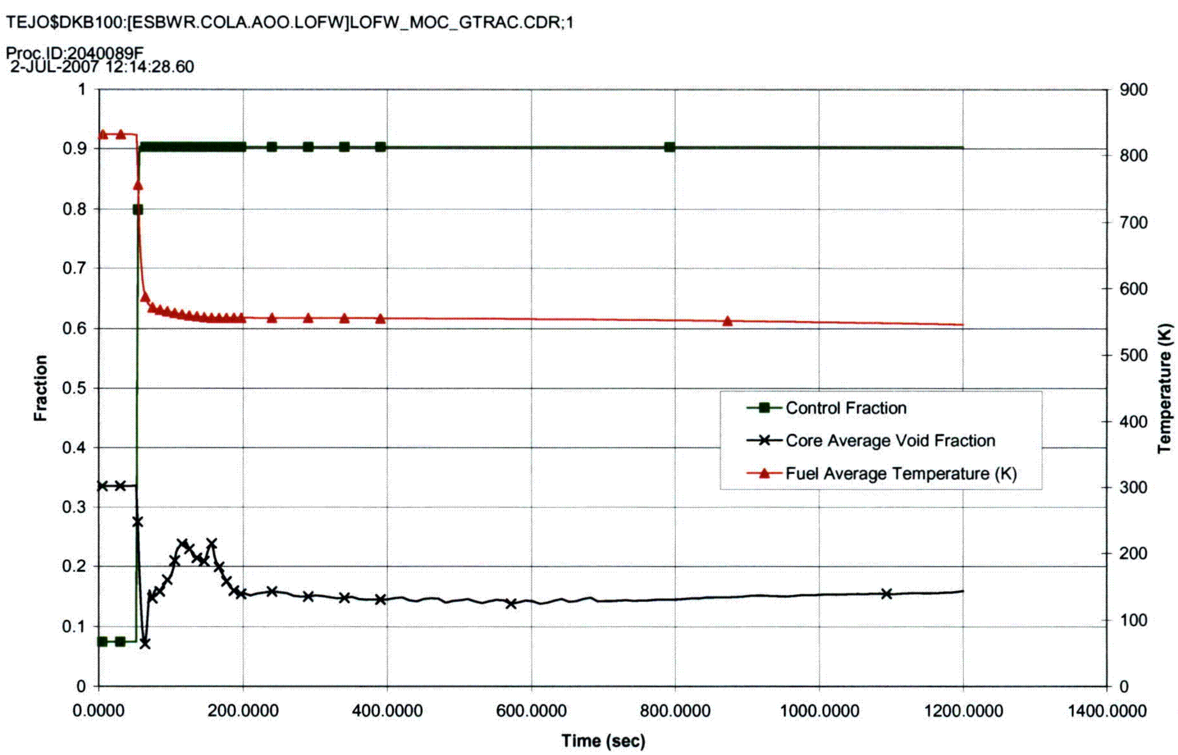


Figure 2.3-16f. Loss of All Feedwater Flow

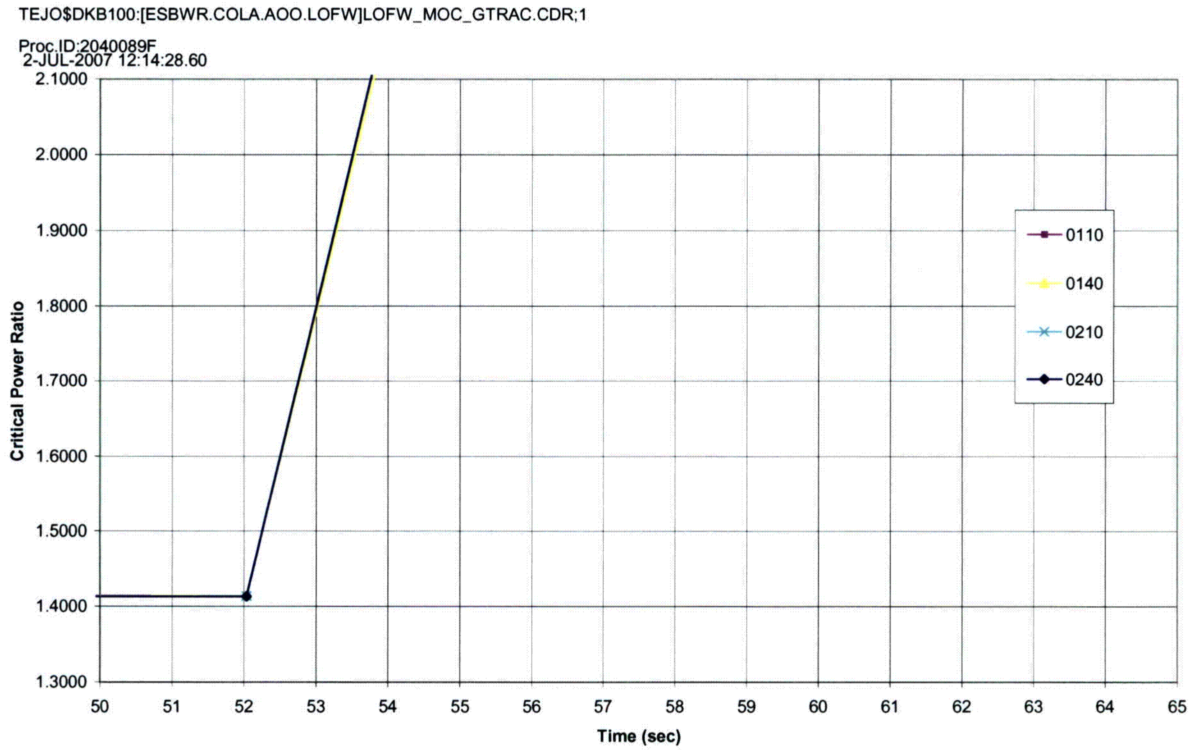


Figure 2.3-16g. Loss of All Feedwater Flow

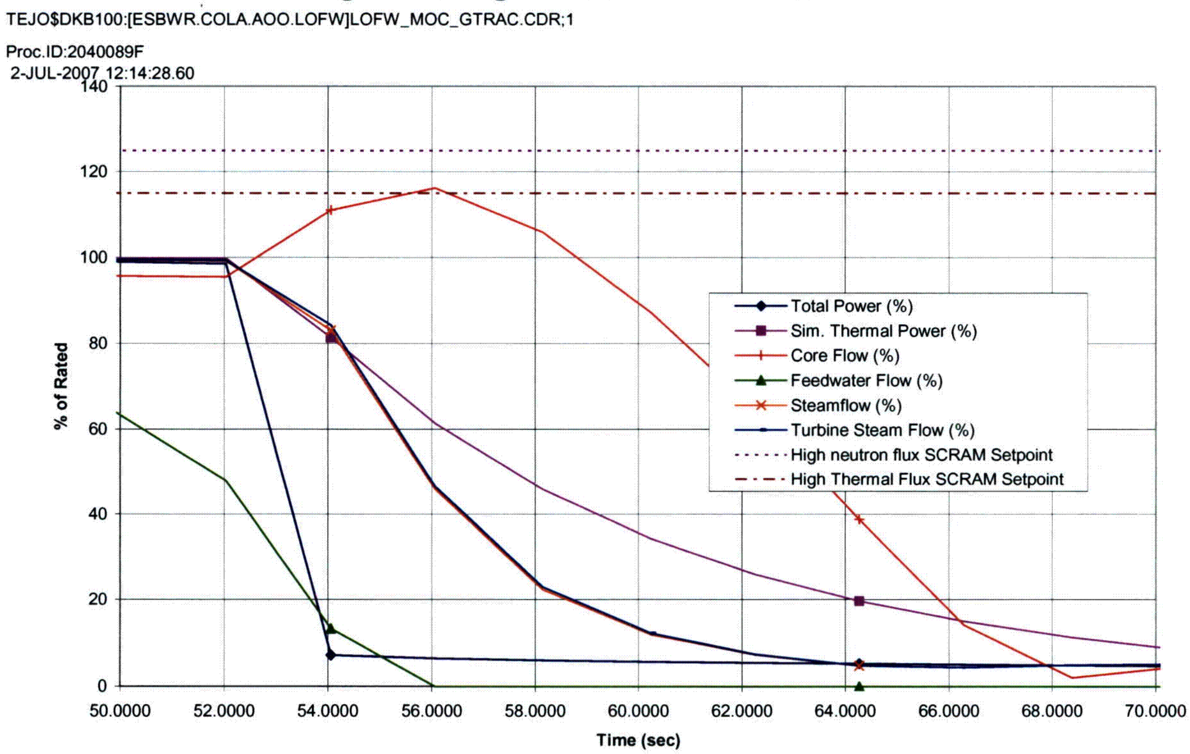


Figure 2.3-16h. Loss of All Feedwater Flow (fig a from 50 to 70s)

2.4 ANALYSIS OF INFREQUENT EVENTS

In Reference 2.4-3, GE Nuclear Energy, "ESBWR Design Control Document" 26A6642BP, Revision 3, February 2007, appendix 15A provides a determination of event frequency to categorize AOOs as defined in 10 CFR 50 Appendix A, and Infrequent Events. Section 15.0 of the DCD describes the licensing basis for this categorization.

The input parameters, initial conditions, and assumptions in Tables 2.3-1, 2 and 3 are applied in the TRACG calculations, based on the initial core in Reference 2.4-1 for the Infrequent Events addressed in Subsections 2.4.1 through 2.4.6 and Subsections 2.4.13 and 2.4.15. The summary of the Infrequent Events analyses is given in Table 2.4-1.

The results of the system response analyses for the initial core design documented in Reference 2.4-1 are provided here. System response analyses bounding operation in the feedwater temperature operating domain is documented in Reference 2.4-2.

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

2.4.1.1 Identification of Causes

Identification of causes is documented in the DCD (Reference 2.4-3).

2.4.1.2 Sequence of Events and Systems Operation

Sequence of Events

Table 2.4-2 lists the sequence of events for Figure 2.4-1.

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 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.

2.4.1.3 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 2.3-1.

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.

2.4.1.4 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 is assumed that all rods entering transition boiling fail.

2.4.1.5 Radiological Consequences

With an initial core OLMCPR of 1.28 or higher, the radiological analysis performed in the DCD (Reference 2.3-3, Section 15.3.1.5) is applicable in the initial core analysis.

2.4.2 Feedwater Controller Failure – Maximum Demand

2.4.2.1 Identification of Causes

See Subsection 2.3.5.2. This event assumes multiple control system failures, to simultaneously increase the flow in multiple FW pumps to their maximum limit.

2.4.2.2 Sequence of Events and Systems Operation

Sequence of Events

With excess FW flow, the water level rises to the high water level reference point (Level 8), at which time the FW pumps are run back, the main turbine is tripped and a scram is initiated.

Because Level 8 is located near the top of the separators, some moisture entrainment and carry-over to the turbine and bypass valve may occur. While this is potentially harmful to the turbine's integrity, it has no safety implications for the plant.

Identification of Operator Actions

The operator should:

- Follow the scram procedure.
- Observe that FW flow runback due to high water level has terminated the failure event.
- Switch the FW controller from auto to manual control to try to regain a correct output signal.
- Identify causes of the failure and report all key plant parameters during the event.

2.4.2.2.1 System Operation

To properly simulate the expected sequence of events, the analysis of this event assumes normal functioning of plant instrumentation and controls, plant protection and reactor protection systems. Important system operational actions for this event are tripping of the main turbine, FW flow runback, and scram due to high water level (Level 8).

2.4.2.3 Core and System Performance

2.4.2.3.1 Input Parameters and Initial Conditions

The total FW flow for all pumps runout is provided in Table 2.3-1.

2.4.2.3.2 Results

The simulated runout of all FW pumps is shown in Figure 2.4-2, the figure shows the changes in important variables during this event. The high water level turbine trip and FW pump runback are initiated early in the event as shown in Table 2.4-3. Scram occurs and limits the neutron flux peak and fuel thermal transient so that no fuel damage occurs. The Turbine Bypass System opens to limit peak pressure in the steamline near the SRVs and the peak pressure at the bottom of the vessel. The peak pressure in the bottom of the vessel remains below the ASME code upset limit. Peak steam line pressure near the SRVs remains below the setpoint of the SRVs.

The water level gradually drops, and can reach the Low Level reference point (Level 2), activating the IC system for long-term level control and the HP_CRD system to permit a slow recovery to the desired level

This event is reanalyzed for each specific initial core configuration.

2.4.2.4 **Barrier Performance**

As previously noted, the effect of this event does 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, there are no fuel rods that enter transition boiling.

2.4.2.5 **Radiological Consequences**

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

2.4.3 Pressure Regulator Failure – Opening of All Turbine Control and Bypass Valves

2.4.3.1 **Identification of Causes**

Identification of causes is documented in the DCD (Reference 2.4-3).

2.4.3.2 **Sequence of Events and Systems Operation**

2.4.3.2.1 ***Sequence of Events***

Table 2.4-4 lists the sequence of events for Figure 2.4-3.

2.4.3.2.2 ***Identification of Operator Actions***

If the reactor scrams as a result of the isolation caused by the low pressure at the turbine inlet in the run mode, the following sequence of operator actions is expected during the course of the event.

The operator should:

- Verify that all rods are inserted;
- Follow the scram procedure;
- Monitor reactor water level and pressure;
- Monitor turbine coastdown and break vacuum before the loss of steam seals (check turbine auxiliaries for proper operation);
- Observe that ICS is initiated on low-water level or MSIV closure;
- Use ICS to control reactor pressure and level;
- Cooldown the reactor per standard procedure if a restart is not intended; and

- Complete the scram report and initiate a maintenance survey of the SB&PC system before reactor restart.

2.4.3.2.3 Systems Operations

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

2.4.3.3 Core and System Performance

2.4.3.3.1 Input Parameters and Initial Condition

A five-second isolation valve closure (maximum isolation valve closing time plus instrument delay) instead of a three second closure is assumed when the turbine pressure decreases below the turbine inlet low-pressure setpoint for main steamline isolation initiation. This is within the specification limits of the valve and represents a conservative assumption.

2.4.3.3.2 Results

Figure 2.4-3 presents graphically how the low steam line pressure trips the isolation valve closure, stops vessel depressurization and produces a normal shutdown of the isolated reactor.

Depressurization results in formation of voids in the reactor coolant and causes a decrease in reactor power almost immediately. Position switches on the isolation valves initiate reactor scram.

The isolation limits the duration and severity of the depressurization so that no significant thermal stresses are imposed on the reactor coolant pressure boundary.

2.4.3.4 Barrier Performance

Barrier performance analyses were not required because the consequences of this event do not result in any temperature or pressure transient in excess of the criteria for which fuel, pressure vessel or containment are designed. The peak pressure in the bottom of the vessel remains below its ASME code faulted limit for the RCPB. Peak steam line pressure near the SRVs remains below the setpoint of the SRVs.

2.4.3.5 Radiological Consequences

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

2.4.4 Pressure Regulator Failure–Closure of All Turbine Control and Bypass Valves

2.4.4.1 Identification of Causes

Identification of causes is documented in the DCD (Reference 2.4-3).

2.4.4.2 Sequence of Events and Systems Operation

2.4.4.2.1 Sequence of Events

Table 2.4-5 lists the sequence of events for Figure 2.4-4.

2.4.4.2.2 Identification of Operator Actions

The operator should:

- Verify that all rods are inserted;
- Follow the scram procedure
- Monitor reactor water level and pressure;
- Monitor turbine coastdown and break vacuum before the loss of steam seals (check turbine auxiliaries for proper operation);
- Cool down the reactor per standard procedure if a restart is not intended; and
- Complete the scram report and initiate a maintenance survey of pressure regulator before reactor restart.

2.4.4.2.3 Systems Operation

Except for the failures in the SB&PC system, normal plant instrumentation and controls and plant protection and reactor protection systems are assumed to function normally. Specifically, this event takes credit for high neutron flux scram to shut down the reactor.

The turbine control valves, in their servo mode, have a full stroke closure time, from fully open to fully closed, from 2.5 seconds to 3.5 seconds. The worst case of 2.5 seconds is assumed in the analysis.

2.4.4.3 Core and System Performance

Neutron flux increases rapidly because of the void reduction caused by the pressure increase. When the sensed neutron flux reaches the high neutron flux scram setpoint, a reactor scram is initiated. The neutron flux and pressure increase is limited by the reactor scram.

2.4.4.4 Barrier Performance

The peak pressure in the bottom of the vessel remains below the ASME code limit for the RCPB. The peak vessel bottom pressure is below its ASME Code faulted pressure limit. The peak pressure at the SRVs is below the SRV setpoints. Therefore, there is no steam discharged to the suppression pool. In this event, there are no fuel rods that enter transition boiling.

2.4.4.5 Radiological Consequences

Because the ΔCPR/ICPR for this event is bounded by the limiting AOO value, no fuel failures are expected; therefore, no radiological analysis is required.

2.4.5 Generator Load Rejection With Total Turbine Bypass Failure

2.4.5.1 Identification of Causes

Identification of causes is documented in the DCD (Reference 2.4-3).

2.4.5.2 Sequence of Events and System Operation

2.4.5.2.1 Sequence of Events

A loss of generator electrical load at high power with failure of all bypass valves produces the sequence of events listed in Table 2.4-6.

2.4.5.2.2 Identification of Operator Actions

The operator should:

- Verify that all rods are inserted;
- Follow the scram procedure;
- Verify proper bypass valve performance;
- Observe that the feedwater/level controls have maintained the reactor water level at a satisfactory value;
- Observe that the pressure regulator is controlling reactor pressure at the desired value; and
- Observe reactor peak power and pressure.

2.4.5.2.3 Systems Operation

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

All plant control systems maintain normal operation unless specifically designated to the contrary, except that failure of all turbine bypass valves is assumed for the entire event.

2.4.5.3 Core and System Performance

2.4.5.3.1 Input Parameters and Initial Conditions

The turbine electrohydraulic control system (EHC) detects load rejection before a measurable turbine speed change takes place.

The closure characteristics of the TCVs are assumed conservatively such that the valves operate in the full arc (FA) mode. The TCVs have a full stroke closure time, from fully open to fully closed, from 0.15 seconds to 0.20 seconds. The worst case value (see Table 2.4-6) is assumed in the analysis.

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

2.4.5.3.2 Results

The results are shown in Figure 2.4-5.

2.4.5.4 Barrier Performance

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

2.4.5.5 **Radiological Consequences**

Because the ΔCPR/ICPR for this event is bounded by the limiting AOO value, no fuel failures are expected; therefore, no radiological analysis is required.

2.4.6 Turbine Trip With Total Turbine Bypass Failure

2.4.6.1 **Identification of Causes**

Identification of causes is documented in the DCD (Reference 2.4-3).

2.4.6.2 **Sequence of Events and System Operation**

2.4.6.2.1 ***Sequence of Events***

Turbine trip at high power with failure of all bypass valves produces the sequence of events listed in Table 2.4-7.

2.4.6.2.2 ***Identification of Operator Actions***

The operator should:

- Verify that all rods are inserted;
- Follow the scram procedure;
- Verify that the generator breaker is automatically opened to allow electrical buses originally supplied by the generator to be supplied by the incoming power;
- Monitor reactor water level and pressure;
- Check turbine for proper operation of all auxiliaries during coastdown;
- Manually initiate ICS, if necessary, to control reactor pressure;
- Depending on conditions, maintain pressure for restart purposes, or initiate normal operating procedures for cooldown;
- Put the mode switch in the startup position before the reactor pressure decays to below 6 MPa;
- Cool down the reactor per standard procedure if a restart is not intended; and
- Investigate the cause of the trip, make repairs as necessary, and complete the scram report.

2.4.6.2.3 ***Systems Operation***

All plant control systems maintain normal operation unless specifically designated to the contrary, except that failure of all main turbine bypass valves is assumed for the entire transient time period analyzed. Credit is taken for successful operation of the RPS.

2.4.6.3 **Core and System Performance**

2.4.6.3.1 ***Input Parameters and Initial Conditions***

Turbine stop valves full stroke closure time is in the range of 0.10 second to 0.15 second. The worst case (see Table 2.4-7) is assumed in the analysis. A reactor scram is initiated by the turbine stop valve position switch, and after the confirmation of no availability of the turbine bypass.

2.4.6.3.2 Results

A turbine trip with failure of the bypass system is simulated at 100% NBR power conditions in Figure 2.4-6.

The severity of this transient is similar to the generator load rejection with failure of bypass event presented in Subsection 2.4.5.

2.4.6.4 Barrier Performance

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

2.4.6.5 Radiological Consequences

Because the ΔCPR/ICPR for this event is bounded by the limiting AOO value, no fuel failures are expected; therefore, no radiological analysis is required.

2.4.7 Control Rod Withdrawal Error During Refueling

Analysis in the DCD applies to the initial core.

2.4.8 Control Rod Withdrawal Error During Startup

Analysis in the DCD applies to the initial core.

2.4.9 Control Rod Withdrawal Error During Power Operation

Analysis in the DCD applies to the initial core.

2.4.10 Fuel Assembly Loading Error, Mislocated Bundle

Analysis in the DCD applies to the initial core.

2.4.11 Fuel Assembly Loading Error, Misoriented Bundle

Analysis in the DCD applies to the initial core.

2.4.12 Inadvertent SDC Function Operation

Analysis in the DCD applies to the initial core.

2.4.13 Inadvertent Opening of a Safety-Relief Valve

2.4.13.1 Identification of Causes

Identification of causes is documented in the DCD (Reference 2.4-3).

2.4.13.2 Sequence of Events and Systems Operation

Sequence of Events

Table 2.4-11 lists the sequence of events for this event. In Figure 2.4-8 the response of the main parameters of this event are presented.

Identification of Operator Actions

The plant operator must re-close the valve as soon as possible and check that reactor and TG output return to normal. If the valve cannot be closed, plant shutdown should be initiated.

Systems Operation

This event assumes normal functioning of normal plant instrumentation and controls, specifically the operation of the pressure regulator and level control systems.

2.4.13.3 **Core and System Performance**

The opening of one SRV allows steam to be discharged into the suppression pool. The sudden increase in the rate of steam flow leaving the reactor vessel causes a mild depressurization transient. In this event, the open SRV capacity is assumed to be 1.1 times the capacity of the SV in DCD Table 5.2-2 (Reference 2.4-3). This increased capacity, 154.2 kg/s, (340 lbm/s) is assumed to observe bounding depressurization results.

The SB&PC system senses the nuclear system pressure decrease and within a few seconds closes the TCVs far enough to stabilize the reactor vessel pressure at a slightly lower value and the reactor settles at nearly the initial power level. Eventually, the plant automatically scrams on high suppression pool temperature.

Thermal margins decrease only slightly through the transient and no fuel damage results from the event.

2.4.13.4 **Barrier Performance**

The transient resulting from the inadvertent SRV opening is a mild depressurization, which is within the range of normal load following and has no significant effect on RCPB and containment design pressure limits.

2.4.13.5 **Radiological Consequences**

While the effect of this event does not result in fuel failure, it does result in the discharge of normal coolant activity to the suppression pool. Because this activity is contained in the primary containment, there is no exposure to operating personnel. Because this event does not result in an uncontrolled release to the environment, the plant operator can choose to leave the activity confined within containment, use the fuel and auxiliary pools cooling system (FAPCS) to remove radioactivity from the pool, or discharge it to the environment under controlled release conditions. If purging containment is chosen, the release is performed in accordance with Technical Specifications; therefore, this event, results in a small increase in the yearly integrated exposure level.

2.4.14 Inadvertent Opening of a Depressurization Valve

Analysis in the DCD applies to the initial core.

2.4.15 Stuck Open Safety-Relief Valve

2.4.15.1 **Identification of Causes**

Cause of a stuck open safety relief valve is attributed to the malfunction of the valve after it has opened either inadvertently or in response to a high pressure signal. It is simply postulated that a failure occurs and the event is analyzed accordingly. Detailed discussion of the valve design is provided in Chapter 5 of Reference 2.4-3.

In this analysis, after any event that produces a reactor scram, it is assumed that a SRV remains open without any possibility of closure. The operations of the ICs produce a depressurization, with the HP_CRD operating to recover the level after the scram. The event is analyzed with 4 ICs available and with bounding capacity, to observe the maximum possible depressurization rate. Finally, the reactor reaches near atmospheric pressure.

2.4.15.2 Sequence of Events and Systems Operation

Sequence of Events

Table 2.4-12 lists the sequence of events for this event. In Figure 2.4-9 the response of the main parameters of this event are presented. If auxiliary power is not available, the sequence of events is similar to the Main Steam Line Break sequence given in Table 6.3-8 of Reference 2.4-3.

Identification of Operator Actions

The plant operator must re-close the valve as soon as possible and check that the reactor and TG output return to normal. If the valve cannot be closed and the reactor has scrammed because of some other reason (if the SRVs are in the open condition, the reactor has scrammed previously, except for the Inadvertent SRV opening analyzed previously), manual activation of the IC and other systems can be initialized to reduce the amount of steam reaching the suppression pool.

Systems Operation

This event assumes normal functioning of the plant instrumentation and controls, specifically the operation of the pressure regulator and water level control systems. In this event the open SRV capacity is assumed to be 1.1 times the capacity of the SV in DCD Table 5.2-2 (Reference 2.4-3). This increased capacity, 154.2 kg/s, (340 lbm/s) is assumed to observe bounding depressurization results.

2.4.15.3 Core and System Performance

The opening of one SRV allows steam to be discharged to the suppression pool. The sudden increase in the rate of steam flow leaving the reactor vessel causes a depressurization transient, with the vessel pressure slowly decreasing until reaching atmospheric pressure. The SRV steam discharge also results in a slight heating of the suppression pool.

Thermal margins decrease only slightly through the transient and no fuel damage is predicted for this event.

2.4.15.4 Barrier Performance

As presented previously, the transient resulting from a stuck open relief valve is the total depressurization of the pressure vessel, which is within the range of normal plant operation and therefore has no significant effect on RCPB and containment design pressure limits.

2.4.15.5 Radiological Consequences

While the effect of this event does not result in fuel failure, it does result in the discharge of normal coolant activity to the suppression pool. Because this activity is contained in the primary containment, there is no exposure to operating personnel. Because this event does not result in an uncontrolled release to the environment, the plant operator can choose to leave the activity confined inside containment, use FAPCS to remove radioactivity from the pool, or discharge it to the environment in a controlled manner. If purging of the containment is chosen, the release is performed in accordance with Technical Specifications. Consequently, this event results in a small increase in the yearly integrated exposure level.

2.4.16 Liquid Containing Tank Failure

Analysis in the DCD applies to the initial core.

2.4.17 References

2.4-1 Global Nuclear Fuel, "ESBWR Initial Core Nuclear Design Report", NEDC-33326-P, Class III (Proprietary), Revision 0, July 2007, NEDO-33326, Class I (Non-proprietary), Revision 0, July 2007.

2.4-2 GE-Hitachi Nuclear Energy, "ESBWR Feedwater Temperature Operating Domain Transient and Accident Analysis", NEDO-33338 Class I, Revision 0, Scheduled October 2007.

2.4-3 GE Nuclear Energy, "ESBWR Design Control Document" 26A6642, Revision 4, September 2007.

Table 2.4-1

Results Summary of Infrequent Events

Sub-section I.D.	**Description**	**Max. Neutron Flux,** % NBR	**Max. Dome Pressure,** MPaG (psig)	**Max. Vessel Bottom Pressure,** MPaG (psig)	**Max. Steamline Pressure,** MPaG (psig)	**Max. Core Average Surface Heat Flux,** % of Initial	**ΔCPR/ ICPR**
2.4.1	Loss of Feedwater Heating with SCRRI failure	122	7.13 (1034)	7.27 (1054)	7.09 (1028)	122	0.12
2.4.2	FWCF – Maximum Demand	115	7.24 (1050)	7.38 (1070)	7.26 (1053)	107	0.04
2.4.3	Pressure Regulator Failure – Opening of all TCVs and BPVs	100	7.08 (1027)	7.21 (1046)	6.99 (1014)	100	<0.01
2.4.4	Pressure Regulator Failure – Closing of all TCVs and BPVs	136	8.02 (1163)	8.15 (1182)	8.02 (1163)	104	0.03
2.4.5	Load Rejection with total bypass failure	257	8.09 (1173)	8.22 (1192)	8.10 (1175)	106	0.08
2.4.6	Turbine Trip with total bypass failure	231	8.08 (1172)	8.21 (1191)	8.08 (1172)	106	0.07
2.4.13	Inadvertent SRV open	101	7.08 (1027)	7.21 (1046)	6.99 (1014)	101	<0.01
2.4.15	Stuck open SRV (1)	--	7.82 (1134)	7.96 (1155)	7.83 (1156)	--	<0.01

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

Table 2.4-2

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°F) temperature reduction in the FW system
25 (est.)	Initial effect of unheated FW starts to raise core power level
72	High thermal simulated Scram is reached but it is not credited
300 (est.)	New Steady State Reached

* See Figure 2.4-1

Table 2.4-3

Sequence of Events for Feedwater Controller Failure – Maximum Demand

Time (sec)	Event *
0	Initiate simulated runout of all FW pumps (170% at rated vessel pressure)
12.8	Main turbine bypass valves opened to control vessel pressure
14.1	L8 vessel level setpoint is reached
14.98	Scram, trip of main turbine and FW pump runback is activated
15.12	Turbine Bypass fast opening activation limits the pressurization of the vessel
15.2	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 2.4-2

Table 2.4-4

Sequence of Events for Pressure Regulator Failure – Opening of All Turbine Control and Bypass Valves

Time (sec)	Event*
0	Simulate all turbine control valves and bypass valves to open
20.1	Low turbine inlet pressure trip initiates main steamline isolation
21.4	MSIV position switch at 85% initiates scram and activates the IC
24.9	Main steam isolation valves closed. Bypass valves remain open, exhausting steam in steamlines downstream of isolation valves
32.1	L2 setpoint is reached
37.3	The IC begins to remove heat from the vessel
42.2	HP_CRD is activated, this recovers the level

* See Figure 2.4-3

Table 2.4-5

Sequence of Events for Pressure Regulator Failure – Closure of All Turbine Control and Bypass Valves

Time (sec)	Event*
0	Simulate zero steam flow demand to main turbine and bypass valves
0	Turbine control valves start to close
1.92	Neutron flux reaches high flux scram setpoint and initiates a reactor scram
2.17	The rods begin to enter inside the core
2.5	TCV is closed
3.2	L3 setpoint is reached
19.3	L2 setpoint is reached
29.5 Long term	HP_CRD is activated on L2 to recover the level

* See Figure 2.4-4

Table 2.4-6

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
0.09	Turbine control valves closed
0.20	After detection of not enough bypass availability the RPS initiates a reactor scram
0.45	The rods begin to enter inside the core
1.79	L3 setpoint is reached
16.4	L2 setpoint is reached
26.6 Long term	HP_CRD is activated on L2 to recover the level

* See Figure 2.4-5

Table 2.4-7

Sequence of Events for Turbine Trip With Total Turbine Bypass Failure

Time (sec)	Event*
0.0	Turbine trip initiates closure of main stop valves
0.0	Turbine bypass valves fail to operate
0.10	Turbine stop valves close
0.20	After detection of not enough bypass availability the RPS initiates a reactor scram
0.45	The rods begin to enter inside the core
1.83	L3 setpoint is reaches
Long term	HP_CRD is activated on L2 to recover the level

* See Figure 2.4-6. This figures show a short term response, for longer term response of a similar transient see Figure 2.4-5

Table 2.4-8

Sequence of Events for Continuous Control Rod Withdrawal Error During Reactor Startup

Data in the DCD applies to the initial core.

Table 2.4-9

Sequence of Events for the Mislocated Bundle

Data in the DCD applies to the initial core.

Table 2.4-10

Sequence of Events for the Misoriented Bundle

Data in the DCD applies to the initial core.

Table 2.4-11

Sequence of Events for Inadvertent SRV Opening

Time (sec)	Event*
0	Spurious opening of one SRV
5.0	Relief valve flow reaches full flow
30.0	System establishes new steady-state operation
326	Suppression pool temperature reaches the setpoint; suppression pool cooling function is initiated. (Not Credited)
326	Suppression pool temperature reaches setpoint; reactor scram is automatically initiated. (scram is conservatively assumed at pool cooling initiation temperature)

*See Figure 2.4-8

Table 2.4-12

Sequence of Events for Stuck Open Safety Relief Valve

Time (sec)	Event*
0	Event happens, the reactor is scrammed and one SRV stuck open
14.0	Relief valve flow reaches one SRV flow
12.0	The vessel begins depressurization
25.4	HP_CRD is activated on L2
97.1	Low steamline pressure setpoint is reached
97.9	MSIV at 85%
100.	MSIV is closed
129.0	ICs discharge valves fully open[(1)]
300.0	HP_CRD is deactivated because of L8
Long term	Suppression pool temperature reaches the setpoint; suppression pool cooling function is initiated. (Not Credited) Atmospheric pressure is reached

*See Figure 2.4-9

(1) Four ICS at a bounding capacity are actuating during the depressurization.

Table 2.4-13

1000 Fuel Rod Failure Parameters

Data in the DCD applies to the initial core.

Table 2.4-14

1000 Fuel Rod Failure Offgas Release Rate

Data in the DCD applies to the initial core.

Table 2.4-15

1000 Fuel Rod Failure Core Fission Product Inventory

Data in the DCD applies to the initial core.

Table 2.4-16

1000 Fuel Rod Failure Dose Results

Data in the DCD applies to the initial core.

Table 2.4-17

Radwaste System Failure Accident Parameters

Data in the DCD applies to the initial core.

Table 2.4-18

Radwaste System Failure Accident Isotopic Release to Environment (megabecquerel)

Data in the DCD applies to the initial core.

Table 2.4-19

Radwaste System Failure Accident Meteorology and Dose Results

Data in the DCD applies to the initial core.

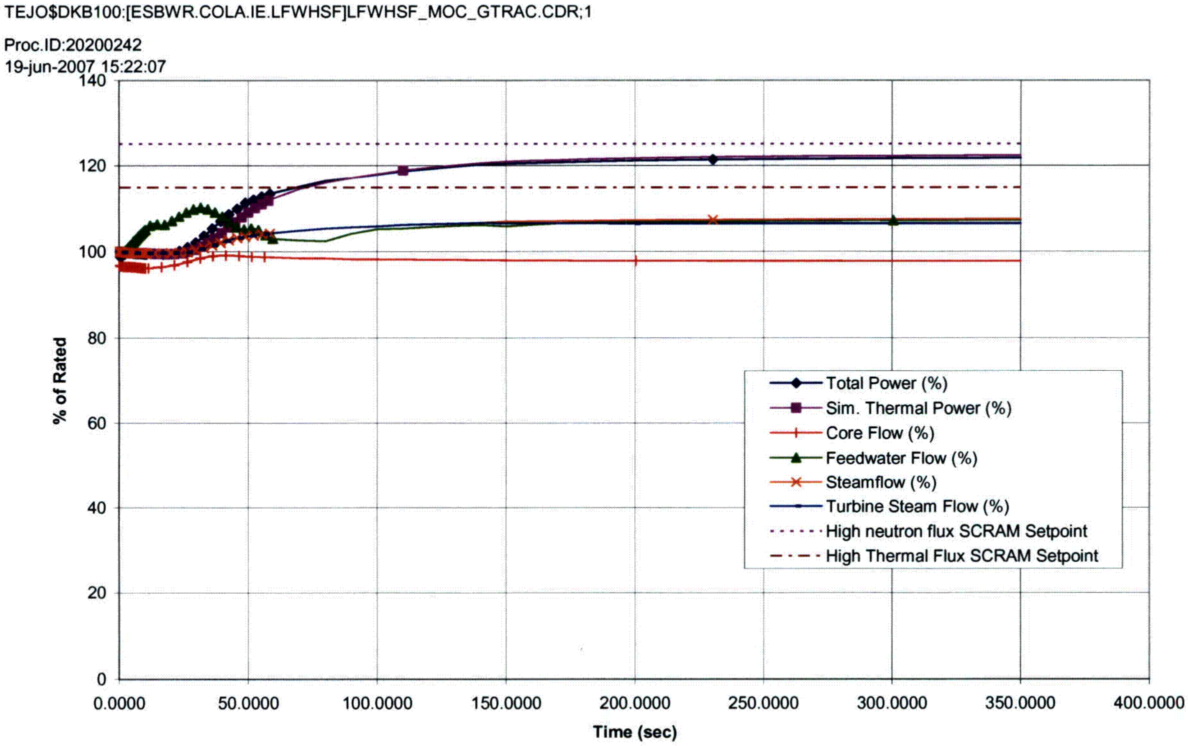


Figure 2.4-1a. Loss of Feedwater Heating with SCRRI/SRI Failure

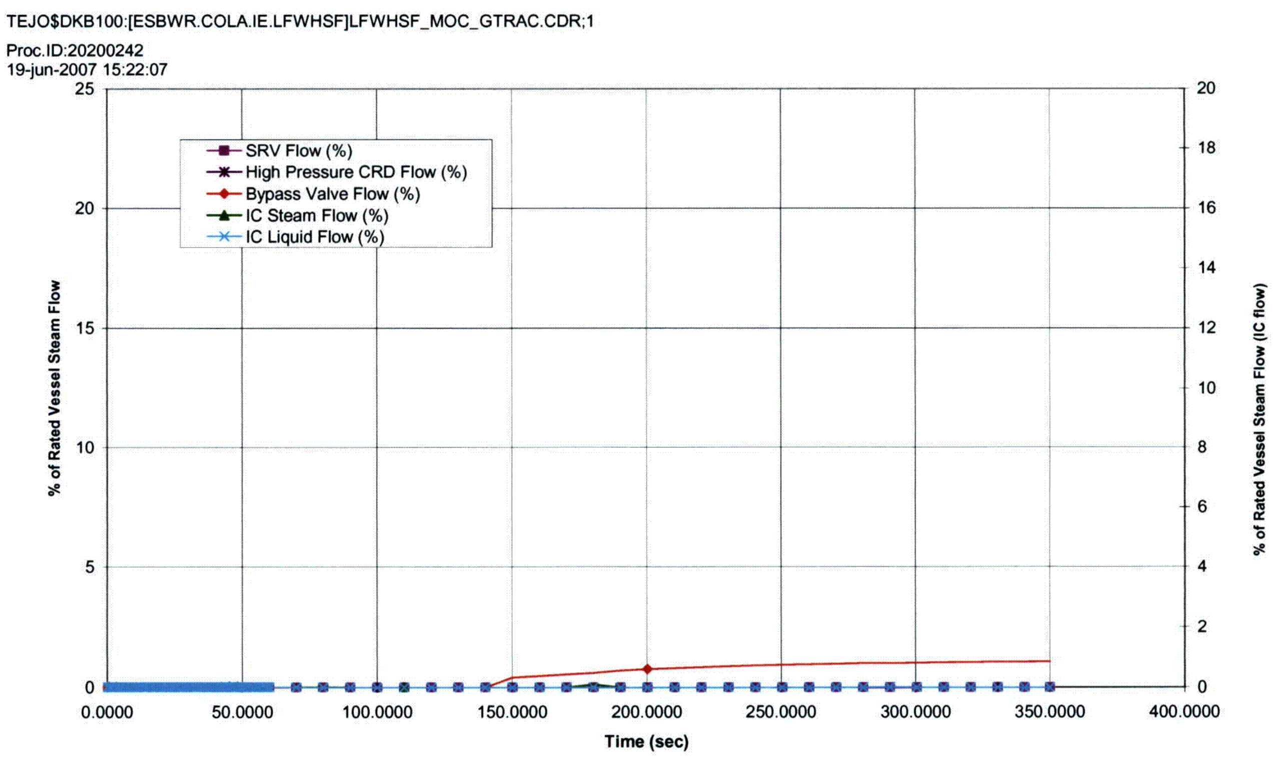


Figure 2.4-1b. Loss of Feedwater Heating with SCRRI/SRI Failure

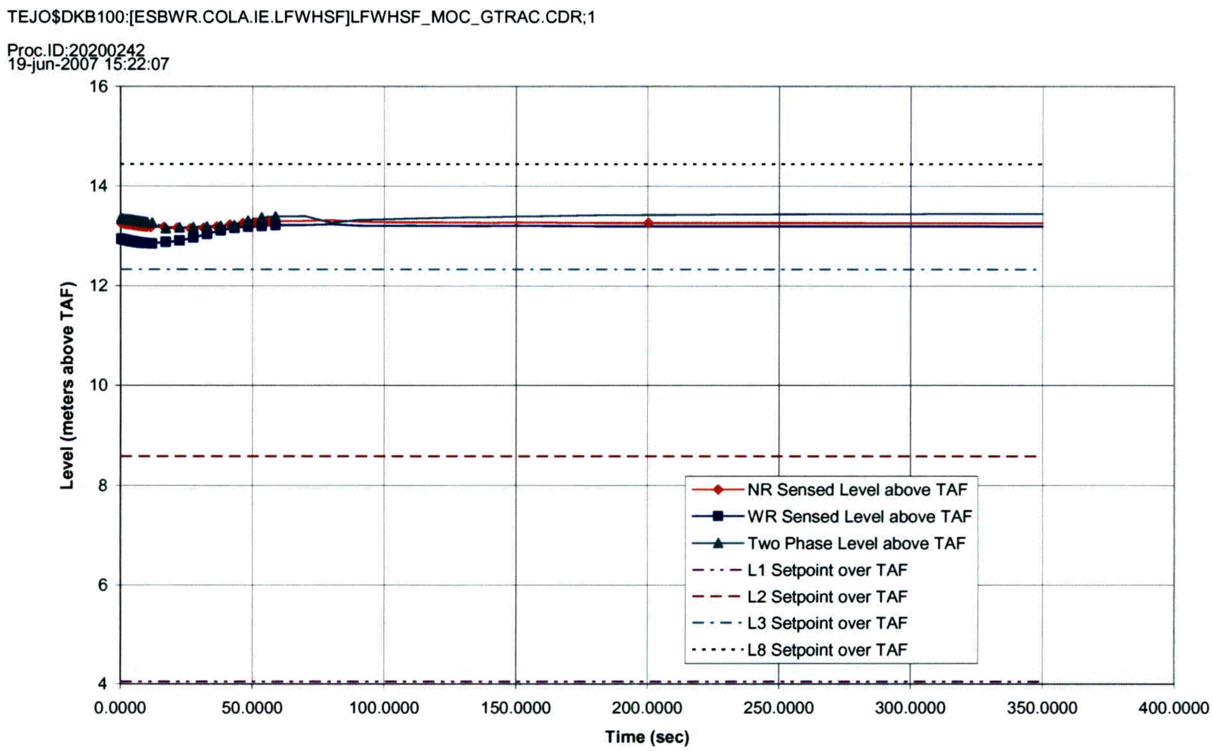


Figure 2.4-1c. Loss of Feedwater Heating with SCRRI/SRI Failure

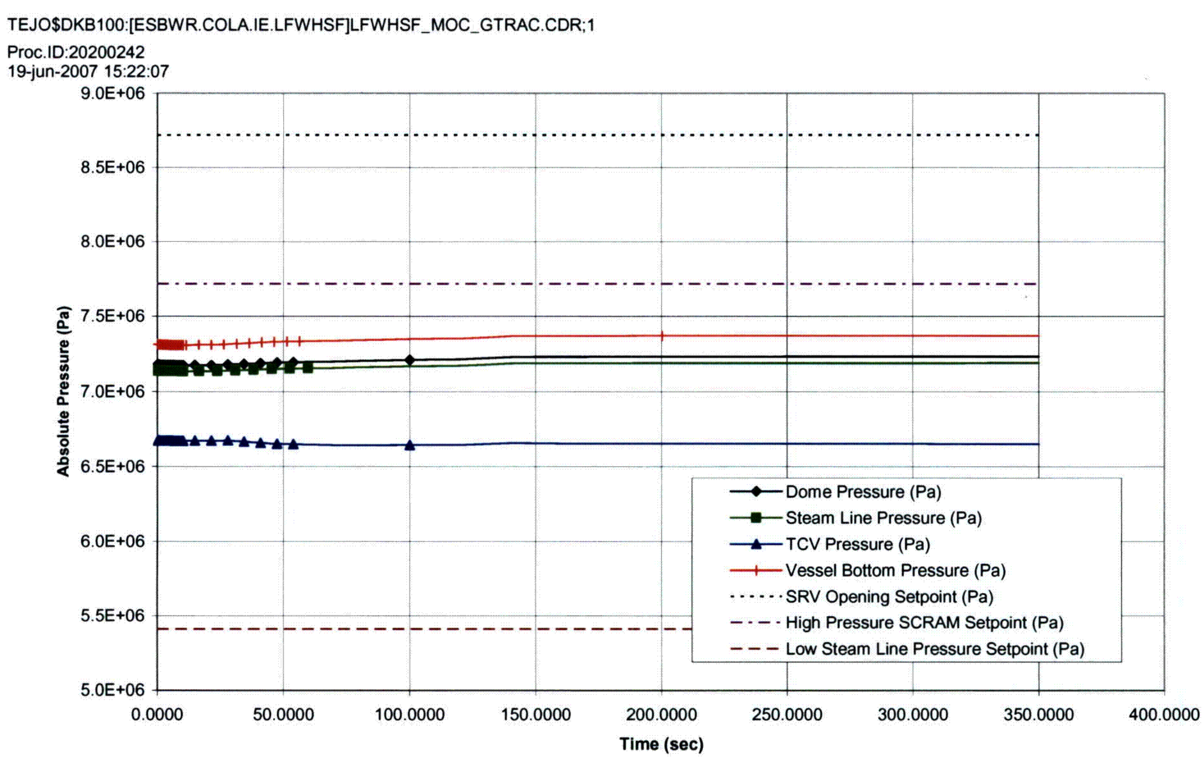


Figure 2.4-1d. Loss of Feedwater Heating with SCRRI/SRI Failure

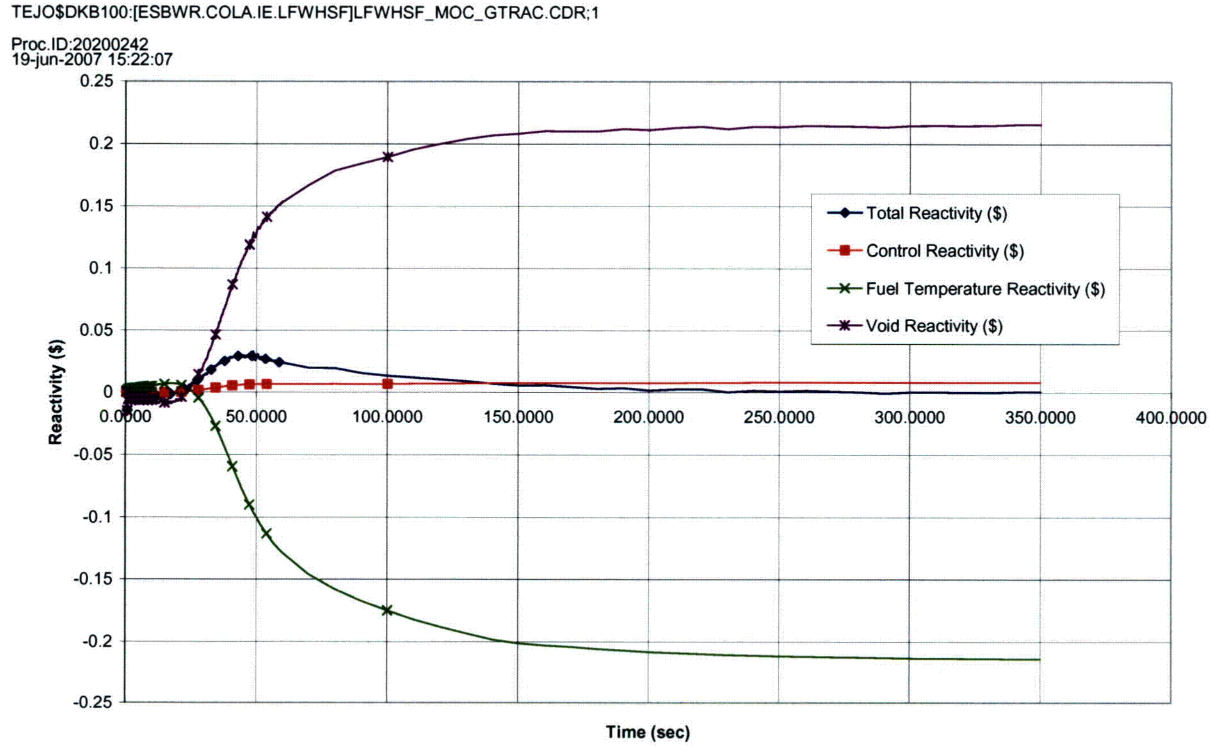


Figure 2.4-1e. Loss of Feedwater Heating with SCRRI/SRI Failure

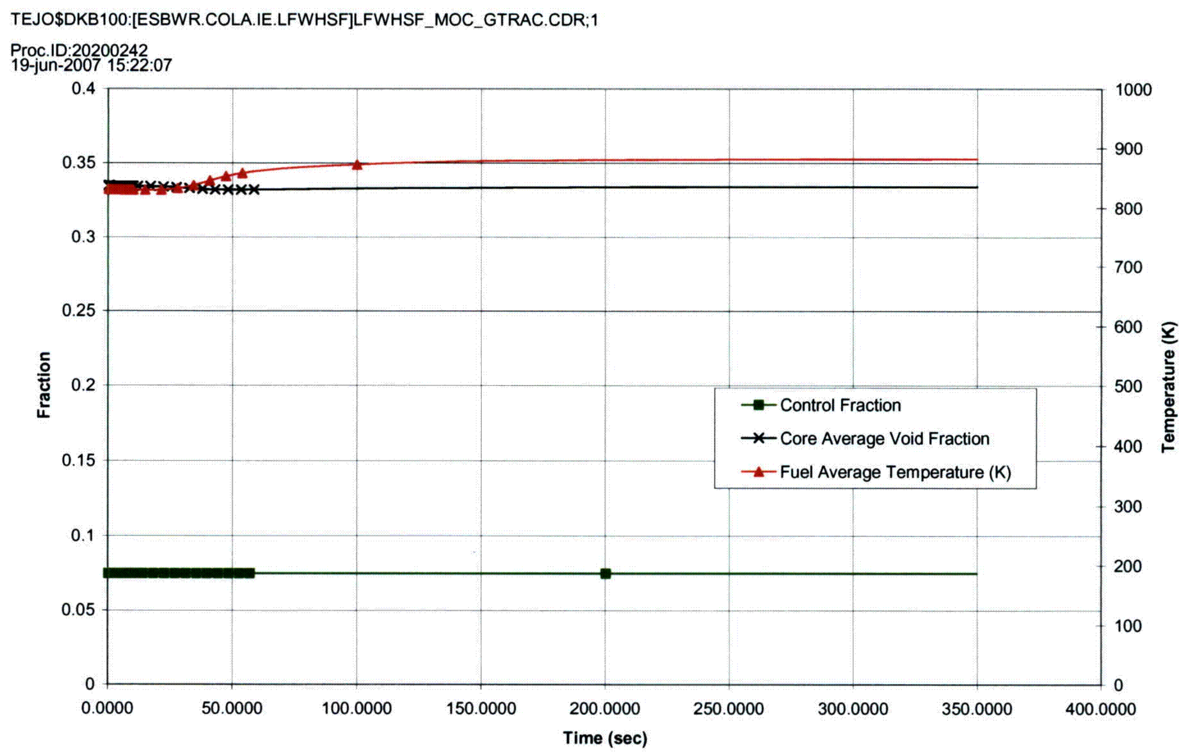


Figure 2.4-1f. Loss of Feedwater Heating with SCRRI/SRI Failure

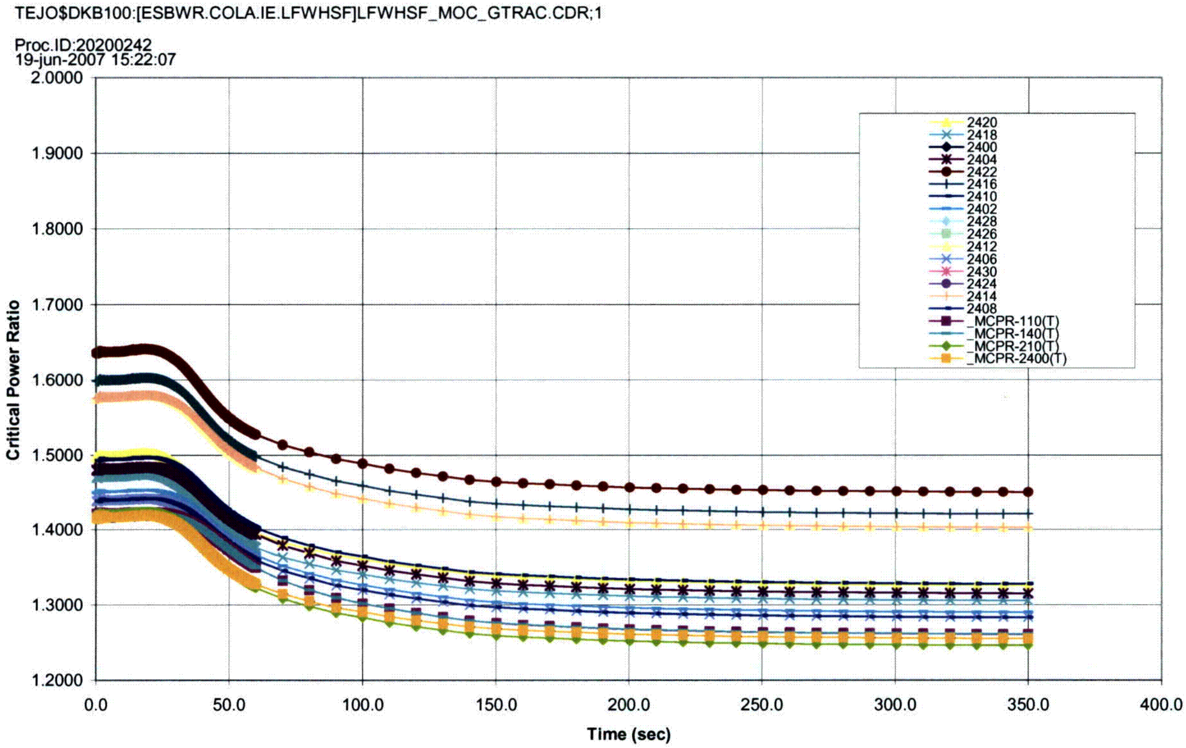


Figure 2.4-1g. Loss of Feedwater Heating with SCRRI/SRI Failure

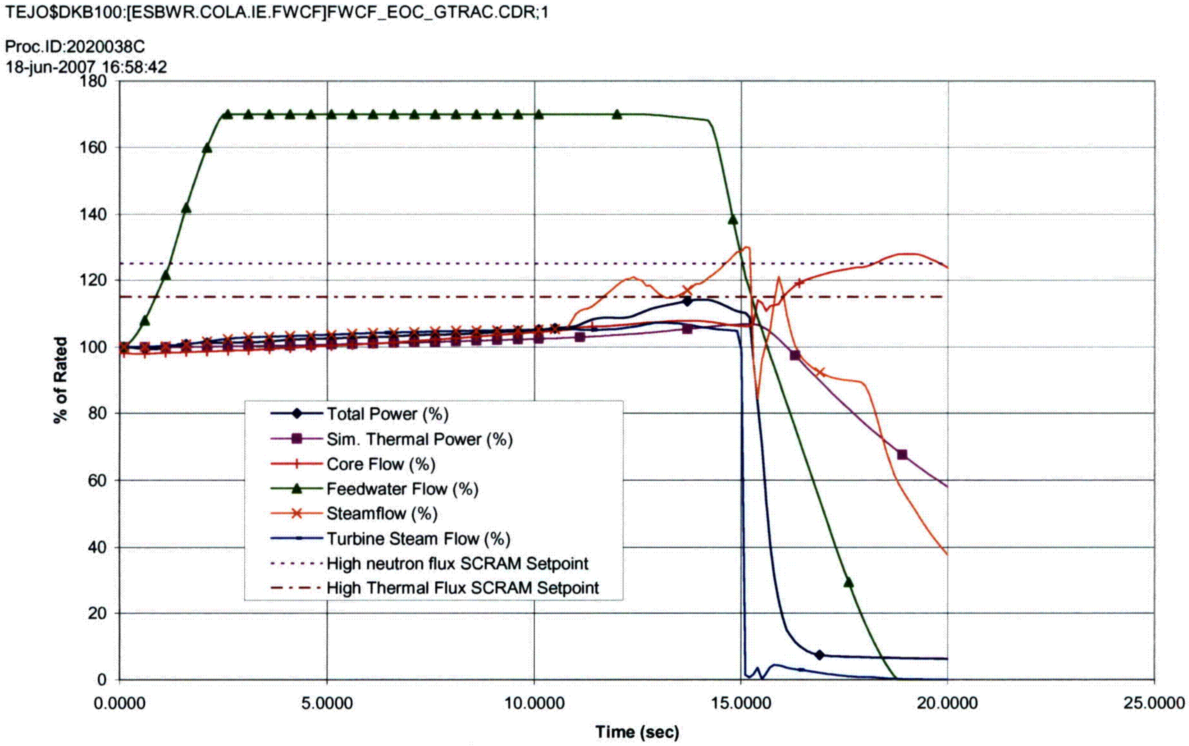


Figure 2.4-2a. Feedwater Controller Failure – Maximum Demand

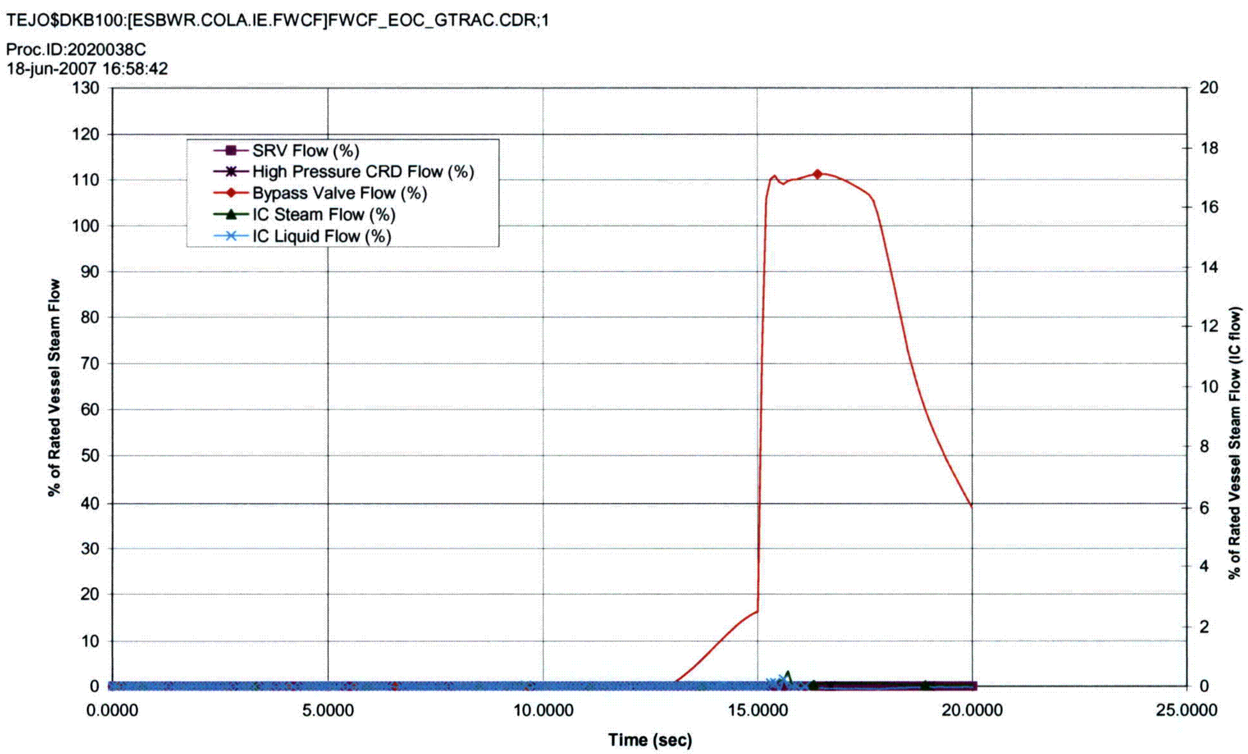


Figure 2.4-2b. Feedwater Controller Failure – Maximum Demand

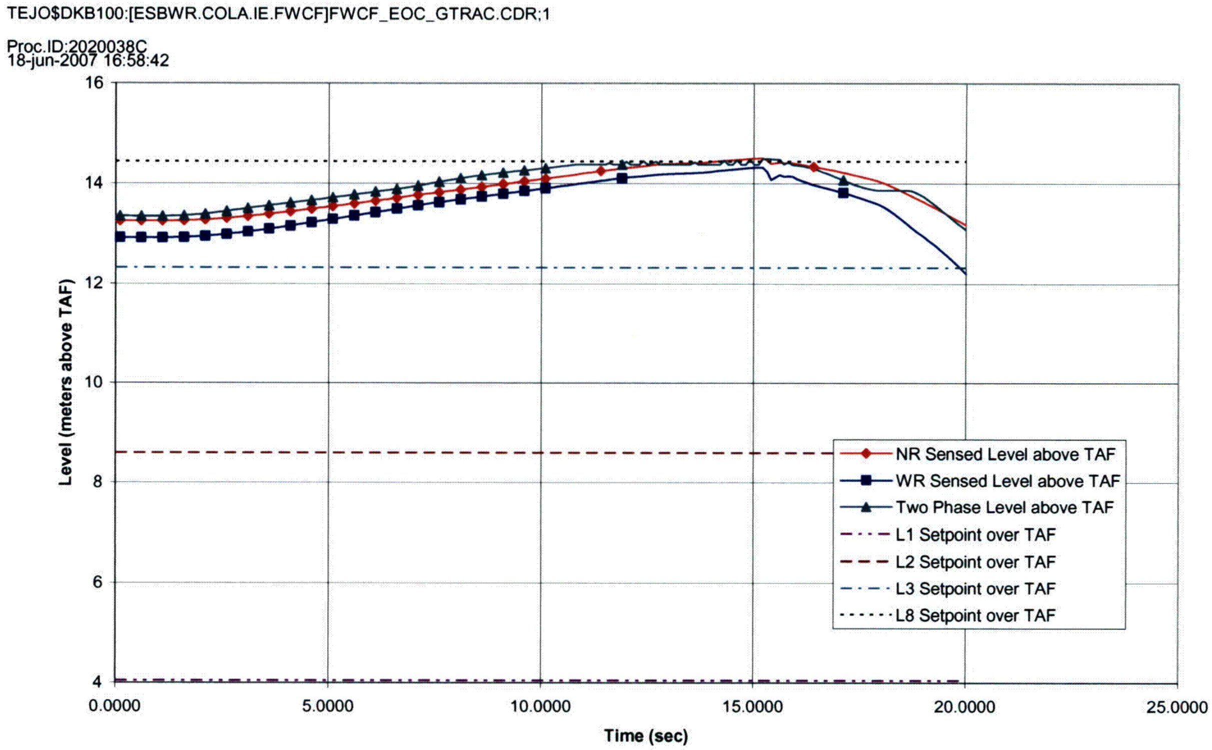


Figure 2.4-2c. Feedwater Controller Failure – Maximum Demand

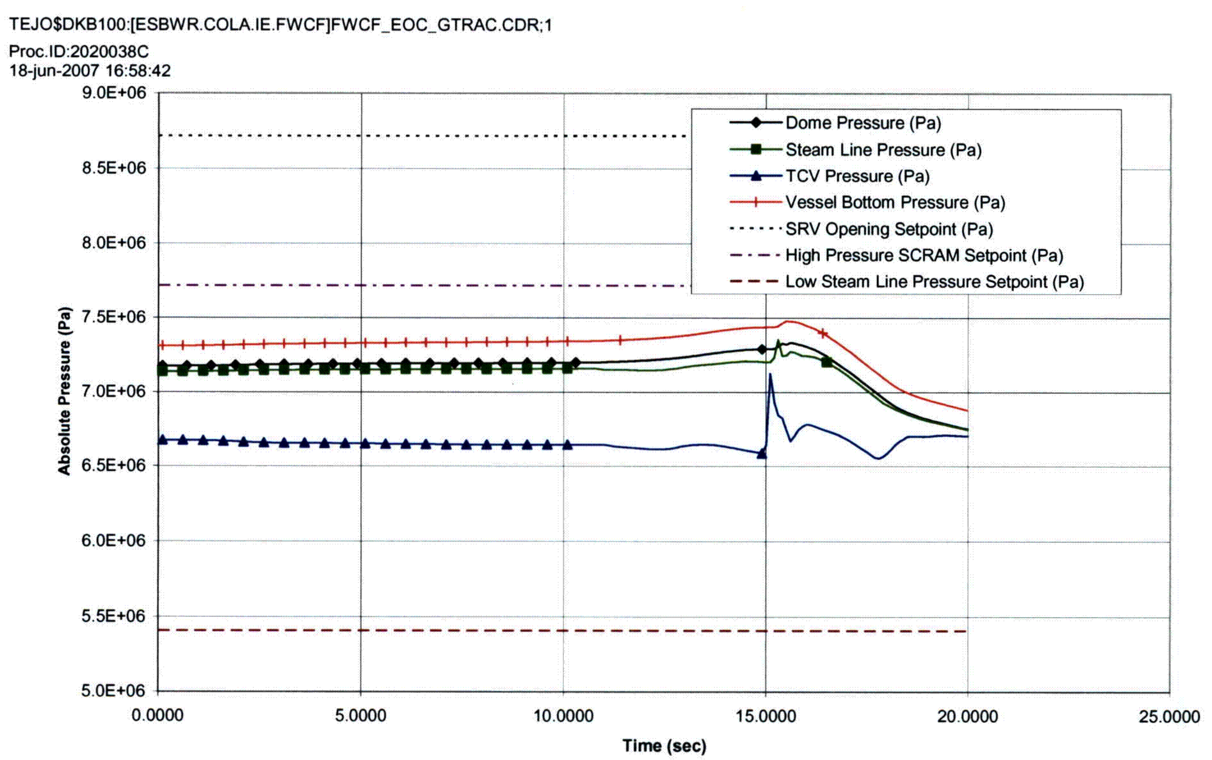


Figure 2.4-2d. Feedwater Controller Failure – Maximum Demand

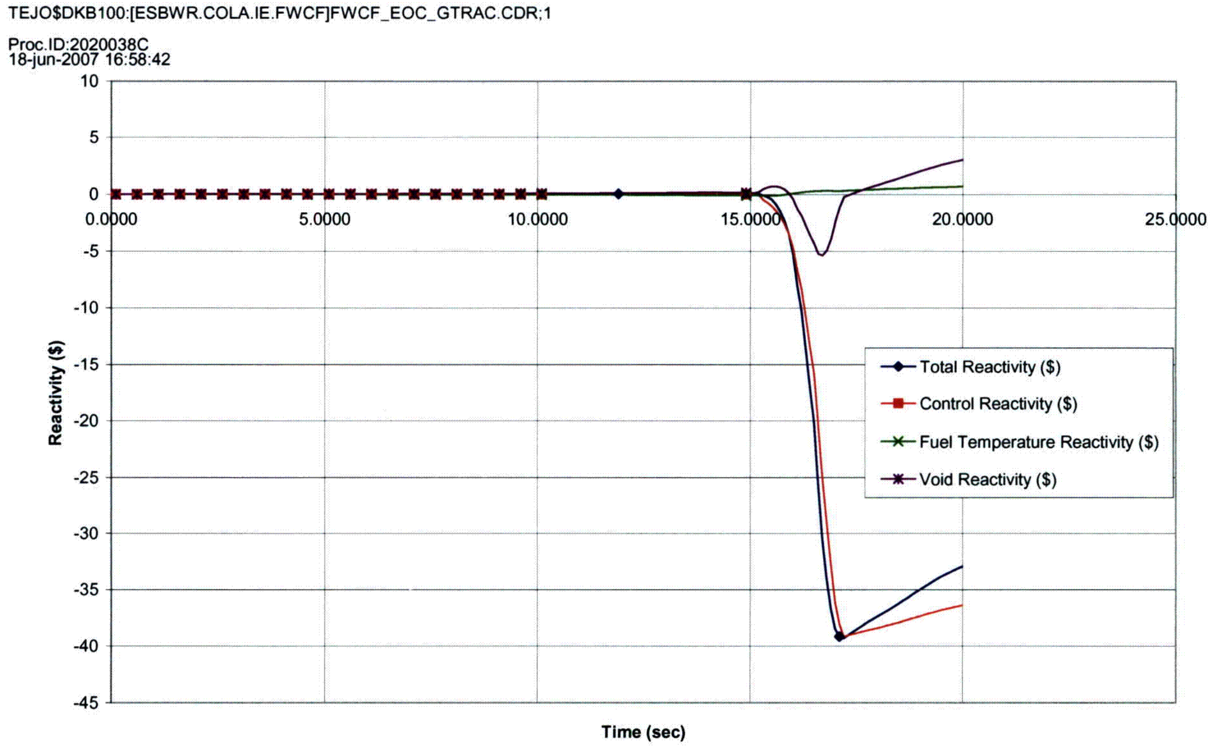


Figure 2.4-2e. Feedwater Controller Failure – Maximum Demand

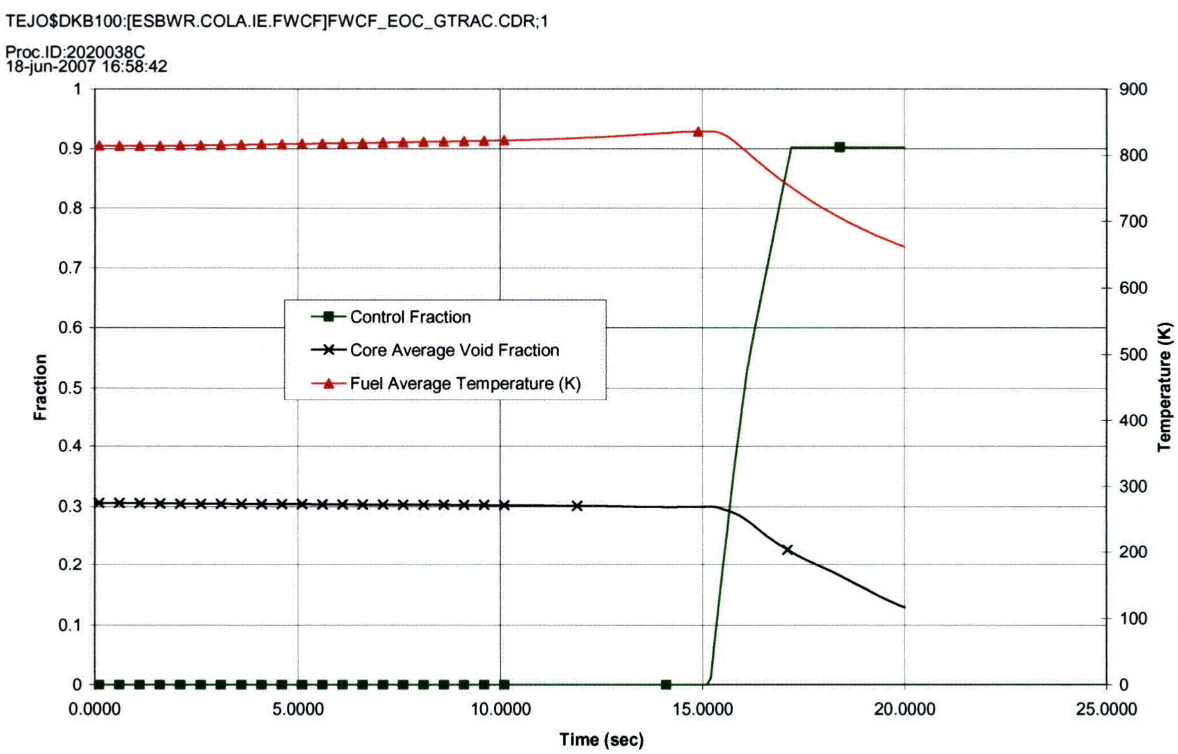


Figure 2.4-2f. Feedwater Controller Failure – Maximum Demand

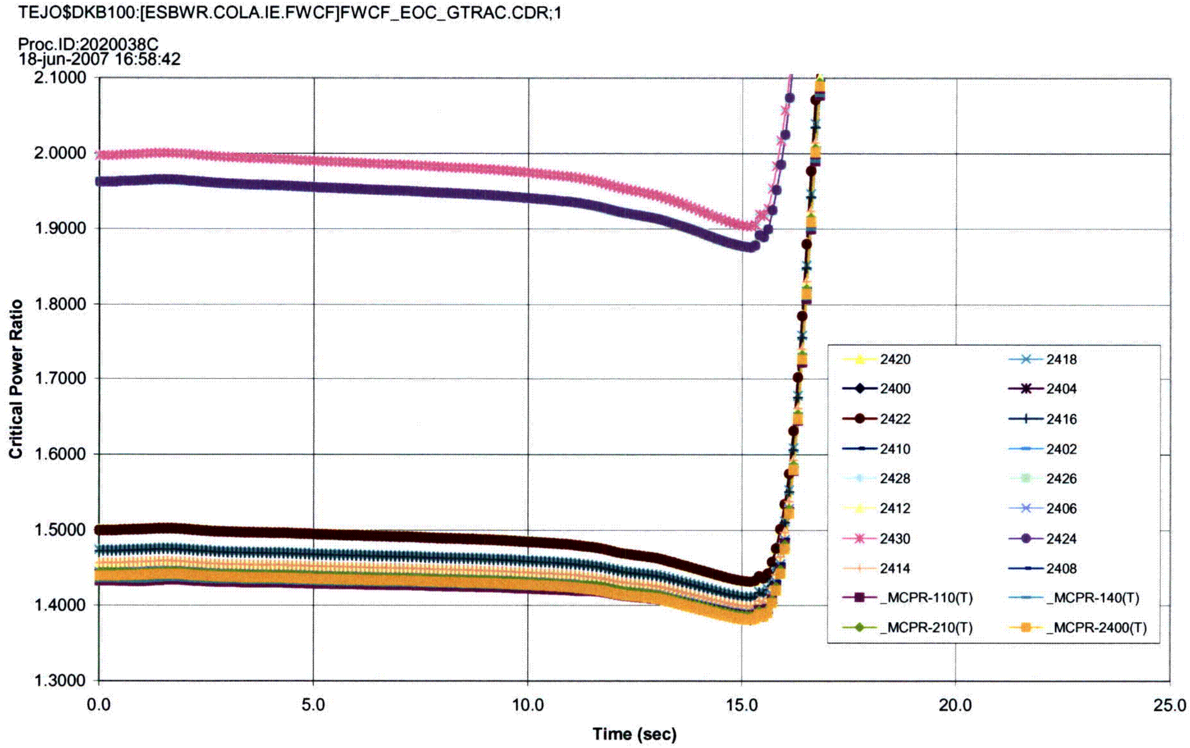


Figure 2.4-2g. Feedwater Controller Failure – Maximum Demand

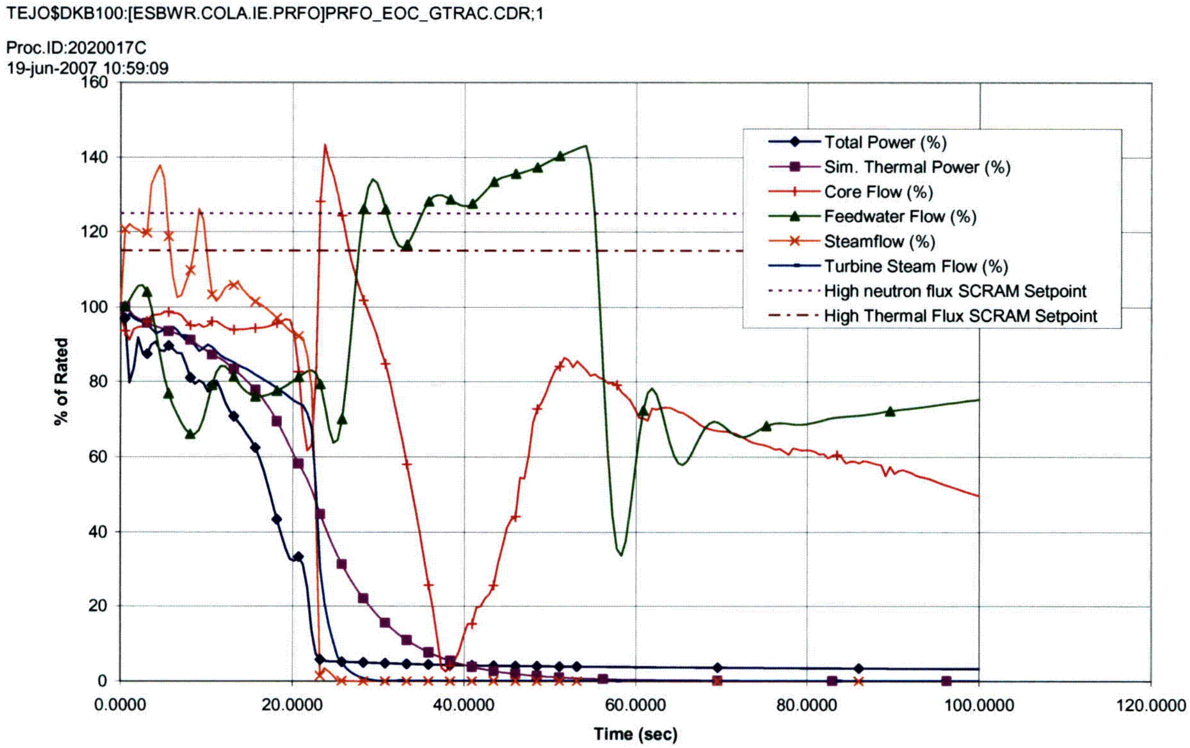


Figure 2.4-3a. Pressure Regulator Failure – Opening of All Turbine Control and Bypass Valves

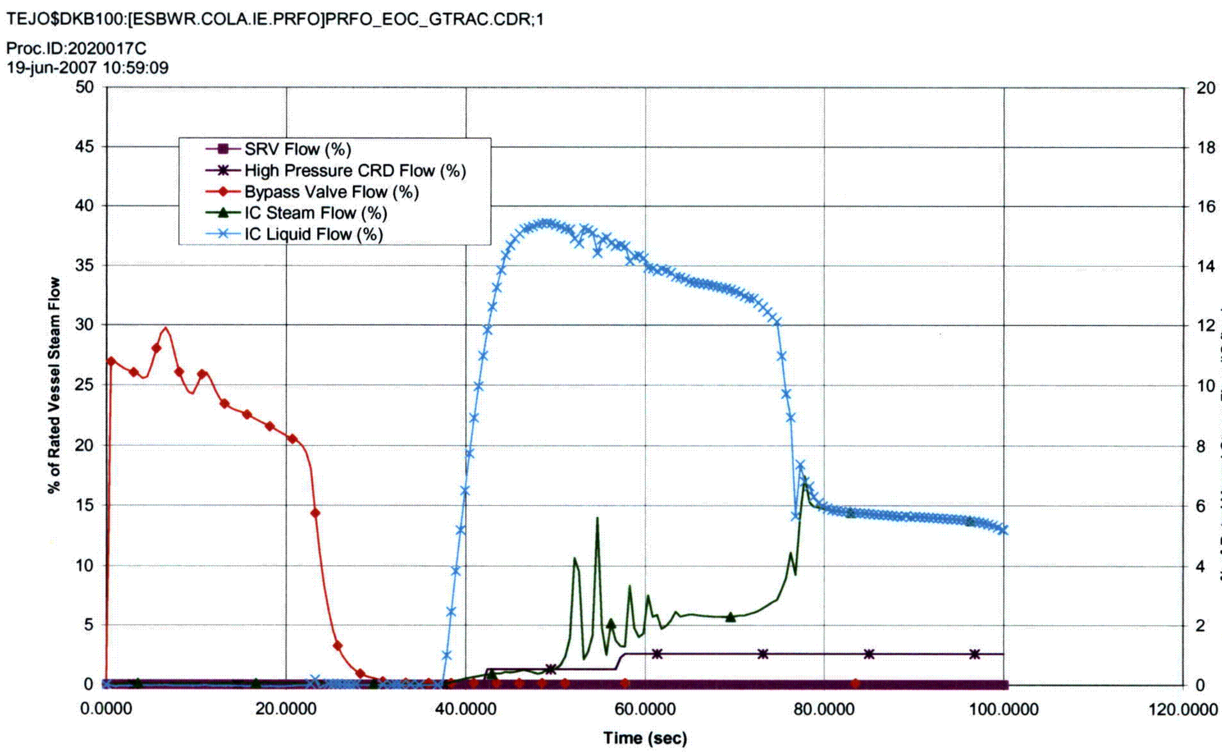


Figure 2.4-3b. Pressure Regulator Failure – Opening of All Turbine Control and Bypass Valves

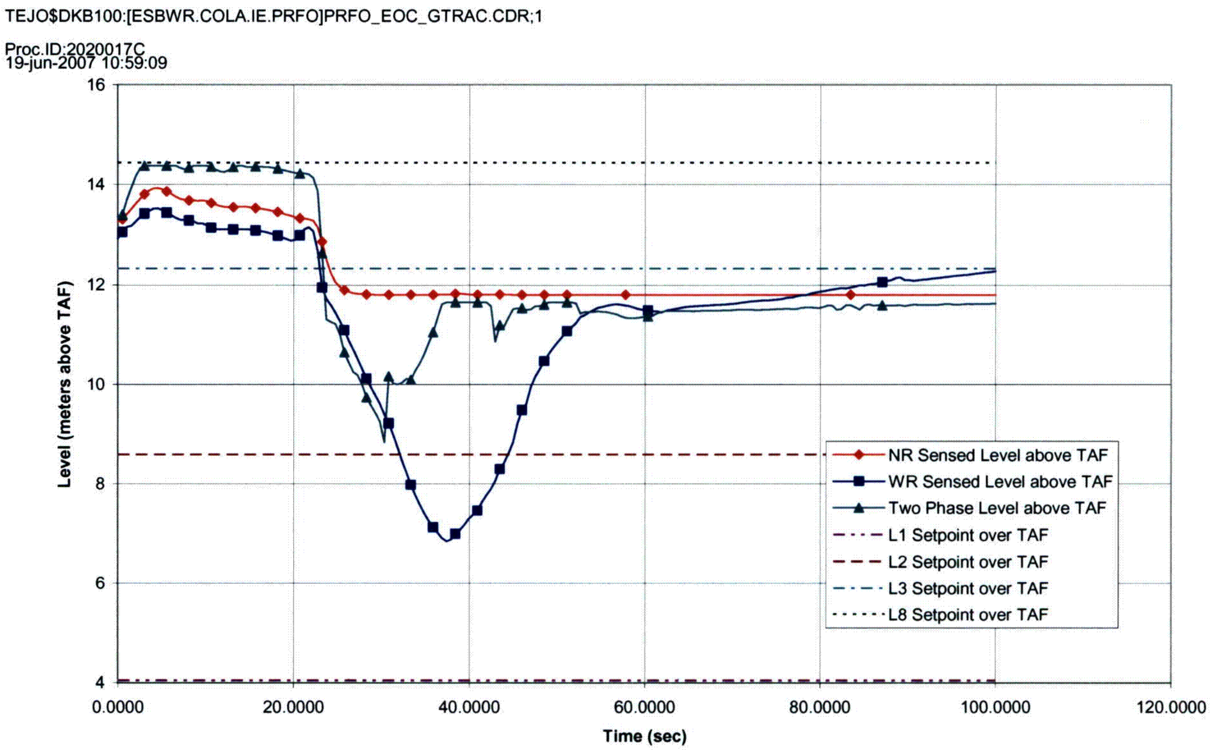


Figure 2.4-3c. Pressure Regulator Failure – Opening of All Turbine Control and Bypass Valves

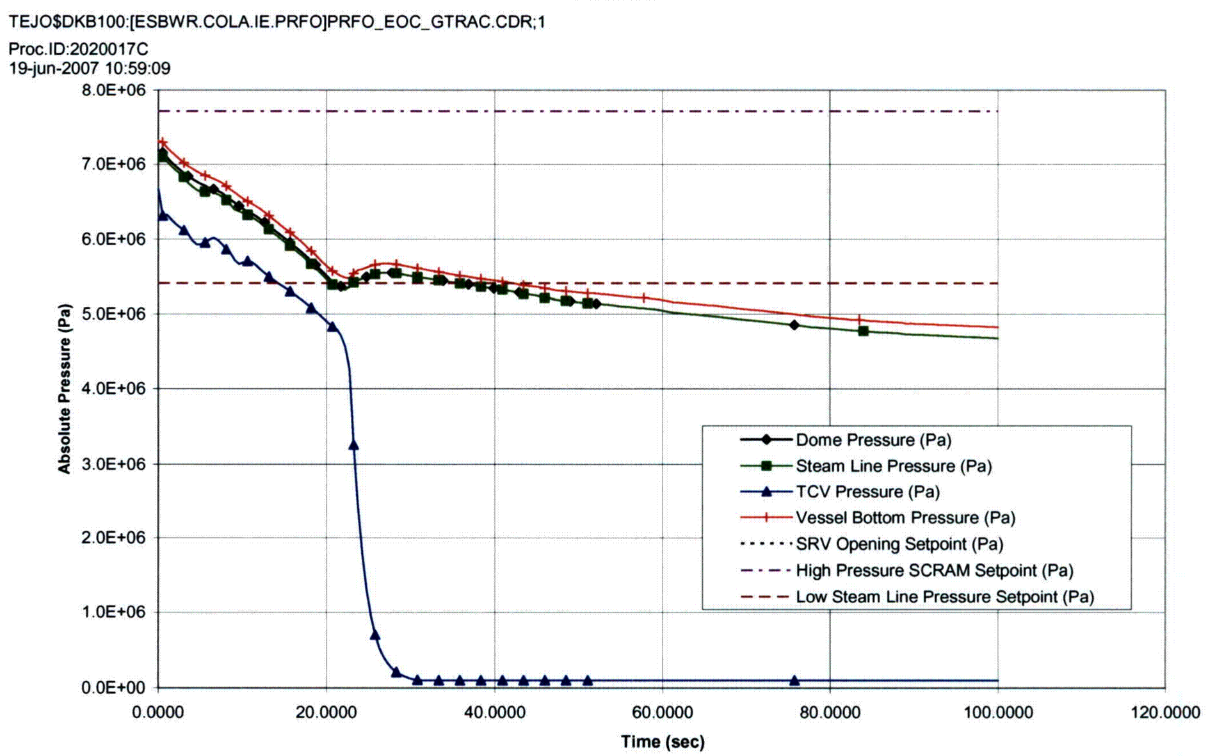


Figure 2.4-3d. Pressure Regulator Failure – Opening of All Turbine Control and Bypass Valves

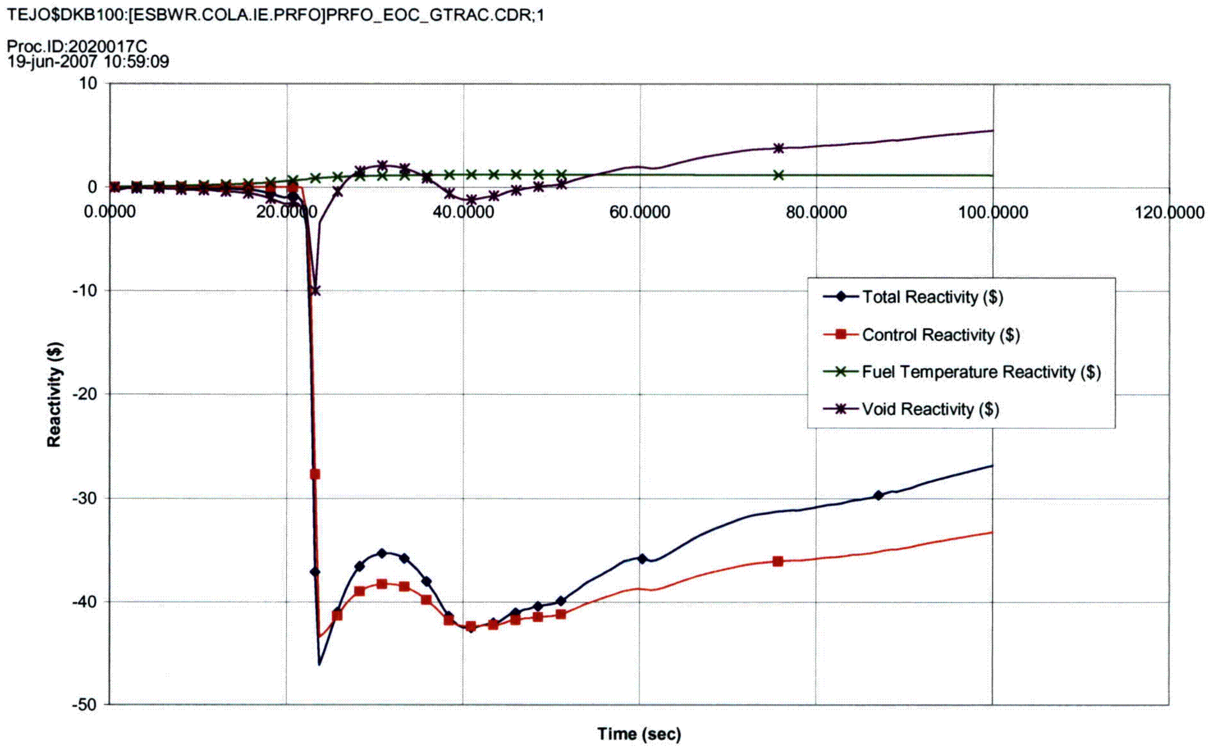


Figure 2.4-3e. Pressure Regulator Failure – Opening of All Turbine Control and Bypass Valves

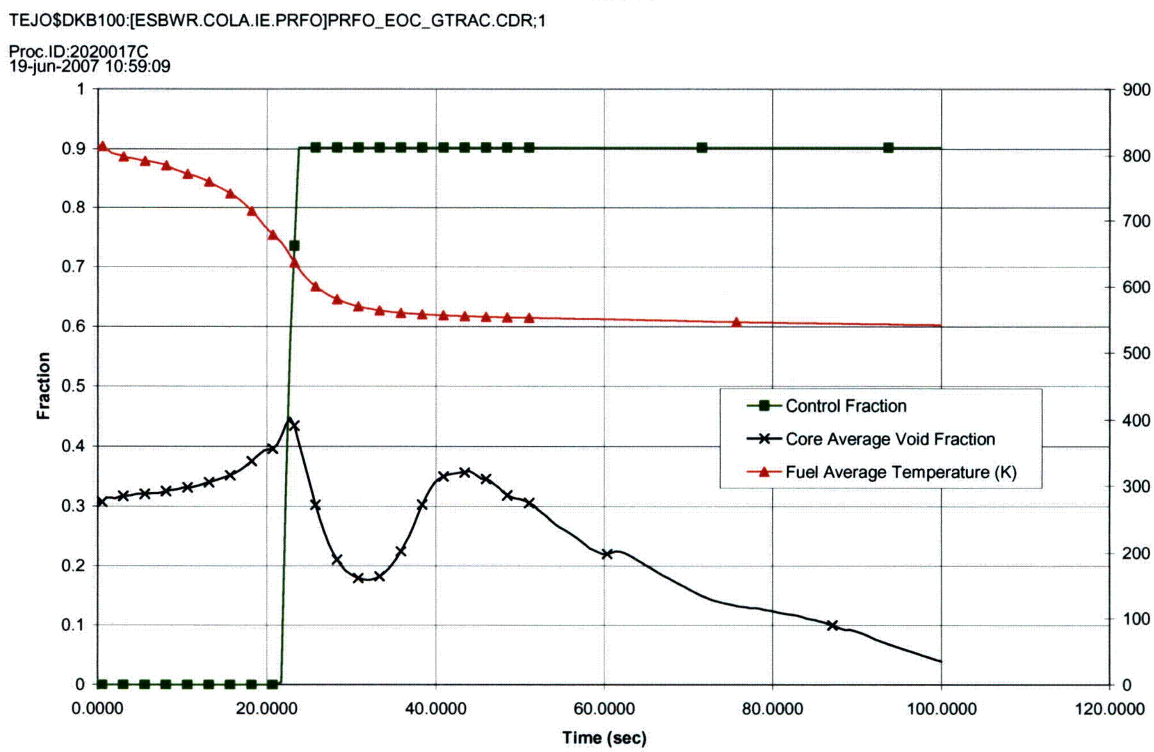


Figure 2.4-3f. Pressure Regulator Failure – Opening of All Turbine Control and Bypass Valves

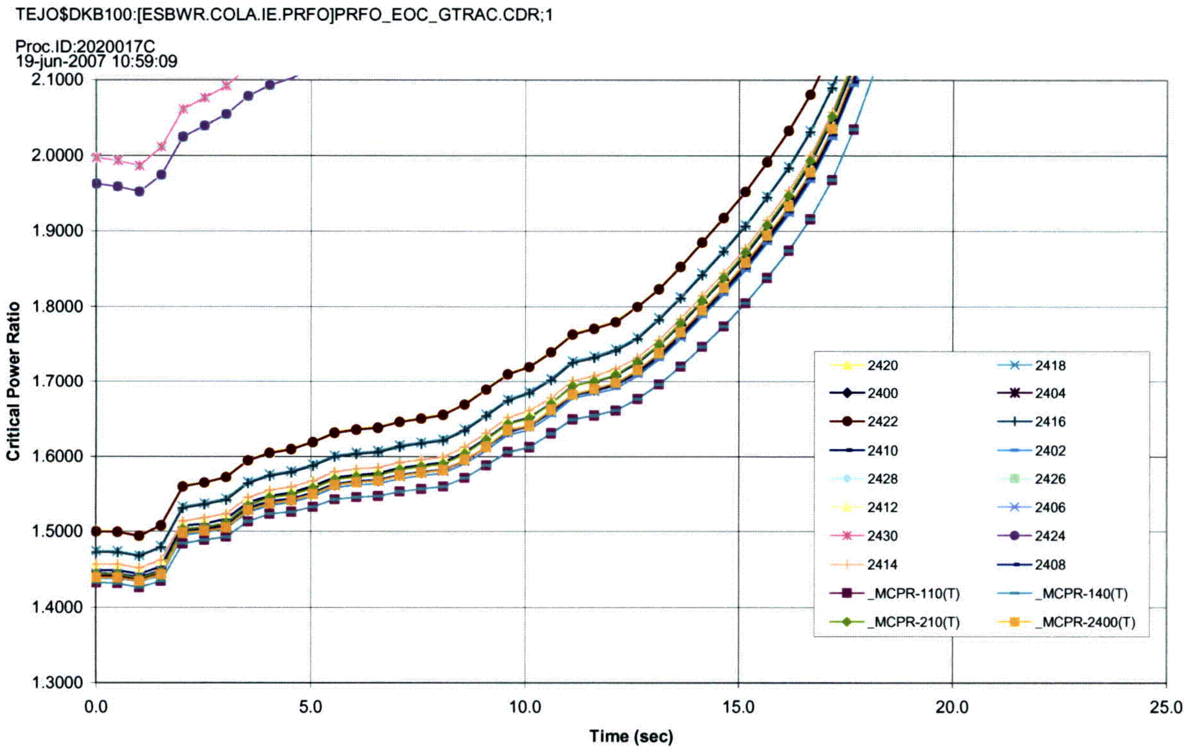


Figure 2.4-3g. Pressure Regulator Failure – Opening of All Turbine Control and Bypass Valves

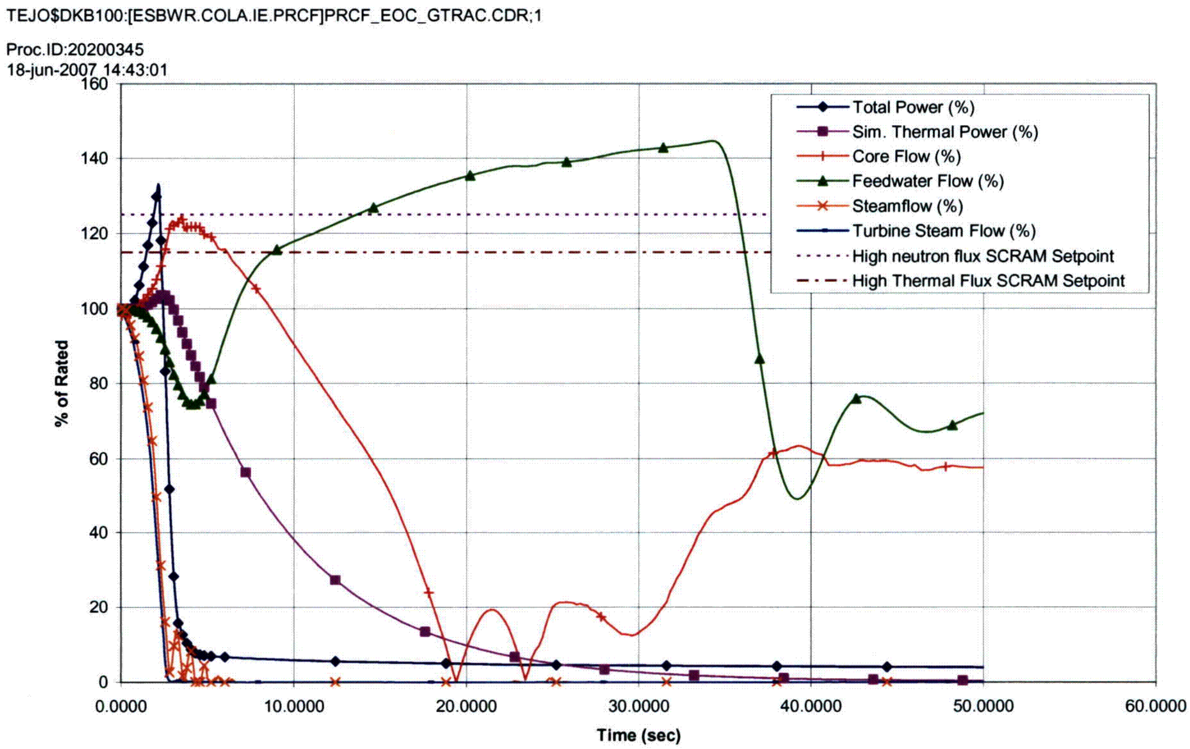


Figure 2.4-4a. Pressure Regulator Failure – Closure of All Turbine Control and Bypass Valves

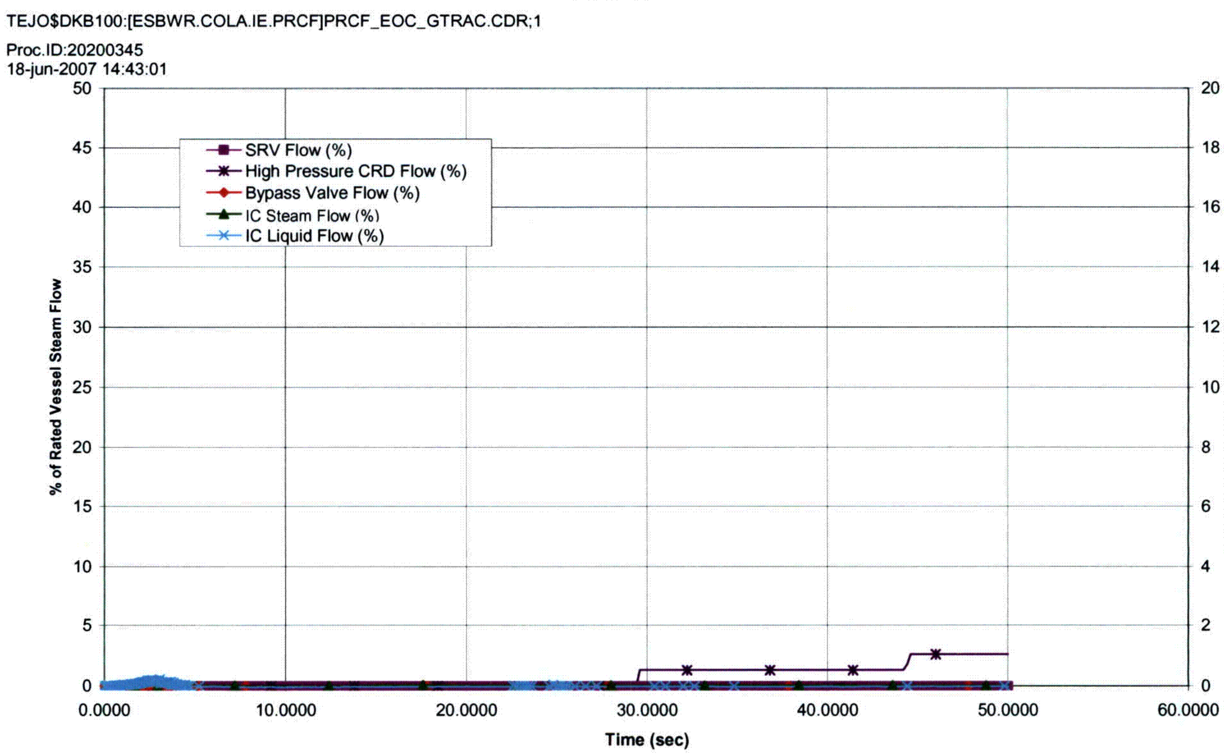


Figure 2.4-4b. Pressure Regulator Failure – Closure of All Turbine Control and Bypass Valves

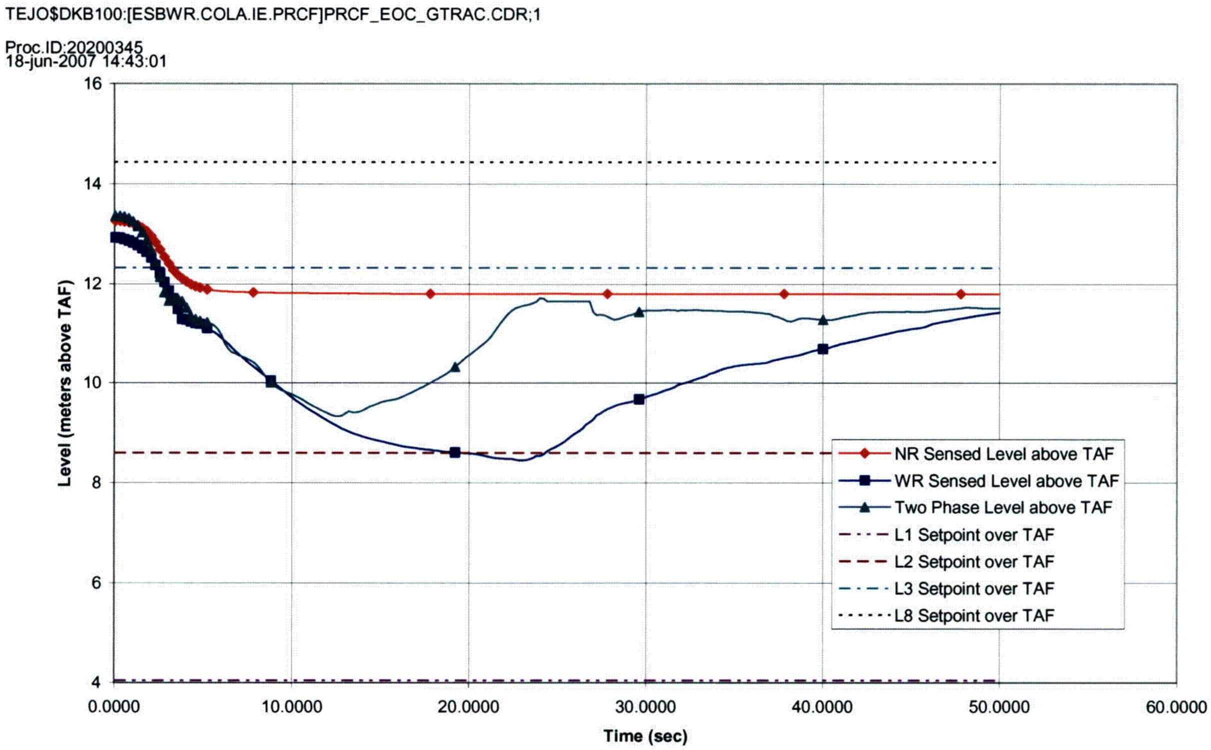


Figure 2.4-4c. Pressure Regulator Failure – Closure of All Turbine Control and Bypass Valves

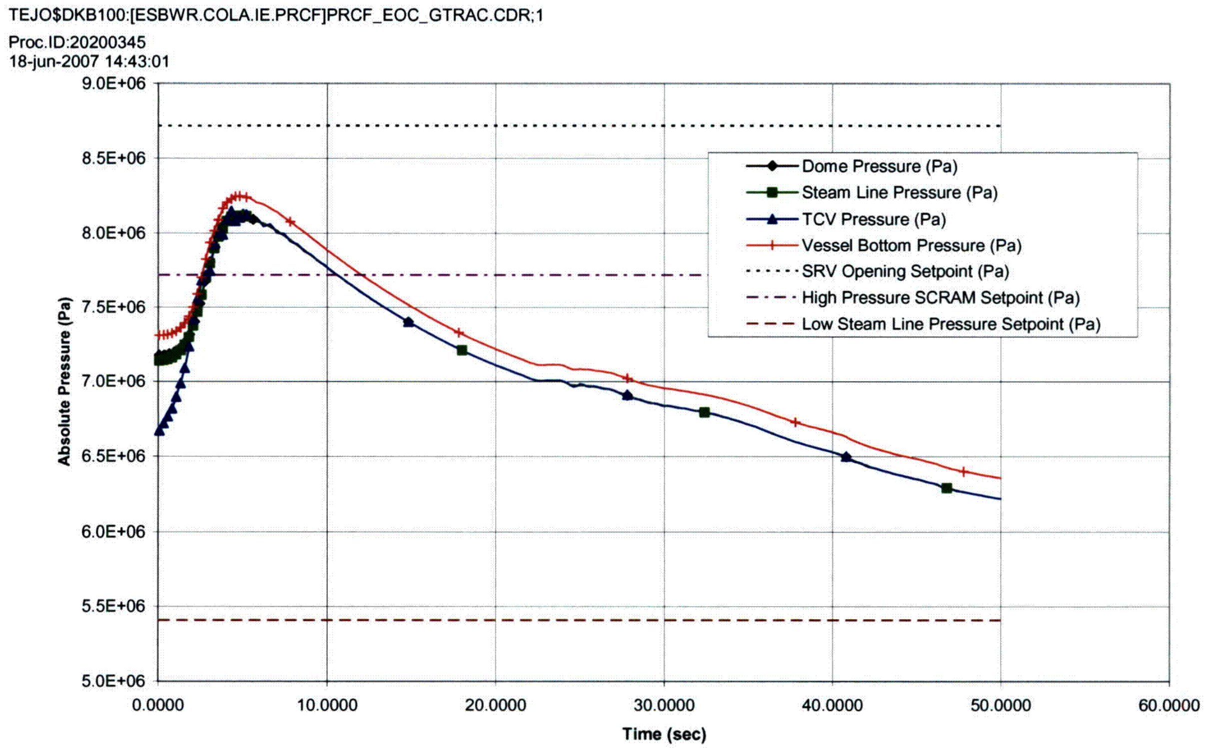


Figure 2.4-4d. Pressure Regulator Failure – Closure of All Turbine Control and Bypass Valves

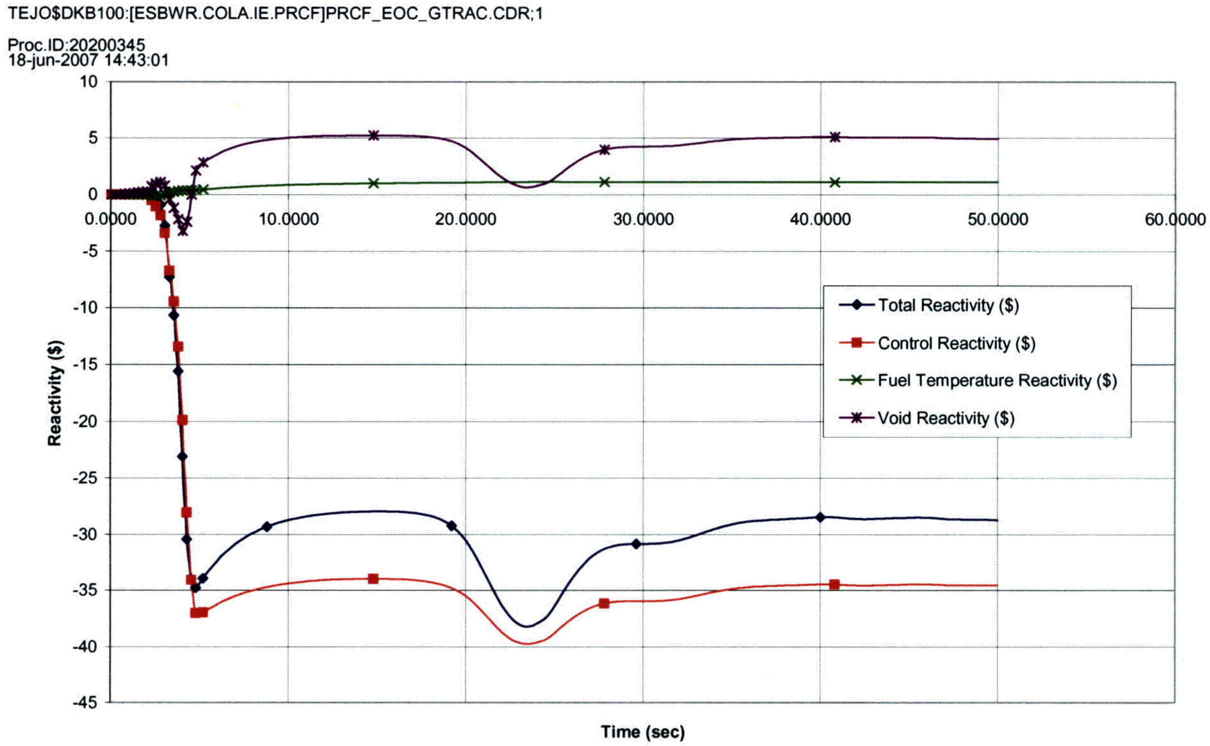


Figure 2.4-4e. Pressure Regulator Failure – Closure of All Turbine Control and Bypass Valves

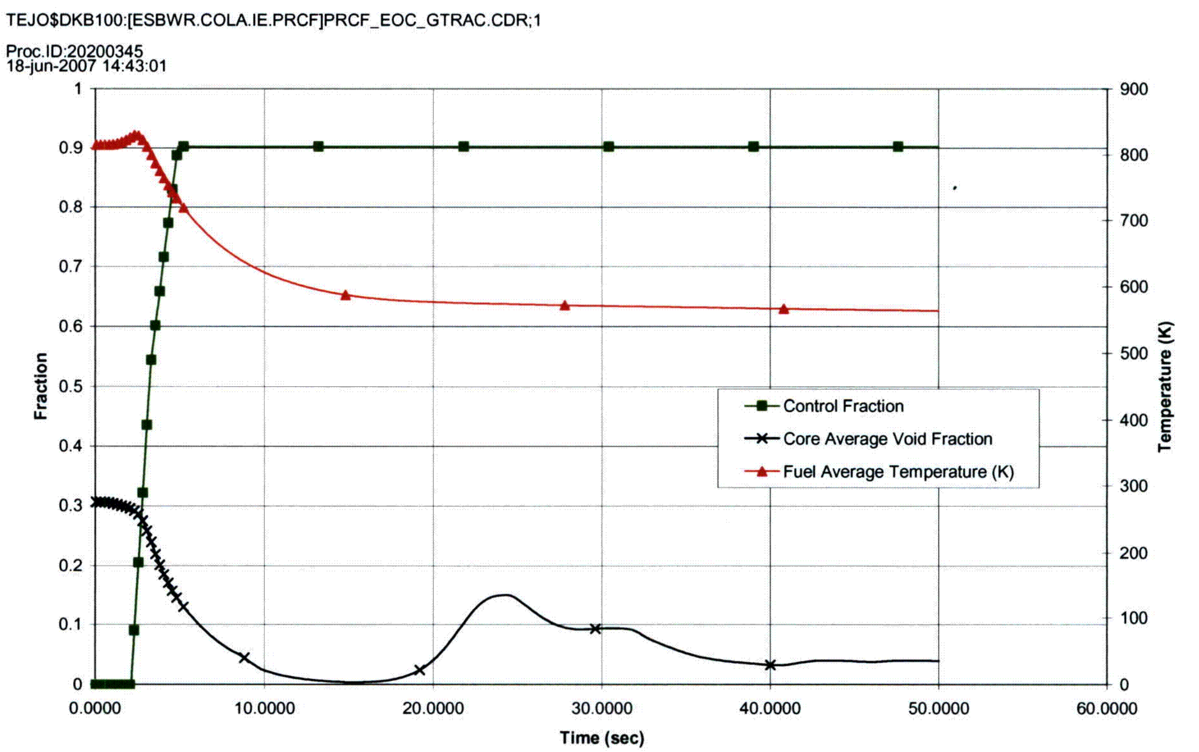


Figure 2.4-4f. Pressure Regulator Failure – Closure of All Turbine Control and Bypass Valves

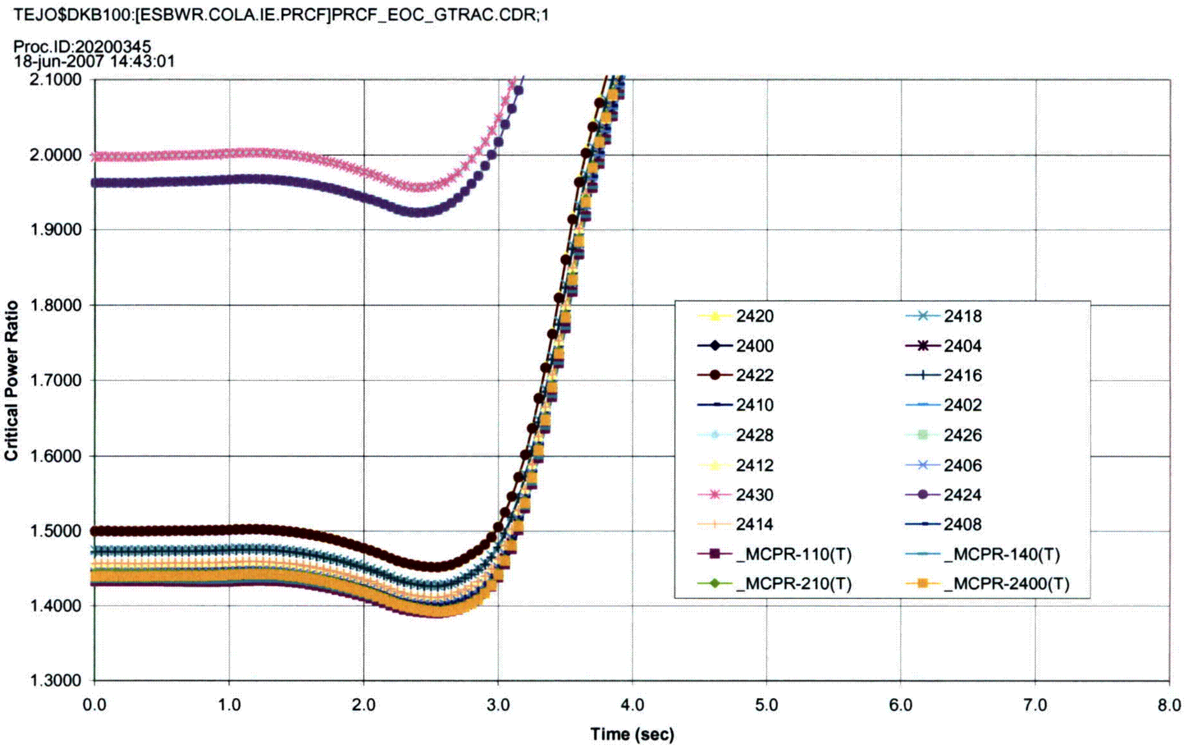


Figure 2.4-4g. Pressure Regulator Failure – Closure of All Turbine Control and Bypass Valves

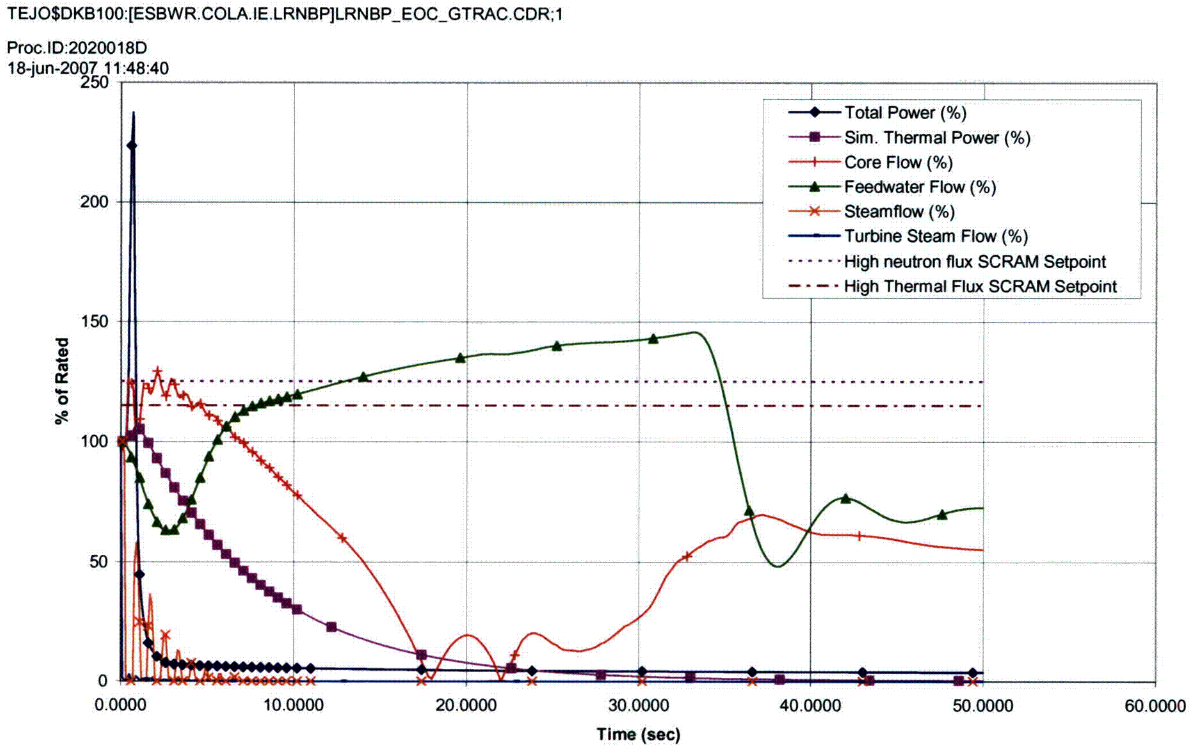


Figure 2.4-5a. Generator Load Rejection With Total Turbine Bypass Failure

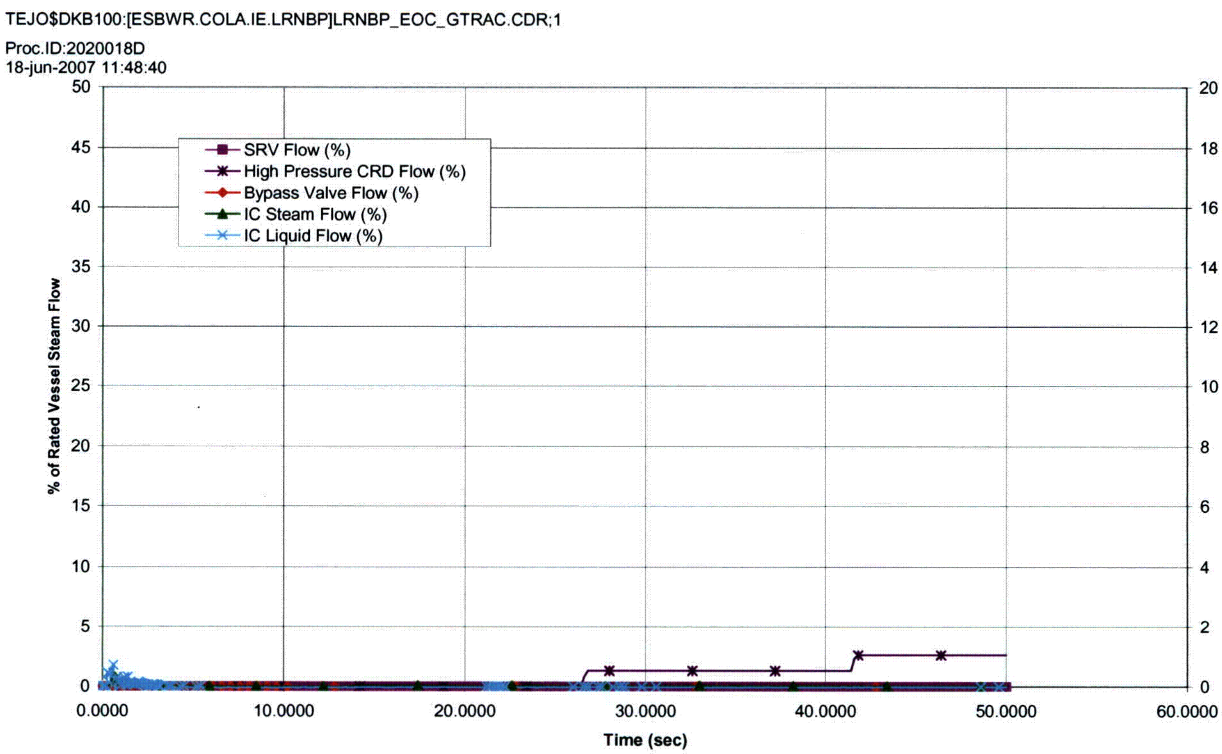


Figure 2.4-5b. Generator Load Rejection With Total Turbine Bypass Failure

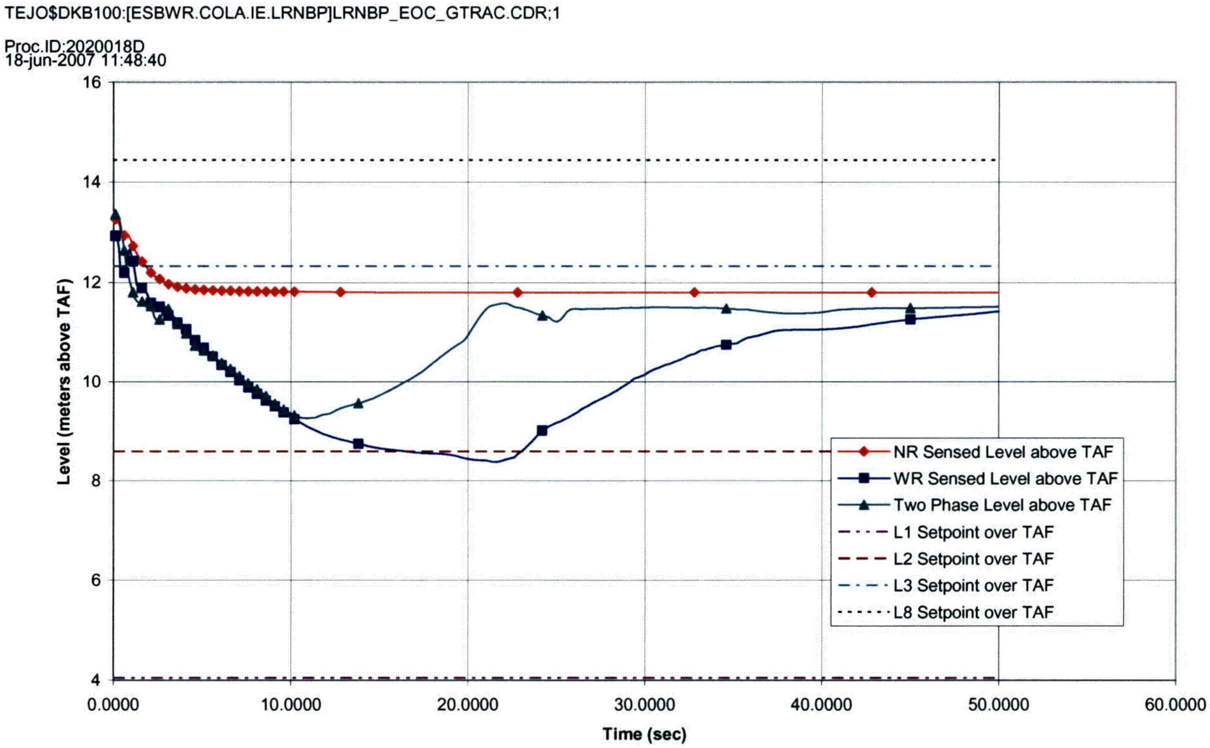


Figure 2.4-5c. Generator Load Rejection With Total Turbine Bypass Failure

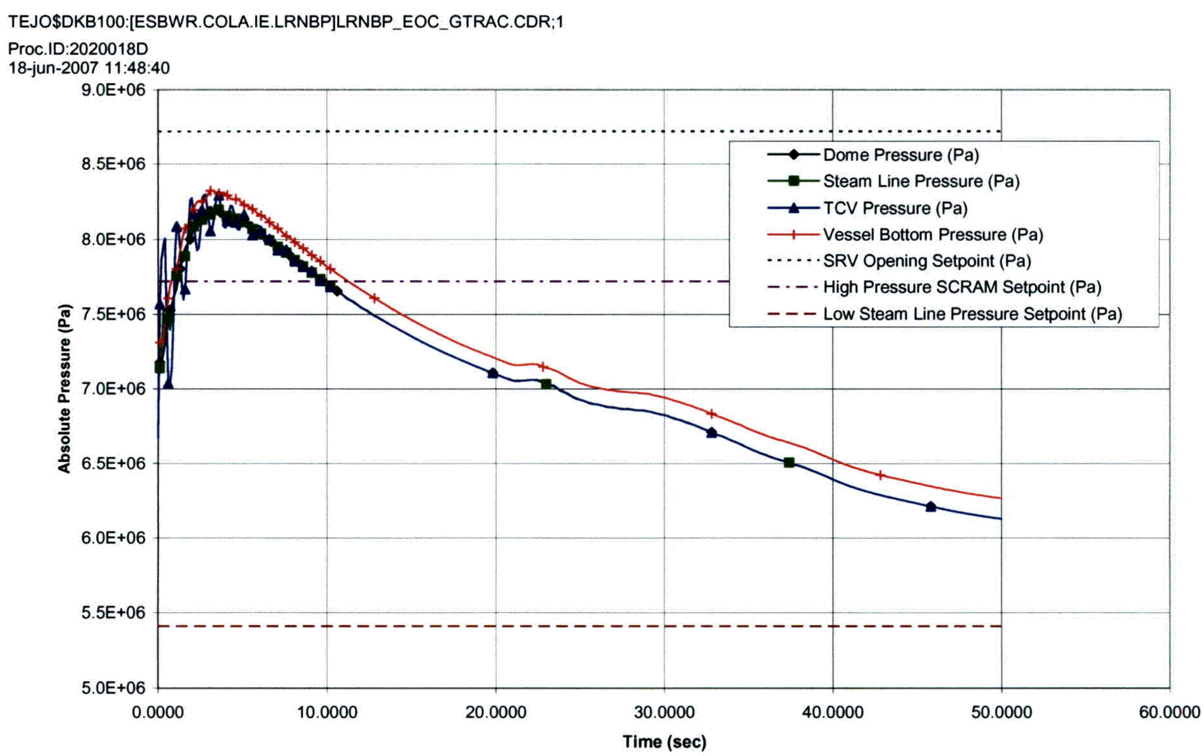


Figure 2.4-5d. Generator Load Rejection With Total Turbine Bypass Failure

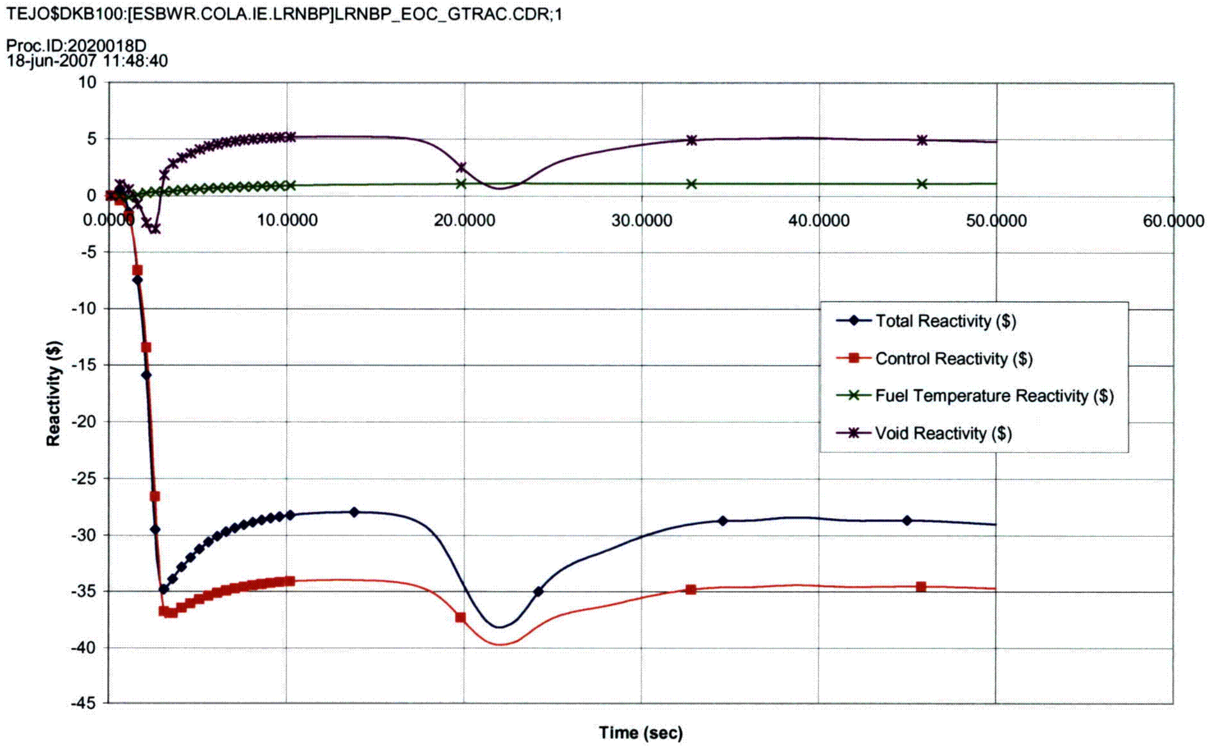


Figure 2.4-5e. Generator Load Rejection With Total Turbine Bypass Failure

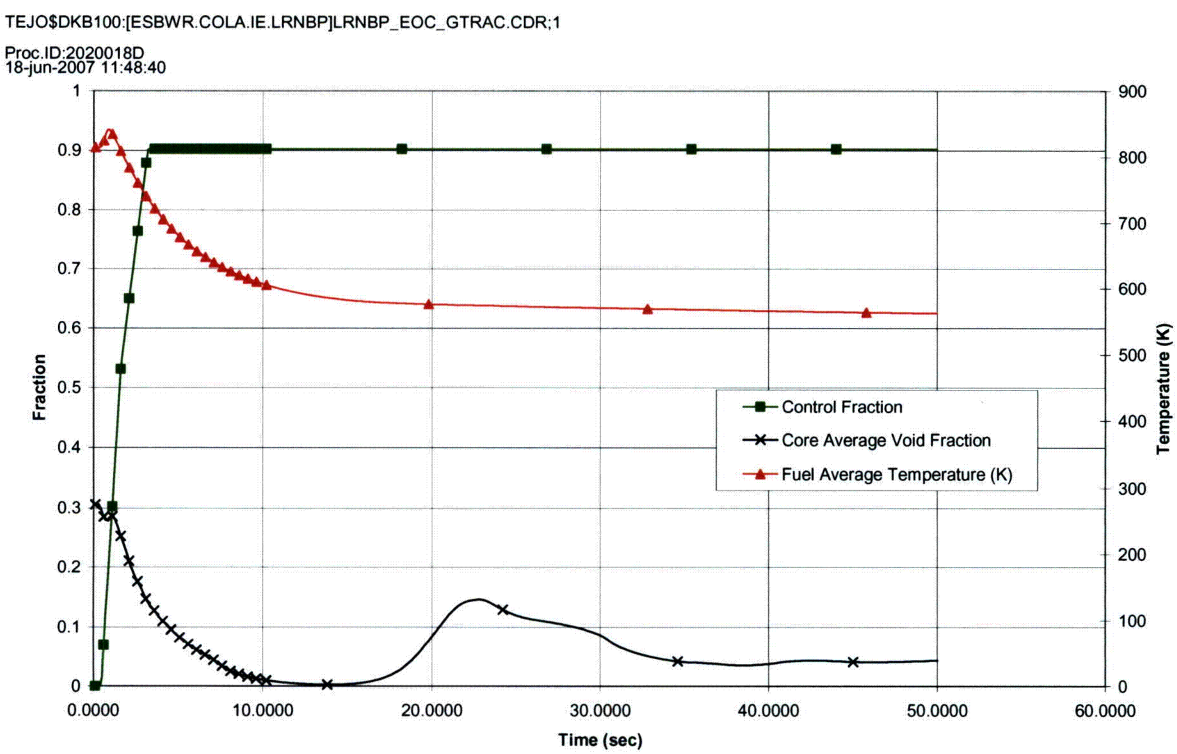


Figure 2.4-5f. Generator Load Rejection With Total Turbine Bypass Failure

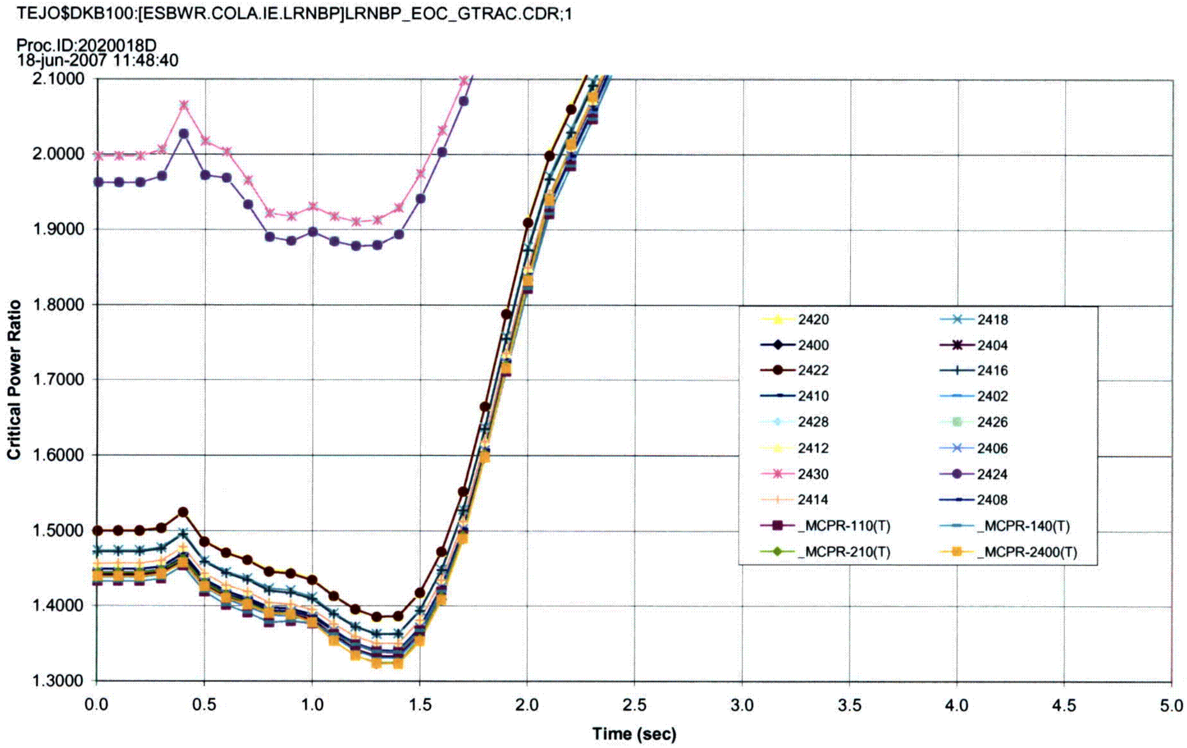


Figure 2.4-5g. Generator Load Rejection With Total Turbine Bypass Failure

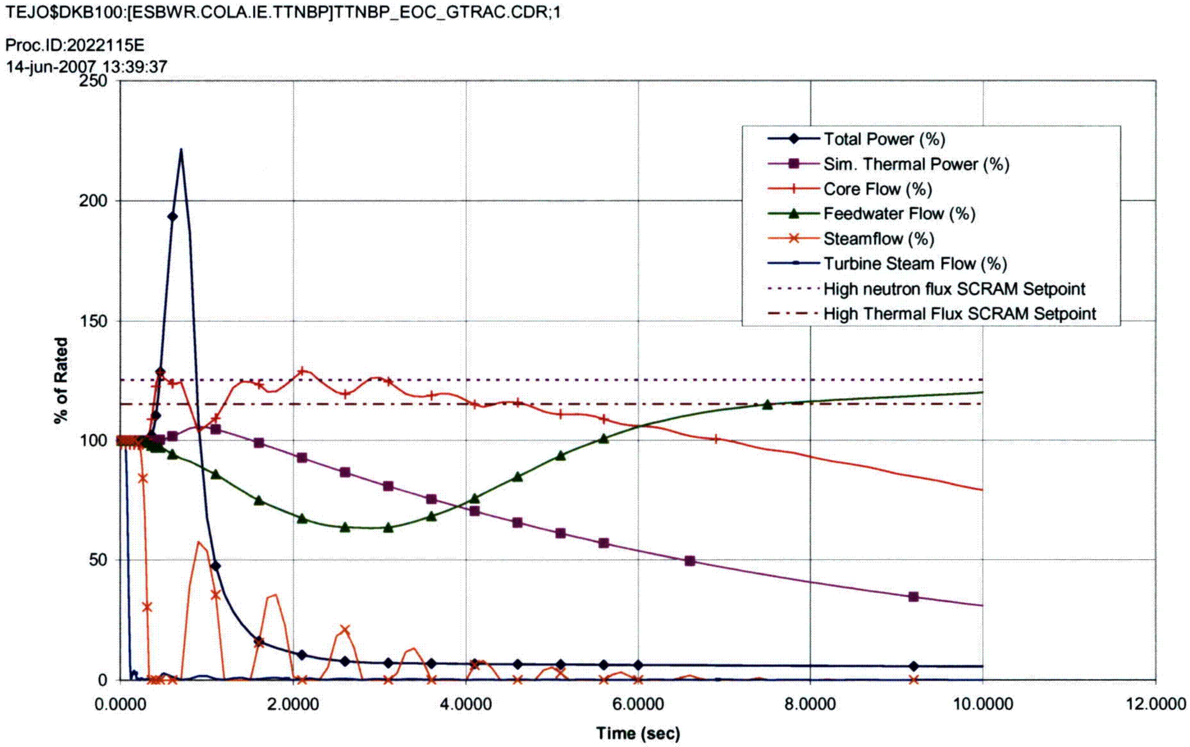


Figure 2.4-6a. Turbine Trip With Total Turbine Bypass Failure

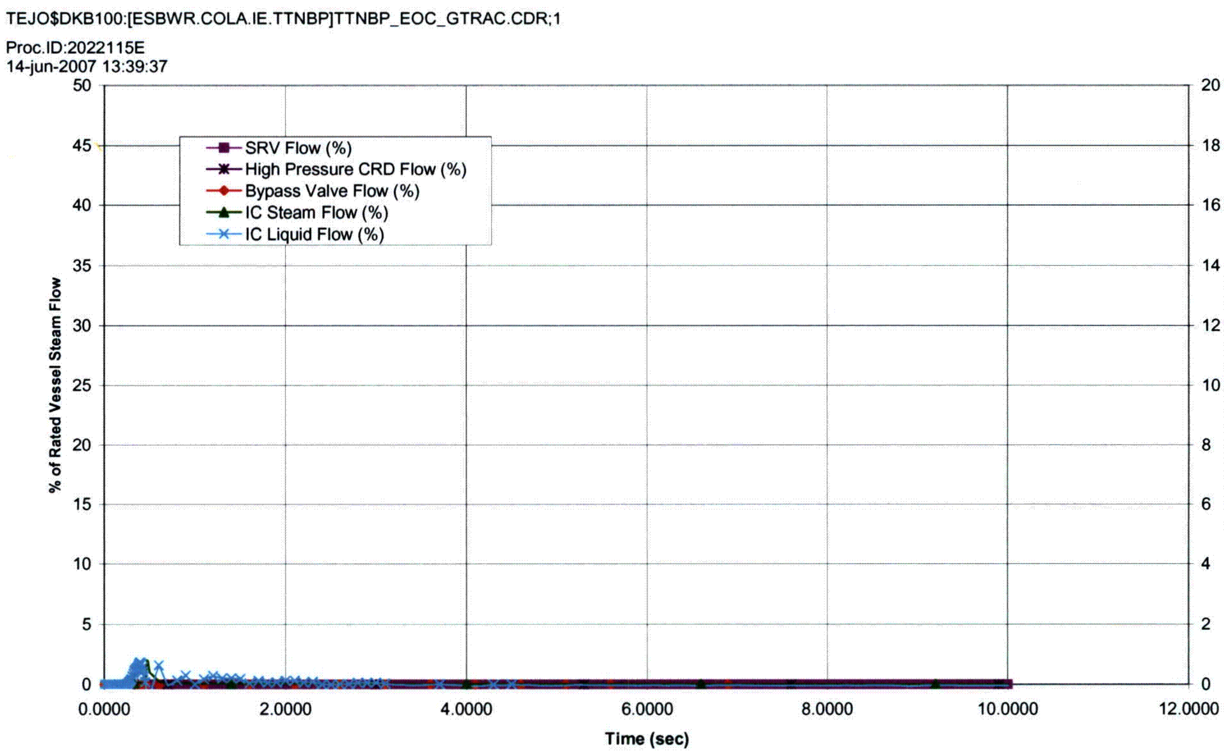


Figure 2.4-6b. Turbine Trip With Total Turbine Bypass Failure

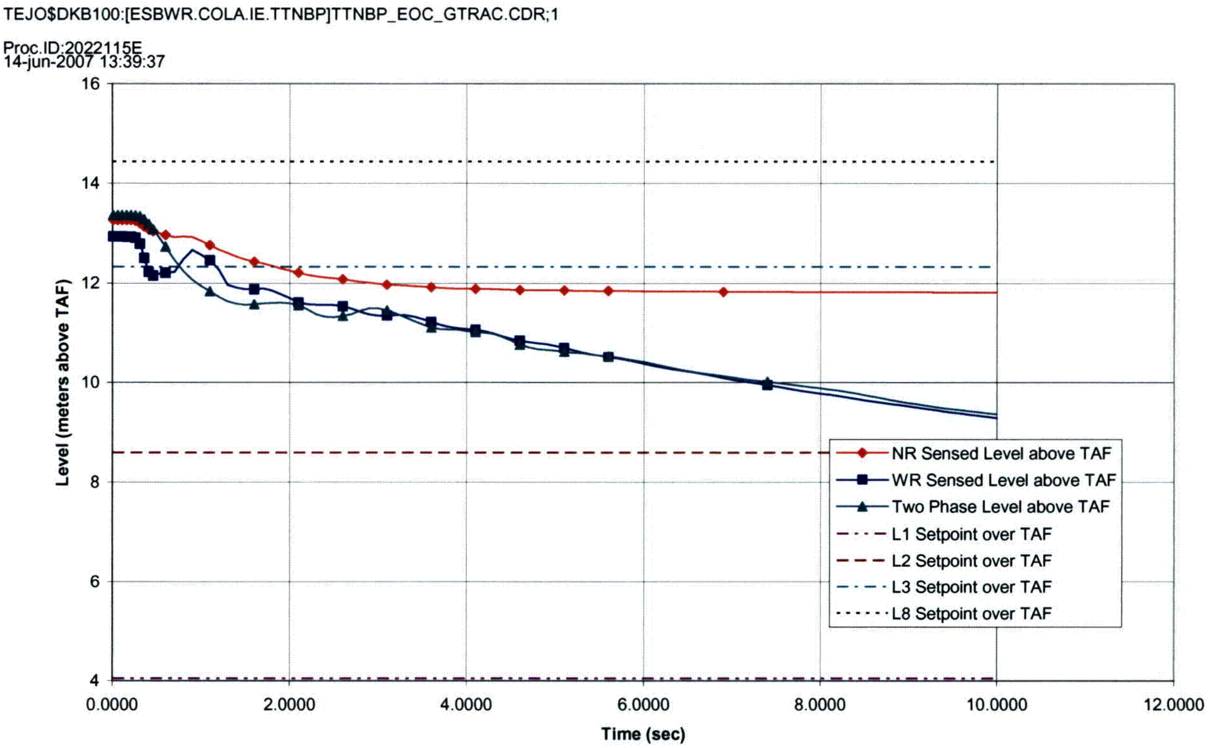


Figure 2.4-6c. Turbine Trip With Total Turbine Bypass Failure

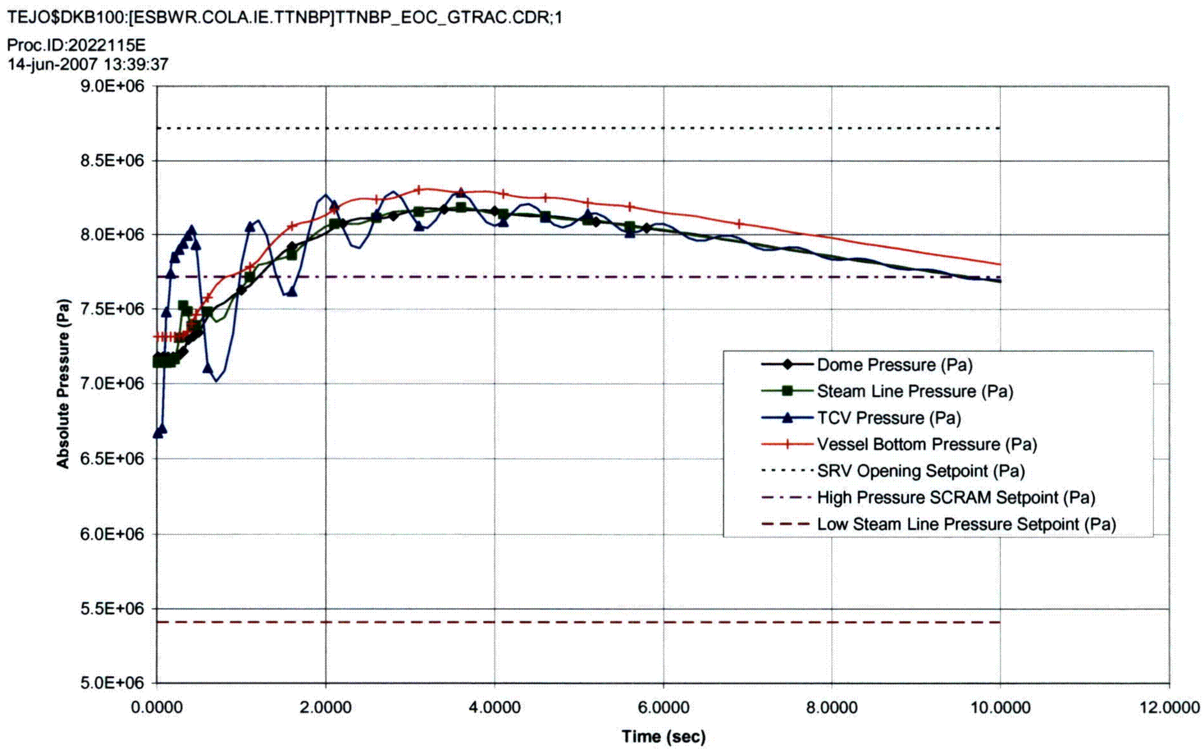


Figure 2.4-6d. Turbine Trip With Total Turbine Bypass Failure

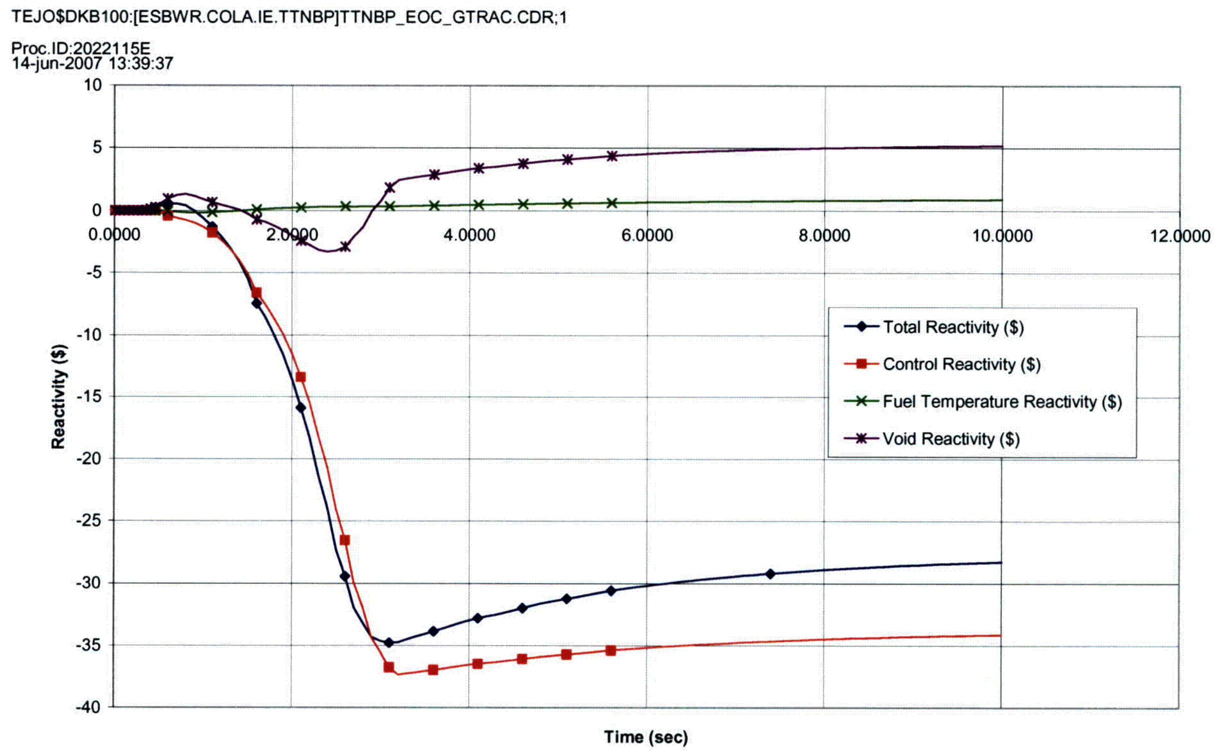


Figure 2.4-6e. Turbine Trip With Total Turbine Bypass Failure

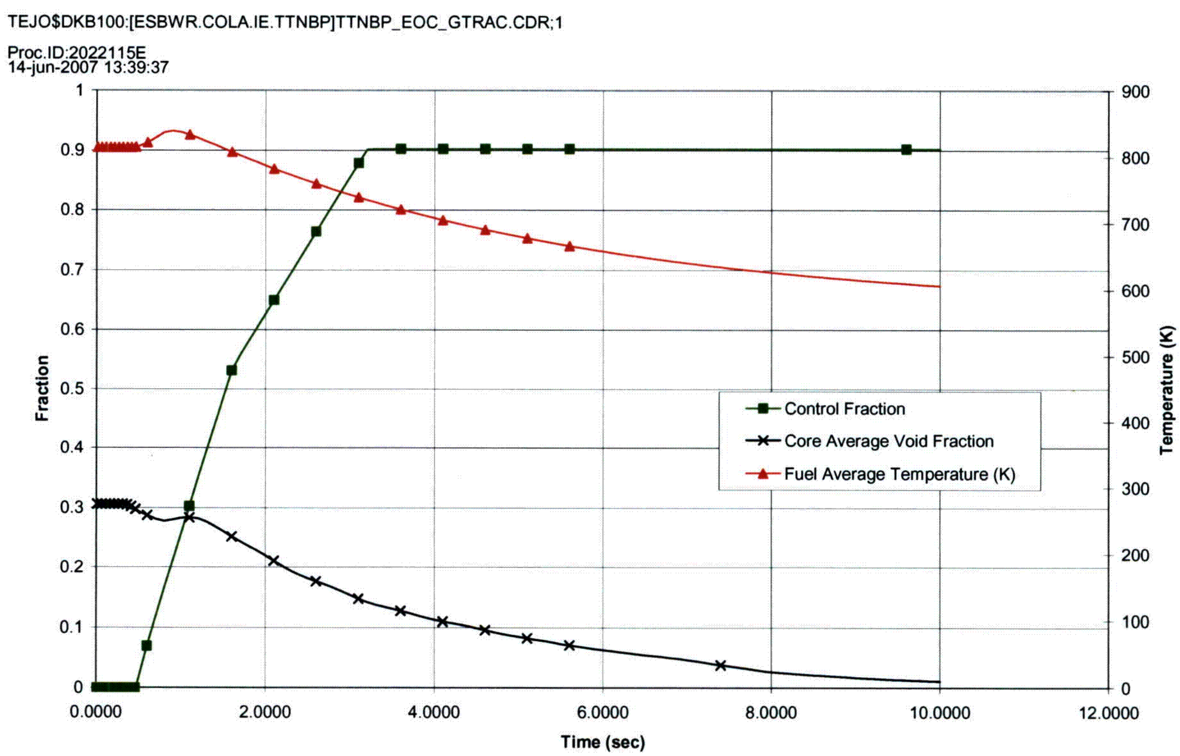


Figure 2.4-6f. Turbine Trip With Total Turbine Bypass Failure

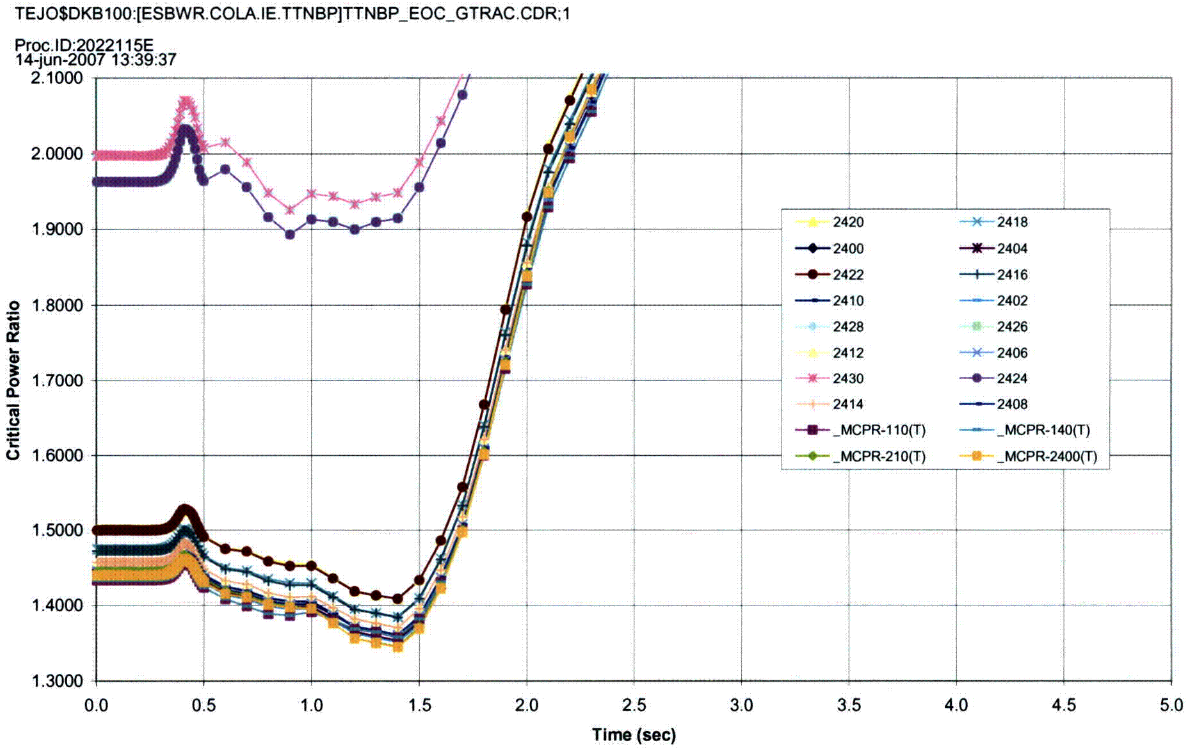


Figure 2.4-6g. Turbine Trip With Total Turbine Bypass Failure

Figure 2.4-7. Transient Changes for Control Rod Withdrawal Error During Startup (Representative BWR Analysis)

Data in the DCD applies to the initial core.

Figure 2.4-7a. Causes of Control Rod Withdrawal Error

Data in the DCD applies to the initial core.

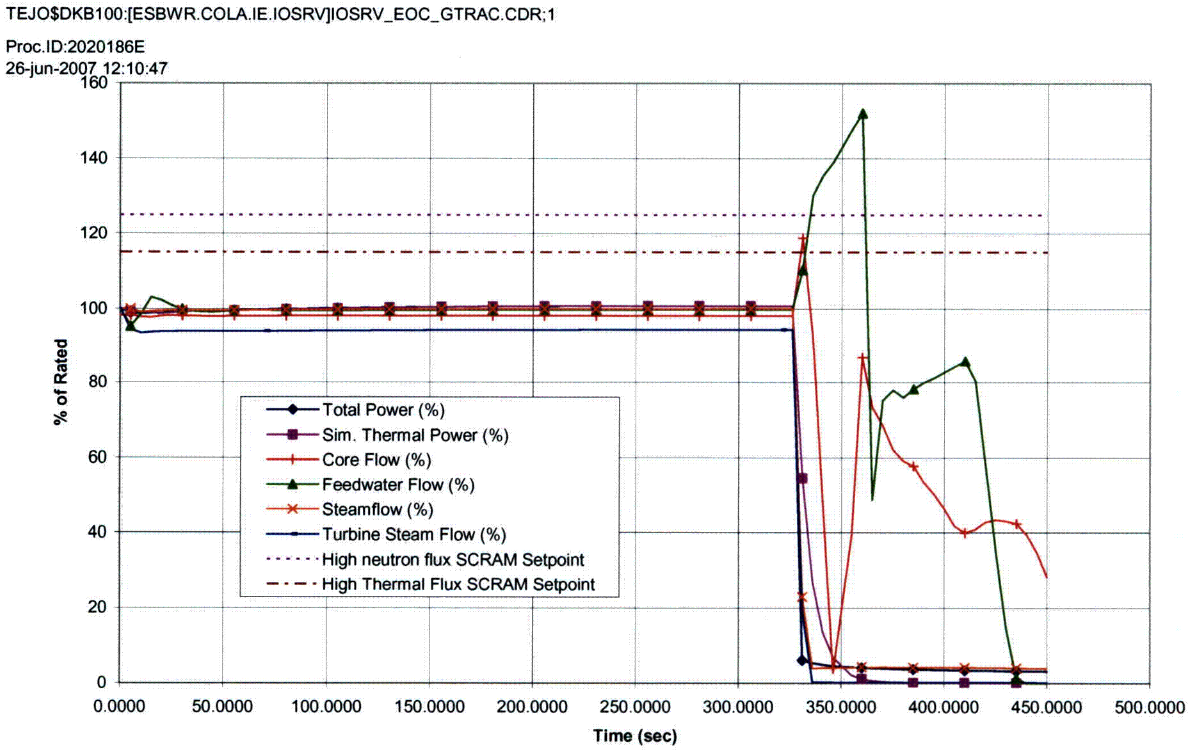


Figure 2.4-8a. Inadvertent SRV opening

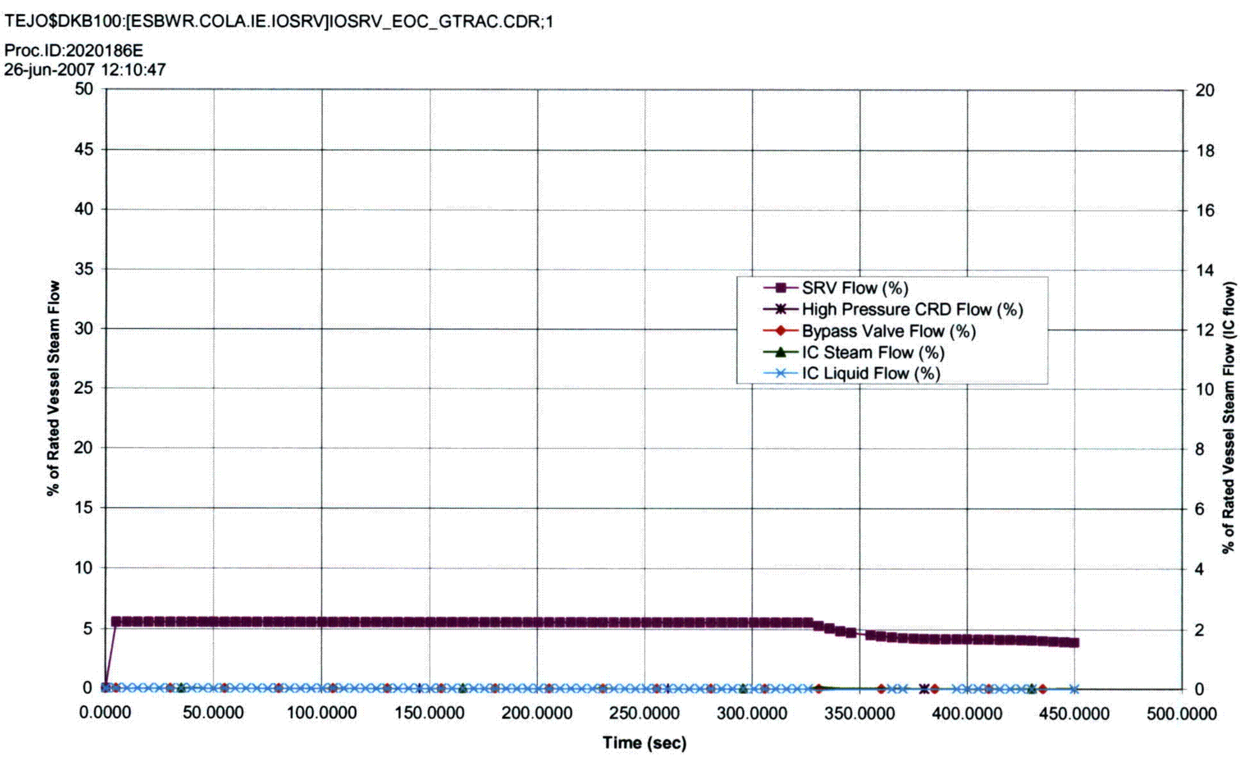


Figure 2.4-8b. Inadvertent SRV opening

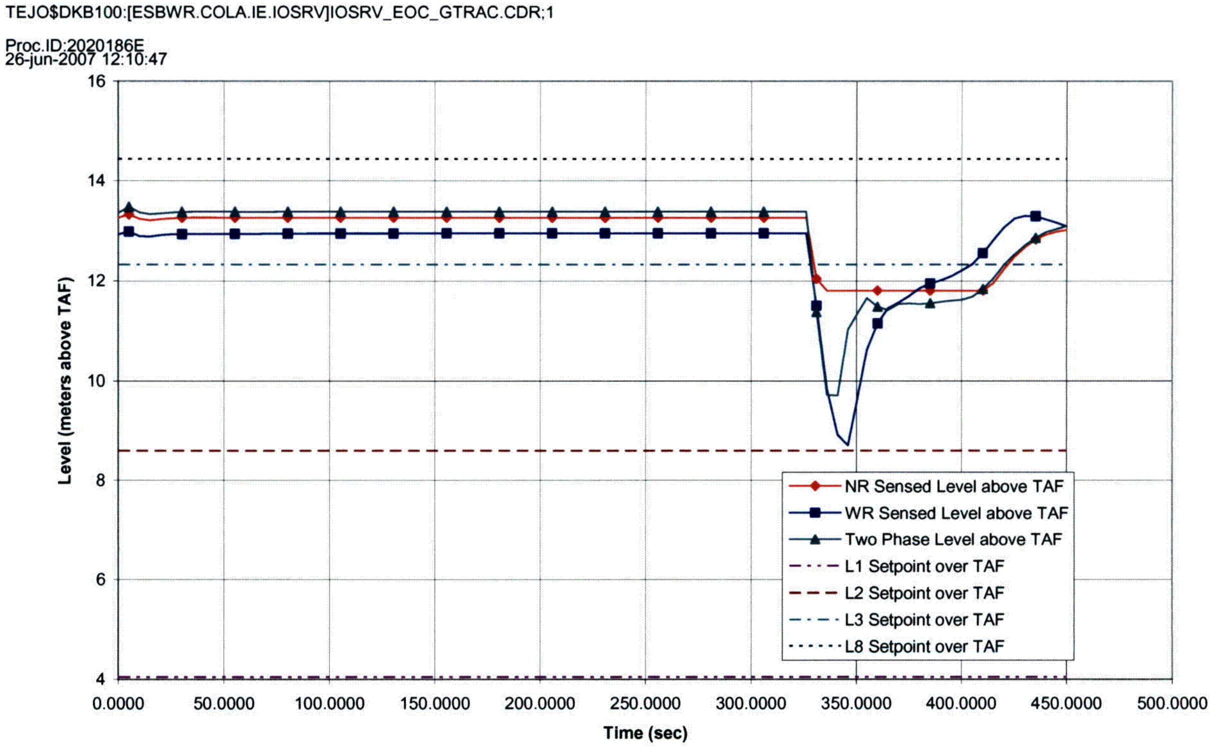


Figure 2.4-8c. Inadvertent SRV opening

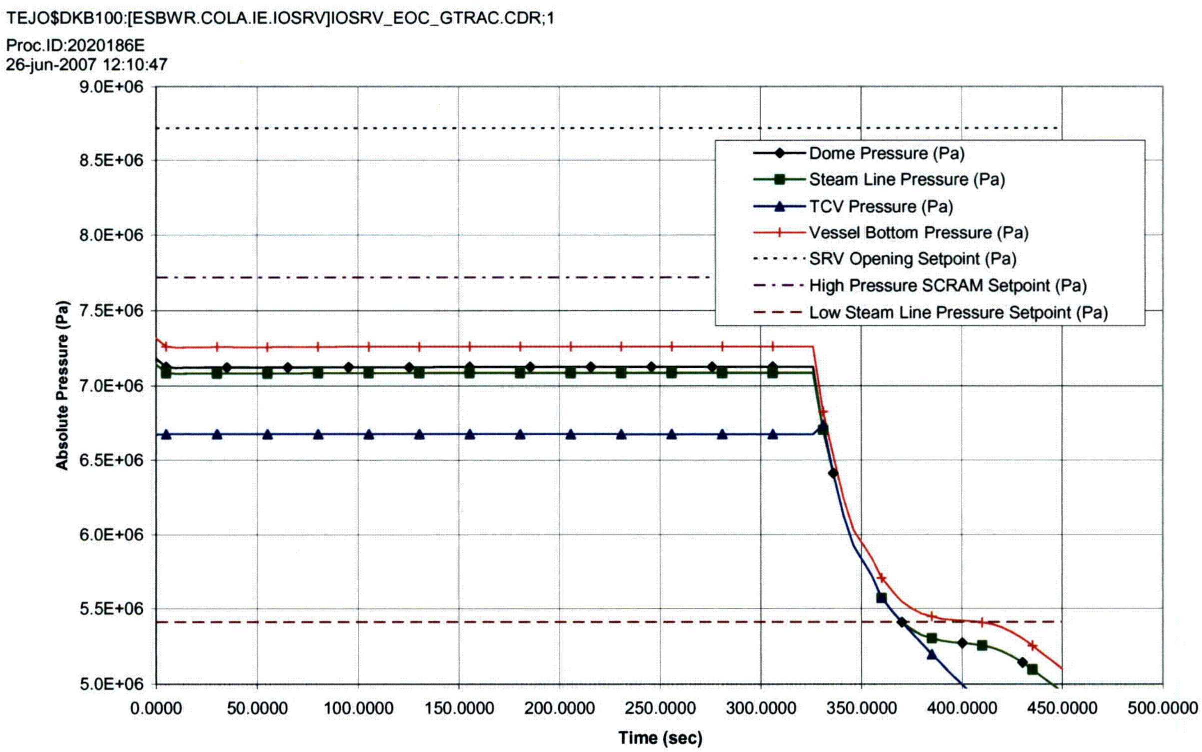


Figure 2.4-8d. Inadvertent SRV opening

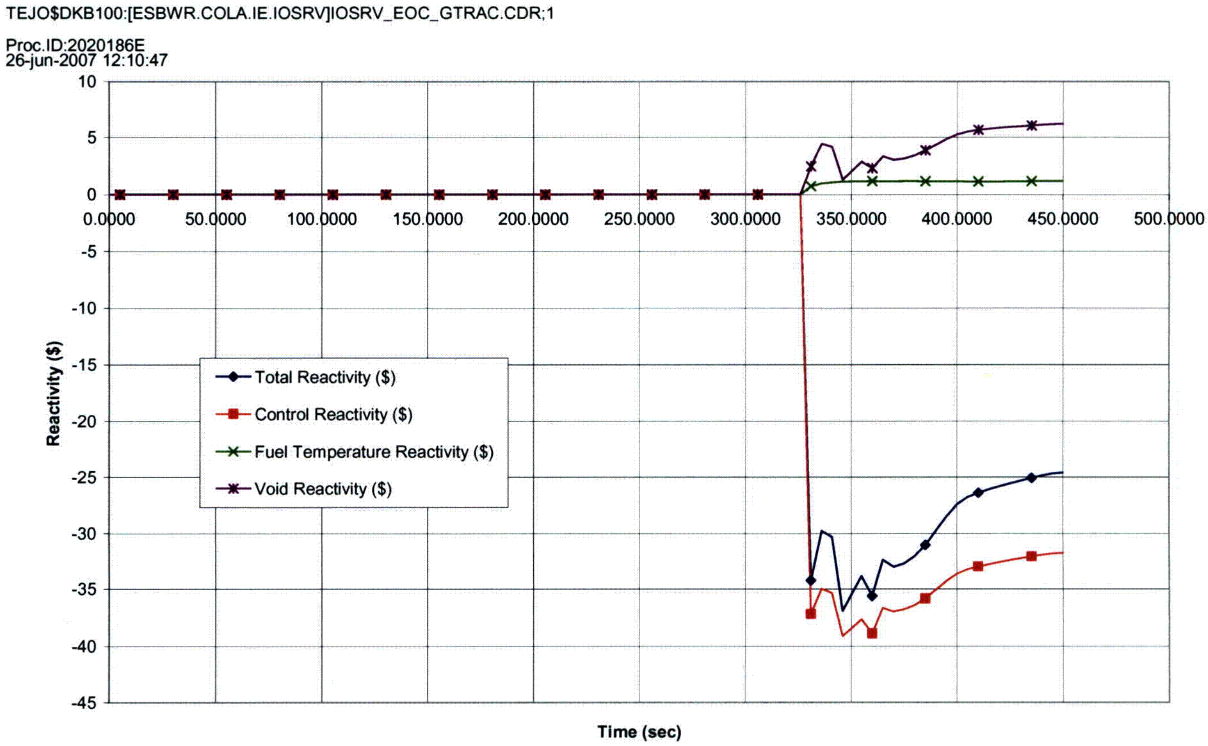


Figure 2.4-8e. Inadvertent SRV opening

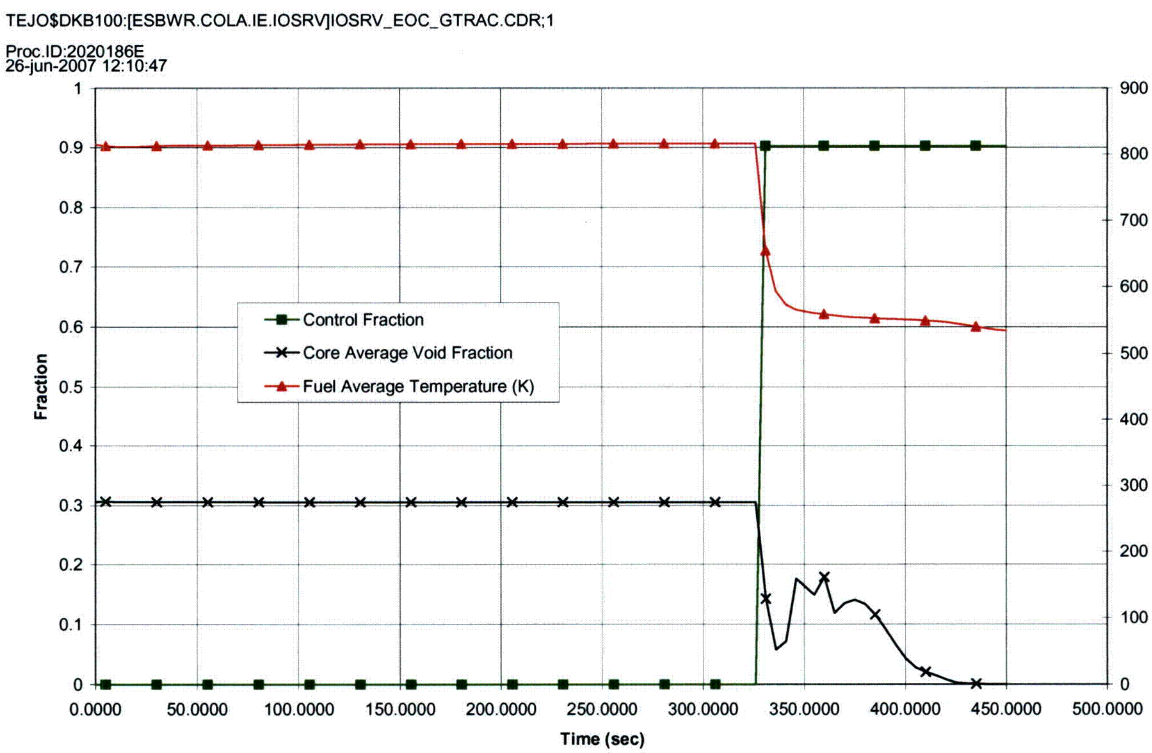


Figure 2.4-8f. Inadvertent SRV opening

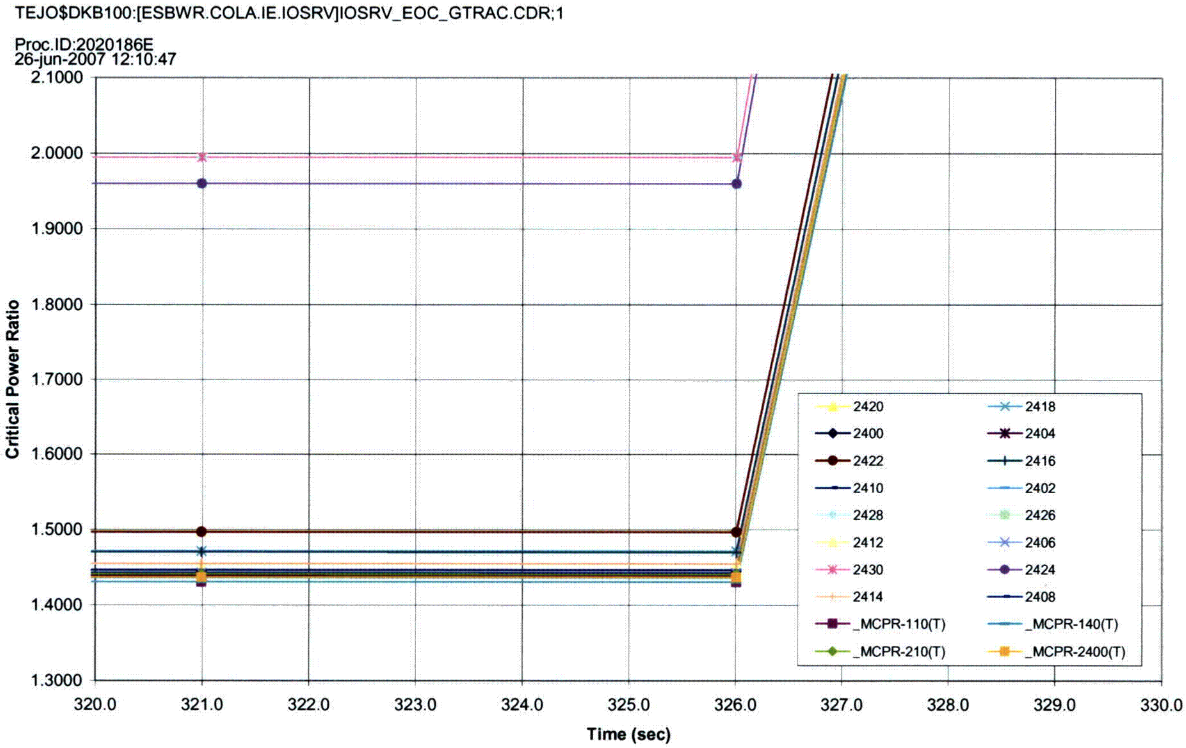


Figure 2.4-8g. Inadvertent SRV opening

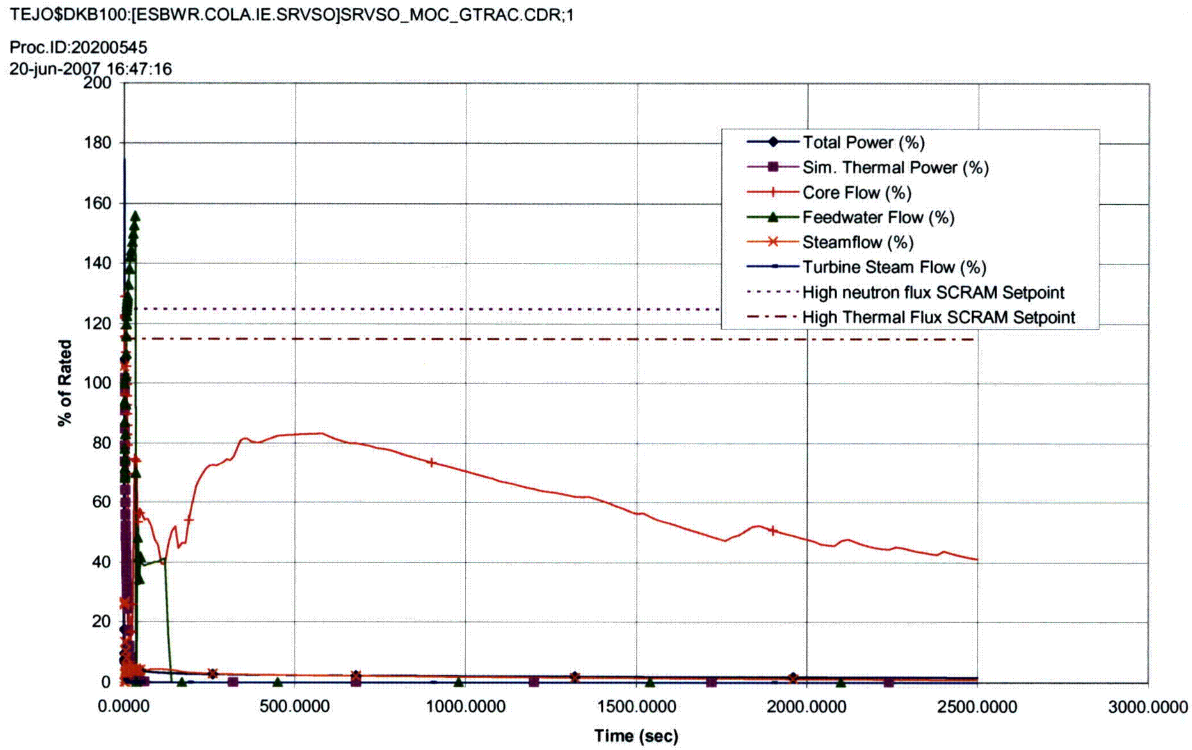


Figure 2.4-9a. Stuck Open Safety Relief Valve

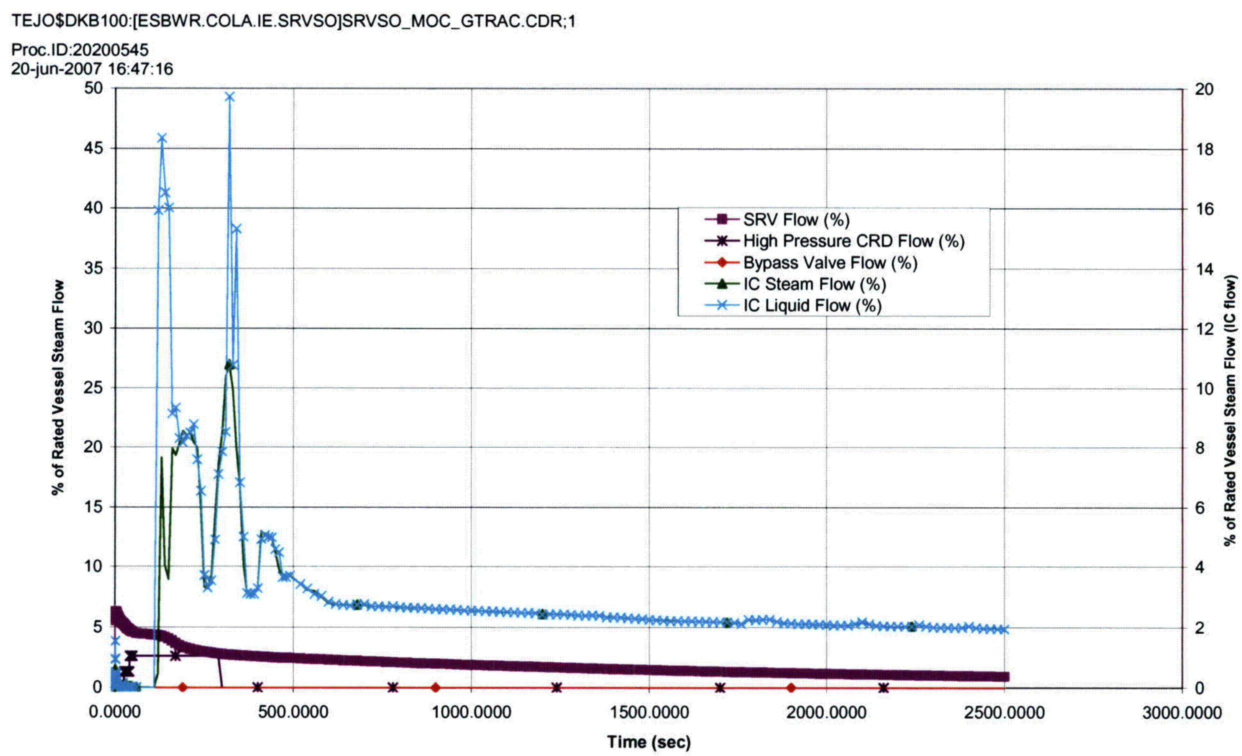


Figure 2.4-9b. Stuck Open Safety Relief Valve

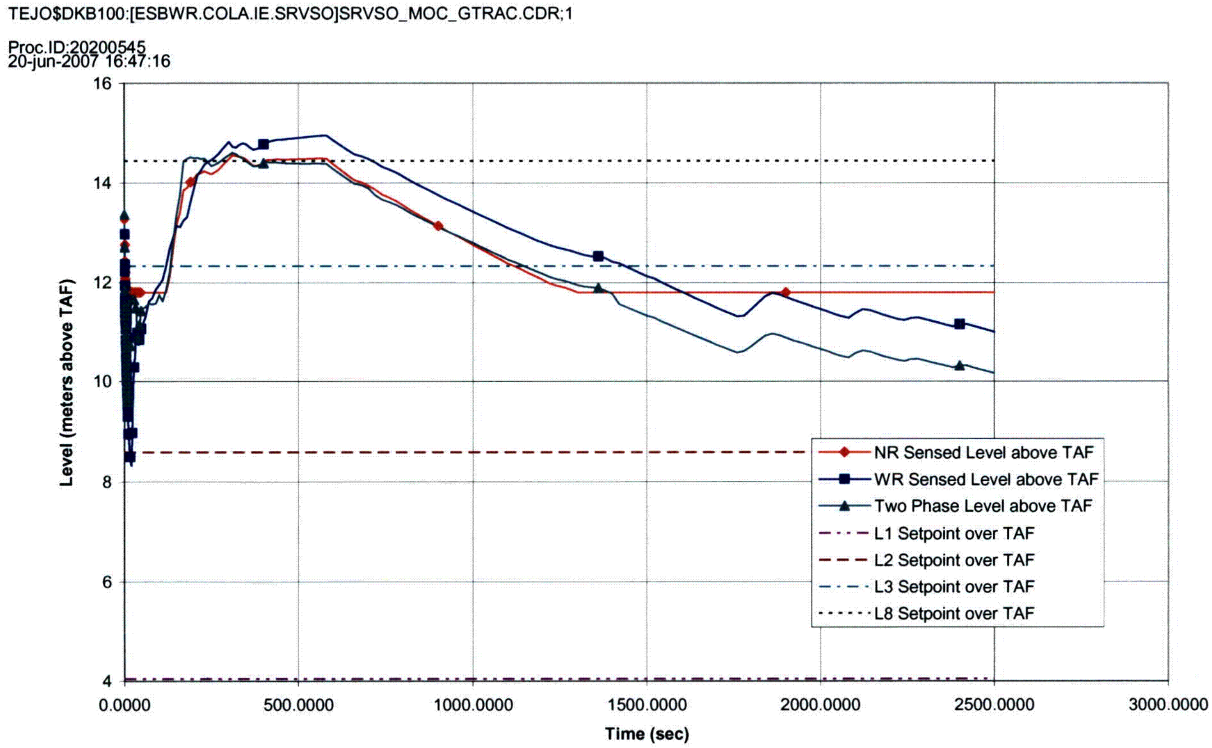


Figure 2.4-9c. Stuck Open Safety Relief Valve

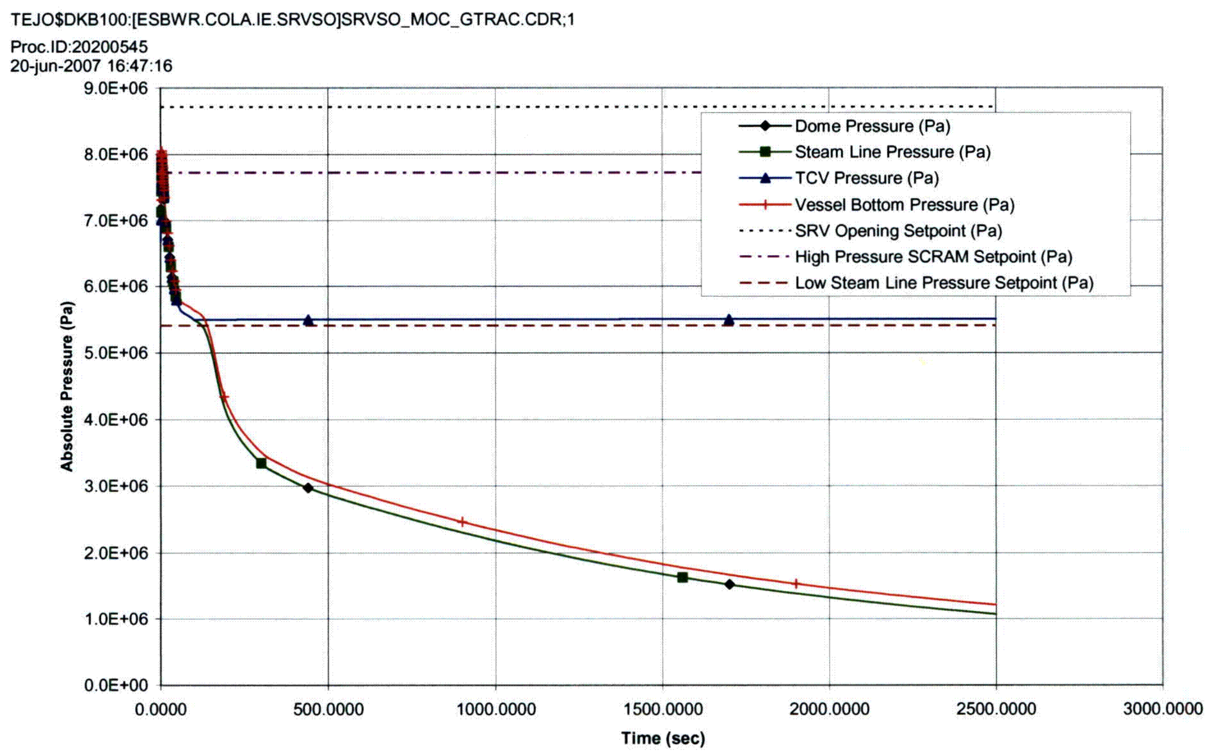


Figure 2.4-9d. Stuck Open Safety Relief Valve

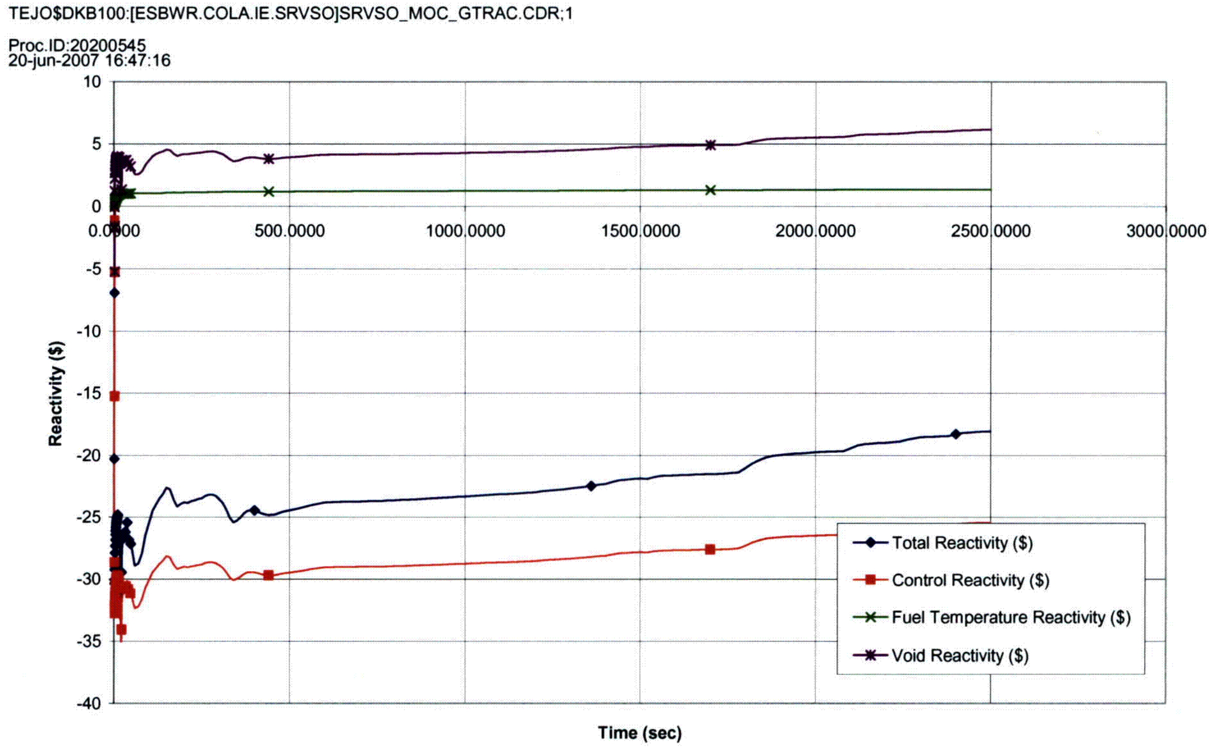


Figure 2.4-9e. Stuck Open Safety Relief Valve

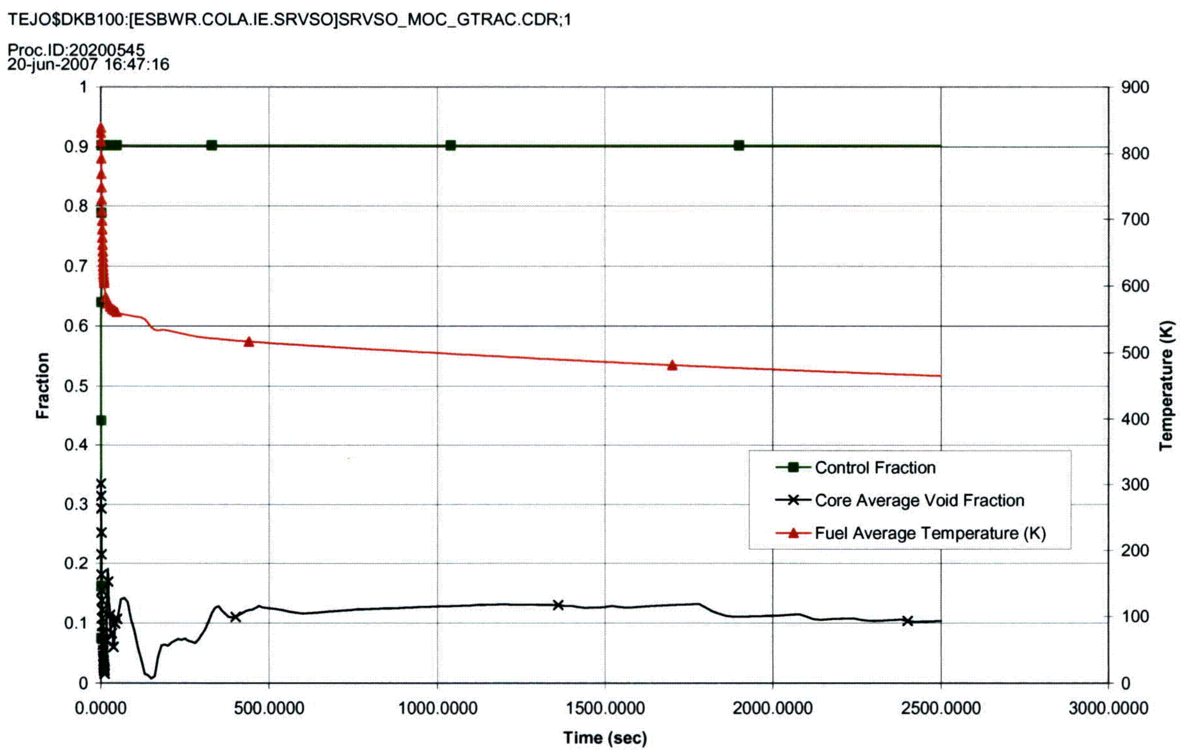


Figure 2.4-9f. Stuck Open Safety Relief Valve

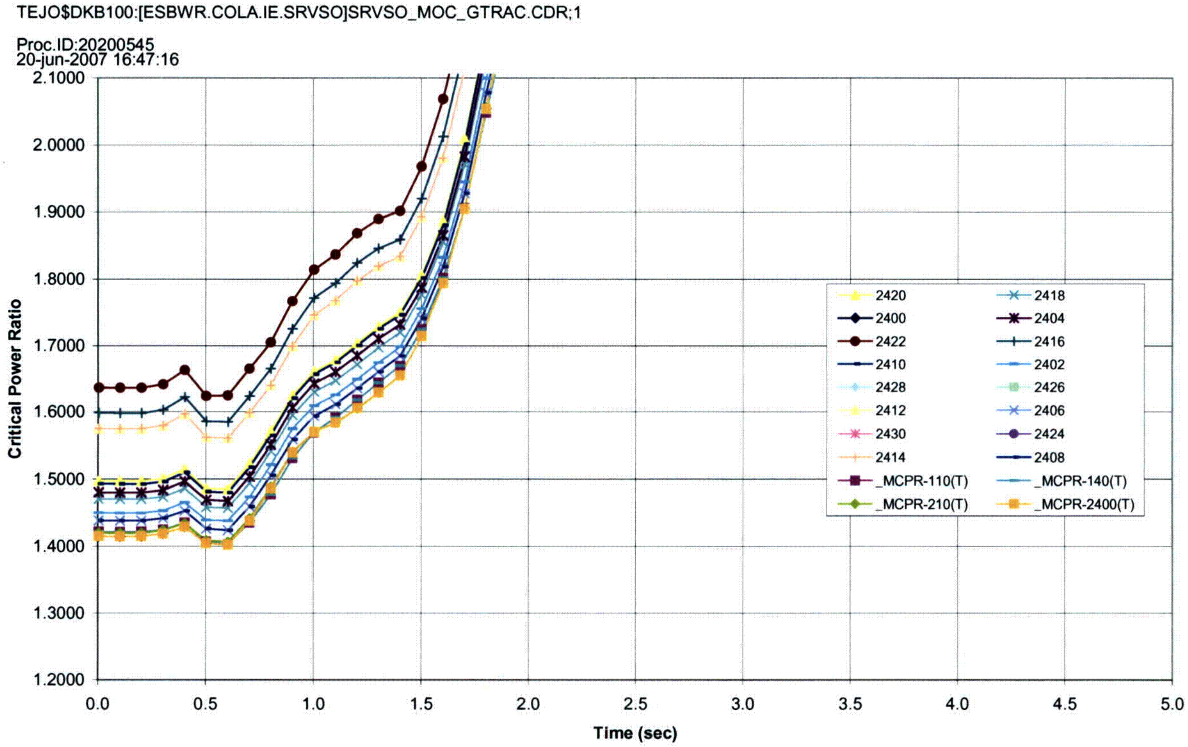


Figure 2.4-9g. Stuck Open Safety Relief Valve

2.5 ANALYSIS OF SPECIAL EVENTS

2.5.1 Overpressure Protection

This Subsection is equivalent to DCD Tier 2 Subsection 5.2.2 (Reference 2.5-4).

Discussion in DCD applies to the initial core.

2.5.1.1.1 Design Basis

Discussion in DCD applies to the initial core.

2.5.1.1.2 System Description

Discussion in DCD applies to the initial core.

2.5.1.1.3 Safety Evaluation

2.5.1.1.3.1 Method of Analysis

The method of analysis is approved by the United States Nuclear Regulatory Commission (NRC), or is developed using criteria approved by the NRC.

It is recognized that the protection of vessels in a nuclear power plant is dependent upon many protective systems to relieve or terminate pressure transients. Installation of pressure-relieving devices may not independently provide complete protection. The safety valve sizing evaluation gives credit for operation of the scram protective system which may be tripped by either one of three sources: (1) a direct valve position signal; (2) a flux signal; or (3) a high vessel pressure signal. The direct scram trip signal is derived from position switches mounted on the MSIVs. The position switches are actuated when the valves are closing and following 15% travel of full stroke. The flux signal is derived from the Neutron Monitoring System and is actuated at 125% of rated nuclear boiler power. The pressure signal is derived from pressure transmitters piped to the vessel steam space.

Full account is taken of the pressure drop on both the inlet and discharge sides of the valves. All combination SRVs discharge into the suppression pool through a discharge pipe from each valve which is designed to achieve sonic flow conditions through the valve, thus providing flow independence to discharge piping losses.

2.5.1.1.3.2 System Design

A parametric study was conducted to determine the required steam flow capacity of the SRVs based on the following assumptions

Operating Conditions

- Operating power = 4590 MWt (102% of nuclear boiler rated power);
- Absolute vessel dome pressure ≤ 7.27 MPa (1055 psia); and
- Steam flow = 2484 kg/s (19.71 Mlbm/hr).

These conditions are the most severe because maximum stored energy exists. At lower power conditions, the transient would be less severe.

Pressurization Events

The overpressure protection system is capable of accommodating the most severe pressurization event. The ESBWR pressurization is mild relative to previous other BWR designs because of the large steam volume in the chimney and vessel head, which mitigates the pressurization. The scram and initial pressurization drops the water level below the feedwater sparger; when the feedwater system performs as expected, the spray of subcooled water condenses steam in the vessel steam space and immediately terminates the pressurization. For purposes of overpressure protection analyses, the feedwater system is assumed to trip at the initiation of the event. The analyses of increase-in-reactor-pressure events are evaluated Subsection 2.3.2, where the performance of the ICS is credited to prevent a lift of the SRVs or SVs. In order to evaluate the overpressure protection capability of the SRVs, no credit is taken in this evaluation for the ICS.

No credit is taken for the first scram signal that would occur (e.g., valve position for MSIV isolation). This is in accordance with NUREG-0800, Subsection 5.2.2, which requires that the reactor scram be initiated by the second safety-related signal from the Reactor Protection System (neutron flux for MSIV isolation, turbine trip and load rejection).

The evaluation of event behavior demonstrates that MSIV closure, with scram occurring on high flux, (i.e., MSIV Closure With Flux Scram special event, MSIVF) is the most severe pressurization event.

Evaluation Method

The evaluation method is described in the DCD.

2.5.1.1.3.3 ***Evaluation of Results***

Total SRV Capacity

The adequacy of one SRV's capacity is demonstrated by analyzing the pressure rise from a MSIVF special event. Results of this analysis are given in Figure 2.5.1-4a through Figure 2.5.1-4f. The peak vessel bottom pressure calculated is below the acceptance limit of 9.481 MPa gauge (1375 psig). Figure 2.5.1-4a through Figure 2.5.1-4f show the MSIVF special event. The pressurization is not dynamic and does not significantly overshoot the relief valve setpoint. Vessel pressurization ceases to increase following relief valve opening when the steam discharge capacity exceeds the stored energy of the vessel plus rate of decay heat addition Figure 2.5.1-4d shows that peak vessel pressure is only a function of the valve setpoint. This is because the higher steam volume-to-power ratio of the ESBWR causes the pressure rate prior to scram to be much lower than operating BWRs. After a scram, the pressure rates due to core decay energy release are correspondingly lower.

Statistical Evaluation of MSIV closure event with flux scram

The statistical analysis is performed based on the equilibrium core, to calculate the upper bound for the Maximum Vessel Pressure during MSIVF, perturbing the physical correlations and operating conditions. These results are applicable to the initial core because the same fuel design is used. Considering the calculated Maximum Vessel Pressure uncertainty in the initial core analysis results in a maximum vessel pressure below 9.481 MPa (1375 psia) with a 95/95 confidence with sufficient margin to ensure that one SRV is sufficient produce acceptable results.

2.5.2 Shutdown Without Control Rods (Standy Liquid Control System Capability)

Analysis in the DCD applies to the initial core.

2.5.3 Shutdown from Outside Main Control Room

Analysis in the DCD applies to the initial core.

2.5.4 Anticipated Transients Without Scram

2.5.4.1 Requirements

The requirements are documented in the DCD.

2.5.4.2 Plant Capabilities

The plant capabilities are documented in the DCD.

2.5.4.3 Performance Evaluation

2.5.4.3.1 Introduction

Two limiting ATWS events are analyzed to evaluate the initial core design for ESBWR.

The procedure and assumptions used in this analysis are documented in Reference 2.5-2.

All transient analyses, unless otherwise specified, were performed with the TRACG code.

2.5.4.3.2 Performance Requirements

The performance requirements are documented in the DCD.

These performance requirements are summarized in Table 2.5.4-1.

2.5.4.3.3 Analysis Conditions

The probability of the occurrence of an ATWS is low. Thus, historically nominal parameters and initial conditions have been used in these analyses as specified in Reference 2.5-1.

As the processes for definition of allowable operational flexibility and margin improvement options expanded the analysis process transitioned to a basis that required use of bounding initial conditions. This was done because the frequency of operation within the allowable optional configurations could not be defined. In other words, "nominal" could not be defined. Some initial conditions, the most important being reactor power, are still analyzed without consideration of instrument uncertainties. Those that are applied conservatively include core exposure, core axial power shape, and Safety Relief Valve operability. All events analyzed assume reduced IC heat removal capacity to add a further measure of conservatism. The peak containment pressure presented is estimated in a conservative manner assuming that all the non-condensable gas from the drywell is in the wetwell airspace at the time of the peak pool temperature.

Selected inputs that affect the critical safety parameters are set to bounding values. The most important parameters for peak vessel pressure are Safety Relief Valve capacities and setpoints. These inputs are set to analytical limits. The most important parameters for clad and suppression pool temperature are initial Critical Power Ratio and boron flow rate respectively. These inputs are set to analytical limits.

Tables 2.5.4-2 and 2.5.4-3 list the initial conditions and equipment performance characteristics, which are used in the analysis.

2.5.4.3.4 *ATWS Logic and Setpoints*

The ATWS logic and setpoints are documented in the DCD.

2.5.4.3.5 *Selection of Events*

Discussion in DCD applies to initial core. However, only the two limiting events from the equilibrium core analysis are performed to demonstrate acceptable results.

2.5.4.4 Transient Responses

Main Steam Isolation Valve Closure

A bounding case with SLC system injection is analyzed for the limiting transient-MSIV closure. This bounding case is analyzed to show the in-depth ATWS mitigation capability of the ESBWR.

The MSIV closure with scram, ARI and FMCRD run-in failure, bounding case shows the ATWS performance with input parameters set per Reference 2.5-2 to produce a conservatively high reactor pressure, peak clad temperature, and a conservatively high suppression pool temperature. This case is intended to show that the peak RPV pressure, peak clad temperature, peak suppression pool temperature and the wetwell pressures are below the acceptance criteria. In this case, both ARI and FMCRD run-in are assumed to fail. Automatic boron injection with a total delay of 191-seconds (180 second timer + 10 second boron transportation delay in the SLC system + 1 second sensor and logic delay in the DCIS system) is relied upon to mitigate the transient event.

The bounding case is composed of five major elements that are intended to conservatively bound the key ATWS safety parameters for ESBWR. First, the reactor power used in the bounding analysis is 102% of the normal operating value. Second, the feedwater enthalpy is conservatively chosen to be 105% of the nominal value. Third, the SRV capacity input, shown in Table 2.5.4-3, chosen for the analysis is set to be conservatively bounding for the vessel bottom pressure response. Fourth, the analysis value of Feedwater Runback (FWRB) coastdown time of 15 seconds with an additional delay in the analysis of 10 seconds for the FWRB activation is chosen to conservatively bound the peak suppression pool temperature. Fifth, the initial Minimum Critical Power Ratio (MCPR) of the hot bundle is set to a value of 1.18 to conservatively bound the Peak Cladding Temperature (PCT), and is conservatively lower than the nominal value of 1.3.

If the ARI and FMCRD run-in fail at the same time, which has extremely low probability of occurrence, the peak reactor pressure would still be controlled by the SRVs. However, the nuclear shutdown then relies on the automatic SLC system injection. The boron would reach the core in about 11 seconds after the initiation. The operation of accumulator driven SLC system produces the initial volumetric flow rate of sodium pentaborate shown in Table 2.5.4-2. The nuclear shutdown would begin when boron reaches the core.

For the bounding MSIV closure case, a short time after the MSIVs have closed completely, the ATWS high-pressure setpoint is reached, which triggers the initiation of the feedwater runback. In the case that control rods fail to insert, the reactor is brought to hot shutdown by automatic

SLC boron injection. Operator actions during this event include reestablishing high-pressure makeup to control the water level at 1.5 m (5 ft) above the top of active fuel (TAF). If the Heat Capacity Temperature Limit (HCTL) is reached, the operator depressurizes the reactor via the SRVs to maintain margin to suppression pool limits.

The reactor system responses are presented in Figures 2.5.4-1a through 2.5.4-1d for the MSIV closure bounding case. The transient behavior of the SLCS bounding case is listed in Table 2.5.4-4a. A sequence of the main events that occur during these transients is presented in Table 2.5.4-4b.

Loss of Condenser Vacuum

This transient starts with a turbine trip because of the low condenser vacuum; therefore, the initial part of the transient is the same as the turbine trip event. However, the MSIVs and turbine bypass valves also close after the condenser vacuum has further dropped to their closure setpoints. Hence, this event is similar to the MSIV closure event for all the key parameters. Similar to the bounding case for the MSIV closure with SLC system described earlier, a bounding case is also analyzed for the Loss of Condenser Vacuum event with input parameters set per Reference 2.5-2. The bounding Loss of Condenser Vacuum case is composed of five major elements that are intended to conservatively bound the key ATWS safety parameters for ESBWR. First, the reactor power used in the bounding analysis is 102% of the normal operating value. Second, the feedwater enthalpy is conservatively chosen to be 105% of the nominal value. Third, the SRV capacity input, shown in Table 2.5.4-3, chosen for the analysis is set to be conservatively bounding for the vessel bottom pressure response. Fourth, the analysis value of Feedwater Runback (FWRB) coastdown time of 15 seconds with an additional delay in the analysis of 10 seconds for the FWRB activation is chosen to conservatively bound the peak suppression pool temperature. Fifth, the initial Minimum Critical Power Ratio (MCPR) of the hot bundle is set to a value of 1.18 to conservatively bound the Peak Cladding Temperature (PCT), which is lower than the nominal value of 1.3.

Table 2.5.4-5a shows the summary of peak values of key parameters for the bounding Loss of Condenser Vacuum case and Table 2.5.4-5b presents a sequence of main events that occur during this transient. Transient behavior is shown in Figures 2.5.4-2a through 2.5.4-2d for the bounding case. The high pressure ATWS setpoint is reached shortly after the closure of MSIVs. The high pressure initiates ARI, FMCRD run-in and the SLC timer. The SLC system trip is activated upon APRM not downscale and high-pressure signals and boron flow starts 3 minutes following the trip with a transportation delay time of 10 seconds, and sensor and logic delay of 1 second. As the poison reaches sufficient concentration in the core, the reactor achieves hot shutdown.

2.5.4.4.1 ***Conclusion***

Bounding system response analysis results for two Category 1 limiting events, the MSIV closure, and the Loss of Condenser Vacuum are provided for the initial core loading documented in Reference 2.5-3. Based upon the results of the analysis, the proposed ATWS design for the ESBWR is satisfactory in mitigating the consequences of an ATWS. All performance requirements specified in Subsection 2.5.4.3.2 are met.

It is also concluded from results of the above analysis that automatic boron injection could mitigate the most limiting ATWS event with margin. Therefore, an automatic SLCS injection as a backup for ATWS mitigation is acceptable.

2.5.5 Station Blackout

The performance evaluation for Station Blackout (SBO) shows conformance to the requirements of 10 CFR 50.63 and is presented in this subsection.

2.5.5.1 Acceptance Criteria

The acceptance criteria are documented in the DCD.

2.5.5.2 Analysis Assumptions

The analysis assumptions and inputs are summarized below.

- Reactor is operating initially at 102% of rated power/100% rated nominal core flow, nominal dome pressure and water level at L4. The reactor has been operating at 102% of rated power for at least 100 days.
- The nominal ANSI/ANS 5.1-1994 decay heat model is assumed but an initial core power of 102% of the nominal is assumed.
- SBO starts with loss of all alternating current (AC) power, which occurs at time zero. Auto bus transfer is assumed to fail.
- Loss of AC power trips reactor, feedwater, condensate and circulating water pumps , and initiates a turbine load rejection.
- The reactor scrams occurs at 2.0 seconds due to loss of power supply to feedwater pumps. When feedwater flow is lost, there is a scram signal with a delay time of 2.0 seconds.
- BPV open on load rejection signal, 6s later the closure because of the loss of condenser vacuum is not credited and the turbine bypass closes to control the reactor pressure when it begins to drop because of the scram of the reactor.
- Closure of all Main Steam Isolation Valves (MSIVs) is automatically initiated when the reactor water level reaches Level 2 after 30 s time delay. The closure, because of the loss of condenser vacuum, is not credited. The closure has no effect on the response because the BPV and TCV are almost closed when MSIV began to close, and the valves are fully closed at 5.0 seconds after signal.
- CRD pumps are not available due to loss of all AC power. The systems available for initial vessel inventory and pressure control, containment pressure/temperature control and suppression pool temperature control are:
  - Three Isolation Condensers (ICs)
  - The rest of the safety systems are not credited or they do not actuate during the calculated sequence of events.

- The passive IC system is automatically initiated upon the loss of feedwater pump power buses at 3s, to remove decay heat following scram and isolation and IC drain flow provides initial reactor coolant inventory makeup to the reactor pressure vessel.
- When the water level reaches Level 2 or 3 no automatic or manual action is credited.
- The vessel despressurizes, the vessel and other components inventory remains constant; however, the measured level evolves because reactor pressure and liquid temperature changes.
- Other assumptions, given in Tables 2.3-1, 2 and 3, are applied to the TRACG calculation.

2.5.5.3 Analysis Results

As shown in Figure 2.5.5-1 and Table 2.5.5-1a , during the first 20000s of depressurization , the level is maintained above the Level 1, also from vessel inventory analysis the level will remain above Level 1 during the first 72h of the transient: Therefore the requirement for reactor vessel coolant integrity is satisfied. As shown in Table 2.5.5-1b, considering a constant mass balance, and increased liquid density, the wide range measured level is above 13.5 m above vessel zero, which provides adequate margin to L1 ADS analytical limit (11.5 m above vessel zero). The collapsed water level remains well above TAF.

Subsequent to a SBO event, hot shutdown condition can be achieved and maintained by operation of IC systems. Therefore, the requirement for achieving and maintaining hot shutdown condition is met.

With operation of IC system, the containment and suppression pool pressures and temperatures are maintained within their design limits, since there is no release into the wetwell or the drywell. Therefore, the integrity for containment is maintained.

RPV leakage is expected to be minimal for three reasons: 1) There are no recirculation pumps in the design. 2) Isolation occurs on L2. 3) The pressure is reduced significantly by ICS. However, If leakage is significant and power has not been restored, the level could drop below the L1 setpoint. In this case, ADS, GDCS and PCCS are available to provide core cooling, inventory control and containment heat removal. Because significant depressurization is provided by ICS the impact of depressurization due to ADS initiation would not be as significant as initiation from rated pressure.

As demonstrated above, each acceptance criterion in Subsection 2.5.5.1 is met. Therefore ESBWR can successfully mitigate a SBO event to meet the requirements of 10 CFR 50.63.

This event bounds AOOs with respect to maintaining water level above the top of active fuel.

2.5.6 Safe Shutdown Fire

Analysis in the DCD applies to the initial core.

2.5.7 Waste Gas System Leak or Failure

Analysis in the DCD applies to the initial core.

2.5.8 References

2.5-1 General Electric Company, "Assessment of BWR Mitigation of ATWS," NEDE-24222, September 1979.

2.5-2 GE Energy Nuclear, "TRACG Application for ESBWR," NEDE-33083P Supplement 2, Class III, (Proprietary), January 2006.

2.5-3 Global Nuclear Fuel, "ESBWR Initial Core Nuclear Design Report", NEDC-33326-P, Class III (Proprietary), Revision 0, July 2007, NEDO-33326, Class I (Non-proprietary), Revision 0, July 2007.

2.5-4 GE Nuclear Energy, "ESBWR Design Control Document" 26A6642, Revision 4, September 2007.

Table 2.5.1-1

Sequence of events for MSIV Closure flux Scram

Time (s)	Event
1. MSI valves start to close	0.0
2. FW trip	0.02
3. MSIV position < 85%	0.78
4. High neutron flux setpoint reached	1.65
5. MSIV position < 1%	1.7
6. Reactor SCRAM on high flux	1.70
7. L3	2.81
8. MSIV closed	3.0
9. L2	9.06
10. HP CRD Initiation	Not credited
11. SRV opening	51.65

Table 2.5.1-2

Safety/-Relief Valve and Depressurization Valve Settings and/or Capacities

Valve Type SRV /DPV	**Number of Valves**[(1)]	**SRV Maximum Analytical Pressure Limit** MPa gauge (psig)	**ASME Rated Capacity at Setpoint with 3% Accumulation Equivalent to Analytical Pressure Limit** (kg/s each)
ADS SRV [(3)]	10	8.618 (1250)	138 minimum
Non-ADS SRV [(4)]	8	8.756 (1270)	140.2 minimum
DPV	8	NA	239 [(2)]

(1) The SRVs also perform the automatic depressurization function.

(2) Minimum capacity in ADS mode. The DPVs are not needed to mitigate the overpressure event.

(3) Nominal Trip Setpoint is 8.367 ± 0.251 MPa gauge (1213.6 ± 36.4 psig). Following testing, lift settings shall be within ± 1%.

(4) Nominal Trip Setpoint is 8.501 ± 0.255 MPa gauge (1233.0 ± 37 psig). Following testing, lift settings shall be within ± 1%.

Table 2.5.4-1

ATWS Performance Requirements

RPV Peak Pressure MPag (psig)	**Maximum Pool Temperature** °C (°F)	**Fuel Integrity**	**Peak Cladding Temperature** °C (°F)	**Local Oxidation of Cladding**	**Maximum Containment Pressure** kPa (psia)
10.34 (1500)	121 (250)	Coolable Geometry	Less than 1204.40 (2200)	Not to exceed 17% of total cladding thickness	414 (60)

Table 2.5.4-2

ATWS Initial Operating Conditions

Parameters	Bounding Value
Power, MWt/% NBR	4590/102
Vessel Diameter, m (ft)	7.1 (23.3)
Numbers of Fuel Bundles	1132
Bounding Initial Conditions Used in ATWS Analysis	
Parameters	**Bounding Value**
Dome Pressure, MPaG (psig)	6.98 (1013)
Natural Circulation Core Flow, Mkg/hr (Mlb/hr)*	37.03 (81.64)
Steam/Feed Flow, kg/s (Mlbm/hr)	2531 (20.09)
Feedwater Temperature, °C (°F)	224.9 (436.8)
Nuclear Characteristics Used in TRACG Simulations Condition Exposure	Reference 2.5-3 EOC
Suppression Pool Volume, m^3 (ft^3)	4354 (153,760)
3 Isolation Condensers volume, 3 Units, from steam box to discharge at vessel, m^3 (ft^3)	42.1 (1485.8)
Initial Suppression Pool Temperature, °C (°F)	43.3 (109.9)
SLCS accumulator driven initial flow, m^3/s (gpm)	0.03 (475)

*The required measurement accuracy is less than or equal to 7.5% of rated core flow for one standard deviation (1σ).

Table 2.5.4-3

ATWS Equipment Performance Characteristics

Parameters	Value
MSIV Closure Time, s	≥3.0
Delay before start of Electro-Hydraulic Rod Insertion, s	≤1
Electro-Hydraulic Control Rod Insertion Time, s	≤130
Maximum time for start of motion of ARI rods, s	15
Maximum time for all ARI rods to be fully inserted, s	25
SLC system transportation and DCIS logic delay time, s	≤11
Safety Relief Valve (SRV) System Capacity, % NBR Steam Flow/No. of Valves –Bounding Cases[1]	≥102/18
High Reactor Pressure Vessel (RPV) Dome pressure setpoint, MPaG (psig)	7.76 (1125)
SRV Setpoint Range, MPaG (psig)	8.618 to 8.756 (1250-1270)
SRV Opening Time, s	<0.5
Pressure Drop Below Setpoint for SRV Closure, % nameplate	≤96
Low Water Level (Level 2) Trip setpoint (from vessel bottom reference zero), cm (in)	1605.0 (631.9)
CRD (High Pressure Make-Up Function) Low Water Level Initiation Setpoint, cm (in)	1605.0 (631.9)
CRD (High Pressure Make-Up Function) Flow Rate, m^3/s (gal/min)	0.065 (1035)
ATWS Dome Pressure Sensor Time Constant, s	≤0.5
ATWS Logic Time Delay, s	≤1
Pool Cooling Capacity, kW/C	430.6
Low Water Level For Closure of MSIVs, cm (in)	1605.0 (631.9)
Low Steamline Pressure For Closure of MSIVs, MPaG (psig)	5.412 (785)
Temperature For Automatic Pool Cooling, °C (°F)	48.9 (120)

(1) The SRV capacity used in the analysis is 102% of the ASME rated capacity noted in DCD Tier 2, Table 5.2-2.

Table 2.5.4-4a

ATWS MSIV Closure Summary – SLC System Bounding Case (Initial Core, SP0)

Parameter	Value	Time (s)
Maximum Neutron Flux, %	203.7	3.3
Maximum Vessel Bottom Pressure, MPaG (psig)	9.38 (1360)	21
Maximum Bulk Suppression Pool Temperature, °C (°F)	74.2 (165.6)	368
Associated Containment Pressure, kPaG (psig)	208.8 (30.3)	368
Peak Cladding Temperature, °C (°F)	791.1(1455.9)	28

Table 2.5.4-4b

ATWS MSIV Closure Bounding Case Sequence of Events (Initial Core, SP0)

Time (s)	Event
0	MSIV Closure starts
0.2	IC initiates
3.6	ATWS trip set at high pressure
5.3	SRVs open
42.1	Level drops below L2 set point
52.0	HP_CRD flow starts
194.6	SLCS injection starts
715.1	High pressure design volume of borated solution injected into bypass

Table 2.5.4-5a

ATWS Loss of Condenser Vacuum Summary – SLC System Bounding Case

(Initial Core, SP0)

Parameter	Value	Time (s)
Maximum Neutron Flux, %	211.71	9.3
Maximum Vessel Bottom Pressure, MPaG (psig)	9.38 (1361)	27
Maximum Bulk Suppression Pool Temperature, °C (°F)	74.5 (166.1)	360
Associated Containment Pressure, kPaG (psig)	209.6 (30.4)	360
Peak Cladding Temperature, °C (°F)	790.2 (1454.3)	34

Table 2.5.4-5b

ATWS Loss of Condenser Vacuum Sequence of Events Bounding Case

(Initial Core, SP0)

Time (s)	Event
0	Loss of Condenser Vacuum
0	Turbine Trip initiated
6	MSIV closure trip set
7.5	IC initiates
9.6	ATWS trip set at high pressure
11.3	SRVs open
49.8	Level drops below L2 set point
60.2	HP_CRD flow starts
200.6	SLCS injection starts
725.6	High pressure design volume of borated solution injected into bypass

Table 2.5.5-1a

Sequence of Events for Station Blackout

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 with one second delay
5.0	Feedwater flow decay to 0
7.0	Vessel water level reaches Level 3
9.7	Vessel water level reaches Level 2
18.0	ICs begins to drop cold water inside the vessel
33.0	ICs drainage valve is fully open
39.8	MSIV valve begins to close
44.8	MSIV is totally closed
72 hours	The system reached the conditions described in Table 2.5.5-1b

Table 2.5.5-1b

Theoretical Vessel conditions at 72 hours after SBO

Parameter	Value
Dome pressure, PaG (psig)	0 (0)
Vessel Bottom Pressure, PaG (psig)	123000 (17.48)
Decay heat, Mw	19.5
Wide range measured level over TAF, m (ft)	6.40 (20.99)
Collapsed Level over TAF, m (ft)**	5.66 (18.57)
IC flow, kg/s (lb/hr)	8.64 (68,573)

Figure 2.5.1-1. through 2.5.1-3 Deleted

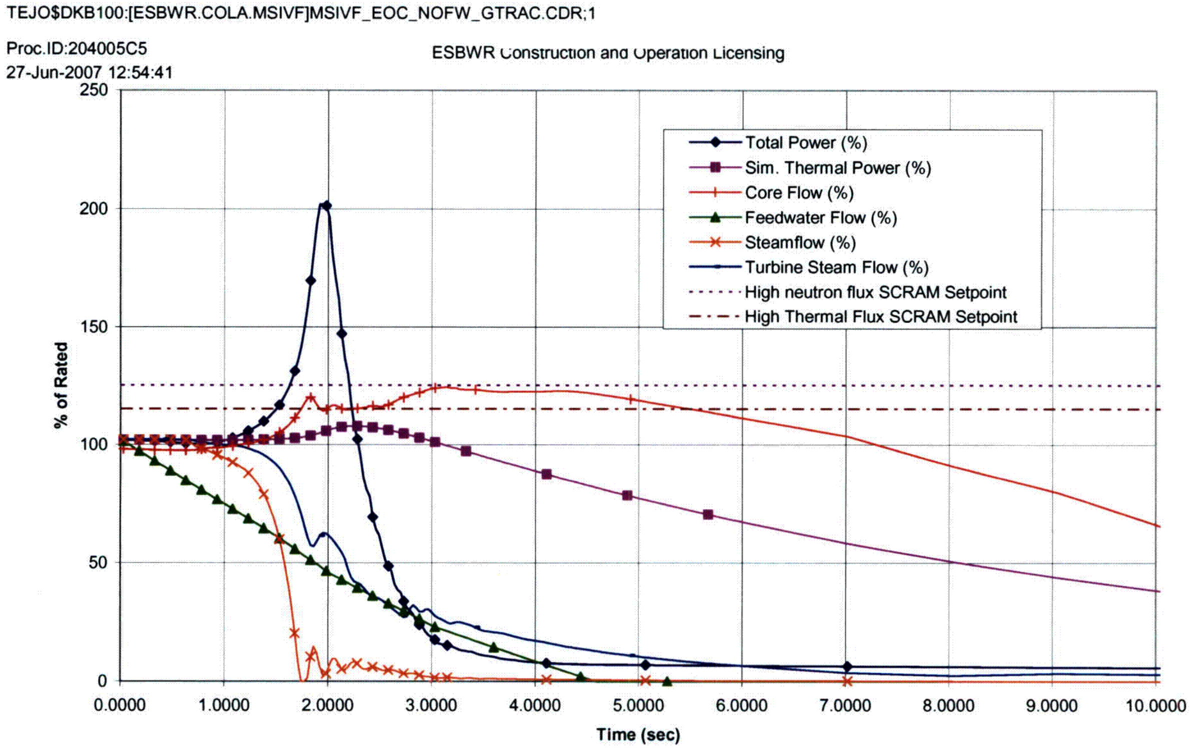


Figure 2.5.1-4a. MSIV Closure With Flux Scram

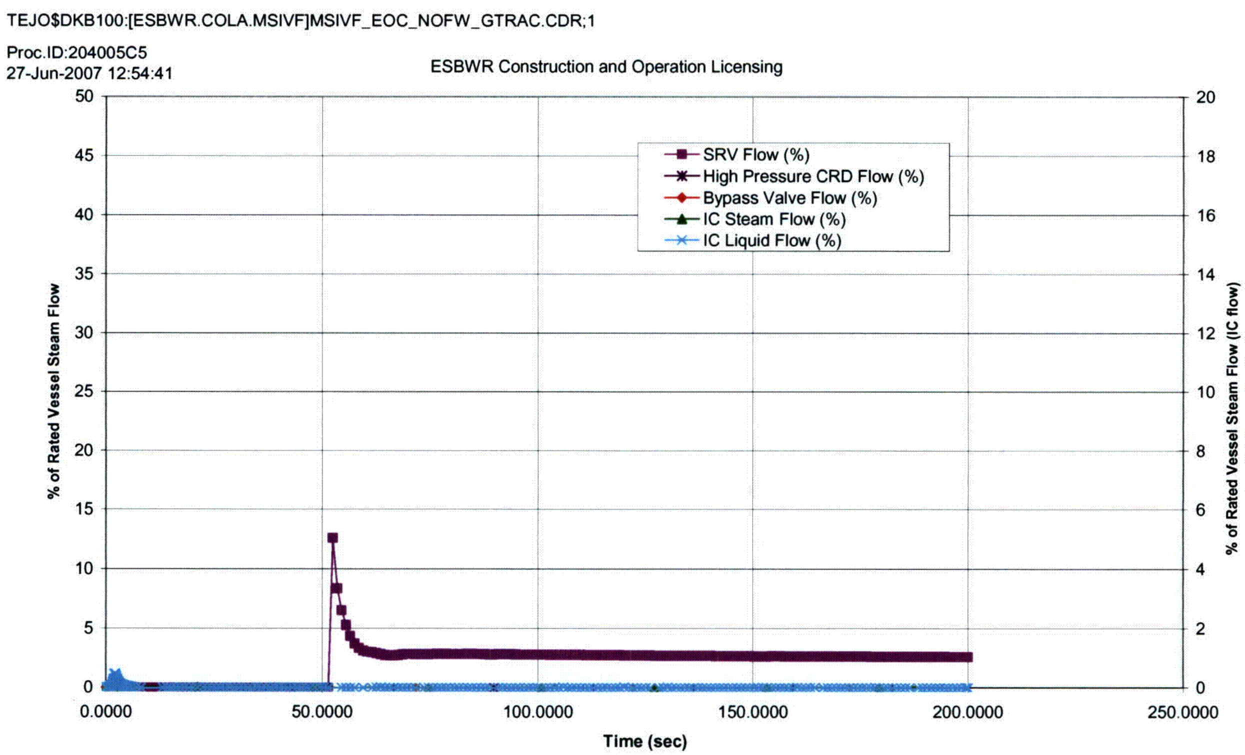


Figure 2.5.1-4b. MSIV Closure With Flux Scram

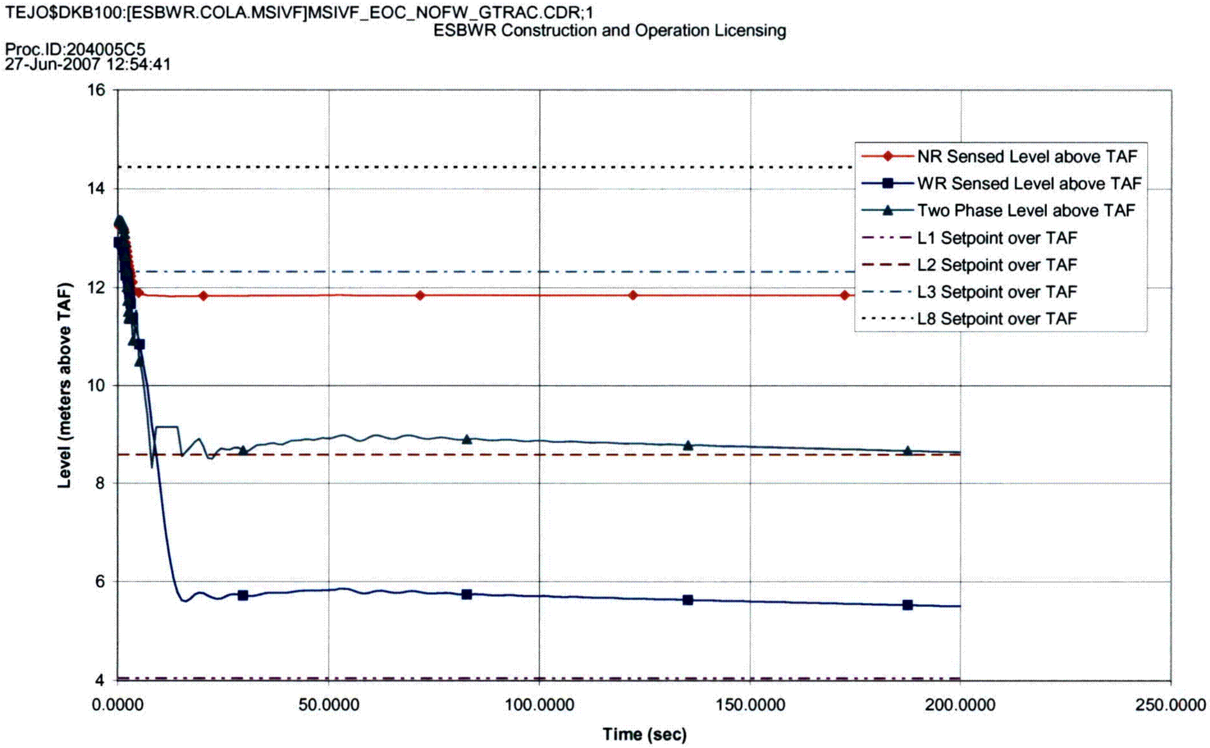


Figure 2.5.1-4c. MSIV Closure With Flux Scram

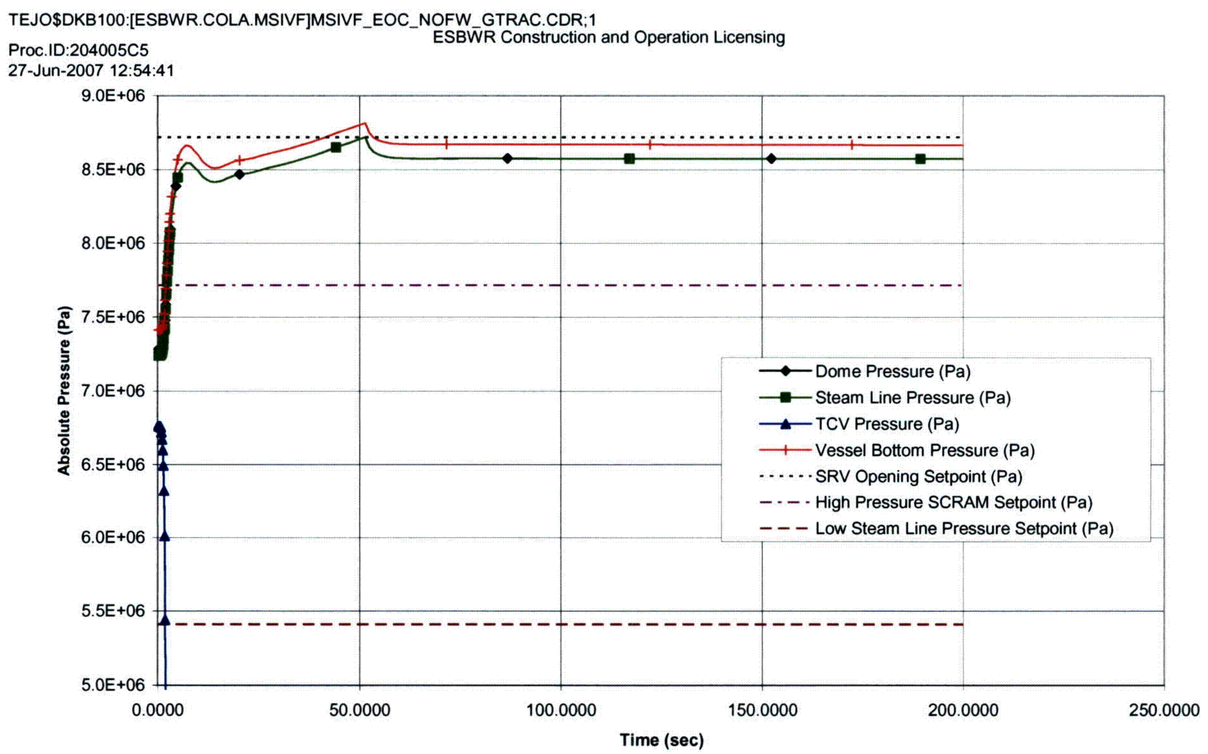


Figure 2.5.1-4d. MSIV Closure With Flux Scram

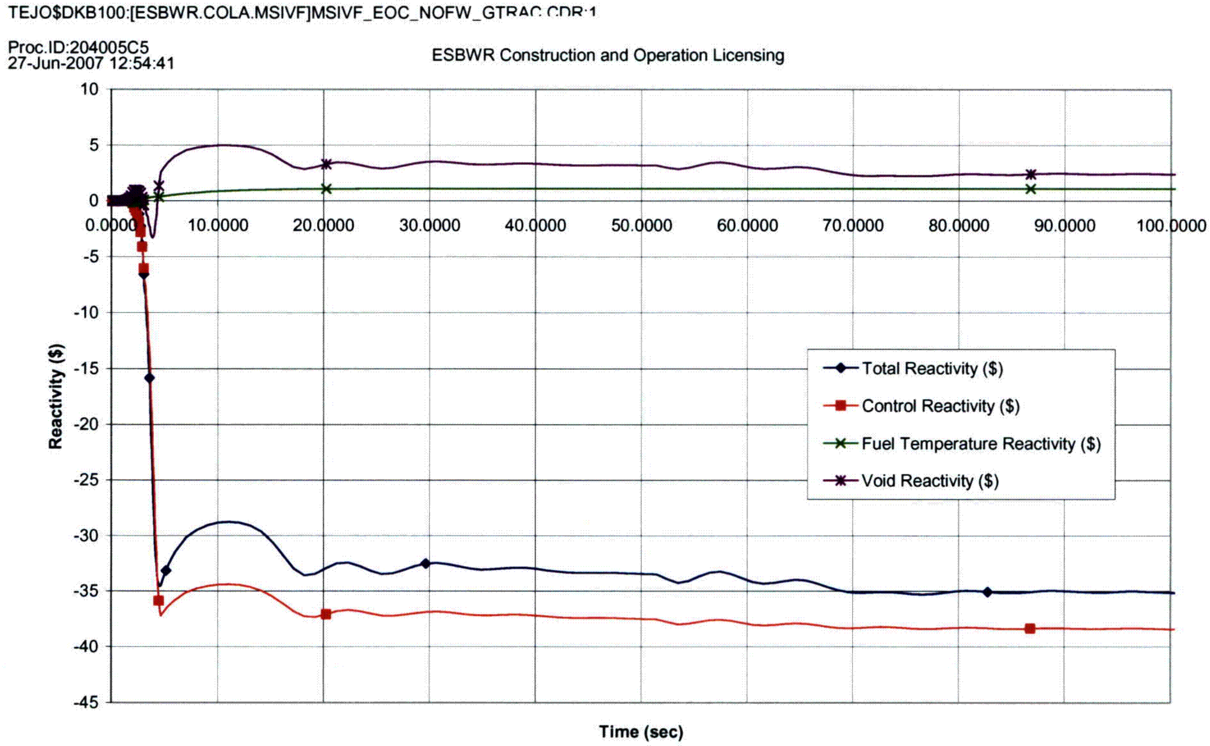


Figure 2.5.1-4e. MSIV Closure With Flux Scram

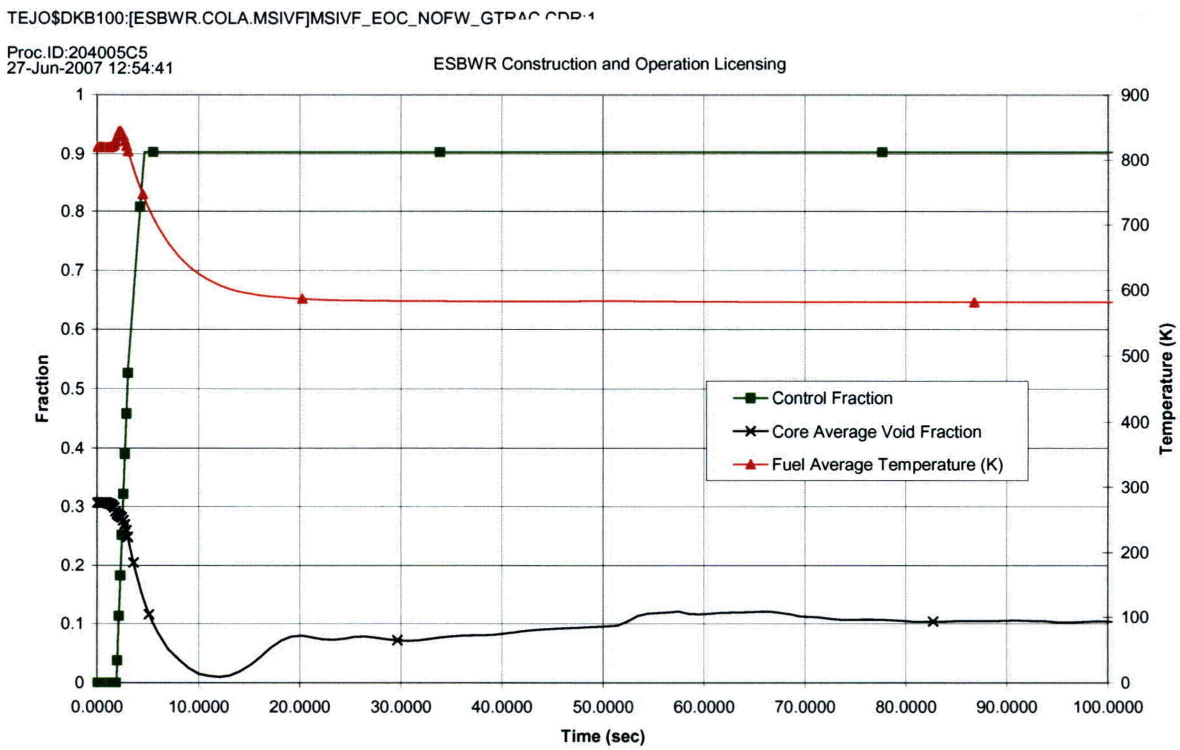


Figure 2.5.1-4f. MSIV Closure With Flux Scram

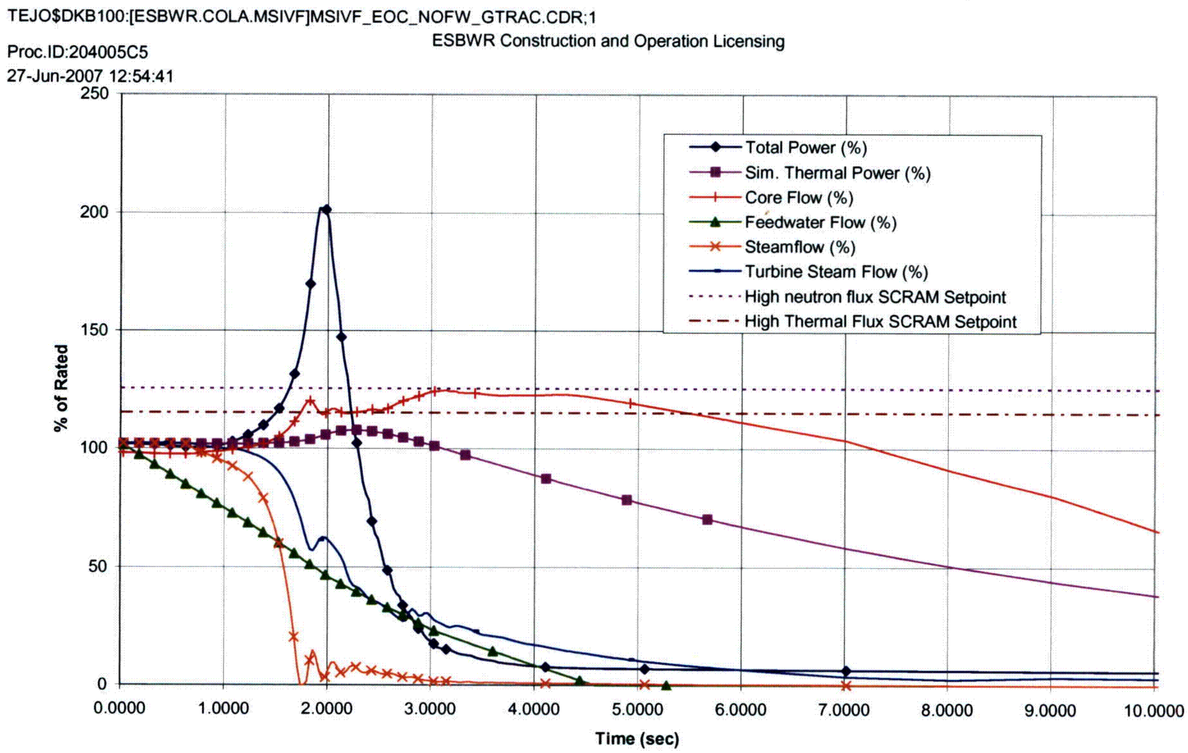
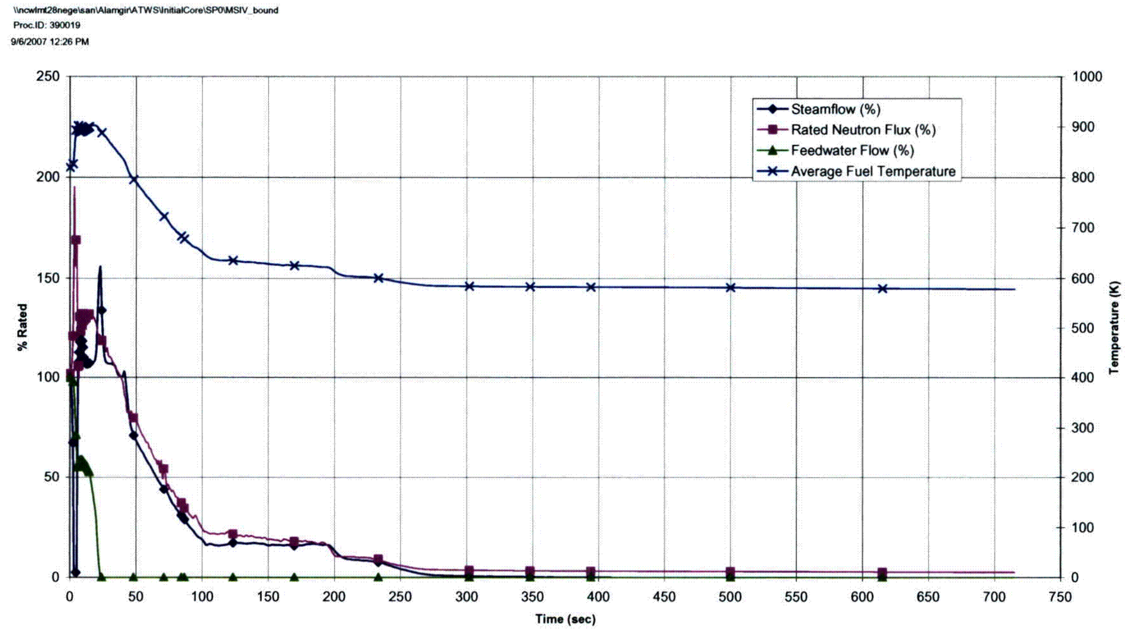


Figure 2.5.1-4g. MSIV Closure With Flux Scram



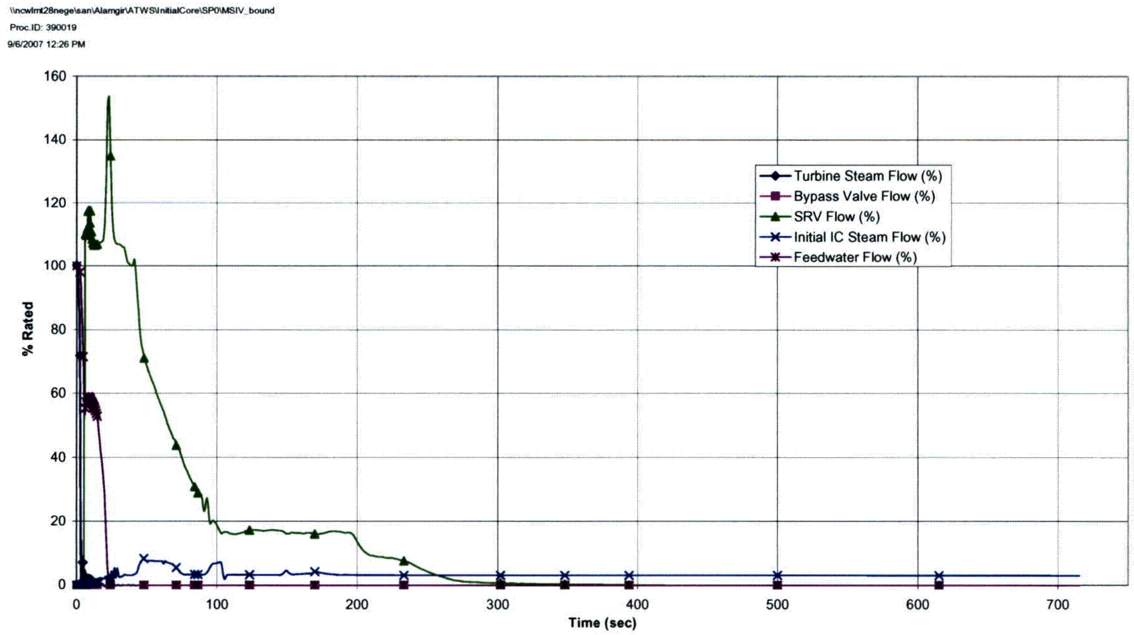


Figure 2.5.4-1a. MSIV Closure - SLC System Bounding Case

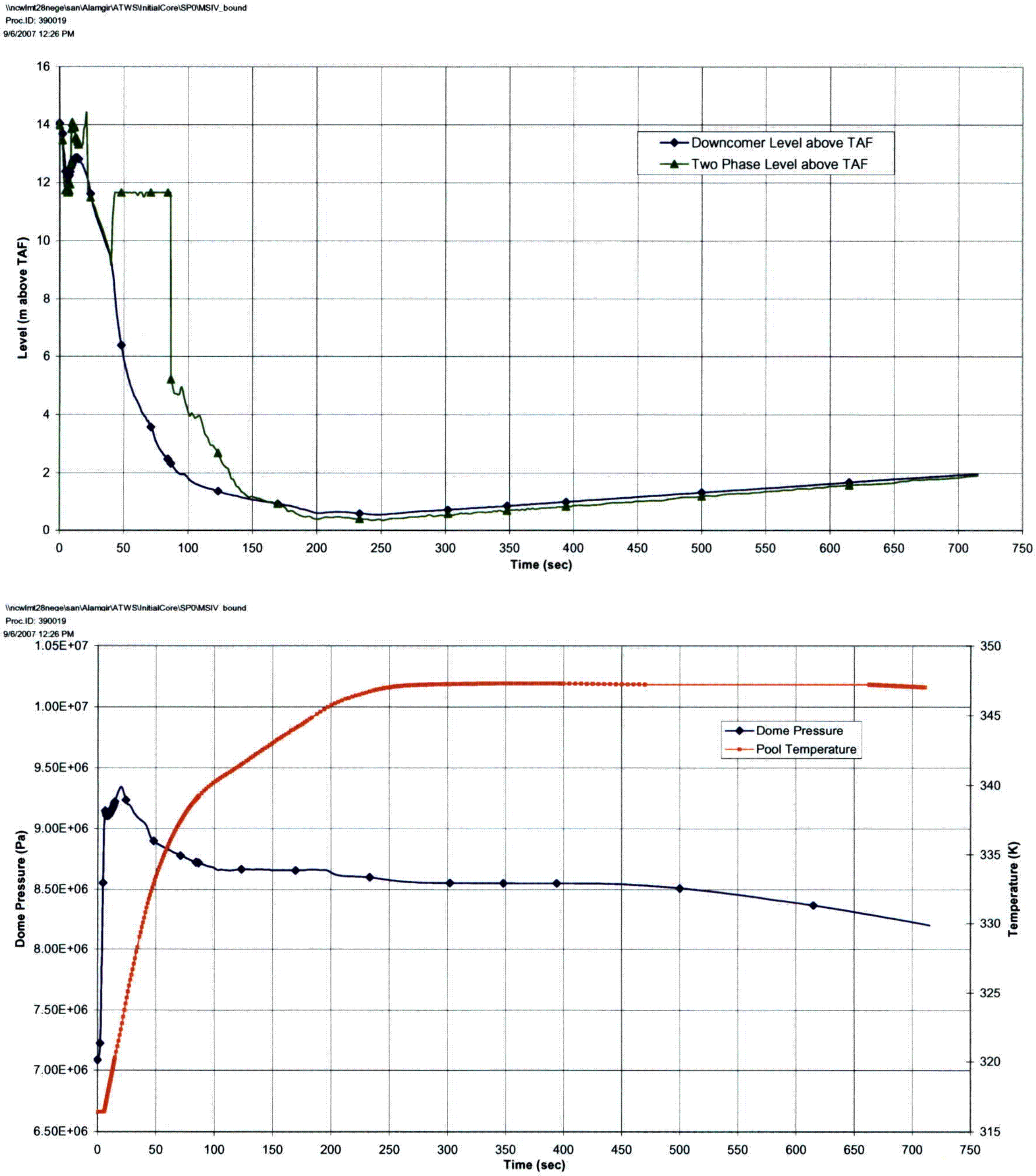


Figure 2.5.4-1b. MSIV Closure - SLC System Bounding Case

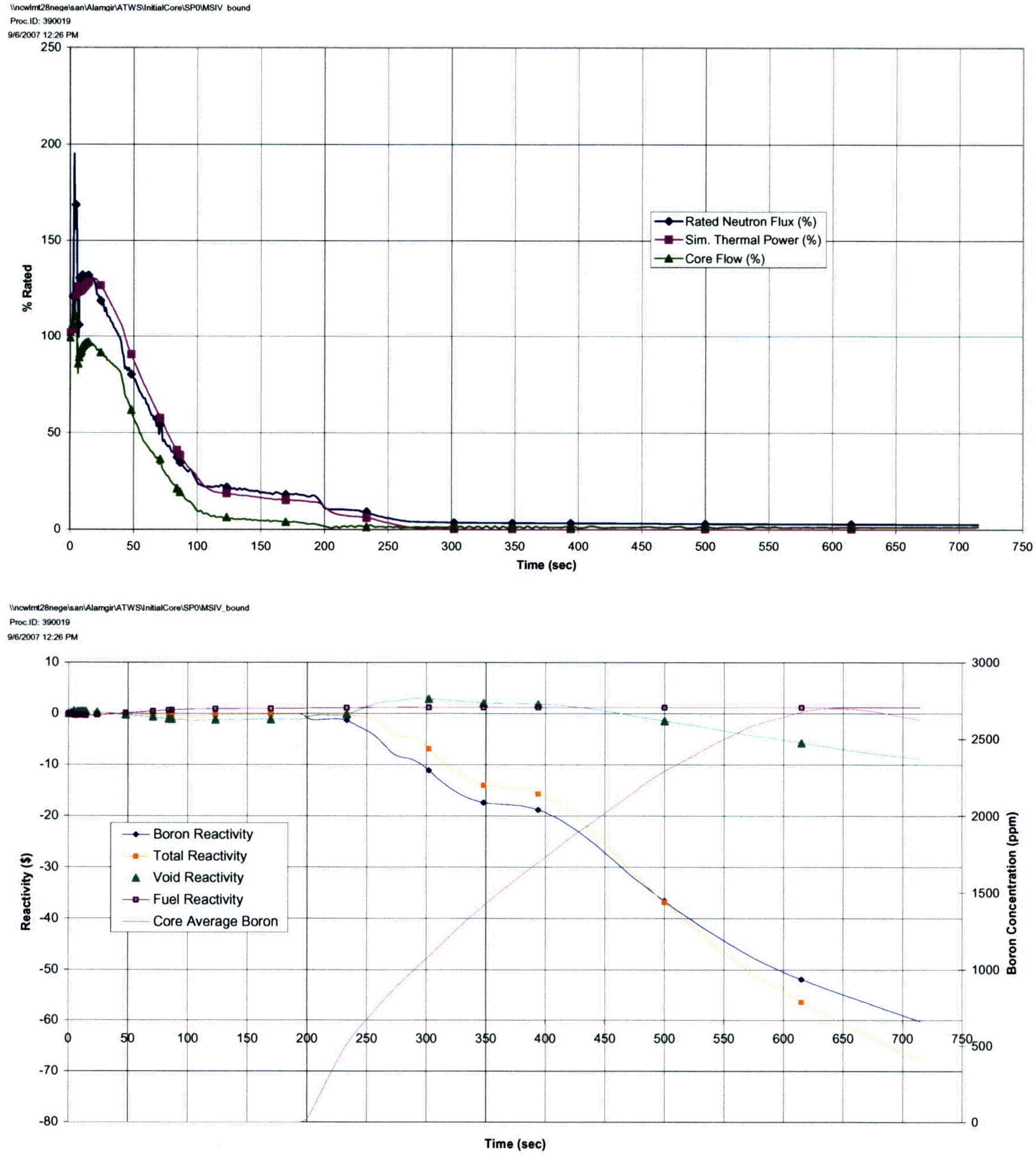


Figure 2.5.4-1c. MSIV Closure - SLC System Bounding Case

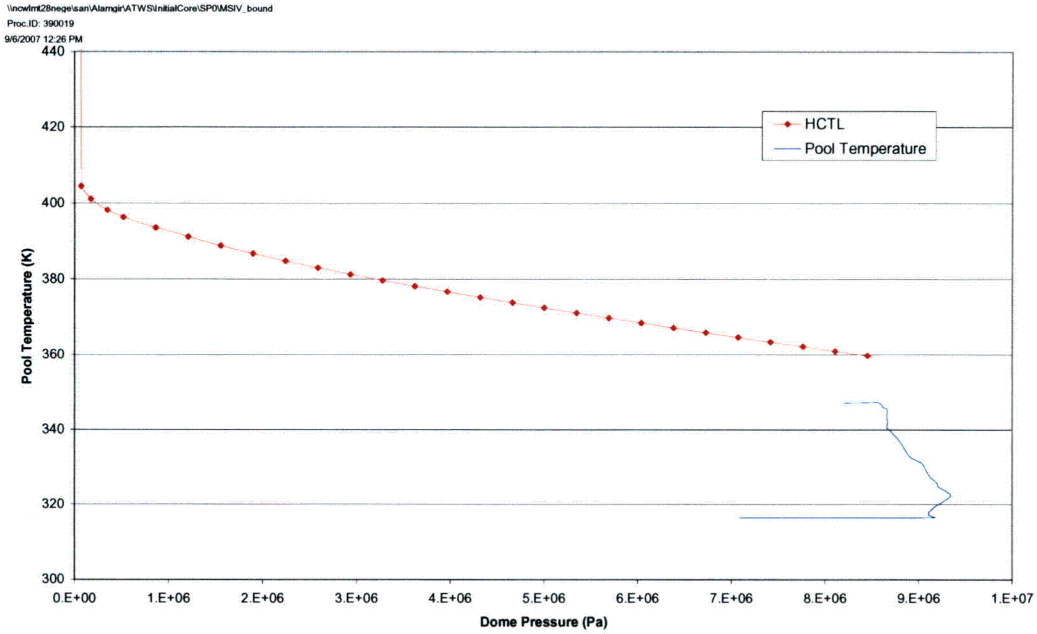
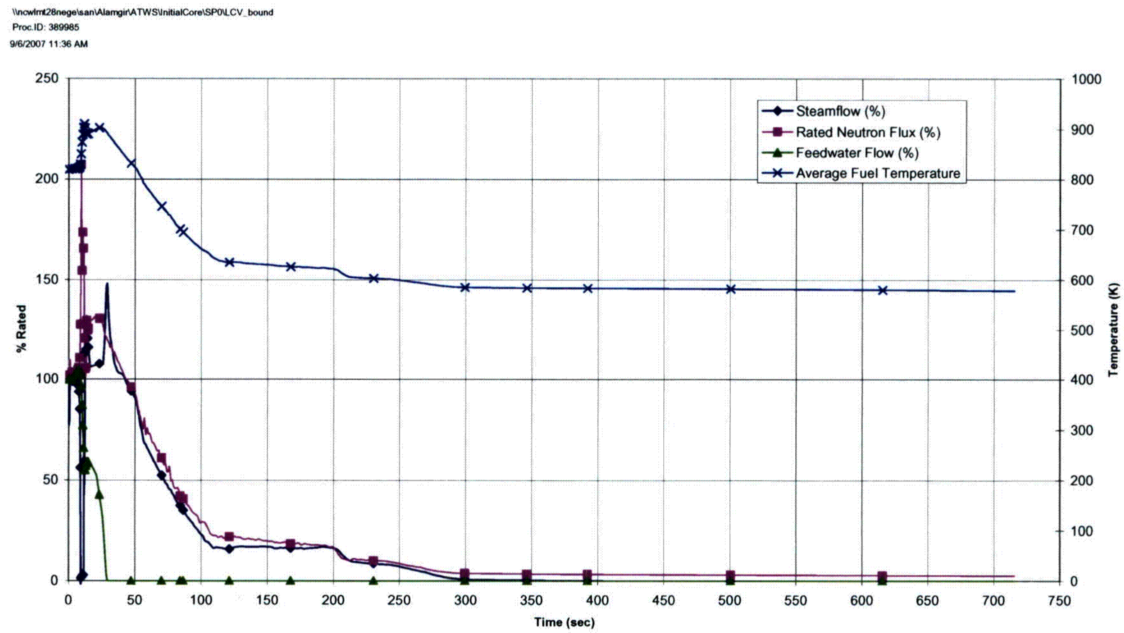


Figure 2.5.4-1d. MSIV Closure - SLC System Bounding Case



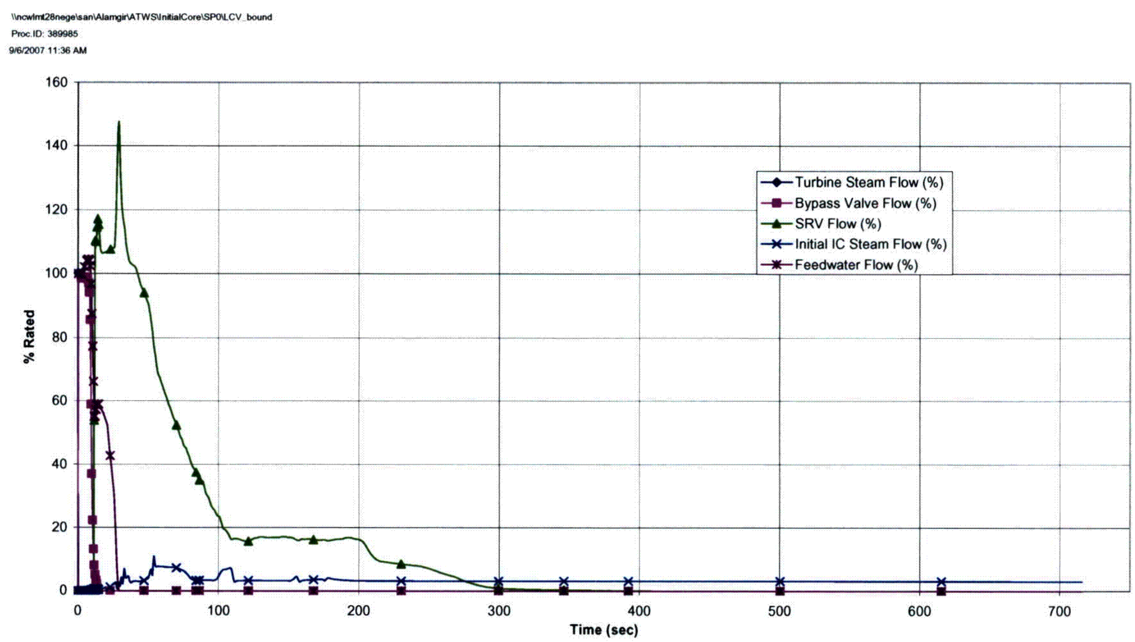


Figure 2.5.4-2a. Loss of Condenser Vacuum SLC System Bounding Case

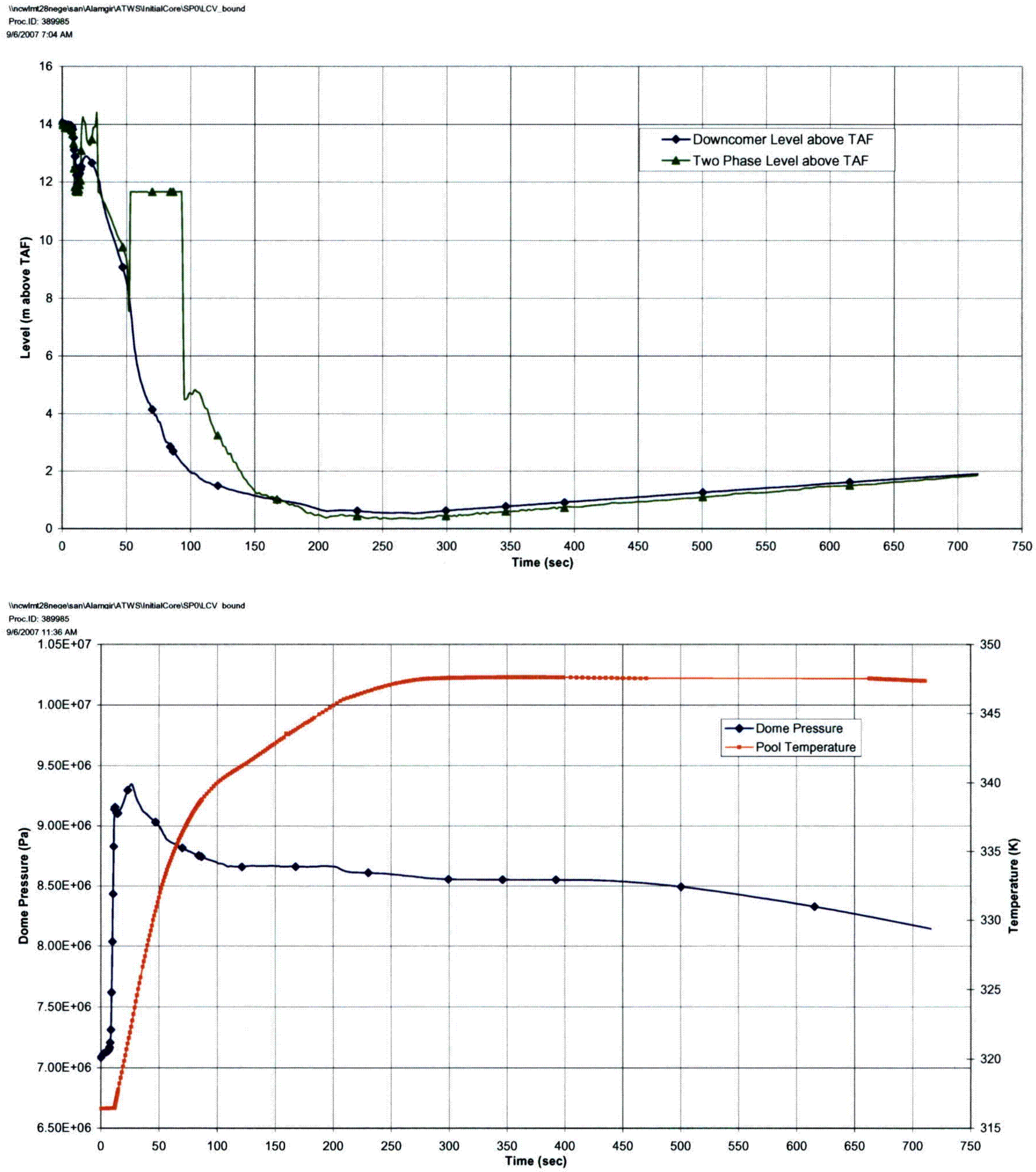


Figure 2.5.4-2b. Loss of Condenser Vacuum SLC System Bounding Case

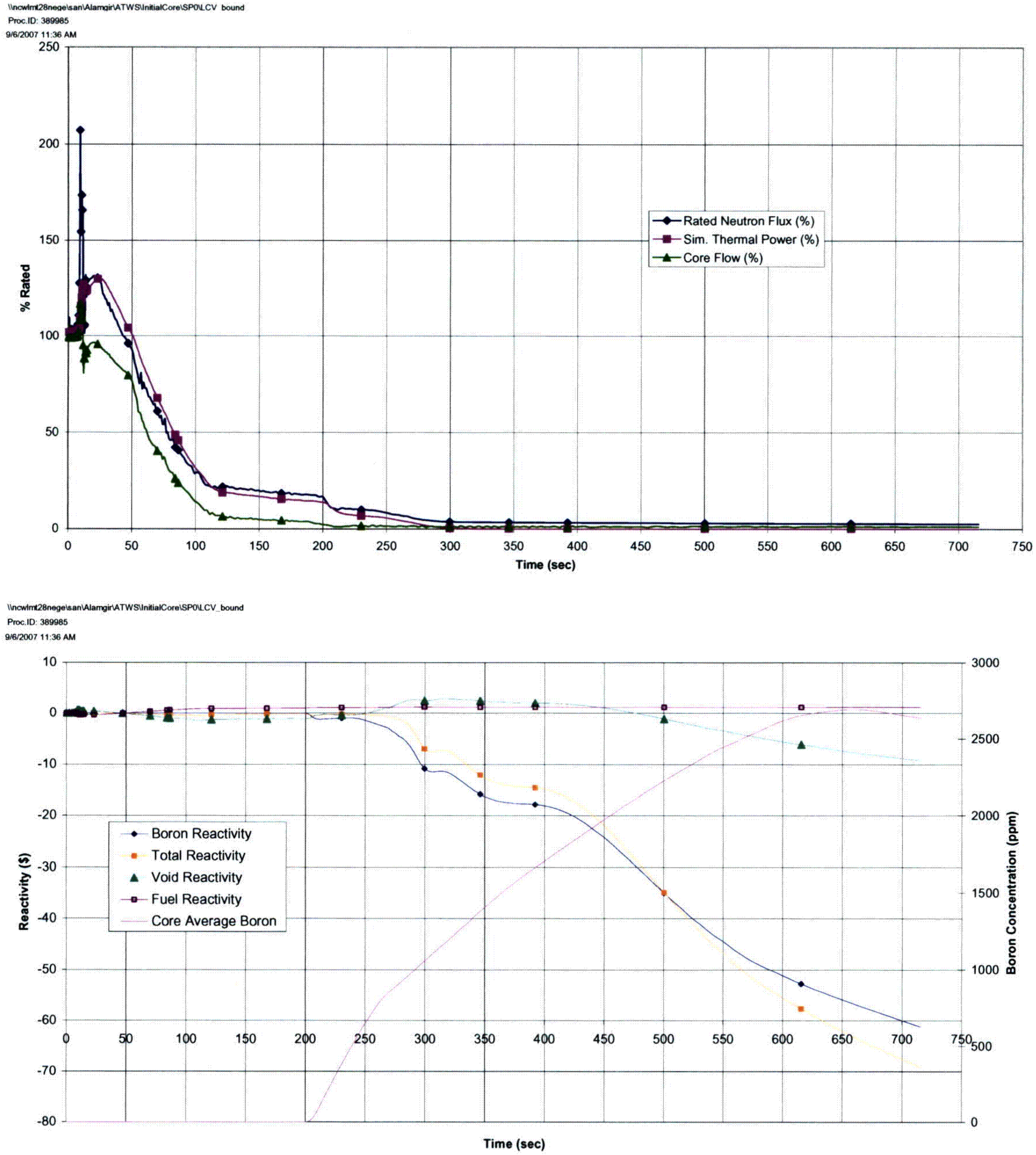


Figure 2.5.4-2c. Loss of Condenser Vacuum SLC System Bounding Case

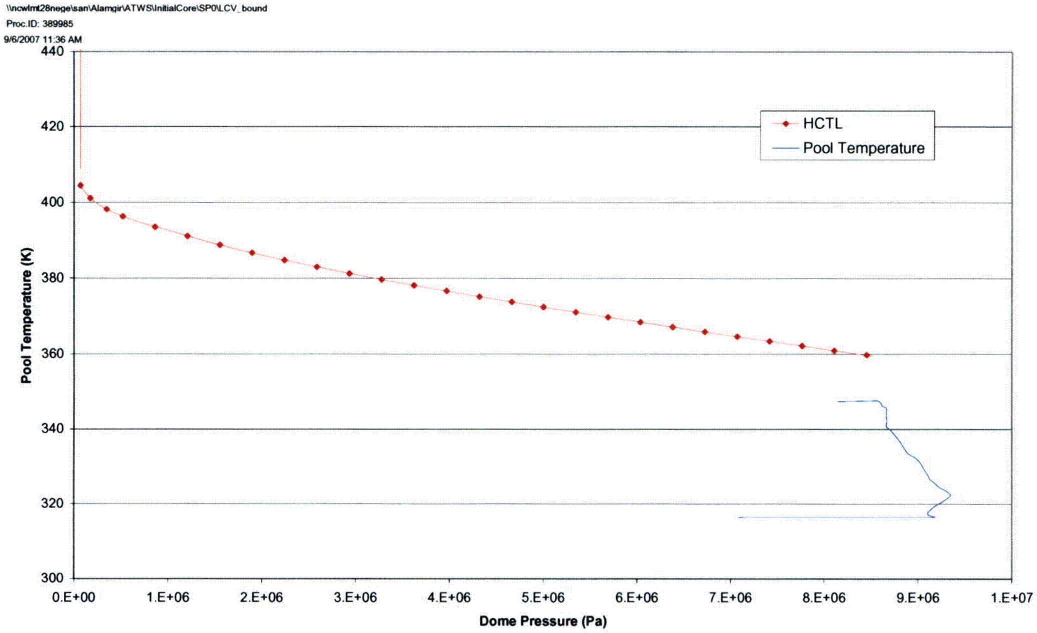


Figure 2.5.4-2d. Loss of Condenser Vacuum SLC System Bounding Case

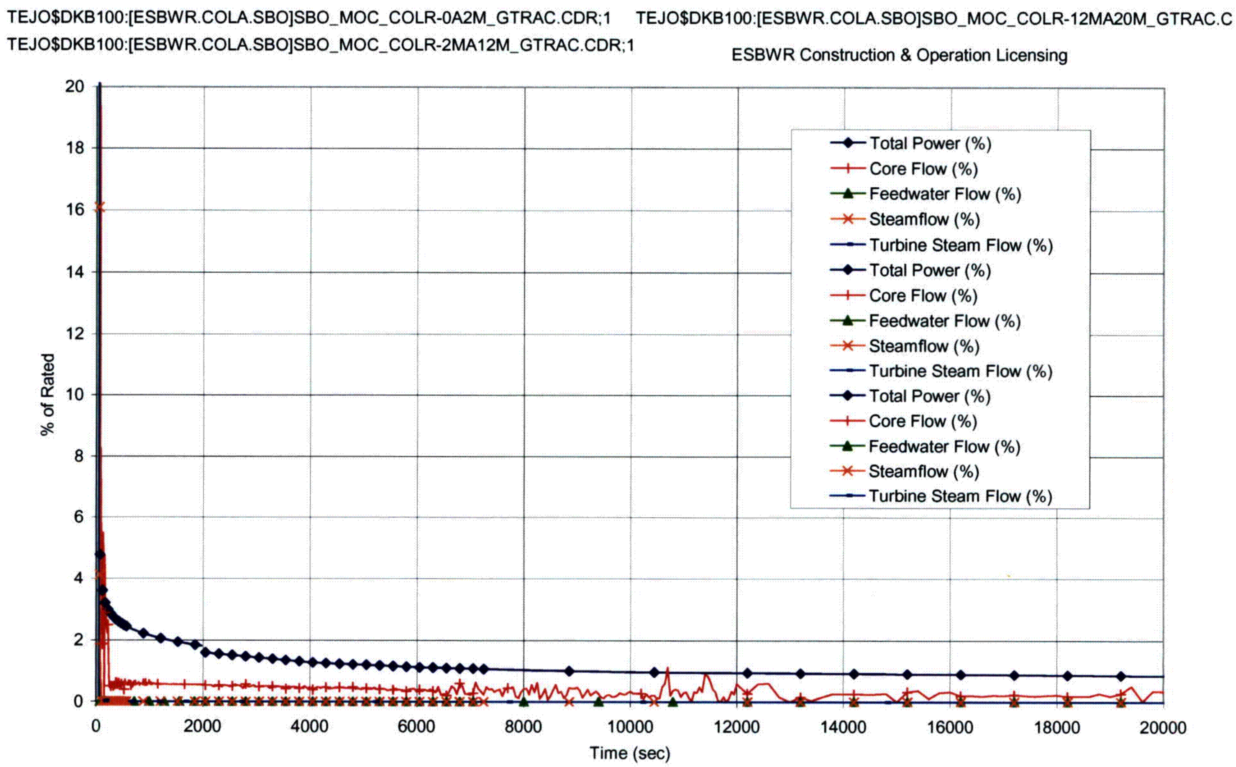


Figure 2.5.5-1a. Pressure Vessel Response for SBO

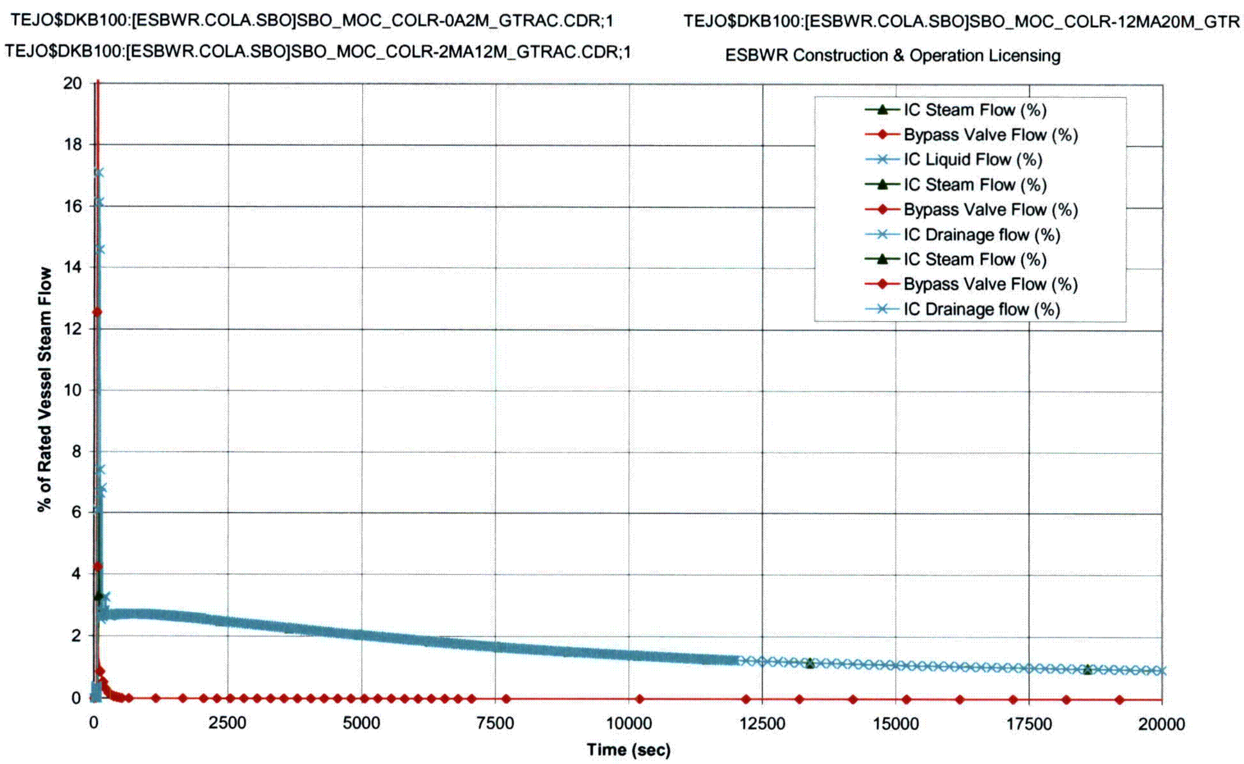


Figure 2.5.5-1b. Vessel inventory Makeup Flow Response for SBO

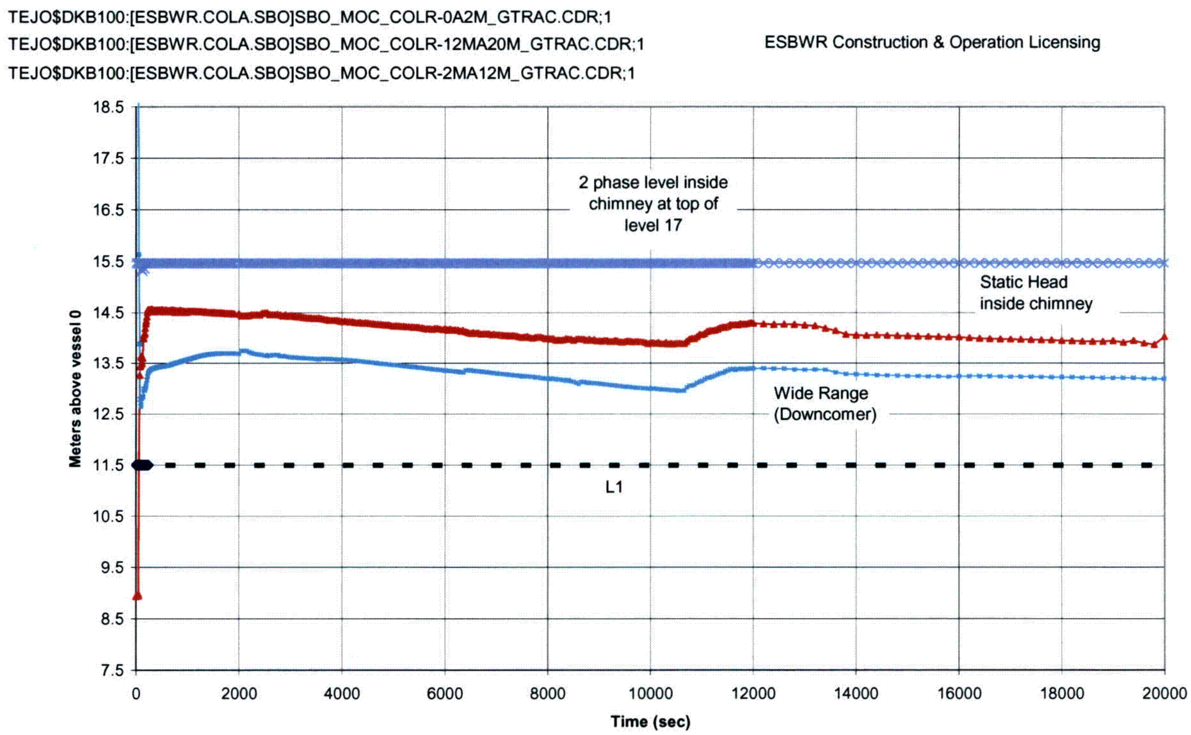


Figure 2.5.5-1c. Water Level Response for SBO

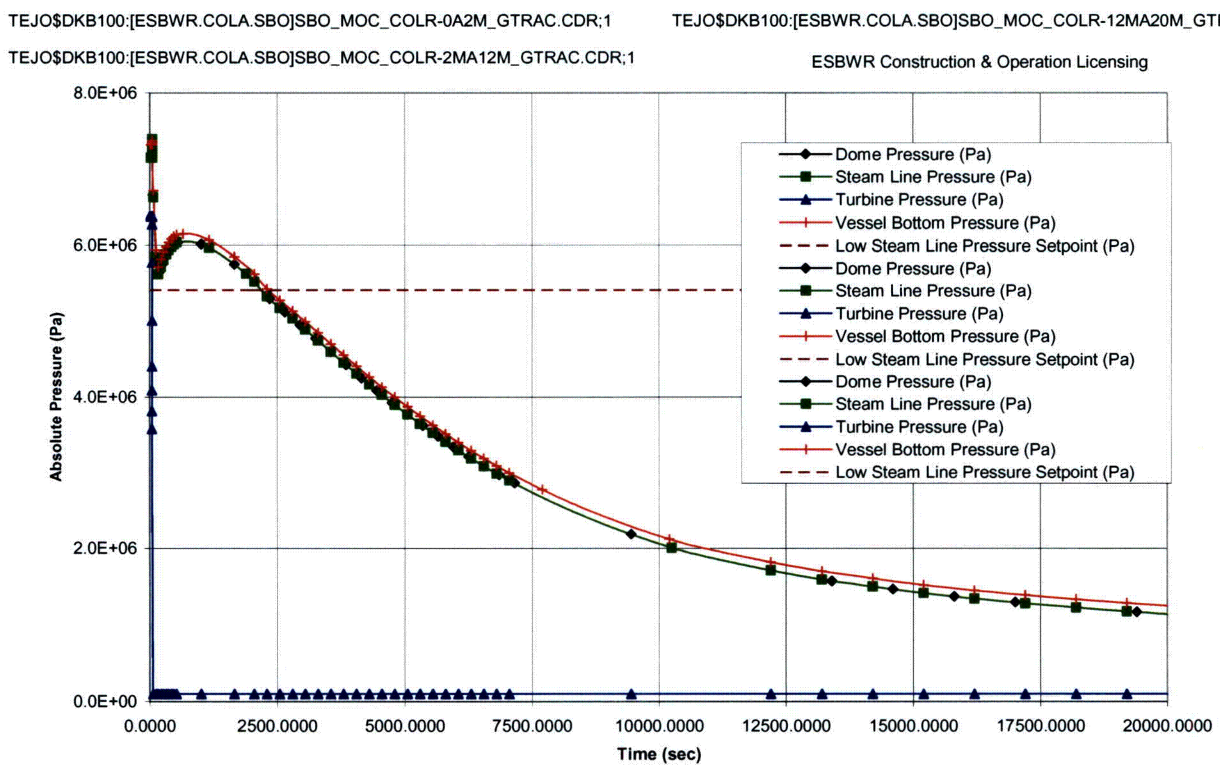


Figure 2.5.5-1d. Pressure Response for SBO

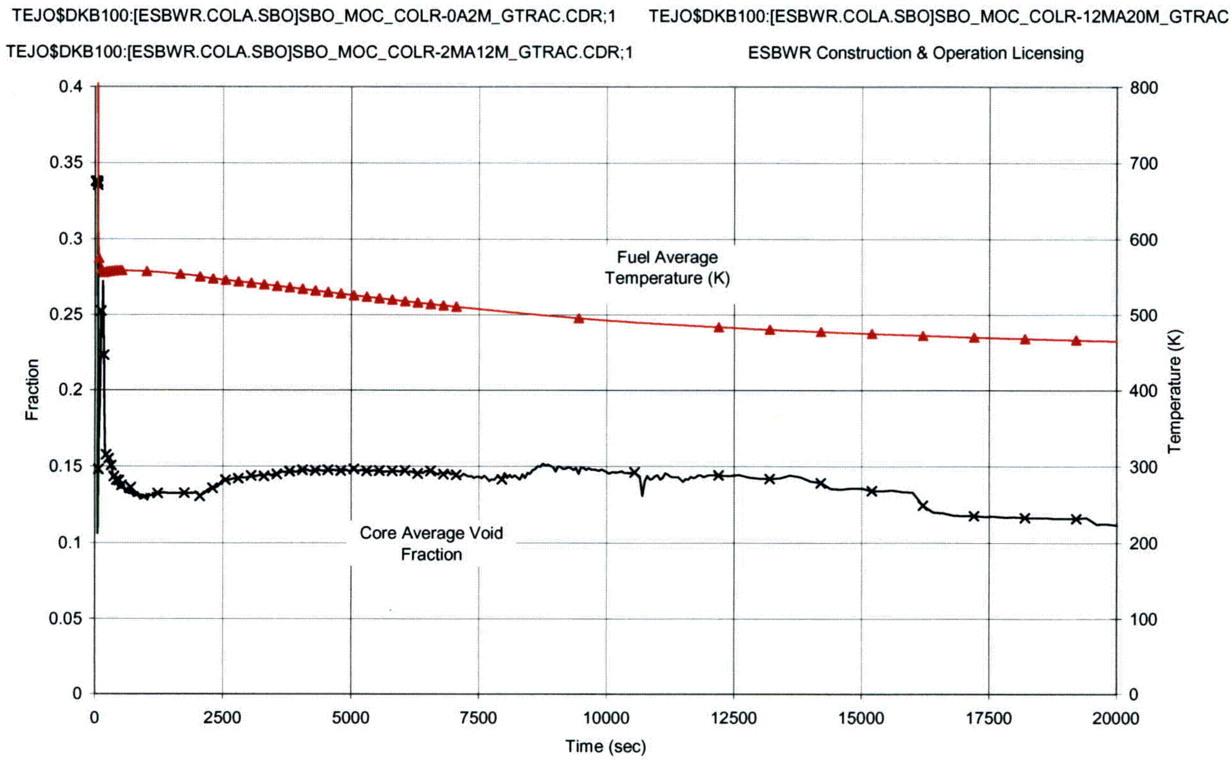


Figure 2.5.5-1e. Core void fraction and fuel temperature Response for SBO

3.0 COMPARISONS AND CONCLUSION

3.1 COMPARISON OF TRACG RESULTS BETWEEN THE EQUILIBRIUM AND INITIAL CORES

The evaluations performed in the report with the initial core are compared to the results documented in the ESBWR DCD with the equilibrium core. Specifically, the tables showing the results of the various events that are documented in this report are repeated here with the additional information showing the results that are documented in the DCD. Comparisons are provided for the stability analyses, Anticipated Operational Occurrences, Infrequent Events, Station Blackout Event, MSIVF, and ATWS. A comparison is not provided for the startup case because the startup procedures assumptions are different from that provided in the DCD in that the MSIVs are open for the initial-core analysis.

3.2 CONCLUSION

The analyses for the ESBWR limiting events with initial core are documented in this report and the results are compared to the equilibrium core documented in the DCD, Tier 2 Tables 3.1-1, through 3.1-6. All aspects of ESBWR safety which are specific to the core design/fuel loading pattern have been evaluated. The analyses show that the initial core design can be safely operated because the relevant acceptance criteria are met in all cases.

Table 3.1-1

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

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

Table 3.1-2

Comparison of Results Summary of Anticipated Operational Occurrence Events Between Equilibrium and Initial Core Cases

Description	Equilib. Core (Equil.) or Initial Core (IC)	Max. Neutron Flux, % NBR	Max. Dome Pressure, MPaG (psig)	Max. Vessel Bottom Pressure, MPaG (psig)	Max. Steamline Pressure, MPaG (psig)	Max. Core Average Surface Heat Flux, % of Initial	ΔCPR/ICPR or Minimum Water Level (m over TAF)
Loss of Feedwater Heating*	Equil	100.2	7.08 (1027)	7.21 (1046)	7.04 (1024)	100	0.04
	IC	116	7.11 (1031)	7.24 (1050)	7.06 (1024)	119	0.09
Closure of One Turbine Control Valve. FAST/SLOW	Equil	124	7.20 (1043)	7.33 (1063)	7.16 (1038)	102	0.04
		112	7.20 (1043)	7.33 (1063)	7.16 (1038)	102	0.03
	IC	125	7.20 (1043)	7.33 (1063)	7.16 (1038)	102	0.04
		110	7.20 (1043)	7.33 (1063)	7.16 (1038)	102	0.03
Generator Load Rejection with Turbine Bypass**	Equil	135	7.15 (1037)	7.29 (1057)	7.28 (1056)	102	0.03
	IC	128	7.15 (1037)	7.29 (1057)	7.28 (1056)	101	0.07
Generator Load Rejection with a Single Failure in the Turbine Bypass System	Equil	168	7.39 (1072)	7.53 (1091)	7.39 (1070)	103	0.03
	IC	151	7.37 (1070)	7.50 (1088)	7.37 (1069)	102	0.02
Turbine Trip with Turbine Bypass**	Equil	120	7.12 (1033)	7.26 (1053)	7.20 (1043)	101	0.02
	IC	116	7.12 (1033)	7.26 (1053)	7.20 (1043)	101	0.07
Turbine Trip with a Single Failure in the Turbine Bypass System	Equil	146	7.37 (1069)	7.50 (1088)	7.37 (1067)	102	0.02
	IC	131	7.34 (1065)	7.48 (1085)	7.34 (1065)	101	0.01
Closure of One MSIV	Equil	114	7.16 (1038)	7.30 (1059)	7.13 (1033)	101	0.02
	IC	114	7.16 (1038)	7.30 (1059)	7.13 (1033)	102	0.03

Table 3.1-2 (cont)
Comparison of Results Summary of Anticipated Operational Occurrence Events Between Equilibrium and Initial Core Cases

Description	**Equilib. Core (Equil.) or Initial Core (IC)**	**Max. Neutron Flux,** % NBR	**Max. Dome Pressure,** MPaG (psig)	**Max. Vessel Bottom Pressure,** MPaG (psig)	**Max. Steamline Pressure,** MPaG (psig)	**Max. Core Average Surface Heat Flux,** % of Initial	**ΔCPR/ICPR or Minimum Water Level** (m over TAF)
Closure of All MSIV	Equil	103	7.76 (1126)	7.89 (1143)	7.76 (1126)	100	≤ 0.01
	IC	102	7.67 (1112)	7.80 (1131)	7.67 (1112)	100	≤ 0.01
Loss of Condenser Vacuum	Equil	110	7.11 (1031)	7.26 (1053)	7.20 (1044)	100	≤ 0.01
	IC	107	7.12 (1032)	7.26 (1053)	7.20 (1044)	100	≤ 0.01
Inadvertent Isolation Condenser Initiation	Equil	113	7.08 (1027)	7.22 (1047)	7.04 (1021)	109	0.08
	IC	111	7.08 (1027)	7.22 (1047)	7.04 (1021)	109	0.09
Runout of One Feedwater Pump	Equil	103	7.08 (1027)	7.22 (1047)	7.05 (1023)	100	<0.01>
	IC	103	7.08 (1027)	7.22 (1047)	7.04 (1021)	101	≤ 0.01
Opening of One Turbine Control or Bypass Valve	Equil	102	7.08 (1027)	7.21 (1046)	7.04 (1021)	100	≤ 0.01
	IC	101	7.08 (1027)	7.21 (1046)	7.04 (1021)	100	≤ 0.01
Loss of Non-Emergency AC Power to Station Auxiliaries	Equil	136	7.13 (1035)	7.28 (1056)	7.28 (10546)	102	5.40 m
	IC	139	7.13 (1035)	7.28 (1056)	7.28 (1056)	102	5.37m
Loss of Feedwater Flow	Equil	100	7.08 (1027)	7.21 (1046)	7.04 (1021)	100	5.28 m
	IC	100	7.08 (1027)	7.21 (1046)	7.04 (1021)	100	5.28m

* Results are very different because the SRI/SCRRI is conservatively not credited for the case with the initial core.

** Differences in the ΔCPR/ICPR is due to a change in the SCRRI/SRI rod pattern

Table 3.1-3
Comparison of Results Summary of Infrequent Events Between Equilibrium and Initial Core Cases

Description	Equilib. Core (Equil.) or Initial Core (IC)	Max. Neutron Flux, % NBR	Max. Dome Pressure, MPaG (psig)	Max. Vessel Bottom Pressure, MPaG (psig)	Max. Steamline Pressure, MPaG (psig)	Max. Core Average Surface Heat Flux, % of Initial	ΔCPR/ ICPR
Loss of Feedwater Heating with SCRRI failure	Equil	122	7.13 (1034)	7.27 (1054)	7.09 (1028)	121	0.11
	IC	122	7.13 (1034)	7.27 (1054)	7.09 (1028)	122	0.12
FWCF – Maximum Demand	Equil	117	7.29 (1057)	7.43 (1078)	7.25 (1052)	109	0.04
	IC	115	7.24 (1050)	7.38 (1070)	7.26 (1053)	107	0.04
Pressure Regulator Failure – Opening of all TCVs and BPVs	Equil	100	7.08 (1027)	7.21 (1046)	7.04 (1021)	100	0.00
	IC	100	7.08 (1027)	7.21 (1046)	6.99 (1014)	100	<0.01
Pressure Regulator Failure – Closing of all TCVs and BPVs	Equil	137	8.06 (1169)	8.19 (1188)	8.06 (1169)	104	0.05
	IC	136	8.02 (1163)	8.15 (1182)	8.02 (1163)	104	0.03
Load Rejection with total bypass failure *	Equil	339	8.14 (1181)	8.27 (1199)	8.15 (1182)	108	0.11
	IC	257	8.09 (1173)	8.22 (1192)	8.10 (1175)	106	0.08
Turbine Trip with total bypass failure *	Equil	295	8.13 (1179)	8.26 (1198)	8.13 (1179)	108	0.11
	IC	231	8.08 (1172)	8.21 (1191)	8.08 (1172)	106	0.07
Inadvertent SRV open	Equil	101	7.08 (1027)	7.21 (1046)	6.99 (1014)	101	<0.01
	IC	101	7.08 (1027)	7.21 (1046)	6.99 (1014)	101	<0.01
Stuck open SRV	Equil	100.0	7.08 (1027)	7.21 (1046)	7.04 (1021)	100.0	<0.1
	IC	--	7.82 (1134)	7.96 (1155)	7.83 (1156)	--	<0.01

* The reduction in ΔCPR/ICPR is due to the lower initial core void coefficient.

Table 3.1-4

Comparison of the Theoretical Vessel Conditions at 72 hours after SBO Between Equilibrium and Initial Core Cases

Parameter	Equilib. Core	Initial Core
	Value	Value
Dome pressure, PaG (psig)	0 (0)	0 (0)
Vessel Bottom Pressure, PaG (psig)	123000 (17.8)	123000 (17.8)
Decay heat, MW	19.5	19.5
Wide range measured level over TAF, m (ft)	6.3 (20.7)	6.4 (21.0)
Collapsed Level over TAF, m (ft)	5.6 (18.4)	5.66 (18.6)
IC flow, kg/s (lb/hr)	8.6 (68,573)	8.6 (68,573)

Table 3.1-5

Comparison of ATWS MSIV Closure Summary – SLC System Bounding Between Equilibrium and Initial Core Cases

Parameter	Equilib. Core (Equil.) or Initial Core (IC)	Value	Time (s)
Sensed Maximum Neutron Flux, %	Equil	213	3
	IC	203.7	3.3
Maximum Vessel Bottom Pressure, MPaG (psig)	Equil	9.41 (1364)	21
	IC	9.38 (1360)	21
Maximum Bulk Suppression Pool Temperature, °C (°F)	Equil	73.1 (163)	370
	IC	74.2 (165.6)	368
Associated Containment Pressure, kPaG (psig)	Equil	206.2 (29.91)	370
	IC	208.8 (30.3)	368
Peak Cladding Temperature, °C (°F)	Equil	850.3 (1562.5)	28
	IC	791.1(1455.9)	28

Table 3.1-6

Comparison of ATWS Loss of Condenser Vacuum Summary – SLC System Bounding Case Between Equilibrium and Initial Core Cases

Parameter	Equilib. Core (Equil.) or Initial Core (IC)	Value	Time (s)
Sensed Maximum Neutron Flux, %	Equil	229	9
	IC	211.71	9.3
Maximum Vessel Bottom Pressure, MPaG (psig)	Equil	9.40 (1364)	26
	IC	9.38 (1361)	27
Maximum Bulk Suppression Pool Temperature, °C (°F)	Equil	73.0 (164)	364
	IC	74.5 (166.1)	360
Associated Containment Pressure, kPaG (psig)	Equil	206.1 (29.89)	364
	IC	209.6 (30.4)	360
Peak Cladding Temperature, °C (°F)	Equil	850.0 (1562)	34
	IC	790.2 (1454.3)	34