GE Hitachi Nuclear Energy

NEDO-33337

Revision 1

Class I

DRF Section 0000-0074-9757-03

April 2009

Licensing Topical Report

# ESBWR INITIAL CORE TRANSIENT ANALYSES

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# 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:

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## Changes From Revision 0 to Revision 1

Most changes are made to update limiting analyses with the PC-level-2 version of the TRACG code and with TRACG model updates. The TRACG model was updated to include design change in the shroud leg, bounding steamline model, and updated channel spacer loss coefficients. Analyses were also performed with the updated fuel loading pattern (Reference 1.1-2).

In addition, some changes were made to be consistent with similar changes made in the ESBWR DCD.

## Global Abbreviations and Acronyms List

| Term | Definition |
|---|---|
| 10 CFR | Title 10, Code of Federal Regulations |
| ABWR | Advanced Boiling Water Reactor |
| AC | Alternating Current |
| ADS | Automatic Depressurization System |
| AOO | Anticipated Operational Occurrence |
| APRM | Average Power Range Monitor |
| ARI | Alternate Rod Insertion |
| ASME | American Society of Mechanical Engineers |
| ATLM | Automated Thermal Limit Monitor |
| ATWS | Anticipated Transients Without Scram |
| BAF | Bottom of Active Fuel |
| BOC | Beginning of Cycle |
| BOP | Balance of Plant |
| BPV | Bypass Valve |
| BWR | Boiling Water Reactor |
| CPR | Critical Power Ratio |
| CRD | Control Rod Drive |
| CSAU | Code Scaling Applicability and Uncertainty |
| DCD | Design Control Document |
| 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 |
| FMCRD | Fine Motion Control Rod Drive |
| FW | Feedwater |
| 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 |
| HP CRD | High Pressure Control Rod Drive |
| ICPR | Initial Critical Power Ratio |
| LCV | Loss Condenser Vacuum |
| LOCA | Loss-of-Coolant-Accident |
| LOFW | Loss of Feedwater |

| Term | Definition |
|---|---|
| LOFWH | Loss of Feedwater Heating |
| LPRM | Local Power Range Monitor |
| MCPR | Minimum Critical Power Ratio |
| MOC | Middle of Cycle |
| MRBM | Multi-Channel Rod Block Monitor |
| 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 |
| OLMCPR | Operating Limit Minimum Critical Power Ratio |
| OSUTL | One Sided Upper Tolerance Limit |
| PCCS | Passive Containment Cooling System |
| RC&IS | Rod Control and Information System |
| RCPB | Reactor Coolant Pressure Boundary |
| 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 |

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# 1. 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 420ºF 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 or SP1M) 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 Analyses
  - 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.
- Analyses of Anticipated Operational Occurrences (AOOs) – All of the AOOs that are documented in the DCD are addressed with the initial core.
- Analyses of Infrequent Events (IEs) – All of the IEs that are documented in the DCD are addressed 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.
- Analyses 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;
- There is no difference in temperature;
- The core neutronic feedback is not important because the reactor is immediately scrammed during a LOCA event; and
- 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**

The codes shown in DCD, Tier 2, Table 15.0-8 applies to the same events analyzed with the initial core.

## 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. The analyses are provided in Section 2, and the conclusion is provided in Section 3.1.

## 1.4 REFERENCES

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

1.1-2 Global Nuclear Fuel, "ESBWR Initial Core Nuclear Design Report", NEDC-33326-P, Class III (Proprietary), Revision 1, March 2009, NEDO-33326, Class I (Non-Proprietary), Revision 1, March 2009.

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

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# 2. 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 one of the hottest channels in 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 determination 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 “Critical Power and Pressure Drop Tests of Simulated 10x10 Bundle Designs Applicable to GE14, NEDC-32874P, March 2000.

**Table 2.1-1a**

**Void Distribution for MOC Analyzed Core - TRACG Average Channel**

| **Channel Power = 4.036 MW, CPR = 1.79<br>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) | 0.41 |
| 2 (BAF+0.06) | 0.00 | 18 (BAF+0.84) | 0.51 |
| 3 (BAF+0.10) | 0.00 | 19 (BAF+0.99) | 0.58 |
| 4 (BAF+0.13) | 0.00 | 20 (BAF+1.14) | 0.64 |
| 5 (BAF+0.17) | 0.01 | 21 (BAF+1.30) | 0.68 |
| 6 (BAF+0.21) | 0.02 | 22 (BAF+1.45) | 0.71 |
| 7 (BAF+0.25) | 0.03 | 23 (BAF+1.60) | 0.73 |
| 8 (BAF+0.29) | 0.04 | 24 (BAF+1.75) | 0.74 |
| 9 (BAF+0.32) | 0.06 | 25 (BAF+1.91) | 0.74 |
| 10 (BAF+0.36) | 0.10 | 26 (BAF+2.06) | 0.760 |
| 11 (BAF+0.40) | 0.12 | 27 (BAF+2.21) | 0.77 |
| 12 (BAF+0.44) | 0.15 | 28 (BAF+2.36) | 0.79 |
| 13 (BAF+0.48) | 0.19 | 29 (BAF+2.51) | 0.80 |
| 14 (BAF+0.51) | 0.22 | 30 (BAF+2.67) | 0.81 |
| 15 (BAF+0.55) | 0.25 | 31 (BAF+2.82) | 0.82 |
| 16 (BAF+0.59) | 0.28 | 32 (BAF+2.97) | 0.82 |

**Table 2.1-1b**

**Void Distribution for MOC Analyzed Core - TRACG Hot Channel**

| **Channel Power = 5.062 MW, CPR = 1.35<br>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) | 0.56 |
| 2 (BAF+0.06) | 0.00 | 18 (BAF+0.84) | 0.66 |
| 3 (BAF+0.10) | 0.00 | 19 (BAF+0.99) | 0.71 |
| 4 (BAF+0.13) | 0.00 | 20 (BAF+1.14) | 0.74 |
| 5 (BAF+0.17) | 0.03 | 21 (BAF+1.30) | 0.74 |
| 6 (BAF+0.21) | 0.05 | 22 (BAF+1.45) | 0.76 |
| 7 (BAF+0.25) | 0.08 | 23 (BAF+1.60) | 0.79 |
| 8 (BAF+0.29) | 0.10 | 24 (BAF+1.75) | 0.82 |
| 9 (BAF+0.32) | 0.14 | 25 (BAF+1.91) | 0.84 |
| 10 (BAF+0.36) | 0.19 | 26 (BAF+2.06) | 0.85 |
| 11 (BAF+0.40) | 0.23 | 27 (BAF+2.21) | 0.87 |
| 12 (BAF+0.44) | 0.27 | 28 (BAF+2.36) | 0.88 |
| 13 (BAF+0.48) | 0.31 | 29 (BAF+2.51) | 0.89 |
| 14 (BAF+0.51) | 0.35 | 30 (BAF+2.67) | 0.89 |
| 15 (BAF+0.55) | 0.38 | 31 (BAF+2.82) | 0.90 |
| 16 (BAF+0.59) | 0.42 | 32 (BAF+2.97) | 0.90 |

**Table 2.1-2a**

**Flow Quality Distribution for MOC Analyzed Core - TRACG Average Channel**

| **Channel Power = 4.036 MW, CPR = 1.79<br>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) | 0.05 |
| 2 (BAF+0.06) | 0.00 | 18 (BAF+0.84) | 0.07 |
| 3 (BAF+0.10) | 0.00 | 19 (BAF+0.99) | 0.09 |
| 4 (BAF+0.13) | 0.00 | 20 (BAF+1.14) | 0.12 |
| 5 (BAF+0.17) | 0.00 | 21 (BAF+1.30) | 0.14 |
| 6 (BAF+0.21) | 0.00 | 22 (BAF+1.45) | 0.16 |
| 7 (BAF+0.25) | 0.00 | 23 (BAF+1.60) | 0.18 |
| 8 (BAF+0.29) | 0.00 | 24 (BAF+1.75) | 0.19 |
| 9 (BAF+0.32) | 0.00 | 25 (BAF+1.91) | 0.21 |
| 10 (BAF+0.36) | 0.01 | 26 (BAF+2.06) | 0.23 |
| 11 (BAF+0.40) | 0.01 | 27 (BAF+2.21) | 0.24 |
| 12 (BAF+0.44) | 0.01 | 28 (BAF+2.36) | 0.26 |
| 13 (BAF+0.48) | 0.01 | 29 (BAF+2.51) | 0.27 |
| 14 (BAF+0.51) | 0.02 | 30 (BAF+2.67) | 0.28 |
| 15 (BAF+0.55) | 0.02 | 31 (BAF+2.82) | 0.28 |
| 16 (BAF+0.59) | 0.03 | 32 (BAF+2.97) | 0.29 |

**Table 2.1-2b**

**Flow Quality Distribution for MOC Analyzed Core - Hot Channel**

| **Channel Power = 5.062 MW, CPR = 1.35<br>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) | 0.12 |
| 3 (BAF+0.10) | 0.00 | 19 (BAF+0.99) | 0.16 |
| 4 (BAF+0.13) | 0.00 | 20 (BAF+1.14) | 0.19 |
| 5 (BAF+0.17) | 0.00 | 21 (BAF+1.30) | 0.22 |
| 6 (BAF+0.21) | 0.00 | 22 (BAF+1.45) | 0.25 |
| 7 (BAF+0.25) | 0.00 | 23 (BAF+1.60) | 0.27 |
| 8 (BAF+0.29) | 0.01 | 24 (BAF+1.75) | 0.30 |
| 9 (BAF+0.32) | 0.01 | 25 (BAF+1.91) | 0.32 |
| 10 (BAF+0.36) | 0.01 | 26 (BAF+2.06) | 0.34 |
| 11 (BAF+0.40) | 0.02 | 27 (BAF+2.21) | 0.36 |
| 12 (BAF+0.44) | 0.02 | 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) | 0.04 | 31 (BAF+2.82) | 0.41 |
| 16 (BAF+0.59) | 0.05 | 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 = 4.036 MW, CPR = 1.79<br>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) | 0.09 (0.36) | 17 (BAF+0.69) | 1.55 |
| 2 (BAF+0.06) | 0.09 | 18 (BAF+0.84) | 1.46 |
| 3 (BAF+0.10) | 0.09 | 19 (BAF+0.99) | 1.36 |
| 4 (BAF+0.13) | 0.09 | 20 (BAF+1.14) | 1.27 |
| 5 (BAF+0.17) | 0.27 (1.07) | 21 (BAF+1.30) | 1.19 |
| 6 (BAF+0.21) | 0.27 | 22 (BAF+1.45) | 1.13 |
| 7 (BAF+0.25) | 0.27 | 23 (BAF+1.60) | 1.11 |
| 8 (BAF+0.29) | 0.27 | 24 (BAF+1.75) | 1.11 |
| 9 (BAF+0.32) | 0.36 (1.46) | 25 (BAF+1.91) | 1.02 |
| 10 (BAF+0.36) | 0.36 | 26 (BAF+2.06) | 0.96 |
| 11 (BAF+0.40) | 0.36 | 27 (BAF+2.21) | 0.90 |
| 12 (BAF+0.44) | 0.36 | 28 (BAF+2.36) | 0.79 |
| 13 (BAF+0.48) | 0.40 (1.59) | 29 (BAF+2.51) | 0.65 |
| 14 (BAF+0.51) | 0.40 | 30 (BAF+2.67) | 0.50 |
| 15 (BAF+0.55) | 0.40 | 31 (BAF+2.82) | 0.35 |
| 16 (BAF+0.59) | 0.40 | 32 (BAF+2.97) | 0.15 |

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 = 5.062 MW, CPR = 1.35<br>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 (0.38) | 17 (BAF+0.69) | 1.55 |
| 2 (BAF+0.06) | 0.10 | 18 (BAF+0.84) | 1.49 |
| 3 (BAF+0.10) | 0.10 | 19 (BAF+0.99) | 1.42 |
| 4 (BAF+0.13) | 0.10 | 20 (BAF+1.14) | 1.35 |
| 5 (BAF+0.17) | 0.28 (1.12) | 21 (BAF+1.30) | 1.28 |
| 6 (BAF+0.21) | 0.28 | 22 (BAF+1.45) | 1.20 |
| 7 (BAF+0.25) | 0.28 | 23 (BAF+1.60) | 1.13 |
| 8 (BAF+0.29) | 0.28 | 24 (BAF+1.75) | 1.06 |
| 9 (BAF+0.32) | 0.37 (1.47) | 25 (BAF+1.91) | 0.96 |
| 10 (BAF+0.36) | 0.37 | 26 (BAF+2.06) | 0.89 |
| 11 (BAF+0.40) | 0.37 | 27 (BAF+2.21) | 0.82 |
| 12 (BAF+0.44) | 0.37 | 28 (BAF+2.36) | 0.73 |
| 13 (BAF+0.48) | 0.39 (1.57) | 29 (BAF+2.51) | 0.60 |
| 14 (BAF+0.51) | 0.39 | 30 (BAF+2.67) | 0.48 |
| 15 (BAF+0.55) | 0.39 | 31 (BAF+2.82) | 0.35 |
| 16 (BAF+0.59) | 0.39 | 32 (BAF+2.97) | 0.16 |

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 = 5.062 MW, CPR = 1.35<br>Active Fuel Length = 3.048 m / 120.00 inches** | | |
|---|---|---|
| **Node<br>(m above BAF)** | **Axial Power Factor** | **Void Fraction** |
| 1 (BAF+0.08) | 0.38 | 0.00 |
| 2 (BAF+0.23) | 1.12 | 0.01 |
| 3 (BAF+0.38) | 1.47 | 0.13 |
| 4 (BAF+0.53) | 1.57 | 0.33 |
| 5 (BAF+0.69) | 1.55 | 0.48 |
| 6 (BAF+0.84) | 1.49 | 0.58 |
| 7 (BAF+0.99) | 1.42 | 0.65 |
| 8 (BAF+1.14) | 1.35 | 0.70 |
| 9 (BAF+1.30) | 1.28 | 0.74 |
| 10 (BAF+1.45) | 1.20 | 0.76 |
| 11 (BAF+1.60) | 1.13 | 0.79 |
| 12 (BAF+1.75) | 1.06 | 0.81 |
| 13 (BAF+1.91) | 0.96 | 0.82 |
| 14 (BAF+2.06) | 0.89 | 0.84 |
| 15 (BAF+2.21) | 0.82 | 0.85 |
| 16 (BAF+2.36) | 0.73 | 0.86 |
| 17 (BAF+2.51) | 0.60 | 0.87 |
| 18 (BAF+2.67) | 0.48 | 0.88 |
| 19 (BAF+2.82) | 0.35 | 0.88 |
| 20 (BAF+2.97) | 0.16 | 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 limiting stability condition in the ESBWR normal operating region is at the rated power 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 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) at the Peak Hot Excess (PHE) reactivity point, 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 is analyzed by perturbing the inlet flow to the group of 16 bundles while maintaining constant power. The calculation is performed at BOC and MOC conditions because this is limiting for channel hydrodynamic stability for the initial core.

**Core wide Stability**

Core stability is evaluated at MOC condition. The calculations are made with the 3-D kinetics model interacting with the thermal hydraulics parameters. The response to a pressure perturbation in the steam line is 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 BOC, MOC and EOC conditions. The channel decay ratios are the highest at BOC and MOC because of the bottom peaked axial flux shape. The decay ratios 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 BOC 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 evaluated at BOC and MOC conditions because of the bottom peaked axial power shapes. The super bundle decay ratios as tabulated in Table 2.2-2 meet the design goal of approximately 0.4.

The core decay ratio was evaluated 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 ratios 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 BOC conditions as an example. A total of 59 trials was 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 and an exposure point near the PHE conditions as an example. As for channel stability, a total of 59 trials was 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 in Reference 2.2-1 at rated power and flow and MOC conditions because regional mode oscillation at these conditions is limiting. A total of 59 trials was 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 limiting channel group from the transient response. This constitutes one trial in the Monte Carlo process. An OSUTL95/95 results from applying the conservative bias and uncertainty determined from the calculated Monte Carlo distribution, and is given in Table 2.2-3.

#### 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 criteria are satisfied for the ESBWR core. The demonstration of stability margins has been performed for an initial core based on the GE14E 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 OSUTL95/95 value for the regional decay ratio, which is limiting compared with other stability modes (channel and core wide), is compared against the design limits in Table 2.2-3. It should be noted that the decay ratio uncertainty determined for initial core is equivalent to the uncertainty documented for the equilibrium core in Reference 2.2-1.

### *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 a heat up rate at the thermal stress limit of 55.5°C/hr (100°F/hr). 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 the 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 that the trajectory of power closely follows the 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 ~ 55ºC/hour (~ 100 ºF/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 withdrawn, 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 caused by flashing in the chimney region (Figure 2.2-14). 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.

Figures 2.2-15a and 2.2-15b show core voiding in the highest power bundles. Vapor generation begins at the top of some 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 Bounding Heatup Rate

Effects of heatup rates higher than the thermal stress limit of 55.5 °C/hr (100 °F/hr) have been studied to establish a bounding heatup rate that will not be exceeded during the low pressure (cold) startup of ESBWR. A number of simulations were done along the lines described in the previous section and employing discreet heatup rates varying from moderate to extreme values with respect to thermal stress limit. The main features common among those simulations are the following:

- The core analyzed is the same one employed in the previous section.
- Neutronic feedback is simulated.
- MSIV's are kept open over the entire startup period.
- Steam line, SJAE (Steam Jet Air Ejector) and Steam Seal Glands warm-up loads are taken into account. This simulates BOP.
- Rod pulling sequence employed is such that the high worth control rods are pulled at an early startup stage (when the coolant began to change phase).
- Application of a power peaking multiplication factor (PIRT46) to the hot channel[*] was made to cause rapid reduction in the 'minimum CPR'; a consequence of local power peaking.

The above measures were taken to push the overall reactor performance closer to stability limits.

A number of TRACG cases, at different heat-up rates, were prepared. These are listed in the Tables 2.2-4 and 2.2-5.

Cases listed in the Table 2.2-4 and accompanying plots (Figures 2.2-20 thru 2.2-59) follow the conventions and considerations given below:

(1) Case names: A case name (say '90MW_242_no_pirt') indicates; expected average power (90MW), hot channel number (242) and status of 'PIRT46' parameter (no_pirt, meaning no adjustment of channel power).

(2) Hot channel no. '242': This is a single bundle channel located in a core region of high enrichment. In 3 out of 4 cases in the Table 2.2-4, 'PIRT46' parameter was applied to '242'. A discussion on this follows.

(3) Parameters listed in the Table 2.2-4 were evaluated over a *defined span of time*; beginning from average core power of ~ 90 MW and ending when the steam dome pressure attained the turbine roll pressure ~ 7 MPa (1015 psia).

(4) Plots have been arranged to facilitate comparison among different cases; they are arranged in <u>ascending order of heat-up rate.</u>

(5) These cases, with heat-up rate varying, from 38 K/hr (68 °F/hr) to 128 K/hr (230 °F/hr), were prepared to help identify the region of interest wherein the upper limit was to be sought.

[*] **Hot channel:** A set of bundles, consisting of 1 or multiple bundles, having the highest radial peaking factor.

Cases listed in the Table 2.2-5 and accompanying plots (Figures 2.2-60 thru 2.2-99), follow the conventions and considerations explained below:

(1) 'PIRT46' parameter was not applied to the hot channel.

(2) Parameters were evaluated over a span of time ranging from 1,000 – 3,000 seconds (~17 – 50 minutes). The reasons for this selection are discussed in the next section.

(3) Plots are arranged in ascending order on heat-up rate.

(4) These cases were prepared to seek the 'limiting' heat-up rate by performing a search in small steps of reactivity over a domain of time from 1,000 to 3,000 seconds.

Table 2.2-4 shows ESBWR low-pressure startup simulations using TRACG. Reactor heat-up rate, in each of these cases, was kept constant until rated pressure of 7 MPa was attained. The time range over which these cases were simulated varied from 8,000 – 21,000 seconds, but the parameters listed there were evaluated over a span of time; beginning from average core power of ~ 90 MW and ending with the approach of dome pressure to turbine roll pressure ~7 MPa (1015 psia).

Table 2.2-5 summarizes the search for limiting heat-up rate. Table 2.2-4 and the corresponding plots suggest the bounding heat-up rate occurs beyond 140 MW (see drop in CPR and increasing oscillations in the core parameters). The parameters listed in the Table 2.2-5 were evaluated over the time range of 1,000 – 3,000 seconds, only.

Review of plots and information listed in the Table 2.2-4, for the heat-up rates of 38, 57, 81 and 123 K/hr (68, 103, 146 and 221 °F/hr) leads to the following observations:

(1) The trace of core flow has two separated regions of noise:

- The first one appears at the initiation of void generation (mainly in the hot regions) in the core and their subsequent collapse in the subcooled chimney. It lasts until dome pressure reaches ~ 0.25 MPa (36 psia). On the time axis this corresponds to 1,000 – 3,000 seconds.
- The second one appears when sustained, though fluctuating, vapor formation is initiated in separators. Those fluctuations are eventually leveled off and a steady void fraction is established at the top of the separators. On time axis appearance of this phenomenon depends on heat-up rate; the higher it goes the sooner it appears. It lasts about 4,000 seconds.

(2) With the exception of the extreme case (123 K/hr, 191 MW), no voiding appears in the separator region within first 3,000 seconds.

(3) Inspection of power trace for '200MW_242_pirt_3' (the extreme case), shows appearance of noise after step increase in power. On this ground alone this case was dropped from consideration.

(4) There is some low amplitude noise, riding the power trace for a duration of ~ 4,000 seconds, appearing together with void fluctuations in the separators.

(5) Study of normalized changes in the core flow, over entire startup time, show the highest amplitude noise occurs between 1,000 – 3,000 seconds. That is when the RPV pressure is less than 0.25 MPa and coolant phase change begins to take place.

In view of above observations the limiting heat-up rate was searched between 80 K/hr (144°F/hr) and 120 K/hr (216 °F/hr) by watching sensitive parameters over the 1,000 – 3,000 interval. The last 4 TRACG cases (case-1 thru case-4) were run to make this search. A review of the tabulated information and plots for these cases reveal:

(1) Fluctuations in the power trace, beyond 'case-2', are ~ 10% of the power. Though, any sustained oscillations of comparable magnitude are absent in the steam dome pressure.

(2) Core flow exhibits noise due to void formation in the hot channel with no matching vapor formation seen in the separators for 'case-1' and 'case-2'. Reversal can also be seen in the 'hot channel exit flow'. These are all indications of vapor condensation in the subcooled chimney, which in turn indicates excessive and pre-mature heat generation in the hot core regions.

(3) CPR plots indicate progressive (from 'case-1' thru 'case-4') shrinking of thermal margins, however, it can be observed that the safety limit is still a large distance away.

(4) In concert with the overall core flow, 'peripheral channel exit flow' is oscillating between 3,000 – 6,000 seconds. However the peripheral bundles are always in up flow and large margin to safety limits are maintained throughout the transient.

(5) Application of 'PIRT46' to the hot channel was made to cause rapid reduction in the 'minimum CPR'. It has little impact on the average core power as applied to a single-bundle channel (see cases with power ~140 MW). However, trend indicates, that high local peaking can limit startup heat-up rate to lower values. Beyond this observation, use of 'PIRT46' per se was of no other consideration in this study.

To conclude, simulation of low pressure ESBWR startup has been carried out by employing discreet heat-up rates over a range; 38 K/hr (68 °F/hr) - 128 K/hr (230 °F/hr), i.e. ~ 91 – 191 MW. After examining stability-sensitive parameters, it is determined, that a maximum heat-up rate twice-the-magnitude of limit placed by thermal stress considerations, is acceptable. This gives a startup heat-up rate in terms of MW of ~ 145, or ~ 110 K/hr (198 °F/hr). *The startup procedures to be employed in operating ESBWR plants are to use a heat-up rate that will exclude initiation of early vapor formation in the core, ensuring a smooth ascension to operating pressure and power.*

### 2.2.4 Stability Effects of Xenon Transient

The possible effects of Xe transient on the stability of ESBWR during startup from cold conditions until attainment of turbine roll pressures are examined. A Xe transient occurs following a change in the reactor power or after a recovery from a scram. Distribution of Xe in the core depends on the power history prior to a restart effort. After a startup when the core power has increased to a significant level (>10% of the core power), shallow rods are withdrawn, if necessary, to compensate for Xe effects (on the power shape), in the regions where its concentrations are high. Subsequently, burn-up of Xe causes local power peaking (increase in RPF) in the core zones where its (Xe) concentration was high. That power peaking is controlled by re-inserting shallow control rods.

In the subsequent sections it is demonstrated that in ascension to turbine roll pressure, transient Xe poses no stability challenge as it is achieved at ~ 2% of the rated power level.

The effect of burn-up of transient Xe is to cause local power peaking in a thermal hydraulic-channel (consisting of one or multiple bundles). TRACG does not calculate time varying Xe, but, in the 'PIRT46' parameter, it provides a capability to increase or decrease local power peaking, simulating Xe effects. Effects on stability margins can then be estimated.

A study using PANACEA code, evaluating startup conditions for the ESBWR Initial Core, MOC, shows that a RPF of 8 is limiting and conservative in ascension to turbine roll pressure. That holds for both situations, vis-a-vis, presence of transient Xe.

The startup simulation method used and the ESBWR core analyzed for this study are the same described in section 2.2-3, albeit, with some channel regrouping done to well scrutinize the hottest region.

Like previous studies, 269 control rods were divided into 10 groups. Rod pulling sequence employed was such that the <u>high worth control rods were pulled at an early startup stage</u> (when the coolant begins to change phase). *Together with 'PIRT46' these measures were taken to push the overall reactor performance closer to stability limits.*

Three TRACG cases were prepared, which are listed in the Table 2.2-6; corresponding plots of interest were also made (Figures 2.2-100 thru 2.2-129). Entries in the Table 2.2-6 and attached plots follow the following conventions and considerations:

(1) Case names: A case name (say '90MW_242_no_pirt') indicates; expected average power (90 MW), hot channel number (242) and status of 'PIRT46' parameter (no_pirt, meaning no adjustment of channel power).

(2) Hot channel # '242' is a single bundle channel located in a core region of high enrichment. Hot channel # '241' is a 16 bundle channel. Both of these channels are located in the same core zone.

(3) Case '95MW_241_pirt_2.5' was run by almost doubling the average power of 'channel 241'. The purpose of raising power of all 16 bundles was to be more conservative.

(4) Stability related parameters were evaluated over a defined span of time; beginning from average core power of ~ 90 MW and ending when the steam dome pressure approached the value at which the turbine roll evolution begins (6.6 MPa (960 psia)).

For each case, the set of plots (related to ESBWR startup) presented in the Appendix 4D of ESBWR DCD (Rev 5) have also been made for this study are provided. Plots are arranged to facilitate comparison among different cases; they are arranged in the same order as the case entries in Table 2.2-6.

Table 2.2-6 provides stability related parameters from ESBWR low-pressure startup simulations using TRACG. 'PIRT46' factor was applied to hot channels to increase local peaking, imitating an effect caused by the burn-up of Xe in high concentration regions. The TRACG cases were run for 21,000 seconds (5 hours and 50 minutes), but the listed parameters were evaluated over a span of time; beginning from average core power of ~ 90 MW and ending with the approach of dome pressure to the turbine roll value of ~ 6.6 MPa (960 psia).

Review of plots and information listed in the Table 2.2-6 leads to the following observations:

(1) Steam dome pressure shows steady and smooth ascension to the operating value after running for ~ 20,000 seconds (~ 5 and-a-half hours) at an average power of 91 MW, which is ~ 2% of the rated power.

(2) With RPF varying between 5 and 11, CPR plots for hot and peripheral channels, show good margins to safety limits during the startup transient.

(3) The largest value of RPF (over the entire startup domain) occurs after step increase in power. This is understandable on physical grounds, since in the early phase of startup, in-core control rod densities are high, causing power to be derived from a smaller fraction of fuel.

(4) Trace of core flow has two separated regions of noise:

- The first one appears at the initiation of void generation (mainly in the hot regions) in the core and their subsequent collapse in the subcooled chimney. It lasts until dome pressure reaches ~ 0.25 MPa (36 psia). On the time axis this corresponds to 1,000 – 3,000 seconds.

- The second one appears when sustained, though fluctuating, vapor formation is initiated in separators. Those fluctuations are eventually leveled off and a steady void fraction is established at the top of the separators. On time axis this phenomenon appears between 6,000 to 10,000 seconds.

In ESBWR startup transient, the rated pressure is achieved at low power levels (~ 2% of the core power). Neutron flux at these levels does not cause any considerable burning of Xe. Furthermore, response of the plant was simulated by raising hot-channel radial peaking to a high value of 11, which is beyond the *conservative* limit of 8 determined by a PANACEA analysis, considering control rod density and transient Xe. No stability challenges were observed.

### 2.2.5 Defense-In-Depth Stability Solution

This discussion in the DCD for equilibrium core also applies to the initial core.

### 2.2.6 References

2.2-1 GE Nuclear Energy, B.S.Shiralkar, et al, "TRACG Application for ESBWR Stability Analysis," NEDE-33083P-A, Supplement 1, Revision 1, January 2008.

2.2-2 GE Nuclear Energy, "ESBWR Design Control Document" 26A6642AP, Chapter 4.

**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)* | 9,752<br>(77.40 Mlbm/hr) | 9,734<br>(77.26 Mlbm/hr) | 10,040<br>(79.68 Mlbm/hr) |
| Feedwater temperature (ºC) | 216<br>(~420 ºF) | 216<br>(~420 ºF) | 216<br>(~420 ºF) |
| Narrow range water level (m) | 20.7 (67.9 ft) | 20.7 (67.9 ft) | 20.7 (67.9 ft) |
| Feedwater flow (kg/s)* | 2429<br>(19.28 Mlbm/hr) | 2428<br>(19.27 Mlbm/hr) | 2428<br>(19.27 Mlbm/hr) |
| Core inlet subcooling* (ºC) | 16.9<br>(30.4 ºF) | 16.9<br>(30.4 ºF) | 16.2<br>(29.2 ºF) |
| Steam dome pressure (MPa) | 7.17<br>(1040 psia) | 7.17<br>(1040 psia) | 7.17<br>(1040 psia) |
| ICPR* | 1.41 | 1.44 | 1.45 |
| Hot Bundle Power (MWt)* | 5.04 | 5.07 | 5.06 |
| Hot Bundle flow (kg/s)* | 6.91<br>(54.84 klbm/hr) | 7.04<br>(55.87 klbm/hr) | 8.8<br>(69.84 klbm/hr) |

* Calculated parameter

** Analysis results obtained with core loading and pressure drop as specified in NEDC-33326P, Rev0.

**Table 2.2-2**

**Baseline Stability Analysis Results**

| Mode | BOC | MOC | EOC |
|---|---|---|---|
| | Decay Ratio | Decay Ratio | Decay Ratio |
| Channel | 0.24 | 0.21 | |
| Superbundle | 0.28 | 0.26 | |
| Core | | 0.30 | |
| Regional | 0.50 | 0.50 | 0.20* |

* Non-limiting analysis results obtained with core loading and pressure drop as specified in NEDC-33326P, Rev 0.

**Table 2.2-3**

**Statistical Stability Analysis Results**

| | Decay Ratio - Baseline Result | Bias and Uncertainty | Decay Ratio – One Sided Upper Tolerance Limit (95/95) | Decay Ratio – Design Limit |
|---|---|---|---|---|
| Regional | 0.50 | 0.20 | 0.70 | 0.8 |

**Table 2.2-4**

**Bounding Heatup Rate Studies**

| Case Name | Core Wide | | Hot Channel* | |
|---|---|---|---|---|
| | Average Heat-up Rate K/hr (°F/hr) | Average Power (MW) | Max. Power (kW) | Min. CPR |
| 90MW_242_no_pirt | 38.06 (68.5) | 91 | 479 | 7.2 |
| 110MW_242_pirt_3 | 57.03 (102.6) | 111 | 1,008 | 5.3 |
| 140MW_242_pirt_3 | 80.97 (145.7) | 141 | 1,821 | 2.6 |
| 200MW_242_pirt_3 | 122.97 (221.3) | 191 | 1,361 | 3.85 |

* A set of bundles, consisting of 1 or multiple bundles, having the highest radial peaking factor.

**Table 2.2-5**
**Bounding Heatup Rate Studies**

| Case Name | Core Wide | | Hot Channel* | |
|---|---|---|---|---|
| | Average Heat-up Rate K/hr (°F/hr) | Average Power (MW) | Max. Power (kW) | Min. CPR |
| Case_1 | 108.4 (195.1) | 140 | 652 | 6.8 |
| Case_2 | 115.6 (208.1) | 153 | 702 | 5.9 |
| Case_3 | 123.1 (221.6) | 168 | 791 | 4 |
| Case_4 | 128 (230) | 173 | 791 | 4 |

**Table 2.2-6**

**Stability Effects of Transient Xe**

| Case Name | Hot Channel* | | | | Core Wide |
|---|---|---|---|---|---|
| | Channel Number | Max. RPF | Max. Power (kW) | Min. CPR | Average Power (MW) |
| 90MW_242_no_pirt | 242 | 5 | 479 | 7.2 | 91 |
| 90MW_242_pirt_3 | 242 | 11 | 1,440 | 5.3 | 90 |
| 95MW_241_pirt_2.5 | 241 | 9 | 1,180 | 4.2 | 95 |

* A set of bundles, consisting of 1 or multiple bundles, having the highest radial peaking factor.

**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.

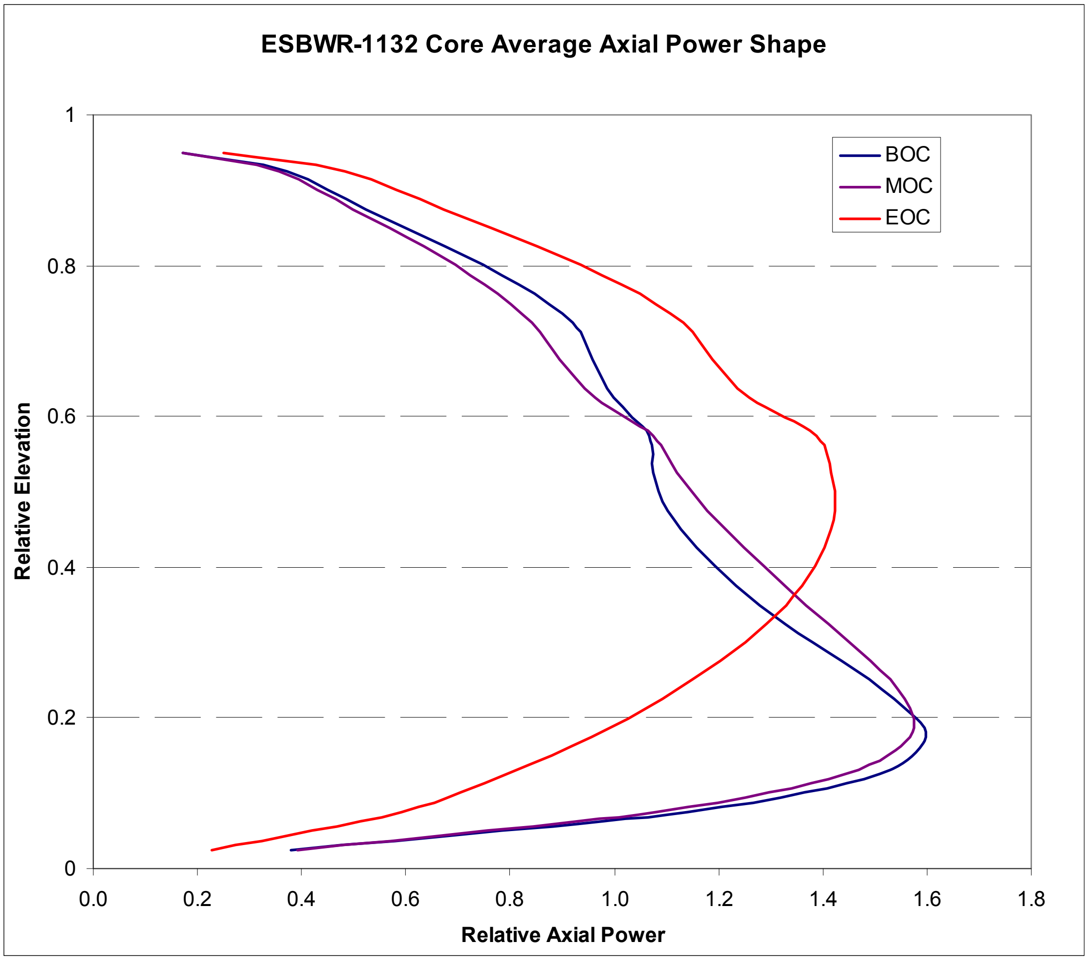



**Figure 2.2-3. Core Average Axial Power Shape at Different Exposures**

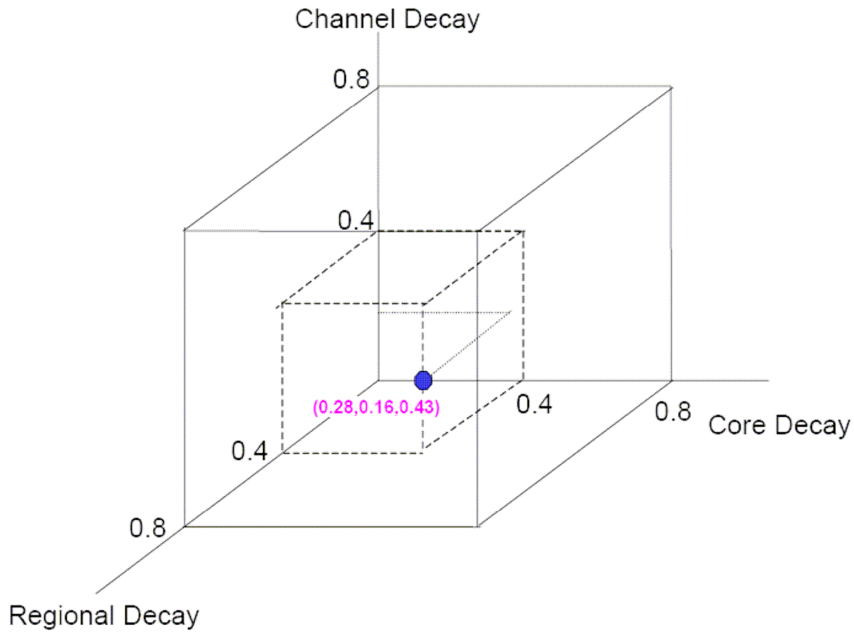



**Figure 2.2-4. Decay Ratio Results Compared to Design Criteria**

**Figure 2.2-5. Stability in Expanded Operating Map**

Data in the DCD applies to the initial core.

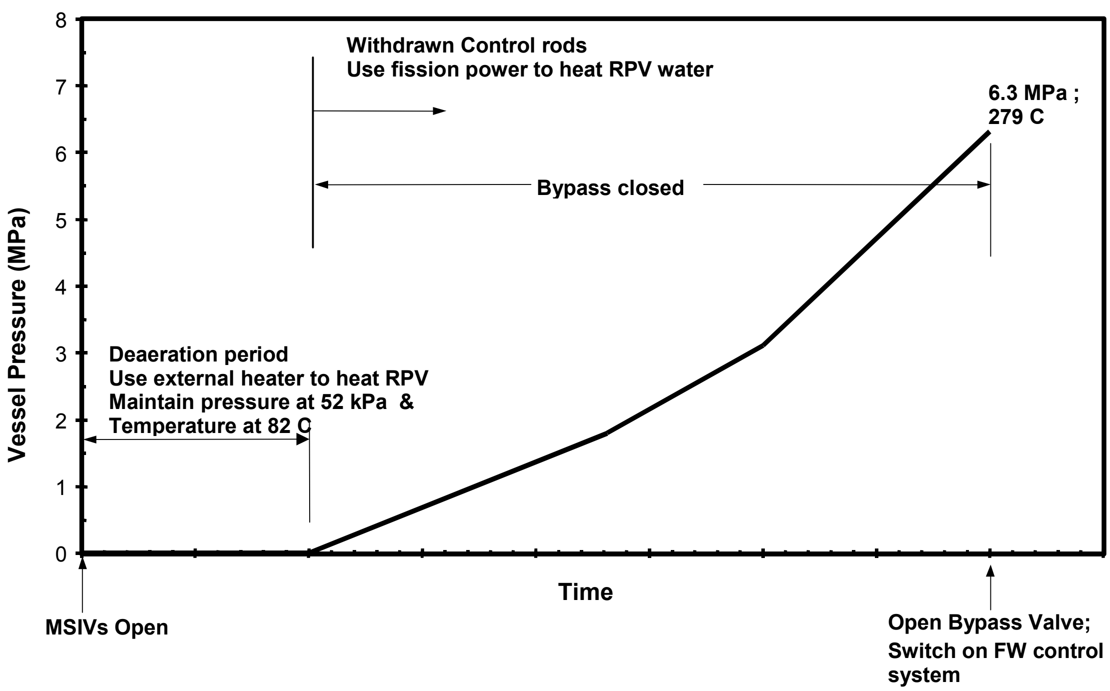



**Figure 2.2-6. ESBWR Startup Trajectory with MSIVs Open**

| 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**

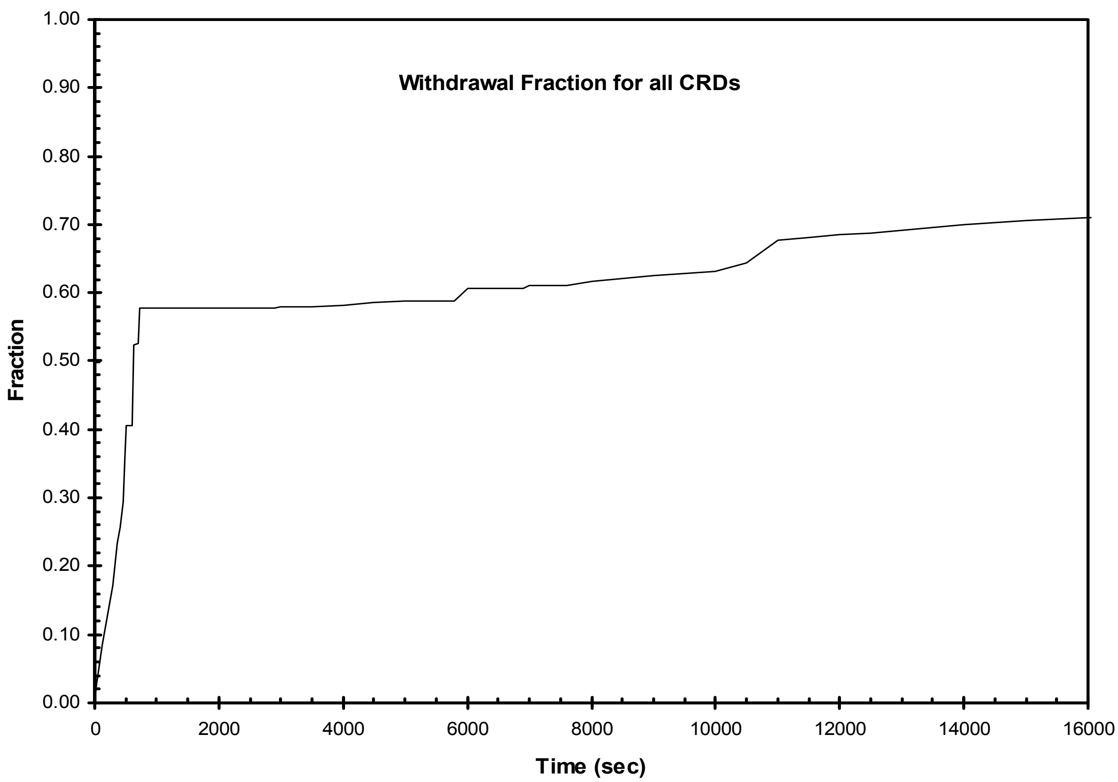



**Figure 2.2-8. Withdrawal Fraction for all Control Rods**

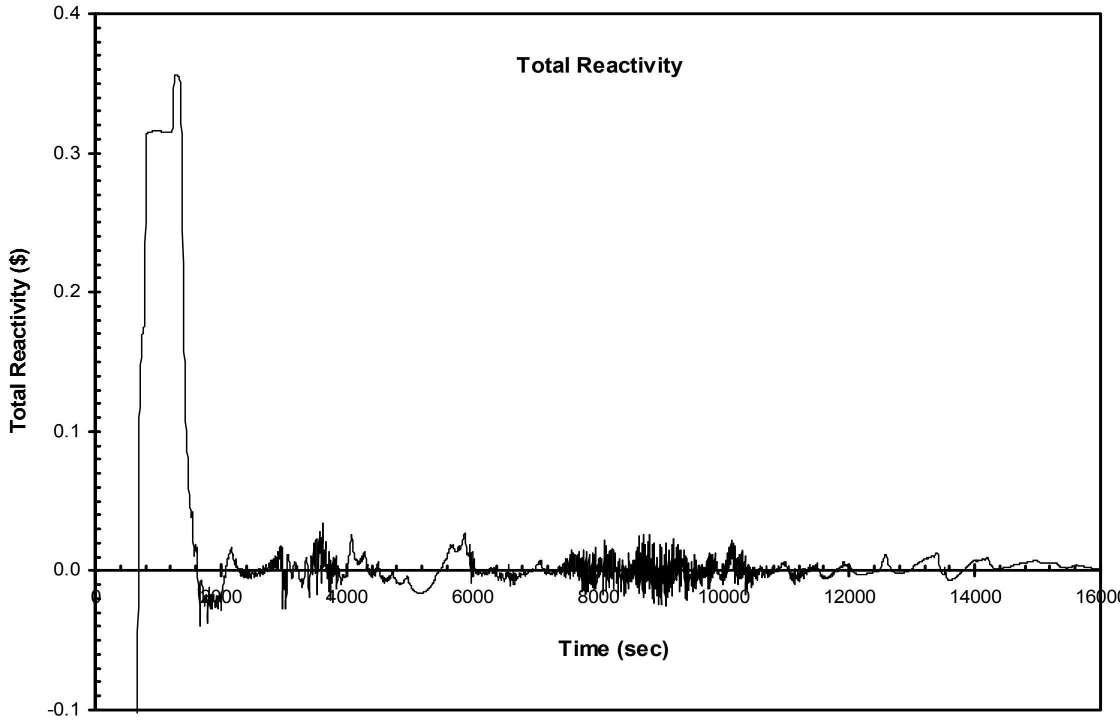



Figure 2.2-9. Total Reactivity

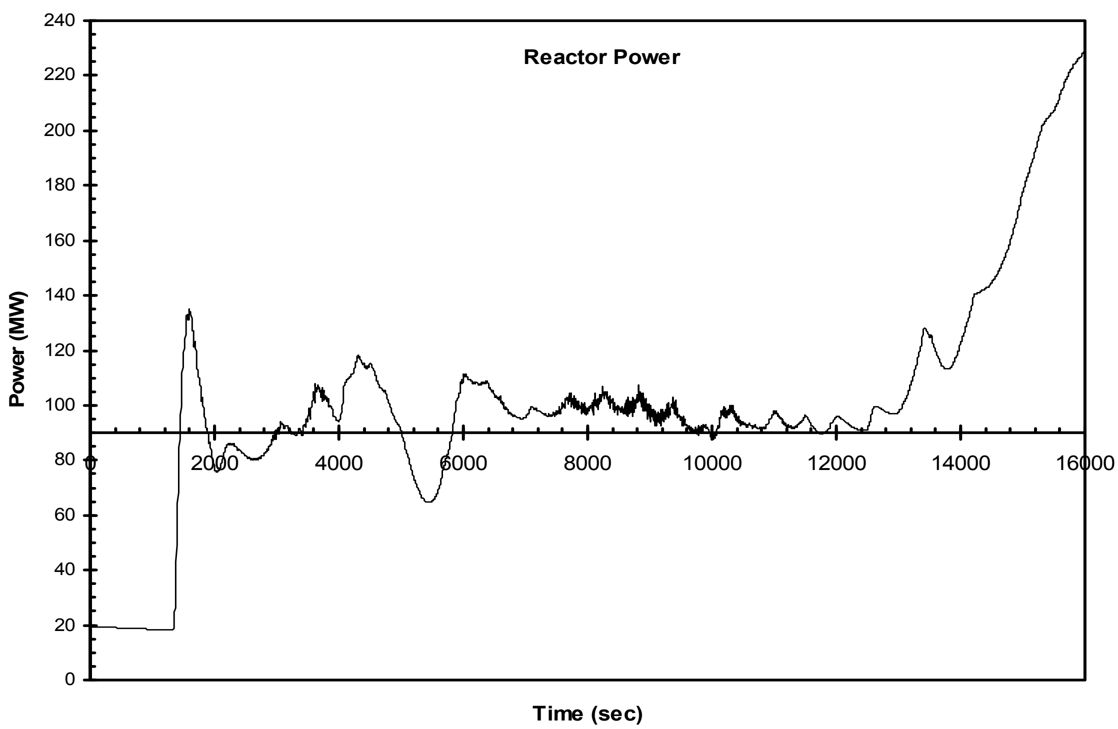



Figure 2.2-10. Reactor Power

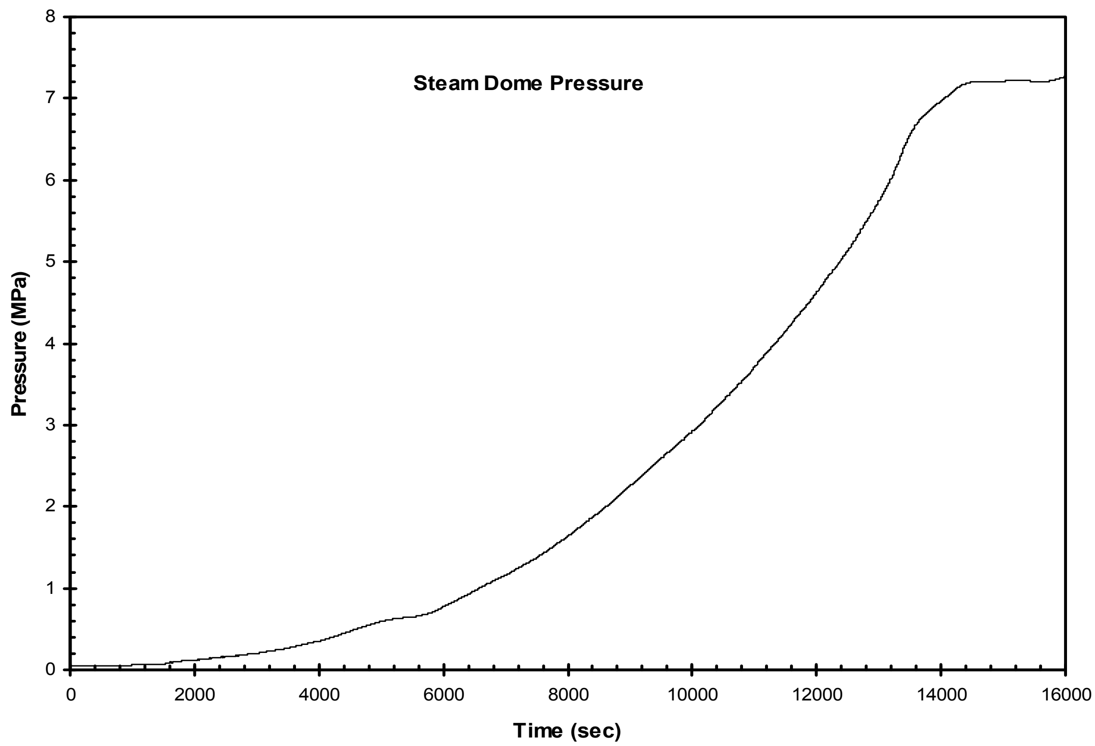



**Figure 2.2-11. Steam Dome Pressure**

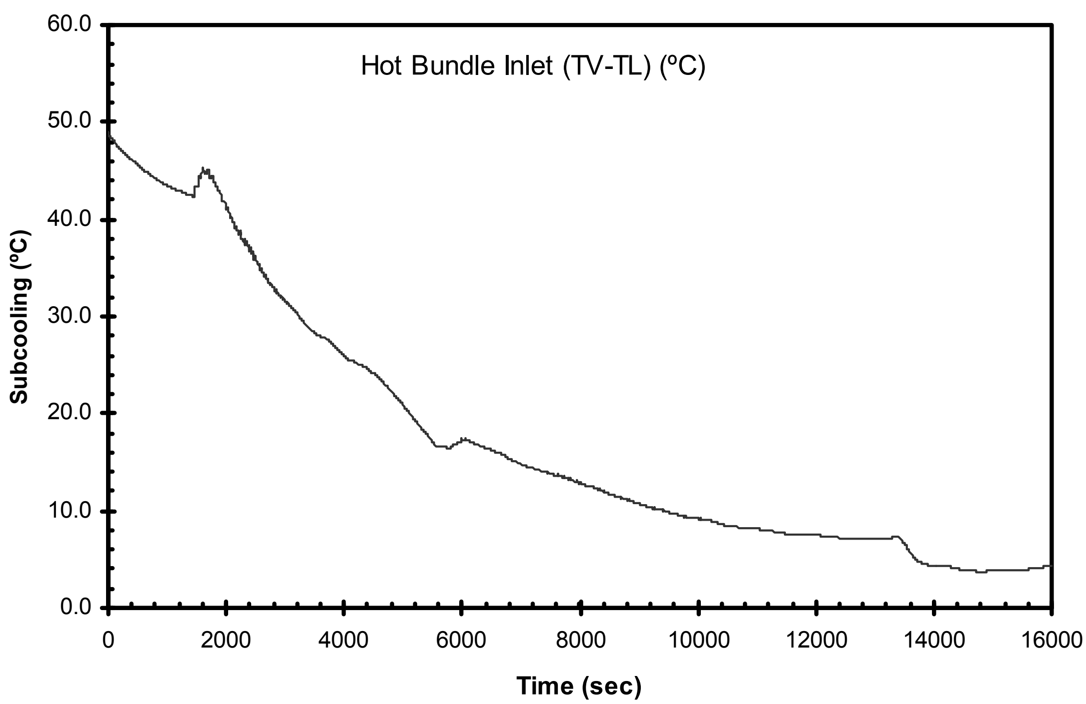



**Figure 2.2-12. Core Inlet Subcooling**

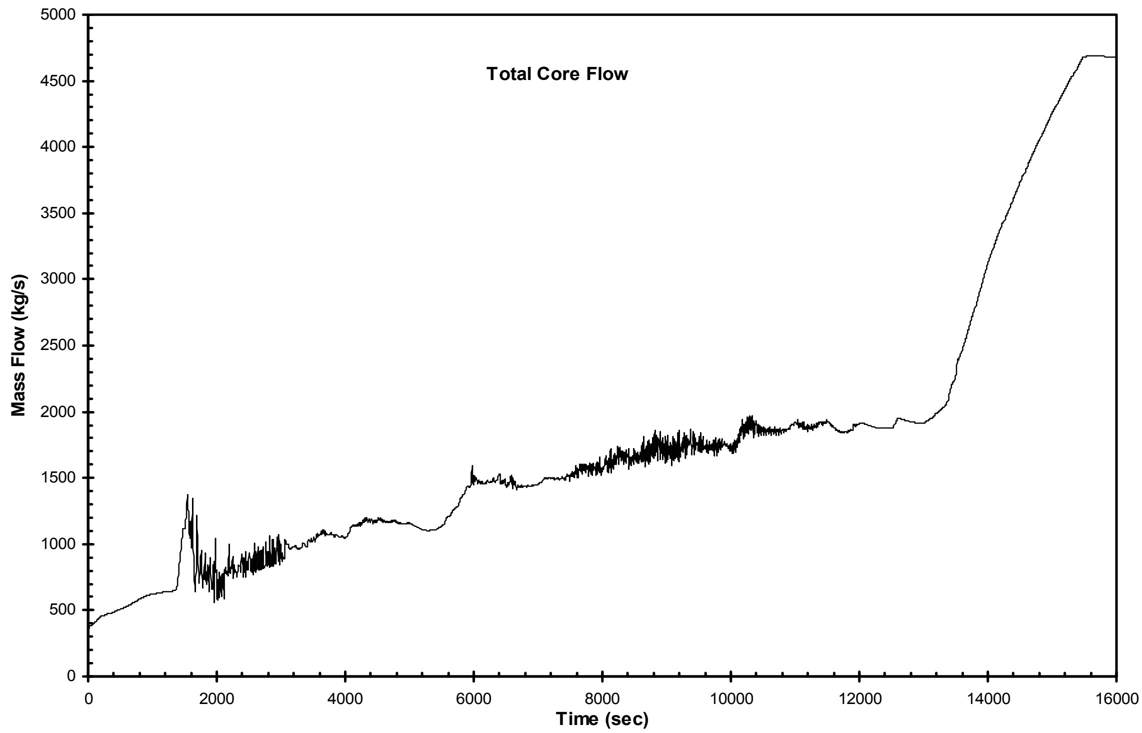



**Figure 2.2-13. Core Inlet Flow**

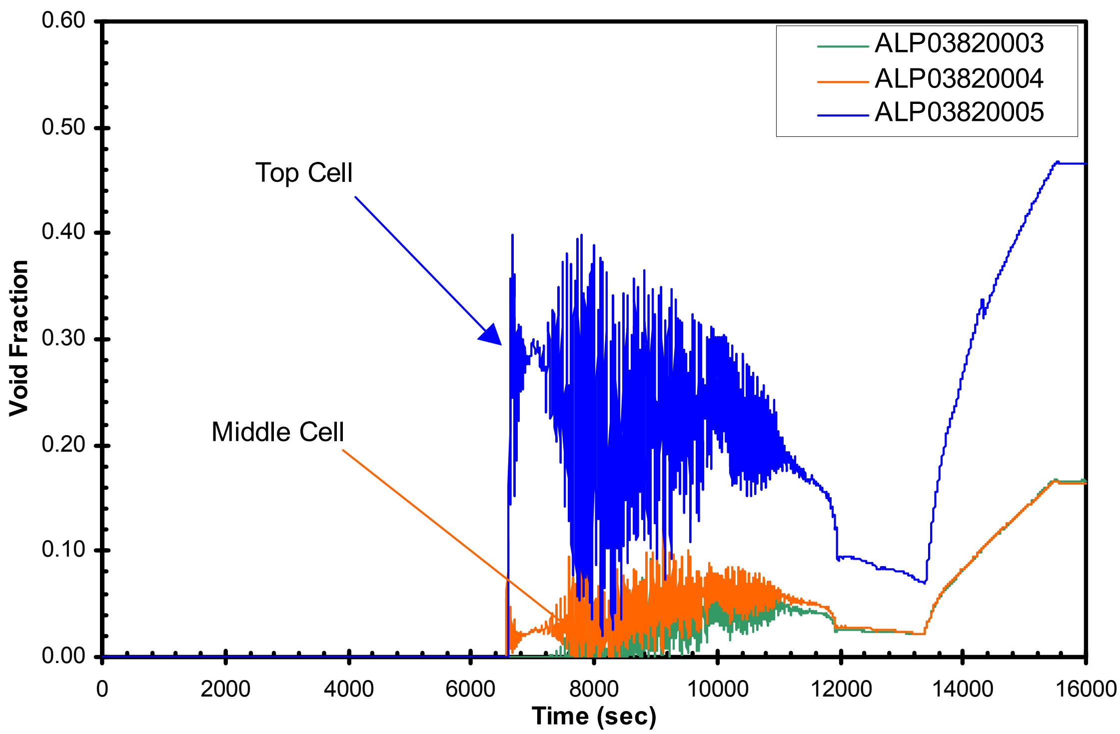



**Figure 2.2-14. Separator Void Fraction**

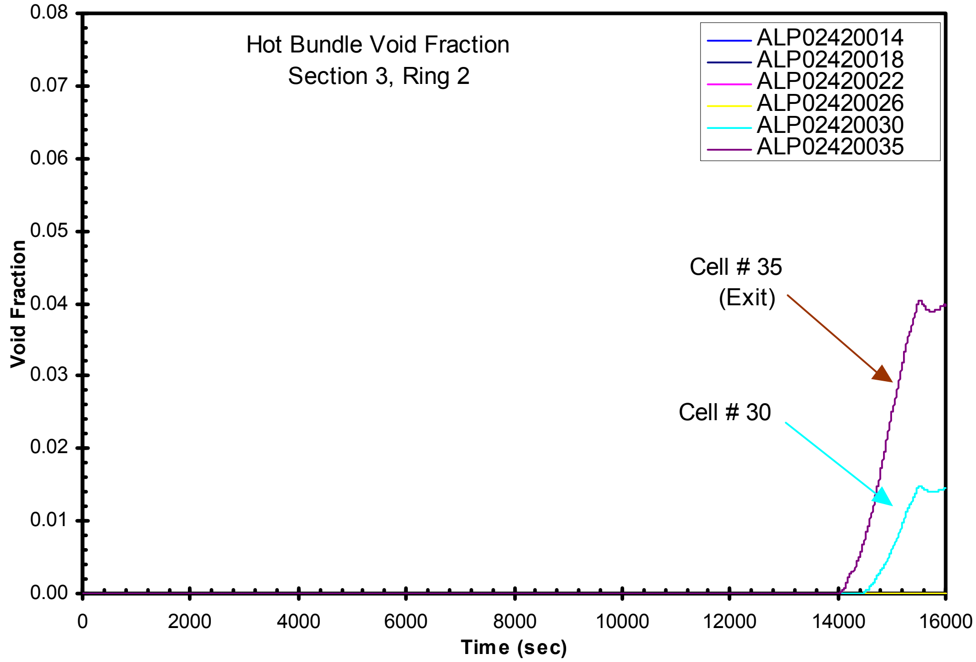



**Figure 2.2-15a. Hot Bundle Void Fraction**

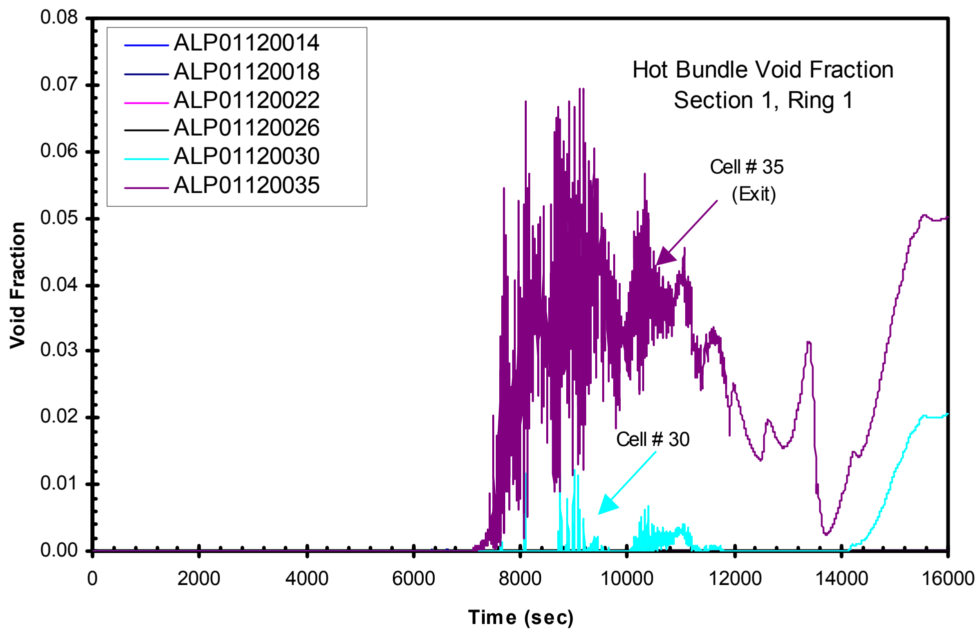



**Figure 2.2-15b. Hot Bundle Void Fraction**

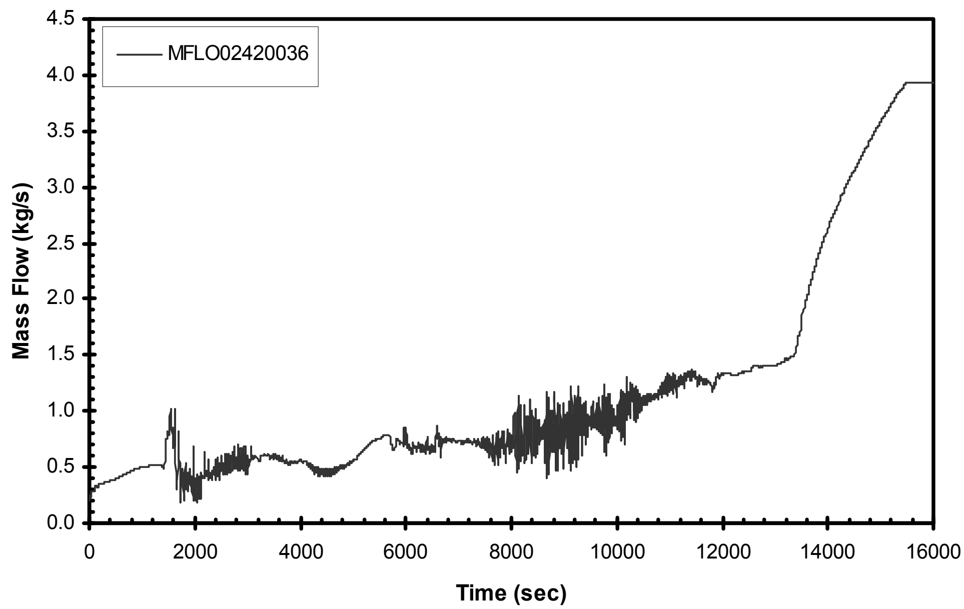



**Figure 2.2-16. Hot Bundle Exit Flow**

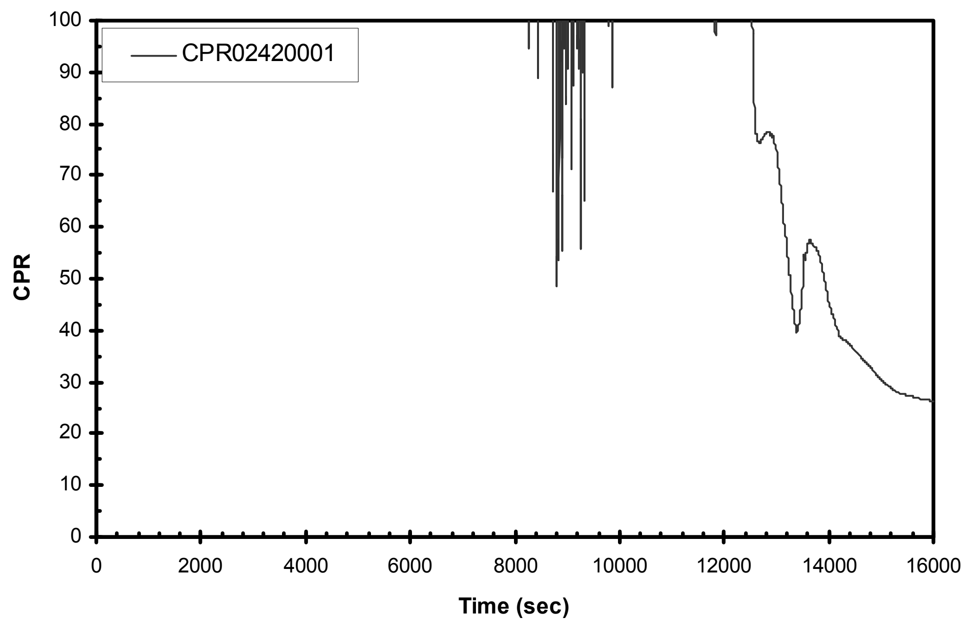



**Figure 2.2-17. Hot Bundle CPR**

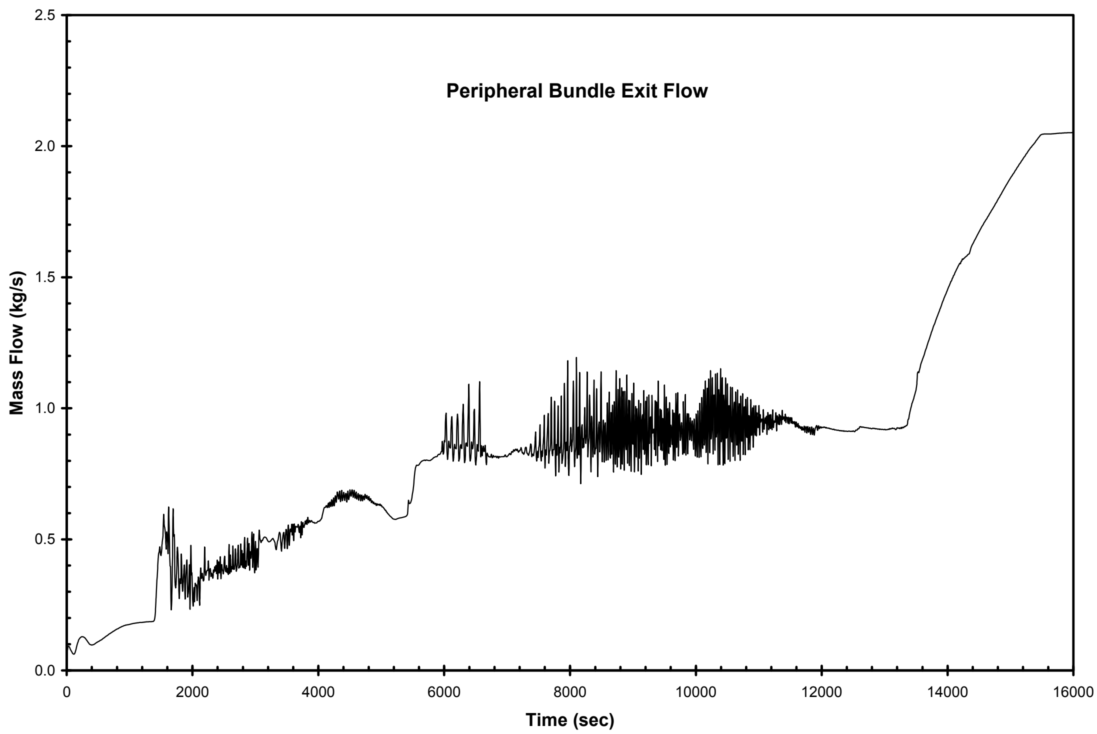



**Figure 2.2-18. Peripheral Bundle Exit Flow**

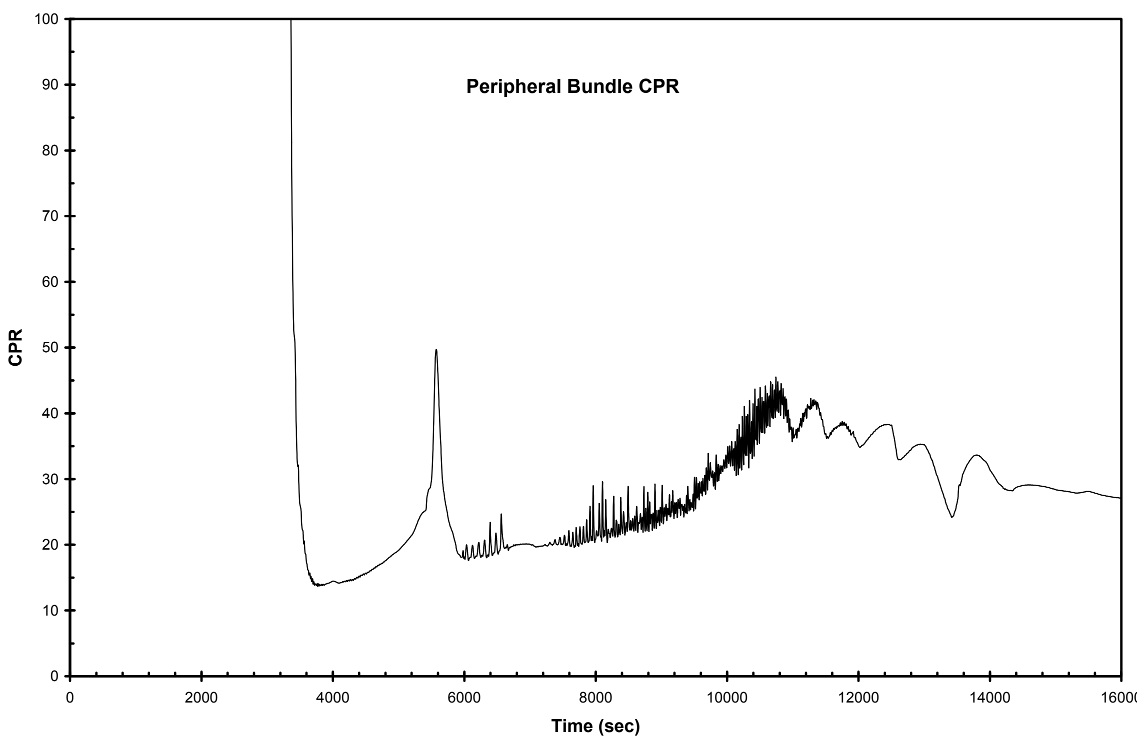



**Figure 2.2-19. Peripheral Bundle CPR**

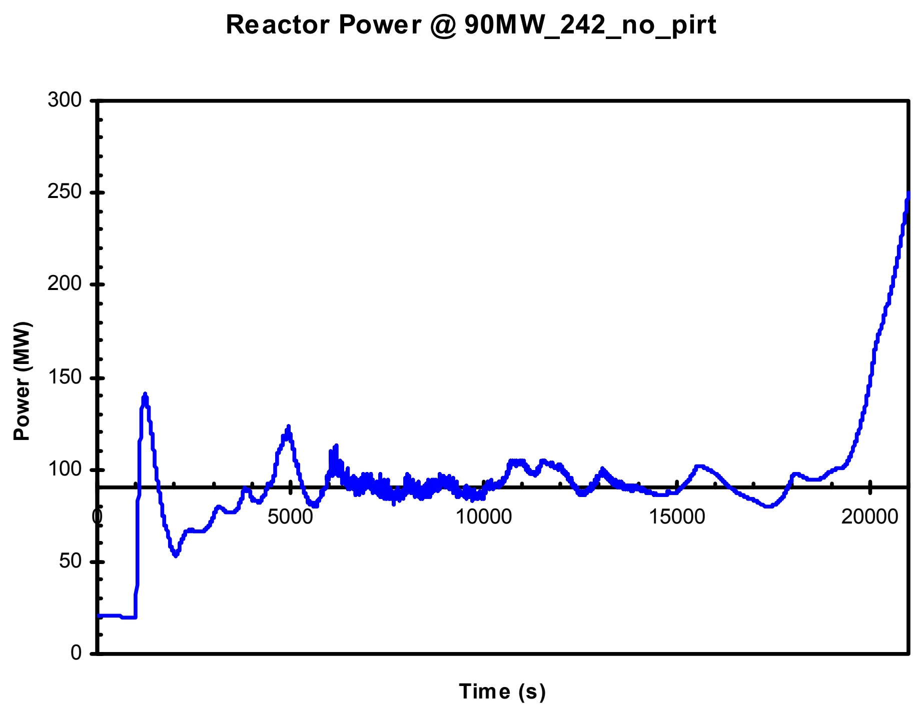



Figure 2.2-20. Reactor Power

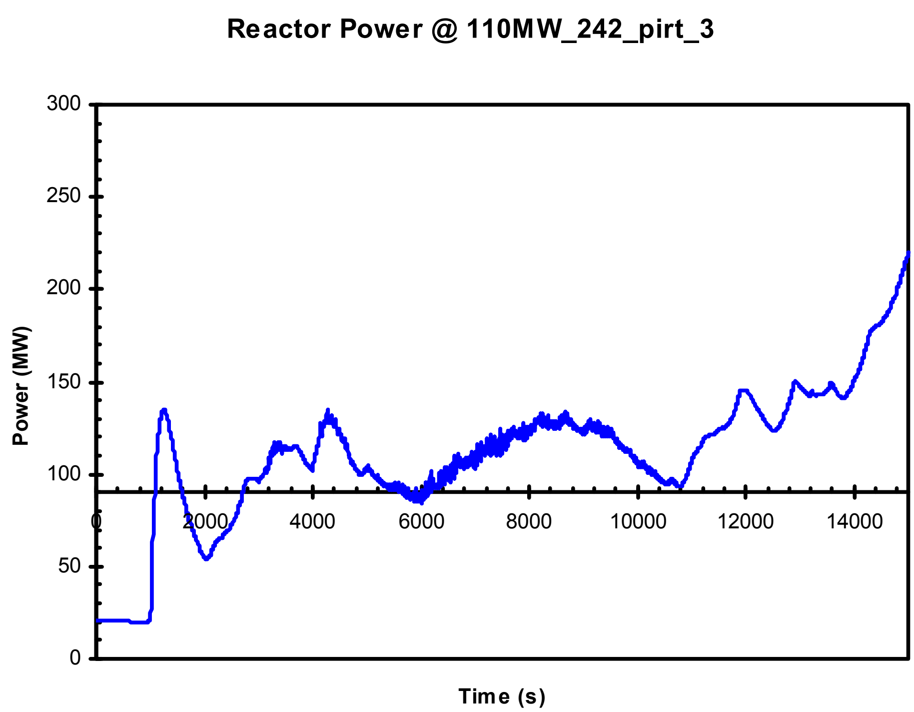



Figure 2.2-21. Reactor Power

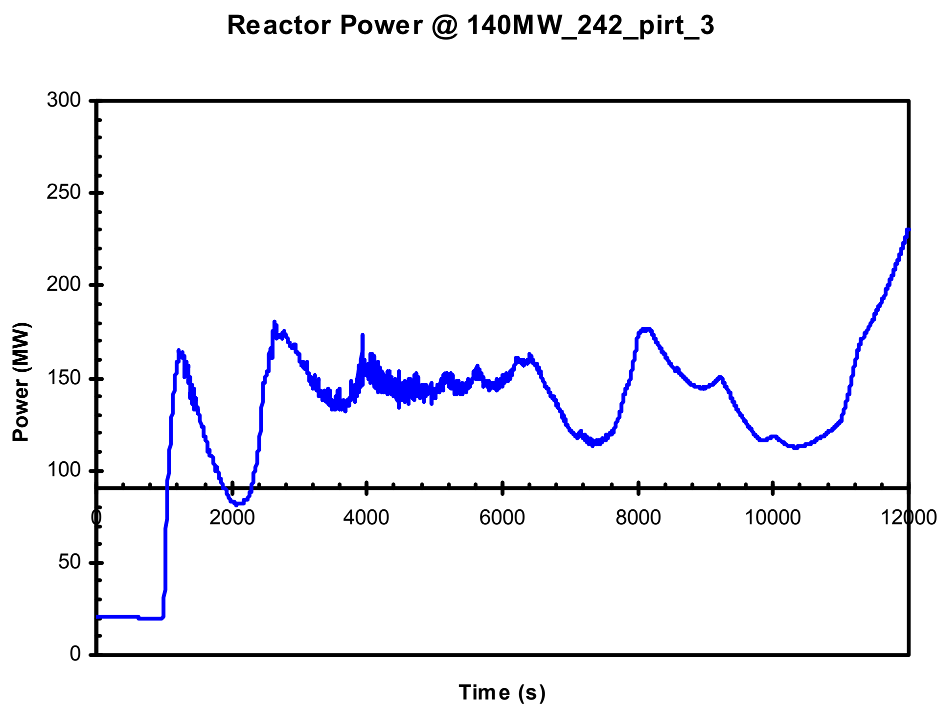



Figure 2.2-22. Reactor Power

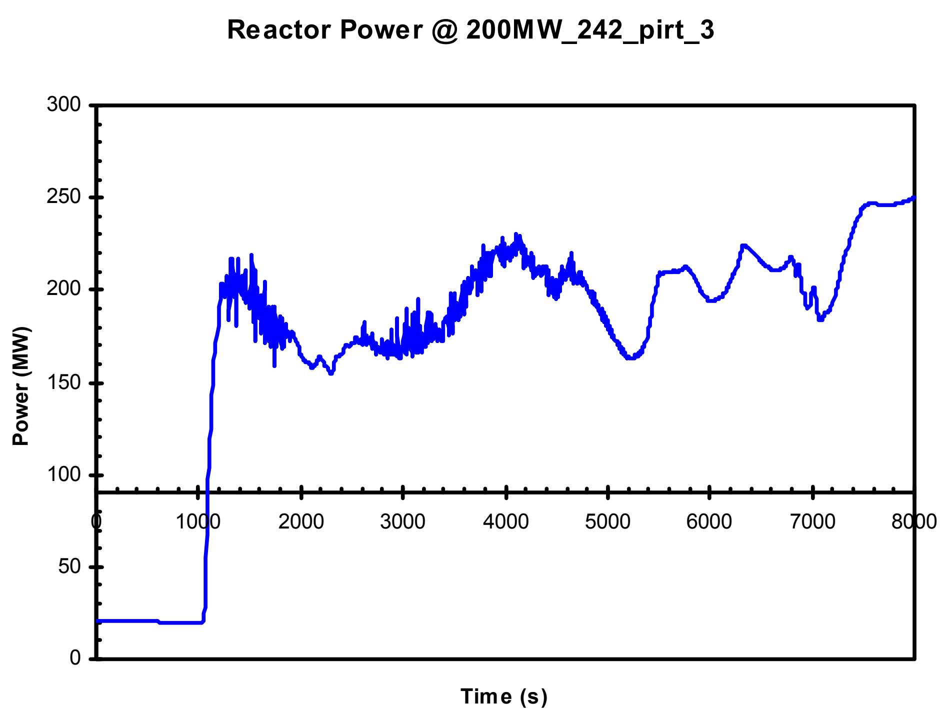



Figure 2.2-23. Reactor Power

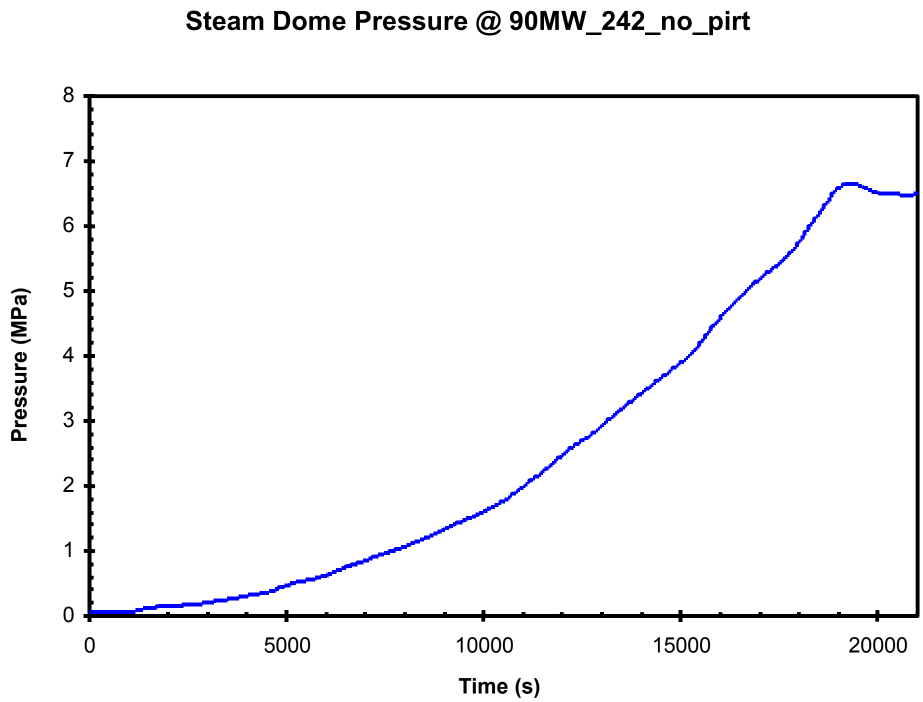



Figure 2.2-24. Steam Dome Pressure

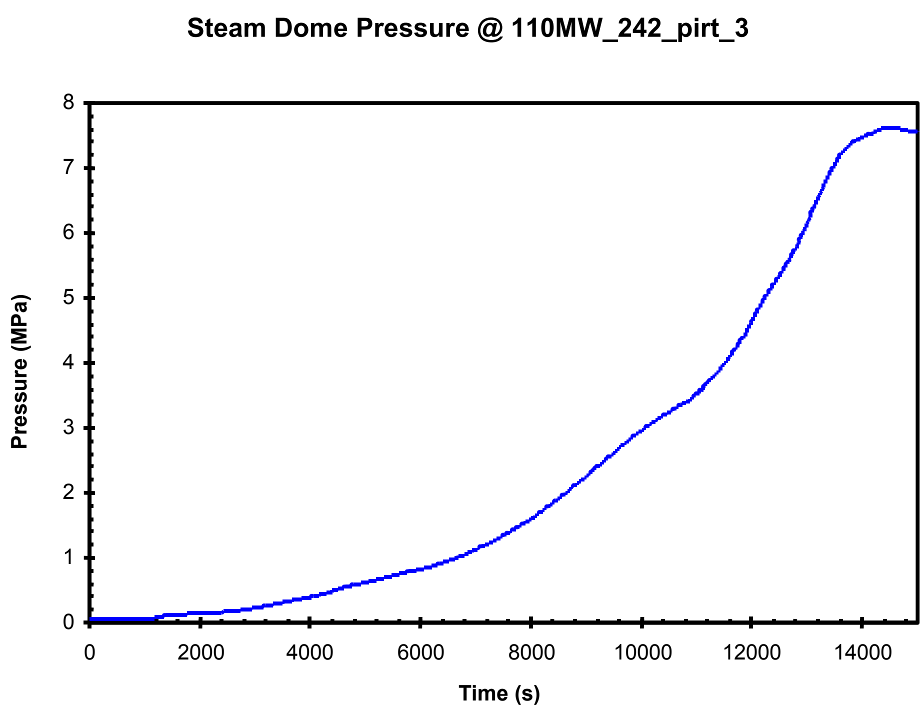



Figure 2.2-25. Steam Dome Pressure

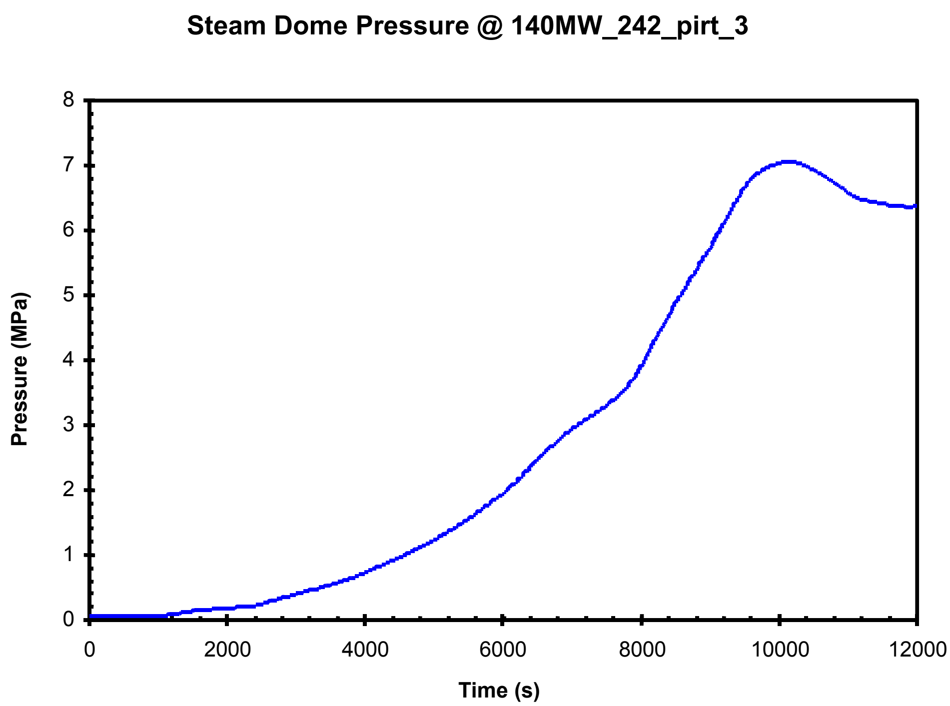



Figure 2.2-26. Steam Dome Pressure

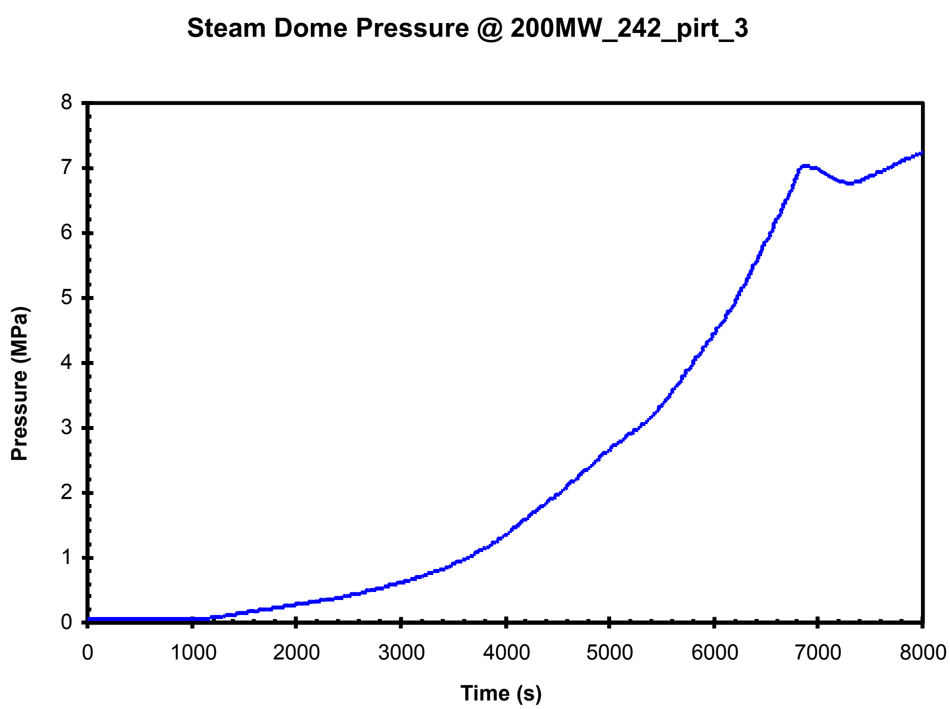



Figure 2.2-27. Steam Dome Pressure

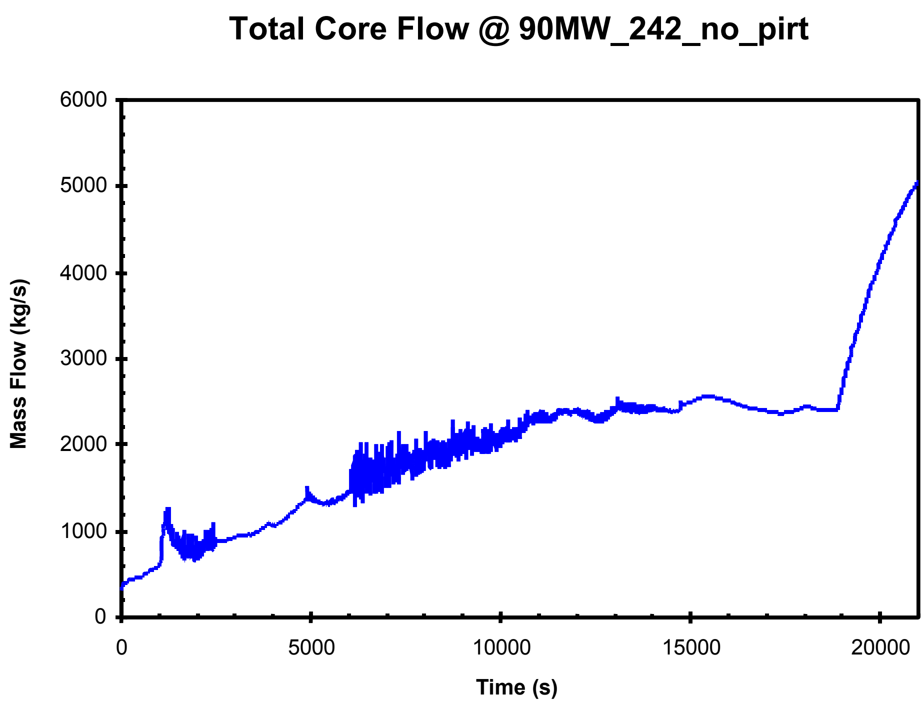



Figure 2.2-28. Total Core Flow

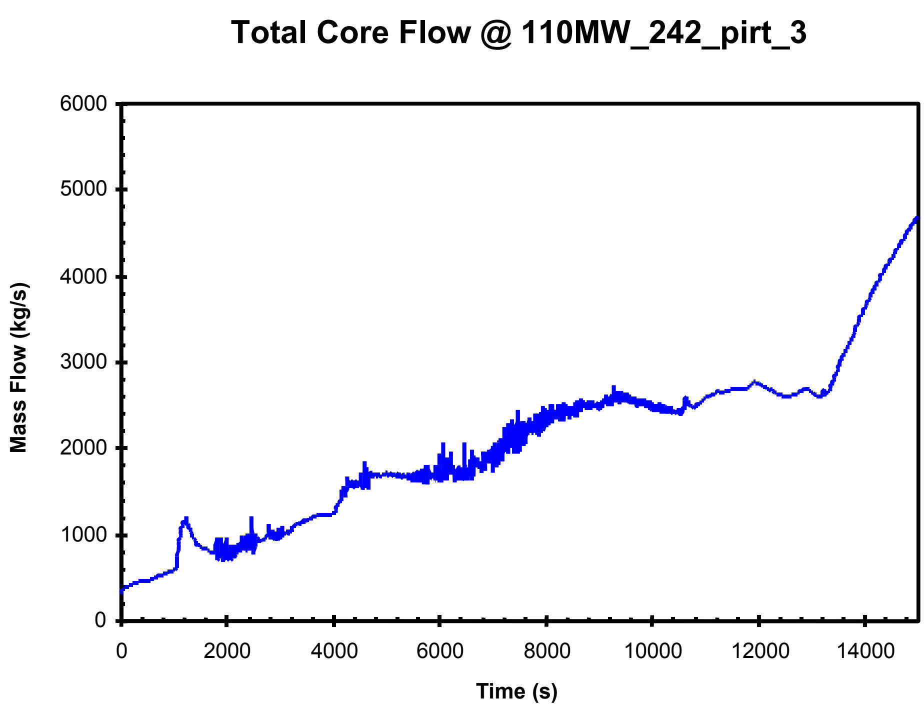



Figure 2.2-29. Total Core Flow

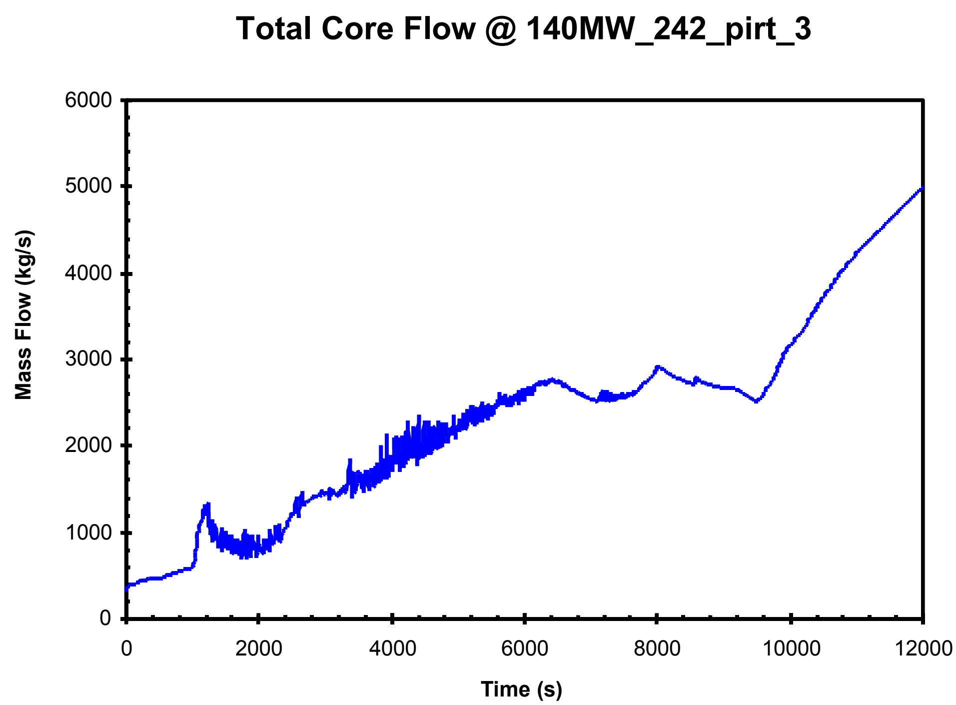



**Figure 2.2-30.  Total Core Flow**

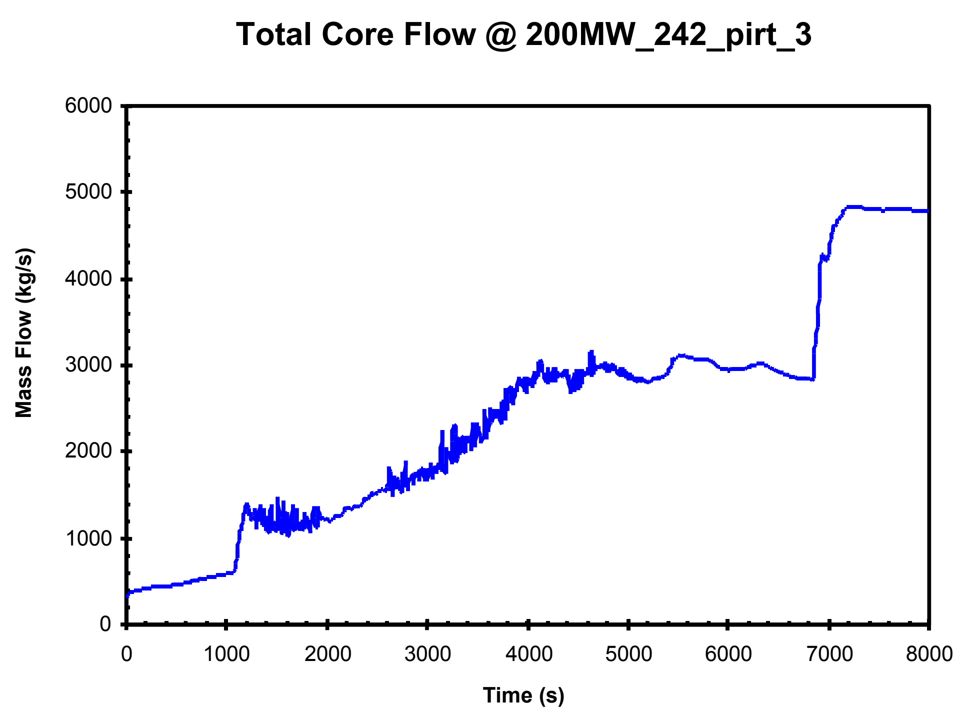



**Figure 2.2-31.  Total Core Flow**

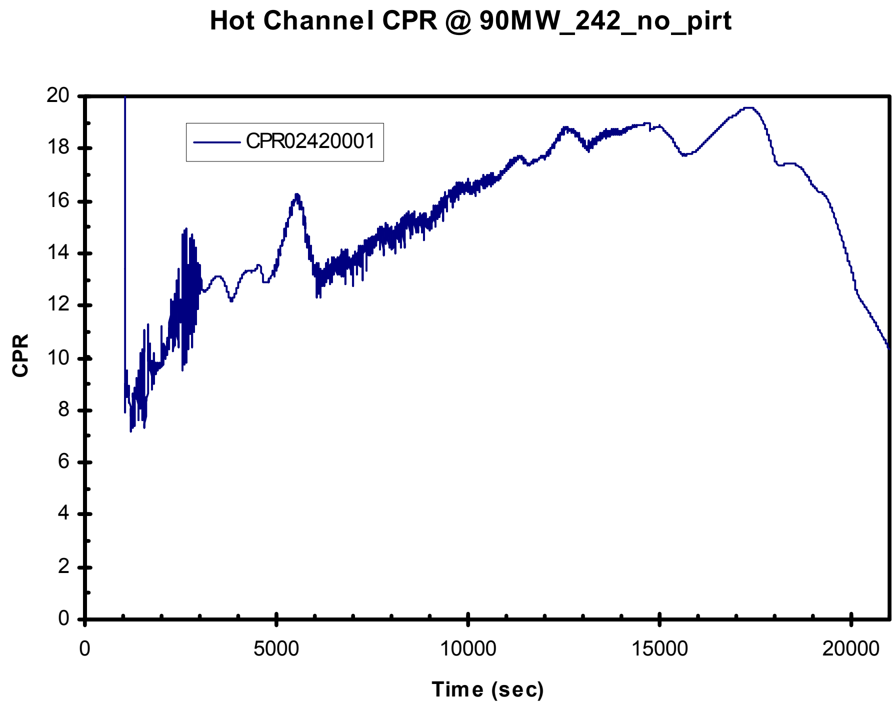



Figure 2.2-32. Hot Channel CPR

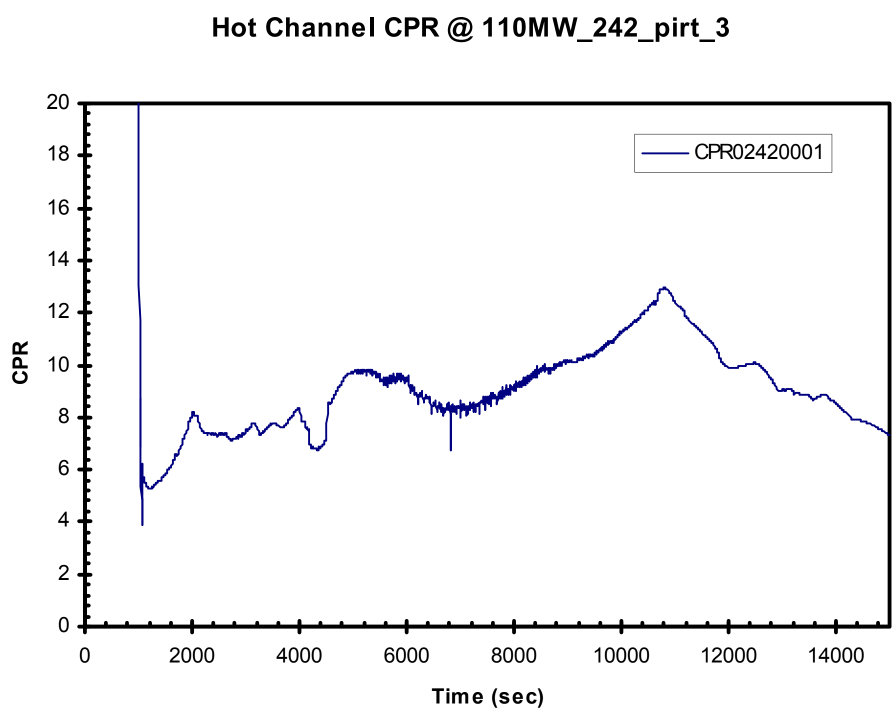



Figure 2.2-33. Hot Channel CPR

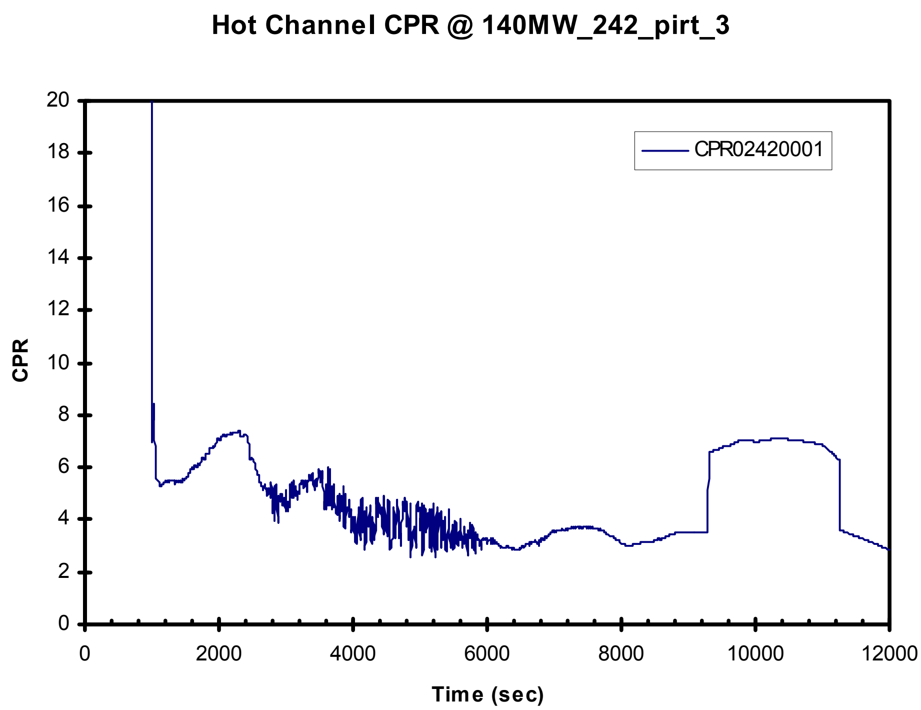



Figure 2.2-34. Hot Channel CPR

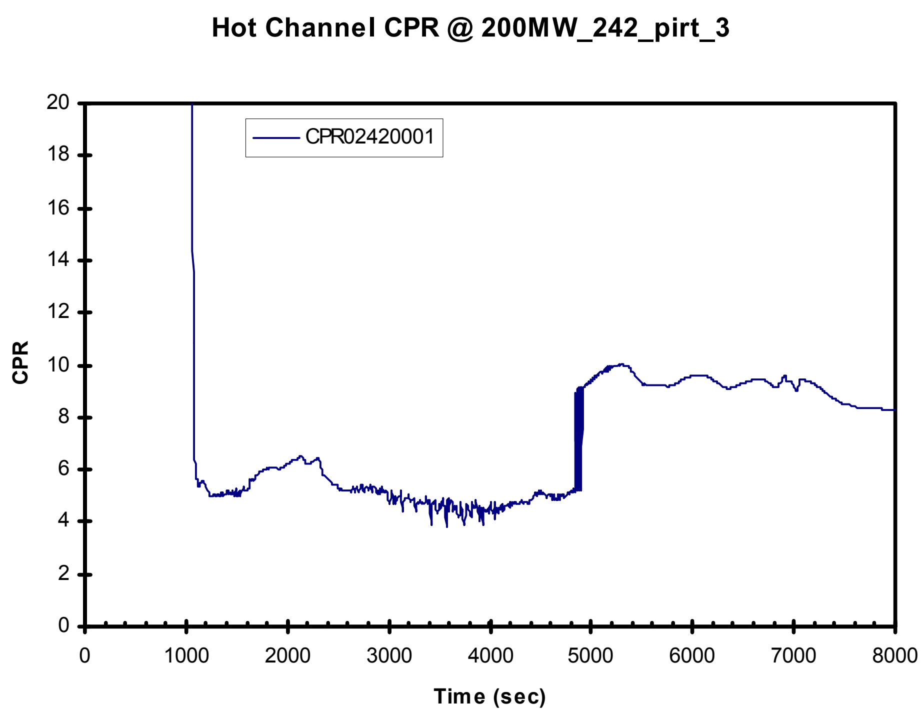



Figure 2.2-35. Hot Channel CPR

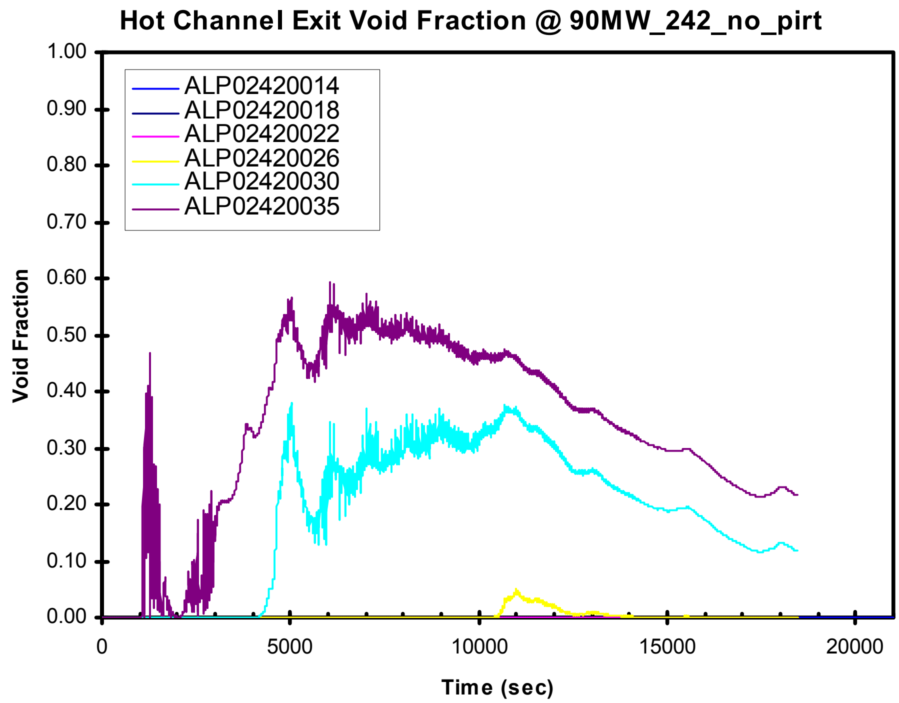



**Figure 2.2-36. Hot Channel Exit Void Fraction**

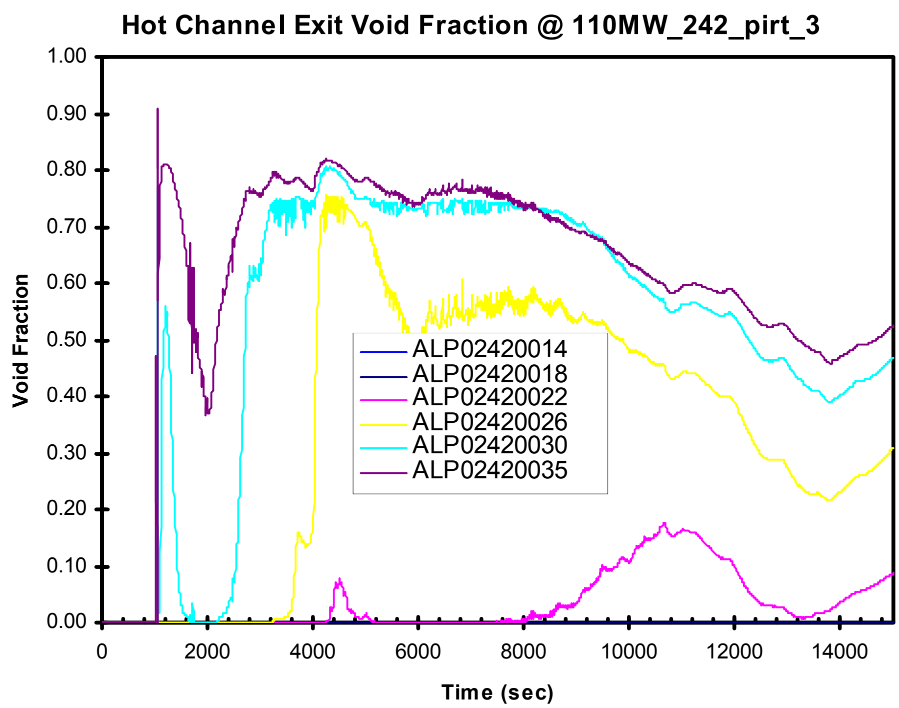



**Figure 2.2-37. Hot Channel Exit Void Fraction**

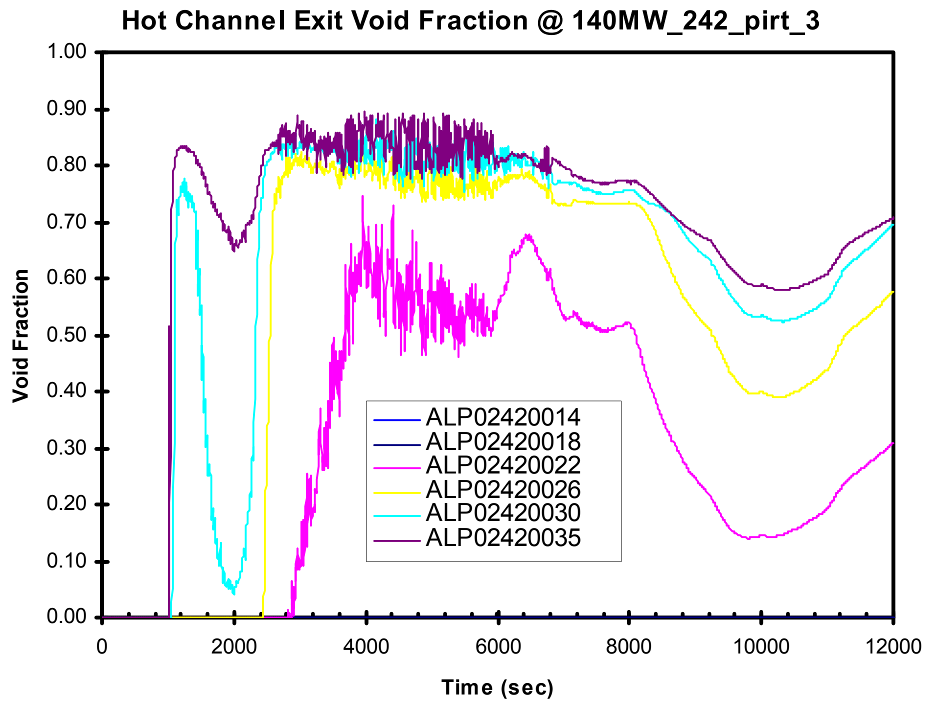



**Figure 2.2-38. Hot Channel Exit Void Fraction**

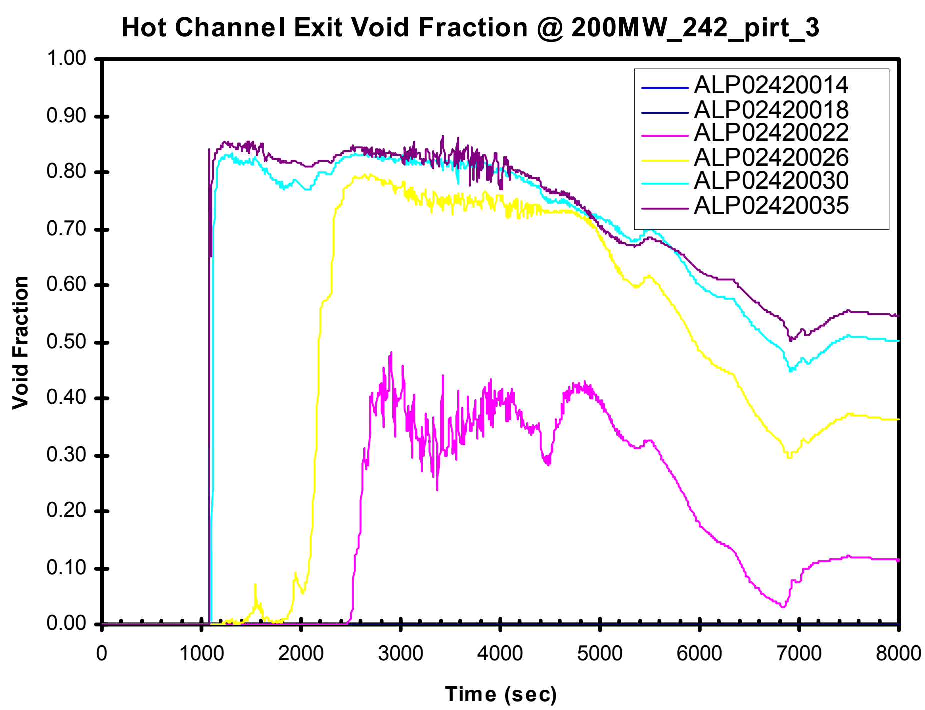



**Figure 2.2-39. Hot Channel Exit Void Fraction**

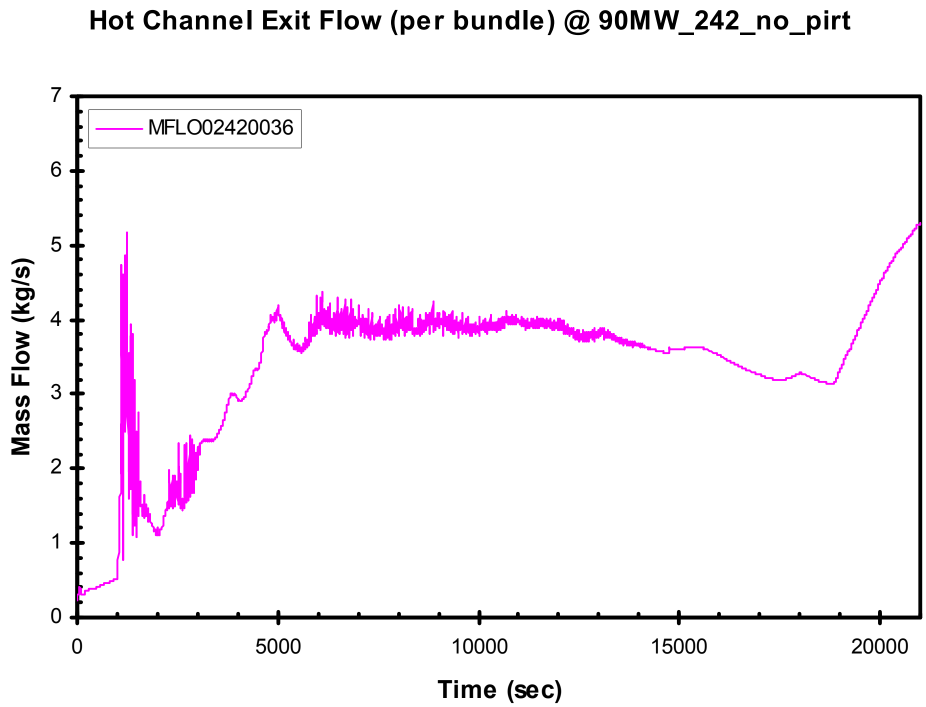



Figure 2.2-40. Hot Channel Exit Flow (per Bundle)

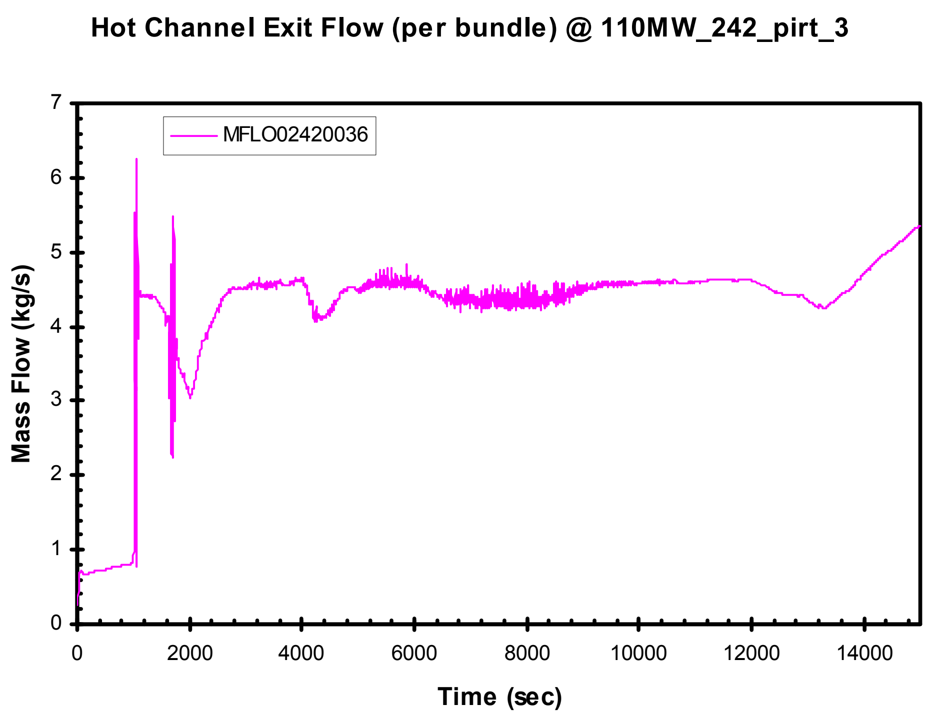



Figure 2.2-41. Hot Channel Exit Flow (per Bundle)

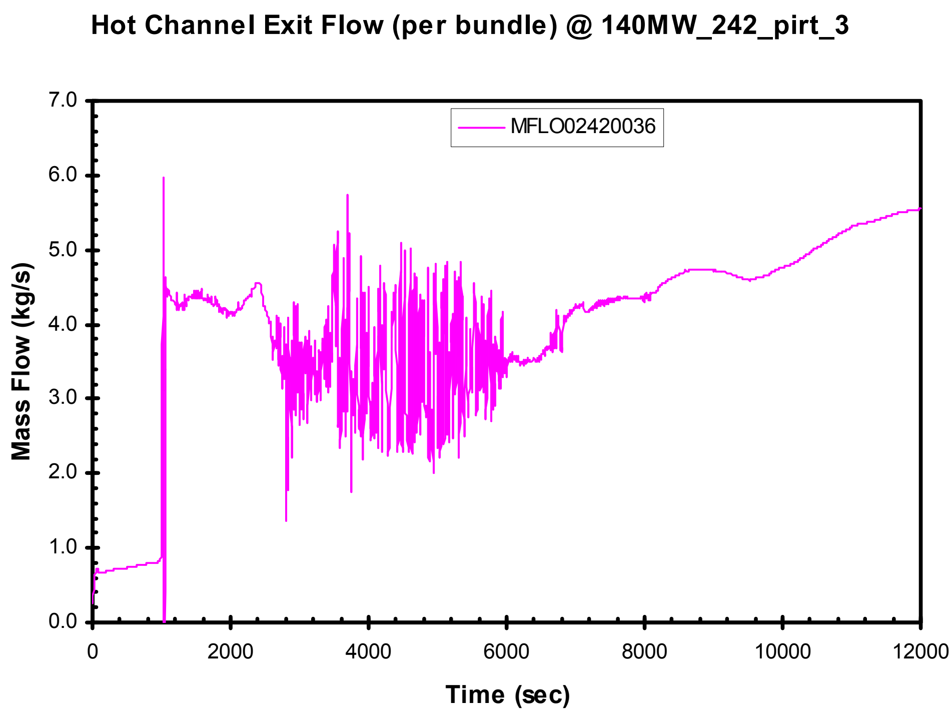



Figure 2.2-42. Hot Channel Exit Flow (per Bundle)

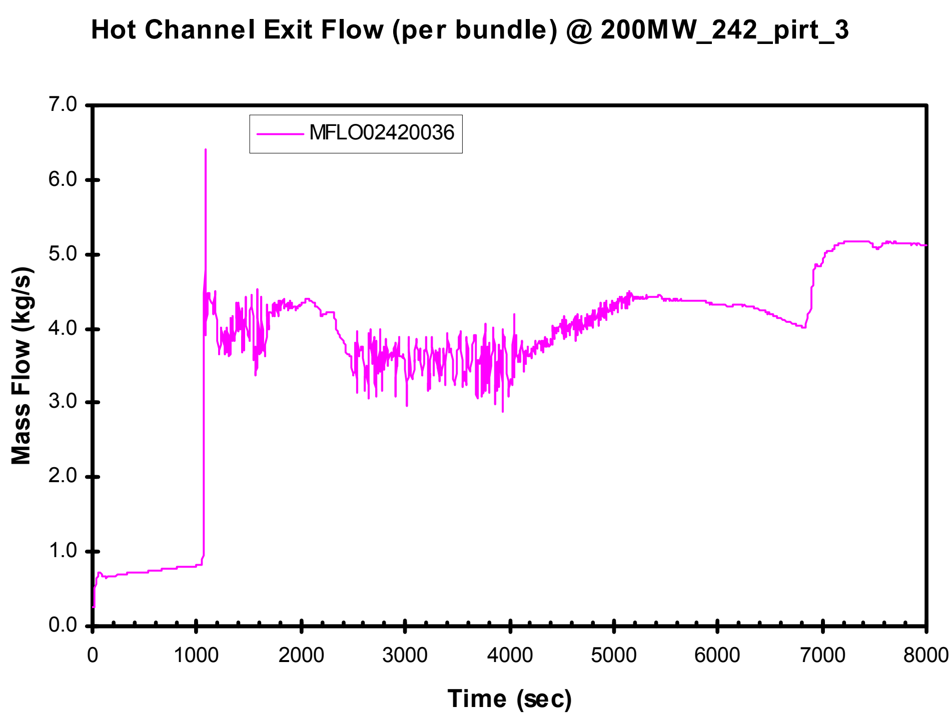



Figure 2.2-43. Hot Channel Exit Flow (per Bundle)

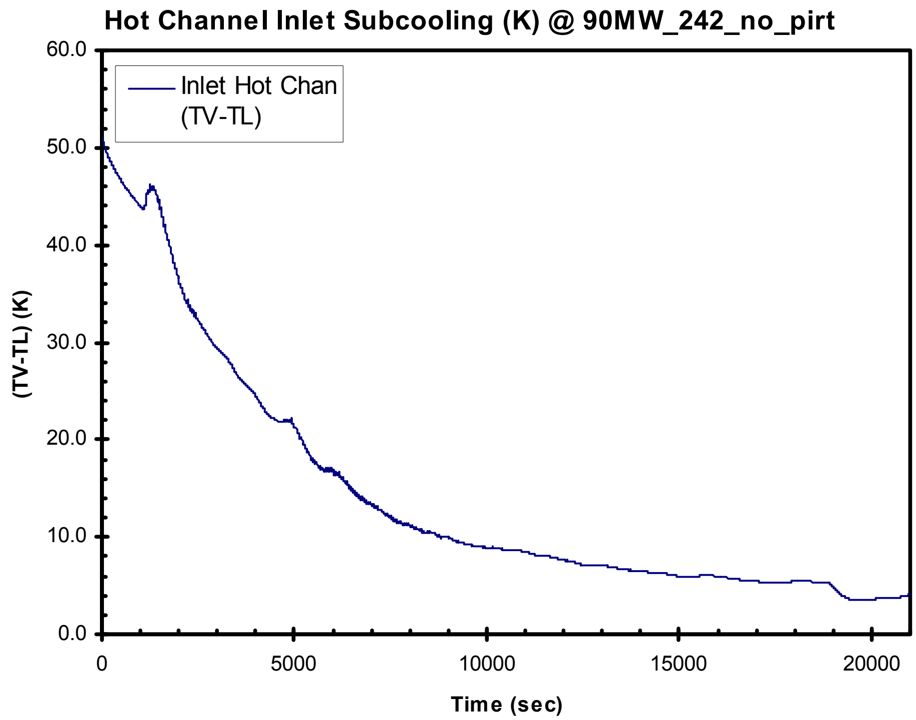



Figure 2.2-44. Hot Channel Inlet Subcooling

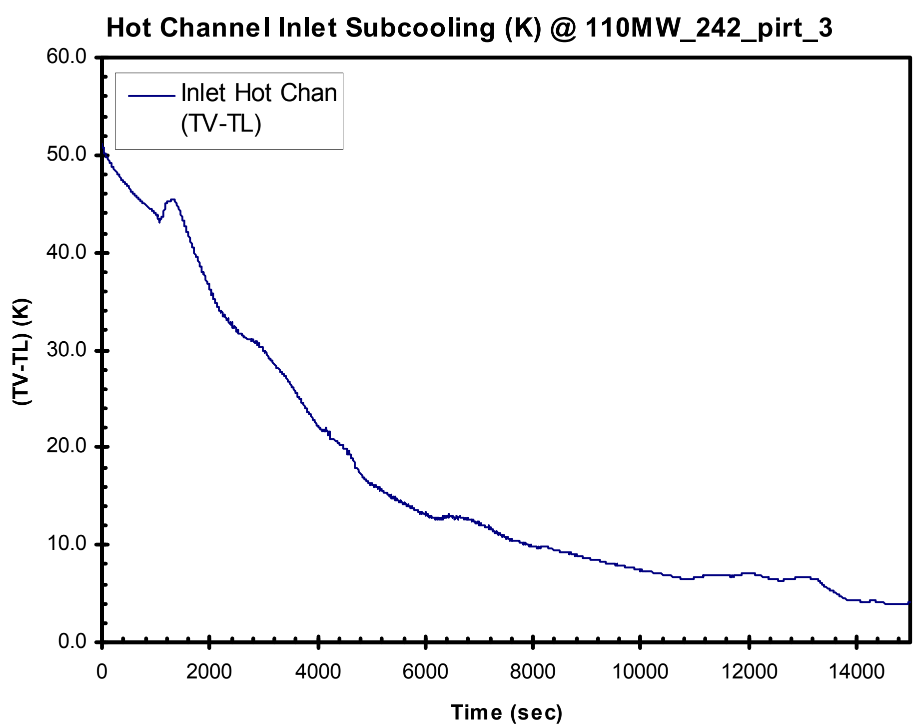



Figure 2.2-45. Hot Channel Inlet Subcooling

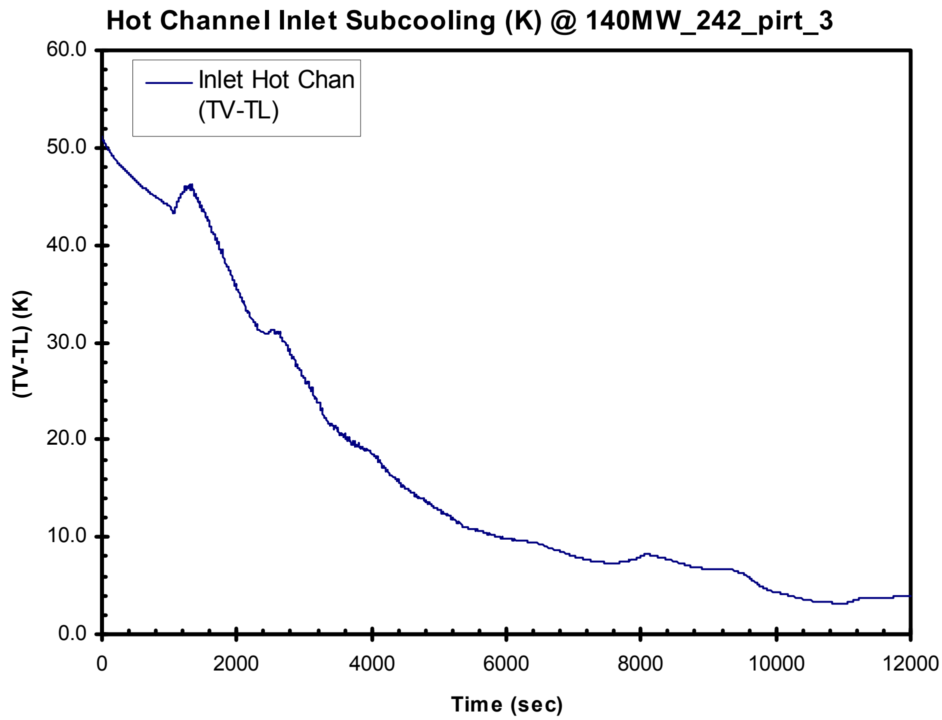



**Figure 2.2-46. Hot Channel Inlet Subcooling**

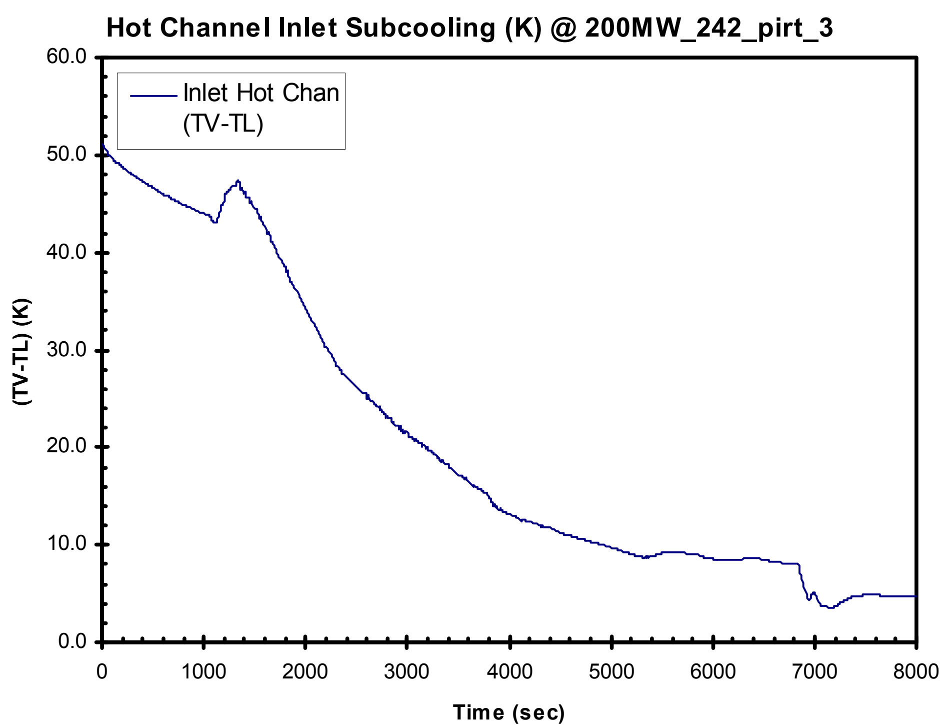



**Figure 2.2-47. Hot Channel Inlet Subcooling**

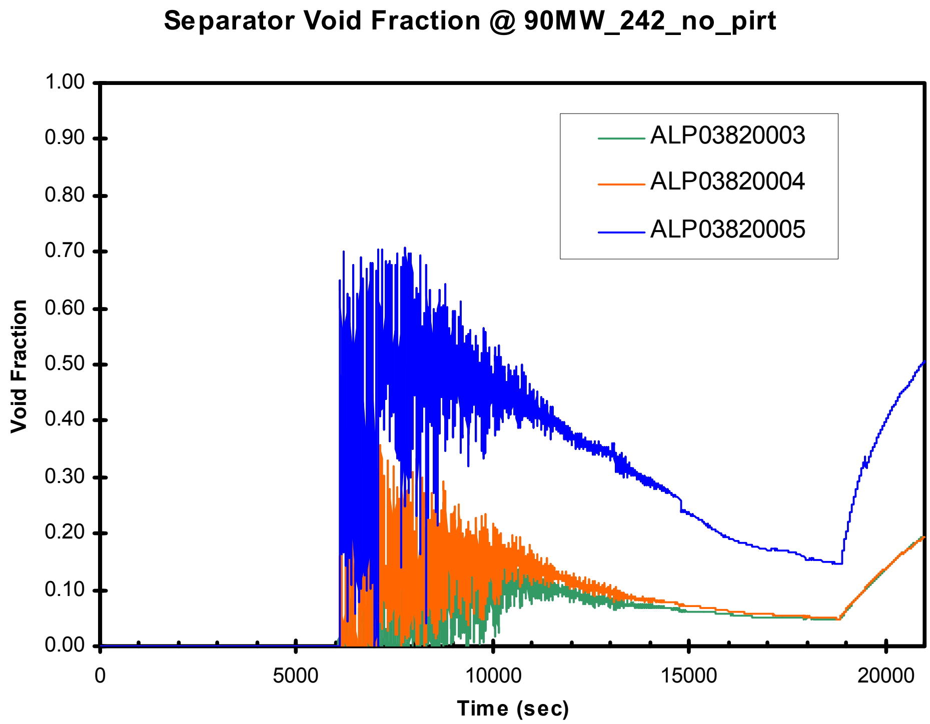



**Figure 2.2-48. Separator Void Fraction**

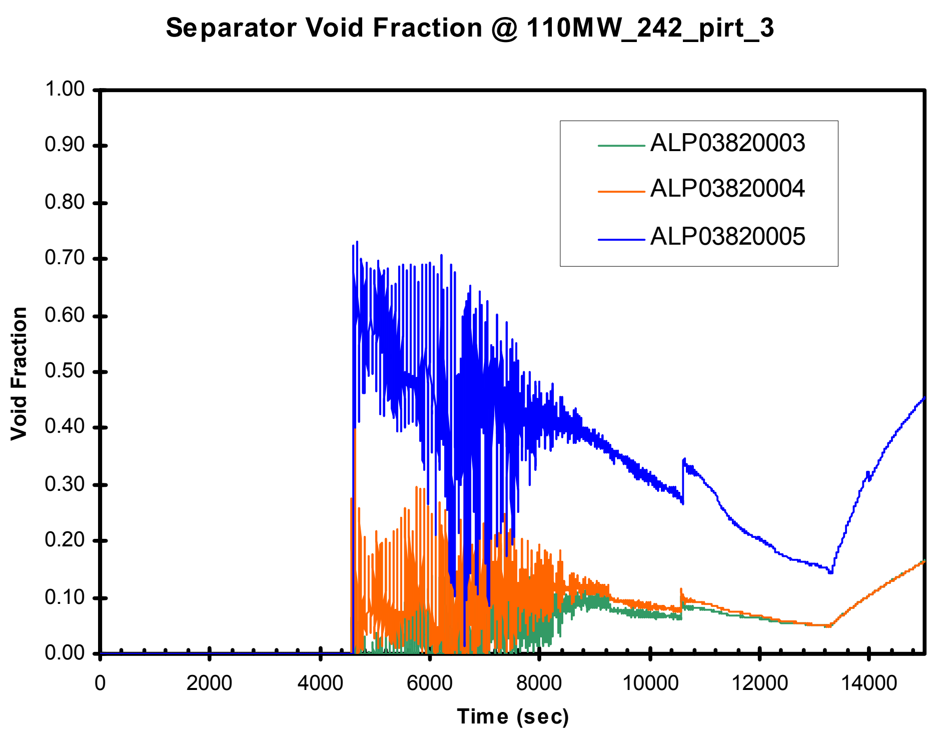



**Figure 2.2-49. Separator Void Fraction**

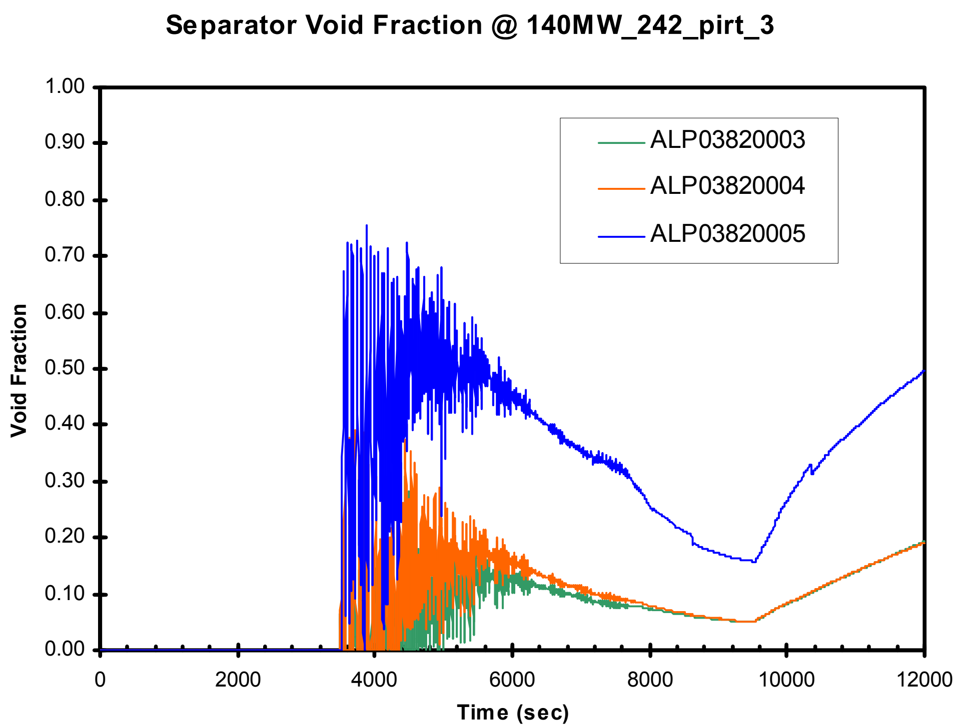



**Figure 2.2-50. Separator Void Fraction**

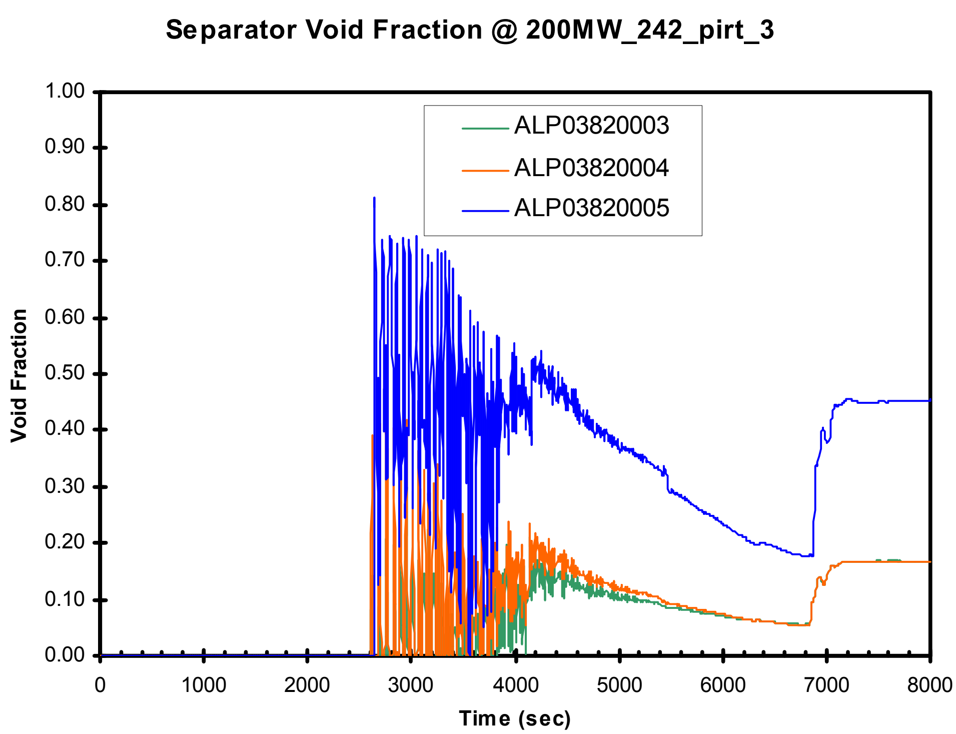



**Figure 2.2-51. Separator Void Fraction**

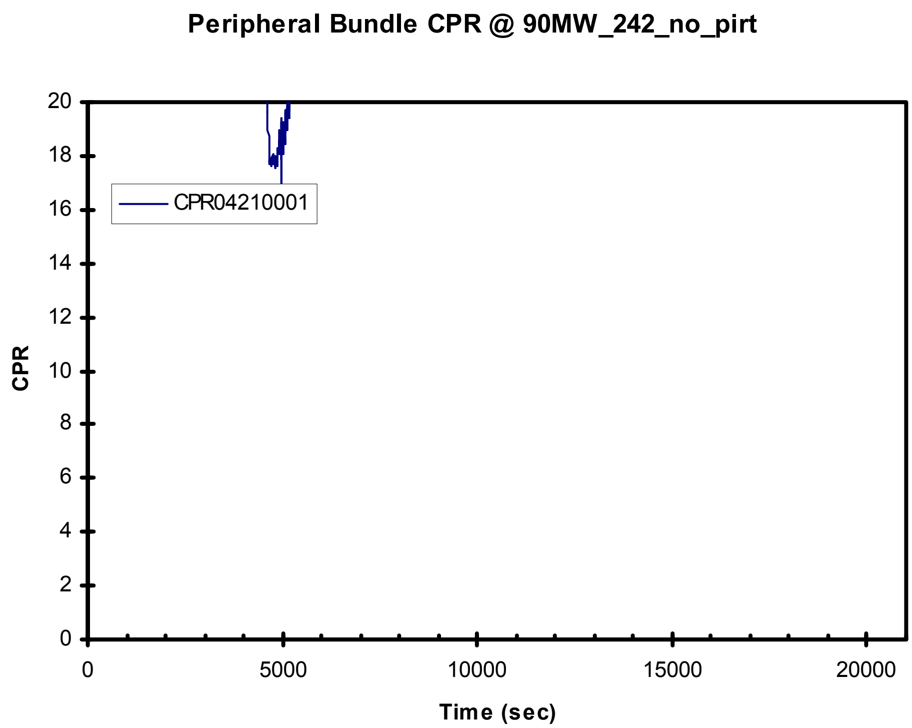



Figure 2.2-52. Peripheral Bundle CPR

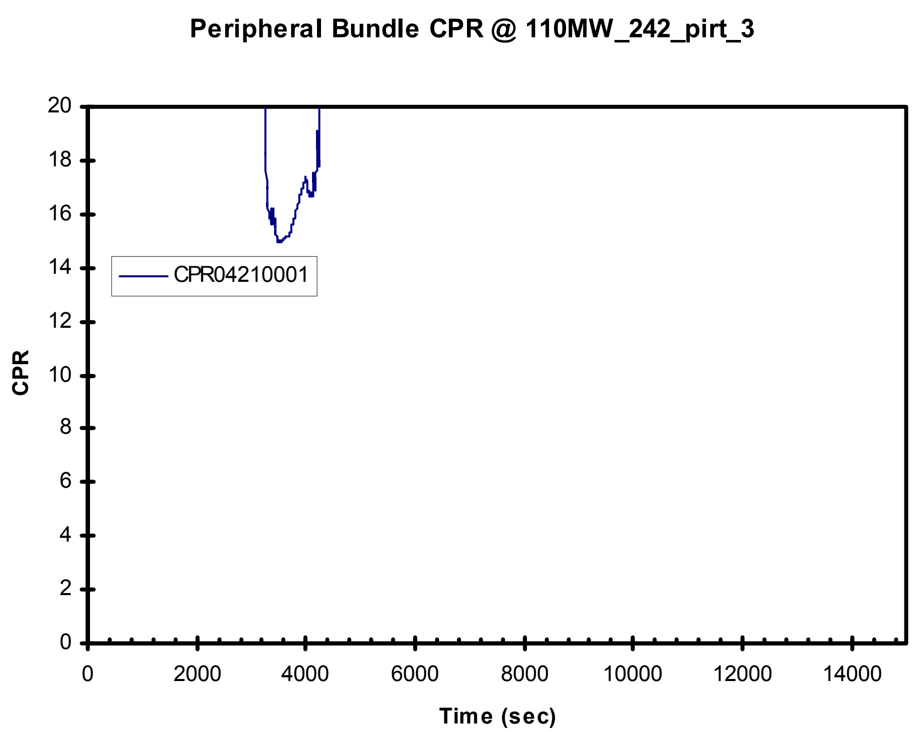



Figure 2.2-53. Peripheral Bundle CPR

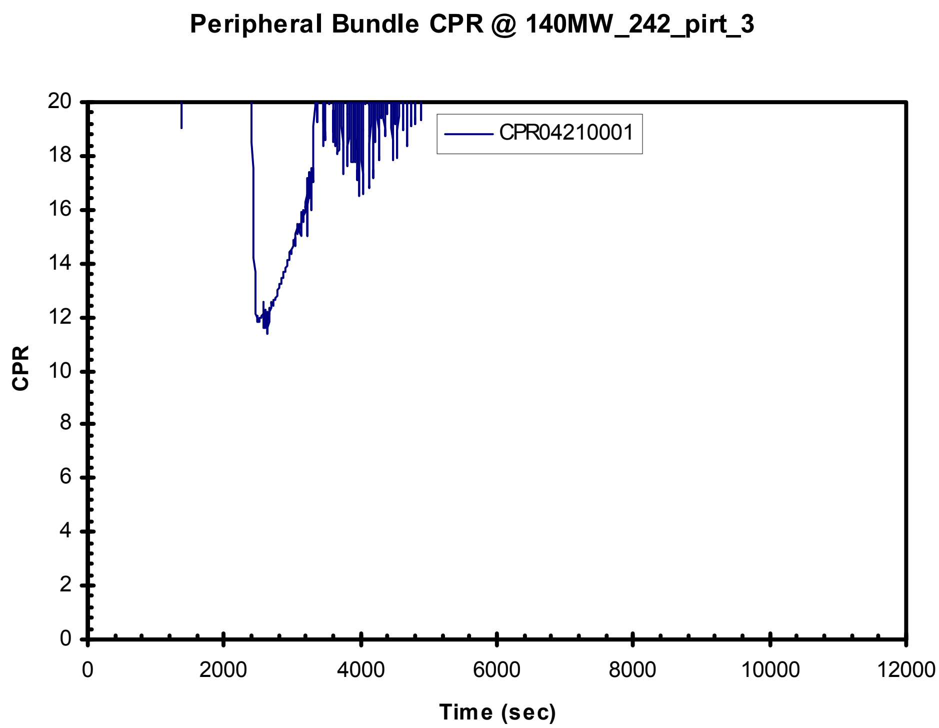



Figure 2.2-54. Peripheral Bundle CPR

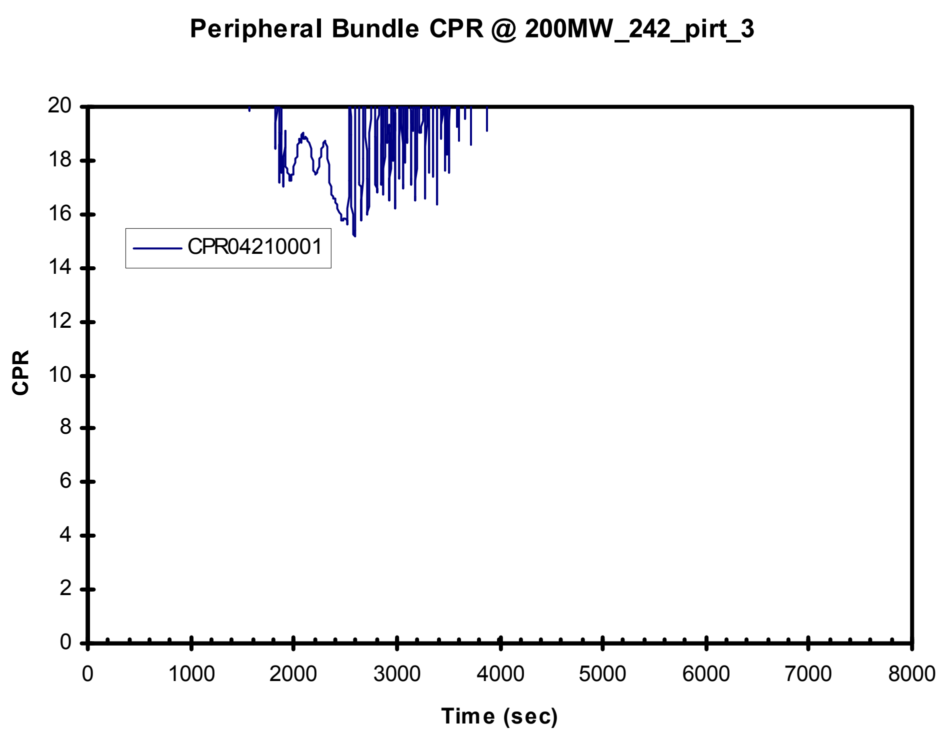



Figure 2.2-55. Peripheral Bundle CPR

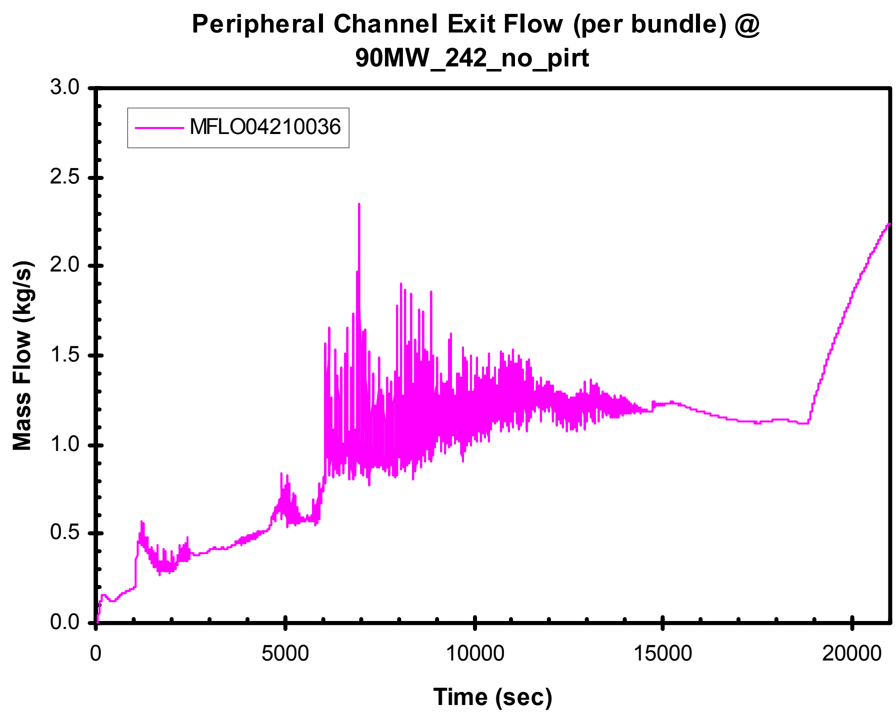



Figure 2.2-56. Peripheral Channel Exit Flow (per bundle)

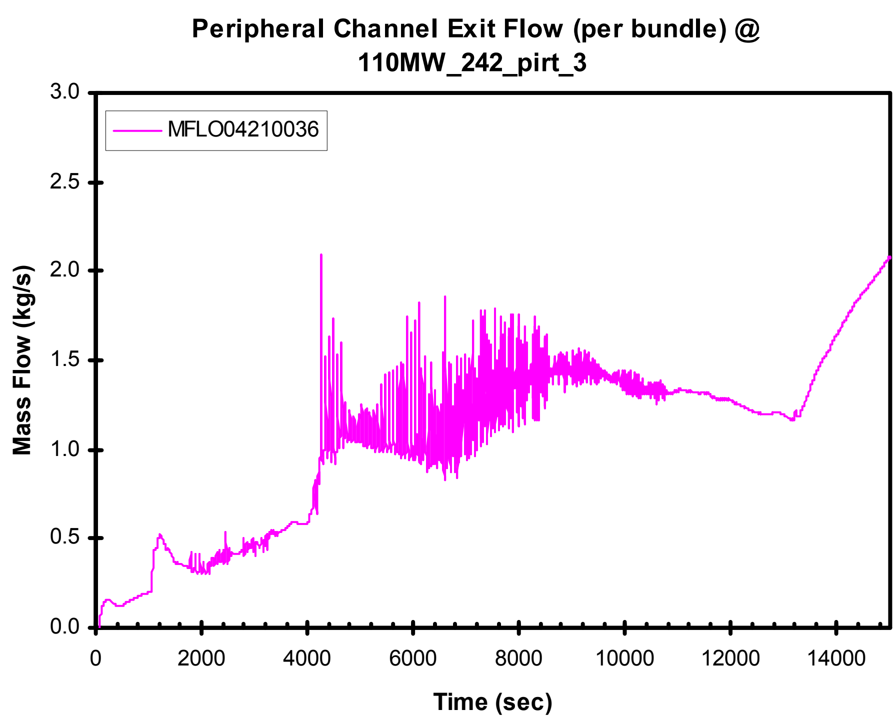



Figure 2.2-57. Peripheral Channel Exit Flow (per bundle)

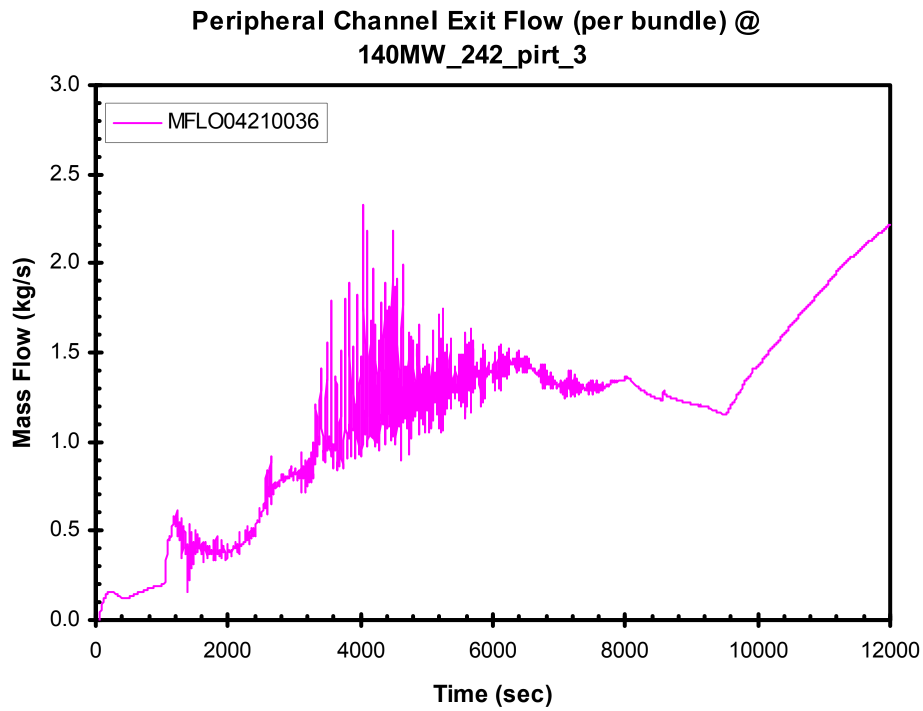



Figure 2.2-58. Peripheral Channel Exit Flow (per bundle)

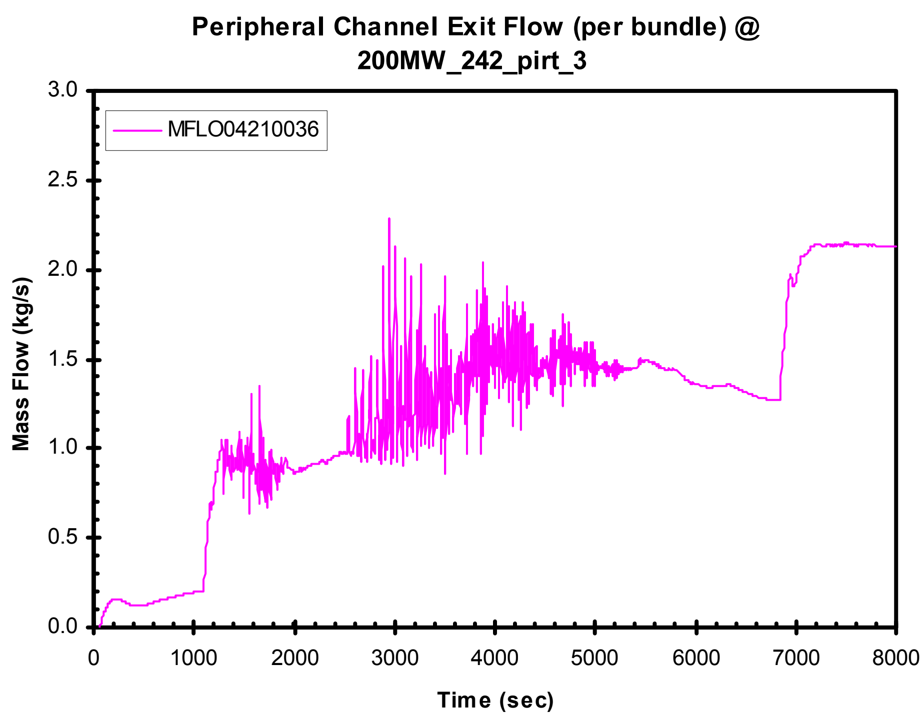



Figure 2.2-59. Peripheral Channel Exit Flow (per bundle)

# Limiting Case Plots

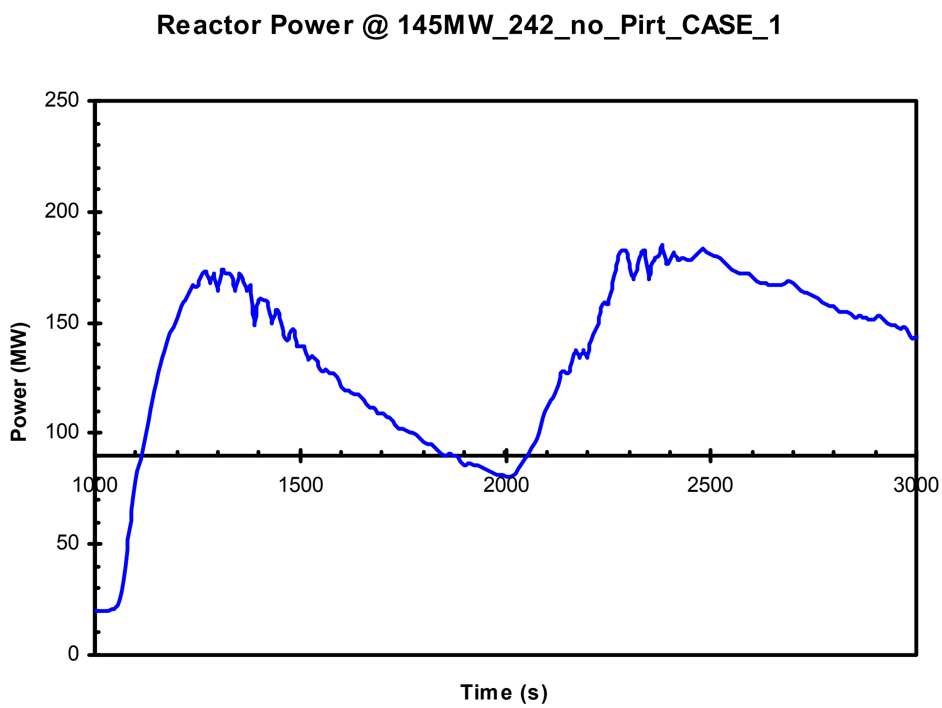



Figure 2.2-60. Reactor Power

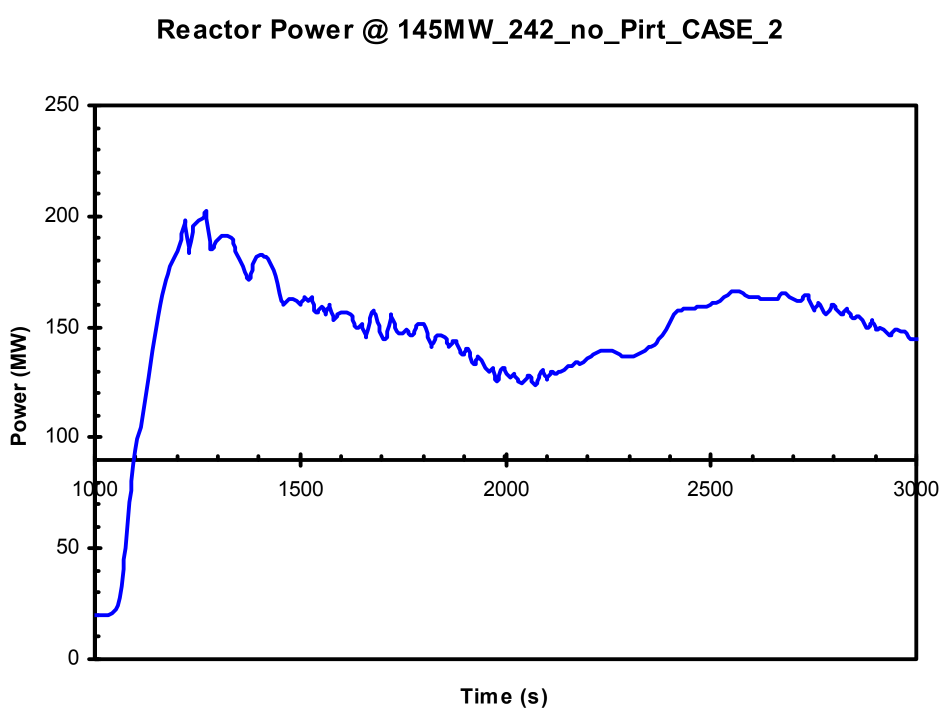



Figure 2.2-61. Reactor Power

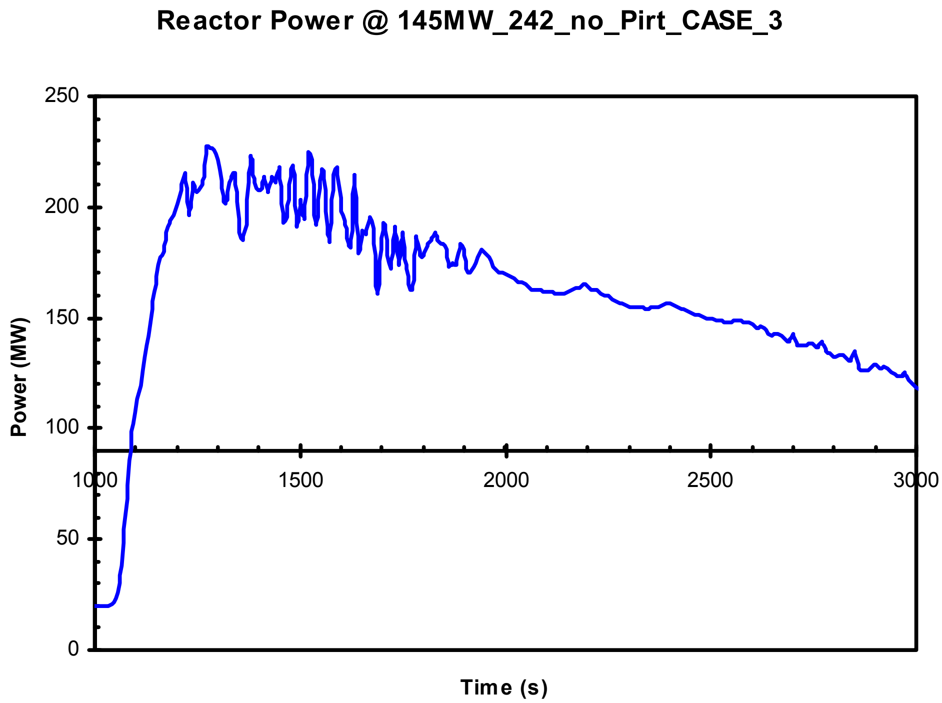



**Figure 2.2-62. Reactor Power**

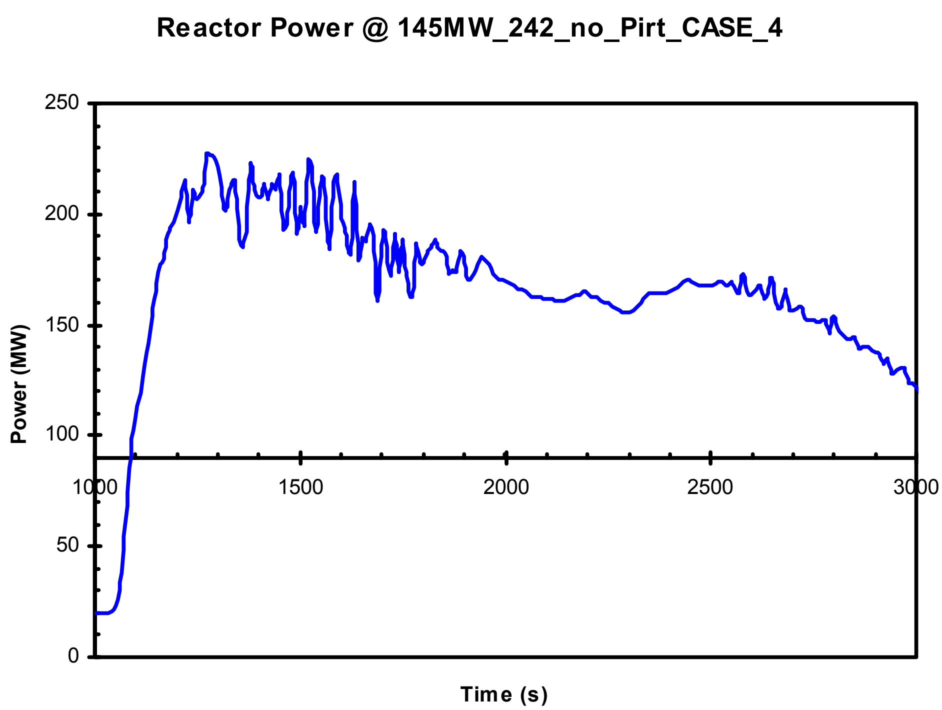



**Figure 2.2-63. Reactor Power**

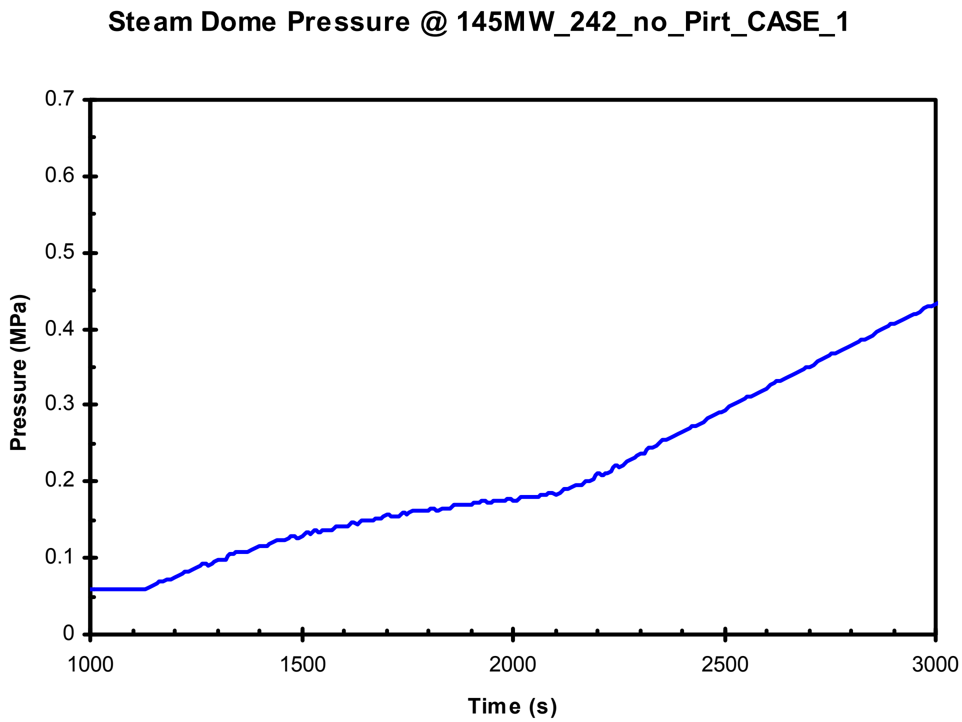



Figure 2.2-64. Steam Dome Pressure

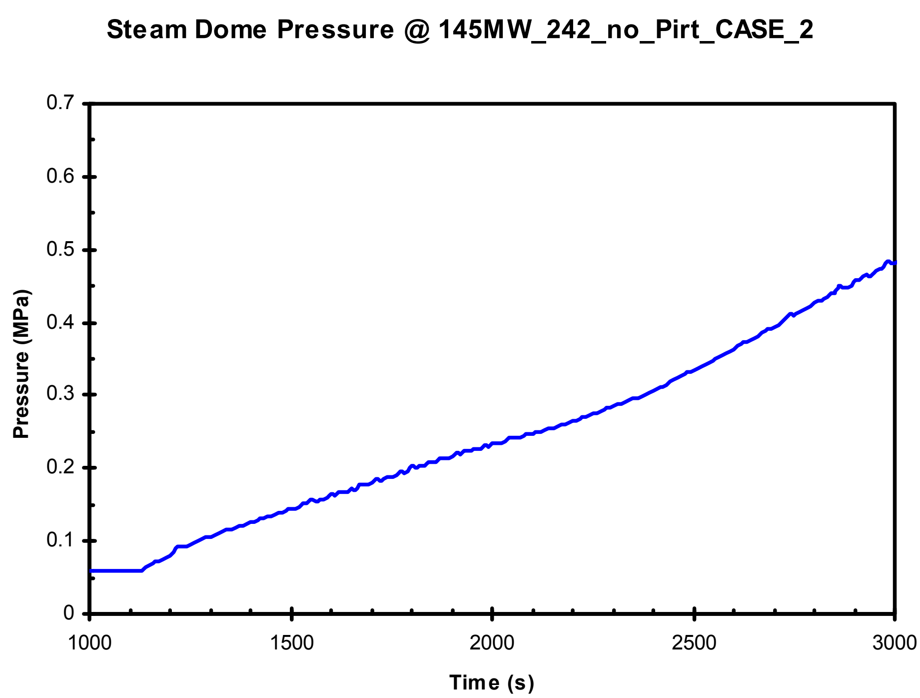



Figure 2.2-65. Steam Dome Pressure

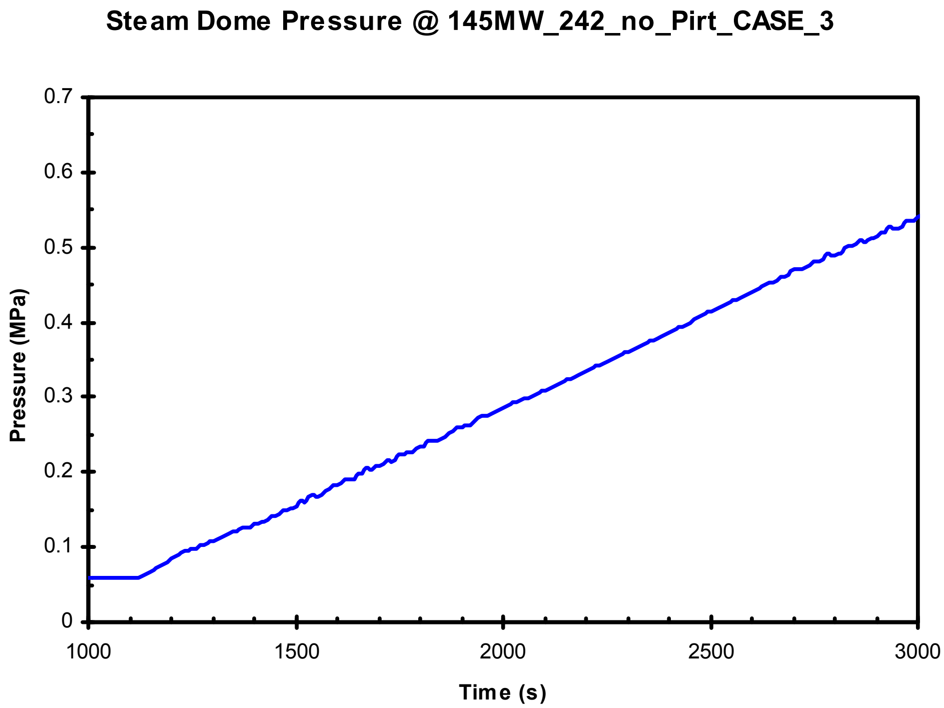



Figure 2.2-66. Steam Dome Pressure

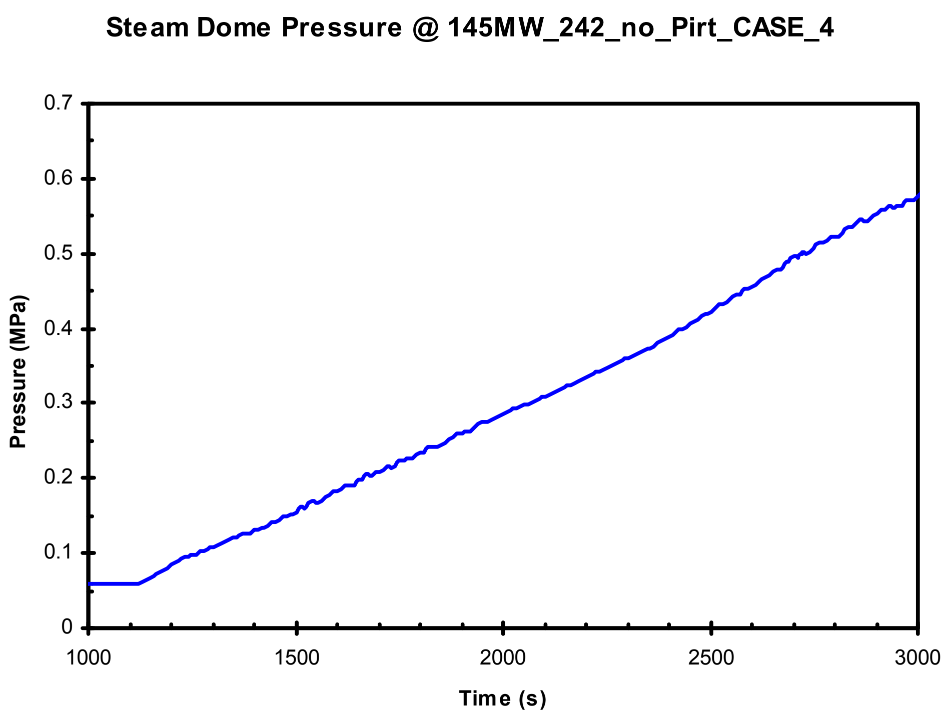



Figure 2.2-67. Steam Dome Pressure

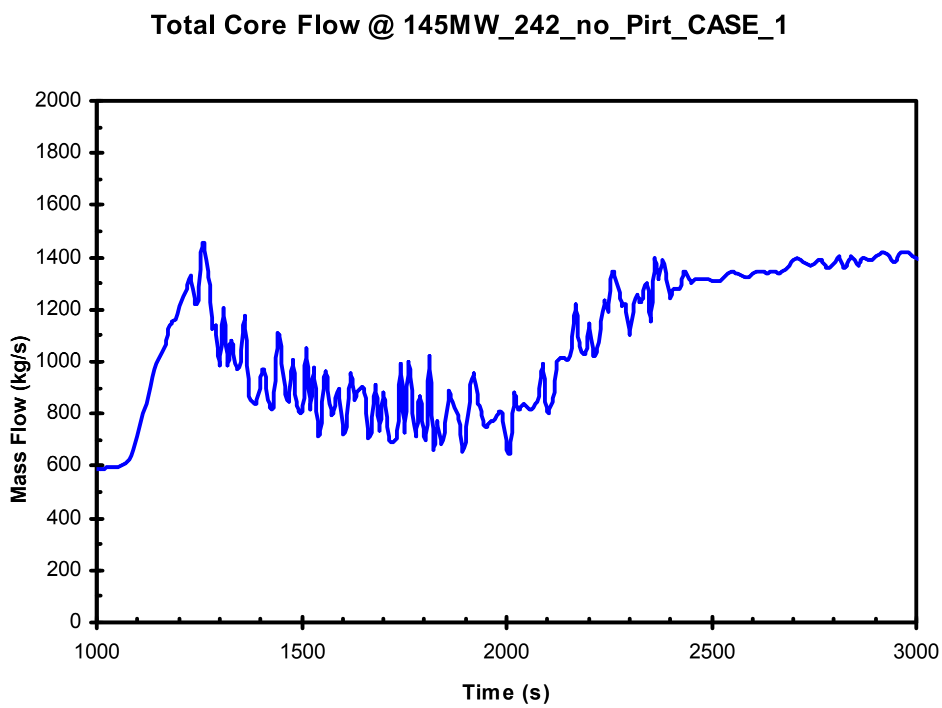



Figure 2.2-68. Total Core Flow

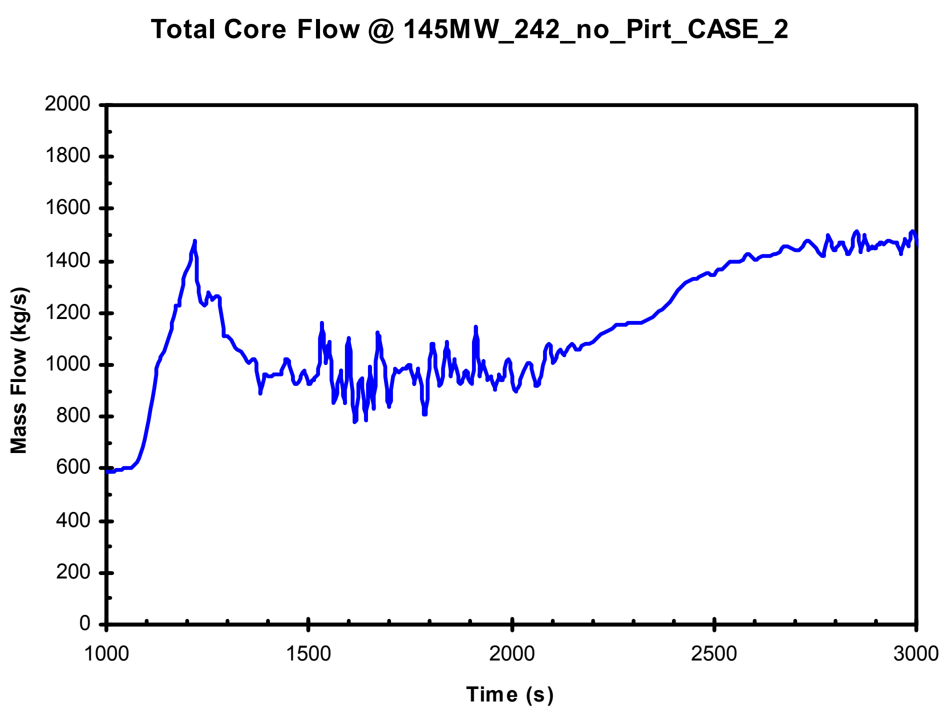



Figure 2.2-69. Total Core Flow

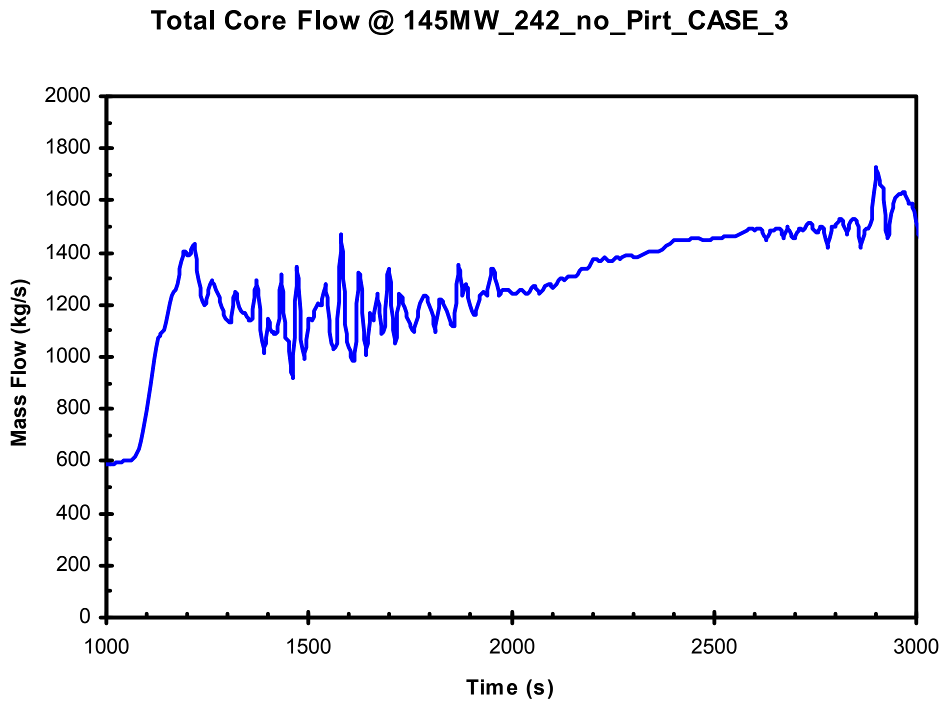



Figure 2.2-70. Total Core Flow

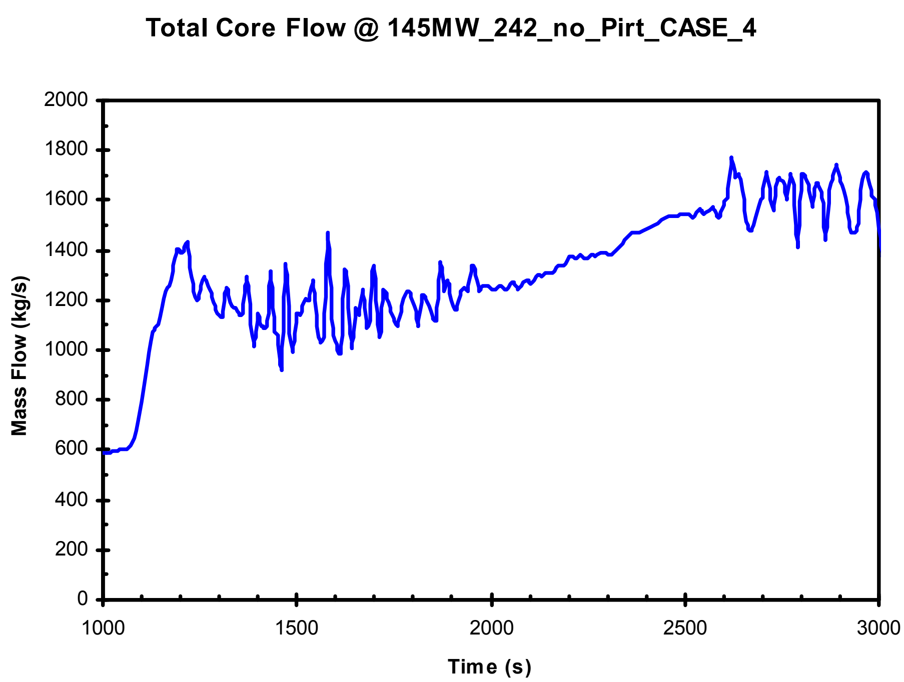



Figure 2.2-71. Total Core Flow

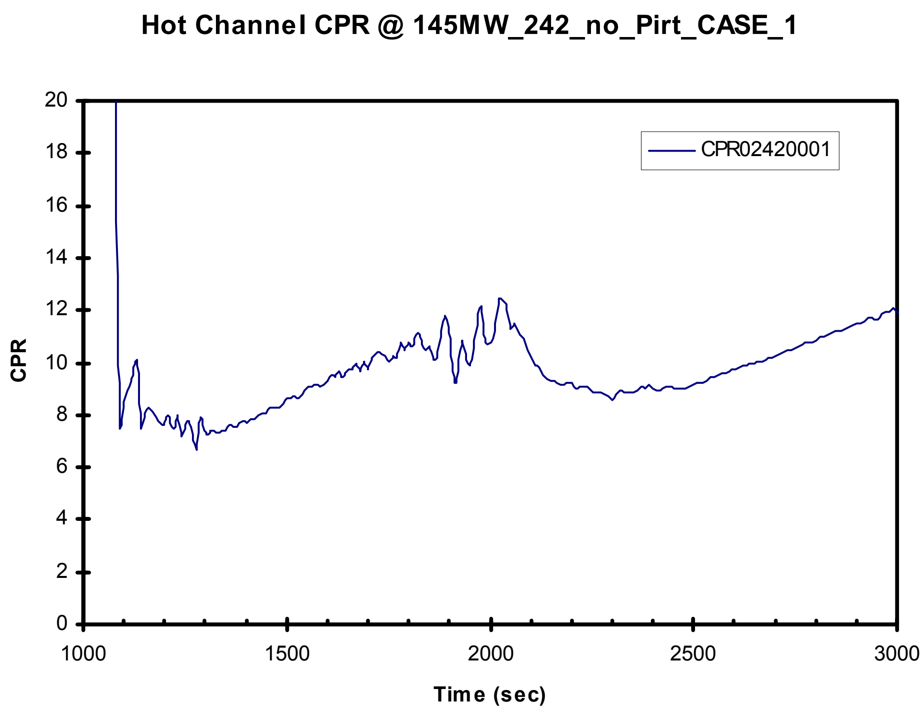



**Figure 2.2-72. Hot Channel CPR**

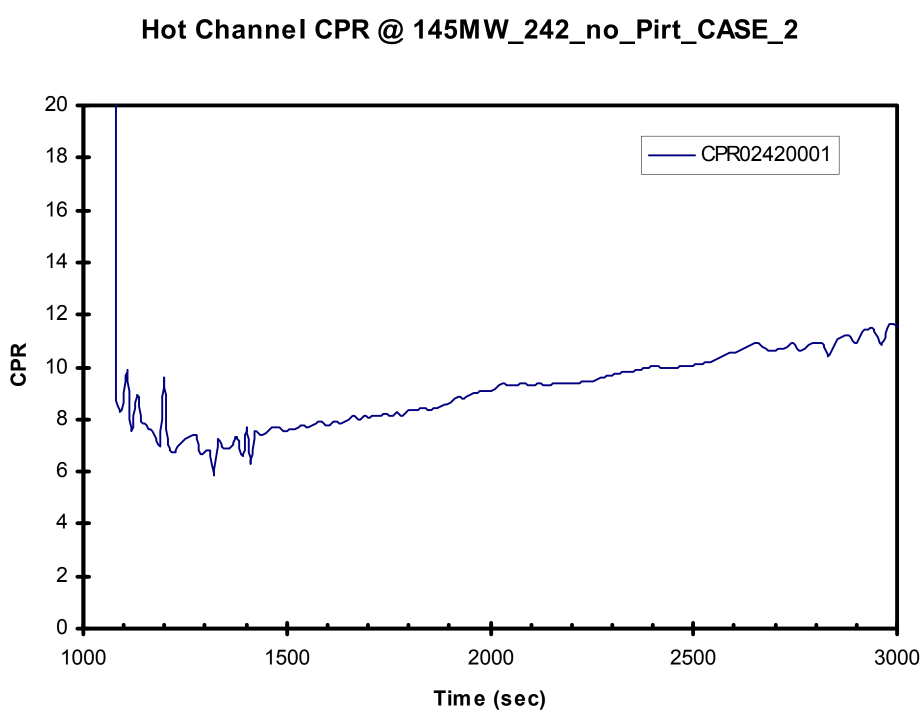



**Figure 2.2-73. Hot Channel CPR**

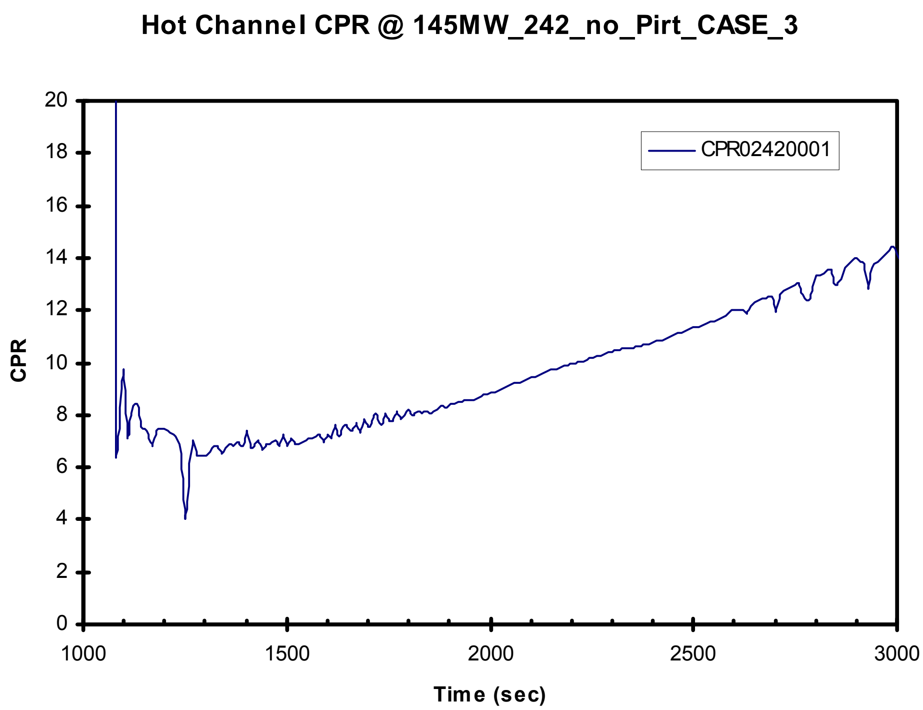



Figure 2.2-74. Hot Channel CPR

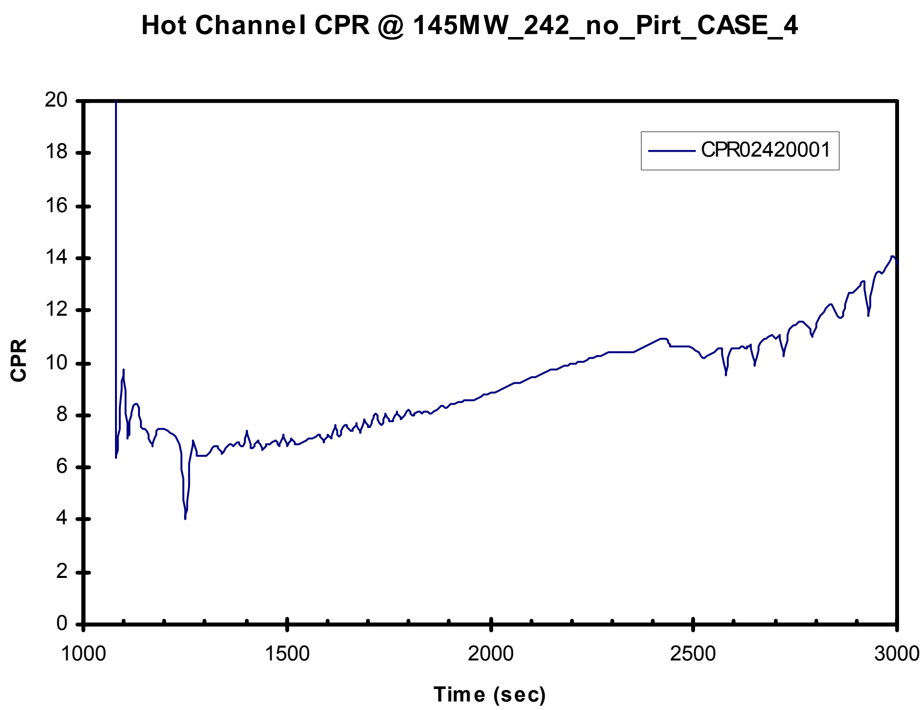



Figure 2.2-75. Hot Channel CPR

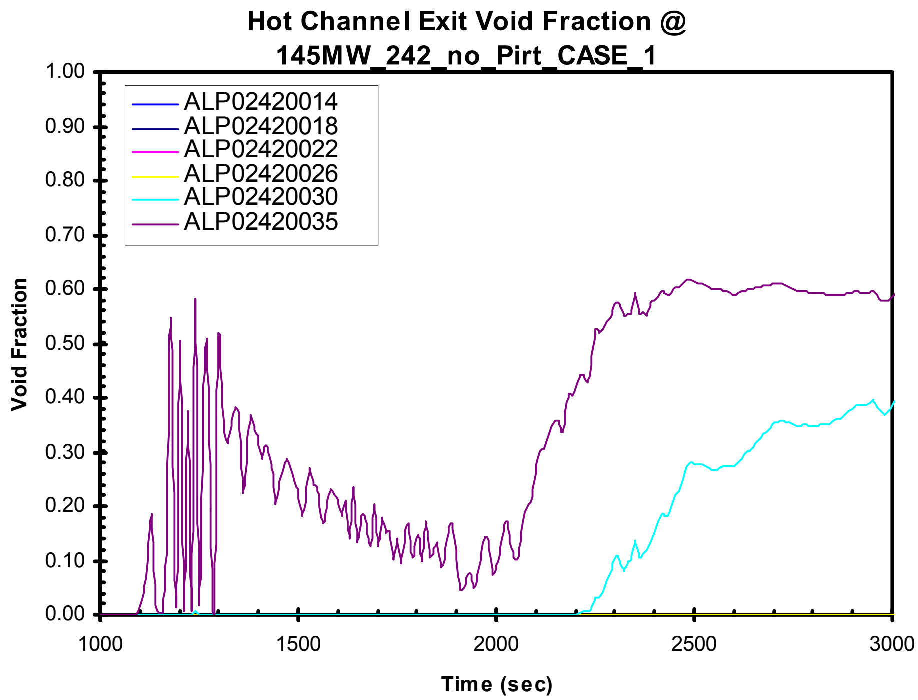



Figure 2.2-76. Hot Channel Exit Void Fraction

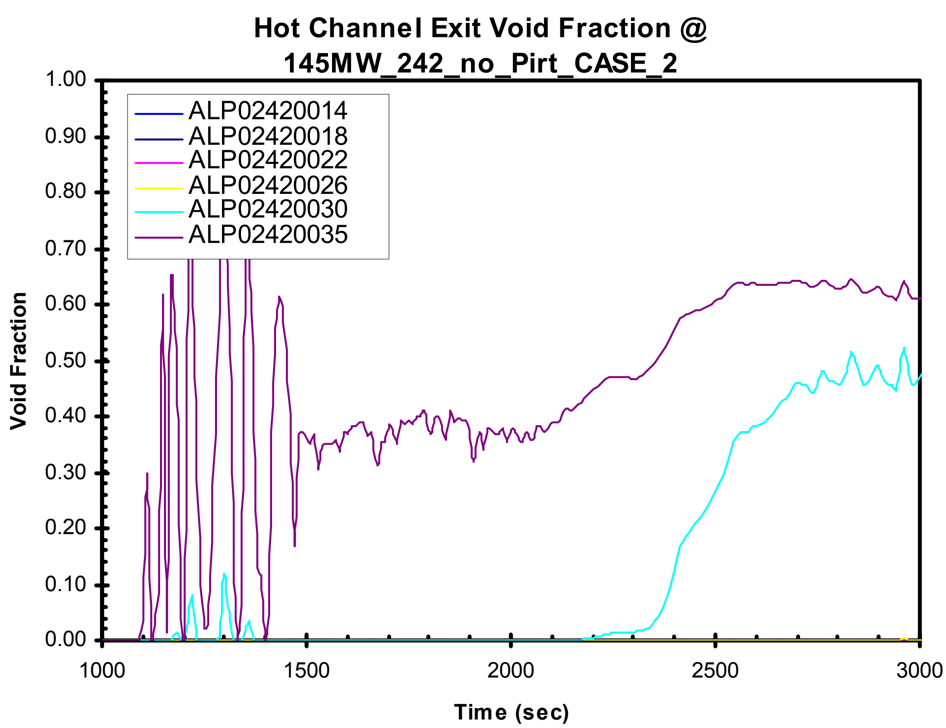



Figure 2.2-77. Hot Channel Exit Void Fraction

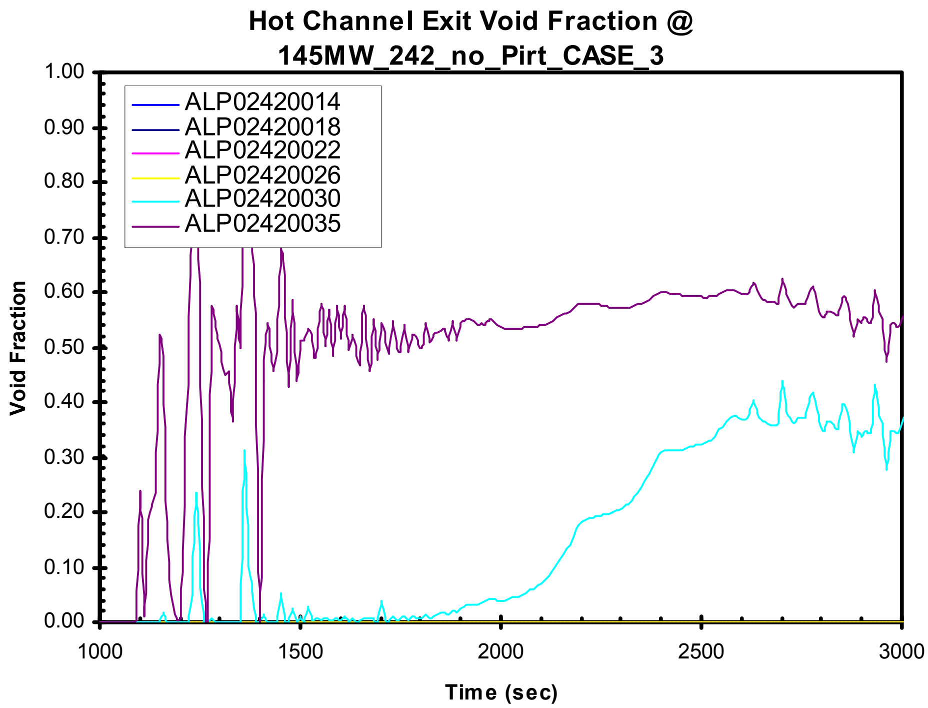



**Figure 2.2-78. Hot Channel Exit Void Fraction**

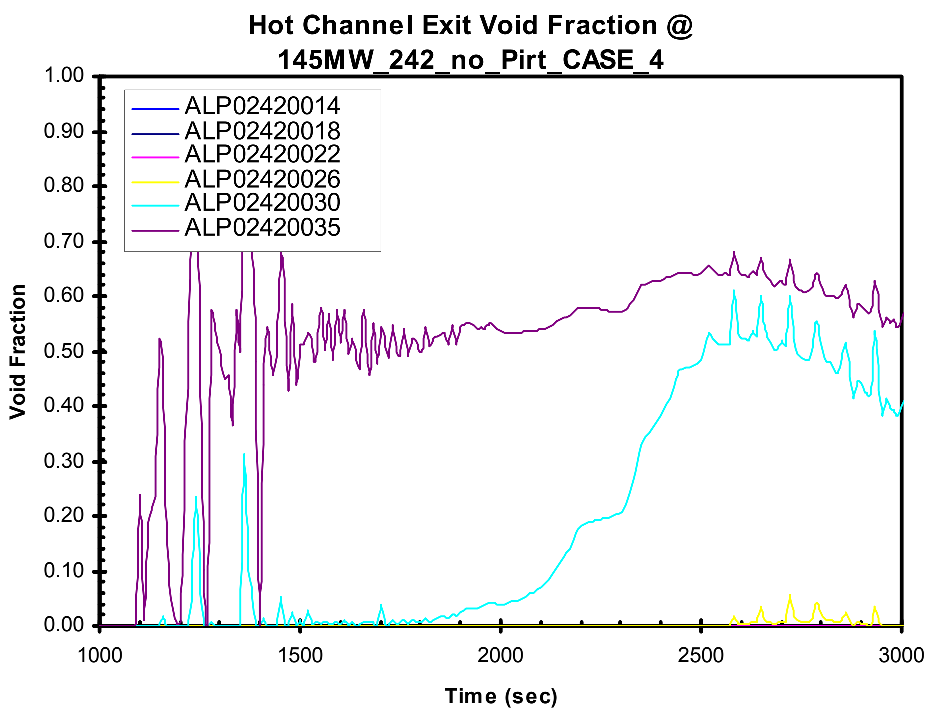



**Figure 2.2-79. Hot Channel Exit Void Fraction**

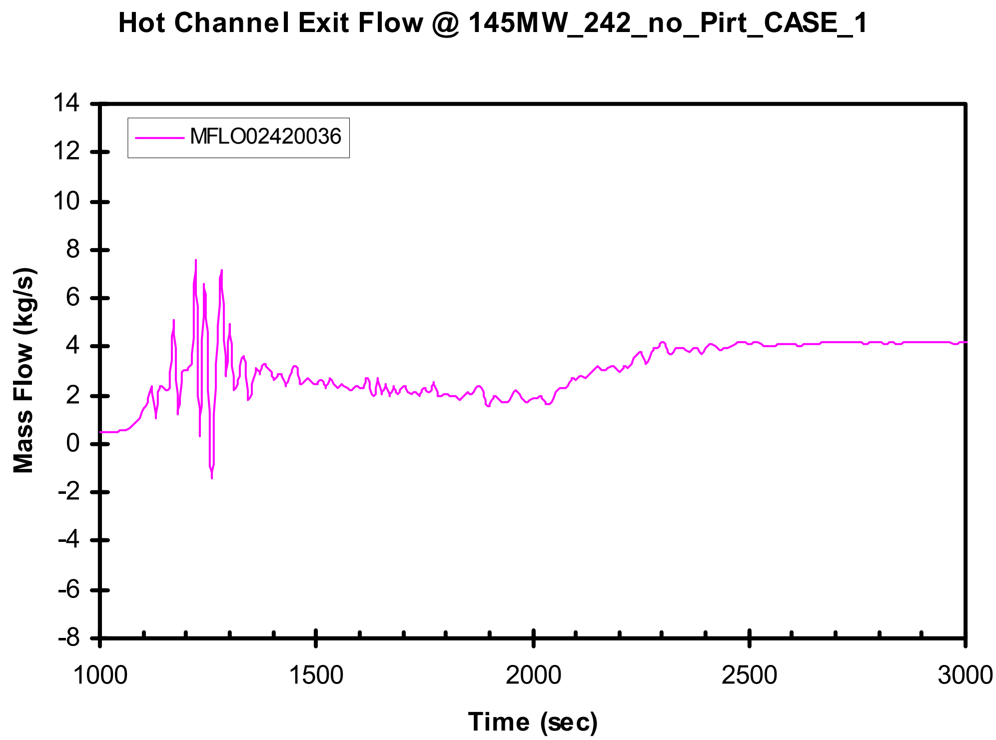



Figure 2.2-80. Hot Channel Exit Flow

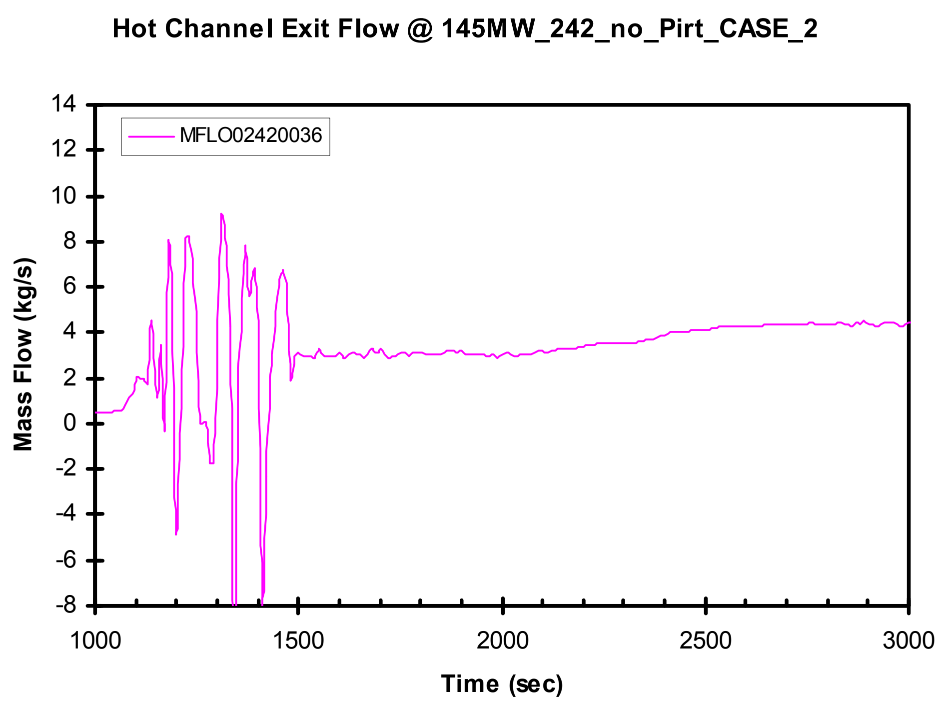



Figure 2.2-81. Hot Channel Exit Flow

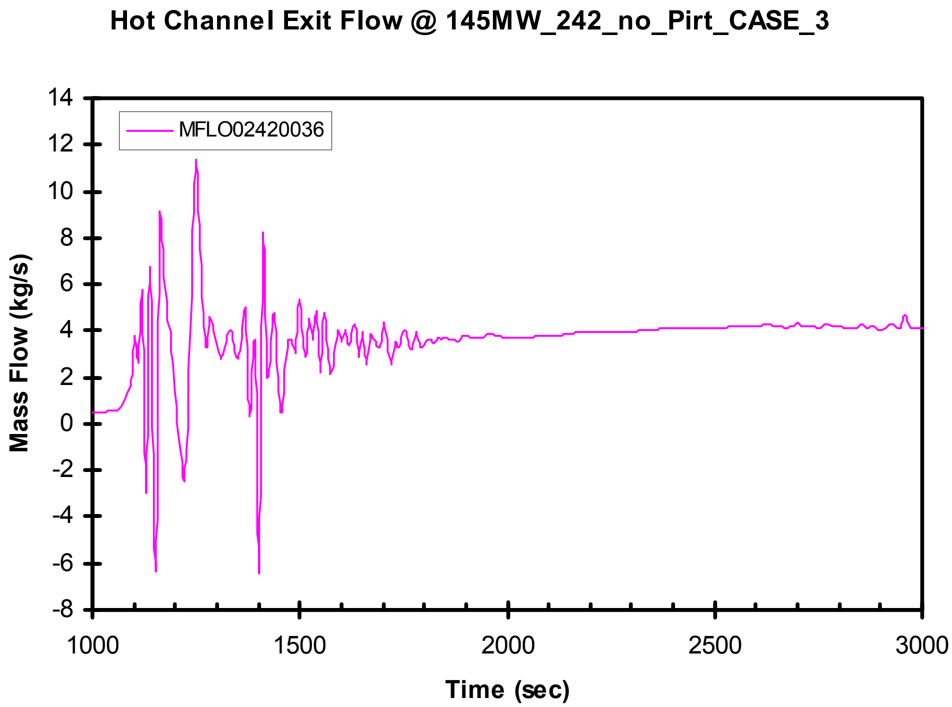



Figure 2.2-82. Hot Channel Exit Flow

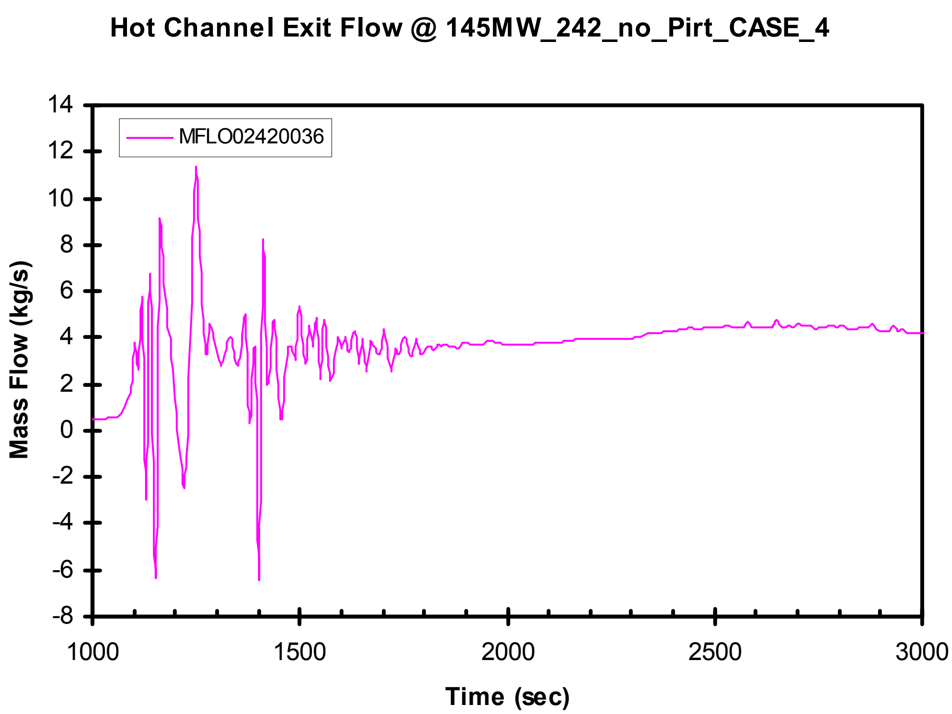



Figure 2.2-83. Hot Channel Exit Flow

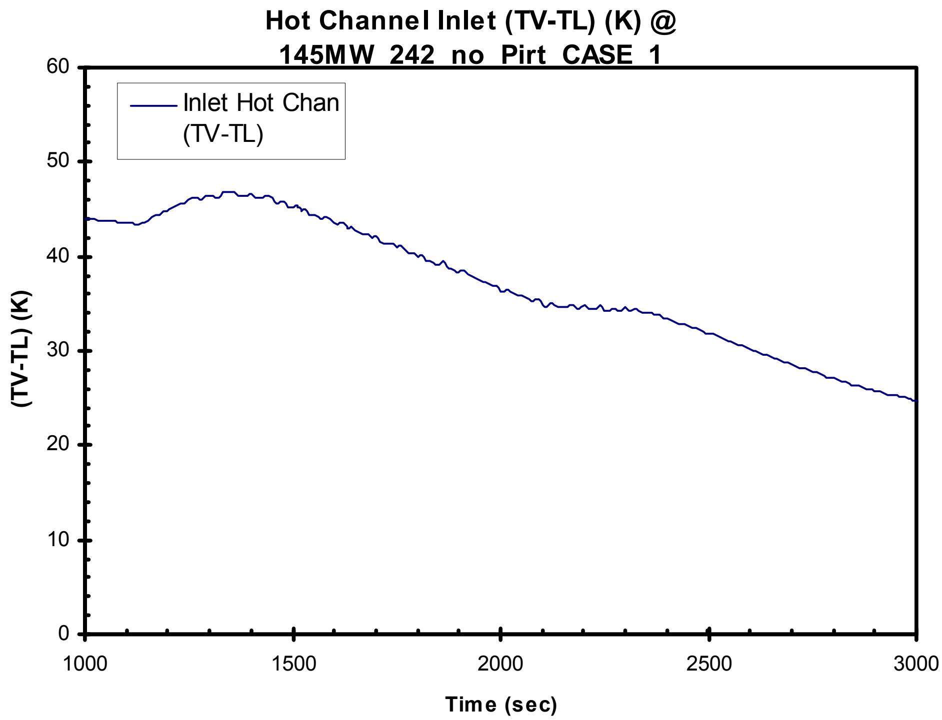



Figure 2.2-84. Hot Channel Inlet Subcooling

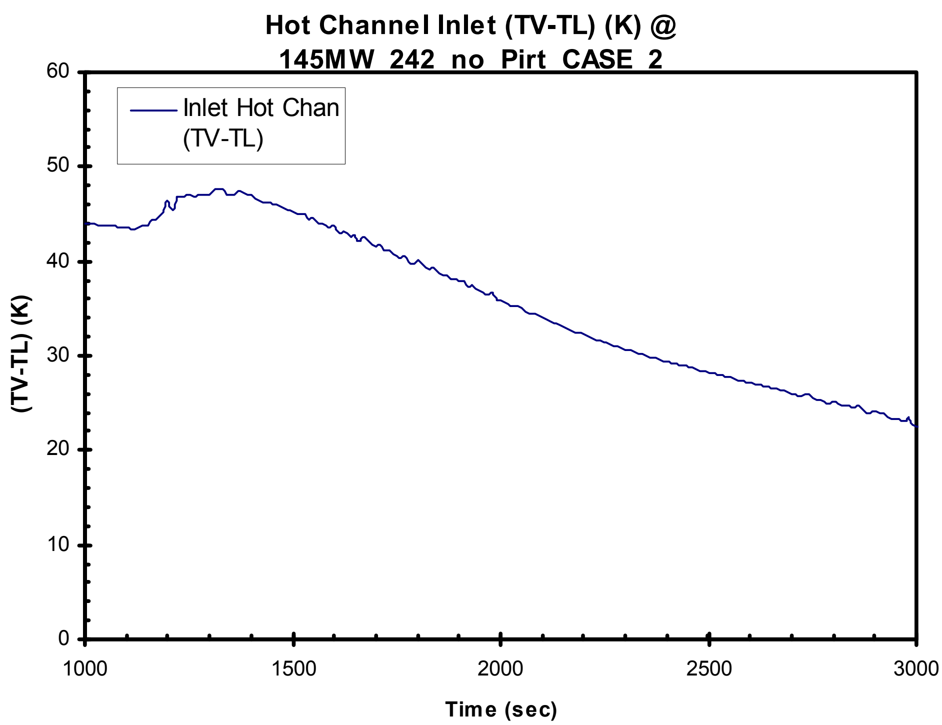



Figure 2.2-85. Hot Channel Inlet Subcooling

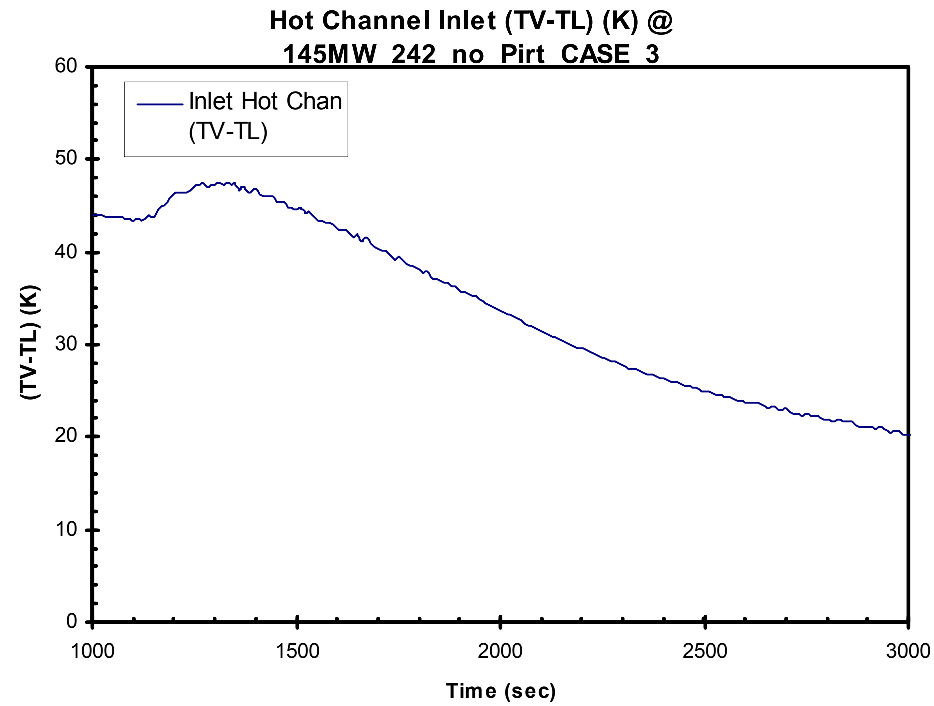



Figure 2.2-86. Hot Channel Inlet Subcooling

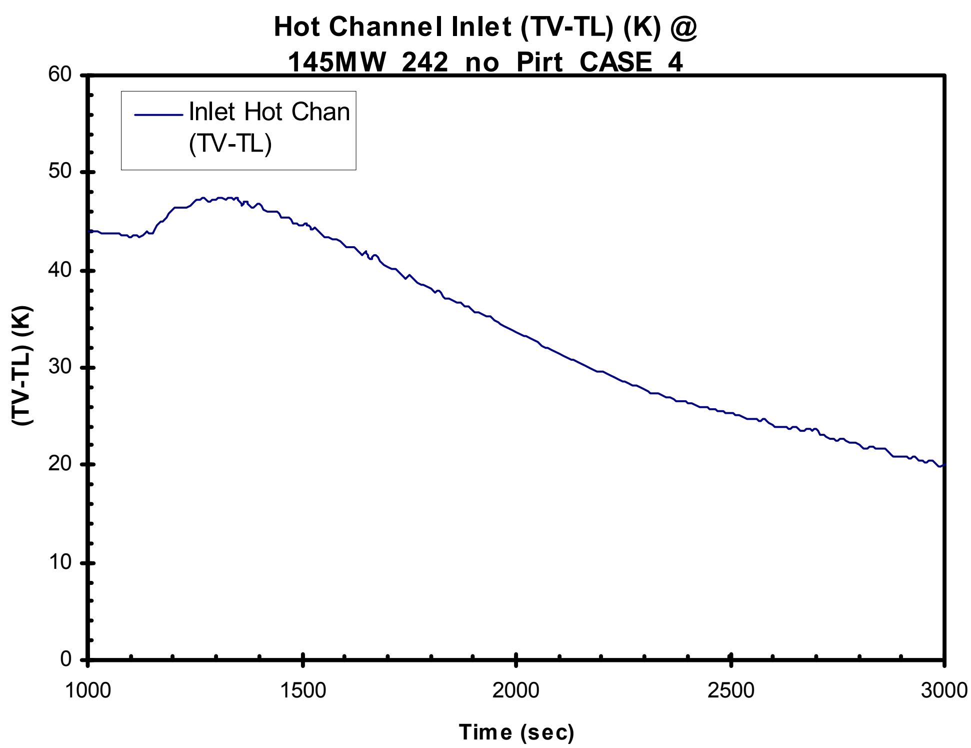



Figure 2.2-87. Hot Channel Inlet Subcooling

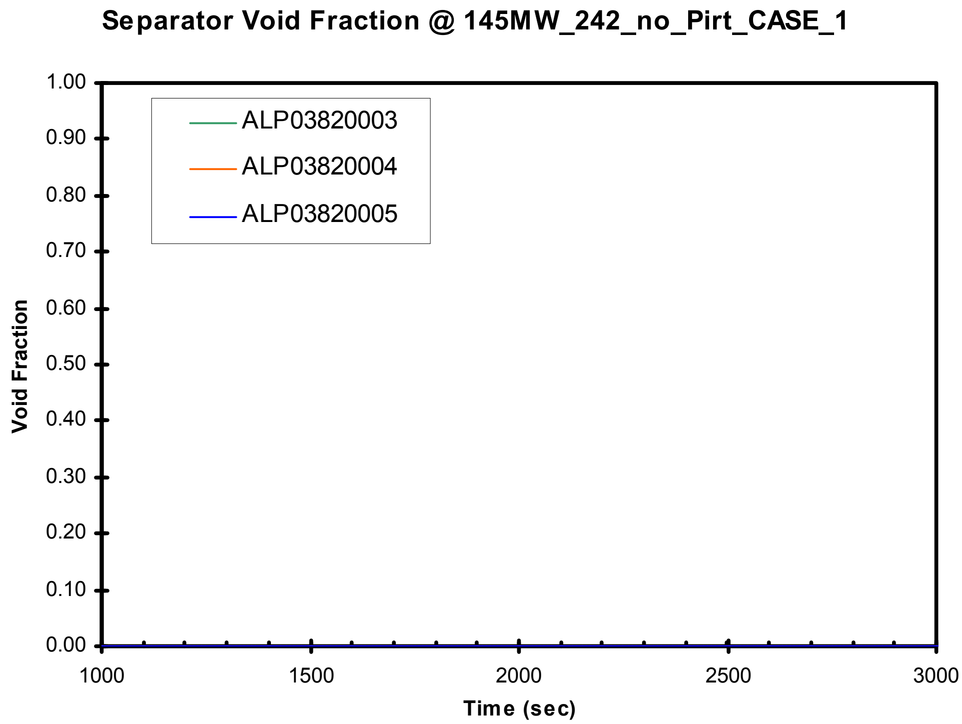



Figure 2.2-88. No voids are seen in separators.

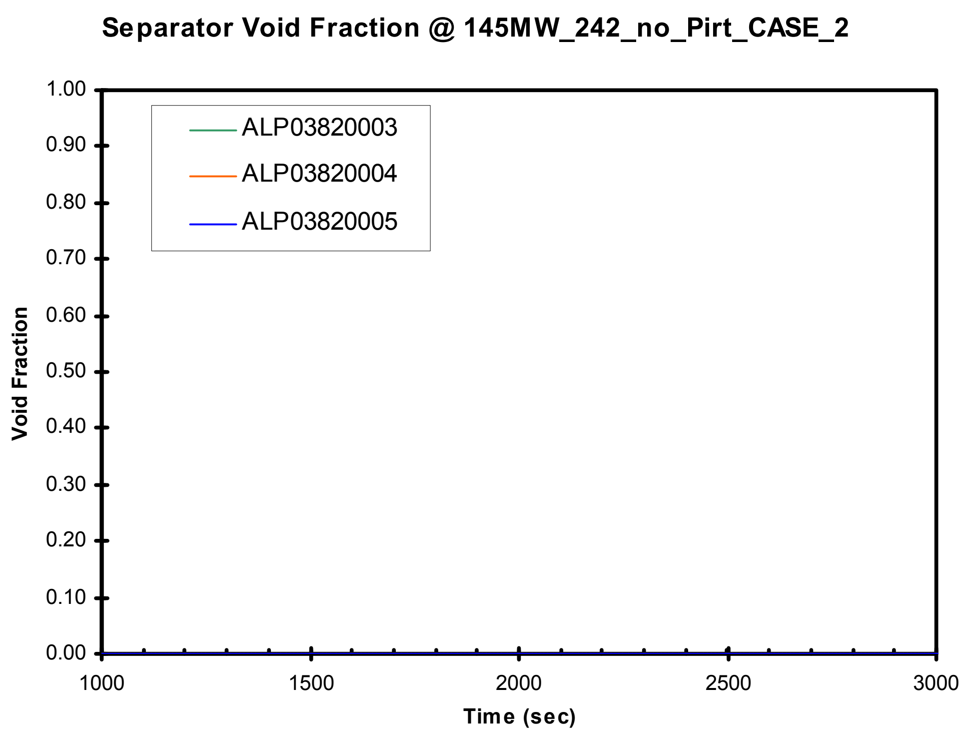



Figure 2.2-89. No voids are seen in separators.

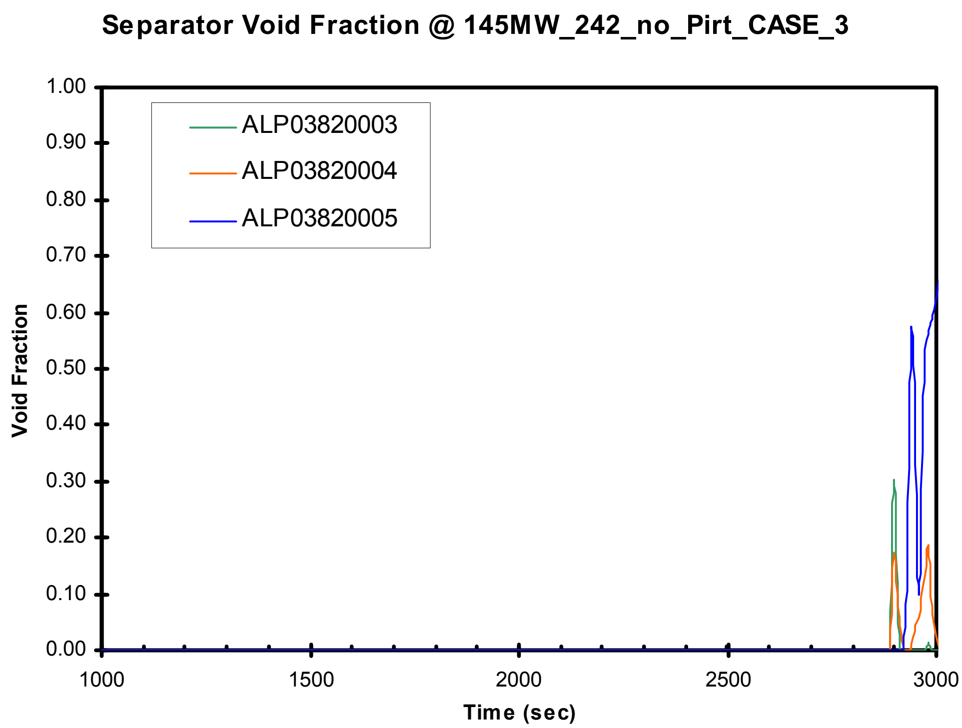



Figure 2.2-90. Separator Void Fraction

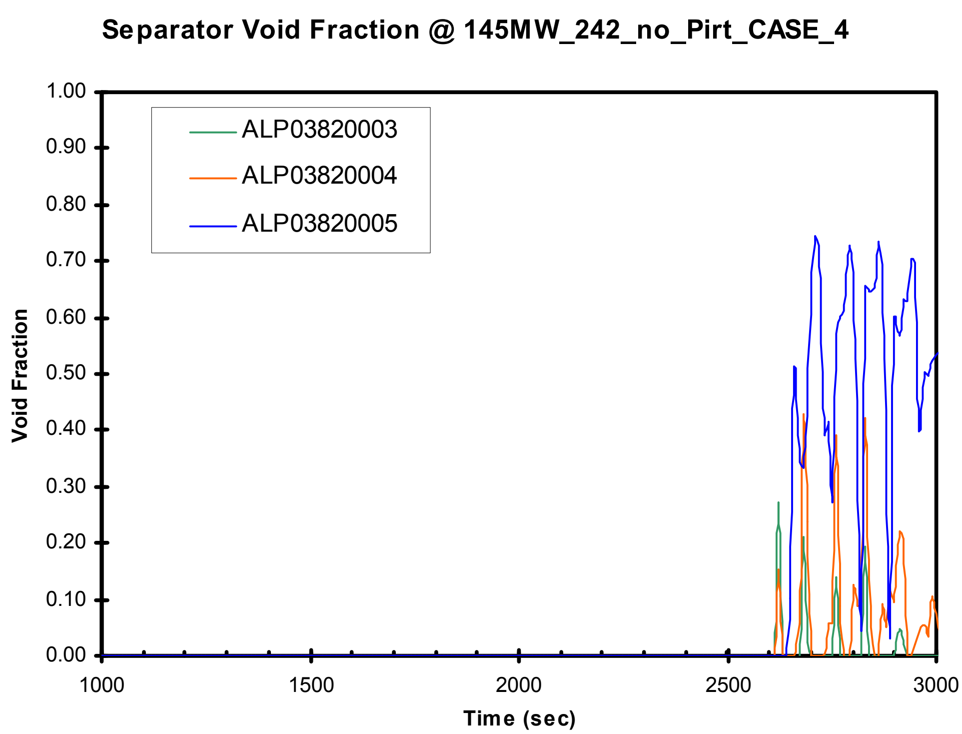



Figure 2.2-91. Separator Void Fraction

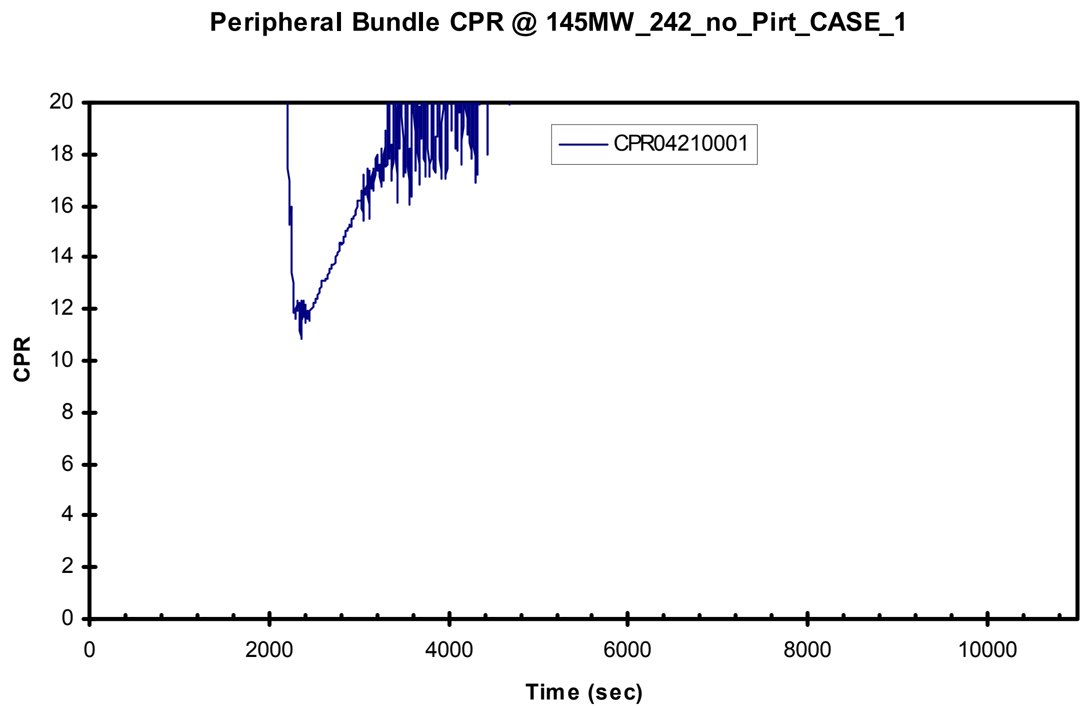



Figure 2.2-92. Peripheral Bundle CPR

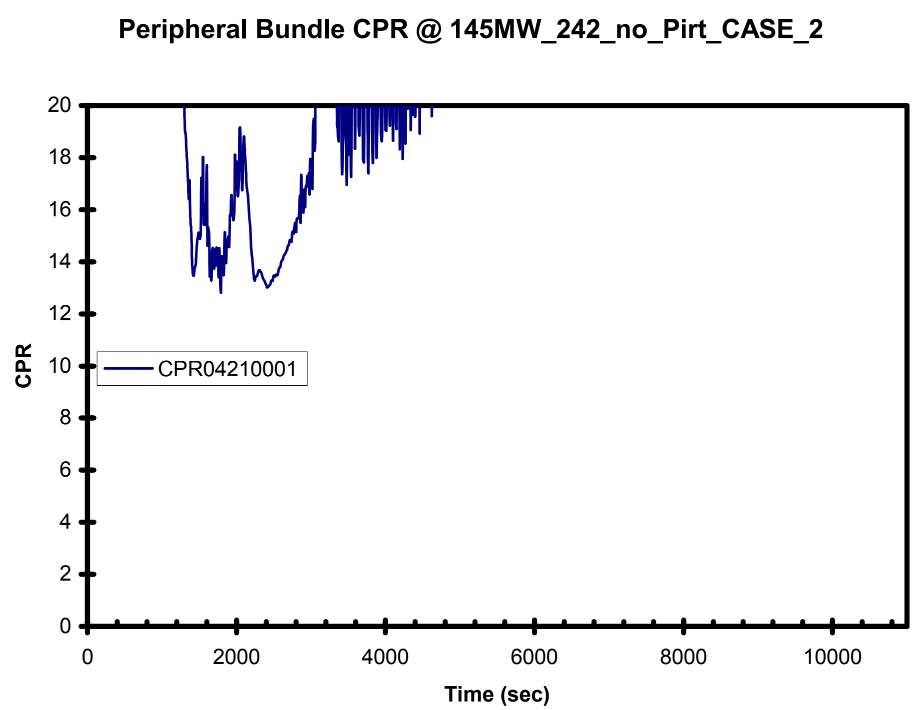



Figure 2.2-93. Peripheral Bundle CPR

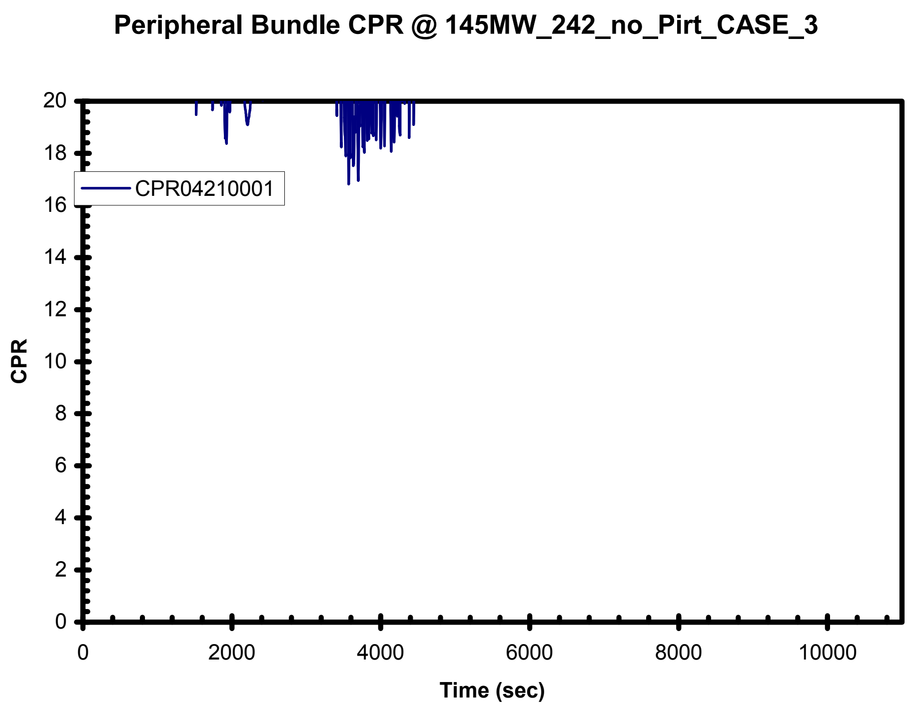



Figure 2.2-94. Peripheral Bundle CPR

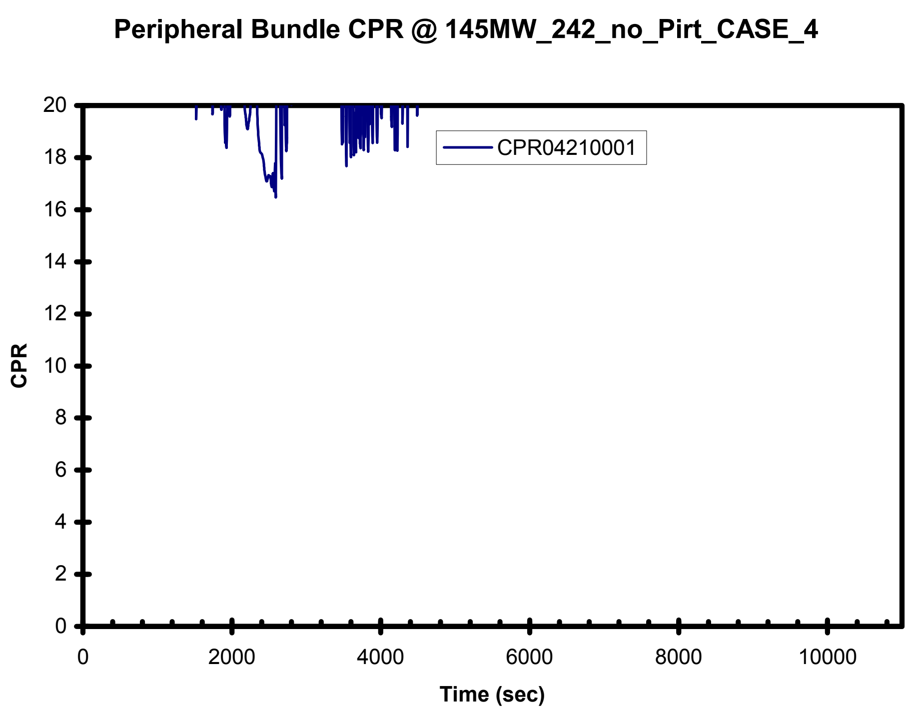



Figure 2.2-95. Peripheral Bundle CPR

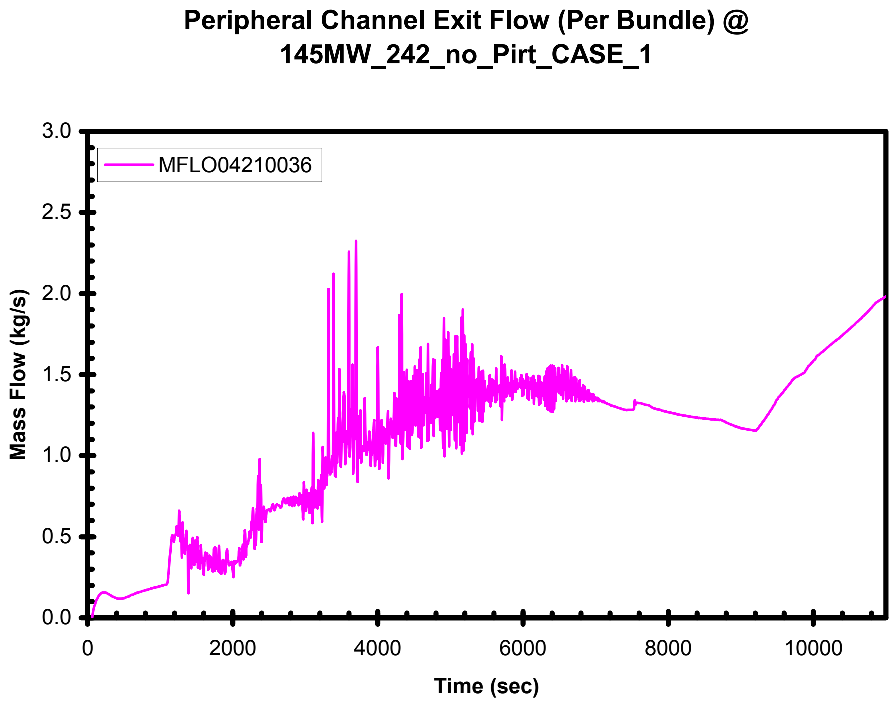



**Figure 2.2-96. Peripheral Channel Exit Flow (per Bundle)**

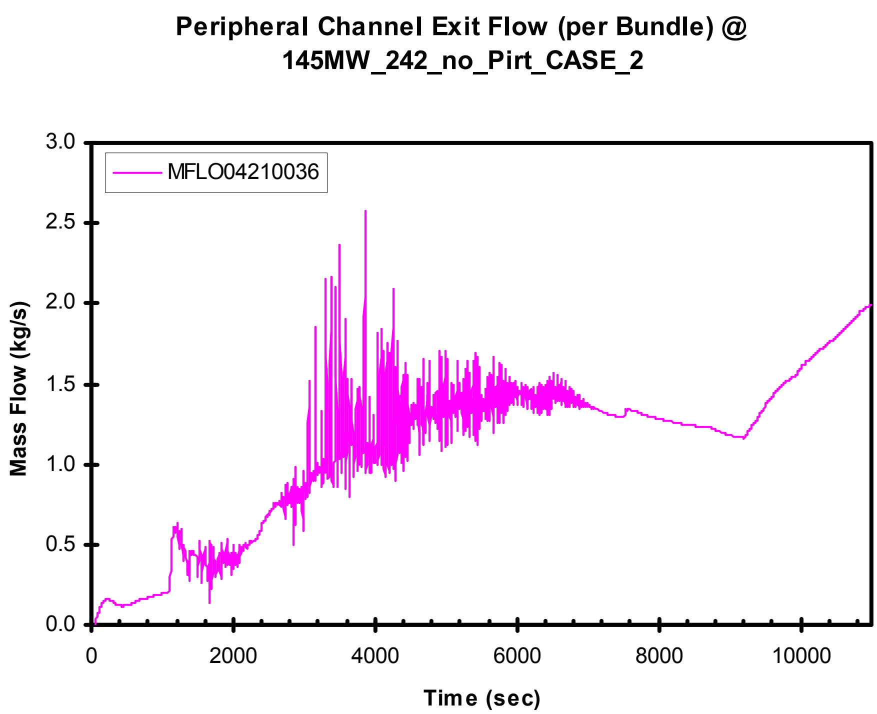



**Figure 2.2-97. Peripheral Channel Exit Flow (per Bundle)**

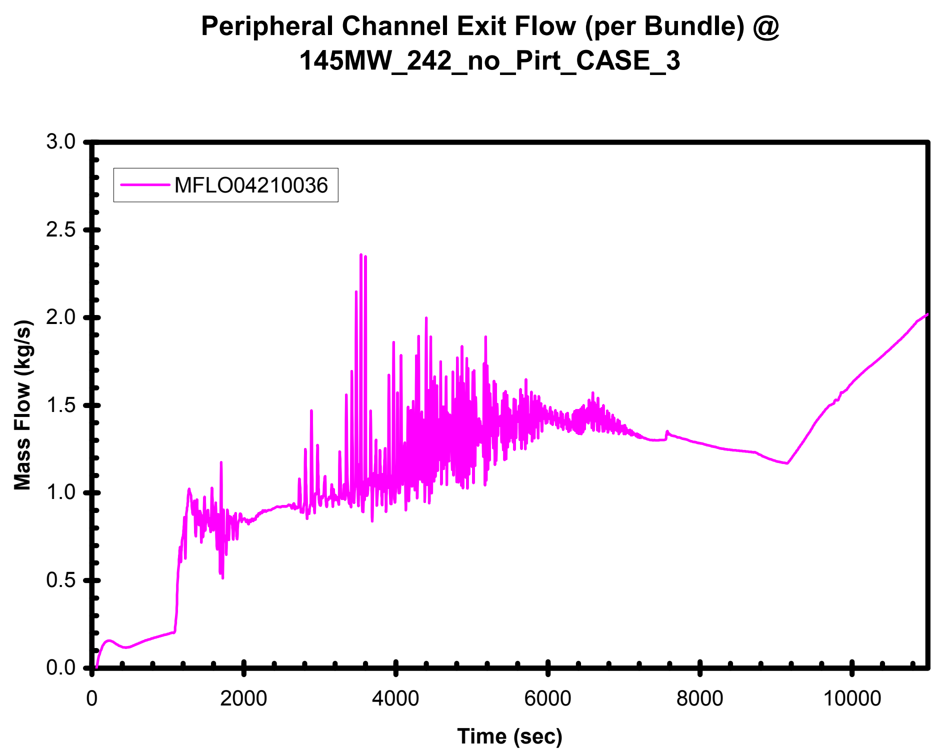



**Figure 2.2-98. Peripheral Channel Exit Flow (per Bundle)**

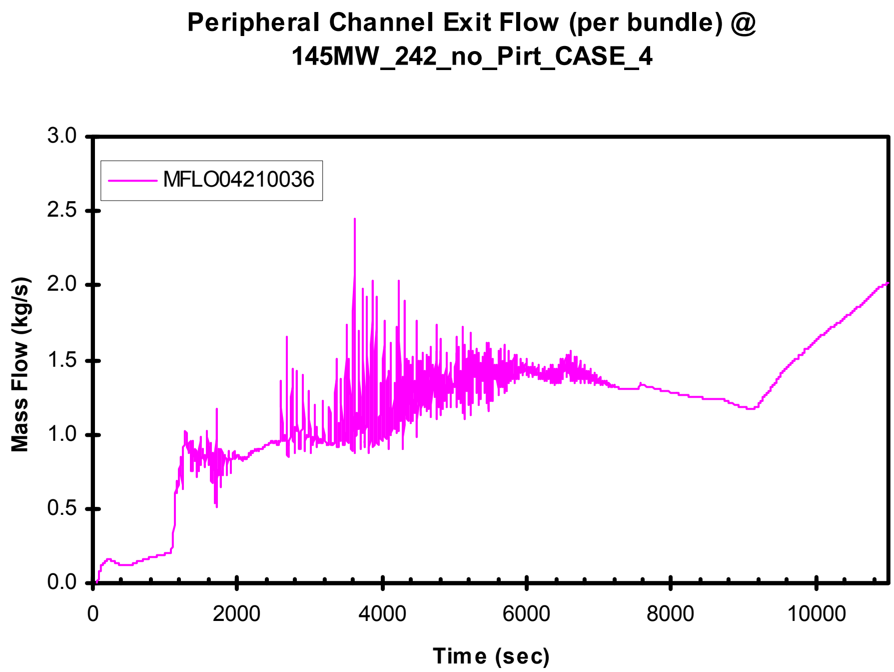



**Figure 2.2-99. Peripheral Channel Exit Flow (per Bundle)**

## Plots for Transient Xe Effects

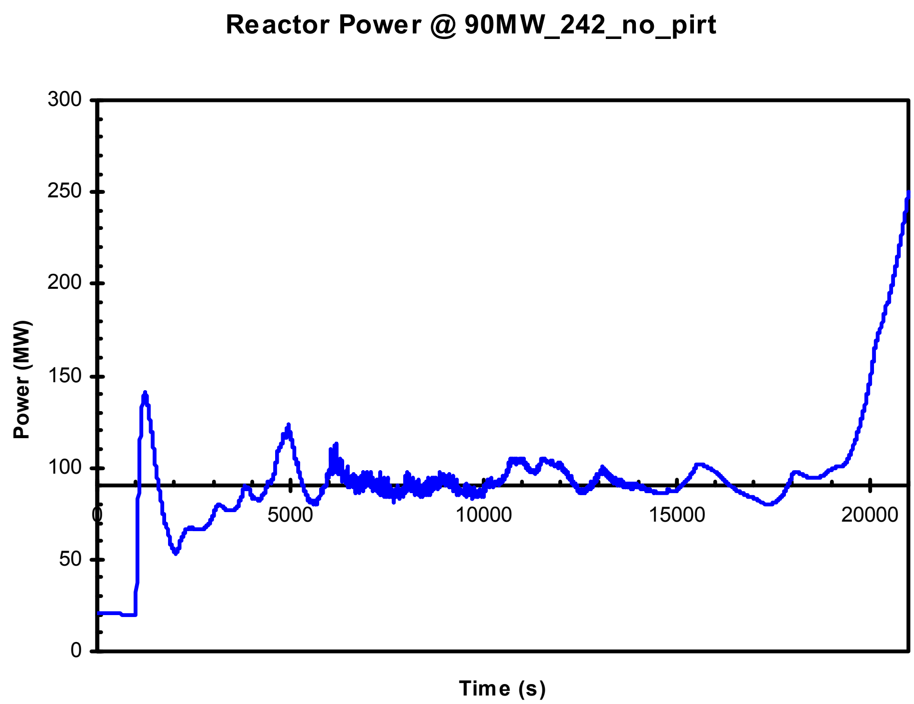



Figure 2.2-100. Reactor Power

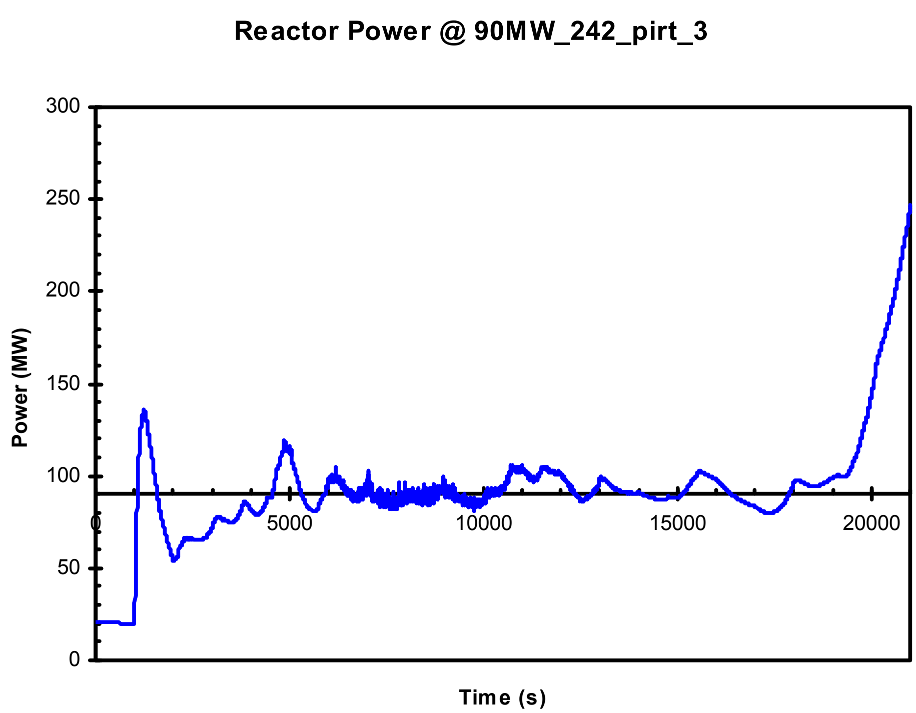



Figure 2.2-101. Reactor Power

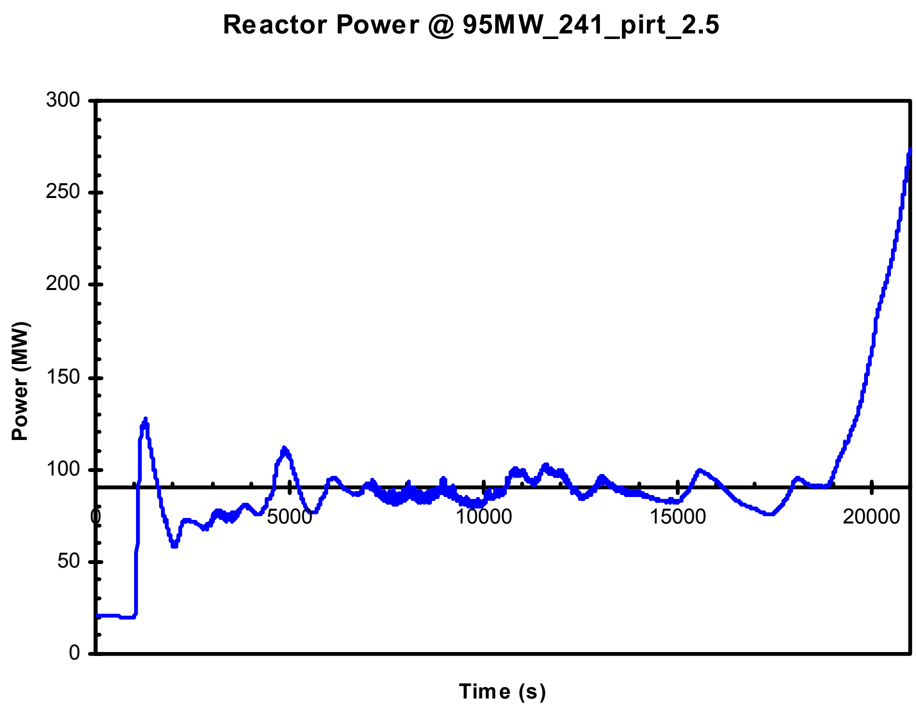



**Figure 2.2-102. Reactor Power**

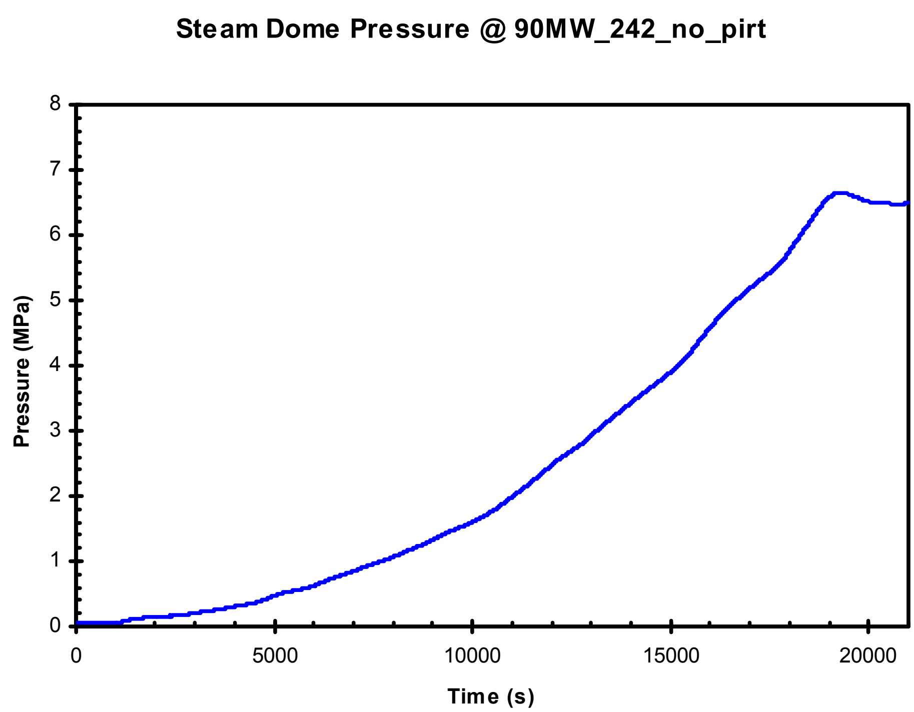



**Figure 2.2-103. Steam Dome Pressure**

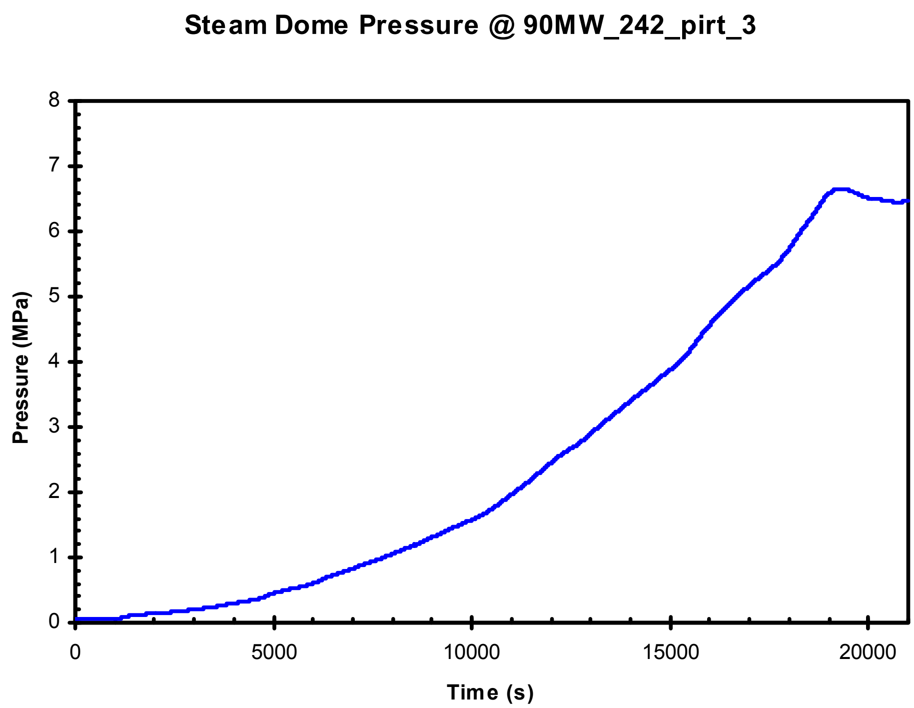



Figure 2.2-104. Steam Dome Pressure

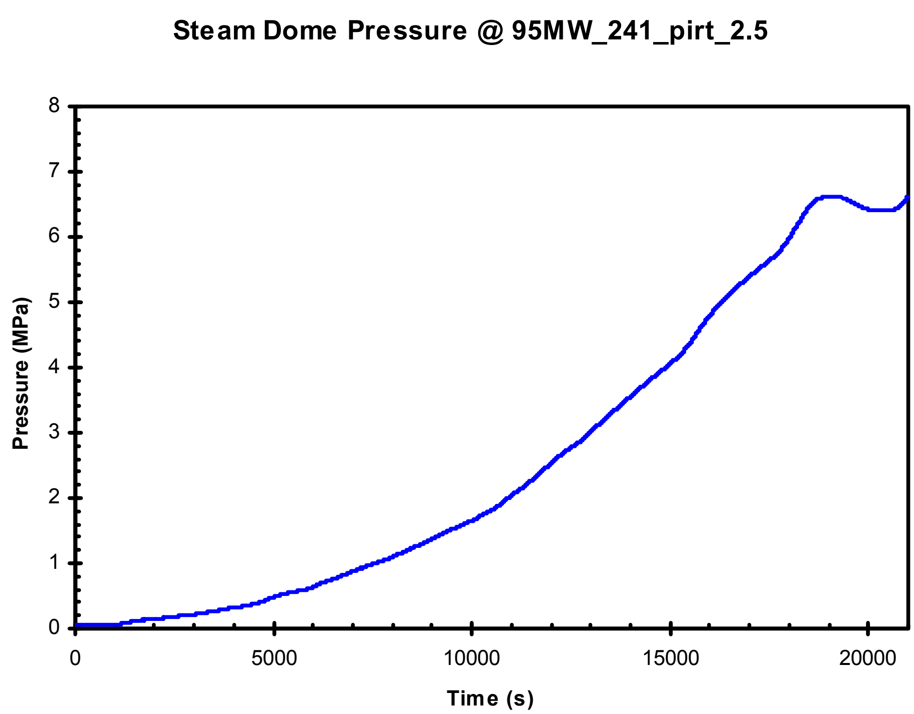



Figure 2.2-105. Steam Dome Pressure

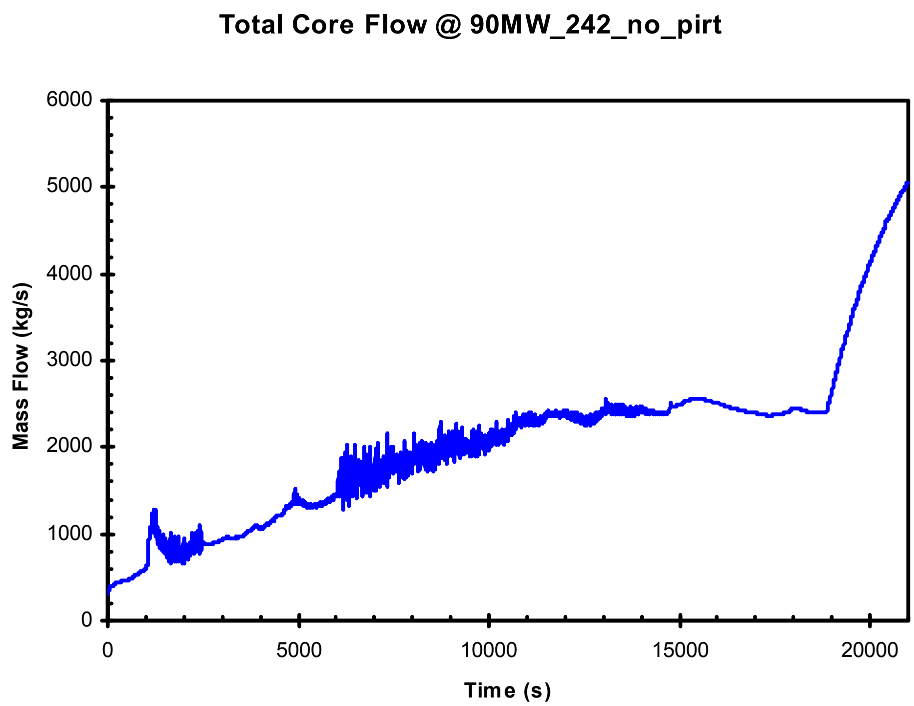



**Figure 2.2-106. Total Core Flow**

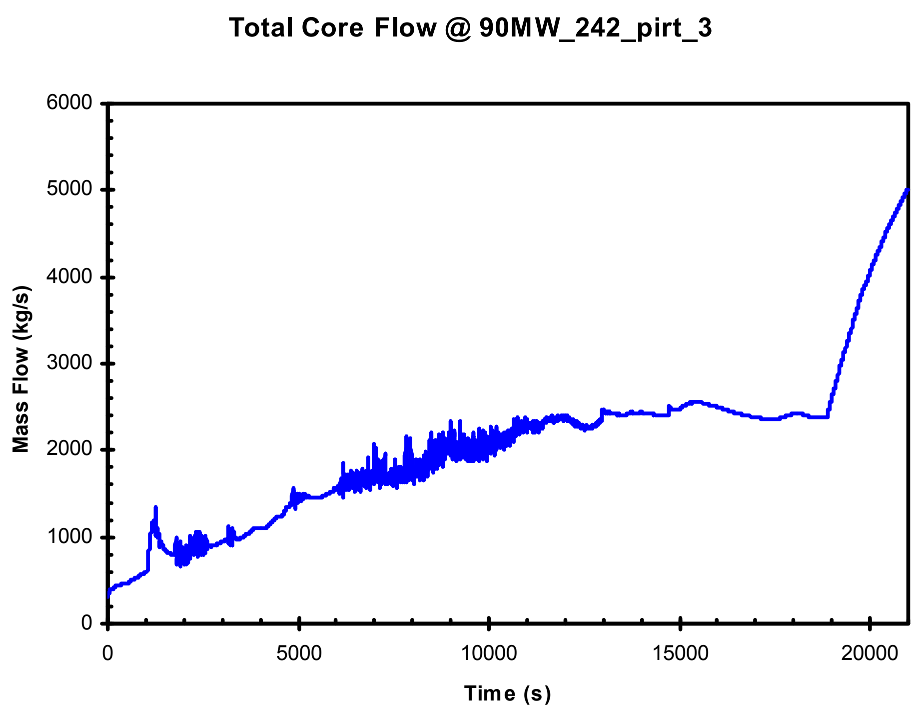



**Figure 2.2-107. Total Core Flow**

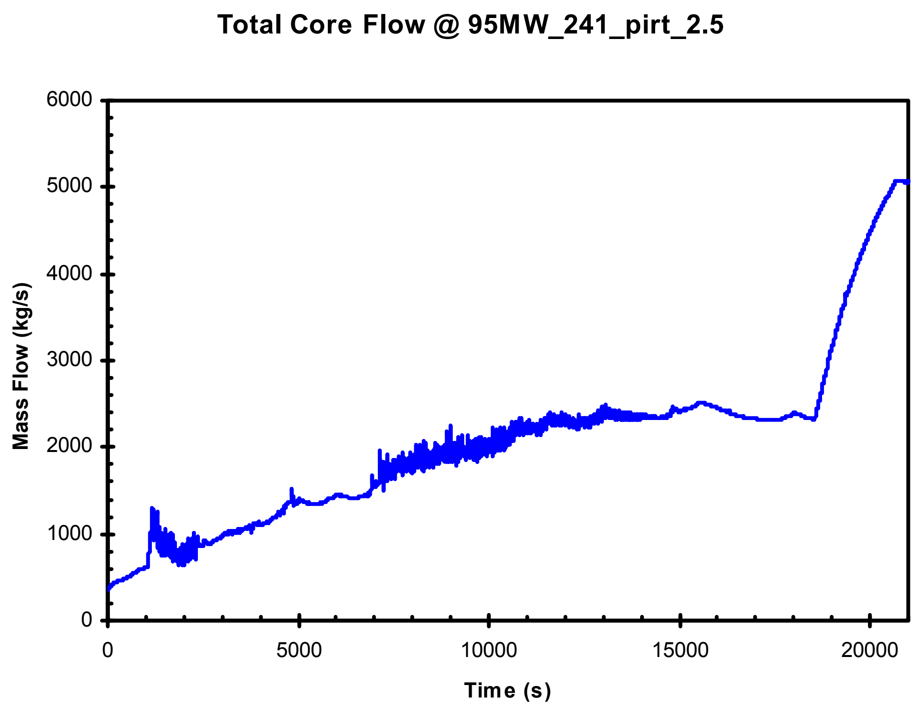



Figure 2.2-108. Total Core Flow

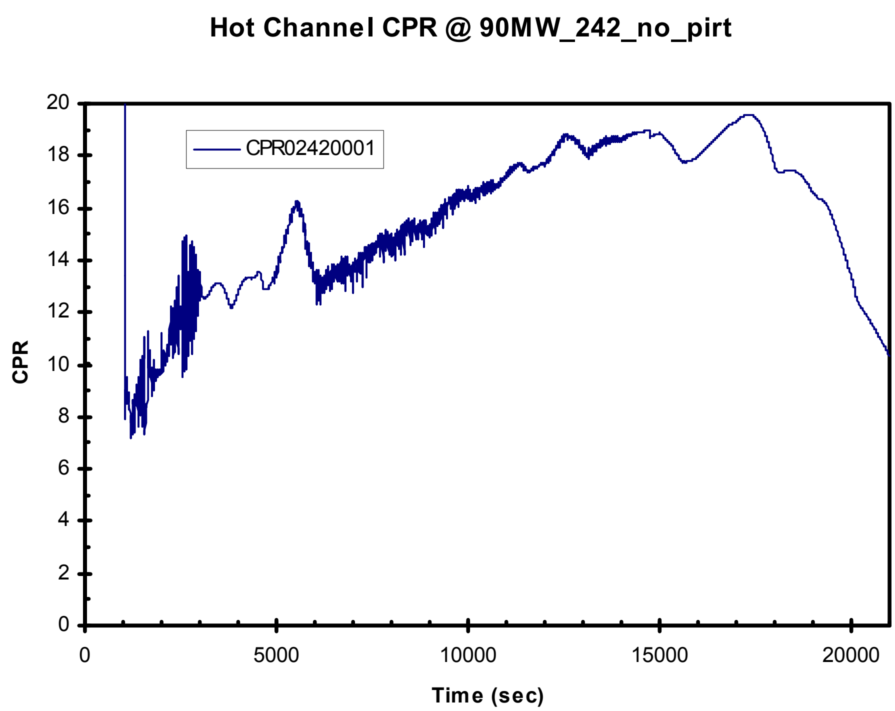



Figure 2.2-109. Hot Channel CPR

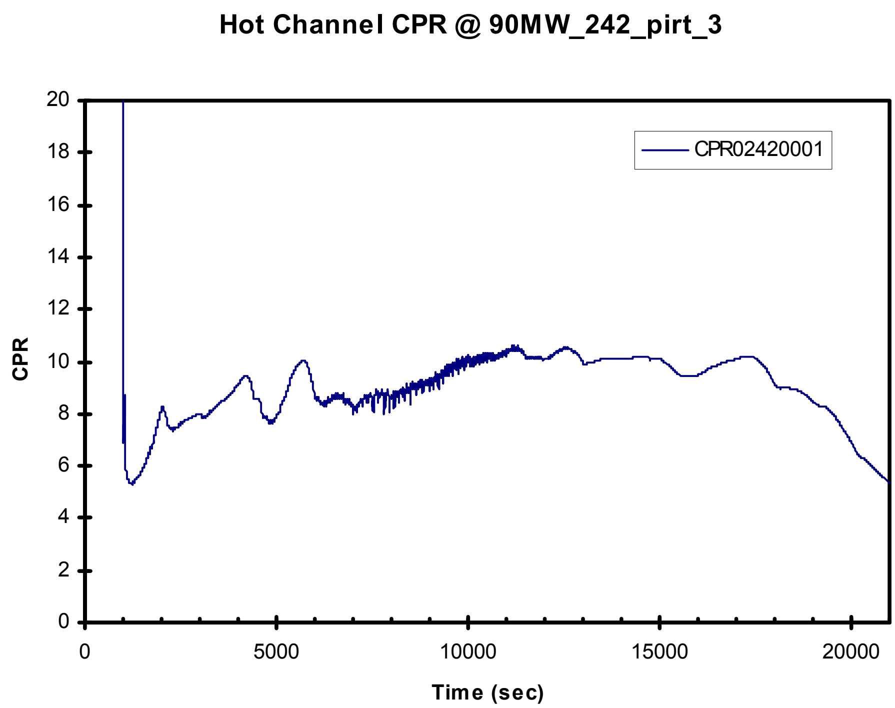



Figure 2.2-110. Hot Channel CPR

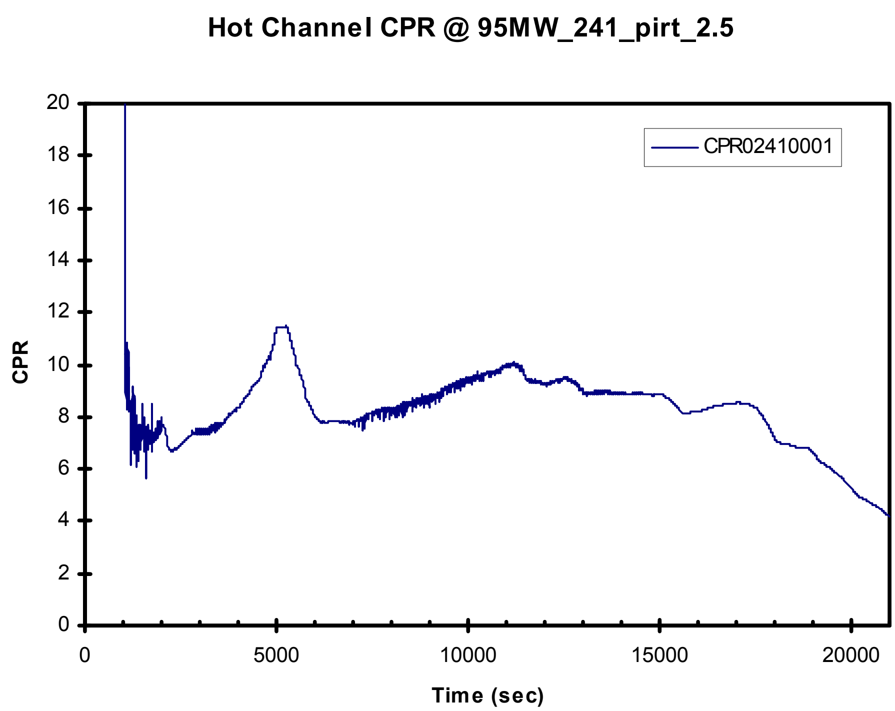



Figure 2.2-111. Hot Channel CPR

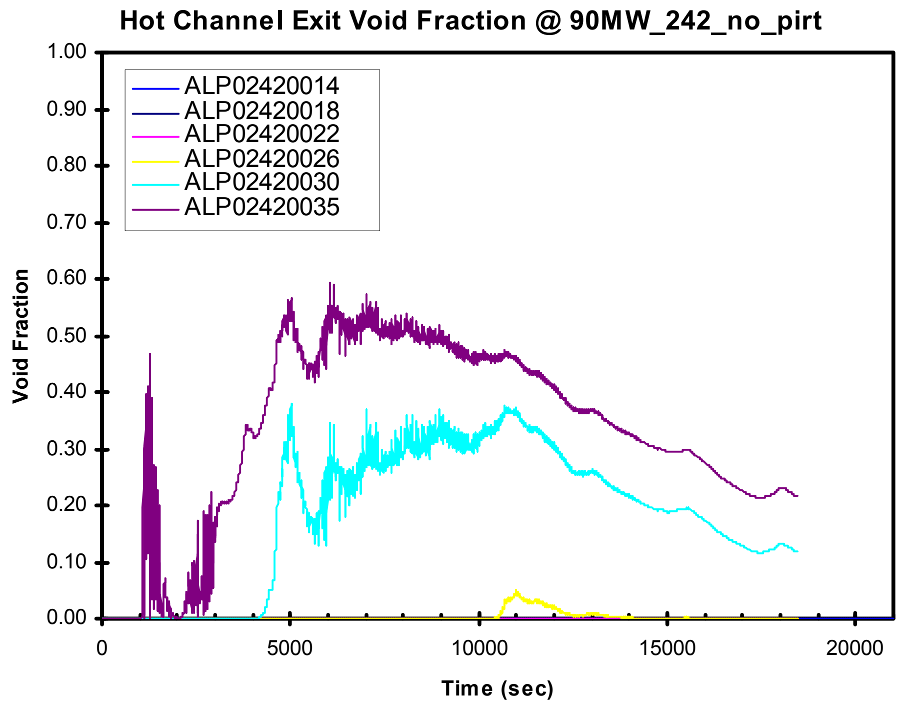



Figure 2.2-112. Hot Channel Exit Void Fraction

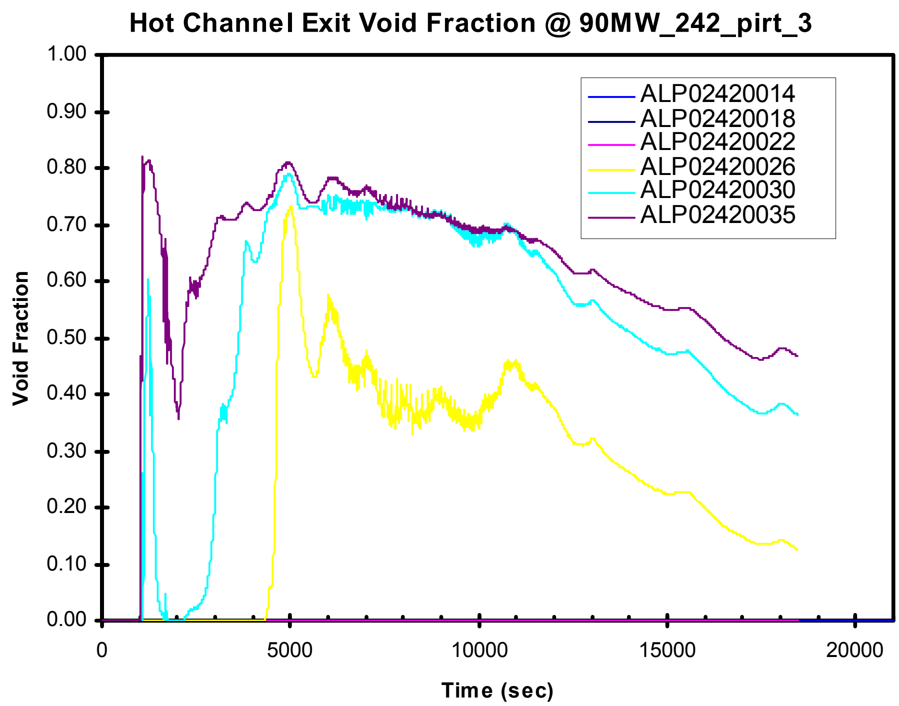



Figure 2.2-113. Hot Channel Exit Void Fraction

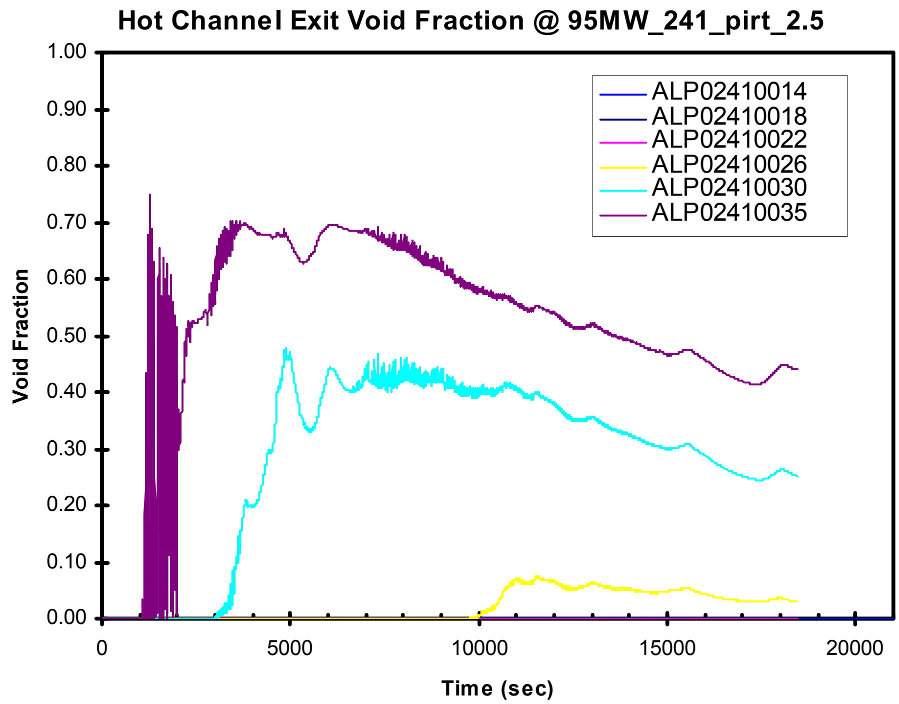



Figure 2.2-114. Hot Channel Exit Void Fraction

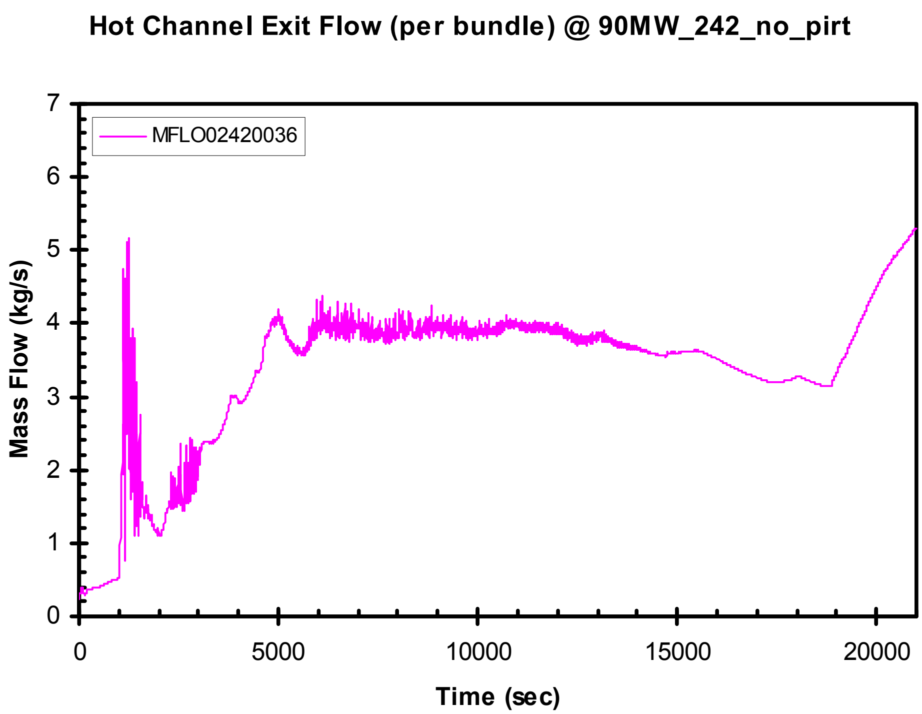



Figure 2.2-115. Hot Channel Exit Flow (per Bundle)

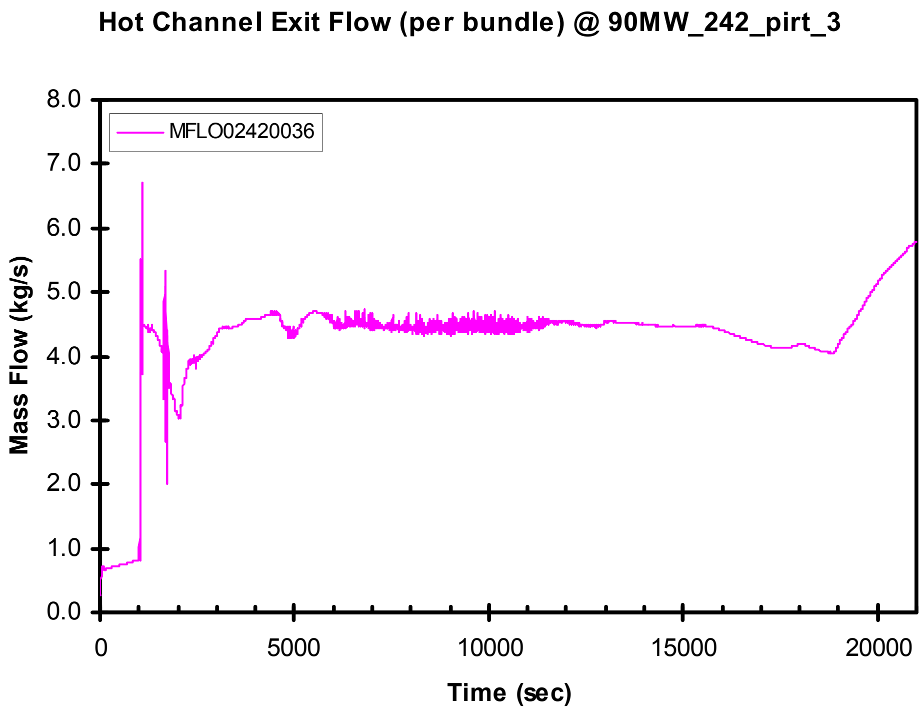



Figure 2.2-116. Hot Channel Exit Flow (per Bundle)

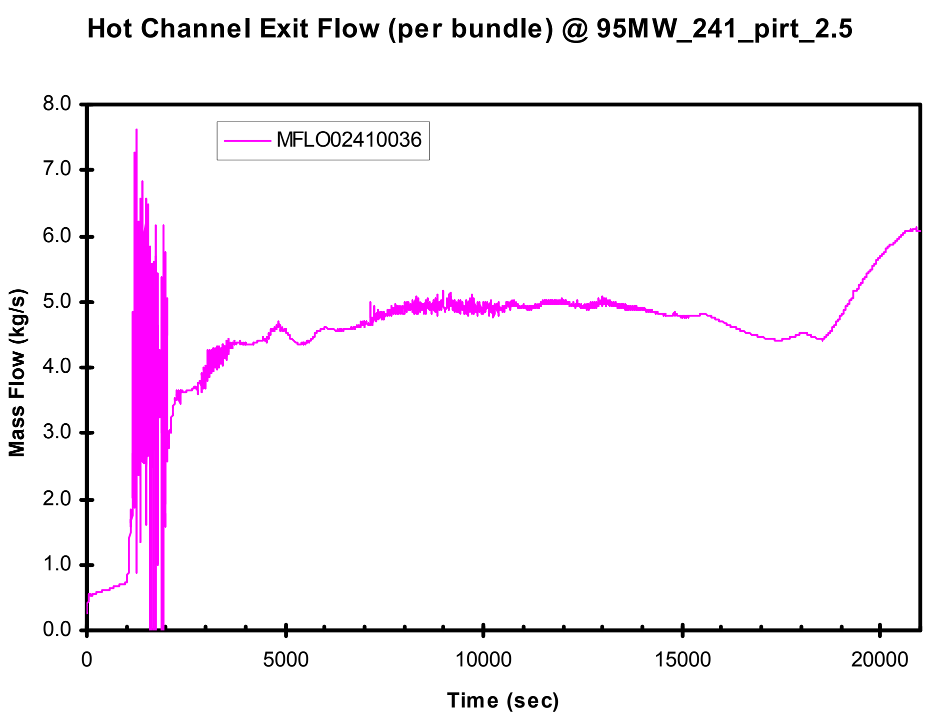



Figure 2.2-117. Hot Channel Exit Flow (per Bundle)

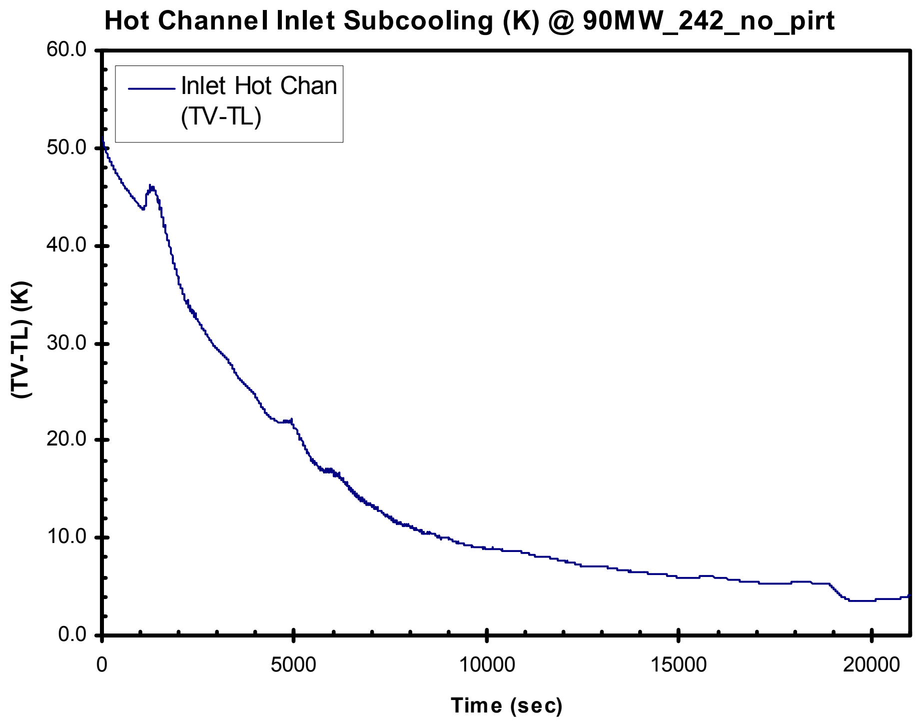



Figure 2.2-118. Hot Channel Inlet Subcooling

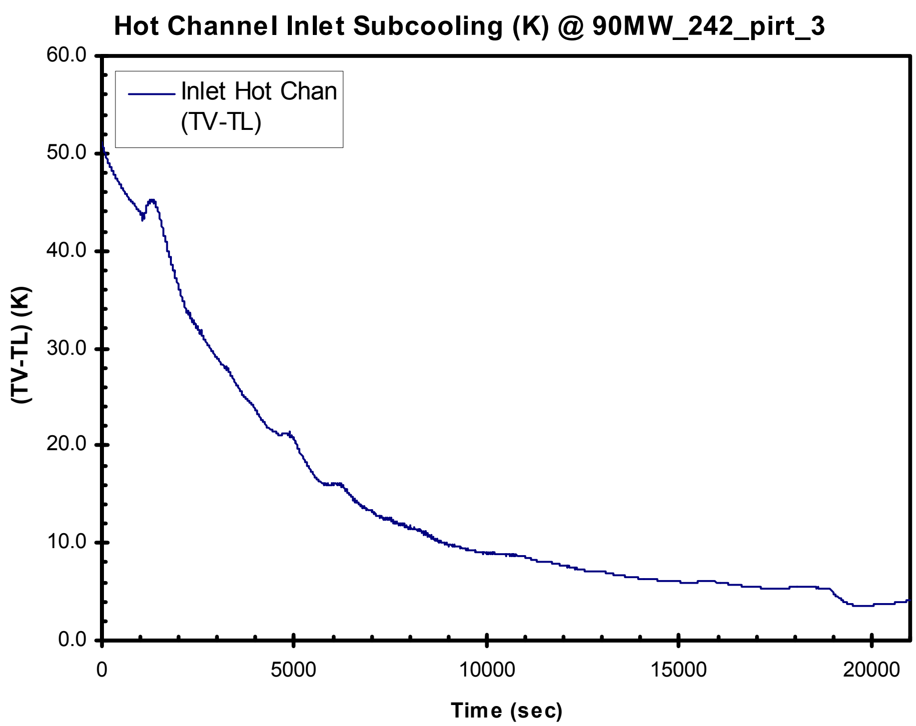



Figure 2.2-119. Hot Channel Inlet Subcooling

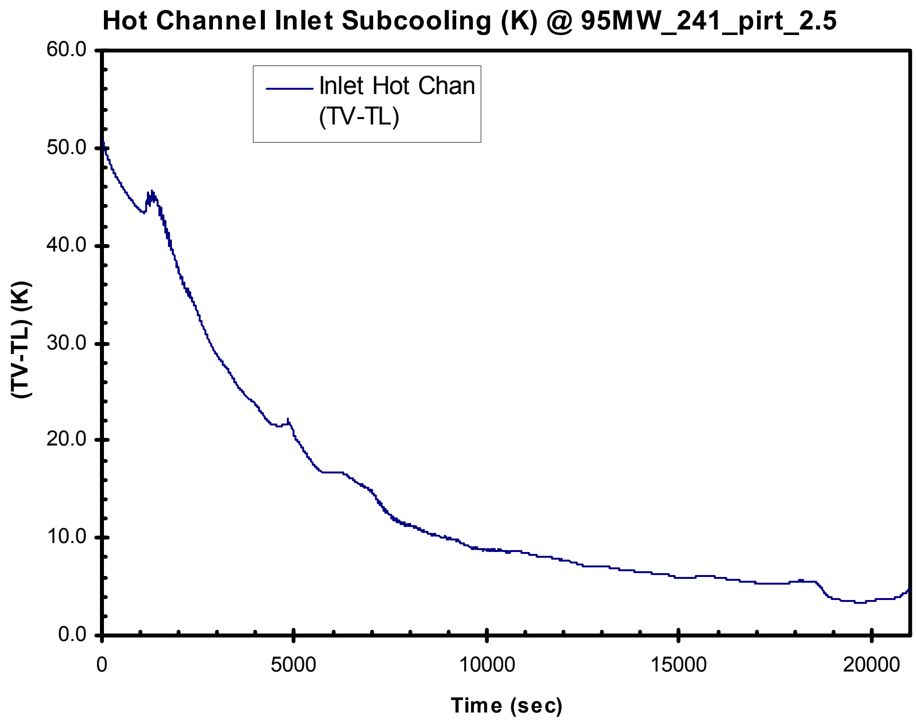



**Figure 2.2-120. Hot Channel Inlet Subcooling**

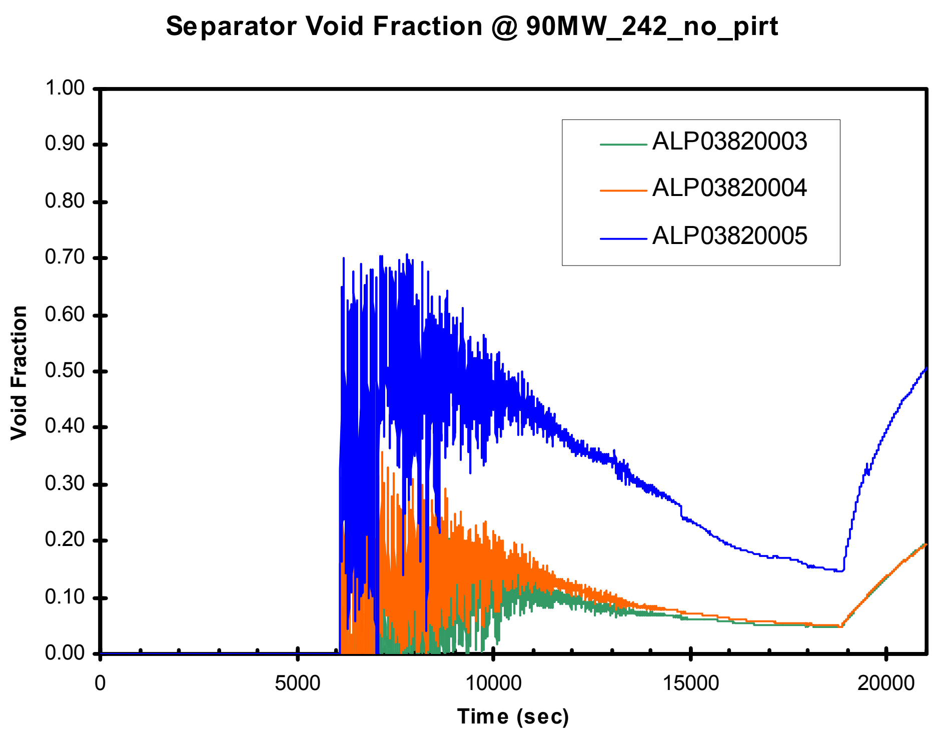



**Figure 2.2-121. Separator Void Fraction**

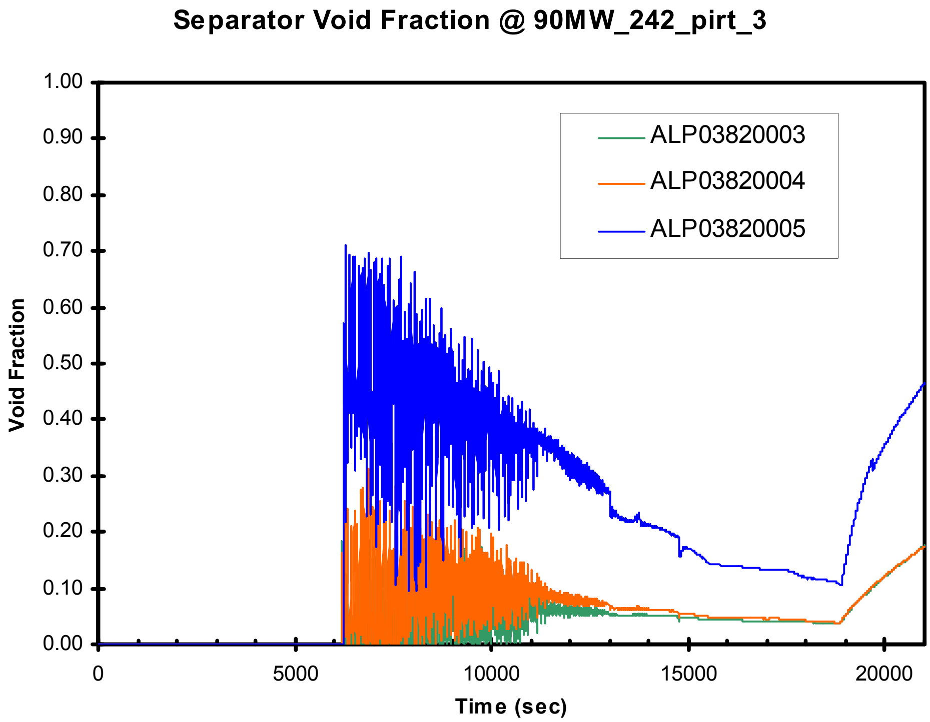



**Figure 2.2-122. Separator Void Fraction**

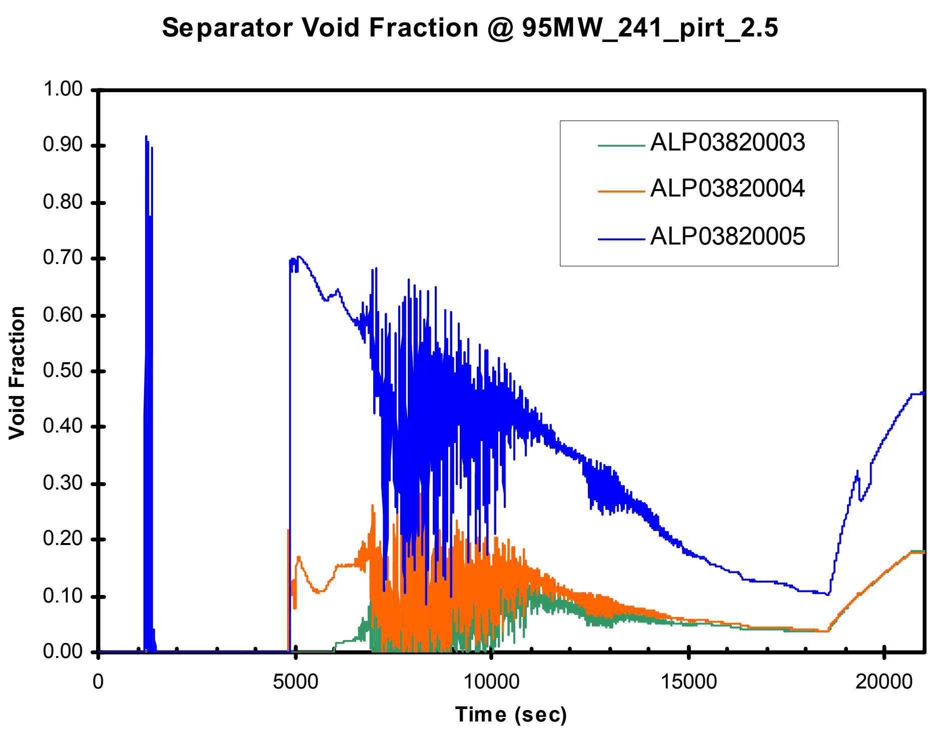



**Figure 2.2-123. Separator Void Fraction**

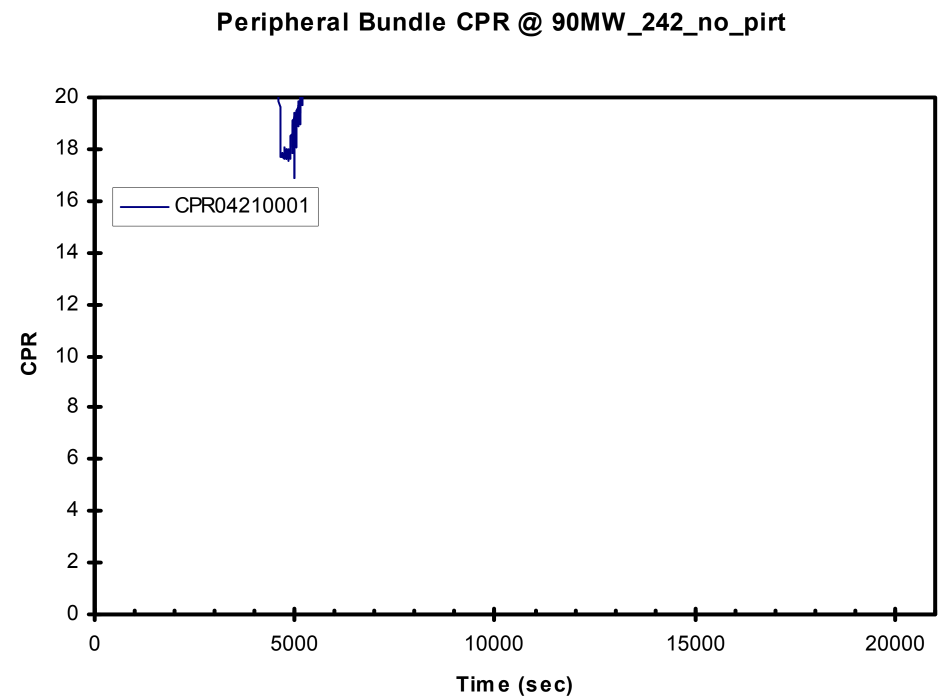



**Figure 2.2-124. Peripheral Bundle CPR**

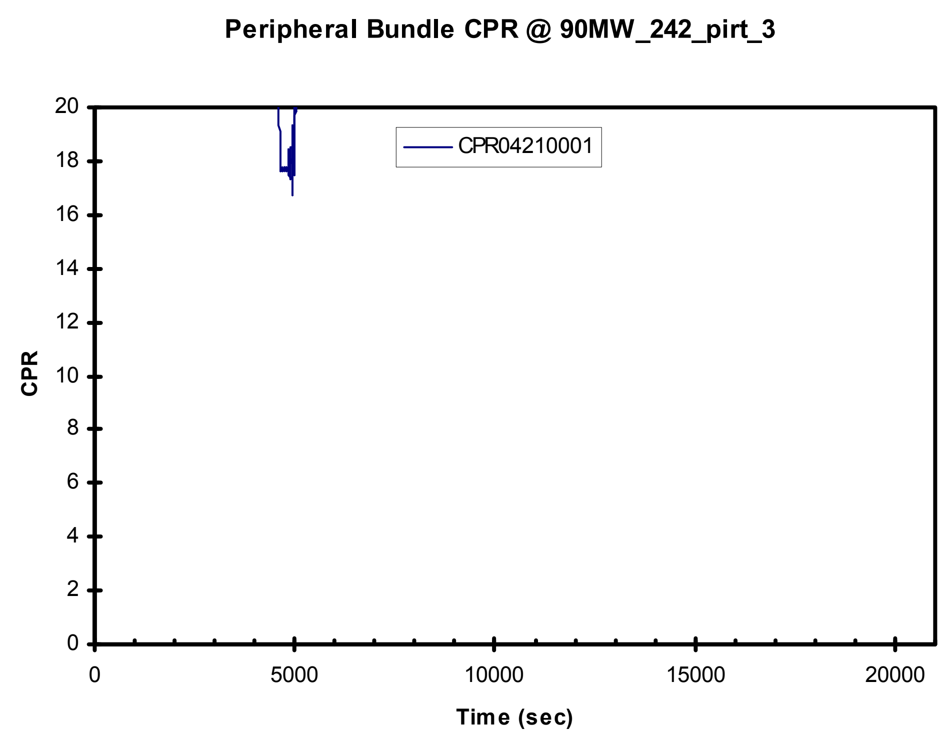



**Figure 2.2-125. Peripheral Bundle CPR**

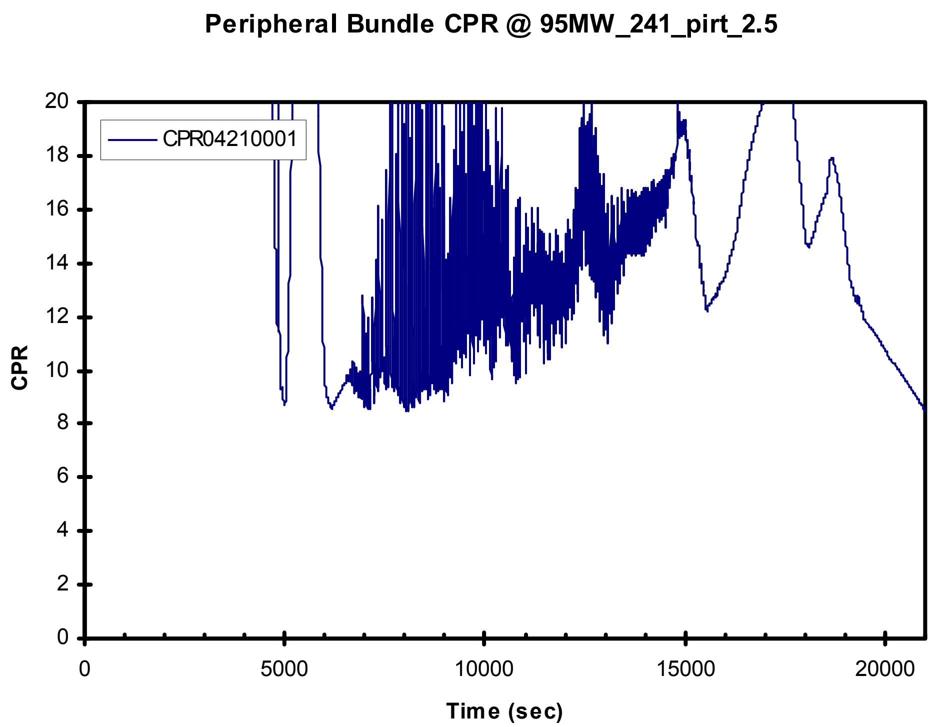



Figure 2.2-126. Peripheral Bundle CPR

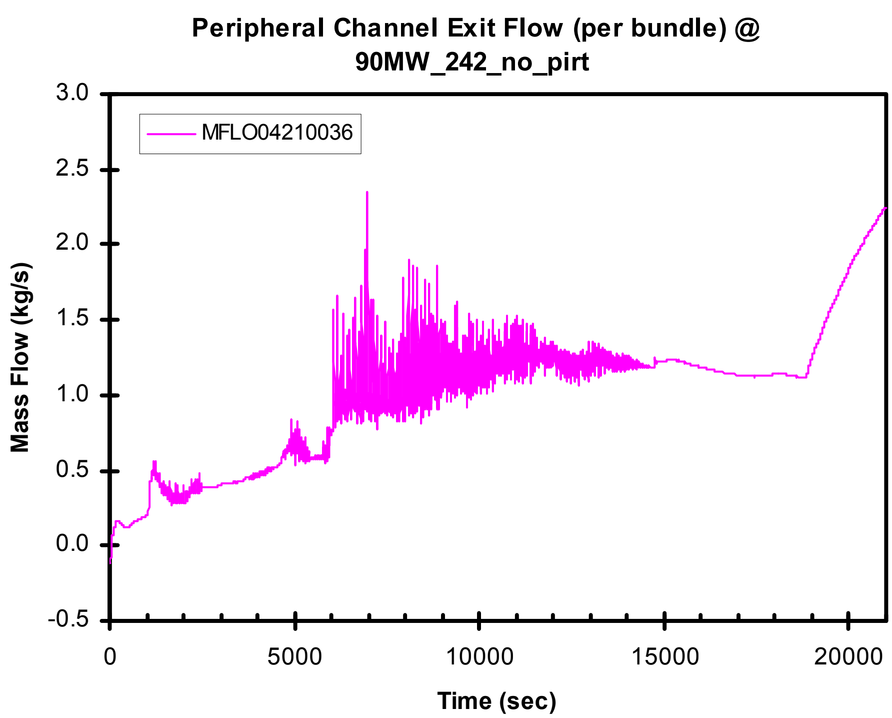



Figure 2.2-127. Peripheral Channel Exit Flow (per Bundle)

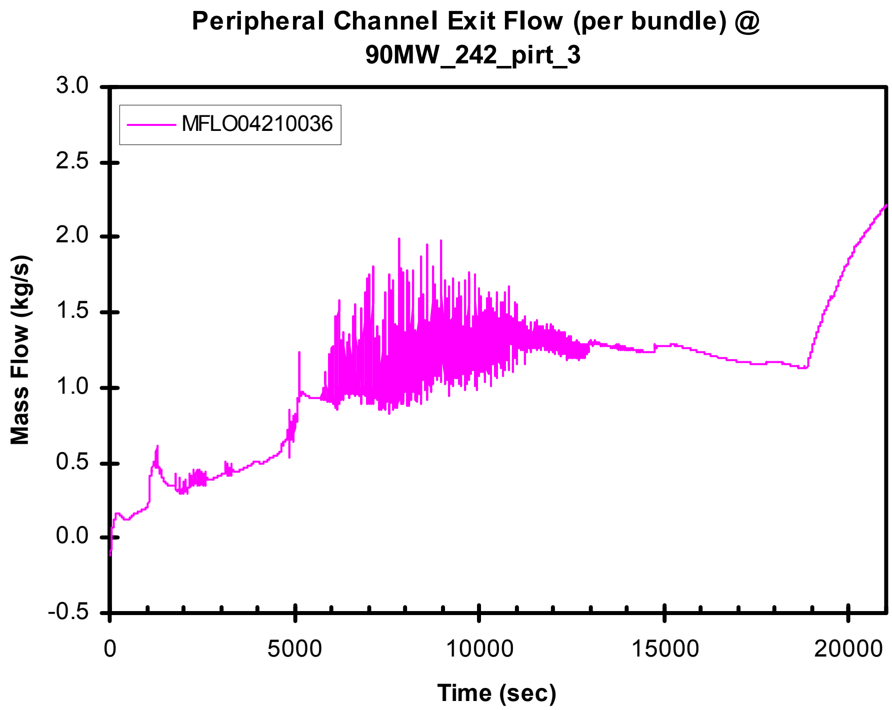



**Figure 2.2-128. Peripheral Channel Exit Flow (per Bundle)**

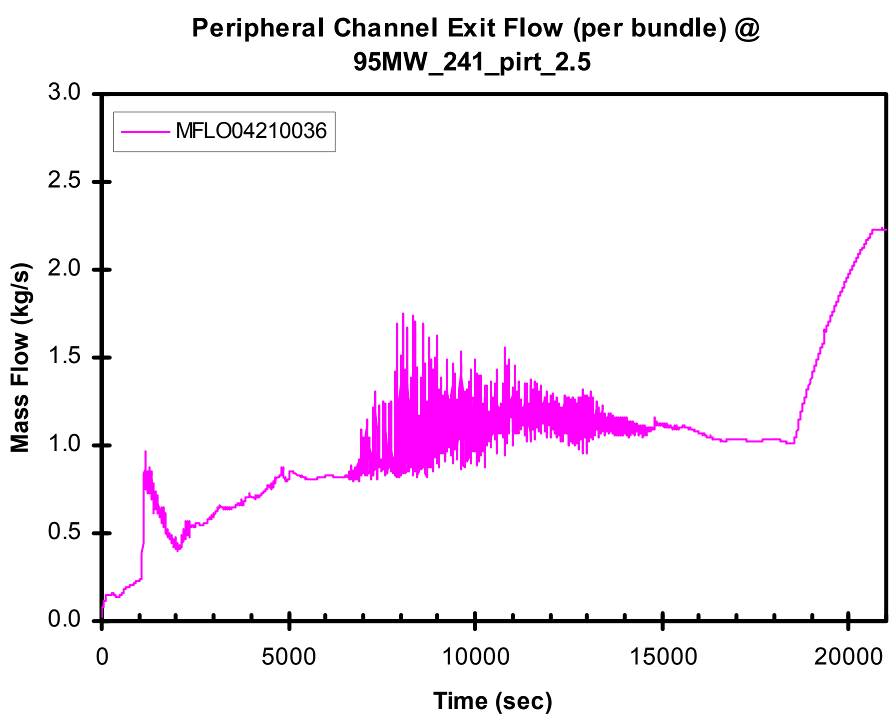



**Figure 2.2-129. Peripheral Channel Exit Flow (per Bundle)**

## 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, ESBWR DCD, 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. The summary of the AOO analyses is given in Tables 2.3-4a and Table 2.3-4b.

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.

A LOFWH that results in a significant decrease in feedwater temperature is independently detected by the ATLMs and by the Diverse Protection System (DPS), either of which mitigates the event by initiating SCRRI and SRI functions, as discussed in Subsections 7.7.2.2.7.7, 7.7.3.3 and 7.8.1.1.3 Reference 2.3-5. This prevents violating any reactor thermal limits. These functions are also collectively referred to as SCRRI/SRI.

Control rod insertion is conservatively assumed to start only when the temperature difference setpoint is reached in the FW nozzle. The SCRRI/SRI is able to suppress the neutron power increase and ensure 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. The rod pattern analyzed is divided in seven control rod groups: six SRI groups, with scattered insertion times (a separation of 15 seconds between each subgroup), and one SCRRI group. However, no SCRRI rods were assigned for this rod pattern; they are not required to show acceptable CPR results.

**2.3.1.1.2 Sequence of Events and Systems Operation**

***Sequence of Events***

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

There is no scram during this event. There is no operator action required to mitigate the event.

***Systems Operation***

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

**2.3.1.1.3 Core and System Performance**

***Input Parameters and Initial Conditions***

The event is simulated by programming a change in FW enthalpy corresponding to the assumed loss of feedwater heating, shown in Table 2.3-1.

***Results***

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

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

**2.3.1.1.4 Barrier Performance**

As noted previously, the effects of this event do not result in any temperature or pressure transient in excess of the criteria for which the fuel 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

The 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. There is no scram for this event. There is no operator action required to mitigate the event.

***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 that results in the neutron flux peak just below the scram setpoint. Faster closing times result in scram actuation and are bounded by the analyzed closure rate;
- 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 (using the bounding steamline inputs in Table 2.3-1) is presented in Figure 2.3-2. Neutron flux increases, because of the void reduction caused by the pressure increase. Using bounding steamline inputs, the calculated neutron flux reaches the high neutron flux scram setpoint; however, the scram is not credited because the peak is close to the high flux scram analytical limit. When the sensed reactor pressure increases, the pressure regulator opens the bypass valves, keeping the reactor pressure at a constant level. The calculated peak simulated thermal power 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. The neutron flux increase does not reach the high neutron flux scram setpoint. 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 bases are 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 2.3.5.1.1, 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. 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***

There is no scram during this event. There is no operator action required to mitigate the event.

*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 Generator Control System (TGCS)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 power 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

The 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***

There is no operator action required to mitigate the event.

***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 TGCS 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 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. Table 2.3-4 shows the results for this event. Although the peak neutron flux and average simulated thermal power 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. 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***

There is no scram during the event. There is no operator action required to mitigate the event.

***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 power does not increase significantly. The control system activates the SCRRI/SRI to reduce the power below 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

The 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***

There is no operator action required to mitigate the event.

***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 power 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

The 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. There is no operator action required to mitigate the event.

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 power 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

The 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.

There is no operator action required to mitigate the event.

***Systems Operation***

MSIV closure initiates a reactor scram trip via position signals to the RPS. The same signal also initiates the operation of isolation condensers, 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 power shows no increase. The FW injection and the isolation condenser 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

The 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***

There is no operator action required to mitigate the event.

***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 and the peak average simulated thermal power are shown in Table 2.3-4. 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

#### *2.3.3.1 Control Rod Withdrawal Error During Startup*

Analysis in the DCD is performed with the initial core.

#### *2.3.3.2 Control Rod Withdrawal Error During Power Operation*

Analysis in the DCD applies to the initial core. ATLM stops rod movements and prevents violation of thermal operating limits.

### 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 ICS 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***

There is no scram during the event. There is no operator action required to mitigate the event.

***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 automatic reactor water level control 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 ICS water temperature and enthalpy, startup time and flow rate are provided in Table 2.3-1.

Inadvertent startup of all loops of the ICS 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 isolation condenser loop flow is established. No delays are considered because they 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 isolation condenser 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

#### 2.3.4.1.4 Barrier Performance

Inadvertent Startup of the isolation condenser 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

The 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***

There is no scram during the event. There is no operator action required to mitigate the event.

***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 maximum flow for a runout of one FW pump 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

The 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. There is no scram during the event. There is no operator action required to mitigate the event.

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

The 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 Non-Emergency AC Power to Station Auxiliaries, Table 2.3-21 lists the sequence of events for Figure 2.3-15.

***Identification of Operator Actions***

There is no operator action required to mitigate the event.

***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 isolation condensers. 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

The 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***

There is no operator action required to mitigate the event.

***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 in Subsection 15.2.6 of the DCD. 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 of Reference 2.3-5 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 Hitachi Nuclear Energy, "TRACG Application for ESBWR Transient Analysis," NEDE-33083P Supplement 3, Class III, Revision 0, December 2007, NEDO-33083 Supplement 3, Class I, Revision 0, December 2007.

2.3-2 Deleted

2.3-3 Global Nuclear Fuel, "ESBWR Initial Core Nuclear Design Report", NEDC-33326-P, Class III (Proprietary), Revision 1, March 2009, NEDO-33326, Class I (Non-proprietary), Revision 1, March 2009.

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

2.3-5 GE Nuclear Energy, "ESBWR Design Control Document" 26A6642.

**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 |
| Core Flow kg/s (Mlbm/hr) (1) | 9920 (78.73) |
| Reference Core Flow kg/s (Mlbm/hr) (1) | 10250 (81.35) |
| Steam Flow, kg/s (Mlbm/hr) (1)<br>Analysis Value | 2434 (19.32) |
| Reference Steam Flow, kg/s (Mlbm/hr) (1) | 2432 (19.3) |
| Feedwater Flow Rate, kg/s (Mlbm/hr) (1)<br>Analysis Value<br>Total Flow For All Pumps Runout, % of rated at 7.34 MPaG (1065 psig) (At rated dome pressure, 7.07 MPaG [1025 psig]).<br>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.<br>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. | <br>2427 (19.3)<br>155 (164) |
| Feedwater Temperature<br>Rated, °C (°F)<br>FW Heating Temperature Loss, Δ°C (Δ°F)<br>Loss of FW Heating Setpoint (SCRRI/SRI Initiation), Δ°C (Δ°F) | <br>216 (420)<br>55.6 (100)<br>16.67 (30) |
| Vessel Dome Pressure, MPaG (psig) | 7.07 (1025) |

**Table 2.3-1**

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

| Parameter | Value |
|---|---|
| Turbine Bypass Capacity, % of rated | 110 |
| Total Delay Time from TSV or TCV motion to the start of BPV Main Disc Motion, s | 0.02 |
| Total Delay Time from TSV or TCV motion to 80% of Total Bypass Valve Capacity, s | 0.17 |
| TCV Closure Times, s<br>Fast Closure Analysis Value (Bounding)<br>Fast Closure Analysis Calculated Value (Bounding) that results in the neutron flux peak just below the scram setpoint<br>Assumed Minimum Servo (Slow) Closure Analysis Value | <br>0.08<br>1.0<br>2.5 |
| TSV Closure Times, s | 0.100 |
| % of Rated Steam Flow that can pass through 3 Turbine Control Valves | 85 (Partial Arc) |
| Minimum Steamline Pressure Difference Between the Vessel Dome Pressure and the Turbine Throttle Pressure at rated conditions, MPa (psi)(2) | 0.179<br>(26) |
| Fuel Lattice | N |
| Core Leakage Flow, % (1) | 10.6 |
| Control Rod Drive Position versus Time | Table 2.3-2 & 3 |
| Core Design used in TRACG Simulations<br>Exposure: | Reference 2.3-3<br>Middle of Cycle and End of Cycle |

**Table 2.3-1**

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

| Parameter | Value |
|---|---|
| Safety Relief Valve (SRV) and Safety Valve (SV) capacity<br><br>At design pressure, MPaG (psig)<br>Number of SRVs<br>Number of SVs | See Subsections 2.4.13.3 & 2.4.15.2<br><br>8.618 (1250)<br>10<br>8 |
| Analysis values for SRV and SV setpoints<br>SRV Setpoint, MPaG (psig)<br>SV Setpoint, MPaG (psig) | <br>8.618 (1250)<br>8.756 (1270) |
| Closure Scram Position of 2 or More MSIVs, % open<br>Maximum delay time, s | 85<br>0.06 |
| MSIV Minimum Closure Time, s<br>MSIV Maximum Closure Time, s | 3.0<br>5.0 |
| MSIV Closure Profile used to Bound Minimum Closure Time, s<br>100% open<br>100% open<br>1% open<br>0% open | <br>0.0<br>0.6<br>1.7<br>3.0 |
| High Flux Trip, % NBR,<br>Sensor Time Constant | 125.0<br>0.03 |
| High Flux Trip setpoint as a linear function of feedwater temperature for feedwater temperatures above 222.2°C (432°F)(4)<br><br>Rate of Change limit on the High Flux Trip setpoint (5) | 125% at 222.2°C (432°F)<br>111% at 252.2°C (486°F)<br>26% / hour |
| TSV Closure Scram Position of 2 or more TSV, % open<br>Trip Time delay, s | 85<br>0.06 |
| TCV Fast Closure Scram Trip, s | 0.08 |

**Table 2.3-1**

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

| Parameter | Value |
|---|---|
| High Pressure Scram, MPaG (psig).<br>Maximum scram delay, s | 7.619 (1105)<br>0.7 |
| High Suppression Pool Temperature Scram trip, °C (°F),<br>Maximum Delay Time, s | 48.9(120)<br>1.05 |
| High Suppression Pool Temperature FAPCS actuation, °C (°F) | 43.3 (110) |
| Vessel level Trips (above bottom vessel)<br>Level 9—(L9), m (in)<br>Level 8—(L8), m (in)<br>Level 7 – (L7), m (in), high level alarm<br>Normal Water Level, m (in)<br>Level 4—(L4), m (in), low level alarm<br>Level 3—(L3), m (in)<br>Level 2—(L2), m (in)<br>Level 1—(L1), m (in)<br>Level 0.5 – (L0.5) m (in) | <br>22.39 (881.5)<br>21.89 (861.8)<br>20.83 (820.3)<br>20.72 (815.7)<br>20.60 (811.2)<br>19.78 (778.7)<br>16.05 (631.9)<br>11.50 (452.8)<br>8.45 (332.7) |
| Maximum APRM Simulated Thermal Power Trip<br>Scram, % NBR<br>Time Constant, s | <br>115<br>7 |
| Simulated Thermal Power setpoint as a linear function of feedwater temperature for feedwater temperatures above 222.2°C (432°F)(4)<br><br>Rate of Change limit on simulated thermal power setpoint(5) | 115% at 222.2°C (432°F)<br>101% at 252.2°C (486°F)<br>26% / hour |
| Minimum Steamline Volume, (total of all lines, including header): Vessel to TSV, $m^3$ ($ft^3$)(2) | 103.3<br>(3648) |
| Minimum Steamline Length (average of all lines): Flow Path from Vessel to TSV, m (ft)(2) | 65.26 (214.1) |
| CRD Hydraulic System minimum capacity, $m^3$/hr (gpm),<br>Capacity in kg/s (Mlbm/hr) for 990 kg/$m^3$ density (61.8 lbm/$ft^3$) | 235.1 (1035)<br>64.6 (0.513) |

## Table 2.3-1

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

| Parameter | Value |
|---|---|
| Maximum time delay from Initiating Signal (Pump 1 & 2), s<br>If offsite power is not available, s | 10 & 25<br>145 |
| Isolation Condensers<br>Max Initial Temperature, °C (°F)<br>Minimum Initial Temperature, °C (°F)<br>Time To injection valve full open (Max), s(3)<br>Heat Removal Capacity for 4 Isolation Condensers, MW (% Rated Power)<br>Isolation Condensers volume, 4 Units, from steam box to discharge at vessel $m^3$ ($ft^3$) | <br>40 (104)<br>10 (50)<br>31<br>135 (3%)<br>56.1 (1981) |

(1) These are calculated steady state values not inputs or assumptions, and may change for different initial condition assumptions. "Reference" values are provided as they form the basis for the percentages shown on the figures.

(2) These values are used in potentially limiting pressurization transients and bound the turbine throttle pressure in DCD, Tier 2, Table 10.1-1. Events that use the bounding steamline inputs use a different fuel bundle CPR R-factor than used in other AOO and infrequent event analyses. This changes the CPR values shown on the plots but does not significantly affect the ΔCPR/ICPR result. Historical values for turbine throttle pressure, 6.57 MPaG, (953 psig) and steamline volume, 135 $m^3$ (4767 $ft^3$) and larger steamline length (consistent with volume) are used in non-limiting events and non-pressurization events.

(3) In the analysis, after 1 s logic delay, the isolation condenser opening valve curve began to open at 15 s for a total opening time of 30 s. For inadvertent isolation condenser operation, the valve begins to open at 15 s with an opening time of 7.5 s.

(4) As the reactor power changes with changes in the feedwater temperature (DCD Tier 2, Figure 4.4-1), the simulated thermal power trip and high flux trip setpoints also change with feedwater temperature.

(5) Rate of change of the simulated thermal power and high flux trip setpoints is established to ensure that the trip setpoints do not rapidly change with unexpected changes in the feedwater temperature. This trip rate of change limit is based on the maximum planned rate in the feedwater temperature setpoint discussed in DCD, Tier 2, Subsection 7.7.3.2.3

**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 2.3-4a

### Results Summary of Anticipated Operational Occurrence Events[1]

| Sub-section I.D. | Description | Max. Neutron Flux, % NBR | Max. Dome Pressure, MPaG (psig) | Max. Vessel Bottom Pressure, MPaG (psig) | Max. Steamline Pressure, MPaG (psig) | Max. Simulated Thermal Power, % NBR | ΔCPR/ICPR or Minimum Water Level (m [ft] over TAF) |
|---|---|---|---|---|---|---|---|
| 2.3.2.1 | Loss of Feedwater Heating | 100 | 7.07 (1025) | 7.21 (1046) | 6.95 (1009) | 100 | 0.02 |
| 2.3.2.1 | Closure of One Turbine Control Valve. FAST/SLOW | 135<br>110 | 7.25 (1052)<br>7.20 (1043) | 7.39 (1072)<br>7.33 (1063) | 7.15 (1037)<br>7.16 (1038) | 103<br>102 | 0.05<br>0.03 |
| 2.3.2.2 | Generator Load Rejection with Turbine Bypass | 127 | 7.16 (1038) | 7.29 (1057) | 7.39 (1074) | 101 | 0.03 |
| 2.3.2.3 | Generator Load Rejection with a Single Failure in the Turbine Bypass System | 206 | 7.34 (1064) | 7.48 (1084) | 7.54 (1093) | 102 | 0.04 |
| 2.3.2.4 | Turbine Trip with Turbine Bypass | 114 | 7.14 (1035) | 7.25 (1052) | 7.38 (1070) | 100 | 0.01 |
| 2.3.2.5 | Turbine Trip with a Single Failure in the Turbine Bypass System | 155 | 7.3 (1059) | 7.44 (1079) | 7.54 (1094) | 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 | 112 | 7.07 (1026) | 7.21 (1046) | 6.96 (1009) | 109 | 0.07 |
| 2.3.4.2 | Runout of One Feedwater Pump | 103 | 7.08 (1027) | 7.22 (1047) | 7.04 (1021) | 101 | ≤ 0.01 |
| 2.3.5.1 | Opening of One Turbine Control or Bypass Valve | 101 | 7.08 (1027) | 7.21 (1046) | 7.04 (1021) | 100 | ≤ 0.01 |
| 2.3.5.2 | Loss of Non-Emergency AC Power to Station Auxiliaries | 139 | 7.13 (1035) | 7.28 (1056) | 7.28 (1056) | 102 | 5.37m (17.6 ft) |
| 2.3.5.3 | Loss of Feedwater Flow | 100 | 7.08 (1027) | 7.21 (1046) | 7.04 (1021) | 100 | 5.28m (17.3 ft) |

(1) This table summarizes the events calculated with the TRACG code. Table 2.3-4b contains the summary of the remaining AOO Events.

## Table 2.3-4b

## Results Summary of Anticipated Operational Occurrence Events (1)

Data in the DCD applies to the initial core.

(1) Table 2.3-4a summarizes the events calculated with the TRACG code for the initial core.

**Table 2.3-5**

**Sequence of Events for Loss of Feedwater Heating**

| Time (s) | Event * |
|---|---|
| 0 - 20 | Steady state |
| 20 | Initiate a 55.6°C (100°F) temperature reduction in the FW system |
| 43 (est) | RC&IS initiates Selected Rod Insertion plus Selected Control Rod Run-In (SCRRI/SRI). |
| 44 (est.) | Initial effect of unheated FW starts to raise core power level |
| 44 | First SRI group inserts (one HCU, 2 control rods, fails to actuate) and SCRRI starts |
| 59 | Second SRI group inserts |
| 74 | Third SRI group inserts |
| 89 | Fourth SRI group inserts |
| 92 | Steam flow below 60% rated |
| 104 | Fifth SRI group inserts |
| 119 | Sixth SRI group inserts |
| 122 | Power below 60% rated |
| 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 |
| 1.0 | TCV closed |
| 1.7 | Turbine bypass valves start to open |
| >50.0 | 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 | SCRRI/SRI activated (no SCRRI rods assigned) |
| 1.5 | First SRI group inserts (one HCU, 2 control rods, fails to actuate) |
| 16.5 | Second SRI group inserts |
| 31.5 | Third SRI group inserts |
| 46.5 | Fourth SRI group inserts |
| ≈ 48 | Steam flow below 60% of rated |
| 61.5 | Fifth SRI group inserts |
| 76.5 | Sixth SRI group inserts |
| ≈77 | 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 |

* See Figure 2.3-4.

**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 |
| 0.02 | Turbine bypass valves start to open (Half fail to open) |
| 0.08 | Turbine control valves closed |
| 0.15 | Not enough turbine bypass availability is detected and the plant is scrammed |
| 0.35 | Control Rod insertion begins |
| 2.7 | 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 | SCRRI/SRI activated (no SCRRI rods assigned) |
| 1.5 | First SRI group inserts (one HCU, 2 control rods, fails to actuate) |
| 16.5 | Second SRI group inserts |
| 31.5 | Third SRI group inserts |
| 46.5 | Fourth SRI group inserts |
| ≈ 48 | Steam flow below 60% of rated |
| 61.5 | Fifth  SRI group inserts |
| 76.5 | Sixth SRI group inserts |
| ≈77 | Core power reaches 60% |
| 0.0-400.0 | FW temperature is decreasing because of reduced turbine steam flow |
| 400 | New steady state is established |

* See Figure 2.3-6.

## 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 |
| 0.25 | Not enough turbine bypass availability is detected and the plant is scrammed |
| 0.35 | Control Rod insertion begins |
| 2.7 | L3 level is reached |
| Long term | L2 is not reached, new steady state |

* 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 | ICS 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 isolation condenser 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 ICS (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 IC 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 |
| ≈55 | Full power established for ICS |
| ≈180 | 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 |
| <1 | TCV begins to close 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 | Isolation condenser 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 ICS 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 isolation condenser 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.

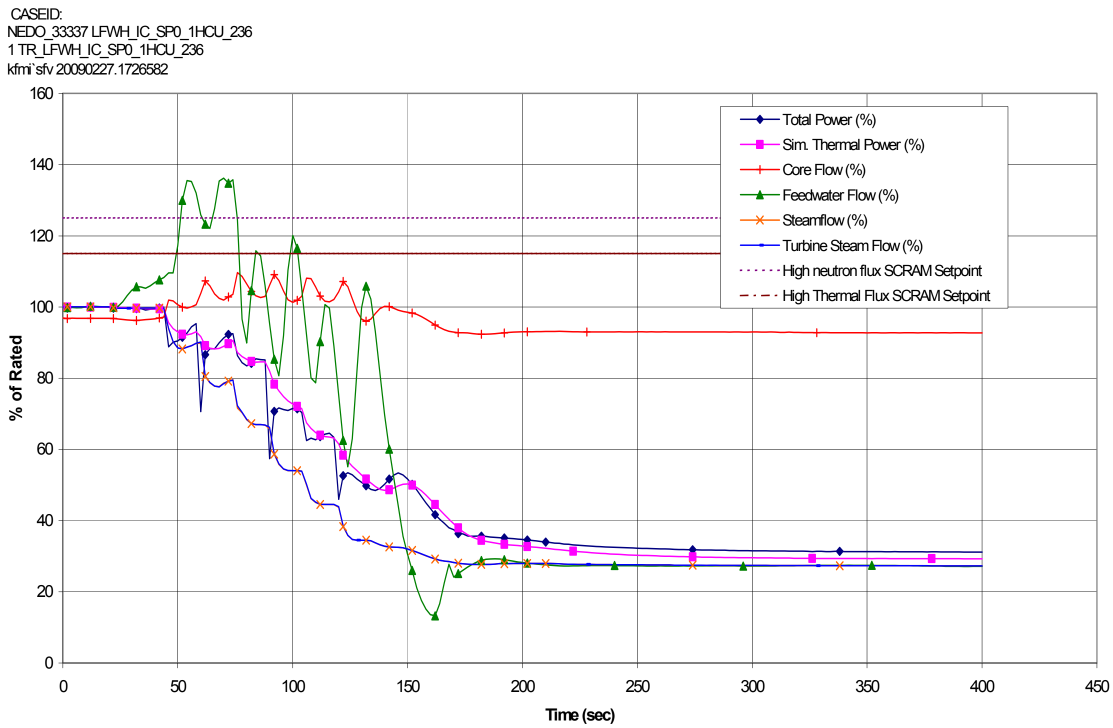



**Figure 2.3-1a. Loss of Feedwater Heating**

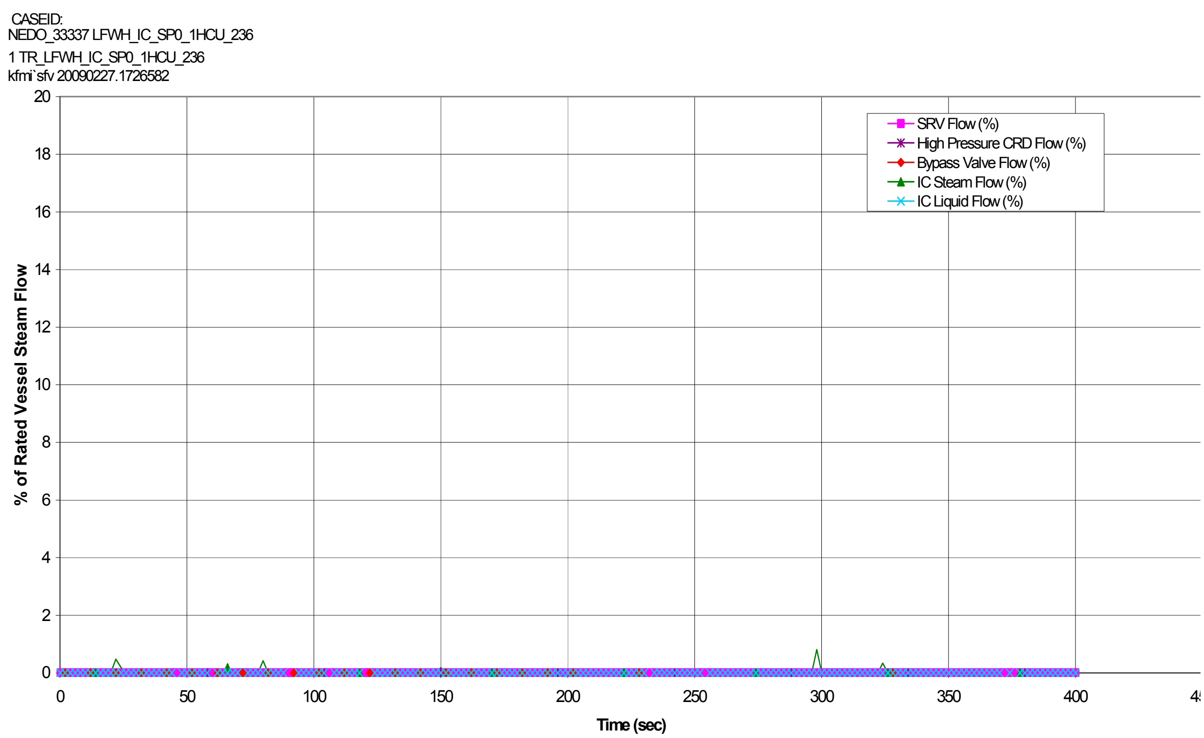



**Figure 2.3-1b. Loss of Feedwater Heating**

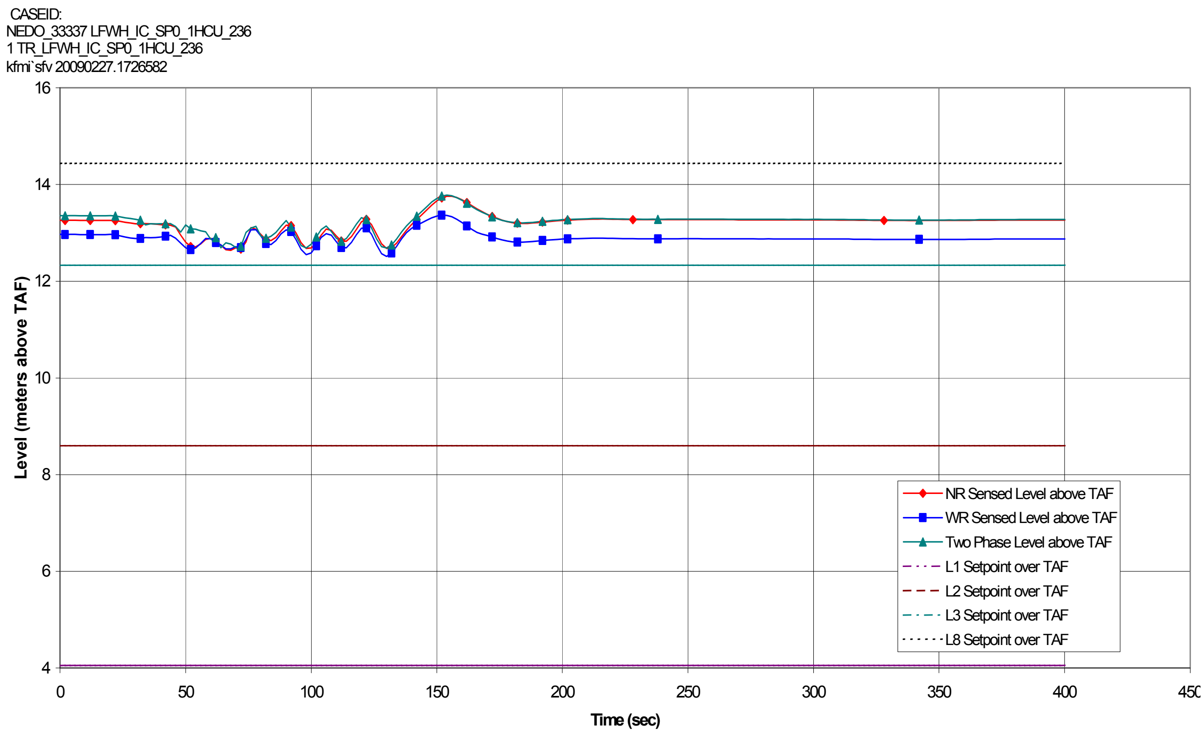



**Figure 2.3-1c. Loss of Feedwater Heating**

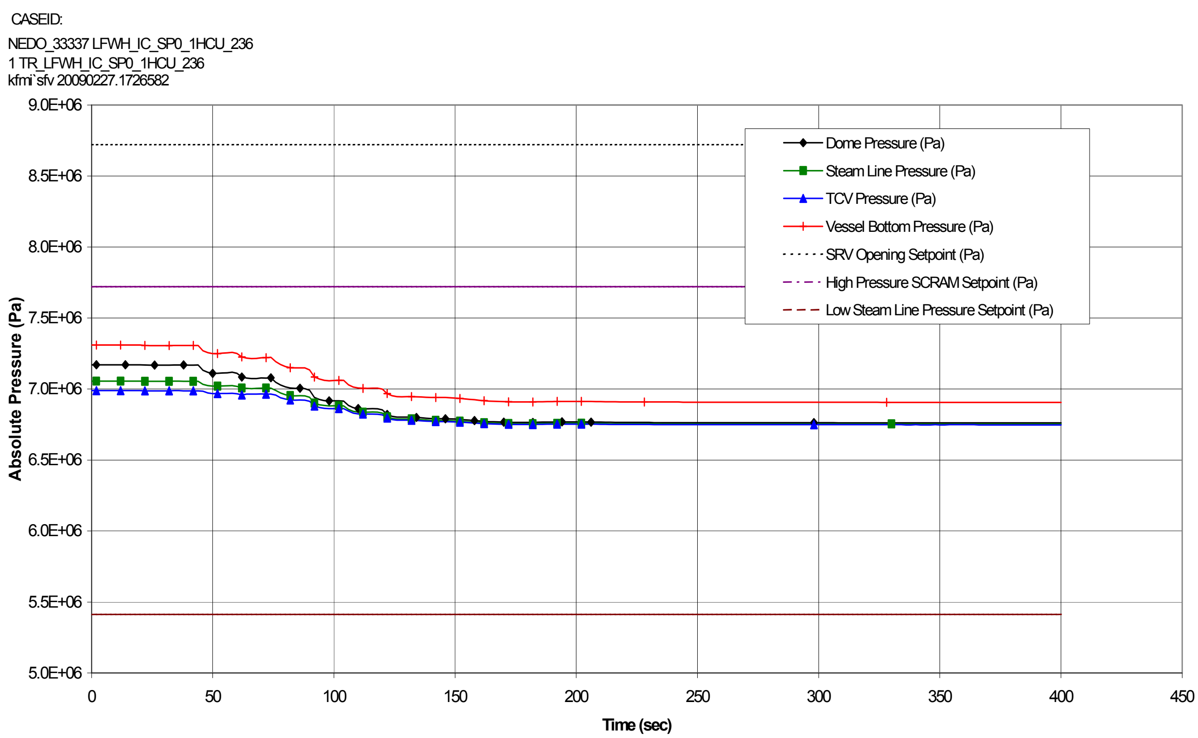



**Figure 2.3-1d. Loss of Feedwater Heating**

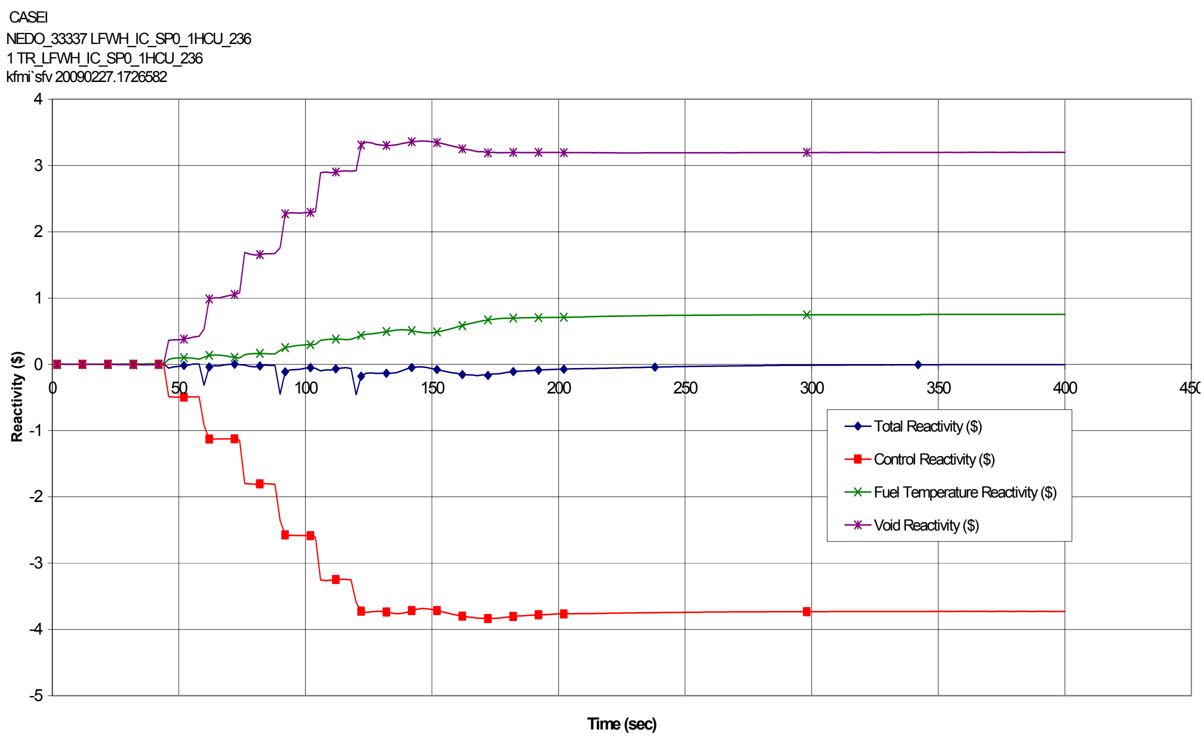



Figure 2.3-1e. Loss of Feedwater Heating

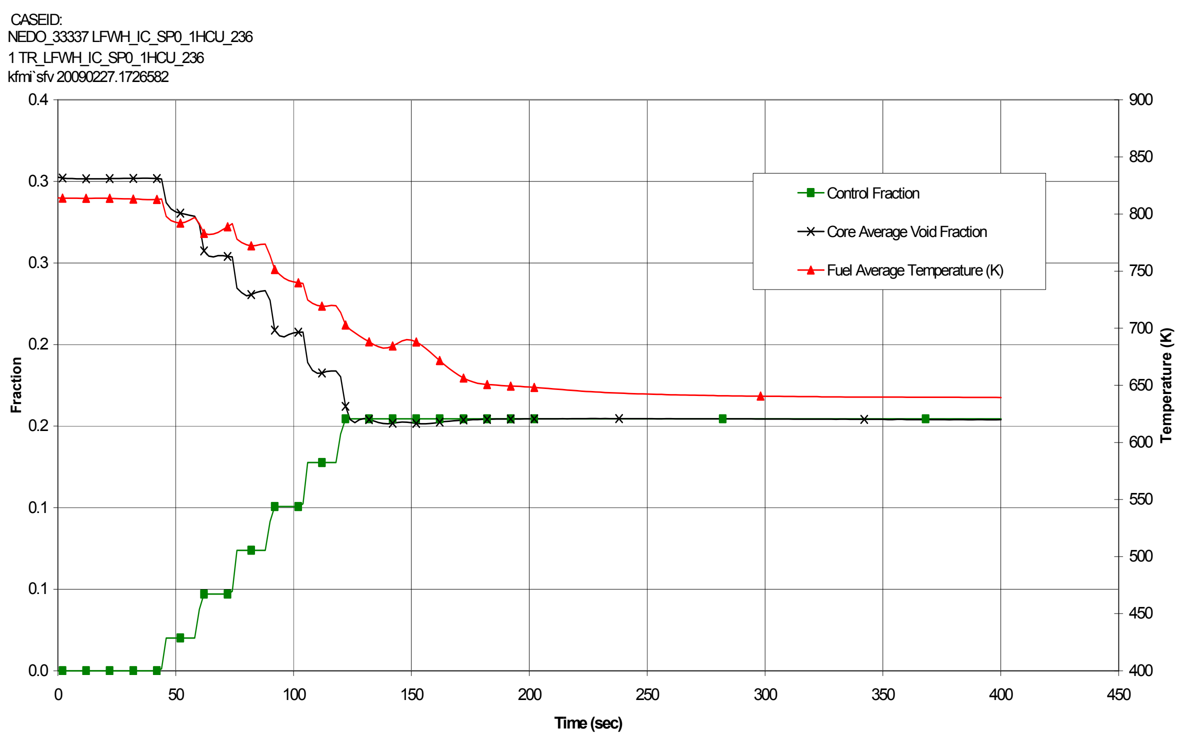



Figure 2.3-1f. Loss of Feedwater Heating

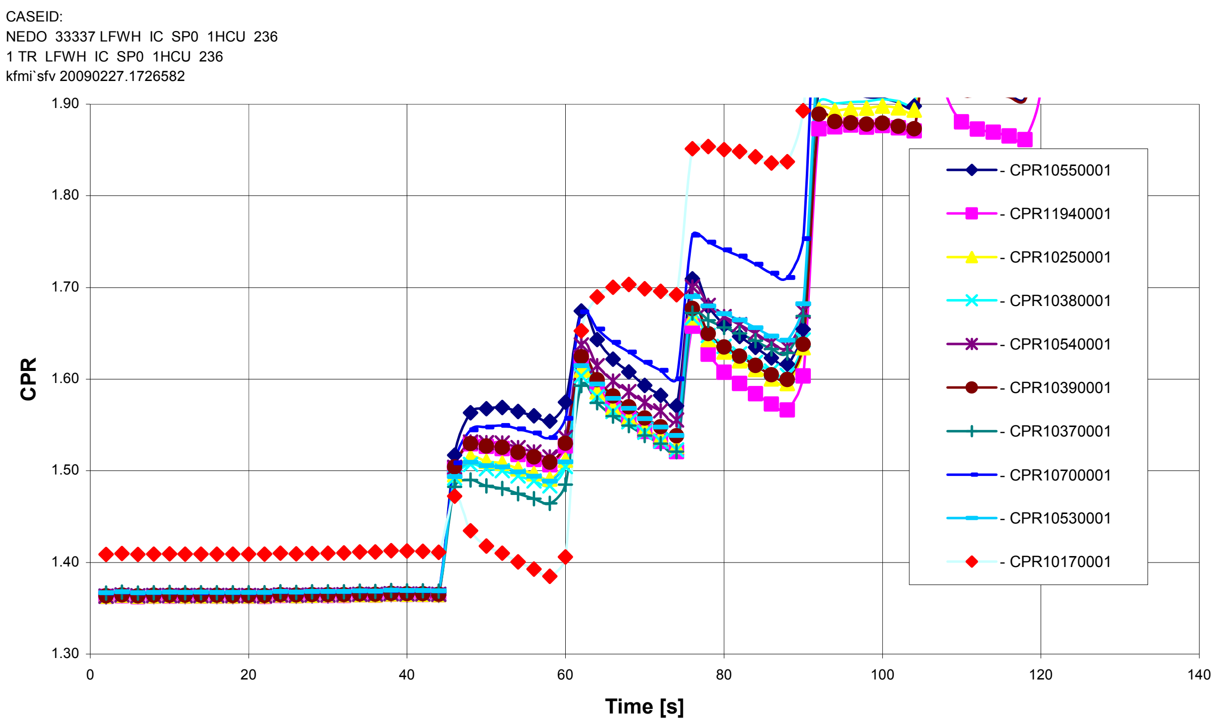



Figure 2.3-1g. Loss of Feedwater Heating

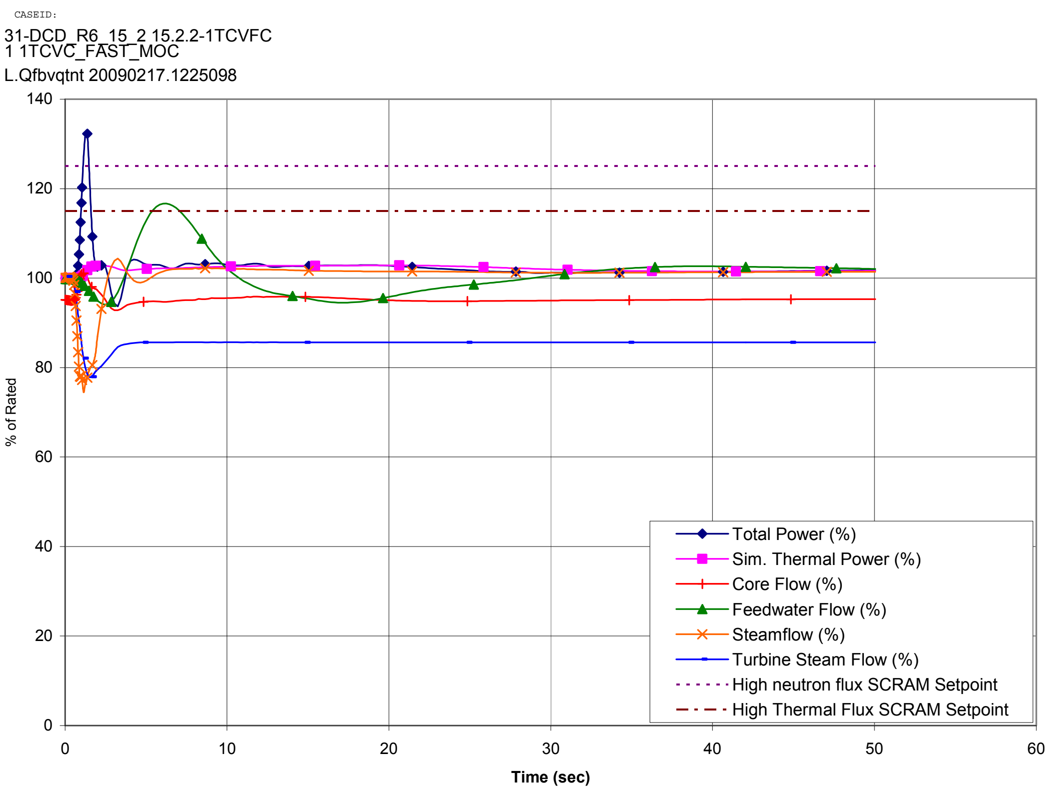



**Figure 2.3-2a. Fast Closure of One Turbine Control Valve**

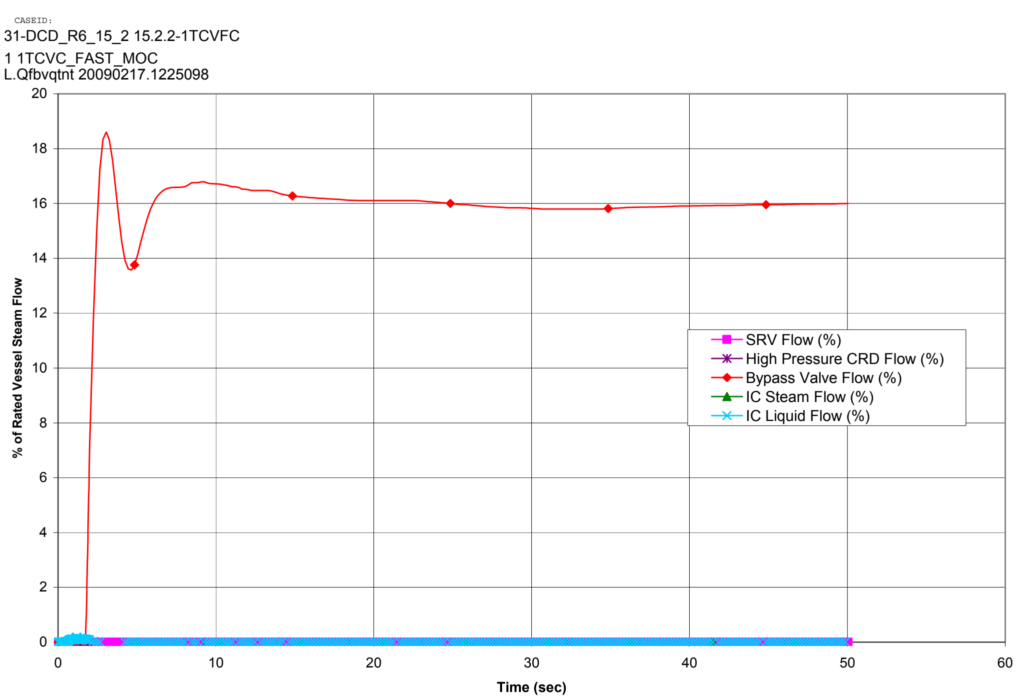



**Figure 2.3-2b. Fast Closure of One Turbine Control Valve**

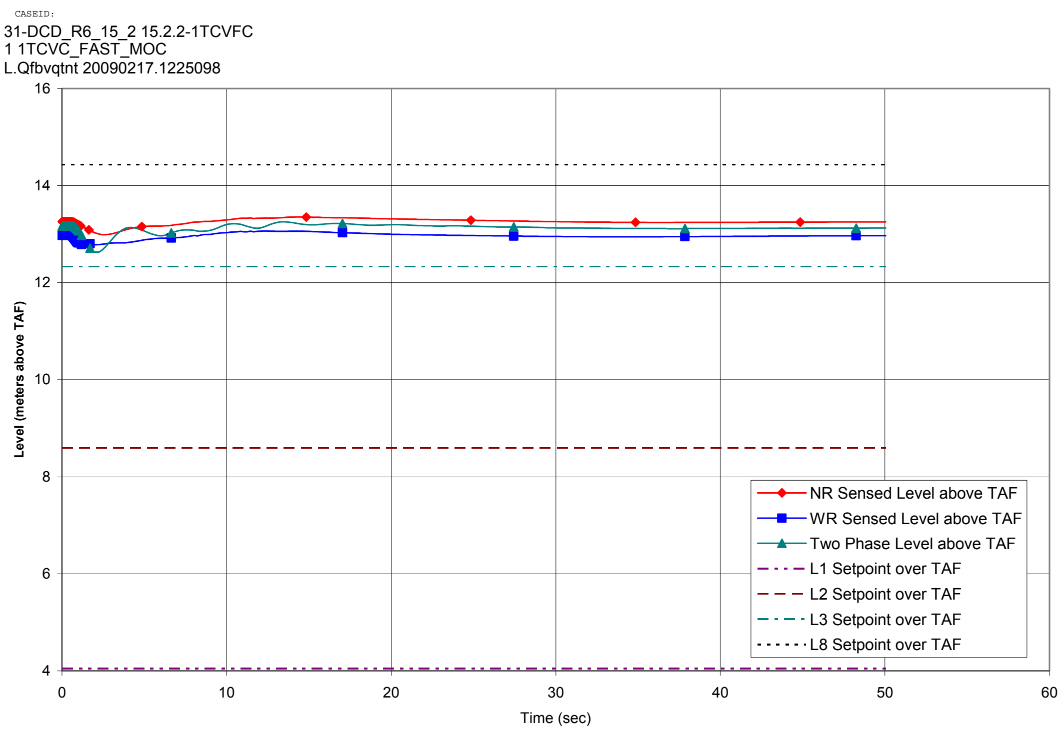



**Figure 2.3-2c. Fast Closure of One Turbine Control Valve**

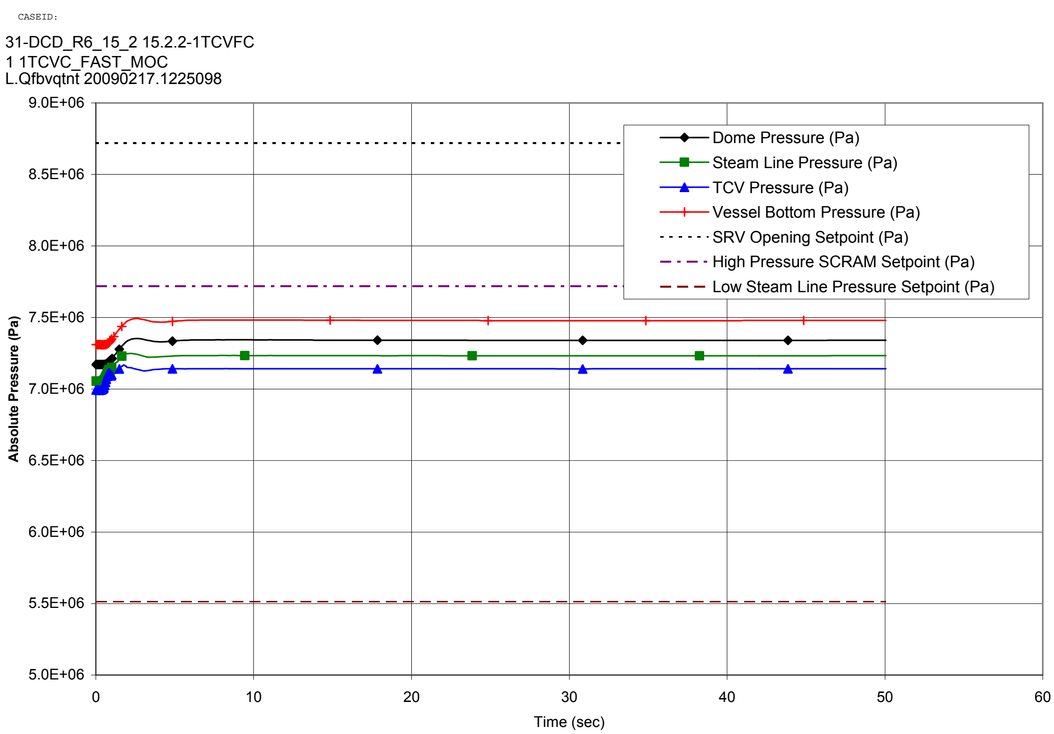



**Figure 2.3-2d. Fast Closure of One Turbine Control Valve**

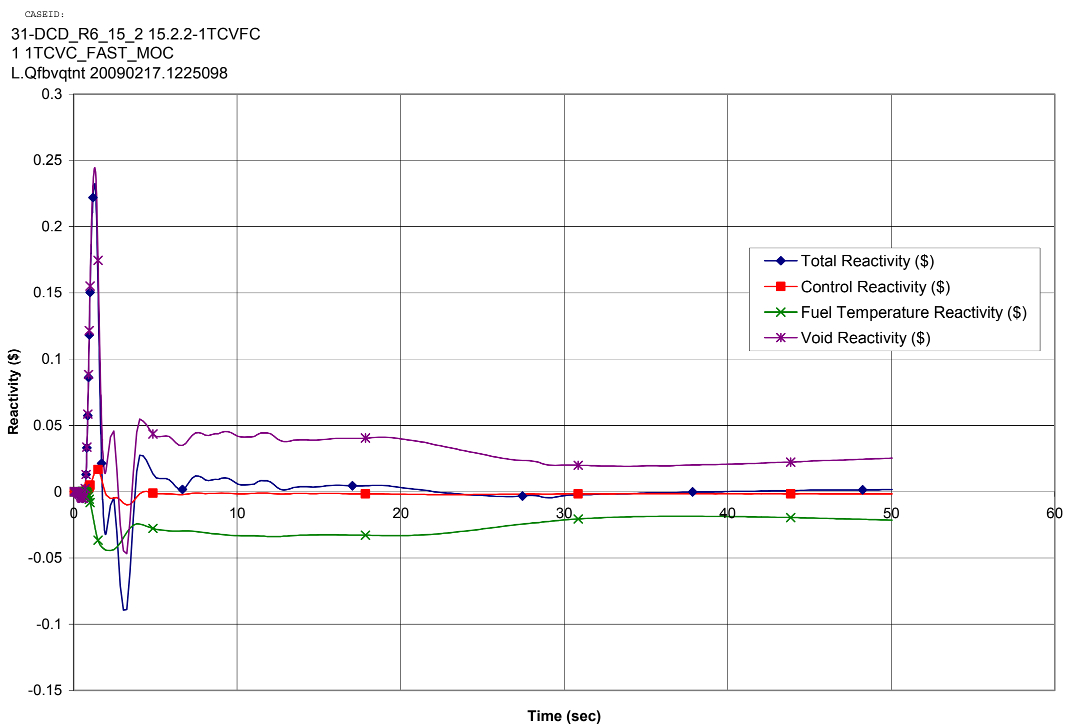



**Figure 2.3-2e. Fast Closure of One Turbine Control Valve**

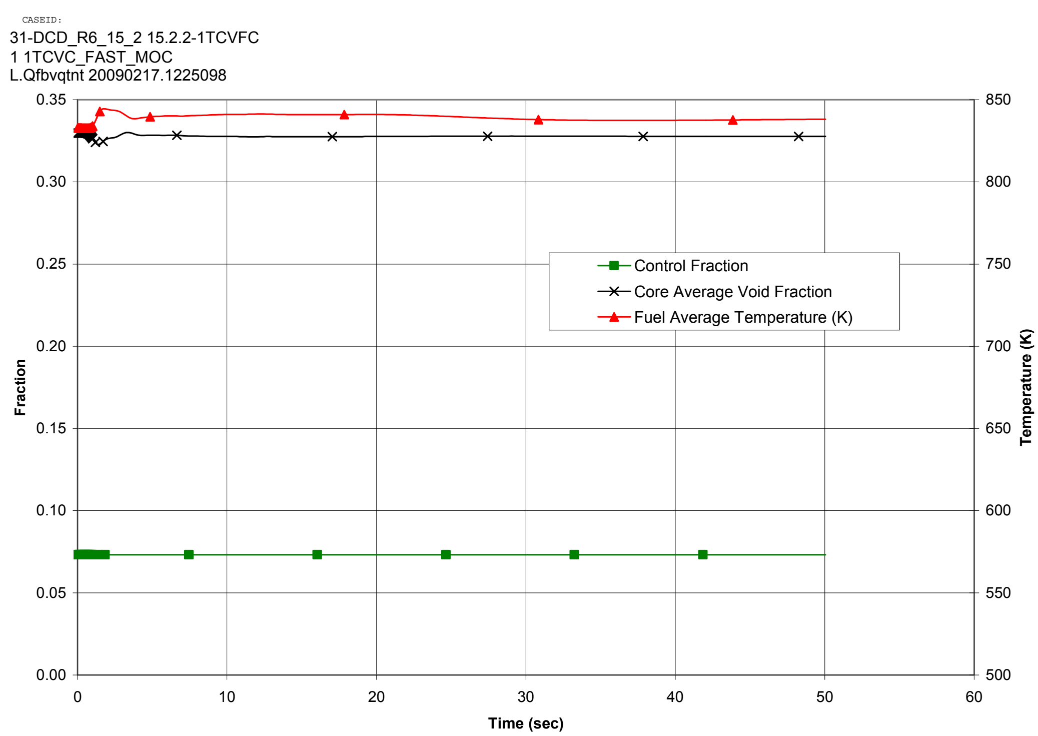



**Figure 2.3-2f. Fast Closure of One Turbine Control Valve**

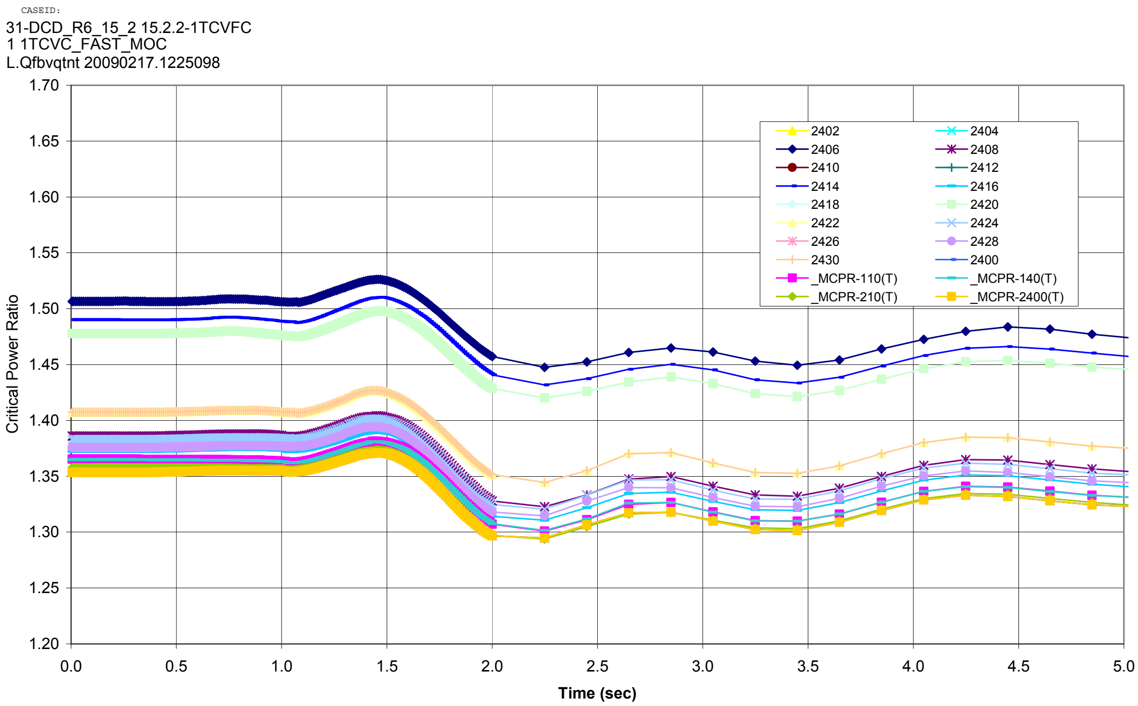



**Figure 2.3-2g. Fast Closure of One Turbine Control Valve**

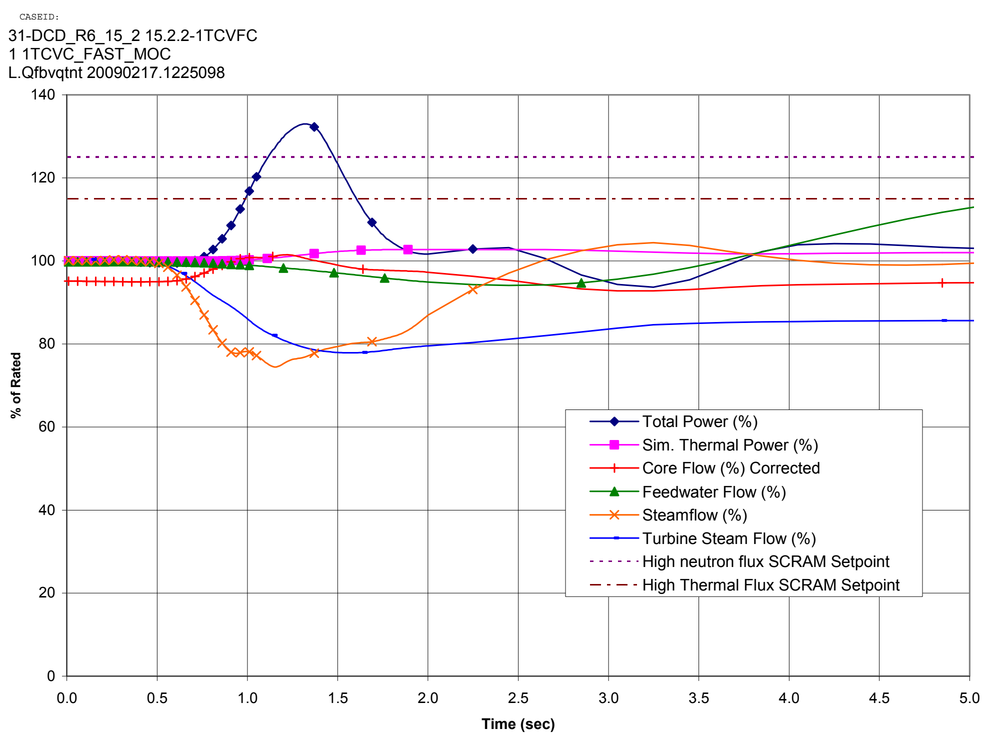



**Figure 2.3-2h. Fast Closure of One Turbine Control Valve (Figure 2.3-2a from 0 to 5s)**

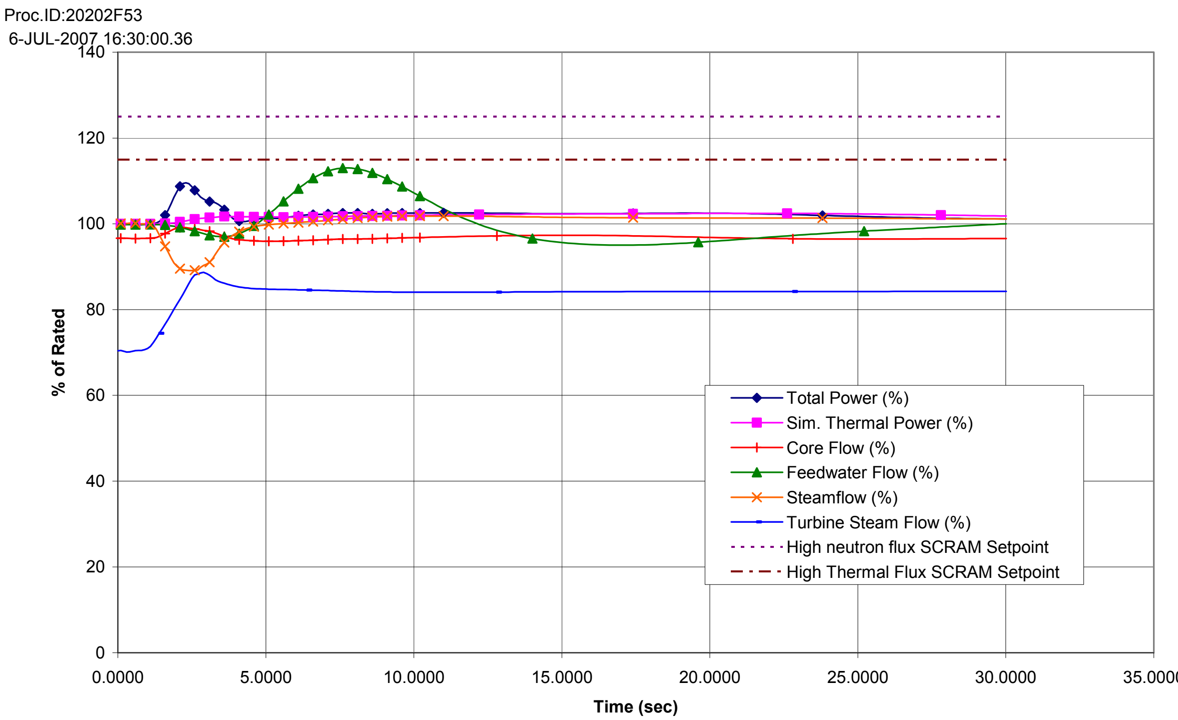



**Figure 2.3-3a. Slow Closure of One Turbine Control Valve**

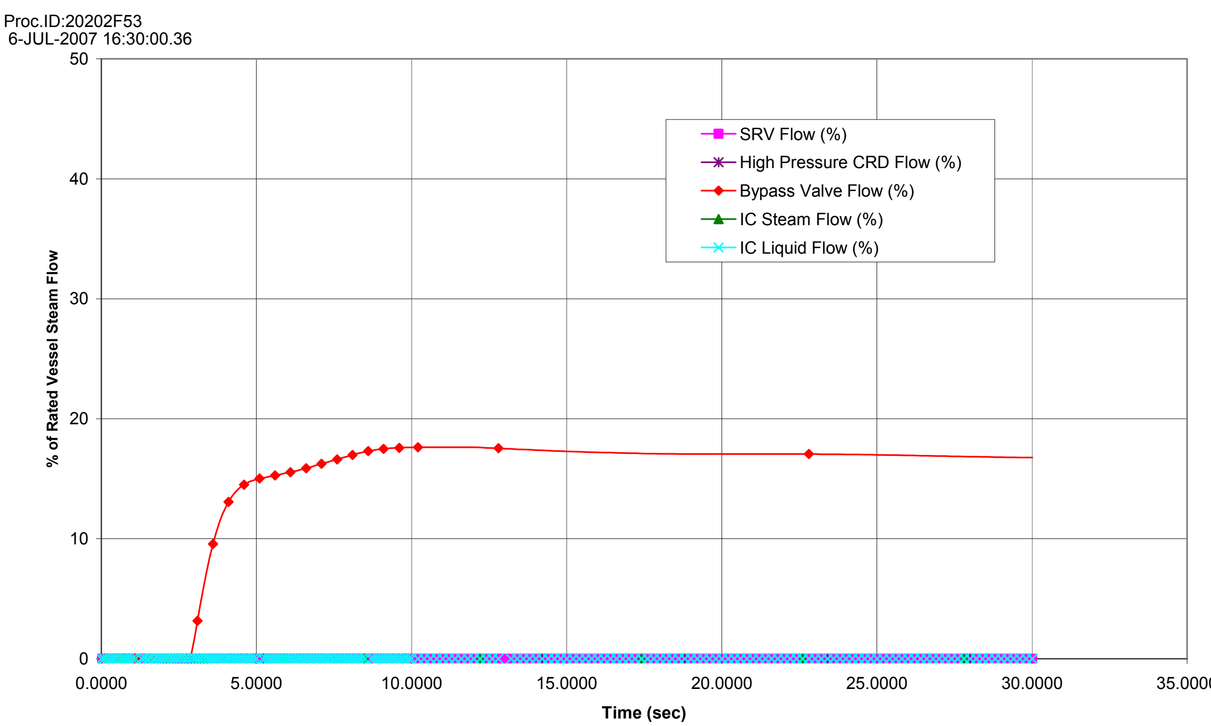



**Figure 2.3-3b. Slow Closure of One Turbine Control Valve**

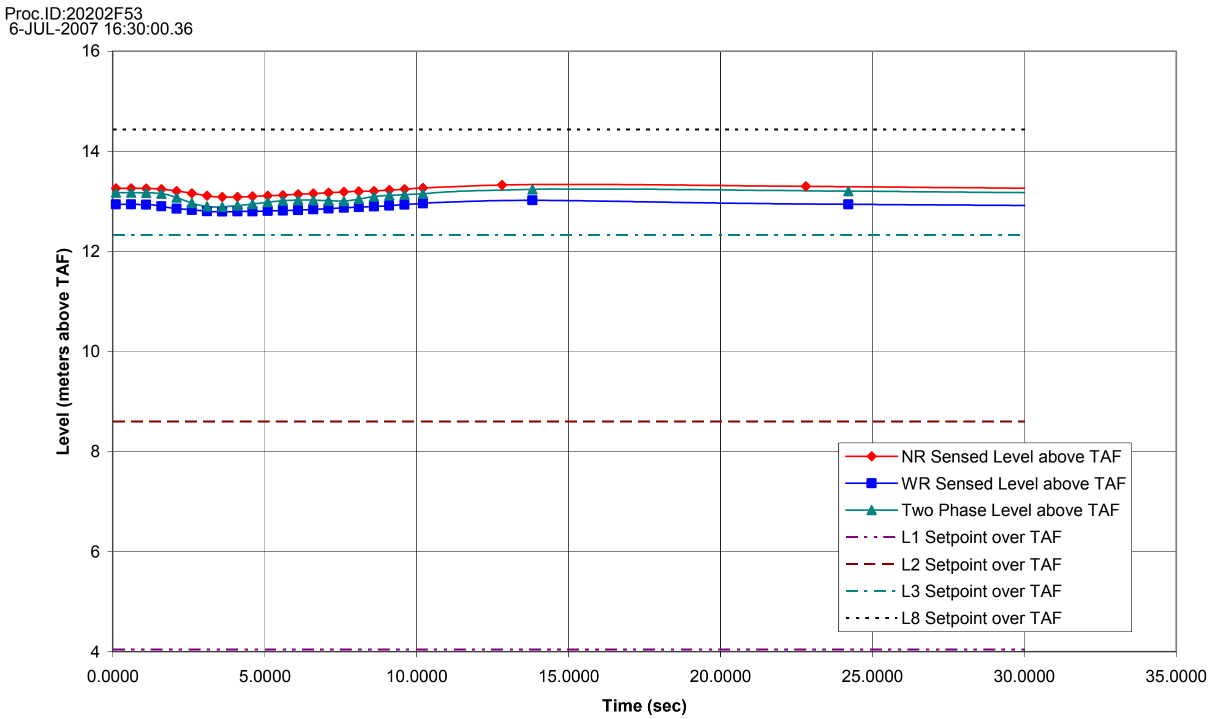



**Figure 2.3-3c. Slow Closure of One Turbine Control Valve**

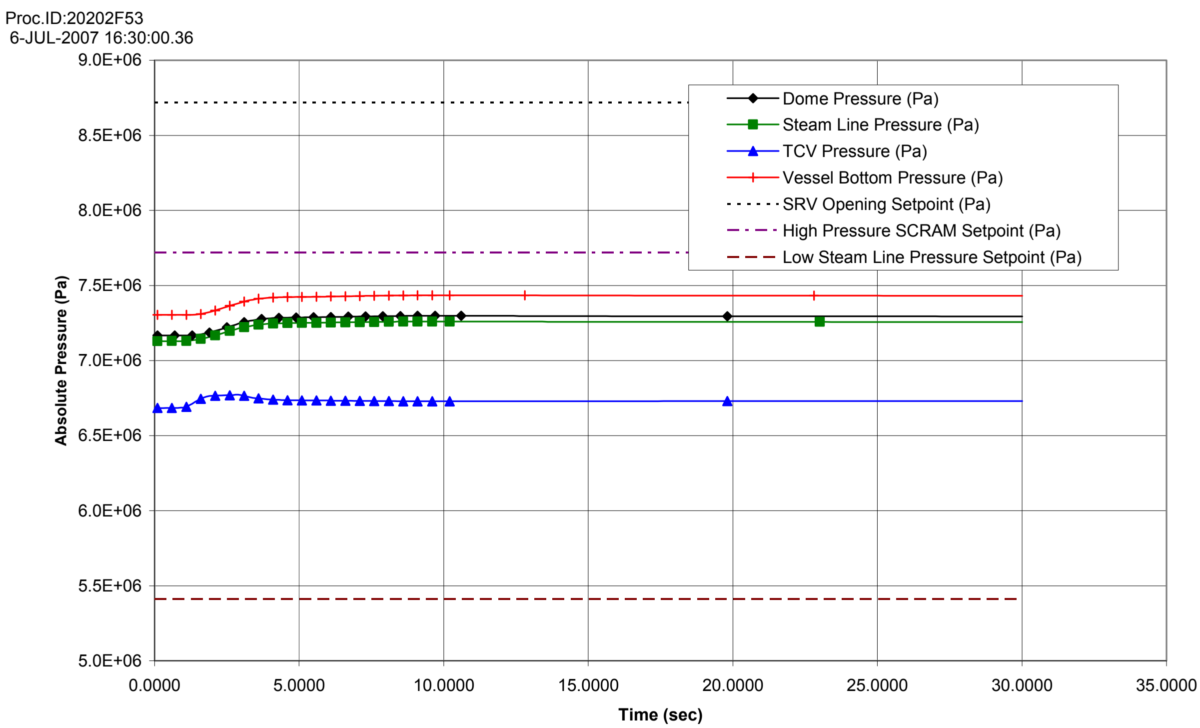



**Figure 2.3-3d. Slow Closure of One Turbine Control Valve**

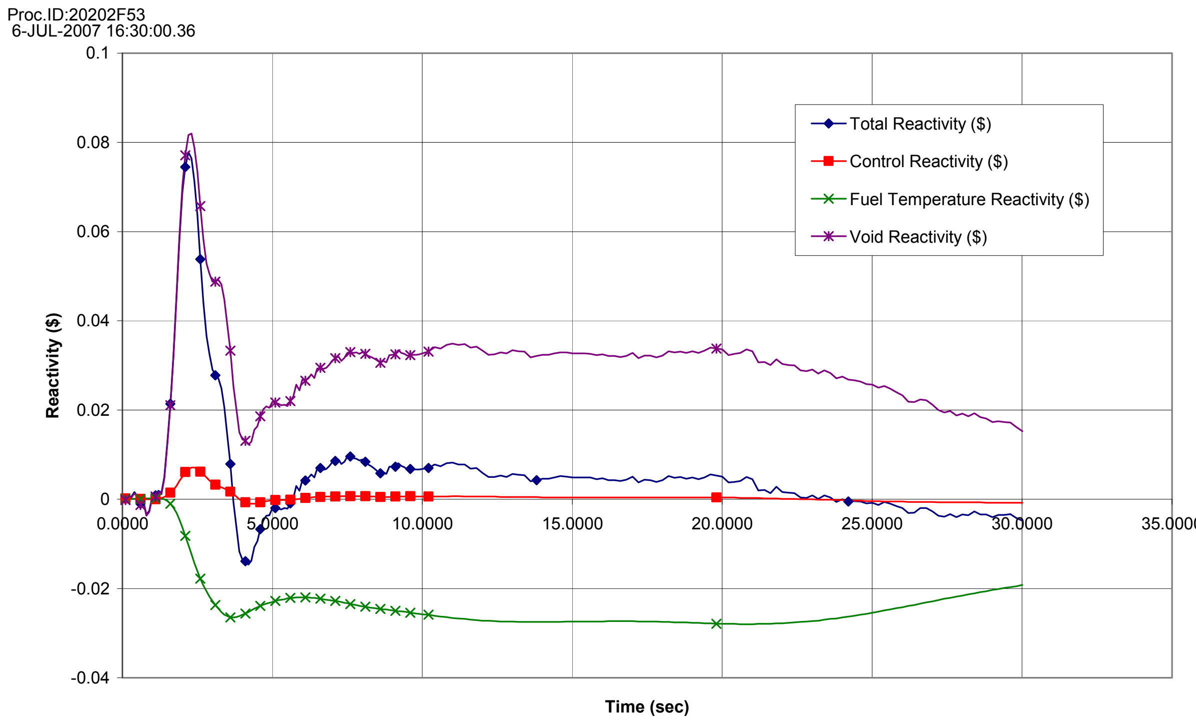



**Figure 2.3-3e. Slow Closure of One Turbine Control Valve**

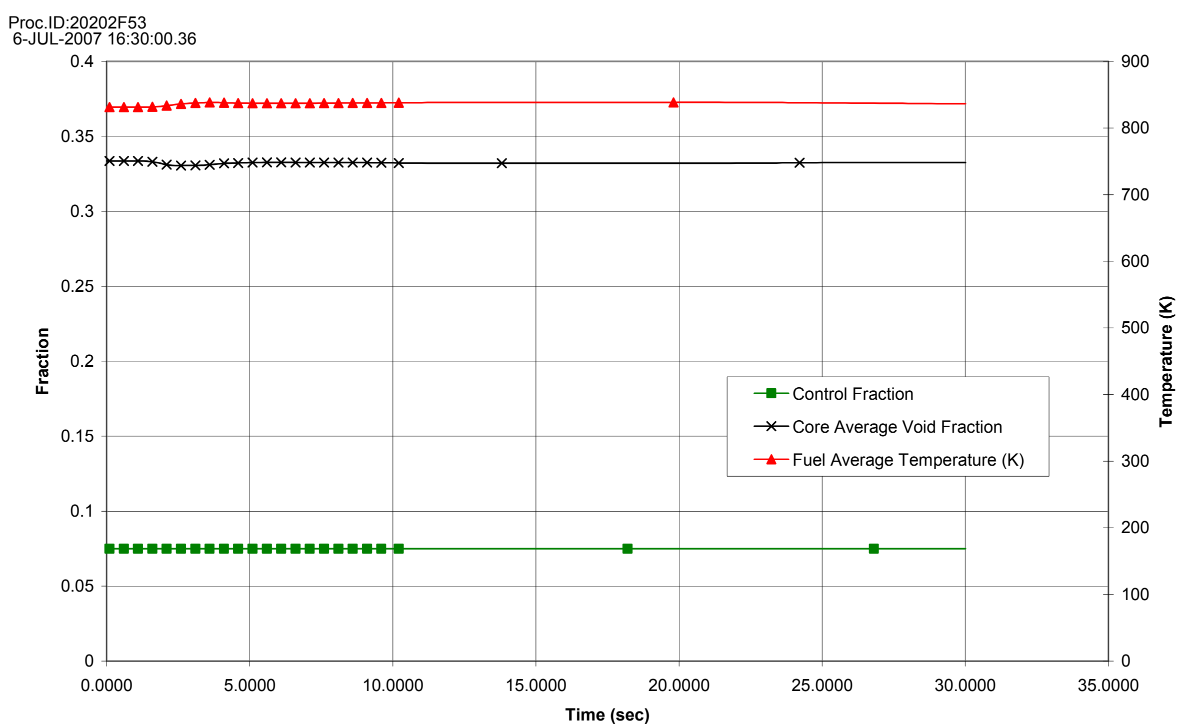



**Figure 2.3-3f. Slow Closure of One Turbine Control Valve**

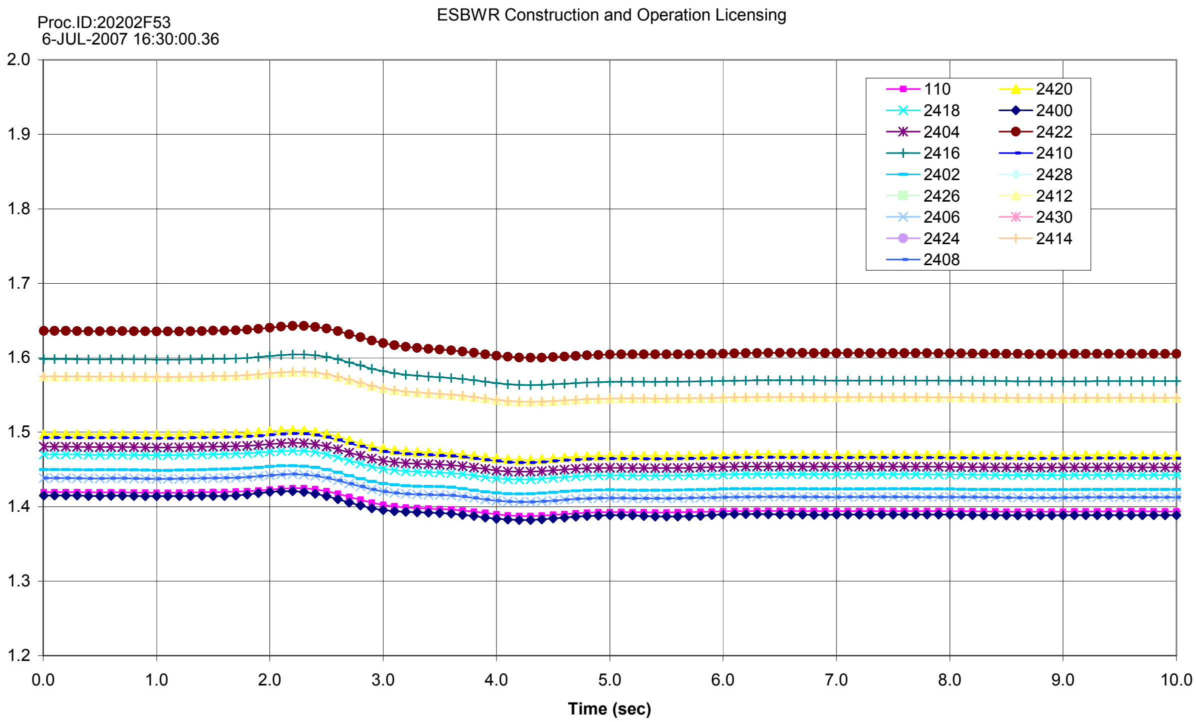



**Figure 2.3-3g. Slow Closure of One Turbine Control Valve**

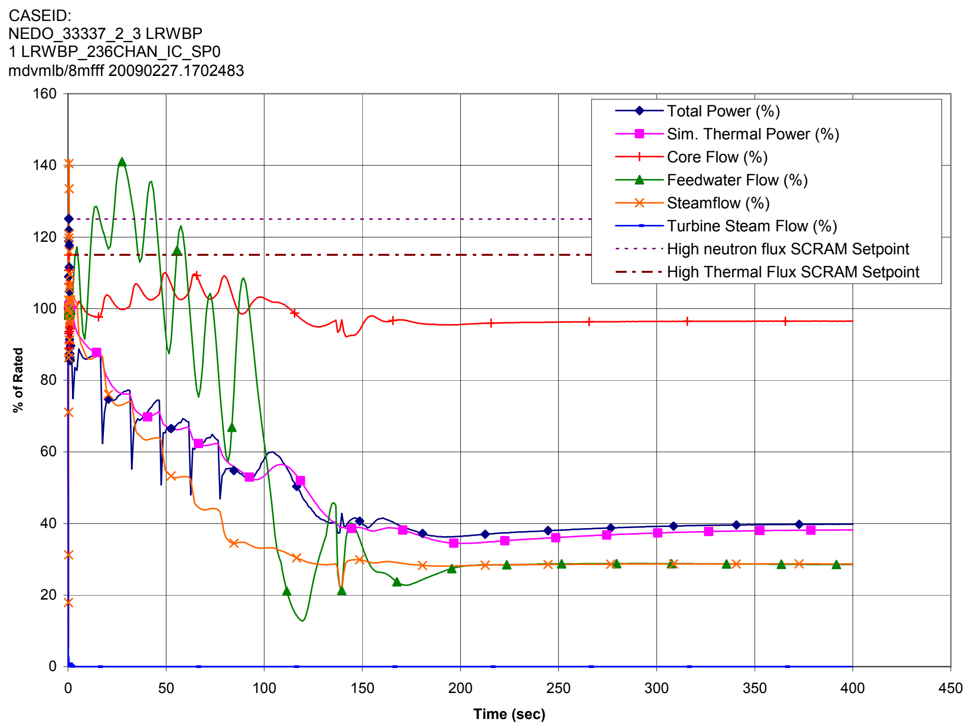



Figure 2.3-4a. Generator Load Rejection with Turbine Bypass

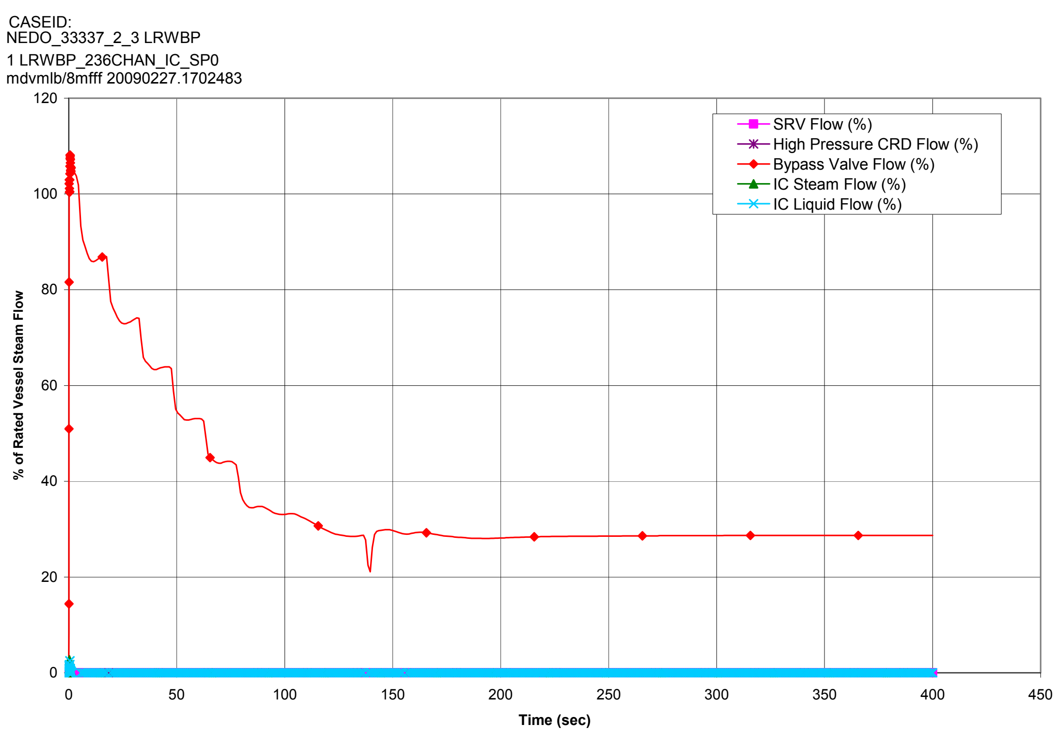



Figure 2.3-4b. Generator Load Rejection with Turbine Bypass

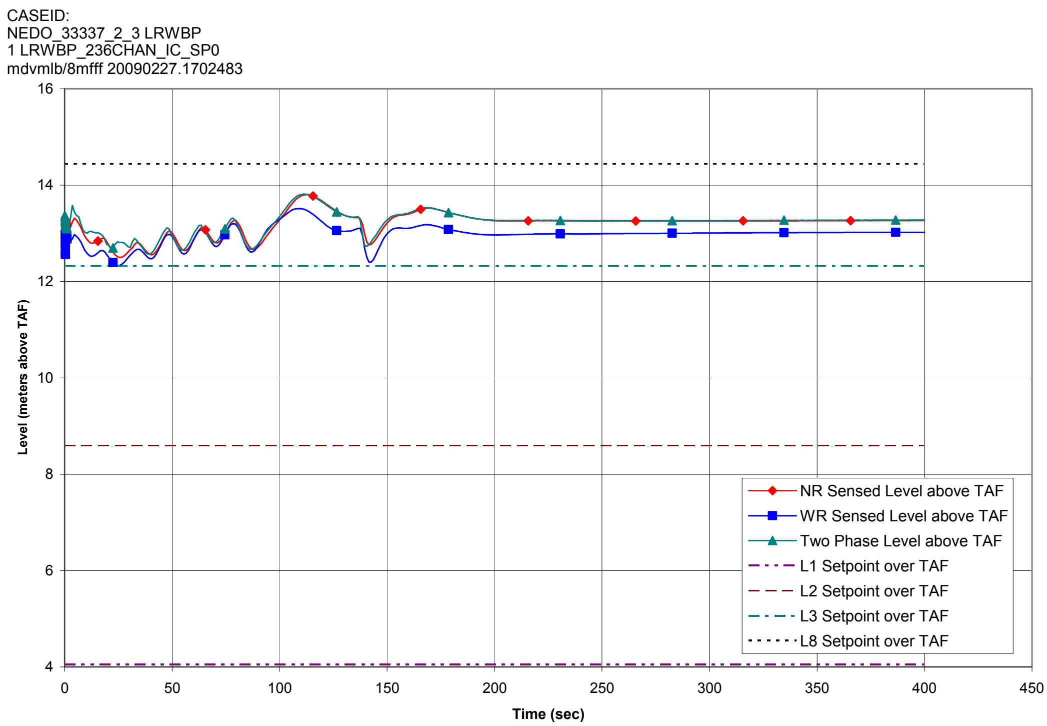



**Figure 2.3-4c. Generator Load Rejection with Turbine Bypass**

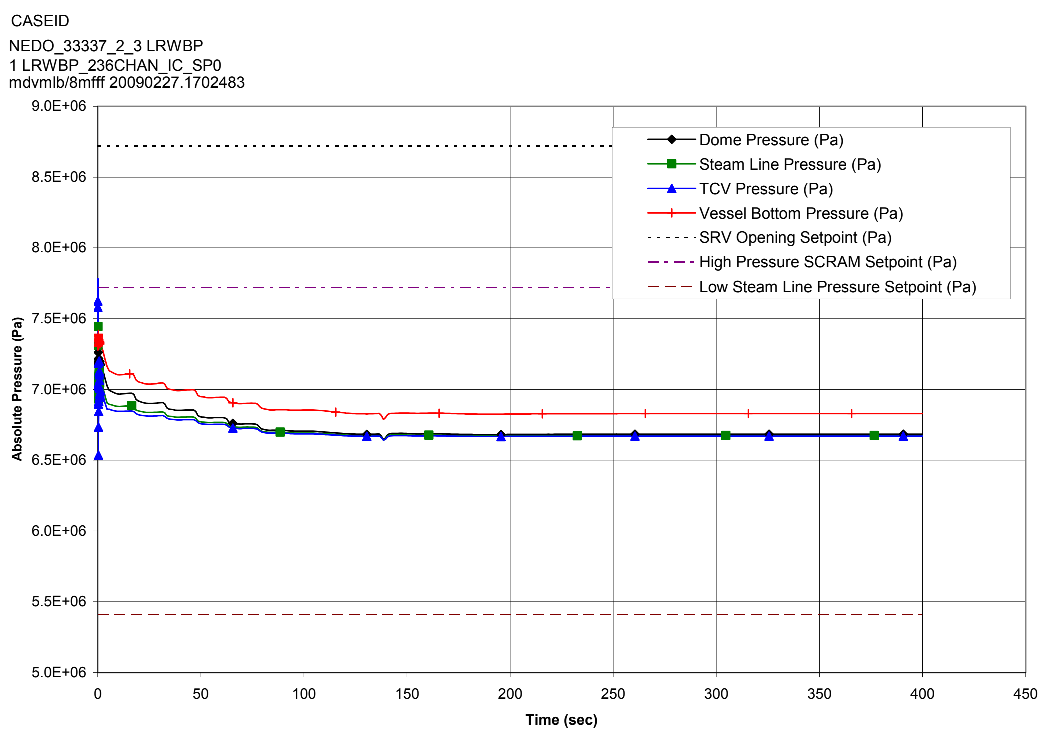



**Figure 2.3-4d. Generator Load Rejection with Turbine Bypass**

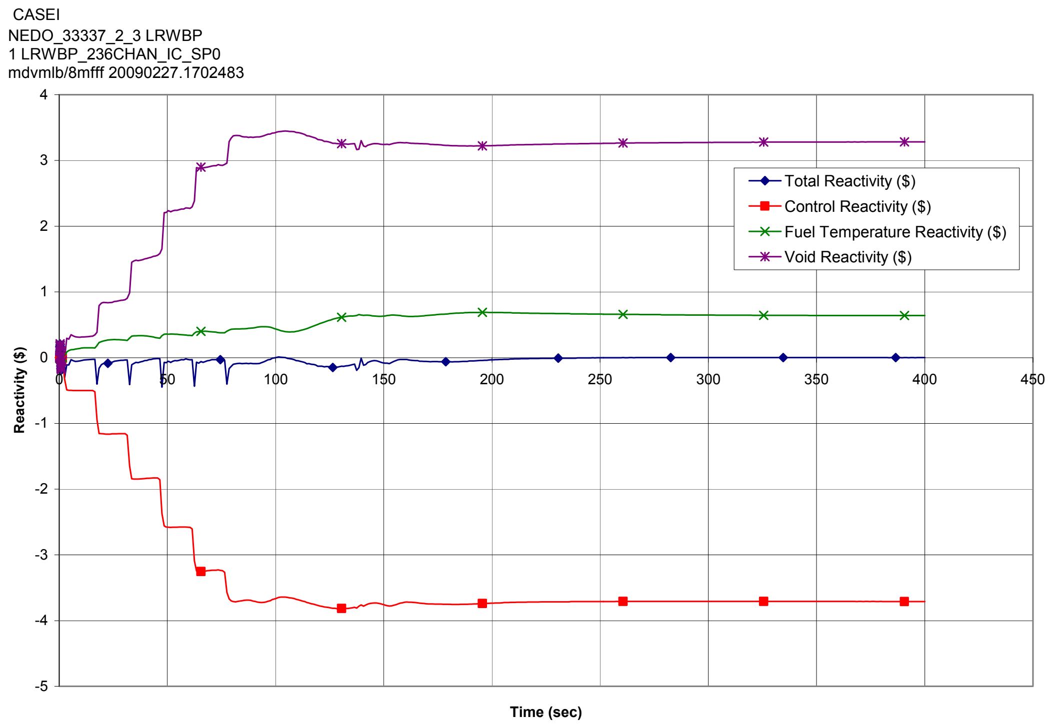



**Figure 2.3-4e. Generator Load Rejection with Turbine Bypass**

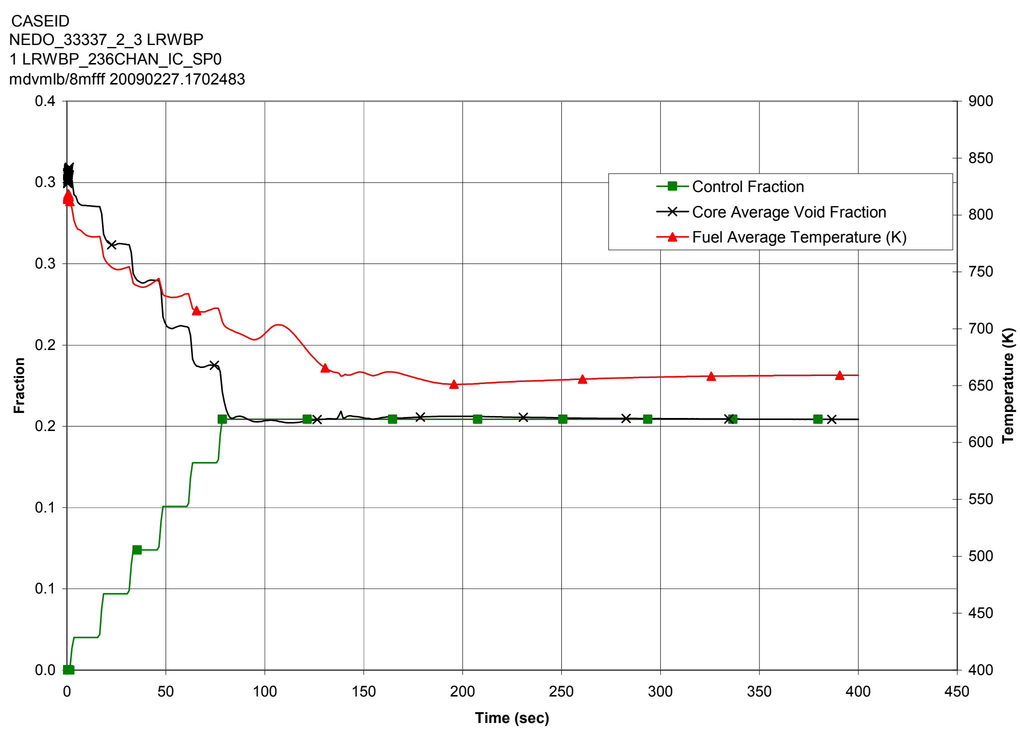



**Figure 2.3-4f. Generator Load Rejection with Turbine Bypass**

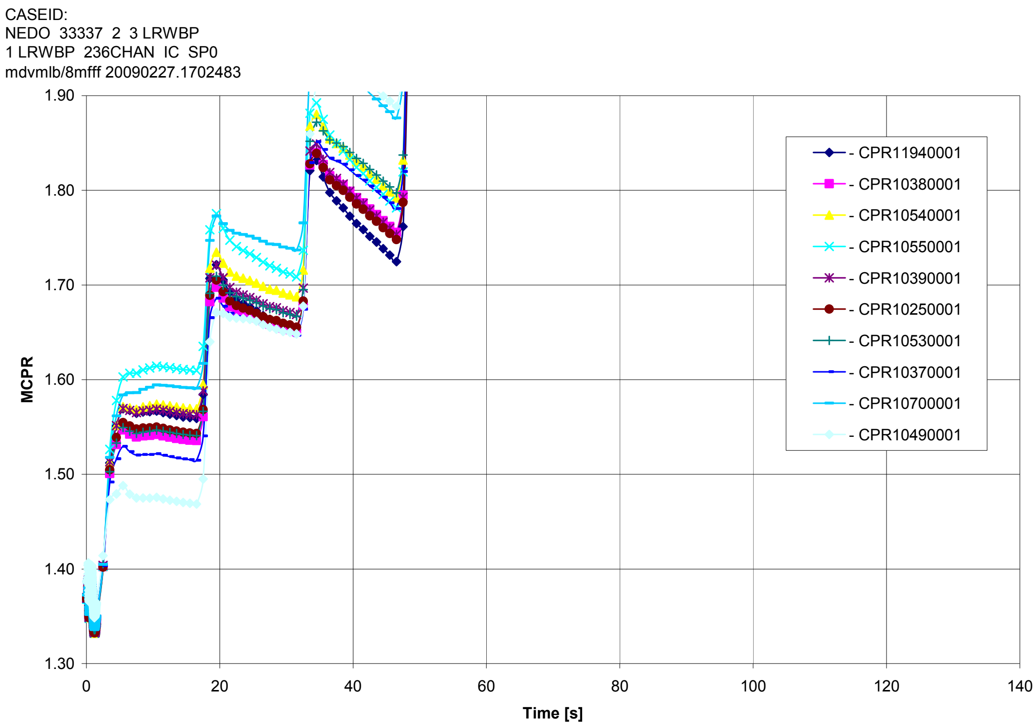



**Figure 2.3-4g. Generator Load Rejection with Turbine Bypass**

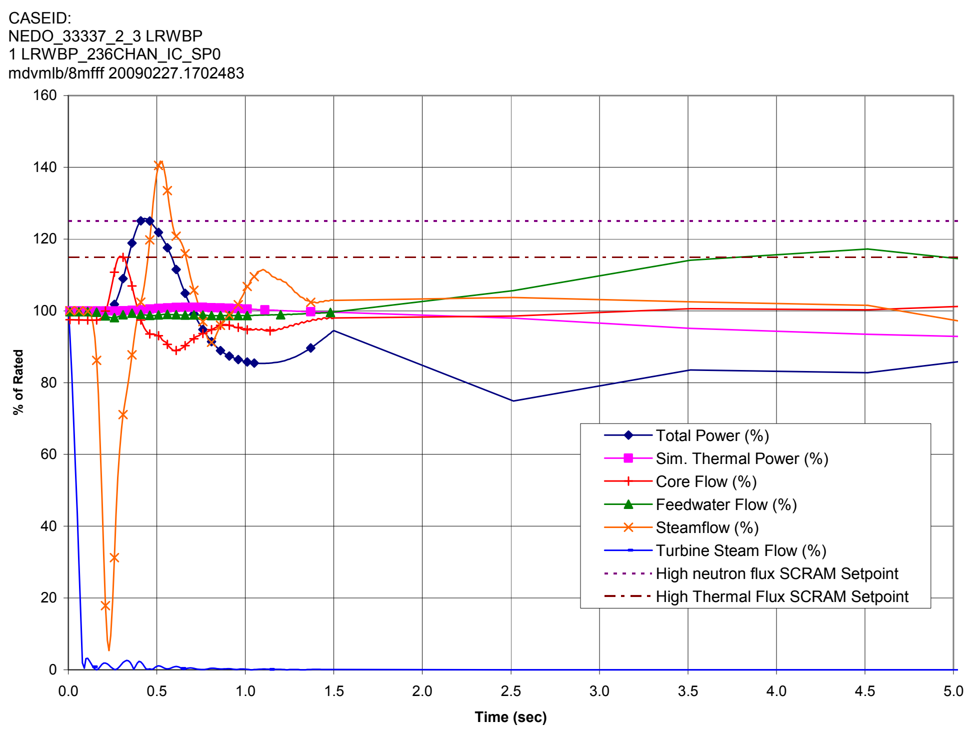



**Figure 2.3-4h. Generator Load Rejection with Turbine Bypass (Figure 2.3-4a from 0 to 5s)**

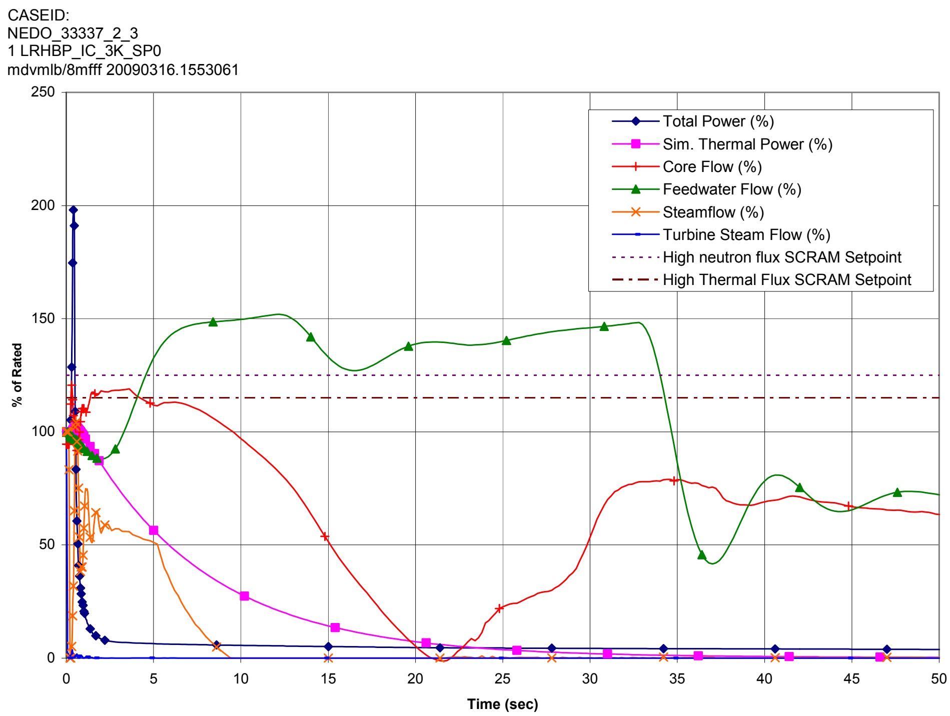



Figure 2.3-5a. Generator Load Rejection with a Single Failure in the Turbine Bypass System

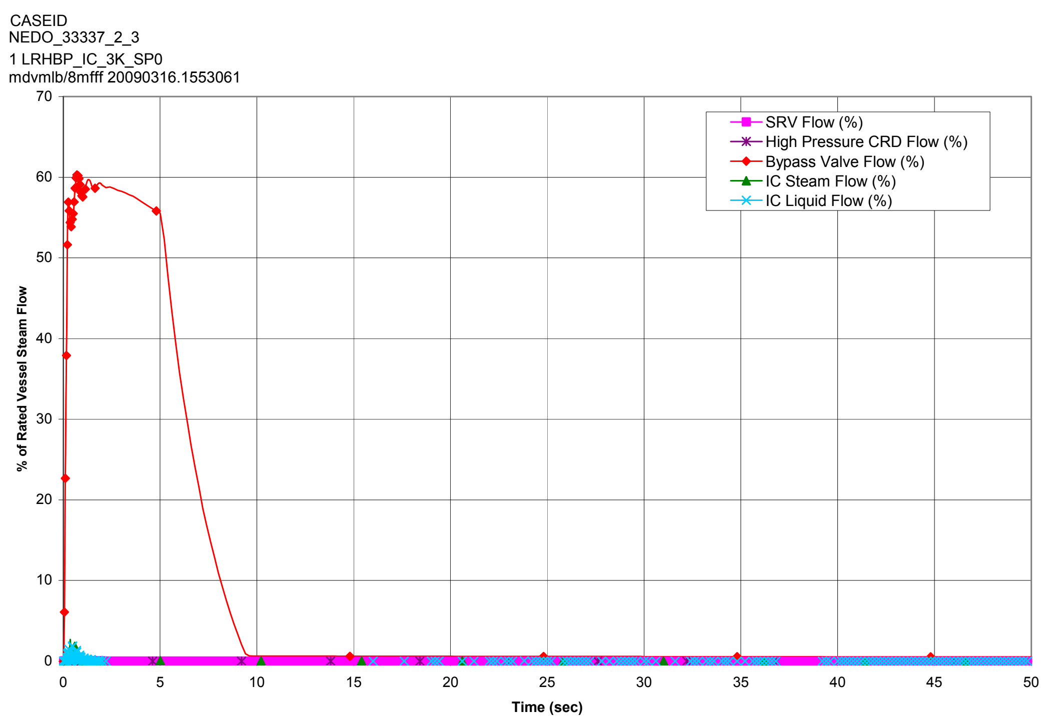



Figure 2.3-5b. Generator Load Rejection with a Single Failure in the Turbine Bypass System

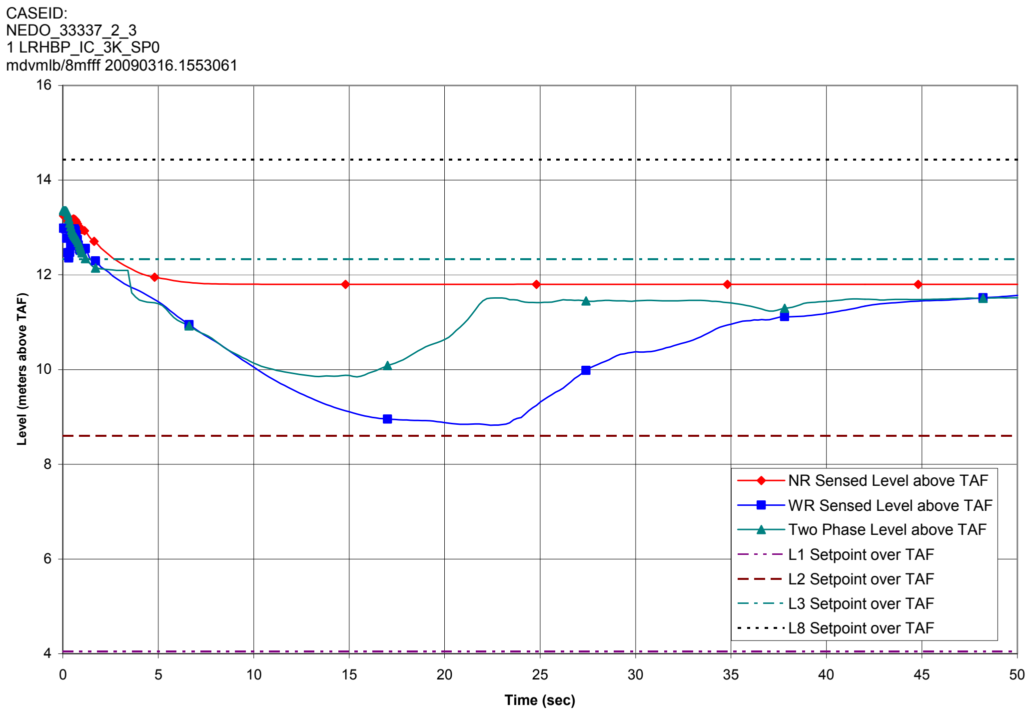



**Figure 2.3-5c. Generator Load Rejection with a Single Failure in the Turbine Bypass System**

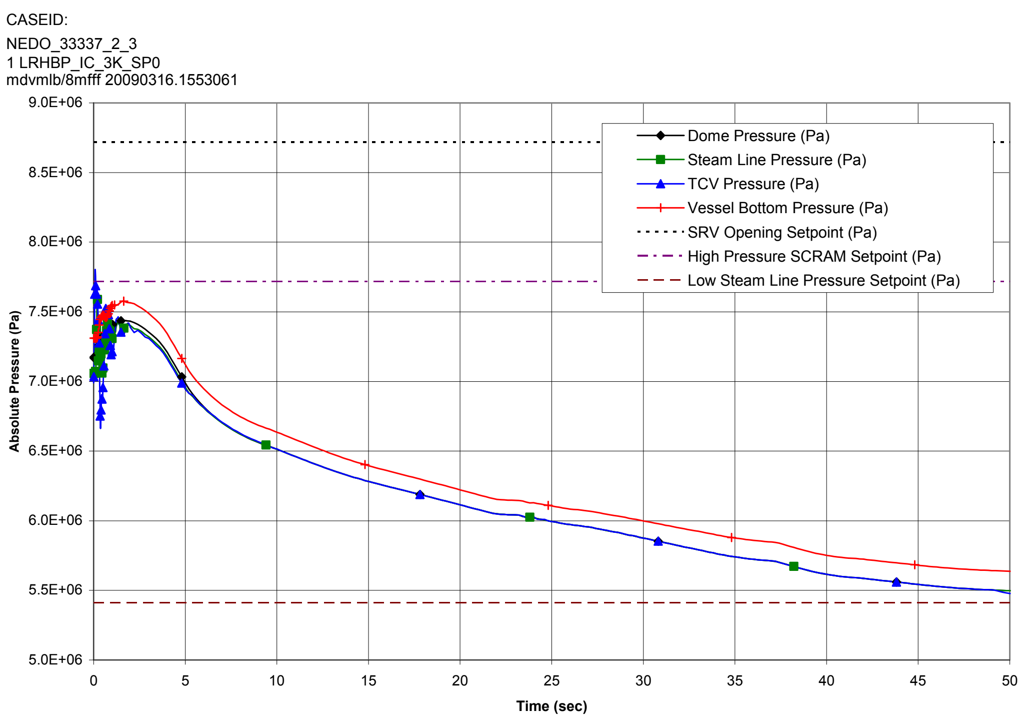



**Figure 2.3-5d. Generator Load Rejection with a Single Failure in the Turbine Bypass System**

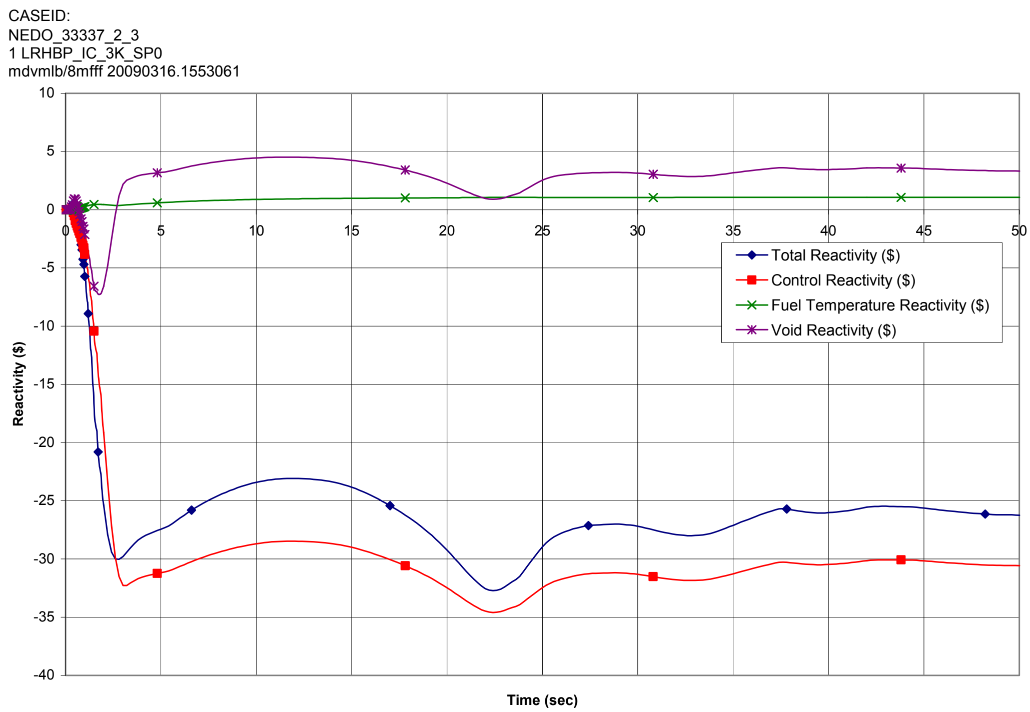



**Figure 2.3-5e. Generator Load Rejection with a Single Failure in the Turbine Bypass System**

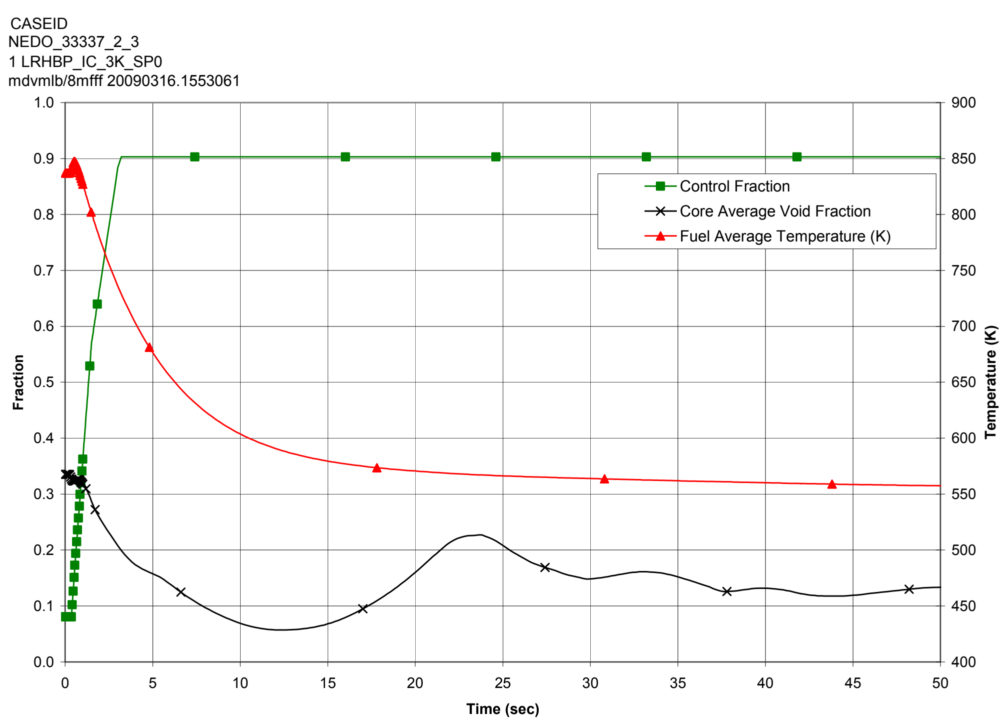



**Figure 2.3-5f. Generator Load Rejection with a Single Failure in the Turbine Bypass System**

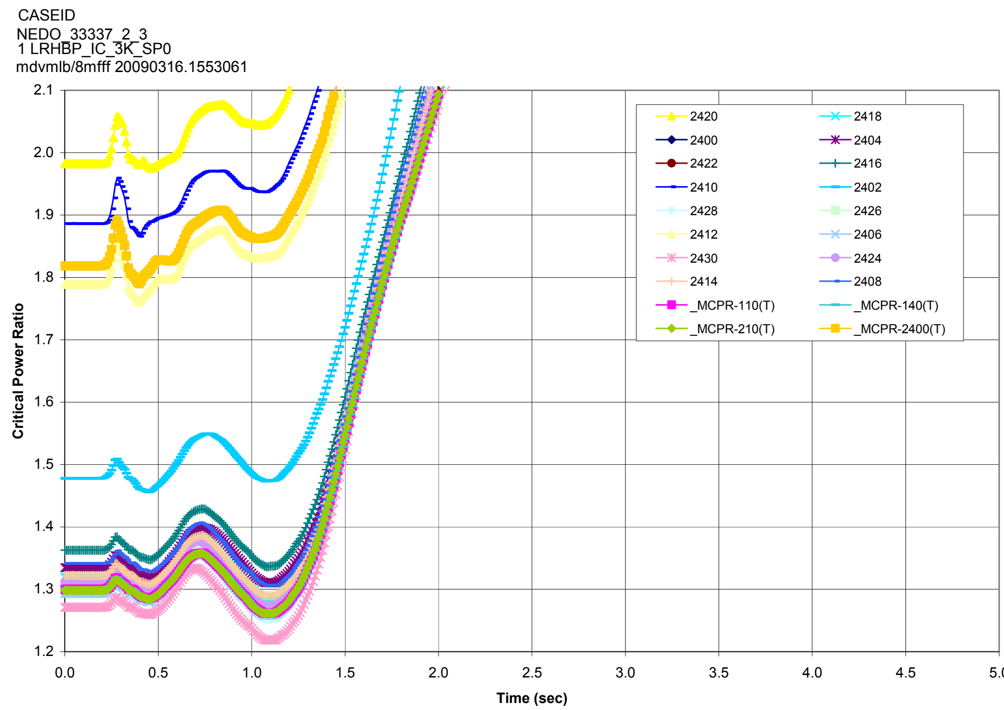



**Figure 2.3-5g. Generator Load Rejection with a Single Failure in the Turbine Bypass System**

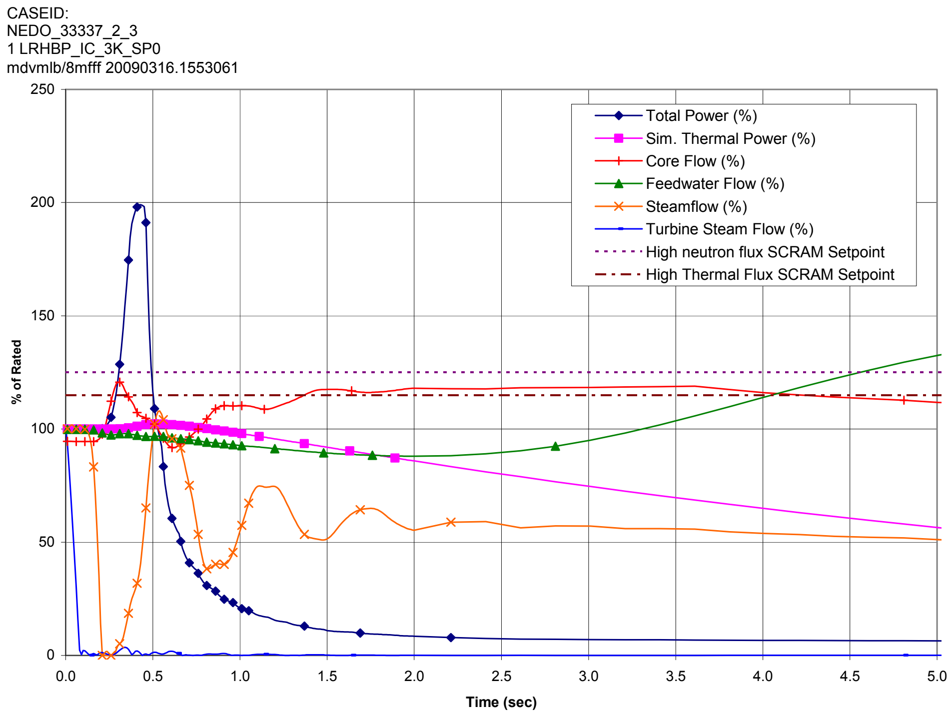



**Figure 2.3-5h. Generator Load Rejection with a Single Failure in the Turbine Bypass System (Figure 2.3-5a from 0 to 5s)**

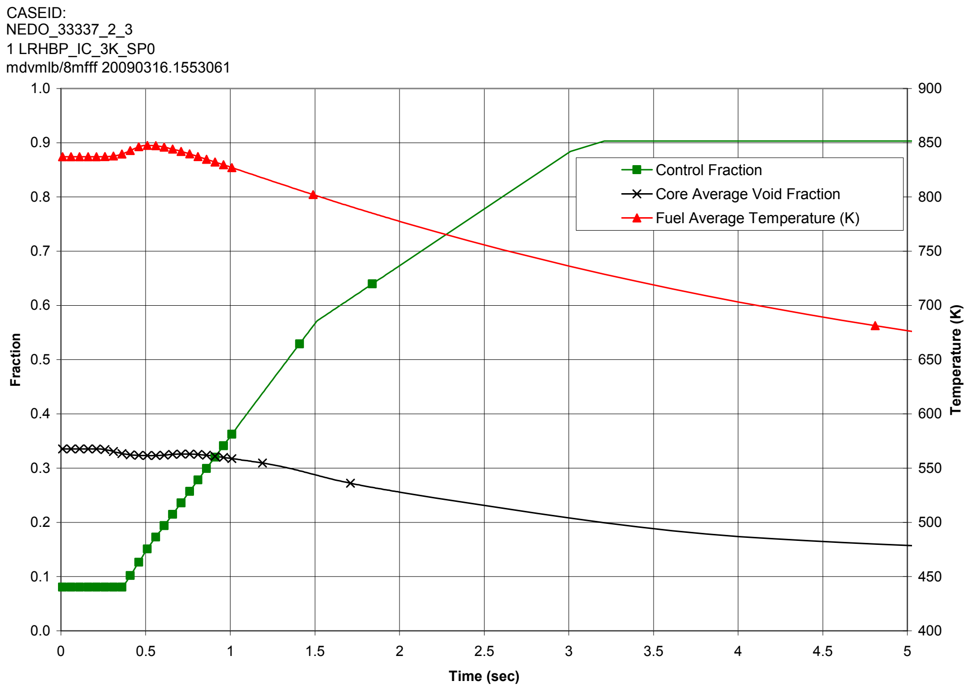



**Figure 2.3-5i. Generator Load Rejection with a Single Failure in the Turbine Bypass System (Figure 2.3-5f from 0 to 5s)**

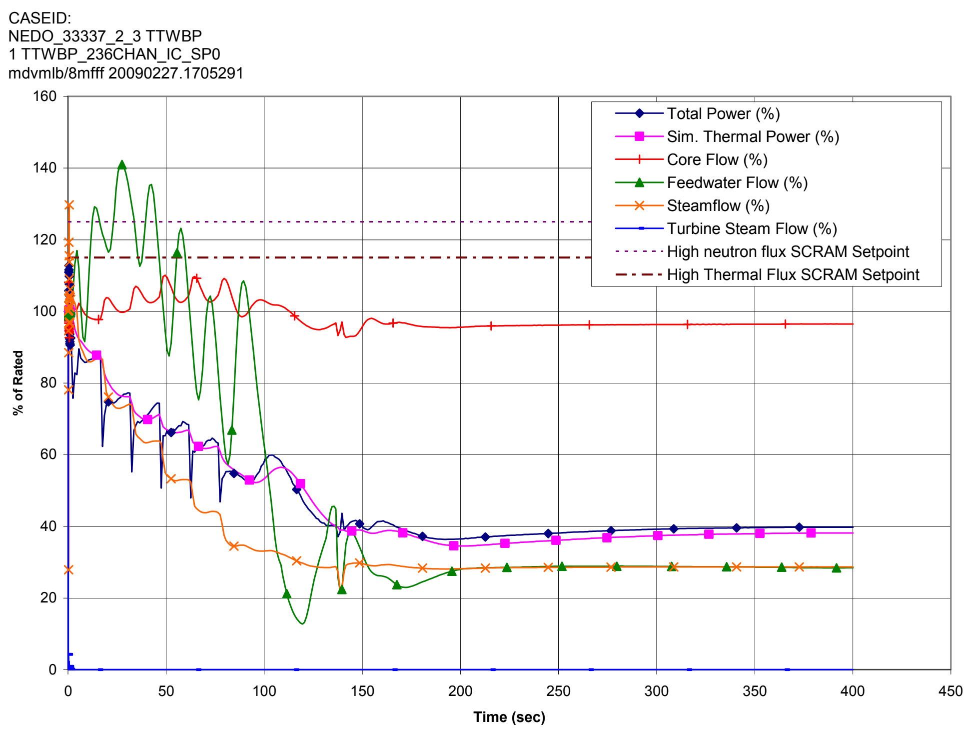



**Figure 2.3-6a. Turbine Trip with Turbine Bypass**

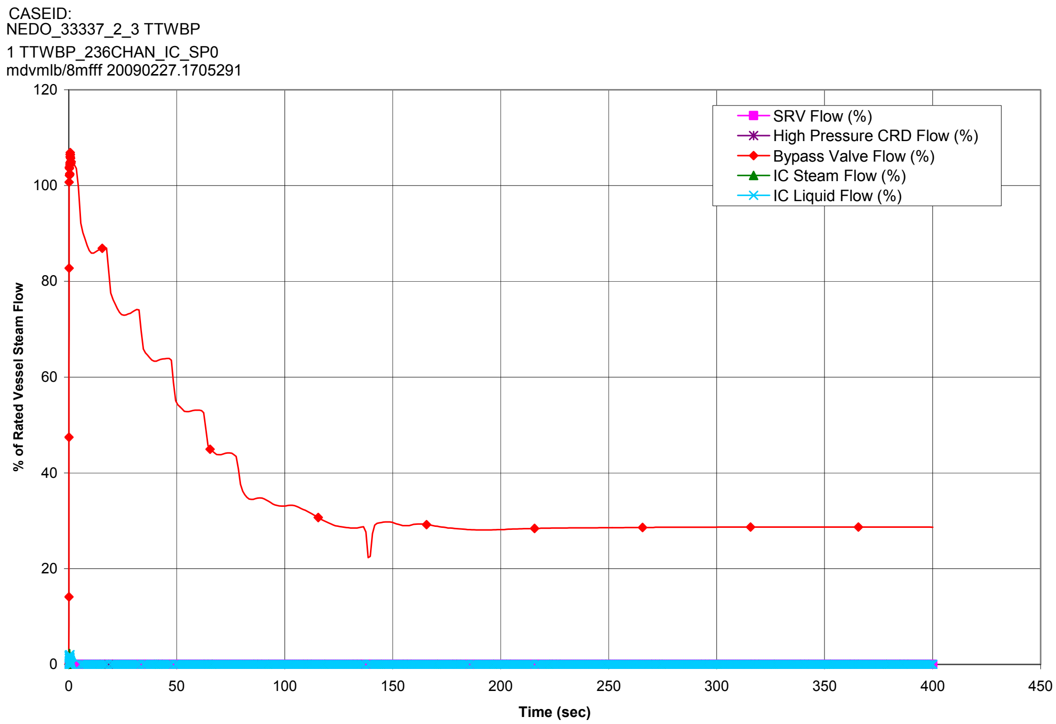



**Figure 2.3-6b. Turbine Trip with Turbine Bypass**

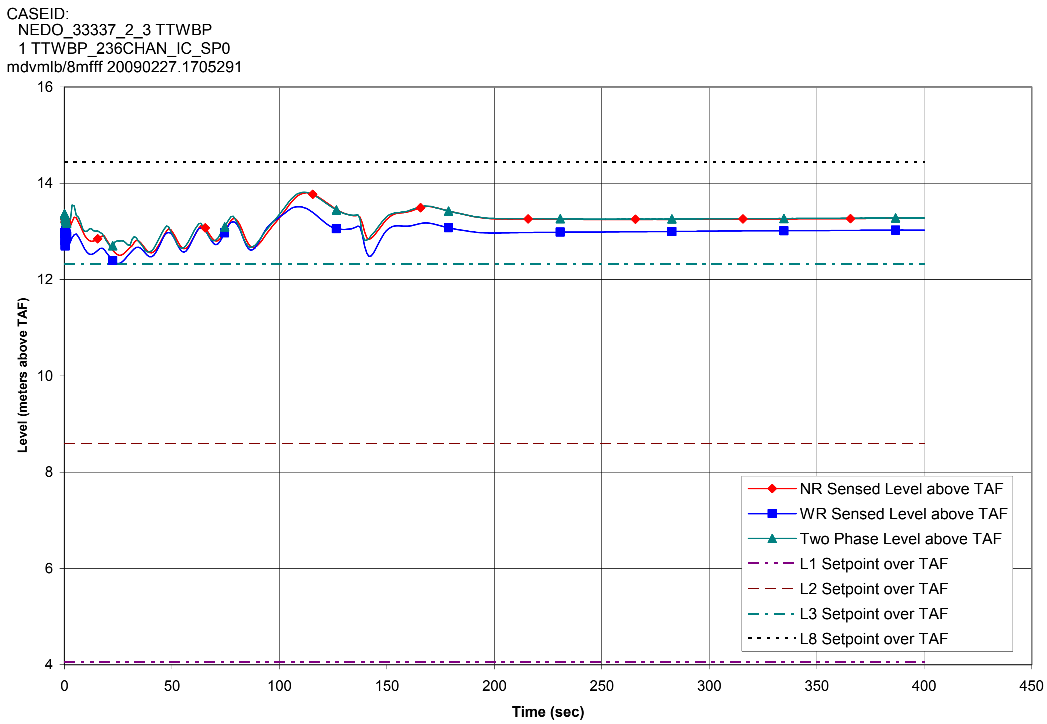



**Figure 2.3-6c. Turbine Trip with Turbine Bypass**

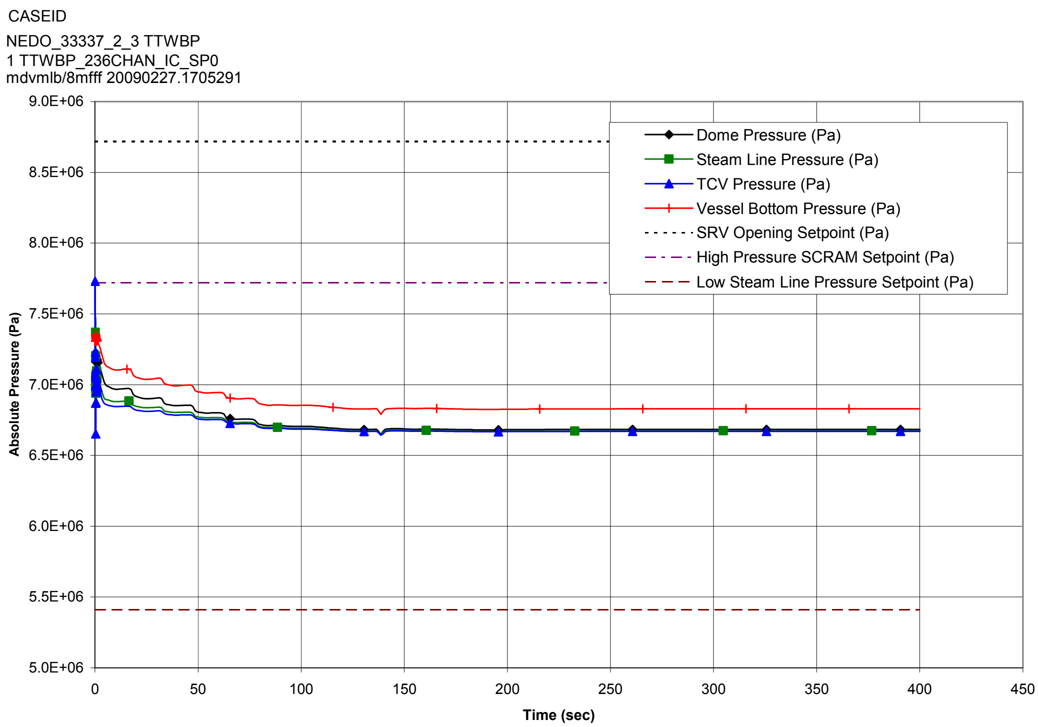



**Figure 2.3-6d. Turbine Trip with Turbine Bypass**

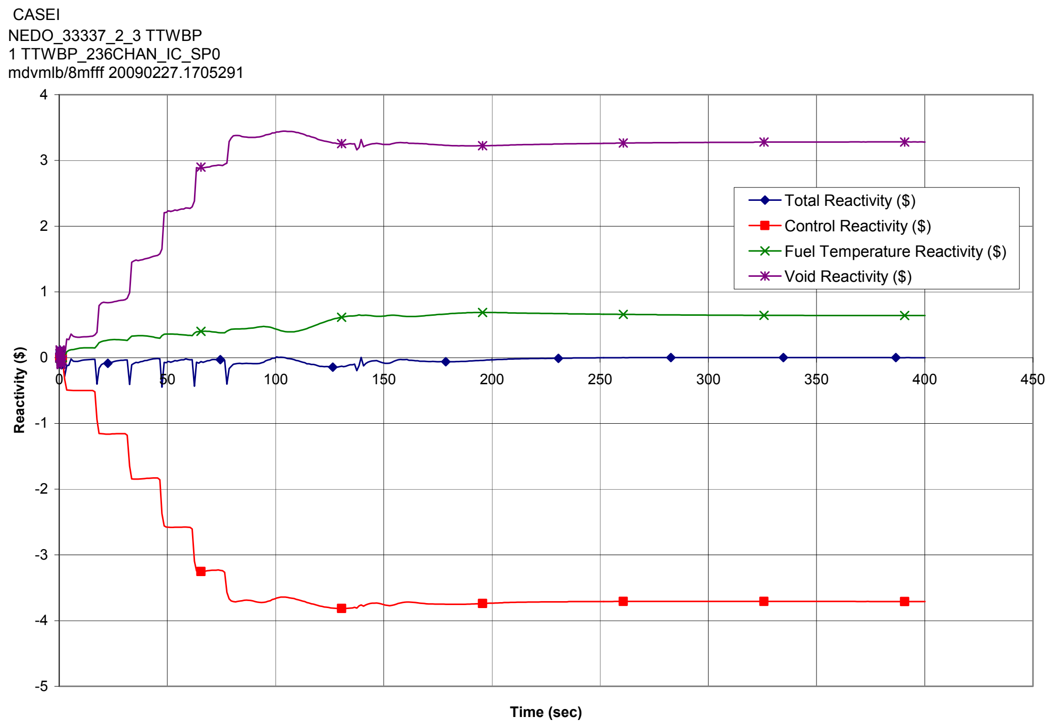



**Figure 2.3-6e. Turbine Trip with Turbine Bypass**

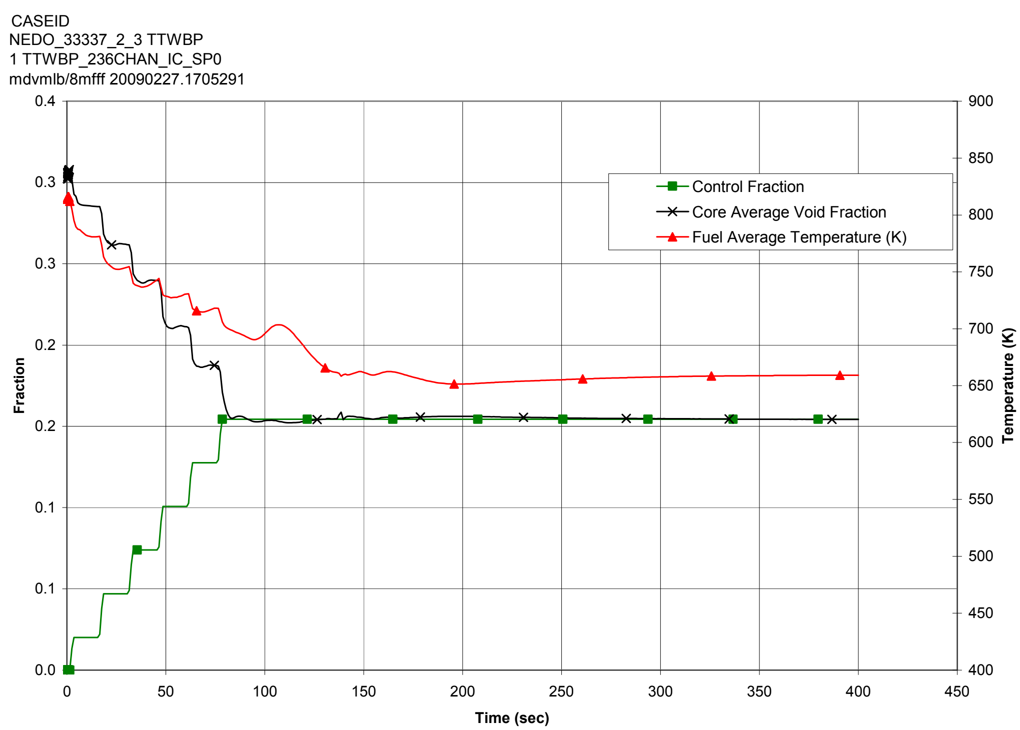



**Figure 2.3-6f. Turbine Trip with Turbine Bypass**

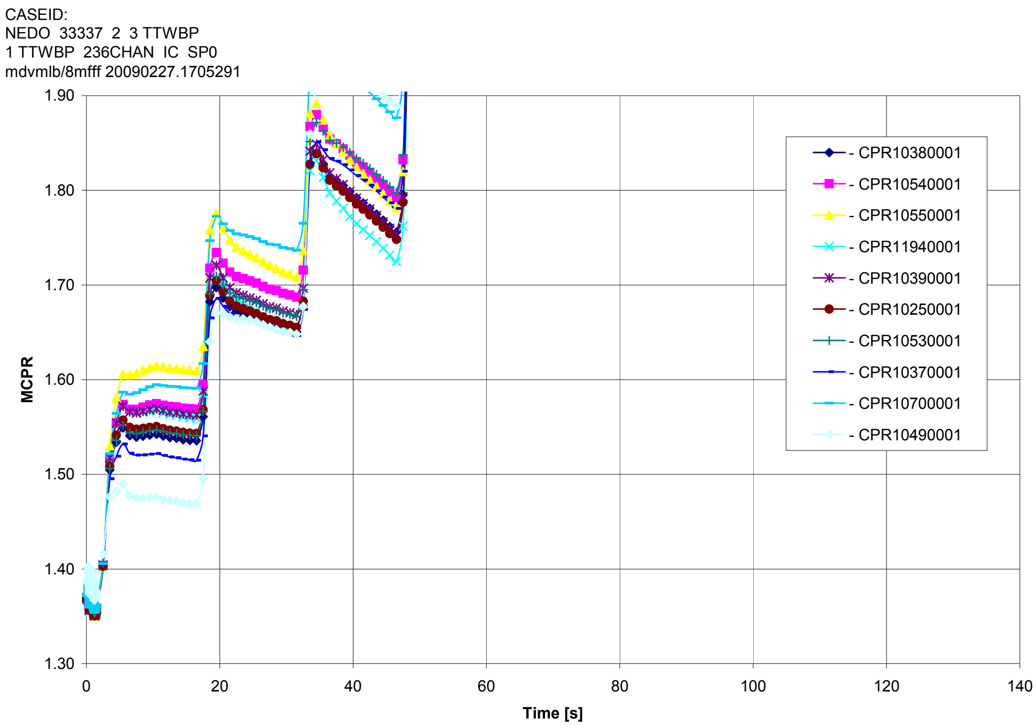



**Figure 2.3-6g. Turbine Trip with Turbine Bypass**

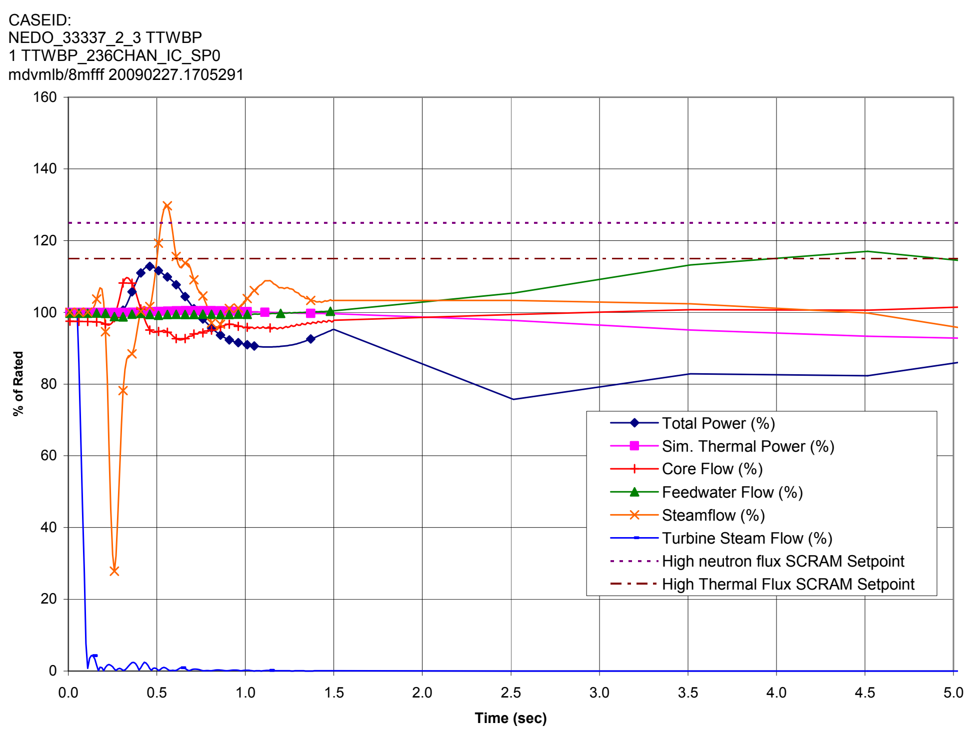



**Figure 2.3-6h. Turbine Trip with Turbine Bypass (Figure 2.3-6a from 0 to 5s)**

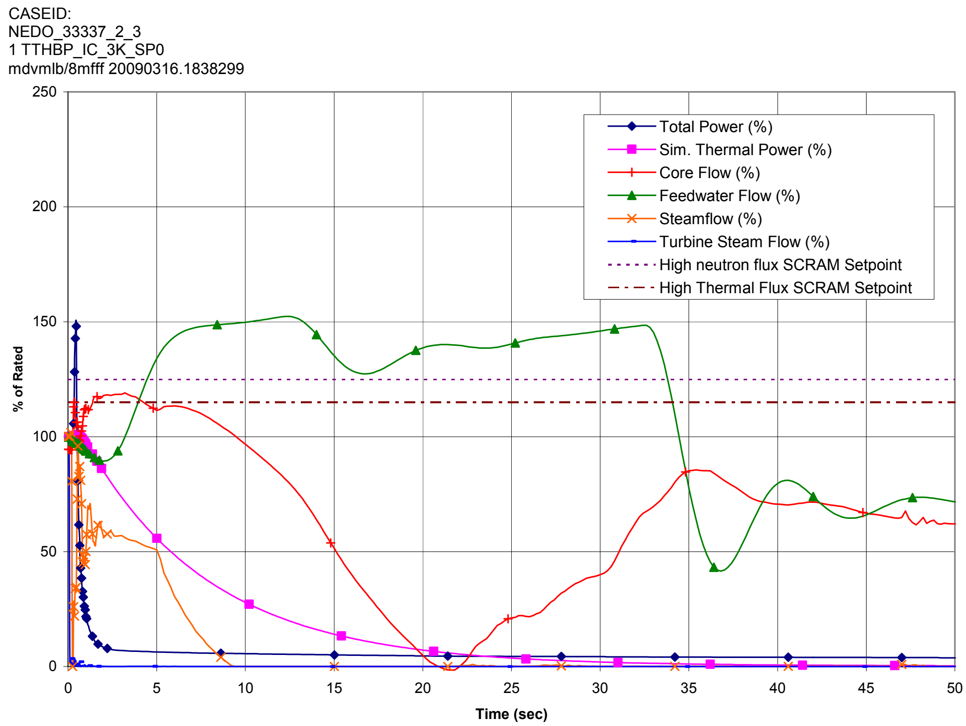



Figure 2.3-7a. Turbine Trip with a Single Failure in the Turbine Bypass System

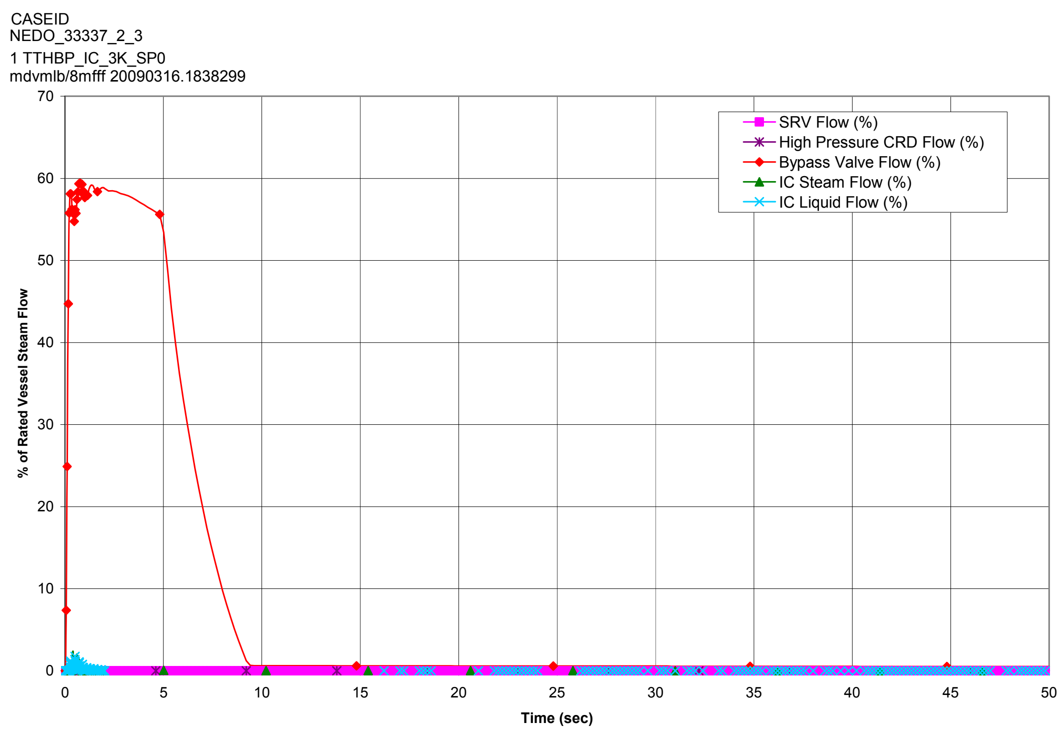



Figure 2.3-7b. Turbine Trip with a Single Failure in the Turbine Bypass System

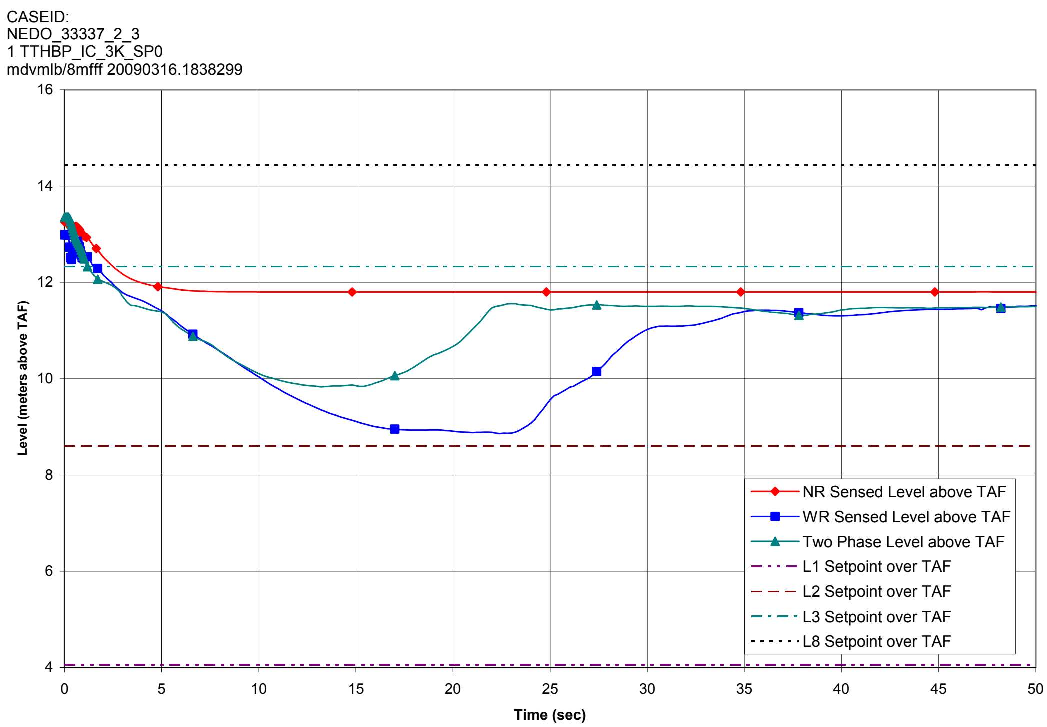



**Figure 2.3-7c. Turbine Trip with a Single Failure in the Turbine Bypass System**

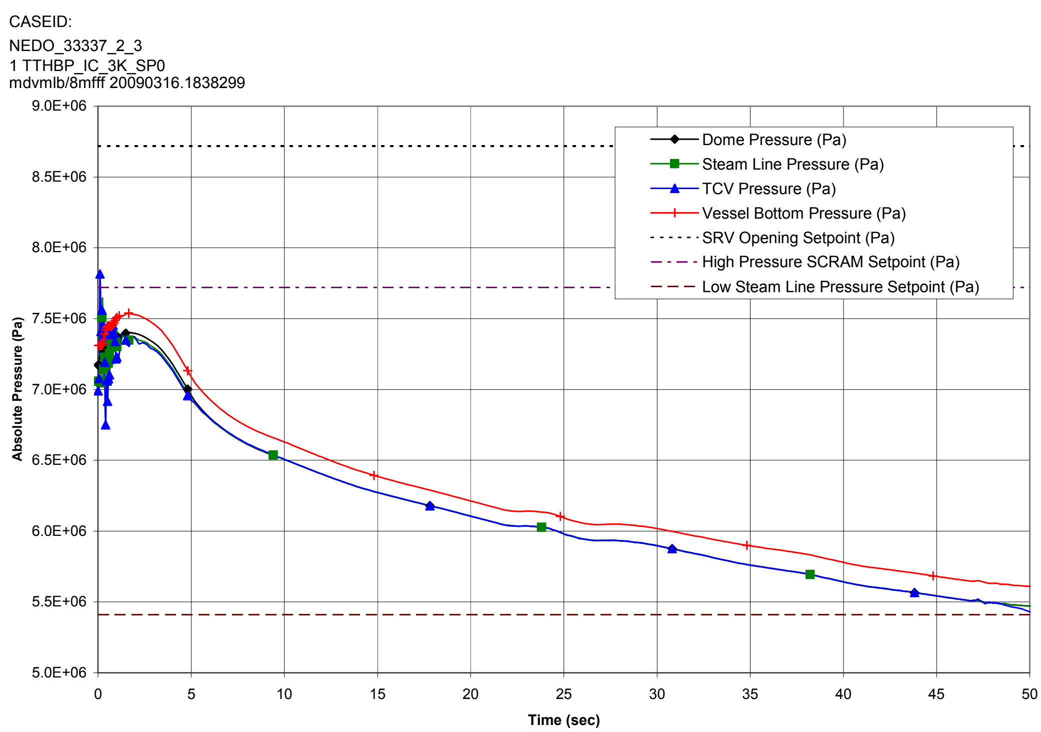



**Figure 2.3-7d. Turbine Trip with a Single Failure in the Turbine Bypass System**

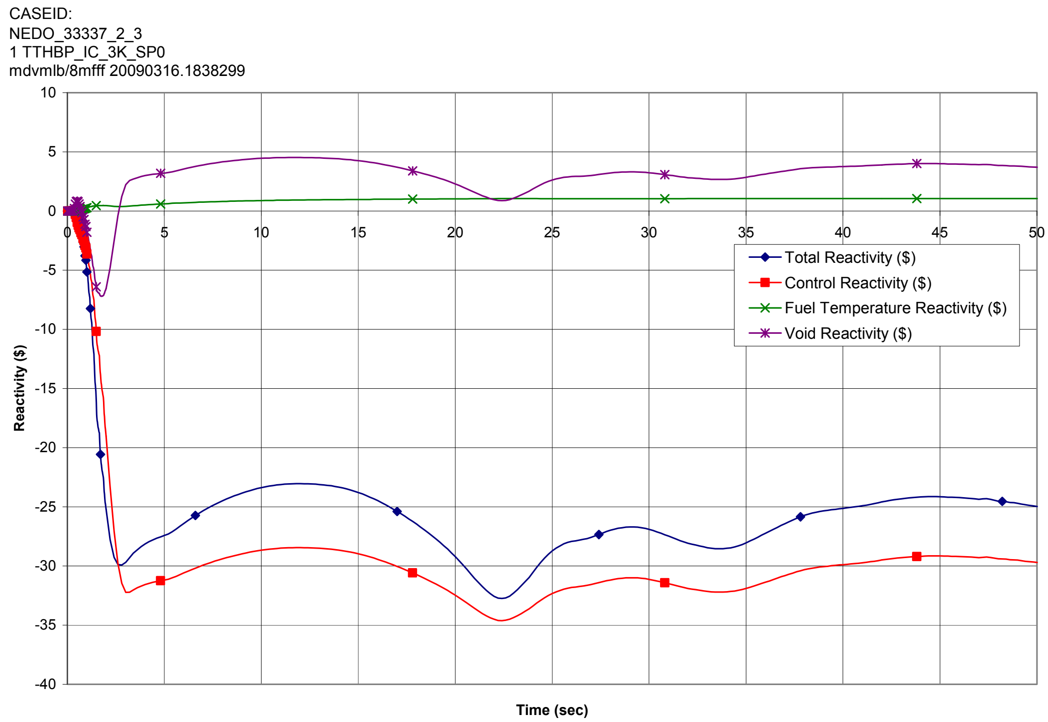



**Figure 2.3-7e. Turbine Trip with a Single Failure in the Turbine Bypass System**

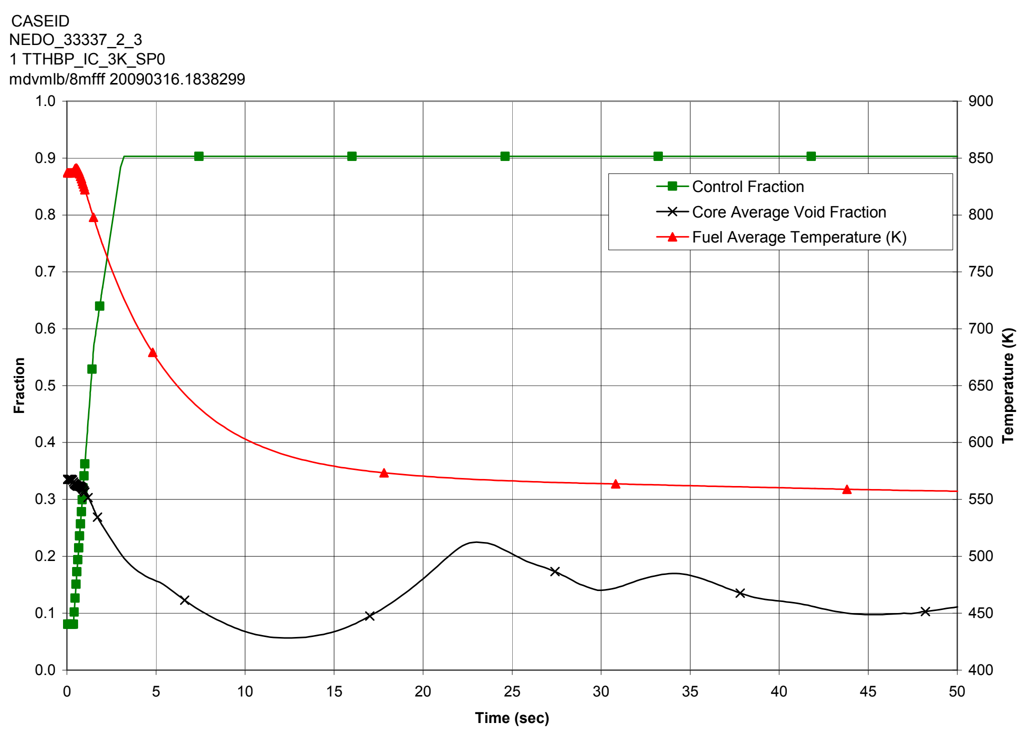



**Figure 2.3-7f. Turbine Trip with a Single Failure in the Turbine Bypass System**

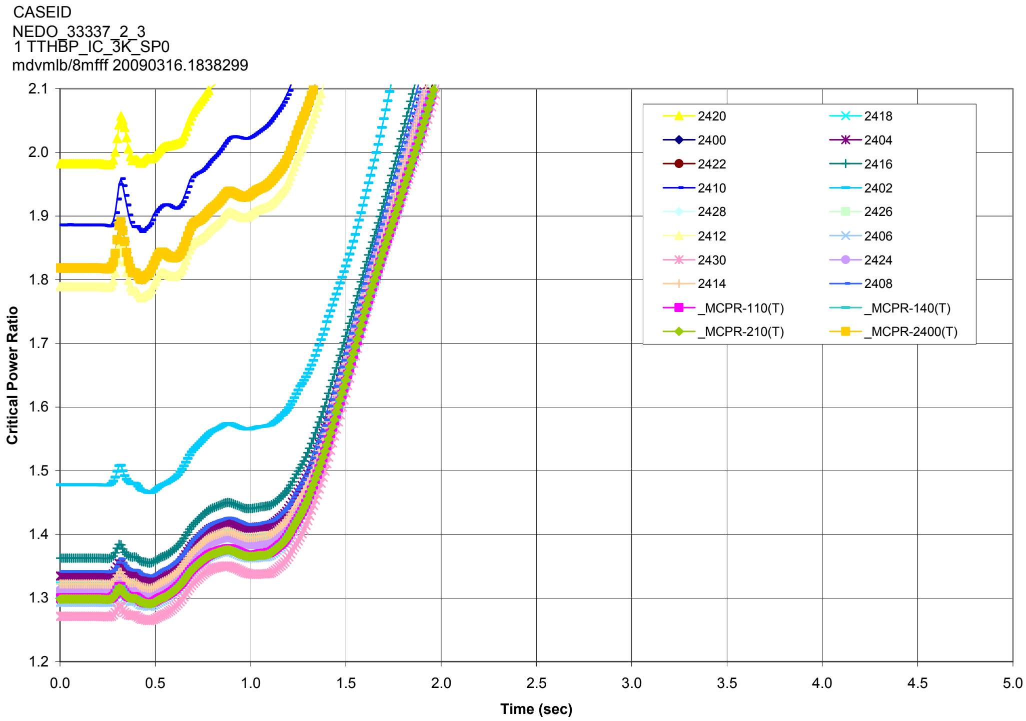



**Figure 2.3-7g. Turbine Trip with a Single Failure in the Turbine Bypass System**

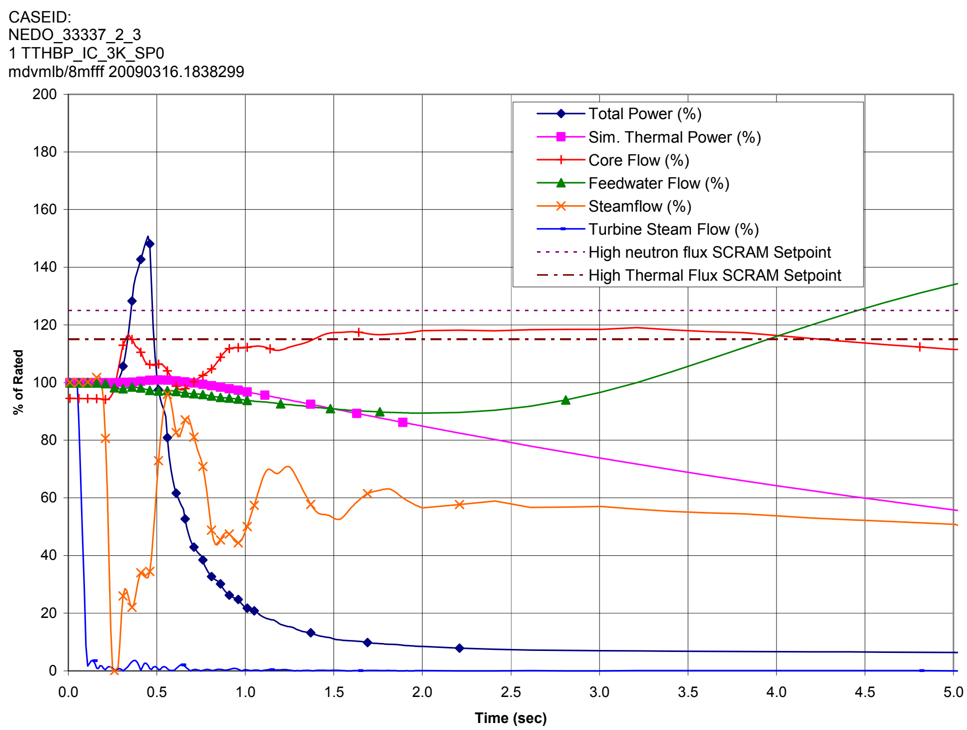



**Figure 2.3-7h. Turbine Trip with a Single Failure in the Turbine Bypass System (Figure 2.3-7a from 0 to 5s)**

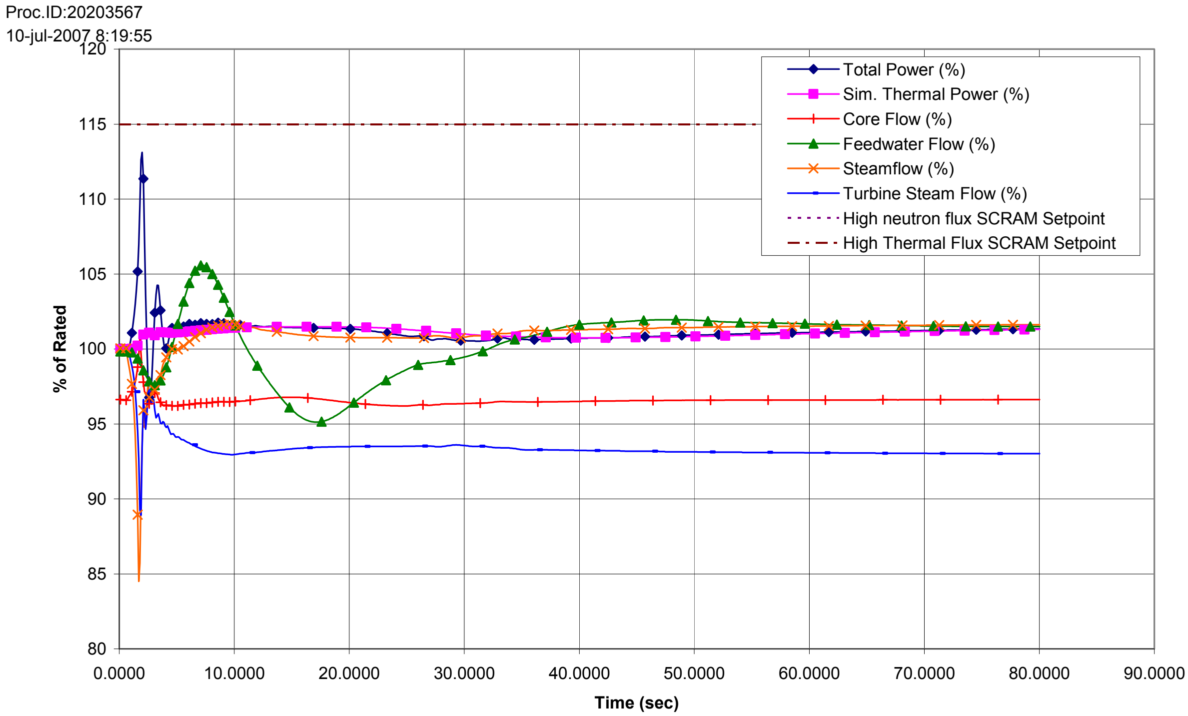



**Figure 2.3-8a. One MSIV Closure**

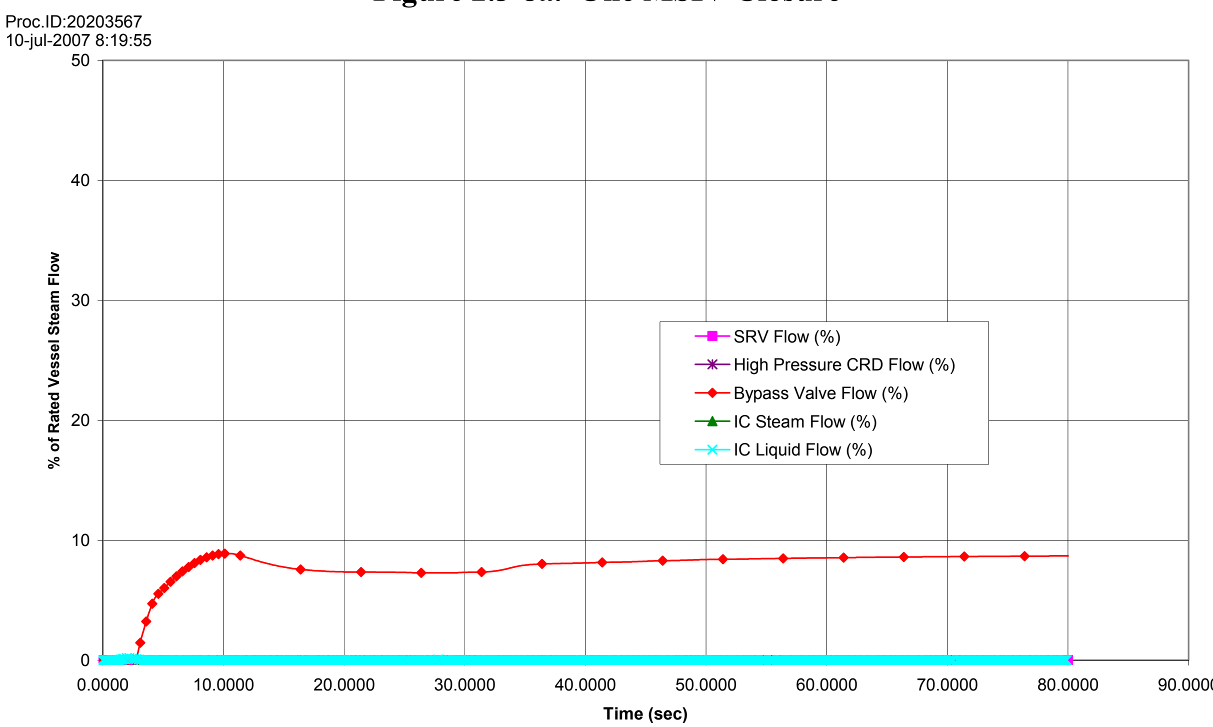



**Figure 2.3-8b. One MSIV Closure**

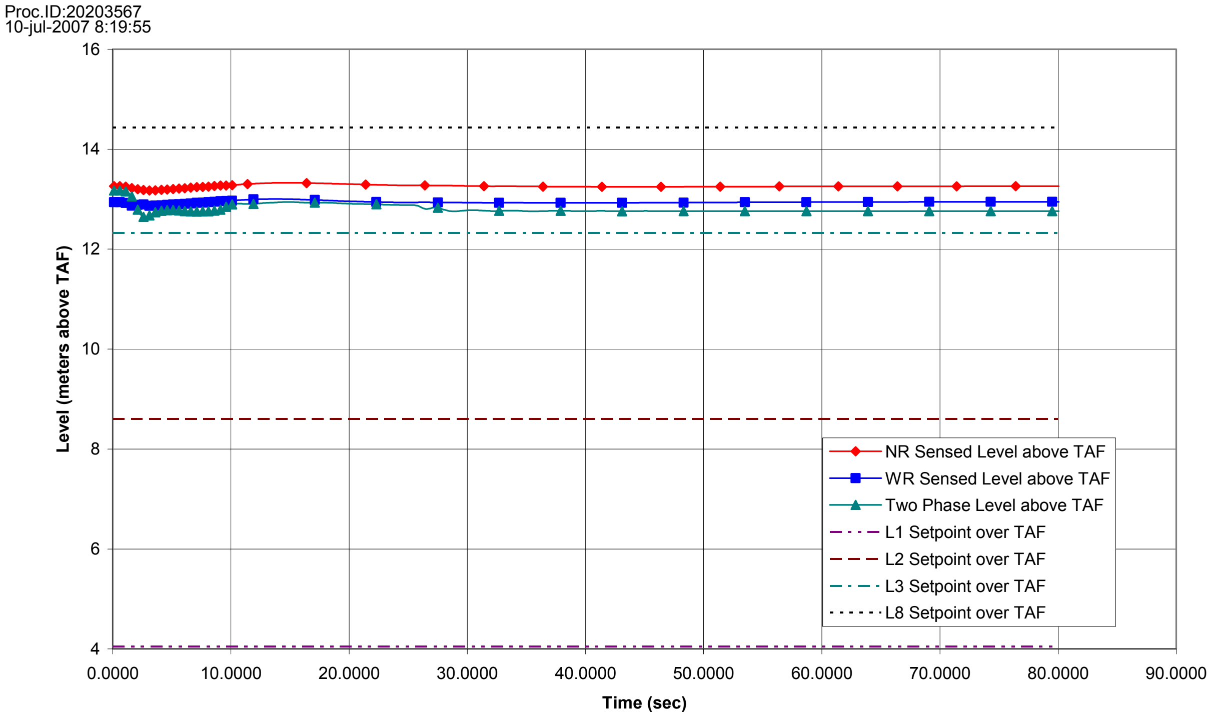



**Figure 2.3-8c. One MSIV Closure**

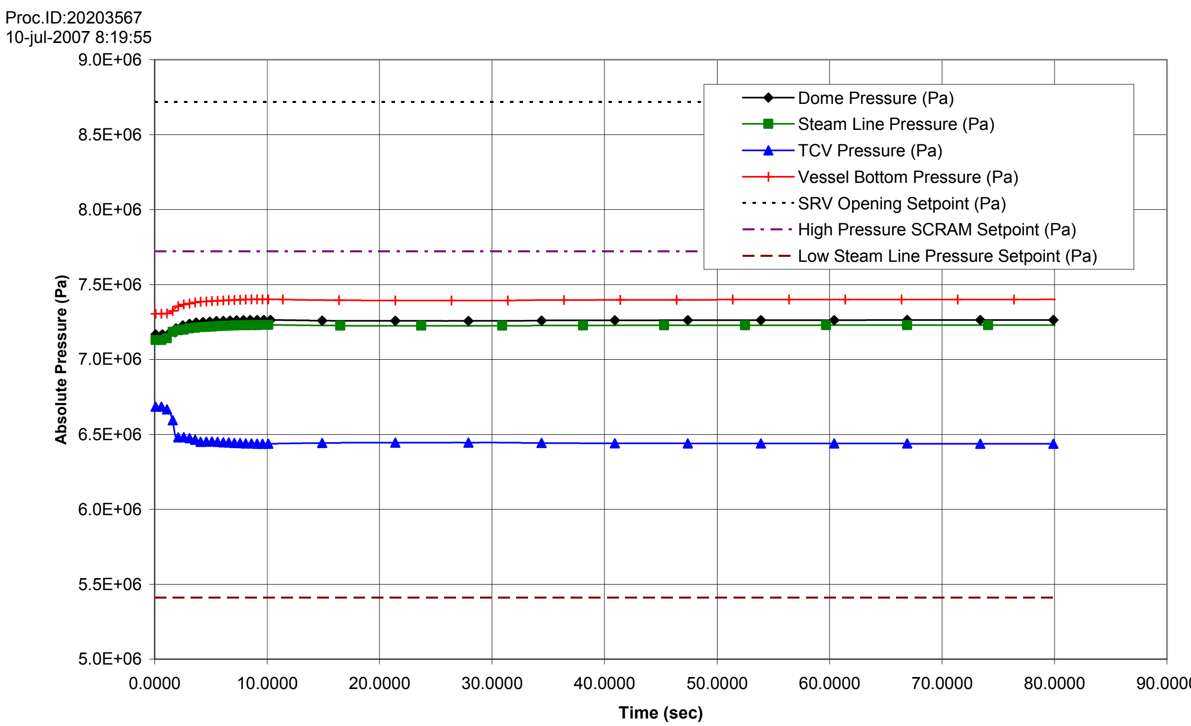



**Figure 2.3-8d. One MSIV Closure**

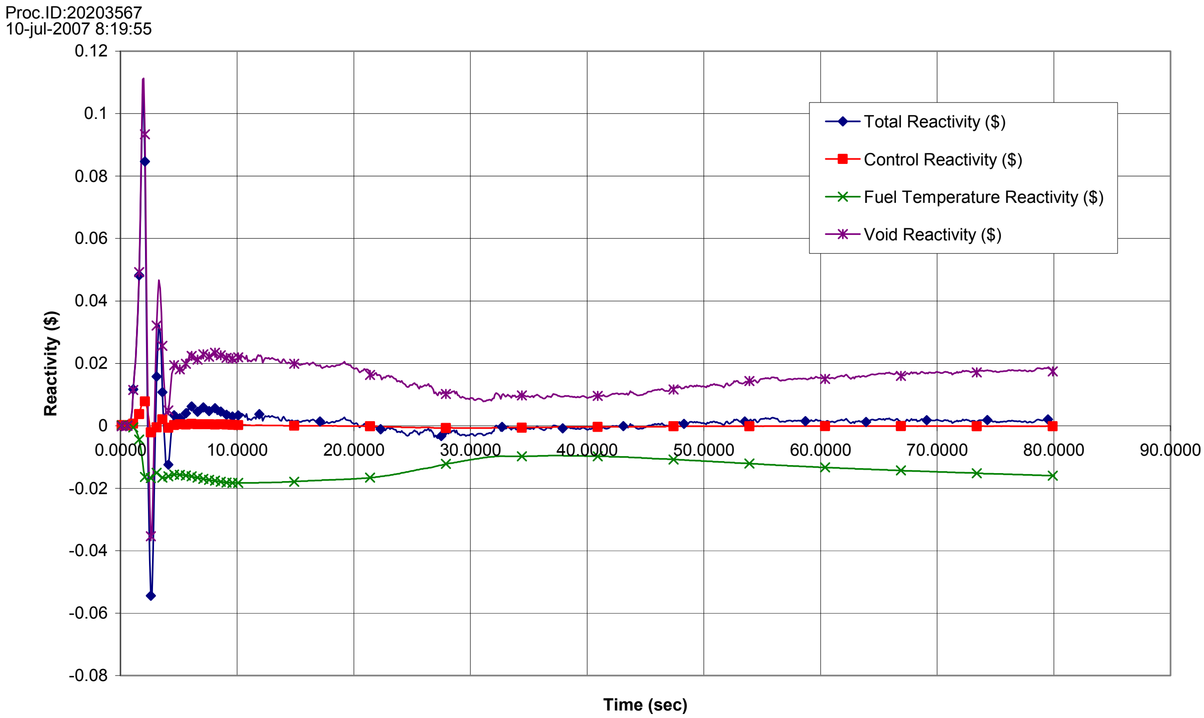



**Figure 2.3-8e. One MSIV Closure**

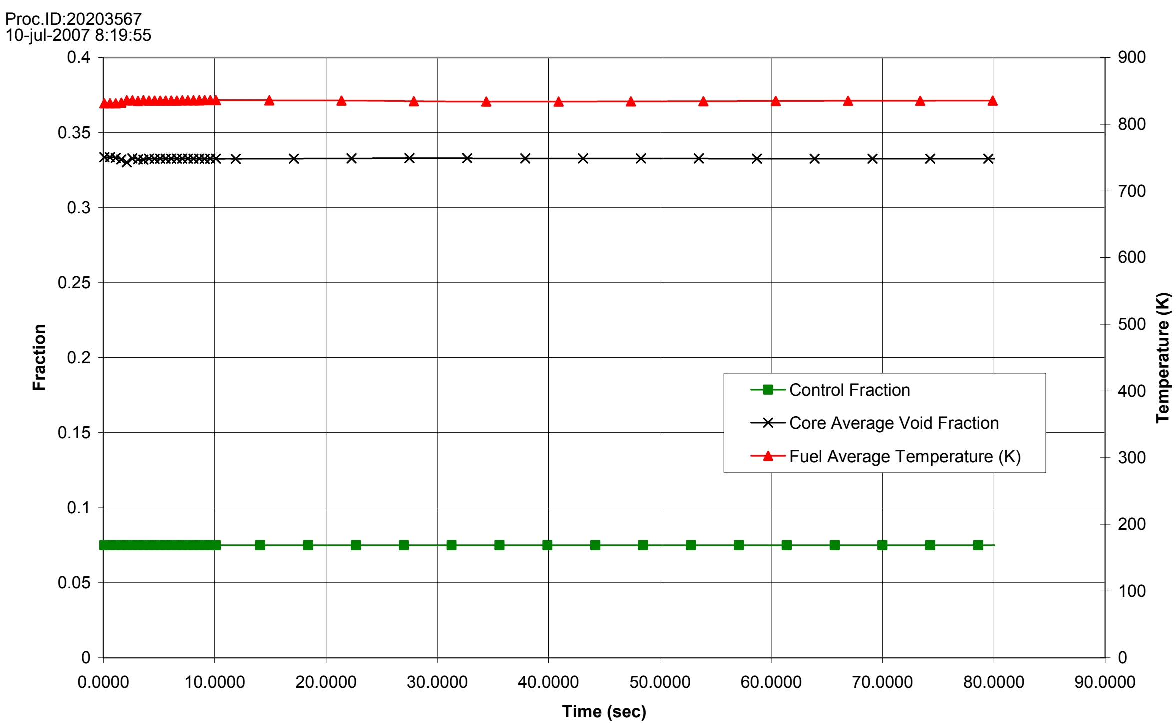



**Figure 2.3-8f. One MSIV Closure**

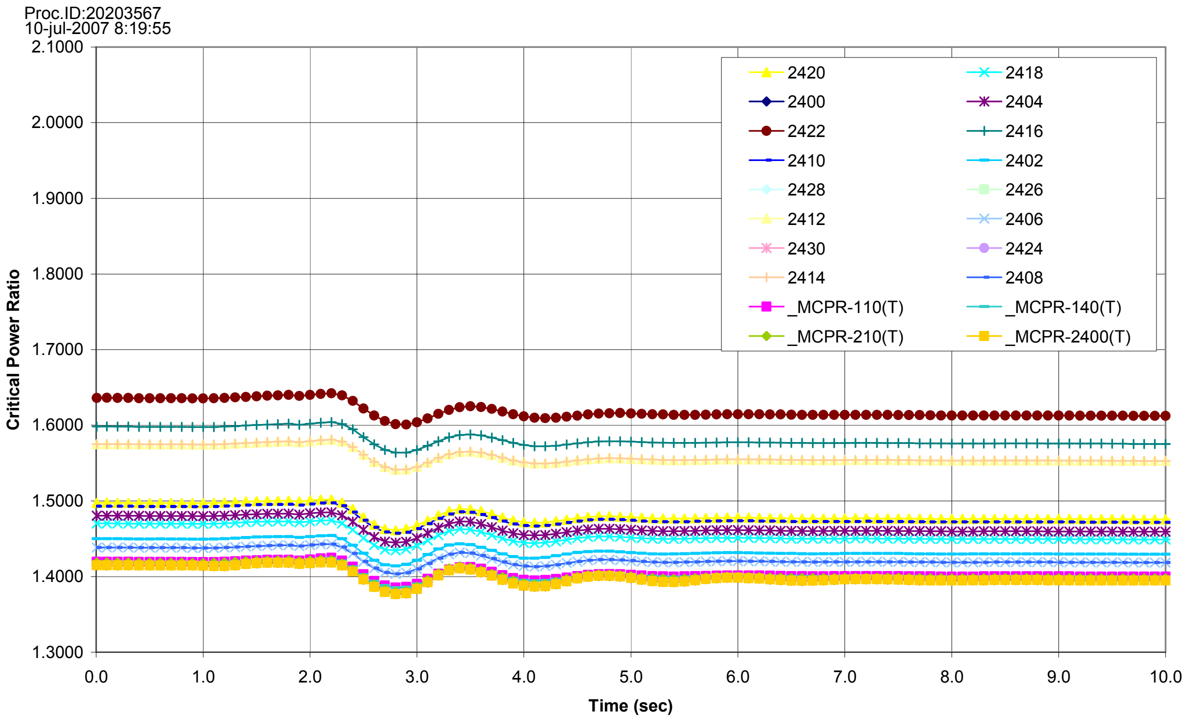



**Figure 2.3-8g. One MSIV Closure**

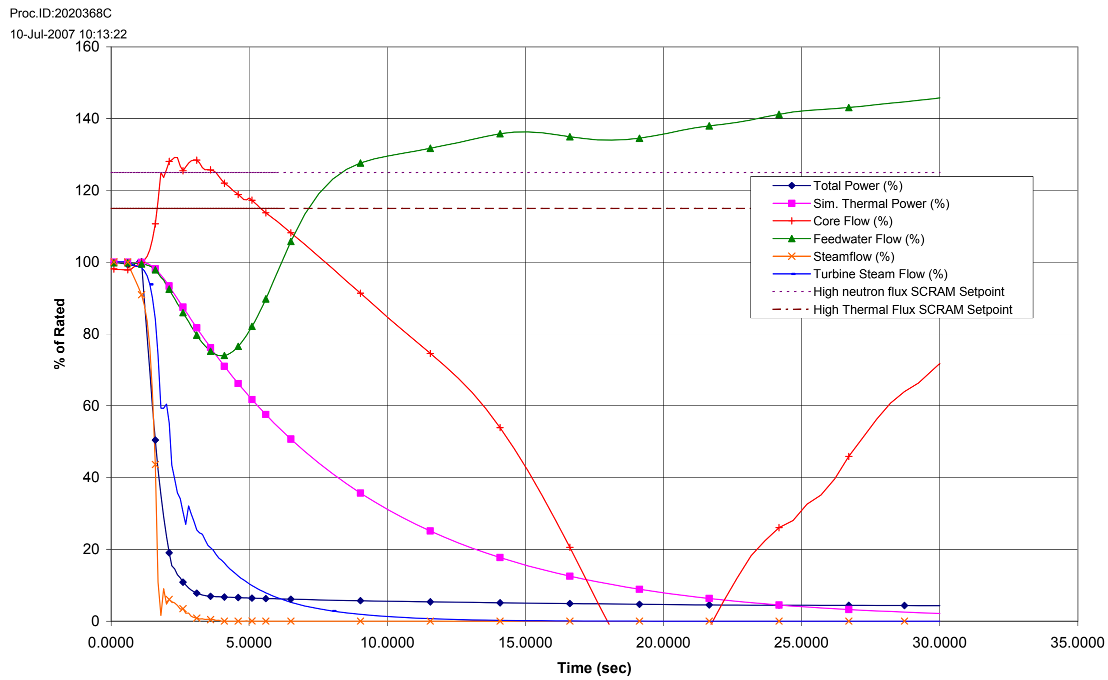



**Figure 2.3-9a. MSIV Closure**

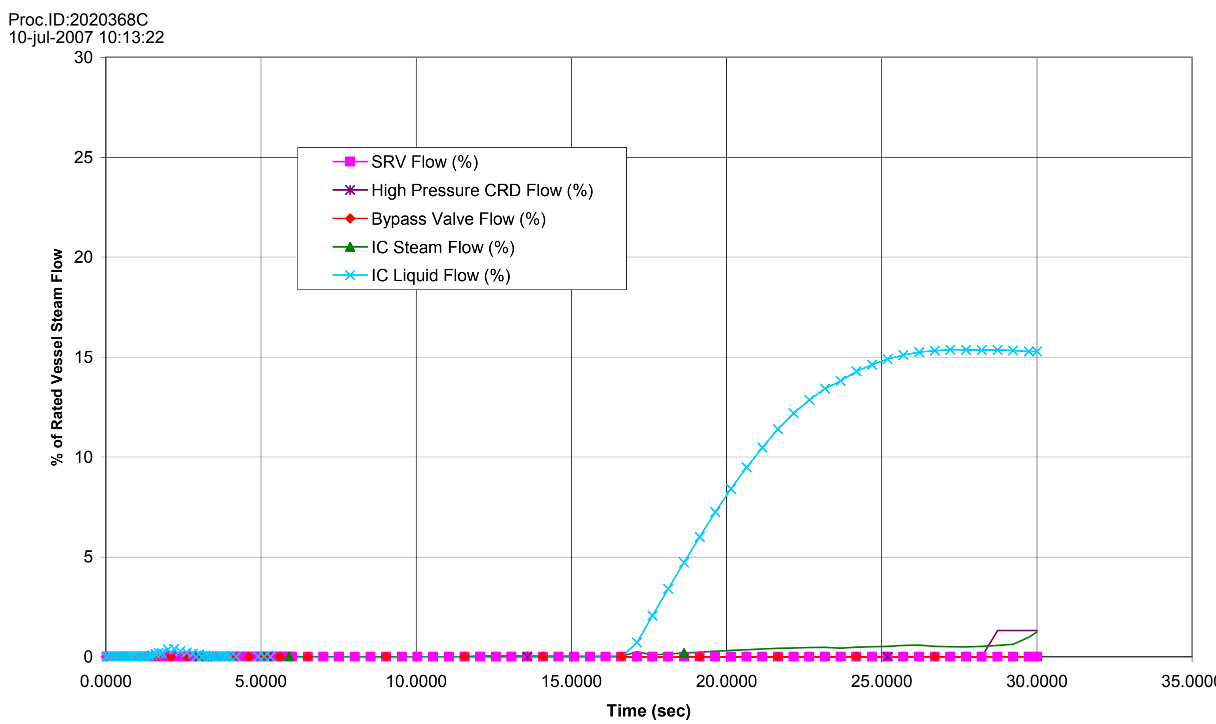



**Figure 2.3-9b. MSIV Closure**

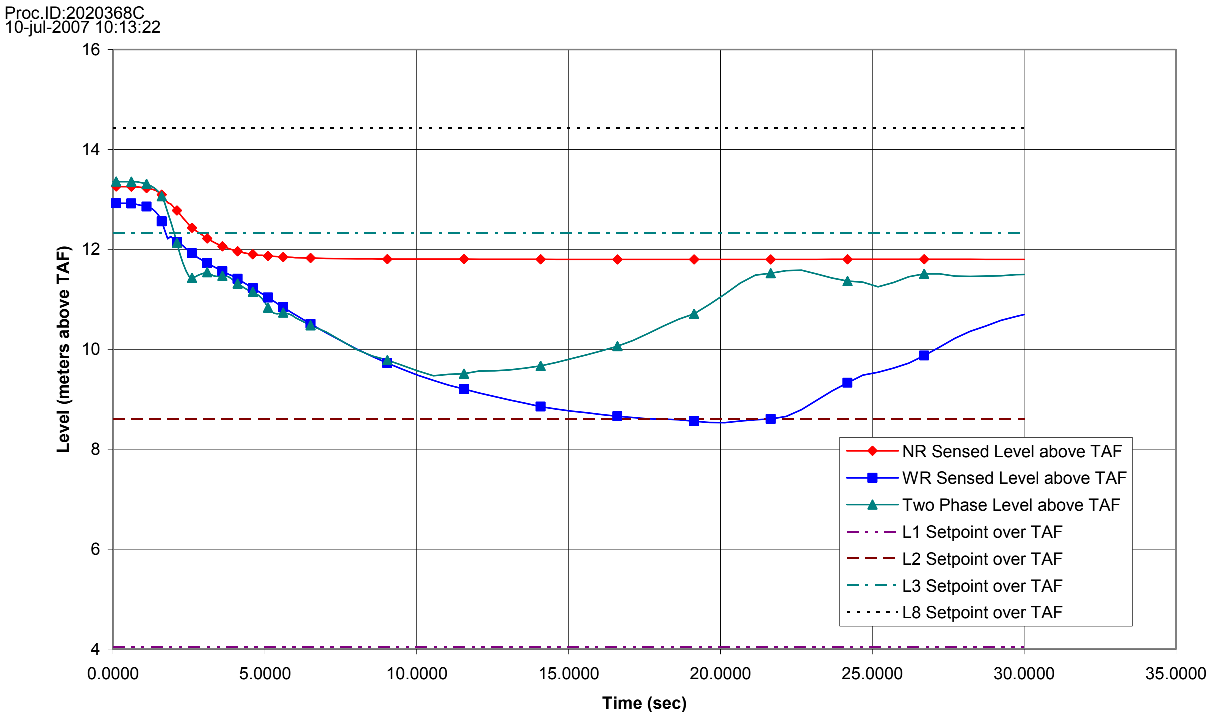



**Figure 2.3-9c. MSIV Closure**

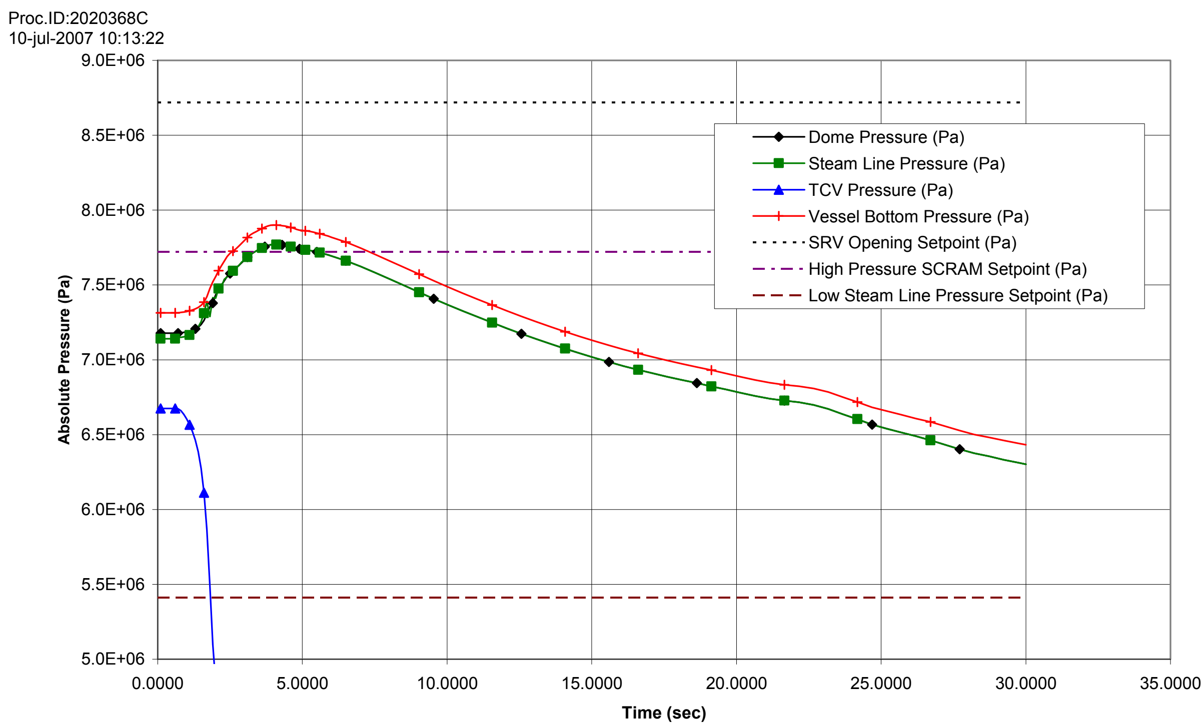



**Figure 2.3-9d. MSIV Closure**

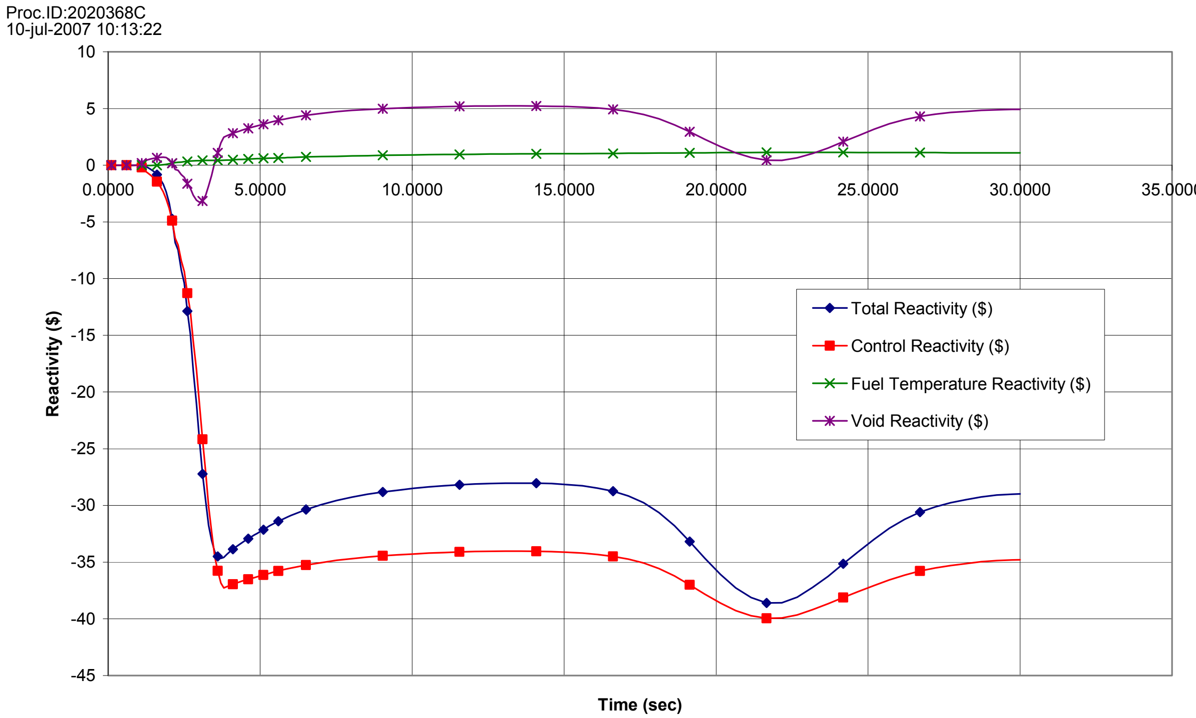



**Figure 2.3-9e. MSIV Closure**

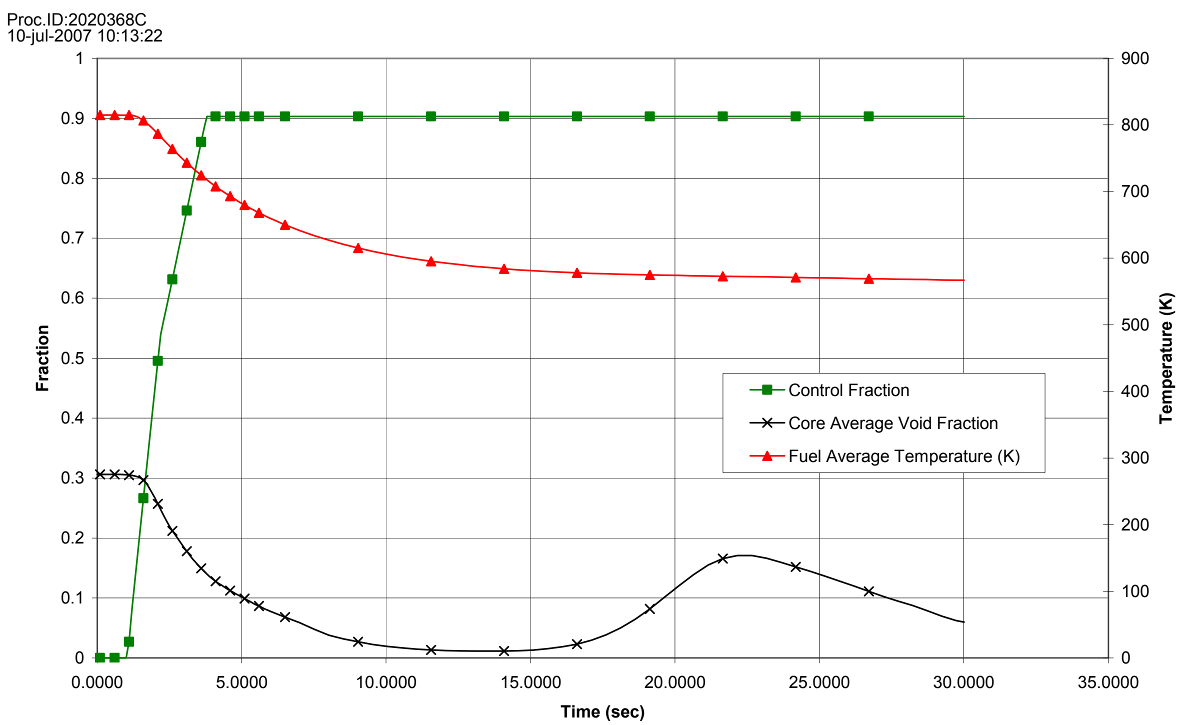



**Figure 2.3-9f. MSIV Closure**

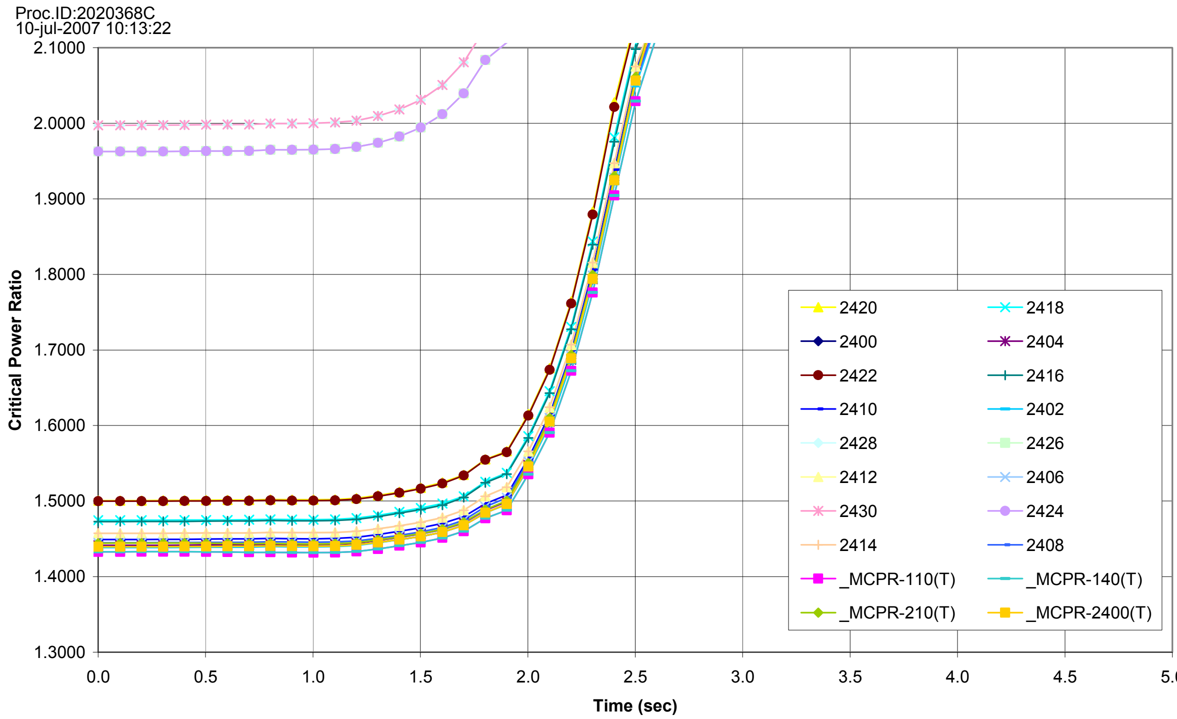



**Figure 2.3-9g. MSIV Closure**

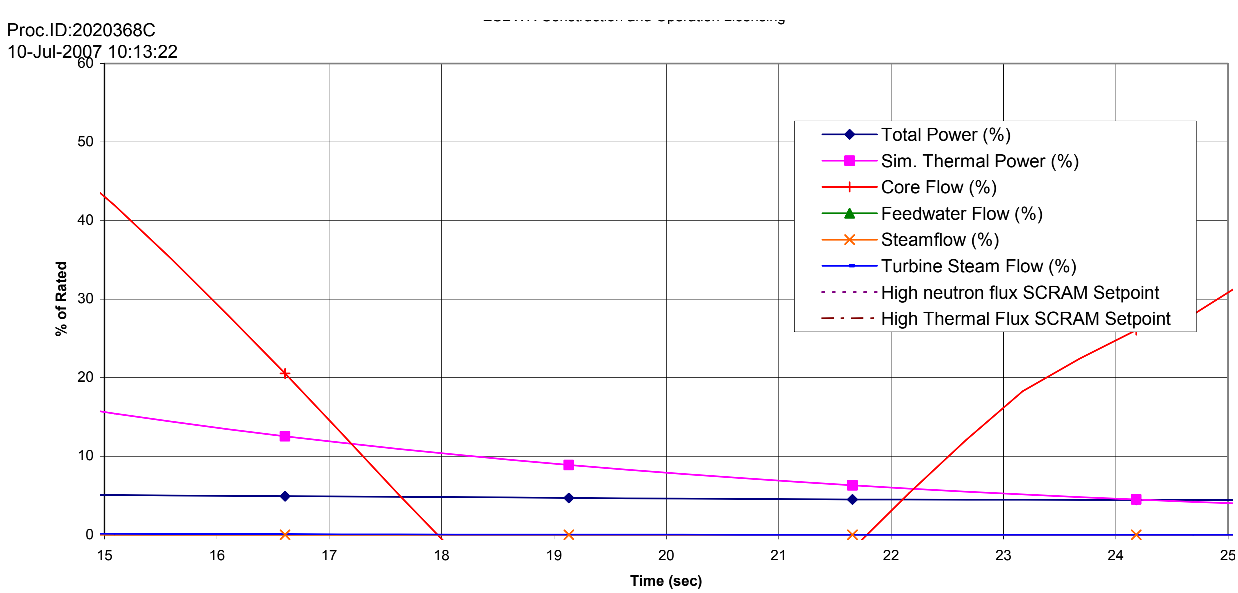



**Figure 2.3-9h. MSIV Closure**
**(Figure 2.3-9a from 15 to 25s)**

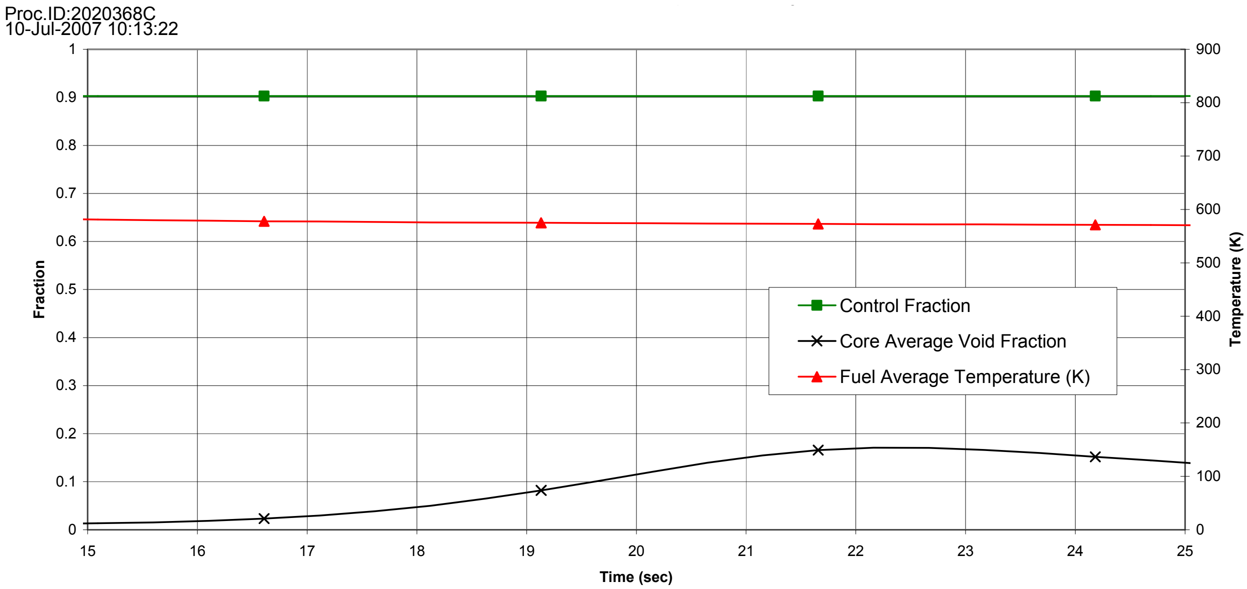



**Figure 2.3-9i. MSIV Closure (Figure 2.3-9f from 15 to 25s)**

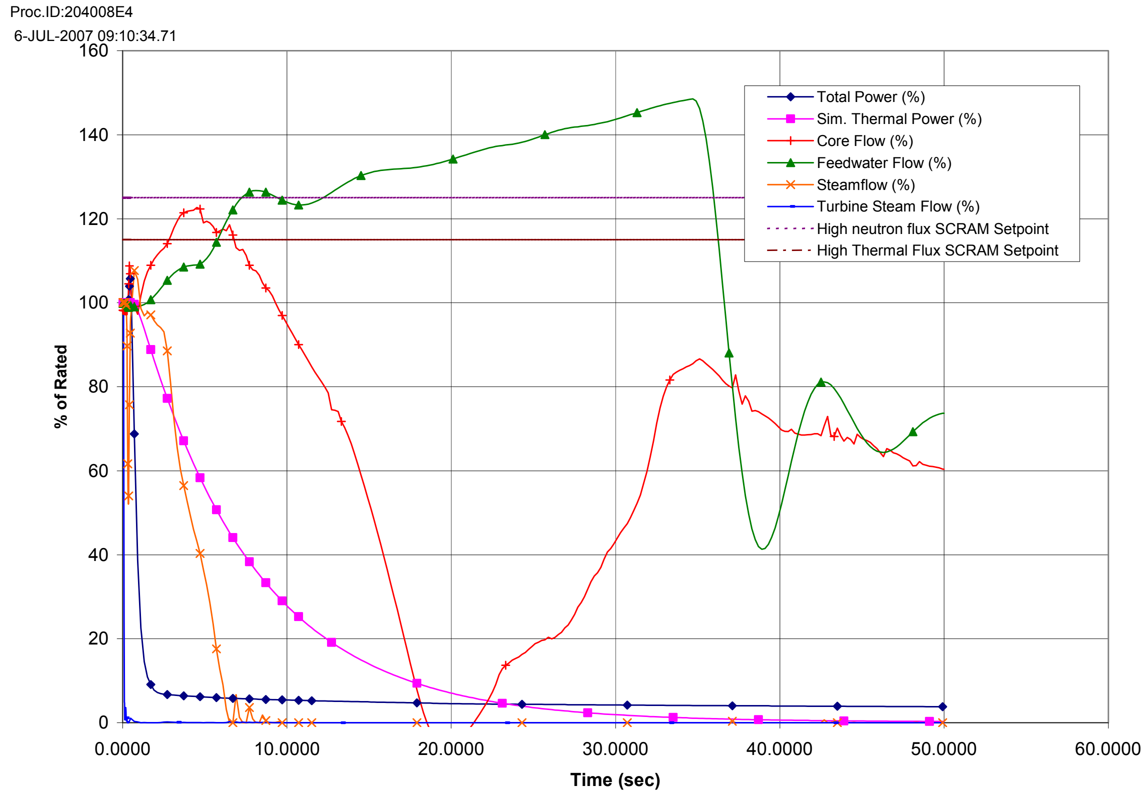



**Figure 2.3-10a. Loss of Condenser Vacuum**

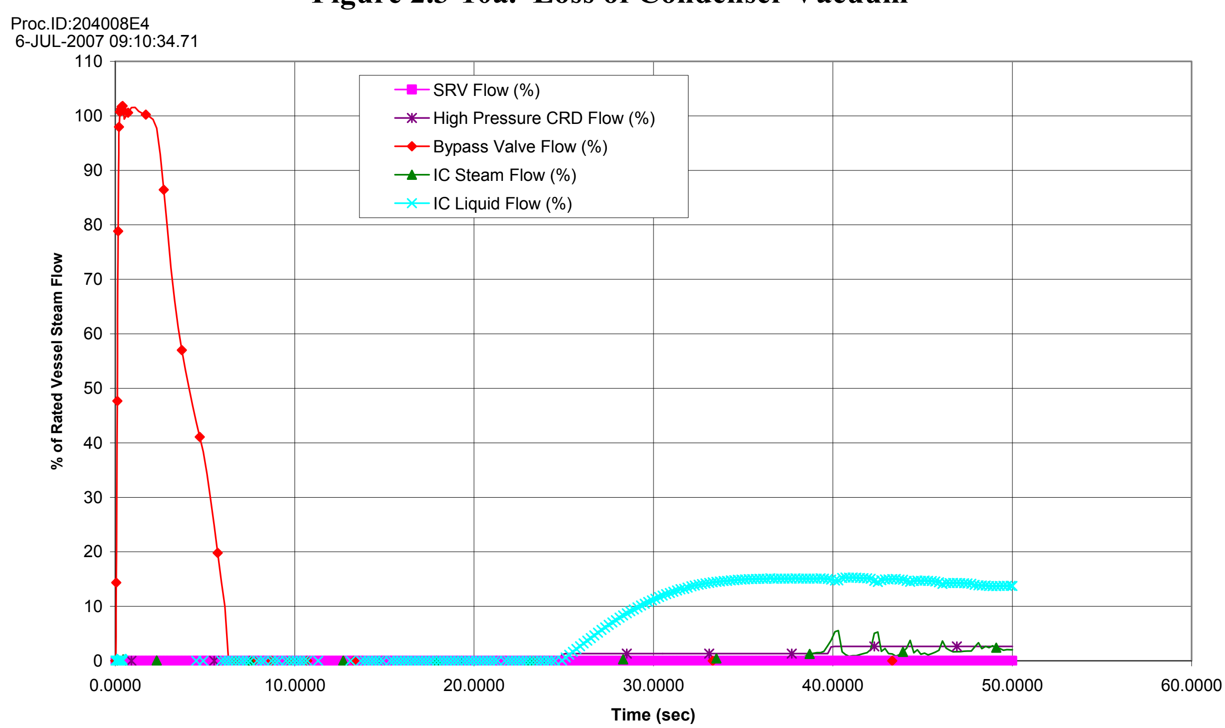



**Figure 2.3-10b. Loss of Condenser Vacuum**

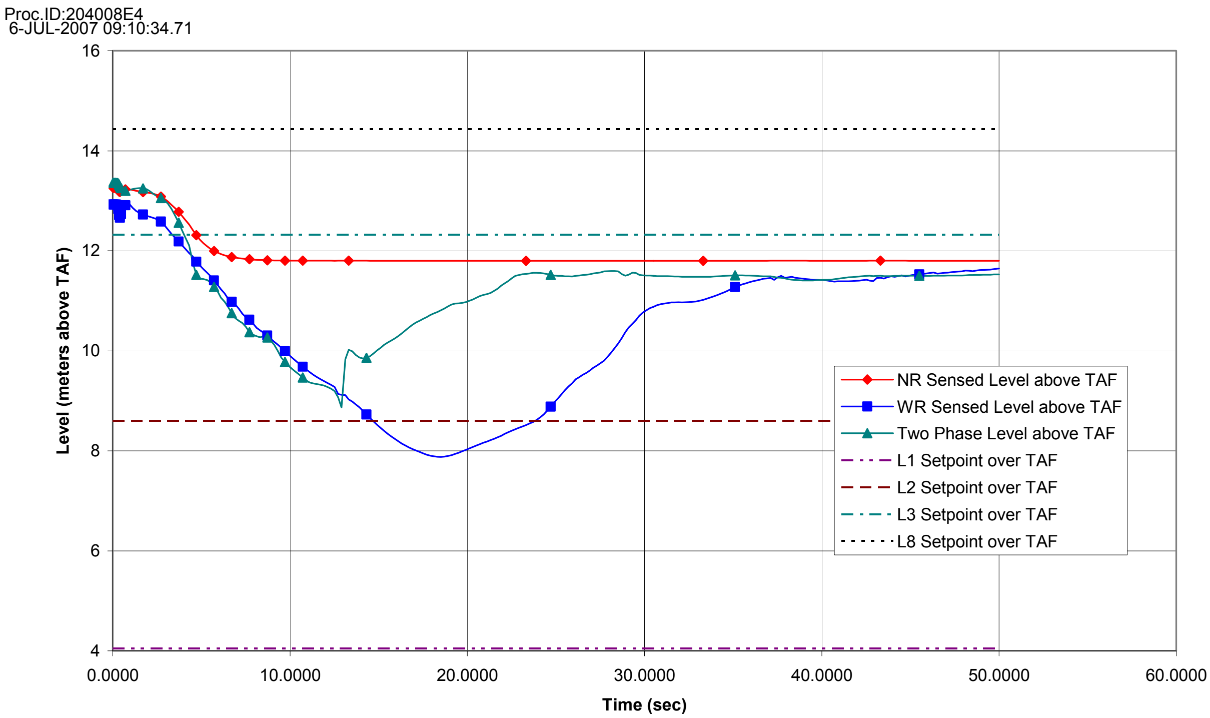



**Figure 2.3-10c. Loss of Condenser Vacuum**

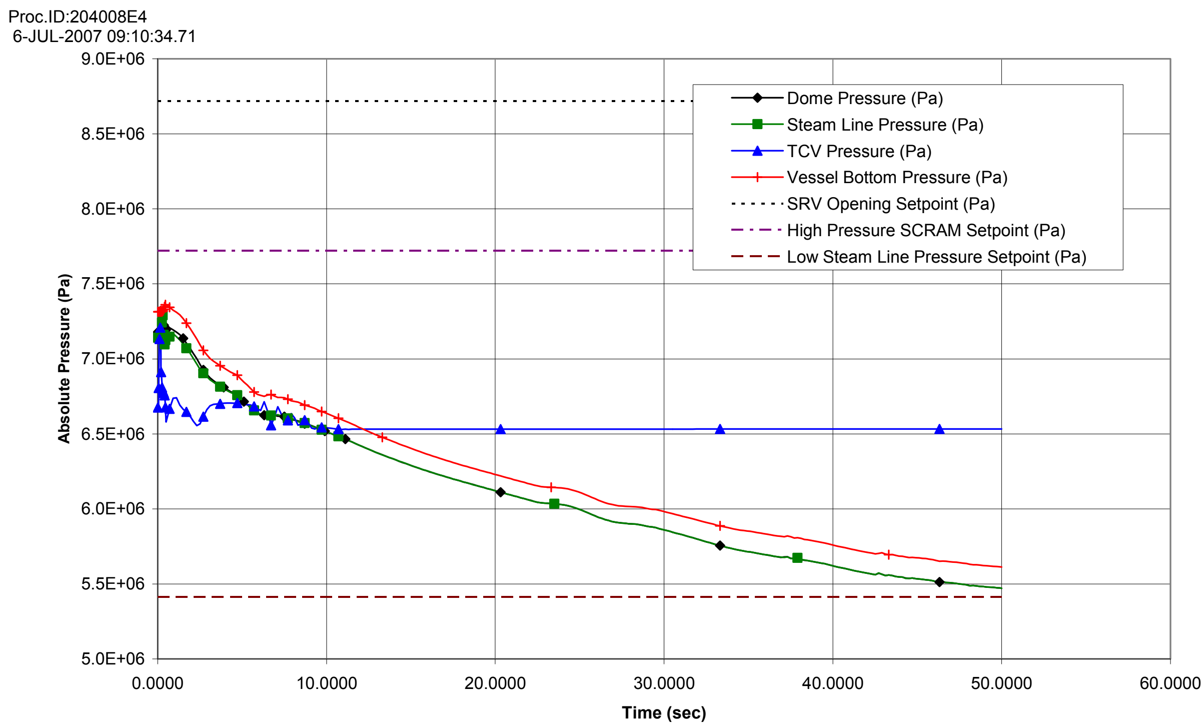



**Figure 2.3-10d. Loss of Condenser Vacuum**

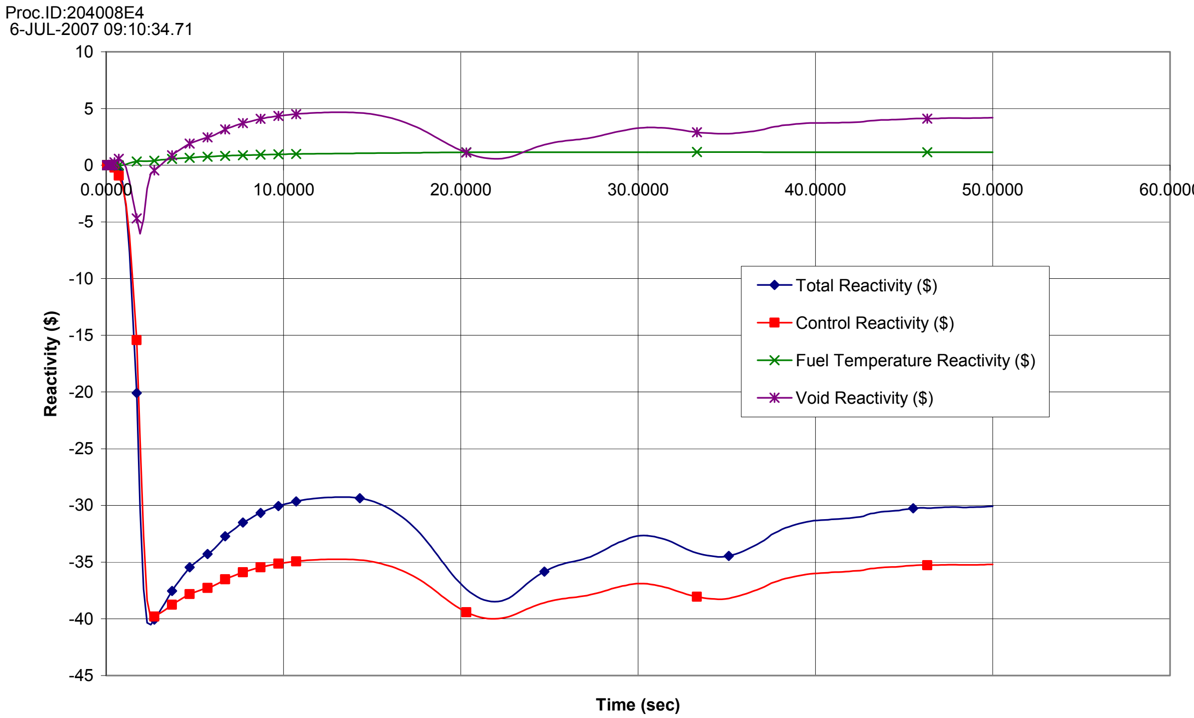



**Figure 2.3-10e. Loss of Condenser Vacuum**

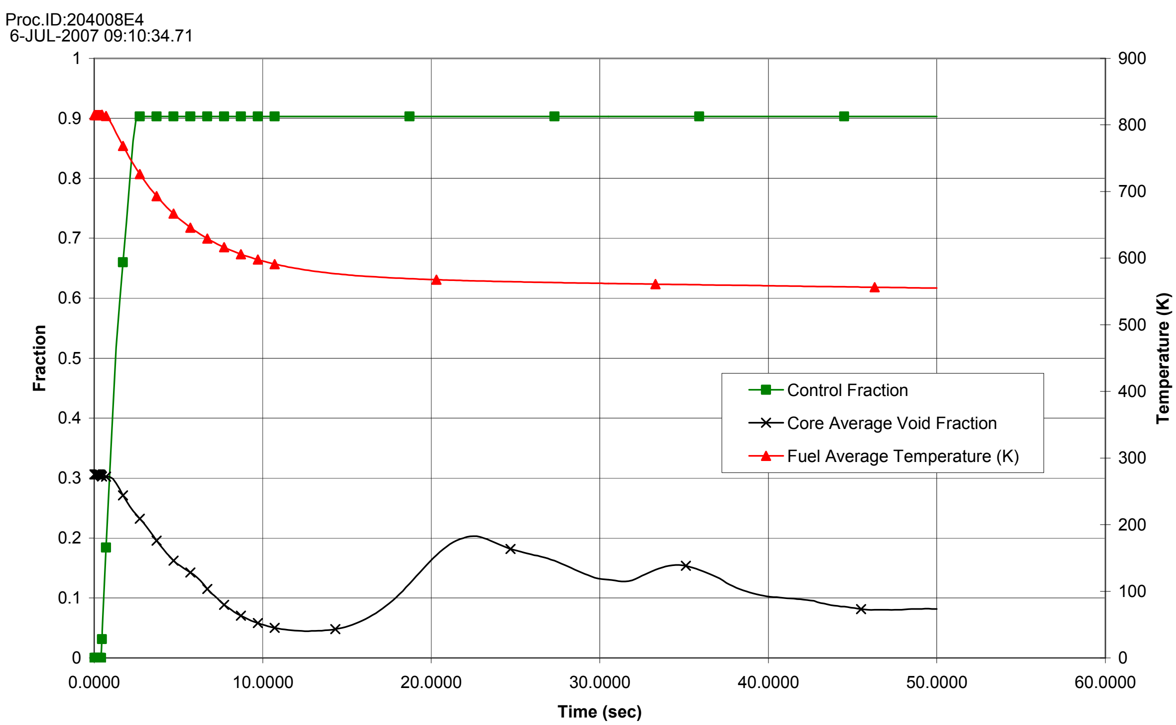



**Figure 2.3-10f. Loss of Condenser Vacuum**

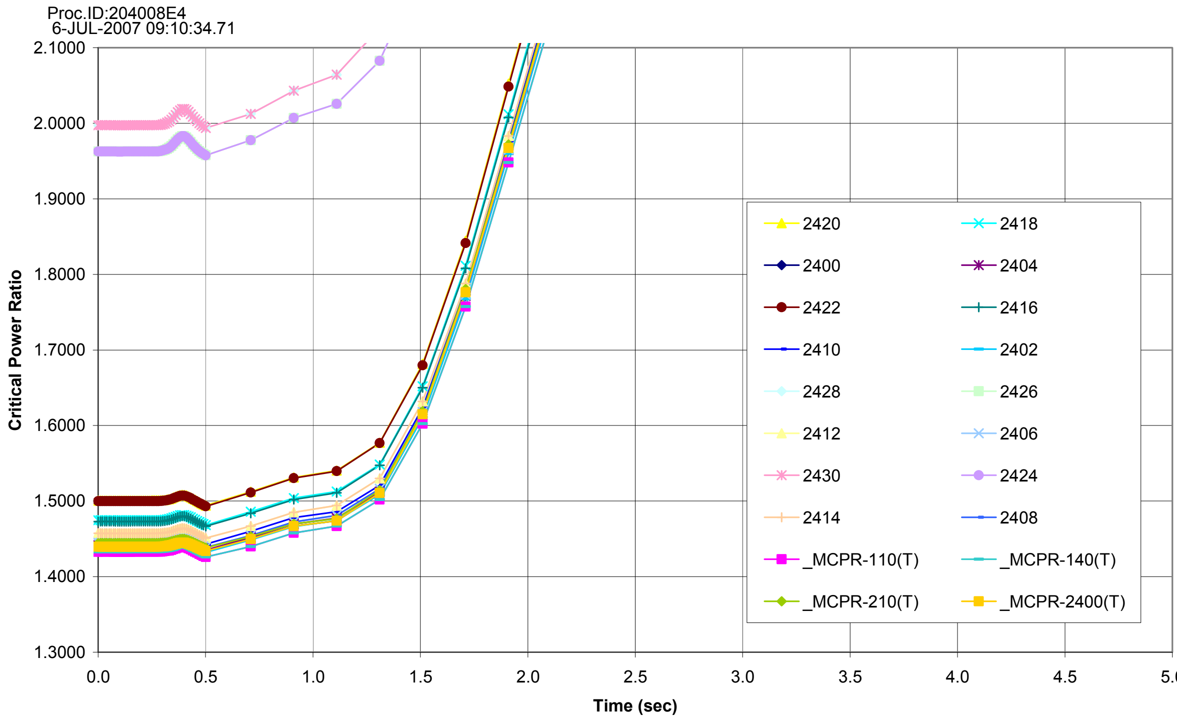



Figure 2.3-10g. Loss of Condenser Vacuum

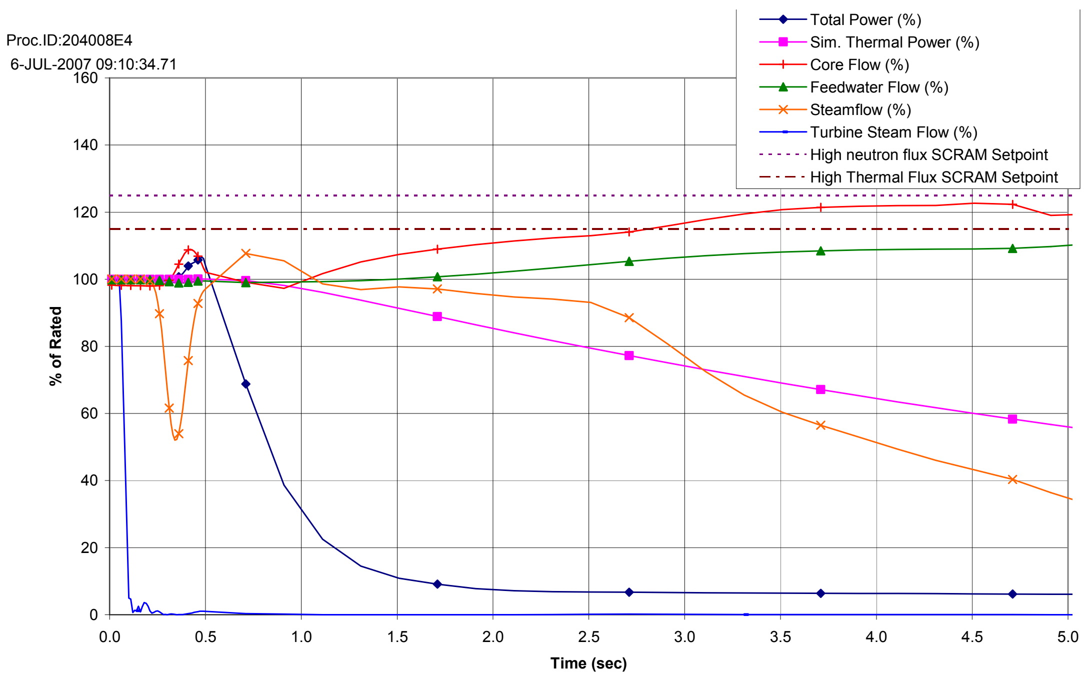



Figure 2.3-10h. Loss of Condenser Vacuum (Figure 2.3-10a from 0 to 5s)

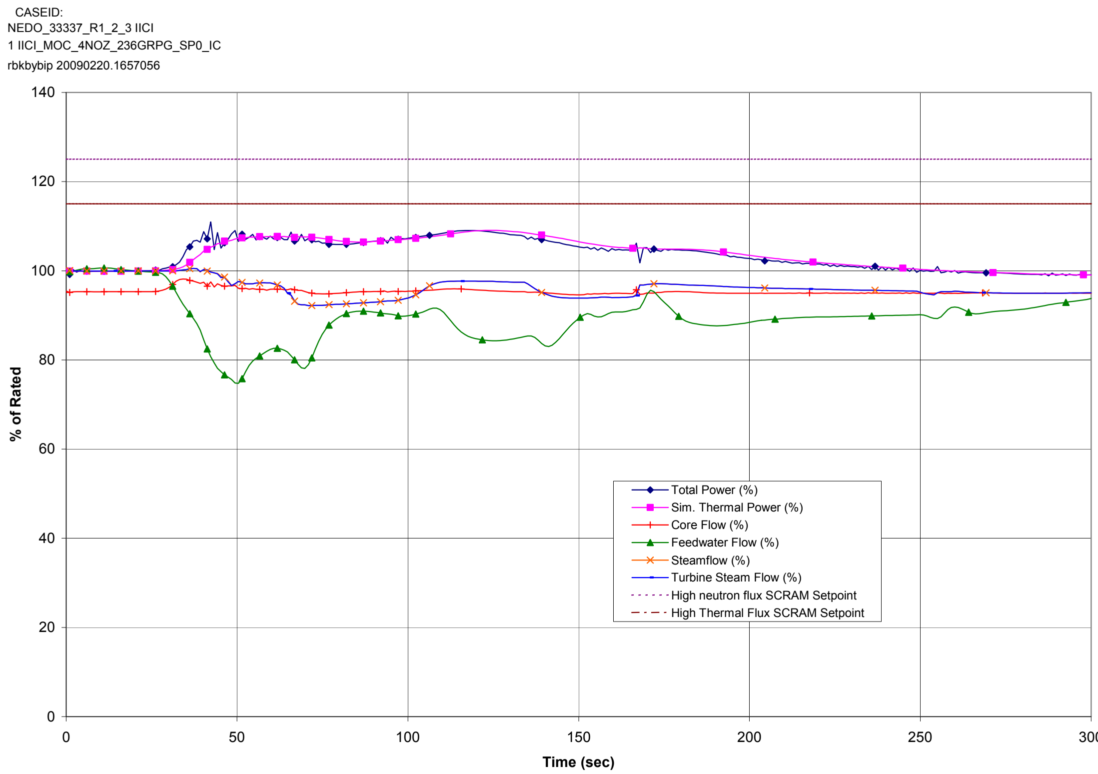



**Figure 2.3-11a. Inadvertent Isolation Condenser Initiation**

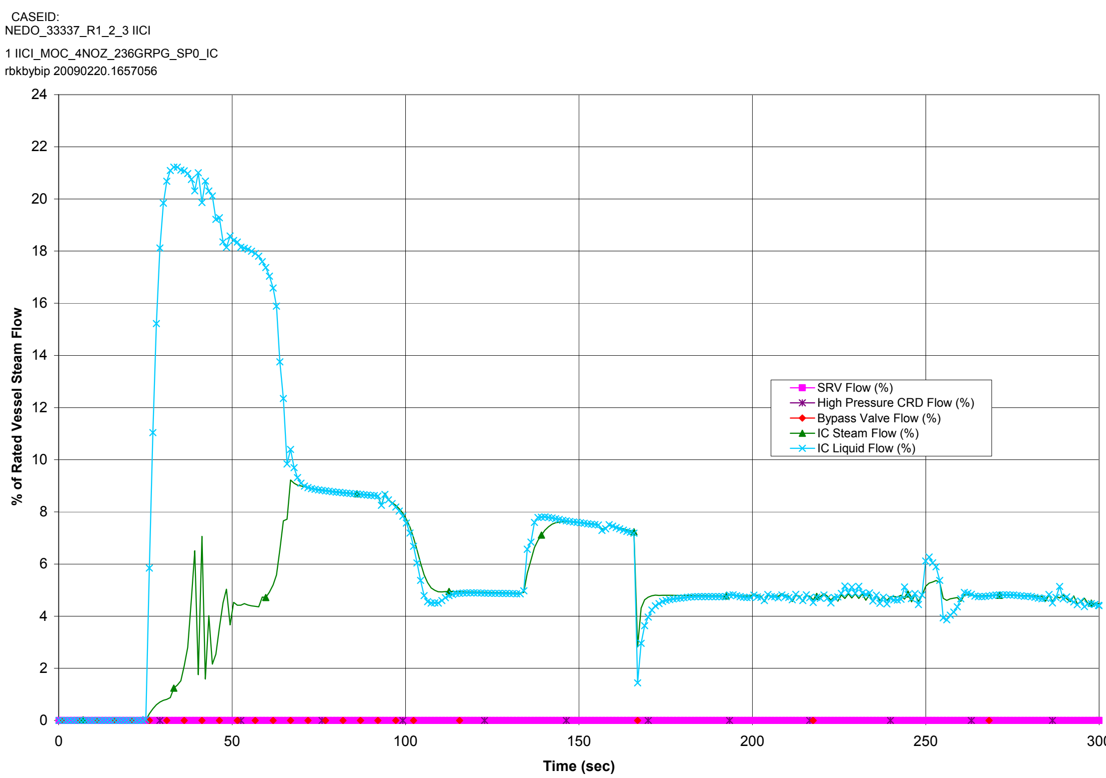



**Figure 2.3-11b. Inadvertent Isolation Condenser Initiation**

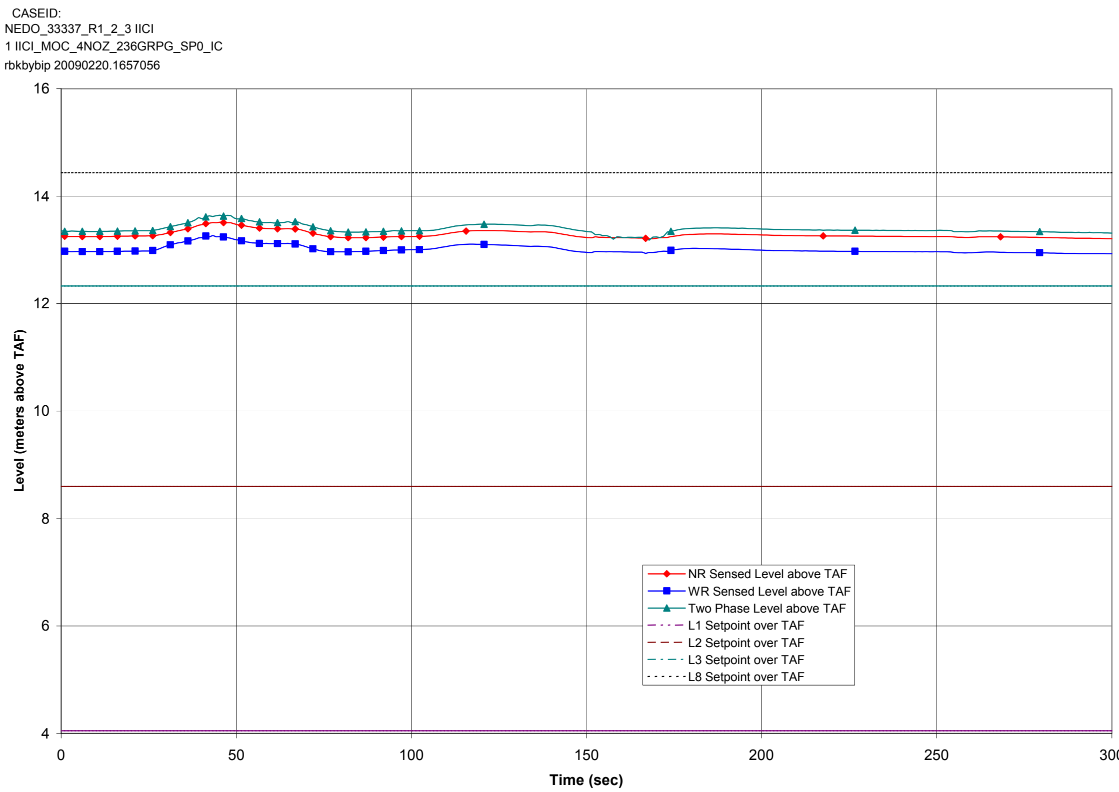



**Figure 2.3-11c. Inadvertent Isolation Condenser Initiation**

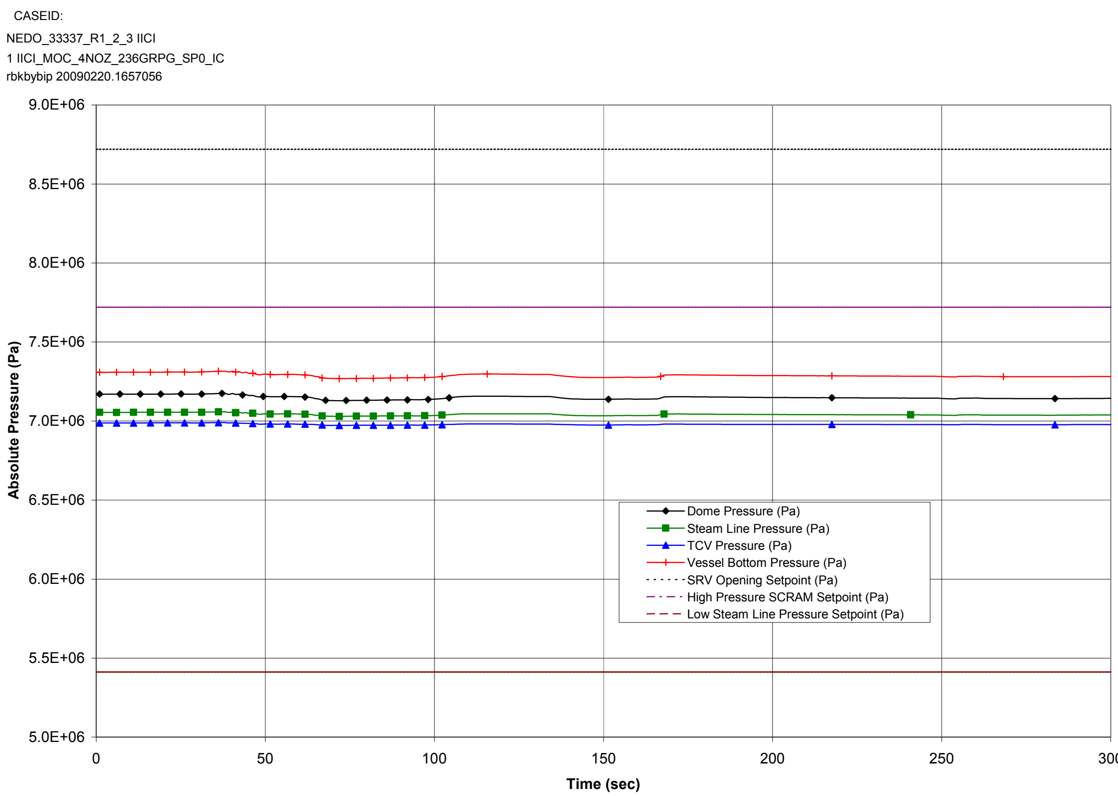



**Figure 2.3-11d. Inadvertent Isolation Condenser Initiation**

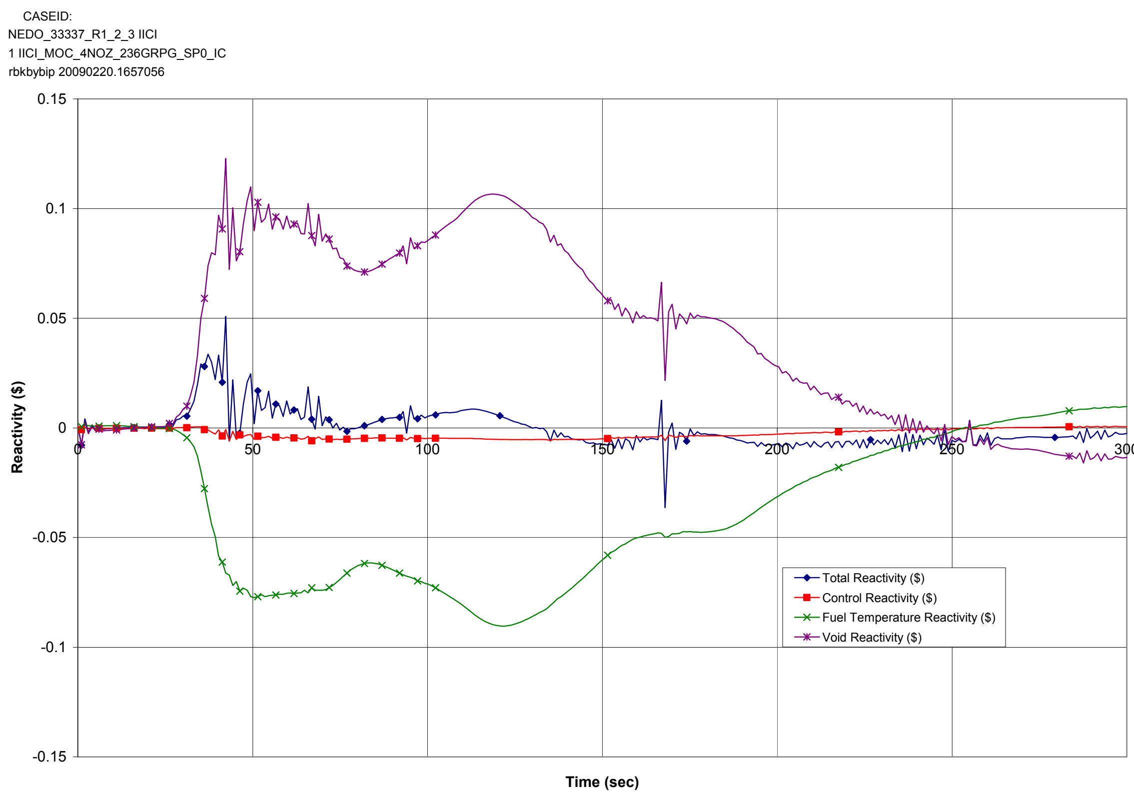



**Figure 2.3-11e. Inadvertent Isolation Condenser Initiation**

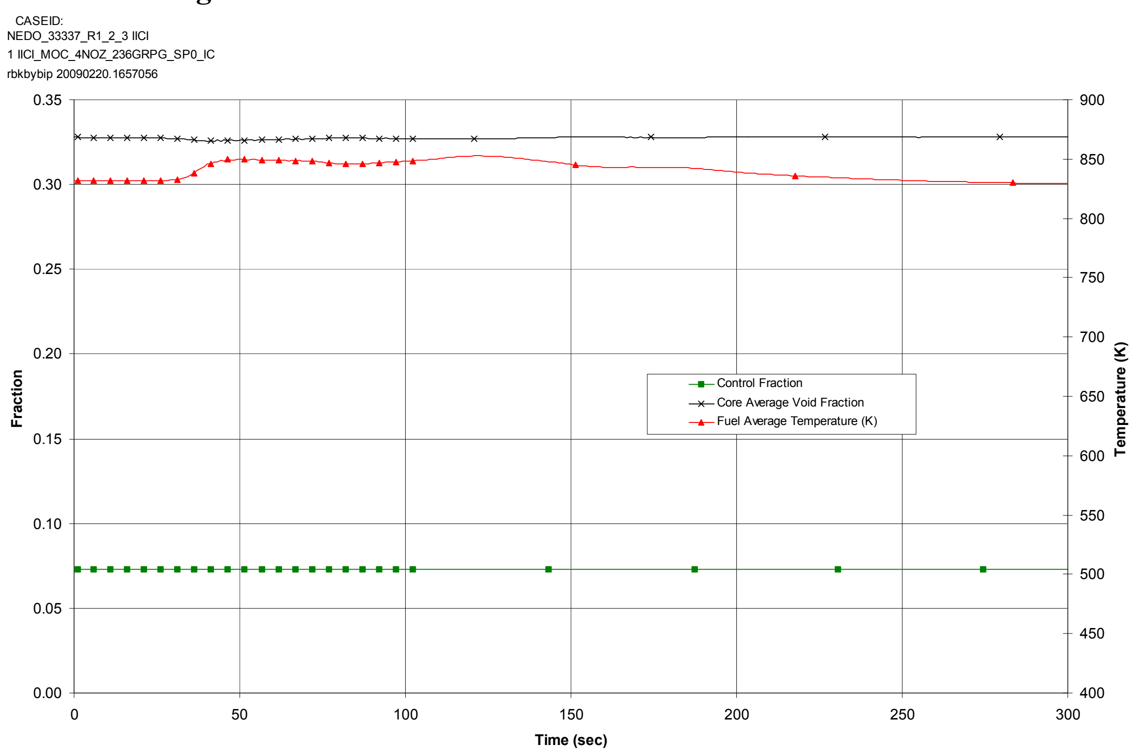



**Figure 2.3-11f. Inadvertent Isolation Condenser Initiation**

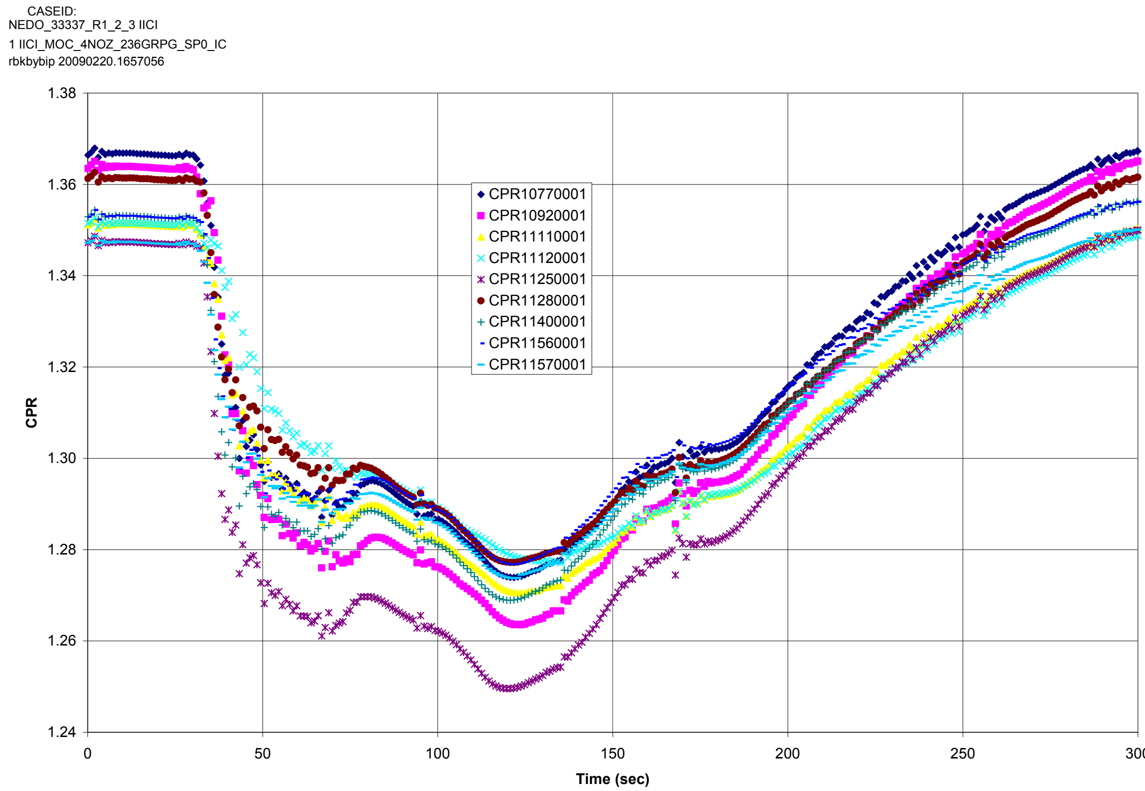



**Figure 2.3-11g. Inadvertent Isolation Condenser Initiation**

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

Data in the DCD applies to the initial core.

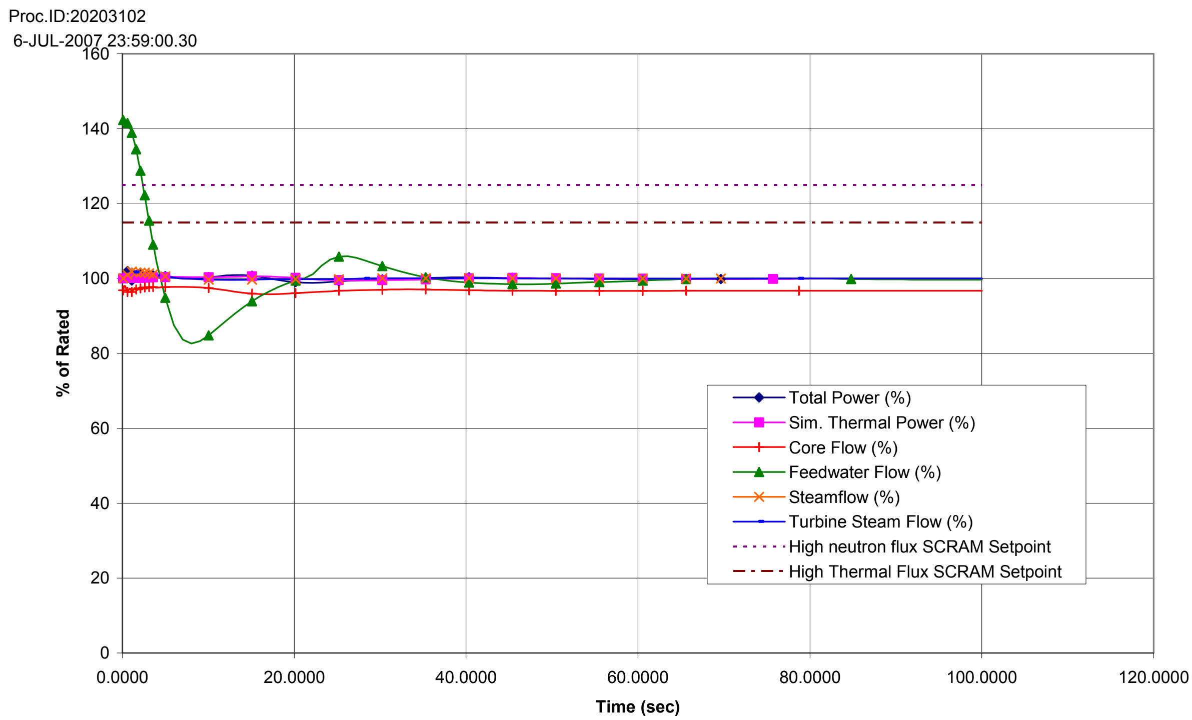



**Figure 2.3-13a. Runout of One Feedwater Pump**

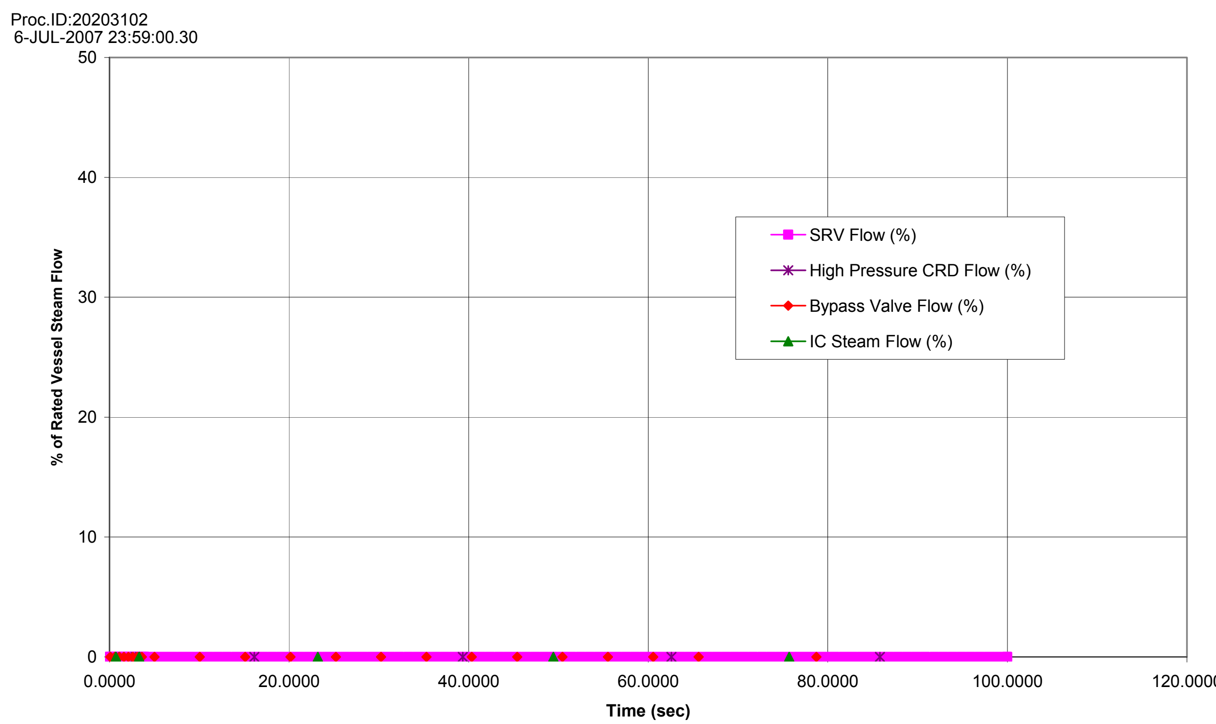



**Figure 2.3-13b. Runout of One Feedwater Pump**

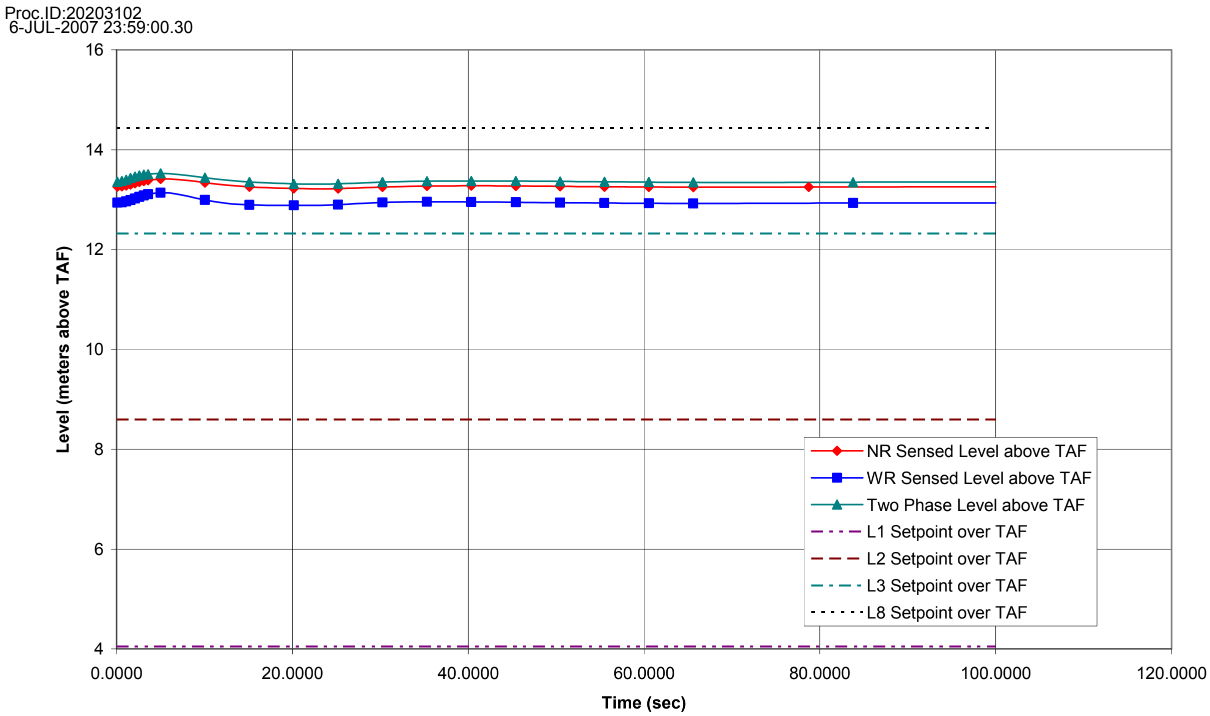



**Figure 2.3-13c. Runout of One Feedwater Pump**

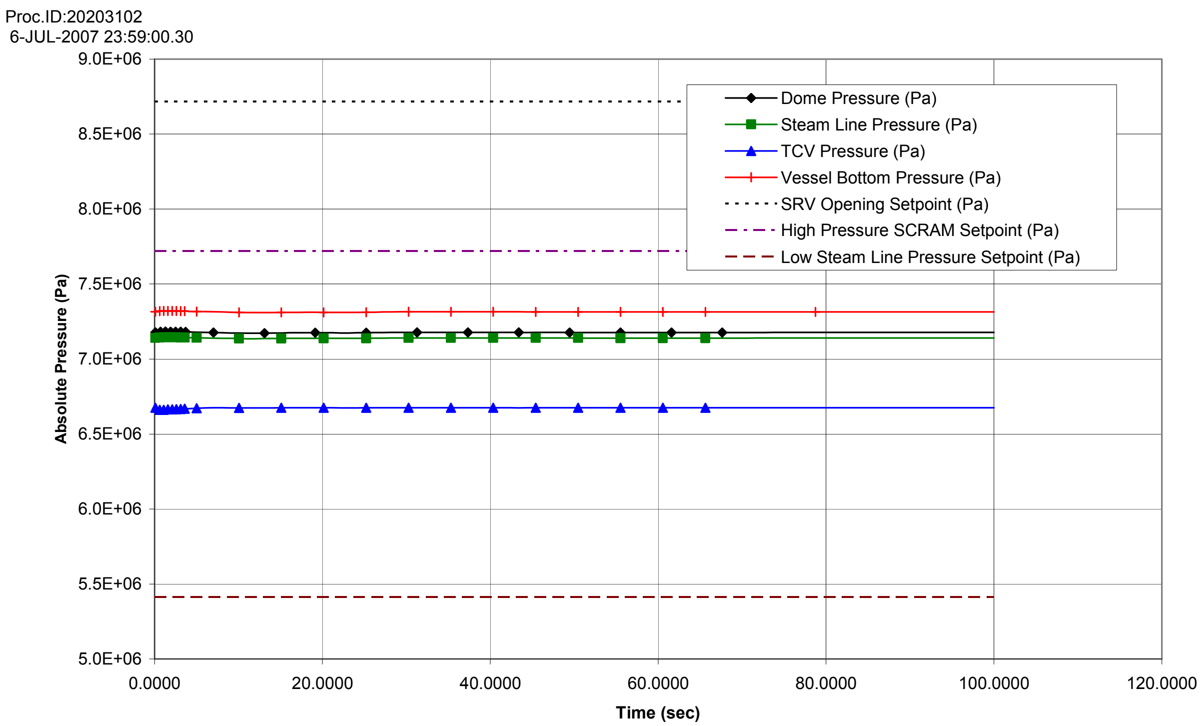



**Figure 2.3-13d. Runout of One Feedwater Pump**

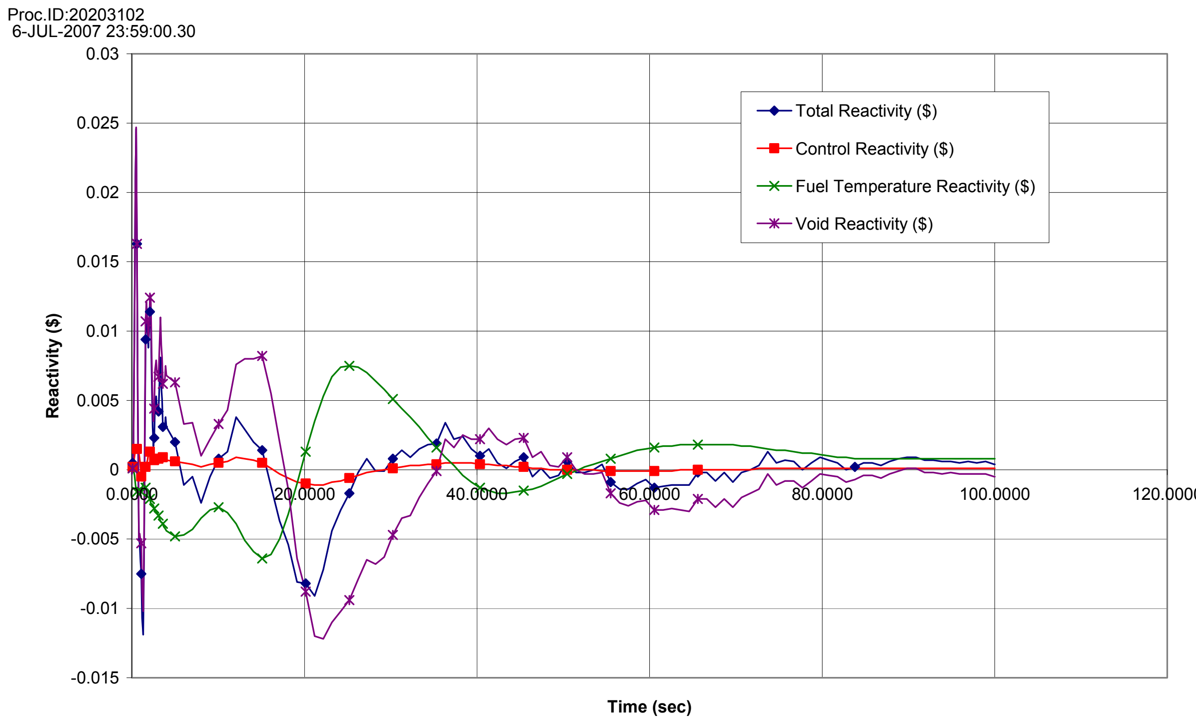



**Figure 2.3-13e. Runout of One Feedwater Pump**

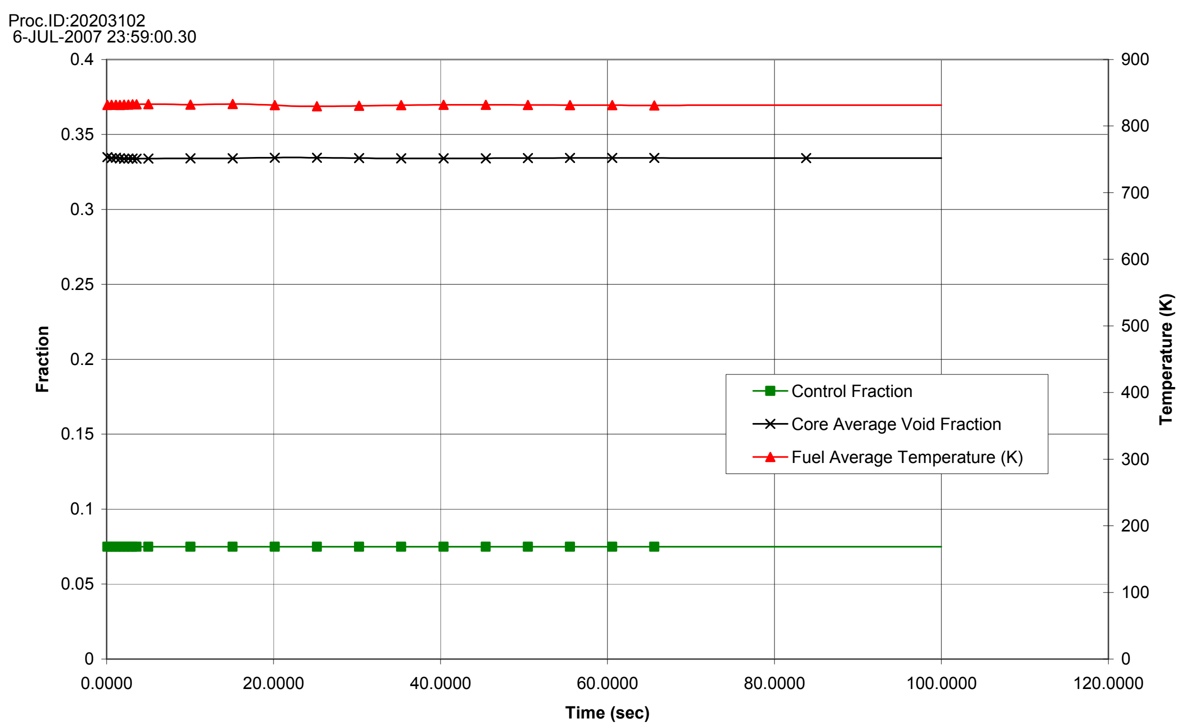



**Figure 2.3-13f. Runout of One Feedwater Pump**

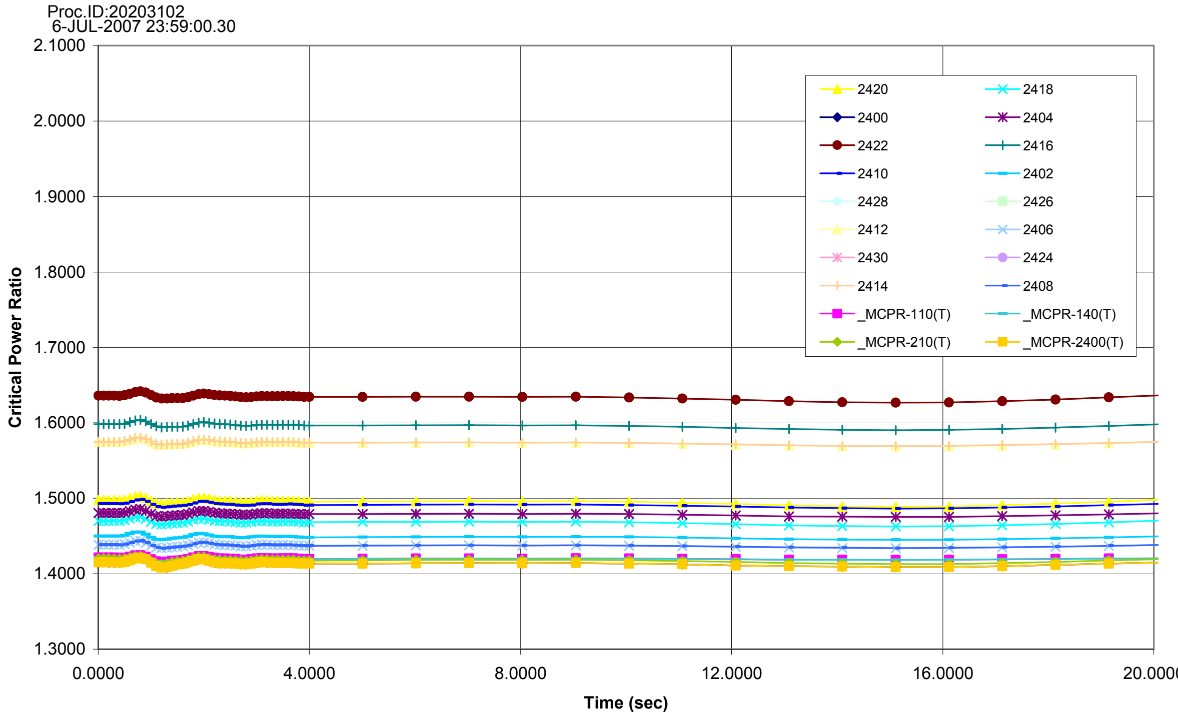



**Figure 2.3-13g. Runout of One Feedwater Pump**

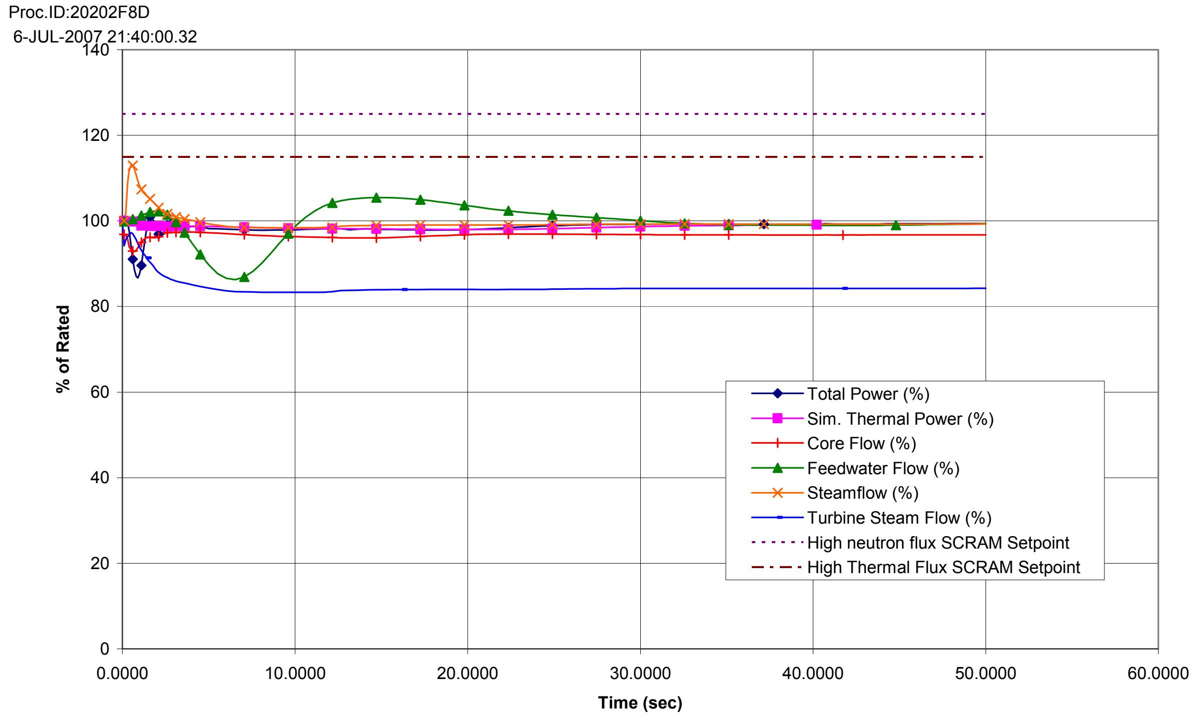



**Figure 2.3-14a. Opening of One Turbine Control or Bypass Valve**

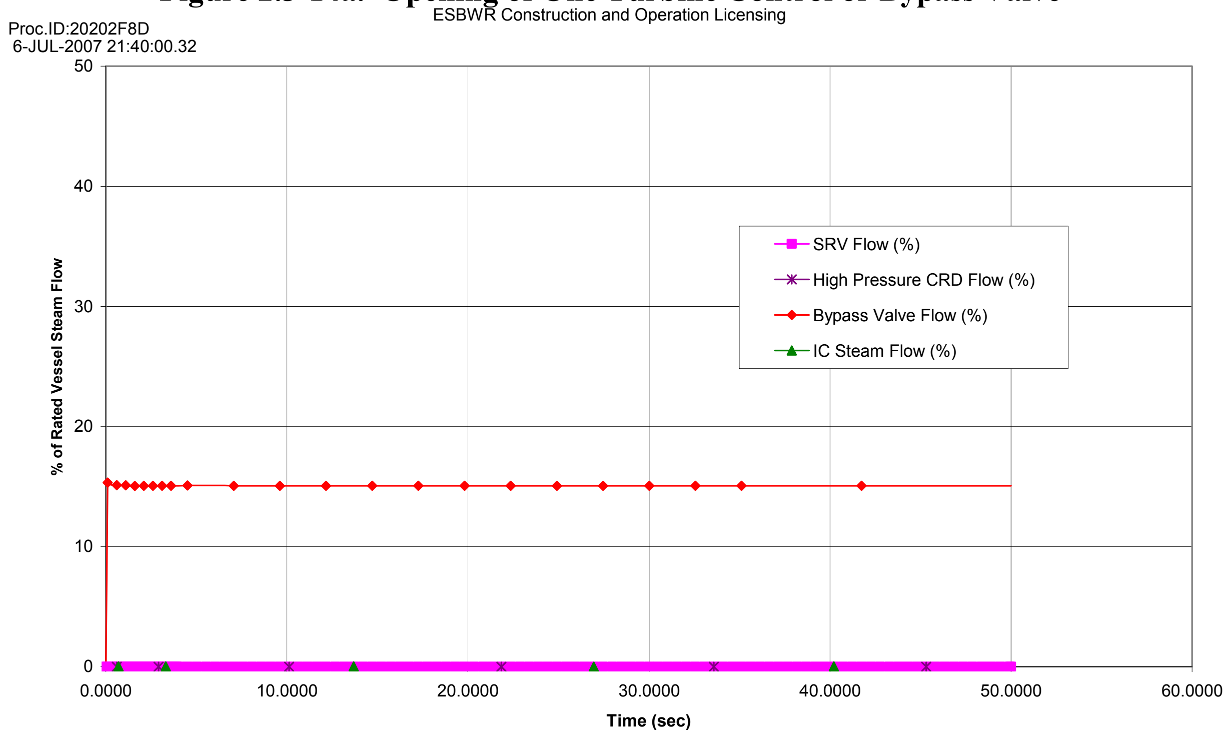



**Figure 2.3-14b. Opening of One Turbine Control or Bypass Valve**

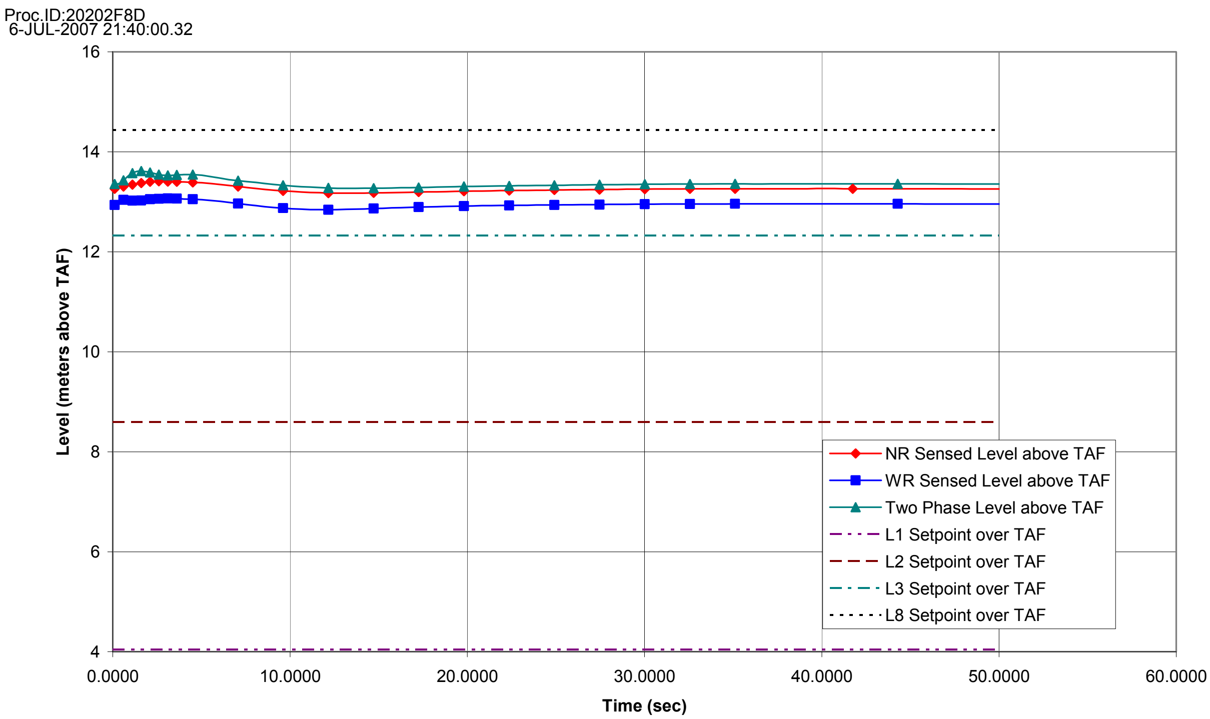



**Figure 2.3-14c. Opening of One Turbine Control or Bypass Valve**

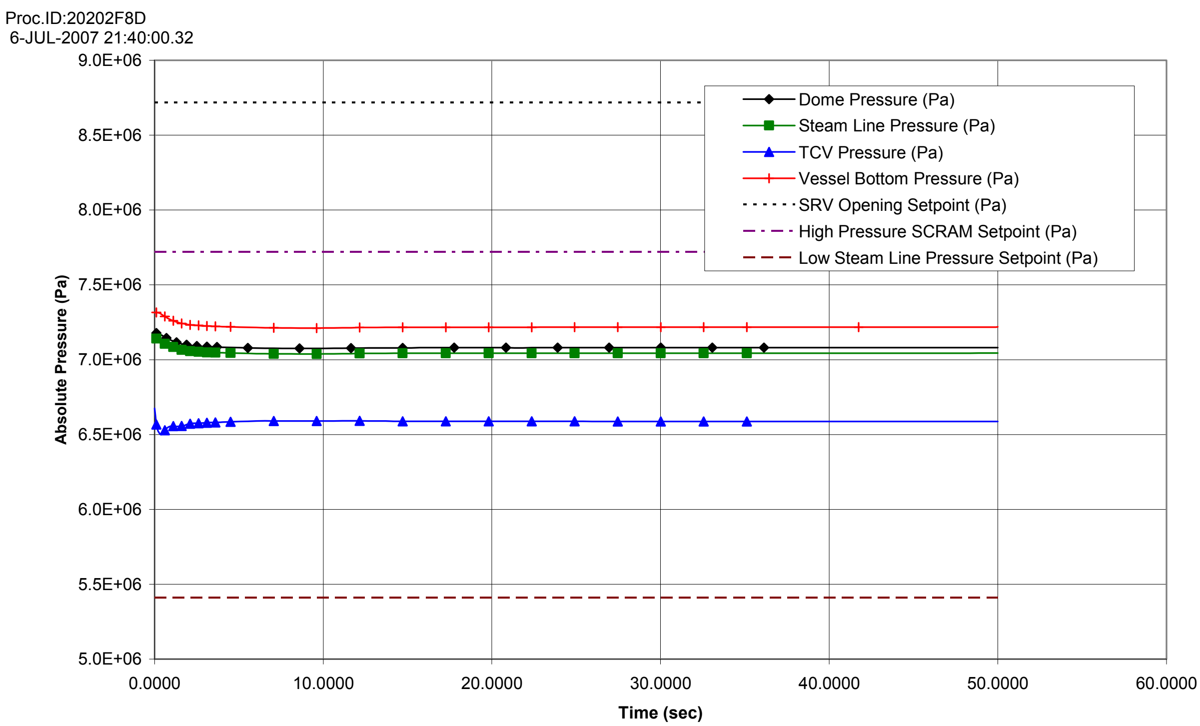



**Figure 2.3-14d. Opening of One Turbine Control or Bypass Valve**

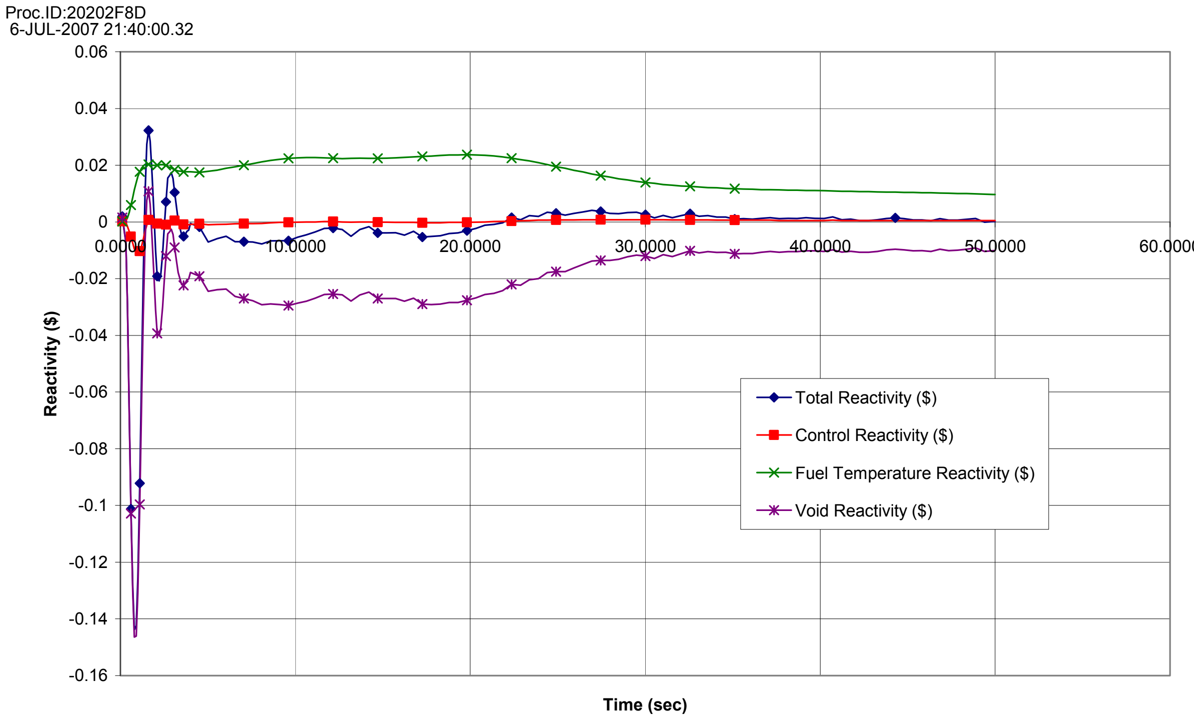



Figure 2.3-14e. Opening of One Turbine Control or Bypass Valve

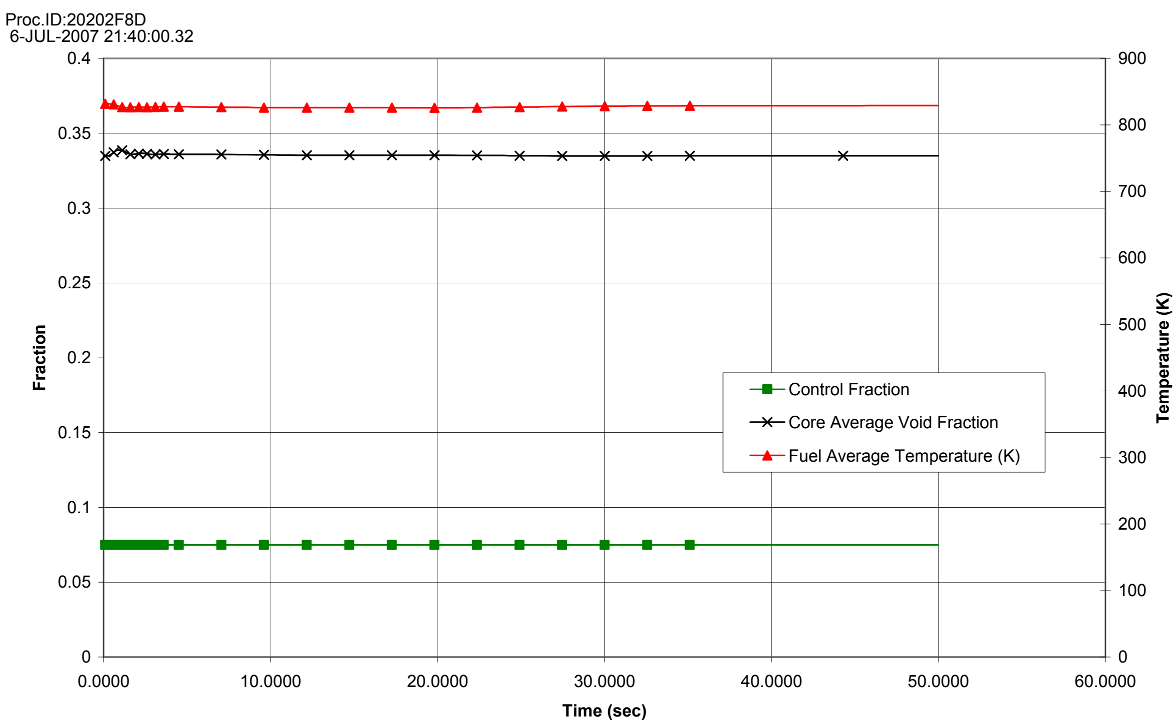



Figure 2.3-14f. Opening of One Turbine Control or Bypass Valve

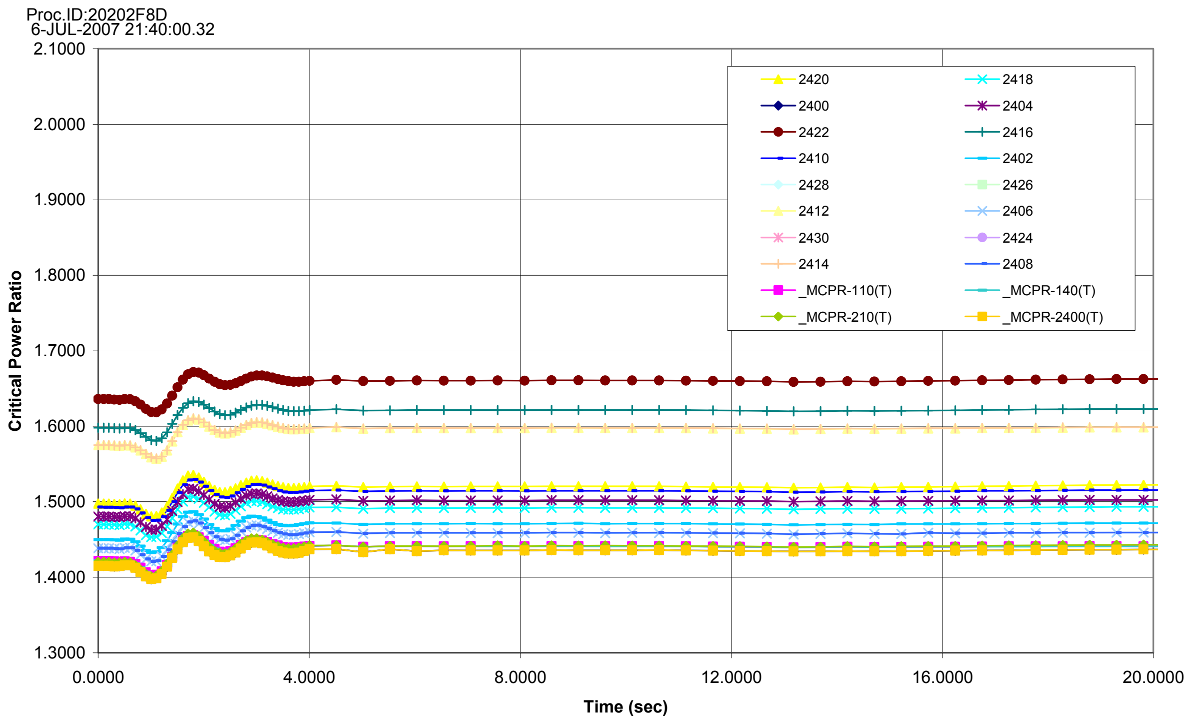



**Figure 2.3-14g. Opening of One Turbine Control or Bypass Valve**

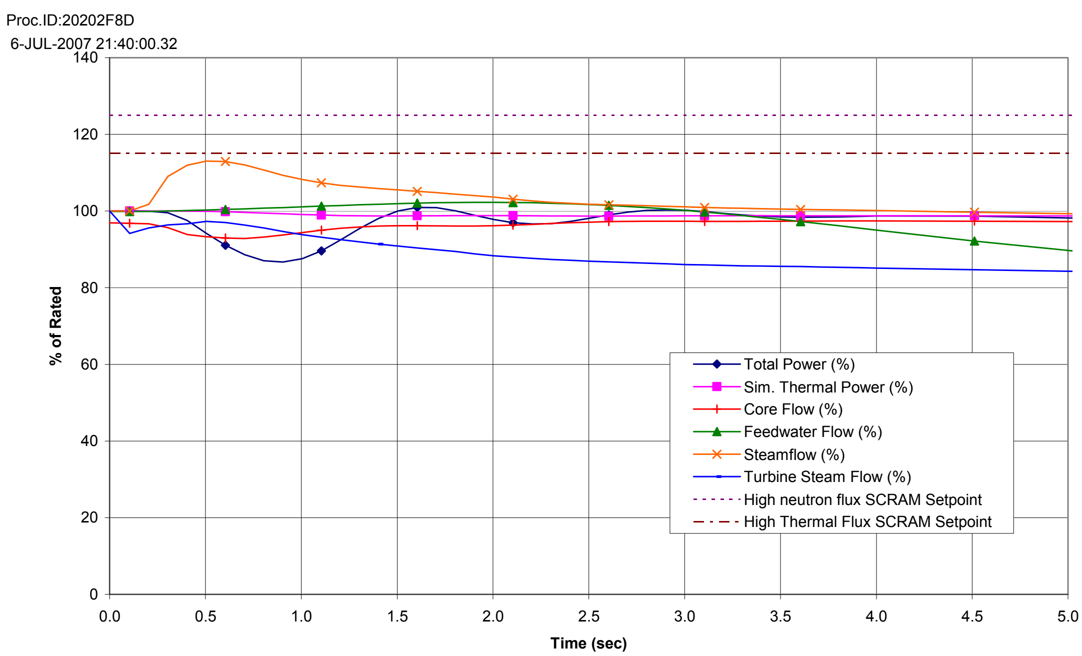



**Figure 2.3-14h. Opening of One Turbine Control or Bypass Valve (Figure 2.3-14a from 0 to 5s)**

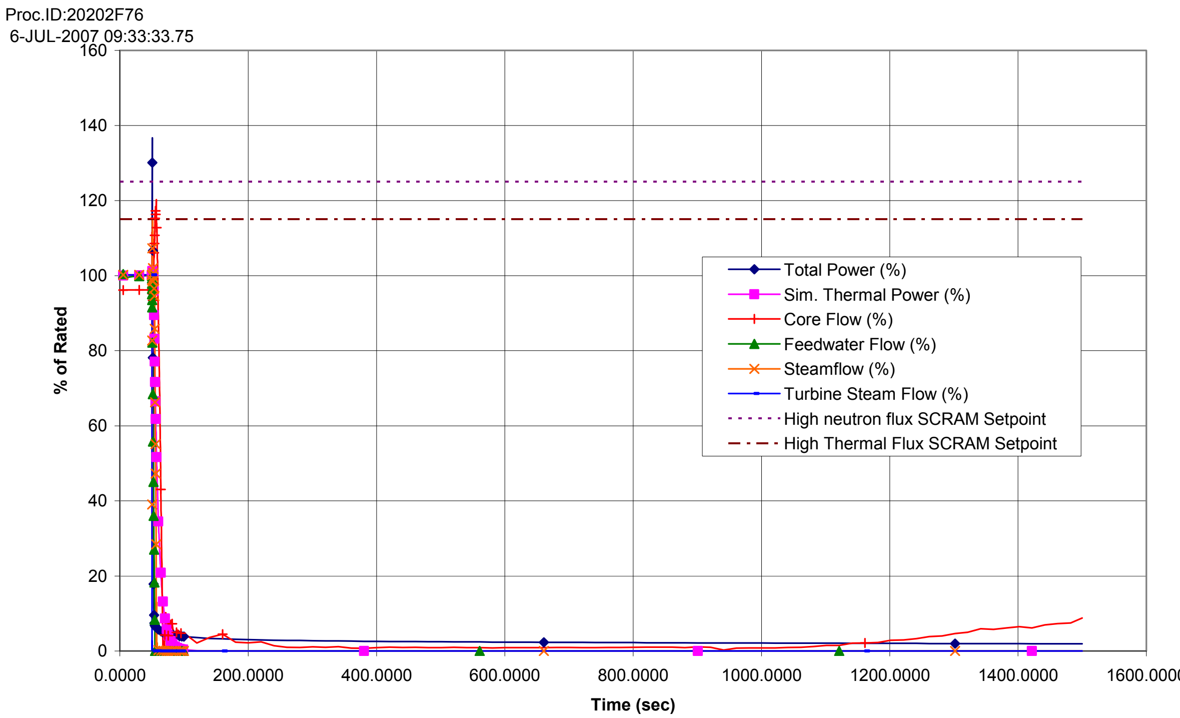



**Figure 2.3-15a. Loss of Non-Emergency AC Power to Station Auxiliaries**

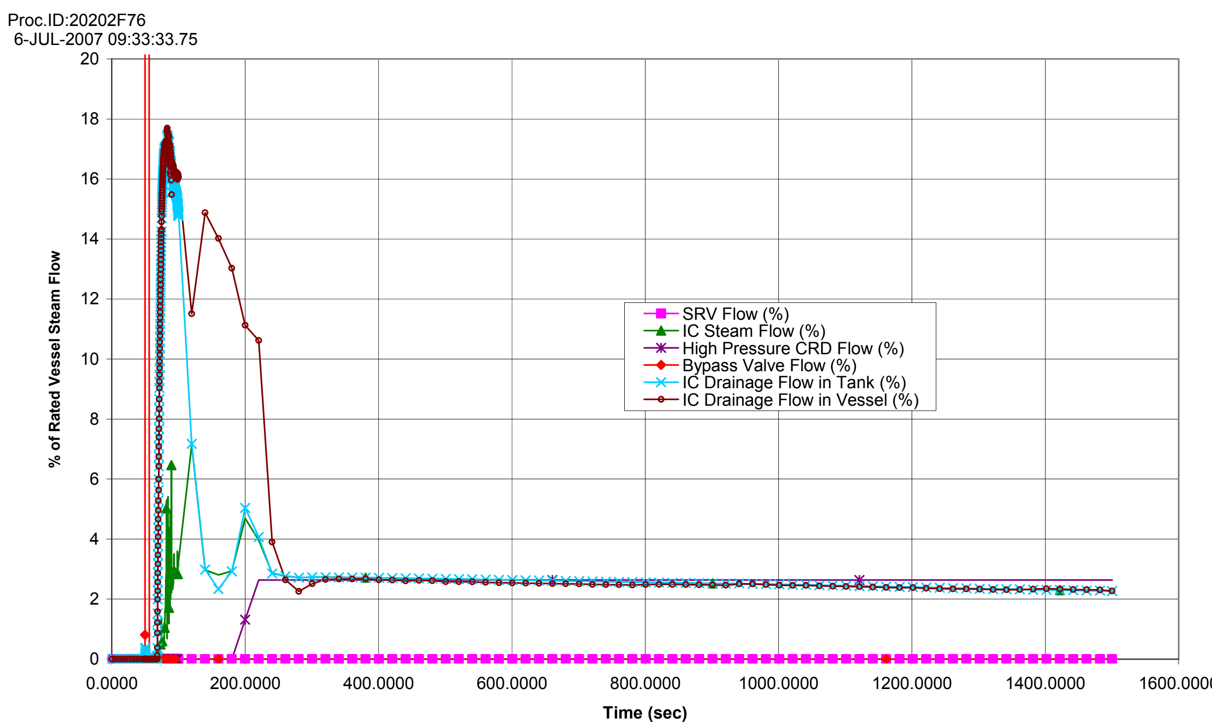



**Figure 2.3-15b. Loss of Non-Emergency AC Power to Station Auxiliaries**

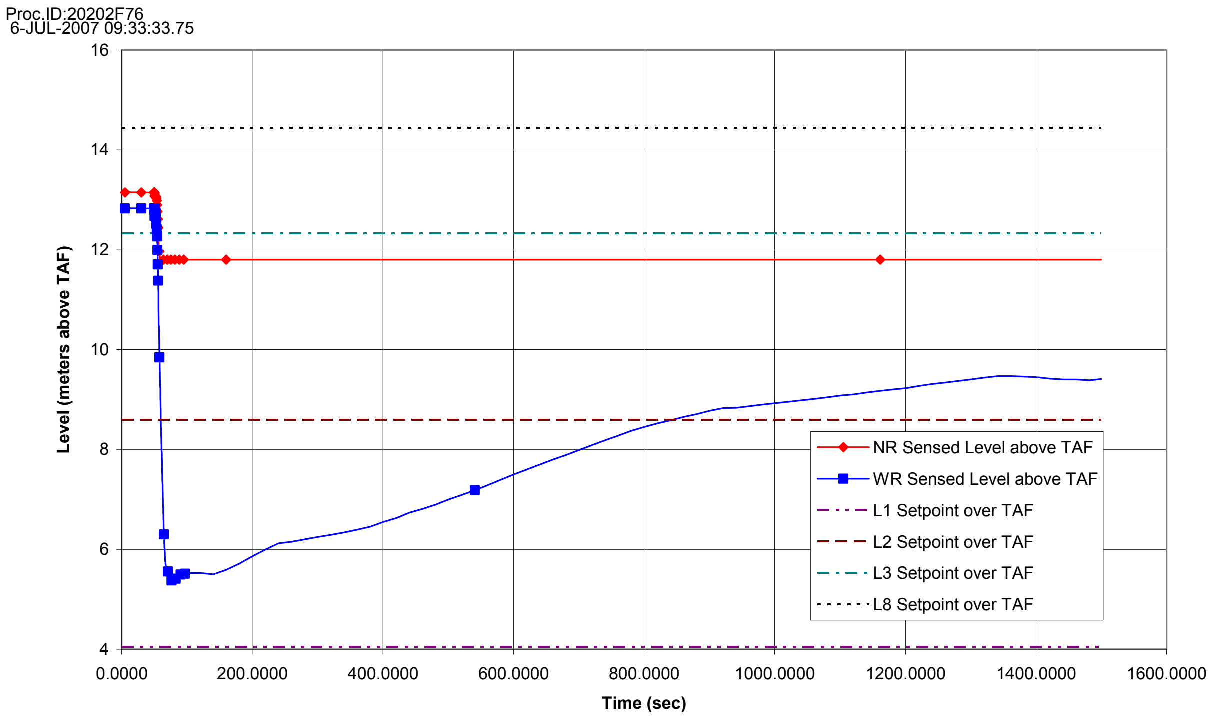



**Figure 2.3-15c. Loss of Non-Emergency AC Power to Station Auxiliaries**

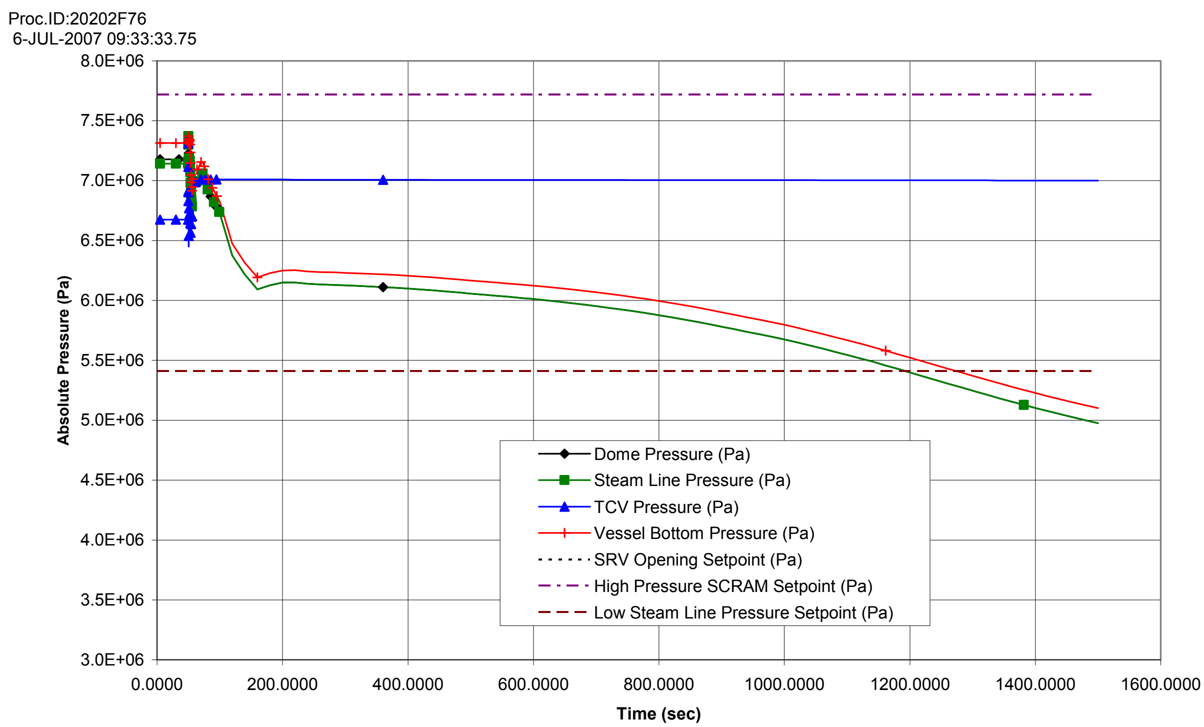



**Figure 2.3-15d. Loss of Non-Emergency AC Power to Station Auxiliaries**

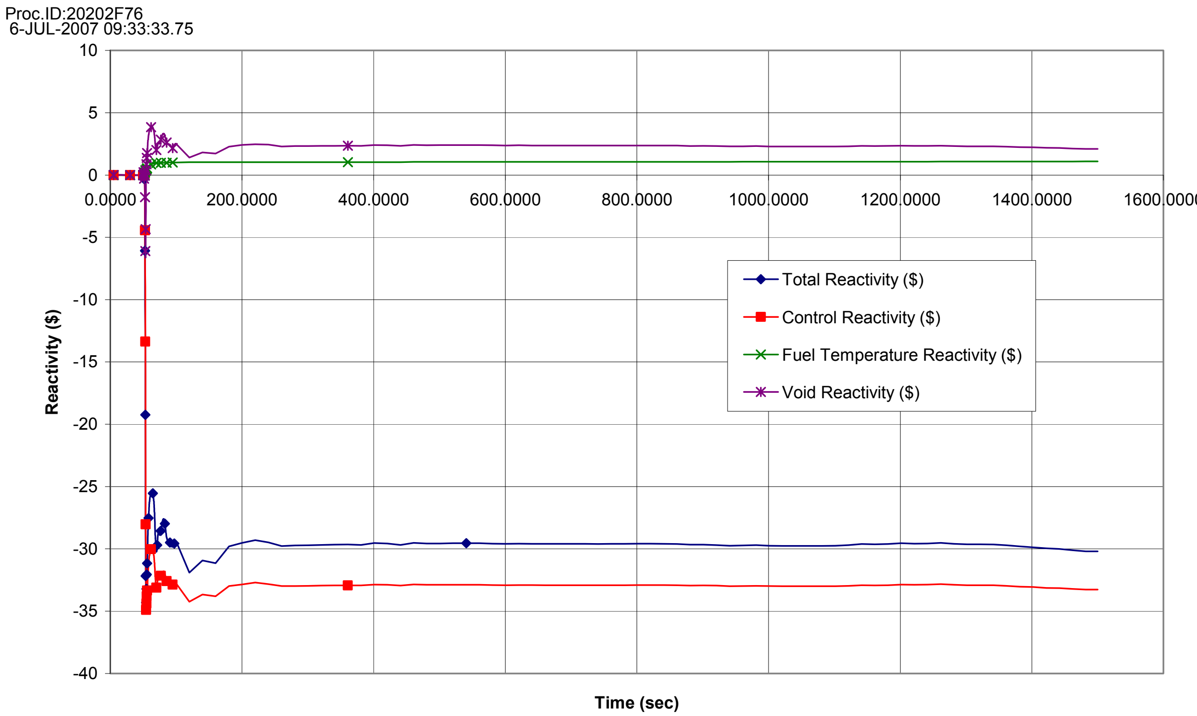



**Figure 2.3-15e. Loss of Non-Emergency AC Power to Station Auxiliaries**

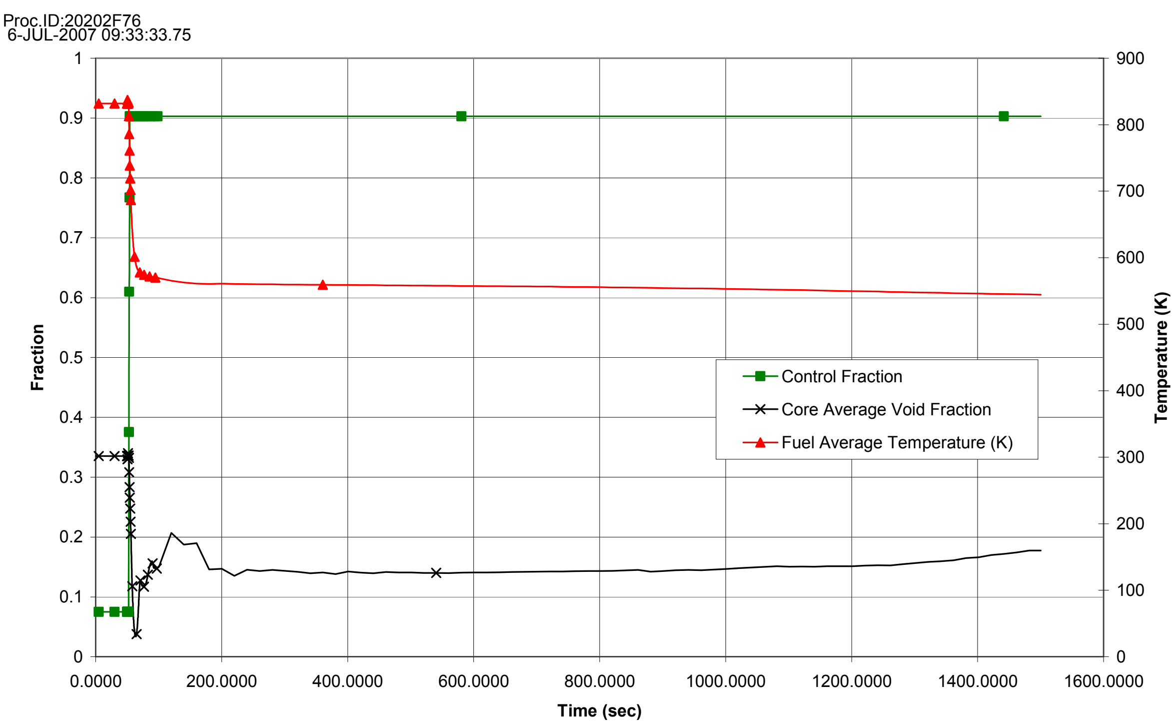



**Figure 2.3-15f. Loss of Non-Emergency AC Power to Station Auxiliaries**

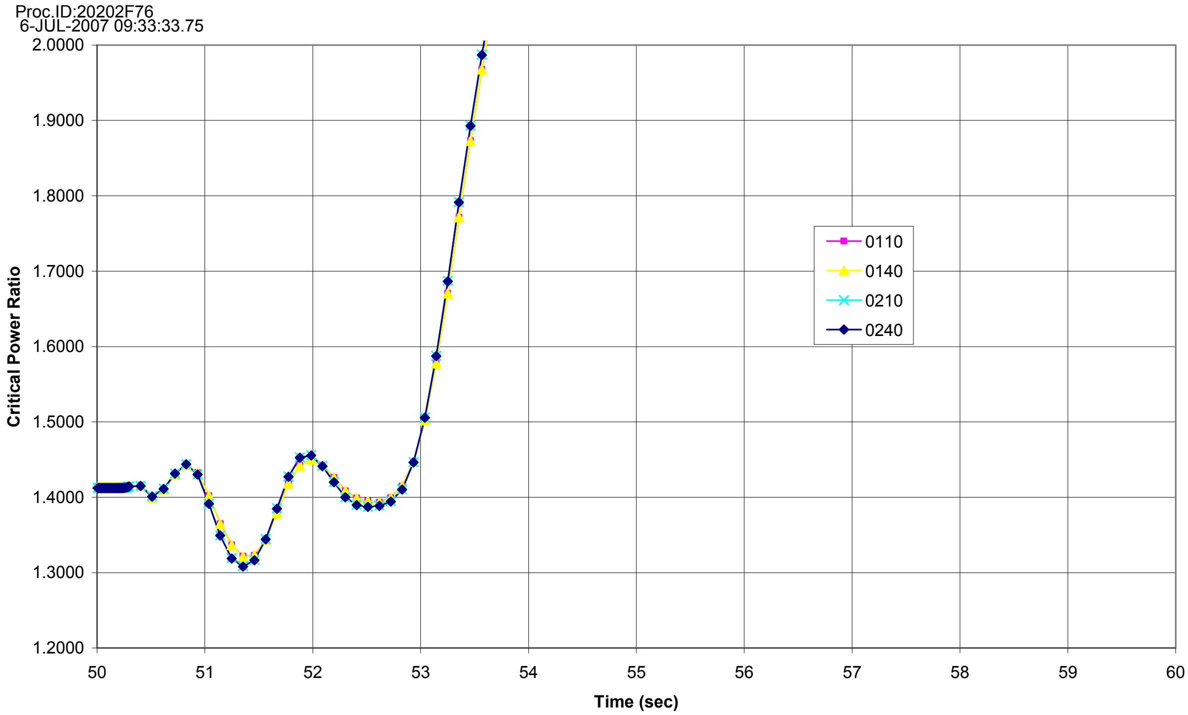



**Figure 2.3-15g. Loss of Non-Emergency AC Power to Station Auxiliaries**

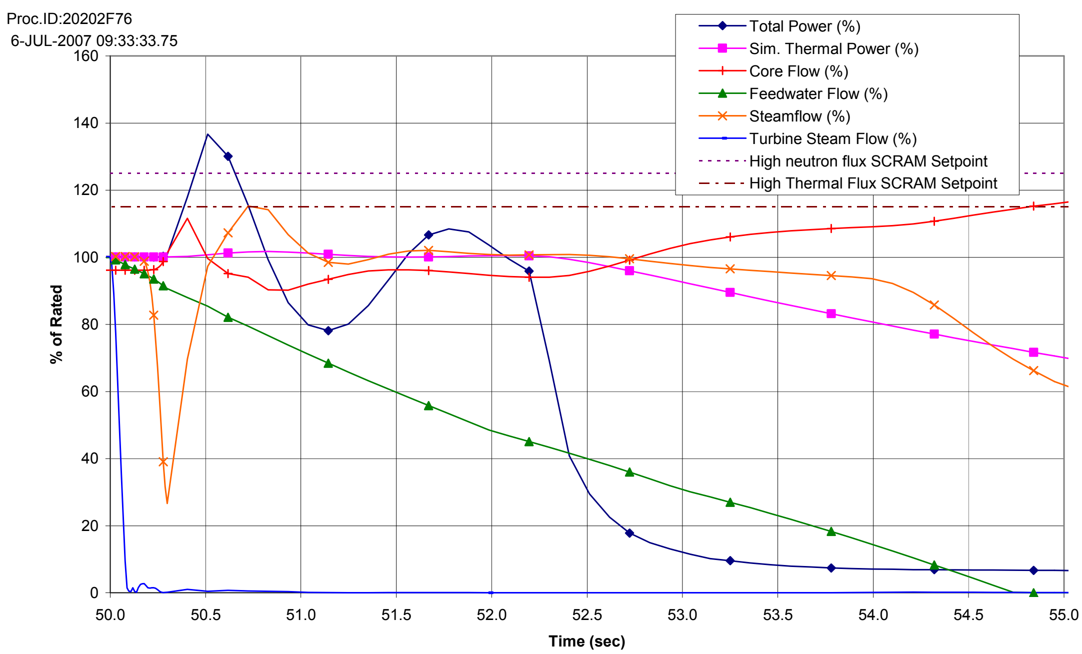



**Figure 2.3-15h. Loss of Non-Emergency AC Power to Station Auxiliaries (Figure 2.3-15a from 50 to 55s)**

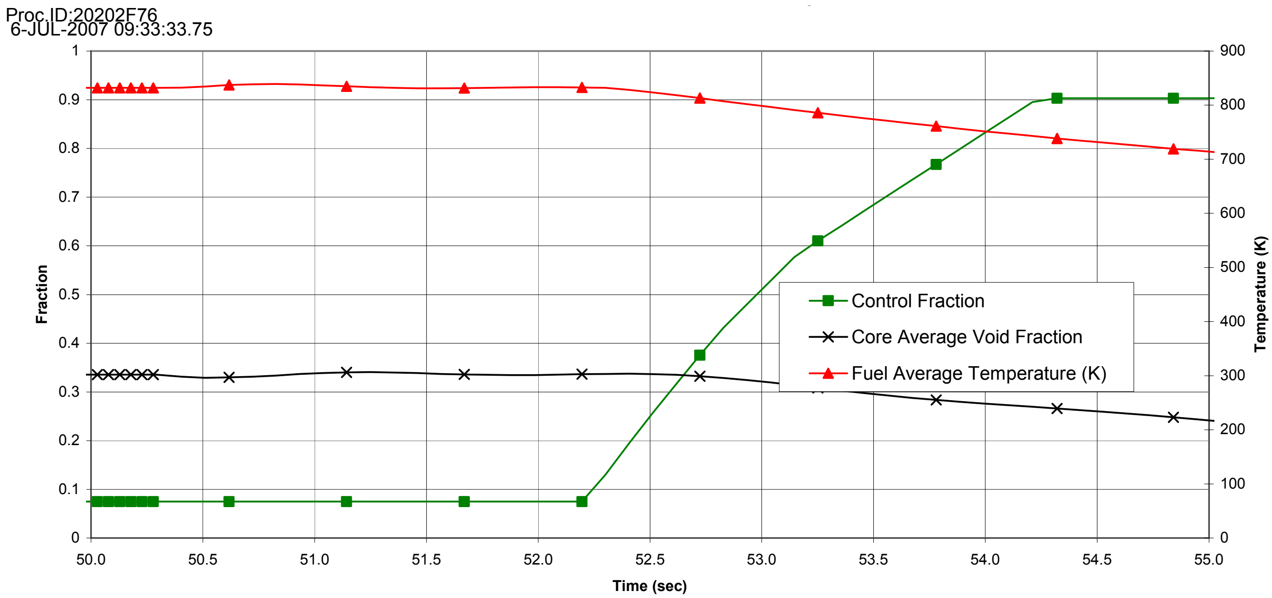



**Figure 2.3-15i. Loss of Non-Emergency AC Power to Station Auxiliaries (Figure 2.3-15f from 50 to 55s)**

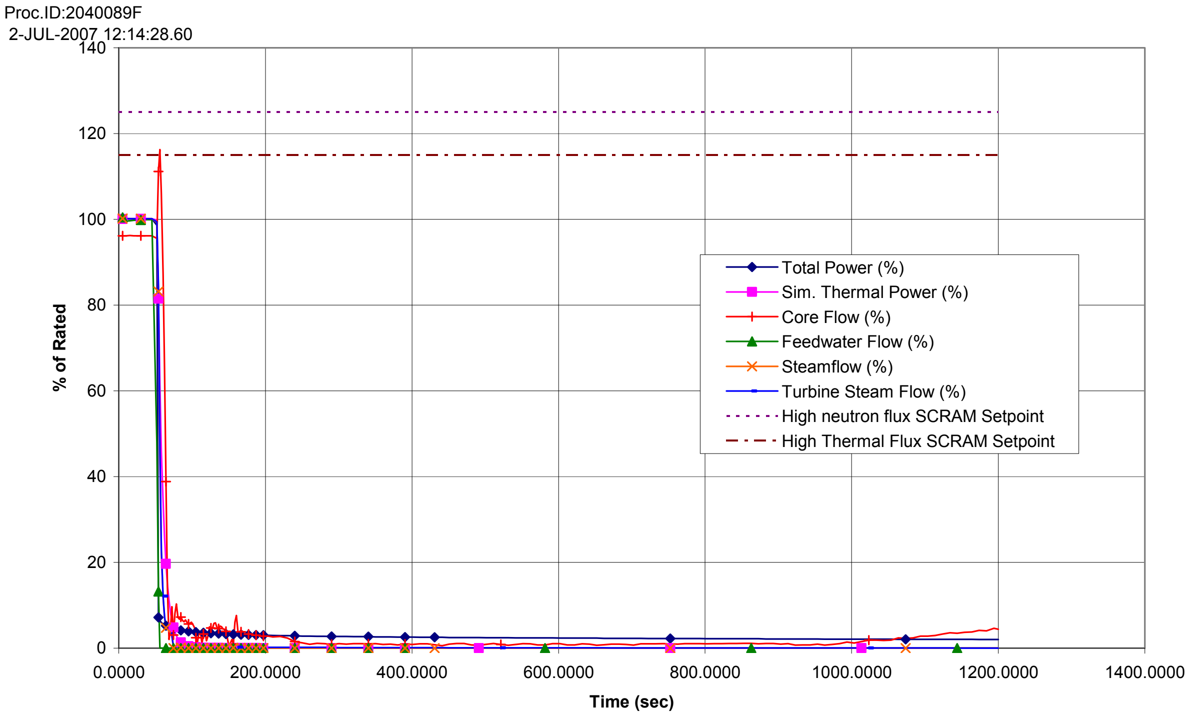



Figure 2.3-16a.  Loss of All Feedwater Flow

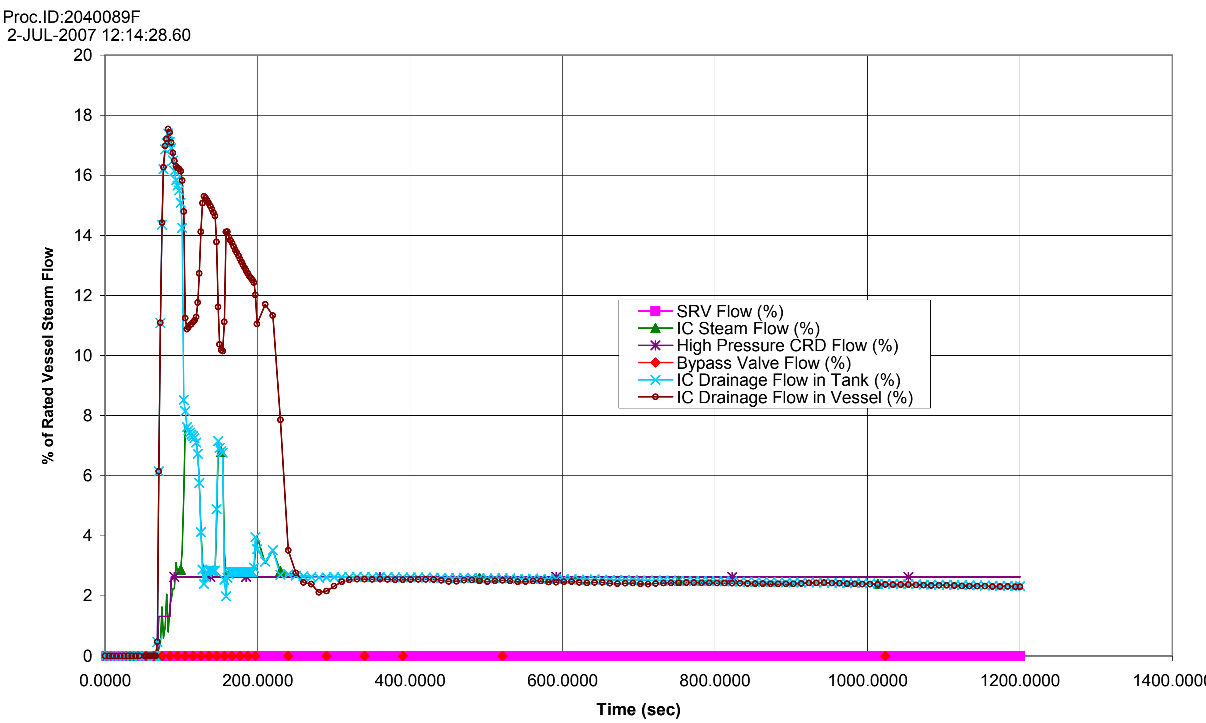



Figure 2.3-16b.  Loss of All Feedwater Flow

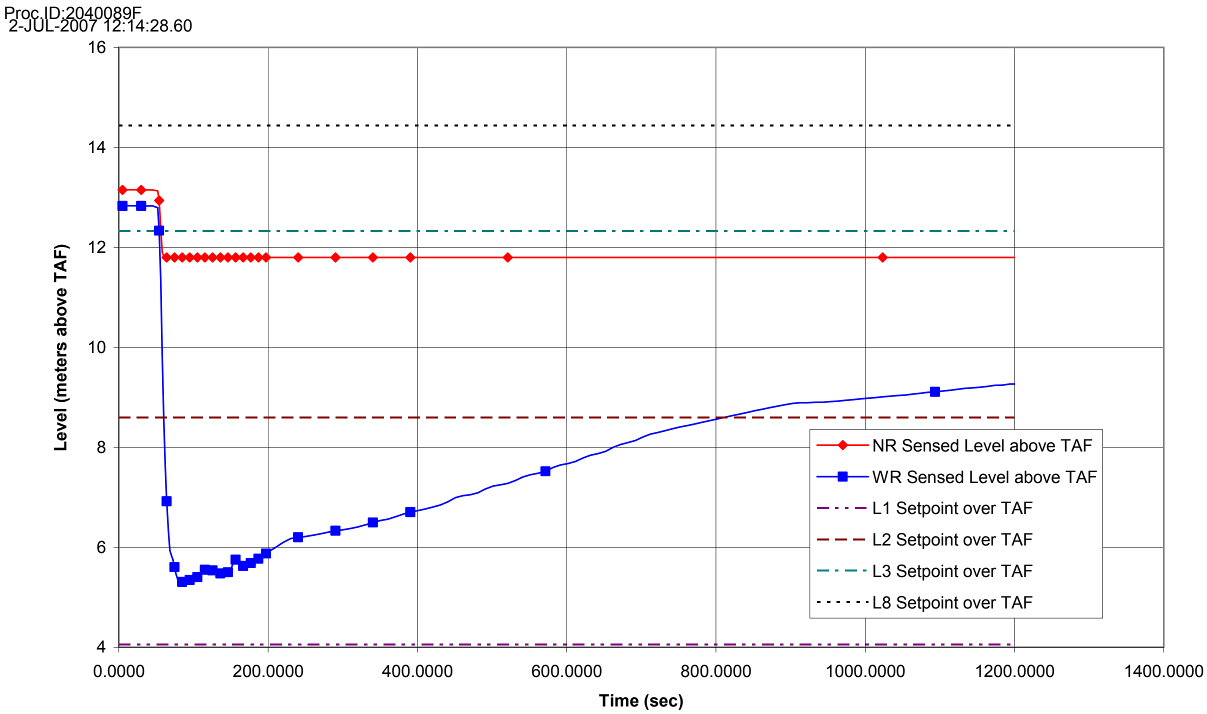



**Figure 2.3-16c. Loss of All Feedwater Flow**

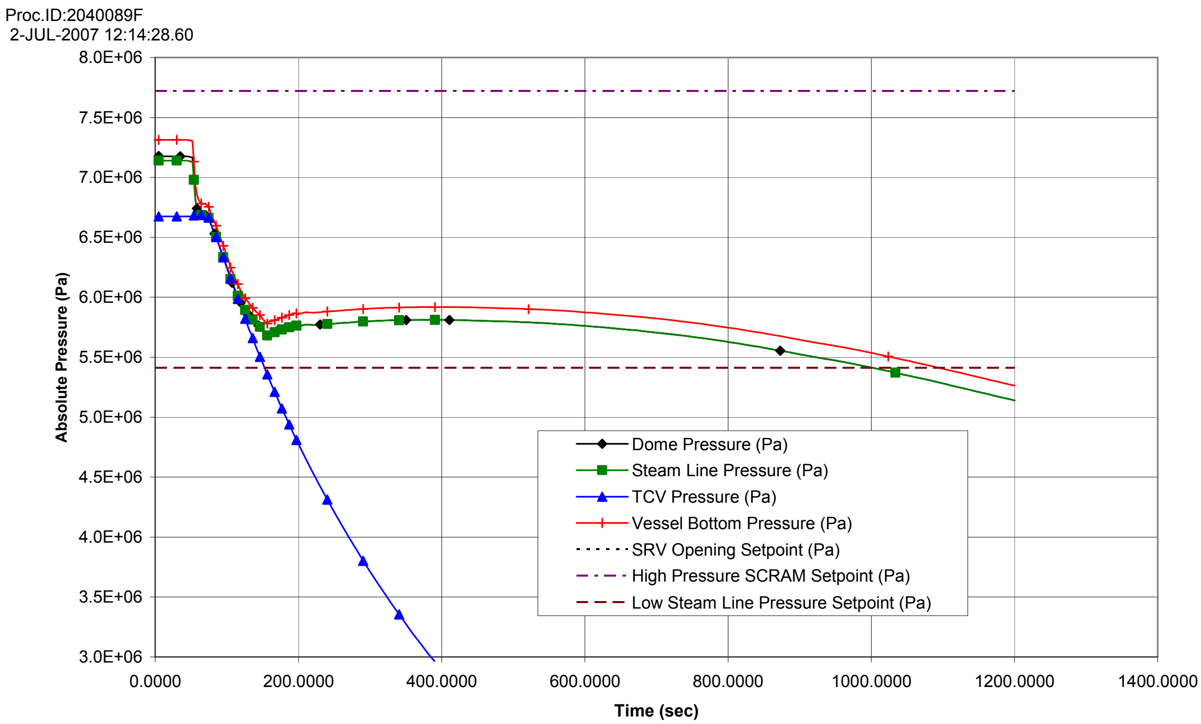



**Figure 2.3-16d. Loss of All Feedwater Flow**

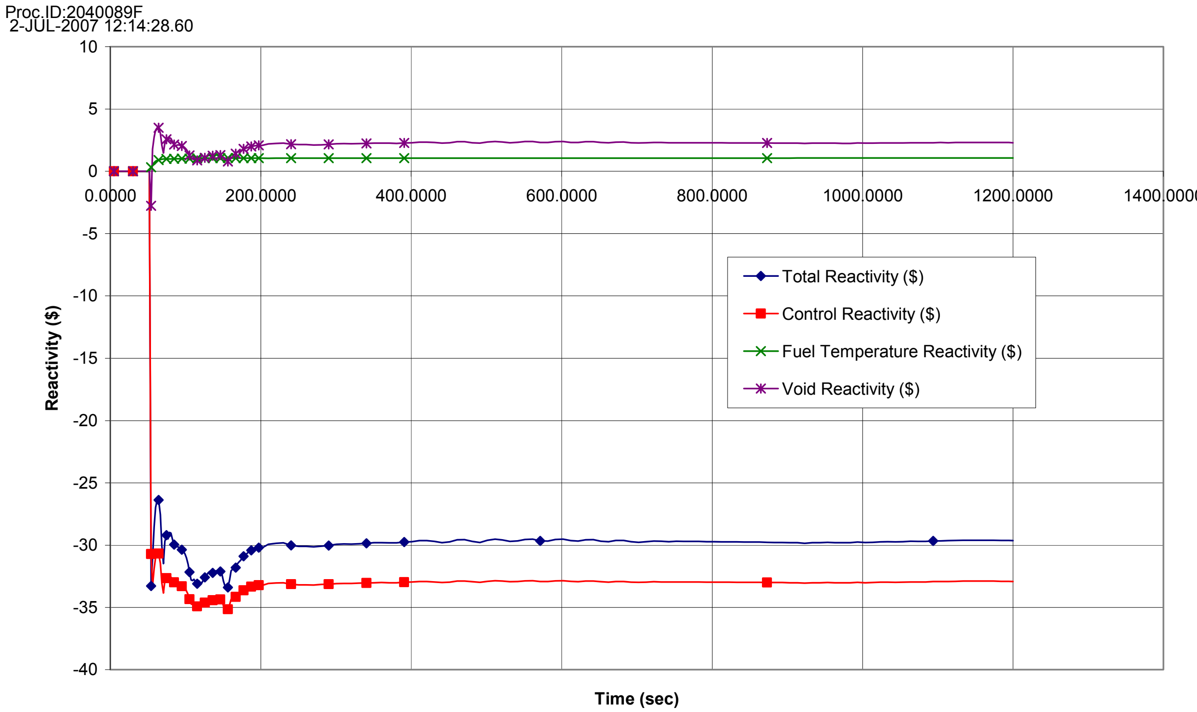



**Figure 2.3-16e. Loss of All Feedwater Flow**

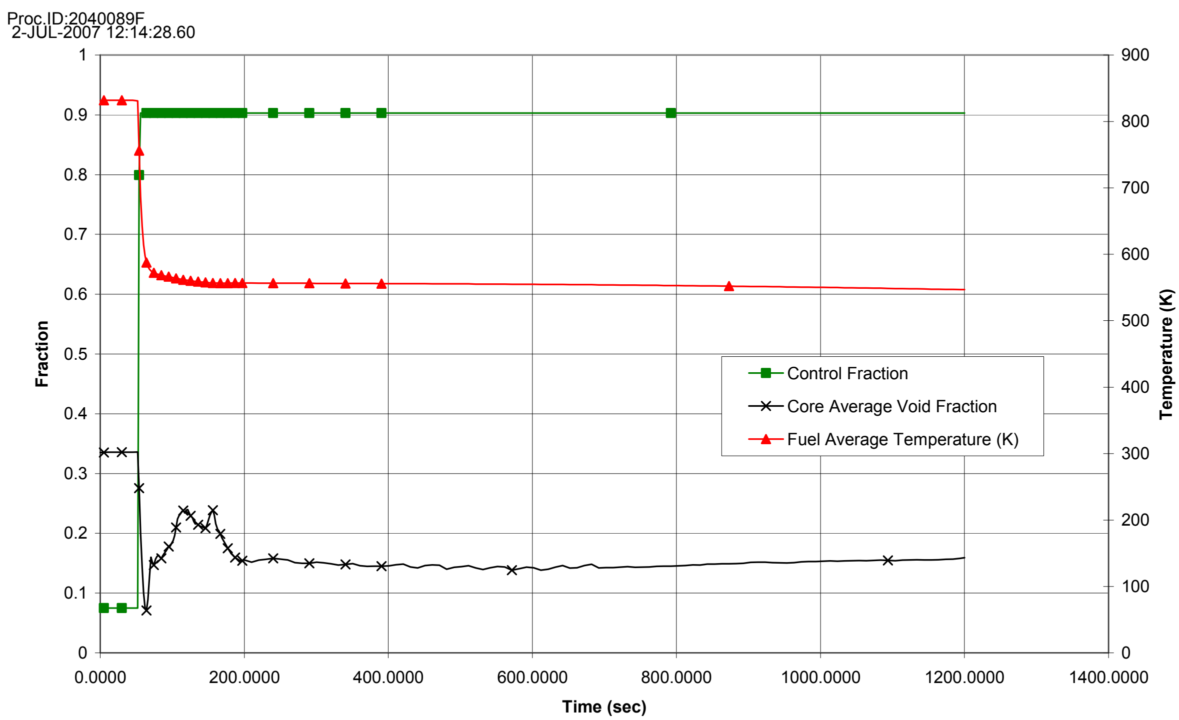



**Figure 2.3-16f. Loss of All Feedwater Flow**

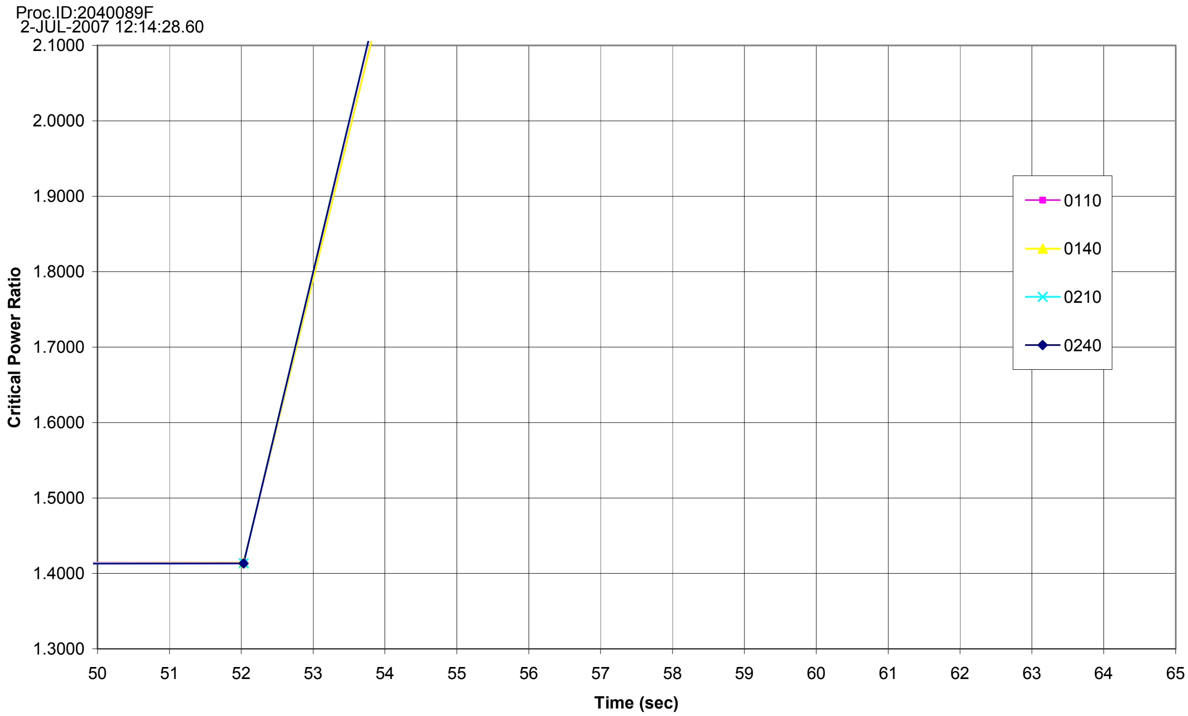



Figure 2.3-16g. Loss of All Feedwater Flow

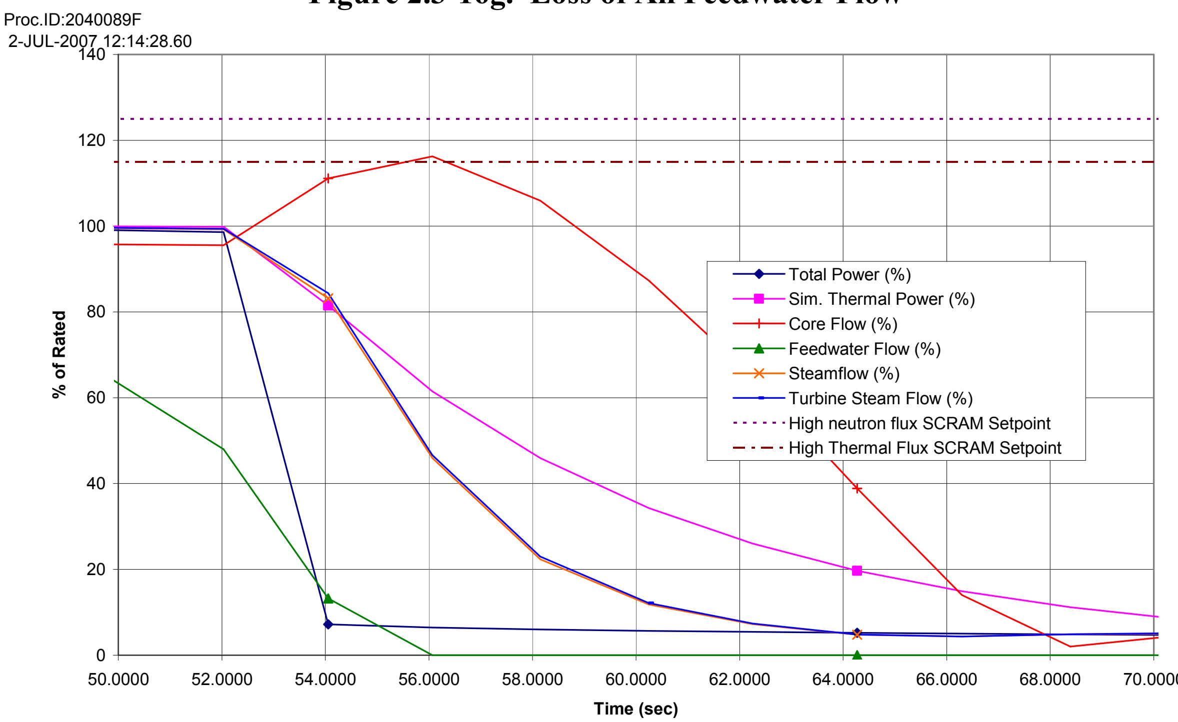



Figure 2.3-16h. Loss of All Feedwater Flow (Figure 2.3-16a from 50 to 70s)

## 2.4 ANALYSIS OF INFREQUENT EVENTS

In Reference 2.4-3, ESBWR DCD, 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, Tier 2, Chapter 15 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 Tables 2.4-1a and Table 2.4-1b.

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 SCRRI and SRI

#### *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. There is no operator action required to mitigate the event. However, operators will not permit reactor operation at elevated powers, and will lower power in accordance with the applicant's license and regulations.

***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.

#### *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. FW enthalpy resulting in the maximum possible power allowed by the STPT setpoint is assumed.

***Results***

Figure 2.4-1 shows the results of the event simulation. Table 2.4-1 provides a summary of the results. 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 at or higher than the OLMCPR value documented in Subsection 2.3.6, the radiological analysis performed in the DCD, Tier 2, Subsection 15.3.1.5 (Reference 2.3-3) is applicable in the initial core analysis.

### 2.4.2 Feedwater Controller Failure – Maximum Flow Demand

#### *2.4.2.1 Identification of Causes*

See Subsection 2.3.4.2. 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. There is a FW isolation function on high water level, but it is not assumed in this analysis because it does not have a significant effect on this event evaluation.

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**

There is no operator action required to mitigate 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. Table 2.4-1a provides a summary of the results. 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 ICS 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

There is no operator action required to mitigate the event.

##### 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. Table 2.4-1a provides a summary of the results.

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

There is no operator action required to mitigate the event.

#### 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*

A pressure regulator downscale failure is simulated as shown in Figure 2.4-4. Table 2.4-1a provides a summary of the results.

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. The fuel thermal-mechanical response shows margin to the AOO cladding strain and fuel melt criteria; therefore, no thermal-mechanical related fuel failures are expected.

#### *2.4.4.5 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.

### 2.4.5 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

There is no operator action required to mitigate the event.

##### 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.

#### *2.4.5.3 Core and System Performance*

##### 2.4.5.3.1 Input Parameters and Initial Conditions

The TGCS 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. Table 2.4-1 provides a summary of the results.

### *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*

With an initial core OLMCPR at or higher than the OLMCPR value documented in Subsection 2.3.6, the radiological analysis performed in the DCD, Tier 2, Subsection 15.3.1.5 (Reference 2.3-3) is applicable in the initial core analysis.

## 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

There is no operator action required to mitigate the event.

#### 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. Table 2.4-1a provides a summary of the results.

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. The withdrawal of a control rod (or highest worth pair of control rods associated with the same HCU) does not result in criticality.

### 2.4.8 Control Rod Withdrawal Error During Startup with Failure of Control Rod Block

Analysis in the DCD is performed with the initial core.

### 2.4.9 Control Rod Withdrawal Error During Power Operation with ATLM Failure

Analysis in the DCD applies to the initial core. The MRBM prevents more than 1000 fuel rod failures.

### 2.4.10 Fuel Assembly Loading Error, Mislocated Bundle

Analysis in the DCD applies to the initial core. The potential exists in this scenario that one or more fuel rods could experience cladding failure. The detection of the fuel leak is followed by power suppression of the damaged rod.

### 2.4.11 Fuel Assembly Loading Error, Misoriented Bundle

Analysis in the DCD applies to the initial core. The potential exists in this scenario that one or more fuel rods could experience cladding failure. The detection of the fuel leak is followed by power suppression of the damaged rod.

### 2.4.12 Inadvertent SDC Function Operation

Analysis in the DCD applies to the initial core. The increased subcooling caused by misoperation of the RWCU/SDC shutdown cooling mode could result in a slow power increase due to the reactivity insertion. During power operation, the reactor settles in a new steady state. During startup, if the power rises such that the neutron flux setpoint is reached, the power rise is terminated by a flux scram before approaching fuel thermal limits.

### 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 operator will monitor suppression pool temperature and water level, and isolate makeup from sources external to the containment if necessary.

#### *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*

Figure 2.4-8 shows the results of the event simulation. Table 2.4-1 provides a summary of the results. The opening of one SRV allows steam to be discharged into the suppression pool. The valve is assumed to open instantaneously, and 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, Tier 2 Table 5.2-2 (Reference 2.4-3). This increased capacity, about 5% rated flow, 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. No fuel damage results from the event, and containment pressure does not reach containment design pressure limits.

### 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 isolation condensers produce a depressurization, with the HP CRD operating to recover the level after the scram and to maximize depressurization. The event is analyzed with four isolation condensers 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 operator will monitor suppression pool temperature and water level, and isolate makeup from sources external to the containment if necessary

***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, Tier 2, 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*

Figure 2.4-9 shows the results of the event simulation. Table 2.4-1a provides a summary of the results. 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. Most of the depressurization results from the operation of four isolation condensers.

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. The analysis is only analyzed for radiological consequences of the spill and is independent of the fuel.

### 2.4.17 References

2.4-1 Global Nuclear Fuel, "ESBWR Initial Core Nuclear Design Report", NEDC-33326-P, Class III (Proprietary), Revision 1March 2009, NEDO-33326, Class I (Non-proprietary), Revision 1, March 2009.

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

2.4-3 GE Nuclear Energy, "ESBWR Design Control Document" 26A6642.

## Table 2.4-1a

## Results Summary of Infrequent Events (1) (2)

| Sub-section I.D. | Description | Max. Neutron Flux, % NBR | Max. Dome Pressure, MPaG (psig) | Max. Vessel Bottom Pressure, MPaG (psig) | Max. Steamline Pressure, MPaG (psig) | Max. Simulated Thermal Power, % NBR | ΔCPR/ ICPR | Maximum Calculated TEDE |
|---|---|---|---|---|---|---|---|---|
| 2.4.1 | Loss of Feedwater Heating with SCRRI failure | 115 | 7.10 (1030) | 7.24 (1050) | 6.97 (1011) | 115 | 0.09 | (Note 4) |
| 2.4.2 | FWCF – Maximum Flow 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 | 432.5 | 8.03 (1165) | 8.16 (1184) | 8.05 (1167) | 105.8 | 0.16 | (Note 4) |
| 2.4.6 | Turbine Trip with total bypass failure | 320.9 | 8.01 (1162) | 8.15 (1182) | 8.02 (1164) | 103.6 | 0.05 | (Note 4) |
| 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 (3) | -- | -- | -- | -- | -- | <0.01 | -- |

(1) The input parameters and initial conditions used to perform the analysis in this table are located in Table 2.3-1.

(2) This table summarizes the events calculated with the TRACG code. Table 2.4-1b contains the summary of the remaining Infrequent Events.

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

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

## Table 2.4-1b

## Results Summary of Infrequent Events [1]

**Data in DCD, Tier 2, Table 15.3-1b applies to the initial core.**

(1) Table 2.4-1a summarizes the events calculated with the TRACG code for the initial core.

**Table 2.4-2**

**Sequence of Events for Loss of Feedwater Heating With Failure of SCRRI and SRI**

| 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); however, SCRRI/SRI fails. |
| 25 (est.) | Initial effect of unheated FW starts to raise core power level |
| ≈225 | The STPT setpoint (115%) is reached, the activation of Scram is not credited. |
| 300 (est.) | Reactor variables settle into new steady state. |

* See Figure 2.4-1.

Note: This event is analyzed with FW temperature reduction that corresponds to a maximum thermal power at the STPT. Larger FW temperature reductions result in a scram terminating the event and a similar ΔCPR/ICPR.

**Table 2.4-3**

**Sequence of Events for Feedwater Controller Failure – Maximum Flow Demand**

| Time (sec) | Event * |
|---|---|
| 0 | Initiate simulated runout of all FW pumps (170% at rated vessel pressure ) |
| 12.8 | 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 |
| >20.0 | L2 is reached because no FW availability, activating ICS 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 ICS |
| 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 isolation condensers 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<br>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 |
| 0.08 | Turbine control valves closed |
| 0.15 | After detection of not enough bypass availability the RPS initiates a reactor scram |
| 0.35 | Control Rod insertion begins |
| 1.7 | L3 setpoint is reached |
| 12 | L2 setpoint is reached |
| 22 | 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 |
| 0.15 | After detection of not enough bypass availability the RPS initiates a reactor scram |
| 0.35 | Control Rod insertion begins |
| 1.8 | L3 setpoint is reaches |
| 12 | L2 setpoint is reached |
| 22 | HP CRD is activated on L2 to recover the level |

* See Figure 2.4-6.

**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 |
| 10 | The vessel begins depressurization |
| 27 | HP CRD is activated on L2 |
| 96 | Low steamline pressure setpoint is reached |
| 97 | MSIV at 85% |
| 101 | MSIV is closed |
| 127 | Isolation condenser discharge valves fully open[(1)] |
| 270 | 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.

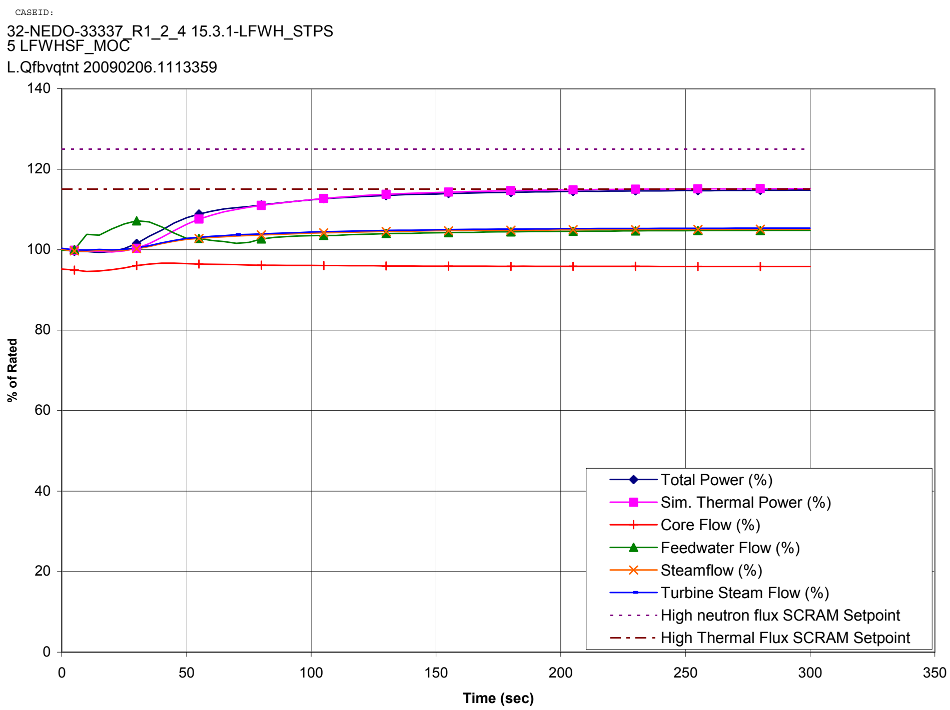



**Figure 2.4-1a. Loss of Feedwater Heating with SCRRI/SRI Failure**

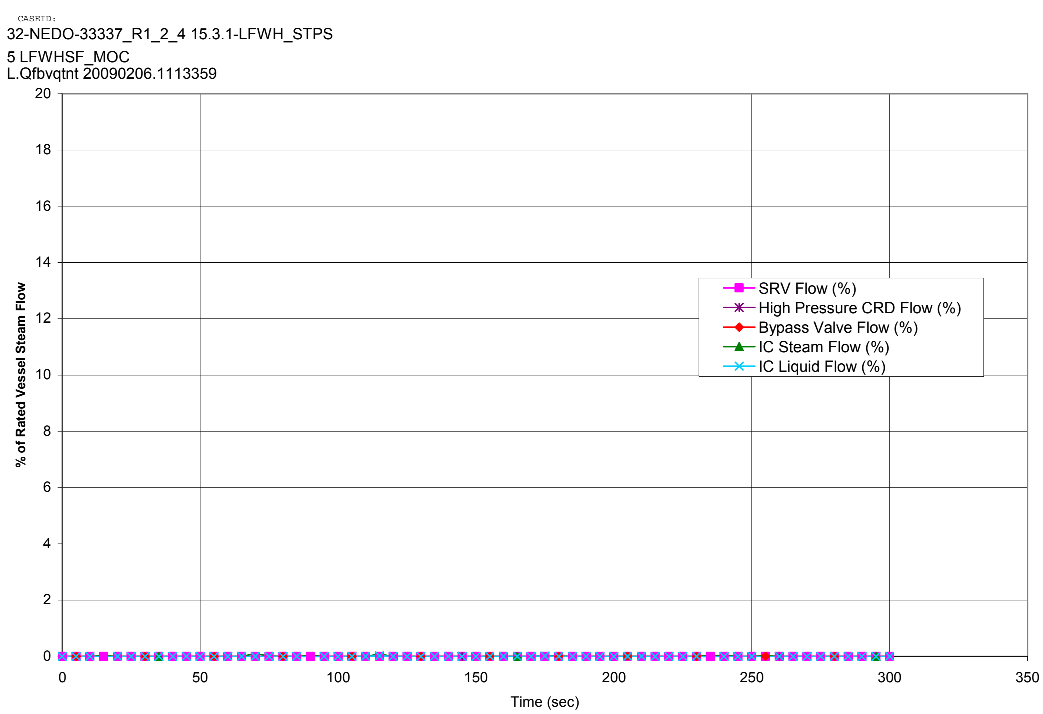



**Figure 2.4-1b. Loss of Feedwater Heating with SCRRI/SRI Failure**

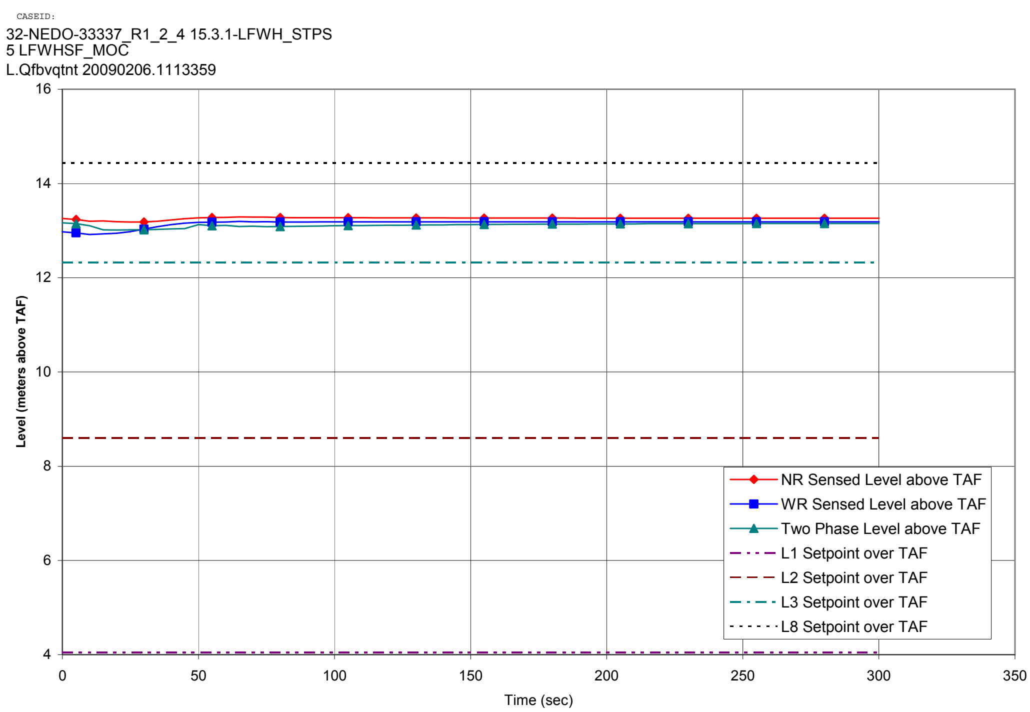



**Figure 2.4-1c.  Loss of Feedwater Heating with SCRRI/SRI Failure**

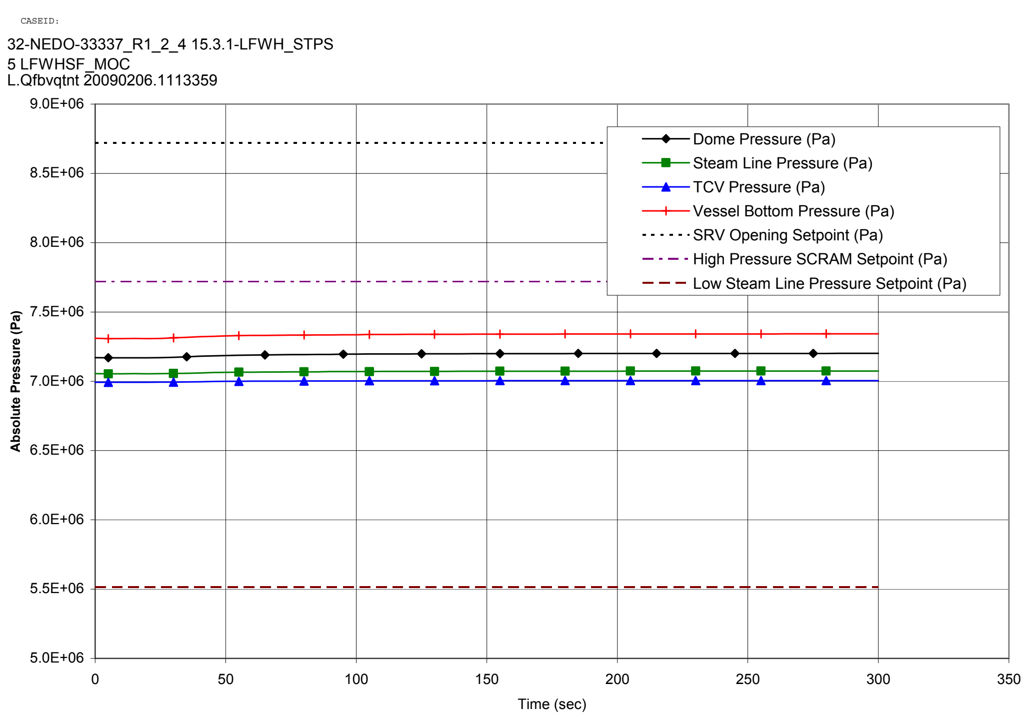



**Figure 2.4-1d.  Loss of Feedwater Heating with SCRRI/SRI Failure**

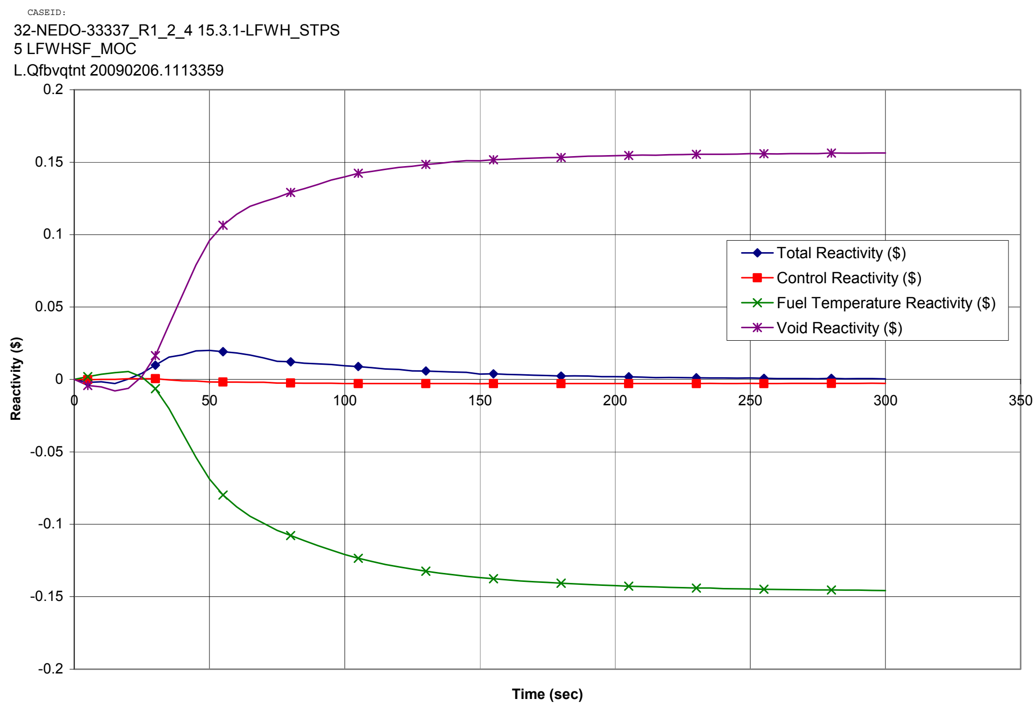



**Figure 2.4-1e. Loss of Feedwater Heating with SCRRI/SRI Failure**

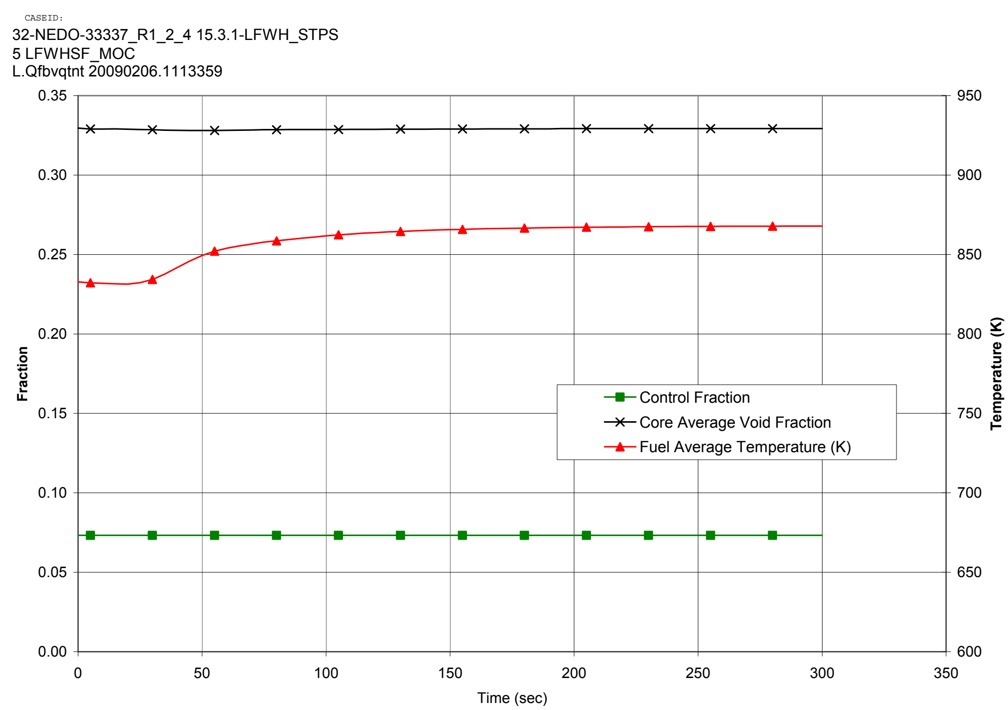



**Figure 2.4-1f. Loss of Feedwater Heating with SCRRI/SRI Failure**

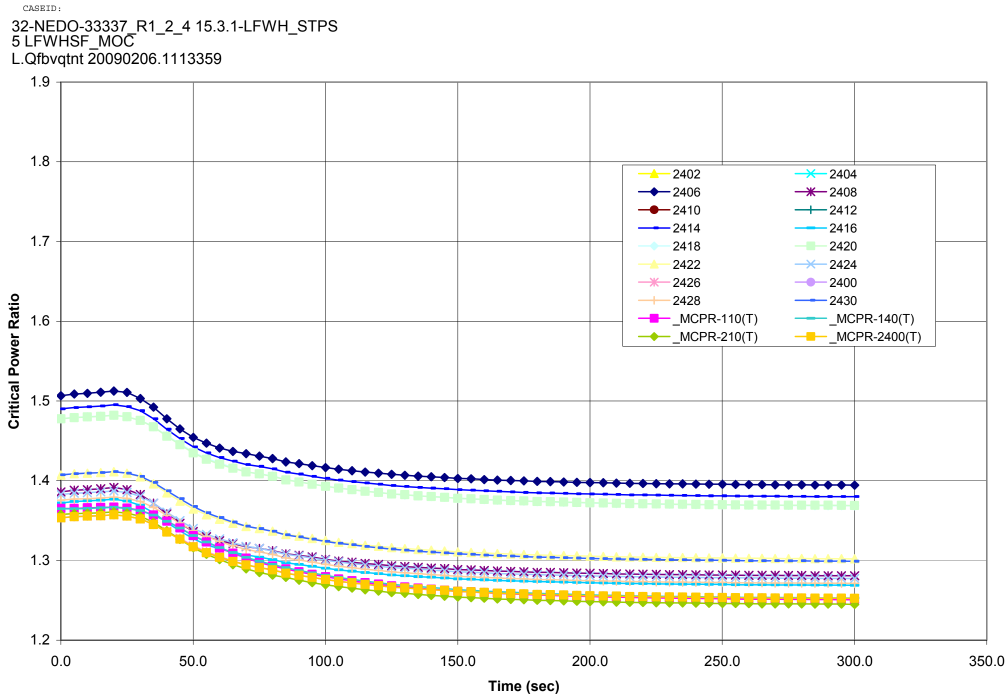



**Figure 2.4-1g. Loss of Feedwater Heating with SCRRI/SRI Failure**

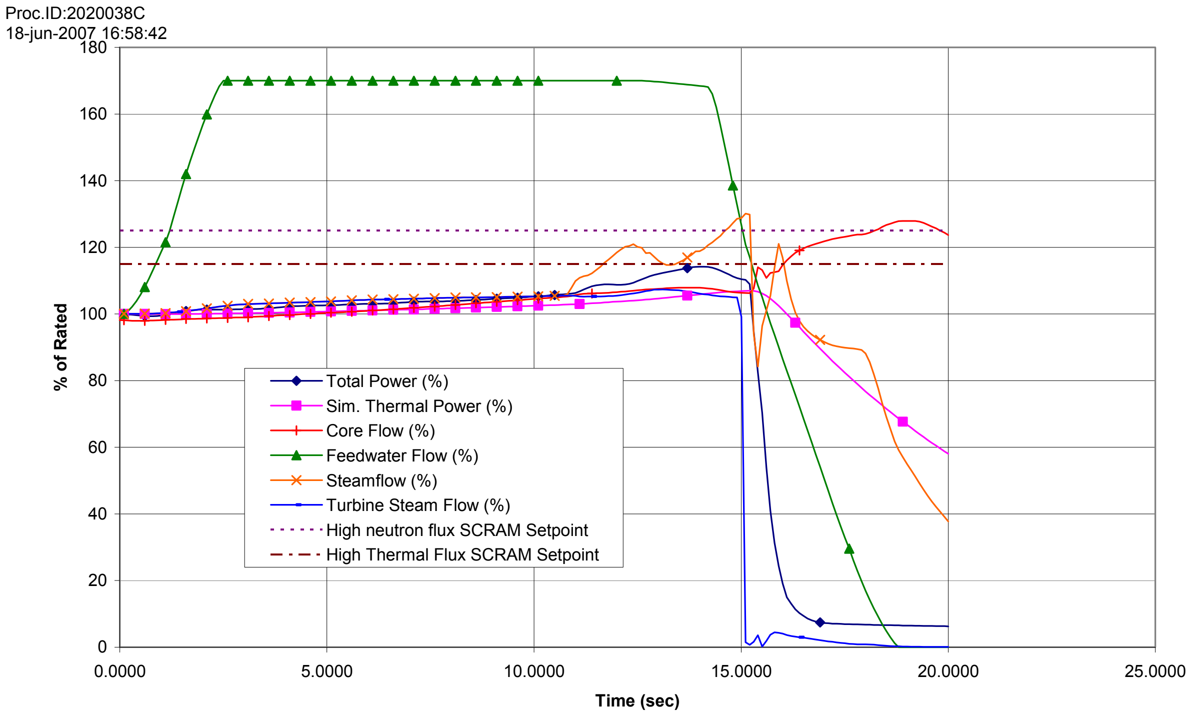



**Figure 2.4-2a. Feedwater Controller Failure – Maximum Flow Demand**

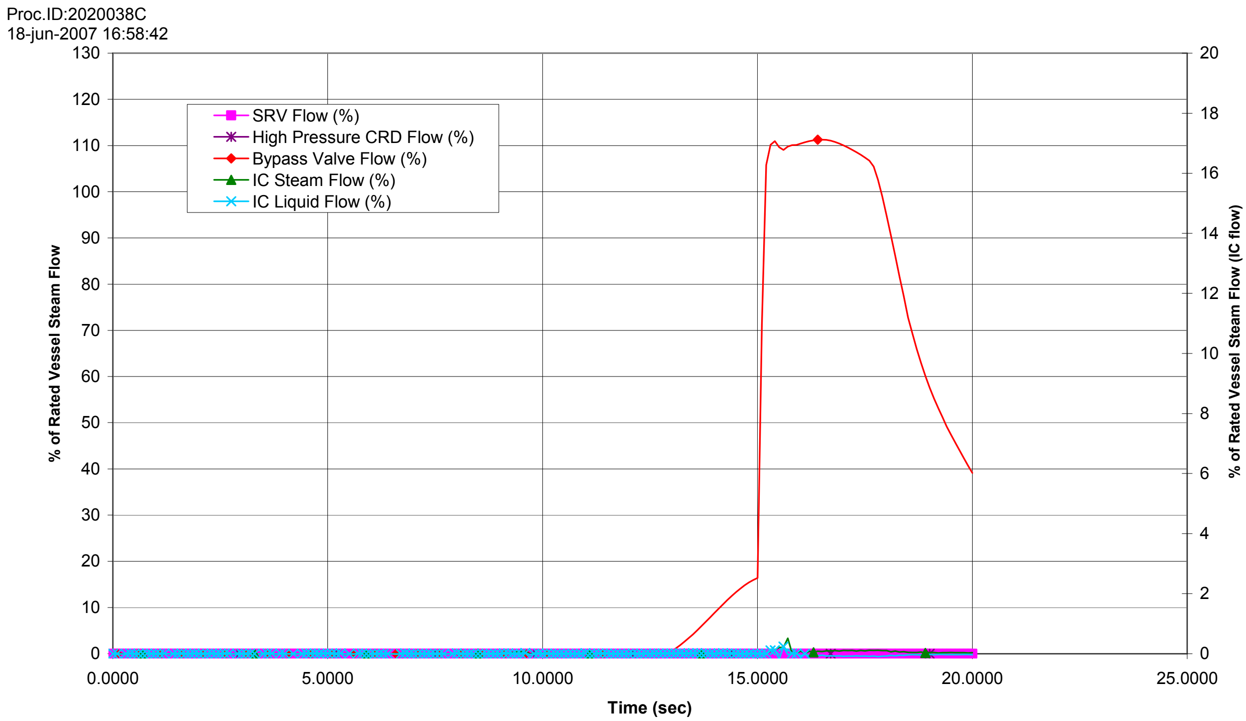



**Figure 2.4-2b. Feedwater Controller Failure – Maximum Flow Demand**

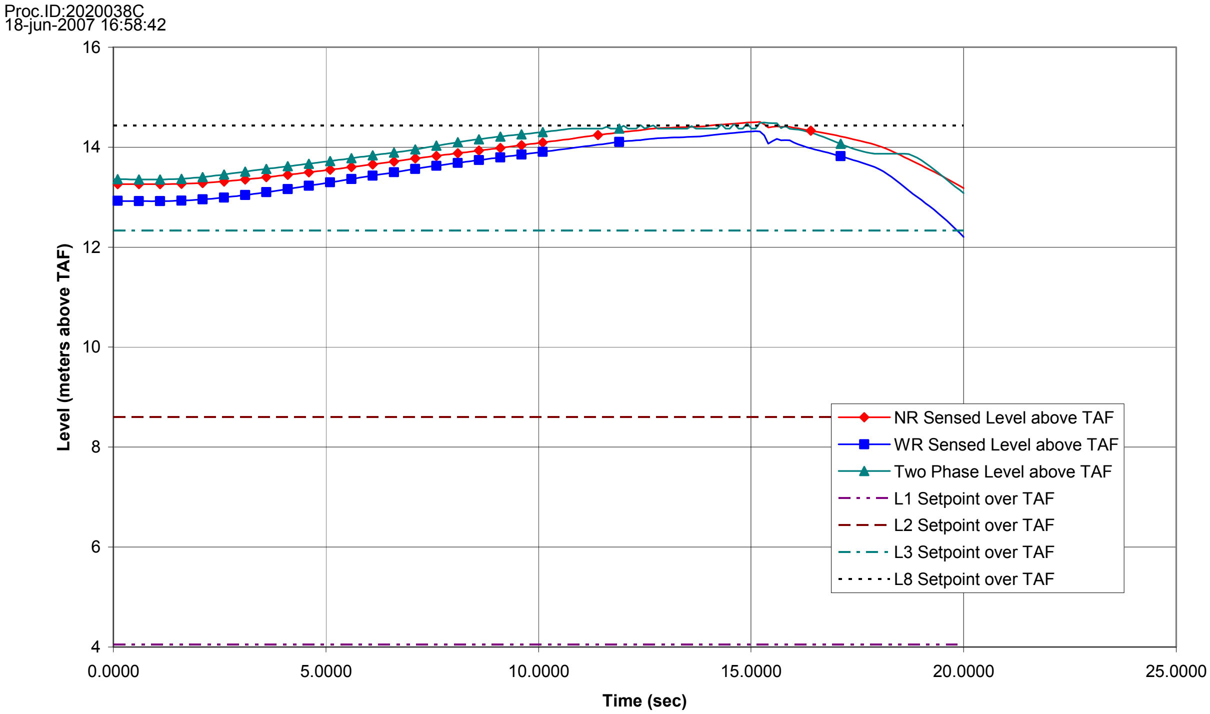



**Figure 2.4-2c. Feedwater Controller Failure – Maximum Flow Demand**

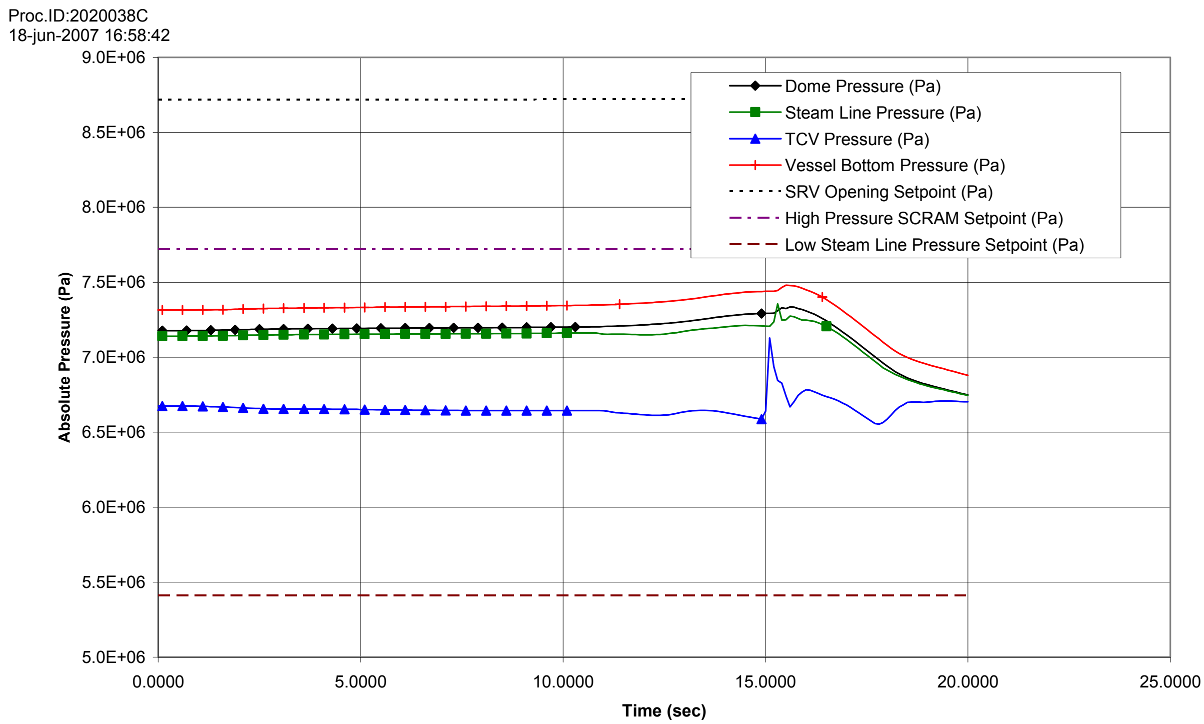



**Figure 2.4-2d. Feedwater Controller Failure – Maximum Flow Demand**

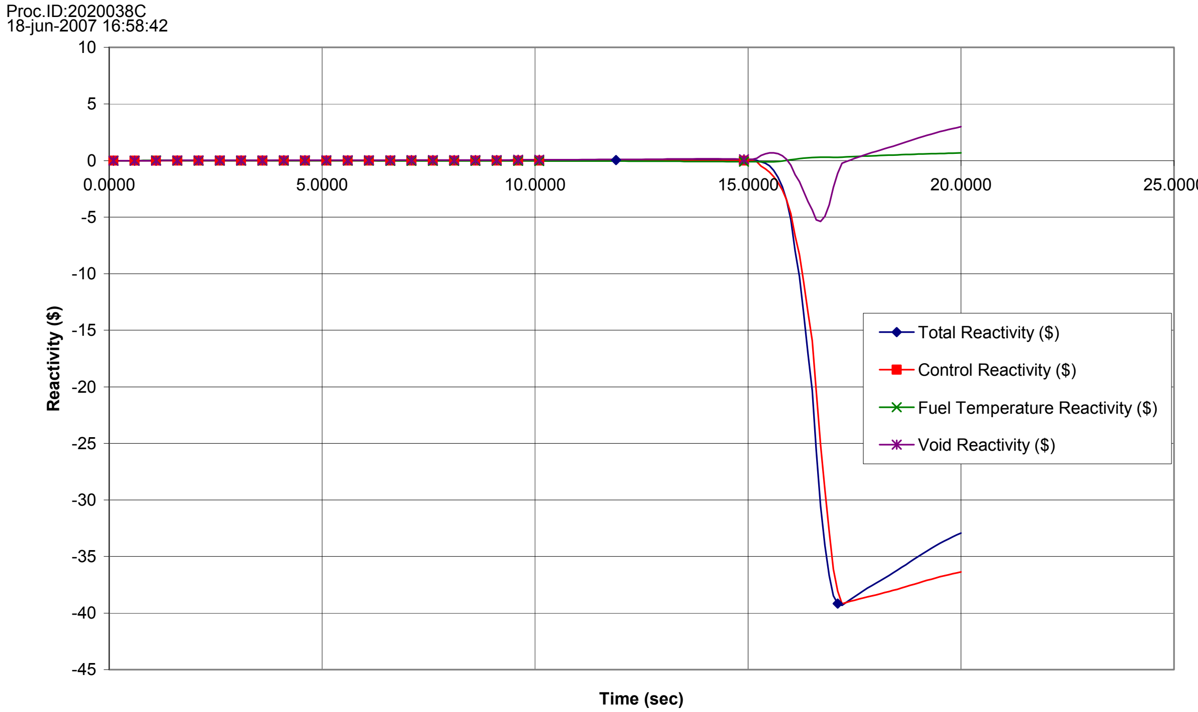



**Figure 2.4-2e. Feedwater Controller Failure – Maximum Flow Demand**

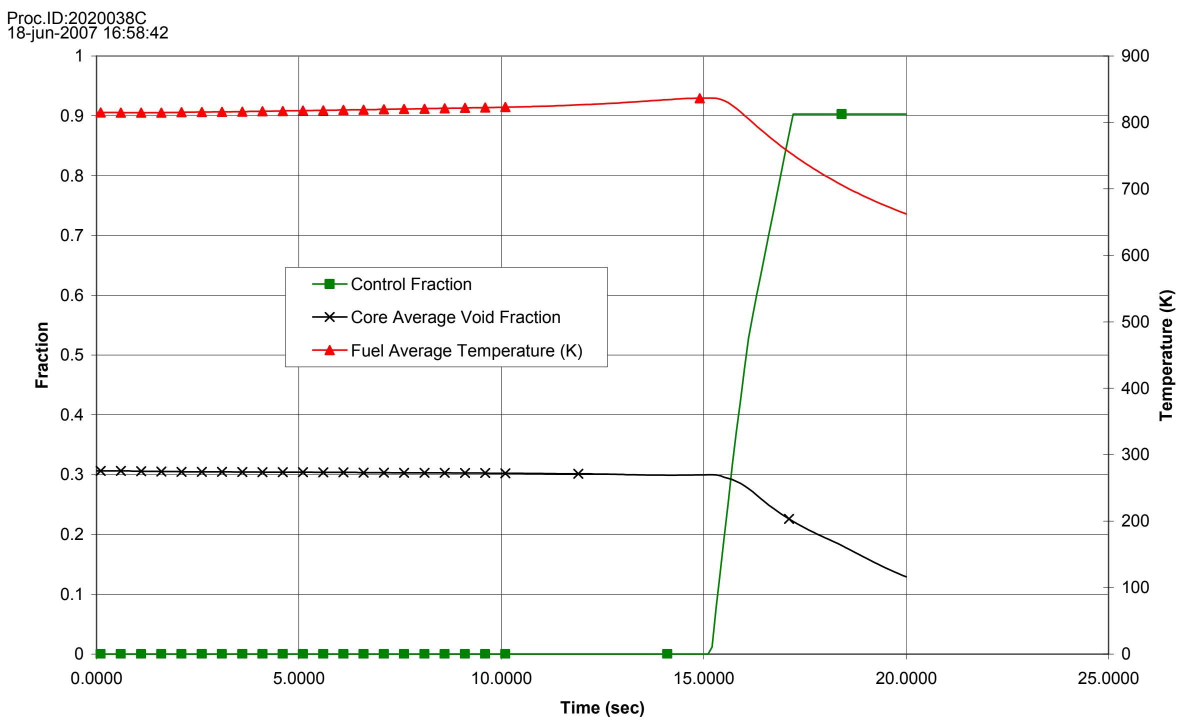



**Figure 2.4-2f. Feedwater Controller Failure – Maximum Flow Demand**

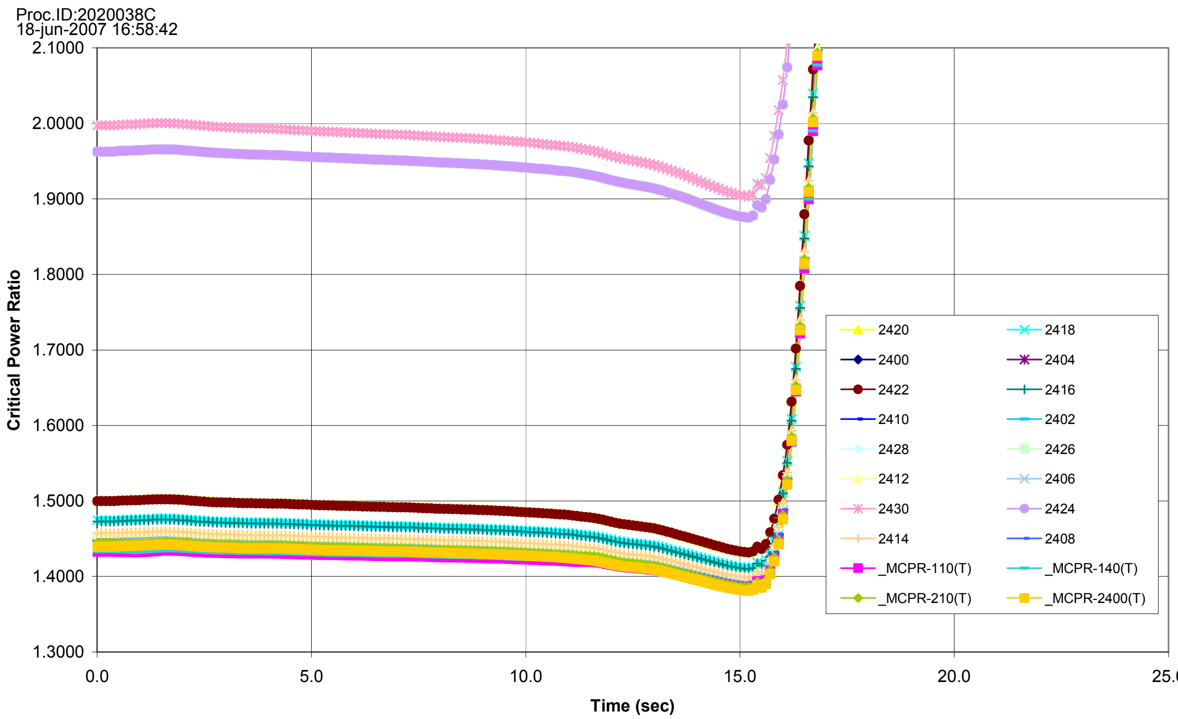



**Figure 2.4-2g. Feedwater Controller Failure – Maximum Flow Demand**

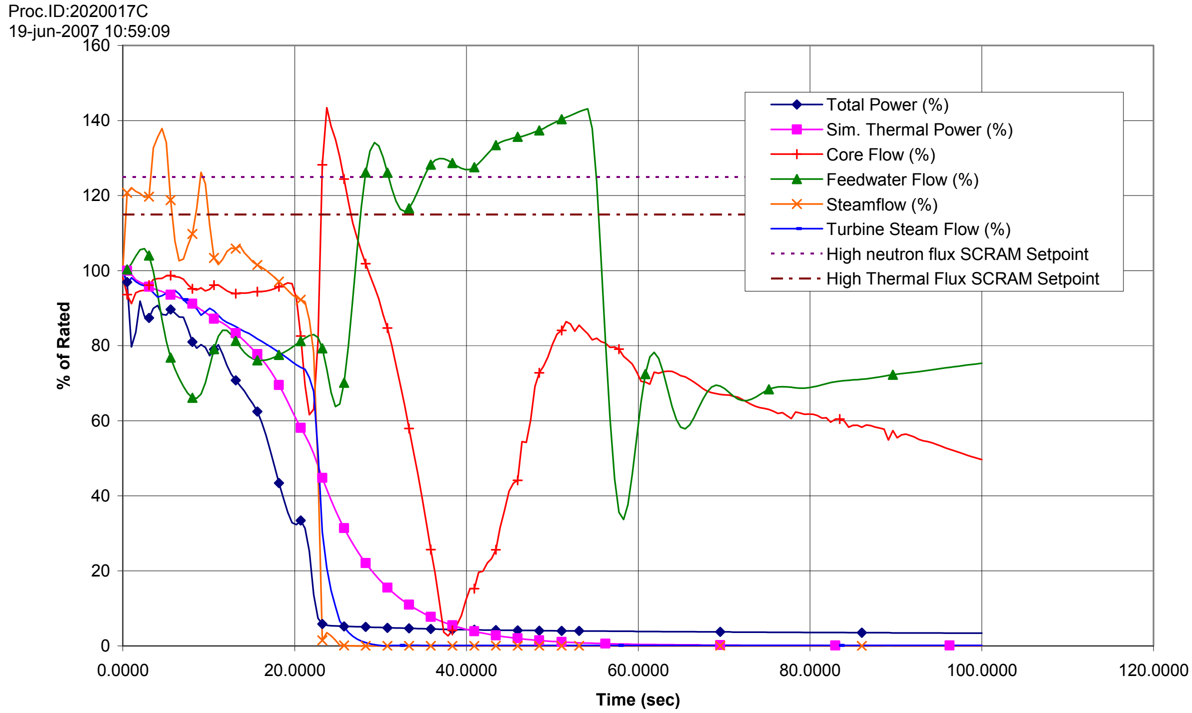



**Figure 2.4-3a. Pressure Regulator Failure – Opening of All Turbine Control and Bypass Valves**

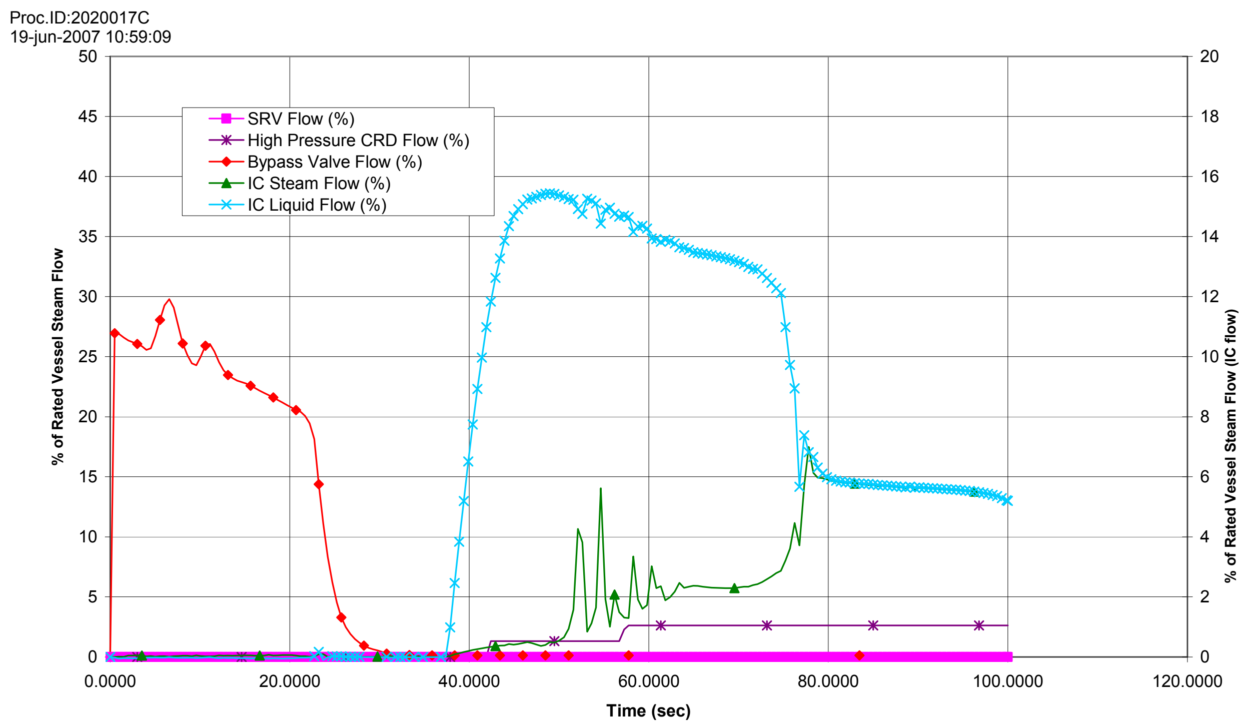



**Figure 2.4-3b. Pressure Regulator Failure – Opening of All Turbine Control and Bypass Valves**

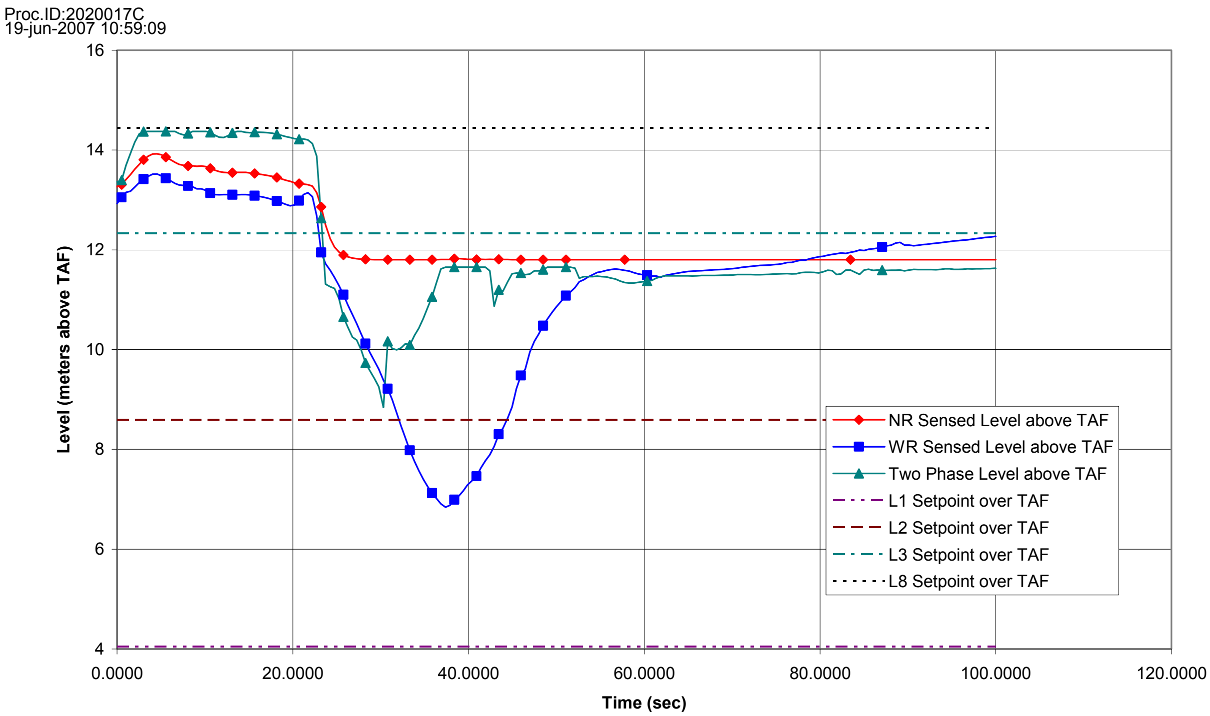



**Figure 2.4-3c. Pressure Regulator Failure – Opening of All Turbine Control and Bypass Valves**

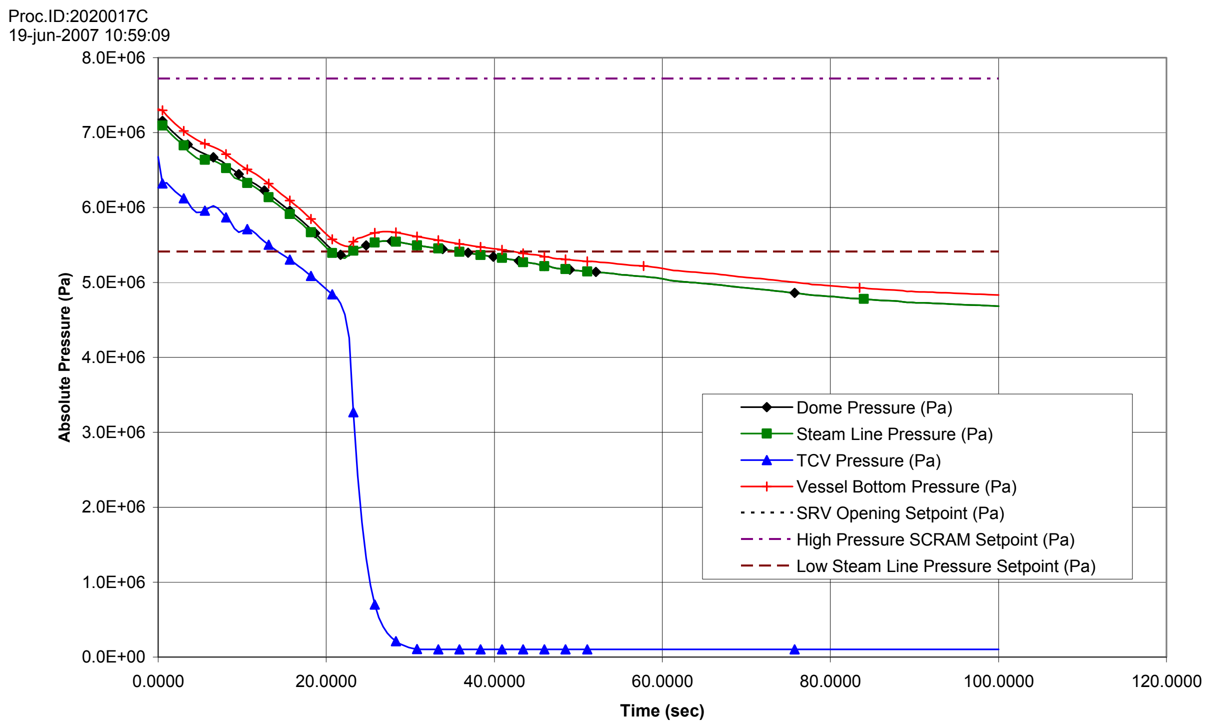



**Figure 2.4-3d. Pressure Regulator Failure – Opening of All Turbine Control and Bypass Valves**

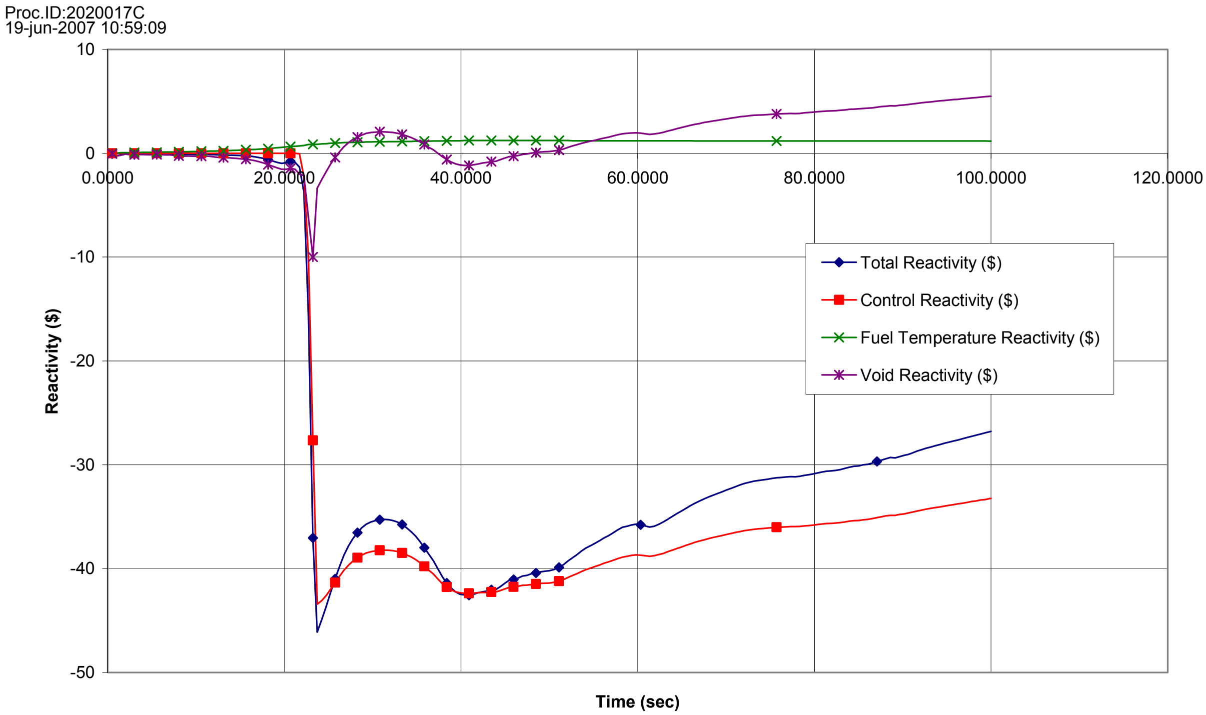



**Figure 2.4-3e. Pressure Regulator Failure – Opening of All Turbine Control and Bypass Valves**

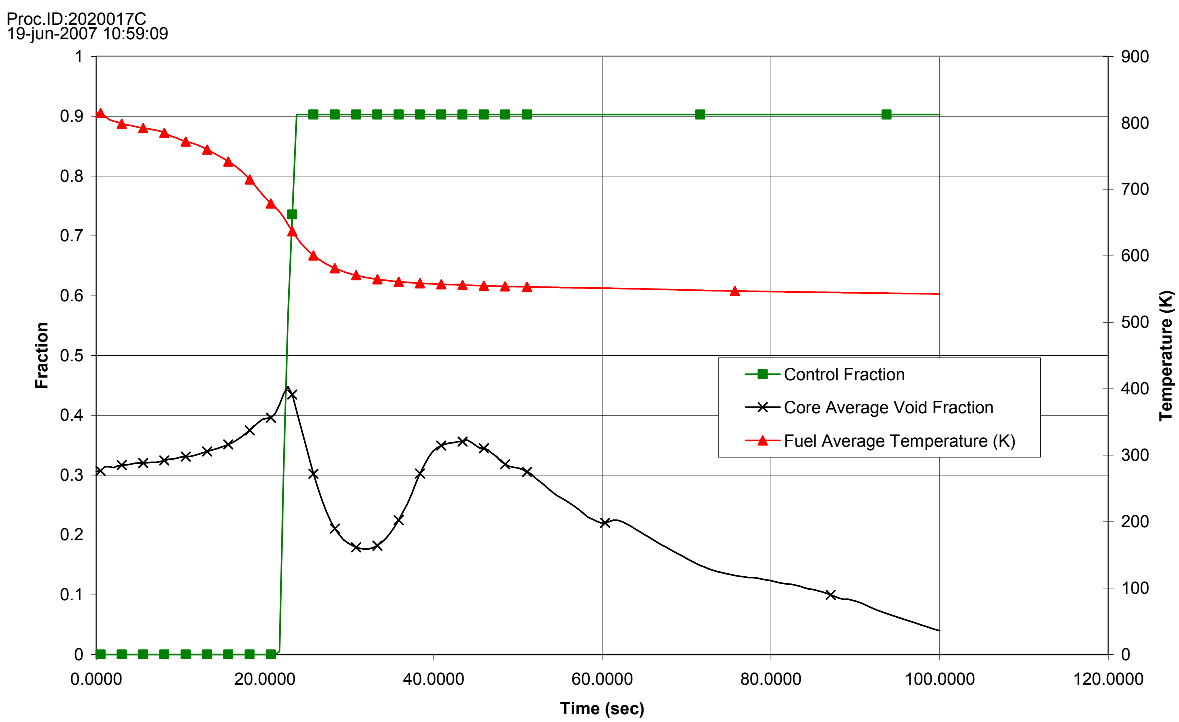



**Figure 2.4-3f. Pressure Regulator Failure – Opening of All Turbine Control and Bypass Valves**

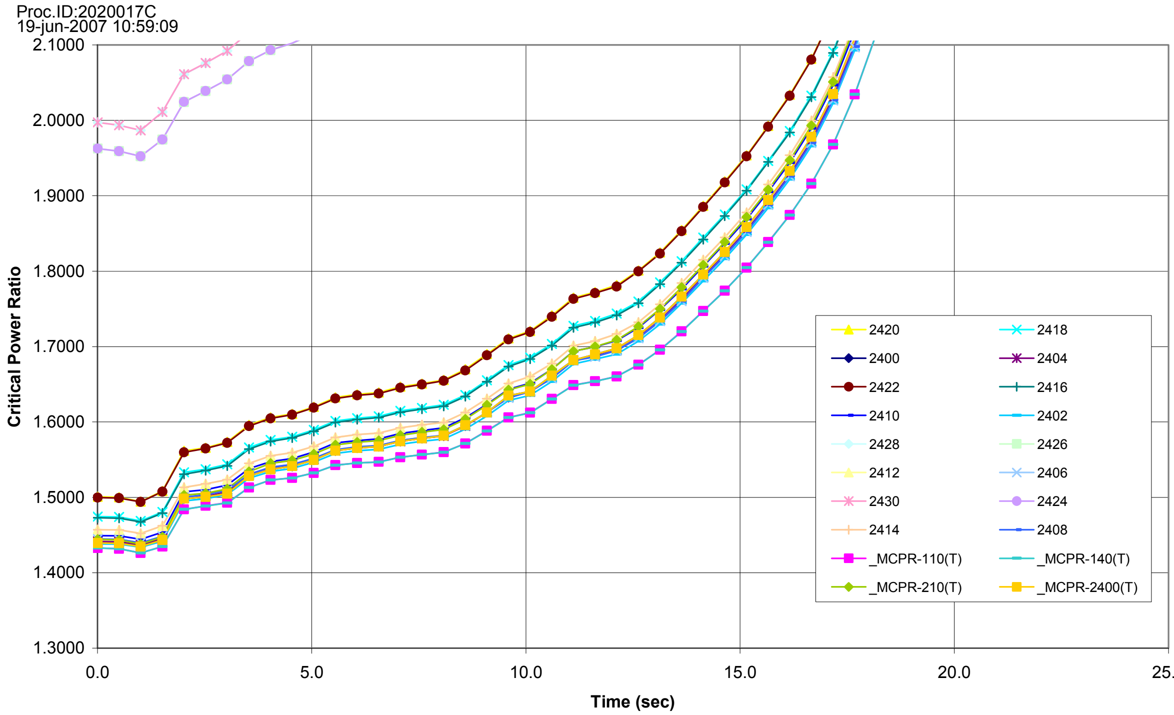



**Figure 2.4-3g. Pressure Regulator Failure – Opening of All Turbine Control and Bypass Valves**

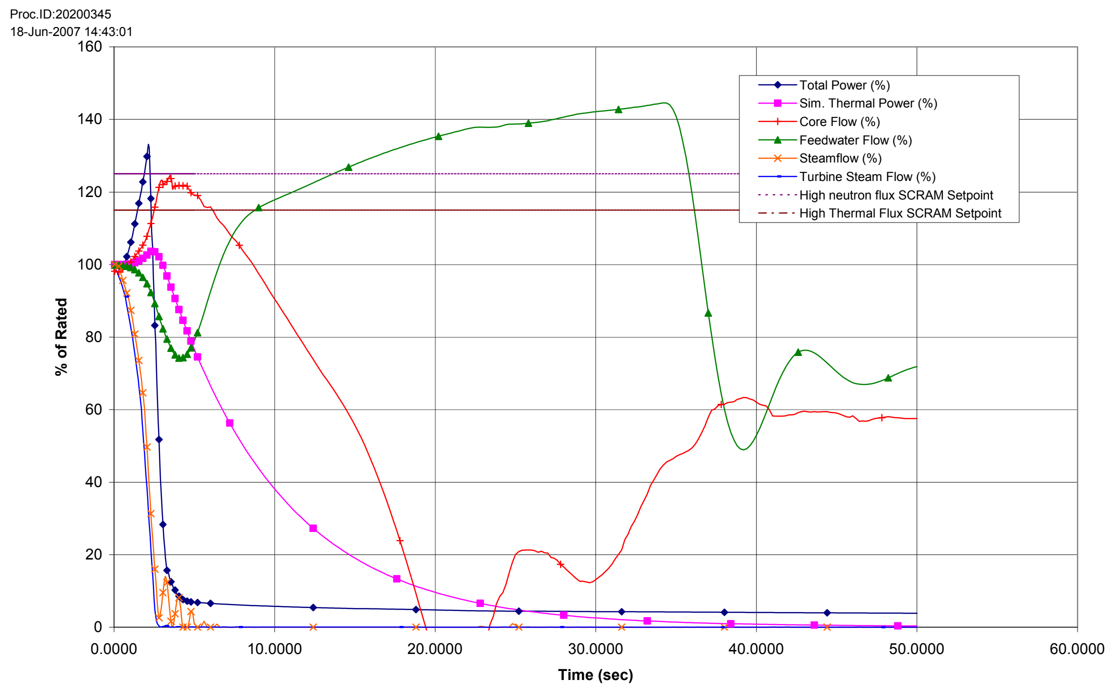



**Figure 2.4-4a. Pressure Regulator Failure – Closure of All Turbine Control and Bypass Valves**

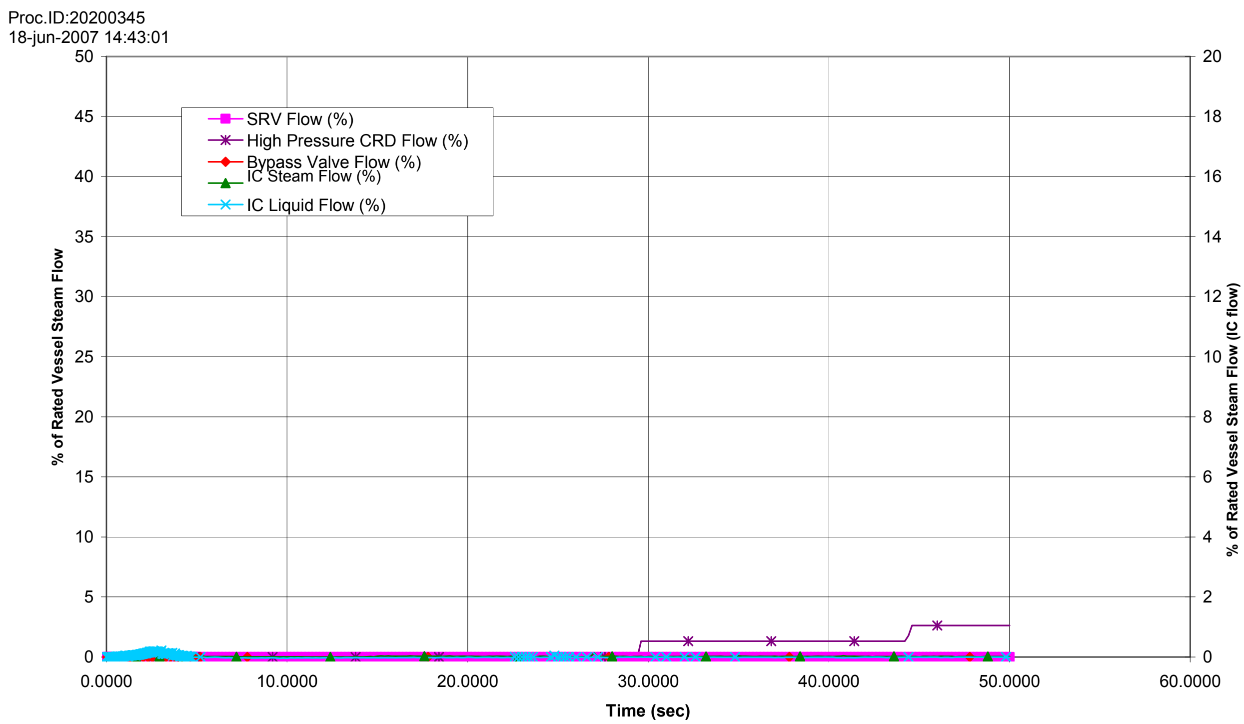



**Figure 2.4-4b. Pressure Regulator Failure – Closure of All Turbine Control and Bypass Valves**

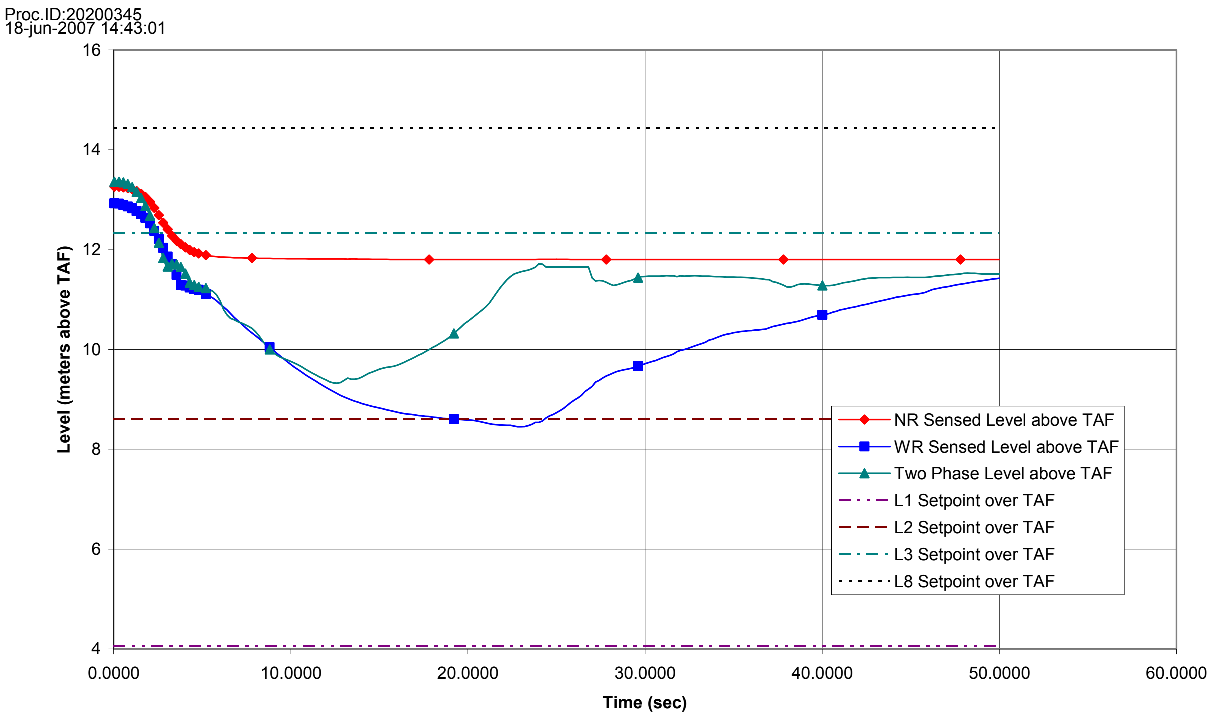



**Figure 2.4-4c. Pressure Regulator Failure – Closure of All Turbine Control and Bypass Valves**

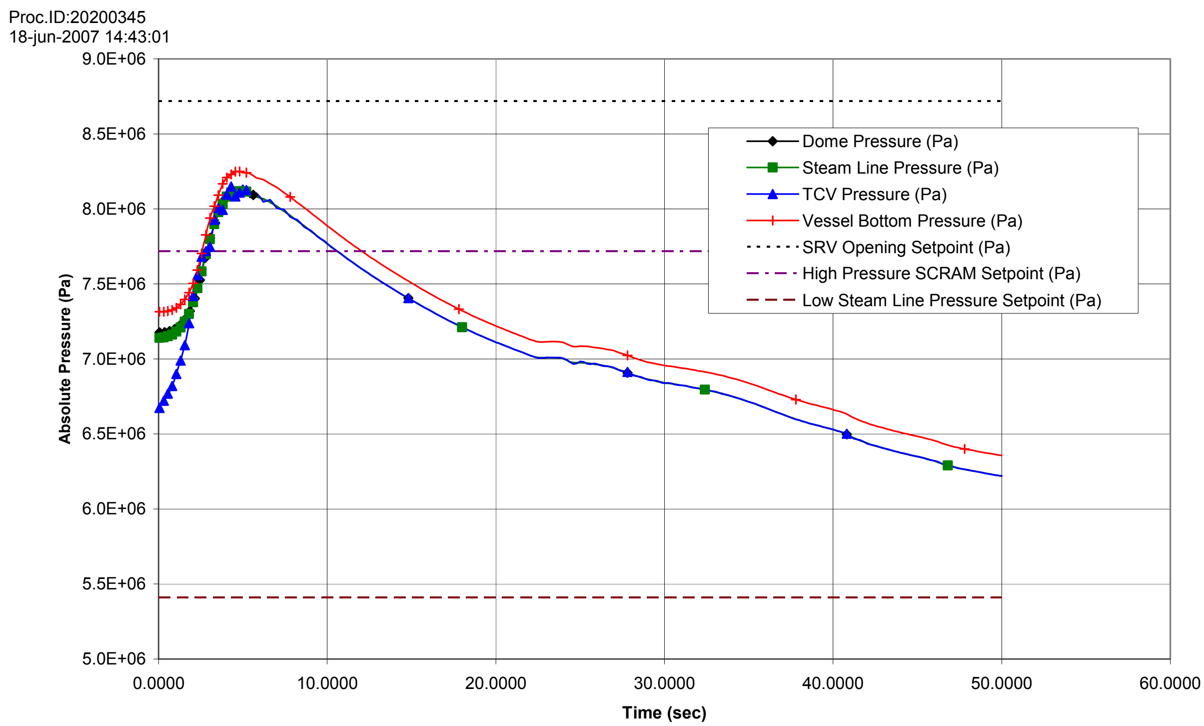



**Figure 2.4-4d. Pressure Regulator Failure – Closure of All Turbine Control and Bypass Valves**

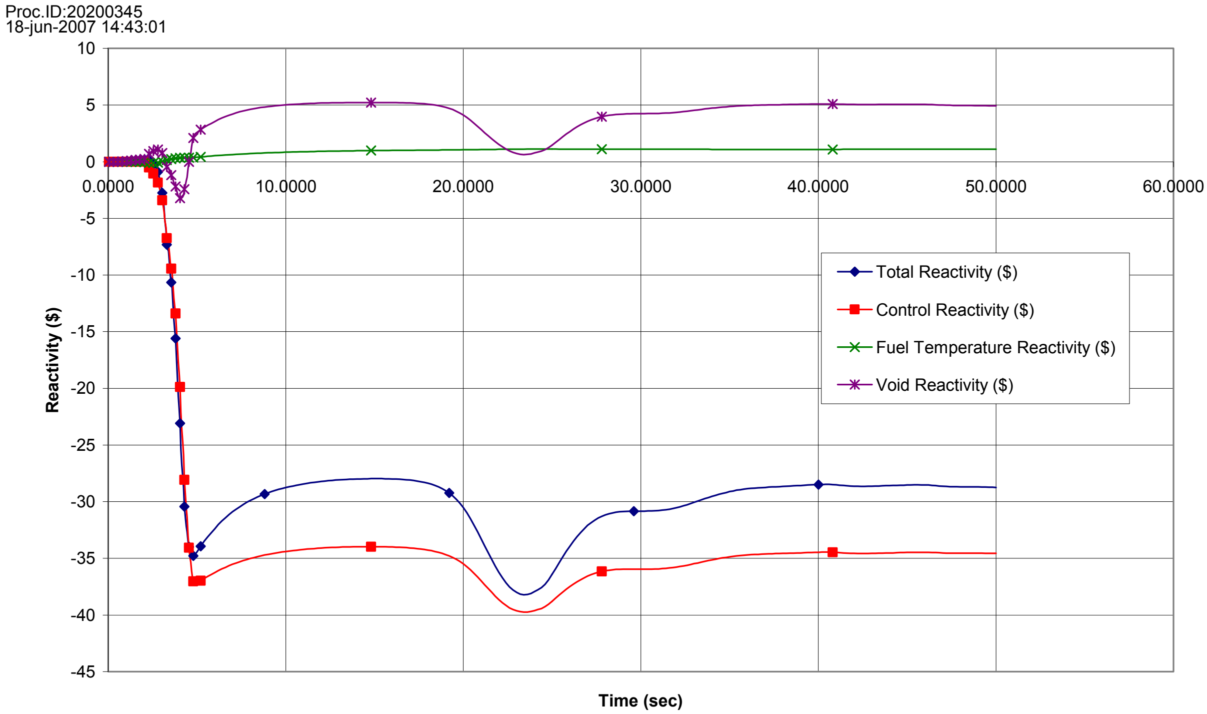



**Figure 2.4-4e. Pressure Regulator Failure – Closure of All Turbine Control and Bypass Valves**

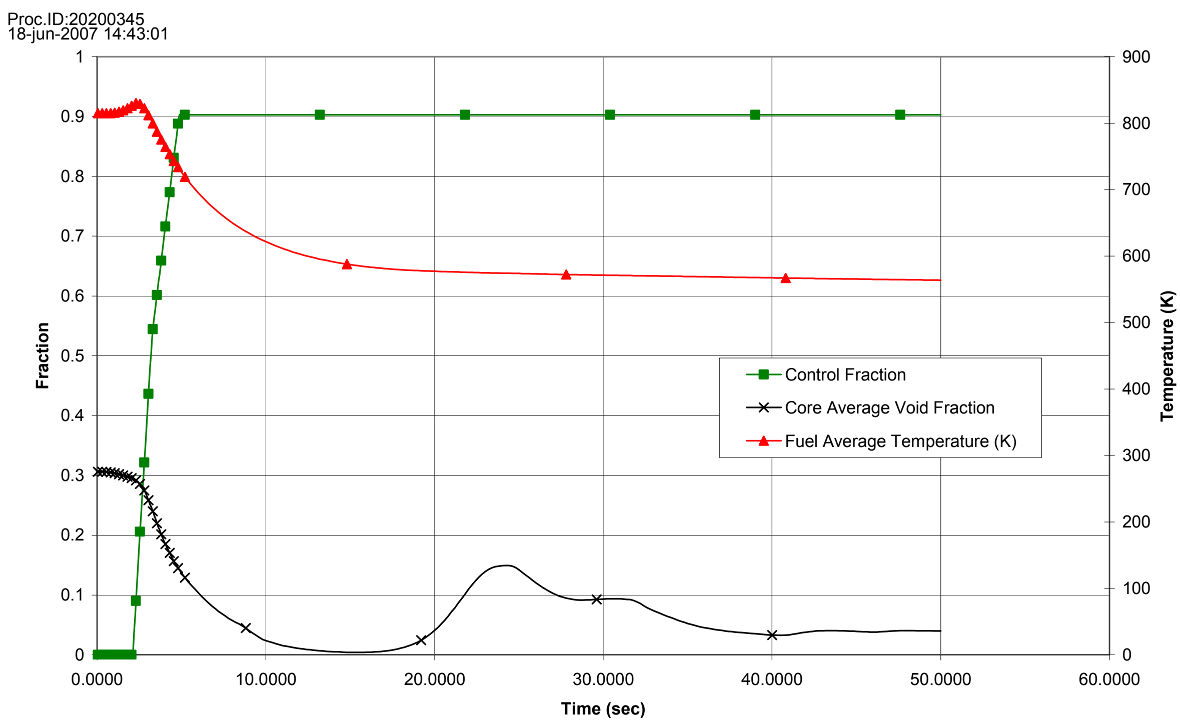



**Figure 2.4-4f. Pressure Regulator Failure – Closure of All Turbine Control and Bypass Valves**

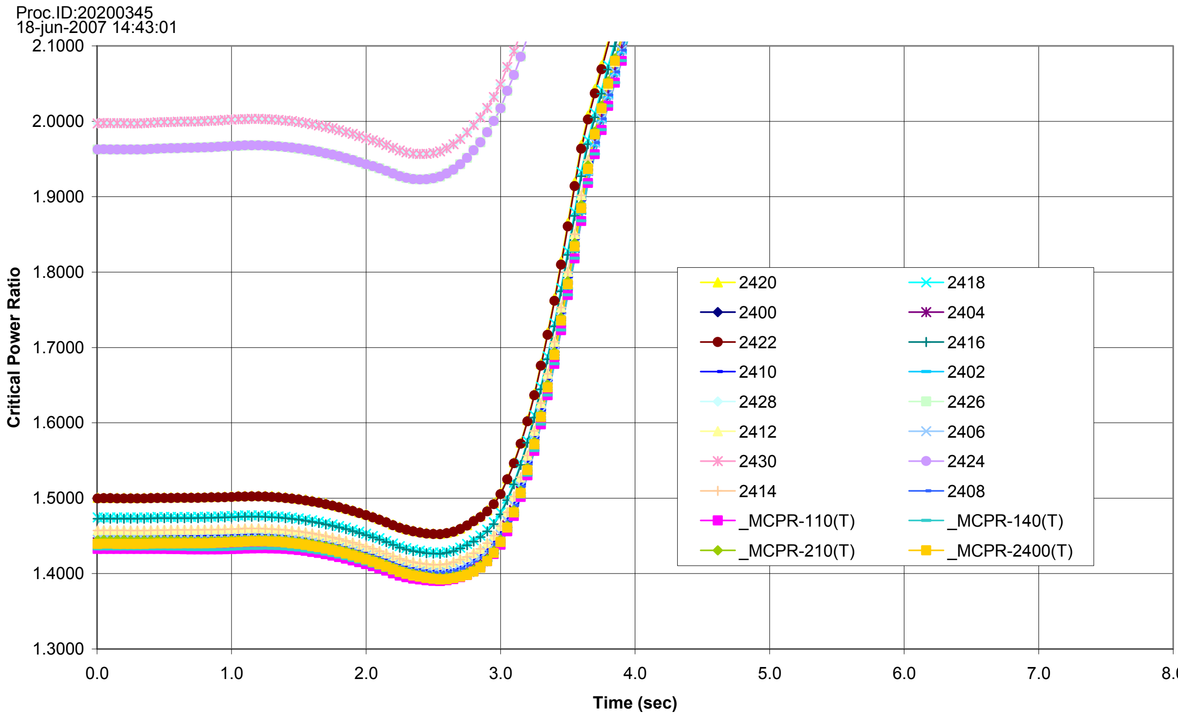



**Figure 2.4-4g. Pressure Regulator Failure – Closure of All Turbine Control and Bypass Valves**

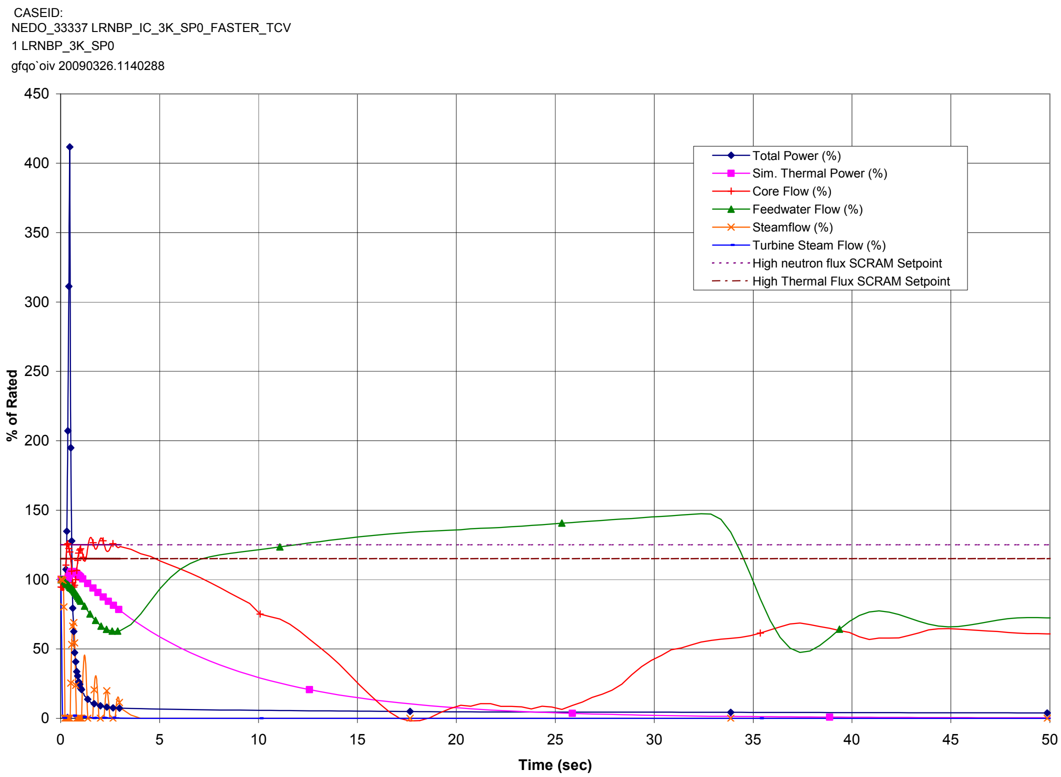



**Figure 2.4-5a. Generator Load Rejection With Total Turbine Bypass Failure**

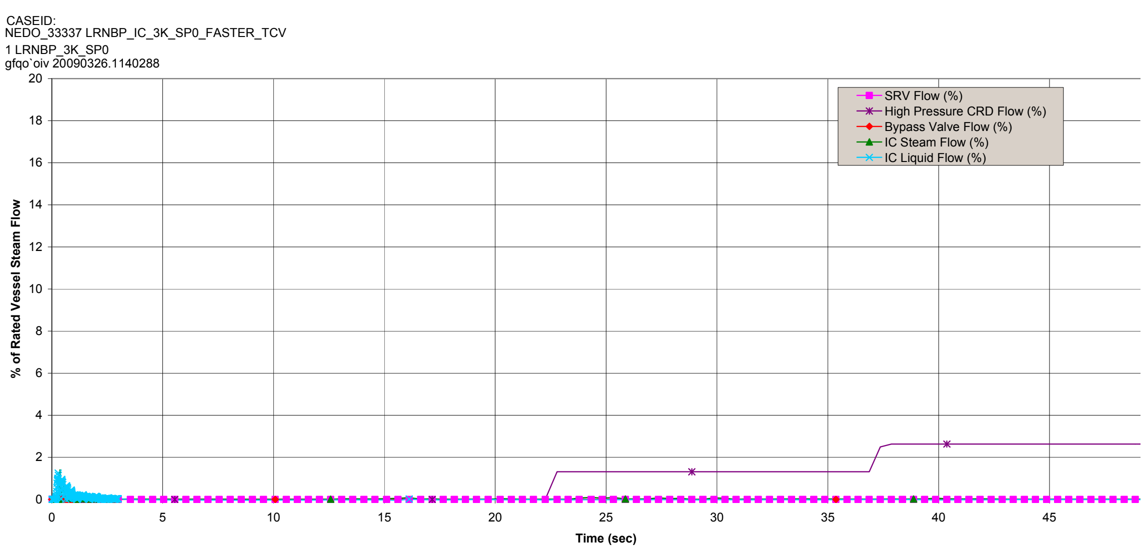



**Figure 2.4-5b. Generator Load Rejection With Total Turbine Bypass Failure**

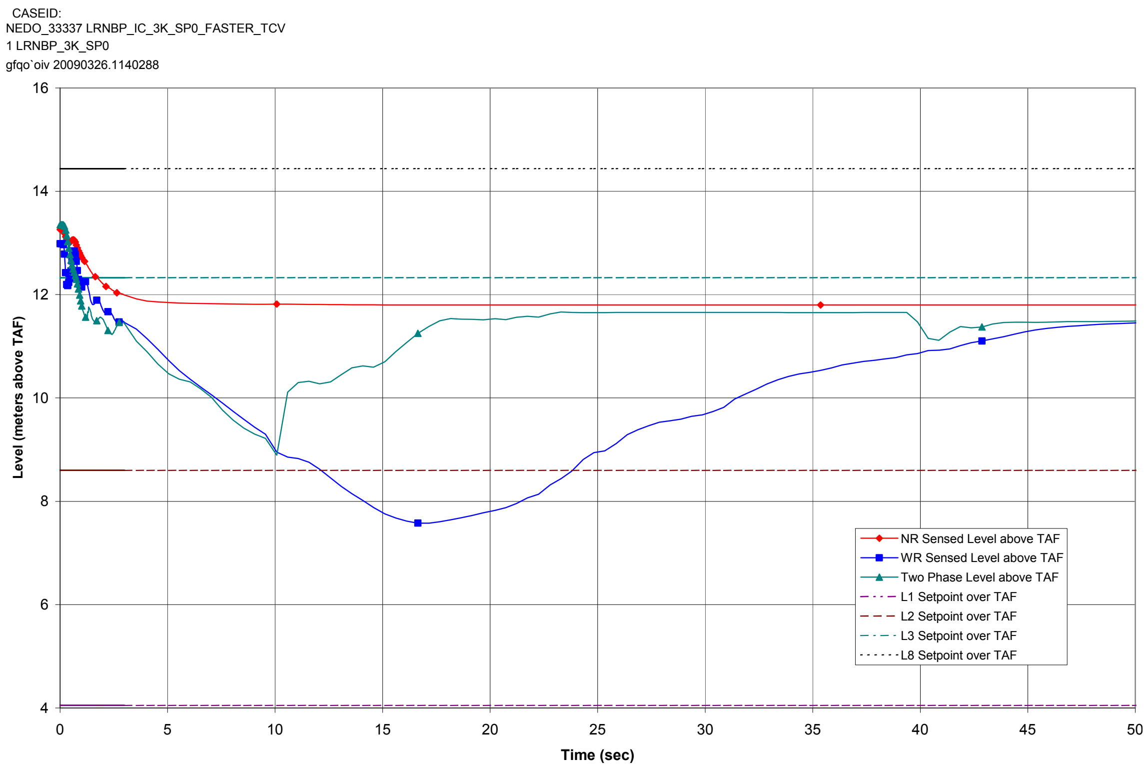



**Figure 2.4-5c. Generator Load Rejection With Total Turbine Bypass Failure**

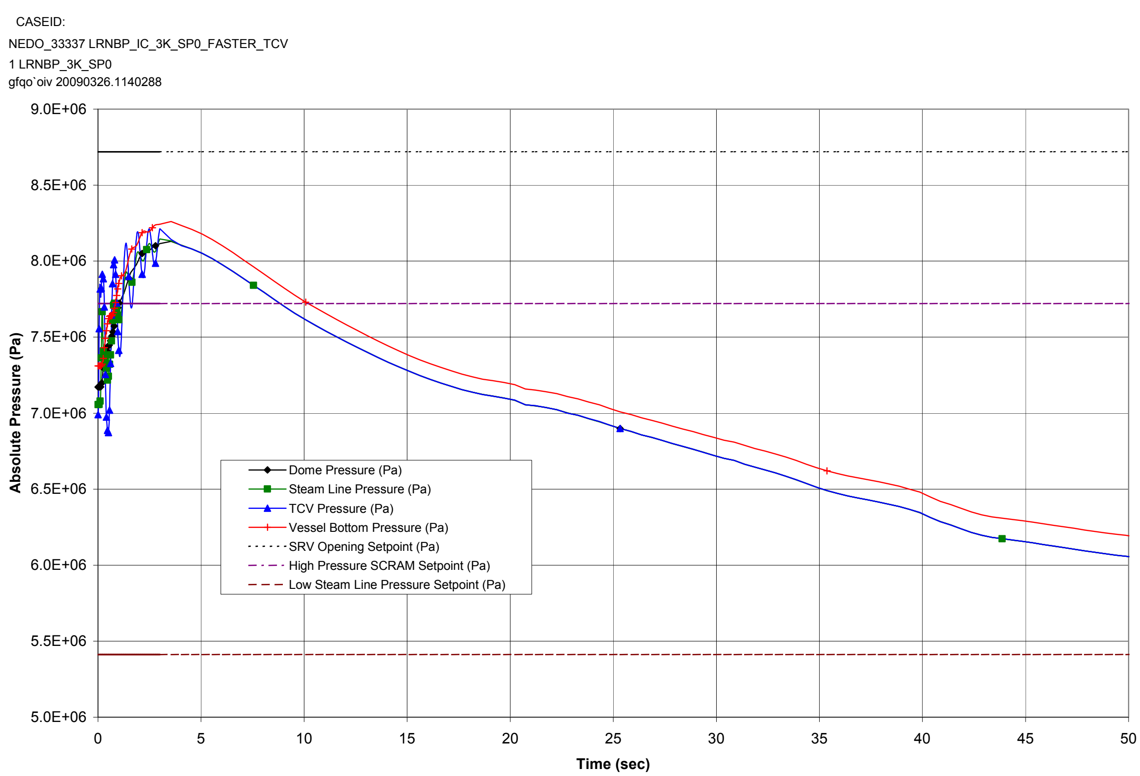



**Figure 2.4-5d. Generator Load Rejection With Total Turbine Bypass Failure**

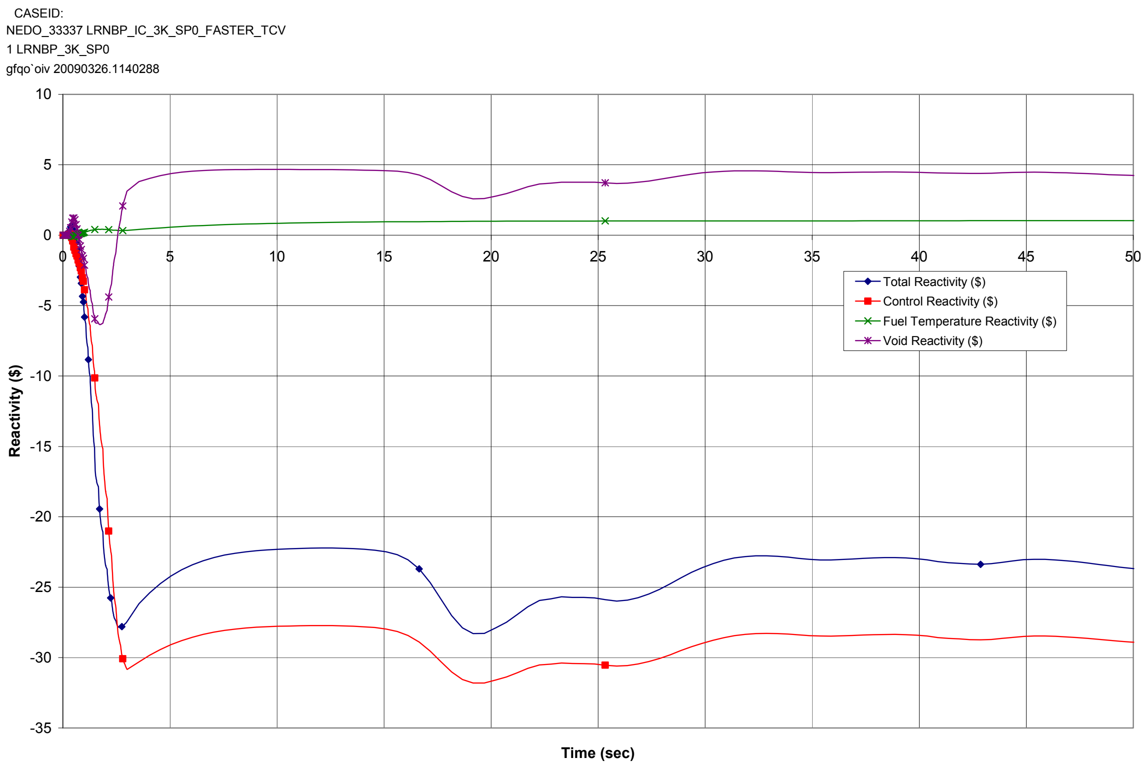



**Figure 2.4-5e. Generator Load Rejection With Total Turbine Bypass Failure**

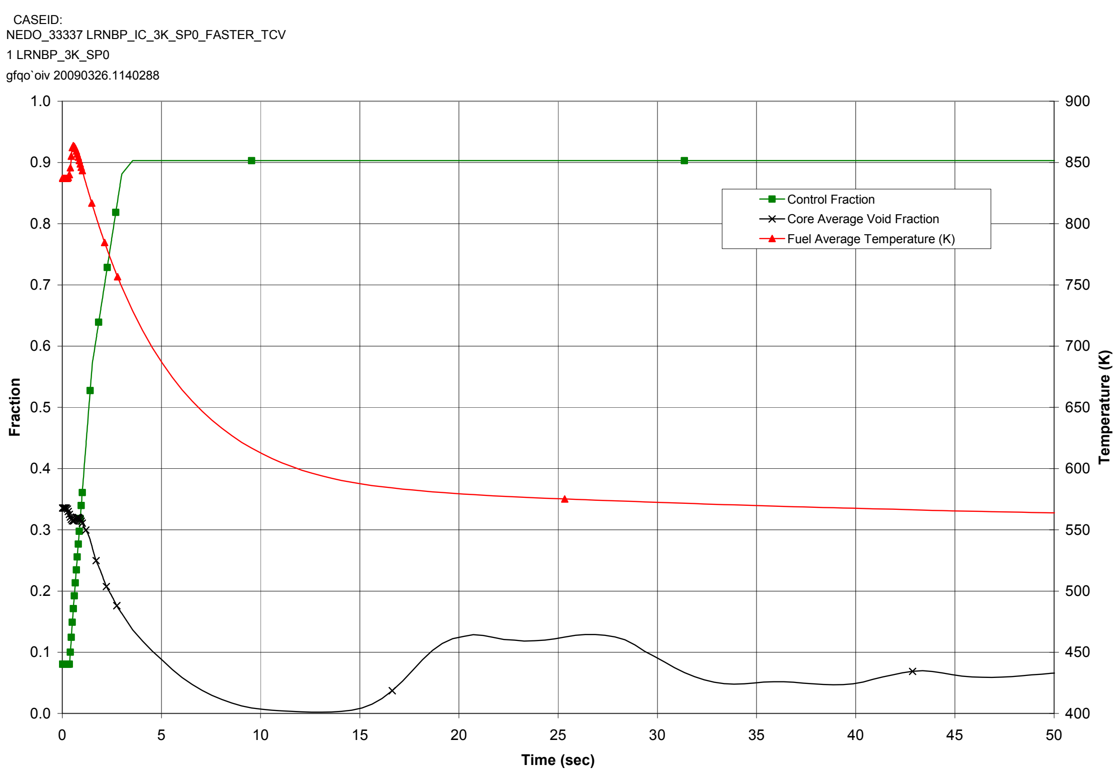



**Figure 2.4-5f. Generator Load Rejection With Total Turbine Bypass Failure**

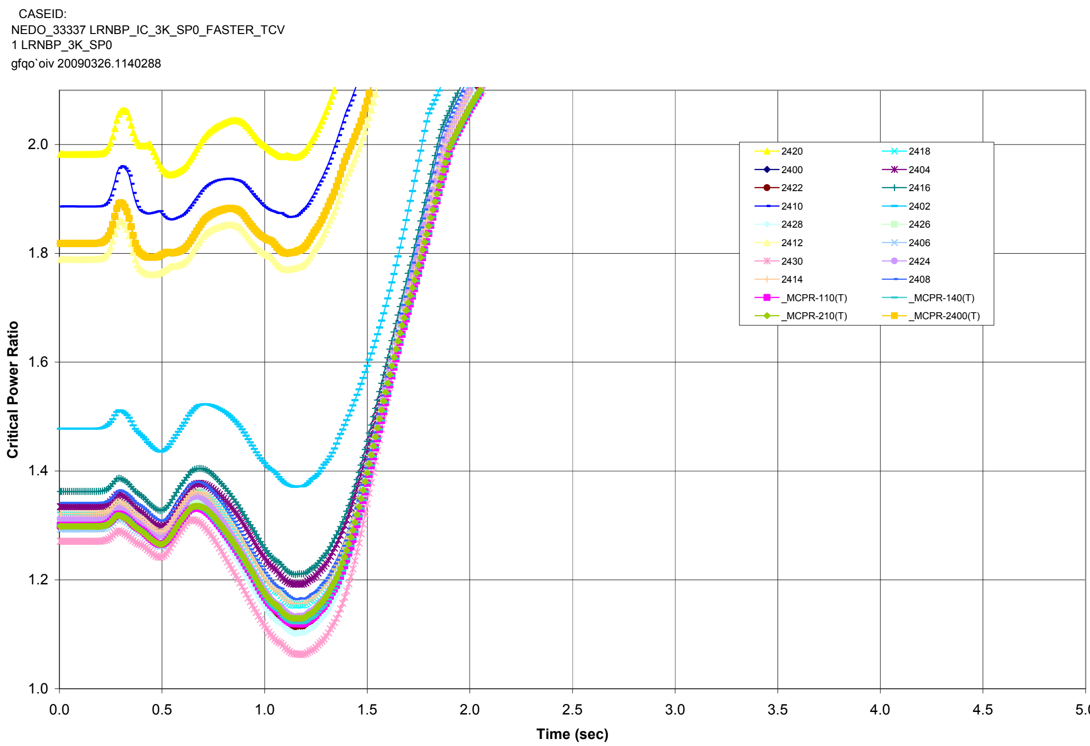



**Figure 2.4-5g. Generator Load Rejection With Total Turbine Bypass Failure**

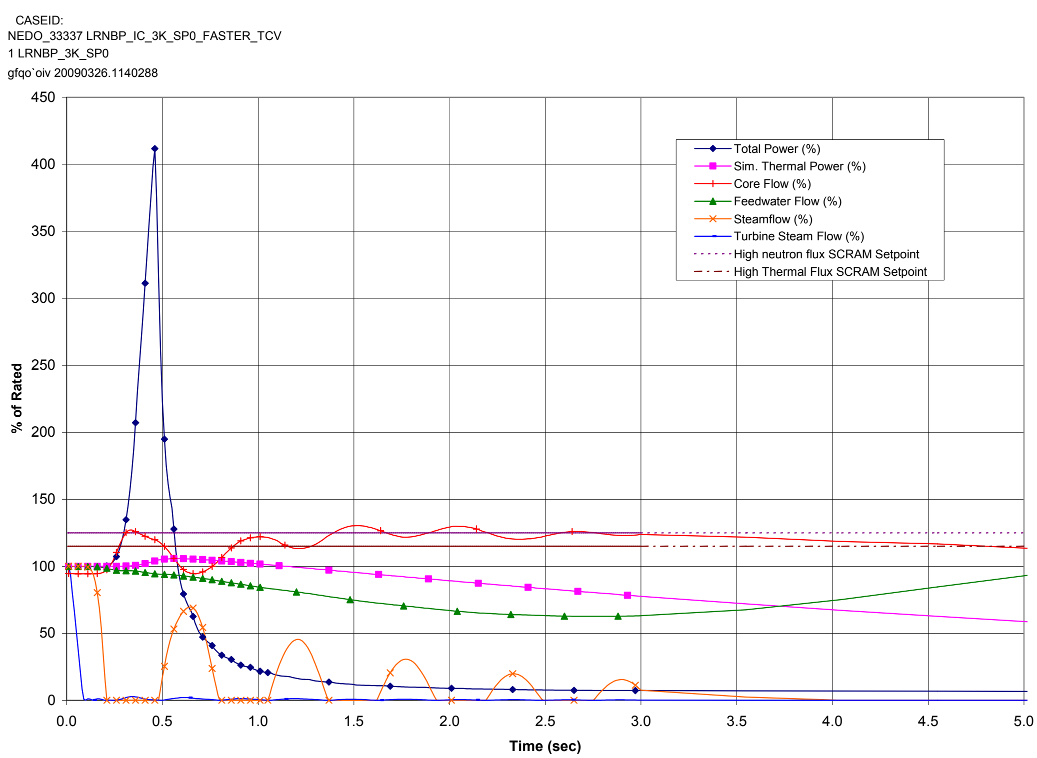



**Figure 2.4-5h. Generator Load Rejection With Total Turbine Bypass Failure (Figure 2.4-5a from 0 to 5 s)**

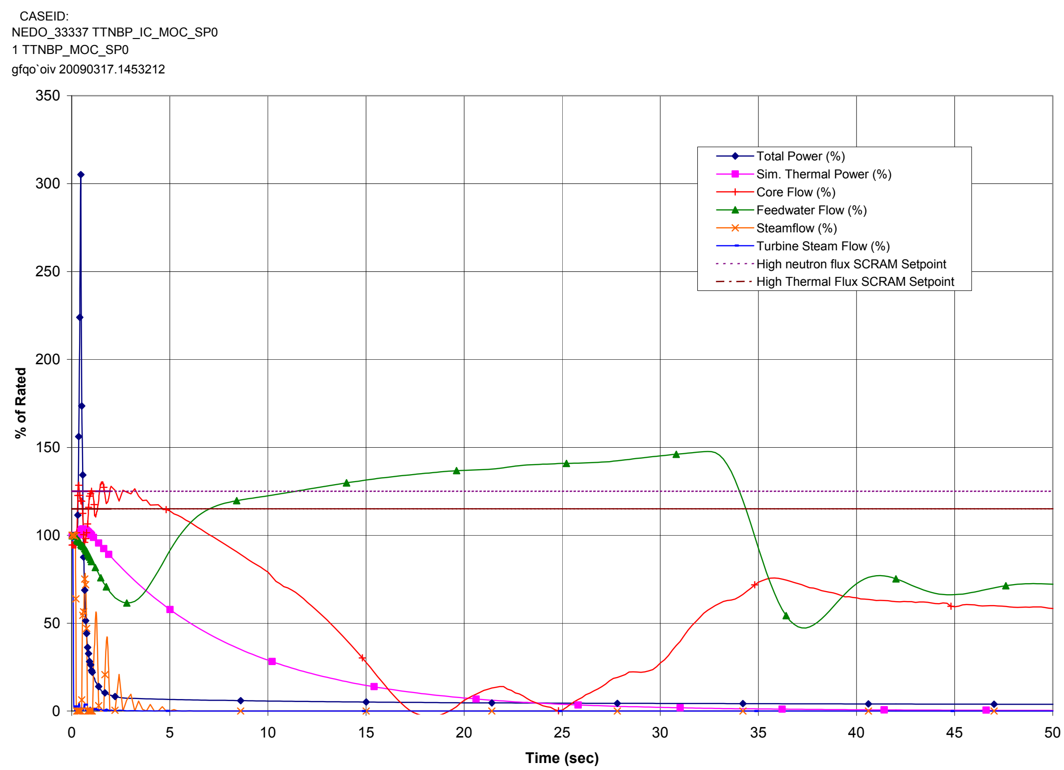



**Figure 2.4-6a. Turbine Trip With Total Turbine Bypass Failure**

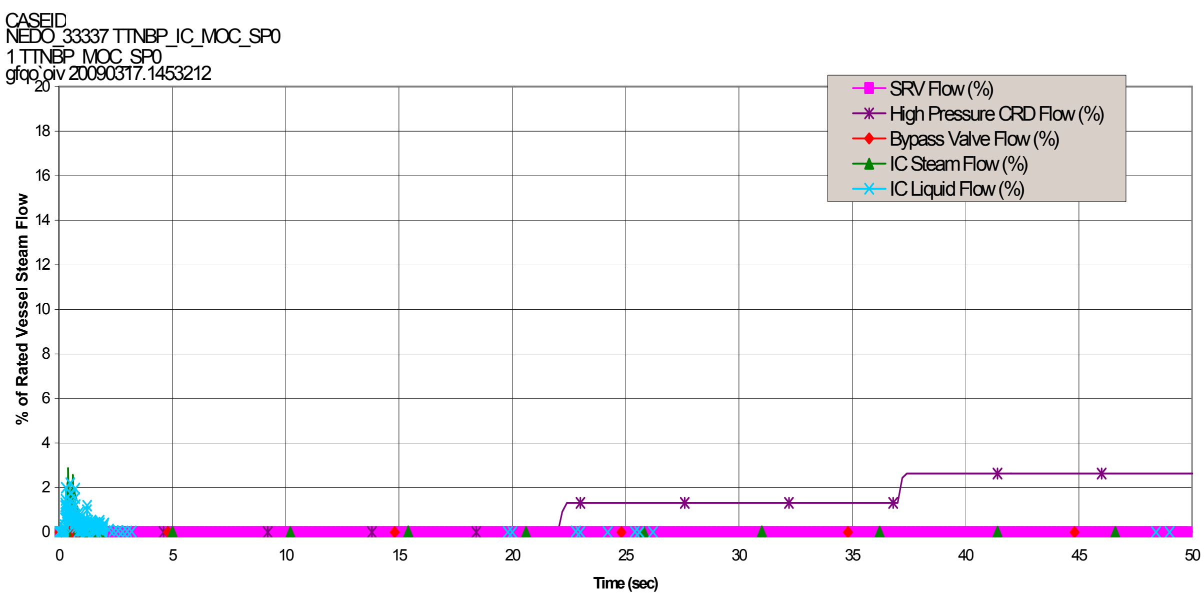



**Figure 2.4-6b. Turbine Trip With Total Turbine Bypass Failure**

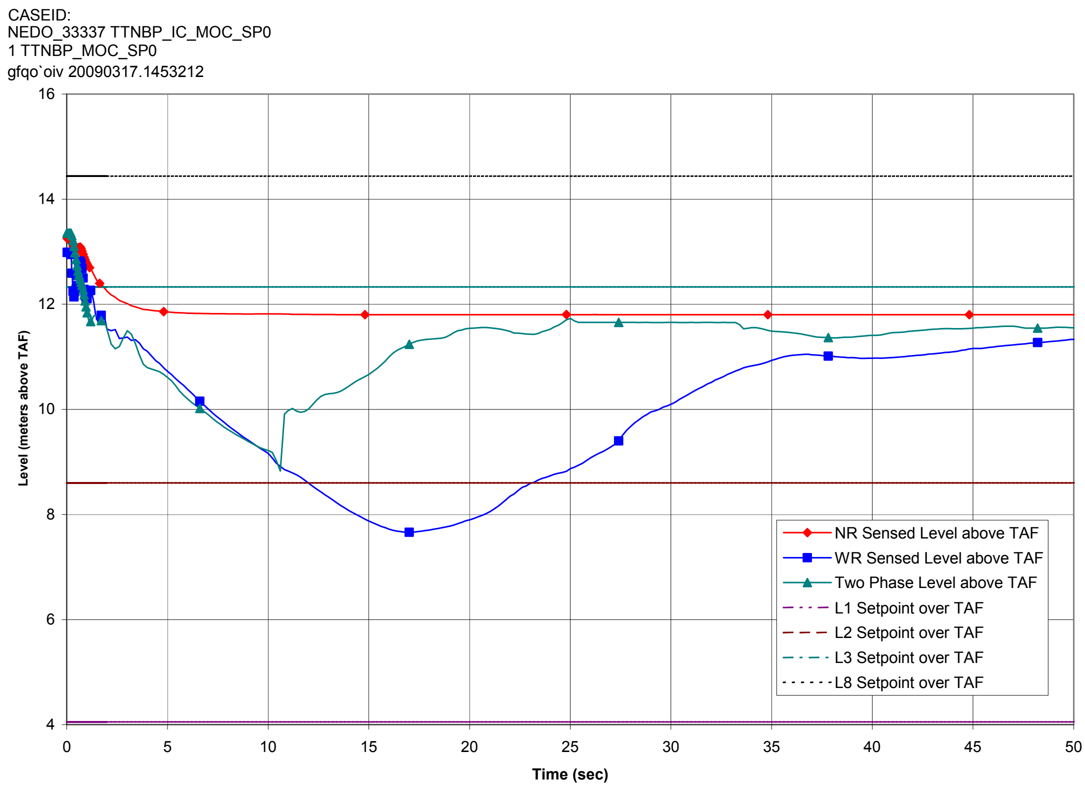



**Figure 2.4-6c. Turbine Trip With Total Turbine Bypass Failure**

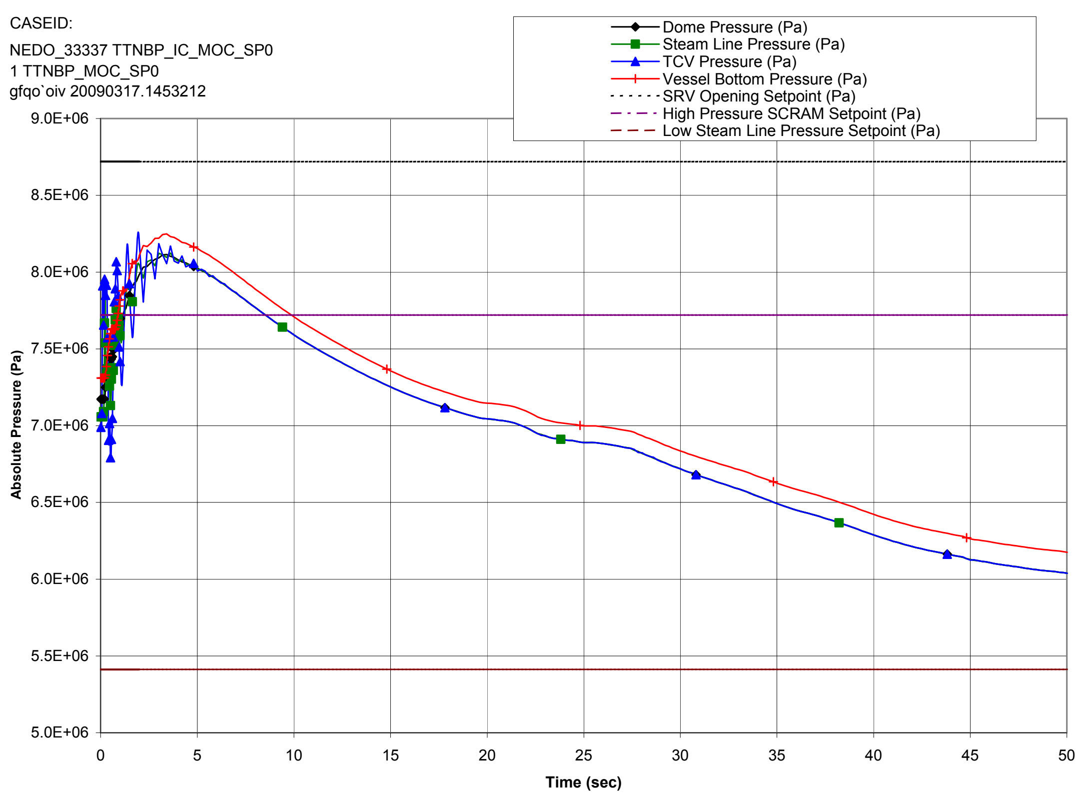



**Figure 2.4-6d. Turbine Trip With Total Turbine Bypass Failure**

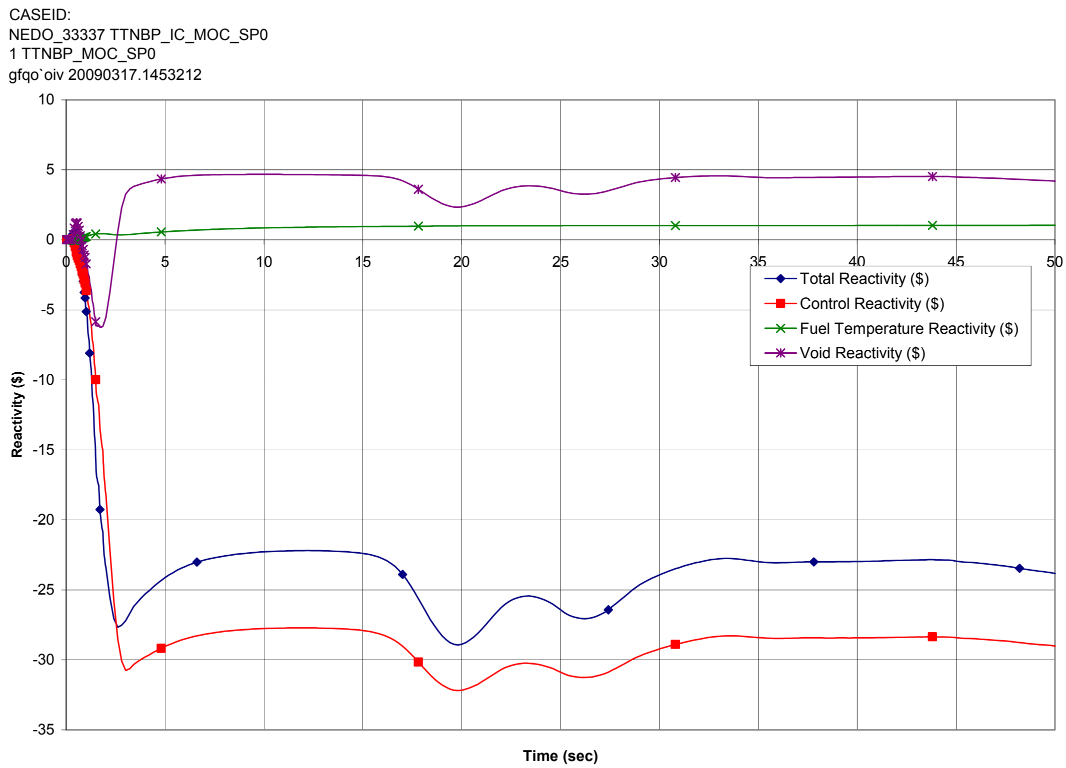



**Figure 2.4-6e. Turbine Trip With Total Turbine Bypass Failure**

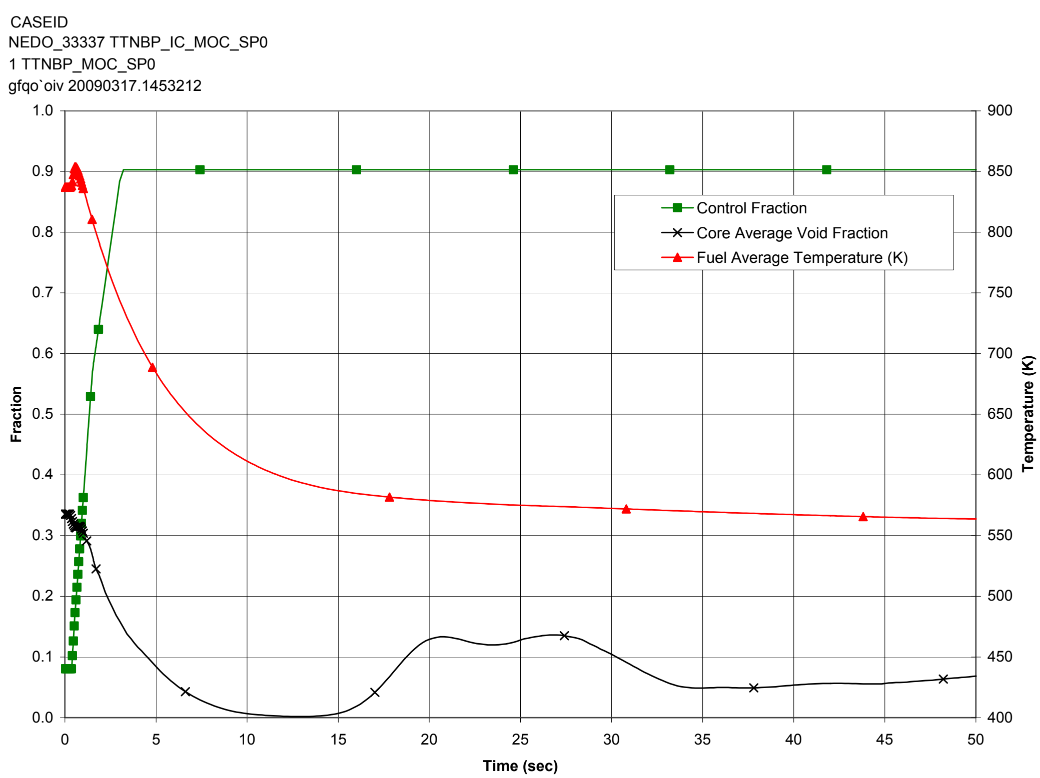



**Figure 2.4-6f. Turbine Trip With Total Turbine Bypass Failure**

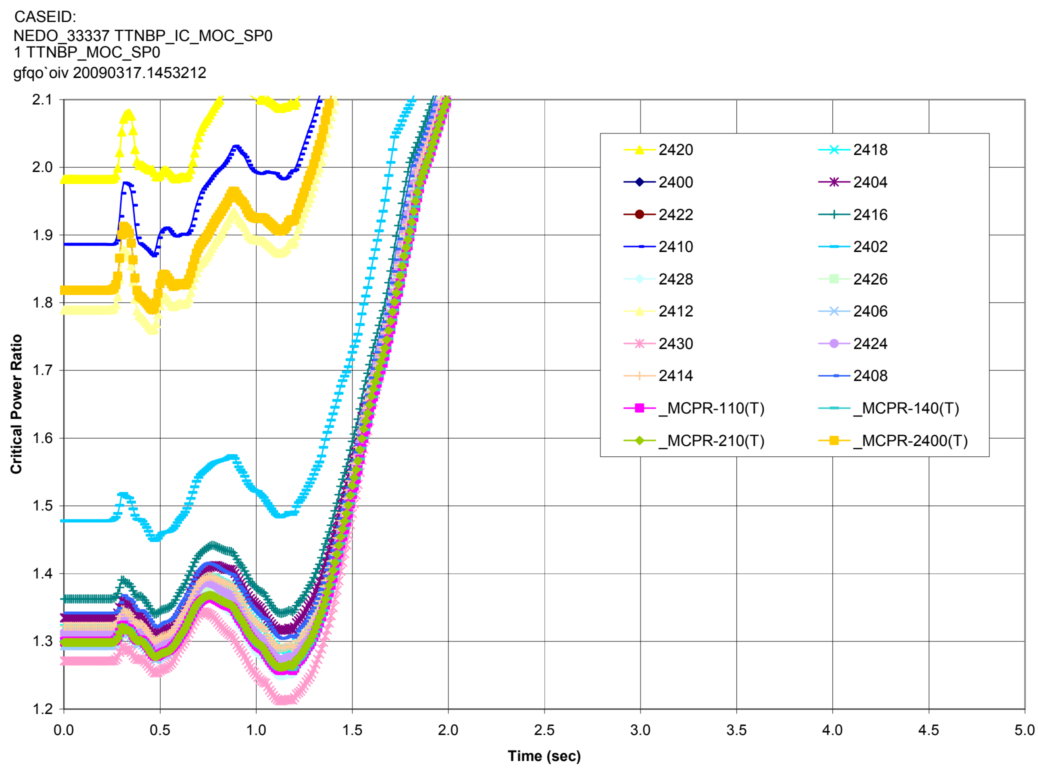



**Figure 2.4-6g. Turbine Trip With Total Turbine Bypass Failure**

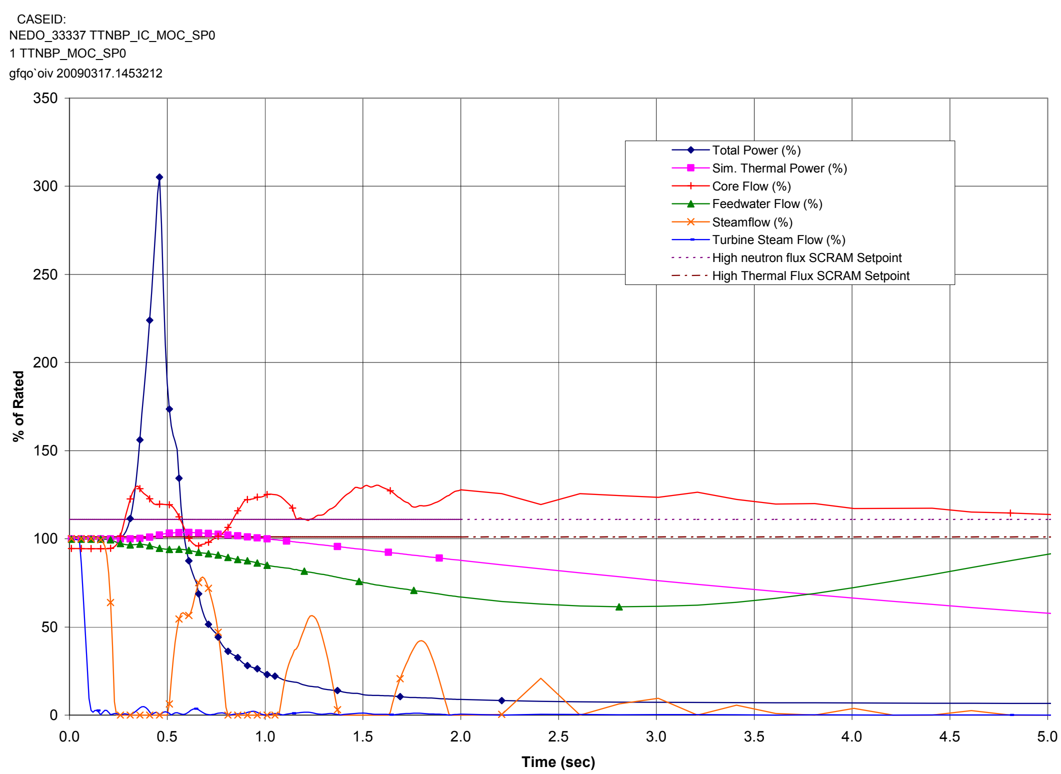



**Figure 2.4-6h. Turbine Trip with Total Bypass Failure (Figure 2.4-6a from 0 to 5 s)**

**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.

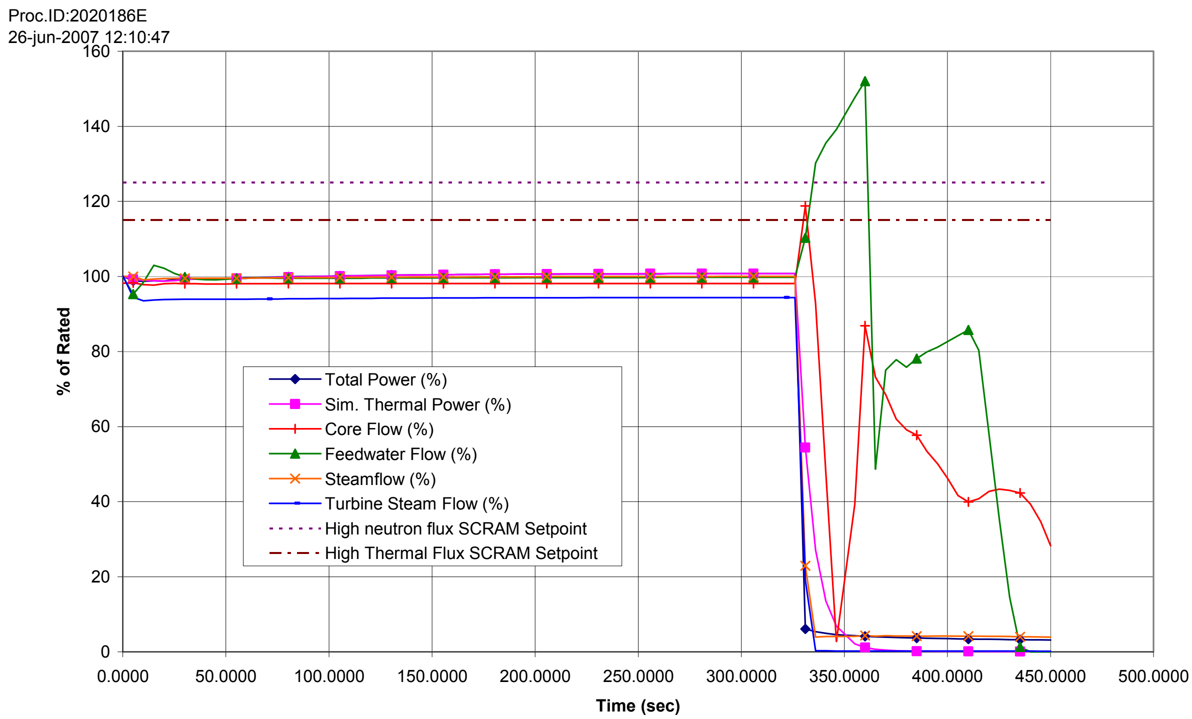



**Figure 2.4-8a. Inadvertent SRV opening**

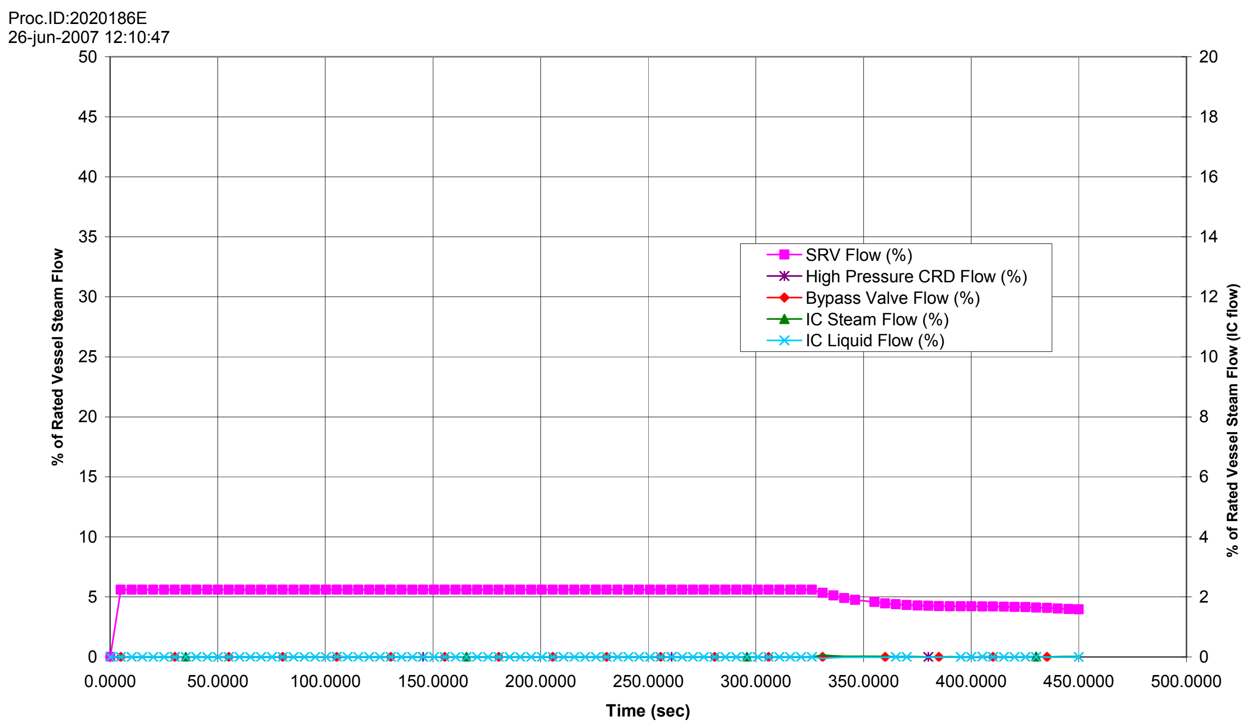



**Figure 2.4-8b. Inadvertent SRV opening**

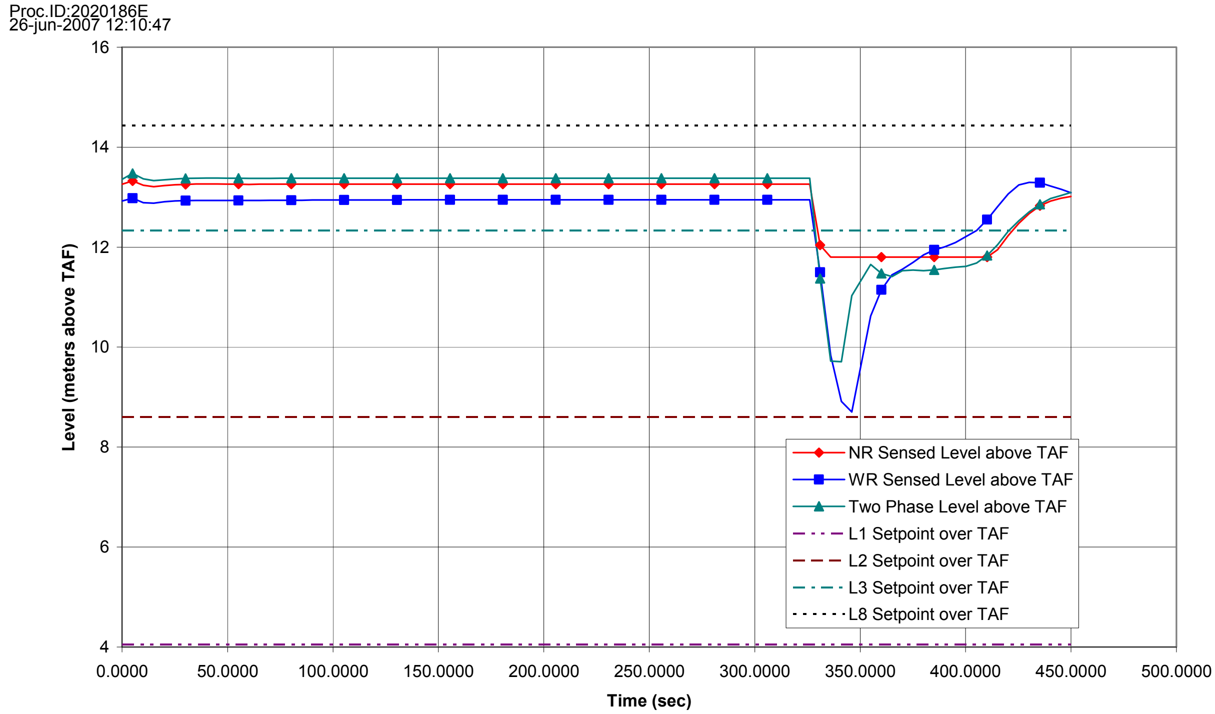



**Figure 2.4-8c. Inadvertent SRV opening**

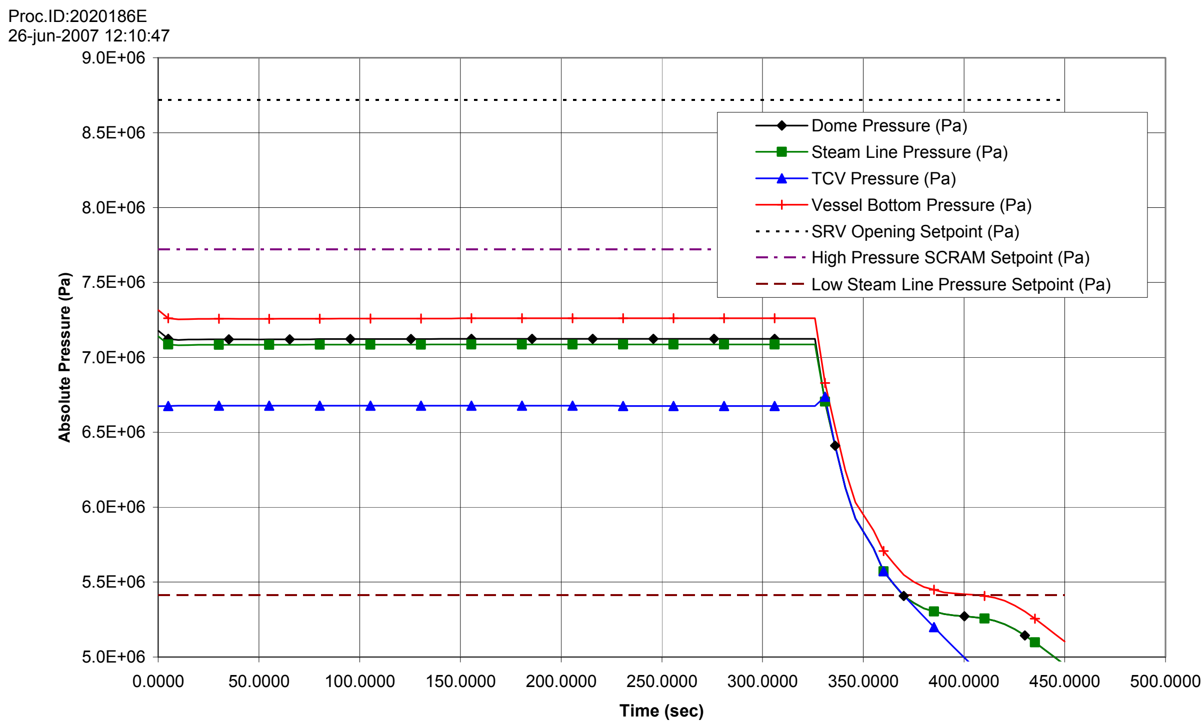



**Figure 2.4-8d. Inadvertent SRV opening**

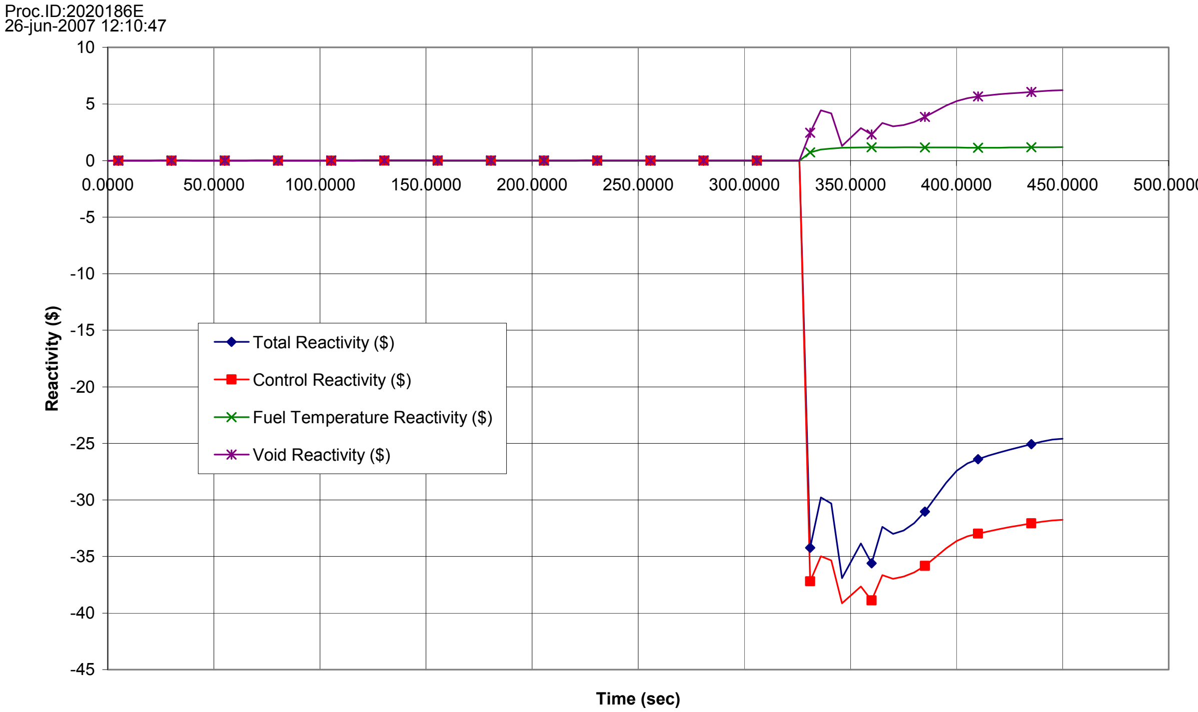



**Figure 2.4-8e. Inadvertent SRV opening**

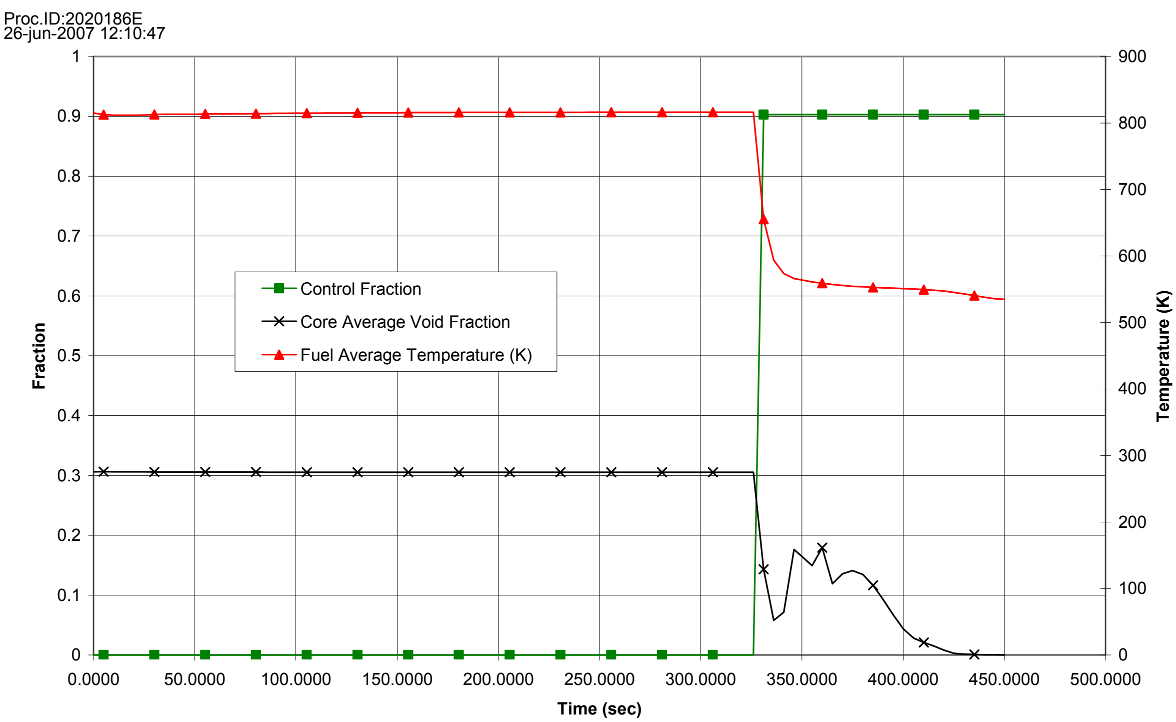



**Figure 2.4-8f. Inadvertent SRV opening**

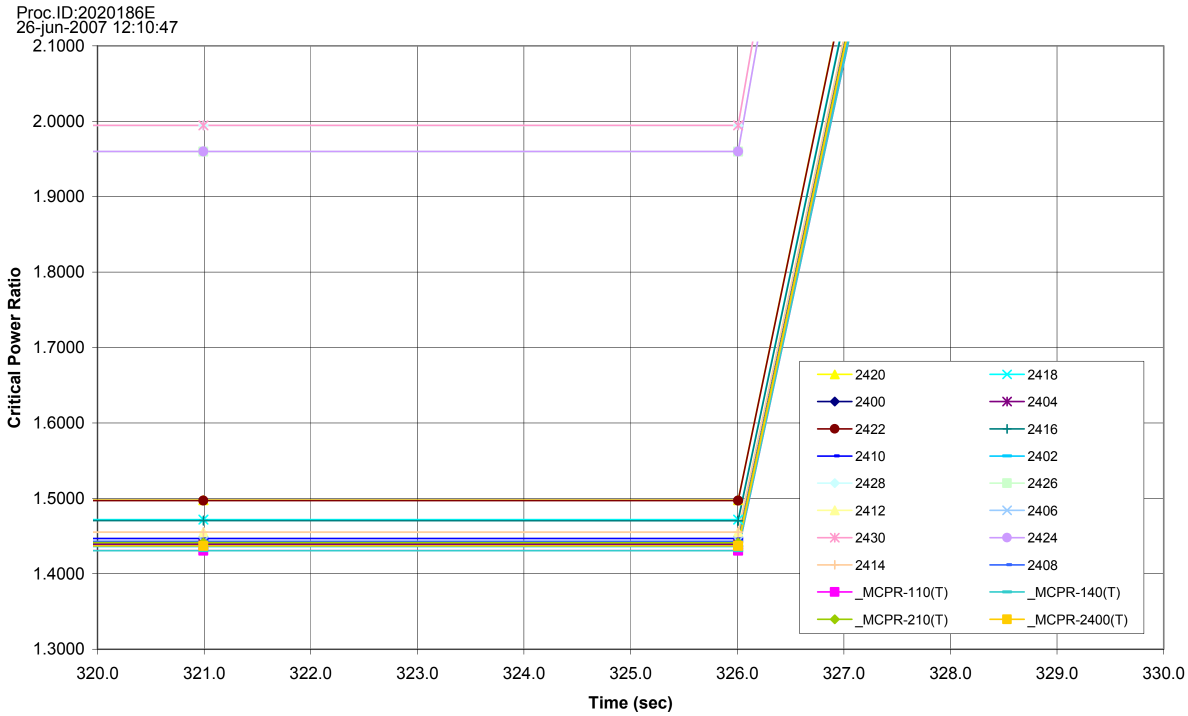



**Figure 2.4-8g. Inadvertent SRV opening**

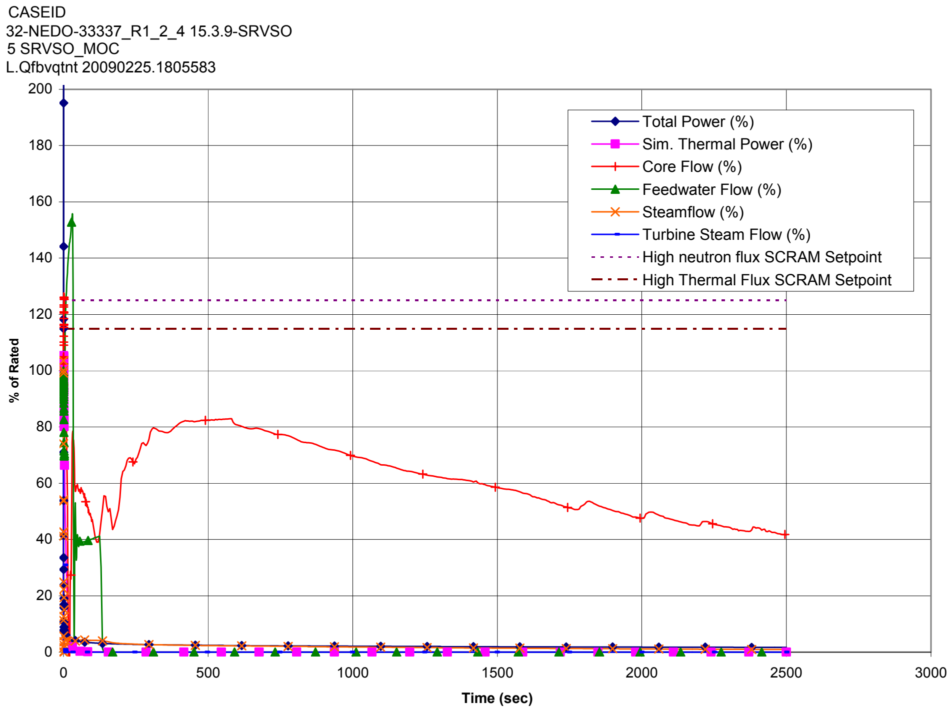



**Figure 2.4-9a. Stuck Open Safety Relief Valve**

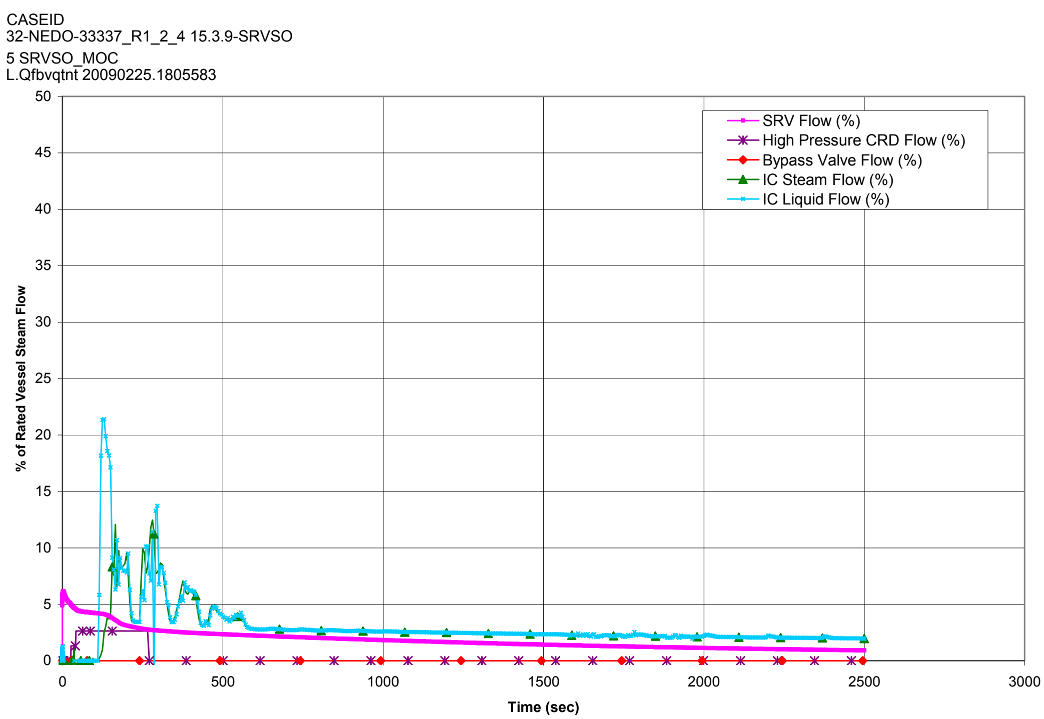



**Figure 2.4-9b. Stuck Open Safety Relief Valve**

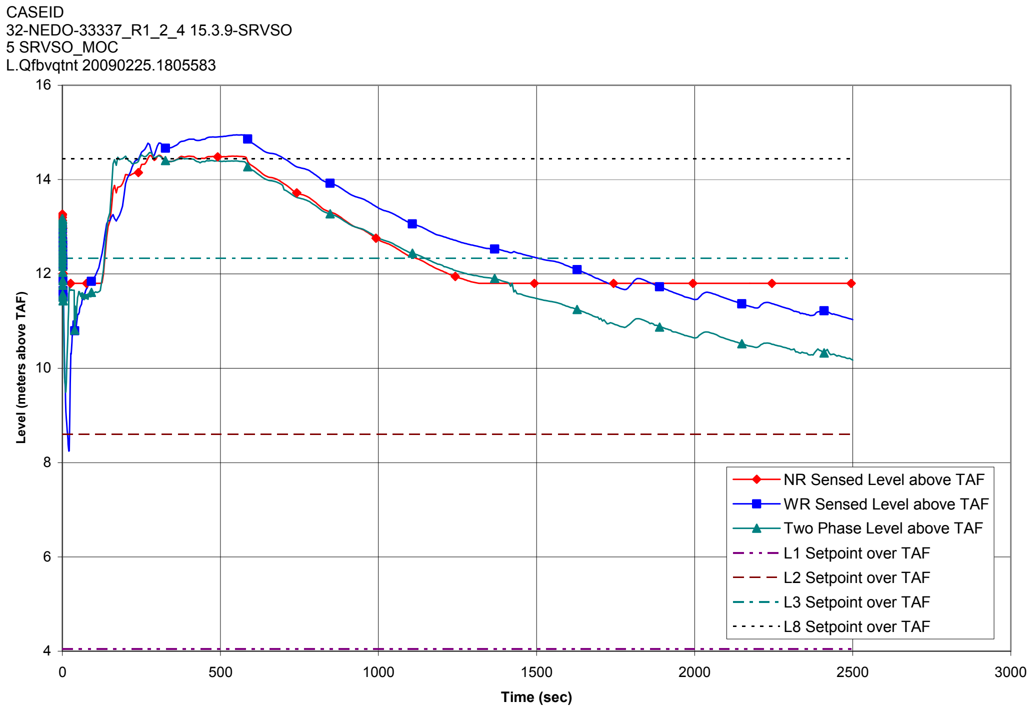



**Figure 2.4-9c. Stuck Open Safety Relief Valve**

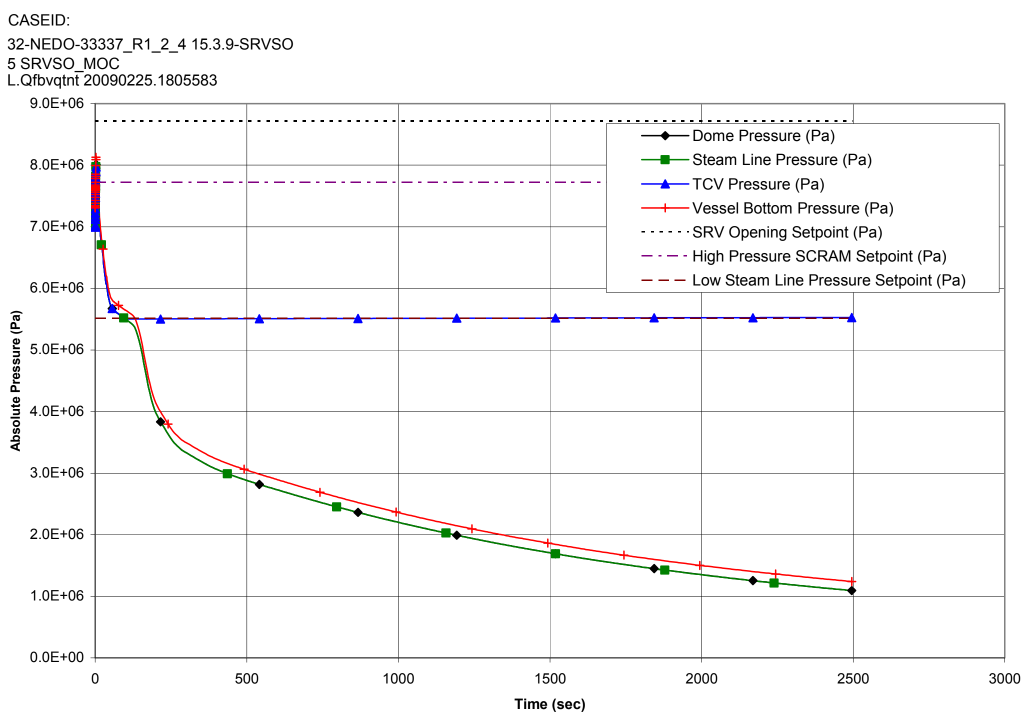



**Figure 2.4-9d. Stuck Open Safety Relief Valve**

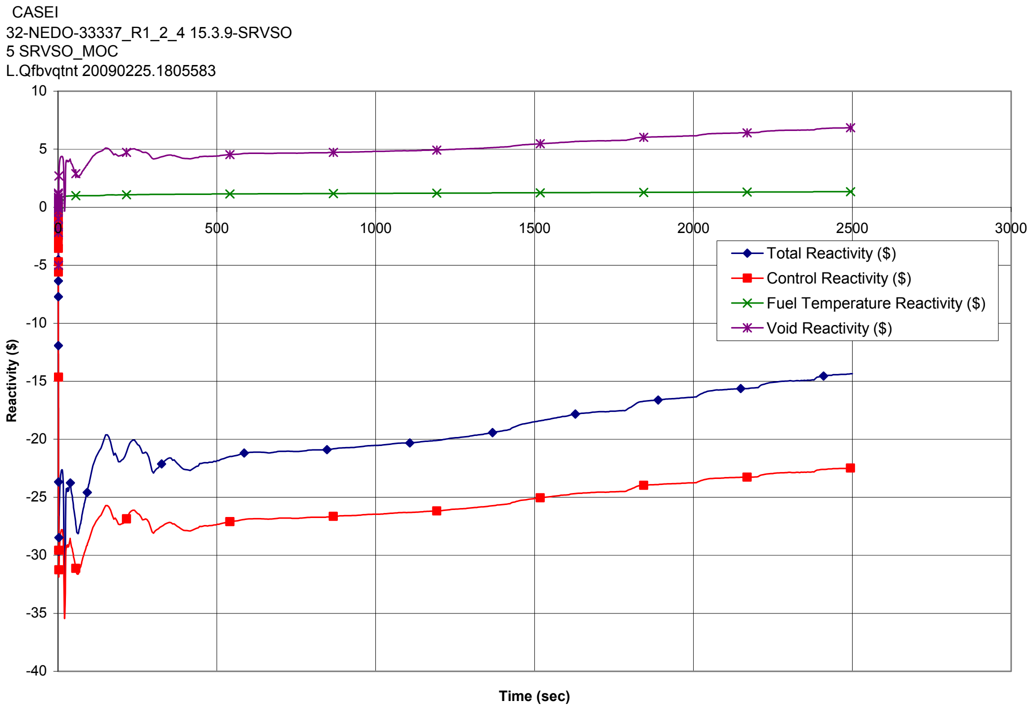



**Figure 2.4-9e. Stuck Open Safety Relief Valve**

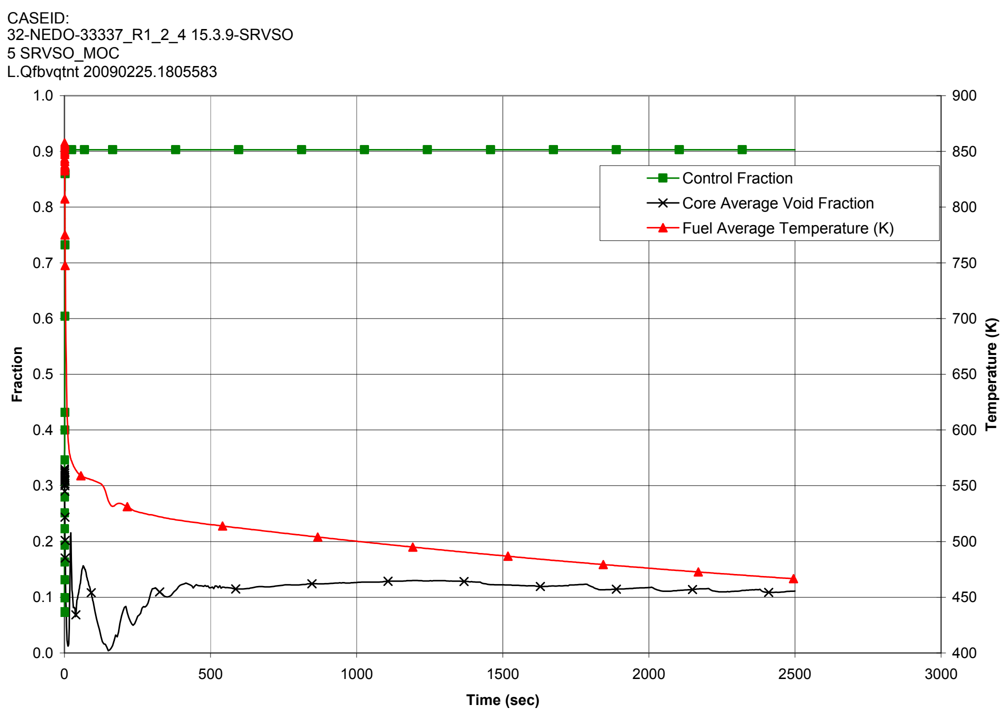



**Figure 2.4-9f. Stuck Open Safety Relief Valve**

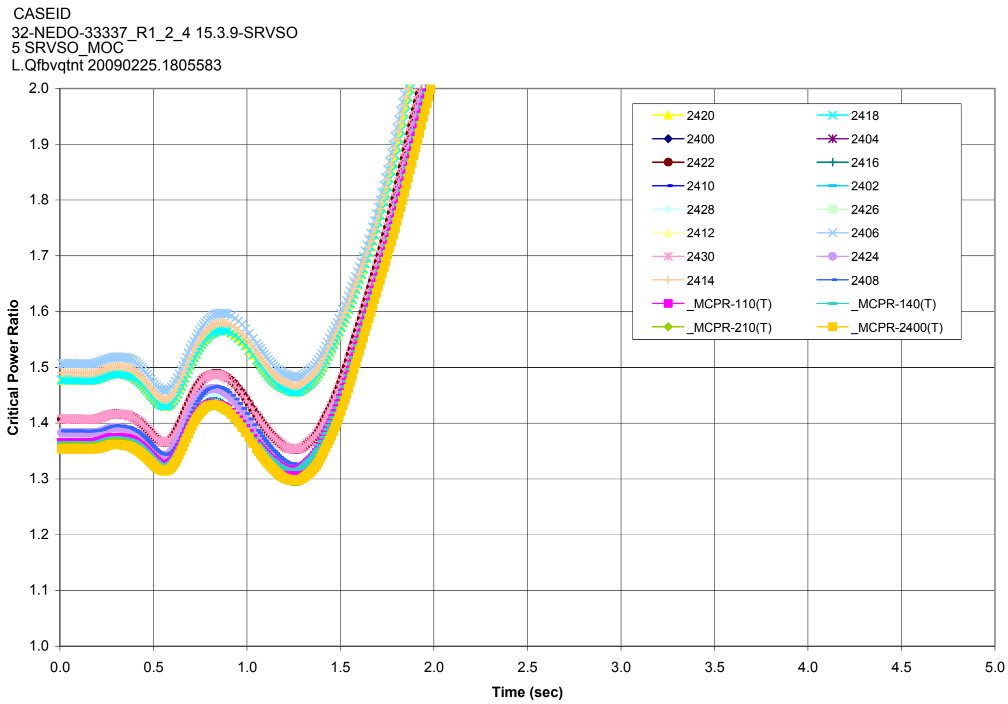



**Figure 2.4-9g. Stuck Open Safety Relief Valve**

## 2.5 ANALYSIS OF SPECIAL EVENTS

### 2.5.1 Overpressure Protection

This Subsection is equivalent to the DCD, Tier 2, Subsection 15.5.1 (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 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 = 2494 kg/s (19.79 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 FWCS 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 in 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 with bounding steamline inputs and a single SRV 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**

SRV capacities are based on establishing an adequate margin from the peak vessel bottom pressure to the vessel code limit in response to pressurization events.

The analysis method assumes that whenever the system pressure increases to the valve mechanical lift set pressure of a valve, the valve begins opening and reaches full open at 103% of set pressure. Only one SRV is required to open to prevent exceeding the ASME limit in the ASME overpressure protection event. Ten SRVs and eight SVs are included in the ESBWR design. The additional SRVs and SVs are used to mitigate the ATWS event.

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 2.5.1-4. The peak vessel bottom pressure calculated is below the acceptance limit of 9.481 MPa gauge (1375 psig). The pressurization is not dynamic and does not significantly overshoot the relief valve setpoint, therefore as demonstrated a single SRV is able to cope with the pressure increase. 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 discussed in Subsection 15.5.1 of Reference 2.5-4 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.

### 2.5.2 Shutdown Without Control Rods (Standby 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 ATWS occurrence 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 isolation condenser 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 bottom pressure, peak clad temperature, peak suppression pool temperature and the peak containment pressure 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 Distributed Control and Information System [DCIS]) 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 feedwater runback 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 1.16 to conservatively bound the Peak Cladding Temperature (PCT), and which 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 coastdown time of 15 seconds with an additional delay in the analysis of 10 seconds for the feedwater runback 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

- The rest of the safety systems are not credited or they do not actuate during the calculated sequence of events.

- The passive ICS is automatically initiated upon the loss of feedwater pump power buses at 3s, to remove decay heat following scram and isolation and isolation condenser 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 2,000 s 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 13.5 m (44.4 ft) above vessel zero, which provides 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 ICS. Therefore, the requirement for achieving and maintaining hot shutdown condition is met.

With operation of ICS, 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, Revision1, Class III, (Proprietary), February 2008.

2.5-3 Global Nuclear Fuel, “ESBWR Initial Core Nuclear Design Report”, NEDC-33326-P, Class III (Proprietary), Revision 1, March 2009, NEDO-33326, Class I (Non-proprietary), Revision 1, March 2009.

2.5-4 GE Nuclear Energy, “ESBWR Design Control Document” 26A6642.

**Table 2.5.1-1**

**Sequence of Events for MSIV Closure with Flux Scram**

| Time (s) | Event * |
|---|---|
| 0.0 | Closure of all MSIVs |
| 0.0 | FW trip |
| 0.8 | MSIV position < 85% |
| 1.7 | MSIV position < 1% |
| 1.8 | High neutron flux setpoint reached |
| 1.9 | Reactor SCRAM on high flux |
| 3.0 | L3 is reached |
| 3.0 | MSIVs are closed |
| 9.4 | L2 is reached |
| 50.6 | A single SRV opens |

* See Figure 2.5.1-4.

**Table 2.5.1-2**

**(Deleted)**

**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** kPaG (psig) |
|---|---|---|---|---|---|
| 10.34 (1500) | 121 (250) | Coolable Geometry | Less than 1204.4 (2200) | Not to exceed 17% of total cladding thickness | 310 (45) |

**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) | 6.97 (1011) |
| Natural Circulation Core Flow, Mkg/hr (Mlb/hr)* | 35.54 (78.36) |
| Steam Flow, kg/s (Mlbm/hr)<br>Feed Flow, kg/s (Mlbm/hr) | 2538 (20.14)<br>2532 (20.10) |
| Feedwater Temperature, °C (°F) | 224.9 (436.8) |
| Nuclear Characteristics Used in TRACG Simulations Condition<br>Exposure | Reference 2.5-3<br>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 (110) |
| 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) | 8.62 to 8.76 (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) | 0.07 (1035) |
| ATWS Dome Pressure Sensor Time Constant, s | ≤0.5 |
| ATWS Logic Time Delay, s | ≤1 |
| Pool Cooling Capacity, kW/ºC (kW/ºF) | 430.6 (239.2) |
| NRHX Shell Side Water Temperature for Pool cooling, ºC (ºF) | 38.3 (101) |
| Low Water Level For Closure of MSIVs, cm (in) | 1605.0 (631.9) |
| Low Steamline Pressure For Closure of MSIVs, MPaG (psig) | 5.41 (785) |
| Temperature For Automatic Pool Cooling, ºC (ºF) | 48.9 (120) |
| Max time delay from SCRRI/SRI command to FW runback due to persisting elevated power levels, s[(2)] | 30 |

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

(2) The purpose of this requirement is to ensure a FW runback is initiated before the power increases significantly above rated power following a generator load rejection or turbine trip with bypass with a SCRRI/SRI failure.

**Table 2.5.4-4a**

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

| Parameter | Value | Time (s) |
|---|---|---|
| Sensed Maximum Neutron Flux, % | 242 | 3.4 |
| Maximum Vessel Bottom Pressure, MPaG (psig) | 9.60 (1391) | 20 |
| Maximum Bulk Suppression Pool Temperature, ºC (ºF) | 73.9 (165) | 346 |
| Associated Containment Pressure, kPaG (psig) | 208 (30.2) | 346 |
| Peak Cladding Temperature, ºC (ºF) | 835 (1535) | 27 |

**Table 2.5.4-4b**

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

| Time (s) | Event |
|---|---|
| 0 | MSIV Closure starts |
| 0.5 | ICS initiates |
| 4 | ATWS trip set at high pressure |
| 5.6 | SRVs open |
| 40 | Level drops below L2 set point |
| 50 | HP CRD flow starts |
| 195 | SLCS injection starts |
| 715 | 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) |
|---|---|---|
| Sensed Maximum Neutron Flux, % | 237.9 | 9.4 |
| Maximum Vessel Bottom Pressure, MPaG (psig) | 9.48 (1375) | 28 |
| Maximum Bulk Suppression Pool Temperature, ºC (ºF) | 73.8 (164.9) | 356 |
| Associated Containment Pressure, kPaG (psig) | 208 (30.2) | 356 |
| Peak Cladding Temperature, ºC (ºF) | 803.5 (1478.2) | 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 |
| 7.4 | ICS initiates |
| 10 | ATWS trip set at high pressure |
| 11.3 | SRVs open |
| 47 | Level drops below L2 set point |
| 57 | HP CRD flow starts |
| 201 | SLCS injection starts |
| 725 | 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 |
| 0.04 | Turbine bypass valves start to open |
| 0.08 | Turbine control valves closed |
| 2 | 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 | Feedwater flow decay to 0 |
| 6.3 | Vessel water level reaches Level 3 |
| 10 | Vessel water level reaches Level 2 |
| 18 | Isolation condensers begins to drop cold water inside the vessel |
| 33 | Isolation condensers drainage valve is fully open |
| 40 | MSIV valve begins to close |
| 45 | 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) | 113000 (16.4) |
| Decay heat, Mw | 19.5 |
| Wide range measured level over TAF, m (ft) | 6.1 (20.0) |
| Collapsed Level over TAF, m (ft)** | 4.55 (14.9) |
| Isolation condenser flow, kg/s (lb/hr) | 8.64 (68,600) |

**Figure 2.5.1-1. through 2.5.1-3 Deleted**

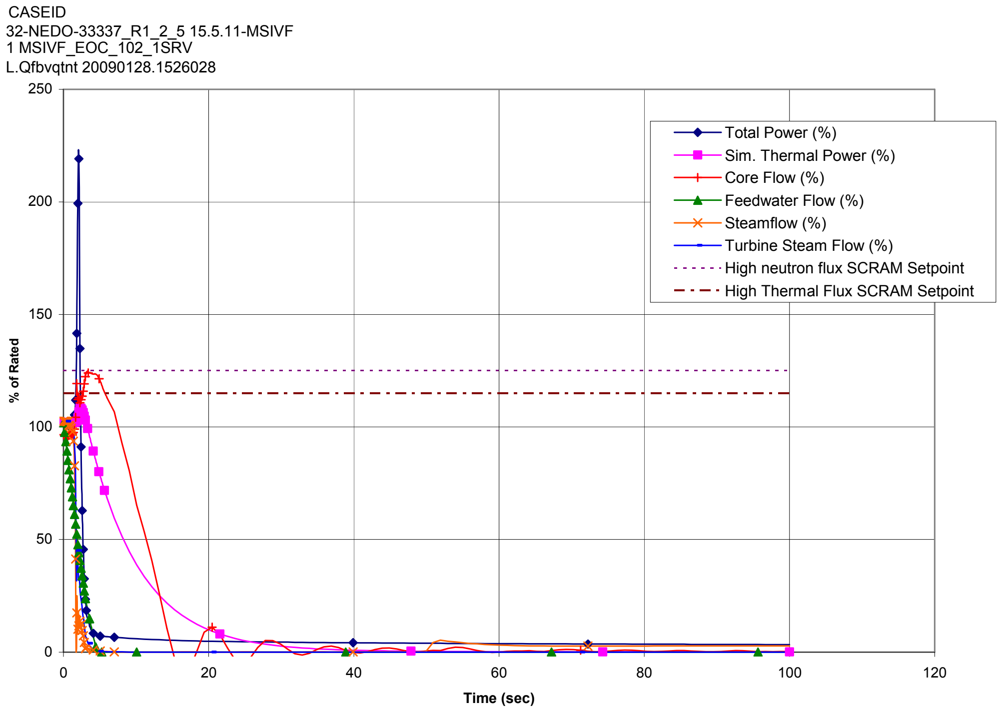



**Figure 2.5.1-4a. MSIV Closure With Flux Scram**

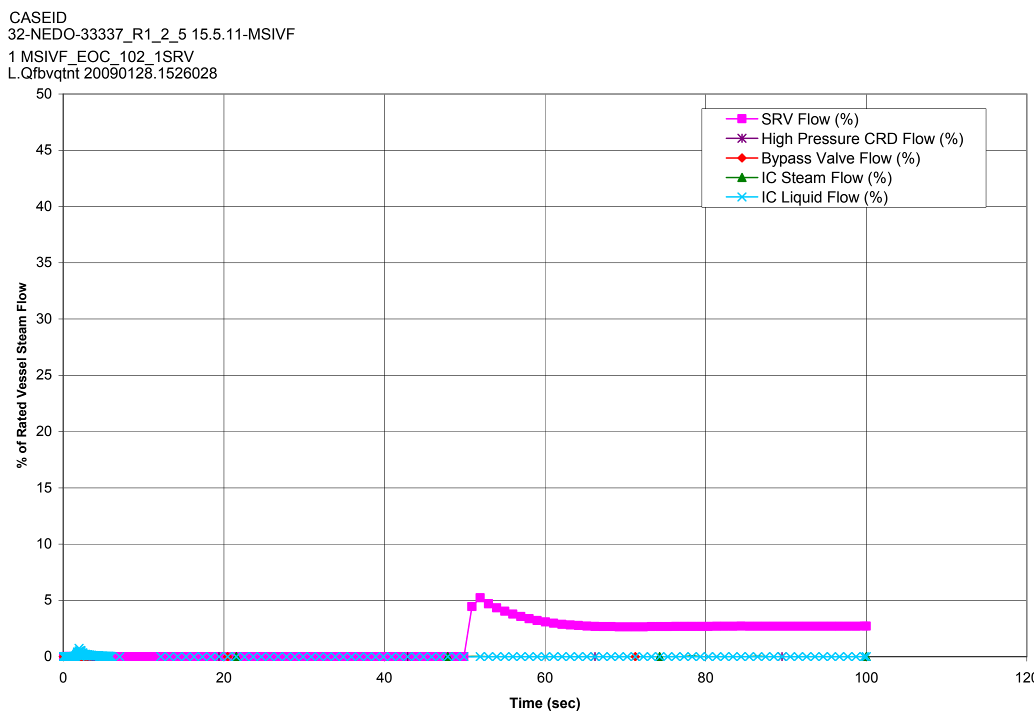



**Figure 2.5.1-4b. MSIV Closure With Flux Scram**

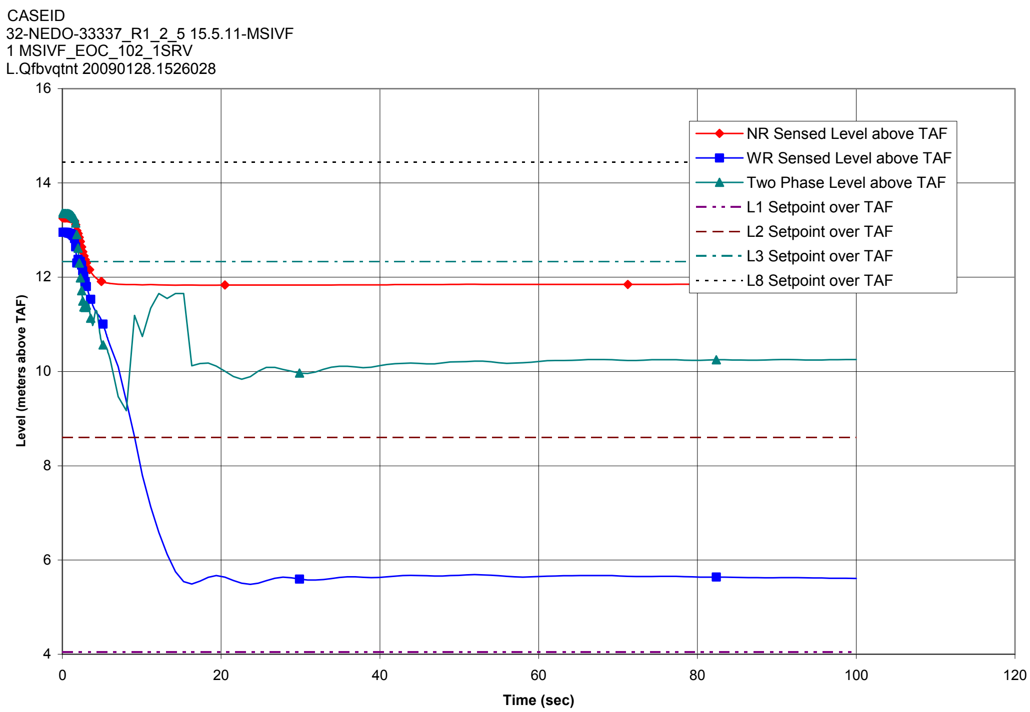



**Figure 2.5.1-4c. MSIV Closure With Flux Scram**

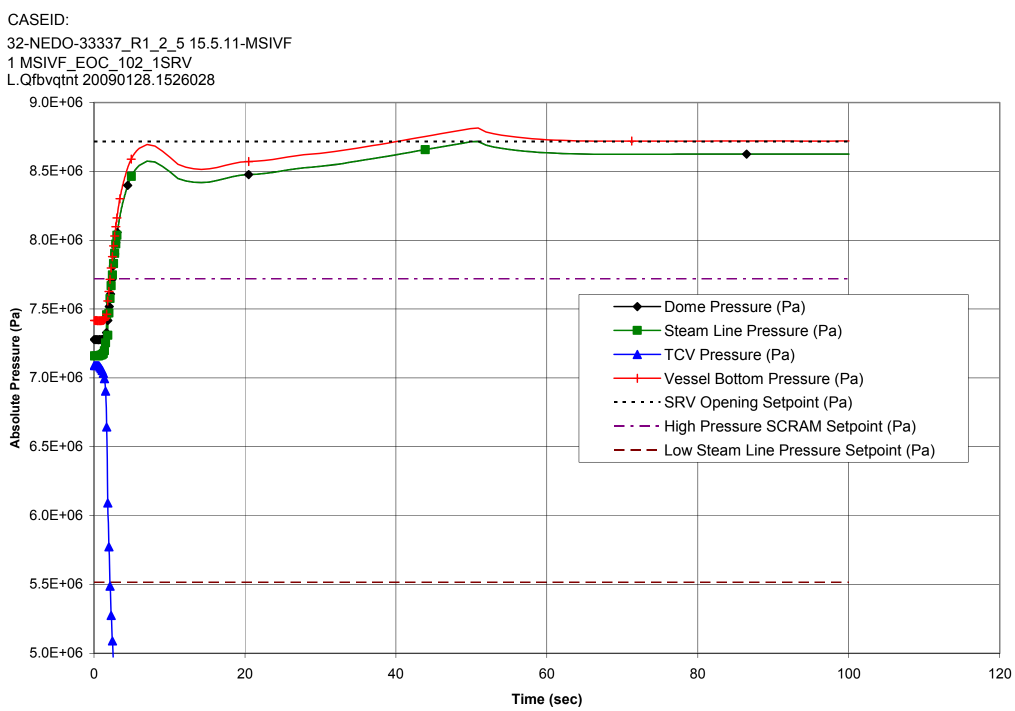



**Figure 2.5.1-4d. MSIV Closure With Flux Scram**

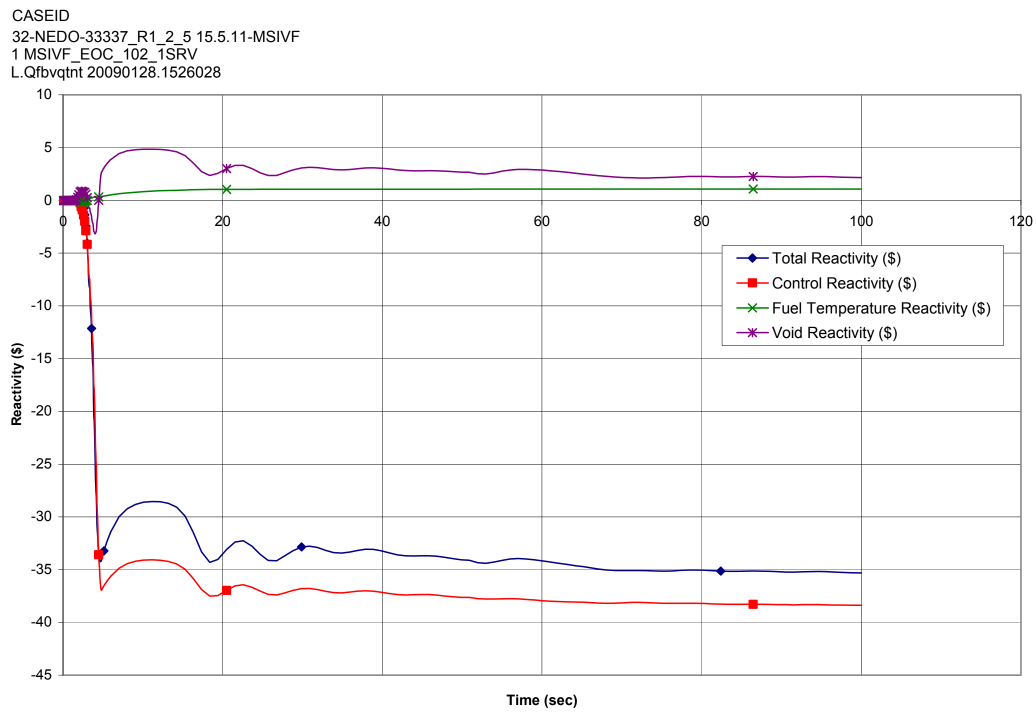



Figure 2.5.1-4e. MSIV Closure With Flux Scram

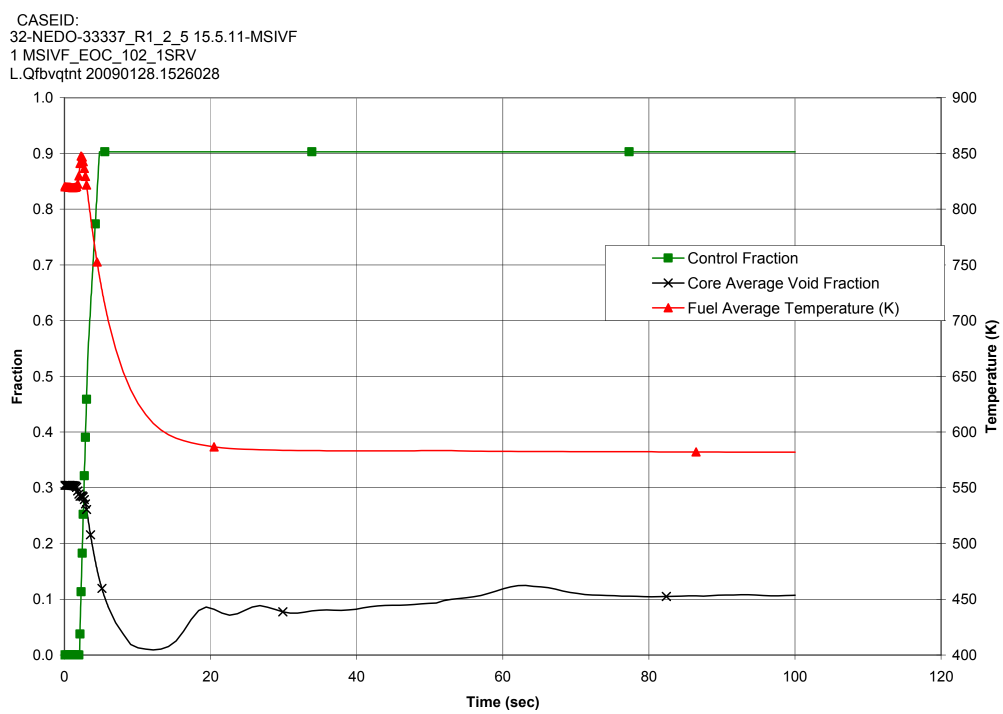



Figure 2.5.1-4f. MSIV Closure With Flux Scram

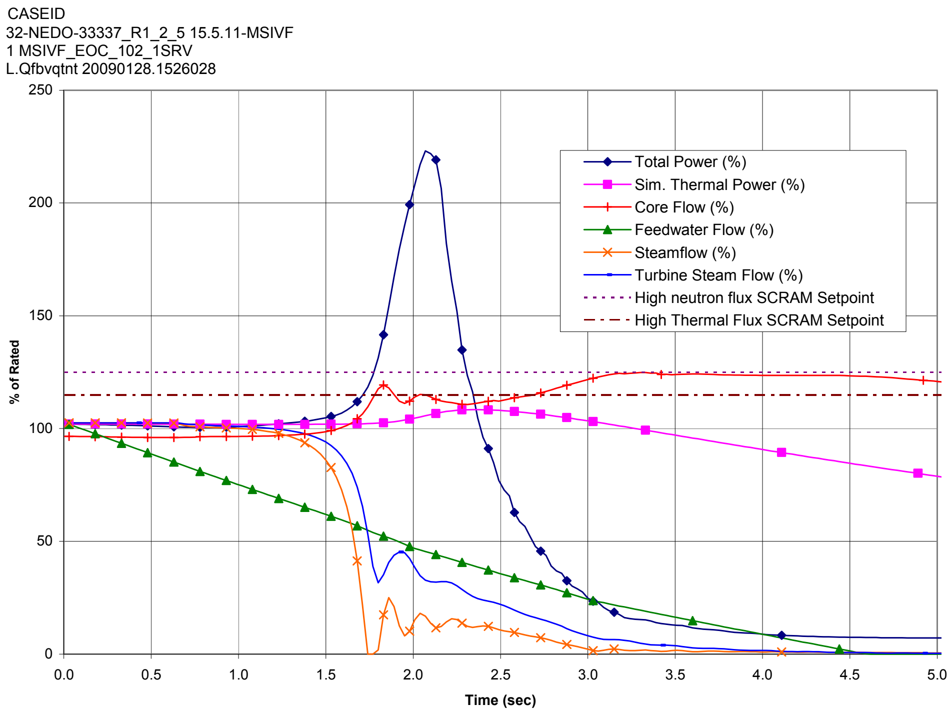
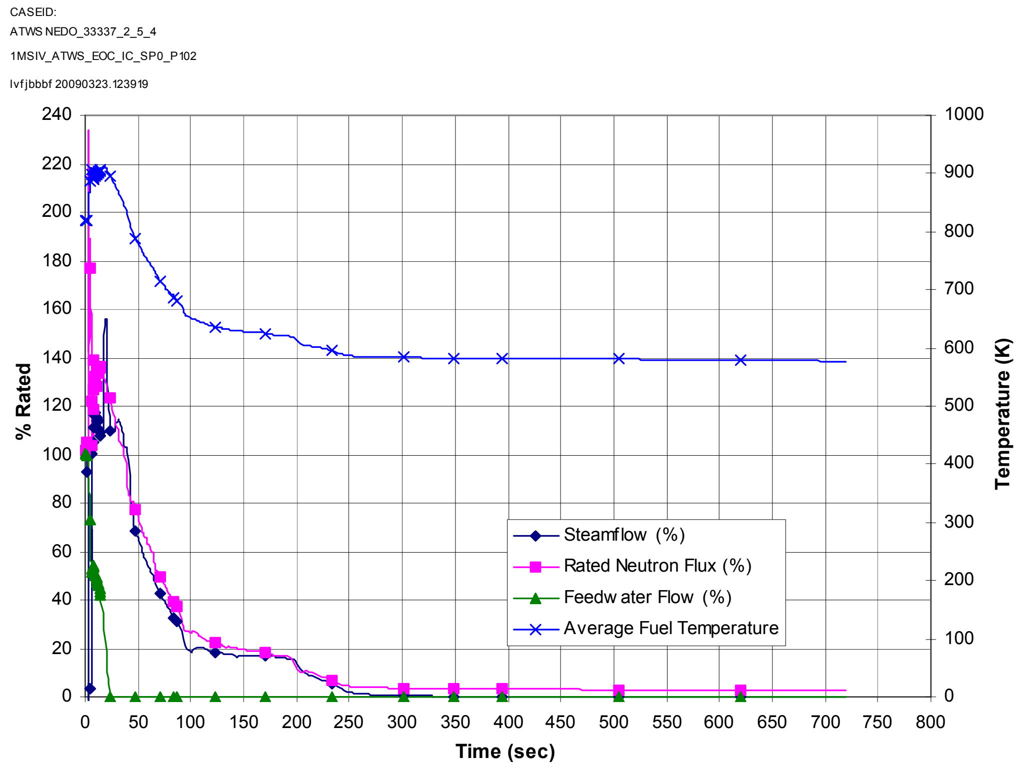



**Figure 2.5.1-4g. MSIV Closure With Flux Scram (Figure 2.5.1-4a from 0 to 5 s)**



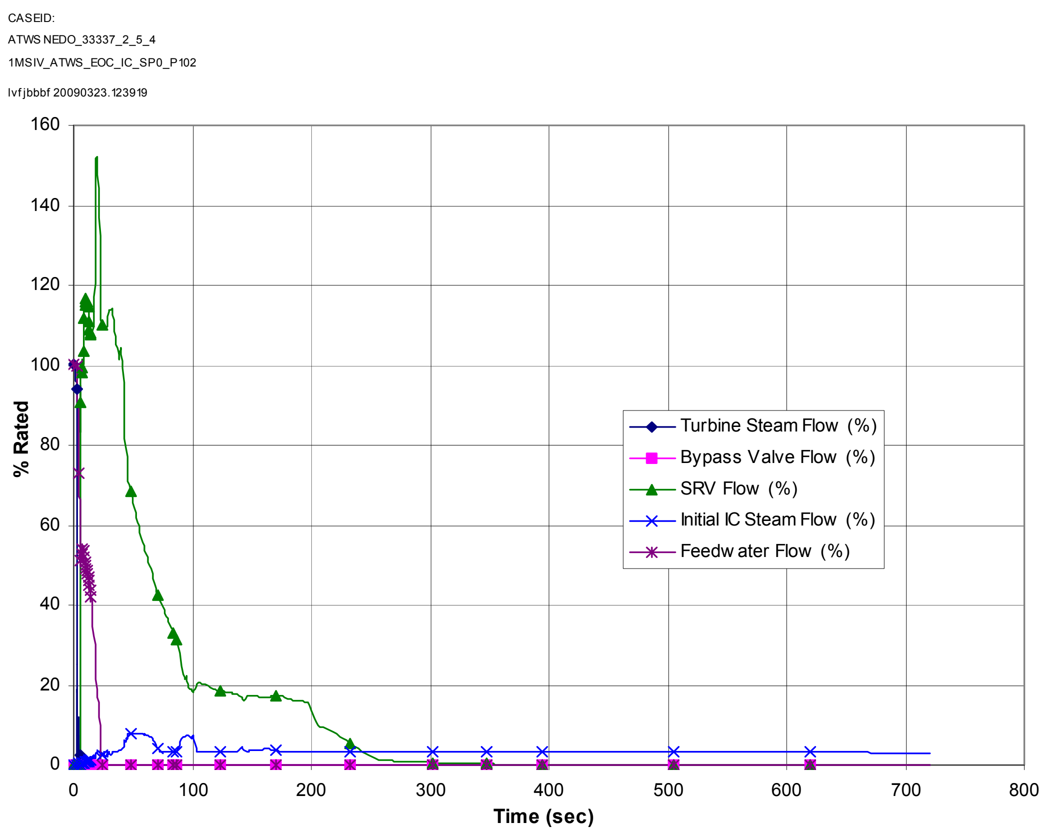
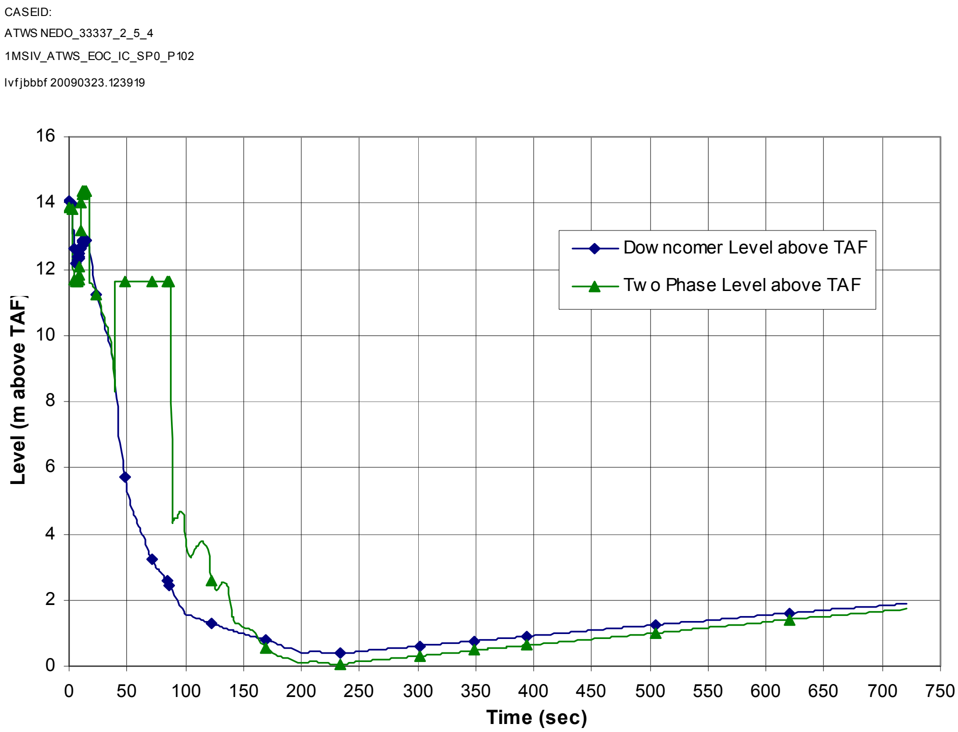




**Figure 2.5.4-1a. MSIV Closure - SLC System Bounding Case**



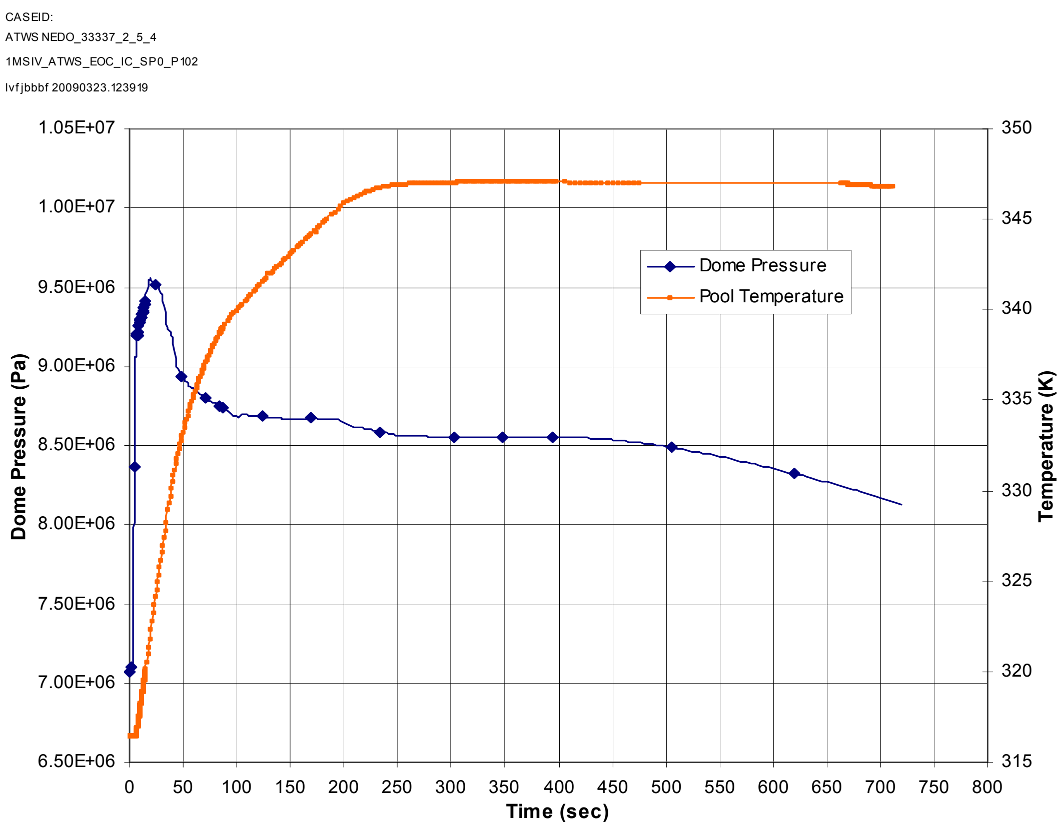
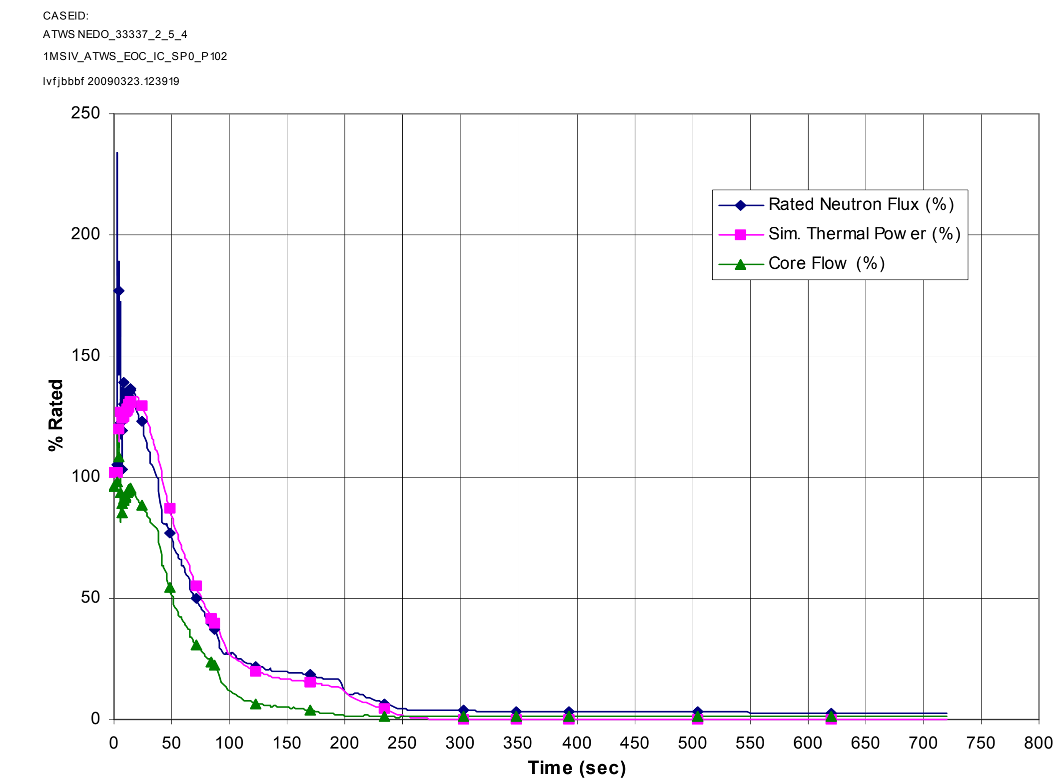




**Figure 2.5.4-1b. MSIV Closure - SLC System Bounding Case**



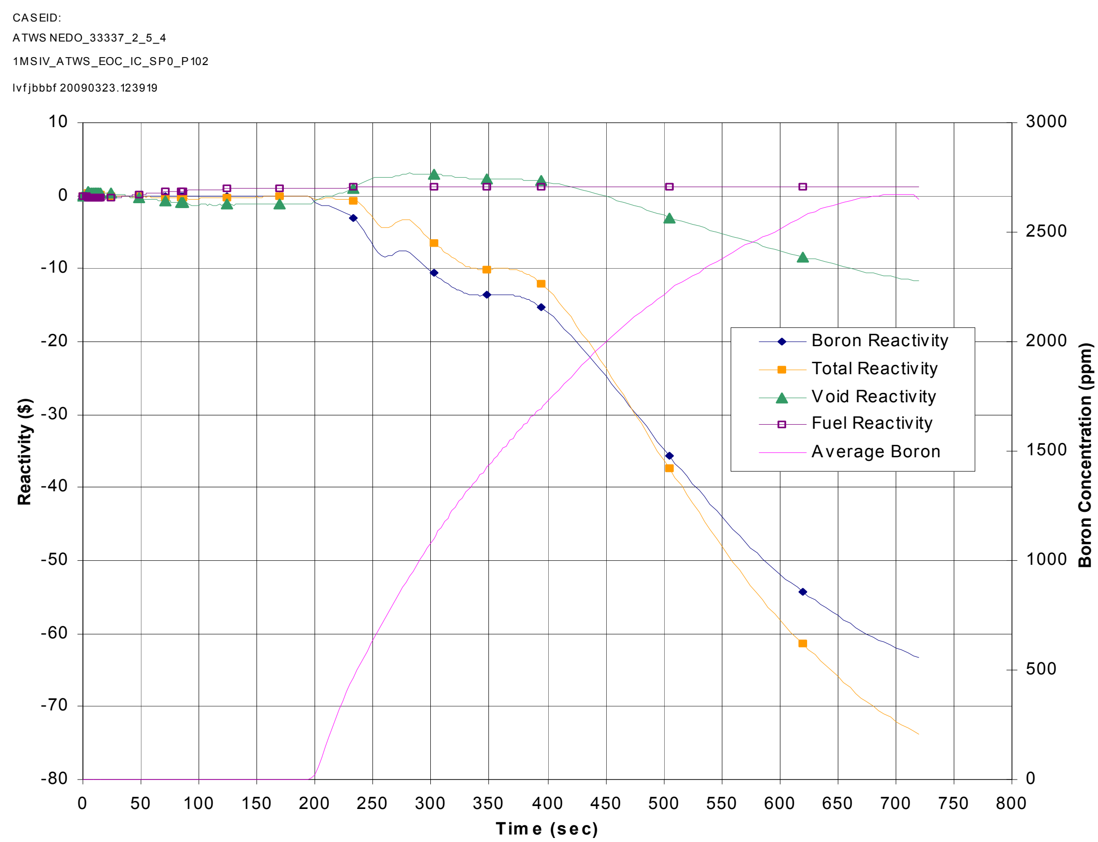




**Figure 2.5.4-1c. MSIV Closure - SLC System Bounding Case**

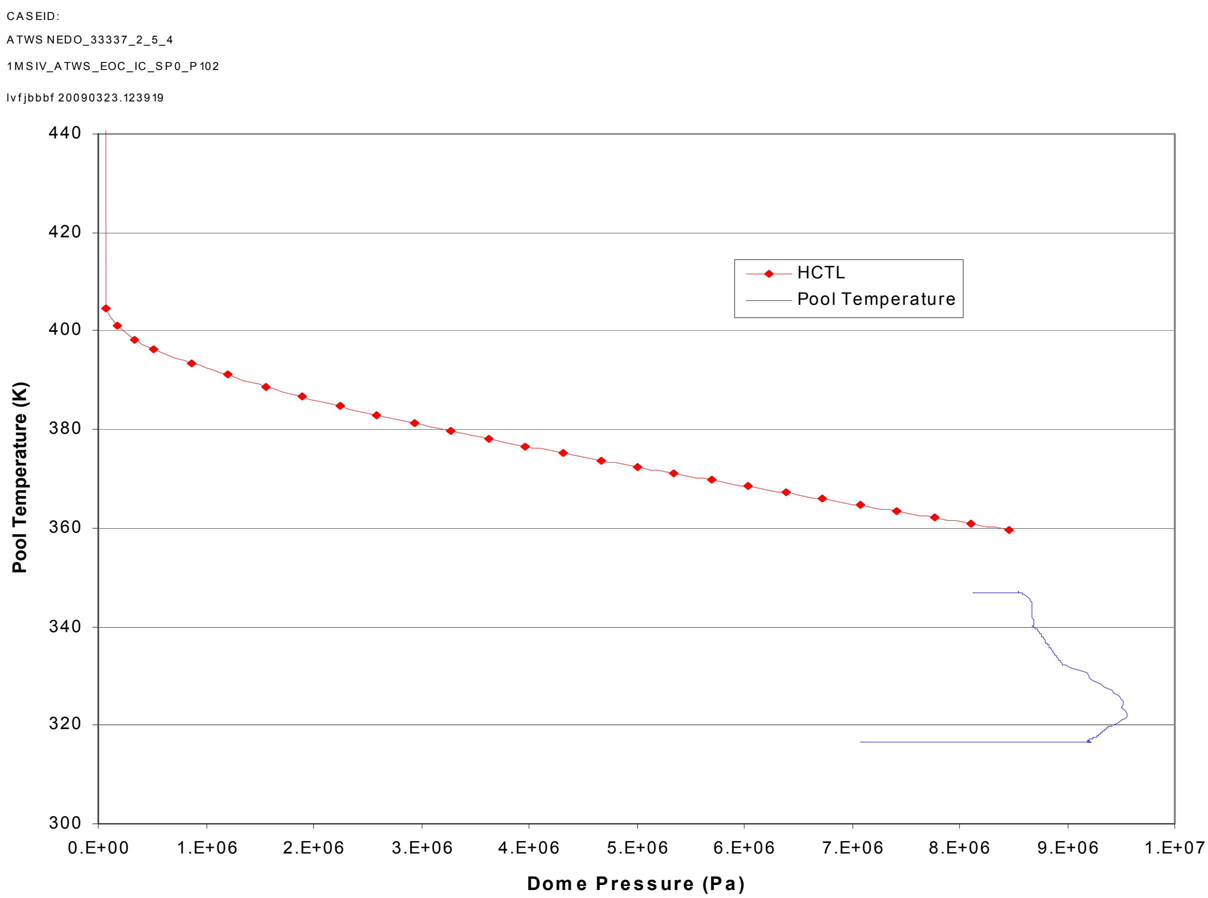



**Figure 2.5.4-1d. MSIV Closure - SLC System Bounding Case**

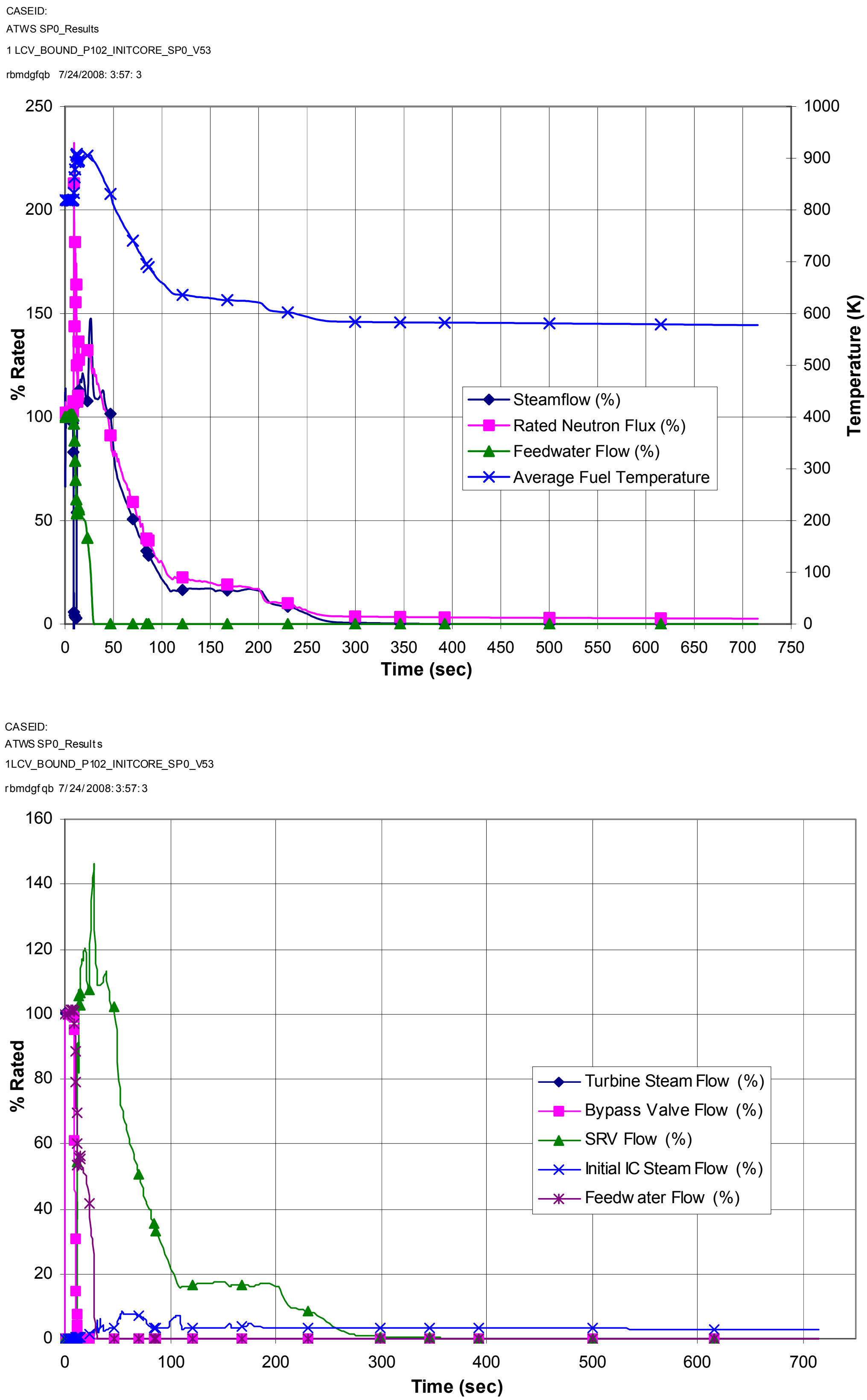
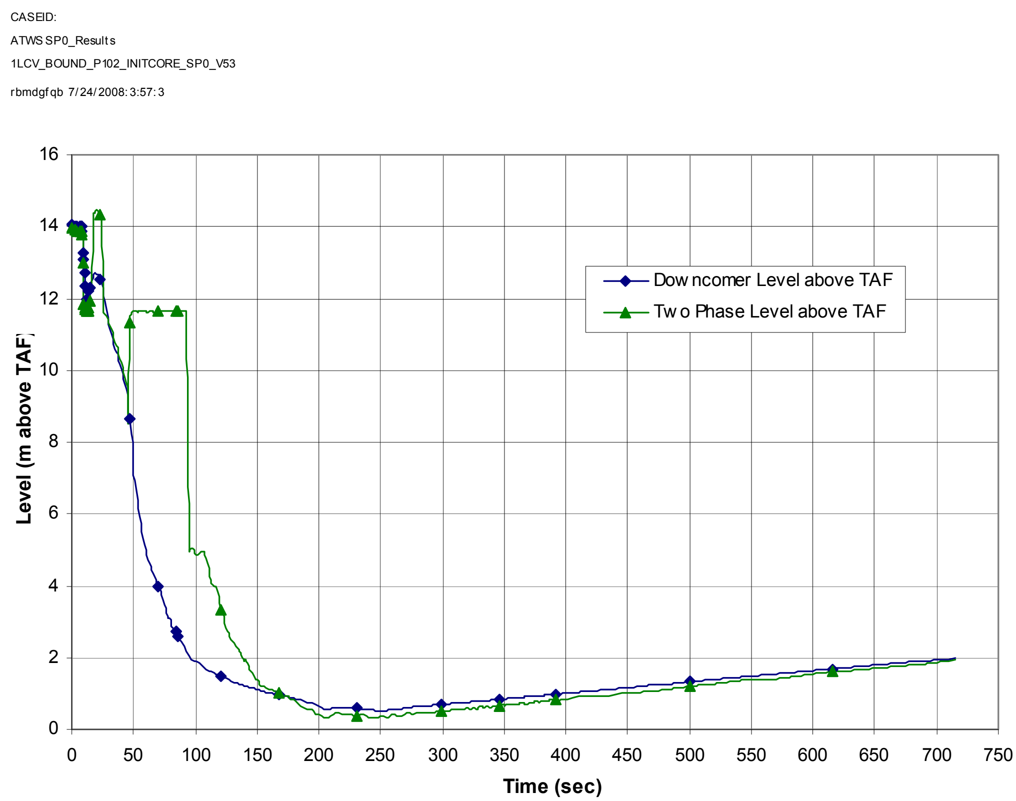



**Figure 2.5.4-2a. Loss of Condenser Vacuum SLC System Bounding Case**



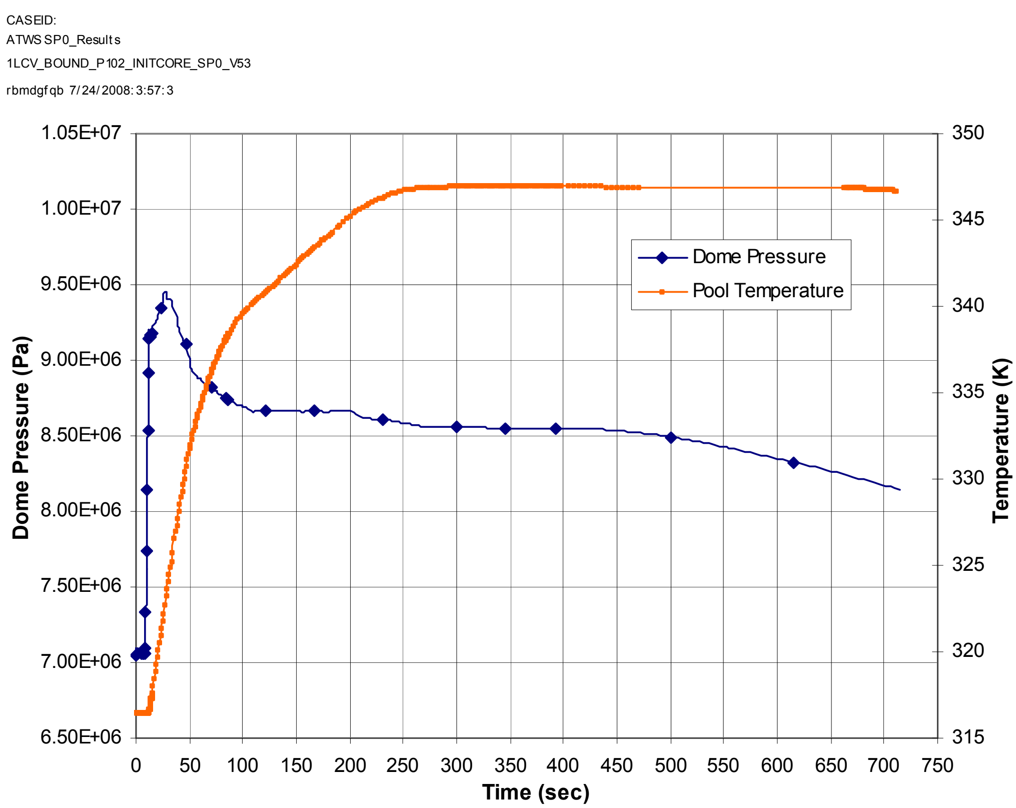
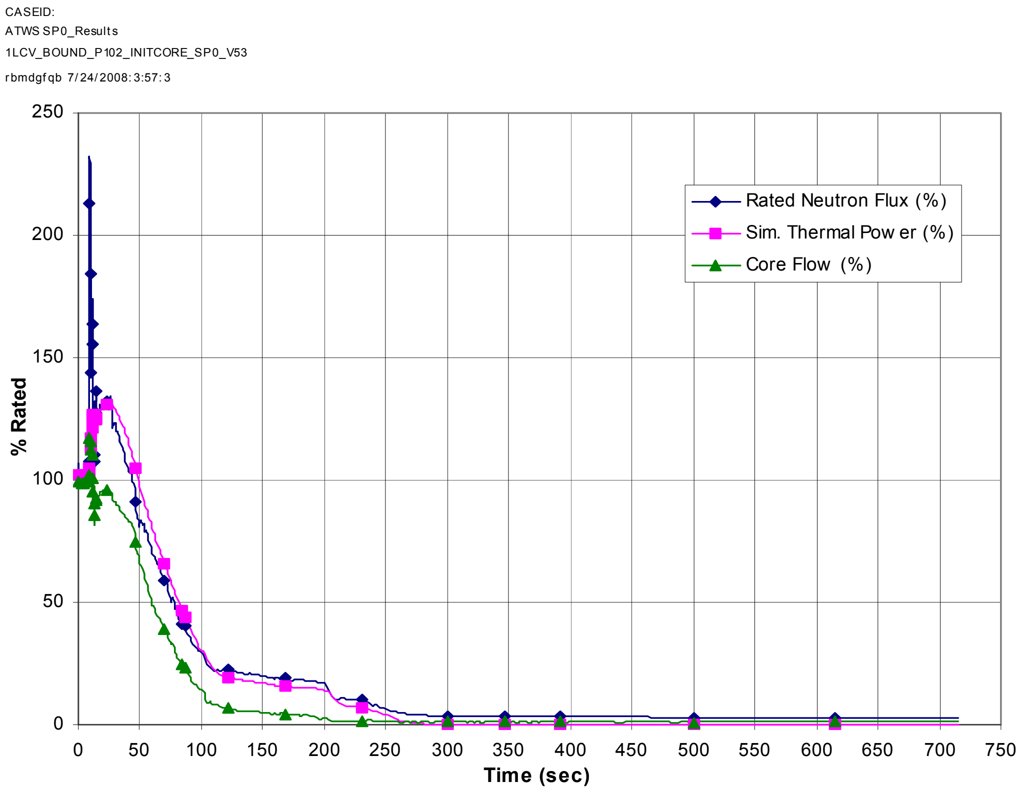




**Figure 2.5.4-2b. Loss of Condenser Vacuum SLC System Bounding Case**



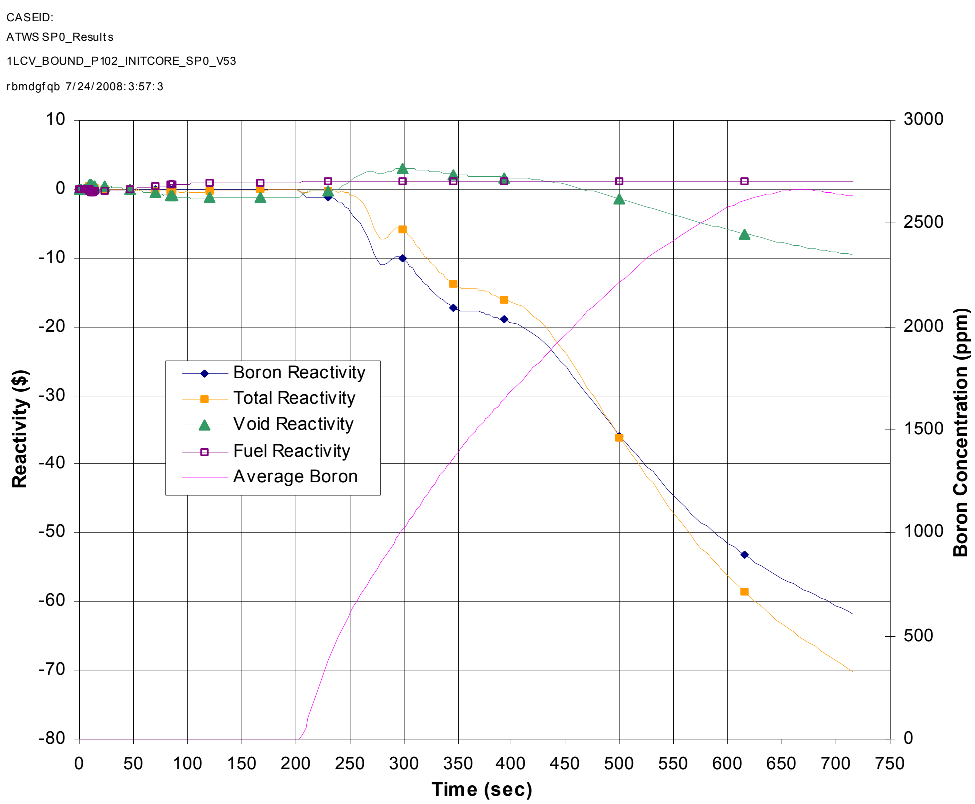




**Figure 2.5.4-2c. Loss of Condenser Vacuum SLC System Bounding Case**

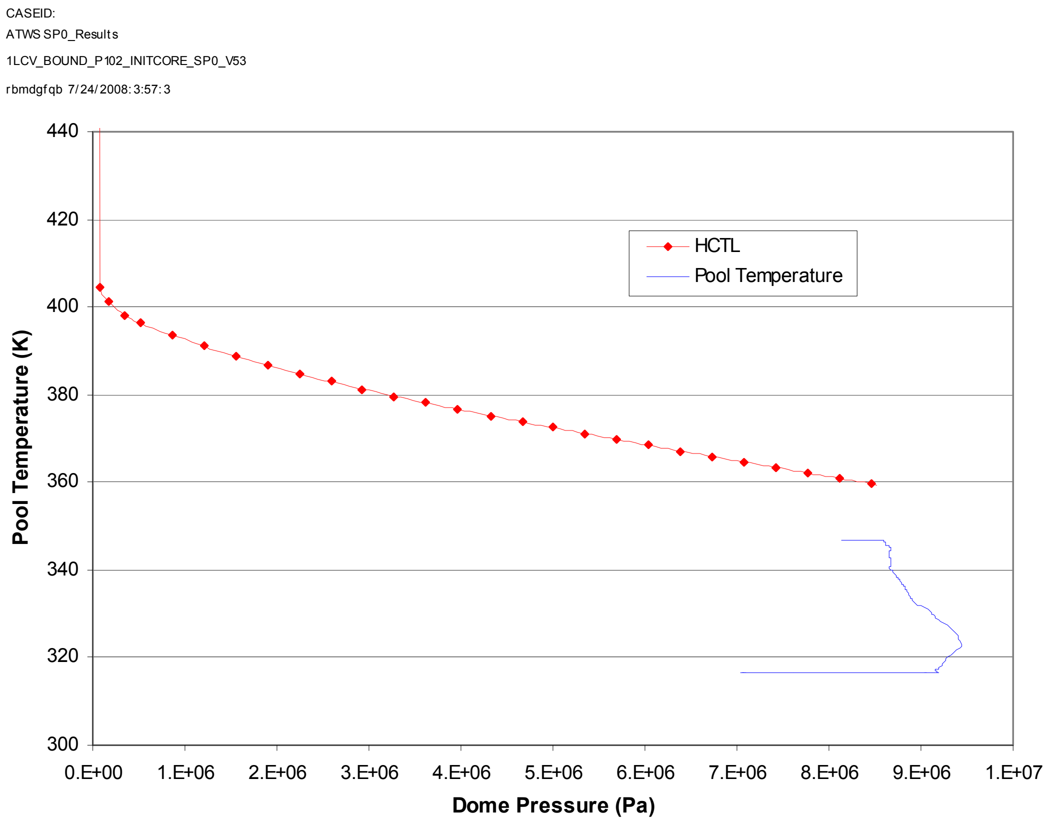



**Figure 2.5.4-2d. Loss of Condenser Vacuum SLC System Bounding Case**

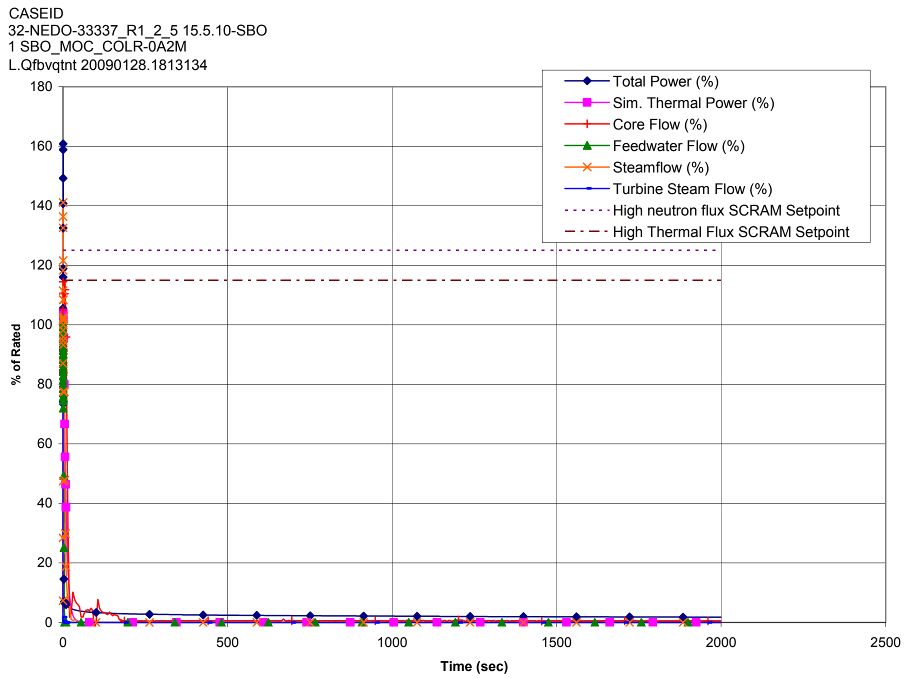



**Figure 2.5.5-1a. Station Blackout**

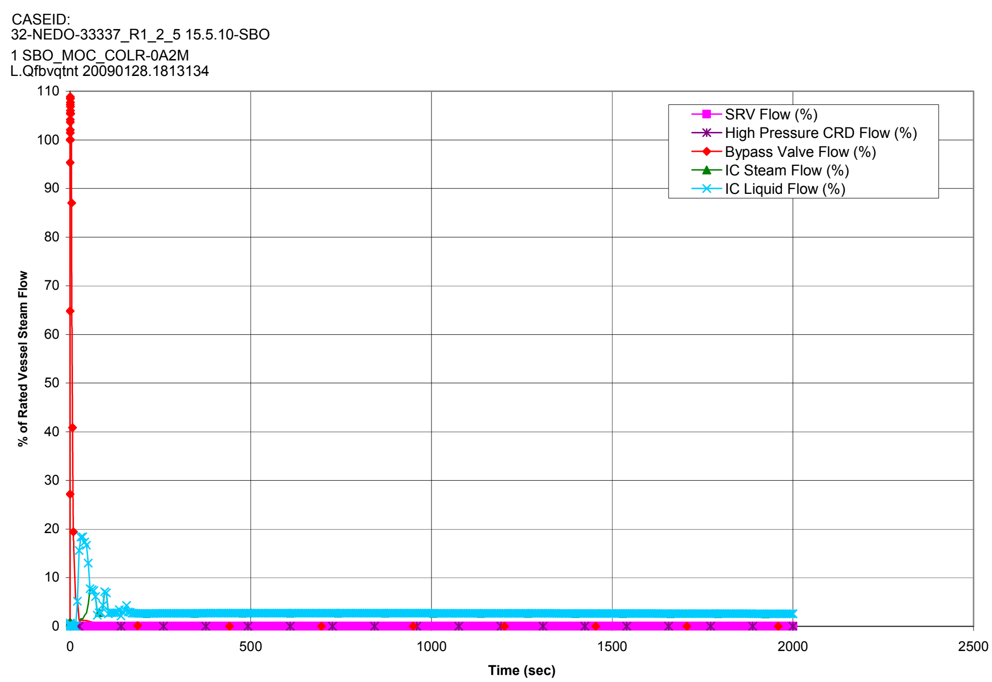



**Figure 2.5.5-1b. Station Blackout**

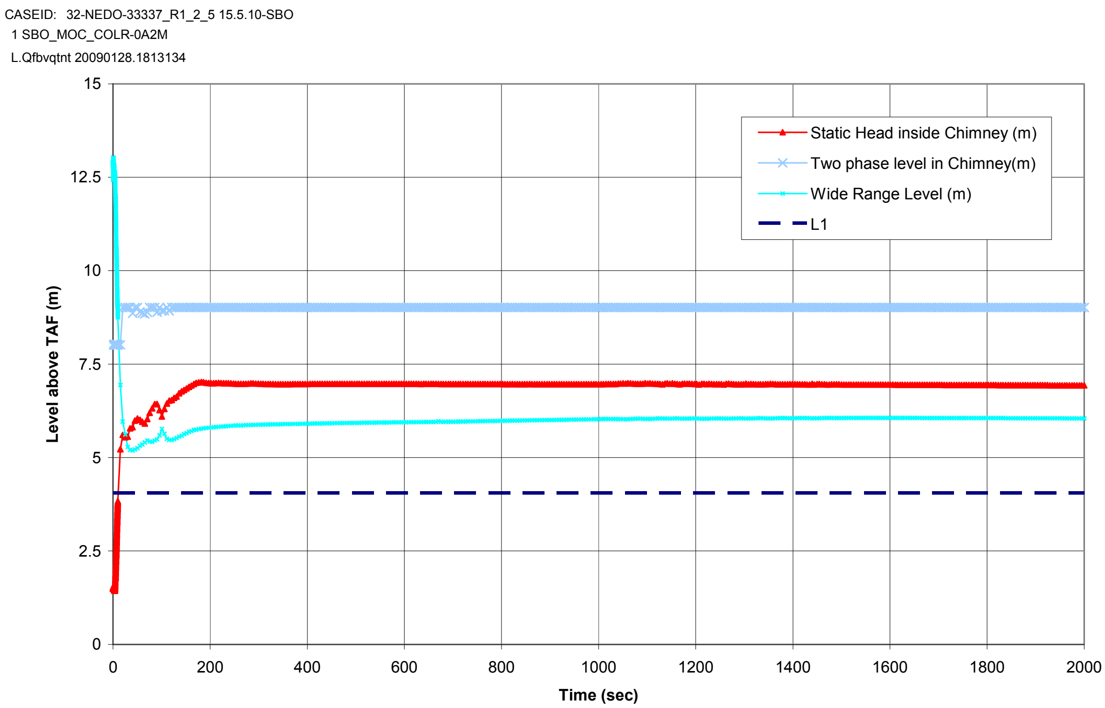



Figure 2.5.5-1c. Station Blackout

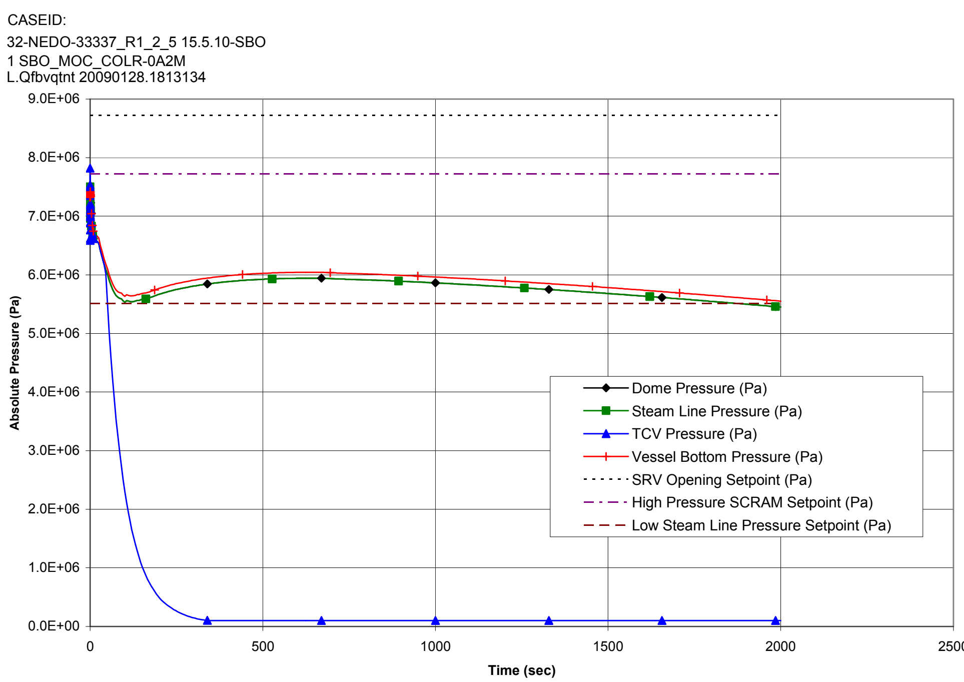



Figure 2.5.5-1d. Station Blackout

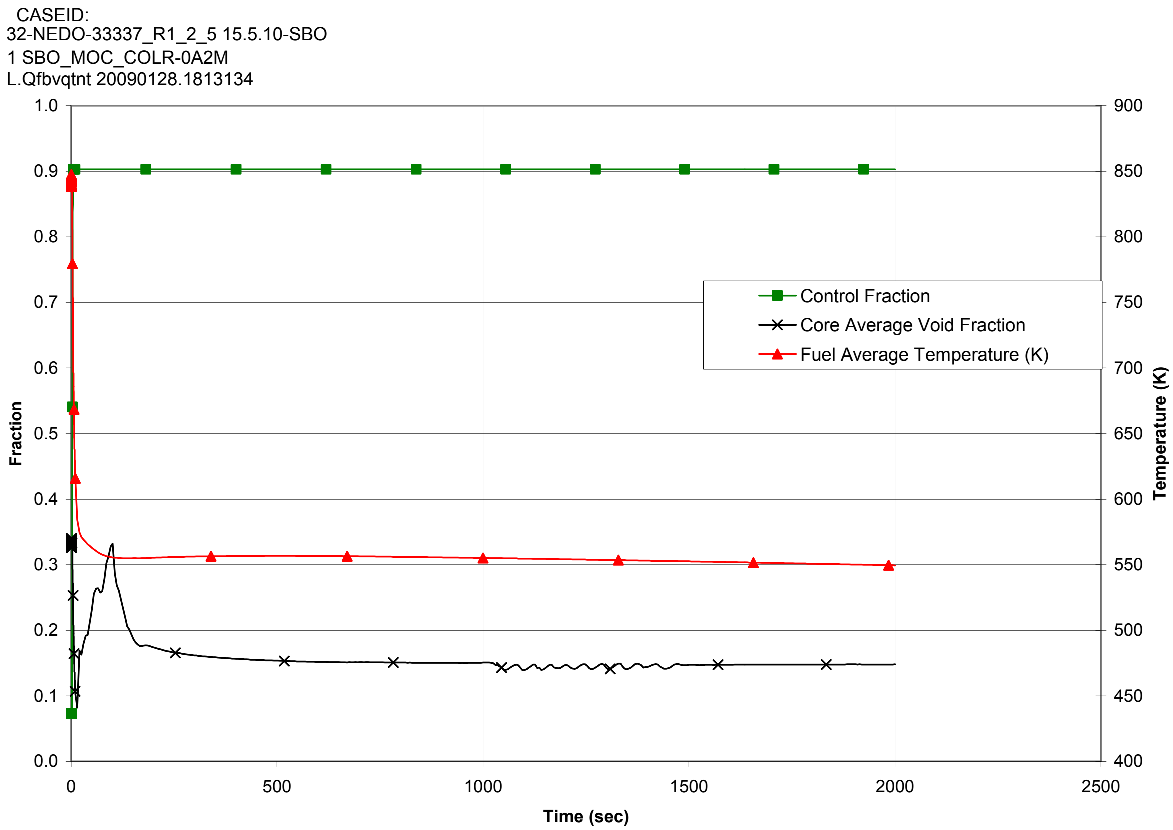



**Figure 2.5.5-1e. Station Blackout**

# 3. CONCLUSION

## 3.1 CONCLUSION

The analyses for the ESBWR limiting events with initial core are documented in this report. 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.