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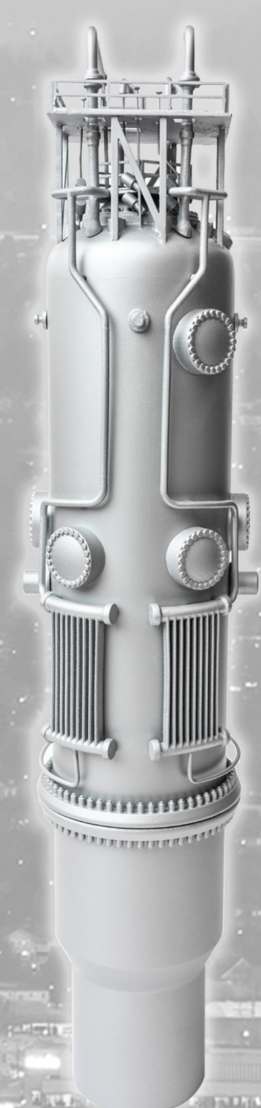
NuScale Standard Plant
Design Certification Application

Chapter Fifteen

Transient and Accident Analyses



PART 2 - TIER 2

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TABLE OF CONTENTS

CHAPTER 15 TRANSIENT AND ACCIDENT ANALYSES 15.0-1

15.0 Transient and Accident Analyses 15.0-1

15.0.0 Classification and Key Assumptions 15.0-1

15.0.1 Radiological Consequence Analyses Using Alternative Source Terms 15.0-12

15.0.2 Review of Transient and Accident Analysis Methods 15.0-12

15.0.3 Design Basis Accident Radiological Consequence Analyses for Advanced Light Water Reactors. 15.0-21

15.0.4 Safe, Stabilized Condition 15.0-33

15.0.5 Long Term Decay and Residual Heat Removal 15.0-33

15.0.6 Evaluation of a Return to Power 15.0-35

15.0.7 References 15.0-39

15.1 Increase in Heat Removal by the Secondary System 15.1-1

15.1.1 Decrease in Feedwater Temperature 15.1-1

15.1.2 Increase in Feedwater Flow 15.1-6

15.1.3 Increase in Steam Flow. 15.1-11

15.1.4 Inadvertent Opening of Steam Generator Relief or Safety Valve 15.1-16

15.1.5 Steam Piping Failures Inside and Outside of Containment. 15.1-17

15.1.6 Loss of Containment Vacuum/Containment Flooding 15.1-25

15.2 Decrease in Heat Removal by the Secondary Side 15.2-1

15.2.1 Loss of External Load 15.2-2

15.2.2 Turbine Trip 15.2-6

15.2.3 Loss of Condenser Vacuum 15.2-8

15.2.4 Closure of Main Steam Isolation Valve(s) 15.2-10

15.2.5 Steam Pressure Regulator Failure (Closed) 15.2-14

15.2.6 Loss of Non-Emergency AC Power to the Station Auxiliaries 15.2-14

15.2.7 Loss of Normal Feedwater Flow 15.2-18

15.2.8 Feedwater System Pipe Breaks Inside and Outside of Containment 15.2-22

15.2.9 Inadvertent Operation of the Decay Heat Removal System 15.2-29

15.3 Decrease in RCS Flow Rate. 15.3-1

15.4 Reactivity and Power Distribution Anomalies. 15.4-1

15.4.1 Uncontrolled Control Rod Assembly Withdrawal from a Subcritical or Low Power or Startup Condition 15.4-1

15.4.2 Uncontrolled Control Rod Assembly Withdrawal at Power 15.4-5

TABLE OF CONTENTS

15.4.3 Control Rod Misoperation (System Malfunction or Operator Error) 15.4-9

15.4.4 Startup of an Inactive Loop or Recirculation Loop at an Incorrect Temperature 15.4-17

15.4.5 Flow Controller Malfunction Causing an Increase in Core Flow Rate (Boiling Water Reactor) 15.4-17

15.4.6 Inadvertent Decrease in Boron Concentration in the Reactor Coolant System 15.4-17

15.4.7 Inadvertent Loading and Operation of a Fuel Assembly in an Improper Position 15.4-24

15.4.8 Spectrum of Rod Ejection Accidents 15.4-26

15.4.9 Spectrum of Rod Drop Accidents 15.4-35

15.4.10 References 15.4-35

15.5 Increase in Reactor Coolant Inventory 15.5-1

15.5.1 Chemical and Volume Control System Malfunction 15.5-1

15.6 Decrease in Reactor Coolant Inventory 15.6-1

15.6.1 Inadvertent Opening of a Reactor Safety Valve 15.6-1

15.6.2 Failure of Small Lines Carrying Primary Coolant Outside Containment 15.6-1

15.6.3 Steam Generator Tube Failure (Thermal Hydraulic) 15.6-6

15.6.4 Main Steam Line Failure Outside Containment (BWR) 15.6-12

15.6.5 Loss-of-Coolant Accidents Resulting from Spectrum of Postulated Piping Breaks within the Reactor Coolant Pressure Boundary 15.6-12

15.6.6 Inadvertent Operation of Emergency Core Cooling System 15.6-19

15.6.7 References 15.6-26

15.7 Radioactive Release from a Subsystem or Component 15.7-1

15.7.1 Gaseous Waste Management System Leak or Failure 15.7-1

15.7.2 Liquid Waste Management System Leak or Failure 15.7-1

15.7.3 Postulated Radioactive Releases Due to Liquid-Containing Tank Failures 15.7-1

15.7.4 Fuel Handling Accidents 15.7-1

15.7.5 Spent Fuel Cask Drop Accidents 15.7-2

15.7.6 NuScale Power Module Drop Accident 15.7-2

15.8 Anticipated Transients Without Scram (ATWS) 15.8-1

15.8.1 References 15.8-1

15.9 Stability 15.9-1

15.9.1 Consideration of Thermal-Hydraulic Stability 15.9-1

TABLE OF CONTENTS

15.9.2 Stability Analyses 15.9-2

15.9.3 Stability Analysis for Operational Occurrences 15.9-4

15.9.4 Demonstration of Module Protection System Functions to Preclude Instability. 15.9-9

15.9.5 Conclusions 15.9-10

15.9.6 References 15.9-10

15.10 Core Damage Event 15.10-1

15.10.1 Inputs, Methods, and Assumptions 15.10-1

15.10.2 Radiological Consequences of the Core Damage Source Term. 15.10-4

LIST OF TABLES

Table 15.0-1: Design Basis Events 15.0-40
Table 15.0-2: Acceptance Criteria-Thermal Hydraulic and Fuel 15.0-44
Table 15.0-3: Acceptance Criteria Specific to Rod Ejection Accidents 15.0-45
Table 15.0-4: Acceptance Criteria Specific to Loss of Coolant Accidents. 15.0-46
Table 15.0-5: Acceptance Criteria-Radiological 15.0-47
Table 15.0-6: Module Initial Conditions Ranges for Design Basis Event Evaluation. 15.0-48
Table 15.0-7: Analytical Limits and Time Delays. 15.0-49
Table 15.0-8: Reactivity Coefficients. 15.0-51
Table 15.0-9: Assumed Single Failures and Credited Nonsafety-Related Systems. 15.0-54
Table 15.0-10: Referenced Topical and Technical Reports 15.0-57
Table 15.0-11: Summary of Applicable Radiological Events to the NuScale Design 15.0-60
Table 15.0-12: Radiological Dose Consequences for Chapter 15 Analyses 15.0-61
Table 15.0-13: Onsite and Offsite Receptor Variables 15.0-62
Table 15.0-14: Primary Coolant Source Term. 15.0-63
Table 15.0-15: Control Room Parameters 15.0-64
Table 15.0-16: Return to Power Calculation Nominal Results Summary 15.0-65
Table 15.0-17: Return to Power Calculation Limiting Results. 15.0-66
Table 15.0-18: Technical Support Center Parameters. 15.0-67
Table 15.0-19: 10x Limiting Accident Radiation Post-Filtration Monitor Analytical Limits 15.0-68
Table 15.0-20: Assumptions for Accident Airborne Effluent Release Point Characteristics for Offsite Receptors 15.0-69
Table 15.0-21: Assumptions for Control Room χ/Q. 15.0-70
Table 15.0-22: Results of Long Term Cooling Analysis 15.0-71
Table 15.1-1: Sequence of Events Limiting MCHFR Case (15.1.1 Decrease in Feedwater Temperature) 15.1-31
Table 15.1-2: Decrease in Feedwater Temperature - Inputs (Limiting Minimum Critical Heat Flux Ratio Case) 15.1-32
Table 15.1-3: Decrease in Feedwater Temperature (15.1.1) - Limiting Analysis Results. 15.1-33
Table 15.1-4: Sequence of Events for Limiting MCHFR Case (15.1.2 Increase in Feedwater Flow) 15.1-34
Table 15.1-5: Increase in Feedwater Flow - Inputs (Limiting Minimum Critical Heat Flux Ratio Case) 15.1-35
Table 15.1-6: Increase in Feedwater Flow (15.1.2) - Limiting Analysis Results. 15.1-36
Table 15.1-7: Sequence of Events (15.1.3 Increase in Steam Flow) 15.1-37

LIST OF TABLES

Table 15.1-8: Increase in Steam Flow - Inputs (Limiting Minimum Critical Heat Flux Ratio Case) 15.1-38

Table 15.1-9: Increase in Steam Flow (15.1.3) - Limiting Analysis Results 15.1-39

Table 15.1-10: Sequence of Events (15.1.5 Steam Line Break, Limiting Reactor Pressure Vessel Pressure Case) 15.1-40

Table 15.1-11: Sequence of Events (15.1.5 Steam Line Break, Limiting Minimum Critical Heat Flux Ratio Case) 15.1-41

Table 15.1-12: Sequence of Events (15.1.5 Steam Line Break, Radiological Input Case for Iodine Spiking) 15.1-42

Table 15.1-13: Steam Piping Failure - Inputs 15.1-43

Table 15.1-14: Steam Piping Failure - Inputs to Radiological Analysis 15.1-44

Table 15.1-15: Steam Piping Failure (15.1.5) - Limiting Analysis Results 15.1-45

Table 15.1-16: Loss of Containment Vacuum/Containment Flooding - Inputs (Limiting Minimum Critical Heat Flux Ratio Case) 15.1-46

Table 15.1-17: Loss of Containment Vacuum/Containment Flooding (15.1.6) - Limiting Analysis Results 15.1-47

Table 15.1-18: Sequence of Events (15.1.6 Loss of Containment Vacuum/Containment Flooding) MCHFR Limiting Case 15.1-48

Table 15.1-19: Sequence of Events (15.1.5 Steam Line Break, Limiting Steam Generator Pressure Case) 15.1-49

Table 15.2-1: Loss of External Load -Turbine Trip - Loss of Condenser Vacuum - Initial Conditions Peak RCS Pressure 15.2-35

Table 15.2-2: Loss of External Load -Turbine Trip - Loss of Condenser Vacuum - Initial Conditions Peak SG Pressure 15.2-36

Table 15.2-3: Loss of External Load -Turbine Trip - Loss of Condenser Vacuum - Initial Conditions MCHFR 15.2-37

Table 15.2-4: Loss of External Load -Turbine Trip - Loss of Condenser Vacuum - RCS Maximum Pressure Sequence of Events. 15.2-38

Table 15.2-5: Loss of External Load -Turbine Trip - Loss of Condenser Vacuum - SG Maximum Pressure Sequence of Events. 15.2-39

Table 15.2-6: Loss of External Load -Turbine Trip - Loss of Condenser Vacuum - MCHFR Sequence of Events. 15.2-40

Table 15.2-7: Loss of External Load -Turbine Trip - Loss of Condenser Vacuum - Limiting Analysis Results 15.2-41

Table 15.2-8: Main Steam Isolation Valve Closure - Initial Conditions for RCS Pressure Limiting Case 15.2-42

Table 15.2-9: Main Steam Isolation Valve Closure - Initial Conditions for SG Pressure Limiting Case 15.2-43

LIST OF TABLES

Table 15.2-10: Main Steam Isolation Valve Closure - Initial Conditions for MCHFR Limiting Case . . . 15.2-44
Table 15.2-11: Main Steam Isolation Valve Closure - Sequence of Events for RCS Pressure Limiting Case . . . 15.2-45
Table 15.2-12: Main Steam Isolation Valve Closure - Sequence of Events for SG Pressure Limiting Case . . . 15.2-46
Table 15.2-13: Main Steam Isolation Valve Closure - Sequence of Events for MCHFR Limiting Case . . . 15.2-47
Table 15.2-14: Main Steam Isolation Valve Closure - Limiting Analysis Results . . . 15.2-48
Table 15.2-15: Loss of Non-Emergency AC Power- RCS Pressure Limiting Event Sequence of Events . . . 15.2-49
Table 15.2-16: Loss of Non-Emergency AC Power -SG Pressure and MCHFR Limiting Event - Sequence of Events . . . 15.2-50
Table 15.2-17: Not Used . . . 15.2-51
Table 15.2-18: Input Parameters Loss of Non-Emergency AC Power -Limiting Cases . . . 15.2-52
Table 15.2-19: Loss of Non-Emergency AC Power - Limiting Analysis Results . . . 15.2-53
Table 15.2-20: Loss of Feedwater Event - RCS Overpressurization/Limiting MCHFR Sequence of Events . . . 15.2-54
Table 15.2-21: Loss of Feedwater Event - Maximum SG Pressure - Sequence of Events . . . 15.2-55
Table 15.2-22: Input Parameters Loss of Feedwater - Limiting Cases . . . 15.2-56
Table 15.2-23: Loss of Feedwater - Limiting Analysis Results . . . 15.2-57
Table 15.2-24: Feedwater Line Break Sequence of Events - Peak RCS Pressure Case . . . 15.2-58
Table 15.2-25: Feedwater Line Break Sequence of Events - Peak Steam Generator Pressure Case . . . 15.2-59
Table 15.2-26: Feedwater Line Break Sequence of Events - Limiting MCHFR Case . . . 15.2-60
Table 15.2-27: Feedwater Line Break Sequence of Events - Limiting DHRS Case . . . 15.2-61
Table 15.2-28: Biases and Uncertainties - Feedwater Line Break . . . 15.2-62
Table 15.2-29: Feedwater Line Break- Limiting Analysis Results . . . 15.2-63
Table 15.2-30: Sequence of Events for Inadvertent Operation of Decay Heat Removal System - Limiting MCHFR Case . . . 15.2-64
Table 15.2-31: Sequence of Events for Inadvertent Operation of Decay Heat Removal System - Limiting Peak SG Pressure Case . . . 15.2-65
Table 15.2-32: Sequence of Events for Inadvertent Operation of Decay Heat Removal System - Limiting Peak RCS Pressure Case . . . 15.2-66

LIST OF TABLES

Table 15.2-33: Inadvertent Operation of Decay Heat Removal System - Limiting Analysis Results 15.2-67

Table 15.4-1: Sequence of Events for Limiting MCHFR Case (15.4.1 Uncontrolled CRA Withdrawal from Subcritical or Low Power Condition) 15.4-36

Table 15.4-2: Key Inputs for Limiting Centerline Fuel Temperature, MCHFR Case (15.4.1 Uncontrolled CRA Withdrawal from Subcritical or Low Power Condition) 15.4-37

Table 15.4-3: Uncontrolled Control Rod Assembly Withdrawal from a Subcritical or Low Power or Startup Condition (15.4.1) - Limiting Analysis Results 15.4-38

Table 15.4-4: Sequence of Events MCHFR Case - 75% Power (15.4.2 Uncontrolled Control Rod Assembly Withdrawal at Power) 15.4-39

Table 15.4-5: Key Inputs for Limiting MCHFR Case (15.4.2 Uncontrolled CRA Withdrawal at Power) 15.4-40

Table 15.4-6: Uncontrolled Control Rod Assembly Withdrawal at Power (15.4.2) - Limiting Analysis Results 15.4-41

Table 15.4-7: Sequence of Events (15.4.3 Control Rod Misoperation, Single Control Rod Assembly Withdrawal) 15.4-42

Table 15.4-8: Key Inputs for Single CRA Withdrawal with Limiting MCHFR 15.4-43

Table 15.4-9: Sequence of Events (15.4.3 Control Rod Misoperation, Control Rod Assembly Drop) 15.4-44

Table 15.4-10: Key Inputs for CRA Drop with Limiting MCHFR and LHGR 15.4-45

Table 15.4-11: Control Rod Misoperation (15.4.3) - Limiting Analysis Results 15.4-46

Table 15.4-12: Internal Flooding Sources (15.4.6 Inadvertent Decrease in Boron Concentration in the Reactor Coolant System) 15.4-47

Table 15.4-13: Bounding Critical Boron Concentrations and Boron Reactivity Coefficients (15.4.6 Inadvertent Decrease in Boron Concentration in the Reactor Coolant System) 15.4-48

Table 15.4-14: Mode 1, Hot Full Power Results (15.4.6 Inadvertent Decrease in Boron Concentration in the Reactor Coolant System) 15.4-49

Table 15.4-15: Mode 1, Hot Zero Power Results (15.4.6 Inadvertent Decrease in Boron Concentration in the Reactor Coolant System) 15.4-50

Table 15.4-16: Mode 2 Results (15.4.6 Inadvertent Decrease in Boron Concentration in the Reactor Coolant System) 15.4-51

Table 15.4-17: Not Used 15.4-52

Table 15.4-18: Mode 3 Results (15.4.6 Inadvertent Decrease in Boron Concentration in the Reactor Coolant System) 15.4-53

Table 15.4-19: Mode 5 Results (15.4.6 Inadvertent Decrease in Boron Concentration in the Reactor Coolant System) 15.4-54

LIST OF TABLES

Table 15.4-20: Inadvertent Loading and Operation of a Fuel Assembly in an Improper Position (15.4.7) - Limiting Analysis Results . . . 15.4-55
Table 15.4-21: Sequence of Events - Maximum Reactor Coolant System Pressure Case (15.4.8 Rod Ejection Accident) . . . 15.4-56
Table 15.4-22: REA Maximum Power Pulses and Reactivity Insertions . . . 15.4-57
Table 15.4-23: REA Fuel Temperatures and Enthalpies for Limiting S3K Cases . . . 15.4-58
Table 15.4-24: Spectrum of Rod Ejection Accidents (15.4.8) - Limiting Analysis Results . . . 15.4-59
Table 15.4-25: Not Used . . . 15.4-60
Table 15.4-26: Not Used . . . 15.4-61
Table 15.4-27: Not Used . . . 15.4-62
Table 15.4-28: Not Used . . . 15.4-63
Table 15.4-29: Not Used . . . 15.4-64
Table 15.4-30: Not Used . . . 15.4-65
Table 15.4-31: Key Inputs for Limiting Linear Heat Generation Rate Case (15.4.2 Uncontrolled Control Rod Assembly Withdrawal at Power) . . . 15.4-66
Table 15.4-32: Key Inputs for Limiting Linear Heat Generation Rate Case (15.4.3 Control Rod Misoperation, Single Control Rod Assembly Withdrawal). . . . 15.4-67
Table 15.4-33: Key Inputs for Limiting Control Rod Assembly Misalignment (15.4.3 Control Rod Misoperation, Control Rod Assembly Misalignment) . . . 15.4-68
Table 15.5-1: Sequence of Events CVCS Malfunction - Limiting SG Pressure (Pressurizer Spray Available). . . . 15.5-6
Table 15.5-2: Sequence of Events CVCS Malfunction - Limiting RCS Pressure (No Pressurizer Spray) . . . 15.5-7
Table 15.5-3: Sequence of Events CVCS Malfunction - Limiting MCHFR (No Pressurizer Spray) . . . 15.5-8
Table 15.5-4: Initial Conditions CVCS Malfunction . . . 15.5-9
Table 15.5-5: Summary of Results CVCS Malfunction . . . 15.5-10
Table 15.6-1: Failure of Lines Carrying Primary Coolant Outside Containment - Sequence of Events - Maximum Mass Release . . . 15.6-27
Table 15.6-2: Failure of Lines Carrying Primary Coolant Outside Containment - Sequence of Events - Maximum Iodine Spiking Time . . . 15.6-28
Table 15.6-3: Failure of Lines Carrying Primary Coolant Outside Containment - Sequence of Events - Maximum Reactor Pressure Vessel Pressure . . . 15.6-29
Table 15.6-4: Failure of Lines Carrying Primary Coolant Outside Containment - Inputs . . . 15.6-30

LIST OF TABLES

Table 15.6-5: Failure of Lines Carrying Primary Coolant Outside Containment - Results 15.6-31
Table 15.6-6: Steam Generator Tube Failure - Sequence of Events - Limiting Mass Release 15.6-32
Table 15.6-7: Steam Generator Tube Failure - Sequence of Events - Limiting Reactor Pressure Vessel Pressure 15.6-33
Table 15.6-8: Steam Generator Tube Failure - Sequence of Events - Limiting Steam Generator Pressure 15.6-34
Table 15.6-9: Steam Generator Tube Failure - Inputs 15.6-35
Table 15.6-10: Steam Generator Tube Failure - Results 15.6-36
Table 15.6-11: Steam Generator Tube Failure Inputs to Radiological Consequences 15.6-37
Table 15.6-12: Loss-of-Coolant Accident - Sequence of Events - Minimum Collapsed Level Above TAF 15.6-38
Table 15.6-13: Loss-of-Coolant Accident - Inputs 15.6-39
Table 15.6-14: Loss-of-Coolant Accident - Results Minimum Collapsed Level Above TAF 15.6-40
Table 15.6-15: Inadvertent Operation of an Emergency Core Cooling System Valve - Sequence of Events 15.6-41
Table 15.6-16: Inadvertent Operation of an Emergency Core Cooling System Valve - Inputs 15.6-42
Table 15.6-17: Inadvertent Operation of an Emergency Core Cooling System Valve - Results 15.6-43
Table 15.6-18: Loss-of-Coolant Analysis - Summary of Break Spectrum 15.6-44
Table 15.6-19: Loss-of-Coolant Analysis - Discharge Line Break Spectrum with All Electric Power Available 15.6-45
Table 15.6-20: Loss-of-Coolant Analysis - Injection Line Break Spectrum with All Electric Power Available 15.6-46
Table 15.6-21: Loss-of-Coolant Analysis - High Point Vent Line Break Spectrum with All Electric Power Available 15.6-47
Table 15.6-22: Loss-of-Coolant Analysis - Pressurizer Spray Supply Line Break Spectrum with All Electric Power Available 15.6-48
Table 15.6-23: Loss-of-Coolant Analysis - Five-percent Injection Line Break with Evaluation of Electric Power Available 15.6-49
Table 15.6-24: Loss-of-Coolant Analysis - Five-percent Injection Line Break with Loss of Normal AC Power Evaluation of Single Failure 15.6-50
Table 15.6-25: Steam Generator Tube Failure - Sequence of Events - Limiting Iodine Spiking Time 15.6-51
Table 15.9-1: Initial Conditions (100 Percent of Rated Power Cases) 15.9-11

LIST OF TABLES

Table 15.9-2: Initial Conditions (20 Percent of Rated Power Cases) . . . 15.9-12
Table 15.9-3: Initial Conditions (1 Percent of Rated Power Cases) . . . 15.9-13
Table 15.9-4: Normal Power Operation - Limiting Analysis Results (1 Percent of Rated Power Case) . . . 15.9-14
Table 15.9-5: Increase in Heat Removal by the Secondary System - Limiting Analysis Results (Rated Power Case) . . . 15.9-15
Table 15.9-6: Increase in Heat Removal by the Secondary System - Limiting Analysis Results (20 Percent of Rated Power Case) . . . 15.9-16
Table 15.9-7: Decrease in Heat Removal by the Secondary System - Limiting Analysis Results (Rated Power Case) . . . 15.9-17
Table 15.9-8: Decrease in Heat Removal by the Secondary System - Limiting Analysis Results (20 Percent of Rated Power Case) . . . 15.9-18
Table 15.9-9: Reactivity and Power Distribution Anomalies - Limiting Analysis Results (20 Percent of Rated Power Case) . . . 15.9-19
Table 15.10-1: Core Damage Source Term Release Timing . . . 15.10-5
Table 15.10-2: Core Inventory Release Fractions . . . 15.10-6
Table 15.10-3: Containment Aerosol Removal Rates . . . 15.10-7

LIST OF FIGURES

Figure 15.0-1: Control Rod Position vs. Time 15.0-72
Figure 15.0-2: SCRAM Reactivity Worth vs. Time 15.0-73
Figure 15.0-3: Schematic of the RADTRAD Model Nodalization 15.0-74
Figure 15.0-4: Schematic of the RADTRAD Model Nodalization for Steam Generator Tube Failure and Main Steam Line Break 15.0-75
Figure 15.0-5: Not Used 15.0-76
Figure 15.0-6: Not Used 15.0-77
Figure 15.0-7: Not Used 15.0-78
Figure 15.0-8: Nominal Return to Power Results - 65 Degree F Pool Temperature (15.0.6) 15.0-79
Figure 15.0-9: Analytical Operating Limits 15.0-80
Figure 15.0-10: Nominal Return to Power Results -140 Degree F Pool Temperature (15.0.6) 15.0-81
Figure 15.0-11: Not Used 15.0-82
Figure 15.0-12: Not Used 15.0-83
Figure 15.0-13: Not Used 15.0-84
Figure 15.0-14: Not Used 15.0-85
Figure 15.0-15: Not Used 15.0-86
Figure 15.0-16: Not Used 15.0-87
Figure 15.0-17: Not Used 15.0-88
Figure 15.0-18: Not Used 15.0-89
Figure 15.0-19: Not Used 15.0-90
Figure 15.1-1: Feedwater Temperature (15.1.1 Decrease in Feedwater Temperature) 15.1-50
Figure 15.1-2: Core Inlet Temperature (15.1.1 Decrease in Feedwater Temperature) 15.1-51
Figure 15.1-3: Core Inlet Density (15.1.1 Decrease in Feedwater Temperature) 15.1-52
Figure 15.1-4: Total Reactivity - Entire Transient (15.1.1 Decrease in Feedwater Temperature) 15.1-53
Figure 15.1-5: Reactor Power (15.1.1 Decrease in Feedwater Temperature) 15.1-54
Figure 15.1-6: Reactor Coolant System Pressure (15.1.1 Decrease in Feedwater Temperature) 15.1-55
Figure 15.1-7: Core Outlet Temperature (15.1.1 Decrease in Feedwater Temperature) 15.1-56
Figure 15.1-8: Main Steam System Pressure (15.1.1 Decrease in Feedwater Temperature) 15.1-57
Figure 15.1-9: Critical Heat Flux Ratio (15.1.1 Decrease in Feedwater Temperature) 15.1-58

LIST OF FIGURES

Figure 15.1-10: Reactor Coolant System Flow Rate (15.1.1 Decrease in Feedwater Temperature) 15.1-59

Figure 15.1-11: Feedwater Flow Rate (15.1.2 Increase in Feedwater Flow) 15.1-60

Figure 15.1-12: Core Inlet Temperature (15.1.2 Increase in Feedwater Flow) 15.1-61

Figure 15.1-13: Core Inlet Density (15.1.2 Increase in Feedwater Flow) 15.1-62

Figure 15.1-14: Total Core Reactivity (15.1.2 Increase in Feedwater Flow) 15.1-63

Figure 15.1-15: Reactor Power (15.1.2 Increase in Feedwater Flow) 15.1-64

Figure 15.1-16: Reactor Coolant System Pressure (15.1.2 Increase in Feedwater Flow) 15.1-65

Figure 15.1-17: Core Outlet Temperature (15.1.2 Increase in Feedwater Flow) 15.1-66

Figure 15.1-18: Main Steam System Pressure (15.1.2 Increase in Feedwater Flow) 15.1-67

Figure 15.1-19: Critical Heat Flux Ratio (15.1.2 Increase in Feedwater Flow) 15.1-68

Figure 15.1-20: Reactor Coolant System Flow (15.1.2 Increase in Feedwater Flow) 15.1-69

Figure 15.1-21: Steam Generator Level (15.1.2 Increase in Feedwater Flow) 15.1-70

Figure 15.1-22: Steam Flow (15.1.3 Increase in Steam Flow) 15.1-71

Figure 15.1-23: Core Inlet Temperature (15.1.3 Increase in Steam Flow) 15.1-72

Figure 15.1-24: Core Inlet Density (15.1.3 Increase in Steam Flow) 15.1-73

Figure 15.1-25: Total Core Reactivity (15.1.3 Increase in Steam Flow) 15.1-74

Figure 15.1-26: Reactor Power (15.1.3 Increase in Steam Flow) 15.1-75

Figure 15.1-27: Reactor Coolant System Pressure (15.1.3 Increase in Steam Flow) 15.1-76

Figure 15.1-28: Core Outlet Temperature (15.1.3 Increase in Steam Flow) 15.1-77

Figure 15.1-29: Main Steam System Pressure (15.1.3 Increase in Steam Flow) 15.1-78

Figure 15.1-30: Critical Heat Flux Ratio (15.1.3 Increase in Steam Flow) 15.1-79

Figure 15.1-31: Reactor Coolant System Flow (15.1.3 Increase in Steam Flow) 15.1-80

Figure 15.1-32: Break Flow Rate (15.1.5 Steam Piping Failure, Limiting Reactor Coolant System Pressure) 15.1-81

Figure 15.1-33: Reactor Coolant System Pressure (15.1.5 Steam Piping Failure, Limiting Reactor Coolant System Pressure) 15.1-82

Figure 15.1-34: Reactor Power (15.1.5 Steam Piping Failure, Limiting Reactor Coolant System Pressure) 15.1-83

Figure 15.1-35: Core Outlet Temperature (15.1.5 Steam Piping Failure, Limiting Reactor Coolant System Pressure) 15.1-84

Figure 15.1-36: Core Inlet Temperature (15.1.5 Steam Piping Failure, Limiting Reactor Coolant System Pressure) 15.1-85

LIST OF FIGURES

Figure 15.1-37: Maximum Steam Generator Pressure (15.1.5 Steam Piping Failure, Limiting Reactor Coolant System Pressure) . . . 15.1-86

Figure 15.1-38: Break Flowrate (15.1.5 Steam Piping Failure, Limiting Minimum Critical Heat Flux Ratio) . . . 15.1-87

Figure 15.1-39: Total Core Reactivity (15.1.5 Steam Piping Failure, Limiting Minimum Critical Heat Flux Ratio) . . . 15.1-88

Figure 15.1-40: Reactor Power (15.1.5 Steam Piping Failure, Limiting Minimum Critical Heat Flux Ratio) . . . 15.1-89

Figure 15.1-41: Reactor Coolant System Pressure (15.1.5 Steam Piping Failure, Limiting Minimum Critical Heat Flux Ratio) . . . 15.1-90

Figure 15.1-42: Volume Average Core Coolant Temperature (15.1.5 Steam Piping Failure, Limiting Minimum Critical Heat Flux Ratio) . . . 15.1-91

Figure 15.1-43: Failed Steam Generator Train Pressure (15.1.5 Steam Piping Failure, Limiting Minimum Critical Heat Flux Ratio) . . . 15.1-92

Figure 15.1-44: Critical Heat Flux Ratio (15.1.5 Steam Piping Failure, Limiting Minimum Critical Heat Flux Ratio) . . . 15.1-93

Figure 15.1-45: Break Flow Rate (15.1.5 Steam Piping Failure, Limiting Radiological) . . . 15.1-94

Figure 15.1-46: Integrated Break Flow (15.1.5 Steam Piping Failure, Limiting Radiological) . . . 15.1-95

Figure 15.1-47: Reactor Power (15.1.6 Containment Flooding) . . . 15.1-96

Figure 15.1-48: Reactor Component Cooling Water System Break Flow Rate (15.1.6 Containment Flooding) . . . 15.1-97

Figure 15.1-49: Reactor Pressure Vessel Heat Transfer (15.1.6 Containment Flooding) . . . 15.1-98

Figure 15.1-50: Containment Vessel Collapsed Liquid Level (15.1.6 Containment Flooding) . . . 15.1-99

Figure 15.1-51: Containment Evacuation System Evacuation Mass Flow (15.1.6 Containment Flooding) . . . 15.1-100

Figure 15.1-52: Containment Vessel Pressure (15.1.6 Containment Flooding) . . . 15.1-101

Figure 15.1-53: Critical Heat Flux Ratio (15.1.6 Loss of Containment Vacuum/Containment Flooding) . . . 15.1-102

Figure 15.1-54: Core Outlet Temperature (15.1.1 Decrease in Feedwater Temperature) . . . 15.1-103

Figure 15.1-55: Total Reactivity (15.1.1 Decrease in Feedwater Temperature) . . . 15.1-104

Figure 15.1-56: Steam Generator Pressure - Maximum SGS Pressure Case (15.1.2 Increase in Feedwater Flow) . . . 15.1-105

Figure 15.1-57: Not Used . . . 15.1-106

Figure 15.1-58: Not Used . . . 15.1-107

Figure 15.1-59: Not Used . . . 15.1-108

LIST OF FIGURES

Figure 15.1-60: Not Used 15.1-109

Figure 15.1-61: Intact Steam Generator Pressure (15.1.5 Steam Piping Failure, Limiting Steam Generator Pressure) 15.1-110

Figure 15.1-62: Faulted Steam Generator Pressure (15.1.5 Steam Piping Failure, Limiting Steam Generator Pressure) 15.1-111

Figure 15.2-1: Pressurizer Level - Peak RCS Pressure Case (15.2.1-15.2.3 LOEL-TT-LOCV) 15.2-68

Figure 15.2-2: Reactor Power - Peak RCS Pressure Case (15.2.1-15.2.3 LOEL-TT-LOCV) 15.2-69

Figure 15.2-3: Net Reactivity - Peak RCS Pressure Case (15.2.1-15.2.3 LOEL-TT-LOCV) 15.2-70

Figure 15.2-4: RCS Average Temperature - Peak RCS Pressure Case (15.2.1-15.2.3 LOEL-TT-LOCV) 15.2-71

Figure 15.2-5: RCS Flowrate - Peak RCS Pressure Case (15.2.1-15.2.3 LOEL-TT-LOCV) 15.2-72

Figure 15.2-6: DHRS Flow Typical - Peak RCS Pressure Case (15.2.1-15.2.3 LOEL-TT-LOCV) 15.2-73

Figure 15.2-7: Reactor Pressure Vessel Pressure - Peak RCS Pressure Case (15.2.1-15.2.3 LOEL-TT-LOCV) 15.2-74

Figure 15.2-8: Steam Generator Pressure Typical - Peak SG Pressure Case (15.2.1-15.2.3 LOEL-TT-LOCV) 15.2-75

Figure 15.2-9: Hot Channel Node MCHFR - Limiting MCHFR Case (15.2.1-15.2.3 LOEL-TT-LOCV) 15.2-76

Figure 15.2-10: Reactor Power - Peak RCS Pressure Case (15.2.4 MSIV Closure) 15.2-77

Figure 15.2-11: Reactor Pressure Vessel Lower Plenum Pressure - Peak RCS Pressure Case (15.2.4 MSIV Closure) 15.2-78

Figure 15.2-12: Volume Average Fuel Temperature - Peak RCS Pressure Case (15.2.4 MSIV Closure) 15.2-79

Figure 15.2-13: Reactor Coolant System Flowrate - Peak RCS Pressure Case (15.2.4 MSIV Closure) 15.2-80

Figure 15.2-14: Primary Temperatures - Peak RCS Pressure Case (15.2.4 MSIV Closure) 15.2-81

Figure 15.2-15: Pressurizer Level - Peak RCS Pressure Case (15.2.4 MSIV Closure) 15.2-82

Figure 15.2-16: Steam Generator Pressure (Typical) - Peak SG Pressure Case (15.2.4 MSIV Closure) 15.2-83

Figure 15.2-17: Steam Generator Level - Peak SG Pressure Case (15.2.4 MSIV Closure) 15.2-84

Figure 15.2-18: Minimum Critical Heat Flux Ratio - Limiting MCHFR Case (15.2.4 MSIV Closure) 15.2-85

Figure 15.2-19: Reactor Power - Peak RCS Pressure Case (15.2.6 Loss of AC Power) 15.2-86

Figure 15.2-20: Total Reactivity - Peak RCS Pressure Case (15.2.6 Loss of AC Power) 15.2-87

Figure 15.2-21: Pressurizer Level - Peak RCS Pressure Case (15.2.6 Loss of AC Power) 15.2-88

LIST OF FIGURES

Figure 15.2-22: Reactor Pressure Vessel Pressure - Peak RCS Pressure Case (15.2.6 Loss of AC Power) . . . 15.2-89

Figure 15.2-23: Reactor Coolant System Flowrate - Peak RCS Pressure Case (15.2.6 Loss of AC Power) . . . 15.2-90

Figure 15.2-24: Reactor Coolant System Average Temperature - Peak RCS Pressure Case (15.2.6 Loss of AC Power) . . . 15.2-91

Figure 15.2-25: Steam Generator Pressure - Peak SG Pressure Case (15.2.6 Loss of AC Power) . . . 15.2-92

Figure 15.2-26: Hot Channel Node Critical Heat Flux Ratio - MCHFR Case (15.2.6 Loss of AC Power) . . . 15.2-93

Figure 15.2-27: Reactor Power - Peak RCS Pressure Case (15.2.7 Loss of Feedwater) . . . 15.2-94

Figure 15.2-28: Reactor Pressure Vessel Pressure - Peak RCS Pressure Case (15.2.7 Loss of Feedwater) . . . 15.2-95

Figure 15.2-29: DHRS Flow - Peak RCS Pressure Case (15.2.7 Loss of Feedwater) . . . 15.2-96

Figure 15.2-30: Reactor Coolant System Flowrate - Peak RCS Pressure Case (15.2.7 Loss of Feedwater) . . . 15.2-97

Figure 15.2-31: Reactor Coolant System Temperature - Peak RCS Pressure Case (15.2.7 Loss of Feedwater) . . . 15.2-98

Figure 15.2-32: Pressurizer Level - Peak RCS Pressure Case (15.2.7 Loss of Feedwater) . . . 15.2-99

Figure 15.2-33: Steam Generator Inlet Pressure (Typical) - Peak SG Pressure Case (15.2.7 Loss of Feedwater) . . . 15.2-100

Figure 15.2-34: Minimum Critical Heat Flux Ratio - Limiting MCHFR Case (15.2.7 Loss of Feedwater) . . . 15.2-101

Figure 15.2-35: Reactor Pressure Vessel Lower Plenum Pressure - Peak RCS Pressure Case (15.2.8 Feedwater Line Break) . . . 15.2-102

Figure 15.2-36: Reactor Power - Peak RCS Pressure Case (15.2.8 Feedwater Line Break) . . . 15.2-103

Figure 15.2-37: Steam Generator Train 1 Pressure - Peak RCS Pressure Case (15.2.8 Feedwater Line Break) . . . 15.2-104

Figure 15.2-38: Steam Generator Train 2 Pressure - Peak RCS Pressure Case (15.2.8 Feedwater Line Break) . . . 15.2-105

Figure 15.2-39: Core Outlet Temperature - Peak RCS Pressure Case (15.2.8 Feedwater Line Break) . . . 15.2-106

Figure 15.2-40: Reactor Coolant System Flowrate - Peak RCS Pressure Case (15.2.8 Feedwater Line Break) . . . 15.2-107

Figure 15.2-41: Integrated Break Flow - Peak RCS Pressure Case (15.2.8 Feedwater Line Break) . . . 15.2-108

Figure 15.2-42: Steam Generator Train 1 Pressure - Peak SG Pressure Case (15.2.8 Feedwater Line Break) . . . 15.2-109

LIST OF FIGURES

Figure 15.2-43: Steam Generator Train 2 Pressure - Peak SG Pressure Case (15.2.8 Feedwater Line Break) 15.2-110

Figure 15.2-44: Hot Channel Node MCHFR - Limiting MCHFR Case (15.2.8 Feedwater Line Break) 15.2-111

Figure 15.2-45: DHRS Heat Removal - Limiting DHRS Case (15.2.8 Feedwater Line Break) 15.2-112

Figure 15.2-46: Steam Generator Train 1 Pressure - Limiting DHRS Case (15.2.8 Feedwater Line Break) 15.2-113

Figure 15.2-47: Steam Generator Train 2 Pressure - Limiting DHRS Case (15.2.8 Feedwater Line Break) 15.2-114

Figure 15.2-48: Reactor Coolant System Flowrate - Limiting DHRS Case (15.2.8 Feedwater Line Break) 15.2-115

Figure 15.2-49: Reactor Pressure Vessel Lower Plenum Pressure - Peak RCS Pressure Case (15.2.9 Inadvertent Operation of DHRS) 15.2-116

Figure 15.2-50: Core Inlet Temperature - Peak RCS Pressure Case (15.2.9 Inadvertent Operation of DHRS) 15.2-117

Figure 15.2-51: Core Outlet Temperature - Peak RCS Pressure Case (15.2.9 Inadvertent Operation of DHRS) 15.2-118

Figure 15.2-52: Reactor Coolant System Flow - Peak RCS Pressure Case (15.2.9 Inadvertent Operation of DHRS) 15.2-119

Figure 15.2-53: Total Reactivity - Peak RCS Pressure Case (15.2.9 Inadvertent Operation of DHRS) 15.2-120

Figure 15.2-54: Steam Plenum Pressure (Typical) - Peak SG Pressure Case (15.2.9 Inadvertent Operation of DHRS) 15.2-121

Figure 15.2-55: Minimum Critical Heat Flux Ratio - Limiting MCHFR Case (15.2.9 Inadvertent Operation of DHRS) 15.2-122

Figure 15.4-1: Reactor Power (15.4.1 Uncontrolled Control Rod Assembly Withdrawal from a Subcritical or Low Power Condition) 15.4-69

Figure 15.4-2: Net Reactivity (15.4.1 Uncontrolled Control Rod Assembly Withdrawal from a Subcritical or Low Power Condition) 15.4-70

Figure 15.4-3: Core Outlet Temperature (15.4.1 Uncontrolled Control Rod Assembly Withdrawal from a Subcritical or Low Power Condition) 15.4-71

Figure 15.4-4: Primary Pressure (15.4.1 Uncontrolled Control Rod Assembly Withdrawal from a Subcritical or Low Power Condition) 15.4-72

Figure 15.4-5: Not Used 15.4-73

Figure 15.4-6: Withdrawn CRA Reactivity Insertion (15.4.2 Uncontrolled Control Rod Assembly Withdrawal at Power) 15.4-74

LIST OF FIGURES

Figure 15.4-7: Reactor Power (15.4.2 Uncontrolled Control Rod Assembly Withdrawal at Power) 15.4-75

Figure 15.4-8: RCS Pressure for Limiting MCHFR Case (15.4.2 Uncontrolled Control Rod Assembly Withdrawal at Power) 15.4-76

Figure 15.4-9: Average RCS Temperature (15.4.2 Uncontrolled Control Rod Assembly Withdrawal at Power) 15.4-77

Figure 15.4-10: RCS Flow (15.4.2 Uncontrolled Control Rod Assembly Withdrawal at Power) 15.4-78

Figure 15.4-11: Critical Heat Flux Ratio (15.4.2 Uncontrolled Control Rod Assembly Withdrawal at Power) 15.4-79

Figure 15.4-12: Not Used 15.4-80

Figure 15.4-13: Withdrawn Control Rod Assembly Reactivity Insertion (15.4.3 Control Rod Misoperation, Single Control Rod Assembly Withdrawal) 15.4-81

Figure 15.4-14: Reactor Power (15.4.3 Control Rod Misoperation, Single Control Rod Assembly Withdrawal) 15.4-82

Figure 15.4-15: Reactor Coolant System (RPV Lower Plenum) Pressure (15.4.3 Control Rod Misoperation, Single Control Rod Assembly Withdrawal) 15.4-83

Figure 15.4-16: Total Core Reactivity (15.4.3 Control Rod Misoperation, Single Control Rod Assembly Withdrawal) 15.4-84

Figure 15.4-17: RCS Average Temperature (15.4.3 Control Rod Misoperation, Single Control Rod Assembly Withdrawal) 15.4-85

Figure 15.4-18: Reactor Coolant System Flow (15.4.3 Control Rod Misoperation, Single Control Rod Assembly Withdrawal) 15.4-86

Figure 15.4-19: Doppler Reactivity (15.4.3 Control Rod Misoperation, Single Control Rod Assembly Withdrawal) 15.4-87

Figure 15.4-20: Not Used 15.4-88

Figure 15.4-21: Dropped CRA Reactivity (15.4.3 Control Rod Misoperation, Control Rod Assembly Drop) 15.4-89

Figure 15.4-22: Absolute Power Rate (15.4.3 Control Rod Misoperation, Control Rod Assembly Drop) 15.4-90

Figure 15.4-23: Reactor Power (15.4.3 Control Rod Misoperation, Control Rod Assembly Drop) 15.4-91

Figure 15.4-24: RCS Pressure (RPV Lower Plenum) (15.4.3 Control Rod Misoperation, Control Rod Assembly Drop) 15.4-92

Figure 15.4-25: Average RCS Temperature (15.4.3 Control Rod Misoperation, Control Rod Assembly Drop) 15.4-93

Figure 15.4-26: Reactor Coolant System Flow (15.4.3 Control Rod Misoperation, Control Rod Assembly Drop) 15.4-94

LIST OF FIGURES

Figure 15.4-27: Doppler Reactivity (15.4.3 Control Rod Misoperation, Control Rod Assembly drop) . . . 15.4-95
Figure 15.4-28: Spectrum of Fuel Misload Arrangements (15.4.7 Inadvertent Loading and Operation of a Fuel Assembly in an Improper Position) . . . 15.4-96
Figure 15.4-29: REA Peak Power Pulse at 25% Initial Power, EOC (15.4.8 Rod Ejection Accident) . . . 15.4-97
Figure 15.4-30: REA Peak Power Pulse at 70% Initial Power, MOC (15.4.8 Rod Ejection Accident) . . . 15.4-98
Figure 15.4-31: REA Peak Power Pulse at 70% Initial Power, BOC (15.4.8 Rod Ejection Accident) . . . 15.4-99
Figure 15.4-32: Not Used . . . 15.4-100
Figure 15.4-33: Reactor Coolant System Pressure (15.4.8 Rod Ejection Accident) . . . 15.4-101
Figure 15.4-34: Not Used . . . 15.4-102
Figure 15.4-35: Critical Heat Flux Ratio (15.4.3 Control Rod Misoperation, Single Control Rod Assembly Withdrawal) . . . 15.4-103
Figure 15.4-36: Critical Heat Flux Ratio (15.4.3 Control Rod Misoperation, Control Rod Assembly Drop) . . . 15.4-104
Figure 15.4-37: Critical Heat Flux Ratio (15.4.8 Rod Ejection Accident) . . . 15.4-105
Figure 15.5-1: Pressurizer Level - Increase in RCS Inventory (No PZR Spray) . . . 15.5-11
Figure 15.5-2: Reactor Power - Increase in RCS Inventory (No PZR Spray) . . . 15.5-12
Figure 15.5-3: Total Reactivity - Increase in RCS Inventory (No PZR Spray) . . . 15.5-13
Figure 15.5-4: Reactor Coolant System Average Temperature - Increase in RCS Inventory (No PZR Spray) . . . 15.5-14
Figure 15.5-5: Makeup Flowrate - Increase in RCS Inventory (No PZR Spray) . . . 15.5-15
Figure 15.5-6: Reactor Coolant System Flow - Increase in RCS Inventory (No PZR Spray) . . . 15.5-16
Figure 15.5-7: Decay Heat Removal System Flow (Typical) - Increase in RCS Inventory (No PZR Spray) . . . 15.5-17
Figure 15.5-8: Reactor Pressure Vessel Lower Plenum Pressure - Increase in RCS Inventory (No PZR Spray) . . . 15.5-18
Figure 15.5-9: Steam Generator Pressure (Typical) - Increase in RCS Inventory (PZR Spray Available) . . . 15.5-19
Figure 15.5-10: Minimum Critical Heat Flux Ratio - Increase in RCS Inventory (No PZR Spray) . . . 15.5-20
Figure 15.6-1: Failure of Lines Carrying Primary Coolant Outside Containment - Maximum Mass Release Scenario Break Mass Flow . . . 15.6-52

LIST OF FIGURES

Figure 15.6-2: Failure of Lines Carrying Primary Coolant Outside Containment - Maximum Mass Release Scenario - Integrated Break Mass Flow 15.6-53

Figure 15.6-3: Failure of Lines Carrying Primary Coolant Outside Containment - Maximum Reactor Pressure Vessel Pressure Scenario - Pressure 15.6-54

Figure 15.6-4: Failure of Lines Carrying Primary Coolant Outside Containment - Maximum Reactor Pressure Vessel Pressure Scenario - Pressure 15.6-55

Figure 15.6-5: Failure of Lines Carrying Primary Coolant Outside Containment - Maximum Reactor Pressure Vessel Pressure Scenario - Reactor Power 15.6-56

Figure 15.6-6: Failure of Lines Carrying Primary Coolant Outside Containment - Maximum Reactor Pressure Vessel Pressure Scenario - Decay Heat Removal System Flow 15.6-57

Figure 15.6-7: Failure of Lines Carrying Primary Coolant Outside Containment - Maximum Reactor Pressure Vessel Pressure Scenario - Reactor Coolant System Temperatures 15.6-58

Figure 15.6-8: Failure of Lines Carrying Primary Coolant Outside Containment - Maximum Reactor Pressure Vessel Pressure Scenario- Reactor Coolant System Temperatures 15.6-59

Figure 15.6-9: Failure of Lines Carrying Primary Coolant Outside Containment - Maximum Reactor Pressure Vessel Pressure Scenario - Pressurizer Level. 15.6-60

Figure 15.6-10: Failure of Lines Carrying Primary Coolant Outside Containment - Maximum Reactor Pressure Vessel Pressure Scenario - Pressurizer Level. 15.6-61

Figure 15.6-11: Failure of Lines Carrying Primary Coolant Outside Containment - Maximum Reactor Pressure Vessel Pressure Scenario - Level above the Top of the Core 15.6-62

Figure 15.6-12: Failure of Lines Carrying Primary Coolant Outside Containment - Maximum Reactor Pressure Vessel Pressure Scenario - Fuel Volume Average Temperature 15.6-63

Figure 15.6-13: Failure of Lines Carrying Primary Coolant Outside Containment - Maximum Reactor Pressure Vessel Pressure Scenario - Fuel Volume Average Temperature 15.6-64

Figure 15.6-14: Failure of Lines Carrying Primary Coolant Outside Containment - Maximum Reactor Pressure Vessel Pressure Scenario - Reactor Coolant System Mass Flow. 15.6-65

Figure 15.6-15: Failure of Lines Carrying Primary Coolant Outside Containment - Maximum Reactor Pressure Vessel Pressure Scenario - Reactor Coolant System Mass Flow. 15.6-66

Figure 15.6-16: Failure of Lines Carrying Primary Coolant Outside Containment - Maximum Reactor Pressure Vessel Pressure Scenario - Total Reactivity 15.6-67

Figure 15.6-17: Steam Generator Tube Failure - Limiting Mass Release Scenario - Break Mass Flow 15.6-68

LIST OF FIGURES

Figure 15.6-18: Steam Generator Tube Failure - Limiting Mass Release Scenario - Mass Release to Steam Generator before Steam Generator Isolation 15.6-69

Figure 15.6-19: Steam Generator Tube Failure - Limiting Mass Release Scenario - Steam Generator Level 15.6-70

Figure 15.6-20: Steam Generator Tube Failure - Limiting Mass Release Scenario - Steam Generator Level 15.6-71

Figure 15.6-21: Steam Generator Tube Failure - Limiting Mass Release Scenario - RCS and SG Pressure 15.6-72

Figure 15.6-22: Steam Generator Tube Failure - Limiting Reactor Pressure Vessel Pressure Scenario - Reactor Pressure Vessel and Steam Generator Pressures 15.6-73

Figure 15.6-23: Steam Generator Tube Failure – Limiting Reactor Pressure Vessel Pressure Scenario – Reactor Pressure Vessel and Steam Generator Pressures 15.6-74

Figure 15.6-24: Steam Generator Tube Failure – Limiting Reactor Pressure Vessel Pressure Scenario – Core and Steam Generator Power 15.6-75

Figure 15.6-25: Steam Generator Tube Failure – Limiting Reactor Pressure Vessel Pressure Scenario – Core and Steam Generator Power 15.6-76

Figure 15.6-26: Steam Generator Tube Failure – Limiting Reactor Pressure Vessel Pressure Scenario – Decay Heat Removal System Flow 15.6-77

Figure 15.6-27: Steam Generator Tube Failure – Limiting Reactor Pressure Vessel Pressure Scenario – Decay Heat Removal System Flow 15.6-78

Figure 15.6-28: Steam Generator Tube Failure – Limiting Reactor Pressure Vessel Pressure Scenario – Steam Generator Level 15.6-79

Figure 15.6-29: Steam Generator Tube Failure – Limiting Iodine Spiking Time - Mass Release to SG 15.6-80

Figure 15.6-30: Steam Generator Tube Failure – Limiting Iodine Spiking Time - SG Levels 15.6-81

Figure 15.6-31: Steam Generator Tube Failure – Limiting Reactor Pressure Vessel Pressure Scenario – Reactor Coolant System Temperature 15.6-82

Figure 15.6-32: Steam Generator Tube Failure – Limiting Reactor Pressure Vessel Pressure Scenario – Reactor Coolant System Temperature 15.6-83

Figure 15.6-33: Steam Generator Tube Failure – Limiting Reactor Pressure Vessel Pressure Scenario – Pressurizer Level 15.6-84

Figure 15.6-34: Steam Generator Tube Failure – Limiting Reactor Pressure Vessel Pressure Scenario – Pressurizer Level 15.6-85

Figure 15.6-35: Steam Generator Tube Failure – Limiting Reactor Pressure Vessel Pressure Scenario – Level above the Top of the Core 15.6-86

LIST OF FIGURES

Figure 15.6-36: Steam Generator Tube Failure – Limiting Reactor Pressure Vessel Pressure Scenario – Fuel Volume Average Temperature 15.6-87

Figure 15.6-37: Steam Generator Tube Failure – Limiting Reactor Pressure Vessel Pressure Scenario – Integrated Mass Release to SG 15.6-88

Figure 15.6-38: Steam Generator Tube Failure – Limiting SG Pressure Case - RCS and SG Pressures 15.6-89

Figure 15.6-39: Steam Generator Tube Failure – Limiting SG Pressure Case - DHRS Flow 15.6-90

Figure 15.6-40: Steam Generator Tube Failure – Limiting SG Pressure Case - Primary Temperatures 15.6-91

Figure 15.6-41: Loss of Coolant Accident - Break Flow 15.6-92

Figure 15.6-42: Loss of Coolant Accident - Reactor Pressure Vessel and Containment Pressure 15.6-93

Figure 15.6-43: Loss of Coolant Accident - Reactor Power 15.6-94

Figure 15.6-44: Loss of Coolant Accident - Reactivity 15.6-95

Figure 15.6-45: Loss of Coolant Accident - Collapsed Liquid Level Above Active Core 15.6-96

Figure 15.6-46: Loss of Coolant Accident - Reactor Coolant System Average Temperature 15.6-97

Figure 15.6-47: Loss of Coolant Accident - Reactor Coolant System Temperatures 15.6-98

Figure 15.6-48: Loss of Coolant Accident - Reactor Vent Valve Flow 15.6-99

Figure 15.6-49: Loss of Coolant Accident - Reactor Recirculation Valve Flow 15.6-100

Figure 15.6-50: Loss of Coolant Accident - Reactor Coolant System Flowrate 15.6-101

Figure 15.6-51: Loss of Coolant Accident - Steam Generator Pressure 15.6-102

Figure 15.6-52: Loss of Coolant Accident - Containment Wall Temperature 15.6-103

Figure 15.6-53: Loss of Coolant Accident - Fuel Volume Average Temperature 15.6-104

Figure 15.6-54: Loss of Coolant Accident – MCHFR 15.6-105

Figure 15.6-55: Inadvertent Operation of an Emergency Core Cooling System Valve – Reactor Power 15.6-106

Figure 15.6-56: Inadvertent Operation of an Emergency Core Cooling System Valve – Inadvertent Reactor Recirculation Valve Flow 15.6-107

Figure 15.6-57: Inadvertent Operation of an Emergency Core Cooling System Valve – Pressures 15.6-108

Figure 15.6-58: Inadvertent Operation of an Emergency Core Cooling System Valve – Reactor Coolant System Flow 15.6-109

Figure 15.6-59: Inadvertent Operation of an Emergency Core Cooling System Valve – Reactor Coolant System Flow 15.6-110

LIST OF FIGURES

Figure 15.6-60: Inadvertent Operation of an Emergency Core Cooling System Valve – Reactor Power 15.6-111

Figure 15.6-61: Inadvertent Operation of an Emergency Core Cooling System Valve – Steam Generator Pressure 15.6-112

Figure 15.6-62: Inadvertent Operation of an Emergency Core Cooling System Valve – Collapsed Liquid Level Above Active Core 15.6-113

Figure 15.6-63: Inadvertent Operation of an Emergency Core Cooling System Valve – Reactor Coolant System Temperature. 15.6-114

Figure 15.6-64: Inadvertent Operation of an Emergency Core Cooling System Valve – Reactor Coolant System Temperature. 15.6-115

Figure 15.6-65: Inadvertent Operation of an Emergency Core Cooling System Valve – Fuel Volume Average Temperature 15.6-116

Figure 15.6-66: Inadvertent Operation of an Emergency Core Cooling System Valve – Fuel Cladding Temperature 15.6-117

Figure 15.6-67: Inadvertent Operation of an Emergency Core Cooling System Valve – Critical Heat Flux Ratio 15.6-118

Figure 15.6-68: Inadvertent Operation of an Emergency Core Cooling System Valve – Critical Heat Flux Ratio 15.6-119

Figure 15.9-1: Illustration of Decay Ratio versus Riser Subcooling 15.9-20

Figure 15.9-2: Time Trace of Primary Coolant Flow Response to a Perturbation at 1 Percent of Rated Power and Beginning of Cycle Reactivity 15.9-21

Figure 15.9-3: Time Trace of Heat Addition and Heat Removal Response to a Perturbation at 1 Percent of Rated Power and Beginning of Cycle Reactivity 15.9-22

Figure 15.9-4: Time Trace of Primary Coolant Flow Response to an Increase in Heat Removal by the Secondary System at Rated Power and End of Cycle Reactivity 15.9-23

Figure 15.9-5: Time Trace of Heat Addition and Heat Removal Response to an Increase in Heat Removal by the Secondary System at Rated Power and End of Cycle Reactivity 15.9-24

Figure 15.9-6: Time Trace of Primary Coolant Flow Response to an Increase in Heat Removal by the Secondary System at 20 Percent of Rated Power and End of Cycle Reactivity 15.9-25

Figure 15.9-7: Time Trace of Heat Addition and Heat Removal Response to an Increase in Heat Removal by the Secondary System at 20 Percent of Rated Power and End of Cycle Reactivity 15.9-26

Figure 15.9-8: Time Trace of Primary Coolant Flow Response to a Decrease in Heat Removal by the Secondary System at Rated Power and End of Cycle Reactivity 15.9-27

LIST OF FIGURES

Figure 15.9-9: Time Trace of Heat Addition and Heat Removal Response to a Decrease in Heat Removal by the Secondary System at Rated Power and End of Cycle Reactivity . . . 15.9-28

Figure 15.9-10: Time Trace of Primary Coolant Flow Response to a Decrease in Heat Removal by the Secondary System at 20 Percent of Rated Power and End of Cycle Reactivity . . . 15.9-29

Figure 15.9-11: Time Trace of Heat Addition and Heat Removal Response to a Decrease in Heat Removal by the Secondary System at 20 Percent of Rated Power and End of Cycle Reactivity . . . 15.9-30

Figure 15.9-12: Time Trace of Primary Coolant Flow Response to Reactivity and Power Distribution Anomalies at 20 Percent of Rated Power and End of Cycle Reactivity . . . 15.9-31

Figure 15.9-13: Time Trace of Heat Addition and Heat Removal Response to Reactivity and Power Distribution Anomalies at 20 Percent of Rated Power and End of Cycle Reactivity . . . 15.9-32

Figure 15.9-14: Time Trace of Pressure Boundary Function Representing the Effect of Decrease in Reactor Coolant Inventory . . . 15.9-33

Figure 15.9-15: Time Trace of Flow Response to a Decrease in Reactor Coolant Inventory at 100 Percent of Rated Power and Beginning of Cycle Reactivity . . . 15.9-34

Figure 15.9-16: Time Trace of Power Response to a Decrease in Reactor Coolant Inventory at 100 Percent of Rated Power and Beginning of Cycle Reactivity . . . 15.9-35

Figure 15.9-17: Time Trace of CHFR Response to a Decrease in Reactor Coolant Inventory at 100 Percent of Rated Power and Beginning of Cycle Reactivity . . . 15.9-36

CHAPTER 15 TRANSIENT AND ACCIDENT ANALYSES

15.0 Transient and Accident Analyses

15.0.0 Classification and Key Assumptions

This subsection addresses the classification of potential plant events that are considered in the NuScale Power, LLC design basis. This section also identifies key assumptions used in the design basis event (DBE) evaluations including initial conditions, trip setpoints, and the limiting failure identification process.

As described in Chapter 1, the NuScale Power Module (NPM) is a natural circulation pressurized water reactor (PWR) with steam generators (SGs) integral to the reactor vessel. Many of the events analyzed for traditional PWRs are applicable to the NuScale Power Plant design, while some events are not applicable. However, it is also necessary to consider the specifics of the NuScale Power Plant design to evaluate a sufficiently broad spectrum of transients, postulated accidents, and initiating events in the design basis analyses presented in Chapter 15.

15.0.0.1 Initiating Event Selection

Initiating events are considered for internal events occurring in a single NPM while at power. Design basis analyses consider the range of power operation, especially for events where reactivity feedback is important, because events from a low-power or zero-power state could be more limiting than from full-power conditions. Hazards such as floods and fires in the plant, earthquakes, severe weather conditions, external fires, and external floods are evaluated in Chapters 3, 9, and 19. Section 19.1 and Chapter 21 discuss the consideration of multi-module events.

Initiating events are identified by considering the similarities and differences of the NuScale Power Plant design to current generation plants. Many DBEs are the same as current generation plants. Other events reflect a unique NuScale Power Plant design feature, e.g., loss of containment vacuum is a contributor to the "increase in secondary system heat removal" event type. The probabilistic risk assessment (PRA) in Section 19.1 evaluates the risk associated with the operation of a single NPM at full power as well as at low-power and shutdown modes of operation for internal and external initiating events. This broad scope allows a wide spectrum of events to be considered as possible DBEs. Therefore, several sources, including the PRA analyses, are included to identify the subset of possible DBEs. The events are categorized into one of seven categories:

1) increase in heat removal by the secondary system

2) decrease in heat removal by the secondary system

3) decrease in reactor coolant system (RCS) flow rate

4) reactivity and power distribution anomalies

5) increase in reactor coolant inventory

6) decrease in reactor coolant inventory

7) radioactive release from a subsystem or component

Table 15.0-1 lists the events selected for evaluation in Sections 15.1 through 15.7 and a list of computer codes used for analyzing each event. Special events, such as the beyond-design-basis core damage event (CDE), are also included in Table 15.0-1.

15.0.0.2 Design Basis Event Classification

NuScale DBEs are classified by frequency of occurrence, including those events that are expected to occur within the NPM lifetime as well as those that are postulated but not expected to occur during the NPM lifetime. The NuScale DBE spectrum is developed by considering DBEs associated with current generation plants and unique events resulting from NuScale Power Plant design features, including review of PRA initiators. This approach ensures the design considers a broad spectrum of potential events. Classification by frequency of occurrence is used to assign the analysis acceptance criteria for the event.

The set of DBEs establishes the design adequacy of the NPM and NuScale Power Plant to limit radiological releases below regulatory guidelines.

15.0.0.2.1 Classification by Event Frequency and Type

Design basis event classification by frequency is based on three distinct categories:

- anticipated operational occurrences (AOOs)
- infrequent events (IEs)
- postulated accidents

Events that are expected to occur one or more times during an NPM lifetime are classified as AOOs. Events that are not expected to occur during an NPM lifetime are classified as IEs or postulated accidents or may be conservatively classified as AOOs. In general, events that are not considered to be within the design basis are evaluated in Chapter 19; however, those beyond design basis events (BDBEs) that are explicitly defined by regulation are addressed in this chapter. These events are termed special events. For example, the CDE described in Section 15.10 is a special event.

Special events also encompass defense-in-depth and common cause failures (CCFs) of digital control systems, as described in Branch Technical Position 7-19. The IE category accommodates the anticipated lower frequency of NuScale event occurrence that results from the unique design features. These features include digital control systems that are redundant and fault tolerant. Infrequent events are also considered to occur assuming worst case single-failure or single-operator error. Digital control system errors caused by a single instance error are treated in this chapter as AOOs. The multiple instance events (i.e., CCFs) are treated as BDBEs,

and addressed with Branch Technical Position 7-19 realistic methods and alternate acceptance criteria as described in Section 7.1.

The NuScale Power Plant design life is 60 years and the criterion "one or more times in NPM life" is conservatively interpreted as including any transient with a frequency of 1E-2 per year or more. Because of the increased reliability of plant systems, notably the digital control system, many event initiators traditionally categorized as AOOs have a frequency of occurrence longer than an NPM lifetime. To recognize this characteristic of advanced plant design, the classification of IE is used to identify transients that have a frequency of less than 1E-2 per year. Design basis accidents that have very low frequencies and are not expected to occur during plant lifetime are used to establish design criteria for safety-related structures, systems, and components (SSC). The IE category accounts for nonsafety-related design features that reduce the occurrence of initiating events. Lowering the initiating event frequency reduces the probability of initiating events that can lead to more serious transients and accidents. Refer to Chapter 19 for discussions regarding the reliability of nonsafety-related equipment that contribute to the event frequency.

The single-NPM events are also categorized both by the thermal-hydraulic effects on the NPM and the radiological consequences. These basic conditions relate to the capability to remove heat from the RCS and retain radionuclide inventory within the fission product barriers. The conditions apply to the current generation of light water reactor (LWR) plants and remain applicable to the NuScale Power Plant design. The associated plant systems may be similar to those used in current generation plants or may differ; a notable difference is the use of natural circulation rather than an active system to remove heat from the core area. Infrequent events have more restrictive radiological acceptance criteria than postulated accidents to maintain the overall product of risk times consequence approximately constant.

To establish the appropriate event frequency for NuScale DBEs, typical data bases and industry event classifications are examined for applicability to the NuScale Power Plant design. The initiating event frequency is only calculated for a small subset of the AOOs. In most cases, AOO consequences are too low to warrant a quantification of the event frequency. The event classification is simplified by substituting a deterministic classification where the consequences are small. Quantification of frequency is performed with NPM-specific PRA data to inform event classification only where the design is unique or where event consequences are large enough to warrant a quantification of the event frequency.

15.0.0.2.2 Acceptance Criteria

Acceptance criteria relate to the plant thermal-hydraulic and neutronic response and to potential radiological consequence associated with a DBE. The DBEs that have a higher occurrence frequency have more restrictive acceptance criteria, while those events with a low-occurrence frequency have less restrictive acceptance criteria. Table 15.0-2 provides thermal-hydraulic acceptance criteria. Thermal-hydraulic acceptance criteria for the rod ejection accident and loss-of-coolant accidents are provided in Table 15.0-3 and Table 15.0-4, respectively. Table 15.0-5 provides radiological acceptance criteria.

These acceptance criteria provide the bases for conclusions regarding the integrity of the three radiological barriers (fuel cladding, reactor coolant pressure boundary, and containment) and whether the occurrence of a DBE can generate a more serious condition.

15.0.0.3 Licensing Methodology

The set of DBEs in Table 15.0-1 establishes the capability of the NPM to limit radiological releases within regulatory guidelines. The analyses of these events are performed with assumptions about plant equipment availability, combinations with external events, acceptance criteria and other parameter information necessary to meet the requirements of the General Design Criteria (GDC) of 10 CFR 50, Appendix A. These conditions, assumptions and conclusions establish the acceptability of the evaluation models used for analysis as discussed in Section 15.0.2. The GDCs used in developing these conditions, assumptions and conclusions are discussed in Section 3.1.

The application of the GDCs and PDCs described in Section 3.1 establishes the SSC available for event mitigation of a DBE.

15.0.0.4 Initial Conditions

The NPM parameters that are used in DBE evaluation are identified in Table 15.0-6 and in the Chapter 15 sections detailing the analyzed events. These parameters represent limiting analysis conditions and, unless otherwise noted in a specific evaluation, are common to the evaluation of the DBEs evaluated in this chapter. Parameter uncertainty to bound the range of potential plant conditions at the start of an event is also identified. In general, limiting conditions are associated with full power operation because the energy available from fission, sensible heat and decay heat represent the largest challenge to fission product barriers at this power level. However, the spectrum of initial power conditions and time in life are considered for each event to ensure that limiting initial conditions have been identified.

The module protection system (MPS) is credited in the DBE analyses. Table 15.0-7 lists the signals, the analytical limits, and their associated time delays used in the Chapter 15 analyses.

The NPM response to DBEs is dependent on core power distribution and reactivity characteristics provided by the moderator temperature coefficient and the fuel (Doppler) coefficient. Section 4.3.2 provides details about the calculation of core coefficients and power distributions. Table 15.0-8 provides the reactivity coefficients used in each DBE evaluation. Table 4.3-2 provides the values calculated for the core equilibrium design.

The spectrum of plant conditions is evaluated for development of the radiological source term presented in Section 15.0.3.

15.0.0.5 Limiting Single Failures

10 CFR 50, Appendix A describes a single failure as an occurrence that results in the loss of capability of a component to perform its intended safety functions. Multiple failures resulting from a single occurrence are considered to be a single failure. A component that changes position or state to achieve its safety function is considered an "active" component while a component that does not change position or state to achieve its safety function is considered a "passive" component.

The principal considerations in applying the single-failure criterion to NPM design basis event evaluations are discussed below.

1) Active failures are considered for mechanical components.

- Design basis event mitigation credits valves that are classified as safety related. Valves move to their safety or "fail safe" position when the externally-applied motive force is removed.
- Check valves in the feedwater system are used to mitigate the consequences of a DBE.

There is one safety-related check valve and one nonsafety-related backup check valve in each feedwater line (four total check valves per NPM). The feedwater system check valves are credited to mitigate the consequences of the feedwater line break event.

The feedwater system check valves are not credited for containment isolation but for short-term retention of decay heat removal system (DHRS) inventory, until the feedwater isolation valve (FWIV) or its backup (nonsafety-related) feedwater regulating valve closes. The FWIV performs the containment isolation function and the feedwater regulating valve serves as a backup to the FWIV.

2) Passive failure of a single SSC is considered a potential event initiator, but not as a single failure in the short term.

- Passive failures of fluid systems are considered only on a long-term basis except for check valves whose failure must be postulated coincident with its required response to a DBE.
- For the purpose of considering passive single failures, the short term is defined as the period up to 24 hours following an initiating event.
- Components whose proper function has been demonstrated and documented are not considered a credible single failure (e.g., American Society of Mechanical Engineers code safety valves).
- The ECCS valve inadvertent actuation block (IAB) is a sub-component feature of the ECCS valve. The IAB feature prevents the ECCS valves from opening until RCS pressure drops below the IAB threshold established by the IAB opening spring. The IAB operates based on stored energy resulting from the differential pressure between the RCS and containment. The IAB requires no AC or DC power, actuation signals or external motive force to maintain the ECCS valves closed. The IAB provides a simple safety-related mechanical backup to avoid inadvertent actuation of an ECCS valve, in the event DC power to the trip valve

solenoid or the solenoid itself fail and depressurize the trip line. As such, the IAB closing function has been determined not to be subject to consideration as a single active failure for Chapter 15 analyses. This determination is consistent with the position in Reference 15.0-12.

3) Active and passive failures are considered for electrical components.

- Protective actions must be accomplished in the presence of a single detectable failure. The effects of non-detectable failures are considered concurrently as part of the most-limiting single failure.
- Both loss of and availability of power is assumed in the analysis of each event and are not considered as a single failure.

4) Single human errors are considered. In this regard, a key consideration is whether the potential error is an error of "omission" or an error of "commission." An error of omission is one in which an operator action is required to achieve a safety function, but the operator fails to perform the function. A error of commission is one in which no operator action is required, but an erroneous action is taken.

- Operator errors are considered in identifying event initiators. The NuScale Power Plant design limits operator errors to consequences that are no more severe than the worst-case single failure.

  Therefore, there are no operator errors that have to be analyzed in the accident analysis. Because the NuScale Power Plant is a passive plant, no operator actions are required for 72 hours. Operator actions allowed by procedure make the consequences less severe. Failure to take one of these actions cannot make the consequences worse than the bounding Chapter 15 analysis.
- Multiple operator errors or errors that result in common mode failures are beyond design basis. These events are analyzed in Chapter 19.

5) The effect of the single failure on a particular transient is assessed in the context of the non-LOCA transient calculation. The assessment is qualitative, with the use of engineering assessment, quantitative with the use of sensitivity calculations, or both. The assessment considers applicable acceptance criteria. For a particular transient, the limiting single failure for one acceptance criterion may be different than the limiting single failure for a different acceptance criterion for the same DBE. The limiting single failures for Chapter 15 events are described with the event analysis and are identified in Table 15.0-9.

15.0.0.6 Equipment Response and Physical Parameter Assumptions

The following sections address control rod assembly insertion characteristics, decay heat, engineered safety feature characteristics and required operator actions.

15.0.0.6.1 Control Rod Assembly Insertion Characteristics

The time for inserting control rods directly affects the amount of heat that has to be removed from the core in response to a DBE. Section 4.3 describes the analytical basis for the control rod assembly insertion rates and reactivity effect as a function

of time. The analyses for the Chapter 15 DBEs apply additional conservatism to the reactivity insertion rate provided in Section 4.3 to bound potential plant conditions. Figure 15.0-1 shows the normalized control rod position versus time, and Figure 15.0-2 shows the normalized SCRAM reactivity worth versus time. The use of bounding insertion times provides conservative results for DBE analyses. Drop time testing requirements are specified in the Technical Specifications.

15.0.0.6.2 Decay Heat

Bounding values for decay heat are designated to represent the maximum decay heat of the core following an event. The 1973 ANS decay heat standard is used in NRELAP5 to represent bounding decay heat. The LOCA methodology calculates fission product decay heat using a bounding form of the 1973 American Nuclear Society (ANS) decay heat standard with a 20 percent uncertainty added to the base value. A bounding form of the 1973 ANS standard in NRELAP5 is conservative relative to the 1971 ANS standard specified in 10 CFR 50, Appendix K. The NuScale design supports an exemption from 10 CFR 50, Appendix K. See Reference 15.0-3 for additional information.

For non-LOCAs, the model also uses the conservative 1973 ANS decay heat standard, which is varied by utilizing different decay heat multipliers and specifying whether or not to include the actinide contribution.

The following decay heat contribution values are examples for the NuScale Power Module. A review of the applicable core physics parameters is performed each cycle to confirm the bounding nature of the values utilized for the non-LOCA transient analyses.

- Minimum = use multiplier of 0.8 while excluding the actinide contribution
- Maximum = use multiplier of 1.0 while including the actinide contribution

15.0.0.6.3 Engineered Safety Features Characteristics

NuScale engineered safety feature systems include the containment systems (Section 6.2), ECCS (Section 6.3), and DHRS (Section 5.4.3). The DHRS provides cooling for non-LOCA design basis events when normal secondary-side cooling is unavailable or otherwise not used. The DHRS is designed to remove post-reactor trip residual and core decay heat from operating conditions and transition the NPM to safe shutdown conditions without reliance on external power. Section 5.4.3 provides additional description of the DHRS.

In conjunction with the containment heat removal function of containment, the ECCS provides a means of core decay heat removal for LOCAs that exceed makeup capability or during loss of both trains of the DHRS. The DHRS provides an additional capacity to remove decay heat during the initial blowdown period of a LOCA, but is not credited in the Appendix K LOCA model.

The ECCS, in conjunction with the containment vessel (CNV), has unique design provisions that ensure sufficient coolant inventory is retained to maintain the core covered and cooled. The ECCS consists of three reactor vent valves (RVVs) mounted

on the head of the reactor pressure vessel (RPV) and two reactor recirculation valves (RRVs) mounted on the side of the RPV in the downcomer region at a height above the core, as shown in Figure 6.3-1. All five valves are closed during normal operation and open to actuate the system during accident conditions. Upon ECCS actuation, the RVVs and the RRVs open creating a steam flow path from the pressurizer to the containment, and an RPV downcomer flow path to and from containment. Water that is vaporized in the core leaves as steam through the RVVs, is condensed and collected in the CNV, and is returned to the downcomer region inside the RPV through the RRVs. The CNV is sized such that the displacement of liquid from the RPV into containment establishes a liquid level above the RRVs, establishing the natural circulation loop. The natural circulation loop removes decay and residual heat from the core and RPV into the containment. Heat in the containment is then transferred by conduction and convection to the water in the reactor pool. Because the ECCS does not replace or add inventory after a LOCA, it does not require boron addition to maintain reactivity control caused by the addition of an external source of water. Section 15.6.5 provides additional information on the NPM response during a LOCA.

The ECCS valves and the DHRS do not rely on electrical power or on nonsafety-related support systems for actuation. After actuation, the valves do not require a subsequent change of state or continuous availability of power to maintain their intended safety functions. The RVVs and RRVs are the only active components in the ECCS. No single failure prevents the ECCS from performing its safety function, including single failures in electrical power (single failures in onsite power and offsite power, busses, electrical and mechanical parts, cabinets and wiring), initiation logic, and single active or passive component failure. One RRV and two RVVs are required for successful ECCS operation. If the redundant direct current (DC) power to the MPS or the ECCS and DHRS valve actuators is lost, the valves actuate. The ECCS valves open once RCS pressure goes below the inadvertent actuation block (IAB) pressure locking threshold.

An ECCS actuation would occur in the event of an AOO or IE such as an inadvertent opening of an RSV, and inadvertent opening of an ECCS valve, or a loss of AC power for more than 24 hours. An ECCS actuation would also occur during an AOO that includes an assumption of a loss of DC power. An analysis of these events was conducted and consistent with Condition 4.4 of Reference 8.3-1, ECCS actuation in response to an AOO or IE is expected to occur much less than once in the lifetime of an NPM.

Long-term cooling requirements that call for the removal of decay heat by the passive containment heat removal are discussed in Section 6.2.

15.0.0.6.4 Required Operator Actions

There are no operator actions credited in the evaluation of NuScale DBEs. After a DBE, automated actions place the NPM in a safe-state and it remains in the safe-state condition for at least 72 hours without operator action, even with assumed failures.

15.0.0.6.5 Availability of Offsite Power

Normal alternating current (AC) power systems are nonsafety-related and not credited to mitigate Chapter 15 events. The normal AC power systems consist of:

- EHVS - High voltage (13.8 kV) AC electrical system and switchyard
- EMVS - Medium voltage (4.16 kV) AC electrical distribution system, and
- ELVS - Low voltage (480V and 120V) AC electrical distribution system.

The onsite direct current (DC) power systems are nonsafety-related and not credited to mitigate Chapter 15 events. The DC power systems consist of:

- EDSS - highly reliable DC power system supply essential loads
- EDNS - normal DC power system supply non-essential loads.

The EDNS and the EDSS power systems are designed with battery backups to allow a continuous power supply in the event the power supply to the chargers is not available, as described in Section 8.1. The battery backup for EDSS is sized to supply loads, with the exception of the ECCS valves, for greater than 24 hours. The ECCS valves are unique because the MPS acts to shed the load for these valves at 24 hours after the loss of normal AC power to the EDSS battery chargers. The loss of normal AC power to the EDSS chargers also causes the MPS to initiate a reactor trip, actuate DHRS, and close the containment isolation valves.

As discussed in Section 8.2.1, offsite power is not needed to achieve and maintain safety functions and a failure within EHVS does not prevent safety-related functions. A loss of the plant external power connections is referred to as a loss of normal AC power. As discussed in Section 8.2.1, a transmission grid connection is not the normal power source for the NuScale Power Plant; rather, a connection to an external transmission grid represents a primary plant load.

The analysis of DBEs considers that a connection to normal AC power is unavailable if the event involves a turbine trip. An initiating event with a subsequent automatic turbine trip is conservatively assumed to disrupt the electrical connection to an external transmission grid. It should be emphasized, however, that loss of a single NPM is a small disturbance unlikely to cause external connection instability. As discussed in Section 8.1.2, the NuScale Power Plant has the capability to become self-sustaining while separated from the offsite electrical grid. However, this capability is not credited in the DBE analyses.

Based on the electrical system design for the NPM, a loss of normal AC power means a loss of power from the ELVS. Such a condition could be due to: 1) a failure within the ELVS; or, 2) a loss of power from the EHVS or EMVS. Since a failure of the EHVS or EMVS results in the same response from the plant electrical system for transient analyses, these failures are not considered separately. With respect to the battery backups for the DC power supply systems, the availability of battery backups is not considered because the EDNS and EDSS are not safety-related. Therefore, the following scenarios are considered for loss of power assumptions in DBEs:

- Loss of normal AC either at the time of the initiating event or at the time of the turbine trip. After 24 hours, the ECCS valves move to their fail-safe open position.
- Loss of normal DC power (EDNS) and normal AC - Power to the reactor trip breakers is provided via the EDNS, so this scenario is the same as a loss of normal AC with the addition of reactor trip at the time at which power is lost.
- Loss of the highly reliable DC power system (EDSS), EDNS, and normal AC - this scenario results in a reactor trip, actuation of DHRS, and closure of containment isolation valves. The ECCS valves move to their fail-safe open position upon RCS pressure dropping below the IAB pressure threshold.

Power is assumed available for events if consequences of the event are more limiting.

15.0.0.6.6 Treatment of Nonsafety-Related Systems

Nonsafety-related systems are considered in establishing the initial plant conditions for DBEs and during the initial plant response to those events. The treatment of nonsafety-related equipment in DBEs is as follows:

- Nonsafety-related system normal operation that increases the consequences of the event are modeled.
- Nonsafety-related system normal operation that improves (decreases) the consequences of the event is not modeled.
- Nonsafety-related system normal operation that does not significantly alter the consequences of the event may be modeled.
- Nonsafety-related equipment is evaluated considering the licensing basis assumptions defining the event. These assumptions can include external events, environmental effects, offsite power availability, and onsite power availability.
- A nonsafety-related system failing to perform its function is considered, but not the failure to a worst-state condition except as an event initiator.

The reliability of nonsafety-related systems is also considered when categorizing events by frequency (refer to Section 15.0). Nonsafety-related mechanical and electrical systems are treated in an analogous manner.

Nonsafety-related equipment may be used for event mitigation for the following two circumstances:

- when a detectable and nonconsequential random and independent failure must occur in order to disable the system
- when nonsafety-related components are used as backup protection.

There are three occurrences where nonsafety-related equipment is credited for event mitigation because the nonsafety-related component is used for backup protection. Listed below is the equipment associated with these occurrences.

Table 15.0-9 identifies the events in which nonsafety-related equipment is credited for event mitigation.

1) The nonsafety-related secondary main steam isolation valve (MSIV) serves as the backup isolation device to the safety-related MSIV for isolation of the main steam piping penetrating containment when the safety-related MSIV is assumed to fail.

2) The nonsafety-related feedwater regulating valve (FWRV) serves as the backup isolation device to the safety related feedwater isolation valve (FWIV) for isolation of the feedwater system (FWS) piping penetrating the containment when the FWIV is assumed to fail.

3) The nonsafety-related feedwater check valve serves as the backup isolation device to the safety-related feedwater check valve for isolation of the DHRS when reverse flow is experienced during a break in the FWS piping.

The nonsafety-related secondary main steam bypass isolation valve (MSIBV) serves a similar function. The nonsafety-related MSIBV serves as a backup isolation device to the safety-related MSIBV for isolation of the main steam piping penetrating containment. Since the MSIBVs are only used during initial startup operations (e.g. for heatup the main steam lines) and are closed during power operations, failure of the safety related MSIBV is not considered in safety analyses. In addition, failure of an MSIV bounds MSIBV failure as the MSIV is a larger valve. So the nonsafety-related MSIBV is not credited for event mitigation based on potential failure of the safety-related MSIBVs and for these reasons the nonsafety-related MSIBV is not identified in Table 15.0-9 as nonsafety-related equipment credited for event mitigation.

Classification information for the secondary MSIVs, MSIBVs, FWRVs, and the nonsafety-related feedwater check valves are listed in Section 3.2, Table 3.2-1. The secondary MSIVs and MSIBVs are described in Section 10.3.2. The FWRVs and nonsafety-related feedwater check valves are described in Section 10.4.7.

The reactor pool liner, described in Section 9.2.5, is a nonsafety-related component of the reactor pool used as the ultimate heat sink (UHS). Section 9.2.5 describes how the pool liner meets the criteria for event mitigation in that water leakage from the liner is detectable and leakage is a nonconsequential random and independent failure. Therefore, any event that progresses to using DHRS, or convection cooling through the containment vessel to the reactor pool with the use of RVVs and RRVs uses the UHS and the pool liner.

15.0.0.7 Multiple Module Events

Chapter 15 DBEs are analyzed for a single NPM. Chapter 21 discusses the suitability of shared components and the design measures taken to ensure these components do not introduce multi-module risks. Section 19.1 discusses consideration of multi-module events.

15.0.1 Radiological Consequence Analyses Using Alternative Source Terms

A modified version of the alternative source term methodology is used to evaluate radiological consequences of DBEs and the beyond-design-basis event CDE. The source term methodology and the application of that methodology are described in Section 15.0.3.

15.0.2 Review of Transient and Accident Analysis Methods

This section summarizes the principal computer codes used in transient analyses and describes the evaluation model development and assessment process (EMDAP). A roadmap with references to NuScale topical and technical reports required to develop those evaluation models is provided in Table 15.0-10.

Several different licensing methodologies are required to provide the neutronic, thermal-hydraulic, and radiological response of the plant to postulated accidents, IEs, and AOOs. The NuScale Power Plant licensing methodologies include the computer programs and the calculation framework for a specific transient or accident such as the mathematical models used, assumptions included in the programs, and procedures for treating the program input and output information. The licensing methodology also includes required assumptions about the plant equipment availability, combinations with external events, and other information necessary to specify the calculation procedure and to meet the requirements of the GDC of 10 CFR 50, Appendix A.

The licensing methodologies address the following elements:

- identification of the DBEs to be analyzed
- design basis licensing assumptions
- determination of the requirements for the licensing methodology, code assessment, uncertainty analysis, and framework qualification

The events that are considered for evaluation in the NuScale Power Plant design, the categorization of DBEs and the description of the licensing assumptions used in the analyses of these events are described in Section 15.0.0.

The design basis analyses of accidents and transients presented in this chapter are performed in accordance with the requirements of 10 CFR 50.34, 10 CFR 50.46, and where applicable, per NUREG-0737. This section describes the review process and acceptance criteria for analytical models and computer codes used for analyzing the accident and transient behavior of the NuScale Power Plant. The models simulate the event under consideration and demonstrate conservatism of the analysis by one of several methods.

1) Demonstrating from the analysis results that the chosen bounding parameters and licensing assumptions are conservative. The 10 CFR 50 Appendix K LOCA analyses are performed in this manner.

2) Using sensitivity analyses in the calculation to demonstrate that values are conservative. Many of the AOOs and postulated accident analyses performed in Chapter 15 use this approach.

3) Performing best-estimate analyses.

Table 15.0-1 provides a tabular summary of the computer codes used, and Table 15.0-8 lists the reactivity coefficients (e.g., moderator temperature and Doppler coefficients) and initial thermal power assumed in the analysis of each transient or accident. Acceptance criteria for the events analyzed in Chapter 15 are listed in Table 15.0-2 through Table 15.0-5.

15.0.2.1 Licensing Methodology (Evaluation Models)

The computer codes represent the important phenomena and components necessary to simulate the events identified in Section 15.0. Mathematical models and the numerical solution of those models predict the important physical phenomena to calculate the safety consequences of the events being analyzed and to determine if adequate safety margin has been provided. The LOCA mathematical models and emergency core cooling system (ECCS) evaluation models meet the applicable requirements contained in Appendix K to 10 CFR 50. The NuScale design supports an exemption from 10 CFR 50 Appendix K requirements that are not applicable. Non-LOCA methodologies are treated by choosing suitably conservative inputs.

The EMDAP establishes the adequacy of a methodology for evaluating complex events that are postulated to occur in nuclear power plant systems. The EMDAP described here has been developed for simulating LOCAs in the NPM. While there are differences between the NuScale Power Plant and current generation PWRs, many of the basic physical phenomena are essentially the same. Also, the NuScale Power Plant EMDAP for LOCA uses a computer code based on the well-established RELAP5 computer code.

By design, the margins to regulatory limits for the NPM are greater than for current generation PWRs, such that a number of phenomena that occur and are important in the PWR loss-of-coolant accident do not occur in the NPM. Examples of phenomena that do not occur for NPM design basis LOCAs include loop seal clearing, pump coastdown, two-phase pump performance, entry of significant amounts of noncondensable gases into the system, core reflooding at low reflood rates, clad swelling and rupture, metal-water reaction, and ECCS bypass. A more detailed examination of differences and a detailed proposed reconciliation of existing LWR regulatory requirements and guidance with the characteristics of the NPM design are presented in Reference 15.0-3.

Reference 15.0-3 provides a description of the NuScale LOCA evaluation model which has been developed following the guidelines in the EMDAP of Regulatory Guide (RG) 1.203. The NuScale LOCA evaluation model meets the applicable requirements of 10 CFR 50 Appendix K and 10 CFR 50.46 acceptance criteria. Differences in geometry and operating mode between current PWRs and the NuScale Power Plant include natural circulation, a high pressure containment, and helical coil steam generators. Therefore, a graded approach to development and assessment of the NuScale EMDAP for LOCA is appropriate with emphasis on the differences between the NuScale Power Plant and current PWRs. Most of the effort is focused on demonstrating the applicability and assessment of the NRELAP5 code for a new application to the LOCA event in the NPM.

The non-LOCA analysis methodology follows the intent of the EMDAP and builds on the NRELAP5 LOCA model, as described in Reference 15.0-5. A non-LOCA phenomena identification and ranking table (PIRT) was used to establish the requirements for this model in recognition of the substantial NRELAP5 EMDAP effort.

The non-LOCA analysis covers analysis of AOOs, IEs, and postulated accidents with the exception of the inadvertent operation of ECCS discussed in Section 15.6.6. The non-LOCA methodology uses NRELAP to evaluate the system response and provide the results for the pressure acceptance criteria. Additional analyses are performed to ensure that other acceptance criteria are met. Therefore, the non-LOCA methodology interfaces with the radiological assessment methodology described in Section 15.0.3 and Reference 15.0-4, the subchannel methodology described in Section 4.4 and Reference 15.0-1, and the containment methodology described in Section 6.2. Long term cooling following non-LOCA events is discussed in Section 15.0.5 and Reference 15.0-7. Additional details about the code capabilities, limitations, modeling details, and verification and validation are provided in the topical or technical reports listed in Table 15.0-10.

Chapter 15 design basis analyses require licensing evaluation models for subchannel analysis, radiological analysis and prediction of neutronic behavior. The primary distinction between information provided in this section and in Sections 4.3 and 4.4 is that model development and qualification for neutronic and subchannel analysis is discussed in Chapter 4, while application of these evaluation models in providing transient and accident analysis results is discussed in Chapter 15.

15.0.2.2 Thermal-Hydraulic Response

15.0.2.2.1 Loss-Of-Coolant Accident Methodology

The evaluation model used to analyze design-basis LOCAs for the NPM uses the NRELAP5 code which addresses the unique features and phenomena of the NPM design and comply with the applicable requirements of 10 CFR 50, Appendix K. An exemption from 10 CFR 50, Appendix K requirements that are not applicable to the NuScale design was documented with the application. Details of the NuScale LOCA methodology and code qualification are discussed in Reference 15.0-3.

Appendix K

The ECCS cooling performance must be calculated in accordance with an acceptable evaluation model as specified in Appendix K to 10 CFR 50, and must be calculated for a number of cases to ensure that the most limiting LOCAs are identified. Two options for ECCS performance calculations allow for ensuring that the most limiting design-basis accident has been evaluated. 10 CFR 50.46(a)(i) endorses the best-estimate approach detailed in RG 1.157 and 10 CFR 50.46(a)(ii) endorses the conservative deterministic approach detailed in Appendix K. In view of the large safety margins in the NPM with respect to 10 CFR 50.46 LOCA acceptance criteria, the deterministic bounding approach in 10 CFR 50.46(a)(ii) is used by NuScale. Details of the NuScale LOCA evaluation model and code qualification are discussed in Reference 15.0-3.

10 CFR 50.46 Loss-Of-Coolant Accident Acceptance Criteria

10 CFR 50.46 requires that LWRs fueled with uranium oxide pellets within zircaloy cladding be provided with an ECCS such that their calculated core cooling performance after a LOCA conforms to the following acceptance criteria:

1) Peak fuel cladding temperature shall not exceed 2200 degrees F.

2) Maximum cladding oxidation shall not exceed 0.17 times the total cladding thickness before oxidation.

3) The maximum hydrogen generation shall not exceed 0.01 times the hypothetical amount that would be generated if all of the metal in the cladding were to react.

4) Calculated changes in core geometry shall be such that the core remains amenable to cooling.

5) The calculated core temperature shall be maintained at an acceptably low value and decay heat shall be removed for an extended period of time after successful initial operation of the ECCS.

The NuScale LOCA licensing methodology (Reference 15.0-3) addresses the first four criteria and the long-term cooling methodology (Reference 15.0-7) addresses the fifth crterion. The methodology described in Reference 15.0-3 covers ECCS performance in the NPM up to the time when recirculation flow is established, and the pressures and levels in containment and RPV approach a stabilized condition (i.e., flow is recirculating in through the RRVs, core heat is removed by boiling in the core, and steam exits through the RVVs).

The NPM is designed so that there is no core uncovery or heatup for a design-basis LOCA, and peak cladding temperature (PCT) is well below the acceptance criterion. For the NuScale LOCA evaluation model, the relevant figures of merit are the collapsed liquid water level in the core, the critical heat flux ratio (CHFR), and containment pressure and temperature. These figures of merit are more sensitive than PCT in the NPM design. Maintaining primary inventory, ensuring the core does not go into post-critical heat flux (CHF) heat transfer, and ensuring that the containment pressure and temperature for the limiting collapsed liquid level case remains below design limits ensures that the Appendix K limits for PCT, oxidation, and hydrogen production are not violated.

There is no oxidation of the cladding as a result of a LOCA in the NPM. There are no changes in core geometry resulting from a LOCA that would prevent the core from being amenable to cooling. Therefore, the first four 10 CFR 50.46 acceptance criteria are met when the collapsed liquid level remains above the top of the core, the critical heat flux ratio is greater than 1.29. The NuScale specific LOCA acceptance criteria are listed in Table 15.0-4.

The calculated core temperature is maintained at an acceptably low value and decay heat is removed in both the short-term and long-term of a LOCA in the NPM.

The long-term evaluation of core temperature and decay heat removal is assessed in Reference 15.0-7.

15.0.2.2.2 Non-Loss-Of-Coolant Accident Methodology

The main steps of the non-LOCA system transient analysis process are:

1) Perform steady state and transient system analysis calculation with NRELAP5.

2) Evaluate results to confirm margin to RCS and steam generator pressure acceptance criteria.

3) Identify if a subchannel analysis is necessary based on system response.

4) Perform a subchannel analysis for those events identified in step 3.

For step 1, NRELAP5 is the thermal-hydraulics code used to calculate the NPM system response short-term transient event progression. The NuScale LOCA evaluation model was developed following the EMDAP guidelines of RG 1.203, as outlined in Reference 15.0-3. The NuScale non-LOCA EM starts with the LOCA EM and modifies it for use for non-LOCA events, as described in Reference 15.0-5. The requirements of the non-LOCA evaluation model capability are established based on the analysis purpose and plant design.

The EMDAP defined in RG 1.203 provides a four-element structured process to establish the adequacy of a methodology for evaluating complex events that are postulated to occur in nuclear power plant systems using the guidance of RG 1.203. The evaluation model has been developed using the guidance of RG 1.203 for simulating the NPM system transient response to non-LOCA events. Reference 15.0-5 describes the modifications made to the LOCA evaluation model to develop the non-LOCA evaluation model.

The short-term non-LOCA transient calculations presented in Reference 15.0-5 cover transient initiation and reactor trip, and demonstrate stable natural circulation is achieved and effective DHRS operation has been established. The transient progression from this point is similar regardless of the specific initiating event, and the subsequent transient progression is treated as part of long-term decay and residual heat removal analysis discussed in Section 15.0.5.

The NPM parameters used in DBE evaluations are identified in Table 15.0-6. Table 15.0-7 lists the analytical limits and the associated time delays used in the Chapter 15 DBEs. Results of the DBE analyses are compared to the acceptance criteria identified in Table 15.0-2 through Table 15.0-5. System response is evaluated with respect to depressurization rates, to determine if the event is bounded by another DBE. Events with a slow depressurization rate that tend toward increasing CHFR, which are bounded by events with a rapid depressurization rate, are not specifically analyzed for CHF with a subchannel analysis. VIPRE-01 is used to perform the subchannel analysis, and Table 15.0-1 identifies the DBEs for which a subchannel analysis is performed.

For the rod ejection calculations, a combination of CASMO5, SIMULATE5, and SIMULATE-3K (S3K) are used to calculate the core response and reactivity-related inputs. S3K is used to calculate fuel energy deposition and temperatures. The power response for the accident is determined by S3K for both NRELAP5 and VIPRE-01.

NRELAP5 is used to calculate system response including data such as flow rates, pressures and temperatures. NRELAP5 results are used as boundary conditions for the subchannel analysis.

VIPRE-01 is used to perform the detailed subchannel calculations to determine the minimum critical heat flux ratio (MCHFR) for each event. Peak fuel temperatures, clad temperatures, radially averaged fuel enthalpy, and CHF calculations are performed with VIPRE-01 to address the event acceptance criteria. The description of the codes and the methodology for rod ejection accident analysis are described in more detail in Reference 15.0-11.

15.0.2.2.3 Flow Stability

The NPM system response is obtained by the NuScale proprietary computer code, PIM, which is used to demonstrate system stability at steady-state operation. The PIM code is described in Section 4.4.7. The PIM code relies on the published description of the theory and numerical methods of RAMONA, but is not a direct derivative of the coding. The PIM code has been developed independently to suit the geometry and specific needs of the NPM. The main advantage of the RAMONA-type algorithm is the absence or insignificance of numerical damping that affects other time-domain codes and requires extensive studies and adjustments before they can be successfully benchmarked and reliably used. Reference 15.0-10 provides details about the process used to select and qualify the PIM code.

15.0.2.3 Subchannel Analysis

VIPRE-01 is a subchannel analysis tool designed for general-purpose thermal-hydraulic analysis under normal operating conditions, operational transients, and events of moderate severity.

VIPRE-01 is used to generate local thermal-hydraulic conditions for CHF tests in developing a CHF correlation. VIPRE-01 provides local thermal-hydraulic conditions throughout the reactor core used in calculating the minimum critical heat flux ratio (MCHFR). VIPRE-01 also provides more realistic boundary conditions, such as the axial profiles for the coolant temperature and wall heat transfer coefficient in a limiting subchannel, for the fuel rod performance analyses. VIPRE-01 qualification is applicable to the NuScale implementation. VIPRE-01 is validated against applicable test data that spans the plant range and establishes the code accuracy and uncertainty. Reference 15.0-1 provides an applicability assessment of the models, correlations, and features in the VIPRE-01 code for the NPM design.

NuScale-specific CHF correlations have been developed to better represent the NuScale core and fuel assembly design. The NuScale-specific CHF correlations are

described in Section 4.4.4 and detailed in Reference 15.0-2. NuScale-specific CHF correlations have been added to the existing suite of VIPRE-01 CHF correlations as an enhancement to VIPRE-01.

15.0.2.4 Radiological Analyses Methodology

The computer codes used in calculating DBE, and beyond-design-basis event CDE, doses are described below. Reference 15.0-4 provides additional details on each of the computer codes described below.

15.0.2.4.1 SCALE 6.1, TRITON, and ORIGEN-SCALE

SCALE 6.1 modular code package, developed by Oak Ridge National Laboratory, is used for development of reactor core and primary coolant fission product source terms. The TRITON and ORIGEN-ARP analysis sequences of the SCALE 6.1 modular code package, and ORIGEN-S (ORIGEN-SCALE), run as a standalone module, are used to generate radiation source terms for the fuel assemblies and primary coolant. This software has been extensively used in the evaluation of operating large LWRs. The operating environment, nuclear fuel, and structural materials in the NuScale Power Plant design are similar to, or bounded by, those typically found in large PWRs.

TRITON is used to generate burnup-dependent cross sections for NuScale fuel assemblies for subsequent use in the ORIGEN-ARP depletion module. The TRITON sequence of the SCALE code package is a multipurpose control module for nuclide transport and depletion, including sensitivity and uncertainty analysis. TRITON can generate problem-dependent and exposure dependent cross sections as well as perform multi-group transport calculations in one-dimensional, two-dimensional, or three-dimensional geometries.

ORIGEN-ARP is a SCALE depletion analysis sequence used to perform point-depletion and decay calculations with the ORIGEN-S module using problem-dependent and burnup-dependent cross sections.

The ORIGEN-S module of SCALE is used to calculate the time-dependent isotopic concentrations of materials in a NuScale fuel assembly by modeling the fission, transmutation, and radioactive decay of fuel isotopes, fission products, and activation products in the assembly. The input isotopic concentrations for those calculations take into account the various chemical and physical processes occurring in the reactor systems and the processing of the liquid, solid, and gaseous waste streams. As a part of the ORIGEN-S decay calculations, time-dependent radiation source terms, (i.e., the activities, neutron spectra, and gamma spectra due to the radioactive isotopes) present in the fuel and waste streams are calculated for use in subsequent dose rate evaluations.

15.0.2.4.2 ARCON96

Onsite and offsite atmospheric dispersion factors for DBEs, and the beyond-design-basis event CDE, are calculated with ARCON96. The program implements the guidance provided in RG 1.194. The code implements:

- a building wake dispersion algorithm
- an assessment of ground level, building vent, elevated, and diffuse source release modes
- hour-by-hour meteorological observations
- sector averaging and directional dependence of dispersion conditions

NuScale uses ARCON96 for various time periods at the exclusion area boundary (EAB) and the outer boundary of the low population zone (LPZ) as well as the control room and technical support center (TSC). Justification for utilizing ARCON96 for offsite locations is provided in Reference 15.0-4.

15.0.2.4.3 RADTRAD

RADTRAD is used to estimate radionuclide transport and removal of radionuclides and dose at selected receptors for the various DBEs and the beyond-design-basis event CDE. Given the radionuclide inventory, release fractions, and timing, RADTRAD estimates doses at the EAB and LPZ, and inside the control room and TSC. As material is transported from the point of release, the input model can account for processes that may reduce the quantity of radioactive material. Material can flow between buildings, from buildings to the environment, or into the control room and TSC through filters, piping or other connectors. An accounting of the amount of radioactive material retained in these pathways is maintained. Decay and in-growth of daughter products can be calculated over time as material is transported. Reference 15.0-4 describes the use of RADTRAD for NuScale application.

15.0.2.4.4 MELCOR

MELCOR is used to model the progression of severe accidents through modeling the major systems of the reactor plant and their coupled interactions (NUREG/CR-6119, Rev. 2). Specific use relevant to the application of the CDE described in Section 15.10 includes:

- thermal-hydraulic response of the primary coolant system and containment vessel
- core uncovering, fuel heatup, cladding oxidation, fuel degradation, and core material melting and relocation
- aerosol generation
- in-vessel and ex-vessel hydrogen production and transport
- fission product release (aerosol and vapor), transport, and deposition

15.0.2.4.5 STARNAUA

This code is an aerosol transport and removal software program that is an enhanced version of NAUAHYGROS and was developed to perform aerosol removal calculations in support of work to develop and apply a more realistic source term for advanced and operating LWRs.

STARNAUA models natural removal of containment aerosols by gravitational settling and diffusiophoresis, and considers the effect of hygroscopicity (growth of hygroscopic aerosols due to steam condensation on the aerosol particles) on aerosol removal. STARNAUA enhancements of NAUAHYGROS include addition of:

- a model for thermophoresis
- a model for spray removal
- the capability to directly input either steam condensation rate or condensation heat transfer rate and total heat transfer rate such as would be provided from an external containment thermal-hydraulics code calculation

The NuScale realistic source term methodology, used to support the radiological consequence analysis of the CDE described in Section 15.10, is consistent with existing industry practice used for large passive plant design certification. More detail on the application of STARNAUA is provided in Reference 15.0-4.

15.0.2.4.6 NuScale pH_T Code

The NuScale pH_T code is used to calculate post-accident aqueous molar concentration of hydrogen ions utilizing the methodology described in Reference 15.0-4 to support the radiological consequence analysis of the CDE described in Section 15.10. Calculation of the extent of iodine re-evolution inside containment is dependent on the pH_T. This program takes inputs for:

- initial boron and lithium concentrations
- total core inventory of iodine and cesium
- integrated photon dose to the containment and total dose to the coolant
- initial mass of coolant
- mass of coolant
- temperature of coolant

The program then calculates the pH_T as a function of time.

15.0.2.4.7 MCNP6

The MCNP6 is used for evaluating potential shine radiological exposures or doses to operators in the control room following a radiological release event. Both sky-shine and shine from filters are evaluated. MCNP is a general-purpose tool used for neutron, photon, electron, or coupled neutron, photon, and electron transport. MCNP treats an arbitrary three-dimensional configuration of materials in geometric cells bounded by first- and second-degree surfaces and fourth-degree elliptical tori. The code is well-suited to performing fixed source calculations.

MCNP uses continuous energy cross-section data. For photons, the code accounts for incoherent and coherent scattering, the possibility of fluorescent emission after photoelectric absorption, and absorption in electron-positron pair production. Electron and positron transport processes account for angular deflection through

multiple Coulomb scattering, collisional energy loss with optional straggling, and the production of secondary particles including x-rays, knock-on and Auger electrons, bremsstrahlung, and annihilation gamma rays from positron annihilation at rest. The MCNP code is commercially-grade dedicated under the NuScale NQA-1 program described in Reference 15.0-4.

15.0.3 Design Basis Accident Radiological Consequence Analyses for Advanced Light Water Reactors

This section presents the methodology used to perform the calculations associated with the radiological consequences of the DBSTs and the core damage source term (CDST) associated with the beyond-design-basis CDE. Table 15.0-11 identifies the list of events analyzed for radiological consequences. Results from the application of this methodology are provided in Table 15.0-12.

15.0.3.1 Introduction

This section describes the NuScale conservative methodology for developing accident source terms and performing the corresponding radiological consequence analyses. Key unique features of the NuScale methodology are the:

- use of ARCON96 to calculate off-site atmospheric dispersion factors
- development of an iodine spike DBST and a beyond-design-basis CDST that are each assessed against the radiological criteria of 10 CFR 52.47(a)(2)(iv). If both the design-basis iodine spike DBST and the beyond-design-basis CDST analyses show acceptable dose results, then 10 CFR 52.47(a)(2)(iv) is met.

10 CFR 52.47(a)(2)(iv) requires nuclear power reactor design certification applicants to evaluate the consequences of a fission product release into the containment assuming the facility is being operated at the maximum licensed power level and to describe those design features intended to mitigate the radiological consequences of an accident. NuScale follows the approach of the 2012 Nuclear Energy Institute (NEI) position paper on small modular reactor source terms (Reference 15.0-6) by referring to the scenario described in footnote 3 of 10 CFR 52.47(a)(2)(iv) as the maximum hypothetical accident (MHA).

The MHA has historically been linked to a large-break LOCA in large LWRs. The NPM has no large diameter primary coolant system piping; therefore, a large-break LOCA cannot be postulated as the basis for the MHA radiological consequence analysis for NuScale.

As stated in RG 1.183, "the design basis accidents were not intended to be actual event sequences, but rather, were intended to be surrogates to enable deterministic evaluation of the response of a facility's engineered safety features." The NuScale design has DBEs that result in primary coolant entering the containment and the iodine spike DBST described in Section 15.0.3.8.6 is used to bound the radiological consequences of these events. The beyond-design-basis CDE described in Section 15.10, with its associated CDST that is composed of a set of key parameters derived from a spectrum of surrogate accident scenarios, is also postulated. The design-basis iodine spike DBST and the beyond-design-basis CDST are each assessed against the radiological criteria of 10 CFR 52.47(a)(2)(iv). If both the design-basis iodine spike DBST

and the beyond-design-basis CDST analyses show acceptable dose results, then 10 CFR 52.47(a)(2)(iv) is met. The analysis of the beyond-design-basis CDST against the acceptance criteria of 10 CFR 52.47(a)(2)(iv) provides reasonable assurance that, even in the extremely unlikely event of a severe accident, the facility's design features and site characteristics provide adequate protection of the public.

Table 15.0-11 identifies the list of events evaluated for radiological consequences, cross references them to RG 1.183, and identifies the primary source of radiation for the event. Table 15.0-12 provides the iodine spike DBST and the CDST dose results. The results meet acceptance criteria and therefore, 10 CFR 52.47(a)(2)(iv) is met.

15.0.3.2 Methodology Overview

The DBE radiological consequence analyses follow the guidance of RG 1.183 methodology modified to reflect the difference in the NuScale Power Plant design from large PWRs, as described in Reference 15.0-4. This methodology addresses the submersion and inhalation doses and the direct shine doses from contained or external sources. The key elements of this methodology are:

- Thermal-hydraulic conditions are modeled using NRELAP.
- Source term and dose evaluations are calculated using RADTRAD.
- Meteorological dispersion is calculated using ARCON96.

Section 15.0.2.4 summarizes the computer codes used for calculating DBE doses.

15.0.3.3 General Dose Analysis Inputs

The following sections summarize the key aspects for calculating the radiological consequences of the DBEs.

15.0.3.3.1 Core Radionuclide Inventory

The isotopic inventories of fuel assemblies are calculated using SCALE 6.1 which is described in Section 15.0.2.4. Isotopic concentrations are based on the detailed geometry of a fuel assembly, rated power plus uncertainty, maximum assembly average exposure, and a range of U-235 enrichments. The isotopic inventory is calculated at a number of time steps in the fuel cycle. Table 11.1-1 provides the maximum end of cycle core isotopic inventory.

15.0.3.3.2 Primary Coolant Radionuclide Inventory

For the radiological consequence analysis, the radioiodine concentrations in the primary coolant system are set at the maximum dose equivalent values permitted by design basis limits. Table 15.0-14 provides the primary coolant radionuclides and nominal inventory assumed in the dose analyses presented in Section 15.0.3.8. The iodine appearance rates, including the pre-incident appearance rates, are described in Section 15.0.3.8, where used.

15.0.3.3.3 Secondary Coolant Activity

Large PWR designs contain a large volume of secondary system water on the "shell" side of the SG. Through primary-to-secondary leakage limits and monitoring by sampling, this water volume contains levels of iodine that are limited operationally. A sensitivity study was performed in Reference 15.0-4 for the steam generator tube failure (SGTF) and main steam line break (MSLB) events assuming the liquid secondary coolant in the SG was at the primary coolant design basis limit concentration. The sensitivity study demonstrated dose results are not sensitive to the initial secondary side activity. This conclusion is supported by comparing the secondary coolant source terms shown in Table 11.1-5 with the primary coolant source terms shown in Table 15.0-14.

15.0.3.3.4 Not Used

15.0.3.3.5 Not Used

15.0.3.3.6 Not Used

15.0.3.3.7 Not Used

15.0.3.3.8 RADTRAD Modeling

Consistent with RG 1.183:

- The RADTRAD decay and daughter product modeling option is used to include progeny from the decay of parent radionuclides that are significant with regard to radiological consequences and the released radioactivity. The calculated total effective dose equivalent (TEDE) is the sum of the committed effective dose equivalent from inhalation and the deep dose equivalent from external exposure from tracked isotopes.
- RADTRAD does not include corrections for depletion of the effluent plume by deposition on the ground.
- RADTRAD determines the maximum two-hour TEDE by calculating the postulated dose for a series of small time increments and performing a "sliding" sum over the increments of successive two-hour periods.

15.0.3.3.9 Not Used

15.0.3.3.10 Not Used

15.0.3.3.11 Atmospheric Dispersion Factors (χ/Q), Breathing Rates, and Occupancy Factors

Atmospheric dispersion factor (χ/Q) inputs to RADTRAD are derived as described in Reference 15.0-4 with assumptions shown in Table 15.0-20 and Table 15.0-21. Table 2.0-1 provides the accident release χ/Q values.

Control room and offsite breathing rate and control room occupancy factor inputs to RADTRAD, consistent with RG 1.183, are listed in Table 15.0-13.

15.0.3.3.12 Dose Conversion Factors

Consistent with RG 1.183, dose conversion factors from Environmental Protection Agency Federal Guidance Report No. 11 (Reference 15.0-8) and Report No. 12 (Reference 15.0-9) are used for dose analysis.

15.0.3.4 Containment Leakage

Containment leakage is described in Reference 15.0-4 and is consistent with the recommendations of RG 1.183. The design-basis containment leak rate is provided in Table 6.5-1.

15.0.3.5 Secondary-Side Decontamination

The helical coil steam generators of the NuScale Power Plant design are different than that of a large PWR because the primary coolant is on the outside of the tubes. As a result, there is no bulk water volume in which decontamination can easily occur. Reference 15.0-4 provides the details about the decontamination factor used in the helical coil steam generators as well as the treatment of iodine deposition in the main steam piping and the condenser.

15.0.3.6 Reactor Building Decontamination Factors

Reactor Building (RXB) decontamination factors are described in Reference 15.0-4.

15.0.3.7 Receptor Location Considerations

Potential on-site radiological receptor locations considered in this evaluation are the control room and TSC; potential off-site locations are the EAB and LPZ. Figure 15.0-3 shows the schematic of the RADTRAD code nodalization used to model these locations for leakage paths from the containment or RXB. Figure 15.0-4 shows the RADTRAD code nodalization for the SGTF and MSLB events in which the principal release path is through the steam generator.

A summary of control room and TSC characteristics are provided in Section 15.0.3.7.1 and Section 15.0.3.7.2, respectively. The variables associated with the derivation of these receptors are presented in Table 15.0-13.

15.0.3.7.1 Control Room Design

Accident analyses are performed for two control room emergency modes as follows:

- Uninterrupted power supply with continuous filtered airflow to the control room envelope for the event duration.

- Immediate loss of power with control room habitability system (CRHS) activation, and restored filtered airflow to the control room envelope at the time of CRHS depletion (72 hours).

Simplifying assumptions are made for the control room ventilation system design. Figure 15.0-3 and Figure 15.0-4 show the control room RADTRAD code nodalization used in the dose analyses. The key design features assumed for the control room are summarized as follows:

- The nonsafety-related normal control room ventilation system inlet filters remove 99 percent of iodine.
- The nonsafety-related normal control room ventilation is isolated by a non safety-related control system once the radioactivity measured at the duct intake reaches the isolation signal setpoint. The setpoint for the radiation monitor to redirect air through the air filtration unit is 10-times background. The setpoint for CRHS initiation and CRE isolation is 10-times the expected radiation out of the filtration unit following a DBE, which indicates a failure of the filtration unit to remove sufficient radioactivity. The time between when the radiation concentration reaches the detector setpoint and radiation enters the control room or technical support center (TSC) envelopes is assumed to be 30 seconds. Ten times the expected post-accident radiation analytical limits for noble gases, particulate and iodine are shown in Table 15.0-19.
- An emergency source of pressurized air with the control room habitability system (CRHS) provides clean air for 72 hours.
- After 72 hours of CRHS operation, the normal control room ventilation system is available for use (except after a seismic event).
- After a seismic event the normal control room ventilation system is not available for use. However, dose consequences are not applicable because mitigating SSCs are Seismic Category I and capable of performing their safety-related and nonsafety-related functions during and following the event.
- The control room is habitable during a loss of normal AC power as the CRHS automatically activates after 10 minutes without normal AC power, as described in Section 6.4.3.
- Control room ventilation is designed to minimize in-leakage.
- The control room is designed with a two-door air lock system. Therefore, in-leakage of 5 cfm is assumed for ingress and egress. An additional 147-cfm of in-leakage is also assumed.

The control room ventilation system design modeling assumptions are provided in Table 15.0-15. Details about system operation with CRHS are provided in Section 6.4 and Section 9.4.1.

No credit is taken for the use of personal protective equipment, such as beta radiation resistant protective clothing, eye protection, or self-contained breathing apparatus. No credit is taken for prophylactic drugs such as potassium iodide pills.

Potential shine radiological exposures to operators within the control room following a radiological release event are evaluated. Direct shine, sky-shine and shine from filters are evaluated using MCNP, as described in Section 15.0.2.4.7. Reference 15.0-4 provides additional details regarding the calculation of shine doses. The 30-day cumulative doses due to either recirculation filter or cloud-shine in the control room are added to the dose results provided in Table 15.0-12.

Shine doses are well below the regulatory limit of 5 rem because of the heavy shielding provided by the wall and floors of the Control Building.

15.0.3.7.2 Technical Support Center Design

Accident analyses are performed for one emergency mode: that of uninterrupted power supply with continuous filtered airflow to the Technical Support Center (TSC) envelope for the event duration. In the event of immediate loss of power with control room habitability system (CRHS) activation, TSC personnel are evacuated and the TSC functions are transferred to an alternate site-specific location. With loss of power with CRHS activation, the TSC is evacuated since it is not serviced by the CRHS.

The key design features assumed for the technical support center are summarized as follows:

- The nonsafety-related normal TSC ventilation system filters remove 99 percent of iodine under accident conditions.
- The nonsafety-related normal TSC ventilation is isolated by a non safety-related control system once the radioactivity measured at the duct intake reaches the isolation signal setpoint. The setpoint for the radiation monitor to redirect air through the air filtration unit is ten-times background. The setpoint for CRHS initiation and control room envelope isolation is ten-times the expected radiation out of the filtration unit following a DBE, which indicates a failure of the filtration unit to remove sufficient radioactivity. The time between when the radiation concentration reaches the detector setpoint and radiation enters the technical support center (TSC) envelope is assumed to be 30 seconds. Ten times the expected post-accident radiation analytical limits for noble gases, particulate and iodine are shown in Table 15.0-19.
- 10-cfm of in-leakage is assumed for ingress and egress. An additional 56 cfm of in-leakage is also assumed.

The technical support center ventilation system design modeling assumptions are provided in Table 15.0-18.

No credit is taken for the use of personal protective equipment, such as beta radiation resistant protective clothing, eye protection, or self-contained breathing apparatus. No credit is taken for prophylactic drugs such as potassium iodide pills.

Potential shine radiological exposures to operators within the TSC following a radiological release event are evaluated. Direct shine, sky-shine and shine from filters are evaluated using MCNP, as described in Section 15.0.2.4.7.

Reference 15.0-4 provides additional details regarding the calculation of shine doses.

15.0.3.7.3 Reactor Building Pool Boiling Radiological Consequences

Without available power for the active cooling systems, the addition of makeup water, or operator action, the sensible and decay heat from the NPMs and spent fuel would heat the pool water and could eventually cause the water in the UHS pools to boil. Table 9.2.5-2 shows that it takes longer than 61 hours for the pool to reach boiling after a loss of normal AC power event. However, if the pool were to boil, the dose would be less than 0.5 rem TEDE onsite and offsite.

15.0.3.8 Consequence Analyses of Design-Basis Source Terms

15.0.3.8.1 Failure of Small Lines Carrying Primary Coolant Outside Containment

Failure of small lines carrying primary coolant outside containment is not an event addressed in RG 1.183. The methodology used for determining dose consequences, including the iodine spiking assumptions for this event, is similar to that used for the MSLB and SGTF. The event-specific transient analysis described in Section 15.6.2 defines the time-dependent release of activity into the RXB.

The small-line break outside containment can be a break in the chemical and volume control system (CVCS) letdown line or makeup line, or the pressurizer spray line. A non-mechanistic line break occurs in the RXB allowing primary coolant from the reactor to be released into the RXB. In addition, primary coolant in the CVCS equipment (heat exchangers, filters, etc.) and piping within the RXB flows out of the other side of the break contributing less than 15,000 lbm additional primary coolant to the release. The limiting radiological scenarios identified in Section 15.6.2 are:

- maximum mass release - double-ended break of the CVCS letdown line
- maximum time of iodine spiking - equivalent 100 percent cross-sectional area break of the CVCS makeup line

Table 15.6-5 provides the assumed mass released from the reactor and the break isolation times for the two scenarios. The total mass released from the event is the sum of the mass released from the reactor provided in Table 15.6-5 and the primary coolant from CVCS equipment and piping discussed above.

Before containment isolation occurs, primary coolant flows out of the reactor vessel through the break at a rate and duration as described in Section 15.6.2. The coolant flow results in a time-dependent release of activity in the RXB that is conservatively modeled as a direct release to the environment. After containment isolation, primary coolant leaks through one containment isolation valve (the redundant in-series valve is assumed to fail open) at the maximum leak rate allowed by design basis limits. The activity from this leak path is also assumed to flow directly to the environment with no mitigation or reduction by intervening structures. After 30 hours, the reactor is assumed to be shut down and depressurized, and releases through the containment isolation valve stop.

The following is a summary of the assumptions used from Appendix E (main steam line break) of RG 1.183:

- coincident iodine spiking factor- 500
- duration of coincident iodine spike- 8 hours
- iodine chemical form- 97 percent elemental iodine and 3 percent organic iodide
- activity released from the fuel due to the iodine spike is assumed to mix instantaneously and homogeneously within the primary coolant in the reactor vessel
- no reduction or mitigation of noble gas radionuclides released from the primary system

The primary coolant in the reactor vessel and CVCS equipment and piping in the RXB initially contains the allowable concentration of dose equivalent (DE) I-131 of 3.7E-02 μCi/gm and DE Xe-133 of 10 μCi/gm.

There are no single failures for this event that affect the thermal-hydraulic response of the NPM. However, the failure of one of the two containment isolation valves on the faulted line is assumed in the dose consequence analysis.

RADTRAD is used to determine the dose, as outlined in Section 15.0.3.3.8. The control room model is described in Section 15.0.3.7.1. The potential radiological consequences of the small lines carrying primary coolant break outside containment event are presented in Table 15.0-12.

15.0.3.8.2 Steam Generator Tube Failure

Radiological consequences of the SGTF are calculated based on the guidance provided in Appendix F of RG 1.183.

Section 15.6.3 describes the sequence of events and thermal-hydraulic response to an SGTF. The SGTF analysis shows that the reactor core remains covered and no fuel failures occur.

This radiological consequence analysis considers the SGTF event with two different initial iodine concentrations, one based on a pre-incident iodine spike and the other based on a coincident iodine spike. A description of the scenario evaluated is summarized as follows:

1) An SGTF occurs in one of the two SGs.

2) For each of the iodine spiking scenarios, the iodine and noble gas coolant activity is calculated based on the maximum concentrations allowed by design basis limits. The primary coolant contains a concentration of 3.7E-02 μCi/gm DE I-131 for the coincident Iodine spike scenario and 2.2 μCi/gm DE I-131 for the pre-incident Iodine spike scenario. For both iodine spiking scenarios, the primary coolant contains 10 μCi/gm DE Xe-133.

3) Primary coolant flows into the secondary coolant through the failed SG tube at a rate and duration defined by the transient analysis described in Section 15.6.3.

4) Primary coolant leaks into the secondary side of the intact SGs at the maximum leak rate of 150 gallons per day allowed by design basis limits. The leakage continues until the primary system pressure is less than the secondary system pressure.

5) A time-dependent release is modeled that results in releasing the activity directly to the environment through the break.

6) Once secondary system isolation occurs, both steam lines continue to release small quantities of radioactivity through valve leakage into the RXB which is assumed to go directly into the environment without any source term reduction.

7) At 30 hours, the primary and secondary systems equalize and valve leakage stops.

Assumptions used from Appendix F of RG 1.183 are:

- coincident iodine spiking factor- 335
- duration of coincident iodine spike- 8 hr
- density for leak rate conversion- 62.4 lbm/ft^3
- iodine chemical form- 97 percent elemental iodine and 3 percent organic iodide
- no reduction or mitigation of noble gas radionuclides released from the primary system

Doses are determined at the EAB, LPZ, and for personnel in the control room and TSC. The control room model is described in Section 15.0.3.7.1. The dose results for the SGTF event are presented in Table 15.0-12.

15.0.3.8.3 Main Steam Line Break Outside Containment Accident

Radiological consequences of the MSLB outside containment accident are calculated based on the guidance provided in Appendix E of RG 1.183. Section 15.1.5 describes the sequence of events and thermal-hydraulic response to a MSLB outside containment.

The radiological dose consequence analysis considers the MSLB event with two different initial iodine concentrations, one based on a pre-incident iodine spike and the other based on a coincident iodine spike. A description of the scenario evaluated is summarized as follows

1) An MSLB occurs in one of the two main steam lines.

2) The iodine and noble gas coolant activity is calculated based on the maximum concentrations allowed by design basis limits for each of the iodine spiking scenarios. The primary coolant contains a concentration of 3.7E-02 µCi/gm DE I-131 for the coincident Iodine spike scenario and 2.2 µCi/gm DE I-131 for the pre-incident Iodine spike scenario. For both iodine spiking scenarios, the primary coolant contains 10 µCi/gm DE Xe-133.

3) Primary coolant leaks into the secondary side of the intact SGs at the maximum leak rate of 150 gallons per day allowed by design basis limits. The leakage continues until the primary system pressure is less than the secondary system pressure.

4) A time-dependent release is modeled that effectively releases the activity directly to the environment through the break.

5) The non-faulted steam line continues to release a small quantity of radiation through valve leakage.

The assumptions used from Appendix E of RG 1.183 are:

- coincident iodine spiking factor- 500
- duration of coincident iodine spike- 8 hr
- density for leak rate conversion- 62.4 lbm/ft^3
- iodine chemical form of 97 percent elemental iodine and 3 percent organic iodide
- no reduction or mitigation of noble gas radionuclides released from the primary system

Doses are determined at the EAB, LPZ, and for personnel in the control room and TSC. The control room model is described in Section 15.0.3.7.1. The potential radiological consequences of a steam system piping failure outside the primary containment are summarized in Table 15.0-12.

15.0.3.8.4 Rod Ejection Accident

Radiological consequences of a rod ejection accident (REA) are calculated based on the guidance provided in Appendix H of RG 1.183. Section 15.4.8 describes the sequence of events and thermal-hydraulic response to an REA which shows that the REA does not result in fuel failure. Therefore, per Appendix H of RG 1.183, a radiological analysis is not required as the consequences of this event are bounded by the consequences of other analyzed events.

15.0.3.8.5 Fuel Handling Accident

A fuel handling accident is postulated to occur during the movement of the fuel resulting in a dropped assembly onto the spent fuel racks, in the reactor vessel during refueling, in a spent fuel cask during loading, or on the weir wall between the reactor pool and SFP. The weir wall provides the highest point in the reactor

pool on which a fuel assembly could come to rest. Therefore, it is assumed that the dropped fuel assembly lands horizontally on the top of the weir wall providing the minimum water depth above the dropped assembly. The methodology for determining fuel handling accident radiological consequences is consistent with the guidance provided in Appendix B of RG 1.183.

The inventory of fission products available for release at the time of the accident is dependent on a number of factors, such as the power history of the fuel assembly, the time delay between reactor shutdown and the beginning of fuel handling operations, the volatility of the nuclides, and the number of fuel rods damaged in a fuel assembly handling accident. The activity available for release is based on 102 percent power, bounding core inventory provided in Table 11.1-1, and a 1.4 radial peaking factor with 48 hours decay from time of reactor shutdown to the beginning of fuel handling operation. Activity is instantaneously released into the pool water from all fuel rods in the dropped assembly.

The following is a summary of the assumptions used from Appendix B of RG 1.183:

- radionuclides considered include xenon, krypton, halogen, cesium, and rubidium
- release fractions are from RG 1.183, Table 3
- depth of water above the damaged fuel of 23 feet is assumed
- overall effective decontamination factor of 200 is assumed
- iodine chemical form released from the pool is 57 percent elemental iodine and 43 percent organic iodide
- no reduction or mitigation of noble gas radionuclides released from the fuel is assumed
- radionuclides are released to the environment over a two-hour period

There are no single failures assumed for this event. Noble gases and iodines are released from the pool, while the cesiums and rubidiums are particulates and remain in the pool. The activity released from the pool to the RXB is assumed to be instantaneously released to the environment without holdup or mitigation. Doses are determined at the EAB, LPZ, and for personnel in the control room and TSC. The control room model is described in Section 15.0.3.7.1. The potential radiological consequences of a fuel handling accident are summarized in Table 15.0-12.

15.0.3.8.6 Radiological Analysis of the Iodine Spike Design-Basis Source Term

Section 15.0.3.1 discusses how a MHA has historically been linked to a large-break LOCA in large LWRs and that, for the NPM, a large-break LOCA cannot physically be postulated as the basis for the MHA radiological consequence analysis. Section 15.6.5 presents the LOCA analysis, which shows that no fuel failures occur. The NuScale design has DBEs that result in primary coolant entering an intact containment and the iodine spike DBST is used to bound the radiological consequences of these events. The design-basis iodine spike DBST and the beyond-design-basis CDST described in Section 15.10 are each assessed against the radiological criteria of

10 CFR 52.47(a)(2)(iv). If both the design-basis iodine spike DBST and the beyond-design-basis CDST analyses show acceptable dose results 10 CFR 52.47(a)(2)(iv) is met.

Reference 15.0-4 provides the methodology for the radiological consequences of the iodine spike DBST.

This radiological consequence analysis considers the iodine spike DBST with two different initial iodine concentrations, one based on a pre-incident iodine spike and the other based on a coincident iodine spike. A description of the evaluated scenario is summarized as follows:

1) A generic failure is assumed to occur inside the CNV, resulting in the release of all 46,700 kg of primary coolant from the RCS to the CNV.

2) The iodine and noble gas coolant activity is calculated based on the maximum concentrations allowed by design basis limits for each of the iodine spiking scenarios. The primary coolant contains a concentration of 3.7E-02 μCi/gm DE I-131 for the coincident iodine spike scenario and 2.2 μCi/gm DE I-131 for the pre-incident iodine spike scenario. For both iodine spiking scenarios, the primary coolant contains 10 μCi/gm DE Xe-133.

3) Primary coolant flows into the CNV through a nonspecific release point with an instantaneous release of activity into the CNV. The release is homogenously mixed as vapor throughout the entire CNV free volume.

4) Activity is then assumed to leak into the environment at the design basis leakage rate for 24 hours, then at 50 percent of the design basis leakage rate thereafter. The activity from this leak path is also assumed to flow directly to the environment with no mitigation or reduction by intervening structures.

5) At 30 hours, it is assumed the reactor is shut down and depressurized and releases through the containment to the environment stop.

The following is a summary of the assumptions used from Appendix E (main steam line break) of RG 1.183:

- Coincident iodine spiking factor - 500
- Duration of coincident iodine spike - 8 hours
- Iodine chemical form of 97 percent elemental iodine and 3 percent organic iodide
- Activity released from the fuel due to the pre-incident iodine spike is assumed to mix instantaneously and homogeneously within the primary coolant in the CNV; activity released from the fuel due to the coincident iodine spike is assumed to mix instantaneously and homogeneously within the fuel volume, then release to the CNV over the 8 hour coincident spiking duration
- No reduction or mitigation of noble gas radionuclides released from the primary system

RADTRAD is used to determine the dose, as outlined in Section 15.0.3.3.8. There are no single failures assumed for this event. The control room model is described in Section 15.0.3.7.1. The potential radiological consequences of the iodine spike DBST are presented in Table 15.0-12.

15.0.4 Safe, Stabilized Condition

Safety analyses of design basis events are performed from event initiation until a safe, stabilized condition is reached. A safe, stabilized condition is reached when the initiating event is mitigated, the acceptance criteria are met and system parameters (for example inventory levels, temperatures and pressures) are trending in the favorable direction. For events that involve a reactor trip, system parameters continue changing slowly as decay and residual heat are removed and the RCS continues to cool down. No operator action is required to reach or maintain a safe, stabilized condition.

Two additional considerations are discussed to show that Chapter 15 acceptance criteria are not challenged beyond the safe, stabilized condition. Long term decay and residual heat removal is discussed in Section 15.0.5 and a potential return to power is discussed in Section 15.0.6.

15.0.5 Long Term Decay and Residual Heat Removal

There are two systems that perform the safety-related function of decay and residual heat removal from the NPM following a DBE. The DHRS, described in Section 5.4.3, provides decay and residual heat removal while RCS inventory is retained inside the RPV, the containment is maintained in partially evacuated dry conditions, and power is available. The ECCS, described in Section 6.3, provides decay and residual heat removal when RCS inventory has been redistributed between the RPV and the CNV after the RVVs and RRVs are opened.

The DBEs listed in Table 15.0-1 progress from initiation of the event to effective DHRS or ECCS operation demonstrating that the NPM has reached a safe, stabilized condition, as described in Section 15.0.4. The decay heat removal process continues into the long-term phase, either with DHRS, natural circulation between the CNV and RPV through the RRVs and RVVs, or a combination of the two.

There are four decay and heat removal scenarios:

1) DHRS,

2) DHRS with the RVVs and RRVs opening 24 hours after a loss of normal AC power,

3) DHRS with the RVVs and RRVs opening after a loss of normal AC and normal DC power when the IAB pressure threshold is reached, and

4) ECCS actuation following an inadvertent opening of a reactor coolant pressure boundary (RCPB) valve or a LOCA.

Scenario 1 - Decay and Residual Heat Removal using DHRS

Non-LOCA events progress from event initiation to the point where DHRS actuation valves open and the MSIVs and FWIVs close to allow DHRS operation. The progression of decay heat removal using DHRS depends on the availability of AC power.

With AC power available, DHRS cools the NPM and provides long term removal of decay heat while the RRVs and RVVs remain closed. Section 5.4.3 describes the operation of DHRS, including actuation, cooling to the safe, stabilized condition, and long term residual and decay heat removal.

Scenarios 2 and 3 - Decay and Residual Heat Removal using DHRS followed by Natural Circulation through the RVVs and RRVs

For non-LOCA events that results in DHRS actuation, if onsite AC power is lost, DC power to the RVVs and RRVs is automatically removed after 24 hours and the RVVs and RRVs go to a fail-safe open position. If the non-LOCA event analysis assumes that AC and DC power are lost, which results in power removed from the RVVs and RRVs, then the RVVs and RRVs are maintained closed by the IAB mechanism. The IAB mechanism prevents RVV and RRV actuation at high RCS pressures. The RVVs and RRVs go to a fail-safe open position when the RCS pressure decreases below the IAB release pressure. Therefore, long-term decay and residual heat removal is accomplished with DHRS followed by natural circulation through the RVVs and RRVs.

Opening the RVVs and RRVs to depressurize the RCS and establish long term cooling is not considered an event escalation because the functions of the RCS barrier are not lost. The progression of cooling function from DHRS to natural circulation using the RVVs and RRVs is an inherent function in the passive design of the NPM. The RCS barrier continues to provide a confined volume for reactor coolant which allows a flow path for cooling the core and thus, confining fission products to the fuel and preventing an escalation of a DBE, including an AOO.

Scenario 4: Decay and Residual Heat Removal using ECCS following an Inadvertent Opening of an RCPB valve or LOCA

The system response in terms of potential challenge to the fuel from an inadvertent opening of an RVV, as described in Section 15.6.6, bounds other RCPB valve opening events as well as other non-LOCA events that transition from DHRS to natural circulation through the RVVs and RRVs. The rate of depressurization after an inadvertent opening of an RVV is more rapid compared to the rate of depressurization after opening other RCPB valves at full power or the RVVs and RRVs following other non-LOCA events. After the RVVs and RRVs open, RCS inventory is redistributed between the RPV and CNV and the NPM enters the same cooling configuration, irrespective of the initiating event. The results of the long term cooling analysis are summarized in Table 15.0-22.

The LOCA analysis, including the analysis of long term cooling following a LOCA per 10 CFR 50.46(b)(5), is discussed in Section 15.6.5.

15.0.6 Evaluation of a Return to Power

Having all control rods inserted provides the safety-related means to maintain the reactor shut down for internal events and for hazards such as floods and fires in the plant, earthquakes, severe weather conditions, external fires, and external floods. With all control rods inserted, a return to power is precluded. For design basis analysis of internal events for which the worst control rod is assumed stuck out, a return to power is highly unlikely. However, a return to power is evaluated for various cooldown progressions to demonstrate that fuel design limits are not challenged. As described in Section 4.3, a failure in reactivity control system reliability to ensure long term shutdown is calculated to be less than 1E-5 per NPM-reactor year. With the highest worth control rod assembly stuck out and the chemical and volume control system unavailable, subcritical core conditions (k_{eff}<1.0) are demonstrated, for 72 hours after a DBE using nominal analysis assumptions, except for the condition where initial boron concentration is very low. The probability of reactivity control systems failing during the first 72 hours after shutdown within the small window of initial conditions that can lead to a return to power is conservatively calculated to be less than 1E-6 per NPM-reactor year.

In the unlikely event of a return to power, shutdown with margin for stuck rods is not required to demonstrate adequate fuel protection. Fuel is protected through physical processes inherent to the NuScale design that control reactivity and limit power compared to a design in which shutdown is required to limit power production to protect fuel integrity. In the NPM design, additional protection is provided by limiting power and passively removing heat. The means for limiting the power produced if the reactor does not remain shut down is dependent on the heat removal system used.

15.0.6.1 Identification of Causes and Accident Description

Design basis events are analyzed with an assumed highest worth control rod stuck fully withdrawn in order to evaluate the immediate shutdown capability of the negative reactivity insertion due to a reactor trip with the control rods inserting into the core, consistent with GDC 26 (See Section 3.1). In the event of an extended cooldown, when the RCS is at low boron concentrations and the CVCS is unavailable to add boron, it may be possible to cool the core to the point of reestablishing some level of critical neutron power if the most reactive control rod stuck out is assumed. This potential overcooling could cause a unique reactivity event similar to a steam line break for traditional multi-loop PWRs. Therefore, this event is specifically evaluated for specified acceptable fuel design limits (SAFDLs).

The purpose of this analysis is to evaluate the thermal hydraulic and core neutronic response of the NPM for an extended overcooling return to power. This analysis is intended to provide a generic bounding evaluation of the extended cooling that could result following any DBE, therefore AOO acceptance criteria and conservative analysis assumptions are applied. The limiting return to power event occurs when operating conditions are biased to maximize initial core fission product poisons which gradually decay resulting in reactivity insertion. The timing of this reactivity insertion occurs well after equilibrium DHRS or ECCS passive cooling modes will have been established following an initial transient and reactor trip. Therefore, analysis of the return to power is limited to the equilibrium thermal hydraulic and neutronic conditions with

appropriate biases and conservatisms to ensure a conservative CHF analysis is performed.

15.0.6.2 Sequence of Events and Systems Operation

For the overcooling return to power event, it is assumed that a reactor trip occurs at end of cycle (EOC) with the most reactive control rod stuck out of the core. The decay of xenon slowly adds positive reactivity during the cooldown. The subsequent cooldown is left unmitigated and boron addition does not occur. While there are simple operational means for mitigating the extended cooldown and thereby eliminating the need for boron addition, operator action is not credited for either mitigating the cooldown or adding boron, consistent with Section 15.0.0.6.4.

15.0.6.3 Thermal Hydraulic and Critical Heat Flux Analyses

15.0.6.3.1 Evaluation Models

The overcooling return to power analysis is performed using the following analysis procedure:

- The core average RCS temperature is determined using the long term cooling statepoint analysis approach described in the LTC technical report.
- The worst rod stuck out, EOC critical power level is determined using the SIMULATE5 core physic analysis model.
- CHF margin is evaluated using the zero flow CHF correlation described in the LOCA EM topical report.

The MCHFR analysis uses the CHF correlation applied in the LOCA evaluation model, evaluated against the 95/95 CHFR acceptance criterion of an AOO, as described in Reference 15.0-3.

SIMULATE5 is an advanced three-dimensional (3D), steady-state, multi-group nodal reactor analysis code capable of multi-dimensional nuclear analyses of reactors. A discussion of SIMULATE5 is provided in Section 4.3.

15.0.6.3.2 Input Parameters and Initial Conditions

As stated above, this event is analyzed specifically for the parameters that generate the most severe overcooling return to power core power event. The following assumptions ensure that the equilibrium power results have sufficient conservatism.

- The core is assumed to be at hot full power and end of cycle (5 ppm boron concentration) conditions prior to the transient initiation.
- A critical boron concentration (CBC) nuclear reliability factor (NRF) is used in this analysis.
- The ECCS valve capacity is maximized to increase the efficiency of heat transfer from the RPV to the UHS.

- The DHRS heat transfer is increased by 30 percent to ensure the consequences of the cooldown are maximized after DHRS actuation.
- A reactor pool level of 69 feet and a temperature of 65 degrees Fahrenheit is used leading to a conservatively high cooldown rate, which adds the maximum positive reactivity.
- A time-dependent xenon worth is used in this analysis for the purposes of calculating timing of return to power only. The core is assumed to be at EOC conditions at the time of event initiation with equilibrium fission products. The xenon worth specified in the input is determined from the time dependent decay of the fission products that are present in the EOC core.

No single failure is assumed. Failure of the main steam or feedwater isolation valves to close could result in a reduction of DHRS cooling, which would be non-conservative for the overcooling return to power event. Full ECCS actuation will be more limiting for CHF, therefore, an ECCS valve failure to open is not considered.

For the limiting MCHFR portion of the analysis, the following conservatisms are applied:

- A dynamic return to power factor of 2.0 is applied to the equilibrium power level to bound any potential overshoot of the equilibrium power.
- The maximum radial peaking ($F_{\Delta H}$) due to the stuck control rod is 7.5. The return to power is driven by the lack of necessary negative reactivity insertion due to the postulated most reactive control rod stuck in a fully withdrawn position. The critical power will be localized in this region of large radial peaking.
- A maximum F_Q was chosen with additional penalty for variation in axial peaking.
- A critical boron concentration (CBC) nuclear reliability factor (NRF) is used in the determination of the critical power level for the limiting MCHFR analysis.

15.0.6.3.3 Results

This analysis provides a conservative characterization of the equilibrium power and corresponding critical heat flux ratio, should a return to power occur. Additionally, the time of return to power is evaluated based on time-dependent xenon and thermal-hydraulic conditions.

For several different cooldown modes and pool temperature conditions, the nominal equilibrium power level and MCHFR are summarized in Table 15.0-16. The limiting equilibrium power level and MCHFR are provided in Table 15.0-17. The nominal results for the limiting pool temperature are included in Figure 15.0-8. The results for a pool temperature of 140 degrees F are provided in Figure 15.0-10.

- The maximum equilibrium power level occurs for the ECCS cooldown mode with a 65 degrees F pool.

- The maximum equilibrium power is approximately 2.9 MW.
- Several of the cases do not return to a critical condition within the 72 hour window analyzed. The earliest return to power occurs at approximately 40 hours post scram.
- The timing of the initial recriticality demonstrates that the return to power event does not occur during the short-term RCS de-energizing phase, but instead is the result of the slow decay of xenon in the long-term equilibrium phase between decay heat and RCS temperature.
- Results show that the equilibrium power decreases with increasing pool temperature.
- The MCHFR is well above the analytical limit, therefore it is concluded that the SAFDLs are ensured should a limited return to power occur following an unmitigated cooldown, regardless of initiating event or time in cycle in which it occurs.

15.0.6.3.4 Conclusions

The AOO acceptance criteria outlined in Table 15.0-2 are used as the basis for the overcooling return to power event. The acceptance criteria, followed by how the NuScale design meets them, are listed below:

1) Potential core damage is evaluated on the basis that it is precluded if the minimum departure from nucleate boiling ratio (DNBR) remains above the 95/95 DNBR limit. Minimum critical heat flux ratio is used instead of minimum DNBR, as described in Section 4.4.2.

   Fuel integrity is not challenged by an overcooling return to power event. The limiting MCHFR is shown in Table 15.0-17. The MCHFR is evaluated using the stagnant flow CHF correlation, therefore the 95/95 design limit is 1.37. The CHF analysis confirms that the overcooling return to power event does not challenge MCHFR limits.

2) RCS pressure should be maintained below 110 percent of the design value.

   Due to the nature of the overcooling return to power event, primary pressure is not challenged and is non-limiting for this event.

3) The main steam pressure should be maintained below 110 percent of the design value.

   Due to the nature of the overcooling return to power event, main steam pressure is not challenged and is non-limiting for this event.

4) The event should not generate a more serious plant condition without other faults occurring independently.

   The overcooling return to power analysis demonstrates that DBEs, where a most reactive control rod is assumed stuck out upon reactor trip, can be safely cooled by DHRS or ECCS, without challenging MCHFR limits.

The evaluation of an overcooling return to power event demonstrates that design limits are not exceeded and the overcooling return to power event is non-limiting with respect to DBEs.

15.0.7 References

15.0-1 NuScale Power, LLC, "Subchannel Analysis Methodology," TR-0915-17564-P-A, Rev. 2.

15.0-2 NuScale Power, LLC, "NuScale Power Critical Heat Flux Correlations," TR-0116-21012-P-A, Rev. 1.

15.0-3 NuScale Power, LLC, "Loss-of-Coolant Accident Evaluation Model," TR-0516-49422, Rev. 0.

15.0-4 NuScale Power, LLC, "Accident Source Term Methodology," TR-0915-17565, Rev. 3.

15.0-5 NuScale Power, LLC, "Non-Loss-of-Coolant Accident Transient Analysis Methodology," TR-0516-49416, Rev. 1.

15.0-6 Nuclear Energy Institute, "Small Modular Reactor Source Terms," [Position Paper] December 27, 2012, Washington, DC.

15.0-7 NuScale Power, LLC, "Long-Term Cooling Methodology," TR-0916-51299, Rev. 1.

15.0-8 U.S. Environmental Protection Agency, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion," Federal Guidance Report 11, EPA-520/1-88-020, 1988.

15.0-9 U.S. Environmental Protection Agency, "External Exposure to Radionuclides in Air, Water, and Soil," Federal Guidance Report 12, EPA-402-R-93-081, 1993.

15.0-10 NuScale Power, LLC, "Evaluation Methodology for Stability Analysis of NuScale Power Module," TR-0516-49417, Rev. 1.

15.0-11 NuScale Power, LLC, "NuScale Rod Ejection Accident Methodology," TR-0716-50350, Rev. 0.

15.0-12 Nuclear Regulatory Commission, "Staff Requirements - SECY-19-0036 - Application of the Single Failure Criterion to NuScale Power LLC's Inadvertent Actuation Block Valves," dated July 2, 2019.

Table 15.0-1: Design Basis Events

Section	Type	Classification	Computer Code Used
15.0	**Transient and Accident Analysis**		
15.0.3	Iodine Spike Design-Basis Source Term (10 CFR 52.47(a)(2)(iv))	N/A[7]	RADTRAD ORIGEN ARCON96
15.0.6	Return to Power Event - NuScale specific event progression	N/A[6]	NRELAP5
15.1	**Increase in Heat Removal by Secondary System**[8]		
15.1.1	Decrease in Feedwater Temperature	AOO	NRELAP5 VIPRE-01
15.1.2	Increase in Feedwater Flow	AOO	NRELAP5 VIPRE-01
15.1.3	Increase in Steam Flow	AOO	NRELAP5 VIPRE-01
15.1.4	Inadvertent Opening of Steam Generator Relief or Safety Valve	AOO	NRELAP5 VIPRE-01
15.1.5	Steam Piping Failures Inside and Outside of Containment	Postulated Accident	NRELAP5 VIPRE-01 RADTRAD ORIGEN ARCON96
15.1.6	Loss of Containment Vacuum/Containment Flooding	AOO	NRELAP5 VIPRE-01
15.2	**Decrease in Heat Removal by the Secondary System**[8]		
15.2.1	Loss of External Load	AOO	NRELAP5 VIPRE-01
15.2.2	Turbine Trip	AOO	NRELAP5 VIPRE-01
15.2.3	Loss of Condenser Vacuum	AOO	NRELAP5 VIPRE-01
15.2.4	Closure of Main Steam Isolation Valve	AOO	NRELAP5 VIPRE-01
15.2.5	Steam Pressure Regulator Failure (Closed)	N/A[1]	N/A
15.2.6	Loss of Non-Emergency AC to the Station Auxiliaries	AOO	NRELAP5 VIPRE-01

Table 15.0-1: Design Basis Events (Continued)

Section	Type	Classification	Computer Code Used
15.2.7	Loss of Normal Feedwater Flow	AOO	NRELAP5 VIPRE-01
15.2.8	Feedwater System Pipe Breaks Inside and Outside Containment	Postulated Accident	NRELAP5 VIPRE-01 RADTRAD ORIGEN ARCON96
15.2.9	Inadvertent Operation of the Decay Heat Removal System	AOO	NRELAP5 VIPRE-01
15.3	**Decrease in RCS Flow Rate (not applicable)**		
15.4	**Reactivity and Power Distribution Anomalies**[8]		
15.4.1	Uncontrolled Control Rod Assembly Withdrawal from a Subcritical or Low Power or Startup	AOO	NRELAP5 VIPRE-01
15.4.2	Uncontrolled Control Rod Assembly Withdrawal at Power	AOO	NRELAP5 VIPRE-01
15.4.3	Control Rod Misoperation (System Malfunction or Operator Error)	AOO	VIPRE-01 SIMULATE5
15.4.4	Startup of an Inactive Loop or Recirculation Loop at an Incorrect Temperature	N/A[1]	N/A
15.4.5	Flow Controller Malfunction Causing an Increase in Core Flow Rate (Boiling Water Reactor)	N/A[1]	N/A
15.4.6	Inadvertent Decrease in Boron Concentration in Reactor Coolant System	AOO	N/A
15.4.7	Inadvertent Loading and Operation of a Fuel Assembly in an Improper Position	IE	SIMULATE5, VIPRE-01
15.4.8	Spectrum of Rod Ejection Accidents	Postulated Accident	SIMULATE-3K NRELAP5 VIPRE-01 RADTRAD ORIGEN ARCON96
15.5	**Increase in Reactor Coolant Inventory**[8]		
15.5.1	Chemical and Volume Control System Malfunction	AOO	NRELAP5 VIPRE-01

Table 15.0-1: Design Basis Events (Continued)

Section	Type	Classification	Computer Code Used
15.6	**Decrease in Reactor Coolant Inventory**[8]		
15.6.1	Inadvertent Opening of Reactor Safety Valve	AOO	See 15.6.6
15.6.2	Failure of Small Lines Carrying Primary Coolant Outside Containment	IE	NRELAP5 RADTRAD ORIGEN ARCON96
15.6.3	Steam Generator Tube Failure	Postulated Accident	RADTRAD NRELAP5 ORIGEN ARCON96
15.6.4	Main Steam Line Failure Outside Containment (BWR)	N/A[1]	N/A
15.6.5	Loss-of-Coolant Accidents Resulting From a Spectrum of Postulated Piping Breaks Within the Reactor Coolant Pressure Boundary	Postulated Accident	NRELAP5
15.6.6	Inadvertent Operation of Emergency Core Cooling System	AOO	NRELAP5
15.7	**Radioactive Release from a Subsystem or Component**		
15.7.1	Gaseous Waste Management System Leak or Failure	N/A[2]	N/A
15.7.2	Liquid Waste Management System Leak or Failure	N/A[2]	N/A
15.7.3	Postulated Radioactive Releases Due to Liquid Containing Tank Failures	N/A[2]	RADTRAD, ORIGEN, ARCON96
15.7.4	Fuel Handling Accidents	Postulated Accident	RADTRAD, ORIGEN, ARCON96
15.7.5	Spent Fuel Cask Drop Accident	Postulated Accident	Not analyzed
15.7.6	NuScale Power Module Drop Accident	N/A[3]	Not analyzed
	Special Events		
15.8	Anticipated Transient Without Scram(10 CFR 50.62)	Special Event	No analysis required.
15.9	Stability - note that stability is not an event. The NPM is protected from this phenomenon by MPS trips and technical specification initial conditions.	N/A[4]	PIM

Table 15.0-1: Design Basis Events (Continued)

Section	Type	Classification	Computer Code Used
15.10	Core Damage Source Term (10 CFR 52.47(a)(2)(iv))	Special Event	RADTRAD ORIGEN STARNAUA pH_T ARCON96 MELCOR
8.4	Station Blackout (10 CFR 50.63)	N/A(5)	NRELAP5

Notes:

(1) Design feature is not part of NuScale design.
(2) Events are described in Chapter 11.
(3) Module drop is considered a Beyond Design Basis Event.
(4) Event is analyzed to AOO Acceptance Criteria.
(5) Event is included in the loss of non-emergency AC power analysis described in Section 15.2.6.
(6) This is not an initiating event, however, AOO acceptance criteria are met. See Section 15.0.6 for details.
(7) The iodine spike DBST is not an event, rather it serves as a bounding surrogate for design-basis loss of primary coolant into containment events described in Section 15.6.
(8) A return to power can occur during the progression of events that involve a cooldown using DHRS or ECCS cooling when decay heat levels and boron concentration are low and control rods are not fully inserted.

Table 15.0-2: Acceptance Criteria-Thermal Hydraulic and Fuel

Classification[5]	Fuel Clad[1]	RCS Pressure	Main Steam System Pressure	Containment	Event Progression
AOO	Fuel cladding integrity shall be maintained by ensuring that SAFDLs are met.[6]	≤ 110% of system design pressure	≤ 110% of system design pressure	Peak pressure ≤ design pressure[4]	An AOO should not develop into a more serious plant condition without other faults occurring independently. Satisfaction of this criterion precludes the possibility of a more serious event during the lifetime of the plant.
IE	Fuel cladding integrity shall be maintained by ensuring that SAFDLs are met.[6]	≤ 120% of system design pressure	≤ 120% of system design pressure	Peak pressure ≤ design pressure[4]	Shall not, by itself, cause a consequential loss of required functions of systems needed to cope with the fault, including those of the RCS and the reactor containment system.
Postulated Accidents[2],[3]	Fuel cladding integrity shall be maintained by ensuring that SAFDLs are met.[6][7]	≤ 120% of system design pressure	≤ 120% of system design pressure	Peak pressure ≤ design pressure[4]	Shall not, by itself, cause a consequential loss of required functions of systems needed to cope with the fault, including those of the RCS and the reactor containment system.
Special Event (SBO)	Core cooling	refer to Section 8.4	N/A	N/A	N/A

Notes:

(1) Minimum critical heat flux ratio (MCHFR) is used instead of minimum DNBR, as described in Section 4.4.2.

(2) See Table 15.0-3 for acceptance criteria for the Rod Ejection Accident.

(3) See Table 15.0-4 for acceptance criteria for Loss of Coolant Accidents.

(4) See Section 6.2.1.1 for containment pressure design limits.

(5) The iodine spike DBST and core damage event associated CDST do not have thermal hydraulic or fuel acceptance criteria.

(6) Specified Acceptable Fuel Design Limits (SAFDLs) are met by assuring that MCHFR is maintained above the 95/95 limit.

(7) SAFDLs are met during postulated accidents to ensure fuel cladding integrity is maintained should a return to power occur during the progression of the event.

Table 15.0-3: Acceptance Criteria Specific to Rod Ejection Accidents

Purpose	Conditions	Parameter	Acceptance Criteria
To assure no fuel failure occurs	Zero power	Maximum peak radial average fuel enthalpy	≤ 100 cal/g
	5% to 100% power	Minimum DNBR(1)	≥ 95/95 DNBR limit(1)
	Function of cladding oxide/wall thickness ratio	Change in radial average fuel enthalpy	See Reference 15.0-11, Figure 5-2
	N/A	Maximum peak fuel temperature	≤ melting temperature
For assessment of core coolability	N/A	Maximum peak radial average fuel enthalpy	≤ 230 cal/g and MCHFR ≥ 95/95 DNBR limit(1)
	N/A	Maximum peak fuel temperature	≤ melting temperature
	N/A	Fuel pellet cladding fragmentation and dispersal	Coolable geometry maintained
	N/A	Fuel rod ballooning	Coolable geometry maintained

Notes:

(1) Minimum critical heat flux ratio (MCHFR) is used instead of minimum DNBR, as described in Section 4.4.2.

Table 15.0-4: Acceptance Criteria Specific to Loss of Coolant Accidents

Parameter[1]	Acceptance Criteria
Maximum fuel element cladding temperature	MCHFR > 1.29 and collapsed water level above the top of active fuel.
Maximum total oxidation of cladding	MCHFR > 1.29 and collapsed water level above the top of active fuel.
Maximum total hydrogen generated from chemical reaction of cladding with water or steam	MCHFR > 1.29 and collapsed water level above the top of active fuel.
Core geometry	MCHFR > 1.29 and collapsed water level above the top of active fuel.[2]
Long term cooling	Following initial successful operation of the ECCS, the calculated core temperature is maintained at an acceptably low value and decay heat is removed for the extended time required by the long-lived radioactivity remaining in the core.

Notes:

(1) From 10 CFR 50.46.

(2) Note that this is typically met by demonstrating that the acceptance criteria are met, and in addition, by assuring that the fuel deformation due to combined LOCA and seismic loads is specifically addressed.

Table 15.0-5: Acceptance Criteria-Radiological

Event	Analysis Release Duration	Exclusion Area Boundary And Low Population Zone[1][2] (Rem-TEDE)	Control Room[3] (Rem-TEDE)
Loss of Coolant Accident (iodine spike DBST)	30 days for leakage pathways.	25	5
Rod Ejection Accident	Per Appendix H of RG 1.183, a radiological analysis is not required as the consequences of this event are bounded by the consequences of other analyzed events.	6.3	5
Steam Generator Tube Failure	Affected steam generator: until time to isolation. Unaffected steam generator: until reactor shut down and depressurized.	25 (Fuel Damage or pre-incident spike) 2.5 (Coincident Iodine Spike)	5
Main Steam Line Break	Until reactor shut down and depressurized.	25 (Fuel Damage or pre-incident spike) 2.5 (Coincident Iodine Spike)	5
Fuel Handling Accidents	2 hours	6.3	5
Small Lines Carrying Primary Coolant Outside Containment	Until isolation, if capable, or until reactor shut down and depressurized.	2.5	5
Feedwater System Pipe Breaks Inside and Outside Containment	Radiological analysis is not required as the radiological consequences of the FWLB event are bounded by the consequences of a steam line break discussed in Section 15.0.3.	2.5[4]	5
Core Damage Event	30 days for leakage pathways.	25	5

Notes:

(1) Based on 10 CFR 52.47 (LOCA), RG 1.183, and 10 CFR 20.1301.
(2) Individual at the EAB shall not receive dose limit for any 2 hour period flowing the onset of release.

(3) Based on 10 CFR 52.47 and is for the duration of the event.

(4) Small fraction (10%) of regulatory dose reference value (25 rem TEDE).

Table 15.0-6: Module Initial Conditions Ranges for Design Basis Event Evaluation

Plant Parameter	Units	Value	Uncertainty (Bias)	Basis
Design core power	%	100	+2	Maximum initial core power is assumed to be 102% power due to uncertainty. Rated power is 160 MW.
RCS T_{avg} at operating conditions.	°F	545	± 10	RCS average temperature is held constant above 15% power. See Table 5.1-2.
RCS T_{avg} at startup	°F	420 - 555	N/A	The minimum temperature for criticality is 420° F. The temperature range is given as the minimum temperature for criticality up to the RCS T_{avg} at operating conditions plus the high bias (545+10).
Pressurizer Pressure	psia	1850	± 70	Nominal operating pressure is 1850 psia in the pressurizer. The analysis range is 70 psia from the nominal condition.
Pressurizer level at core power ≥15% Rated thermal power (RTP)	%	60	± 8	Hot full power (HFP) pressurizer level is 60%. The analysis range of ±8% is applied to the HFP level.
Pressurizer level at core power ≤15% RTP	%	42 - 68	N/A	Nominal level is 50% at hot zero power (HZP). A low range of 8% is applied to the nominal HZP level. The upper level is defined as the level at HFP plus the high bias (60+8).
Containment pressure	psia	0 - 3 psia	N/A	Nominal operating pressure is less than 1 psia. The maximum of the analytical range is 3.0 psia, which is chosen to bound the initial pressure at which the leak detection system will not be available.
Main steam pressure at 100% RTP	psia	500	± 35	This value is a function of reactor power.
Feedwater temperature at 100% RTP	°F	300	± 10	This value is a function of reactor power.
RCS flow at 100% RTP	lbm/s	1180 - 1480*	N/A	Flow range intended to slightly bound the minimum and maximum flowrates calculated. RCS natural circulation flow is primarily a function of operating power and the loop hydraulic resistances. The specified flow range is not related to control deadband or instrumentation uncertainty.
RCS flow minimum	ft^3/s	1.7	N/A	Minimum flow rate for low power operation. This flow rate ensures a maximum RCS loop time. The specified flow rate is not related to control deadband or instrumentation uncertainty.

Note:

*Flow rate 535-670 kg/s converted to lbm/s.

Table 15.0-7: Analytical Limits and Time Delays

Signal	Analytical Limit	Basis and Event Type	Actuation Delay
High Power	120%[5] rated thermal power (RTP) (≥ 15% RTP) 25% RTP (<15% RTP)	This signal is designed to protect against exceeding critical heat flux (CHF) limits for reactivity and overcooling events.	2.0 sec
Source and Intermediate Range Log Power Rate	3 decades/min	This signal is designed to protect against exceeding CHF and energy deposition limits during startup power excursions	Variable
High Power Rate	±15% RTP/min	This signal is designed to protect against exceeding CHF limits for reactivity and overcooling events.	2.0 sec
High Startup Range Count Rate	5.0 E+05 counts per second[6]	This signal is designed to protect against exceeding CHF and energy deposition limits during rapid startup power excursions.	3.0 sec
High Subcritical Multiplication	3.2	This signal is designed to detect and mitigate inadvertent subcritical boron dilutions in operating modes 2 and 3.	150.0 sec
High Reactor Coolant System (RCS) Hot Temperature	610°F	This signal is designed to protect against exceeding CHF limits for reactivity and heatup events.	8.0 sec
High Containment Pressure	9.5 psia	This signal is designed to detect and mitigate RCS or secondary leaks above the allowable limits to protect RCS inventory and emergency core cooling system (ECCS) function during these events.	2.0 sec
High Pressurizer Pressure	2000 psia	This signal is designed to protect against exceeding reactor pressure vessel (RPV) pressure limits for reactivity and heatup events.	2.0 sec
High Pressurizer Level	80%	This signal is designed to detect and mitigate chemical and volume control system (CVCS) malfunctions to protect against overfilling the pressurizer.	3.0 sec
Low Pressurizer Pressure	1720 psia[1]	This signal is designed to detect and mitigate primary high energy line break (HELB) outside containment and protect RCS subcooled margin for protection against instability events.	2.0 sec
Low Low Pressurizer Pressure	1600 psia[2]	This signal is designed to detect and mitigate primary HELB outside containment and protect RCS subcooled margin for protection against instability events.	2.0 sec
Low Pressurizer Level	35%	This signal is designed to protect the pressurizer heaters from uncovering and overheating during decrease in RCS inventory events.	3.0 sec
Low Low Pressurizer Level	20%	This signal is designed to detect and mitigate loss-of-coolant accidents (LOCAs) to protect RCS inventory and ECCS functionality during LOCA and primary HELB outside containment events.	3.0 sec
Low Low Main Steam Pressure	20 psia (at ≤15% RTP)	This signal is designed to detect and mitigate secondary HELB outside containment to protect steam generator inventory and decay heat removal system (DHRS) functionality.	2.0 sec
Low Main Steam Pressure	300 psia (at >15% RTP)	This signal is designed to detect and mitigate secondary HELB outside containment to protect steam generator inventory and DHRS functionality.	2.0 sec

Table 15.0-7: Analytical Limits and Time Delays (Continued)

Signal	Analytical Limit	Basis and Event Type	Actuation Delay
High Main Steam Pressure	800 psia	This signal is designed to detect and mitigate loss of main steam demand to protect primary and secondary pressure limits during heatup events.	2.0 sec
High Main Steam Superheat	150°F	This signal is designed to detect and mitigate steam generator boil off to protect DHRS functionality during at power and post trip conditions.	8.0 sec
Low Main Steam Superheat	0.0°F	This signal is designed to detect and mitigate steam generator overfilling to protect DHRS functionality during at power and post trip conditions.	8.0 sec
Low RCS Flow	1.7 ft^3/s	This signal is designed to ensure boron dilution cannot be performed at low RCS flowrates where the loop time is too long to be able to detect the reactivity change in the core within sufficient time to mitigate the event.	6.0 sec
Low Low RCS Flow	0.0 ft^3/s	This signal is designed to ensure flow remains measureable and positive during low power startup conditions.	6.0 sec
High CNV Water Level	300-264"(3) (elevation)	This signal is designed to protect water level above the core in LOCA events.	3.0 sec
Low AC voltage	Note 4	This signal is designed to ensure appropriate load shedding occurs to EDSS in the event of extended loss of normal AC power to the EDSS battery chargers.	60.0 sec
High Under-the-Bioshield Temperature	250°F	This signal is designed to detect high energy leaks or breaks at the top of the NuScale Power Module under the bioshield to reduce the consequences of high energy line breaks on the safety related equipment located on top of the module.	8.0 sec

Notes:

1. If RCS hot temperature is above 600°F. See Figure 15.0-9.
2. If RCS hot temperature is below 600°F. See Figure 15.0-9.
3. CNV water level is presented in terms of elevation where reference zero is the bottom of the reactor pool. The range allows ±18" from the nominal ECCS level setpoint of 282".
4. Normal AC voltage is monitored at the bus(es) supplying the battery chargers for the highly reliable DC power system. The analytical limit is based on Loss of Normal AC Power to plant buses (0 volts) but the actual bus voltage is based upon the voltage ride-thru characteristics of the EDSS battery chargers.
5. The overcooling event analyses account for a cooldown event specific process error analytical limit of 0.5%/°F.
6. The high count rate trip is treated as a source range over power trip that occurs at a core power analytical limit of 500kW which functionally equates neutron monitoring system counts per second to core power. This trip is bypassed once the intermediate range signal has been established.

Table 15.0-8: Reactivity Coefficients

Section	Design Basis Event	Power Level % HFP (160 MWt)	Moderator Temperature Coefficient(1)	Doppler Coefficient(1)
15.1	**Increase in Heat Removal by the Secondary System**			
15.1.1	Decrease in Feedwater Temperature	102%	-43.0 pcm/°F	-2.50 pcm/°F
15.1.2	Increase in Feedwater Flow	102%	-43.0 pcm/°F	-1.40 pcm/°F
15.1.3	Increase in Steam Flow	102%	-43.0 pcm/°F	-2.50 pcm/°F
15.1.4	Inadvertent Opening of Steam Generator Relief or Safety Valve	102%	-43.0 pcm/°F	-2.50pcm/°F
15.1.5	Steam Piping Failures Inside and Outside of Containment	102%	0.0 pcm/°F MCHFR, SG pressure, limiting iodine spiking cases -7.70 to -41.47 pcm/°F (function of temperature, RCS pressure and maximum mass release cases)	-1.40 pcm/°F -1.81 pcm/°F
15.1.6	Loss of Containment Vacuum/Containment Flooding	102%	-43.0 pcm/°F	-2.50 pcm/°F
15.2	**Decrease in Heat Removal by the Secondary System**			
15.2.1	Loss of External Load	102%	0.0 pcm/°F	-1.40 pcm/°F
15.2.2	Turbine Trip	102%	0.0 pcm/°F	-1.40 pcm/°F
15.2.3	Loss of Condenser Vacuum	102%	0.0 pcm/°F	-1.40 pcm/°F
15.2.4	Closure of Main Steam Isolation Valve	102%	0.0 pcm/°F	-1.40 pcm/°F
15.2.5	Steam Pressure Regulator Failure (Closed)	N/A	N/A	N/A
15.2.6	Loss of Non-Emergency AC to the Station Auxiliaries	102%	0.0 pcm/°F	-1.40 pcm/°F
15.2.7	Loss of Normal Feedwater Flow	102%	0.0 pcm/°F	-1.40 pcm/°F
15.2.8	Feedwater System Pipe Breaks Inside and Outside Containment	102%	0.0 pcm/°F	-1.40 pcm/°F
15.2.9	Inadvertent Operation of the Decay Heat Removal System	102%	0.0 pcm/°F	-1.40 pcm/°F
15.3	**Decrease in RCS Flow Rate (not applicable)**			
15.4	**Reactivity and Power Distribution Anomalies**			
15.4.1	Uncontrolled Control Rod Assembly Withdrawal from a Subcritical or Low Power or Startup Condition	1W - 15%	+6 pcm/°F	-1.40 pcm/°F

Table 15.0-8: Reactivity Coefficients (Continued)

Section	Design Basis Event	Power Level % HFP (160 MWt)	Moderator Temperature Coefficient[1]	Doppler Coefficient[1]
15.4.2	Uncontrolled Control Rod Assembly Withdrawal at Power	102% 75% 50% 25%	0.0 pcm/°F	-1.377 pcm/°F
15.4.3	Control Rod Misoperation (System Malfunction or Operator Error)	See Section 15.4.3	See Section 15.4.3	See Section 15.4.3
15.4.4	Startup of an Inactive Loop or Recirculation Loop at an Incorrect Temperature	N/A	N/A	N/A
15.4.5	Flow Controller Malfunction Causing an Increase in Core Flow Rate (BWR)	N/A	N/A	N/A
15.4.6	Inadvertent Decrease in Boron Concentration in Reactor Coolant System	See 15.4.1 and 15.4.2[2]	See 15.4.1 and15.4.2[2]	See 15.4.1 and 15.4.2[2]
15.4.7	Inadvertent Loading and Operation of a Fuel Assembly in an Improper Position	N/A	N/A	N/A
15.4.8	Spectrum of Rod Ejection Accidents	See Section 15.4.8	See Section 15.4.8	See Section 15.4.8
15.5	**Increase in Reactor Coolant Inventory**			
15.5.1	Chemical and Volume Control System Malfunction	102%	-43.0 pcm/°F	-2.50 pcm/°F
15.6	**Decrease in Reactor Coolant Inventory[3]**			
15.6.1	Inadvertent Operation of A Reactor Safety Valve	See Section 15.6.6	See Section 15.6.6	See Section 15.6.6
15.6.2	Failure of Small Lines Carrying Primary Coolant Outside Containment	102%	0.0 pcm/°F	-1.40 pcm/°F
15.6.3	Steam Generator Tube Failure	102%	0.0 pcm/°F	-1.40 pcm/°F
15.6.4	Main Steam Line Failure Outside Containment (BWR)	N/A	N/A	N/A
15.6.5	Loss-of-Coolant Accidents Resulting From a Spectrum of Postulated Piping Breaks Within the Reactor Coolant Pressure Boundary	102%	See Section 15.6.5	See Section 15.6.5
15.6.6	Inadvertent Operation of Emergency core Cooling System	102%	See Section 15.6.6	See Section 15.6.6
15.7	**Radioactive release from a subsystem or component**			
15.7.1	Gaseous Waste Management System Leak or Failure	N/A	N/A	N/A
15.7.2	Liquid Waste Management System Leak or Failure	N/A	N/A	N/A
15.7.3	Postulated Radioactive Releases Due to Liquid Containing Tank Failures	N/A	N/A	N/A
15.7.4	Fuel Handling Accidents	N/A	N/A	N/A
15.7.5	Spent Fuel Cask Drop Accident	N/A	N/A	N/A
15.7.6	NuScale Power Module Drop Accident	N/A	N/A	N/A

Table 15.0-8: Reactivity Coefficients (Continued)

Section	Design Basis Event	Power Level % HFP (160 MWt)	Moderator Temperature Coefficient(1)	Doppler Coefficient(1)
Special Events				
15.8	Anticipated Transient Without Scram	N/A	N/A	N/A
15.9	Stability	See Section 15.9	See Section 15.9	See Section 15.9
15.10	Core Damage Event	N/A	N/A	N/A

Note:

(1) Reactivity coefficients are often referred to as being "BOC" or "EOC" in the analyses. BOC parameters generally involve the least negative Doppler coefficient and least negative moderator temperature coefficient. EOC parameters generally involve the most negative Doppler coefficient and most negative moderator temperature coefficient. While these characterizations are generally true, the reactivity coefficients are selected to be conservative for the analysis and may not be selected strictly based on time in the cycle. The reactivity coefficients include calculational uncertainty.

(2) The reactivity insertions possible for an inadvertent decrease in Boron concentration are bounded by the reactivity insertions analyzed in Section 15.4.1 and Section 15.4.2.

(3) The iodine spike DBST, which is a surrogate for the Section 15.6 DBEs that result in primary coolant entering the containment, does not have associated reactivity coefficients.

Table 15.0-9: Assumed Single Failures and Credited Nonsafety-Related Systems

Section	Design Basis Event	Assumed Single Failure	Credited Nonsafety-Related System
15.1	**Increase in Heat Removal by the Secondary System**		
15.1.1	Decrease in Feedwater Temperature	No adverse single failures	None
15.1.2	Increase in Feedwater Flow	Failure of one FWIV to close - SG overfill analysis	FWRV
15.1.3	Increase in Steam Flow	No adverse single failures	None
15.1.4	Inadvertent Opening of Steam Generator Relief or Safety Valve	No adverse single failures	None
15.1.5	Steam System Piping Failures Inside and Outside of Containment	Failure of FWIV to close on the impacted secondary train for break inside containment. or Failure of MSIV to close on the impacted secondary train for break outside containment.	FWRV or Secondary MSIV
15.1.6	Loss of Containment Vacuum/ Containment flooding	No adverse single failures	None
15.2	**Decrease in Heat Removal by the Secondary System**		
15.2.1	Loss of External Load	Failure of one FWIV to close - secondary side pressure	FWRV
15.2.2	Turbine Trip	Failure of one FWIV to close - secondary side pressure	FWRV
15.2.3	Loss of Condenser Vacuum	Failure of one FWIV to close - secondary side pressure	FWRV
15.2.4	Closure of Main Steam Isolation Valve	No adverse single failures for primary or secondary pressure	None
15.2.5	Steam Pressure Regulator Failure (Closed)	N/A	N/A
15.2.6	Loss of Non-Emergency AC to the Station Auxiliaries	No adverse single failures for primary or secondary pressure	None
15.2.7	Loss of Normal Feedwater Flow	No adverse single failures	None
15.2.8	Feedwater System Pipe Breaks Inside and Outside Containment	Failure of FWIV check valve for isolation of the DHRS.	Nonsafety-related FW check valve
15.2.9	Inadvertent Operation of the Decay Heat Removal System	No adverse single failures for primary pressure Failure of one FWIV to close - secondary side pressure	None FWRV
15.3	**Decrease in RCS Flow Rate (not applicable)**		
15.4	**Reactivity and Power Distribution Anomalies**		
15.4.1	Uncontrolled Control Rod Assembly Withdrawal from a Subcritical or Low Power or Startup Condition	No adverse single failures	None
15.4.2	Uncontrolled Control Rod Assembly Withdrawal at Power	No adverse single failures	None

Table 15.0-9: Assumed Single Failures and Credited Nonsafety-Related Systems (Continued)

Section	Design Basis Event	Assumed Single Failure	Credited Nonsafety-Related System
15.4.3	Control Rod Misoperation (System Malfunction or Operator Error)	Failure of an ex-core flux detector with respect to power-related trips	None
15.4.4	Startup of an Inactive Loop or Recirculation Loop at an Incorrect Temperature	N/A	N/A
15.4.5	Flow Controller Malfunction Causing an Increase in Core Flow Rate (BWR)	N/A	N/A
15.4.6	Inadvertent Decrease in Boron Concentration in Reactor Coolant System	No adverse single failures	None
15.4.7	Inadvertent Loading and Operation of a Fuel Assembly in an Improper Position	No adverse single failures	None
15.4.8	Spectrum of Rod Ejection Accidents	No adverse single failures	None
15.5	**Increase in Reactor Coolant Inventory**		
15.5.2	Chemical and Volume Control System Malfunction	No adverse single failures	None
15.6	**Decrease in Reactor Coolant Inventory(1)**		
15.6.1	Inadvertent Opening of Reactor Safety Valve	See 15.6.6	See 15.6.6
15.6.2	Failure of Small Lines Carrying Primary Coolant Outside Containment	No adverse single failures	None
15.6.3	Steam Generator Tube Failure	Failure of MSIV for faulted steam generator - Radiological. No adverse single failures - thermal hydraulic acceptance criteria	Secondary MSIV None
15.6.4	Main Steam Line Failure Outside Containment (BWR)	N/A	N/A
15.6.5	Loss-of-Coolant Accidents Resulting From a Spectrum of Postulated Piping Breaks Within the Reactor Coolant Pressure	No adverse single failures	None
15.6.6	Inadvertent Operation of Emergency Core Cooling System	No adverse single failures	None
15.7	**Radioactive release from a subsystem or component**		
15.7.1	Gaseous Waste Management System Leak or Failure	N/A	N/A
15.7.2	Liquid Waste Management System Leak or Failure	N/A	N/A
15.7.3	Postulated Radioactive Releases Due to Liquid Containing Tank Failures	N/A	N/A
15.7.4	Fuel Handling Accidents	N/A	N/A
15.7.5	Spent Fuel Cask Drop Accident	N/A	N/A
15.7.6	NuScale Power Module Drop Accident	N/A	N/A

Table 15.0-9: Assumed Single Failures and Credited Nonsafety-Related Systems (Continued)

Section	Design Basis Event	Assumed Single Failure	Credited Nonsafety-Related System
Special Events			
15.8	Anticipated Transient Without Scram	N/A	N/A
15.9	Stability	No adverse single failures	None
15.10	Core Damage Event	N/A	N/A

Note:

(1) The iodine spike DBST, which is a surrogate for the Section 15.6 DBEs that result in primary coolant entering the containment, does not have assumed single failures or credited nonsafety-related systems.

Table 15.0-10: Referenced Topical and Technical Reports

Topical or Technical Report	Report	Rev No	Description	NRC SER Reference
NuScale LOCA Evaluation Model (Topical)	TR-0516-49422	0	Summarizes the NuScale LOCA Phenomena Identification and Ranking Table (PIRT), the NRELAP5 code features and modifications, assessments of NRELAP5 against Separate Effects Tests (SETs) and Integral Effects Tests (IETs), and the applicability evaluation of NRELAP5 to NuScale LOCA analysis. Provides sample cases to demonstrate application of the evaluation model (EM) to the NPM design. The transition between the initial LOCA event and long term cooling is defined in this document	Not issued
Long Term Cooling Methodology (Technical)	TR-0916-51299	1	This report summarizes the long term NuScale design basis event progression following ECCS actuation, the regulatory requirements and NuScale-specific design requirements applicable to long term core cooling (LTCC), the LTCC acceptance criteria, the NuScale LTCC PIRT, the analysis tools, qualification of the tools, and methodology for demonstrating that the LTCC acceptance criteria are met, and the results of LTCC analyses.	Not issued
Evaluation Methodology for Stability Analysis of NuScale Power Module (Topical)	TR-0516-49417	1	Presents a methodology for addressing thermal-hydraulic stability in the NPM. The basis of the NPM stability study is a detailed phenomenological review. Provides generic representations of anticipated transients where unstable oscillations may occur. Identifies the limiting instability mode as natural circulation instability. The adiabatic riser response dominates the response rather than wave propagation in the core. The dynamics of the SG and the fission power response to reactivity feedback influence stability. Describes the computational method for the analysis of the postulated instability modes of the NPM during steady state normal operation and anticipated transients. Identifies that potential instability from loss of subcooling in the riser is excluded by MPS protective actions.	Not issued

Table 15.0-10: Referenced Topical and Technical Reports (Continued)

Topical or Technical Report	Report	Rev No	Description	NRC SER Reference
Accident Source Term Methodology (Topical)	TR-0915-17565	3	Describes assumptions, codes, and methodologies used to calculate the radiological consequences of design basis accidents, including the iodine spike design-basis source term, and the beyond-design basis core damage event. Describes the methodologies associated with performing the beyond-design-basis core damage source term radiological analysis and associated aerosol transport and iodine re-evolution assessment methodologies. Describes the STARNAUA aerosol modeling to the range of post-accident containment conditions and justifies the assumption that no elemental iodine decontamination factor limit should be applied to natural aerosol removal phenomenon in the NuScale containment. Describes the use of ARCON96 for establishing offsite atmospheric dispersion.	Not issued
Subchannel Analysis Methodology (Topical)	TR-0915-17564-P-A	2	Discusses how NuScale Power, LLC, meets the NRC requirements for use of VIPRE-01 Safety Evaluation Reports (SERs), the modelling methodology for performing steady state and transient subchannel analyses, and the qualification of the code for application to the NuScale Power Plant design. Explains why the methodology is independent of any one CHF correlation and may be used for NuScale applications if methodology requirements are satisfied. Describes methodology for treatment of uncertainties in the NuScale subchannel methodology.	ML18338A030
NuScale Power Critical Heat Flux Correlations (Topical)	TR-0116-21012-P-A	1	Provides the bases for use of critical heat flux (CHF) correlations in VIPRE-01within its range of applicability, along with its associated correlation limit, for the NuScale Power, LLC, Design Certification Application and for the safety analysis of the NPM with NuFuel-HTP2™ fuel. The report describes the tests, test facilities, statistical methods, base CHF correlation development, NSPX factor development, and final validation for the development of the CHF correlation.	ML18214A478

Table 15.0-10: Referenced Topical and Technical Reports (Continued)

Topical or Technical Report	Report	Rev No	Description	NRC SER Reference
NuScale Rod Ejection Accident Methodology	TR-0716-50350	0	Describes the codes and methodology used to analyze the rod ejection accident (REA). Describes the three-dimensional behavior using SIMULATE5 and SIMULATE-3K, the reactor system response using NRELAP5, and the subchannel thermal-hydraulic behavior and fuel response using VIPRE-01.	Not Issued
Non-LOCA Transient Analysis Methodology	TR-0516-49416	1	Describes evaluation model that simulates the NPM transient response to non-LOCA events. Addresses the EMDAP process used to establish the adequacy of the non-LOCA methodology. Uses a graded approach to the EMDAP for development of the non-LOCA system transient evaluation model considering the overlap in high-ranked phenomena and conservatism applied to input and boundary conditions of the LOCA EM in the non-LOCA plant transient calculations. Describes the non-LOCA PIRT assessment of the relative importance of phenomena and processes that may occur in the NuScale Power Module during non-LOCA events in relation to specified figures of merit. Describes the requirements for evaluation model capability developed from the non-LOCAPIRT. Explains how NRELAP5 assessments performed for LOCA EM development demonstrate NRELAP5 qualification for high rank/low knowledge-level non-LOCA PIRT phenomena: 1. Describes the separate effects testing of the full-length DHRS at the NIST facility to address DHRS heat transfer. 2. Presents the NRELAP5 assessments against the SIET TF-1 and TF-2 data to validate adequacy of SG heat transfer from the DHRS loop during non-LOCA transients. 3. Describes the integral effects test of the DHRS operation at the NIST facility. 4. Provides a code-to-code benchmark assessing the NRELAP5 prediction of the NPM response to reactivity insertion events using RETRAN-3D.	Not issued

Table 15.0-11: Summary of Applicable Radiological Events to the NuScale Design

Event	Dose Consequence Analysis Section	Thermal Hydraulic Analysis Section	Regulatory Guide 1.183 Appendix (1)	Primary Source of Radiation
Failure of Small Lines Carrying Primary Coolant Outside Containment	15.0.3.8.1	15.6.2	N/A	Coolant activity (with pre-existing and coincident iodine spiking)
Steam Generator Tube Failure	15.0.3.8.2	15.6.3	F	Coolant activity (with pre-existing and coincident iodine spiking)
Main Steam Line Break	15.0.3.8.3	15.1.5	E	Coolant activity (with pre-existing and coincident iodine spiking)
Rod Ejection Accident	15.0.3.8.4	15.4.8	H	No fuel damage, radiological consequences bounded by other events
Fuel Handling Accident	15.0.3.8.5	N/A	B	Damaged fuel
Iodine Spike Design-Basis Source Term(3)	15.0.3.8.6	N/A	N/A	Coolant activity (with pre-existing and coincident iodine spiking)
Core Damage Event(2)	15.10.2	N/A	N/A	Damaged fuel

Notes:

(1) Appendices C, D, and G were not included because they are not applicable to the NuScale design.
(2) The CDE is a beyond-design-basis special event.
(3) The iodine spike DBST is not an event, rather it serves as a bounding surrogate for design-basis loss of primary coolant into containment events described in Section 15.6.

Table 15.0-12: Radiological Dose Consequences for Chapter 15 Analyses

Event	Location	Acceptance Criteria (rem TEDE)	Dose (rem TEDE)
Failure of Small Lines Carrying Primary Coolant Outside Containment	EAB	2.5	0.02
	LPZ	2.5	0.04
	CR	5.0	0.09
Steam Generator Tube Failure (pre-incident iodine spike)	EAB	25.0	0.07
	LPZ	25.0	0.08
	CR	5.0	0.13
Steam Generator Tube Failure (coincident iodine spike)	EAB	2.5	<0.01
	LPZ	2.5	0.01
	CR	5.0	0.03
Main Steam Line Break (pre-incident iodine spike)	EAB	25.0	<0.01
	LPZ	25.0	<0.01
	CR	5.0	0.01
Main Steam Line Break (coincident iodine spike)	EAB	2.5	0.01
	LPZ	2.5	0.06
	CR	5.0	0.13
Fuel Handling Accident	EAB	6.3	0.55
	LPZ	6.3	0.55
	CR	5.0	0.89
Iodine Spike Design Basis Source Term(1) (pre-incident iodine spike)	EAB	25.0	<0.01
	LPZ	25.0	<0.01
	CR	5.0	<0.01
Iodine Spike Design-Basis Source Term(1) (coincident iodine spike)	EAB	25.0	<0.01
	LPZ	25.0	<0.01
	CR	5.0	0.02
Core Damage Event(2)	EAB	25.0	0.63
	LPZ	25.0	1.37
	CR	5.0	2.14

Notes:

(1) The iodine spike DBST is not an event, rather it serves as a bounding surrogate for design-basis loss of primary coolant into containment events described in Section 15.6.

(2) The CDE is a beyond-design-basis special event.

Table 15.0-13: Onsite and Offsite Receptor Variables

	Control Room	Technical Support Center	Exclusion Area Boundary	Low Population Zone
Atmospheric dispersion				
0-8 hours	Table 2.0-1	Table 2.0-1	Table 2.0-1	Table 2.0-1
8-24 hours	Table 2.0-1	Table 2.0-1	Table 2.0-1	Table 2.0-1
24-720 hours	Table 2.0-1	Table 2.0-1	Table 2.0-1	Table 2.0-1
Breathing Rates (sec/m^3)				
0-8 hours	3.50E-04	3.50E-04	3.50E-04	3.50E-04
8-24 hours	3.50E-04	3.50E-04	1.80E-04	1.80E-04
24-720 hours	3.50E-04	3.50E-04	2.30E-04	2.30E-04
Occupancy factor %				
0-8 hours	100	100	N/A	N/A
8-24 hours	60	60	N/A	N/A
24-720 hours	40	40	N/A	N/A

Table 15.0-14: Primary Coolant Source Term

Nuclide	Primary Activity (Ci)
I-131	1.322E+00
I-132	6.031E-01
I-133	1.992E+00
I-134	3.545E-01
I-135	1.254E+00
Kr-85m	9.660E-01
Kr-85	2.865E+02
Kr-87	5.275E-01
Kr-88	1.536E+00
Xe-133	2.536E+02
Xe-135	8.569E+00

Table 15.0-15: Control Room Parameters

Parameter	Units	Value
Control Room Envelope Volume	ft^3	74,680
Control Room Emergency Duration	hr	72
Control Room Emergency Flow Rate	cfm	100
Control Room Normal Flow Rate	cfm	742
Control Room Recirculation Flow Rate	cfm	7011
Control Room Unfiltered Ingress/Egress	cfm	5
Control Room Unfiltered Inleakage	cfm	147

Table 15.0-16: Return to Power Calculation Nominal Results Summary

Cooldown Mode	Pool Temperature (°F)	Equilibrium Power (%RTP)	MCHFR
DHRS Uncovered Riser	65.0	Subcritical	N/A
DHRS Covered Riser	65.0	0.49	29
DHRS Covered Riser	100.0	0.43	33
DHRS Covered Riser	140.0	Subcritical	N/A
ECCS Actuated	65.0	1.01	>8
ECCS Actuated	100.0	0.65	13
ECCS Actuated	140.0	Subcritical	N/A

Table 15.0-17: Return to Power Calculation Limiting Results

Equilibrium Power (%RTP)	Peak Heat Flux (kW/m^2)	MCHFR
1.84	164	>4

Table 15.0-18: Technical Support Center Parameters

Parameter	Units	Value
TSC normal flow rate	cfm	659
TSC recirculation flow rate	cfm	11,891
TSC unfiltered ingress/egress	cfm	10
TSC unfiltered inleakage	cfm	56
TSC volume	ft^3	85,614

Table 15.0-19: 10x Limiting Accident Radiation Post-Filtration Monitor Analytical Limits

Principal Radionuclides Measured	Value	Unit
Noble gas (Kr-85)	5.11E-01	µCi/cc
Noble gas (Xe-133)	6.13E+00	µCi/cc
Particulate (Cs-137)	6.24E-06	µCi/cc
Iodine (I-131)	9.10E-04	µCi/cc

Table 15.0-20: Assumptions for Accident Airborne Effluent Release Point Characteristics for Offsite Receptors

Parameter	Value
Release location	Any point on Reactor Building or Turbine Generator Building wall
Release height	Ground level (0.0 meters)
Intake height	0.0 meters
Adjacent building cross-sectional area	Negligible (0.01 square meters)

Table 15.0-21: Assumptions for Control Room χ/Q

Parameter	Value
Release height	Ground level (0.0 meters)
Intake height	0.0 meters
Adjacent building cross-sectional area	Negligible (0.01 square meters)
Distance from source to receptor	34.1 meters

Table 15.0-22: Results of Long Term Cooling Analysis

Case Description	Core Inlet Temperature (°F)		Collapsed Riser Level (ft)		Boron Precipitation Margin (°F)	
	Transient at 12.5 Hours	State-point at 72 Hours	Transient at 12.5 Hours	State-point at 72 Hours	Transient at 12.5 Hours	State-point at 72 Hours
Maximum Temperature Injection Line (IL) Break	292.8	270.4	8.9	9.1	208.9	187.8
Minimum Temperature IL Break	152.8	140.4	10.0	10.4	73.1	62.3
Minimum Level IL Break	165.3	154.5	7.3(1)	8.0	76.2	69.0
Maximum Temperature IL Break, 45 ft reactor pool level(2)	-	280.3	-	9.2	-	197.6
Minimum Temperature IL Break, 13% initial power(2)	-	94.3	-	10.4	-	16.6
Minimum Temperature SGTF, 13% initial power(2)	-	112.1	-	10.1	-	33.3
Minimum Temperature DHRS cooldown, 13% initial power(2)	-	116.8	-	10.4	-	39.2

Notes:

(1) Minimum collapsed riser level was 2.8 feet and occurred approximately 3.6 hours after ECCS actuation.

(2) A 12 hour transient simulation for these cases was not performed. Limiting conditions are only important at the end of the LTC phase at 72 hours.

Figure 15.0-1: Control Rod Position vs. Time

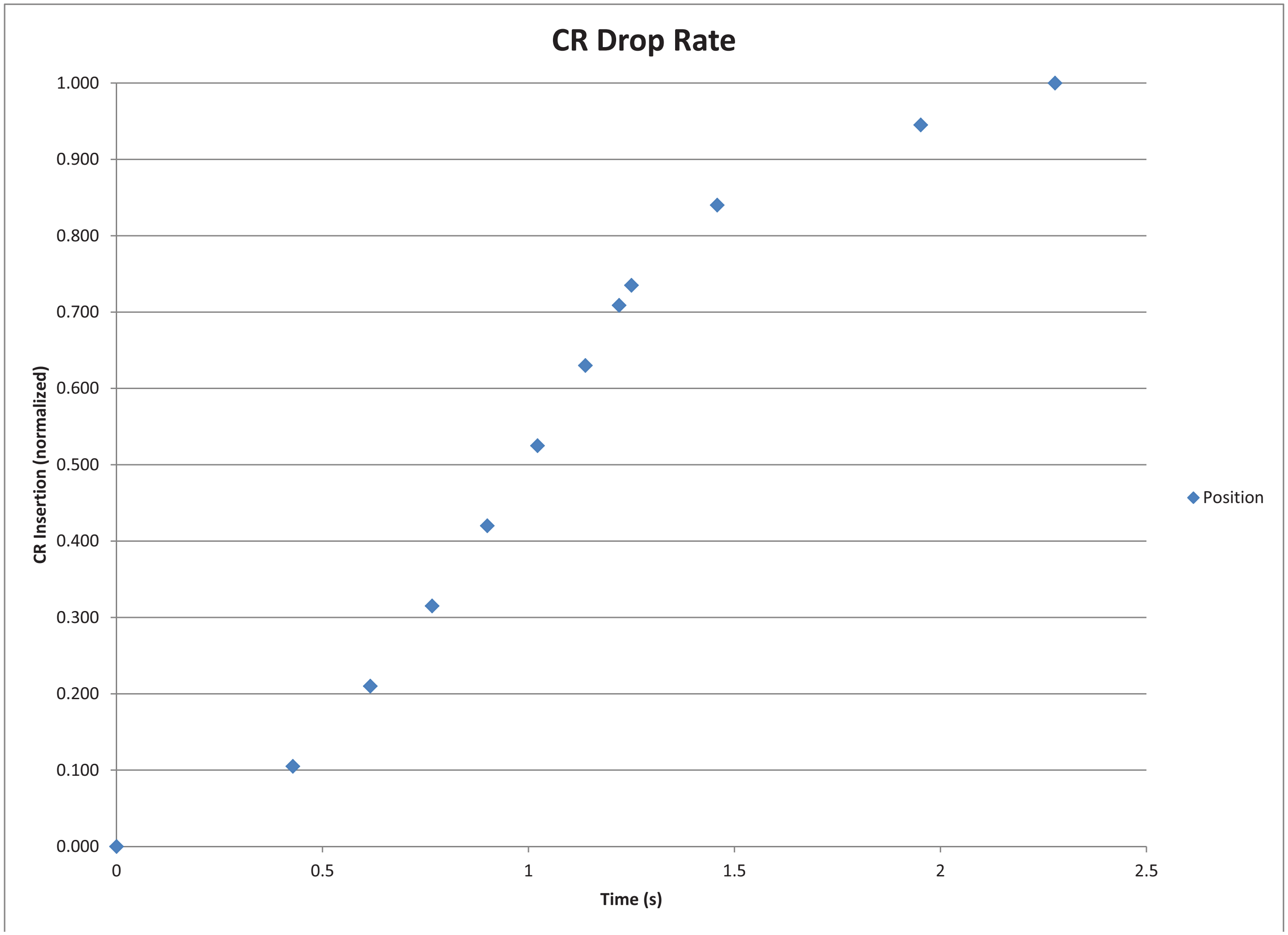


Figure 15.0-2: SCRAM Reactivity Worth vs. Time

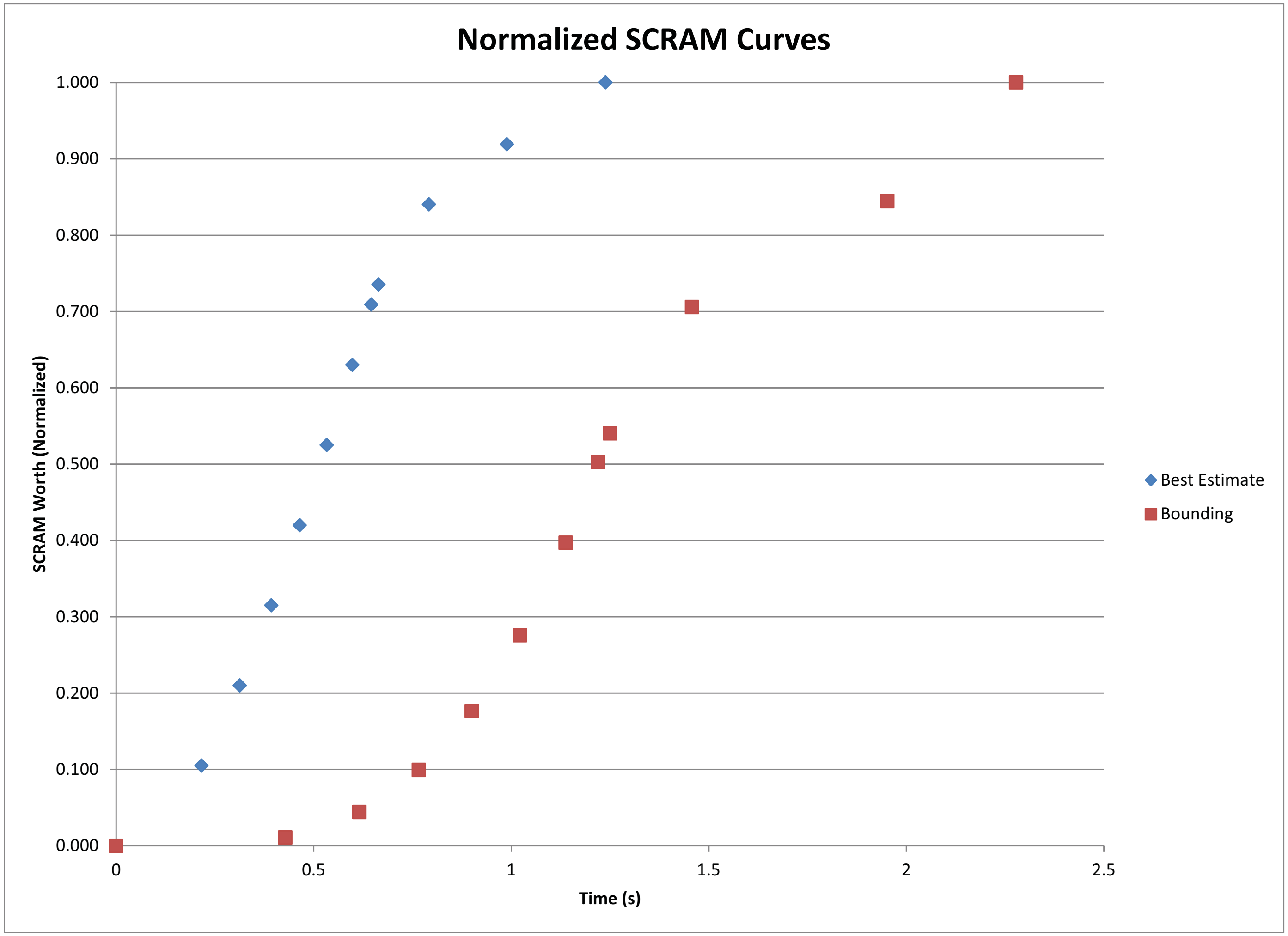


Figure 15.0-3: Schematic of the RADTRAD Model Nodalization

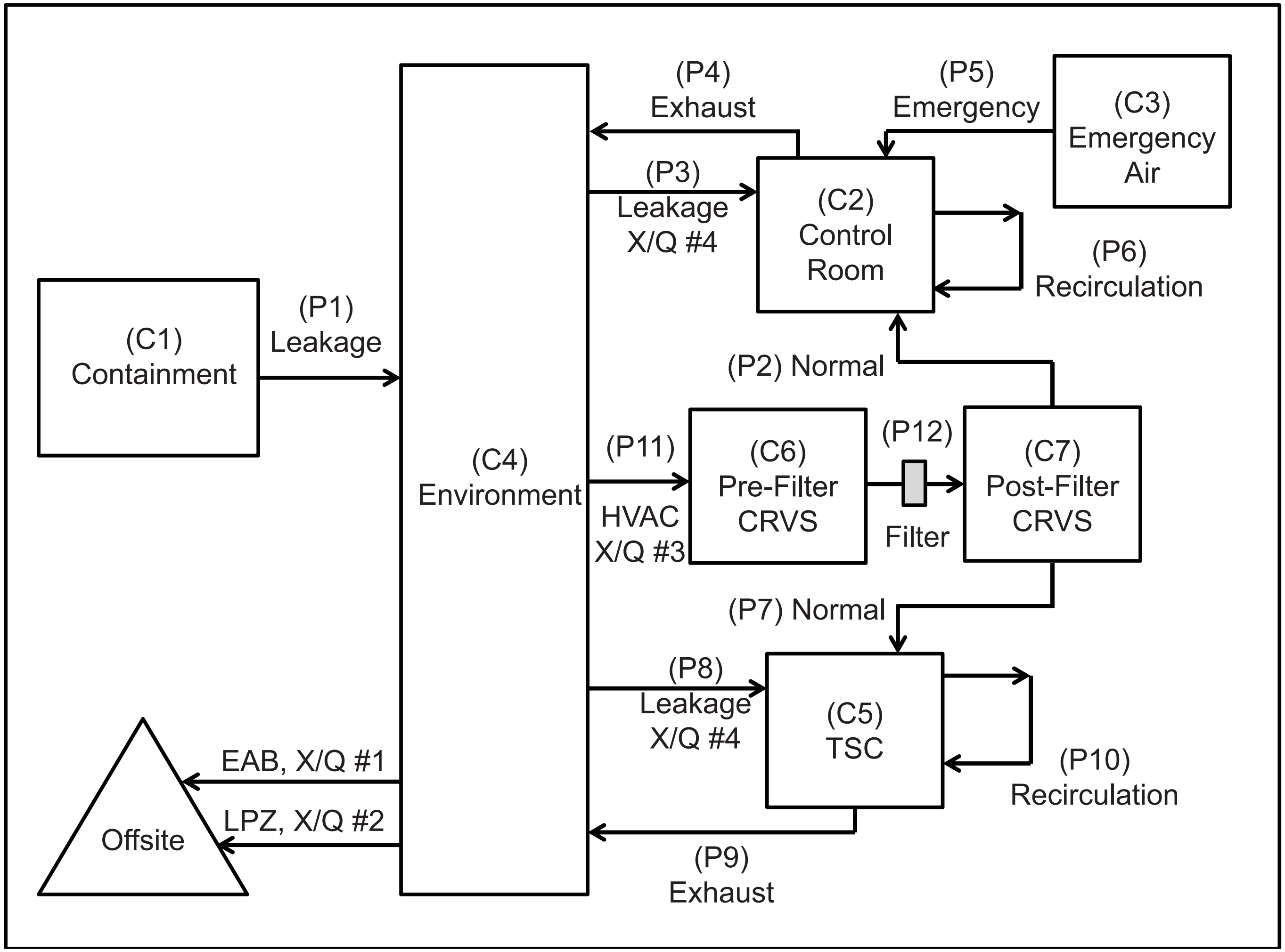


Figure 15.0-4: Schematic of the RADTRAD Model Nodalization for Steam Generator Tube Failure and Main Steam Line Break

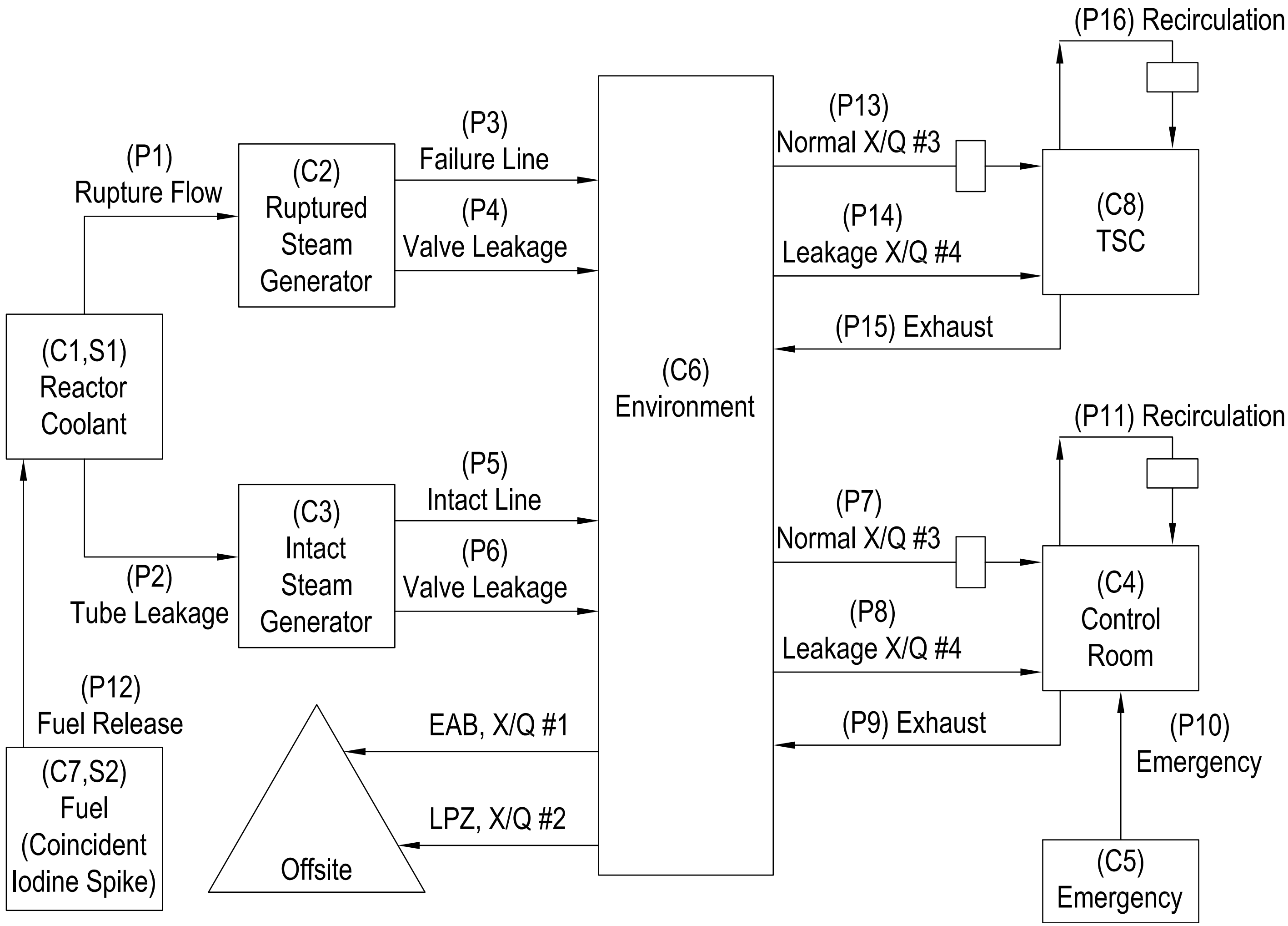


Figure 15.0-5: Not Used

Figure 15.0-6: Not Used

Figure 15.0-7: Not Used

Figure 15.0-8: Nominal Return to Power Results - 65 Degree F Pool Temperature (15.0.6)

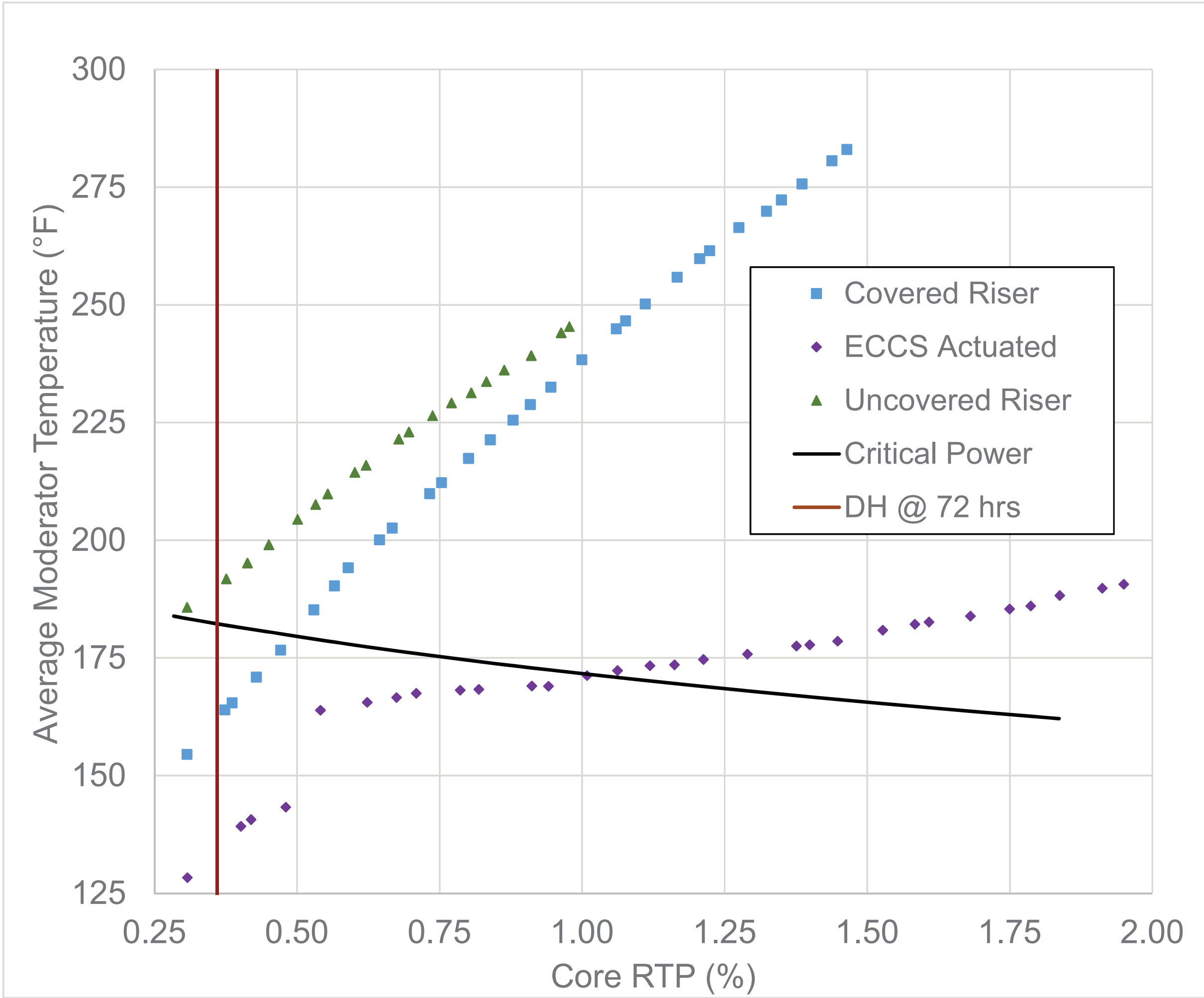


Figure 15.0-9: Analytical Operating Limits

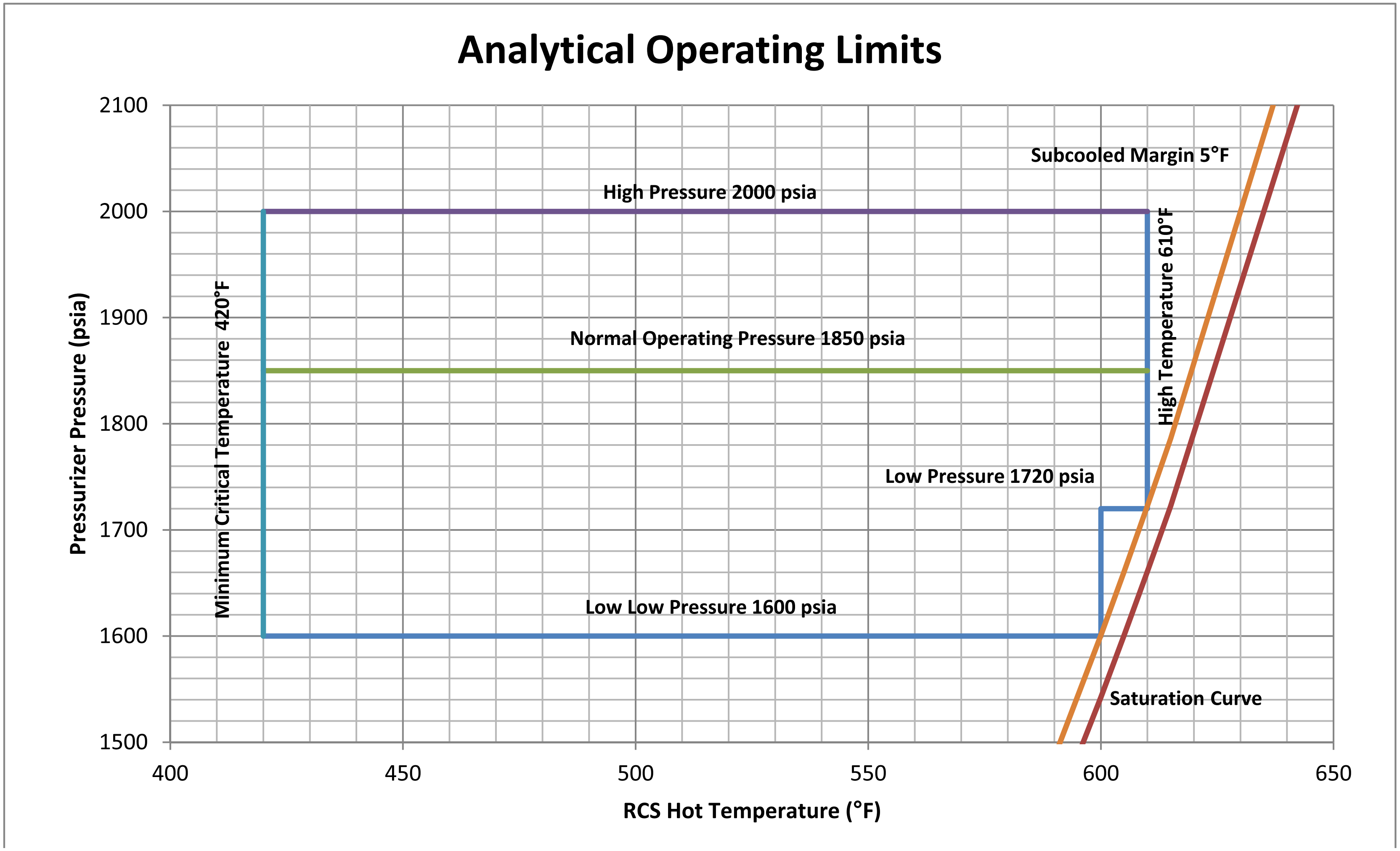


Figure 15.0-10: Nominal Return to Power Results -140 Degree F Pool Temperature (15.0.6)

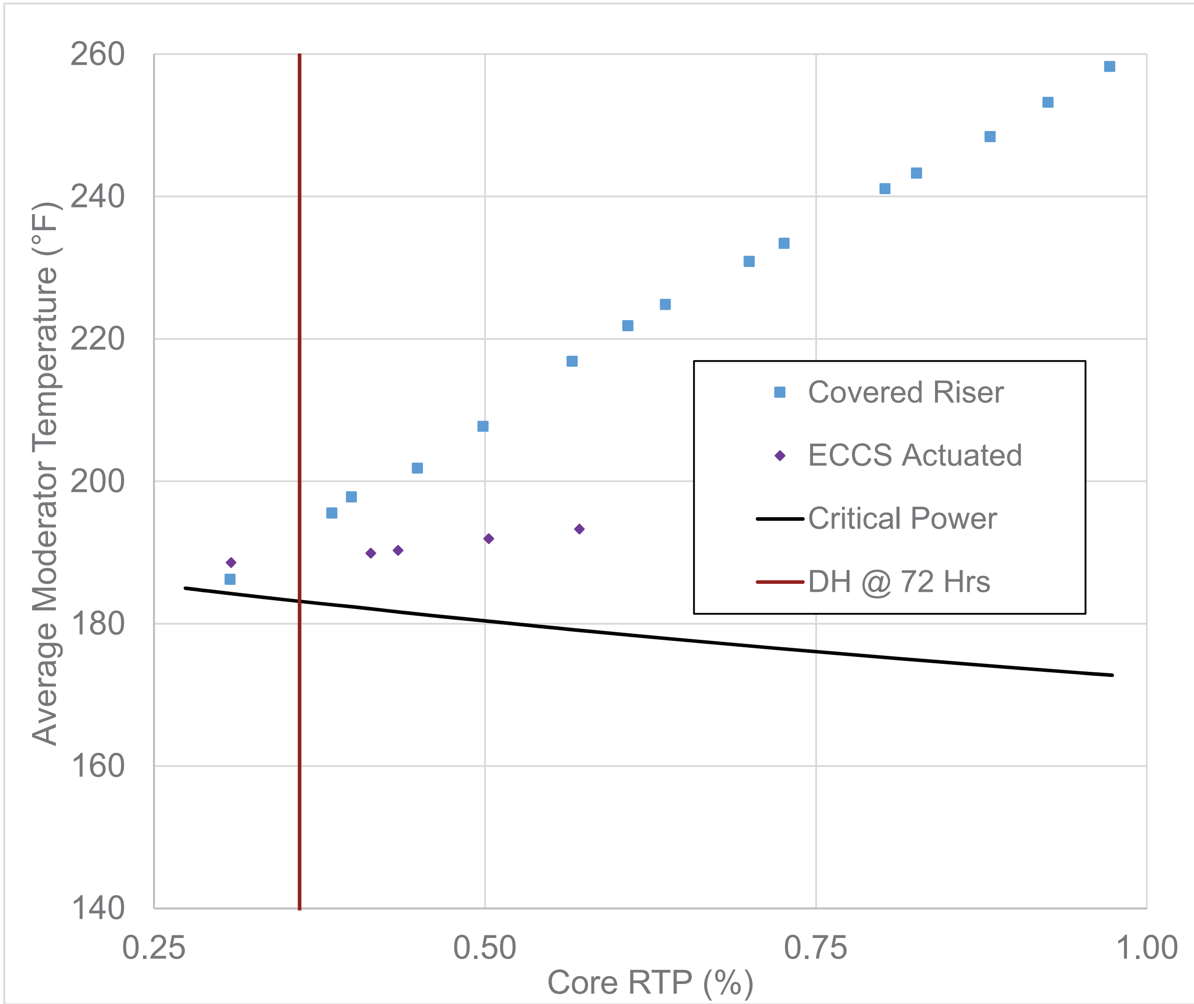


Figure 15.0-11: Not Used

Figure 15.0-12: Not Used

Figure 15.0-13: Not Used

Figure 15.0-14: Not Used

Figure 15.0-15: Not Used

Figure 15.0-16: Not Used

Figure 15.0-17: Not Used

Figure 15.0-18: Not Used

Figure 15.0-19: Not Used

15.1 Increase in Heat Removal by the Secondary System

There are several events that could result in an increase in primary heat removal by the secondary system. There is also a NuScale design-specific event that causes an overcooling of the primary system due to a loss of the containment vacuum or flooding of containment. These events are classified as anticipated operational occurrences (AOOs), infrequent events, or accidents as shown in Table 15.0-1. An increase in primary heat removal results in an increase in core reactivity which leads to an increase in power. The increase in reactor power leads to a decrease in the minimum critical heat flux ratio (MCHFR). None of the increase in heat removal events present a significant challenge to primary or secondary pressure limits. Sensitivities were not run for these events to maximize primary or secondary pressure. Therefore, figures of merit are only provided for the limiting MCHFR case for this class of events. The maximum RCS and SG pressures from the cases analyzed are provided in the results summary table for each section. The transient response of the plant to these events is described in the following sections:

- Section 15.1.1 - Decrease in Feedwater Temperature
- Section 15.1.2 - Increase in Feedwater Flow
- Section 15.1.3 - Increase in Steam Flow
- Section 15.1.4 - Inadvertent Opening of Steam Generator Relief or Safety Valve
- Section 15.1.5 - Steam Piping Failures Inside and Outside of Containment
- Section 15.1.6 - Loss of Containment Vacuum/Containment Flooding

15.1.1 Decrease in Feedwater Temperature

15.1.1.1 Identification of Causes and Accident Description

A decrease in feedwater temperature could be caused by a failure in the feedwater system. A lower feedwater temperature would increase the heat removal from the primary system, leading to a higher moderator density. As the reactor coolant system (RCS) cools, if the reactivity control system is in an automatic mode, it inserts positive reactivity by pulling the regulating control rods from the core in an attempt to maintain RCS temperature. The increase in reactor power due to the insertion of positive reactivity results in an increase in core power, and a decrease in the minimum critical heat flux ratio (MCHFR).

A decrease in feedwater temperature is expected to occur one or more times in the life of the reactor, so it is classified as an AOO. The categorization of the NuScale Power Plant design basis events (DBEs) is discussed in Section 15.0.

15.1.1.2 Sequence of Events and Systems Operation

The sequence of events for the limiting decrease in feedwater temperature case is provided in Table 15.1-1.

Unless specified below, the analysis of a decrease in feedwater temperature assumes the plant control systems (PCSs) and engineered safety features (ESFs) perform as

designed, with allowances for instrument inaccuracy. No operator action is credited to mitigate the effects of a decrease in feedwater temperature event.

Normally, as the colder feedwater enters the steam generator (SG), the feedwater controls would reduce feedwater flow, helping to mitigate the overcooling event. This event is conservatively analyzed with the feedwater controls disabled, forcing a constant feedwater flow rate, and is not credited to mitigate the event. Similarly, the turbine throttle and stop valves are not credited, allowing pressurization of the steam line.

The reactivity control system is assumed to be in normal automatic mode. Operator action is not credited for regulating control rod movement or increasing boron concentration. This ensures that the maximum reactivity insertion is reached as the control system attempts to maintain RCS temperature by pulling the regulating control rods from the core.

The module protection system (MPS) is credited to protect the plant in the event of a decrease in feedwater temperature. If the feedwater temperature were to drop to a level that causes a high enough power excursion, the MPS high power signal would trip the reactor, preventing the reactor from reaching a power level where the acceptance criteria could be challenged. The following MPS signals provide the plant with protection during a decrease in feedwater temperature:

- high core power (5 percent uncertainty added)
- high steam line pressure
- high hot leg temperature
- high steam superheat

Due to the cooling of the RCS during a decrease in feedwater temperature event, the coolant in the downcomer increases in density. This increase in density can affect the power level detection by the excore neutron detectors. In order to account for this effect, the high core power rate trip is not credited in the analyses, and a 5 percent uncertainty is added to the high core power trip.

In a decrease in feedwater temperature event that results in a reactor trip, the subsequent actuation secondary system isolation (SSI) and the decay heat removal system (DHRS) are credited with isolating the main steam and feedwater systems and maintaining reactor cooling.

There are no single failures that could make the limiting decrease in feedwater temperature MCHFR case more severe. A single failure of one of the main steam isolation valves (MSIVs) is considered. If the non-safety related turbine trip is not credited with mitigating this single failure, the closure of the backup main steam isolation valve at the same time as the MSIV is credited. Regardless, this single failure does not affect the MCHFR because the reactor has already tripped when the failure occurs. Therefore, this single failure is not modeled in the case presented in this section, which demonstrates the limiting decrease in feedwater temperature with respect to MCHFR.

Normal alternating current (AC) power is assumed to be available for this event. A loss of AC power is not a conservative condition for a decrease in feedwater temperature event. The loss of normal power scenarios are listed below:

- Loss of normal AC – In this scenario, MPS remains powered so none of the safety systems are automatically actuated, but feedwater is lost and the turbine is tripped.
  - Loss of normal AC at the time of the decrease in feedwater temperature is non-limiting because feedwater is lost which reduces the overcooling event.
  - Loss of normal AC at the time of reactor trip is non-limiting because feedwater is lost which reduces the overcooling event.
- Loss of the normal DC power system (EDNS) and normal AC – Power to the reactor trip breakers is provided via the EDNS, so this scenario is the same as discussed above with addition of reactor trip at the time at which power is lost. For the decrease in feedwater temperature events, this scenario is non-limiting for the reasons listed above and from the immediate reactor trip.
- Loss of the highly reliable DC power system (EDSS), EDNS and normal AC – Power to the MPS is provided via the EDSS so this scenario results in an actuation of reactor trip and all of the ESFs. In terms of the overcooling event, this scenario is non-limiting for the reasons discussed above.

15.1.1.3 Thermal Hydraulic and Subchannel Analyses

15.1.1.3.1 Evaluation Models

The thermal hydraulic analysis of the plant response to a decrease in feedwater temperature is performed using NRELAP5. The NRELAP5 model is based on the design features of a NuScale Power Module (NPM). The non-loss-of-coolant accident (non-LOCA) NRELAP5 model is discussed in Section 15.0.2. The relevant boundary conditions from the NRELAP5 analyses are provided to the subsequent subchannel critical heat flux (CHF) analysis.

The subchannel core CHF analysis is performed using VIPRE-01. VIPRE-01 is a subchannel analysis tool designed for general-purpose thermal-hydraulic analysis under normal operating conditions, operational transients, and events of moderate severity. See Section 15.0.2. for a discussion of the VIPRE-01 code and evaluation model.

15.1.1.3.2 Input Parameters and Initial Conditions

The lowest feedwater temperature that could result from a failure in the feedwater system (FWS) is 100 degrees F. This is the temperature of the feedwater before it passes through the feedwater heaters. Sensitivities on a spectrum of feedwater temperature reductions reveal that the limiting case with respect to MCHFR is a reduction to 100 degrees F in 86 seconds.

The initial conditions used in the evaluation of the limiting decrease in feedwater temperature event result in a conservative calculation. Table 15.1-2 provides key inputs for the limiting decrease in feedwater temperature case. The following initial

conditions are assumed in the analysis to ensure that the results have sufficient conservatism.

- Initial power level is assumed to be 102 percent of nominal to account for a 2 percent instrumentation uncertainty.
- Conservative scram characteristics are used, including a maximum time delay, holding the most reactive rod out of the core, and utilizing a bounding control rod drop rate.
- The most limiting end-of-cycle core parameters are used to provide a limiting power response. The most negative moderator temperature coefficient (MTC) of -43.0 pcm/degrees F and the most negative Doppler temperature coefficient (DTC) of -2.5 pcm/degrees F are used to provide the largest power response for this event.
- Allowances for instrument inaccuracy are accounted for in the analytical limits of mitigating systems in accordance with the guidance provided in Regulatory Guide (RG) 1.105.

The results from the thermal hydraulic evaluation are used as input to the subchannel analysis to determine the MCHFR for this event. The subchannel evaluation model is discussed in Section 15.0.2.3.

15.1.1.3.3 Results

The sequence of events for the limiting decrease in feedwater temperature event is provided in Table 15.1-1. Figure 15.1-1 through Figure 15.1-10 and Figure 15.1-54 through Figure 15.1-55 show the transient behavior of key parameters for the limiting decrease in feedwater temperature event. A spectrum of feedwater temperature cases were analyzed, and the limiting temperature decrease is a case in which the cooldown rate yields nearly simultaneous reactor trips on high power and high hot leg temperature. The limiting event initiates with a linear decrease in feedwater temperature to the minimum possible temperature of 100 degrees F over 86 seconds. The RCS response to the overcooling event begins once the cold feedwater front propagates through the secondary system piping and reaches the SG. This decrease in feedwater temperature leads to an increase in the heat removal rate from the RCS via the SG. During the over-cooling phase of the transient, the RCS temperature steadily decreases, while the RCS density increases. The colder, denser RCS causes the regulating control rod bank to withdraw in an attempt to maintain RCS temperature. The withdrawal of the control rods causes a positive reactivity insertion that increases reactor power. If the regulating control rod bank were disabled, a similar power response would be driven by moderator feedback instead.

The decrease in reactor coolant temperature causes the primary coolant volume to shrink, which initially reduces the pressure of the RCS. However, as core power increases, the RCS pressure begins to rise. The hot leg temperature rises in response to the increase in reactor power. The high hot leg temperature reactor limit is reached at about 184 seconds into the transient. During the assumed reactor trip signal delay, the rise in reactor power initiates a high power reactor trip at approximately 187 seconds into the transient. The peak RCS pressure occurs just

prior to the reactor trip. The high hot leg temperature trip also actuates the DHRS valves to open. The feedwater isolation valves (FWIVs) and MSIVs close, isolating the SG from the rest of the secondary system.

Steam generator pressure does not change significantly during the initial phase of the transient. However, after DHRS and SSI actuation at 192 seconds, the closure of the FWIVs and MSIVs causes pressurization of the SG. Steam generator pressure increase resulting from main steam isolation is expected and is not a direct consequence of the decrease in feedwater temperature event itself. The maximum secondary pressure is reached after main steam and feedwater isolation.

The CHFR decreases as reactor power and RCS pressure increase. The automatic protection systems terminate this transient before the CHFR reaches the design limit. The MCHFR for the limiting decrease in feedwater temperature does not violate the CHF limit.

During the overcooling phase, RCS flow steadily increases in response to rising reactor power. The reactor scram causes a rapid decrease in flow as the heat source driving natural circulation is reduced. The flow oscillates until RCS temperatures re-equilibrate. At approximately 1200 seconds, RCS flow stabilizes and passive DHRS cooling dominates (Figure 15.1-10). The reactor trip, subsequent actuation of DHRS, and stabilization of RCS flow demonstrate the plant response to a decrease in feedwater temperature, and a return to a stable condition with no operator actions. For a discussion on possible return to power scenarios, see Section 15.0.6.

15.1.1.4 Radiological Consequences

The normal leakage related radiological consequences of this event are bounded by the design basis accident analyses presented in Section 15.0.3.

15.1.1.5 Conclusions

The five Design Specific Review Standard (DSRS) acceptance criteria for this AOO are met for the limiting decrease in feedwater temperature case. These acceptance criteria, followed by how the NuScale Power Plant design meets them are listed below:

1) Pressure in the reactor coolant and main steam systems should be maintained below 110 percent of the design values.

    - The pressure responses in the reactor pressure vessel (RPV) and in the main steam system (MSS) are less severe than those of the AOOs presented in Section 15.2, decrease in heat removal by the secondary side. Therefore, this acceptance criterion is met for the decrease in feedwater temperature event. The maximum pressure values for the cases analyzed are shown in Table 15.1-3

2) Fuel cladding integrity shall be maintained by ensuring that the minimum departure from nucleate boiling ratio (DNBR) remains above the 95/95 DNBR limit based on acceptable correlations (see DSRS Section 4.4).

    - The MCHFR for this event is above the 95/95 limit as shown in Table 15.1-3. Therefore this acceptance criterion is met.

3) An incident of moderate frequency should not generate a more serious plant condition without other faults occurring independently.

   - The analysis presented for this event shows that stable DHRS cooling is reached, and the acceptance criterion is met.

4) The guidance provided in RG 1.105, "Instrument Spans and Setpoints," can be used to analyze the impact of instrument spans and setpoints on the plant response to the type of transient addressed in this DSRS section, in order to meet the requirements of General Design Criteria 10, 13, 15, 20, and 26.

   - The instrument spans and setpoints discussed in Section 15.1.1.3.2, address the guidance in RG 1.105.

5) The most limiting plant systems single failure, as defined in the "Definitions and Explanations" of Appendix A to 10 CFR Part 50, shall be identified and assumed in the analysis and shall satisfy the positions of RG 1.53.

   - There is no limiting single failure that could occur during a decrease in feedwater temperature event that could result in more severe conditions with respect to the acceptance criteria.

15.1.2 Increase in Feedwater Flow

15.1.2.1 Identification of Causes and Accident Description

An increase in feedwater flow event could occur due to a failure in the feedwater system. For the NuScale Power Plant design, the increased feedwater flow increases heat transfer from the primary side, which in turn, results in a power increase due to the moderator density increase. As the RCS cools, if the reactivity control system is in an automatic mode, it will insert positive reactivity by pulling the regulating control rods from the core in an attempt to maintain RCS temperature. The increase in reactor power due to the insertion of positive reactivity result in an increase in reactor power, and a decrease in the MCHFR.

An increase in feedwater flow is expected to occur one or more times in the life of the reactor, so it is classified as an AOO. The categorization of the NuScale DBEs are discussed in Section 15.0.

15.1.2.2 Sequence of Events and Systems Operation

The sequence of events for an increase in feedwater flow event is provided in Table 15.1-4.

Unless specified below, the analysis of an increase in feedwater flow event assumes the PCSs and ESFs perform as designed, with allowances for instrument inaccuracy. No operator action is credited to mitigate the effects of an increase in feedwater event.

The FWS could malfunction and increase feedwater flow by increasing the speed of two normally operating feedwater pumps, turning on a feedwater pump, opening a feedwater regulator valve, or opening a DHRS valve at low RCS power. The inadvertent

opening of a DHRS valve at low RCS power is addressed by the analysis in Section 15.2.9. The feedwater system includes two normally operating pumps with a third for backup. If the backup pump were to spuriously turn on at its maximum flow rate, the total increase in feedwater flow is bounded by assuming a malfunction of the two operating feedwater pumps increasing to maximum speed. In order to conservatively bound all possible feedwater flow increase scenarios, a spectrum of feedwater flow increases are analyzed to demonstrate that limiting conditions for MCHFR are reached. For the limiting MCHFR case, the steam outlet is modeled as a constant steam pressure, allowing the steam flow to increase providing an increase in steam generator heat transfer in response to the increase in feedwater flow.

Operator action is not credited for regulating control rod movement or increasing boron concentration. This ensures that the maximum reactivity insertion is reached as the control system attempts to maintain RCS temperature by pulling the regulating control rods from the core.

The MPS is credited to protect the plant in the event of an increase in feedwater flow. If the feedwater flow increases to a level that causes a high enough power excursion, the MPS high power signal trips the reactor, preventing the reactor from reaching a power level where the acceptance criteria could be challenged. The following MPS signals provide the plant with protection during an increase in feedwater flow:

- high core power (5 percent uncertainty added)
- high core power rate (not credited in the safety analysis of this event)
- high steam superheat
- high steam pressure
- low steam superheat
- high RCS temperature

Due to the cooling of the RCS during an increase in feedwater flow event, the coolant in the downcomer increases in density. This increase in density can affect the power level detection by the excore neutron detectors. In order to account for this effect, the high core power rate trip is not credited in the analyses, and a 5 percent uncertainty is added to the high core power trip.

In increase in feedwater flow events that result in a reactor trip, the subsequent actuation of SSI is credited for mitigation of changes to the secondary system inventory, and the subsequent actuation of the DHRS is credited with maintaining reactor cooling. The MPS signals credited for SSI actuation are high steam superheat, low steam superheat, high steam pressure and high RCS temperature. The MPS signals credited for DHRS actuation are high RCS temperature and high steam line pressure.

There are no single active failures that could occur during the limiting increase in feedwater flow event that could result in more severe conditions with respect to the acceptance criteria. In order to evaluate the potential for an increase in feedwater flow to overfill the SG, a case is analyzed with the failure of one FWIV to close. This maximizes pressure in the secondary system and contributes to SG overfill.

Normal AC power is assumed to be available for this event. A loss of AC power is not a conservative condition for an increase in feedwater flow event. The loss of normal power scenarios are listed below:

- Loss of normal AC - In this scenario, MPS remains powered so none of the safety systems are automatically actuated, but feedwater is lost and the turbine is tripped.
  - Loss of normal AC at the time of the steam flow increase is non-limiting because feedwater is lost which reduces the overcooling event.
  - Loss of normal AC at the time of reactor trip is non-limiting because feedwater is lost which reduces the overcooling event.
- Loss of EDNS and normal AC - Power to the reactor trip breakers is provided via the EDNS, so this scenario is the same as discussed above with addition of reactor trip at the time at which power is lost. For the increase in feedwater flow events, this scenario is non-limiting for the reasons listed above and from the immediate reactor trip.
- Loss of EDSS, EDNS and normal AC - Power to the MPS is provided via the EDSS, so this scenario results in an actuation of reactor trip system (RTS) and all of the ESFs. In terms of the overcooling event, this scenario is non-limiting for the reasons discussed above.

15.1.2.3 Thermal Hydraulic and Subchannel Analyses

15.1.2.3.1 Evaluation Model

The thermal hydraulic analysis of the plant response to an increase in feedwater flow is performed using NRELAP5. The NRELAP5 model is based on the design features of an NPM. The non-LOCA transient modifications to the NRELAP5 model are discussed in Section 15.0.2. The relevant boundary conditions from the NRELAP5 analyses are provided to the downstream subchannel CHF analysis.

The subchannel core CHF analysis is performed using VIPRE-01. VIPRE-01 is a subchannel analysis tool designed for general-purpose thermal-hydraulic analysis under normal operating conditions, operational transients, and events of moderate severity. See Section 15.0.2 for a discussion of the VIPRE-01 code and evaluation model.

15.1.2.3.2 Input Parameters and Initial Conditions

The initiating amount of feedwater flow increase is a 100 percent increase from normal flow. Feedwater flow is assumed to linearly increase from its initial steady state value to its final value over a time span of 0.1 seconds. This conservatively bounds the possible rates of feedwater flow increases.

The initial conditions used in the evaluation of the limiting increase in feedwater flow event result in a conservative calculation. Table 15.1-5 provides key inputs for the limiting increase in feedwater flow case. The following initial conditions are assumed in the analysis to ensure that the results have sufficient conservatism.

- Initial power level is assumed to be 102 percent of nominal to account for a 2 percent measurement uncertainty.
- Conservative scram characteristics are used, including a maximum time delay, holding the most reactive rod out of the core, and utilizing a bounding control rod drop rate.
- The most limiting combination of end-of-cycle core parameters is used to provide a limiting power response. The most negative MTC of -43.0 pcm/degrees F and the least negative DTC of -1.40 pcm/degrees F are used to provide the largest power response for this event.
- Allowances for instrument inaccuracy are accounted for in the analytical limits of mitigating systems in accordance with the guidance provided in RG 1.105.

The results from the thermal hydraulic evaluation are used as input to the subchannel analysis to determine the MCHFR for this event. The subchannel evaluation model is discussed in Section 15.0.2.3.

15.1.2.3.3 Results

The sequence of events for the limiting MCHFR increase in feedwater flow event is provided in Table 15.1-4. Figure 15.1-11 through Figure 15.1-21 show the transient behavior of key parameters for the limiting MCHFR increase in feedwater flow event. A spectrum of feedwater flow increase cases are analyzed, and the limiting feedwater flow increase is a near instantaneous 15 percent increase of normal feedwater flow.

A feedwater system malfunction that causes an increase in feedwater flow results in an unplanned overcooling of the RCS. The subsequent decrease in RCS temperature increases core reactivity due to moderator feedback which raises reactor power. Decreasing average RCS temperature will also prompt the CR controller to withdraw the regulating bank from the core if automatic control is enabled. Rising reactor power will potentially cause CHF conditions to develop in the core. The feedwater flow increase causes RTS actuation on low steam line superheat, high steam line pressure, high RCS temperature, or high reactor power. Closure of the FWIVs and MSIVs on secondary system isolation isolates the steam generator from the feedwater source, ending the overcooling event. DHRS actuates on the high RCS temperature or high steam line pressure signals. Core decay heat drives natural circulation which transfers thermal energy from the RCS to the reactor pool via the DHRS. Passive DHRS cooling is established and the transient calculation is terminated with the NPM in a safe, stable condition.

The MCHFR for the limiting increase in feedwater flow event does not fall below the 95/95 acceptance criterion discussed in Section 4.4.4. The MCHFR for the limiting increase in feedwater flow case does not violate the design limit.

During the overcooling phase, RCS flow steadily increases in response to rising reactor power. The reactor scram causes a rapid decrease in flow as the heat source driving natural circulation is reduced. The flow oscillates until RCS temperatures

re-equilibrate. Eventually, RCS flow stabilizes and passive DHRS cooling dominates. The reactor trip, subsequent actuation of SSI and DHRS, and stabilization of RCS flow demonstrate the plant response to increase in feedwater flow, and a return to a stable condition with no operator actions. For a discussion on possible return to power scenarios, see Section 15.0.6.

Figure 15.1-56 shows the SGS pressure using the maximum SGS pressure case.

15.1.2.4 Radiological Consequences

The normal leakage related radiological consequences of this event are bounded by the design basis accident analyses presented in Section 15.0.4.

15.1.2.5 Conclusions

The five DSRS acceptance criteria for this AOO are met for the limiting increase in feedwater flow case. These acceptance criteria, followed by how the NuScale Power Plant design meets them are listed below:

1) Pressure in the reactor coolant and main steam systems should be maintained below 110 percent of the design values.

    - The pressure responses in the RPV and in the MSS are less severe than those of the AOOs presented in Section 15.2, decrease in heat removal by the secondary side. Therefore, this acceptance criterion is met for increase in steam flow event. The maximum pressure values for the cases analyzed are shown in Table 15.1-6.

2) Fuel cladding integrity shall be maintained by ensuring that the minimum DNBR remains above the 95/95 DNBR limit based on acceptable correlations (see DSRS Section 4.4).

    - The MCHFR for this event is above the 95/95 limit as shown in Table 15.1-6. Therefore, this acceptance criterion is met.

3) An incident of moderate frequency should not generate a more serious plant condition without other faults occurring independently.

    - The analysis presented for this event shows that stable DHRS cooling is reached, and the acceptance criterion is met.

4) The guidance provided in RG 1.105, "Instrument Spans and Setpoints," can be used to analyze the impact of instrument spans and setpoints on the plant response to the type of transient addressed in this DSRS section, in order to meet the requirements of General Design Criteria 10, 13, 15, 20, and 26.

    - The instrument spans and setpoints discussed in Section 15.1.2.3.2 address the guidance in RG 1.105.

5) The most limiting plant systems single failure, as defined in the "Definitions and Explanations" of Appendix A to 10 CFR Part 50, shall be identified and assumed in the analysis and shall satisfy the positions of RG 1.53.

- The timing of the reactor trip precludes possible single failures from negatively affecting the acceptance criteria of the transient, as discussed in Section 15.1.2.2.

15.1.3 Increase in Steam Flow

15.1.3.1 Identification of Causes and Accident Description

An increase in steam flow event could occur due to a spurious opening of the turbine bypass valve. The inadvertent opening of the main steam safety valves (MSSVs), described in Section 15.1.4, could also cause an increase in steam flow. The opening of a MSSV creates a similar plant response to the inadvertent opening of the turbine bypass valve, and is bounded by the analysis presented in this section. For the NuScale Power Plant design, the increased steam flow increases heat transfer from the primary side, which in turn results in a power increase due to the moderator density increase. As the RCS cools, the reactivity control system will insert positive reactivity by pulling the regulating control rods from the core in an attempt to maintain RCS temperature. The insertion of positive reactivity results in an increase in reactor power, and a decrease in the MCHFR.

An increase in steam flow is expected to occur one or more times in the life of the reactor, so it is classified as an AOO. The categorization of the NuScale DBEs are discussed in Section 15.0.0.

15.1.3.2 Sequence of Events and Systems Operation

The sequence of events for the limiting increase in steam flow case is provided in Table 15.1-7.

Unless specified below, the analysis of an increase in steam flow event assumes the PCSs and ESFs perform as designed, with allowances for instrument inaccuracy. No operator action is credited to mitigate the effects of an increase in steam flow event.

The maximum possible increase in steam flow is a 100 percent increase of normal full power steam flow, following a spurious opening of the turbine bypass valve. In order to bound the maximum steam flow increase, a maximum increase of 125 percent of normal steam flow is analyzed. However, this condition results in rapid RTS actuation before limiting CHF conditions can develop. Less extreme steam flow increases delay RTS actuation, given the slower power response and slower SG depressurization. A steam flow increase of approximately 12 percent allows more time for the system thermal response to develop, which is more limiting in terms of MCHFR. The FWS is not credited with mitigating the overcooling effect of an increase in steam flow. It is assumed to have an unlimited makeup capacity to provide more than 150 percent of the nominal full power capacity. The analysis of an increase in steam flow event assumes that the feedwater pump speed remains constant and that the maximum increase in feedwater flow is 0.5 lbm for every one psi decrease in SG pressure. These assumptions maximize the overcooling event by increasing the available source of secondary coolant.

Operator action is not credited for regulating control rod movement or increasing boron concentration. This ensures that the maximum reactivity insertion is reached as the control system attempts to maintain RCS temperature by pulling the regulating control rods from the core.

The MPS is credited to protect the plant in the event of an increase in steam flow. When the steam flow increases to a level that causes a high enough power excursion, the MPS high power signal trips the reactor, preventing the reactor from reaching a power level where the acceptance criteria could be challenged. For the limiting MCHFR case the reactor trip is delayed by imposing a steam flow increase of 12 percent. A 12 percent flow increase yields the liming combination of reactor power and hot leg temperature to minimize CHFR. The following MPS signals provide the plant with protection during an increase in steam flow:

- high core power (5 percent uncertainty added)
- high core power rate (not credited in the safety analysis of this event)
- high RCS temperature
- high steam superheat
- high PZR pressure
- low PZR pressure
- low steam pressure
- low PZR level

Due to the cooling of the RCS during an increase in steam flow event, the coolant in the downcomer increases in density. This increase in density can affect the power level detection by the excore neutron detectors. In order to account for this effect, the high core power rate trip is not credited in the analyses, and a 5 percent uncertainty is added to the high core power trip.

In increased steam flow events that result in a reactor trip, the subsequent actuation of SSI and the DHRS is credited with maintaining reactor cooling. The MPS signals credited for SSI actuation are high RCS temperature, high PZR pressure, low PZR pressure and high superheat. The MPS signals credited for DHRS actuation are, high RCS temperature, high steam pressure, and high PZR pressure.

There are no single failures that could result in a more severe outcome of the limiting increase in steam flow event with respect to the acceptance criteria. Two possible failures that could affect the transient are the failure of one of the MSIVs or one of the FWIVs. However, as discussed in Section 15.1.1 and Section 15.1.2, these single failures could only occur after secondary side isolation or DHRS actuation, which occurs coincident with or after the reactor trip, and the MCHFR has already occurred. Therefore, neither single failure is modeled in the case presented in this section.

Normal AC power is assumed to be available for this event. A loss of AC power is not a conservative condition for an increase in steam flow event. The loss of normal power scenarios are listed below.

- Loss of normal AC - In this scenario, MPS remains powered, so none of the safety systems are automatically actuated, but feedwater is lost, and the turbine is tripped.
  - Loss of normal AC at the time of the steam flow increase is non-limiting because feedwater is lost, which reduces the overcooling event.
  - Loss of normal AC at the time of reactor trip is non-limiting because feedwater is lost, which reduces the overcooling event.
- Loss of EDNS and Normal AC - Power to the reactor trip breakers is provided via the EDNS, so this scenario is the same as discussed above with addition of reactor trip at the time at which power is lost. For the increase in steam flow events, this scenario is non-limiting for the reasons listed above and from the immediate reactor trip.
- Loss of EDSS, EDNS and Normal AC - Power to the MPS is provided via the EDSS, so this scenario results in an actuation of RTS and all of the ESFs. In terms of the overcooling event, this scenario is non-limiting for the reasons discussed above.

15.1.3.3 Thermal Hydraulic and Subchannel Analyses

15.1.3.3.1 Evaluation Model

The thermal hydraulic analysis of the plant response to an increase in steam flow is performed using NRELAP5. A description of the NRELAP5 model is provided in Section 15.0.2. The NRELAP5 model is based on the design features of a NPM. The non-LOCA transient modifications to the NRELAP5 model are discussed in Section 15.0.2. The relevant boundary conditions from the NRELAP5 analyses are provided to the downstream subchannel CHF analysis.

The subchannel core CHF analysis is performed using VIPRE-01. VIPRE-01 is a subchannel analysis tool designed for general-purpose thermal-hydraulic analysis under normal operating conditions, operational transients, and events of moderate severity. See Section 15.0.2 for a discussion of the VIPRE-01 code and evaluation model.

15.1.3.3.2 Input Parameters and Initial Conditions

As discussed in Section 15.1.3.2, the limiting increase in steam flow is an increase of 12 percent normal steam flow. Steam flow is assumed to linearly increase from its initial steady state value to its final value over a time span of 0.1 seconds. This conservatively bounds the possible rates of steam flow increases.

The initial conditions used in the evaluation of the limiting increase in steam flow event result in a conservative calculation. Table 15.1-8 provides key inputs for the limiting increase in steam flow case. The following initial conditions are assumed in the analysis to ensure that the results have sufficient conservatism.

- Initial power level is assumed to be 102 percent of nominal to account for a 2 percent measurement uncertainty.

- Conservative scram characteristics are used, including a maximum time delay, holding the most reactive rod out of the core, and utilizing a bounding control rod drop rate.
- The end-of-cycle core parameters are used to provide a limiting power response. The most negative MTC of -43.0 pcm/degrees F and DTC of -2.5 pcm/degrees F are used to provide the largest power response for this event.
- Allowances for instrument inaccuracy are accounted for in the analytical limits of mitigating systems in accordance with the guidance provided in RG 1.105.

The results from the thermal hydraulic evaluation are used as input to the subchannel analysis to determine the MCHFR for this event. The subchannel model is discussed in Section 15.0.2.3.

15.1.3.3.3 Results

An increase in steam flow increases secondary system heat removal causing RCS temperature to decrease. In response, the regulating bank controller is assumed to withdraw the regulating bank to compensate for positive moderator feedback, and reactor power increases. As power and subsequently core outlet temperature increase, margin to the CHF limit decreases.

The sequence of events for the limiting MCHFR increase in steam flow event is provided in Table 15.1-7. Figure 15.1-22 through Figure 15.1-31 show the transient behavior of key parameters for an increase in steam flow event. A spectrum of steam flow increase cases were analyzed. In general, low steam flow increases are limiting for MCHFR since hot leg temperature and pressure increase to reactor trip actuation limits slower given the slow increase in power. When the steam flow increase becomes greater in magnitude, RTS actuation occurs earlier on high reactor power and then high steam superheat signals. These cases are not limiting since hot leg temperature remains at a relatively low value.

The limiting event initiates with an increase in steam flow by approximately 12 percent of full power steam flow in 0.1 seconds. The feedwater pump flow rate is allowed to increase with the steam flow to maximize the overcooling effect. The RCS response to the overcooling event begins once the steam flow through the SG begins to increase. This increase in steam flow leads to an increase in the heat removal rate from the RCS via the SG. During the overcooling phase of the transient, the RCS temperature steadily decreases, while the RCS density increases. The colder, denser RCS causes the regulating control rod bank to withdraw in an attempt to maintain RCS temperature. The withdrawal of the control rods causes a positive reactivity insertion that increases reactor power. This leads to an RTS actuation on high reactor power.

The decrease in reactor coolant temperature causes the primary coolant volume to shrink, which initially reduces the pressure of the RCS. However, as core power increases, the RCS pressure begins to rise. The hot leg temperature rises in response to the increase in reactor power. The high reactor power limit is reached

at 63 seconds into the transient, tripping the reactor at 65 seconds. The peak RCS pressure occurs around the time of the reactor scram.

Steam generator pressure does not change significantly during the initial phase of the transient. The low PZR pressure limit is reached at 123 seconds which causes a SSI to occur. An SSI actuation causes closure of the FWIVs and MSIVs which causes pressurization of the SG. Steam generator pressure increase resulting from main steam isolation is expected and is not a direct consequence of the increase in steam flow. The high steam pressure limit is reached at 1694 seconds which actuates DHRS. The maximum secondary pressure is reached after main steam and feedwater isolation and DHRS actuation.

The CHFR decreases as reactor power and RCS pressure increase. The automatic protection systems terminate this transient before the CHFR reaches the design limit. The MCHFR for the limiting increase in steam flow case does not violate the design limit.

During the overcooling phase, RCS flow steadily increases in response to rising reactor power. The reactor scram causes a rapid decrease in flow as the heat source driving natural circulation is reduced. The flow oscillates until RCS temperatures re-equilibrate. After 2000 seconds, RCS flow stabilizes and passive DHRS cooling dominates. The reactor trip, subsequent secondary side isolation and actuation of DHRS, and stabilization of RCS flow demonstrate the plant response to an increase in steam flow, and a return to a stable condition with no operator actions. For a discussion on possible return to power scenarios, see Section 15.0.6.

The highest RCS pressure and temperature occurs when an 11 percent increase in steam flow is evaluated. This occurs prior to reaching DHRS actuation and thus has no effect on the limiting MCHFR case.

15.1.3.4 Radiological Consequences

The normal leakage-related radiological consequences of this event are bounded by the design basis accident analyses presented in Section 15.0.3.

15.1.3.5 Conclusions

The five DSRS acceptance criteria for this AOO are met for the limiting increase in steam flow case. These acceptance criteria, followed by how the NuScale design meets them are listed below.

1) Pressure in the RCS and MSS should be maintained below 110 percent of the design values.

    - The pressure responses in the RPV and in the MSS are less severe than those of the AOOs presented in Section 15.2, decrease in heat removal by the secondary side. Therefore, this acceptance criterion is met for increase in steam flow event. The maximum pressure values for the cases analyzed are shown in Table 15.1-9.

2) Fuel cladding integrity shall be maintained by ensuring that the minimum DNBR remains above the 95/95 DNBR limit based on acceptable correlations (see DSRS Section 4.4).

    - The MCHFR for this event is above the 95/95 limit as shown in Table 15.1-9. Therefore, this acceptance criterion is met.

3) An incident of moderate frequency should not generate a more serious plant condition without other faults occurring independently.

    - The analysis presented for this event shows that stable DHRS cooling is reached, and the acceptance criterion is met.

4) The guidance provided in RG 1.105, "Instrument Spans and Setpoints," can be used to analyze the impact of instrument spans and setpoints on the plant response to the type of transient addressed in this DSRS section, in order to meet the requirements of General Design Criteria 10, 13, 15, 20, and 26.

    - The instrument spans and setpoints discussed in Section 15.1.3.3.2, address the guidance in RG 1.105.

5) The most limiting plant systems single failure, as defined in the "Definitions and Explanations" of Appendix A to 10 CFR Part 50, shall be identified and assumed in the analysis and shall satisfy the positions of RG 1.53.

    - The limiting single failure for an increase in steam flow event could be a failure of one of the MSIVs or one of the FWIVs. However, the prior reactor trip precludes this failure from negatively affecting the limiting MCHFR case presented, as discussed in Section 15.1.3.2.

15.1.4 Inadvertent Opening of Steam Generator Relief or Safety Valve

This section is typically used to provide an analysis for an inadvertent opening of a steam generator relief or MSSV. The NuScale Power Plant design does not have a SG relief valve, but does have two MSSVs. The event is initiated by the spurious opening of one of the two main MSSVs, which are located downstream of the secondary MSIVs. The MSSVs are sized to accommodate 100 percent of the full power steam flow. As both valves are required to meet this flow requirement, the spurious opening of one MSSV following a mechanical failure would yield a steam flow increase less than 100 percent of the flow at full power conditions. A spurious opening of the turbine bypass valve would yield a similar system response as it is also located downstream of the secondary MSIVs. However, the increase in steam flow due to a spurious opening of the turbine bypass valve could result in a 100 percent increase in steam flow. Therefore, the spectrum of possible steam increases due to a full or partial turbine bypass valve opening covers and bounds the steam increases due to an inadvertent opening of a MSSV. The analysis of a limiting increase in steam flow is presented in Section 15.1.3.

15.1.5 Steam Piping Failures Inside and Outside of Containment

15.1.5.1 Identification of Causes and Accident Description

A steam line break (SLB) event for the NuScale Power Plant design could range from a small break to a double ended rupture of the main steam line. This event could occur inside or outside of the containment vessel (CNV). A spectrum of SLB locations with varied core and plant conditions are analyzed to determine the scenarios with the most severe results.

A SLB inside the CNV would increase the pressure inside containment, reaching the high containment pressure analytical limit. The high containment pressure signal trips the core, isolates the CNV, and actuates SSI. The break flow would decrease due to SG depressurization until dryout due to feedwater isolation. The containment pressure is sensitive to any SLB size, so the protection system detects the break sooner than a comparable break outside of containment. A spectrum of breaks inside containment is evaluated to ensure that containment pressure is acceptable. The peak containment pressure remains below the design limit for all postulated events, as shown in Section 6.2. Aside from containment pressure, the plant conditions for a SLB inside containment are bounded by the analysis presented in this section for a SLB outside of containment.

A SLB outside the CNV would cause an increase in steam flow event that could either cause a low SG pressure signal or a high core power trip due to the reactor power response from the decreased RCS temperature. The break flow would be stopped by the closure of the MSIV and depressurization of the steam system piping. The largest steam line break outside containment could occur from a double-ended rupture of the portion of the main steam line located outside of the CNV. However, a double-ended rupture results in a low steam pressure signal that occurs earlier in the transient than for a small break outside of containment. A small SLB outside of containment is the most limiting type of SLB because it provides the longest event progression before detection by the protection system. A smaller break can result in a significant delay in detection time relative to a larger break, producing more limiting primary and secondary pressures, MCHFR, and integral mass and energy release.

A significant steam piping failure is not expected to occur during the life of the plant, so the event is classified as an accident. A smaller or secondary piping failure has a higher probability of occurrence, so the SLB accident will be evaluated against the 95/95 MCHFR AOO acceptance criteria.

15.1.5.2 Sequence of Events and Systems Operation

There are separate SLB cases outside of containment that are limiting with respect to primary pressure, SGS pressure, MCHFR, and radiological consequences. The SLB event is a secondary depressurization event. The sequence of events for the limiting SLB cases are provided in Table 15.1-10, Table 15.1-11, Table 15.1-12, and Table 15.1-19.

Unless specified below, the analysis of an SLB event assumes the PCSs and ESFs perform as designed, with allowances for instrument inaccuracy. No operator action is credited to mitigate the effects of a steam piping failure.

The FWS is not credited with mitigating the overcooling effect of an increase in steam flow. The feedwater flow increases to a limit of 240 lbm/s which exceeds the maximum pump runout capacity for two trains of feedwater aligned in parallel. The analysis of an increase in steam flow event assumes that the feedwater pump speed remains constant and that the pump curve conservatively simulates a 0.4 lbm/s increase in feedwater flow for every one psi decrease in SG pressure, which is over twice the nominal rate of flow increase. These simplifying assumptions maximize the overcooling event by increasing the available source of secondary coolant.

Operator action is not credited for regulating control rod movement or increasing boron concentration. This ensures that the maximum reactivity insertion is reached as the control system attempts to maintain RCS temperature by pulling the regulating control rods from the core.

The MPS is credited to protect the plant in the event of a SLB. The following MPS signals provide the plant with protection during a SLB:

- high core power (5 percent uncertainty added)
- high core power rate (not credited in the safety analysis of this event)
- low SG pressure
- low PZR pressure (not credited in the safety analysis of this event)
- low PZR level
- high CNV pressure
- high steam superheat
- low steam superheat
- high RCS hot temperature
- high PZR pressure

Due to the cooling of the RCS during a SLB, the coolant in the downcomer increases in density. This increase in density can affect the power level detection by the excore neutron detectors. In order to account for this effect, the high core power rate trip is not credited in the analyses, and a 5 percent uncertainty is added to the high core power trip.

The DHRS is a safety-related system credited to actuate and mitigate the effects of this transient. The redundancy and passive nature of the DHRS ensure that the system will perform its intended function despite a single failure. The operation of a DHRS train is challenged when a SLB is located on the steam line downstream of the MSIV, but upstream of the backup isolation valve, with a single failure of the MSIV, or when a SLB occurs inside containment. These scenarios prevent one train of the DHRS from functioning by preventing isolation of one SG train.

The limiting single failure for a SLB inside containment is a failure of the FWIV to close on the impacted secondary train. In this scenario, the feedwater regulating valve is credited to perform the isolation function. However, the regulating valve has a significantly longer closure time than the FWIV. This maximizes the runout of the

feedwater pumps which increases the mass and energy released and increases containment pressure. This scenario also increases the energy removed by the break making the overcooling event worse.

The limiting single failure for a SLB outside of containment is the failure of one MSIV to close. If the break location is between the MSIV and backup isolation valve, a failure of the first MSIV to isolate on demand would result in a complete blowdown of the affected SG train.

Normal AC power is assumed to be available for the SLB case that is limiting for MCHFR. A loss of AC power is not a conservative condition for the overcooling conditions of an SLB that minimizes the MCHFR. The loss of normal power scenarios are listed below.

- Loss of Normal AC - In this scenario, MPS remains powered such that none of the safety systems are automatically actuated, but feedwater is lost, and the turbine is tripped.
  - Loss of normal AC at the time of the break is non-limiting for the SLB case that minimizes the MCHFR because feedwater is lost, which reduces the severity of the overcooling event and moderates the core power increase.
  - Loss of normal AC at the time of reactor trip is non-limiting because the loss of feedwater reduces the severity of the overcooling event. The core power increase due to the cooldown is moderated.
- Loss of EDNS and Normal AC - Power to the reactor trip breakers is provided via the EDNS, so this scenario is the same as discussed above with addition of reactor trip at the time at which power is lost. For the SLB events, this scenario is non-limiting due to the immediate reactor trip.
- Loss of EDSS, EDNS and Normal AC - Power to the MPS is provided via the EDSS, so this scenario results in an actuation of RTS and all of the ESFs. In terms of the SLB events, this scenario is non-limiting for the reasons discussed above.

15.1.5.3 Thermal Hydraulic and Subchannel Analyses

15.1.5.3.1 Evaluation Model

The thermal hydraulic analysis of the plant response to a SLB is performed using NRELAP5. A description of the NRELAP5 model is provided in Section 15.0.2. The NRELAP5 model is based on the design features of a NPM. The non-LOCA transient modifications to the NRELAP5 model are discussed in Section 15.0.2. The steam piping breaks are modeled in NRELAP5 as valves that instantly open at transient initiation and have a sudden infinite expansion loss. This modeling is appropriate for a SLB because the break will vent either to a relatively large CNV or an even larger reactor or turbine building. The relevant boundary conditions from the NRELAP5 analyses are provided to the downstream subchannel CHF analysis.

The subchannel core CHF analysis is performed using VIPRE-01. VIPRE-01 is a subchannel analysis tool designed for general-purpose thermal-hydraulic analysis under normal operating conditions, operational transients, and events of moderate

severity. See Section 15.0.2 for a discussion of the VIPRE-01 code and evaluation model.

The radiological analysis of the SLB event is performed using RADTRAD. A discussion of RADTRAD and the radiological evaluation model is provided in Section 15.0.3.

15.1.5.3.2 Input Parameters and Initial Conditions

Each of the SLB cases for the limiting RCS pressure, limiting SGS pressure, limiting MCHFR and limiting radiological consequences are provided below. Table 15.1-13 provides key inputs including biases for each of the limiting SLB cases. The initial conditions and assumptions used in the evaluation of a SLB event result in a conservative calculation.

Steam Line Break Case Resulting in a Limiting Reactor Coolant System Pressure

The limiting RCS pressure case results from a 7.5 percent split break located between the primary and backup MSIVs at 90 percent power in the middle of cycle (MOC) condition. The initial power level allows core power to increase to 121 percent of nominal without reactor trip or SSI. Break flow continues at the elevated core power level until 50,000 lbm of steam has been lost through the break, at which time feedwater flow is terminated due to exhaustion of the hotwell condensate inventory. The following initial conditions are assumed in the analysis of the SLB.

- Initial power level is assumed to be 90 percent of nominal.
- Conservative scram characteristics are used, including a maximum time delay, holding the most reactive rod out of the core, and utilizing a bounding control rod drop rate.
- Nominal MTC and DTC representative of middle-of-cycle core parameters are used.
- The beginning-of-life SG is assumed with no tube plugging. SG heat transfer efficiency is reduced by assuming SG tube fouling and by applying a -30 percent uncertainty to the SG heat transfer value. This biasing increases SG secondary inventory and maximizes mass and energy release. The reduced heat transfer efficiency also increases transient RCS pressure.
- The single failure identified for this event is the failure of an MSIV to close.

Steam Line Break Case Resulting in Limiting Steam Generator Pressure

The SLB event is a secondary depressurization event; however, due to the nature of DHRS operation, the combination of secondary inventory and RCS temperature can cause significant SGS pressurization post trip. This is a normal behavior of the system, and does not challenge the SGS design pressure. The limiting case is a double ended guillotine break of the steam line in the common steam line just upstream of the turbine. This case includes the following assumptions:

- The limiting case is initialized at 102% reactor power with conservatively high RCS temperature and SG pressure.
- BOC core exposure with a high fuel temperature bias, least negative MTC and least negative DTC reactivity coefficient are applied.
- This case assumes a failed FWIV on the impacted SG train, which allows additional FW into the impacted SG.

Steam Line Break Cases Resulting in the Limiting Minimum Critical Heat Flux Ratio

A thermal hydraulic analysis is performed to provide the limiting boundary conditions for the downstream subchannel analysis, which evaluates the final CHF value. The limiting MCHFR case results from a 2.5 percent split break in the MS piping just outside containment between the primary and secondary MSIVs. The following initial conditions are assumed in the analysis of a range of small SLB cases to ensure that the boundary conditions calculated for the subchannel analysis will result in the limiting MCHFR.

- Initial power level is assumed to be 102 percent of nominal to account for a 2 percent measurement uncertainty.
- Conservative scram characteristics are used, including a maximum time delay, holding the most reactive rod out of the core, and utilizing a bounding control rod drop rate.
- The beginning-of-life SG is assumed, which includes no SG tube plugging and no SG fouling. A 30 percent uncertainty is added to the SG heat transfer to maximize the cooldown event.
- The single failure identified for this event is the failure of an MSIV to close, which results in more severe plant conditions. However, this single failure does not affect MCHFR because it occurs after CHFR reaches a minimum.

The results from the thermal hydraulic evaluations are used as input to the subchannel analysis to determine the limiting MCHFR for this event. The subchannel evaluation model is discussed in Section 15.0.2.3.

SLB Cases Resulting in Limiting Radiological Consequences

There are two different SLB cases that are limiting from a radiological release perspective. The first limiting radiological case is an 18 percent break from 50 percent power at MOC conditions for which no reactor trip or SSI occurs during the first 30 minutes of the event. The other limiting radiological case is a 7.5 percent break from 80 percent power at BOC conditions that maximizes the time between the high power reactor trip and secondary isolation. These cases are initialized at reactor conditions that are biased to maximize SG secondary inventory and mass release. SG tube fouling is assumed, a SG heat transfer uncertainty of -30 percent is applied, and feedwater temperature is biased 10 degrees F low. A single failure of the MSIV on the affected train is modeled to maximize the mass and energy release after the isolation signal.

15.1.5.3.3 Results

Steam Line Break Case Resulting in the Limiting Reactor Coolant System Pressure

The sequence of events for the limiting RCS pressure SLB case event is provided in Table 15.1-10. Figure 15.1-32 through Figure 15.1-37 show the transient behavior of key parameters for this SLB case. The transient initiates with a 7.5 percent split break of the main steam piping outside containment between the primary and backup MSIVs at 90 percent power in the middle of cycle (MOC) condition. The effects of the postulated SLB on other systems are considered in Section 3.6 consistent with Branch Technical Position (BTP) 3-3 and BTP 3-4. Most SLB events have a cooldown effect on the RCS, where there is little or no pressurization of the RCS. However, this SLB case represents a small break in the steam line that causes core power to increase and reach a new stable condition without a reactor trip or secondary isolation. Feedwater is terminated due to assumed hotwell inventory depletion after 50,000 lbm of steam has been lost through the break, causing additional RCS heatup and pressurization after the reactor has reached conditions well above rated power. Feedwater termination causes a reactor trip on low steam pressure. This scenario resulted in the highest RCS pressure for the steam piping failure events analyzed.

Due to reactivity feedback from the cooldown, core power rises to ~121 percent and holds at that level until feedwater is exhausted. The RCS pressure remains below the RSV lift point. This scenario challenges the ability to transition to DHRS cooling on the intact SG following a total loss of feedwater after sustained operation at very high reactor power. The analysis results show that sufficient condensate inventory remains in the intact SG/DHRS loop following secondary isolation for the intact SG DHRS to establish core cooling and effective decay heat removal.

After DHRS is actuated, the heat removal on the steam side is limited by the DHRS condenser. This causes a pressurization of the secondary side. This pressurization is an expected function of the DHRS actuation, and not a direct consequence of the steam pipe break accident. The peak main steam pressure is plotted (Figure 15.1-37) to demonstrate that peak pressure remains well below 110 percent of design pressure.

The limiting single failure assumed in this SLB case is the failure of the primary MSIV on the impacted train to close on demand, which allows the impacted SG to completely empty and depressurize after reactor trip and DHRS actuation. This single failure results in a more severe plant condition by releasing more mass through the break. It also causes a higher heat load on the intact DHRS train. The MSIV failure could be non-limiting on RCS pressure by allowing more steam flow for cooling. However, the failed MSIV has little effect on the RCS pressure because the failed steam line is already nearly empty at the time of DHRS actuation.

The result of this SLB case is a stable plant condition, where the DHRS maintains core cooling. For a discussion of a possible return to power scenario, see Section 15.0.6.

Steam Line Break Case Resulting in Limiting Steam Generator Pressure

The sequence of events for the limiting SGS pressure SLB case event is provided in Table 15.1-19. Figure 15.1-61 through Figure 15.1-62 show the transient behavior of key parameters for this SLB case. The limiting SGS pressure case is a double ended guillotine break of the steam line in the common steam line just upstream of the turbine. A FWIV failure to close is assumed as the worst case single failure. The FWIV failure on the impacted SG allows additional inventory to be added to the impacted loop following the reactor and turbine trip. The additional SG inventory in the impacted loop exacerbates the SG pressure increase following turbine trip and MSIV closure. The acceptance criterion for peak secondary pressure is satisfied for the steam pipe break accident.

Steam Line Break Case Resulting in the Limiting Minimum Critical Heat Flux Ratio

The sequence of events for a representative limiting MCHFR SLB case is provided in Table 15.1-11. Figure 15.1-38 through Figure 15.1-44 show the transient behavior of key parameters for this SLB case. The transient initiates with a 2.5 percent split break of the main steam piping outside containment between the primary and secondary MSIVs. The effects of the postulated SLB on other systems are considered in Section 3.6 consistent with BTP 3-3 and BTP 3-4. This small break minimizes MCHFR by allowing the system to reach the core power limits rather than the low main steam pressure or high main steam superheat limits.

The SLB causes an increase in steam flow through the SG. The feedwater pump flow rate is allowed to increase with the steam flow to maximize the overcooling effect. This increase in steam flow leads to an increase in the heat removal rate from the RCS via the SG. During the overcooling phase of the transient, the RCS temperature steadily decreases, while the RCS density increases. The colder, denser RCS causes the regulating control rod bank to withdraw in an attempt to maintain RCS temperature. The withdrawal of the control rods causes a positive reactivity insertion that increases reactor power. If the regulating control rod bank were disabled or kept in insertion-only mode, a similar power response would be driven by moderator feedback instead. Reactor power reaches the high power limit, tripping the reactor.

In this SLB case, the overcooling effect initially lowers RCS pressure. However, as core power rises, pressure increases until the reactor trip. The pressure in the intact SG increases following secondary isolation and DHRS actuation, for the same reasons as the main steam pressure increase shown in the limiting RPV pressure SLB case. Steam pressure in the impacted SG decreases to near atmospheric following secondary isolation and SG dryout. The peak primary and secondary pressures are bounded by the pressures presented for the maximum pressure cases.

The CHFR decreases as reactor power and RCS pressure increase. The automatic protection systems terminate this transient before the CHFR reaches the design limit. The MCHFR for the limiting steam pipe break case does not violate the design limit.

As in the SLB case with limiting RPV pressure, the limiting single failure assumed in this SLB case is the failure of the primary MSIV on the impacted train to close on demand, which allows the impacted SG to completely empty and depressurize after reactor trip and DHRS actuation. This single failure results in a more severe plant condition by releasing more mass through the break, increasing the total amount of cooldown, but not the cooldown rate. It also causes a higher heat load on the intact DHRS train. However, this failure does not affect the limiting CHFR because it occurs after the time of MCHFR.

The result of this SLB case is a stable plant condition, where the DHRS maintains core cooling. For a discussion of a possible return to power scenario, see Section 15.0.6.

SLB Cases Resulting in Limiting Radiological Consequences

The sequence of events for a representative limiting SLB case for radiological consequences is provided in Table 15.1-12. Figure 15.1-45 and Figure 15.1-46 demonstrate the break flow rate and integral break flow for this case. This SLB scenario maximizes the time between the reactor trip and secondary isolation to maximize iodine spiking time. The break is small enough to avoid the low SG pressure trip, but eventually causes a high power trip based on rod bank withdrawal due to the cooldown. The break continues to cool the RCS until the temperature drops sufficiently that the low steam superheat SSI actuation signal isolates the break. The calculated break flow characteristics and reactor trip timing for this case, as well as an undetected small break case are used in the downstream radiological consequences analysis. Table 15.1-14 provides the integrated mass flows from these cases with additional margin that is used in the downstream radiological analysis presented in Section 15.0.3.

15.1.5.4 Radiological Consequences

The radiological consequences of the SLB event are discussed in Section 15.0.3. The results are summarized in Table 15.0-12.

15.1.5.5 Conclusions

The four DSRS acceptance criteria for this accident are met for the limiting SLB cases. These acceptance criteria, followed by how the NuScale design meets them are listed below.

1) Pressure in the RCS and MSS should be maintained below acceptable design limits, considering potential brittle, as well as ductile failures.

    - The limiting RPV pressure for a SLB is under the more conservative AOO acceptance criterion of 110 percent of design values. The limiting MSS pressures for a SLB must be less than or equal to 110 percent of the design value. The calculated RPV and MSS pressures demonstrate margin to the acceptance criterion. Therefore, the acceptance criteria for pressures is met for this event. The maximum primary and SG pressure values for the cases analyzed are shown in Table 15.1-15.

2) The potential for core damage is evaluated on the basis that it is acceptable if the minimum DNBR remains above the 95/95 DNBR limit for pressurized water reactors based on acceptable correlations (see DSRS Section 4.4).

   - The MCHFR for this event is above the 95/95 limit as shown in Table 15.1-15. Therefore, this acceptance criterion is met.

3) The radiological criteria used in the evaluation of steam system pipe break accidents (pressurized water reactors only) appear in DSRS Section 15.0.3.

   - The radiological analysis of the SLB accident is presented in Section 15.0.3 and demonstrates that the acceptance criteria are met.

4) System(s) provided for decay heat removal must be highly reliable and, when required, automatically initiated. For the NuScale Power Plant design, the DHRS provides the safety-related means of decay heat removal.

   - The results of the analysis show that the DHRS initiates and provides heat removal during a SLB, ensuring that acceptance criteria are not challenged.

15.1.6 Loss of Containment Vacuum/Containment Flooding

15.1.6.1 Identification of Causes and Accident Description

A loss of containment vacuum and containment flooding events that result in an increase in RCS cooling are NuScale Power Plant design-specific events. The NuScale containment net volume is less than conventional designs and the module is partially immersed in a pool of borated water during normal operation. Since the containment operates at a vacuum during normal operation, air or water ingress into containment could increase heat transfer from the RPV to the reactor pool. This overcooling could lead to a higher reactor power, higher RCS pressure, and reduced MCHFR.

The containment evacuation system (CES) maintains the containment volume at a vacuum during normal operation. A failure in the CES could result in loss of vacuum since containment pressure would increase due to evaporation of any RCS fluid leaking into containment. If the failure of the CES or RCS fluid leakage is sufficiently severe, this could result in a loss of vacuum event. If the containment vacuum is lost, heat transfer from the reactor vessel will increase. The analysis of a loss of containment vacuum shows a negligible effect on reactor power, and is therefore bounded by a containment flooding event.

The reactor component cooling water system (RCCWS) provides heat removal to the control rod drive system. The RCCWS supplies RCCW to CNTS that then conducts RCCW to CRDS piping that passes through containment to provide this function. If piping containing RCCW were to leak or rupture inside the CNV, a containment flooding event would occur. Other potential containment flooding sources include: feedwater containing line break, main steam containing line break, CVCS fluid containing line break, high point vent fluid containing pipe break, and RCCWS fluid containing line break. The feedwater fluid containing line break event is evaluated in Section 15.2.7, the SLB event is evaluated in Section 15.1.5, and the CVCS fluid containing line break is evaluated in Section 15.6.2. The RCCWS fluid line break is a more limiting containment

flooding event than a high point vent fluid pipe because it has a temperature lower than the containment saturation temperature. If the lower temperature RCCWS fluid line ruptures, there would be no immediate boiling, preventing the high containment pressure limit from being reached. The flooding of the CNV could cause an increase in heat transfer from the RPV to containment, cooling the RCS. As the RCS cools, reactor power increases due to the negative moderator coefficient. This unexpected rise in core power would decrease the MCHFR, and lead to an over pressurization of the RPV.

A loss of containment vacuum event is categorized as an AOO. Typically, pipe system failures are categorized as accidents, but the containment flooding event is conservatively categorized as an AOO. The categorization of the NuScale DBEs is discussed in Section 15.0.

15.1.6.2 Sequence of Events and Systems Operation

The event sequence table for the Loss of Containment Vacuum/Containment Flooding event is provided in Table 15.1-18.

Unless specified below, the analysis of the containment flooding event assumes the PCSs and ESFs perform as designed, with allowances for instrument inaccuracy. No operator action is credited to mitigate the effects a containment flooding event.

The containment flooding is initiated by a break in a CRDS line that conducts RCCW inside containment. The total RCCWS volume is conservatively assumed to be approximately 501 ft^3. The representative containment flooding case assumes one RCCWS pump continues operating after the break.

The containment evacuation pump could malfunction to cause a loss of containment vacuum scenario, but it is assumed to operate at capacity for a containment flooding scenario in order to delay the high containment pressure limit.

Operator action is not credited for regulating control rod movement or increasing boron concentration. This ensures that the maximum reactivity insertion is reached as the control system attempts to maintain RCS temperature by pulling the regulating control rods from the core.

The MPS high containment pressure signal is credited to provide protection against loss of containment vacuum and containment flooding events. In loss of containment vacuum/containment flooding cases that result in a reactor trip, the same high containment pressure signal actuates the SSI to maintain reactor cooling. DHRS is subsequently actuated on high steam line pressure after containment system isolation is completed.

There are no single failures that could make a containment flooding event more severe with respect to the acceptance criteria.

The potential loss of normal power scenarios for a containment flooding event are discussed below:

- Loss of Normal AC - This scenario causes a loss of power to the EDS chargers which automatically trips the reactor, actuates DHRS and isolates containment within 60 seconds. The timing of these actuations is bounded by the high containment pressure trip that occurs in a containment flooding event.
- Loss of EDNS and Loss of Normal AC - Power to the reactor trip breakers is provided via the nonsafety DC power distribution (EDNS) so this scenario is the same as discussed above with the addition of the reactor trip at the time at which power is lost. For the containment flooding event, this scenario is bounded for the same reason a loss of normal AC power is bounded.
- Loss of EDSS, EDNS, and Loss of Normal AC - Power to the MPS is provided via the highly reliable DC power distribution (EDSS) so this scenario results in an actuation of RTS and all of the ESFs. The flooding event is a cooldown event and as such, the limiting pressure is dependent upon the timing that the MSIVs and FWIVs close. This scenario results in the quickest timing for valve isolations and allows for the actuation of ECCS following module depressurization below the IAB pressure. This case is analyzed, but is not limiting with respect to MCHFR.

15.1.6.3 Thermal Hydraulic and Subchannel Analyses

15.1.6.3.1 Evaluation Model

The thermal hydraulic analysis of the plant response to containment flooding is performed using NRELAP5. A description of the NRELAP5 model is provided in Section 15.0.2. The NRELAP5 model is based on the design features of a NPM. The non-LOCA transient modifications to the NRELAP5 model are discussed in Section 15.0.2. The relevant boundary conditions from the NRELAP5 analyses are provided to the downstream subchannel CHF analysis.

The subchannel core CHF analysis is performed using VIPRE-01. VIPRE-01 is a subchannel analysis tool designed for general-purpose thermal-hydraulic analysis under normal operating conditions, operational transients, and events of moderate severity. See Section 15.0.2 for a complete discussion of the VIPRE-01 code and evaluation model.

15.1.6.3.2 Input Parameters and Initial Conditions

Table 15.1-16 provides key inputs for the limiting CNV case. The initial conditions used in the evaluation of this containment flooding case ensure that the results have sufficient conservatism.

- RCCWS break flow conditions:
  - One RCCWS pump operating to produce a break flow rate of 660 gpm.
  - RCCWS water assumed to be 141 degrees F.
- Initial power level is assumed to be 102 percent of nominal to account for a 2 percent measurement uncertainty.

- Conservative scram characteristics are used, including a maximum time delay, holding the most reactive rod out of the core, and utilizing a bounding control rod drop rate.
- The most negative EOC core parameters are used to provide a limiting power response. The most negative MTC of -43.0 pcm/degrees F and the most negative DTC of -2.5 pcm/degrees F are used to provide the largest power response for this event.
- The containment conditions are biased to provide the longest delay from transient initiation to detection by the high containment pressure limit. The containment initial conditions are biased so as to empty the entire volume of RCCW into containment.
- Allowances for instrument inaccuracy are accounted for in the analytical limits of mitigating systems in accordance with the guidance provided in RG 1.105.

The results from the thermal hydraulic evaluation are used as input to the subchannel analysis to determine the MCHFR for this event. The subchannel evaluation model is discussed in Section 15.0.2.3.

15.1.6.3.3 Results

Figure 15.1-47 through Figure 15.1-53 show the transient behavior of key parameters for the containment flooding case with the limiting MCHFR.

The containment flooding event is initiated by a rupture in a CRDS line that conducts RCCW inside the CNV. As the RCCWS fluid enters containment, there is an initial rise in containment pressure due to the fluid and small amount of vapor generation due to the lower pressure inside containment relative to the RCCWS fluid pressure. This rise in containment pressure results in the CES pumping vapor out of the CNV to maintain containment pressure. This delays the containment pressure from reaching the high containment pressure limit, which results in a reactor trip and terminates the event. The remaining RCCWS fluid enters containment, raising the liquid level in the CNV to a level that allows fluid to replace the space between the RPV and the CNV walls. The fluid allows an increase in heat transfer from the RPV, to the CNV.

For the containment flooding and loss of vacuum events, the volume of RCCW is limited to that which is originally contained within the system 501 ft^3 and fills containment to a level that is well below the lower end of the range for the CNV level analytical limit. Since the high CNV level analytical limit is not reached, ECCS is not actuated and continued heating of the water within containment results in containment isolation and reactor trip when the high containment pressure analytical limit of 9.5 psia is reached.

As the heat transfer increases from the RPV to the CNV, the RCS is overcooled. The slightly cooler core inlet temperature introduces an increase in reactivity due to the moderator temperature reactivity feedback, resulting in an increase in reactor power. The magnitude of the power increase is smaller than the other overcooling events presented in Section 15.1. The smaller power increase ensures that the

pressures of the RCS and MSS do not challenge the design pressures of the RPV and main steam piping, respectively. As with other overcooling events, if a containment flooding event trips the reactor, the subsequent pressure rises in the primary and secondary systems are a result of the isolation functions and not a direct consequence of the containment flooding.

Most containment flooding scenarios are terminated by the protection system before any significant change in CHFR. The MCHFR calculated for this event does not challenge the design limit as shown in Table 15.1-17 and is bounded by the other overcooling events presented inSection 15.1.

This containment flooding event that is limiting for MCHFR results in containment isolation, subsequent DHRS actuation and establishment of DHRS cooling, demonstrating a stable condition.

15.1.6.4 Radiological Consequences

The normal leakage-related radiological consequences of this event are bounded by the design basis accident analyses presented in Section 15.0.3.

15.1.6.5 Conclusions

The five DSRS acceptance criteria for this AOO are met for the limiting containment flooding case. These acceptance criteria, followed by how the NuScale design meets them are listed below.

1) Pressure in the RCS and MSS should be maintained below 110 percent of the design values.

    - The pressure responses in the RPV and in the MSS are less severe than those of the AOOs presented in Section 15.2, decrease in heat removal by the secondary side. Therefore, this acceptance criterion is met. The maximum primary and SG pressure values for the cases analyzed are shown in Table 15.1-17.

2) Fuel cladding integrity shall be maintained by ensuring that the minimum DNBR remains above the 95/95 DNBR limit based on acceptable correlations (see DSRS Section 4.4).

    - The MCHFR for this event is above the 95/95 limit as shown in Table 15.1-17, and bounded by the other overcooling events presented in Section 15.1. Therefore, this acceptance criterion is met.

3) An incident of moderate frequency should not generate a more serious plant condition without other faults occurring independently.

    - The analysis presented for this event demonstrates that stable DHRS cooling is reached, and the acceptance criterion is met.

4) The guidance provided in RG 1.105, "Instrument Spans and Setpoints," can be used to analyze the impact of instrument spans and setpoints on the plant response to the type of transient addressed in this DSRS section, in order to meet the requirements of General Design Criteria 10, 13, 15, 20, and 26.

- The instrument spans and setpoints discussed in Section 15.1.6.3.2, address the guidance in RG 1.105.

5) The most limiting plant systems single failure, as defined in the "Definitions and Explanations" of Appendix A to 10 CFR Part 50, shall be identified and assumed in the analysis and shall satisfy the positions of RG 1.53.

    - There is no single failure that adversely affects the transient results of a containment flooding event.

Table 15.1-1: Sequence of Events Limiting MCHFR Case (15.1.1 Decrease in Feedwater Temperature)

Event	Time [s]
Feedwater temperature begins to decrease	0
Feedwater temperature reaches 100 °F	86
Regulating bank begins to withdraw in response to a decrease in average RCS temperature	100
High hot leg temperature analytical limit is reached	184
High reactor power analytical limit is reached.	187
Peak reactor power/ Limiting MCHFR/control rod insertion begins	189
DHRS and SSI actuation	192

Table 15.1-2: Decrease in Feedwater Temperature - Inputs (Limiting Minimum Critical Heat Flux Ratio Case)

Parameter	Nominal	Bias
Core power	160 MWt	+2%
Pressurizer pressure	1850 psia	+70psia
RCS flow rate	See Table 15.0-6 for range	Minimum -1174 lbm/s
RCS average temperature	545 °F	+10 °F
SG pressure	500 psia	-35psia
FW temperature	300 °F	-10 °F
Pressurizer level	60%	+8%
SG heat transfer bias	Nominal	N/A
Reactor pool temperature	100 °F	Nominal (100 °F)
MTC	EOC	Most Negative
DTC	EOC	Most Negative

Table 15.1-3: Decrease in Feedwater Temperature (15.1.1) - Limiting Analysis Results

Acceptance Criteria	Limit	Analysis Value
Maximum RCS pressure	2310 psia	2005 psia
Maximum SG pressure	2310 psia	1541 psia
MCHFR	1.284	1.847

Table 15.1-4: Sequence of Events for Limiting MCHFR Case (15.1.2 Increase in Feedwater Flow)

Event	Time [s]
Feedwater flow begins to increase	0
Regulating bank begins to withdraw in response to a decrease in average RCS temperature	13
Reactor trips on high reactor power	66
Control rods begin to insert	68
High steam line superheat trip	68
Peak reactor power	69
Secondary System Isolation	76
DHRS actuation	78

Table 15.1-5: Increase in Feedwater Flow - Inputs (Limiting Minimum Critical Heat Flux Ratio Case)

Parameter	Nominal	Bias
Core power	160 MWt	+2%
Pressurizer pressure	1850 psia	+70psia
RCS flow rate	See Table 15.0-6 for range	Minimum - 1179 lbm/s
RCS average temperature	545 °F	+10 °F
SG pressure	500 psia	-35psia
Pressurizer level	60%	+8%
SG heat transfer	Nominal	None
Feedwater temperature	300 °F	-10 °F
Reactor pool temperature	40 °F - 200 °F	Minimum (40 °F)
DHRS heat transfer	Nominal	None
MTC	EOC	Most Negative
DTC	EOC	Least Negative

Table 15.1-6: Increase in Feedwater Flow (15.1.2) - Limiting Analysis Results

Acceptance Criteria	Limit	Analysis Value
Maximum RCS pressure	2310 psia	2002 psia
Maximum SG pressure	2310 psia	1491 psia
MCHFR	1.284	1.854

Table 15.1-7: Sequence of Events (15.1.3 Increase in Steam Flow)

Event	Time [s]
Steam flow begins to increase	0
Peak reactor power is reached	63
Reactor trips on high reactor power	65
Peak RCS pressure is reached	66
Low PZR pressure limit reached	123
Secondary system isolation	125
FWIVs and MSIVs fully close	132
High steam pressure limit is reached	1692
DHRS actuation	1694
Maximum SG pressure	1697

Table 15.1-8: Increase in Steam Flow - Inputs (Limiting Minimum Critical Heat Flux Ratio Case)

Parameter	Nominal	Bias
Core power	160 MWt	+2%
Pressurizer pressure	1850 psia	+70psia
RCS flow rate	See Table 15.0-6 for range	1174 lbm/s
RCS average temperature	545 °F	+10 °F
SG pressure	500 psia	+35psia
Pressurizer level	60%	+8%
SG heat transfer	Nominal	None
FW temperature	300 °F	None
Reactor pool temperature	65 °F - 110 °F	Minimum (65 °F)
DHRS heat transfer	Nominal	None
MTC	EOC	Most Negative
DTC	EOC	Most Negative

Table 15.1-9: Increase in Steam Flow (15.1.3) - Limiting Analysis Results

Acceptance Criteria	Limit	Analysis Value
Maximum RCS pressure	2310 psia	1981 psia
Maximum SG pressure	2310 psia	804 psia
MCHFR	1.284	1.881

Table 15.1-10: Sequence of Events (15.1.5 Steam Line Break, Limiting Reactor Pressure Vessel Pressure Case)

Event	Time [s]
SLB occurs	0
Peak reactor power reached	73
Hotwell inventory depleted - Feedwater terminated	1318
Low steam pressure limit reached (MSL 2)	1327
High PRZ pressure limit reached	1328
Reactor trips	1329
Secondary System Isolation	1329
DHRS actuated	1330
Peak RCS pressure reached	1334

Table 15.1-11: Sequence of Events (15.1.5 Steam Line Break, Limiting Minimum Critical Heat Flux Ratio Case)

Event	Time [s]
SLB occurs	0
High reactor power limit is reached	67
Peak reactor power reached	69
Reactor Trip	69
High pressurizer pressure limit is reached	71
Control rods fully inserted	72
Secondary System Isolation Signal (FWRV, FWIVs and MSIVs close signal)	73
DHRS Actuation	73
Peak RCS pressure reached	75
Peak MSS pressure reached	122

Table 15.1-12: Sequence of Events (15.1.5 Steam Line Break, Radiological Input Case for Iodine Spiking)

Event	Time [s]
SLB occurs	0
High power limit reached	75
Reactor trip	77
Low steam superheat limit reached	197
Secondary System Isolation actuation	205
FWIV and MSIV fully closed	212
Dryout of affected SG line	~700
High Steam Line Pressure actuation	1687
DHRS actuated	1689

Table 15.1-13: Steam Piping Failure - Inputs

Parameter	Nominal	Biases				
		RPV Pressure Limiting Case	SGS Pressure Limiting Case	MCHFR Limiting Case	Radiological Max Iodine Spiking Case	Radiological Max Mass Release Case
Core power (MWt)	160	-10%	+2%	+2%	-20%	-50%
Pressurizer pressure (psia)	1850	+70	+0	+70	+70	+70
Pressurizer Level (%)	60	+8	+0	+8	+8	+8
RCS flow rate	See Table 15.0-6 for range	Low (1140 lbm/s)	Low (1175 lbm/s)	Low (1155 lbm/s)	Low (1085 lbm/s)	Low (895 lbm/s)
RCS average temperature (°F)	545	+10	+10	+10	+10	+10
SG pressure (psia)	500	+35	+35	+35	+35	+35
Core Exposure	MOC (Nominal MTC and DTC)	MOC (Nominal MTC and DTC)	BOC (Most Positive MTC and DTC)	BOC (Most Positive MTC and DTC)	BOC (Most Positive MTC and DTC)	MOC (Nominal MTC and DTC)

Table 15.1-14: Steam Piping Failure - Inputs to Radiological Analysis

Parameter	Units	SLB with Maximum Mass Release	SLB with Maximum Spiking Time
Integrated mass through break - pre-trip	lbm	144,818	3077
Integrated mass through break - from trip to SG line empty	lbm	N/A	11,706
Integrated secondary flow - intact steam line - pre-trip	lbm	N/A	5731
Integrated secondary flow - intact steam line - from trip to isolation	lbm	N/A	4228

Table 15.1-15: Steam Piping Failure (15.1.5) - Limiting Analysis Results

Acceptance Criteria	Limit	Analysis Value
Maximum RCS pressure	2310 psia	2081 psia
Maximum SG pressure	2310 psia	1495 psia
MCHFR	1.284	1.866

Table 15.1-16: Loss of Containment Vacuum/Containment Flooding - Inputs (Limiting Minimum Critical Heat Flux Ratio Case)

Parameter	Nominal	Bias
Core power	160 MWt	+2%
Initial Pressurizer pressure	1850 psia	+70psia
Initial Pressurizer Level	60%	None
RCS flow rate	See Table 15.0-6 for range	Minimum - 1166 lbm/s
RCS average temperature	545 °F	+10 °F
Reactor pool temperature	100 °F	+100 °F
MTC	EOC	Most Negative
DTC	EOC	Most Negative

Table 15.1-17: Loss of Containment Vacuum/Containment Flooding (15.1.6) - Limiting Analysis Results

Acceptance Criteria	Limit	Analysis Value
Maximum RCS pressure	2310 psia	1937 psia
Maximum SG pressure	2310 psia	1426 psia
MCHFR	1.284	2.66

Table 15.1-18: Sequence of Events (15.1.6 Loss of Containment Vacuum/Containment Flooding) MCHFR Limiting Case

Event	Time [s]
Malfunction that causes loss of containment vacuum/flooding	0
RCCW tank emptied	364
Maximum RCS pressure	695
High Containment pressure limit reached - Reactor trip	781
Reactor trip - Control rod insertion begins	783
Secondary system isolation valves fully closed	790
High steam line pressure limit reached	790
DHRS actuation valves fully open	822
Maximum SGS pressure	871

Table 15.1-19: Sequence of Events (15.1.5 Steam Line Break, Limiting Steam Generator Pressure Case)

Event	Time [s]
SLB occurs	0
Low steam pressure limit is reached	1
Reactor trip actuated	3
Secondary System Isolation actuated	3
Control Rods Inserted	5
DHRS actuation	19
Peak SGS pressure reached	66

Figure 15.1-1: Feedwater Temperature (15.1.1 Decrease in Feedwater Temperature)

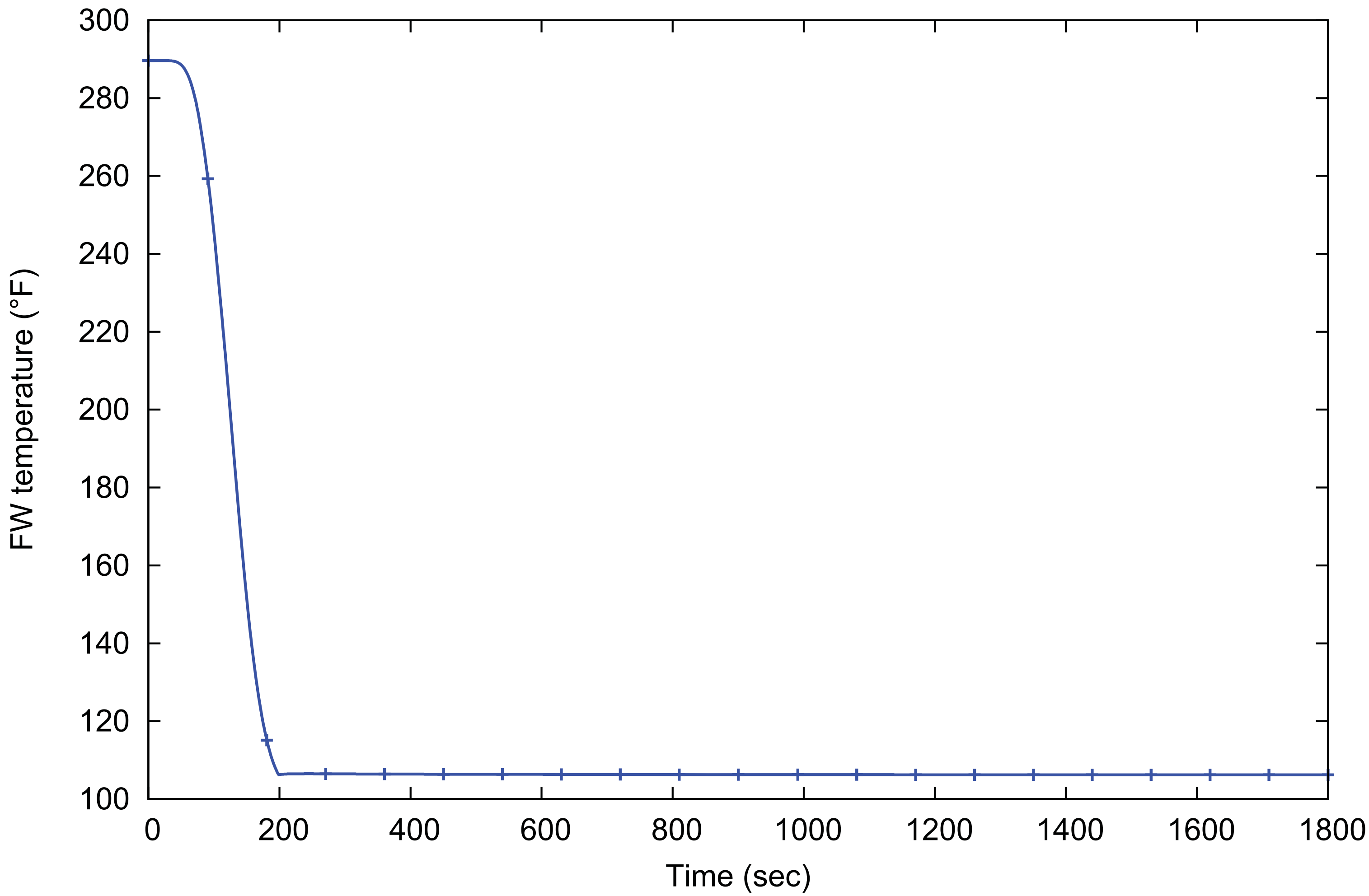


Figure 15.1-2: Core Inlet Temperature (15.1.1 Decrease in Feedwater Temperature)

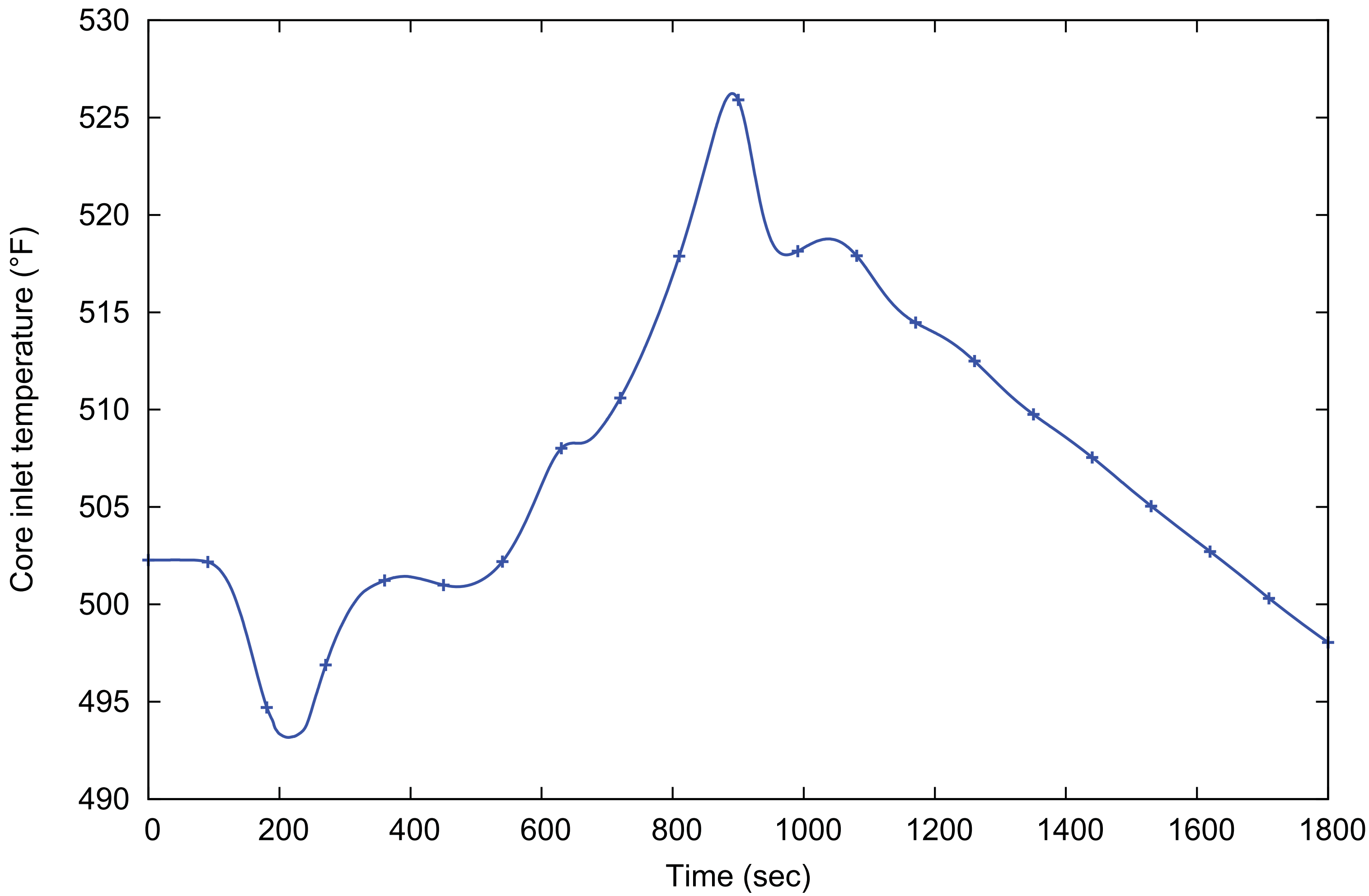


Figure 15.1-3: Core Inlet Density (15.1.1 Decrease in Feedwater Temperature)

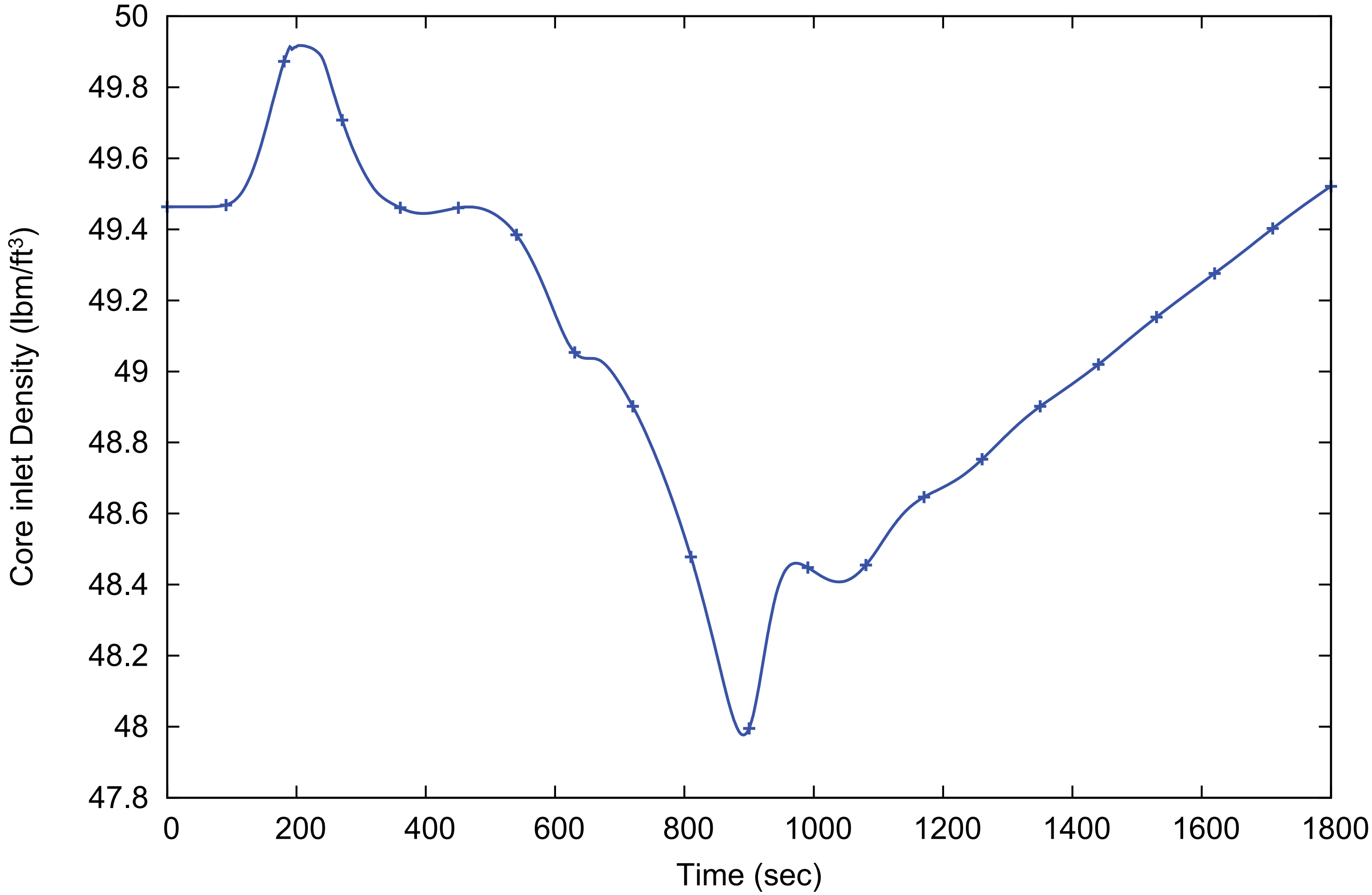


Figure 15.1-4: Total Reactivity - Entire Transient (15.1.1 Decrease in Feedwater Temperature)

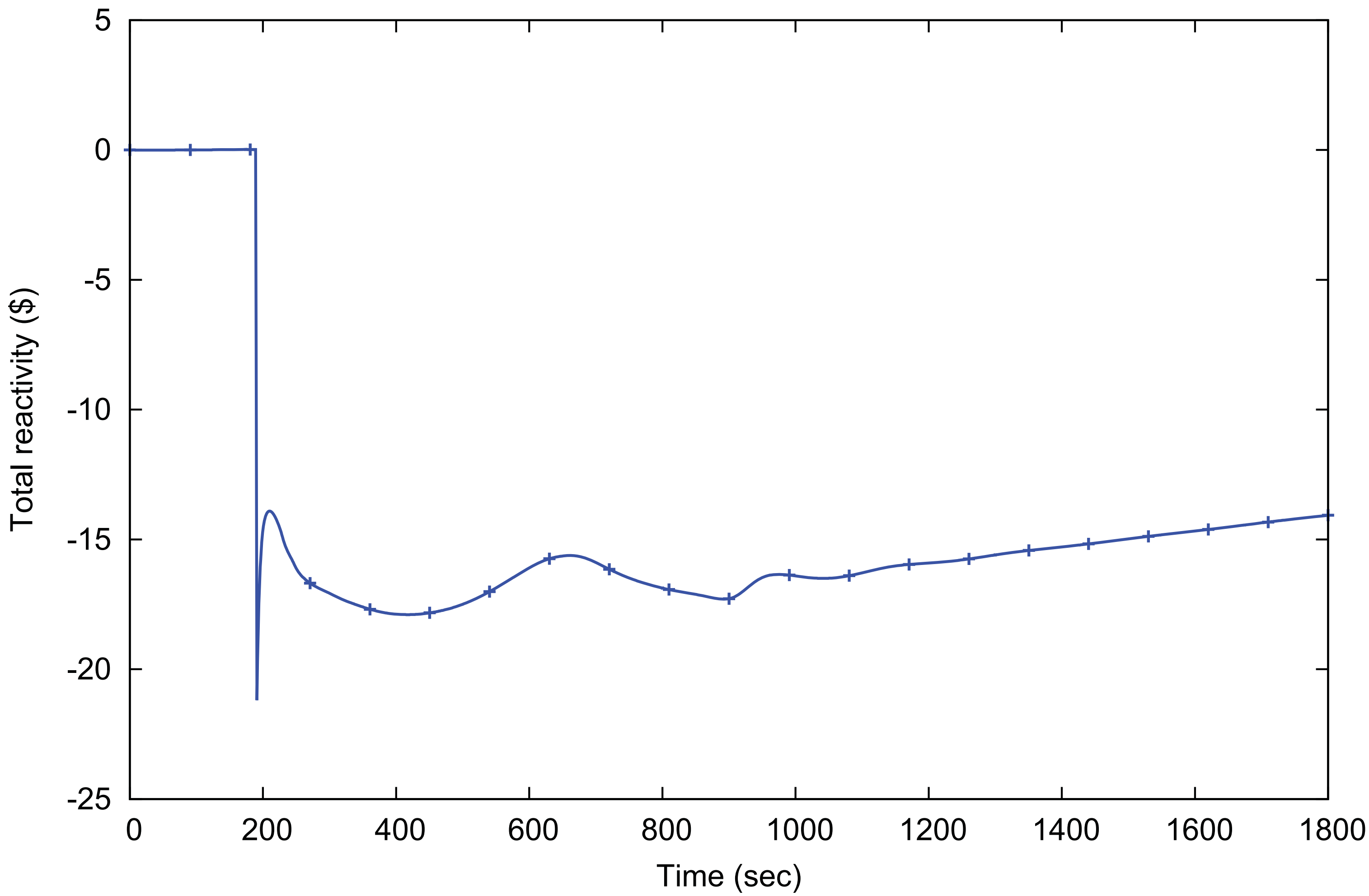


Figure 15.1-5: Reactor Power (15.1.1 Decrease in Feedwater Temperature)

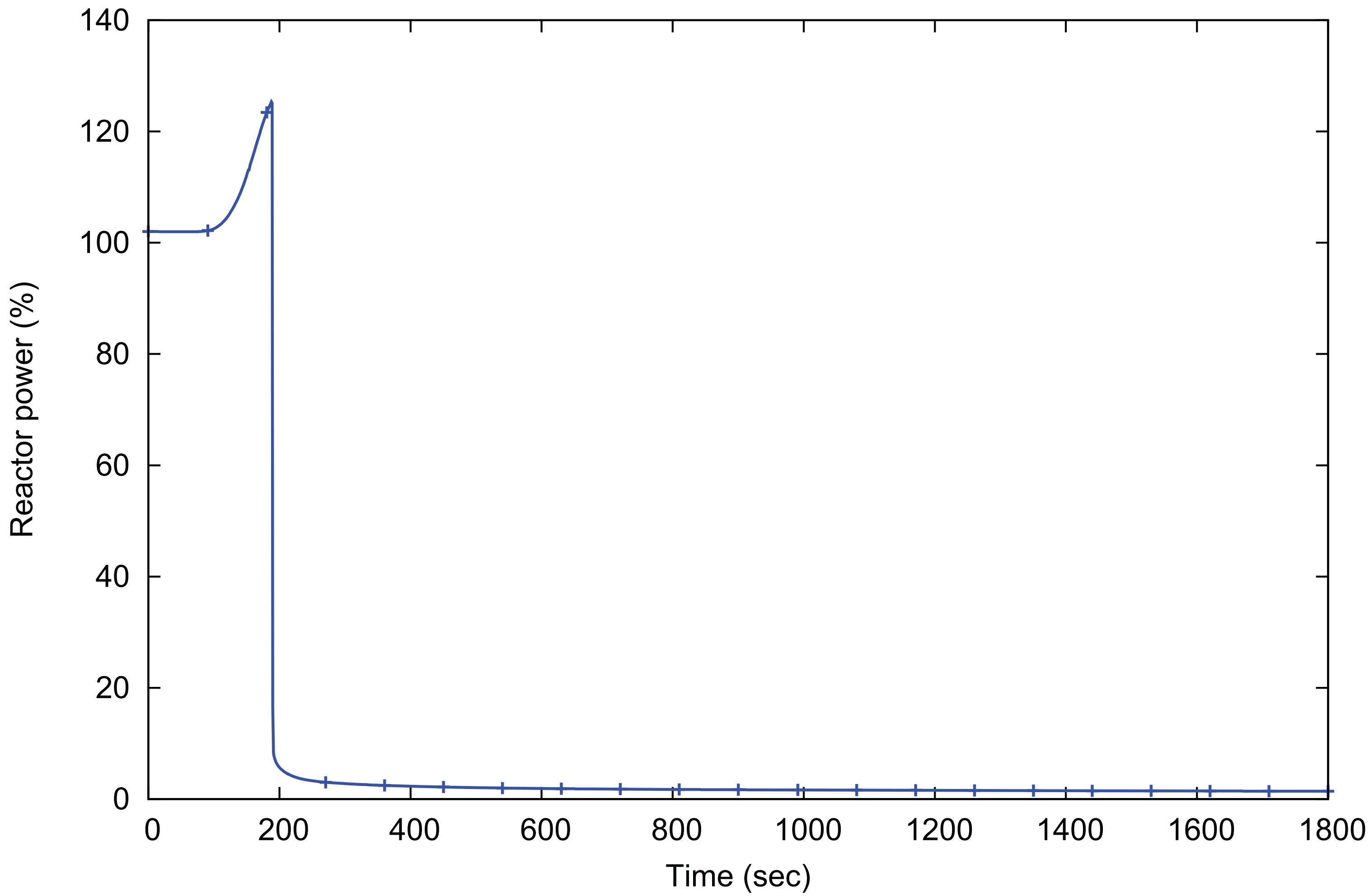


Figure 15.1-6: Reactor Coolant System Pressure (15.1.1 Decrease in Feedwater Temperature)

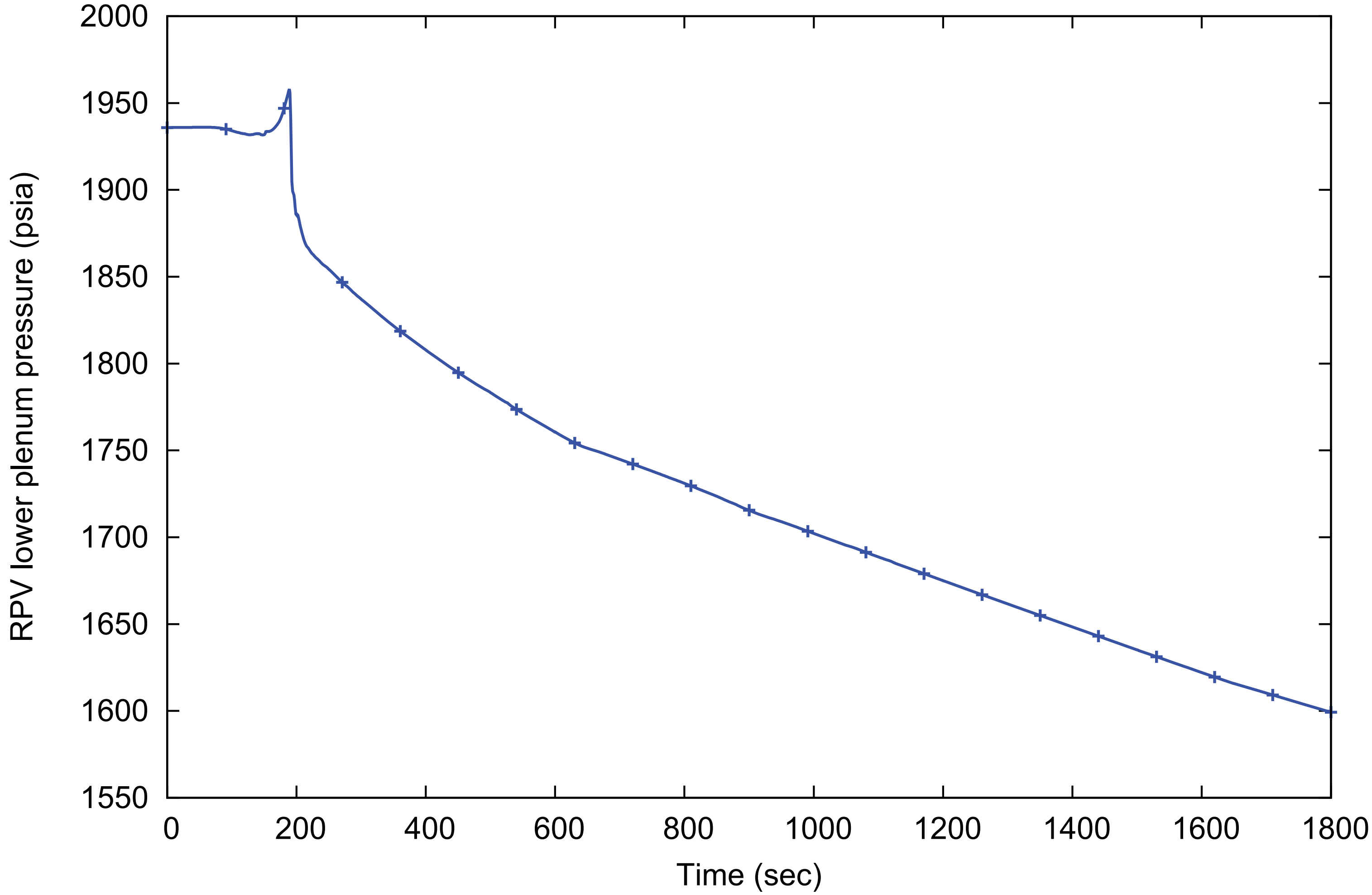


Figure 15.1-7: Core Outlet Temperature (15.1.1 Decrease in Feedwater Temperature)

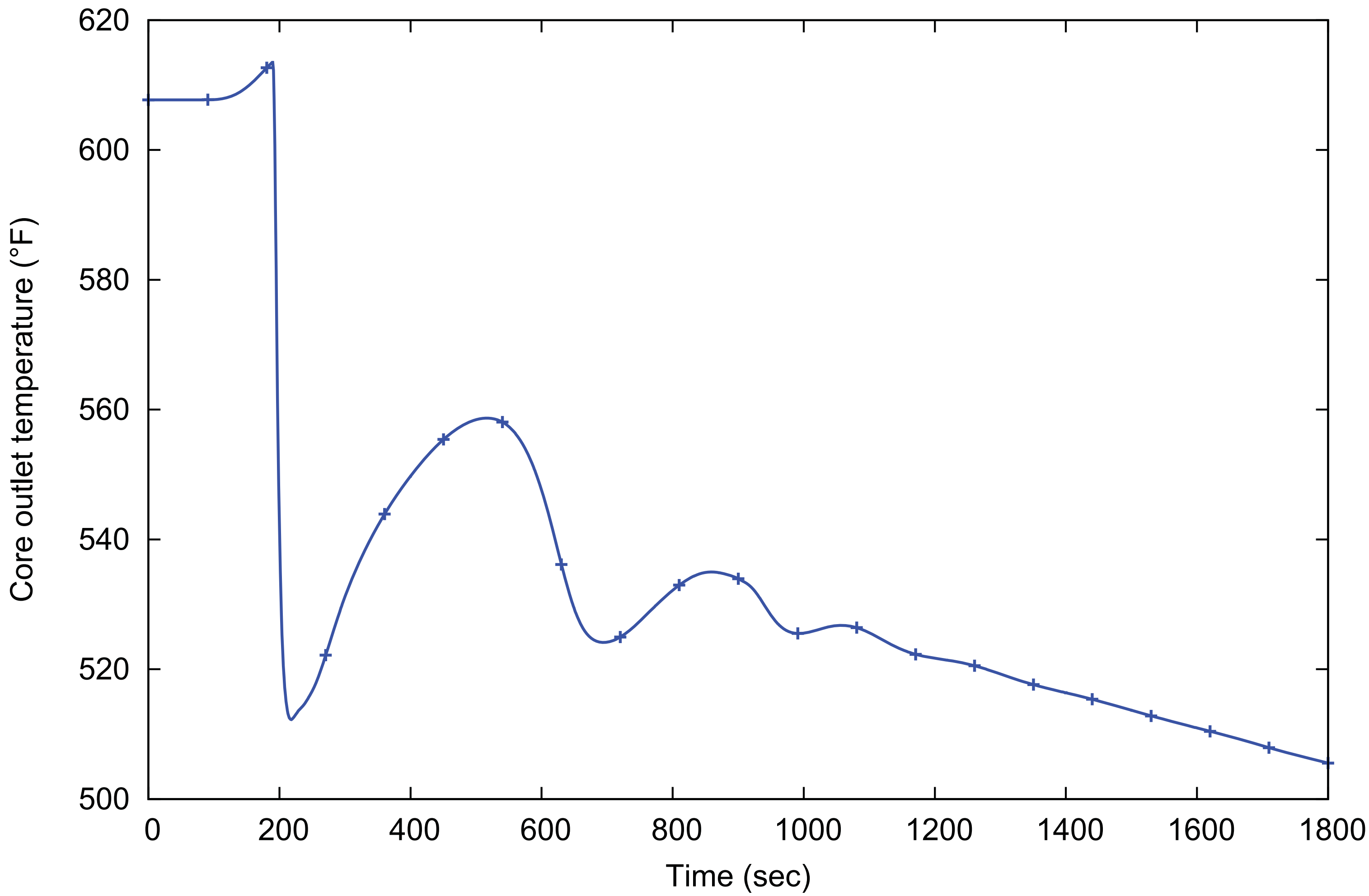


Figure 15.1-8: Main Steam System Pressure (15.1.1 Decrease in Feedwater Temperature)

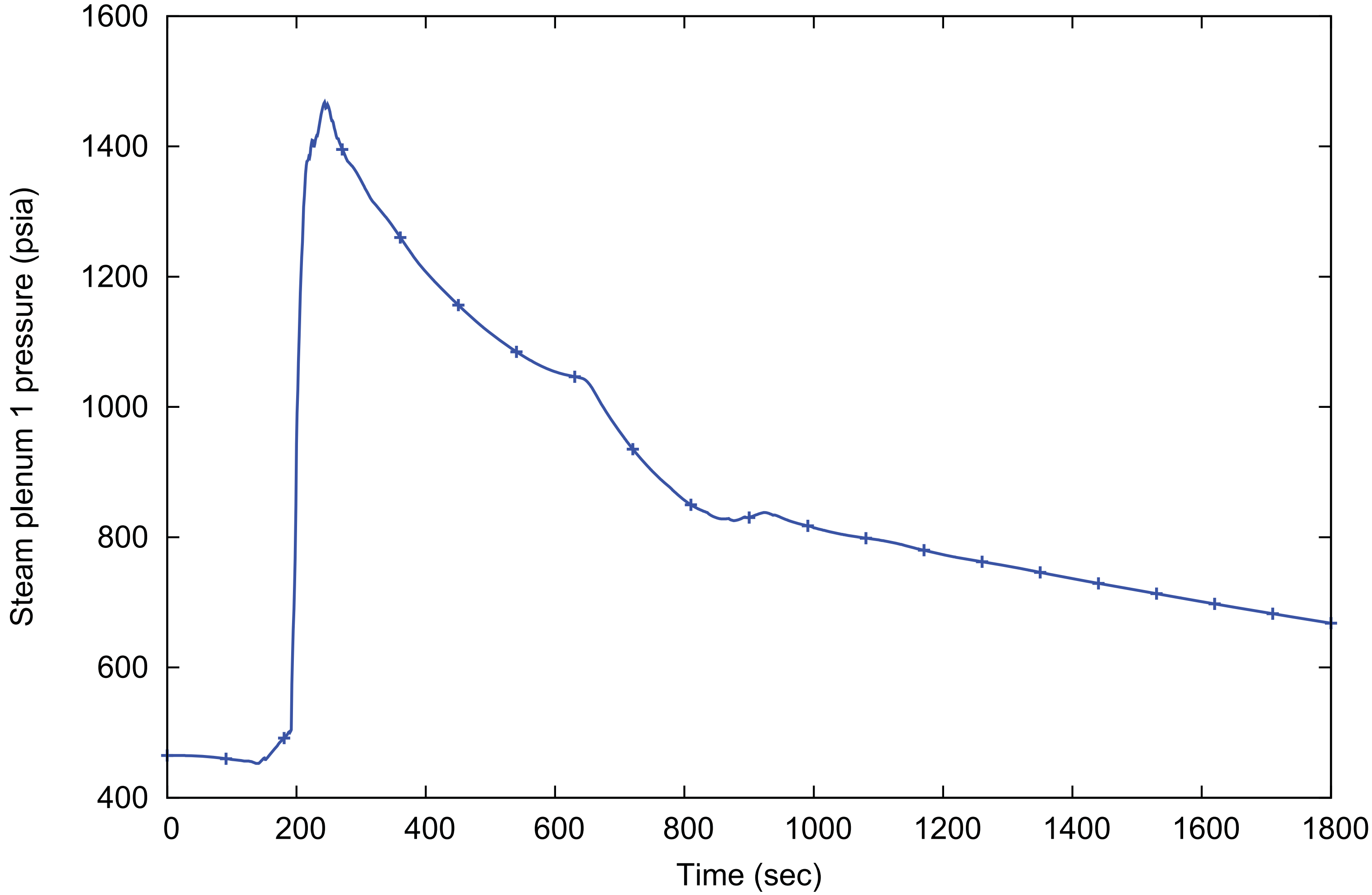


Figure 15.1-9: Critical Heat Flux Ratio (15.1.1 Decrease in Feedwater Temperature)

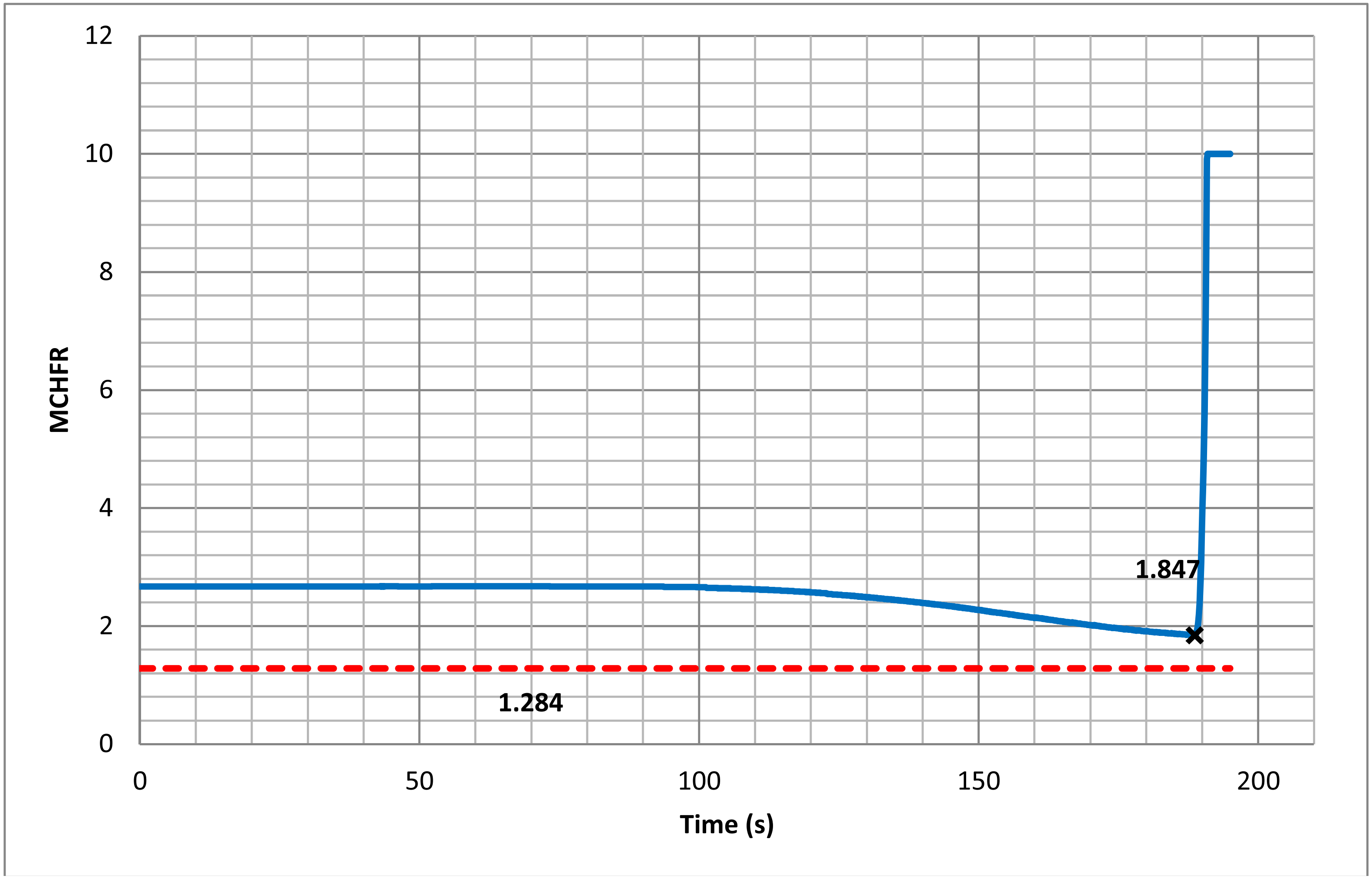


Figure 15.1-10: Reactor Coolant System Flow Rate (15.1.1 Decrease in Feedwater Temperature)

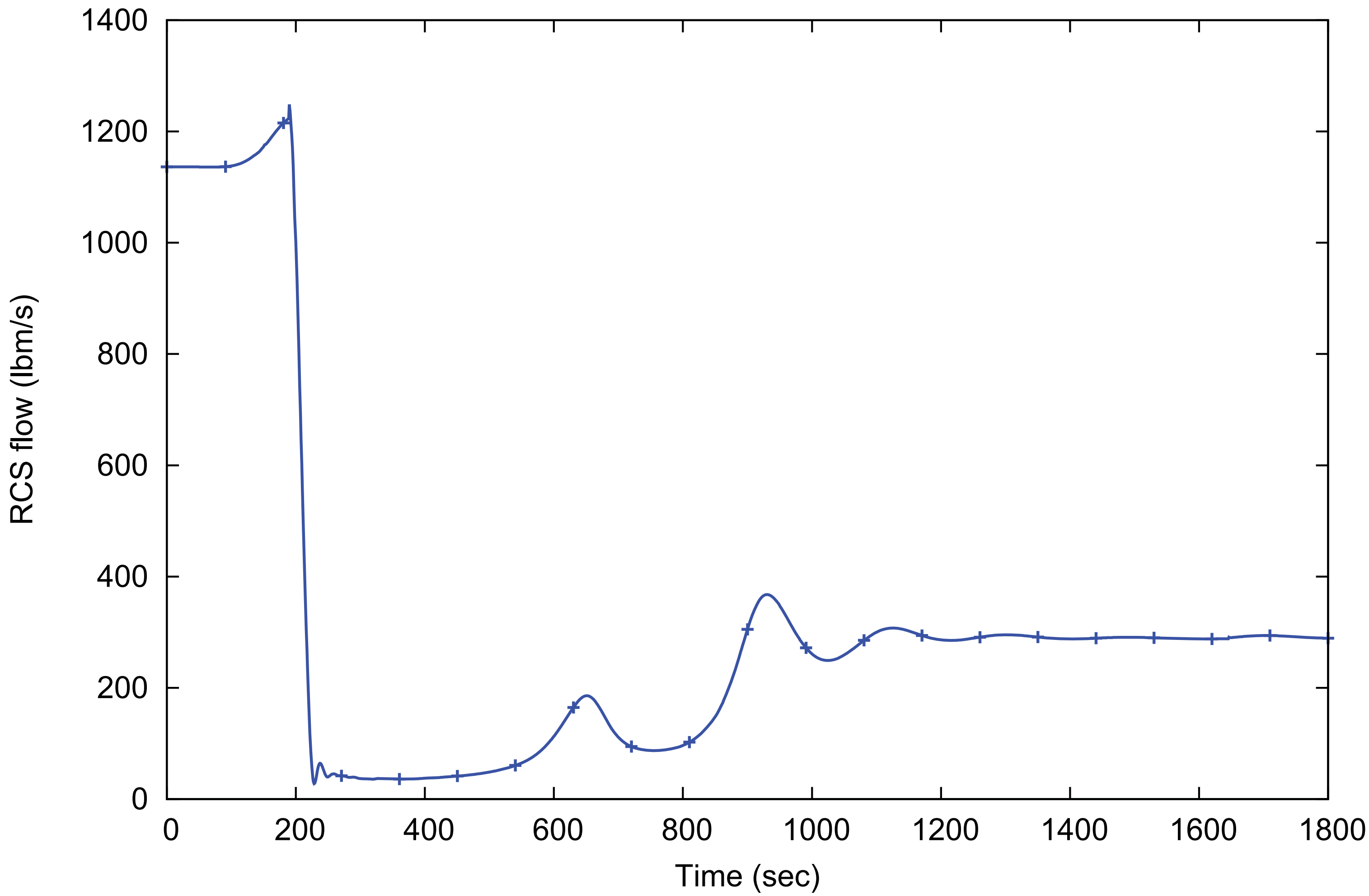


Figure 15.1-11: Feedwater Flow Rate (15.1.2 Increase in Feedwater Flow)

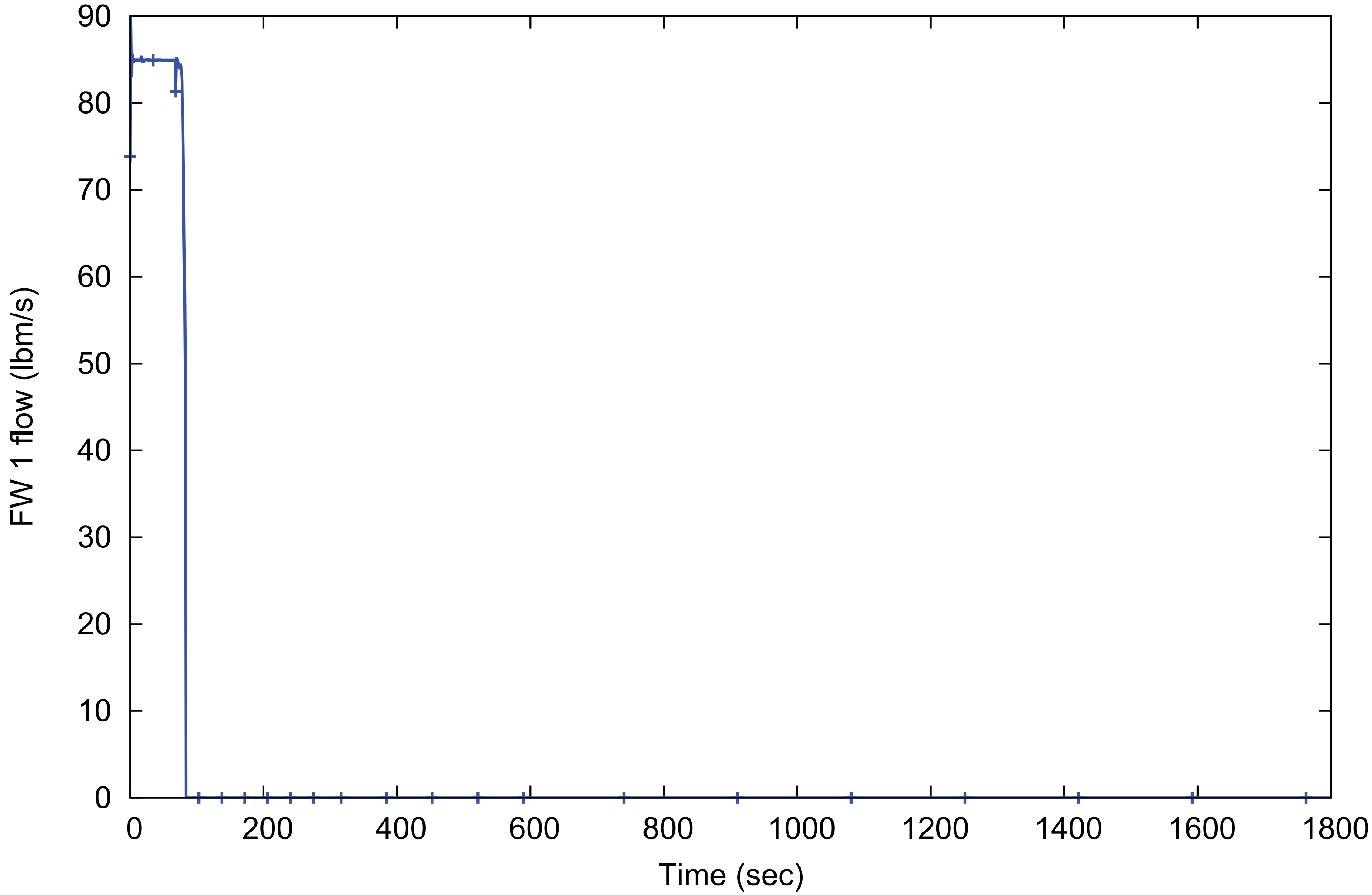


Figure 15.1-12: Core Inlet Temperature (15.1.2 Increase in Feedwater Flow)

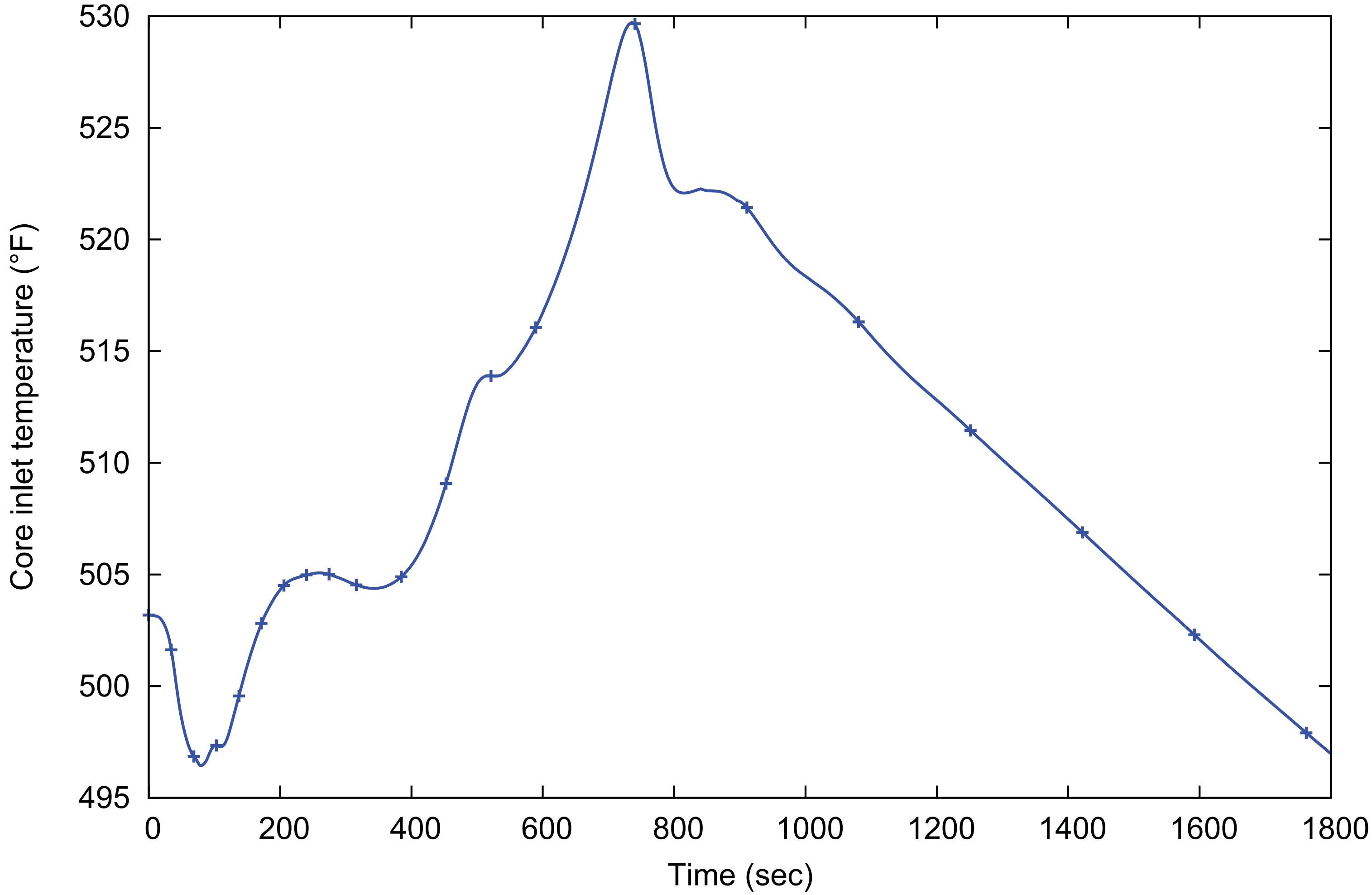


Figure 15.1-13: Core Inlet Density (15.1.2 Increase in Feedwater Flow)

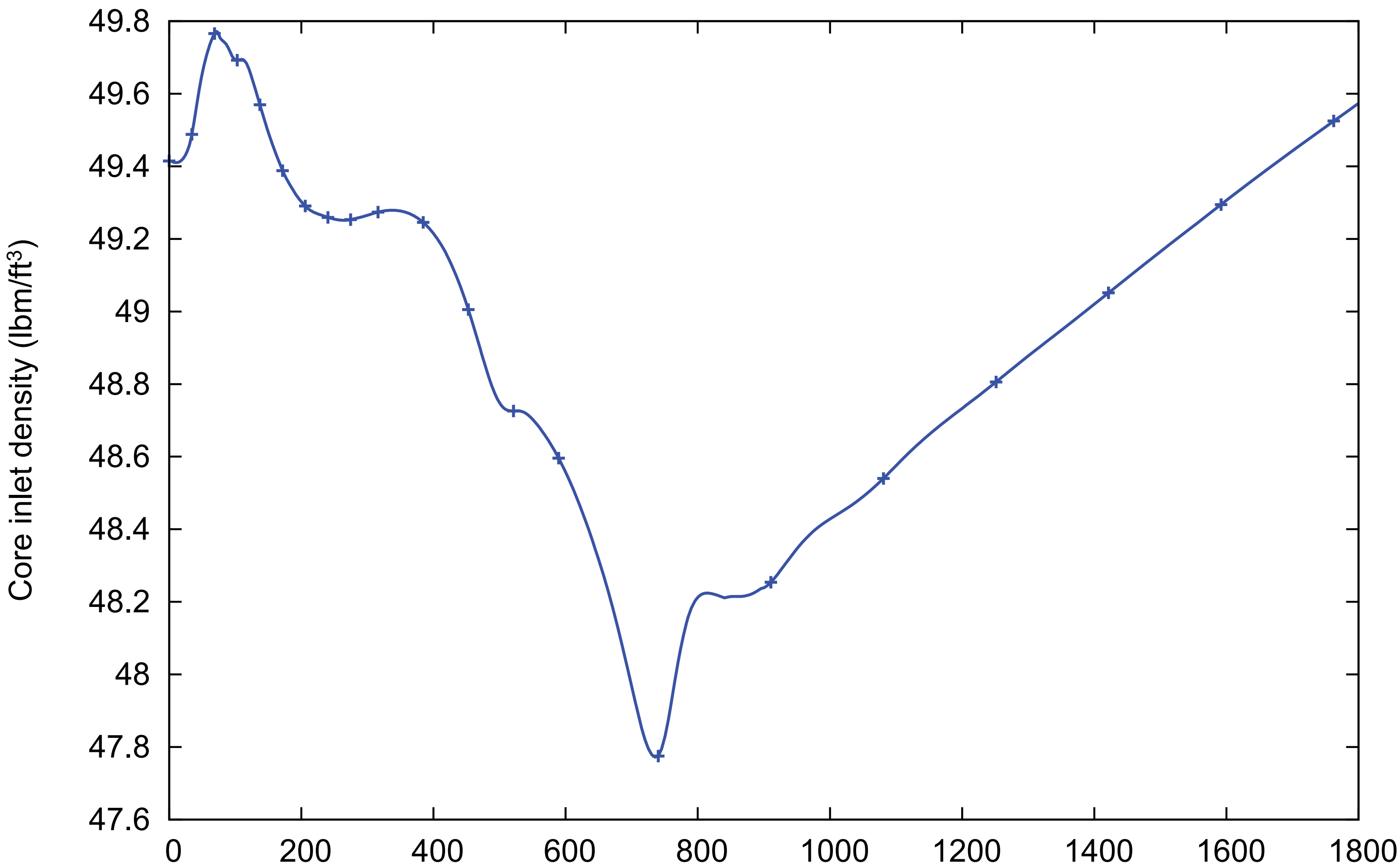


Figure 15.1-14: Total Core Reactivity (15.1.2 Increase in Feedwater Flow)

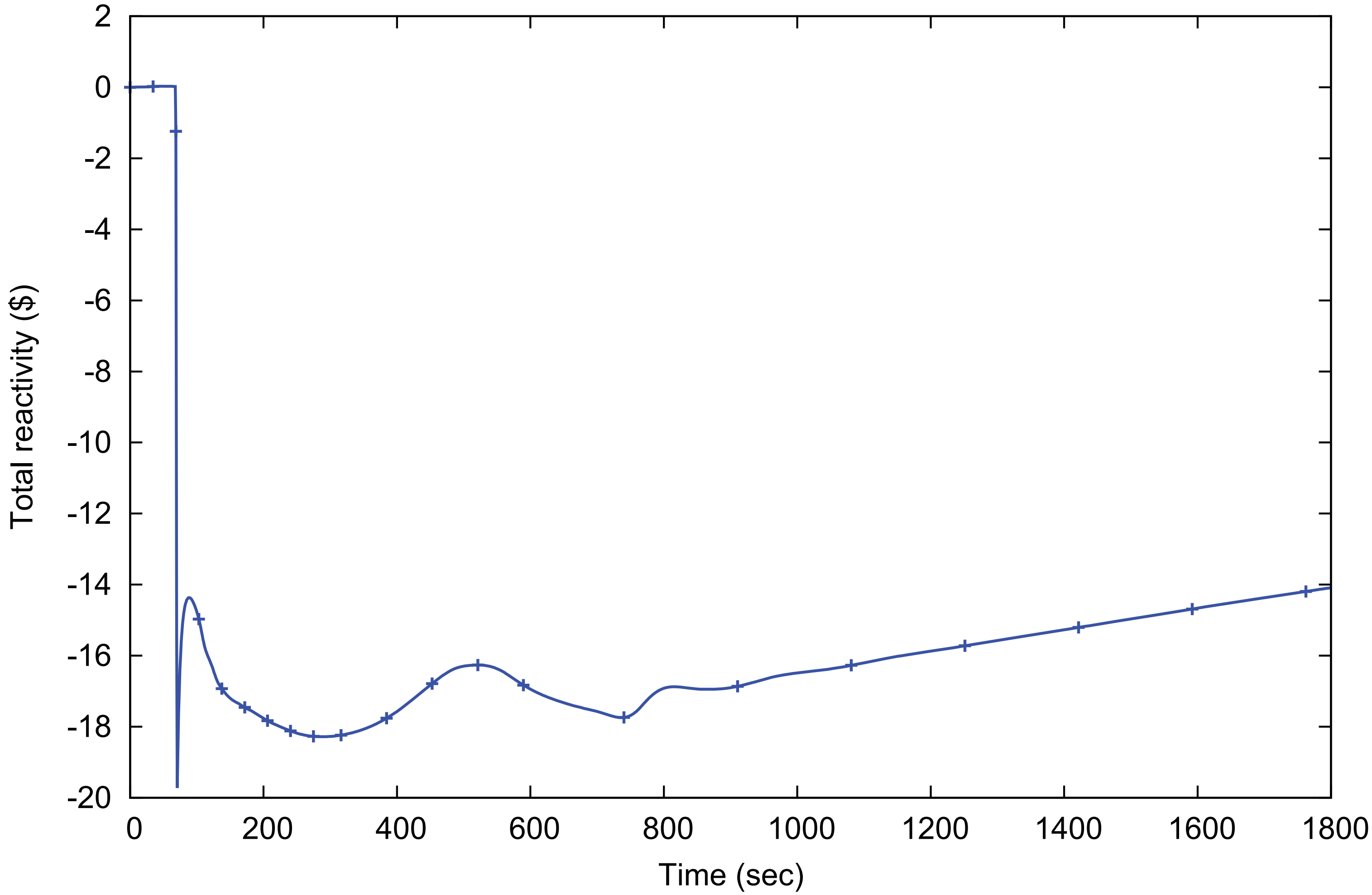


Figure 15.1-15: Reactor Power (15.1.2 Increase in Feedwater Flow)

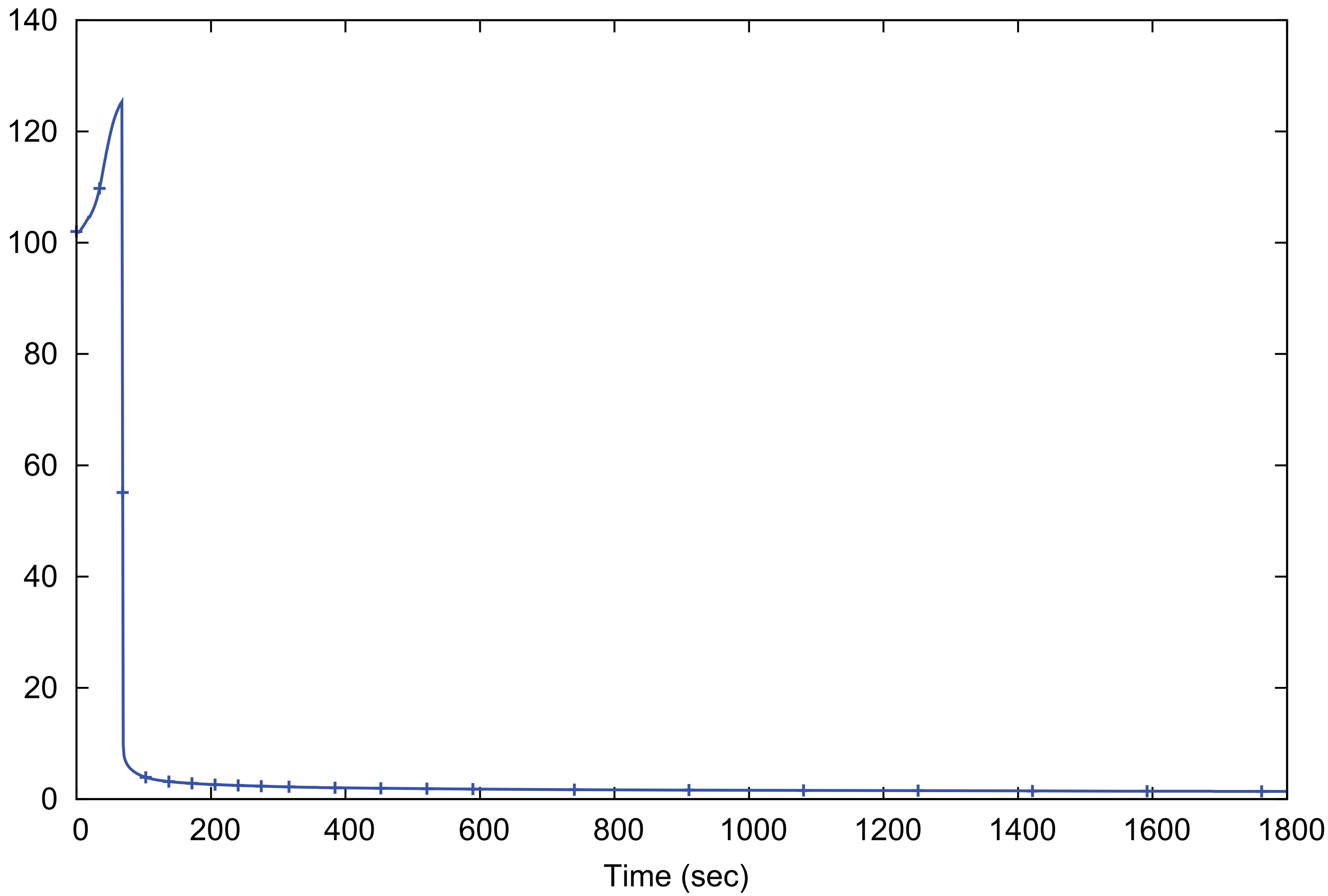


Figure 15.1-16: Reactor Coolant System Pressure (15.1.2 Increase in Feedwater Flow)

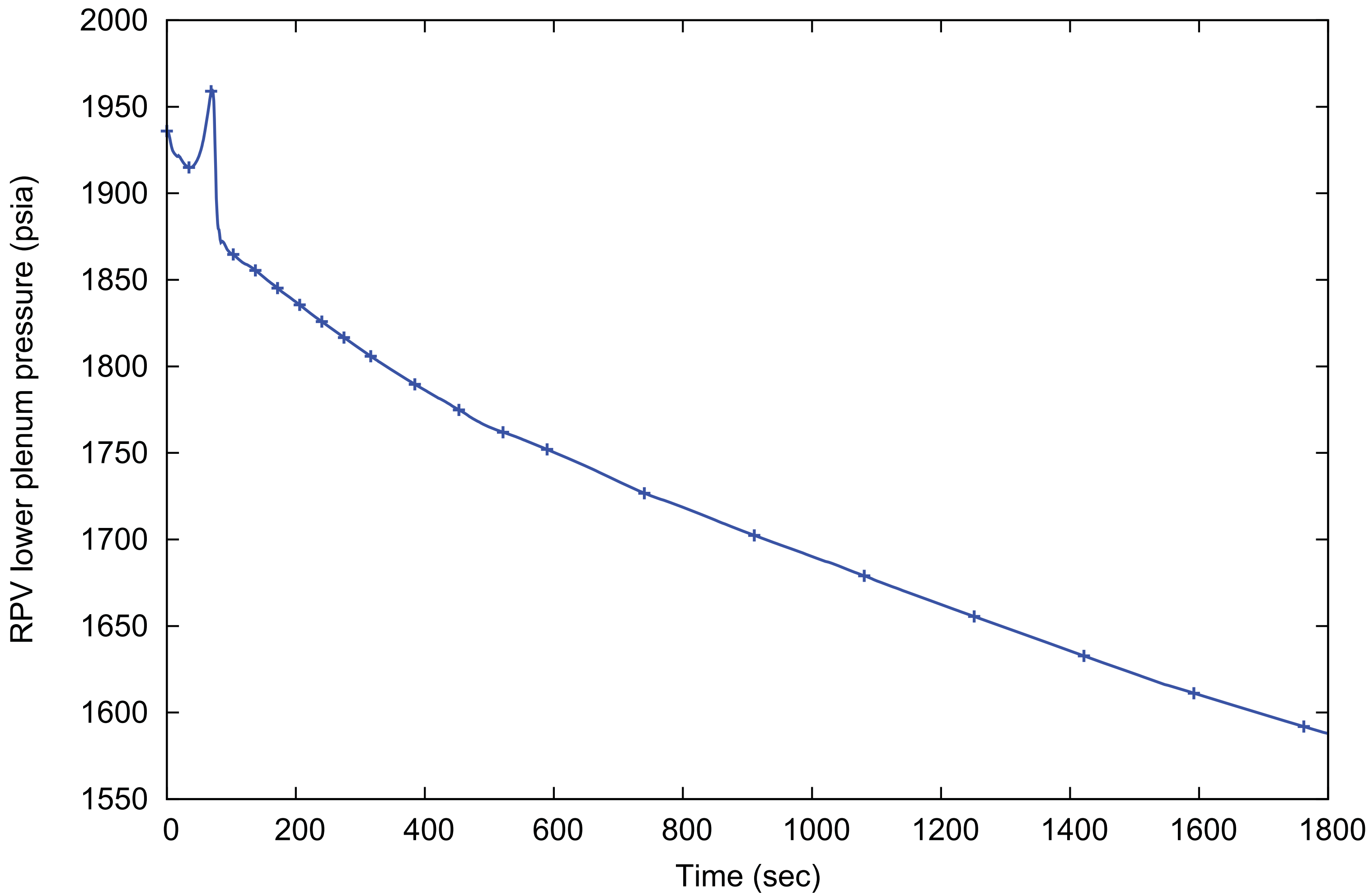


Figure 15.1-17: Core Outlet Temperature (15.1.2 Increase in Feedwater Flow)

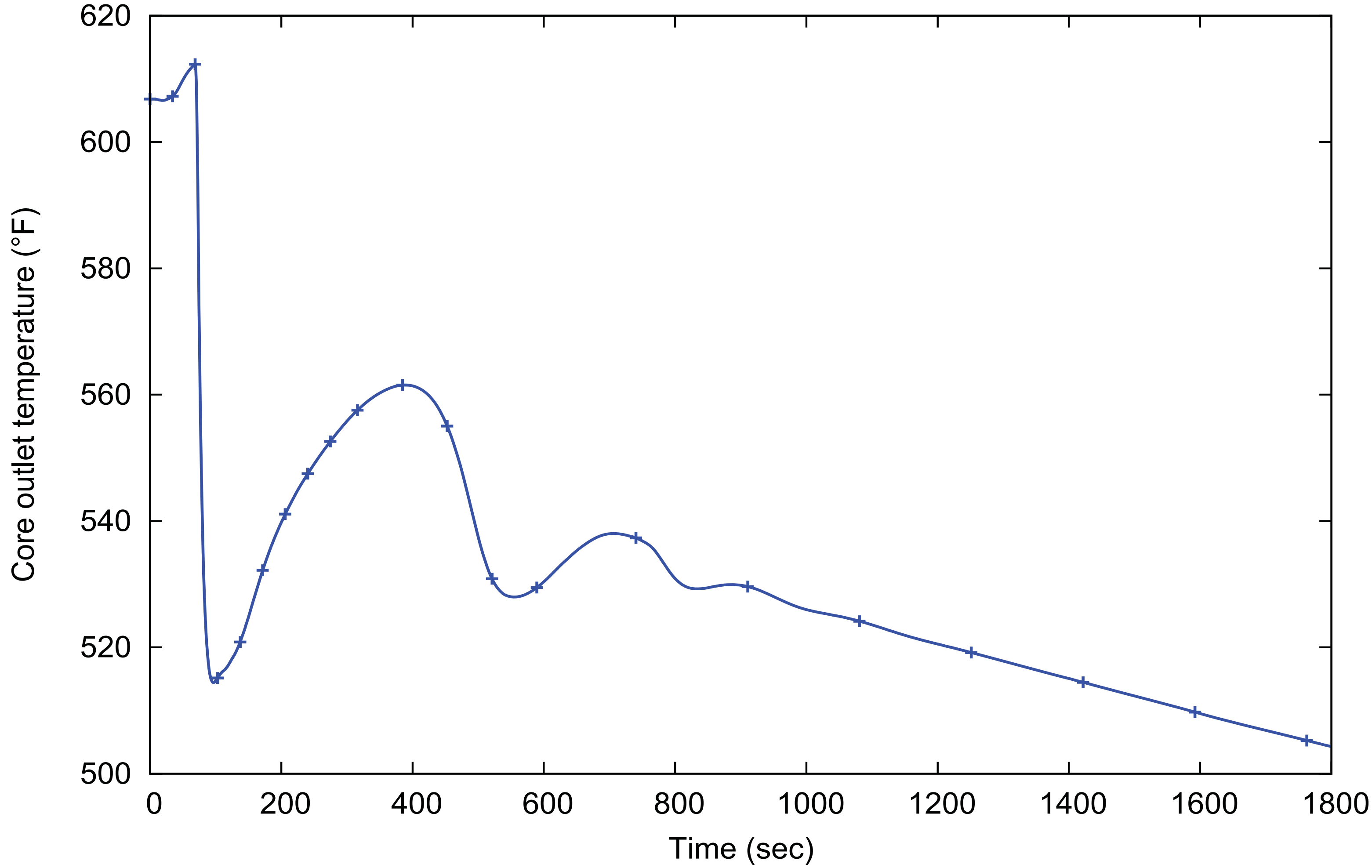


Figure 15.1-18: Main Steam System Pressure (15.1.2 Increase in Feedwater Flow)

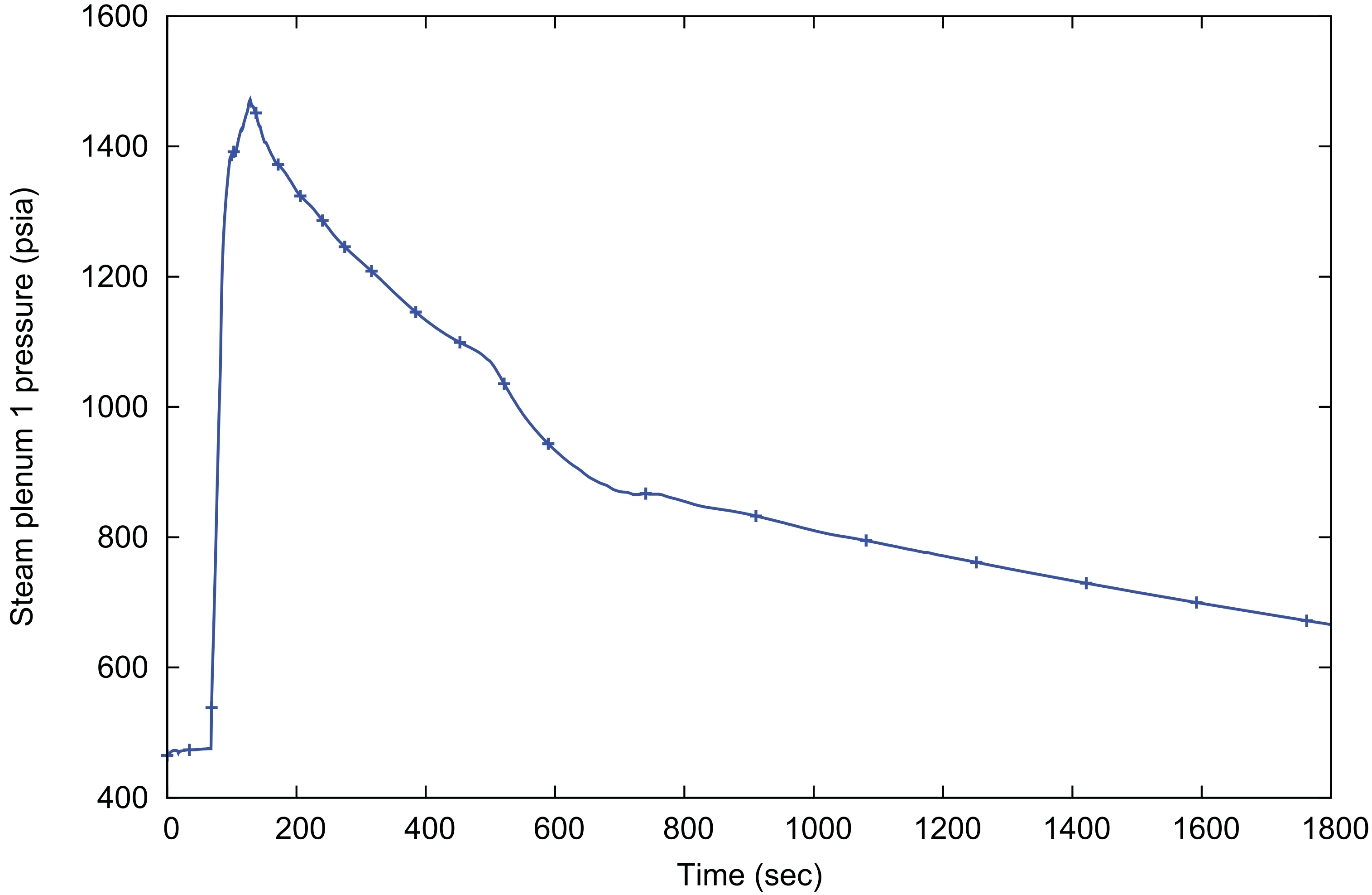


Figure 15.1-19: Critical Heat Flux Ratio (15.1.2 Increase in Feedwater Flow)

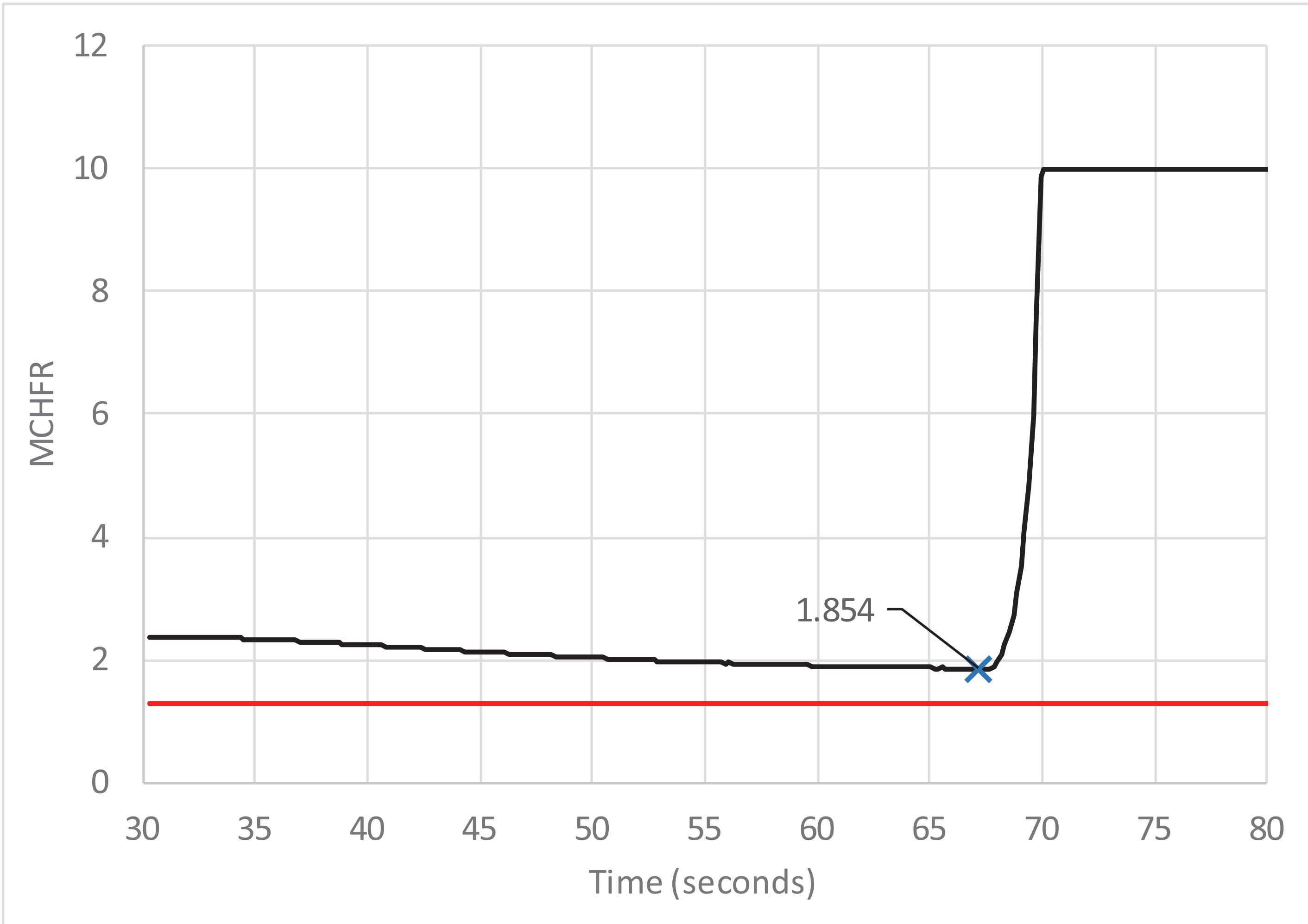


Figure 15.1-20: Reactor Coolant System Flow (15.1.2 Increase in Feedwater Flow)

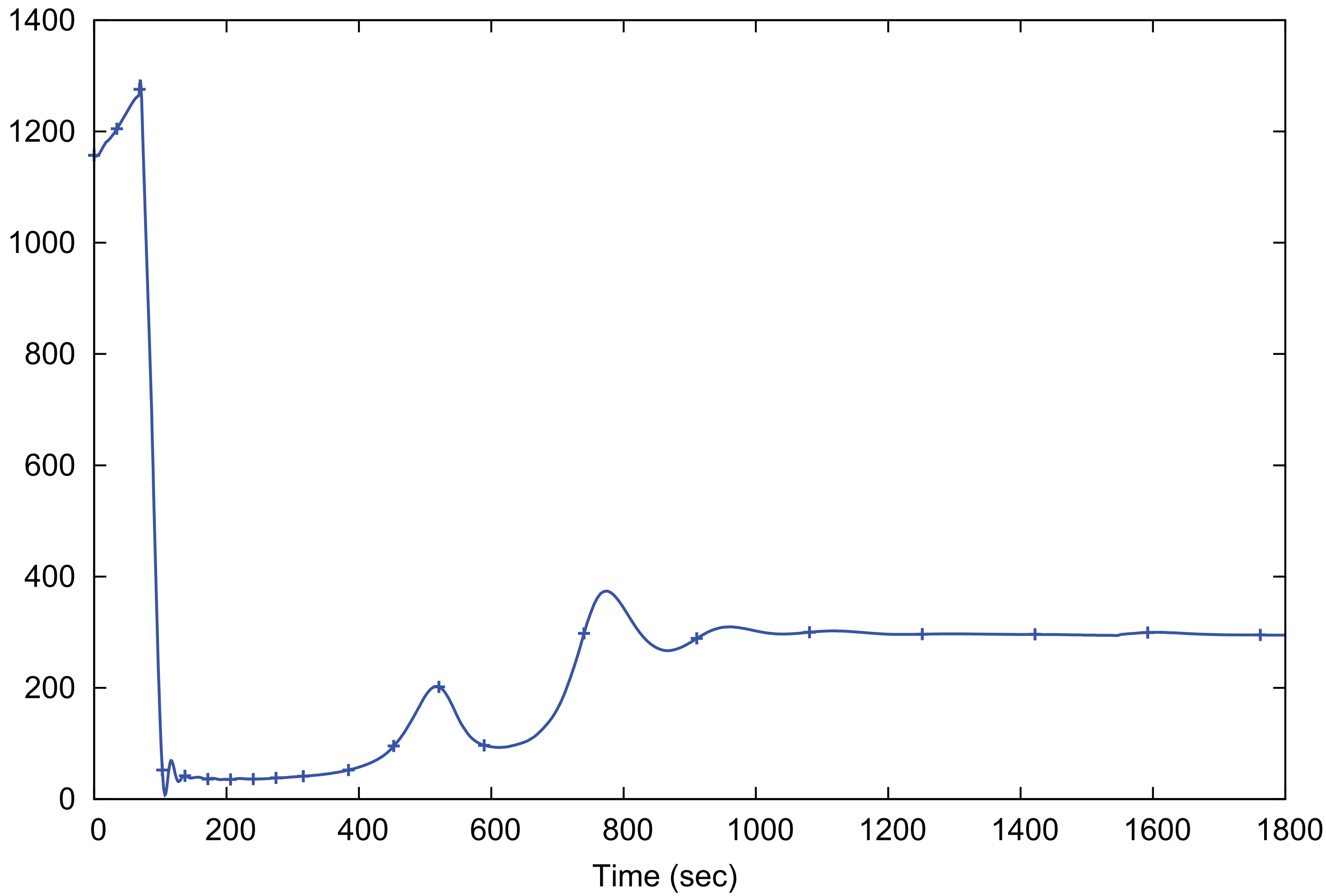


Figure 15.1-21: Steam Generator Level (15.1.2 Increase in Feedwater Flow)

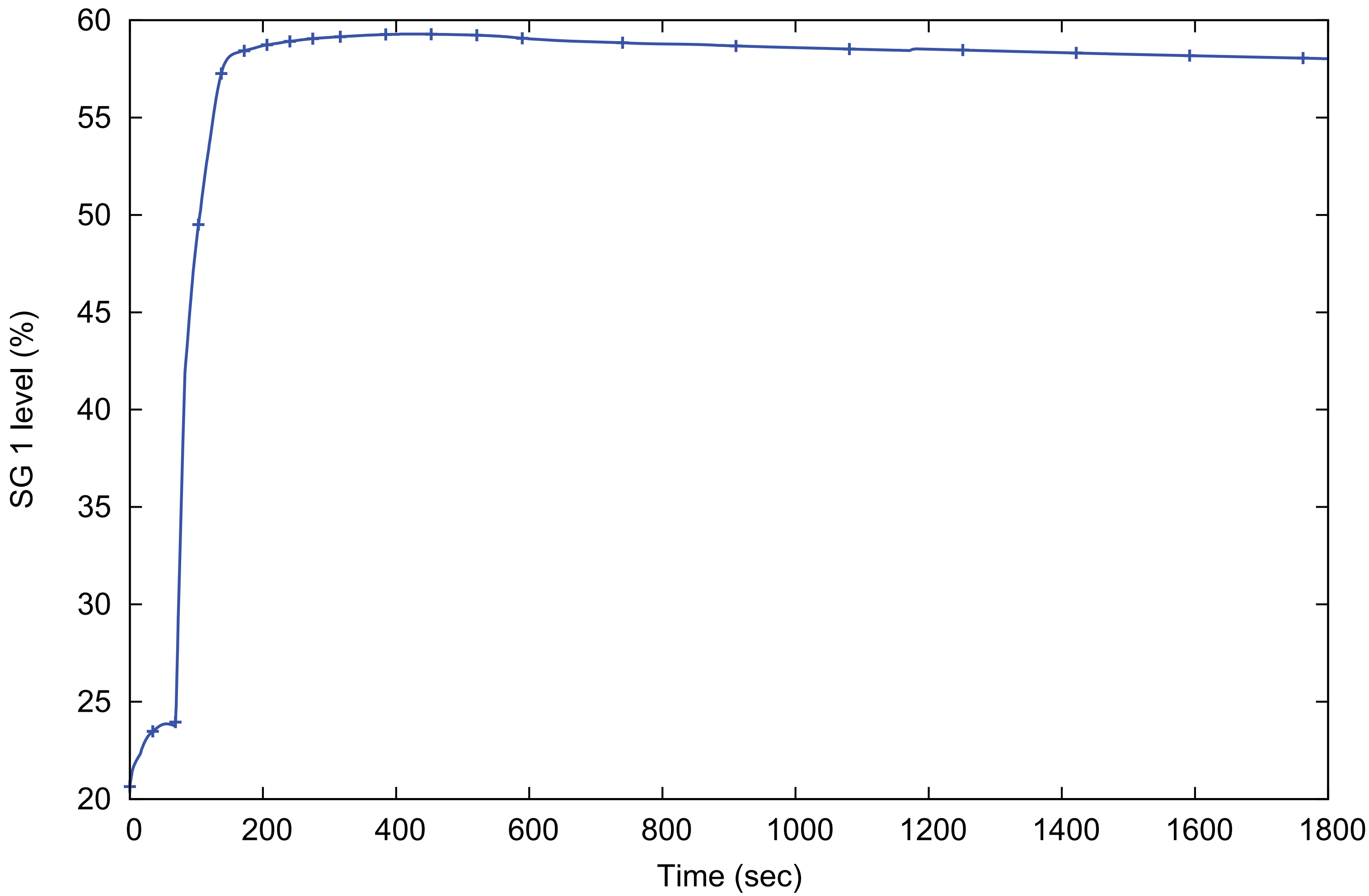


Figure 15.1-22: Steam Flow (15.1.3 Increase in Steam Flow)

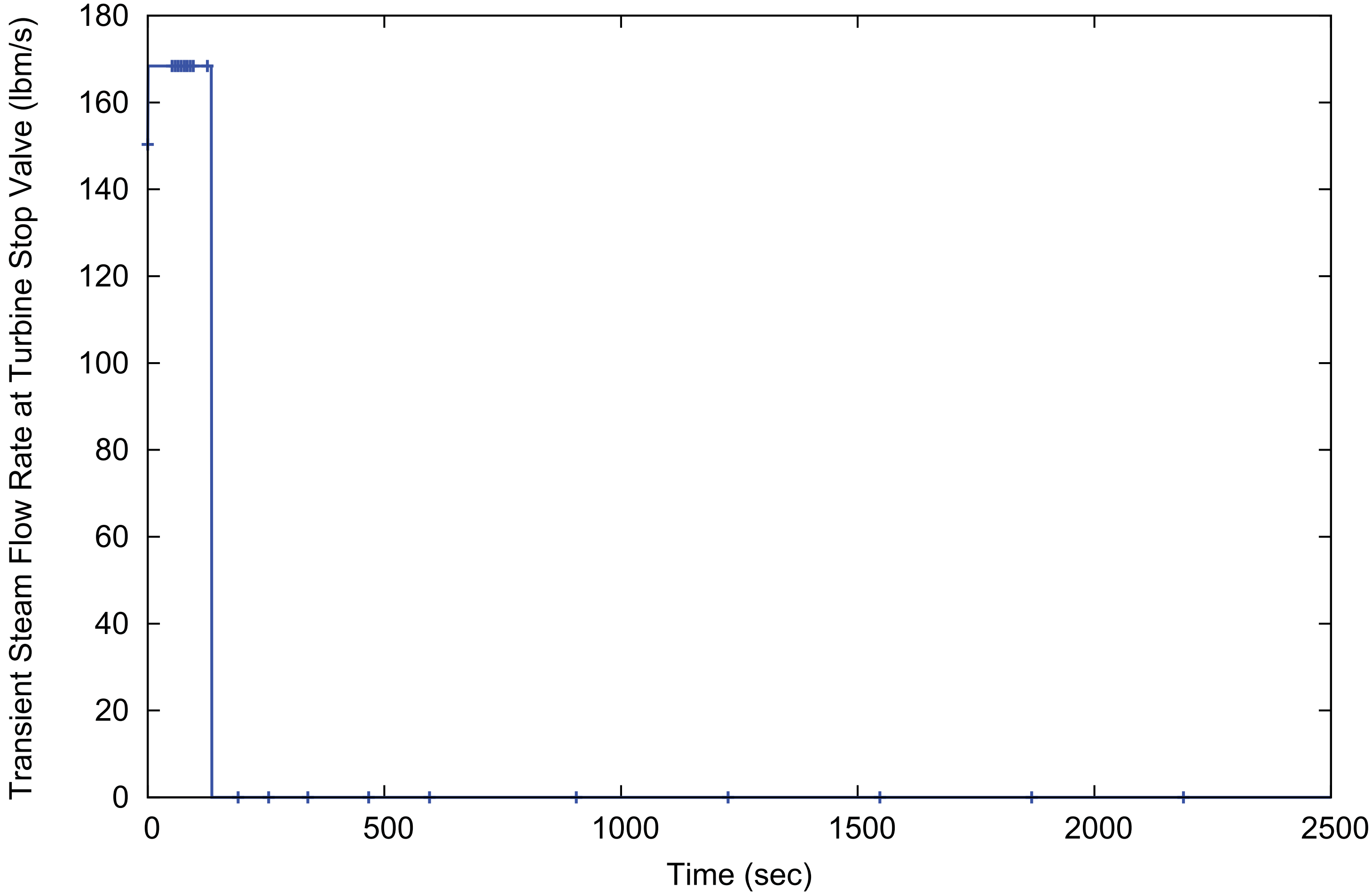


Figure 15.1-23: Core Inlet Temperature (15.1.3 Increase in Steam Flow)

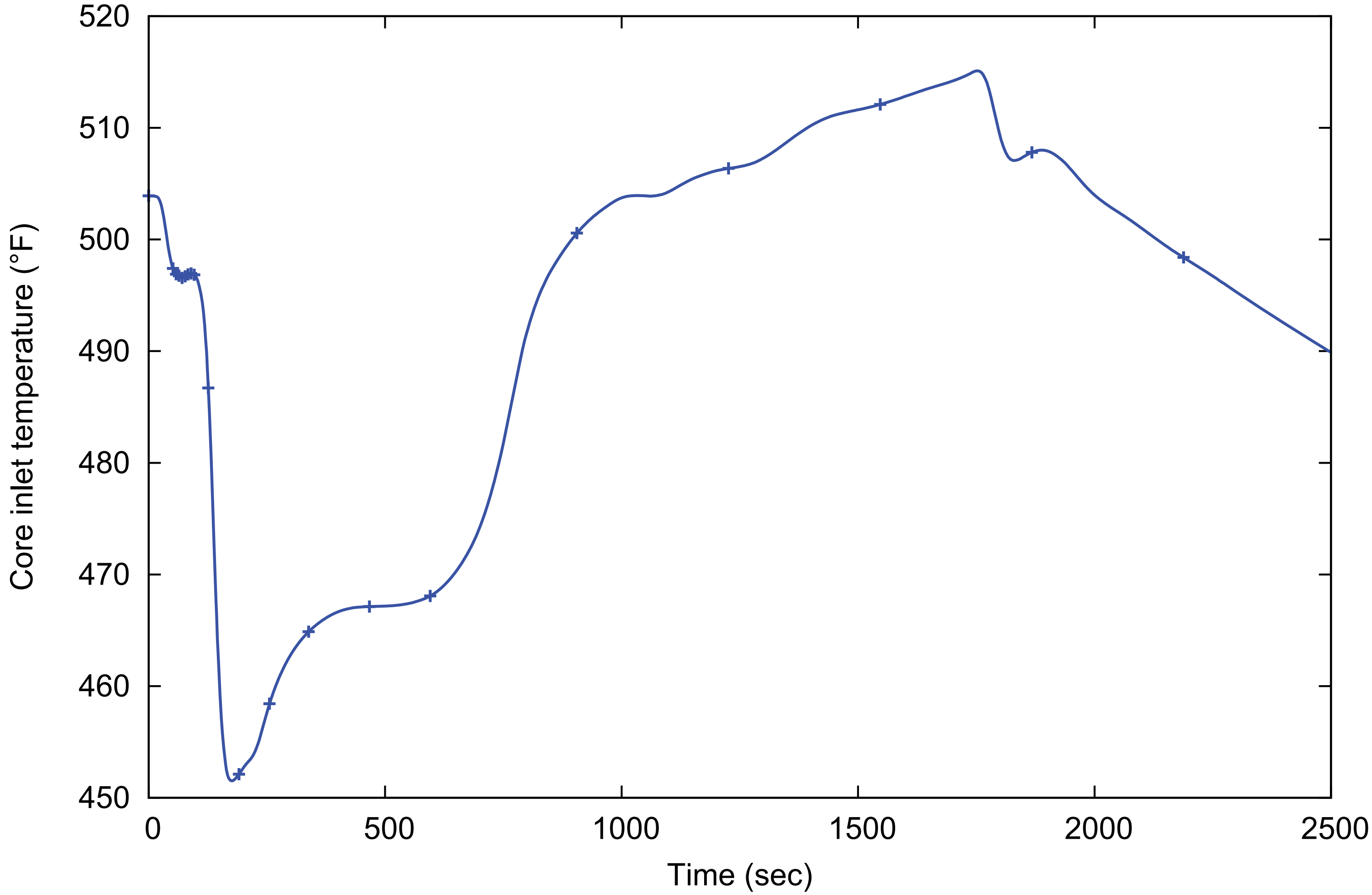


Figure 15.1-24: Core Inlet Density (15.1.3 Increase in Steam Flow)

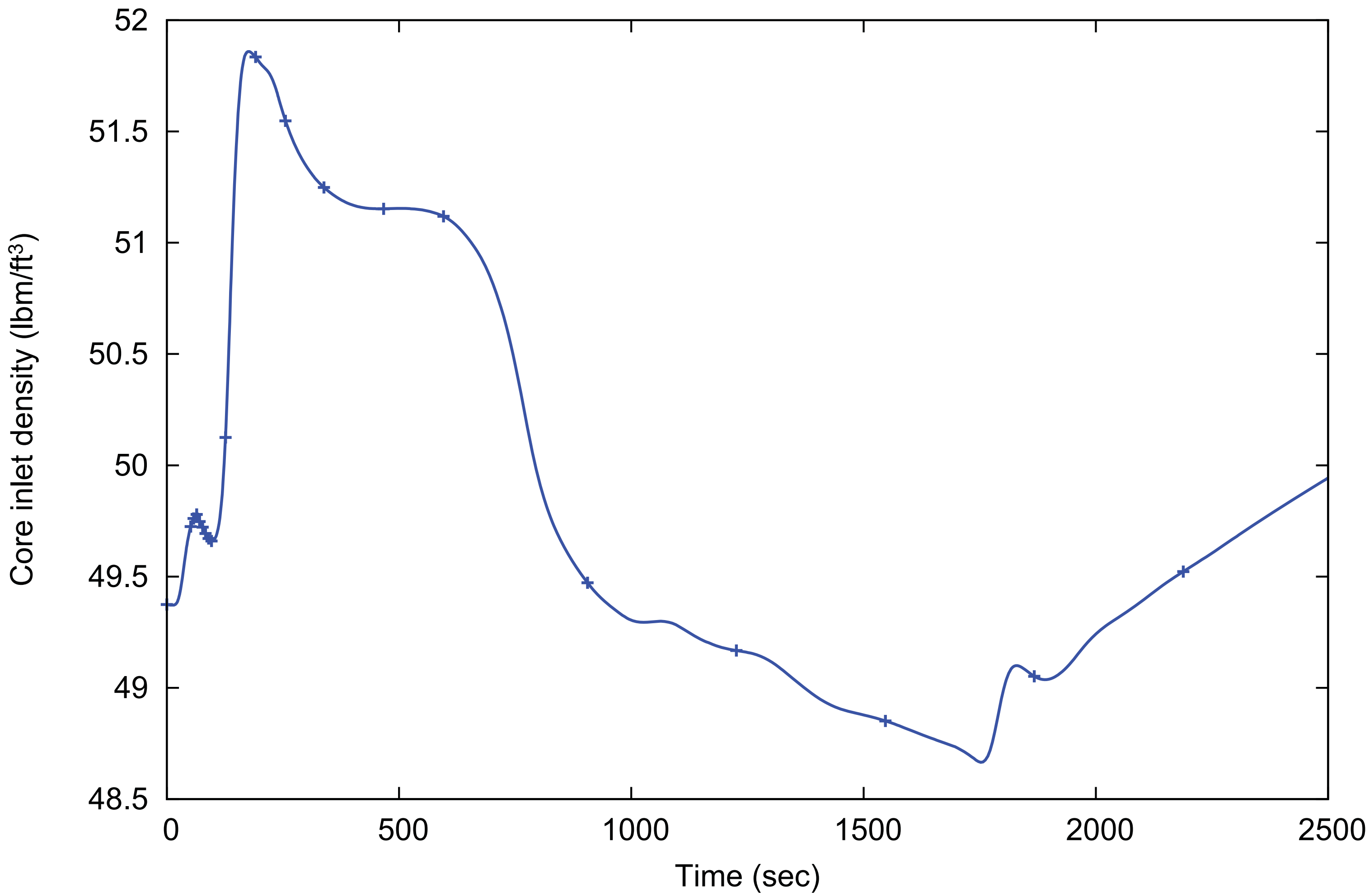


Figure 15.1-25: Total Core Reactivity (15.1.3 Increase in Steam Flow)

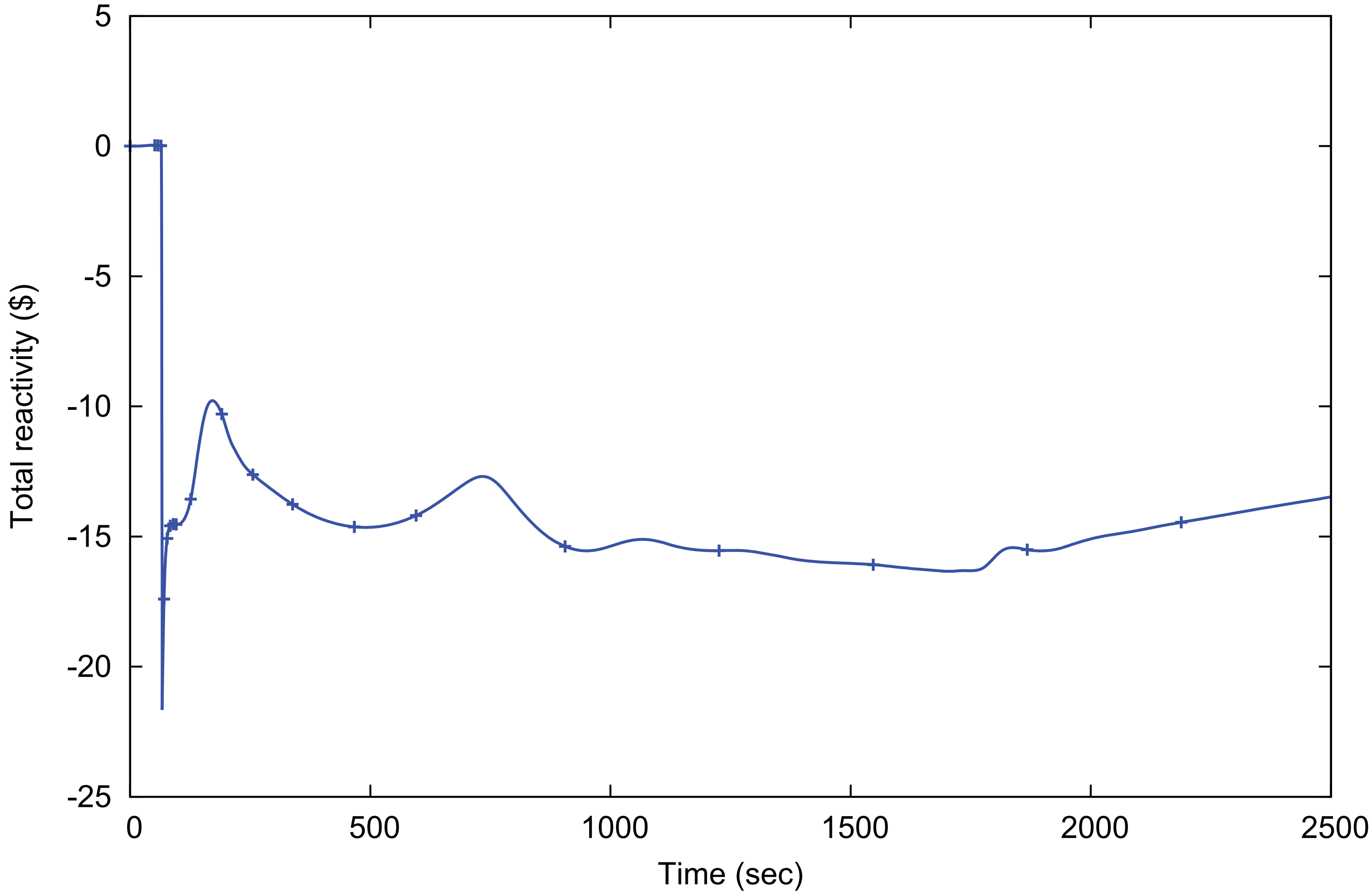


Figure 15.1-26: Reactor Power (15.1.3 Increase in Steam Flow)

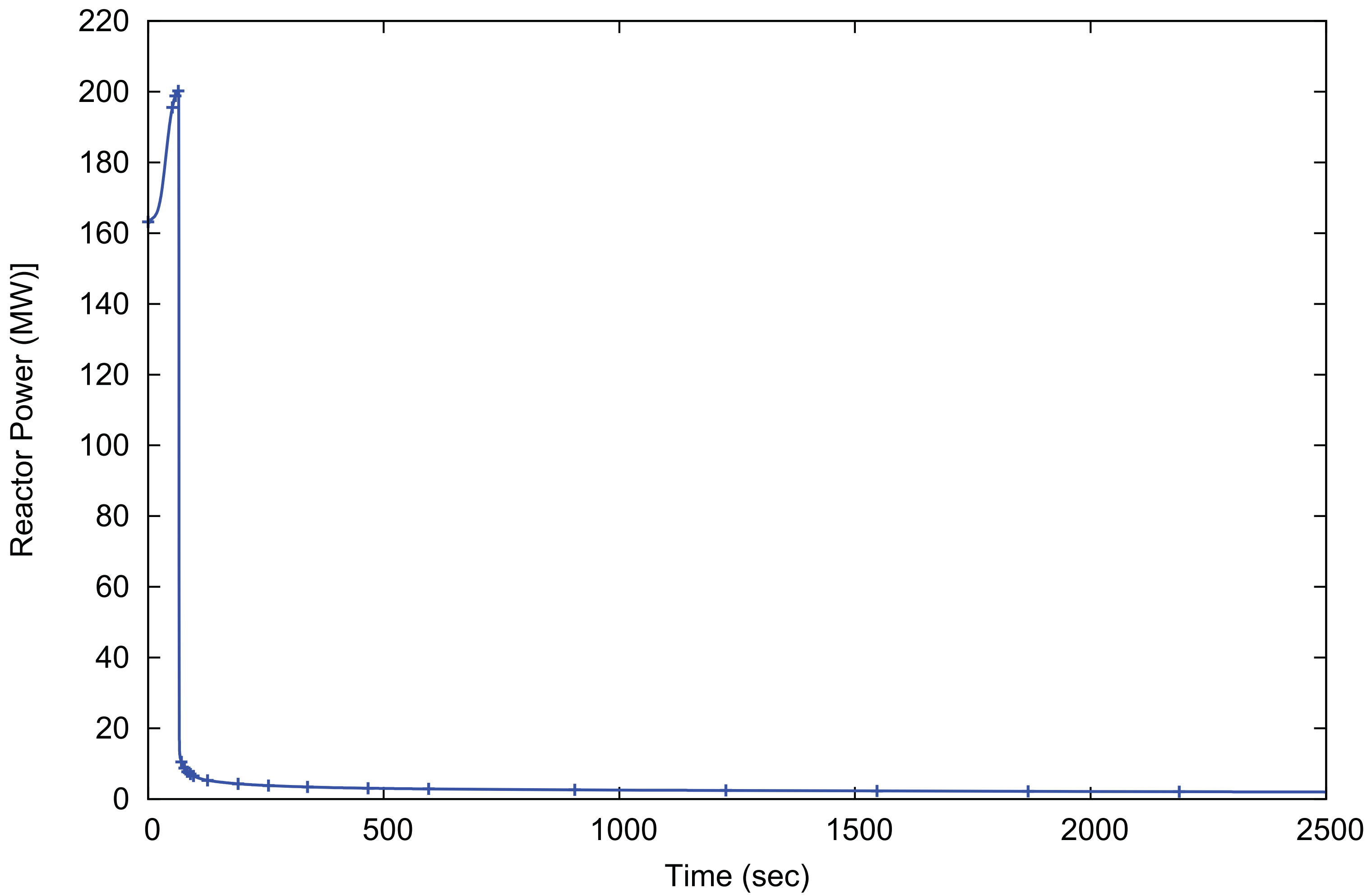


Figure 15.1-27: Reactor Coolant System Pressure (15.1.3 Increase in Steam Flow)

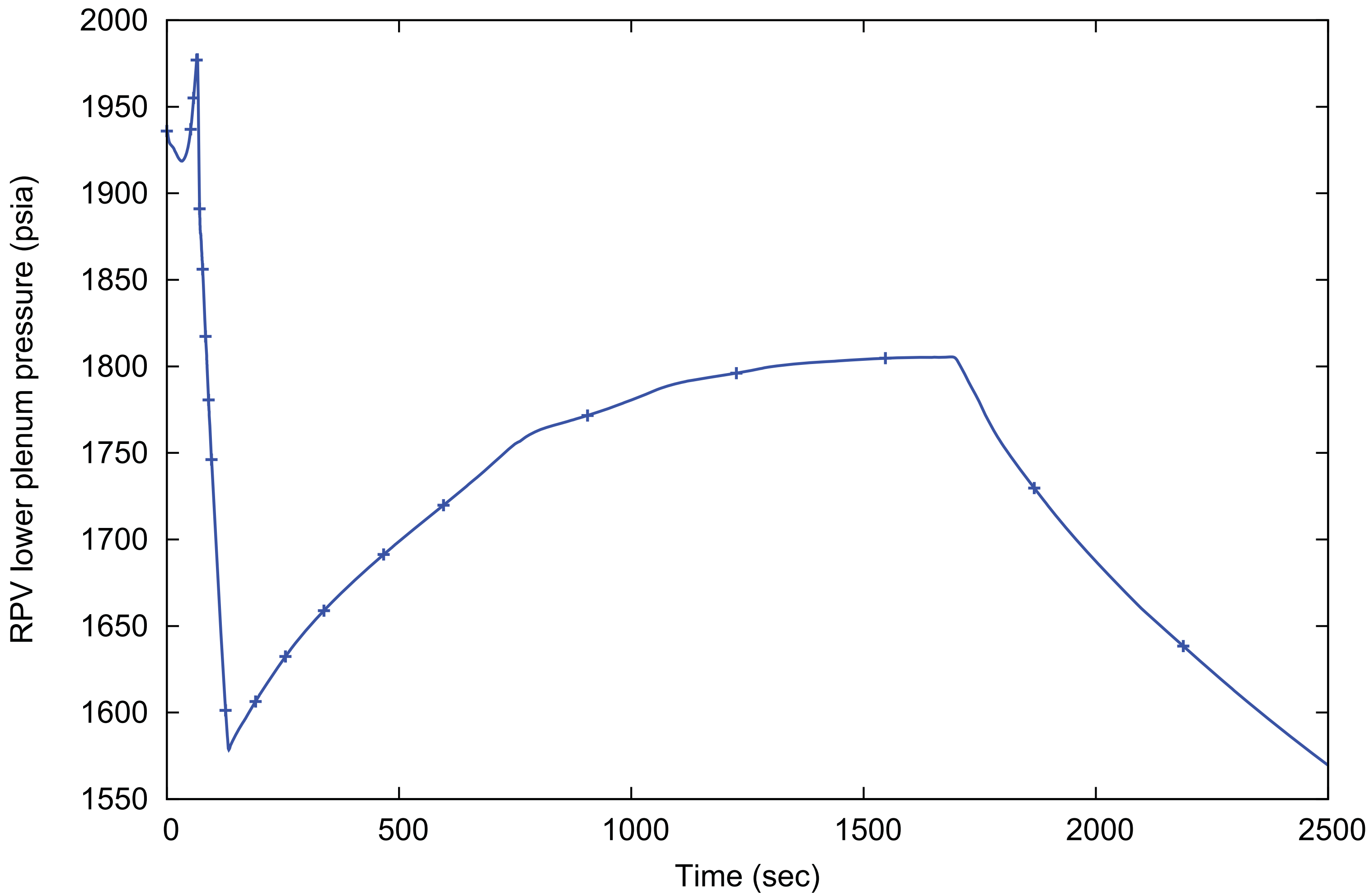


Figure 15.1-28: Core Outlet Temperature (15.1.3 Increase in Steam Flow)

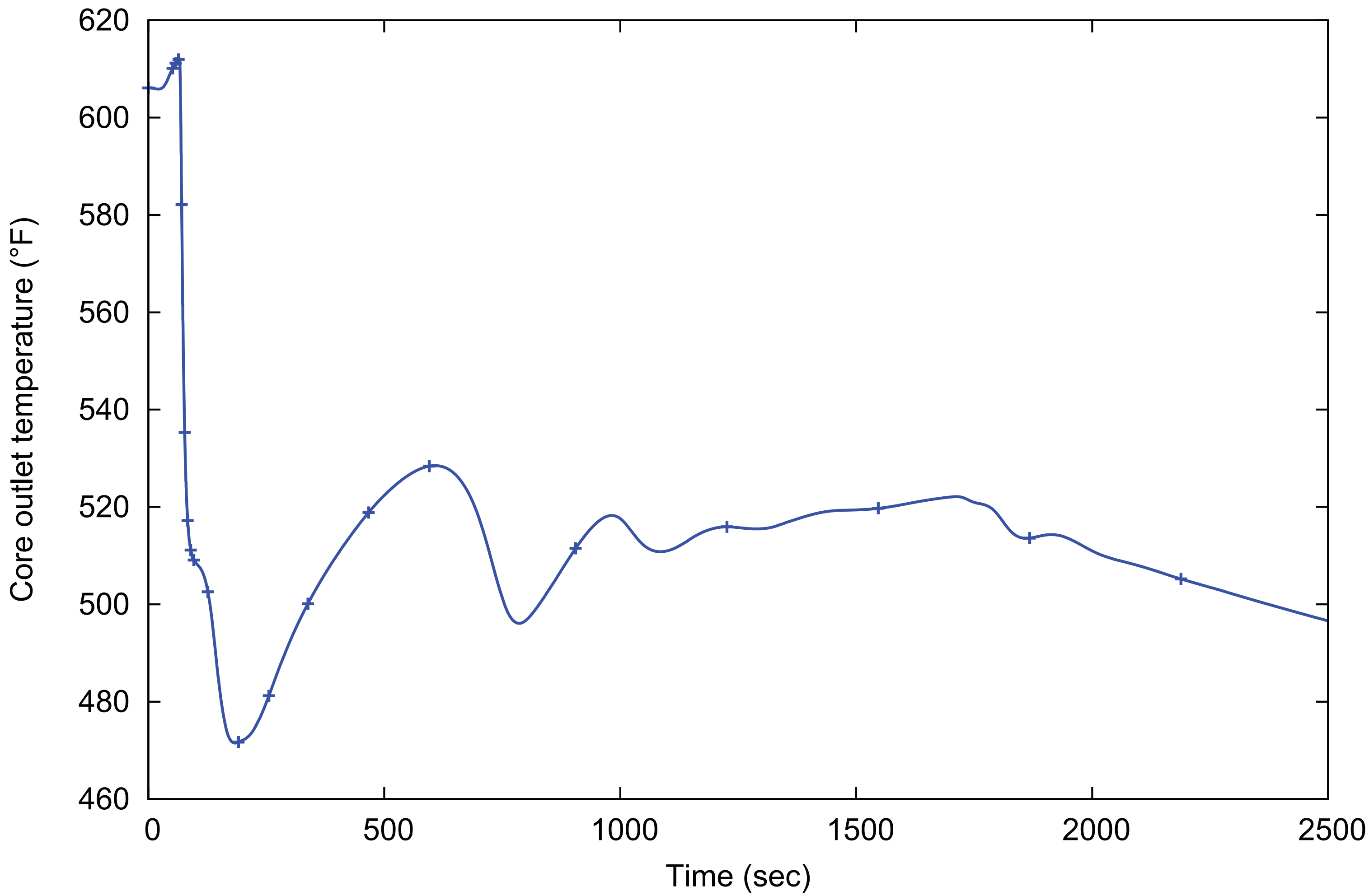


Figure 15.1-29: Main Steam System Pressure (15.1.3 Increase in Steam Flow)

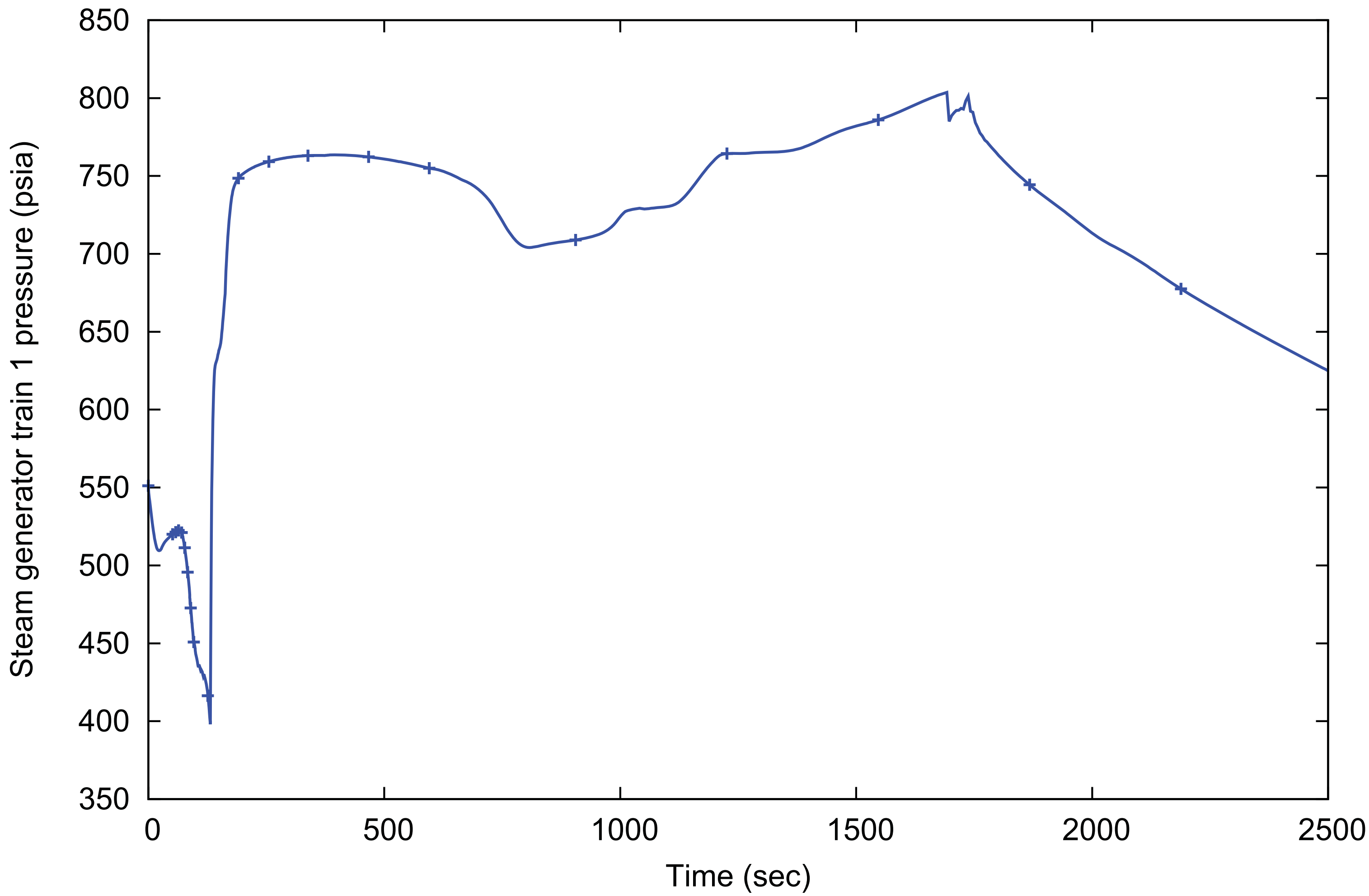


Figure 15.1-30: Critical Heat Flux Ratio (15.1.3 Increase in Steam Flow)

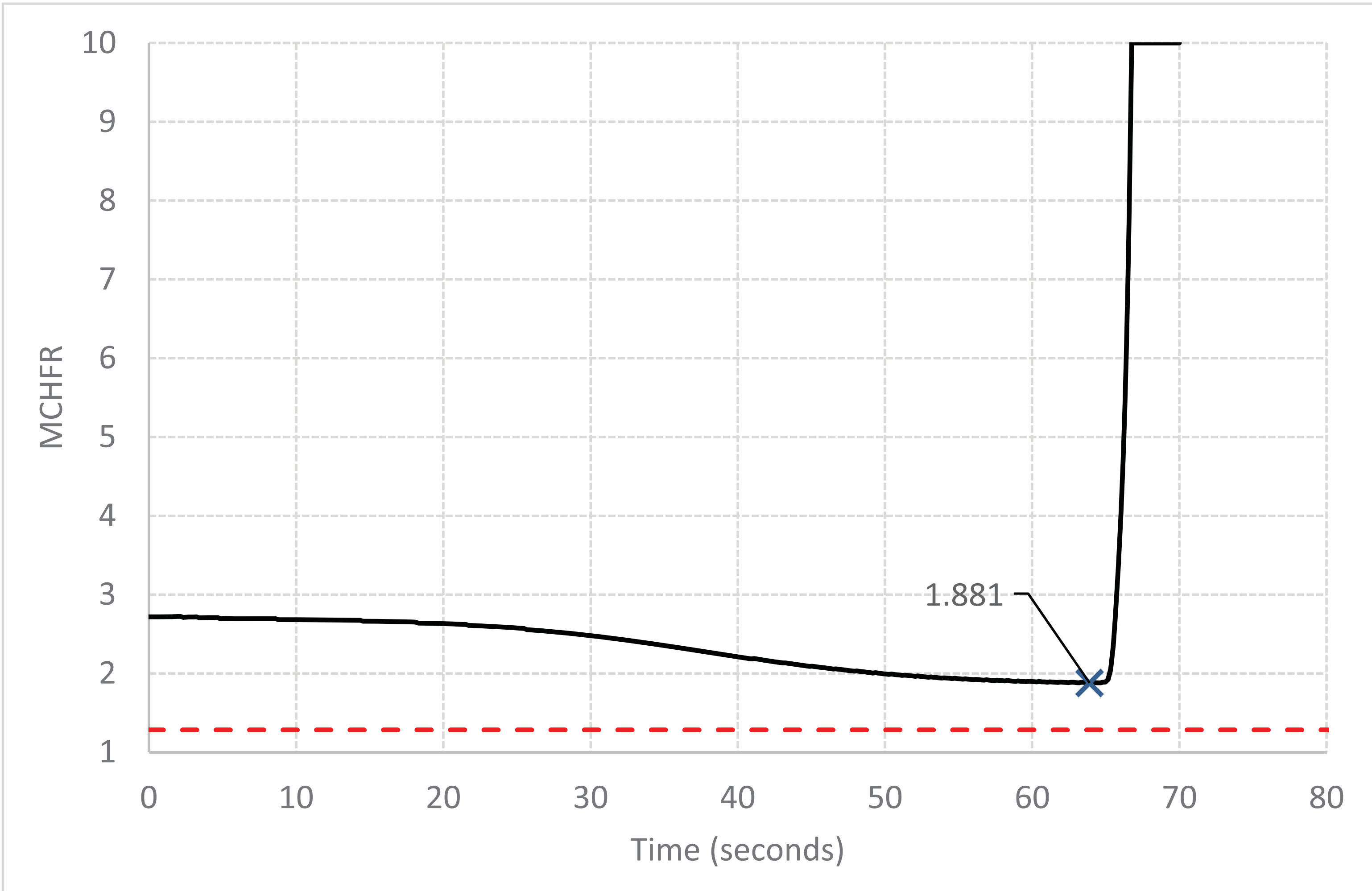


Figure 15.1-31: Reactor Coolant System Flow (15.1.3 Increase in Steam Flow)

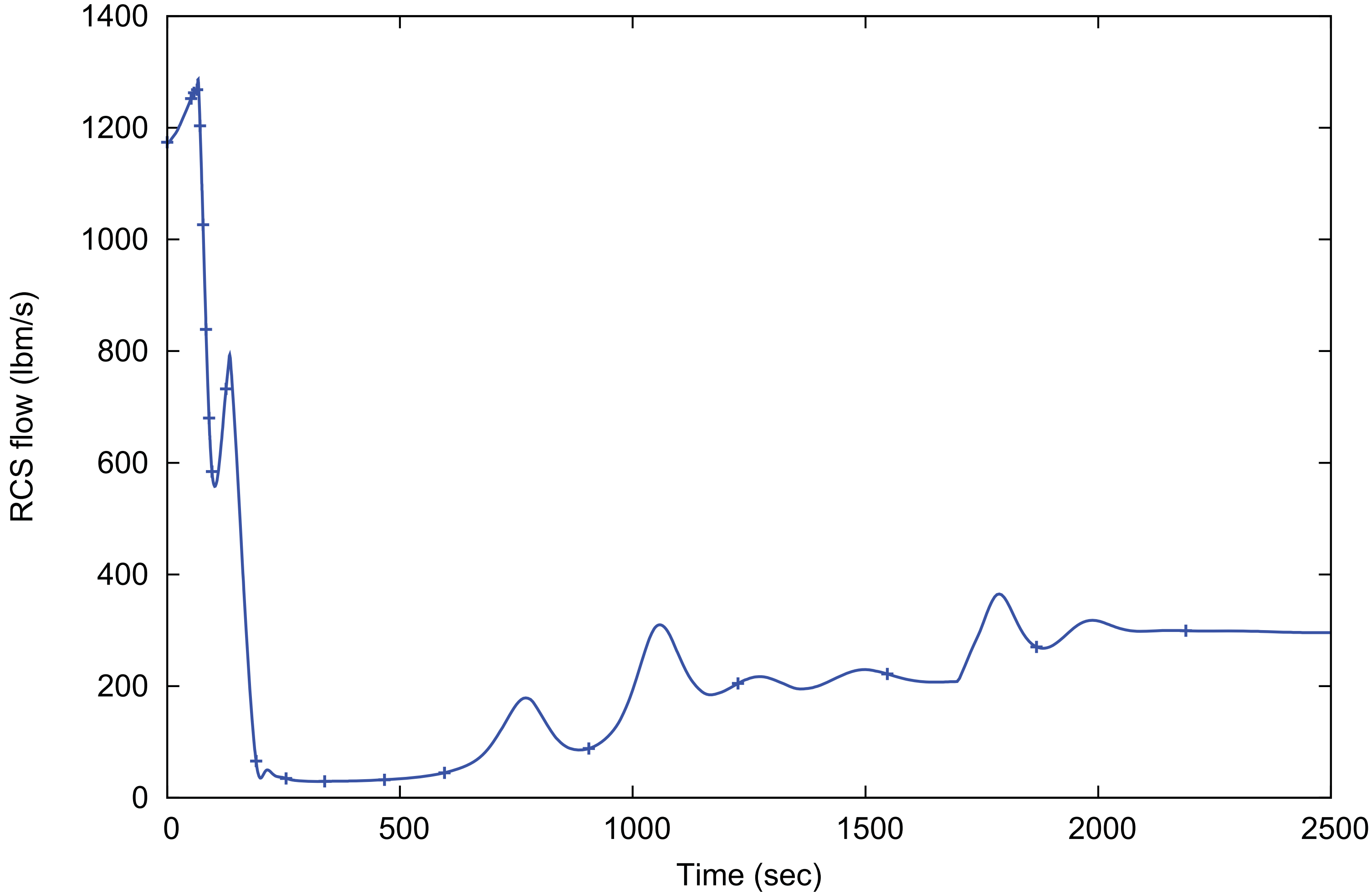


Figure 15.1-32: Break Flow Rate (15.1.5 Steam Piping Failure, Limiting Reactor Coolant System Pressure)

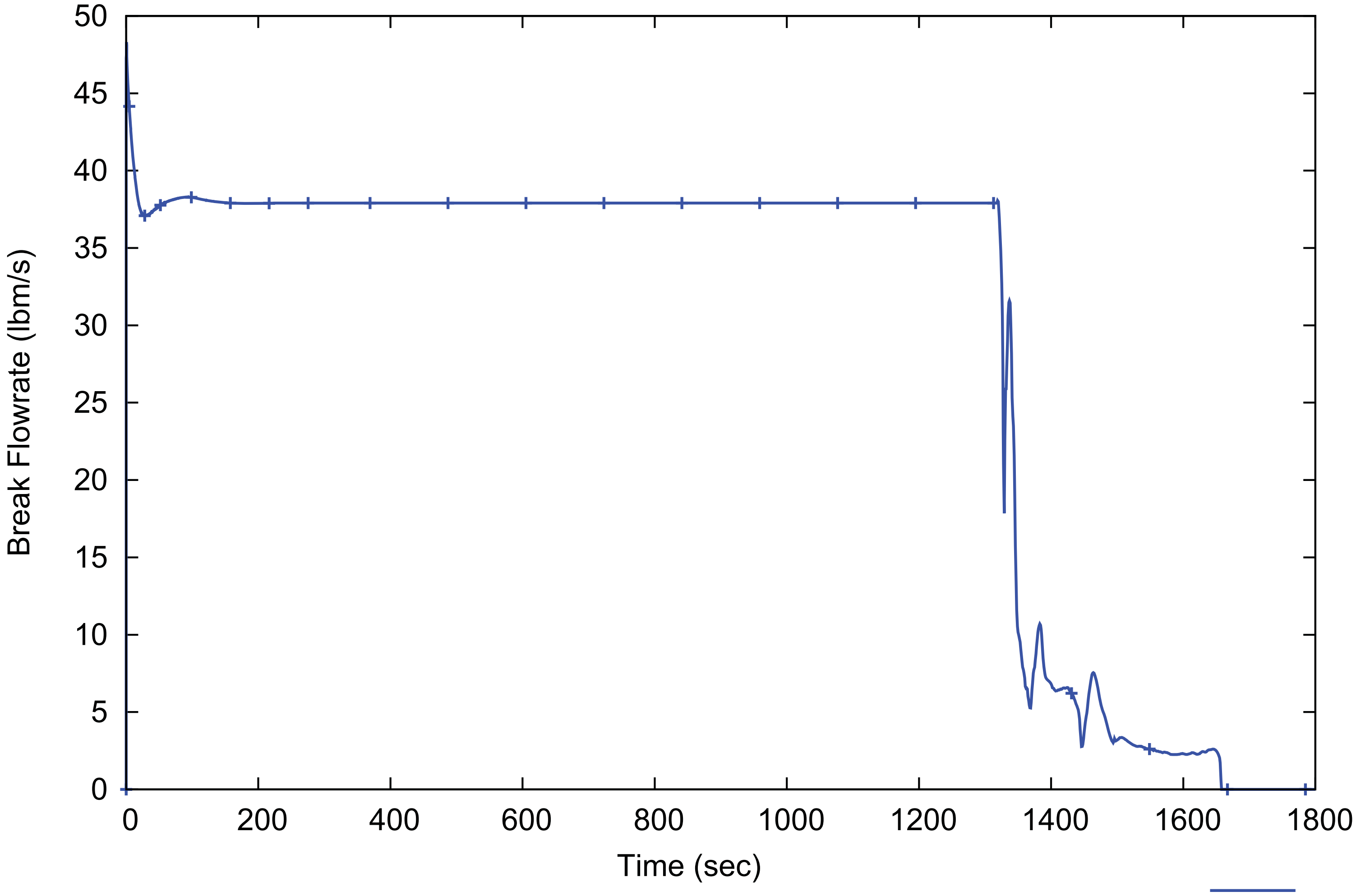


Figure 15.1-33: Reactor Coolant System Pressure (15.1.5 Steam Piping Failure, Limiting Reactor Coolant System Pressure)

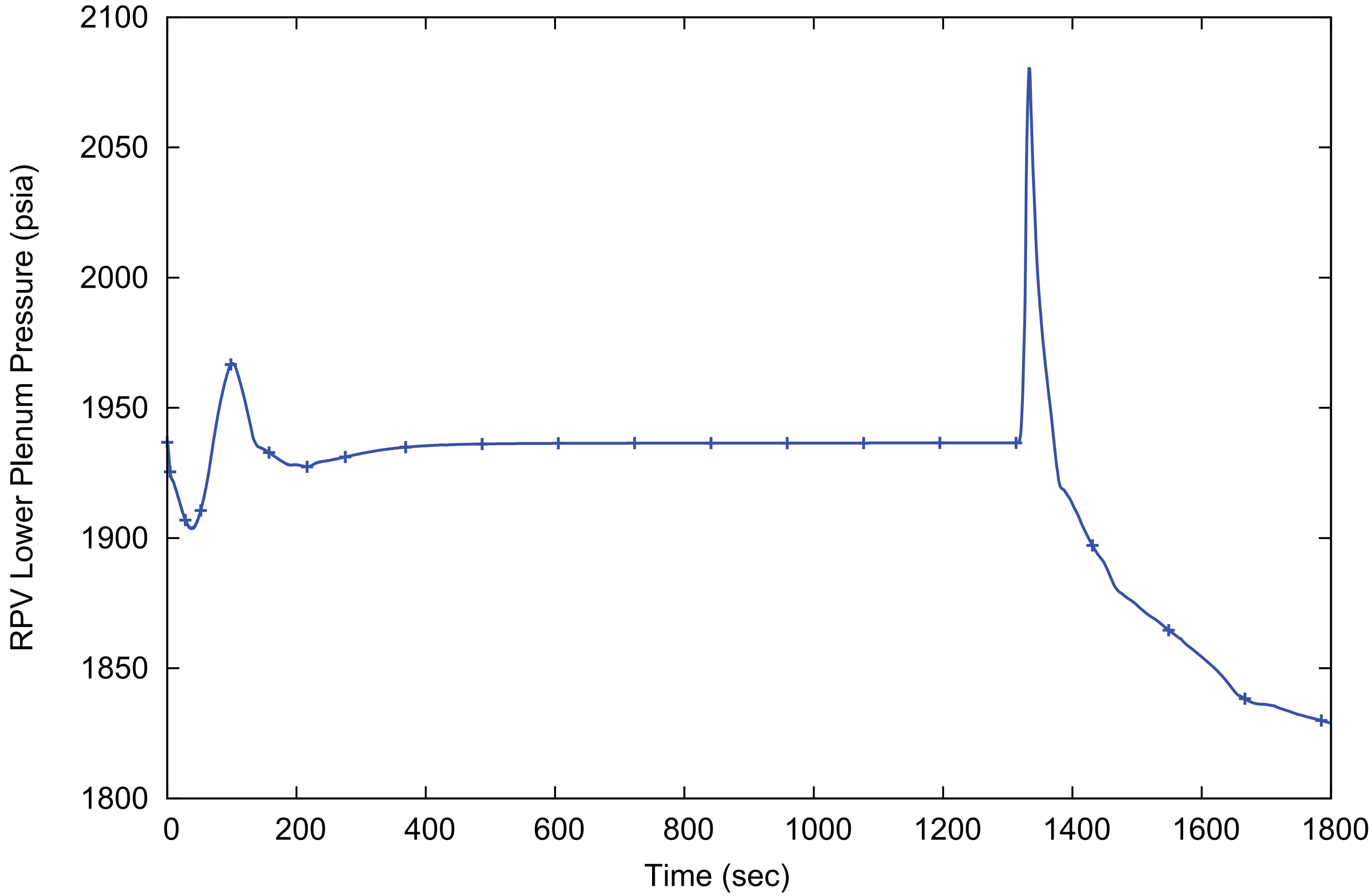


Figure 15.1-34: Reactor Power (15.1.5 Steam Piping Failure, Limiting Reactor Coolant System Pressure)

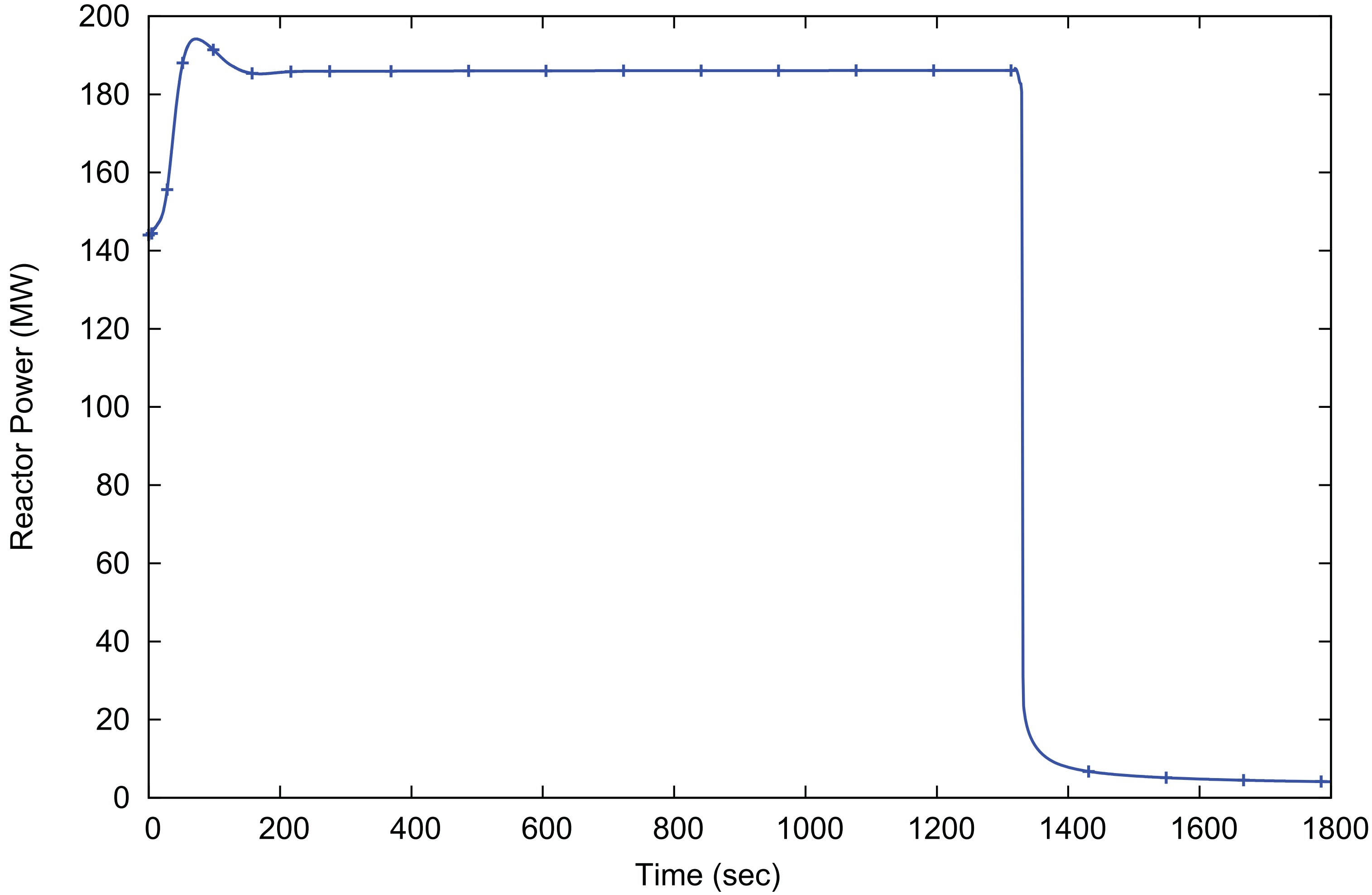


Figure 15.1-35: Core Outlet Temperature (15.1.5 Steam Piping Failure, Limiting Reactor Coolant System Pressure)

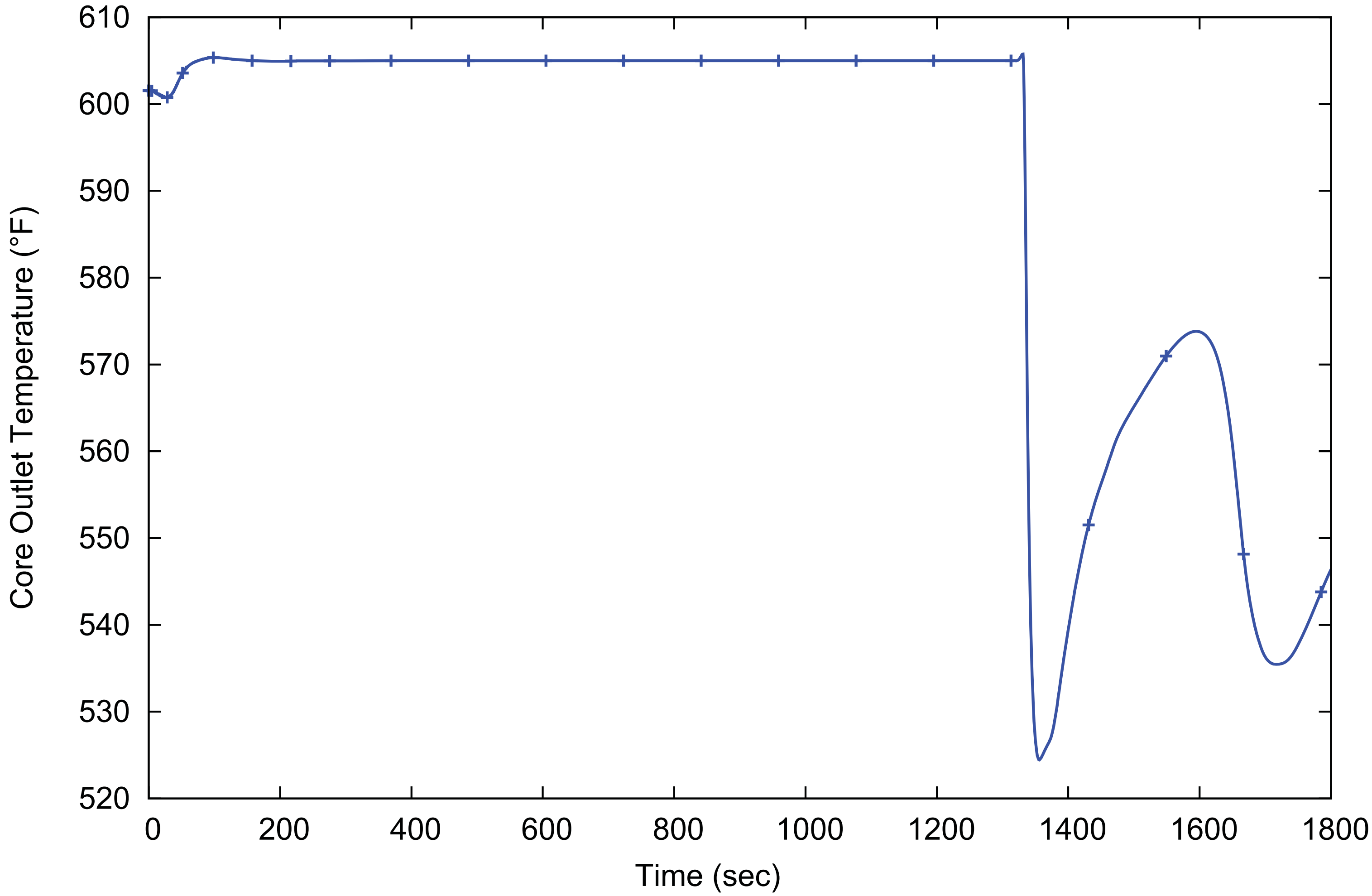


Figure 15.1-36: Core Inlet Temperature (15.1.5 Steam Piping Failure, Limiting Reactor Coolant System Pressure)

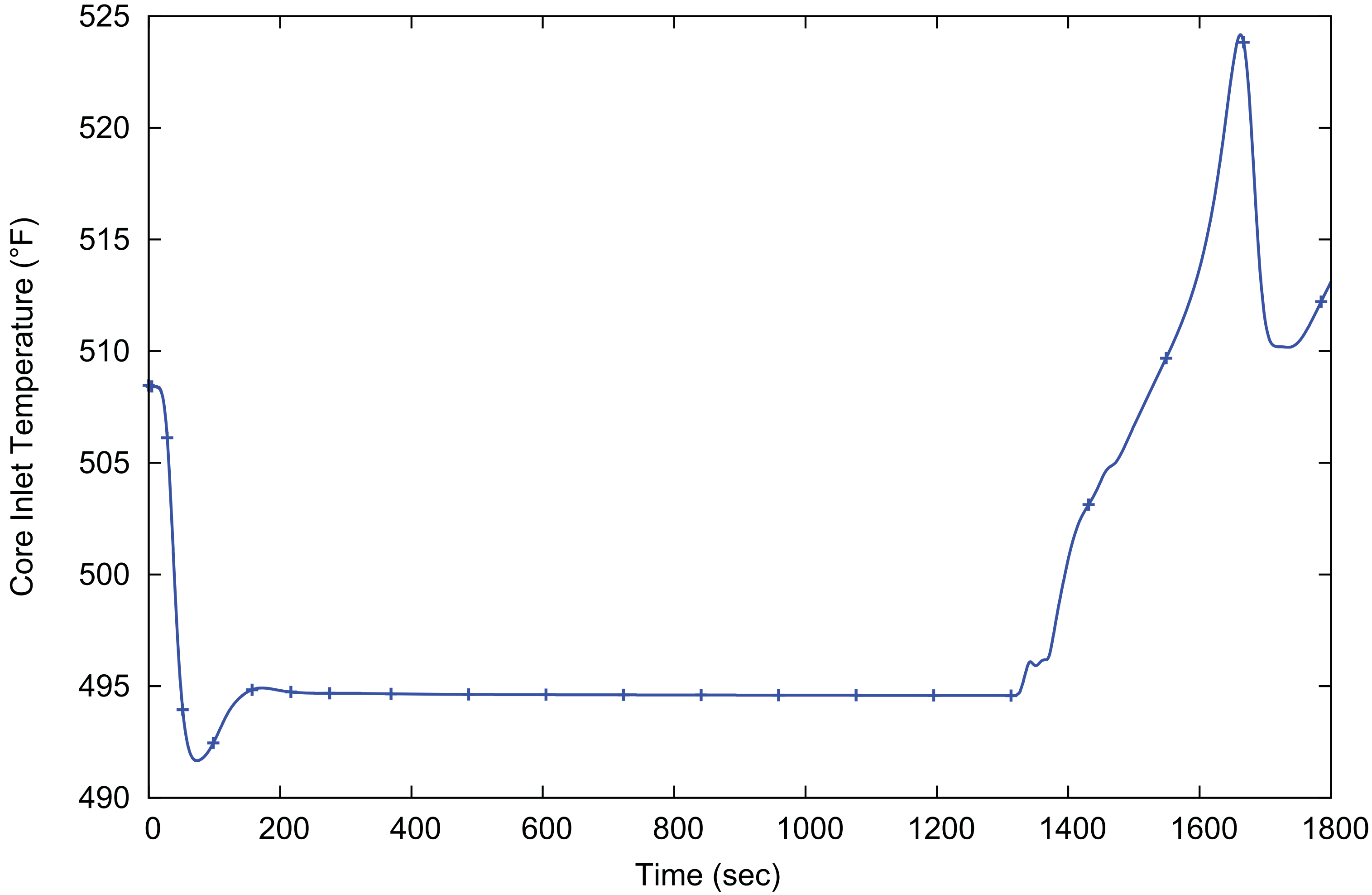


Figure 15.1-37: Maximum Steam Generator Pressure (15.1.5 Steam Piping Failure, Limiting Reactor Coolant System Pressure)

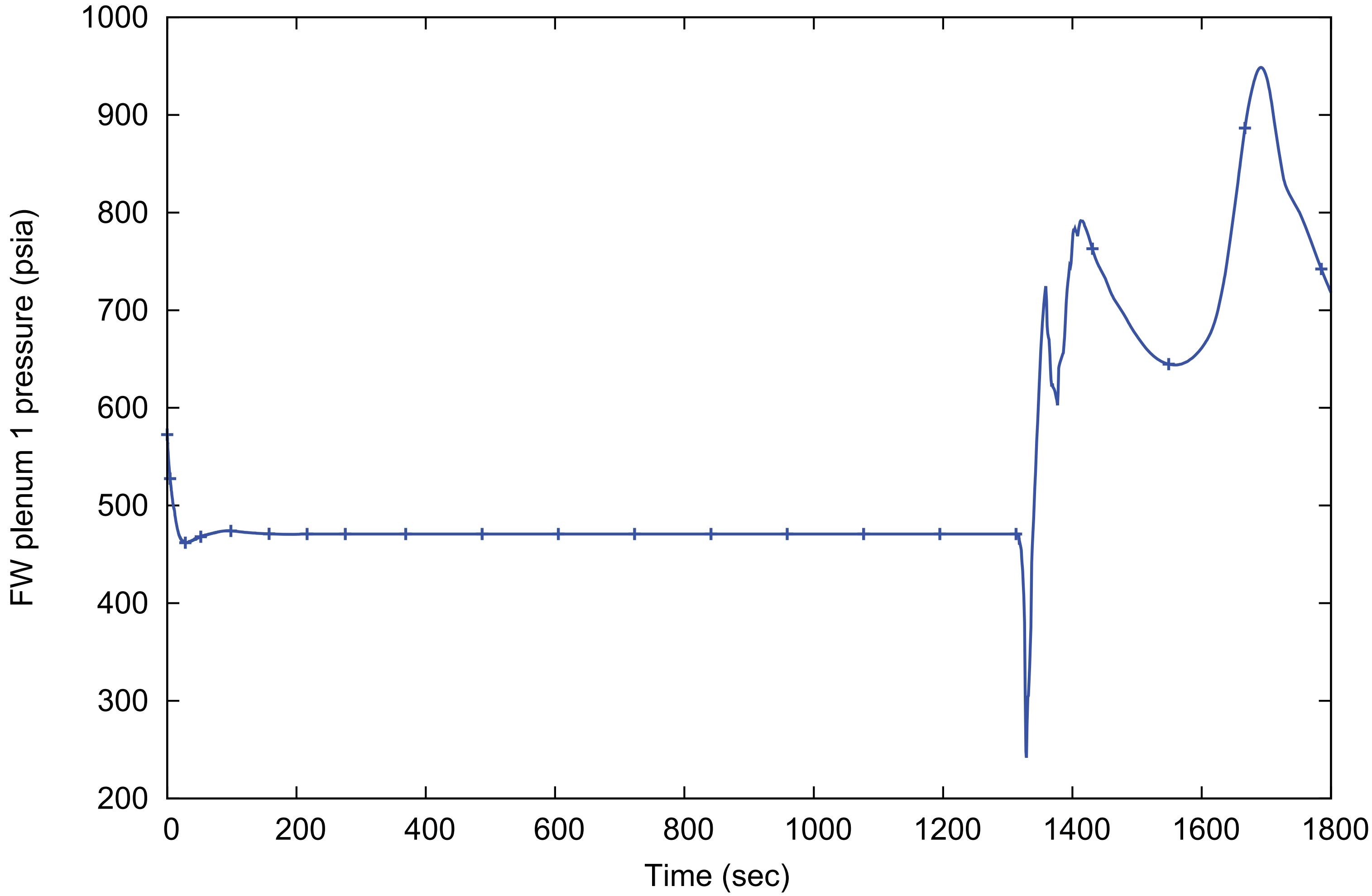


Figure 15.1-38: Break Flowrate (15.1.5 Steam Piping Failure, Limiting Minimum Critical Heat Flux Ratio)

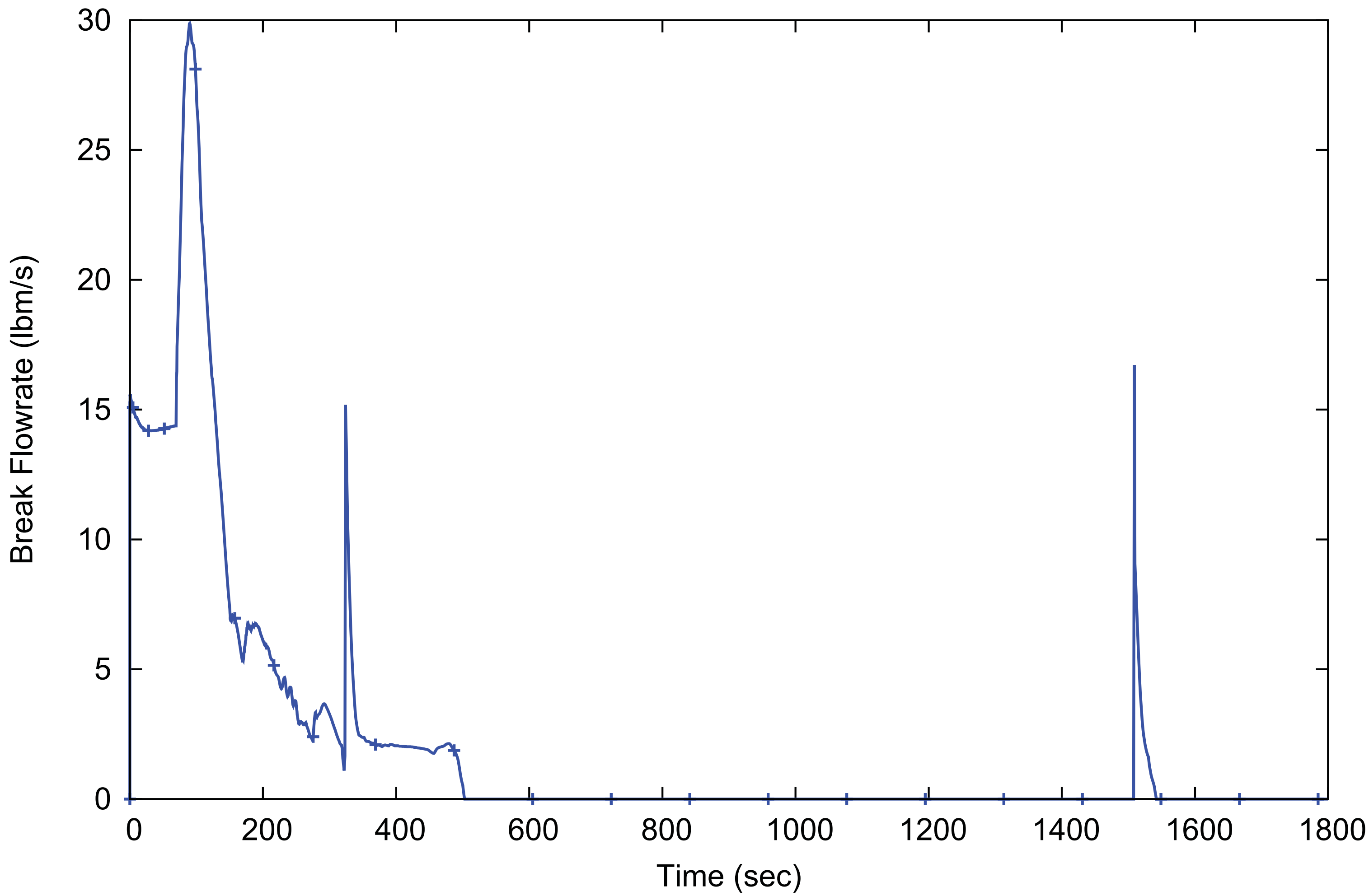


Figure 15.1-39: Total Core Reactivity (15.1.5 Steam Piping Failure, Limiting Minimum Critical Heat Flux Ratio)

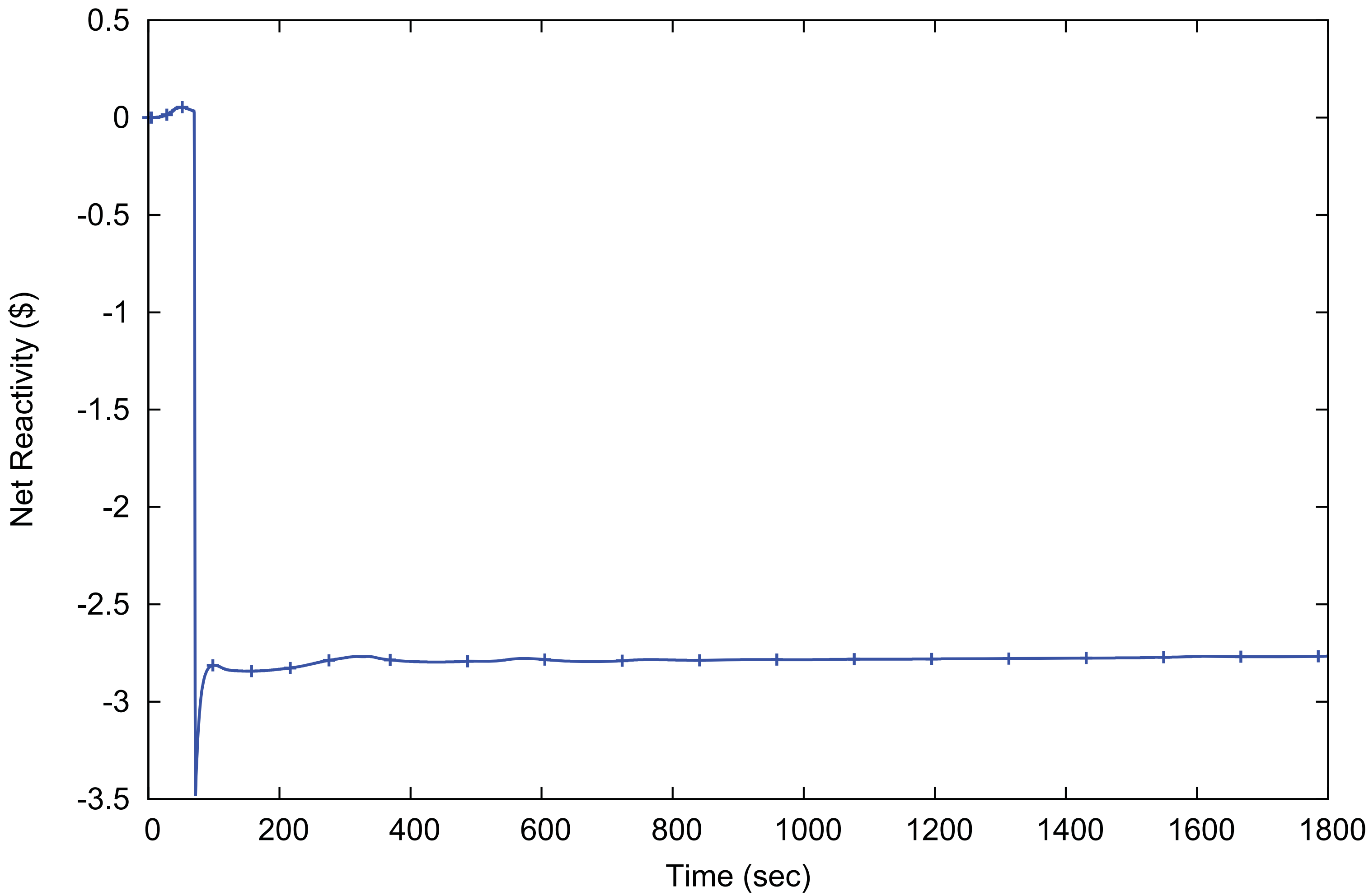


Figure 15.1-40: Reactor Power (15.1.5 Steam Piping Failure, Limiting Minimum Critical Heat Flux Ratio)

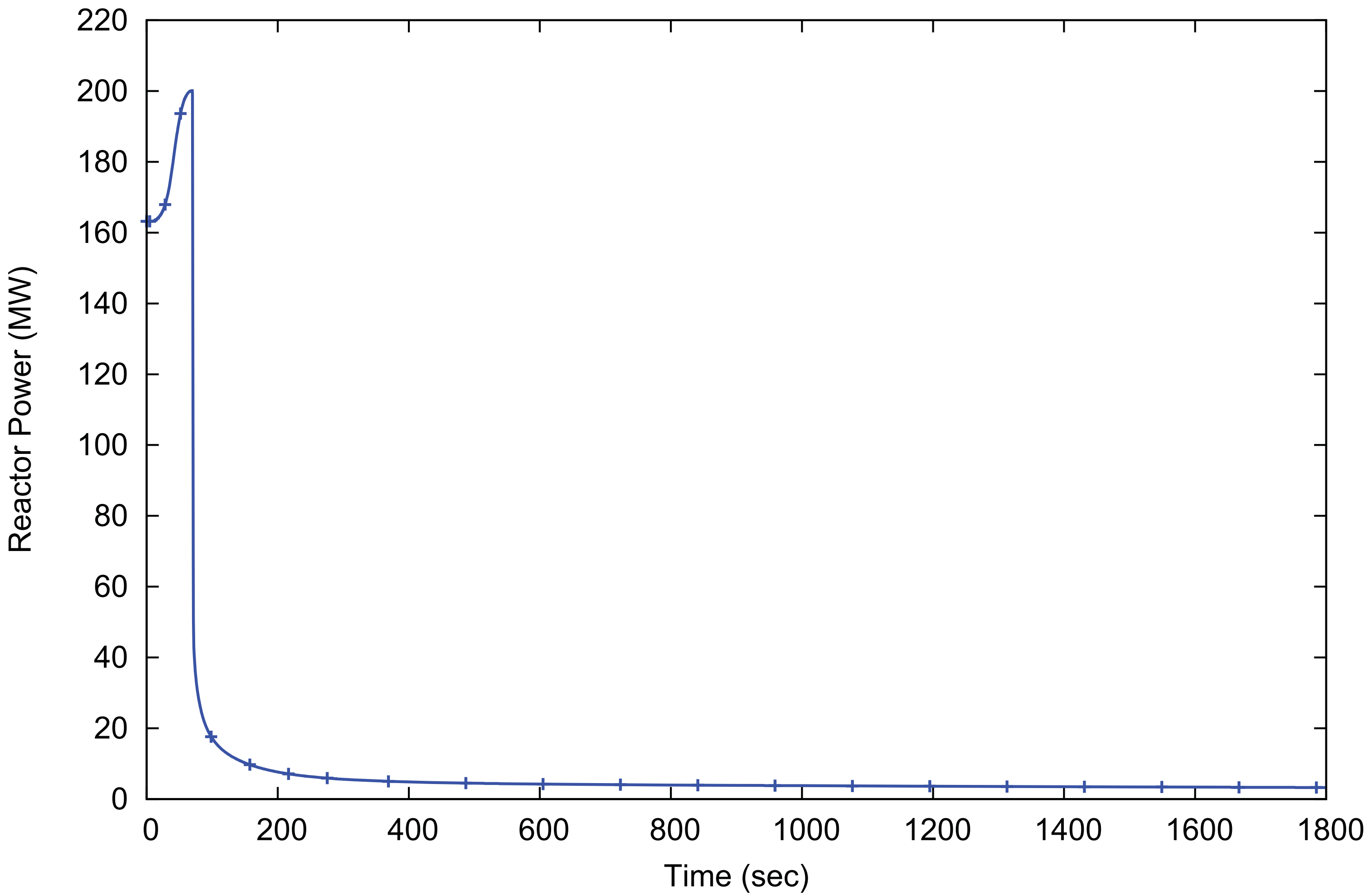


Figure 15.1-41: Reactor Coolant System Pressure (15.1.5 Steam Piping Failure, Limiting Minimum Critical Heat Flux Ratio)

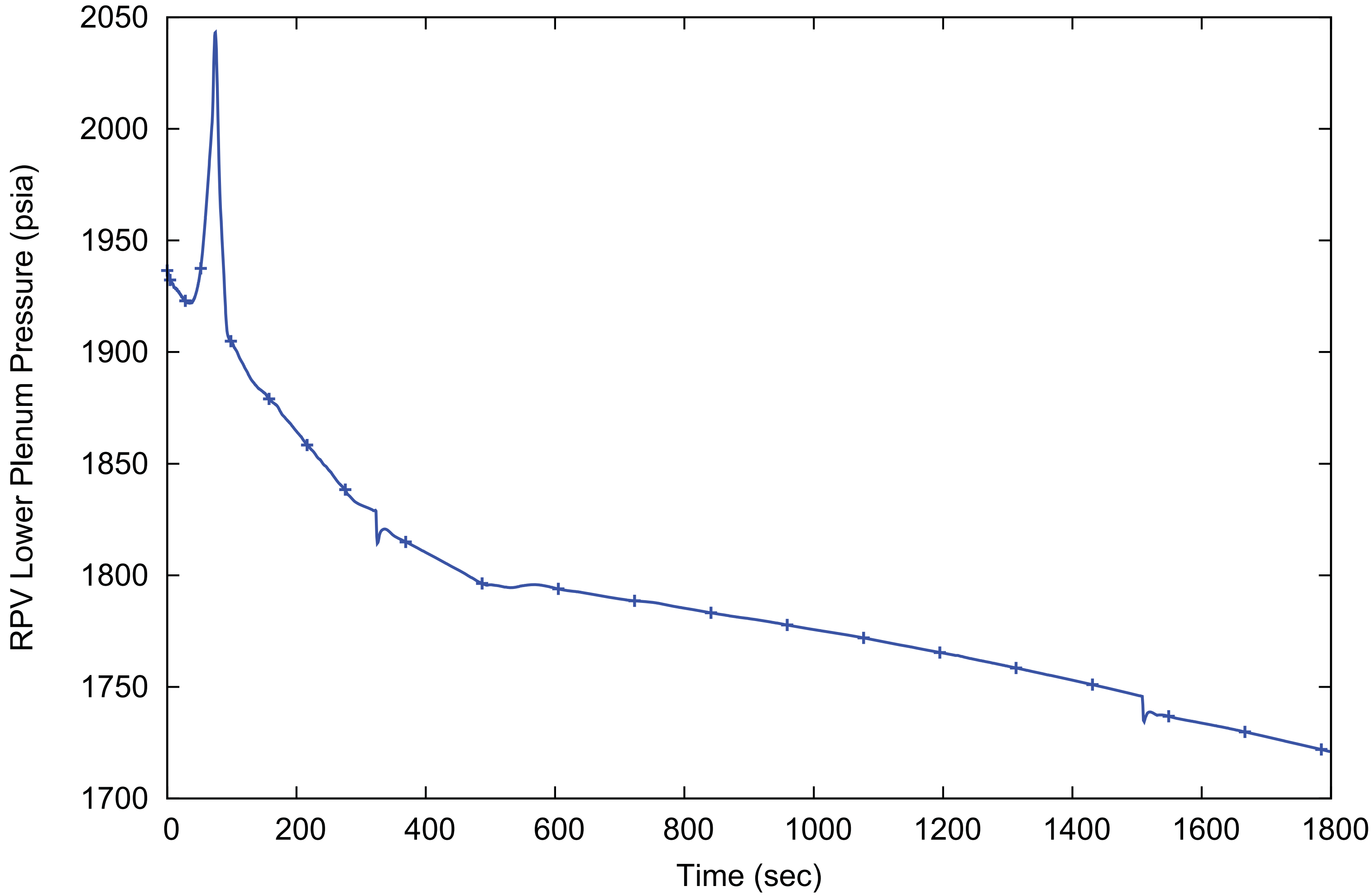


Figure 15.1-42: Volume Average Core Coolant Temperature (15.1.5 Steam Piping Failure, Limiting Minimum Critical Heat Flux Ratio)

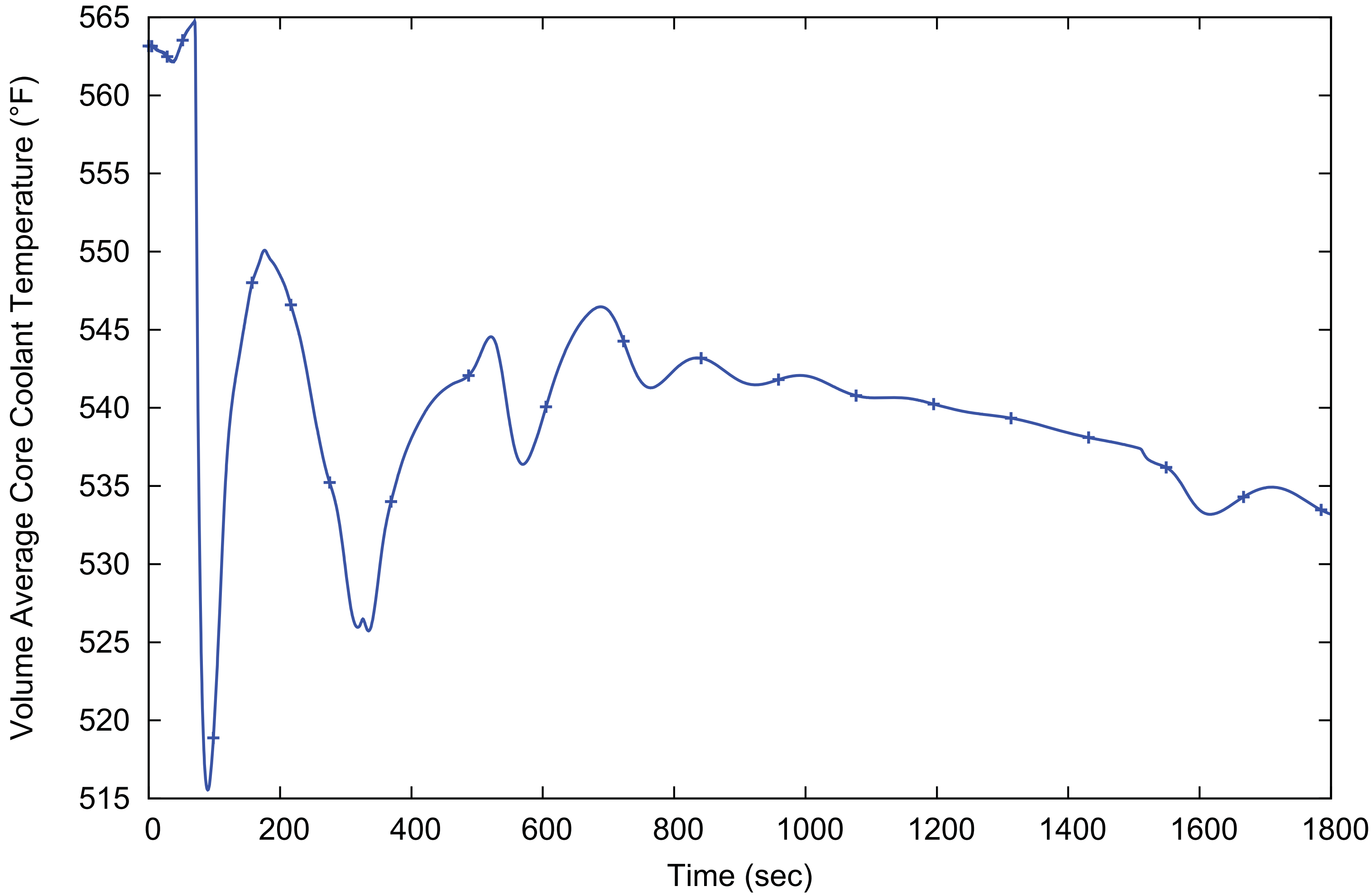


Figure 15.1-43: Failed Steam Generator Train Pressure (15.1.5 Steam Piping Failure, Limiting Minimum Critical Heat Flux Ratio)

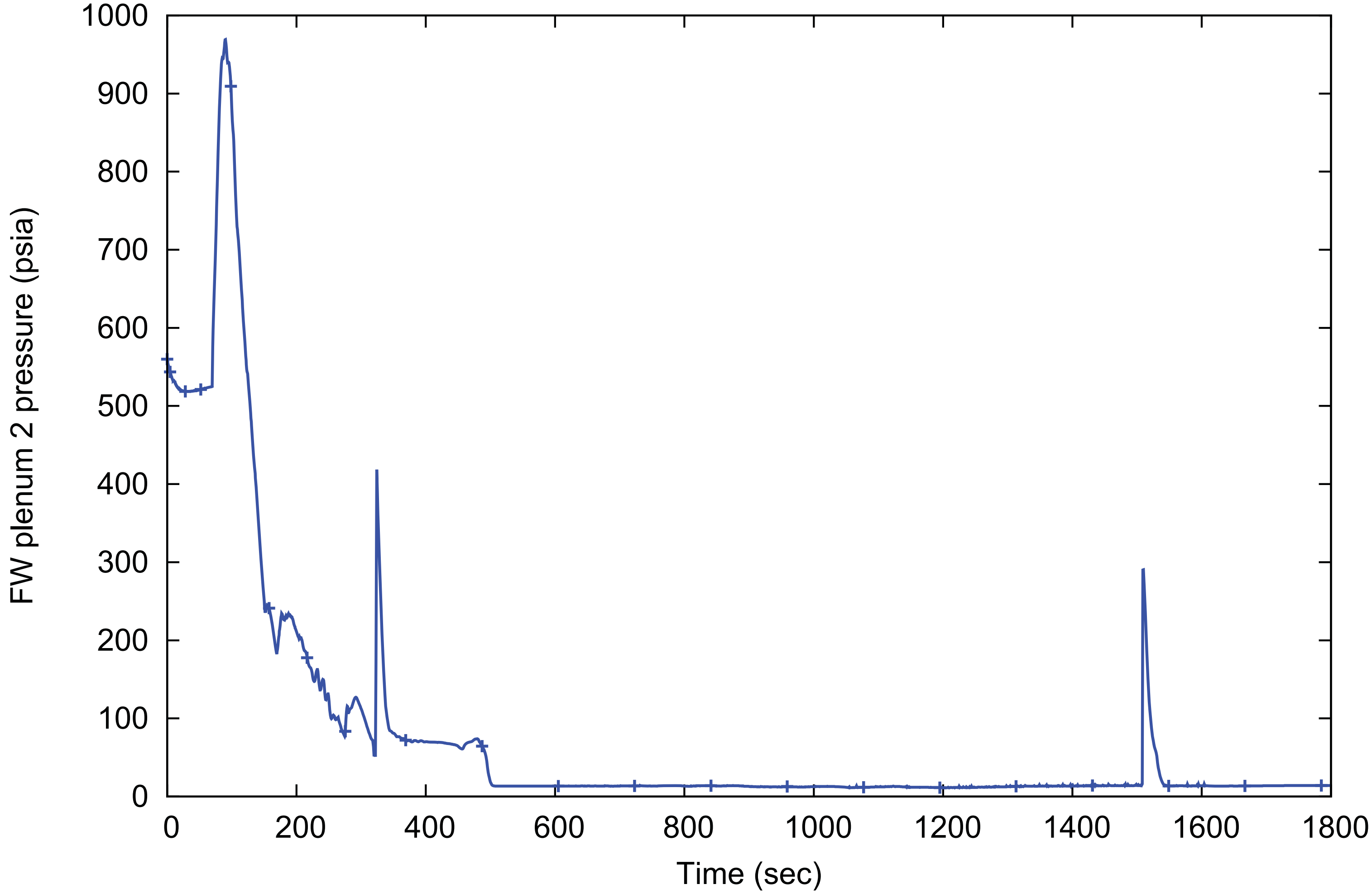


Figure 15.1-44: Critical Heat Flux Ratio (15.1.5 Steam Piping Failure, Limiting Minimum Critical Heat Flux Ratio)

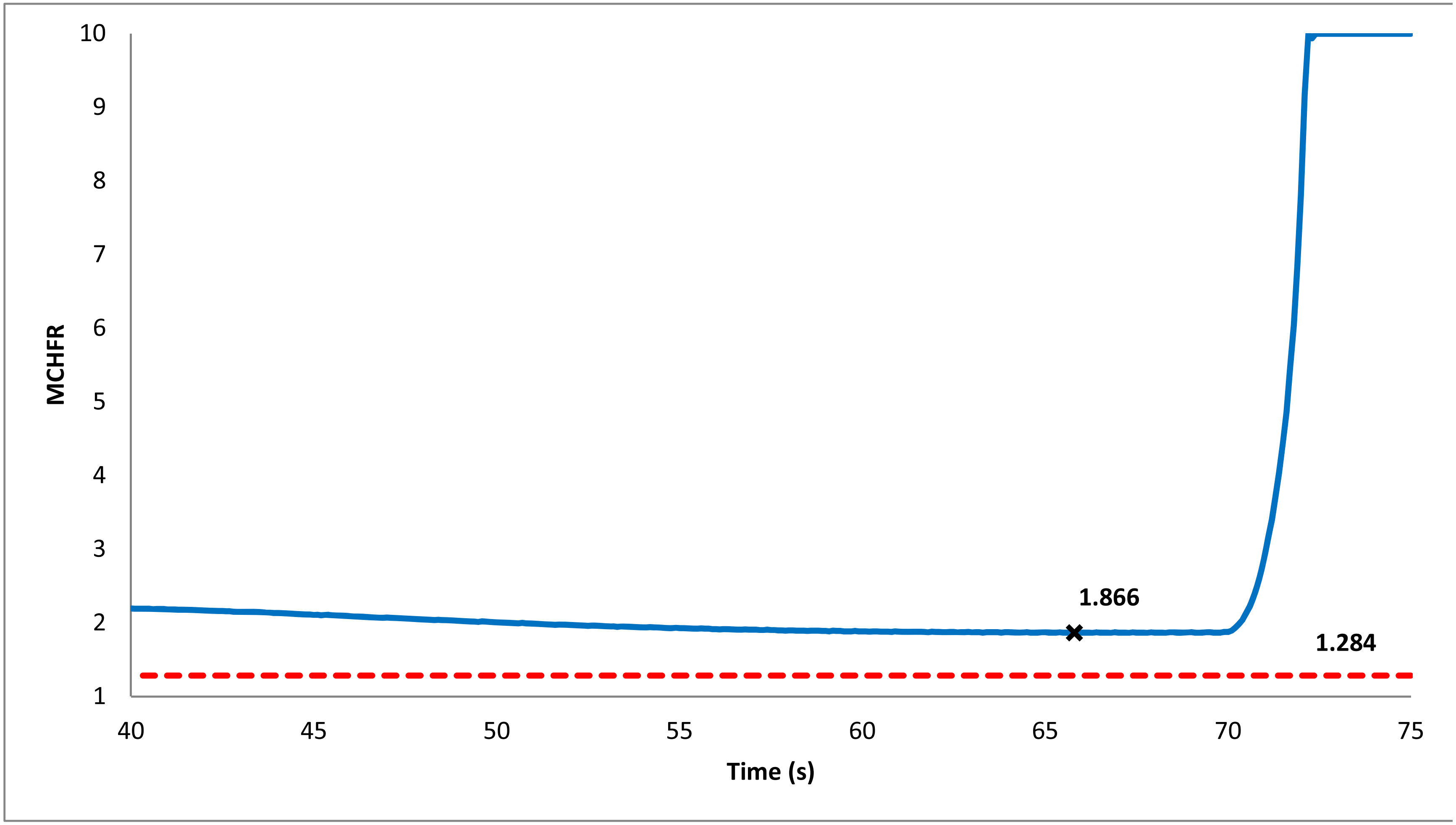


Figure 15.1-45: Break Flow Rate (15.1.5 Steam Piping Failure, Limiting Radiological)

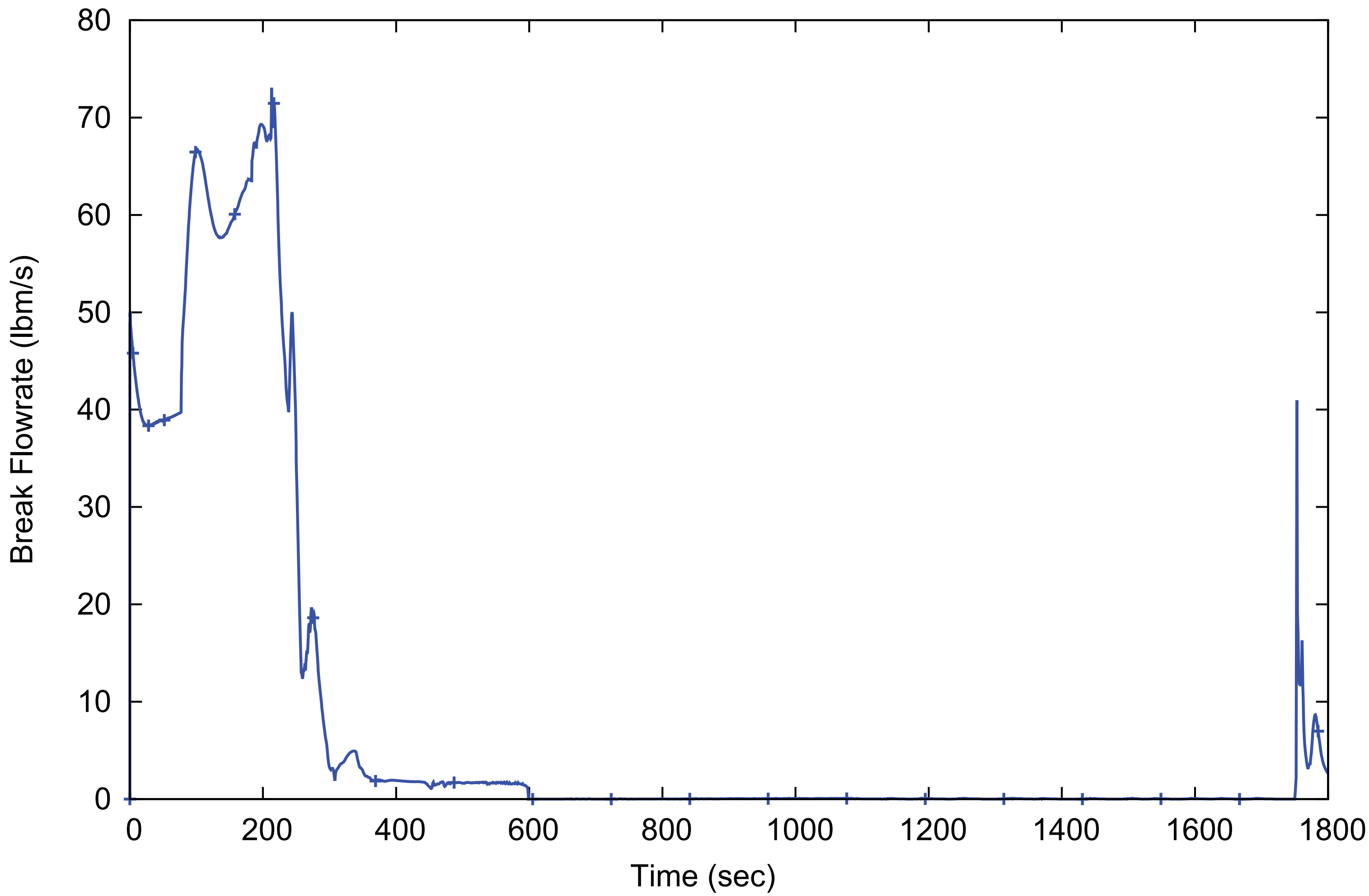


Figure 15.1-46: Integrated Break Flow (15.1.5 Steam Piping Failure, Limiting Radiological)

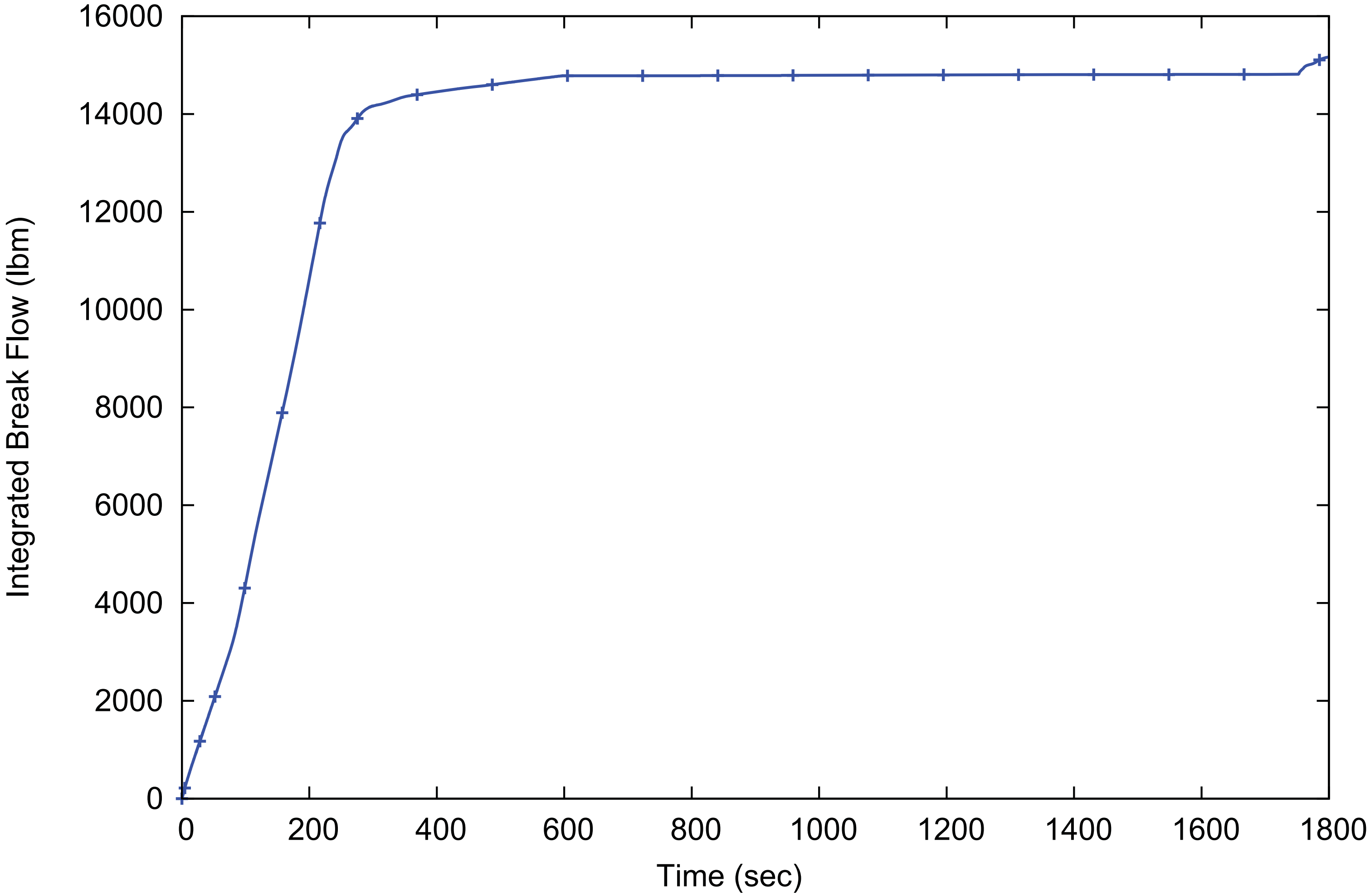


Figure 15.1-47: Reactor Power (15.1.6 Containment Flooding)

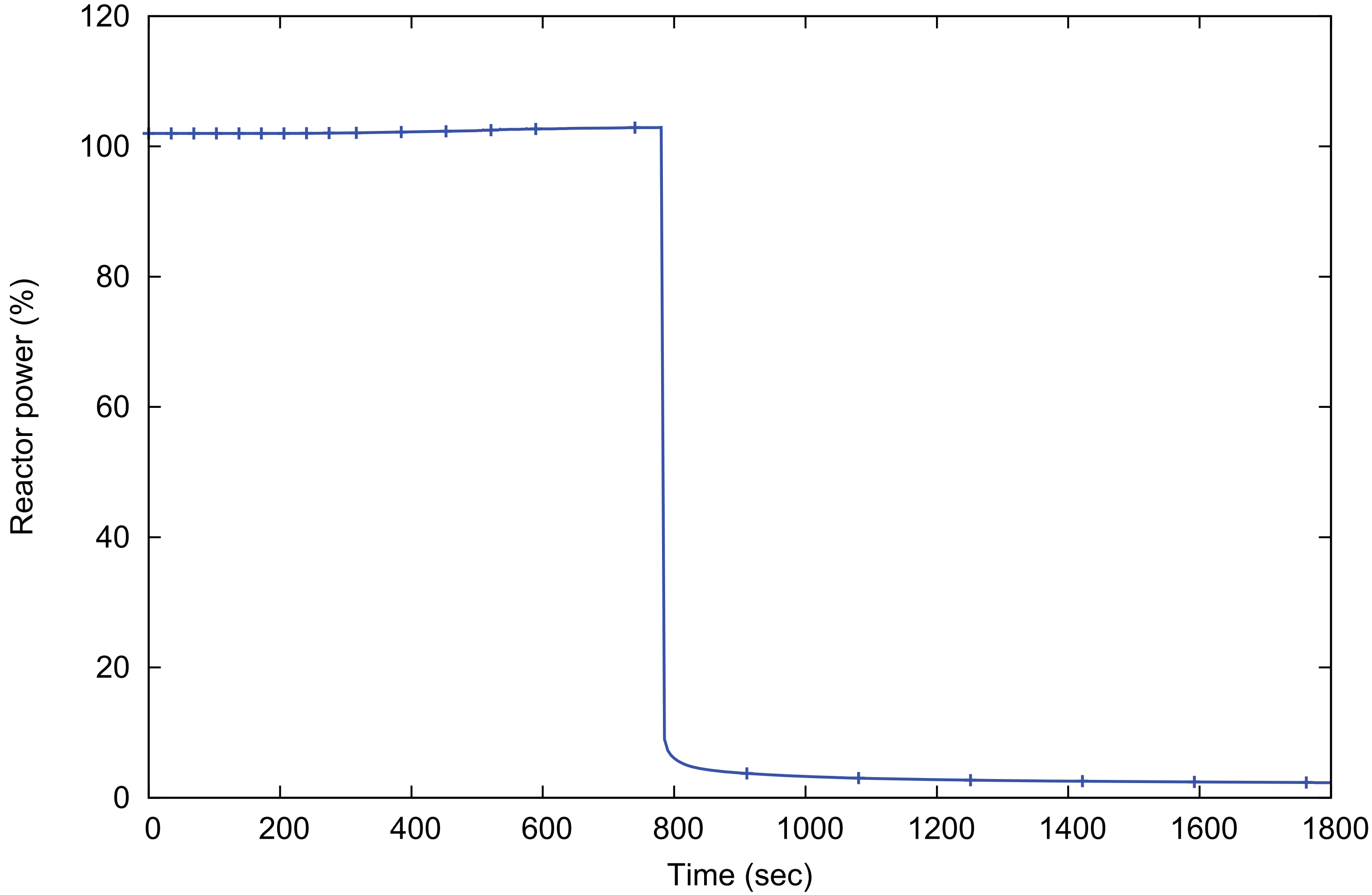


Figure 15.1-48: Reactor Component Cooling Water System Break Flow Rate (15.1.6 Containment Flooding)

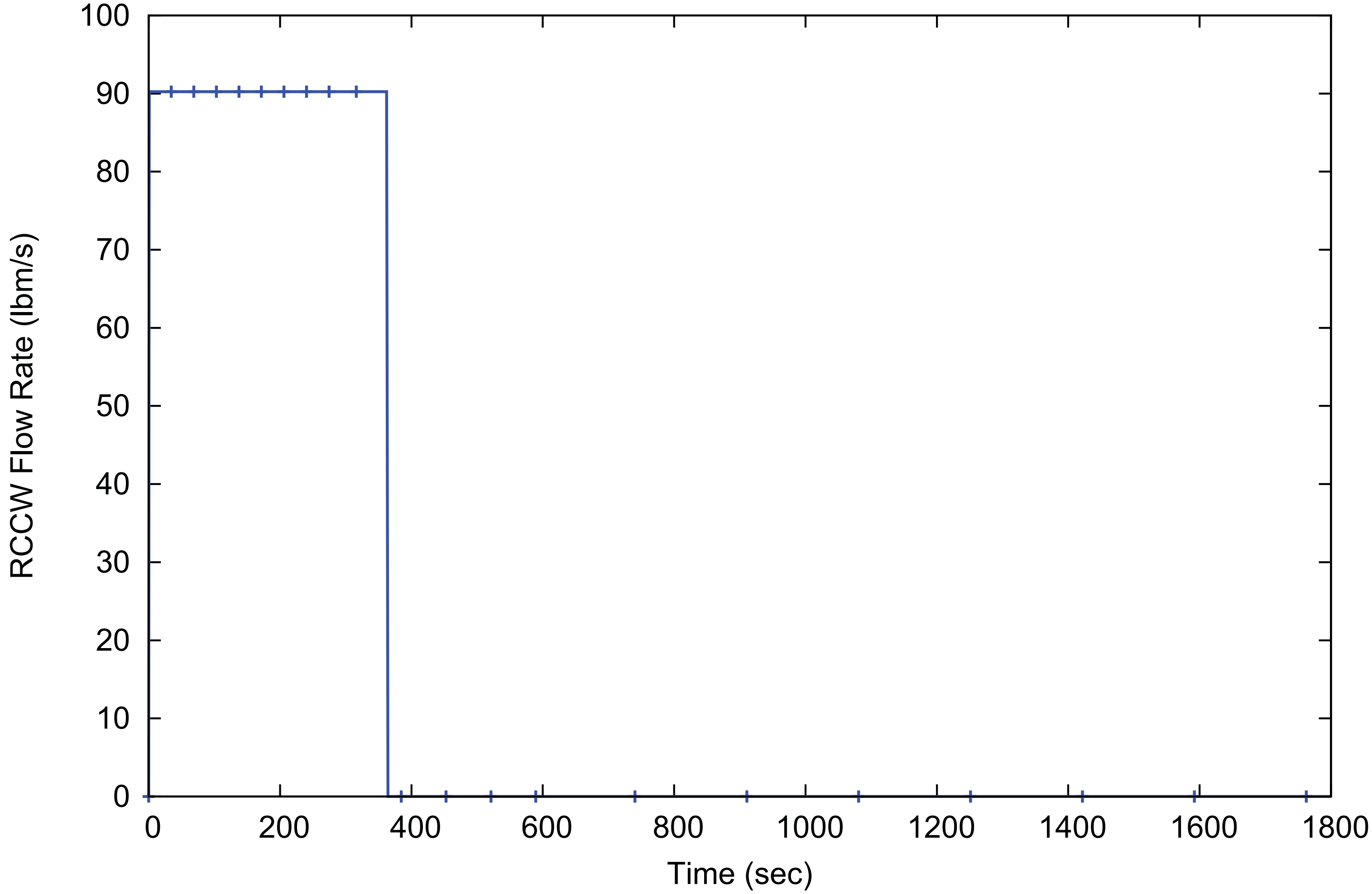


Figure 15.1-49: Reactor Pressure Vessel Heat Transfer (15.1.6 Containment Flooding)

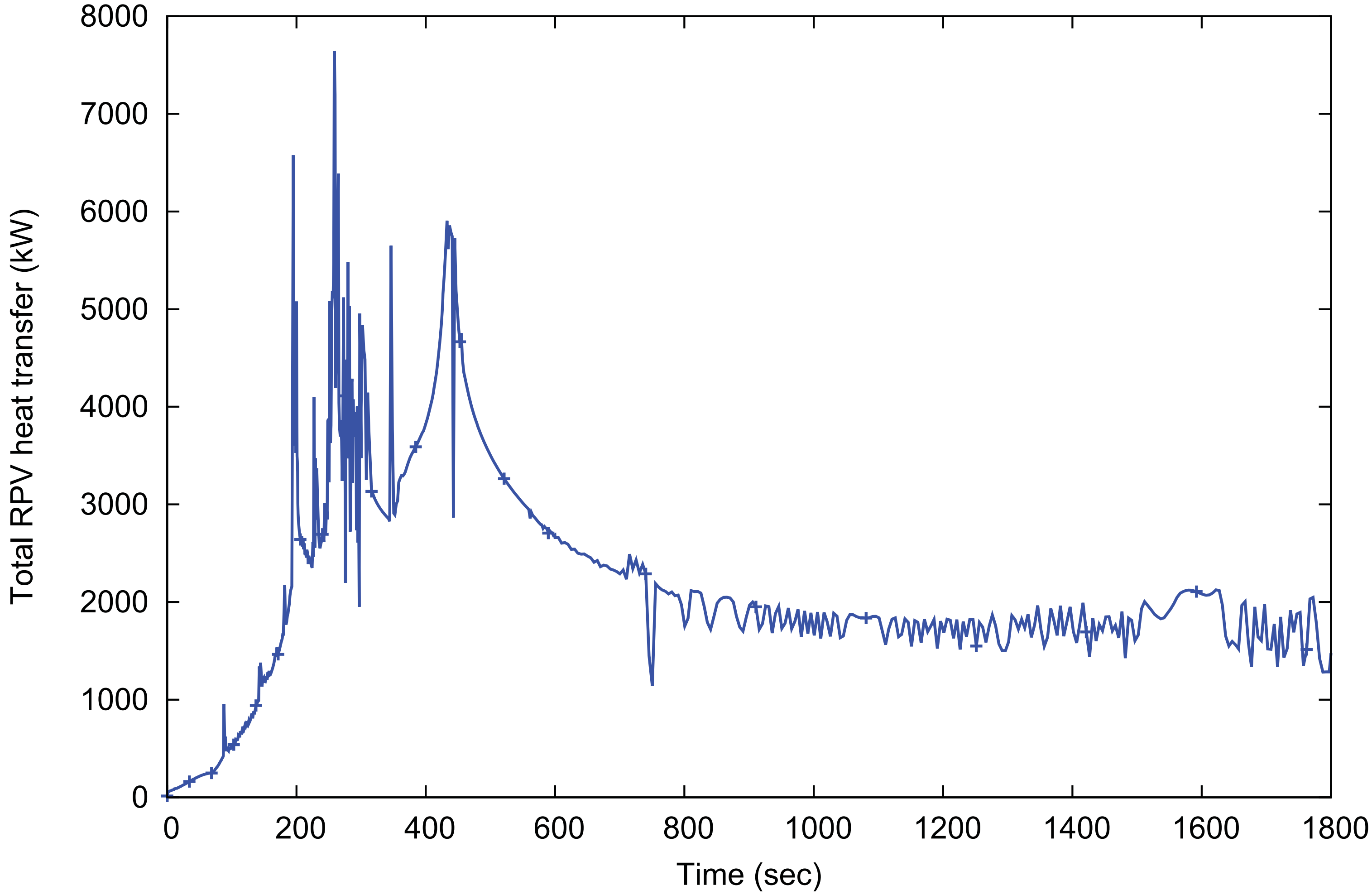


Figure 15.1-50: Containment Vessel Collapsed Liquid Level (15.1.6 Containment Flooding)

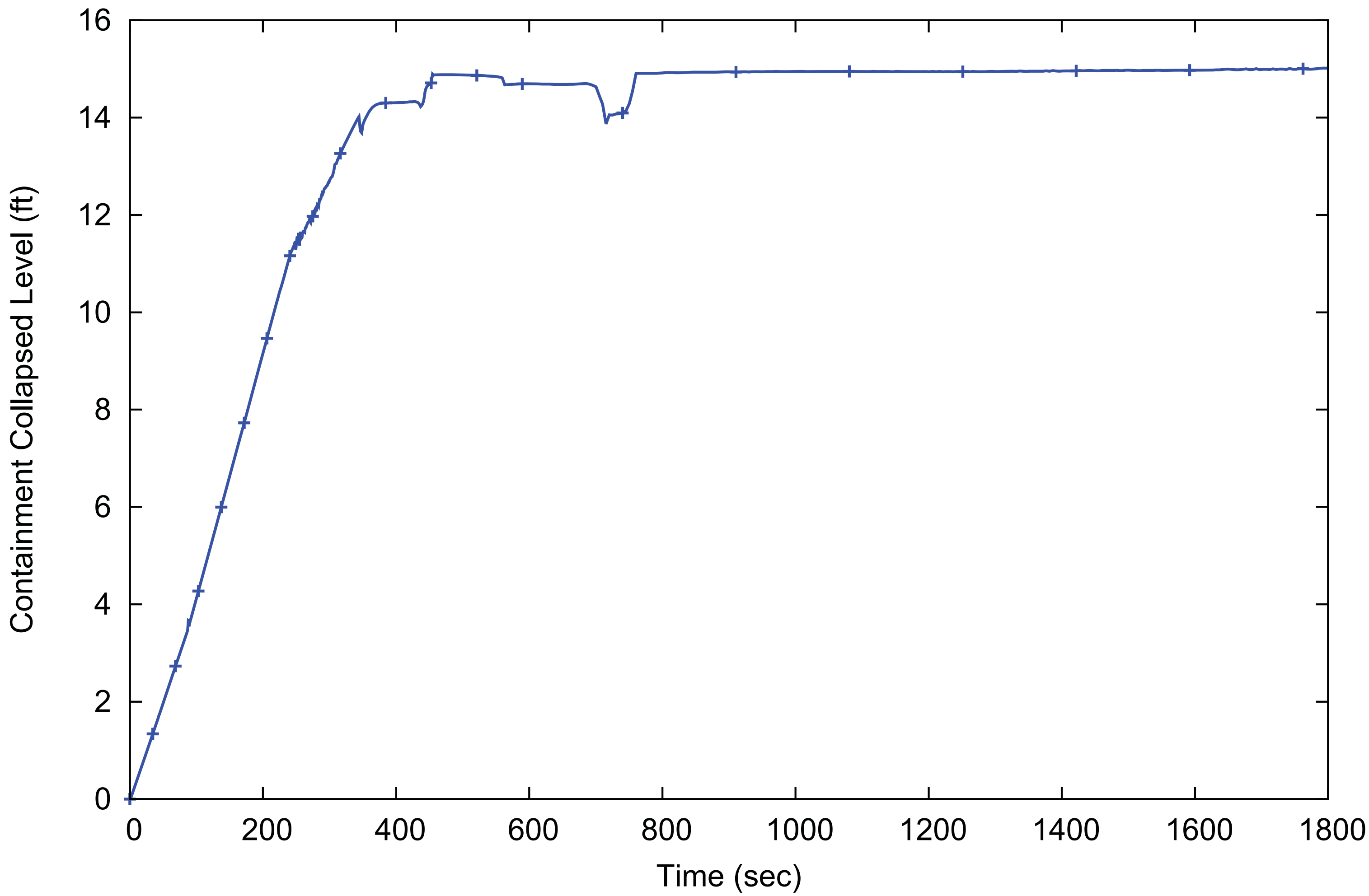


Figure 15.1-51: Containment Evacuation System Evacuation Mass Flow (15.1.6 Containment Flooding)

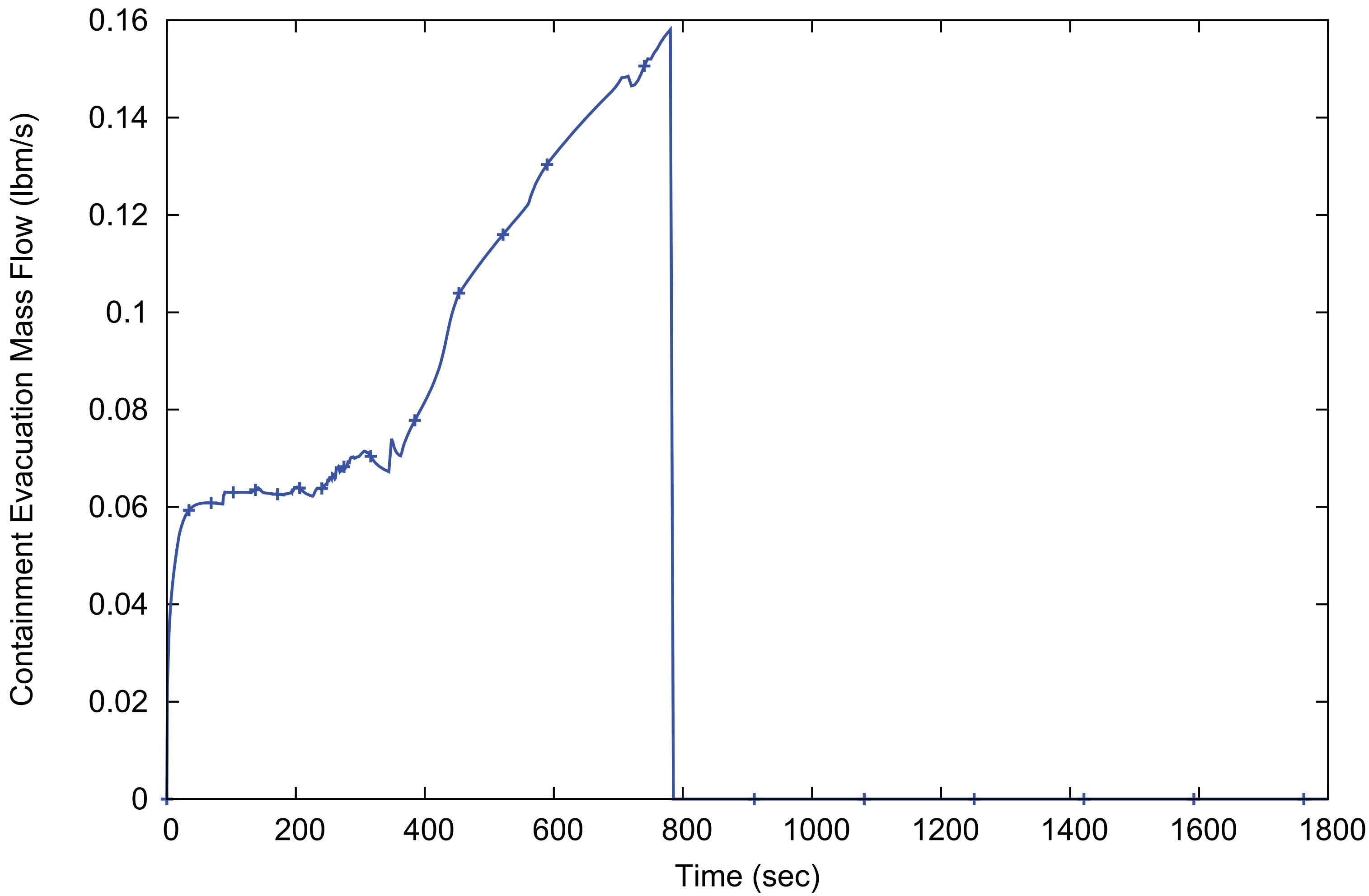


Figure 15.1-52: Containment Vessel Pressure (15.1.6 Containment Flooding)

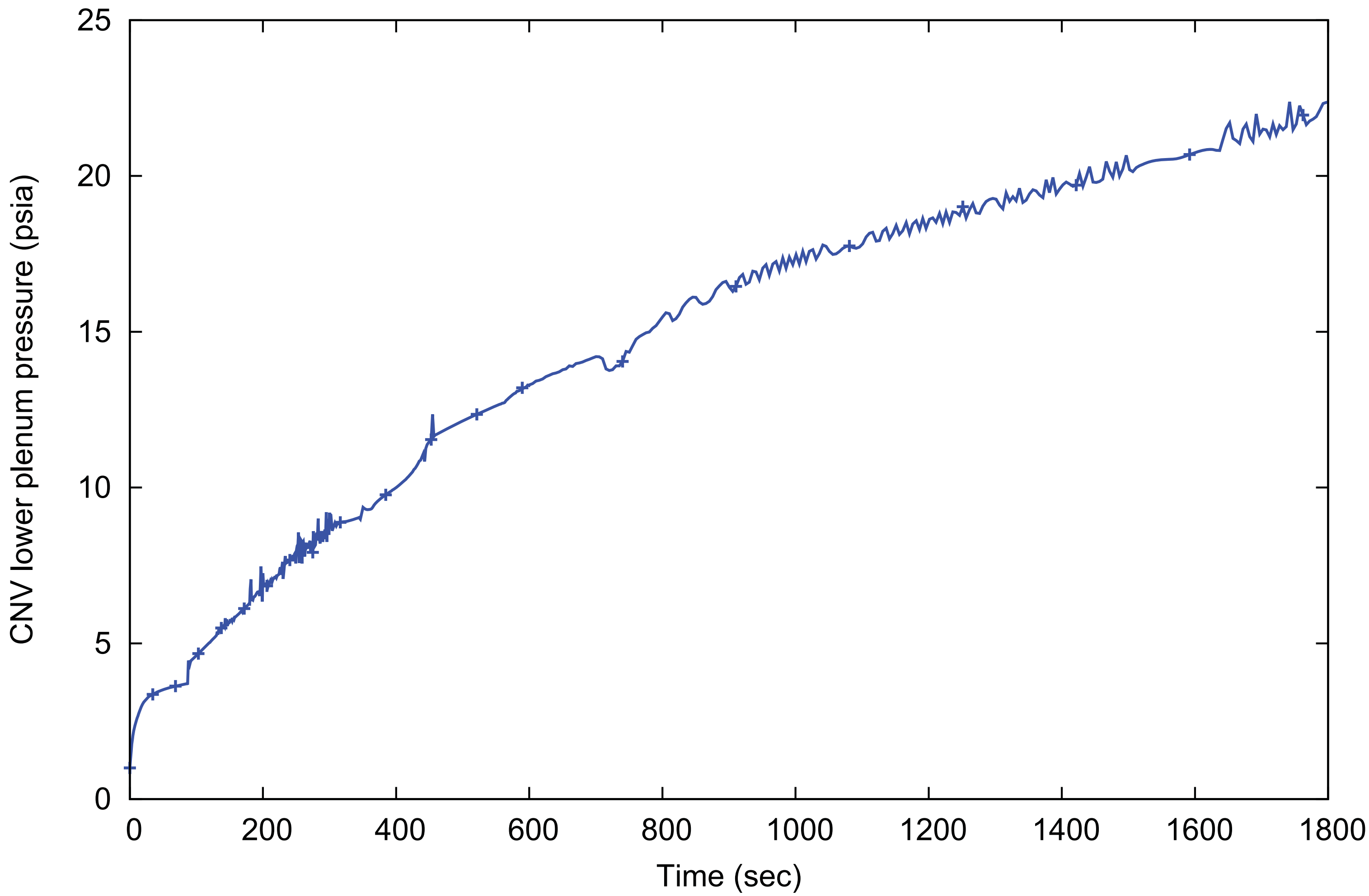


Figure 15.1-53: Critical Heat Flux Ratio (15.1.6 Loss of Containment Vacuum/Containment Flooding)

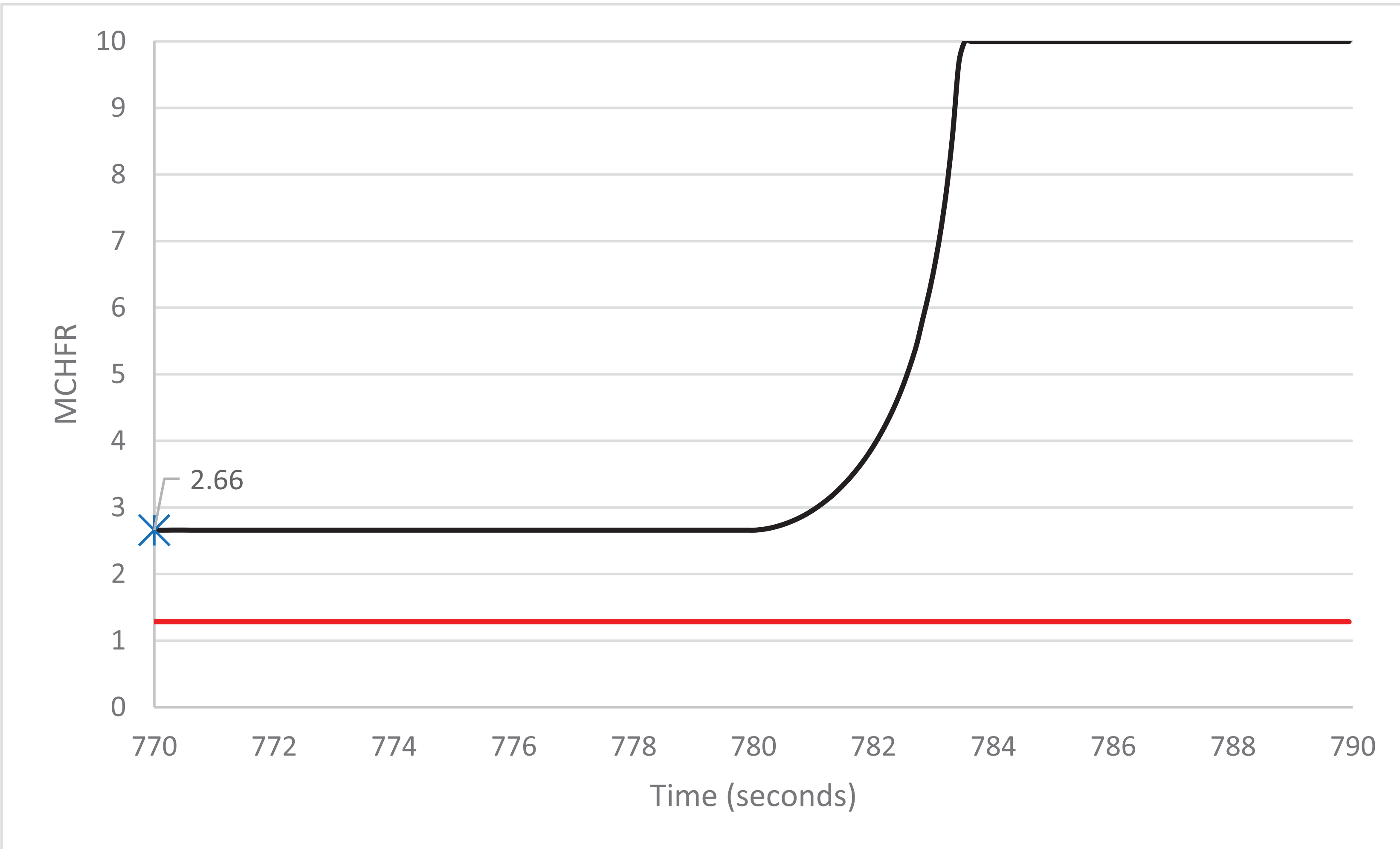


Figure 15.1-54: Core Outlet Temperature (15.1.1 Decrease in Feedwater Temperature)

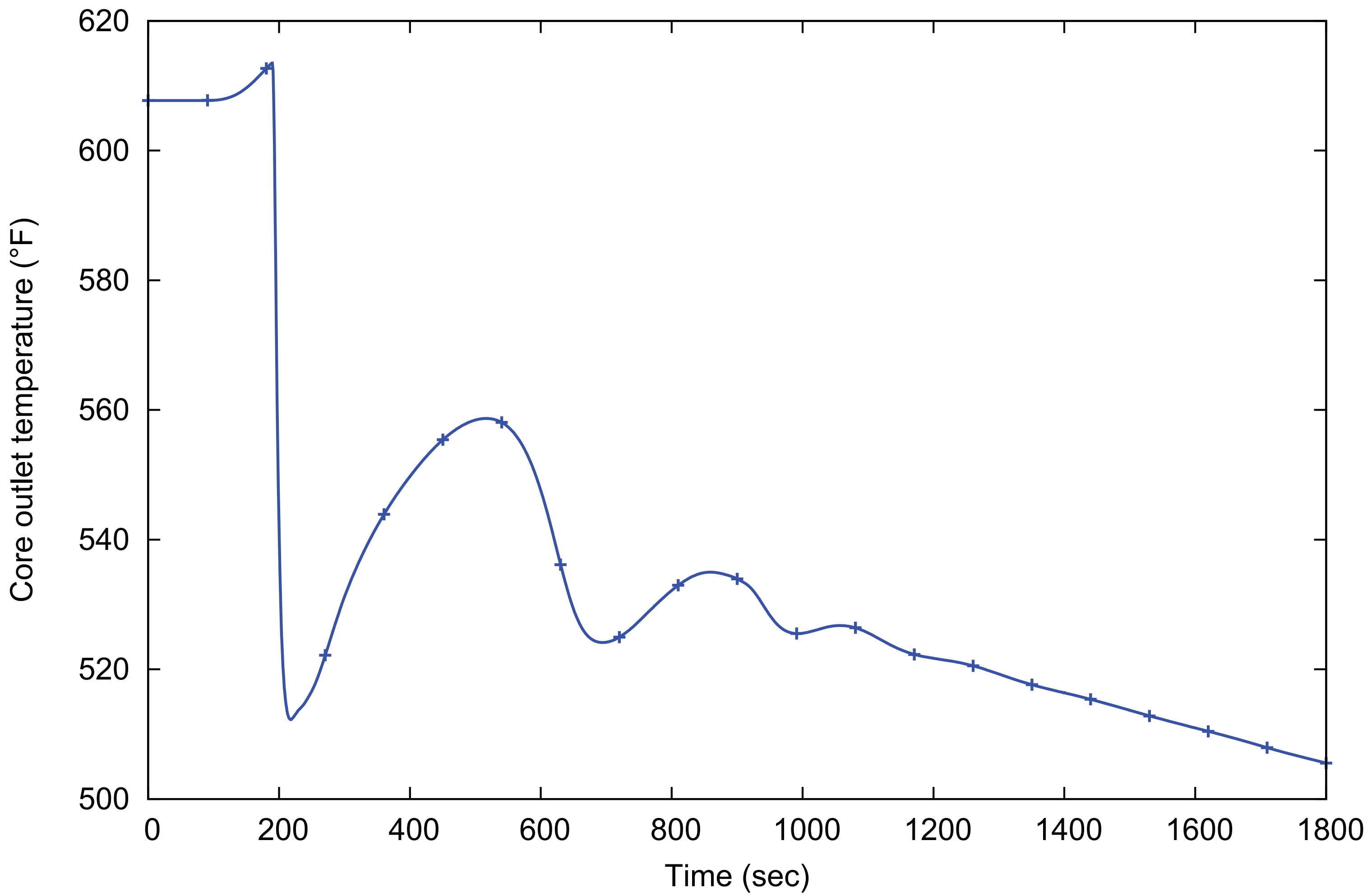


Figure 15.1-55: Total Reactivity (15.1.1 Decrease in Feedwater Temperature)

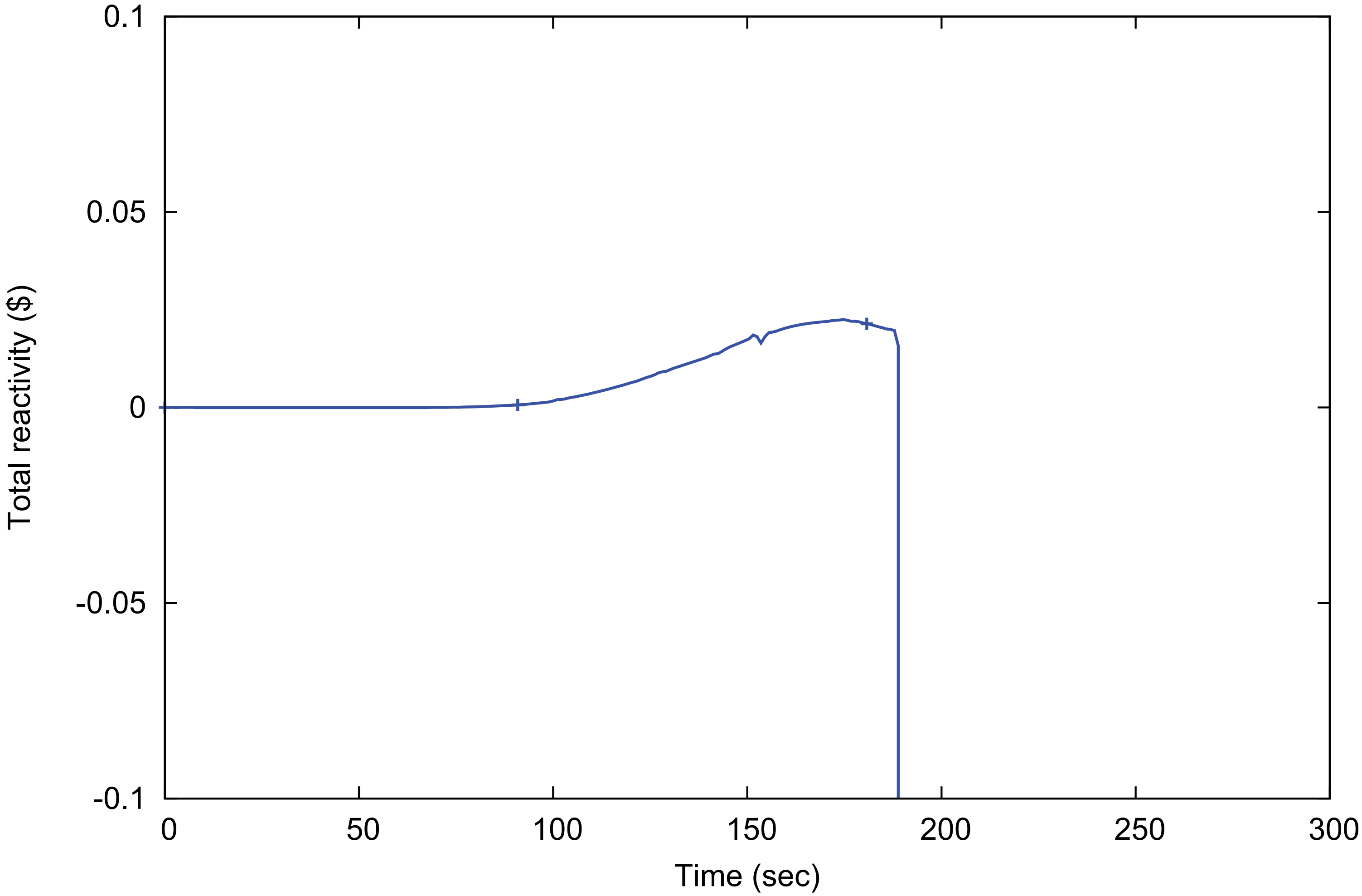


Figure 15.1-56: Steam Generator Pressure - Maximum SGS Pressure Case (15.1.2 Increase in Feedwater Flow)

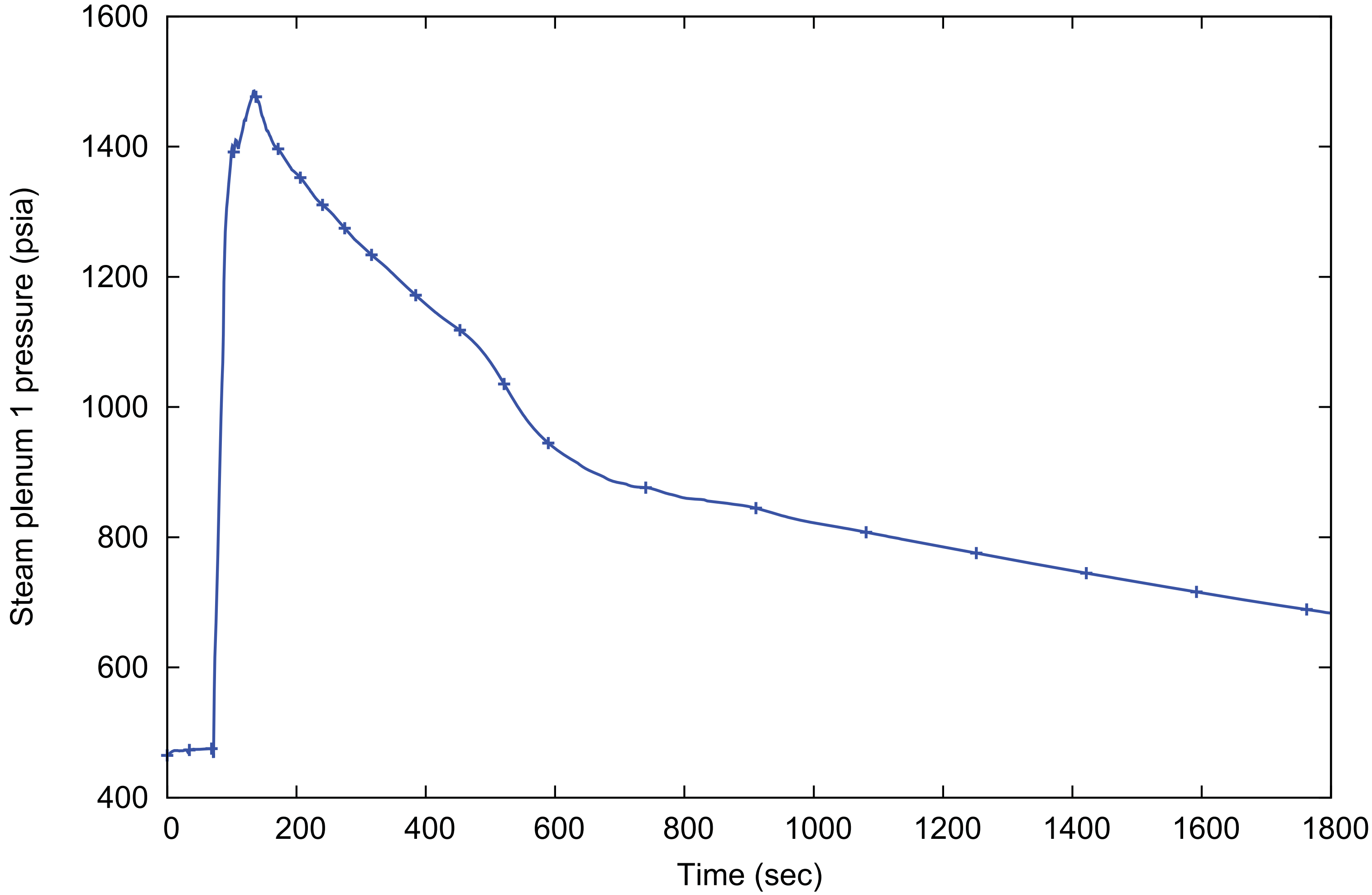


Figure 15.1-57: Not Used

Figure 15.1-58: Not Used

Figure 15.1-59: Not Used

Figure 15.1-60: Not Used

Figure 15.1-61: Intact Steam Generator Pressure (15.1.5 Steam Piping Failure, Limiting Steam Generator Pressure)

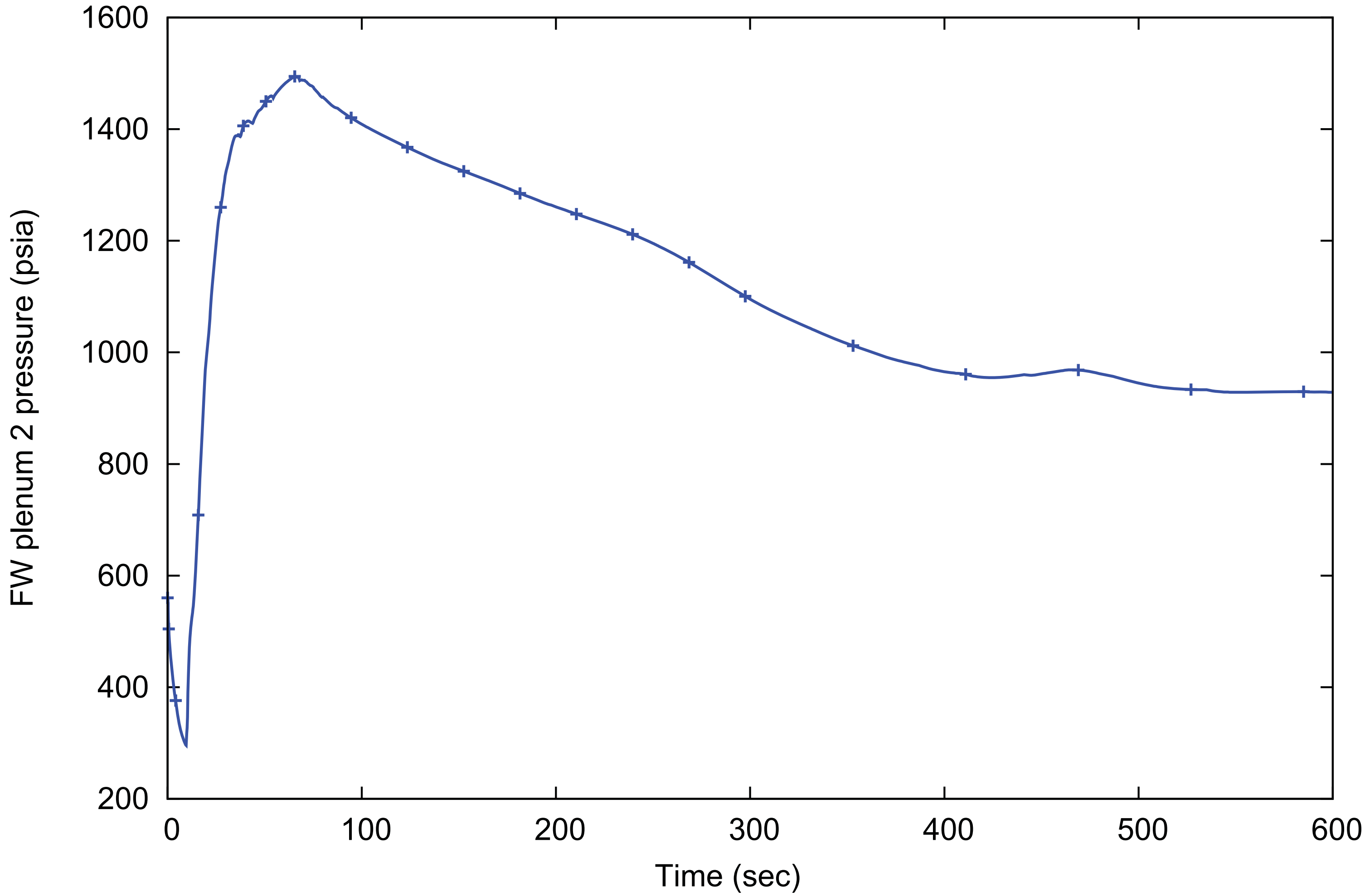
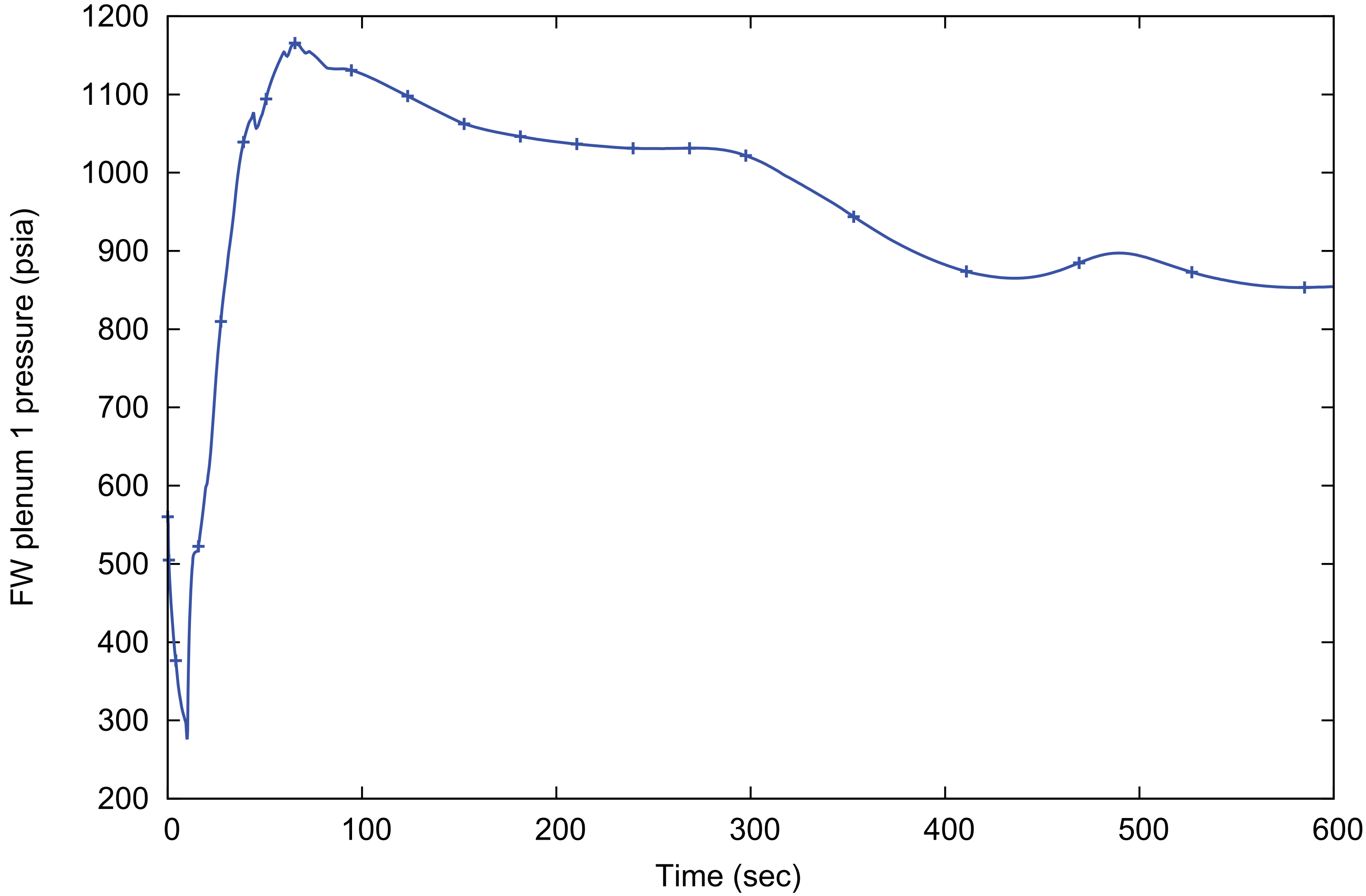


Figure 15.1-62: Faulted Steam Generator Pressure (15.1.5 Steam Piping Failure, Limiting Steam Generator Pressure)



15.2 Decrease in Heat Removal by the Secondary Side

This section addresses design basis events associated with a potential unplanned decrease in primary system heat removal through the steam generators (SGs). The decrease in heat removal causes the primary side temperature and pressure to rise and the pressurizer level to increase. SG pressure also increases. There are eight events that are defined for this category by the NuScale DSRS. One unique NuScale Power Module (NPM) event has been identified for this event type involving the decay heat removal system (DHRS) (Section 15.2.9).

The FSAR subsections are as follows:

- Section 15.2.1 - Loss of External Load
- Section 15.2.2 - Turbine Trip
- Section 15.2.3 - Loss of Condenser Vacuum
- Section 15.2.4 - Closure of Main Steam Isolation Valve (MSIV)
- Section 15.2.5 - Steam Pressure Regulator Failure
- Section 15.2.6 - Loss of Non-Emergency AC Power to the Station Auxiliaries
- Section 15.2.7 - Loss of Normal Feedwater Flow
- Section 15.2.8 - Feedwater System Pipe Breaks Inside and Outside of Containment
- Section 15.2.9 - Inadvertent Operation of the Decay Heat Removal System

The evaluation of the first three events in Section 15.2, Loss of External Load, Turbine Trip and Loss of Condenser Vacuum (LOEL, TT and LOCV) produced essentially identical responses for the primary and secondary system and minimum critical heat flux ratio (MCHFR). Therefore, a single set of figures based on an enveloping analysis of these events is provided to show the bounding cases for these events. The Inadvertent Closure of a Main Steam Isolation Valve (MSIV) figures are presented separately. Section 15.2.5, Steam Pressure Regulator Failure, is not applicable to the NPM and is kept primarily to maintain Section organization and numbering. Figures for Section 15.2.6 through Section 15.2.9 are also presented individually for each section.

The NuScale DSRS states that for new applications the loss of offsite power (LOOP) must be considered in addition to a single active failure. For NuScale, a LOOP is determined by a loss of AC power at the ELVS level (480V). The highly reliable DC power system (EDSS) and the non-safety DC power (EDNS) are not safety-related and their loss of function to provide power is considered as a possible concurrent event with the loss of ELVS. If the EDSS system fails, a reactor trip and containment isolation will occur and the emergency core cooling system (ECCS) valves will open when RCS pressure drops below the ECCS inadvertent actuation block (IAB) threshold. The timing of ECCS operation is after the time period of concern for evaluation of maximum RCS pressure, maximum steam pressure or MCHFR for decrease in heat removal events presented in this section. Therefore, the potential for ECCS operation is addressed in Section 15.0.5, Long Term Decay and Residual Heat Removal.

While there are various integrated, automatic control systems that are expected to keep the reactor at power when a turbine trip or loss of external load occurs, sensors, signal processing and final control elements that support these automated controls are not safety-related.

Therefore, mitigating control system responses are not credited for the events in this section, but their potential adverse impact to safety functions are considered.

15.2.1 Loss of External Load

15.2.1.1 Identification of Causes and Event Description

A loss of external load (LOEL) event is initiated by an electrical disturbance that results in the loss of a significant portion, or all, of the turbine generator load, leading to a turbine trip. The turbine trip causes the primary and secondary side temperatures and pressures to increase because energy is not being removed through the steam generators to the condenser. The reactor trip signal and secondary system isolation (SSI) are initiated on high pressurizer pressure. DHRS actuates on high pressurizer pressure. The reactor trip reduces power to decay heat levels. The SSI isolates the feedwater and steam systems and DHRS actuates and transfers decay heat to the reactor pool. If offsite power is lost, with a coincident loss of DC power (EDSS and EDNS), the reactor trip, DHRS actuation and SSI actuation happen concurrently.

A LOEL event is expected to occur one or more times in the life of the plant. Therefore, a LOEL event is an AOO as indicated in Table 15.0-1.

15.2.1.2 Sequence of Events and Systems Operation

The severity of a LOEL event is dictated by the time it takes DHRS to initiate and establish a stable cooldown rate. Secondary pressure is initially driven by the speed of the closure of the turbine control valves. However, following the valve closures, SG pressure continues to increase until DHRS establishes natural circulation. DHRS establishes cooling and begins to depressurize the SGs and the RCS approximately 80 seconds into the event. Key parameters (pressurizer level, reactor power, net reactivity, RCS average temperature, RCS flowrate, DHRS flow, and RCS reactor vessel pressure) are shown in Figure 15.2-1 through Figure 15.2-7 for the peak RCS pressure case.

Unless stated otherwise, the plant control systems (PCSs) and the engineered safety features (ESFs) perform as designed with allowances for instrument inaccuracy. For the LOEL event, containment pressure control is enabled because its operation is benign with respect to the event consequences. In contrast, most PCSs are disabled because their operation is beneficial with respect to the consequences for an LOEL event. The disabled PCSs are for reactor coolant system (RCS) temperature control, pressurizer pressure control, pressurizer level control, steam pressure control, and feedwater and turbine load control. No operator action is credited to mitigate the effects of an LOEL event.

For the limiting RCS pressure case, the module protection system (MPS) initiates a reactor trip, actuates DHRS and SSI, and deenergizes the pressurizer heaters on a high pressurizer pressure signal. The FW regulating valves and secondary MSIVs are nonsafety-related and close in 30 seconds and 7 seconds, respectively. These valves are credited as having redundant isolation capability in the case of the failure of the FWIVs or MSIVs to close. The limiting assumption for RCS pressure during an LOEL event is a loss of AC power, with DC power available, primarily due to the heatup caused by an

immediate loss of feedwater. No single failure resulted in a more limiting RCS pressure for the LOEL event. The RCS pressure increase is mitigated by opening one of the two redundant RSVs.

For the limiting SG pressure case, the MPS initiates a reactor trip and actuates SSI and DHRS on high steam pressure. The limiting single failure for an LOEL for peak SG pressure is the failure of a FWIV to close with AC and DC power available. Loss of AC and DC power would initiate MPS and ESF functions earlier in the event and therefore are not limiting. If a FWIV fails to close, FW flow will be provided to the SG until the FW pumps are secured or the FW regulating valves close. The feedwater regulating valves are nonsafety-related but are credited to close within 30 seconds in the event of a failure of the safety-related FWIV. The feedwater regulation valves get a close signal on SSI actuation. The valves also close on a loss of DC power (EDSS). The FWIV failure to close results in the highest SG peak pressure and in the worst case would result in degrading performance of one DHRS train due to overfilling. The remaining DHRS train is adequate for heat removal. The peak SG pressure is shown on Figure 15.2-8.

For the limiting MCHFR case, the MPS initiates a reactor trip and actuates the DHRS and SSI on high pressurizer pressure. The limiting assumption for MCHFR during an LOEL event is with AC power and DC power available. Loss of DC power would initiate MPS and ESF functions earlier in the event and, therefore, is not limiting. No single failure resulted in a more limiting MCHFR for the LOEL event. The RCS pressure reaches the RSV actuation setpoint in this event. The limiting MCHFR versus time is shown in Figure 15.2-9.

For the peak RCS pressure case, a loss of AC power provides the limiting results. Loss of DC power would initiate MPS functions earlier in the event and therefore is not limiting.

For the peak SG pressure and MCHFR cases, normal AC and DC power are assumed to be available. A loss of AC or DC power is not conservative for this event because the loss of power would terminate feedwater flow and actuate MPS functions earlier in the event sequence.

The enveloping sequence of events for either the LOEL, TT or LOCV transients are described in Table 15.2-4 for the limiting RCS pressure event, Table 15.2-5 for the limiting secondary pressure event, and Table 15.2-6 for the limiting MCHFR event.

15.2.1.3 Thermal Hydraulic and Subchannel Analyses

15.2.1.3.1 Evaluation Models

The thermal hydraulic analysis of the NPM response to a LOEL is performed using NRELAP5. The NRELAP5 model is based on the design features of a NuScale module. The non-LOCA NRELAP5 model is discussed in Section 15.0.2.2.2. The relevant boundary conditions from the NRELAP5 analyses are provided to the downstream subchannel Critical Heat Flux (CHF) analysis.

The subchannel core CHF analysis is performed using VIPRE-01. VIPRE-01 is a subchannel analysis tool designed for general-purpose thermal-hydraulic analysis under normal operating conditions, operational transients, and events of moderate

severity. See Section 15.0.2.3 for a discussion of the VIPRE-01 code and evaluation model.

15.2.1.3.2 Input Parameters and Initial Conditions

The reactivity feedback from the moderator temperature and fuel temperature is taken from the beginning of cycle (BOC). The reactivity coefficients for moderator temperature are least negative at BOC. Thus, they are conservative for undercooling events such as LOEL, since they minimize the negative reactivity insertion from the increase in coolant temperature.

Input conditions include an assumed power level of 102 percent. The most reactive rod is assumed to remain out of the core, with a delay of 2 seconds between the reactor trip signal and scram initiation. The most limiting combination of reactivity coefficients for moderator temperature and fuel temperature is applied. Instrument inaccuracy is accounted for by examining the sensitivity to the setpoints over the given margin of error.

The values for key input parameters and initial conditions for the evaluation of the LOEL event are listed in Table 15.2-1 for the maximum RCS pressure event, Table 15.2-2 for the maximum steam pressure event and Table 15.2-3 for the MCHFR event.

15.2.1.3.3 Results

As the RCS heats up, the expansion of water volume increases pressurizer level and pressure as shown in Figure 15.2-1 and Figure 15.2-7, respectively. Upon the reactor trip, power decreases as shown in Figure 15.2-2. Figure 15.2-3 presents the net reactivity from the control rod insertion. RCS temperature increases due to the heat up from loss of secondary cooling followed by a decrease due to the reactor trip and then increases due to the reduction in heat removal until DHRS begins to cool the primary system as plotted in Figure 15.2-4. RCS flow (Figure 15.2-5) drops due to the reactor trip and is reestablished as DHRS flow is established (Figure 15.2-6). Steam generator pressure for the peak SG pressure case is presented in Figure 15.2-8.

LOEL results in increased temperatures in the RCS which could potentially challenge fuel parameters. Although RCS fluid and fuel temperatures increase, the core remains covered throughout the event, such that the MCHFR limits are not challenged. The limiting MCHFR is demonstrated in Figure 15.2-9.

15.2.1.4 Radiological Consequences

The radiological consequences of an LOEL event are bounded by the design basis accident analyses presented in Section 15.0.3.

15.2.1.5 Conclusions

The six DSRS acceptance criteria for this AOO are met for the enveloping analysis which includes: Loss of External Load, Turbine Trip and Loss of Condenser Vacuum cases.

These acceptance criteria, followed by how the NuScale design meets them are listed below:

1) Pressure in the reactor coolant and main steam systems should be maintained below 110 percent of the design values.

    - The limiting RCS pressure for this event, shown in Table 15.2-7, is below 110 percent of the design value for the reactor coolant system.
    - The limiting steam generator pressure, shown in Table 15.2-7, is below 110 percent of the design value for the main steam system up to the MSIVs.

2) Fuel cladding integrity should be maintained by ensuring the minimum DNBR remains above the 95/95 DNBR limit based on acceptable correlations (See DSRS Section 4.4)

    - The MCHFR for this event, shown in Table 15.2-7, is above the 95/95 limit.

3) An incident of moderate frequency should not generate a more serious plant condition without other faults occurring independently.

    - The analyses presented for this event shows that stable DHRS cooling is reached, and the acceptance criteria are met. The LOEL event does not lead to a more serious event.

4) The guidance provided in RG 1.105, "Instrument Spans and Setpoints," can be used to analyze the impact of the instrument spans and setpoints on the plant response to the type of transient addressed in this DSRS section, in order to meet the requirements of GDCs 10 and 15.

    - Allowances for instrument inaccuracy are accounted for in the analytical limits of mitigating systems in accordance with the guidance provided in Regulatory Guide 1.105.

5) The most limiting plant systems single failure, as defined in the "Definitions and Explanations" of Appendix A to 10 CFR Part 50, shall be identified as assumed in the analysis and shall satisfy the positions of RG 1.53.

    - The limiting single failure is a failure of a feedwater isolation valve to close for the limiting SG pressure case. Results from this scenario do not challenge the identified limits.

6) The guidance provided in SECY-77-439, SECY-94-084 and RG 1.206 with respect to the consideration of the performance of nonsafety-related systems during transients and accidents, as well as the consideration of single failures of active and passive systems (especially as they relate to the performance of check valves in passive systems) must be evaluated and verified.

    - The inputs and assumptions for the operation of nonsafety-related systems and single failures as discussed in Section 15.2.1.2 and Section 15.2.1.3 and ensure the guidance provided is met.

15.2.2 Turbine Trip

15.2.2.1 Identification of Causes and Event Description

A turbine trip event is initiated by the closure of the turbine stop valves (TSVs). For the NPM design, the effect of a turbine trip is similar to LOEL and LOCV. No credit is taken for the turbine bypass system or other control systems and, therefore, numerous secondary side or electrical perturbations can result in a trip of the turbine generator. The turbine trip causes the primary and secondary side temperatures and pressures to increase because energy is not being removed through the steam generators to the condenser. The reactor trip signal, Decay Heat Removal System (DHRS) and secondary system isolation (SSI) actuation signals are initiated on high pressurizer pressure or high steam line pressure. The reactor trip reduces power to decay heat levels. The DHRS actuates and transfers decay heat to the reactor pool. If offsite power is lost, with a coincident loss of DC power, the reactor trip, DHRS actuation and SSI occur simultaneously.

A turbine trip event is expected to occur one or more times in the life of the plant. Therefore, a turbine trip event is an AOO as indicated in Table 15.0-1.

15.2.2.2 Sequence of Events and Systems Operation

The severity of a turbine trip is dictated by the time it takes DHRS to initiate and establish a stable cooldown rate. Secondary pressure is initially driven by the speed of the closure of the turbine stop valve (TSV). Following the valve closure, SG pressure continues to increase until DHRS establishes natural circulation. DHRS establishes cooling and begins to depressurize the SGs and the RCS approximately 80 seconds into the event.

Unless stated otherwise, the PCSs and the ESFs perform as designed with allowances for instrument inaccuracy. For the turbine trip event, containment pressure control is enabled because its operation is benign with respect to the event consequences. In contrast, most PCSs are disabled because their operation is beneficial with respect to the consequences for a turbine trip event. The disabled PCSs are for RCS temperature control, pressurizer pressure control, pressurizer level control, steam pressure control, and feedwater and turbine load control. No operator action is credited to mitigate the effects of a turbine trip event.

The description for the remaining sequence of the turbine trip event is the same as presented for the LOEL event in Section 15.2.1.2, including the loss of power scenarios and also including the failure of one feedwater isolation value (FWIV) to close. The enveloping sequence of events for either the LOEL, TT, or LOCV transients are described in Table 15.2-4 for the limiting RCS pressure event, Table 15.2-5 for the limiting secondary pressure events and Table 15.2-6 for the limiting MCHFR event.

15.2.2.3 Thermal Hydraulic and Subchannel Analyses

15.2.2.3.1 Evaluation Models

The thermal hydraulic analysis of the NPM response to a turbine trip event is performed using NRELAP5. The NRELAP5 model is based on the design features of a NuScale module. The non-LOCA NRELAP5 model is discussed in Section 15.0.2.2.2. The relevant boundary conditions from the NRELAP5 analyses are provided to the downstream subchannel Critical Heat Flux (CHF) analysis.

The subchannel core CHF analysis is performed using VIPRE-01. VIPRE-01 is a subchannel analysis tool designed for general-purpose thermal-hydraulic analysis under normal operating conditions, operational transients, and events of moderate severity. See Section 15.0.2.3 for a discussion of the VIPRE-01 code and evaluation model.

15.2.2.3.2 Input Parameters and Initial Conditions

The reactivity feedback from the moderator temperature and fuel temperature is taken from the Beginning of Cycle (BOC). The reactivity coefficients for moderator temperature are least negative at BOC. Thus, they are conservative for undercooling events such as turbine trip, since they minimize the negative reactivity insertion from the increase in coolant temperature.

Input conditions include an assumed power level of 102 percent. The most reactive rod is assumed to remain out of the core, along with a delay of 2 seconds between the reactor trip signal and the scram initiation. The most limiting combination of reactivity coefficients for moderator temperature and fuel temperature is applied. Instrument inaccuracy is accounted for by examining the sensitivity to the setpoints over the given margin of error. The key parameters are listed in Table 15.2-1 for the maximum RCS pressure event, Table 15.2-2 for the maximum SG pressure event and Table 15.2-3 for the limiting MCHFR event.

15.2.2.3.3 Results

The results for the turbine trip event are essentially the same as those presented in Section 15.2.1.3.3 for the LOEL event. However, as shown in Table 15.2-7, the TT event results in slightly higher maximum SGS pressure, but the TT event does not challenge the limits for RCS pressure, SGS pressure or MCHFR.

15.2.2.4 Radiological Consequences

The radiological consequences of a turbine trip event are bounded by the design basis accident analyses presented in Section 15.0.3.

15.2.2.5 Conclusions

The regulatory acceptance criteria for an AOO are met for the limiting turbine trip event. The results show significant margin between the system response and the design limits. The MCHFR for the limiting turbine trip event meets the acceptance

criterion discussed in Section 4.4.4, demonstrating that this event does not result in any fuel damage. The high steam line pressure or high pressurizer pressure reactor trips and subsequent actuation of the SSI and DHRS terminate this transient by inserting the control rods and removing the decay heat, resulting in a safe stable condition following the event. No operator actions were credited for mitigation of the turbine trip event. A turbine trip does not lead to a more serious event.

The analysis for the turbine trip shows that the NPM design is acceptable with respect to transients resulting in unplanned decreases in heat removal by the secondary system, transients expected with moderate frequency, and transients where the predicted response meets the acceptance criteria to maintain RCS and secondary piping pressure below 110 percent of the design value and MCHFR is maintained above the design limit. Details of the DSRS acceptance criteria are discussed in Section 15.2.1 for the LOEL Event but are also applicable for the turbine trip event. The numerical values for the acceptance criteria are listed in Table 15.2-7.

15.2.3 Loss of Condenser Vacuum

15.2.3.1 Identification of Causes and Event Description

A loss of condenser vacuum (LOCV) event involves a disturbance that results in an increase in condenser pressure due to air inleakage or a reduction in cooling to the condenser. For the NPM design, the effect of a loss of condenser vacuum results in a trip of the turbine generator and a loss of feedwater flow, which causes the primary side temperature and pressure to increase because energy is not being removed through the steam generators to the condenser. The reactor trips on high pressurizer pressure or high steam line pressure, reducing power to decay heat levels. The Decay Heat Removal System (DHRS) and secondary system isolation (SSI) actuate to isolate the steam and feedwater systems and transfer decay heat to the reactor pool.

A LOCV event is expected to occur one or more times in the life of the plant. Therefore, a LOCV event is an AOO as indicated in Table 15.0-1.

15.2.3.2 Sequence of Events and Systems Operation

The severity of a LOCV event is dictated by the time it takes DHRS to initiate and establish a stable cooldown rate. Secondary pressure is initially driven by the speed of the closure of the turbine stop valve (TSV). Following the valve closure, SG pressure continues to increase until DHRS establishes natural circulation. DHRS establishes cooling and begins to depressurize the SGs and the RCS approximately 80 seconds into the event.

Unless stated otherwise, the PCSs and the ESFs perform as designed with allowances for instrument inaccuracy. For the LOCV event, containment pressure control is enabled because its operation is benign with respect to the event consequences. In contrast, most PCSs are disabled because their operation is beneficial with respect to the consequences for an LOCV event. The disabled PCSs are for RCS temperature control, pressurizer pressure control, pressurizer level control, steam pressure control, and feedwater and turbine load control. No operator action is credited to mitigate the effects of an LOCV event.

The description for the remaining sequence of the LOCV event are the same as presented for the LOEL event in Section 15.2.1.2, including the loss of power scenarios and also including the failure of one feedwater isolation value (FWIV) to close. The enveloping sequence of events for either the LOEL, TT, or LOCV transients are presented in Table 15.2-4 for the limiting RCS pressure event, Table 15.2-5 for the limiting secondary pressure events and Table 15.2-6 for the limiting MCHFR event.

15.2.3.3 Thermal Hydraulic and Subchannel Analyses

15.2.3.3.1 Evaluation Models

The thermal hydraulic analysis of the NPM response to a LOCV event is performed using NRELAP5. The NRELAP5 model is based on the design features of a NuScale module. The non-LOCA NRELAP5 model is discussed in Section 15.0.2.2.2. The relevant boundary conditions from the NRELAP5 analyses are provided to the downstream subchannel Critical Heat Flux (CHF) analysis.

The subchannel core CHF analysis is performed using VIPRE-01. VIPRE-01 is a subchannel analysis tool designed for general-purpose thermal-hydraulic analysis under normal operating conditions, operational transients, and events of moderate severity. See Section 15.0.2.3 for a discussion of the VIPRE-01 code and evaluation model.

15.2.3.3.2 Input Parameters and Initial Conditions

The reactivity feedback from the moderator temperature and fuel temperature is taken from the Beginning of Cycle (BOC). The reactivity coefficients for moderator temperature are least negative at BOC. Thus, they are conservative for undercooling events such as turbine trip, since they minimize the negative reactivity insertion from the increase in coolant temperature.

Input conditions include an assumed power level of 102 percent. The most reactive rod is assumed to remain out of the core,along with a delay of 2 seconds between the reactor trip signal and scram initiation. The most limiting combination of reactivity coefficients for moderator temperature and fuel temperature is applied. Instrument inaccuracy is accounted for by examining the sensitivity to the setpoints over the given margin of error. The key parameters are listed in Table 15.2-1 for the maximum RCS pressure event, Table 15.2-2 for the maximum SG pressure event and Table 15.2-3 for the limiting MCHFR event.

15.2.3.3.3 Results

The results for the LOCV event are essentially the same as those presented in Section 15.2.1.3.3 for the LOEL event. However, as shown in Table 15.2-7, the LOCV event results in slightly higher maximum RCS pressure and slightly lower MCHFR, but the LOCV event does not challenge the limits on RCS pressure, SGS pressure or MCHFR.

15.2.3.4 Radiological Consequences

The radiological consequences of a LOCV event are bounded by the design basis accident analyses presented in Section 15.0.3.

15.2.3.5 Conclusions

The regulatory acceptance criteria for an AOO are met for the limiting LOCV event. The results show significant margin between the system response and the design limits. The MCHFR for the limiting LOCV event meets the acceptance criterion discussed in Section 4.4.4, demonstrating that this event does not result in any fuel damage. The high steam line pressure or high pressurizer pressure reactor trips and subsequent actuation of the DHRS and SSI terminate this transient by inserting the control rods and removing the decay heat, resulting in a safe stable condition following the event. No operator actions were credited for mitigation of the LOCV event. An LOCV event does not lead to a more serious event.

The analysis for LOCV shows that the NPM design is acceptable with respect to transients resulting in unplanned decreases in heat removal by the secondary system, transients expected with moderate frequency, and transients where the predicted response meets the acceptance criteria to maintain RCS and secondary piping pressure below 110 percent of the design value and MCHFR is maintained above the design limit.

Details for meeting the DSRS acceptance criteria are discussed in Section 15.2.1 for the LOEL Event but are also applicable to the LOCV Event.

15.2.4 Closure of Main Steam Isolation Valve(s)

15.2.4.1 Identification of Causes and Accident Description

Closure of one or more MSIVs increases secondary side pressure and temperature in the affected steam generator(s) (SG). For the NPM design, the pressure and temperature also increase on the primary side because of the decrease in cooling. Closure of either one or two MSIVs causes SG pressures increase rapidly to the module protection system (MPS) setpoint for secondary pressure to resulting in a reactor trip and actuation of the decay heat removal system (DHRS) and secondary system isolation (SSI).

Unintended actuation of a single MSIV could be caused by an inadvertent MPS actuation signal to the valve or a loss of DC (EDSS) power to the valve. A single valve could also close on a valid MSIV closure signal with a failure of the opposite train MSIV to close. An inadvertent MPS signal to both trains of valves or a loss of EDSS power to both MSIVs could cause both valves to close. The MSIVs could also be inadvertently closed due to operator error.

A MSIV closure event is expected to occur one or more times in the life of the plant. Therefore, a MSIV closure event is an AOO as indicated in Table 15.0-1.

15.2.4.2 Sequence of Events and Systems Operation

The inadvertent MSIV closure event was analyzed for one and both MSIVs in parallel lines closing. The safety-related MSIVs are designed to close within 5 seconds. For conservatism, this analysis assumes the MSIVs close immediately at the event initiation, which maximizes the primary and secondary temperature and pressure transients. Each steam line also has a backup nonsafety-related MSIV that has a designed closure time of less than 7 seconds. The nonsafety-related MSIVs are credited to close within 7 seconds if the safety-related MSIV fails to close.

The peak RCS pressure occurs following closure of two MSIVs with a loss of offsite power. The peak SG pressure was also closure of two MSIVs with offsite power available. The differences between the peak primary and secondary pressure cases is due to the assumed biases and initial conditions. The lowest MCHFR also occurs with the closure of two MSIVs with a loss of AC power. DC power sources (EDNS and EDSS) were assumed to be available for all scenarios as the loss of DC power resulted in earlier reactor trip and actuation of ESF mitigating functions.

The sequence of events for the limiting scenarios is shown in Table 15.2-11, Table 15.2-12 and Table 15.2-13. The single MSIV closure was not limiting for any of the acceptance criteria.

The Module Protection System (MPS) is credited to protect the plant in the event of MSIV closure. The MPS actuation trips the reactor and initiates SSI and DHRS, preventing the plant from reaching conditions where the acceptance criteria could be challenged. Both high pressurizer pressure and high SG pressure MPS signals are credited in this event. The sensitivity of the transient response to the initial pressurizer pressure and level bias is evaluated. One of the two redundant RSVs is required to open to mitigate the increase in RCS pressure.

The analysis of the MSIV closure event assumes that plant control systems perform as designed, with allowance for instrument inaccuracy and biases as provided in Table 15.2-8, Table 15.2-9 and Table 15.2-10.

No single failures resulted in higher RCS or SG pressure or lower MCHFR. No operator action is credited for the MSIV closure event.

For cases with AC power available, the Chemical and Volume Control System was assumed to maintain normal recirculation flow with no makeup. The pressurizer spray system was not assumed to function for this event as it reduced the severity of RCS pressurization.

The inadvertent MSIV closure event is a module-specific event and does not affect other modules. The trip of the turbine is not expected to affect the electrical grid or result in loss of offsite power.

15.2.4.3 Core and System Performance

15.2.4.3.1 Evaluation Model

The thermal hydraulic analysis of the NPM response to a MSIV closure event is performed using NRELAP5. The NRELAP5 model is based on the design features of a NuScale module. The non-LOCA NRELAP5 model is discussed in Section 15.0.2.2.2. The relevant boundary conditions from the NRELAP5 analyses are provided to the downstream subchannel Critical Heat Flux (CHF) analysis.

The subchannel core CHF analysis is performed using VIPRE-01. VIPRE-01 is a subchannel analysis tool designed for general-purpose thermal-hydraulic analysis under normal operating conditions, operational transients, and events of moderate severity. See Section 15.0.2.3 for a discussion of the VIPRE-01 code and evaluation model.

15.2.4.3.2 Input Parameters and Initial Conditions

The initial power level was assumed to be 102 percent of reactor thermal power (163.2 MW). A signal delay of 2 seconds was assumed on the reactor trip, SSI, DHRS actuation (30-second valve opening time), and FWIV closure. MSIV closure was assumed to occur at the beginning of the event with no signal delay. The input parameters for the evaluation model for the MSIV closure events are provided in Table 15.2-8, Table 15.2-9 and Table 15.2-10. The input parameters were developed to provide a conservative result for the events evaluated for RCS pressure, SG pressure and MCHFR. No credit is taken for pressurizer spray or decreasing pressurizer heater power.

15.2.4.3.3 Results

For all of the MSIV closure events steam pressure rises quickly and the MPS initiates a reactor trip, SSI and DHRS actuation. No single failures cause the results of the limiting events to become more severe. The assumption of a loss of offsite power does result in the limiting primary pressure and MCHFR. For the other acceptance criteria, the initial condition biases provide the limiting scenarios.

Following the closure of both MSIVs, the RCS heats up and RCS pressure increases as shown in Figure 15.2-11. The expansion of water volume increases pressurizer level as shown in Figure 15.2-15. Upon the reactor trip, power decreases as shown in Figure 15.2-10. Figure 15.2-12 presents the volume average fuel temperature. RCS temperature initially increases followed by a decrease due to the reactor trip and then increases due to the reduction in heat removal until DHRS begins to cool the primary system as plotted in Figure 15.2-14. Steam generator pressure and level for the peak SG pressure case are presented in Figure 15.2-16 and Figure 15.2-17.

For the peak RCS pressure case, the reactor safety valve lifts to maintain primary pressure below design limits. The RSV lifts for a short time and containment pressure increases only slightly and is not expected to reach the containment isolation setpoint.

The single MSIV closure was not limiting for any of the acceptance criteria. For the single MSIV closure event, the slower valve closing time for the nonsafety-related MSIVs was also evaluated and did not result in any limiting values. The results of the events for the inadvertent closure of MSIVs were well within the AOO acceptance criteria.

The subchannel analysis indicates that the MCHFR is above the limit and is acceptable. Figure 15.2-18 shows the MCHFR for the most limiting event, concurrent closure of both MSIVs with a loss of AC power.

15.2.4.4 Radiological Consequences

The radiological consequences of a MSIV closure event are bounded by the design basis accident analyses presented in Section 15.0.3.

15.2.4.5 Conclusions

The six DSRS acceptance criteria for this AOO are met for the MSIV closure cases. These acceptance criteria, followed by how the NuScale design meets them are listed below:

1) Pressure in the reactor coolant and main steam systems should be maintained below 110 percent of the design values.

    - The limiting RCS pressure for this event, listed in Table 15.2-14, is below 110 percent of the design value for the reactor coolant system.
    - The limiting steam generator pressure, listed in Table 15.2-14, is below 110 percent of the design value for the main steam system up to the MSIVs.

2) Fuel cladding integrity should be maintained by ensuring the minimum DNBR remains above the 95/95 DNBR limit based on acceptable correlations (See DSRS Section 4.4)

    - The MCHFR for this event, listed in Table 15.2-14, is above the 95/95 limit.

3) An incident of moderate frequency should not generate a more serious plant condition without other faults occurring independently.

    - The analyses presented for this event shows that stable DHRS cooling is reached, and the acceptance criteria are met.

4) The guidance provided in RG 1.105, "Instrument Spans and Setpoints," can be used to analyze the impact of the instrument spans and setpoints on the plant response to the type of transient addressed in this DSRS section, in order to meet the requirements of GDCs 10 and 15.

    - Allowances for instrument inaccuracy are accounted for in the analytical limits of mitigating systems in accordance with the guidance provided in Regulatory Guide 1.105.

5) The most limiting plant systems single failure, as defined in the "Definitions and Explanations" of Appendix A to 10 CFR Part 50, shall be identified as assumed in the analysis and shall satisfy the positions of RG 1.53.

- No single failure for an MSIV Closure event resulted in greater challenges to the acceptance criteria. Results from this scenario do not challenge the identified limits as described in this Section.

6) The guidance provided in SECY-77-439, SECY-94-084 and RG 1.206 with respect to the consideration of the performance of nonsafety-related systems during transients and accidents, as well as the consideration of single failures of active and passive systems (especially as they relate to the performance of check valves in passive systems) must be evaluated and verified.

    - The inputs and assumptions for the operation of nonsafety-related systems and single failures as discussed in Section 15.2.4.2 and Section 15.2.4.3.2 ensure the guidance provided is met.

15.2.5 Steam Pressure Regulator Failure (Closed)

The event classification report does not include this event because there is no steam pressure regulator in the NuScale design. The Section number will be retained as a placeholder.

15.2.6 Loss of Non-Emergency AC Power to the Station Auxiliaries

15.2.6.1 Identification of Causes and Event Description

A loss of AC power to Station Auxiliaries can occur from the following:

- Failures in the electrical grid
- Failures in the plant equipment
- Failures in switchyard equipment
- External weather events.

For the NuScale design, the loss of AC power results in the turbine generator tripping and a loss of pumps in the secondary. The primary side temperature and pressure increase due to heat no longer being removed through the steam generators. The pressure increase on the primary or secondary initiates a reactor trip which reduces power to decay heat levels and actuates SSI and DHRS to transfer heat from the primary system to the reactor pool.

A loss of AC power event is expected to occur one or more times in the life of the NPM, so it is classified as an AOO. The categorization of the NuScale design basis events are discussed in Section 15.0.0, Table 15.0-1.

15.2.6.2 Sequence of Events and Systems Operation

Unless specified below, the analysis of a loss of AC power event assumes the plant control systems and MPS perform as designed, with allowances for instrument inaccuracy. No operator action is credited to mitigate the effects of a loss of AC power event.

The loss of AC power event for the NuScale design assesses the loss of the low voltage AC power distribution system (ELVS), which supplies power to plant motors, heaters, packaged equipment and battery chargers. The severity of the loss of AC power event is dictated by the timing of the reactor trip and the loss of power. For all limiting cases, the transient is more severe if there is a loss of AC power with both the EDSS and EDNS DC power supplies still available. Loss of DC power initiates a reactor trip and ESF actuations earlier in the transient and results in less limiting conditions.

In the limiting RCS pressure case, power is lost to pressurizer heaters; condensate, feed, and CVCS pumps; and the turbine trips immediately. The loss of cooling in the secondary causes a heatup and increase in pressure in the RCS which initiates a reactor trip and actuates SSI and DHRS on the high pressurizer pressure MPS signal. Following closure of the MSIVs and FWIVs, the peak RCS pressure occurs coincident with the lifting of an RSV, after which system pressure quickly decreases. Secondary pressure increases until the DHRS valves are fully open and natural circulation is established.

Single failures were considered, however, there was no limiting failure that affected either primary or secondary pressure. No single failure of a FWIV or MSIV to close increased peak RCS pressure as secondary flow is lost at event initiation. No single failure reduced the MCHFR as the limiting value occurs prior to secondary system isolation signal being generated.

Two event sequences for the loss of AC power event are provided in Table 15.2-15 and Table 15.2-16 for the limiting scenarios considering biased boundary conditions.

15.2.6.3 Thermal Hydraulic and Subchannel Analyses

15.2.6.3.1 Evaluation Models

The thermal hydraulic analysis of the NPM response to a loss of AC power event is performed using NRELAP5. The NRELAP5 model is based on the design features of a NuScale module. The non-LOCA NRELAP5 model is discussed in Section 15.0.2.2.2. The relevant boundary conditions from the NRELAP5 analyses are provided to the downstream subchannel Critical Heat Flux (CHF) analysis.

The subchannel core CHF analysis is performed using VIPRE-01. VIPRE-01 is a subchannel analysis tool designed for general-purpose thermal-hydraulic analysis under normal operating conditions, operational transients, and events of moderate severity. See Section 15.0.2.3 for a discussion of the VIPRE-01 code and evaluation model.

15.2.6.3.2 Input Parameters and Initial Conditions

The initial conditions used in the evaluation of the limiting loss of AC power event result in a conservative calculation. The following initial conditions are assumed in the analysis to ensure that the results have sufficient conservatism:

- Initial power level is assumed to be 102 percent of nominal to account for a 2 percent measurement uncertainty.

- Conservative scram characteristics are used, including a maximum time delay, holding the most reactive rod out of the core, and utilizing a bounding control rod drop rate.
- The most limiting beginning of cycle core parameters are used to provide a limiting power response. The most positive MTC of 0.0 pcm/degrees F and least negative DTC of -1.40 pcm/degrees F are used to provide the least negative reactivity insertion as temperature increases due to undercooling.
- Allowances for instrument inaccuracy are accounted for in the analytical limits of mitigating systems in accordance with the guidance provided in RG 1.105

The results from the thermal hydraulic evaluation are used as input to the subchannel analysis to determine the MCHFR for this event. Other key inputs and assumptions used in the thermal hydraulic and subchannel analyses are provided in Table 15.2-18.

15.2.6.3.3 Results

Sensitivity studies on a loss of AC power event were performed. Cases were performed with different loss of power scenarios based on nominal and biased boundary conditions in an attempt to maximize the primary and secondary pressures. For cases that did not result in an immediate reactor trip upon loss of power, the pressurization was found to be higher.

Upon the reactor trip, power decreases as shown in Figure 15.2-19. Figure 15.2-20 presents the total reactivity following the control rod insertion. As the RCS heats up, the expansion of water volume increases pressurizer level (Figure 15.2-21). RCS pressure increases as shown in Figure 15.2-22. The increase in reactor coolant temperature causes the primary coolant volume to expand, raising RCS pressure. The peak RCS pressure occurs with the lifting of the RSV, which quickly reduces pressure below the RSV reset pressure and ensures RCS pressure is maintained below the acceptance criteria. RCS temperature decreases due to the reactor trip and then increases due to the reduction in heat removal because of SSI until DHRS begins to cool the primary system as plotted in Figure 15.2-24. RCS flow (Figure 15.2-23) drops initially due to the reactor trip and is reestablished as DHRS flow is established.

Steam generator pressure increases significantly during the initial phase of the transient following the tripping of the turbine stop valves and SSI, but is limited by the cooling provided by DHRS. The maximum secondary pressure is well below the design pressure of the steam system piping. Steam generator pressure for the peak SG pressure case is presented in Figure 15.2-25.

The predicted lowest MCHFR was greater than the acceptance criteria for this event, no fuel failure is predicted to occur. The limiting MCHFR versus time is plotted in Figure 15.2-26.

15.2.6.4 Radiological Consequences

The radiological consequences of a loss of AC power event are bounded by the design basis accident analyses presented in Section 15.0.3.

15.2.6.5 Conclusions

The five DSRS acceptance criteria for this AOO are met for the limiting loss of AC power cases. These acceptance criteria, followed by how the NuScale design meets them are listed below:

1) Pressure in the reactor coolant and main steam systems should be maintained below 110 percent of the design values.
   - The limiting RCS pressure for this event, listed in Table 15.2-19, is below 110 percent of the design value for the reactor coolant system.
   - The limiting steam generator pressure, listed in Table 15.2-19, is below the 110 percent design value for the main steam system up to the MSIV.
2) Fuel cladding integrity should be maintained by ensuring the minimum DNBR remains above the 95/95 DNBR limit based on acceptable correlations.
   - The MCHFR for this event, listed in Table 15.2-19, is above the 95/95 limit.
3) An incident of moderate frequency should not generate a more serious plant condition without other faults occurring independently.
   - The analyses presented for this event shows that stable DHRS cooling is reached, and the acceptance criteria for an AOO are met.
4) The guidance provided in RG 1.105, "Instrument Spans and Setpoints," can be used to analyze the impact of the instrument spans and setpoints on the plant response to the type of transient addressed in this DSRS section, in order to meet the requirements of GDCs 10 and 15.
   - Allowances for instrument inaccuracy are accounted for in the analytical limits of mitigating systems in accordance with the guidance provided in Regulatory Guide 1.105.
5) The most limiting plant systems single failure, as defined in the "Definitions and Explanations" of Appendix A to 10 CFR Part 50, shall be identified and assumed in the analysis and shall satisfy the positions of RG 1.53.
   - No single failures resulted in increased RCS or SG pressure for the limiting events.

The results of this analysis show that a loss of AC power event does not adversely affect the core, the reactor coolant system, or the steam system, and the regulatory acceptance criteria for AOOs are met.

15.2.7 Loss of Normal Feedwater Flow

15.2.7.1 Identification of Causes and Event Description

A loss of normal feedwater flow could occur from the following scenarios:

- Pump failures
- Valve malfunctions
- Loss of AC power

For NuScale design, a loss of normal feedwater is a decrease in heat removal via the steam generators resulting in an increase in RCS temperature and pressure which lead to a reactor trip. A loss of feedwater can occur due to a feedwater system pipe break; however, such an event is addressed as an accident, separately in Section 15.2.8.

The loss of normal feedwater flow event is expected to occur one or more times in the life of the plant, so it is classified as an AOO. The categorization of the NuScale design basis events are discussed in Section 15.0.0, Table 15.0-3.

15.2.7.2 Sequence of Events and Systems Operation

Unless specified below, the analysis of a loss of normal feedwater flow event assumes the plant control systems and engineered safety features perform as designed, with the allowances for instrument inaccuracy. No operator action is credited to mitigate the effects of a loss or normal feedwater flow event. DC power (EDNS and EDSS) is assumed to be available throughout the events evaluated because the loss of DC power actuates MPS and RTS functions earlier in the transient and produces less limiting results for all acceptance criteria.

Two event sequences provide limiting results for the applicable acceptance criteria. The first LOFW limiting case results in the limiting RCS pressure and MCHFR event. Water level in the steam generators decreases causing primary- to-secondary heat transfer rates to decrease and RCS temperature and pressures to increase. The reactor is tripped and DHRS and SSI are actuated on the high pressurizer pressure MPS signal. The MCHFR is reached before the time of reactor trip. The loss of normal AC power is coincident with the turbine trip. Insertion of control rods reduces reactor power and subsequently, core flow. The conservative delay in DHRS valve actuation results in continued pressure increase in the primary. The peak RCS pressure coincides with the lifting of the RSV, which decreases primary pressure and maintains it less than 110 percent of the design pressure. The establishment of DHRS flow results in cooling the RCS.

The second limiting case results in the peak secondary pressure. The event is initiated from a fault that results in a partial loss of feedwater flow. The reduced feedwater flow causes the steam generator water level to fall, resulting in a decrease in primary-to-secondary heat transfer; increasing both RCS temperature and pressure. The hot leg temperature reaches the high analytical limit for reactor trip, SSI, and DHRS actuation after a considerable instrument response delay. Normal power is assumed available during the transient. RCS pressure reaches a maximum and begins to

decrease prior to lifting an RSV and before DHRS is fully actuated. The maximum secondary pressure is reached while stable DHRS flow is being established, but is over 500 psia from exceeding the design pressure of the secondary. The establishment of DRHS flow results in restored cooling of the module.

The event sequences for the LOFW event are provided in Table 15.2-20 and Table 15.2-21 for the limiting scenarios.

The MPS is credited to protect the plant in the event of a loss of normal feedwater flow. The high pressurizer pressure signal and the high hot leg temperature are the only MPS signals that are credited in providing the plant protection. One of the two RSVs is credited for mitigating RCS overpressure.

No single failures were found to have adverse impact on the primary or secondary peak pressure or the MCHFR.

15.2.7.3 Thermal-Hydraulic and Subchannel Analysis

15.2.7.3.1 Evaluation Model

The thermal hydraulic analysis of the NPM response to a LOFW event is performed using NRELAP5. The NRELAP5 model is based on the design features of a NuScale module. The non-LOCA NRELAP5 model is discussed in Section 15.0.2.2.2. The relevant boundary conditions from the NRELAP5 analyses are provided to the downstream subchannel Critical Heat Flux (CHF) analysis.

The subchannel core CHF analysis is performed using VIPRE-01. VIPRE-01 is a subchannel analysis tool designed for general purpose thermal hydraulic analysis under normal operation conditions, operational transients, and events of moderate severity. See Section 15.0.2 for a discussion of the VIPRE-01 code and evaluation model.

15.2.7.3.2 Input Parameters and Initial Conditions

The initial conditions used in the evaluation of the limiting loss of normal feedwater flow events result in a conservative calculation. These parameters are listed in Table 15.2-22. The following initial conditions are assumed in the analysis to ensure that the results have sufficient conservatism:

- Initial power level is assumed to be 102 percent of nominal to account for a 2 percent measurement uncertainty
- Conservative scram characteristics are used, including a maximum time delay, holding the most reactive rod out of the core, and utilizing a bounding control rod drop rate.
- The most limiting combination of beginning of cycle core parameters is used to provide a limiting power response. The least negative MTC of 0.0pcm/degrees F and least negative DTC of -1.40pcm/degrees F are used to provide the largest power response for this event.

- Allowances for instrument inaccuracy are accounted for in the analytical limits of mitigating systems in accordance with the guidance provided in RG 1.105.
- Loss of normal AC power without the loss of highly reliable DC power at the time of turbine trip, but before the reactor trip would cause the MSIVs to fail closed, thus closing slightly sooner than if closure had been actuated along with DHRS. Closure of the MSIVs with the turbine stop valve reduces the volume of steam that must absorb the over pressure caused by arresting main steam flow, thus maximizing steam generator (and RCS) over pressurization.

The results from the thermal hydraulic evaluation are used as input to the subchannel analysis to determine the MCHFR for this event.

15.2.7.3.3 Results

Sensitivity studies on the LOFW were performed in an attempt to maximize the primary and secondary pressure response and minimize the CHF ratio. The cases were assessed with normal boundary conditions as well as with biased conditions.

Two cases were identified as providing limiting results: the first case included a complete LOFW flow and resulted in the maximum RCS pressure and MCHFR. The second case assessed a partial loss of feedwater flow, and resulted in the maximum steam generator pressure.

Upon the reactor trip, power decreases as shown in Figure 15.2-27. As the RCS heats up, the expansion of water volume increases pressurizer level (Figure 15.2-32). RCS pressure increases as shown in Figure 15.2-28. The increase in reactor coolant temperature causes the primary coolant volume to expand, raising RCS pressure. The peak RCS pressure occurs with the lifting of the RSV, which quickly reduces pressure below the RSV reset pressure and ensures RCS pressure is maintained below the acceptance criteria. RCS temperature decreases due to the reactor trip and then increases due to the reduction in heat removal until DHRS begins to cool the primary system as plotted in Figure 15.2-31. RCS flow (Figure 15.2-30) drops initially due to the reactor trip and stabilizes as DHRS flow is established Figure 15.2-29. The limiting SG pressure is presented in Figure 15.2-33.

The predicted lowest MCHFR was greater than the acceptance criteria for this event as shown in Figure 15.2-34. No fuel failure is predicted to occur.

15.2.7.4 Radiological Consequences

The radiological consequences of a LOFW event are bounded by the design basis accident analyses presented in Section 15.0.3.

15.2.7.5 Conclusions

The six DSRS acceptance criteria for this AOO are met for the limiting decrease in normal feedwater flow cases. These acceptance criteria, followed by how the NuScale design meets them are listed below:

1) Pressure in the reactor coolant and main steam systems should be maintained below 110 percent of the design values.

    - The limiting RCS pressure for this event, listed in Table 15.2-23, is below 110 percent of the design value for the reactor coolant system.
    - The limiting main steam system pressure, listed in Table 15.2-23, is below 110 percent of the design value for the main steam system

2) Fuel cladding integrity should be maintained by ensuring the minimum DNBR remains above the 95/95 DNBR limit based on acceptable correlations

    - The MCHFR for this event, listed in Table 15.2-23, is above the 95/95 limit.

3) An incident of moderate frequency should not generate a more serious plant condition without other faults occurring independently.

    - The analyses presented for this event shows that stable DHRS cooling is reached, and the acceptance criteria are met.

4) The guidance provided in RG 1.105, "Instrument Spans and Setpoints," can be used to analyze the impact of the instrument spans and setpoints on the plant response to the type of transient addressed in this DSRS section, in order to meet the requirements of GDCs 10 and 15.

    - Allowances for instrument inaccuracy are accounted for in the analytical limits of mitigating systems in accordance with the guidance provided in Regulatory Guide 1.105.

5) The most limiting plant systems single failure, as defined in the "Definitions and Explanations" of Appendix A to 10 CFR Part 50, shall be identified as assumed in the analysis and shall satisfy the positions of RG 1.53.

    - No single failures were identified that have adverse impact on the acceptance criteria. Results from this scenario do not challenge the identified limiting parameters as described in this Section.

6) The guidance provided in SECY-77-439, SECY-94-084 and RG 1.206 with respect to the consideration of the performance of nonsafety-related systems during transients and accidents, as well as the consideration of single failures of active and passive systems (especially as they relate to the performance of check valves in passive systems) must be evaluated and verified.

    - The inputs and assumptions for the operation of nonsafety-related systems and single failures as discussed in Section 15.2.7.2 and Section 15.2.7.3 ensure the guidance provided is met.

15.2.8 Feedwater System Pipe Breaks Inside and Outside of Containment

15.2.8.1 Identification of Causes and Event Description

A feedwater line break (FWLB) event for the NuScale design could range from a small split crack to a double ended rupture of the feedwater line. This event can occur both inside and outside of the containment vessel (CNV) due to seismic events, thermal stress, or cracking of the feedwater piping. A spectrum of FWLB locations and break sizes, with varied core and plant conditions, are analyzed to determine the scenarios with the most severe results.

A FWLB inside the CNV will increase the pressure in the evacuated atmosphere resulting in a loss of containment vacuum, and actuating the high containment pressure module protection system (MPS) signal. The high containment pressure MPS signal actuates the reactor trip system (RTS), isolates containment, and actuates the Secondary System Isolation (SSI). The decay heat removal system (DHRS) is subsequently actuated. The break will depressurize the impacted steam generator train and drain the DHRS piping and condenser in the affected loop. The non-impacted steam generator system and DHRS loop will continue to provide cooling to the RCS.

Feedwater breaks outside of containment will cause a loss of feedwater flow to the steam generators and a heatup and subsequent pressure increase in the RCS. Larger breaks will cause a rapid heatup and will trip the reactor and actuate SSI on low steam line pressure, whereas smaller breaks will cause a gradual heatup and loss of pressure in the main steam system resulting in a low steam line pressure signal, a high pressurizer pressure signal, or a high steam superheat signal that will actuate the RTS and SSI. Assuming a loss of AC power will cause a rapid heatup and will trip the reactor and actuate SSI on high pressurizer pressure. Coincident or subsequent actuation of DHRS provides cooling and depressurization to the RCS via the intact loop.

A break in the feedwater line is not expected to occur during the life of the NPM, so it has been classified as an accident as shown in Table 15.0-1.

15.2.8.2 Sequence of Events and Systems Operation

Unless specified below, the analysis of a FWLB event assumes the plant control systems and engineered safety features perform as designed, with allowances for instrument inaccuracy. No operator action is credited to mitigate the consequences of a FWLB event.

For some FWLBs with AC power available, the loss of FW flow causes steam pressure to decrease resulting in an RTS and SSI actuation on low steam pressure. Depending on the size of the break and initial conditions, the high steam superheat trip is reached in some events before SG pressure drops below the low pressure setpoint.

For FWLB events where AC power is lost, the turbine and feedwater pumps trip immediately resulting in a loss of RCS cooling and rapid pressurization. In these cases, the high PZR pressure signal will cause a reactor trip and actuate DHRS and SSI.

For cases where DC power (EDSS) is lost, a reactor trip and CNV isolation occur at the time of the FWLB, which results in less limiting break scenarios as mitigating actions are accomplished sooner. Therefore, DC power is assumed to be available for all cases presented in this section.

Breaks inside containment cause a rapid pressurization of the CNV resulting in a reactor trip and CNV isolation with SSI actuation and subsequent DHRS actuation. These break locations are not limiting for RCS pressure, SG pressure or MCFHR, but are the most challenging to the DHRS as one heat exchanger is disabled with the inventory released inside the CNV. The CNV response is addressed in Section 6.2.1.4.

The limiting FWLB for RCS pressure is a 10 percent split break in a FW line just outside containment with a coincident loss of AC power. The combination of turbine trip, feedwater pump trip and the feedwater pipe break result in a loss of cooling to the primary and a heatup and pressurization of the RCS. Reactor trip, DHRS and SSI actuation occur from the high pressurizer pressure MPS signal followed by the lifting of an RSV that quickly reduces primary pressure below the RSV reset pressure. Secondary pressure increases during the event until stable DHRS cooling is established, at which point the temperature and pressure in both the primary and secondary decrease. Table 15.2-24 provides the sequence of events for the limiting RCS pressure case.

The limiting MCHFR FWLB event is a double ended guillotine break in a FW line just outside of containment coincident with a loss of AC power. The decrease in heat removal from the loss of AC power at the time of the break had a more significant impact on MCHFR than the size of the break. Loss of AC power and turbine trip is more limiting for MCHFR due the increased heatup of the RCS. FWLBs inside containment are not limiting for MCHFR because the CNV pressure trip is reached very quickly which trips the reactor and isolates containment. The limiting MCHFR sequence of events is provided in Table 15.2-26.

The limiting SG pressure case occurs from a 4.0 percent split break in a FW line just outside containment coincident with a loss of AC power at the time of the break. The peak secondary pressurization is a function of the delay in DHRS actuation and establishing heat removal. Table 15.2-25 provides the sequence of events for the limiting secondary pressure case resulting from the 4.0-percent FW line break with the failure of the FWIV backflow check valve.

The limiting DHRS cooling case involves a DEG break of a feedwater line inside containment. Unlike breaks outside of containment, this break results in the complete loss of one train of DHRS. Upon break initiation, pressure inside of the CNV rapidly increases, reaching the high CNV pressure analytical limit and actuating reactor trip, containment isolation, and SSI. DHRS actuation occurs later in the event due to meeting the high pressurizer pressure limit. The remaining DHRS loop provides cooling to the module and is sufficient to remove 100 percent of decay heat and drive flow through the core. This event is not limiting for any of the acceptance criteria. The sequence of events for this case is provided in Table 15.2-27.

The MPS is credited to protect the NPM in the event of a FWLB. The following MPS signals provide the plant with protection during a FWLB:

- Low steam pressure
- High pressurizer pressure
- High CNV pressure
- High steam superheat
- High steam pressure

The actuation of a single RSV is credited for ensuring pressures in the RCS do not exceed the acceptance criteria.

No single failures have an impact on the limiting MCHFR results. The failure of the safety-related check valve (FWIV backflow check valve) to close on the failed SG did result in a limiting value for SG pressure and RCS pressure. For FWLB inside containment, in the event of the failure of the safety-related check valve, the second nonsafety-related check valve is credited to ensure that adequate inventory is maintained in the intact steam generator and DHRS condenser. Therefore, there are no single active failures that cause the FWLB event to have unacceptable results.

15.2.8.3 Thermal Hydraulic and Subchannel Analyses

15.2.8.3.1 Evaluation Model

The thermal hydraulic analysis of the NPM response to an FWLB is performed using NRELAP5. The NRELAP5 model is based on the design features of a NuScale module. The non-LOCA NRELAP5 model is discussed in Section 15.0.2.2.2. The relevant boundary conditions from the NRELAP5 analyses are provided to the downstream subchannel Critical Heat Flux (CHF) analysis.

The subchannel core CHF analysis is performed using VIPRE-01. VIPRE-01 is a subchannel analysis tool designed for general-purpose thermal hydraulic analysis under normal operating conditions, operational transients, and events of moderate severity. See Section 15.0.2.3 for a discussion of the VIPRE-01 code and evaluation model.

15.2.8.4 Input Parameters and Initial conditions

The FWLB inside containment scenarios are bounded by the consequences of the FWLB outside of containment from a primary and secondary system response perspective. The initial conditions and assumptions used in the evaluation of a FWLB event result in a conservative calculation.

The limiting peak RCS pressure case is a 10 percent split break in a FW line just outside of containment. The following initial conditions are assumed in the analysis of the FWLB to ensure that the transient results in the limiting RCS pressure.

- Initial power level is assumed to be 102 percent of nominal to account for a 2 percent measurement uncertainty.

- Conservative scram characteristics are used, including a maximum time delay, holding the most reactive rod out of the core, and utilizing a bounding control rod drop rate.
- The limiting beginning of cycle (BOC) core parameters are used to provide a limiting power response. The least negative MTC (0.0 pcm/degrees F) and DTC (-1.4 pcm/degrees F) are used to minimize the power response for this event.
- A loss of AC power is assumed to occur at the time of the break, causing an immediate turbine and feedwater pump trip. The normal and highly reliable DC power systems are assumed to be available, which delays the reactor trip and MPS actuations that would occur immediately on a loss of DC power.
- The FWIV backflow check valve is assumed to fail to close on the faulted FW line.
- The FWIVs are assumed to close at the design limit closure rate while the MSIVs are assumed to close rapidly to maximize the heatup affect.
- System biases include: high RCS temperature, high fuel temperature, high pressurizer pressure, high pressurizer level, and minimum RCS flow.
- The end of life steam generator was assumed which includes the design limit tube plugging and fouling in addition to a negative 30 percent uncertainty for the primary-to-secondary heat transfer.

The limiting MCHFR event is a double ended guillotine break of a FW line just outside of containment with a loss of AC power. This FWLB involves a loss of cooling that heats up and pressurizes the RCS, resulting in the limiting MCHFR. The following initial conditions are assumed in the analysis to ensure that the transient results in the limiting CHF ratio.

- Initial power level is assumed to be 102 percent of nominal to account for a 2 percent measurement uncertainty
- Conservative scram characteristics are used, including a maximum time delay, holding the most reactive rod out of the core, and utilizing a bounding control rod drop rate.
- The limiting BOC core parameters are used to provide a limiting power response. The least negative MTC (0.0 pcm/degrees F) and DTC (-1.4 pcm/degrees F) are used to minimize the power dampening response for this heatup event.
- A loss of AC power is assumed to occur at the time of the break, causing an immediate turbine and feedwater pump trip. The normal and highly reliable DC power systems are assumed to be available, which delays a reactor trip from mitigating the power excursion.
- The FWIV check valve is assumed to close on the faulted FW line to minimize fluid loss, which maximizes the RCS heatup. The FWIVs are assumed to close at the design limit closure rate while the MSIVs are assumed to close rapidly to maximize the heatup affect.
- System biases include: high RCS temperature, high fuel temperature, low pressurizer pressure, low pressurizer level, and minimum RCS flow.

- The end of life steam generator was assumed which includes the design limit tube plugging and fouling in addition to a negative 30-percent uncertainty for the primary-to-secondary heat transfer.

The limiting event for SG pressure is the 4.0-percent split break in a FW line just outside of containment. The following initial conditions are assumed in the analysis to ensure that the transient results in the limiting secondary pressure.

- Initial power level is assumed to be 102 percent of nominal to account for a 2 percent measurement uncertainty.
- Conservative scram characteristics are used, including a maximum time delay, holding the most reactive rod out of the core, and utilizing a bounding control rod drop rate.
- The limiting BOC core parameters are used to provide a limiting power response. The least negative MTC (0.0 pcm/degrees F) and DTC (-1.4 pcm/degrees F) are used to minimize the power response for this event.
- A loss of AC power is assumed to occur at the time of the break, causing an immediate turbine and feedwater pump trip. The normal and highly reliable DC power systems are assumed to be available, which delays the reactor trip and MPS actuations that would occur immediately on a loss of DC power.
- The FWIVs are assumed to close at the design limit closure rate while the MSIVs are assumed to close rapidly to maximize the heatup affect.
- The single failure assumed in this case was the failure of a FWIV check valve to seat, reducing the inventory in the second SG train.
- System biases include: high RCS temperature, high fuel temperature, high pressurizer pressure and pressurizer level, and minimum RCS flow.
- The end of life steam generator was assumed which includes the design limit tube plugging and fouling in addition to a negative 30 percent uncertainty for the primary to secondary heat transfer.

The limiting DHRS cooling case is a double ended guillotine break in a FW line inside containment. The following initial conditions are assumed in the analysis to ensure that the transient results are conservative in the assessment of DHRS functionality.

- Initial power level is assumed to be 102 percent of nominal to account for a 2 percent measurement uncertainty.
- Conservative scram characteristics are used, including a maximum time delay, holding the most reactive rod out of the core, and utilizing a bounding control rod drop rate.
- The limiting BOC core parameters are used to provide a limiting power response. The least negative MTC (0.0 pcm/degrees F) and DTC (-1.4 pcm/degrees F) are used to minimize the power response for this event.
- The FWIVs and MSIVs are assumed to close rapidly to maximize the heatup affect and limit the mass released to containment.
- System biases include: high RCS temperature, high fuel temperature, high pressurizer pressure and level, and minimum RCS flow.

- DHRS performance is assumed to be low (-30 percent) with a high biased pool temperature.
- CNV heat transfer is disabled to ensure FW liquid released to the CNV does not assist the heat removal.
- AC power is assumed to be available with the turbine trip occurring at the time of reactor trip.
- The end of life steam generator was assumed which includes the design limit tube plugging and fouling in addition to a negative 30-percent uncertainty for the primary-to-secondary heat transfer.

The results from the thermal hydraulic evaluations are used as input to the subchannel analysis to determine the limiting MCHFR for this event. Other key inputs and assumptions used in the analysis are included in Table 15.2-28.

15.2.8.5 Results

For the limiting RCS pressure case, the FW line is assumed to have a 10 percent split break in a FW line just outside containment. AC power is assumed to be lost, causing the turbine and feedwater pumps to immediately trip, resulting in a rapid loss of cooling and large RCS pressurization. In this scenario, the reactor is tripped by the high pressurizer pressure MPS signal also leading to a secondary system isolation and DHRS actuation. Pressure in the RCS continues to increase following reactor trip until the reactor safety valve lift limit is reached. Once the RSV lifts, system pressure quickly returns to below the RSV reset pressure. The limiting peak RCS pressure from FWLB is provided in Figure 15.2-35. DHRS cooling is sufficient to begin to depressurize the system such that a second RSV lift does not occur. The reactor power drops sharply following the reactor trip and follows the typical decay heat curve as shown on Figure 15.2-36. SG pressure rises in the unaffected (intact FW loop) SG as shown in Figure 15.2-37. The affected (FW pipe break loop) SG pressure response is shown in Figure 15.2-38. RCS temperature decreases and RCS flow stabilizes as shown in Figure 15.2-39 and Figure 15.2-40, respectively. The SSI and DHRS actuation closes the FWIVs and stops the loss of secondary fluid. The integrated break flow for the event is shown in Figure 15.2-41.

The peak SG pressure case results from a 4.0 percent split break in a FW line just outside containment. AC power is assumed to be lost at the time of the break. The FWIV backflow check valve on the affected SG train is assumed to fail to seat upon reverse flow. This condition results in asymmetrical pressure in the SGs as shown in Figure 15.2-42 and Figure 15.2-43. The intact SG has the peak SG pressure (Figure 15.2-42).

In the limiting MCHFR case, the FWLB is a double ended guillotine break in a FW line just outside containment. AC power is assumed to be lost at the time of the break. The MCHFR versus time is presented on Figure 15.2-44.

For the limiting DHRS cooling case, the FWLB is a doubled ended guillotine break of a FW line inside containment. The break flow will flash in the evacuated CNV atmosphere resulting in a loss of containment vacuum signal and a high containment pressure

signal. The high CNV pressure signal causes an actuation of the reactor trip system and a containment isolation signal which includes the actuation of the secondary system isolation. The DHRS actuation will rapidly actuate subsequent to the SSI. The break will cause a loss of FW pressure and decrease flow to both steam generators causing a heatup of the RCS. The break results in the loss of an entire DHRS loop as the fluid from the feedwater line and steam generator empty into containment. The second loop of DHRS remains intact throughout the event providing cooling to the RCS. DHRS heat removal is shown in Figure 15.2-45. The RCS flowrate decreases and then stabilizes as shown in Figure 15.2-48. The intact SG pressure is shown in Figure 15.2-46 and the depressurized (faulted) SG pressure is shown in Figure 15.2-47.

15.2.8.6 Radiological Consequences

The radiological consequences of the FWLB event are bounded by the consequences of a steam line break discussed in Section 15.0.3.

15.2.8.7 Conclusions

The five DSRS acceptance criteria for this accident are met for the limiting FWLB cases. These acceptance criteria, followed by how the NuScale design meets them are listed below:

1) Pressure in the reactor coolant and main steam systems should be maintained below 110 percent of the design pressures for low probability events and below 120 percent for very low probability events.

   - The pressure responses in the RPV and in the MSS for this accident are shown in Table 15.2-29 to be under the more conservative AOO acceptance criterion of 110 percent of design values. Therefore, the acceptance criteria for pressure are met for this event.

2) The potential for core damage is evaluated for an acceptable minimum DNBR remaining above the 95/95 DNBR limit for pressurized-water reactors based on acceptable correlations (See DSRS Section 4.4)

   - The MCHFR for this event is above the 95/95 limit as presented in Table 15.2-29.

3) Calculated doses at the site boundary from any activity release must be a small fraction of the 10 CFR Part 100 guidelines.

   - The radiological analysis of the SLB accident bounds the consequences for the FWLB and is presented in Section 15.0.3 and demonstrates that the acceptance criteria are met.

4) The DHRS must be safety grade and automatically initiated when required.

   - The results of this analysis demonstrate that, even with a failed DHRS train (due to FWLB inside containment), the second DHRS actuates and provides heat removal during a FWLB, ensuring that acceptance criteria are met.

5) Certain assumptions should be in the analysis of important parameters that describe initial plant conditions and postulated system failures.

- The assumptions included in this analysis for biasing and sensitivities were developed to consider sufficient conservatism to ensure the acceptance criteria are met.

15.2.9 Inadvertent Operation of the Decay Heat Removal System

15.2.9.1 Identification of Causes and Event Description

This section addresses actuation of DHRS at higher power which is a heatup event due to the decrease in cooling. The limiting cases for this event are at full power. The events evaluated include the following scenarios:

- Inadvertent opening of a single DHRS actuation valve
- Inadvertent isolation of one SG and actuation of one DHRS train
- Inadvertent isolation of both SGs and actuation of both DHRS trains
- Inadvertent isolation of one SG
- Inadvertent isolation of both SGs

At low power and reduced feedwater flow rates, the inadvertent operation of DHRS (IODHRS) is a cooldown event. The pressure drop along the secondary decreases such that it no longer exceeds the hydrostatic pressure of the liquid inventory in the DHRS piping. If the DHRS actuation valve opens under these conditions, a portion of the DHRS liquid inventory drains into the feedwater line and momentarily increases steam generator flow. This unique variant of the IODHRS event leads to an increase in heat removal from the RCS. This IODHRS event is bounded by more limiting overcooling events addressed in Section 15.1, such as an increase in feedwater flow.

The Decay Heat Removal System (DHRS) has several scenarios that can initiate part or all of the system functions. The DHRS actuation valves are normally energized and powered by independent DC power sources from the EDSS system. Loss of power or an inadvertent control signal to one actuation valve on either DHRS system will open the flow path to the associated DHRS heat exchanger, providing a short circuit flow path for feedwater through the DHRS piping instead of through the steam generator. A spurious actuation could occur on one or both trains of DHRS.

A full actuation of DHRS opens the two actuation valves on each of the two trains and closes the feedwater isolation valves (FWIVs) and the main steam isolation valves (MSIVs). An inadvertent signal to isolate one or both steam generators by closure of the FWIV and MSIV on the affected train(s) is also evaluated.

The inadvertent operation of DHRS is expected to occur one or more times in the life of the NPM, it is classified as an AOO. The categorization of the NuScale design basis events are discussed in Section 15.0, Table 15.0-1.

15.2.9.2 Sequence of Events and Systems Operation

Unless specified below, the analysis of the inadvertent operation of the DHRS event assumes the plant control systems and engineered safety features perform as

designed, with the allowances for instrument inaccuracy. No operator action is credited to mitigate the effects of the inadvertent operation of the DHRS event.

DHRS actuates on a loss of AC (ELVS) or DC (EDSS) power. A loss of ELVS or EDSS also causes a reactor trip and containment isolation, which mitigate the event sooner than the MPS would with power available. Therefore, loss of power scenarios are not limiting and are not addressed further in this section.

The inadvertent opening of a single DHRS actuation valve provides a short circuit fluid pathway between the feedwater and main steam systems that bypasses the steam generator. When operating at full power, this causes a portion of the subcooled feedwater flow to flow through the DHRS piping instead of through the steam generator. This leads to a reduction in heat removal from the RCS but is normally compensated for with the turbine load controller, which increases the total feedwater flow rate to return to the demanded turbine load. Under these nominal conditions, this is a relatively minor transient that does not result in a MPS actuation of safety related systems.

If the turbine load controller is unable to compensate for the reverse DHRS flow, a turbine trip occurs due to the reduced temperature steam coming from the main steam system. This event progression (not crediting turbine bypass) results in an MPS high steam superheat or high steam pressure reactor trip signal that also initiates a full SSI actuation, which closes the FWIVs and MSIVs.

Limiting RCS Pressure Case

The maximum RCS pressure case occurs with inadvertent isolation of both SGs. In this case all the FWIVs, FWRVs, MSIVs and backup MSIVs close, but the DHRS actuation valves do not open. RCS pressure continues to increase until the high pressurizer pressure MPS reactor trip is initiated. Control rods are inserted and then the RSV lifts relieving the RCS pressure. Subsequently, the DHRS valves open and RCS cooling begins.

Limiting SGS Pressure Case

For the opening of a single DHRS valve, the reduction in heat removal leads to a continuous heatup of the RCS until the RCS high hot temperature MPS reactor trip and SSI signals are generated. This gradual RCS heatup is limiting for addressing peak SG pressure. Therefore, this scenario results in the most limiting case for peak SG pressure.

Limiting MCHFR Case

A DHRS actuation signal opening the valves on both DHRS trains and isolating both SGs rapidly leads to an MPS reactor trip signal for high pressurizer pressure or high steam pressure. Since the loss of heat transfer to both trains of the SG occurs simultaneously, this scenario provides the most rapid reduction in RCS heat removal and is the limiting MCHFR scenario. The RSV lifts to relieve the RCS pressure. The RSV is sized to relieve the maximum pressure and inventory ramp of the RCS that occurs in this event. No single failure of safety-related equipment results in higher a peak RCS pressure. After the DHRS valves are fully opened, SG pressure begins to decline.

The timing of the sequence of events for the limiting cases is provided in Table 15.2-30 through Table 15.2-32.

15.2.9.3 Thermal Hydraulic and Subchannel Analyses

15.2.9.3.1 Evaluation Model

The thermal hydraulic analysis of the NPM response to an inadvertent opration of the DHRS event is performed using NRELAP5. The NRELAP5 model is based on the design features of a NuScale module. The non-LOCA NRELAP5 model is discussed in Section 15.0.2.2.2. The relevant boundary conditions from the NRELAP5 analyses are provided to the downstream subchannel Critical Heat Flux (CHF) analysis.

The subchannel core CHF analysis is performed using VIPRE-01. VIPRE-01 is a subchannel analysis tool designed for general purpose thermal hydraulic analysis under normal operation conditions, operational transients, and events of moderate severity. See Section 15.0.2 for a discussion of the VIPRE-01 code and evaluation model.

15.2.9.3.2 Input Parameters and Initial Conditions

<u>Limiting MCHFR event</u>

- Initial power level is assumed to be 102 percent (163.2 MW) of nominal to account for a 2 percent measurement uncertainty.
- Limiting BOC reactivity coefficients are applied which minimizes the decrease in reactor power following the increase in RCS temperature.
- Regulating rod bank reactivity control is disabled to prevent the insertion of negative reactivity and decrease in reactor power following the increase in RCS temperature.
- The initial SG pressure is biased low.
- Minimum RCS design flow is assumed which minimizes MCHFR.
- Initial fuel temperature and decay heat are biased high.
- RCS coolant temperature is initialized to the high condition to result in the highest temperature during the transient calculation.
- High steam superheat signal is disabled to prevent an MPS actuation before the maximum RCS temperature is reached.
- Pressurizer pressure and pressurizer level are biased high for the event.
- No loss of AC power or failure of a FWIV is considered as this reduces the consequences of the event.
- Initial reactor pool temperature is biased high at 200 degrees F.
- Pressurizer spray is disabled and pressurizer heater output is constant.

<u>Limiting SG Pressure</u>

- Core power is initialized at 102 percent of rated power (163.2 MW).

- Limiting BOC reactivity coefficients are applied which minimizes the decrease in reactor power following the increase in RCS temperature.
- Initial RCS design flow is biased low.
- Initial pressurizer pressure and pressurizer level are biased low.
- RCS coolant temperature is initialized to the high condition to facilitate reaching the maximum analytical temperature during the transient calculation.
- Loss of AC Power is not assumed to occur, but single failure of a FWIV maximizes the effects of the event.
- Initial fuel temperature and decay heat are biased high.
- Initial SG pressure is increased to the high condition to maximize peak SG pressure.
- Pressurizer spray is disabled and pressurizer heater output is constant.
- The reactor pool temperature is increased to 200 degrees F.

<u>Limiting RCS Pressure</u>

- Core power is initialized at 102 percent of rated power (163.2 MW).
- Limiting BOC reactivity coefficients are applied which minimizes the decrease in reactor power following the increase in RCS temperature.
- Initial RCS pressure is biased to the high condition to maximize the peak RCS pressure during the transient.
- RCS temperature is biased high and RCS flow is biased low.
- Initial pressurizer level is biased to the high condition to maximize the peak RCS pressure following the in-surge of coolant into the pressurizer during the transient.
- Pressure control is disabled during the transient. This includes disabling the pressurizer heater response (i.e. set heater power to a constant value) and disabling the pressurizer spray.
- No loss of AC power or failure of a FWIV is considered as this reduces the consequences of the event.
- Initial fuel temperature and decay heat are biased high.
- Initial SG pressure is biased low.
- Initial reactor pool temperature is biased high at 200 degrees F.

15.2.9.3.3 Results

The limiting RCS pressure event occurs in the inadvertent isolation of both SGs at full power. In this event RCS pressure quickly escalates and leads to a high pressurizer pressure trip of the reactor. Subsequently the RSV lifts to reduce RCS pressure. The maximum RCS pressure reached in this analysis occurs after the RSV lift setting is reached but is well below the RCS maximum design pressure as shown in Figure 15.2-49. The RCS heatup is shown in the average core inlet and outlet

temperatures provided in Figure 15.2-50 and Figure 15.2-51, respectively. RCS flow is shown in Figure 15.2-52. The total reactivity is negative throughout the event as shown in Figure 15.2-53. This event is terminated after the MPS reactor trip, actuation of DHRS, and establishing of stable DHRS cooling.

The limiting case identified for MCHFR occurs for the inadvertent isolation of both SGs and actuation of both DHRS trains. As shown in Figure 15.2-55, the limit for MCHFR is not challenged.

The inadvertent opening of a single DHRS valve results in the highest SG pressure because it results in a gradual RCS heatup which is limiting for SG pressure. This event results in the highest SG pressure of the analyzed AOOs but as shown in Figure 15.2-54, pressure is maintained well below the SG design pressure.

Based on the above evaluation, these inadvertent operation of DHRS events do not result in pressure or temperature transients that exceed the criteria for which the reactor pressure vessel, SG, containment vessel, or fuel are designed. Therefore, these barriers to the transport of radionuclides to the environment function as designed.

15.2.9.4 Radiological Consequences

The radiological consequences of an inadvertent operation of the DHRS event are bounded by the design basis accident analyses presented in Section 15.0.3.

15.2.9.5 Conclusion

The five DSRS AOO acceptance criteria are met for the inadvertent operation of the DHRS. These acceptance criteria, followed by how the NuScale design meets them are listed below:

1) Pressure in the reactor coolant and main steam systems should be maintained below 110 percent of the design values.

    - The limiting RCS pressure for this event, listed in Table 15.2-33, is below 110 percent of the design value for the reactor coolant system.
    - The limiting main steam system pressure, listed in Table 15.2-33, is below 110 percent of the design value for the main steam system

2) Fuel cladding integrity should be maintained by ensuring the minimum DNBR remains above the 95/95 DNBR limit based on acceptable correlations (See DSRS Section 4.4)

    - The MCHFR for this event, listed in Table 15.2-33, is above the 95/95 limit.

3) An incident of moderate frequency should not generate a more serious plant condition without other faults occurring independently.

    - The analyses presented for this event shows that stable DHRS cooling is reached, and the acceptance criteria are met.

4) The guidance provided in RG 1.105, "Instrument Spans and Setpoints," can be used to analyze the impact of the instrument spans and setpoints on the plant response to the type of transient addressed in this DSRS section, in order to meet the requirements of GDCs 10 and 15.

    - Allowances for instrument inaccuracy are accounted for in the analytical limits of mitigating systems in accordance with the guidance provided in Regulatory Guide 1.105.

5) The most limiting plant systems single failure, as defined in the "Definitions and Explanations" of Appendix A to 10 CFR Part 50, shall be identified as assumed in the analysis and shall satisfy the positions of RG 1.53.

    - Postulated single failures for this event would provide results that are less limiting than those identified in the limiting case. As such, there are no limiting single failures identified for this event that would adversely affect the results.

This evaluation demonstrates that RCS pressure, SG pressure and MCHFR limits are not challenged. The regulatory acceptance criteria for an AOO are met for the inadvertent operation of DHRS event cases evaluated. A comparison of the limiting pressure values and the associated acceptance criteria is shown in Table 15.2-33. The system stabilizes to a safe condition.

Table 15.2-1: Loss of External Load -Turbine Trip - Loss of Condenser Vacuum - Initial Conditions Peak RCS Pressure

Parameter	Bias	Analysis Value
Initial Reactor Power	+2%	102%
Reactor Pool Temperature	+100°F	200°F
Feedwater Temperature	nominal	300°F
Moderator Temperature Coefficient	BOC	0.0 pcm/°F
Doppler Reactivity Coefficient	BOC	-1.4 pcm/°F
Decay Heat	High	various
RCS Flow Rate	minimum	1216 lbm/s
Delayed Neutron Fraction	BOC	0.008
Pressurizer Level	+8%	68%
RSV Lift Setpoint	+3%	2137 psia
RCS Average Coolant Temperature	+10°F	555°F
Steam Generator Pressure	+35 psia	535 psia
Pressurizer Pressure	-70 psia	1780 psia
Turbine Stop Valve Stroke Time	n/a	0.0001 s
Time of Loss of AC Power	n/a	coincident

Table 15.2-2: Loss of External Load -Turbine Trip - Loss of Condenser Vacuum - Initial Conditions Peak SG Pressure

Parameter	Bias	Analysis Value
Initial Reactor Power	+2%	102%
Reactor Pool Temperature	+100°F	200°F
Feedwater Temperature	nominal	300°F
Moderator Temperature Coefficient	BOC	0.0 pcm/°F
Doppler Reactivity Coefficient	BOC	-1.4 pcm/°F
Decay Heat	high	various
RCS Flow Rate	minimum	1164 lbm/s
Delayed Neutron Fraction	BOC	0.008
Pressurizer Level	+8%	68%
RSV Lift Setpoint	+3%	2137 psia
RCS Average Coolant Temperature	+10°F	555°F
Steam Generator Pressure	+35 psia	535 psia
Turbine Stop Valve Stroke Time	n/a	0.0001 s
Pressurizer Pressure	low	1780 psia

Table 15.2-3: Loss of External Load -Turbine Trip - Loss of Condenser Vacuum - Initial Conditions MCHFR

Parameter	Bias	Analysis Value
Initial Reactor Power	+2%	102%
Reactor Pool Temperature	+100°F	200°F
Feedwater Temperature	nominal	300°F
Moderator Temperature Coefficient	BOC	0.0 pcm/°F
Doppler Reactivity Coefficient	BOC	-1.4 pcm/°F
Decay Heat	high	various
RCS Flow Rate	minimum	1188 lbm/s
Delayed Neutron Fraction	BOC	0.008
Pressurizer Level	+8%	68%
RSV Lift Setpoint	+3%	2137 psia
RCS Average Coolant Temperature	+10°F	555°F
SG Pressure	-35	465 psia
Pressurizer Pressure	-70 psia	1780 psia
Turbine Stop Valve Stroke Time	n/a	0.0001 s

Table 15.2-4: Loss of External Load -Turbine Trip - Loss of Condenser Vacuum - RCS Maximum Pressure Sequence of Events

Event	Time (sec)
Event initiator -Turbine Trip and loss of FW flow due to loss of condenser vacuum with assumed loss of offsite power	0
Turbine Stop Valves Fully Closed (assumption)	0
FW flow is secured (assumption)	0
Pressurizer heater power secured	0
CVCS flow secured (assumption)	0
Reactor Trip, SSI and DHRS Actuation analytical limit (High Pressurizer Pressure)	10
Reactor Trip issued	12
DHRS and SSI Actuation signals issued	12
RSV Lift Point (2137 psia)	15
Peak RCS Pressure	16
Secondary system isolation complete	12
RSV Reseats	24
DHRS actuation valves full open	42

Table 15.2-5: Loss of External Load -Turbine Trip - Loss of Condenser Vacuum - SG Maximum Pressure Sequence of Events

Event	Time (sec)
Event initiator -Turbine Trip	0
Turbine Stop Valves Fully Closed (assumption)	0
Reactor Trip, SSI and DHRS Actuation analytical limit (High Steam Pressure)	6
Reactor Trip issued	8
DHRS and SSI Actuation signals issued	8
Secondary system isolation complete	9
DHRS actuation valves full open	38
Time of Peak Secondary Pressure	86

Table 15.2-6: Loss of External Load -Turbine Trip - Loss of Condenser Vacuum - MCHFR Sequence of Events

Event	Time (sec)
Event initiator -Loss of condenser vacuum results in Turbine Trip and loss of FW flow	0
Turbine Stop Valves Fully Closed (assumption)	0
FW flow is secured (assumption)	0
MCHFR reached	0
Reactor Trip, SSI and DHRS Actuation analytical limit (High Pressurizer Pressure)	11
Reactor Trip, SSI and DHRS signals issued	13
RSV Lift Point (2137 psia)	16
Secondary system isolation complete	13
RSV reseats	24
DHRS valves open	38

Table 15.2-7: Loss of External Load -Turbine Trip - Loss of Condenser Vacuum - Limiting Analysis Results

Acceptance Criteria	Transient	Limit	Analysis Value
Maximum RCS Pressure	LOCV	2310 psia	2161 psia
Maximum SG Pressure	TT	2310 psia	1545 psia
MCHFR	LOCV	1.284	2.441

Table 15.2-8: Main Steam Isolation Valve Closure - Initial Conditions for RCS Pressure Limiting Case

Parameter	Analysis Value
Initial Reactor Power (includes 2% uncertainty)	102% RTP (163.2 MW_{th})
Initial RCS T_{ave} (low bias)	535 °F
Initial Pressurizer Pressure (low bias)	1780 psia
Initial Pressurizer Level (high bias)	68%
Feedwater Temperature (nominal)	300 °F
Steam Generator Pressure (low bias)	465 psia
Initial RCS Flowrate (low bias)	1209 lbm/s
Moderator Temperature Coefficient (MTC) (BOC)	0.0
Doppler Reactivity Coefficient (BOC)	-1.4
Delayed Neutron Fraction (ß) (BOC)	0.008
Safety Relief Valve Open Setpoint	2137 psia
Pool Temperature (high bias)	200°F

Table 15.2-9: Main Steam Isolation Valve Closure - Initial Conditions for SG Pressure Limiting Case

Parameter	Analysis Value
Initial Reactor Power (includes 2% uncertainty)	102% RTP (163.2 MW_{th})
Initial RCS Tave (high bias)	555 °F
Initial Pressurizer Pressure (high bias)	1920 psia
Initial Pressurizer Level (high bias)	68%
Feedwater Temperature (nominal)	300 °F
Steam Generator Pressure (low bias)	465 psia
Initial RCS Flowrate (low bias)	1145 lbm/s
Moderator Temperature Coefficient (MTC) (BOC)	0.0
Doppler Reactivity Coefficient (BOC)	-1.4
Delayed Neutron Fraction (ß) (BOC)	0.008
Safety Relief Valve Open Setpoint	2137 psia
Pool Temperature (high bias)	200°F

Table 15.2-10: Main Steam Isolation Valve Closure - Initial Conditions for MCHFR Limiting Case

Parameter	Analysis Value
Initial Reactor Power (includes 2% uncertainty)	102% RTP (163.2 MW_{th})
Initial RCS Tave (high bias)	555 °F
Initial Pressurizer Pressure (high bias)	1920 psia
Initial Pressurizer Level (high bias)	68%
Feedwater Temperature (nominal)	300 °F
Steam Generator Pressure (low bias)	465 psia
Initial RCS Flowrate (low bias)	1145 lbm/s
Moderator Temperature Coefficient (MTC) (BOC)	0.0
Doppler Reactivity Coefficient (BOC)	-1.4
Delayed Neutron Fraction (ß) (BOC)	0.008
Safety Relief Valve Open Setpoint	2137 psia
Pool Temperature (high bias)	200°F

Table 15.2-11: Main Steam Isolation Valve Closure - Sequence of Events for RCS Pressure Limiting Case

Event	Time (s)
Event initiator - MSIV Closure - concurrent loss of AC power assumed	0
High pressurizer pressure MPS setpoint reached	9
RTS, SSI and DHRS actuation initiates	11
RSV1 lift setpoint reached	14
Peak RPV pressure is reached	14
RSV1 reseat setpoint reached	22
DHRS actuation valves are fully open	41

Table 15.2-12: Main Steam Isolation Valve Closure - Sequence of Events for SG Pressure Limiting Case

Event	Time (s)
Event initiator - MSIV Closure	0
High steam line pressure and high pressurizer pressure setpoints reached	4
RTS, SSI and DHRS actuation initiates	6
FWIVs are fully closed	13
DHRS actuation valves are fully open	36
Peak secondary pressure	81

Table 15.2-13: Main Steam Isolation Valve Closure - Sequence of Events for MCHFR Limiting Case

Event	Time (s)
Event initiator - MSIV Closure	0
Loss of AC power occurs (FW pumps trip)	0
High pressurizer pressure is reached	4
RTS, SSI and DHRS actuation initiates	6
RSV1 lift setpoint reached	9
RSV1 reseat setpoint reached	17
DHRS actuation valves are fully open	36

Table 15.2-14: Main Steam Isolation Valve Closure - Limiting Analysis Results

Acceptance Criteria	Limit	Analysis Value
Maximum RCS Pressure	2310 psia	2161
Maximum SG Pressure	2310 psia	1512
MCHFR	1.284	2.670

Table 15.2-15: Loss of Non-Emergency AC Power- RCS Pressure Limiting Event Sequence of Events

Event	Time [s]
Loss of AC power	0
High pressurizer pressure analytical limit is reached	5
Control rod insertion begins	7
Secondary system isolation occurs	7
DHRS actuation valves begin to open	7
RSV1 lift setpoint reached	10
Maximum RCS pressure (2160 psia)	11
Maximum containment pressure	18
DHRS actuation valves fully open	37
RSV1 reseat setpoint reached	49

Table 15.2-16: Loss of Non-Emergency AC Power -SG Pressure and MCHFR Limiting Event - Sequence of Events

Event	Time [s]
Loss of AC power	0
Minimum CHFR	0.3
High pressurizer pressure analytical limit is reached	8
Control rod insertion begins	10
Secondary system isolation occurs	10
DHRS actuation valves begin to open	10
RSV1 lift setpoint reached	13
Maximum containment pressure	21
DHRS actuation valves fully open	40
RSV1 reseat setpoint reached	66
Maximum SGS pressure (1415 psia)	82

Table 15.2-17: Not Used

Table 15.2-18: Input Parameters Loss of Non-Emergency AC Power -Limiting Cases

Parameter	RCS Overpressure	SG Overpressure and MCHFR[1]
Initial PZR pressure	1920 psia	1850 psia
Initial RCS temperature	535°F	555°F
Initial PZR level	68%	68%
Initial Feedwater temperature	300°F	300°F
Initial SG Pressure	535 psia	535 psia
RSV setpoint (2075 psia + 3% drift)	2137 psia	2137 psia
RCS Flowrate	1186 lbm/s	1155 lbm/s
Pool Temperature	110°F	110°F
SG Tube Heat Transfer	Nominal	Nominal
DC Power Available	EDNS, EDSS	EDNS, EDSS

[1] Since Heatup events are not challenging for MCHFR, the limiting SG pressure case is selected as representative for MCHFR.

Table 15.2-19: Loss of Non-Emergency AC Power - Limiting Analysis Results

Acceptance Criteria	Limit	Analysis Value
Maximum RCS Pressure	2310 psia	2160 psia
Maximum SG Pressure	2310 psia	1415 psia
MCHFR	1.284	2.539

Table 15.2-20: Loss of Feedwater Event - RCS Overpressurization/Limiting MCHFR Sequence of Events

Event	Time [s]
Loss of Feedwater initiation	0
High pressurizer pressure (>2000 psia)	12
Loss of Normal AC	12
SSI actuation	14
RTS actuation (rods begin to drop)	14
DHRS actuation	14
RSV #1 lifts	16
Peak RCS pressure	17
DHRS valves complete opening	44
RSV #1 reseats	73

Table 15.2-21: Loss of Feedwater Event - Maximum SG Pressure - Sequence of Events

Event	Time [s]
Loss of feedwater initiation Feedwater flow begins 0.1 second ramp down to 98.3% of initial value	0
RCS hot leg high temperature MPS signal	834
SSI actuation	842
RTS actuation	842
DHRS actuation	842
DHRS valve fully open	872
Peak secondary pressure	919

Table 15.2-22: Input Parameters Loss of Feedwater - Limiting Cases

Parameter	RCS Overpressure and MCHFR	SG Overpressure
Initial RCS pressure	1780 psia	1780 psia
Initial RCS temperature	555°F	555°F
Initial PZR level	68%	52%
Initial Feedwater temperature	310°F	290°F
Initial SG Pressure	465 psia	535 psia
Drift on RSV setpoint	2137 psia (+3%)	2137 psia (+3%)
Moderator and Doppler coefficients of reactivity	0.0/-1.40pcm/°F	0.0/-1.40pcm/°F
RCS Flowrate	1490 lbm/s	1163 lbm/s
Pool Temperature	200°F	200°F
Loss of FW Flow at initialization	100%	1.7%

Table 15.2-23: Loss of Feedwater - Limiting Analysis Results

Acceptance Criteria	Limit	Analysis Value
Maximum RCS Pressure	2310 psia	2171 psia
Maximum SG Pressure	2310 psia	1528 psia
MCHFR	1.284	2.426

Table 15.2-24: Feedwater Line Break Sequence of Events - Peak RCS Pressure Case

Event	Time [s]
Failure that initiates event.	0
AC power is lost resulting in turbine trip and FW pump trip	0
High PZR pressure limit reached	5
Control rod insertion begins	7
Secondary system isolation and DHRS actuation	7
RSV lift point is reached	9
Maximum RCS pressure reached	10
RSV reseat setpoint reached	19
DHRS actuation valve fully open	37
Maximum SG Pressure (non-faulted train)	79

Table 15.2-25: Feedwater Line Break Sequence of Events - Peak Steam Generator Pressure Case

Event	Time [s]
Failure that initiates event.	0
AC power is lost resulting in turbine trip and FW pump trip	0
High PZR pressure limit reached	5
Control rod insertion begins	7
Secondary system isolation and DHRS actuation	7
RSV lift point is reached	10
Maximum RCS pressure reached	10
RSV reseat setpoint reached	19
DHRS actuation valve fully open	37
Peak pressure reached in SG (non-faulted train)	79

Table 15.2-26: Feedwater Line Break Sequence of Events - Limiting MCHFR Case

Event	Time [s]
Failure that initiates event	0
AC power is lost resulting in turbine trip and FW pump trip	0
High PZR pressure limit reached	13
Secondary system isolation and DHRS actuation	15
Control rod insertion begins	15
RSV lift point is reached	19
Maximum RCS pressure reached	20
RSV reseat setpoint reached	29
Maximum CNV pressure reached	30
DHRS actuation valve fully open	45
Maximum SGS pressure reached	87

Table 15.2-27: Feedwater Line Break Sequence of Events - Limiting DHRS Case

Event	Time [s]
Failure that initiates event	0
High CNV pressure limit reached	1
RTS actuated - control rod insertion begins	3
Secondary system isolation actuation	3
High pressurizer pressure limit reached (DHRS actuation signal)	7
Maximum containment pressure	11
RSV lift point is reached	25
Maximum RCS pressure reached	25
RSV reseat setpoint reached	31
DHRS actuation valves open	39
Maximum SGS pressure reached	81

Table 15.2-28: Biases and Uncertainties - Feedwater Line Break

Region	Parameter	Value
Core	Power	100%
	Calorimetric Uncertainty	2%
	Neutronics Input	BOC
	Decay Heat	High
RCS	Primary mass flow rate	1179 lbm/s
	Max core average temperature	555°F
	Pressurizer pressure(1)	1850+70 psia
	Pressurizer level(1)	60% + 8%
	SG tube plugging	10%
	SG fouling	$1.0e^{-4}$ hr-ft^2-°F/BTU
	SG heat transfer(2)	-30%
	FW temperature	310°F
	MS pressure	500+35 psia
Reactor Pool	Temperature	200°F
DHRS	DHRS heat transfer(2)	-30%

(1) For the MCHFR limiting event pressurizer pressure is biased low to 1780 psia and pressurizer level is biased low to 52%.
(2) Margin to the acceptance criteria is not sensitive to this bias.

Table 15.2-29: Feedwater Line Break- Limiting Analysis Results

Acceptance Criteria	Limit	Analysis Value
Maximum RCS Pressure	2310 psia	2164 psia
Maximum SG Pressure	2310 psia	1391 psia
MCHFR	1.284	2.49

Table 15.2-30: Sequence of Events for Inadvertent Operation of Decay Heat Removal System - Limiting MCHFR Case

Event	Time (s)
Transient initiation (inadvertent isolation of both SGs and actuation of both DHRS trains)	0
High pressurizer pressure	4
RTS actuation - Control rod insertion begins	6
RSV lift setpoint reached	8
Maximum RCS pressure reached	9
RSV reseat setpoint reached	16
DHRS actuation valves fully open	30
Maximum SGS pressure reached	71

Table 15.2-31: Sequence of Events for Inadvertent Operation of Decay Heat Removal System - Limiting Peak SG Pressure Case

Event	Time (s)
Transient initiation (inadvertent opening of one DHRS actuation valve)	0
High RCS temperature limit reached	104
RTS actuation - Control rod insertion begins	112
Secondary System isolation valves closed	112
Maximum RCS pressure	121
DHRS actuation valves fully open	142
Maximum SG Pressure reached	190

Table 15.2-32: Sequence of Events for Inadvertent Operation of Decay Heat Removal System - Limiting Peak RCS Pressure Case

Event	Time (s)
Transient initiation (Inadvertent isolation of both SGs)	0
Secondary system isolation valves closed	2
High Pressurizer Pressure limit reached	4
RTS actuation - Control rod insertion begins	6
RSV lift point is reached	8
Maximum RCS pressure reached	9
RSV reseat setpoint reached	16
DHRS actuation valves fully open	36
Maximum SG pressure reached	78

Table 15.2-33: Inadvertent Operation of Decay Heat Removal System - Limiting Analysis Results

Acceptance Criteria	Limit	Analysis Value
Maximum RCS Pressure	2310 psia	2161 psia
Maximum SG Pressure	2310 psia	1592 psia
MCHFR	1.284	2.671

Figure 15.2-1: Pressurizer Level - Peak RCS Pressure Case (15.2.1-15.2.3 LOEL-TT-LOCV)

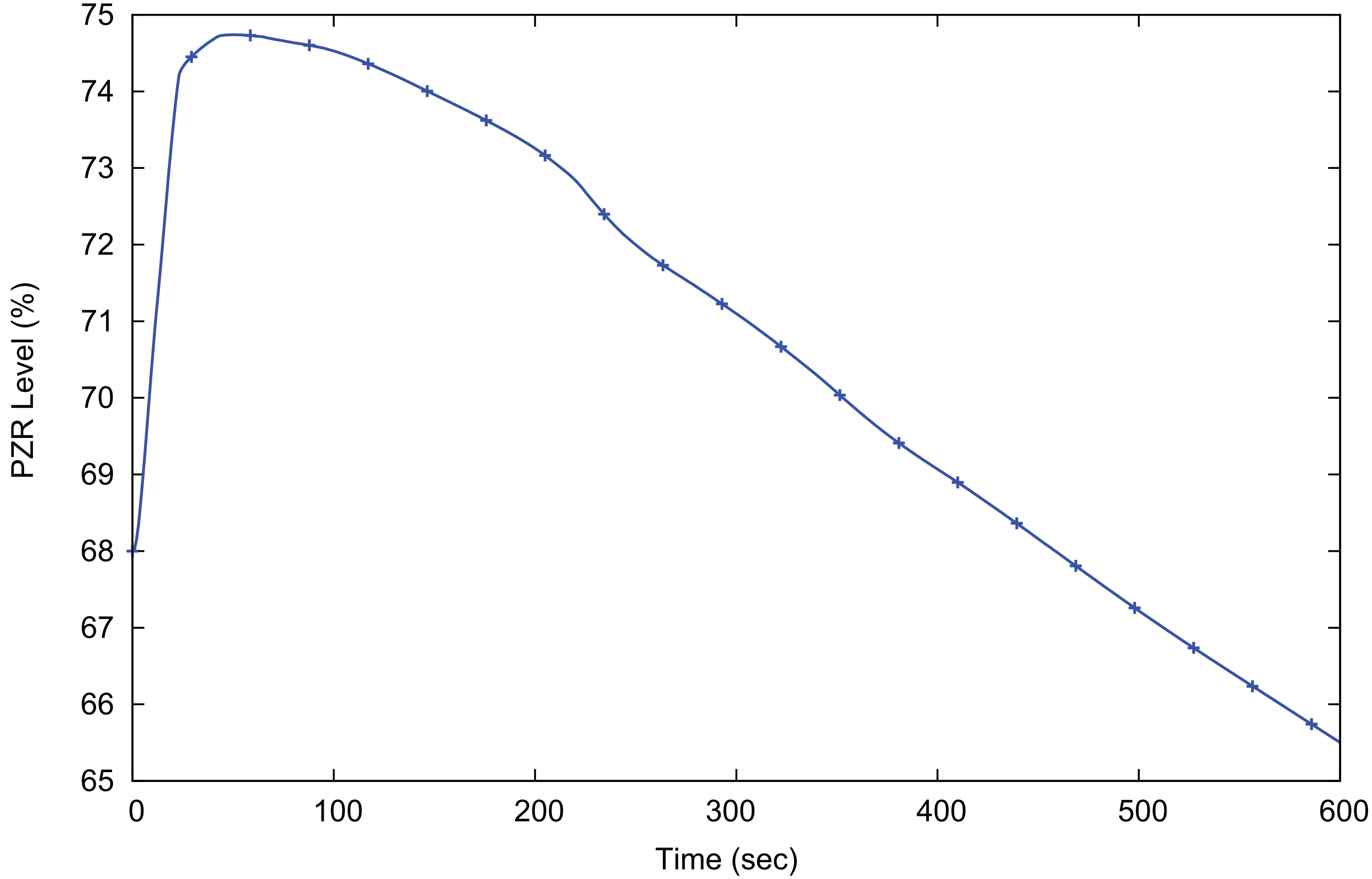


Figure 15.2-2: Reactor Power - Peak RCS Pressure Case (15.2.1-15.2.3 LOEL-TT-LOCV)

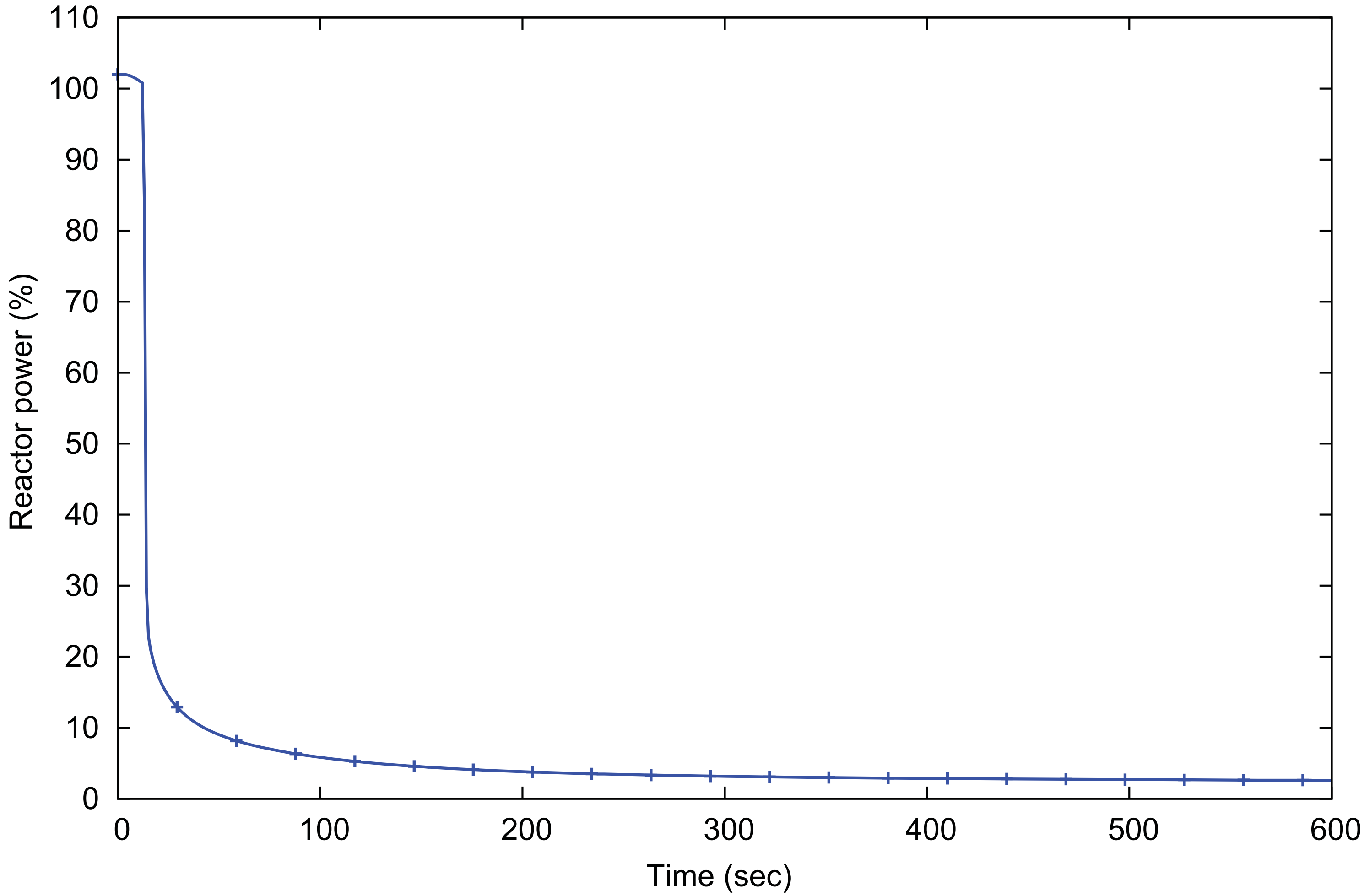


Figure 15.2-3: Net Reactivity - Peak RCS Pressure Case (15.2.1-15.2.3 LOEL-TT-LOCV)

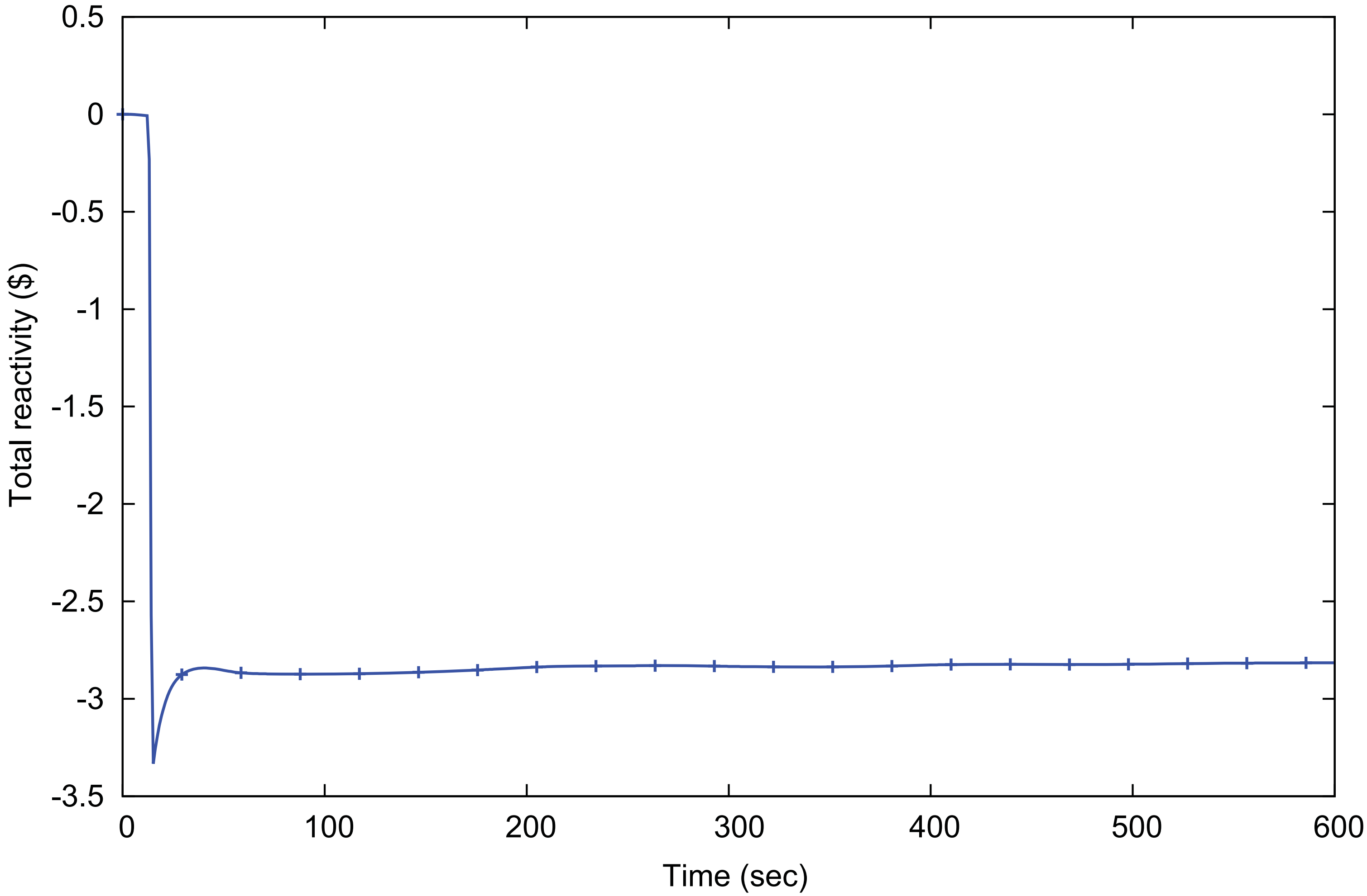


Figure 15.2-4: RCS Average Temperature - Peak RCS Pressure Case (15.2.1-15.2.3 LOEL-TT-LOCV)

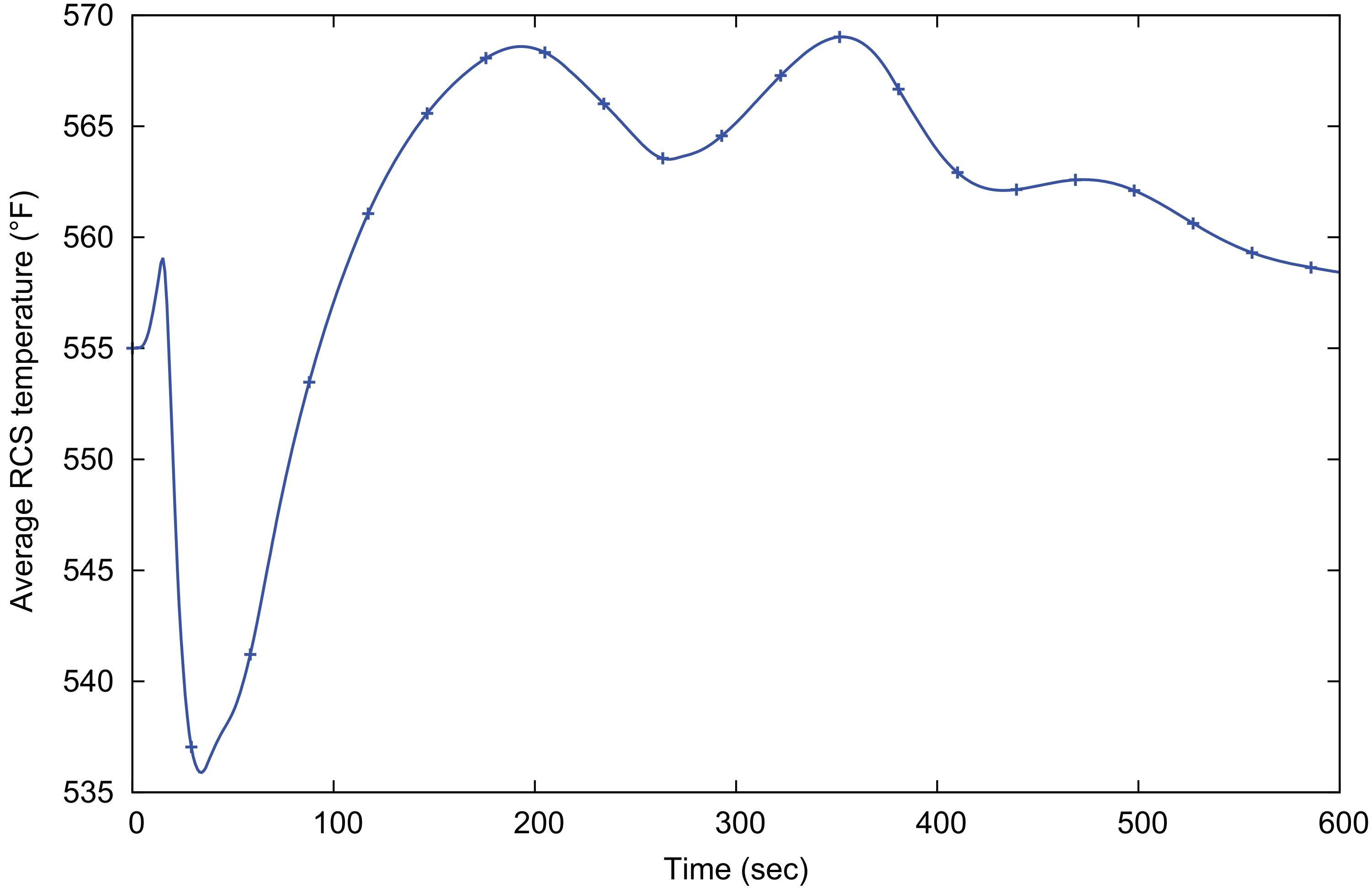


Figure 15.2-5: RCS Flowrate - Peak RCS Pressure Case (15.2.1-15.2.3 LOEL-TT-LOCV)

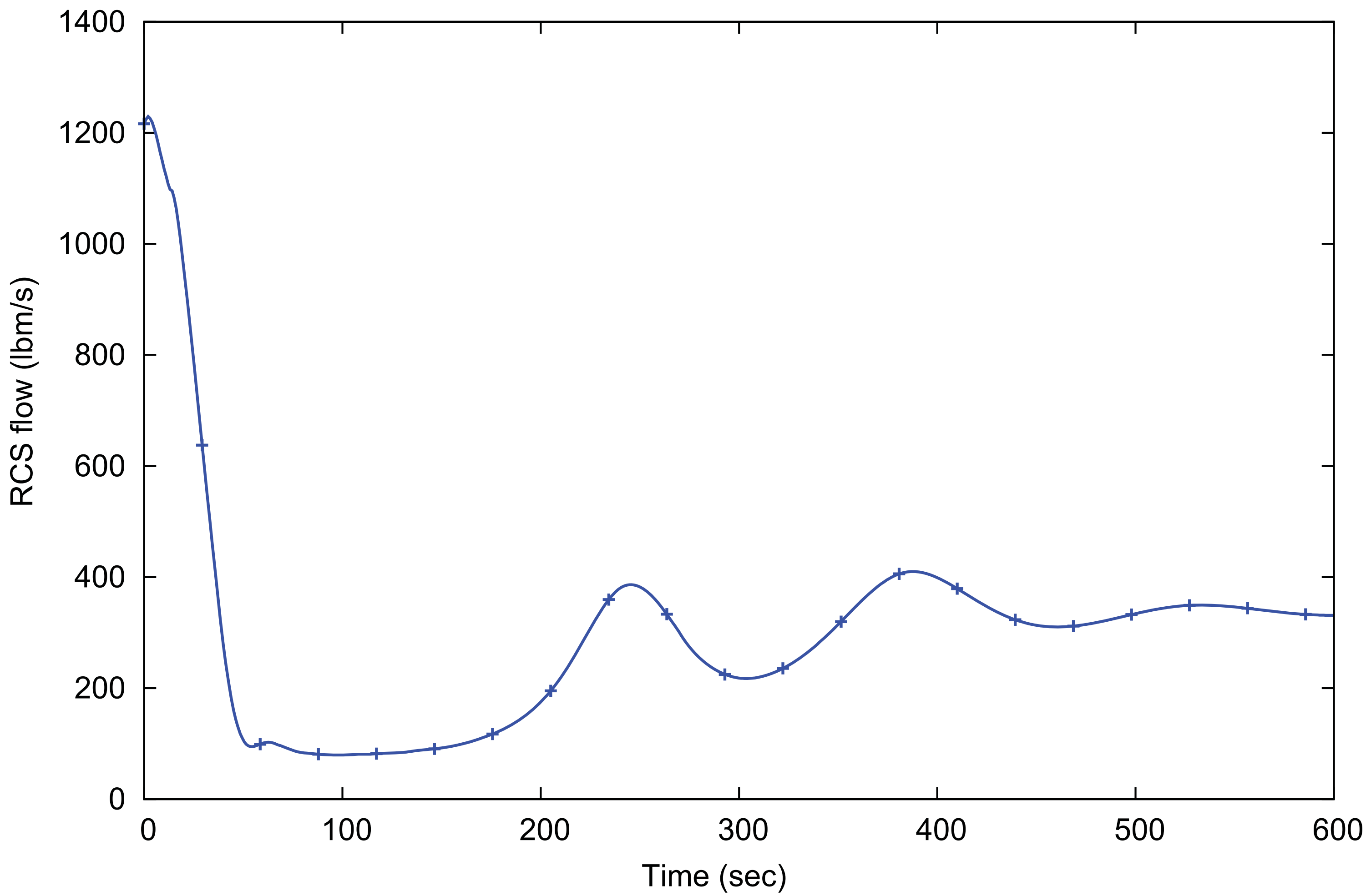


Figure 15.2-6: DHRS Flow Typical - Peak RCS Pressure Case (15.2.1-15.2.3 LOEL-TT-LOCV)

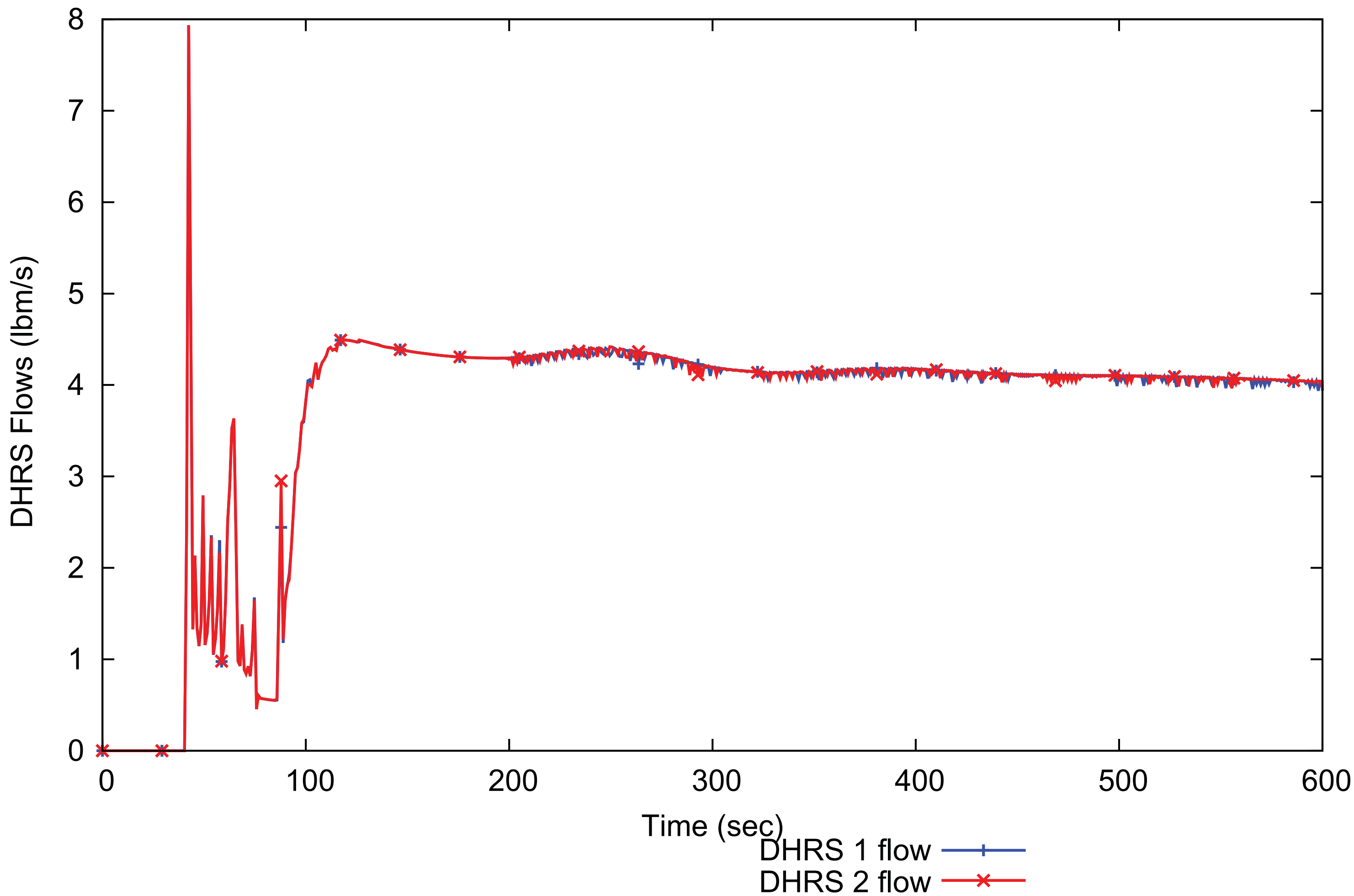


Figure 15.2-7: Reactor Pressure Vessel Pressure - Peak RCS Pressure Case (15.2.1-15.2.3 LOEL-TT-LOCV)

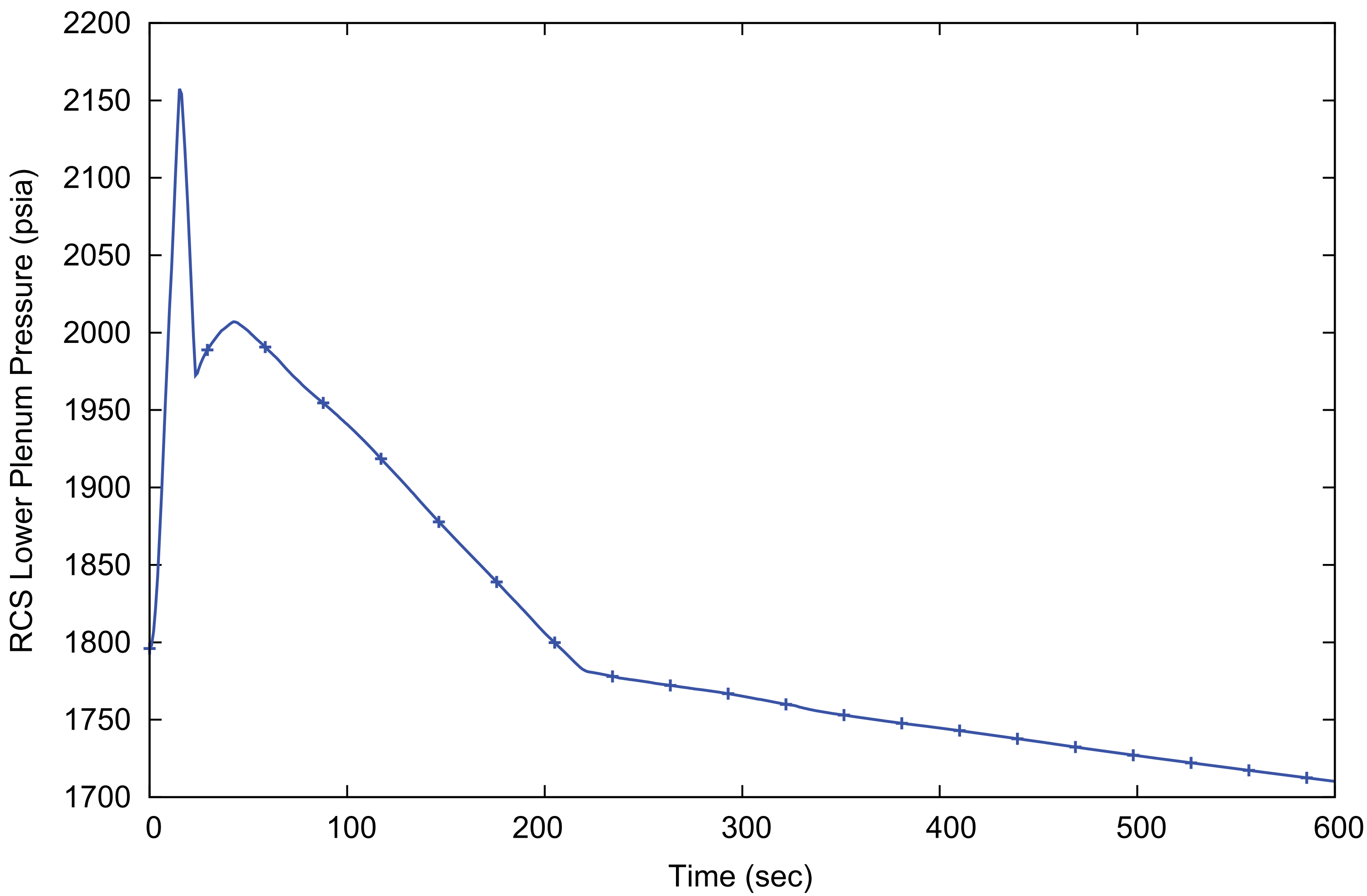


Figure 15.2-8: Steam Generator Pressure Typical - Peak SG Pressure Case (15.2.1-15.2.3 LOEL-TT-LOCV)

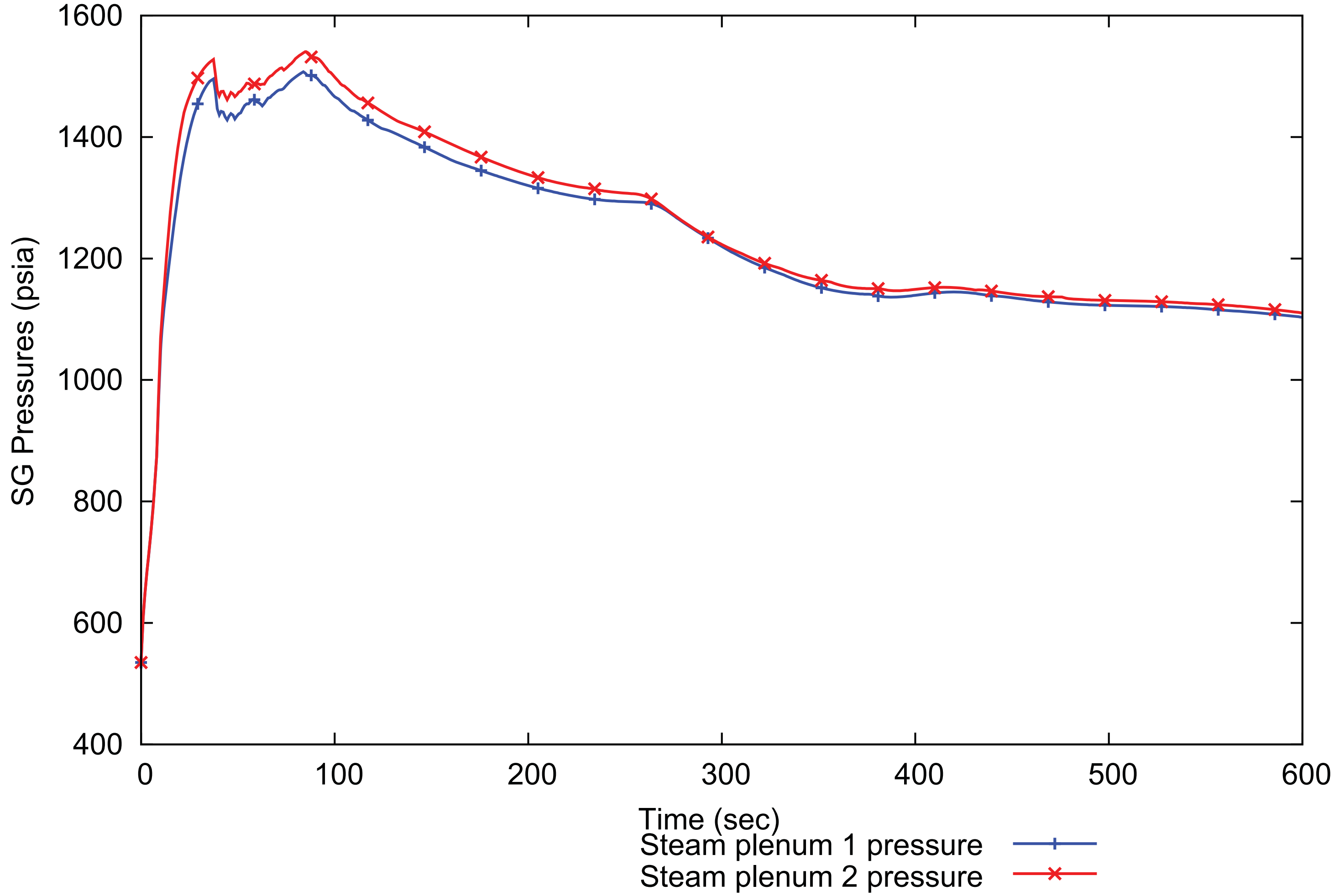


Figure 15.2-9: Hot Channel Node MCHFR - Limiting MCHFR Case (15.2.1-15.2.3 LOEL-TT-LOCV)

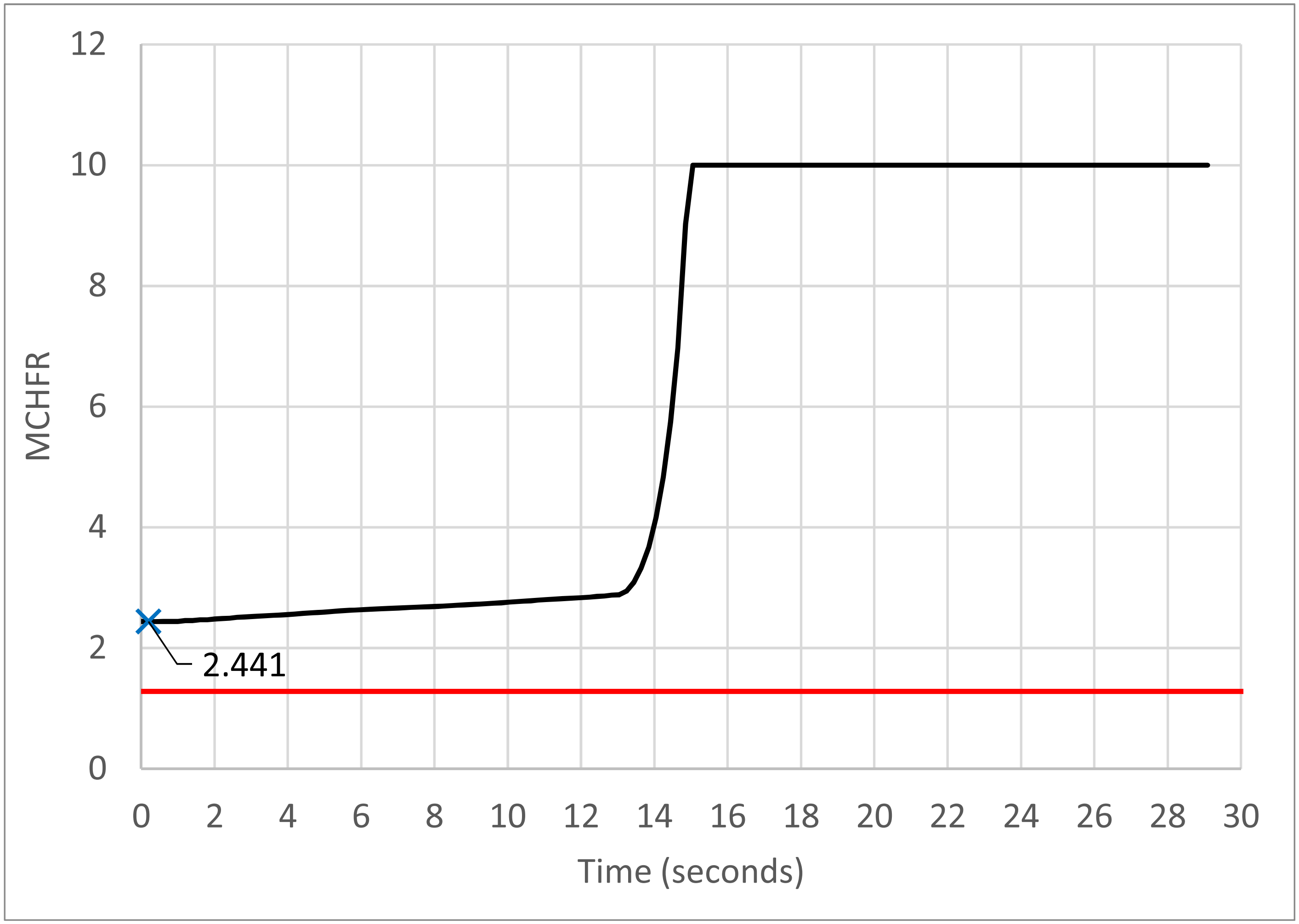


Figure 15.2-10: Reactor Power - Peak RCS Pressure Case (15.2.4 MSIV Closure)

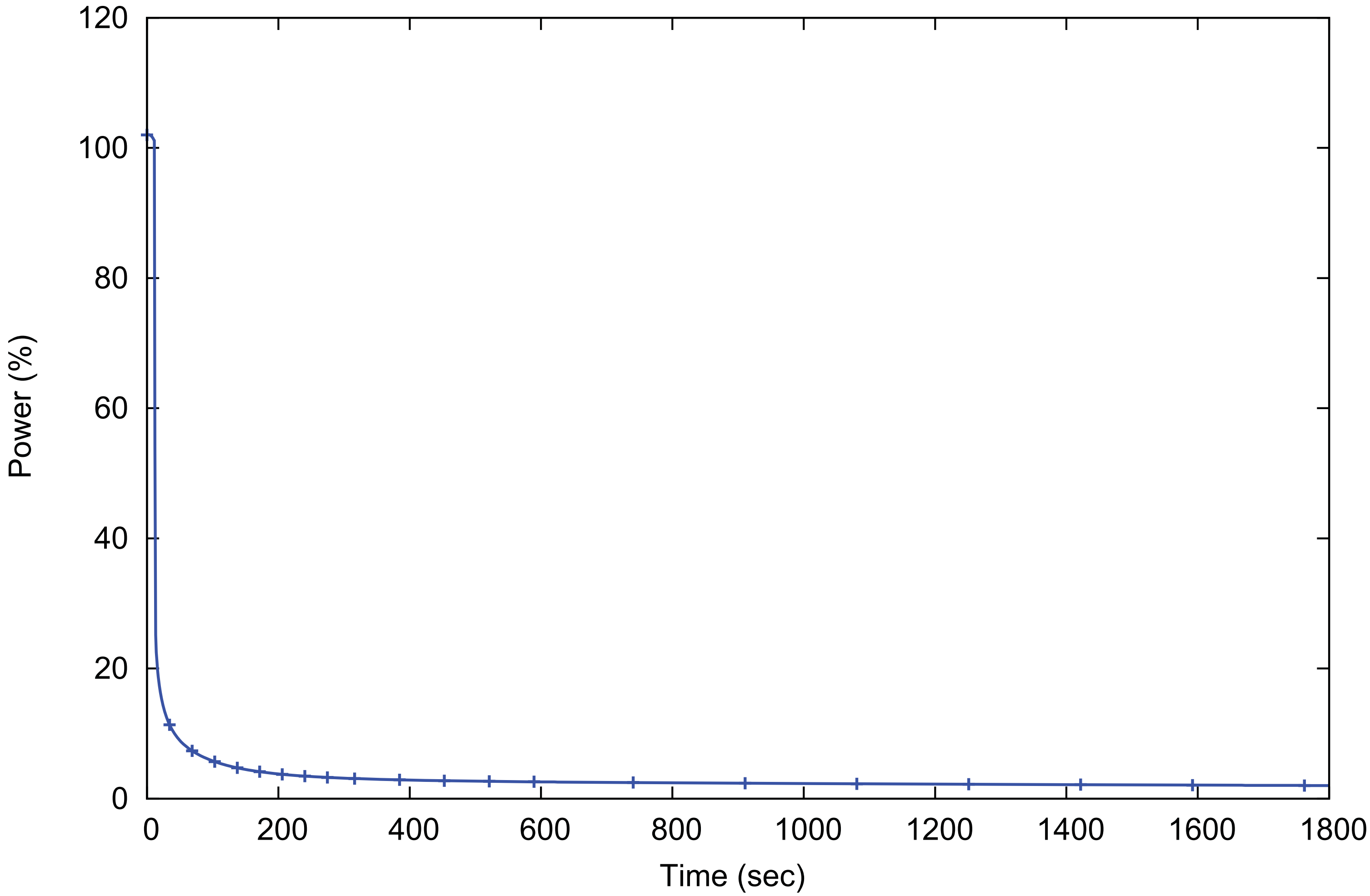


Figure 15.2-11: Reactor Pressure Vessel Lower Plenum Pressure - Peak RCS Pressure Case (15.2.4 MSIV Closure)

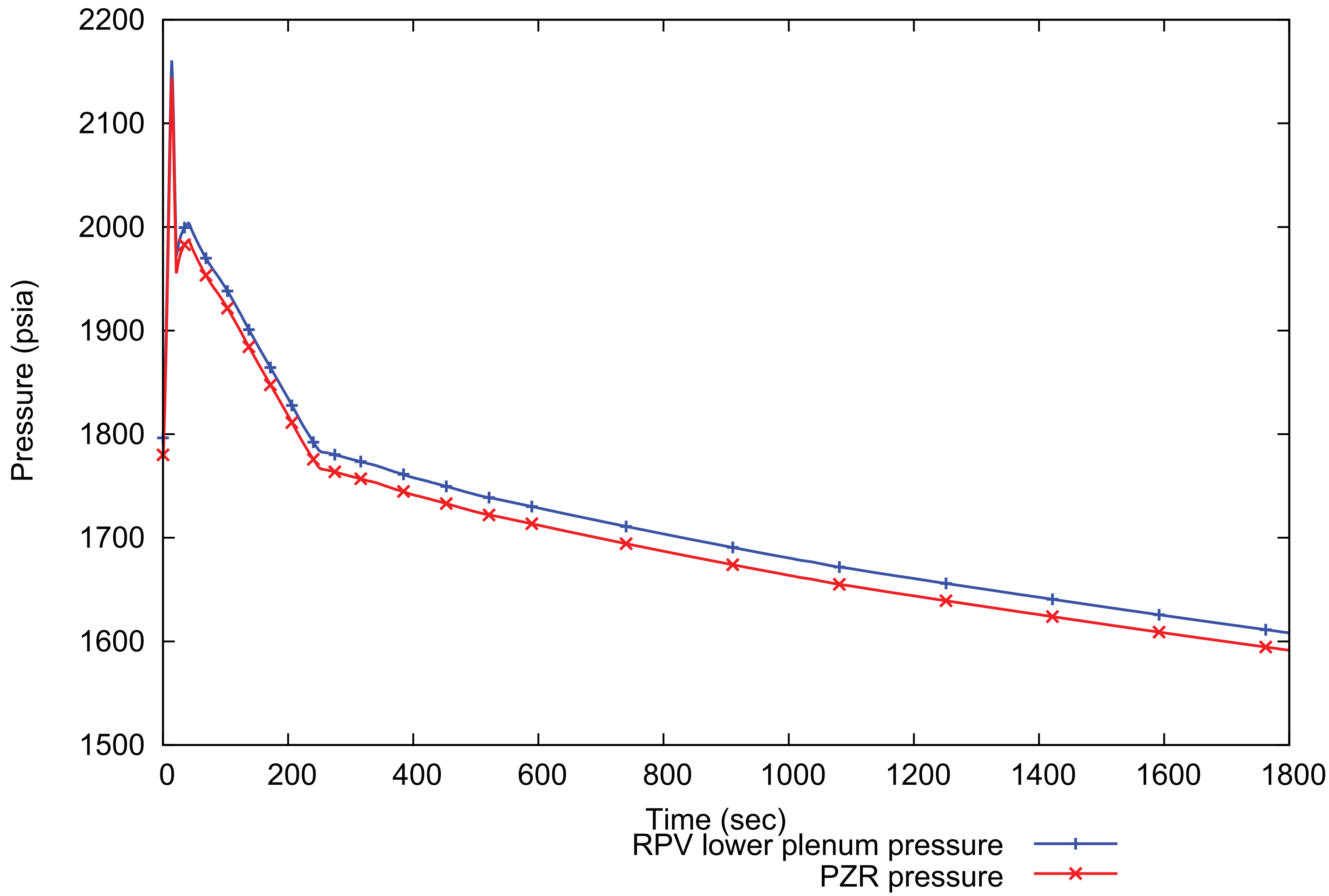


Figure 15.2-12: Volume Average Fuel Temperature - Peak RCS Pressure Case (15.2.4 MSIV Closure)

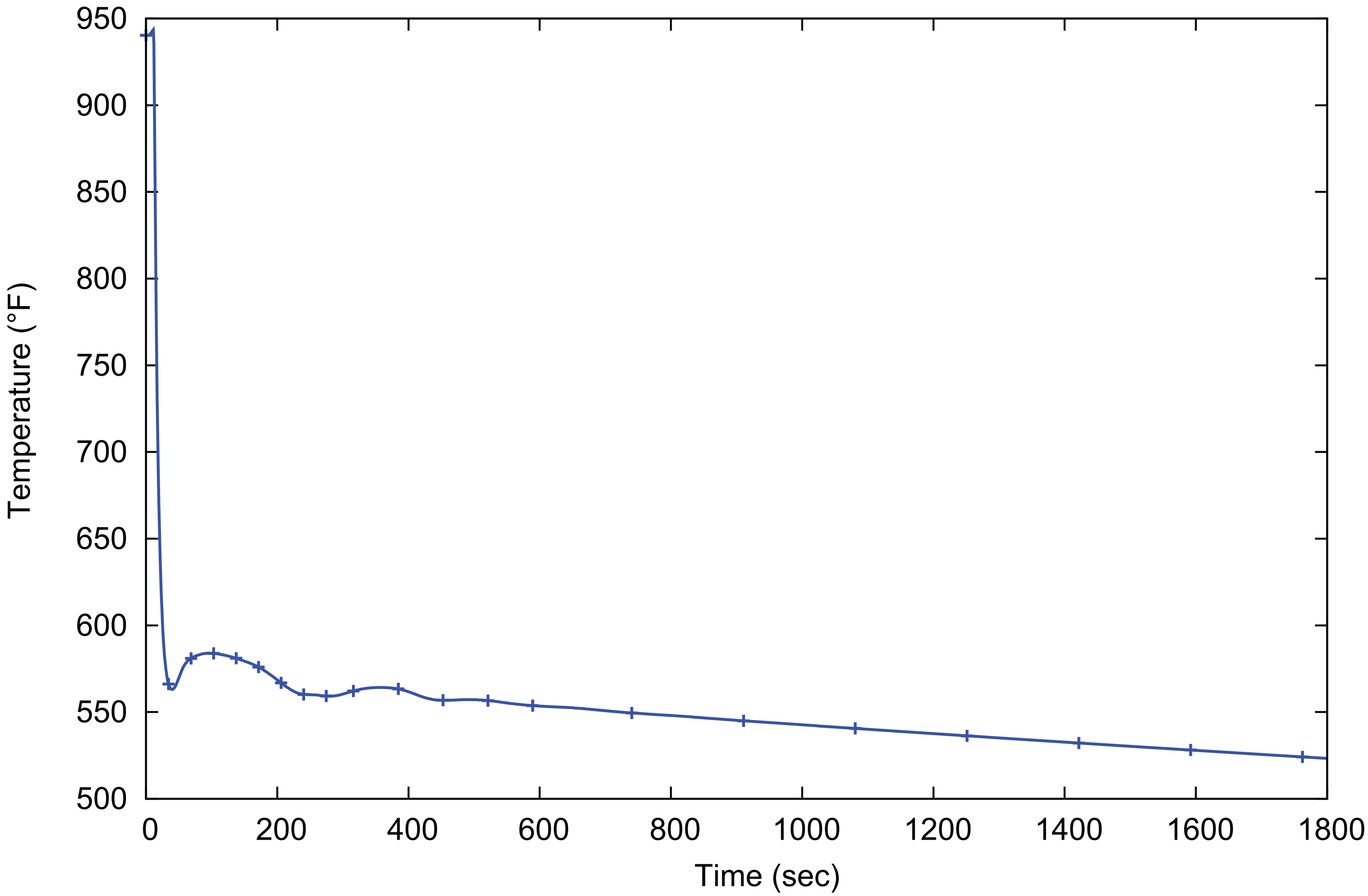


Figure 15.2-13: Reactor Coolant System Flowrate - Peak RCS Pressure Case (15.2.4 MSIV Closure)

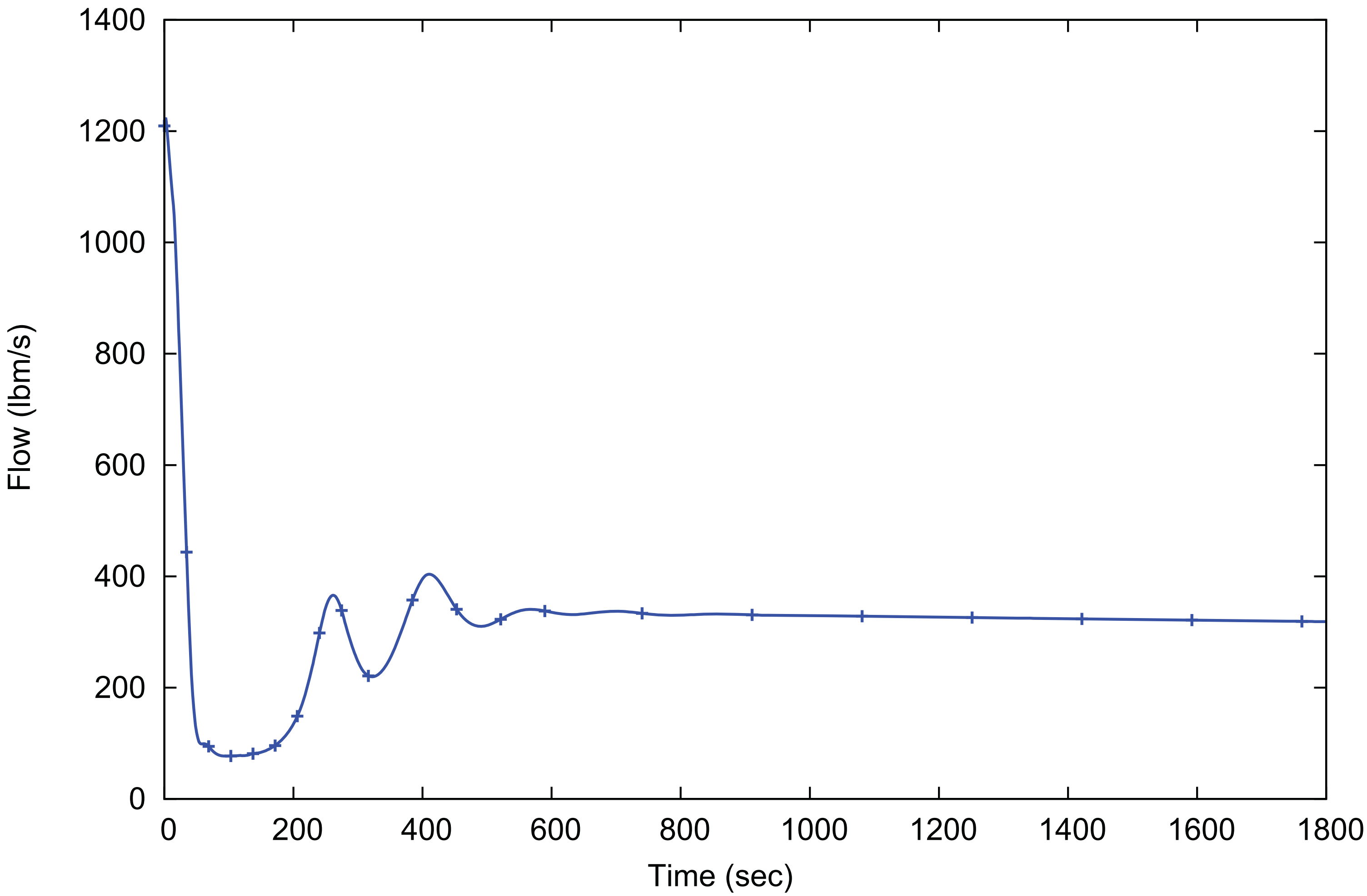


Figure 15.2-14: Primary Temperatures - Peak RCS Pressure Case (15.2.4 MSIV Closure)

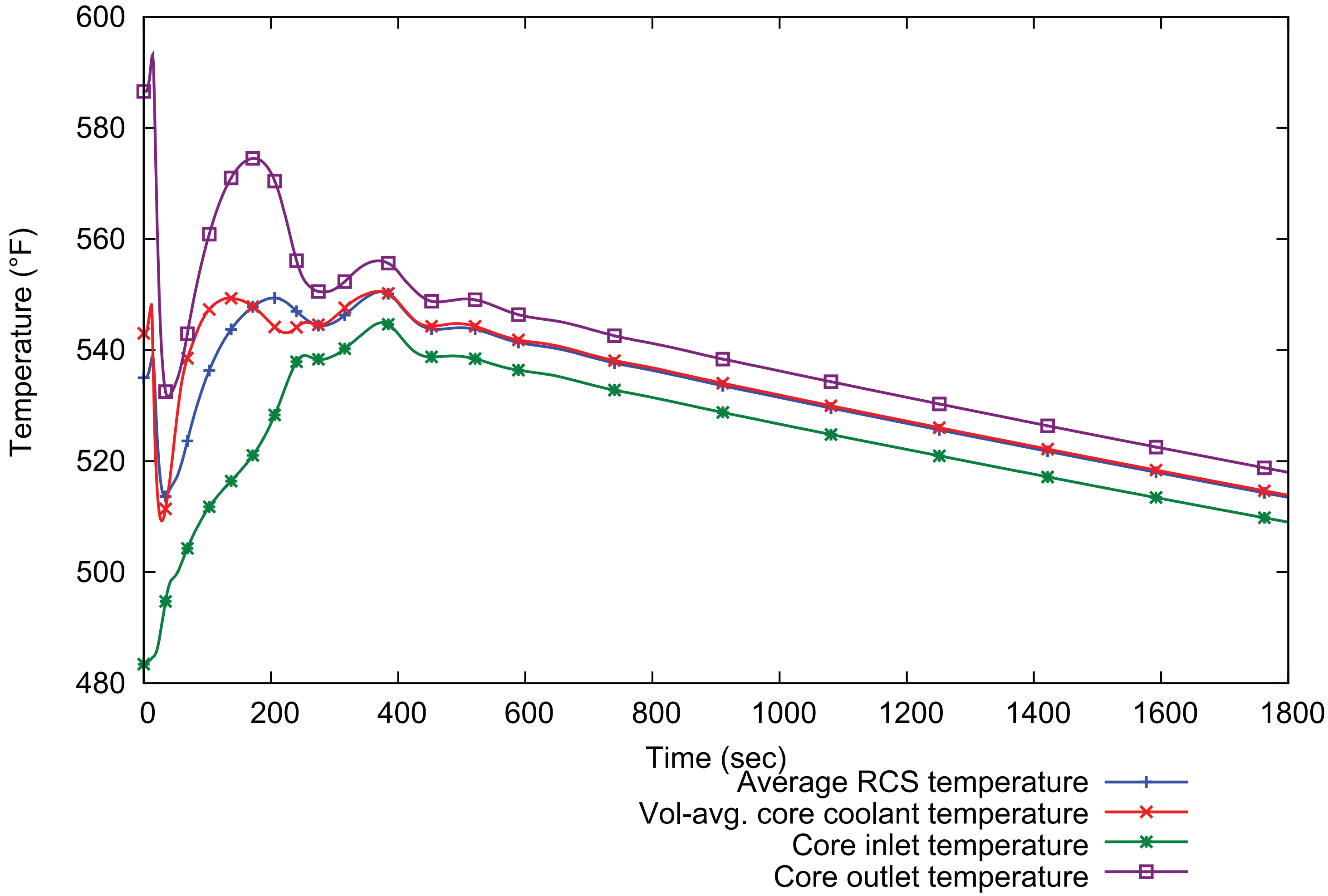


Figure 15.2-15: Pressurizer Level - Peak RCS Pressure Case (15.2.4 MSIV Closure)

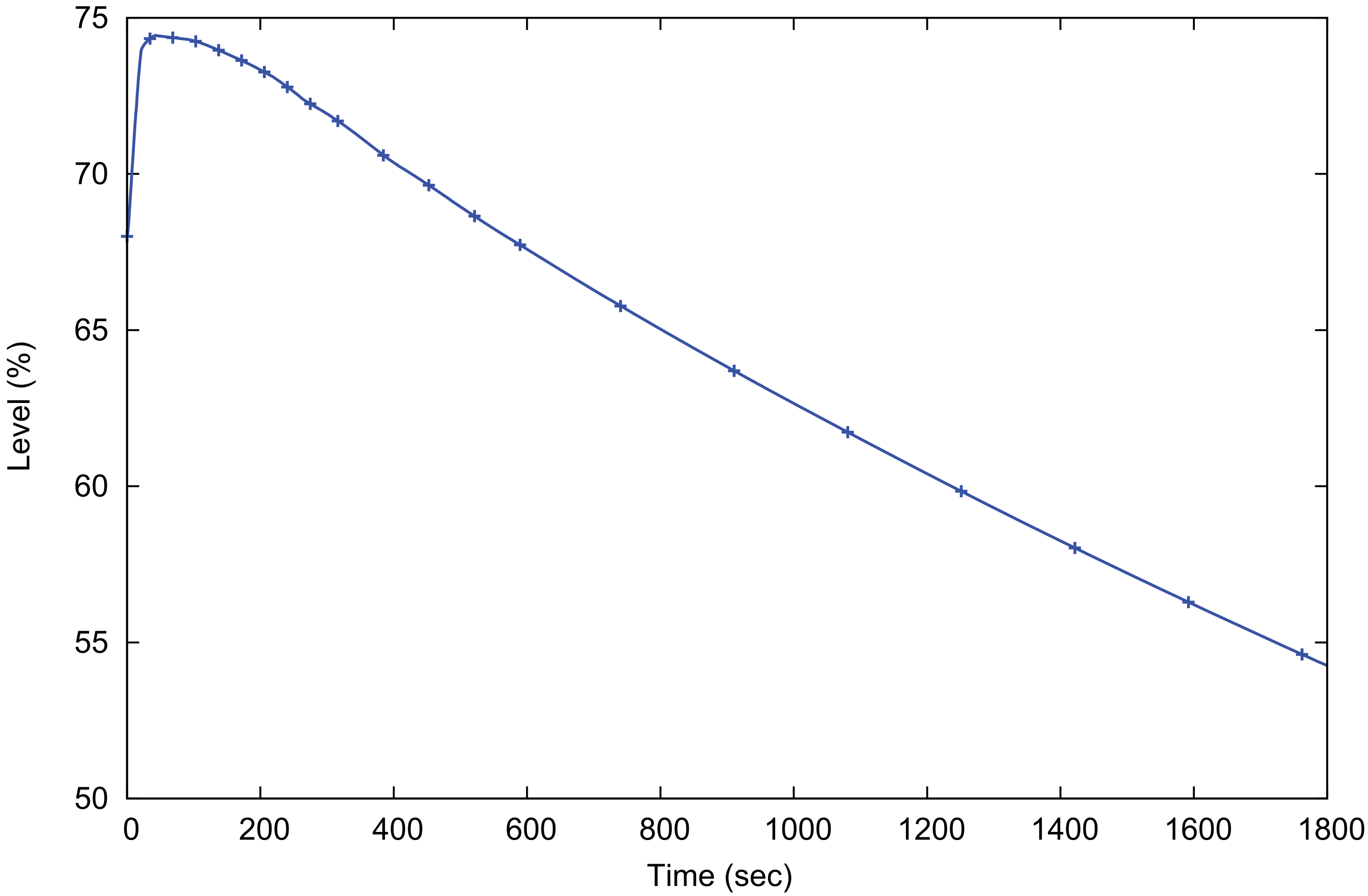


Figure 15.2-16: Steam Generator Pressure (Typical) - Peak SG Pressure Case (15.2.4 MSIV Closure)

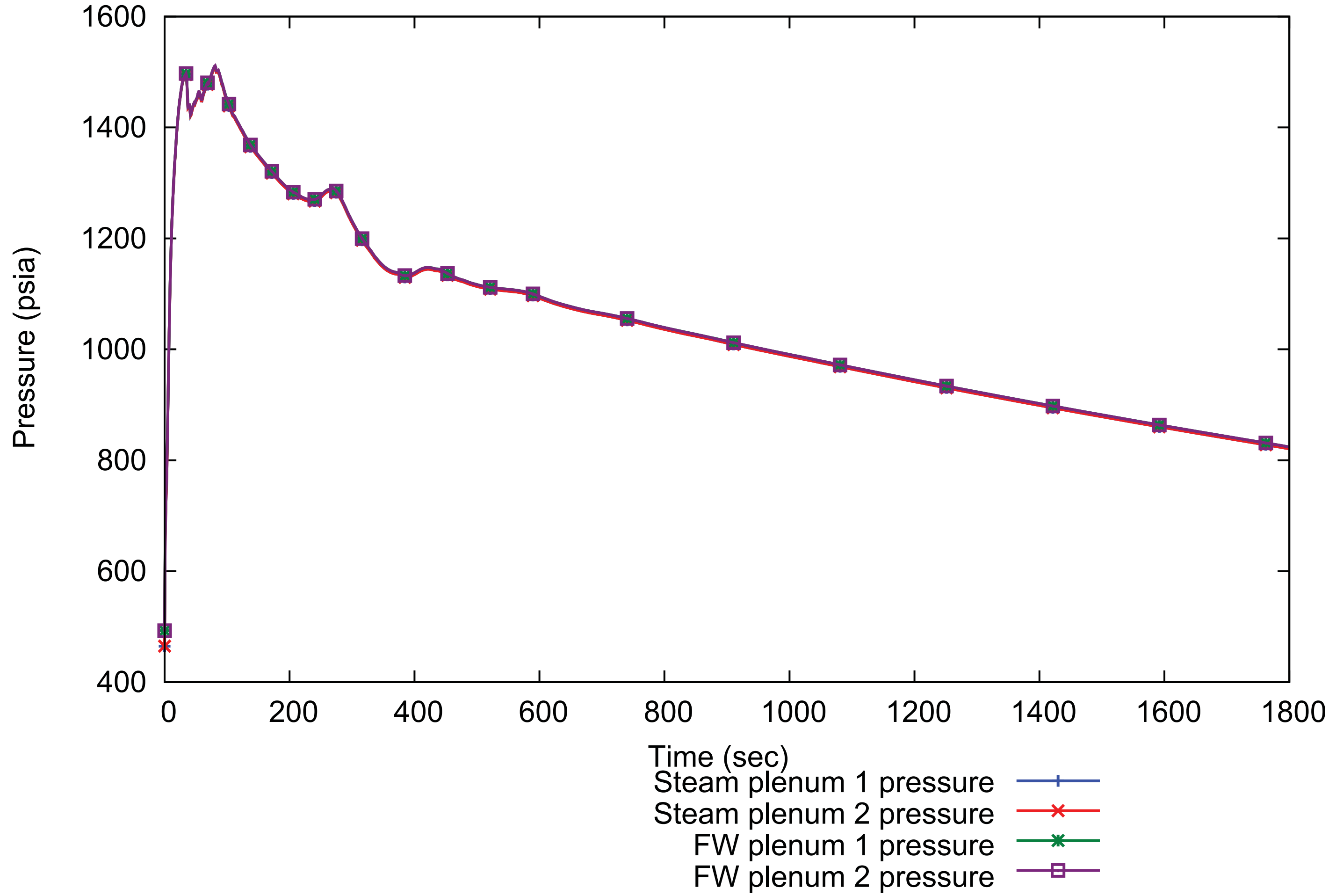


Figure 15.2-17: Steam Generator Level - Peak SG Pressure Case (15.2.4 MSIV Closure)

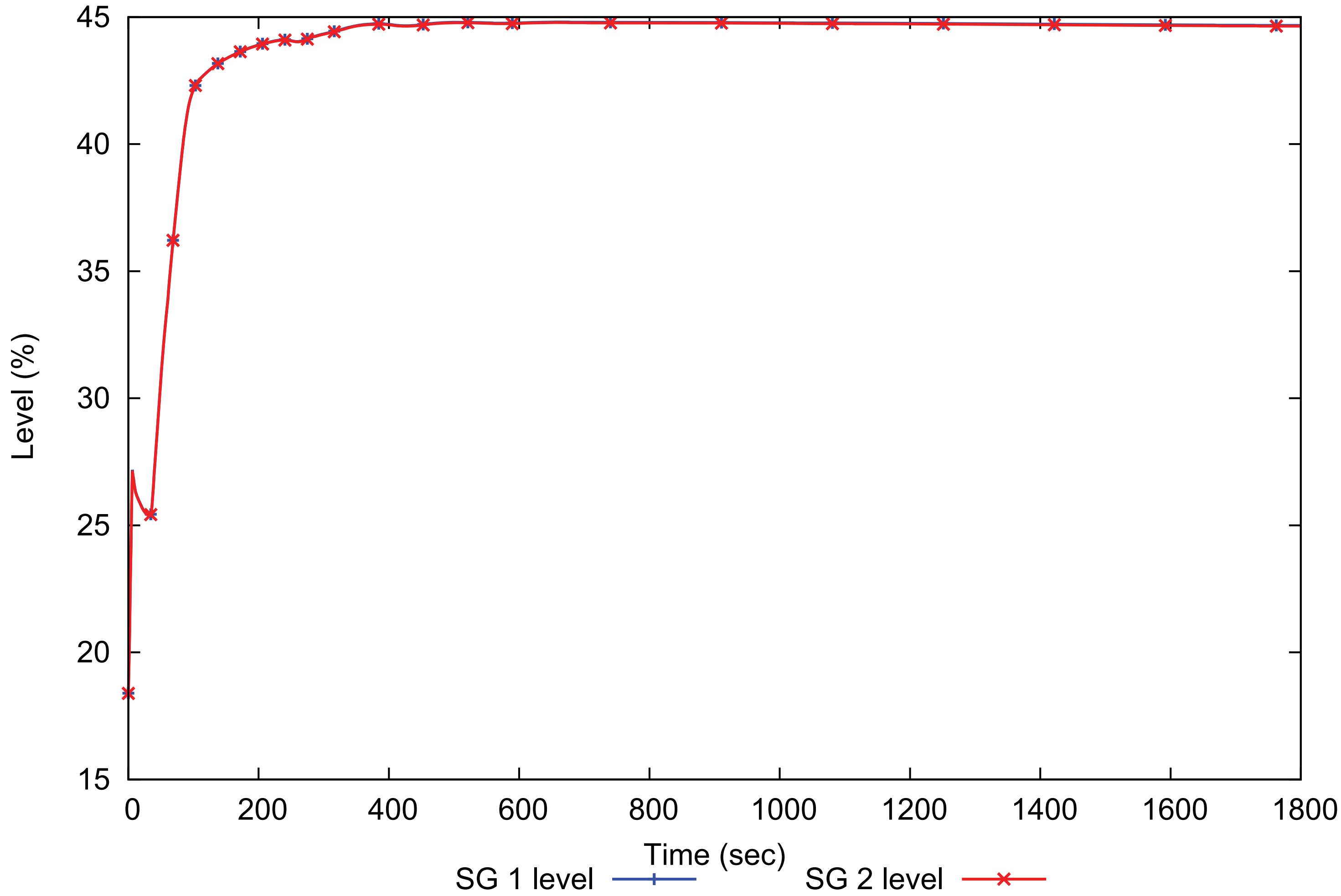


Figure 15.2-18: Minimum Critical Heat Flux Ratio - Limiting MCHFR Case (15.2.4 MSIV Closure)

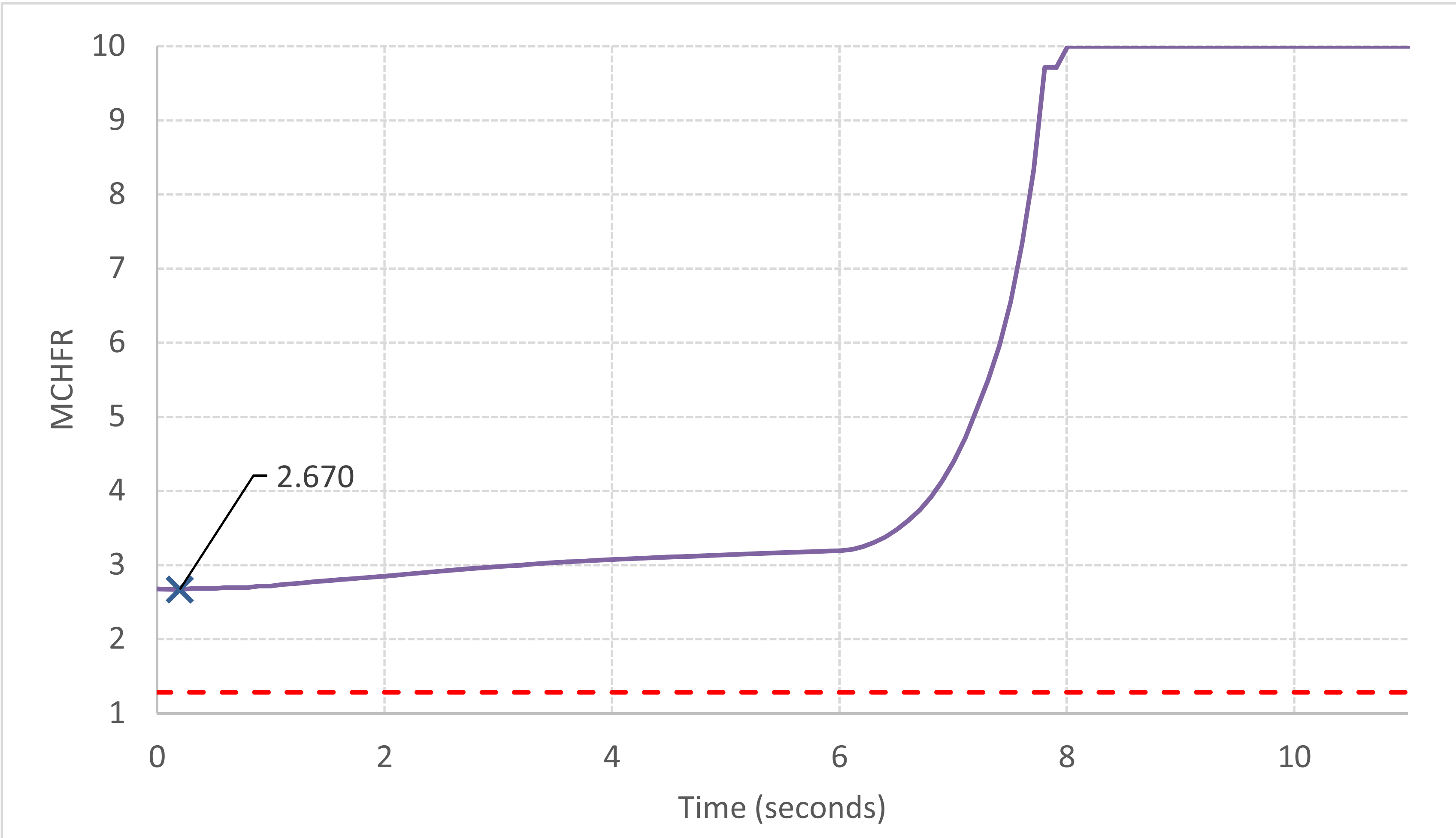


Figure 15.2-19: Reactor Power - Peak RCS Pressure Case (15.2.6 Loss of AC Power)

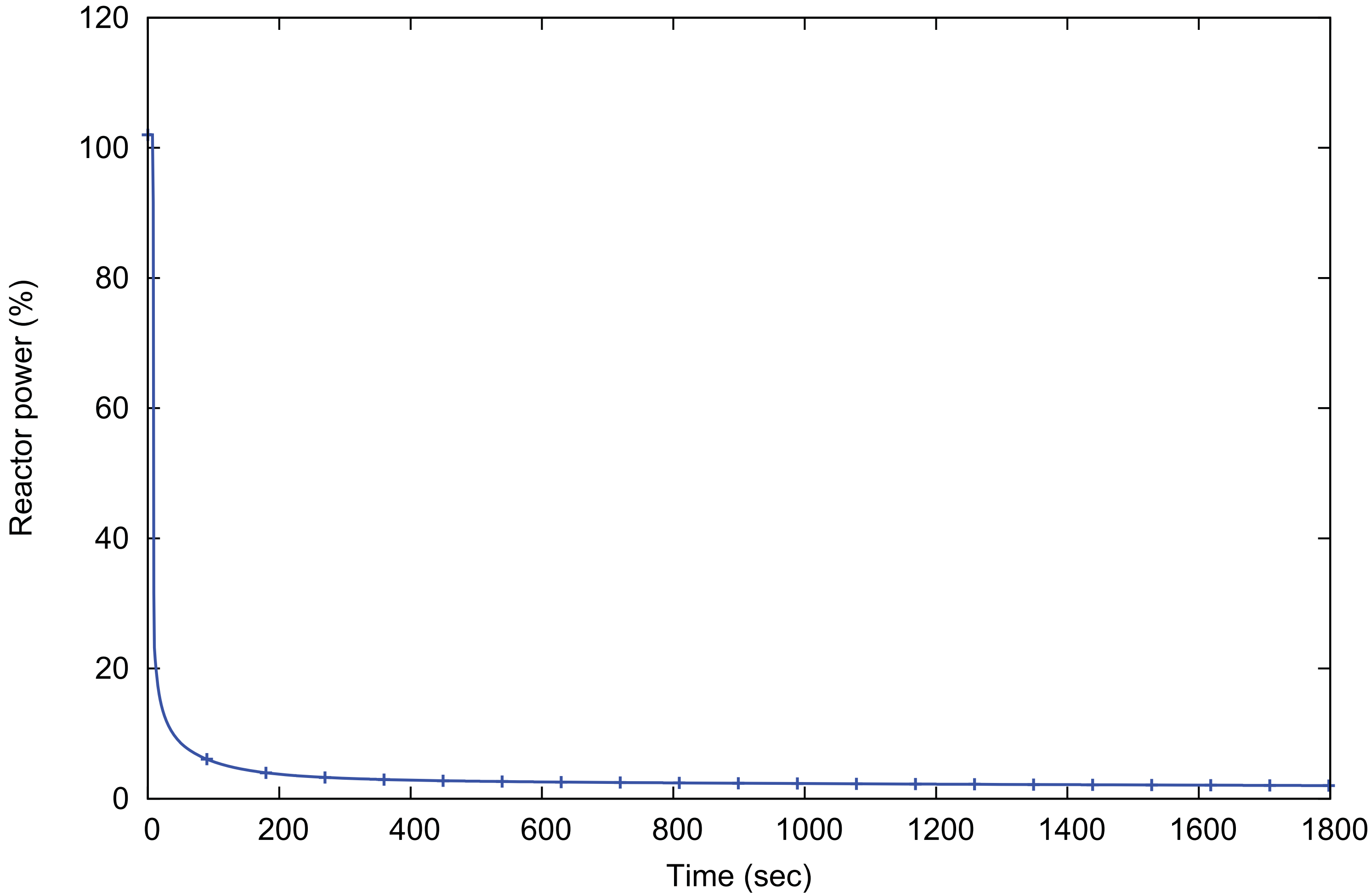


Figure 15.2-20: Total Reactivity - Peak RCS Pressure Case (15.2.6 Loss of AC Power)

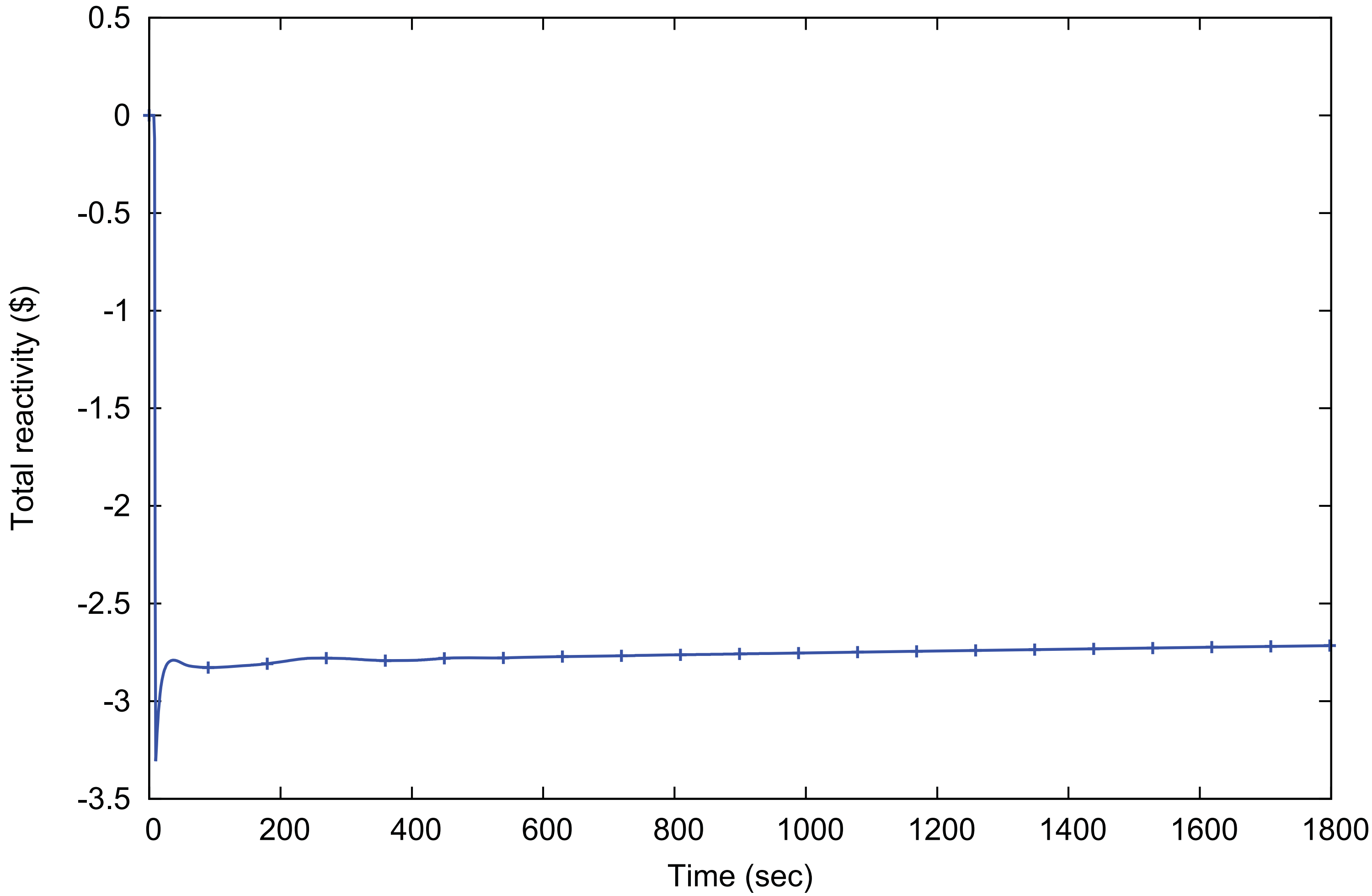


Figure 15.2-21: Pressurizer Level - Peak RCS Pressure Case (15.2.6 Loss of AC Power)

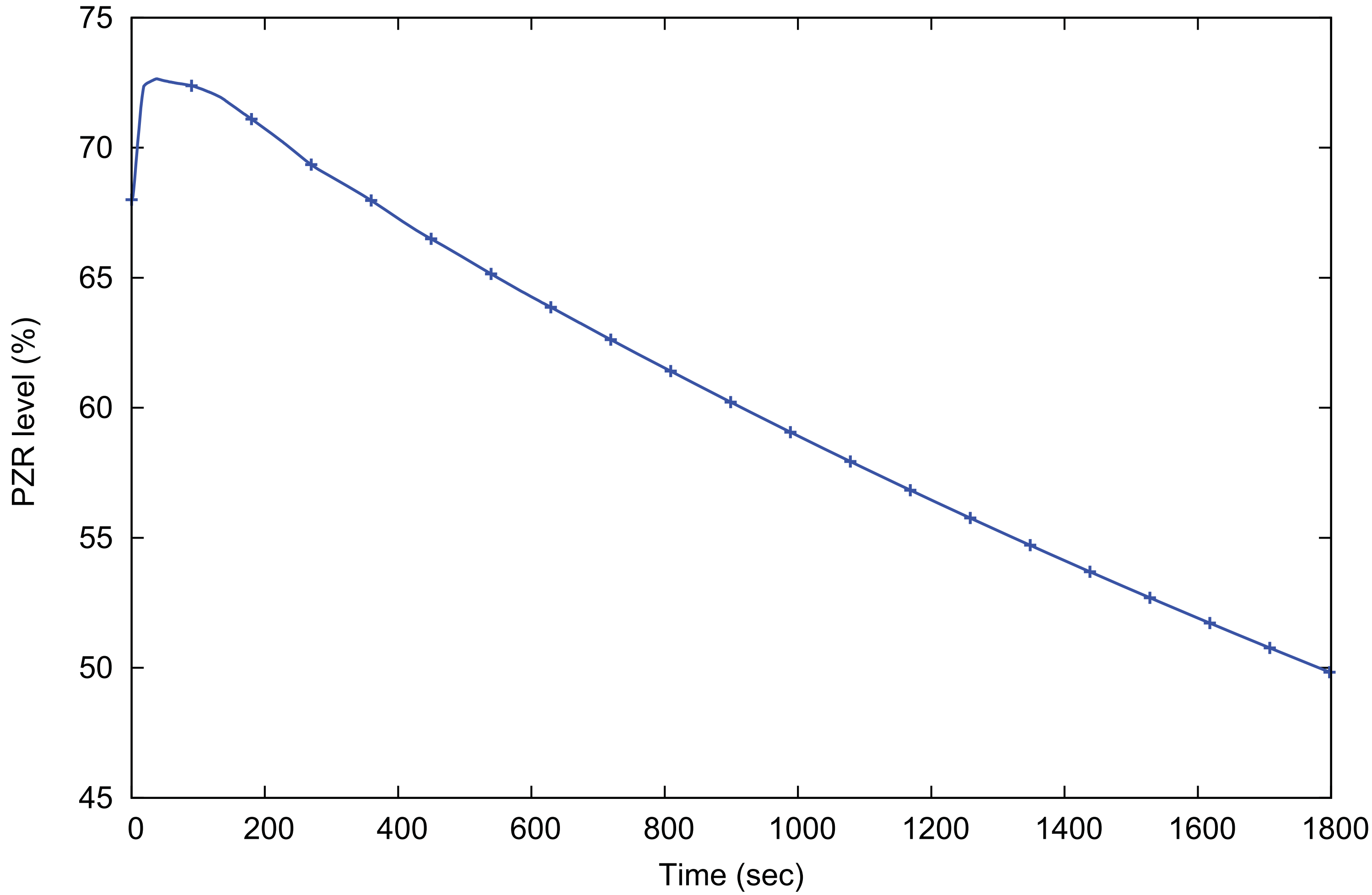


Figure 15.2-22: Reactor Pressure Vessel Pressure - Peak RCS Pressure Case (15.2.6 Loss of AC Power)

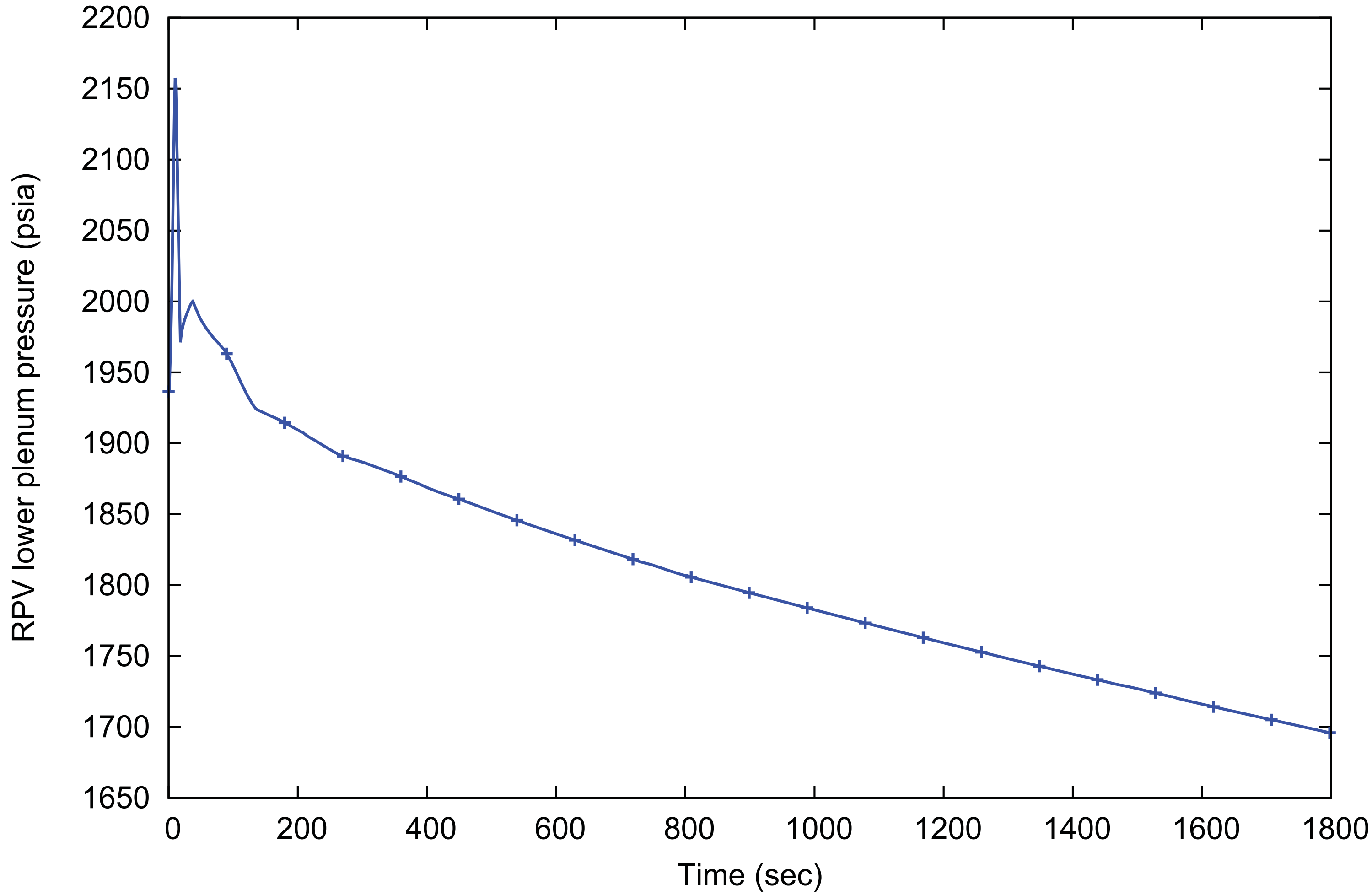


Figure 15.2-23: Reactor Coolant System Flowrate - Peak RCS Pressure Case (15.2.6 Loss of AC Power)

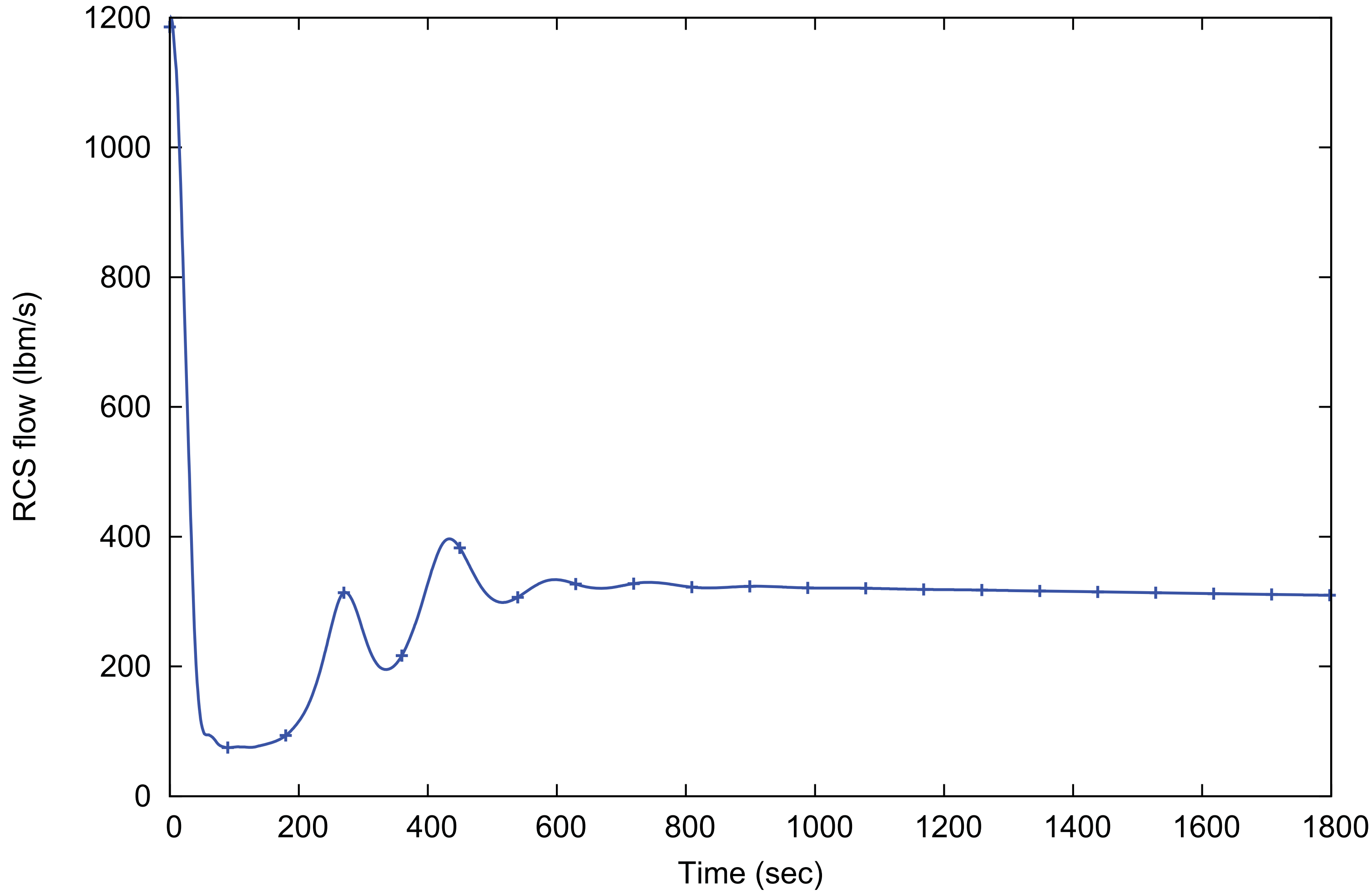


Figure 15.2-24: Reactor Coolant System Average Temperature - Peak RCS Pressure Case (15.2.6 Loss of AC Power)

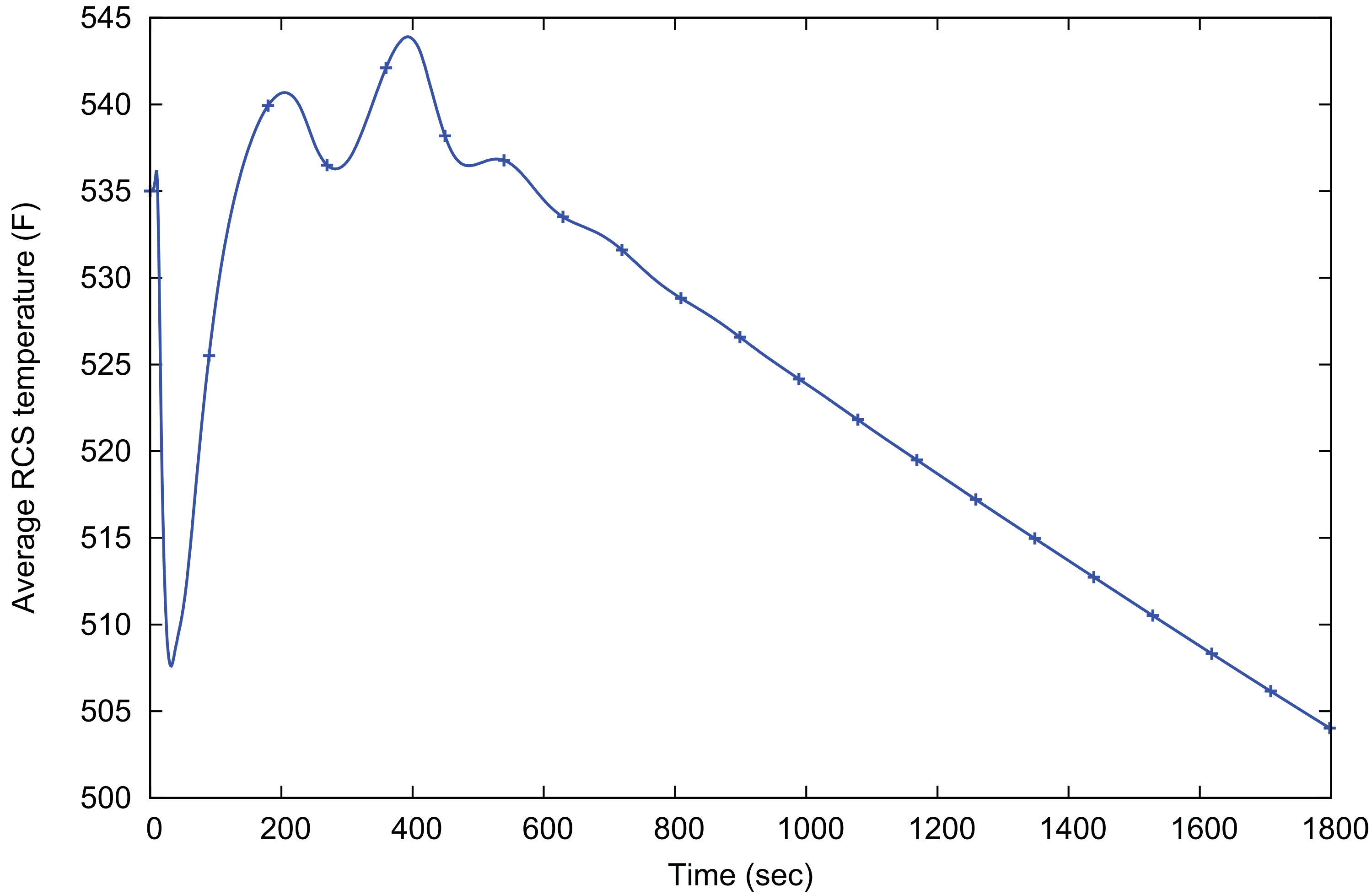


Figure 15.2-25: Steam Generator Pressure - Peak SG Pressure Case (15.2.6 Loss of AC Power)

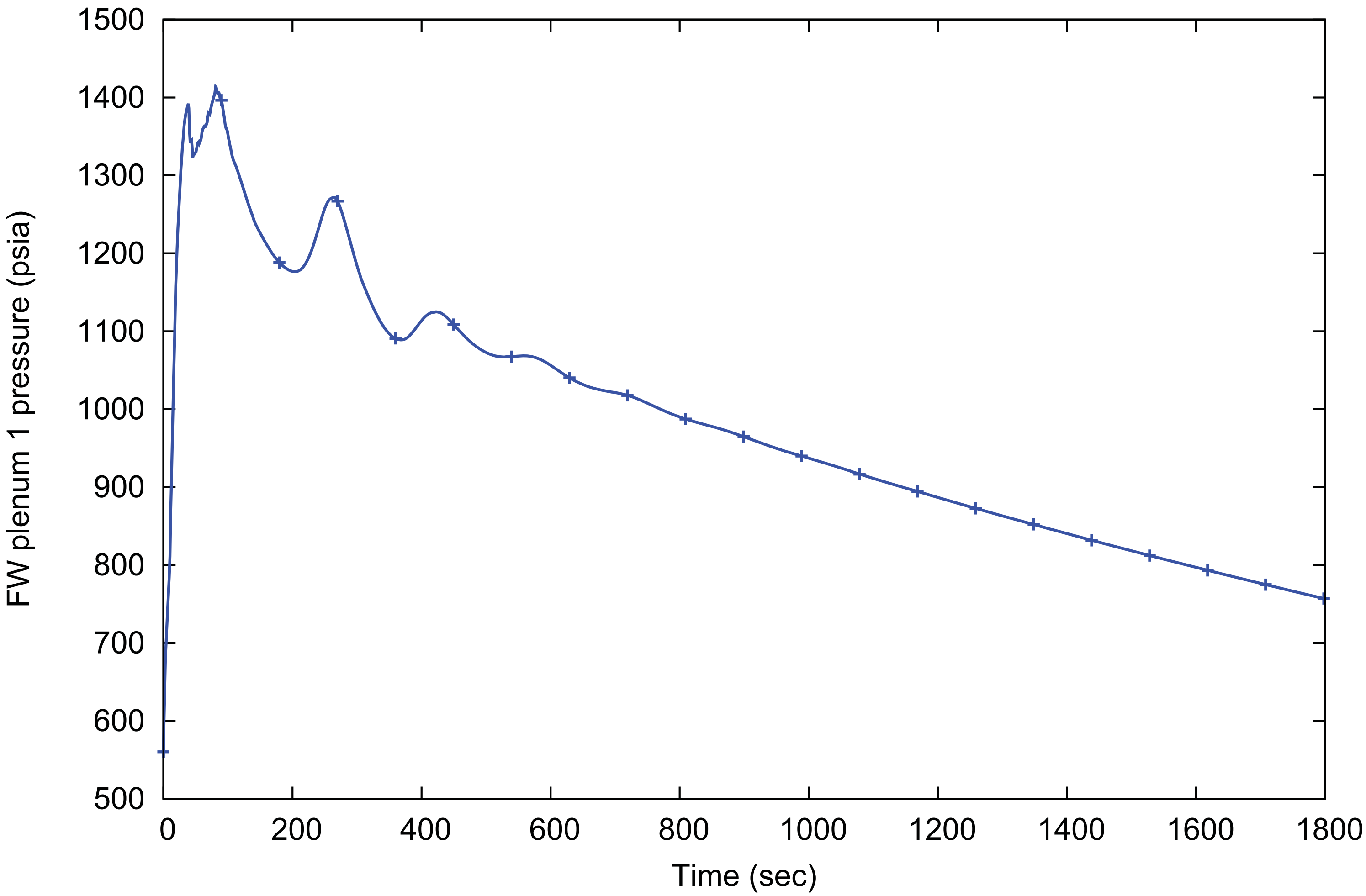


Figure 15.2-26: Hot Channel Node Critical Heat Flux Ratio - MCHFR Case (15.2.6 Loss of AC Power)

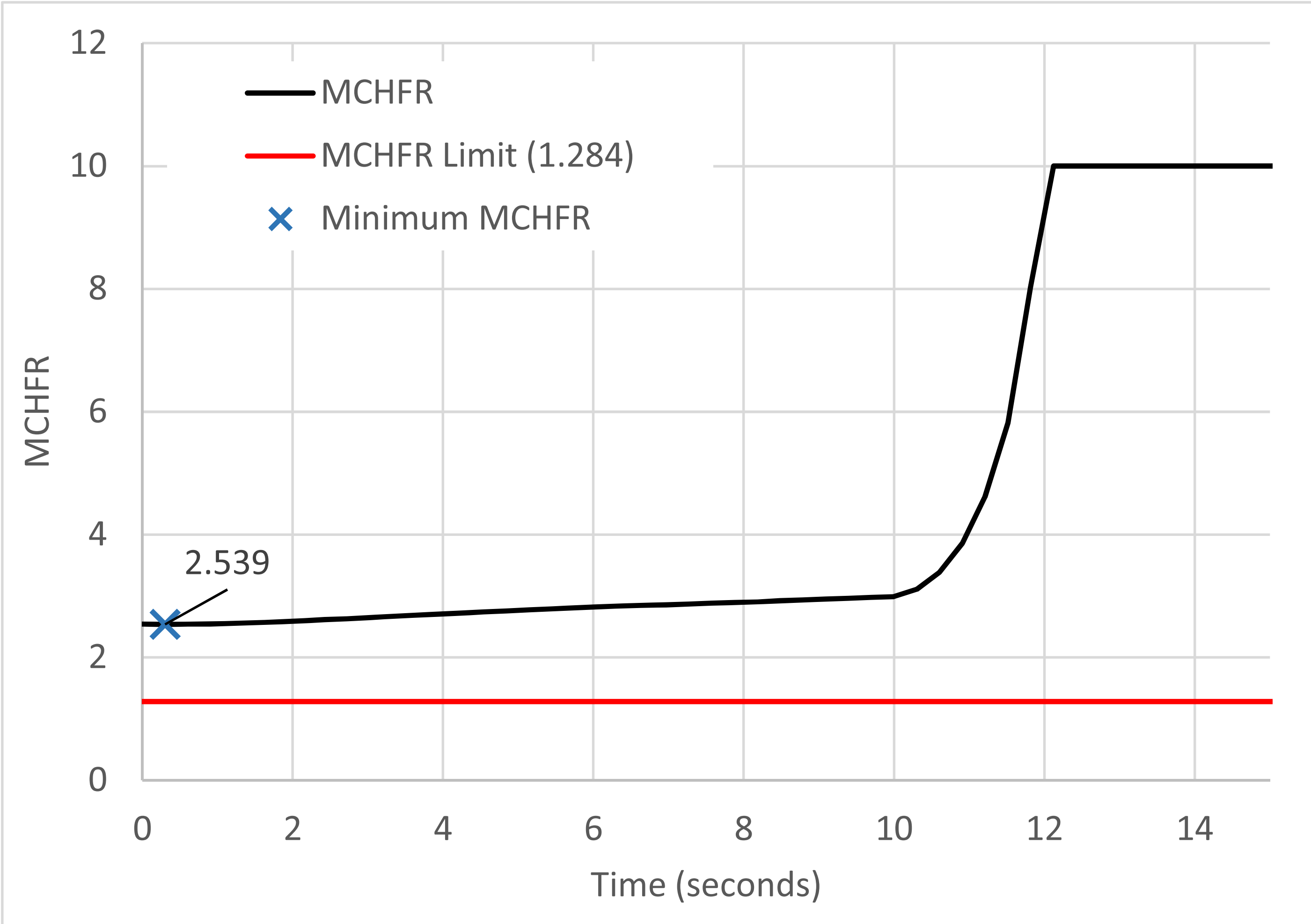


Figure 15.2-27: Reactor Power - Peak RCS Pressure Case (15.2.7 Loss of Feedwater)

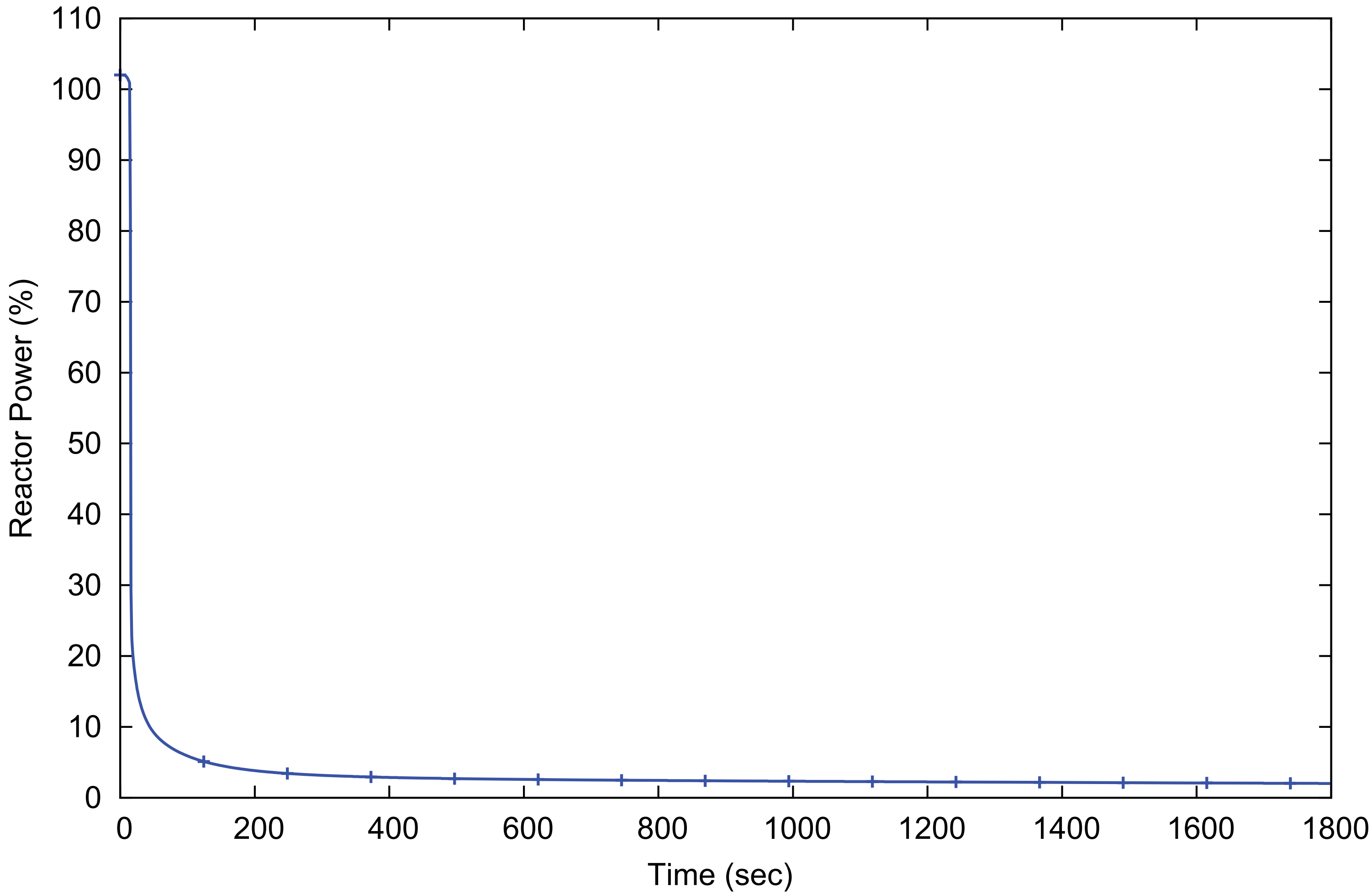


Figure 15.2-28: Reactor Pressure Vessel Pressure - Peak RCS Pressure Case (15.2.7 Loss of Feedwater)

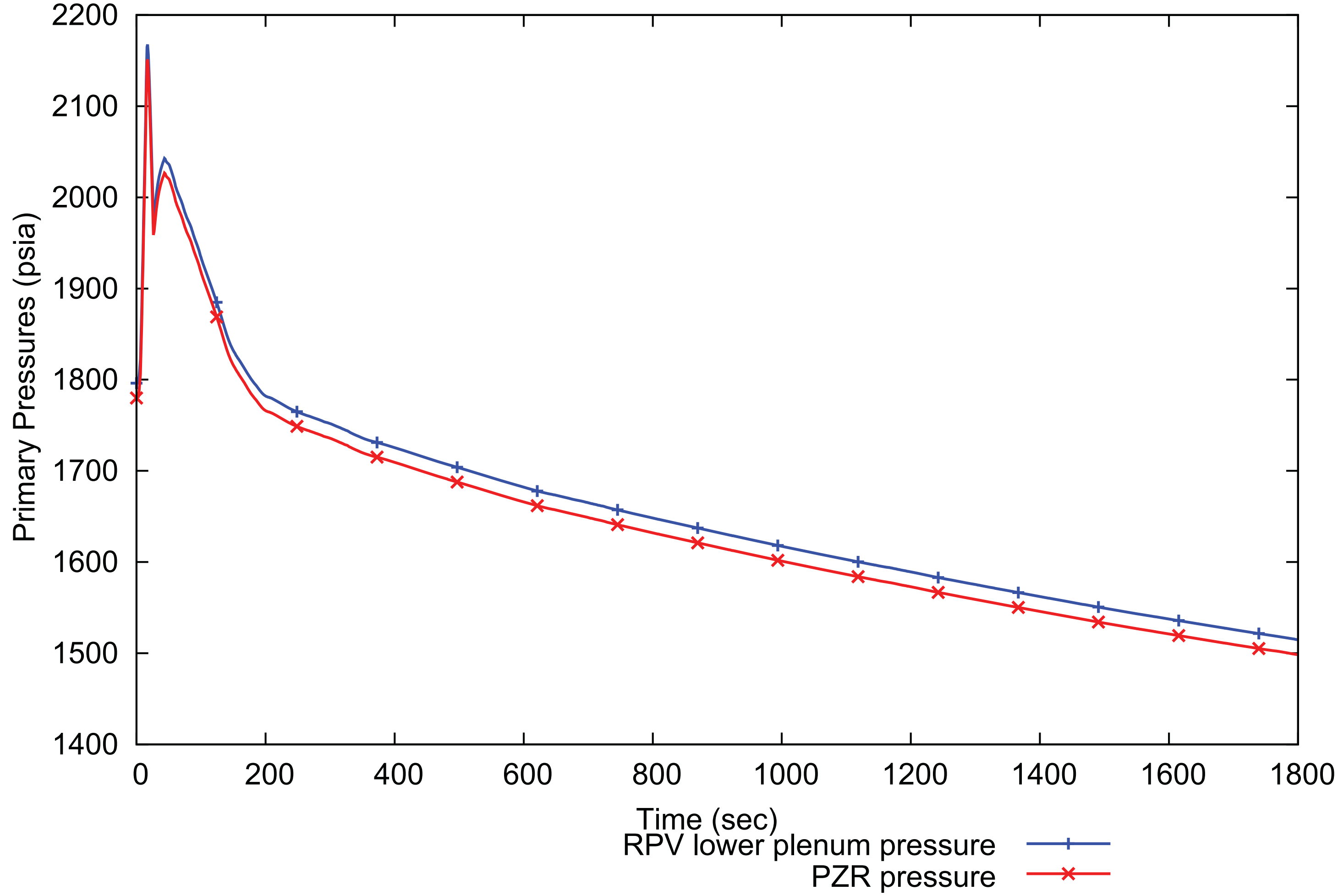


Figure 15.2-29: DHRS Flow - Peak RCS Pressure Case (15.2.7 Loss of Feedwater)

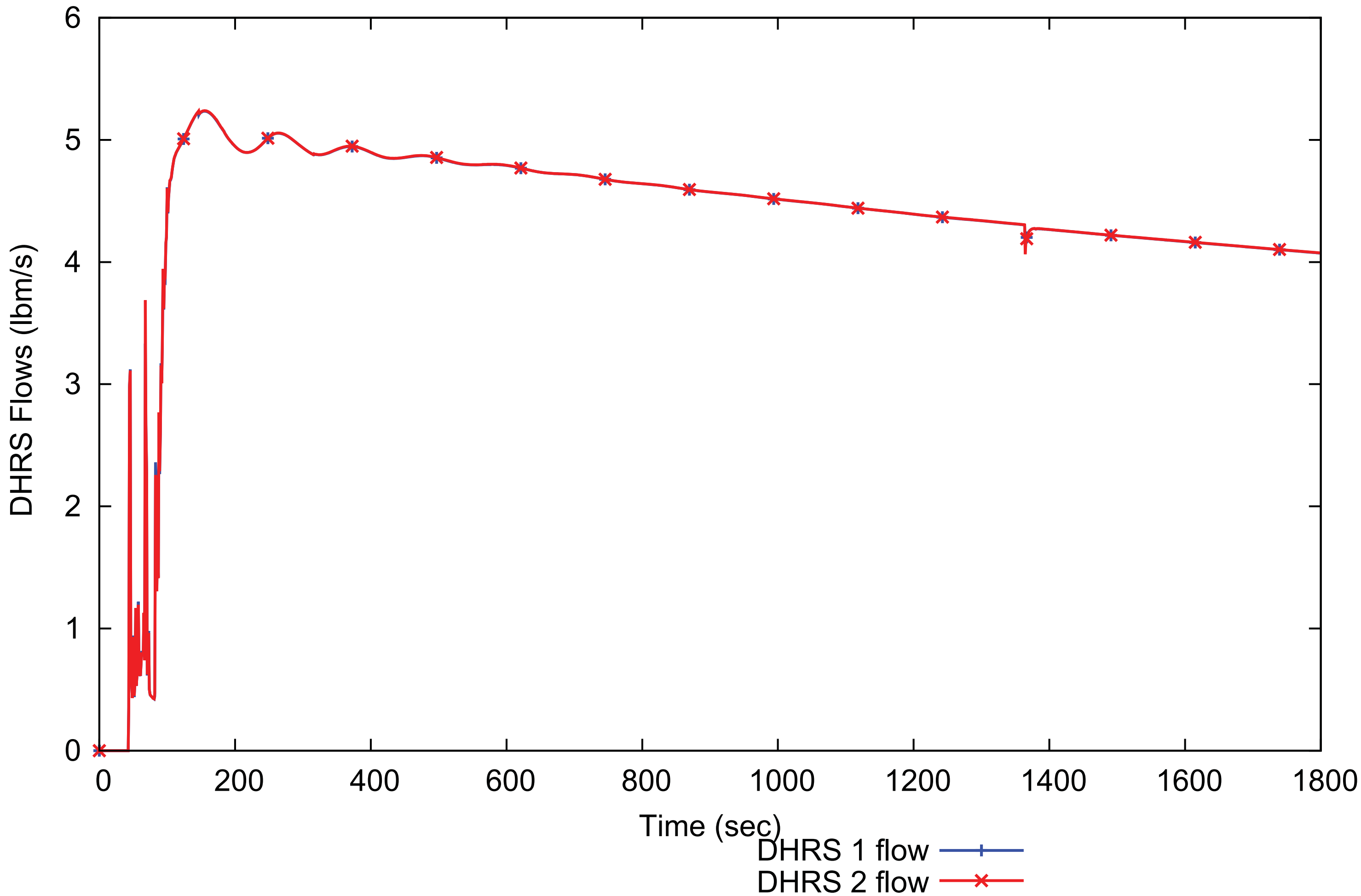


Figure 15.2-30: Reactor Coolant System Flowrate - Peak RCS Pressure Case (15.2.7 Loss of Feedwater)

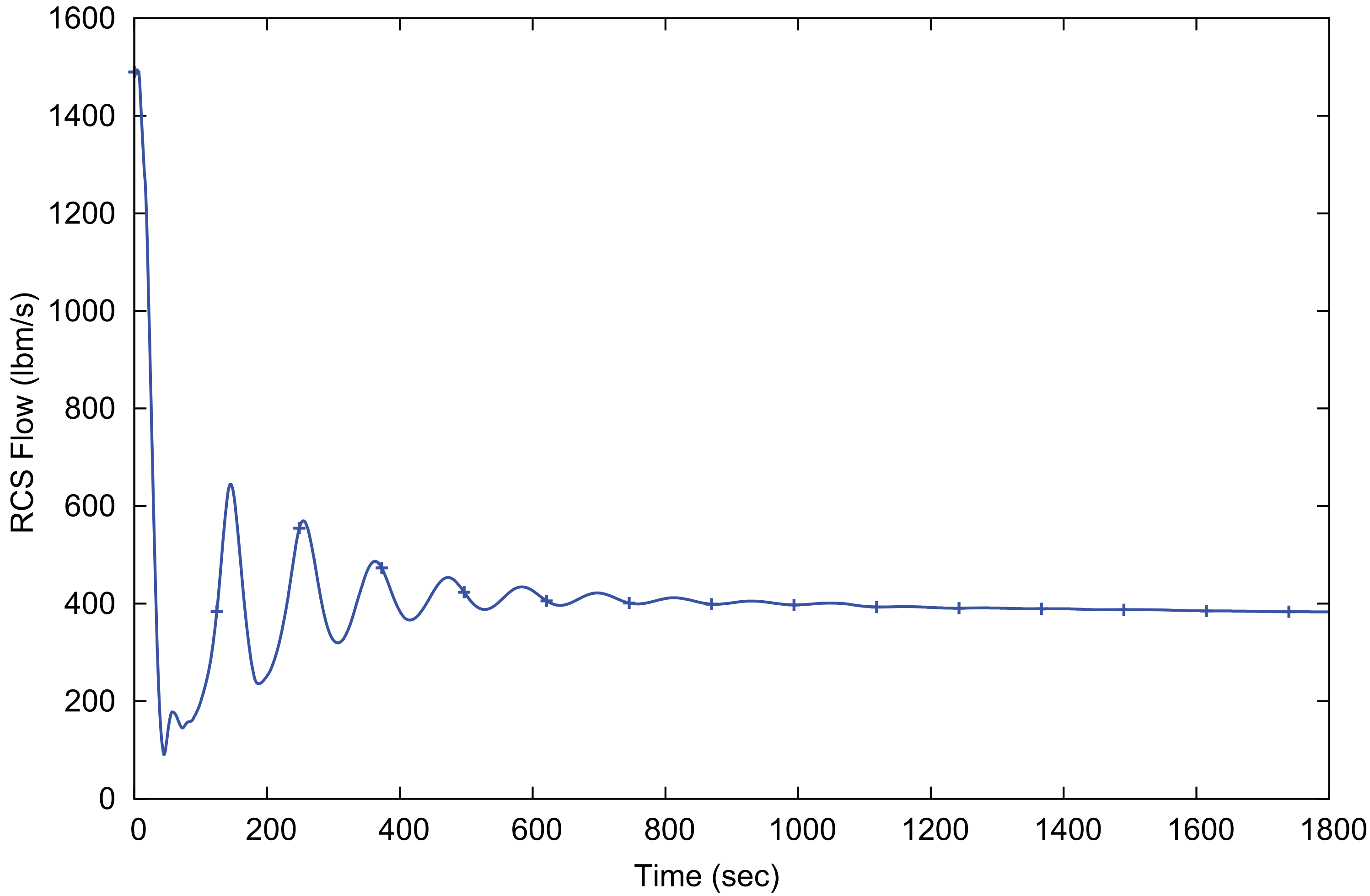


Figure 15.2-31: Reactor Coolant System Temperature - Peak RCS Pressure Case (15.2.7 Loss of Feedwater)

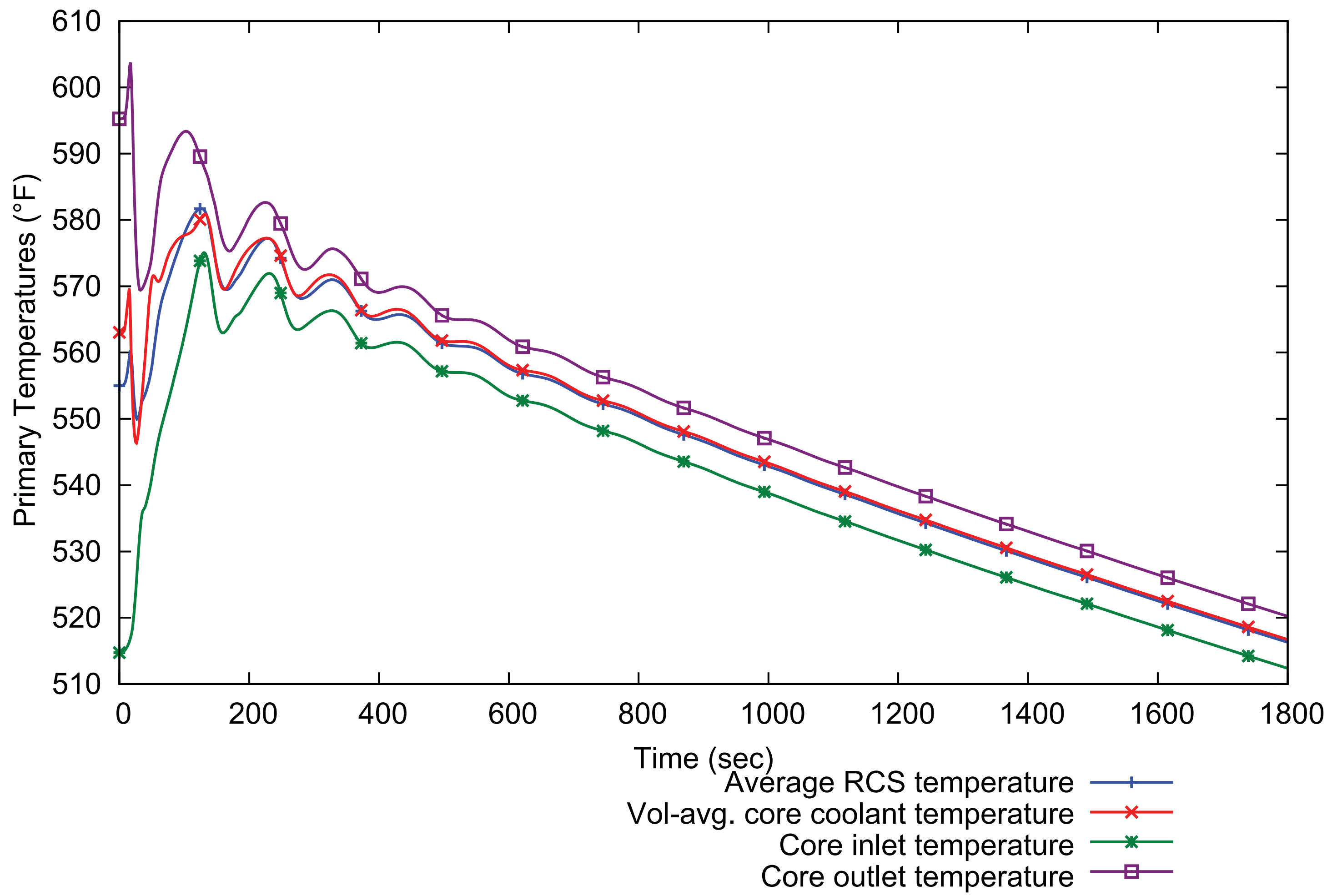


Figure 15.2-32: Pressurizer Level - Peak RCS Pressure Case (15.2.7 Loss of Feedwater)

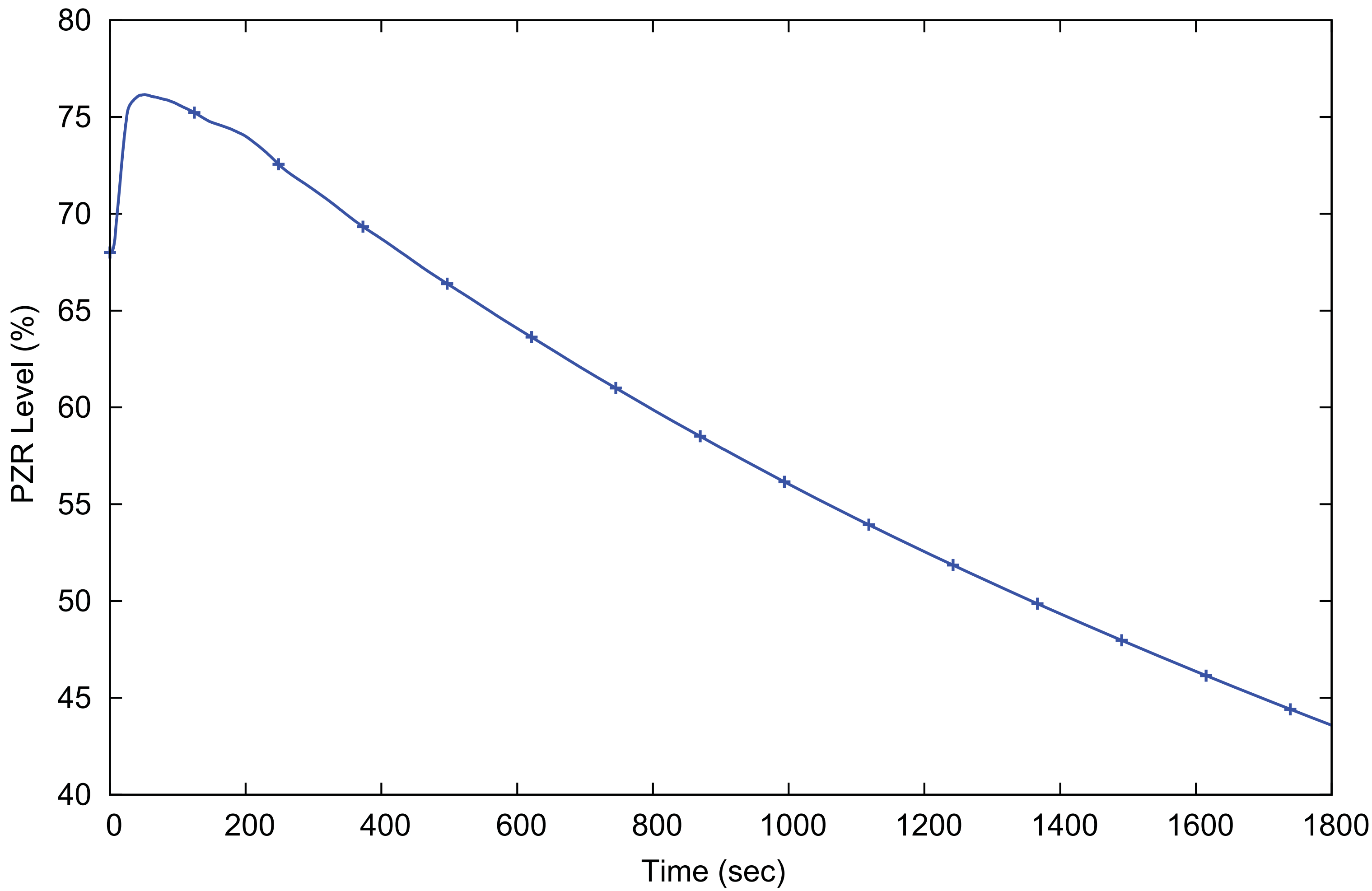


Figure 15.2-33: Steam Generator Inlet Pressure (Typical) - Peak SG Pressure Case (15.2.7 Loss of Feedwater)

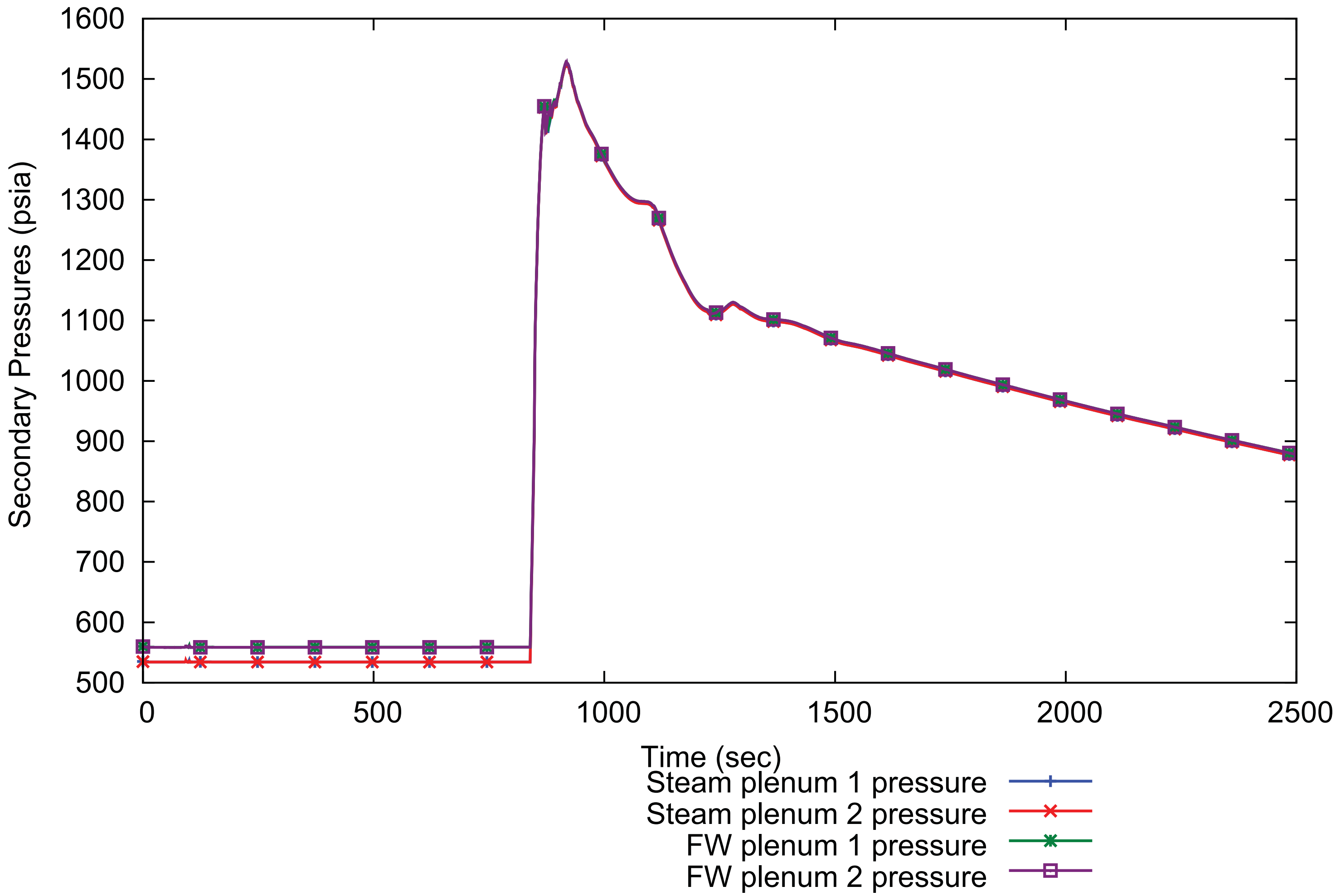


Figure 15.2-34: Minimum Critical Heat Flux Ratio - Limiting MCHFR Case (15.2.7 Loss of Feedwater)

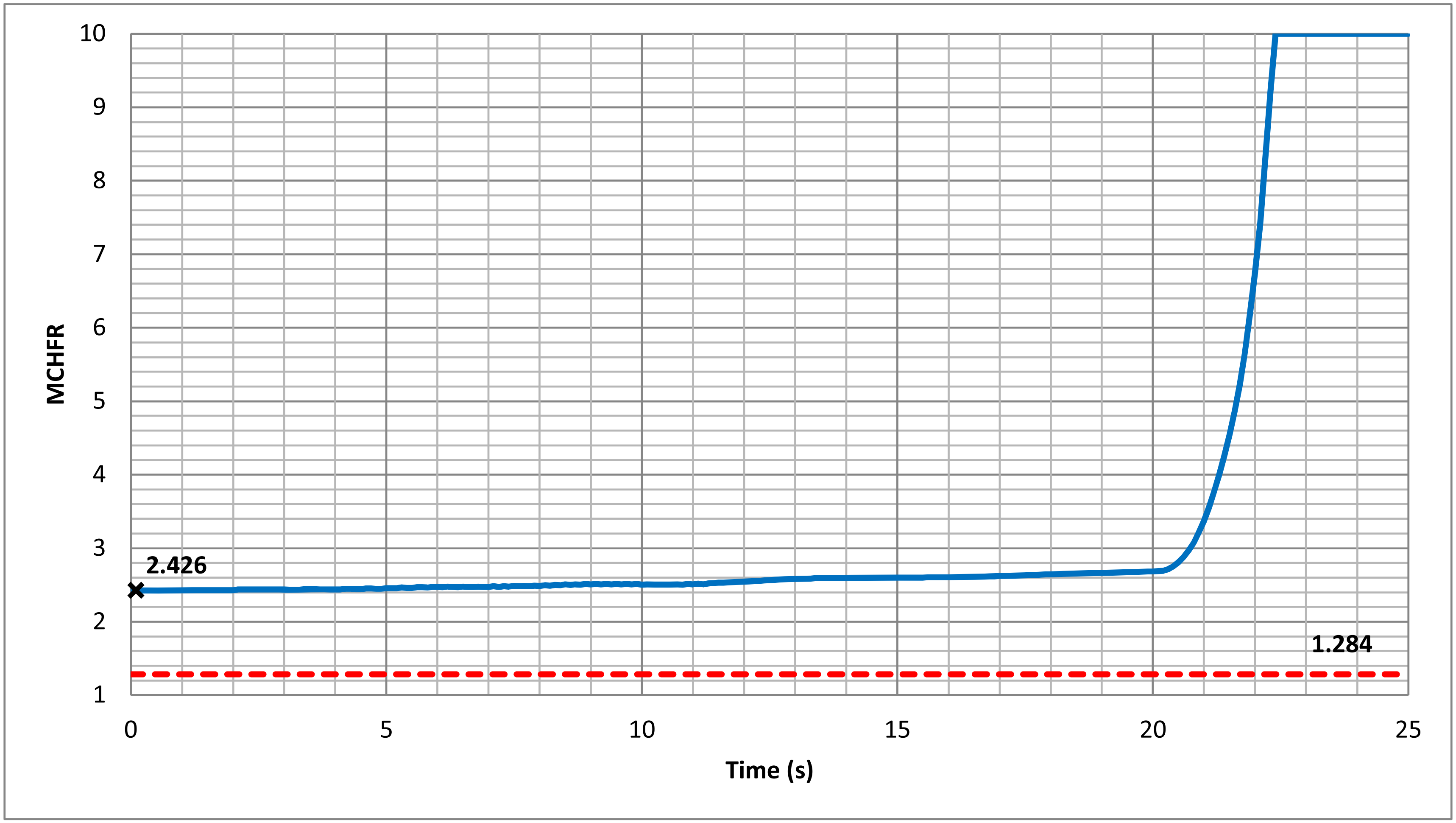


Figure 15.2-35: Reactor Pressure Vessel Lower Plenum Pressure - Peak RCS Pressure Case (15.2.8 Feedwater Line Break)

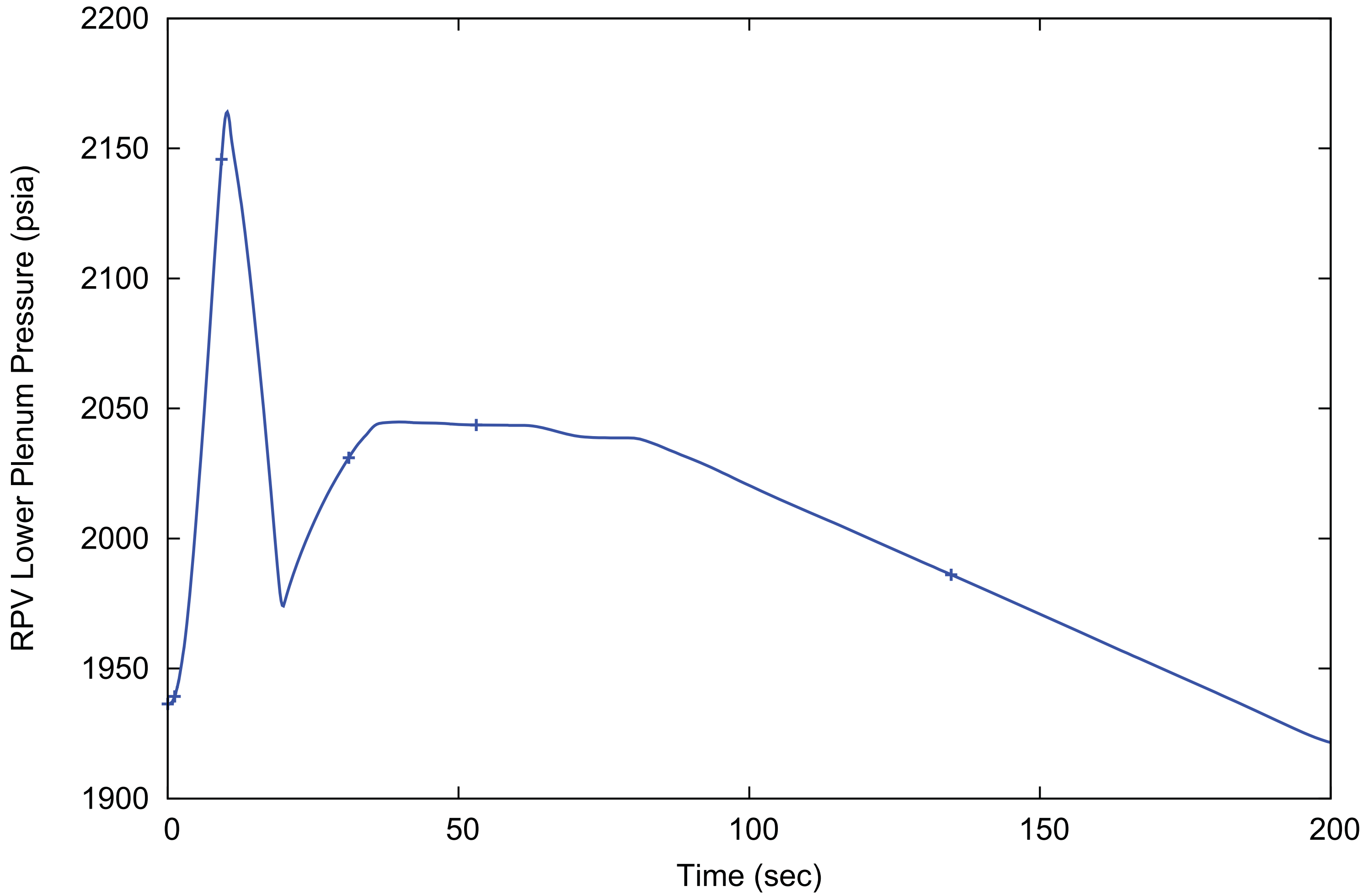


Figure 15.2-36: Reactor Power - Peak RCS Pressure Case (15.2.8 Feedwater Line Break)

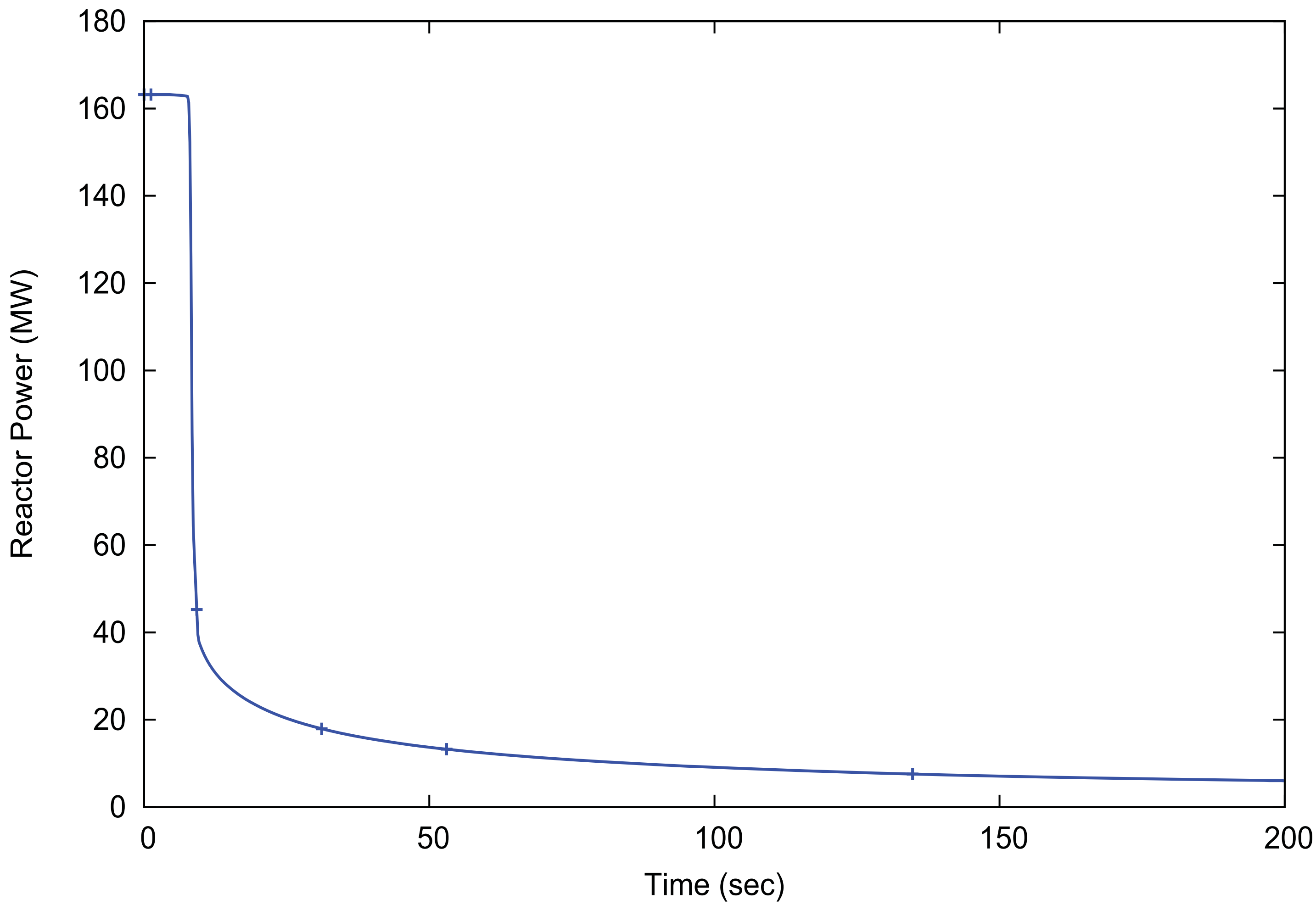


Figure 15.2-37: Steam Generator Train 1 Pressure - Peak RCS Pressure Case (15.2.8 Feedwater Line Break)

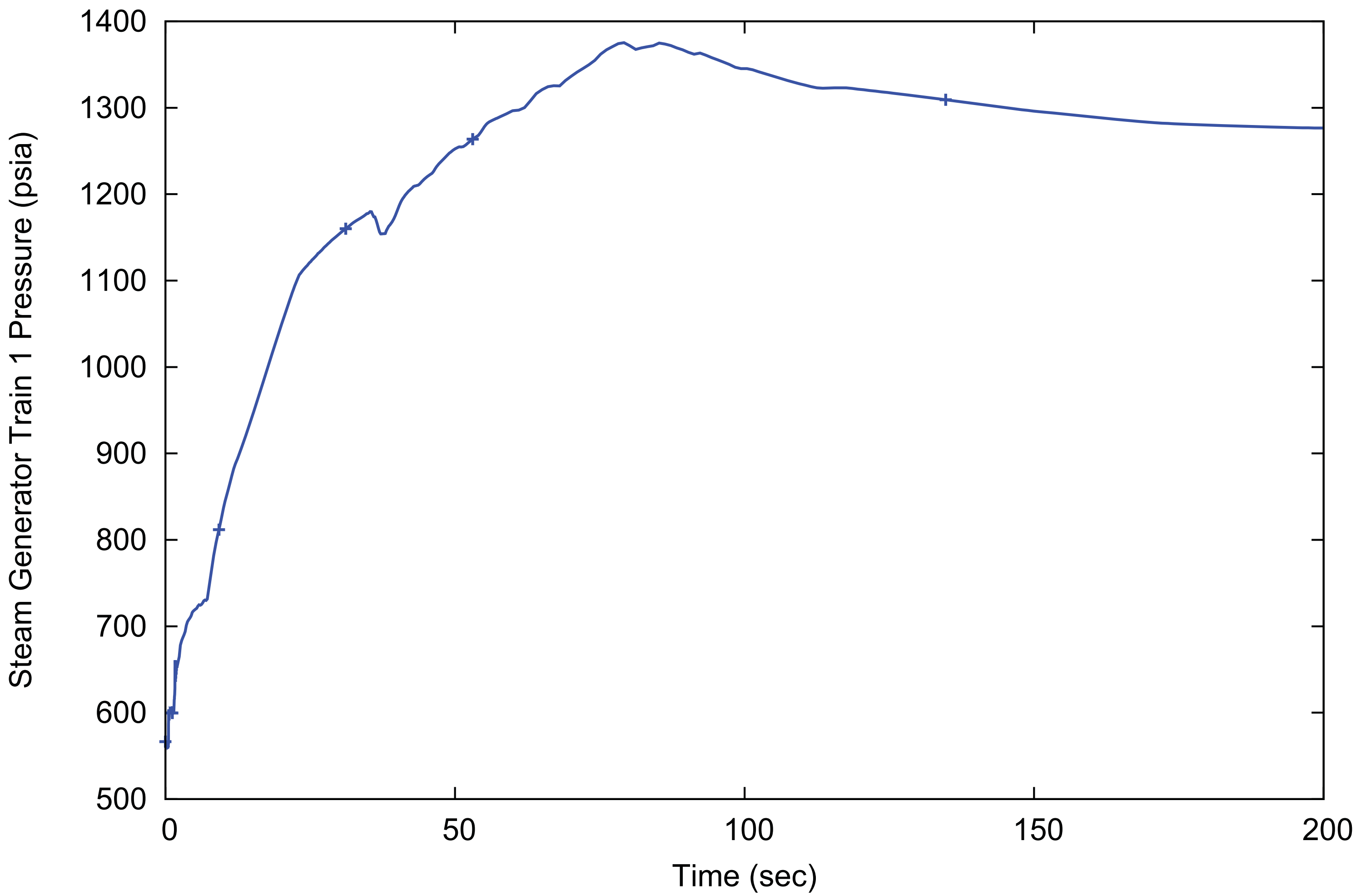


Figure 15.2-38: Steam Generator Train 2 Pressure - Peak RCS Pressure Case (15.2.8 Feedwater Line Break)

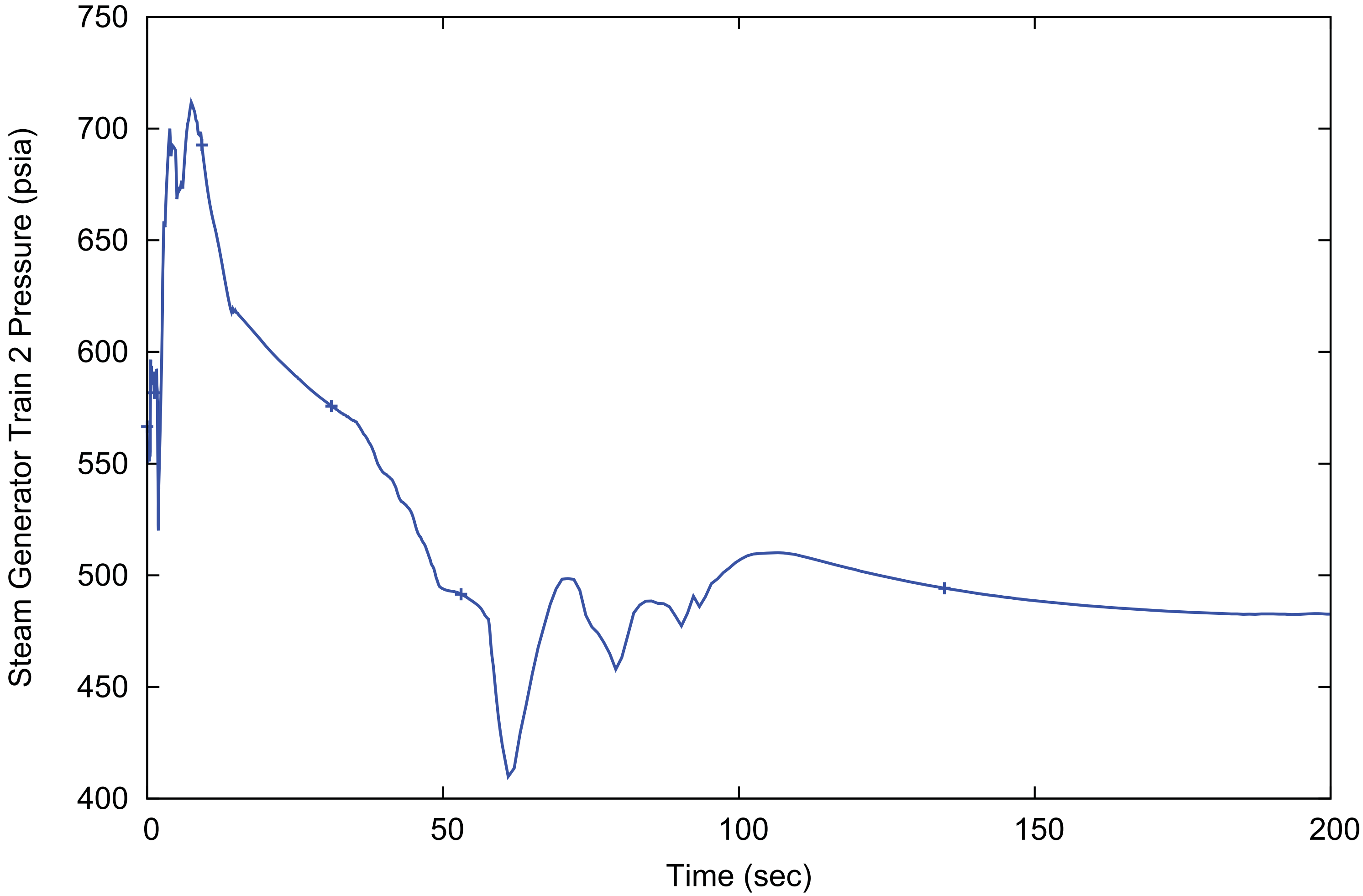


Figure 15.2-39: Core Outlet Temperature - Peak RCS Pressure Case (15.2.8 Feedwater Line Break)

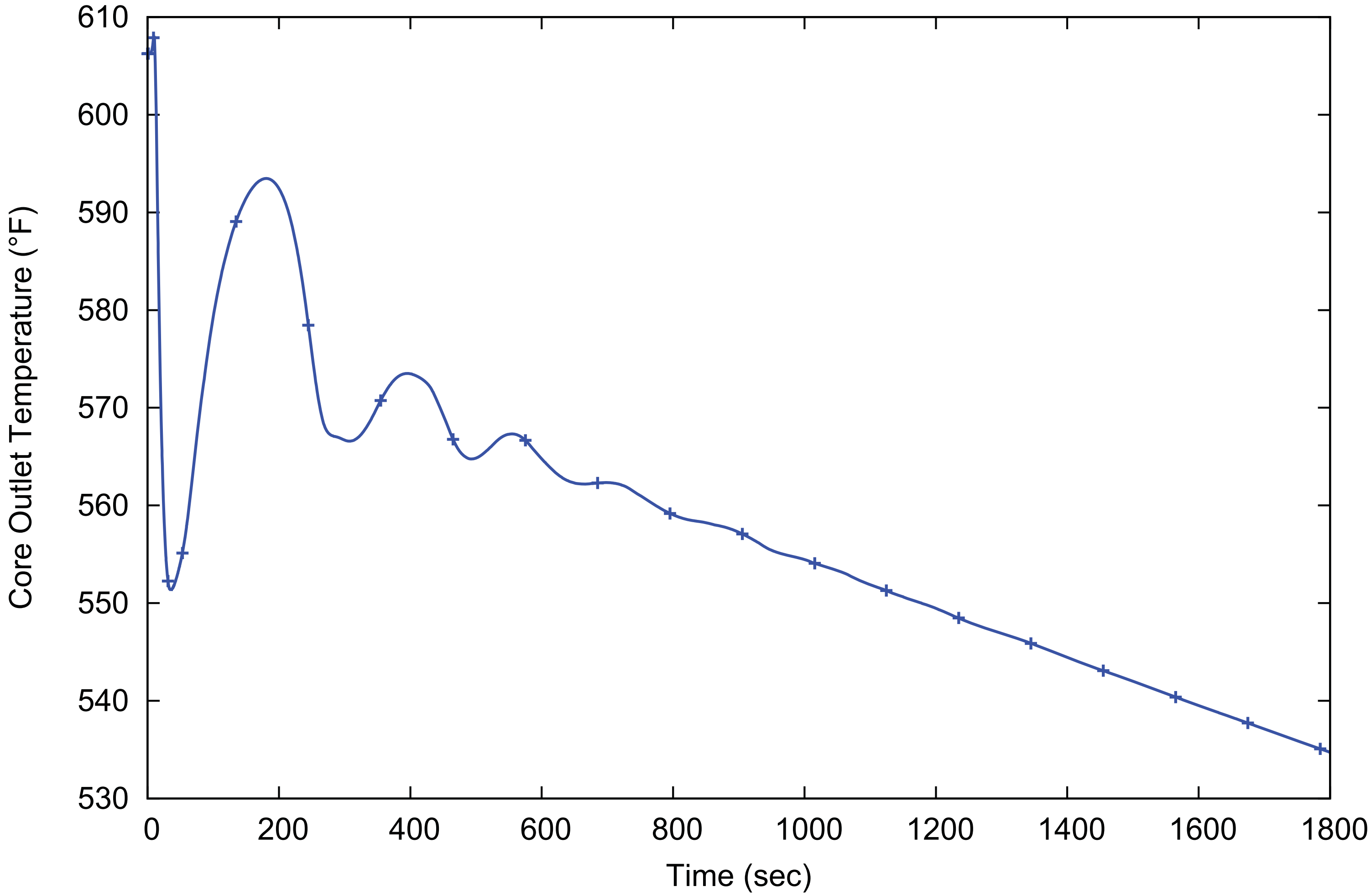


Figure 15.2-40: Reactor Coolant System Flowrate - Peak RCS Pressure Case (15.2.8 Feedwater Line Break)

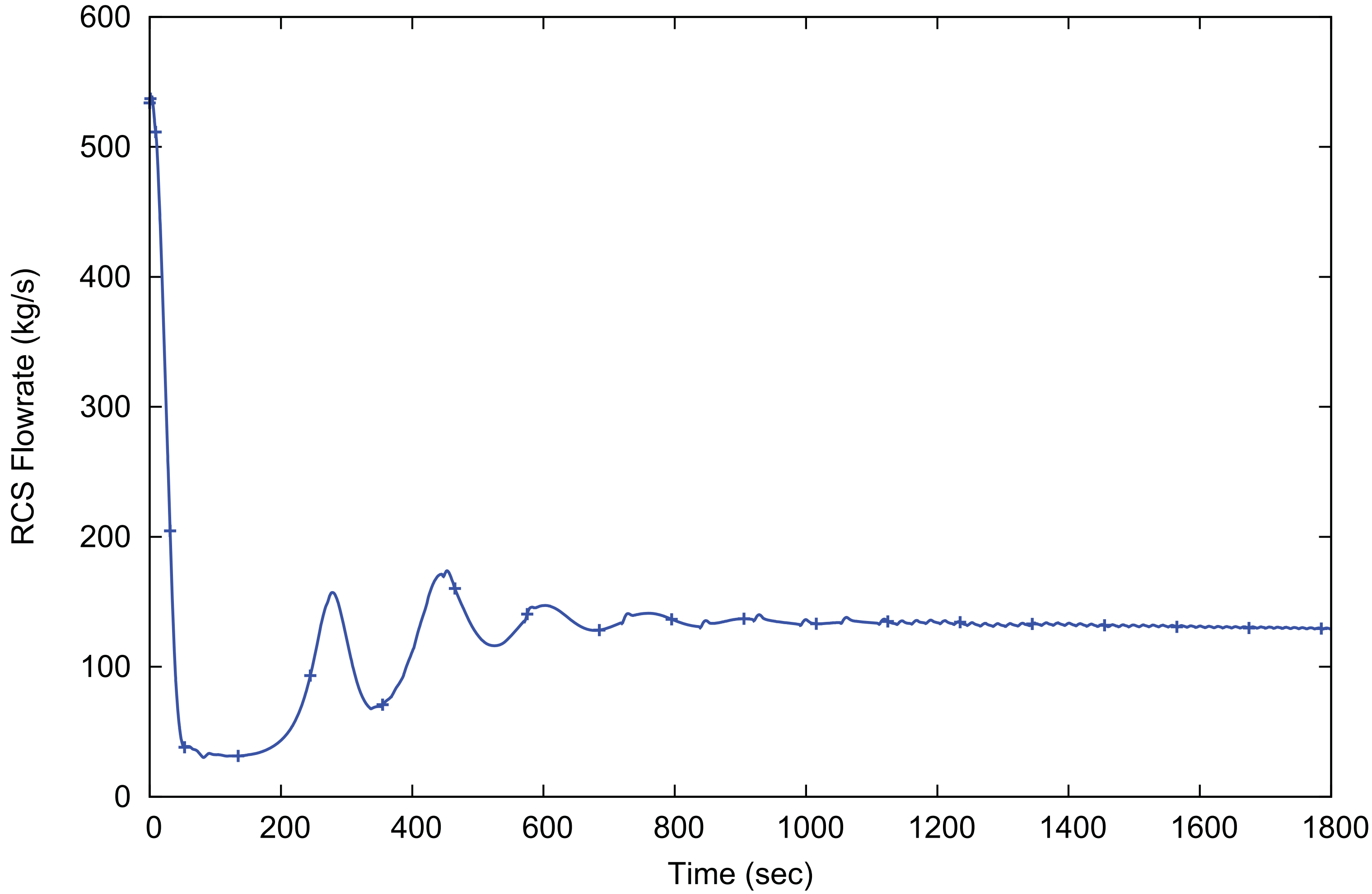


Figure 15.2-41: Integrated Break Flow - Peak RCS Pressure Case (15.2.8 Feedwater Line Break)

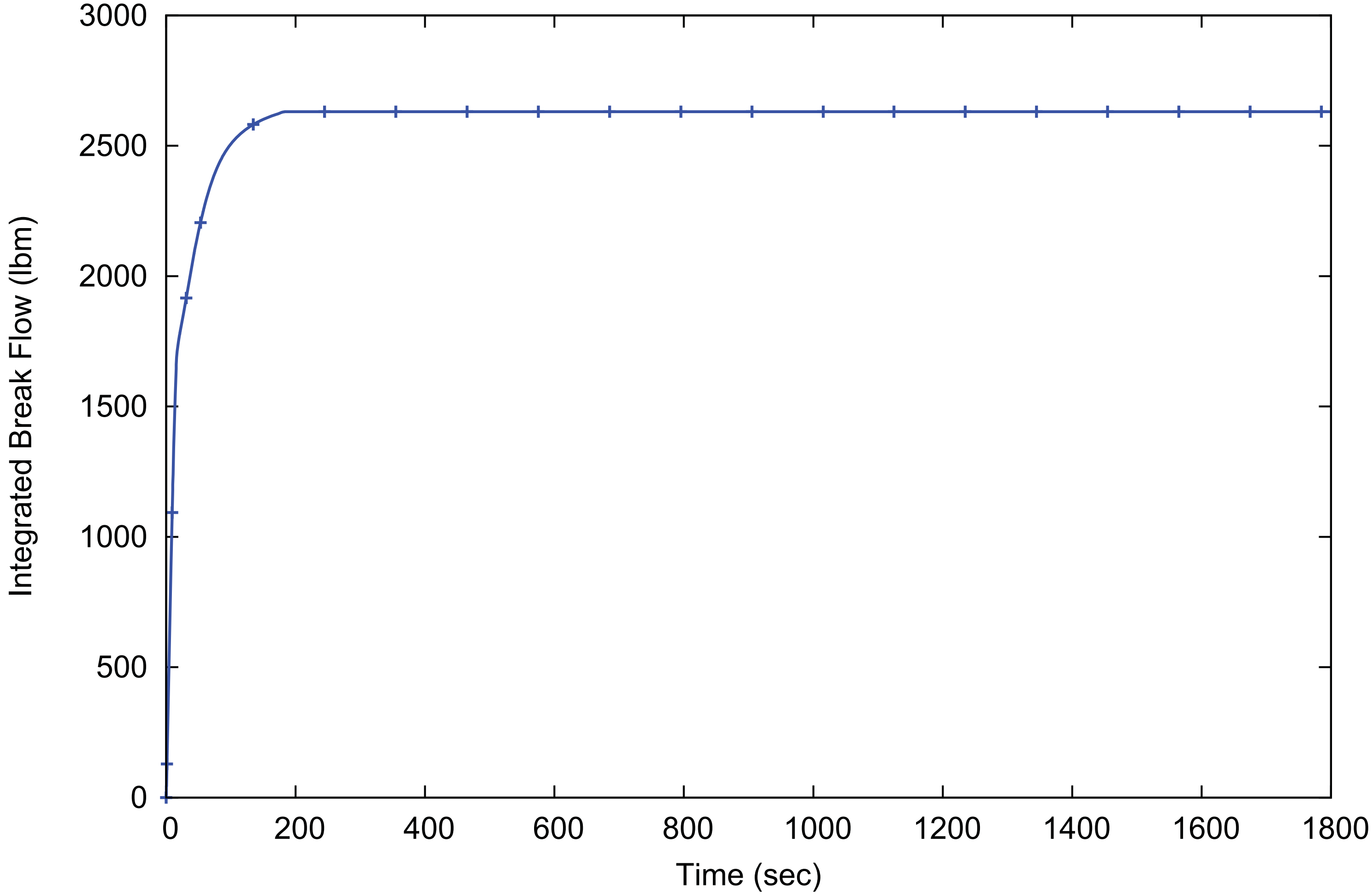


Figure 15.2-42: Steam Generator Train 1 Pressure - Peak SG Pressure Case (15.2.8 Feedwater Line Break)

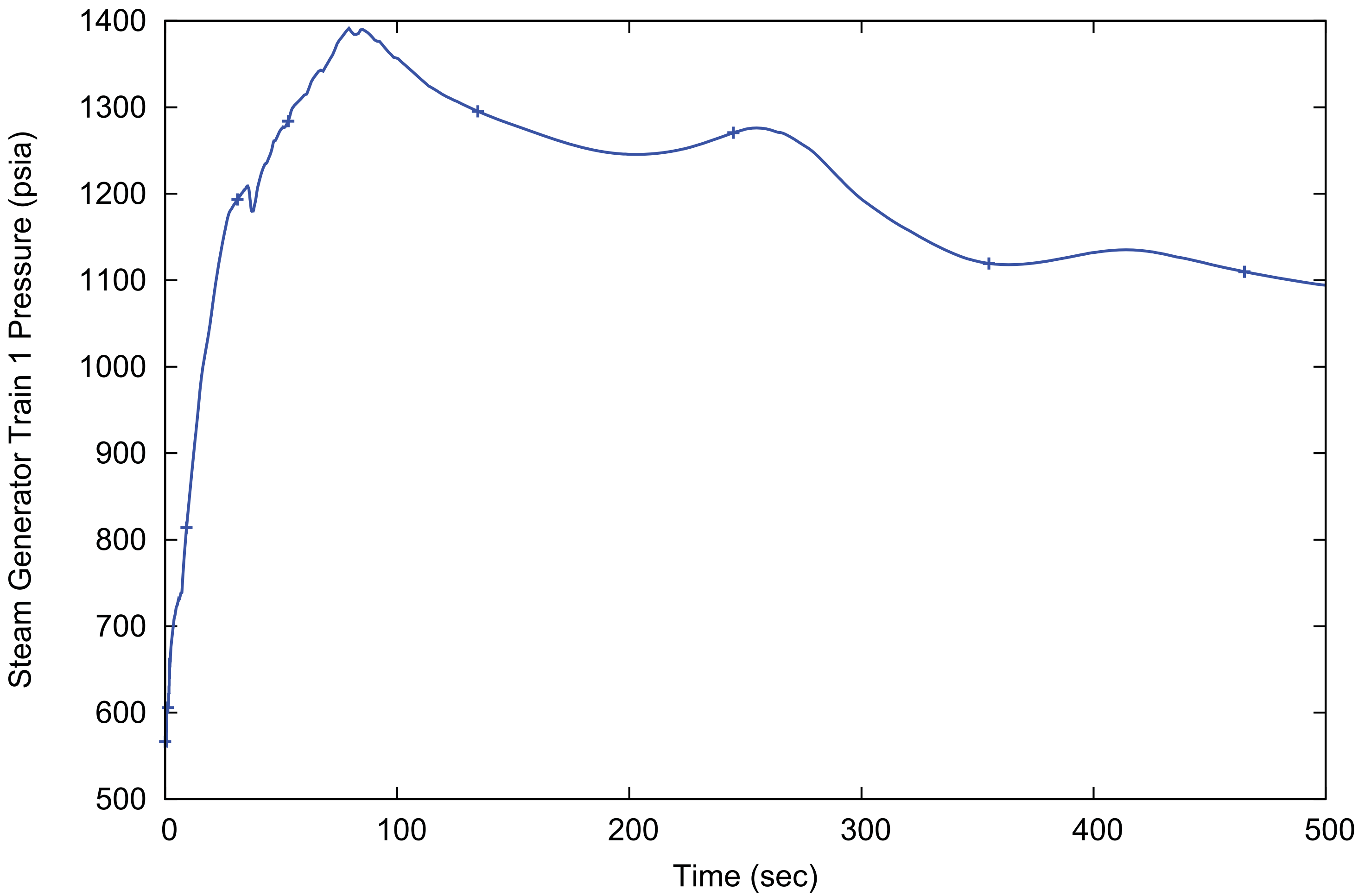


Figure 15.2-43: Steam Generator Train 2 Pressure - Peak SG Pressure Case (15.2.8 Feedwater Line Break)

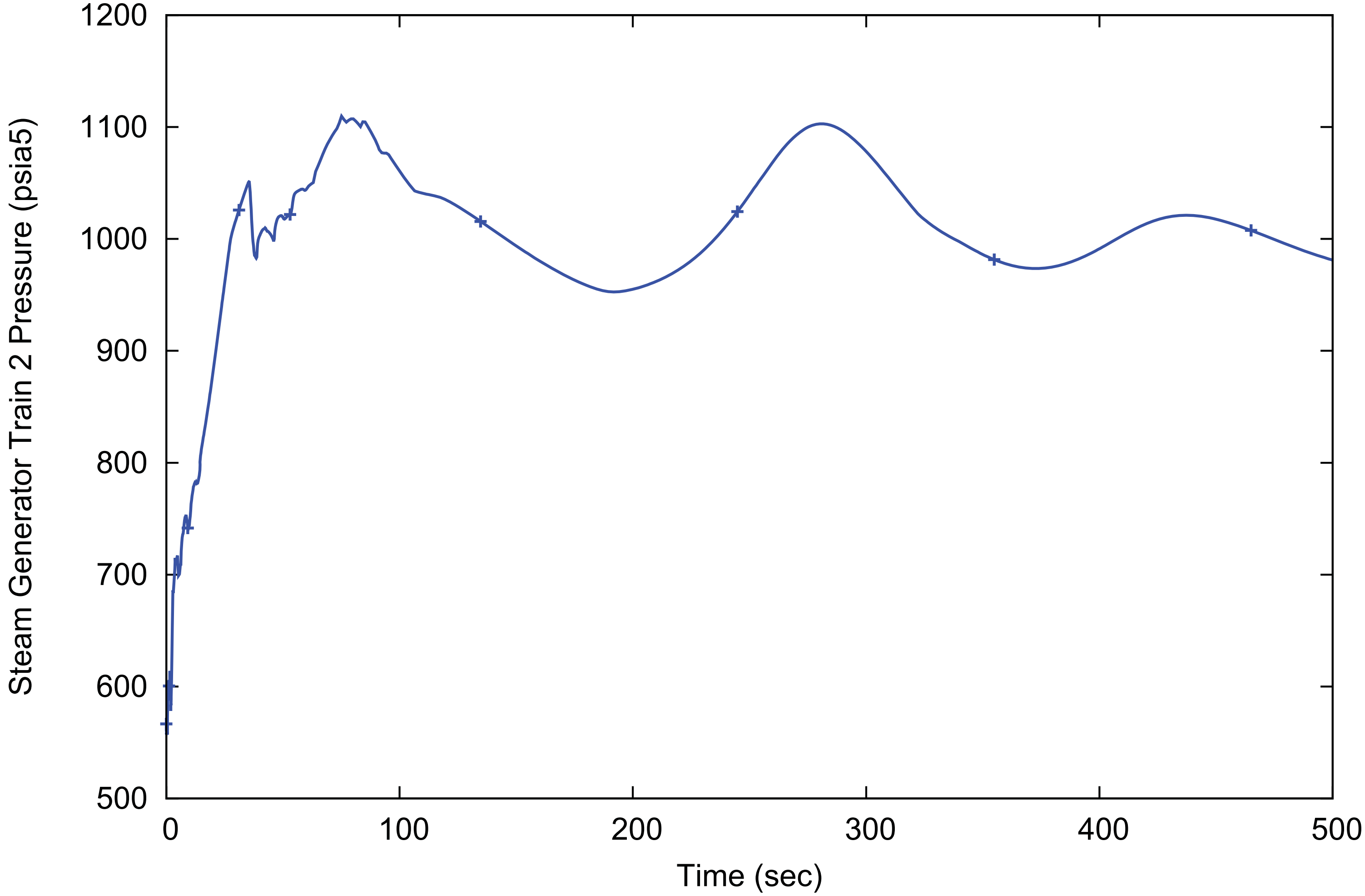


Figure 15.2-44: Hot Channel Node MCHFR - Limiting MCHFR Case (15.2.8 Feedwater Line Break)

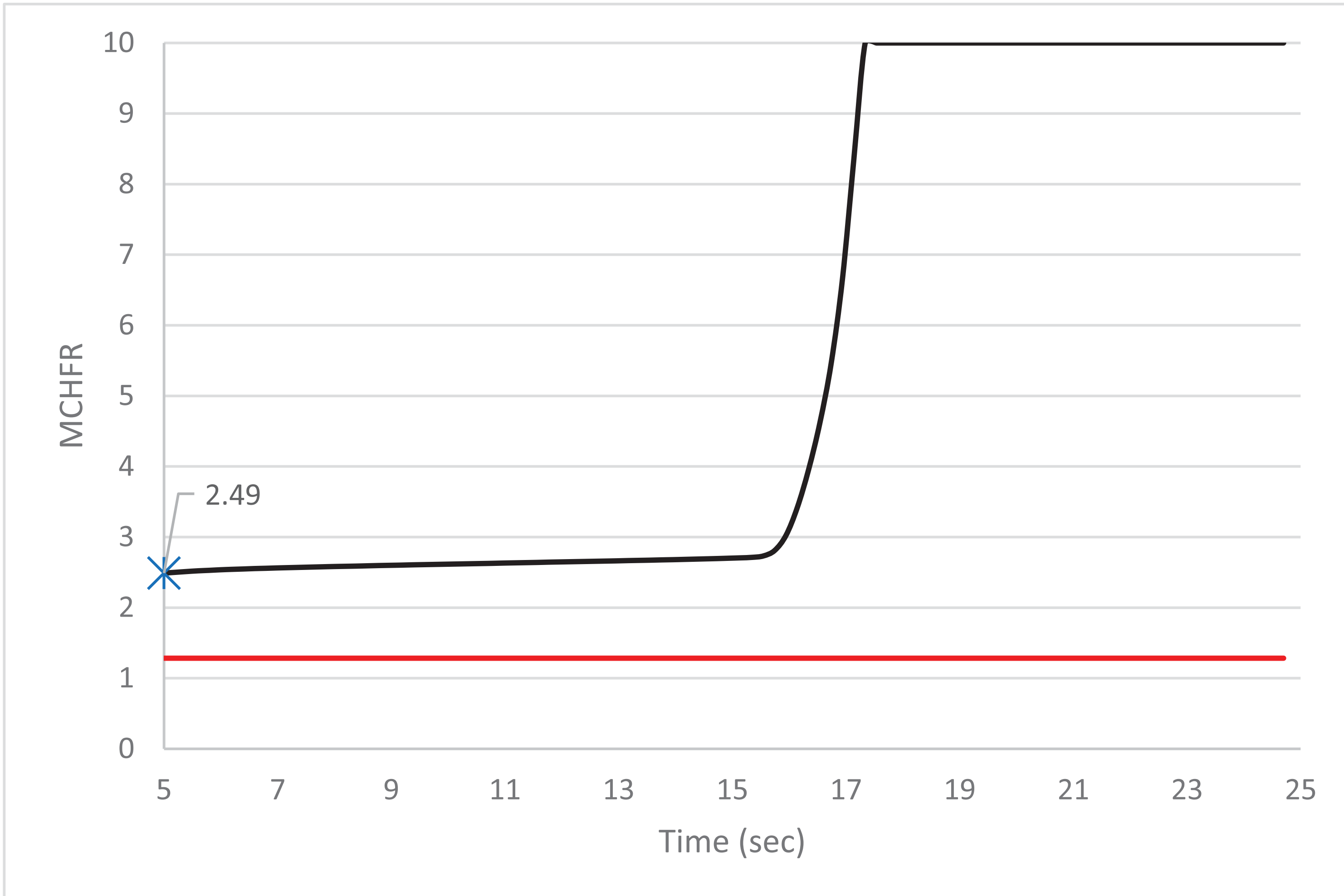


Figure 15.2-45: DHRS Heat Removal - Limiting DHRS Case (15.2.8 Feedwater Line Break)

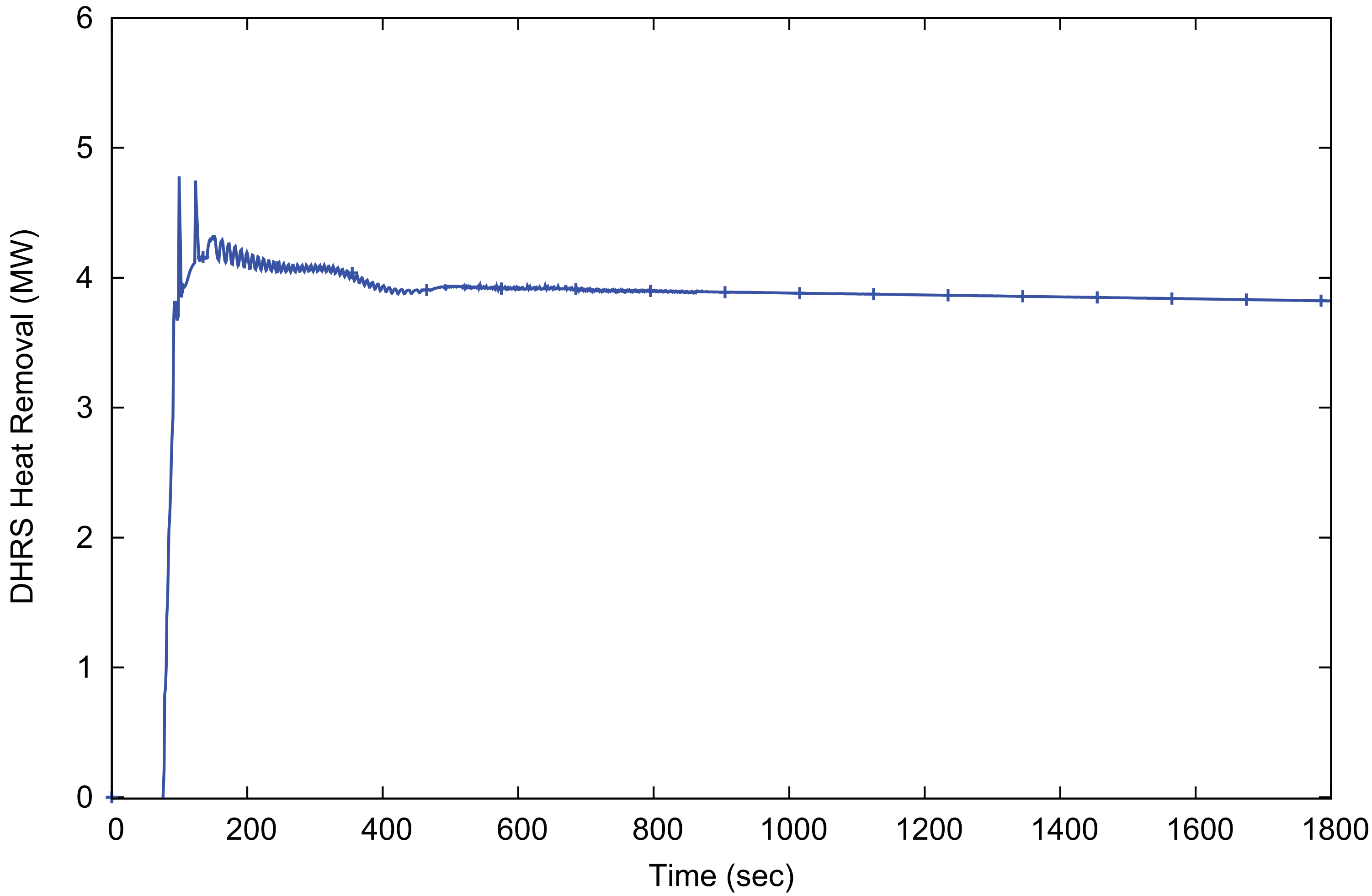


Figure 15.2-46: Steam Generator Train 1 Pressure - Limiting DHRS Case (15.2.8 Feedwater Line Break)

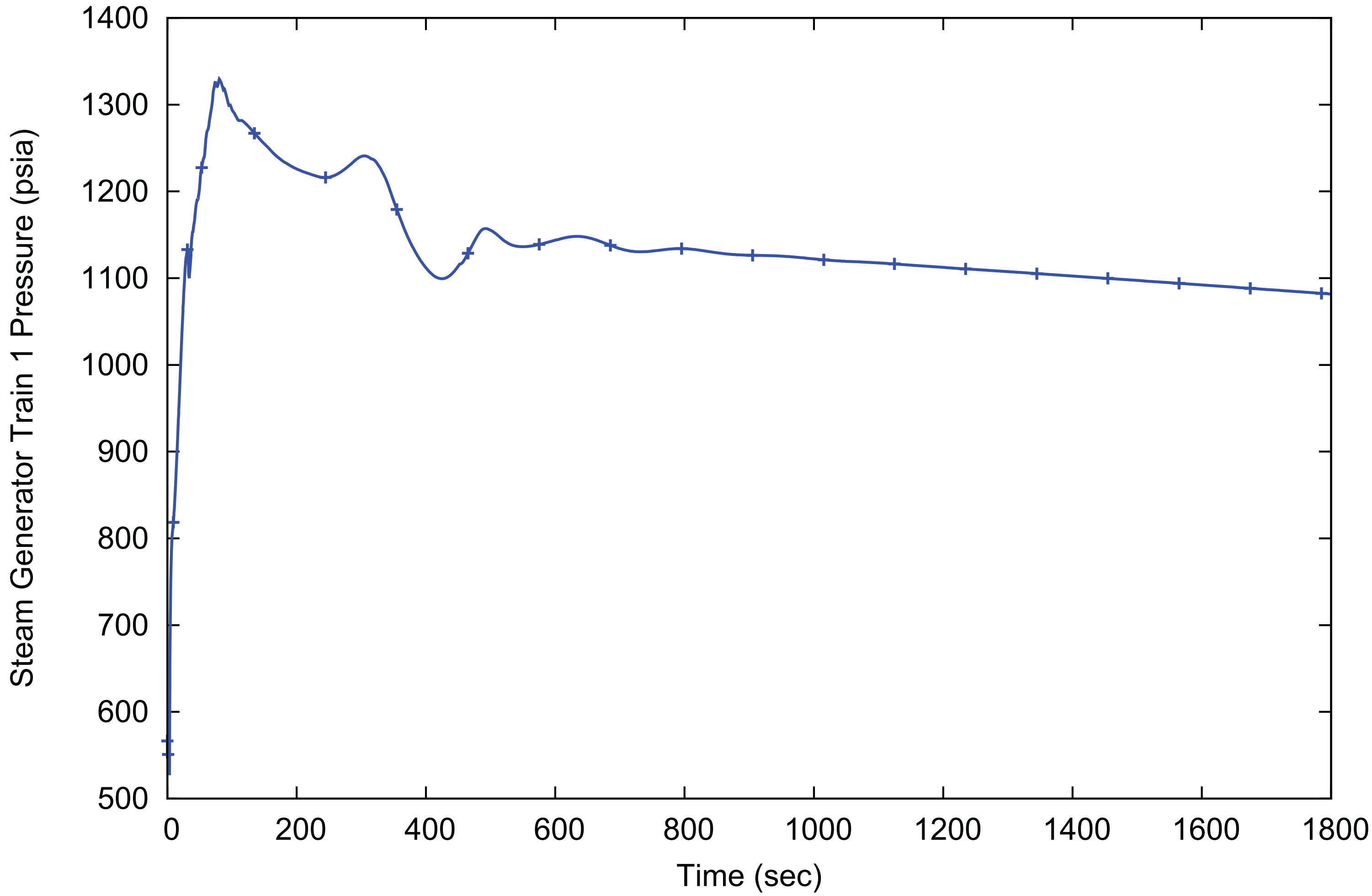


Figure 15.2-47: Steam Generator Train 2 Pressure - Limiting DHRS Case (15.2.8 Feedwater Line Break)

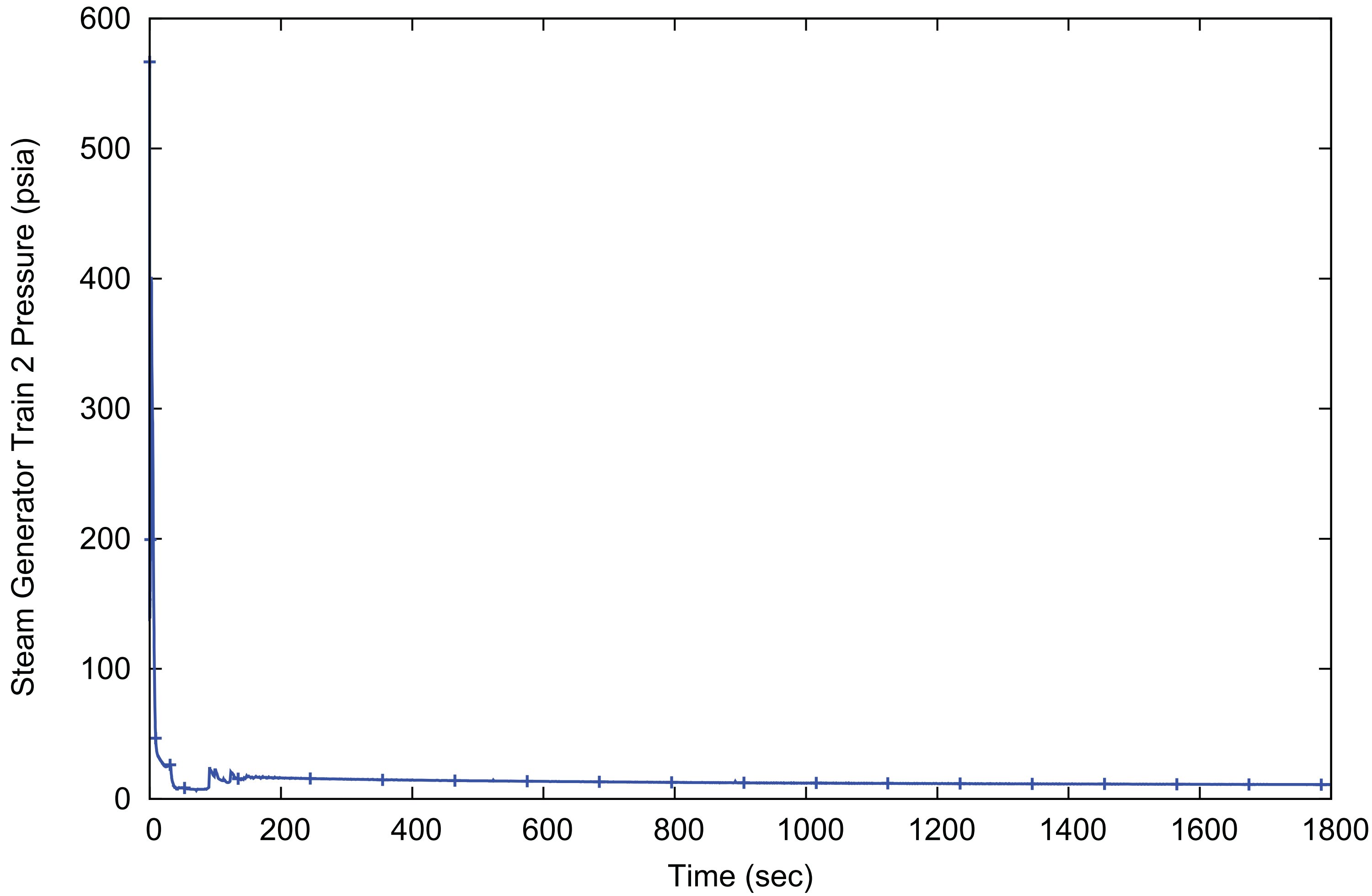


Figure 15.2-48: Reactor Coolant System Flowrate - Limiting DHRS Case (15.2.8 Feedwater Line Break)

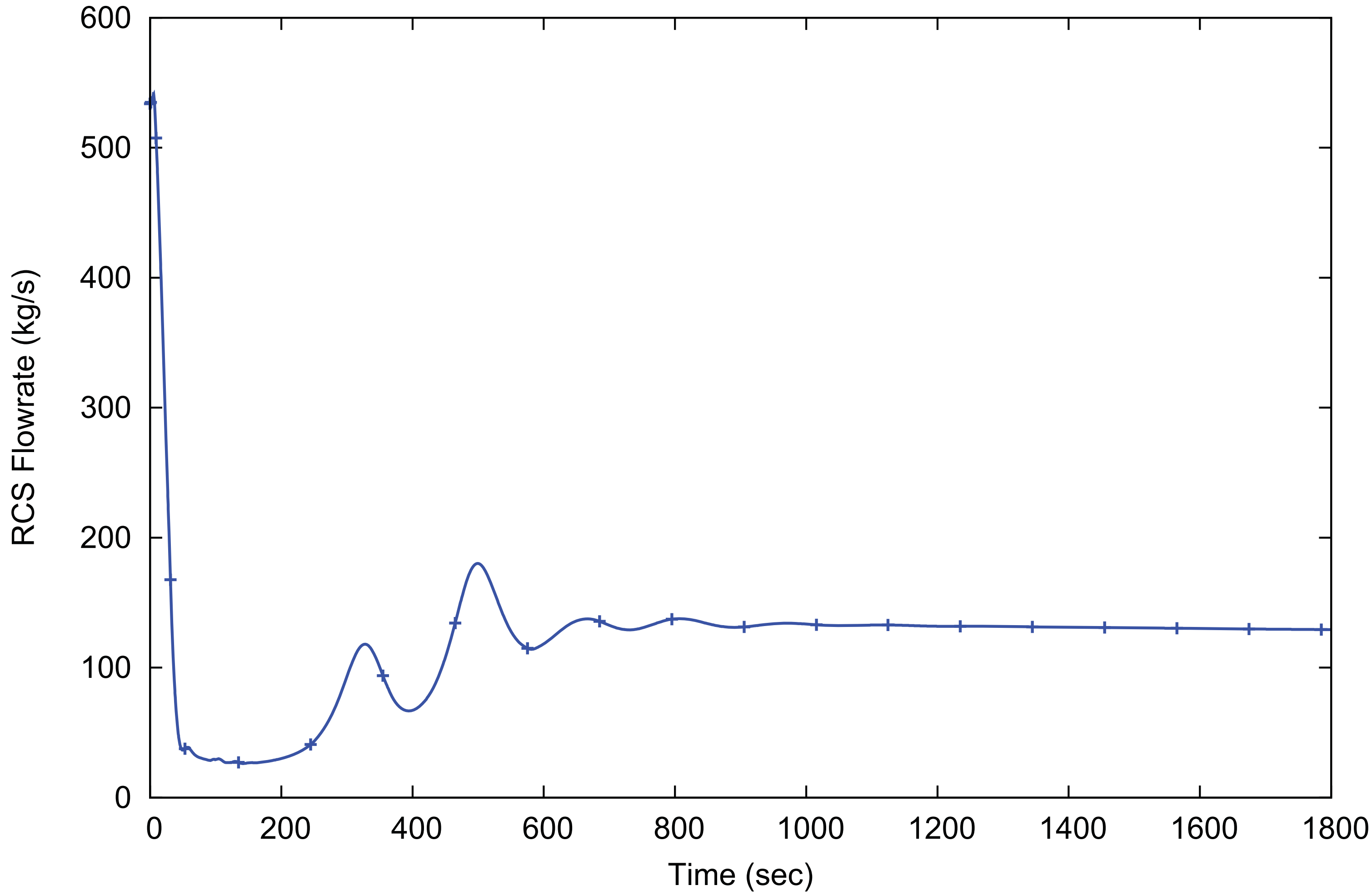


Figure 15.2-49: Reactor Pressure Vessel Lower Plenum Pressure - Peak RCS Pressure Case (15.2.9 Inadvertent Operation of DHRS)

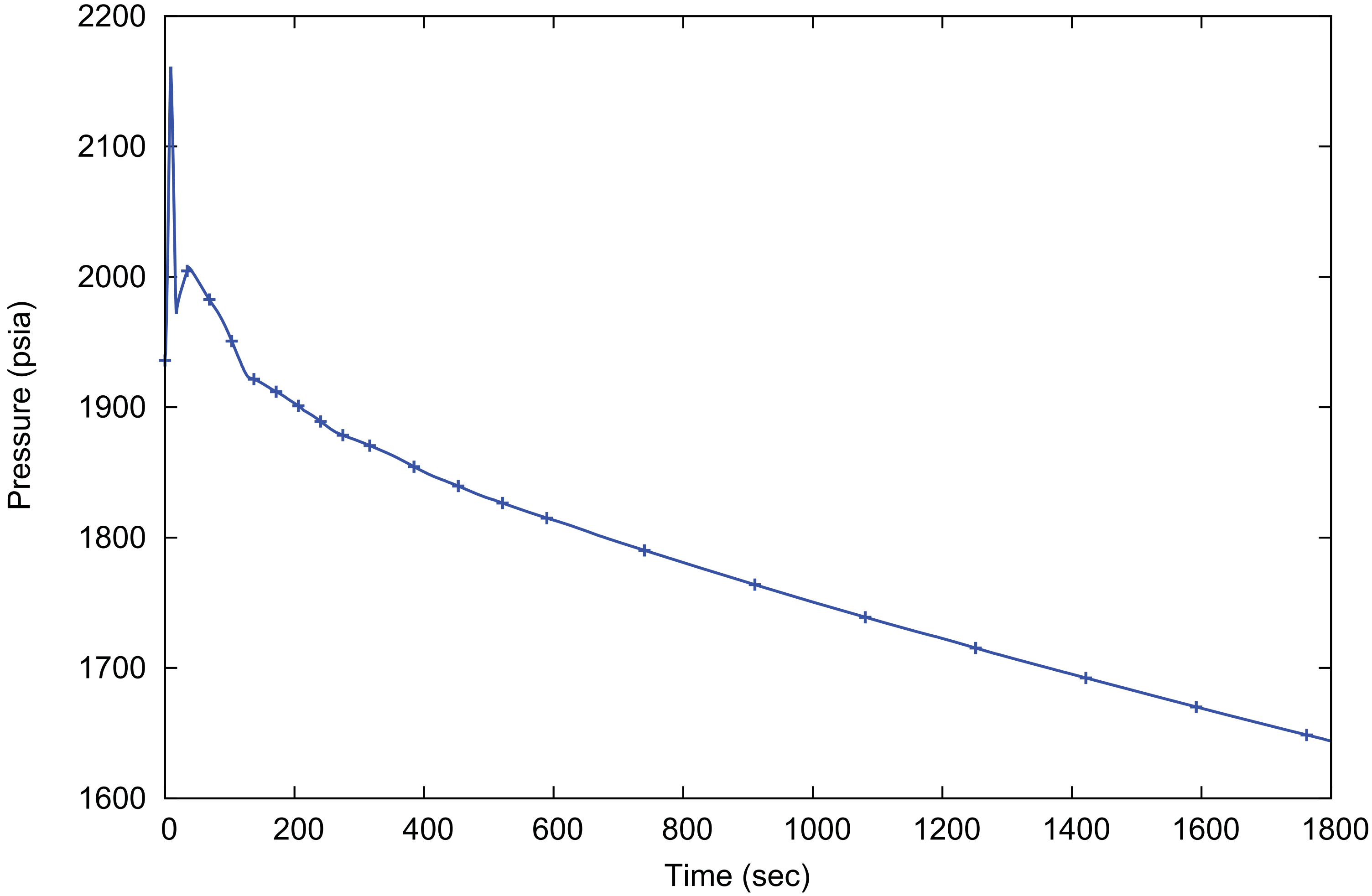


Figure 15.2-50: Core Inlet Temperature - Peak RCS Pressure Case (15.2.9 Inadvertent Operation of DHRS)

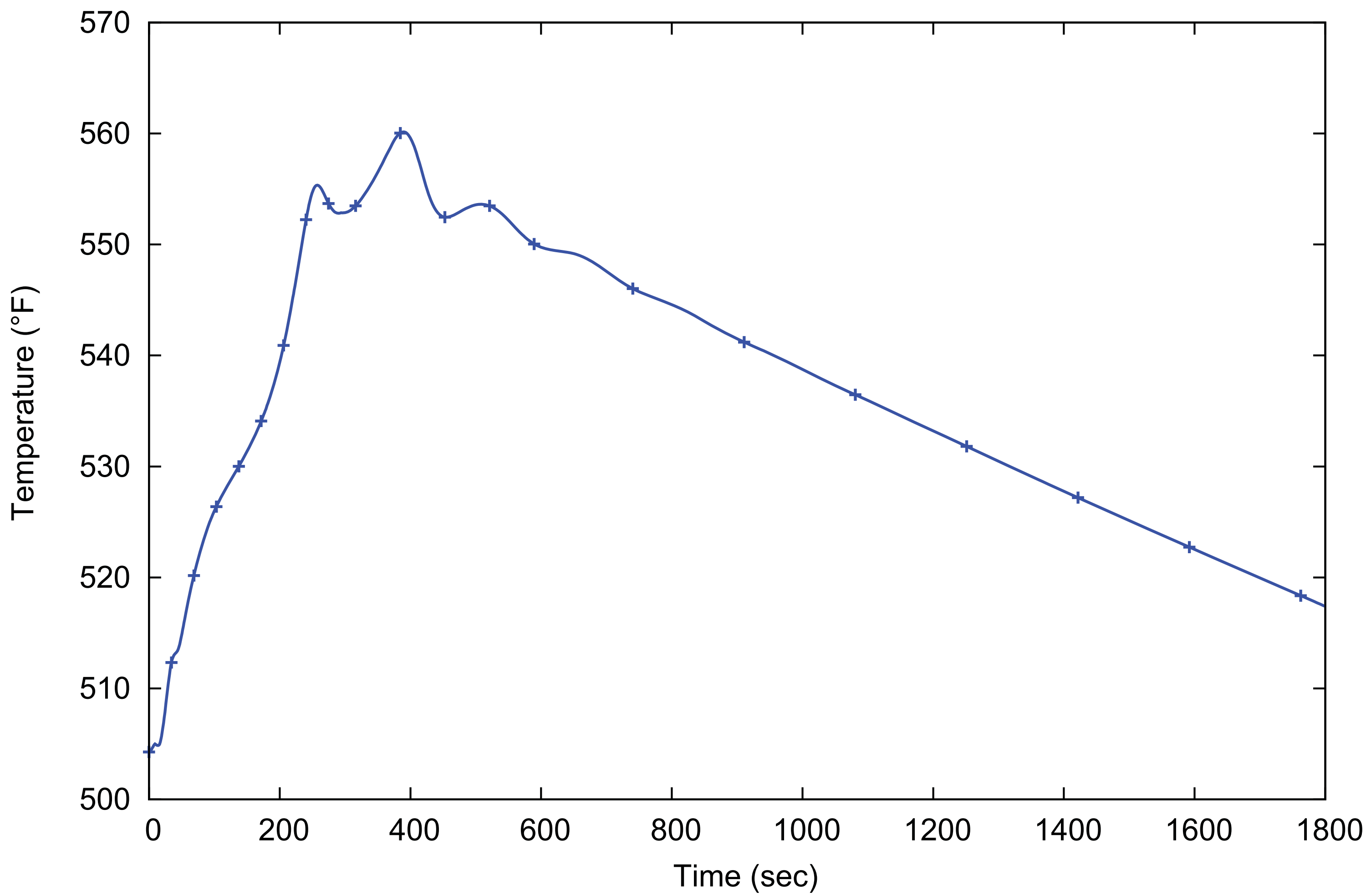


Figure 15.2-51: Core Outlet Temperature - Peak RCS Pressure Case (15.2.9 Inadvertent Operation of DHRS)

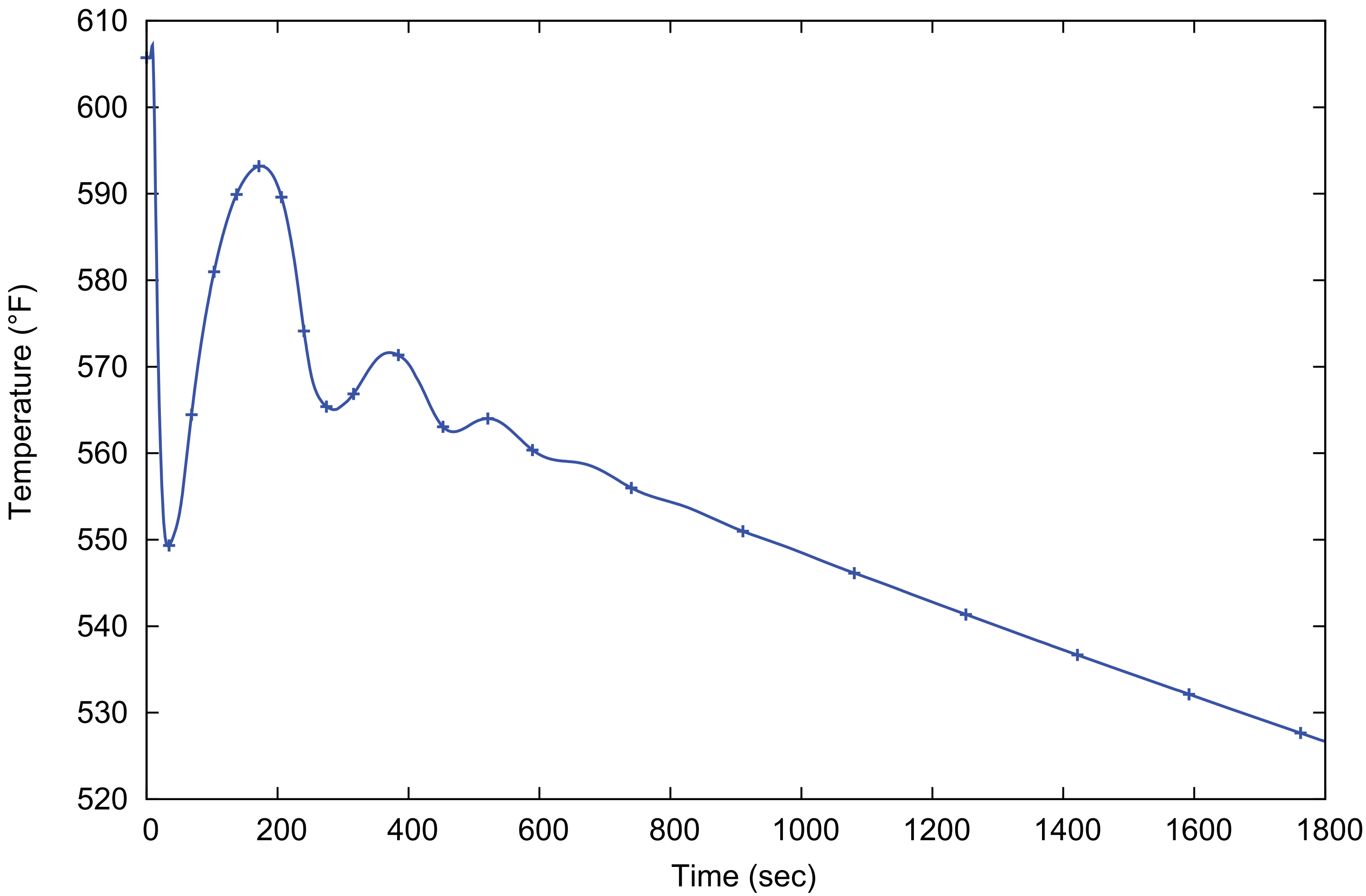


Figure 15.2-52: Reactor Coolant System Flow - Peak RCS Pressure Case (15.2.9 Inadvertent Operation of DHRS)

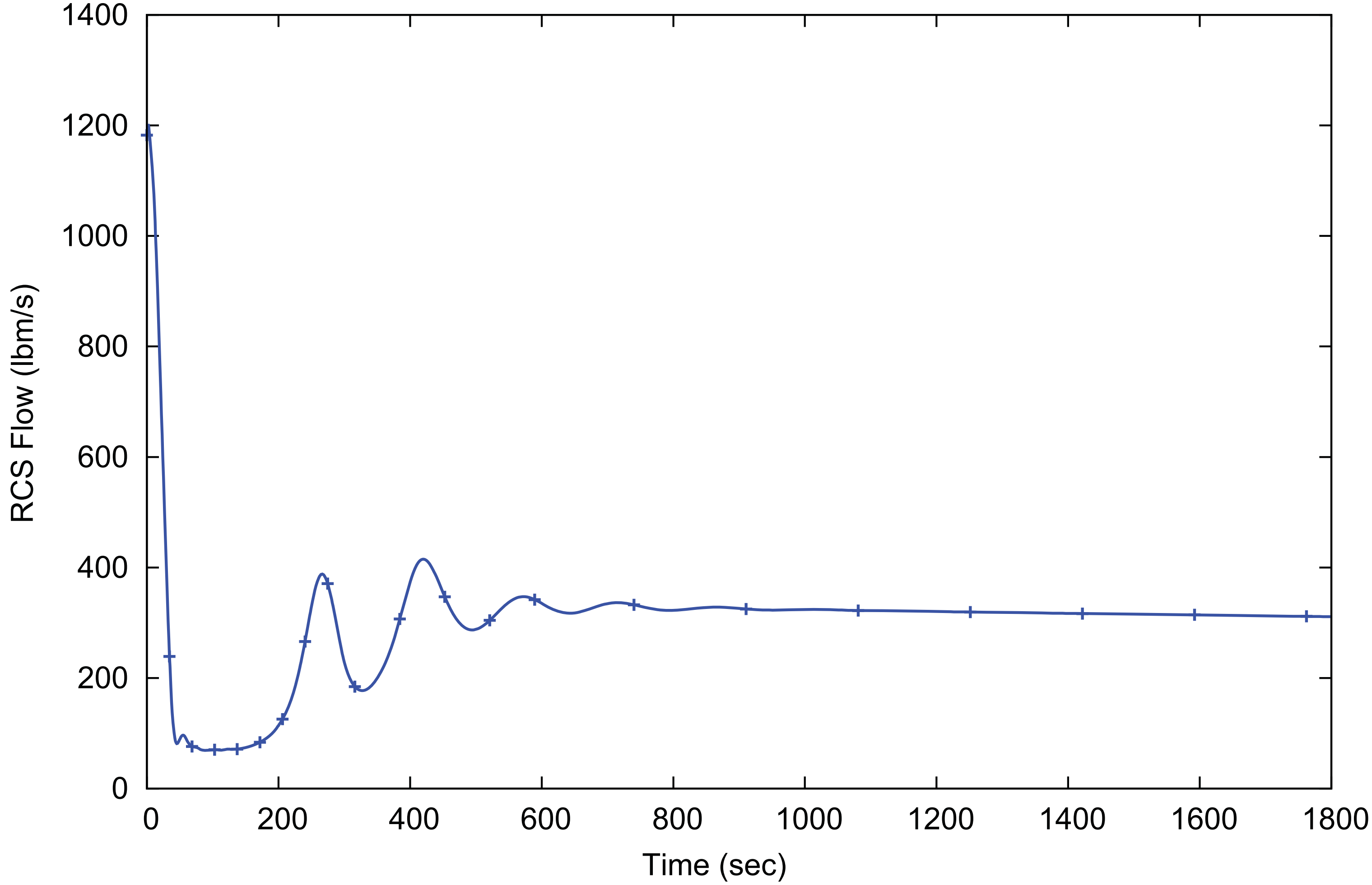


Figure 15.2-53: Total Reactivity - Peak RCS Pressure Case (15.2.9 Inadvertent Operation of DHRS)

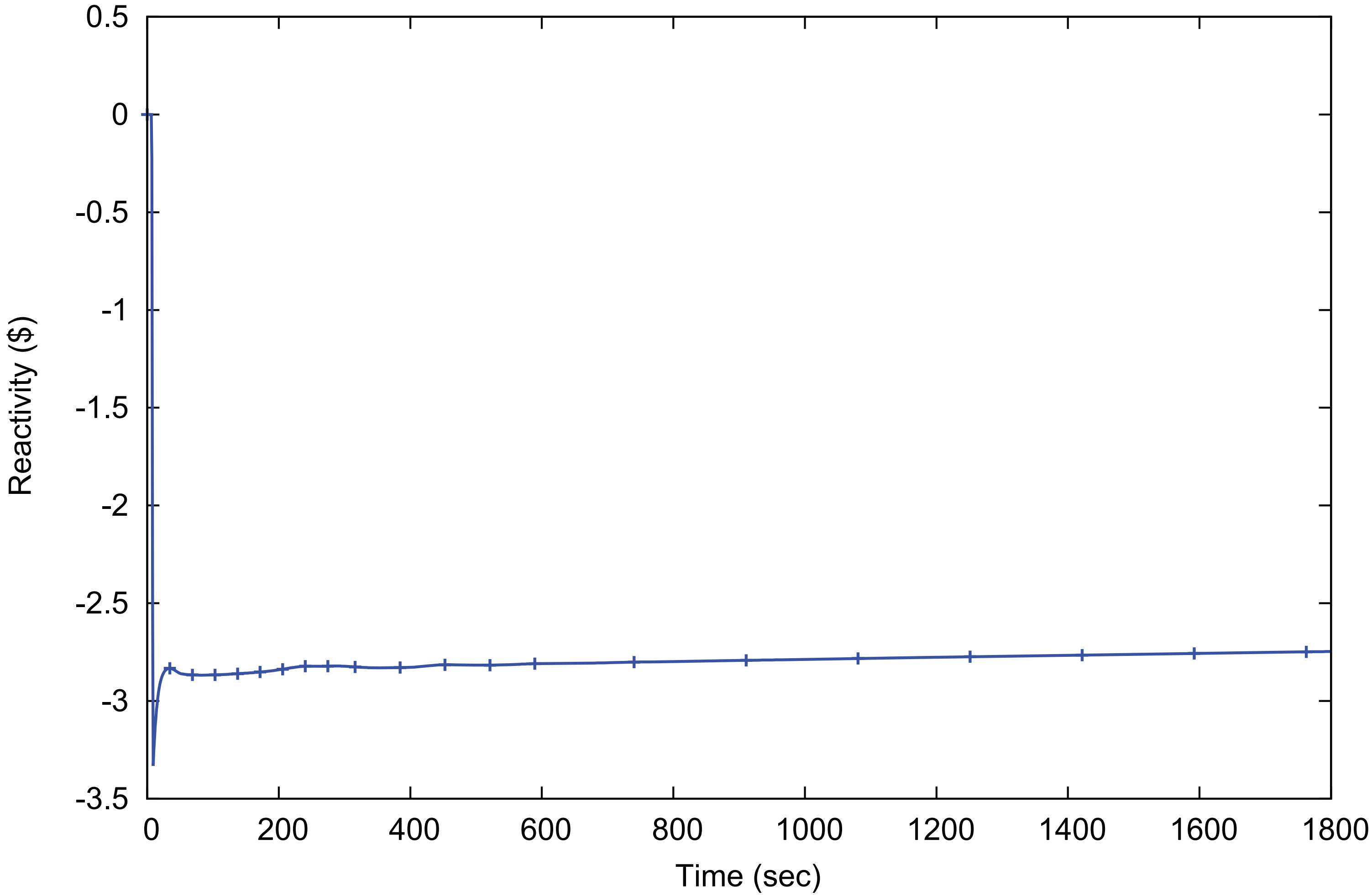


Figure 15.2-54: Steam Plenum Pressure (Typical) - Peak SG Pressure Case (15.2.9 Inadvertent Operation of DHRS)

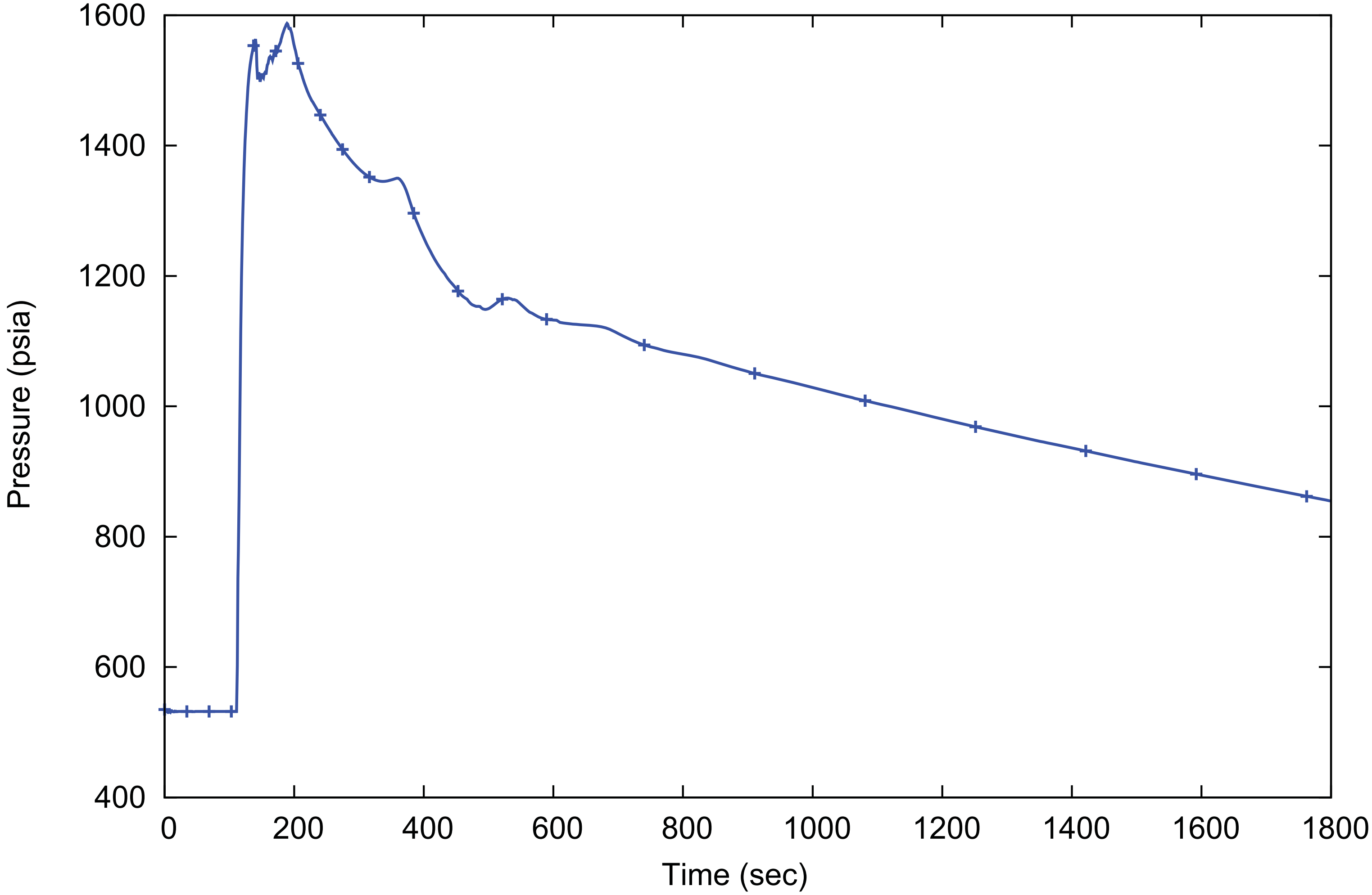
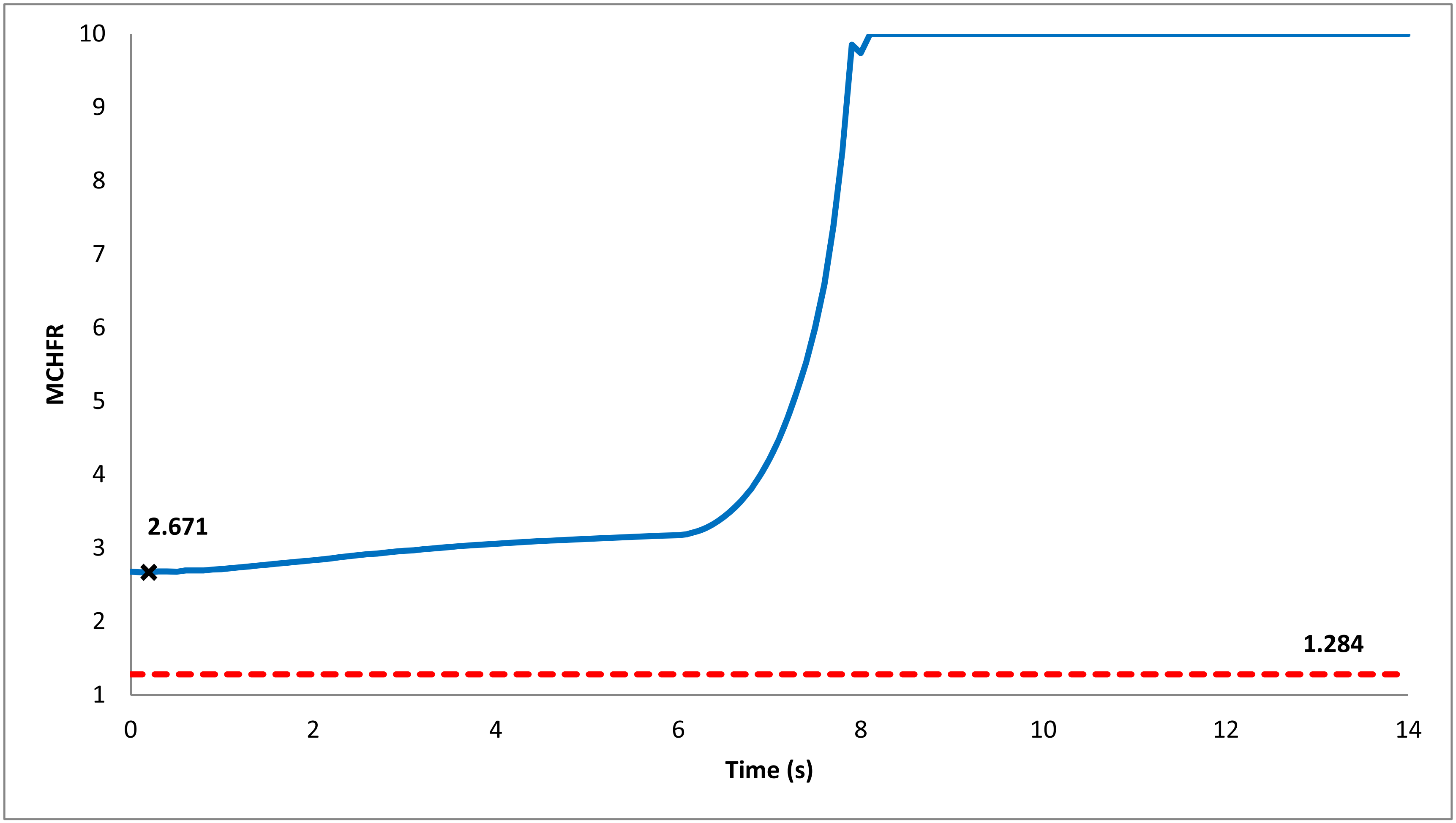


Figure 15.2-55: Minimum Critical Heat Flux Ratio - Limiting MCHFR Case (15.2.9 Inadvertent Operation of DHRS)



15.3 Decrease in RCS Flow Rate

These events do not apply to the NPM design and no new unique NPM events are identified for this event type.

15.4 Reactivity and Power Distribution Anomalies

Section 15.4 covers the various events that are caused by reactivity and power distribution anomalies. The sections for these events will cover the following:

- Section 15.4.1 - uncontrolled control rod assembly (CRA) withdrawal from a subcritical or low power startup condition.
- Section 15.4.2 - uncontrolled CRA withdrawal at power.
- Section 15.4.3 - control rod misoperation (CRM) (system malfunction or operator error).
- Section 15.4.4 - startup of an inactive loop or recirculation loop at an incorrect temperature. Although this event is not applicable to NuScale, this event will be addressed in Section 15.4.4 for consistency with the Standard Review Plan (SRP).
- Section 15.4.5 - flow controller malfunction causing an increase in boiling water reactor core flow rate. Although this event is not applicable to NuScale, this event will be addressed in Section 15.4.5 for consistency with the SRP.
- Section 15.4.6 - inadvertent decrease in boron concentration in the reactor coolant system (RCS).
- Section 15.4.7 - inadvertent loading and operation of a fuel assembly in an improper position.
- Section 15.4.8 - spectrum of rod ejection accidents (REAs).
- Section 15.4.9 - spectrum of rod drop accidents. Although this event is not applicable to NuScale, this event will be addressed in Section 15.4.9 for consistency with the SRP.

15.4.1 Uncontrolled Control Rod Assembly Withdrawal from a Subcritical or Low Power or Startup Condition

15.4.1.1 Identification of Causes and Accident Description

An uncontrolled CRA withdrawal from a subcritical or low power or startup condition event could result in a rapid insertion of reactivity into the reactor core. There is an increase in reactor power due to the unexpected addition of reactivity as the CRA is withdrawn from the core. The core power increases at a faster rate than heat can be removed, resulting in an increase in RCS temperature and a decrease in minimum critical heat flux ratio (MCHFR).

An uncontrolled CRA withdrawal from a subcritical or low power or startup condition is expected to occur one or more times in the life of the reactor, and it is classified as an anticipated operational occurrence (AOO). The categorization of the NuScale design basis events are provided in Table 15.0-1.

15.4.1.2 Sequence of Events and Systems Operation

The sequence of events for an uncontrolled CRA withdrawal from a subcritical or low power or startup condition is provided in Table 15.4-1 for the limiting MCHFR case. RCS pressure and Secondary pressure are not acceptance criteria for this event. RCS and Secondary pressure are bounded by other AOO events.

Unless specified below, the analysis of an uncontrolled CRA withdrawal from a subcritical or low power or startup condition event assumes the plant control systems and engineered safety features perform as designed, with allowances for instrument inaccuracy. No operator action is credited to mitigate the effects of the CRA withdrawal.

The rod control function of the control rod drive system (CRDS) provides reactivity control to compensate for rapid, short term variations in the reactivity of the core. The rod control function is also used to maintain the measured RCS temperature at or near the programmed average coolant temperature. The CRDS rod control operational modes include manual mode, automatic mode, and insertion-only automatic mode. The CRDS rod control function could be in manual mode during startup conditions. An operator error or malfunction in the CRDS would have to occur to initiate a uncontrolled CRA withdrawal from a subcritical or low power or startup condition.

The maximum allowed withdrawal rate of a CRA is 15 in./min, with a step size no greater than three-eighths inch. A spectrum of reactivity insertion rates that includes the maximum rate and bounds possible boron dilution scenarios is included in the analysis.

The module protection system (MPS) is credited to protect the plant in the event of an uncontrolled CRA withdrawal from a subcritical or low power or startup condition. The following MPS signals provide the plant with protection in the event of an uncontrolled CRA withdrawal from a subcritical or low power or startup condition:

- high power (at 25 percent of full power for startup conditions)
- source range (SR) and intermediate range (IR) power rate
- high SR countrate
- high pressure

RCS pressure and inventory controls are disabled to ensure a maximum pressure and pressurizer level, which is conservative. The feedwater and steam flows are held constant for this analysis to prevent the secondary side from mitigating the primary side heat increase.

The potential loss of power scenarios for the NuScale Power Plant are non-limiting for this event. These scenarios are discussed below:

- Loss of normal AC power: in this scenario, the MPS remains powered so none of the safety systems are automatically actuated; however, feedwater flow is lost and the turbine is tripped. For this event, the initial power is low, so a loss of feedwater and turbine trip do not significantly impact RCS conditions. The secondary side pressure rises slowly, and the SG inventory boils off gradually, and consequently the overall effect is not significant.
- Loss of normal DC power system (EDNS), in addition to loss of normal AC power: power to the trip breakers is provided via the EDNS. This results in a reactor trip, which terminates bank withdrawal, and therefore is non-limiting.

- Loss of the highly reliable DC power system (EDSS), in addition to loss of both EDNS and Normal AC power: this scenario results in reactor trip and actuation of all ESFs. This terminates the bank withdrawal, and therefore is non-limiting.

There are no single failures that could occur during an uncontrolled CRA withdrawal from a subcritical or low power or startup condition event that would result in more severe conditions for the limiting case.

15.4.1.3 Thermal Hydraulic and Subchannel Analyses

15.4.1.3.1 Evaluation Models

The thermal hydraulic analysis of the plant response to an uncontrolled CRA withdrawal from a subcritical or low power or startup condition is performed using NRELAP5. The NRELAP5 model is based on the design features of a NuScale module. The non-loss-of-coolant accident (LOCA) NRELAP5 model is discussed in Section 15.0.2. The relevant boundary conditions from the NRELAP5 analyses are provided to the downstream subchannel critical heat flux (CHF) analysis.

The subchannel core CHF analysis is performed using VIPRE-01. VIPRE-01 is a subchannel analysis tool designed for general-purpose thermal-hydraulic analysis under normal operating conditions, operational transients, and events of moderate severity. Limiting axial and radial power shapes are used in the subchannel analysis to ensure a conservative MCHFR result, in accordance with the methodology described in Reference 15.4-1. See Section 15.0.2.3 for a discussion of the VIPRE-01 code and evaluation model.

15.4.1.3.2 Input Parameters and Initial Conditions

A spectrum of initial conditions is analyzed to find the limiting reactivity insertion due to an uncontrolled CRA withdrawal from a subcritical or low power or startup condition. The initial conditions of the transient evaluation result in a conservative calculation. Table 15.4-2 provides key inputs and the associated biases for the limiting MCHFR case. The following initial conditions and assumptions ensure that the results have sufficient conservatism.

- The minimum initial power assumed for this analysis is 1 watt. The transient analyses for this event evaluate cases with initial powers ranging from this minimum power of 1 watt to 15 percent of full power (consistent with use of the low setting for the high power analytical limit). The SRP guidance states that minimizing initial power provides the most conservative conditions for a CRA withdrawal from a subcritical or low power because it provides the maximum power peak. However, this is not the case for the NuScale design. The SR power rate trip signal prevents a significant power increase at lower powers.
- The least negative/most positive reactivity feedback coefficients are used to minimize the reactivity feedback. A positive MTC for low power (<25 percent) conditions is assumed. The lower reactivity feedback coefficients provide less reactivity feedback to mitigate the surge in power.

- A direct moderator heating value of 2.5 percent is investigated to determine if there is a significant impact on the peak power. The input had a negligible effect on the transient, and is not presented in this section.
- The maximum worth is assumed in the bank withdrawal to provide the highest possible peak power.
- Reactivity insertion rate: The positive reactivity inserted by the CRA withdrawal is modeled as a constant reactivity addition beginning at the transient initiation. The maximum rod speed of 15 inches/min corresponds to a maximum reactivity insertion of 17.84 pcm/s. However to bound the reactivity insertion from possible boron dilution scenarios, a maximum reactivity insertion of 35 pcm/s is analyzed.
  - The reactivity insertion rate for the limiting MCHFR and centerline fuel temperature cases is 0.014 pcm/s.
- Conservative scram characteristics are used, including a maximum time delay, holding the most reactive rod out of the core, and utilizing a bounding control rod drop rate.

The results from the thermal hydraulic evaluation are used as input to the subchannel analysis to determine the MCHFR for this event. The subchannel model is discussed in Section 15.0.2.

15.4.1.3.3 Results

The sequence of events for a representative uncontrolled CRA withdrawal from a low power or startup condition is provided in Table 15.4-1 for the limiting MCHFR and centerline fuel temperature case. Figure 15.4-1 through Figure 15.4-4 show the transient behavior of key parameters for the case that is limiting with respect to MCHFR and centerline fuel temperature.

The CRA bank begins to withdraw at the transient initiation, which begins to raise power, RCS temperature, and RCS pressure. The cases that are limiting for centerline fuel temperature and MCHFR have an initial power of 24MW. This initial power, coupled with a slow withdrawal rate, demonstrates that initially avoiding the high power and power rate trips allows a heatup and pressurization prior to scram. The reactor power rises until the high power (25 percent) limit is reached. This initiates a scram 2 seconds later, when the peak power is achieved. The maximum pressure occurs after the scram has completed.

The limiting cases for an uncontrolled CRA withdrawal from a low power or startup condition demonstrate margin to the acceptance criteria. The limiting MCHFR for this event is above the design limit, and the fuel centerline temperature is below the fuel melting temperature. The limiting values for these acceptance criteria are shown in Table 15.4-3.

15.4.1.4 Radiological Consequences

The normal leakage related radiological consequences of this event are bounded by the design basis accident analyses presented in Section 15.0.3.

15.4.1.5 Conclusions

The two applicable acceptance criteria for this AOO are met for the limiting cases. These acceptance criteria, followed by how the NuScale design meets them are listed below:

- The thermal margin limits departure from nucleate boiling ratio for PWRs as specified in SRP Section 4.4, subsection II.1, are met.
  - The MCHFR for the limiting case is above the design limit, as shown in Table 15.4-3.
- Fuel centerline temperatures as specified in SRP Section 4.2, subsection II.A.2(a) and (b), do not exceed the melting point.
  - The fuel centerline temperature of the limiting case is below the fuel melting temperature, as shown in Table 15.4-3.

15.4.2 Uncontrolled Control Rod Assembly Withdrawal at Power

15.4.2.1 Identification of Causes and Accident Description

A spurious CRA withdrawal that occurs when the reactor is at power leads to an unexpected addition of positive reactivity into the reactor. An uncontrolled CRA withdrawal at power results in an increase in core power with a corresponding increase in heat flux. Due to the time lag in the response of the secondary system, the heat removal from the steam generators follows the heat increase in the primary system. The result is an increase in RCS temperature and pressure. These conditions could challenge design pressures and the SAFDLs. The power range neutron excore detectors provide high power and high flux rate core protection. For cases where the reactivity insertion is sufficiently slow, the high pressurizer pressure and high hot leg temperature limits provide protection. These MPS limits are analyzed for a spectrum of uncontrolled CRA withdrawal conditions to ensure that protection functions are actuated to prevent the violation of the design safety limits.

An uncontrolled CRA withdrawal is expected to occur one or more times in the life of the reactor, and it is classified as an AOO. The categorization of the NuScale design basis events are provided in Table 15.0-1.

15.4.2.2 Sequence of Events and Systems Operation

The sequence of events for a representative uncontrolled CRA withdrawal at power is provided in Table 15.4-4 for the limiting MCHFR case.

Unless specified below, the analysis of an uncontrolled CRA withdrawal event assumes the plant control systems and engineered safety features perform as designed, with allowances for instrument inaccuracy. No operator action is credited to mitigate the effects of an uncontrolled CRA withdrawal event.

The regulating CRA banks contain the only CRAs that are not fully withdrawn during power operation. The power dependent insertion limit (PDIL) restricts the amount of insertion steps that the regulating banks can achieve during power operation. The

uncontrolled CRA withdrawal analysis assumes that both regulating banks are inserted to the PDILs with an uncertainty of six insertion steps added to the position. The expected normal travel rate of the CRAs is 6 in./min. However, the maximum allowed withdrawal rate of a CRA is 15 in./min, with a step size no greater than three-eighths inch. This corresponds to a maximum possible reactivity insertion of 20 pcm/s. A spectrum of constant reactivity insertion rates that includes the maximum rate and bounds possible boron dilution scenarios is included in the uncontrolled CRA withdrawal analysis.

The effect of a reactivity insertion event on the RCS is an increase in temperature, which decreases density and causes flow into the pressurizer, increasing RCS pressure. As a result, the normal module control system response would be to decrease pressurizer heater power. In uncontrolled CRA withdrawal cases in which a trip occurs on high pressurizer pressure, the heaters are disabled; in other cases the heater output is held constant at the steady-state value. The pressurizer spray is assumed to function normally in cases in which this delays a trip on high pressurizer pressure; in other cases the spray response to increasing pressurizer pressure is disabled.

The MPS is credited to protect the plant in the event of an uncontrolled CRA withdrawal. The following MPS signals provide the plant with protection during an uncontrolled CRA withdrawal:

- high core power
- high core power rate
- high RCS hot temperature
- high pressurizer pressure

In uncontrolled CRA withdrawal events that result in a reactor trip, the subsequent actuation of the decay heat removal system (DHRS) is credited with maintaining reactor cooling. The MPS signals credited for DHRS actuation are high RCS hot temperature, high pressurizer pressure and high steam line pressure.

There are no single failures that could occur during an uncontrolled CRA withdrawal event that result in a more severe outcome for the limiting uncontrolled CRA withdrawal cases. The diversity, redundancy, and independence of the MPS ensure the system will perform its intended function despite a single failure.

The loss of normal AC power is analyzed for an uncontrolled CRA withdrawal at power. It was determined that loss of AC power at event initiation is non-limiting, and loss of AC power at reactor trip has a negligible impact on the analysis results.

- Loss of Normal AC - In this scenario, the MPS remains powered, so none of the safety systems are automatically actuated.
- Loss of EDNS and Loss of normal AC - Power to the control rod drive mechanisms is provided via the nonsafety DC power distribution (EDNS), so this scenario is the same as discussed above, with the addition of the CRAs dropping at the time at which power is lost. For this event, this scenario is non-limiting because of the immediate loss of power to the CRDMs, resulting in the drop of the CRAs.

- Loss of EDSS, EDNS and Loss of normal AC - Power to the MPS is provided by the highly-reliable DC power distribution system (EDSS), so this scenario results in an actuation of RTS and all of the engineered safety features. This scenario is non-limiting because of the immediate reactor trip.

15.4.2.3 Thermal Hydraulic and Subchannel Analyses

15.4.2.3.1 Evaluation Models

The thermal hydraulic analysis of the plant response to an uncontrolled CRA withdrawal is performed using NRELAP5. The NRELAP5 model is based on the design features of a NuScale module. The non-LOCA NRELAP5 model is discussed in Section 15.0.2. The relevant boundary conditions from the NRELAP5 analyses are provided to the downstream subchannel CHF analysis.

The subchannel core CHF analysis is performed using VIPRE-01. VIPRE-01 is a subchannel analysis tool designed for general-purpose thermal-hydraulic analysis under normal operating conditions, operational transients, and events of moderate severity. Limiting axial and radial power shapes are used in the subchannel analysis to ensure a conservative MCHFR result, in accordance with the methodology described in Reference 15.4-1. See Section 15.0.2 for a discussion of the VIPRE-01 code and evaluation model.

15.4.2.3.2 Input Parameters and Initial Conditions

A spectrum of initial conditions is analyzed to find the limiting reactivity insertion due to an uncontrolled CRA withdrawal. Key inputs of the uncontrolled CRA withdrawal evaluation are provided in Table 15.4-5 for the limiting MCHFR case, and Table 15.4-31 for the limiting linear heat generation rate (LHGR) case. The following initial conditions and assumptions ensure that the results have sufficient conservatism.

- Initial power level: 25 percent, 50 percent, 75 percent, and 102 percent of nominal power are analyzed in the uncontrolled CRA withdrawal evaluation. The power level for the limiting MCHFR case is 75 percent of nominal power. The power level for the limiting LHGR case is 102 percent of nominal power.
- Reactivity insertion rate: The positive reactivity inserted by the CRA withdrawal is modeled as a constant reactivity addition beginning at the transient initiation. The maximum rod speed of 15 inches/min corresponds to a maximum reactivity insertion of 20 pcm/s. However, to bound the reactivity insertion from possible boron dilution scenarios, a maximum reactivity insertion of 35 pcm/s is analyzed.
  - The reactivity insertion rate for the limiting MCHFR case is 0.92 pcm/s.
  - The reactivity insertion rate for the limiting LHGR case is 35 pcm/s.
- Reactivity Feedback: The least negative reactivity coefficients are implemented in the limiting uncontrolled CRA withdrawal cases. The least negative reactivity coefficients provide the least amount of feedback to mitigate the power increase due to an uncontrolled CRA withdrawal.

- The turbine bypass system is not credited in this analysis to minimize heat removal by the secondary side.
- Conservative scram characteristics are used, including a maximum time delay, holding the most reactive rod out of the core, and using a bounding control rod drop rate.
- Allowances for instrument inaccuracy are accounted for in the analytical limits of mitigating systems in accordance with the guidance provided in Regulatory Guide (RG) 1.105.

The results from the thermal hydraulic evaluation are used as input to the subchannel analysis to determine the MCHFR for this event. The subchannel evaluation model is discussed in Section 15.0.2.

15.4.2.3.3 Results

The sequence of events for a limiting uncontrolled CRA withdrawal with respect to MCHFR is provided in Table 15.4-4. Figure 15.4-6 through Figure 15.4-11 show the transient behavior of key parameters for an uncontrolled CRA withdrawal.

The withdrawal of the regulating bank results in a reactivity insertion that increases reactor power. The power increase leads to a rise in RCS temperature, pressurizer level, and RCS pressure. Feedback from the rising fuel and moderator temperatures partially counteracts the reactivity insertion, slowing the power increase. For uncontrolled CRA withdrawal cases with higher reactivity insertion rates, the MPS trips the reactor on high pressurizer pressure or high power rate. These cases are non-limiting for MCHFR because the reactor is tripped before the maximum amount of reactivity can be inserted. The limiting combination of reactivity insertion and reactivity feedback produces the maximum possible power increase prior to trip. The power increase in the limiting MCHFR case is terminated by a reactor trip after a signal delay. The high RCS hot temperature limit is reached during the reactor trip delay time. The MPS trips the reactor and actuates secondary side isolation and the DHRS during this event. The most limiting MCHFR occurs near the time of the power peak. The MCHFR remains above the design limit, and no fuel centerline melting is predicted for the uncontrolled CRA withdrawal.

The maximum LHGR case for an uncontrolled CRA withdrawal at power occurs for the case with the highest reactivity insertion rate. The rapid increase in power causes a high power rate trip. Sensitivity calculations show that the maximum power is not sensitive to the assumption of loss of normal AC power at reactor trip. This case results in the limiting power peak and thus a higher fuel temperature as evidenced by the LHGR. The LHGR remains below the design limit, so no fuel centerline melting is predicted for the uncontrolled CRA withdrawal.

The uncontrolled CRA withdrawal at power cases that result in a reactor trip, secondary side isolation, and DHRS actuation, demonstrate that stable core cooling is maintained.

15.4.2.4 Radiological Consequences

The normal leakage related radiological consequences of this event are bounded by the design basis accident analyses presented in Section 15.0.3.

15.4.2.5 Conclusions

The two applicable acceptance criteria for this AOO are met for the limiting uncontrolled CRA withdrawal cases. These acceptance criteria, followed by how the NuScale Power Plant design meets them are listed below.

1) The thermal margin limits departure from nucleate boiling ratio for pressurized water reactors as specified in SRP Section 4.4, subsection II.1, are met.

   - The MCHFR for the limiting uncontrolled CRA withdrawal is above the design limit, as shown in Table 15.4-6. Therefore, this criterion is met.

2) Fuel centerline temperatures as specified in SRP Section 4.2, subsection II.A.2(a) and (b), do not exceed the melting point.

   - As discussed in Reference 15.4-1, a steady-state linear heat generation rate (LHGR) protection limit can be applied to an uncontrolled CRA withdrawal at power event to ensure that the fuel centerline temperatures do not exceed the melting point. The LHGR for the limiting CRA withdrawal is below the limit, as shown in Table 15.4-6.

15.4.3 Control Rod Misoperation (System Malfunction or Operator Error)

15.4.3.1 Identification of Causes and Accident Description

A control rod misoperation (CRM) could introduce an unexpected change of reactivity in the core. The resulting change in power distribution could lead to a decrease in the CHF ratio. CRM events analyzed in this section are:

- Control rod assembly misalignments. A single CRA or multiple CRAs are displaced in a position relative to the bank while the other rods in its bank are in another allowed position.
- Single CRA withdrawals. A single CRA is inadvertently withdrawn from the bank insertion limit. The withdrawal of an entire bank of CRAs is analyzed in Section 15.4.1 and Section 15.4.2.
- Control rod assembly drop. A single CRA or multiple CRAs are dropped inadvertently into the reactor core.

The CRA misalignment is an event where a single or multiple CRAs are out of alignment with the remaining rods in the bank. The alignment may be higher or lower than the expected rod position. A misalignment can occur based on the uncertainty in the rod position from its indicated or expected position. A limiting misalignment occurs when the core is operating at steady-state full power with the rods inserted to the PDILs except one rod is left withdrawn. This is a postulated condition as the rod position indicators will alarm when the rods are out of alignment beyond the position

uncertainty. The effects of this misalignment are bounded by the analysis of the withdrawal of a single CRA. Another limiting misalignment that is postulated is where all CRAs are withdrawn, except one that is misaligned in to the 25 percent rated power PDIL position. The analysis of this misalignment is presented in this section. The postulated misalignment of the rods inserted to the PDIL but with one CRA fully inserted is not a credible condition for the NuScale core. Reactor hold points will prohibit the movement of rods for that severe of a peaking distortion and therefore is not analyzed for NuScale. The CRA misalignments are classified as AOOs.

The single rod withdrawal transient occurs when a control rod is set at the bank position PDIL and is postulated to withdraw. This event may occur due to wiring failures or operator error in which one rod is pulled with disregard for rod position information. The single rod withdrawal adds reactivity and initiates a power increase transient. The power distribution in the core becomes asymmetric and peaking can challenge the MCHFR safety limit. A CRM that results in a withdrawal is classified as an AOO.

The CRA or bank drop occurs when a single rod or entire group from the control or shutdown banks drops into the core. This can be caused by a mechanical or electrical failure. The event is characterized by a sudden drop in reactor core power. When the rod worth is not significant enough to shut the core down, the constant demand of the secondary side causes a decrease in core inlet temperature, which could result in a power increase. The control system for the regulating control rod bank could withdraw the regulating bank to restore power. The resulting power overshoot with asymmetric peaking could challenge the MCHFR safety limit. A single CRA drop event is classified as an AOO.

15.4.3.2 Sequence of Events and Systems Operation

The sequence of events for each CRM transient is discussed in the respective results section.

Unless specified below, the analysis of a CRM event assumes the plant control systems and engineered safety features perform as designed, with allowances for instrument inaccuracy. No operator action is credited to mitigate the effects of a CRM event.

The regulating CRA groups contain the only CRAs that are not fully withdrawn during power operation. The PDIL restricts the amount of insertion steps that the regulating groups can achieve during power operation. The CRM analyses assume that the regulating bank CRAs have a position uncertainty of six insertion or withdrawal steps. The expected normal travel rate of the CRAs is 6 in./min. However, the maximum allowed withdrawal rate of a CRA is 15in./min, with a step size no greater than three-eighths inch. The conservative analysis of the rod drop event does not credit the reduction in the worth of the regulating control rod bank due to the dropped rod.

The effect of a reactivity insertion event on the RCS is an increase in temperature, which decreases density and causes flow into the pressurizer, increasing RCS pressure. As a result, the normal module control system response would be to decrease pressurizer heater power.

For CRM cases analyzing MCHFR, the heater is disabled in cases in which a trip occurs on high pressurizer pressure; in other cases the heater output is assumed to remain constant at its initial value. Pressurizer spray is allowed to function normally in cases in which this delays a trip on high pressurizer pressure; in other cases, spray is disabled.

The MPS is credited to protect the plant in the event of a CRM. The following MPS reactor trip signals provide the plant with protection during a CRM:

- high pressurizer pressure
- high power
- high power rate
- high RCS hot temperature

In CRM events that result in a reactor trip and SSI, the subsequent actuation of the DHRS is credited with maintaining reactor cooling. The MPS signals credited for SSI are high RCS hot temperature, high pressurizer pressure, high steam line pressure, and high steam line superheat. The MPS signals credited for DHRS actuation are high RCS hot temperature, high pressurizer pressure, and high steam pressure.

The CRM event analyses assume a single failure of an ex-core flux detector. This failure could affect the power-related reactor trips (high power and high power change rate). Due to the power asymmetry resulting from the CRA drop and withdrawal transients, it is possible for a failed detector to cause an error in the detected power. This error could occur if the remaining detectors are located in regions with relative flux that has either increased or decreased due to the power asymmetry. This effect is captured in the analyses by assuming that the remaining detectors see the lowest possible flux in the CRA withdrawal cases, and the highest possible flux in the CRA drop cases.

The potential loss of power scenarios for the NPP during a CRM are discussed below:

- Loss of Normal AC - In this scenario, the MPS remains powered, so none of the safety systems are automatically actuated. However, power is lost to the feedwater pumps, CVCS recirculation pumps, pressurizer heaters, and the condenser, resulting in a turbine trip. For a CRM, the loss of AC power is non-limiting for evaluation of MCHFR.
- Loss of EDNS and loss of normal AC - Power to the control rod drive mechanisms is provided via the nonsafety DC power distribution (EDNS), so this scenario is the same as discussed above, with addition of the CRAs dropping at the time at which power is lost. For this event, this scenario is non-limiting because of the immediate loss of power to the CRDMs. This results in the drop of the CRAs, terminating the event.
- Loss of EDSS, EDNS and LOAC - Power to the MPS is provided by the highly-reliable DC power distribution system (EDSS), so this scenario results in an actuation of RTS and all of the engineered safety features. This scenario is non-limiting because of the immediate reactor trip.

15.4.3.3 Control Rod Assembly Misalignment Analysis

15.4.3.3.1 Evaluation Models

There is no plant thermal hydraulic analysis required for the CRA misalignments given the steady-state nature of the plant thermal hydraulic conditions of the events. The steady-state core analyses are performed using SIMULATE5 to provide power distributions as input to the subchannel analyses. A discussion of SIMULATE5 is provided in Section 4.3.

The subchannel core CHF analysis is performed using VIPRE-01. VIPRE-01 is a subchannel analysis tool designed for general-purpose thermal-hydraulic core analysis under normal operating conditions, operational transients, and events of moderate severity. See Section 15.0.2.3 for a discussion of the VIPRE-01 code and evaluation model.

15.4.3.3.2 Input Parameters and Initial Conditions

The key inputs for the primary side conditions for the subchannel analysis of the CRA misalignment are provided in Table 15.4-33. The limiting CRA misalignment presented in this section occurs when all CRAs are withdrawn, except for one that is inserted to the 25 percent power PDIL. The following input parameters and initial conditions are considered in this CRA misalignment analysis to ensure a conservative calculation:

- The misaligned CRA is assumed to be inserted 6 steps past the PDIL to account for CRA position indication uncertainty. The added insertion steps maximize the misalignment, providing the most conservative power asymmetry.
- The full range of initial power conditions are analyzed for this misalignment scenario, including: 25, 50, 75, and 100 percent initial power.
- The most positive and negative axial offset values are considered in the analysis.
- The BOC, MOC, and EOC core conditions are analyzed.

15.4.3.3.3 Results

A static misalignment results in a change in local power shapes and peaking, but the overall power of the core does not change. The thermal hydraulic boundary conditions of the core are at steady-state such that there is no plant thermal hydraulic transient analysis required for input to the subchannel analysis. Instead, the misalignment scenarios are analyzed to ensure that the applicable SAFDLs are not violated.

The MCHFR for the limiting static misalignment of a CRA is above the design limit and bounded by the MCHFR of the single CRA withdrawal, presented in Section 15.4.3.4.

15.4.3.4 Single Control Rod Assembly Withdrawal Analysis

15.4.3.4.1 Evaluation Models

The thermal hydraulic analysis of the plant response to a single CRA withdrawal is performed using NRELAP5. The NRELAP5 model is based on the design features of a NuScale module. The non-LOCA NRELAP5 model is discussed in Section 15.0.2. The relevant boundary conditions from the NRELAP5 analyses are provided to the downstream subchannel CHF analysis.

The subchannel core CHF analysis is performed using VIPRE-01. VIPRE-01 is a subchannel analysis tool designed for general-purpose thermal-hydraulic analysis under normal operating conditions, operational transients, and events of moderate severity. See Section 15.0.2 for a discussion of the VIPRE-01 code and evaluation model.

15.4.3.4.2 Input Parameters and Initial Conditions

A spectrum of initial conditions is analyzed to find the limiting reactivity insertion due to a single CRA withdrawal. Key inputs and the associated biases for the limiting single CRA withdrawal analyses are provided in Table 15.4-8 with respect to MCHFR, and Table 15.4-32 with respect to LHGR. The following initial conditions and assumptions ensure that the results have sufficient conservatism.

- Initial power level: 25 percent, 50 percent, 75 percent, and 102 percent of nominal power are analyzed to find the limiting cases.
  - The initial power level for the limiting MCHFR case is 75 percent of nominal power.
  - The initial power level for the limiting LHGR case is 75 percent of nominal power.
- Reactivity insertion rate: The positive reactivity inserted by the CRA withdrawal is modeled as a constant reactivity addition beginning at the transient initiation. The uncontrolled CRA withdrawal evaluation considers reactivity addition rates up to 12 pcm/s. This value bounds the 9.77 pcm/s that corresponds to the maximum CRA withdrawal rate of 15 in./min. The reactivity insertion rate for the limiting MCHFR case is 2.5 pcm/s, and the limiting RCS pressure and LHGR cases is 1.32 pcm/s.
- Reactivity Feedback: The least negative reactivity coefficients are implemented in the limiting CRA withdrawal cases. The least negative reactivity coefficients provide the least amount of feedback to mitigate the power increase due to a CRA withdrawal.
- Conservative scram characteristics are used, including a maximum time delay and holding the most reactive rod out of the core.
- The turbine bypass system is not credited in this analysis to minimize heat removal by the secondary side.
- Allowances for instrument inaccuracy are provided for setpoints of mitigating systems in accordance with RG 1.105.

- The limiting axial and radial power shapes are used in the subchannel analysis to ensure a conservative evaluation of the SAFDLs.

The results from the thermal hydraulic evaluation are used as input to the subchannel analysis to determine the limiting MCHFR and LHGR for this event. The subchannel evaluation model is discussed in Section 15.0.2.

15.4.3.4.3 Results

The sequence of events for a single CRA withdrawal that results in the minimum MCHFR is provided in Table 15.4-7. Figure 15.4-13 through Figure 15.4-19 and Figure 15.4-35 show the transient behavior of key parameters for this event.

The withdrawal of a single CRA that results in a limiting MCHFR has an initial power of 75 percent. The withdrawal of the CRA results in a reactivity insertion that increases reactor power. The power increase leads to a rise in RCS temperature, pressurizer level, and RCS pressure. The CRA misalignment with the rest of the bank causes an asymmetry in the core, where power peaking increases in the location of the withdrawn CRA. Reactivity feedback from the rising fuel and moderator temperatures partially counteracts the reactivity insertion, slowing the power increase. For CRM cases with higher reactivity insertion rates, the MPS trips the reactor on high reactor power or high power rate. These cases are non-limiting because the reactor is tripped before the maximum amount of reactivity can be inserted. The limiting combination of reactivity insertion and reactivity feedback produces the maximum possible power increase without reaching the high reactor power or high power rate limits. The power increase is terminated by MPS actuation of reactor trip, when the high hot temperature limit and the RCS pressure high pressurizer pressure limit are reached simultaneously inclusive of each respective trip delay.

The MPS high pressurizer pressure signal trips the reactor and actuates the DHRS. The most limiting MCHFR (Figure 15.4-35) occurs at the moment before the power begins to decrease. The MCHFR remains above the design limit, and no fuel centerline melting is predicted for the withdrawal of a single CRA. The LHGR calculated for the single CRA withdrawal is below the calculated limits for cladding strain and fuel centerline melting. The maximum RCS pressure occurs approximately 1 second after MCHFR occurs, and is followed by decreasing RCS temperature and pressure. The limiting LHGR occurs in a single CRA withdrawal case that has an initial power of 75 percent. The limiting reactivity insertion rate is the same as that for the MCHFR case, and does not cause a high power rate trip like cases at lower initial powers. The pressurizer spray is assumed to function normally, which delays the trip on high pressure until after the power has peaked. This case maximizes power in combination with asymmetric peaking, resulting in a limiting LHGR. The LHGR remains below the design limit, so no fuel centerline melting is predicted for the single CRA withdrawal.

15.4.3.5 Control Rod Assembly Drop Analysis

15.4.3.5.1 Evaluation Models

The thermal hydraulic analysis of the plant response to a CRA drop is performed using NRELAP5. The NRELAP5 model is based on the design features of a NuScale module. The non-LOCA NRELAP5 model is discussed in Section 15.0.2. The relevant boundary conditions from the NRELAP5 analyses are provided to the downstream subchannel CHF analysis.

The subchannel core CHF analysis is performed using VIPRE-01. VIPRE-01 is a subchannel analysis tool designed for general-purpose thermal-hydraulic analysis under normal operating conditions, operational transients, and events of moderate severity. See Section 15.0.2.3 for a discussion of the VIPRE-01 code and evaluation model.

15.4.3.5.2 Input Parameters and Initial Conditions

A spectrum of initial conditions is analyzed to find the limiting MCHFR and limiting LHGR conditions for a single or multiple CRA drop. The initial conditions of the CRA drop evaluation result in a case that produces a conservative calculation of both MCHFR and LHGR. Key inputs and the associated biases for the limiting CRA drop analysis are provided in Table 15.4-10. The following initial conditions and assumptions ensure that the results have sufficient conservatism.

- The initial power for the limiting MCHFR and LHGR case is 102 percent of nominal power. Twenty-five percent, 50 percent, 75 percent, and 102 percent of nominal power are analyzed to find the limiting conditions for MCHFR and LHGR.
- Dropped CRA conditions: The CRA drop scenarios are analyzed to find the worst power peaking change after the drop. These relative peaking changes are applied to the entire transient in the subchannel analysis to ensure a conservative calculation of MCHFR. Due to the size of the NuScale core and relative rod worth, a CRA drop from higher power conditions will result in a substantial loss of power and subsequent negative power rate trip such that the transient power level remains well below the initial power level. Since the worst power peaking conditions are applied to the entire limiting CRA drop transient, the limiting MCHFR occurs at the transient initiation when initialized at 102 percent power. This case also results in the maximum LHGR.
- Time in cycle: The EOC core conditions are implemented in the limiting CRA drop cases. The most negative reactivity coefficients occur at the EOC and provide the most reactivity feedback to mitigate the power decrease due to a CRA drop.
- Conservative scram characteristics are used, including a maximum time delay and holding the most reactive rod out of the core.
- The turbine bypass system is not credited in this analysis to minimize heat removal by the secondary side.

- Allowances for instrument inaccuracy are provided for setpoints of mitigating systems in accordance with RG 1.105.
- The limiting axial and radial power shapes are used in the subchannel analysis to ensure a conservative evaluation of the SAFDLs.

The results from the thermal hydraulic evaluation are used as input to the subchannel analysis to determine the limiting MCHFR and LHGR for this event. The subchannel evaluation model is discussed in Section 15.0.2.

15.4.3.5.3 Results

The sequence of events for the bounding single CRA drop is provided in Table 15.4-9. Figure 15.4-21 through Figure 15.4-27 and Figure 15.4-36 show the transient behavior of key parameters for a single CRA drop. Following a CRA drop in the NuScale reactor, there is a rapid drop in the core reactivity and power. The high power rate limit is reached at just under 1 second into the transient. The MPS sends a reactor trip signal, terminating the event. At lower powers, the power decrease is less pronounced, and the reactor does not trip. In the lower power cases, the regulating CRA bank brings the reactor back to the initial power after an initial power overshoot. However, these cases are non-limiting with respect to MCHFR and LHGR.

The MCHFR for the limiting case, Figure 15.4-36, remains above the design limit. The LHGR calculated for the limiting rod drop case is below the limits for fuel melting and cladding strain.

15.4.3.6 Radiological Consequences

The normal leakage related radiological consequences of these events are bounded by the design basis accident analyses presented in Section 15.0.3.

15.4.3.7 Conclusions

The CRM events meet the SRP 15.4.3 acceptance criteria as follows:

1) The thermal margin limits departure from nucleate boiling ratio, as specified in SRP Section 4.4, subsection II.1, are met.

    - The MCHFR for the limiting CRA misalignment is above the CHF design limit, as shown in Table 15.4-11.
    - The MCHFR for the limiting single CRA withdrawal is above the CHF design limit, as shown in Table 15.4-11.
    - The MCHFR for the limiting CRA drop is above the CHF design limit, as shown in Table 15.4-11.

2) Fuel centerline temperatures as specified in SRP Section 4.2, subsection II.A.2(a) and (b), do not exceed the melting point.

- As discussed in Reference 15.4-1, a steady-state linear heat generation rate protection limit can be applied to the CRM events to ensure that the fuel centerline temperatures do not exceed the melting point. The linear heat generation rate for the limiting CRA misalignment is below the limit for fuel melt, as shown in Table 15.4-11.
- The linear heat generation rate for the limiting single CRA withdrawal is below the limit for fuel melt, as shown in Table 15.4-11.
- The linear heat generation rate for the limiting CRA drop is below the limit for fuel melt, as shown in Table 15.4-11.

3) Uniform cladding strain as specified in SRP Section 4.2, subsection II.A.2(b), does not exceed 1 percent.

- A core design evaluation provides cladding strain limits for NuScale fuel in terms of LHGR. The LHGR values for each of the CRM events is below these limits, demonstrating that cladding strain is not predicted.

15.4.4 Startup of an Inactive Loop or Recirculation Loop at an Incorrect Temperature

There are not multiple coolant loops in the NuScale design. This event is not applicable to the NuScale design.

15.4.5 Flow Controller Malfunction Causing an Increase in Core Flow Rate (Boiling Water Reactor)

There is no steam pressure regulator in the NuScale design. This event is not applicable to the NuScale design.

15.4.6 Inadvertent Decrease in Boron Concentration in the Reactor Coolant System

15.4.6.1 Identification of Causes and Accident Description

A malfunction in the chemical volume and control system (CVCS) or an operator error could result in unborated or diluted water being inadvertently added to the RCS. In the NuScale design, a boric acid blend system allows the operator to match or adjust the boron concentration of the reactor coolant makeup water during normal operation. Boron dilution can be either an automatic or a manual operation. In either case, the dilution is governed by administrative controls with procedures that establish the limits on the rate and duration of dilution. An unintended decrease in boron concentration increases the reactivity of the core and decreases the shutdown margin.

The module protection system (MPS) is designed to isolate the demineralized water source prior to the loss of a significant portion of the technical specification minimum shutdown margin. The MPS automatically isolates the demineralized water source on high subcritical multiplication, low RCS flow, and any reactor trip system (RTS) actuation. The CVCS is disconnected from the RCS during Mode 4 (transition) and Mode 5 (refueling), but the refueling mode is evaluated to make sure that it bounds the effects of other possible dilution sources present during the refueling process.

An inadvertent decrease in boron concentration in the RCS is expected to occur one or more times during the lifetime of the reactor, and is classified as an AOO.

15.4.6.2 Sequence of Events and Systems Operation

An inadvertent decrease in boron concentration in the RCS is evaluated for Modes 1, 2, 3, and 5. Boron dilution causes an increase in reactivity, and the plant response to the event is similar to an uncontrolled CRA withdrawal, presented in Section 15.4.1 and Section 15.4.2. The limiting CVCS dilution source considered in this analysis is the demineralized water system (DWS) supply. To reduce the overall probability of boron dilution events, administrative controls are placed on the boron addition system supply to the CVCS makeup pumps, assuring that it is not a dilution source for the RCS or the refueling pool.

Unless specified in this section, the RCS boron dilution evaluation assumes the plant control systems and engineered safety features perform as designed, with allowances for instrument inaccuracy. No operator action is credited to mitigate the effects of an RCS boron dilution event.

The CVCS has two variable speed positive displacement makeup pumps that supply the RCS with makeup coolant and change RCS boron concentration by supplying blended makeup water. Each makeup pump has a maximum capacity of 20 gpm. Normally one pump is operated at a time and the other is provided for redundancy. In most cases, normal power operations require no more than 20 gpm but it is possible to run both makeup pumps at the same time above 50 percent power. Therefore, the maximum possible CVCS makeup flow rate is supplied by running two makeup pumps with a maximum capacity of 40 gpm. For conservatism, the analysis applies an additional 5 gpm uncertainty to the makeup flow rates for each pump (25 gpm/pump). Below 50 percent power, the demineralized water flow rate from the CVCS makeup is limited to 25 gpm. To prevent reactor trip on high pressurizer pressure or level, the CVCS letdown mass flow rate is maintained equal to the makeup flow rate.

The regulating CRA bank is not credited with mitigating the reactivity insertion associated with a boron dilution of the RCS. Each of the two regulating bank groups is assumed to be at their respective PDIL so that rods do not insert automatically as a result of the reactivity addition of an RCS boron dilution.

Technical Specifications preclude the possibility of boron dilution when a CRA is stuck out during Modes 2 and 3 by enforcing shutdown margin requirements. Plant procedures provide controls for connected sources that could cause dilution of the reactor pool.

A boron dilution event could occur during refueling operation (Mode 5) if unborated water is unexpectedly introduced to the reactor building pool. Table 15.4-12 provides possible reactor building internal flooding sources with estimated water volumes that could introduce unborated water to the pool. It is assumed that the flooding source immediately enters the pool and mixes perfectly and instantaneously to provide the maximum boron dilution rate, except if a loss of AC power is assumed.

A loss of normal power is considered during Modes 1 and 5. The loss of alternating current (AC) power during Mode 5 results in the loss of the reactor pool cooling and recirculation system. To accommodate the loss of pool cooling circulation, the initial minimum pool mixing volume assumed in the analysis is further reduced for conservatism. In Mode 1 operation, the loss of power scenarios are non-limiting.

- Loss of normal AC - In this scenario, MPS remains powered so none of the safety systems are automatically actuated. The feedwater pumps, pressurizer heaters, the turbine stop valve, and CVCS pumps (recirculation and makeup) are assumed to fail. By securing CVCS, the dilution event is terminated and ends the event earlier than if the pumps were allowed to continue to operate.
- Loss of normal direct current (DC) power system and loss of normal AC - Power to the reactor trip breakers is provided by the normal DC power system so this scenario is the same as a loss of normal AC scenario with the addition of the reactor trip at the time at which power is lost. For the boron dilution event, this scenario is non-limiting for the reasons listed above.
- Loss of highly reliable DC power system, normal DC power system, and loss of normal AC - Power to the MPS is provided via the highly reliable DC power system so this scenario results in an actuation of a reactor trip and all of the engineered safety functions. In terms of boron dilution event, this scenario is non-limiting for the reasons discussed above. Also, the CVCS demineralized water supply isolation valves are normally held open with instrument air. Upon loss of instrument air or power, the valve returns to a closed state by a passive force. The closed position is the safe position for the demineralized water isolation valves since it isolates the dilution path.

During Modes 2 and 3, the reactor is subcritical and there is no power produced by the turbine. Therefore, a loss of AC power due to a possible grid disturbance following a turbine trip is not postulated to occur during Modes 2 and 3.

The MPS signals to trip the reactor and isolate the CVCS are credited with protecting the plant in the event of a boron dilution of the RCS. The demineralized water supply to the CVCS makeup pumps is isolated by two series safety-related isolation valves on the following MPS signals:

- any RTS
- high subcritical multiplication
- low RCS flow

These MPS signals provide protection in Modes 1 through 3.

There are no single failures that could occur during a boron dilution of the RCS that result in a more severe outcome for the limiting cases. The diversity, redundancy, and independence of the MPS and CVCS isolation valves ensure the plant protection is provided for a boron dilution of the RCS despite a single failure.

15.4.6.3 Boron Mixing, Thermal Hydraulic, and Subchannel Analyses

15.4.6.3.1 Evaluation Methodology Summary

Two calculation techniques are used to analyze the boron dilution event for the NuScale module, which provide conservative boron dilution assumptions for the evaluation of both reactivity insertion and loss of shutdown margin. The first method evaluates the boron dilution by assuming an instantaneous perfect (complete) mixing model. The second method evaluates the boron dilution by assuming a slug flow or dilution front (wave front) mixing model. In the instantaneous perfect mixing model, unborated water injected into the RCS is assumed to mix instantaneously with the effective system volume. The change in core boron concentration with time is continuous and homogeneous, corresponding to the increasing amount of dilution water entering the RCS. In the dilution front model, unborated water injected into the RCS is assumed to mix with a slug of borated water at the injection point. The diluted slug is assumed to move through the RCS (i.e. through the riser, steam generators, downcomer, and finally though the reactor core). The change in core boron concentration with time depends on the location of the diluted slug.

The two calculation techniques provide the reactivity insertion rate due to the boron dilution. To ensure that the SRP 15.4.6 acceptance criteria are met, the reactivity insertion rate in Mode 1 operation is compared to the spectrum of reactivity insertion rates evaluated in the uncontrolled CRA withdrawal from a subcritical or low power or startup condition and uncontrolled CRA withdrawal analyses at power in Section 15.4.1 and Section 15.4.2, respectively. The CRA withdrawal thermal-hydraulic analyses provide reactor trip timing to estimate when the MPS terminates the boron dilution event. Adequate shutdown margin must remain when the boron dilution event is terminated. For Mode 2 and Mode 3 operation, the boron dilution scenarios are evaluated at the time of DWS isolation to ensure that adequate shutdown margin remains at the time of automatic DWS isolation. Boron dilution of the reactor pool is estimated using the perfect mixing equation (Equation 15.4-1). The boron dilution scenarios in Mode 5 operation are performed with conservative assumptions of water volumes to demonstrate that shutdown margin is maintained.

15.4.6.3.2 Boron Dilution Assuming Perfect Mixing

The perfect (complete) mixing method evaluates the boron concentration of the RCS with the following equation:

$$\frac{dC}{dt} = \frac{Q_{in}\rho_{in}}{V_r\rho_r}C(t) \qquad \text{Equation 15.4-1}$$

where,

Q_{in} = dilution flow rate of unborated water (gpm). The maximum dilution flow rate [gpm] is used for this parameter based on the ability of the makeup pumps to deliver water to the CVCS injection line.

ρ_{in} = dilution water density (lb_m/cu.ft). The density value at 14.7psia, 40 °F (Min value) is used in all cases. The heat addition by the regenerative heat exchangers is not credited, so the analysis assumes that the recirculation pumps are not operational.

V_r = effective water volume of the RCS (gal). A conservatively small value is used which removes the volume of the pressurizer.

ρ_r = density of the water in the RCS (lb_m/cu.ft), and

C(t) = time dependent concentration of boron in the RCS (ppm).

The reactivity insertion rate associated with a given boron dilution rate is calculated with the following equation:

$$\frac{dR}{dt} = \alpha_B \frac{dC}{dt} = -\frac{\alpha_B Q_{in} \rho_{in}}{60 V_r \rho_r} C(t) \qquad \text{Equation 15.4-2}$$

where,

dR/dt = reactivity insertion rate [pcm/sec], and

α_B = differential boron worth [pcm/ppm].

15.4.6.3.3 Boron Dilution Assuming Dilution Front or Slug Flow Model

The dilution front or slug (wave front) model uses the following equation for boron concentration:

$$C(N) = C_0 \left[\frac{W_{NC}}{(W_D + W_{NC})}\right]^N \qquad \text{Equation 15.4-3}$$

$$t = \frac{M_{RCSI}}{W_D + W_{NC}} + (N-1)\frac{M_{RCS}}{W_D + W_{NC}}$$

where,

C(N) = the Nth front boron concentration,

C_0 = initial boron concentration,

W_D = dilution mass flow rate,

W_{NC} = natural circulation mass flow rate,

M_{RCS} = RCS fluid mass minus the pressurizer,

M_{RCSI} = Initial pass RCS fluid mass (mass between the CVCS injection point to core inlet), and

N = number of times the wave front passes through the core (1, 2, etc.).

In this model, the boron concentration in the RCS is reduced in discrete steps at each time, t, corresponding to the time the wave front passes through the core. Using these equations, the ratio $C_0/C(t)$ is calculated which corresponds to discrete times after dilution begins.

15.4.6.3.4 Input Parameters and Initial Conditions

The initial conditions and input parameters for the boron dilution of the RCS analysis are selected to insure a conservative calculation.

- The shutdown margin threshold in this analysis for Mode 1 is when k_{eff} = 0.98. The shutdown margin threshold for Modes 2 and 3 in this analysis is when k_{eff} = 0.93. The shutdown margin threshold in this analysis for Modes 4 and 5 is when k_{eff} = 0.90. Therefore, the shutdown margin reactivity credited in this analysis is 2041 pcm for Mode 1, 7527 pcm for Modes 2 and 3 and 11,111 pcm for Modes 4 and 5.
- For Mode 1 operation, the initial power levels considered for a boron dilution of the RCS include: hot zero power (HZP), 25 percent power, 50 percent power, 75 percent power and full (100 percent) power. The limiting Mode 1 cases of HZP and full power are provided in this section.
- Maximum bounding boron concentrations and boron coefficients are assumed because the rate of change of concentration and associated reactivity is greater for an initially higher concentration. The critical boron concentrations and boron reactivity coefficients assumed for each mode of operation are provided in Table 15.4-13.
- The maximum makeup flow rates assumed in the analysis are 50 gpm for power levels above 50 percent power and 25 gpm for power levels below 50 percent power. The letdown flow rates are assumed to be equal to the makeup flow rates assumed in the analysis.
- A minimum makeup temperature of 40 degrees F is assumed for the analysis of boron dilution of the RCS.
- The minimum RCS flow rates are assumed to increase loop transit time, which increases the timing for detection and isolation.
- The minimum possible reactor pool volume is used to provide a limiting time to loss of shutdown margin for Mode 5.

15.4.6.3.5 Results

The results for a boron dilution of the RCS during Mode 1 operation are presented in Table 15.4-14 for hot full power and Table 15.4-15 for HZP. The tabulated results for the hot full power scenario demonstrate that the reactivity insertion rates are bounded by the range of the reactivity insertion rates that are evaluated in the uncontrolled CRA withdrawal analysis, presented in Section 15.4.2. The tabulated results for the HZP scenario demonstrate that the reactivity insertion rates are bounded by the range of the reactivity insertion rates that are evaluated in the uncontrolled CRA withdrawal from a subcritical or low power or startup condition analysis, presented in Section 15.4.1.

The results for a boron dilution of the RCS during Mode 2 operation and Mode 3 operation are presented in Table 15.4-16 and Table 15.4-18, respectively. The tabulated results for the Mode 2 and Mode 3 scenarios demonstrate that shutdown margin is maintained at the time of DWS isolation.

The results for a boron dilution of the RCS during Mode 5 operation are presented in Table 15.4-19. Based on the comparison of the total dilution volume required to achieve criticality against the maximum flooding source volumes presented in Table 15.4-12, it is concluded that pool flooding as a result of pipe breaks and potential flooding sources is not limiting and can be accommodated by the initial reactivity condition of k_{eff} of 0.90 or less.

15.4.6.4 Radiological Consequences

The plant conditions after the limiting decrease in boron concentration cases during Mode 1 operation are bounded by the uncontrolled CRA withdrawal from a subcritical or low power or startup condition and uncontrolled CRA withdrawal analyses presented in Section 15.4.1 and Section 15.4.2. Based on the uncontrolled CRA withdrawal from a subcritical or low power or startup condition and uncontrolled CRA withdrawal results, no fuel failures are predicted and radionuclide barriers maintain integrity during a decrease in boron concentration event. The results for the non-power modes of operation show that shutdown margin is maintained for a decrease in boron concentration event. The normal leakage related radiological consequences of this event are bounded by the design basis accident analyses presented in Section 15.0.3.

15.4.6.5 Conclusions

The plant conditions after the limiting decrease in boron concentration cases during Mode 1 operation are bounded by the uncontrolled CRA withdrawal from a subcritical or low power or startup condition and uncontrolled CRA withdrawal analyses presented in Section 15.4.1 and Section 15.4.2. The results for the non-power modes of operation (Modes 2 and 3) show that shutdown margin is maintained for a decrease in boron concentration event. The results of flooding based dilutions during Mode 5 operation demonstrate that subcriticality is maintained following the most limiting flooding based dilution scenario. Based on these results, it is concluded that the SRP 15.4.6 acceptance criteria are met.

15.4.7 Inadvertent Loading and Operation of a Fuel Assembly in an Improper Position

15.4.7.1 Identification of Causes and Accident Description

An inadvertent loading and operation of a fuel assembly in an improper position is an event that could affect the power distribution and power peaking of the NuScale reactor core. If undetected, a fuel assembly loading error could lead to a reduced CHF ratio and reduced margin to fuel centerline melt.

Fuel loading controls are established to prevent a fuel assembly loading error. The fuel loading operation is conducted in accordance with detailed approved procedures. The fuel loading safety measures and procedures are discussed in Section 14.2.

An inadvertent loading and operation of a fuel assembly in an improper position is not expected to occur during the lifetime of the reactor, and is classified as an infrequent event.

15.4.7.2 Sequence of Events and Systems Operation

The core monitoring system detects a fuel loading error if it causes a relative power shape deviation higher than a detection threshold. The overpower fraction detection threshold is 1.44 and the underpower fraction detection threshold is 0.65. These fractions mean that if an assembly is 44 percent above its predicted power or 35 percent below its predicted power it will be detected by the core monitoring system. Some assembly misloads can be detected before these power fraction detection thresholds are reached. Fuel assembly manufacturing practices and quality assurance techniques ensure that assemblies containing un-prescribed enrichments or burnable poison loadings are not available on site for initial loading or reload of the core. Therefore, no misloads related to un-prescribed enrichments or burnable poisons loadings for a given core loading plan are considered. The entire spectrum of potential shuffle and rotational fuel assembly misloads are considered.

Shuffle Misloads

The spectrum of potential fuel assembly misloads for the NuScale reactor core is examined to assess the impacts of an undetectable fuel assembly misload being present during normal operations. Figure 15.4-28 shows the full spectrum of fuel misloads considered.

The fuel assembly misloads are evaluated in the three categories shown in Figure 15.4-28: quarter-core, half-core, and cross-core. Each of the assembly locations that are evaluated as misloads are numbered for each of the three categories. For the quarter-core misloads, assembly locations 1 through 13 could potentially be shuffled into any of the other numbered assembly locations, which results in a total of 78 potential fuel misloads to be considered on a quarter-core basis. For the half-core misloads, assembly locations 1 through 9 could potentially be shuffled into assembly locations 10 through 18, a total of 81 potential fuel misloads to be considered on a half-core basis. For the cross-core misloads, assembly locations 1 through 12 could potentially be shuffled into assembly locations 13 through 18, resulting a total of

72 potential fuel misloads to be considered on a half-core basis. There are 231 potential misloads that are analyzed.

The center assembly in the equilibrium cycle is a fresh assembly. Therefore, exchanges of the center assembly are only examined on a quarter-core basis because the faces of the center assembly do not have a different depletion history than each other and exchanges in the other quadrants will be consistent with those performed in a single quadrant.

Rotational Misloads

The fuel assembly top nozzle has two holes that mate with pins in the upper core plate, and a third alignment hole that mates with the fuel handling equipment (Section 4.2). These features collectively, prevent fuel assembly rotational misloads. Nevertheless, 180 degree rotational misloads are conservatively examined.

15.4.7.3 Core and System Performance

15.4.7.3.1 Evaluation Model

The design and analysis of the NuScale Power Module reactor core is performed with the Studsvik Scandpower Core Management Software suite of reactor simulation tools. A discussion of the analysis tools and analytical methods is provided in Section 4.3.3.

SIMULATE5 is an advanced three-dimensional (3D), steady-state, multi-group nodal reactor analysis code capable of multi-dimensional nuclear analyses of reactors. SIMULATE5 is used to determine the limiting undetectable fuel misload, and to provide peaking factors to the subchannel analysis.

The subchannel core CHF analysis is performed using VIPRE-01. VIPRE-01 is a subchannel analysis tool designed for general-purpose thermal-hydraulic analysis under normal operating conditions, operational transients, and events of moderate severity. See Section 15.0.2.3 and Section 4.4 for a discussion of the VIPRE-01 code and evaluation model.

15.4.7.3.2 Input Parameters and Initial Conditions

The fuel misload event changes the power distribution of the core, but the thermal hydraulic boundary conditions remain the same. Therefore, there is no need for an NRELAP5 analysis to ensure that the RCS pressure remains below the design limit of the RPV. The power distribution of the equilibrium core analysis is discussed in Section 4.3.

The power peaking augmentation factors for the limiting undetectable misload are calculated using SIMULATE5, and provided as input to the steady-state subchannel analysis to determine the MCHFR for this event. The limiting undetectable misload is a swap of two adjacent assemblies in the 10 and 13 locations as seen in the 'Quarter Core' portion of Figure 15.4-28. This limiting misload power peaking

augmentation factor is bounded by the analysis value of 1.25. Other key inputs and assumptions used in the subchannel analysis are provided in Reference 15.4-1.

15.4.7.3.3 Results

The limiting undetectable fuel misloading event results in an MCHFR which is above the 95/95 CHFR limit. Fuel temperature margin to centerline melt is calculated for the worst case fuel assembly misloading event. The calculated value of Linear Heat Generation Rate (LHGR) for the worst misload is below the limiting LHGR. These results are provided in Table 15.4-20. Because MCHFR is above the limit and fuel centerline melting is not expected to occur, no fuel damage is expected. These events change the power distribution within the core, not overall core power. Therefore, there is no power increase associated with the fuel misloading events that could challenge the radionuclide barriers.

15.4.7.4 Radiological Consequences

The normal leakage related radiological consequences of this event are bounded by the design basis accident analyses presented in Section 15.0.3.

15.4.7.5 Conclusions

The results from the evaluation of the limiting undetectable fuel misloading events show that no fuel damage is expected. There is no pressure transient associated with this event, so the RCS pressure boundary is not challenged. With no fuel damage and no challenge to radionuclide boundaries, the normal leakage related radiological consequences of this event are bounded by the design basis accident analyses in Section 15.0.3. Therefore, all SRP 15.4.7 acceptance criteria are met.

15.4.8 Spectrum of Rod Ejection Accidents

15.4.8.1 Identification of Causes and Accident Description

A postulated failure of the CRDM pressure housing could cause a control rod to be ejected from the core. The unexpected and rapid increase in positive reactivity demonstrates the effects of a limiting reactivity insertion event.

The power spike resulting from the CRA ejection is quickly countered by the fuel reactivity feedback as the fuel temperature begins to increase. The sudden increase in power is detected by the MPS, resulting in a reactor trip. The sudden ejection of a CRA adds positive reactivity to a localized region of the core in a very short period of time. This CRA ejection results in a power excursion in the region near the affected fuel assembly and results in a highly asymmetric power distribution in the radial dimension. This adverse power distribution subsequently leads to overheating of the affected fuel assemblies and possible fuel damage.

There is a low probability of a rod ejection accident occurring, and it is not expected to occur during the life of the plant. The REA is classified as a postulated accident.

15.4.8.2 Sequence of Events and Systems Operation

The sequence of events for the limiting REA case with respect to primary pressure is shown in Table 15.4-21.

Unless specified in this section, the analysis of an REA assumes the plant control systems and engineered safety features perform as designed, with allowances for instrument inaccuracy. No operator action is credited to mitigate the effects of an REA.

Pressure and inventory control is disabled for the maximum pressure REA cases. This ensures maximum pressure, which is conservative with respect to both peak RCS pressure and MCHFR.

The feedwater flow control would normally raise feedwater flow in response to increasing RCS temperature. In order to maximize peak RCS pressure and minimize MCHFR, the feedwater flow is held constant for the maximum pressure REA cases.

Steam pressure is held constant for the maximum pressure cases. Steam pressure controls would normally relieve steam pressure as reactor power increases. This would enhance heat transfer and lower both the RCS temperature and pressure. Therefore, steam pressure is held constant to maximize RCS pressure.

If the reactivity control system is in an automatic mode, the regulating CRAs could insert prior to a reactor trip in response to an REA. The regulating CRAs are not credited to mitigate the reactivity insertion of the REA prior to a reactor trip, to provide the most conservative power response.

The thermal hydraulic analysis of the REA considers the loss of power scenarios as follows:

- Loss of Normal AC - In this scenario, the MPS remains powered, so none of the safety systems are actuated automatically. However, power is lost to the feedwater pumps, CVCS recirculation pumps, pressurizer heaters, and the condenser, resulting in a turbine trip.
  - Loss of normal AC at the time of the event initiation is analyzed in NRELAP5.
  - Loss of normal AC at the time of reactor trip is analyzed in NRELAP5.
- Loss of EDNS and loss of normal AC - Power to the control rod drive mechanisms is provided via the nonsafety DC power distribution (EDNS), so this scenario is the same as discussed above, with addition of the CRAs dropping at the time at which power is lost. For this event, this scenario is non-limiting because of the immediate loss of power to the CRDMs, resulting in the drop of the CRAs.
- Loss of EDSS, EDNS and loss of normal AC - Power to the MPS is provided by the highly-reliable DC power distribution system (EDSS), so this scenario results in an actuation of RTS and all of the engineered safety features. This scenario is non-limiting because of the immediate reactor trip.

There is no single failure that will result in more severe conditions for the limiting REA cases.

15.4.8.2.1 Mechanical Design

The CRDM pressure housing is not designed to be an exterior feature, but an integral portion of the RPV with an extremely low probability of failure. The control rod drive housings are welded to nozzles that are integrally forged as part of the RPV head. The safe end to nozzle welds and safe end to control rod housing welds are inspected to American Society of Mechanical Engineers Class 1 requirements. The control rod drive system functional design is discussed in Section 4.6.

15.4.8.2.2 Effects on Other Control Rod Housings

The damaged control rod housing is postulated to provide a limiting reactivity insertion event. However, the mechanical failure of a control rod housing that would result in a missile is non-credible, as discussed in Section 3.5. Nevertheless, the reactivity effects of a neighboring CRDM housing being damaged are bounded by assuming a rod is stuck out for a reactor trip.

15.4.8.2.3 Nuclear Design

The NuScale design uses two reactivity control mechanisms: power regulating control rods and RCS boron concentration. The use of boron in the RCS to control reactivity limits the insertion of regulating control rods during power operation to a predefined PDIL. With the rods mostly withdrawn from the core, the amount of reactivity associated with an REA is limited.

The NuScale core is also designed with a negative DTC, which limits the magnitude of a power pulse associated with an REA. A discussion of the negative reactivity feedback design of the core is provided in Section 4.3.

15.4.8.2.4 Module Protection System

The MPS protects the NuScale plant by tripping the reactor in the event of an REA. The MPS reactor trip signals that provide protection during an REA are the following:

- high power
- high power rate
- high steam line superheat
- high pressurizer pressure
- high RCS hot temperature
- high steam line pressure

In the event of a reactor trip, the subsequent actuation of the DHRS would maintain reactor cooling.

15.4.8.3 Fuel, Thermal Hydraulic, and Subchannel Analyses

15.4.8.3.1 Evaluation Models

The fuel analyses for an REA are performed using SIMULATE-3K (S3K). S3K is an advanced, nodal code for transient analysis of both pressurized water reactors and boiling water reactors. S3K explicitly couples both the neutronics and the thermal-hydraulic calculations for each assembly in the core. A discussion of the S3K code can be found in Section 4.3.3. The S3K output provides the reactor power as a function of time to the downstream NRELAP5 analysis. The S3K output is also used for an adiabatic fuel calculation to determine if there is any fuel failures predicted for an REA.

The maximum fuel temperature is conservatively calculated using the following adiabatic equation:

$$\Delta T = \frac{E_T \cdot F_{q,\,max}}{C_p \cdot V_{node} \cdot n_{nodes}}$$

where,

ΔT = temperature increase

E_T = total energy created during accident

$F_{q,max}$ = maximum pin peaking factor before control rods move

C_p = fuel heat capacity

V_{node} = nodal volume

n_{nodes} = total number of nodes in the core

The peak radial average fuel enthalpy is determined using the following adiabatic equation:

$$h_i = \frac{Cp \cdot T_{f,\,max}}{\rho_f}$$

where,

h_i = maximum initial radial average fuel enthalpy

$T_{f,max}$ = maximum initial fuel centerline temperature

ρ_f = fuel density

The following equation defines the conservative radial average fuel enthalpy increase:

$$\Delta h = \frac{E_T \cdot F_{q,\,max}}{V_{node} \cdot \rho_f \cdot n_{nodes}}$$

where,

Δh = radial average fuel enthalpy increase

The thermal hydraulic analysis of the plant response to an REA is performed using NRELAP5. The NRELAP5 model is based on the design features of a NuScale module. The non-LOCA NRELAP5 model is discussed in Section 15.0.2. The relevant boundary conditions from the NRELAP5 analyses are provided to the downstream subchannel CHF analysis.

The subchannel core analysis is performed using VIPRE-01 to predict any fuel failure due to CHF cladding failure. VIPRE-01 is a subchannel analysis tool designed for general-purpose thermal-hydraulic analysis under normal operating conditions, operational transients, and events of moderate severity. See Section 15.0.2.3 for a discussion of the VIPRE-01 code and evaluation model.

15.4.8.3.2 Input Parameters and Initial Conditions

<u>S3K Model</u>

The power and reactivity data for an REA is calculated using S3K for the downstream thermal hydraulic and subchannel transient analyses. The power response calculated by S3K is also used to determine the temperature and enthalpy responses of the fuel. The subchannel analysis and the temperature and enthalpy responses of the fuel indicate if there is any fuel failure during an REA. The inputs, initial conditions, and conservatisms of the S3K REA model are discussed in this section.

In order to maximize the possible reactivity insertion from an ejected rod, the non-shutdown CRAs are assumed to be at the PDIL with an uncertainty of 6 steps. The shutdown bank is positioned all rods out. Conservative scram characteristics are applied to the REA model including:

- highest worth CRA (other than the ejected rod) remains stuck out of the core
- reactor trip delay of 2 seconds
- maximum CRA drop time after scram

Conservative core characteristics are applied to the REA model to ensure the maximum reactivity insertion with minimum feedback. A top peaked power shape is applied to the REA model to maximize the effects of the ejected rod. The uncertainty values of the DTC and MTC in the REA analysis are applied to the S3K values in the conservative (less negative) direction to minimize the fuel feedback

effects that could mitigate the power response of an REA. For a discussion on specific core parameter values and the associated biases used in the REA methodology, see Reference 15.4-2.

The S3K analysis provides REA power response calculations at the following power levels and times in cycle:

- Power (%) - 0, 25, 50, 70, 80, 100
- Time in life - BOC, MOC (4 GWD/T), EOC

The S3K fuel response data is provided as input to an adiabatic fuel response calculation to determine if the fuel enthalpy and fuel centerline temperature remain below the SAFDLs. The adiabatic calculation is conservative because it assumes all of the energy generated during the transient is deposited into the fuel. This maximizes enthalpy and temperature increases.

NRELAP5 Model

The NRELAP5 thermal hydraulic analysis utilizes the power response calculated by S3K to simulate the power pulse associated with an REA. The NRELAP5 analysis evaluates the limiting pressure response to an REA. The inputs, initial conditions, and conservatisms of the NRELAP5 REA model are discussed in this section.

The power functions generated by the S3K analysis maximize the power pulse to create limiting conditions to evaluate the SAFDLs. However, to find the maximum pressure response to an REA, the NRELAP5 analysis evaluates an REA that results in the maximum power pulse that does not result in a reactor trip. Reference 15.4-2 discusses key inputs and the associated biases for the NRELAP5 REA model. Several of the inputs and biases are discussed below:

- Initial power of 102 percent HFP - This represents full power with an additional 2 percent calorimetric error.
- Ejected rod simulated by increasing the power to 117 percent - This power peak avoids the 120 percent high power trip and the 15 percent/min high power rate trip.
- Least negative reactivity coefficients minimize reactivity feedback - MTC of 0.0 pcm/degrees F and DTC of -1.40 pcm/degrees F
- Direct moderator heating fraction of 0.025 - Direct moderator heating maximizes heat deposition to coolant during REA power peak.
- The initial pressure is biased low and the RCS temperature is biased high. These conditions result in a conservative evaluation of the maximum pressure because the conditions maximize the liquid expansion coefficient.
- The pressurizer level is biased high to minimize the steam space to produce the maximum pressure.
- The steam pressure is biased high to delay the high steam superheat reactor trip.

The NRELAP5 analysis also evaluates the thermal hydraulic conditions for the limiting CHF conditions. These cases provide boundary conditions to the downstream subchannel analysis to evaluate CHF. In order to minimize the MCHFR, the NRELAP5 model uses the power functions generated by the S3K analysis, which provides the maximum power pulse at various statepoints. The NRELAP5 case that results in the most limiting MCHFR conditions is initialized with the following inputs:

- Initial power of 70 percent HFP - This initial power results in one of the higher S3K power responses.
- EOC conditions - This time in cycle results in one of the higher S3K power responses.
- Average RCS temperature biased high - The higher temperature corresponds to a higher coefficient of expansion. This exacerbates the REA-induced core pressure pulse and inlet flow slow-down, minimizing MCHFR.
- RCS flow biased low - The lower core flow minimizes MCHFR.
- Fuel and gap conductivities are maximized - Maximizing the conductivities increases the energy flow into the coolant, which maximizes the inlet flow slow-down.

The results from the thermal hydraulic evaluation are used as input to the subchannel analysis to determine if the MCHFR design limit is met for this event. Other key inputs and assumptions used in the subchannel analysis are provided in Reference 15.4-1. The results of the subchannel analysis and adiabatic fuel energy calculation determine if there is any potential fuel damage resulting from an REA. The REA event-specific methodology is provided in Reference 15.4-2.

15.4.8.3.3 Regulatory Criteria for NuScale

Reference 15.4-2 discusses the various REA regulatory acceptance criteria and how they apply to the NuScale design. A summary of these acceptance criteria are provided in this section.

<u>Fuel Cladding Failure</u>

- For zero power conditions, the high temperature cladding failure threshold is expressed in cladding differential pressure. The peak radial average fuel enthalpy must be below 100 cal/g. For NuScale, the 100 cal/g limit is applied at all peak rod differential pressures.
- For intermediate and full power conditions, fuel cladding failure is presumed if local heat flux exceeds the critical heat flux (CHF) thermal design limit.
- The PCMI failure limit is a change in radial average fuel enthalpy of 75 cal/g, based on the corrosion-dependent limit depicted in Figure 5-2 of Reference 15.0-11.
- If fuel temperature anywhere in the pellet exceeds incipient fuel melting conditions, then fuel cladding failure is presumed.

Core Coolability

- Peak radial average fuel enthalpy shall remain below 230 cal/g.
- No fuel melt shall occur.
- For intermediate and full power conditions, local heat flux shall not exceed the critical heat flux (CHF) thermal design limit.

RCS Pressure

The maximum RCS pressure must remain below 120 percent of design pressure. Therefore, the peak pressure during an REA is limited below 2520 psia.

15.4.8.3.4 Fuel and Cladding Integrity Results

S3K provides the power and reactivity response to an REA for each statepoint discussed in Section 15.4.8.3.2. Each initial power level is evaluated for BOC, MOC (4.0 GWD/T), and EOC conditions. The S3K analysis assumes the maximum reactivity insertion from ejecting the highest worth CRA for each of these statepoints. Table 15.4-22 provides the maximum power in percent HFP as well as the inserted reactivity of the ejected rod for a spectrum of initial power levels and times in cycle. Figure 15.4-29, Figure 15.4-30, and Figure 15.4-31 provide the maximum power pulse for EOC, MOC, and BOC conditions, respectively. The plots show a rapid reactivity excursion when the rod is ejected, but the power pulse is mitigated, due to fuel feedback effects. The reactor is tripped when the power reaches 15 percent above the initial power level, and the rods are inserted after a 2.0 second delay. At BOC and 4.0 GWD/T, the peak power pulse occurred at 70 percent power with the ejection of a rod from the inner bank of control rods. At EOC, the initial 25 percent power case provided the largest power pulse from an ejection from the inner bank of control rods. Sensitivities of power cases around the EOC, 25 percent case affirmed that the 25 percent power case at EOC produced the largest power pulse at 524 percent of HFP.

An adiabatic fuel response calculation evaluates these power responses to determine if any fuel failures occur due to the fuel temperature or enthalpy increase. Table 15.4-23 provides the peak fuel temperature, change in fuel enthalpy, and net fuel enthalpy for a spectrum of initial power levels and times in cycle. A summary of the limiting conditions below is in Table 15.4-24:

- The limiting peak radial average fuel enthalpy at zero power conditions
- The limiting peak radial average fuel enthalpy for intermediate and full power conditions
- The limiting change in radial average fuel enthalpy
- The limiting peak fuel temperature

The subchannel analysis evaluates a spectrum of REA conditions that are provided by the S3K and NRELAP5 analyses. The REA case that results in the limiting MCHFR is an REA that occurs at an initial power of 70 percent at EOC. The limiting MCHFR is above the design limit as demonstrated in Figure 15.4-37.

The maximum possible mass and energy release to containment due to a postulated control rod housing failure is bounded by an inadvertent opening of an RVV. A postulated control rod housing failure would represent a maximum break size of 2.375 inch (inner diameter). This break size is smaller than the opening of an RVV. The additional energy from the power excursion of an REA is not sufficient to exceed the energy release of an inadvertent opening of an RVV. The inadvertent opening of an RVV is discussed in Section 15.6.6. The limiting containment peak pressures and temperatures for design basis events are discussed in Section 6.2.

The REA acceptance criteria for fuel cladding failure and core coolability are met by the NuScale design, which indicates that no fuel failures are predicted in the event of an REA.

15.4.8.3.5 Maximum Reactor Coolant System Pressure Results

The sequence of events provided in Table 15.4-21 is for the REA that results in the limiting RCS pressure. Figure 15.4-33 shows the pressure for this REA case. The pressure rises until it peaks a few seconds after the reactor trip. The maximum pressure shown in Table 15.4-24 is below the safety valve opening limit and well below 120 percent of the RPV design pressure. Note that Figure 15.4-33 includes a ten-second steady state time period at the beginning of the plot, while the Table 15.4-21 sequence of events starts the event at zero seconds.

15.4.8.4 Radiological Consequences

No fuel damage is predicted for the limiting REA. Therefore, the radiological consequences of a REA are bounded by the consequences of other accidents presented in Section 15.0.3.

15.4.8.5 Conclusions

The applicable acceptance criteria for this accident are met for the limiting REA cases. These acceptance criteria are provided in Section 15.4.8.3.3. The NuScale Power Plant design meets these criteria as discussed in the summary below.

<u>Fuel Cladding Failure</u>

- The limiting peak radial average fuel enthalpy at zero power conditions is below the fuel cladding failure limit, as shown in Table 15.4-24.
- For intermediate and full power conditions, the limiting MCHFR is above the design limit, as shown in Table 15.4-24.
- The limiting change in radial average fuel enthalpy is below 75 cal/g (A conservative value on Figure B-1 of NUREG-0800 SRP 4.2).
- The limiting peak fuel temperature is below the fuel melting temperature, as shown in Table 15.4-24.

These limiting fuel cladding results indicate no fuel failures.

Core Coolability

- The limiting peak radial average fuel enthalpy is below the limit, as shown in Table 15.4-24.
- The limiting peak fuel temperature is below the fuel melting temperature, as shown in Table 15.4-24.

These limiting results indicate that core coolability will be maintained for the limiting REA.

RCS Pressure

- The limiting RCS peak pressure is below the RPV design limit, as shown in Table 15.4-24.

This limiting result indicates that RPV integrity is maintained during the limiting REA.

15.4.9 Spectrum of Rod Drop Accidents

15.4.9.1 Identification of Causes and Accident Description

This event is specific to boiling water reactors and not applicable to the NuScale design. The pressurized water reactor equivalent of a rod drop, the rod ejection, is addressed in Section 15.4.8. Control rod mioperations, including a dropped control rod assembly, are addressed in Section 15.4.3.

15.4.10 References

15.4-1 NuScale Power LLC, "Subchannel Analysis Methodology," TR-0915-17564-P-A, Rev. 2.

15.4-2 NuScale Power LLC, "Rod Ejection Methodology," TR-0716-50350, Rev. 0.

Table 15.4-1: Sequence of Events for Limiting MCHFR Case (15.4.1 Uncontrolled CRA Withdrawal from Subcritical or Low Power Condition)

Event	Time [s]
Rod withdrawal initiates	0
High power (25%) limit reached	446
Reactor trip actuation	448
Maximum power	449
MCHFR	449
Scram complete	450

Table 15.4-2: Key Inputs for Limiting Centerline Fuel Temperature, MCHFR Case (15.4.1 Uncontrolled CRA Withdrawal from Subcritical or Low Power Condition)

Parameter	Nominal	Bias
Initial power	24 MW	N/A[1]
Initial RCS flow rate	553 lbm/s	Low[2]
Pressurizer pressure	1850 psia	Nominal
Pressurizer level	57%	Nominal
RCS Average temperature	545 °F	Nominal
MTC	+6 pcm/°F	Most positive
FTC	-1.40 pcm/°F	Least negative

[1]A spectrum of initial powers is analyzed, and this value provided the limiting MCHFR results.

[2]The initial RCS flow rate varies as a function of the initial power. RCS flow is minimized by applying a high bias to RCS form losses.

Table 15.4-3: Uncontrolled Control Rod Assembly Withdrawal from a Subcritical or Low Power or Startup Condition (15.4.1) - Limiting Analysis Results

Acceptance Criteria	Limit	Analysis Value
MCHFR	1.284	10.0
Maximum fuel centerline temperature	4816 °F	1051.8 °F

Table 15.4-4: Sequence of Events MCHFR Case - 75% Power (15.4.2 Uncontrolled Control Rod Assembly Withdrawal at Power)

Event	Time [s]
CRA bank begins to withdraw	0
High RCS hot temperature limit reached	144
High pressurizer pressure limit reached	150
Reactor trip actuated	152
MCHFR occurs	151
Maximum RCS pressure occurs	154
DHRS valves fully open	182

Table 15.4-5: Key Inputs for Limiting MCHFR Case (15.4.2 Uncontrolled CRA Withdrawal at Power)

Parameter	Nominal	Bias
Initial power	160 MW	Analyzed 75%
RCS Flowrate	See Table 15.0-6 for range	1039 lbm/s (low[1])
RCS Pressure	1850 psia	-70 psia
Pressurizer Level	60%	-8%
MTC	0.0 pcm/°F	Most positive
FTC	-1.377 pcm/°F	Least negative

[1] RCS flow rate is near the minimum for 75% power.

Table 15.4-6: Uncontrolled Control Rod Assembly Withdrawal at Power (15.4.2) - Limiting Analysis Results

Acceptance Criteria	Limit	Analysis Value
MCHFR (75% power)	1.284	1.499
Peak LHGR	19.7 kW/ft	9.16 kW/ft

Table 15.4-7: Sequence of Events (15.4.3 Control Rod Misoperation, Single Control Rod Assembly Withdrawal)

Event	Time [s]
Single CRA begins to withdraw	0
High hot leg temperature limit reached	125
High RCS pressure limit reached	130
Reactor trip actuation	132
Lowest MCHFR	132
Maximum RCS pressure occurs	134
DHRS valves fully open	162

Table 15.4-8: Key Inputs for Single CRA Withdrawal with Limiting MCHFR

Parameter	Normal	Bias
Initial power	75% of full power	Nominal
RCS flowrate	See Table 15.0-6 for range	1036 lbm/s(low[1])
Pressurizer level	60%	-8%
RCS pressure	1850 psia	-70 psia
RCS average temperature	545 °F	+1.5 °F
MTC	0 pcm/°F	Least Negative
DTC	-1.4 pcm/°F	Least Negative

[1] RCS flow rate is near the minimum for 75% power.

Table 15.4-9: Sequence of Events (15.4.3 Control Rod Misoperation, Control Rod Assembly Drop)

Event	Time [s]
CRA begins to drop	0
High power rate change limit reached	1
Reactor trip actuation	3
High steam line 1 and 2 superheat limit reached	4
High steam line 1 and 2 pressure limit reached	11
DHRS actuation valves open	43

Table 15.4-10: Key Inputs for CRA Drop with Limiting MCHFR and LHGR

Parameter	Nominal	Bias
Initial power	160 MW	+2%
RCS flow rate	See Table 15.0-6 for range	1166 lbm/s (low)
RCS pressure	1850 psia	+70 psia
RCS average temperature	545 °F	+10 °F
MTC	-43.0 pcm/°F	Most Negative
DTC	-2.5 pcm/° F	Most Negative

Table 15.4-11: Control Rod Misoperation (15.4.3) - Limiting Analysis Results

Acceptance Criteria	Limit	Analysis Value
MCHFR CRA misalignment	1.284	1.437
MCHFR Single CRA withdrawal	1.284	1.375
MCHFR CRA drop	1.284	1.432
Peak LHGR CRA misalignment	19.7 kW/ft	8.39 kW/ft
Peak LHGR Single CRA withdrawal	19.7 kW/ft	8.29 kW/ft
Peak LHGR CRA drop	19.7 kW/ft	6.71 kW/ft

Table 15.4-12: Internal Flooding Sources (15.4.6 Inadvertent Decrease in Boron Concentration in the Reactor Coolant System)

Break Source	Volume [gal]
Fire suppression header	100,000
Automatic fire suppression water	54,000
Site cooling water header piping	200,000
Site cooling water heating ventilation and air conditioning support piping	40,000
Demin/utility water piping	12,000
Main steam piping	77,000
Feedwater piping	24,000

Table 15.4-13: Bounding Critical Boron Concentrations and Boron Reactivity Coefficients (15.4.6 Inadvertent Decrease in Boron Concentration in the Reactor Coolant System)

Operation Mode	Critical Boron Concentration (ppm)	Boron Reactivity Coefficient (pcm/ppm)
Mode 1, ≥50 percent power	1600	-10
Mode 1, < 50 percent power	1800	-10
Mode 2	800	-11
Mode 3	1100	-12.5
Mode 4 and 5	1800	-11.5

Table 15.4-14: Mode 1, Hot Full Power Results (15.4.6 Inadvertent Decrease in Boron Concentration in the Reactor Coolant System)

Dilution rate (gpm)	5	25 (1 pump)	50 (2 pumps)
Reactivity insertion rate (complete mixing model) pcm/second	0.11	0.56	1.11
Initial reactivity insertion rate (wave front model) pcm/second	3.44	17.18	34.35
Range of reactivity insertion rates analyzed in uncontrolled control rod assembly withdrawal (Section 15.4.2) pcm/sec(1)	0.05 to 35		
Time to Loss of SDM - Complete Mixing Model (Minutes)	305.2	61.0	30.5

[1] Reactivity insertion rates from all dilution rates are bounded by the range of reactivity insertion rates assumed in uncontrolled control rod assembly withdrawal at power condition (Section 15.4.2).

Table 15.4-15: Mode 1, Hot Zero Power Results (15.4.6 Inadvertent Decrease in Boron Concentration in the Reactor Coolant System)

<table>
<tr><td>Dilution rate (gpm)</td><td>5</td><td>25
(1 pump)</td><td>50
(2 pumps)</td></tr>
<tr><td>Initial reactivity step (pcm)</td><td>141.29</td><td>684.95</td><td rowspan="7">N/A[1]</td></tr>
<tr><td>Initial reactivity insertion rate (Wave Front model) pcm/second</td><td>3.51</td><td>17.57</td></tr>
<tr><td>Reactivity insertion rate (Complete Mixing Model) pcm/second</td><td>0.1149</td><td>0.5744</td></tr>
<tr><td>Duration of the reactivity insertion rate for each wave (seconds)</td><td>40.2</td><td>38.99</td></tr>
<tr><td>Range of reactivity insertion rates assumed in Section 15.4.1 (pcm/sec)[2]</td><td colspan="2">0.005 to 35</td></tr>
<tr><td>Reactor Trip/DWS Isolation Time From Initiation of Dilution (minutes)</td><td>36.17</td><td>35.05</td></tr>
<tr><td>Shutdown Margin Remaining (Dilution Front Model) at the time of RX Trip (pcm)</td><td>1,772</td><td>771</td></tr>
</table>

[1]Two pump operation not allowed for power levels less than 50% power.

[2]Reactivity insertion rates from all dilution rates are bounded by the range of reactivity insertion rates assumed in uncontrolled control rod assembly withdrawal from a subcritical or low power startup condition (Section 15.4.1).

Table 15.4-16: Mode 2 Results (15.4.6 Inadvertent Decrease in Boron Concentration in the Reactor Coolant System)

Dilution rate[1] (gpm)	5	25 (1 pump)	50 (2 pumps)
Initial wave reactivity step (pcm)	128.2	621.3	N/A
Time to loss of shutdown margin, minutes	1523	297	
Time of DWS isolation, (minutes)	1349	222	
Shutdown margin remaining at DWS isolation trip (pcm)	604.5	1450.0	

[1]Two pump operation (50 gpm) is prohibited below 50% power.

Table 15.4-17: Not Used

Table 15.4-18: Mode 3 Results (15.4.6 Inadvertent Decrease in Boron Concentration in the Reactor Coolant System)

Dilution rate[1] (gpm)	5	25 (1 pump)	50 (2 pumps)[1]
Initial wave reactivity step (pcm)	162.9	790.3	N/A
Time to loss of shutdown margin, minutes	1098	222	
Time of DWS isolation, (minutes)	944	166	
Shutdown margin remaining at DWS isolation trip (pcm)	850	1384.3	

[1]Two pump operation (50 gpm) is prohibited below 50% power.

**Table 15.4-19: Mode 5 Results
(15.4.6 Inadvertent Decrease in Boron Concentration in the Reactor Coolant System)**

Assumed initial mixing volume (ft^3)	143,652
Time to loss of shutdown margin (minutes)[1]	445
Time to loss of shutdown margin (hours)[1]	7.41
Total dilution volume to reduce shutdown margin to zero (gallons)	444,650

Note 1: A range of flooding sources were evaluated.

Table 15.4-20: Inadvertent Loading and Operation of a Fuel Assembly in an Improper Position (15.4.7) - Limiting Analysis Results

Acceptance Criteria	Limit	Analysis Value
MCHFR	1.284	1.437
Peak LHGR	19.7 kW/ft	8.39 kW/ft

Table 15.4-21: Sequence of Events - Maximum Reactor Coolant System Pressure Case (15.4.8 Rod Ejection Accident)

Event	Time [s]
Rod ejection begins	0
High RCS pressure limit reached	9
Reactor trip actuates on high RCS pressure signal	11
Scram complete	13
Maximum RCS pressure reached	13

Table 15.4-22: REA Maximum Power Pulses and Reactivity Insertions

Time in Cycle	Initial Power (%)	Maximum Power (%)	Maximum Reactivity Insertion ($)
BOC	100	112	0.114
BOC	80	131	0.400
BOC	70	164	0.584
BOC	50	123	0.599
BOC	25	76	0.657
BOC	0	7	0.547
MOC	100	116	0.143
MOC	80	148	0.475
MOC	70	205	0.670
MOC	50	158	0.691
MOC	25	103	0.756
MOC	0	17	0.670
EOC	100	125	0.211
EOC	80	228	0.670
EOC	70	515	0.905
EOC	50	460	0.930
EOC	25	524	1.008
EOC	0	63	1.032

Table 15.4-23: REA Fuel Temperatures and Enthalpies for Limiting S3K Cases

Time in Cycle[1]	Initial Power (%)	Peak Fuel Temperature (°F)	Delta Radially Averaged Fuel Enthalpy (cal/g)	Net Radially Averaged Fuel Enthalpy
BOC	25	1637	18.3	57.4
BOC	50	2028	24.6	71.2
BOC	70	2288	27.8	80.3
BOC	80	2346	26.8	82.2
MOC	25	1650	18.7	57.8
MOC	50	2009	24.0	70.5
MOC	70	2280	27.5	80.0
MOC	80	2312	25.5	81.0
EOC	0	1399	17.5	49.1
EOC	25	1694	20.3	59.5
EOC	50	1996	23.5	70.0
EOC	70	2246	26.3	78.8
EOC	80	2300	25.1	80.6
EOC	100	2287	18.3	79.7

[1] The results for the HZP and full power cases for BOC and MOC conditions are covered by the analysis, but the magnitudes of these power pulses are not high enough to trip the reactor and are non-limiting, relative to the other cases.

Table 15.4-24: Spectrum of Rod Ejection Accidents (15.4.8) - Limiting Analysis Results

Acceptance Criteria	Limit	Analysis Value
Peak radial average fuel enthalpy at zero power	100 cal/g	49.1 cal/g
Change in radial average fuel enthalpy	75 cal/g	27.8 cal/g
Peak radial average fuel enthalpy	230 cal/g	82.2 cal/g
Maximum RCS pressure	2520 psia	2160 psia
Peak fuel temperature	4791 °F	2345 °F
MCHFR	1.284	1.838

Table 15.4-25: Not Used

Table 15.4-26: Not Used

Table 15.4-27: Not Used

Table 15.4-28: Not Used

Table 15.4-29: Not Used

Table 15.4-30: Not Used

Table 15.4-31: Key Inputs for Limiting Linear Heat Generation Rate Case (15.4.2 Uncontrolled Control Rod Assembly Withdrawal at Power)

Parameter	Nominal	Bias
Initial power	160 MW	+2%
RCS flowrate	See Table 15.0-6 for range	1168.0 lbm/s (low[1])
RCS pressure	1850 psia	Nominal
Pressurizer level	60%	Nominal
MTC	0.0 pcm/°F	Most Positive
FTC	-1.377 pcm/°F	Least Negative

[1] RCS flow rate is near the minimum for 102% power, and conservatively below the nominal range for 100% power.

Table 15.4-32: Key Inputs for Limiting Linear Heat Generation Rate Case (15.4.3 Control Rod Misoperation, Single Control Rod Assembly Withdrawal)

Parameter	Nominal	Bias
Initial power	75% rated power	Nominal
RCS flowrate	See Table 15.0-6 for range	1037 lbm/s (low[1])
RCS pressure	1850 psia	-70 psia
Pressurizer level	60%	-8%
MTC	0 pcm/°F	Most Positive
FTC	-1.40 pcm/°F	Least Negative

[1] RCS flow rate is near the minimum for 75% power.

Table 15.4-33: Key Inputs for Limiting Control Rod Assembly Misalignment (15.4.3 Control Rod Misoperation, Control Rod Assembly Misalignment)

Parameter	Nominal	Bias
Initial power	160 MW	+2%
RCS flowrate	See Table 15.0-6 for range	1180 lbm/s (low)
RCS pressure	1850 psia	+70 psia
Core inlet temperature	510 °F (high)[1]	Nominal

[1] This nominal core inlet temperature corresponds to the biased RCS average temperature of 550 °F and biased low RCS flowrate.

Figure 15.4-1: Reactor Power (15.4.1 Uncontrolled Control Rod Assembly Withdrawal from a Subcritical or Low Power Condition)

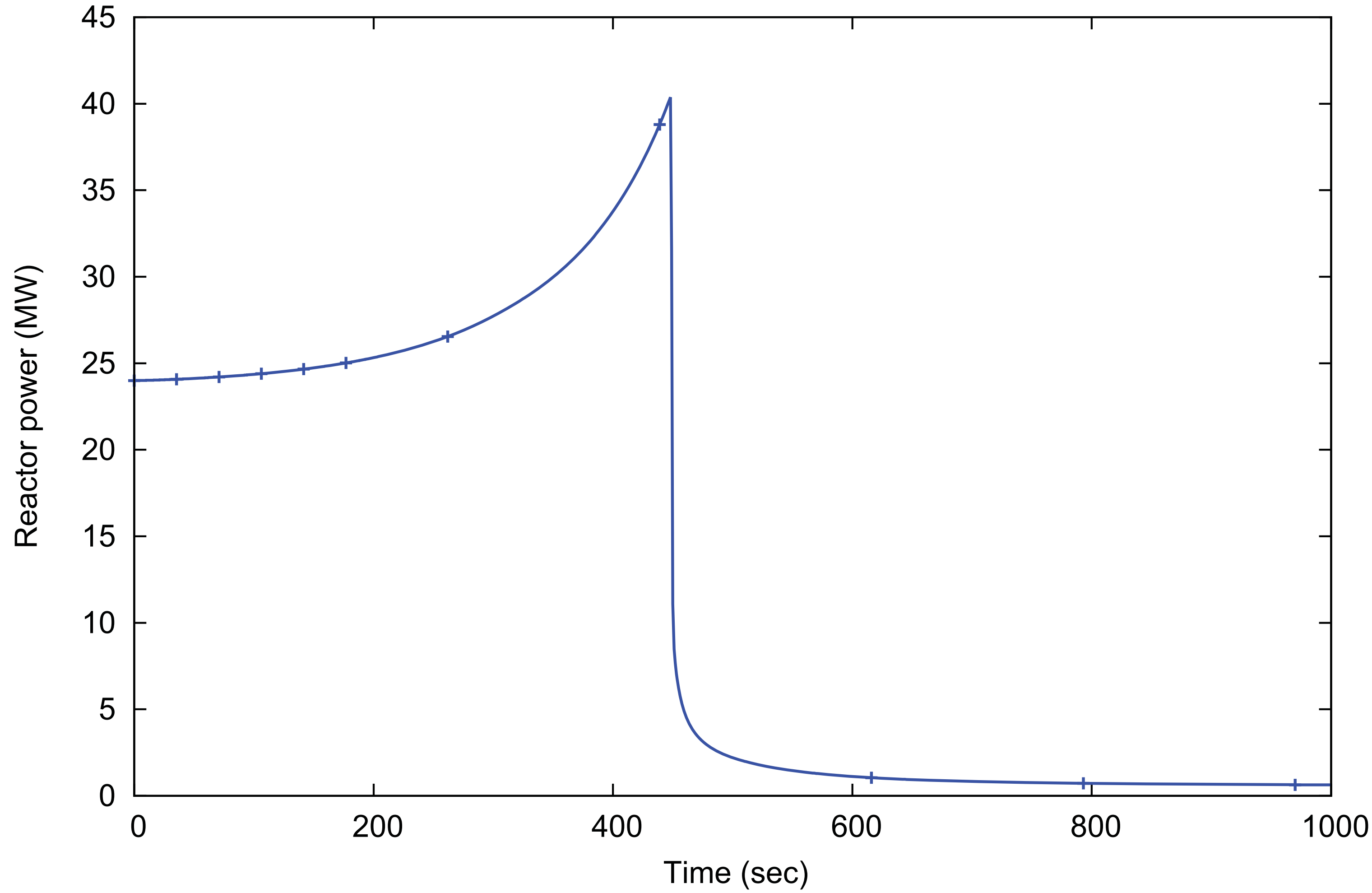


Figure 15.4-2: Net Reactivity (15.4.1 Uncontrolled Control Rod Assembly Withdrawal from a Subcritical or Low Power Condition)

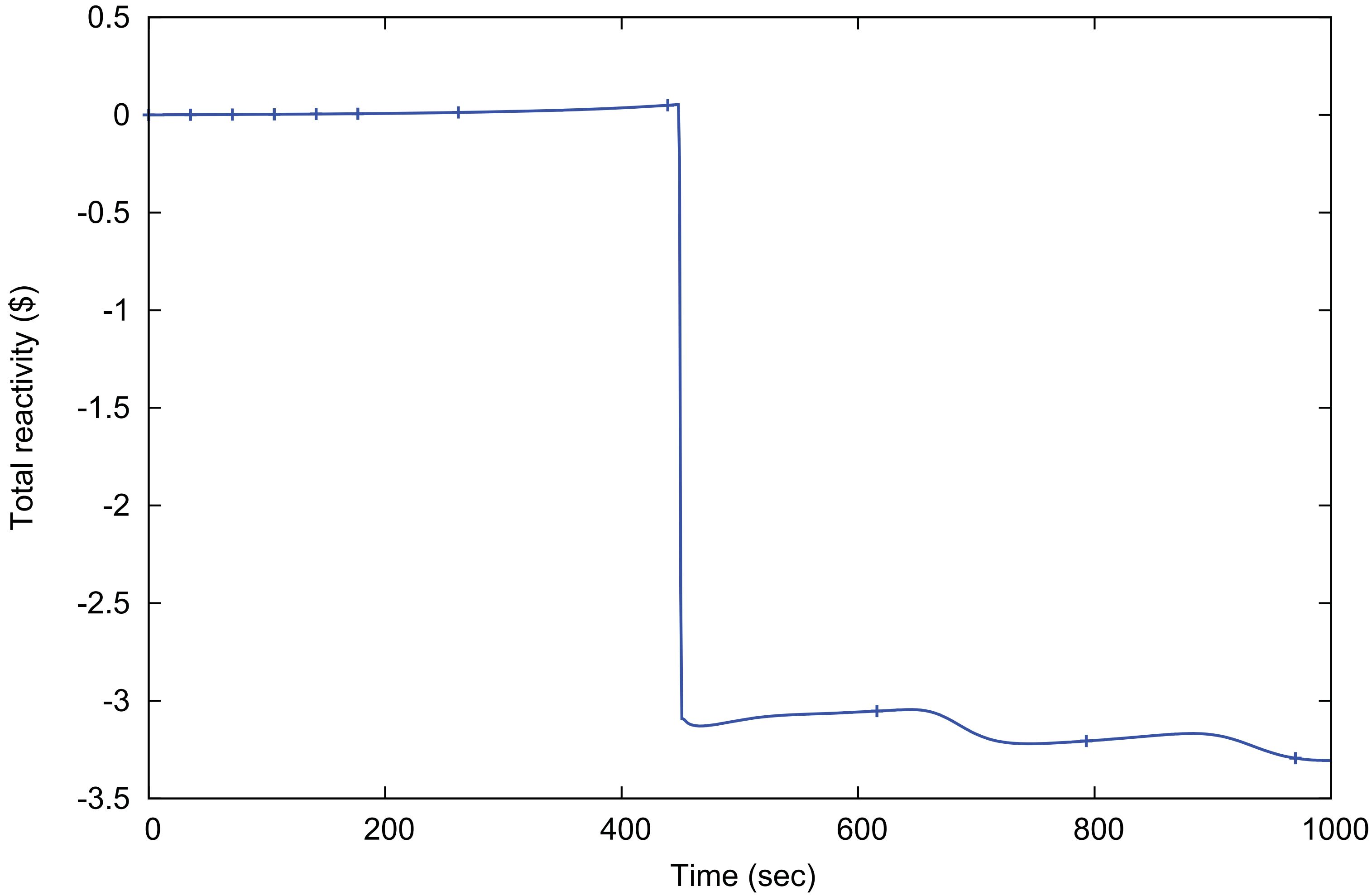


Figure 15.4-3: Core Outlet Temperature (15.4.1 Uncontrolled Control Rod Assembly Withdrawal from a Subcritical or Low Power Condition)

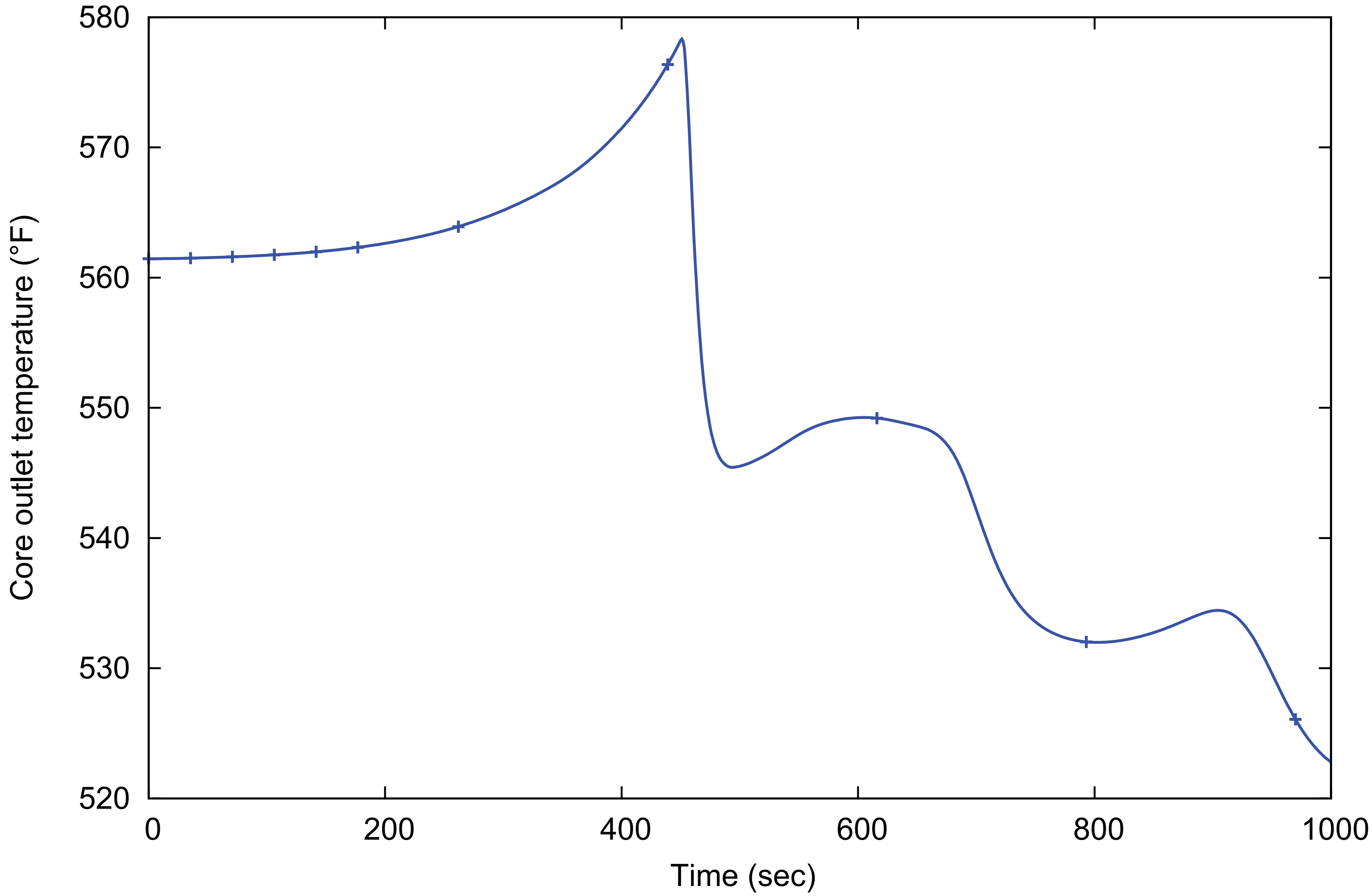


Figure 15.4-4: Primary Pressure
(15.4.1 Uncontrolled Control Rod Assembly Withdrawal from a Subcritical or Low Power Condition)

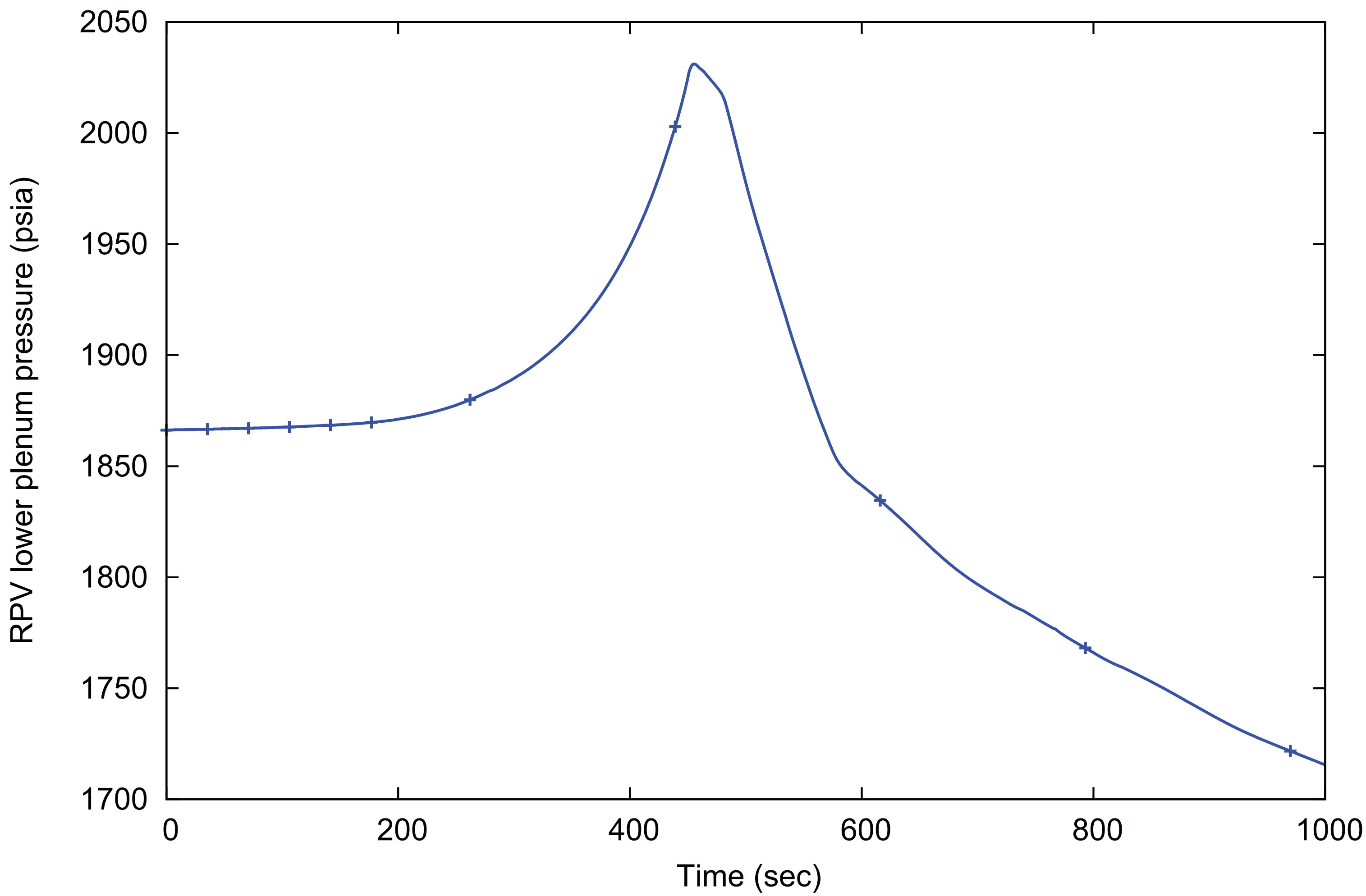


Figure 15.4-5: Not Used

Figure 15.4-6: Withdrawn CRA Reactivity Insertion
(15.4.2 Uncontrolled Control Rod Assembly Withdrawal at Power)

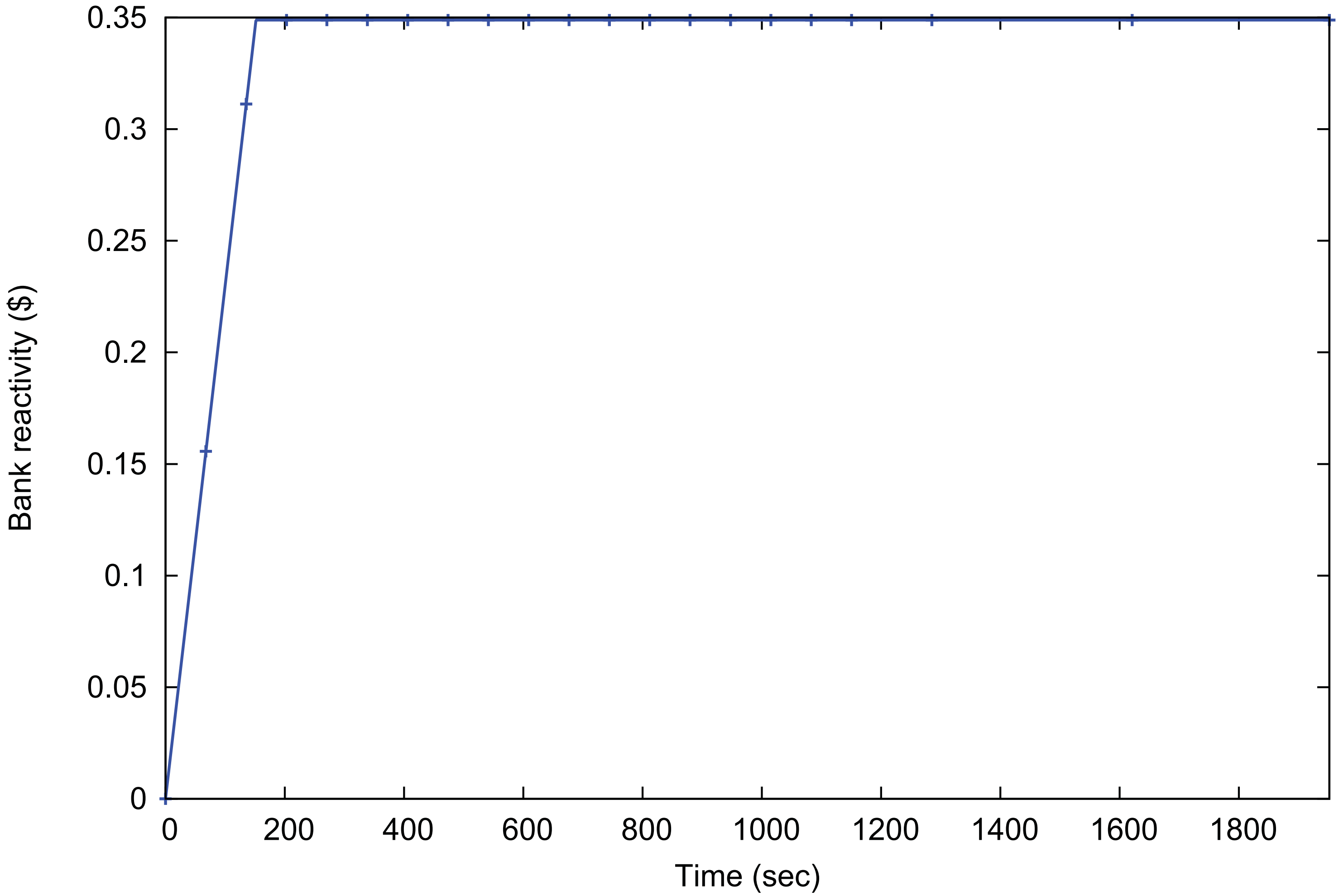


Figure 15.4-7: Reactor Power
(15.4.2 Uncontrolled Control Rod Assembly Withdrawal at Power)

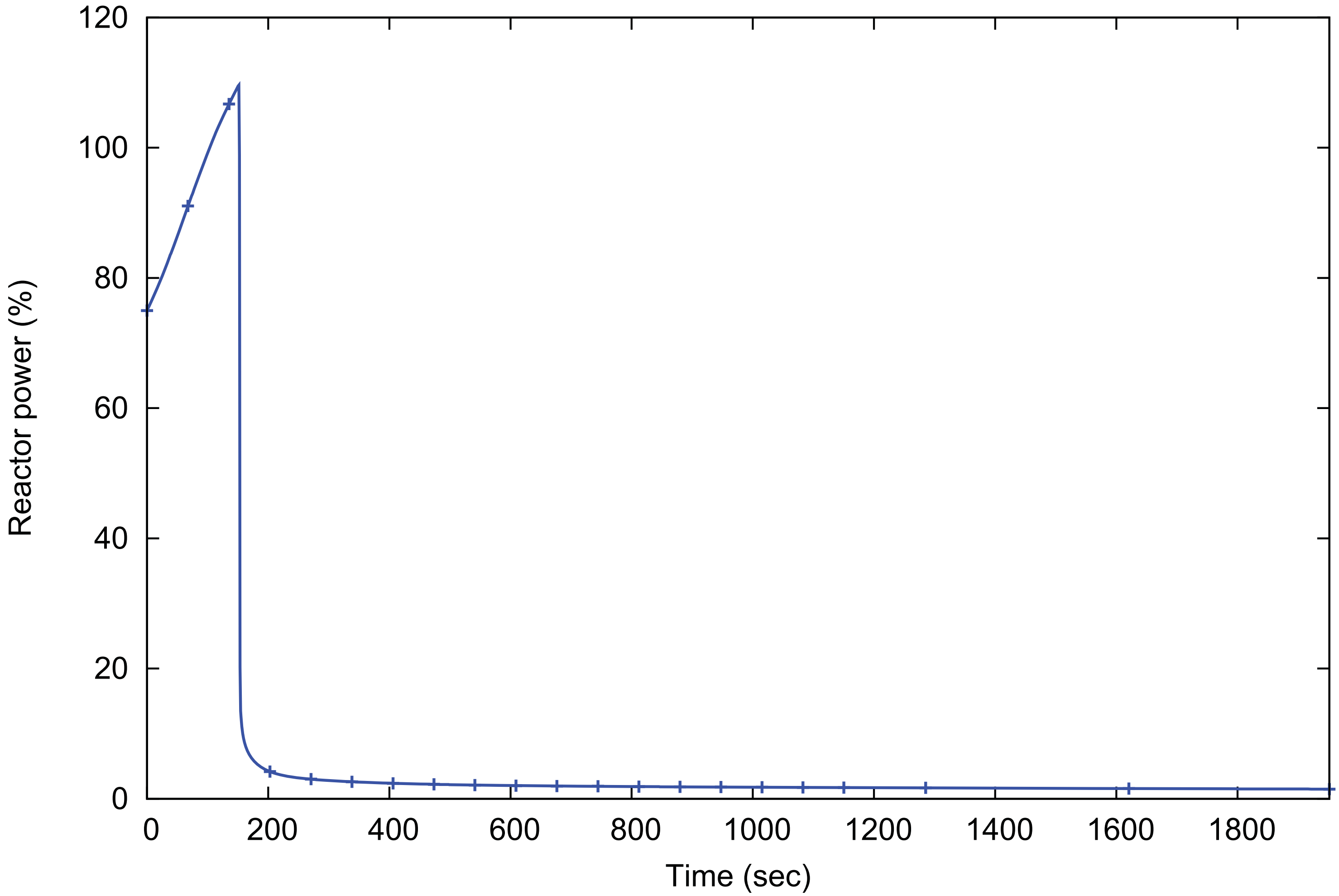


Figure 15.4-8: RCS Pressure for Limiting MCHFR Case
(15.4.2 Uncontrolled Control Rod Assembly Withdrawal at Power)

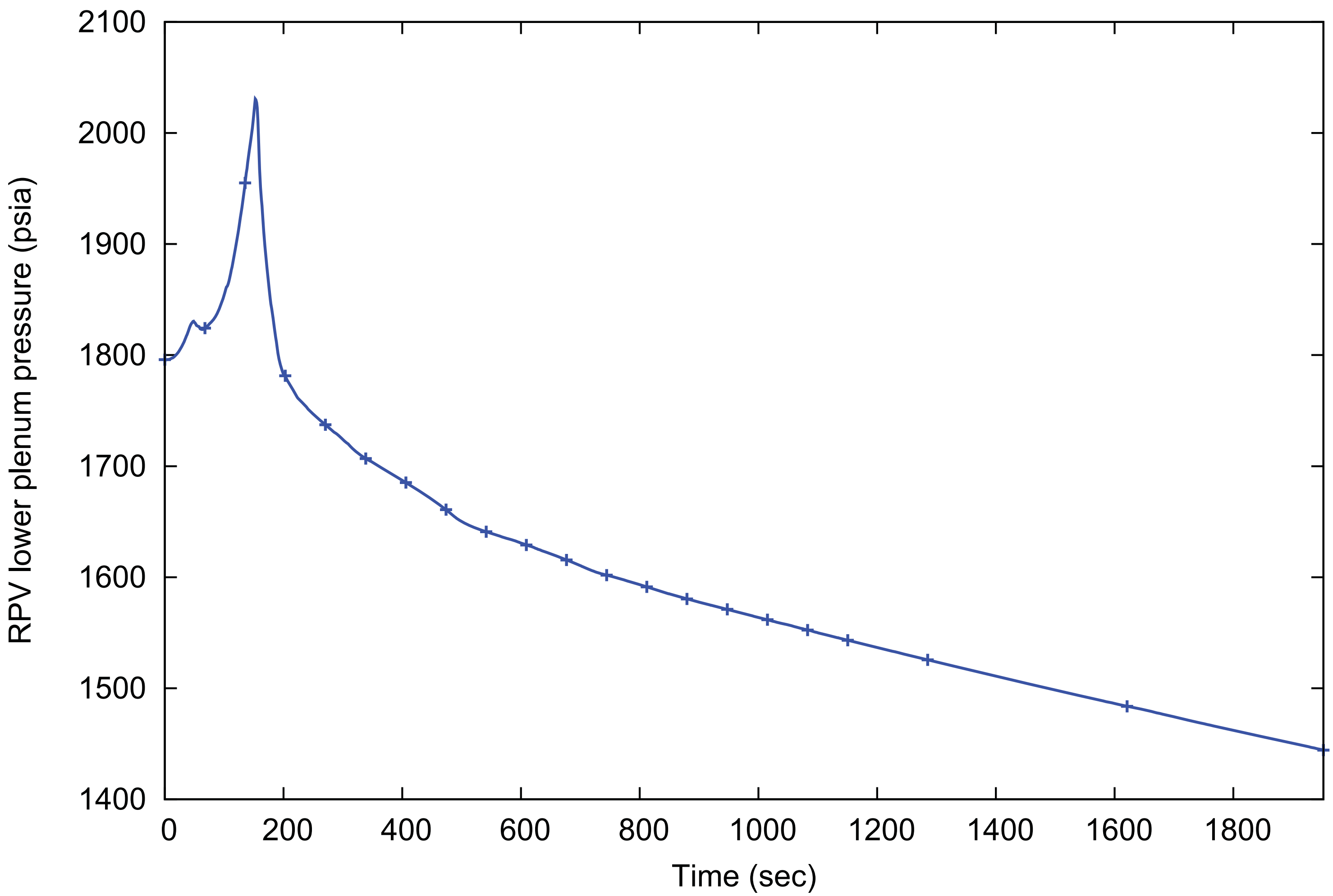


Figure 15.4-9: Average RCS Temperature
(15.4.2 Uncontrolled Control Rod Assembly Withdrawal at Power)

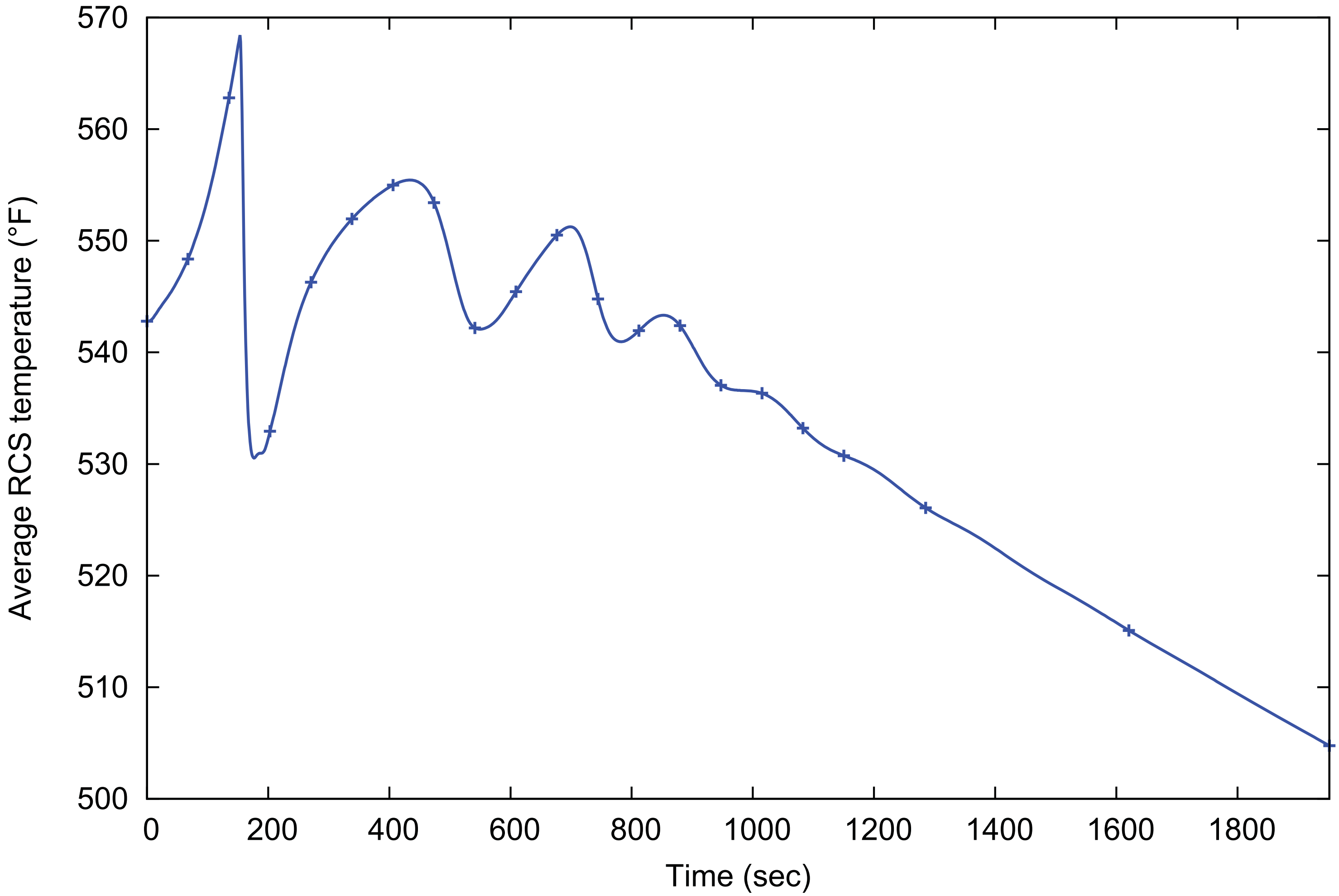


Figure 15.4-10: RCS Flow
(15.4.2 Uncontrolled Control Rod Assembly Withdrawal at Power)

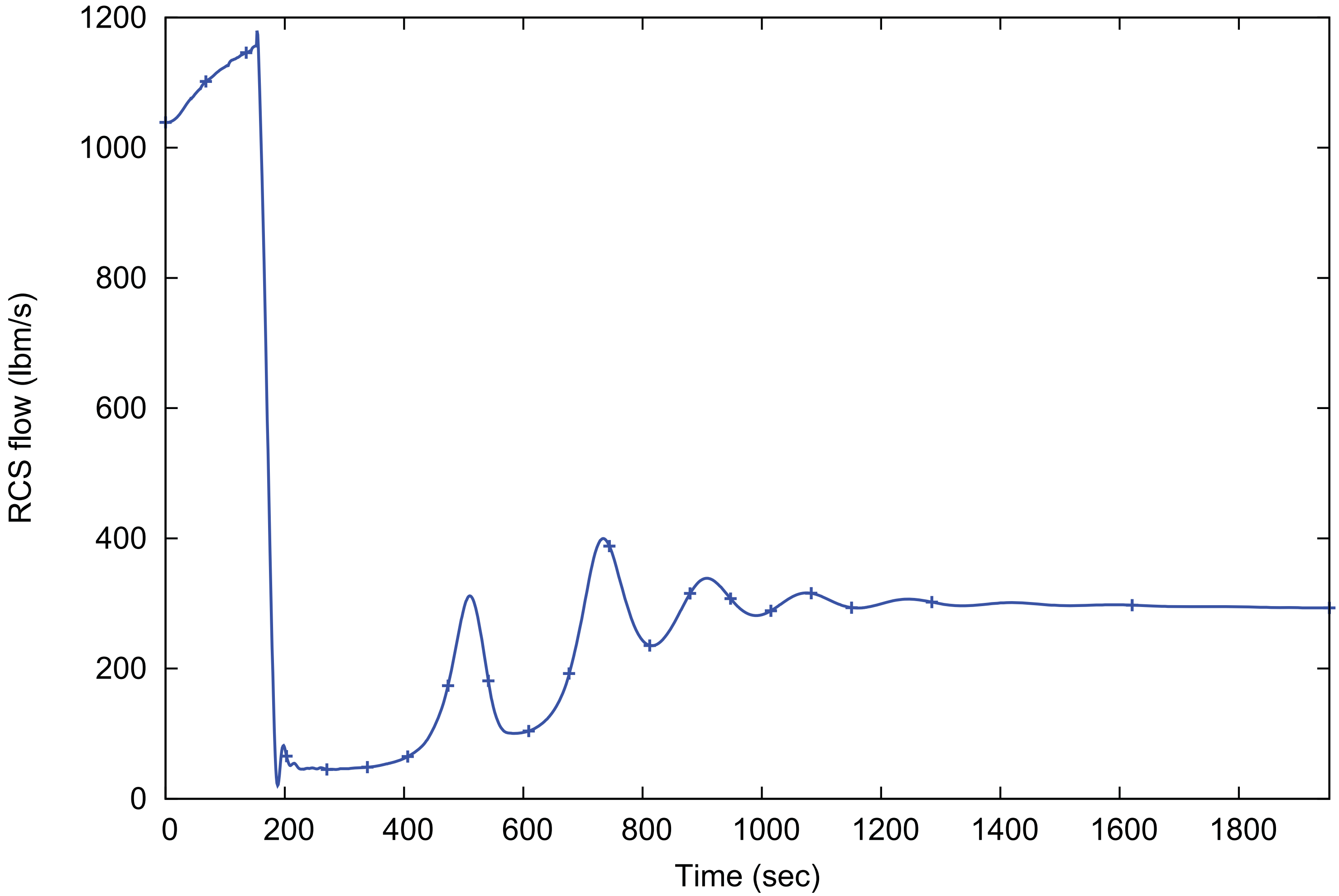


Figure 15.4-11: Critical Heat Flux Ratio
(15.4.2 Uncontrolled Control Rod Assembly Withdrawal at Power)

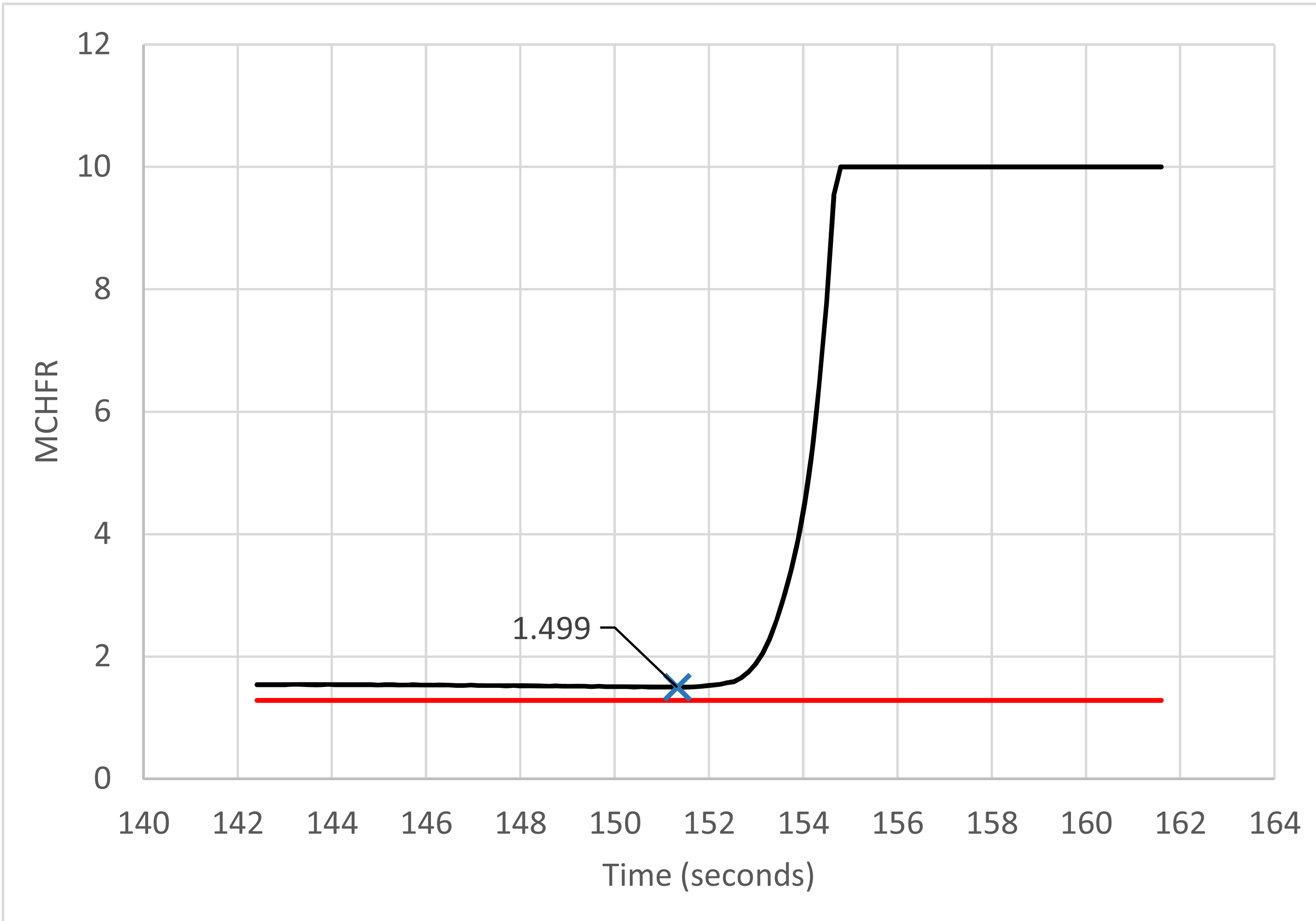


Figure 15.4-12: Not Used

Figure 15.4-13: Withdrawn Control Rod Assembly Reactivity Insertion
(15.4.3 Control Rod Misoperation, Single Control Rod Assembly Withdrawal)

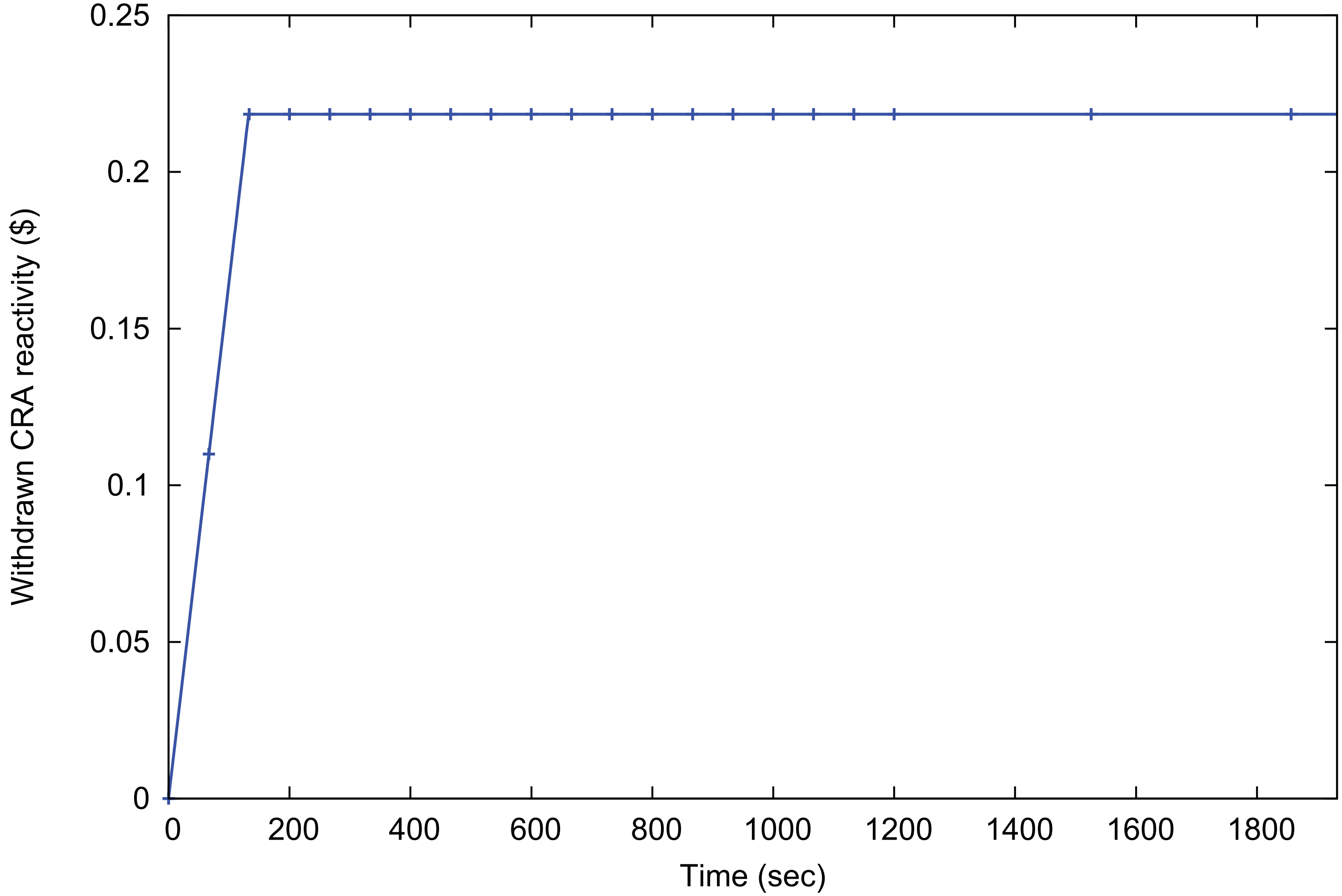


Figure 15.4-14: Reactor Power
(15.4.3 Control Rod Misoperation, Single Control Rod Assembly Withdrawal)

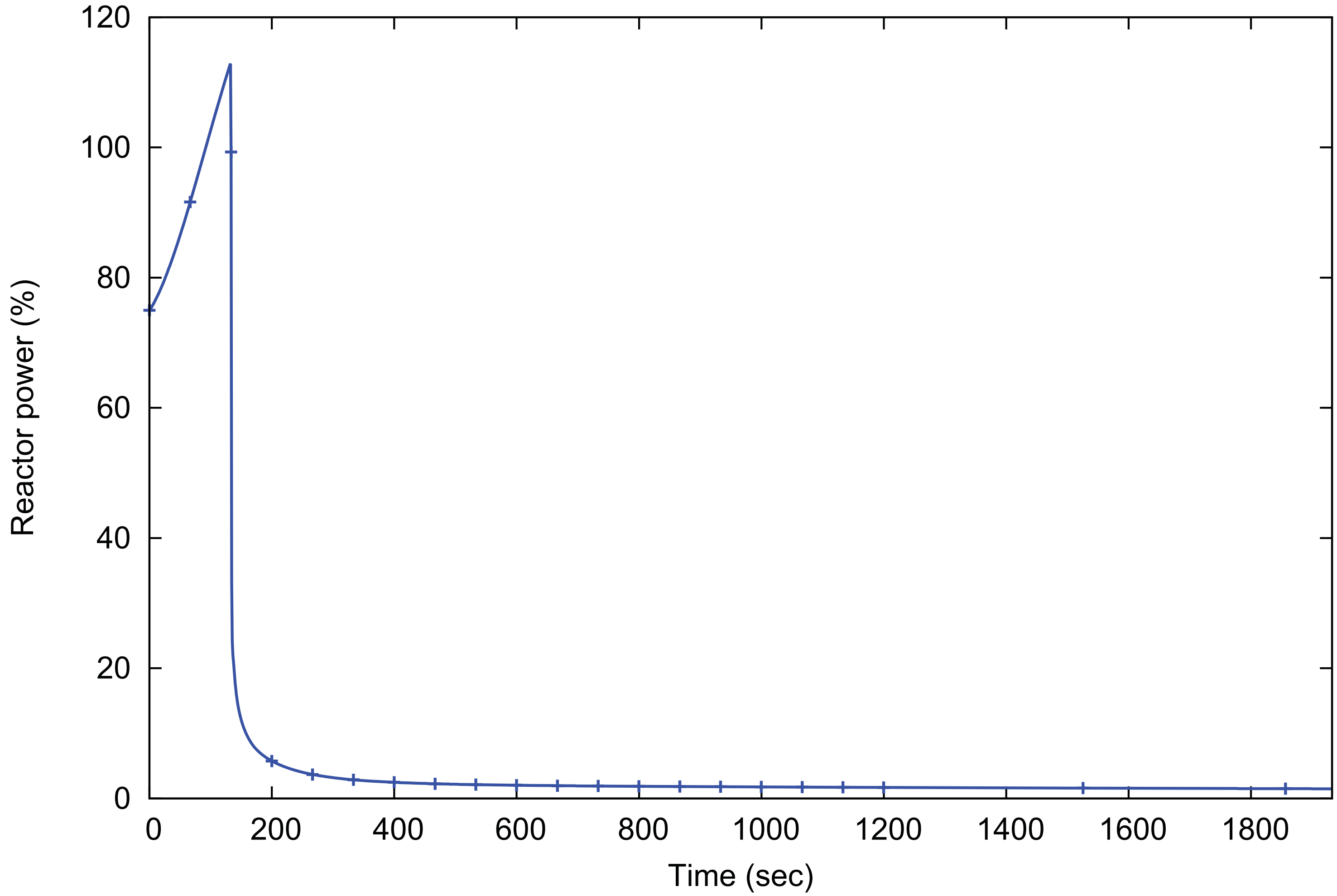


Figure 15.4-15: Reactor Coolant System (RPV Lower Plenum) Pressure (15.4.3 Control Rod Misoperation, Single Control Rod Assembly Withdrawal)

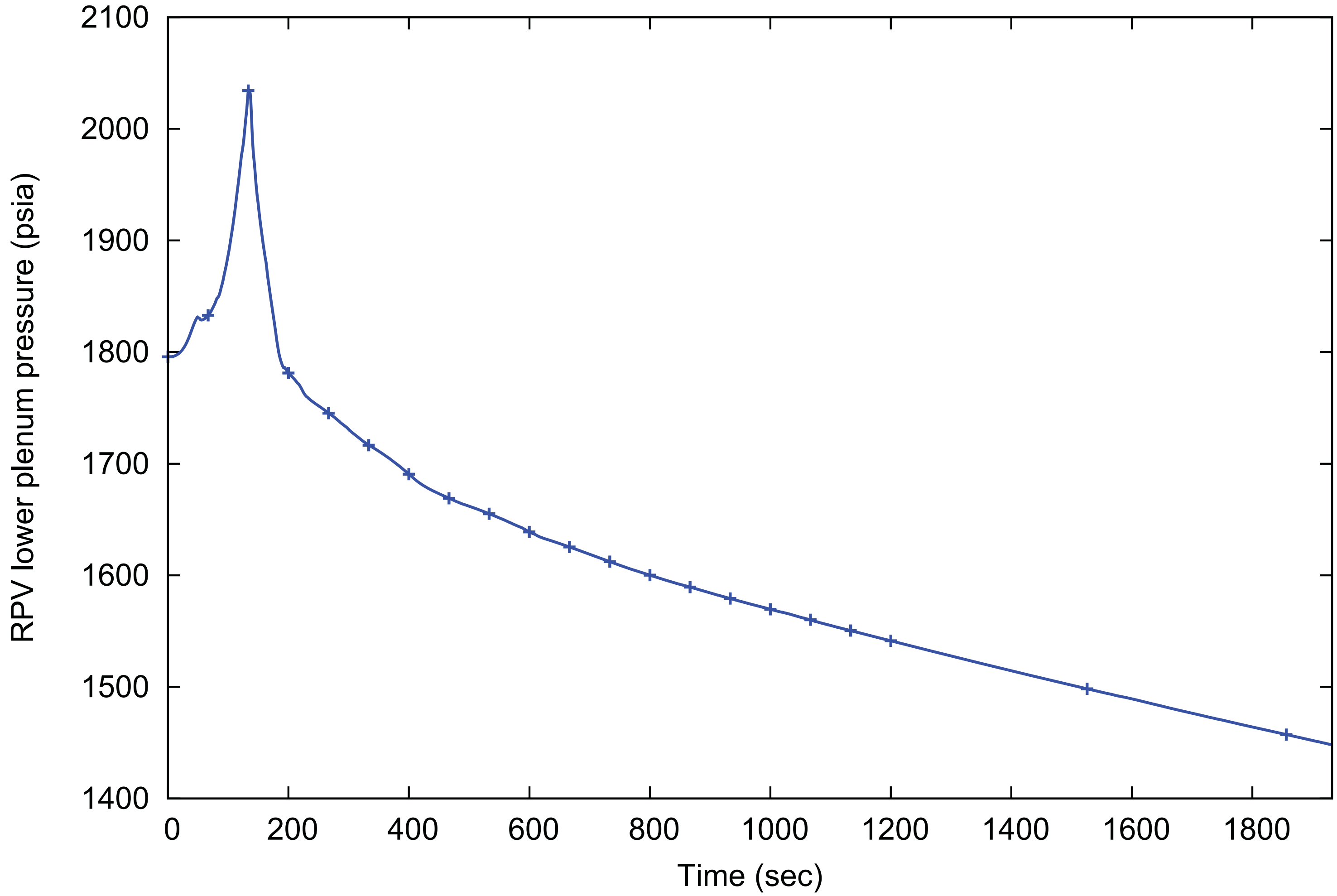


Figure 15.4-16: Total Core Reactivity
(15.4.3 Control Rod Misoperation, Single Control Rod Assembly Withdrawal)

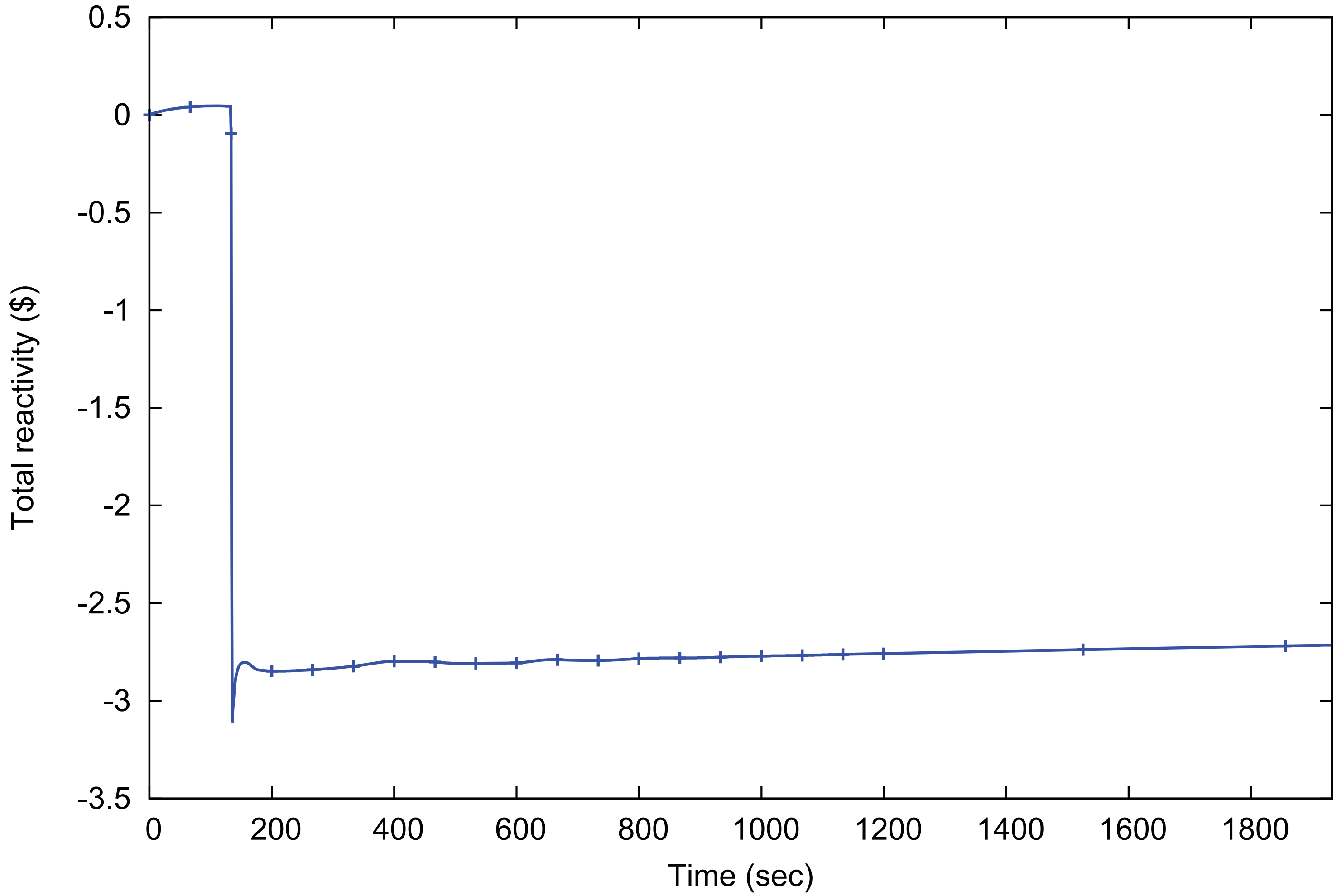


Figure 15.4-17: RCS Average Temperature (15.4.3 Control Rod Misoperation, Single Control Rod Assembly Withdrawal)

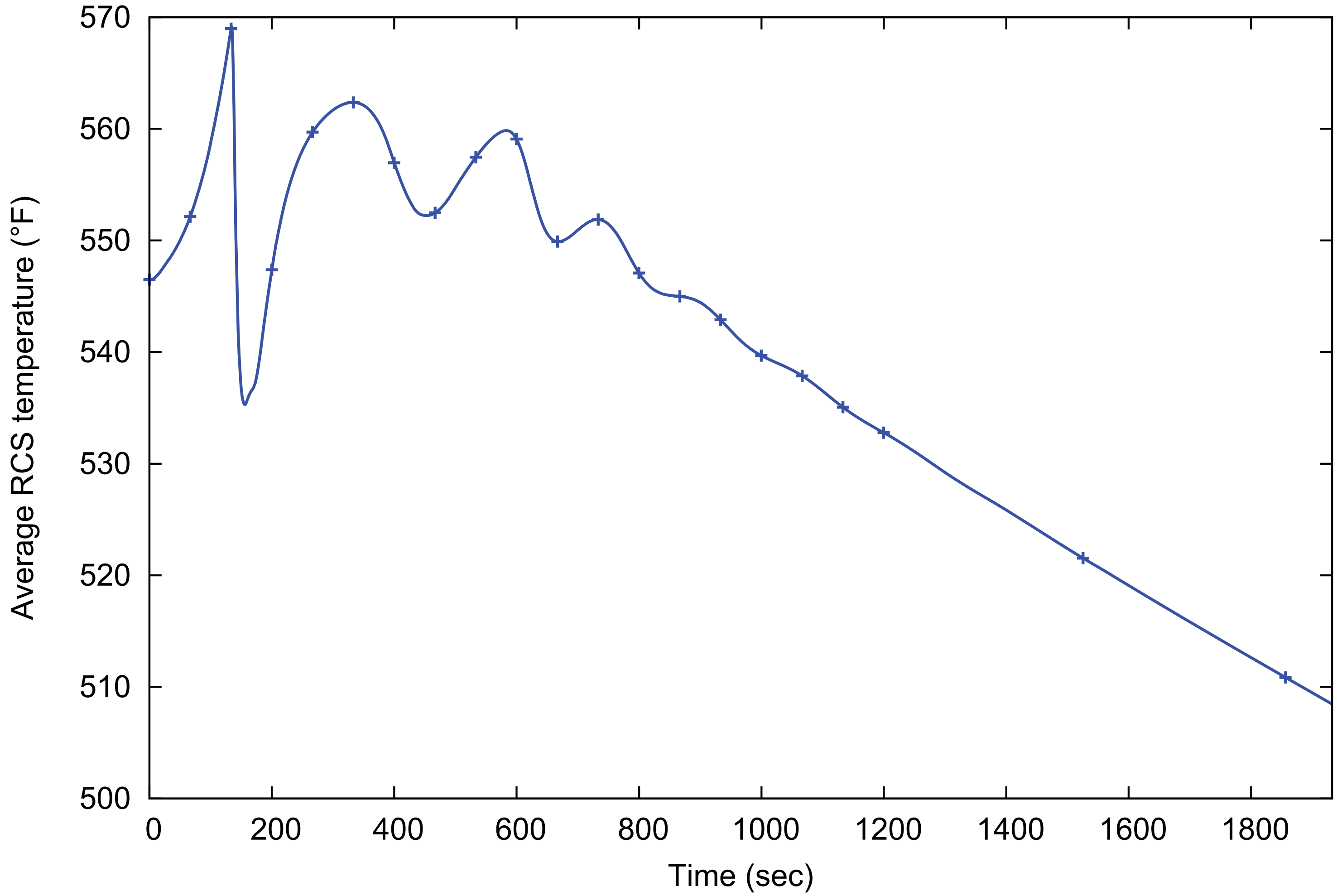


Figure 15.4-18: Reactor Coolant System Flow (15.4.3 Control Rod Misoperation, Single Control Rod Assembly Withdrawal)

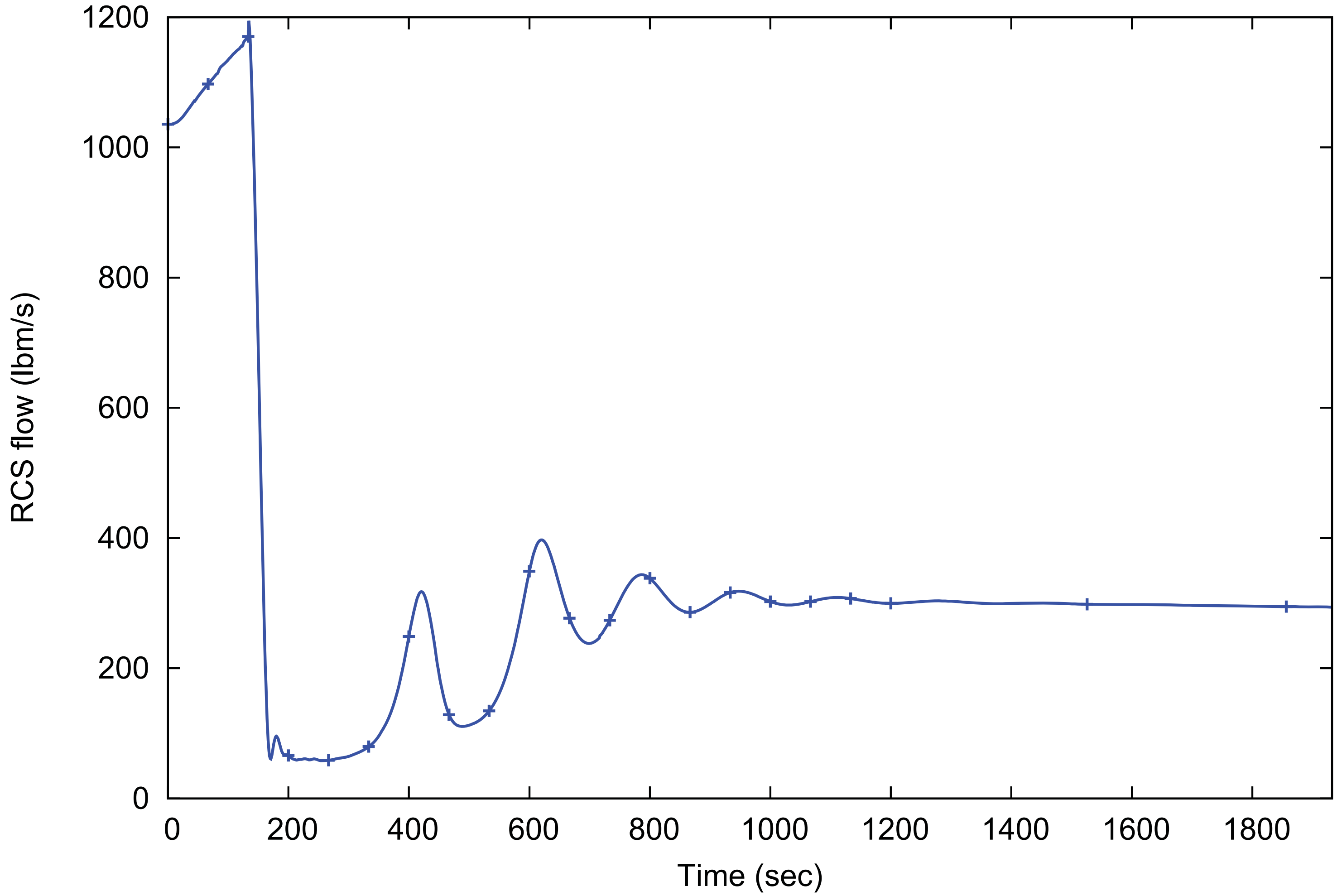


Figure 15.4-19: Doppler Reactivity
(15.4.3 Control Rod Misoperation, Single Control Rod Assembly Withdrawal)

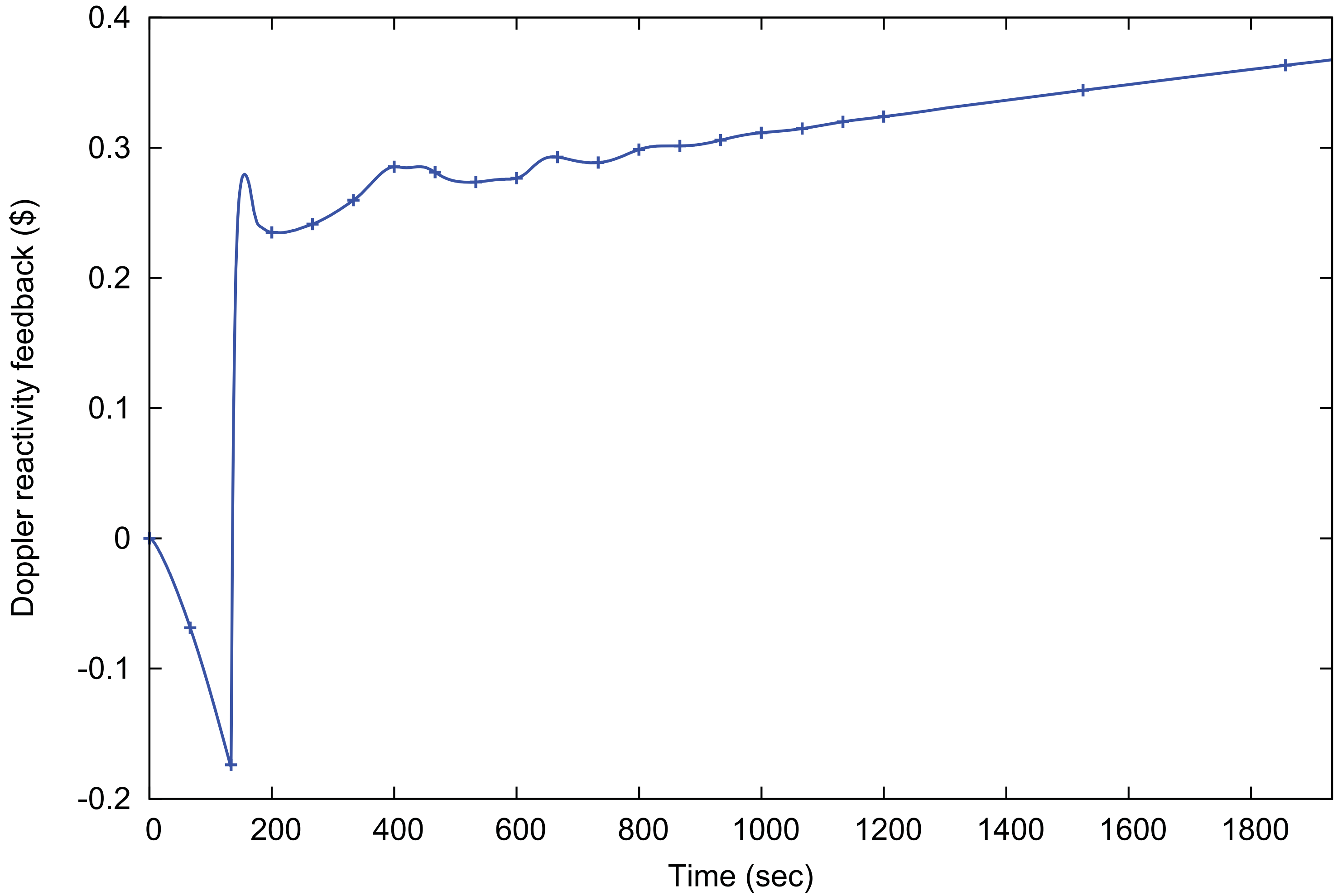


Figure 15.4-20: Not Used

Figure 15.4-21: Dropped CRA Reactivity
(15.4.3 Control Rod Misoperation, Control Rod Assembly Drop)

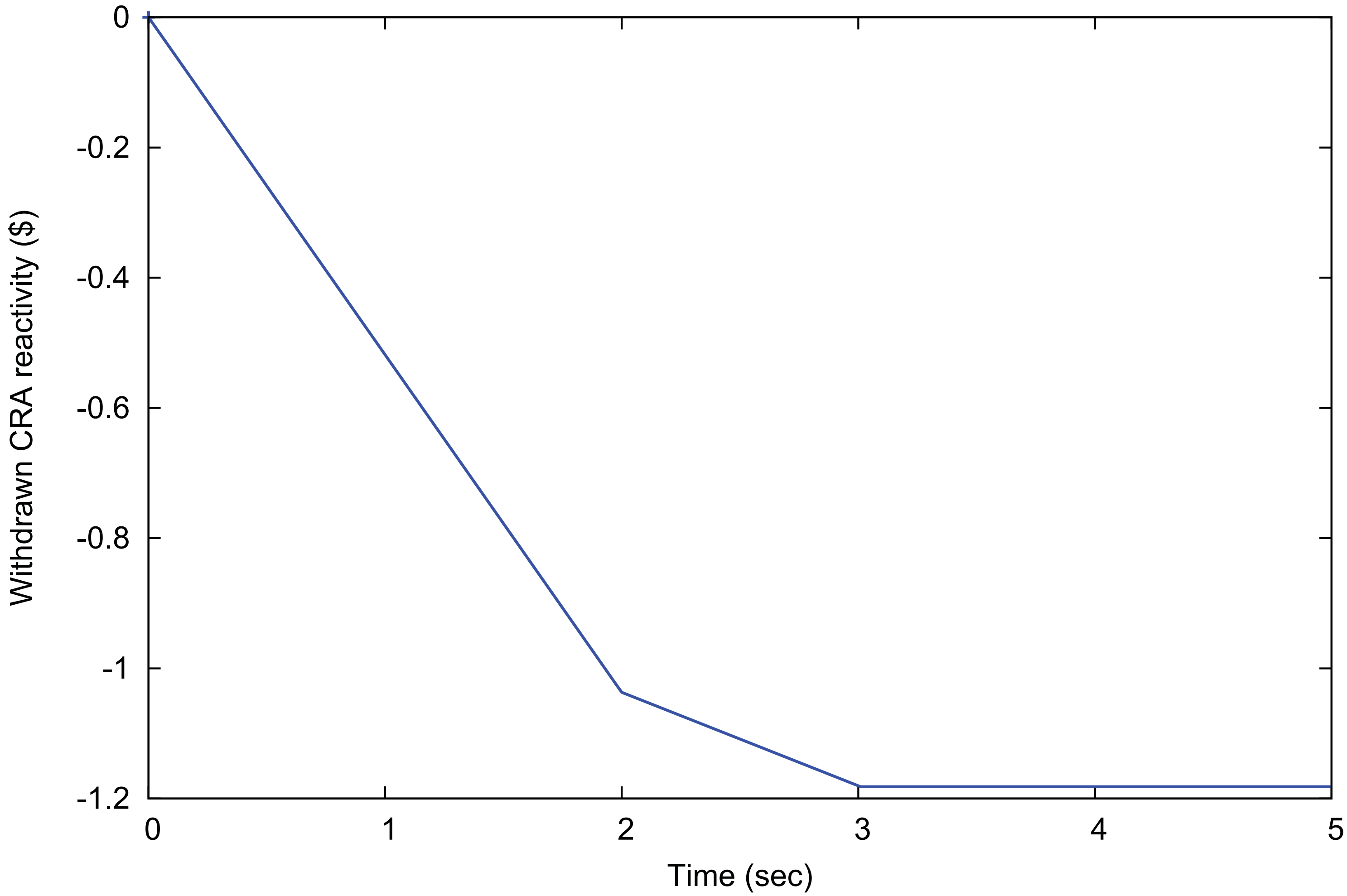


Figure 15.4-22: Absolute Power Rate
(15.4.3 Control Rod Misoperation, Control Rod Assembly Drop)

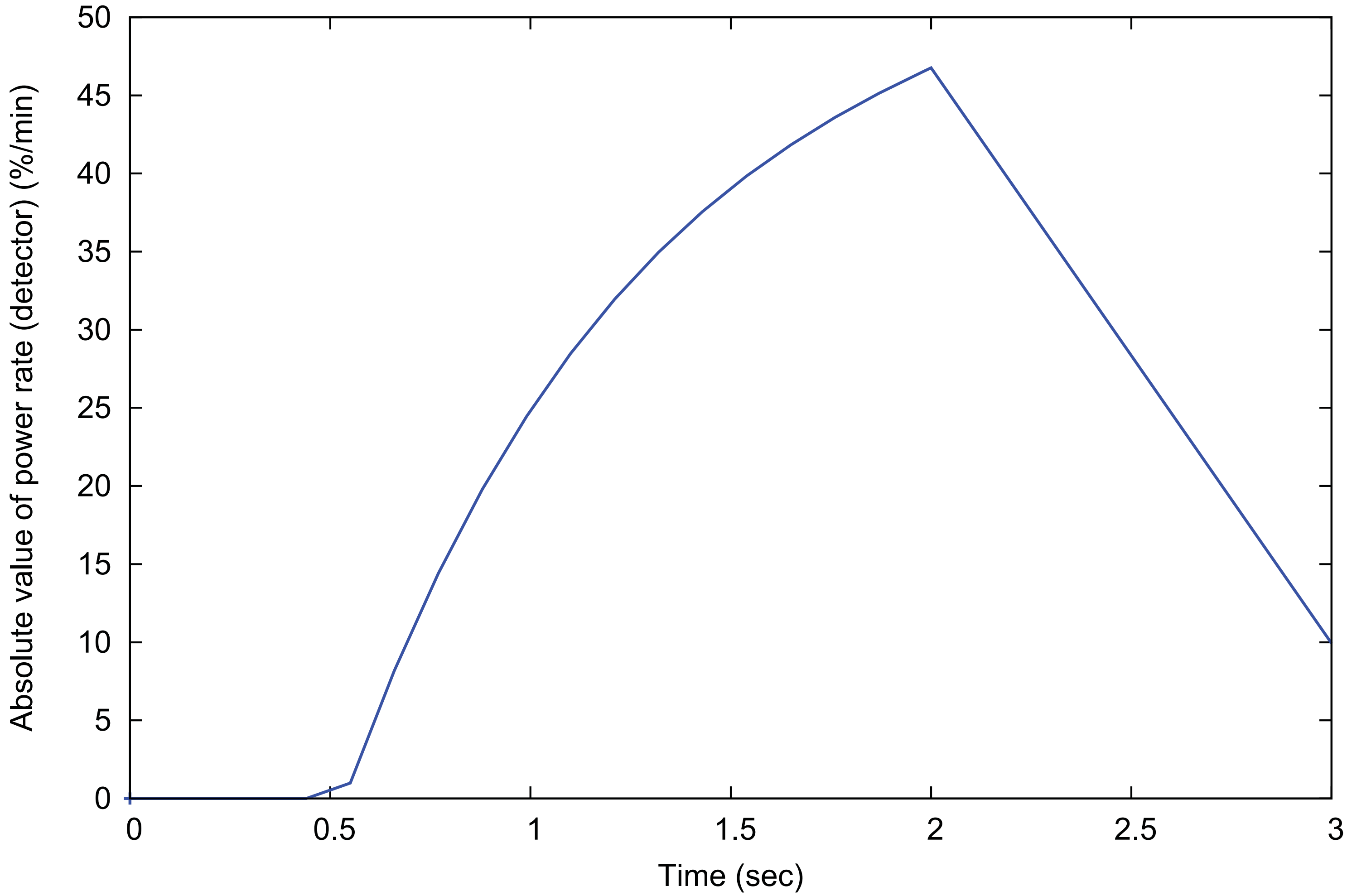


Figure 15.4-23: Reactor Power
(15.4.3 Control Rod Misoperation, Control Rod Assembly Drop)

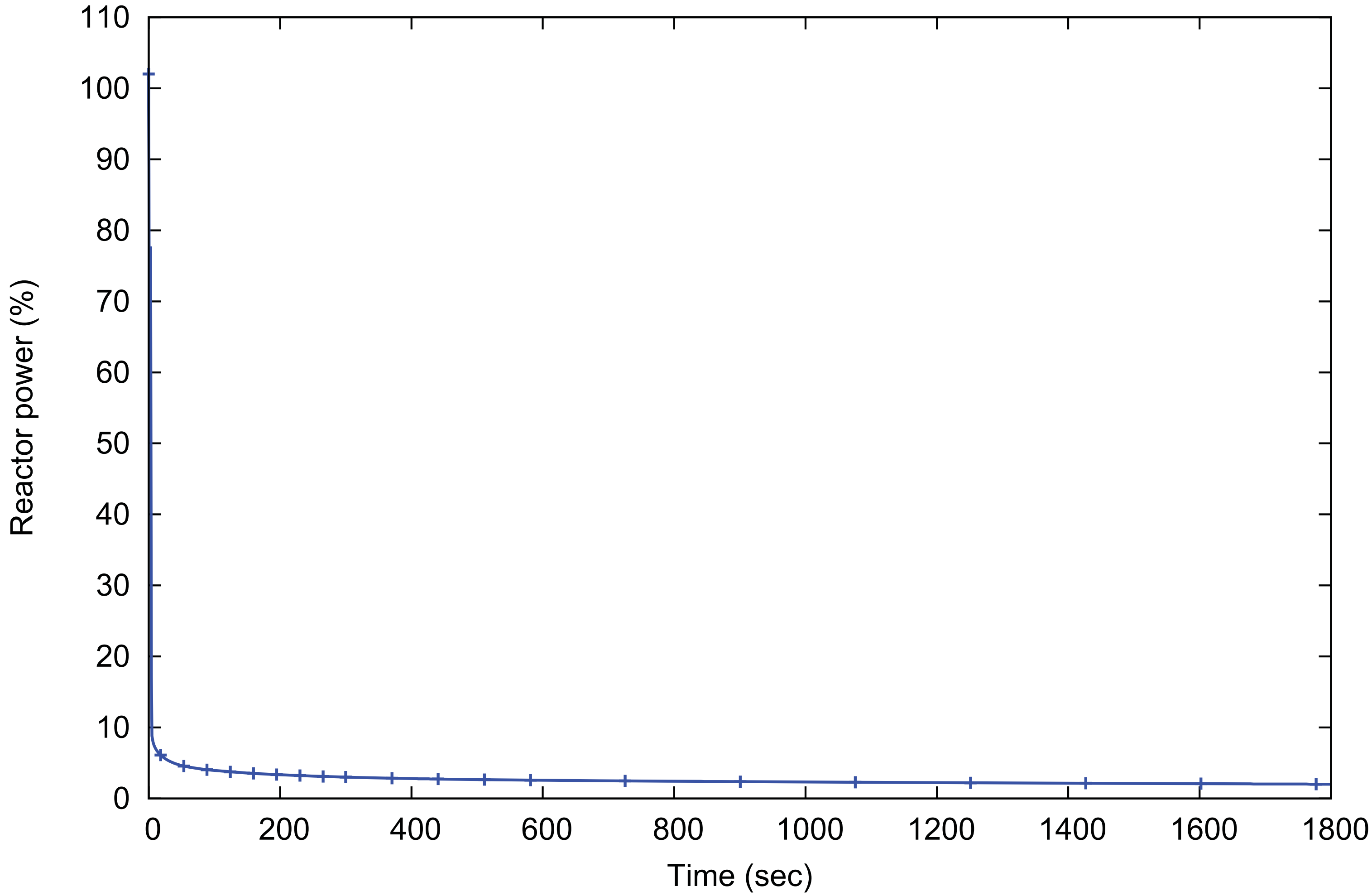


Figure 15.4-24: RCS Pressure (RPV Lower Plenum)
(15.4.3 Control Rod Misoperation, Control Rod Assembly Drop)

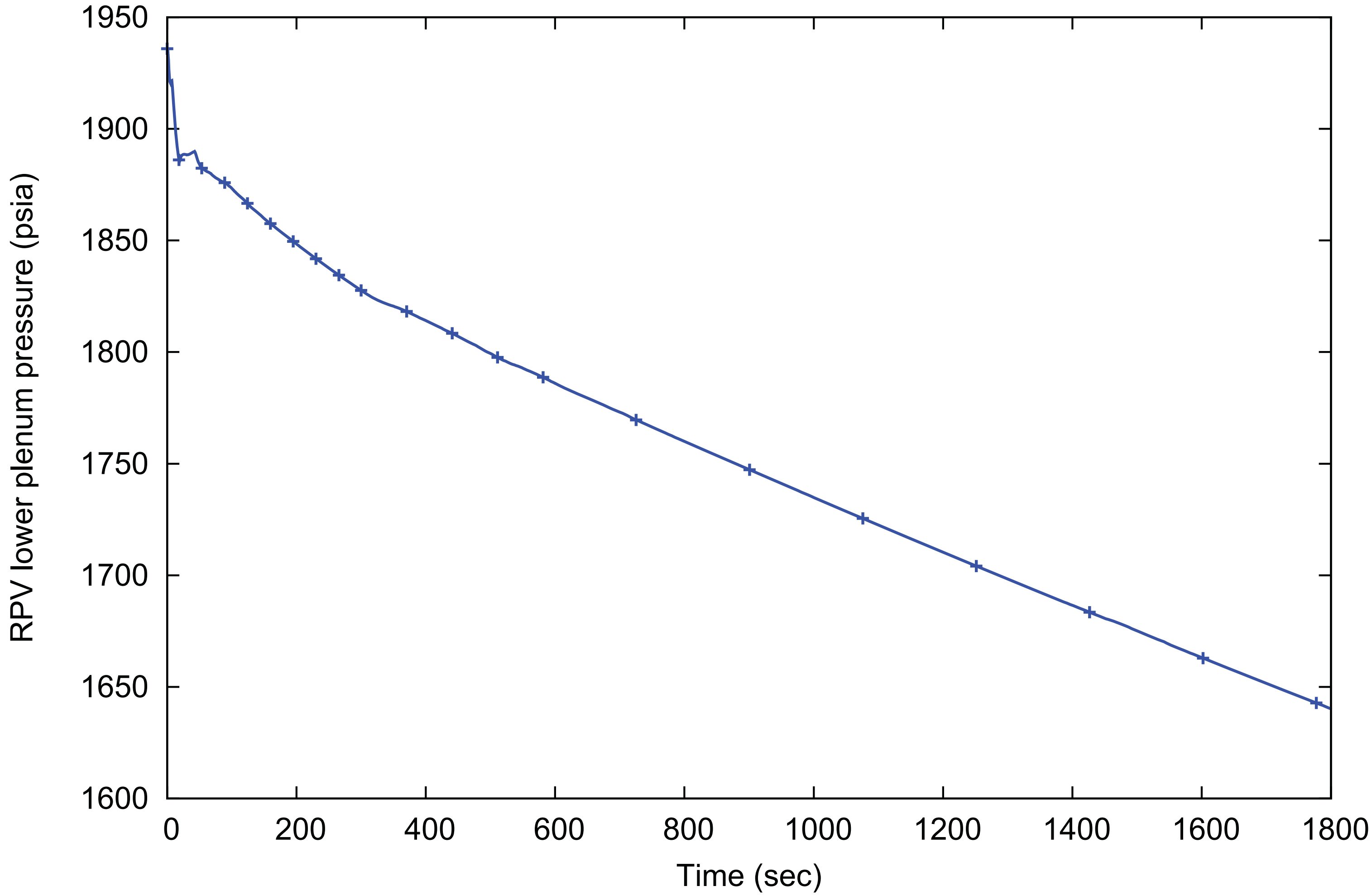


Figure 15.4-25: Average RCS Temperature
(15.4.3 Control Rod Misoperation, Control Rod Assembly Drop)

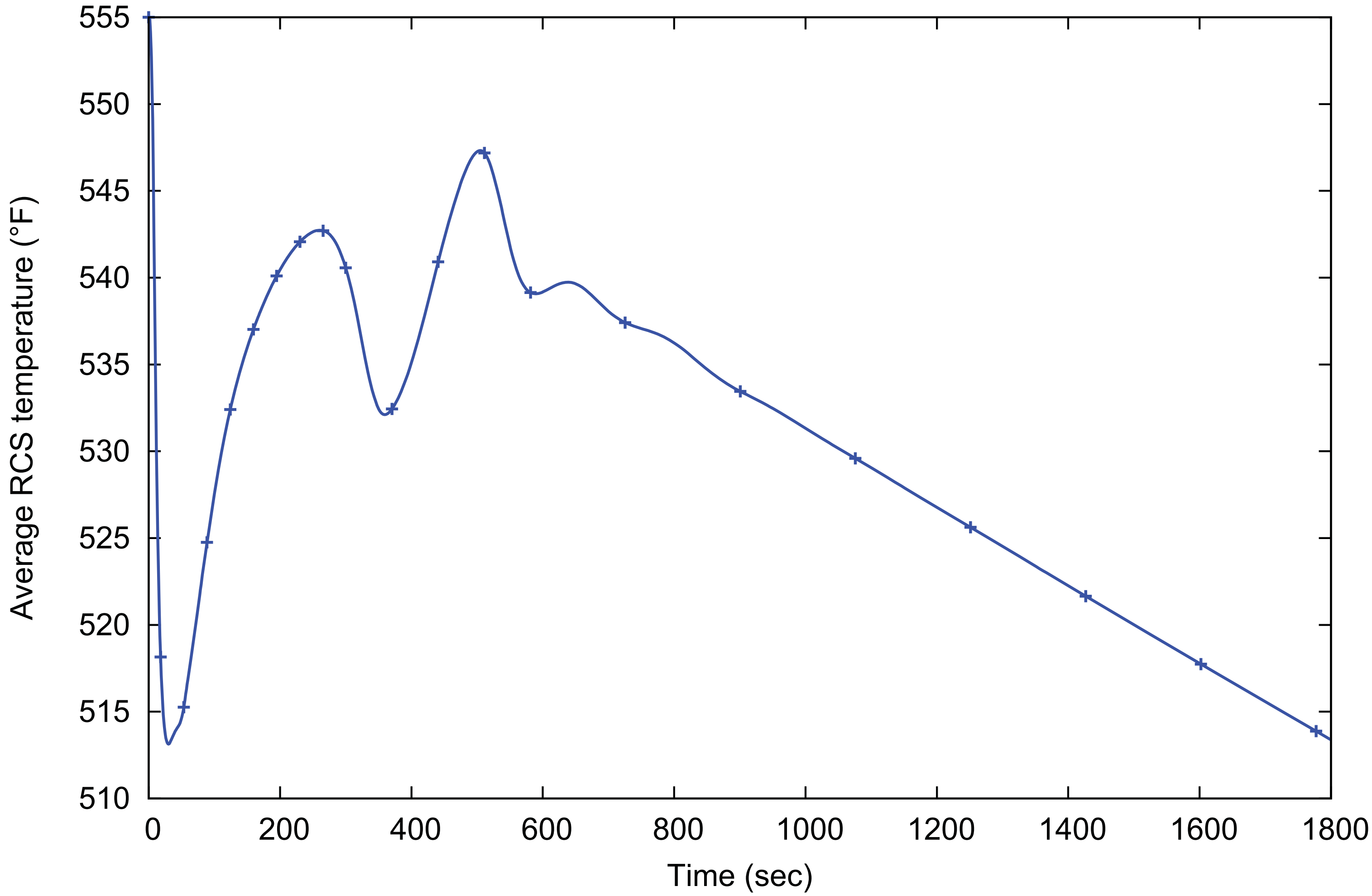


Figure 15.4-26: Reactor Coolant System Flow
(15.4.3 Control Rod Misoperation, Control Rod Assembly Drop)

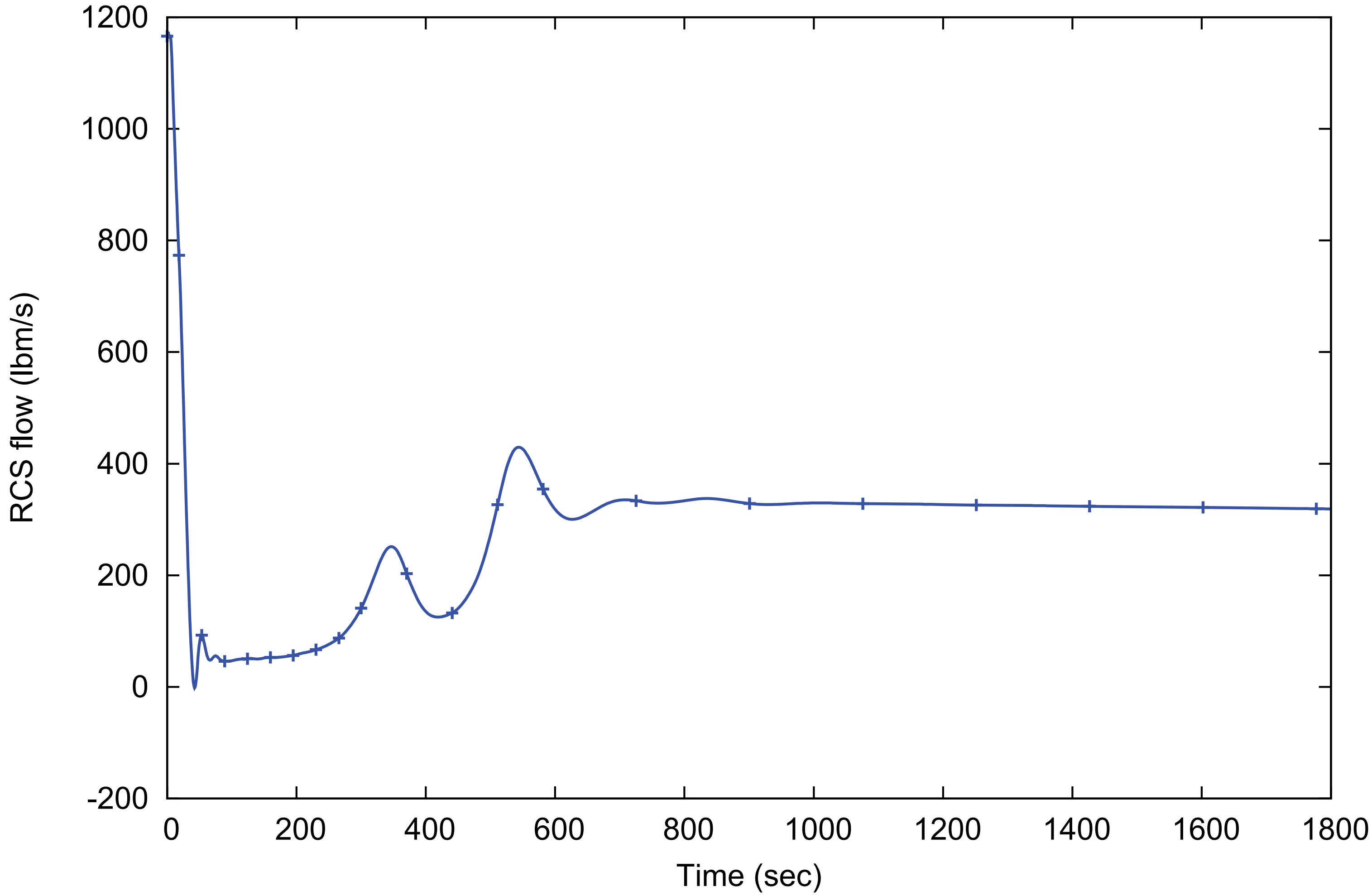


Figure 15.4-27: Doppler Reactivity
(15.4.3 Control Rod Misoperation, Control Rod Assembly drop)

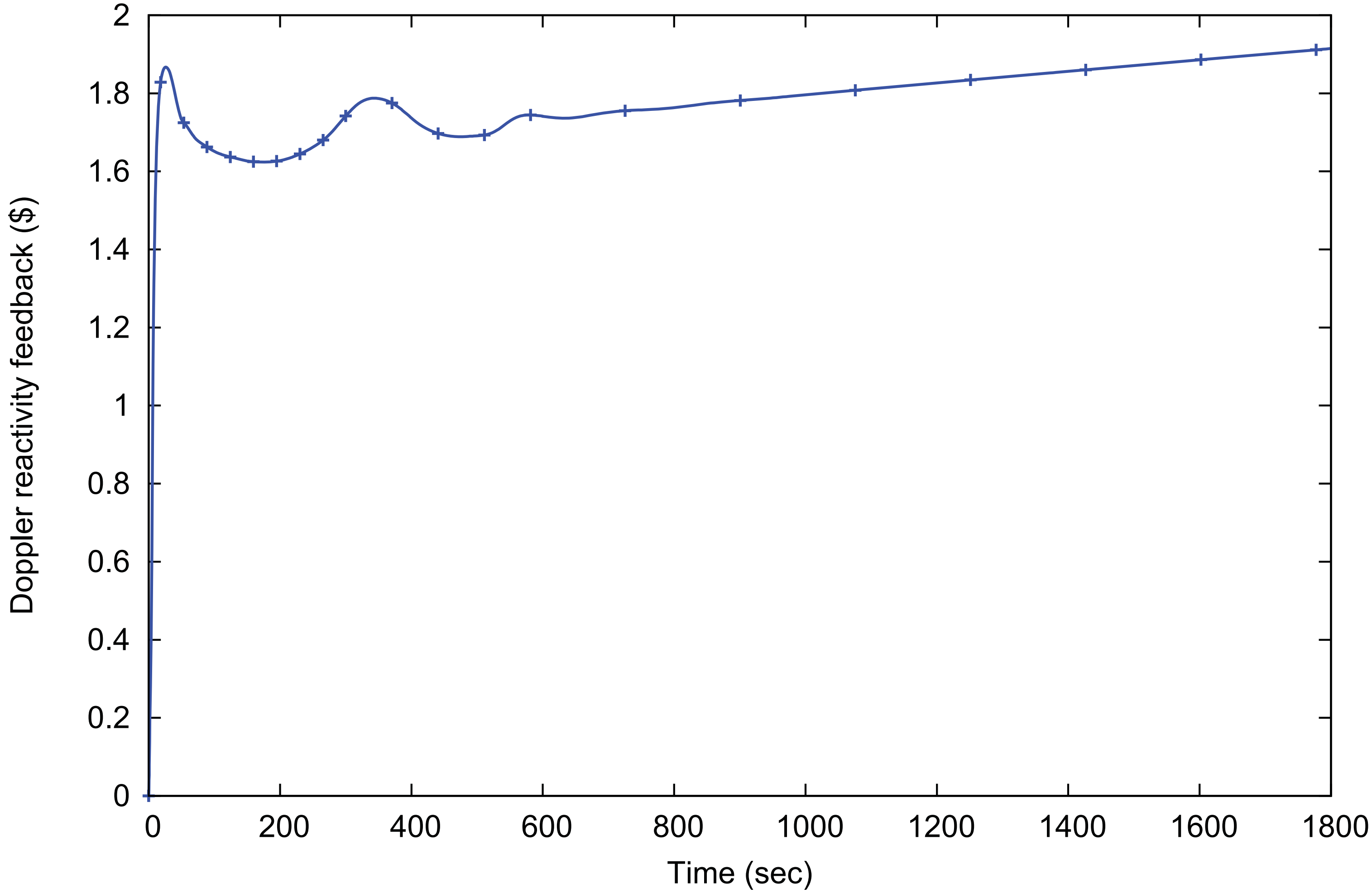


Figure 15.4-28: Spectrum of Fuel Misload Arrangements
(15.4.7 Inadvertent Loading and Operation of a Fuel Assembly in an Improper Position)

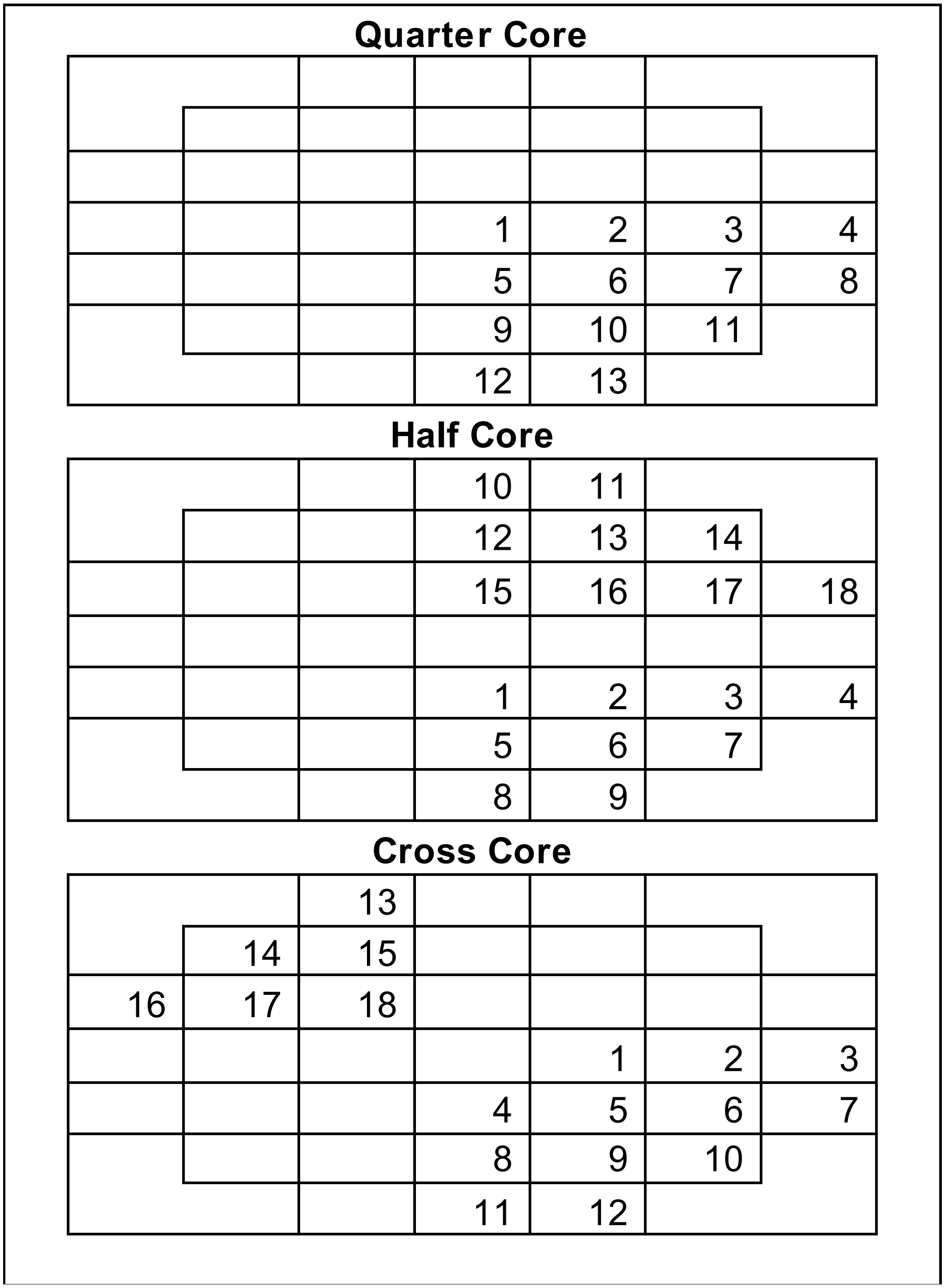


Figure 15.4-29: REA Peak Power Pulse at 25% Initial Power, EOC (15.4.8 Rod Ejection Accident)

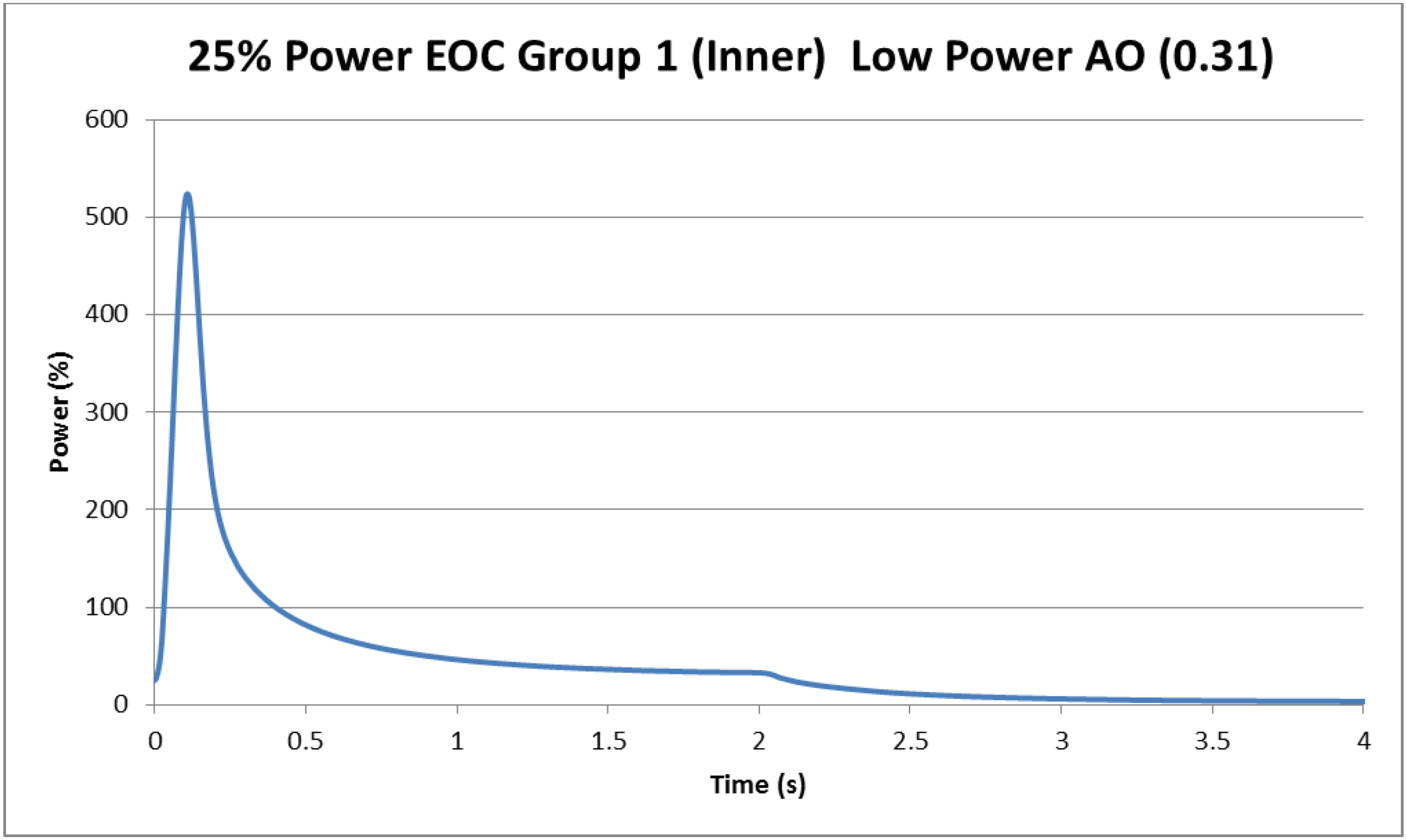


Figure 15.4-30: REA Peak Power Pulse at 70% Initial Power, MOC (15.4.8 Rod Ejection Accident)

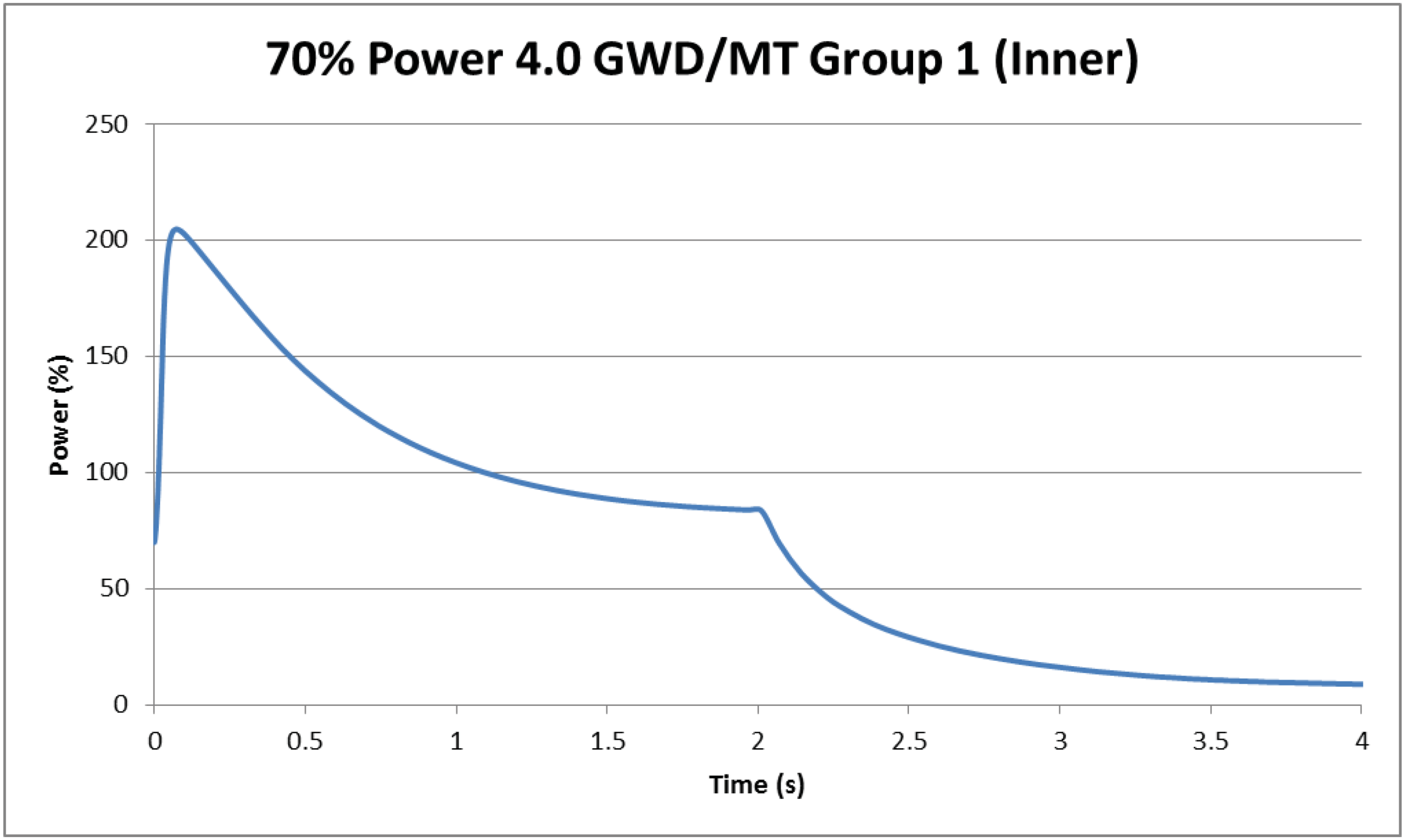


Figure 15.4-31: REA Peak Power Pulse at 70% Initial Power, BOC (15.4.8 Rod Ejection Accident)

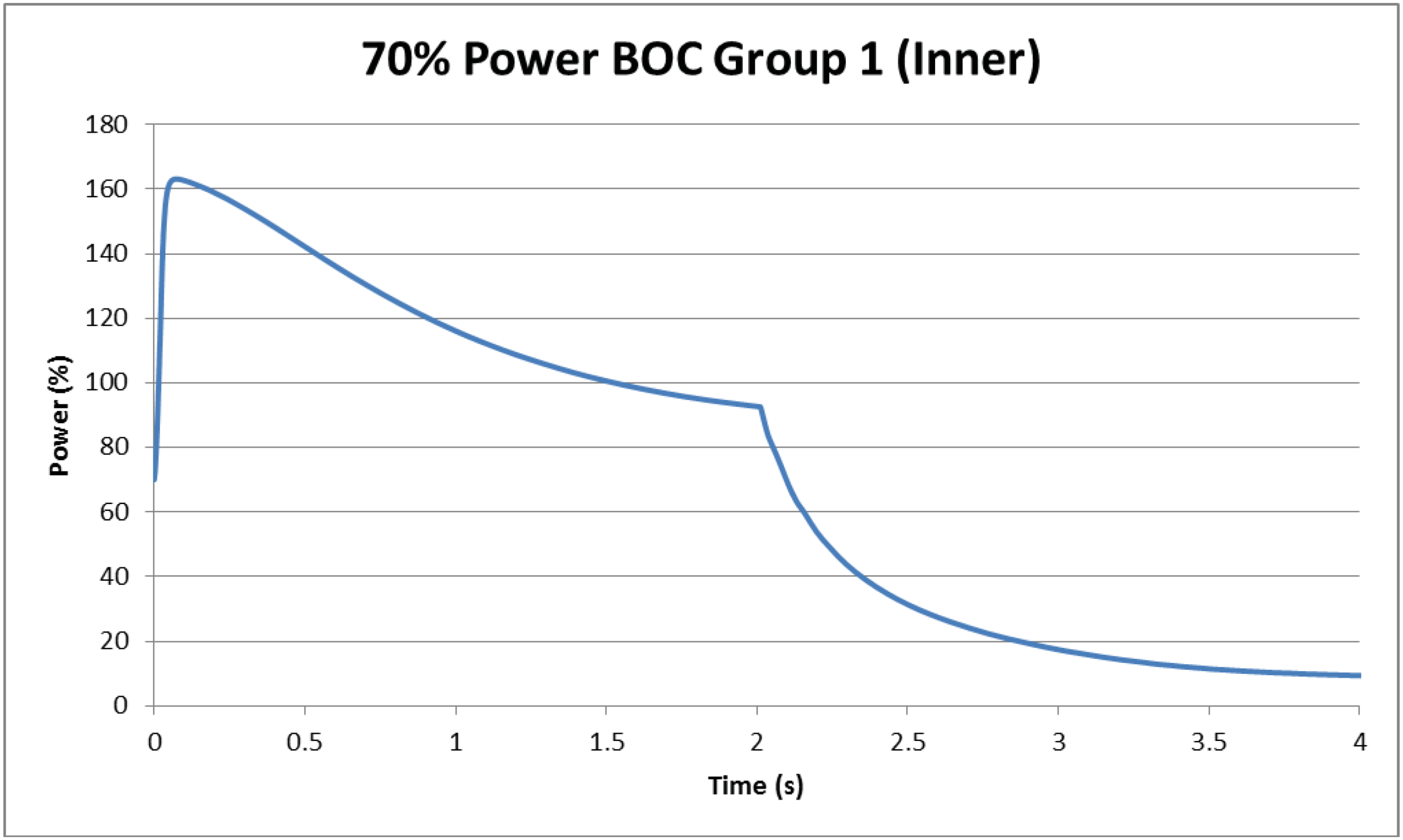


Figure 15.4-32: Not Used

Figure 15.4-33: Reactor Coolant System Pressure (15.4.8 Rod Ejection Accident)

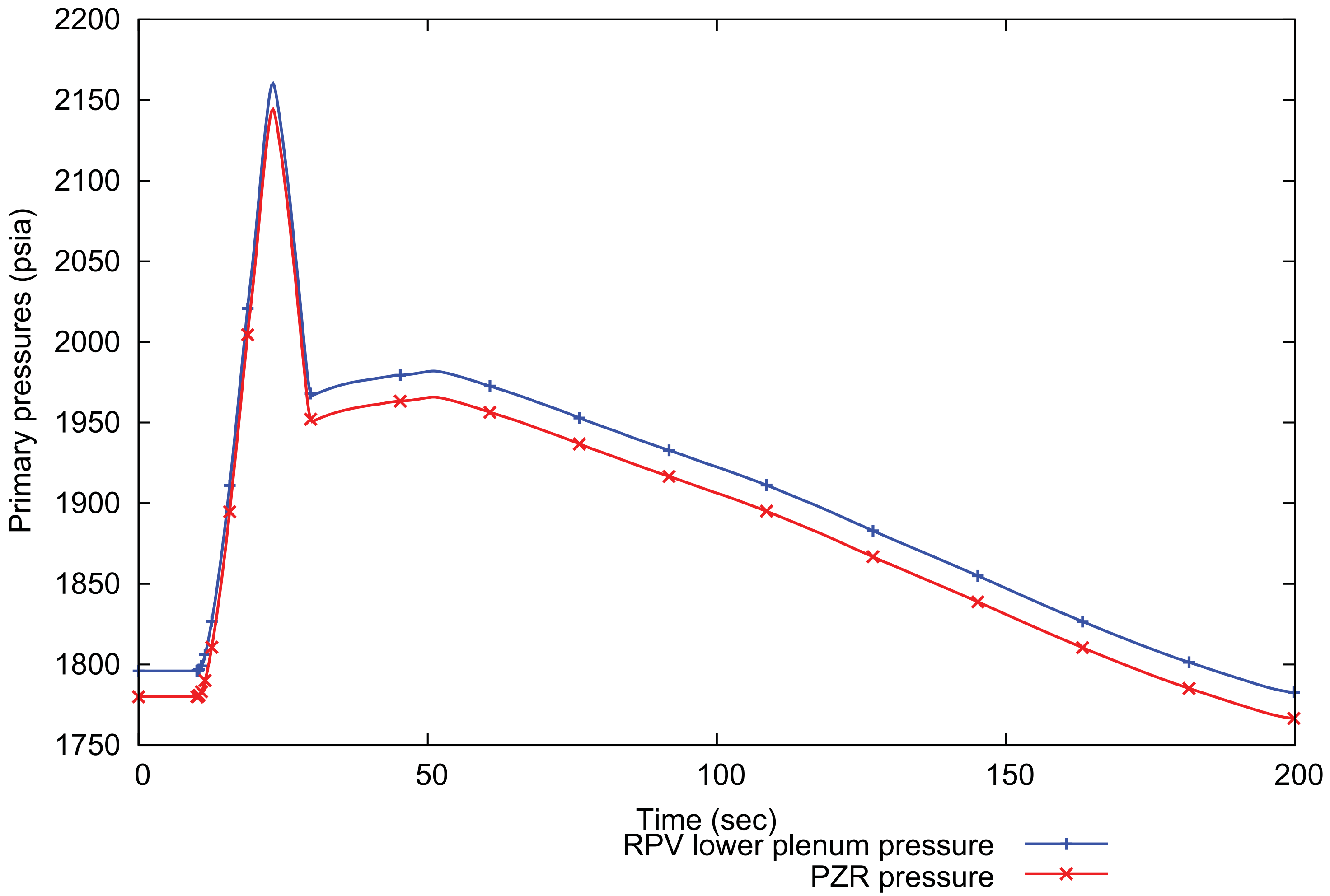


Figure 15.4-34: Not Used

Figure 15.4-35: Critical Heat Flux Ratio
(15.4.3 Control Rod Misoperation, Single Control Rod Assembly Withdrawal)

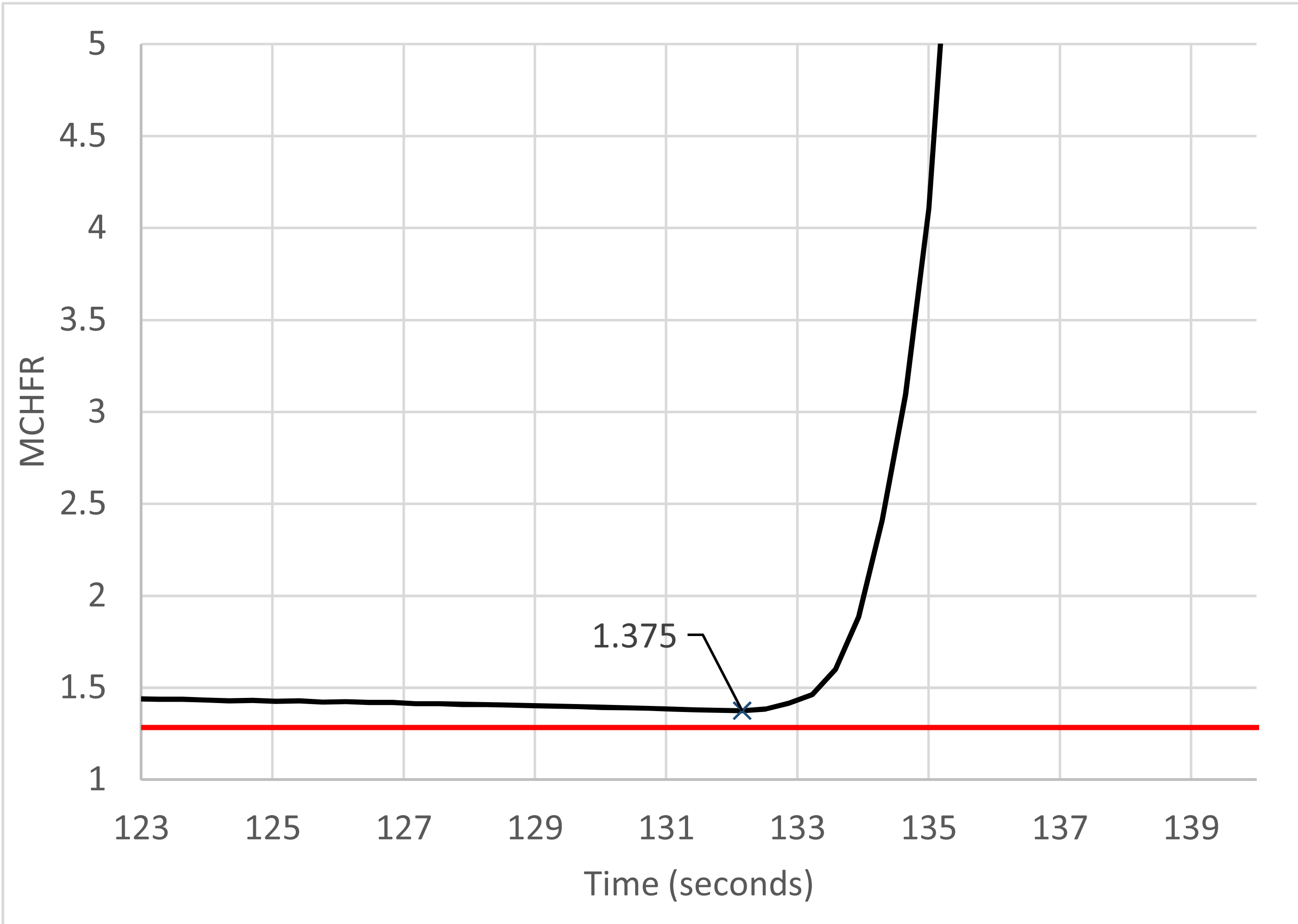


Figure 15.4-36: Critical Heat Flux Ratio
(15.4.3 Control Rod Misoperation, Control Rod Assembly Drop)

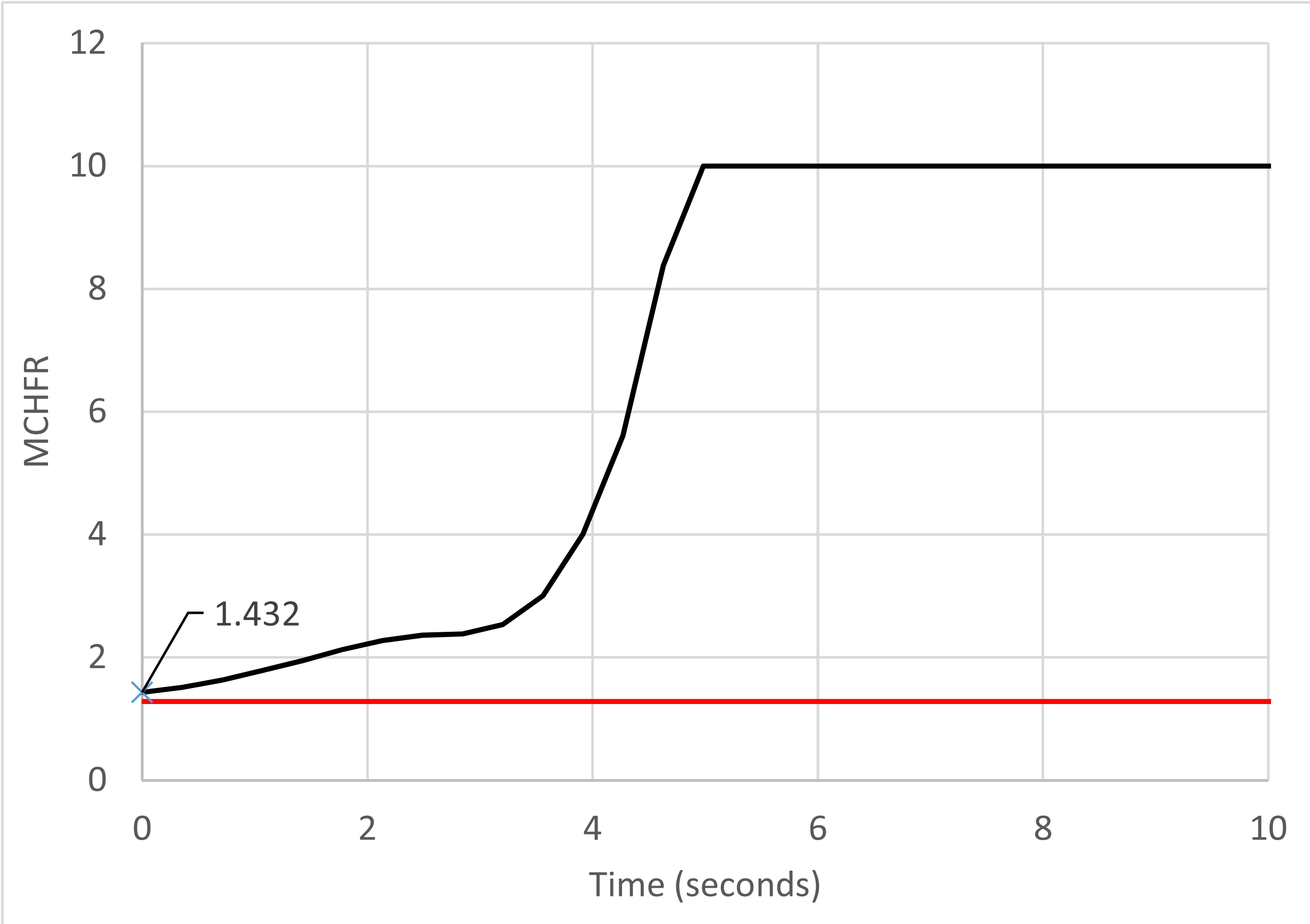
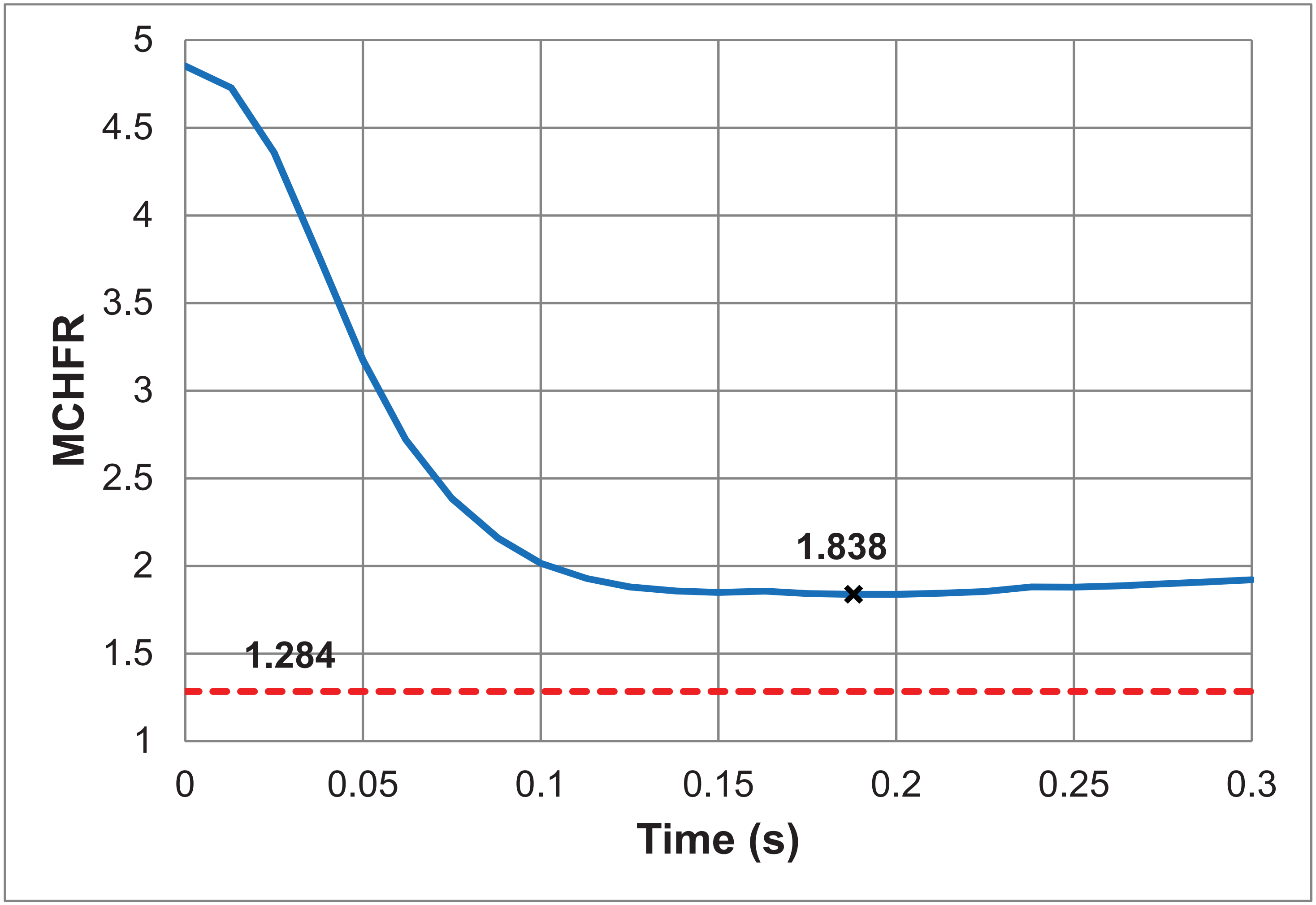


Figure 15.4-37: Critical Heat Flux Ratio (15.4.8 Rod Ejection Accident)



15.5 Increase in Reactor Coolant Inventory

This section addresses design basis events associated with a potential unplanned increase in reactor coolant system (RCS) inventory. In the NuScale design, such an event could occur due to unplanned chemical and volume control system (CVCS) operation. The effect of such events on the NuScale Power Module (NPM) is similar to typical PWRs, i.e., such events, depending on the boron concentration, temperature of injected water and automatic control system operation, have the potential for causing a positive reactivity addition. The event addressed in this section is:

- Chemical and Volume Control System Malfunction. This event is addressed in Section 15.5.1.

15.5.1 Chemical and Volume Control System Malfunction

15.5.1.1 Identification of Causes and Accident Description

An increase in RCS inventory could be caused by a malfunction of the CVCS makeup pumps resulting in an excessive increase in RCS inventory. If borated water at the same concentration of the primary is added to the RCS, the addition of large amounts of cold water to the primary system will generate a reactor trip on high pressurizer water level or high pressurizer pressure.

An increase in inventory is expected to occur one or more times in the life of the plant. Therefore, the increase in RCS inventory event is classified as an AOO, as indicated in Table 15.0-1.

If water is added to the RCS at a lower boron concentration, the event would be a boron dilution transient. The boron dilution event is addressed in Section 15.4.6.

15.5.1.2 Sequence of Events and Systems Operation

The CVCS is a nonsafety-related system that allows flow to be recirculated from the RCS through a regenerative and a nonregenerative heat exchangers and demineralizers. The CVCS discharge line takes suction from the RCS downcomer region below the steam generator. The CVCS injection line discharges inside the riser above the core. Some or all of the discharge flow can be diverted to the liquid radioactive waste system through the letdown line. The remaining flow is returned to the RCS by the recirculation pumps through the injection line and pressurizer spray. In addition, borated water, demineralized water or both can be added to the injection line from two 20 gpm makeup pumps. The system is maintained pressurized at approximately RCS pressure throughout the recirculation cycle. During normal recirculation operation, the volume of RCS withdrawn from the RCS through the CVCS discharge line is equal to the amount pumped through the CVCS injection line and pressurizer spray lines, unless water from the makeup pumps is added.

The increase in RCS inventory event is terminated by MPS actuation of a reactor trip and CVCS isolation. The MPS signal is initiated by a high pressurizer level or high pressurizer pressure condition. The CVCS is isolated from the RCS by two redundant safety-related containment isolation valves located on both the exit and entry

containment penetrations. These valves are also isolated on a containment isolation signal or a loss of DC power (EDSS).

For the purposes of this analysis, the amount of water directed to letdown from the discharge line is assumed to be zero gpm and the maximum makeup of 40 gpm, with the same boron concentration as RCS, is added to the injection line at the beginning of the event.

The sequence of events for the increase in RCS inventory is provided in Table 15.5-1, Table 15.5-2, and Table 15.5-3 for the limiting scenarios considering biased boundary conditions and pressurizer spray operation. All scenarios assume the availability of AC Power (ELVS) and DC Power (EDNS and EDSS). The CVCS cannot function without ELVS or EDNS power and the CVCS flow pathways are isolated on a loss of EDSS. Therefore, the loss of any of these power supplies would terminate the CVCS flow addition event.

Three event sequences provided limiting results for the applicable acceptance criteria. In each scenario, fast closure (0.1 seconds) of the secondary system containment isolation valves is assumed. The isolation valve closure isolates the secondary side, resulting in a loss of secondary side heat sink until the DHRS actuates and begins cooling. The fast closure of the valves increase the heatup and pressurization of the primary and secondary systems.

The first limiting case maintained normal pressurizer spray flow which delays the pressurizer high pressure reactor trip. This event is terminated by the high pressurizer level reactor trip and concurrent automatic CVCS isolation. The DHRS and SSI initiate on high steam pressure after the reactor trip. Maximum steam generator pressure occurs after the closure of the MSIVs. Stable DHRS cooling is established. This event resulted in the highest steam generator pressure for the increase in RCS inventory events.

The second limiting event sequence was evaluated without pressurizer spray flow and the event resulted in a reactor trip on high pressurizer pressure. Concurrently, DHRS and SSI are actuated on high pressurizer pressure. Pressure continues to increase and is limited by lifting of an RSV. The level in the pressurizer continues to increase until the high pressurizer level setpoint automatically isolates CVCS. The pressurizer level trip is delayed in this sequence because the reactor trip and DHRS actuation begin cooling the RCS, increasing fluid system density and decreasing RCS volume. The termination of pressurizer level increase and the establishment of stable DHRS cooling are considered the ending point for the simulation. This event resulted in the highest RCS pressure for the increase in RCS inventory events.

The third limiting event sequence was evaluated without pressurizer spray flow and the event resulted in a reactor trip on high pressurizer pressure. Concurrently, DHRS is actuated on high pressurizer pressure. Pressure continues to increase but is maintained below the RSV setpoint. The level in the pressurizer continues to increase until the RCS low-low flow setpoint automatically isolates CVCS. The establishment of stable DHRS cooling is considered the ending point for the simulation. This event resulted in the lowest MCHFR for the increase in RCS inventory events.

There are no single failures that would result in a more serious outcome for the increase in RCS inventory events. The diversity, redundancy, and independence of the MPS

ensure the system will perform its intended function despite a single failure. The redundancy and passive nature of the DHRS ensure that the system will perform its intended function despite a single failure. The reactor safety relief valves and containment isolation valves also have redundancy to accommodate a single failure. No cases involving loss of AC or DC power were evaluated as the loss of either AC or DC power terminates this event. No operator action was credited to mitigate this event.

15.5.1.3 Evaluation Models

The thermal hydraulic analysis of the NPM response to an increase in RCS inventory event is performed using NRELAP5. The NRELAP5 model is based on the design features of a NuScale module. The non-LOCA NRELAP5 model is discussed in Section 15.0.2.2.2. The relevant boundary conditions from the NRELAP5 analyses are provided to the downstream subchannel Critical Heat Flux (CHF) analysis.

The subchannel CHF analysis is performed using VIPRE-01. VIPRE-01 is a subchannel analysis tool designed for general-purpose thermal-hydraulic analysis under normal operating conditions, operational transients, and events of moderate severity. Section 15.0.2.3 provides a discussion of the VIPRE-01 code and evaluation model.

15.5.1.3.1 Input Parameters and Initial Conditions

The initiating event for the increase in RCS inventory is the malfunction of the CVCS by a spurious pressurizer water level signal that requires maximum CVCS makeup flow of 40 gpm.

The initial power level for this event was the licensed core thermal power of the module with 2 percent uncertainties (102 percent). Other parameters were biased to maximize primary and secondary pressure and minimize CHFR including:

- The initial RCS pressure
- The initial RCS temperature
- The initial PZR level
- The drift on RSV setpoint
- Delay on CVCS valve closure time
- CVCS makeup fluid temperature
- Moderator and Doppler coefficients of reactivity
- Maximum regulating control rod speed
- Minimum RCS flowrate

The input parameters for the increase in RCS inventory for the limiting cases are provided in Table 15.5-4 for the limiting scenarios considering biased boundary conditions and pressurizer spray operation.

15.5.1.3.2 Results

Sensitivity studies on an increase in reactor coolant inventory were performed. Cases were performed with normal boundary conditions, and other cases were performed with biased conditions in an attempt to maximize the primary mass increase and pressurization rate.

For cases without spray, primary system pressurization is higher than the cases with spray. The limiting case results summaries for the CVCS malfunction are shown in Table 15.5-5. Figures 15.5-1 through 15.5-7 provide the salient information for the limiting scenarios for the maximum RCS pressure case (no PZR spray) including pressurizer level, reactor power, total reactivity, RCS average temperature, RCS mass addition, RCS flow, and DHRS flow. The maximum RCS pressure met the acceptance criteria and is shown in Figure 15.5-8.

The maximum SG pressure met the acceptance criteria and is shown in Figure 15.5-9. The lowest MCHFR met the acceptance criteria and is shown on Figure 15.5-10. This event is not limiting for any of the three AOO acceptance criteria. This event does not lead to a more serious fault condition.

As the predicted MCHFR was greater than the acceptance criteria for this event, no fuel failure is predicted to occur. As a result, the radiological consequences for this event are acceptable as no fuel failure or release of primary coolant outside containment is predicted to occur.

15.5.1.4 Radiological Consequences

The radiological consequences of an increase in RCS inventory event are bounded by the design basis accident analyses presented in Section 15.0.3.

15.5.1.5 Conclusions

The regulatory acceptance criteria for AOO are met for the inadvertent increase in RCS inventory event. A comparison of the limiting pressure values and the associated acceptance criteria is shown in Table 15.5-5.

The DSRS acceptance criteria for this AOO are met for the increase in RCS inventory event. These acceptance criteria, followed by how the NuScale design meets them are listed below:

1) Pressure in the reactor coolant and main steam systems should be maintained below 110 percent of the design values.

    - The limiting RCS pressure for this event, as shown in Table 15.5-5, is below 110 percent of the design value for the reactor coolant system.
    - The limiting steam generator pressure, as shown in Table 15.5-5, is below 110 percent of the design value for the main steam system up to the MSIVs.

2) Fuel cladding integrity should be maintained by ensuring the minimum CHFR remains above the 95/95 CHFR limit based on acceptable correlations (See DSRS Section 4.4)

   - The MCHFR for this event, as shown in Table 15.5-5, is above the 95/95 limit.

3) An incident of moderate frequency should not generate a more serious plant fault.

   - The analyses presented for this event shows that a safe stabilized condition is reached, and the acceptance criteria are met.

4) The guidance provided in RG 1.105, "Instrument Spans and Setpoints," can be used to analyze the impact of the instrument spans and setpoints on the plant response to the type of transient addressed in this DSRS section, in order to meet the requirements of GDCs 10 and 15.

   - Allowances for instrument inaccuracy are accounted for in the analytical limits of mitigating systems in accordance with the guidance provided in Regulatory Guide 1.105.

5) The most limiting plant systems single failure, as defined in the "Definitions and Explanations" of Appendix A to 10 CFR Part 50, shall be identified as assumed in the analysis and shall satisfy the positions of RG 1.53.

   - No limiting single failure was identified that provided more limiting RCS pressure, SG pressure, or MCHFR. Results from this scenario do not challenge the identified limits.

6) The guidance provided in SECY-77-439, SECY-94-084 and RG 1.206 with respect to the consideration of the performance of nonsafety-related systems during transients and accidents, as well as the consideration of single failures of active and passive systems (especially as they relate to the performance of check valves in passive systems) must be evaluated and verified.

   - The inputs and assumptions for the operation of nonsafety-related systems and single failures as discussed in Section 15.5.1.2 ensure the guidance provided is met.

Table 15.5-1: Sequence of Events CVCS Malfunction - Limiting SG Pressure (Pressurizer Spray Available)

Event	Time [s]
Makeup Initiated	0
Analytical limit for high pressurizer level (80%) is reached	567.0
CVCS Isolation and RTS Actuation following high pressurizer Level analytical limit	570.0
Analytical Limit for High Steam Superheat Reached	570.8
CVCS Isolation Valves Shut	577.0
High steam line pressure analytical limit reached	578.1
Secondary System Isolation Signal	578.8
Secondary System Isolated	578.9
DHRS Actuation following high steam line pressure	580.1
Maximum secondary pressure reached	655.5

Table 15.5-2: Sequence of Events CVCS Malfunction - Limiting RCS Pressure (No Pressurizer Spray)

Event	Time [s]
Makeup Initiated	0
Analytical limit for high pressurizer pressure (2000 psia) is reached	188.7
RTS, DHRS, and Secondary System Isolation actuation following high pressurizer pressure analytical limit	190.7
Secondary System Isolated	190.8
RSV Actuates First Time	2700
RSV Actuates Second Time	3170
RSV Actuates Third Time	3531
Peak Primary Pressure Reached	3531
Analytical limit for pressurizer level (80%) is reached	3533
CVCS Isolation Actuation following pressurizer level analytical limit reached	3536
CVCS Isolation Valves Shut following high pressurizer level analytical limit	3543

Table 15.5-3: Sequence of Events CVCS Malfunction - Limiting MCHFR (No Pressurizer Spray)

Event	Time [s]
Makeup Initiated	0
Analytical limit for high pressurizer pressure (2000 psia) is reached	145.2
RTS, DHRS, and Secondary System Isolation actuation following high pressurizer pressure analytical limit	147.2
Low low RCS flow (0%) analytical limit reached	180.4
CVCS Isolation Signal	186.4
CVCS Isolation Valves Shut	193.4

Table 15.5-4: Initial Conditions CVCS Malfunction

Parameter	Limiting SG Pressure (With PZR Spray)	Limiting RCS Pressure (Without PZR Spray)	Limiting MCHFR (Without PZR Spray)
The initial RCS pressure	1920 psia	1920 psia	1920 psia
The initial RCS temperature	555°F	535°F	555°F
The initial PZR level	68% (+8%)	52% (-8%)	68% (+8%)
The initial feedwater temperature	300 °F	300 °F	300 °F
The drift on RSV setpoint	2137 psia (+3%)	2137 psia (+3%)	2137 psia (+3%)
CVCS isolation valve closure time	7 seconds	7 seconds	7 seconds
CVCS makeup fluid temperature	150°F	150°F	40°F
CVCS makeup flowrate	40 gpm	40 gpm	40 gpm
Moderator and Doppler coefficients of reactivity	-43.0/-1.40 pcm/°F	-43.0/-1.40 pcm/°F	-43.0/-1.40 pcm/°F
Maximum regulating control rod speed	15 in/min	15 in/min	15 in/min
RCS flow rate	1166 lbm/s	1193 lbm/s	1166 lbm/s

Table 15.5-5: Summary of Results CVCS Malfunction

Acceptance Criteria	Limit	Analysis Value
Maximum RCS Pressure (no PZR spray)	2310 psia	2160 psia
Maximum SG Pressure (PZR spray available)	2310 psia	1430 psia
MCHFR (no PZR spray)	1.284	2.702

Figure 15.5-1: Pressurizer Level - Increase in RCS Inventory (No PZR Spray)

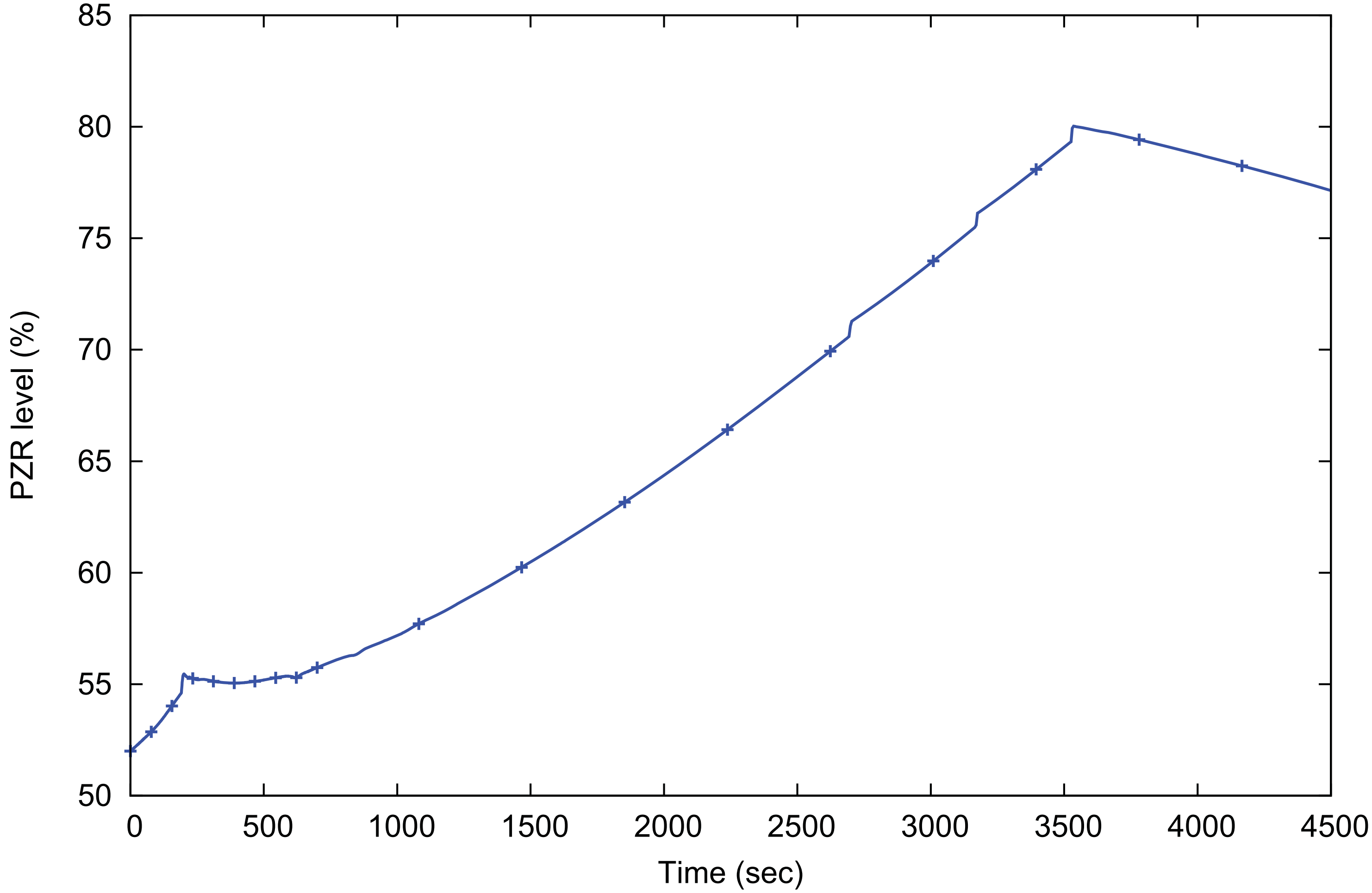


Figure 15.5-2: Reactor Power - Increase in RCS Inventory (No PZR Spray)

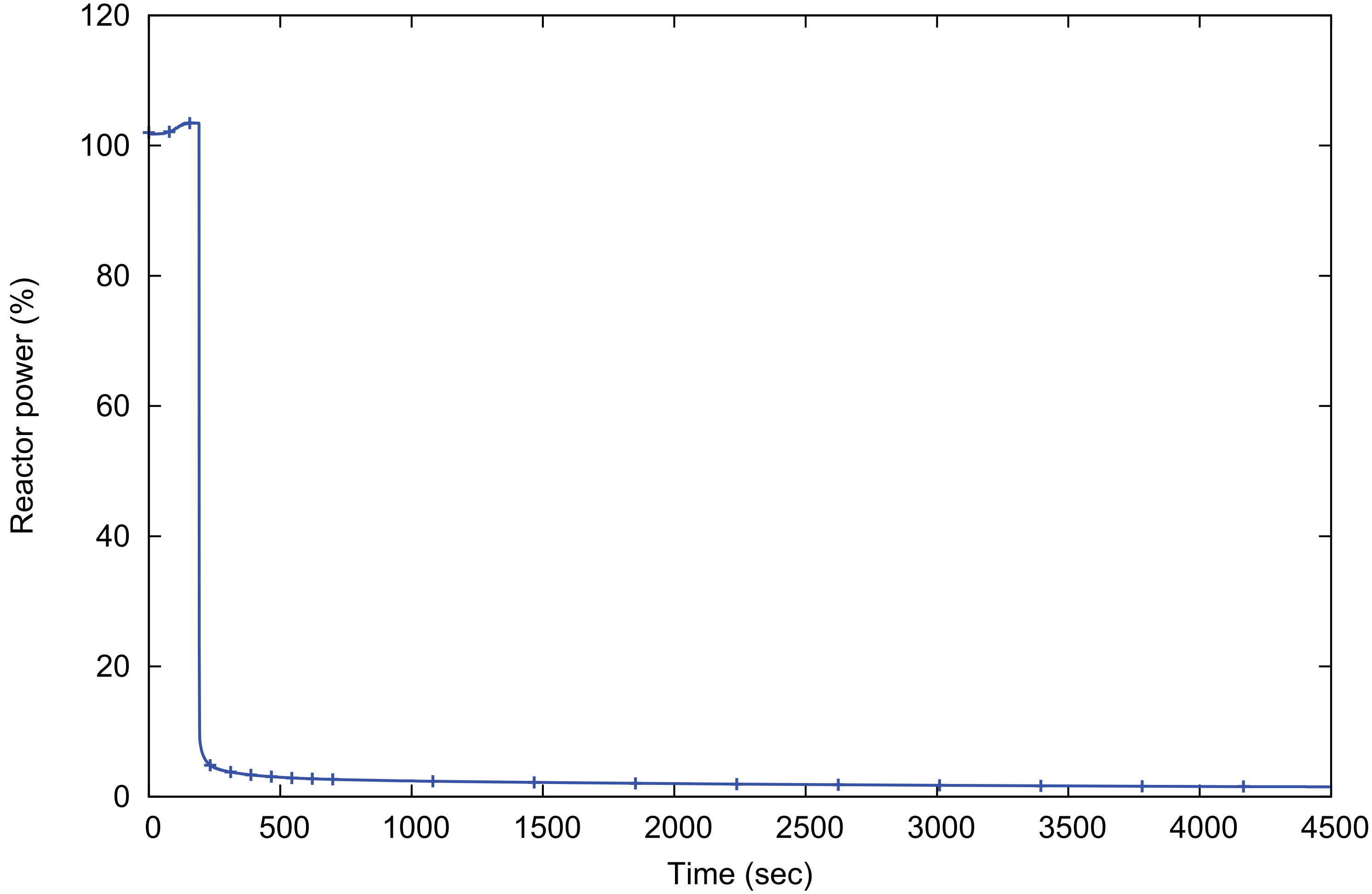


Figure 15.5-3: Total Reactivity - Increase in RCS Inventory (No PZR Spray)

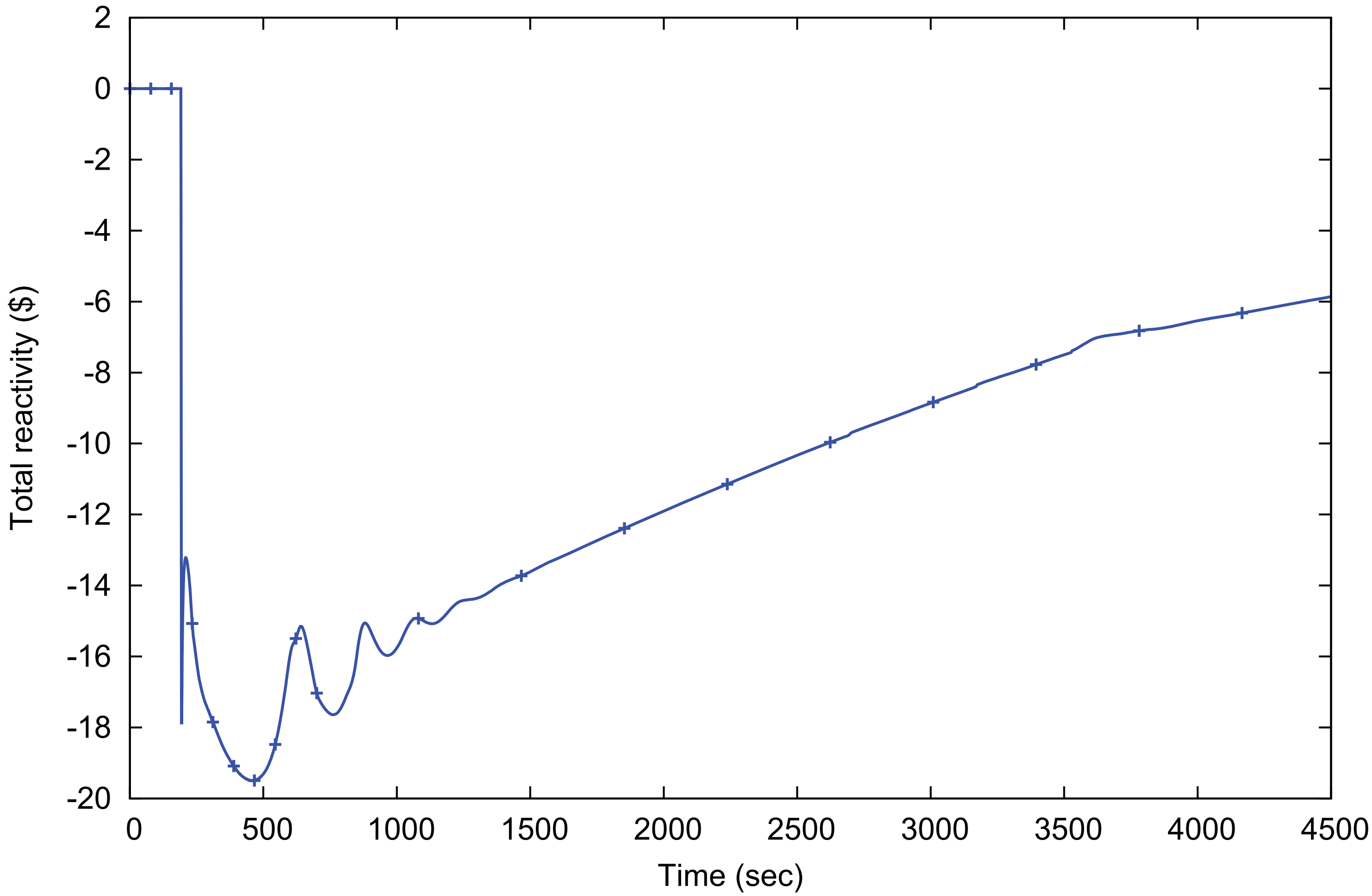


Figure 15.5-4: Reactor Coolant System Average Temperature - Increase in RCS Inventory (No PZR Spray)

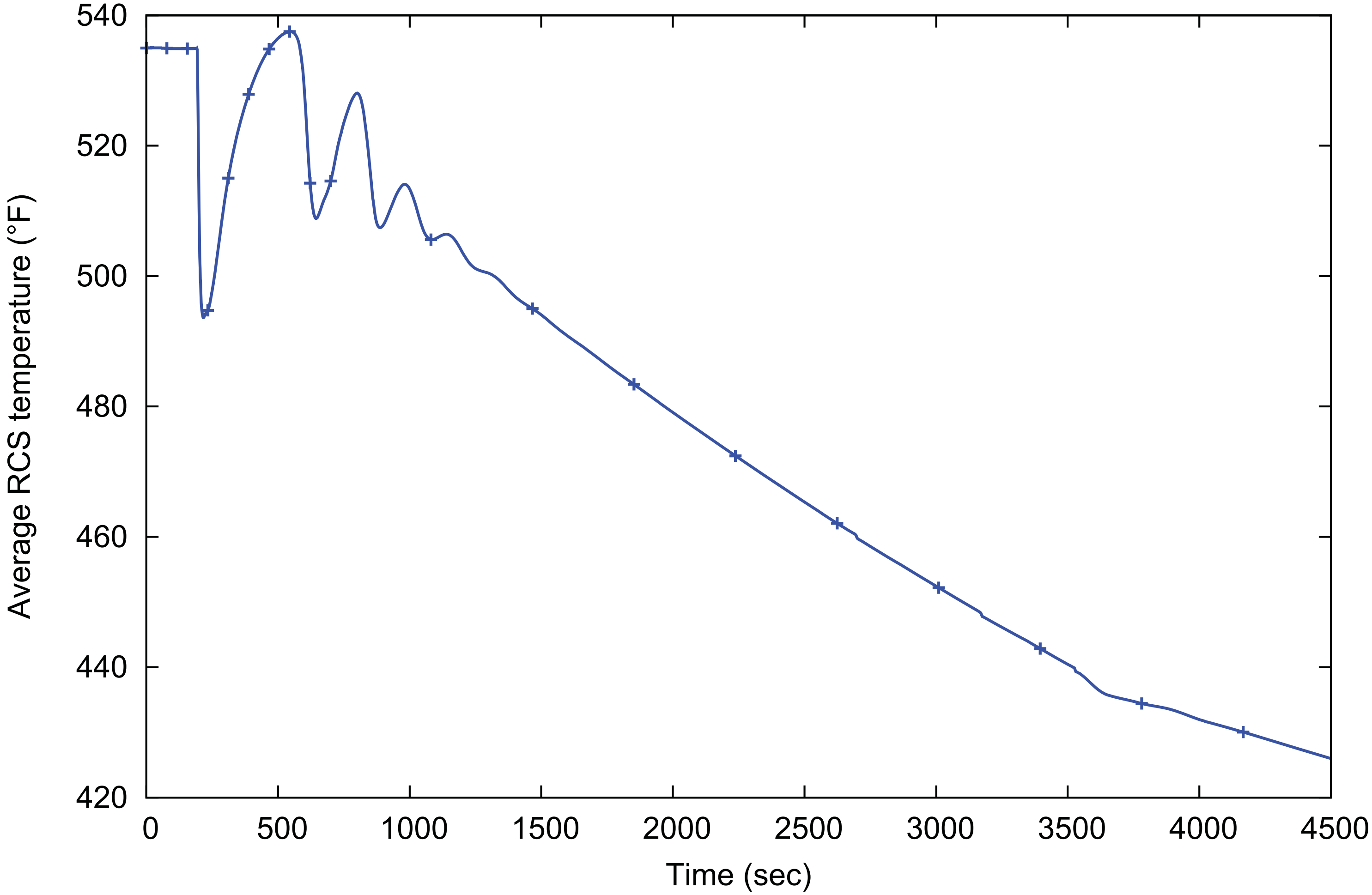


Figure 15.5-5: Makeup Flowrate - Increase in RCS Inventory (No PZR Spray)

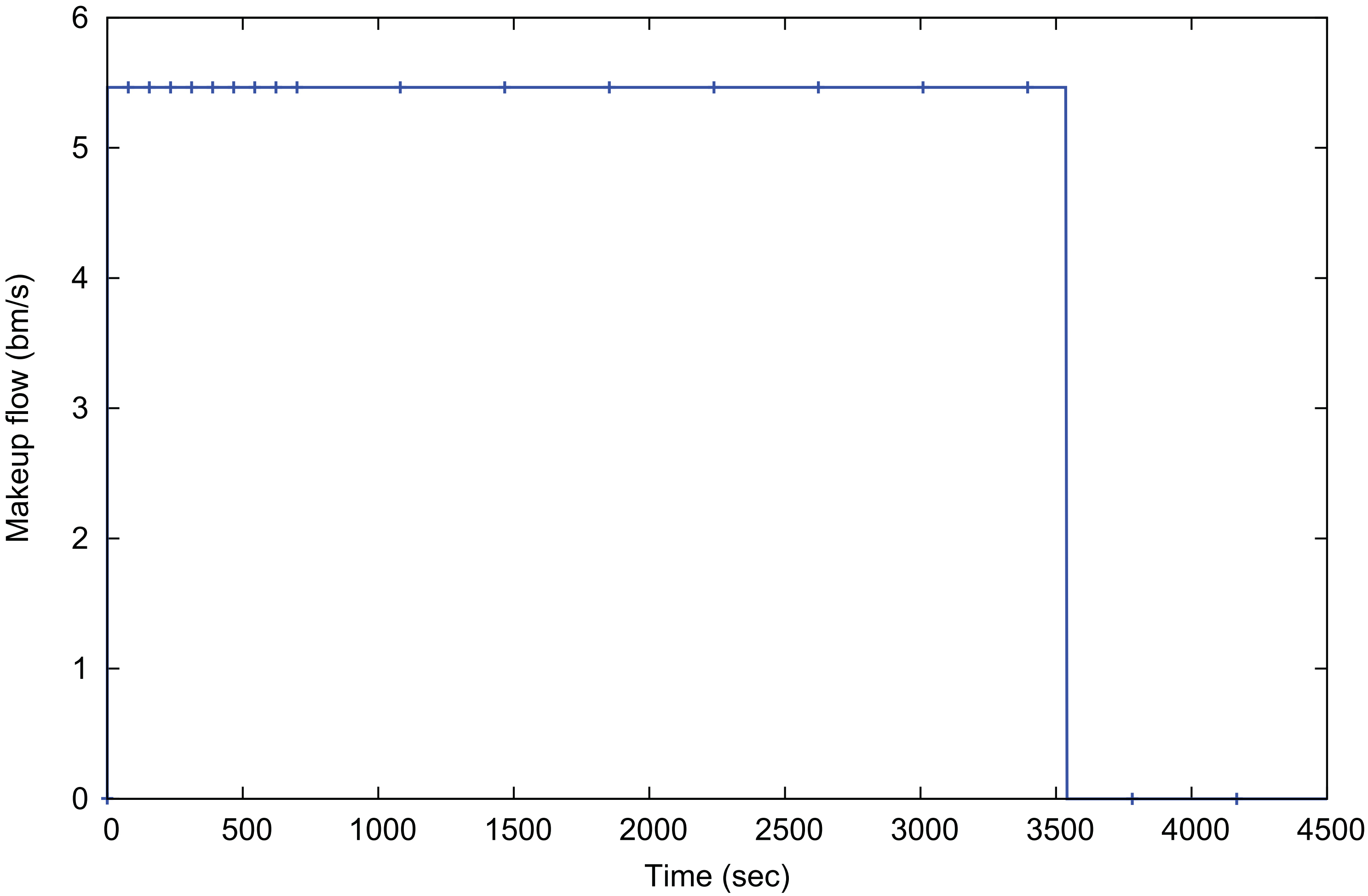


Figure 15.5-6: Reactor Coolant System Flow - Increase in RCS Inventory (No PZR Spray)

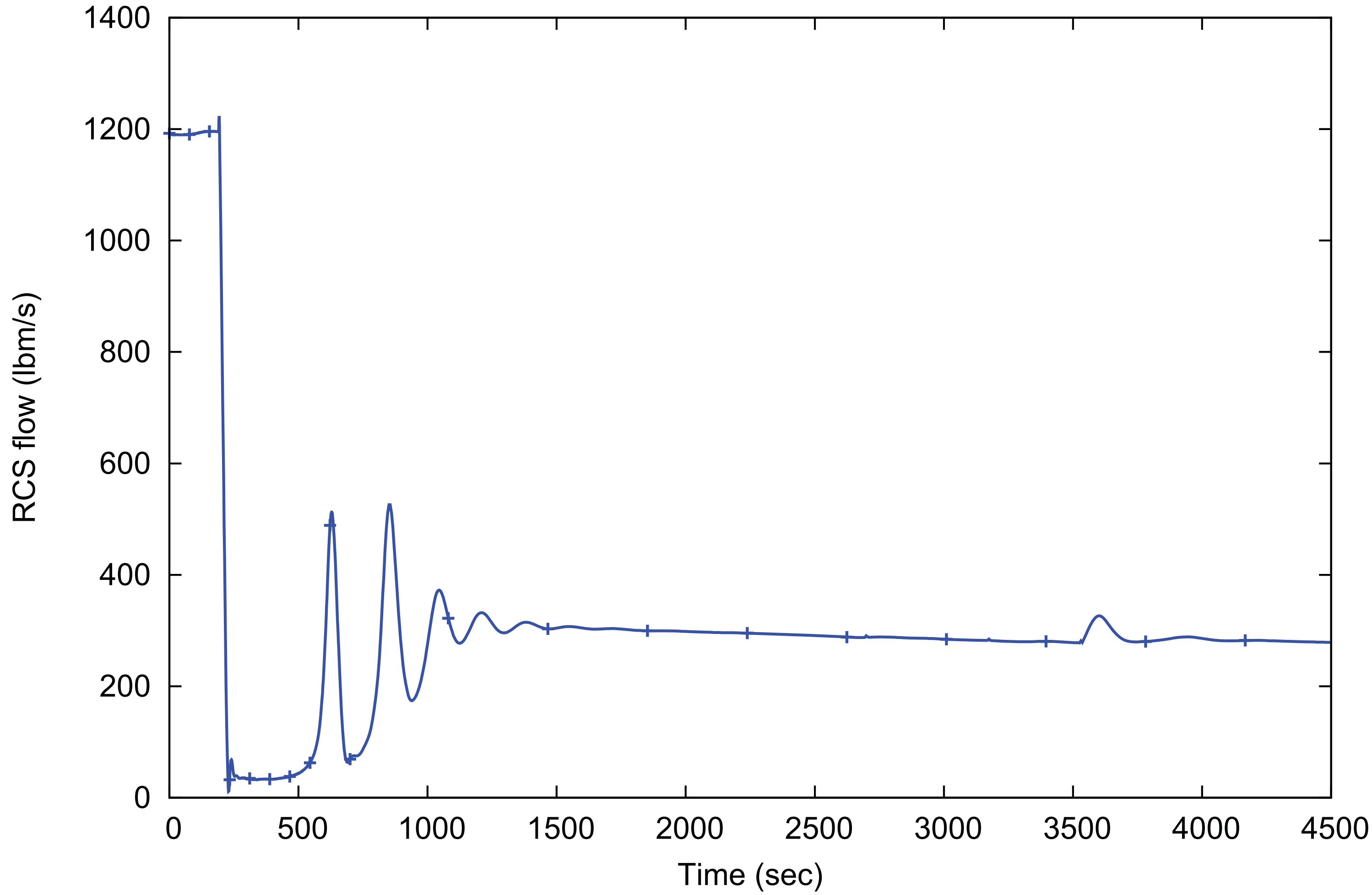


Figure 15.5-7: Decay Heat Removal System Flow (Typical) - Increase in RCS Inventory (No PZR Spray)

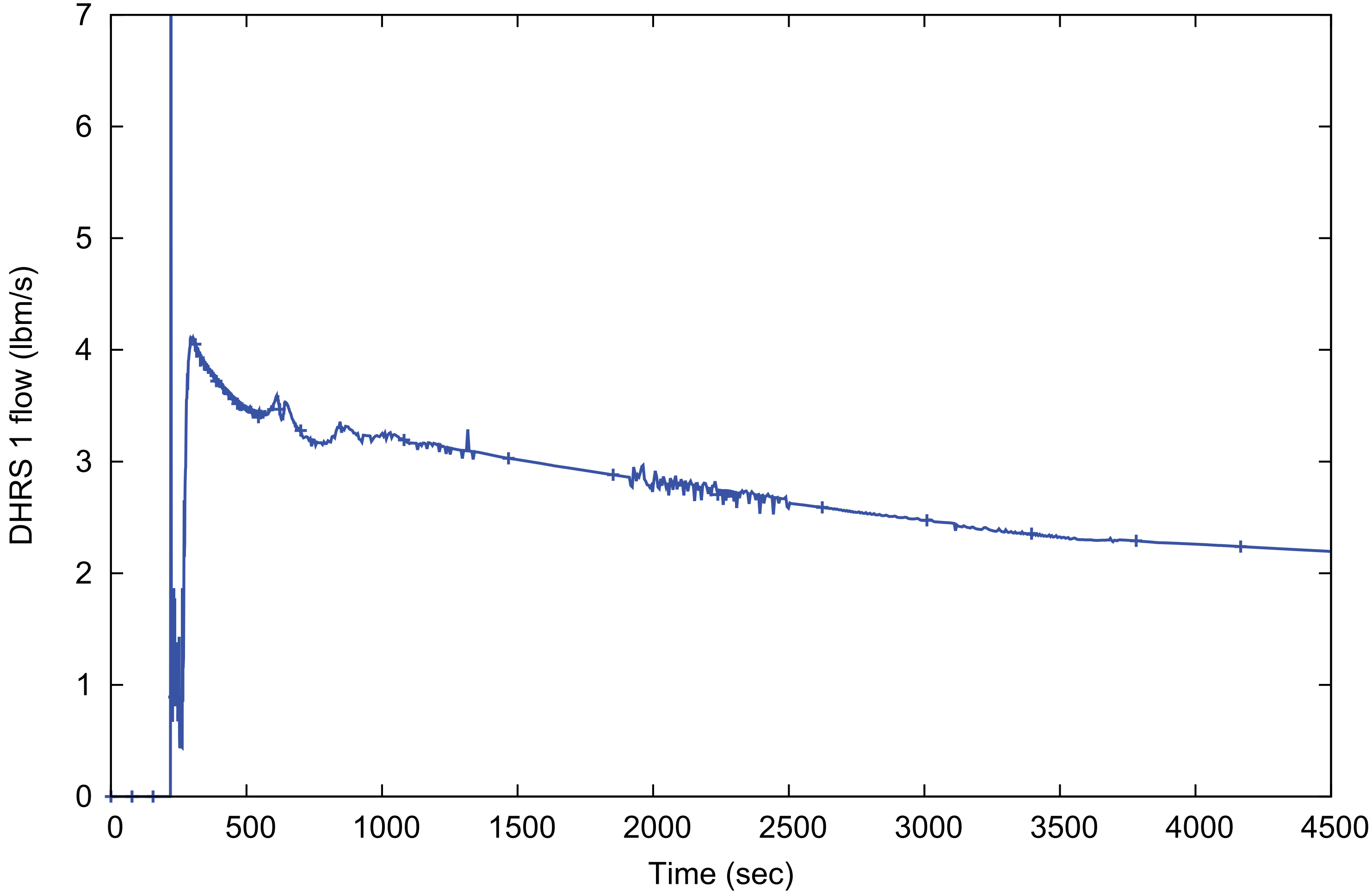


Figure 15.5-8: Reactor Pressure Vessel Lower Plenum Pressure - Increase in RCS Inventory (No PZR Spray)

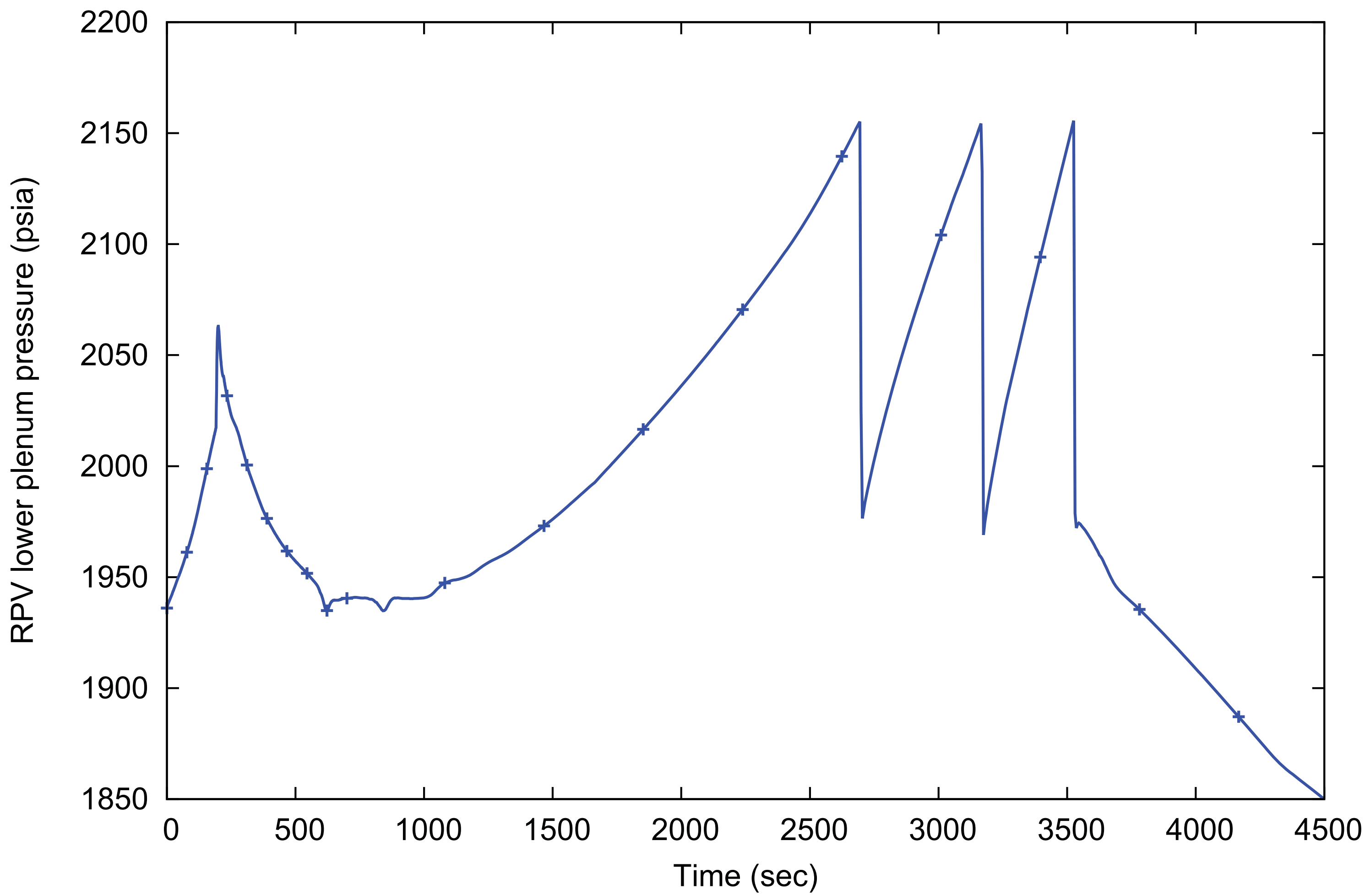


Figure 15.5-9: Steam Generator Pressure (Typical) - Increase in RCS Inventory (PZR Spray Available)

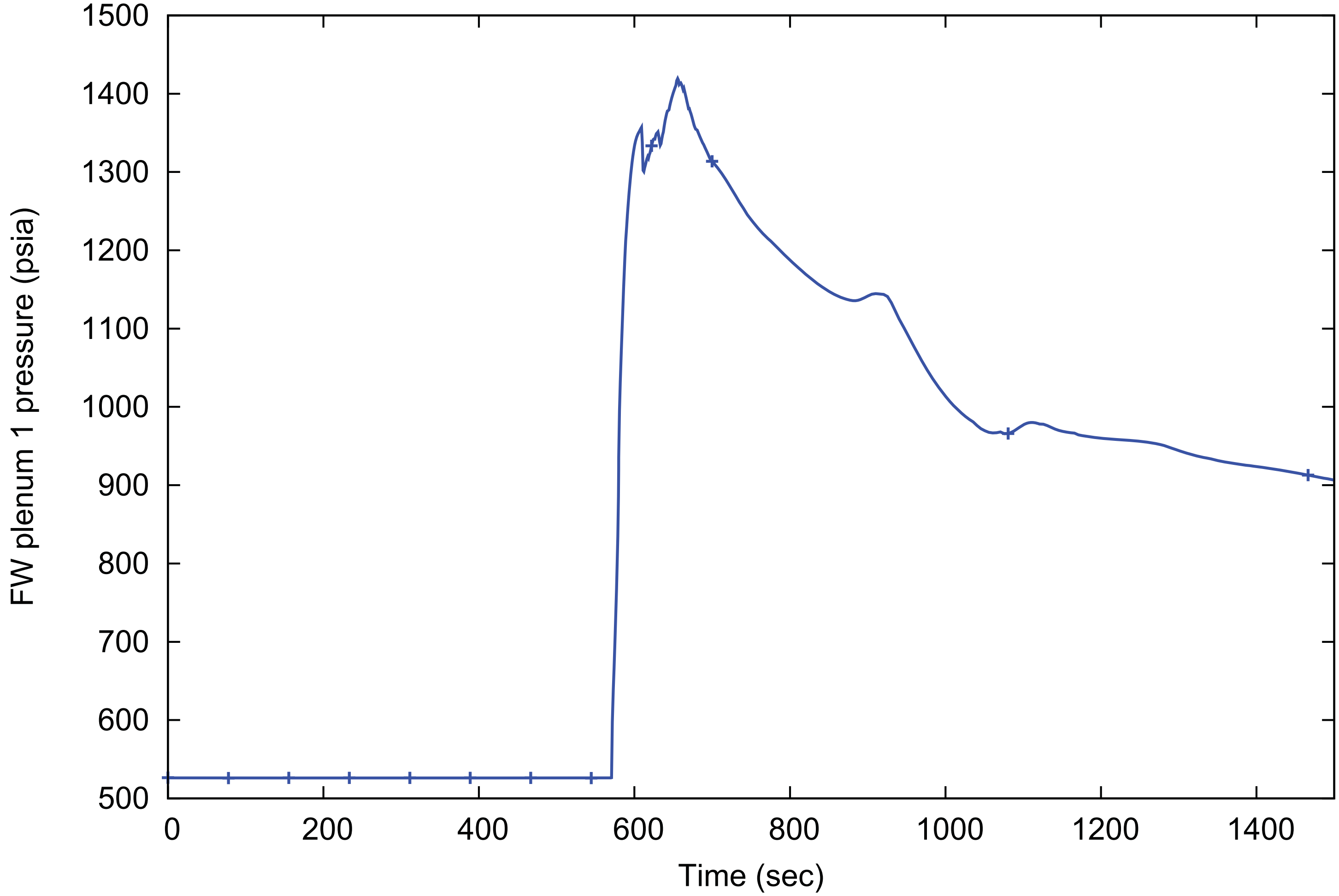
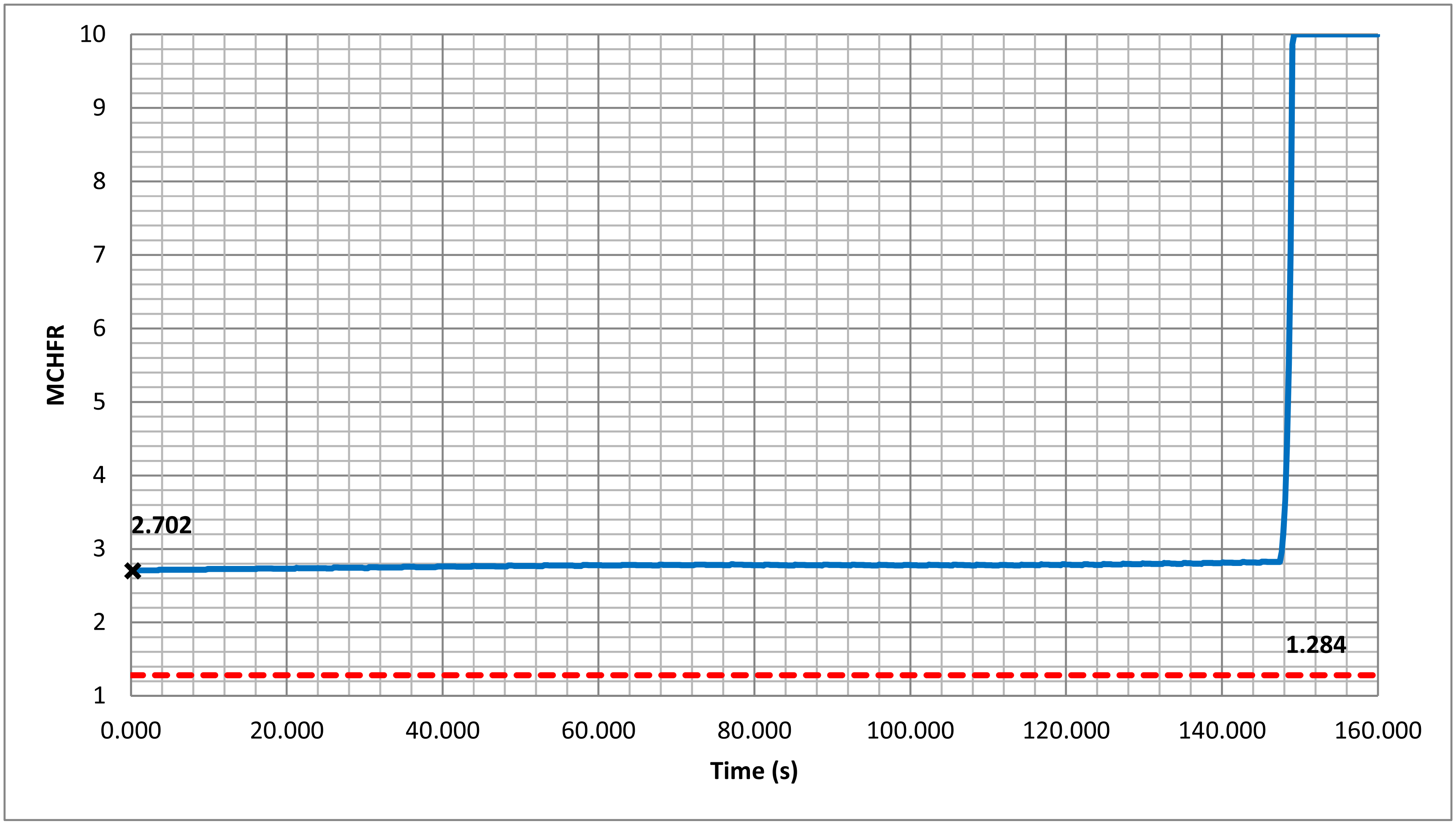


Figure 15.5-10: Minimum Critical Heat Flux Ratio - Increase in RCS Inventory (No PZR Spray)



15.6 Decrease in Reactor Coolant Inventory

This section addresses sign basis events associated with a potential unplanned decrease in reactor coolant system (RCS) inventory.

The following events are addressed in this section:

- Section 15.6.1 - Inadvertent Opening of a Reactor Safety Valve
- Section 15.6.2 - Failure of Small Lines Carrying Primary Coolant Outside Containment
- Section 15.6.3 - Steam Generator Tube Failure
- Section 15.6.4 - Main Steam Line Failure Outside Containment (BWR)
- Section 15.6.5 - Loss-of-Coolant Accidents Resulting From Spectrum of Postulated Piping Breaks Within the Reactor Coolant Pressure Boundary
- Section 15.6.6 - Inadvertent Operation of Emergency Core Cooling System

15.6.1 Inadvertent Opening of a Reactor Safety Valve

The reactor safety valves (RSVs) provide over-pressure protection of the NuScale Power Module (NPM). An inadvertent opening of an RSV has the same thermal hydraulic effects as an inadvertent opening of a reactor vent valve (RVV). This event can be caused by a mechanical valve failure. This event is classified as an anticipated operational occurrence (AOO) in Table 15.0-1.

The inadvertent RSV actuation event is bounded by the inadvertent opening of an RVV, which is a component of the emergency core cooling system (ECCS), due to the RVV opening being significantly larger. The larger opening size challenges the reactor pressure vessel (RPV) more in terms of mass and energy releases.

The inadvertent ECCS valve operation event is presented in Section 15.6.6.

15.6.2 Failure of Small Lines Carrying Primary Coolant Outside Containment

Lines that carry primary coolant outside containment are the chemical and volume control system (CVCS) lines: makeup and letdown lines, pressurizer spray lines, and RPV high point degasification line. The CVCS lines extend from the RPV and exit the containment vessel (CNV) through double containment isolation valves (CIVs). Failure of lines carrying primary coolant outside containment is analyzed for thermal hydraulic effects and radiological consequences. This event is classified as an infrequent event, as shown in Table 15.0-1.

15.6.2.1 Identification of Causes and Accident Description

Failure of lines carrying primary coolant outside containment is a non-mechanistic break in the CVCS makeup line, CVCS letdown line, or pressurizer spray line. The CIVs on the RPV high point degasification line are normally closed, therefore, a break in this line outside containment is not considered. Also, the pressurizer spray line and degasification line are the same size, are located at the top of the pressurizer, and exit through the containment head. A break in either of these lines was determined to be

bounded by the CVCS makeup and letdown lines, and therefore, are not addressed further.

To determine the most severe consequences of the failure of lines carrying primary coolant outside containment, a spectrum of break sizes and locations is evaluated. Primary coolant is released from the break into the Reactor Building (RXB) until CVCS CIVs close. The piping carrying primary coolant outside containment are not expected to fail during the life of the plant, so this event is classified as an infrequent event, as indicated in Table 15.0-1.

15.6.2.2 Sequence of Events and Systems Operation

The analysis considers the rupture of the CVCS makeup and letdown lines outside containment. Primary coolant is discharged from the line break into the RXB until a CVCS isolation signal occurs. The CVCS can be isolated from the RCS by two redundant safety-related CIVs located outside containment on the exit and entry containment penetrations. These valves isolate on a containment isolation signal, or a low-low pressurizer level, low-low pressurizer pressure or low-low-RCS flow signals. The fluid in the CVCS components in the RXB is assumed to drain out of the break and contribute to the radiological consequences of the event.

A spectrum of break sizes and break locations are analyzed to determine the most severe consequences. The analyses show that the reactor trips, and the decay heat removal system (DHRS) actuates to remove decay heat, but ECCS actuation setpoints are not reached. This means that once the CVCS CIVs close, the reactor coolant remains in the RPV, as opposed to discharging into containment for recirculation, and shutdown of the NPM proceeds using DHRS.

Therefore, the CVCS line breaks outside containment focuses on maximizing the primary coolant mass and energy release to the RXB to maximize the radiological consequences of the event and maximizing the RCS pressure for addressing acceptance criteria.

Three scenarios are identified:

- maximum mass release
- maximum iodine spiking time
- maximum RCS pressure

Table 15.6-1, Table 15.6-2, and Table 15.6-3 provide the sequence of events for the three scenarios.

15.6.2.3 Core and System Performance

15.6.2.3.1 Evaluation Model

The thermal hydraulic analysis of the plant response to the failure of lines carrying primary coolant outside containment is performed using NRELAP5. The NRELAP5

model is based on the design features of an NPM. The non-loss-of-coolant accident (non-LOCA) NRELAP5 model is discussed in Section 15.0.2.

15.6.2.3.2 Input Parameters and Initial Conditions

This evaluation considers the rupture of the CVCS makeup line or CVCS letdown line located outside the containment boundary. The assumptions and initial conditions of the evaluations are selected to maximize the severity of the accident by maximizing the mass and energy release out of the break, maximize the duration of the resultant iodine spike, and maximize RCS pressure. Unless specified below, the analyses assume the control systems and engineered safety features perform as designed, with allowances for instrument inaccuracy. No operator action is credited to mitigate the effects of a CVCS line break outside containment.

Table 15.6-4 provides inputs and assumptions for the three scenarios. The maximum mass release scenario is a double ended CVCS letdown line break and the maximum iodine spiking time scenario is an equivalent 100 percent cross-sectional area makeup line break. The maximum RCS pressure scenario is an equivalent 100 percent cross-sectional area makeup line break. The following are key input parameters:

- core power (102 percent) - The enthalpy in the riser, where the makeup line is located, and the density in the downcomer, where the letdown line is located, is maximized at the maximum power of 102 percent.
- pressurizer pressure (1920 psia) - In order to delay the low pressurizer pressure signal, which initiates CVCS isolation and terminates the break flow from the NPM, the nominal steady state pressure of 1850 psia is increased by the pressure uncertainty of 70 psia.
- pressurizer level (68 percent) - The pressurizer level is increased by the level measurement uncertainty of eight percent in order to delay the low pressurizer level trip, which could cause the reactor trip to occur. When the reactor trip occurs, the resulting cooling of the primary side water increases the rate of depressurization, which leads to a low pressurizer pressure trip and CVCS isolation, ending the transient. So, delaying the reactor trip delays the CVCS isolation resulting in more primary coolant flow from the break.
- A combination of core parameters is used to provide a limiting power response. Sensitivity cases show the beginning-of cycle (BOC) core parameters maximize mass release, iodine spiking time and RCS pressure. Table 15.0-8 provides the EOC and BOC moderator temperature and Doppler coefficients.
- Loss of power - Loss of power conditions, as described below, and no loss of power conditions are examined at the start of the event and concurrent with a reactor trip.
    - Loss of normal AC - The turbine is tripped and feedwater is lost. The module protection system (MPS) remains powered so safety systems are not automatically actuated. The small line failure outside of containment is detected by the MPS on low pressurizer pressure or low pressurizer level. When the turbine is tripped, the turbine stop valves close, leading to a decreased capacity of the steam generators to remove heat from the RPV.

This causes the pressurizer pressure to increase, the water density to decrease, and the pressurizer level to increase, which delays the event detection and maximizes the mass release, iodine spiking time, and RCS pressure. It also leads to a reactor trip followed by MPS signals to initiate containment isolation. By having the turbine available, as in the case with power available, the mass release and RCS pressure are lower than if a loss of normal AC occurs. Therefore, a loss of AC power at the start of the event is conservative, as confirmed in sensitivity studies.

  – Loss of the normal DC power system (EDNS) and normal AC - Power to the reactor trip breakers is provided via the EDNS, so, in addition to the above, a reactor trip occurs. Having the reactor trip closer to the time of event initiation leads to quicker containment isolation and reduced mass release. Thus, it is conservative to extend the reactor trip.
  – Loss of the highly reliable DC power system (EDSS), EDNS, and normal AC - Power to the module protection system (MPS) is provided via the EDSS, so this scenario results in an actuation of DHRS, the 24 hr timer for ECCS, and containment isolation. This scenario is non-conservative for the reasons outlined above.

- A single failure of the main steam isolation valve (MSIV) on one steam generator to close is included as a sensitivity case. The sensitivity shows that assuming no single failure is more conservative.

15.6.2.3.3 Results

Figure 15.6-1 to Figure 15.6-16 show the system response to the failure of lines carrying primary coolant outside containment. Table 15.6-5 contains the results of the event. The three limiting scenarios begin with a break of a CVCS line outside containment with a coincident loss of normal AC power. The system response from the breaks is similar, only the timing of the MPS signals varies as a result of the inputs and assumptions used to maximize the parameter of interest.

The maximum mass release scenario starts with a double-ended CVCS letdown line break outside containment with a coincident loss of normal AC power. The turbine stop valves close as a result of the loss of normal AC power increasing steam line pressure. A high steam line superheat signal initiates the reactor trip and actuates SSI which causes the MSIVs and feedwater isolation valves (FWIVs) to close. DHRS is actuated on high steam line pressure which causes the DHRS valves to open. To maximize the release for radiological purposes, a break in the makeup line is modeled at the time of reactor trip to increase the flow from the RCS to simulate the double-ended break. A low-low pressurizer pressure signal initiates closure of the containment isolation valves on the CVCS lines, isolating the break flow from the RCS. Figure 15.6-1 shows the break mass flow for the limiting case and Figure 15.6-2 shows the integrated break mass flow.

The maximum iodine spiking duration scenario starts with an equivalent 100 percent cross-sectional area break of the CVCS makeup line with a coincident loss of normal AC power. The turbine stop valves close as a result of the loss of normal AC power, increasing the steam line pressure and RPV pressure. The reactor

trips on high steam superheat signal and SSI is actuated. Subsequently, a high steam pressure signal actuates DHRS. As the system cools due to DHRS, a low pressurizer level signal occurs. The low-low pressurizer pressure signal initiates the CVCS containment isolation valves to close, isolating the break flow from the RCS.

The maximum RPV pressure scenario starts with an equivalent 100 percent cross-sectional area break of the CVCS makeup line with a coincident loss of normal AC power. The turbine stop valves close as a result of the loss of normal AC power, increasing the steam line pressure and RPV pressure (Figure 15.6-3 and Figure 15.6-4). A high pressurizer pressure signal occurs, initiating a reactor trip, SSI and DHRS actuation. The reactor trip is evident in the reactor power decrease depicted in Figure 15.6-5 and DHRS flow shown in Figure 15.6-6. The MSIV closure as a result of the SSI actuation signal causes a high steam line pressure signal. As the system cools due to DHRS, as shown in the pressure, temperature, and level response of Figure 15.6-3 and Figure 15.6-4, and Figure 15.6-7 through Figure 15.6-13, a low-low pressurizer pressure signal initiates the CVCS containment isolation valves to close, isolating the break flow from the RCS. The system continues to cool using DHRS (Figure 15.6-6), RCS flow stabilizes (Figure 15.6-14 and Figure 15.6-15), RCS temperature (Figure 15.6-7 and Figure 15.6-8) and fuel temperature (Figure 15.6-12 and Figure 15.6-13) stabilize and continue to decline, the core remains subcritical (Figure 15.6-16), and the water level is well above the top of the active fuel (Figure 15.6-11). The system response shows that the event has terminated and the NPM reaches a safe, stabilized condition.

15.6.2.4 Radiological Consequences

Section 15.0.3 provides the radiological consequences of a failure in small lines carrying primary coolant outside containment.

15.6.2.5 Conclusions

The acceptance criteria for an infrequent event are listed in Table 15.0-2. These acceptance criteria, followed by how the NuScale Power Plant design meets them are listed below.

1) Potential core damage is evaluated on the basis that it is acceptable if the minimum DNBR remains above the 95/95 DNBR limit. Minimum critical heat flux ratio (CHFR) is used instead of minimum DNBR, as described in Section 4.4.2.

   The fuel integrity is not challenged by a break of a CVCS line outside containment. The fuel temperatures decrease upon the reactor trip and DHRS actuation, as shown in Figure 15.6-13, and the water level remains above the top of the active fuel, as shown in Figure 15.6-11. In addition, the event is bounded by the rapid depressurization predicted during the inadvertent RVV opening event, which is analyzed for critical heat flux as presented in Section 15.6.6.

2) RCS pressure should be maintained below 120 percent of the design value. The design pressure for the reactor vessel is 2100 psia, thus the acceptance criterion is 2520 psia.

Table 15.6-5 presents the results of the three limiting scenarios. The RCS pressure, even for the limiting RCS pressure scenario, is below the acceptance criterion.

3) The main steam pressure should be maintained below 120 percent of the design value. The design pressure for the reactor vessel is 2100 psia, thus the acceptance criterion is 2520 psia.

   Table 15.6-5 presents the results of the three limiting scenarios. The main steam pressure, presented as steam generator pressure, is below the acceptance criterion.

4) The containment pressure should be maintained below the design pressure of 1050 psia.

   The failure of small lines carrying primary coolant outside containment is not an event that challenges containment pressure. Events that discharge RCS fluid directly inside containment bound this event. The peak containment pressure for design basis events is evaluated in Section 6.2.

5) The event should not generate a more serious plant condition without other faults occurring independently.

   The analysis presented for this event shows that stable DHRS cooling is reached and therefore the acceptance criterion is satisfied.

15.6.3 Steam Generator Tube Failure (Thermal Hydraulic)

15.6.3.1 Identification of Cause and Accident Description

A steam generator tube failure (SGTF) could be caused by a rapid propagation of a circumferential crack that leads to a double-ended rupture of the tube. Reactor coolant passes from the primary side of the SG to the secondary side where steam is produced and travels through the main steam lines to the turbine. Radionuclides contained in the primary coolant are discharged through the failed tube into the atmosphere until the faulted SG is isolated by automatic closure of the MSIVs. The design of the helical coil steam generators, described in Section 5.4, is different from the design of SGs in conventional pressurized water reactors because primary coolant is located on the outside, or shell side, of the tubes. Thus, following a tube failure, the primary coolant travels from the shell side of the SG into the tube through the break.

The categorization of the design basis events are discussed in Section 15.0.0.2. An SGTF is classified as a potential accident because it is not expected to occur during the lifetime of the NPM.

The SGTF analysis evaluates the primary and secondary system response to the transient to verify that the event meets the acceptance criteria specified in Table 15.0-2. The SGTF analysis also determines the mass of primary coolant that is released to the environment. The released mass is used to determine the radiological consequences of the SGTF event, which are addressed in Section 15.0.3.

15.6.3.2 Sequence of Events and Systems Operation

Sensitivity analyses are performed to identify limiting scenarios for the four areas of interest for the SGTF event. The four scenarios are:

- limiting mass release
- limiting iodine spiking time
- limiting RCS pressure
- limiting SG pressure.

The sequence of events for each of the four limiting SGTF scenarios are provided in Table 15.6-6, Table 15.6-7, Table 15.6-8, and Table 15.6-25. Unless otherwise specified, the analysis of an SGTF event assumes the plant control systems and engineered safety features perform as designed, with allowances for instrument uncertainty. No operator action is credited to mitigate the effects of a SGTF.

15.6.3.3 Core and System Performance

15.6.3.3.1 Evaluation Model

The thermal hydraulic analysis of the plant response to a SGTF is performed using NRELAP5. The NRELAP5 model is based on the design features of an NPM. The non-LOCA NRELAP5 model is discussed in Section 15.0.2.

15.6.3.3.2 Input Parameters and Initial Conditions

The input parameters and initial conditions used in the evaluation of the SGTF event are selected to provide a conservative calculation and to maximize the mass release out of the failed SG tube, maximize iodine spiking time, and maximize the RCS and SG pressure. Unless otherwise specified, the analysis of a SGTF event assumes that the plant control systems and engineered safety features perform as designed, with allowances for instrument uncertainty. No operator action is credited to mitigate the effects of a SGTF.

Table 15.6-9 provides inputs and assumptions for the four scenarios. The following are key input parameters:

- core power (102 percent) - A high biased power is conservative with respect to the mass release in that it leads to higher pressure differences between the primary and secondary system, thus resulting in a high break flow. It is also conservative with respect to primary and secondary side pressures.
- pressurizer pressure (1920 psia) - A higher initial pressure leads to a higher secondary pressure and a longer mass release before the low pressurizer pressure signal actuates a reactor trip signal and delays the low-low pressurizer pressure actuation of SSI that would isolate the faulted SG. Thus, the nominal steady state pressure of 1850 psia is increased by the pressure uncertainty of 70 psia.

- pressurizer level (68 percent) - The pressurizer level is increased by the level measurement uncertainty of eight percent in order to delay the low pressurizer level trip, which could cause the reactor trip to occur. When the reactor trip occurs, the resulting cooling of the primary side water increases the rate of depressurization, which leads to a low pressurizer pressure trip discussed above. Delaying the reactor trip delays SSI actuation on low-low pressurizer level or pressure, resulting in more primary coolant flow from the faulted SG.
- feedwater temperature - Sensitivity analyses show that applying a low bias to the feedwater temperature maximizes the mass releases. and maximizes SG pressure. Thus, the nominal feedwater temperature of 300 degrees F is reduced to 290 degrees F by the bias for the mass release and SG pressure cases. For the iodine spiking case, a high feedwater temperature bias (310 degrees F) maximized the spiking time.
- fuel kinetics - A combination of core parameters is used to provide a limiting power response. Sensitivity cases show that the beginning of cycle (BOC) core parameters maximize mass release, iodine spiking time and SG pressure. Table 15.0-8 provides the EOC and BOC moderator temperature and Doppler coefficients.
- RCS average temperature - The nominal RCS average temperature of 545 degrees F is either biased high or low by 10 degrees F, depending on which bias maximizes the parameter of interest. A higher RCS average temperature results in a higher SG pressure and more mass released and a lower RCS temperature results in higher iodine spiking times.
- SG pressure - Sensitivity analyses show that applying a low bias maximizes iodine spiking time. Thus, the nominal SG pressure of 500 psia is reduced by the bias to 465 psia for that case. Maximum mass release and SG pressure is achieved with a high SG pressure bias (535 psia).
- SG tube plugging - Sensitivity is performed on the impact of 10 percent SG tube plugging. Results show that no tube plugging leads to higher mass releases and higher iodine spiking time, while 10 percent tube plugging leads to higher SG pressures.
- Loss of power - No loss of power and a loss of normal AC power at the start of the event and concurrent with a reactor trip are examined.
  - Loss of normal AC - The turbine is tripped and feedwater is lost. The module protection system (MPS) remains powered so none of the safety systems are automatically actuated. When the turbine is tripped, the turbine stop valves close, leading to a decreased capacity of the steam generators to remove heat from the RPV. This causes the pressure to increase and the water density to decrease, which maximizes RCS and SG pressure. Closing the turbine stop valves leads to closure of the MSIVs. For the limiting mass release and iodine spiking time scenario, extending the time before isolating the faulted SG maximizes the mass released. Thus, it is conservative to assume that a loss of normal AC has not occurred for these scenarios and that the normal turbine bypass system controls steam pressure.

- Loss of the normal DC power system (EDNS) and normal AC - Power to the reactor trip breakers is provided via the EDNS, so, in addition to the above, a reactor trip occurs. A loss of EDNS does not change the system response resulting from a loss of normal AC power.
- Loss of the highly reliable DC power system (EDSS), EDNS, and normal AC - Power to the MPS is provided via the EDSS, so this scenario results in an actuation of a reactor trip, DHRS, SSI, the 24 hr timer for ECCS, and containment isolation. A loss of EDSS does not change the system response resulting from a loss of normal AC power.

- single failure - A single failure of the MSIV on the faulted steam generator to close is assumed for the limiting mass release and iodine spiking time scenarios. For the pressure scenarios, the sensitivity analyses show that steam line pressure and RPV pressure is greater when the MSIV closes. Therefore, assuming no single failure is more conservative for the pressure scenarios.
- SG tube failure location - Sensitivity analyses show that a tube failure at the bottom of the SG results in higher RCS and SG pressure, higher mass releases and higher iodine spiking time.
- Allowances for instrument inaccuracy are accounted for in the analytical limits of mitigating systems in accordance with the guidance provided in Regulatory Guide 1.105.
- The highest worth control rod is assumed to be stuck at the fully withdrawn position.

15.6.3.3.3 Results

Figure 15.6-17 to Figure 15.6-40 show the system response to the SGTF. Table 15.6-10 and Table 15.6-11 contain the results of the event. The four limiting scenarios begin with a failure of a SG tube. The system response from the tube failure is similar in the four scenarios, only the timing of the MPS signals vary as a result of the inputs and assumptions used to maximize the parameter of interest.

The limiting mass release scenario starts with a partial tube failure at the bottom of the steam generator. The mass flow from the failed tube rapidly rises to the maximum mass flow rate. The reactor pressure and pressurizer level decreases, as well as the mass flow from the failed tube, until the MPS initiates a reactor trip and pressurizer heater trip on a low pressurizer level signal, and SSI is actuated at the pressurizer low-low level setpoint. The primary to secondary flow through the failed SG tube is shown in Figure 15.6-17. Figure 15.6-18 shows the integrated break mass release from the faulted SG. The RCS and SG pressure are shown in Figure 15.6-21.

SSI actuation occurs and includes the coincident closure signal of the MSIVs, secondary MSIVs, FWIVs, and FWRVs. The MSIV on the faulted SG is assumed to fail open, extending the RCS mass flow from the faulted SG until the secondary MSIV closes 7 seconds after the SSI actuation signal. Figure 15.6-19 and Figure 15.6-20 show the steam generator level, which is below 25 percent at the time of the secondary MSIV closure, thus the valve closes in a steam environment and SG

overfill occurs well after secondary MSIV isolation. The system response continues as described for the limiting pressure cases below.

The limiting RPV pressure scenario begins with a partial tube failure at the bottom of the SG with a coincident loss of normal AC power, closing the turbine stop valves. The integrated mass flow through the break is shown in Figure 15.6-37. The SG heat removal capability is degraded after the SGTF. The water in the RPV expands causing a pressure increase until the MPS actuates a reactor trip, SSI and DHRS actuation on a high pressurizer pressure signal. The RCS and SG pressure response is shown in Figure 15.6-22 and Figure 15.6-23. The reactor trip is evident in the core power decrease depicted in Figure 15.6-24 and Figure 15.6-25 and successful DHRS actuation is evident in DHRS flow shown in Figure 15.6-26 and Figure 15.6-27. The SSI actuation includes closure of the MSIVs, isolating the break flow through the tube to the environment. Because MSIVs close before secondary MSIVs resulting in maximizing the system pressure, it is more conservative to assume that the MSIVs function as designed. Figure 15.6-28 shows the SG level, which is below 25 percent at the time of the MSIV closure, thus the valve closes in a steam environment and SG overfill occurs well after MSIV closure.

As the NPM cools with DHRS, as shown in the pressure, temperature, and level response depicted in Figure 15.6-22 and Figure 15.6-23, and Figure 15.6-31 through Figure 15.6-37, the pressurizer level decreases. The system continues to cool using DHRS (Figure 15.6-27), RCS flow stabilizes, RCS temperature (Figure 15.6-31 and Figure 15.6-32) and fuel temperature (Figure 15.6-36 stabilize and continue to decline, the core remains subcritical and the water level is well above the top of the active fuel (Figure 15.6-35). The system response shows that the event has terminated and the NPM reaches a safe, stabilized condition.

The limiting SG pressure scenario begins with a tube failure at the bottom of the steam generator with a concurrent loss of normal AC power. The progression of the scenario is similar to the limiting RPV pressure scenario described above. The timing of the MPS signals, reactor trip, SSI actuation, DHRS actuation and MSIV closure changes due to the parameters and assumptions that maximize the SG pressure. The RCS and SG pressure response is shown in Figure 15.6-38. The RCS temperature response and DHRS mass flow are shown in Figure 15.6-39 and Figure 15.6-40, respectively.

The limiting iodine spiking time case provides the longest time between reactor trip and secondary system isolation. The limiting iodine spiking time case is a partial SGTF split break of a tube located at the bottom of the SG, and assumes no loss of offsite power and a single failure of one primary MSIV. The turbine bypass valves were enabled to control steam pressure post reactor trip, which delays secondary side isolation. The sequence of events for this case is provided in Table 15.6-25. The mass release to the SG and the SG levels are shown in Figure 15.6-29 and Figure 15.6-30 respectively.

The MPS is credited to protect the plant in the event of SGTF. Note that the high steam superheat and high RCS temperature signals were not credited in this event. The following MPS signals provide the plant with protection during an SGTF:

- high pressurizer pressure,
- high steam line pressure
- low low pressurizer pressure
- low low pressurizer level
- low pressurizer pressure, and
- low pressurizer level.

The MSIVs and the secondary MSIVs are credited for isolating the faulted SG, depending on the scenario. The MSIVs and secondary MSIVs are designed for the conditions analyzed. The MSIVs and secondary MSIVs are designed to close in design basis conditions. Classification information for the MSIVs and secondary MSIVs are listed in Section 3.2, Table 3.2-1.

15.6.3.4 Radiological Consequences

Table 15.6-11 provides the inputs to the SGTF radiological consequence analysis presented in Section 15.0.3.

15.6.3.5 Conclusions

The acceptance criteria for a potential accident are listed in Table 15.0-2. These acceptance criteria, followed, by how the NuScale Power Plant design meets them, are listed below.

1) Potential core damage is evaluated on the basis that it is acceptable if the minimum DNBR remains above the 95/95 DNBR limit. Minimum critical heat flux ratio (CHFR) is used instead of minimum DNBR, as described in Section 4.4.2.

   The fuel integrity is not challenged by a SGTF. The fuel temperatures decrease upon the reactor trip and DHRS actuation, as shown in Figure 15.6-36, and the water level remains above the top of the active fuel, as shown in Figure 15.6-35. In addition, the event is bounded by the rapid depressurization predicted during the inadvertent RVV opening event, which is analyzed for critical heat flux and presented in Section 15.6.6.

2) RCS pressure should be maintained below 120 percent of the design value. The design pressure for the reactor vessel is 2100 psia, thus the acceptance criterion is 2520 psia.

   Table 15.6-10 presents the results of the four limiting scenarios. The RCS pressure is below the acceptance criterion.

3) The main steam pressure should be maintained below 120 percent of the design value. The design pressure for the reactor vessel is 2100 psia, thus the acceptance criterion is 2520 psia.

   Table 15.6-10 presents the results of the four limiting scenarios. The main steam pressure, presented as steam generator pressure, is below the acceptance criterion.

4) The containment pressure should be maintained below the design pressure of 1000 psia.

   An SGTF is not an event that challenges containment pressure. Events that discharge RCS fluid directly inside containment bound this event. The peak containment pressure for design basis events is evaluated in Section 6.2.

5) The event should not generate a more serious plant condition without other faults occurring independently.

   The analysis presented for this event shows that stable DHRS cooling is reached and therefore the acceptance criterion is satisfied.

15.6.4 Main Steam Line Failure Outside Containment (BWR)

This event is a BWR-specific event and not applicable to the NuScale Design.

15.6.5 Loss-of-Coolant Accidents Resulting from Spectrum of Postulated Piping Breaks within the Reactor Coolant Pressure Boundary

A LOCA is an event that compromises the reactor coolant pressure boundary (RCPB), resulting in RCS inventory loss at a rate that exceeds the capacity of normal makeup flow. A spectrum of break sizes and locations of the RCS pressure boundary piping are assessed. Table 15.6-18 provides the spectrum of break sizes and locations.

A LOCA for the NuScale design is unique compared to traditional large light water reactors because the diameters of the RCS piping are small so there is no distinction between "large break LOCA" and "small break LOCA" scenarios. In addition, RCS inventory is preserved within containment and available for recirculation soon after event initiation. A LOCA is analyzed for thermal hydraulic effects and is classified as a potential accident, as shown in Table 15.0-1. Inadvertent opening of an ECCS valve is not considered a LOCA and is addressed in Section 15.6.6.

15.6.5.1 Identification of Causes and Accident Description

A LOCA is a postulated accident that is initiated by a non-mechanistic break in a pipe inside containment connected to the RPV. The break location and break size determines the rate of RCS inventory loss and depressurization rate. Thus a spectrum of break sizes is postulated to occur at various locations in the pipelines penetrating the RCPB, as shown in Table 15.6-18. The postulated break sites are the rupture of the RCS injection and discharge lines, high point vent line, and pressurizer spray supply lines inside of containment.

15.6.5.2 Sequence of Events and Systems Operation

The initiating event for this transient is a rupture of the RCS injection or discharge line, RPV high point vent line, or pressurizer spray supply line inside of containment. The LOCA break spectrum is separated into two categories: a liquid space break consisting of the RCS injection line and discharge line; and a steam space break consisting of the high point vent line and pressurizer spray supply line.

A steam space break initiates a blowdown of the RCS inventory into the CNV from the top of the RPV. A liquid space break causes blowdown of the RCS inventory into the CNV from the liquid filled region of the RPV. The progression of the event, including the actuation of the engineered safety features, is similar to a steam space break with the exception of different timing of the key events and the liquid/steam composition of the break flow.

Table 15.6-12 shows the sequence of events for the limiting LOCA. The MPS is credited to initiate the reactor trip, isolate containment, and initiate DHRS, SSI and ECCS. DHRS is not credited for cooling following a LOCA. No operator action is credited in this event analysis.

The transition from the LOCA analysis to the post-LOCA long-term core cooling phase occurs when natural circulation between the RPV and the containment through the RVVs and RRVs has reached a stable state and decay and residual heat is being removed. The purpose of the post-LOCA long term cooling evaluation is to show that continued cooling occurs without boron precipitation for at least 72-hours after the initiation of a LOCA.

15.6.5.3 Core and System Performance

15.6.5.3.1 Evaluation Model

The thermal hydraulic analysis of the plant response to a LOCA uses NRELAP5. Section 15.0.2 provides details on the modeling requirements and code modifications needed to appropriately capture the phenomena and features of the LOCA evaluation model. Section 15.0.2 discusses the LOCA Evaluation Model Development and Assessment Process (EMDAP). Utilizing the results of the break spectrum, the methodology demonstrates that the design and operating conditions analyzed will result in a safe condition of a NPM for postulated design basis LOCAs.

The post-LOCA long-term core cooling analysis is performed using the NRELAP5 model to support the ECCS long term cooling methodology. A spectrum of cases is performed to encompass minimum and maximum cooldown scenarios. The results of the long-term core cooling analysis are then compared to the acceptance criteria developed for evaluating the margin to boron precipitation to show that boron precipitation is avoided during the long-term core cooling phase.

For the boron precipitation portion of the analysis, the following methodology is used. The determination of the boron precipitation temperature for a given mixing volume starts with the calculation of the entire mass of boron in the RCS. A corresponding concentration is calculated for the mixing volume assumption. Finally, the precipitation temperature is obtained for the mixing volume concentration using the boron precipitation curve. These calculations are performed for various mixing volumes corresponding to various elevation of liquid levels above the core. Temperature and level results from the long-term core cooling calculation are compared to the boron precipitation results to determine if boron precipitation could occur.

15.6.5.3.2 Input Parameters and Initial Conditions

The input parameters and initial conditions used in the LOCA analysis are selected to provide a conservative calculation. The parameter of interest for the LOCA is the collapsed liquid level above the core. Thus, inputs and assumptions are chosen to determine the minimum collapsed liquid level above the core. As shown in Table 15.6-19 through Table 15.6-24, the 5-percent RCS injection line break is limiting for maintaining the collapsed liquid level above the core.

Unless otherwise specified, the LOCA analysis assumes that the plant control systems and engineered safety features perform as designed, with allowances for instrument uncertainty. No operator action is credited to mitigate the effects of a LOCA for the duration of the event, including the post-LOCA long-term core cooling phase.

Table 15.6-13 provides inputs and assumptions for the limiting-break LOCA analysis. The following are key input parameters common to the spectrum of breaks analyzed in Table 15.6-18:

- Initial power level is assumed to be 102 percent of nominal to account for a 2 percent instrumentation uncertainty.
- RCS average temperature is initialized at a temperature of 555 degrees F to maximize RCS energy.
- Pressurizer pressure is biased high to maximize the RCS energy and containment peak pressure.
- Pressurizer level is biased low to minimize inventory availability.
- Containment pressure is biased high to maximize containment peak pressure.
- Main steam pressure is biased high to maximize overall system energy.
- Feedwater temperature is biased high to maximize the overall system energy and limit heat transfer to the steam generators.
- RCS flow is biased low to maximize RCS energy.
- The ECCS IAB release pressure is assumed to be at the lowest value of 1000 psid. Using the minimum value of the release pressure results in the lowest minimum collapsed water level.
- Bypass flow through the reflector and guide tubes is maximized to 8.5 percent of the total core flow to minimize flow through the hot assembly. This value is consistent with the subchannel analysis methodology discussed in Section 15.0.2.
- Reactor pool temperature is assumed to be 140 degrees F to reduce heat sink potential. Sensitivities demonstrate that the initial reactor pool temperature has negligible impact on the LOCA acceptance criteria.
- A minimum reactor pool level is applied to reduce heat sink potential.
- Maximum radial peaking factors are applied to the hot channel and hot pin models, consistent with Reference 15.6-1.

- A bounding bottom peaked axial power shape is applied to the hot channel and the average channel. Sensitivities show that different axial power shapes have a negligible impact on the LOCA acceptance criteria.
- Minimal reactivity feedback coefficients are conservatively applied, consistent with Reference 15.6-1.
- Beginning-of-cycle kinetics parameters with an additional 6 percent biasing are used in order to prolong the fission power transient, consistent with Reference 15.6-1.
- An energy deposition factor of 1.0 is implemented such that all the core power is conservatively deposited in the fuel.
- The decay heat standard for the LOCA evaluation model are described in Section 15.0.2.
- The reactor trip (SCRAM) reactivity insertion is calculated to account for the moderator and Doppler temperature defects with 2 percent shutdown margin at a temperature of 420 degrees F. This includes the assumption of the highest worth rod stuck at the fully withdrawn position.
- The following loss of power scenarios are considered:
  - No loss of power - In this scenario, MPS actuations occur as designed.
  - Loss of normal AC power - When normal AC power is lost, the feedwater pumps coast down and a turbine trip is initiated, thus limiting RCS cooling via the secondary system. Reactor trip, containment isolation, and DHRS actuation occur after a 60-second delay following a loss of normal AC power and ECCS actuation occurs after a 24-hour delay following the loss of normal AC power. However, because DC power is still available, the MPS can still actuate these safety functions, including reactor trip, earlier if a separate actuation limit is reacted. DHRS is not credited in the LOCA analysis. The event sequence for a loss of normal AC power is similar to that when no power is assumed lost. The primary difference is an earlier termination of secondary cooling. Thus, a loss of normal AC power conservatively maximizes the RCS thermal conditions after event initiation.
  - Loss of normal DC power (EDNS) and normal AC- Power to the reactor trip breakers is provided via the EDNS, so the primary difference to a loss of normal AC power is that the reactor trip will occur sooner. Delaying the reactor trip maximizes the RCS thermal conditions. Therefore, a loss of normal AC is more conservative.
  - Loss of the highly reliable DC power system (EDSS), EDNS, and normal AC - Power to the MPS is provided via the EDSS, so this scenario results in an immediate actuation of the reactor trip system, DHRS (although not credited in the LOCA analysis), the 24-hour timer for the ECCS valves, and containment isolation. This assumption is less conservative for the reasons discussed above.
- Single failure evaluations of the failure of a single RVV to open, a failure of a single RRV to open, and failure of one ECCS division (one RVV and one RRV) to open were performed to determine the most conservative scenario. The

evaluations show that the sensitivity to single failure is small with no clear limiting case. Therefore, the LOCA analysis is performed with no single failure and the limiting result is conservatively rounded down.

- Allowances for instrument inaccuracy are accounted for in the analytical limits of mitigating systems in accordance with the guidance provided in Regulatory Guide 1.105.

The input parameters and initial conditions used in the limiting case for the post-LOCA long-term core cooling analysis are also selected to provide a conservative calculation. The limiting criterion is minimum collapsed liquid level above the core during the long-term cooling period and occurs for a 100-percent RCS injection line break with the following inputs and assumptions.

- A 1.2 multiplier to decay heat.
- ECCS valve flow capacity biased low.
- A minimum reactor pool temperature of 65 degrees F to increase heat transfer.
- The reactor pool level of 69 feet.
- A single failure of ECCS division (one RRV and one RVV).
- A loss of EDSS, EDNS, and normal AC.
- Zero non-condensable gases are modeled.

15.6.5.3.3 Results

The LOCA analysis is performed for a spectrum of break sizes and break locations (Table 15.6-18) to determine the location and size of the break that is limiting for maintaining the collapsed liquid level above the core. Table 15.6-19 through Table 15.6-22 provide the results for each of the analyzed breaks. From these results, the 5-percent cross-sectional area break of the RCS injection line has the minimum collapsed level above the top of active fuel (TAF).

The 5-percent injection line break was then analyzed for different power scenarios, as shown in Table 15.6-23. The limiting power scenario is the loss of normal AC power, which is then analyzed for different single failure scenarios, as shown in Table 15.6-24.

Thus, the limiting scenario begins with an equivalent 5 percent cross-sectional area break of the RCS injection line with a loss of normal AC and no single failure. The results presented for the LOCA are for the limiting event. Figure 15.6-41 to Figure 15.6-54 show the system response to a LOCA. Table 15.6-14 contains the results of the limiting LOCA event.

Upon initiation of a LOCA, the RCS inventory flows out of the break into the containment (Figure 15.6-41). A coincident loss of normal AC power is assumed at time zero. A loss of normal AC power stops feedwater flow, thus terminating RCS cooling via the secondary system. The reactor trip does not occur until 60 seconds after the loss of normal AC power or until a separate MPS analytical limit is reached. Given the small break size and the loss of secondary cooling, the RCS undergoes a

short-term pressurization while the reactor is still at power (Figure 15.6-42). The increasing RCS pressure reaches the MPS high pressurizer pressure setpoint causing the reactor trip, as evident in Figure 15.6-43 and Figure 15.6-44. A high pressurizer pressure signal also initiates the isolation of the SGs by closing the MSIVs and the feedwater isolation valves. This also opens the valves in the DHRS system, which completes the recirculation loop between the steam generators in the RPV and the DHRS heat exchangers in the reactor pool. However, to provide a bounding LOCA analysis, DHRS cooling is not credited.

As primary coolant is discharged into the containment from the break (Figure 15.6-41), the inventory level inside the RPV continues to decrease and the inventory level and pressure inside the containment continues to increase (Figure 15.6-45) until the high containment level limit generates the MPS ECCS actuation signal. However, as the differential pressure between the RPV and containment exceeds the IAB threshold pressure, the IAB feature prevents the ECCS valves from opening. Pressure and temperature inside the RPV continues a gradual downward trend as primary inventory continues to flow into the containment through the break, as shown in Figure 15.6-42, Figure 15.6-46 and Figure 15.6-47.

The RVVs and RRVs open once the differential pressure between the RPV and containment decreases below the IAB pressure release setpoint, as shown in Figure 15.6-48 and Figure 15.6-49. With all ECCS valves open, the RPV pressure (Figure 15.6-42) and RCS temperature (Figure 15.6-47) rapidly drop, causing voiding in the core and a temporary reduction in the collapsed liquid level above the top of the core (Figure 15.6-45). Core CHFR remains above the safety limit, ensuring that fuel damage due to local dry out conditions does not occur. Inventory released to the containment is allowed to flow back into the RPV downcomer through the RRVs, increasing the collapsed level in the RPV. RCS flowrate is shown in Figure 15.6-50. Steam generator pressure is shown in Figure 15.6-51.

Pressure and temperature inside the containment also experience a rapid increase, as shown in Figure 15.6-42 and Figure 15.6-52. Containment pressure and temperature reaches a maximum value after the ECCS valves open and then decreases as thermal energy is transferred to the reactor pool through the containment wall. The peak containment pressure for design basis events is evaluated in Section 6.2.

At this point, the LOCA event transitions to the post-LOCA long term core cooling phase. A gradual cool down and depressurization of the containment and RPV is occurring, as evident in the pressure, temperature, and level response depicted in Figure 15.6-42, Figure 15.6-45, Figure 15.6-47, and Figure 15.6-52 through Figure 15.6-54. Stable ECCS cooling is established and the module remains in a safe condition with liquid level maintained above the core throughout the entire transient duration. Collapsed liquid level has recovered to an equilibrium level of approximately 10 feet above the top of the core by the end of the transient.

During the post-LOCA long-term core cooling phase, the containment and RPV temperatures and pressures continue to decrease, indicating that the decay and residual heat are being removed from the RPV and containment. Sensitivities show

that the scenario consisting of a 100-percent injection line break with a reactor pool temperature of 65 degrees F, low pressurizer level and a 1.2 multiplier on decay heat results in the RCS minimum collapsed level. Boron precipitation does not occur at the time of the minimum collapsed liquid level, based on the core temperature being less than the highest boron precipitation temperatures for the highest boron concentration. Boron precipitation is also evaluated for the minimum RCS temperature during the 72-hour time following the LOCA, indicating that boron precipitation does not occur.

The MPS is credited to protect the NPM in the event of a LOCA. The following MPS signals provide the plant with protection during a LOCA:

- high pressurizer pressure
- high containment pressure
- low pressurizer level
- low pressurizer pressure
- low low pressurizer level
- high containment water level

15.6.5.4 Radiological Consequences

Section 15.0.3 presents the iodine spike design basis source term (DBST) methodology and the radiological consequences of the iodine spike DBST. The LOCA does not result in fuel failure, therefore the iodine spike DBST bounds the source term, and thus the dose consequences, of a LOCA.

15.6.5.5 Conclusions

The acceptance criteria for a LOCA, per 10 CFR 50.46(b), are listed in Table 15.0-4. These acceptance criteria, followed, by how the NuScale Power Plant design meets them, are listed below.

1) Peak cladding temperature - The calculated maximum fuel element cladding temperature shall not exceed 2200 degrees F.

2) Maximum cladding oxidation - The calculated maximum total oxidation of cladding shall not exceed 17 percent of the total cladding thickness before oxidation.

3) Maximum hydrogen generation - The calculated total amount of hydrogen generated from the chemical reaction of the cladding with water or steam shall not exceed 0.01 times the hypothetical amount that would be generated if all of the metal in the cladding surrounding the fuel, excluding the cladding surrounding the fuel rod plenum volume, were to react.

4) Coolable geometry - Calculated changes in core geometry shall be such that the core remains amenable to cooling.

5) Long-term cooling - After any calculated successful initial operation of the ECCS, the calculated core temperature shall be maintained at an acceptably low value and decay heat shall be removed for the extended period of time required by the long-lived radioactivity remaining in the core.

As discussed in Section 15.0.2, acceptance criteria 1 through 4 are met and no fuel failure occurs by demonstrating that the collapsed liquid level remains above the top of the active fuel, MCHFR remains greater than the safety limit, and containment pressure and temperature remains within design limits. Section 15.6.5.3.3 and Table 15.6-14 demonstrate that the collapsed level remains above the top of the active fuel (Figure 15.6-45), MCHFR remains greater than the safety limit, and containment pressure and temperature remain within design limits. Therefore, acceptance criteria 1 through 4 are met.

The long-term cooling acceptance criterion 5 is also met. As discussed in Section 15.6.5.3.3, the core temperature is maintained at an acceptably low value and decay heat is removed for 72 hours after the initiation of the event. The resultant core temperature and inventory is sufficient to preclude boron precipitation.

15.6.6 Inadvertent Operation of Emergency Core Cooling System

15.6.6.1 Identification of Causes and Accident Description

An inadvertent operation of emergency core cooling system (ECCS) is defined as an accidental reactor vessel depressurization and decrease of reactor vessel coolant inventory that could be caused by a spurious electrical signal, hardware malfunction, or operator error. The NuScale design of ECCS is described in Section 6.3. The ECCS consists of three RVVs exiting the top of the RPV and two RRVs creating an opening to the RPV in the downcomer region above the core. Each ECCS valve includes an inadvertent actuation block (IAB) feature to reduce the frequency of inadvertent opening of the valve during power operation. Section 3.9.1 provides a description of possible failures of the ECCS valves, and concludes that the inadvertent opening of more than one ECCS valve is considered a beyond design basis event due to the ECCS valve IAB feature. Thus, the inadvertent operation of ECCS consists of the inadvertent opening of one RVV or one RRV. The failure of an ECCS valve to a partially open position was evaluated and determined not to be a credible initiating event.

An ECCS valve will open when the force from the pressure in the valve control chamber is less than the opening force of the main valve spring plus the pressure force on the underside of the disc. Depressurization of the control chamber occurs when coolant is lost from the control chamber at a rate greater than is made up through the control chamber orifice that connects to the RCS. The principle method to depressurize the control chamber is by opening the associated ECCS trip valve, which drains the RCS fluid in the chamber to the containment unless it is blocked by the IAB function. The control chamber fluid can also be drained as a result of a mechanical failure of the valve assembly.

If an ECCS trip valve opens, the IAB feature will stop the loss of fluid from the control chamber by blocking the trip line flow path if the differential pressure between the RCS and containment is greater than the IAB threshold. The threshold is determined by the

opening force of a spring internal to the IAB device. The flowpath from the control chamber through the trip line is blocked by a rod in the IAB arming valve moving into its seat. The IAB is actuated by the differential pressure between the RCS on one side of the rod and the pressure in the trip line. When a trip valve opens, fluid drains into containment and the pressure in the trip line decreases, which creates a large differential pressure across the rod. When the force from the differential pressure across the rod is greater than the IAB spring force, the rod moves into its seat and blocks the control chamber fluid from exiting through the trip line. The pressure in the control chamber is maintained by fluid entering through the orifice from the RCS, which prevents the ECCS valve from opening.

The IAB function is a sub-component feature of an ECCS valve as discussed in Section 15.0.0.5. A failure of one of the IAB features on an ECCS valve could result in the opening of a single ECCS valve if an ECCS actuation signal is present or DC power (EDSS) is not available (causes trip valve to fail open). Since the IAB is treated as a component not subject to single failure (Reference 15.6-4), failure of this device is an initiating event. Depressurization of the valve control chamber by a mechanical failure of the valve assembly is a similar initiating event. The mechanical failure results in an ECCS valve opening independent of the status of an ECCS signal or DC power availability. Single active failures, discussed in Section 15.6.6.3, were considered in each of these events but did not result in more limiting results for the acceptance criteria. The limiting event analyzed is a mechanical failure of the valve that depressurizes the control chamber at operating pressure.

The inadvertent opening of a single ECCS valve is not expected to occur during the lifetime of a module. However the event is conservatively categorized as an AOO, as indicated in Table 15.0-1.

The inadvertent opening of an RPV valve analysis evaluates the primary system response to the transient to verify that the event meets the acceptance criteria specified in Table 15.0-2.

15.6.6.2 Sequence of Events and Systems Operation

Sensitivity analyses are performed to identify the limiting event for the inadvertent operation of an ECCS valve. The limiting initiating event for this transient is the inadvertent opening of one RRV. However, it is of note that the resulting MCHFR is similar to that of the inadvertent opening of an RVV. For the following reasons, it is concluded that the RRV event is more challenging for MCHFR and is selected as the limiting event for this calculation:

- The RVV event leads to enhanced cooling of the fuel and cladding.
- The RRV event leads to a more extreme mismatch between power and flow than in the RVV event.
- The RRV event drives a more significant reduction in local hot-channel flow at the axial elevation of MCHFR. A reduction in flow would typically increase the fuel and cladding temperature.

The sequence of events is provided in Table 15.6-15. Unless otherwise specified, the analysis of an inadvertent opening of an RRV assumes the plant control systems and engineered safety features perform as designed, with allowances for instrument uncertainty. No operator action is credited to mitigate the effects of the event.

15.6.6.3 Core and System Performance

15.6.6.3.1 Evaluation Model

The thermal hydraulic response to an inadvertent opening of an ECCS valve event exhibits unique transient progression relative to other AOO events analyzed for the NPM. This progression is divided into two phases:

- The first phase is initiated with an inadvertent opening of an RPV valve (RSV, RVV, or RRV) that results in a blowdown of the RCS into the containment vessel. This breach can be characterized as a steam region breach (i.e., opening of an RSV or RVV) or a liquid region breach (i.e., opening of an RRV). For the limiting event of an inadvertent opening of an RRV, this phase ends when the remaining ECCS valves are actuated as designed by the MPS.
- The second phase begins with ECCS actuation through designed MPS operation and ends when the NPM reaches a safe, stable condition and transitions to long-term ECCS cooling.

These two phases align with the two phases of the LOCA transient progression for the NPM. The LOCA evaluation model and Reference 15.6-1 have:

- identified and ranked important phenomena which occur during these transient phases for the NPM,
- assessed NRELAP5 against separate effects tests and integral effects tests related to these phenomena,
- determined NRELAP5 to be applicable for evaluating these phenomena, and
- developed a conservative NRELAP5 input model for transient analyses which involve an un-isolatable decrease in the RCS inventory event (See Section 15.6.5).

Due to the phenomenological similarities to the LOCA pipe break events described in Section 15.6.5, the LOCA evaluation model, with modifications, is conservatively used in this analysis to evaluate the inadvertent opening of an RPV valve event, consistent with Appendix B of Reference 15.6-1.

15.6.6.3.2 Input Parameters and Initial Conditions

The input parameters and initial conditions used in the evaluation of an inadvertent opening of an RRV are selected to provide a conservative calculation and to minimize the MCHFR. Unless otherwise specified, the analysis assumes that the plant control systems and engineered safety features perform as designed, with allowances for instrument uncertainty. No operator action is credited to mitigate the effects of an inadvertent opening of an RRV.

Table 15.6-16 provides inputs and assumptions. The following are key input parameters:

- Initial power level is assumed 102 percent of nominal power. A high biased power is conservative with respect to MCHFR.
- The RCS average temperature is biased high to 555°F, per Table 15.0-6, to maximize initial RCS energy.
- RCS flow is biased low to minimize MCHFR conditions in the core.
- Pressurizer pressure was analyzed for two conditions: biased high and biased low. The limiting case was found with pressure biased high.
- The initial pressurizer level is biased high to 68 percent. This pressurizer level resulted in a slightly more limiting MCHFR than an initial pressurizer level biased low.
- Beginning-of-cycle core parameters for Doppler temperature coefficient and moderator temperature coefficient are used for calculating scram worth.
- The following conservative scram characteristics are assumed.
  - The maximum time delay from the MPS signal to control rod movement (scram) is applied.
  - The most reactive control rod is assumed to be stuck in the fully withdrawn position.
  - The bounding control rod drop rate, shown in Figure 15.0-2, is applied.
- Beginning-of-cycle kinetic parameters with an additional 6 percent biasing are used in order to prolong the fission power transient, consistent with Reference 15.6-1.
- Minimal reactivity feedback coefficents are conservatively applied in order to minimize negative feedback, consistent with Reference 15.6-1.
- A bounding middle peaked axial power shape is applied to maximize the highest axial peaking factor. Sensitivity studies confirm that this shape is limiting.
- An energy deposition factor of 1.0 is implemented such that all the core power is conservatively deposited in the fuel, consistent with Reference 15.6-1.
- The following loss of power scenarios are considered.
  - No loss of power - In this scenario, all MPS and ESFs actuate as designed. The ECCS valve opening is dependent on both the MPS ECCS actuation setpoints on high CNV water level and the IAB release pressure setpoint.
  - Loss of normal AC - When normal AC power is lost, the feedwater pumps coast down and a turbine trip is initiated, thus limiting RCS cooling via the secondary system. Reactor trip, containment isolation, secondary system isolation, and DHRS actuation occur after a 60-second delay following a loss of normal AC power. ECCS actuation occurs after a 24-hour delay following a loss of normal AC power. However, because DC power is still available, the MPS can still actuate a reactor trip, secondary system isolation, containment isolation, and DHRS, earlier if a separate actuation limit is

reached. However, DHRS is not credited in this analysis. The event sequence for a loss of normal AC power is similar to that when no power is assumed lost. The primary difference is an earlier termination of secondary cooling. This scenario is non-limiting for the reasons described above.

  – Loss of the normal DC power system (EDNS) and normal AC - Power to the reactor trip breakers is provided via the EDNS, so the primary difference to a loss of normal AC power is that the reactor trip will occur sooner. This scenario is non-limiting for the reasons described above.

  – Loss of the highly reliable DC power system (EDSS), EDNS, and normal AC - This scenario results in an immediate actuation of the reactor trip system, secondary system isolation, DHRS (although not credited in the analysis), and containment isolation. As power to the MPS is lost, the ECCS valve opening is dependent only on the IAB pressure release threshold. This scenario is identified as limiting although it is similar to the power available scenarios due to the rapid nature of the transient.

- The single failure evaluation considered one RVV failing to open, one RRV failing to open, or failure of one ECCS division causing one RVV and one RRV to fail to open. The evaluation compared the results to a scenario with no single failure. The limiting MCHFR occurs within the first one second of the RRV opening. No failures occur in a timeframe that affect this result. The evaluation showed that the single failure cases have no adverse impact on the limiting MCHFR or other acceptance criteria evaluated in this analysis. Therefore, the scenario with no single failure is limiting for this analysis.

- The failure modes that could lead to a partial opening of an ECCS valve were characterized as having a remote probability of occurrence or were determined to not be credible. None of the credible component failure mechanisms that could prevent a full stroke of the ECCS valve have the potential to cause an ECCS valve to open. Therefore, a partial opening of an ECCS valve is not a credible initiating event.

- Allowances for instrument inaccuracy are accounted for in the analytical limits of mitigating systems in accordance with the guidance provided in Regulatory Guide 1.105.

- No operator action is credited.

15.6.6.3.3 Results

Figure 15.6-55 to Figure 15.6-68 show the system response to an inadvertent RRV opening event. Table 15.6-17 contains the results of the event. The limiting case is initiated by an inadvertent opening of an RRV. Sensitivity analysis show that the limiting scenario has a loss of normal AC and DC power and no single failure occurs.

Upon the inadvertent RRV opening, the large blowdown of the RCS into the containment causes rapid depressurization of the RCS and rapid pressurization of the containment. Inadvertent RRV flow is shown in Figure 15.6-56. The assumed loss of AC and DC power at time zero result in an immediate reactor scram, secondary system isolation, and DHRS actuation. However, DHRS operation is conservatively not credited in this analysis. The RCS and containment pressures are

shown in Figure 15.6-57. The high containment pressure analytical limit is reached shortly after event initiation.

The rapid RCS depressurization causes voiding in the core and a momentary decrease in RCS flow (Figure 15.6-58 and Figure 15.6-59), leading to a reduction in CHFR (Figure 15.6-67 and Figure 15.6-68). Reactor power decreases during this time due to control rod insertion and negative void feedback, as seen in Figure 15.6-55 and Figure 15.6-60. Following the occurrence of transient MCHFR (Figure 15.6-67), a temporary increase in RCS flow is observed due to the increased density gradient from voiding in the riser (Figure 15.6-58).

The isolation of the secondary system following the loss of normal AC and DC power at transient initiation causes an increase in steam generator pressure, as seen in Figure 15.6-61. Heat transfer from the RCS to the secondary coolant isolated in the steam generator region is limited due to the decreasing RCS temperatures associated with decreasing pressure and saturation temperature. Steam generator pressure is not limiting for an inadvertent opening of an RPV valve event.

As primary coolant is released to the containment through the open RRV, the inventory level inside the containment increases (Figure 15.6-62). The minimum collapsed liquid level remains approximately 10 feet above the top of the active fuel throughout the event. As the RPV continues to depressurize, the differential pressure between the RPV and containment drops below the IAB threshold pressure, allowing the ECCS valves to open. Pressure and temperature inside the RPV continue a gradual downward trend, as shown in Figure 15.6-57, Figure 15.6-63, and Figure 15.6-64.

After the remaining ECCS valves open and pressure equalizes across the RRVs, liquid coolant from the containment begins to flow into the RPV downcomer region. This establishes a two phase natural circulation loop through the ECCS valves with steam exiting the pressurizer area into containment through the RVVs and liquid returning from the containment to the RPV through the RRVs. Decay heat and residual heat is transferred from the containment to the reactor pool resulting in the pressure and the temperature inside the RPV and containment continuing to decrease.

The transient continues until stable ECCS cooling has been established and RCS pressure and temperature continues to decrease. The module remains in a safe condition with liquid level maintained above the top of the core through the entire transient. The fuel volume average temperature is shown in Figure 15.6-65 and fuel cladding temperature is shown in Figure 15.6-66.

The MPS is credited to protect the module in the event of an inadvertent opening of an RRV by the following MPS signals:

- high containment pressure, and
- high containment water level

No operator actions are credited for this event.

The event transitions to long-term cooling, similar to that described in Section 15.6.5.

15.6.6.4 Radiological Consequences

Section 15.0.3 provides the radiological consequences for the NuScale infrequent events and postulated accidents. Radiological consequence analyses are not required for AOOs. Section 15.0.3 also presents the iodine spike DBST methodology and the radiological consequences of the iodine spike DBST. The inadvertent opening of an RPV valve does not result in fuel failure, therefore the iodine spike DBST bounds the source term, and thus the dose consequences, of this event.

15.6.6.5 Conclusions

The acceptance criteria for an AOO are listed in Table 15.0-2. These acceptance criteria, followed, by how the NuScale Power Plant design meets them, are listed below. Table 15.6-17 provides the results of the limiting scenario of an inadvertent opening of an RRV.

1) Fuel cladding integrity shall be maintained by ensuring that minimum DNBR remains above the 95/95 DNBR limit. Minimum critical heat flux ratio (MCHFR) is used instead of minimum DNBR, as described in Section 4.4.2.

   The fuel integrity is not challenged by the inadvertent opening of the RRV. The fuel temperatures decrease upon the reactor trip, as shown in Figure 15.6-65 and Figure 15.6-66, and the water level remains above the top of the active fuel, as shown in Figure 15.6-62. The MCHFR is above the acceptance criterion as shown in Table 15.6-17 and as shown in Figure 15.6-67 and Figure 15.6-68. The MCHFR occurs in the high flow correlation range shortly after event initiation as reactor power is still elevated at this time.

2) RCS pressure should be maintained below 110 percent of the design value. The design pressure for the reactor vessel is 2100 psia, thus the acceptance criterion is 2310 psia.

   The RCS pressure is below the acceptance criterion, as shown in Table 15.6-17.

3) The main steam pressure should be maintained below 110 percent of the design value. The design pressure for the reactor vessel is 2100 psia, thus the acceptance criterion is 2310 psia.

   The main steam pressure, presented as steam generator pressure, is below the acceptance criterion, as shown in Table 15.6-17.

4) The event should not generate a more serious plant condition without other faults occurring independently.

   The analysis presented for this event shows that the NPM continues to be cooled with natural circulation through the ECCS valves and the event terminates in a safe, stabilized condition.

The response of the NPM during the long-term cooling phase following the inadvertent opening of an RPV valve is similar to the response of the NPM following a LOCA. The long-term cooling analysis, results and conclusions are discussed in Section 15.6.5.

15.6.7 References

15.6-1 NuScale Power, LLC, "Loss-of-Coolant Accident Evaluation Model," TR-0516-49422, Rev. 0.

15.6-2 NuScale Power, LLC, "NuScale Power Critical Heat Flux Correlations," TR-0116-21012-P-A, Rev. 1.

15.6-3 NuScale Power, LLC, "Subchannel Analysis Methodology," TR-0915-17564-P-A, Rev. 2.

15.6-4 Nuclear Regulatory Commission, “Staff Requirements - SECY-19-0036 - Application of the Single Failure Criterion to NuScale Power LLC’s Inadvertent Actuation Block Valves,” dated July 2, 2019.

Table 15.6-1: Failure of Lines Carrying Primary Coolant Outside Containment - Sequence of Events - Maximum Mass Release

Event	Time (s)(1)
Letdown Line Break	0
Loss of AC	0
Turbine Trip	0
Pressurizer Heater Isolation	0
Loss of Feedwater	0
High Steam Line Superheat Analytical Limit	1
RTS Actuation Signal	7
Secondary Isolation Signal	7
RTS Actuation	9
Makeup Line Break (assumed to maximize release)	9
High Steam Line Pressure Analytical Limit	10
DHRS Actuation Signal	10
DHRS Actuation	12
Low-Low PZR Pressure Analytical Limit	94
CVCS Isolation Signal	94
CVCS Isolation Actuation	96
End of Spiking Time	103
Total Integrated break flow maximum	223

Notes: (1) Time rounded to the nearest second

Table 15.6-2: Failure of Lines Carrying Primary Coolant Outside Containment - Sequence of Events - Maximum Iodine Spiking Time

Event	Time (s)[(1)]
Makeup Line Break	0
Loss of AC	0
Turbine Trip	0
Pressurizer Heater Isolation	0
Loss of Feedwater	0
High Steam Line Superheat Analytical Limit	1
RTS Actuation Signal	7
Secondary Isolation Signal	7
RTS Actuation	9
High Steam Line Pressure Signal	10
DHRS Actuation Signal	10
DHRS Actuation	12
Low-Low Pressurizer Pressure Analytical Limit	170
CVCS Isolation Signal	170
CVCS Isolation Actuation	172
End of Spiking Time	180

Notes: (1) Time rounded to the nearest second

Table 15.6-3: Failure of Lines Carrying Primary Coolant Outside Containment - Sequence of Events - Maximum Reactor Pressure Vessel Pressure

Event	Time (s)(1)
Makeup Line Break	0
Loss of AC	0
Turbine Trip	0
Pressurizer Heater Isolation	0
Loss of Feedwater	0
High Pressurizer Pressure Analytical Limit	13
Secondary Isolation Signal	13
DHRS Actuation Signal	13
RTS and DHRS Actuation	15
Peak RCS Pressure	19
Low-Low PZR Pressure Analytical Limit	155
CVCS Isolation Signal	155
CVCS Isolation Actuation	157
End of Spiking Time	165

Notes: (1) Time rounded to the nearest second

Table 15.6-4: Failure of Lines Carrying Primary Coolant Outside Containment - Inputs

Parameter	Nominal	Maximum Mass Release Bias	Maximum Iodine Spiking Time Bias	Maximum RCS Pressure Bias
Break Location	N/A	Letdown line	Makeup line	Makeup line
Core power	100% (160 MWt)	+2% (163.2 MWt)	+2% (163.2 MWt)	+2% (163.2 MWt)
RCS average temperature	545 °F	+10 °F	+10 °F	-10 °F
Pressurizer pressure	1850 psia	+70 psia	+70 psia	+70 psia
Pressurizer level	60%	+8%	+8%	+8%
SG pressure	500 psia	-35 psia	-35 psia	-35 psia
Feedwater temperature	300 °F	-10 °F	-10 °F	-10 °F
RCS flow	See Note (1)(2)	1180 lbm/s(1)	1180 lbm/s(1)	No bias applied.
Fuel kinetics	N/A	BOC	BOC	BOC
Loss of normal AC power	N/A	Start of Event	Start of Event	Start of Event

Notes:

(1) RCS flow is a function of power. See Table 15.0-6 for description of RCS flow range.

(2) RCS flow converted from kg/s. Flow is biased low.

Table 15.6-5: Failure of Lines Carrying Primary Coolant Outside Containment - Results (1)

Parameter	Iodine Spiking Time (sec)	Mass Released (lbm)	Peak Reactor Pressure (psia)	Peak Steam Generator Pressure (psia)
Maximum Mass Release	94	782 (Pre-Trip) 12,170 (Post-Trip)	1962	1397
Maximum Iodine Spiking Time	170	599 (Pre-Trip) 10,930 (Post-Trip)	2017	1369
Maximum RCS Pressure	N/A	N/A	2067	1224
Maximum SG Pressure(2)	N/A	N/A	N/A	1473

Notes:

(1) Values are rounded

(2) The peak SG pressure case is not presented separately because the SG has the same design pressure as the RPV. An increase in SG pressure is driven by heat transfer from the RPV, which requires a higher primary side temperature and saturation pressure. Therefore, the SG peak pressure cannot exceed the RPV peak pressure.

Table 15.6-6: Steam Generator Tube Failure - Sequence of Events - Limiting Mass Release

Event	Time (s)[1]
SGTF (55% tube area split break) at bottom of SG	0
Maximum RCS pressure	1
Low PZR level trip signal	1246
PZR Heater Isolation	1248
Reactor Trip	1249
Low-low PZR level	1282
Secondary System Isolation Signal	1283
MSIV Closure Signal	1285
TSV Closure	1285
High SG-2 Steam Line Pressure	1293
DHRS Actuation	1295
Maximum SG pressure reached	1365

Notes: (1) Time rounded up to second.

Table 15.6-7: Steam Generator Tube Failure - Sequence of Events - Limiting Reactor Pressure Vessel Pressure

Event	Time (s)[1]
SGTF (16% tube area split break) at bottom of SG	0
Loss of AC Power	0
TSV Closure	0
High PZR Pressure	6
DHRS Actuation	8
PZR Heater Isolation	8
MSIV Closure Signal	8
Reactor Trip	8
Maximum RCS pressure	12
Maximum SG pressure reached	1385

Notes: (1) Time rounded up to second.

Table 15.6-8: Steam Generator Tube Failure - Sequence of Events - Limiting Steam Generator Pressure

Event	Time (s)(1)
SGTF (DEG break) at bottom of SG	0
Loss of AC Power	0
TSV Closure	0
High PZR Pressure	9
High SG Steam Line Pressure	9
Reactor Trip	11
DHRS Actuation	11
PZR Heater Isolation	11
MSIV Closure Signal	11
Maximum RCS pressure	16
Maximum SG pressure reached	106

Notes: (1) Time rounded up to second.

Table 15.6-9: Steam Generator Tube Failure - Inputs

Description	Nominal	Limiting Mass Release Scenario Bias	Limiting RCS Pressure Scenario Bias	Limiting Steam Generator Pressure Scenario Bias	Limiting Iodine Spiking Time Scenario Bias
Core Power	160 MWt	163.2 (102%)	163.2 (102%)	163.2 (102%)	163.2 (102%)
Pressurizer Pressure	1850 psia	+70 psia	+70 psia	+70 psia	+70
Pressurizer Level	60%	+8%	+8%	+8%	+8%
Feedwater Temperature	300 °F	-10 °F	-10 °F	-10 °F	+10 °F
Fuel Kinetics	N/A	BOC	BOC	BOC	BOC
RCS Average Temperature	545 °F	+10 °F	+10 °F	+10 °F	-10 °F
Steam Generator Pressure	500 psia	+35 psia	+35 psia	+35 psia	-35 psia
Steam Generator Tube Plugging	N/A	0%	0%	10%	0%
Loss of Normal AC Power	N/A	None	Start of Event	Start of Event	None
Failure of MSIV on faulted SG	N/A	Yes	No	No	Yes
Steam Generator Tube Failure Location	N/A	Bottom of SG	Bottom of SG	Bottom of SG	Bottom of SG

Table 15.6-10: Steam Generator Tube Failure - Results (1)

Parameter	Acceptance Criteria	Value	Limiting Case
Peak Reactor Pressure	≤2520 psia	2158 psia	Limiting Reactor Pressure
Peak Steam Generator Pressure	≤2520 psia	1871 psia	Limiting Steam Generator Pressure

Notes: (1) Values increased to the whole number.

Table 15.6-11: Steam Generator Tube Failure Inputs to Radiological Consequences

Description	Units	Maximum Mass Release	Maximum Iodine Spiking Time
Time of Reactor Trip	s	1249	1543
Time of Isolation for the broken SG-1	s	1292	1592
Time of Isolation for the intact SG-2	s	1285	1585
Iodine Spiking Time	s	43	50
Initial Mass Released through Failed Tube – Pre-Trip	lbm	10213	9116
Initial Mass Released through Failed Tube – Trip to Isolation	lbm	327	282
Secondary Flow – Failed SG-1 – Pre-Trip	lbm	99963	124340
Secondary Flow – Failed SG-1 – Trip to Isolation	lbm	2781	3259
Secondary Flow – Intact SG-1 – Pre-Trip	lbm	92021	117352
Secondary Flow – Intact SG-1 – Trip to Isolation	lbm	2352	2871

Table 15.6-12: Loss-of-Coolant Accident - Sequence of Events - Minimum Collapsed Level Above TAF

Event	Time (sec)*
Line break	0
Loss of normal AC	0
High pressurizer pressure	8
Reactor Trip System actuation signal	10
Reactor trip	12
High containment pressure	16
Containment isolation signal	18
Containment isolation	20
Low pressurizer level (35%)	968
Low Low pressurizer level (20%)	1729
High containment water level (264 inches)	7372
High CNV water level	7372
Low pressurizer pressure (1600 psia)	9380
ECCS actuates (RCS pressure drops below IAB threshold)	12724

*Time rounded to the nearest second.

Table 15.6-13: Loss-of-Coolant Accident - Inputs

Description	Units	Nominal	Analyzed Value
Core power - beginning of cycle	MWt	160	163.2 (102%)
RCS Average Temperature	F	545	555
Pressurizer pressure	psia	1850	1920
Pressurizer level	%	60	52
Containment Pressure	psia	<1 psia	2
Main steam pressure	psia	500	535
Feedwater temperature	F	300	310
Bypass flow (reflector and guide tubes)	%	n/a	8.5
Reactor pool temperature	F	n/a	140
Reactor pool level	ft	n/a	55

Table 15.6-14: Loss-of-Coolant Accident - Results Minimum Collapsed Level Above TAF(1)

Parameter	Acceptance Criteria	Value
Minimum Collapsed Liquid Level	Above top of core	1.5 ft(3)
Minimum critical heat flux ratio	>1.29	1.72
Containment pressure(2)	<1050 psia	538 psia
Containment temperature(2)	<550 F	454 F

Notes:

(1) Values rounded

(2) Section 6.2 contains the limiting containment analysis. The containment pressure and temperature reported here is from the limiting minimum collapsed liquid level scenario.

(3) Minimum collapsed level conservatively rounded down to 1.5 ft (lowest sensitivity calculated 1.7 ft).

Table 15.6-15: Inadvertent Operation of an Emergency Core Cooling System Valve - Sequence of Events

Event	Time (s)*
RRV opens	0
Loss of normal AC and DC power	0
Control rods begin to fall	0
Minimum CHFR occurs	0.5
Control rods fully inserted into core	2.3
Remaining ECCS valves open	50
Natural circulation from containment to reactor pressure vessel is established	483
Peak steam generator pressure is reached	490
Minimum collapsed liquid level above the core	630

*Time rounded to the nearest tenth of a second.

Table 15.6-16: Inadvertent Operation of an Emergency Core Cooling System Valve - Inputs

Description	Units	Nominal	Analyzed Value
Core power	MWt	160	163.2 (102%)
Pressurizer pressure	psia	1850	1920
Pressurizer level	%	60	68
Reactor Coolant System Flow	lbm/sec	See Table 15.0-6	1179

Table 15.6-17: Inadvertent Operation of an Emergency Core Cooling System Valve - Results*

Parameter	Acceptance Criteria	Value
Peak reactor pressure	≤2310 psia	1936 psia
Peak steam generator pressure	≤2310 psia	588 psia
MCHFR	1.13	1.41
Fuel cladding integrity maintained	Yes	Yes
Generate more serious plant condition?	No	No

*Values increased to the whole number.

Table 15.6-18: Loss-of-Coolant Analysis - Summary of Break Spectrum

Break Area %	Break Area in^2	Equivalent Diameter in	Break Area Analyzed			
			Discharge Line	Injection Line	High Point Vent	Pressurizer Spray Supply
100	2.23	1.69	X	X	X	-
75	1.67	1.46	X	X	X	-
50	1.12	1.19	X	X	X	-
35	0.78	1.00	X	X	X	X
20	0.45	0.75	X	X	X	-
10	0.22	0.53	X	X	X	-
5	0.11	0.38	X	X	X	-
2.2	0.05	0.25	X	X	X	-

Table 15.6-19: Loss-of-Coolant Analysis - Discharge Line Break Spectrum with All Electric Power Available

Break Size (%)	Time of RTS (s)	Time of ECCS (s)	MCHFR	Peak CNV Pressure (psia)	Min Collapsed Level above TAF (ft)
100	5	934	1.77	649	7.7
75	6	1170	1.78	641	7.4
50	7	1588	1.79	636	6.9
35	8	2066	1.79	638	6.6
20	12	2997	1.79	676	5.9
10	34	5652	1.79	656	4.8
5	206	13187	1.80	531	4.3
2.2	674	26337	1.80	511	4.5

Table 15.6-20: Loss-of-Coolant Analysis - Injection Line Break Spectrum with All Electric Power Available

Break Size (%)	Time of RTS (s)	Time of ECCS (s)	MCHFR	Peak CNV Pressure (psia)	Min Collapsed Level above TAF (ft)
100	8	938	1.80	867	5.0
75	8	1012	1.79	869	4.9
50	9	1208	1.79	874	4.7
35	10	1466	1.80	863	4.0
20	15	2570	1.80	804	3.7
10	33	5993	1.80	636	2.6
5	170	12526	1.80	536	2.0
2.2	842	24684	1.80	497	3.4

Table 15.6-21: Loss-of-Coolant Analysis - High Point Vent Line Break Spectrum with All Electric Power Available

Break Size (%)	Time of RTS (s)	Time of ECCS (s)	MCHFR	Peak CNV Pressure (psia)	Min Collapsed Level above TAF (ft)
100	6	8868	1.80	497	7.6
75	7	9440	1.80	458	7.5
50	8	10330	1.80	398	7.2
35	11	11643	1.80	335	7.0
20	19	14825	1.80	240	7.1
10	81	20825	1.80	317	6.3
5	529	N/A	1.80	92	11.6
2.2	2342	N/A	1.80	87	22.9

Table 15.6-22: Loss-of-Coolant Analysis - Pressurizer Spray Supply Line Break Spectrum with All Electric Power Available

Break Size (%)	Time of RTS (s)	Time of ECCS (s)	MCHFR	Peak CNV Pressure (psia)	Min Collapsed Level above TAF (ft)
35	7	9528	1.80	449	7.4

Table 15.6-23: Loss-of-Coolant Analysis - Five-percent Injection Line Break with Evaluation of Electric Power Available

Case	Time of RTS (s)	Time of ECCS (s)	MCHFR	Peak CNV Pressure (psia)	Min Collapsed Level Above TAF (ft)
All power available	170	12526	1.80	536	2.0
Loss of Normal AC Power	10	12690	1.72	538	1.8
Loss of normal AC and DC power	2	12542	1.72	537	2.0

Table 15.6-24: Loss-of-Coolant Analysis - Five-percent Injection Line Break with Loss of Normal AC Power Evaluation of Single Failure

Single Failure	Time of RTS (s)	Time of ECCS (s)	MCHFR	Peak CNV Pressure (psia)	Min Collapsed Level Above TAF (ft)
No failure	10	12691	1.72	538	1.8
Failure of one RVV to open	10	12691	1.72	526	1.7
Failure of one RRV to open	10	12691	1.72	557	1.8
Failure of one RVV and RRV to open	10	12691	1.72	544	1.7

Table 15.6-25: Steam Generator Tube Failure - Sequence of Events - Limiting Iodine Spiking Time

Event	Time (s)$^{(1)}$
SGTF (DEG break) at bottom of SG	0
Low PZR level	1540
PZR Heater Isolation	1542
Reactor Trip	1543
Low-low PZR level	1582
MSIV Closure Signal	1585
TSV Closure	1585
High SG-1 Steam Line Pressure (faulted)	1603
DHRS Actuation	1605

Note: (1) Time rounded up to second.

Figure 15.6-1: Failure of Lines Carrying Primary Coolant Outside Containment - Maximum Mass Release Scenario Break Mass Flow

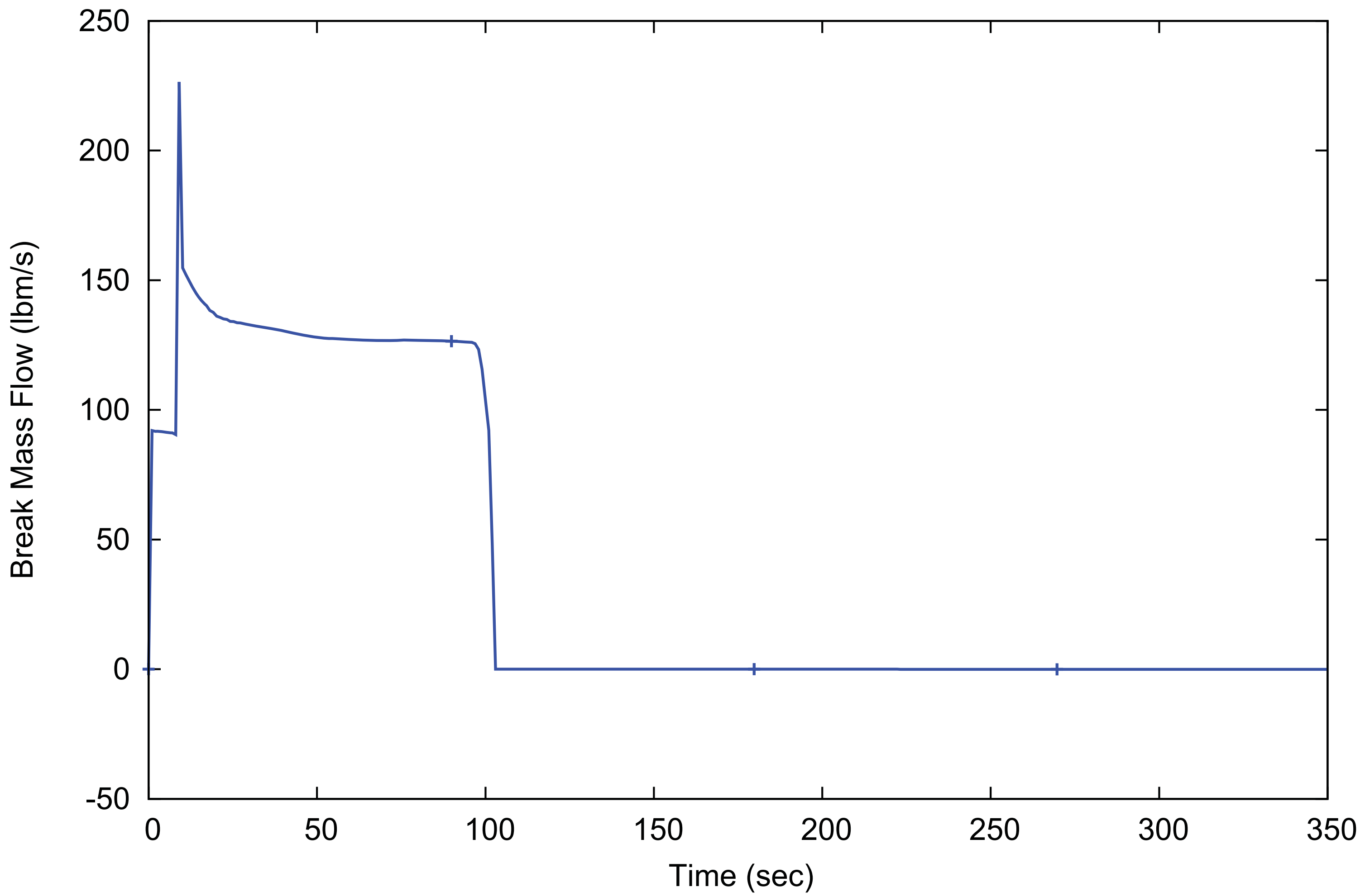


Figure 15.6-2: Failure of Lines Carrying Primary Coolant Outside Containment - Maximum Mass Release Scenario - Integrated Break Mass Flow

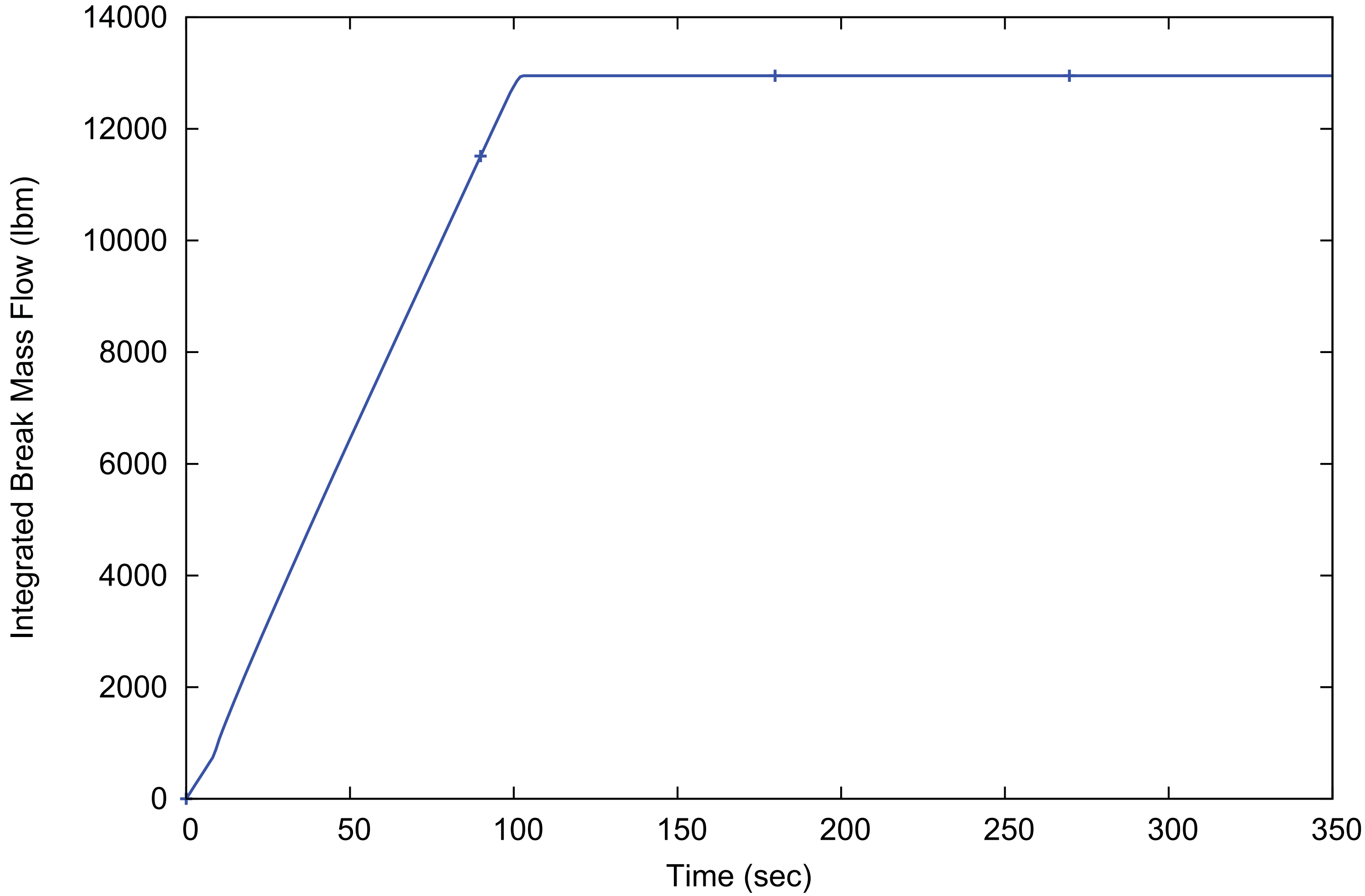


Figure 15.6-3: Failure of Lines Carrying Primary Coolant Outside Containment - Maximum Reactor Pressure Vessel Pressure Scenario - Pressure

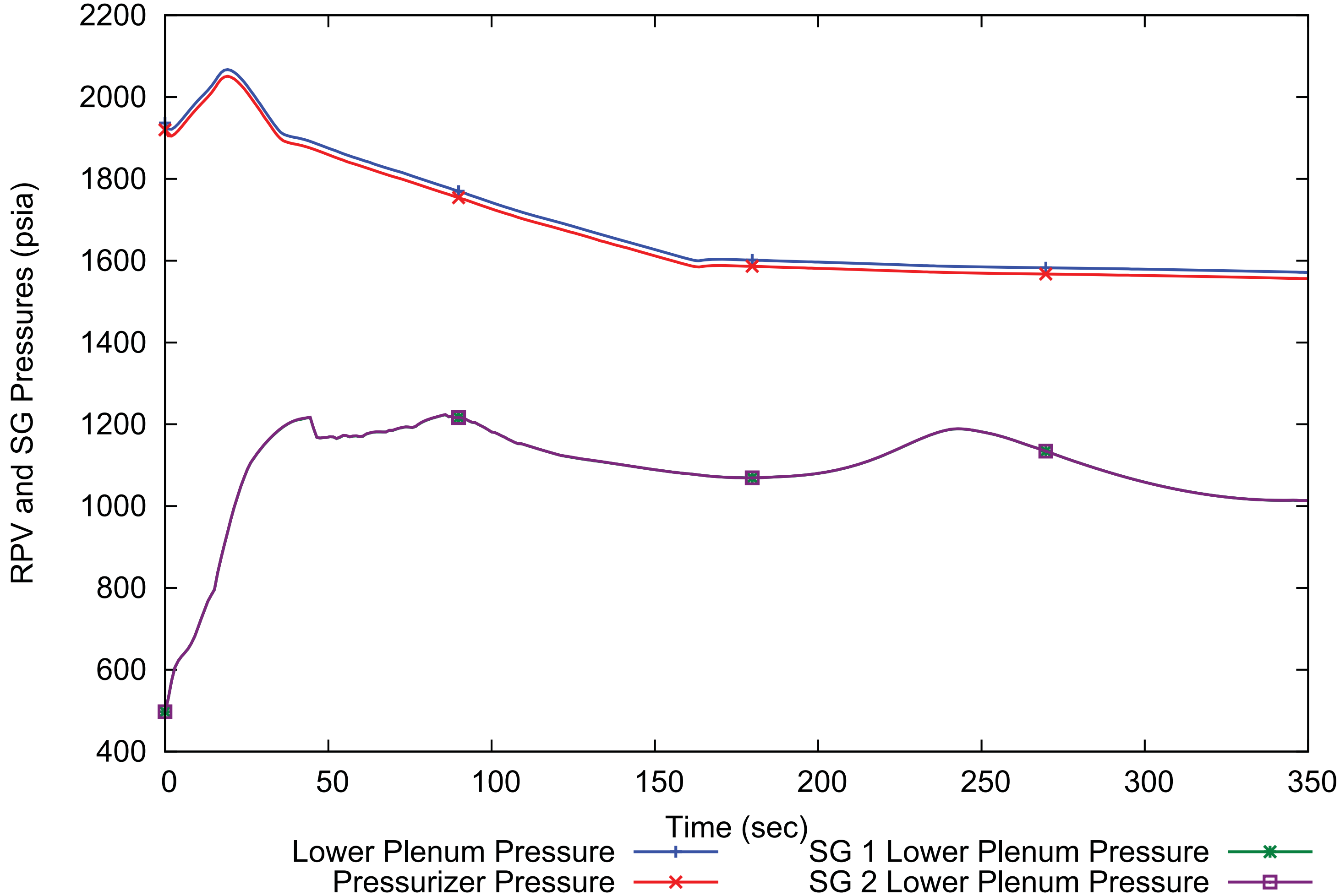


Figure 15.6-4: Failure of Lines Carrying Primary Coolant Outside Containment - Maximum Reactor Pressure Vessel Pressure Scenario - Pressure

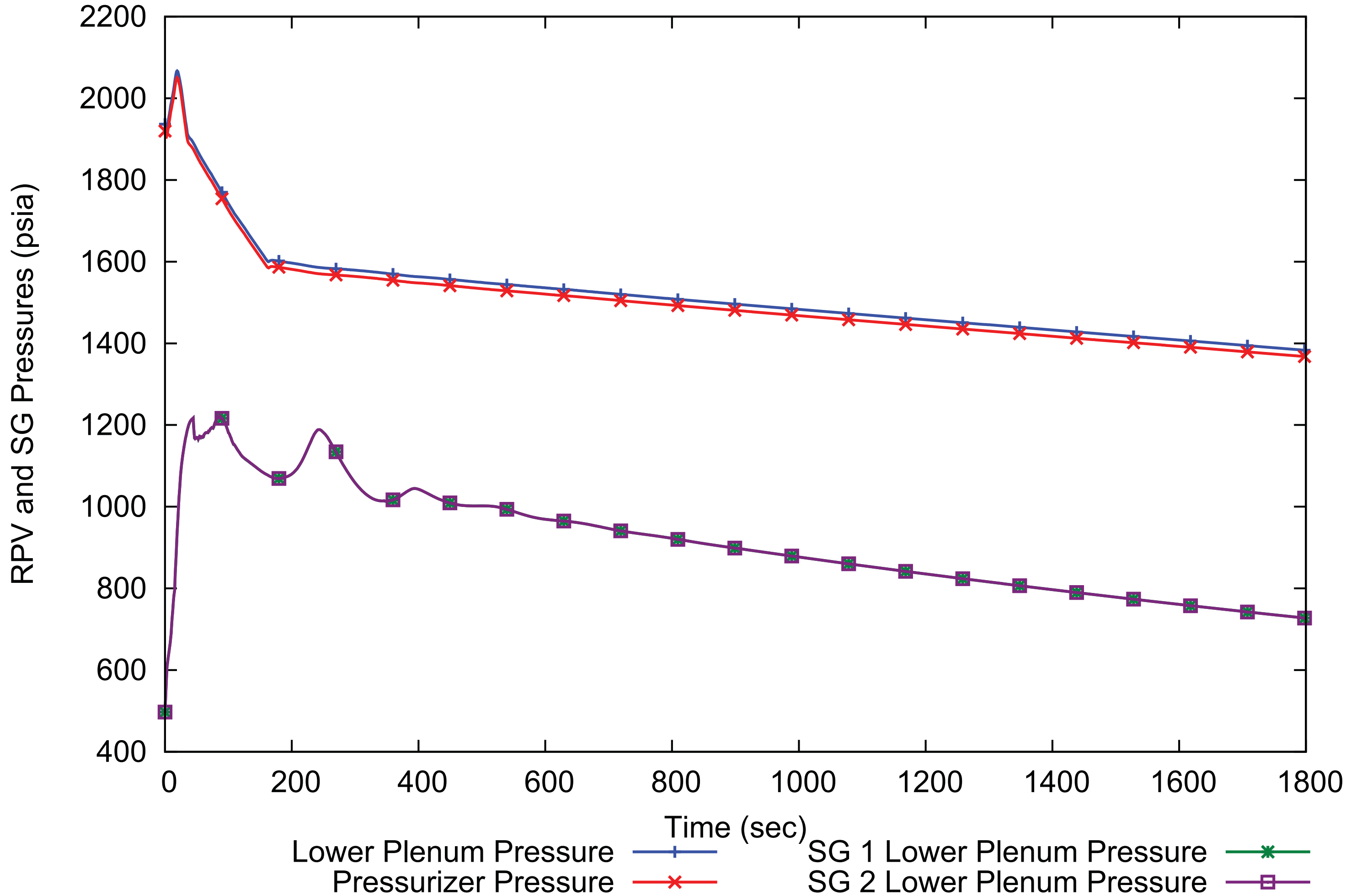


Figure 15.6-5: Failure of Lines Carrying Primary Coolant Outside Containment - Maximum Reactor Pressure Vessel Pressure Scenario - Reactor Power

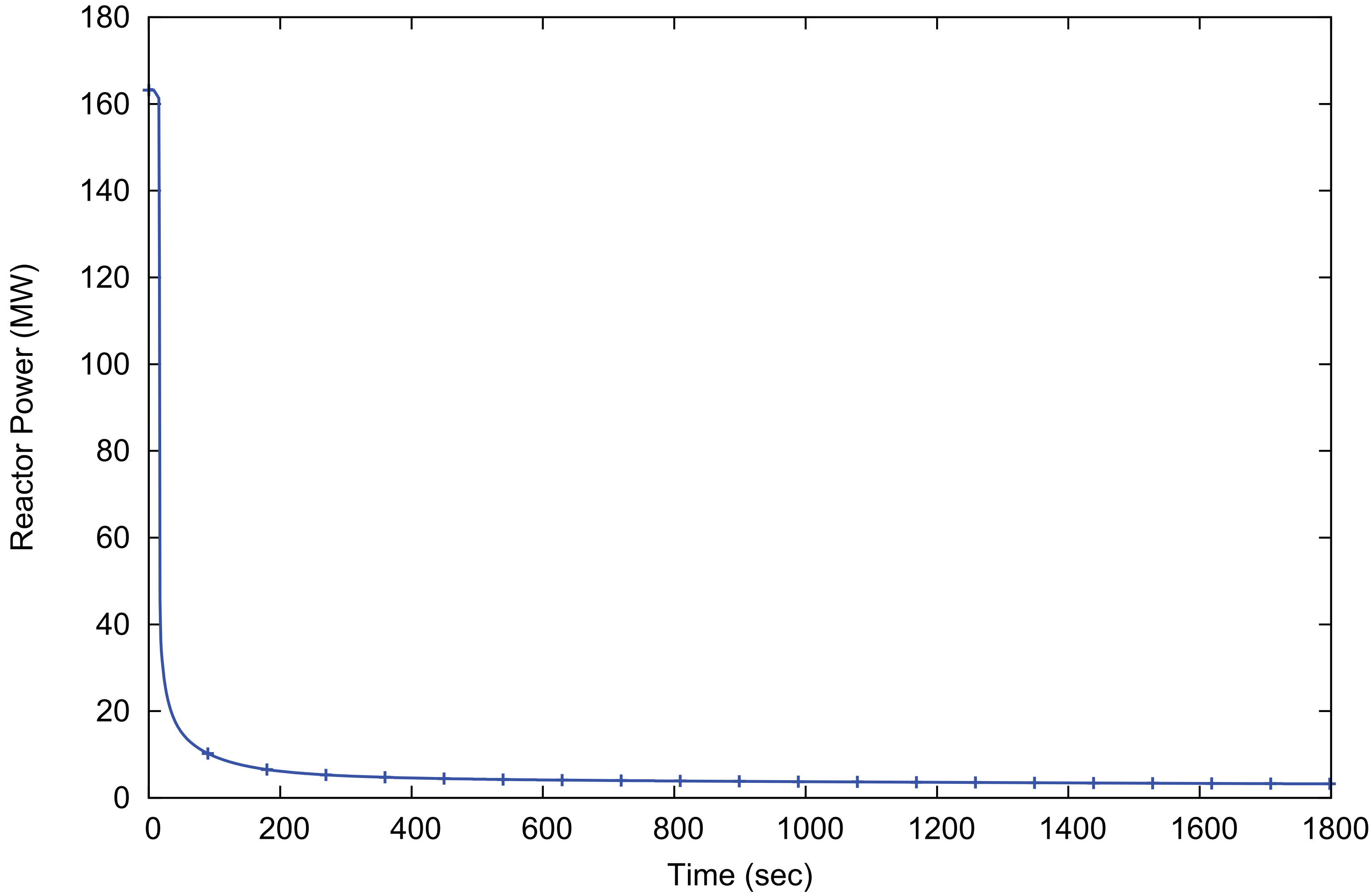


Figure 15.6-6: Failure of Lines Carrying Primary Coolant Outside Containment - Maximum Reactor Pressure Vessel Pressure Scenario - Decay Heat Removal System Flow

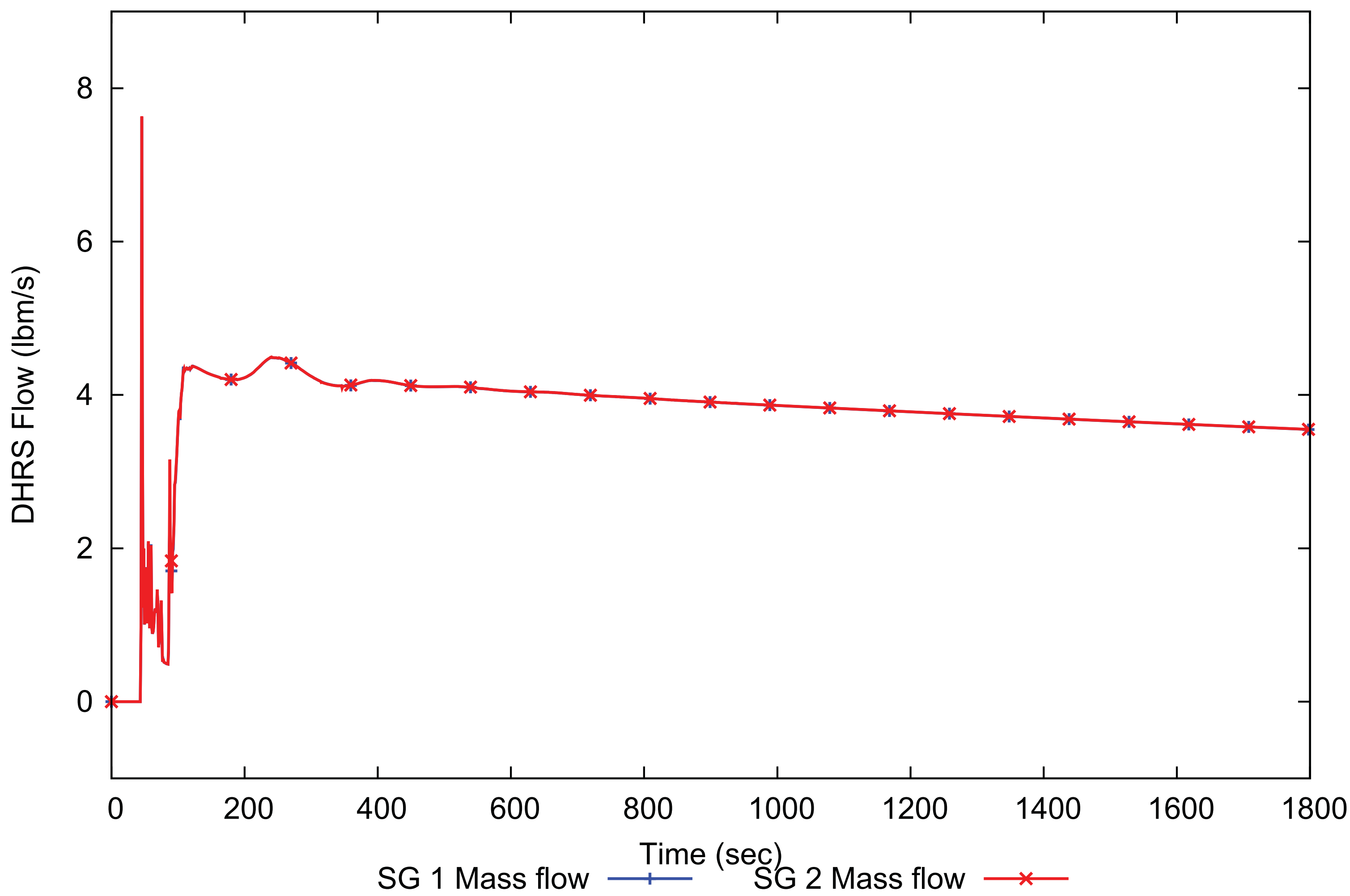


Figure 15.6-7: Failure of Lines Carrying Primary Coolant Outside Containment - Maximum Reactor Pressure Vessel Pressure Scenario - Reactor Coolant System Temperatures

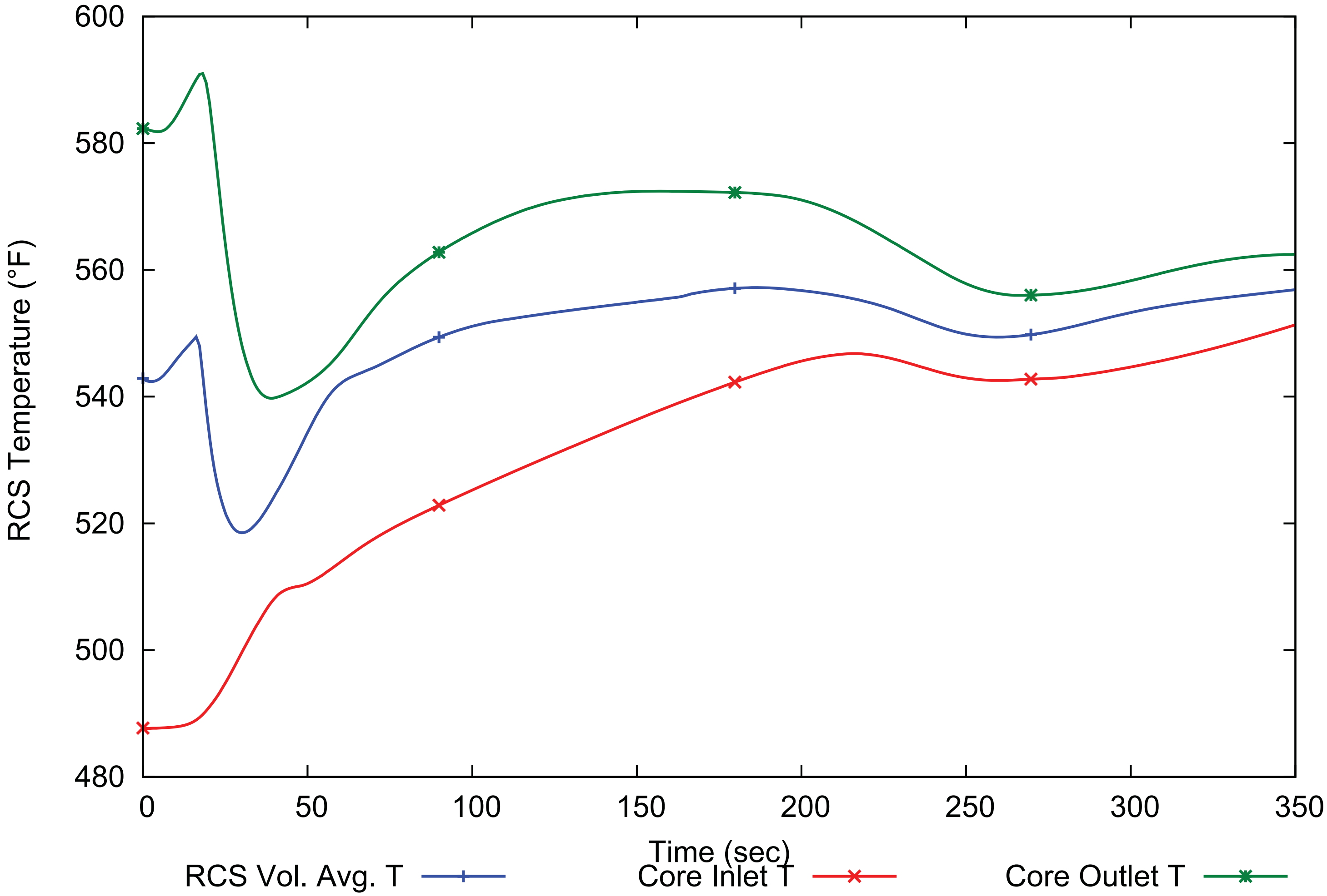


Figure 15.6-8: Failure of Lines Carrying Primary Coolant Outside Containment - Maximum Reactor Pressure Vessel Pressure Scenario- Reactor Coolant System Temperatures

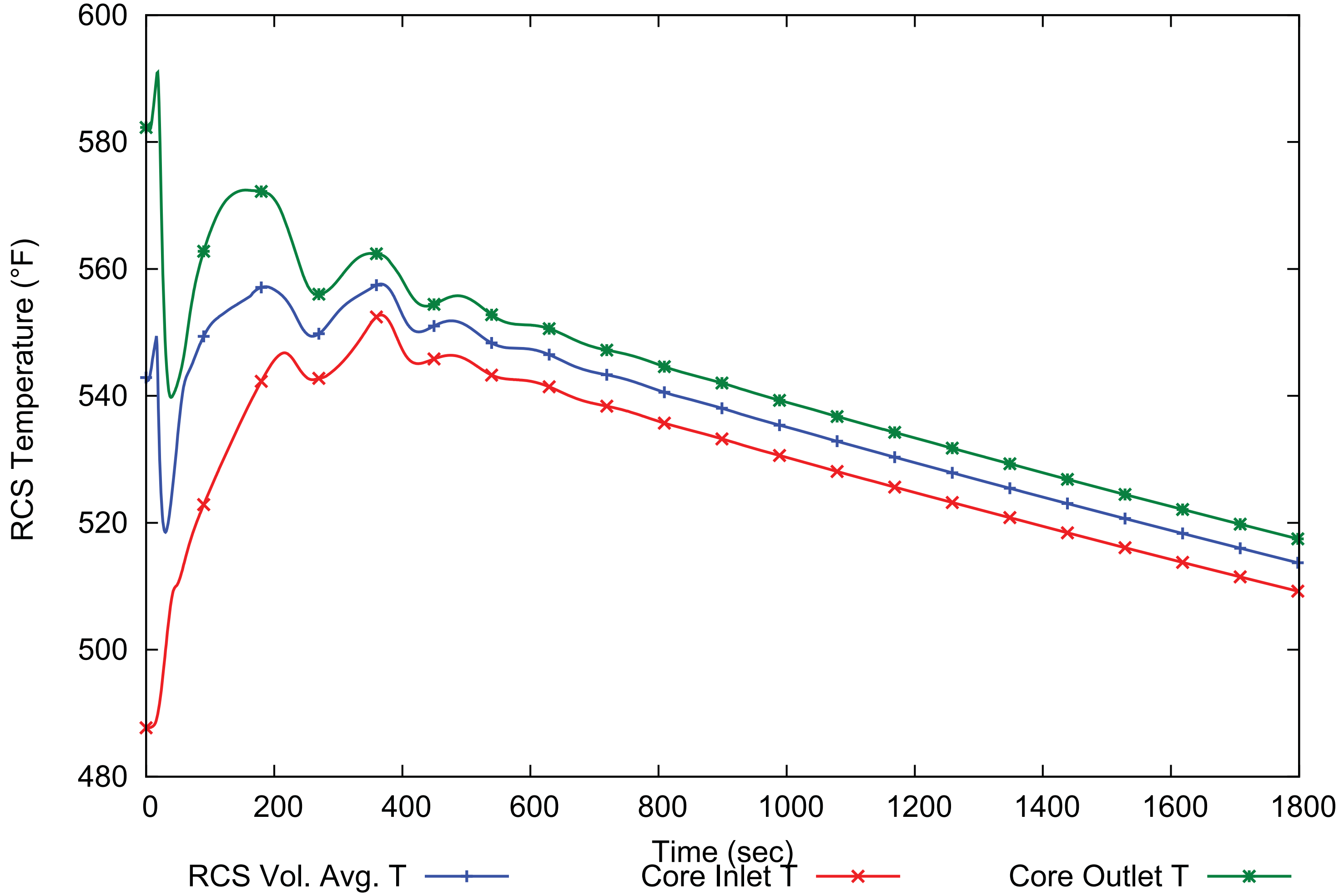


Figure 15.6-9: Failure of Lines Carrying Primary Coolant Outside Containment - Maximum Reactor Pressure Vessel Pressure Scenario - Pressurizer Level

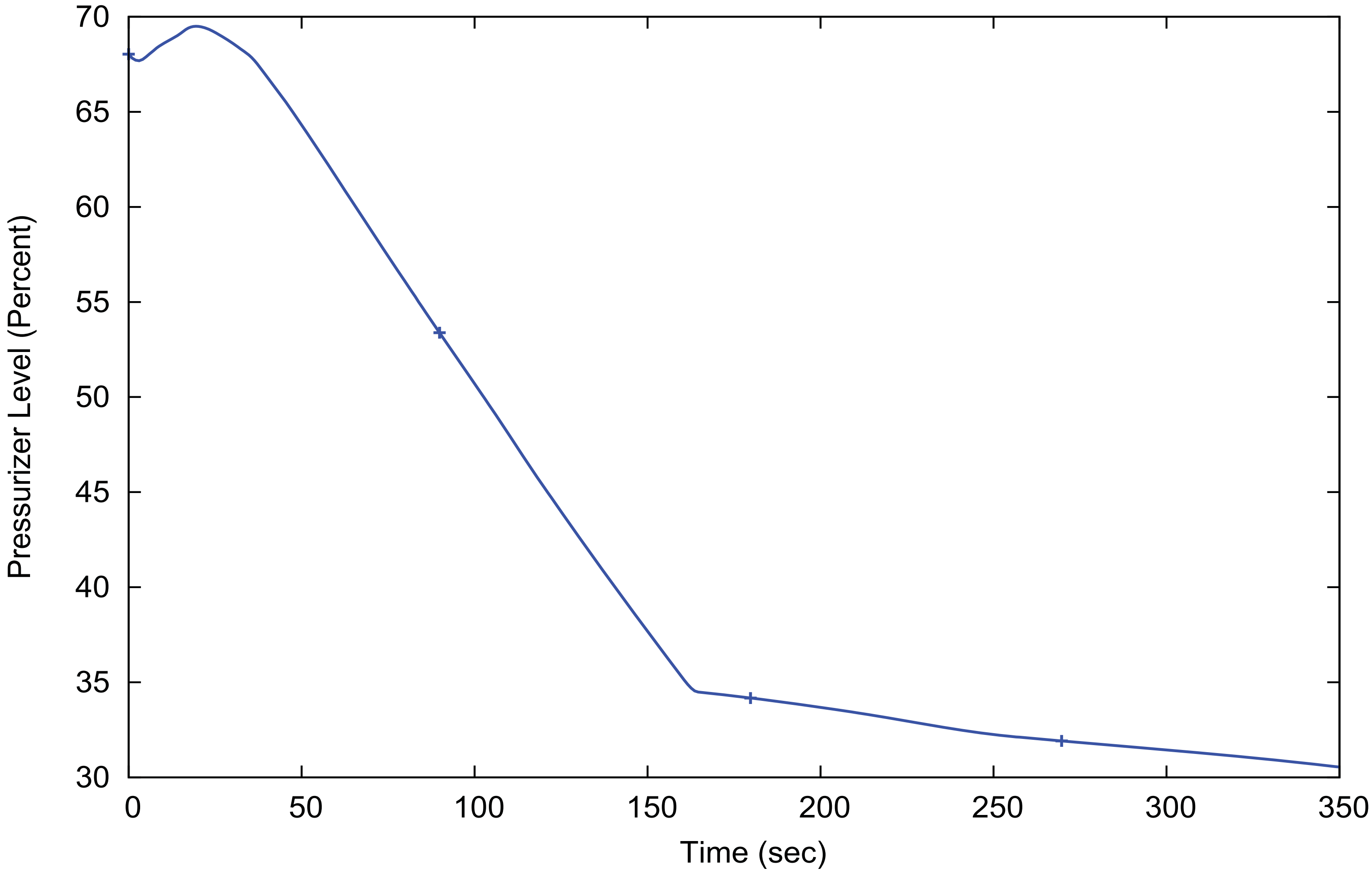


Figure 15.6-10: Failure of Lines Carrying Primary Coolant Outside Containment - Maximum Reactor Pressure Vessel Pressure Scenario - Pressurizer Level

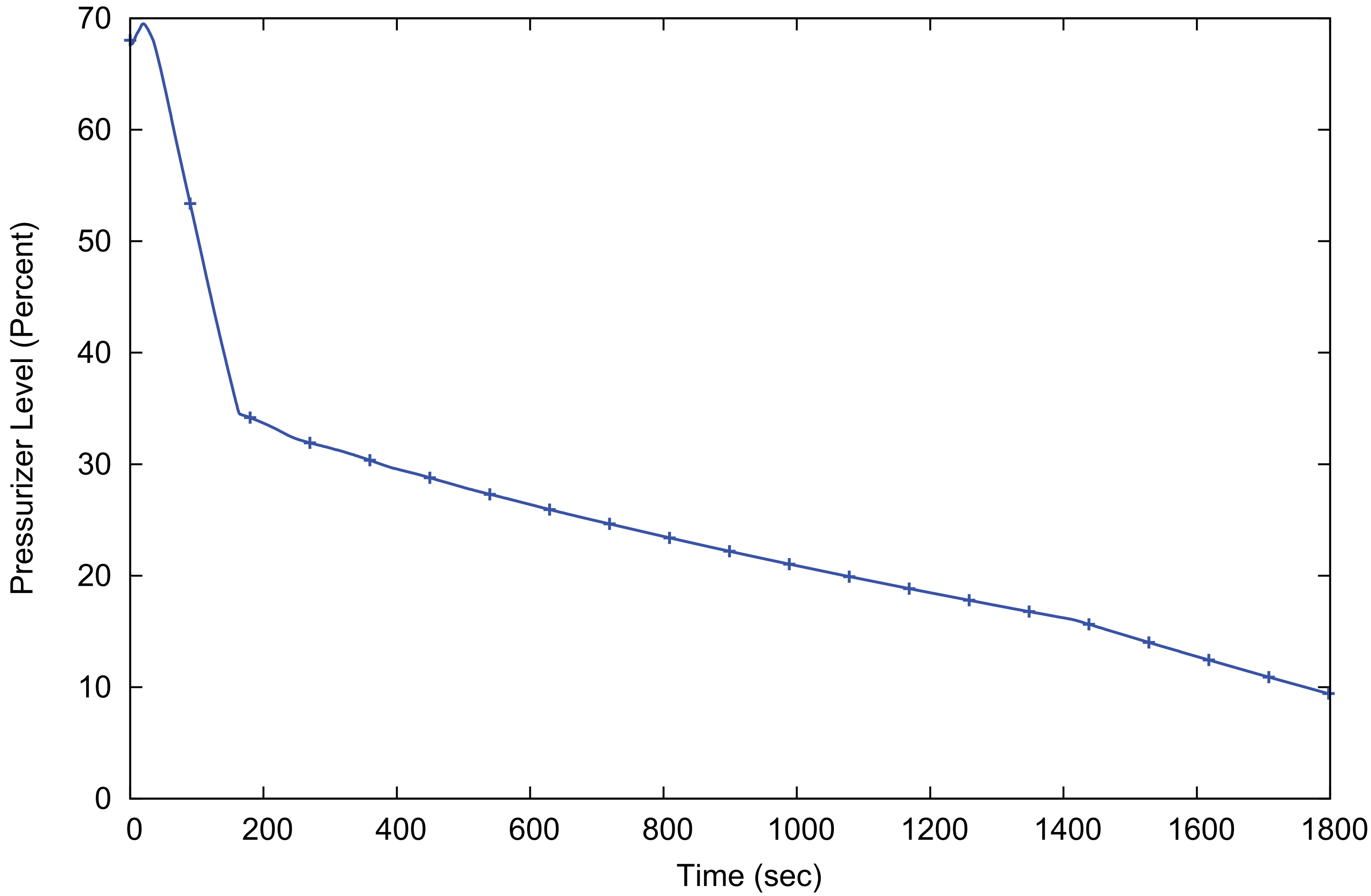


Figure 15.6-11: Failure of Lines Carrying Primary Coolant Outside Containment - Maximum Reactor Pressure Vessel Pressure Scenario - Level above the Top of the Core

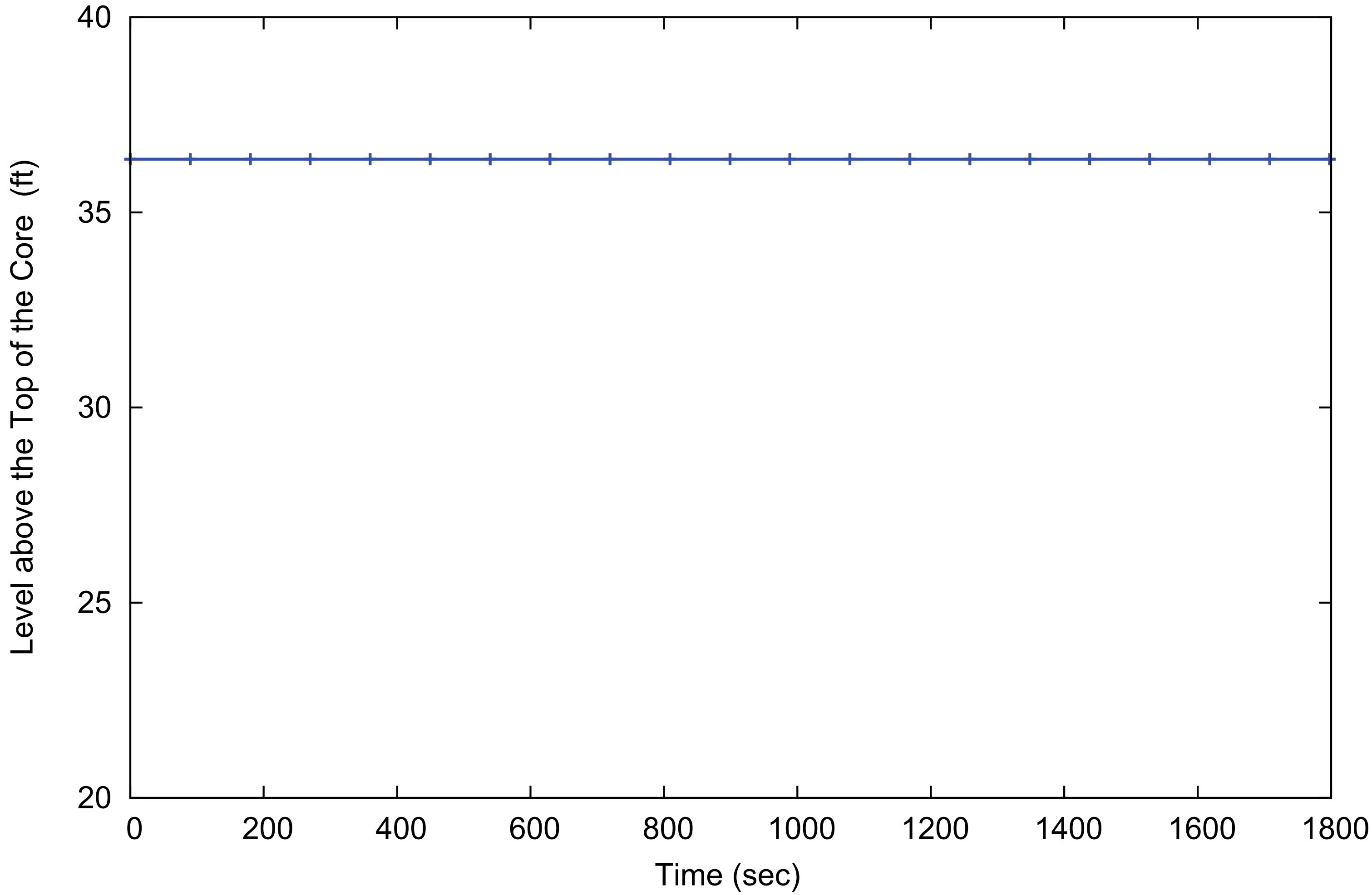


Figure 15.6-12: Failure of Lines Carrying Primary Coolant Outside Containment - Maximum Reactor Pressure Vessel Pressure Scenario - Fuel Volume Average Temperature

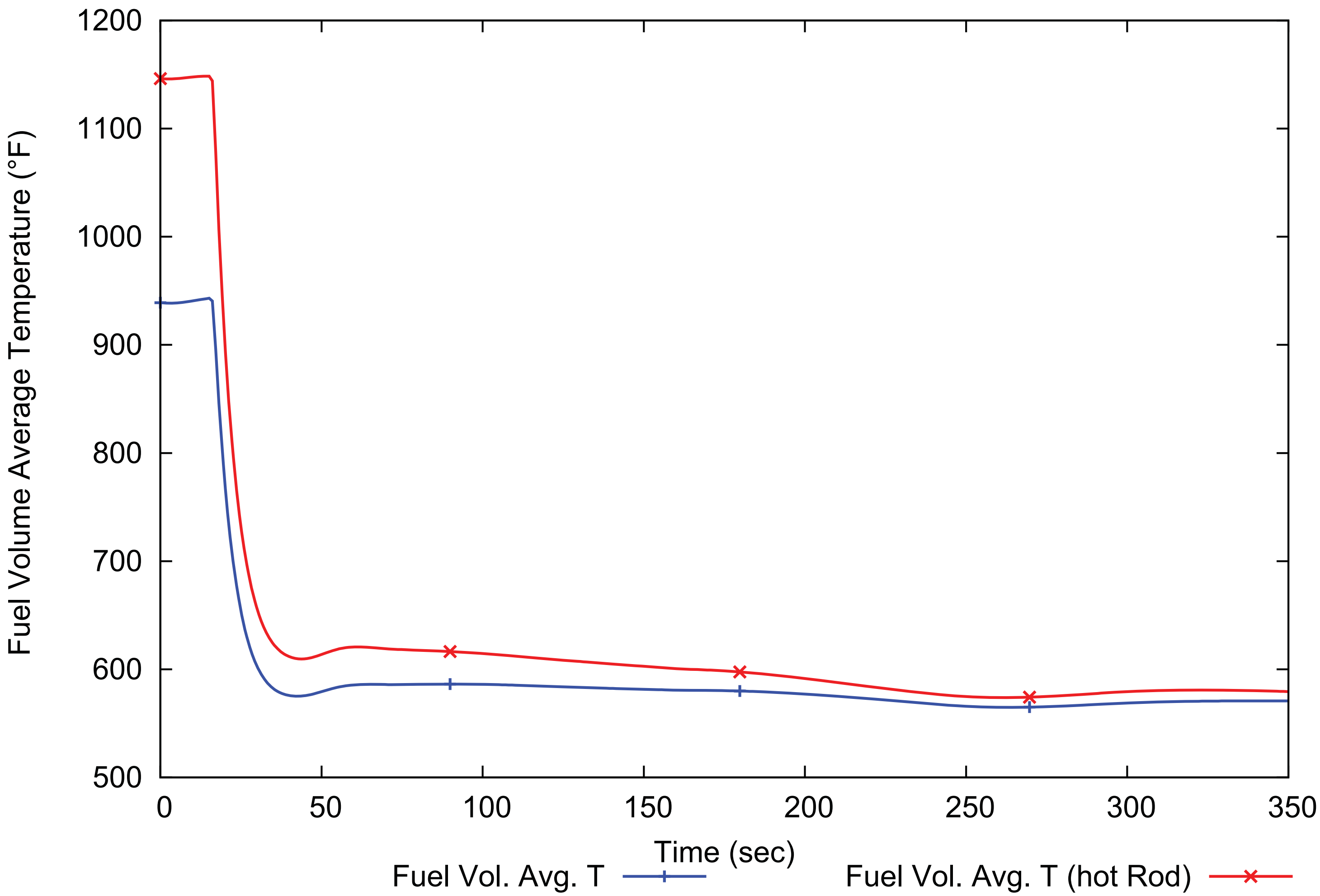


Figure 15.6-13: Failure of Lines Carrying Primary Coolant Outside Containment - Maximum Reactor Pressure Vessel Pressure Scenario - Fuel Volume Average Temperature

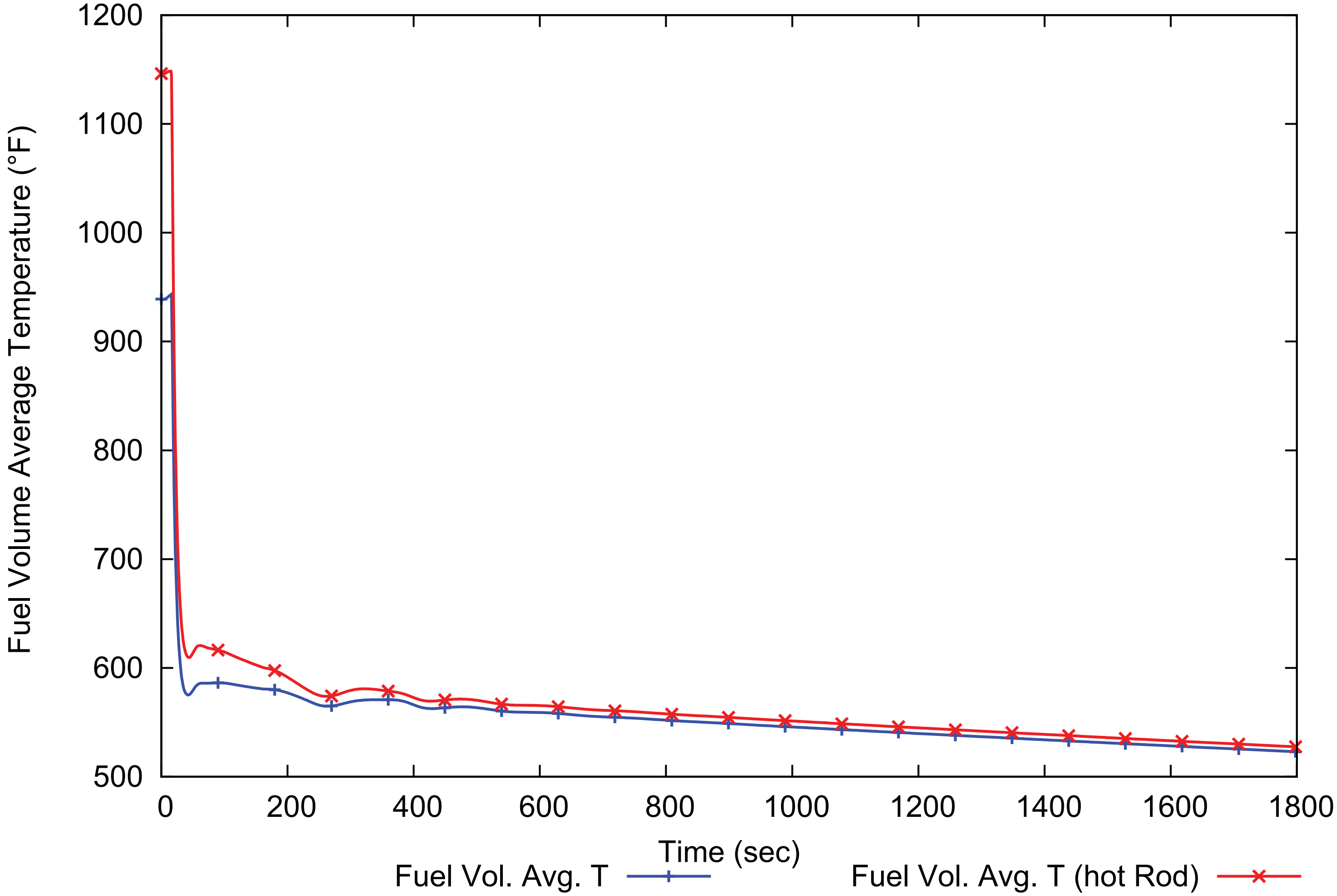


Figure 15.6-14: Failure of Lines Carrying Primary Coolant Outside Containment - Maximum Reactor Pressure Vessel Pressure Scenario - Reactor Coolant System Mass Flow

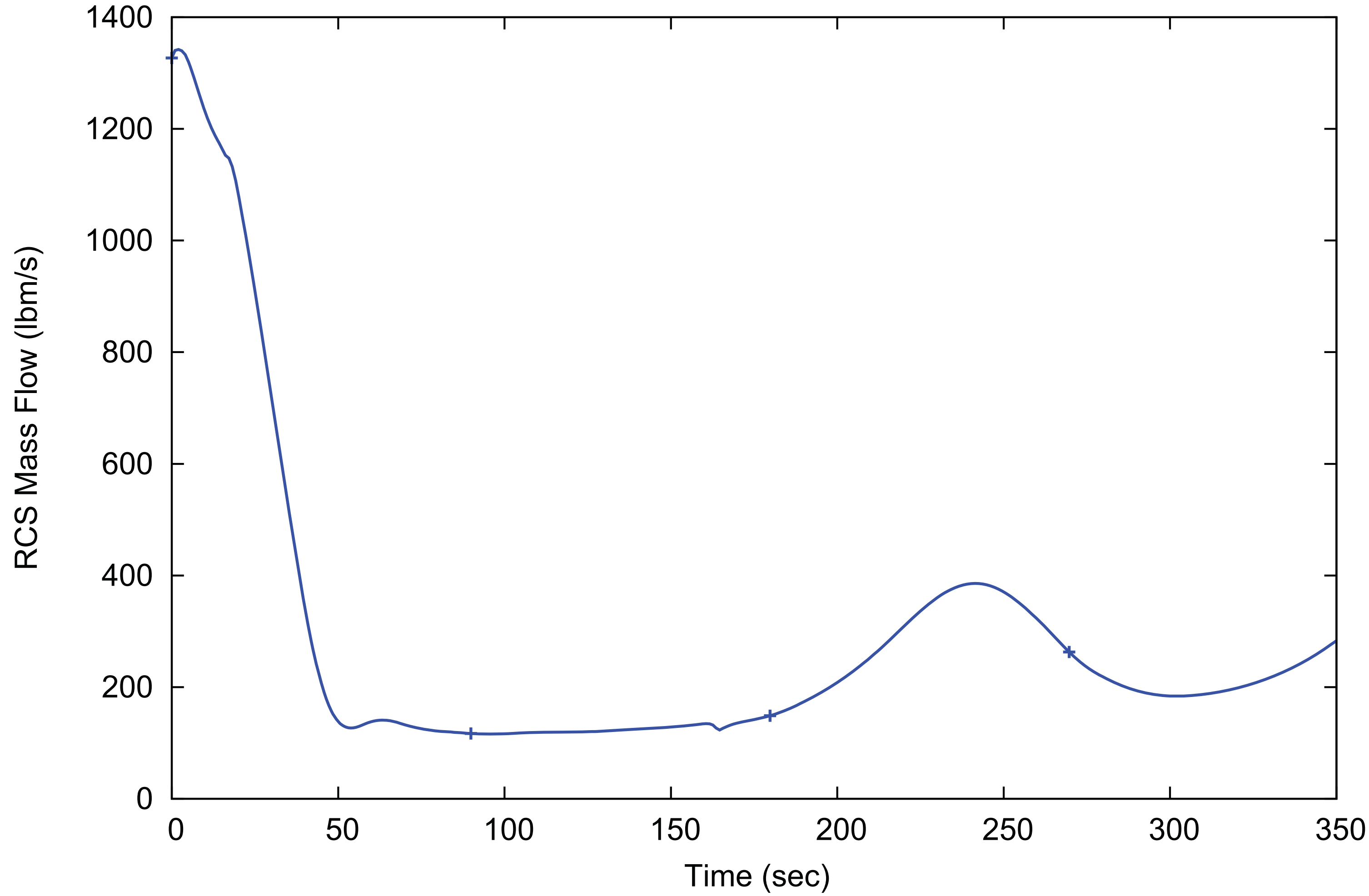


Figure 15.6-15: Failure of Lines Carrying Primary Coolant Outside Containment - Maximum Reactor Pressure Vessel Pressure Scenario - Reactor Coolant System Mass Flow

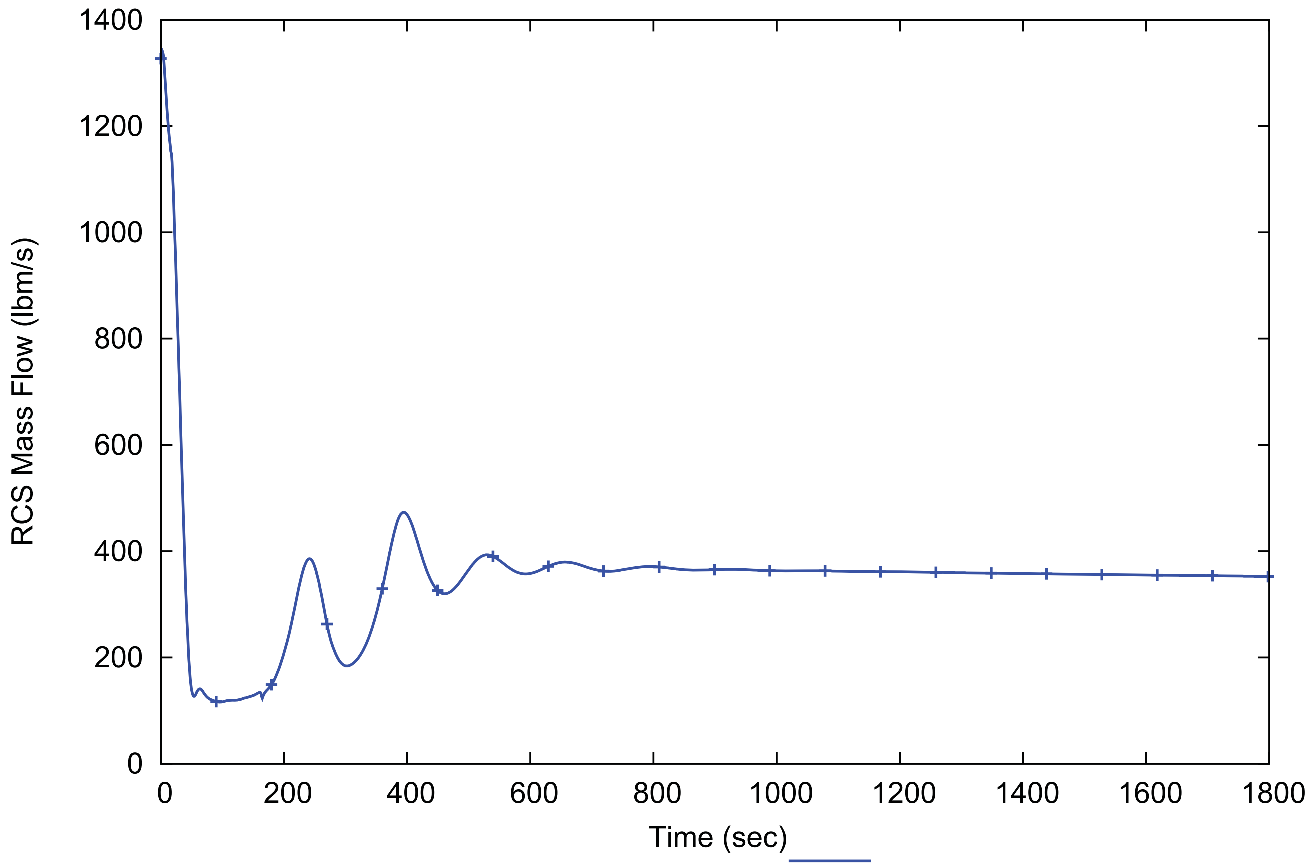


Figure 15.6-16: Failure of Lines Carrying Primary Coolant Outside Containment - Maximum Reactor Pressure Vessel Pressure Scenario - Total Reactivity

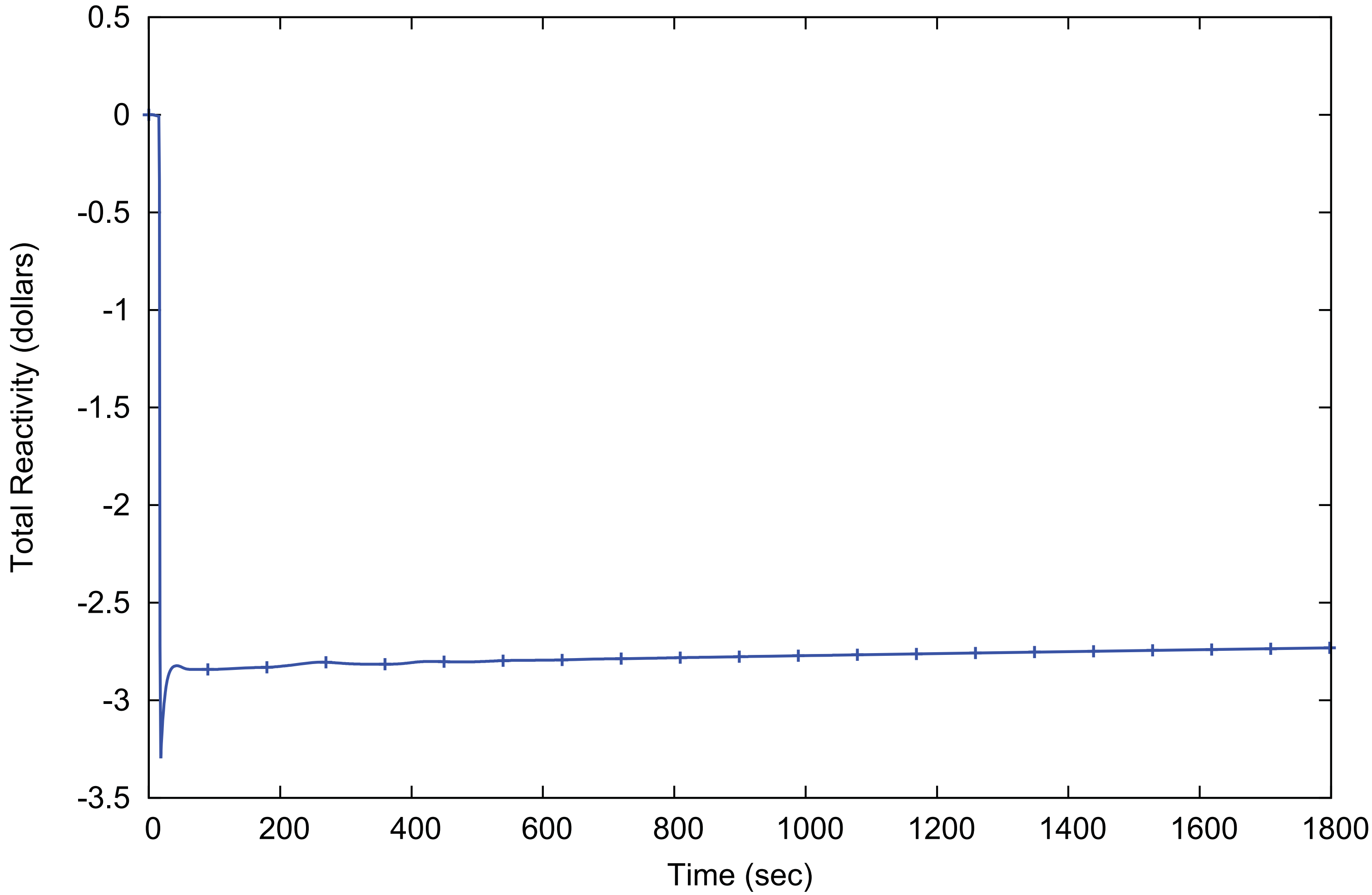


Figure 15.6-17: Steam Generator Tube Failure - Limiting Mass Release Scenario - Break Mass Flow

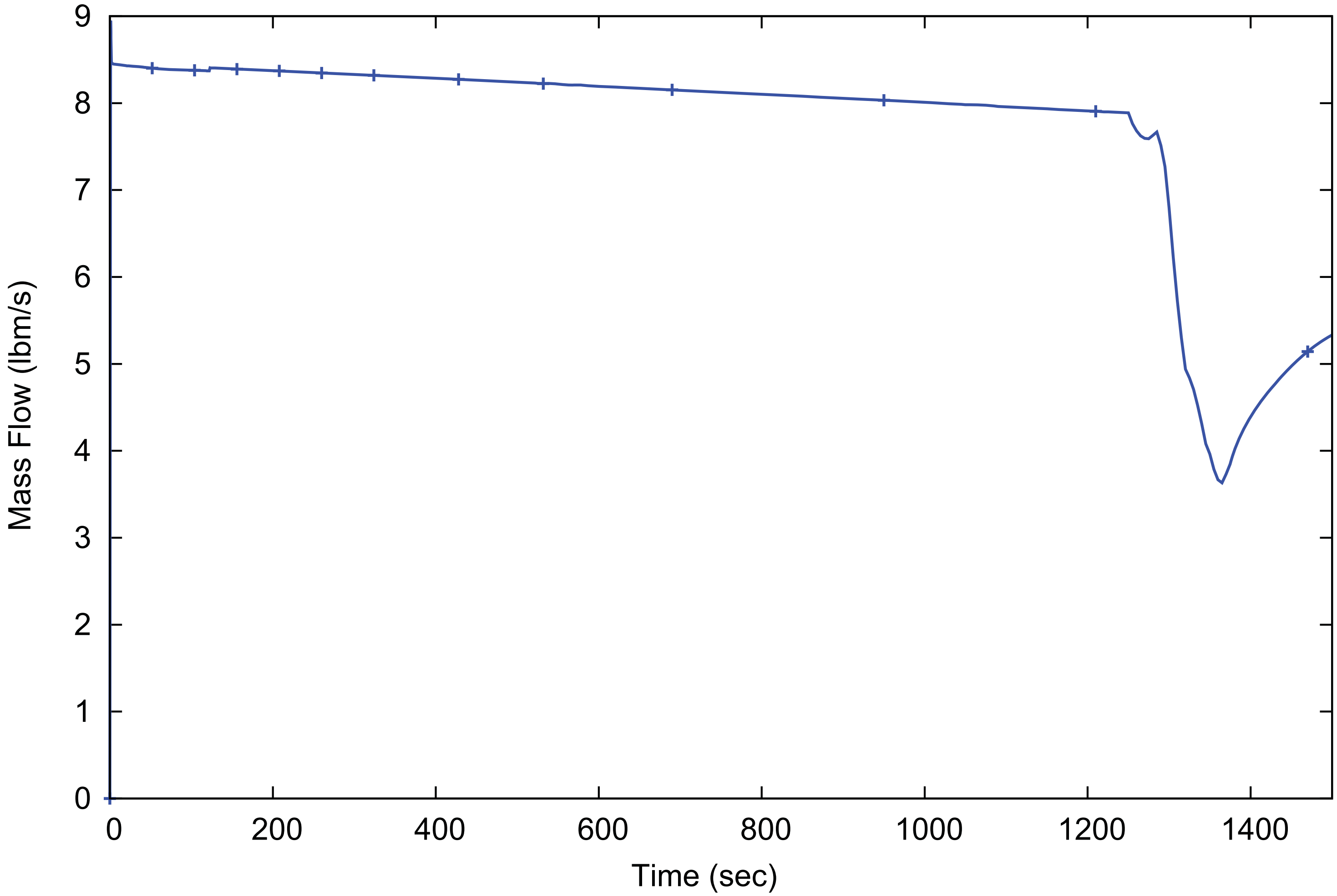


Figure 15.6-18: Steam Generator Tube Failure - Limiting Mass Release Scenario - Mass Release to Steam Generator before Steam Generator Isolation

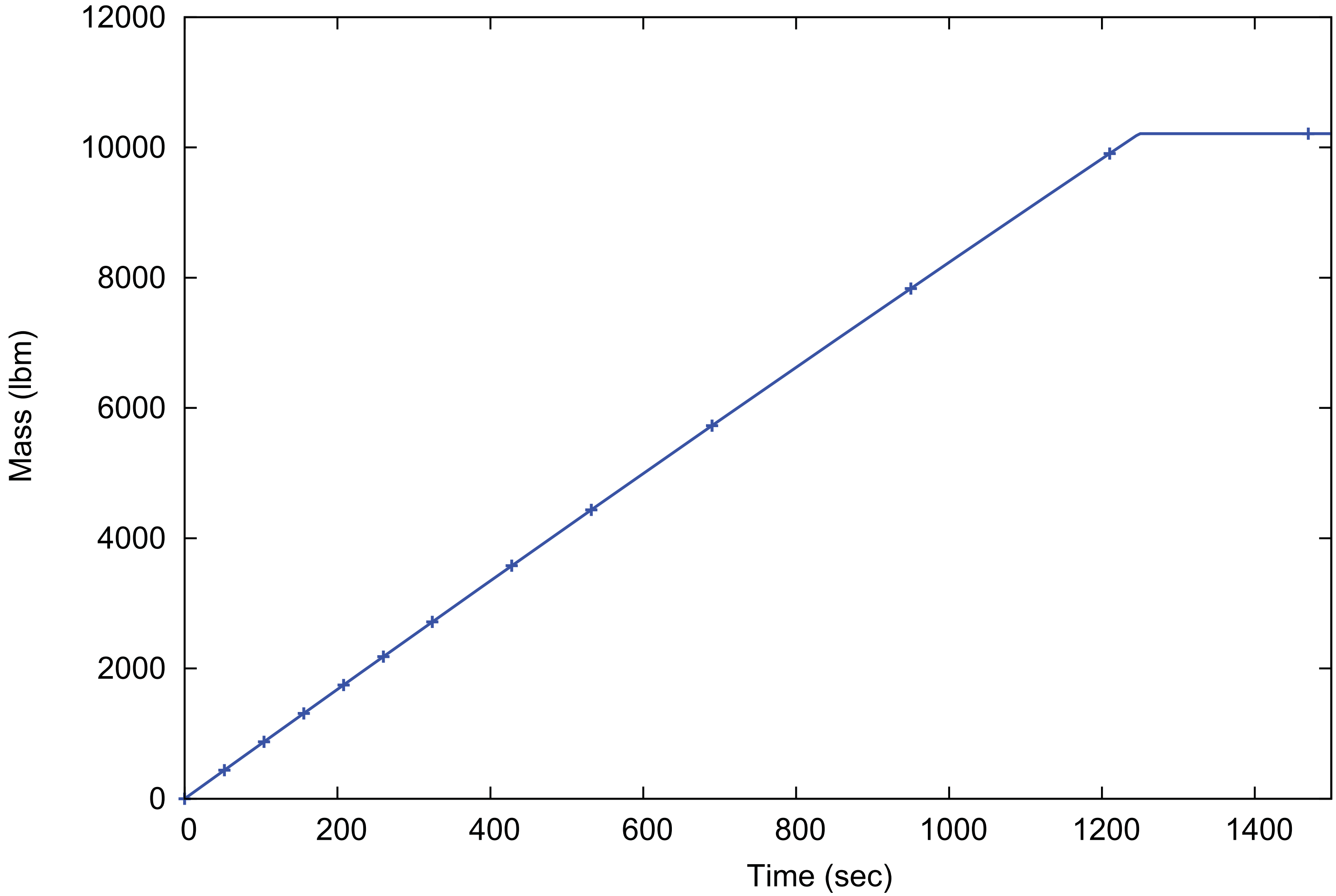


Figure 15.6-19: Steam Generator Tube Failure - Limiting Mass Release Scenario - Steam Generator Level

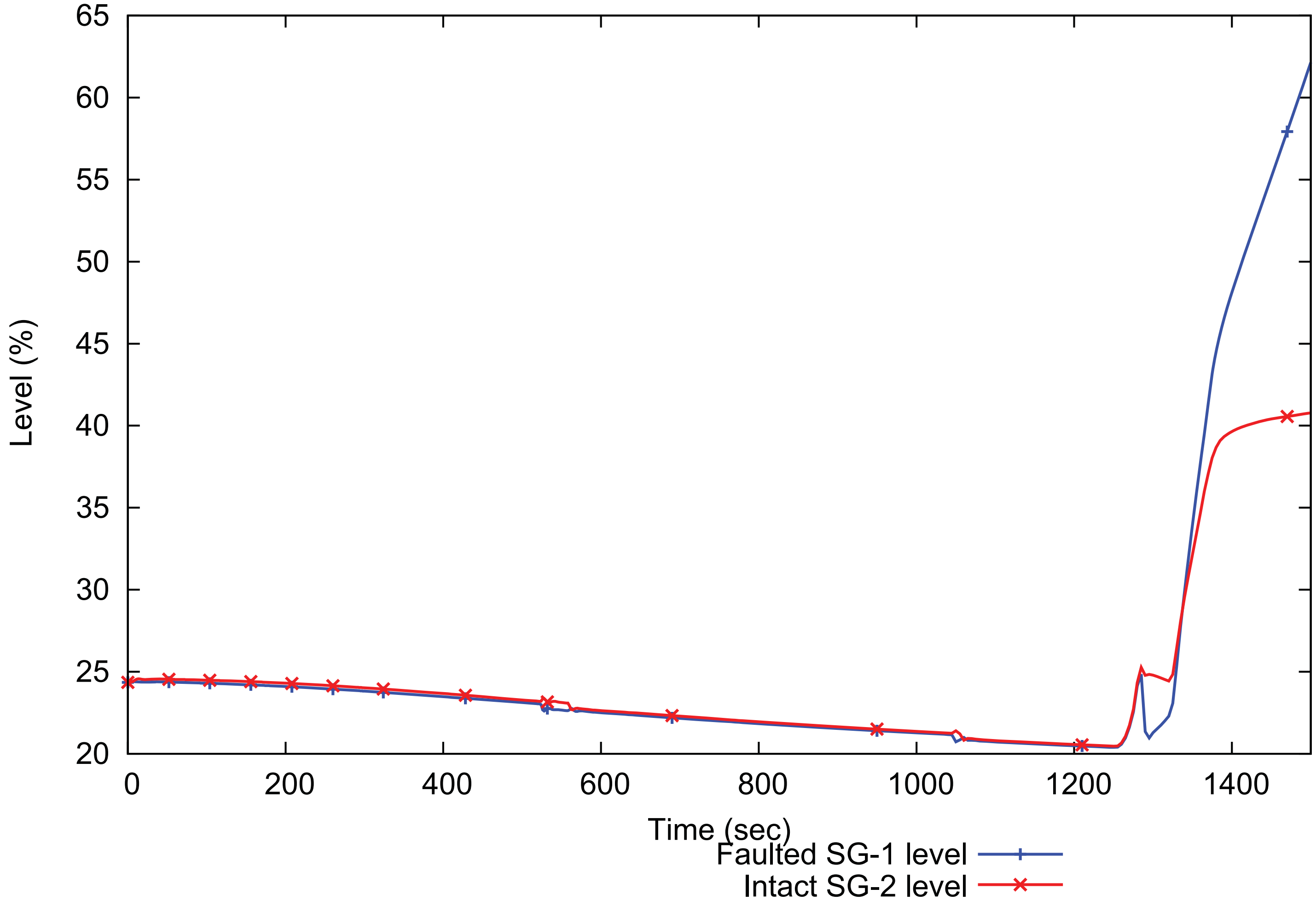


Figure 15.6-20: Steam Generator Tube Failure - Limiting Mass Release Scenario - Steam Generator Level

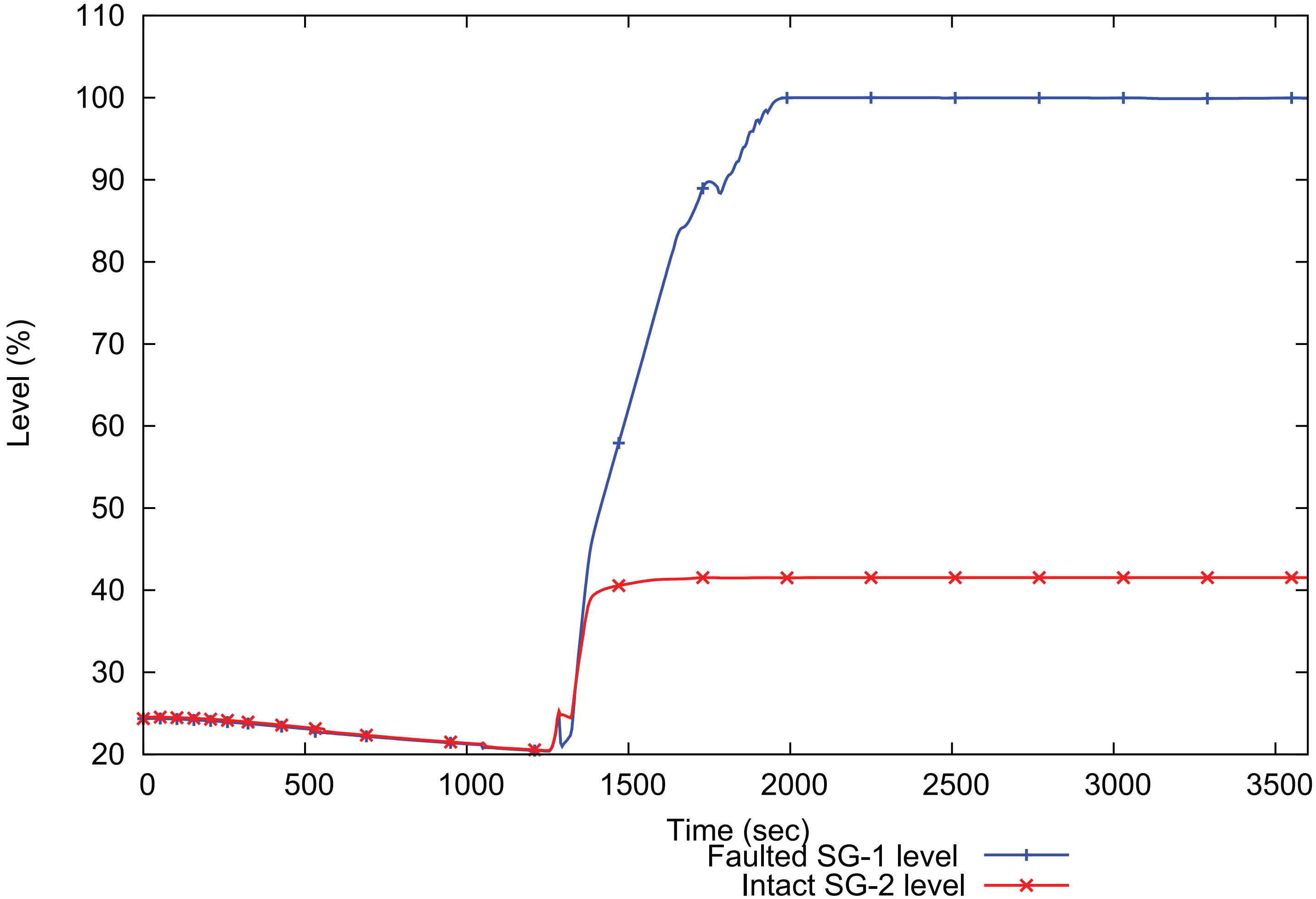


Figure 15.6-21: Steam Generator Tube Failure - Limiting Mass Release Scenario - RCS and SG Pressure

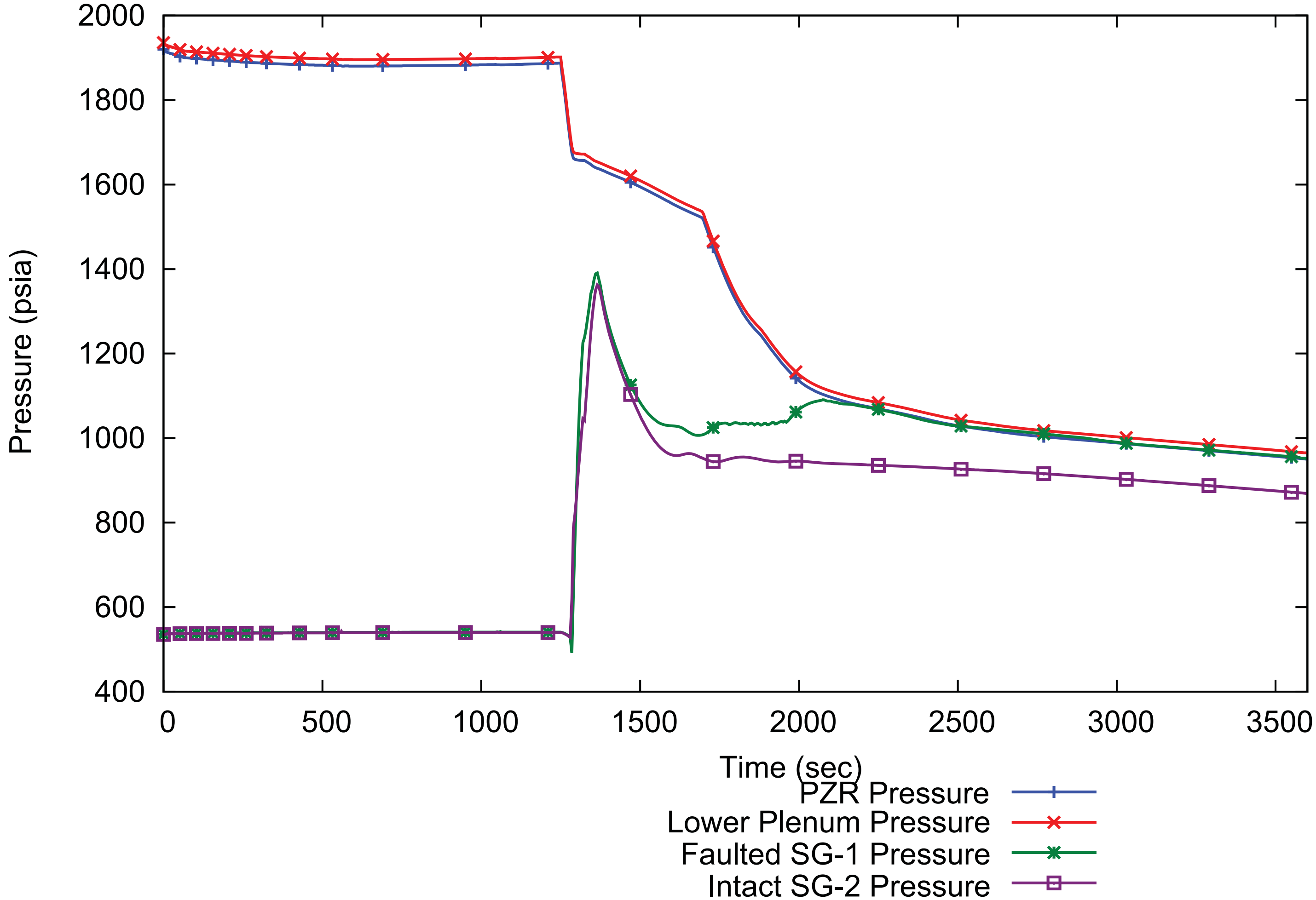


Figure 15.6-22: Steam Generator Tube Failure - Limiting Reactor Pressure Vessel Pressure Scenario - Reactor Pressure Vessel and Steam Generator Pressures

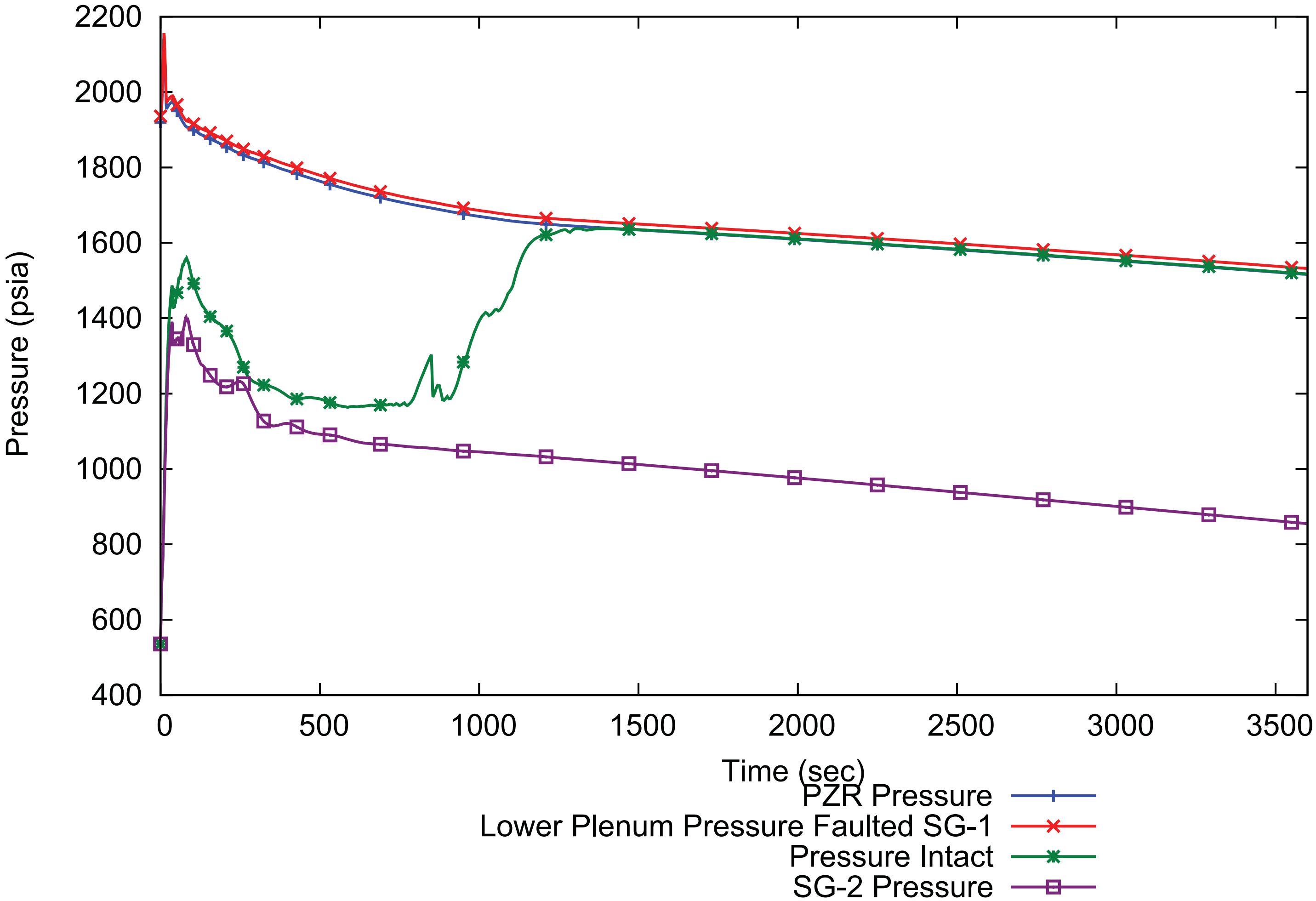


Figure 15.6-23: Steam Generator Tube Failure – Limiting Reactor Pressure Vessel Pressure Scenario – Reactor Pressure Vessel and Steam Generator Pressures

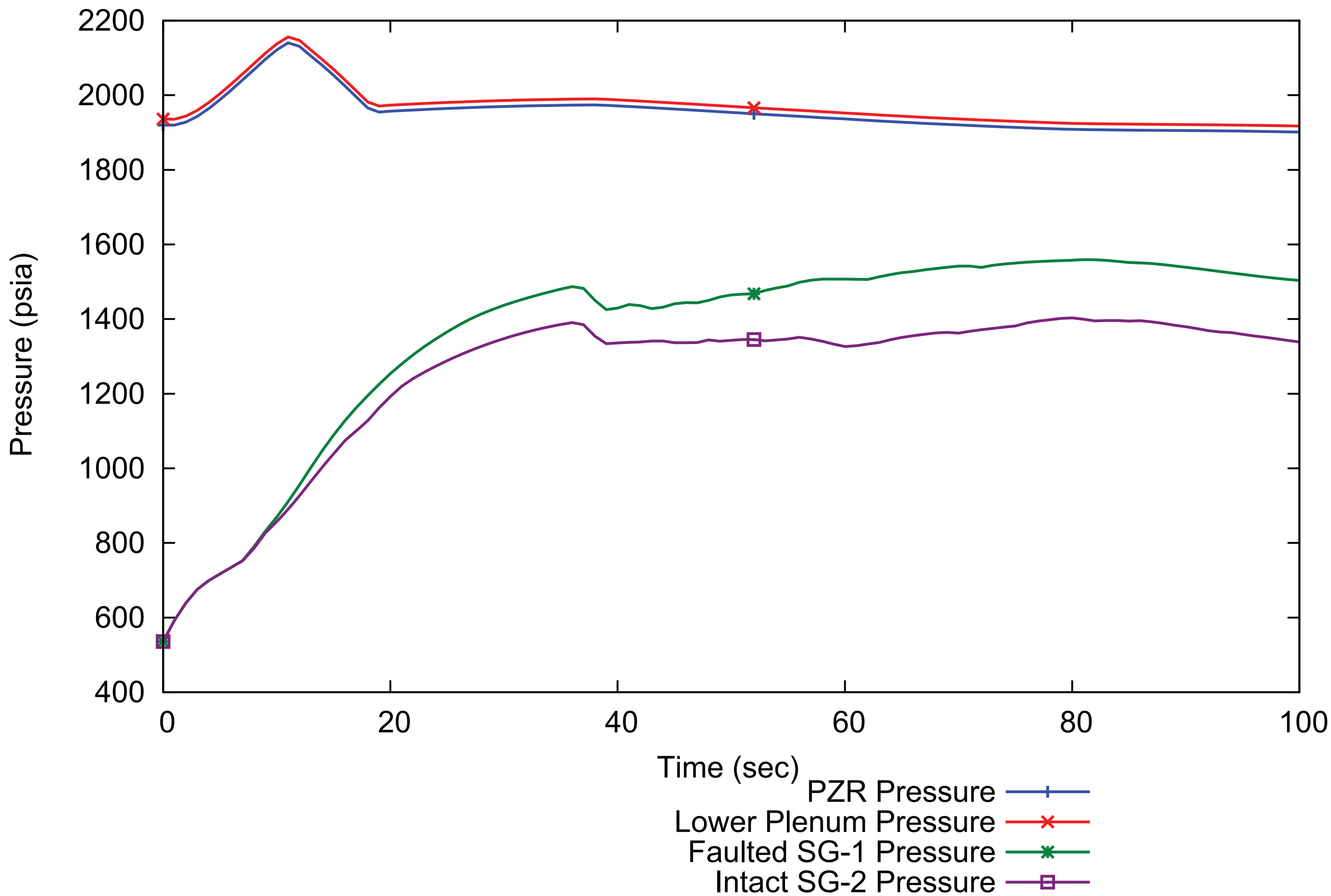


Figure 15.6-24: Steam Generator Tube Failure – Limiting Reactor Pressure Vessel Pressure Scenario – Core and Steam Generator Power

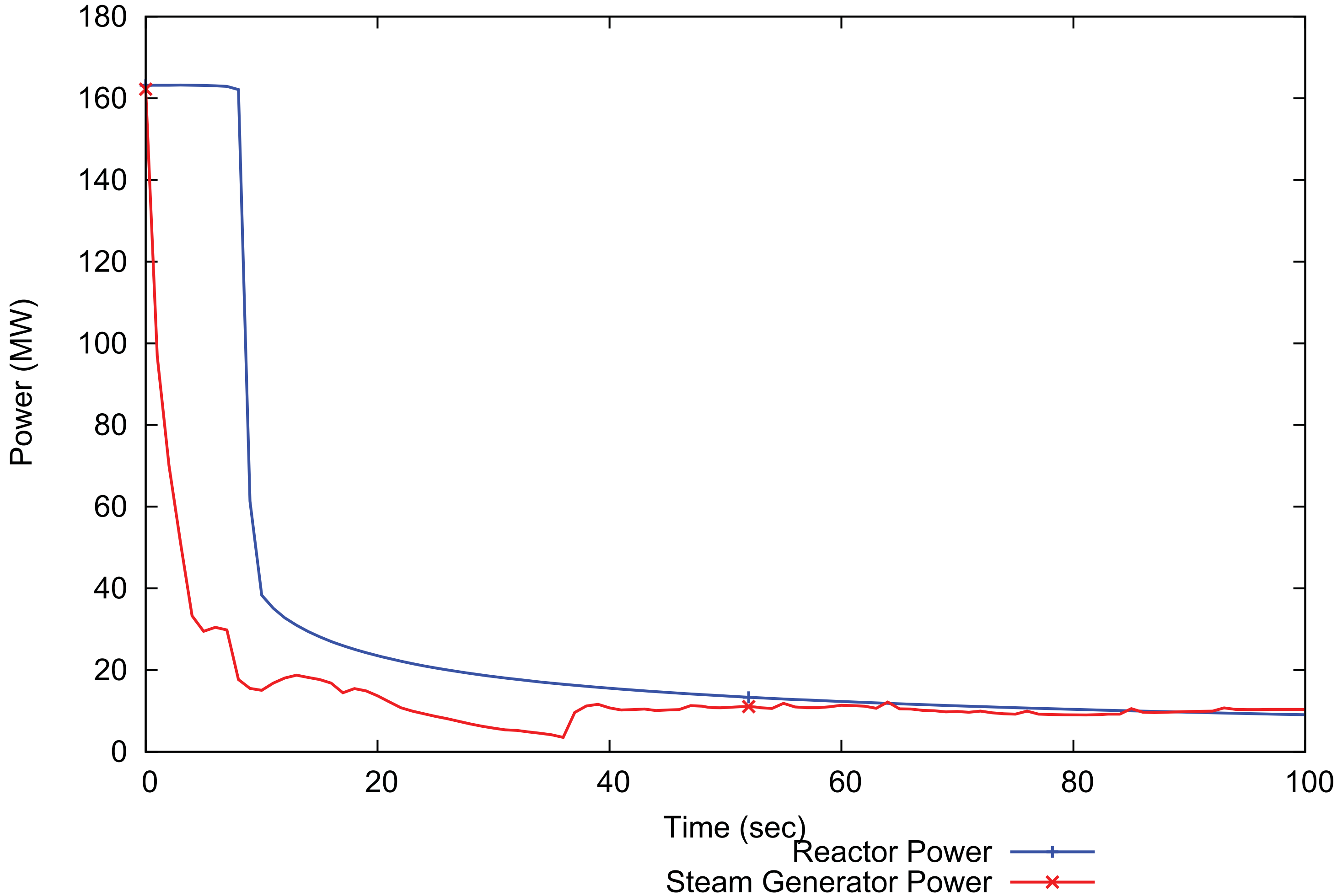


Figure 15.6-25: Steam Generator Tube Failure – Limiting Reactor Pressure Vessel Pressure Scenario – Core and Steam Generator Power

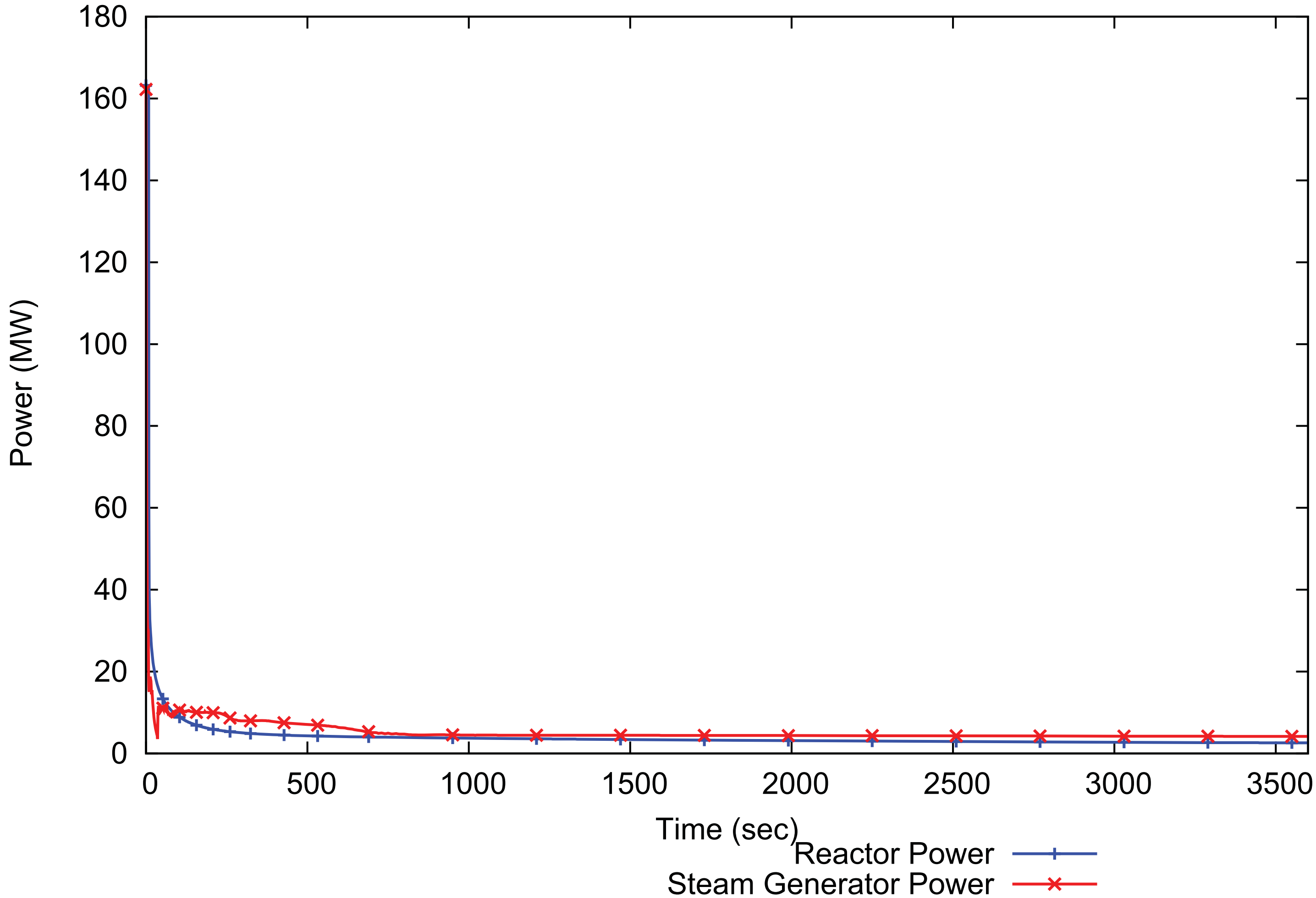


Figure 15.6-26: Steam Generator Tube Failure – Limiting Reactor Pressure Vessel Pressure Scenario – Decay Heat Removal System Flow

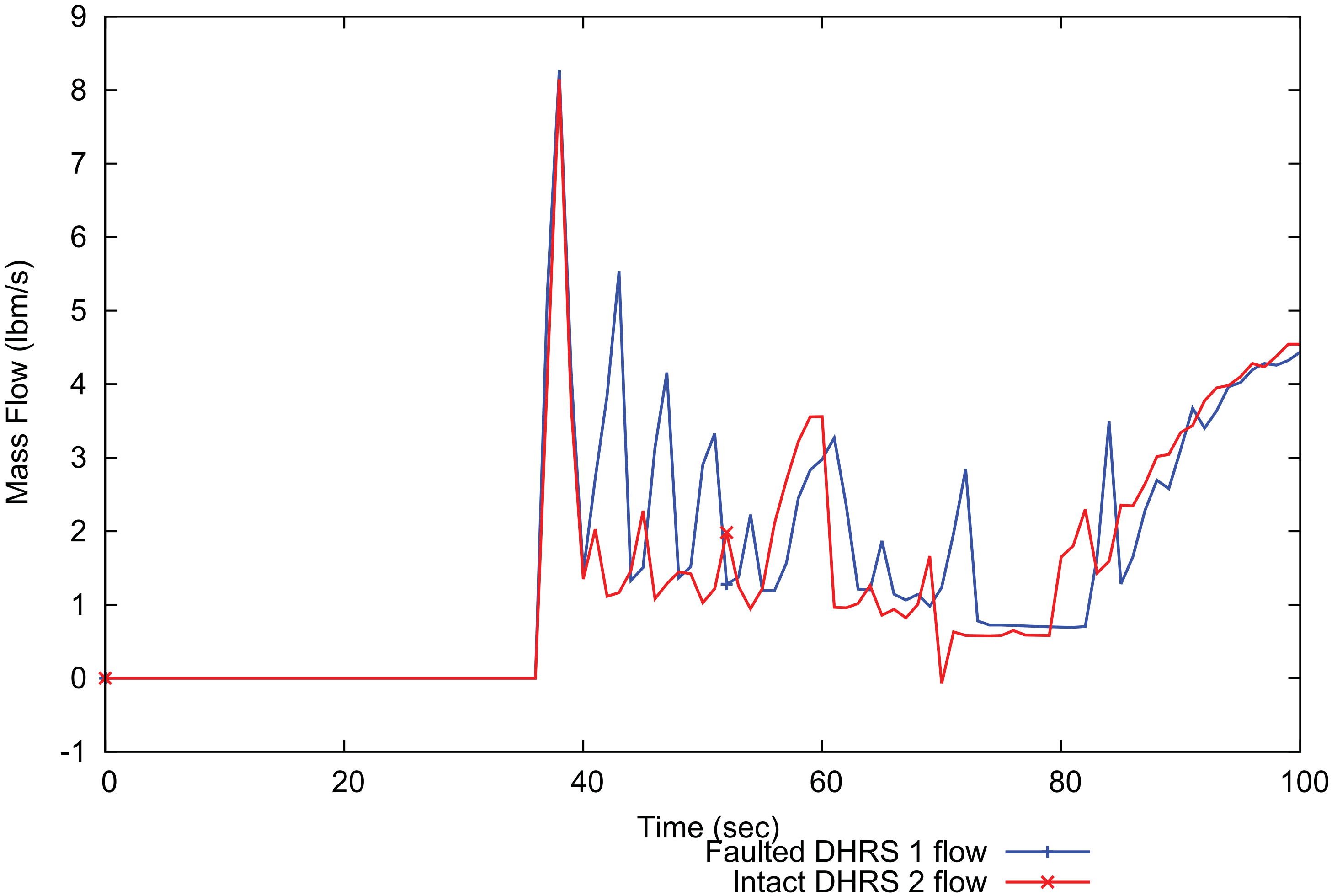


Figure 15.6-27: Steam Generator Tube Failure – Limiting Reactor Pressure Vessel Pressure Scenario – Decay Heat Removal System Flow

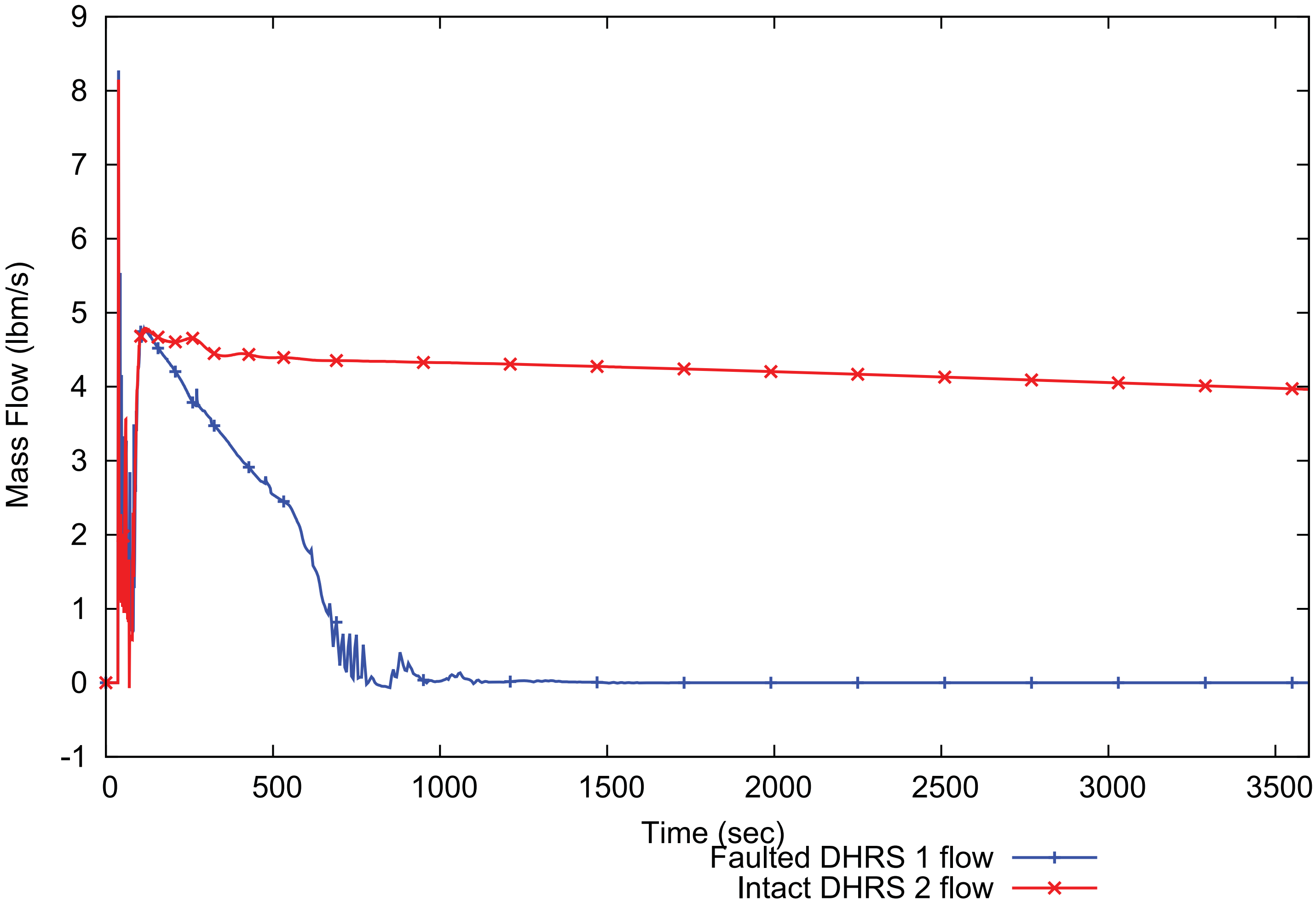


Figure 15.6-28: Steam Generator Tube Failure – Limiting Reactor Pressure Vessel Pressure Scenario – Steam Generator Level

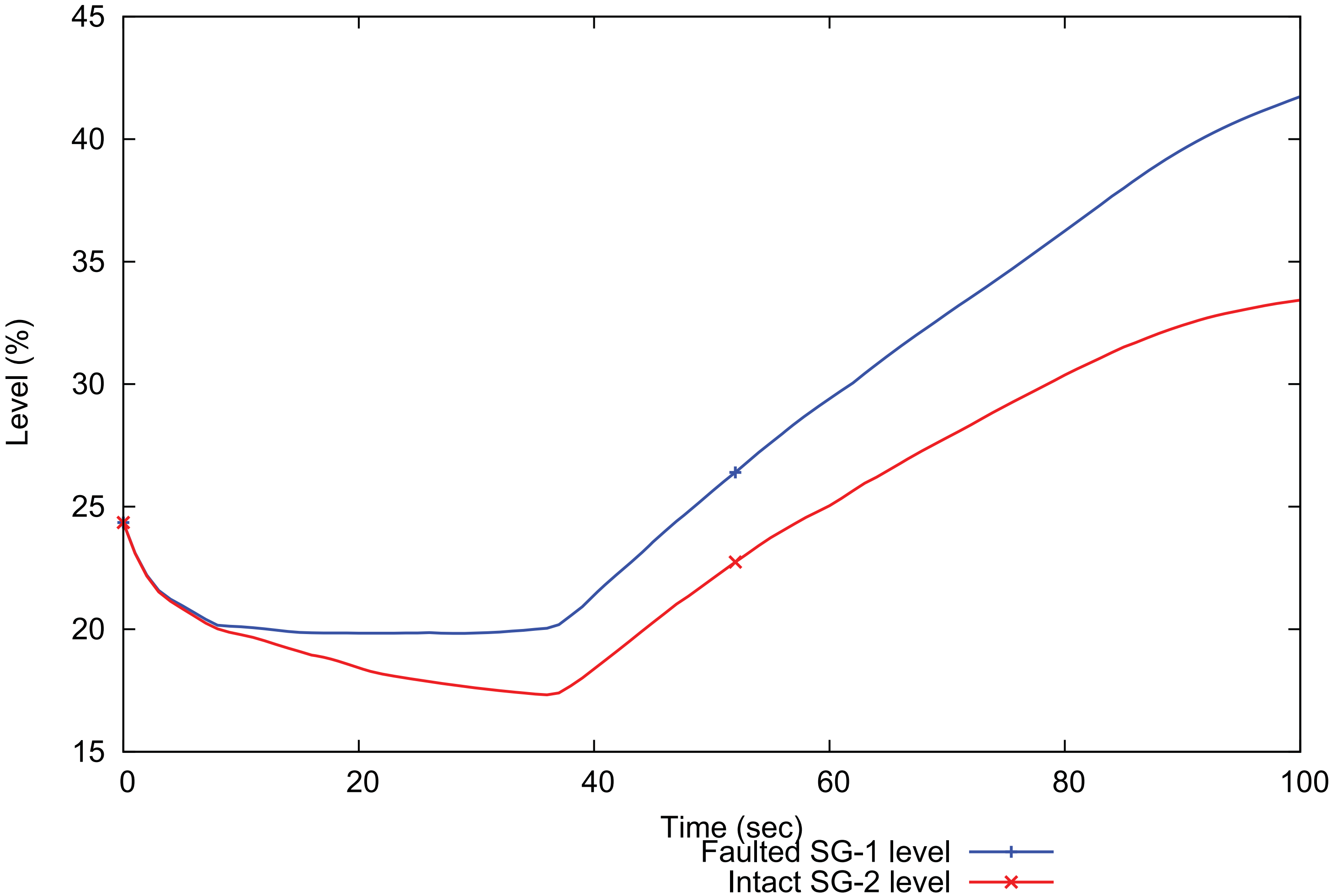


Figure 15.6-29: Steam Generator Tube Failure – Limiting Iodine Spiking Time - Mass Release to SG

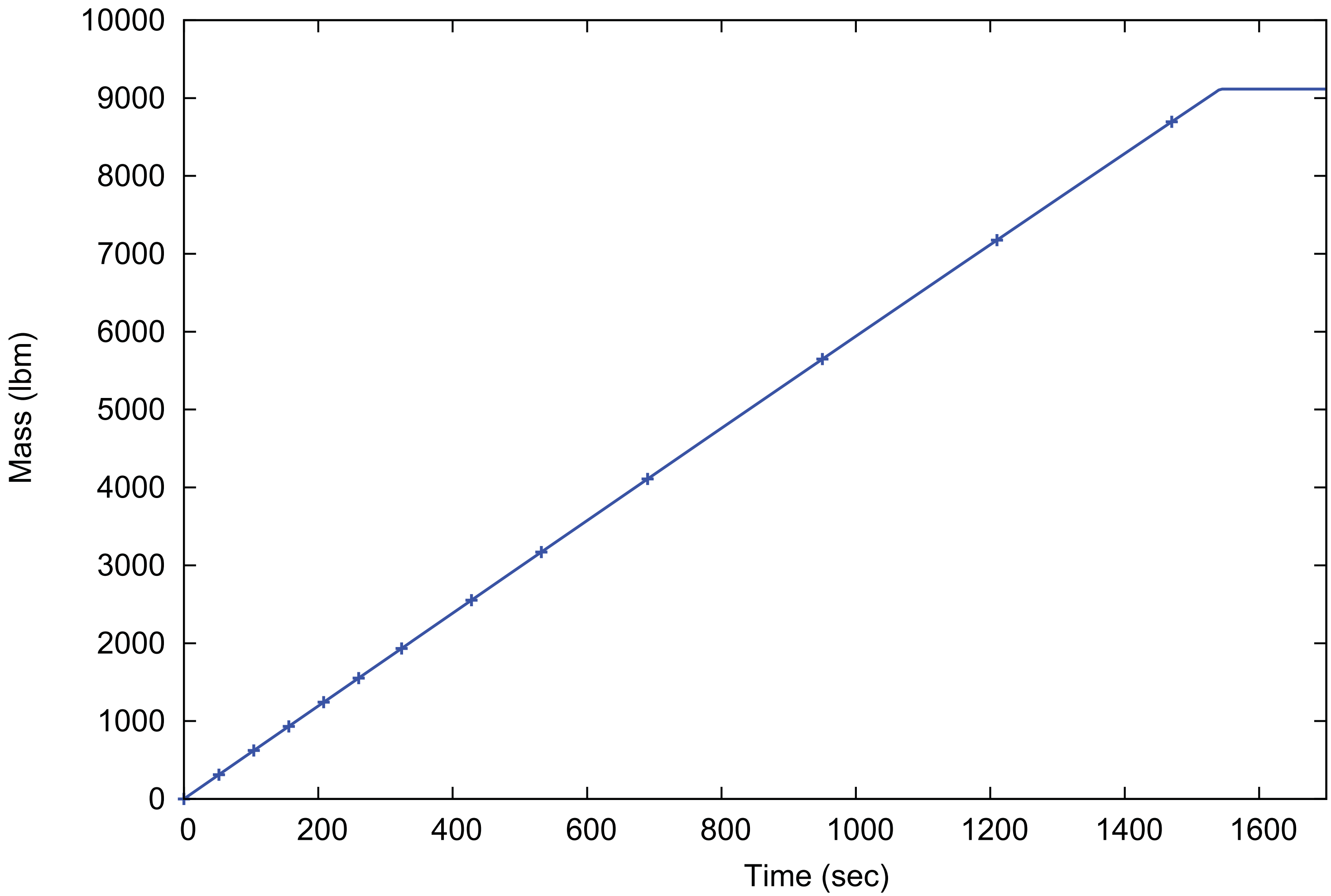


Figure 15.6-30: Steam Generator Tube Failure – Limiting Iodine Spiking Time - SG Levels

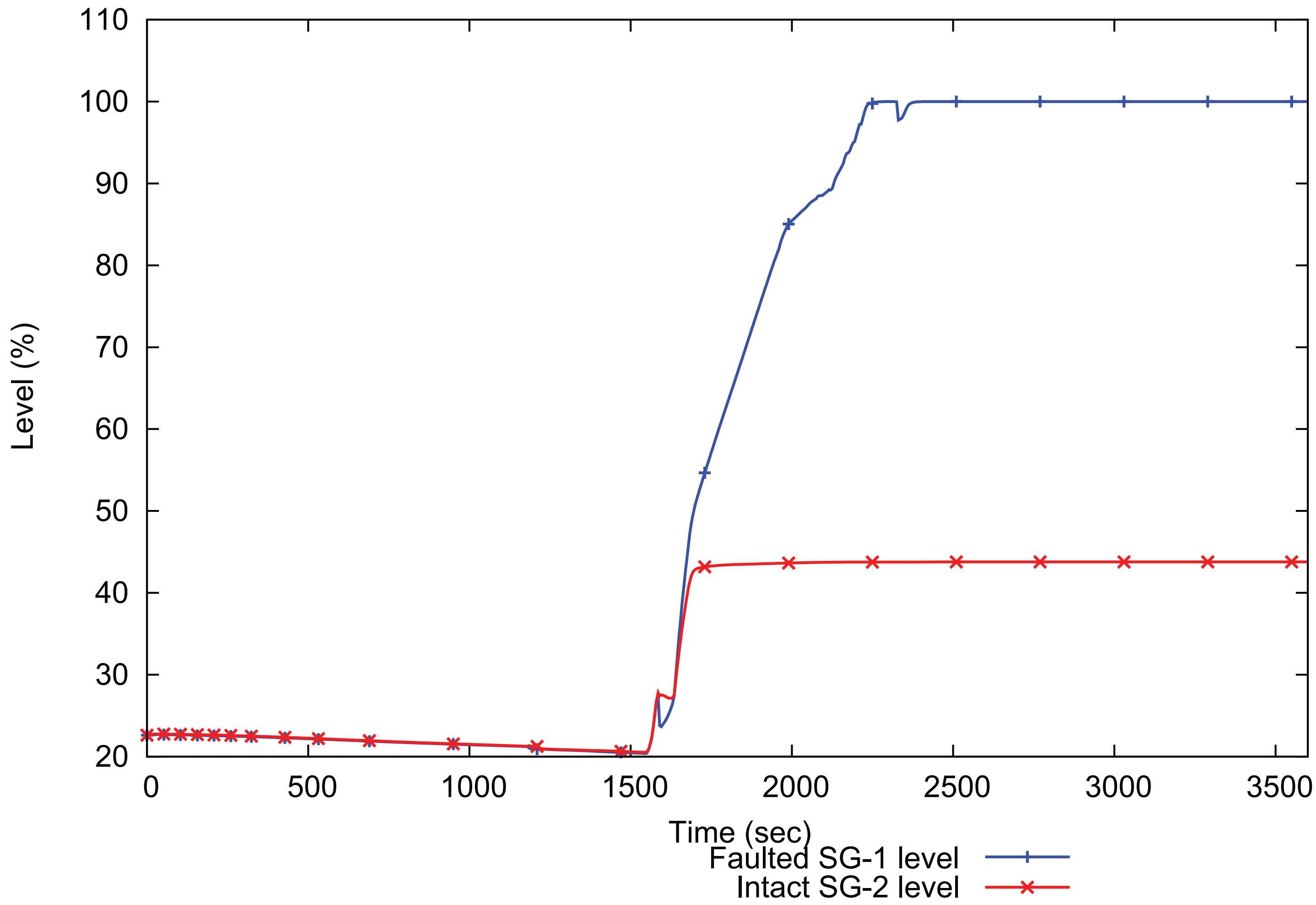


Figure 15.6-31: Steam Generator Tube Failure – Limiting Reactor Pressure Vessel Pressure Scenario – Reactor Coolant System Temperature

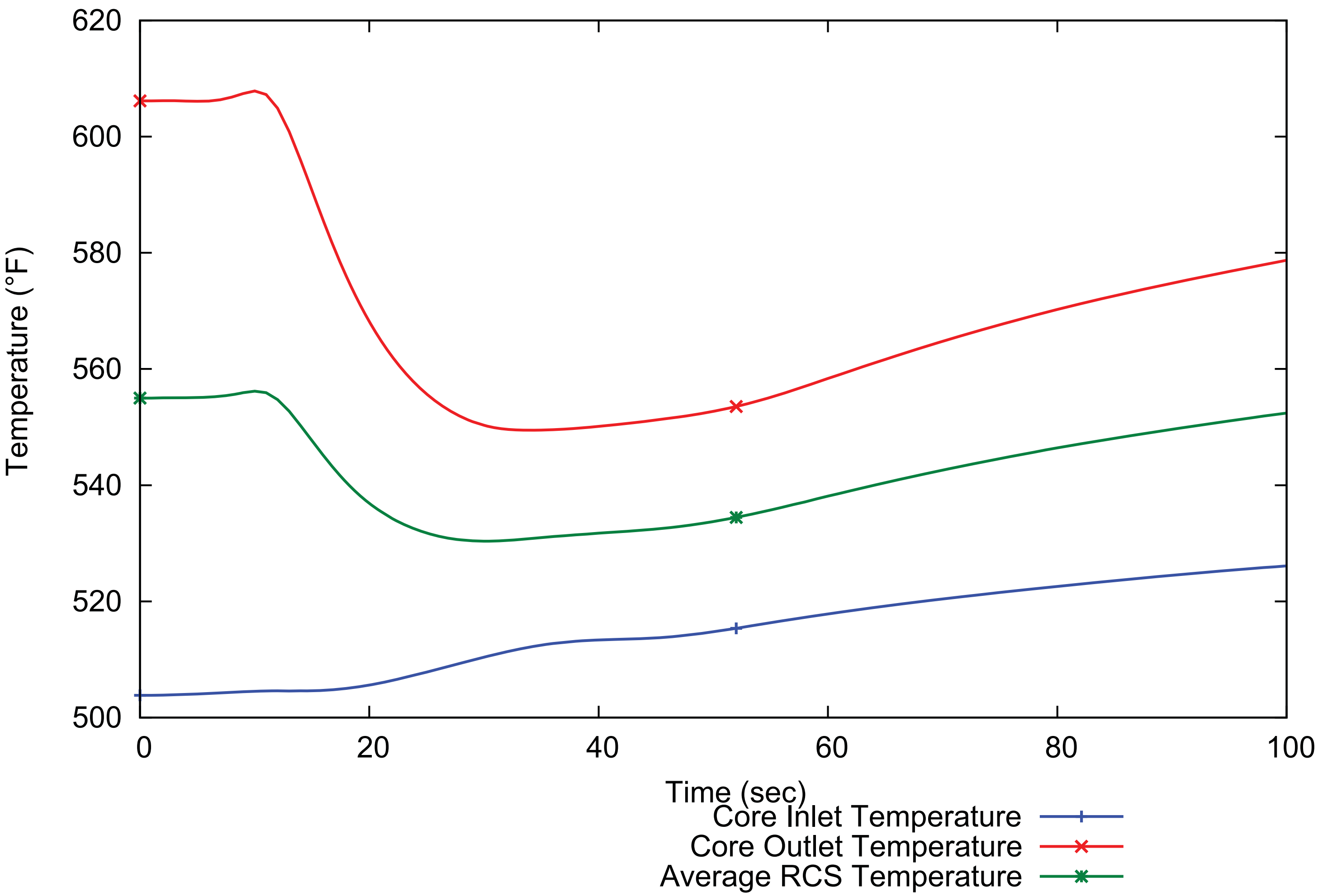


Figure 15.6-32: Steam Generator Tube Failure – Limiting Reactor Pressure Vessel Pressure Scenario – Reactor Coolant System Temperature

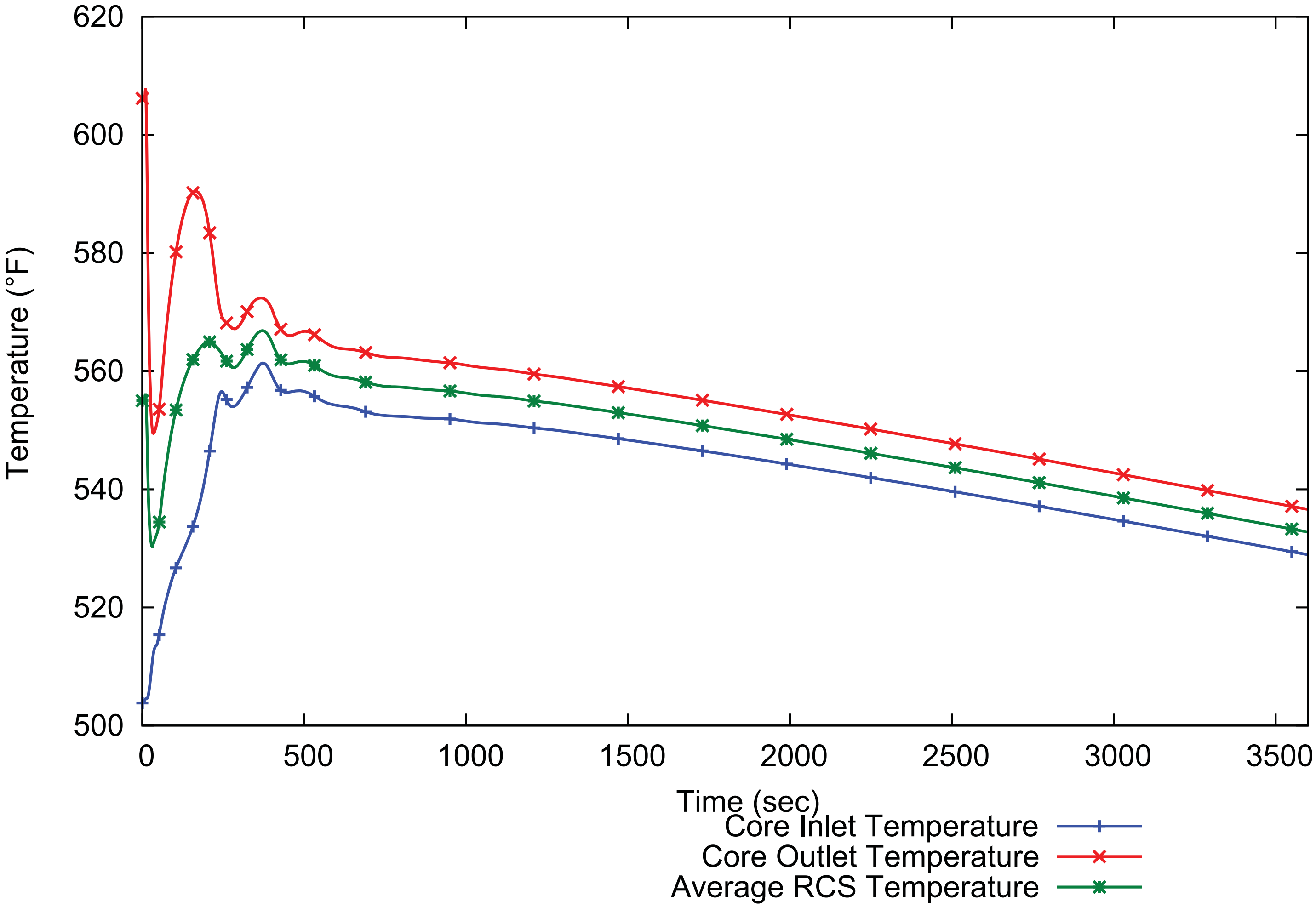


Figure 15.6-33: Steam Generator Tube Failure – Limiting Reactor Pressure Vessel Pressure Scenario – Pressurizer Level

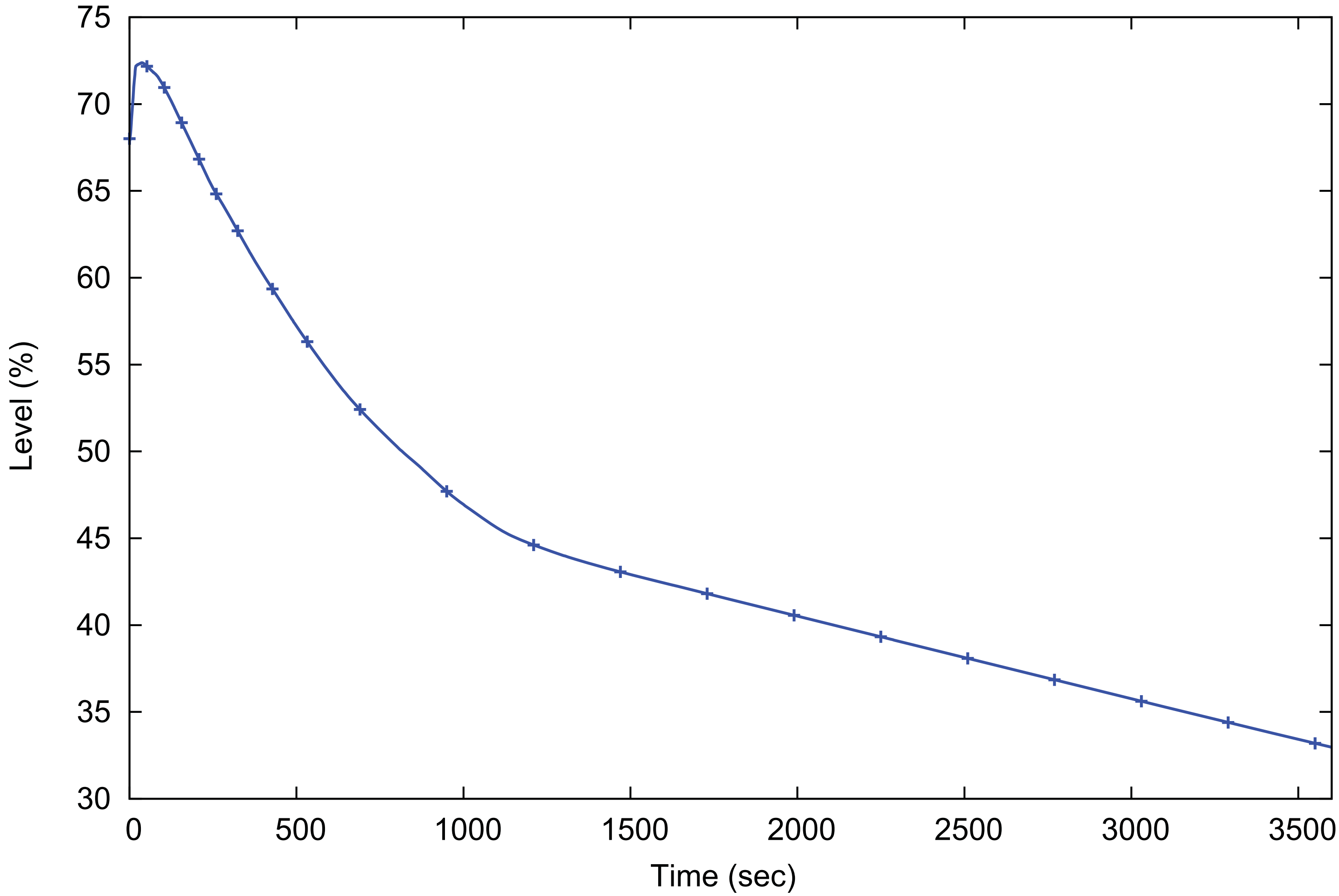


Figure 15.6-34: Steam Generator Tube Failure – Limiting Reactor Pressure Vessel Pressure Scenario – Pressurizer Level

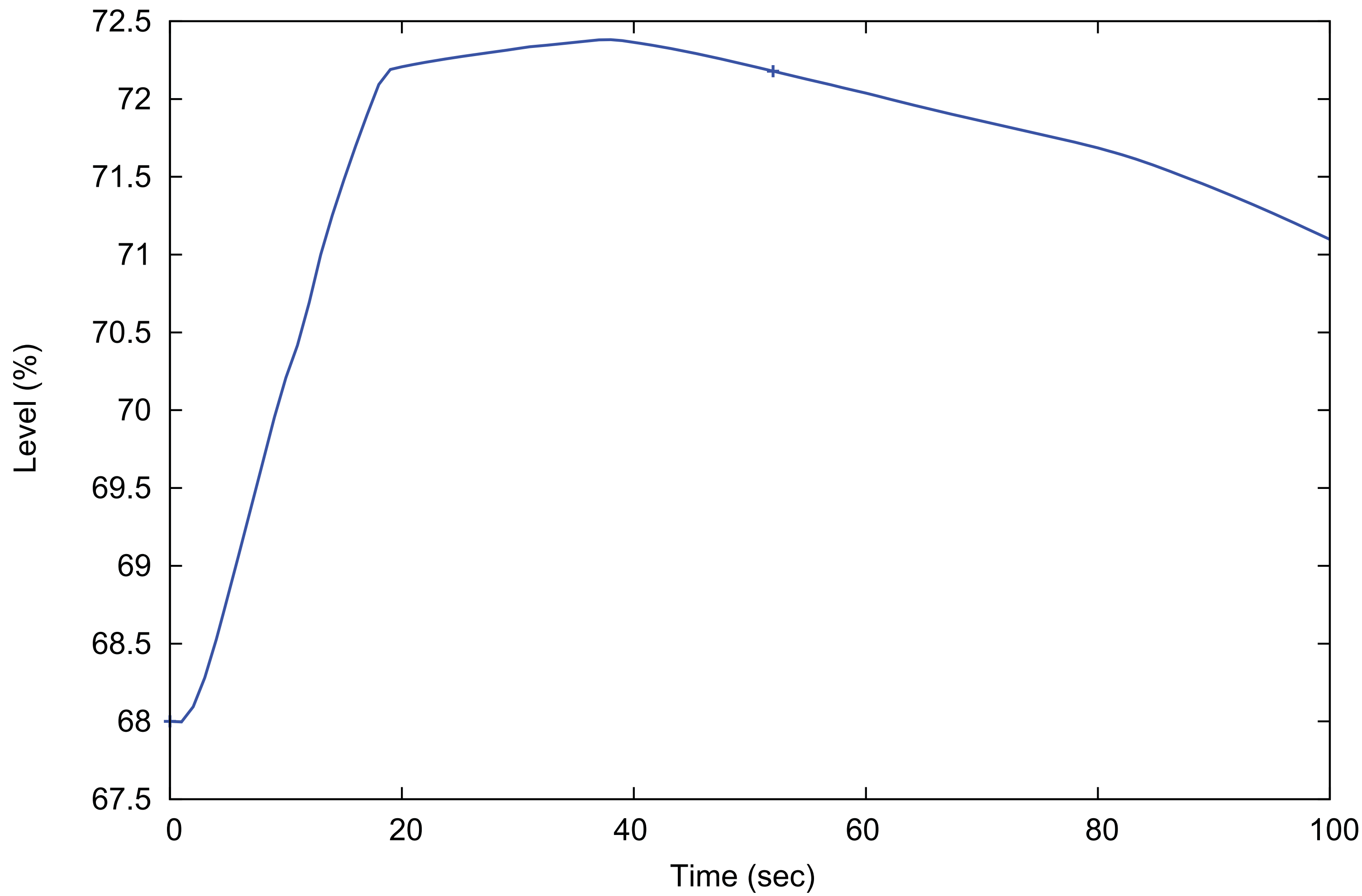


Figure 15.6-35: Steam Generator Tube Failure – Limiting Reactor Pressure Vessel Pressure Scenario – Level above the Top of the Core

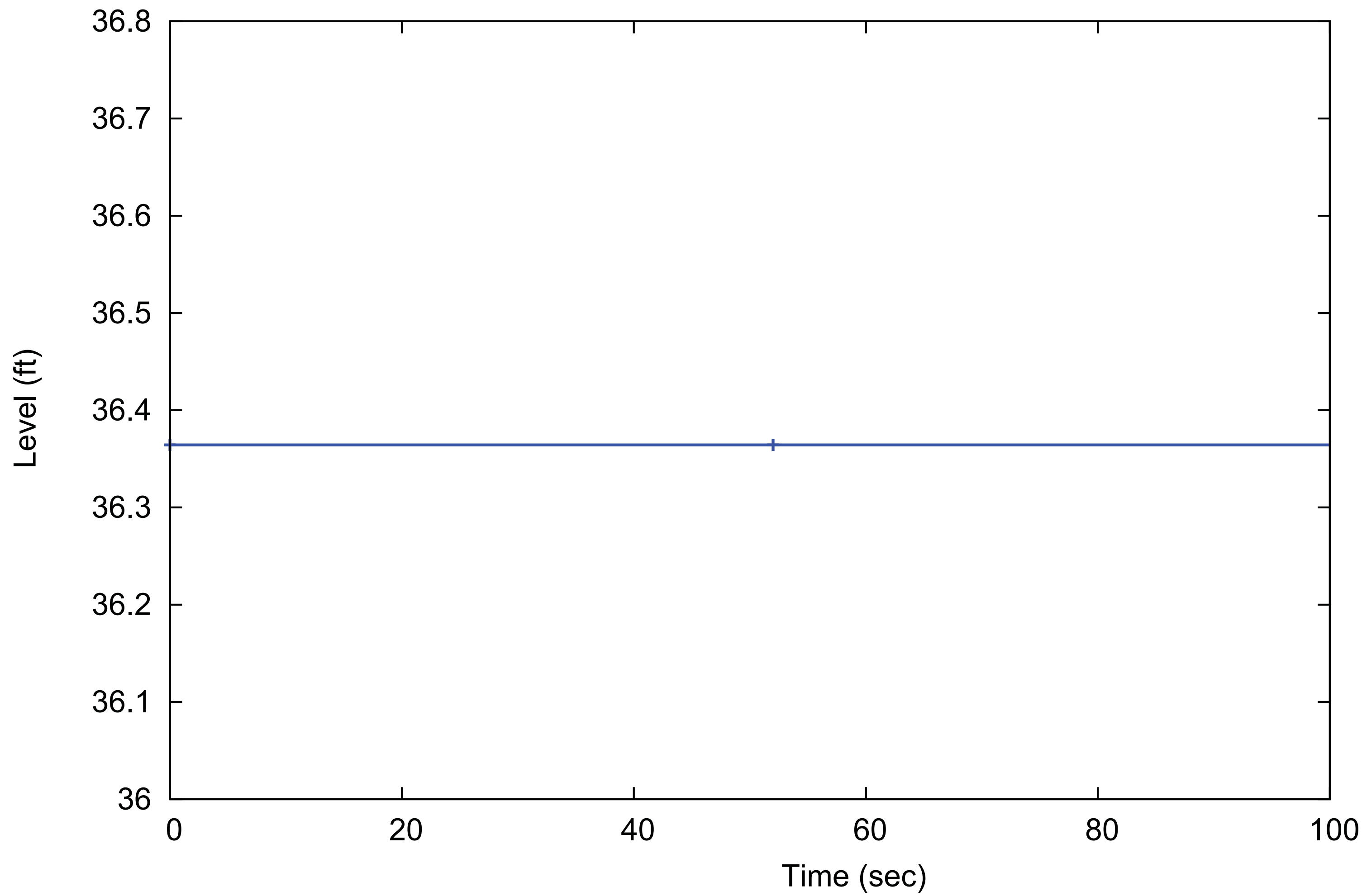


Figure 15.6-36: Steam Generator Tube Failure – Limiting Reactor Pressure Vessel Pressure Scenario – Fuel Volume Average Temperature

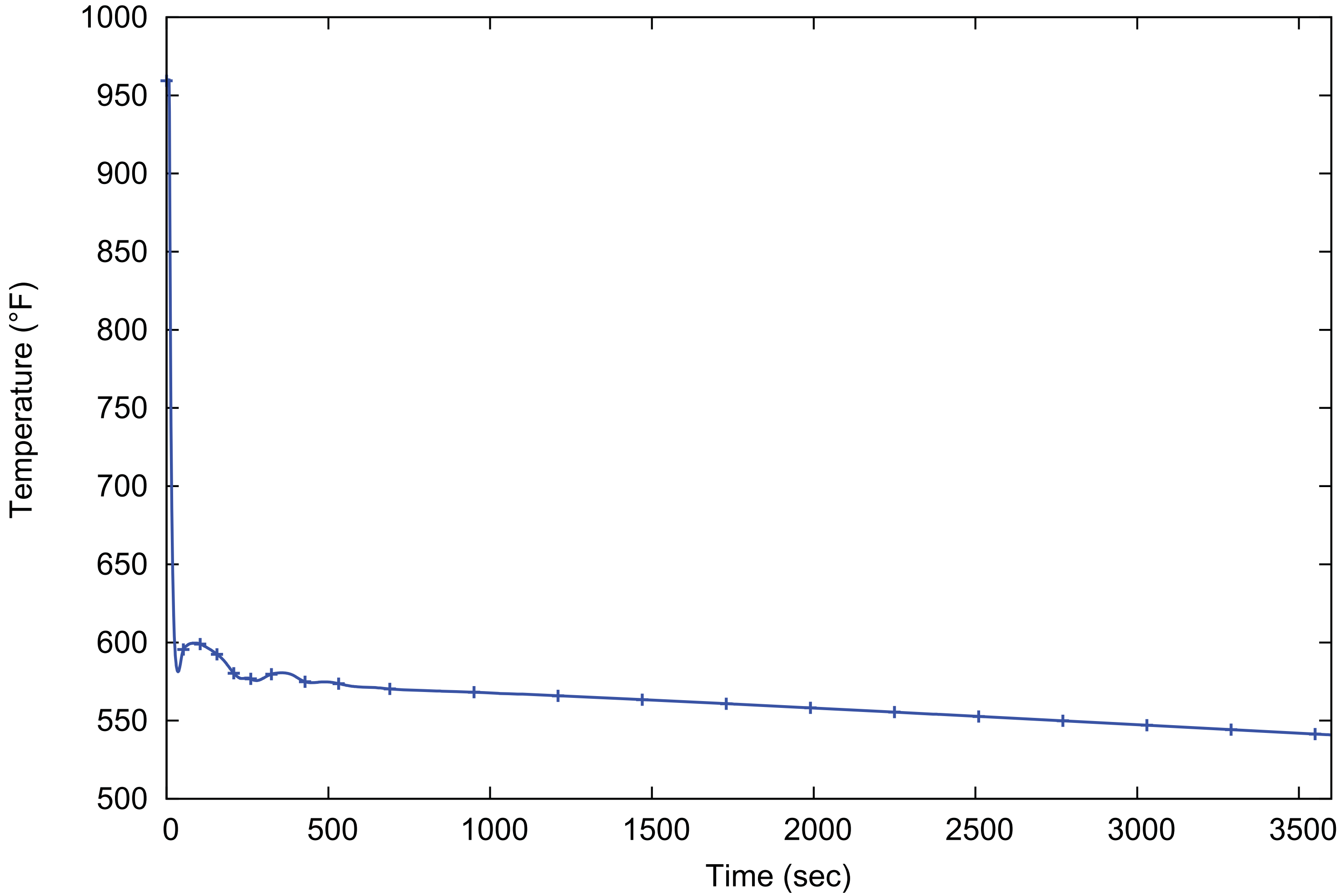


Figure 15.6-37: Steam Generator Tube Failure – Limiting Reactor Pressure Vessel Pressure Scenario – Integrated Mass Release to SG

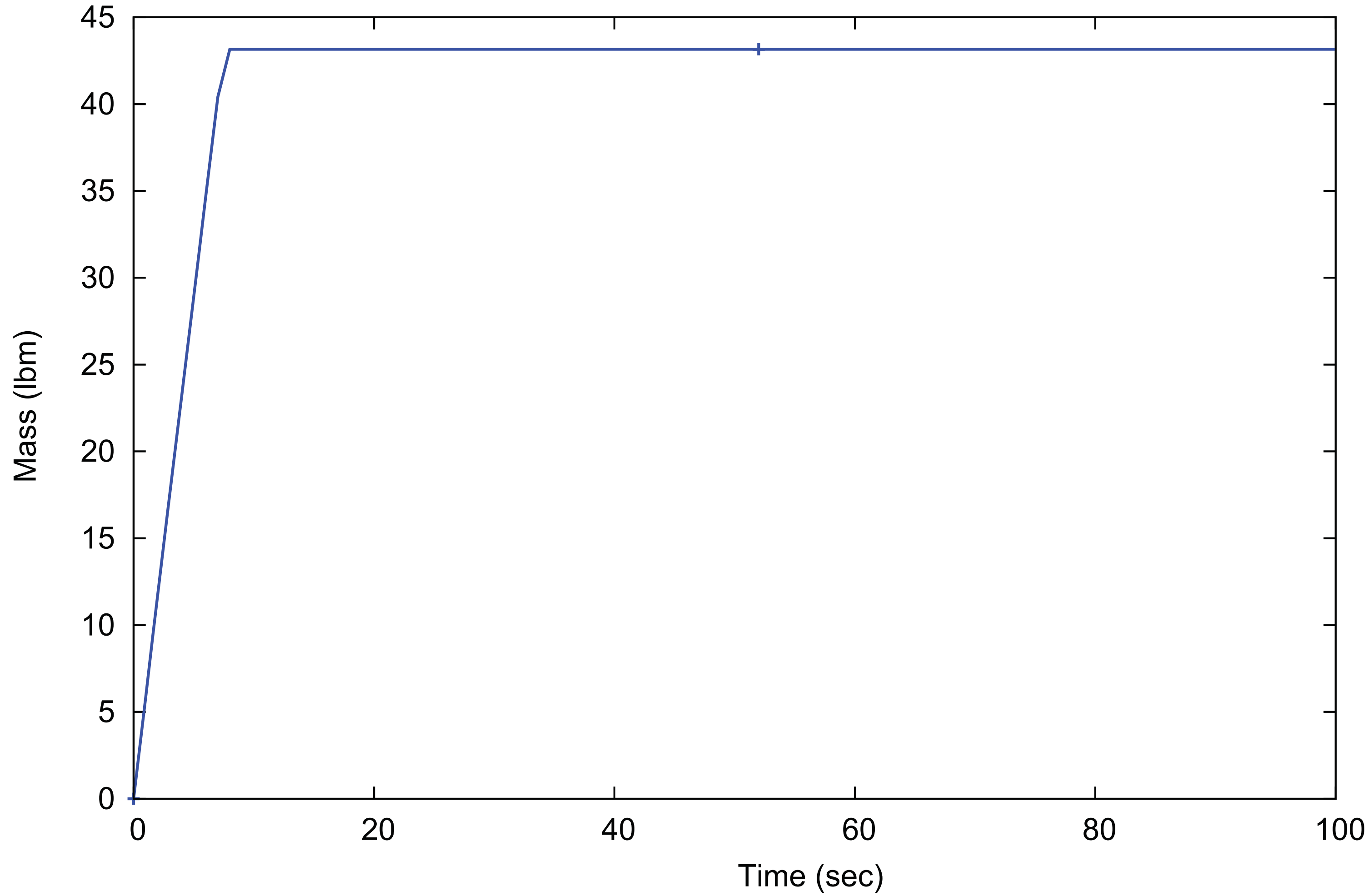


Figure 15.6-38: Steam Generator Tube Failure – Limiting SG Pressure Case - RCS and SG Pressures

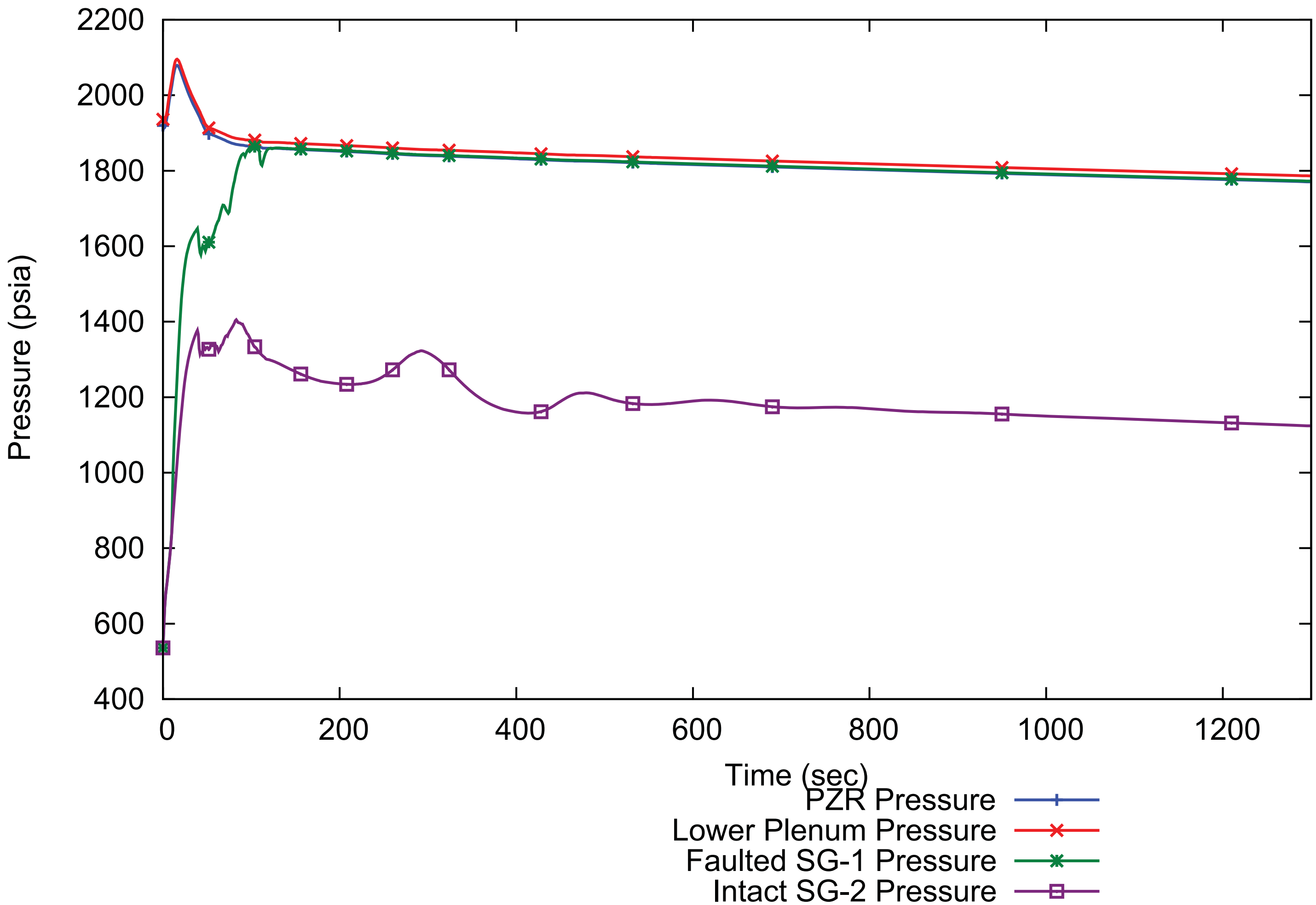


Figure 15.6-39: Steam Generator Tube Failure – Limiting SG Pressure Case - DHRS Flow

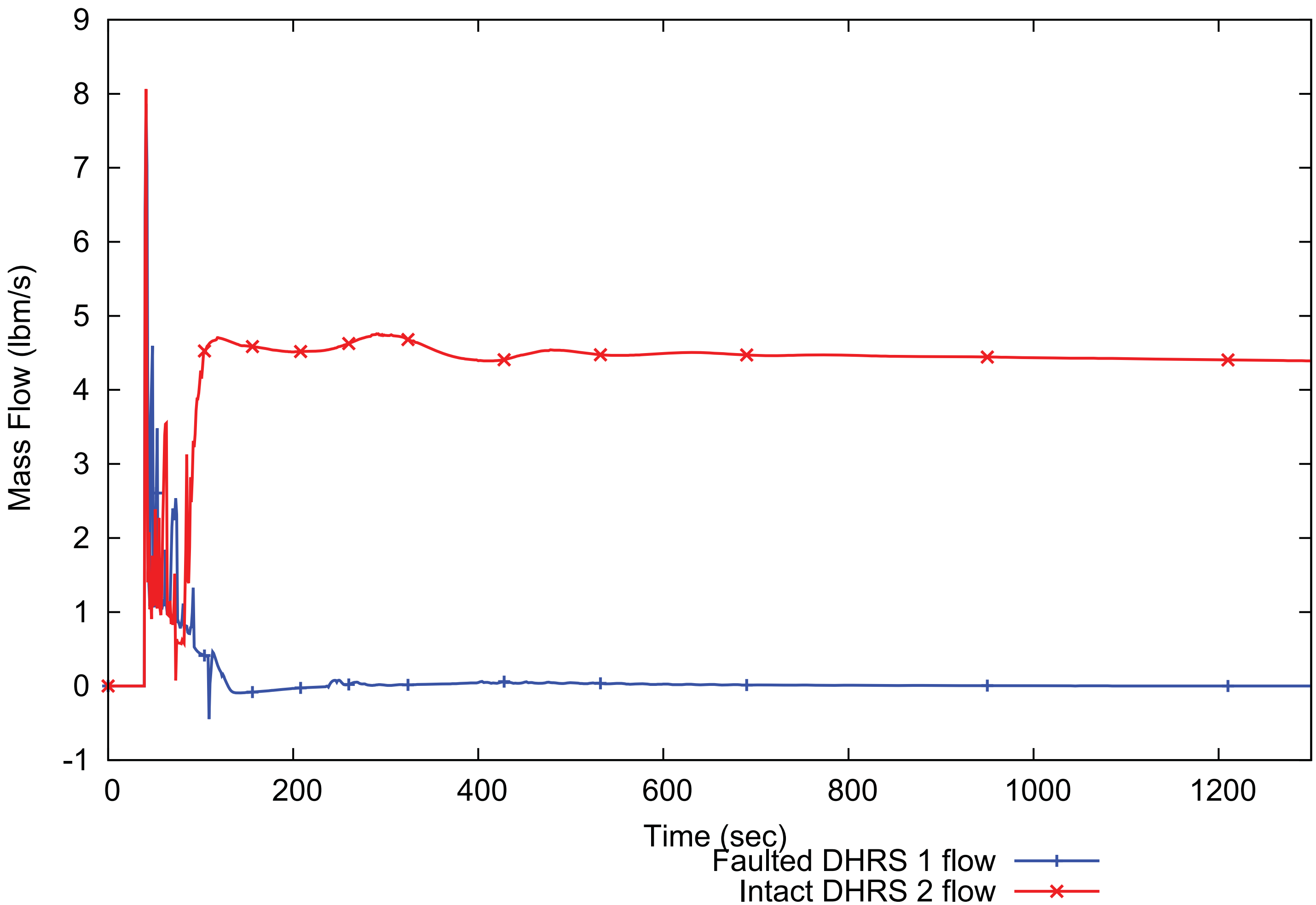


Figure 15.6-40: Steam Generator Tube Failure – Limiting SG Pressure Case - Primary Temperatures

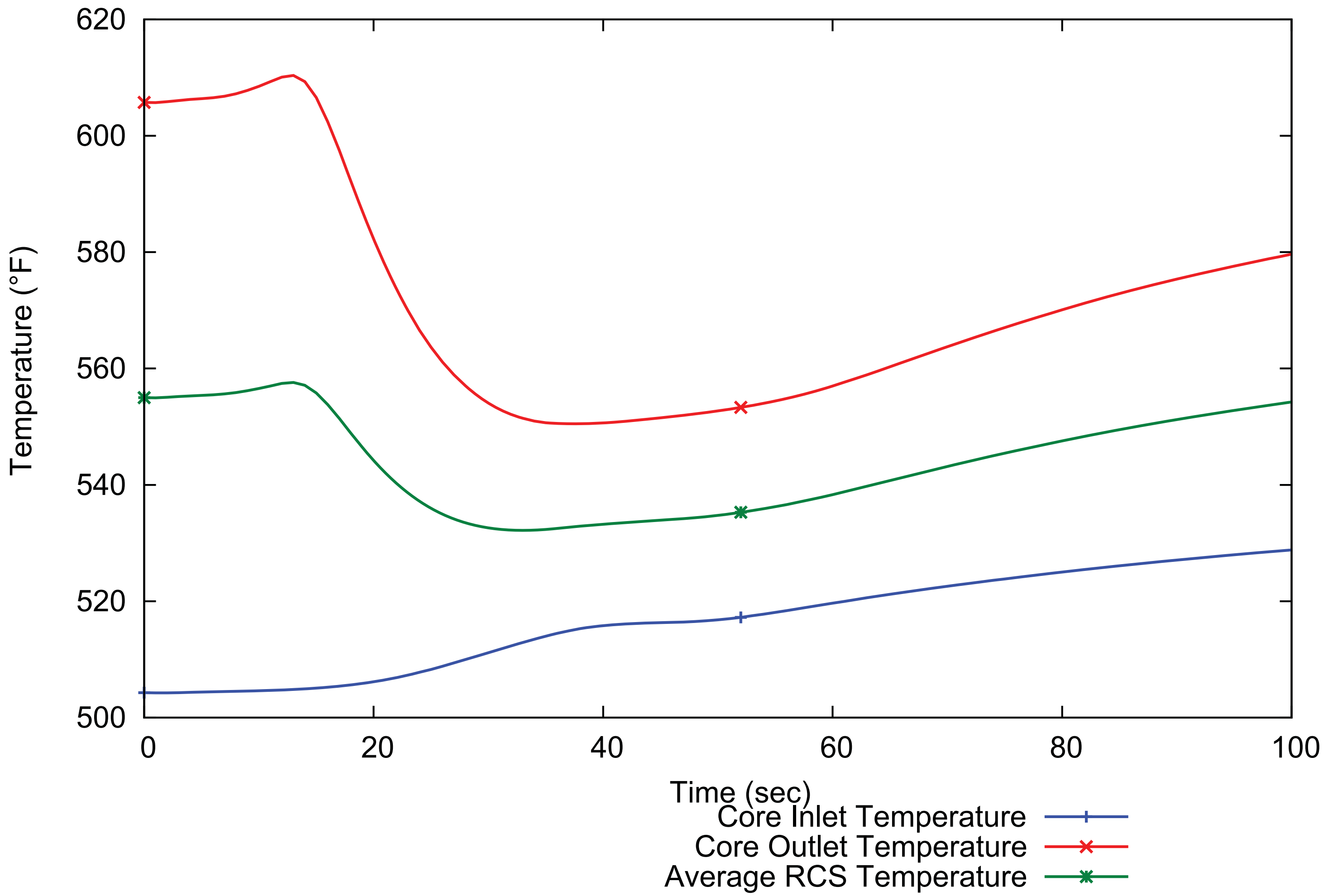


Figure 15.6-41: Loss of Coolant Accident - Break Flow

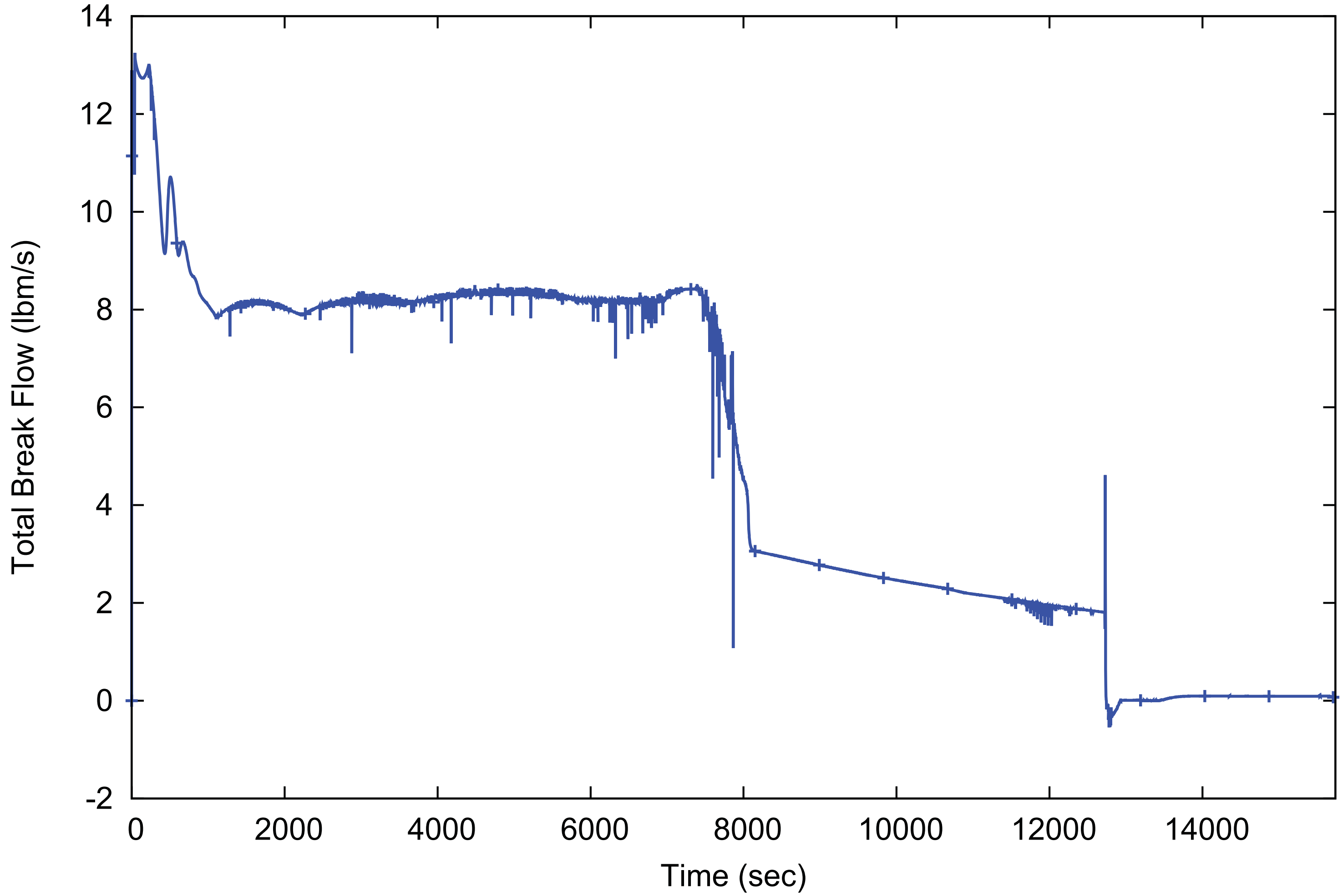


Figure 15.6-42: Loss of Coolant Accident - Reactor Pressure Vessel and Containment Pressure

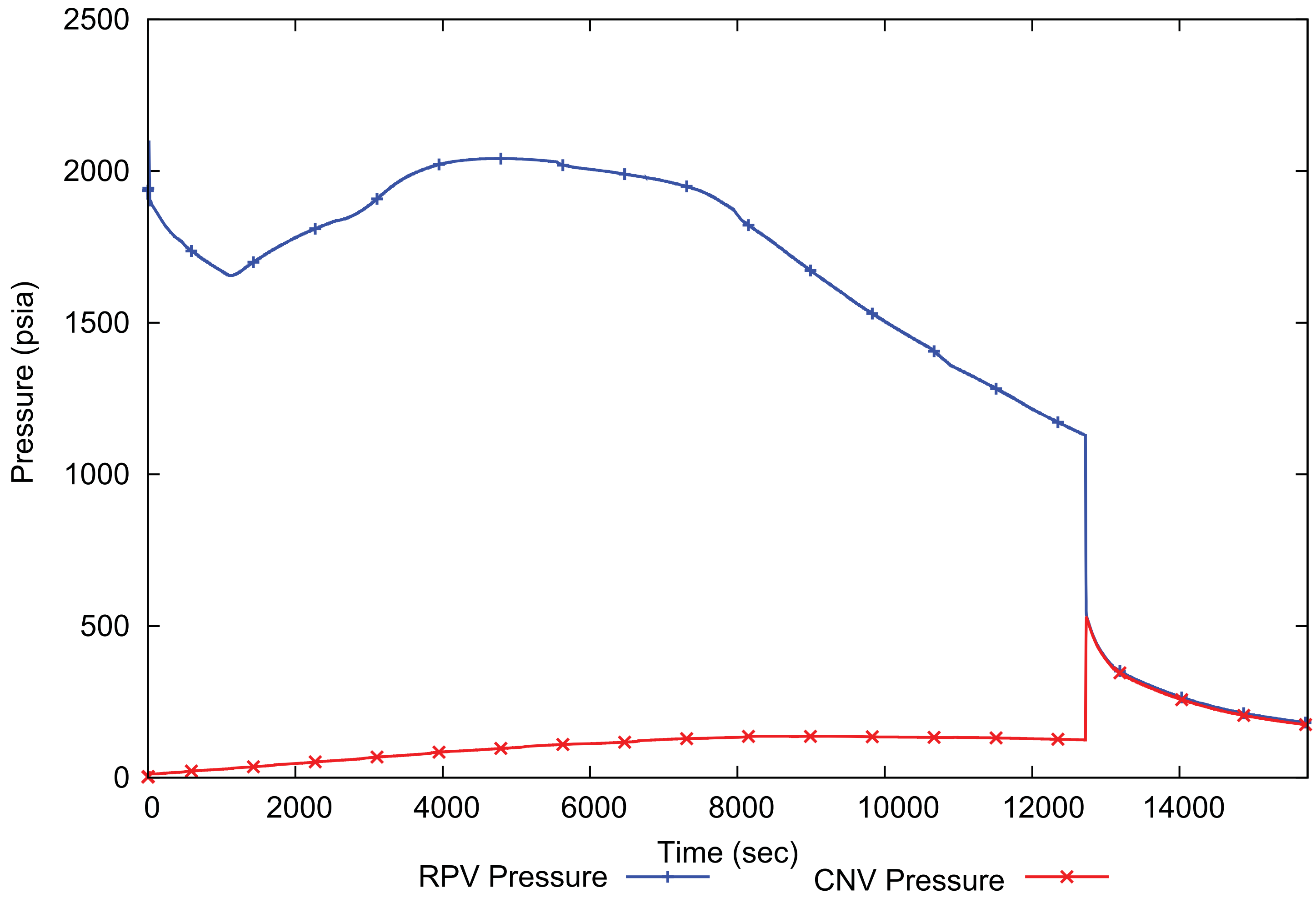


Figure 15.6-43: Loss of Coolant Accident - Reactor Power

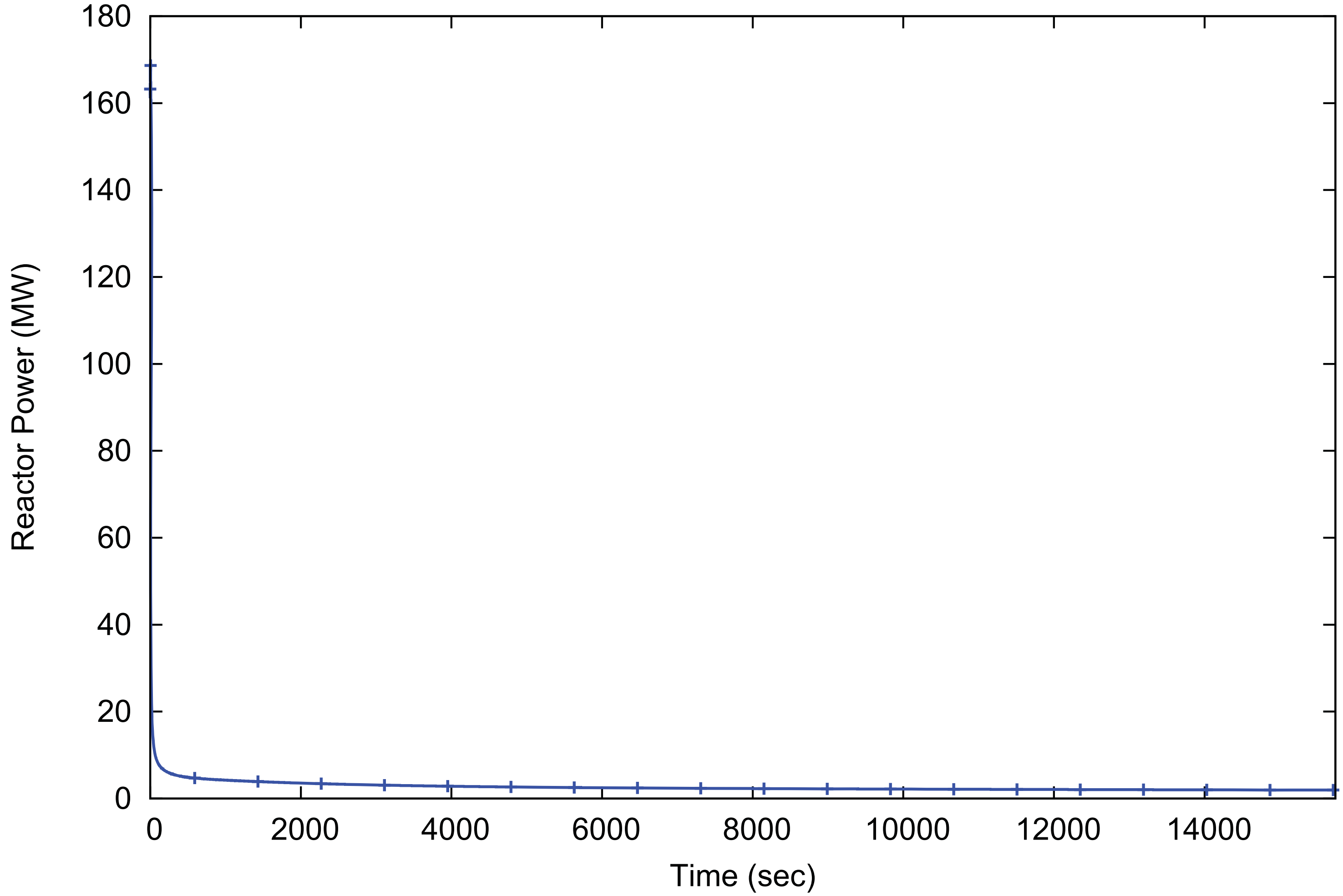


Figure 15.6-44: Loss of Coolant Accident - Reactivity

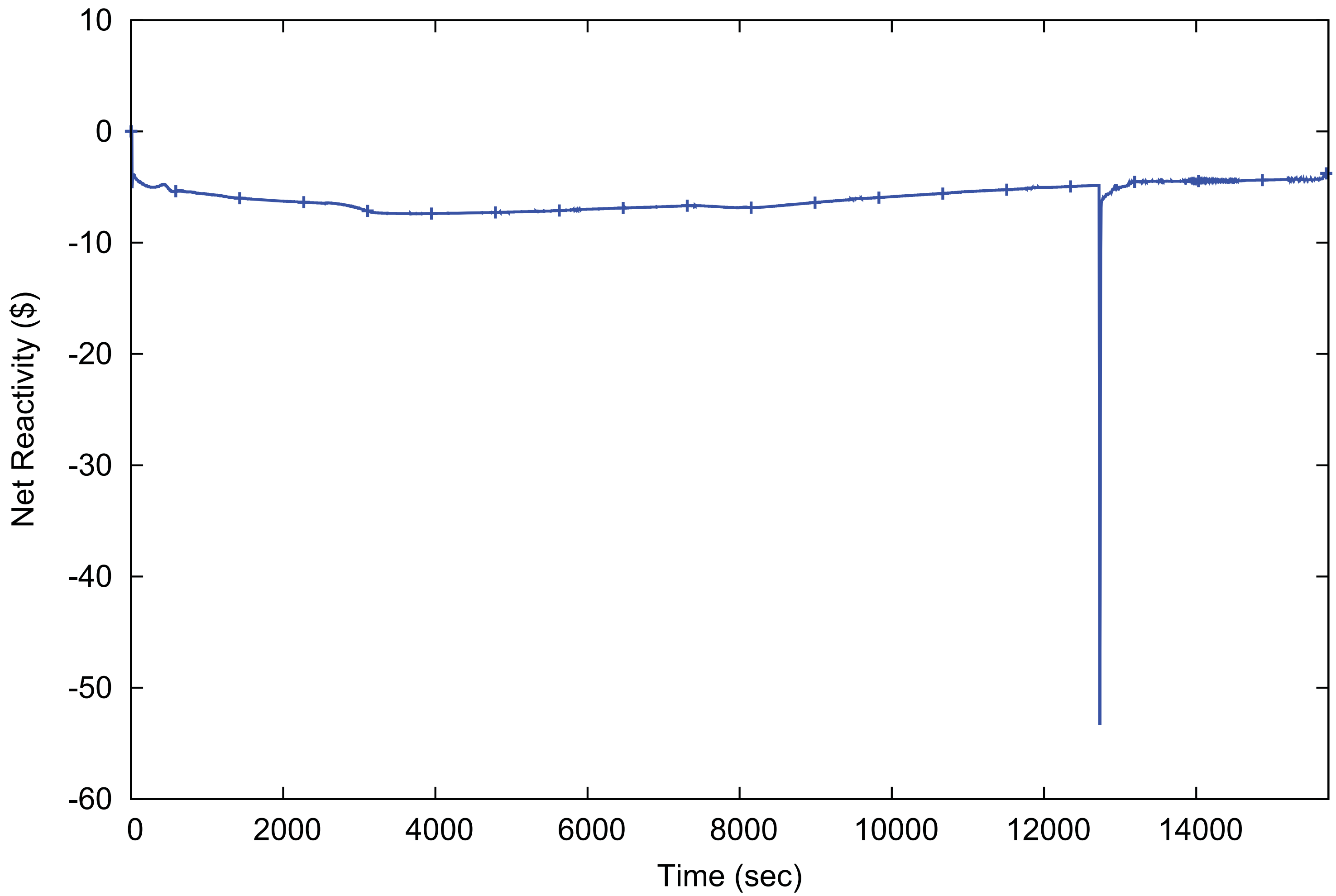


Figure 15.6-45: Loss of Coolant Accident - Collapsed Liquid Level Above Active Core

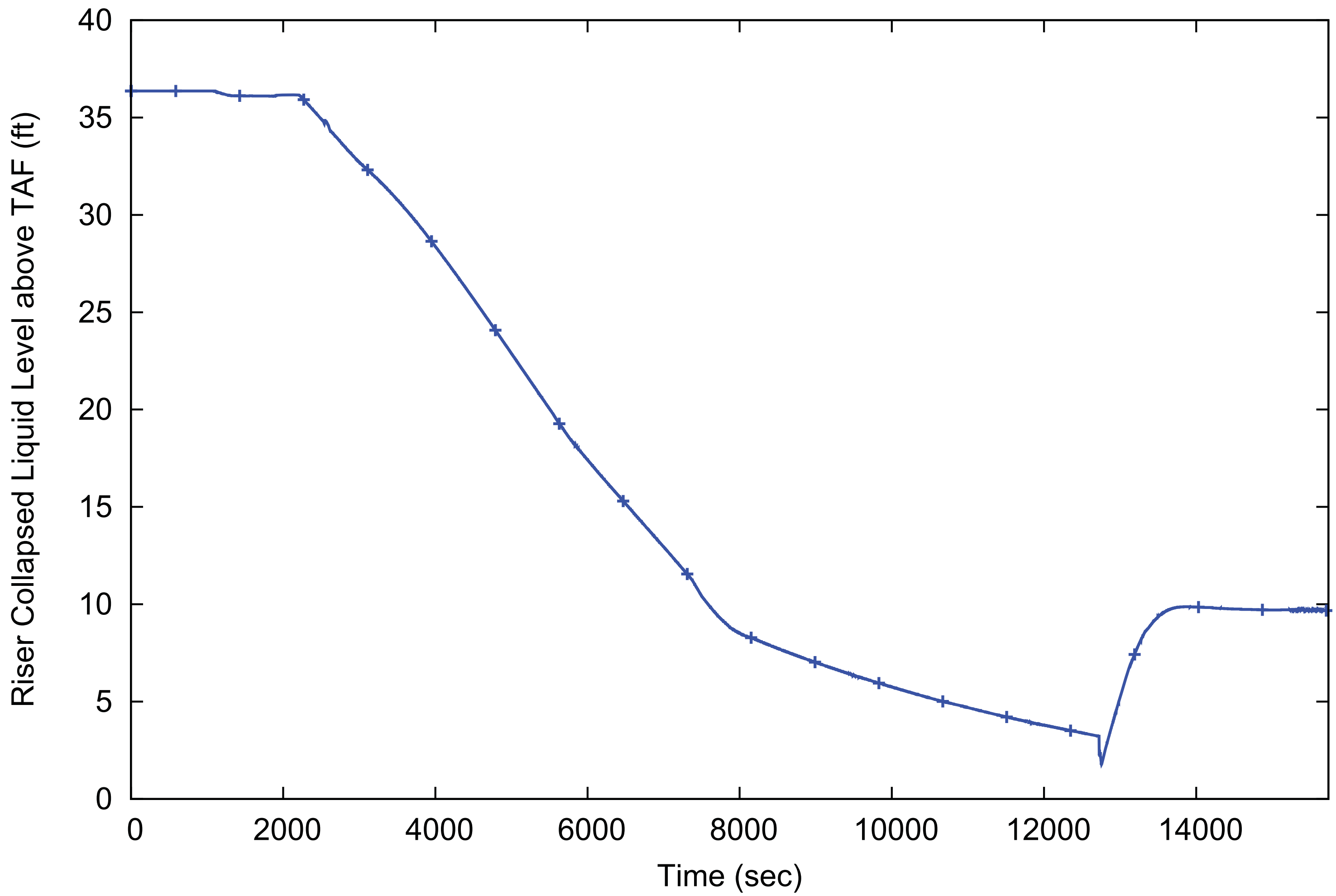


Figure 15.6-46: Loss of Coolant Accident - Reactor Coolant System Average Temperature

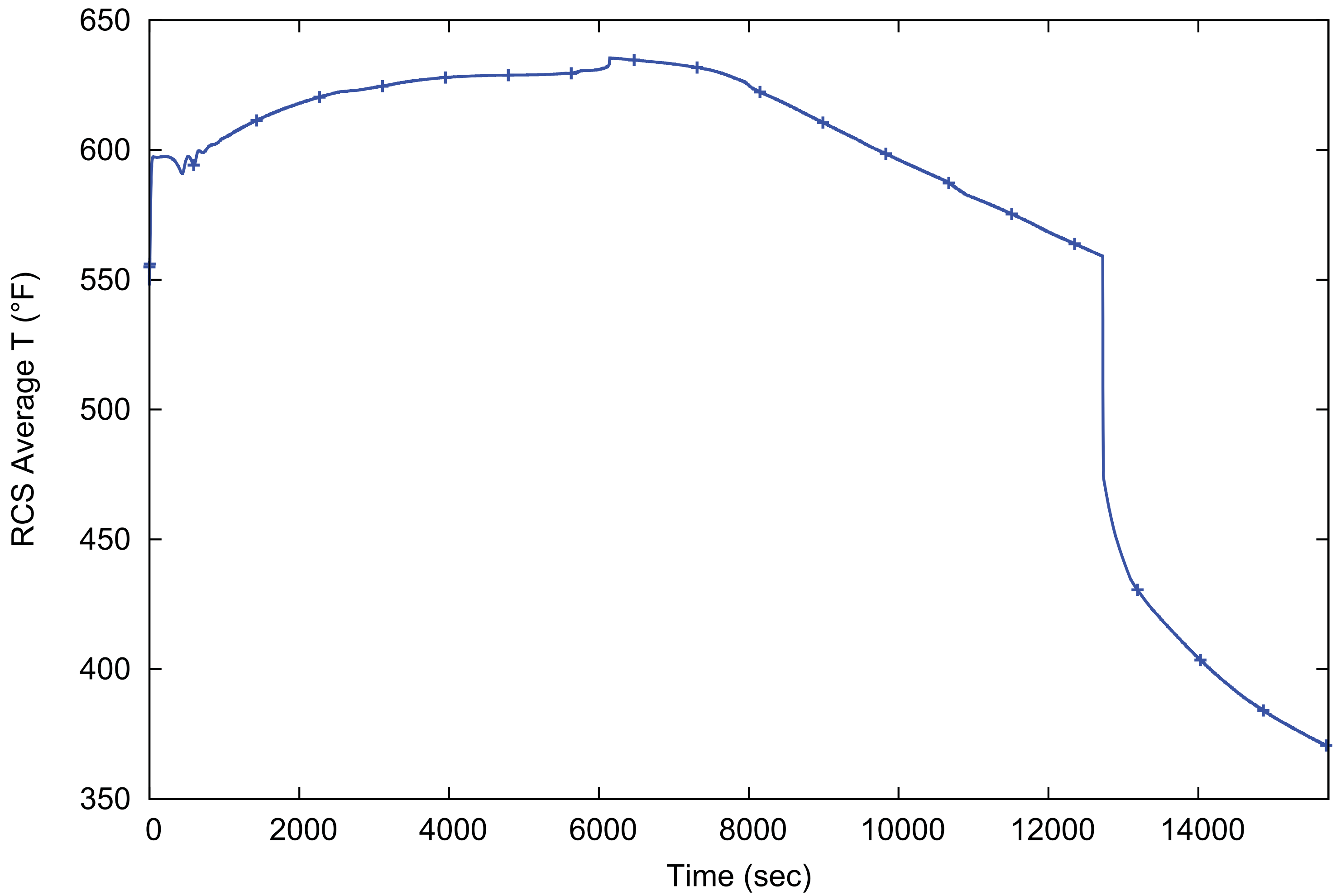


Figure 15.6-47: Loss of Coolant Accident - Reactor Coolant System Temperatures

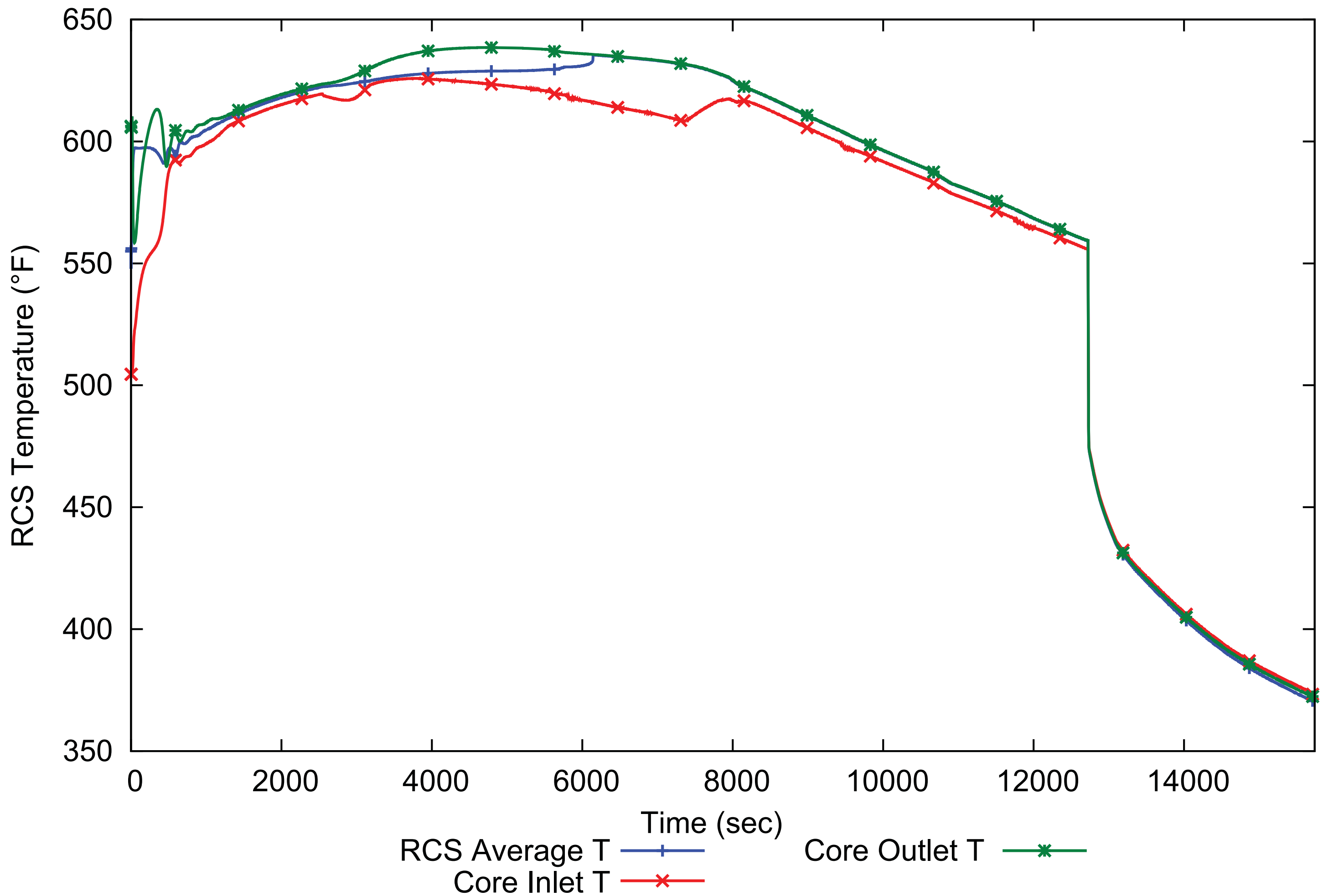


Figure 15.6-48: Loss of Coolant Accident - Reactor Vent Valve Flow

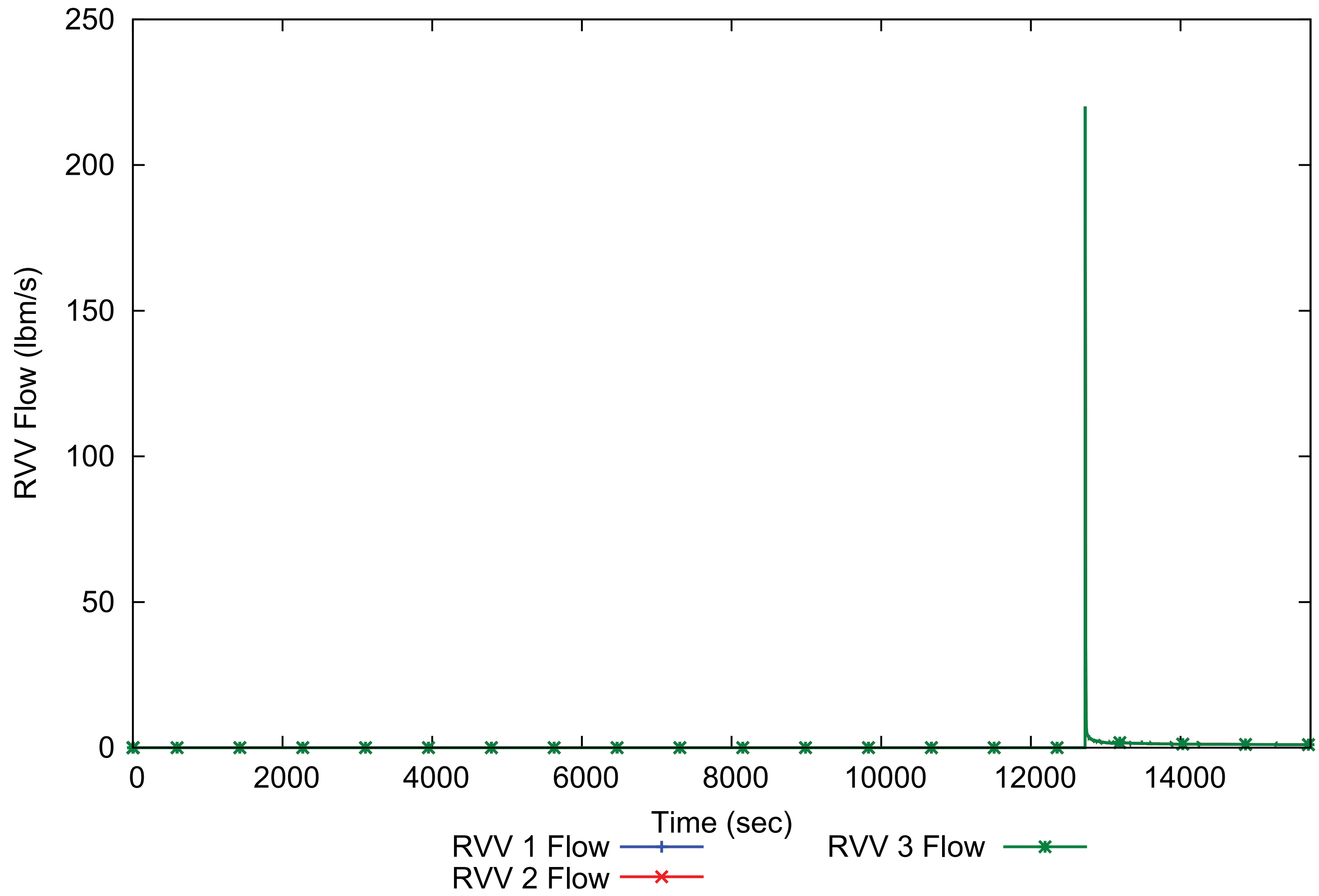


Figure 15.6-49: Loss of Coolant Accident - Reactor Recirculation Valve Flow

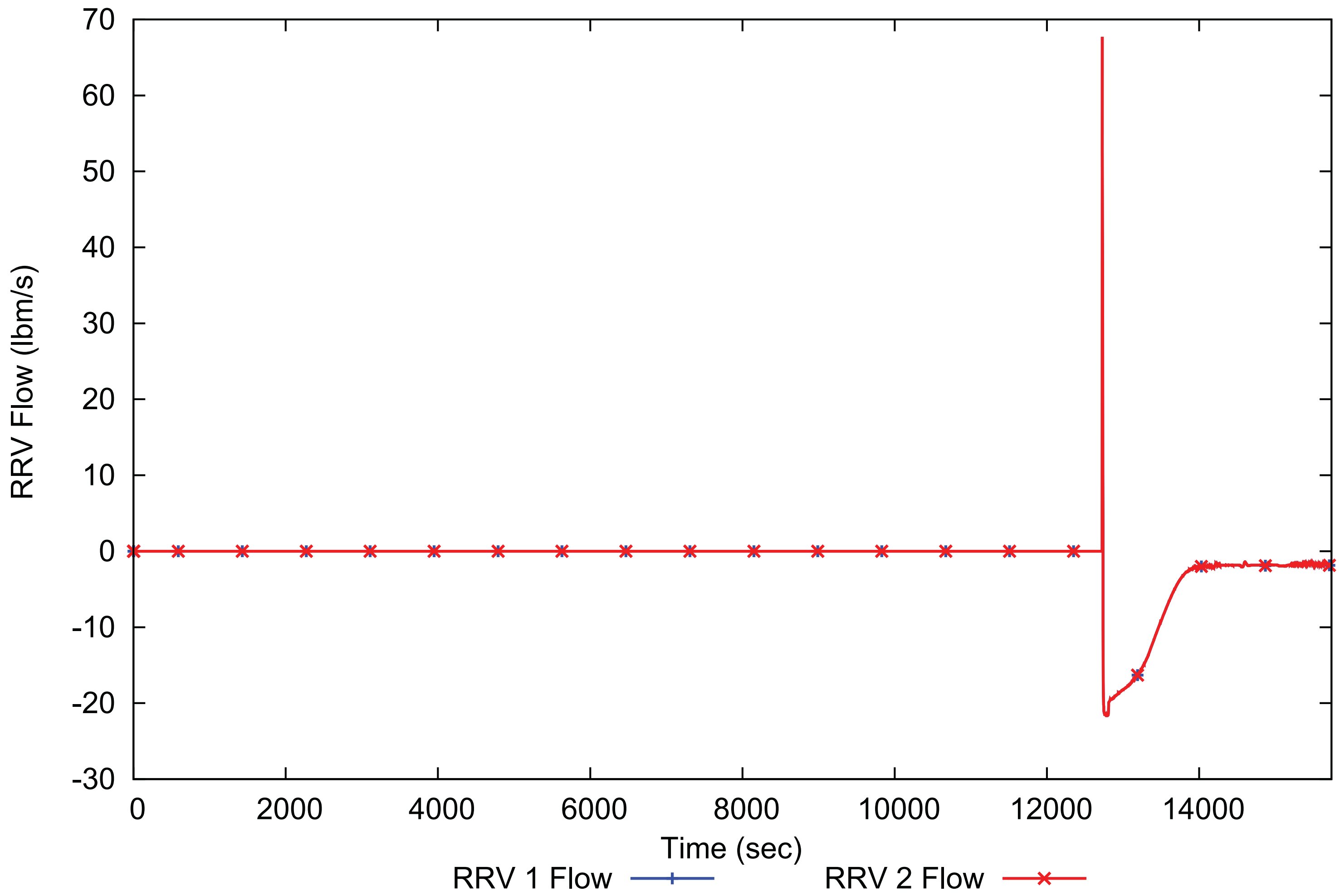


Figure 15.6-50: Loss of Coolant Accident - Reactor Coolant System Flowrate

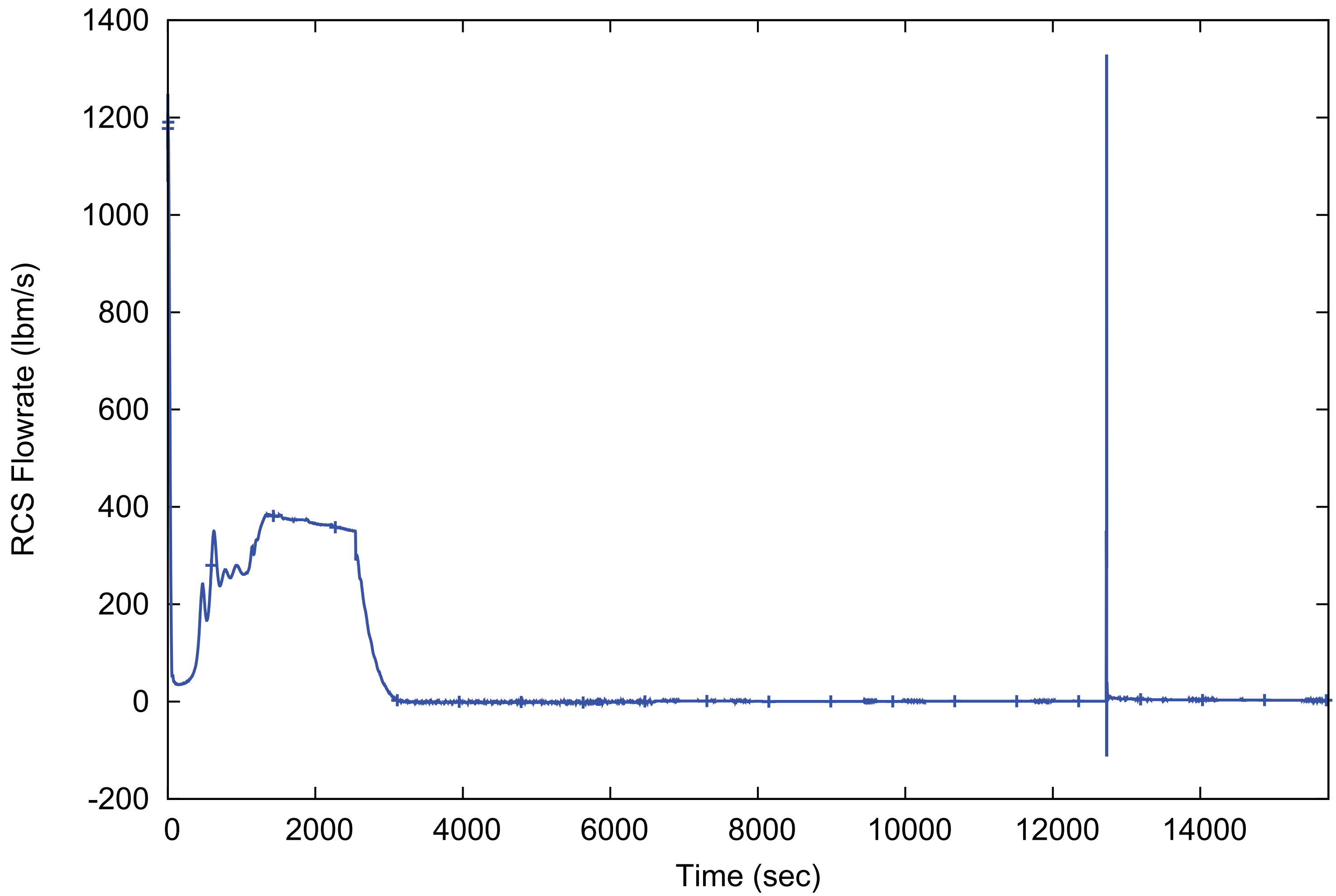


Figure 15.6-51: Loss of Coolant Accident - Steam Generator Pressure

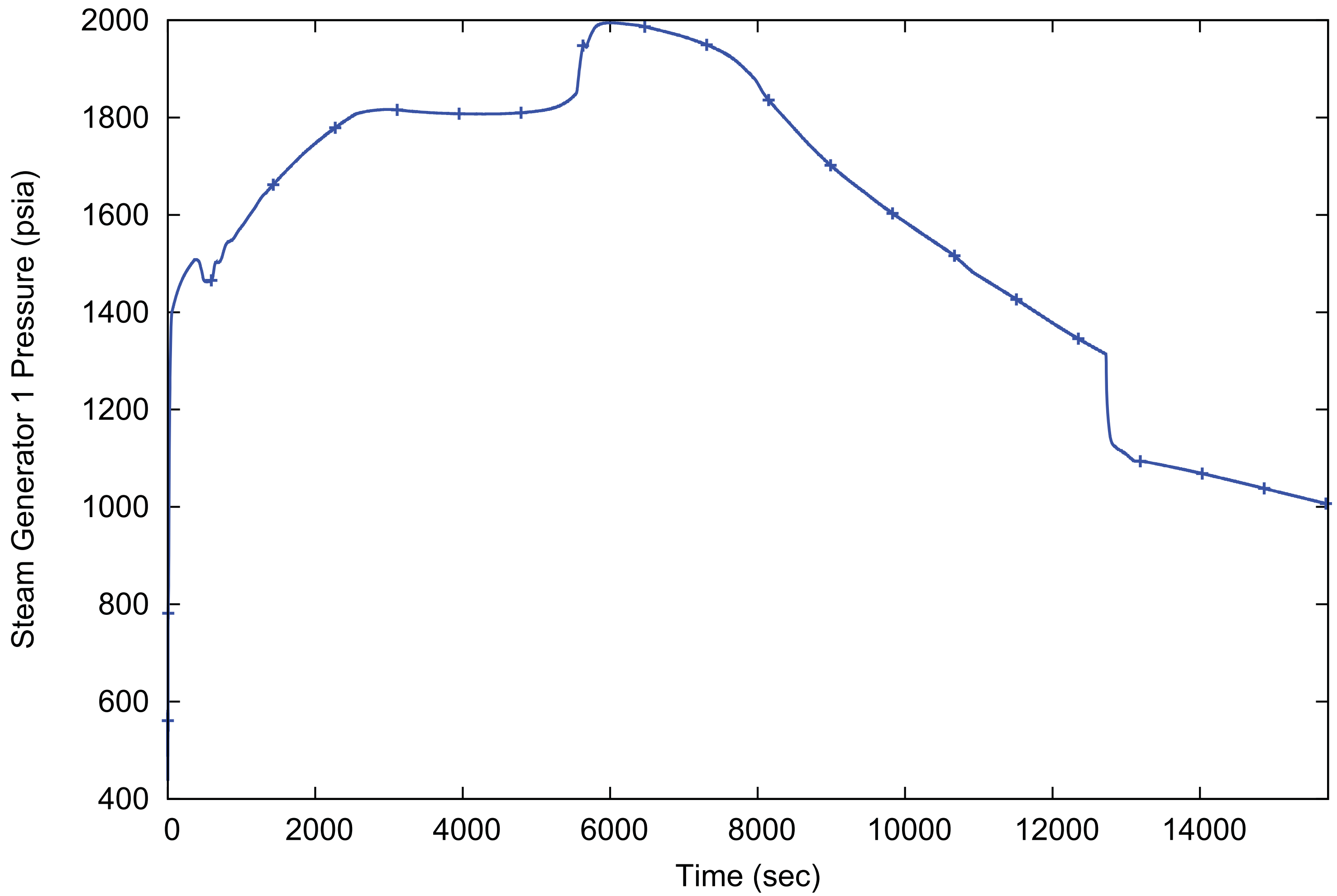


Figure 15.6-52: Loss of Coolant Accident - Containment Wall Temperature

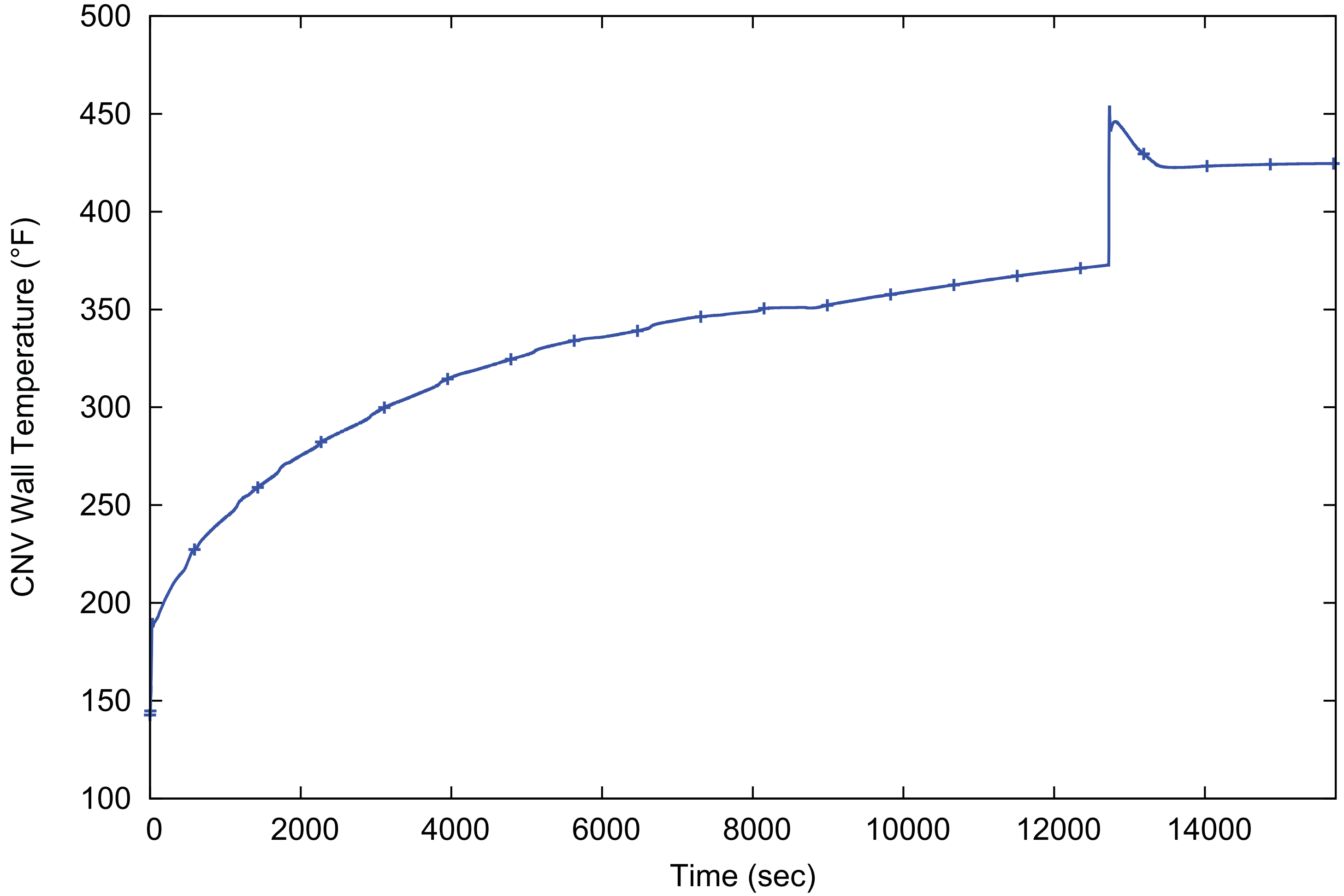


Figure 15.6-53: Loss of Coolant Accident - Fuel Volume Average Temperature

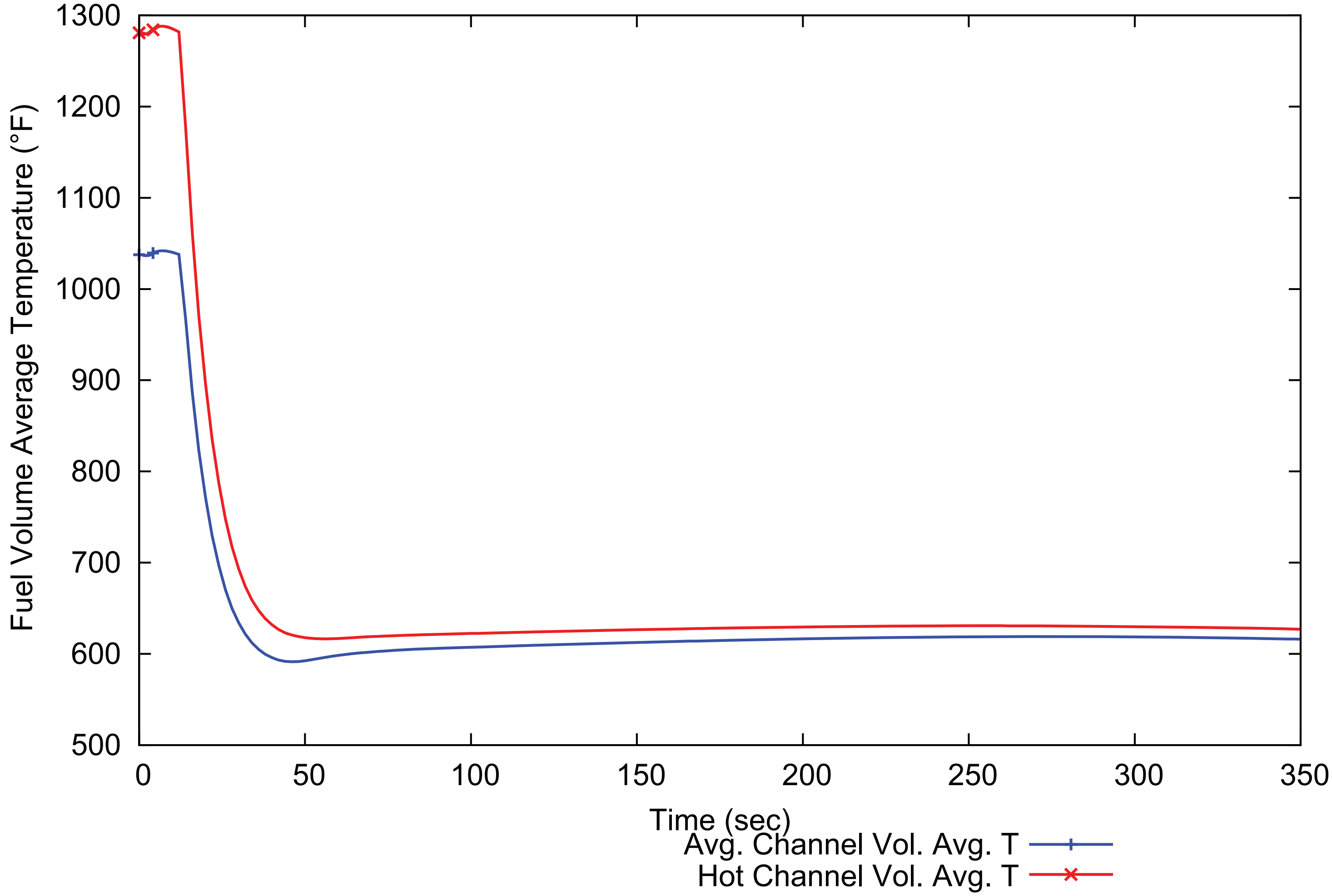


Figure 15.6-54: Loss of Coolant Accident – MCHFR

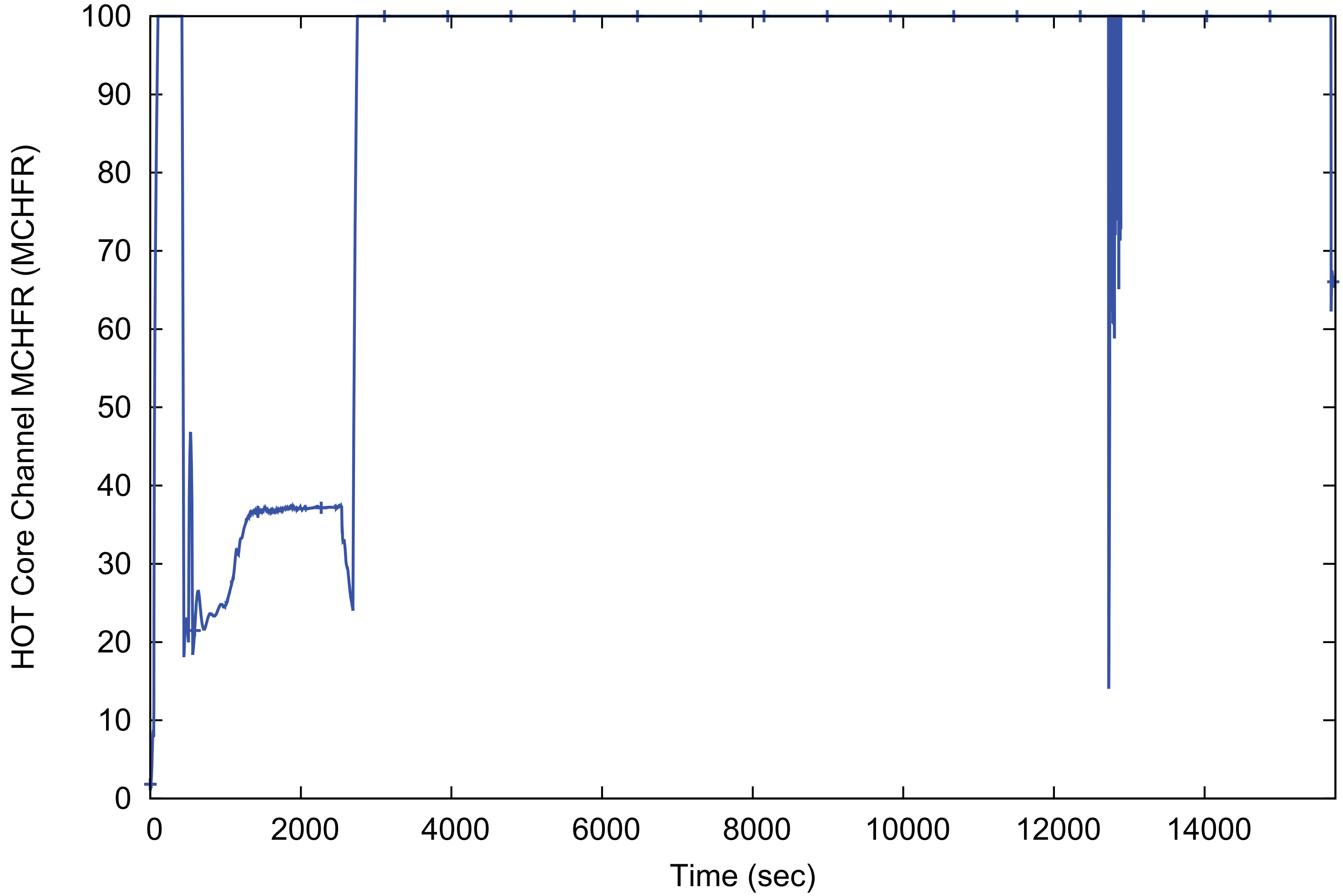


Figure 15.6-55: Inadvertent Operation of an Emergency Core Cooling System Valve – Reactor Power

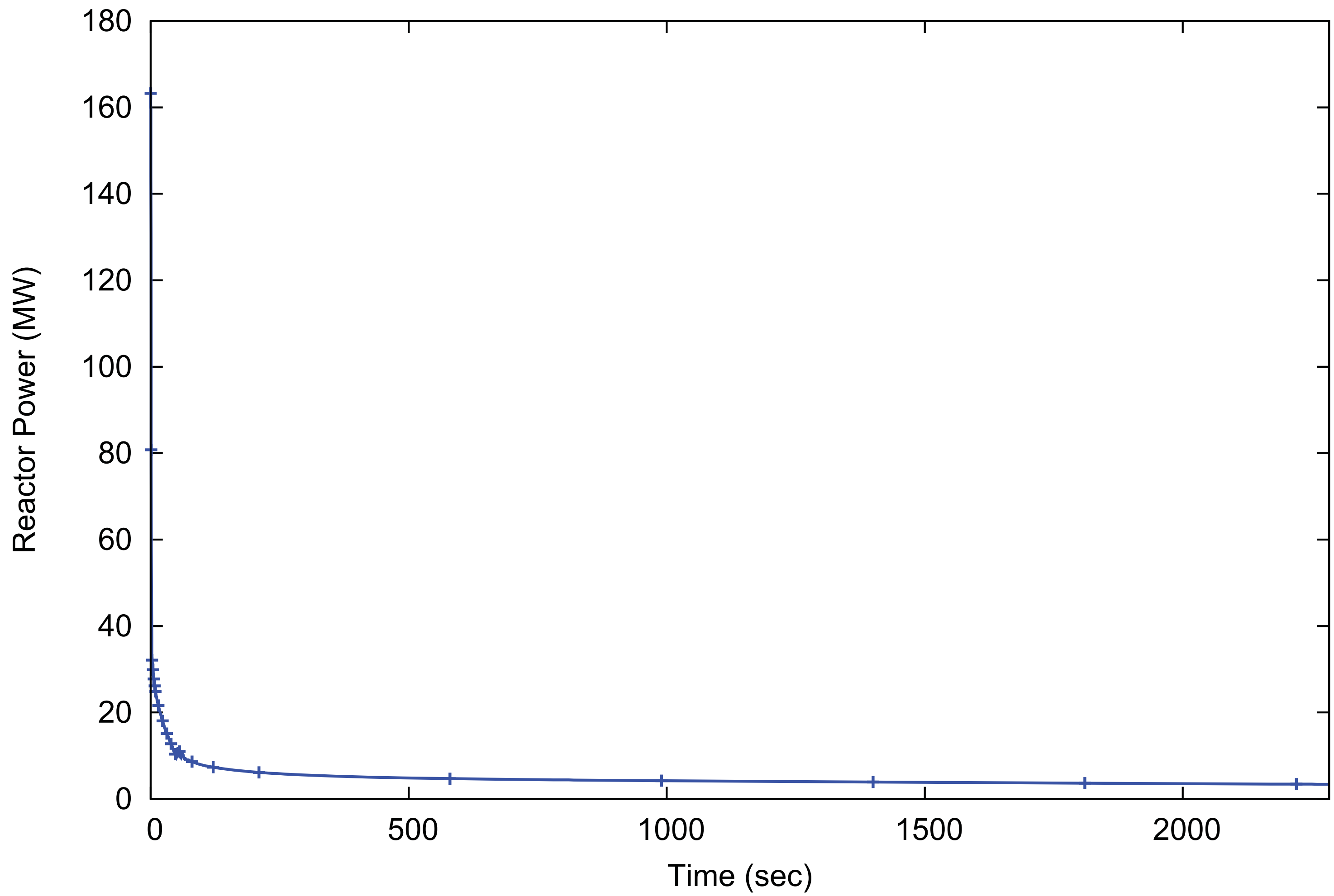


Figure 15.6-56: Inadvertent Operation of an Emergency Core Cooling System Valve – Inadvertent Reactor Recirculation Valve Flow

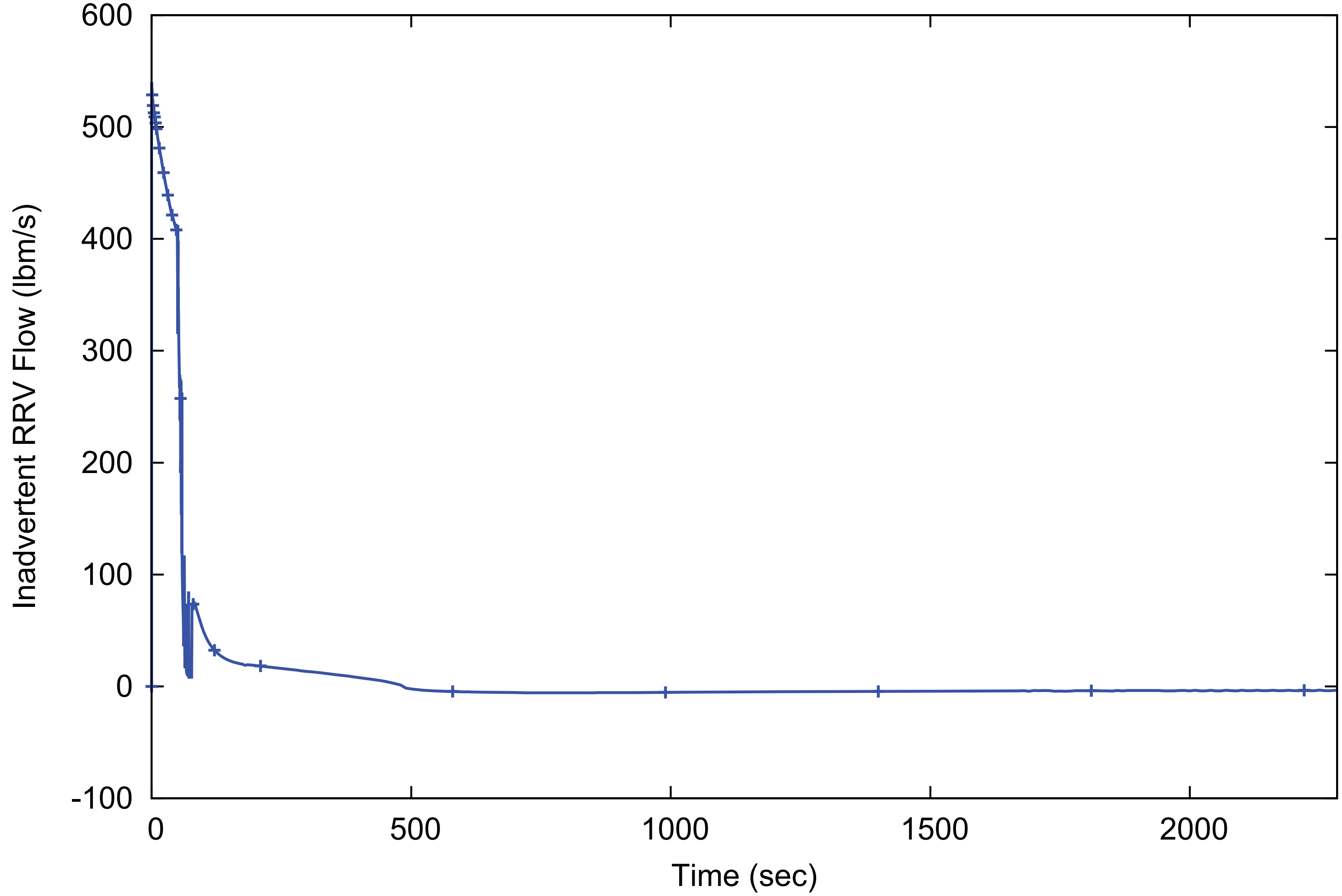


Figure 15.6-57: Inadvertent Operation of an Emergency Core Cooling System Valve – Pressures

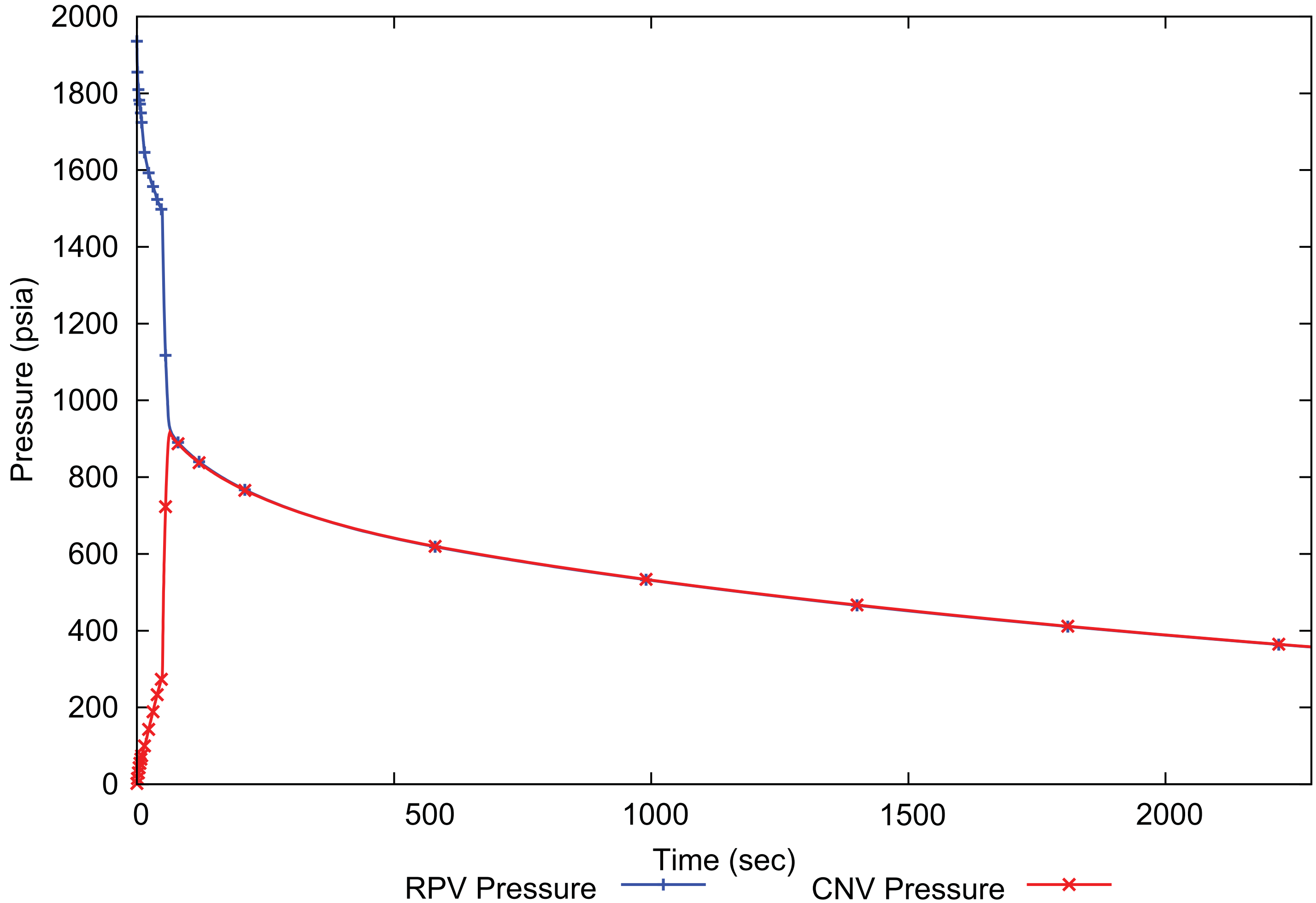


Figure 15.6-58: Inadvertent Operation of an Emergency Core Cooling System Valve – Reactor Coolant System Flow

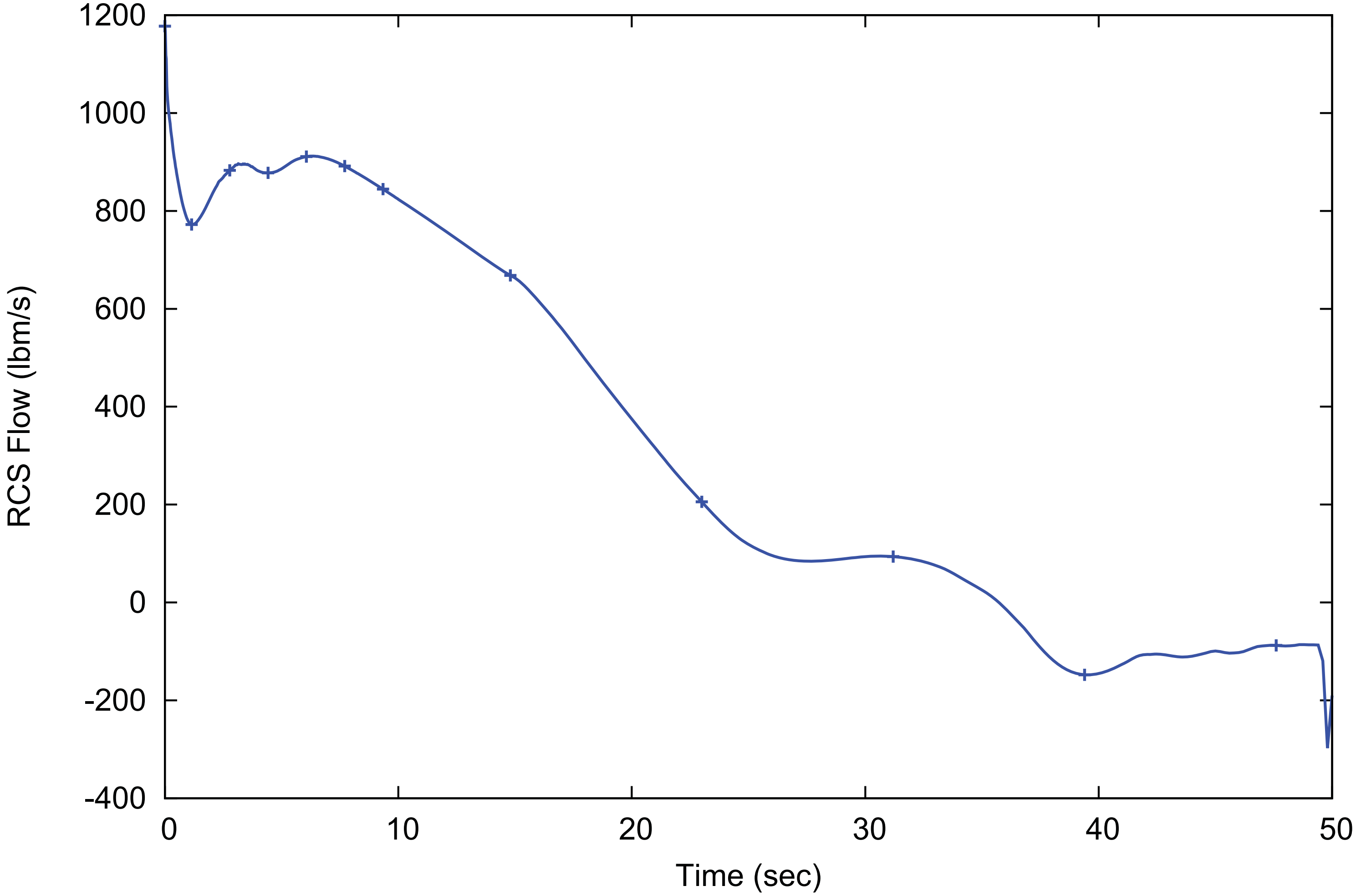


Figure 15.6-59: Inadvertent Operation of an Emergency Core Cooling System Valve – Reactor Coolant System Flow

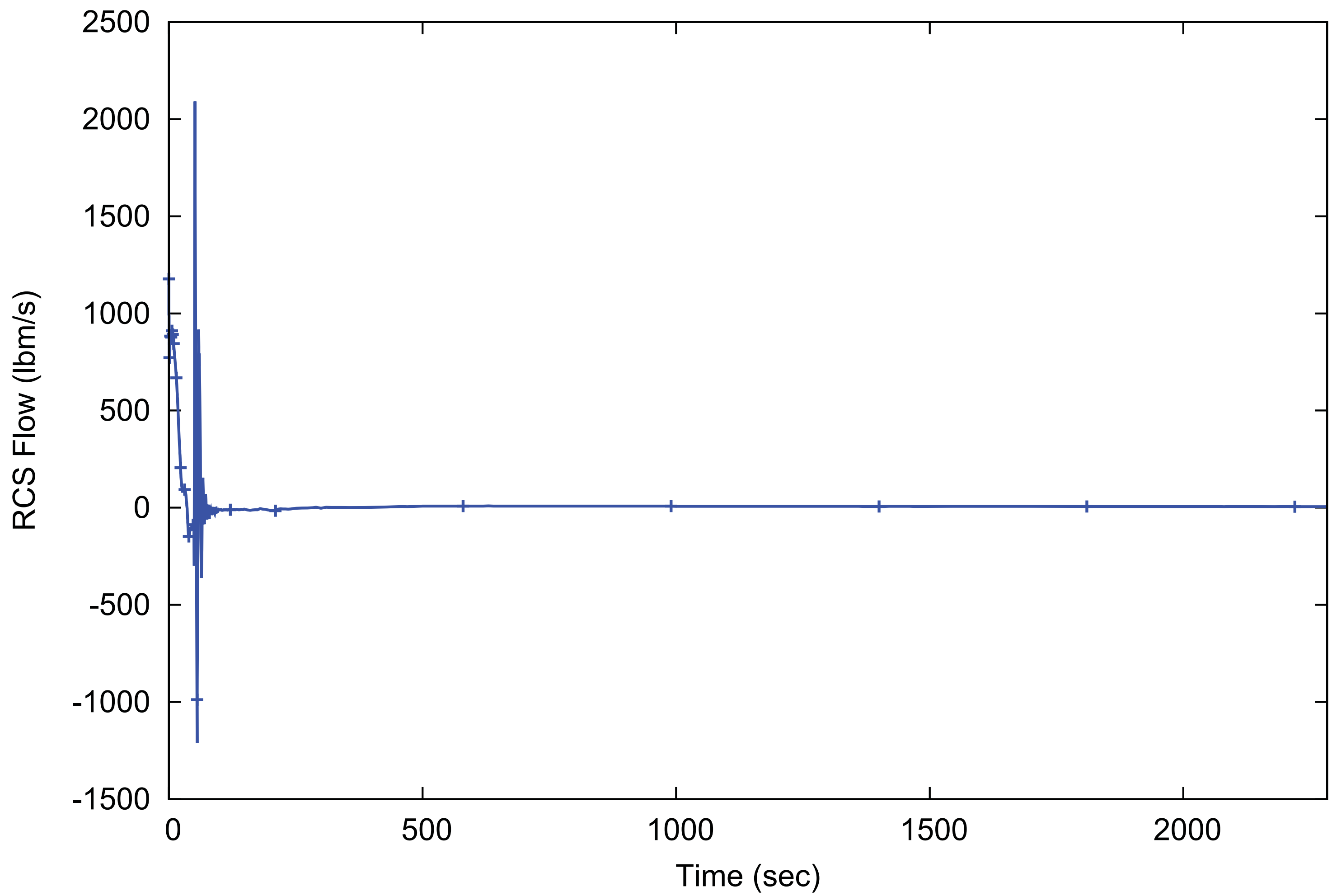


Figure 15.6-60: Inadvertent Operation of an Emergency Core Cooling System Valve – Reactor Power

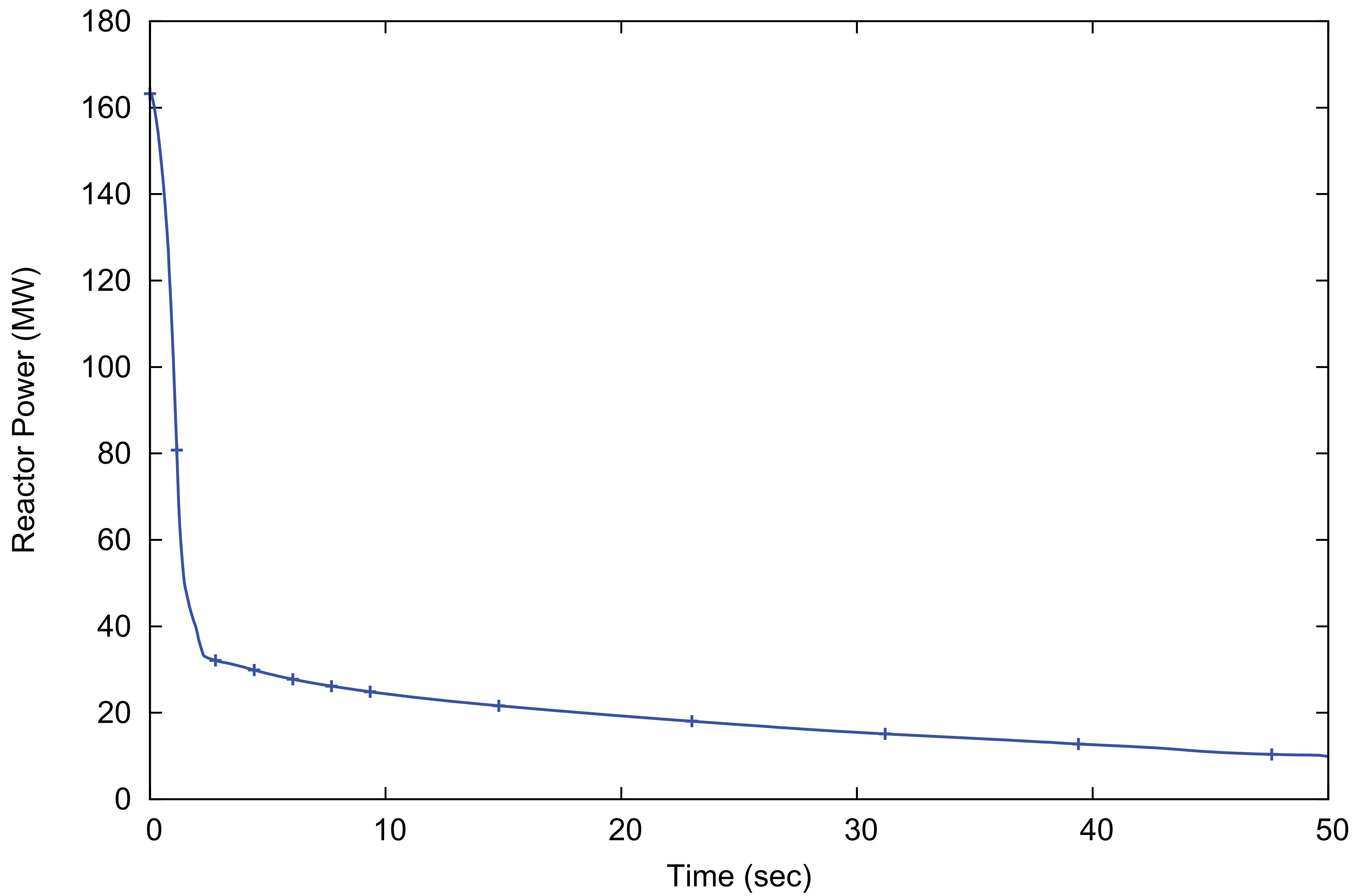


Figure 15.6-61: Inadvertent Operation of an Emergency Core Cooling System Valve – Steam Generator Pressure

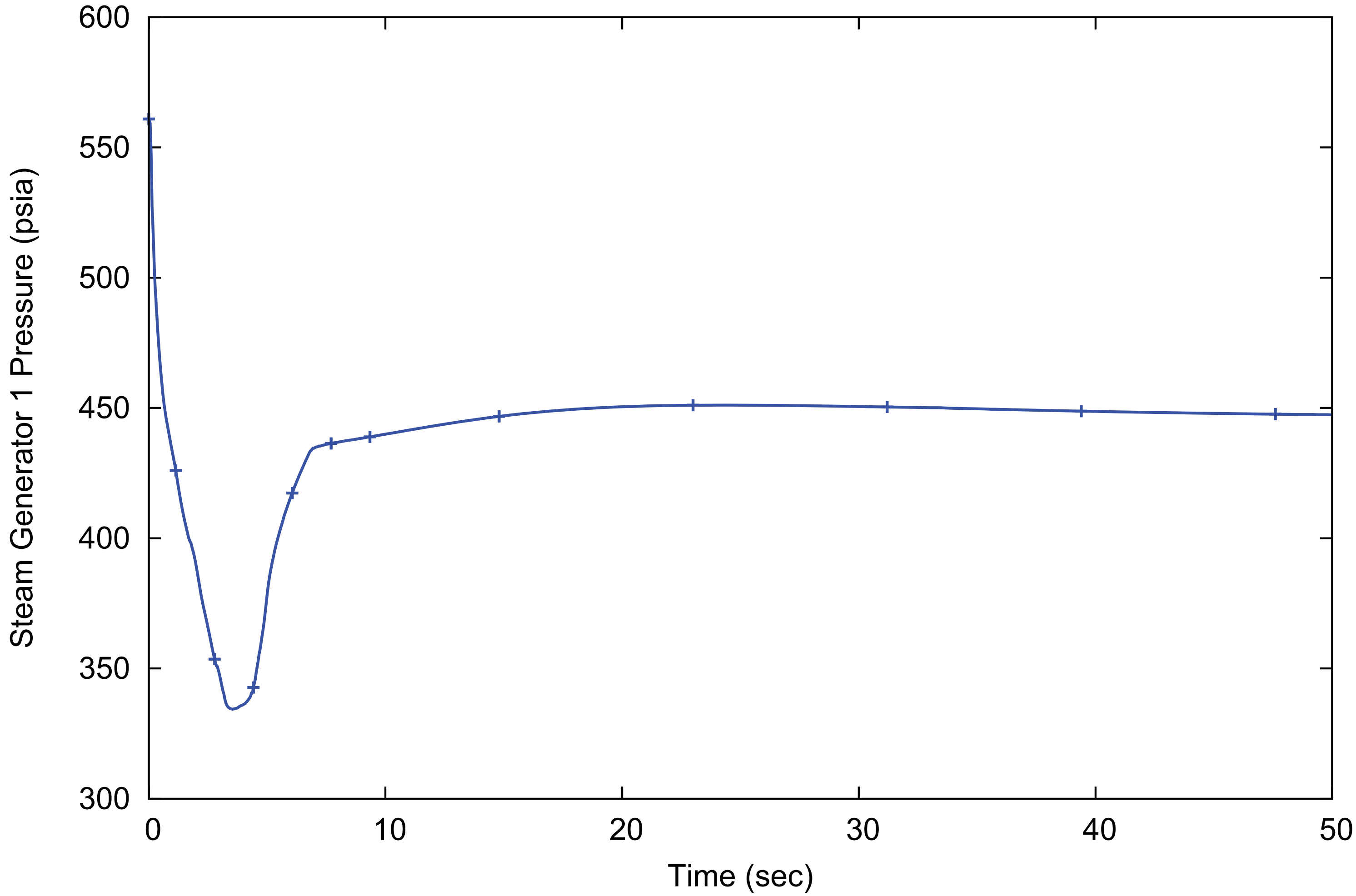


Figure 15.6-62: Inadvertent Operation of an Emergency Core Cooling System Valve – Collapsed Liquid Level Above Active Core

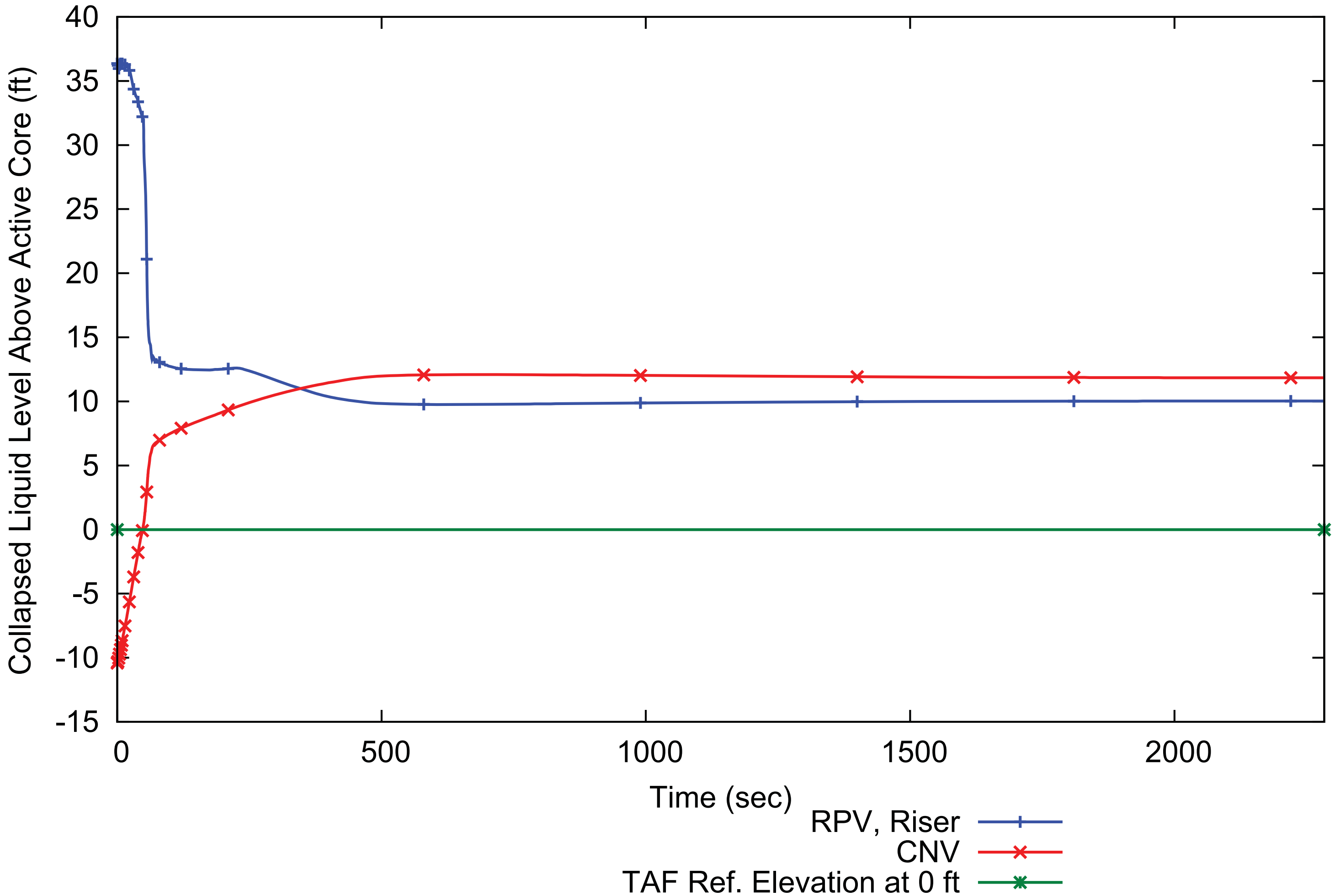


Figure 15.6-63: Inadvertent Operation of an Emergency Core Cooling System Valve – Reactor Coolant System Temperature

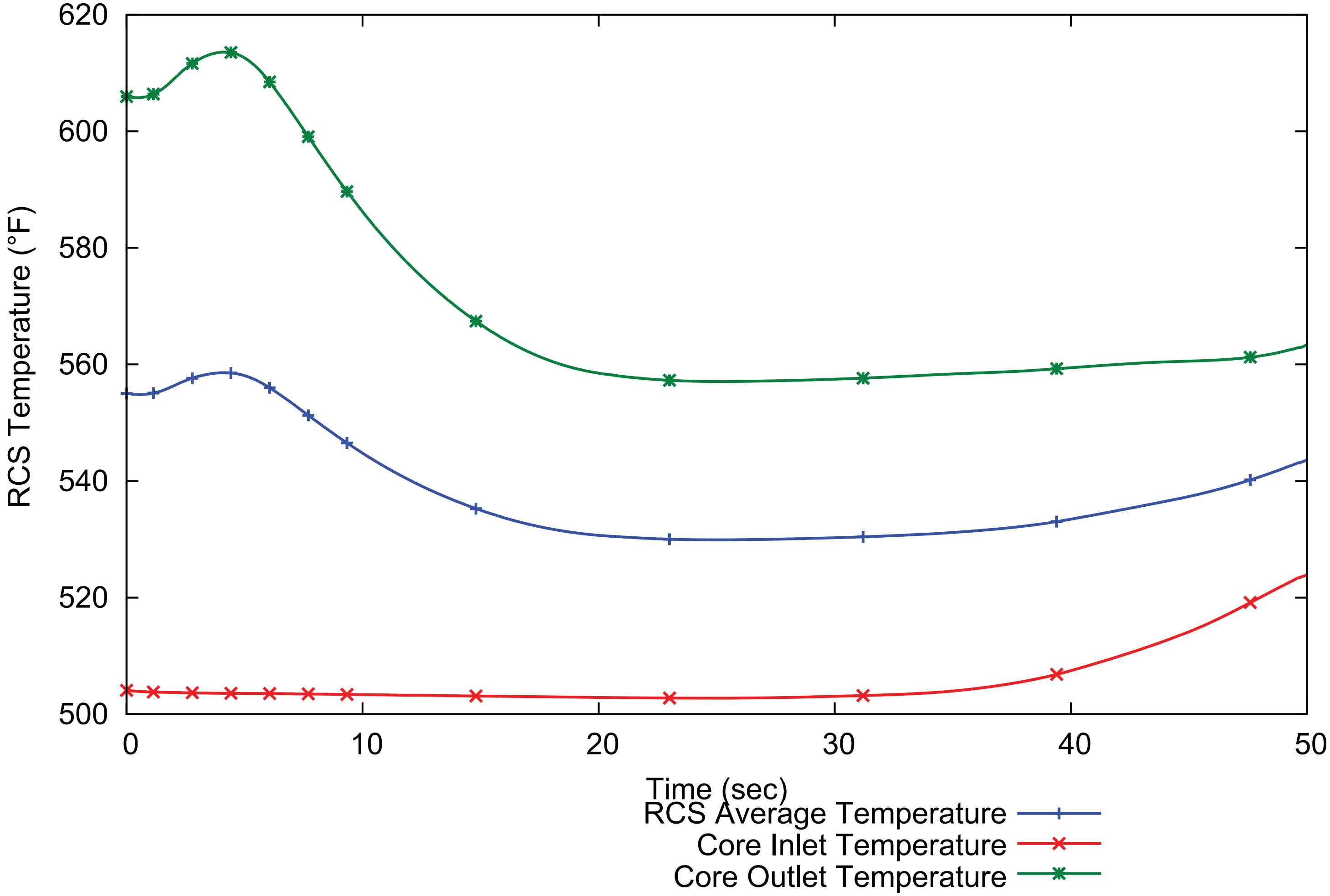


Figure 15.6-64: Inadvertent Operation of an Emergency Core Cooling System Valve – Reactor Coolant System Temperature

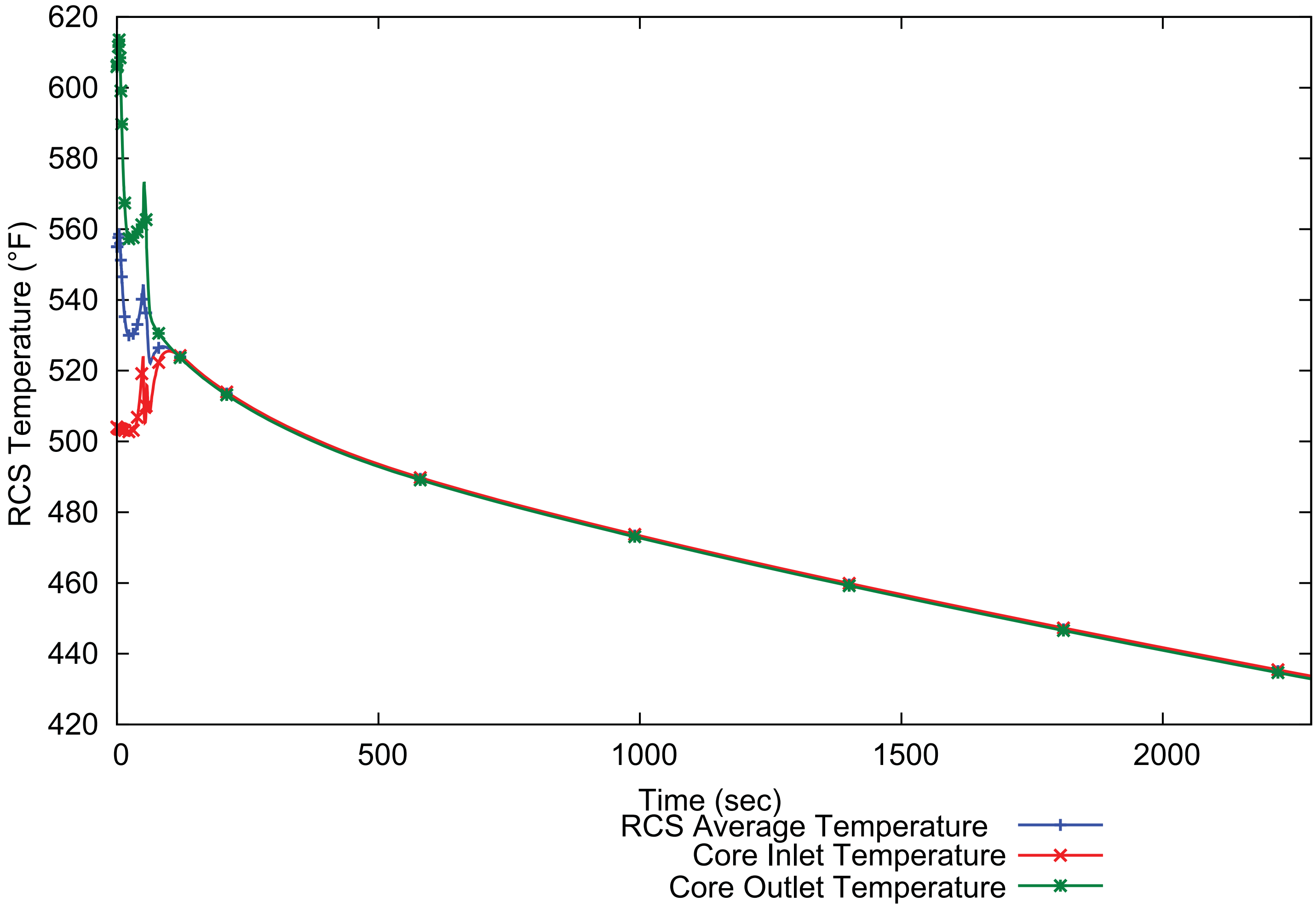


Figure 15.6-65: Inadvertent Operation of an Emergency Core Cooling System Valve – Fuel Volume Average Temperature

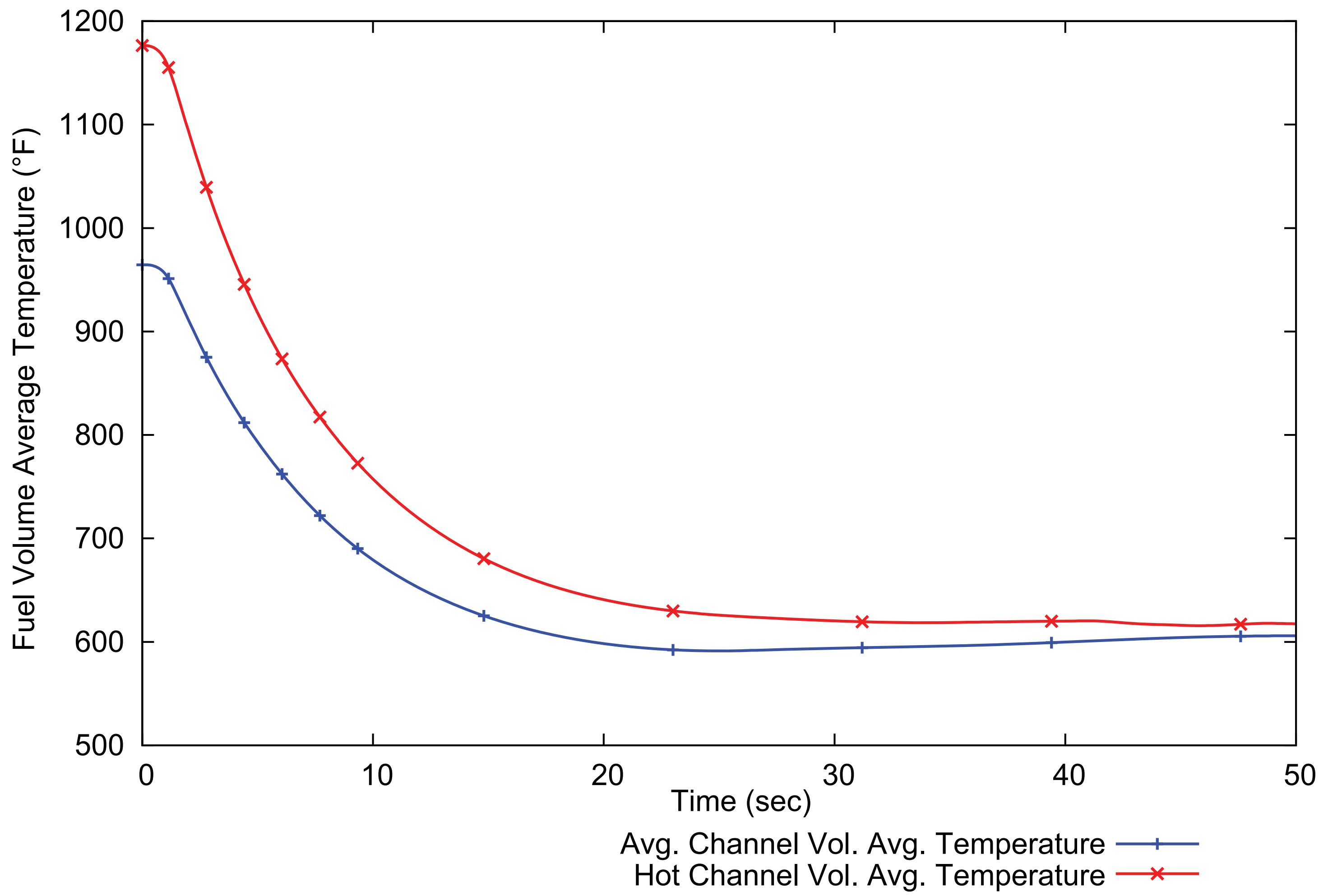


Figure 15.6-66: Inadvertent Operation of an Emergency Core Cooling System Valve – Fuel Cladding Temperature

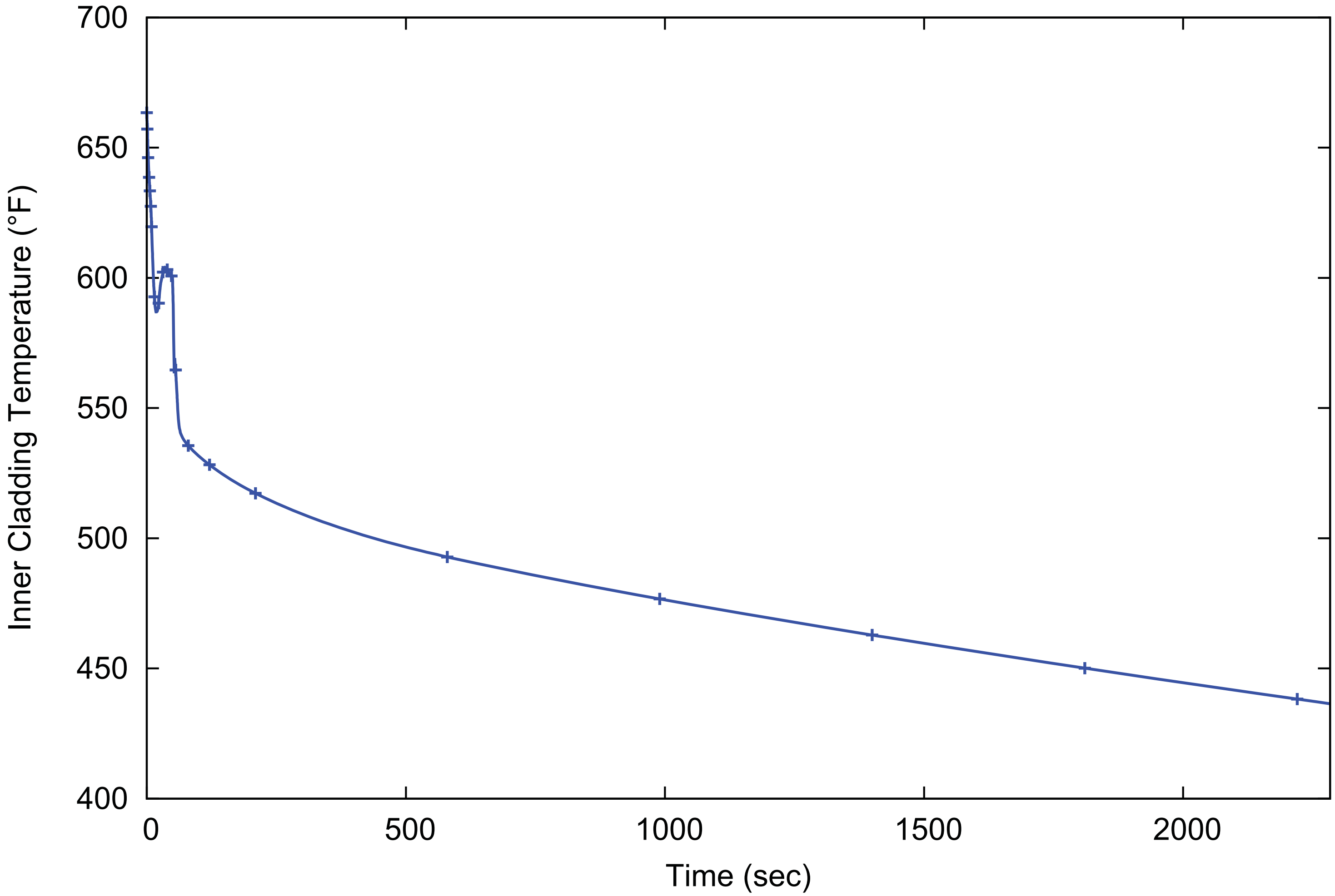


Figure 15.6-67: Inadvertent Operation of an Emergency Core Cooling System Valve – Critical Heat Flux Ratio

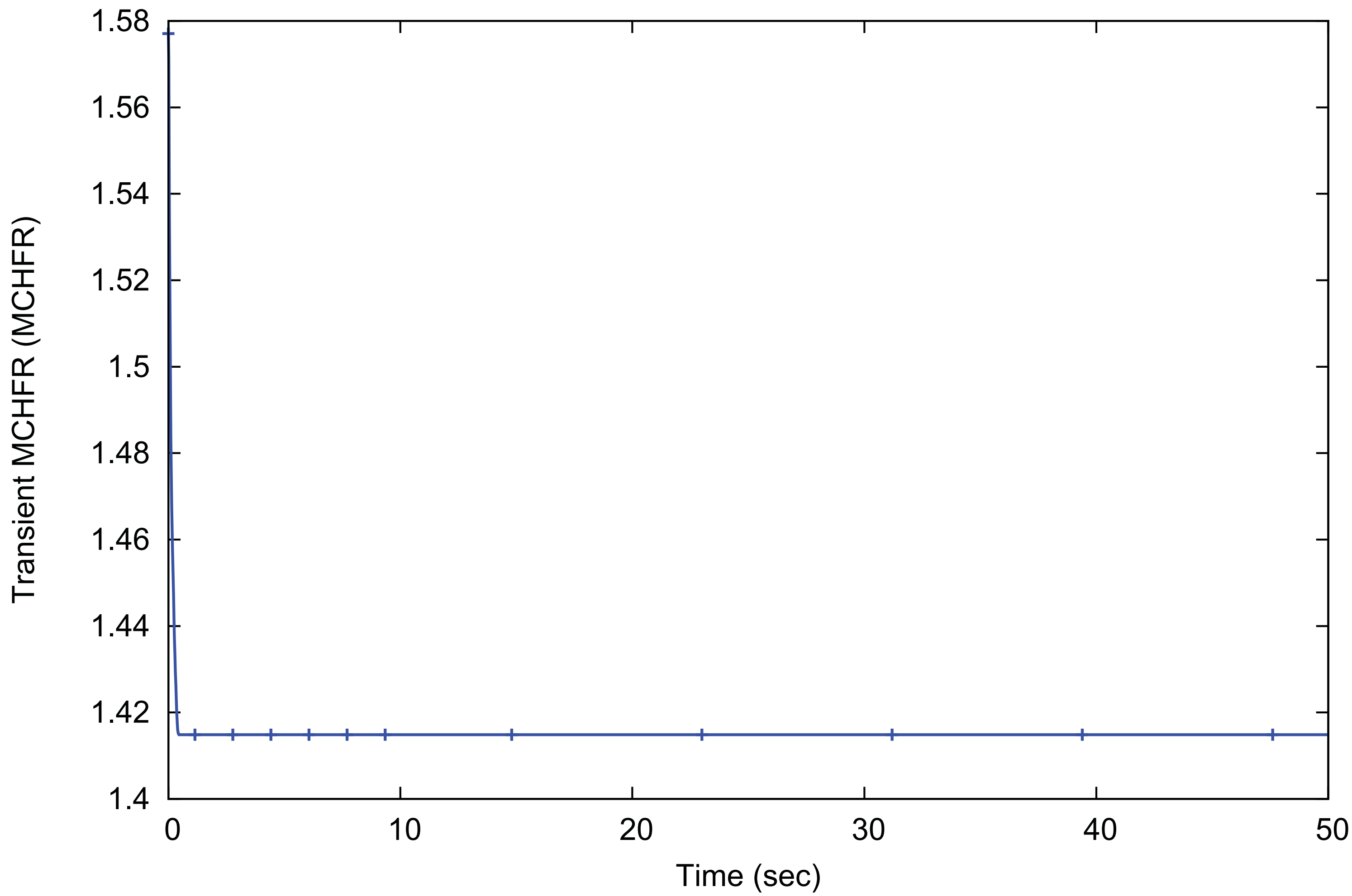
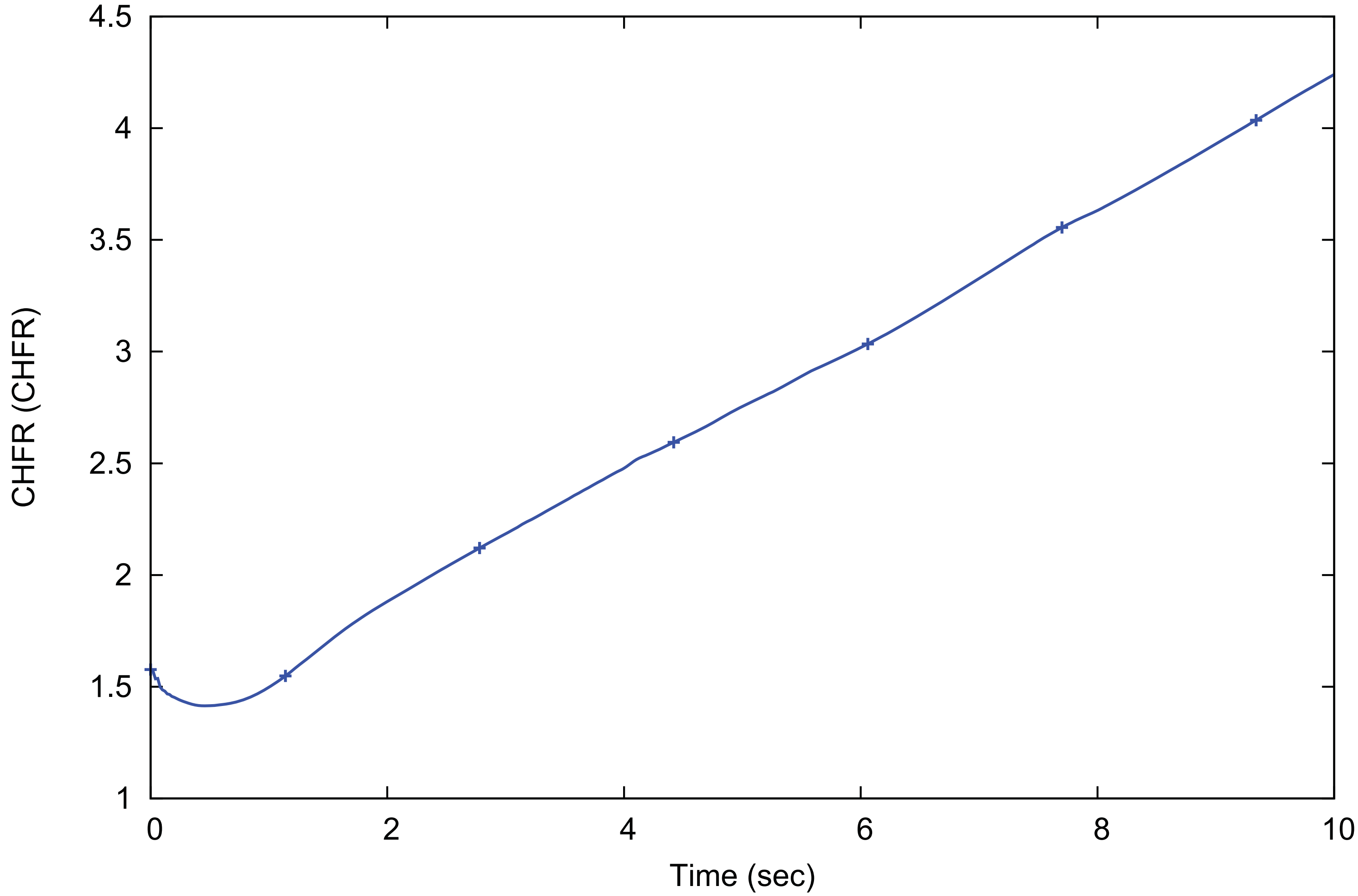


Figure 15.6-68: Inadvertent Operation of an Emergency Core Cooling System Valve – Critical Heat Flux Ratio



15.7 Radioactive Release from a Subsystem or Component

This section addresses events that could result in a radioactive release from a component or system other than the reactor coolant system. The sources of such releases are waste processing systems and fuel handling systems. The NuScale Power Plant design is similar to current generation pressurized water reactors with respect to waste processing and fuel handling systems. The NuScale Power Plant design is unique in that the entire NuScale Power Module (NPM) is moved for refueling.

Releases from waste processing systems are no longer treated as design basis events and are not addressed in Chapter 15. Reference is made to Chapter 11 where the postulated waste processing release events are treated. Fuel handling accidents are addressed in this chapter.

The following events are considered for the NuScale Power Plant design:

- Section 15.7.1 - Gaseous Waste Management System Leak or Failure
- Section 15.7.2 - Liquid Waste Management System Leak or Failure
- Section 15.7.3 - Postulated Radioactive Releases Due to Liquid-Containing Tank Failures
- Section 15.7.4 - Fuel Handling Accidents
- Section 15.7.5 - Spent Fuel Cask Drop Accidents
- Section 15.7.6 - NuScale Power Module Drop Accident

15.7.1 Gaseous Waste Management System Leak or Failure

An evaluation of a gaseous waste management system leak or failure event is provided in Section 11.3.

15.7.2 Liquid Waste Management System Leak or Failure

An evaluation of a liquid waste management system leak or failure event is provided Section 11.2.

15.7.3 Postulated Radioactive Releases Due to Liquid-Containing Tank Failures

An evaluation of a postulated radioactive release due to liquid-containing tank failures event is provided in Section 11.2.

15.7.4 Fuel Handling Accidents

A fuel handling accident may occur during the movement of the fuel. The failure of one entire assembly is assumed to occur when an assembly is dropped in the reactor pool above the spent fuel racks, dropped in the reactor core during refueling, or when an assembly impacts a spent fuel cask during loading. Activity is instantaneously released into the pool water from all irradiated fuel rods in the assembly. Specific isotopes remain in the pool water or enter the reactor building atmosphere instantaneously either fully or partially and are directly released into the environment over a two-hour period.

This event is classified as an accident as in Table 15.0-2. The radiological consequence analysis for the fuel handling accident is provided in Section 15.0.3.

15.7.5 Spent Fuel Cask Drop Accidents

The NuScale Reactor Building crane (RBC) is used to move the spent fuel cask in the Reactor Building refueling area. The RBC system design conforms to the single-failure-proof guidelines of NUREG-0612 so that a credible failure of a single component will not result in the loss of capability to stop and hold a critical load.

The use of this single-failure-proof crane precludes the need to perform load drop evaluations. As a result no design basis accident analysis has been performed to assess the radiological consequences of a spent fuel cask drop accident.

Section 9.1.5 provides additional information regarding the RBC system design and capabilities, including a description of system interlocks, safe load paths, and load exclusion zones.

15.7.6 NuScale Power Module Drop Accident

The RBC is used to move the NPMs in the Reactor Building refueling area. The RBC system design conforms to the single-failure-proof guidelines of NUREG-0612 so that a credible failure of a single component will not result in the loss of capability to stop and hold a critical load.

The use of this single-failure-proof crane precludes the need to perform load drop evaluations. As a result no design basis accident analysis has been performed to assess the radiological consequences of an NPM drop accident.

Section 9.1.5 provides additional information regarding the RBC system design and capabilities, including a description of system interlocks, safe load paths, and load exclusion zones.

Chapter 19 provides a description of the low power and shut down (LPSD) probability risk assessment (PRA). All stages of a nominal refueling outage are included in the LPSD PRA, including movement and disassembly of an NPM during refueling. A NPM drop event is also evaluated in the LPSD PRA. See Section 19.1.6 for additional information.

15.8 Anticipated Transients Without Scram (ATWS)

An ATWS is characterized as a failure of the module protection system (MPS) to initiate a reactor trip in response to an anticipated operational occurrence (AOO). The probability of an AOO, in coincidence with a failure to scram, is much lower than the probability of any other events evaluated in Chapter 15. Therefore, an ATWS event is classified as a beyond design basis event.

10 CFR 50.62(c)(1) states:

> Each pressurized water reactor must have equipment from sensor output to final actuation device, that is diverse from the reactor trip system, to automatically initiate the auxiliary (or emergency) feedwater system and initiate a turbine trip under conditions indicative of an ATWS. This equipment must be designed to perform its function in a reliable manner and be independent (from sensor output to the final actuation device) from the existing reactor trip system.

The underlying purpose of the specific design features required by 10 CFR 50.62 is to reduce the risk associated with ATWS events by providing an alternative means of reactor scram or by mitigating the consequences of such events. Section 15.8 of the standard review plan (SRP 15.8) provides two options for evolutionary plants to reduce the risks associated with ATWS. The first option is to provide a diverse scram system, which would reduce the probability of a failure to scram. SECY-83-293 suggests that the safety goal of the specific design features in 10 CFR 50.62 is to reduce the expected core damage frequency (CDF) associated with ATWS to about 1 E-5/year. Therefore, a diverse scram system or other design feature should reduce the ATWS CDF to a level close to 1E-5/year to reduce the risks of ATWS to an acceptable level to satisfy this option. The second option is to demonstrate that SRP 15.8 ATWS safety criteria are met when evaluating the consequences of an ATWS occurrence.

The ATWS contribution to CDF for the NuScale design is significantly below the safety goal of 1E-5/year, as demonstrated in Section 19.1. This low contribution is based on the reliability of the reactor trip function of the MPS. The MPS, described in Section 7.1 and 7.2, includes a robust reactor protection system with internal diversity, which avoids common cause failures and reduces the probability of a failure to scram. The MPS utilizes the highly integrated protection system (HIPS) platform. Reference 15.8-1 describes integration of fundamental I&C design principles into the HIPS design. The HIPS platform encompasses the principles of independence, redundancy, predictability and repeatability, and diversity and defense-in-depth. The redundancy and diversity of the MPS design ensures that an ATWS occurrence is a very low probability event for the NuScale Power Plant, which meets the intent of the first criterion of SRP 15.8 for evolutionary plants. The NuScale design supports an exemption from the portion of 10 CFR 50.62(c)(1) requiring diverse turbine trip capabilities because the NuScale design does not rely on diverse turbine trip functionality to reduce the risk associated with ATWS events. Additionally, the NuScale design does not include an auxiliary feedwater system; therefore, the portion of 50.62(c)(1) that requires diverse capability to initiate AFWS is not applicable to the NuScale design. An analysis of the beyond design basis ATWS event is described in Section 19.2.

15.8.1 References

15.8-1 NuScale Power LLC, "Design of the Highly Integrated Protection System Platform," TR-1015-18653-P-A, Rev. 2.

15.9 Stability

In current generation plants, events that could result in thermal-hydraulic instability within the reactor vessel are considered significant only for boiling water reactors (BWRs). Individual fuel assemblies with high power-to-flow ratios may undergo instabilities or the neutronic conditions may lead to power oscillation. Current generation pressurized water reactors use pumps for forced circulation, which keeps core flow essentially constant with power level. The NuScale Power Module (NPM) employs natural circulation. With this design feature, flow through the core is not held constant by pumps providing forced circulation. Thus, variations in flow may result in changes in power level and vice versa. The identification and evaluation of the significance of these mechanisms is addressed in Section 4.4.7. The analysis of the NPM to representative perturbations and the behavior to bounding flow instability are evaluated in this section. The evaluation is based on reactivity coefficients that span the full range associated with beginning to end of cycle and demonstrates that the NPM is protected from unstable flow oscillations provided that operation is limited by a defined pressure-temperature exclusion zone such that no boiling in the riser area above the core is allowed. A large negative moderator reactivity coefficient may stabilize the flow even if riser boiling occurs but this is conservatively not credited in the stability methodology.

15.9.1 Consideration of Thermal-Hydraulic Stability

The NuScale Stability Evaluation Methodology Topical Report (Reference 15.9-1) presents a comprehensive analysis of the thermal hydraulic stability of the NPM and demonstrates compliance with 10 CFR 50, Appendix A, General Design Criteria (GDC) 10 and GDC 12. The topical report considers potential power and hydraulic stability mechanisms during anticipated operational occurrences (AOOs) and normal operating conditions. Thermal-hydraulic instability during infrequent events or accidents is not considered because the acceptance criteria for such events allow for conditions beyond the specified acceptable fuel design limits imposed on AOO events experiencing instabilities. The topical report considers flow stability from a fundamental conceptual perspective without making assumptions based on similarities to or differences from other nuclear systems. The topical report describes computational methods developed for the analysis of the limiting instability modes for the NPM design during steady state, normal operation and anticipated transients.

The region exclusion stability protection solution is shown in Reference 15.9-1 to be an acceptable approach for preventing the occurrence of instabilities in the NPM. This solution precludes occurrence of instability by tripping the plant before entering a region where instability may occur. The Module Protection System (MPS) trips the NPM at a minimum of five (5) degrees F before reaching the region where instability is possible. This is shown schematically in Figure 15.9-1, which shows decay ratio versus riser subcooling. The operational domain identified with potential instability is characterized by loss of subcooling in the riser (Figure 5.1-3). The riser is considered to be adiabatic in that no heat is added or removed and the liquid entering the riser from the core experiences a gradual decrease in pressure but no change in temperature. The loss of riser subcooling can lead to boiling which can be destabilizing because enthalpy changes to a boiling fluid have a much more significant effect on density than they do on single phase fluids and thus is destabilizing. This phenomenon is described in more detail in Reference 15.9-1. This condition is already excluded by the module protection system (MPS) for considerations other than flow stability.

The limiting instability mode is a natural circulation instability that shares some attributes with density wave instability but is dominated by the adiabatic riser response rather than wave propagation in the core. This mode is unique to the NuScale design and identified as riser instability. Stability is influenced by the dynamics of the helical coil steam generator (SG) and the fission power response of the core to reactivity feedback.

Important distinctions from the typical density wave instabilities in a BWR are:

- negative moderator reactivity feedback is stabilizing in the NPM.
- increasing core inlet subcooling is not destabilizing.
- the period of flow and power oscillations in the NPM is one to two orders of magnitude higher than the oscillation period in a BWR, hence the preference for a regional exclusion stability solution instead of a detect and suppress solution.

15.9.2 Stability Analyses

Several cases were analyzed over a wide range of power and primary system flow operating conditions and possible scenarios to demonstrate that stability is maintained during routine power operations in the NPM, and the limiting case is presented in Section 15.9.2.1. Adequate stability performance when the plant systems remain within MPS settings during transient scenarios is demonstrated in Section 15.9.3. Finally, scenarios in which MPS setpoints are exceeded and a reactor trip occurs are discussed in Section 15.9.4.

The operating states and events addressed include:

- stability of various steady-state operating power levels (at the corresponding natural circulation flow) is analyzed to demonstrate the operating behavior with regard to the stability of the NPM during power operations. Stability at beginning of cycle (BOC) and end of cycle (EOC) conditions are verified to address moderator reactivity variations.
- stability during transients is analyzed to demonstrate the operating behavior of the NPM during operational events, such as minor changes in feedwater flow, which may occur during normal operations and during AOOs. The NPM is demonstrated to return to stable operations, possibly at a new power level or flow condition, for situations in which the riser subcooling is maintained.

Since primary system flow is driven by natural circulation, the range of flow for which the NPM can operate in steady-state at a given power level is narrow, and is influenced by effects such as pressure losses and the SG pressure and level. The system stability performance representative of fixed points along the power and flow operating line are presented in Section 15.9.2.1. The NPM response to transients is discussed in Section 15.9.3. These sections address the behavior of the NPM as it transitions through power, flow, and other state variables that may not be experienced in steady state operation and shows that these transitory conditions and the endpoint condition are stable. Alternatively, the MPS mitigates any potentially unstable conditions before instability can occur. Such situations are identified in Section 15.9.3.1 through Section 15.9.3.6 and discussed in Section 15.9.4.

15.9.2.1 Stability Analysis for Power Operations

The first demonstration of the stability performance of the NPM is for power operations over a range of power level and flow conditions in the presence of a small perturbation in operating conditions. Primary system flow, core inlet temperature, secondary inlet flow and temperature, and the secondary steam pressure conditions are specified at each power level. Modeling incorporates the effects of ambient heat losses and heat loss through the non-regenerative heat exchanger in the chemical and volume control system (CVCS) to assure consistent thermodynamic modeling of NPM operations. Primary system coolant is withdrawn by the CVCS from the downcomer and returned to the riser at all power levels. The water returned to the riser is colder than water removed from the downcomer as a result of heat removal by the non-regenerative heat exchanger.

Calculations are performed at representative thermal power levels of 160, 120, 80, 40, 32, and 1.6 MW. These conditions are equivalent to 100 percent, 75 percent, 50 percent, 25 percent, 20 percent, and 1 percent of rated power, respectively. The power level of 32 MW is considered in order to address effects related to activation of the turbine and feedwater heater system. Only the limiting case is presented in detail in Section 15.9.2.1.1 through Section 15.9.2.1.4.

After reaching steady state conditions in each calculation, a small perturbation is applied to the steady conditions. In applying the perturbation for determining stability performance, the magnitude of the resulting initial disturbance is not important as long as the disturbance is small enough to not introduce nonlinear effects or cause flow regime or heat transfer transitions. The important variable is the relative change of the perturbation as the disturbance propagates in time. The perturbation is applied to the steady conditions by the following approach:

- Momentary increase in pressure loss residual (pressure in the system after allowance for pressure losses) in the natural circulation primary coolant circuit. The momentary pressure residual perturbation is the main approach because of its reliable effect on initiating a system-wide response and for exciting possible modes in the NPM that may produce oscillations.

After the primary system flow is perturbed, the stability is determined from the core inlet flow as function of time. There are two different considerations in interpreting the transient response.

- The short window immediately after the perturbation highlights the apparent decay ratio of the system to a perturbation. This apparent decay ratio illustrates the rapid response of the system to a perturbation. The system quickly attempts to return to the initial conditions.
- The relatively long-term transient response of the system is to show very small magnitude oscillations relative to the initial response to a sharp perturbation. These oscillations are related to loop dynamics, where the longest period oscillation is characterized by the overall time for fluid to transit the natural circulation loop.

Analyses are performed at each condition for a duration of sufficient time for the short-lived effects to dampen out, leaving a clear indication of the longer-lived effects.

The conclusions from the analyses show that the NPM is highly stable at power levels above 50 percent, with EOC conditions providing more damping (i.e. more stability) than BOC conditions. The NPM is stable between 20 percent and 50 percent power, with some damped, long-term primary system flow oscillations evident. The condition at 20 percent power represents the point at which the turbine is placed on line.

The flow stability condition that is the least stable occurs at 1.6 MW core power (1 percent of rated) with a BOC reactivity condition. The core inlet temperature is slightly above 420 degrees Fahrenheit. The primary coolant flow response shows damped oscillations with a period of several minutes at this power level. The flow is less stable than the higher power cases, but with the low power level (<20 percent), there is no challenge to fuel limits.

15.9.2.1.1 Identification of Causes and Event Description

The 1.6 MW power case at BOC was found to be the least stable of all power operations considered.

15.9.2.1.2 Sequence of Events and Systems Operation

No systems operations occur in response to the event.

15.9.2.1.3 Input Parameters and Initial Conditions

Input parameters and initial conditions for the limiting event are presented in Table 15.9-3.

The event is analyzed for BOC reactivity conditions.

15.9.2.1.4 Results

The analysis that produced the most limiting results is described in Table 15.9-4 and in Figure 15.9-2 and Figure 15.9-3.

15.9.3 Stability Analysis for Operational Occurrences

The nature of the natural circulation system performance narrows the analysis down to examining transients that are credible in the NPM. Several operational events are investigated with externally imposed boundary conditions applied to influence the system response. These boundary conditions include reactivity insertion (either directly in the core or via changes in primary system conditions) and realistic changes in primary and secondary conditions.

The results of these analyses demonstrate an acceptable operating region for the NPM where instability does not occur. Events considered fall into the following general classifications:

- increase in heat removal by the secondary system
- decrease in heat removal by the secondary system
- decrease in reactor coolant system (RCS) flow rate
- increase in reactor coolant inventory
- reactivity and power distribution anomalies
- decrease in reactor coolant inventory

The operational events considered are analogous to licensing basis AOOs. However, typical licensing basis AOO scenarios are chosen to provide a limiting system response and generally result in a reactor trip that mitigates the event. The stability operational events are constructed to initiate a reactor trip, which is not simulated, in order to assess the stability of the NPM. This is a key consideration, because any event that quickly results in an MPS trip does not experience unstable flow oscillations; by not simulating the MPS trip, this effectively bounds any scenario in which MPS trip limits are not reached.

The NPM system response is obtained by the computer code, PIM, which is used in demonstrating system stability at initially steady-state operation. The PIM code is described in Section 4.4.7. An input forcing function is applied to the appropriate boundary condition to initiate the transient, for example, a user-specified feedwater flow changing as a function of time to simulate a decrease in heat removal by the secondary system.

15.9.3.1 Increase in Heat Removal by the Secondary System

15.9.3.1.1 Identification of Causes and Event Description

Stability perturbations can occur from a rapid increase of feedwater flow. The flow increase can be caused by feedwater pump speed increase, valve alignment changes, or other causes. However, the analyzed change is sufficiently small that the MPS does not actuate and control systems, such as those for steam pressure, maintain other parameters at the original value.

Other causes of increased heat removal, such as decreasing feedwater temperature or decreasing steam pressure (that causes increased boiling in the SGs), are generally bounded by changes in feedwater flow. This is because the potential for change in feedwater temperature is more gradual when considering the entire feedwater system train (preheaters, piping lengths, etc.) and large rapid changes in steam pressure are expected to cause either compensating control actions or MPS trips.

15.9.3.1.2 Sequence of Events and Systems Operation

A disturbance results in feedwater flow being rapidly increased by 10 percent in 0.1 seconds. This change is chosen because, while it would normally cause a reactor trip, this trip is not simulated and, thus, it conservatively bounds smaller changes to feedwater flow that would not result in a reactor trip. No systems operations occur in response to the event, so no sequence of events table is generated.

15.9.3.1.3 Input Parameters and Initial Conditions

The event is analyzed for both the reactor at 100 percent power and the reactor at 32 MW to simulate the expected power during startup at which the turbine comes on-line and feedwater heating begins. Input parameters and initial conditions for the 100 percent and 20 percent power cases are presented in Table 15.9-1 and Table 15.9-2 respectively. Both BOC and EOC reactivity conditions were considered in each analysis, but only EOC results are presented as they are the most limiting results.

15.9.3.1.4 Results

The results are presented in Table 15.9-5 and Figure 15.9-4 and Figure 15.9-5 for 100 percent of rated power and EOC reactivity. Additional results are presented in Table 15.9-6 and Figure 15.9-6 and Figure 15.9-7 for 20 percent of rated power and EOC reactivity. These results indicate that the plant is highly stable during a postulated increase in heat removal by the secondary system.

15.9.3.2 Decrease in Heat Removal by the Secondary System

15.9.3.2.1 Identification of Causes and Event Description

Stability following reduction of feedwater flow is addressed in this section. A hypothetical rapid decrease in feedwater flow occurs because of feedwater pump speed change, valve alignment changes, or other causes. However, complete loss of feedwater is not considered because it would result in actuation of the MPS and a trip.

Other causes of decreased heat removal, such as increasing feedwater temperature or increasing steam pressure are generally bounded by changes in feedwater flow because larger changes would result in a trip on high reactor power.

15.9.3.2.2 Sequence of Events and Systems Operation

Feedwater flow is decreased rapidly by 10 percent in 0.1 seconds while maintaining feedwater temperature and steam pressure. This magnitude of change is chosen to determine the acceptability of a partial loss of feedwater. While this magnitude of change would normally cause a reactor trip, this trip is not simulated and, thus, it conservatively bounds smaller changes to feedwater flow that would not result in a reactor trip.

The resulting reduction in the heat removal from the primary coolant flow initiates a transient in which primary coolant temperature starts to rise and negative moderator feedback reduces the fission power. The combined reduction of the heat sink and core power restore the primary coolant temperature to a value above its initial value. The Doppler reactivity compensates for the difference and the net average reactivity is restored to zero. The density head driving the primary coolant flow is also reduced and the flow changes from its initial value to about 90 percent of its initial value.

No systems operations occur in response to the event.

15.9.3.2.3 Input Parameters and Initial Conditions

The event is analyzed for both the reactor at 100 percent power and the reactor at 32 MW to simulate the expected power during startup at which the turbine comes on-line and feedwater heating begins. Input parameters and initial conditions for the 100 percent and 20 percent power cases are presented in Table 15.9-1 and Table 15.9-2 respectively. Both BOC and EOC reactivity conditions were considered in each analysis, but only EOC results are presented as they are the most limiting results.

15.9.3.2.4 Results

The results are presented in Table 15.9-7 and Figure 15.9-8 and Figure 15.9-9 for 100 percent of rated power and EOC reactivity. Results are also presented in Table 15.9-8 and Figure 15.9-10 and Figure 15.9-11 for the case at 20 percent of rated power with EOC reactivity. These results indicate that the plant is highly stable during a postulated decrease in heat removal by the secondary system.

15.9.3.3 Decrease in Reactor Coolant System Flow Rate

The effect of a decrease in primary system flow rate (in isolation from other effects) is not considered a credible event for stability analysis. This is because there is no source for changing the primary system flow without other influences, and because there are no primary system pumps in the NPM to directly influence primary system flow.

15.9.3.4 Increase in Reactor Coolant Inventory

The effects of increasing RCS inventory are not important in the stability assessment because subcooled margin in the riser increases with increasing primary system pressure and overall stability behavior is not sensitive to pressure changes for a single-phase system.

The effect of adding cold water via the CVCS during an increasing RCS inventory event is generally bounded by analyses of increased heat removal by the secondary system. The potential for minor reduction in primary system flow can occur from adding cooler water to the riser. However, the relatively small cooldown that may occur at high power conditions and the very long time for coolant to transit from the CVCS return line located in the riser, around the primary system, and into the core in low power operations makes this a secondary consideration in comparison to secondary side cooldowns.

15.9.3.5 Reactivity and Power Distribution Anomalies

15.9.3.5.1 Identification of Causes and Event Description

The effect on NPM stability from a reactivity anomaly can be caused by changes in boron concentration, by an uncontrolled control rod assembly withdrawal or similar events that result in reactivity insertion.

15.9.3.5.2 Sequence of Events and Systems Operation

Reactor power is 32 MW when enough reactivity is added to the core to initiate a high flux rate trip while other reactivity components perform normally. The choice of 32 MW allows margin to the reactor trip setpoint; the high flux rate trip is not simulated to conservatively bound smaller reactivity insertions that would not initiate this trip.

15.9.3.5.3 Input Parameters and Initial Conditions

Input parameters and initial conditions for the limiting event are presented in Table 15.9-2.

The event is analyzed with the reactor at 32 MW. Both BOC and EOC reactivity conditions are considered, but only EOC conditions are presented as they were the most limiting. At EOC, 0.65 dollars of reactivity is added.

15.9.3.5.4 Results

The effect of a change in boron concentration is slow to develop and is bounded by the applied variations in reactivity conditions.

Reactivity increases that do not result in reactor trip on high flux or high flux rate develop slowly and are bounded by effects of increasing heat removal from the secondary side. These events cause pressurizer insurges that maintain or increase subcooling in the riser.

The results are presented in Table 15.9-9 and Figure 15.9-12 and Figure 15.9-13 for 20 percent of rated power and EOC reactivity. These results indicate that the plant is highly stable during a postulated addition of reactivity event.

15.9.3.6 Decrease in Reactor Coolant Inventory

Decreasing RCS inventory without changes in primary pressure is not important in the stability assessment. Riser subcooling will be maintained and the protection system will trip the NPM on low pressurizer level before any appreciable effect can be seen regarding stability.

Decreasing reactor coolant inventory that results in decreasing pressure but without a level trip is expected to produce no significant effect on stability as long as the primary coolant in the riser remains subcooled. However, further depressurization beyond the trip setpoint that results in riser voiding can destabilize the system. This section provides analysis results using the PIM code that show the effects of depressurization and ability of the MPS to mitigate the event.

15.9.3.6.1 Identification of Causes and Event Description

Stability following a long depressurization is addressed in this section. This simulates a decrease in reactor coolant inventory, though in this analysis no loss of coolant mass is credited. This has no functional impact, as the first trip that would

be reached would be the low-low pressurizer pressure instead of the low pressurizer level trip.

A decrease in RCS inventory without changes in primary pressure is not analyzed, as riser subcooling will be maintained and the protection system will trip the NPM on low pressurizer level before any appreciable effect can be seen regarding stability.

15.9.3.6.2 Sequence of Events and Systems Operation

Reactor power is 160 MW when a slow depressurization of approximately 0.5 psi/second is imposed as a boundary forcing function. This depressurization is done over 1000 seconds, resulting in the pressure reaching 1378 psia. No systems operations occur in response to the event.

15.9.3.6.3 Input Parameters and Initial Conditions

Input parameters and initial conditions for the limiting event is presented in Table 15.9-1.

The event is analyzed at various points throughout the cycle. BOC is analyzed as the least stable exposure point because the magnitude of the moderator reactivity is small. EOC is analyzed as, due to the stronger moderator reactivity feedback, the power response to a given flow oscillation will be larger if the oscillations do occur. Analysis at MOC was performed since the effect of exposure on different parameters is not in the same direction.

Results are presented at BOC as a sufficient example, as unstable oscillations will occur upon loss of riser inlet subcooling.

15.9.3.6.4 Results

The results are presented in Figure 15.9-14 to Figure 15.9-17. The results show that the reactor would be safely shut down well before the development of oscillations due to loss of subcooling in the riser. A reactor trip would be initiated before these oscillations develop, the low-low pressurizer pressure trip once the pressure reaches 1600 psia.

This trip would occur at approximately 530 seconds, while the oscillations begin to develop at approximately 927 seconds. The effect on CHFR was found to be quantitatively similar for all exposure points. This response, as seen in Figure 15.9-17 for BOC, is an increase relative to the initial value. The increase of CHFR is expected as a result of the increased natural circulation flow caused by voiding in the riser. This further confirms that this event is not a limiting event for stability.

15.9.4 Demonstration of Module Protection System Functions to Preclude Instability

At rated power, the minimum loop time for the NPM is more than 60 seconds. The response delay for the MPS is no more than 8.0 seconds for setpoints that are pertinent to stability

analysis and the scram time is less than 2.5 seconds. The time from the first scram setpoint being reached to the control rods being fully inserted is less than 11 seconds, which is significantly less than the minimum loop time for the NPM. Therefore, the MPS will shut down the reactor before any potential instability manifests itself as a divergent primary flow oscillation.

15.9.5 Conclusions

There are two main aspects of the stability methodology. The first is the use of a regional exclusion as the stability solution type and the rationale for its selection. The second aspect is the demonstration that the NPM maintains stability within the region of operation allowed by the MPS. The NPM returns to the original oscillation-free condition after steady state conditions are perturbed.

Operational events do not result in unstable plant behavior. At EOC values, the negative moderator coefficient suppresses the oscillation growth. At BOC, oscillations could occur; however, these oscillations do not occur because events that result in loss of riser subcooling and unstable operation are precluded by the region exclusion solution prior to instability.

The radiological consequences of all events shown in Section 15.9.3 are bounded by the design basis accident analyses presented in Section 15.0.3.

15.9.6 References

15.9-1 NuScale Power, LLC, “Evaluation Methodology for Stability Analysis of the NuScale Power Module,” TR-0516-49417-P, Rev. 1, August 2019.

Table 15.9-1: Initial Conditions (100 Percent of Rated Power Cases)

Parameter		Analysis Value[1]
Initial reactor power		160 MWt
Core inlet temperature		496.6 ° F
Core inlet mass flow rate		587.0 kg/s
Pressurizer pressure		1850 psia
Feedwater temperature		299.7 ° F
Feedwater flow		1155 gpm
Steam generator pressure		461.1 psia
Moderator density coefficients, EOC[2]	b0	0.4850
	b1	1.4877
	b2	-1.8359
	b3	1.3687
	b4	-0.4568
Doppler reactivity coefficient, EOC[3]		2.15×10^{-3} pcm/$K^{1.5}$

1. No biases are considered, as parametric sensitivity studies showed no significant changes in results and nonlinear response.
2. Moderator density coefficients are implemented as described in Section 5.6.1.2 of Reference 15.9-1, but with the coefficients shown in this table instead of the coefficients shown in Reference 15.9-1.
3. Doppler reactivity coefficient is implemented as described in Section 5.6.1.1 of Reference 15.9-1, but with the coefficient shown in this table instead of the coefficients shown in Reference 15.9-1.

Table 15.9-2: Initial Conditions (20 Percent of Rated Power Cases)

Parameter	Analysis Value[1]
Initial reactor power	32 MWt
Core inlet temperature	525.6 °F
Core inlet mass flow rate	312.4 kg/s
Pressurizer pressure	1850 psia
Feedwater temperature	199.9 °F
Feedwater flow	202.5 gpm
Steam generator pressure	628.4 psia
Moderator density coefficients, EOC	See Table 15.9-1
Doppler reactivity coefficient, EOC[2]	1.87×10^{-3} pcm/K$^{1.5}$

1. No biases are considered, as parametric sensitivity studies showed no significant changes in results and nonlinear response.
2. Doppler reactivity coefficient is implemented as described in Section 5.6.1.1 of Reference 15.9-1, but with the coefficient shown in this table instead of the coefficients shown in Reference 15.9-1.

Table 15.9-3: Initial Conditions (1 Percent of Rated Power Cases)

Parameter		Analysis Value[1]
Initial reactor power		1.6 MWt
Core inlet temperature		426.7 °F
Core inlet mass flow rate		94.75 kg/s
Pressurizer pressure		1850 psia
Feedwater temperature		50.0 °F
Feedwater flow		6.226 gpm
Steam generator pressure		331.0 psia
Moderator density coefficients, BOC[2]	a0	0.7789
	a1	0.6342
	a2	-0.7009
	a3	0.4406
	a4	-0.1583
Doppler reactivity coefficient, BOC[3]		1.12×10^{-3} pcm/$K^{1.5}$

1. No biases are considered, as parametric sensitivity studies showed no significant changes in results and nonlinear response.
2. Moderator density coefficients are implemented as described in Section 5.6.1.2 of Reference 15.9-1, but with the coefficients shown in this table instead of the coefficients shown in Reference 15.9-1.
3. Doppler reactivity coefficient is implemented as described in Section 5.6.1.1 of Reference 15.9-1, but with the coefficient shown in this table instead of the coefficients shown in Reference 15.9-1.

Table 15.9-4: Normal Power Operation - Limiting Analysis Results (1 Percent of Rated Power Case)

Acceptance Criteria	Limit	Analysis Value
Decay ratio	$\leq$0.8[1]	0.70

1. Though this limit does not apply to stability calculations below 5 percent power, it is included here because this case showed the highest decay ratio among all normal power operation analyses.

Table 15.9-5: Increase in Heat Removal by the Secondary System - Limiting Analysis Results (Rated Power Case)

Acceptance Criteria	Limit	Analysis Value
Decay ratio	≤0.8	0.03

Table 15.9-6: Increase in Heat Removal by the Secondary System - Limiting Analysis Results (20 Percent of Rated Power Case)

Acceptance Criteria	Limit	Analysis Value
Decay ratio	≤0.8	0.13

Table 15.9-7: Decrease in Heat Removal by the Secondary System - Limiting Analysis Results (Rated Power Case)

Acceptance Criteria	Limit	Analysis Value
Decay ratio	≤0.8	0.04

Table 15.9-8: Decrease in Heat Removal by the Secondary System - Limiting Analysis Results (20 Percent of Rated Power Case)

Acceptance Criteria	Limit	Analysis Value
Decay ratio	≤0.8	0.15

Table 15.9-9: Reactivity and Power Distribution Anomalies - Limiting Analysis Results (20 Percent of Rated Power Case)

Acceptance Criteria	Limit	Analysis Value
Decay ratio	≤0.8	0.14

Figure 15.9-1: Illustration of Decay Ratio versus Riser Subcooling

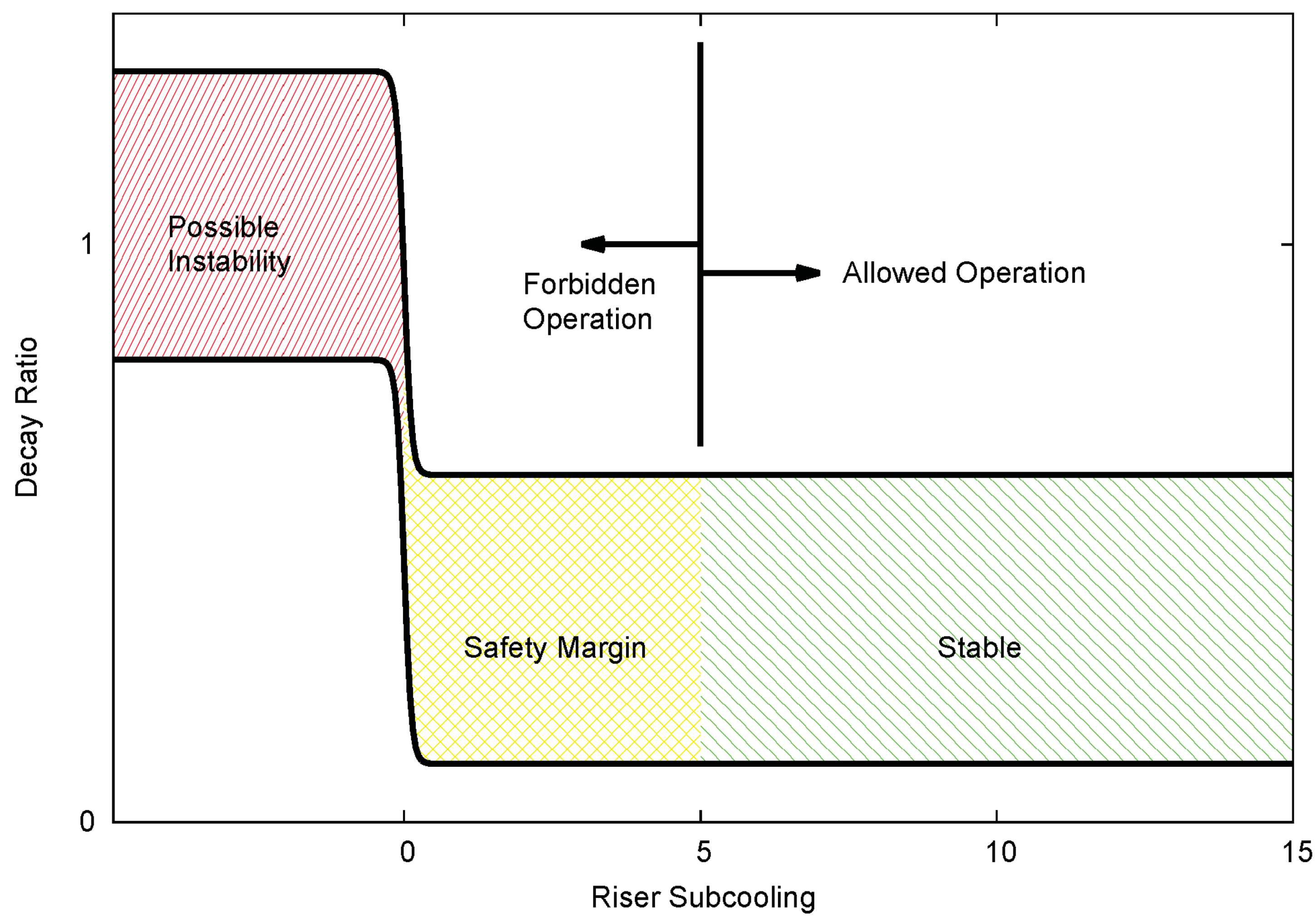


Figure 15.9-2: Time Trace of Primary Coolant Flow Response to a Perturbation at 1 Percent of Rated Power and Beginning of Cycle Reactivity

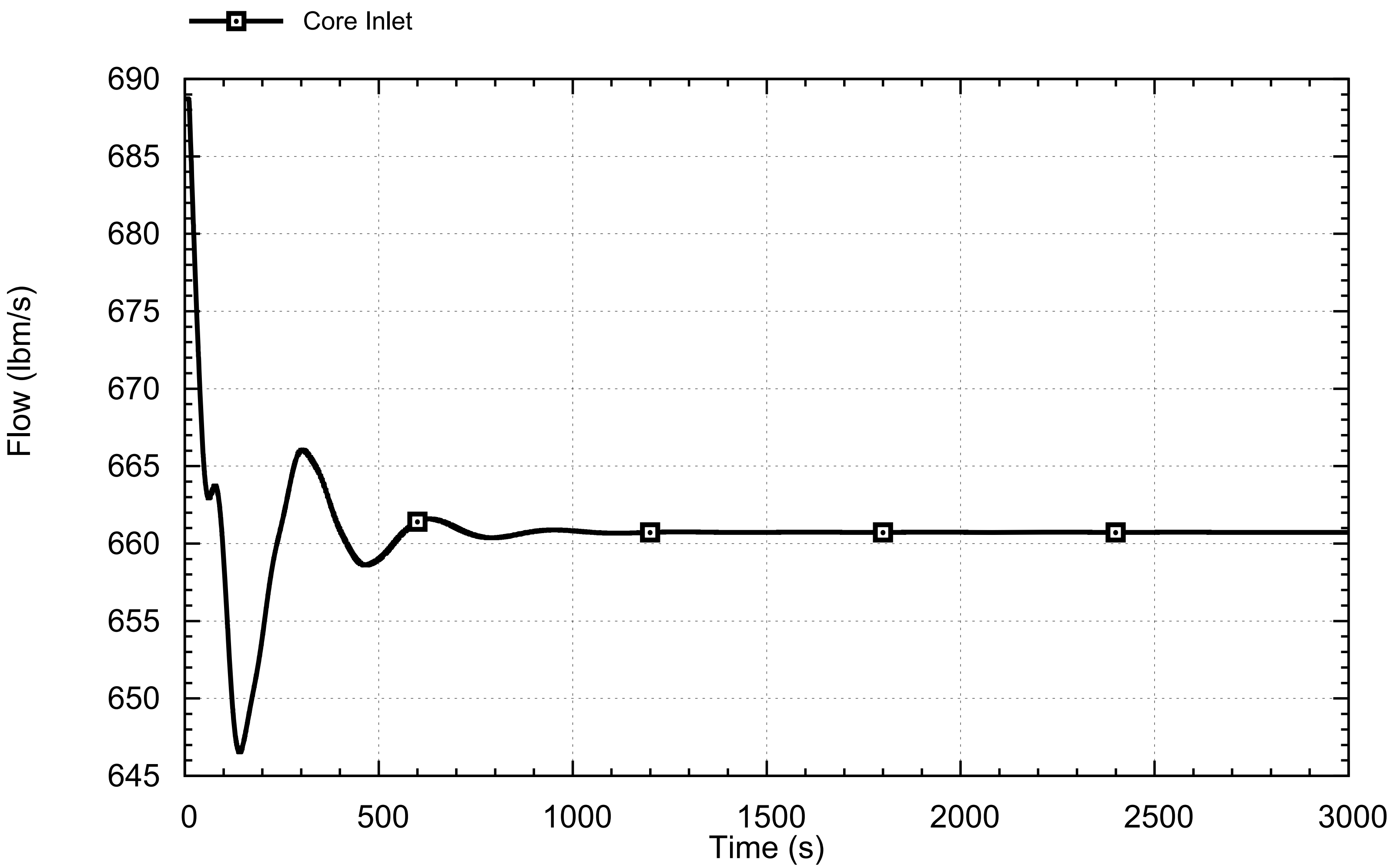


Figure 15.9-3: Time Trace of Heat Addition and Heat Removal Response to a Perturbation at 1 Percent of Rated Power and Beginning of Cycle Reactivity

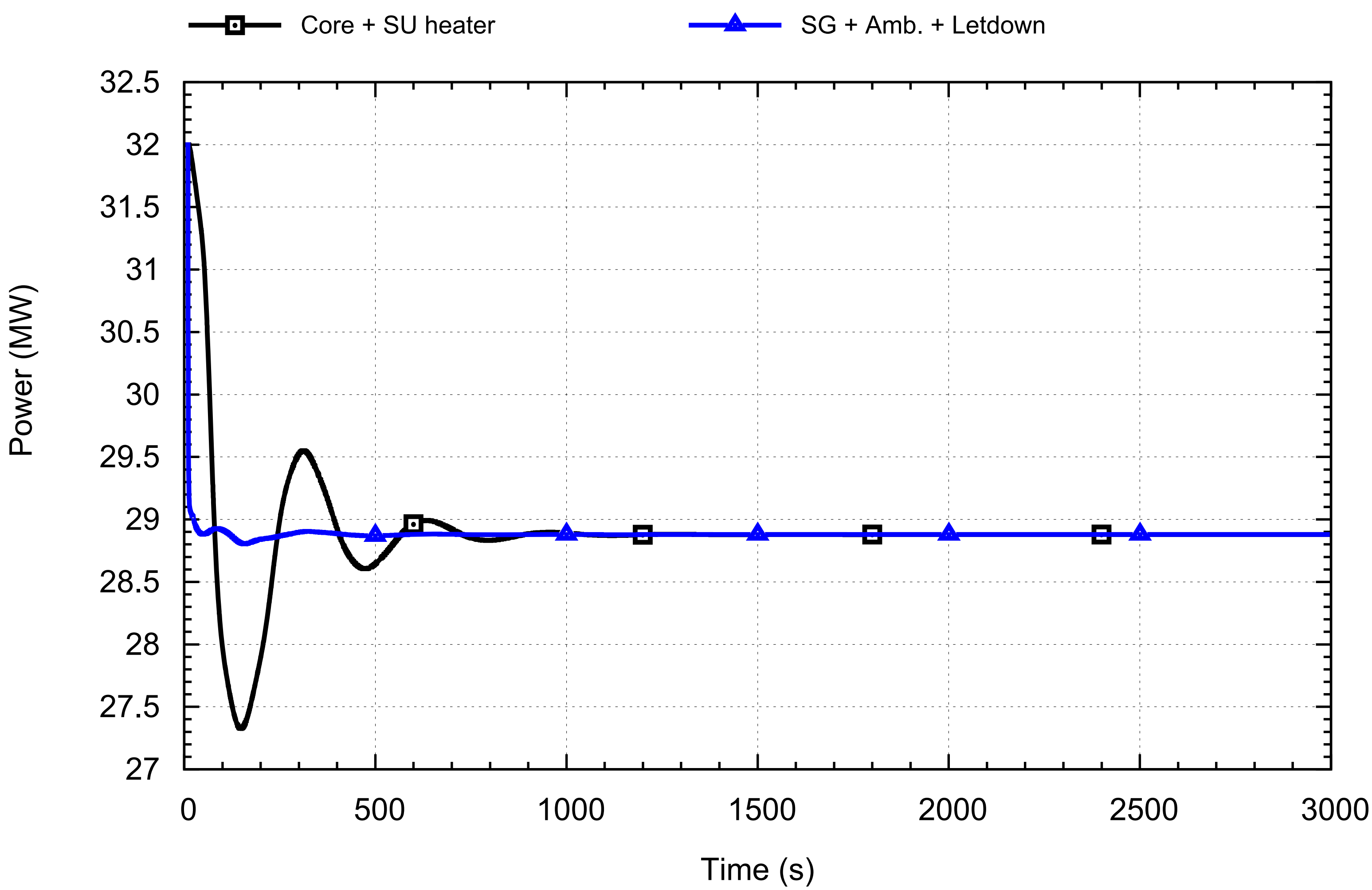


Figure 15.9-4: Time Trace of Primary Coolant Flow Response to an Increase in Heat Removal by the Secondary System at Rated Power and End of Cycle Reactivity

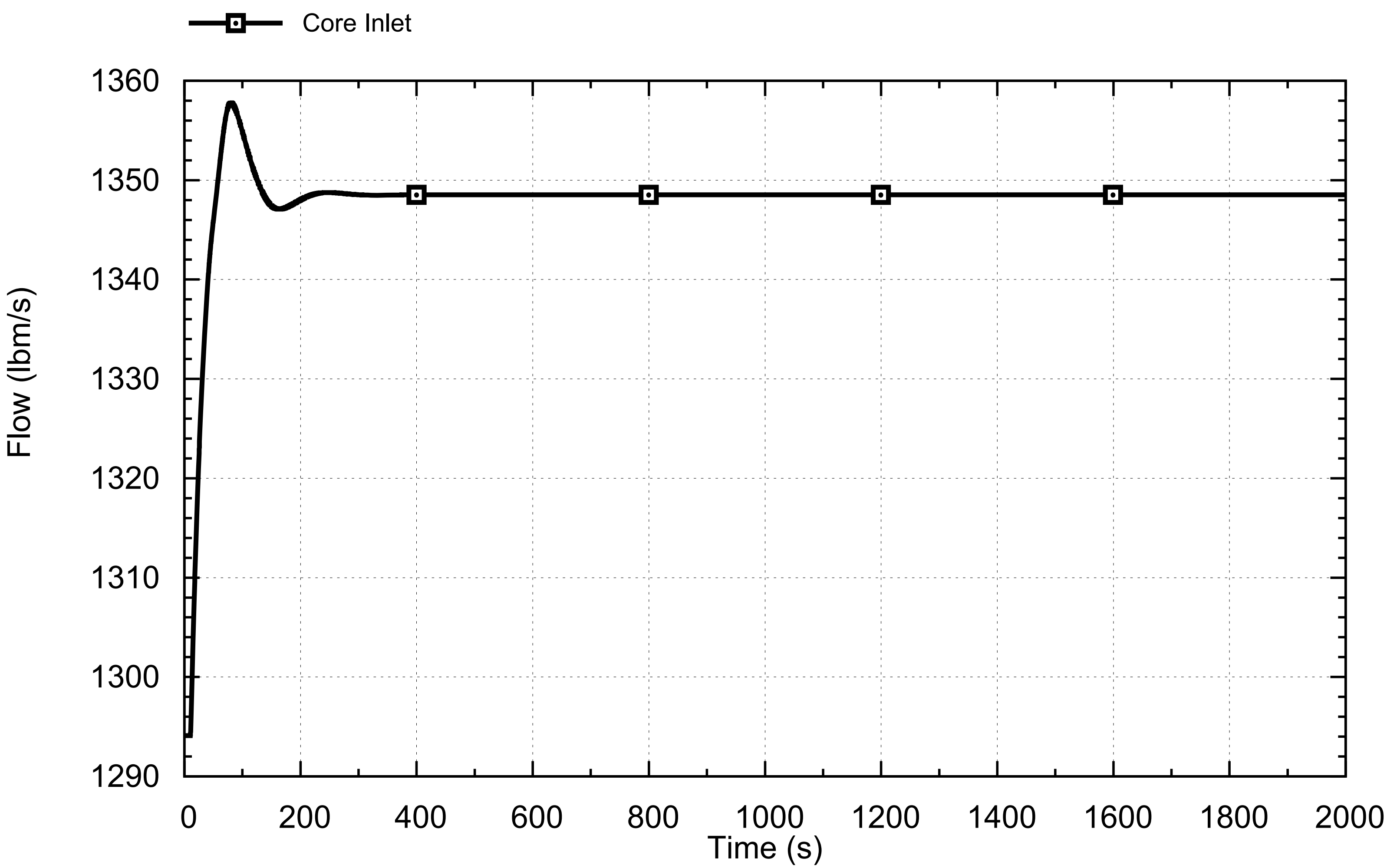


Figure 15.9-5: Time Trace of Heat Addition and Heat Removal Response to an Increase in Heat Removal by the Secondary System at Rated Power and End of Cycle Reactivity

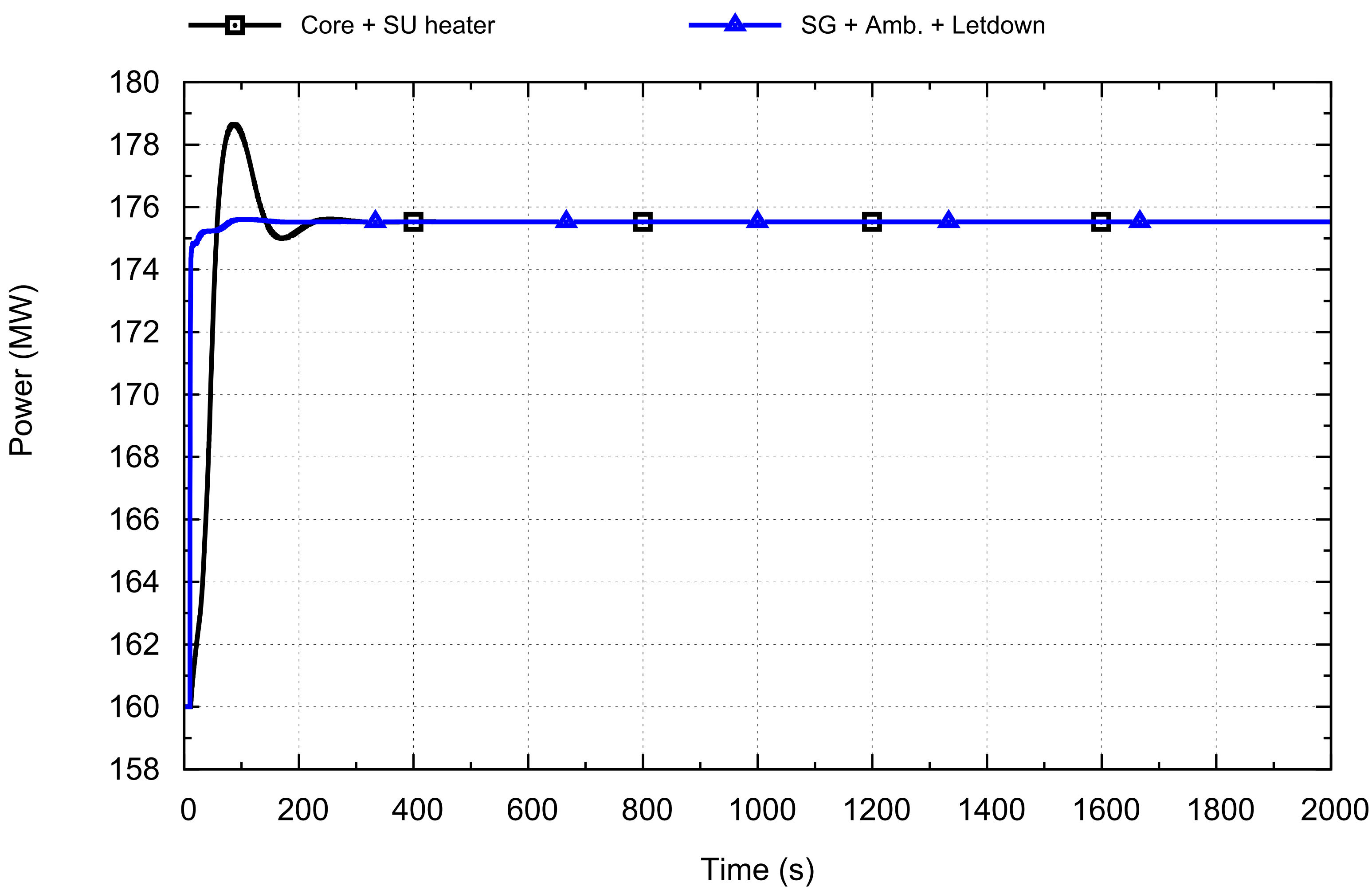


Figure 15.9-6: Time Trace of Primary Coolant Flow Response to an Increase in Heat Removal by the Secondary System at 20 Percent of Rated Power and End of Cycle Reactivity

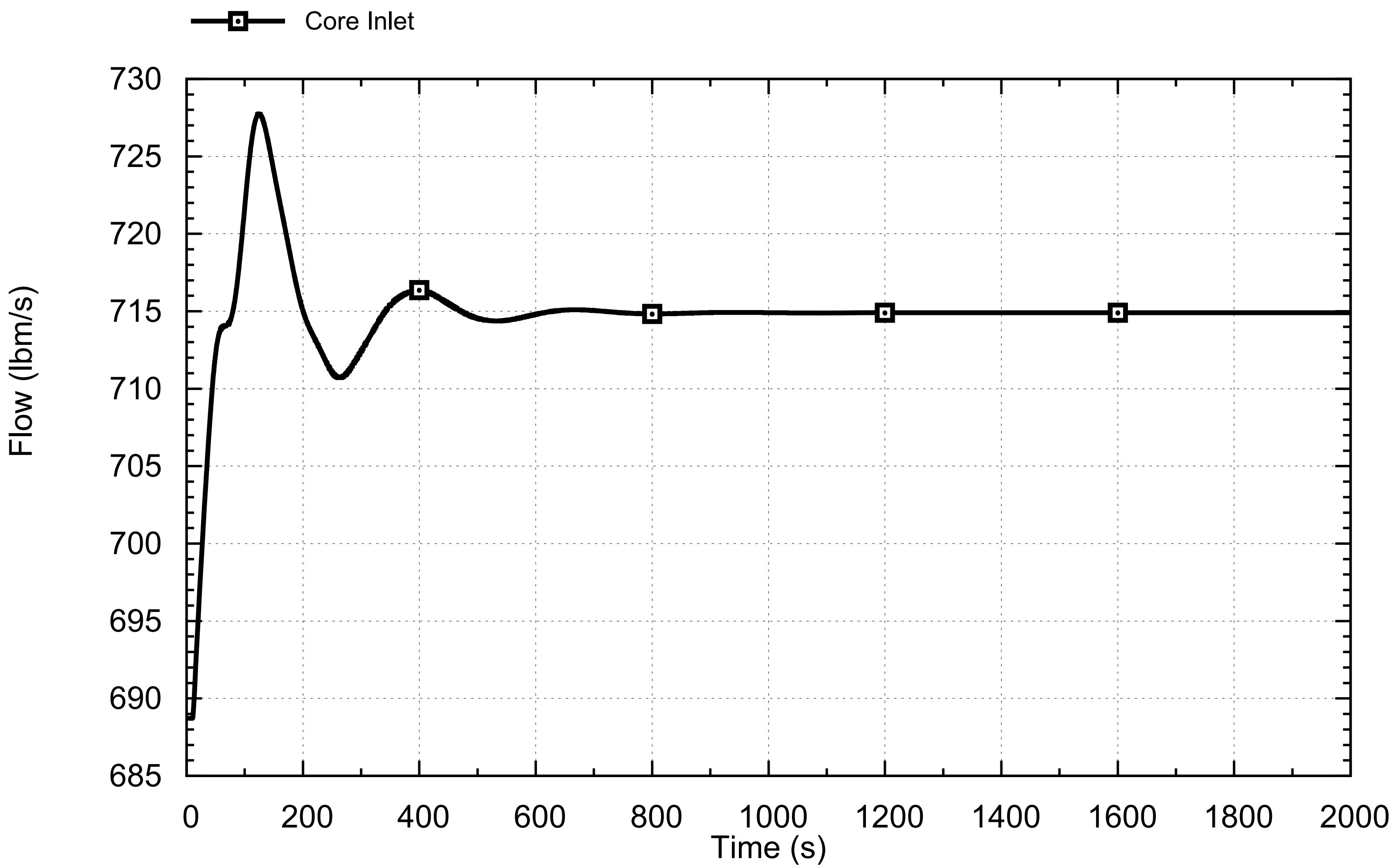


Figure 15.9-7: Time Trace of Heat Addition and Heat Removal Response to an Increase in Heat Removal by the Secondary System at 20 Percent of Rated Power and End of Cycle Reactivity

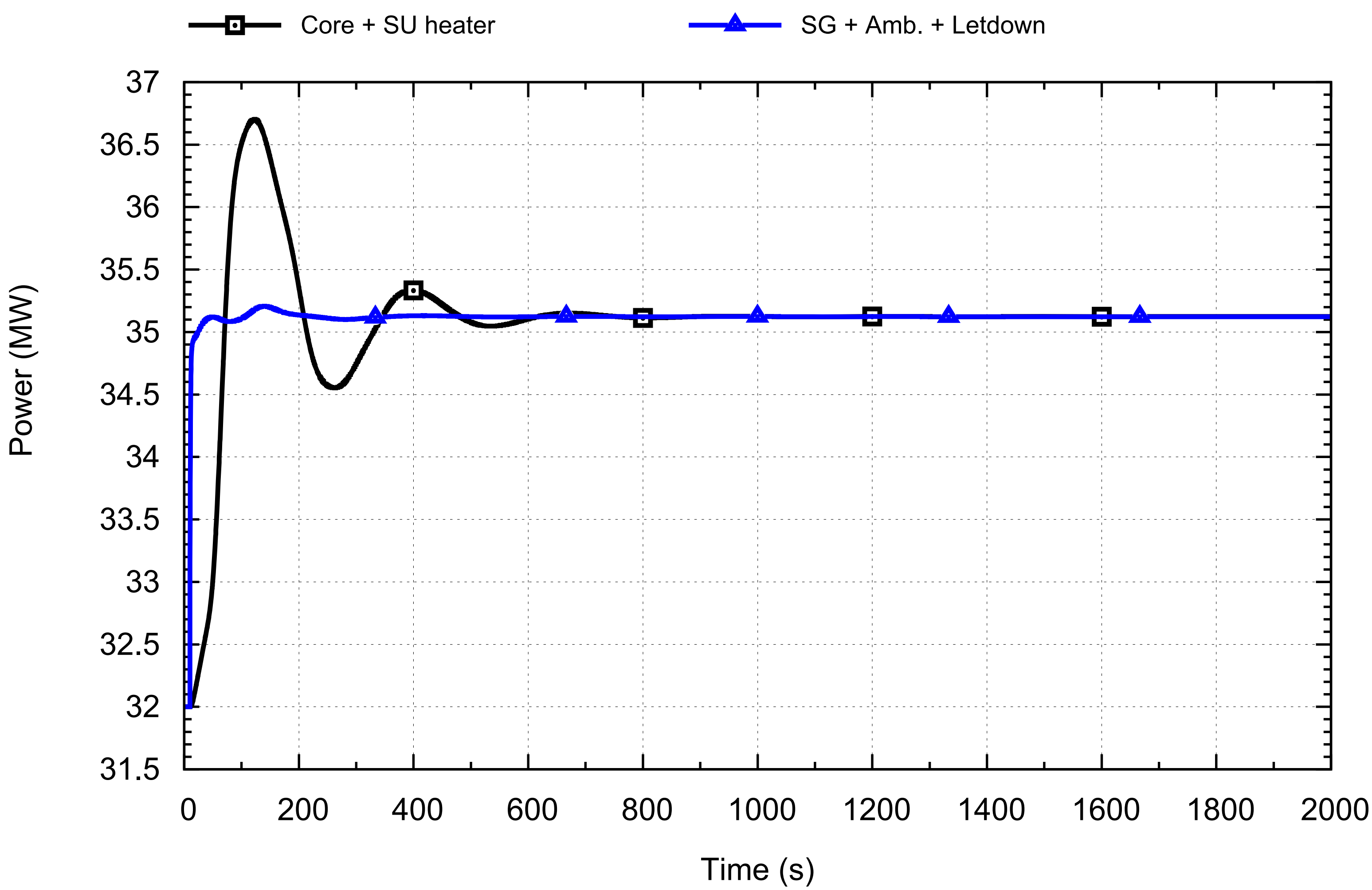


Figure 15.9-8: Time Trace of Primary Coolant Flow Response to a Decrease in Heat Removal by the Secondary System at Rated Power and End of Cycle Reactivity

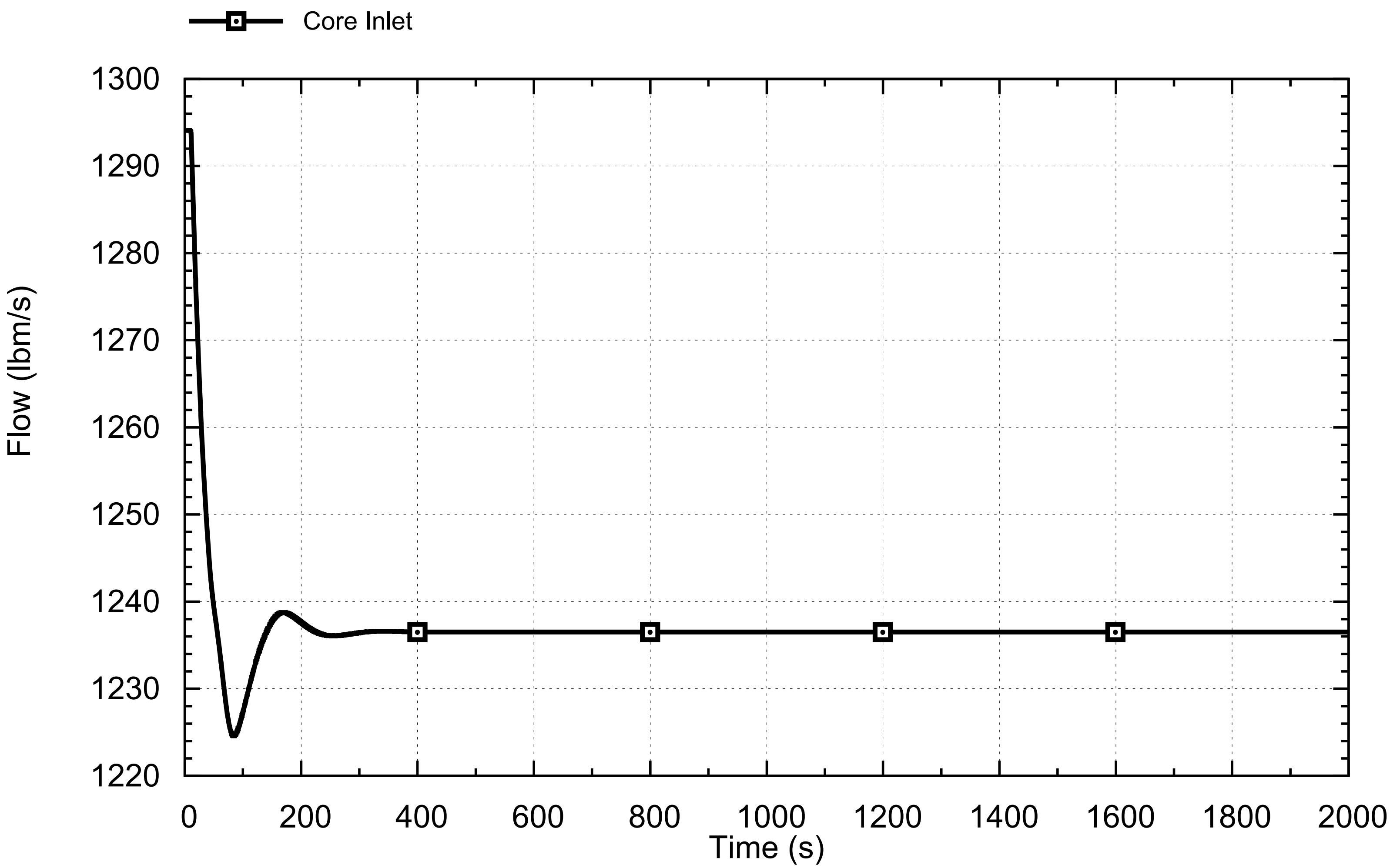


Figure 15.9-9: Time Trace of Heat Addition and Heat Removal Response to a Decrease in Heat Removal by the Secondary System at Rated Power and End of Cycle Reactivity

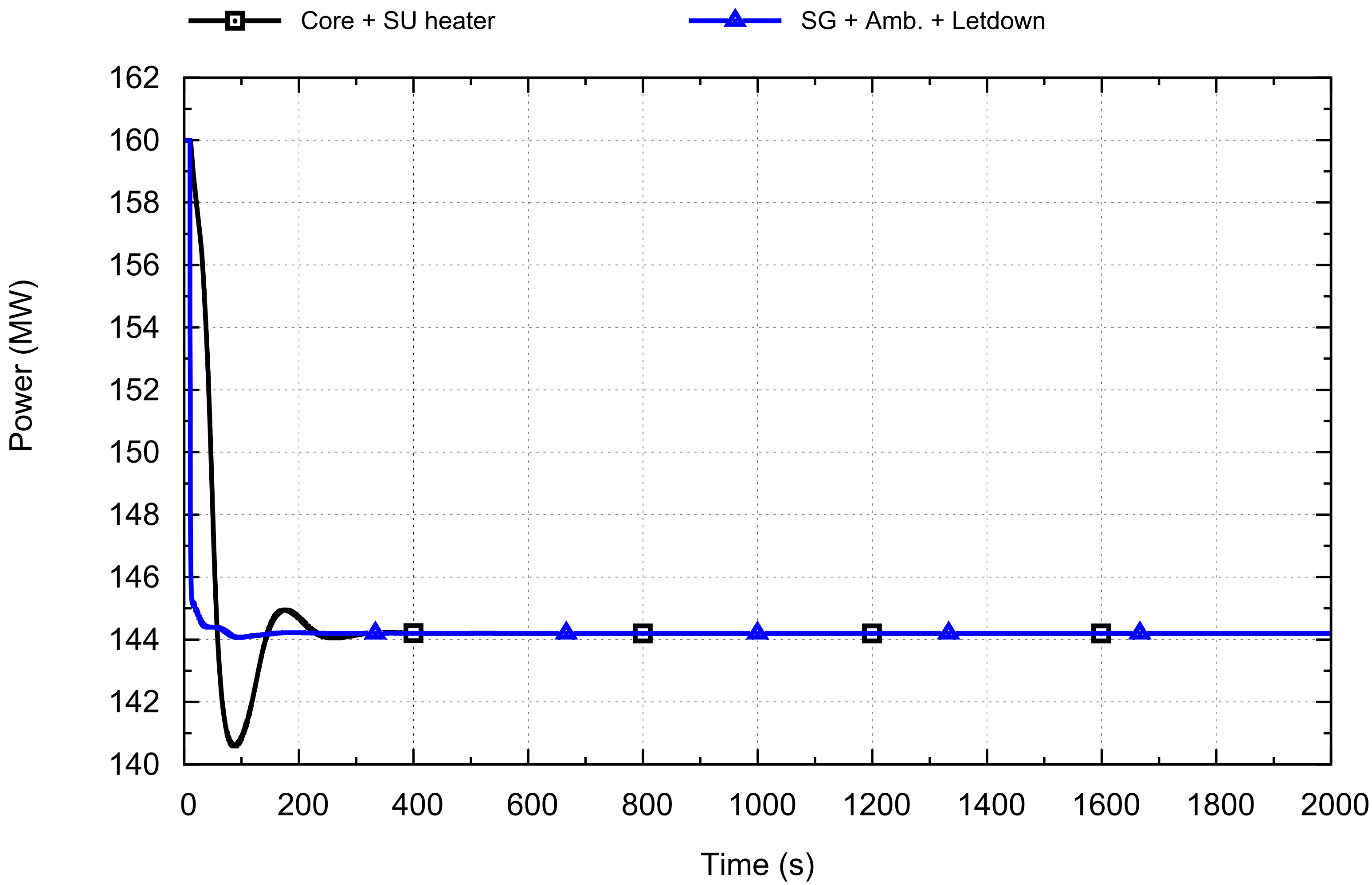


Figure 15.9-10: Time Trace of Primary Coolant Flow Response to a Decrease in Heat Removal by the Secondary System at 20 Percent of Rated Power and End of Cycle Reactivity

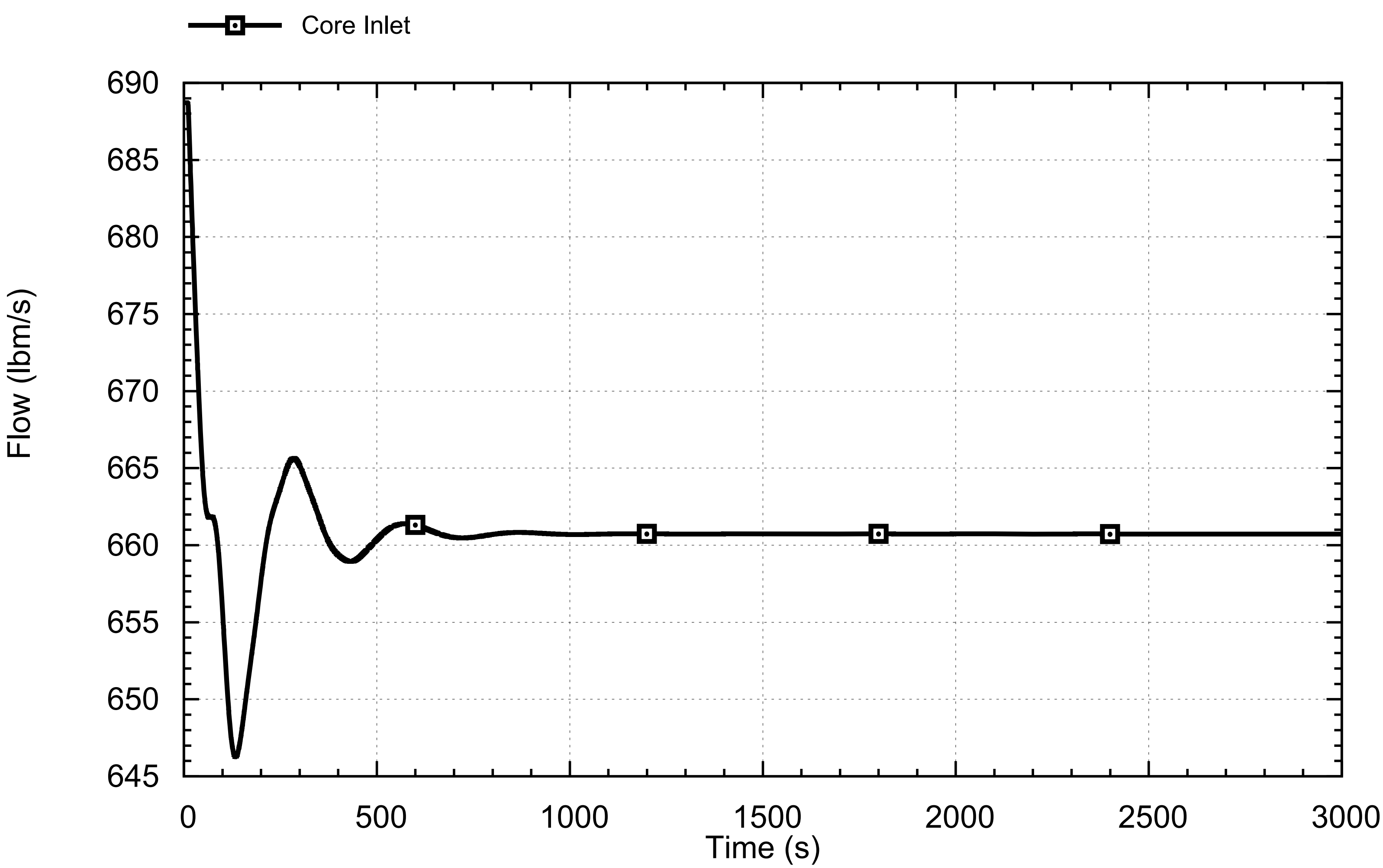


Figure 15.9-11: Time Trace of Heat Addition and Heat Removal Response to a Decrease in Heat Removal by the Secondary System at 20 Percent of Rated Power and End of Cycle Reactivity

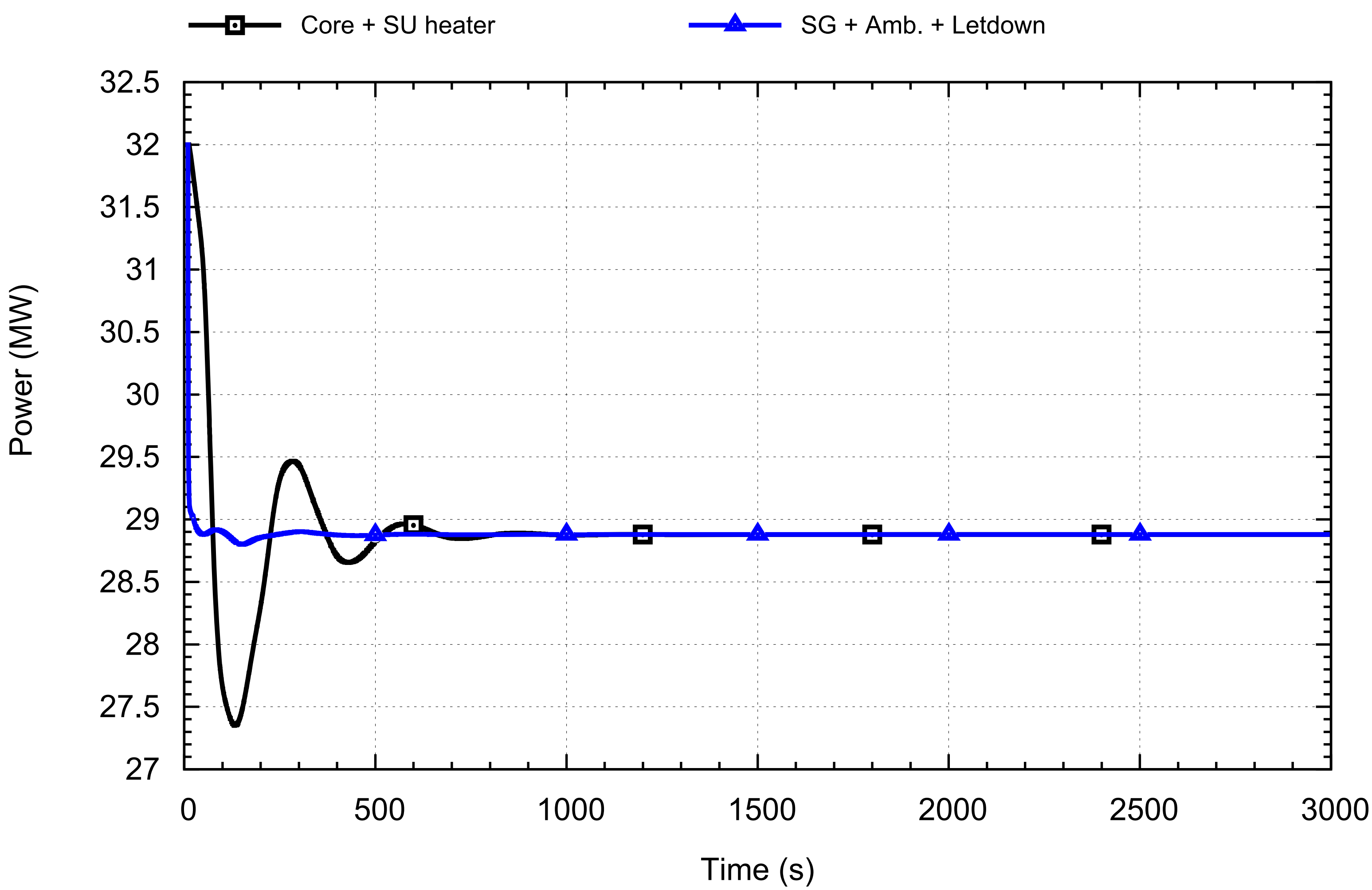


Figure 15.9-12: Time Trace of Primary Coolant Flow Response to Reactivity and Power Distribution Anomalies at 20 Percent of Rated Power and End of Cycle Reactivity

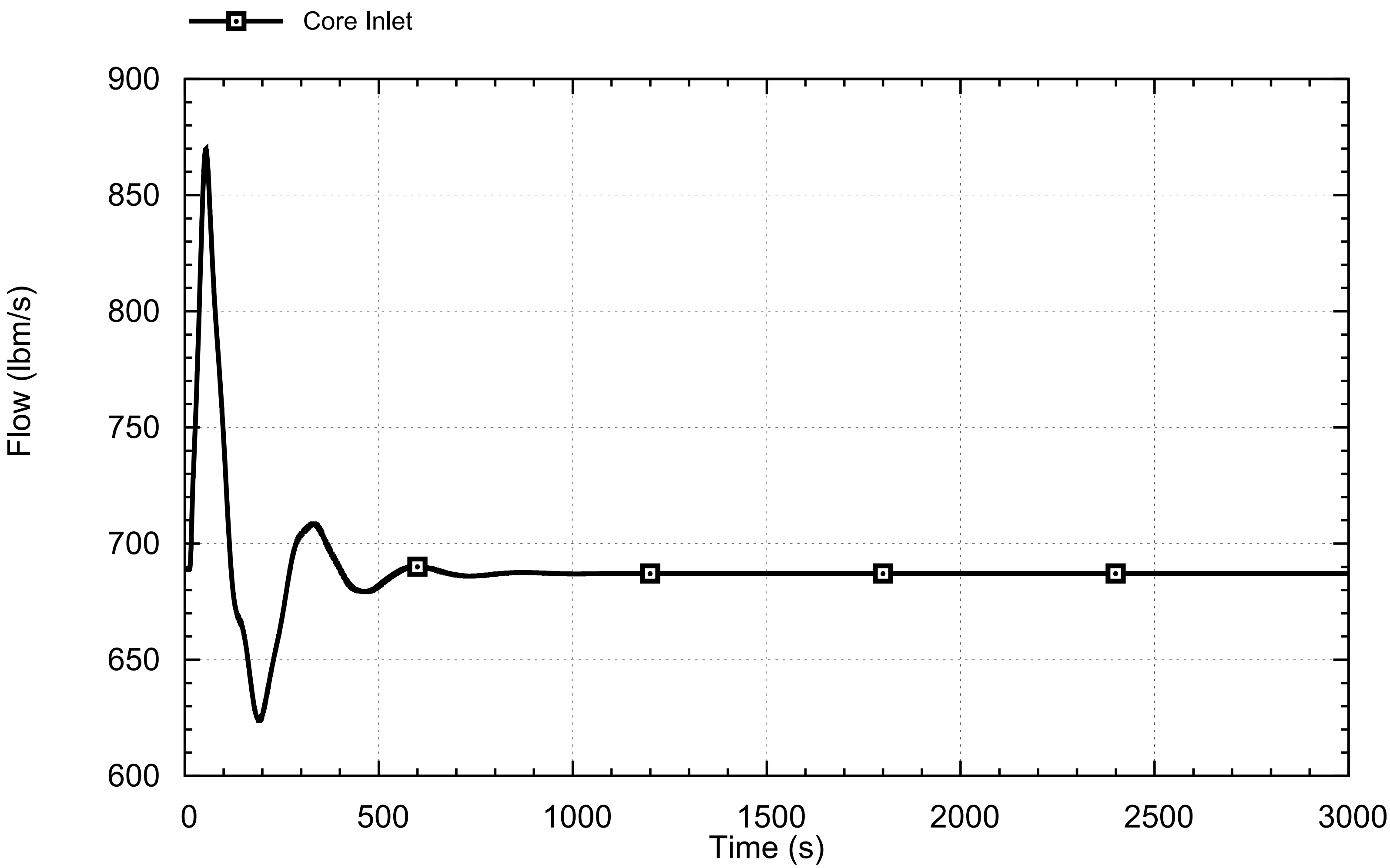


Figure 15.9-13: Time Trace of Heat Addition and Heat Removal Response to Reactivity and Power Distribution Anomalies at 20 Percent of Rated Power and End of Cycle Reactivity

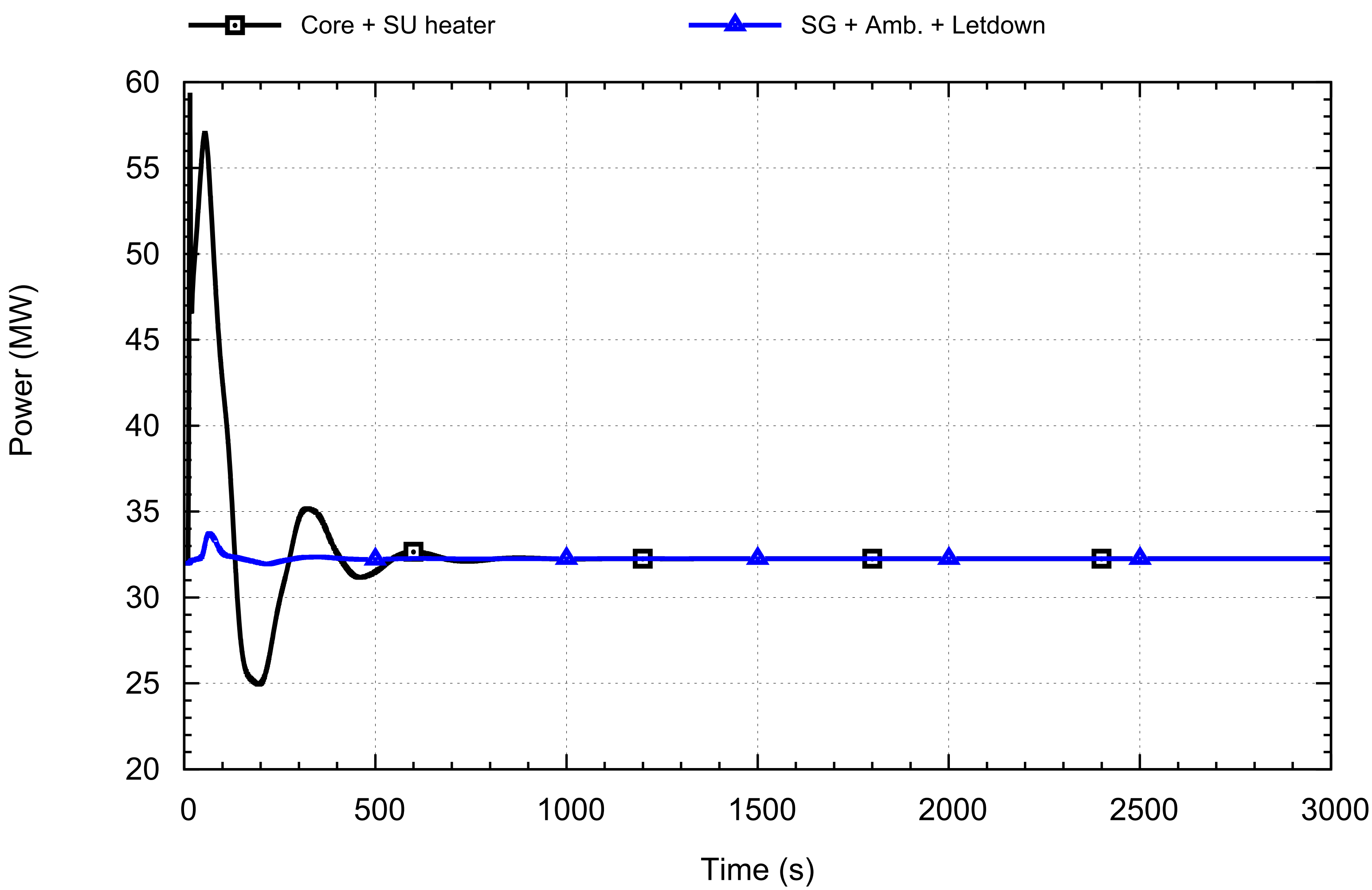


Figure 15.9-14: Time Trace of Pressure Boundary Function Representing the Effect of Decrease in Reactor Coolant Inventory

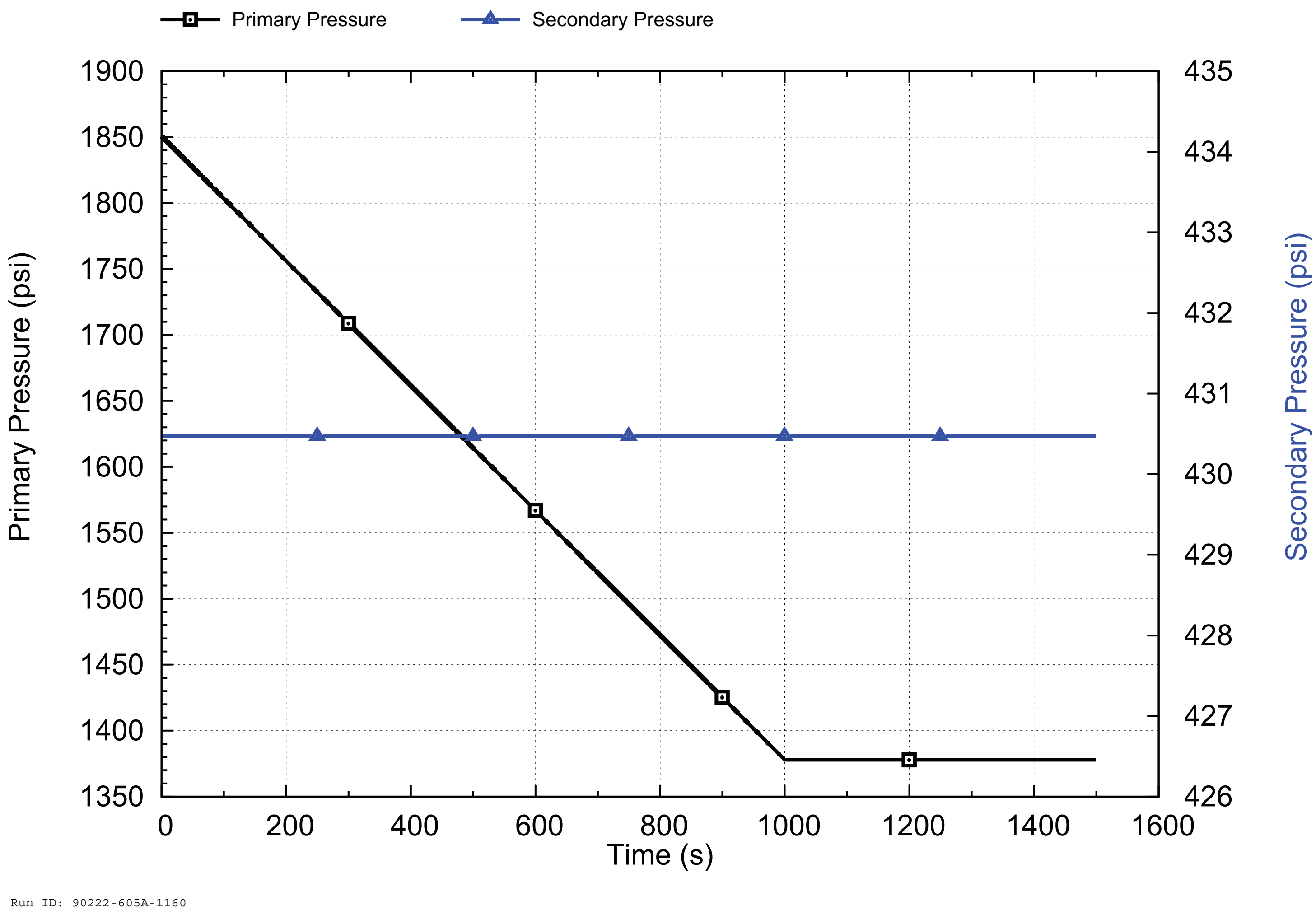
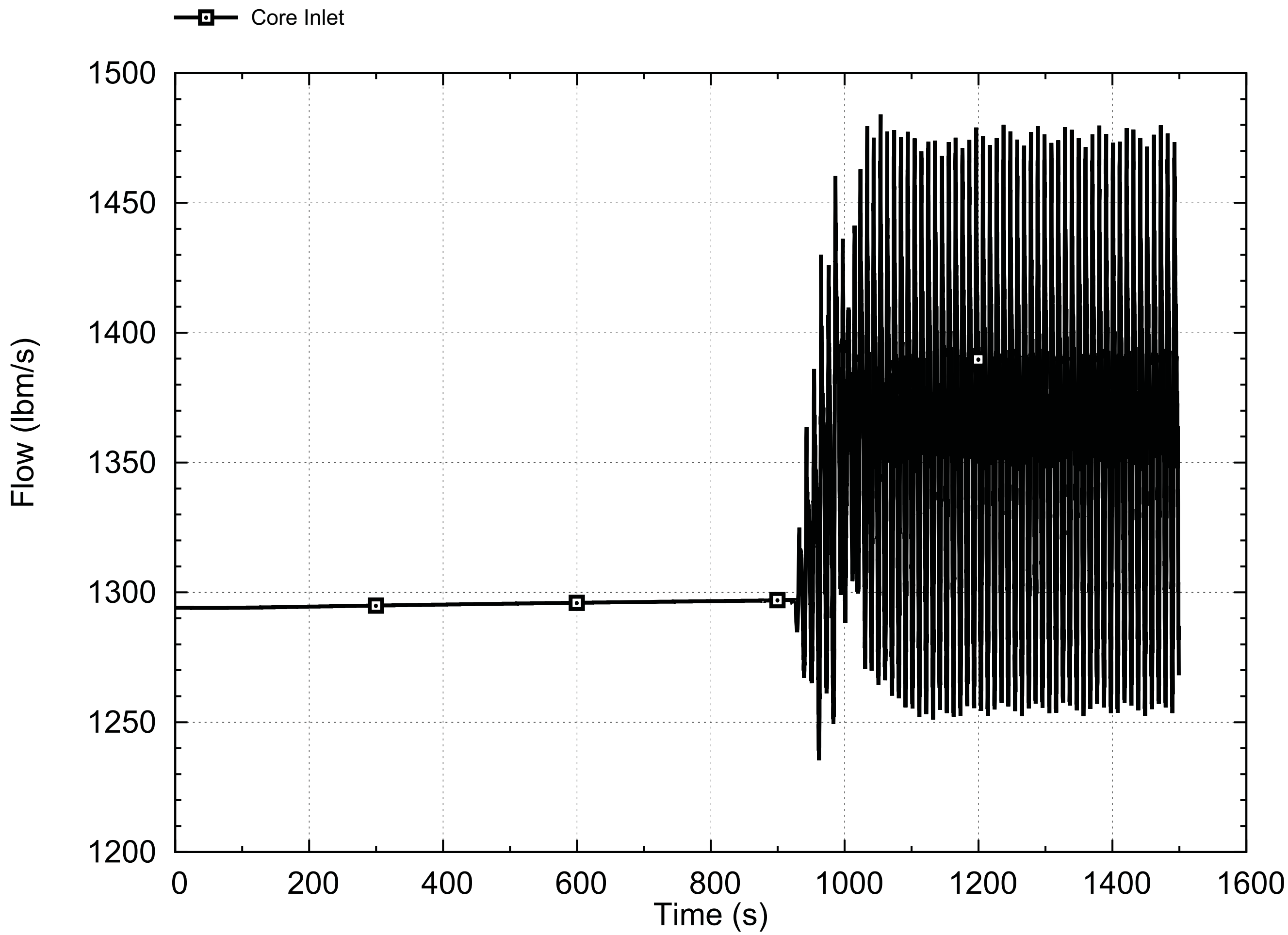
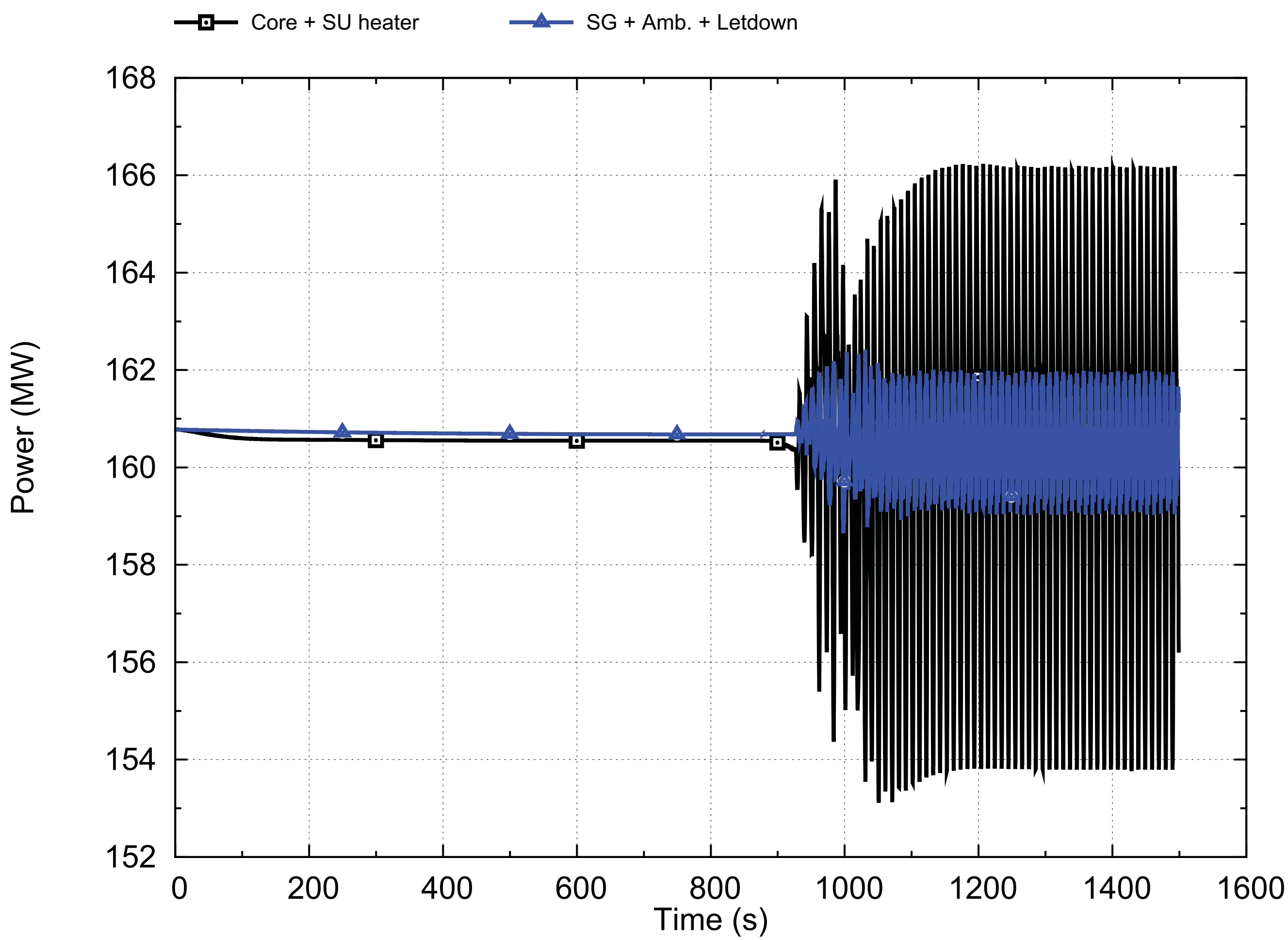


Figure 15.9-15: Time Trace of Flow Response to a Decrease in Reactor Coolant Inventory at 100 Percent of Rated Power and Beginning of Cycle Reactivity



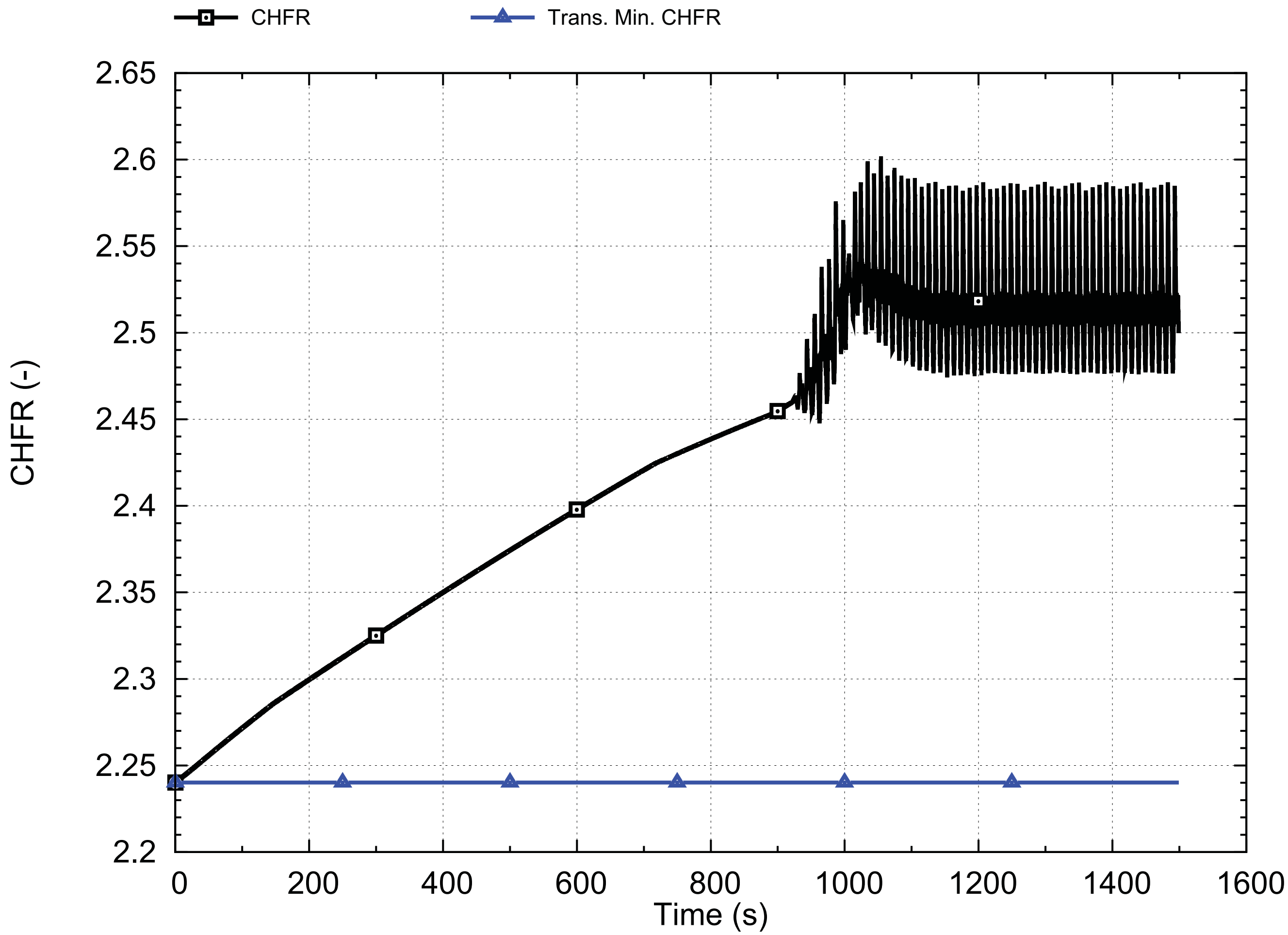
Run ID: 90222-605A-1160

Figure 15.9-16: Time Trace of Power Response to a Decrease in Reactor Coolant Inventory at 100 Percent of Rated Power and Beginning of Cycle Reactivity



Run ID: 90222-605A-1160

Figure 15.9-17: Time Trace of CHFR Response to a Decrease in Reactor Coolant Inventory at 100 Percent of Rated Power and Beginning of Cycle Reactivity



Run ID: 90222-605A-1160

15.10 Core Damage Event

A beyond-design-basis core damage event (CDE), with an associated core damage source term (CDST) composed of a set of key parameters derived from a spectrum of surrogate accident scenarios, is postulated. The beyond-design-basis CDE analysis and the design-basis iodine spike design basis source term (DBST) analysis described in Section 15.0.3 are each assessed against the radiological criteria of 10 CFR 52.47(a)(2)(iv), and if both analyses show acceptable dose results, 10 CFR 52.47(a)(2)(iv) is met. The analysis of the beyond-design-basis CDST against the acceptance criteria of 10 CFR 52.47(a)(2)(iv) provides reasonable assurance that, even in the extremely unlikely event of a severe accident, the facility's design features and site characteristics provide adequate protection of the public.

The inputs, methods, and assumptions used to derive the CDST and analyze its radiological consequences are discussed in Section 15.10.1. The radiological consequences of the CDST are discussed in Section 15.10.2.

15.10.1 Inputs, Methods, and Assumptions

15.10.1.1 Core Radionuclide Inventory

The core radionuclide inventory described in Section 15.0.3, and shown in Table 11.1-1, is assumed for the CDST.

15.10.1.2 Primary Coolant Radionuclide Inventory

The primary coolant radionuclide inventory is assumed to be zero and is not considered as a contributor to dose in the CDE radiological consequence analysis, in accordance with the methodology of Reference 15.0-4.

15.10.1.3 Secondary Coolant Activity

The secondary coolant activity is assumed to be zero and is not considered as a contributor to dose in the CDE radiological consequence analysis, in accordance with the methodology of Reference 15.0-4.

15.10.1.4 Source Term Release Timing and Magnitude

The CDST associated with the CDE is composed of a set of key parameters, such as fuel release fractions and timing, derived from a spectrum of surrogate accident scenarios. A surrogate accident scenario is a postulated event that results in core damage with subsequent release of appreciable quantities of fission products into an intact containment, that serves as a surrogate to the large break loss-of-coolant accident with a substantial meltdown of the core typically evaluated by light water reactors as the maximum hypothetical accident. Five surrogate accident scenarios derived from intact-containment internal events in the Level 1 probabilistic risk assessment were used to establish the CDST in accordance with the methodology of Reference 15.0-4.

Each of the five surrogate accident sequence cases involves various failure of the emergency core cooling system (ECCS), (i.e., all ECCS valves failing to open, the reactor vent valves (RVVs) failing to open, or the reactor recirculation valves (RRVs) failing to

open). In each case, the decay heat removal system is assumed available to remove heat. The five surrogate accident scenario cases are summarized as follows:

Case 1: chemical and volume control system (CVCS) injection line break with all ECCS valves failing to open.

Case 2: CVCS injection line break with RVVs failing to open.

Case 3: CVCS injection line break with RRVs failing to open.

Case 4: loss of direct current (DC) power with the RVVs failing to open.

Case 5: loss of DC power with the RRVs failing to open.

Each surrogate accident scenario case is modeled using MELCOR to provide a representative range of release timing and fractions for the development of the CDST. Section 15.0.2.4 provides a discussion of the MELCOR computer code. The methodology used to identify the release magnitude and timing of the CDST is provided in Reference 15.0-4. Release timing and core inventory release fractions for the CDST are listed in Table 15.10-1 and Table 15.10-2, respectively.

The radioactive source term is calculated by multiplying the maximum core inventory provided in Table 11.1-1 by the release fractions provided in Table 15.10-2. The radioactivity released from the fuel is assumed to mix instantaneously and homogeneously throughout the free air volume of the containment.

15.10.1.5 Aerosol and Elemental Iodine Removal

Natural deposition phenomena, including sedimentation, diffusiophoresis, thermophoresis, and hygroscopicity, result in aerosol removal. The NuScale aerosol removal methodology uses the aerosol removal code STARNAUA to track these various deposition phenomena in calculating time-dependent airborne aerosol mass and removal rates. Section 15.0.2.4 provides a discussion of the STARNAUA computer code. Aerosol removal rates used for the CDE radiological consequence analysis are provided in Table 15.10-3. A key assumption of the NuScale aerosol transport methodology is that there is no maximum limit on iodine decontamination factor because removal is facilitated by natural processes, as opposed to an active spray system. The NuScale removal rate calculation methodology is based on the calculated time-dependent airborne aerosol mass in accordance with Appendix A, Section 3.3 of RG 1.183. NuScale conservatively does not take credit for elemental iodine removal. Rather, only aerosol removal is credited. A summary of the aerosol transport and removal calculation process is described in Reference 15.0-4. Treatment of aerosol resuspension and revaporization is discussed in Reference 15.0-4.

15.10.1.6 Chemical Form of Iodine

Reference 15.0-4 provides the methodology for the radiological consequences of the CDE, based on the guidance provided in Appendix A of RG 1.183. The chemical form of radioiodine released to the containment atmosphere is 95 percent cesium iodide,

4.85 percent elemental iodine, and 0.15 percent organic iodide. Note that the methodology considers cesium iodide as an aerosol.

15.10.1.7 RADTRAD Modeling

The RADTRAD modeling techniques described in Section 15.0.3.3.8 are used to analyze the CDST.

15.10.1.8 pH_T and Iodine Re-Evolution

The CDE radiological consequence methodology calculates the post-accident pH_T. The pH_T code is used to calculate the extent of iodine re-evolution inside containment. During the postulated CDE, additional acids and bases may enter the coolant and cause a change in pH_T. The expected overall pH_T of the coolant is modeled over a period of 30 days. Section 15.0.2 provides a discussion of the NuScale pH_T program used to calculate post-accident pH_T.

Details about the methodologies used for evaluating post-accident pH_T in coolant water following an event that results in significant core damage are presented in Reference 15.0-4. The results of implementing the methodology show the post-accident pH_T inside containment is between 6.0 and 7.0.

The CDE radiological consequence methodology assumes a negligible amount of iodine re-evolution occurs between pH_T values of 6.0 and 7.0, and does not need to be explicitly included in the dose analysis calculation. This assumption simplifies the analysis without an impact to the conservatism of the calculated dose results. The treatment of iodine re-evolution is described in Reference 15.0-4.

15.10.1.9 Atmospheric Dispersion Factors (χ/Q), Breathing Rates, and Occupancy Factors

The atmospheric dispersion factor (χ/Q), breathing rate, and occupancy factor inputs to RADTRAD described in Section 15.0.3.3.11 are assumed in the CDE radiological consequence analysis.

15.10.1.10 Dose Conversion Factors

The dose conversion factors described in Section 15.0.3.3.12 are assumed in the CDE radiological consequence analysis.

15.10.1.11 Containment Leakage

The containment leakage assumptions described in Section 15.0.3.4 are assumed in the CDE radiological consequence analysis. Activity is released to the atmosphere from the containment at the design-basis containment leak rate provided in Table 6.5-1 for 24 hours, and at 50 percent of the design-basis containment leak rate after 24 hours.

15.10.1.12 Secondary-Side Decontamination

The secondary-side decontamination assumptions described in Section 15.0.3.5 are assumed in the CDE radiological consequence analysis.

15.10.1.13 Reactor Building Decontamination Factors

The Reactor Building decontamination assumptions described in Section 15.0.3.6 are assumed in the CDE radiological consequence analysis.

15.10.1.14 Receptor Location Considerations

The receptor location, control room, technical support center, and Reactor Building pool boiling radiological consequence assumptions described in Section 15.0.3.7 are assumed in the CDE radiological consequence analysis. The control room model is described in Section 15.0.3.7.

15.10.2 Radiological Consequences of the Core Damage Source Term

Using the inputs, methods, and assumptions described in Section 15.10.1, the potential radiological consequences of the CDE are calculated and presented in Table 15.0-12. As shown in Table 15.0-12, NuScale meets the radiological acceptance criteria for the CDE. Because NuScale already meets the radiological acceptance criteria, NuScale elected not to exercise the provisions in Reference 15.0-4 that allow less conservative analysis assumptions for the beyond-design-basis CDE (e.g., 50th percentile χ/Q values, Reactor Building decontamination factors, median core radionuclide inventory, etc.).

Table 15.10-1: Core Damage Source Term Release Timing

Parameter	Value
Delay of radionuclide release into containment	3.80 hours
Duration of radionuclide release into containment	1.00 hour

Table 15.10-2: Core Inventory Release Fractions

Radionuclide Group	Release Fraction into Containment Vessel
Noble gases	3.9E-01
Halogens	1.4E-01
Alkali metals	2.0E-01
Alkaline earths	5.3E-03
Tellurium group	1.5E-01
Molybdenum group	4.9E-02
Noble metals	7.9E-04
Lanthanides	2.1E-08
Cerium group	2.1E-08

Table 15.10-3: Containment Aerosol Removal Rates

Time (hours)	Removal Rate ($hour^{-1}$)
0.00E+00	0.00E+00
3.80E+00	2.20E+01
3.99E+00	9.54E+00
4.18E+00	3.66E+00
4.42E+00	2.06E+00
4.67E+00	1.83E+00
4.87E+00	1.73E+00
5.15E+00	1.98E+00
6.19E+00	1.76E+00
2.59E+01	0.00E+00