GE Hitachi Nuclear Energy

NEDO-34183

Revision A

January 2025

US Protective Marking: Non-Proprietary Information
UK Protective Marking: Not Protectively Marked

BWRX-300 UK Generic Design Assessment (GDA) Chapter 15.5 - Deterministic Safety Analysis

US Protective Marking: Non-Proprietary Information
UK Protective Marking: Not Protectively Marked

INFORMATION NOTICE

This document does not contain proprietary information and carries the notations "US Protective Marking: Non-Proprietary Information" and "UK Protective Marking: Not Protectively Marked."

IMPORTANT NOTICE REGARDING CONTENTS OF THIS REPORT

Please Read Carefully

The design, engineering, and other information contained in this document is furnished for the purpose of obtaining the applicable Nuclear Regulatory Authority review and determination of acceptability for use for the BWRX-300 design and licensing basis information contained herein. The only undertakings of GEH with respect to information in this document are contained in the contracts between GEH and its customers or participating utilities, and nothing contained in this document shall be construed as changing those contracts. The use of this information by anyone for any purpose other than that for which it is intended is not authorized; and with respect to any unauthorized use, no representation or warranty is provided, nor any assumption of liability is to be inferred as to the completeness, accuracy, or usefulness of the information contained in this document. Furnishing this document does not convey any license, express or implied, to use any patented invention or any proprietary information of GEH, its customers or other third parties disclosed herein or any right to publish the document without prior written permission of GEH, its customers or other third parties.

UK SENSITIVE NUCLEAR INFORMATION AND US EXPORT CONTROL INFORMATION

This document does not contain any UK Sensitive Nuclear Information (SNI) subject to protection from public disclosure as described in the Nuclear Industries Security Regulations (NISR) 2003, does not contain UK Export Controlled Information (ECI), and does not contain US Export Controlled Information (ECI) subject to the export control laws and regulations of the United States, including 10 CFR Part 810.

EXECUTIVE SUMMARY

This document is Chapter 15.5, Deterministic Safety Analysis (DSA), of the Preliminary Safety Report (PSR) of the GE-Hitachi (GEH) BWRX-300 for the purposes of UK Generic Design Assessment (GDA). It contains the principal description and detailed results of the DSA and contains the related figures and tables.

The faults covered in this chapter are currently restricted to bounding reactor faults at power, supplemented with a bounding Fuel Handling Accident non-reactor fault. The analysis of Design Extension Conditions with core damage and faults associated with the fuel pool is currently out of scope for the deterministic analysis.

The overall results of the DSA, and other safety analyses, are presented in PSR Ch. 15.9.

ACRONYMS AND ABBREVIATIONS

Acronym	Explanation
2D	Two-Dimensional
3D	Three-Dimensional
ABWR	Advanced Boiling Water Reactor
ALARP	As Low As Reasonably Practicable
ALWR	Advanced Light Water Reactor
ANSI	American National Standards Institute
AOO	Anticipated Operational Occurrence
ATLM	Automatic Thermal Limit Monitor
ATS	Anticipatory Trip System
BDBA	Beyond Design Basis Accident / Beyond Design Basis Analysis
BL	Baseline
BL-DBA	Baseline Design Basis Analysis
BL-DSA	Baseline Deterministic Safety Analysis
BOC	Beginning of Cycle
BWR	Boiling Water Reactor
CB	Control Building
CCF	Common Cause Failure
CEDE	Committed Effective Dose Equivalent
CET	Containment Event Tree
CFR	Code of Federal Regulations
CIV	Containment Isolation Valve
COLR	Core Operating Limits Report
CRD	Control Rod Drive
CRDA	Control Rod Drop Accident
CSAU	Code Scaling, Applicability, and Uncertainty
C&I	Control and Instrumentation
CN-DBA	Conservative Design Basis Analysis
CN-DSA	Conservative Deterministic Safety Assessment
DBA	Design Basis Accident / Design Basis Analysis
DCF	Dose Conversion Factor
DDE	Deep Dose Equivalent
DEC	Design Extension Condition
D-in-D	Defence-in-Depth
DL	Defence Line
DL1	Defence Line 1

Acronym	Explanation
DL2	Defence Line 2
DL3	Defence Line 3
DL4a	Defence Line 4a
DL4b	Defence Line 4b
DL5	Defence Line 5
DR	Decay Ratio
DSA	Deterministic Safety Assessment
EAB	Exclusion Area Boundary
EPRI	Electric Power Research Institute
EOR	End of Rated Power
ECCS	Emergency Core Cooling System
ES	Emergency Scenario
ESBWR	Economic Simplified Boiling Water Reactor
EX-DBA	Extended Design Basis Analysis
FAP	Forward Action Plan
FF	Flash Fraction
FHA	Fuel Handling Accident
FMCRD	Fine Motion Control Rod Drive
FSF	Fundamental Safety Function
FW	Feedwater
FWLB	Feedwater Line Break
FWPT	Feedwater Pump Trip
GDA	Generic Design Assessment
GEH	GE Hitachi
GNF	Global Nuclear Fuel
HEPA	High Efficiency Particulate Air
I&C	Instrumentation and Control
ICLB	Isolation Condenser Line Break
ICS	Isolation Condenser System
II	Inventory Increase
ILB	Instrument Line Break
IR	Inventory Reduction
LfE	Learning from Experience
LFWH	Loss of Feedwater Heating
LHGR	Linear Heat Generation Rate
LOCA	Loss of Coolant Accident

Acronym	Explanation
LOCV	Loss of Condenser Vacuum
LOPP	Loss of Preferred Power
LPZ	Low Population Zone
LR-TT	Load Rejection or Turbine Trip
LTR	Licensing Topic Report
MAAP	Modular Accident Analysis Program
MCPR	Minimum Critical Power Ratio
MCR	Main Control Room
MCRE	Main Control Room Envelope
MOC	Middle of Cycle
MRBM	Multi-Channel Rod Block Monitor
MIT	Massachusetts Institute of Technology
MSL	Main Steam Line
MSCIV	Main Steam Containment Isolation Valve
MSRIV	Main Steam Reactor Isolation Valve
OLMCPR	Operating Limit Minimum Critical Power Ratio
ONR	Office for Nuclear Regulation
OPEX	Operational Experience
PCT	Peak Clad Temperature
PI	Pressure Increase
PIE	Postulated Initiating Event
PIRT	Phenomenon Identification and Ranking Table
PSA	Probabilistic Safety Assessment
PSR	Preliminary Safety Report
PWR	Pressurized Water Reactor
QA	Quality Assurance
RAI	Request for Additional Information
RB	Reactor Building
RCPB	Reactor Coolant Pressure Boundary
RGP	Relevant Good Practice
RI	Reactivity Increase
RIV	Reactor Isolation Valve
RLC	Reactor Level Control
RPC	Reactor Pressure Control
RPV	Reactor Pressure Vessel
SA	Severe Accident

Acronym	Explanation
SAA	Severe Accident Analysis
SAMG	Severe Accident Management Guideline
SBWR	Simplified Boiling Water Reactor
SCRRI	Select Control Rod Run-In
SMR	Small Modular Reactor
SSC	Structure, System, Component
STP	Simulated Thermal Power
TB	Turbine Building
TBV	Turbine Bypass Valve
TCV	Turbine Control Valves
TD	Temperature Decrease
TRACG	Transient Reactor Analysis Code General Electric
TSV	Turbine Stop Valve
UK	United Kingdom
U.S.	United States
USNRC	U.S. Nuclear Regulatory Commission

DEFINITIONS AND SYMBOLS

Symbol	Definition
χ/Q	Atmospheric Dispersion Factor
ΔCPR/ICPR	Delta Critical Power Range over Initial Critical Power Ratio

TABLE OF CONTENTS

LIST OF TABLES

LIST OF FIGURES

REVISION SUMMARY

Revision #	Section Modified	Revision Summary
A	All	Initial Issuance

INTRODUCTION

Chapter Route Map

This sub-chapter is part of PSR Ch. 15 which presents the BWRX-300 Safety Analysis and comprises the following subchapters:

- PSR Ch. 15.1 – NEDC-34179P, "BWRX-300 UK GDA Ch. 15.1: Safety Analysis General Considerations," (Reference 15.5-1)
- PSR Ch. 15.2 – NEDC-34180P, "BWRX-300 UK GDA Ch. 15.2: Safety Analysis - Identification, Categorisation, and Grouping of Postulated Initiating Events and Accident Scenarios," (Reference 15.5-2)
- PSR Ch. 15.3 – NEDC-34181P, "BWRX-300 UK GDA Ch. 15.3: Safety Analysis - Objectives and Acceptance Criteria," (Reference 15.5-3)
- PSR Ch. 15.4 – NEDC-34182P, "BWRX-300 UK GDA Ch. 15.4: Safety Analysis - Human Actions," (Reference 15.5-4)
- PSR Ch. 15.5 – NEDC-34183P, "BWRX-300 UK GDA Ch. 15.5: Deterministic Safety Analysis," [*this sub-chapter*] (Reference 15.5-5)
- PSR Ch. 15.6 – NEDC-34184P, "BWRX-300 UK GDA Ch. 15.6: Probabilistic Safety Assessment," (Reference 15.5-6)
- PSR Ch. 15.7 – NEDC-34185P, "BWRX-300 UK GDA Ch. 15.7: Deterministic Safety Analyses-Analysis of Internal Hazards," (Reference 15.5-7)
- PSR Ch. 15.8 – NEDC-34186P, "BWRX-300 UK GDA Safety Analysis External Hazards," (Reference 15.5-8)
- PSR Ch. 15.9 – NEDC-34187P, "BWRX-300 UK GDA Summary of Results of the Safety Analysis," (Reference 15.5-9)

This layout mainly follows the structure set out in the IAEA specific safety guide SSG-61 "Format and Content of the Safety Analysis Report for Nuclear Power Plants," (Reference 15.5-52) with the exception that Internal and External Hazards are discussed in two separate subchapters.

Sub-Chapter Structure

This sub-chapter is presenting the BWRX-300 Deterministic Safety Analysis and comprises the following main sections:

- 15.5.1 – General Description of the Approach
- 15.5.2 – Analysis of Normal Operation
- 15.5.3 – Analysis of Anticipated Operational Occurrences (AOOs)
- 15.5.4 – Analysis of Design Basis Accidents (DBAs)
- 15.5.5 – Analysis of Design Extension Conditions (DECs) without Significant Fuel Degradation
- 15.5.6 – Analysis of Design Extension Conditions with Core Melting
- 15.5.7 – Analysis of Postulated Initiating Events and Accident Scenarios Associated with the Spent Fuel Pool
- 15.5.8 – Analysis of Fuel Handling Events

- 15.5.9 – Analysis of Radioactive Releases from a Subsystem or a Component
- 15.5.10 – Analysis of Internal and External Hazards
- 15.5.11 – Deterministic Safety Analysis Results

This layout mainly follows the structure set out in SSG-61 with two exceptions: the analysis of AOOs and DBAs are presented in two separate sections, and an additional section covering fuel handling events.

The separate presentation of AOO and DBA analysis is a reflection of the Defence Line (DL) concept, which is deeply embedded in the engineering development and safety argument.

In addition, there are two appendices which contain UK GDA specific material and future work items. These are discussed below.

This subchapter presents normal operations analysis and fault analysis. The fault analysis is divided into reactor faults and non-reactor faults (i.e., those faults associated with fuel handling and the spent fuel pool).

Reactor faults are classified according to their frequency of occurrence (being divided into AOOs, DBAs, and DECs) and are analysed accordingly, with each different type of analysis being reported in a separate section of this subchapter. DECs without significant fuel degradation and DECs with core melting are analysed and reported in separate sections. The majority of faults do not challenge the fission product barriers and do not lead to release of activity outside containment: no calculation of the radiological consequences is therefore undertaken at the moment. For the smaller number of faults that can lead to the release of activity outside containment, radiological consequence calculations are presented in a separate section.

Each reactor fault is presented in terms of the following topics:

- Postulated Initiating Event
- Sequence of Event
- Identification of Operator Actions
- Systems Operation
- Input Parameters and Initial Conditions
- Results
- Barrier Performance
- Radiological Consequences

The reactor faults and analysis are divided into the following groups based on the high-level effect on the plant:

- Temperature Decreases (TD) events – decreases in core coolant temperature.
- Pressure Increases (PI) events – increases in reactor pressure.
- Reactivity Increases (RI) events – reactivity and power distribution anomalies.
- Inventory Increase (II) events – increases in reactor coolant inventory.
- Inventory Reduction (IR) events – decreases in reactor coolant inventory.

Interfaces with other Chapters

This sub-chapter interfaces with the following PSR Chapters:

- PSR Ch. 3 – NEDC-34165P, "BWRX-300 UK GDA Ch. 3: Safety Objectives and Design Rules for SSCs," (Reference 15.5-10)
- PSR Ch. 4 – NEDC-34165P, "BWRX-300 UK GDA Ch. 4: Reactor," (Reference 15.5-11)
- PSR Ch. 5 – NEDC-34167P, "BWRX-300 UK GDA Ch. 5: Reactor Coolant System and Associated Systems," (Reference 15.5-12)
- PSR Ch. 6 – NEDC-34168P, "BWRX-300 UK GDA Ch. 6: Engineered Safety Features," (Reference 15.5-13)
- PSR Ch. 7 – NEDC-34169P, "BWRX-300 UK GDA Ch. 7: Instrumentation and Control," (Reference 15.5-14)
- PSR Ch. 8 – NEDC-34170P, "BWRX-300 UK GDA Ch. 8: Electrical Power," (Reference 15.5-15)
- PSR Ch. 9A – NEDC-34171P, "BWRX-300 UK GDA Ch. 9A: Auxiliary Systems," (Reference 15.5-16)
- PSR Ch. 10 – NEDC-34174P, "BWRX-300 UK GDA Ch. 10: Steam and Power Conversion Systems," (Reference 15.5-17)
- PSR Ch. 11 – NEDC-34174P, "BWRX-300 UK GDA Ch. 11: Management of Radioactive Waste," (Reference 15.5-18)
- PSR Ch. 12 – NEDC-34175P, "BWRX-300 UK GDA Ch. 12: Radiation Protection," (Reference 15.5-19)
- PSR Ch. 15 – Safety Analysis Analyses (all other subchapters)
- PSR Ch. 16 – NEDC-34188P, "BWRX-300 UK GDA Ch. 16: Operational Limits and Conditions," (Reference 15.5-20)
- PSR Ch. 17 – NEDC-34189P, "BWRX-300 UK GDA Ch. 17: Management for Safety," (Reference 15.5-21)
- PSR Ch. 18 – NEDC-34190P, "BWRX-300 UK GDA Ch. 18: Human Factors Engineering," (Reference 15.5-22)
- PSR Ch. 19 – NEDC-34191P, "BWRX-300 UK GDA Ch. 19: Emergency Preparedness and Response," (Reference 15.5-23)
- PSR Ch. 27 – NEDC-34199P, "BWRX-300 UK GDA Ch. 27: ALARP Evaluation," (Reference 15.5-24)

Purpose

The purpose of this subchapter is to present the preliminary safety arguments for the BWRX-300 in the area of deterministic safety analysis, which forms part of the safety analysis. The primary objective of the overall safety analysis is to demonstrate that the Fundamental Safety Functions (FSFs):

- Control of Reactivity
- Cooling of the Fuel
- Confinement of Radioactive Materials

- Normal Operations
- Design Basis Conditions (AOOs and DBAs)
- Design Extension Conditions (with and without core damage) and demonstrate that the DL functions are effective in meeting the applicable defined acceptance criteria.

Scope

The BWRX-300 standard design is being developed using a phased design process. The set of PIEs identified thus far is presented in the Fault List accompanying 005N3558, "Fault Evaluation Report," (Reference 15.5-51) and is discussed below. A subset of the faults contained therein is presented in this chapter and was selected based on the bounding event evaluation described in Section 15.2.4 of NEDC-34180P (Reference 15.5-2). These are currently restricted to bounding reactor faults at power, supplemented with a bounding Fuel Handling Accident non-reactor fault. The analysis of Design Extension Conditions with core damage and faults associated with the fuel pool is currently out of scope for the deterministic analysis.

Country Specific Material ‑ UK Step 2 GDA

The PSR is being submitted as part of Step 2 of Generic Design Assessment (GDA) by the United Kingdom (UK) Office for Nuclear Regulation (ONR). GDA is an up-front, non-site-specific assessment of a generic nuclear power plant design. It is intended to determine whether a proposed reactor type could be constructed, operated, and decommissioned in Great Britain. Step 2 is a fundamental assessment of the generic safety, security, and environmental protection cases. It is intended to identify potential showstoppers that may preclude deployment of the design.

Specific nuclear safety requirements differ between countries. Significant unique aspects of the UK regulatory regime are the overriding requirement to demonstrate that risks have been managed and reduced to As Low As Reasonably Practicable (ALARP) and clarity on the Claims, Arguments, and Evidence employed in the safety argument. The demonstration of ALARP is typically achieved through the application of Relevant Good Practice (RGP). One element of RGP in the fault studies area is the production of a fault schedule.

The claims and argument's structure appears in Appendix A. The approach to development of the fault schedule appears in PSR Ch. 15.9. Other UK specific aspects are handled through the identification of future work.

Future Work

The need for future work has been identified during the production of this subchapter. This arises principally for the following reasons:

- Continuing design development.
- Development of the preliminary safety arguments.
- Country-specific requirements.

A schedule of Forward Action Plan (FAP) items is presented in Appendix B. Each FAP item comprises a concise description of the required work along with a project phase for when it is needed by; an outline of the reason for raising the FAP is also presented.

15.5 PRESENTATION OF DETERMINISTIC SAFETY ANALYSIS

As discussed in the Introduction, Section 15.5 closely follows the structure set out in SSG-61 with two exceptions: the analysis of AOOs and DBAs are presented in two separate sections, and an additional section covering fuel handling events has been added.

The separate presentation of AOO and DBA analysis is a reflection of the Defence Line (DL) concept, which is deeply embedded in the engineering development and safety argument.

In addition, there are two appendices which contain UK GDA specific material and future work items.

15.5.1 General Description of the Approach

Categorisation of Events and Defence Lines

As described in 006N5064, "BWRX-300 Safety Strategy," (Reference 15.5-50), there are three layers of DSA performed (see NEDC-34179P (Reference 15.5-1) and Section 15.2.1 of NEDC-34180P (Reference 15.5-2) for additional details) that credit different sets of DLs where the bounding event sequence is performed for the transient or non-LOCA and LOCA events.

Plant states and event categories are established as a framework to organise the various safety analyses and application of Defence-in-Depth (D-in-D) across the complete spectrum of possible plant conditions. These are based on frequency of occurrence and consistent with those defined in IAEA SSG-2 "Deterministic Safety Analysis for Nuclear Power Plants," (Reference 15.5-55). The frequencies of occurrence which delineate transitions between event categories are described in NEDC-34180P (Reference 15.5-2, Section 15.2.2), along with the limited exceptions for this categorisation approach.

There are five Defence Lines (DLs). Defence Line 1 (DL1) and Defence Line 5 (DL5) do not include performance of automatic plant control or automatic event mitigation functions. DL1 minimises potential for PIEs to occur in the first place and minimises potential for failures to occur in subsequent DLs. DL5 involves emergency preparedness measures to protect the public in case a substantial radioactive release does occur. Defence Line 2 (DL2), Defence Line 3 (DL3), Defence Line 4a (DL4a) and Defence Line 4b (DL4b) comprise plant functions that act to prevent PIEs from leading to significant radioactive releases. The DL objectives and D-in-D concepts are described further in NEDC-34179P (Reference 15.5-1, Section 15.1.1). 006N5064 (Reference 15.5-50) ensures performance of the Fundamental Safety Functions (FSF), which maintain integrity of the physical barriers to release, by applying these DLs systematically to mitigate PIEs:

- Among DL2, DL3, and DL4a, two independent and diverse DLs can mitigate any PIE with a frequency greater than 1E-05 per reactor-year for PIEs caused by single failures.
- Among DL2, DL3, and DL4a, at least one DL can mitigate any PIE caused by a Common Cause Failure (CCF) in another DL, with the mitigation means being independent from the effects of the initiating CCF.
- DL4b includes provisions for prevention and/or mitigation of severe accidents. These provisions are independent from failed equipment and adverse conditions present in the event sequences where they are necessary.

The effectiveness of the DLs is demonstrated by layered DSA.

Layers of DSA

The BWRX-300 uses a layered analysis approach that includes these types of DSA evaluations: Baseline DSA (BL-DSA), Conservative DSA (CN-DSA), and Extended DSA (EX-DSA). These DSA are performed to demonstrate the effectiveness of DL mitigation by demonstrating that the AOO, DBA, and DEC acceptance criteria are met. The DSA acceptance criteria are described in Section 15.3.2 of NEDC-34181P (Reference 15.5-3).

The BL-DSA is the formal demonstration of performance to acceptance criteria for AOO PIEs, demonstrates the effectiveness of DL2, and is further described in Section 15.2.1.1 of NEDC-34180P (Reference 15.5-2).

The CN-DSA is the formal demonstration of performance to acceptance criteria for DBA PIEs and event sequences, demonstrates the effectiveness of DL3, and is further described in Section 15.2.1.2 of NEDC-34180P (Reference 15.5-2).

The EX-DSA is the formal demonstration of performance to acceptance criteria for DEC PIEs and event sequences without core damage, demonstrates the effectiveness of DL4a for deterministically postulated event sequences and the effectiveness of DL4b for complex sequences, and is further described in Section 15.2.1.3 of NEDC-34180P (Reference 15.5-2).

Severe Accident Analysis is also performed to assess DEC PIEs and event sequences with core damage and is not discussed further in this Chapter.

Fault Evaluation

The role of the BWRX-300 Fault Evaluation is summarized in NEDC-34179P (Reference 15.5-1, Section 15.1.1) and is described in more detail in NEDC-34180P (Reference 15.5-2), 006N5064 (Reference 15.5-50), and 005N3558 (Reference 15.5-51).

The output of the Fault Evaluation is presented in the Fault List, which presents a single, consolidated route map to all scenarios composing the complete set of deterministic safety analyses for the BWRX-300. Throughout the design development process, it is used to support organised iteration between design and analysis teams. It is updated throughout this process to reflect the current state of design and analysis (both DSA and PSA) maturation. In its final state, it will support the efficient validation that all required DL functions have been identified and provide traceability between the DL functions and those analysis cases that establish their performance bases.

Some regulatory regimes may require additional information to be presented in a tabular format. Appendix E in PSR Ch. 15.9 presents proposals for development of a UK type fault schedule. FAP item PSR15.5-28 pertains.

Presentation of the DSA

The DSA is completed in two parts:

- The plant response to event sequences is evaluated and analysed to confirm the performance of the fission product barriers against the derived acceptance criteria.
- The event dose consequences resulting from a fission product release or other source of radiation, such as the reactor coolant, is radiologically analysed.

Deterministic Safety Analysis Approach for Non-LOCA Events

Non-LOCA or Transient DSA analyses event sequences where the Reactor Coolant Pressure Boundary (RCPB) remains intact. These events are broken down into groups that result in similar core responses. Subsection 15.2.4 of NEDC-34180P (Reference 15.5-2) describes the core response during off-normal conditions, the groups determined for BWRX-300 (Table 15.2-1), and the selected bounding event scenarios for AOOs, DBAs and DECs without core damage analyses (summarised in Table 15.2-2).

The BWRX-300 scenarios are identified through the fault evaluation. The methods and assumptions described in NEDC-34043P, Revision 0, “BWRX-300 TRACG Application,” (Reference 15.5-25) are used to confirm the performance of the fission product barriers for the DSA non-LOCA events. The Transient Reactor Analysis Code General Electric (TRACG) application for both non-LOCA and LOCA event analysis is discussed in Subsection 15.5.1.2.1.

The TRACG Application for BWRX-300 also includes the stability analysis that evaluates potential coupled thermal hydraulic – neutronic instabilities in the reactor core. TRACG is used to perform transient safety analysis and stability analysis for both forced flow and natural circulation BWR designs. Previous TRACG applications as well as BWRX-300 use the systematic approach called Code Scaling, Applicability, and Uncertainty (CSAU) to confirm the applicability of a computer code for DSA that conforms to NUREG/CR-5249, "Quantifying Reactor Safety Margins: Application of Code Scaling, Applicability, and Uncertainty Evaluation Methodology to a Large-Break, Loss-of-Coolant Accident," (Reference 15.5-26). This approach involves systematic evaluation of the phenomena that are important for the plant design and accident scenarios identified. A qualitative process is used to identify and rank the importance of phenomena. Through this process, a Phenomenon Identification and Ranking Table (PIRT) is established that conforms with USNRC's RG 1.203, "Transient and Accident Analysis Methods," (Reference 15.5-27). The PIRT is used together with the TRACG documentation to systematically demonstrate the applicability of TRACG models and the qualification of the TRACG model to predict the phenomena. Defining the nodalisation and evaluation of the effects of scale are included. In addition to code applicability and qualification, the PIRT is also used as the basis to perform quantitative uncertainty analysis of transient scenarios, if needed. Additional information regarding the approach for addressing uncertainty in the DSA is provided in Subsection 15.5.1.1.

The TRACG applicability to model phenomena also requires that the code capability be demonstrated to apply the code in the intended manner with a qualifying result achieved. TRACG capability to model phenomena is important to BWRX-300 simulation and is consistent with modern best practices. TRACG qualification is based upon proven practices for verification and validation using acceptable codes and standards. Experiments and plant events used to validate TRACG provide evidence that TRACG can be applied for the BWRX-300 design.

Integral to the capability of TRACG for transient DSA is the use of three-dimensional nuclear kinetics input. This input comes directly from and essentially uses the same methods as the steady state core simulator, PANAC11. PANAC11 is used in the BWRX-300 as described in Subsections 15.5.1.3. Other code interfaces are described in NEDC-34043P (Reference 15.5-25).

Design control procedures require independent verification of safety analysis calculations to ensure that results are properly summarised from calculations, physically sound/correct, and consistent with expected results when compared to previous calculations. The results are then confirmed to meet the appropriate acceptance criteria. Table 15.5-44 provides additional insight in conservatisms used in the transient analysis.

Deterministic Safety Analysis Approach for LOCA Events

The methods and assumptions of the DSA confirming the performance of the fission product barriers for LOCAs are described in Licensing Topical Report (LTR) NEDC-33922P-A, Revision 3, "BWRX-300 Containment Evaluation Method, Revision 3," (Reference 15.5-28).

TRACG calculates the mass and energy release from modelled breaks of various sizes and locations. Atmospheric pressure is used for the TRACG pressure boundary condition for any breaks. This approach provides no credit for the back pressure from containment. Consequently, the retained Reactor Pressure Vessel (RPV) inventory calculated by TRACG represents the minimum coolant volume. This modeling provides results as if the break occurred outside containment.

Breaks inside containment realistically experience back pressure from containment that reduces the mass and energy calculated by TRACG once the break flow becomes unchoked. However, this effect is not treated explicitly because it requires two-way coupling between the TRACG calculation and the GOTHIC containment calculation. Instead, the methodology has a one-way coupling with the mass and energy release rates conservatively calculated by TRACG that supplies inputs to the GOTHIC calculation up until the point in time when the containment and RPV pressures first equalise. Choked flow naturally satisfies the assumed one-way coupling because choked flow does not depend on the downstream pressure. Select TRACG inputs are specified so that mass and energy release rates are conservatively calculated.

Rapid mass and energy releases into containment occur before a large break is isolated. This leads to the highest containment peak pressure at approximately the same time that the break is isolated. For large breaks, the containment shell is the dominant short-term energy sink, and it causes containment pressure to decrease from its peak value after isolation of the break occurs.

Compared to large breaks, small un-isolated breaks have a much slower mass and energy release rate from the RPV into containment. The lowest break on the RPV that remains un-isolated and occurs outside containment produces the most limiting scenario for minimum RPV inventory. Regardless of break location and whether it is inside or outside containment, break flow slowly decreases with time because the RPV is being depressurised largely due to the ICS and to a much lesser extent by the break flow.

Containment pressure slowly increases and eventually equals the RPV pressure for breaks inside containment. It is not realistic to use the TRACG break flow that was calculated using an atmospheric pressure boundary condition as input to the GOTHIC containment calculation after the point in time when containment and RPV pressures first equal each other. A better approximation is to assume zero break flow after this point in time, but this could potentially be nonconservative with respect to the longer-term calculated containment pressures. The proposed methodology does not require the GOTHIC calculation to continue beyond the point where the containment and RPV pressures equalize, because the longer-term containment pressure is bounded by the RPV pressure calculated. The DL3 functions credited in the conservative DBA LOCA analyses and the DL2 and DL4a functions credited in the DEC LOCA analyses are described in Tables 15.5-45 and 15.5-46, respectively. The methods and assumptions for radiological analyses are described for these events in Section 15.5.9.

15.5.1.1 Safety Margins in Safety Analyses

The DSA demonstrates that the challenges to the physical barriers do not exceed their physical capacity.

Uncertainties in initial conditions and methods are accounted for in the CN-DSA using RG 1.203 (Reference 15.5-27). The BL-DSA and EX-DSA are performed using best-estimate methods based on the purpose of the BL-DSA and EX-DSA analyses. For CN-DSA thermal-hydraulic analysis, a graded approach is used combining uncertainties. The graded approach involves a qualitative assessment of the safety margin on a case-by-case basis and includes a review of the magnitude of results compared to acceptance criteria along with the judgment of conservatism in the derived acceptance criteria.

The DSA confirms the FSFs successfully keep plant radioactive material releases within the acceptance criteria with adequate safety margins.

15.5.1.1.1 Large Margin

For events with large margin or substantially non-limiting, there is no need to apply uncertainty to the analysis methodology. Judgment is used to establish what is “large margin” or “substantially non-limiting.” Instead of a quantitative evaluation of uncertainty, the event is dispositioned qualitatively based on the uncertainty evaluation performed for a limiting event of a similar type, historical analysis of similar type, or other qualitative based disposition.

Large Margin Examples

The inadvertent isolation condenser initiation in Subsection 15.5.4.3.2 and the closure of all MSRIVs and FWRIVs in Subsection 15.5.4.2.4 are examples of large margin events. There are many DBA events that have minimal impacts compared to the acceptance criteria.

Inadvertent Isolation Condenser Initiation

This CN-DBA event results in a peak pressure of 7.32 MPaG, much lower than the acceptance criterion of 12.41 MPaG and is bounded by pressure increase CN-DBA events in Subsection 15.5.4.2.4. Also, the Peak Clad Temperature (PCT) is 586.3°F (308.0°C), much lower than the acceptance criterion in Table 15.3-2 and is bounded by the generator load rejection CN-DBA event in Subsection 15.5.4.2.1. In this event, there is no concern for cladding oxidation, and there is no threat to the containment pressure boundary.

Closure of All MS Reactor Isolation Valves and FW Isolation Valves

This CN-DBA event results in a peak pressure of 8.73 MPaG, much lower than the acceptance criterion of 12.41 MPaG. The peak pressure is bounded by other pressure increase CN-DBA events in Subsection 15.5.4.2. Also, the PCT is 594.9°F (312.7°C), much lower than the acceptance criteria in Table 15.3-2. This event has a large safety margin to the DBA acceptance criteria. In this event, there is no concern for cladding oxidation, and there is no threat to the containment pressure boundary. As a result, for the above events and any other events with similar margin, there is no need for any quantification of uncertainty and a qualitative disposition is adequate.

15.5.1.1.2 Medium Margin

In medium margin scenarios, method uncertainty is addressed by biasing important phenomena in a conservative direction (typically one or two sigma). Input parameters such as power, pressure, level, or temperature are based on using the most limiting normal operating values. In these cases, the selection of key phenomena is dependent on the specific event evaluated. Important phenomena can be different for the output parameters when multiple output parameters are considered for selecting the bias direction. The selection of the important phenomena and determining the bounding bias direction is considered for each output parameter that has medium margin and compared to the derived acceptance criteria.

Medium Margin Example

A medium margin event is the large break inside containment described in Subsection 15.5.4.5.

Large Pipe Breaks Inside Containment

There are multiple large pipe break inside containment scenarios evaluated that have commonalities. These events result in no significant fuel cladding heat up and are not bounding with respect to maintaining inventory above the fuel (to ensure continued cooling). They represent the largest challenge to the containment fission product barrier. These events are treated as medium margin events and the initial conditions and modeling parameters are biased to ensure conservative containment conditions are calculated. The initial conditions used are provided in Table 15.5-2. The modeling parameters biased in this analysis are discussed in NEDC-33922P-A (Reference 15.5-28). The combination of the conservative biased inputs with the observation of the margin available results in conservative analyses.

15.5.1.1.3 Low Margin or Quantitative Evaluation of Uncertainty is Desired

A proven Monte Carlo technique is used to combine the individual biases and uncertainties into an overall bias and uncertainty for low margin events. This process is described in the BWRX-300 TRACG Application Report, NEDC-34043P (Reference 15.5-25).

There are no events identified as low margin in this PSR.

15.5.1.2 Description of the Computer Codes or Standards Used in the Safety Analyses

This section concerns the Verification and Validation of computer codes and calculational methods. It is presented at a relatively high level and some regulators and future licensees/operators may require further detail. FAP item PSR15.5-36 pertains.

There is a large amount of data available from operating BWR plants and from the testing and licensing efforts to licence the predecessor BWR/Advanced Boiling Water Reactor (ABWR)/Economic Boiling Water Reactor (ESBWR) designs and individual plants. The vast database of feature performance in licensed reactors, combined with the recent thorough licensing review of the ABWR and ESBWR, provides an extremely well qualified foundation from which to make the modest extrapolations to the BWRX-300. The following codes, methods, and accompanying assumptions are used in evaluating the performance of the BWRX-300. The radiological consequences following DBAs and DECs presented in NEDC-34187P (Reference 15.5-9) are calculated using the RADTRAD radiological consequence code. Short-term atmospheric dispersion factors at the site boundary and control room intakes are calculated using the PAVAN and ARCON computer codes, respectively, once site information is provided.

15.5.1.2.1 TRACG

TRACG is a GEH proprietary version of the Transient Reactor Analysis Code. TRACG is the primary licensing analysis tool for LOCA and transient analyses for PIEs with a large range of frequencies up to events that do not involve significant core damage (severe accidents). TRACG has been used in a variety of applications for operating BWRs as well as design/analysis for the ESBWR.

TRACG uses advanced realistic one-dimensional and three-dimensional methods to model the phenomena that are important in evaluating the operation of BWRs. It is a best-estimate code for analysis of BWR transients ranging from simple operational transients to design basis LOCAs and failure to scram transients. TRACG has an extensive qualification base for separate effects, BWR fuel and components, and integral tests. It has been reviewed and approved by the US NRC for several analysis applications such as AOOs, ECCS LOCA and failure to scram overpressure (a BDBA event for BWRs) analyses and has been reviewed and approved in NEDC-33922P-A (Reference 15.5-28).

TRACG is used to analyse the challenges to the fuel, RPV, and the mass and energy releases to the containment, for LOCA and non-LOCA DSA. TRACG draws from the licensed BWR database, which includes design features of the BWRX-300 (albeit in various configurations) and appropriate testing and allows direct application to BWRX-300 design and analysis. TRACG is maintained and updated by GEH.

Scope and Capabilities

TRACG is based on a multi-dimensional two-fluid model for the reactor thermal hydraulics and a three-dimensional neutron kinetics model. The TRACG output is also used to provide mass and energy release input to the containment analysis code GOTHIC. GOTHIC solves the conservation equations for mass, momentum, and energy for the gas and liquid phases. TRACG does not include any assumptions of thermal or mechanical equilibrium between phases. The gas phase may consist of a mixture of steam and a non-condensable gas, and the liquid phase may contain dissolved boron. The thermal-hydraulic model is a multi-dimensional formulation for the vessel component and a one-dimensional formulation for all other components.

The conservation equations for mass, momentum, and energy are closed through an extensive set of basic models consisting of constitutive correlations for shear and heat transfer at the gas/liquid interface as well as at the wall. The constitutive correlations are flow regime-dependent and are determined based on a single flow regime map, which is used consistently throughout the code. In addition to the basic thermal-hydraulic models, TRACG contains a set of component models for BWR components, such as fuel channels, steam separators, and can simulate BWR steam dryers as part of its vessel model. TRACG also contains a control system model capable of simulating the major BWR control systems such as those for pressure and water level.

The neutron kinetics model is consistent with the GEH BWR core simulator PANACEA. It solves a modified one-group diffusion model with six delayed neutron precursor groups. Feedback is provided from the thermal-hydraulic model to the kinetics model for moderator density, fuel temperature, boron concentration and control rod position.

The TRACG structure is based on a modular approach. The TRACG thermal-hydraulic model contains a set of basic components, such as pipe, valve, tee, channel, steam separator, heat exchanger and vessel. System simulations are constructed using these components as building blocks. Any number of these components may be combined. The number of components, their interaction, and the detail in each component are specified through code input. Consequently, TRACG has the capability to simulate a wide range of facilities, ranging from simple separate effects tests to complete BWR plants.

TRACG has been extensively qualified against separate effects tests, component performance data, integral system effects tests and full-scale BWR plant data. Detailed documentation of the TRACG qualification is contained in NEDO-32177, “TRACG Qualification,” (Reference 15.5-29) report.

The total effort and extent of qualification performed on TRACG, since its inception in 1979 now exceeds, both in extent and breadth, that of any other engineering computer program GE/GEH has submitted to the USNRC for design application approval.

Scope of Application of TRACG to BWRX-300

The PIRT (discussed in References 15.5-26 and 15.5-27) identify specific governing phenomena in predicting BWRX-300 transient and LOCA performance. Most of these phenomena are common to operating BWRs. This section examines specific Simplified Boiling Water Reactor (SBWR)/ESBWR-related tests and test facilities beyond the previous qualification database. Early in the SBWR program, it was identified that there was no information in the data base for a heat transfer correlation for steam condensation in tubes in the presence of non-condensable gases. A Single Tube Condensation Test Program was conducted to secure this information and reported to the USNRC in TRACG Qualification Report for SBWR (NEDC-32725P, Revision 1) and ESBWR (NEDC-33080P-A, Revision 2) that are used in the TRACG model for the BWRX-300 as described in NEDC-33922P-A (Reference 15.5-28).

The test program was conducted to investigate steam condensation inside tubes in the presence of non-condensable gases. The work was independently conducted at the University of California at Berkeley and at the Massachusetts Institute of Technology (MIT). The work was initiated to obtain a database and a correlation for heat transfer and condensation inside tubes. Three researchers utilised three separate experimental configurations at the University of California at Berkeley, while two researchers utilised one configuration at MIT. The researchers ran tests with pure steam, steam/air, and steam/helium mixtures with representative and bounding flow rates and non-condensable mass fractions. The researchers found the system well-behaved for all tests, with either of the non-condensable gases. The results of the tests at the University of California at Berkeley are the basis for the condensation heat transfer correlation used in the TRACG computer code.

TRACG ICS modeling is qualified by the PANTHERS IC test using a representative configuration. The steady state heat exchanger performance was predicted by the PANTHERS IC prototypical geometry full-scale test.

Because the BWRX-300 RPV and ICS are similar to those of the ESBWR, the TRACG method developed for the ESBWR RPV thermal hydraulics and mass energy release is also used for the BWRX-300 RPV thermal hydraulics and mass and energy release. The TRACG code and the application method developed for ESBWR was reviewed and approved by the USNRC. That application method was developed using the Code Scaling, Applicability and Uncertainty (CSAU) guidance. NEDC-33922P-A (Reference 15.5-28), provides an overview of the TRACG thermal hydraulics method for the mass and energy release and its applicability to the BWRX-300 RPV.

15.5.1.2.2 GOTHIC

Containment analysis is performed by using the GOTHIC "Thermal Analysis Package Qualification Report," (Reference 15.5-30) code.

The GOTHIC computer code is a state-of-the-art program for modeling multiphase, multicomponent fluid flow for performing both containment DBA analyses and analyses to support equipment qualification. The GOTHIC code is developed by Numerical Applications Incorporated, and the development program is sponsored by the Electric Power Research Institute (EPRI).

The GOTHIC code has a node structure that allows both lumped parameter and Three-Dimensional (3D) modeling capabilities. The multidimensional analysis capability facilitates the study of non-condensable gas and stratification and the calculation of flow field details within any given volume. The code has undergone extensive review and validation against a large test array. The validation program scope examines the code capability for predicting pressure and temperature as well as hydrogen distribution and mixing under various conditions.

GOTHIC is a continuously maintained and improved computer code. The GOTHIC code has been developed in compliance with the requirements of 10 CFR 50 Appendix B Quality Assurance requirements (Reference 15.5-46) and GEH's software quality requirements. The PSR results were generated using GOTHIC Version 8.3 which is the latest released version. Future BWRX-300 containment analyses may be performed using newer versions of the GOTHIC code.

15.5.1.2.3 PANAC11

The BWR Core Simulator (PANAC11 or P11) is described in PSR Ch. 4 and is a steady-state, 3D coupled nuclear-thermal-hydraulic computer program representing the BWR core exclusive of the external flow loop.

The analytical methods used in the design and analysis of the BWRX-300 are described in detail in NEDC-34039P, "BWRX-300 GNF2 Steady State Nuclear Methods TGBLA06/PANAC11 Application Methodology," (Reference 15.5-60).

15.5.1.2.4 ANSI/ANS-18.1-2020 Standard

Radiation concentrations in the reactor coolant and steam during normal operations are determined based on American National Standards Institute (ANSI)/American Nuclear Society (ANS)-18.1-2020, "Radioactive Source Term for Normal Operation of Light Water Reactors," (Reference 15.5-33). This standard provides the bases for estimating typical concentrations of the principal radionuclides that may be anticipated over the lifetime of a BWR plant. The source term data is based on the cumulative industry experience at operating BWR plants, including measurements at several stations. The operating data reflects the influence of several observations made during the transition period from operation with fuel of older designs to operation with fuel of current improved designs such as the GNF2 fuel used in the BWRX-300.

15.5.1.2.5 PAVAN

Note the following code applies to analysis of site-specific aspects, and therefore does not apply to the extant UK PSR.

The NUREG/CR-2858, "PAVAN: An Atmospheric-Dispersion Program for Evaluating Design-Basis Accidental Releases of Radioactive Materials from Nuclear Power Stations," code (Reference 15.5-34) used to calculate offsite atmospheric dispersion factors using meteorological data (i.e., wind speed, wind direction, and measurements of atmospheric stability) recorded at the site, and PAVAN uses Joint Frequency Distributions of wind direction, wind speed, and atmospheric stability class to estimate χ/Q values for specific averaging time periods at specified distances. The PAVAN model is based on a straight-line Gaussian model that assumes the release rate is constant for the entire release period. PAVAN calculations implement the models in USNRC RG 1.145, "Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants," (Reference 15.5-47).

The RG 1.145 methods and the PAVAN code calculations incorporate the results of several field tracer tests and diffusion experiments for elevated and ground-level releases from various locations on reactor facility buildings during stable atmospheric conditions with low wind speeds. PAVAN calculated χ/Q values are made for either assumed ground-level releases (e.g., through building penetrations and vents) or elevated releases from free-standing stacks. The dispersion theories applied, user instructions, code algorithms, FORTRAN source code, and test cases for the PAVAN code are documented in NUREG/CR-2858, (Reference 15.5-34).

15.5.1.2.6 RADTRAD

The dose consequences of postulated DBAs are calculated using the RADTRAD Version 3.10 computer code NUREG/CR-6604, "RADTRAD: A Simplified Model for Radionuclide Transport and Removal and Dose Estimation," (Reference 15.5-35). The RADTRAD code was developed by the Accident Analysis and Consequence Assessment Department at Sandia National Laboratories for the USNRC, Office of Nuclear Reactor Regulation, Division of Reactor Program Management.

RADTRAD is a simplified model for Radionuclide Transport and Removal and Dose estimation.

The RADTRAD code uses a combination of tables and numerical models of source term reduction phenomena to determine the time-dependent dose at specified locations both onsite and offsite for a given accident scenario. The RADTRAD code is used to assess occupational radiation time in cycles, typically in the control room and site boundaries. RADTRAD code is capable of estimating dose attenuation due to modification of a facility or accident sequence. RADTRAD is a licensing analysis code used to show compliance with nuclear plant siting criteria for the radiation doses at onsite and offsite locations for various LOCA and non-LOCA DBAs. As radioactive material is transported through the containment, the user can account for sprays and natural deposition that may reduce the quantity of radioactive material. Material flow between buildings, from buildings to the environment, or into control rooms through High Efficiency Particulate Air (HEPA) filters, piping, or other connectors is modelled. An accounting of radioactive material amounts retained due to these tortuous pathways is maintained. Decay and in-growth of daughter isotopes can be calculated over time as the material is transported.

15.5.1.2.7 ARCON

ARCON, NUREG/CR-6331, Revision 1, “Atmospheric Relative Concentrations in Building Wakes,” (Reference 15.5-36) is a computer code used to calculate atmospheric relative concentrations (X/Q) in support of control room habitability assessments required by 10 CFR Part 50, Appendix A, General Design Criterion 19. ARCON uses hourly meteorological data and the atmosphere’s influence (i.e., dilution and dispersion) in the vicinity of buildings to calculate the relative concentration at control room air intakes. These concentrations would be exceeded no more than five percent of the time and calculated for averaging periods ranging from one hour to 30 days in duration.

The model is based on a straight-line Gaussian model that assumes the release rate is constant for the entire period of the release. ARCON can account for both plume meander under low wind speed conditions and the plume dispersion due to building wakes. An expanded description of ARCON code bases, capabilities, and limitations is documented in USNRC Regulatory Guide 1.194, “Atmospheric Relative Concentrations for Control Room Radiological Habitability Assessments at Nuclear Power Plants,” (Reference 15.5-48).

15.5.1.2.8 ORIGEN2

ORIGEN2, Version 2.1, “Isotope Generation and Depletion Code – Matrix Exponential Method,” (Reference 15.5-37) is a computer code system for calculating the buildup, decay, and processing of radioactive materials. ORIGEN2 is typically used for estimating core inventories of nuclides for light water reactors at various stages of power operation.

15.5.1.2.9 MAAP

The BWRX-300 Severe Accident (SA) conditions and radioactive releases are evaluated using the EPRI MAAP5-Modular Accident Analysis Program for LWR Power Plants, Transmittal Document for MAAP5 Code Revision 5.06 (Reference 15.5-49). MAAP5 is a computer code developed by the Electric Power Research Institute (EPRI) used by nuclear utilities and research organisations to predict the progression of light water reactor accidents.

MAAP5 is used to analyse reactor thermal-hydraulic and containment response to transients as well as SAA sequence progressions. MAAP5 is used to predict the timing of key events, evaluate the influence of mitigative systems, evaluate effectiveness of operator actions, predict magnitude and timing of fission product releases, and investigate uncertainties in SA phenomena. It calculates the progression of the postulated accident sequence, including the deposition of the fission products, from initiating events to either a safe, stable state or to an impaired containment condition (due to overpressure or overtemperature). The software also calculates the amount of fission product released to the environment.

The BWRX-300 plant level system model in MAAP5 will account for all necessary flow volumes and structural heat sinks, to best represent the plant thermal-hydraulic response to a SA scenario. Accommodation of the important structural heat sinks in the primary and secondary systems (containment, reactor building) ensures that due credit is given to fission product structural interaction mechanisms and the source terms are accordingly evaluated in the best-estimate manner.

15.5.2 Analysis of Normal Operation

Part of normal operational analysis is to confirm that the core will remain stable during normal operation. The stability considerations during normal operation and AOOs are described in the Thermal Hydraulics Summary Report (Refence 15.5-58).

The BWRX-300 240-bundle core is evaluated for Beginning of Cycle (BOC), Middle of Cycle (MOC) and End of Rated Power (EOR) time in cycles in determining both the core-wide decay ratios and regional mode oscillations.

Core-Wide Decay Ratio

The core-wide Decay Ratios (DR) are evaluated using a step perturbation in pressure while the regional mode oscillations are evaluated using channel velocity perturbations. The primary stability evaluation is performed at nominal conditions including a nominal FW temperature of 467.4°F (241.9°C).

Another stability evaluation is performed for the state that is reached after a Loss of Feedwater Heating (LFWH) AOO analysis as described in Subsection 15.5.3.1.1. A Select Control Rod Run-In (SCRRI) (described in PSR Ch. 7 (Reference 15.5-14)) is initiated as a mitigating response for a LFWH AOO. The analysis of LFWH AOO described in Subsection 15.5.3.1.1 assumes that the FW temperature is reduced to 377.4°F (191.9°C) at BOC, MOC and EOR. Once the reactor achieves a new steady-state condition, a step pressure perturbation is applied to evaluate core stability response.

Core-Wide Dominance

For the regional mode evaluation, based on the harmonic modes distribution of the core, the inlet velocities for all channels were perturbed by ±20% at time = 0. This harmonic power distribution is predicted by the steady state core simulator PANAC11 and results in a line of symmetry between the two halves of the core with higher and lower predicted harmonic power. The velocity perturbations are made positive on one side of the line of symmetry and negative on the other side. This stimulates the potential harmonic oscillations (regional oscillations). The resulting channel power response of limiting channels is evaluated for susceptibility to regional mode oscillations. If the core is not susceptible to regional mode oscillations after a velocity perturbation, the initially symmetric, out of phase channel power responses come into phase after a short duration, confirming the dominance of core-wide oscillations.

Results

DR/SCRRI

The maximum nominal core-wide DR design limit is 0.80. The calculated core-wide DRs at nominal conditions are below the maximum DR allowed. The calculated DR values at the end of the LFWH event are below the maximum allowed DR and are lower than the DR values at nominal conditions. The calculated DR values at nominal temperature and LFWH conditions are presented in Table 15.5-3. The nominal stability response is presented in Figure 15.5-3 for the MOC time in cycle. The LFWH stability response is presented in Figure 15.5-4 for the MOC time in cycle.

Core Wide Dominance

A limiting core-wide dominance evaluation is performed at 115% rated power. At MOC, the limiting channel power values follow the expected behaviour where initially symmetric, out of phase channel power responses come into phase after a short duration. The core is not susceptible to regional mode oscillations at nominal conditions, and this conclusion also applies to normal operation. The regional stability response using the limiting channels is presented in Figure 15.5-5 at the MOC time in cycle. This is a hypothetical evaluation, and any growing core-wide oscillations are mitigated by DL3-05 (see Table 15.5-6).

Based on these stability evaluations, the following stability claims are supported:

1. Power oscillations that result in conditions exceeding specified acceptable fuel design limits are not possible
2. Regional instability is not possible

3. Design features prevent the loss of stability margin for upset events

15.5.3 Analysis of Anticipated Operational Occurrences

This section concerns the analysis of Anticipated Operational Occurrences. Bounding events have been analysed and presented. Some regulators and future licensees / operators may require further analysis to be presented to assist in the safe management of specific events, or to explicitly demonstrate that the presented events are bounding. FAP item PSR15.5-30 pertains. Decoupling criteria have been used to judge the adequacy of the reactor's response to the AOOs. Some regulatory regimes may require the analysis to be extended out from the demonstration of meeting the decoupling criteria (e.g., no cladding failure) to the achievement of a stable, safe state. FAP item PSR15.5-38 pertains. The treatment of frequent faults and therefore the approach to Anticipated Transients Without Scram (ATWSs) may differ between regulatory regimes. FAP item PSR15.5-29 pertains.

15.5.3.1 Decrease in Core Coolant Temperature AOO

This section describes the bounding BL-AOO event for the Temperature Decrease (TD) Group.

15.5.3.1.1 Loss of Feedwater Heating AOO

This event is designated as a BL-AOO event. The event sequence name is LFWH, and the event sequence ID is TD-LFWH_BL-AOO.

Additional LFWH AOO cases support the detailed evaluation and demonstrate the thermal hydraulic stability of the BWRX after a LFWH AOO.

Postulated Initiating Event

The event assumes a LFWH from a single failure of either the closure of one extraction steam valve or the inadvertent bypass of a FW heater. This failure is conservatively modelled as an instantaneous decrease in FW temperature that bounds the maximum FW TD resulting from a single failure. A PIE with AOO frequency results in the maximum FW temperature reduction identified in Table 15.5-4 from a loss of one FW heater.

Sequence of Event

The event sequence comprises in summary:

- Reduction in FW temperature occurs instantly resulting in an increase in power
- Reactor Level Control (RLC) compensates initially by lowering flow rate, minimising the effect on power
- SCRRI inserts control rods on indication of FW temperature reduction
- Reactor Pressure Control (RPC) maintains pressure and RLC maintains level
- A new controlled steady-state condition is achieved with a new power distribution

Table 15.5-7 lists the event sequence.

Identification of Operator Actions

No operator action is required to mitigate the event.

Systems Operation

Credited DL2 functions:

- DL2-27 – SCRRI on FW TD
- DL2-02 – Maintain Target Level

- DL2-01 – Maintain Target Pressure

Core and System Performance

Input Parameters and Initial Conditions

The event is simulated by decreasing the FW temperature. The initial conditions and plant parameters are provided in Tables 15.5-4 through 15.5-6. The analysis is performed using an equilibrium core design. The event is run at BOC, MOC and EOR cycle time in cycle conditions.

Results

The results of the simulated loss of FW heating event are presented in Figures 15.5-6 through 15.5-11. The results are presented in Table 15.9-2 for the time in cycle with the limiting Critical Power Ratio (CPR) response.

The reduced temperature FW enters the core and causes an increase in core inlet subcooling. This increases core power due to the negative void reactivity coefficient. RLC compensates initially by lowering FW flow rate and a SCRRI is initiated to minimise the core power increase and decrease the final steady state power. Steam flow and FW flow then stabilise at a lower level. The RPV water level decreases and then returns to normal level. The pressure and level remain well within the RPV water level and RCPB pressure acceptance criteria in Table 15.3-1.

The core power increase is limited. Thermal-mechanical evaluations confirm there is significant margin to centreline fuel temperature or cladding strain acceptance criteria in Table 15.3-1. Limits on the Linear Heat Generation Rate (LHGR) are included each operating cycle ensuring the centreline fuel temperature and cladding strain acceptance criteria are met. The limits on LHGR are included in the Core Operating Limits Report (COLR).

The calculated Delta Critical Power Ratio Over Initial Critical Power Ratio (ΔCPR/ICPR) is provided. This is used to set an Operating Limit Minimum Critical Power Ratio (OLMCPR) ensuring the CPR remains within the Minimum Critical Power Ratio (MCPR) acceptance criterion. This event is potentially limiting for OLMCPR and is evaluated each operating cycle to determine the core OLMCPR. The resulting limiting event OLMCPR is included in the COLR.

Sensitivity studies were performed on maximum FW pump flow (120% of rated), initial FW temperature-6°C, and FW controller settings. The sensitivity studies demonstrated no significant change in the event sequence or results.

Barrier Performance

The effect of this event does not result in any temperature or pressure transient more than the derived acceptance criteria for the fuel, pressure vessel, or containment. Therefore, these barriers maintain their integrity and function as designed.

Radiological Consequences

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

15.5.3.2 Increase in Reactor Pressure AOOs

15.5.3.2.1 Generator Load Rejection or Turbine Trip AOO

This event is in the Pressure Increase (PI) Group and is designated a BL-AOO event. The event sequence name is Generator Load Rejection or Turbine Trip (LR-TT), and the Event Sequence ID is PI-LR-TT_BL-AOO.

Postulated Initiating Event

The initiating event is either a generator load rejection or a turbine trip. Turbine Control Valves (TCVs) have a fast closure function to protect the turbine during a generator load rejection. The Turbine Stop Valves (TSVs) close at a fast rate following a turbine trip. RPC remains unaffected and demands the Turbine Bypass Valves (TBVs) open to control reactor pressure.

Sequence of Event

The event sequence comprises in summary:

- TCVs and/or TSVs close quickly causing pressure to increase
- Anticipatory scram occurs on indication of a load rejection or turbine trip
- RPC opens TBVs to control pressure
- RLC maintains level
- Controlled state achieved

Table 15.5-8 lists the event sequence.

Identification of Operator Actions

No operator action is required to mitigate the event.

Systems Operation

Credited DL2 functions:

- DL2-02 – Maintain Target Level
- DL2-01 – Maintain Target Pressure
- DL2-08 – Anticipatory Hydraulic Scram on Generator Load Rejection or Turbine Trip Demand
- DL2-09 – TBV Fast Open on Generator Load Rejection or Turbine Trip Demand

Core and System Performance

Input Parameters and Initial Conditions

The event is simulated by initiating a generator load rejection or turbine trip resulting in a fast closure of the TCVs or TSVs. The initial conditions and plant parameters are provided in Tables 15.5-4 through 15.5-6.

The analysis is performed using an equilibrium core design. The event is run at BOC, MOC, and EOR cycle time in cycle conditions.

Results

The simulated generator load rejection / turbine trip is presented in Figures 15.5-12 through 15.5-17, and the results are presented in Table 15.9-1. The results are shown for the case with the limiting CPR result. Automatic reactor scram occurs following indication of a generator load rejection or turbine trip. Pressure increases but is limited by the TBVs opening.

The core thermal power does not increase above the initial power and there is no concern in approaching the centreline fuel temperature or cladding strain acceptance criteria in Table 15.3-1. This event is not limiting and is not considered during LHGR limits development.

The calculated ΔCPR/ICPR is provided. This is used to set an OLMCPR ensuring the CPR remains within the MCPR Acceptance Criterion. This event is potentially limiting for OLMCPR and is evaluated each operating cycle for determining the core OLMCPR. The resulting limiting event OLMCPR is included in the COLR.

Barrier Performance

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

Radiological Consequences

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

15.5.3.2.2 Closure of One Main Steam Reactor Isolation Valve AOO

This event is in the PI Group and is designated as a BL-AOO event. The event sequence name is Closure of one MSRIV and the Event Sequence ID is PI-1MSRIVC_BL-AOO.

Postulated Initiating Event

There are two Main Steam Lines (MSLs). The event is an inadvertent closure of one MSRIV that terminates flow in one of the MSLs. A minimum MSRIV closure time results in the most severe event.

Sequence of Event

The event sequence comprises in summary:

- One MSRIV closes causing RPV pressure and power to increase
- Anticipatory Trip System (ATS) scram occurs on MSRIV position
- RLC controls levels
- Second MSRIV in the second steam line closes on leak detection indication (this is assumed because it makes the event more severe)
- One ICS train initiates on high RPV pressure (the simulation is ended before any ICS initiation because the key mitigation DL functions have already been demonstrated and a single ICS train can control pressure and removing decay heat as demonstrated in the pressure increase DBA analysis)
- Controlled state achieved

Table 15.5-9 lists the event sequence.

Identification of Operator Actions

No operator action is required to mitigate the event.

Systems Operation

Credited DL2 functions:

- DL2-02 – Maintain Target Level
- DL2-21 – Anticipatory Hydraulic Scram on MSRIV/Main Steam Containment Isolation Valve (MSCIV) Position
- DL2-31 – ICS Pressure Control on High Reactor Pressure

Core and System Performance

The event is simulated by initiating a closure of one MSRIV. The initial conditions and plant parameters are provided in Tables 15.5-4 through 15.5-6.

Input Parameters and Initial Conditions

The analysis is performed using an equilibrium core design. The event is run at BOC, MOC, and EOR cycle time in cycle conditions.

Results

The simulated closure of one MSRIV is presented in Figures 15.5-18 through 15.5-23, and the results are presented in Table 15.9-1. The results are shown for the case with the limiting CPR and reactor pressure response. The neutron flux and PI resulting from the closure of one MSRIV are limited by an anticipatory scram on MSRIV position. The PI is also initially limited because the MSRIV remains open in the second steam line. The second steam line is then assumed to close on MSL break indication. This conservative assumption makes the event more severe. Pressure then increases and ICS is initiated on high RPV pressure.

The core thermal power increase is not significant and there is no concern for approaching the centreline fuel temperature or cladding strain acceptance criteria in Table 15.3-1. This event is not limiting and does not need to be considered during development of limits on the LHGR.

The calculated ΔCPR/ICPR is provided. This is used to set an OLMCPR ensuring the CPR remains within the MCPR acceptance criterion. This event is potentially limiting for OLMCPR and is evaluated each operating cycle for determining the core OLMCPR. The resulting limiting event OLMCPR is included in the COLR.

Barrier Performance

This event does not result in any temperature or pressure transient more than the derived acceptance criteria for the fuel, pressure vessel, or containment. Therefore, these barriers maintain their integrity and function as designed.

Radiological Consequences

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

15.5.3.2.3 Loss of Condenser Vacuum AOO

This event is in the PI Group and is designated a BL-AOO event. The event sequence name is Loss of Condenser Vacuum (LOCV), and the Event Sequence ID is PI-LOCV_BL-AOO.

Postulated Initiating Event

There are a few potential causes of a LOCV including loss of one or more circulating water pumps. The LOCV results in a turbine trip. The TSVs close at a fast rate following a turbine trip.

Sequence of Event

The event sequence comprises in summary:

- The TSVs close and main turbine trips on low main condenser vacuum causing pressure increase
- Anticipatory scram occurs on a turbine trip
- RPC opens TBVs to control pressure
- RLC maintains level
- TBVs close on high main condenser pressure and pressure increases slowly due to decay heat (the simulation is ended before TBV closure because the key mitigation DL

functions are demonstrated, and a single ICS train is capable of controlling pressure and removing decay heat)

- One ICS train initiates on high RPV pressure (the simulation is ended before any ICS initiation because the key mitigation DL functions have already been demonstrated and a single ICS train is capable of controlling pressure and removing decay heat as demonstrated in the PI DBA analysis)
- Controlled state achieved

Table 15.5-10 lists the event sequence.

Identification of Operator Actions

No operator action is required to mitigate the event.

Systems Operation

Credited DL2 functions:

- DL2-02 – Maintain Target Level
- DL2-01 – Maintain Target Pressure
- DL2-13 – Turbine Trip on High Main Condenser Pressure Setpoint 2
- Anticipatory Hydraulic Scram on Either:
  - DL2-08 – Generator Load Rejection or Turbine Trip Demand
  - DL2-37 – High Main Condenser Pressure Setpoint 1
- DL2-09 – TBV Fast Open on Generator Load Rejection or Turbine Trip Demand
- DL2-14 – TBV Closure on High Main Condenser Pressure Setpoint 3
- DL2-31 – ICS Pressure Control on High Reactor Pressure

Core and System Performance

Input Parameters and Initial Conditions

The LOCV results in a turbine trip. The event is simulated by initiating a turbine trip resulting in a fast TSV closure. The initial conditions and plant parameters are provided in Tables 15.5-4 through 15.5-6.

The analysis is performed using an equilibrium core design. The event is run at BOC, MOC and EOR cycle time in cycle conditions.

Results

The simulated loss of condenser vacuum is presented in Figure 15.5-24 through 15.5-29 and the results are presented in Table 15.9-1. The results are shown for the case with the limiting CPR result. The pressure increases due to the fast TSV closure and is limited by the anticipatory scram on indication of a turbine trip. A scram on high main condenser pressure may occur sooner but is conservatively not modelled. The PI is also initially limited by the TBVs opening. Condenser vacuum loss is assumed to continue, resulting in TBVs closing. Once TBVs are closed, reactor pressure increases, and one ICS train initiates on high RPV pressure.

The core thermal power does not increase above the initial power and there is no concern for approaching the centreline fuel temperature or cladding strain acceptance criteria in Table 15.3-1. This event is not limiting and is not considered during LHGR limits development.

The calculated ΔCPR/ICPR is provided. This is used to set an OLMCPR ensuring the CPR remains within the MCPR acceptance criterion. This event is potentially limiting for OLMCPR and is evaluated each operating cycle for determining the core OLMCPR. The resulting limiting event OLMCPR is included in the COLR.

Barrier Performance

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

Radiological Consequences

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

15.5.3.2.4 Loss of Preferred Power AOO

This event is in the PI Group and is designated a BL-AOO event. The event sequence name is Loss-of-Preferred Power (LOPP), and the Event Sequence ID is PI-LOPP_BL-AOO.

Postulated Initiating Event

A LOPP is initiated by offsite power supply failure. The loss of power results in the generator output breakers opening and the TCVs fast closure.

Sequence of Event

The event sequence comprises in summary:

- LOPP occurs
- TCVs close quickly causing PI
- FW pumps lose power, FW pump discharge check valves maintain coolant inventory
- Circulating water pumps lose power
- Anticipatory scram occurs on generator load rejection
- RPC opens TBVs to control pressure
- TBVs close on LOPP
- One ICS train initiates on high RPV pressure (the simulation is ended before any ICS initiation because the key mitigation DL functions have already been demonstrated and a single ICS train can control pressure and removing decay heat as demonstrated in the pressure increase DBA analysis)
- Controlled state achieved

Table 15.5-11 lists the event sequence.

Identification of Operator Actions

No operator action is required to mitigate the event.

Systems Operation

Credited DL2 functions:

- DL2-01 – Maintain Target Pressure
- Anticipatory Hydraulic Scram on Either:
  - DL2-08 – Generator Load Rejection or Turbine Trip Demand

  - DL2-26 – Low Electric Bus Voltage
- DL2-09 – TBV Fast Open on Generator Load Rejection or Turbine Trip Demand
- DL2-31 – ICS Pressure Control on High Reactor Pressure
- DL2-43 – FW Check Valve Closure on Reverse FW Flow

Core and System Performance

Input Parameters and Initial Conditions

The LOPP results in the generator output breakers opening and a loss of power to the FW pumps. The event is simulated by initiating a FW trip and a load rejection resulting in fast TCVs closure. Anticipatory scram occurs on a generator load rejection. A scram on low bus voltage may occur sooner but is conservatively not modelled. This scram timing is the same as the anticipatory scram on a load rejection. The initial conditions and plant parameters are provided in Tables 15.5-4 through 15.5-6.

The analysis is performed using an equilibrium core design. The event is run at BOC, MOC, and EOR cycle time in cycle conditions. The results are shown for the case with the limiting result for CPR.

Results

The simulated LOPP is presented in Figure 15.5-30 through 15.5-35 and the results are presented in Table 15.9-1. The results are shown for the case with the limiting result for CPR. The PI due to TCV closure is limited by the anticipatory scram on the generator load rejection. Scram on a low electric bus voltage may occur sooner but is conservatively not credited. The PI is also initially limited by the TBVs opening. The TBVs later close on loss of power. Once the TBVs are closed, reactor PIs and ICS initiates on high RPV pressure. The ICS continues to limit the PI.

The core thermal power increase is not significant and there is no concern for approaching the centreline fuel temperature or cladding strain acceptance criteria in Table 15.3-1. This event is not limiting and does not need to be considered during development of limits on the LHGR.

The calculated ΔCPR/ICPR is provided. This is used to set an OLMCPR ensuring the CPR remains within the MCPR acceptance criterion. This event is potentially limiting for OLMCPR and is evaluated each operating cycle for determining the core OLMCPR. The resulting limiting event OLMCPR is included in the COLR.

Barrier Performance

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

Radiological Consequences

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

15.5.3.3 Decrease in Reactor Coolant Inventory AOO

15.5.3.3.1 Feedwater Pump Trip – One Pump

The section analyses the bounding BL-AOO event for the Inventory Reduction (IR) group. The event sequence name is FW Pump Trip – One Pump (FWPT) and the Event Sequence ID is IR-FWPT_BL-AOO.

Postulated Initiating Event

There is one FW pump normally operating and a second FW pump in standby. This event assumes a failure resulting in a trip of the operating FW pump. The RLC remains unaffected by the failure and increases the flow demand on the standby FW pump to maintain RPV water level.

Sequence of Event

The event sequence comprises in summary:

- One FW pump trips causing RPV water level decrease
- Standby FW pump starts and increases to rated FW flow
- Power decreases temporarily from a reduction in core flow and core inlet subcooling
- RPC maintains pressure
- RLC maintains level
- RPV water low level scram and high-level FW isolation are avoided
- Controlled state achieved

Table 15.5-12 lists the event sequence.

Identification of Operator Actions

No operator action is required to mitigate the event.

Systems Operation

Credited DL2 functions:

- DL2-25 – Start Standby FW pump on Loss of Operating FW pump
- DL2-02 – Maintain Target Level
- DL2-01 – Maintain Target Pressure

Core and System Performance

Input Parameters and Initial Conditions

The event is simulated by initiating a FW pump trip. The initial conditions and plant parameters are provided in Tables 15.5-4 through 15.5-6.

The analysis is performed using an equilibrium core design. The event is run at BOC, MOC, and EOR cycle time in cycle conditions.

Results

The FWPT event is presented in Figures 15.5-36 through 15.5-41 and the results are presented in Table 15.9-1 for the time in cycle with the limiting CPR response. Reduction in FW flow results in a reduction of vessel inventory, causing the vessel water level to drop. The standby FW pump starts on confirmed low FW flow conditions, the RPC throttles TCVs to control pressure, and the RLC increases the FW pump flow to rated conditions to maintain level. Low reactor water level (L3) scram is avoided.

The core thermal power does not increase and there is no concern in approaching the centreline fuel temperature or cladding strain acceptance criteria in Table 15.3-1. This event is not limiting and is not considered during development of LHGR limits.

The calculated ΔCPR/ICPR is provided. The event is not limiting and is not considered in the OLMCPR development.

Barrier Performance

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

Radiological Consequences

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

15.5.3.4 Increase in Reactor Coolant Inventory AOO

15.5.3.4.1 Inadvertent Isolation Condenser Initiating – One Train

This event is designated as a BL-AOO event. The event sequence ID is II-IICI-1_BL-AOO. This event assumes a failure causes a single Isolation Condenser (IC) condensate return valve to open. The event assumes that RLC remains unaffected by the failure and is able to maintain level. The event also assumes that RPC remains unaffected by the failure and is able to maintain pressure.

Postulated Initiating Event

The ICs are normally in standby mode. This event assumes spurious opening of a single IC condensate return valve, resulting in the introduction of cold water into the reactor. The event assumes that RLC and RPC remain unaffected by the failure and are available to control reactor level and reactor pressure.

Sequence of Event

The event sequence comprises in summary:

- ICS condensate return valve on one train opens
- Cold ICS condensate water drains into the chimney
- RLC maintains level
- RPC maintains pressure
- Controlled state achieved

Table 15.5-13 lists the event sequence.

Identification of Operator Actions

No operator action is required to mitigate the event.

Systems Operation

The credited DL2 functions:

- DL2-02 – Maintain Target Level
- DL2-01 – Maintain Target Pressure

Core and Systems Performance

Input Parameters and Initial Conditions

The event is simulated by opening the IC condensate return valve on one ICS train. The initial conditions are provided in Tables 15.5-4 through 15.5-6.

The analysis is performed using an equilibrium core design. The event is run at BOC, MOC, and EOR cycle time in cycle conditions.

Results

The inadvertent initiation of one ICS train event is presented in Figures 15.5-42 through 15.5-47. Table 15.9-1 shows the limiting results for CPR response. When the IC condensate return valve is opened, cold water is introduced into the chimney region. After an initial small perturbation, the increased density in the chimney reduces core flow, water level, and power temporarily. After the initial reduction, core flow, level, and power increase. When the increase in level is sensed, the FW controller starts to demand the operating FW pump to reduce flow. After the initial surge, as condensate water drains into the chimney and IC flow reduces, the FW controller demands the operating FW pump to increase flow. The RPV water level, core flow, and core power settle back to their initial values. RPV pressure increases insignificantly. The level and pressure remain well within the RPV water level and RCPB pressure acceptance criteria in Table 15.3-1.

The core power increase is limited. There is no concern for approaching the centreline fuel temperature or cladding strain acceptance criteria in Table 15.3-1. This event is not limiting and does not need to be considered during development of limits on the LHGR.

The calculated ΔCPR/ICPR is provided. This is used to set an OLMCPR ensuring the CPR remains within the MCPR acceptance criterion. This event is potentially limiting for OLMCPR and is evaluated each operating cycle to determine the core OLMCPR. The resulting limiting event OLMCPR is included in the COLR.

Barrier Performance

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

Radiological Consequences

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

15.5.4 Analysis of Design Basis Accidents

This section concerns the analysis of Design Basis Accidents. Bounding events have been analysed and presented. Some regulators and future licensees / operators may require further analysis to be presented to assist in the safe management of specific events, or to explicitly demonstrate that the presented events are bounding. FAP item PSR15.5-30 pertains. Decoupling criteria have been used to judge the adequacy of the reactor's response to the DBAs. Some regulatory regimes may require the analysis to be extended out from the demonstration of meeting the decoupling criteria (e.g., no cladding failure) to the achievement of a stable, safe state. FAP item PSR15.5-38 pertains. The treatment of frequent faults and therefore the approach to Anticipated Transients Without Scram (ATWSs) may differ between regulatory regimes. FAP item PSR15.5-29 pertains.

This section evaluates the bounding BWRX-300 non-LOCA and LOCA PIEs. Subsections 15.5.4.1 through 15.5.4.4 describe the DSA non-LOCA DBAs, while Subsection 15.5.4.5 describes the DSA LOCAs inside containment. Subsection 15.5.9.1 describes the DSA for LOCAs outside containment.

15.5.4.1 Decrease in Reactor Coolant Temperature Design Basis Accident

15.5.4.1.1 Loss of All Feedwater Heating

This event is in the TD group and is designated a CN-DBA event. The event sequence name is CCF-LFWH, Passive Digital CCF DL2 Technology Platform (CCF-DL2), and the Event Sequence ID is TD-CCF-LFWH_CCF-DL2_CN-DBA.

Postulated Initiating Event

A CCF results in the loss of all FW heating. Any CCF that results in the loss of all FW heating occurs gradually because of the thermal inertia inherent in the FW heaters. The FW temperature lowers to the main condenser temperature with the assumed time constant shown in Table 15.5-4.

Sequence of Event

The event sequence comprises in summary:

- CCF results in the loss of all FW heating
- FW temperature decreases causing positive reactivity insertion
- RLC and RPC fail as-is
- Scram on high Simulated Thermal Power (STP) causing negative reactivity insertion
- RPV pressure decreases. The downcomer level decreases temporarily to lower than L3 because of void collapse
- MS isolation occurs on low RPV pressure
- RLC (FW) continues at initial flow causing RPV water level to increase
- FW isolation occurs on high RPV water level
- An ICS initiates on high RPV pressure. The first ICS train fails to actuate (assumed single failure). One of the two remaining ICS trains sufficiently controls pressure (the simulation is ended before any ICS initiation because the key mitigation DL functions have already been demonstrated and a single ICS train can control pressure and remove decay heat as demonstrated in the pressure increase DBA analysis).
- Controlled state achieved

Table 15.5-14 lists the event sequence.

Identification of Operator Actions

No operator action is required to mitigate the event.

Systems Operation

Credited DL3 functions:

- DL3-05 – Hydraulic Scram on High Simulated Thermal Power
- DL3-23 – FW Isolation on High RPV Water Level
- DL3-17 – MSRIV/MSCIV Isolation on Low RPV Pressure
- DL3-12 – ICS Train 2 Initiation on High RPV Pressure

Core and System Performance

Input Parameters and Initial Conditions

The event is simulated by decreasing the FW temperature. The initial conditions and plant parameters are provided in Tables 15.5-4 through 15.5-6. The initial CPR is conservatively biased low, and the hot rod power is conservatively biased high.

The analysis is performed using an equilibrium core design. The event is run at BOC, MOC, and EOR cycle exposure conditions.

Results

The simulated loss of all FW heating event is presented in Figures 15.5-48 through 15.5-53, and the results are presented in Table 15.9-2 for the time in cycle with the limiting PCT response. The reduced temperature FW enters the core and causes an increase in core inlet subcooling. This increases core power due to the negative void reactivity coefficient, and a scram occurs on high Simulated Thermal Power (STP). MSRIV isolates on low RPV pressure.

FW flow remains at 100% due to the RLC CCF and RPV water level rises until FW isolates on high RPV water level. Decay heat causes RPV pressure to rise and an ICS train initiate. Only one ICS train is needed to prevent further RPV pressure increase and maintain long-term cooling. A single failure of an ICS train starting on high RPV pressure does not affect event mitigation.

The long-term core cooling capability is assured by meeting the acceptance criteria for fuel cladding and the RCPB. The reactor integrity is assured by meeting the pressure criteria provided in Table 15.3-2. The results demonstrate significant margin to the acceptance criteria. The cladding temperature remains well below the temperature at which significant oxidation occurs due to metal water reaction.

Because there is significant margin to the acceptance criteria, this event is considered to have large margin and no additional conservatism is applied.

Barrier Performance

This event does not result in any temperature or pressure transient in excess of the derived acceptance criteria for the fuel, pressure vessel, or containment. No fuel failures occur because there is no significant cladding temperature increase. Therefore, these barriers maintain their integrity and function as designed.

Radiological Consequences

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

15.5.4.2 Increase in Reactor Pressure Design Basis Accidents

15.5.4.2.1 Generator Load Rejection or Turbine Trip

This event is in the PI group and is designated a CN-DBA event. The event sequence name is Generator Load Rejection or Turbine Trip (LR-TT), Passive Digital CCF DL2 Technology Platform (CCF-DL2) and the Event Sequence ID is PI-LR-TT_CCF-DL2_CN-DBA.

Postulated Initiating Event

The PIE is the same as the BL-AOO event. The event sequence assumes a passive CCF of the DL2 functions. The CCF results in RPC and RLC, which are continually operating, failing as-is and failure of the anticipatory scram.

Sequence of Event

The event sequence comprises in summary:

- TCVs and/or TSVs close quickly causing pressure and power increase
- RLC fails as-is at initial condition, the TBVs remain closed, and anticipatory scram fails
- Scram occurs on high neutron flux
- After scram, no immediate challenge to cladding and RCPB integrity
- RPV pressure continues to increase because RPC fails as-is
- RPV water level reduces due to the PI

- One ICS train initiates on high RPV pressure. First ICS train fails to actuate (assumed single failure). One of the two remaining trains is sufficient to control pressure
- With RLC failing as-is, initial FW flow continues causing RPV water level to increase
- FW isolates on high RPV water level
- Controlled state achieved

Table 15.5-15 lists the event sequence.

Identification of Operator Actions

No operator action is required to mitigate the event.

Systems Operation

Credited DL3 functions:

- DL3-04 – Hydraulic Scram on High neutron flux
- DL3-23 – FW Isolation on High RPV Water Level
- DL3-12 – ICS Train 2 Initiation on High RPV Pressure

Core and System Performance

Input Parameters and Initial Conditions

The event is simulated by initiating a fast closure of the TCVs or TSVs. The initial conditions and plant parameters are provided in Tables 15.5-4 through 15.5-6. This event is limiting for PCT.

Therefore, additional initial conditions and phenomena that impact PCT described in NEDC-34043P (Reference 15.5-25) are biased in the limiting direction by at least one standard deviation consistent with the approach for an event with medium margin:

- Core void coefficient
- Channel interfacial shear
- Chimney interfacial shear
- Separator steam carry under
- Critical quality used in boiling transition correlation
- Channel radial peaking factor
- Hot rod power
- Total initial power

Analysis is performed as needed to confirm the conservative PCT direction. Then a bounding case is created with all inputs in the conservative direction. The analysis is performed using an equilibrium core design. The event is run at BOC, MOC, and EOR cycle time in cycle conditions.

Results

The simulated generator load rejection / turbine trip is presented in Figures 15.5-54 through 15.5-59 and the results are presented in Table 15.9-2. The results are presented for the bounding case described above. Reactor scram occurs following high neutron flux. The neutron flux increases rapidly because of the void reduction caused by the PI. TBVs fail to open; however, initiation of ICS limits the PI. Only one ICS train is needed to prevent further

PI and maintain long-term cooling. A single failure of an ICS train to start on high RPV pressure does not affect the event mitigation.

The long-term core cooling capability is assured by meeting acceptance criteria for fuel cladding and RCPB. The reactor integrity is assured by meeting the pressure criteria, provided in Table 15.3-2. The results demonstrate significant margin to the acceptance criteria. The cladding temperature remains below the temperature at which significant oxidation occurs due to metal water reaction.

Because this event is considered to have medium margin to acceptance criteria, additional conservatisms are applied as described above to create a bounding case. The results from applying these conservatisms have little effect on peak pressure because peak pressure is primarily driven by the ICS initiation setpoint. Once the pressure setpoint is reached, ICS initiates, and pressure rapidly reduces. For PCT, these conservatisms result in a PCT 91°C (163°F) higher than the base case with margin similar to other DBA analyses. The results are within the acceptance criteria.

Barrier Performance

The effect of this event does not result in any challenge to the temperature or pressure transient derived acceptance criteria for the fuel, pressure vessel, or containment. No fuel failures occur because there is no significant cladding temperature increase. Therefore, these barriers maintain their integrity and function as designed.

Radiological Consequences

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

15.5.4.2.2 Loss of Preferred Power

This event is in the PI group and is designated a CN-DBA event. The event sequence name is LOPP, and the Event Sequence ID is PI-LOPP_CCF-DL2_CN-DBA.

Postulated Initiating Event

The PIE is the same as the BL-AOO event. The event sequence assumes a passive CCF of the DL2 functions. The CCF results in RPC and RLC failing as-is at the initial condition, which are continually operating, and failure of the anticipatory scram.

Sequence of Event

The event sequence comprises in summary:

- TCV closes slowly due to loss of turbine control hydraulic pumps
- FW pumps lose power and coast down
- RLC fails as-is at the initial condition, TBVs remain closed, and the anticipatory scram fails
- Scram occurs on high neutron flux
- After scram, no immediate challenge to cladding and RCPB integrity
- RPV pressure continues to increase because TBVs remain closed
- An ICS train initiates on high RPV pressure. The first ICS train fails to actuate (assumed single failure). One of the remaining two trains is sufficient to control pressure
- Controlled state achieved

Table 15.5-16 lists the event sequence.

Identification of Operator Actions

No operator action is required to mitigate the event.

Systems Operation

Credited DL3 functions:

- DL3-04 – Hydraulic Scram on High Neutron Flux
- DL3-12 – ICS Train 2 Initiation on High RPV Pressure
- DL3-39 – FW Isolation on Loss of Normal FW Flow

Core and System Performance

Input Parameters and Initial Conditions

The event is simulated by initiating a slow closure of the TCVs and FWPT. The initial conditions and plant parameters are provided in Tables 15.5-4 through 15.5-6. The initial CPR is conservatively biased low, and the hot rod power is conservatively biased high.

The analysis is performed using an equilibrium core design. The event is run at BOC, MOC, and EOR cycle time in cycle conditions.

Results

The simulated LOPP is presented in Figures 15.5-60 through 15.5-65, and the results are presented in Table 15.9-2. The results are shown for the case with the limiting result for cladding temperature and reactor pressure response. The neutron flux increases rapidly because of the void reduction caused by the PI. Reactor scram occurs following high neutron flux. TBVs fail to open; however, the PI is limited by the ICS initiation. Only one ICS train is needed to prevent PI and maintain long-term cooling. Therefore, a single failure of an ICS train to start on high RPV pressure does not affect the event mitigation. The closure of the TCVs in the LOPP AOO is due to DL2 active mitigation. In the LOPP CN-DBA sequence, the TCVs close because of the PIE. If there is no power to maintain hydraulic pressure, the valves slowly close.

The long-term core cooling capability is assured by meeting the acceptance criteria for fuel cladding and RCPB. The reactor integrity is assured by meeting the pressure criteria provided in Table 15.3-2. The results demonstrate significant margin to the acceptance criteria. The cladding temperature remains well below the temperature at which significant oxidation occurs due to metal water reaction.

Because there is significant margin to the acceptance criteria, this event is considered to have large margin and no additional conservatism is applied.

Barrier Performance

The effect of this event does not result in any temperature or pressure transient challenge to the derived acceptance criteria for the fuel, pressure vessel, or containment. No fuel failures occur because there is no significant cladding temperature increase. Therefore, these barriers maintain their integrity and function as designed.

Radiological Consequences

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

15.5.4.2.3 RPV Pressure Control Downscale

This event is in the PI group and is designated a CN-DBA event. The event sequence name is CCF – RPV Pressure Control Downscale (CCF-RPCD), Passive Digital CCF DL2

Technology Platform (CCF-DL2) and the Event Sequence ID is PI-CCF-RPCD_CCF-DL2_CN-DBA.

Postulated Initiating Event

The PIE is a spurious CCF of the RPV pressure control. This failure results in a demand to close the TCVs (normal servo closure). This PIE also prevents the TBVs from opening. The event sequence assumes a passive CCF of the DL2 functions. The CCF results in the RLC failing as-is at initial conditions, and failure of the anticipatory scram.

Sequence of Event

The event sequence comprises in summary:

- RPC demands TCVs to slow close and the TBVs remain closed
- RLC fails as-is at the initial condition and the anticipatory scram fails
- Scram occurs on high neutron flux
- After scram, no immediate challenge to cladding and RCPB integrity
- With RLC failing as-is at the initial condition, the initial FW flow continues causing RPV water level to increase
- FW isolates on high RPV water level
- One ICS train initiates on high RPV pressure. The first ICS train fails to actuate (assumed single failure). One of the two remaining ICS trains is sufficient to control pressure
- Controlled state achieved

Table 15.5-17 lists the event sequence.

Identification of Operator Actions

No operator action is required to mitigate the event.

Systems Operation

Credited DL3 functions:

- DL3-04 – Hydraulic Scram on High Neutron Flux
- DL3-23 – FW Isolation on High RPV Water Level
- DL3-12 – ICS Train 2 Initiation on High RPV Pressure

Core and System Performance

Input Parameters and Initial Conditions

The event is simulated by initiating a slow closure of the TCVs. The initial conditions and plant parameters are provided in Tables 15.5-4 through 15.5-6. The initial CPR is conservatively biased low, and the hot rod power is conservatively biased high.

The analysis is performed using an equilibrium core design. The event is run at BOC, MOC, and EOR cycle time in cycle conditions.

Results

The simulated RPV Pressure Control Downscale is presented in Figures 15.5-66 through 15.5-71, and the results are presented in Table 15.9-2. The results are shown with the limiting result for cladding temperature and reactor pressure response. The neutron flux increases rapidly because of the void reduction caused by the PI. Reactor scram occurs following high neutron flux. TBVs fail to open, and FW pump trips on high RPV water level. Natural circulation continues at a rate consistent with decay heat power. Only one ICS train is needed to prevent PI and maintain long-term cooling. Therefore, a single failure of an ICS train to start on high RPV pressure does not affect the event mitigation.

The long-term core cooling capability is assured by meeting the acceptance criteria for fuel cladding and RCPB. The reactor integrity is assured by meeting the pressure criteria provided in Table 15.3-2. The results demonstrate significant margin to the acceptance criteria. The cladding temperature remains well below the temperature at which significant oxidation occurs due to metal water reaction.

Because there is significant margin to the acceptance criteria, this event is considered to have large margin and no additional conservatism is applied.

Barrier Performance

The effect of this event does not result in any temperature or pressure transient in excess of the derived acceptance criteria for the fuel, pressure vessel, or containment. No fuel failures occur because there is no significant cladding temperature increase. Therefore, these barriers maintain their integrity and function as designed.

Radiological Consequences

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

15.5.4.2.4 Closure of All Main Steam Reactor Isolation Valves and FW Isolation Valves

This event is in the PI group and is designated a CN-DBA event. The event sequence name is CCF- Closure of All MSRIVs and FW isolation valves (CCF-DL4a-MSRIVC-FWIV) and the Event Sequence ID is PI-CCF-DL4a-MSRIVC-FWIV_CN-DBA.

Postulated Initiating Event

The PIE is a spurious CCF DL4a function that affects all MSRIVs and FW isolation valves.

Sequence of Event

The event sequence comprises in summary:

- Closure of all MSRIVs and FW isolation valves
- Scram occurs on high neutron flux
- After scram, no immediate challenge to cladding and RCPB integrity
- An ICS train initiates on high RPV pressure. The first ICS train fails to actuate (assumed single failure). One of the two remaining trains is sufficient to control pressure
- Controlled state achieved

Table 15.5-18 lists the event sequence:

Identification of Operator Actions

No operator action is required to mitigate the event.

Systems Operation

Credited DL3 functions:

- DL3-04 – Hydraulic Scram on High Neutron Flux

- DL3-12 – ICS Train 2 initiation on High RPV Pressure

Core and System Performance

Input Parameters and Initial Conditions

The event is simulated by initiating a closure of the MSRIVs and FW isolation valves. The initial conditions and plant parameters are provided in Tables 15.5-4 through 15.5-6. The initial CPR is conservatively biased low, and the hot rod power is conservatively biased high.

The analysis is performed using an equilibrium core design. The event is run at BOC, MOC, and EOR cycle time in cycle conditions.

Results

The simulated closure of MSRIVs and FW isolation valves is presented in Figures 15.5-72 through 15.5-77, and the results are presented in Table 15.9-2. The results are shown for the limiting case for cladding temperature and reactor pressure response. Reactor scram occurs following high neutron flux. The PI is limited by ICS initiation. Only one ICS train is needed to prevent PI and maintain long-term cooling. A single failure of an ICS train to start on high RPV pressure does not affect event mitigation.

The long-term core cooling capability is assured by meeting the acceptance criteria for fuel cladding and RCPB. The reactor integrity is assured by meeting the pressure criteria provided in Table 15.3-2. The results demonstrate significant margin to the acceptance criteria. The cladding temperature remains well below the temperature at which significant oxidation occurs due to metal water reaction.

Because there is significant margin to the acceptance criteria, this event is considered to have large margin and no additional conservatism is applied.

Barrier Performance

The effect of this event does not result in any temperature or pressure transient that challenges the derived acceptance criteria for the fuel, pressure vessel, or containment. No fuel failures occur because there is no significant cladding temperature increase. Therefore, these barriers maintain their integrity and function as designed.

Radiological Consequences

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

15.5.4.3 Increase in Reactor Coolant Inventory DBAs

This event group is in the Inventory Increase (II) group.

15.5.4.3.1 Feedwater Flow Increase – All Pumps

This event is designated as a CN-DBA event. The event sequence name is Feedwater Flow Increase – All Pumps (CCF-FWFI), Passive Digital CCF DL2 Technology Platform (CCF-DL2) and the Event Sequence ID is II-CCF_FWFI_CCF-DL2_CN-DBA.

Postulated Initiating Event

One FW pump is normally operating and a second FW pump in standby. The RLC adjusts the pump speed to adjust FW flow to maintain RPV water level. This event assumes a spurious CCF that causes both FW pumps to increase flow to maximum speed that results in the maximum FW flow. Although not possible, the increase in flow is assumed to occur instantaneously.

Sequence of Event

The event sequence comprises in summary:

- Both FW pumps increase to maximum flow causing RPV water level increase
- RPC remains as-is at initial condition
- Level, pressure, and power increase
- Automatic FW isolation on high RPV water level
- Scram on high simulated thermal power
- RPV pressure and level decrease
- MS isolation on low RPV pressure
- An ICS train initiates on high RPV pressure. The first ICS train fails to actuate (assumed single failure). One of the two remaining trains is sufficient to control pressure and remove decay heat as demonstrated in the pressure increase DBA analysis (the simulation is ended before any ICS initiation because the key mitigation DL functions have already been demonstrated).
- Controlled state achieved

Table 15.5-19 lists the event sequence.

Identification of Operator Actions

No operator action is required to mitigate the event.

Systems Operation

The credited DL3 functions:

- DL3-05 – Hydraulic Scram on High Simulated Thermal Power
- DL3-23 – FW Isolation on High RPV Water Level
- DL3-17 – MSRIV/MSCIV Isolation on Low RPV Pressure
- DL3-12 – ICS Train 2 Initiation on High RPV Pressure

Core and System Performance

Input Parameters and Initial Conditions

The event is simulated by increasing both FW pumps flow to the maximum speed, resulting in the maximum FW flow. The initial conditions and total maximum flow for both FW pumps are provided in Tables 15.5-4 through 15.5-6. The initial CPR is conservatively biased low, and the hot rod power is conservatively biased high.

The analysis is performed using an equilibrium core design. The event is run at BOC, MOC, and EOR cycle conditions.

Results

The simulated maximum FW pumps flow event is presented in Figures 15.5-78 through 15.5-83, and the results are presented in Table 15.9-2. The results are shown for the case with the limiting PCT response results. The increase in FW flow causes reactor level, pressure, and power to increase as RPC and RLC are unavailable. FW isolation occurs on high reactor level. Scram occurs on high STP. Pressure decreases until MSRIV isolation occurs on low RPV pressure. Pressure then increases and one ICS train is initiated on high RPV pressure. Only one ICS train is needed to prevent RPV pressure from increasing and maintaining long-term cooling. A single failure of an ICS train to start on high RPV pressure does not affect the event mitigation.

The long-term core cooling capability is assured by meeting the acceptance criteria for fuel cladding and RCPB. The reactor integrity is assured by meeting the pressure criteria provided in Table 15.3-2. The results demonstrate significant margin to the acceptance criteria. The cladding temperature remains well below the temperature at which significant oxidation occurs due to metal water reaction.

Because there is significant margin to the acceptance criteria, this event is considered to have large margin and no additional conservatism is applied.

Barrier Performance

The effect of this event does not result in any temperature or pressure transient in excess of the design criteria for the fuel, pressure vessel, or containment. No fuel failures occur because there is no significant cladding temperature increase. Therefore, these barriers maintain their integrity and function as designed.

Radiological Consequences

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

15.5.4.3.2 Inadvertent Isolation Condenser Initiating – All Trains

This event is designated as a CN-DBA event. The event sequence name is Inadvertent Isolation Condenser Initiation – All Trains (CCF-DL4a-IICI), Passive Digital CCF DL2 Technology Platform (CCF-DL2) and the Event Sequence ID is II-CCF-IICI_CCF-DL2_CN-DBA.

Postulated Initiating Event

The ICS is normally in standby mode. This event assumes a spurious CCF that causes all IC condensate return valves to open, resulting in introducing cold water into the reactor.

Sequence of Event

The event sequence comprises in summary:

- All IC condensate return valves open
- RLC and RPC fail as-is at the initial condition
- Cold ICS condensate water drains into the chimney
- Core flow, reactor pressure, and power decrease
- RPV water level increases due to RLC failing as-is
- FW isolation occurs on high RPV water level
- Scram occurs on low RPV pressure
- MS isolation occurs on low RPV pressure
- Controlled state achieved

Table 15.5-20 lists the event sequence.

Identification of Operator Actions

No operator action is required to mitigate the event.

Systems Operation

The credited DL3 functions:

- DL3-23 – FW Isolation on High RPV Water Level
- DL3-02 – Hydraulic Scram on Low RPV Pressure
- DL3-17 – MSRIV/MSCIV Isolation on Low RPV Pressure

Core and System Performance

Input Parameters and Initial Conditions

The event is simulated by opening all IC condensate return valves in one second. The initial conditions are provided in Tables 15.5-4 through 15.5-6. The initial CPR is conservatively biased low, and the hot rod power is conservatively biased high.

The analysis is performed using an equilibrium core design. The event is run at BOC, MOC, and EOR cycle time in cycle conditions.

Results

The inadvertent initiation of all ICS trains event is presented in Figures 15.5-84 through 15.5-89, and the results are presented in Table 15.9-2. The results are shown for the case with the limiting PCT response results. When the IC condensate return valves are opened, cold water is introduced into the chimney region, reducing reactor power, reactor pressure, and core flow. Reactor scram and MSRIV isolation initiation occurs on low RPV pressure. After an initial reduction in reactor level, level begins to rise as RLC is unavailable. FW isolation occurs on high reactor level.

The long-term core cooling capability is assured by meeting the acceptance criteria for fuel cladding and RCPB. The reactor integrity is assured by meeting the RCPB pressure criteria. The acceptance criteria are provided in Table 15.3-2. The results demonstrate significant margin to the acceptance criteria. The cladding temperature remains well below the temperature where significant oxidation occurs due to metal water reaction.

Sensitivity studies were performed at the ICS minimum initial temperature in Table 15.5-4 and for an assumed increase in ICS return line volume of 50%. There was no significant change in the event sequence or results. Because there is significant margin to the acceptance criteria, this event is considered to have large margin and no additional conservatism is applied.

Barrier Performance

The effect of this event does not result in any temperature or pressure transient in excess of the derived acceptance criteria for the fuel, pressure vessel, or containment. No fuel failures occur because there is no significant cladding temperature increase. Therefore, these barriers maintain their integrity and function as designed.

Radiological Consequences

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

15.5.4.4 Decrease in Reactor Coolant Inventory Design Basis Accidents

This event group is in the IR group.

15.5.4.4.1 Loss of Feedwater Flow

This event is designated as a CN-DBA event. The event sequence name is CCF Loss of FW Flow (CCF-LOFW), Passive Digital CCF DL2 Technology Platform (CCF-DL2) and the Event Sequence ID is IR-CCF-LOFW _CCF-DL2_CN-DBA.

Postulated Initiating Event

The event sequence assumes a spurious CCF causes the loss of all FW flow and a passive CCF of the DL2 function results in a fails as-is RPC.

Sequence of Events

The event sequence summary:

- Loss of FW flow causes RPV water level and power to decrease
- RPV pressure decreases due to frozen RPC
- Scram occurs on low RPV water level
- FW isolation on a loss of normal FW flow indication
- MS isolation on low RPV pressure
- All ICS trains initiate on low RPV water level
- Controlled state achieved

Table 15.5-21 lists the event sequence.

Identification of Operator Actions

No operator action is required to mitigate the event.

Systems Operation

Credited DL3 functions:

- DL3-03 – Hydraulic Scram on Low RPV Level
- DL3-17 – MSRIV/MSCIV Isolation on Low RPV Pressure
- DL3-14 – ICS Initiation on Low RPV Water Level
- DL3-39 – FW Isolation on Loss of Normal FW Flow

Core and System Performance

Input Parameters and Initial Conditions

The event is conservatively simulated by initiating a trip of all FW pumps. The initial conditions and plant parameters are provided in Tables 15.5-4 through 15.5-6. The initial CPR is conservatively biased low, and the hot rod power is conservatively biased high. The analysis is performed using an equilibrium core design. The event is run at BOC, MOC, and EOR cycle time in cycle conditions.

Results

The loss of FW flow is presented in Figures 15.5-90 through 15.5-95, and the results are presented in Table 15.9-2 for the time in cycle with the limiting PCT response. The trip of all FW pumps results in a reduction of vessel inventory, causing the pressure and vessel water level to drop. Reactor scram occurs on low RPV water level. FW isolates on loss of normal FW flow.

The MSRIVs close on low RPV pressure. RPV water level continues to decrease until ICS initiates. Three ICS trains are modelled to open. Only one ICS train is needed to prevent RPV pressure increase and maintain long-term cooling. A single failure of an ICS train to start on low RPV water level will not affect event mitigation.

The long-term core cooling capability is assured by meeting the acceptance criteria for fuel cladding and RCPB. The reactor integrity is assured by meeting the RCPB pressure criteria.

The acceptance criteria are provided in Table 15.3-2. The results demonstrate significant margin to the acceptance criteria. The cladding temperature remains well below the temperature where significant oxidation occurs from metal water reaction.

Because there is significant margin to the acceptance criteria, this event is considered to have large margin and no additional conservatism is applied.

Barrier Performance

This event does not result in any temperature or pressure transient in excess of the derived acceptance criteria for the fuel, pressure vessel, or containment. No fuel failures occur because there is no significant cladding temperature increase. Therefore, these barriers maintain their integrity and function as designed.

Radiological Consequences

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

15.5.4.4.2 Reactor Pressure Vessel Pressure Control Open

This event is designated as a CN-DBA event. The event sequence name is RPV Pressure Control Open (CCF-RPCO), Passive Digital CCF DL2 Technology Platform (CCF-DL2) and the Event Sequence ID is IR-CCF-RPCO_CCF-DL2_CN-DBA.

Postulated Initiating Event

The event assumes all TCVs and TBVs are fully opened by a spurious RPC CCF. The event sequence assumes a passive CCF of the DL2 functions resulting in the RLC failing as-is.

Sequence of Events

The event sequence comprises in summary:

- All TCVs and TBVs open causing RPV pressure and power to decrease
- FW flow remains at 100% due to DL2 CCF
- Reactor scram and MS isolation on low RPV pressure
- FW isolates on high RPV water level
- An ICS train initiates on high RPV pressure. The first ICS train fails to actuate (assumed single failure). One of the two remaining trains is sufficient to control pressure and removes decay heat as demonstrated in the PI DBA analysis (the simulation is ended before any ICS initiation because the key mitigation DL functions have already been demonstrated).
- Controlled state achieved

Table 15.5-22 lists the event sequence.

Identification of Operator Actions

No operator action is required to mitigate the event.

Systems Operation

Credited DL3 functions:

- DL3-02 – Hydraulic Scram on Low RPV Pressure
- DL3-17 – MSRIV/MSCIV Isolation on Low RPV Pressure
- DL3-23 – FW Isolation on High RPV Water Level

- DL3-12 – ICS Train 2 Initiation on High RPV Pressure

Core and System Performance

Input Parameters and Initial Conditions

The event is simulated by fully opening all TCVs and TBVs. The initial conditions and plant parameters are provided in Tables 15.5-4 through 15.5-6. The initial CPR is conservatively biased low, and the hot rod power is conservatively biased high.

The analysis is performed using an equilibrium core design. The event is run at BOC, MOC, and EOR cycle time in cycle conditions.

Results

The simulated RPV Pressure Control Open event is presented in Figures 15.5-96 through 15.5-101, and the results are presented in Table 15.9-2 for the time in cycle with the limiting PCT response. The opening of the TCVs and TBVs results in a decrease in reactor pressure causing voids to increase and power to decrease. FW remains at 100% rated flow due to the RLC CCF. The reactor scrams and MS isolates on low RPV pressure. RPV water level rises until FW isolates on high RPV water level. RPV pressure then rises due to decay heat. An ICS train initiates on high RPV pressure after the first ICS train fails to actuate (assumed single failure). One ICS train sufficiently controls pressure.

The long-term core cooling capability is assured by meeting the acceptance criteria for fuel cladding and RCPB. The reactor integrity is assured by meeting the RCPB pressure criteria. The acceptance criteria are provided in Table 15.3-2. The results demonstrate significant margin to the acceptance criteria. The cladding temperature remains well below the temperature where significant oxidation occurs due to metal water reaction.

Because there is significant margin to the acceptance criteria, this event is considered to have large margin and no additional conservatism is applied.

Barrier Performance

The effect of this event does not result in any temperature or pressure transient in excess of the derived acceptance criteria for the fuel, pressure vessel, or containment. No fuel failures occur because there is no significant cladding temperature increase. Therefore, these barriers maintain their integrity and function as designed.

Radiological Consequences

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

15.5.4.5 Loss of Coolant Accidents Design Basis Accidents

The scenarios for LOCA developed in Subsection 15.2.5.6 of NEDC-34180P (Reference 15.5-2) bound the CN-DBA sequences, demonstrating the fuel and containment integrity acceptance criteria are met for at least 72-hours using only passive heat removal systems.

The LOCA method used in containment analyses is described in NEDC-33922P-A (Reference 15.5-28). The initial conditions and the modeling parameters are biased to account for uncertainties. The DLs credited in the conservative LOCA breaks inside containment are identified in Table 15.5-45.

The design of the PCCS and the ICS is discussed in PSR Ch. 6 - Engineered Safety Features (Reference 15.5-13).

Meeting the acceptance criteria for fuel integrity is demonstrated by showing that level does not fall below the TAF, or fuel cladding temperature does not exceed the fuel cladding temperature during normal operating conditions.

As discussed in Subsection 15.2.5.6 of NEDC-34180P (Reference 15.5-2), all large breaks are isolated rapidly (10 seconds). Therefore, RPV inventory loss does not threaten fuel integrity in a large break LOCA. After RPV isolation, decay heat is removed by the ICs directly from the RPV. The limiting parameters for large break LOCA events are containment pressure and temperature. Containment peak pressure reaches its peak value at approximately the time of RPV isolation.

The LOCA analyses demonstrate the core remains covered or fuel cladding temperature remains below the normal operating temperature for at least 72-hours using conservative assumptions for un-isolated small break LOCAs. Therefore, fuel cladding temperature remains well below the fuel acceptance criteria, oxidation does not occur, and there is no hydrogen generation from cladding oxidation.

15.5.4.5.1 Main Steam Pipe Break Inside Containment, Conservative Case

Postulated Initiating Event

A break in the MS pipe occurs at an arbitrary location between the outside RIV, and the containment penetration. The most limiting break is the double-ended instantaneous guillotine break of the MS pipe.

The break flow occurs from both ends of the break. Break flow from the RPV side of the break is choked at the flow limiter inside the main steam RPV nozzle. The TSVs and TCVs are conservatively assumed to close rapidly because this results in retaining more energy in containment. With this assumption, steam flows from the RPV to the steam line header upstream of the closed TSV/TCV through the intact steam pipe and to the break location in the reverse direction to the normal flow through the broken steam pipe. Steam flow from the RPV is choked at the flow limiters in both the broken and intact steam pipes. Steam inventory in the steam pipe also discharges into the containment without being restricted at the flow limiters. Containment Isolation Valve (CIV) closure is conservatively not credited, and the entire steam line volume inventory contributes to containment pressurisation.

Sequence of Event

The bounding scenario analysed:

- Double-ended guillotine rupture of MSL break inside containment concurrent with LOPP
- FWPT and coast down
- TSVs and TCVs close rapidly
- Scram initiated from high containment pressure
- Control rods start to insert
- ICS condensate return valve starts opening
- CUW stops
- Control rods inserted sufficiently to diminish fission from prompt neutrons
- RIVs fully close
- Condensate return valve for one ICS train is fully open
- Peak containment pressure is reached

- Containment pressure starts decreasing

Event timing is summarised in Table 15.5-23.

Identification of Operator Actions

No operator action is required to mitigate the event.

Systems Operation

As shown in Figure 15.5-104, break flow from the turbine side is generally higher than the break flow from the RPV side of the break because of the pipe inventory, and the break flow continues even after the RIVs are fully closed at 10 seconds. FW pumps are assumed to trip concurrent with the break initiation because the bounding scenario assumes LOPP concurrent with the pipe break. One IC is started when the high containment pressure setpoint is reached, which occurs within 1 second. Although two ICs are available, only one IC is credited to bound the large break case in the isolation condenser steam supply pipe. Figure 15.5-102 shows the heat removal rate of one IC exceeds the power generation decay heat after 20 seconds. As a result, reactor pressure decreases rapidly even after the break isolation shown in Figure 15.5-103. RPV water level, labelled as “Collapsed Downcomer Level” in Figure 15.5-105 stabilises well above the TAF in 3-hours. The decrease in the downcomer level during the first 3-hours is due to the gradual decrease in the void fraction in the core and chimney, not due to RPV water inventory losses. There is no RPV water inventory loss after the RIVs close. Fuel never heats up because the core remains covered throughout the event. After the fission power is diminished, fuel cladding temperature remains near saturation temperature.

Figure 15.5-106 shows the containment pressure in response to a large MS pipe break inside containment. The break location is assumed away from the containment shell at the lowest MS pipe elevation and directed upwards as discussed in NEDC-33922P-A (Reference 15.5-28). This configuration maximises containment pressure. The break location is assumed next to the containment shell and directed towards the containment shell in calculating the shell temperature in Figure 15.5-107 as discussed in Containment Evaluation Report. This maximises shell temperature.

The initial containment pressure includes a bias to account for uncertainties and is assumed at the containment high pressure setpoint for scram, reactor isolation and IC initiation. Although it appears that the setpoint is reached as soon as the break occurs, this is an artifact of the conservative initial condition assumption. A finite amount of time would have to elapse for containment pressure to reach the setpoint if the containment initial pressure is at the nominal pressure in normal operation. The pressure trend in Figure 15.5-104 shows that the containment pressure increases by 18.5 kPa in less than 1 second, indicating that the containment high pressure setpoint is reached in less than 1 second when the initial containment pressure is at the nominal pressure in normal operation. This confirms the break flow calculation assumption that containment high pressure setpoint will be reached in less than 1 second.

The initial reactor power in the conservative case calculations is 102% of the rated power to account for the power uncertainty. However, hot shutdown conditions may be more limiting for the mass release from the break because the initial RPV void fraction is lower, resulting in higher liquid carryover to the break location. This also causes the break flow enthalpy to be lower. Both the full rated initial conditions and hot shutdown initial conditions are analysed. Figure 15.5-106 shows that the full rated initial condition is more limiting for containment peak pressure.

PCCS does not require actuation, it is always in service and rejects heat to the equipment pool that is connected to the reactor cavity pool during normal operation. The calculations assume there is no heat loss from the containment shell to the concrete as discussed in

NEDC-33922P-A (Reference 15.5-28). Heat removal from containment atmosphere is by the containment shell heating up, by PCCS and through the containment dome to the pool. After RIV closure for large breaks, the only energy addition to the containment is due to the heat transfer from the RPV and hot piping walls. Because only one IC is sufficient to depressurise the RPV rapidly, heat load to containment also becomes small in the long term.

Core and Barrier Performance

Results

Key results are summarised in Table 15.9-3. The plots for RPV parameters and containment parameters are shown in Figures 15.5-102 through 15.5-107. The peak pressure is less than the design pressure with more than 20% margin. Containment shell temperature is also well below the containment shell design temperature of 166°C. Peak accident pressure is approximately 322 kPaG (423 kPa) and half of the peak pressure is approximately 161 kPaG (262 kPa). As shown in Figure 15.5-106, containment is depressurised rapidly, and containment pressure is reduced to 185 kPa at 6-hours. This meets the acceptance criterion for the containment response to pipe breaks that the containment pressure should be reduced to less than half of the peak pressure in 24-hours.

Barrier Performance

There is no fuel damage as a result of an MSLB inside containment. The only activity available for release is normal reactor coolant concentration in the vessel and piping prior to the break.

Radiological Consequences

The radiological consequences for a MSLB inside containment are bounded by the consequences for MSLB outside containment presented in Subsection 15.5.9.1.1.

15.5.4.5.2 Feedwater Pipe Break Inside Containment, Conservative Case

Postulated Initiating Event

A double-ended guillotine break occurs in the larger diameter segment of one FW pipe. This is more limiting than a break occurring in the smaller diameter FW pipe segments closer to the RPV. The bounding scenario is the same as that described in Subsection 15.5.4.5.1 for MS Pipe Break Inside Containment.

Sequence of Event

The bounding scenario analysed:

- Double-ended guillotine rupture of the FW pipe break inside containment concurrent with LOPP
- FWPT and coast down
- TSVs and TCVs close rapidly
- Scram initiated from high containment pressure
- Control rods start inserting on scram initiation
- Condensate return valves on two ICS trains start opening
- Control rods are inserted sufficiently to diminish fission from prompt neutrons
- FWRIVs and CIVs are fully closed
- Peak containment pressure is reached
- IC valves are fully open

Timing of events is summarised in Table 15.5-24.

Identification of Operator Actions

No operator action is required to mitigate the event.

System Operation

RPV and containment response to a large FW break is similar to the RPV and containment response to a large MS pipe break discussed in Subsection 15.5.4.5.1. Containment pressure reaches the containment high pressure setpoint for scram, reactor and containment isolation, and IC initiation in less than 1 second. Power generated by prompt fission is diminished in 3 seconds after the break. Condensate return valves in two of the three ICS trains start opening in 1 second and fully open in 11 seconds. As shown in Figure 15.5-164, heat removal rate is much larger than the decay heat. As a result (shown in Figure 15.5-165), RPV pressure decreases much faster than the MS pipe break case. Reactor water level is shown in Figure 15.5-166. The indicated water level stabilises above the actual collapsed downcomer level. This is because the wide range level is off scale when the actual level falls below the lower tap and no longer indicates level. The actual collapsed downcomer level stabilises well above TAF.

Break flow from the pump side decreases initially because the break location is far away from the pump and the enthalpy becomes saturated locally right after the break although the pump is still coasting down as shown in Figure 15.5-167. The break flow from the pump side recovers and exceeds that of the RPV break side. This is because of the pipe water inventory and the pump coasting down. The break flow becomes zero when the CIV is closed at 10 seconds. Enthalpy value after this point is not meaningful because there is no break flow.

As in the MS pipe break cases, containment pressure in Figure 15.5-168 and temperature in Figure 15.5-169 are calculated for a break location maximising pressure and temperature.

An additional FW pipe break case was included accounting for the lower initial FW temperature because the containment peak pressure may be higher if the FW pipe break occurs when the plant is operating at reduced FW temperature. Break flow rate is higher at higher subcooling. However, break flow enthalpy is also lower. Figure 15.5-168 shows the containment pressure for normal FW temperature and reduced FW temperature. Normal FW temperature results in a higher containment pressure. The peak pressure for both cases is bounded by the peak pressure resulting from a MS pipe break.

Core and Barrier Performance

Results

Plots for RPV and containment parameters are shown in Figures 15.5-164 through 15.5-169. Key results are summarised in Table 15.9-3 and show that the peak containment pressure and temperature resulting from FW pipe breaks are bounded by the MS pipe breaks. Containment pressure and temperatures are less limiting than the MS pipe cases and meet the acceptance criteria. Containment pressure calculated for FW pipe break at 6-hours is also less than half the peak containment pressure calculated for the MS pipe case and decreasing.

Barrier Performance

There is no fuel damage as a result of an FWLB inside containment. The only activity available for release is normal reactor coolant concentration in the pipe prior to the break.

Radiological Consequences

The radiological consequences for a FWLB inside containment are bounded by the consequences for FWLB outside containment presented in Subsection 15.5.9.1.2.

15.5.4.5.3 Large Isolation Condenser Pipe Breaks Inside Containment

An ICS break larger than the area of a 19 mm equivalent diameter pipe is detected by the leakage detection for each ICS train separately. When a break is detected in one ICS train, both RIVs in the steam supply pipe and the condensate return pipe of the affected ICS train are closed. The stroke time and delay time assumed for the IC isolation valves in the analysis are the same as those for all other RIVs and bound all other equipment initiation delays starting from the time of the pipe break. The other two unaffected ICs are available to remove decay heat. For conservatism, the analysis assumes only one of the two remaining ICs is put in service on high containment pressure. Therefore, the number of ICs available in this case is only one, which is the same as the number of ICs available in the MS pipe break cases as analysed for all breaks larger than a 19 mm diameter.

Although the IC steam supply pipe diameter may be as large as the MS flow limiter diameter, the break flow rate from an IC steam supply pipe break is less than the break flow rate from the MS pipe break. This is due to the much larger inventory in the MS piping connected to both ends of the break.

The liquid in the IC is subcooled and does not contribute to high energy discharge from the break. Therefore, the MS pipe break for containment response is more limiting than the IC steam supply pipe break.

ICS condensate return pipe diameter is much smaller than the FW pipe diameter used in the FWLB analysis. Therefore, large breaks in the IC condensate return pipe are bounded by large breaks in the FW pipe or MS pipe.

Because the IC pipe breaks are bounded by either the MS pipe or the FW pipe breaks, no further analysis of IC pipe breaks is needed.

15.5.4.5.4 Small Steam and Liquid Pipe Breaks Inside Containment

Postulated Initiating Event

A break area of ≤ 19 mm equivalent diameter remains un-isolated. These breaks are analysed for fuel integrity and containment integrity for at least 72-hours using conservative assumptions.

All liquid pipe break nozzles are at least 4 meters above TAF. A small pipe break on instrument lines may remain un-isolated indefinitely.

A small liquid pipe break and a small steam pipe break have similar break flow rates after the level falls to 4 m above TAF. Because the ICs depressurise the RPV, the break flow becomes very small in a few hours. Fuel heat up does not occur even without injection to the RPV. Containment heat removal occurs through the PCCS to the equipment pool and through the containment head to the reactor cavity pool.

Sequence of Event

The bounding scenario analysed:

- Small steam pipe break concurrent with LOPP
- Pressure controller remains as-is; TCV remains at initial position, turbine chest pressure decreases rapidly
- FW pump trips and coasts down
- MS pipe low pressure setpoint is reached
- Reactor scrams
- MSRIVs are fully closed

- RPV water level decreases to Level 2
- Condensate return valves on two ICS trains are fully open
- Peak containment pressure is reached

Tables15.5-25 and Table 15.5-26 list the event sequences for small steam and liquid pipe breaks, respectively.

Identification of Operator Actions

No operator action is required to mitigate the event.

System Operation

The conservative cases assume LOPP concurrent with the pipe break, which is more limiting than the case where preferred power is available (discussed in Subsection 15.2.5.6 of NEDC-34180P (Reference 15.5-2). TCV and TSV closure is expected to occur because of the LOPP. However, TCVs and TSVs are assumed not to close on LOPP. Rather, MSRIV closure on low steam pipe pressure is conservatively credited in the analysis. The back pressure through TSVs and TCVs is assumed to decrease rapidly, maximising the RPV water inventory loss to the turbine. Reactor scram also occurs on low steam line pressure accounting for the delays after the low steam pipe pressure is reached.

Small Liquid Pipe Break

IC condensate return valves start opening when the level falls to Level 2. As shown in Figure 15.5-114 and Figure 15.5-115, decay heat is removed by two ICs. The RPV depressurises initially when the sum of the decay heat removal rate by the ICs and the energy discharge from the break exceeds the decay heat power. ICs remove less power at lower pressure because of the lower temperature difference between the RPV steam and pool water. Reactor pressure stabilises at a low value and the depressurisation rate becomes very small. As shown in Figure 15.5-116, there is a rapid decrease in the collapsed downcomer level. This decrease is primarily due to the void collapse in the RPV. There are two small increases in level at approximately 69000 and 91000 seconds in Figure 15.5-116. These increases are due to void redistribution in the vessel. There is no increase in the RPV water inventory. As shown in Figure 15.5-116, downcomer collapsed level falls below TAF at 206000 seconds. However, the two-phase level in the core remains above TAF. Fuel remains wetted and thus never heats up.

The break mass and energy release are calculated assuming there is no back pressure. This assumption was made to bound breaks outside containment and accounts for the expected lower containment pressure than calculated because of the conservative assumptions used in the containment analyses. Even without break back pressure, Figure 15.5-118 shows that the break flow becomes very small in the long term.

Containment pressure calculated by using conservative assumptions and the small liquid pipe break flow without back pressure is shown in Figure 15.5-119. RPV pressure is also shown on the same figure. The calculated containment pressure increases to the RPV pressure at approximately 232800 seconds. Containment pressure is not higher than the RPV pressure because the break flow stops if the containment pressure becomes equal to the RPV pressure.

However, there is a potential that if ICS depressurises the RPV faster than PCCS depressurises containment in the absence of a break, reverse flow from containment to the RPV may occur. Non-condensables ingested into the RPV may collect in the ICs and reduce their efficiency. Both the RPV and the containment could start repressurising if back flow were to occur. To investigate this possibility, the containment pressure is calculated starting from 232800 seconds until the end of the 72-hour period assuming no break flow. The dashed line in Figure 15.5-117 shows that containment is rapidly depressurised in the absence of break flow. Because containment pressure decreases faster than the RPV pressure when there is no break flow, the RPV cannot depressurise below the containment pressure and reverse flow cannot occur. Energy released from the RPV through the break is a small fraction of the decay heat in the long term. A much larger fraction of the decay heat is removed by the ICs. Therefore, RPV pressure calculated with and without a break are approximately the same at the time the RPV depressurises to near containment pressure.

Figure 15.5-117 shows that containment pressure remains below 262 kPa in the long term, which is 50% of the peak accident pressure calculated in Subsection 15.5.4.5.1.

Small Steam Pipe Break

Figure 15.5-110 shows that level remains above TAF for small steam pipe breaks. Containment response shown in Figure 15.5-112 and Figure 15.5-113 for a small steam pipe break is similar to the containment response in Figure 15.5-119 and Figure 15.5-120 for a small liquid pipe break.

A small pipe break on an IC steam pipe does not cause a more limiting core or containment response than an instrument line break. The small breaks conservatively credit only two of the three ICs even though a break of ≤ 19 mm equivalent diameter on an IC does not cause degradation in the IC. There is sufficient steam in the RPV to feed the condensation in the IC. Insufficient steam in the RPV to feed the IC only occurs if the RPV is depressurised to the point where almost all of the steam escapes the break. This is the case if the RPV pressure is lower than that calculated for an instrument pipe break. However, in this case, the break flow is also less than the break flow calculated for an instrument pipe break, resulting in the IC small break less limiting than the instrument pipe break. Breaks on IC steam pipes are no more limiting than a break on an instrument steam pipe break. This is because the break flow rates are the same for a small steam pipe break regardless of the break location, or the RPV pressure is too low to feed the IC.

Core and Barrier Performance

Results

Key results for small steam and liquid pipe breaks are summarised in Table 15.9-3. The plots for RPV parameters and containment for small steam pipe breaks are shown in Figures 15.5-108 through 15.5-113.

The plots for RPV and containment parameters are shown in Figures 15.5-114 through 15.5-120. A small pipe break on the instrument lines may remain un-isolated indefinitely. Because the ICs depressurise the RPV, the break flow becomes very small in a few hours. Fuel heat up does not occur even without injection to the RPV.

Barrier Performance

There is no fuel damage because of a small liquid or FW break inside containment. The only activity available for release is normal reactor coolant concentration in the pipe prior to the break.

Radiological Consequences

The radiological consequences for a small liquid or FW break inside containment are bounding for the consequences for small liquid and steams pipe breaks outside containment presented in Subsection 15.5.9.1.5.

15.5.5 Analysis of Design Extension Conditions without Significant Fuel Degradation

The bounding transient event selection in Section 15.2.4 determines the list of DEC events evaluated in the following subsections. Tables 15.5-4 through 15.5-6 provide input parameters and initial conditions used in the DEC analyses. Table 15.5-6 provides the DLs used in the

DEC analyses. The analysis is performed consistent with the DSA analysis approach described in NEDC-34179P (Reference 15.5-1, Sections 15.1.3) and in Section 15.2.2 of NEDC-34180P (Reference 15.5-2).

15.5.5.1 Control Rod Drop Accident – Practically Eliminated

The BWRX-300 uses Global Nuclear Fuel (GNF)-2, with a core design that is like the BWR operating fleet. The approved Control Rod Drop Accident (CRDA) methodology of GNF LTR NEDE-33885P-A, Revision 1 “Control Rod Drop Accident Methodology,” (Reference 15.5-57) will be applied to the BWRX-300 to demonstrate that the cladding failures do not occur for the postulated (albeit incredible) CRDA.

Technical Basis:

The combined features of the CRD system and the rod control system incorporate appropriate limits on the potential amount and rate of reactivity increase. The fine motion movement capability of the FMCRD allows reactivity additions from rod withdrawal to be limited. Control rod withdrawal sequences and patterns are selected to achieve optimum core performance and low individual rod worth. The BWRX-300 design prevents rod drop and rod ejection events through positive design means.

Positive Means

The BWRX-300 has DL2 dual separation detection devices that practically eliminate the CRDA. The BWRX-300 features a FMCRD that uses a bayonet style coupling that requires a 45-degree rotation to uncouple. Because the FMCRD is firmly bolted into its position under the reactor vessel and the control rod is constrained from rotation by the fuel assemblies, it is not possible for the control rod to uncouple from the FMCRD during reactor operation.

The hollow piston is the component within the FMCRD coupled to the control rod. The hollow piston normally rests on the ball nut internal to the FMCRD. There are dual FMCRD separation detection switches that sense whether the hollow piston along with the associated control rod are resting on the ball nut. If the sensor detects that the hollow piston is no longer on the ball nut, then control rod withdrawal is blocked. Additionally, the hollow piston has latches that prevent inadvertent withdrawal of the assembly when not attached to the ball nut. This essentially limits possible separation such that it is not physically possible for a CRDA involving a single control rod falling completely out of the core to occur.

Control rod ejection is prevented by physical constraints including the attachment of the control rod guide tube to the core plate and the CRD connection to the control rod guide tube. The FMCRD includes a brake that further prevents inadvertent rod withdrawal. The FMCRD also includes an internal ball check valve, which reduces the likelihood of rapid rod withdrawal. The Safety Class 1 ball check valve performs a Safety Category 1 function in preventing:

- Reverse flow from the scram inlet port against the pressure and flow conditions caused by a break of the scram line
- Loss of pressure from the underside of the hollow piston
- Generation of loads on the drive that could cause a rapid rod withdrawal and associated reactivity insertion.

Normal rod movement and the rod withdrawal rate are limited by the FMCRD. The rod control system controls rod patterns and provides control rod blocks limiting the rate and amount of reactivity addition for control rod movement.

The combined features of the CRD system and the rod control system incorporate appropriate limits on the potential amount and rate of reactivity increase. The fine motion movement capability of the FMCRD allows limited reactivity additions from rod withdrawal. Control rod withdrawal sequences and patterns are selected to achieve optimum core performance and low individual rod worth. The BWRX-300 design prevents rod drop and rod ejection events through positive design means. Control rod drop is prevented using a bayonet style coupling, CRD mechanism latches, and CRD separation switches. As a result, the CRDA and control rod ejection event have been practically eliminated.

Probability Analysis

The BWRX-300 design features two separate and independent functions to detect separation between a control rod and the CRDM. A block is initiated that cuts power to the CRDM, preventing further separation. As a result, if the rod were to drop at a later point in time, it would only drop a short distance (a few centimetres), causing a mild reactivity change compared to what is analysed with the approved methodology of NEDE-33885P-A (Reference 15.5-57).

For a FMCRD, the frequency of a control rod drop is estimated to be 3.47 X 10-6 per year. This estimate assumes the control rod gets stuck in the core with a frequency of 1.0 per year and that the drop can occur at any time in the reactor cycle, making it conservative and not completely representative of a CRDA event. Due to doppler and void feedback characteristics, CRDA is only of concern in low power conditions, i.e., below 5% rated thermal power and below 300 psig steam dome pressure. The probability that a reactor is in a low power state is:

Reactor in a low power state = 3 days per year = 0.0082 years/year = 8.2 X 10-3

When multiplying the FMCRD rod drop frequency by the conditional probability that a reactor is in a low power state, the resulting CRDA frequency is 2.85 X 10-8 per year. This result assumes a stuck rod frequency of 1.0 per year, which is conservative given that there have been 0 incidences of a stuck rod in over 25,000 drive-years of FMCRD European operating experience. Another evaluation performed includes a stuck rod frequency not equal to 1.0 per year and concluded a CRDA frequency on the order of 10^{-11} per year.

The rod control system controls rod patterns and provides control rod blocks to limit the rate and amount of reactivity addition for control rod movement.

Conclusion

A control rod drop or rod ejection event is not possible in the BWRX-300 due to the FMCRD bayonet style coupling, hollow piston latches, and dual separation detection devices that limit separation of the rod from the drive. Even if the accident were to be possible, the event would not be credible due to its mild reactivity insertion and low probability of occurrence. BWR Operating Experience (OPEX) demonstrates that with the FMCRD design, no instance of a CRDA event has ever occurred and probability of occurrence is in the order of 3E-8 per year, which is considered practically eliminated.

15.5.5.2 Pressure Increase – DECs

15.5.5.2.1 Closure of One Main Steam Reactor Isolation Valve

This event is designated as an EX-DEC event. The event sequence name is Closure of One MSRIV (1MSRIVC), and the Event Sequence ID is PI-1MSRIVC_CCF-Hydraulic-Scram_EX-DEC.

Postulated Initiating Event

The PIE is the same as the BL-AOO event discussed in Subsection 15.5.3.2.2. The analysis assumes a CCF hydraulic scram failure. The control rods enter the core using the CRDM run-in function. This event demonstrates that the CRDM run-in function performs the FSF control of reactivity without hydraulic scram.

Sequence of Event

The event sequence summary:

- One MSRIV closes causing pressure and power to increase
- Hydraulic scram signal on MSRIV position fails; scram fails
- Hydraulic scram on any signal fails
- MSRIV in the second steam line closes on leak detection indication (assumed because it makes the event more severe)
- CRDM run-in initiation on high flux after scram signal
- All ICS trains initiate on high flux after scram signal
- FW and condensate pumps trip on high flux after scram signal
- Controlled state achieved.

Table 15.5-27 lists the event sequence.

Identification of Operator Actions

No operator action is required to mitigate the event.

Systems Operation

Credited DL2 functions:

- DL2-21 – Anticipatory Hydraulic Scram Signal on MSRIV/MSCIV Position (scram fails)
- DL2-43 – FW Check Valve Closure on Reverse FW Flow

Credited DL4a functions:

- DL4a-40 – CRD Fast Motor Run-In on High Flux After Scram Signal
- DL4a-41 – FW Pump/Condensate Pump Trip on High Flux after Scram Signal
- DL4a – ICS Trains 1, 2, and 3 Initiations on High Flux after Scram Signal

Core and System Performance

Input Parameters and Initial Conditions

The event is simulated by initiating closure of one MSRIV. The initial conditions and plant parameters are provided in Tables 15.5-4 through 15.5-6. The initial CPR and hot rod power are consistent with the referenced core design.

The analysis is performed using an equilibrium core design. The event is run at BOC, MOC, and EOR cycle time in cycle conditions.

Results

The simulated closure of one MSRIV is presented in Figures 15.5-121 through 15.5-126, and the results are presented in Table 15.9-2. The results are shown for the case with the limiting result for cladding temperature and reactor pressure response. The neutron flux increases rapidly because of the void reduction caused by the pressure increase. The pressure increase is limited due to the initiation of ICS, FWPT and CRDM run-in that occur on high flux after scram signal (i.e., indications of high-power level post scram initiation). The long-term core cooling capability is assured by meeting the DBA event acceptance criteria for fuel cladding and RCPB. The reactor integrity is assured by meeting the DBA event RCPB pressure criteria. The acceptance criteria are provided in Section 15.3. Although the DEC acceptance criteria are described in Section 15.3.2, the results are conservatively compared to the DBA acceptance criteria (described in Section 15.3.1) and demonstrate significant margin. The cladding temperature remains below the temperature at which significant oxidation occurs due to metal water reaction.

This event resulted in the highest peak cladding temperature and peak vessel pressure. Sensitivities are performed to examine cliff edge effects. Sensitivities on key initial conditions and phenomena that impact cladding temperature and peak vessel pressure described in the TRACG Application NEDC-34043P are applied separately by at least one standard deviation:

- Core void coefficient
- Channel interfacial shear
- Chimney interfacial shear
- Separator steam carry under
- Critical quality used in boiling transition correlation
- Hot rod power.

Results indicate no significant cliff edge effects. No excessive vessel pressure and no core damage occurs.

Barrier Performance

The effect of this event does not result in any temperature or pressure transient more than the DBA derived acceptance criteria for the fuel, pressure vessel, or containment. No fuel failures occur. Therefore, these barriers maintain their integrity and function as designed.

Radiological Consequences

DEC events do not have event-specific radiological acceptance criteria. The effects of DEC events are considered in NEDC-34184P (Reference 15.5-6)).

15.5.5.2.2 Complex Sequence of Generator Load Rejection or Turbine Trip

This event is designated as an EX-DEC event. The event sequence name is Complex Sequence Generator Load Rejection or Turbine Trip (LR-TT) plus CCF-Mechanical CRD, and the Event Sequence ID is CSS-LR-TT_CCF-Mechanical-CRD_EX-DEC. This event demonstrates that with multiple failures to insert independent control rods with diverse motive forces combined with a very frequent PIE, that the remaining control rods perform the FSF reactivity control.

Postulated Initiating Event

The PIE is the same as for the LR-TT AOO event. Additionally, the event assumes that half of the control rods with the highest rod worth fail to scram and the CRDM run-in fails to insert the rods that failed to scram. No other failures are assumed.

Sequence of Event

The event sequence summary:

- TCVs and/or TSVs close quickly causing pressure and power increase
- Anticipatory scram occurs on indication of a turbine trip or load rejection signal, but half of the control rods fail to insert
- RPC opens TBVs to control pressure
- RLC maintains level
- Controlled state achieved.

Table 15.5-28 lists the event sequence.

Identification of Operator Actions

No operator action is required to mitigate the event.

Systems Operation

Credited DL2 functions:

- DL2-02 – Maintain Target Level
- DL2-01 – Maintain Target Pressure
- DL2-08 – Anticipatory Hydraulic Scram on Generator Load Rejection or Turbine Trip Demand
- DL2-09 – TBV Fast Open on Generator Load Rejection or Turbine Trip Demand

Core and System Performance

Input Parameters and Initial Conditions

The event is simulated by initiating a generator load rejection or turbine trip that results in a fast closure of the TCVs and/or TSVs. The initial conditions and plant parameters are provided in Tables 15.5-4 through 15.5-6. The initial CPR and hot rod power are consistent with the referenced core design.

The analysis is performed using an equilibrium core design. The event is run at BOC, MOC, and EOR cycle time in cycle conditions.

Results

The simulated generator load rejection or turbine trip is presented in Figures 15.5-127 through 15.5-132, and the results are presented in Table 15.9-2. The results are shown for the case with the limiting result for reactor pressure response. Automatic anticipatory reactor scram occurs following indication of a generator load rejection or turbine trip with half of the rods failing to insert. The neutron flux increases rapidly because of the void reduction caused by the PI.

The long-term core cooling capability is assured by meeting the DBA event acceptance criteria for fuel cladding and RCPB. The reactor integrity is assured by meeting the DBA event RCPB pressure criteria. The acceptance criteria are provided in Section 15.3. Although the DEC acceptance criteria are described in Section 15.3.2, the results are conservatively compared to the DBA acceptance criteria (described in Section 15.3.1) and demonstrate significant margin. The cladding temperature remains well below the temperature at which significant oxidation occurs due to metal water reaction.

A controlled state is achieved with steam bypassed to the main condenser and fed back into the reactor by FW. This is maintained for a significant amount of time as long as power is available. Depending on the conditions, operators may initiate additional CRDM run-in signals or manually insert CRDM to insert the remaining control rods into the core. If operator actions are unsuccessful, operators inject boron to shut the reactor down. Another option available to the operators is to decrease power by reducing FW flow. With reduced FW flow, reactor water level decreases, reducing core flow and reducing reactivity through void reactivity feedback until the steam flow matches the FW flow.

Barrier Performance

The effect of this event does not result in any temperature or pressure transient in excess of the DBA derived acceptance criteria for the fuel, pressure vessel, or containment. No fuel failures occur because there is no significant cladding temperature increase. Therefore, these barriers maintain their integrity and function as designed.

Radiological Consequences

DEC events do not have event specific radiological acceptance criteria. The effects of DEC events are considered in NEDC-34184P (Reference 15.5-6)).

15.5.5.2.3 Loss of Condenser Vacuum with CCF Hydraulic Scram

This event is designated as an EX-DEC event. The event sequence name is LOCV, CCF-Hydraulic-Scram and the Event Sequence ID is PI-LOCV_CCF-Hydraulic-Scram_EX-DEC. This event demonstrates that the CRDM run-in function performs the FSF controlling reactivity without the hydraulic scram.

Postulated Initiating Event

The PIE is the same as the BL-AOO event. A CCF results in failure of hydraulic scram. The control rods enter the core using the CRDM run-in function. No other failures are assumed.

Sequence of Event

The event sequence summary:

- Loss of vacuum results in a turbine trip
- TSVs close quickly causing pressure and power increase
- Hydraulic scram signal on either turbine trip or high main condenser pressure scram fails
- Hydraulic scram fails on any signal
- CRDM run-in initiation occurs on high flux after scram signal
- RPC opens TBVs to control pressure
- RLC maintains RPV water level
- ICS initiates on high flux after scram signal
- FW pumps trip on high flux after scram signal
- TBVs close on high main condenser pressure
- Controlled state achieved

Table 15.5-29 lists the event sequence.

Identification of Operator Actions

No operator action is required to mitigate the event.

Systems Operation

Credited DL functions:

DL2:

- DL2-02 – Maintain Target Level
- DL2-01 – Maintain Target Pressure
- DL2-09 – TBV Fast Open on Generator Load Rejection or Turbine Trip Demand
- DL2-13 – Turbine Trip on High Main Condenser Pressure Setpoint 2
- DL2-14 – TBV Closure on High Main Condenser Pressure Setpoint 3

- Anticipatory Hydraulic Scram on Either:
  - DL2-08 –Generator Load Rejection or Turbine Trip Demand (scram fails)
  - DL2-37 – High Main Condenser Pressure Setpoint 1 (scram fails)

DL4a:

- DL4a-40 – CRD Fast Motor Run-In on High Flux After Scram Signal
- DL4a-41 – FW Pump/Condensate Pump Trip on High Flux after Scram Signal
- DL4a – ICS Trains 1, 2, and 3 Initiations on High Flux after Scram Signal

Core and System Performance

Input Parameters and Initial Conditions

The LOCV results in a turbine trip. The event is simulated by initiating a turbine trip that results in a fast closure of the TSVs. The initial conditions and plant parameters are provided in Tables 15.5-4 through 15.5-6. The initial CPR and hot rod power are consistent with the referenced core design.

The analysis is performed using an equilibrium core design. The event is run at BOC, MOC, and EOR cycle time in cycle conditions.

Results

The simulated LOCV is presented in Figures 15.5-133 through 15.5-138 and the results are presented in Table 15.9-2. The results are shown for the case with the limiting result for cladding temperature and reactor pressure response. The neutron flux increases rapidly because of the void reduction caused by the PI. However, the PI is initially limited by the TBVs opening. The pressure increase is limited due to the initiation of ICS, FW trip, and CRDM run-in that occur on high flux after scram signal.

The long-term core cooling capability is assured by meeting the DBA event acceptance criteria for fuel cladding and RCPB. The reactor integrity is assured by meeting the DBA event RCPB pressure criteria. The acceptance criteria are provided in Section 15.3. Although the DEC acceptance criteria are described in Section 15.3.2, the results are conservatively compared to the DBA acceptance criteria (described in Section 15.3.1) and demonstrate significant margin. The cladding temperature remains below the temperature at which significant oxidation occurs due to metal water reaction.

Barrier Performance

The effect of this event does not result in any temperature or pressure transient in excess of the DBA derived acceptance criteria for the fuel, pressure vessel, or containment. No fuel failures occur because there is no significant cladding temperature increase. Therefore, these barriers maintain their integrity and function as designed.

Radiological Consequences

DEC events do not have event specific radiological acceptance criteria. The effects of DEC events are considered NEDC-34184P (Reference 15.5-6).

15.5.5.2.4 Loss of Preferred Power with CCF Hydraulic Scram

This event is designated as an EX-DEC event. The event sequence name is LOPP, CCF-Hydraulic-Scram and the Event Sequence ID is PI-LOPP_CCF-Hydraulic-Scram_EX-DEC. This event demonstrates that the CRDM run-in function performs the FSF controlling reactivity without the hydraulic scram.

Postulated Initiating Event

The PIE is the same as the BL-AOO event. A CCF results in failure of hydraulic scram. The control rods enter the core using the CRDM run-in function. No other failures are postulated in the event.

Sequence of Event

The event sequence summary:

- LOPP results in generator output breakers opening
- TCVs close quickly
- FW pumps lose power
- Hydraulic scram signal fails on either generator load rejection or low electric bus voltage
- Hydraulic scram fails on any signal
- CRDM run-in initiation on high flux after scram signal
- TBVs close on loss of power
- ICS initiates on high flux after scram signal
- Controlled state achieved

Table 15.5-30 lists the event sequence.

Identification of Operator Actions

No operator action is required to mitigate the event.

Systems Operation

Credited DL functions:

DL2:

- DL2-01 – Maintain Target Pressure
- Anticipatory Hydraulic Scram on Either:
  - DL2-08 –Generator Load Rejection or Turbine Trip Demand (scram fails)
  - DL2-26 – Low Electric Bus Voltage (scram fails)
- DL2-09 – TBV Fast Open on Generator Load Rejection or Turbine Trip Demand
- DL2-43 – FW Check Valve Closure on Reverse FW Flow

DL4a:

- DL4a-40 – CRD Fast Motor Run-In on High Flux After Scram Signal
- DL4a – ICS Trains 1, 2, and 3 Initiations on High Flux after Scram Signal

Core and System Performance

Input Parameters and Initial Conditions

The LOPP results in the generator output breakers opening and a loss of power to the FW pumps. The event is simulated by initiating a FWPT and a load rejection that results in a fast closure of the TCVs. The initial conditions and plant parameters are provided in Tables 15.5-4 through 15.5-6. The initial CPR and hot rod power are consistent with the referenced core design.

The analysis is performed using an equilibrium core design. The event is run at BOC, MOC, and EOR cycle time in cycle conditions.

Results

The simulated loss of preferred power with CCF hydraulic scram failure is shown in Figures 15.5-139 through 15.5-144 and the results are presented in Table 15.9-2. The results are shown for the case with the limiting result for cladding temperature and reactor pressure response. The rapid closure of the TCVs results in a PI. The neutron flux increases rapidly because of the void reduction caused by the PI. However, the PI is initially limited by the opening of the TBVs. The TBVs later close, ICS initiates, and CRDMs run in on high flux after scram signal. The ICS continues to limit the PI.

The long-term core cooling capability is assured by meeting the DBA event acceptance criteria for fuel cladding and RCPB. The reactor integrity is assured by meeting the DBA event RCPB pressure criteria. The acceptance criteria are provided in Section 15.5.3. Although the DEC acceptance criteria are described in Section 15.5.5, the results are conservatively compared to the DBA acceptance criteria (described in Section 15.5.4) and demonstrate significant margin. The cladding temperature remains below the temperature at which significant oxidation occurs due to metal water reaction.

Barrier Performance

The effect of this event does not result in any temperature or pressure transient in excess of the DBA derived acceptance criteria for the fuel, pressure vessel, or containment. No fuel failures occur. Therefore, these barriers maintain their integrity and function as designed.

Radiological Consequences

DEC events do not have event specific radiological acceptance criteria. The effects of DEC events are considered in NEDC-34184P (Reference 15.5-6).

15.5.5.3 Reactivity and Power Distribution Anomalies – DECs

15.5.5.3.1 All Control Rod Withdrawal at Power

This event is designated as an EX-DEC event. The event sequence name is CCF – All Control Rod Withdrawal at Power- All Rods (CCF-ACRW), Passive Digital CCF DL2 Technology Platform (CCF-DL2) and the Event Sequence ID is RI-CCF-ACRW_CCF-DL2_EX-DEC.

Postulated Initiating Event

All control rods inserted in the core start to withdraw due to rod control spurious CCF. A passive CCF of DL2 technology platform results in DL2 function failure.

Sequence of Event

The event sequence summary:

- All control rods start to withdraw resulting in a power increase
- Automatic Thermal Limit Monitor (ATLM) and Multi-Channel Rod Block Monitor (MRBM) fail to block rod withdrawal
- RPC and RLC fail as-is at the initial condition
- Scram on STP power or neutron flux
- After scram, no immediate challenge to cladding and RCPB integrity
- RPV pressure decreases
- RPV water level decreases temporarily due to the void collapse in the core and chimney

- Sensed level increases due to continuing FW flow and flashing in the downcomer
- FW isolation occurs on high RPV water level
- MS isolation occurs on low RPV pressure and pressure slowly increases
- One ICS train initiates on high RPV pressure (the simulation is ended before any ICS initiation because the key mitigation DL functions have already been demonstrated and a single ICS train can control pressure and removing decay heat as demonstrated in the pressure increase DBA analysis)
- Controlled state achieved

Table 15.5-31 lists the event sequence.

Identification of Operator Actions

No operator action is required to mitigate the event.

Systems Operation

Credited DL3 functions:

- DL3-05 – Hydraulic Scram on High Simulated Thermal Power
- DL3-04 – Hydraulic Scram on High Neutron Flux
- DL3-23 – FW Isolation on High RPV Water Level
- DL3-17 – MSRIV/MSCIV Isolation on Low RPV Pressure
- DL3-11 – ICS Train 1 initiation on High RPV Pressure

Core and System Performance

Input Parameters and Initial Conditions

The event is simulated by withdrawing all the control rods in the core using the initial conditions, plant parameters, and control rod speed specified in Tables 15.5-4 through 15.5-6. The initial CPR is conservatively biased low, and the hot rod power is conservatively biased high.

The analysis is performed using an equilibrium core design. The event is run at BOC and MOC cycle time in cycle conditions. EOC time in cycle is not run because all control rods are fully withdrawn.

Results

The simulated All Control Rod Withdrawal at Power (ACRW) is presented in Figures 15.5-145 through 15.5-150. The analysis results are presented in Table 15.9-2. The results are shown for the case with the limiting PCT response result.

When the control rods are withdrawn, the power increases and scram occurs on high simulated thermal power or neutron flux. The RPV water level increases and FW is isolated. RPV pressure decreases and the MSRIVs close. Eventually the RPV pressure increases, and one ICS train initiates. The pressure remains well within the DBA event RCPB pressure acceptance criteria in Table 15.3-2. There is no challenge to containment.

The long-term core cooling capability is assured by meeting the DBA acceptance criteria for fuel cladding and RCPB. The reactor integrity is assured by meeting the DBA event RCPB pressure criteria. The acceptance criteria are provided in Section 15.3 of NEDC-34181P. Although the DEC acceptance criteria are described in Section 15.3.2, the results are conservatively compared to the DBA acceptance criteria (described in Section 15.3.1) and demonstrate significant margin. The cladding temperature remains well below the temperature at which significant oxidation occurs due to metal water reaction.

Barrier Performance

The effect of this event does not result in any temperature or pressure transient in excess of the DBA derived acceptance criteria for the fuel, pressure vessel, or containment. No fuel failures occur because there is no significant cladding temperature increase. Therefore, these barriers maintain their integrity and function as designed and there is no core damage.

Radiological Consequences

DEC events do not have event specific radiological acceptance criteria. The effects of DEC events are considered in NEDC-34184P (Reference 15.5-6).

15.5.5.3.2 Inadvertent Single Control Rod Withdrawal at Power DEC

This event is designated as an EX-DEC event. The event sequence name is Inadvertent Control Rod Withdrawal at Power – Single Rod (ICRW), and the Event Sequence ID is RI-ICRW_CCF_DL2 _EX-DEC.

Postulated Initiating Event

A control rod inserted in the core is withdrawn due to a failure. A passive CCF of DL2 results in failure of the DL2 functions.

Sequence of Event

The event sequence summary:

- Single rod (with highest reactivity worth) starts to withdraw
- ATLM and MRBM fail to block the rod withdrawal
- RPC and RLC are assumed to function normally because this prolongs the event and makes it more severe for cladding temperature effects
- Reactor power increases but the scram level is not reached
- Local power and cladding temperature increase
- Power and the cladding temperature reach a stable level
- Operator action to initiate scram is expected due to the high-power level (But is not credited in analysis simulation as these DL functions demonstrate achieving and maintaining a controlled state)
- After the scram, no credit is taken for RPC or RLC (the simulation is ended because the key mitigation DL functions have already been demonstrated)
- RPV pressure decreases and the RPV water level increases (the simulation is ended because the key mitigation DL functions have already been demonstrated)
- FW isolation occurs on high RPV water level (the simulation is ended because the key mitigation DL functions have already been demonstrated)
- MS isolation occurs on low RPV pressure and pressure slowly increases (the simulation is ended because the key mitigation DL functions have already been demonstrated)
- One ICS train initiates on high RPV pressure (the analytical simulation is ended before any ICS initiation because the key mitigation DL functions have already been demonstrated and a single ICS train can control pressure and removing decay heat as demonstrated in the pressure increase DBA analysis)

- Controlled state achieved

Table 15.5-32 lists the event sequence.

Identification of Operator Actions

No operator action is credited.

Systems Operation

Credited DL3 functions:

- DL3-17 – MSRIV/MSCIV isolation on Low RPV Pressure
- DL3-23 – FW Isolation on High RPV Water Level
- DL3-11 – ICS Train 1 Initiation on High RPV Pressure

Core and System Performance

Input Parameters and Initial Conditions

The event is simulated by withdrawing a single control rod in the core using the initial conditions, plant parameters, and control rod speed specified in Tables 15.5-4 through 15.5-6. The initial CPR and the hot rod power are consistent with the reference core design.

The analysis is performed using an equilibrium core design. The event is run at BOC and MOC cycle time in cycle conditions. EOC time in cycle is not run because all control rods are fully withdrawn.

Results

The simulated ICRW event is presented in Figures 15.5-151 through 15.5-156. The analysis results are presented in Table 15.9-2. The results are shown for the case with the limiting PCT response result.

When the control rod is withdrawn, the power increases. The RPV water level and pressure vary insignificantly because RLC and RPC function to maintain level and pressure preventing scram, thus maximising the impact of fuel cladding temperature increase.

The acceptance criteria are provided in Section 15.3. Although the DEC acceptance criteria are described in Section 15.3.2, the results are conservatively compared to the PCT values of the DBA acceptance criteria (described in Section 15.3.1) and demonstrate significant margin. The long-term core cooling capability is assured because the effects are local. The reactor integrity is assured by meeting the DBA event RCPB pressure criteria.

Sensitivities are performed examining cliff edge effects. Key initial conditions are conservatively biased to cause transition boiling even though this does not occur at the nominal/realistic conditions associated with DEC conditions. With initial CPR conservatively biased low (by approximately 0.05), and the hot rod power (LHGR) conservatively biased high (approximately 30%), local high cladding temperatures occur in hot rods in a few high-power bundles located near the control rod withdrawn in error. This results in fuel cladding failure in a very limited number of rods. However, the fuel failures are localised, the core remains cooled, and no core damage occurs.

Barrier Performance

There is no challenge to the RCPB and containment. The fuel cladding may experience local failures if initial LHGR and CPR are more severe. The predicted number of rod failures is limited to high powered fuel rods in a few high-powered bundles near the control rod withdrawn in error. However, the fuel failures are localised, the core remains cooled, and no core damage occurs.

Radiological Consequences

DEC events do not have event specific radiological acceptance criteria. The effects of DEC events are considered in NEDC-34184P (Reference 15.5-6).

15.5.5.4 Decrease in Reactor Coolant Inventory – DEC

15.5.5.4.1 Feedwater Isolation

This event is in the IR group and is designated as an EX-DEC event. The event sequence name is FW Isolation (CCF-FWI-DL3) and the Event Sequence ID is IR-CCF-FWI-DL3_EX-DEC.

Postulated Initiating Event

The event sequence assumes a spurious CCF isolates all FW flow and a passive CCF of the DL3 functions.

Sequence of Event

The event sequence summary:

- FW flow isolation causes RPV water level and power to decrease
- RPC maintains RPV pressure
- Scram and MSRIV isolation on sustained low FW flow
- ICS pressure control initiates on high RPV pressure
- Controlled state achieved

Table 15.5-33 lists the event sequence.

Identification of Operator Actions

No operator action is required to mitigate the event.

Systems Operation

Credited DL functions:

DL2:

- DL2-01 – Maintains Target Pressure
- DL2-42 – Anticipatory Hydraulic Scram on Sustained Low FW Flow
- DL2-31 – ICS Pressure Control on High Reactor Pressure

DL4:

- DL4a -12 – MSRIV/MSCIV Isolation on Sustained Low FW Flow

Core and System Performance

Input Parameters and Initial Conditions

The event is simulated by initiating a conservatively fast isolation of all FW flow. The initial conditions and plant parameters are provided in Tables 15.5-4 through 15.5-6. The initial CPR is conservatively biased low, and the hot rod power is conservatively biased high.

The analysis is performed using an equilibrium core design. The event is run at BOC, MOC, and EOR cycle time in cycle conditions.

Results

The CCF FW isolation event is presented in Figures 15.5-157 through 15.5-162, and the results are presented in Table 15.9-2 for the time in cycle with the limiting PCT response. The loss of FW flow results in a reduction of vessel inventory, causing the power and RPV water level to decrease. RPC maintains reactor pressure. Reactor scram and MS isolation occurs based on sustained low FW flow. ICS pressure control initiates based on high RPV pressure.

The long-term core cooling capability is conservatively assured by meeting the DBA acceptance criteria for fuel cladding and RCPB. The reactor integrity is conservatively assured by meeting the DBA event RCPB pressure criteria. The acceptance criteria are provided in Section 15.3 of NEDC-34181P. Although the DEC acceptance criteria are described in Section 15.3.2, the results are conservatively compared to the DBA acceptance criteria (described in Section 15.3.1) and demonstrate significant margin. The cladding temperature remains well below the temperature at which significant oxidation occurs from metal water reaction.

Barrier Performance

The effect of this event does not result in any temperature or pressure transient in excess of the DBA derived acceptance criteria for the fuel, pressure vessel, or containment. No fuel failures occur because there is no significant cladding temperature increase. Therefore, these barriers maintain their integrity and function as designed.

Radiological Consequences

DEC events do not have event-specific radiological acceptance criteria. The effects of DEC events are considered in NEDC-34184P (Reference 15.5-6)).

15.5.6 Analysis of Design Extension Conditions with Core Melting

This section concerns Severe Accident Analysis. A detailed methodology is to be provided in support of future work, and additional analysis work will be conducted, and the results reported in a future version of the safety case. FAP PSR15-4 pertains.

Introduction

The SAA:

- Uses best-estimate models and assumptions
- Takes credit for realistic system action and performance beyond original intended functions
- Takes credit for realistic operator actions

The analysis provides conformity with the acceptance criteria in Table 15.3-3 of NEDC-34181P. To achieve the appropriate level of confidence that these acceptance criteria are met, the safety analysis:

- Are performed by qualified analysts in accordance with approved QA process
- Apply a systematic analysis method
  - Use verified data
  - Use justified assumptions
  - Use verified and validated models and computer codes
  - Build in a degree of conservatism
  - Subject to a review process

There is a strong relationship between the deterministic SAA modeling of severe accident sequence progression and the Level 2 PSA described in NEDC-34184P (Reference 15.5-6). Deterministic modeling is used to confirm that the Level 2 containment event categories and release categories are valid.

Severe Accident Sequence Selection

SA sequences are selected using both deterministic and probabilistic methods. The selection process is summarised below:

1. The identification of the potential core damage sequences is initially based on historical experience with BWR severe core damage scenarios, engineering judgement and system understanding of the BWRX-300 reactor.
2. In parallel, Level 1 PSA is conducted to determine the sequences contributing to core damage frequency using event trees for modelled initiating events. The Level 1 PSA evaluates plant events, including BDBAs to identify failures and determine if core damage occurs.
3. Level 2 PSA is conducted to evaluate Level 1 core damage sequences to assess the containment response, determine radioactive release magnitude and timing, and to calculate small release frequency and large release frequency. The Level 2 PSA uses Containment Event Trees (CETs) to evaluate the core damage sequences from the Level 1 PSA.
4. The end states of the CETs are organised into discrete release categories, which are a representation of quantity, timing, and release path taken. The frequency of occurrence for sequences associated with each release category is estimated. This is supported by expert elicitation and various analytic tools.
5. Representative core damage sequences to be modelled in the deterministic SAA are identified for each release category. Corresponding plant damage states for each sequence, which represent the plant conditions at the onset of core damage, are defined as input to the deterministic modeling.

The outputs of the Severe Accident Sequence Selection include:

- Documentation of the selection process, including details of the criteria and methods used for selection of representative core damage sequences from the Level 2 PSA.
- The list of selected event sequences.
- Definition of the plant scenario corresponding to each selected sequence in terms of the initial plant mode, the initiating failure, additional mitigation failures and the corresponding plant conditions at the onset of core damage.
- Identification of existing, or specification of new relevant and feasible design requirements that support Fundamental Safety Properties provisions for DL1; these design requirements should explicitly validate any assumptions in the analysis and underpin justifications made during the selection process.

Severe Accident Analysis

Accident progression analyses for representative cases are performed to obtain the data for the development of CETs such as plant thermal-hydraulic behaviour, chronology of accident progression (the timing of the core damage and the containment failure), and containment loads from SA phenomena. Accident progression analyses include cases for both failure and success of mitigation systems.

Accident progression is modelled with the SAA software code Modular Accident Analysis Program (MAAP) (Reference 15.5-49). This analysis includes models for the important accident phenomena that might occur in the Reactor Pressure Vessel, in the containment, and in the reactor building. MAAP calculates the progression of the postulated accident sequence, including the deposition of the fission products, from initiating events to either a safe, stable state or to an impaired containment condition (due to overpressure or overtemperature).

MAAP calculates the factional release of material in the core. Therefore, to determine the activities of radionuclides releases the total inventory in the core needs to be determined. Radionuclide inventories are defined based on BWRX-300 neutronic analysis which identify the isotopic content of the core at various point in time (beginning, middle and end of life), and thermal-hydraulic analyses which establish the conditions in the fuel, cladding and coolant that determine the migration of specific isotopes through the various barriers to release.

A set of ‘release categories’ are defined to represent the spectrum of possible SA sequences and other release scenarios. For the SA sequences, these are a combination of the magnitude and timing of core damage and the specific release path taken. Representative scenario/conditions for each release category are defined based on the definition of the release category. MAAP calculates the activities of radionuclides released to the environment which along with other parameters that affect the subsequent behaviour in the environment – such as physical and chemical form, energy associated with the release, and height of the release – which make up the source term for each release category. These release categories with their associated source terms which will be the input to the Level 3 PSA.

Event timings from the accident progression analyses are used as inputs to time available for implementation of SAMGs and emergency response procedures. Insights from the SAA will provide inputs to the accident management and emergency preparedness planning as described in PSR Ch. 19 (Reference 15.5-23).

The SA sequences exceeding the radioactive release thresholds for small and large releases, which must be practically eliminated, are considered for design mitigation including identification of complementary design features.

Mitigation of SA sequences

Risk reduction design features are provided to cope with SA sequences. Evaluations are carried out on the capability of risk reduction design features to cope with DECs. The DEC risk reduction design features are described in PSR Ch. 15.9 Appendix D (Reference 15.5-13).

As the design advances to incorporate these mitigations, the analyses will be repeated to ensure results meet the established safety goals.

SA Sequences Considered for Practical Elimination

The BWRX-300 Safety Strategy incorporates the concept of practical elimination as discussed in IAEA SSR-2/1, “Safety of Nuclear Power Plants: Design,” (Reference 15.5-56) which states plant event sequences that could result in high radiation doses or in a large radioactive release have to be ‘practically eliminated.’ As a result of the adequate implementation of DL1, DL2, DL3, DL4a and DL4b features and functions, the likelihood is extremely low of an early or large off-site radioactive release that could potentially result from many PIEs and event sequences. However, these PIEs and event sequences are mitigated by reasonably practicable means (the application of defence-in-depth); therefore, a specific practical elimination claim is not made relative to these PIEs and event sequences.

The aim of the practical elimination concept is to complement the adequate implementation of defence-in-depth. Focused analysis is used to identify specific failures or plant conditions which cannot be practicably mitigated by application of defence-in-depth, and which could lead to unacceptable radiological. When such instances are identified, a specific practical elimination claim is required to substantiate that they are extremely unlikely to arise, with a high degree of confidence.

Practical elimination is achieved by:

- Demonstrating the possibility or occurrence of specific failures or plant conditions, which could lead to an early or large radioactive release, is extremely unlikely to arise with a high degree of confidence. For example, by using design provisions that have a reliability such that their failure does not need to be postulated or mitigated (e.g., the RPV).
- Applying a practical elimination claim only for failures or conditions which cannot be mitigated by reasonably practicable means.
- Providing supporting demonstration which includes elements of engineering judgement, deterministic analyses, and probabilistic assessments. Probabilistic insights should be used in support of deterministic and engineering analyses. Meeting a probabilistic target alone is not a justification to exclude further deterministic and engineering analyses and possible implementation of additional, practicable preventative measures.

The following five general types of SA sequences are considered for practical elimination:

- Events that could lead to prompt reactor core damage and consequent early containment failure, such as:
  a. Failure of the RPV
  b. Uncontrolled reactivity accidents
- SA sequences that could lead to early containment failure, such as:
  c. Highly energetic direct containment heating
  d. Large steam explosion
  e. Explosion of combustible gases, including hydrogen and carbon monoxide
- SA sequences that could lead to late containment failure, such as:
  f. Base mat penetration or containment bypass during molten core containment interaction
  g. Long-term loss of containment heat removal
  h. Explosion of combustible gases, including hydrogen and carbon monoxide
- SA with containment bypass, such as:
  i. LOCA with the potential to drive the leakage outside of the containment via supporting systems (e.g., interfacing system LOCAs). As the containment function may be jeopardised by the IE, an escalation to core damage is analysed and, where relevant, considered for practical elimination.
  j. Containment bypass consequential to SA progression
- SA in which the containment is open (plant in a shutdown state):
  k. Fuel damage in a storage fuel pool and uncontrolled releases

Outputs from the SAA

Outputs from the SAA will include:

- DL1 design requirements specified based on assumptions made during SAA.

- Documentation describing the analytical methods, approach, and assumptions used in the analysis.
- Documentation demonstrating a robust uncertainty evaluation approach.
- For each sequence analysed, documentation recording:
    a. Description of the sequence of events
    b. Results with respect to small and large release thresholds associated with the quantitative plant safety goals
    c. Graphs/plots of the key parameters during each sequence
    d. Definition of the end-state conditions for each sequence.
- Identification of scenarios requiring a practical elimination claim.
- Summary of insights into the effectiveness of those design features provided for severe accident mitigation.
- Summary of insights into important human actions for consideration during development of accident management procedures.

The results of the SAA will include:

- Information describing the analysis method and assumptions.
- Documentation demonstrating a robust uncertainty evaluation approach.
- A description of analysis results and conclusions with respect to radioactive release magnitudes compared to radioactive release thresholds for small and large releases.

SAA and design considerations to mitigate large and small releases are part of an iterative process as the design of the BWRX-300 progresses. Updates to the analysis documentation is made when new results become available.

15.5.7 Analysis of Postulated Initiating Events and Accident Scenarios Associated with the Spent Fuel Pool

This section concerns the deterministic analysis of initiating events associated with the spent fuel pool. There is currently no deterministic analysis presented for such events nor, with the exception of fuel handling events [discussed in the next section], other non-reactor events. Some regulators and future licensees / operators may require deterministic analysis to be presented to assist in the safe management of specific events. FAP item PSR15.5-30 pertains. The requirements of such analyses are often specific to different sites, countries, and regulatory regimes. FAP items PSR15.5-31, PSR15.5-32, and PSR15.5-33 pertain.

The analysis of initiating events associated with the fuel pool are DECs analysed in the Level 1 PSA described in NEDC-34184P (Reference 15.5-6).

15.5.8 Analysis of Fuel Handling Events

This section concerns the analysis of Fuel Handling Events. A single bounding Fuel Handling Accident has been analysed. Some regulators and future licensees / operators may require further analysis to be presented to assist in the safe management of specific events. FAP item PSR15.5-30 pertains. Site-specific assumptions (e.g., atmospheric dispersion factors) will have to be addressed for the relevant site. FAP item 15.5-31 pertains. The assumptions behind radiological calculations (e.g,. fission product release from breached fuel in a FHA) are often specific to different countries and regulatory regimes and will have to be shown to be applicable to the relevant site. FAP item PSR15.5-33 pertains. The criteria used to judge the acceptability of the residual radiological consequences are often specific to different countries and regulatory regimes and will have to be shown to be applicable to the relevant site. FAP item PSR15.5-32 pertains.

The Fuel Handling Accident (FHA) is categorised as a non-reactor group DBA event.

Postulated Initiating Event

The event occurs because of a failure of the fuel assembly lifting mechanism, resulting in the drop of a raised irradiated fuel assembly on the top of the reactor core or into the fuel pool storage racks. The dropped irradiated fuel assembly results in cladding failure in the dropped and impacted bundles and a subsequent release of fission products from the water in two distinct release phases. For the purpose of the PSR, the radiological release consequences are determined for the Main Control Room (MCR).

Off-site consequences are a site-specific assessment, including radiological release consequences for the Exclusion Area Boundary (EAB) and Low Population Zone (LPZ) and will be completed in future work.

Sequence of Event

The event sequence summary:

- Reactor is shut down for refuelling operations that begins 24-hours after shutdown
- During refuelling operation, a fuel assembly is moved over the top of the core or the fuel pool and fuel bundle, grapple, mast, and head fall on top of the core or the spent fuel racks
- Rods in the dropped bundle and impacted bundles fail, releasing fission gases and caesium iodide in the plenum and gap of the damaged rods to the reactor or fuel pool water that initiates a 2-phase fission product release
- In Phase 1 of the release that occurs over the first two hours after the event begins, fission gases rise through reactor or fuel pool water to refuelling operation floor common airspace surrounding the top of the reactor cavity or fuel pool
- The Phase 1 fission gases initially released to the refuelling floor are released to the environment
- The Phase 2 release begins two hours after event initiation, where the caesium iodide initially released to the pool evolves to form elemental iodine. Given the chemical conditions in the fuel pool, a portion of the elemental iodine becomes volatile and releases to the refuelling floor airspace and subsequently to the environment.

To prevent the spread of a fission gas release from a FHA to rooms adjacent to the RB fuel handling area operating deck, the RB truck bay door shall be closed during movement of irradiated fuel bundles. Administratively closing the RB truck bay door during movement of irradiated fuel bundles also ensures that the FHA release to the environment occurs from the RB as a diffuse source release.

The event sequence for the postulated FHA is provided in Table 15.5-34.

Methodology

The dose consequences of the postulated FHA are calculated using the RADTRAD Version 3.10 computer code. RADTRAD shows compliance with nuclear plant siting criteria for the radiation doses at offsite and Control Room receptors for various LOCA and non-LOCA DBAs.

The fission product release from the breached fuel resulting from the FHA is based on RG 1.183, Revision 0, Regulatory Position 3.2, “Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors,” (Reference 15.5-38), and the estimate of the number of fuel rods breached, including the fuel power and burnup limitations specified are based upon RG 1.183, Footnote 11. Radionuclides considered include xenon, krypton, halogens, caesium, and rubidium. The chemical form of radioiodine released from the fuel to the fuel pool is assumed to be 95 percent caesium iodide (CsI), 4.85 percent elemental iodine (I2), and 0.15 percent organic iodide such as CH_3I. All the gap activity from the damaged rods is assumed to be released over two Phases:

Phase 1: The instantaneous release from the rising bubbles of fission gas.

I_2 and organic iodine are conservatively assumed to be in vapor form and subsequently decontaminated by passage through the overlying pool of water into the building atmosphere. The retention of noble gases in the water in the fuel pool or reactor cavity is negligible (i.e., decontamination factor of 1). Particulate radionuclides, such as CsI, released from the fuel are assumed to be retained by the water in the fuel pool or reactor cavity (i.e., infinite decontamination factor).

Phase 2: The projected release of CsI re-evolving as I2.

CsI is conservatively assumed to completely dissociate into the pool water. Due to the low pH of the pool water, CsI (and Phase 1 absorbed I2) then re-evolves as I2 into the building atmosphere.

Identification of Operator Actions

No operator action is credited.

Fuel Damage

Because of the complex nature of the impact and the resulting damage to fuel assembly components, predicting the number of failed rods is not possible. For this reason, a simplified energy approach is taken. Numerous conservative assumptions are made to assure that the number of failed rods is conservatively analysed.

The key assumption for the FHA is that during a refuelling operation, a fuel assembly is moved over the top of the reactor core or fuel pool. While the fuel grapple is in the over-hoist condition with the bottom of the assembly at the maximum height allowed when using the fuel handling equipment, the main hoist cable and a redundant cable fails. A drop height of 67.4 ft (20.544 m) above the core is assumed as it is bounding with respect to plant design. The grapple head and mast are fixed vertically to the dropped assembly so that all the kinetic energy is transferred through the dropped assembly to the group of impacted assemblies. The dropped assembly impacts the core at a slight angle and the rods in this assembly might bend. After the assembly impacts the core, the assembly, grapple head and mast fall onto the core horizontally without contacting the side of the pressure vessel or fuel pool.

Fuel rod failure is assumed at 1% circumferential strain. The associated axial strain is $^{0.01}/_{\nu}$ where ν is Poisson's ratio, plastic deformation is assigned a value of 0.5, and the energy per rod failure is expressed by **Equation EF**:

$$E_f = \sigma_y \times \varepsilon \times V$$

Where:

E_f = energy per rod failure

σ_y = yield stress

ε = axial cladding strain

V = volume of fuel cladding

Kinetic energy is calculated for the dropped fuel bundle that accounts for the influences of buoyancy and resistance from the reactor cavity pool water. Finite element analysis simulations are used to determine the kinetic energy based on the drop distance in air or water. The simulation results revealed that when the drop distance of a fuel bundle in air is greater than 7.5 ft (2.3 m), the kinetic energy of the fuel bundle drop in water is less than 50% of that in air. When the bundle drop height is 34.0 ft (10.4 m) the energy is approximately 22% of that in air. This analysis credits a 50% reduction in the kinetic energy of the dropped bundle although the limiting case drop height correlates to a larger reduction.

The fuel assembly wet weight is 600.03 lbf (272.17 kgf), and the mast wet weight is 430.00 lbf (195.04 kgf). Applying the 50% kinetic energy reduction to the fuel assembly due to dropping through water is expressed by **Equation E1**:

$$E_1 = \frac{h_{drop} \times \left(W_{fuel} + W_{mast}\right)}{2}$$

Where:

E_1 = energy from initial drop

W_{fuel}= weight of fuel bundle

W_{mast}= weight of refuelling mast

h_{drop}= drop height

Substituting numerical values into Equation E1 yields 34,712.01 ft-lbf (4,799.2 kgf-m).

It is assumed that half of the energy is absorbed by the cladding. The ratio of the cladding to the non-fuel mass for GNF2 fuel is 0.4997. The energy per rod failure using the methodology described above (see Equation EF) is 271.26 ft-lbf/rod (37.503 kgf-m/rod). Therefore, the number of failed rods in the impacted assemblies from the initial drop is:

$$\frac{(50\%)(4{,}799.2\ kgf\text{-}m)(0.4997)}{37.503\ kgf\text{-}m\ /\ rod} = 31.97\ rods\ or\ 32\ rods$$

Additional energy is generated in a secondary impact as the bundle falls over from a vertical orientation to a horizontal orientation, and damages additional rods in the impacted bundles. The fuel bundle is assumed to have a height of 160.0 in. (4.064 m). Incorporating the 50% reduction due to the kinetic energy in water is expressed as:

$$E_2 = 50\% \times (h_{fuel} \times W_{mast} + \frac{1}{2} \times h_{fuel} \times W_{fuel})$$

Where:

E_2 = energy of dropped bundle and mast from secondary impact

h_{fuel} = height of refueling mast

Fifty percent of the kinetic energy is absorbed by the impacted assemblies resulting in the number of failed rods in the impacted assemblies from the secondary impact is five rods.

All the rods in the dropped assembly, which contains 92 rods, are assumed to fail and to be full length rods. GNF2 fuel assemblies contain both full-length and part-length rods, resulting in 85.6 effective full-length rods per bundle when accounting for the difference in lengths. Aside from the dropped assembly, the length of the damaged fuel rods is not considered in this evaluation. All the damaged rods are conservatively assumed to be full-length rods.

The number of failed rods used in determining the radiological consequences is:

$$92\ \text{rods} + 32\ \text{rods} + 5\ \text{rods} = 129\ \text{failed rods} = 1.507\ \text{failed fuel bundles}$$

There are 240 fuel bundles in the BWRX-300 core. The fraction of the core damaged in an FHA is determined by:

$$Core\ Damage\ Fraction = \frac{1.507\ bundles\ damaged}{240\ bundles\ in\ the\ core} = 6.28\text{E-}03$$

Core Inventory of Isotopes

A BWRX-300 core inventory of isotopes is calculated using the Oak Ridge National Laboratory code ORIGEN2, Version 2.1 (Reference 15.5-37) and the BWRUE.LIB cross-section library in units of Ci/MWth for the bulk operating parameters in Table 15.5-35.

A subset of the more than 700 isotopes from this inventory are used to model DBA dose consequences. The 60 isotopes used for DBAs are the dominant contributors to immersion and inhalation doses from airborne activity released during a DBA. This set of nuclides consist of 54 isotopes identified in WASH-1400 (NUREG-75/014) and 6 isotopes identified in SAND-85-2575 (NUREG/CR-4467). This is the group of isotopes typically used for Alternate Source Term (AST) dose evaluations.

The BWRX-300 reactor is subcritical for at least 24-hours prior to initiating refuelling operations. The BWRX-300 core inventory of the 60 dose-significant isotopes after 24-hours of decay are shown in Table 15.5-36 with an uncertainty of 1.02 applied to the gap inventory of the damaged fuel to comply with RG 1.183 (Reference 15.5-38), Section 3.1.

Gap Fractions

For events that are non-LOCAs where some fuel damage is postulated like the FHA, the fractions of the core inventory assumed in the fuel rod gap region for the various radionuclides are taken from RG 1.183 and reported in Table 15.5-37. These gap fractions are subject to the power and burnup limitations specified in RG 1.183 (Reference 15.5-38), Footnote 11.

Radial Peaking Factor

The radioactive material available for release in an FHA is assumed in the analysis to come from assemblies with peak inventory. To simulate this assumption, the inventory is scaled up by the maximum power Radial Peaking Factor (RPF). This represents the maximum achievable operational power history immediately preceding shutdown. Based on prior experience with GNF2 cores, a representative RPF value of 1.70 is assumed for this analysis.

Activity Released from the Fuel

All particulate isotopes are retained by the water in the fuel pool or reactor cavity water. Thus, only the noble gases and the gaseous form of iodine are available to escape the water. In accordance with RG 1.183 (Reference 15.5-38) and “Alternative Fuel Handling Accident Transport Methodology,” (Reference 15.5-40), the gaseous activity of iodine present in the fuel rod gap is in the form of 4.85% elemental and 0.15% organic iodine. In Phase 1 of the FHA, only 5% of the iodine in fuel rod gap is considered. The remaining 95% in the form of caesium iodide is retained in the pool water and is accounted for in the Phase 2 release. The total activity initially released from the fuel in Phases 1 and 2 is the mathematical product of the activity, core power (Ef), RPF, power uncertainty factor, gap fractions (Table 15.5-37), and core damage fraction. With this information, the release from the fuel is calculated as shown in Table 15.5-38.

Pool Scrubbing (Decontamination)

Credit is taken for retention of some released fission gas in the water after it is released from the damaged fuel rods. Because the depth of water above the reactor cavity pool and fuel pool is greater than 23.0 ft, the Decontamination Factor (DF) models from the Alternative Fuel Handling Accident Transport Methodology (Reference 15.5-40) are conservatively applied without correction. The alternative iodine transport model described in the Alternative Fuel

Handling Accident Transport Methodology (Reference 15.5-40) evaluates the FHA released from the pool in two distinct Phases:

1. Phase 1 models the initial, instantaneous, release of the gaseous activity available in the fuel rod gap to the airspace above the refuelling pool. This Phase accounts for the release of the noble gases and the gaseous elemental, and organic forms of iodine initially released from the damaged fuel rods.
2. Phase 2 models the subsequent release of elemental iodine that slowly re-evolves from the disassociation and volatilisation of the initially released caesium iodide (CsI) and from the elemental iodine retained in the pool because of the Phase 1 release.

A new model for the overall iodine DF applied to the iodine release in Phase 1 is taken from the alternative iodine transport model described in Reference 15.5-40.

In the Phase 2 release, the volatilisation rate of I2 is used to determine the rate at which the I2 is released from the pool to the airspace above the pool. In this Phase, essentially no further retention of I2 is assumed. The DF is 1 in the transfer of I2 from the water to the airspace above the pool during Phase 2.

Chemical Species

Consistent with RG 1.183 (Reference 15.5-38) and Reference 15.5-40 guidance on FHA, the chemical form of the iodine released from the fuel to the pool water is:

- 95% CsI
- 4.85% I2
- 0.15% organic iodine (methyl iodine)

An overall DF is applied uniformly to the total of the I2 and organic iodine release during Phase 1 while the CsI is retained by the water. Thus, the chemical species of the iodine released from the pool to the airspace above the pool during Phase 1 is in the following form:

- 0% CsI
- 97% I2
- 3% organic iodine (methyl iodine)

In Phase 2, the only iodine released from the pool is in the form of I2, either from the I2 retained in the water initially in Phase 1 or that which evolves from CsI. Therefore, the chemical form of the iodine that is released to the airspace above the pool during Phase 2 can be characterised as:

- 0% CsI
- 100% I2
- 0% organic iodine (methyl iodine)

Other radionuclides such as caesium and rubidium are considered. However, they are assumed to be permanently retained by the water either in particulate form or in solution. This is because there are no mechanisms to cause these radionuclides to evolve from the pool and be released in an airborne form.

Phase 1 Release from the Pool: Initial Gaseous Release and Water Depth

The depth of water is greater than 23 ft, but assumed to be 23 ft, to allow for use of the alternate DF method. The overall iodine DF for the Phase 1 release, DF1, is calculated based on the bubble size and rise time through the fuel pool. The bubble size and rise time are a function of the fuel pin pressure. The Phase 1 DF1 is computed using a best estimate fuel pin pressure per Reference 15.5-40. Therefore, a conservative pin pressure of 1200 psig is conservatively applied that corresponds to the conditions applicable to the FHA scenario. The DF1 takes into consideration the buildup of fission gas in the fuel rod over the time in cycle life as well as the actual temperature condition of the pool water during the FHA. The calculation of the Phase 1 DF1 is given by **Equation DF1**:

$$DF_1 = 81.046 \times e^{0.305\ t/d}$$

Where:

DF_1 = decontamination factor

t = bubble rise time (seconds)

d = bubble diameter (cm)

The bubble rise time and diameter are calculated using the following Equations:

$$t = 9.2261 \times e^{-6E-4 \times x}$$

$$d = -0.0002 \times x + 1.0009$$

Where:

x = internal rod pressure (psig)

The bubble rise time, bubble diameter and DF1 are calculated using a bounding fuel rod pressure and the results are shown in Table 15.5-39c. The DF1 value of 490, is applied to the total iodine released in Phase 1 (5% of the total iodine released to the pool).

Phase 2 Release from the Pool: Re-evolution Phase

During Phase 2 the CsI initially released to the pool water is assumed to disassociate and dissolve in the water. This is assumed to occur instantaneously after release. The dissolved iodine subsequently evolves due to chemical reactions with hydrogen peroxide to form elemental iodine I2.

The I2 formed from the CsI (95% of the total iodine released to the pool) is in addition to the I2 initially retained by the pool in Phase 1. The amount of iodine retained by the pool in Phase 1 is greater than 99% of the iodine released to the pool in Phase 1 (i.e., 1-1/490). As a result, about 5% of the total iodine released from the fuel rod gap inventory in Phase 1 is absorbed in the pool, and available for Phase 2 release. The iodine that remains in the pool for Phase 2 is released directly to the environment using the re-evolution rate.

The Phase 1 activity released from the surface of the reactor cavity pool is shown in Table 15.5-39a and the Phase 2 activity released from reactor cavity pool is shown in Table 15.5-39b.

Transport in the Reactor Building

The radioactivity released from the reactor cavity pool is assumed to mix instantaneously with the free air volume of the refuelling outage floor and crane area, which is the intermediate volume between the FHA release from the water and the environment. The RB is not isolated until after the FHA occurs and there are no mechanical means to ensure the refuelling outage floor and crane area airspace is well mixed and confined. As a result, no credit is taken for holdup and retention by the RB.

The analysis is not sensitive to the building volume because the release occurs over a 2-hour duration per RG 1.183 (Reference 15.5-38), Appendix B in Phase 1, and is instantaneously released to the environment in Phase 2 as the I2 volatises.

Release Assumptions

Phase 1

The Phase 1 no holdup release transport to the environment, simulated transport of 99.9% of the available activity for transport to the environment in 2-hours.

To simulate the leakage to the environment, the following equation is used to calculate a corresponding leakage rate.

This rate is calculated using the Equation below, and setting time (variable “t”) to 120 minutes as follows:

$$\frac{C(t)}{C_{ss}} = e^{-Qt/V}$$

Where:

$C(t)$ = volumetric concentration at time t

C_{ss} = steady state concentration at the beginning of the release

Q = constant flow rate out of space (m^3 / min)

V = RB airspace volume (m^3)

t = Duration of the release period in minutes (minutes)

To achieve a 99.9% release (0.1% retention) from the RB to the environment within 120 minutes, the following values are assumed:

$C(t)/C_{ss}$ = 0.001 (fraction retained in the RB)

V = 2.83E+02 m^3 (1.0E+04 ft^3)

t = 120 minutes (2 hours)

Substituting these values and solving for Q in Equation $C(t)/C_{ss}$ results in a Phase 1 flow rate of 5.753E+02 ft^3/min (1.629E+01 m^3/min). This Phase 1 flow transports 99.9% of the contamination released to the RB refueling outage floor and crane area and is transported and released to the environment in two hours.

Phase 2

The Phase 2 iodine transport parameters are taken from NUREG/CR-6928, “Industry-Average Performance for Components and Initiating Events at U.S. Commercial Nuclear Power Plants,” (Reference 15.5-59). A portion of the I2 inventory volatilises due to the chemistry conditions present in the water, and slowly releases from the water without reduction. The rate of release or evolution is characterised by the evolution removal coefficient that is effectively the rate of release of the volatised I2 from the pool to the airspace above the pool.

The evolution removal coefficient is converted to a volumetric flow rate, Qe(L/s) for use in RADTRAD resulting in a volumetric flow rate of 3.77E-03 cfm.

Dose Calculation

Because the inventory of isotopes released from the surface of the reactor cavity pool is already determined in Tables 15.5-39a and 15.5-39b, the RADTRAD model used to calculate the dose consequences is a simple two compartment model that simulates transport from the RB or fuel pool to the environment.

Breathing Rates

The postulated FHA breathing rates used for the MCR is consistent with RG 1.183 (Reference 15.5-38).

Decay and Daughtering

This analysis assumes a decay time of 24-hours prior to the removal of spent fuel during refuelling, and credit for this decay period is taken in the initial core inventory Table 15.5-36. Decay and daughtering of nuclides during the FHA transport are credited in the RADTRAD model for the duration of the event (i.e., 30 days).

Dose Conversion Factors

Dose Conversion Factors (DCFs) used for inhalation of radioactive material comply with Federal Guidance Report (FGR) 11, “Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion,” (Reference 15.5-42). DCFs used for submergence in a semi-infinite cloud are taken from FGR 12 “External Time in cycle to Radionuclides in Air, Water, and Soil,” (Reference 15.5-43).

Dispersion in the Environment

Modeling of dispersion into the environment requires detailed site information, and therefore not provided at PSR.

Main Control Room Model

Volume

The MCRE has an approximate volume of 48,125 ft^3 (1,363 m^3) and is serviced by the CB HVS. To account for equipment and structures in the MCRE, this volume is conservatively reduced by 20% to a free-air volume of 38,500 ft^3 (1,090 m^3) for this analysis.

Unfiltered MCR Intake Flow

In the event of a FHA, the CB HVS will automatically switch from supplying unfiltered outside air through the CB normal ventilation to emergency filtration of the outside air supply. However, because the control room habitability is not a Safety Category 1 function in the BWRX-300 design, emergency filtration is not credited in the MCRE for mitigation of any DBA. This analysis does not take credit for filtration of the MCRE. Instead, the analysis models the CB HVS only for the purpose of propagating contamination into the MCRE.

The normal CB HVS intake flow rate and emergency CB HVS intake flow rate are the same. The supply of unfiltered outside makeup air that enters the MCRE is at the normal CB HVS flow rate of 1,870 ft^3/min (0.883 m^3/s).

In addition to the CB HVS flow, the potential exists for unfiltered in-leakage of outside air to enter the MCRE from ingress and egress of plant personnel or through small openings in the MCRE such as the bathroom fans, penetrations, and small cracks in the walls. No MCRE unfiltered in-leakage flow rate is established for the BWRX-300 MCRE because the control room habitability is not a Safety Category 1 function. The primary concern radiologically is that increased control room unfiltered in-leakage could result in the reactor operator being exposed to a larger dose than predicted.

To ensure this analysis conservatively bounds unfiltered in-leakage, the intake flow of contaminated outside air is assumed to be at an overall rate of one MCRE volume per minute for the duration of the event (720-hours). Thus, an unfiltered flow rate of 38,500 ft^3/min

(18.2 m^3/s) is assumed for 30 days, and the MCRE exhaust flow is assumed equal to the intake flow.

This flow rate is a factor of more than 20 times the normal intake flow rate ensuring the limiting MCRE operator dose estimate is established. Use of an MCRE unfiltered air intake rate that bounds the actual CB HVS flow rate ensures the results are applicable to situations where the MCRE intake flow may vary and bounds any potential changes to the CB HVS flow rate that might occur during the preliminary design phase of the facility. In addition, the conservatism in this flow rate assumption bounds any unfiltered in-leakage rate. Therefore, the assumed MCRE flow rate established by this design analysis does not require confirmation of the unfiltered in-leakage through testing as it represents an open volume with a maximised inflow and exhaust rate.

Results

The dose result is 1.7 mSv Total Effective Dose Equivalent (TEDE) for an unfiltered, unmitigated fault. This provides confidence for a future UK specific assessment.

15.5.9 Analysis of Radioactive Releases from a Subsystem or a Component

This section concerns the analysis of radiological releases. Four initiating events are analysed. Some regulators and future licensees / operators may require further analysis to be presented to assist in the safe management of specific events, or to explicitly demonstrate that the presented events are bounding. FAP item PSR15.5-30 pertains. Site-specific assumptions (e.g., atmospheric dispersion factors established) will also need to be addressed for a future site-specific safety case. FAP item PSR15.5-31 pertains. The assumptions behind radiological calculations (e.g., radiation concentrations in reactor coolant and steam) are often specific to different countries and regulatory regimes and will have to be shown to be applicable to the relevant site. FAP item PSR15.5-33 pertains. The criteria used to judge the acceptability of the residual radiological consequences are often specific to different countries and regulatory regimes and will have to be shown to be applicable to the relevant site. FAP item PSR15.5-32 pertains.

15.5.9.1 Analysis of Loss of Coolant Accidents Outside Containment

As discussed in Subsection 15.5.4.5, the scram and RPV isolation trips occur for the large breaks outside containment within the same time as breaks inside containment. For LOCAs outside containment, the dose consequence analyses are performed using the NUREG/CR-6604 RADTRAD computer code Version 3.10 and the transport radioactivity assumptions from Appendix D of RG 1.183 (Reference 15.5-38) to various offsite receptors.

For large breaks, timing of the break detection is less than 1 s for breaks outside containment that is the same as the time to reach the containment high pressure setpoint. Because reactor scram and isolation valve closures for breaks inside containment also occur for breaks outside containment, the event progression is no different for breaks outside containment than inside containment. For MS pipe breaks, the break flow rate calculated for breaks inside containment is also used for breaks outside containment because the MSCIV closure is not credited in the MS pipe break inside containment. For FW pipe breaks, the only difference between the pipe breaks inside and outside of containment is the closure of the Feedwater Containment Isolation Valve (FWCIV). For a FW pipe break outside containment, break flow includes flow from the reactor as well as the pipe inventory.

The IC pipe break outside containment is limited to the flow passing through the orifices in the steam distribution pipes. During normal operation prior to the break, condensate return valves are closed and remain closed. Isolation steam supply pipe has a guard pipe outside containment so that break flow in the supply pipe upstream of the orifice is not discharged outside containment. A break in the IC is followed by a discharge of the subcooled water in the supply pipe until the IC RIVs close on break detection.

Because the small break analyses inside the containment do not credit containment back pressure, the mass and energy release calculated for breaks inside the containment are bounding for breaks outside the containment.

The following analytic inputs, assumptions, and criteria are used for all breaks outside containment:

Source Term

There is no fuel damage resulting from the following postulated breaks outside containment:

- MSLB
- FWLB
- Isolation Condenser Line Break (ICLB)
- Instrument Line Break (ILB)

Because there is no fuel damage from any of the breaks postulated outside containment, the activity available for release from the break is the activity present in the reactor coolant when the break occurs. As a result, the source term is the same for all breaks outside containment.

Thermal hydraulic LOCA analyses are performed for both small and large break scenarios that comply with the following fuel rod acceptance criteria:

- Reactor level will not decrease below TAF, or
- Fuel cladding temperature will not exceed the normal operating temperature.

Compliance with these criteria ensures the integrity of the cladding fission product barrier is maintained for small and large pipe break scenarios in containment that are bounding for breaks both inside and outside containment.

The only activity available for release from the break is that which is present in the reactor coolant and steam when the break occurs. Radiation concentrations in BWRX-300 reactor coolant and steam adequate for use in design basis calculations (such as shielding, equipment design, etc.) are determined based on ANSI/ANS-18.1-2020 (Reference 15.5-33).

Using Appendix D from RG 1.183 (Reference 15.5-38), two cases are considered for conditions when the postulated accident occurs:

1. The concentration that is the maximum iodine pre-accident spike value, typically, 4.0 µCi/g dose equivalent I-131, permitted under plant operating limits and conditions.
2. The concentration that is the maximum equilibrium value, typically 0.2 µCi/g dose equivalent I-131, permitted under plant operating limits and conditions.

Dose Conversion Factors

The DCFs used to convert design basis iodine concentrations to dose equivalent I-131 concentrations for all breaks outside containment are those from FGR No. 11, Table 2, Column “effective” (Reference 15.5-42).

TEDE doses are the sum of the Committed Effective Dose Equivalent (CEDE) from inhalation and the Deep Dose Equivalent (DDE) from external time in cycle. CEDE DCFs are taken from FGR No. 11 (Reference 15.5-42). The DDE DCFs used from FGR No. 12 (Reference 15.5-43) assume submergence in a semi-infinite cloud with appropriate credit for attenuation by body tissue. DCFs for Kr-89 and Xe-137 are taken from FGR No. 15, “External Time in cycle to Radionuclides in Air, Water, and Soil,” (Reference 15.5-44). The DCFs used in the analyses are based on data provided by the International Commission on Radiological Protection.

Reactor or Turbine Building Volume

The breaks outside containment analyses are not sensitive to the volume of either building as the release to the environment is instantaneous. The use of a large volume generates a faster flow rate of a more diluted source, and a small volume results in a slower flow rate of a more concentrated source term. In either case, the activity available for release to the environment is transported in one minute, and the flow rate applied is calculated based on the assumed volume to ensure an instantaneous release.

Breathing Rate and Occupancy Factors

The postulated breathing rates and occupancy factors used are taken from RG 1.183 (Reference 15.5-38) for onsite (MCRE) dose receptors.

Decay and Daughtering Nuclides

Decay and daughtering of nuclides are credited in the dose model for the 30-day duration of all the line breaks outside containment except for the small line break outside containment.

Main Control Room Model

The MCR model is the same for all breaks outside containment and is used in assessing the dose to the operators and the technical support centre personnel that are housed in the MCRE.

Volume

The MCRE has a volume of approximately 48,125 ft^3 (1,363 m^3) and is serviced by the CB HVS. To account for equipment, and structures this volume is conservatively reduced by 20% to a free-air volume of 38,500 ft^3 (1,090 m^3) for the analyses.

Unfiltered Intake Flow

In the event of a break outside containment, the CB HVS automatically switches from supplying unfiltered outside air through the CB normal ventilation to emergency filtration of the outside air supply. However, because the MCR habitability is not a Safety Category 1 function, emergency filtration is not credited for mitigation of any DBA. The analyses do not credit MCRE filtration. The analyses model the CB HVS only for the purpose of propagating contamination into the MCR.

The normal CB HVS intake flow rate and emergency CB HVS intake flow rate are the same. The supply of unfiltered outside makeup air that enters the MCRE is at the normal CB HVS flow rate of 1,870 ft3/min (0.883 m3/s).

In addition to the CB HVS flow, the potential exists for unfiltered in-leakage of outside air to enter the MCRE from ingress and egress of plant personnel or through small openings in the MCRE such as the bathroom fans, penetrations, and small cracks in the walls. No MCRE unfiltered in-leakage flow rate is established because the MCR habitability is not a Safety Category 1 function. The primary concern radiologically is that increased MCR unfiltered in-leakage could result in the reactor operator being exposed to a larger dose than predicted.

To ensure the analyses conservatively bound unfiltered in-leakage, the intake flow of contaminated outside air is assumed to be at an overall rate of one MCRE volume per minute for the duration of the event (720-hours). Thus, an unfiltered flow rate of 38,500 ft3/min (18.2 m3/s) is assumed for 30 days, and the MCRE exhaust flow is assumed equal to the intake flow.

This flow rate is a factor of more than 20 times the normal intake flow rate ensuring the limiting MCRE operator dose estimate is established. Use of an MCRE unfiltered air intake rate that bounds the actual CB HVS flow rate ensures the results are applicable to situations where the MCRE intake flow may vary and bounds any potential changes to the CB HVS flow rate that might occur during the preliminary design phase of the facility. In addition, the conservatism in this flow rate assumption bounds any unfiltered in-leakage rate. Therefore, the assumed MCRE flow rate established by the design analyses does not require confirmation of the unfiltered in-leakage through testing as it represents an open volume with a maximised inflow and exhaust rate.

No Holdup Release to the Environment Flow Rate and Release Duration

All activity is released from line breaks outside containment to the Turbine Building (TB) or RB (for ICLB and ILB events). The analyses are not sensitive to the TB or RB volume because the release is assumed instantaneously. No holdup releases to the environment are assumed with a total of 99.9% transport of the TB or RB airspaces to the environment over a period of one minute after the event at a flow rate of 6.91E+04 cfm (1.95E+03 m3/m). The flow rate applied ensures an instantaneous release and is applied for the duration of the event (720 hours). For the line breaks outside containment, the radioactivity in the coolant released in each event is transported to the atmosphere instantaneously without credit for plate out, holdup, or dilution within facility buildings.

Dispersion to the Environment

Modeling of dispersion into the environment requires detailed site information, and therefore not provided at PSR.

Acceptance Criteria

The criteria used to judge the acceptability of the residual radiological consequences are often specific to different countries and regulatory regimes and will have to be shown to be applicable to the relevant site. For the UK, the BWRX-300 design will require future consideration against UK specific acceptance criteria. FAP item PSR15.5-32 pertains.

15.5.9.1.1 Main Steam Line Break Outside Containment

Postulated Initiating Event

A MSLB accident occurs during power operation in the steam tunnel that is the interface between the RB and TB or in the TB. The postulated event assumes that a MSL instantaneously and circumferentially breaks outside containment and downstream of the outermost MSCIV. The plant is designed to immediately detect such an occurrence and initiate isolation of all MSLs, including the broken line. The energetic release of reactor steam and water from the break is assumed to blowdown into the TB airspace where it is released to the atmosphere instantaneously without credit for plate out, holdup, or dilution within the facility.

Mass Release

The total mass of coolant released is the amount in the line at the time of the break plus the amount that passes through the RIVs and the outboard CIV prior to closure. The masses of coolant and steam released from the postulated MSLB based on preliminary thermal hydraulic analysis are:

- Reactor Water Release = 29,659 lbm (13,453 kg)
- Reactor Steam Release = 13,241 lbm (6,006 kg)

Table 15.5-40 provides the activity release from this coolant volume.

This analysis conservatively assumes that the liquid released from the break flashes to steam and is available for transport to the environment along with the steam released from the break. The mass release duration from the MSLB is equal to the maximum closure time of the CIV of 10 seconds. A flashing fraction of 0.4 is conservatively applied to the liquid reactor coolant released as flashing steam.

Dispersion to the Environment

Modeling of dispersion into the environment requires detailed site information, and therefore not provided at PSR.

Results

The analysis results require comparison with suitable radiological acceptance criteria for the UK and will be provided in future work.

15.5.9.1.2 Large Feedwater Line Break Outside Containment

Flow from the RPV side of the break is bounded by FW breaks inside containment because of the longer pipe length. The higher-pressure losses occur for a break outside containment. No operator actions are required to mitigate the event.

Postulated Initiating Event

The FWLB accident occurs in the steam tunnel that interfaces between the RB and TB or in the TB. An instantaneous circumferential break of a FW line at a location downstream of the high-pressure feedwater heaters and upstream of the outermost CIV is conservatively assumed for the postulated event. The plant is designed to immediately detect such an occurrence and initiate FW line isolation. The energetic release of reactor water from the break is assumed to blowdown into the TB airspace where it is released to the atmosphere instantaneously as a ground-level release without credit for plate out, holdup, or dilution within facility. The mass release duration from the FWLB is equal to the maximum CIV closure time of 10 seconds.

Mass Release

The total mass of coolant released is the amount in the line at the time of the break plus the amount that passes through the valves prior to closure. The limiting mass release of coolant and steam from the postulated FWLB are:

- Reactor Water Release = 39,657 lbm (17,988 kg)
- Reactor Steam Release = 3,106 lbm (1,409 kg)

The mass release versus time is shown in Figure 15.5-167. The total FWLB source term activity released to the environment is shown in Table 15.5-41. A flashing fraction of 0.4 is conservatively applied to the liquid reactor coolant released.

Dispersion to the Environment

Modeling of dispersion into the environment requires detailed site information, and therefore not provided at PSR.

Results

The analysis results require comparison with suitable radiological acceptance criteria for the UK and will be provided in future work.

15.5.9.1.3 Shutdown Cooling System Line Break Outside Containment

The SDC (Reference 15.5-12) provides decay heat removal for refuelling or maintenance. The SDC provides decay heat removal at normal and lower reactor operating pressures. The SDC is subjected to high energy conditions for a short time (less than 2% of the plant operating conditions). The system piping is assigned a medium energy line due to the short time that it is subjected to high energy conditions.

The SDC connects to the FW system between the FW containment isolation and FW isolation control valve outside containment. Due to the smaller SDC piping diameter, the FW pipe break outside containment discussed previously in Subsection 15.5.9.1.2 bounds the SDC pipe break outside containment with respect to the dose consequence.

15.5.9.1.4 Large Isolation Condenser Line Break Outside Containment

The mass and energy release from the IC pipe breaks outside containment are still bounded by the IC pipe breaks inside containment. Breaks inside containment remain bounding because the IC pipe breaks do not utilise any DL3 functions that depend on containment parameters, and the containment back pressure is not credited in any of the IC pipe breaks inside containment.

As is the case for IC pipe breaks inside containment, core response is not a concern because the break is isolated rapidly. The consequences of large IC pipe breaks outside containment require an evaluation for the loads, pressures, and temperatures outside containment, and radiological consequences resulting from normal operation coolant activity.

Postulated Initiating Event

The ICLB accident analysed for offsite dose consequences is a postulated break of an ICS steam supply pipe in the ICS pool on the operating deck of the RB due to the largest mass release. The plant is designed to immediately detect such an occurrence, initiate isolation of the broken line within 5 seconds, and fully isolate the break in 10 seconds. Blowdown steam from the break is directed to the two heat exchangers in one ICS unit and the liquid mass in the heat exchangers is expelled to the ICS pool where it mixes with the pool water without flashing. Scrubbing in the ICS pool is conservatively ignored. The mass release duration from the ICLB to the RB is 10 seconds accounting for the 10 sec closure of the CIVs. The energetic release of reactor steam from the break is assumed released to the environment instantaneously as a ground-level release without holdup.

Mass Release

The limiting mass of steam released from the postulated ICLB is 6,684 lbm (3,032 kg). Table 15.5-42 provides the release source term for the ICLB.

Dispersion to the Environment

Modeling of dispersion into the environment requires detailed site information, and therefore not provided at PSR.

15.5.9.1.5 Small Breaks Outside Containment

Because the small break analyses inside containment do not credit containment back pressure, the mass and energy release calculated for breaks inside containment are bounding for breaks outside containment.

Dose analysis is performed for breaks outside containment for the instrument line break.

Instrument Line Break Dose Consequences

Postulated Initiating Event

A circumferential rupture of an instrument line connected to the primary coolant system is postulated to occur outside primary containment in the RB. The ILB analysis assumes that the event cannot be isolated, and no fuel damage occurs. The resulting activity is released to the environment directly from the RB with no credit for holdup or filtration. There is no identified specific event or circumstance that results in the failure of a small line.

Instrument lines penetrating primary containment are required to have an automatically operated CIV, one that can be manually operated from a remote location, or an Excess Flow Check Valve (EFCV) (see PSR Ch. 6 (Reference 15.5-13) for a description of EFCV usage). In addition, instrument lines penetrating containment are sized or include a flow restricting orifice to ensure that any breach of the line outside containment during power operation reduces leakage to the maximum extent practical and ensures the rate and extent of coolant loss are within the capability of the normal reactor coolant makeup system.

The ILB analysis assumes the ruptured instrument line is equipped with a 0.25 in (6.4 mm) flow restricting orifice and a passively actuated EFCV that is typically configured for use in operating BWRs. For evaluating the dose consequences of a small line rupture outside, the EFCV is conservatively treated as an active component that fails to isolate the break, resulting in radiological release to the environment. The results of this analysis are not applicable to instrument lines that are larger than 0.25 in (6.4 mm) in diameter or do not contain a 0.25 in (6.4 mm) flow restricting orifice.

The ILB leakage may result in noticeable increases in radiation, temperature, humidity, or audible noise levels in the RB or abnormal indications of actuations caused by the break. Termination of the break flow is dependent on operator action. The action is initiated upon the discovery of the un-isolatable leak. The action consists of the orderly shutdown and depressurisation of the reactor. Operator action is assumed to occur at 72-hours after the break occurs, after which a controlled shutdown of the reactor is assumed to occur over a 5.2-hour period.

Two cases are considered for coolant conditions that may exist when the postulated accident occurs:

1. The maximum equilibrium iodine concentration permitted for continued full power operation.
2. The iodine concentration corresponding to the conditions of an assumed pre-accident spike.

Mass Release

No credit is taken for operator action for the first 72-hours. After 72-hours, the control room operators begin a controlled shutdown of the plant that takes an additional 5.2 hours. NEDO-32708, Revision 1, "Radiological Accident Evaluation: NEDO-32708, "Radiological Accident Evaluation: The CONAC04A Code," (Reference 15.5-45), Figure 7-1 provides the fluid flow rate data for an ILB before (i.e., 10 minutes) and after controlled shutdown is initiated by the operators. This data is adapted to this evaluation and extends the operator action to 72-hours. The coolant mass release is 761,007 lbm (345,187 kg) calculated over the duration of the event.

The ILB Airborne Release Source Term for equilibrium and iodine spike is provided in Tables 15.5-43A and 15.5-43B, respectively. Only the equilibrium concentrations of iodine and iodine spike activity releases differ.

Flashing of Reactor Coolant to Steam

The total iodine fraction in the liquid that becomes airborne assumed equal to the leakage fraction that flashes to vapor (flash fraction (FF)) is determined using a constant enthalpy process:

$$FF = \frac{h_{f1} - h_{f2}}{h_{fg}}$$

Where:

h_{f1} = Enthalpy of coolant liquid release at different pressure and temperature conditions during shutdown

h_{f2} = Enthalpy at the heat of vaporisation at atmospheric pressure = 180.1 BTU/lbm (418.9 kJ/kg)

h_{fg} = Enthalpy at the heat of vaporisation at 100 °C (212 °F) = 970.2 BTU/lbm (2,256.7 kJ/kg)

NEDO-32708 (Reference 15.5-45), Figure 7-2 provides the reactor pressure during shutdown for the whole duration of the ILB event that is used to calculate the enthalpy of the saturated liquid (hf1) at different times during the shutdown. The mass of coolant that flashes to steam at 72-hours is 279,745 lbm (126,890 kg). The mass of coolant that flashes to steam calculated over the duration of the whole event is 286,627 lbm (130,012 kg). The ILB water concentrations and the mass of coolant flashed are used to calculate the airborne activity released.

Release Duration

Operator action is assumed to occur at 72-hours. After 72-hours, the reactor is taken through a controlled shutdown where it is assumed that the instrument line cannot be isolated and primary coolant continues to flow through the ILB. The controlled shutdown process is assumed to occur over a 5.2-hour period. The coolant mass release occurs over a 77.2-hour period (i.e., 72-hour operator action delay + 5.2-hour controlled shutdown), approximately.

Dispersion to the Environment

Modeling of dispersion into the environment requires detailed site information, and therefore not provided at PSR.

15.5.10 Analysis of Internal and External Hazards

The analysis of internal hazards is reported in PSR Ch. 15.7 – Deterministic Safety Analyses – Analysis of Internal Hazards of this PSR.

The analysis of external hazards is reported in PSR Ch. 15.8 – Deterministic Safety Analyses – Analysis of External Hazards of this PSR.

15.5.11 Deterministic Safety Analysis Results

This section presents the tabulated content of the deterministic safety analysis results in support of the PSR.

Table 15.5-1: Table Not Used

Table 15.5-2: Key Initial Conservative LOCA Evaluations

Parameter	Value	Notes
Thermal Power	887.4 MW	102% of rated power. Hot shutdown initial power is also included in steam pipe break cases
Dome pressure	7308.4 kPa	Upper end of normal operating range
Initial Feedwater (FW) temperature	241.9 °C for Main Steam Line (MSL) Break	Initial temperature of 191.9 °C is also included in FW pipe break cases
Initial temperature of the Isolation Condenser System (ICS) and reactor cavity pools	43.3 °C	
Initial containment pressure	119.7 kPa	Upper end of normal operating range
Initial containment temperature	43.3 °C	Lower end of normal operating range
Initial water level in downcomer	21.1 m (large breaks)	Lower initial level (0.152 m below normal level) is also evaluated for small breaks

Table 15.5-3: Summary of Core-Wide Decay Ratio Results

Cycle Exposure (GWd/ST)	Core-Wide Decay Ratio	
	Nominal (241.9 °C)	Loss of Feedwater Heating (LFWH) to 191.9 °C with Selected Control Rod Run In (SCRRI) mitigation (Anticipated Operational Occurrence (AOO) Conditions)
0 (Begining of Cycle (BOC))	0.56	0.53
4 (Middle of Cycle (MOC))	0.71	0.57
6 (End of Rated Cycle (EOR))	0.64	0.56

Table 15.5-4: Input Parameters and Initial Conditions and Assumptions Used in Non-LOCA Analyses

Parameter	Value
Heat Balance Related Parameters	
Rated thermal power level, MWt	870
Core flow, Mlbm/hr (kg/s) – analysis value	Calculated
Reference rated core flow, Mlbm/hr (kg/s)	15.0 (1890.0)
Steam flow, Mlbm/hr (kg/s) – analysis value	Calculated
Reference rated steam flow, kg/s	503.2
Feedwater (FW) flow, Mlbm/hr (kg/s) – analysis value	Calculated
Reference rated FW flow, kg/s	507.0
Nominal dome pressure, psia (MPa)	1040 (7.171)
Nominal FW temperature, °F	467
Normal Reactor Water Level (NWL), m	21.097 m from inside bottom of the RPV 12.22 m above TAF
Control Rod Drive / High Pressure Injection System Related Parameters	
Control rod position versus time	Table 15.5-4
Control Rod Drive Mechanism (CRDM) run-in fast speed (minimum requirement), mm/s	70
CRDM withdrawal (maximum speed), mm/s	28.0
MSL Related Parameters	
Number of MSLs	2
Minimum MSL length (average of all lines): flow path from vessel to Turbine Stop Valve (TSV), ft (m)	194 (59.0)
Minimum MSL volume (total of all lines including header and piping to TBV), cubic ft (cubic m)	885.3 (25.07)
Minimum MSL pressure difference between the vessel dome pressure and the turbine throttle pressure at rated conditions, psi (kPa)	30.0 (207)
Main Steam Reactor Isolation Valve (MSRIV) minimum closure time, s	3.0
MSRIV closure profile for minimum closure time, s 100% open area 100% open area 1% open area 0% open area	 0.0 0.6 1.7 3.0
Feedwater Related Parameters	
Number of motor-driven FW pumps (One FW pump is operating, and one FW pump is in standby during power operation)	2

Parameter	Value
Maximum flow demanded is between, % Rated	105 - 115
Maximum flow of both FW pumps assuming a Common Cause Failure (CCF), % Rated	220
FW temperature reduction for LFWH AOO, ° F (°C)	90 (50)
Minimum time constant of FW temperature response, s	60
FW pump trip coast down time constant, s	3
ICS Related Parameters	
ICS heat removal capacity per train, MW	33.75
ICS nominal initial temperature, K	294
ICS condensate return valve maximum opening time, s	10
ICS train condensate return line internal diameter, mm	177.9
The elevation difference from centerline of the horizontal connections between the ICS steam line distribution header to the centerline of the ICS return injection to the chimney, minimum, in (m)	622.362 (15.808)
Balance of Plant Related Parameters	
TSV and Turbine Control Valve (TCV) maximum fast closure steam flow shutoff rate from rated power, % rated steam flow / s	667
TCV maximum slow (servo) closure rate of change of steam flow shutoff from rated power % rated steam flow / s	40
Turbine Bypass Valve (TBV) capacity at rated conditions % rated steam flow	25
Miscellaneous Parameters	
Biased initial Critical Power Ratio (CPR), (reduction versus nominal initial CPR) Note: the value is approximate because it is slightly different depending on the initial condition. This is used in CN-PA events and some Design Extension Condition (DEC) events (as noted in event description)	~0.1

Table 15.5-5: CRD Scram Time

Reactor Vessel Bottom Gauge Pressure MPaG (psig)	Rod Insertion Position	Required Maximum Time (s) Note 1
≤ 1269 psig (8.75 MPaG)	10%	≤ 0.46
	40%	≤ 1.20
	60%	≤ 1.71
	100%	≤ 3.70
≤ 1375 psig (9.48 MPaG) Note 2	10%	≤ 0.56
	40%	≤ 1.40
	60%	≤ 2.03
	100%	≤ 4.20

Notes:

The times include 0.2 s delay from de-energising the scram pilot valves to control rods movement.

These times are used in pressurisation increase events in the Design Basis Accident (DBA) category even if the pressure is not greater than 1269 psig (8.75 MPaG) during the rod insertion.

Table 15.5-6: Defence Lines Inputs Used in Non-LOCA Analyses

Defence Line (DL) ID	Function Name	Inputs	Setpoint / Delay Analytical Limits
DL2-01	Maintain Target Pressure (Performed by Reactor Pressure Control (RPC))	Steam flow demand is a function of reactor dome pressure	Target dome pressure is normal dome pressure in Table 15.5-3 Design description of RPC is provided in Subsection 7.3.3.2, Item 5
DL2-02	Maintain Target Level (Performed by Reactor Level Control (RLC))	Reactor Pressure Vessel (RPV) level error (function of core power, reactor water level, steam flow, steam flow enthalpy, FW enthalpy)	Target reactor level is NWL (L5) in Table 15.5-3 Design description of RLC is provided in Subsection 7.3.3.2, Item 3
DL2-04	Control Rod Block on (Automatic Thermal Limit Monitor) ATLM	Determination of the thermal limits or soft duty guidelines violation	Design description of ATLM is provided in Subsection 7.3.3.2, Item 20
DL2-05	Control Rod Block on Multi-Channel Rod Block Monitor (MRBM)	Indication of potential fuel damage thermal limits being exceeded	Design description of MRBM is provided in Subsection 7.3.3.2, Item 21
DL2-08	Anticipatory Hydraulic Scram on Turbine Trip or Generator Load Rejection Demand	Turbine Trip or Generator Load Rejection Demand	Total time delay = 0.05 s after generator load rejection or turbine trip signal
DL2-09	TBV Fast Open on Turbine Trip or Generator Load Rejection Demand	Turbine Trip or Generator Load Rejection Demand	Before or at the same time as TCV or TSV closure
DL2-13	Turbine Trip on High Main Condenser Pressure Setpoint 2	High Main Condenser Pressure (HMCP) Setpoint 2	HMCP2 no greater than 30.5 kPaA Total time delay = 1 s
DL2-14	TBV Closure on High Main Condenser Pressure Setpoint 3	High Main Condenser Pressure Setpoint 3	HMCP3 no greater than 71.1 kPaA Total time delay = 1 s
DL2-21	Anticipatory Hydraulic Scram on MSRIV/ Main Steam Containment Isolation Valve (MSCIV) Position	MSRIV/MSCIV Position	<90% open Total time delay (scram) = 0.05 s
DL2-25	Start Standby FW Pump on Loss of Operating FW Pump	Loss of Operating FW Pump	Analysis allows 10 seconds to start standby pump. Analysis allows 15 seconds to ramp up the standby pump to full flow

Defence Line (DL) ID	Function Name	Inputs	Setpoint / Delay Analytical Limits
DL2-27	Select Control Rod Run-In on FW Temperature Decrease	FW Temperature Decrease	FW temperature reduction of 30°F (16.6°C) or more. Total time delay = 5 s
DL2-31	ICS Initiation on High Reactor Pressure	High Reactor Pressure	Settings are not important for Transient Deterministic Safety Analysis (DSA) results (not simulated)
DL2-43	FW Check Valve Closure on Reverse FW Flow	Reverse FW Flow	No analytical limit.
DL3-01	Hydraulic Scram on High RPV Pressure	High RPV Pressure (HP1)	1129.4 psig (7.787 MPaG) Total time delay = 0.7 s
DL3-02	Hydraulic Scram on Low RPV Pressure	Low RPV Pressure (LP1)	800.0 psig (5.516 MPaG) Total time delay = 0.7 s
DL3-03	Hydraulic Scram on Low RPV Level	Low RPV Level (L3)	L3: 773.425 in (19.645 m) from inside bottom of RPV Total time delay = 1.0 s
DL3-04	Hydraulic Scram on High Neutron Flux	High Neutron Flux	125% of rated reactor power Total time delay = 0.09 s
DL3-05	Hydraulic Scram on High Simulated Thermal Power	High Simulated Thermal Power	115% of rated thermal power Signal Time Constant = 7 s Total time delay = 0.09 s
DL3-11	ICS Train 1 Initiation on High RPV Pressure	High RPV Pressure (HP2)	1202.2 psig (8.289 MPaG) Total time delay = 0.7 s
DL3-12	ICS Train 2 Initiation on High RPV Pressure	High RPV Pressure (HP3)	1233.4 psig (8.504 MPaG) Total time delay = 0.7 s
DL3-13	ICS Train 3 Initiation on High RPV Pressure	High RPV Pressure (HP4)	1264.6 psig (8.719 MPaG) Total time delay = 0.7 s
DL3-14	ICS Initiation on Low RPV Water Level	Low RPV Water Level (L2)	L2: 560.0 in (14.224 m) from inside bottom of RPV Total time delay = 1.0 s
DL3-17	MSRIV/MSCIV Isolation on Low RPV Pressure	Low RPV Pressure (LP1)	800.0 psig (5.516 MPaG) Total time delay = 0.7 s
DL3-23	FW Isolation on High RPV Water Level	High RPV Water level (L9)	L9: 880.984 in (22.377 m) from inside bottom of RPV Total time delay = 1.0 s
DL3-39	FW Isolation on Loss of Normal FW Flow	Loss of Normal FW Flow	Total time delay = 1.0 s

Defence Line (DL) ID	Function Name	Inputs	Setpoint / Delay Analytical Limits
DL4a-12	MSRIV/MSCIV Isolation on Sustained Low FW Flow	Sustained Low FW flow	Analysis allows 70 seconds to confirm sustained low FW flow
DL4a-40	Control Rod Drive (CRD) Fast Motor Run-In on Control Rods Not Full-In Signal	Control Rod Full-In Position Switch	Total time delay = 5 s
DL4a-41	FW Pump/Condensate Pump Trip on Control Rods Not Full-In Signal	Control Rod Full-In Position Switch	Total time delay = 5 s

Table 15.5-7: Sequence of Events for Loss of Feedwater Heating (AOO)

Time (s)	Event
0	Initiate a 90°F (50°C) temperature reduction in the FW system
5.5	SCRRI inserts control rods on indication of FW temperature reduction
~60	Control rods stop moving
~400	New steady state achieved

Table 15.5-8: Sequence of Events for Turbine Trip (AOO)

Time (s)	Events
0.0	Reactor scrams on initiation on turbine trip signal
0.2	Control rods begin to move
0.25	TSVs begin to close
0.25	TBVs begin to open
0.40	TSVs are closed
>6.0	New steady state

Table 15.5-9: Sequence of Events for Closure of One Main Steam Reactor Isolation Valve (AOO)

Time (s)	Event
0.0	Initiate closure of one MSRIV
0.8	Anticipatory scram on MSRIV position
3.0	MSRIV in first steam line is closed
3.0	Closure of MSRIV in second steam line initiated on leak detection indication
6.0	MSRIV in second steam line is closed
>25.0	High RPV pressure reached, ICS train initiated (not simulated)
>25.0	New Steady state

Table 15.5-10: Sequence of Events for Loss of Condenser Vacuum (AOO)

Time (s)	Event
0.0	Anticipatory scram initiated on turbine trip signal caused by loss of condenser vacuum
0.2	Control rods begin to move
0.25	TSVs begin to close
0.25	TBVs begin to open
0.40	TSVs are closed
24.0	TBVs close on high main condenser pressure (not simulated)
>24.0	ICS initiation on high RPV pressure (not simulated)
>24.0	New steady state

Table 15.5-11: Sequence of Events for Loss-of-Preferred Power (AOO)

Time (s)	Event
0.0	FW pumps lose power
0.0	Anticipatory scram initiated on generator load rejection caused by generator output breakers opening on loss of power
0.2	Control rods begin to move
0.25	TCVs begin to close due to generator load rejection signal
0.25	TBVs begin to open
6.25	TBVs close following loss-of-preferred power
>20.0	ICS initiation on high RPV pressure (not simulated)
>20.0	New steady state

Table 15.5-12: Sequence of Events for Feedwater Pump Trip (AOO)

Time (s)	Event
0.0	Initiate FW pump trip
10.0	Standby FW pump starts
25.0	Standby FW pump at 100% rated FW flow
~200	New steady state achieved near 100% power and 100% FW flow

Table 15.5-13: Sequence of Events for Inadvertent Isolation Condenser Initiation One Train (AOO)

Time (s)	Event
0	Initiate opening of IC condensate return valve on one train
>150	New steady state achieved near initial conditions

Table 15.5-14: Sequence of Events for Loss of Feedwater Heating (DBA)

Time (s)	Event
0	Initiate FW temperature reduction
~90	High Simulated Thermal Power (STP) reached, scram initiated
~110	Low RPV pressure reached, Main Steam isolation initiated
~300	High RPV level L9 reached, FW isolation initiated
> 400	High RPV pressure reached; ICS train initiated (not simulated)

Table 15.5-15: Sequence of Events for Generator Load Rejection (DBA)

Time (s)	Event
0.0	TCVs begin to close due to a generator load rejection signal
~0.0	TBVs begin to open
0.42	High neutron flux reached, Scram initiated
0.62	Control rods begin to move
152	High RPV pressure reached, ICS train initiated
166	High RPV level L9 reached, FW isolation initiated
>300	A controlled state is achieved

Table 15.5-16: Sequence of Loss-of-Preferred Power (DBA)

Time (s)	Event
0.0	TCV begin to close, and FW pump trips
0.80	High neutron flux reached; Scram initiated
1.0	Control rods begin to move
152	High RPV pressure reached; ICS train initiated
>300	A controlled state is achieved

Table 15.5-17: RPV Pressure Control Downscale (DBA)

Time (s)	Event
0.0	TCVs begin to close, and TBVs remain closed
0.80	High neutron flux reached, Scram initiated
1.0	Control rods begin to move
155	High RPV pressure is reached, ICS train initiated
167	High RPV level L9 reached, FW isolation initiated
>300	A controlled state is achieved

Table 15.5-18: Closure of All MSRIVs and FW Isolation Valves (DBA)

Time (s)	Event
0.0	MSRIVs begin to close, and FW isolation valves close
1.09	High Neutron Flux reached, Scram initiated
1.29	Control rods begin to move
122	High RPV pressure reached, ICS initiated
>300	A controlled state is achieved

Table 15.5-19: Sequence of Events for Feedwater Flow Increase – All Pumps

Time (s)	Event
0	Initiate instant increase in speed of both FW pumps
~20	High RPV level L9 reached, FW isolation initiated
~30	High STP reached, Scram initiated
~90	Low RPV pressure reached, Main Steam Reactor Pressure Isolation Valve isolation initiated
>200	High RPV pressure reached, ICS train initiated (not simulated)

Table 15.5-20: Sequence of Events for Condenser Initiation – All Trains

Time (s)	Events
0	Initiate opening of all IC condensate return valves
~35	High RPV level L9 reached, FW isolation initiated
~55	Low RPV pressure reached, scram, and MSRIV isolation initiated

Table 15.5-21: Sequence of Events for Loss of Feedwater (DBA)

Time (s)	Event
0.0	Initiate loss of FW flow
~30	Low RPV level L3 reached, scram initiated
~85	Low RPV pressure reached, MSRIV isolation initiated
~130	Low RPV level L2 reached, all IC Trains initiate
>300	New steady state achieved

Table 15.5-22: Sequence of Events for RPV Pressure Control Open (DBA)

Time (s)	Event
0.0	TBVs and TCVs open
~70	Low RPV pressure reached, reactor scram and MSRIV isolation initiated
~225	High RPV level L9 reached, FW isolation initiated
>300	High RPV pressure reached, ICS train initiated (not simulated)

Table 15.5-23: Timing of Events for Main Steam Pipe Break Inside Containment, CN-DSA

Time (seconds)	Event	Notes
0.0	Double-ended guillotine rupture of main steam pipe break inside the containment concurrent with Loss-of-Preferred Power (LOPP)	
0.0	FW pump trip and coast down	This is a consequence of LOPP
0.0	TSV or TCV starts closing rapidly	Conservative assumption
1.0	Control rods start to insert on scram initiation	High containment pressure setpoint for scram, reactor isolation and isolation condenser initiation is reached in less than 1 second.
1.0	ICS condensate return valve starts opening	
1.0	Reactor Water Cleanup System (CUW) stops	
3.0	Control rods are inserted sufficiently to diminish fission from prompt neutrons	This is a conservatively long duration for fission from prompt neutrons to diminish. Fission power starts decreasing when the control rods are partially inserted. In addition, voiding in the core due to rapid depressurisation also causes reactor power to decrease rapidly.
5.0	RIVs start to close	The delay time is significantly conservative given that containment pressure reaches the isolation setpoint in less than 1 second.
10.0	RIVs are fully closed	
11.0	ICS condensate return valve is fully open	
12.2	Peak containment pressure is reached	
>12.2	Containment pressure starts decreasing	

Table 15.5-24: Timing of Events for Feedwater Pipe Break Inside Containment, CN-DSA

Time (seconds)	Event	Notes
0.0	Double-ended guillotine rupture of FW pipe break inside containment concurrent with LOPP	
0.0	FW pump trip and coast down	This is a consequence of LOPP
0.0	TSV or TCV starts closing rapidly	Conservative assumption
1.0	Control rods start being inserted on scram initiation	High containment pressure setpoint for scram, reactor isolation and isolation condenser initiation is reached in less than 1 second
1.0	ICS-A and B condensate return valves start opening	
3.0	Control rods are inserted sufficiently to diminish fission from prompt neutrons	This is a conservatively long duration for fission from prompt neutrons
5.0	Feedwater Reactor Isolation Valves (FWRIVs) and CIVs start to close	
10.0	FWRIVs and CIVs are fully closed	
10.1	Peak containment pressure is reached	
11.0	Isolation condenser valves are fully open	

Table 15.5-25: Timing of Events for Small Steam Pipe Break Inside Containment, CN-DSA

Time (seconds)	Event	Notes
0.0	Small steam pipe break concurrent with LOPP	
0.0	Pressure controller freezes. TCV remains in initial position, turbine chest pressure decreases rapidly	Assuming continued steam discharge to the turbine is conservative. TCV closure as a consequence of LOPP is not credited.
0.0	FW pump trip and coast down	
10.6	Steam line low pressure setpoint is reached	
12.3	Reactor scram, 0.7 s delay after low pressure setpoint is reached and another 1 s for scram delay	
15.6	MSRIVs start closing 5 seconds after steam line low pressure	
20.6	MSRIVs are fully closed	
63.6	Level decreases to Level 2	
64.6	ICS-A and ICS-B condensate return valves start opening	
74.6	ICS-A and ICS-B condensate return valves are fully open	
234000	Peak containment pressure reached	

Table 15.5-26: Timing of Events for Small Liquid Pipe Break Inside Containment, CN-DSA

Time (seconds)	Event	Notes
0.0	Small liquid pipe break concurrent with LOPP	
0.0	Pressure controller freezes. TCV remains at initial position, turbine chest pressure decreases rapidly	Assuming continued steam discharge to the turbine is conservative. TCV closure as a consequence of LOPP is not credited.
0.0	FW pump trip and coast down	
10.6	Steam line low pressure setpoint is reached	
12.3	Reactor scram, 0.7 s delay after low pressure setpoint is reached and another 1 s for scram delay	
15.6	MSRIVs start closing 5 seconds after steam pipe low pressure	
20.6	MSRIVs are fully closed	
58.9	Level decreases to Level 2	
59.9	ICS-A and ICS-B condensate return valves start opening	
69.9	ICS-A and ICS-B condensate return valves are fully open	
232800	Peak containment pressure occurs	

Table 15.5-27: Sequence of Events for Closure of One Main Steam Reactor Isolation Valve (DEC)

Time (s)	Event
0.0	Initiate closure of one MSRIV
0.8	Anticipatory scram signal on MSRIV position and scram fails
3.0	MSRIV in first steam line is closed
3.0	Closure of MSRIV in second steam line initiated on leak detection indication
5.8	Feedwater pump trips on high flux after scram signal
5.8	All ICS trains initiate on high flux after scram signal
5.8	CRDM run-in initiation on high flux after scram signal
6.0	MSRIV in second steam line is closed
>300	A controlled state is achieved

Table 15.5-28: Sequence of Events for Complex Sequence Generator Load Rejection (DEC)

Time (s)	Event
0.0	Scram initiation on load rejection signal Half of the control rods fail to insert
0.2	Control rods begin to move.
0.25	TCVs begin to close
0.25	TBVs begin to open
>200	A controlled state is achieved

Table 15.5-29: Sequence of Events for Loss of Condenser Vacuum (DEC)

Time (s)	Event
0.0	Scram fails on initiation on turbine trip signal caused by loss of condenser vacuum
0.25	TSVs begin to close
0.25	TBVs begin to open
0.40	TSVs are closed
5.0	CRDM run-in initiation on high flux after scram signal
5.0	ICS initiation on high flux after scram signal
5.0	Feedwater pump trips on high flux after scram signal
24.0	TBVs close on high main condenser pressure
>300	A controlled state is achieved

Table 15.5-30: Sequence of Events for Loss of Preferred Power (DEC)

Time (s)	Event
0.0	FW pumps lose power
0.0	Scram fails on generator load rejection caused by generator output breakers opening on loss of power
0.25	TCVs begin to close due to generator load rejection signal
0.25	TBVs begin to open
5.0	CRDM run-in initiation on high flux after scram signal
5.0	ICS initiation on high flux after scram signal
6.25	TBVs close following loss-of-preferred power
>300	A controlled state is achieved

Table 15.5-31: Sequence of Events for All Control Rod Withdrawal at Power (ACRW)

Time (s)	Event
0.1	Initiate withdrawal of all rods
19	High STP reached, scram initiated
63	High RPV Level L9 reached, FW isolation initiated
84	Low RPV pressure reached, Main Steam isolation initiated
> 200	High RPV pressure reached; ICS train initiates on high pressure (not simulated)

Table 15.5-32: Sequence of Events for Inadvertent Control Rod Withdrawal at Power Single Rod (ICRW)

Time (s)	Event
0.1	Initiate withdrawal of one rod
>70	Power reaches a stable level
long term	Operators act to control reactor power

Table 15.5-33: Sequence of Events for Feedwater Isolation (DEC)

Time (s)	Event
0.0	Initiate FW Isolation
5	FW flow at zero
~30	Low RPV level L3 reached, no scram (DL3 failure)
65	Reactor scram initiated on sustained low FW flow
70	MSRIV isolation initiated on sustained low FW flow
250	ICS Trains initiate on high RPV pressure
>400	Controlled state achieved

Table 15.5-34: Fuel Handling Accident Sequence of Events

Sequence of Events	Elapsed Time
The BWRX-300 reactor is shut down for refueling operations that begins 24 hours after shutdown. Over this period the core isotopic inventory decays.	24 hours
During a refueling operation a fuel assembly is moved over the top of the core or fuel pool, and the fuel bundle, grapple, mast, and head fall on top of the core or spent fuel racks.	0
Rods in the dropped bundle and impacted bundles fail, releasing the fission gases and cesium iodide in the plenum and gap of the damaged rods to the pool water.	0
Fission gases rise through the pool water to refueling operation floor common airspace surrounding the top of the reactor cavity and fuel pool.	0
The fission gases initially released to the refueling floor airspace are released to the environment.	2 hours
The cesium iodide initially released to the pool evolves to form elemental iodine. Given the chemical conditions in the pool, a portion of the elemental iodine becomes volatile and releases to the refueling floor airspace and subsequently to the environment. This Phase begins 2 hours after the event	718 hours

Table 15.5-35: BWRX-300 Core Parameters

Parameter/Description	Value
Thermal Power	870(MWth)
Core Size (Number of Bundles)	240
Fuel Type	GNF2
Bundle Average Enrichment	3.84-4.68 $^{w}/_{o}$ U235
Total Core Uranium Mass	44.79 Mt (49.37 St)
Core Average Exposure	38,000 MWd/Mt (41,888 MWd/St)

Table 15.5-36: BWRX-300 Core Inventory 24 Hours After Shutdown

Nuclide	Activity (Ci/MWth)	Activity (MBq/MWth)	Nuclide	Activity (Ci/MWth)	Activity (MBq/MWth)
Co-58	3.08E+02	1.14E+07	Te-131m	2.31E+03	8.55E+07
Co-60	6.51E+02	2.41E+07	Te-132	3.13E+04	1.16E+09
Kr-85	5.40E+02	2.00E+07	I-131	2.54E+04	9.40E+08
Kr-85m	1.78E+02	6.59E+06	I-132	3.23E+04	1.20E+09
Kr-87	2.91E-02	1.08E+03	I-133	2.54E+04	9.40E+08
Kr-88	5.56E+01	2.06E+06	I-134	1.36E-03	5.03E+01
Rb-86	7.40E+01	2.74E+06	I-135	4.16E+03	1.54E+08
Sr-89	2.56E+04	9.47E+08	Xe-133	5.16E+04	1.91E+09
Sr-90	4.48E+03	1.66E+08	Xe-135	1.49E+04	5.51E+08
Sr-91	5.70E+03	2.11E+08	Cs-134	9.46E+03	3.50E+08
Sr-92	7.64E+01	2.83E+06	Cs-136	2.60E+03	9.62E+07
Y-90	4.62E+03	1.71E+08	Cs-137	5.97E+03	2.21E+08
Y-91	3.32E+04	1.23E+09	Ba-139	3.20E-01	1.18E+04
Y-92	1.13E+03	4.18E+07	Ba-140	4.51E+04	1.67E+09
Y-93	8.00E+03	2.96E+08	La-140	4.88E+04	1.81E+09
Zr-95	4.73E+04	1.75E+09	La-141	7.08E+02	2.62E+07
Zr-97	1.84E+04	6.81E+08	La-142	1.03E+00	3.81E+04
Nb-95	4.80E+04	1.78E+09	Ce-141	4.44E+04	1.64E+09
Mo-99	4.00E+04	1.48E+09	Ce-143	2.55E+04	9.44E+08
Tc-99m	3.81E+04	1.41E+09	Ce-144	3.82E+04	1.41E+09
Ru-103	4.33E+04	1.60E+09	Pr-143	4.02E+04	1.49E+09
Ru-105	7.65E+02	2.83E+07	Nd-147	1.70E+04	6.29E+08
Ru-106	2.05E+04	7.59E+08	Np-239	4.53E+05	1.68E+10
Rh-105	2.15E+04	7.96E+08	Pu-238	2.28E+02	8.44E+06
Sb-127	2.63E+03	9.73E+07	Pu-239	1.87E+01	6.92E+05
Sb-129	1.95E+02	7.22E+06	Pu-240	2.70E+01	9.99E+05
Te-127	2.87E+03	1.06E+08	Pu-241	7.83E+03	2.90E+08
Te-127m	4.19E+02	1.55E+07	Am-241	1.47E+01	5.44E+05
Te-129	1.08E+03	4.00E+07	Cm-242	3.03E+03	1.12E+08
Te-129m	1.31E+03	4.85E+07	Cm-244	2.15E+02	7.96E+06

Table 15.5-37: Non-LOCA Fraction of Fission Product Inventory in Gap

Group	Gap Fraction
I-131	0.08
Kr-85	0.10
Other Noble Gases	0.05
Other Halogens	0.05
Alkali Metals	0.12

Table 15.5-38: BWRX-300 Fuel Handling Accident (FHA) Activity Released from Fuel

Nuclide	Activity Released from Fuel (Ci)	Activity Released from Fuel (MBq)
Kr-85m	8.43E+01	3.12E+06
Kr-85	5.12E+02	1.89E+07
Kr-87	1.38E-02	5.10E+02
Kr-88	2.63E+01	9.74E+05
I-131	1.92E+04	7.12E+08
I-132	1.53E+04	5.66E+08
I-133	1.20E+04	4.45E+08
I-134	6.44E-04	2.38E+01
I-135	1.97E+03	7.29E+07
Xe-133	2.44E+04	9.04E+08
Xe-135	7.06E+03	2.61E+08

Table 15.5-39a: BWRX-300 FHA Phase 1 Activity Released from the Reactor Cavity Pool

Nuclide	Activity Released from Fuel (Ci)	Activity Released from Fuel (MBq)	Phase 1 Decontamination Factor (DF)	Activity Released from the Pool (Ci)	Activity Released from the Pool (MBq)
Kr-85m	8.43E+01	3.12E+06	1	8.43E+01	3.12E+06
Kr-85	5.12E+02	1.89E+07	1	5.12E+02	1.89E+07
Kr-87	1.38E-02	5.10E+02	1	1.38E-02	5.10E+02
Kr-88	2.63E+01	9.74E+05	1	2.63E+01	9.74E+05
I-131	9.62E+02	3.56E+07	490	1.96E+00	7.27E+04
I-132	7.65E+02	2.83E+07	490	1.56E+00	5.78E+04
I-133	6.02E+02	2.23E+07	490	1.23E+00	4.54E+04
I-134	3.22E-05	1.19E+00	490	6.57E-08	2.43E-03
I-135	9.85E+01	3.65E+06	490	2.01E-01	7.44E+03
Xe-133	2.44E+04	9.04E+08	1	2.44E+04	9.04E+08
Xe-135	7.06E+03	2.61E+08	1	7.06E+03	2.61E+08

Table 15.5-39b: BWRX-300 FHA Phase 2 Activity Available for Release from the Reactor Cavity Pool

Nuclide	Activity Available for Release from the Pool (Ci)	Activity Available for Release from the Pool (MBq)
I-131	1.92E+04	7.12E+08
I-132	1.53E+04	5.66E+08
I-133	1.20E+04	4.45E+08
I-134	6.44E-04	2.38E+01
I-135	1.97E+03	7.29E+07

Table 15.5-39c: Key Parameters for DBA FHA Analysis

I. Data and Assumptions Used to Estimate Source Terms	
A. Power level, MWth	870
B. Core Radionuclide Inventory	Table 15.5-35
C. Plenum Activity Radioactivity for I-131, % Radioactivity for Kr-85, % Radioactivity for other noble gases, % Radioactivity for other halogens, % Radioactivity for alkali metals, %	 8 10 5 5 12
D. Radial peaking factor for damaged rods	1.7
E. Duration of release, hr Phase 1, initial gaseous release Phase 2, pool iodine re-evolution	 2 718[1]
F. Total Number of Bundles in Core	240
G. Number of damaged bundles	1.507
H. Minimum time after shutdown to accident, hr	24
I. Average fuel exposure, MWd/MT (MWd/ST)	38,000 (41,888)
II. Data and Assumptions Used to Estimate Activity Released	
A. Species fraction, in percentage form released from fuel Organic iodine, % Elemental iodine, % Particulate iodine, % Noble gas, %	 0.15 4.85 95 100
Phase 1 Reactor Building/Fuel Building Atmosphere	
Organic iodine, % Elemental iodine, % Particulate iodine, % Noble gas, %	3 97 0 100
Phase 2 Reactor Building/Fuel Building Atmosphere	
Organic iodine, % Elemental iodine, % Particulate iodine, % Noble gas, %	0 100 0 0
B. Pool Water Level above the core, m (ft) Pool Volume (L)	≥7.01 (23.0) 770,000

C. Pool Retention decontamination factor	
Phase 1 Total Iodine	490
Fuel Rod Pressure, psig	1200
Bubble Rise Time, sec	4.491
Bubble diameter, cm	0.761
Phase 1 Noble gas	1
Phase 2 Iodine	1
Phase 1 and 2 Particulates	Infinite
D. Release rate, %/hr	
Phase 1, 0 to 2 hours	345
Phase 2, 2 to 720 hours	8.33E-04
Pool, pH	4
Total Iodine in pool, mol	5.12E-02
Stable Iodine Concentration, mol/L	1.50E-08
Total Iodine Concentration, mol/L	6.65E-08
I_2 Volatile Fraction	6.32E-03
S_{pool}/V_{pool}, 1/m	0.1
III. Dispersion and Dose Data	
A. Atmospheric Dispersion Factors, s/m^3	
MCRE	
0-2 hours	1.51E-03
2-8 hours	1.06E-03
8-24 hours	4.95E-04
1-4 days	4.68E-04
4-30 days	4.21E-04
IV. MCRE Parameters	
A. Volume m^3 (ft^3)	1,090 (38,500)
B. Flow Rate to Environment m^3/s (ft^3/min)	18.2 (38,500)
V. Breathing Rates m^3/s	
MCRE	
0-30 days	3.5E-04
VI. Occupancy Factors	
MCRE	
0-24 hours	1.0
1-4 days	0.6
4-30 days	0.4

Notes:

1. Phase 2 starts at a 2-hour time delay after the start of the Phase 1 release.

Table 15.5-40: Main Steam Line Break Accident Airborne Release Source Term[1]

Isotope	Equilibrium Iodine		Pre-Incident Iodine Spike	
	MBq	Ci	MBq	Ci
Kr-83m	1.3E+04	3.4E-01	1.3E+04	3.4E-01
Kr-85m	3.6E+02	9.6E-03	3.6E+02	9.6E-03
Kr-85	8.7E+01	2.3E-03	8.7E+01	2.3E-03
Kr-87	2.4E+03	6.6E-02	2.4E+03	6.6E-02
Kr-88	1.3E+03	3.6E-02	1.3E+03	3.6E-02
Kr-89	4.4E+05	1.2E+01	4.4E+05	1.2E+01
Xe-131m	7.1E+01	1.9E-03	7.1E+01	1.9E-03
Xe-133m	4.2E+01	1.1E-03	4.2E+01	1.1E-03
Xe-133	6.4E+02	1.7E-02	6.4E+02	1.7E-02
Xe-135m	1.2E+04	3.1E-01	1.2E+04	3.1E-01
Xe-135	6.4E+03	1.7E-01	6.4E+03	1.7E-01
Xe-137	2.1E+04	5.8E-01	2.1E+04	5.8E-01
Xe-138	6.0E+04	1.6E+00	6.0E+04	1.6E+00
I-131	1.3E+04	3.6E-01	2.6E+05	7.2E+00
I-132	1.3E+05	3.5E+00	2.6E+06	7.1E+01
I-133	9.8E+04	2.6E+00	2.0E+06	5.3E+01
I-134	3.9E+05	1.0E+01	7.7E+06	2.1E+02
I-135	1.9E+05	5.1E+00	3.9E+06	1.0E+02
Rb-89	7.6E+04	2.0E+00	7.6E+04	2.0E+00
Cs-134	1.1E+03	3.0E-02	1.1E+03	3.0E-02
Cs-136	8.6E+02	2.3E-02	8.6E+02	2.3E-02
Cs-137	1.7E+03	4.5E-02	1.7E+03	4.5E-02
Cs-138	8.2E+04	2.2E+00	8.2E+04	2.2E+00
Ba-137m	1.7E+03	4.5E-02	1.7E+03	4.5E-02
H-3	1.2E+04	3.2E-01	1.2E+04	3.2E-01
Na-24	1.6E+02	4.4E-03	1.6E+02	4.4E-03
P-32	6.6E+00	1.8E-04	6.6E+00	1.8E-04
Cr-51	1.6E+02	4.3E-03	1.6E+02	4.3E-03
Mn-54	8.0E+01	2.2E-03	8.0E+01	2.2E-03
Mn-56	3.0E+02	8.1E-03	3.0E+02	8.1E-03
Fe-55	1.7E+02	4.5E-03	1.7E+02	4.5E-03
Fe-59	4.4E+01	1.2E-03	4.4E+01	1.2E-03
Co-58	3.8E+01	1.0E-03	3.8E+01	1.0E-03

Isotope	Equilibrium Iodine		Pre-Incident Iodine Spike	
	MBq	Ci	MBq	Ci
Co-60	7.4E+01	2.0E-03	7.4E+01	2.0E-03
Ni-63	1.7E-01	4.5E-06	1.7E-01	4.5E-06
Cu-64	7.8E+02	2.1E-02	7.8E+02	2.1E-02
Zn-65	3.4E+01	9.2E-04	3.4E+01	9.2E-04
Sr-89	1.2E+02	3.3E-03	1.2E+02	3.3E-03
Sr-90	6.2E+00	1.7E-04	6.2E+00	1.7E-04
Y-90	6.2E+00	1.7E-04	6.2E+00	1.7E-04
Sr-91	7.4E+04	2.0E+00	7.4E+04	2.0E+00
Sr-92	1.6E+05	4.3E+00	1.6E+05	4.3E+00
Y-91	1.6E+03	4.4E-02	1.6E+03	4.4E-02
Y-92	5.0E+04	1.3E+00	5.0E+04	1.3E+00
Y-93	5.4E+03	1.5E-01	5.4E+03	1.5E-01
Zr-95	3.4E+03	9.2E-02	3.4E+03	9.2E-02
Nb-95	3.4E+03	9.2E-02	3.4E+03	9.2E-02
Mo-99	1.6E+04	4.3E-01	1.6E+04	4.3E-01
Tc-99m	1.6E+04	4.3E-01	1.6E+04	4.3E-01
Ru-103	8.2E+02	2.2E-02	8.2E+02	2.2E-02
Rh-103m	8.2E+02	2.2E-02	8.2E+02	2.2E-02
Ru-106	1.2E+02	3.3E-03	1.2E+02	3.3E-03
Rh-106	1.2E+02	3.3E-03	1.2E+02	3.3E-03
Ag-110m	1.7E-01	4.5E-06	1.7E-01	4.5E-06
Te-129m	1.6E+03	4.4E-02	1.6E+03	4.4E-02
Te-131m	2.2E+03	5.9E-02	2.2E+03	5.9E-02
Te-132	3.8E+02	1.0E-02	3.8E+02	1.0E-02
Ba-140	1.7E+04	4.6E-01	1.7E+04	4.6E-01
La-140	1.7E+04	4.6E-01	1.7E+04	4.6E-01
Ce-141	8.2E+02	2.2E-02	8.2E+02	2.2E-02
Ce-144	1.2E+02	3.3E-03	1.2E+02	3.3E-03
Pr-144	1.2E+02	3.3E-03	1.2E+02	3.3E-03
W-187	3.8E+01	1.0E-03	3.8E+01	1.0E-03
Np-239	1.3E+04	3.5E-01	1.3E+04	3.5E-01

Note:

1. The released activity shown in the table is decayed in the RADTRAD model over the one-minute release duration from the building assumed.

Table 15.5-41: Feedwater Line Break (FWLB) Accident Airborne Release Source Term

Isotope	Equilibrium Iodine		Pre-Incident Iodine Spike	
	Ci	MBq	Ci	MBq
Kr-83m	8.03E-02	2.97E+03	8.03E-02	2.97E+03
Kr-85m	2.25E-03	8.34E+01	2.25E-03	8.34E+01
Kr-85	5.50E-04	2.03E+01	5.50E-04	2.03E+01
Kr-87	1.55E-02	5.73E+02	1.55E-02	5.73E+02
Kr-88	8.45E-03	3.13E+02	8.45E-03	3.13E+02
Kr-89	2.88E+00	1.06E+05	2.88E+00	1.06E+05
Xe-131m	4.51E-04	1.67E+01	4.51E-04	1.67E+01
Xe-133m	2.68E-04	9.91E+00	2.68E-04	9.91E+00
Xe-133	4.09E-03	1.51E+02	4.09E-03	1.51E+02
Xe-135m	7.33E-02	2.71E+03	7.33E-02	2.71E+03
Xe-135	4.09E-02	1.51E+03	4.09E-02	1.51E+03
Xe-137	1.36E-01	5.02E+03	1.36E-01	5.02E+03
Xe-138	3.80E-01	1.41E+04	3.80E-01	1.41E+04
I-131	4.70E-01	1.74E+04	9.39E+00	3.47E+05
I-132	4.62E+00	1.71E+05	9.39E+01	3.47E+06
I-133	3.47E+00	1.28E+05	6.93E+01	2.57E+06
I-134	1.37E+01	5.08E+05	2.74E+02	1.02E+07
I-135	6.72E+00	2.49E+05	1.37E+02	5.08E+06
Rb-89	2.73E+00	1.01E+05	2.73E+00	1.01E+05
Cs-134	3.96E-02	1.46E+03	3.96E-02	1.46E+03
Cs-136	3.09E-02	1.14E+03	3.09E-02	1.14E+03
Cs-137	6.05E-02	2.24E+03	6.05E-02	2.24E+03
Cs-138	2.95E+00	1.09E+05	2.95E+00	1.09E+05
Ba-137m	6.05E-02	2.24E+03	6.05E-02	2.24E+03
H-3	2.41E-01	8.91E+03	2.41E-01	8.91E+03
Na-24	5.90E-03	2.18E+02	5.90E-03	2.18E+02
P-32	2.37E-04	8.79E+00	2.37E-04	8.79E+00
Cr-51	5.76E-03	2.13E+02	5.76E-03	2.13E+02
Mn-54	2.88E-03	1.07E+02	2.88E-03	1.07E+02
Mn-56	1.08E-02	3.99E+02	1.08E-02	3.99E+02
Fe-55	5.97E-03	2.21E+02	5.97E-03	2.21E+02
Fe-59	1.58E-03	5.86E+01	1.58E-03	5.86E+01
Co-58	1.37E-03	5.06E+01	1.37E-03	5.06E+01

Isotope	Equilibrium Iodine		Pre-Incident Iodine Spike	
	Ci	MBq	Ci	MBq
Co-60	2.66E-03	9.85E+01	2.66E-03	9.85E+01
Ni-63	5.97E-06	2.21E-01	5.97E-06	2.21E-01
Cu-64	2.81E-02	1.04E+03	2.81E-02	1.04E+03
Zn-65	1.22E-03	4.53E+01	1.22E-03	4.53E+01
Sr-89	4.39E-03	1.62E+02	4.39E-03	1.62E+02
Sr-90	2.23E-04	8.25E+00	2.23E-04	8.25E+00
Y-90	2.23E-04	8.25E+00	2.23E-04	8.25E+00
Sr-91	2.66E+00	9.85E+04	2.66E+00	9.85E+04
Sr-92	5.76E+00	2.13E+05	5.76E+00	2.13E+05
Y-91	5.83E-02	2.16E+03	5.83E-02	2.16E+03
Y-92	1.80E+00	6.66E+04	1.80E+00	6.66E+04
Y-93	1.94E-01	7.19E+03	1.94E-01	7.19E+03
Zr-95	1.22E-01	4.53E+03	1.22E-01	4.53E+03
Nb-95	1.22E-01	4.53E+03	1.22E-01	4.53E+03
Mo-99	5.69E-01	2.10E+04	5.69E-01	2.10E+04
Tc-99m	5.69E-01	2.10E+04	5.69E-01	2.10E+04
Ru-103	2.95E-02	1.09E+03	2.95E-02	1.09E+03
Rh-103m	2.95E-02	1.09E+03	2.95E-02	1.09E+03
Ru-106	4.39E-03	1.62E+02	4.39E-03	1.62E+02
Rh-106	4.39E-03	1.62E+02	4.39E-03	1.62E+02
Ag-110m	5.97E-06	2.21E-01	5.97E-06	2.21E-01
Te-129m	5.83E-02	2.16E+03	5.83E-02	2.16E+03
Te-131m	7.92E-02	2.93E+03	7.92E-02	2.93E+03
Te-132	1.37E-02	5.06E+02	1.37E-02	5.06E+02
Ba-140	6.12E-01	2.26E+04	6.12E-01	2.26E+04
La-140	6.12E-01	2.26E+04	6.12E-01	2.26E+04
Ce-141	2.95E-02	1.09E+03	2.95E-02	1.09E+03
Ce-144	4.39E-03	1.62E+02	4.39E-03	1.62E+02
Pr-144	4.39E-03	1.62E+02	4.39E-03	1.62E+02
W-187	1.37E-03	5.06E+01	1.37E-03	5.06E+01
Np-239	4.68E-01	1.73E+04	4.68E-01	1.73E+04

Table 15.5-42: ICS Line Break Accident Airborne Release Source Term

Isotope	Equilibrium Iodine		Pre-Incident Iodine Spike	
	Ci	MBq	Ci	MBq
Kr-83m	1.73E-01	6.39E+03	1.73E-01	6.39E+03
Kr-85m	4.85E-03	1.79E+02	4.85E-03	1.79E+02
Kr-85	1.18E-03	4.38E+01	1.18E-03	4.38E+01
Kr-87	3.34E-02	1.23E+03	3.34E-02	1.23E+03
Kr-88	1.82E-02	6.73E+02	1.82E-02	6.73E+02
Kr-89	6.19E+00	2.29E+05	6.19E+00	2.29E+05
Xe-131m	9.70E-04	3.59E+01	9.70E-04	3.59E+01
Xe-133m	5.76E-04	2.13E+01	5.76E-04	2.13E+01
Xe-133	8.79E-03	3.25E+02	8.79E-03	3.25E+02
Xe-135m	1.58E-01	5.83E+03	1.58E-01	5.83E+03
Xe-135	8.79E-02	3.25E+03	8.79E-02	3.25E+03
Xe-137	2.92E-01	1.08E+04	2.92E-01	1.08E+04
Xe-138	8.19E-01	3.03E+04	8.19E-01	3.03E+04
I-131	3.94E-03	1.46E+02	8.19E-02	3.03E+03
I-132	3.64E-02	1.35E+03	7.58E-01	2.80E+04
I-133	2.91E-02	1.08E+03	5.76E-01	2.13E+04
I-134	1.15E-01	4.26E+03	2.27E+00	8.41E+04
I-135	5.46E-02	2.02E+03	1.09E+00	4.04E+04
Rb-89	1.15E-03	4.26E+01	1.15E-03	4.26E+01
Cs-134	1.67E-05	6.17E-01	1.67E-05	6.17E-01
Cs-136	1.30E-05	4.82E-01	1.30E-05	4.82E-01
Cs-137	2.55E-05	9.42E-01	2.55E-05	9.42E-01
Cs-138	1.24E-03	4.60E+01	1.24E-03	4.60E+01
Ba-137m	2.55E-05	9.42E-01	2.55E-05	9.42E-01
H-3	8.49E-02	3.14E+03	8.49E-02	3.14E+03
Na-24	2.49E-06	9.20E-02	2.49E-06	9.20E-02
P-32	1.00E-07	3.70E-03	1.00E-07	3.70E-03
Cr-51	2.43E-06	8.97E-02	2.43E-06	8.97E-02
Mn-54	1.21E-06	4.49E-02	1.21E-06	4.49E-02
Mn-56	4.55E-06	1.68E-01	4.55E-06	1.68E-01
Fe-55	2.52E-06	9.31E-02	2.52E-06	9.31E-02
Fe-59	6.67E-07	2.47E-02	6.67E-07	2.47E-02

Isotope	Equilibrium Iodine		Pre-Incident Iodine Spike	
	Ci	MBq	Ci	MBq
Co-58	5.76E-07	2.13E-02	5.76E-07	2.13E-02
Co-60	1.12E-06	4.15E-02	1.12E-06	4.15E-02
Ni-63	2.52E-09	9.31E-05	2.52E-09	9.31E-05
Cu-64	1.18E-05	4.38E-01	1.18E-05	4.38E-01
Zn-65	5.15E-07	1.91E-02	5.15E-07	1.91E-02
Sr-89	1.85E-06	6.84E-02	1.85E-06	6.84E-02
Sr-90	9.40E-08	3.48E-03	9.40E-08	3.48E-03
Y-90	9.40E-08	3.48E-03	9.40E-08	3.48E-03
Sr-91	1.12E-03	4.15E+01	1.12E-03	4.15E+01
Sr-92	2.43E-03	8.97E+01	2.43E-03	8.97E+01
Y-91	2.46E-05	9.09E-01	2.46E-05	9.09E-01
Y-92	7.58E-04	2.80E+01	7.58E-04	2.80E+01
Y-93	8.19E-05	3.03E+00	8.19E-05	3.03E+00
Zr-95	5.15E-05	1.91E+00	5.15E-05	1.91E+00
Nb-95	5.15E-05	1.91E+00	5.15E-05	1.91E+00
Mo-99	2.40E-04	8.86E+00	2.40E-04	8.86E+00
Tc-99m	2.40E-04	8.86E+00	2.40E-04	8.86E+00
Ru-103	1.24E-05	4.60E-01	1.24E-05	4.60E-01
Rh-103m	1.24E-05	4.60E-01	1.24E-05	4.60E-01
Ru-106	1.85E-06	6.84E-02	1.85E-06	6.84E-02
Rh-106	1.85E-06	6.84E-02	1.85E-06	6.84E-02
Ag-110m	2.52E-09	9.31E-05	2.52E-09	9.31E-05
Te-129m	2.46E-05	9.09E-01	2.46E-05	9.09E-01
Te-131m	3.34E-05	1.23E+00	3.34E-05	1.23E+00
Te-132	5.76E-06	2.13E-01	5.76E-06	2.13E-01
Ba-140	2.58E-04	9.54E+00	2.58E-04	9.54E+00
La-140	2.58E-04	9.54E+00	2.58E-04	9.54E+00
Ce-141	1.24E-05	4.60E-01	1.24E-05	4.60E-01
Ce-144	1.85E-06	6.84E-02	1.85E-06	6.84E-02
Pr-144	1.85E-06	6.84E-02	1.85E-06	6.84E-02
W-187	5.76E-07	2.13E-02	5.76E-07	2.13E-02
Np-239	1.97E-04	7.29E+00	1.97E-04	7.29E+00

Table 15.5-43A: Instrument Line Break Accident Airborne Iodine Equilibrium Source Term

Time	2.0 h		8.0 h		24.0 h		72.0 h		78.0 h	
Nuclide	**Ci**	**MBq**	**Ci**	**MBq**	**Ci**	**MBq**	**Ci**	**MBq**	**Ci**	**MBq**
I-131	2.3E-01	8.5E+03	9.2E-01	3.4E+04	2.7E+00	1.0E+05	8.2E+00	3.1E+05	8.5E+00	3.1E+05
I-132	2.3E+00	8.3E+04	9.0E+00	3.3E+05	2.7E+01	1.0E+06	8.1E+01	3.0E+06	8.3E+01	3.1E+06
I-133	1.7E+00	6.3E+04	6.8E+00	2.5E+05	2.0E+01	7.5E+05	6.1E+01	2.3E+06	6.2E+01	2.3E+06
I-134	6.7E+00	2.5E+05	2.7E+01	9.9E+05	8.0E+01	3.0E+06	2.4E+02	8.9E+06	2.5E+02	9.1E+06
I-135	3.3E+00	1.2E+05	1.3E+01	4.8E+05	3.9E+01	1.5E+06	1.2E+02	4.4E+06	1.2E+02	4.5E+06
Rb-89	1.3E+00	4.9E+04	5.4E+00	2.0E+05	1.6E+01	5.9E+05	4.8E+01	1.8E+06	4.9E+01	1.8E+06
Cs-134	1.9E-02	7.2E+02	7.8E-02	2.9E+03	2.3E-01	8.6E+03	7.0E-01	2.6E+04	7.2E-01	2.6E+04
Cs-136	1.5E-02	5.6E+02	6.1E-02	2.2E+03	1.8E-01	6.7E+03	5.5E-01	2.0E+04	5.6E-01	2.1E+04
Cs-137	3.0E-02	1.1E+03	1.2E-01	4.4E+03	3.6E-01	1.3E+04	1.1E+00	3.9E+04	1.1E+00	4.0E+04
Cs-138	1.4E+00	5.3E+04	5.8E+00	2.1E+05	1.7E+01	6.4E+05	5.2E+01	1.9E+06	5.3E+01	2.0E+06
Ba-137m	3.0E-02	1.1E+03	1.2E-01	4.4E+03	3.6E-01	1.3E+04	1.1E+00	3.9E+04	1.1E+00	4.0E+04
H-3	9.9E-02	3.6E+03	3.9E-01	1.5E+04	1.2E+00	4.4E+04	3.6E+00	1.3E+05	3.6E+00	1.3E+05
Na-24	2.9E-03	1.1E+02	1.2E-02	4.3E+02	3.5E-02	1.3E+03	1.0E-01	3.8E+03	1.1E-01	3.9E+03
P-32	1.2E-04	4.3E+00	4.7E-04	1.7E+01	1.4E-03	5.2E+01	4.2E-03	1.5E+02	4.3E-03	1.6E+02
Cr-51	2.8E-03	1.0E+02	1.1E-02	4.2E+02	3.4E-02	1.3E+03	1.0E-01	3.8E+03	1.0E-01	3.9E+03
Mn-54	1.4E-03	5.2E+01	5.6E-03	2.1E+02	1.7E-02	6.3E+02	5.1E-02	1.9E+03	5.2E-02	1.9E+03
Mn-56	5.3E-03	2.0E+02	2.1E-02	7.8E+02	6.3E-02	2.3E+03	1.9E-01	7.0E+03	2.0E-01	7.2E+03
Fe-55	2.9E-03	1.1E+02	1.2E-02	4.3E+02	3.5E-02	1.3E+03	1.1E-01	3.9E+03	1.1E-01	4.0E+03
Fe-59	7.7E-04	2.9E+01	3.1E-03	1.1E+02	9.3E-03	3.4E+02	2.8E-02	1.0E+03	2.9E-02	1.1E+03
Co-58	6.7E-04	2.5E+01	2.7E-03	9.9E+01	8.0E-03	3.0E+02	2.4E-02	8.9E+02	2.5E-02	9.1E+02

Time	2.0 h		8.0 h		24.0 h		72.0 h		78.0 h	
Nuclide	**Ci**	**MBq**	**Ci**	**MBq**	**Ci**	**MBq**	**Ci**	**MBq**	**Ci**	**MBq**
Co-60	1.3E-03	4.8E+01	5.2E-03	1.9E+02	1.6E-02	5.8E+02	4.7E-02	1.7E+03	4.8E-02	1.8E+03
Ni-63	2.9E-06	1.1E-01	1.2E-05	4.3E-01	3.5E-05	1.3E+00	1.1E-04	3.9E+00	1.1E-04	4.0E+00
Cu-64	1.4E-02	5.1E+02	5.5E-02	2.0E+03	1.6E-01	6.1E+03	4.9E-01	1.8E+04	5.1E-01	1.9E+04
Zn-65	6.0E-04	2.2E+01	2.4E-03	8.9E+01	7.2E-03	2.7E+02	2.2E-02	8.0E+02	2.2E-02	8.2E+02
Sr-89	2.1E-03	7.9E+01	8.6E-03	3.2E+02	2.6E-02	9.5E+02	7.7E-02	2.9E+03	7.9E-02	2.9E+03
Sr-90	1.1E-04	4.0E+00	4.4E-04	1.6E+01	1.3E-03	4.9E+01	3.9E-03	1.5E+02	4.0E-03	1.5E+02
Y-90	1.1E-04	4.0E+00	4.4E-04	1.6E+01	1.3E-03	4.9E+01	3.9E-03	1.5E+02	4.0E-03	1.5E+02
Sr-91	1.3E+00	4.8E+04	5.2E+00	1.9E+05	1.6E+01	5.8E+05	4.7E+01	1.7E+06	4.8E+01	1.8E+06
Sr-92	2.8E+00	1.0E+05	1.1E+01	4.2E+05	3.4E+01	1.3E+06	1.0E+02	3.8E+06	1.0E+02	3.9E+06
Y-91	2.9E-02	1.1E+03	1.1E-01	4.2E+03	3.4E-01	1.3E+04	1.0E+00	3.8E+04	1.1E+00	3.9E+04
Y-92	8.8E-01	3.3E+04	3.5E+00	1.3E+05	1.1E+01	3.9E+05	3.2E+01	1.2E+06	3.3E+01	1.2E+06
Y-93	9.5E-02	3.5E+03	3.8E-01	1.4E+04	1.1E+00	4.2E+04	3.4E+00	1.3E+05	3.5E+00	1.3E+05
Zr-95	6.0E-02	2.2E+03	2.4E-01	8.9E+03	7.2E-01	2.7E+04	2.2E+00	8.0E+04	2.2E+00	8.2E+04
Nb-95	6.0E-02	2.2E+03	2.4E-01	8.9E+03	7.2E-01	2.7E+04	2.2E+00	8.0E+04	2.2E+00	8.2E+04
Mo-99	2.8E-01	1.0E+04	1.1E+00	4.1E+04	3.3E+00	1.2E+05	1.0E+01	3.7E+05	1.0E+01	3.8E+05
Tc-99m	2.8E-01	1.0E+04	1.1E+00	4.1E+04	3.3E+00	1.2E+05	1.0E+01	3.7E+05	1.0E+01	3.8E+05
Ru-103	1.4E-02	5.3E+02	5.8E-02	2.1E+03	1.7E-01	6.4E+03	5.2E-01	1.9E+04	5.3E-01	2.0E+04
Rh-103m	1.4E-02	5.3E+02	5.8E-02	2.1E+03	1.7E-01	6.4E+03	5.2E-01	1.9E+04	5.3E-01	2.0E+04
Ru-106	2.1E-03	7.9E+01	8.6E-03	3.2E+02	2.6E-02	9.5E+02	7.7E-02	2.9E+03	7.9E-02	2.9E+03
Rh-106	2.1E-03	7.9E+01	8.6E-03	3.2E+02	2.6E-02	9.5E+02	7.7E-02	2.9E+03	7.9E-02	2.9E+03
Ag-110m	2.9E-06	1.1E-01	1.2E-05	4.3E-01	3.5E-05	1.3E+00	1.1E-04	3.9E+00	1.1E-04	4.0E+00

Time	2.0 h		8.0 h		24.0 h		72.0 h		78.0 h	
Nuclide	**Ci**	**MBq**	**Ci**	**MBq**	**Ci**	**MBq**	**Ci**	**MBq**	**Ci**	**MBq**
Te-129m	2.9E-02	1.1E+03	1.1E-01	4.2E+03	3.4E-01	1.3E+04	1.0E+00	3.8E+04	1.1E+00	3.9E+04
Te-131m	3.9E-02	1.4E+03	1.6E-01	5.7E+03	4.7E-01	1.7E+04	1.4E+00	5.2E+04	1.4E+00	5.3E+04
Te-132	6.7E-03	2.5E+02	2.7E-02	9.9E+02	8.0E-02	3.0E+03	2.4E-01	8.9E+03	2.5E-01	9.1E+03
Ba-140	3.0E-01	1.1E+04	1.2E+00	4.4E+04	3.6E+00	1.3E+05	1.1E+01	4.0E+05	1.1E+01	4.1E+05
La-140	3.0E-01	1.1E+04	1.2E+00	4.4E+04	3.6E+00	1.3E+05	1.1E+01	4.0E+05	1.1E+01	4.1E+05
Ce-141	1.4E-02	5.3E+02	5.8E-02	2.1E+03	1.7E-01	6.4E+03	5.2E-01	1.9E+04	5.3E-01	2.0E+04
Ce-144	2.1E-03	7.9E+01	8.6E-03	3.2E+02	2.6E-02	9.5E+02	7.7E-02	2.9E+03	7.9E-02	2.9E+03
Pr-144	2.1E-03	7.9E+01	8.6E-03	3.2E+02	2.6E-02	9.5E+02	7.7E-02	2.9E+03	7.9E-02	2.9E+03
W-187	6.7E-04	2.5E+01	2.7E-03	9.9E+01	8.0E-03	3.0E+02	2.4E-02	8.9E+02	2.5E-02	9.1E+02
Np-239	2.3E-01	8.5E+03	9.2E-01	3.4E+04	2.7E+00	1.0E+05	8.2E+00	3.1E+05	8.5E+00	3.1E+05

Table 15.5-43B: Instrument Line Break Accident Iodine Spike Release Source Term

Time	2.0 h		8.0 h		24.0 h		72.0 h		78.0 h	
Nuclide	Ci	MBq	Ci	MBq	Ci	MBq	Ci	MBq	Ci	MBq
I-131	4.6E+00	1.7E+05	1.8E+01	6.8E+05	5.5E+01	2.0E+06	1.6E+02	6.1E+06	1.7E+02	6.3E+06
I-132	4.6E+01	1.7E+06	1.8E+02	6.8E+06	5.5E+02	2.0E+07	1.6E+03	6.1E+07	1.7E+03	6.3E+07
I-133	3.4E+01	1.3E+06	1.4E+02	5.0E+06	4.1E+02	1.5E+07	1.2E+03	4.5E+07	1.2E+03	4.6E+07
I-134	1.3E+02	4.9E+06	5.4E+02	2.0E+07	1.6E+03	5.9E+07	4.8E+03	1.8E+08	4.9E+03	1.8E+08
I-135	6.7E+01	2.5E+06	2.7E+02	9.9E+06	8.0E+02	3.0E+07	2.4E+03	8.9E+07	2.5E+03	9.1E+07
Rb-89	1.3E+00	4.9E+04	5.4E+00	2.0E+05	1.6E+01	5.9E+05	4.8E+01	1.8E+06	4.9E+01	1.8E+06
Cs-134	1.9E-02	7.2E+02	7.8E-02	2.9E+03	2.3E-01	8.6E+03	7.0E-01	2.6E+04	7.2E-01	2.6E+04
Cs-136	1.5E-02	5.6E+02	6.1E-02	2.2E+03	1.8E-01	6.7E+03	5.5E-01	2.0E+04	5.6E-01	2.1E+04
Cs-137	3.0E-02	1.1E+03	1.2E-01	4.4E+03	3.6E-01	1.3E+04	1.1E+00	3.9E+04	1.1E+00	4.0E+04
Cs-138	1.4E+00	5.3E+04	5.8E+00	2.1E+05	1.7E+01	6.4E+05	5.2E+01	1.9E+06	5.3E+01	2.0E+06
Ba-137m	3.0E-02	1.1E+03	1.2E-01	4.4E+03	3.6E-01	1.3E+04	1.1E+00	3.9E+04	1.1E+00	4.0E+04
H-3	9.9E-02	3.6E+03	3.9E-01	1.5E+04	1.2E+00	4.4E+04	3.6E+00	1.3E+05	3.6E+00	1.3E+05
Na-24	2.9E-03	1.1E+02	1.2E-02	4.3E+02	3.5E-02	1.3E+03	1.0E-01	3.8E+03	1.1E-01	3.9E+03
P-32	1.2E-04	4.3E+00	4.7E-04	1.7E+01	1.4E-03	5.2E+01	4.2E-03	1.5E+02	4.3E-03	1.6E+02
Cr-51	2.8E-03	1.0E+02	1.1E-02	4.2E+02	3.4E-02	1.3E+03	1.0E-01	3.8E+03	1.0E-01	3.9E+03
Mn-54	1.4E-03	5.2E+01	5.6E-03	2.1E+02	1.7E-02	6.3E+02	5.1E-02	1.9E+03	5.2E-02	1.9E+03
Mn-56	5.3E-03	2.0E+02	2.1E-02	7.8E+02	6.3E-02	2.3E+03	1.9E-01	7.0E+03	2.0E-01	7.2E+03
Fe-55	2.9E-03	1.1E+02	1.2E-02	4.3E+02	3.5E-02	1.3E+03	1.1E-01	3.9E+03	1.1E-01	4.0E+03
Fe-59	7.7E-04	2.9E+01	3.1E-03	1.1E+02	9.3E-03	3.4E+02	2.8E-02	1.0E+03	2.9E-02	1.1E+03
Co-58	6.7E-04	2.5E+01	2.7E-03	9.9E+01	8.0E-03	3.0E+02	2.4E-02	8.9E+02	2.5E-02	9.1E+02

Time	2.0 h		8.0 h		24.0 h		72.0 h		78.0 h	
Nuclide	Ci	MBq	Ci	MBq	Ci	MBq	Ci	MBq	Ci	MBq
Co-60	1.3E-03	4.8E+01	5.2E-03	1.9E+02	1.6E-02	5.8E+02	4.7E-02	1.7E+03	4.8E-02	1.8E+03
Ni-63	2.9E-06	1.1E-01	1.2E-05	4.3E-01	3.5E-05	1.3E+00	1.1E-04	3.9E+00	1.1E-04	4.0E+00
Cu-64	1.4E-02	5.1E+02	5.5E-02	2.0E+03	1.6E-01	6.1E+03	4.9E-01	1.8E+04	5.1E-01	1.9E+04
Zn-65	6.0E-04	2.2E+01	2.4E-03	8.9E+01	7.2E-03	2.7E+02	2.2E-02	8.0E+02	2.2E-02	8.2E+02
Sr-89	2.1E-03	7.9E+01	8.6E-03	3.2E+02	2.6E-02	9.5E+02	7.7E-02	2.9E+03	7.9E-02	2.9E+03
Sr-90	1.1E-04	4.0E+00	4.4E-04	1.6E+01	1.3E-03	4.9E+01	3.9E-03	1.5E+02	4.0E-03	1.5E+02
Y-90	1.1E-04	4.0E+00	4.4E-04	1.6E+01	1.3E-03	4.9E+01	3.9E-03	1.5E+02	4.0E-03	1.5E+02
Sr-91	1.3E+00	4.8E+04	5.2E+00	1.9E+05	1.6E+01	5.8E+05	4.7E+01	1.7E+06	4.8E+01	1.8E+06
Sr-92	2.8E+00	1.0E+05	1.1E+01	4.2E+05	3.4E+01	1.3E+06	1.0E+02	3.8E+06	1.0E+02	3.9E+06
Y-91	2.9E-02	1.1E+03	1.1E-01	4.2E+03	3.4E-01	1.3E+04	1.0E+00	3.8E+04	1.1E+00	3.9E+04
Y-92	8.8E-01	3.3E+04	3.5E+00	1.3E+05	1.1E+01	3.9E+05	3.2E+01	1.2E+06	3.3E+01	1.2E+06
Y-93	9.5E-02	3.5E+03	3.8E-01	1.4E+04	1.1E+00	4.2E+04	3.4E+00	1.3E+05	3.5E+00	1.3E+05
Zr-95	6.0E-02	2.2E+03	2.4E-01	8.9E+03	7.2E-01	2.7E+04	2.2E+00	8.0E+04	2.2E+00	8.2E+04
Nb-95	6.0E-02	2.2E+03	2.4E-01	8.9E+03	7.2E-01	2.7E+04	2.2E+00	8.0E+04	2.2E+00	8.2E+04
Mo-99	2.8E-01	1.0E+04	1.1E+00	4.1E+04	3.3E+00	1.2E+05	1.0E+01	3.7E+05	1.0E+01	3.8E+05
Tc-99m	2.8E-01	1.0E+04	1.1E+00	4.1E+04	3.3E+00	1.2E+05	1.0E+01	3.7E+05	1.0E+01	3.8E+05
Ru-103	1.4E-02	5.3E+02	5.8E-02	2.1E+03	1.7E-01	6.4E+03	5.2E-01	1.9E+04	5.3E-01	2.0E+04
Rh-103m	1.4E-02	5.3E+02	5.8E-02	2.1E+03	1.7E-01	6.4E+03	5.2E-01	1.9E+04	5.3E-01	2.0E+04
Ru-106	2.1E-03	7.9E+01	8.6E-03	3.2E+02	2.6E-02	9.5E+02	7.7E-02	2.9E+03	7.9E-02	2.9E+03
Rh-106	2.1E-03	7.9E+01	8.6E-03	3.2E+02	2.6E-02	9.5E+02	7.7E-02	2.9E+03	7.9E-02	2.9E+03
Ag-110m	2.9E-06	1.1E-01	1.2E-05	4.3E-01	3.5E-05	1.3E+00	1.1E-04	3.9E+00	1.1E-04	4.0E+00

Time	2.0 h		8.0 h		24.0 h		72.0 h		78.0 h	
Nuclide	**Ci**	**MBq**	**Ci**	**MBq**	**Ci**	**MBq**	**Ci**	**MBq**	**Ci**	**MBq**
Te-129m	2.9E-02	1.1E+03	1.1E-01	4.2E+03	3.4E-01	1.3E+04	1.0E+00	3.8E+04	1.1E+00	3.9E+04
Te-131m	3.9E-02	1.4E+03	1.6E-01	5.7E+03	4.7E-01	1.7E+04	1.4E+00	5.2E+04	1.4E+00	5.3E+04
Te-132	6.7E-03	2.5E+02	2.7E-02	9.9E+02	8.0E-02	3.0E+03	2.4E-01	8.9E+03	2.5E-01	9.1E+03
Ba-140	3.0E-01	1.1E+04	1.2E+00	4.4E+04	3.6E+00	1.3E+05	1.1E+01	4.0E+05	1.1E+01	4.1E+05
La-140	3.0E-01	1.1E+04	1.2E+00	4.4E+04	3.6E+00	1.3E+05	1.1E+01	4.0E+05	1.1E+01	4.1E+05
Ce-141	1.4E-02	5.3E+02	5.8E-02	2.1E+03	1.7E-01	6.4E+03	5.2E-01	1.9E+04	5.3E-01	2.0E+04
Ce-144	2.1E-03	7.9E+01	8.6E-03	3.2E+02	2.6E-02	9.5E+02	7.7E-02	2.9E+03	7.9E-02	2.9E+03
Pr-144	2.1E-03	7.9E+01	8.6E-03	3.2E+02	2.6E-02	9.5E+02	7.7E-02	2.9E+03	7.9E-02	2.9E+03
W-187	6.7E-04	2.5E+01	2.7E-03	9.9E+01	8.0E-03	3.0E+02	2.4E-02	8.9E+02	2.5E-02	9.1E+02
Np-239	2.3E-01	8.5E+03	9.2E-01	3.4E+04	2.7E+00	1.0E+05	8.2E+00	3.1E+05	8.5E+00	3.1E+05

Table 15.5-44: Conservatisms Used in the Non-LOCA DSA

Analysis Type	Event Type	Conservatism	
		Code	**Plant Parameters and System Performances**
BL-DSA	AOO	Best Estimate	• Rated power initial conditions • Conservatism in the plant parameters and the derived acceptance criteria is established conservatively such that there is no need to account for uncertainties in the analysis method • DL2 functions are primarily credited to meet acceptance criteria. • If a DL3 function is necessary in the BL-DSA to mitigate a less frequent AOO PIE, a demonstration has to be provided that it is not practicable to implement a DL2 mitigation function, and an EX-DSA has to also be performed assuming CCF of DL3 mitigating functions.
CN-DSA	DBA	Graded Approach	• Conservative initial conditions are established for events that are limiting • Conservative setpoints and plant performance parameters • Analysis method uncertainties are addressed with a graded approach depending on margin to the derived acceptance criteria as described in Subsection 15.5.1.1 • DL3 functions are credited in the analyses • Application of limiting single active component failures among equipment performing DL3 functions

Analysis Type	Event Type	Conservatism	
		Code	Plant Parameters and System Performances
EX-DSA	DEC	Best Estimate	• Rated power initial conditions are used • Nominal setpoints and plant performance parameters are used (conservative setpoints and plant performance may be used for convenience/simplification) • Sensitivity analyses performed to understand cliff edge effects • Any available DL function that is not disabled by the PIE or assumed failures in the event sequences may be credited

Table 15.5-45: DL3 Functions Credited in Conservative LOCA Analyses

Credited Function	Action	Signal	Analytical Setpoint
DL3-02	Hydraulic scram	Low steam RPV pressure	5.617 MPa
DL3-07	Hydraulic scram	High containment pressure	2.8 psig (18.5 kPaG)
DL3-09	Hydraulic scram	Line Break Indication (MS, FW, ICS)	For breaks larger than 19 mm in diameter (Note 1)
DL3-14	ICS initiation	Low RPV water level	14.22 m
DL3-15	ICS initiation	High Containment pressure	2.8 psig (18.5 kPaG)
DL3-16	ICS initiation	Line Break Indication (MS, FW, ICS)	For breaks larger than 19 mm in diameter (Note 1)
DL3-17	MSRIV closure (Note 2)	Low steam RPV pressure	5.617 MPa
DL3-18	MSRIV closure (Note 2)	Low RPV water level	14.22 m
DL3-22	Reactor and Containment Isolation Valve closure (Note 2)	High Containment pressure	2.8 psig (18.5 kPaG) (Note 1)
DL3-20 DL3-21	MSRIV closure	Line break indication in MSL, FW or Shutdown Cooling System (SDC)	Within 1 second for breaks larger than 19 mm in diameter
DL3-25	FW and SDC RIV closure	Line break indication in FW or SDC	Within 1 second for breaks larger than 19 mm in diameter
DL3-26	CUW RIV closure	Line break indication in CUW	Within 1 second for breaks larger than 19 mm in diameter
DL3-27 DL3-28 DL3-29	ICS RIV closure of the broken ICS train	Line break indication in the respective ICS trains	Within 1 second for breaks larger than 19 mm in diameter

Notes:

1. This setpoint is reached in less than 1 s in large break cases.
2. For large breaks, isolation valves are assumed to start closing with a 5 second delay from the time of pipe break and are fully closed in 10 seconds. For small breaks, the isolation valves start closing with a 5 second delay after the setpoint is reached and are fully closed in another 5 seconds. Containment isolation is credited only for FW pipe breaks. Isolation functions may also include CIVs. This table shows only the valves credited to close by the isolation signal.

Table 15.5-46: DL2 and DL4a Functions Credited in DEC LOCA Analyses

DL3 Function Failure	Action	DL2 or DL4a Credited Function	Notes
DL3-02	Hydraulic scram on low steam pipe pressure	DL2-08 on turbine trip demand	DL2-08 is initiated on LOPP, and a faster scram than DL3-02 credited in CN-DSA.
DL3-07	Hydraulic scram on high containment pressure	DL4a-05	
DL3-09	Hydraulic scram on Line Break Indication (MS, FW, ICS)	DL4a-26	
DL3-14	ICS initiation on Low RPV water level	DL4a-33	
DL3-15	ICS initiation on High Containment pressure	DL4a-11	
DL3-16	ICS initiation on Line Break Indication (MS, FW, ICS)	DL4a-27 and DL4a-28	
DL3-17	MSRIV closure on Low steam pipe pressure	DL2-41	DL3-17 is credited in small break cases for limiting inventory losses to the turbine. In DL2, the pressure controller throttles the flow until DL2-41 closes the MSRIVs. This combination of functions results in less inventory losses than the CN-DSA sequences.
DL3-18	MSRIV closure on Low RPV water level	DL4a-34	
DL3-22	Reactor and Containment Isolation Valve closure on High Containment pressure	DL4a-16	
DL3-20 DL3-21	MSRIV closure on Line break indication in MSL, FW or SDC	DL4a-14 and DL4a-15	
DL3-25	FW and SDC RIV closure on line break indication in FW or SDC	DL4a-17	
DL3-27 DL3-28 DL3-29	ICRIV closure of the broken ICS train line break indication in the respective ICS train	None	See discussion in Subsection 15.2.4.6.4

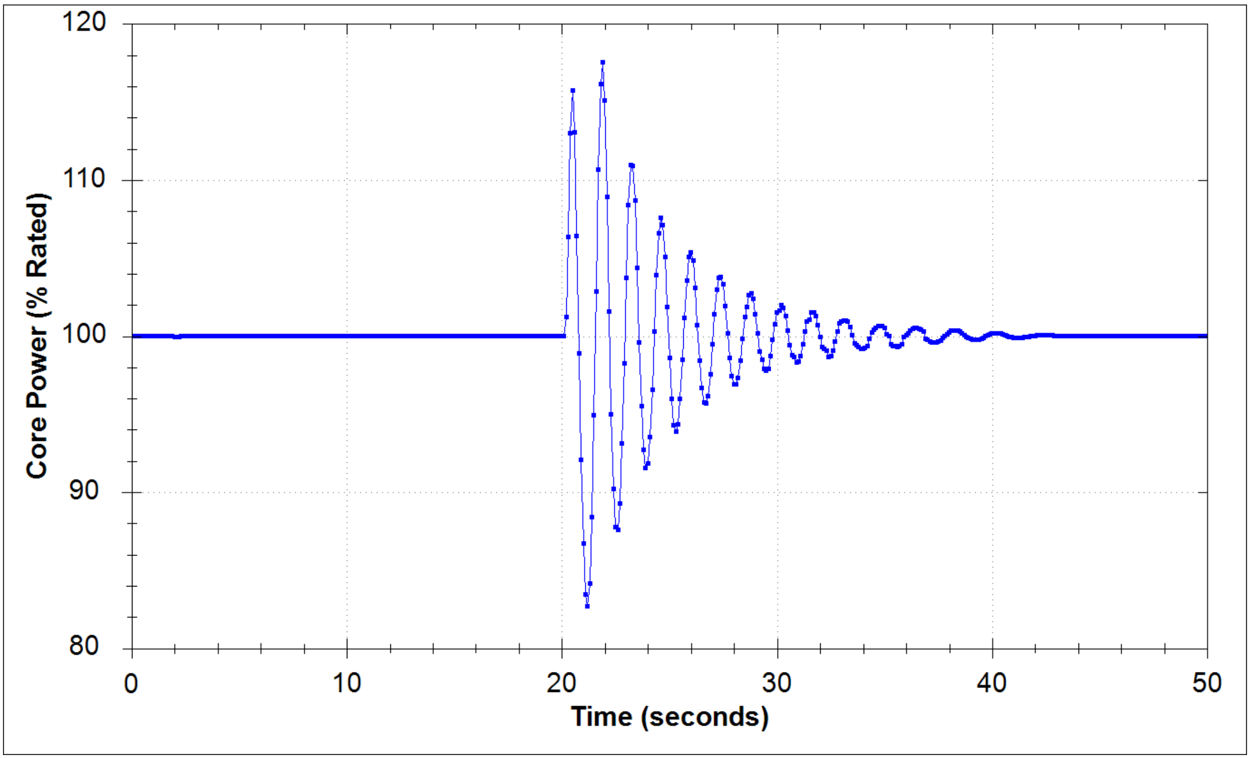

Figure 15.5-3: MOC Transient Results for Core-Wide Stability Analysis

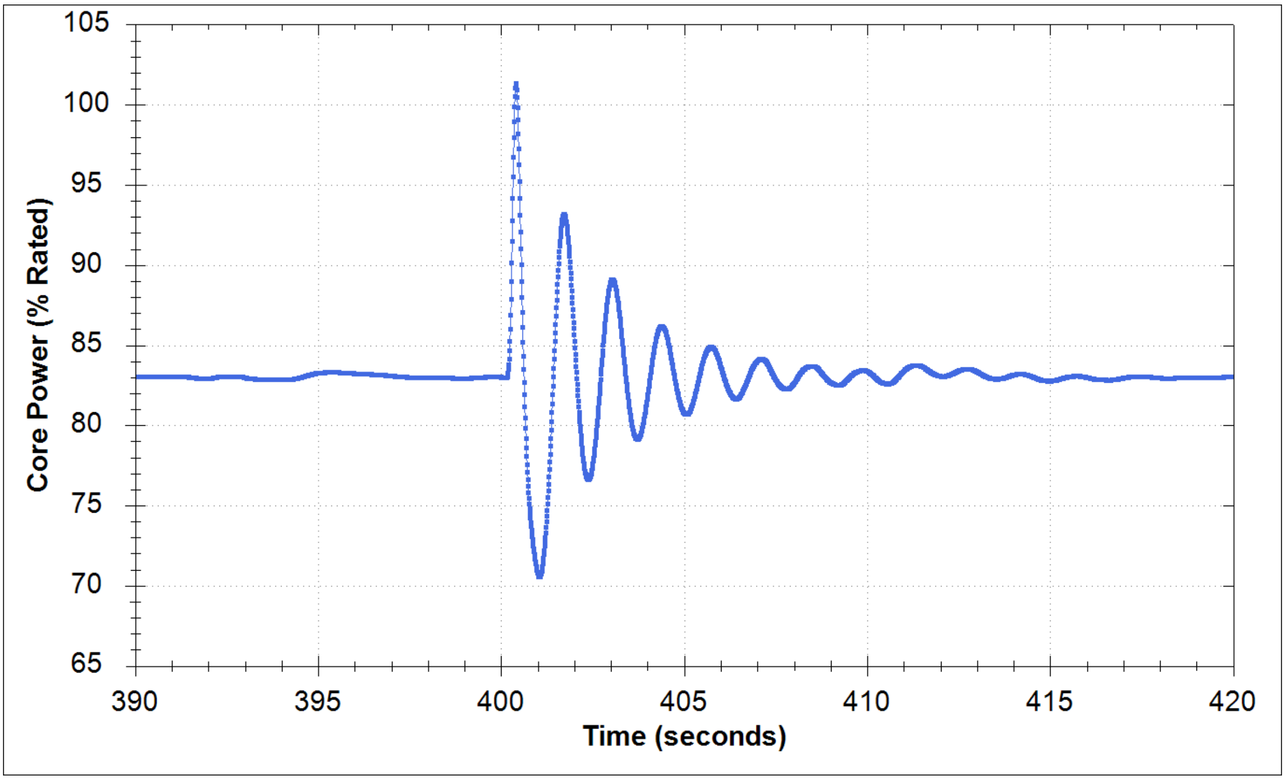

Figure 15.5-4: MOC Transient Results for LFWH with Selected Control Rod Run-in (SCCRI) Core-Wide Stability Analysis

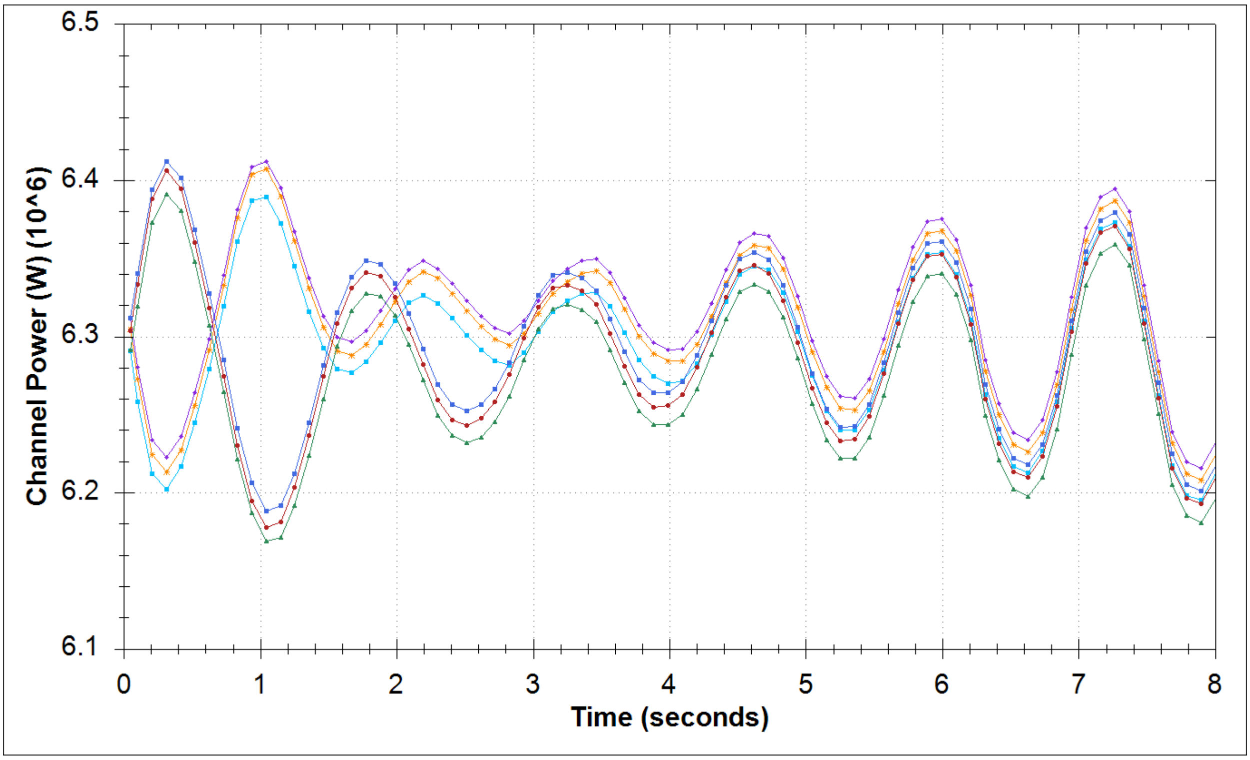

Figure 15.5-5: Regional Mode Stability Response at MOC for FW Temperature of 241.9°C

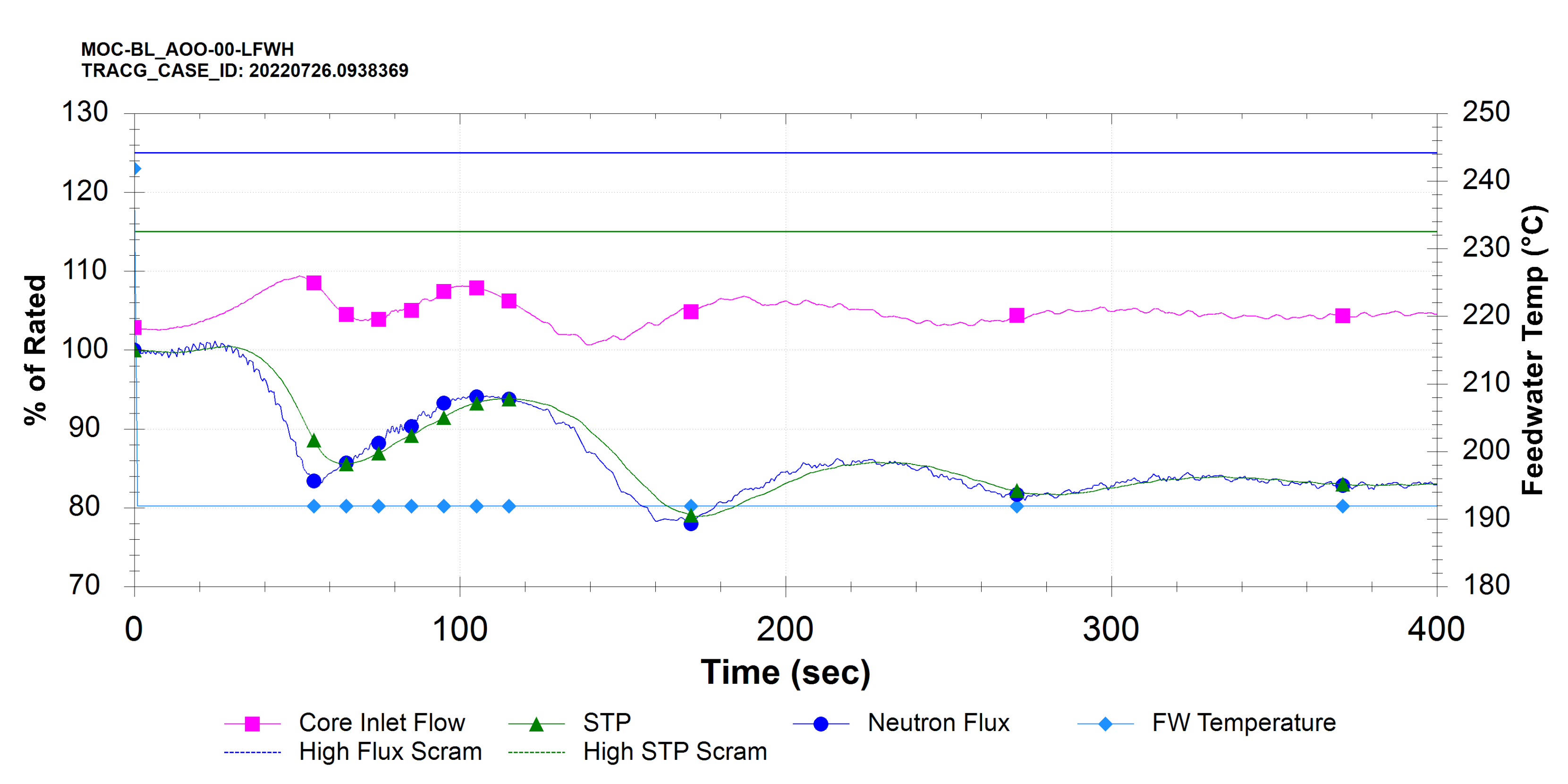

Figure 15.5-6: Loss of Feedwater Heating (AOO)

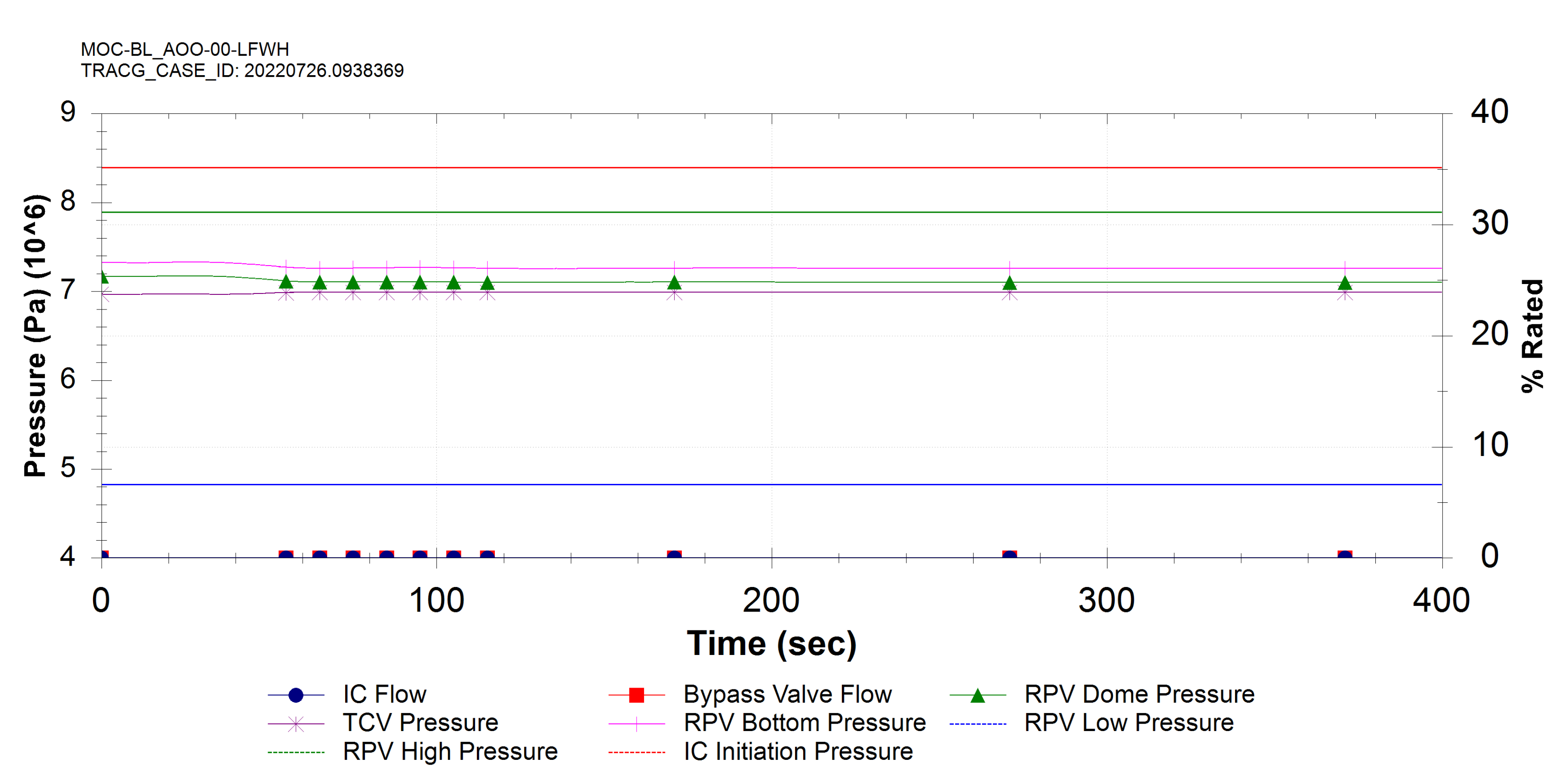

Figure 15.5-7: Loss of Feedwater Heating (AOO)

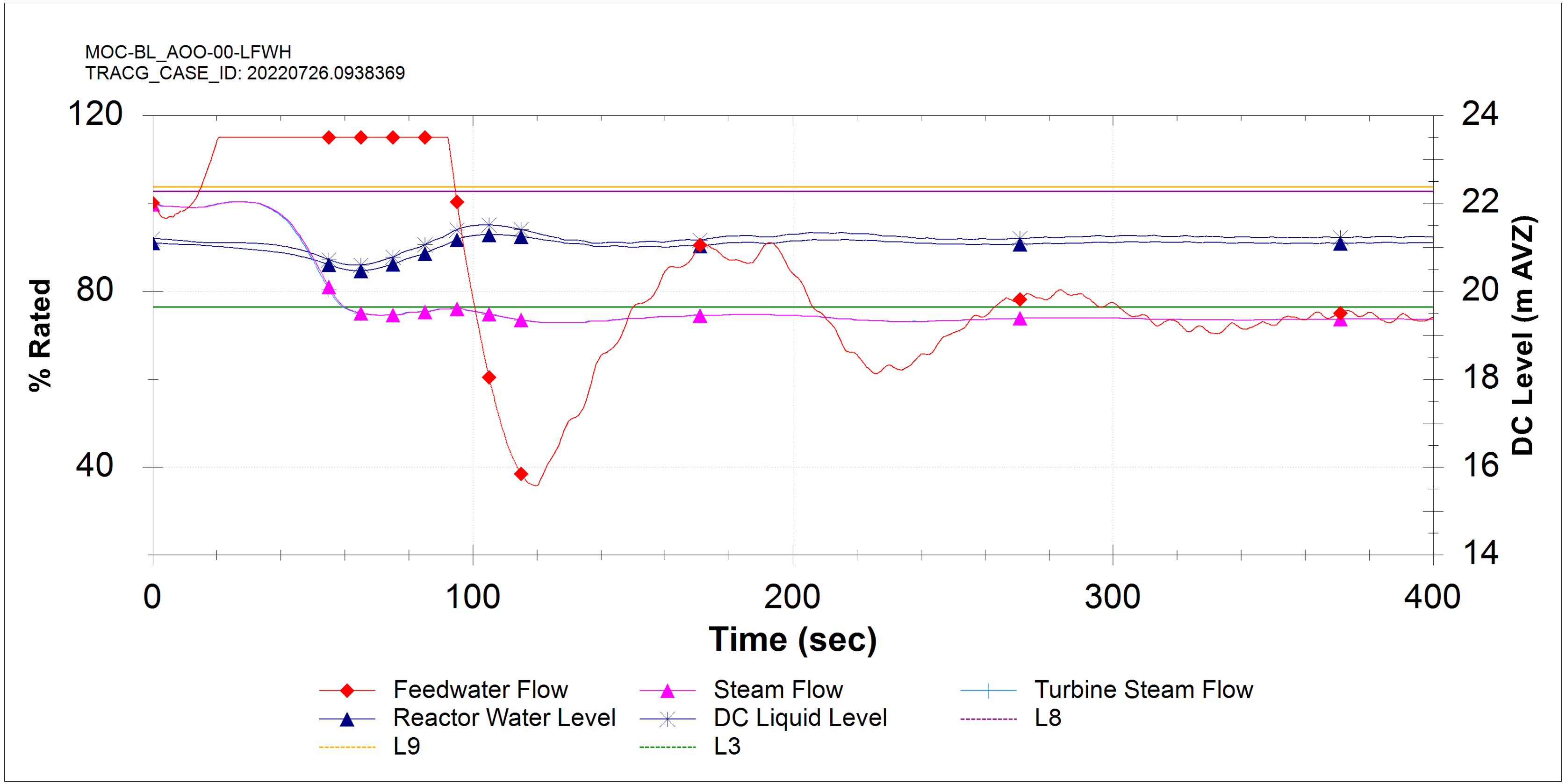

Figure 15.5-8: Loss of Feedwater Heating (AOO)

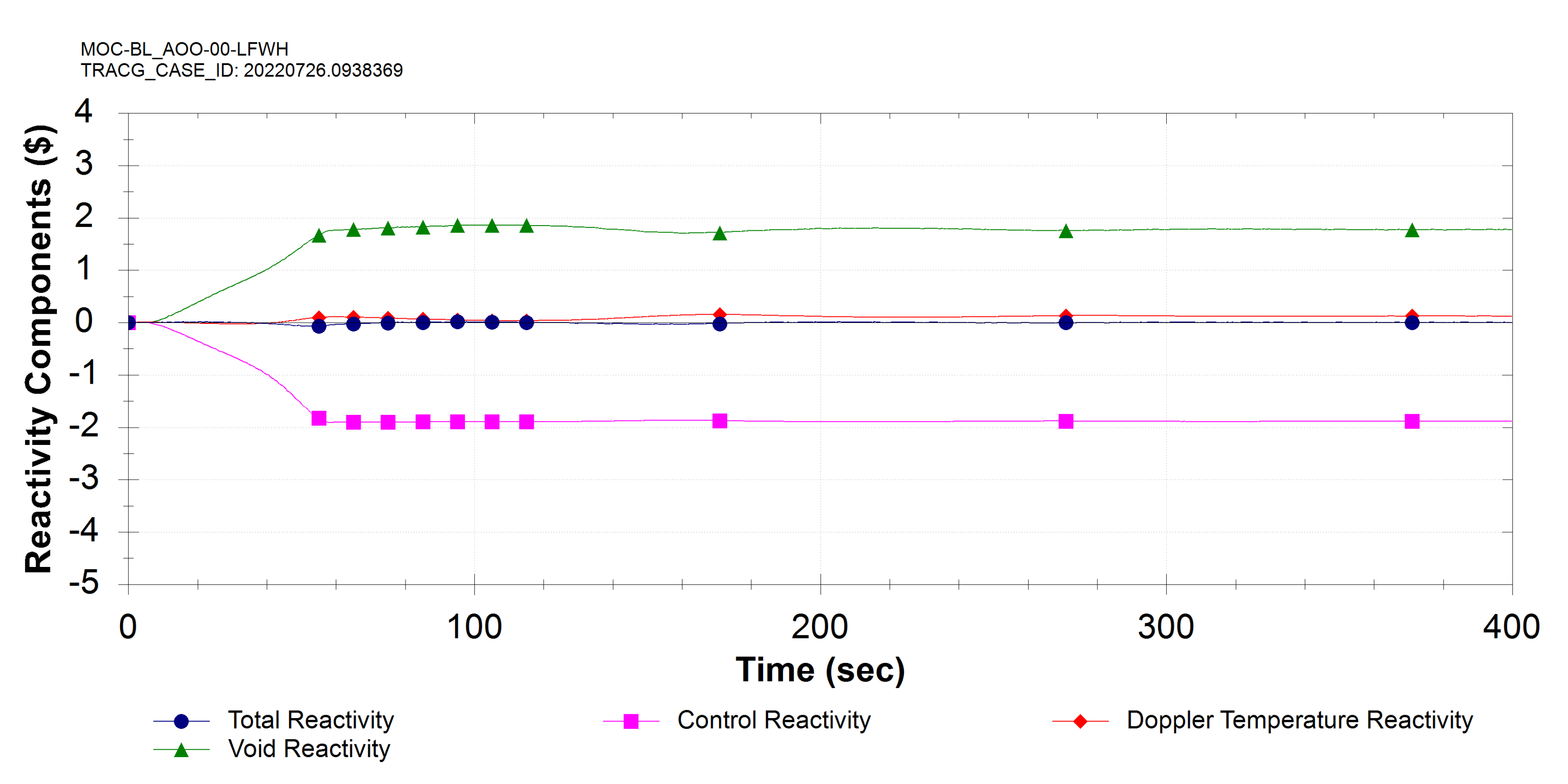

Figure 15.5-9: Loss of Feedwater Heating (AOO)

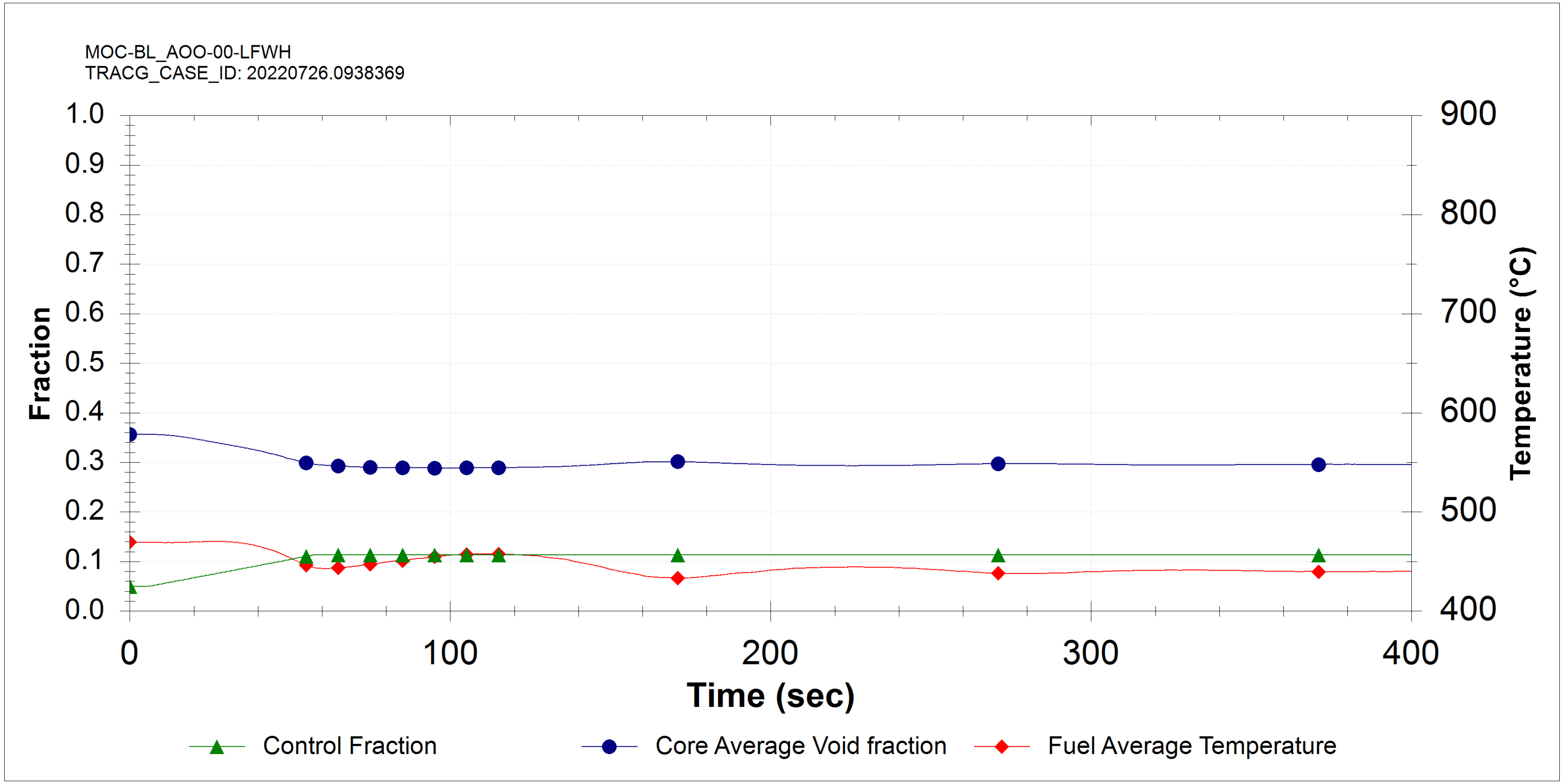

Figure 15.5-10: Loss of Feedwater Heating (AOO)

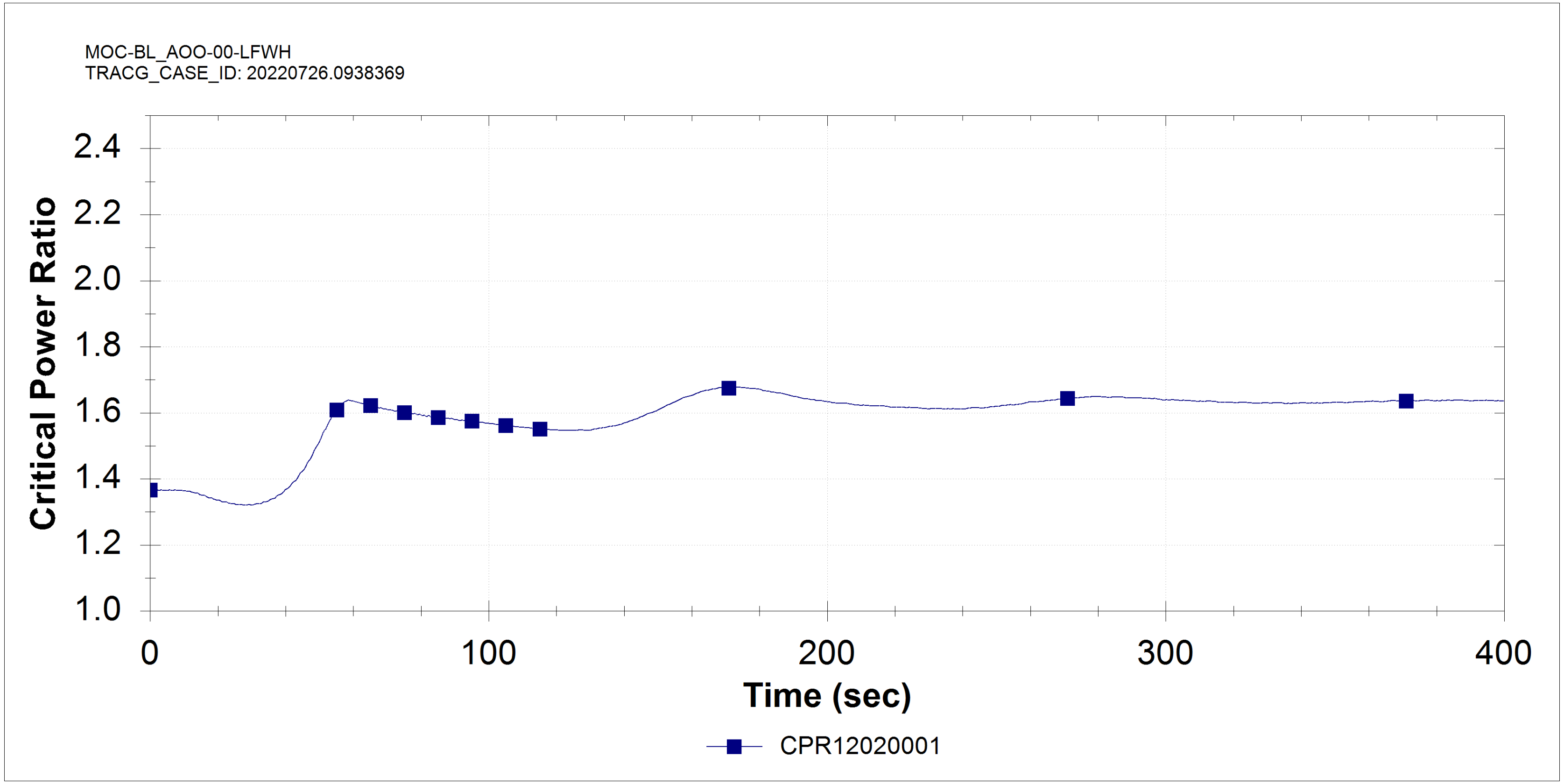

Figure 15.5-11: Loss of Feedwater Heating (AOO)

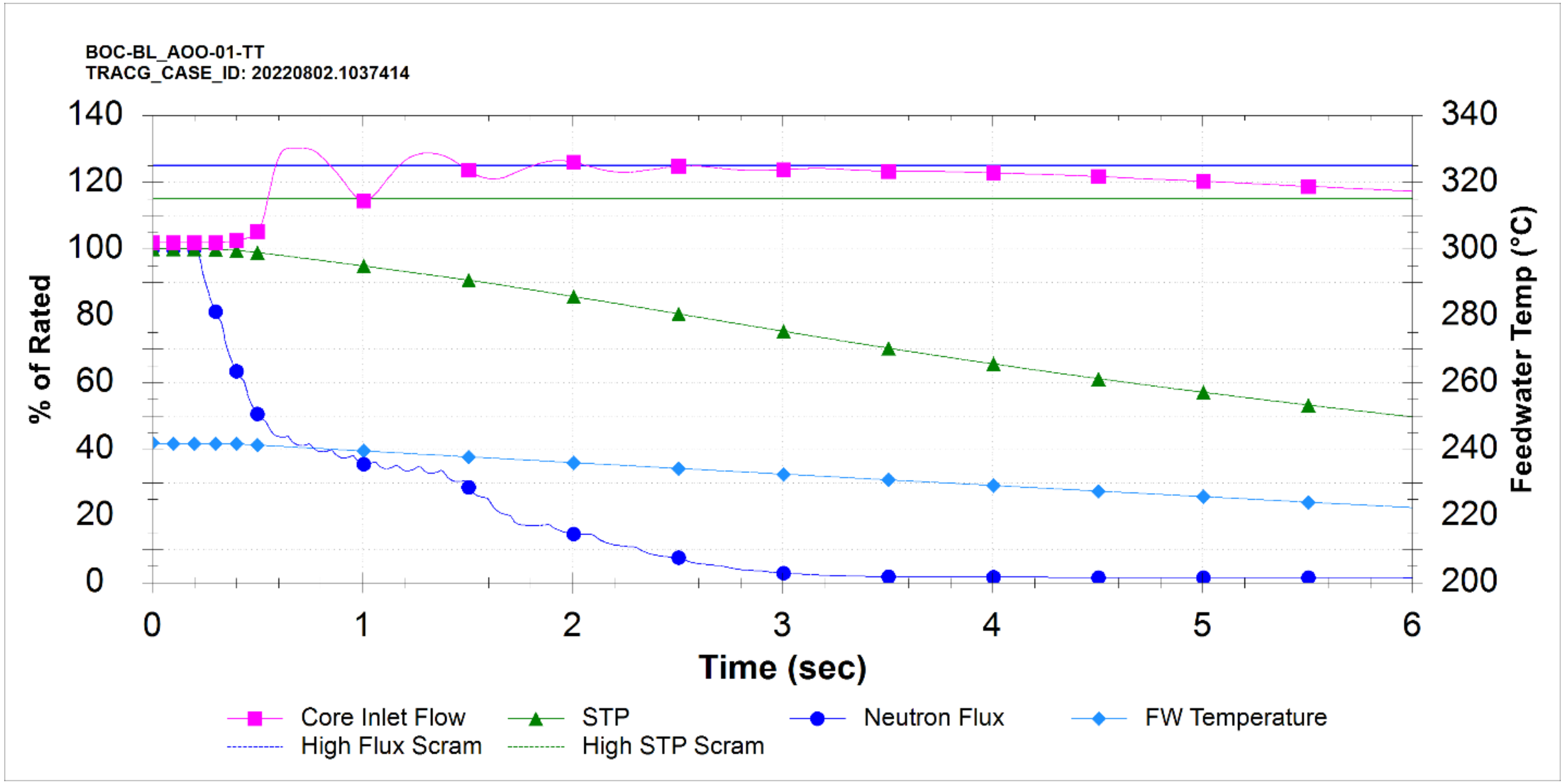

Figure 15.5-12: Turbine Trip (AOO)

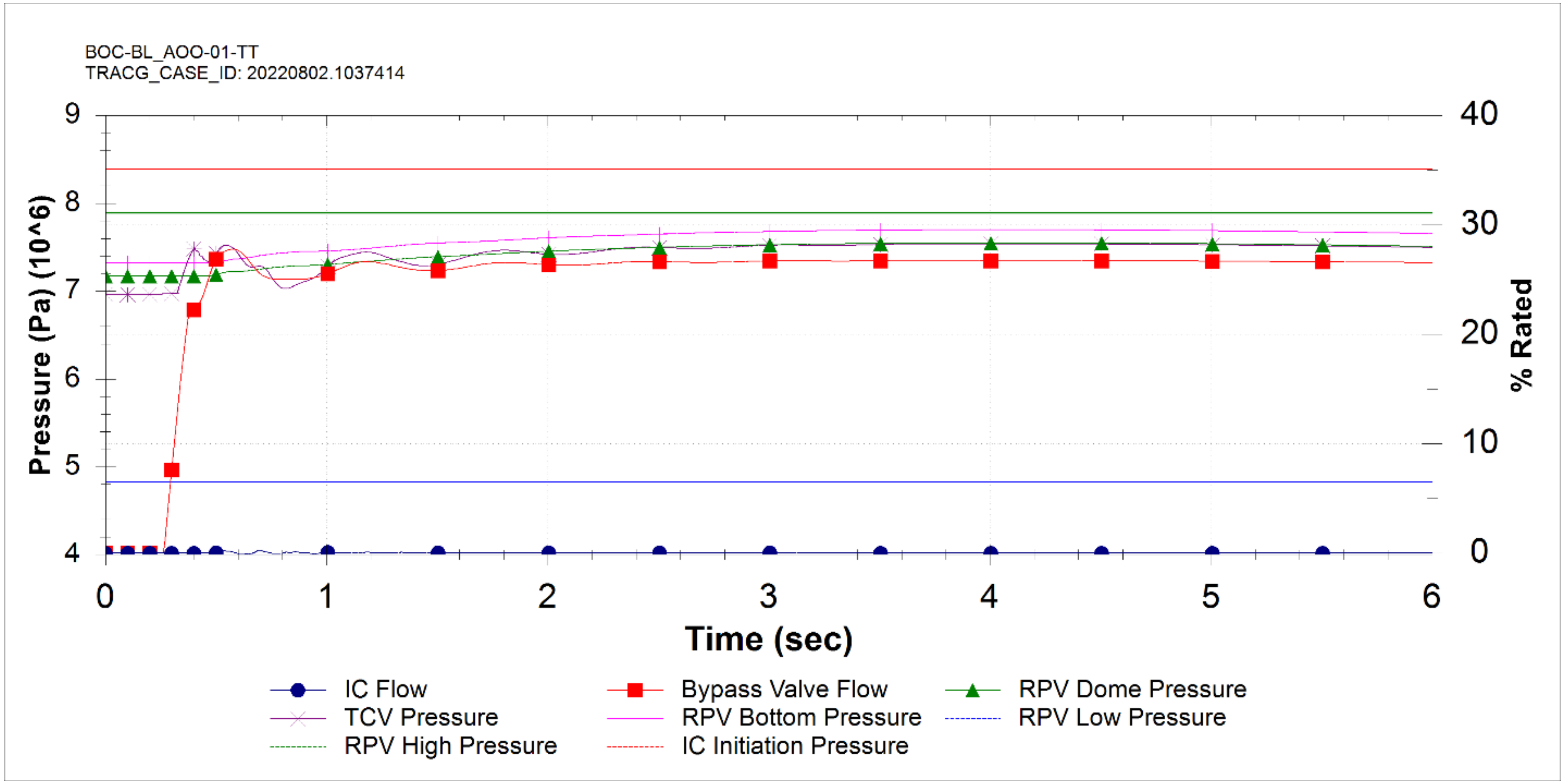

Figure 15.5-13: Turbine Trip (AOO)

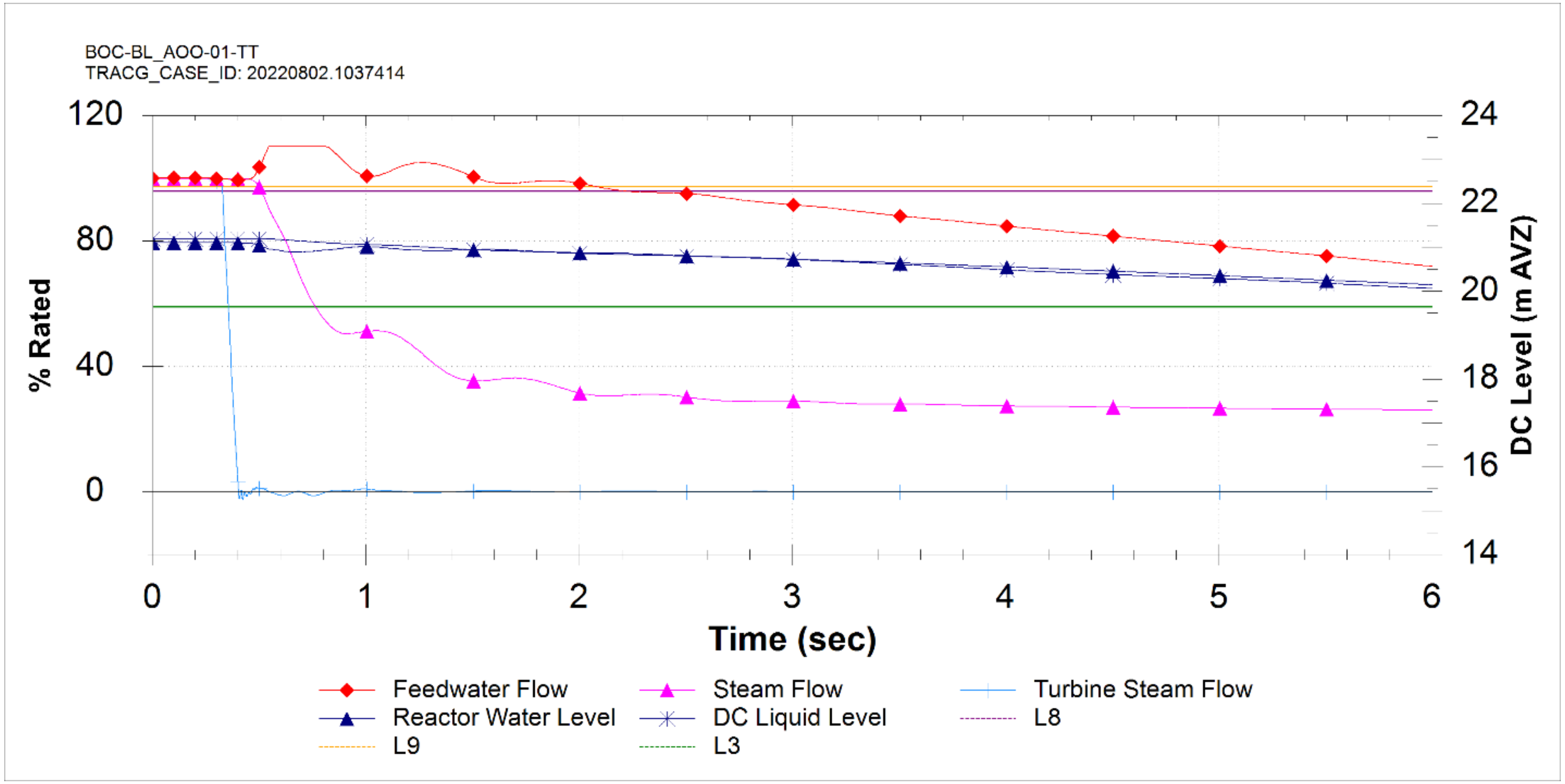

Figure 15.5-14: Turbine Trip (AOO)

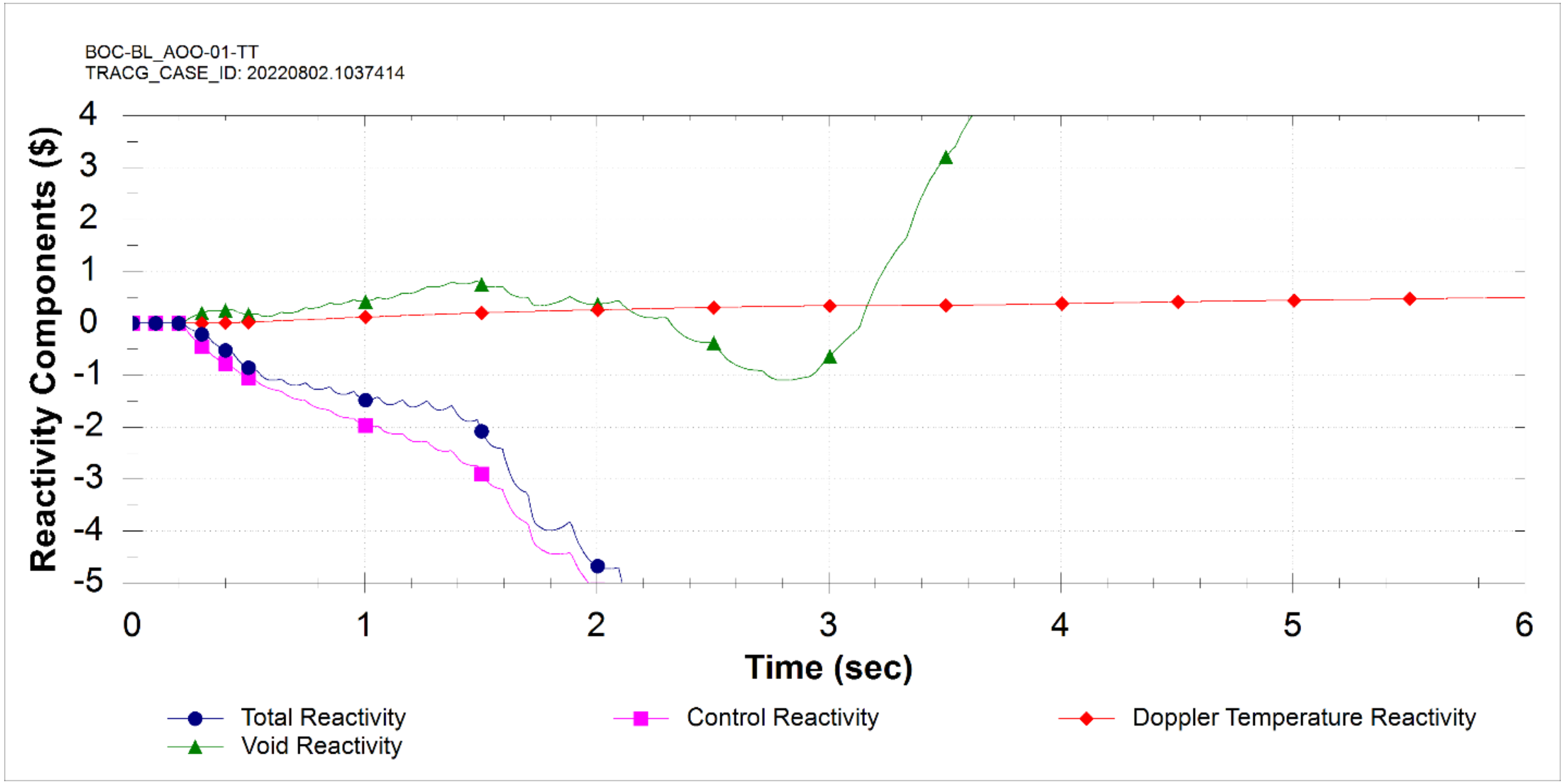

Figure 15.5-15: Turbine Trip (AOO)

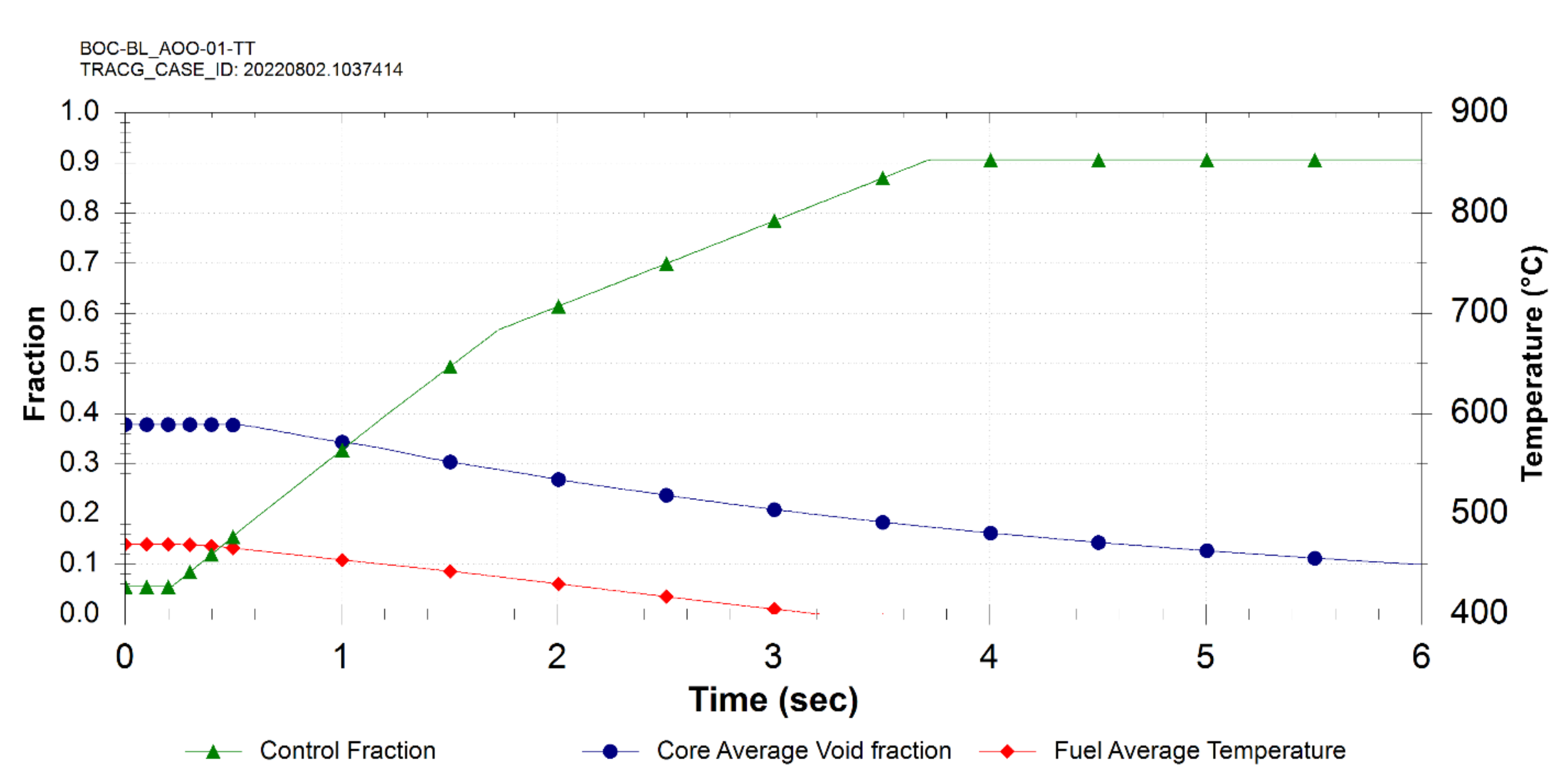

Figure 15.5-16: Turbine Trip (AOO)

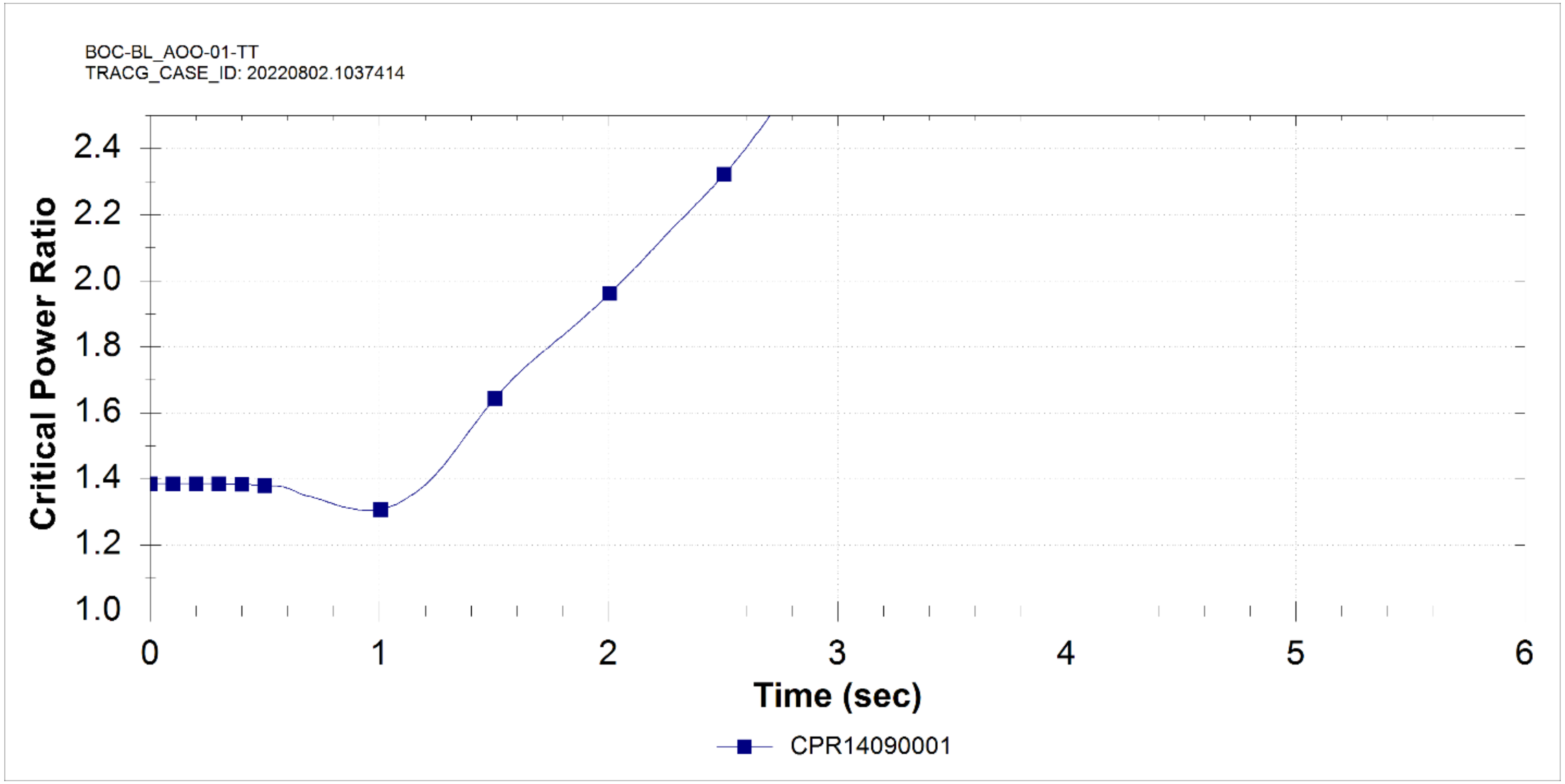

Figure 15.5-17: Turbine Trip (AOO)

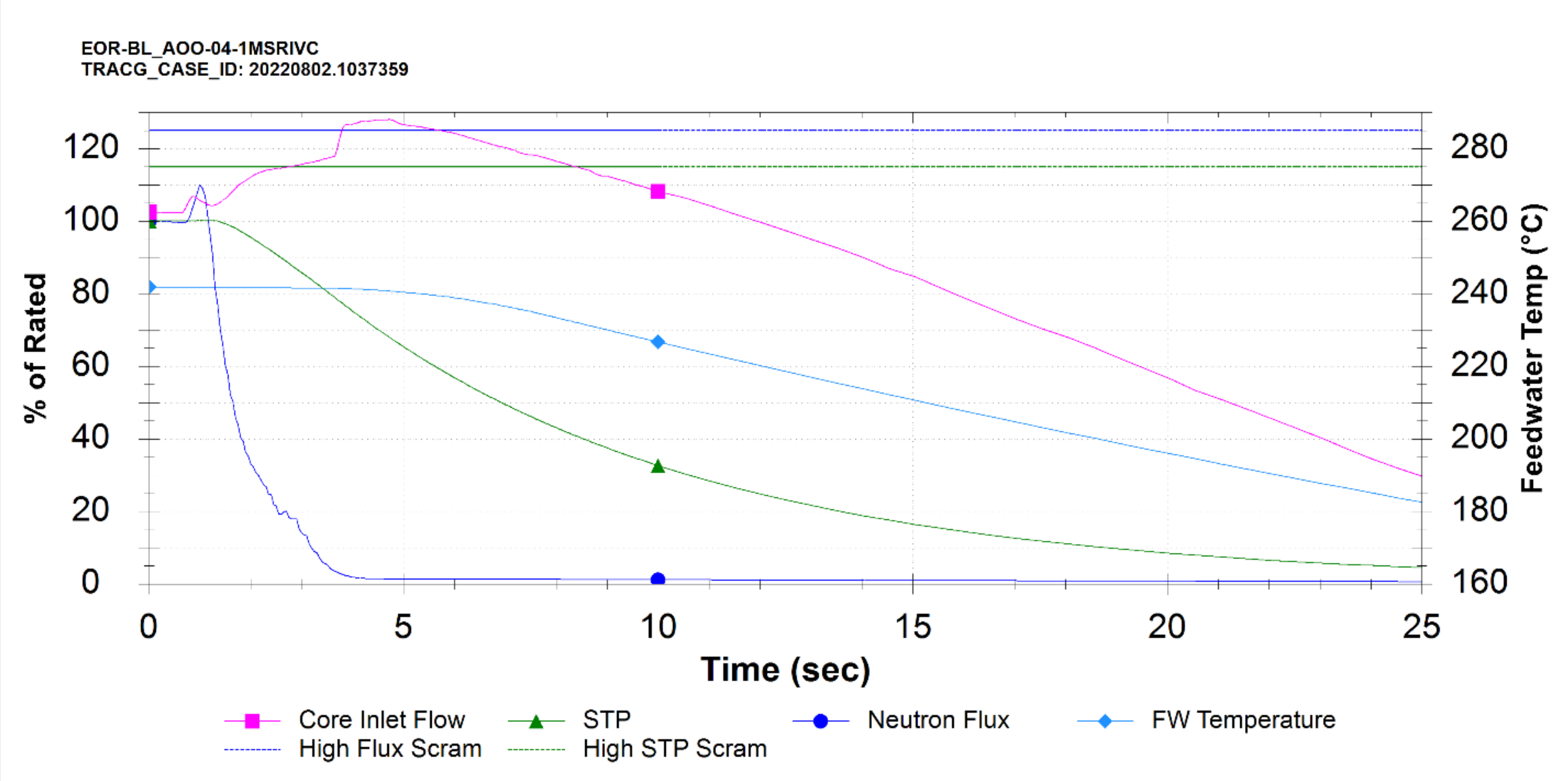

Figure 15.5-18: Closure of One Main Steam Reactor Isolation Valve (AOO)

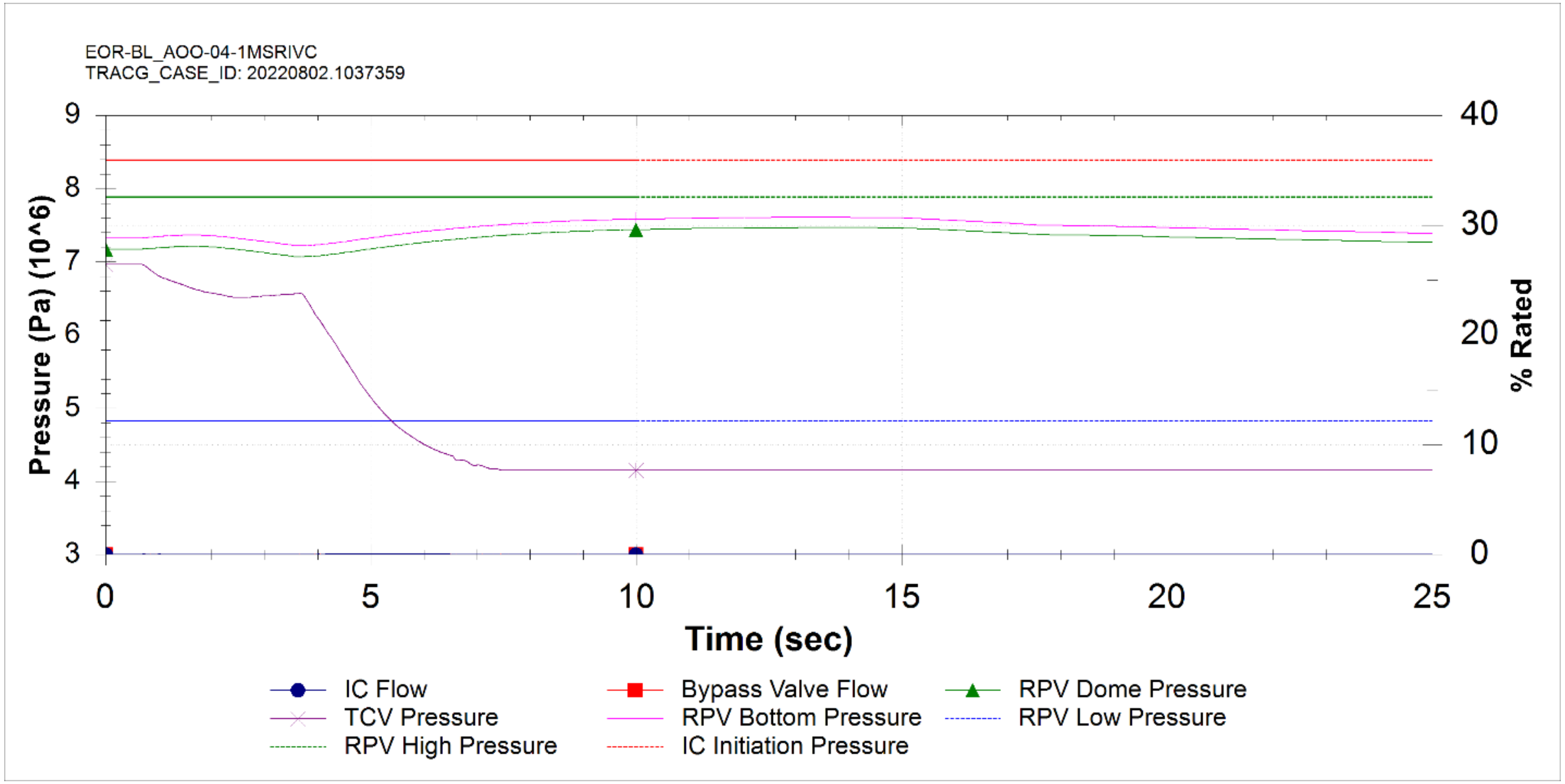

Figure 15.5-19: Closure of One Main Steam Reactor Isolation Valve (AOO)

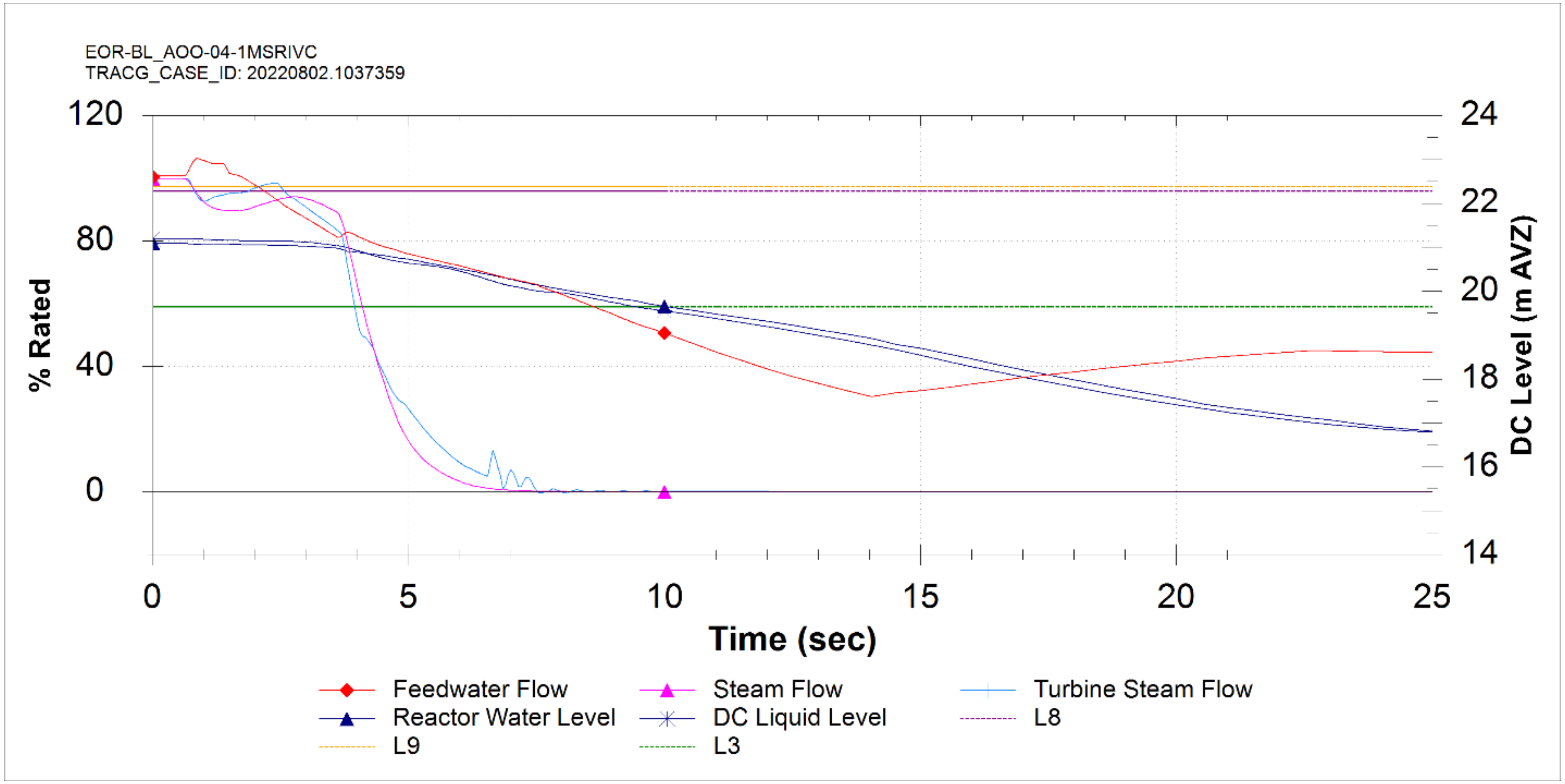

Figure 15.5-20: Closure of One Main Steam Reactor Isolation Valve (AOO)

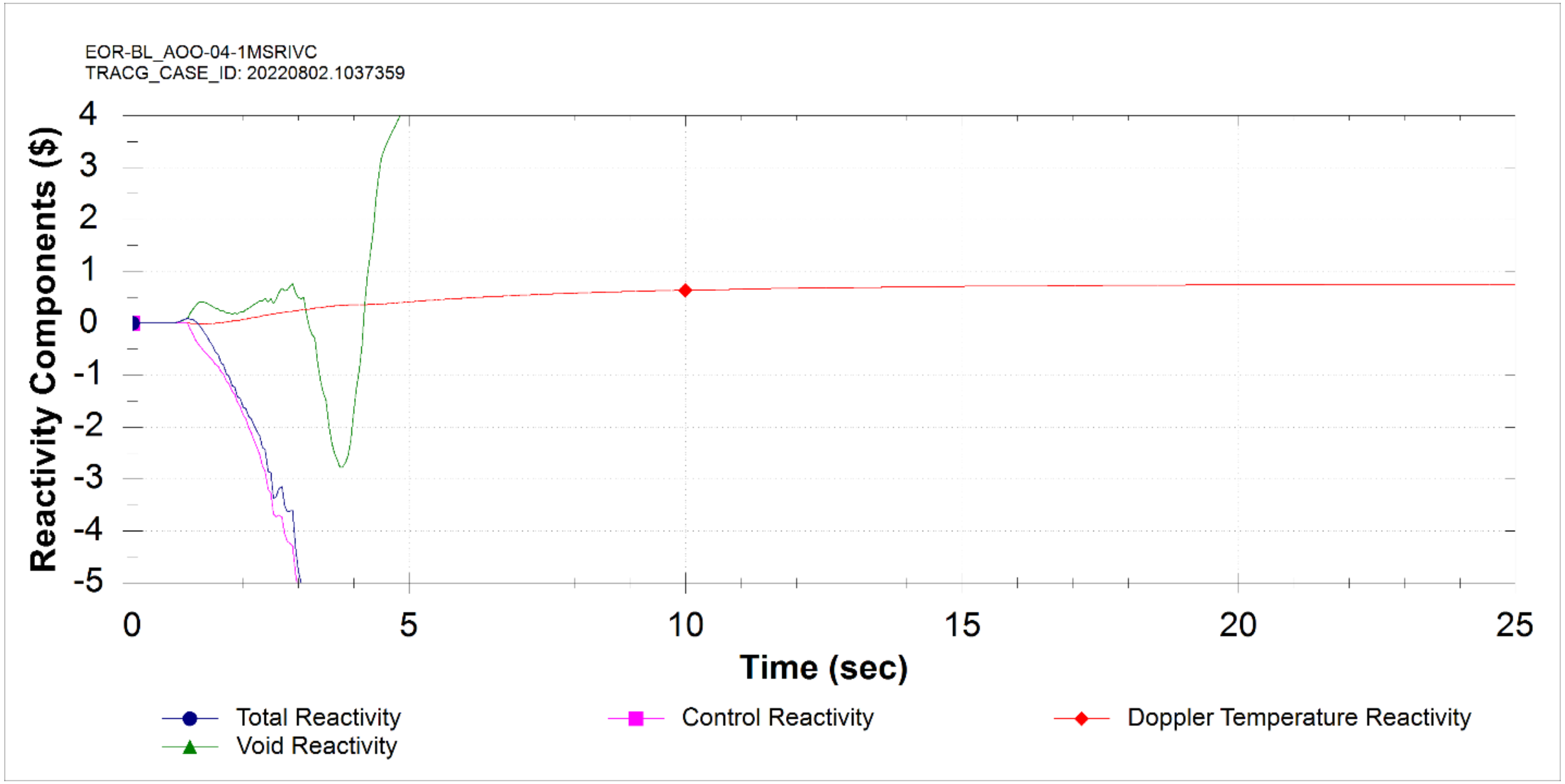

Figure 15.5-21: Closure of One Main Steam Reactor Isolation Valve (AOO)

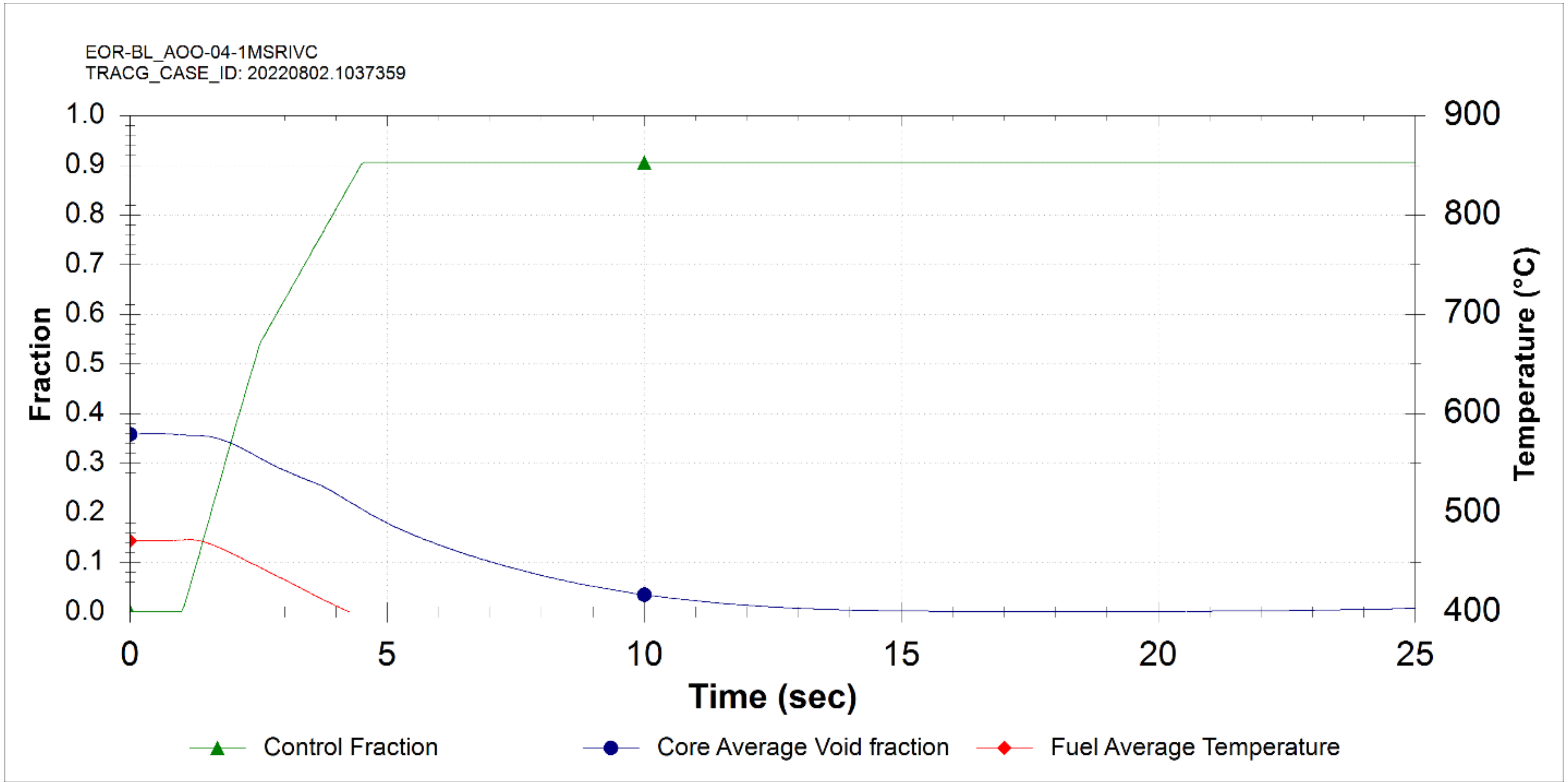

Figure 15.5-22: Closure of One Main Steam Reactor Isolation Valve (AOO)

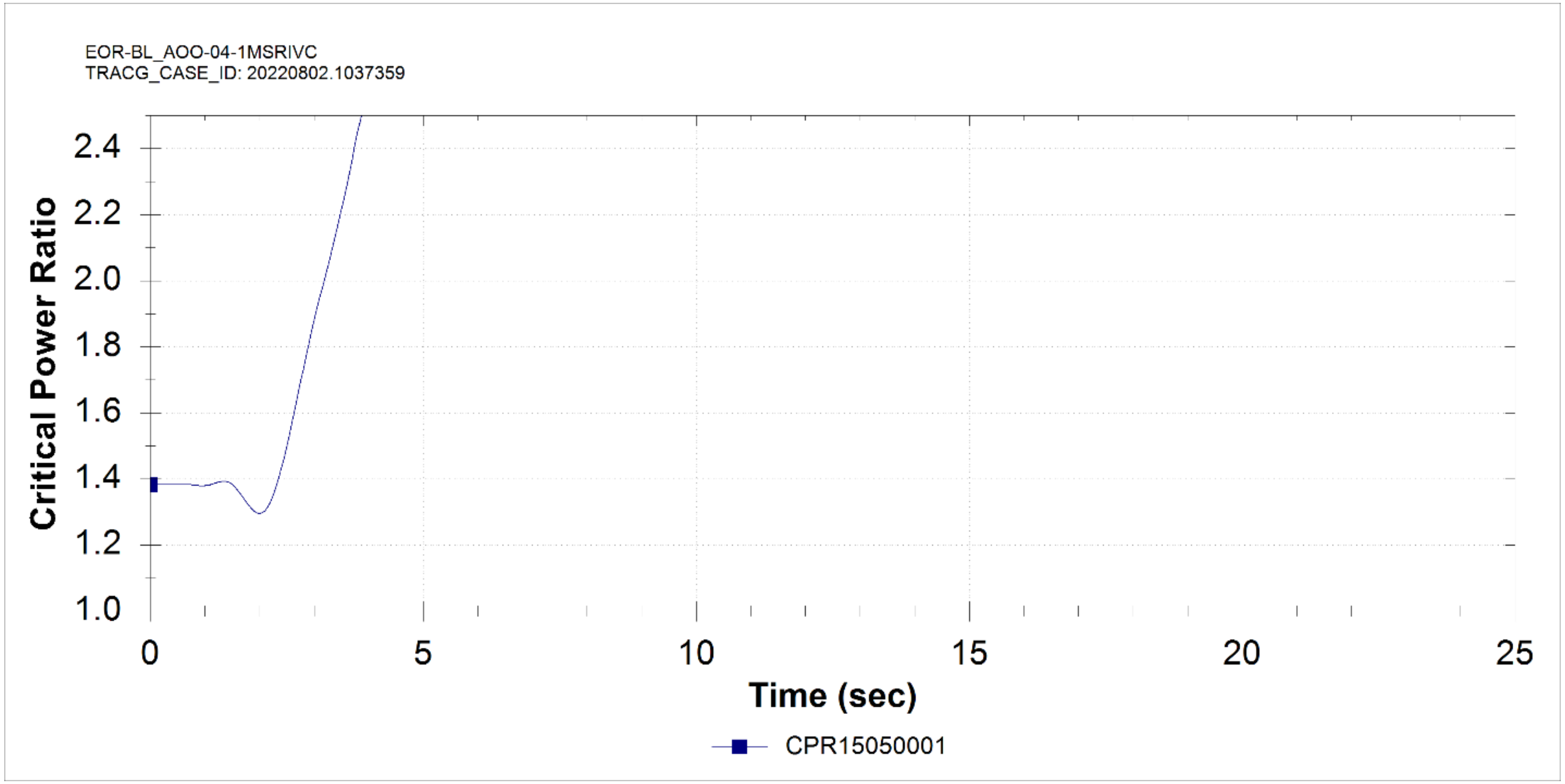

Figure 15.5-23: Closure of One Main Steam Reactor Isolation Valve (AOO)

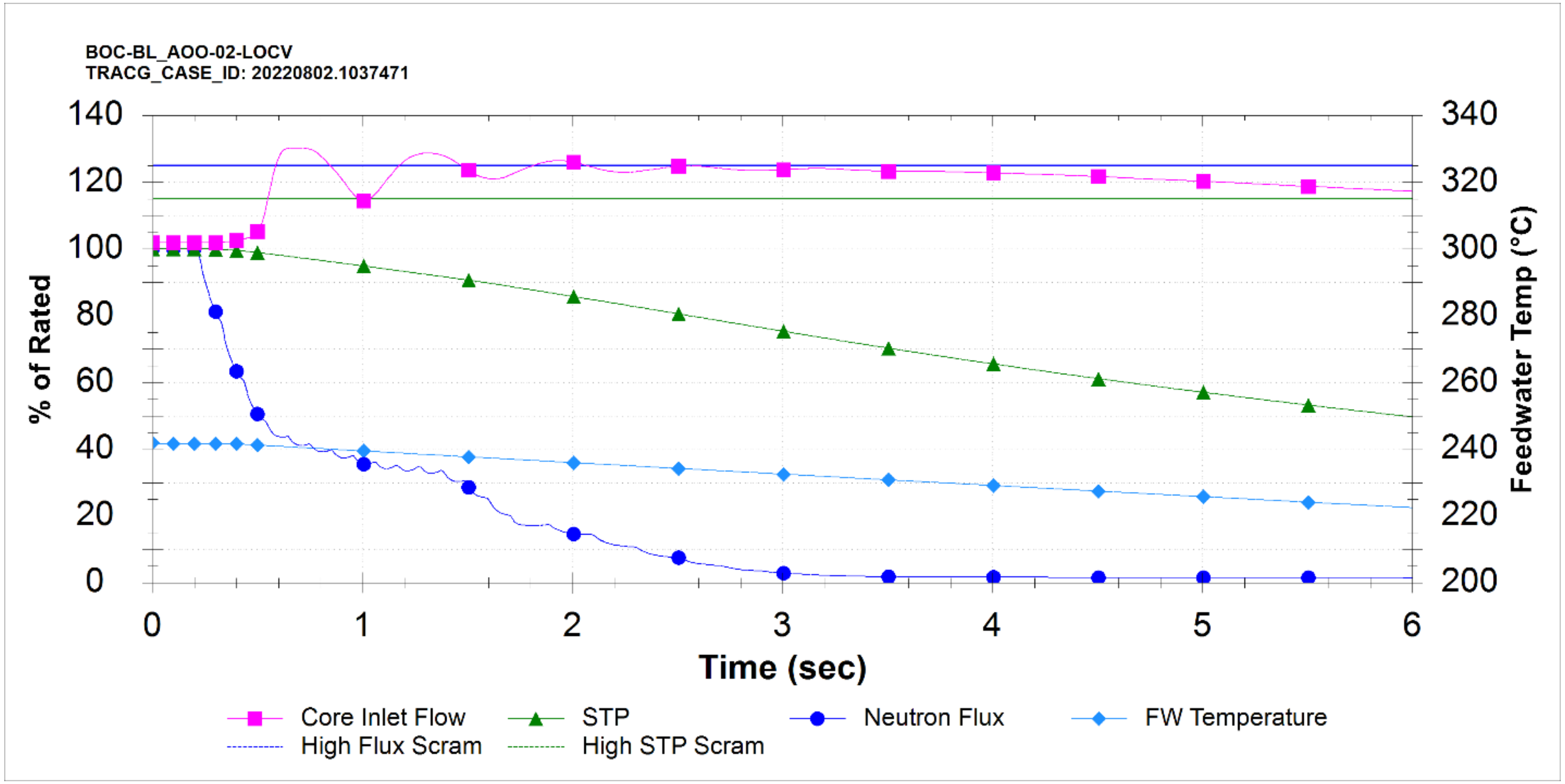

Figure 15.5-24: Loss of Condenser Vacuum (AOO)

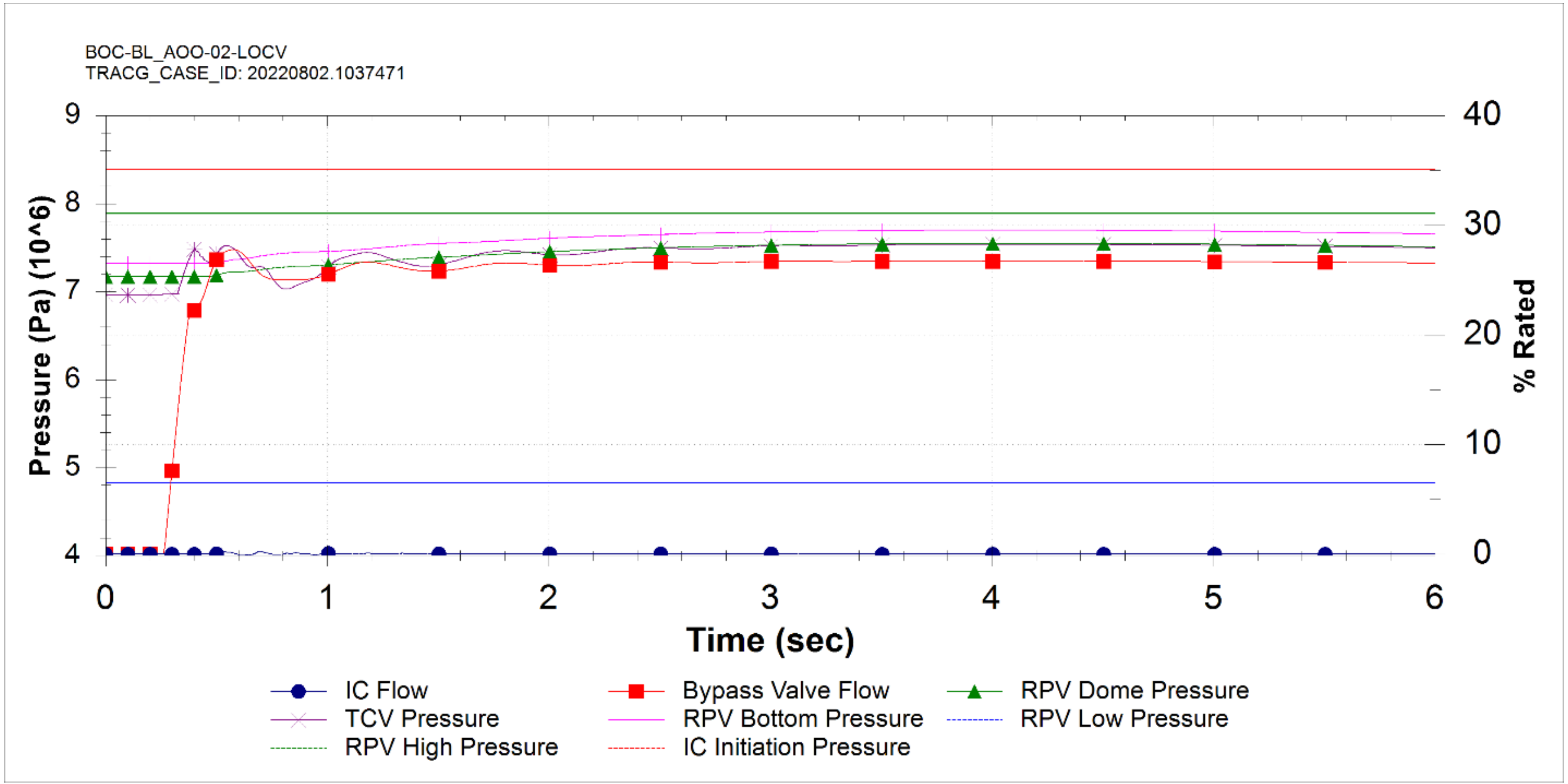

Figure 15.5-25: Loss of Condenser Vacuum (AOO)

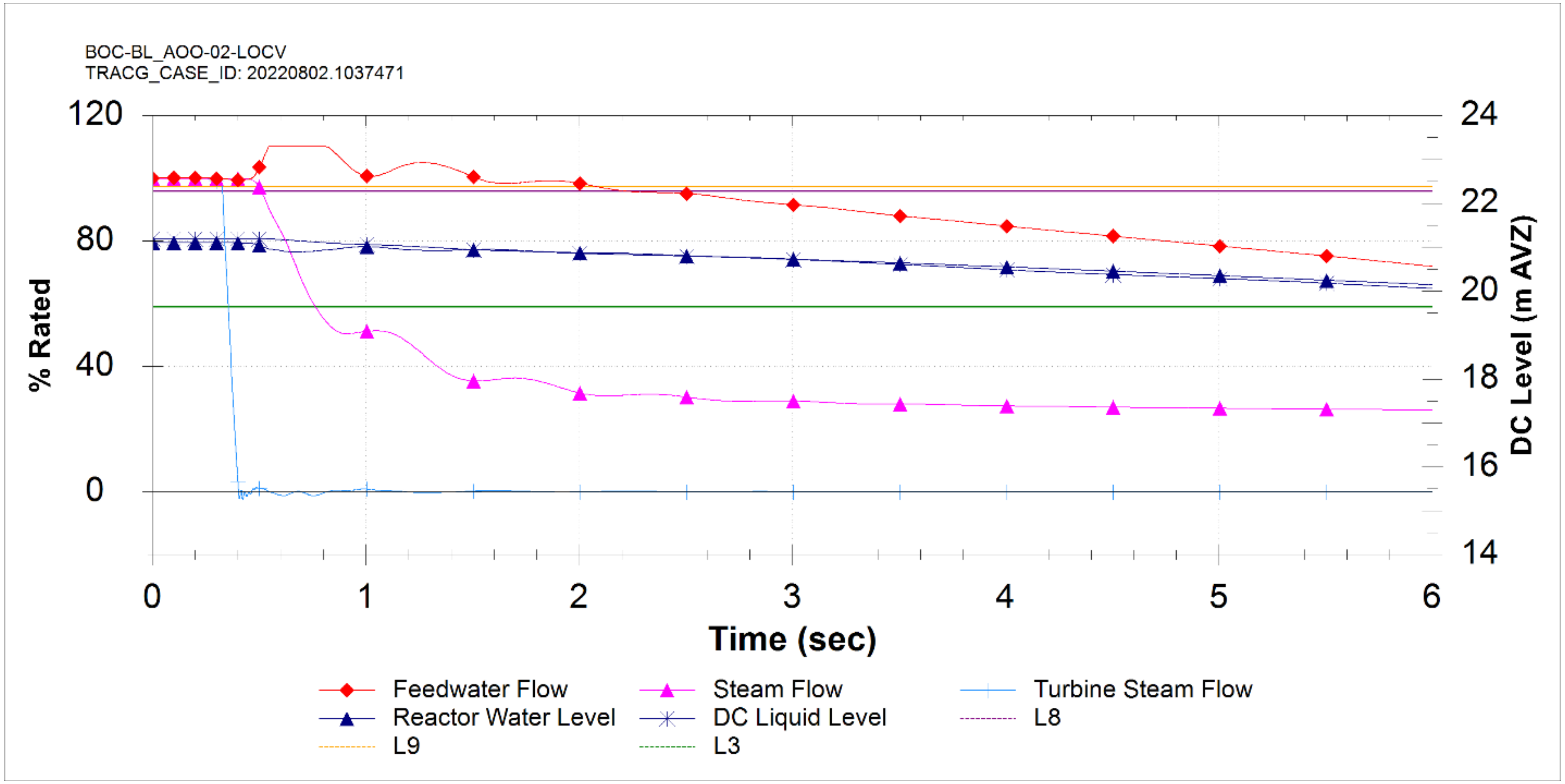

Figure 15.5-26: Loss of Condenser Vacuum (AOO)

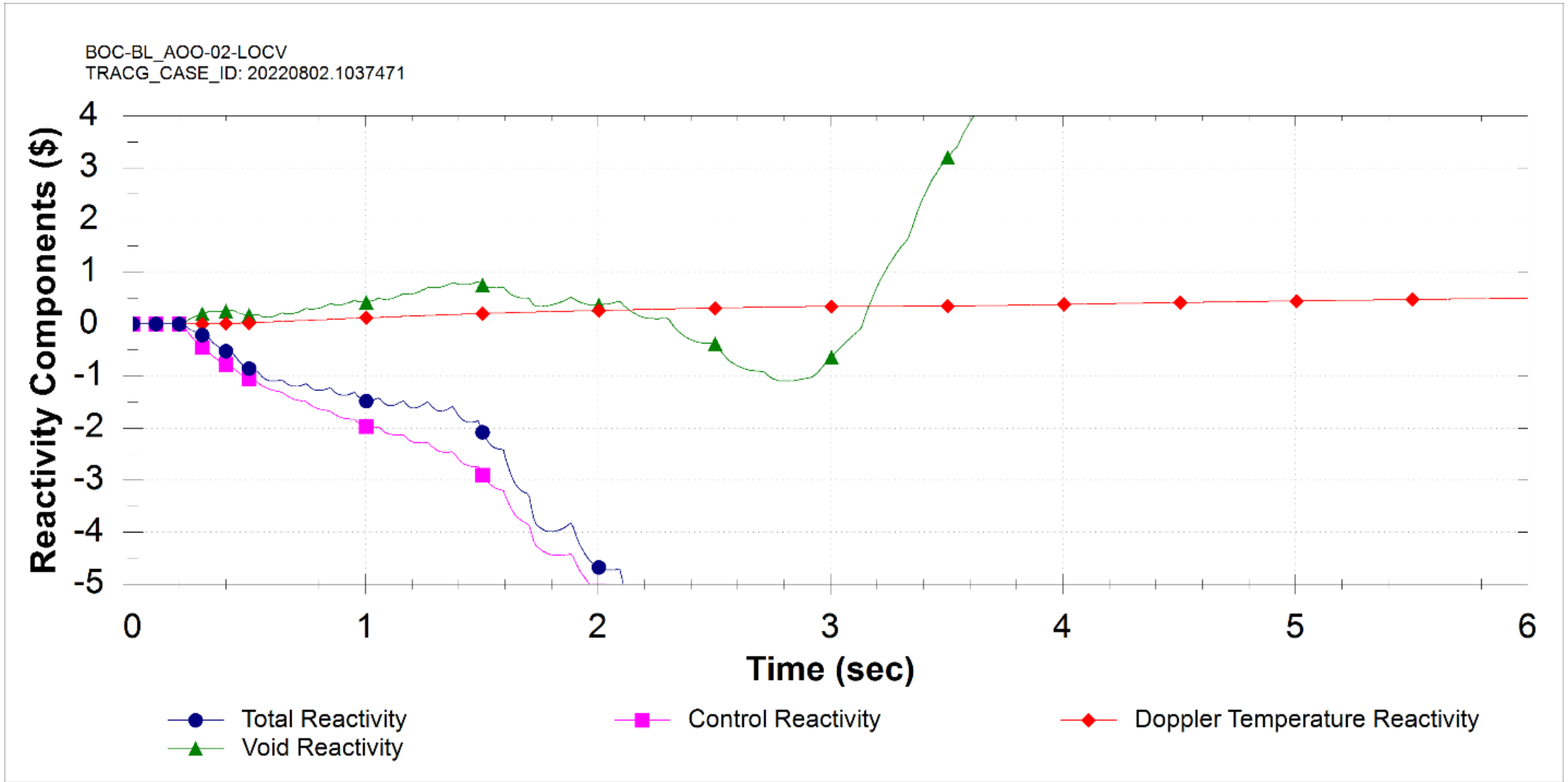

Figure 15.5-27: Loss of Condenser Vacuum (AOO)

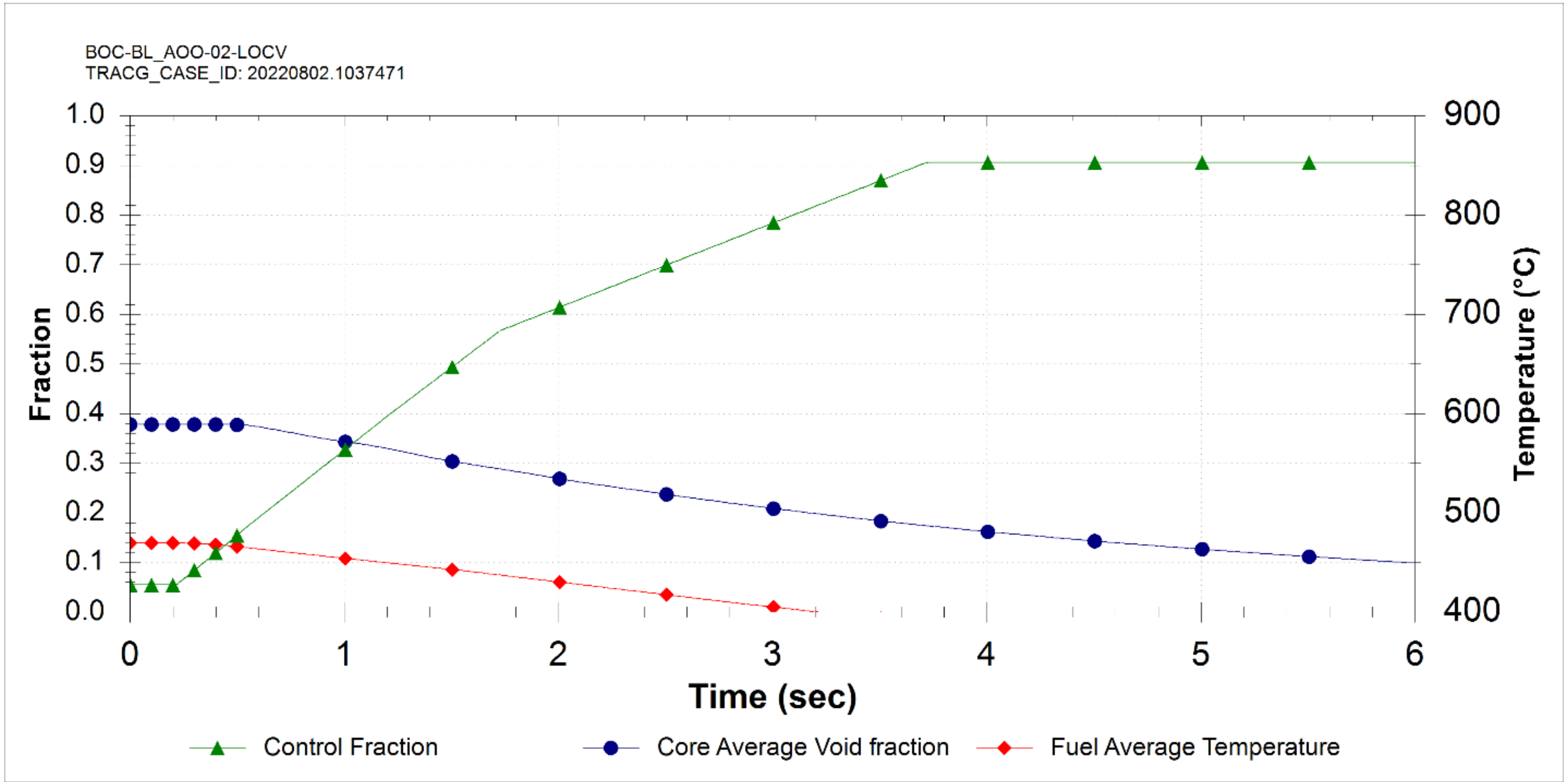

Figure 15.5-28: Loss of Condenser Vacuum (AOO)

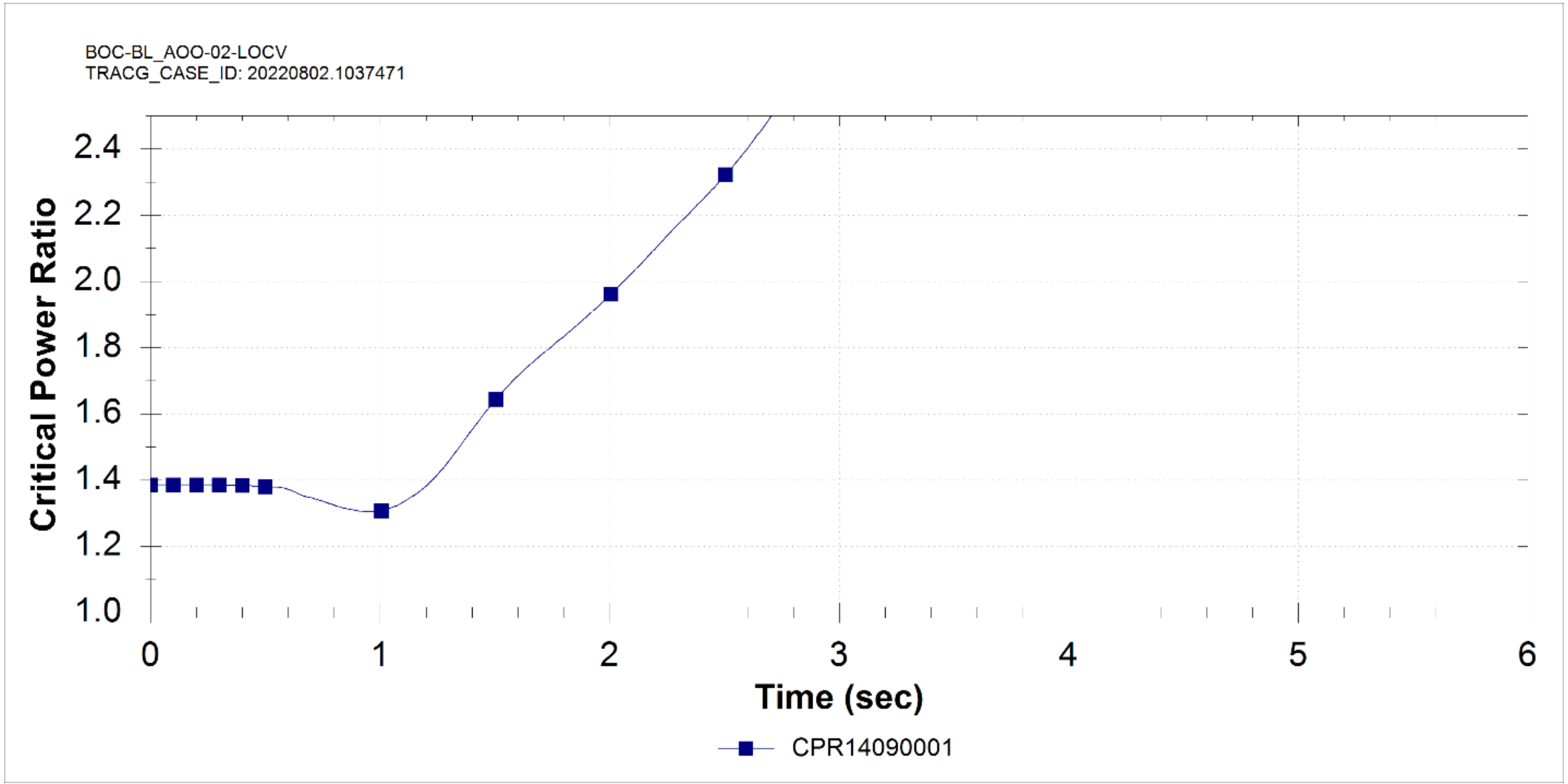

Figure 15.5-29: Loss of Condenser Vacuum (AOO)

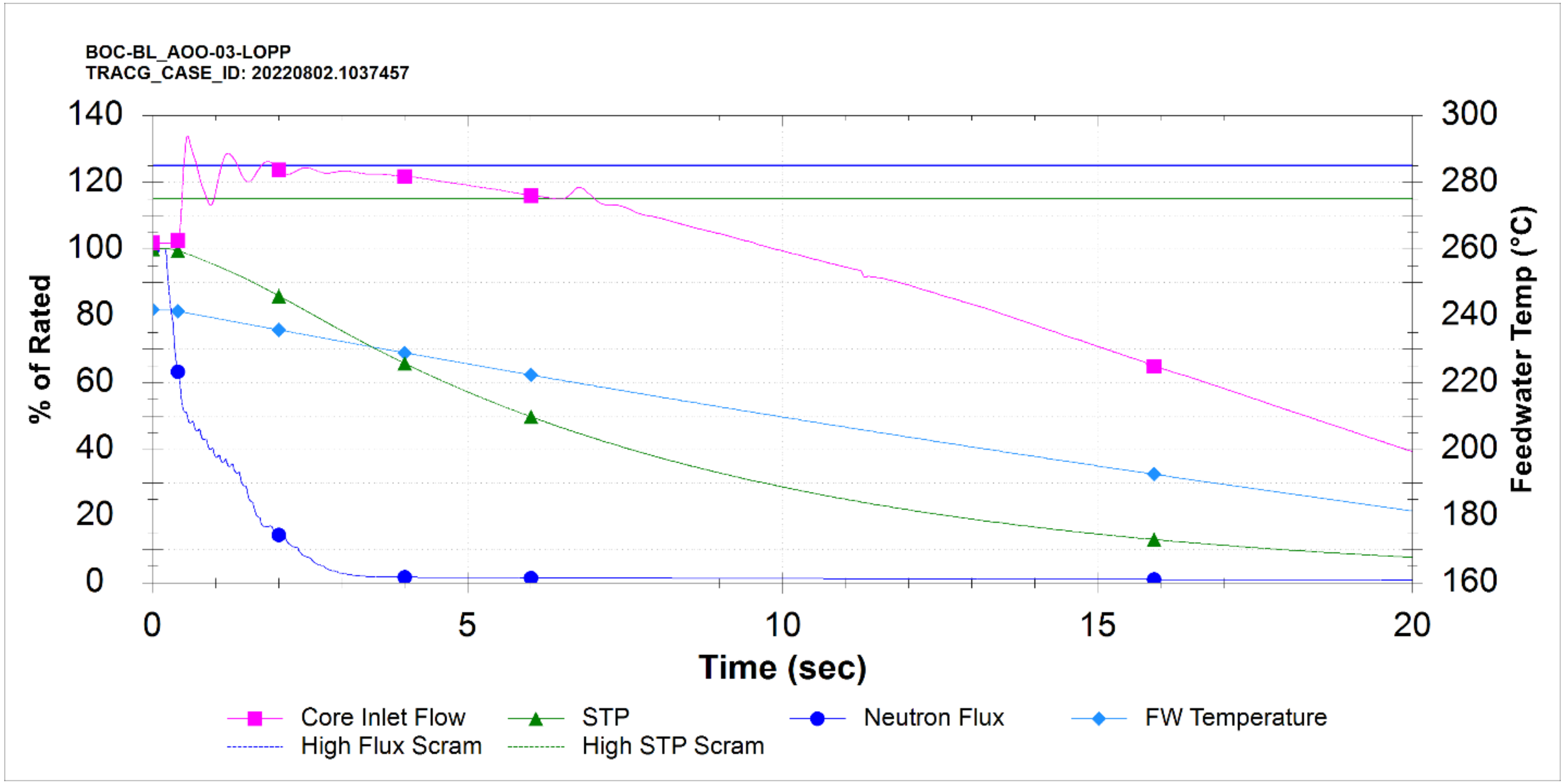

Figure 15.5-30: Loss of Preferred Power (AOO)

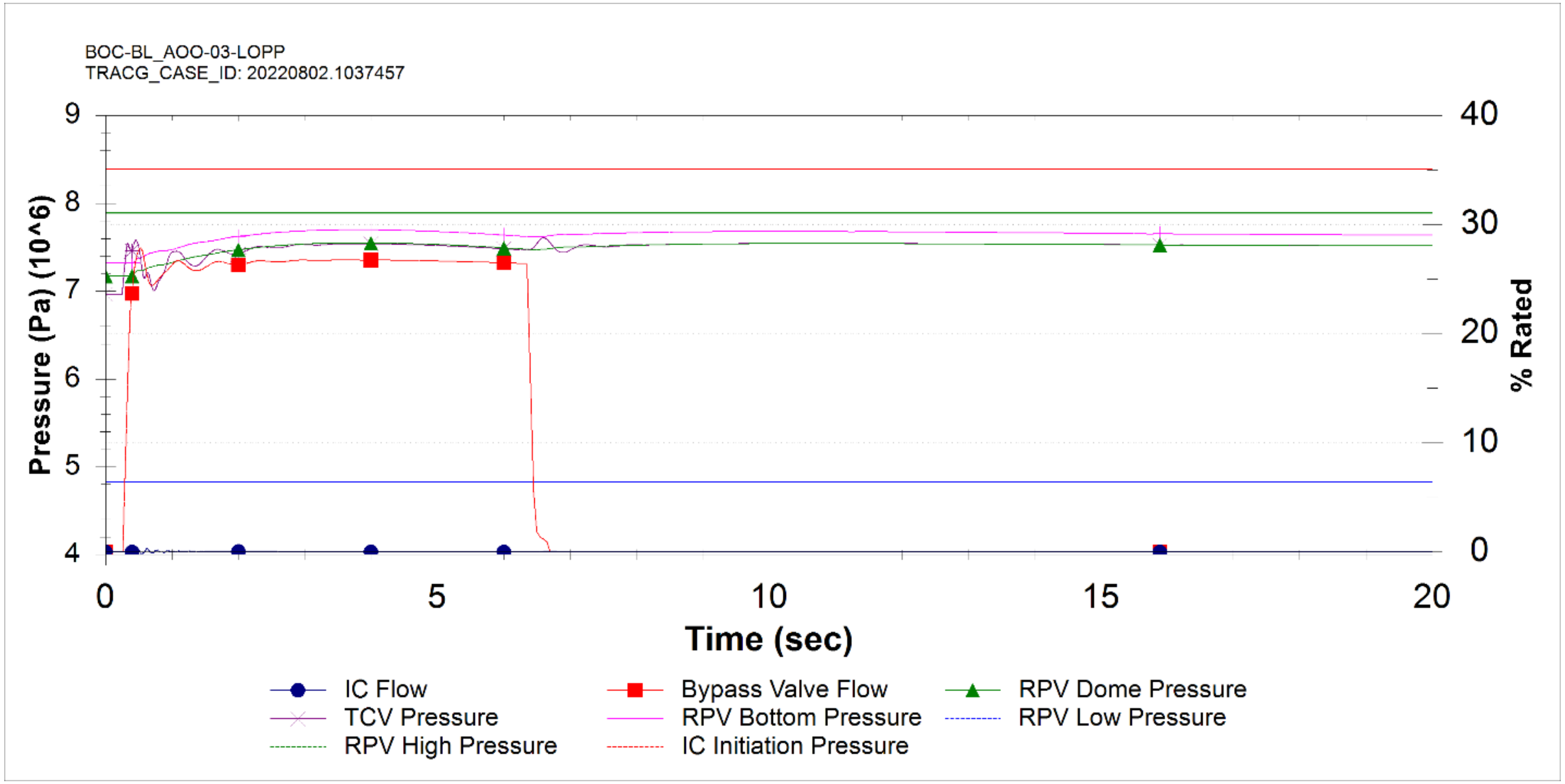

Figure 15.5-31: Loss of Preferred Power (AOO)

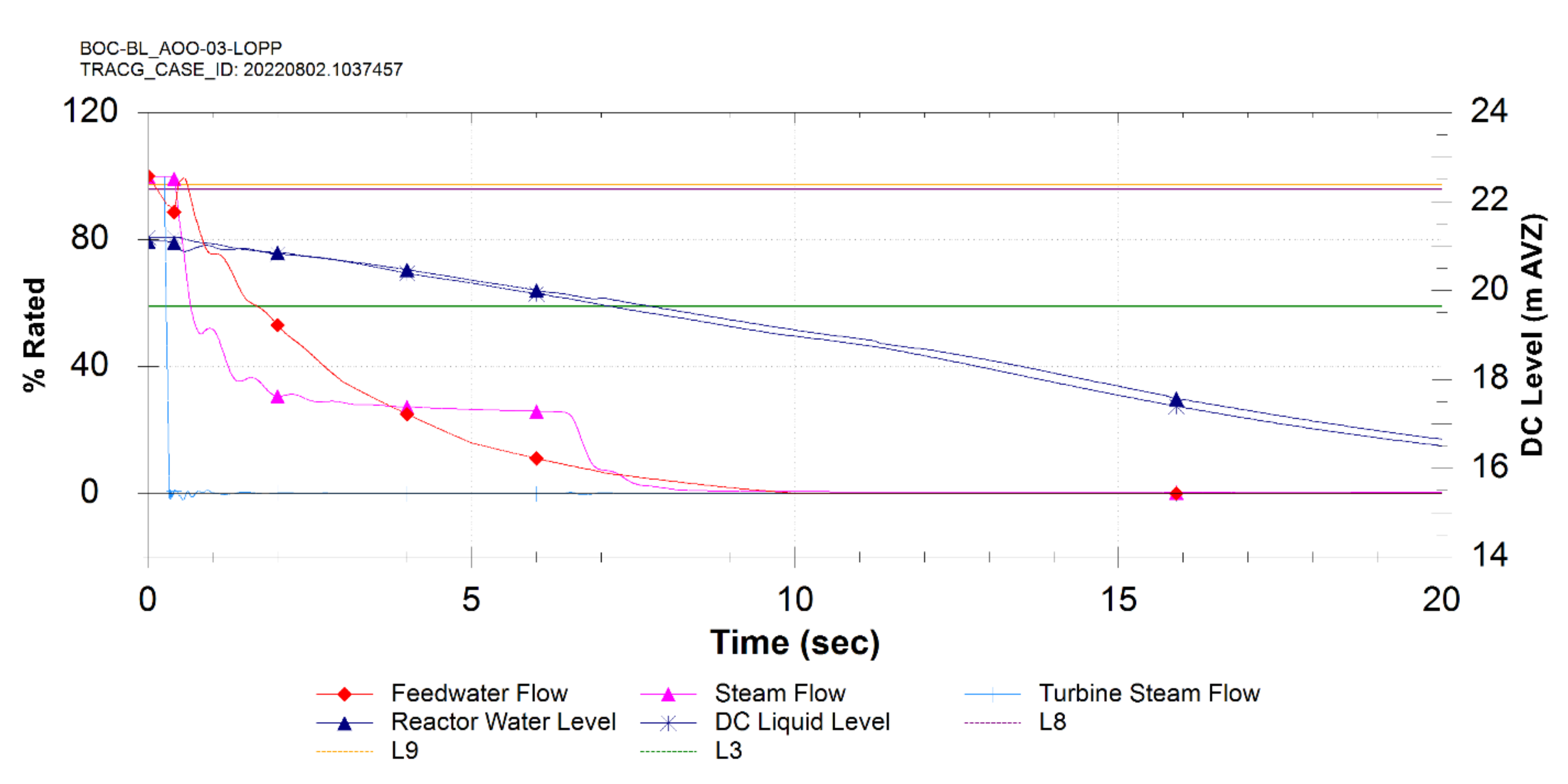

Figure 15.5-32: Loss of Preferred Power (AOO)

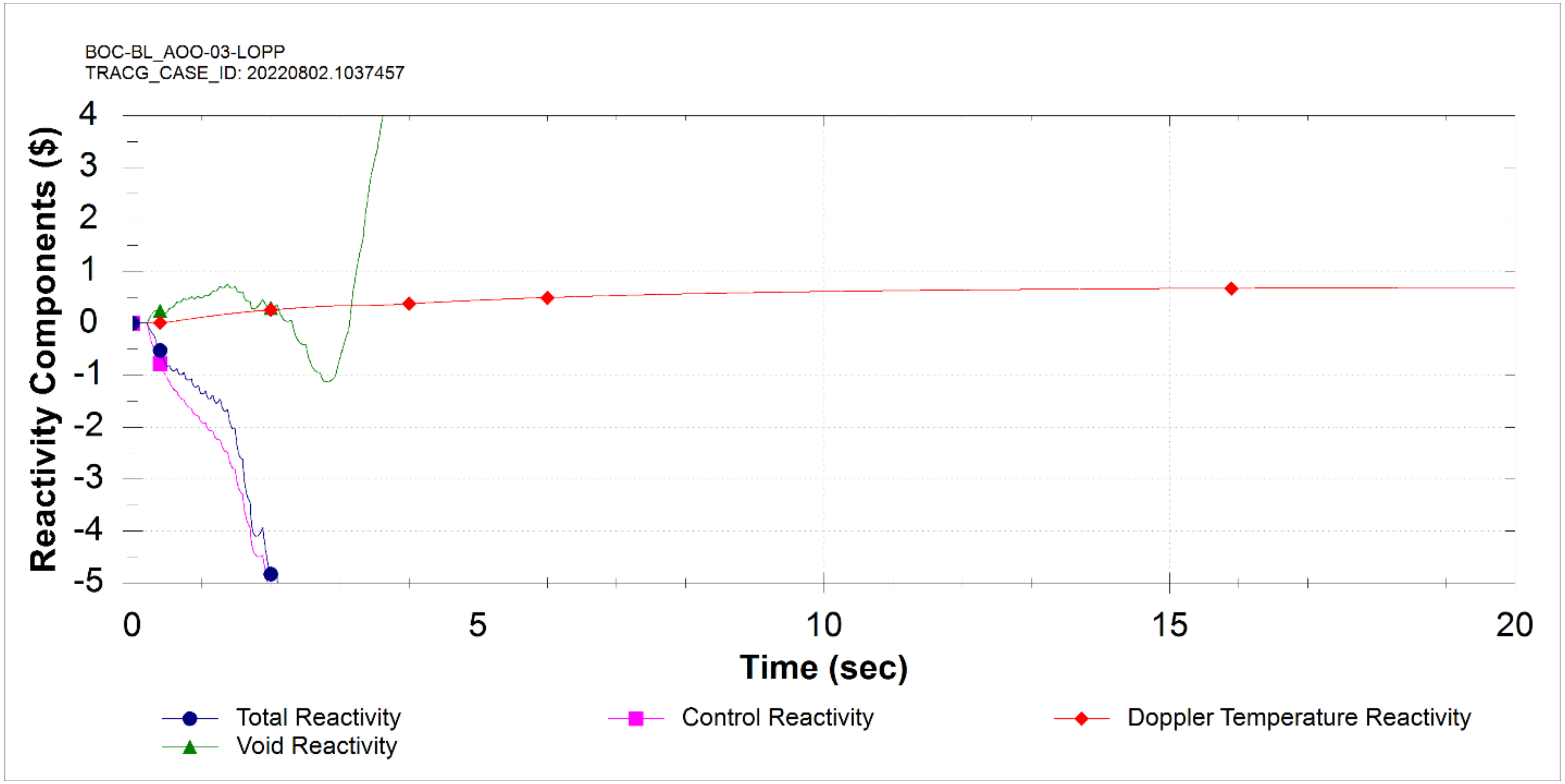

Figure 15.5-33: Loss of Preferred Power (AOO)

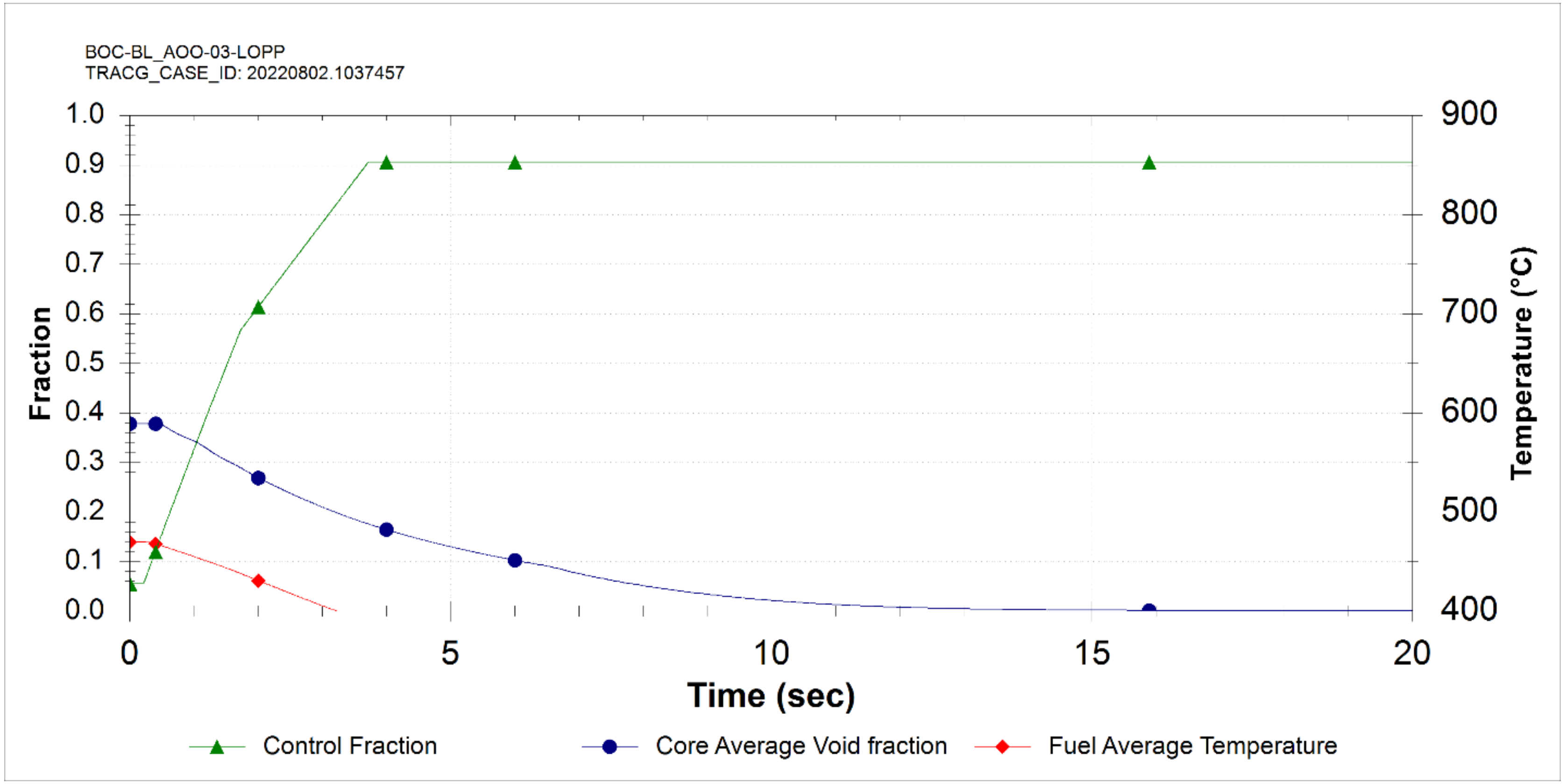

Figure 15.5-34: Loss of Preferred Power (AOO)

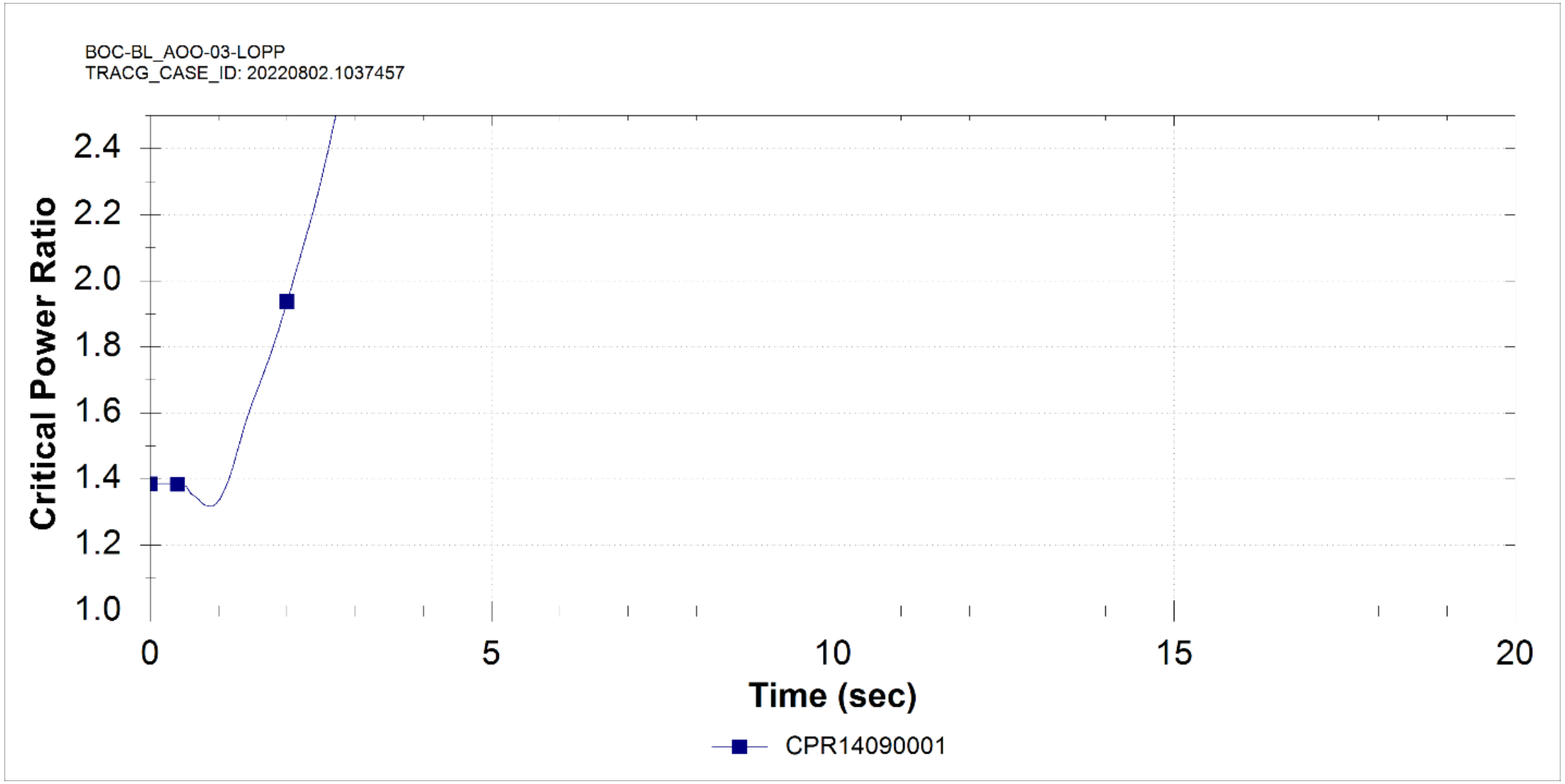

Figure 15.5-35: Loss of Preferred Power (AOO)

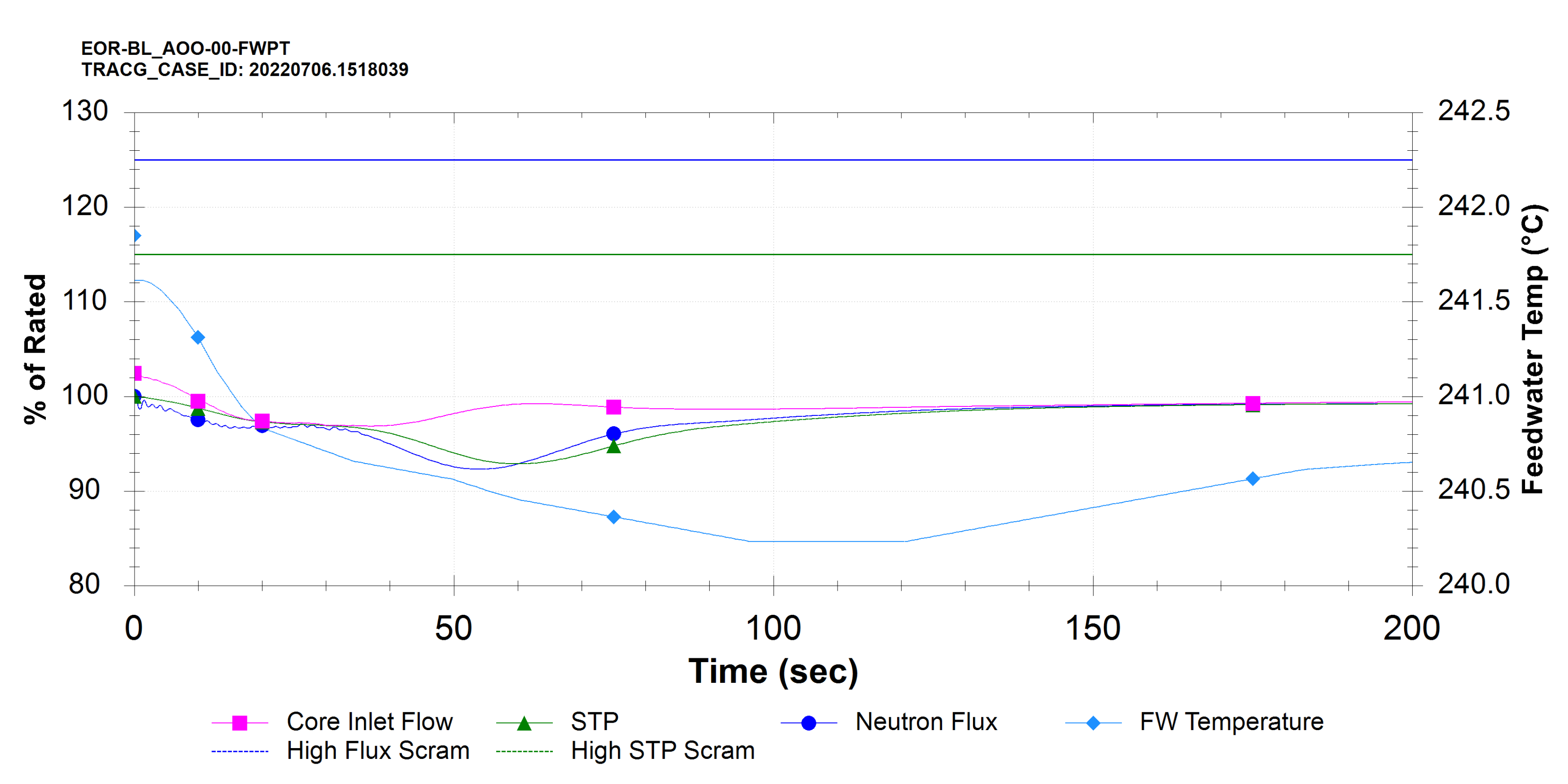

Figure 15.5-36: Feedwater Pump Trip – One Pump (AOO)

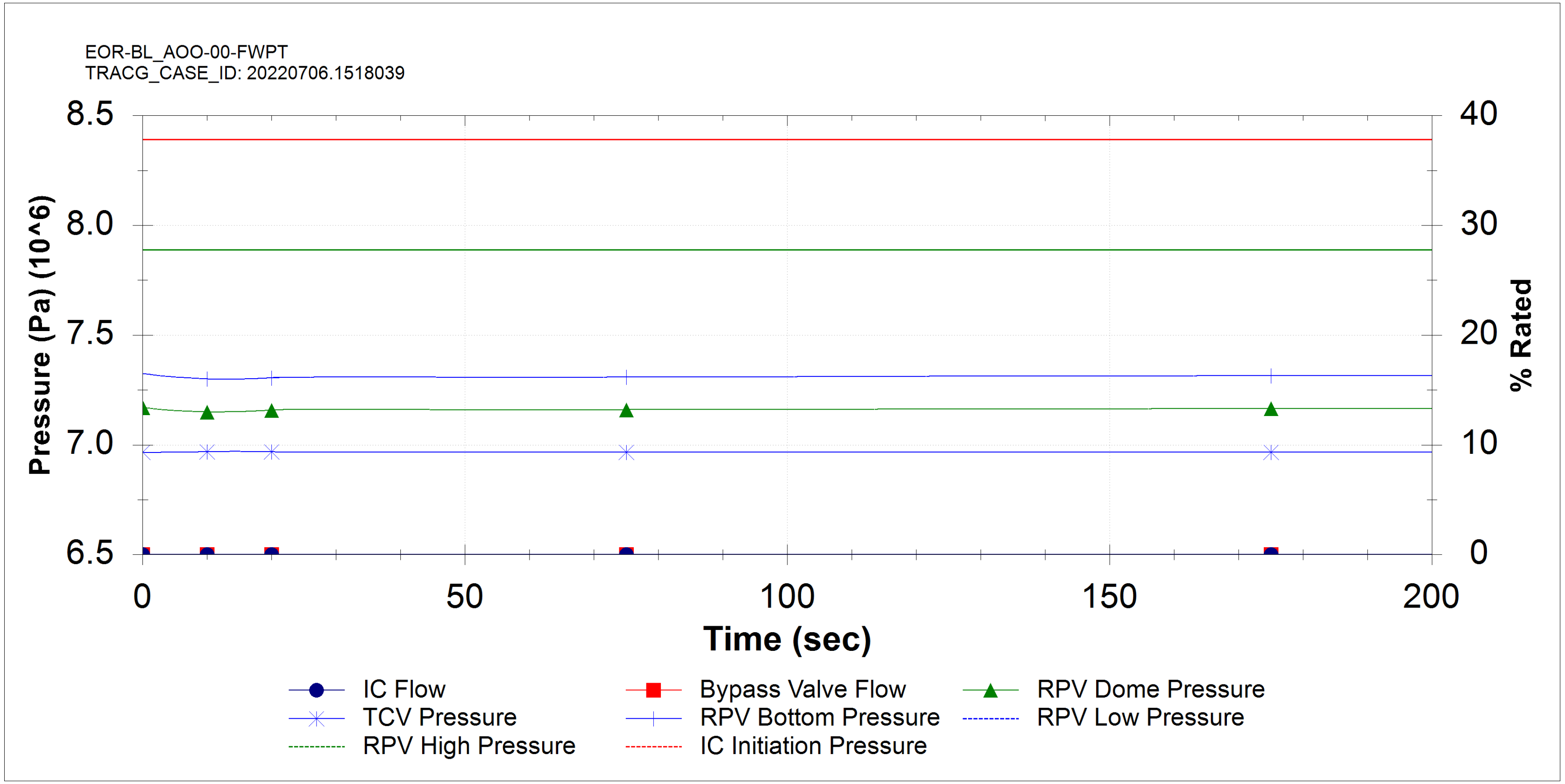

Figure 15.5-37: Feedwater Pump Trip – One Pump (AOO)

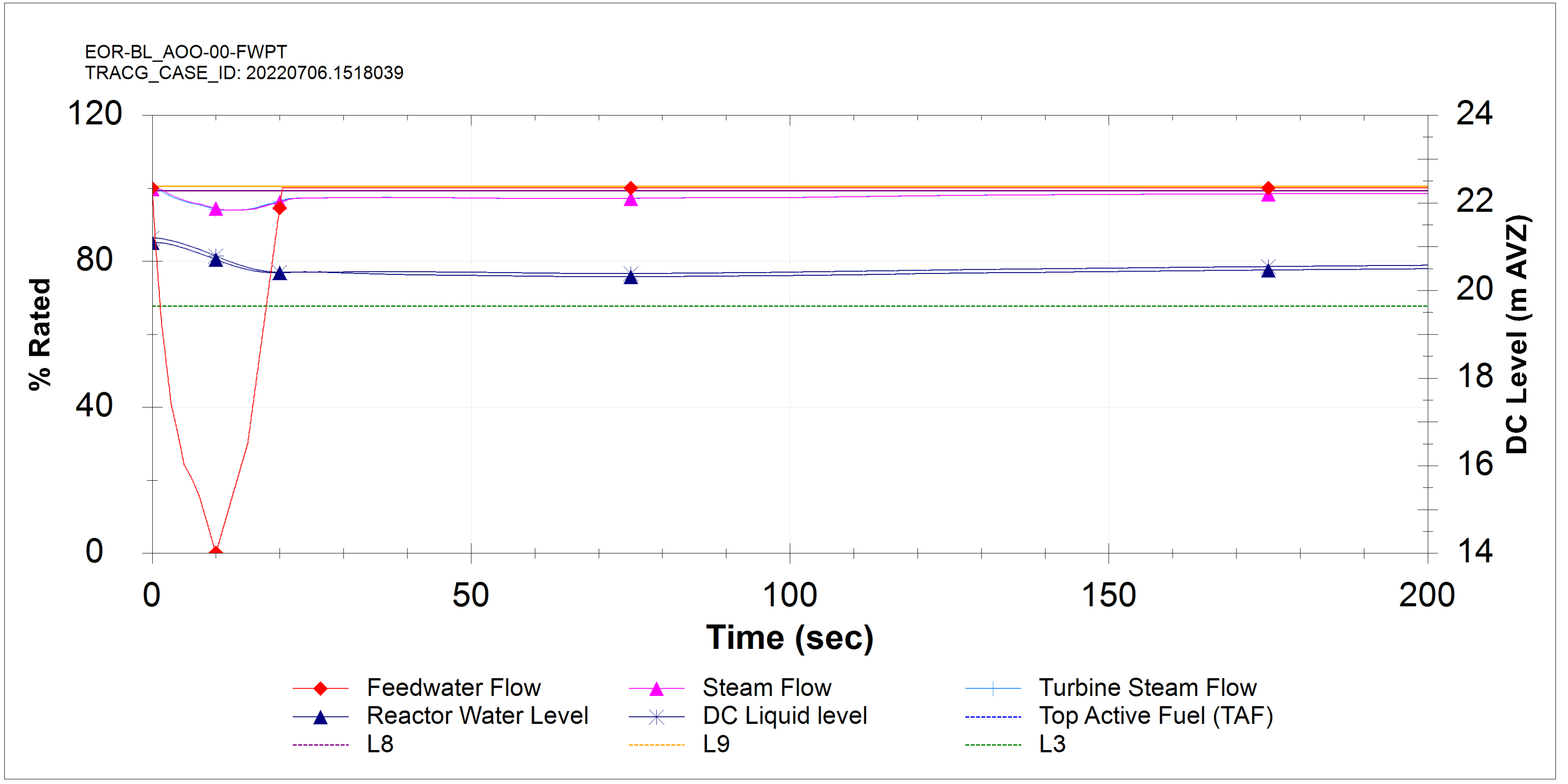

Figure 15.5-38: Feedwater Pump Trip – One Pump (AOO)

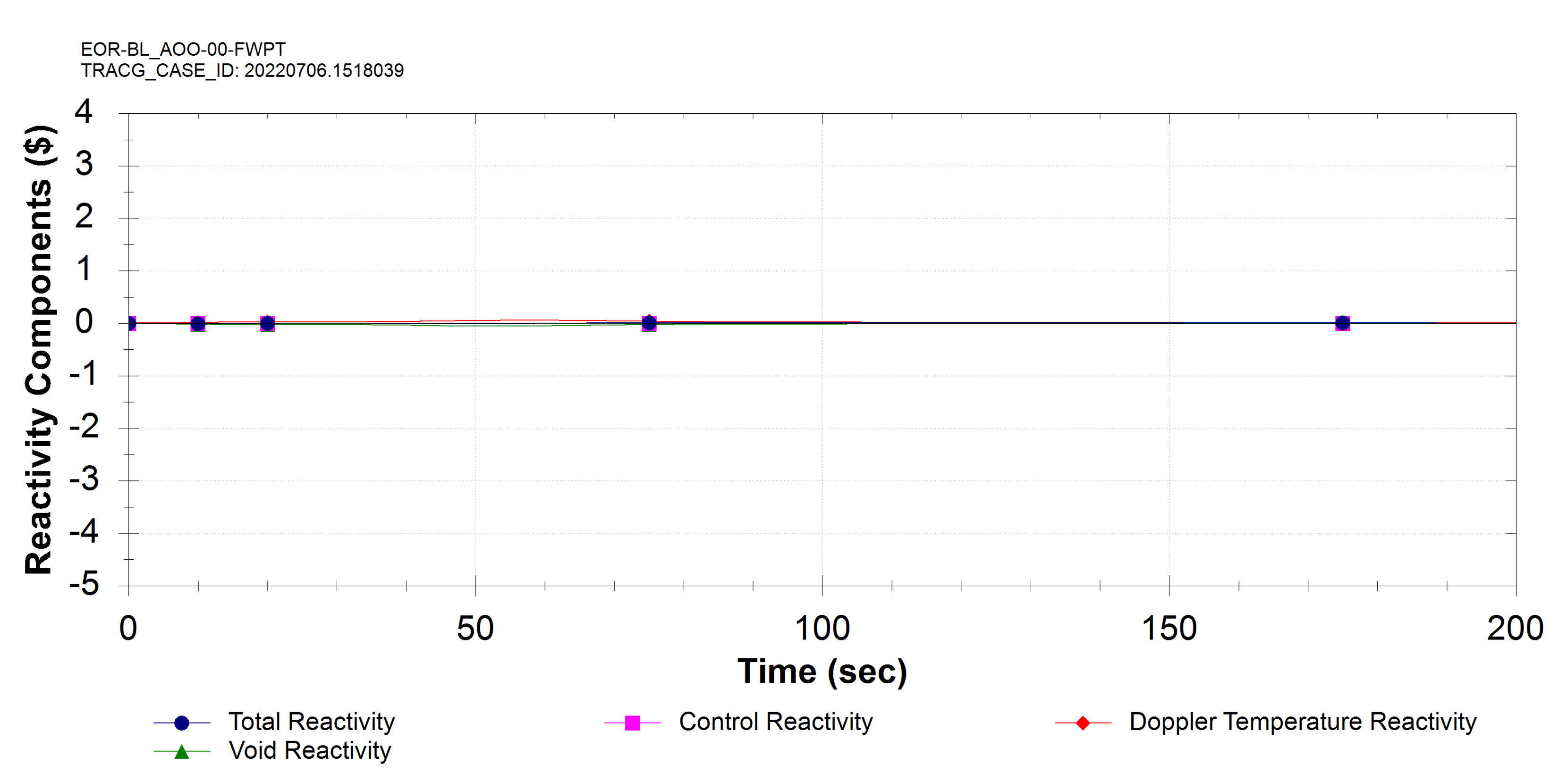

Figure 15.5-39: Feedwater Pump Trip – One Pump (AOO)

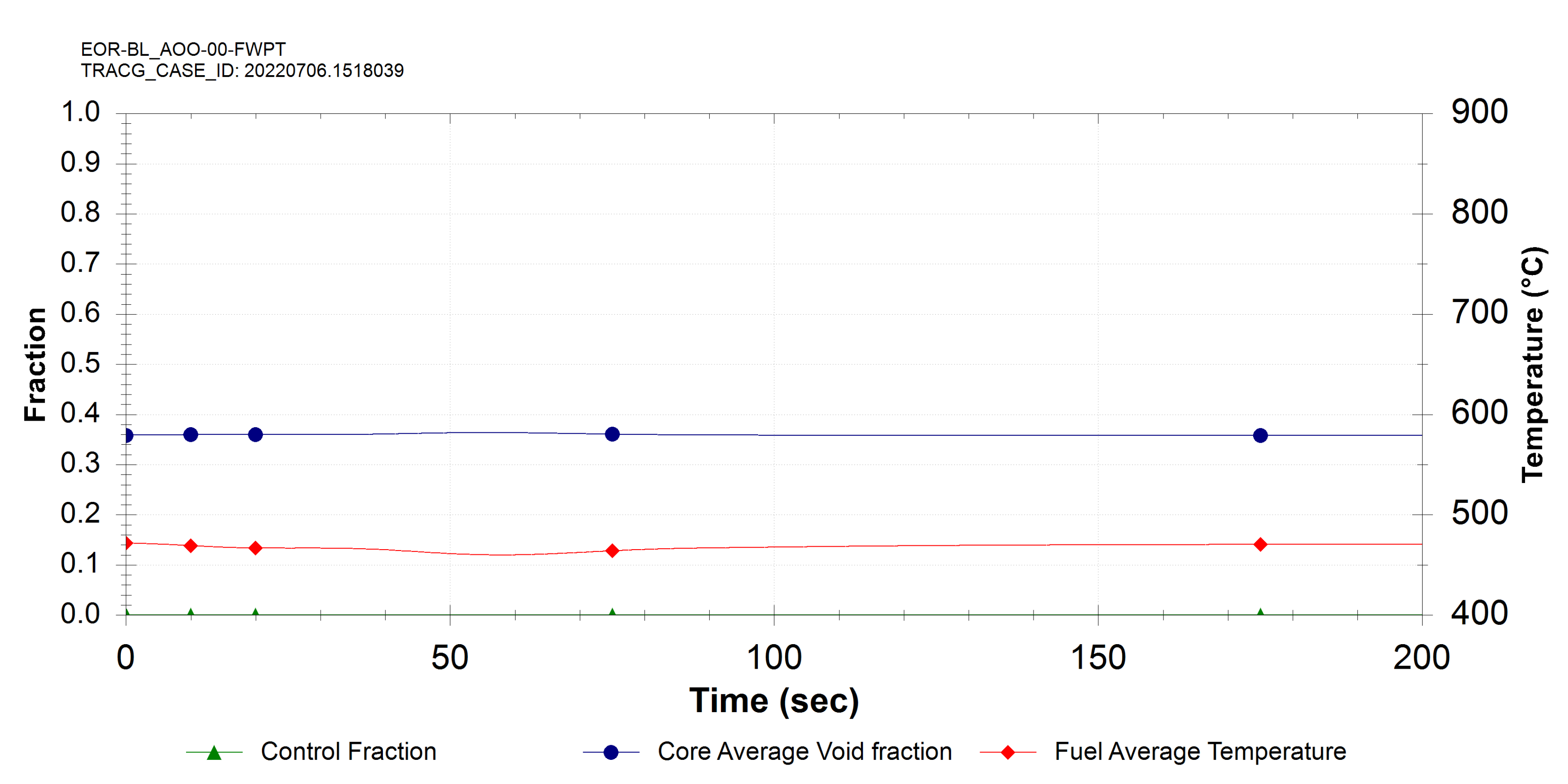

Figure 15.5-40: Feedwater Pump Trip – One Pump (AOO)

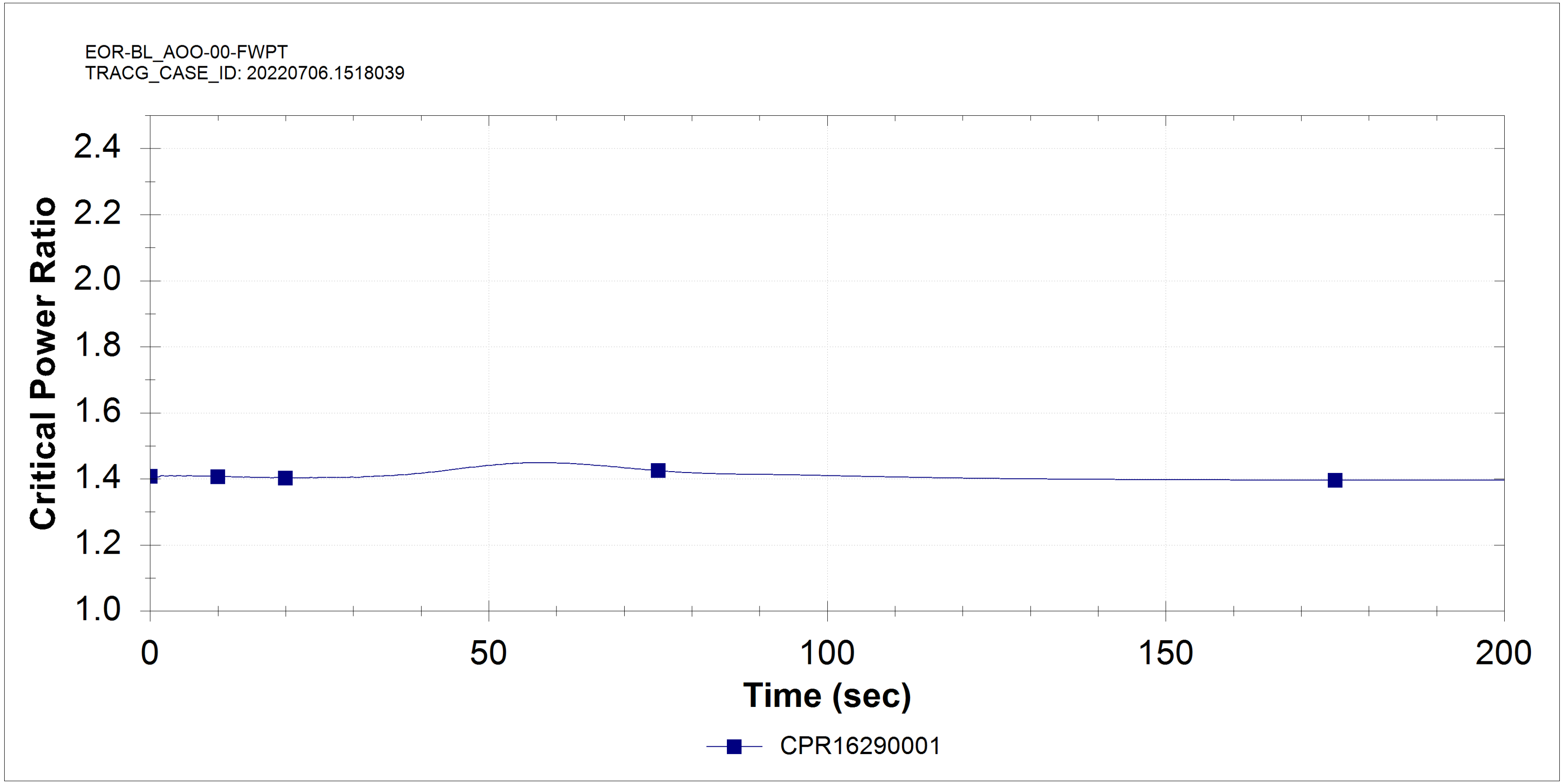

Figure 15.5-41: Feedwater Pump Trip – One Pump (AOO)

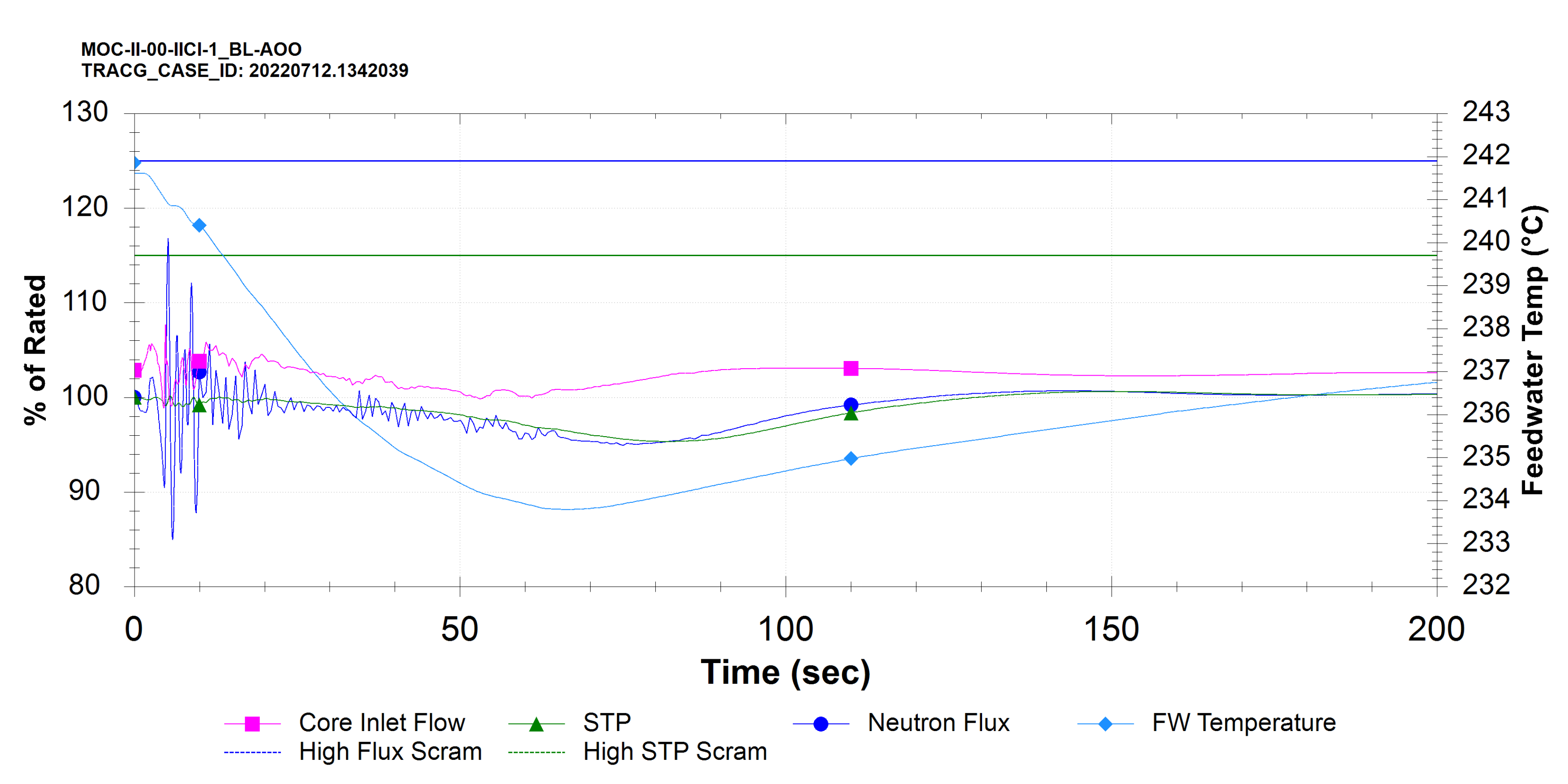

Figure 15.5-42: Inadvertent Isolation Condenser Initiation – One Train (AOO)

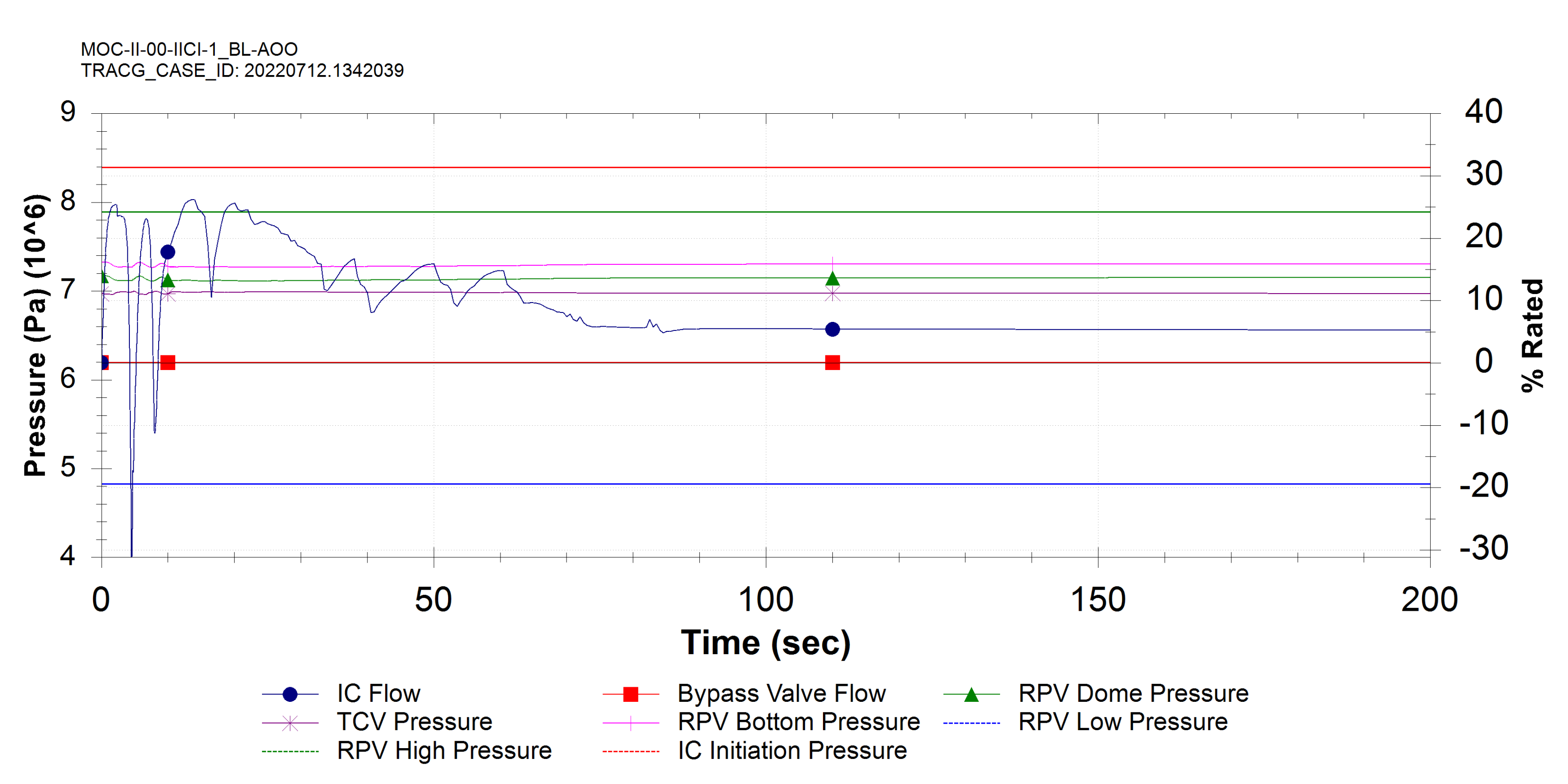

Figure 15.5-43: Inadvertent Isolation Condenser Initiation – One Train (AOO)

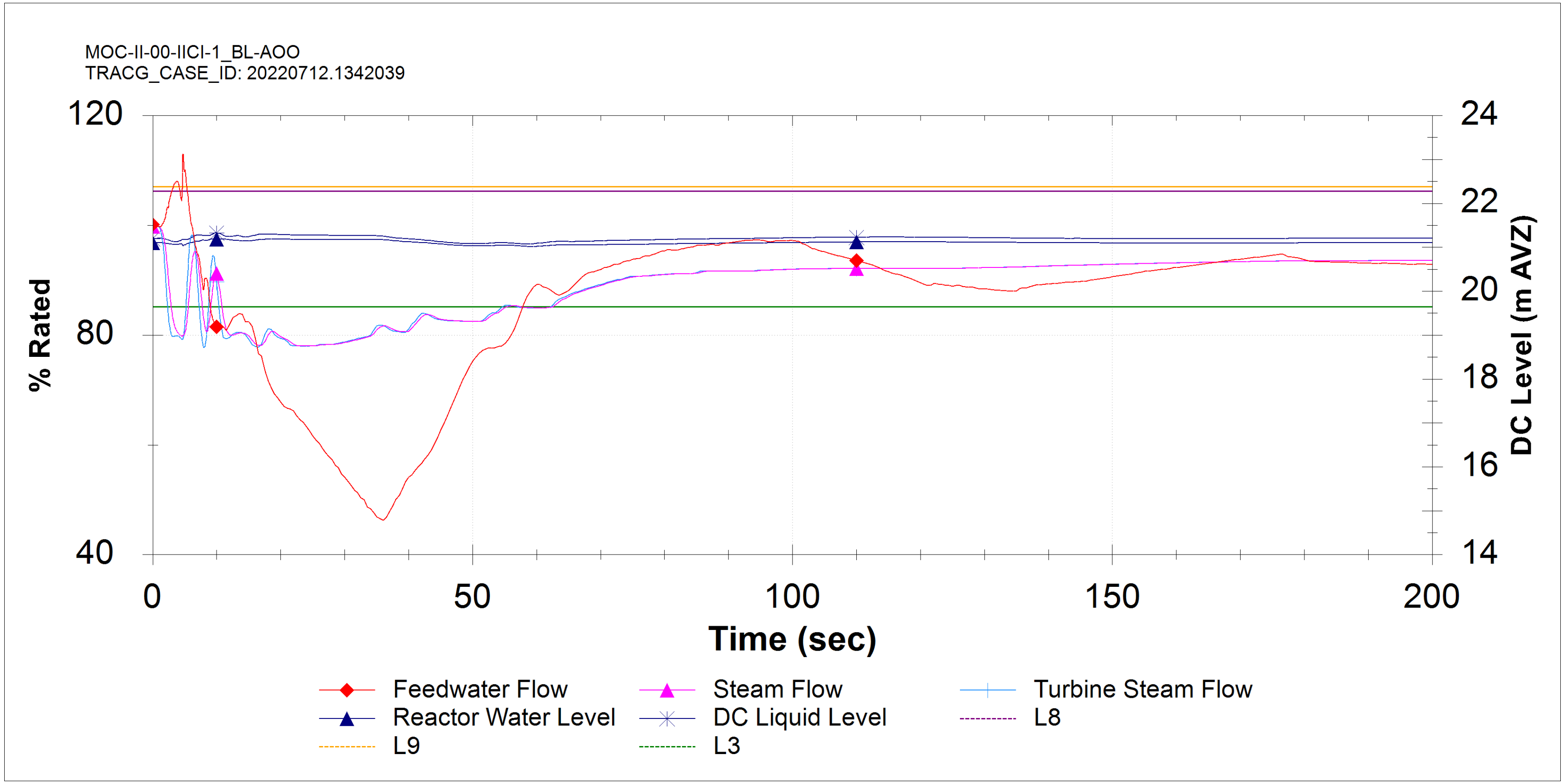

Figure 15.5-44: Inadvertent Isolation Condenser Initiation – One Train (AOO)

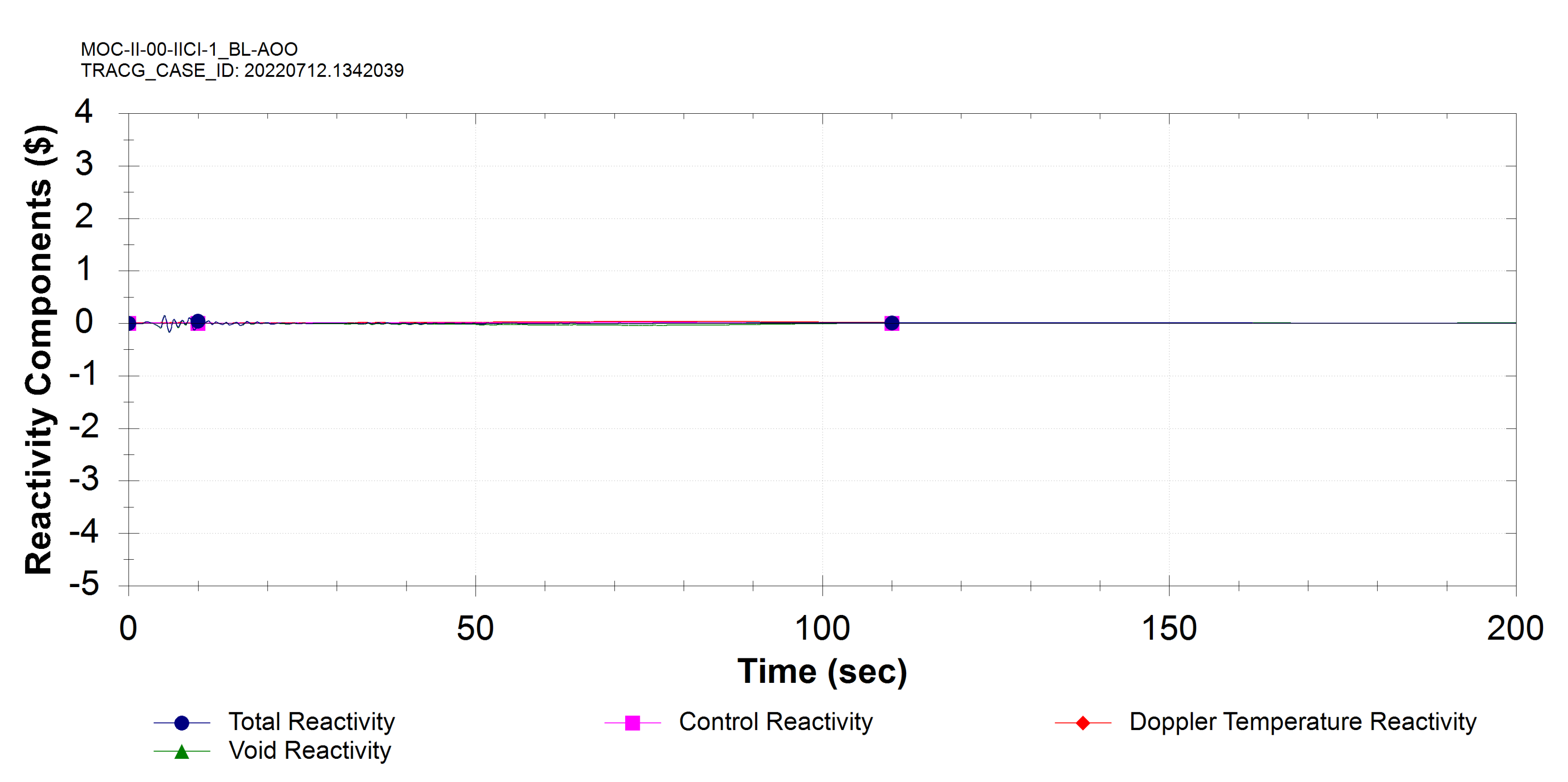

Figure 15.5-45: Inadvertent Isolation Condenser Initiation – One Train (AOO)

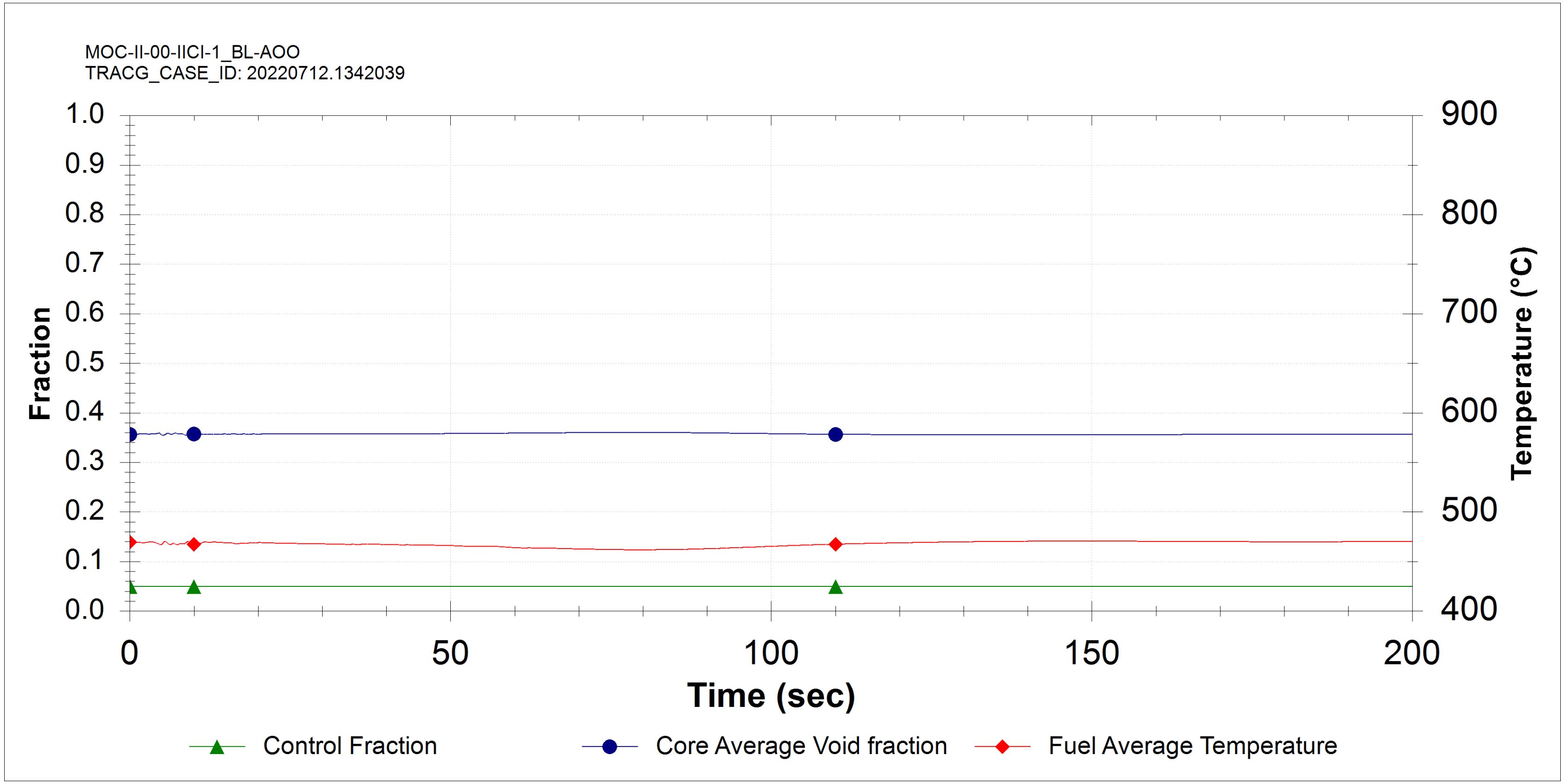

Figure 15.5-46: Inadvertent Isolation Condenser Initiation – One Train (AOO)

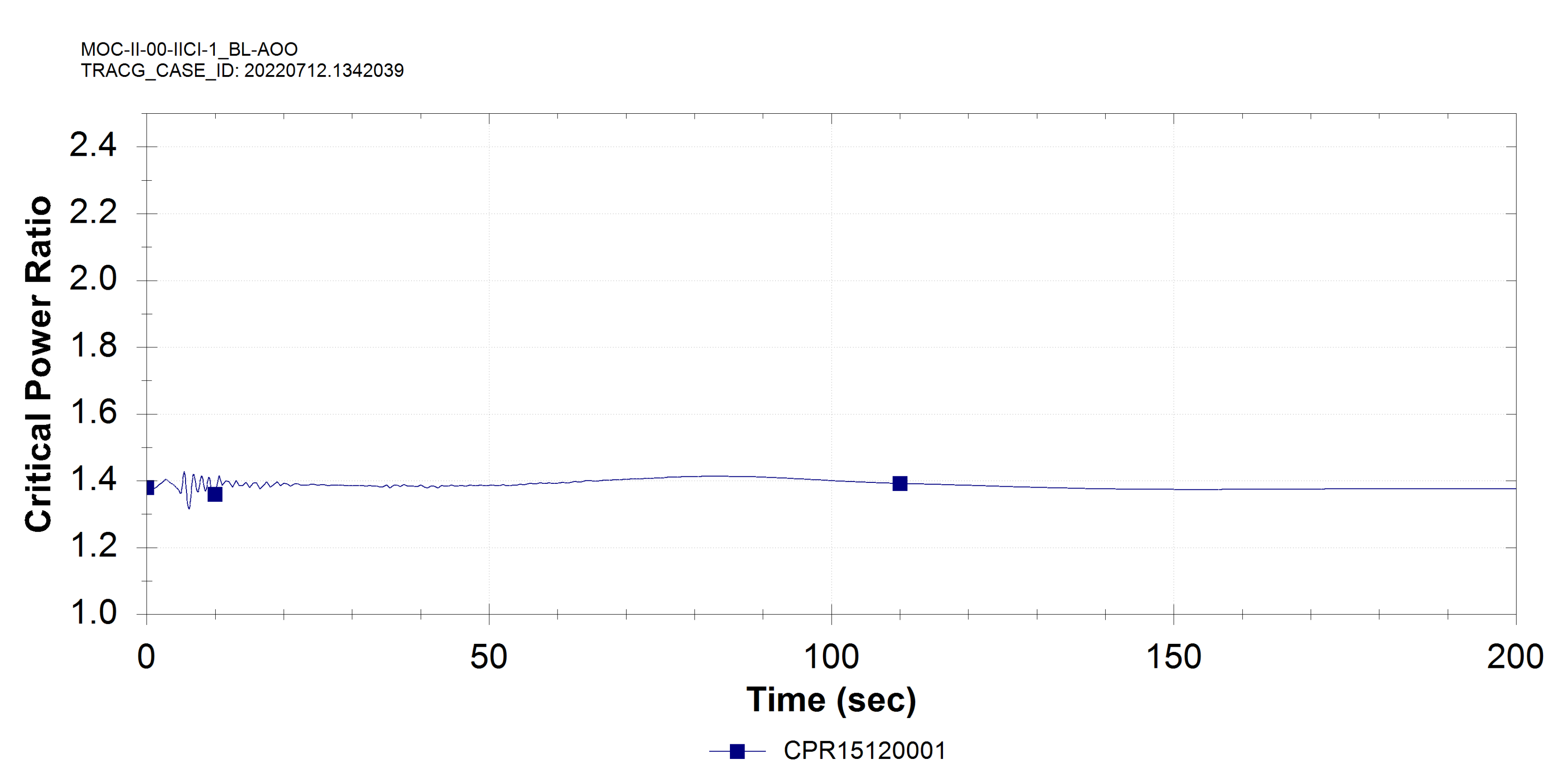

Figure 15.5-47: Inadvertent Isolation Condenser Initiation – One Train (AOO)

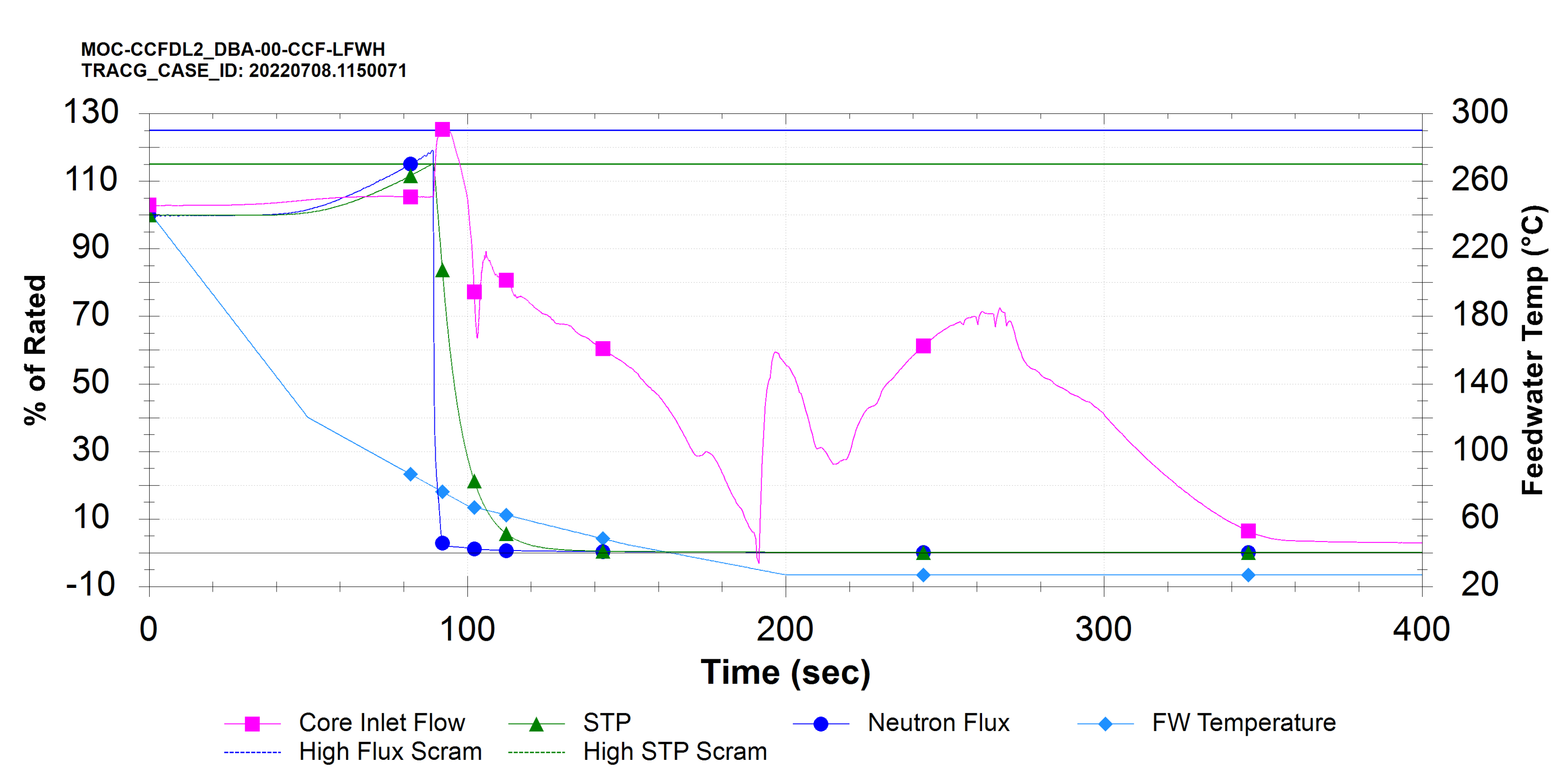

Figure 15.5-48: Loss of Feedwater Heating (DBA)

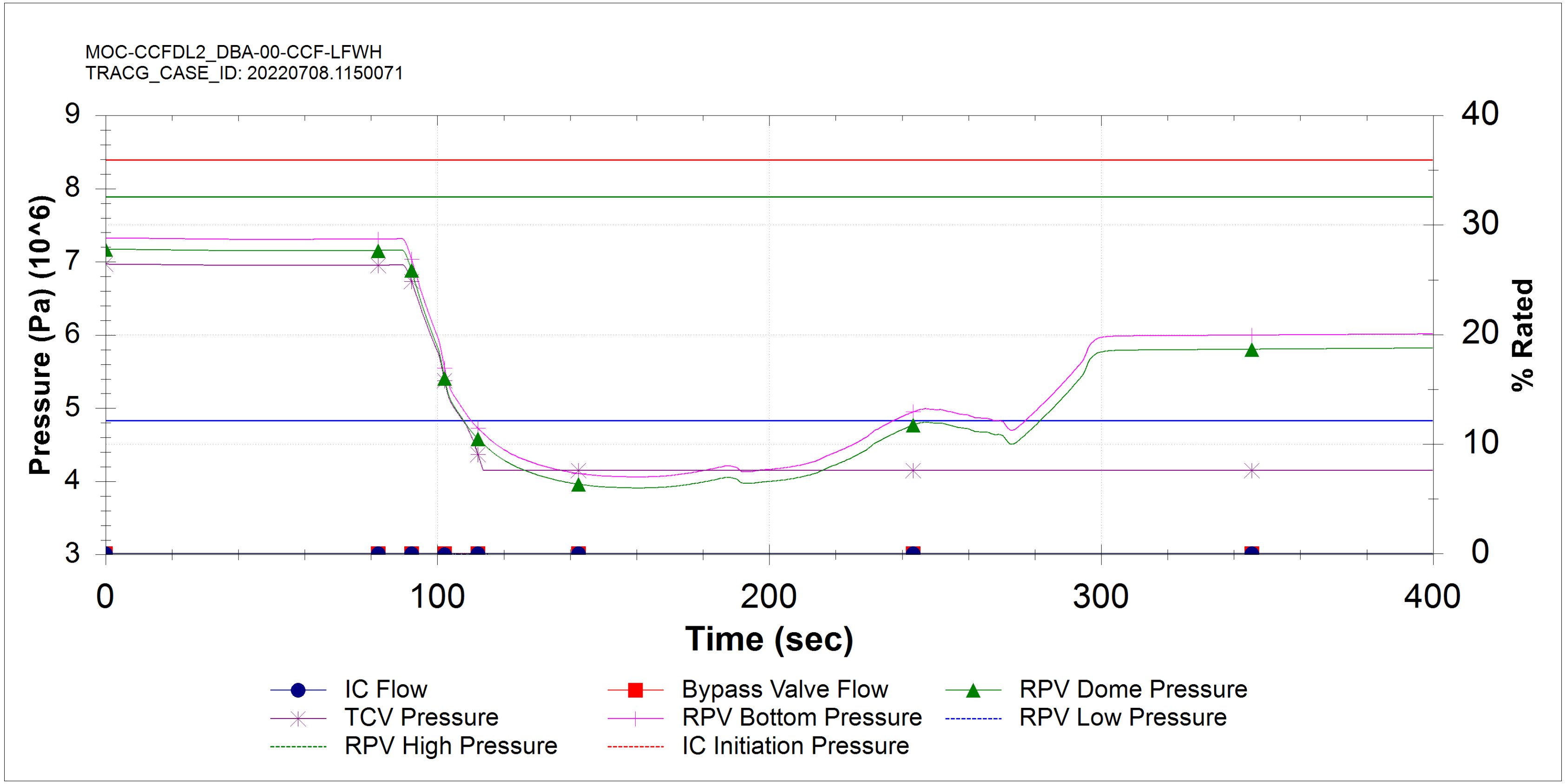

Figure 15.5-49: Loss of Feedwater Heating (DBA)

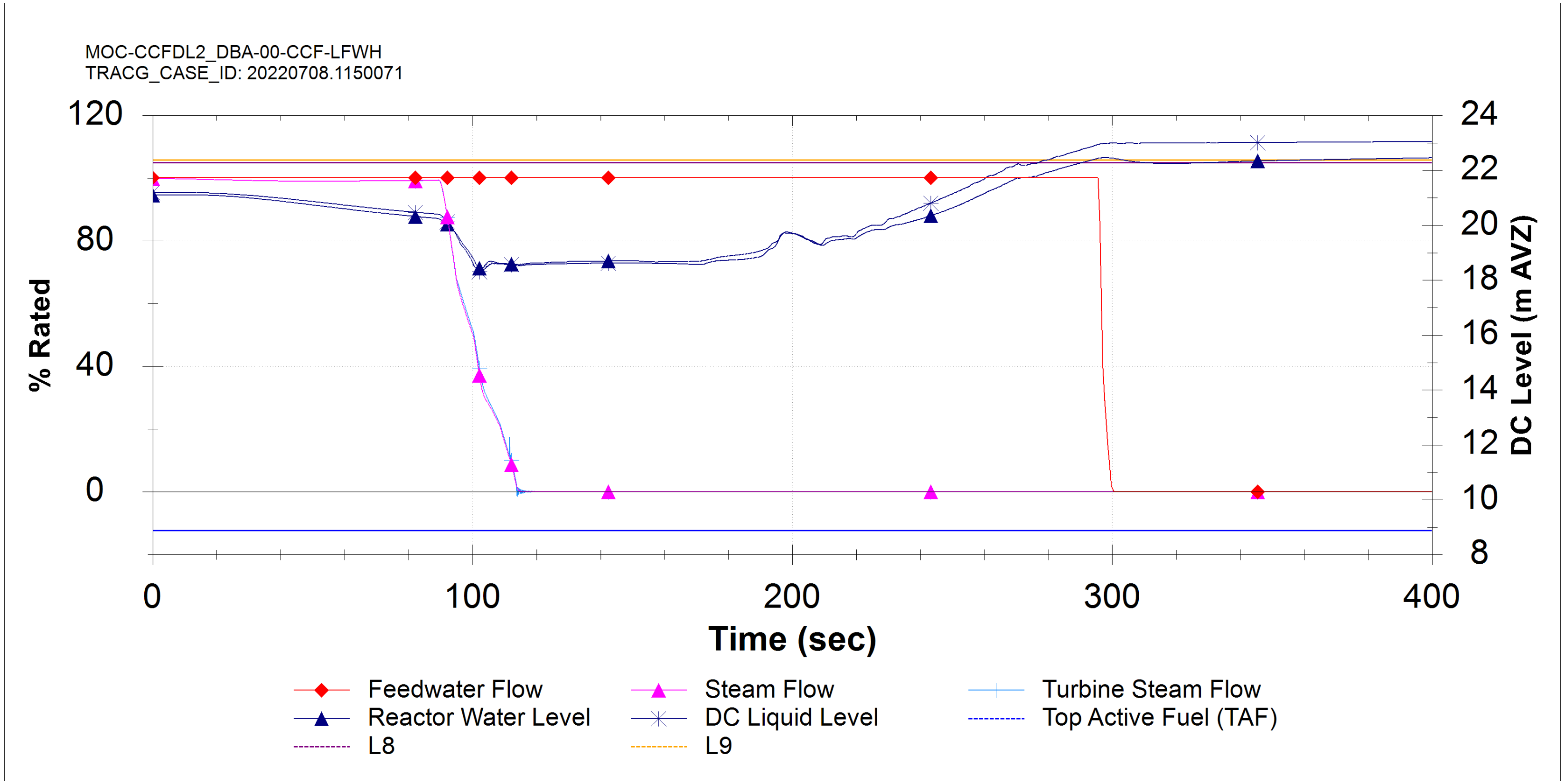

Figure 15.5-50: Loss of Feedwater Heating (DBA)

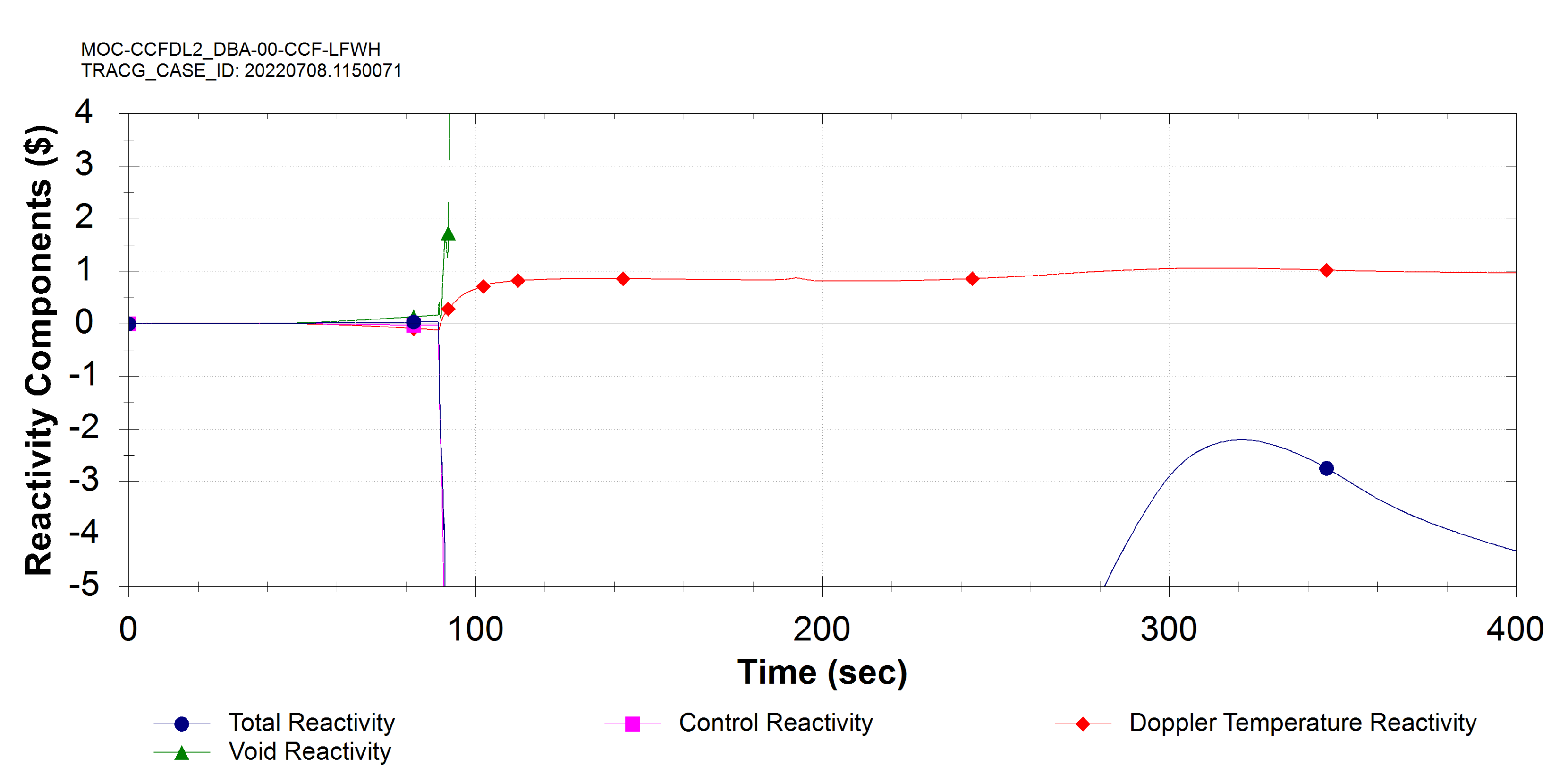

Figure 15.5-51: Loss of Feedwater Heating (DBA)

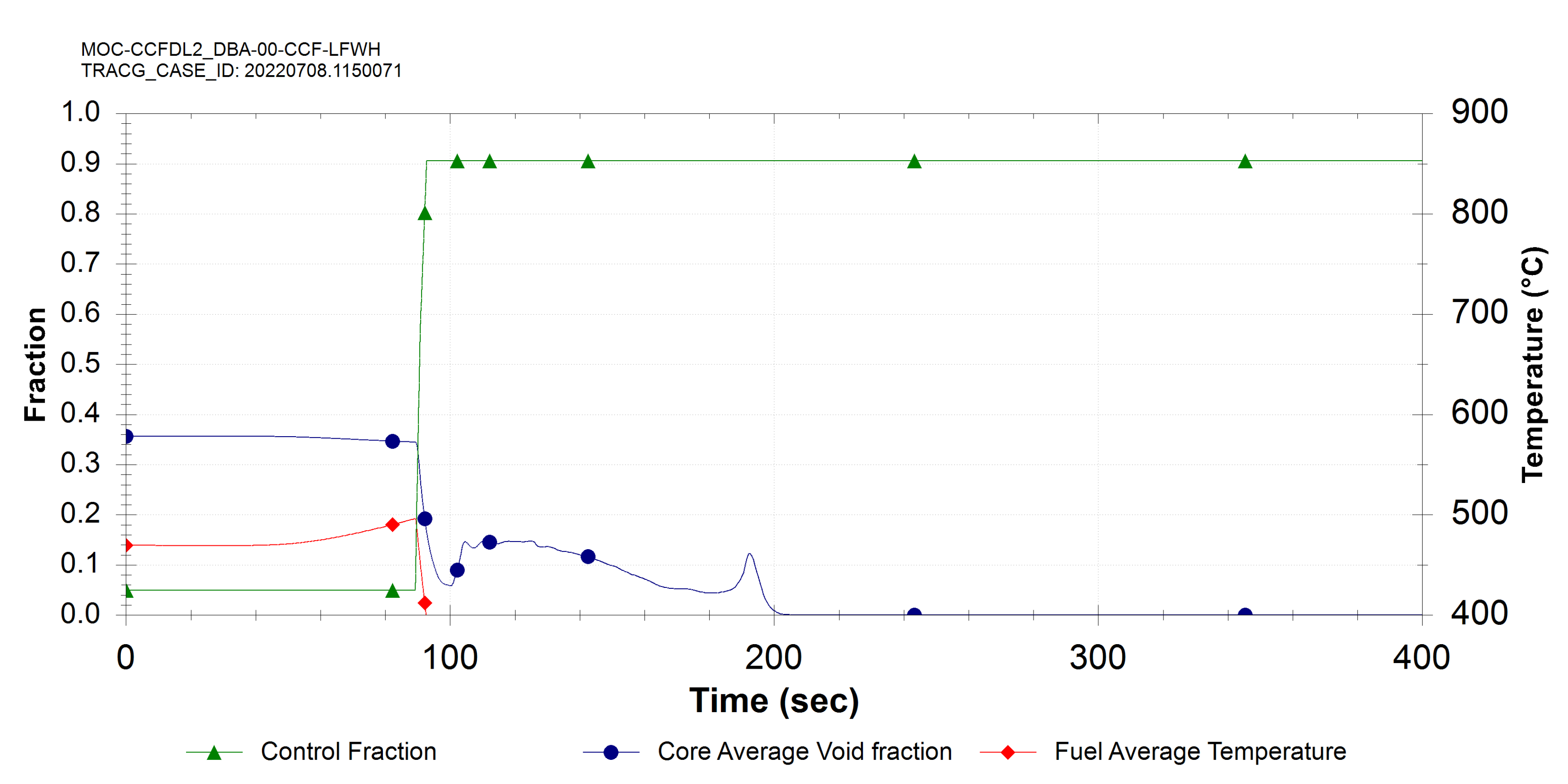

Figure 15.5-52: Loss of Feedwater Heating (DBA)

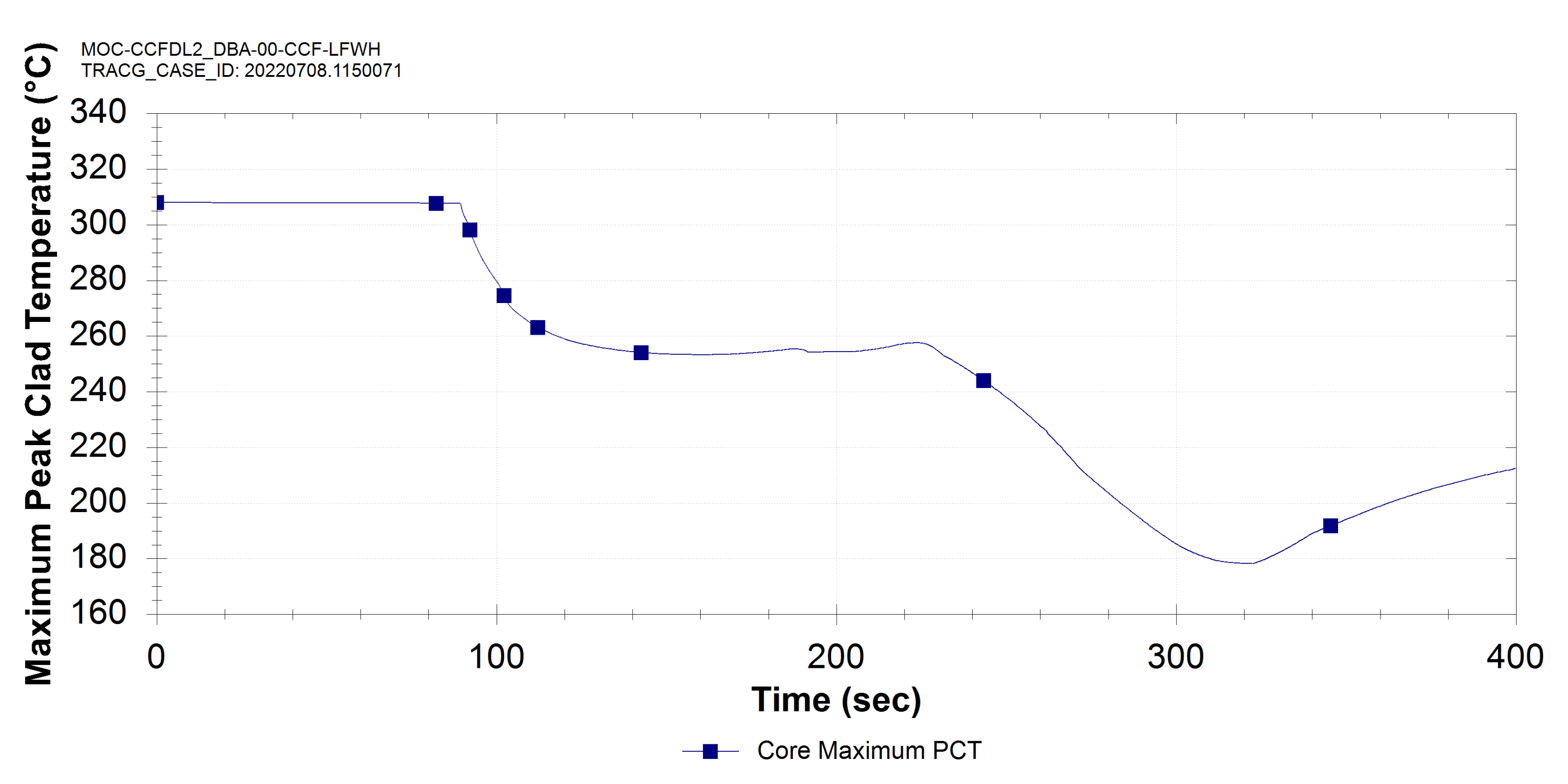

Figure 15.5-53: Loss of Feedwater Heating (DBA)

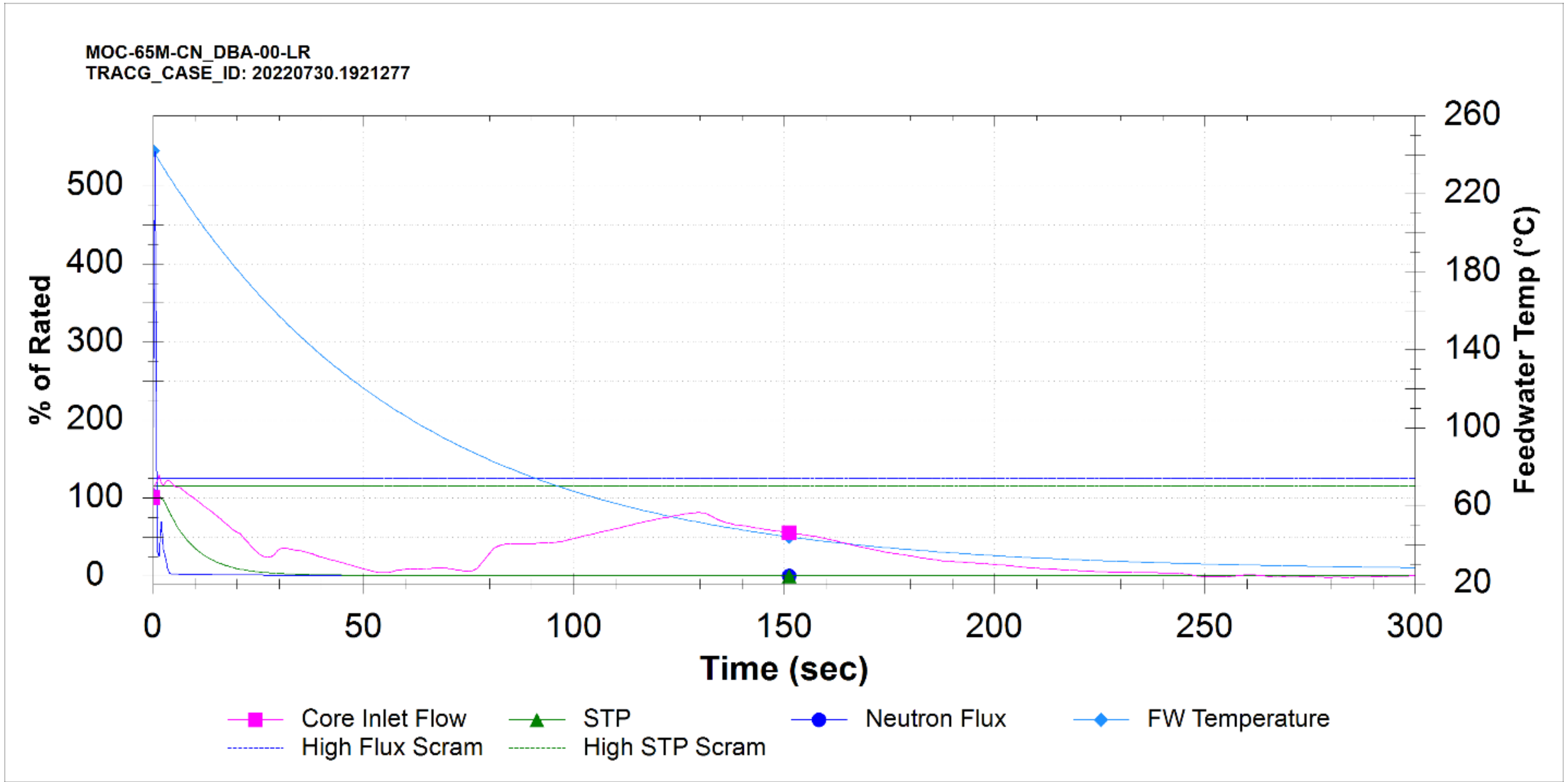

Figure 15.5-54: Generator Load Rejection (DBA)

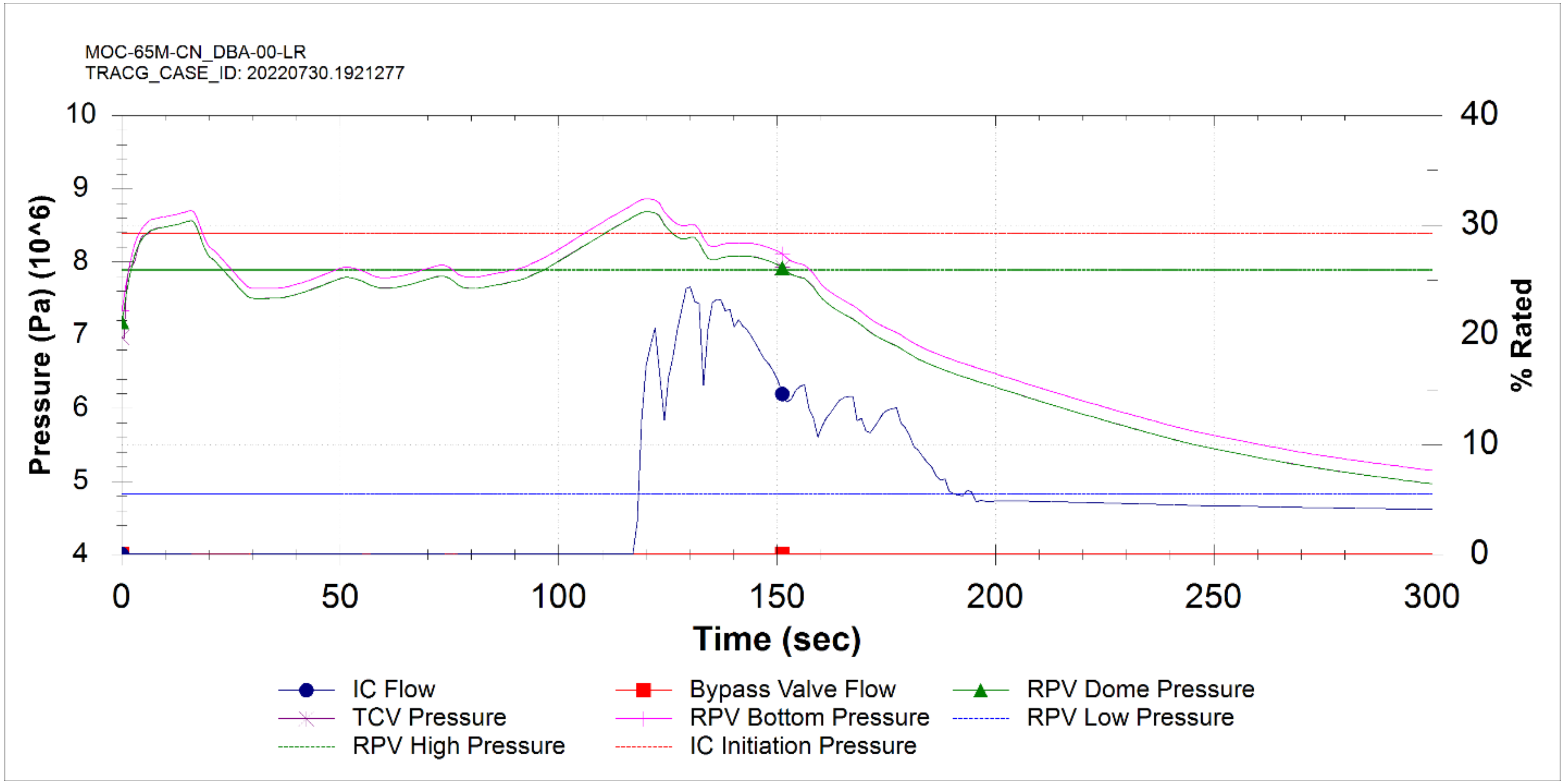

Figure 15.5-55: Generator Load Rejection (DBA)

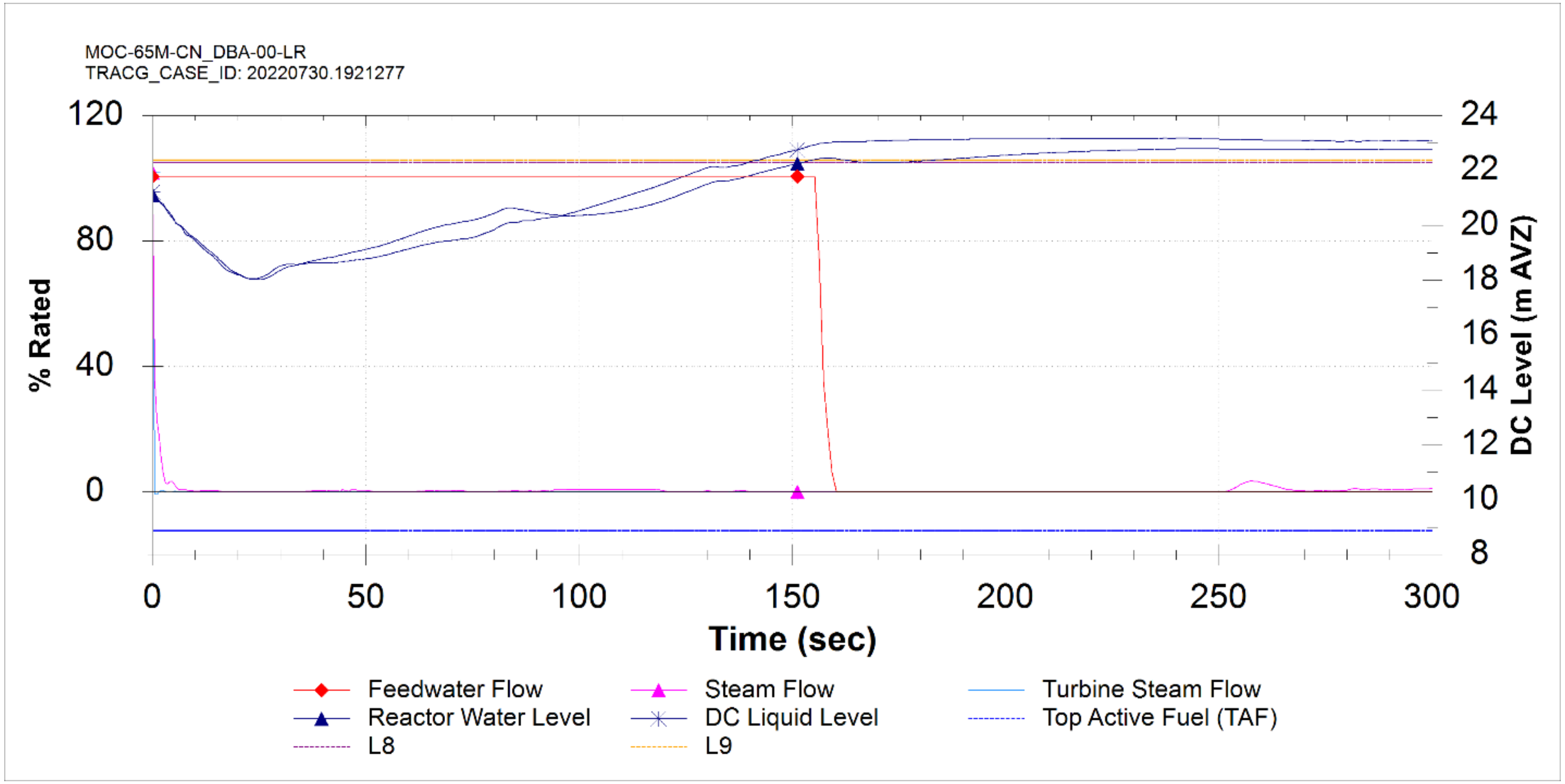

Figure 15.5-56: Generator Load Rejection (DBA)

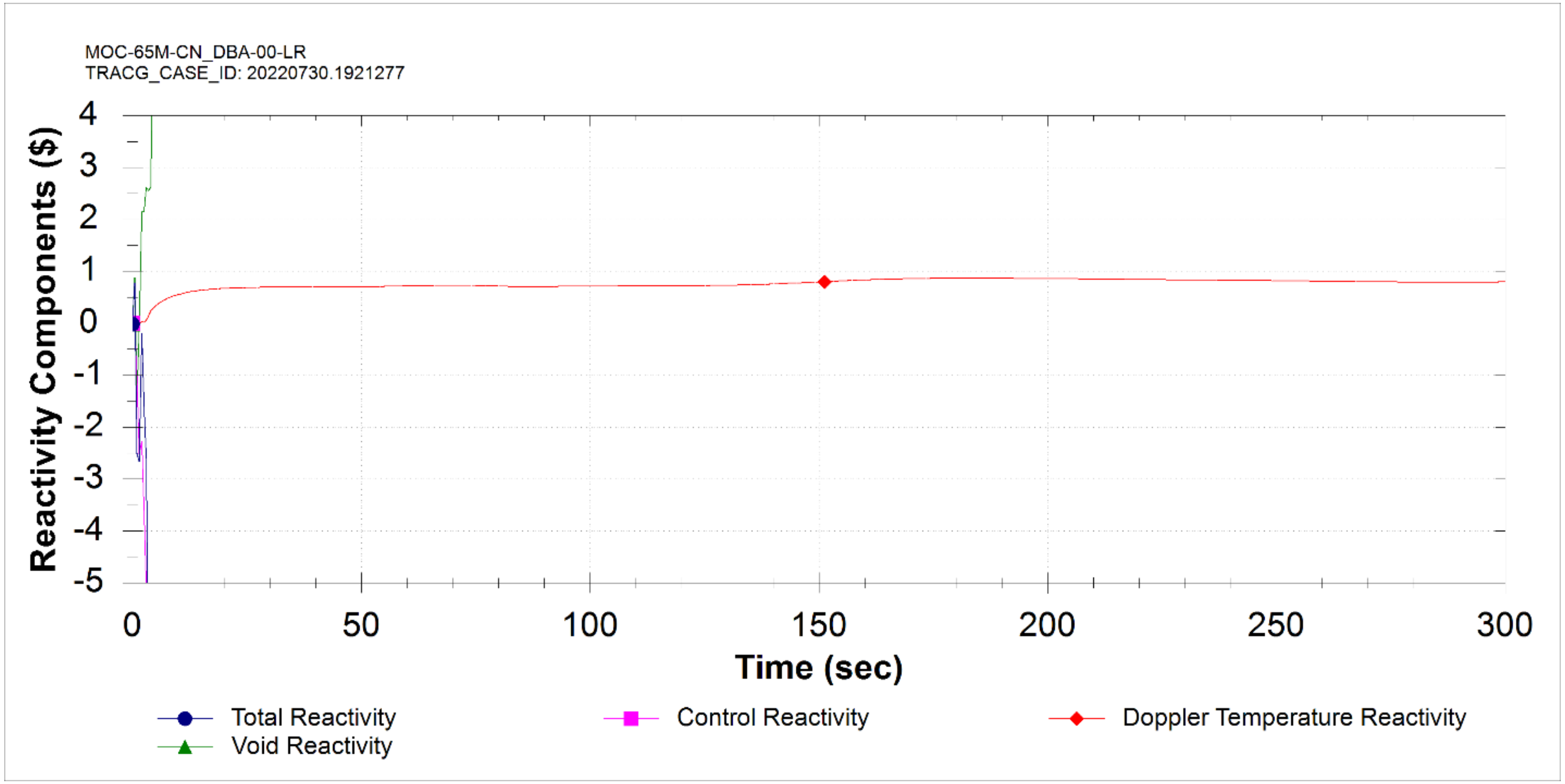

Figure 15.5-57: Generator Load Rejection (DBA)

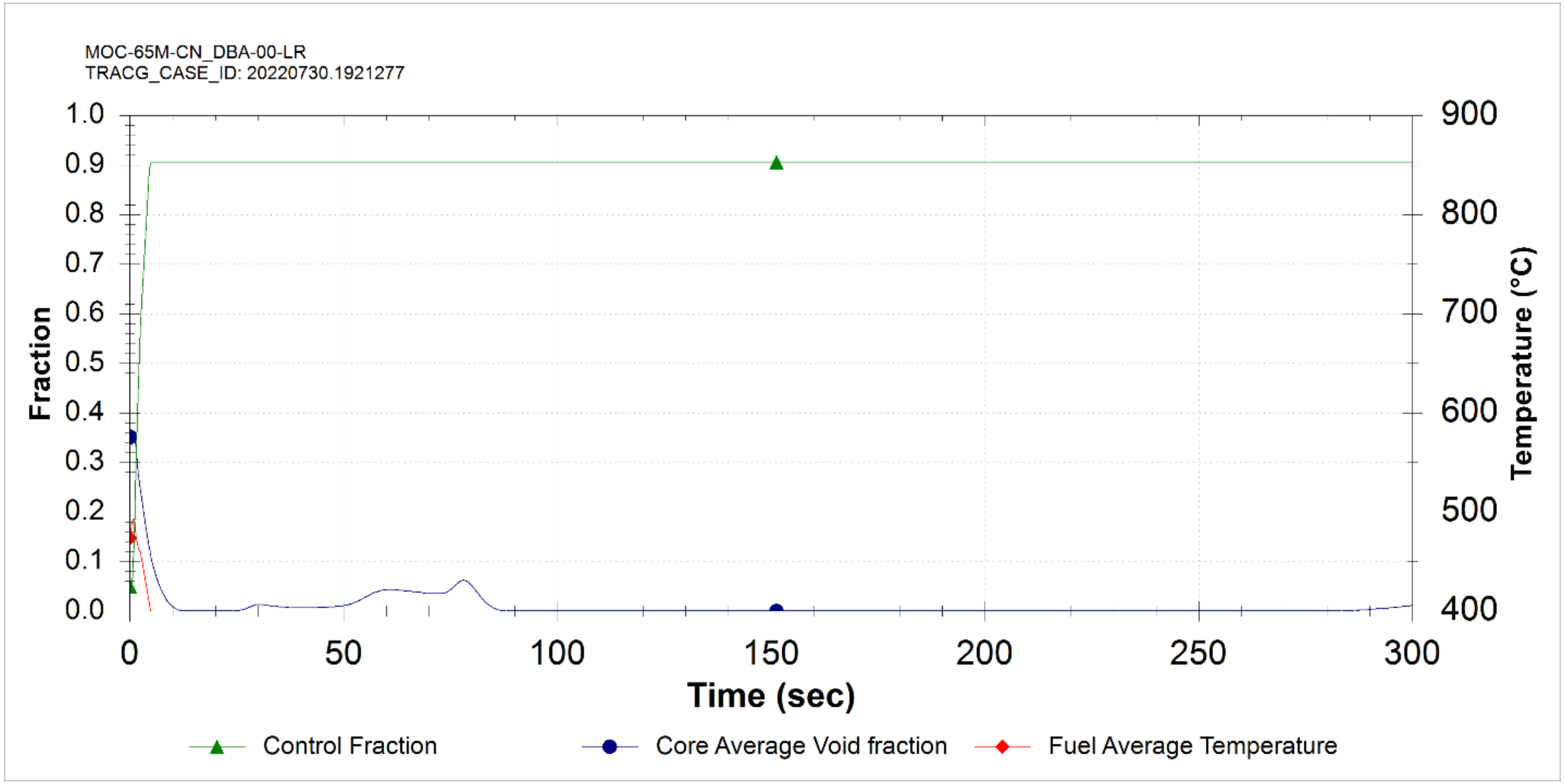

Figure 15.5-58: Generator Load Rejection (DBA)

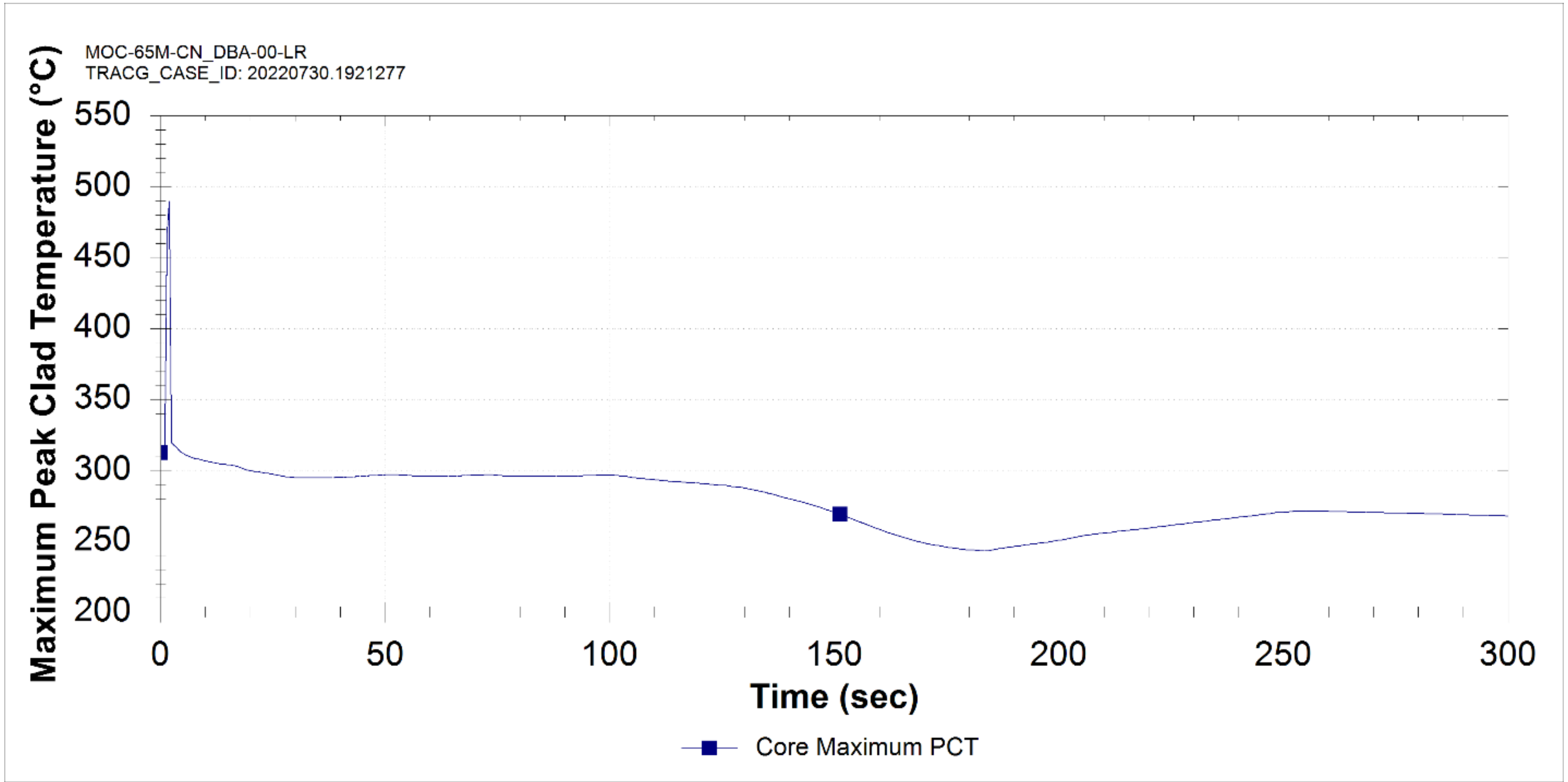

Figure 15.5-59: Generator Load Rejection (DBA)

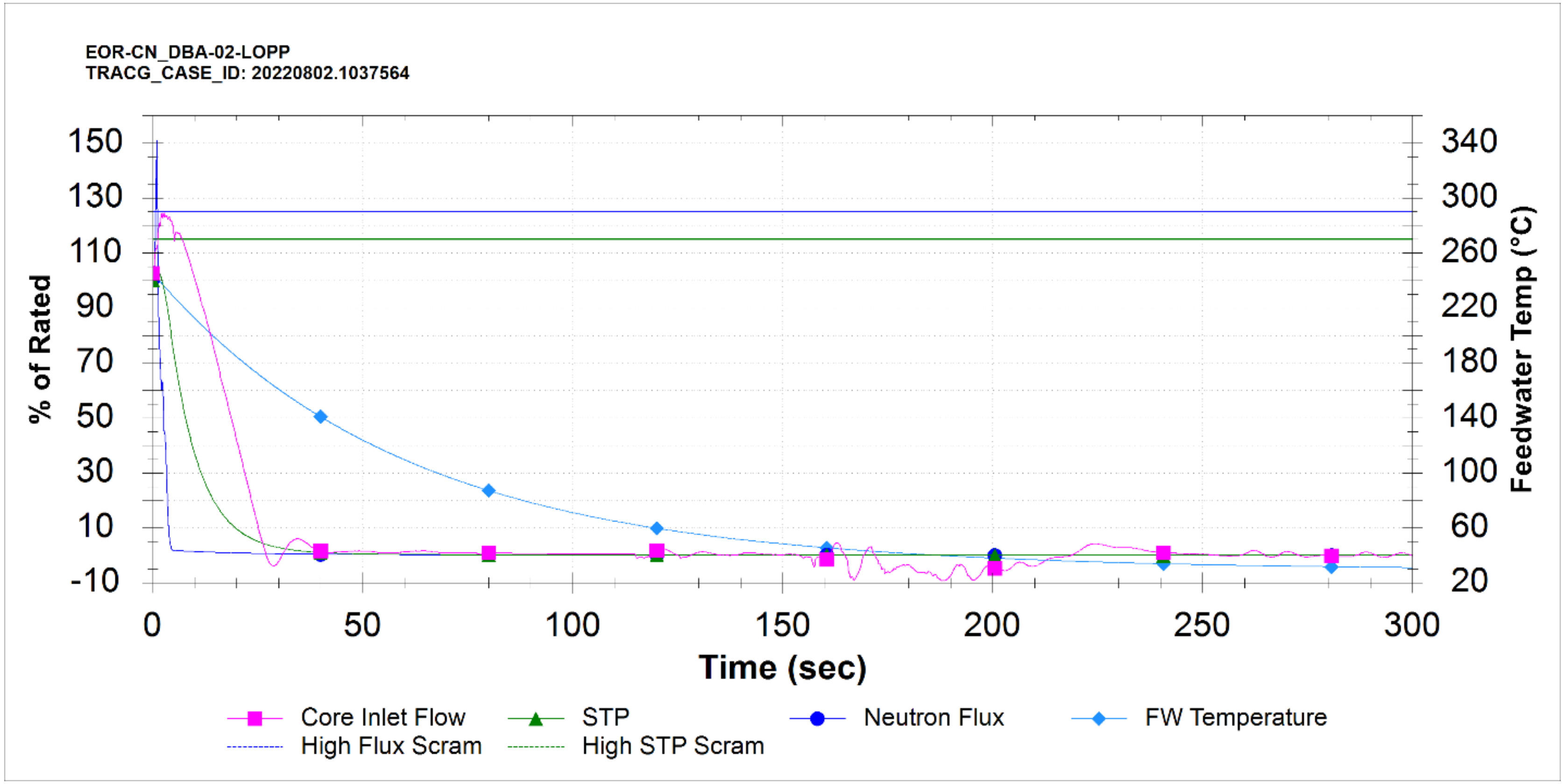

Figure 15.5-60: Loss of Preferred Power (DBA)

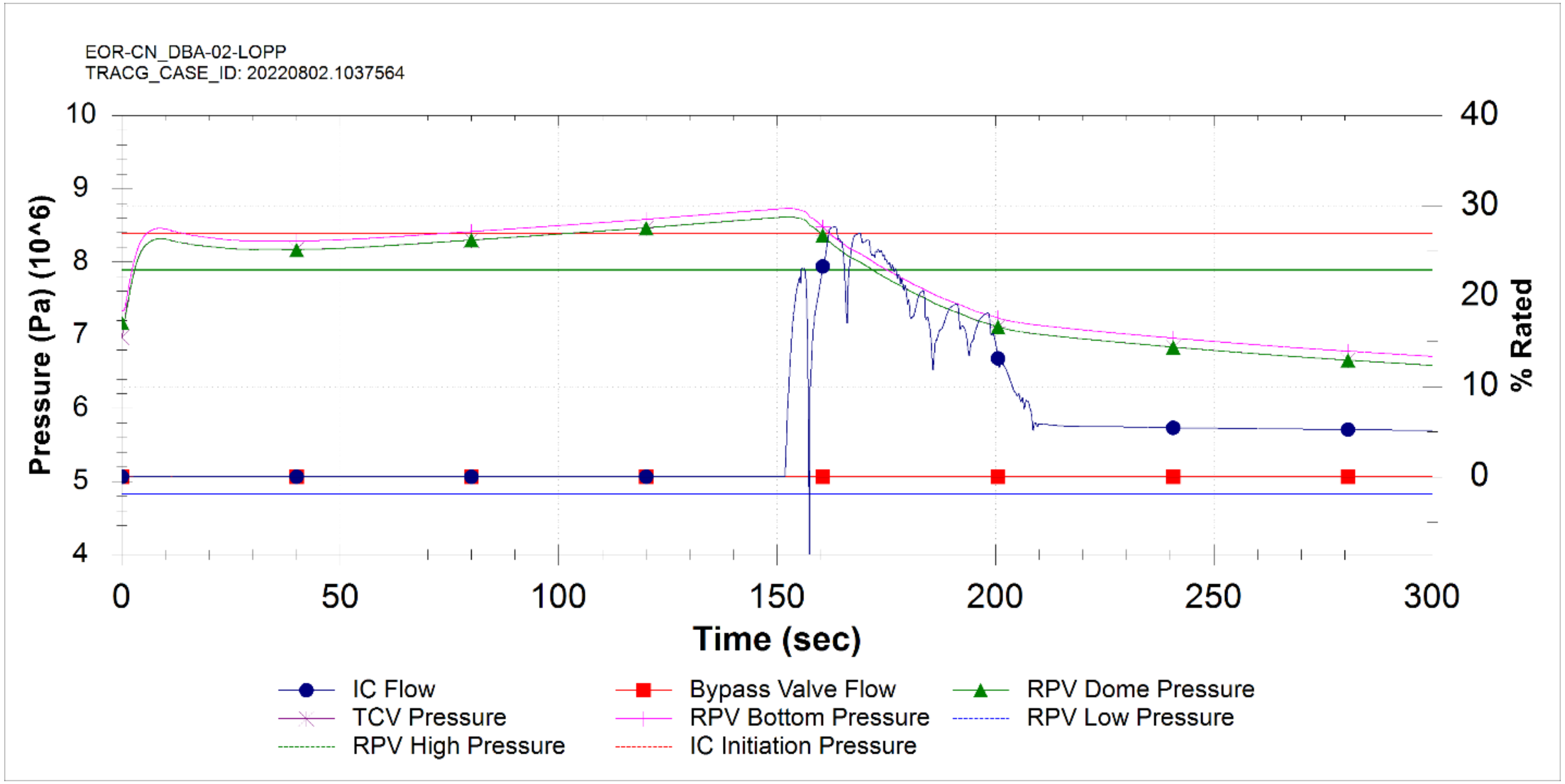

Figure 15.5-61: Loss of Preferred Power (DBA)

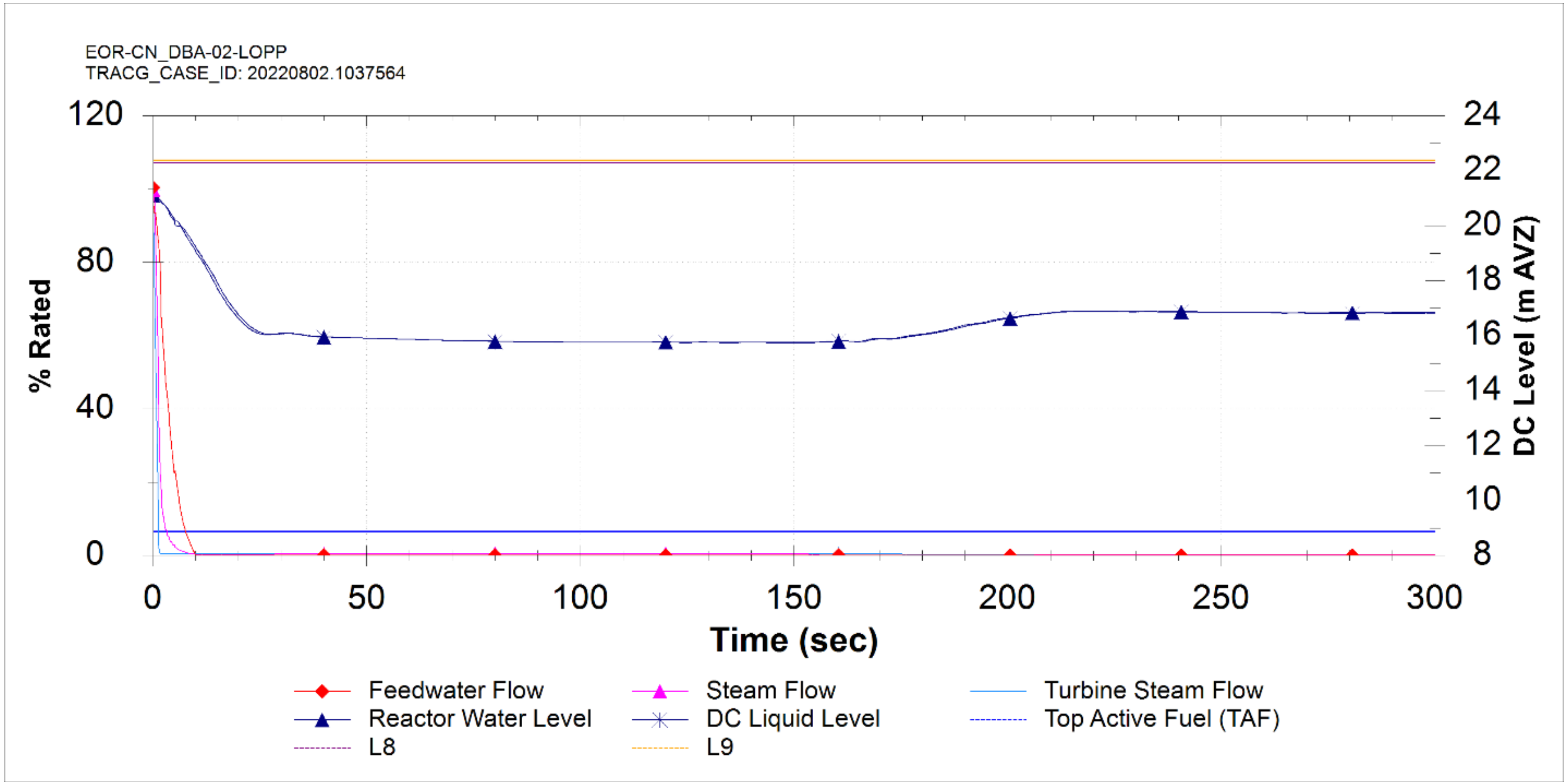

Figure 15.5-62: Loss of Preferred Power (DBA)

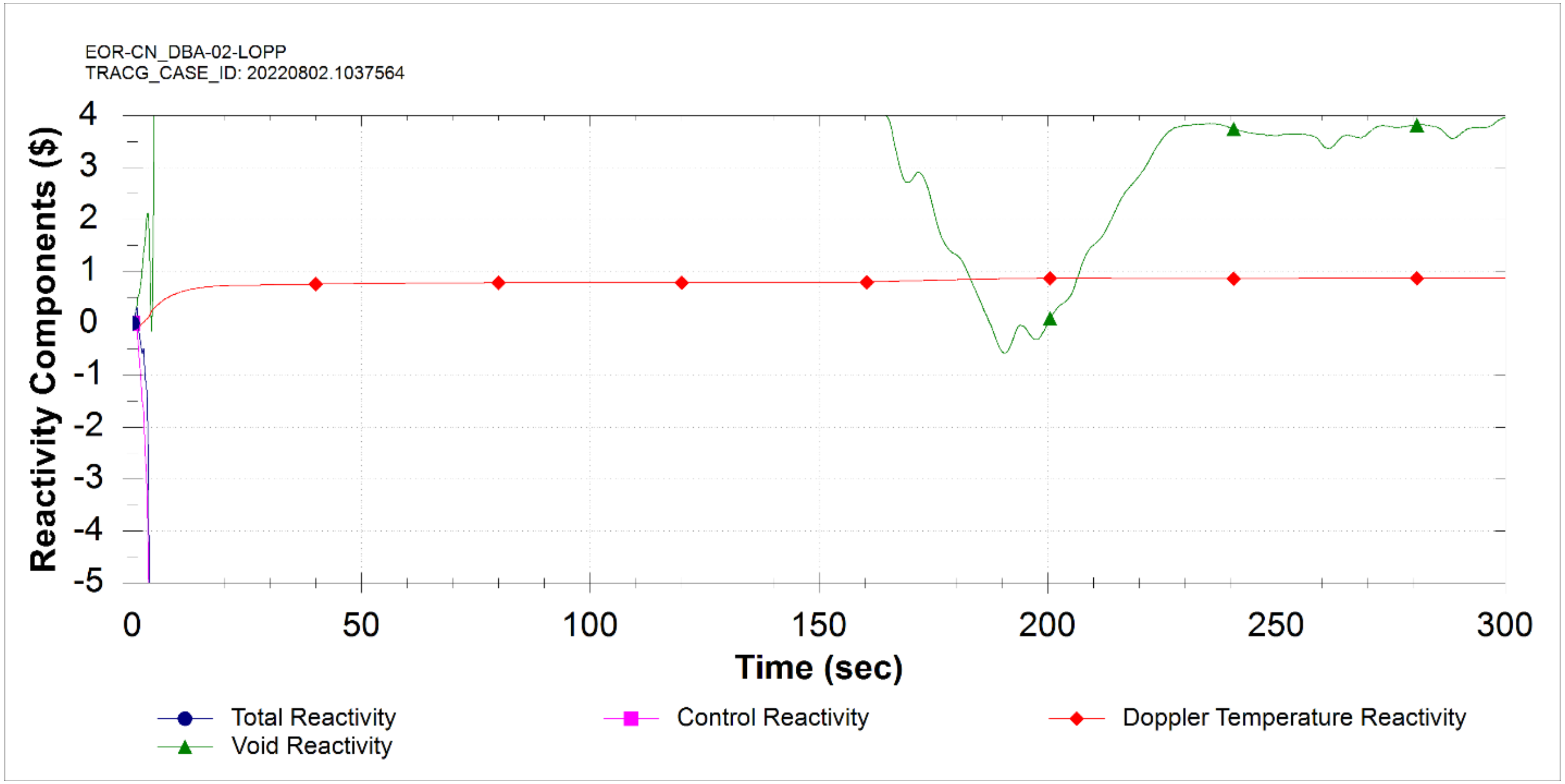

Figure 15.5-63: Loss of Preferred Power (DBA)

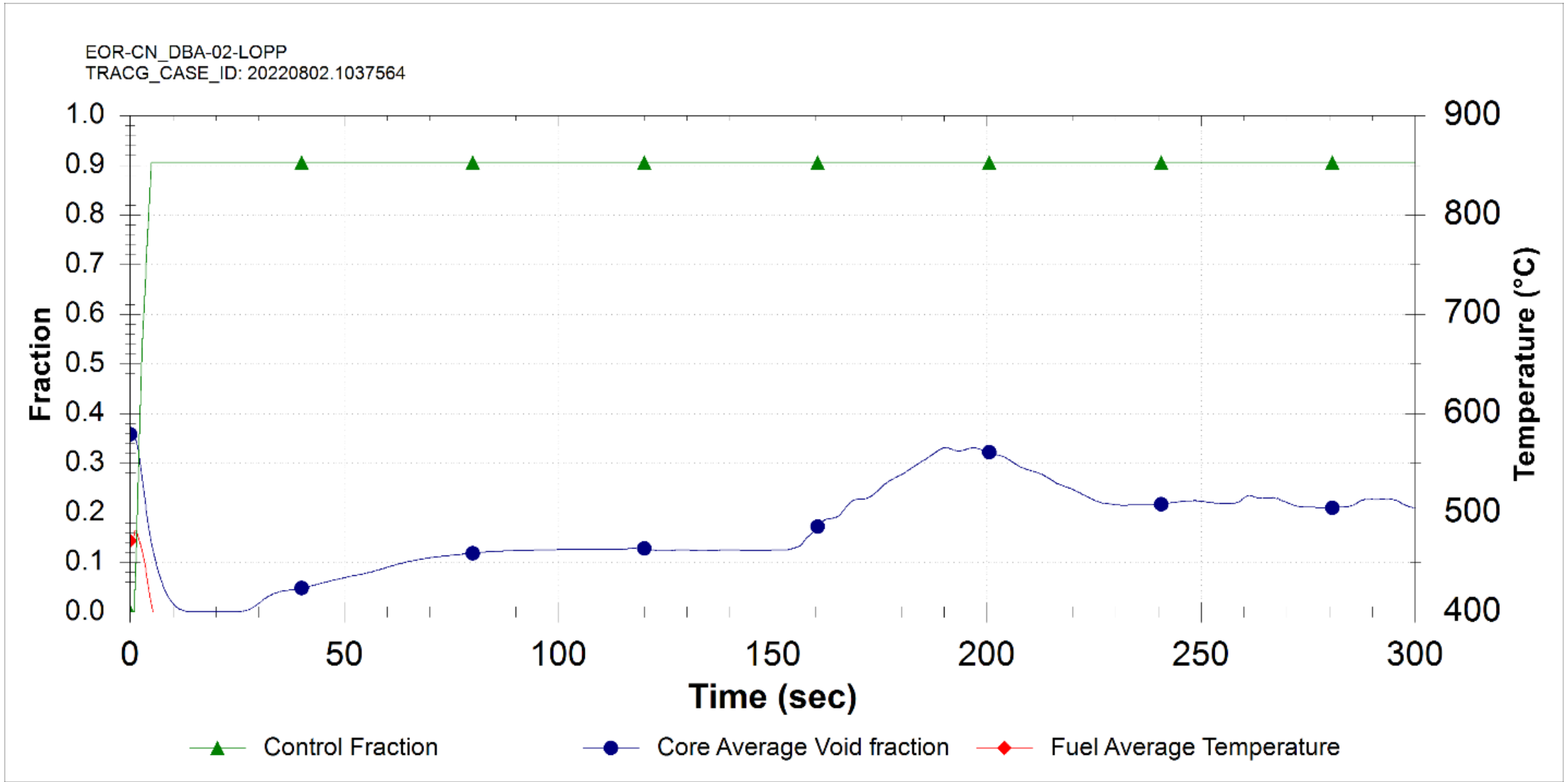

Figure 15.5-64: Loss of Preferred Power (DBA)

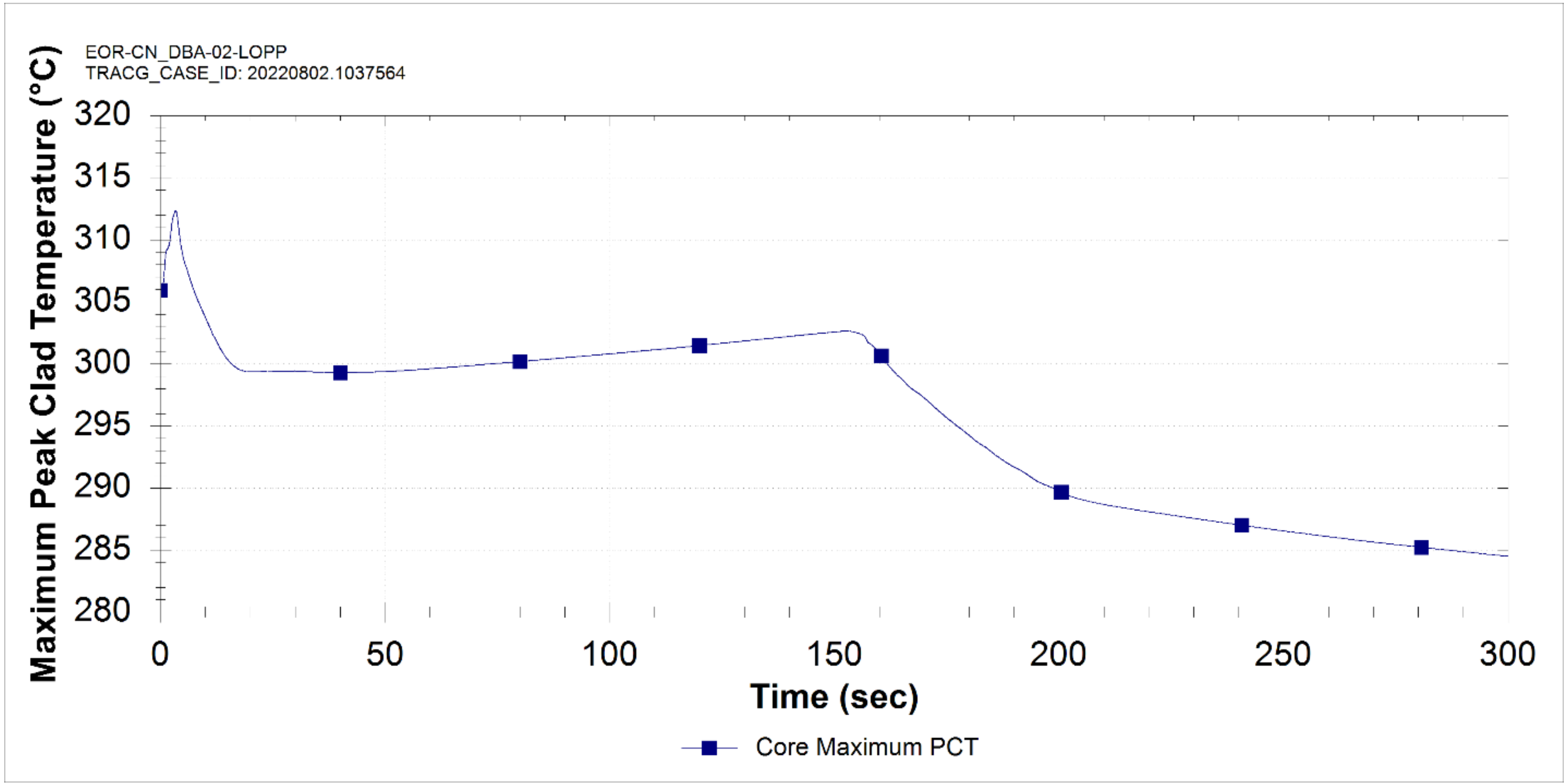

Figure 15.5-65: Loss of Preferred Power (DBA)

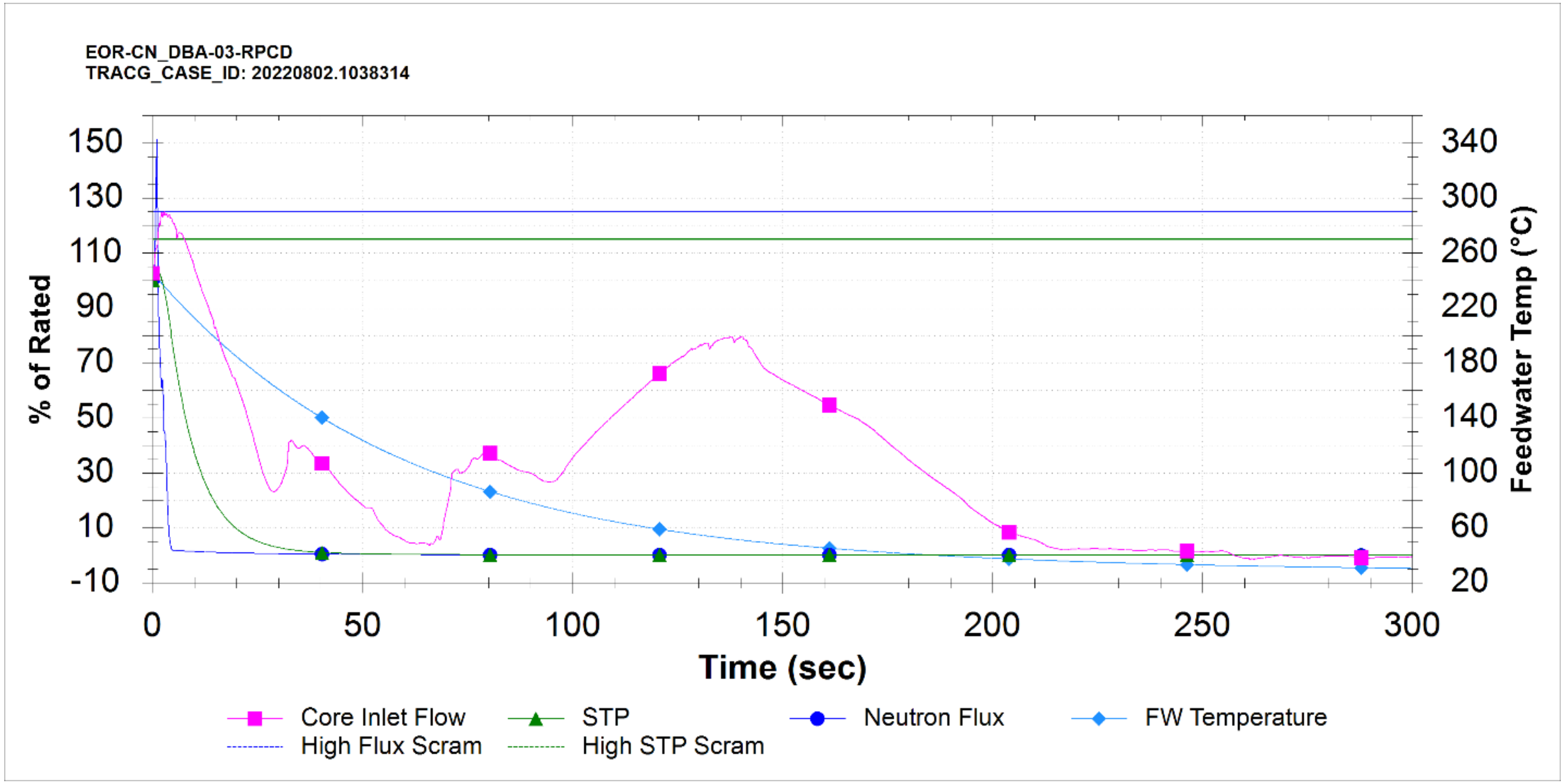

Figure 15.5-66: RPV Pressure Control Downscale (DBA)

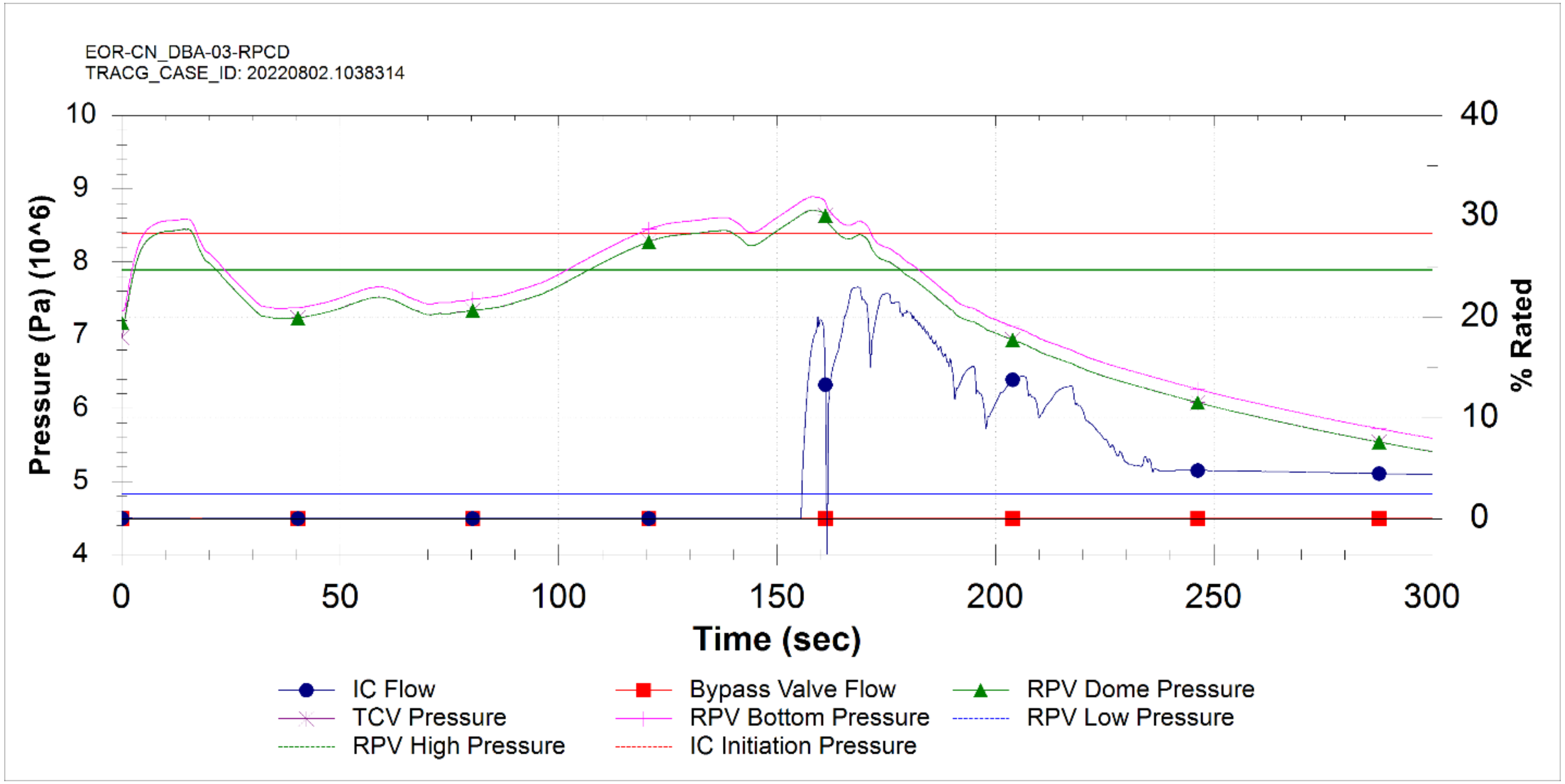

Figure 15.5-67: RPV Pressure Control Downscale (DBA)

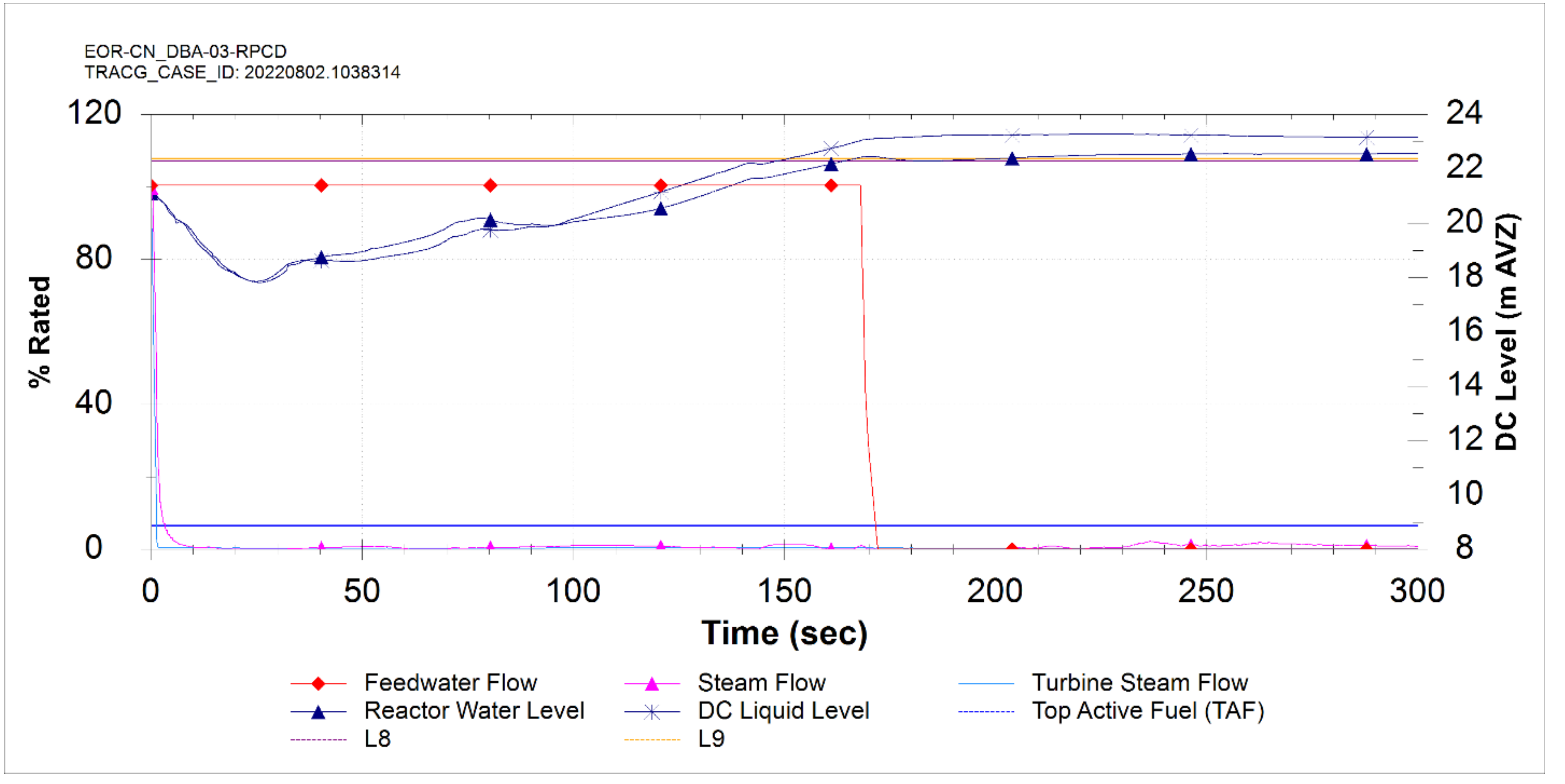

Figure 15.5-68: RPV Pressure Control Downscale (DBA)

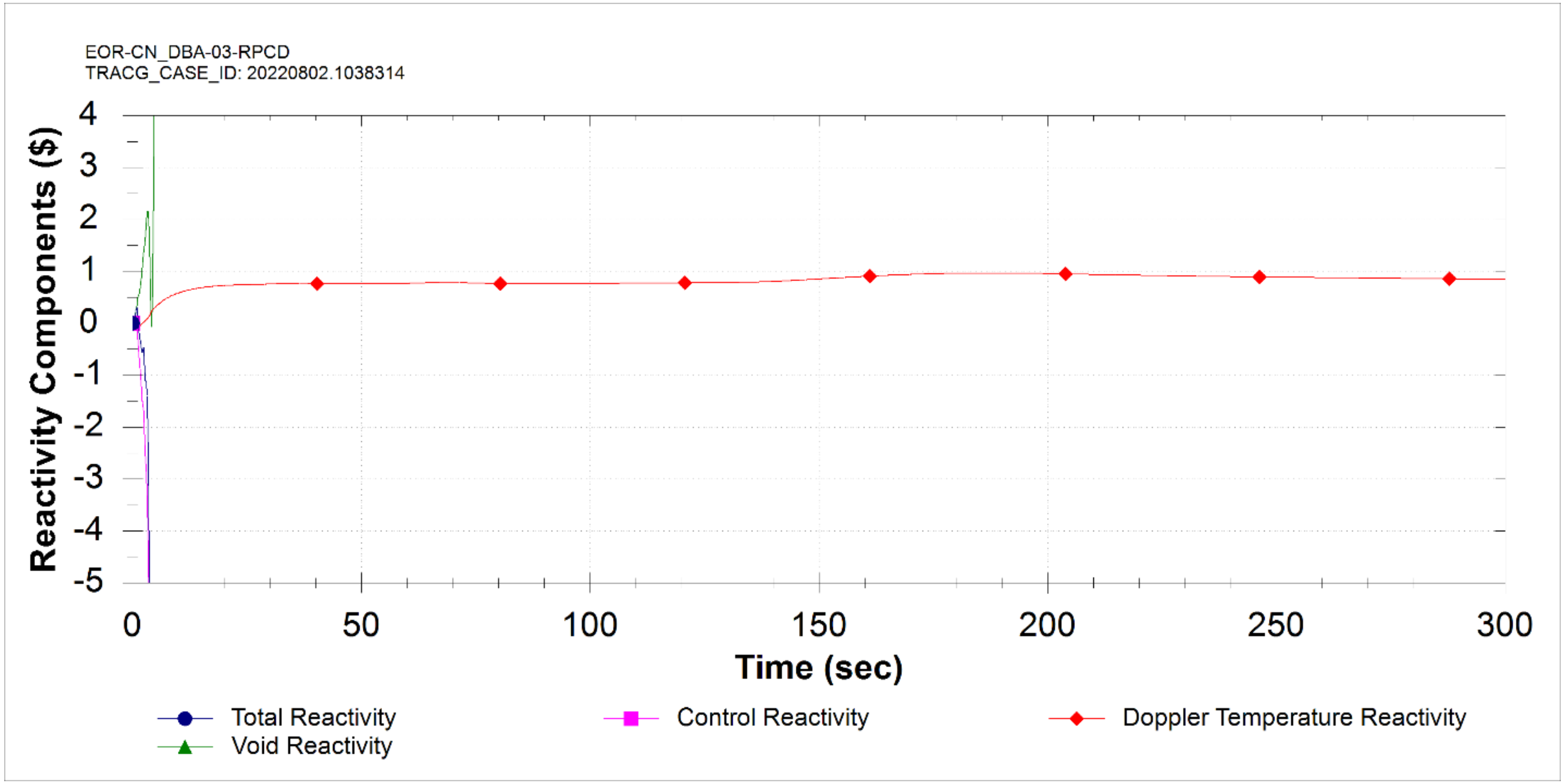

Figure 15.5-69: RPV Pressure Control Downscale (DBA)

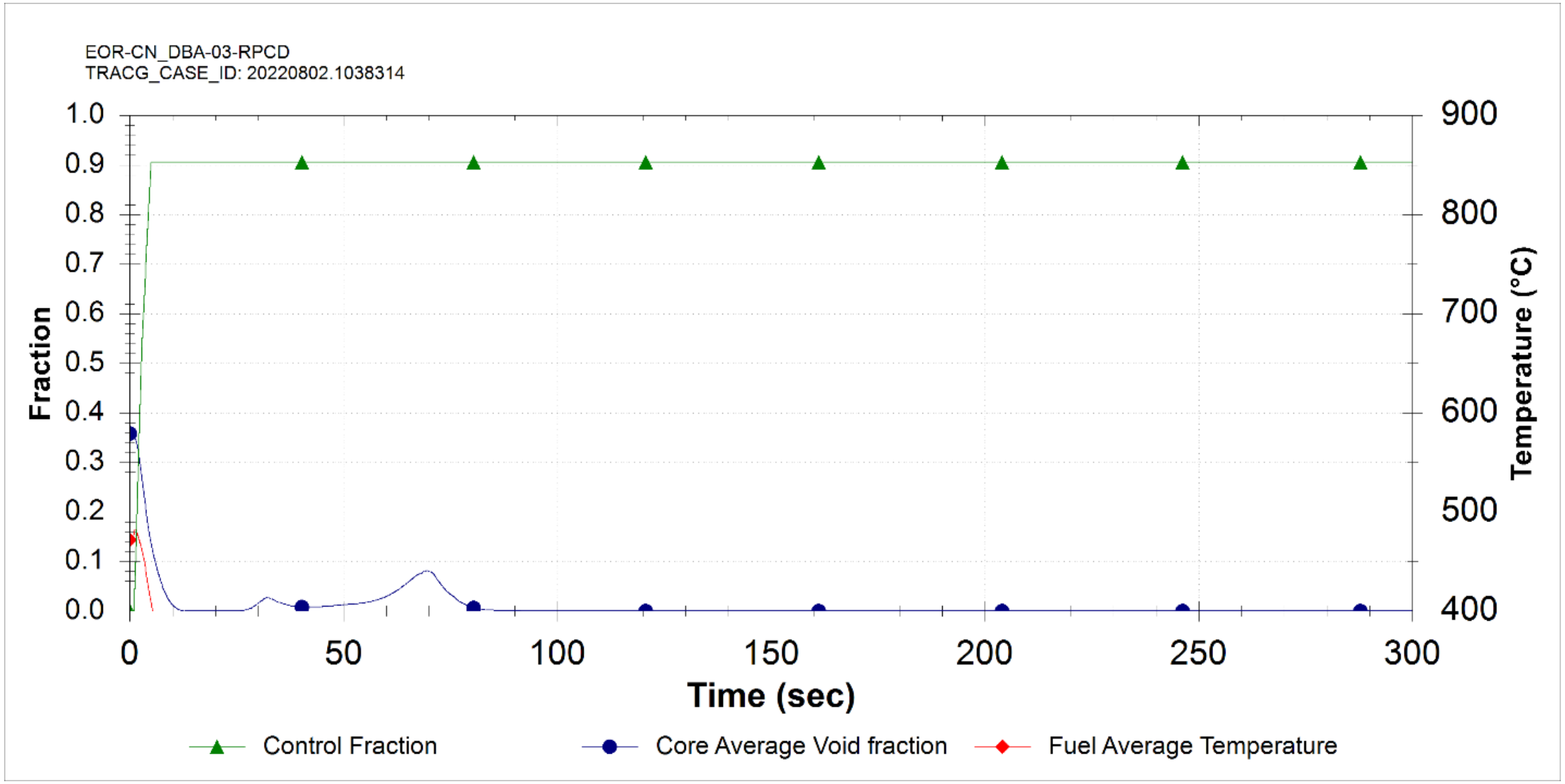

Figure 15.5-70: RPV Pressure Control Downscale (DBA)

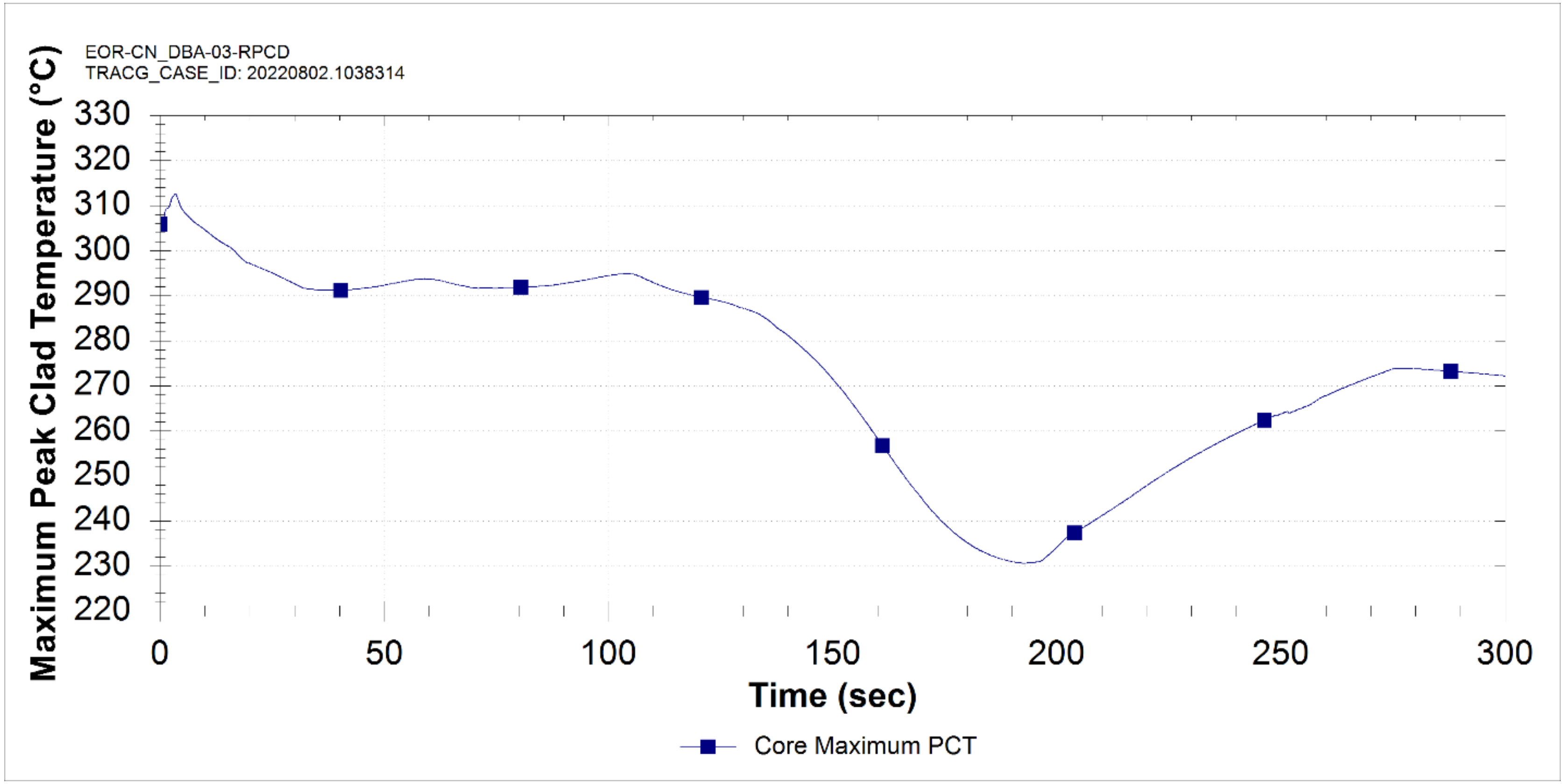

Figure 15.5-71: RPV Pressure Control Downscale (DBA)

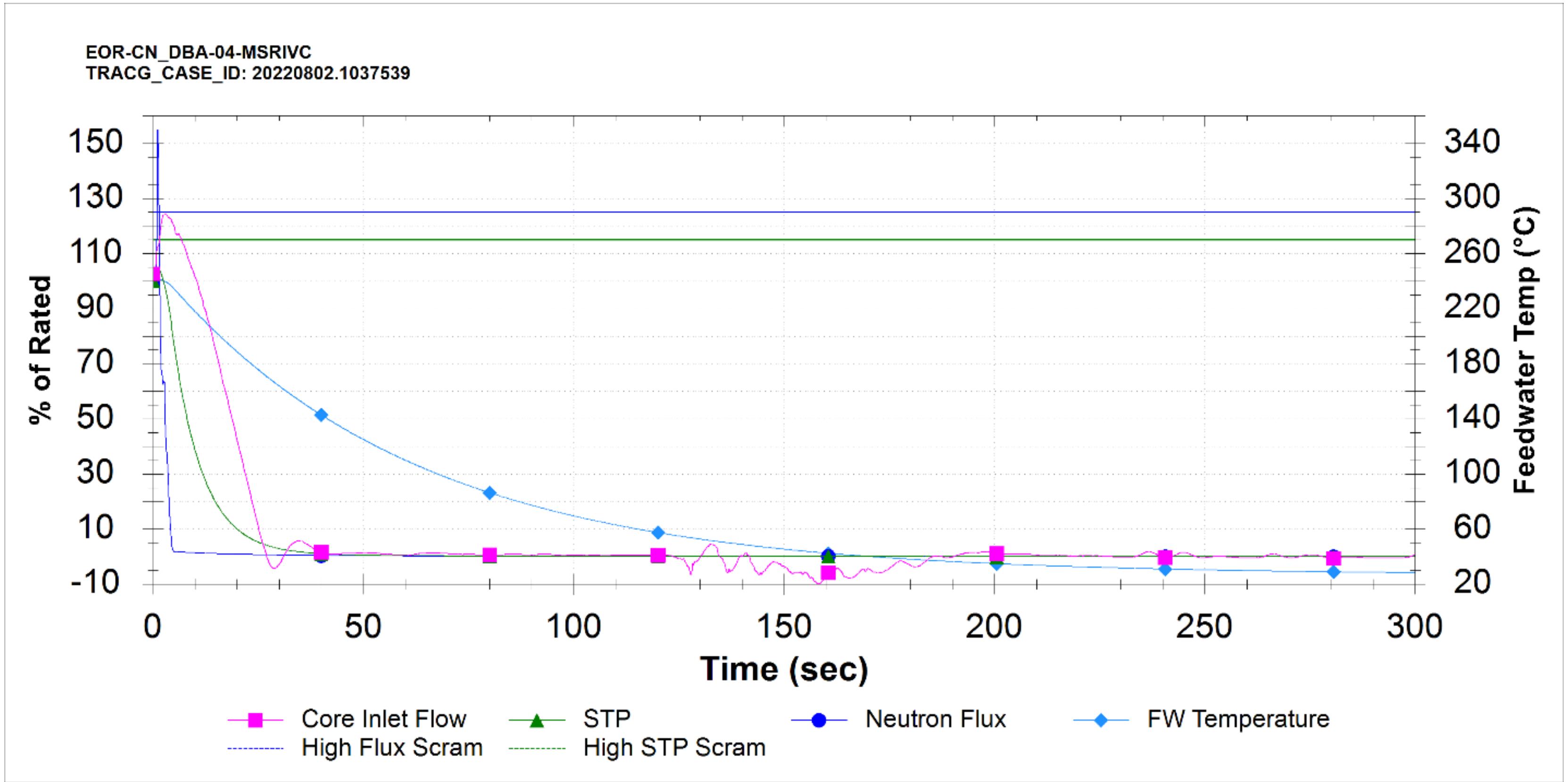

Figure 15.5-72: Closure of All MSRIVs and FW Isolation Valves (DBA)

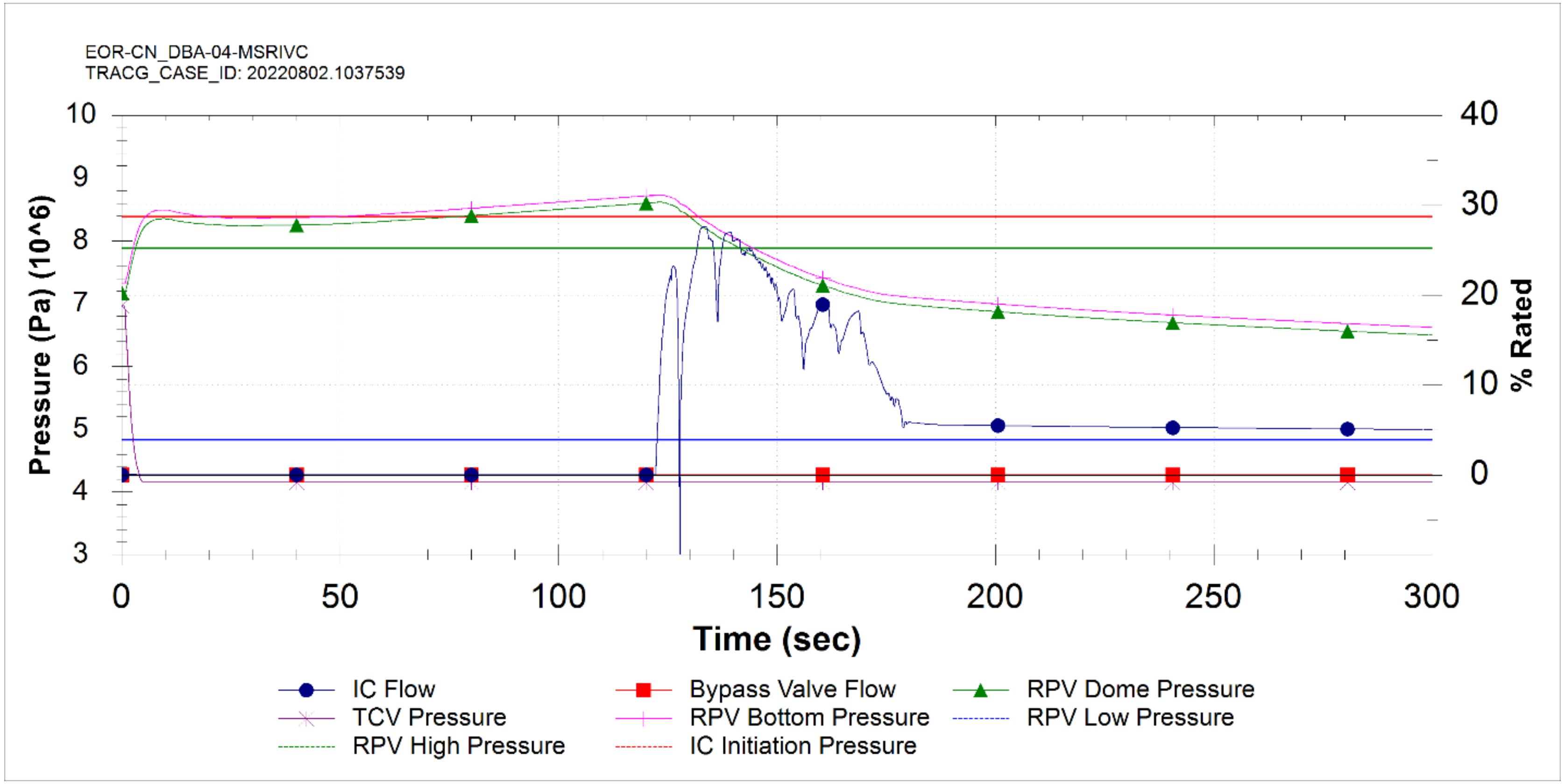

Figure 15.5-73: Closure of All MSRIVs and FW Isolation Valves (DBA)

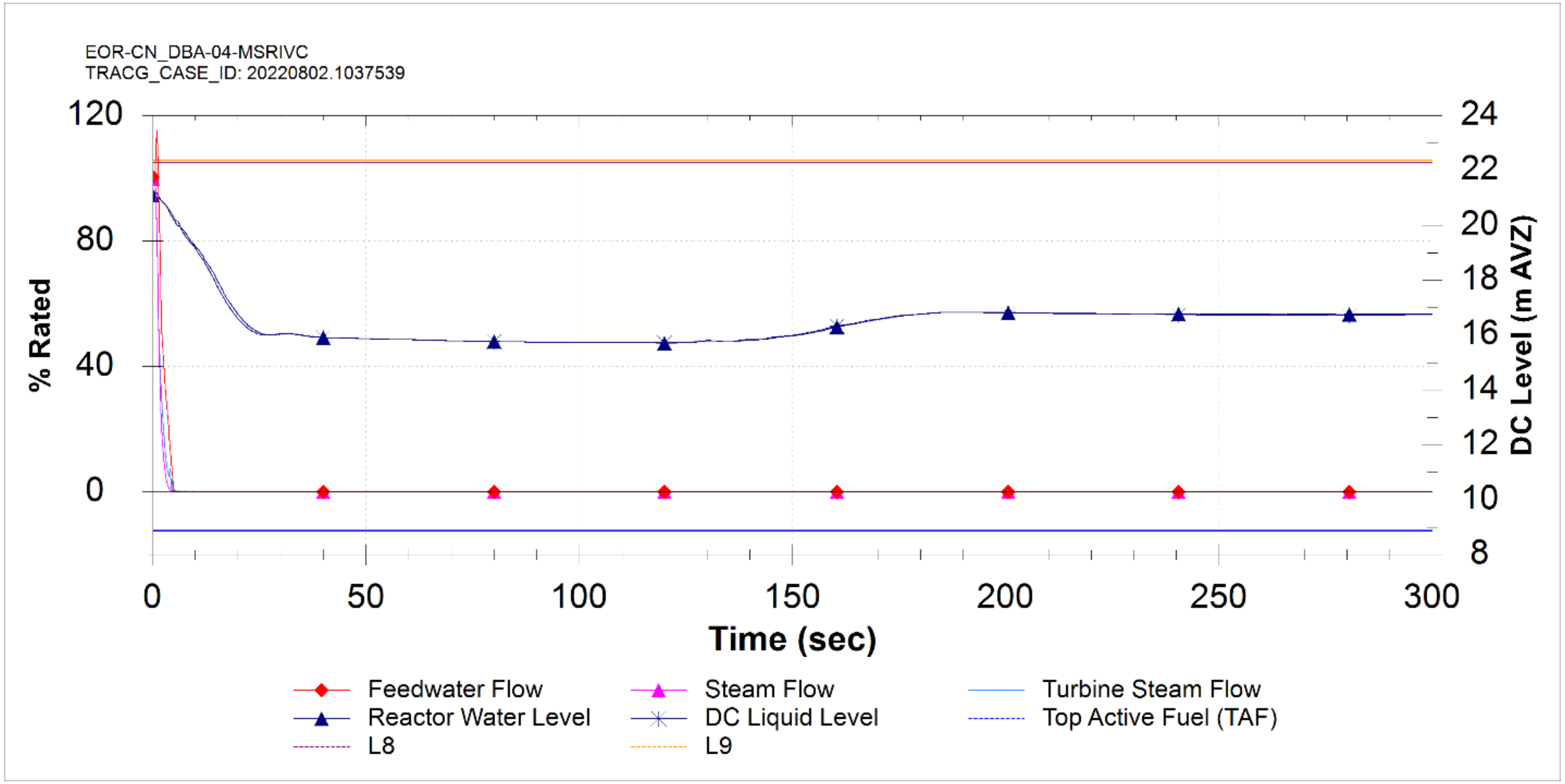

Figure 15.5-74: Closure of All MSRIVs and FW Isolation Valves (DBA)

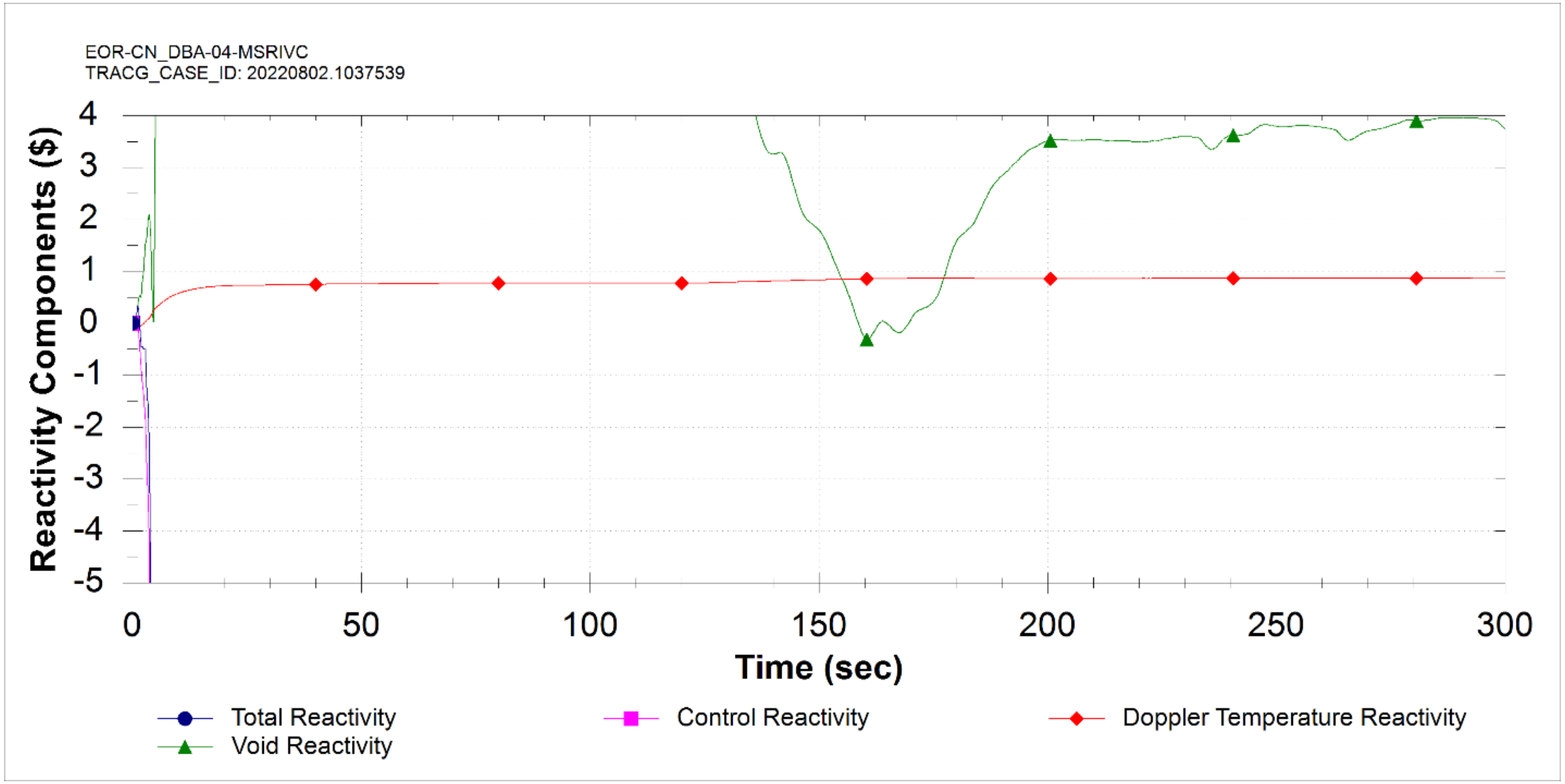

Figure 15.5-75: Closure of All MSRIVs and FW Isolation Valves (DBA)

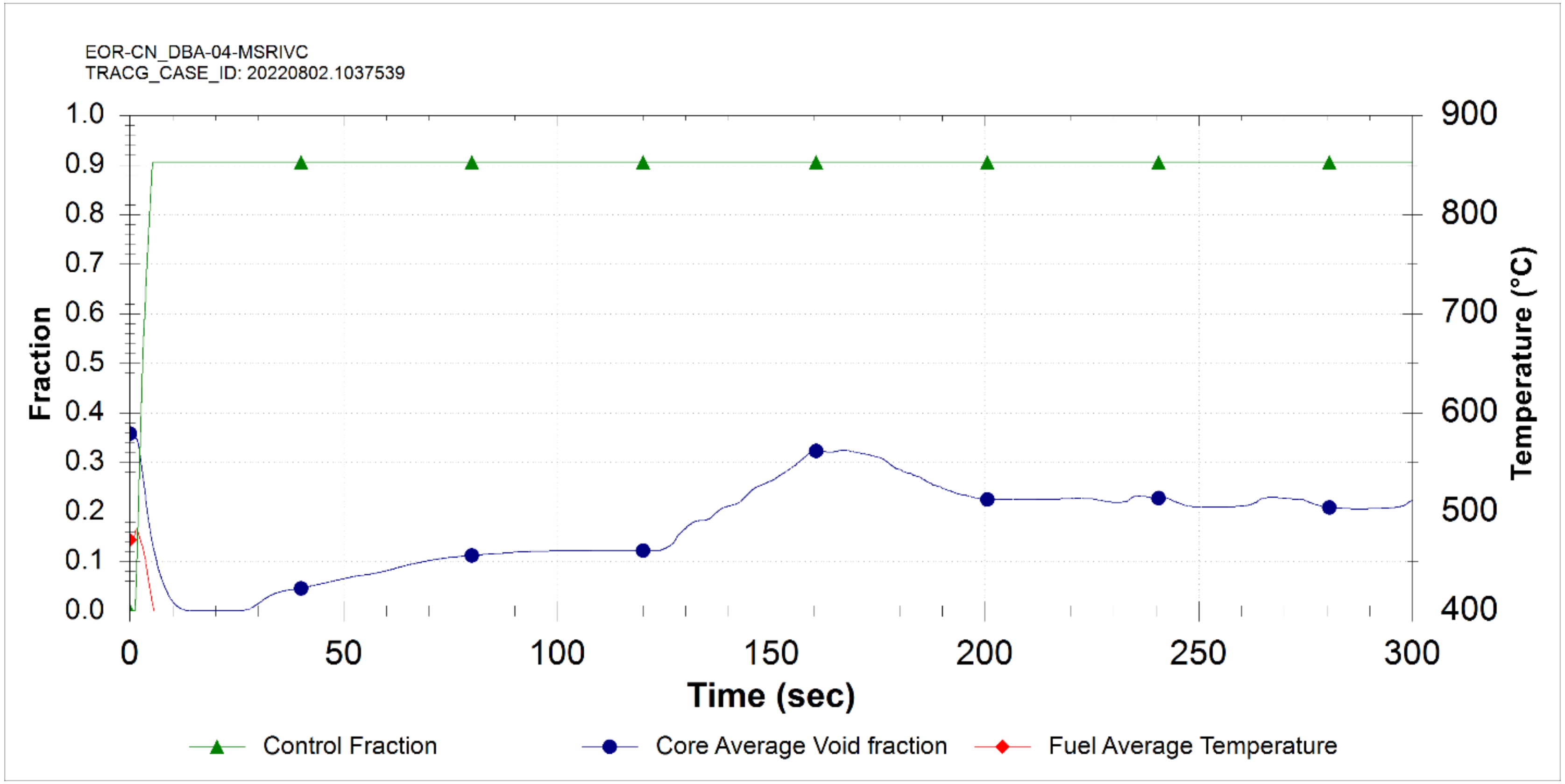

Figure 15.5-76: Closure of All MSRIVs and FW Isolation Valves (DBA)

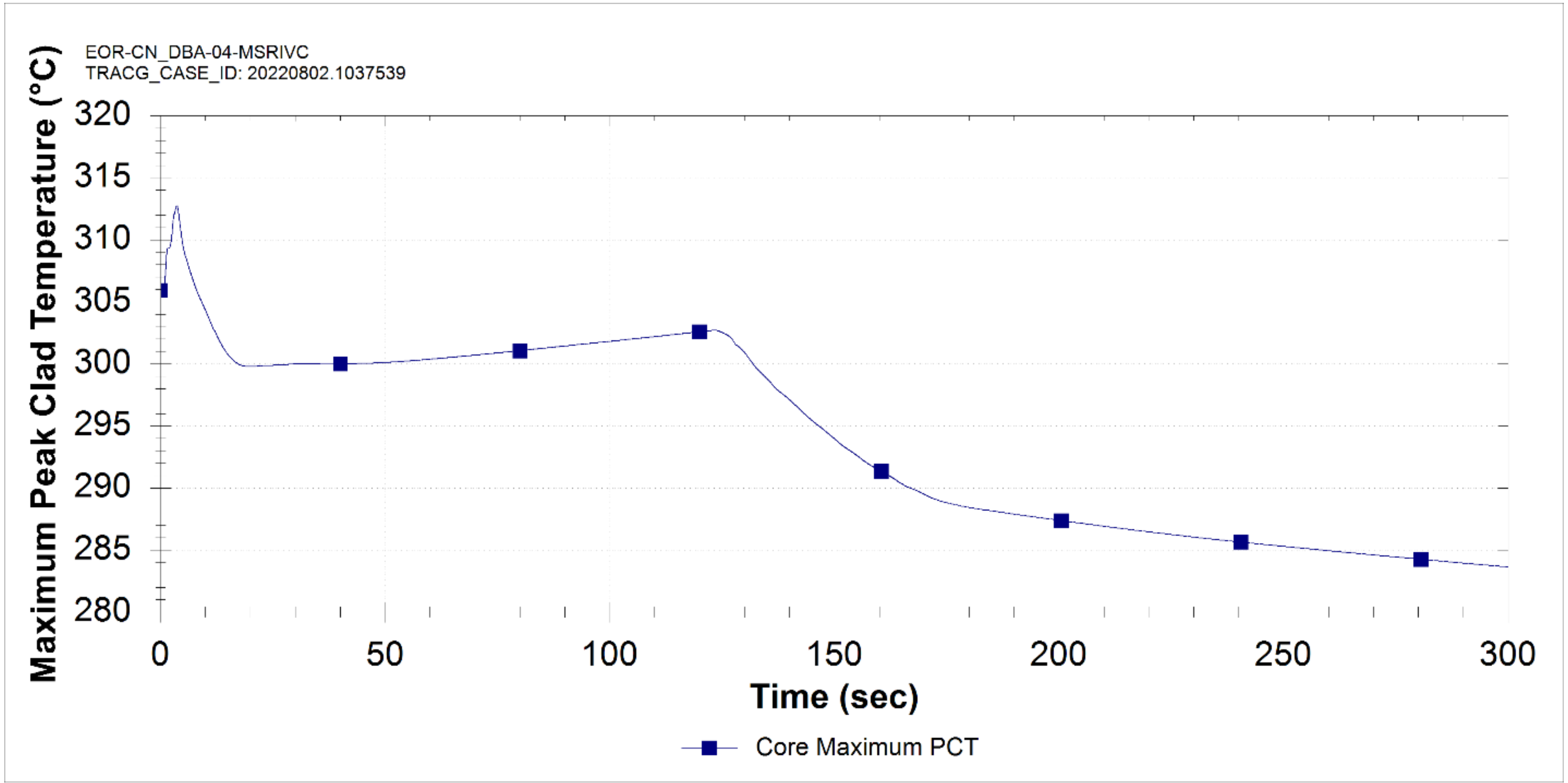

Figure 15.5-77: Closure of All MSRIVs and FW Isolation Valves (DBA)

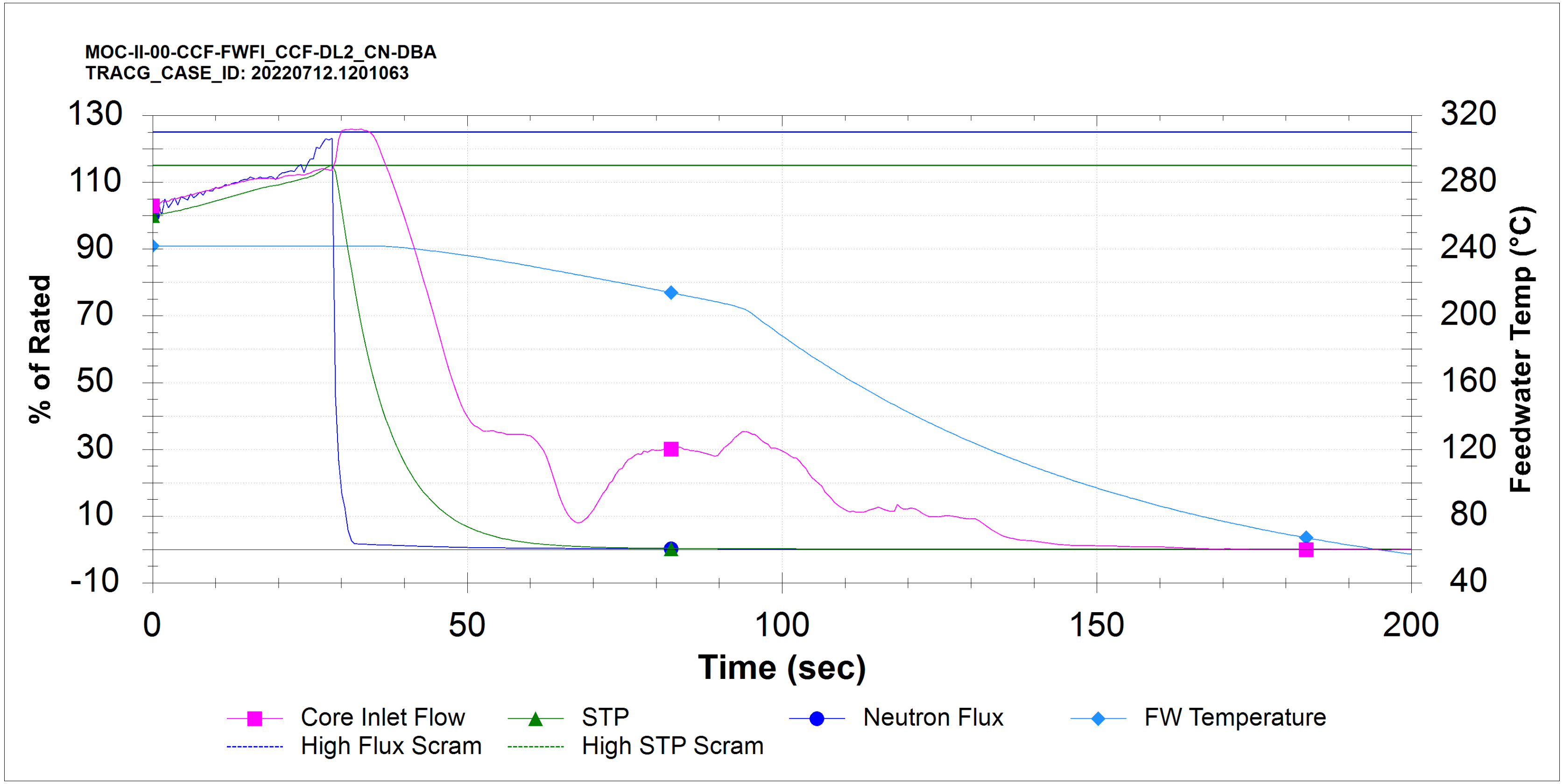

Figure 15.5-78: Feedwater Flow Increase (DBA)

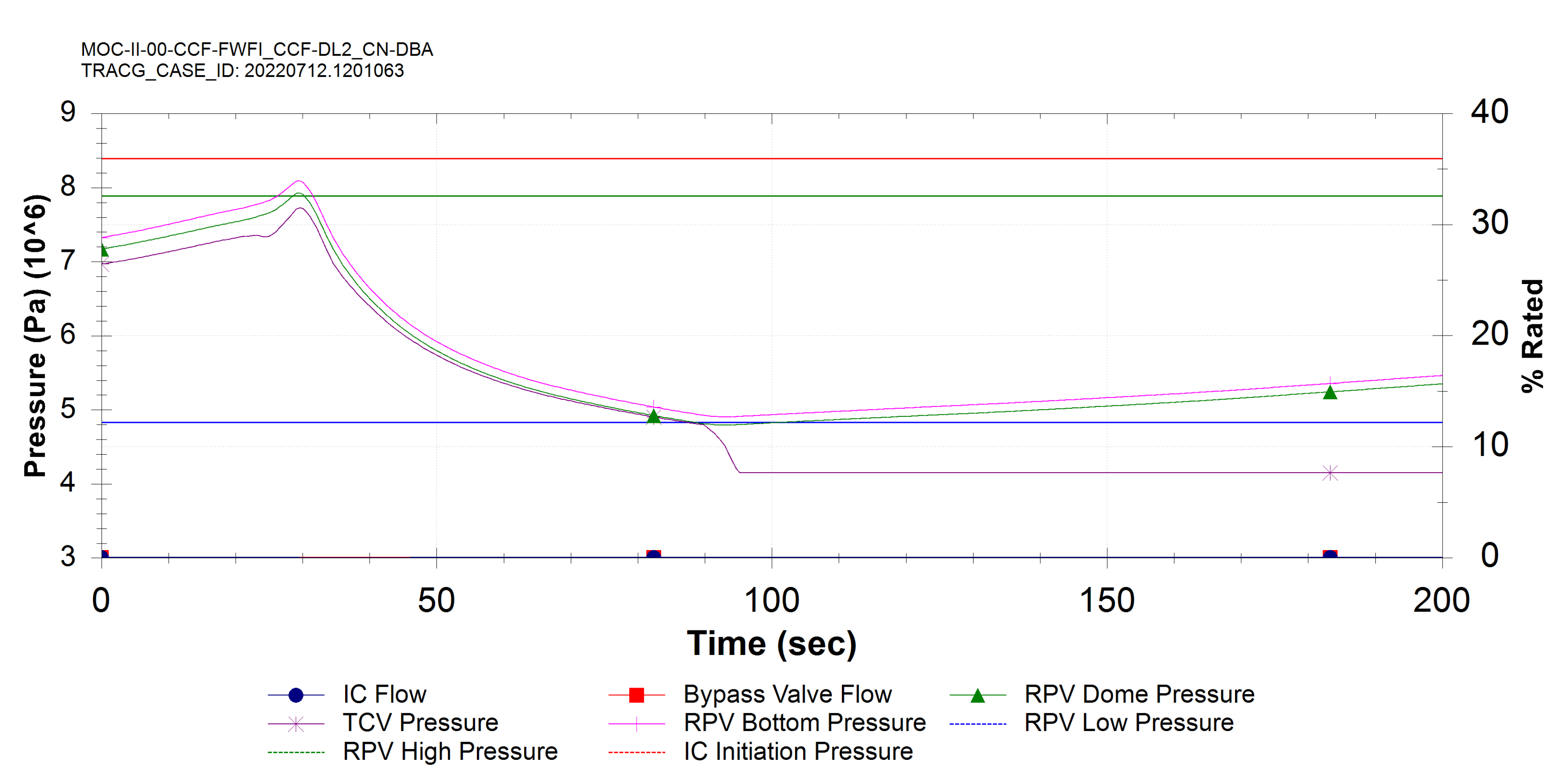

Figure 15.5-79: Feedwater Flow Increase (DBA)

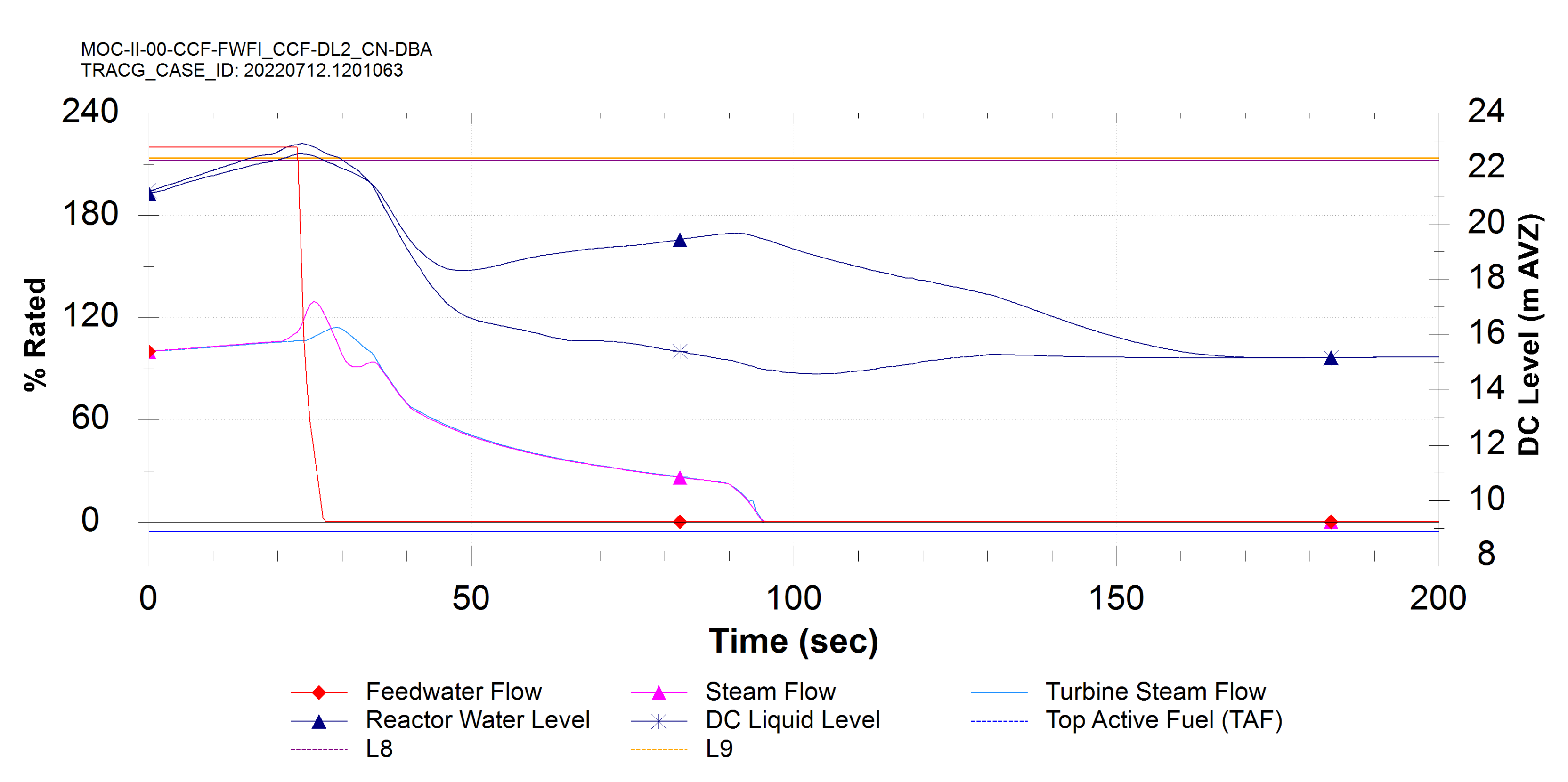

Figure 15.5-80: Feedwater Flow Increase (DBA)

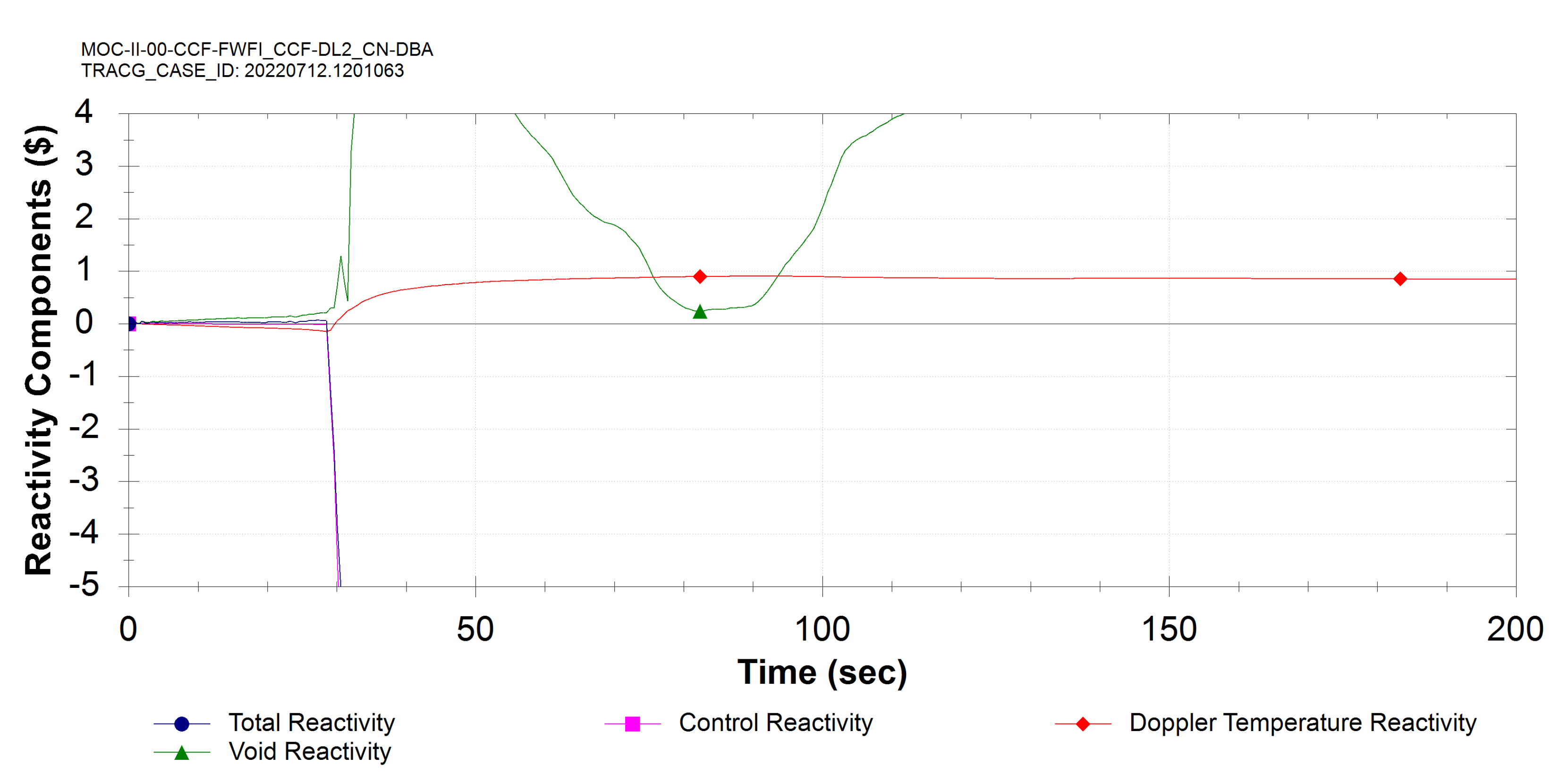

Figure 15.5-81: Feedwater Flow Increase (DBA)

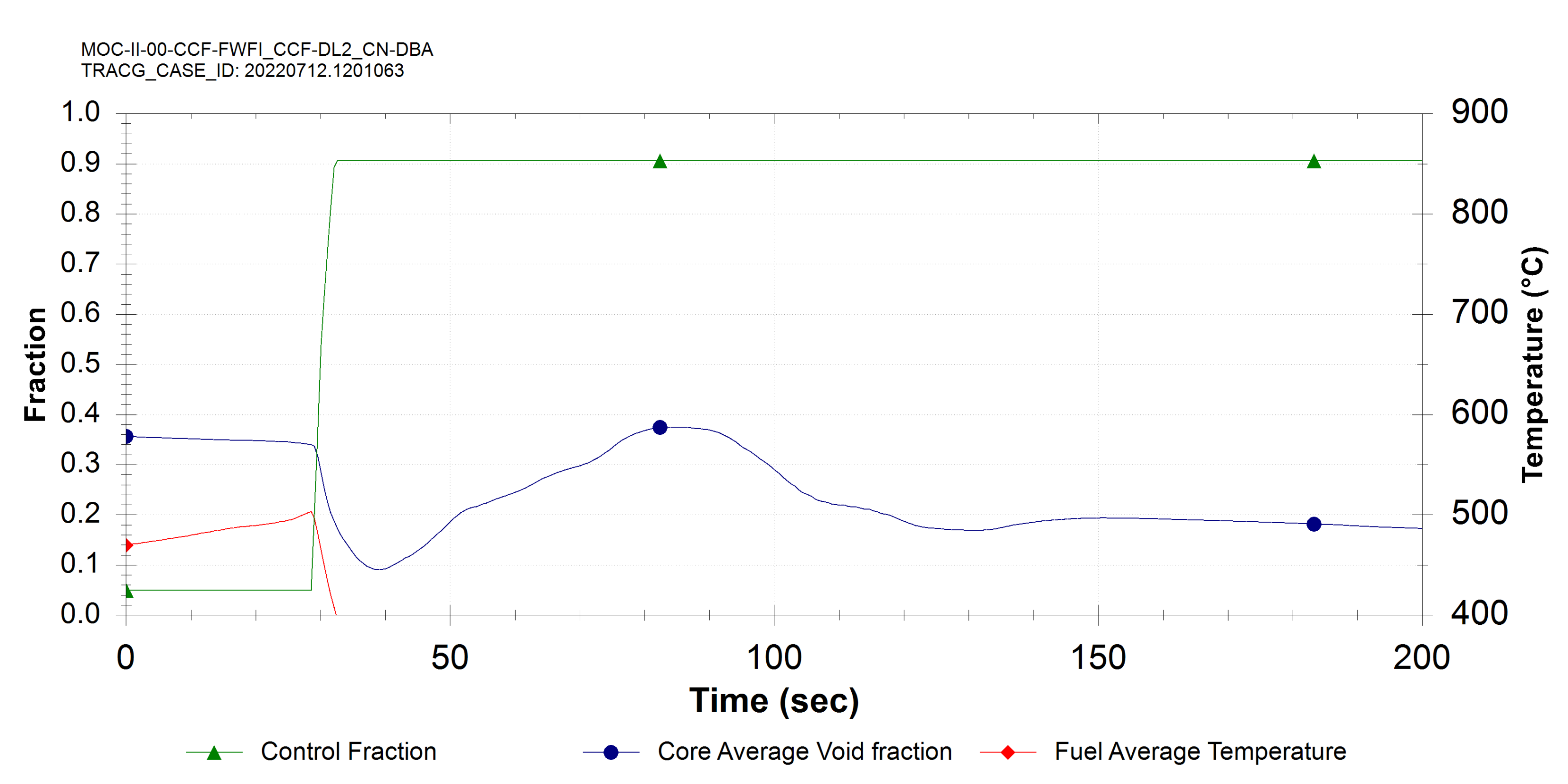

Figure 15.5-82: Feedwater Flow Increase (DBA)

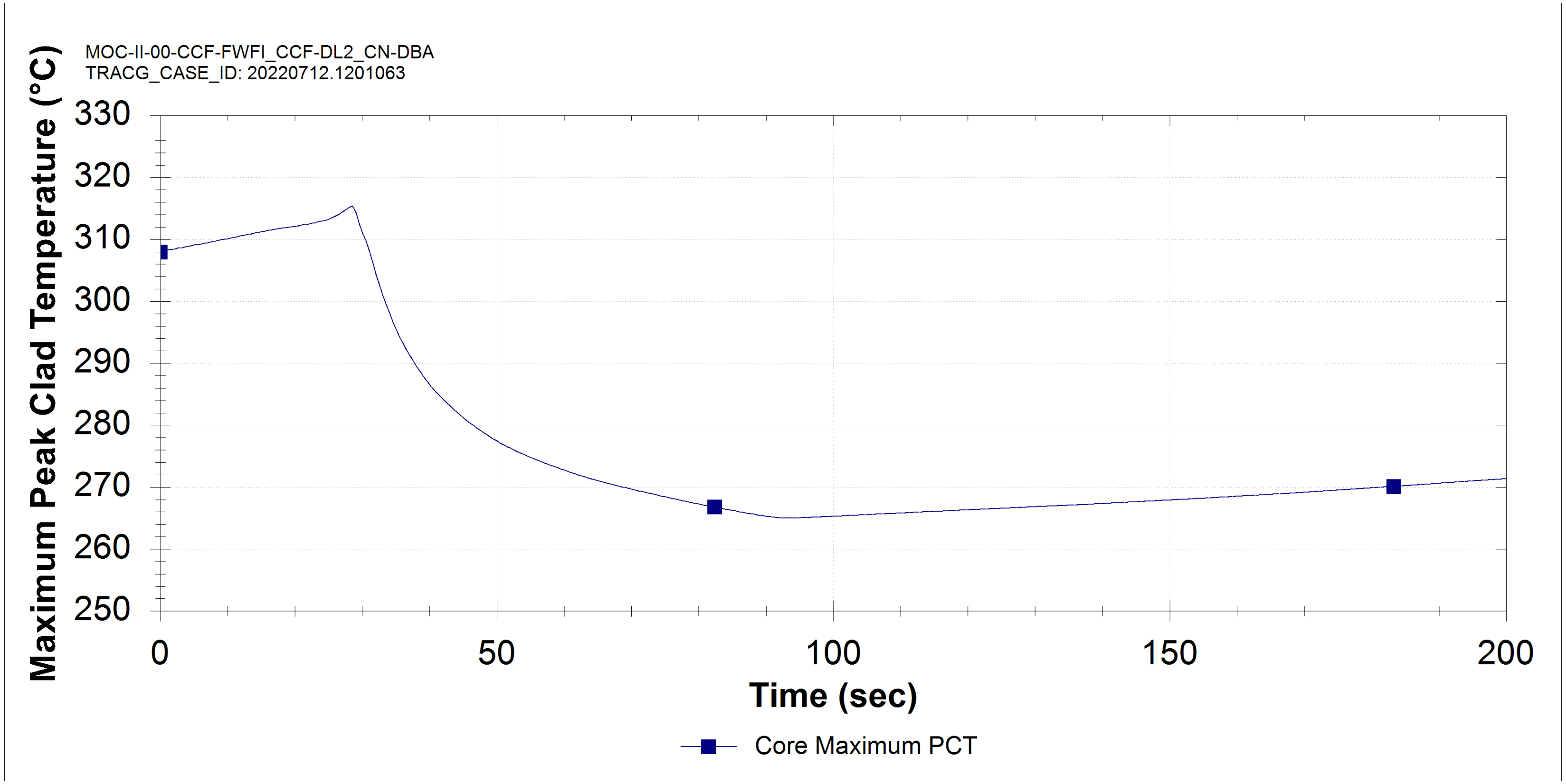

Figure 15.5-83: Feedwater Flow Increase (DBA)

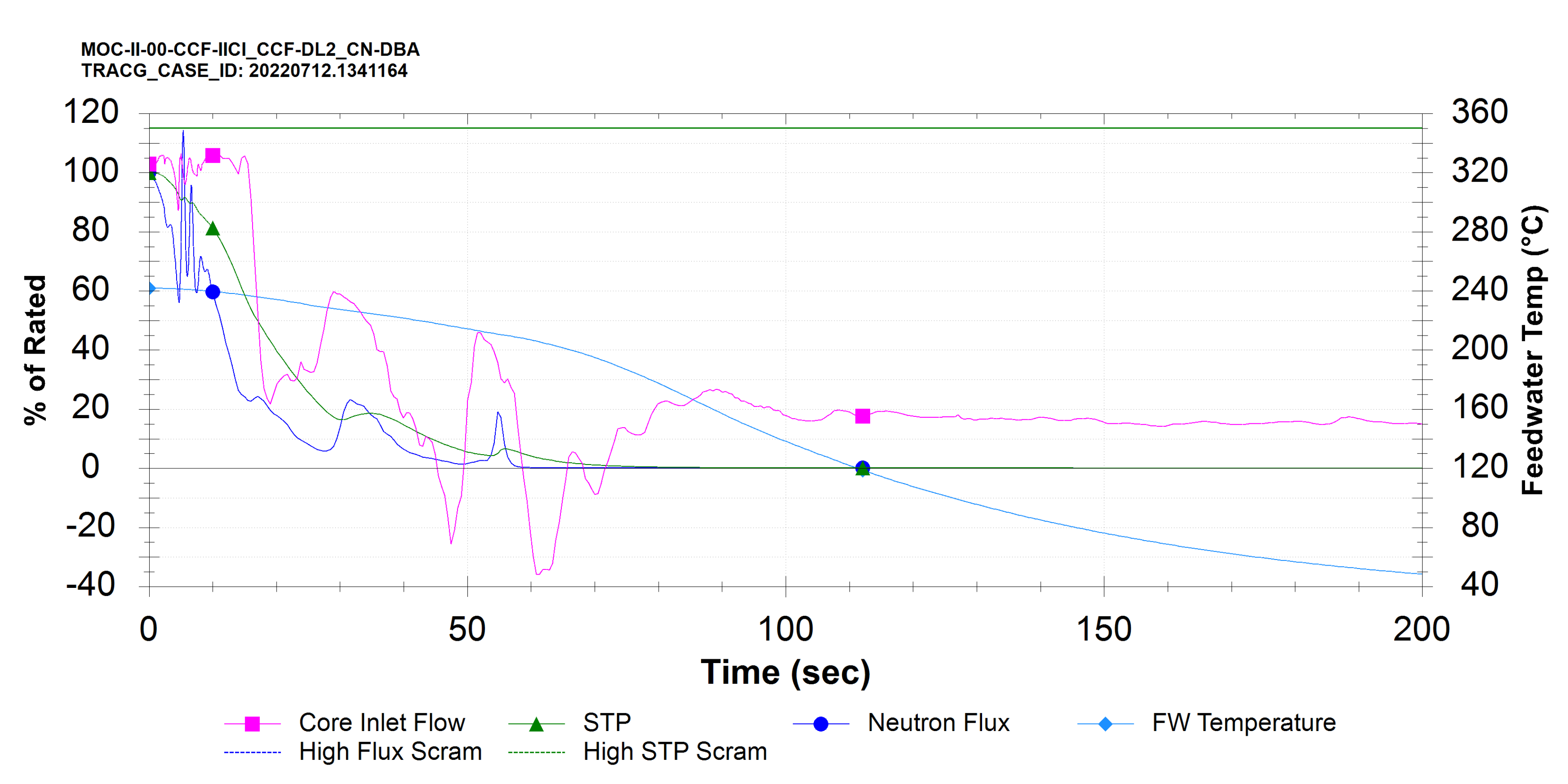

Figure 15.5-84: Inadvertent Isolation Condenser Initiation – All Trains (DBA)

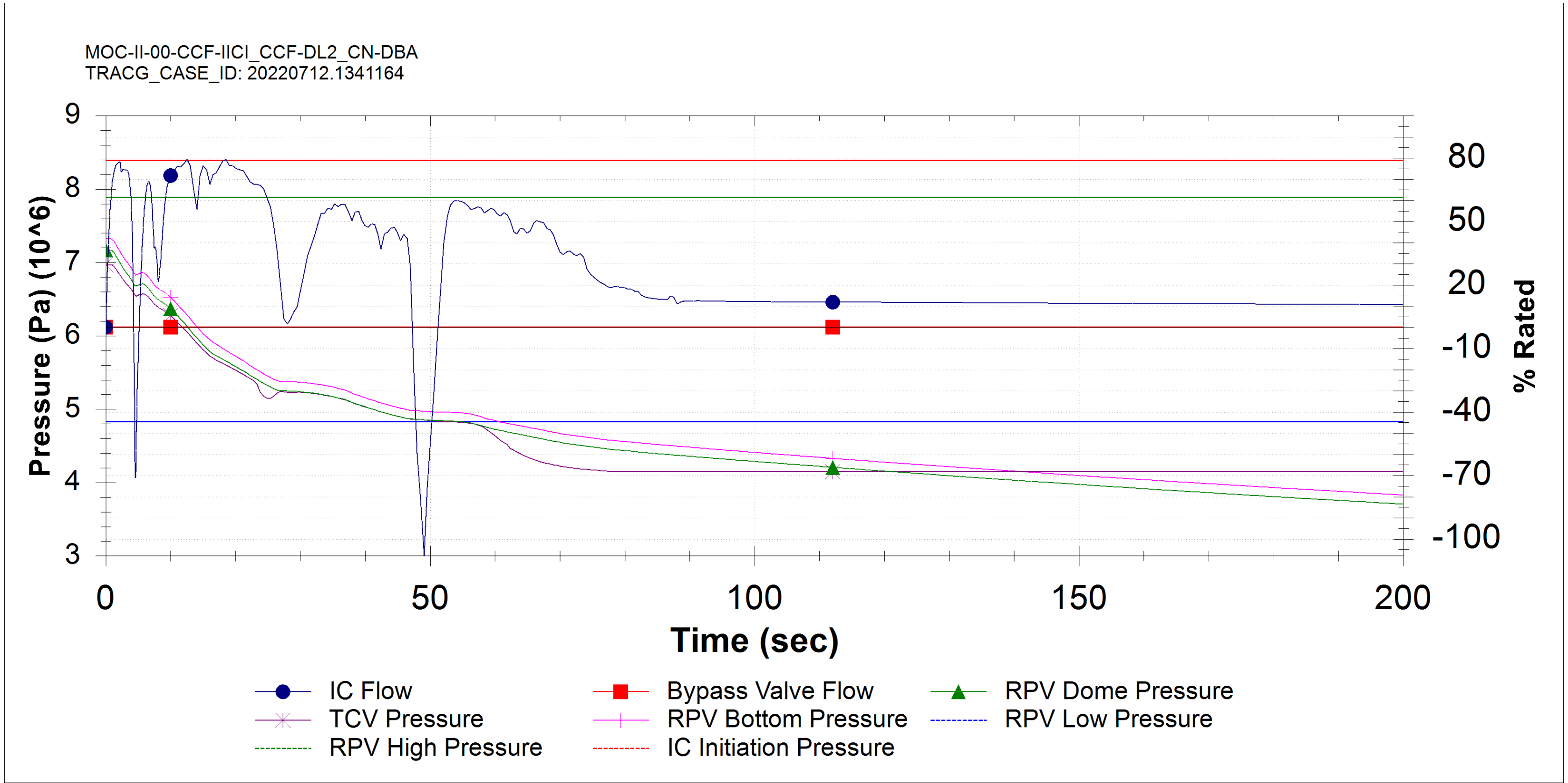

Figure 15.5-85: Inadvertent Isolation Condenser Initiation – All Trains (DBA)

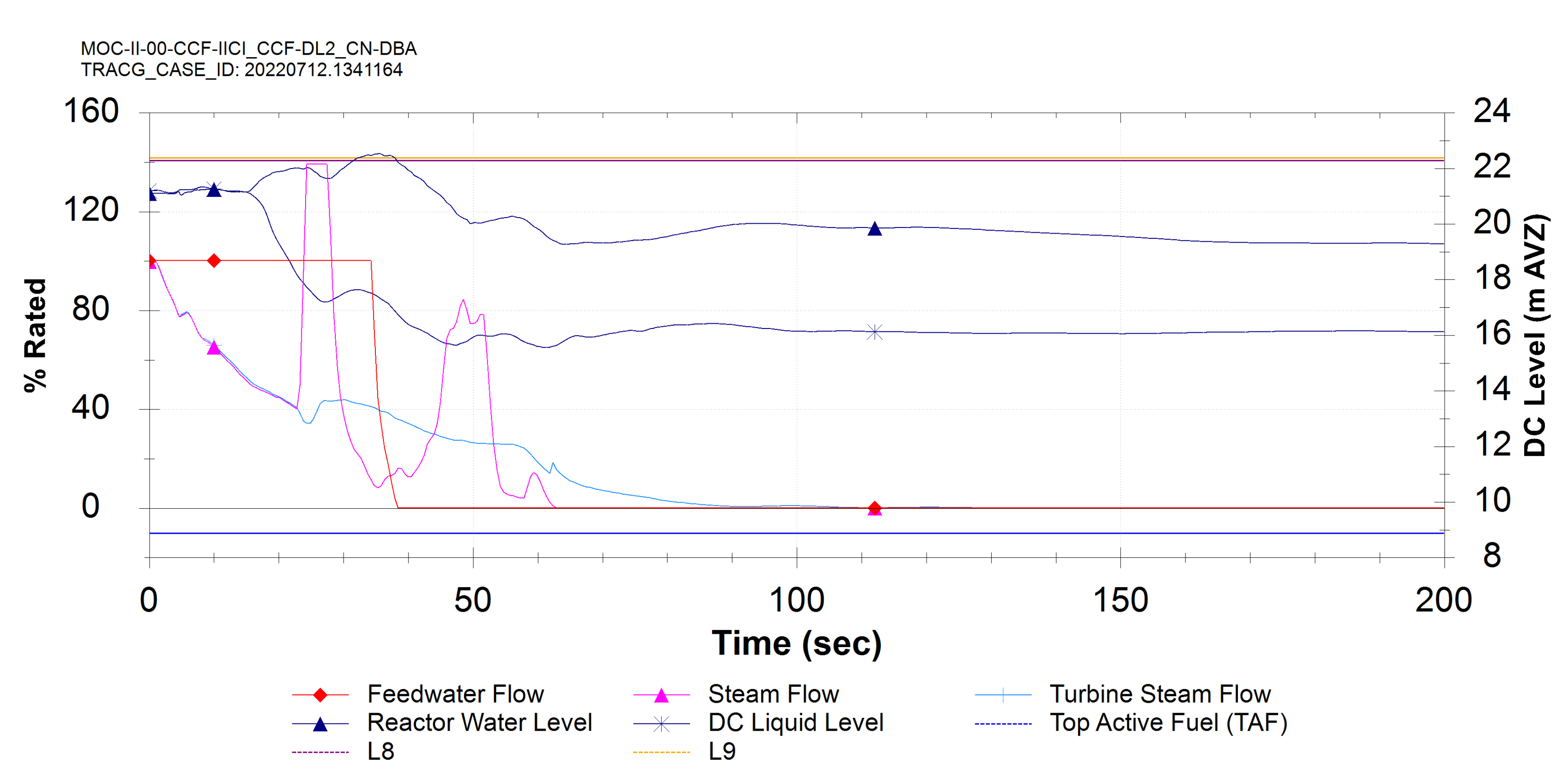

Figure 15.5-86: Inadvertent Isolation Condenser Initiation – All Trains (DBA)

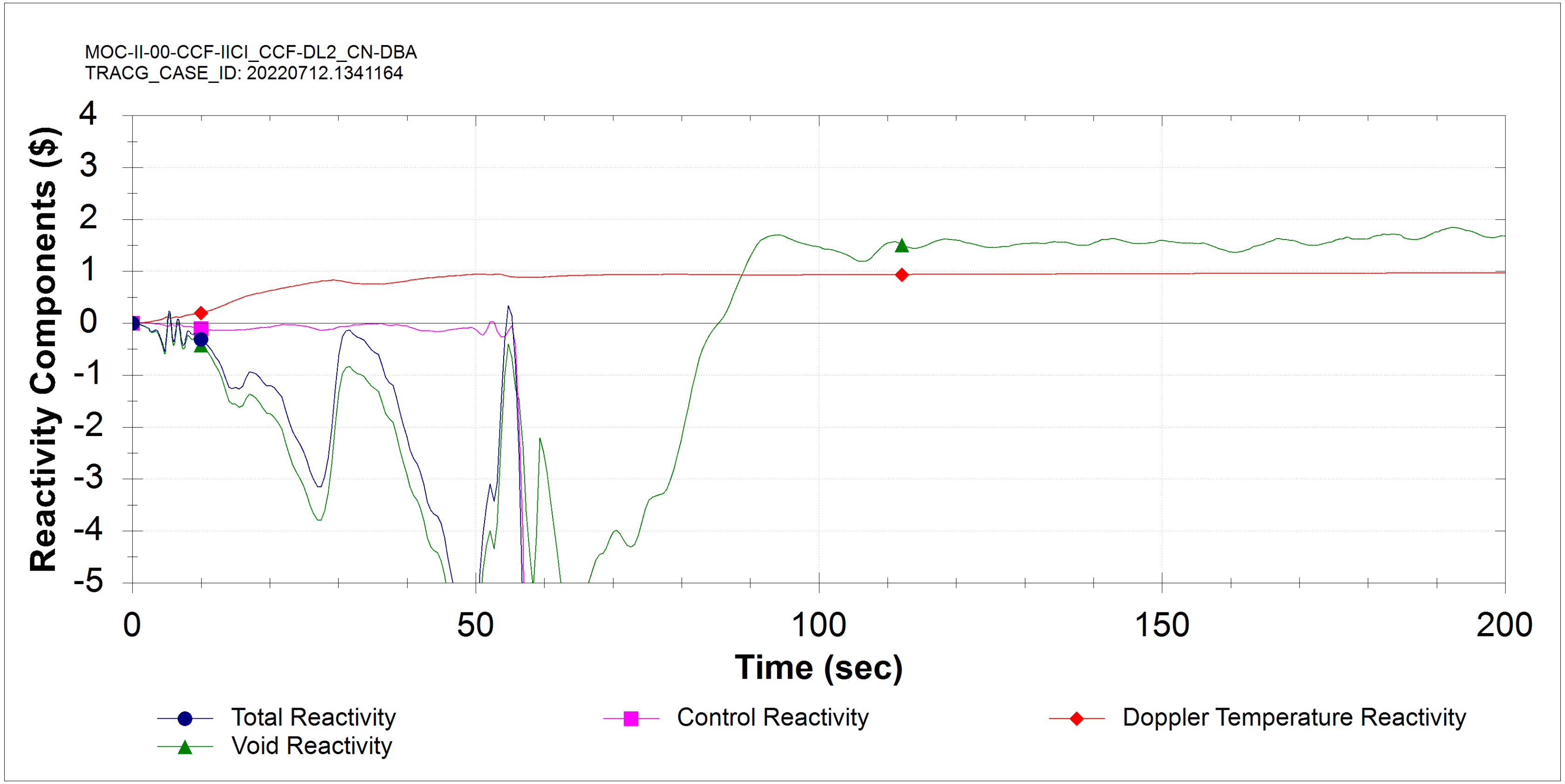

Figure 15.5-87: Inadvertent Isolation Condenser Initiation – All Trains (DBA)

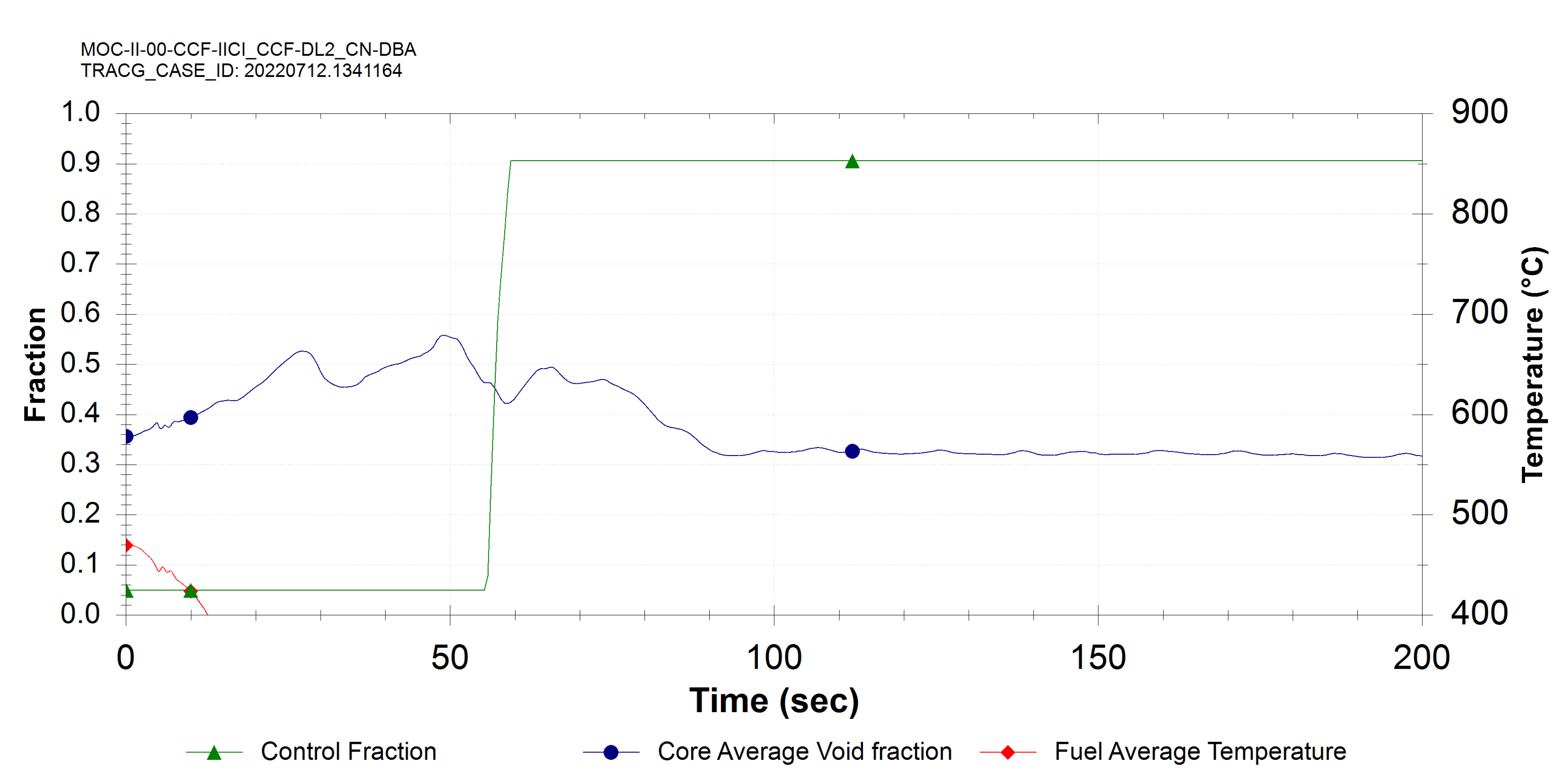

Figure 15.5-88: Inadvertent Isolation Condenser Initiation – All Trains (DBA)

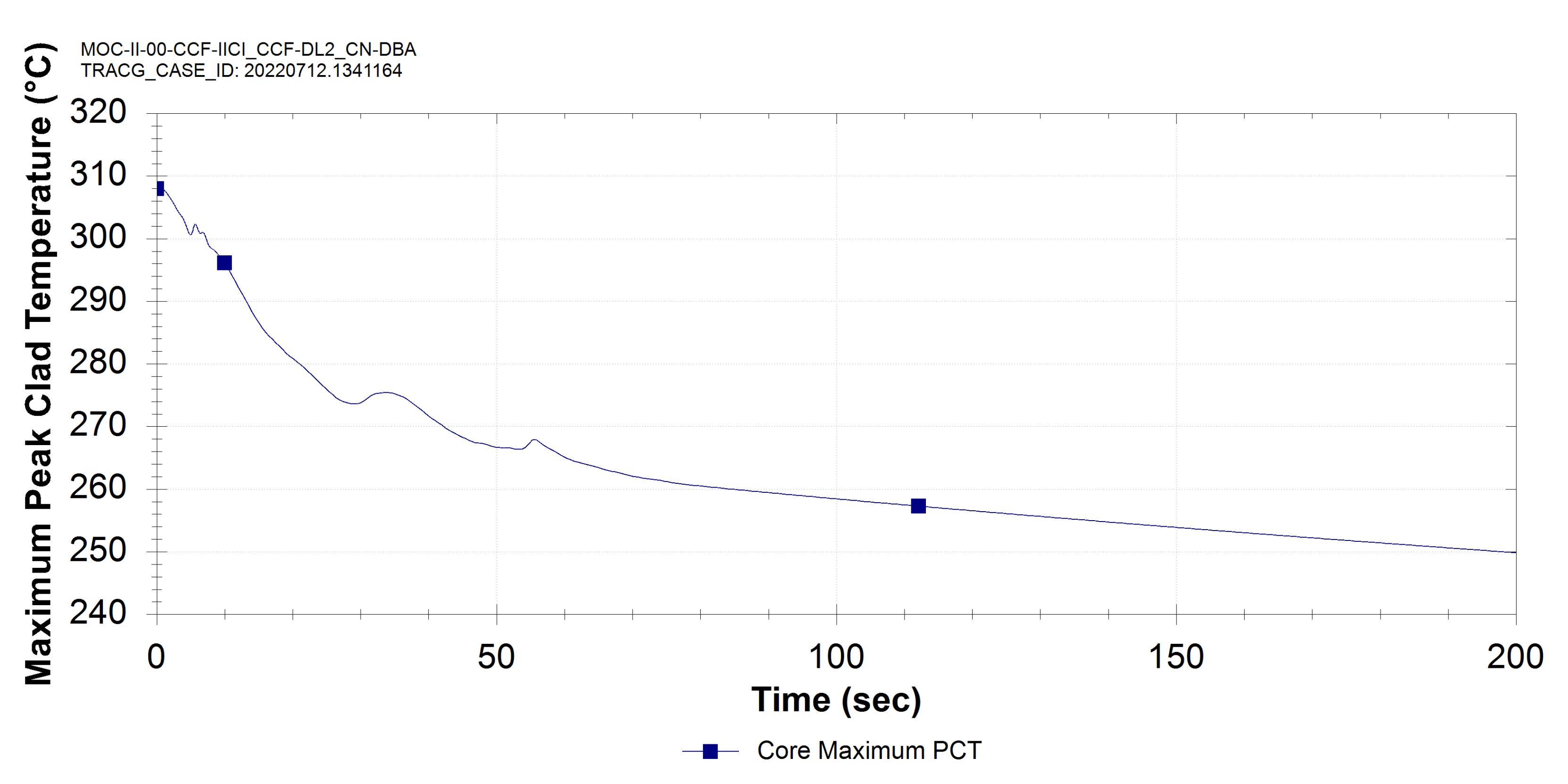

Figure 15.5-89: Inadvertent Isolation Condenser Initiation – All Trains (DBA)

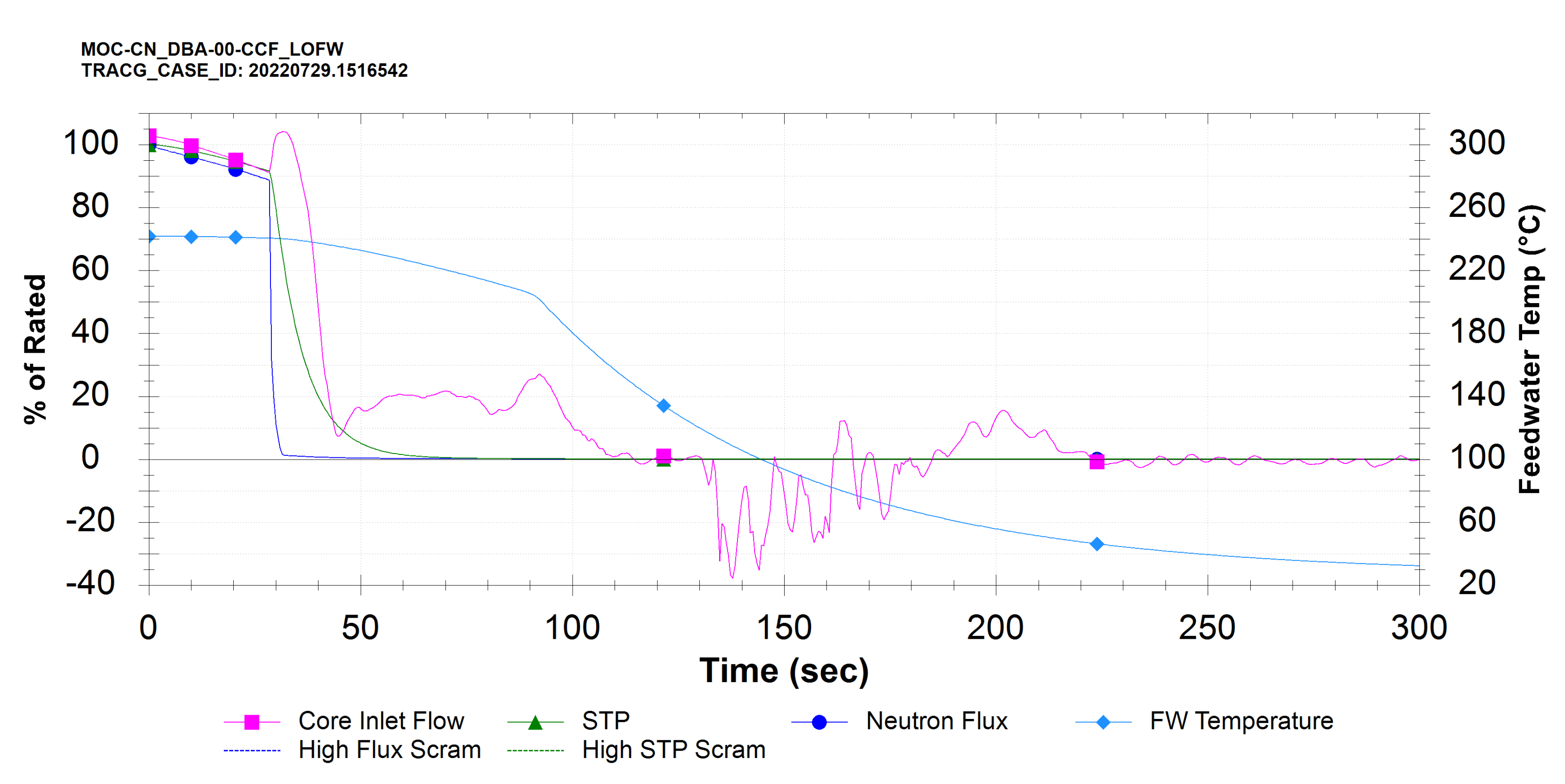

Figure 15.5-90: Loss of Feedwater Flow (DBA)

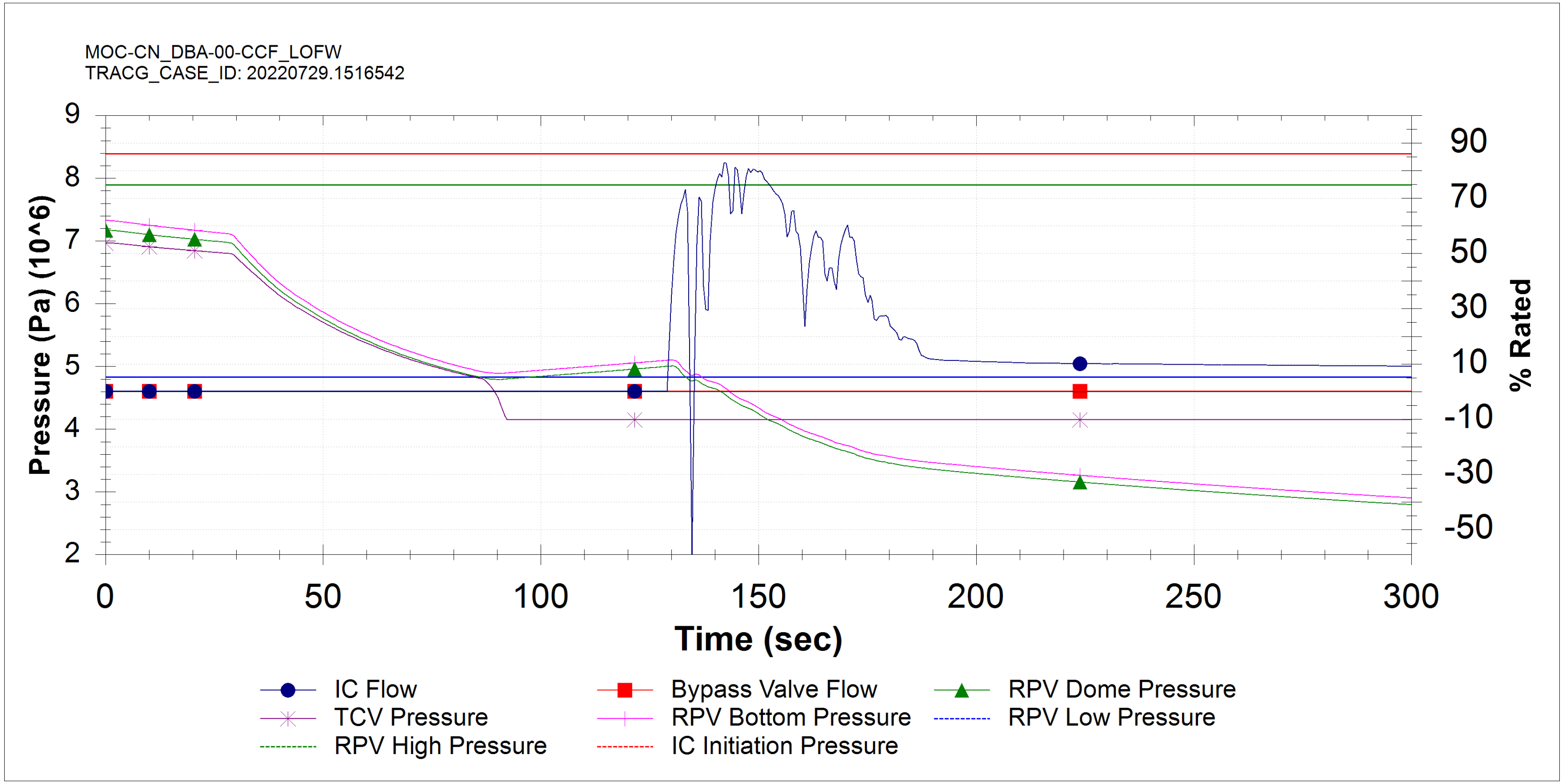

Figure 15.5-91: Loss of Feedwater Flow (DBA)

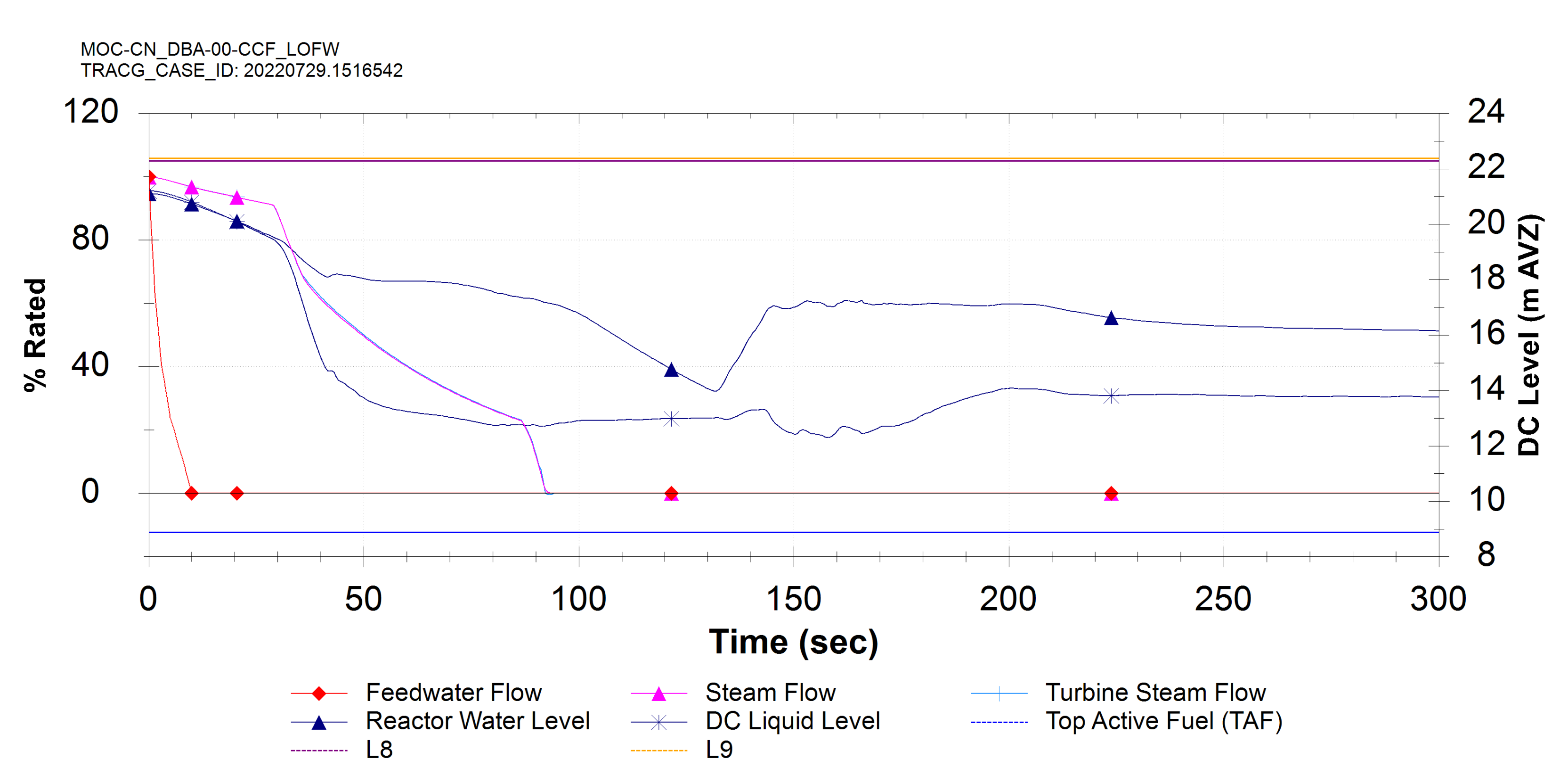

Figure 15.5-92: Loss of Feedwater Flow (DBA)

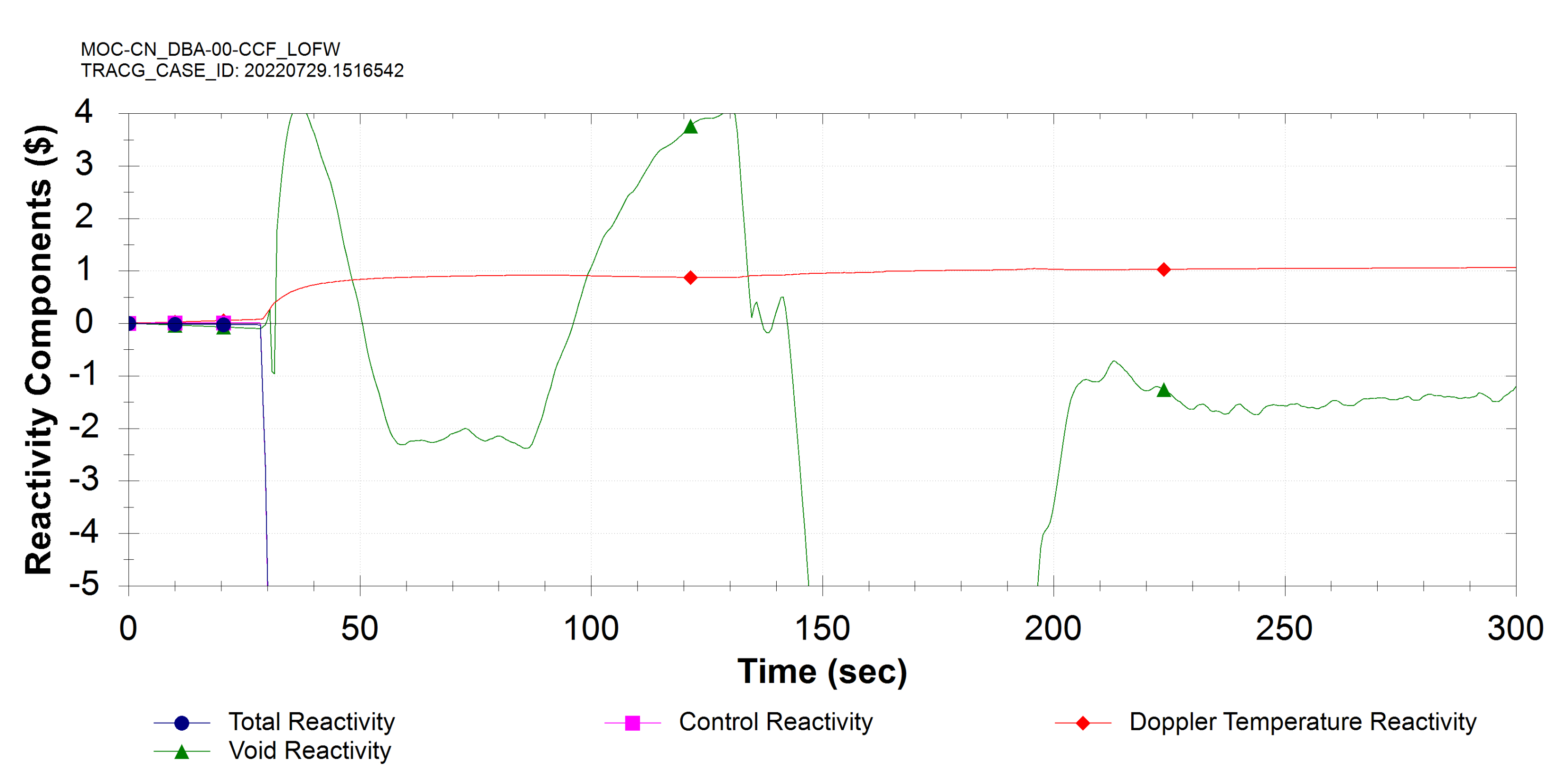

Figure 15.5-93: Loss of Feedwater Flow (DBA)

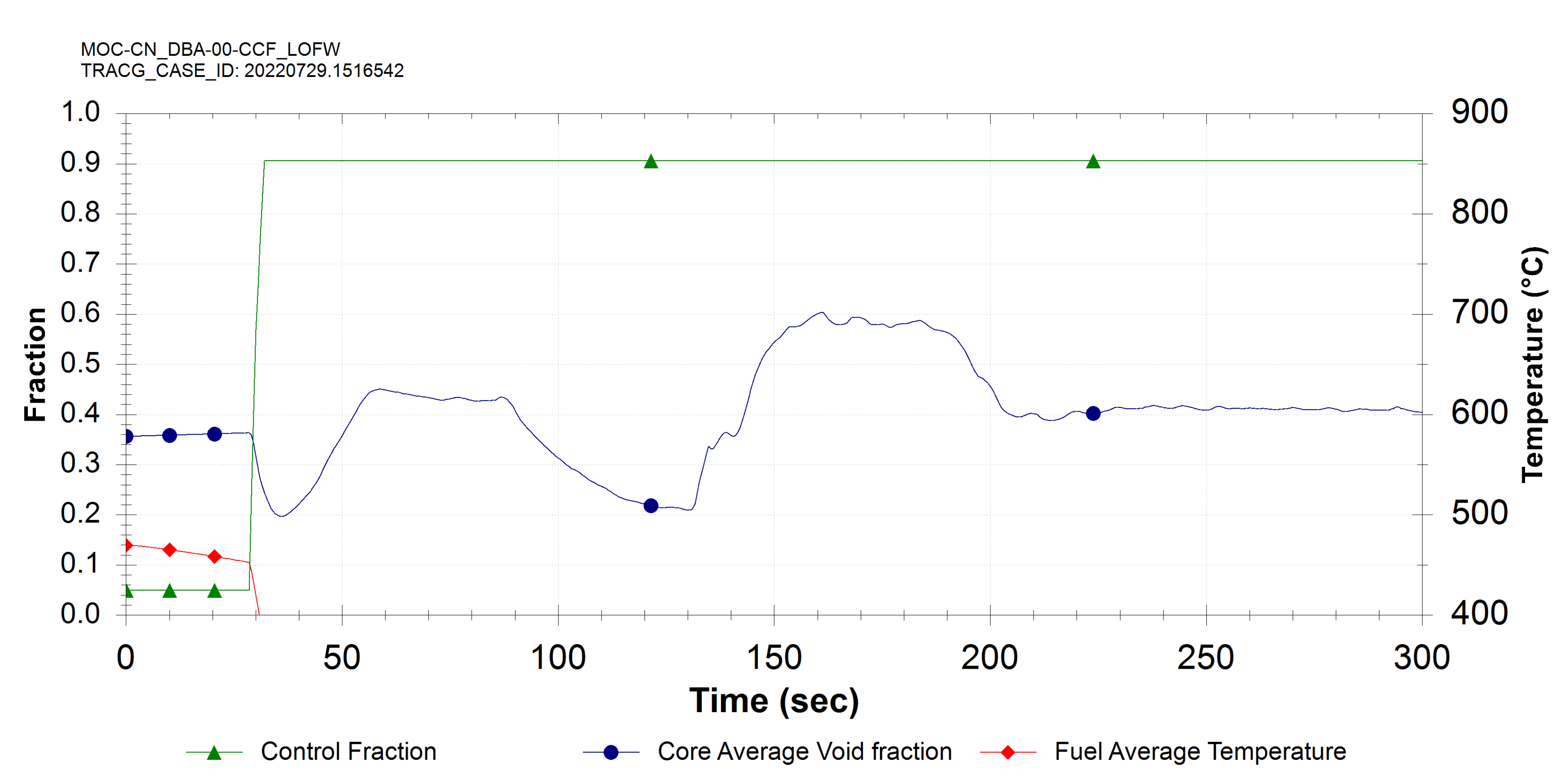

Figure 15.5-94: Loss of Feedwater Flow (DBA)

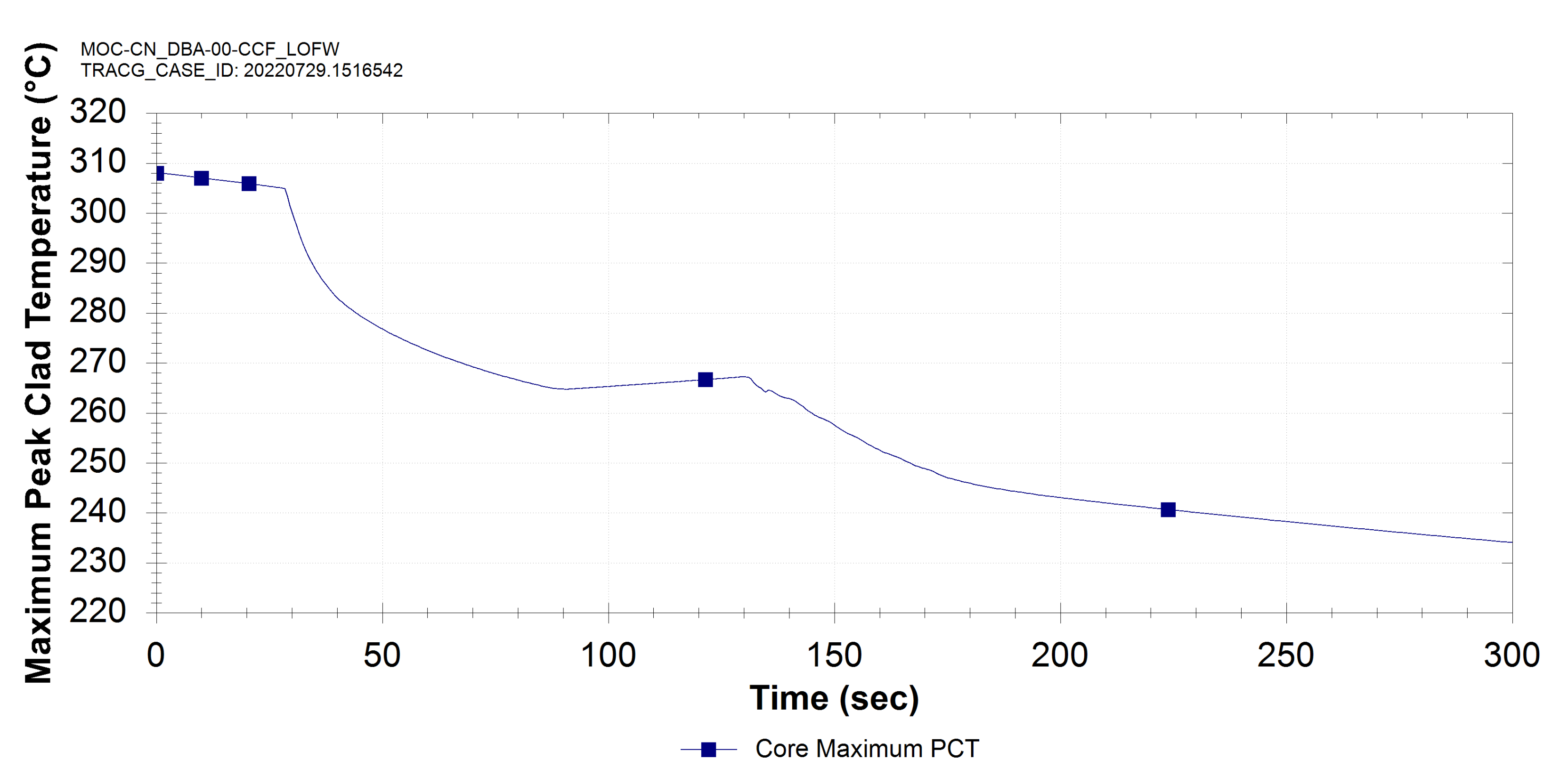

Figure 15.5-95: Loss of Feedwater Flow (DBA)

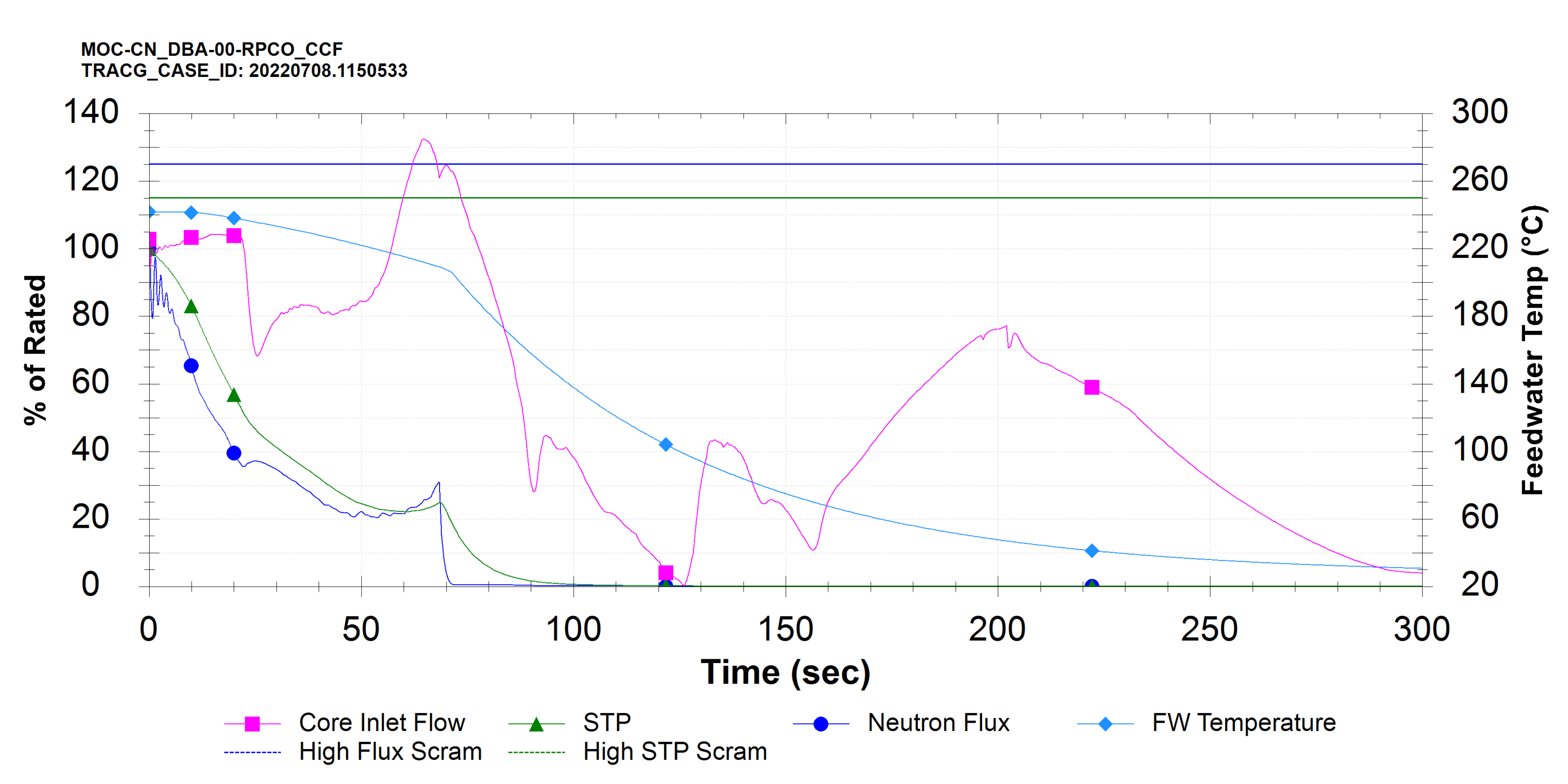

Figure 15.5-96: RPV Pressure Control Open (DBA)

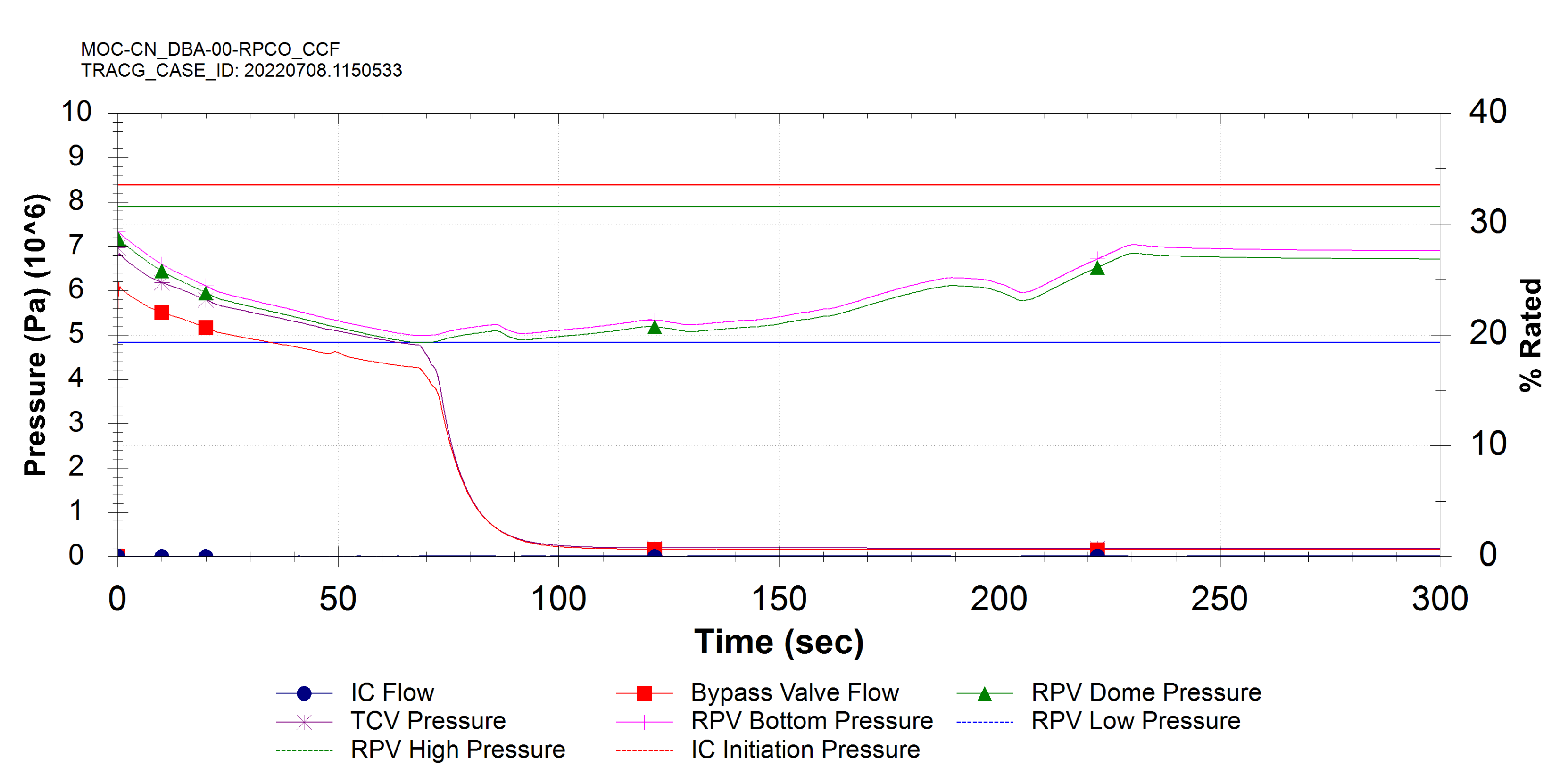

Figure 15.5-97: RPV Pressure Control Open (DBA)

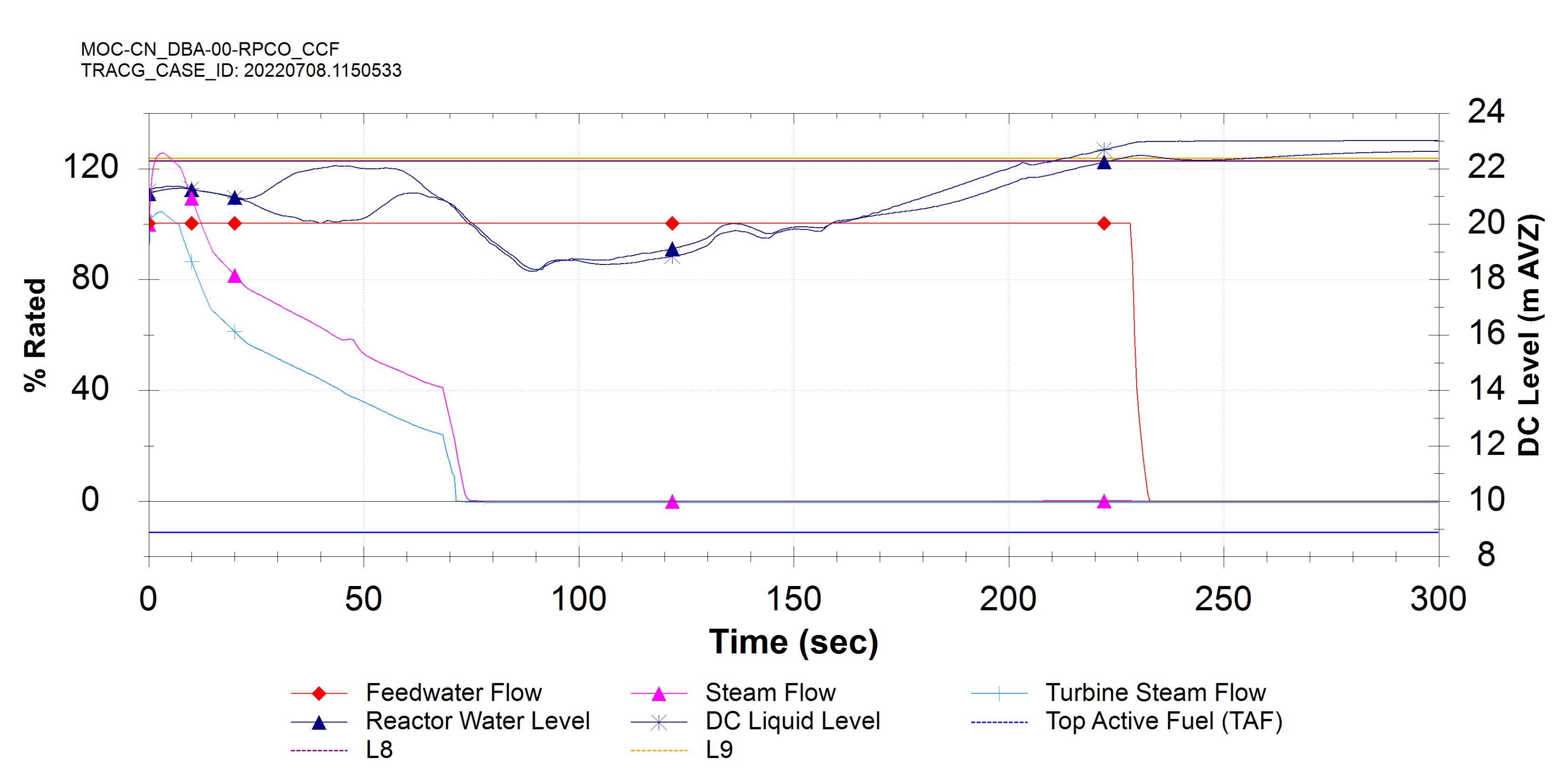

Figure 15.5-98: RPV Pressure Control Open (DBA)

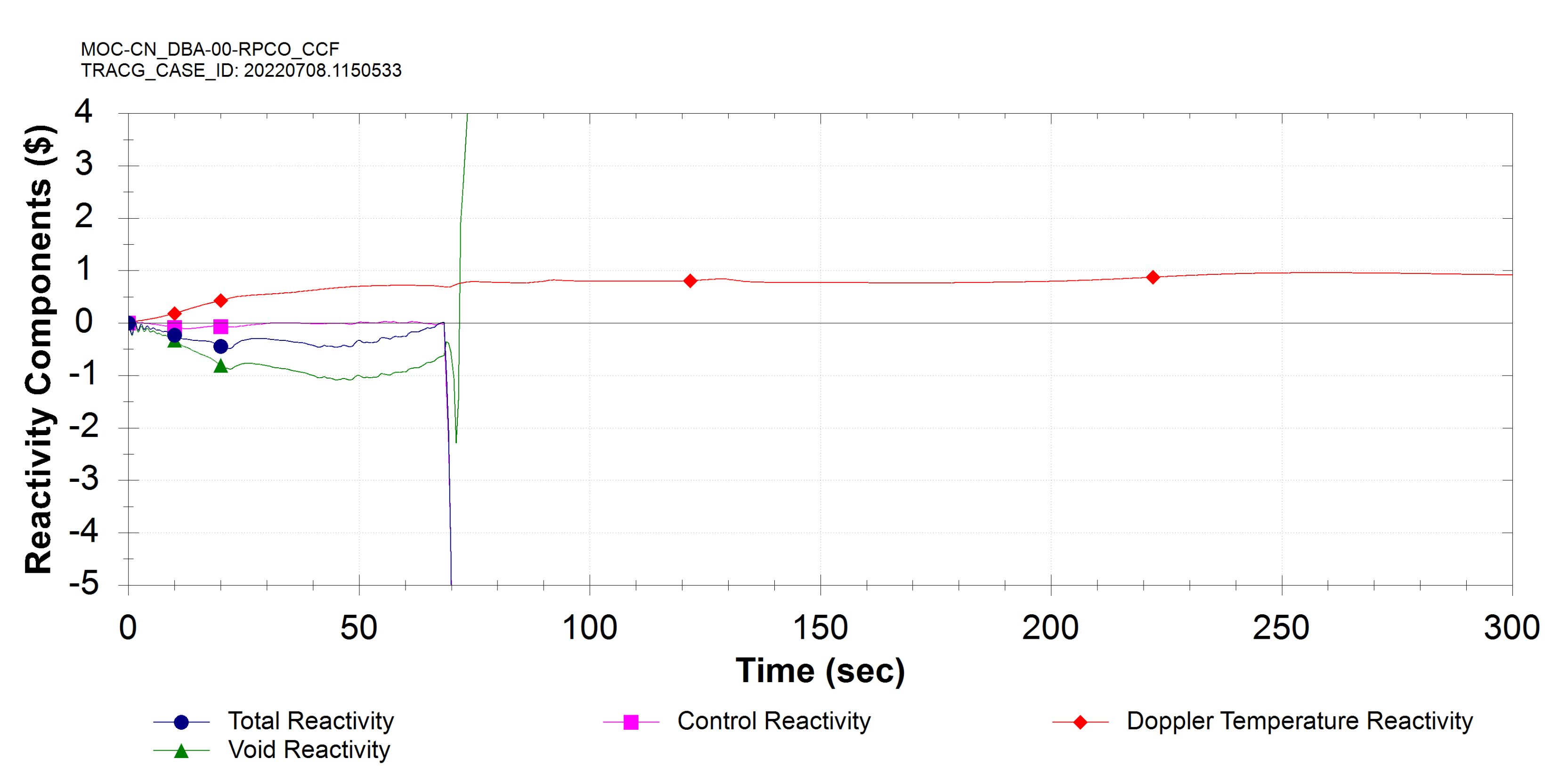

Figure 15.5-99: RPV Pressure Control Open (DBA)

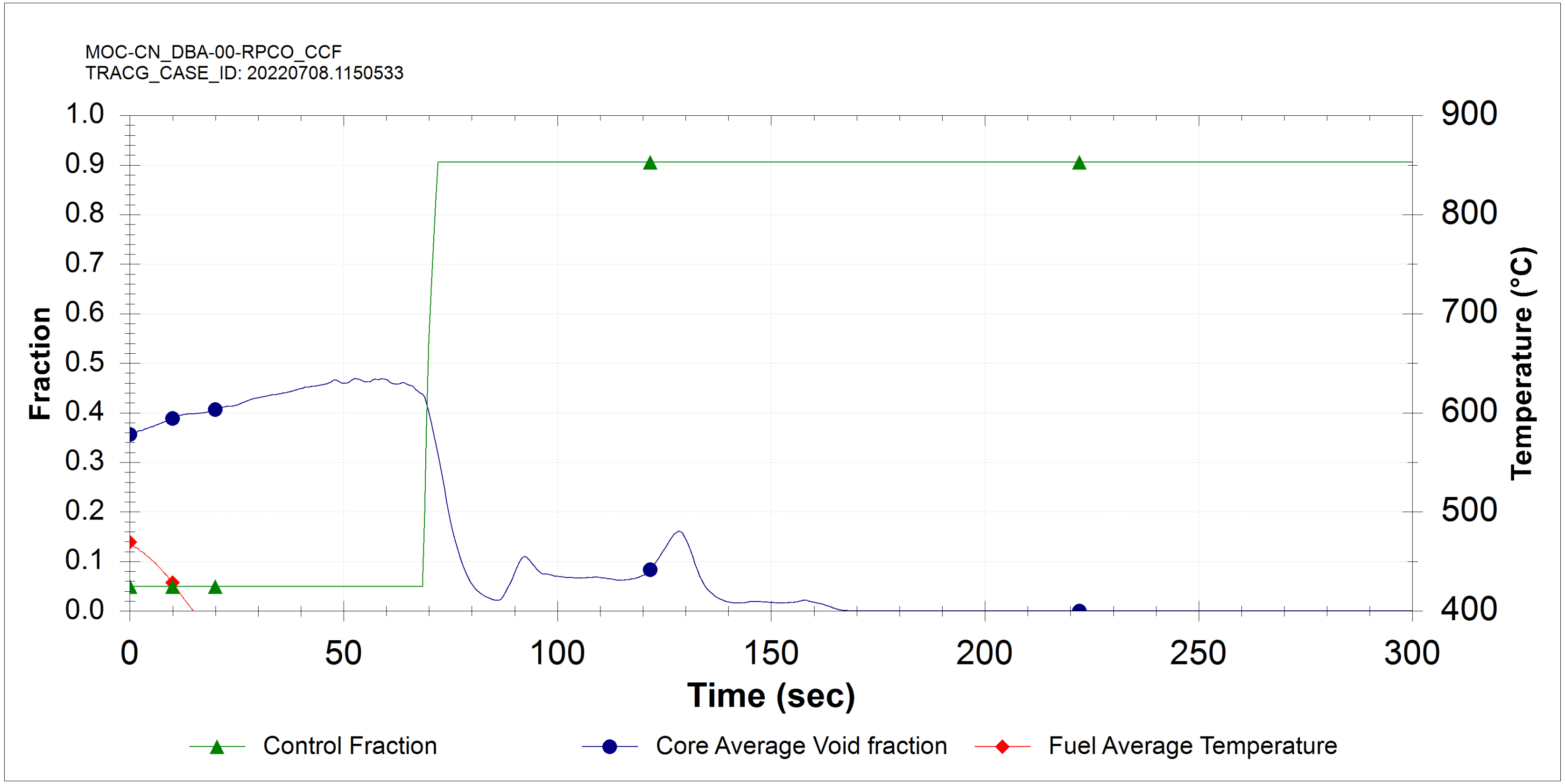

Figure 15.5-100: RPV Pressure Control Open (DBA)

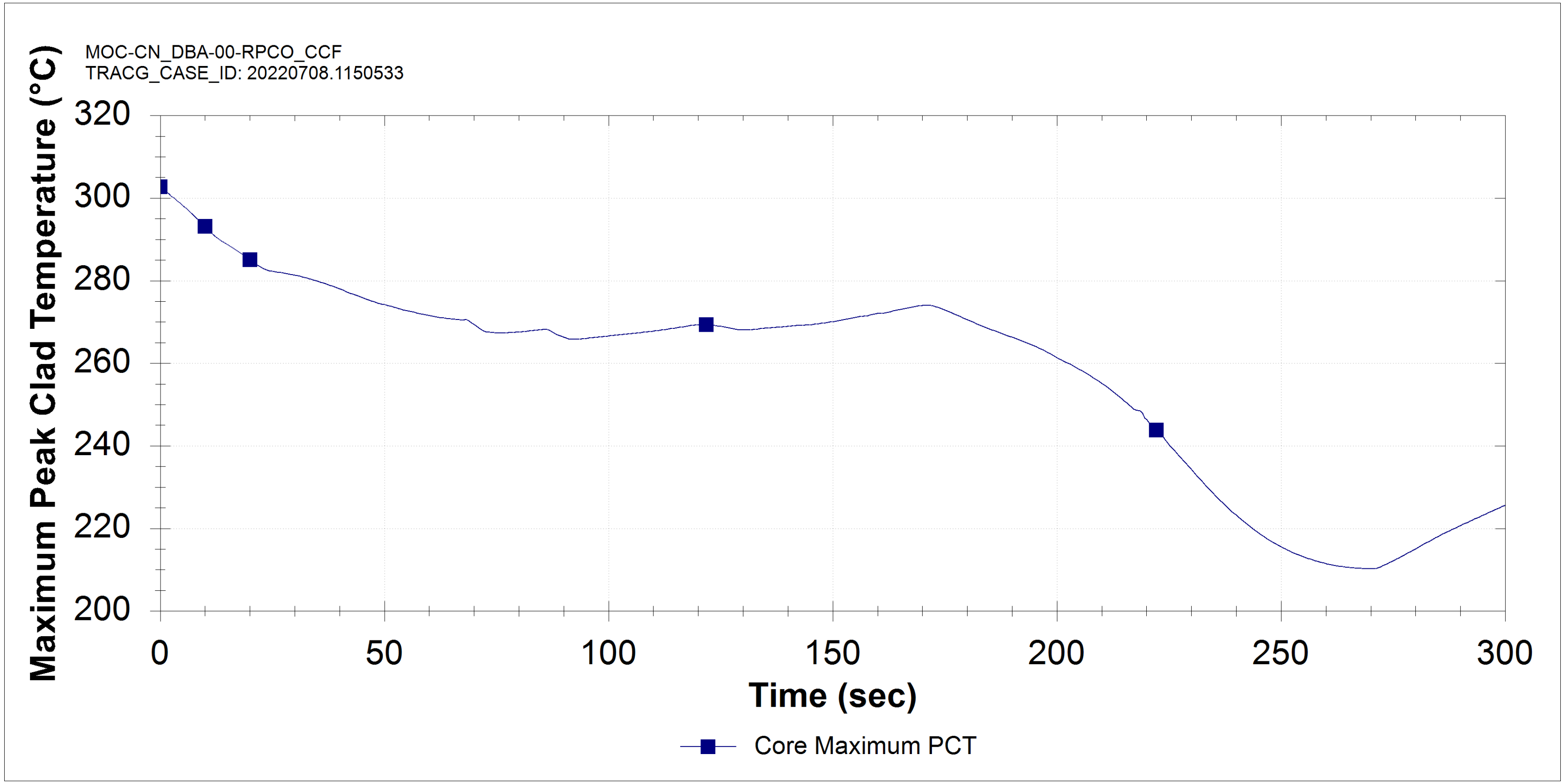

Figure 15.5-101: RPV Pressure Control Open (DBA)

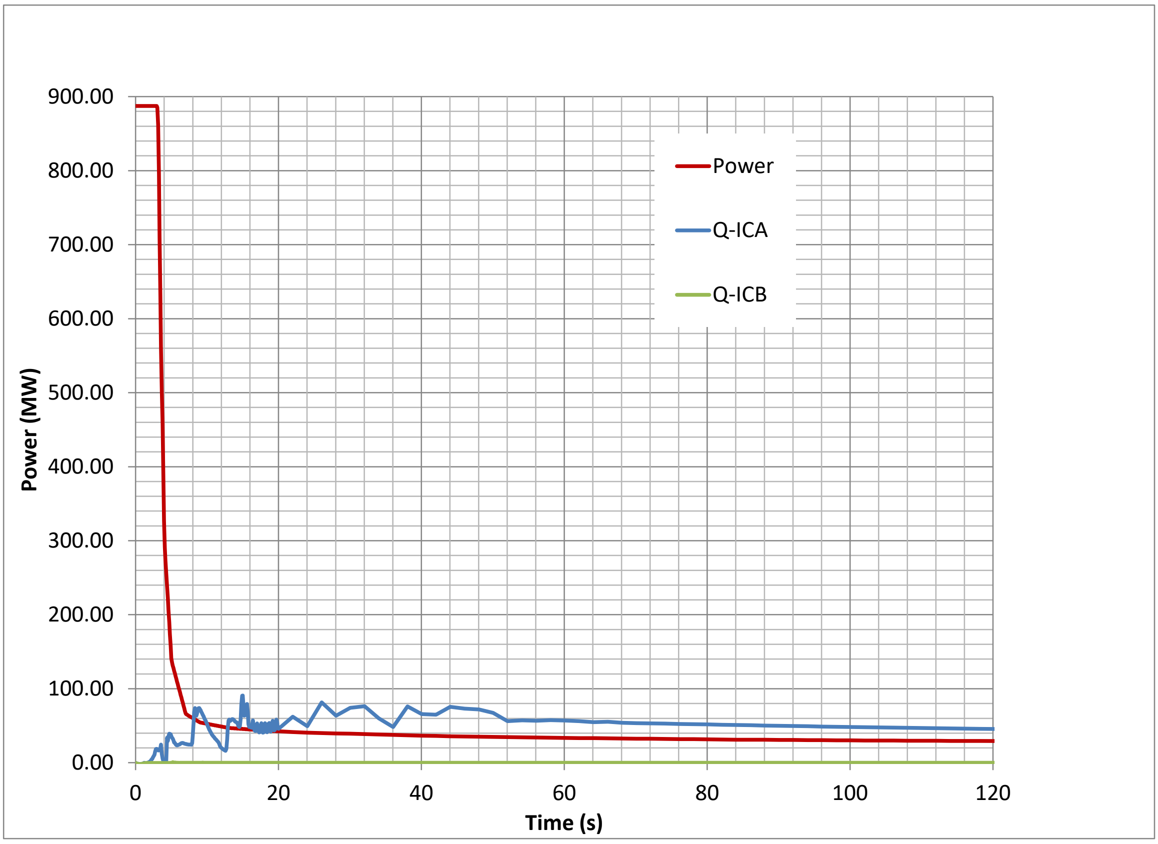

Figure 15.5-102: Reactor Power, Large Main Steam Pipe Break, Conservative Case

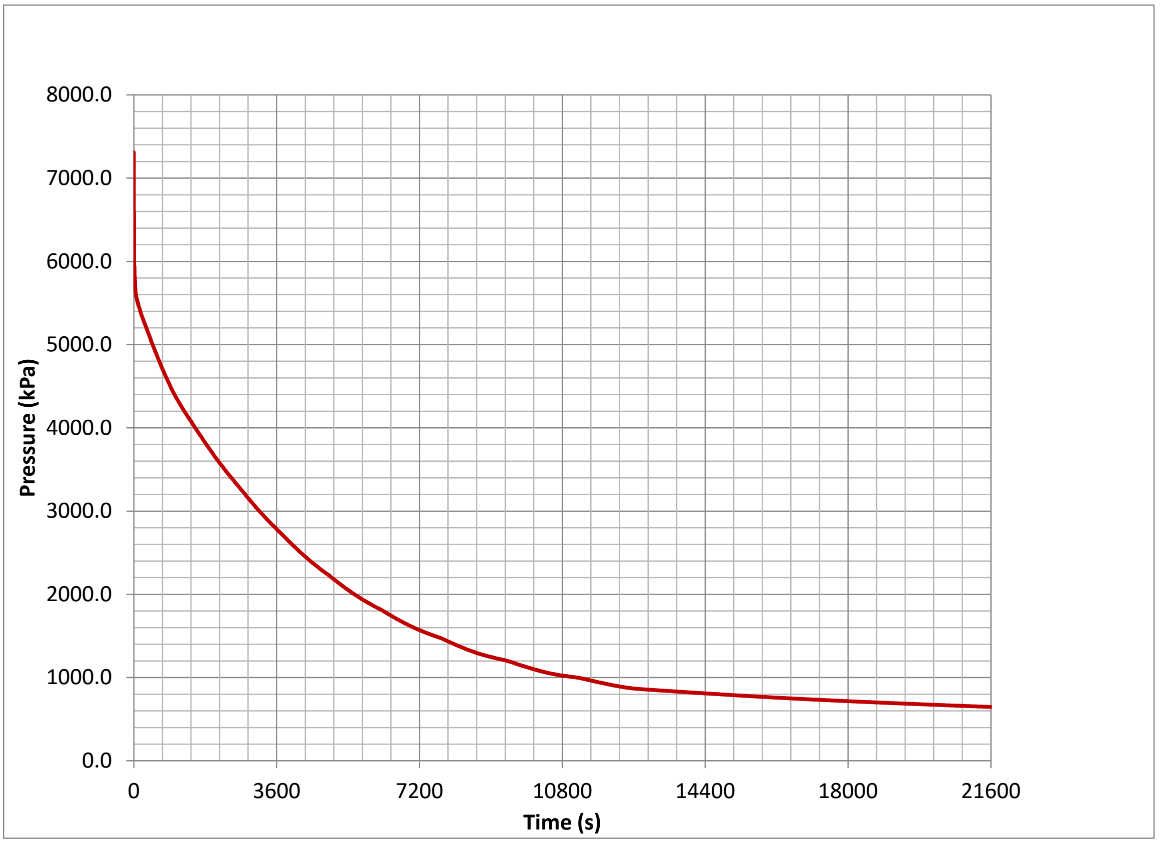

Figure 15.5-103: Reactor Pressure, Large Main Steam Pipe Break, Conservative Case

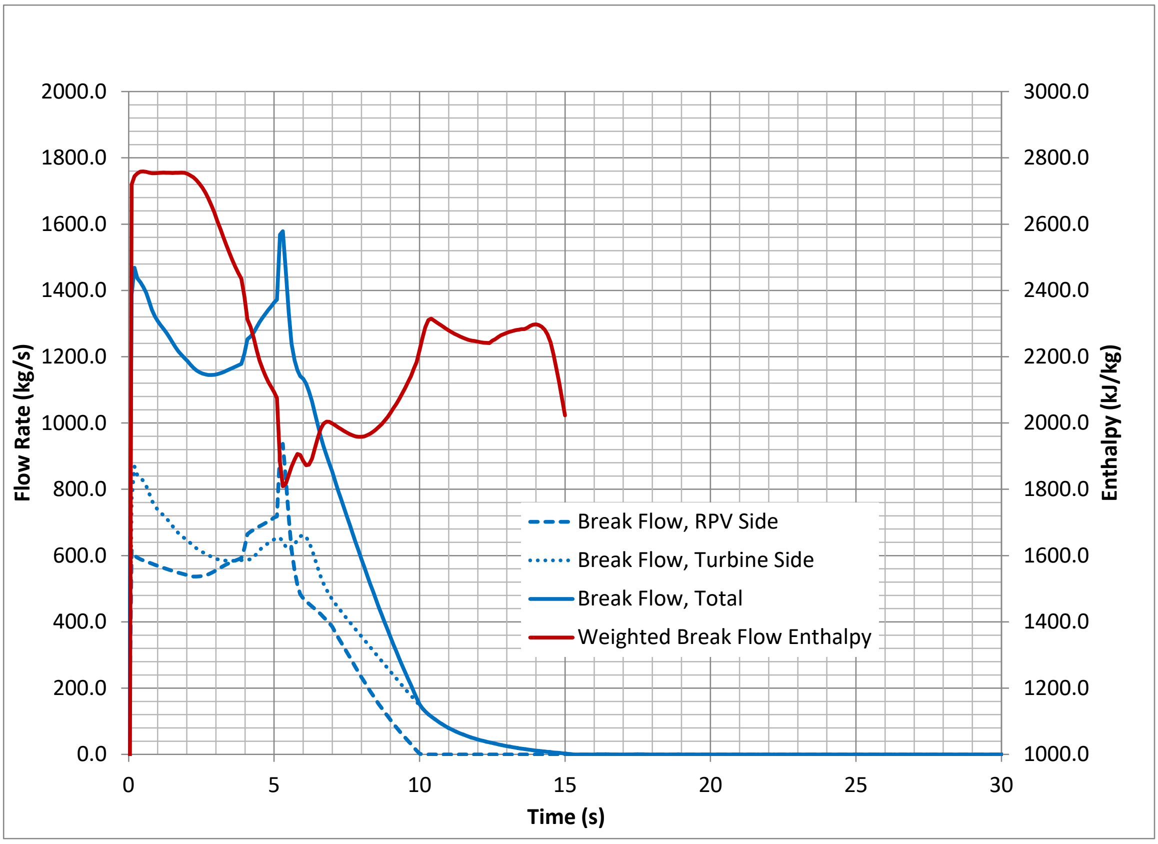

Figure 15.5-104: Break Flow Rate and Enthalpy, Large Main Steam Pipe Break, Conservative Case

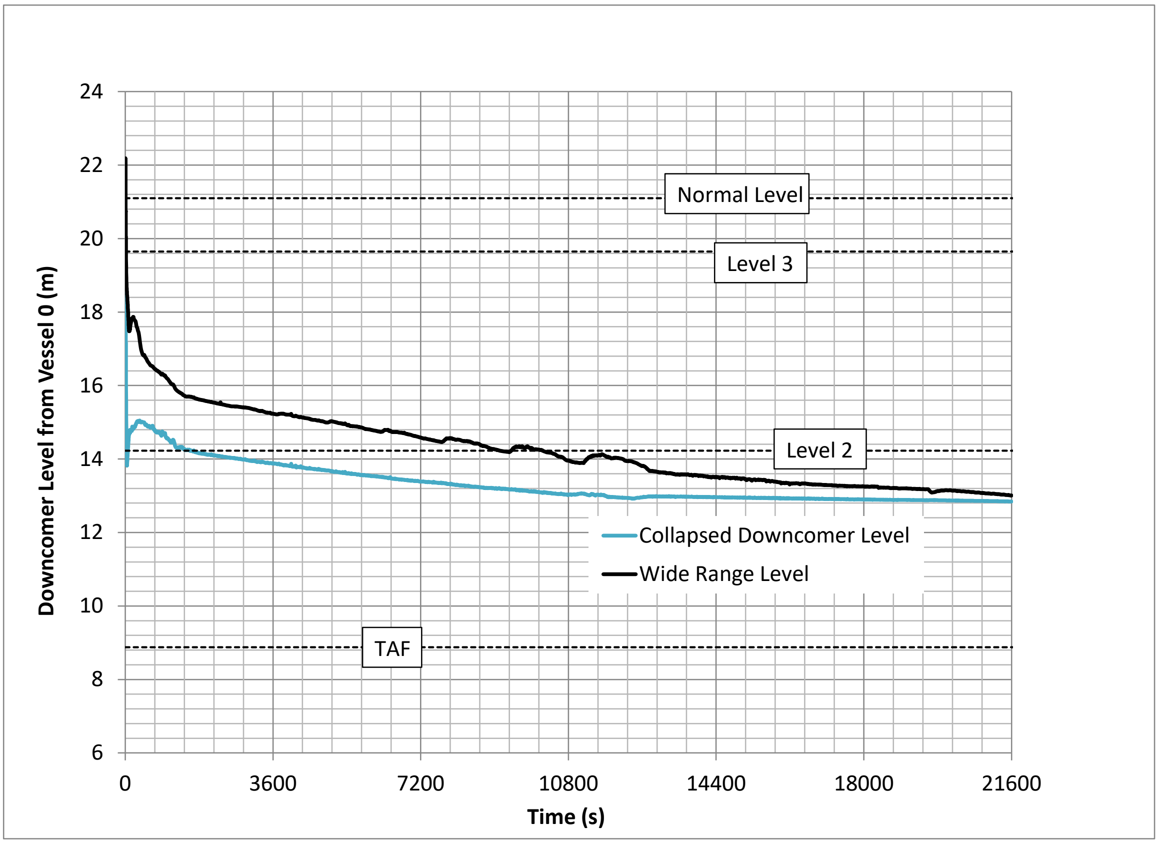

Figure 15.5-105: Reactor Water Level, Large Main Steam Pipe Break, Conservative Case

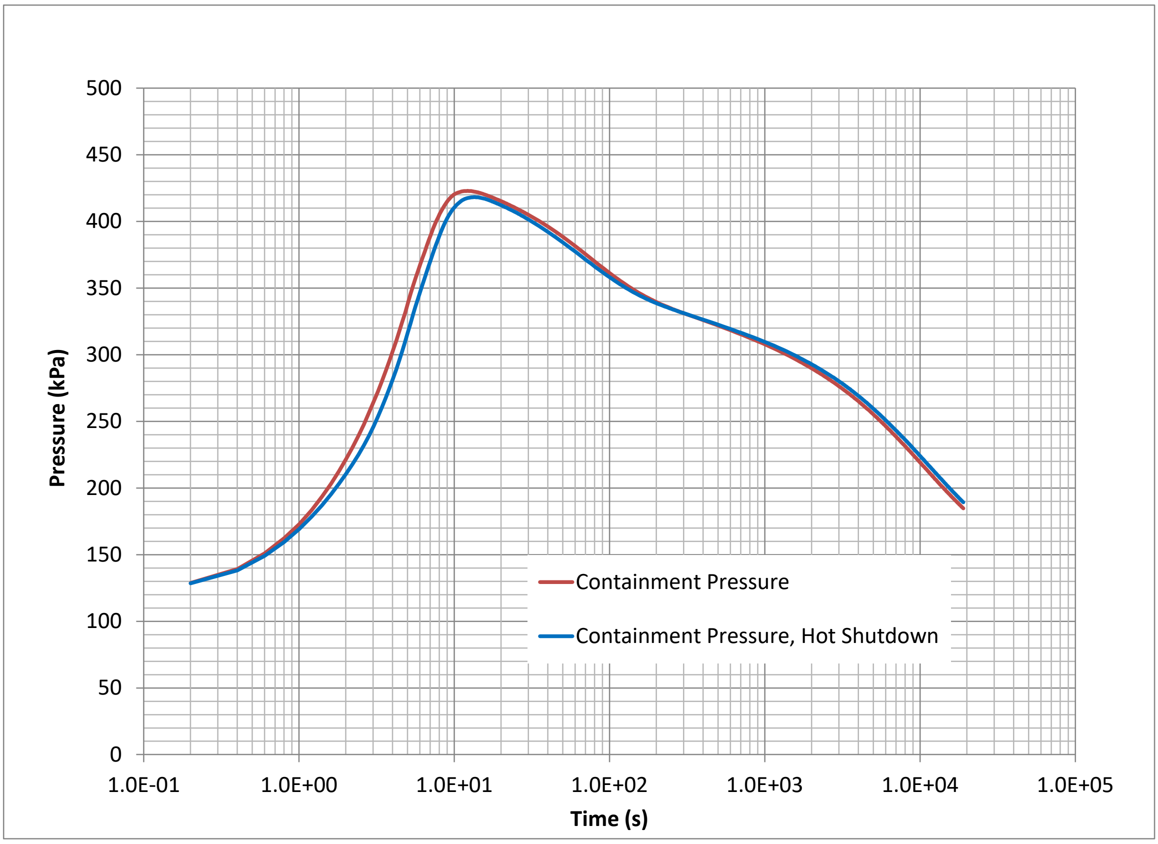

Figure 15.5-106: Containment Pressure, Large Main Steam Pipe Break, Conservative Case

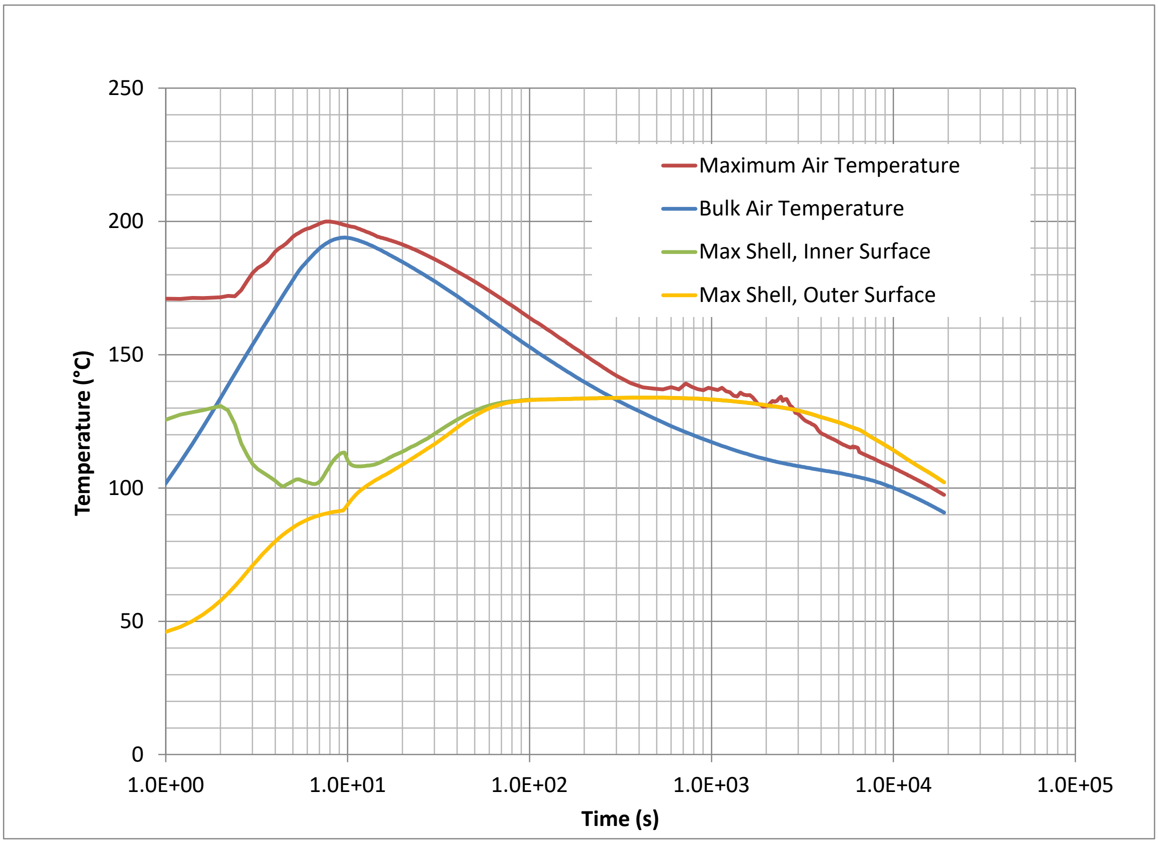

Figure 15.5-107: Containment Temperatures, Large Main Steam Pipe Break, Conservative Case

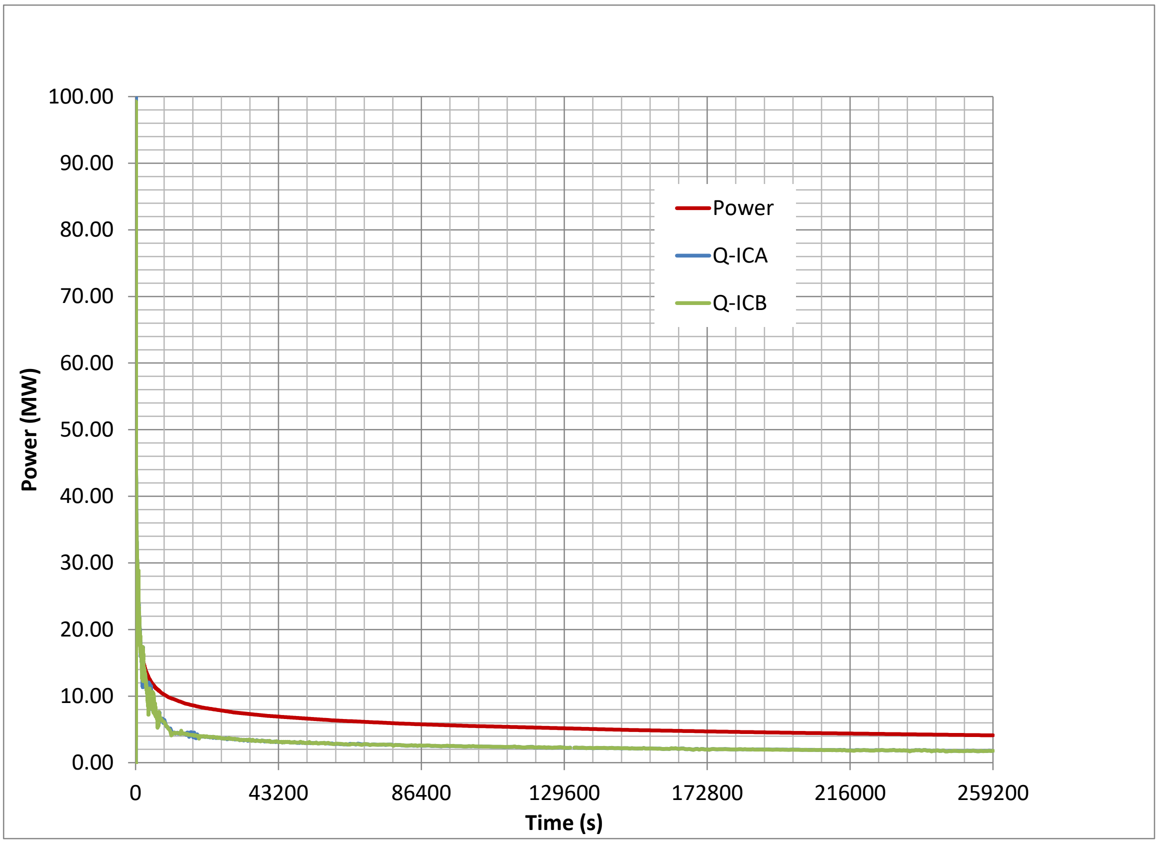

Figure 15.5-108: Reactor Power, Small Steam Break With LOPP, Conservative Case

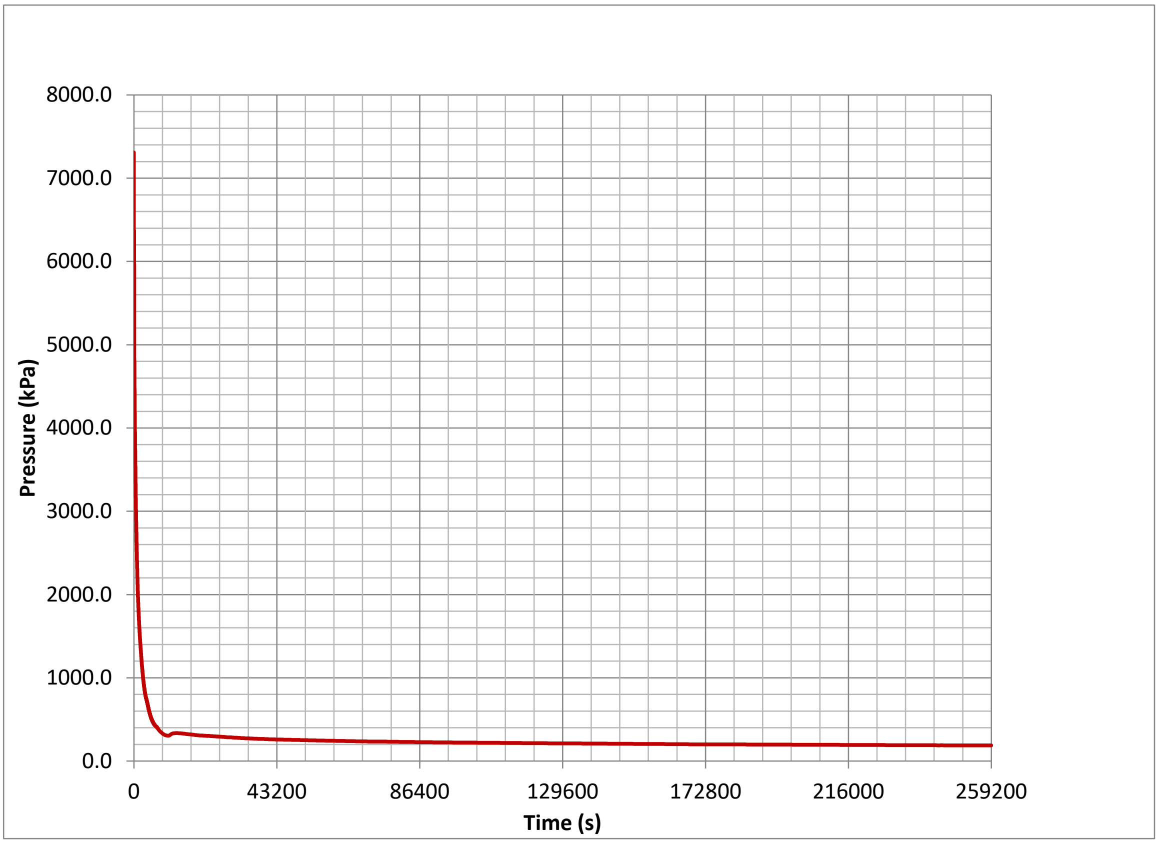

Figure 15.5-109: Reactor Pressure, Small Steam Pipe Break With LOPP, Conservative Case

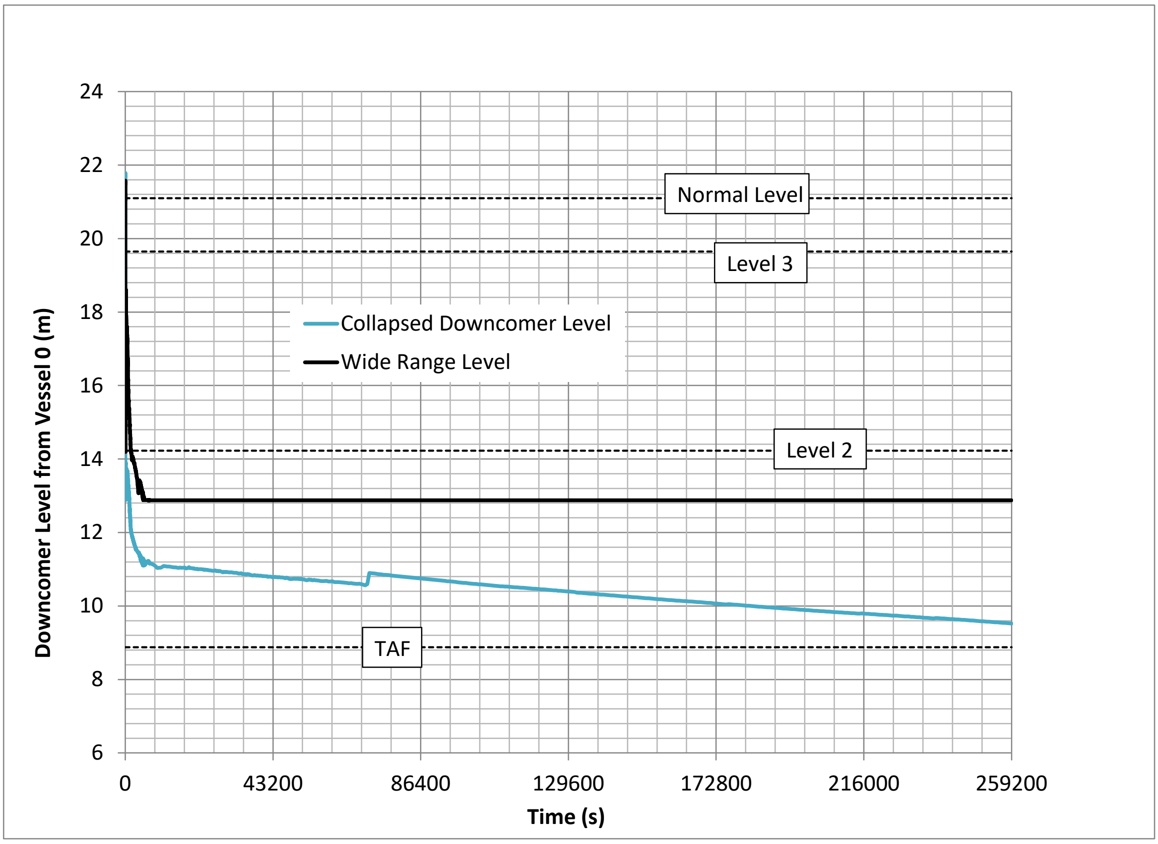

Figure 15.5-110: Reactor Water Level, Small Steam Pipe Break With LOPP, Conservative Case

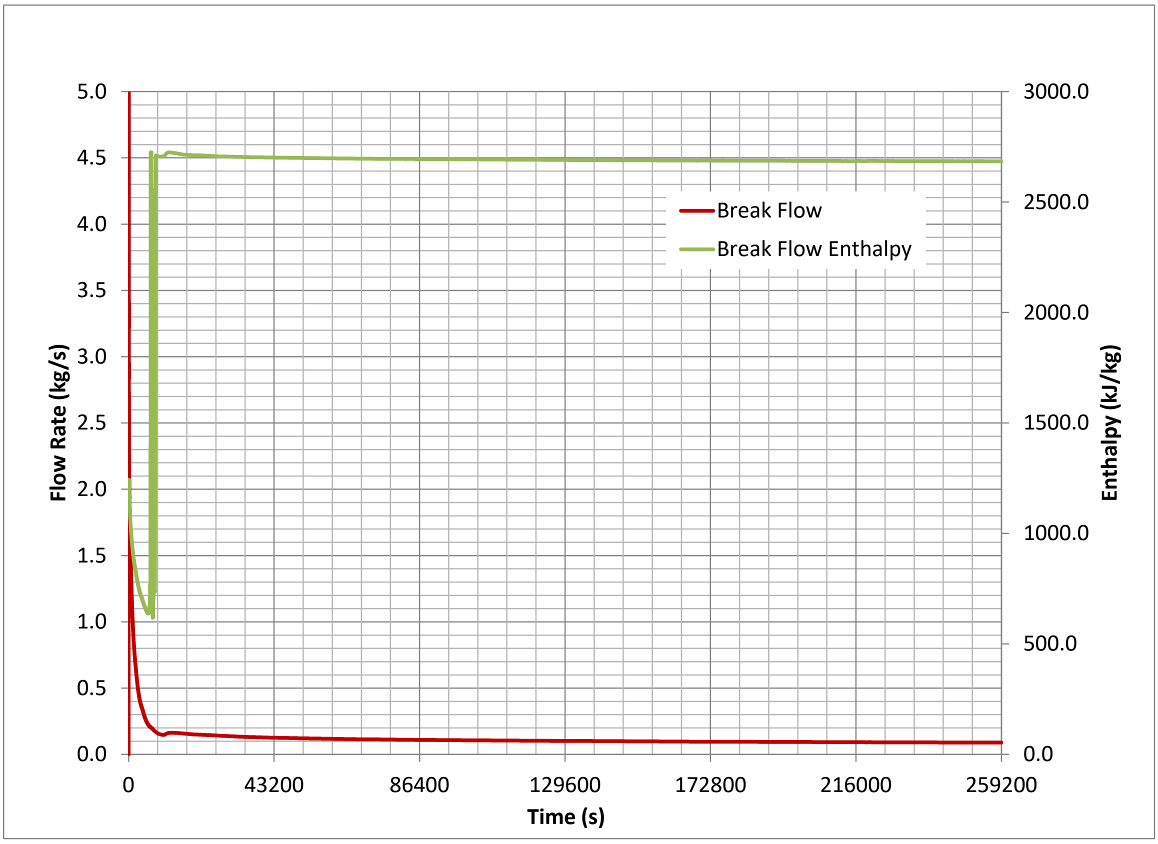

Figure 15.5-111: Break Flow Rate and Enthalpy, Small Steam Pipe Break With LOPP, Conservative Case

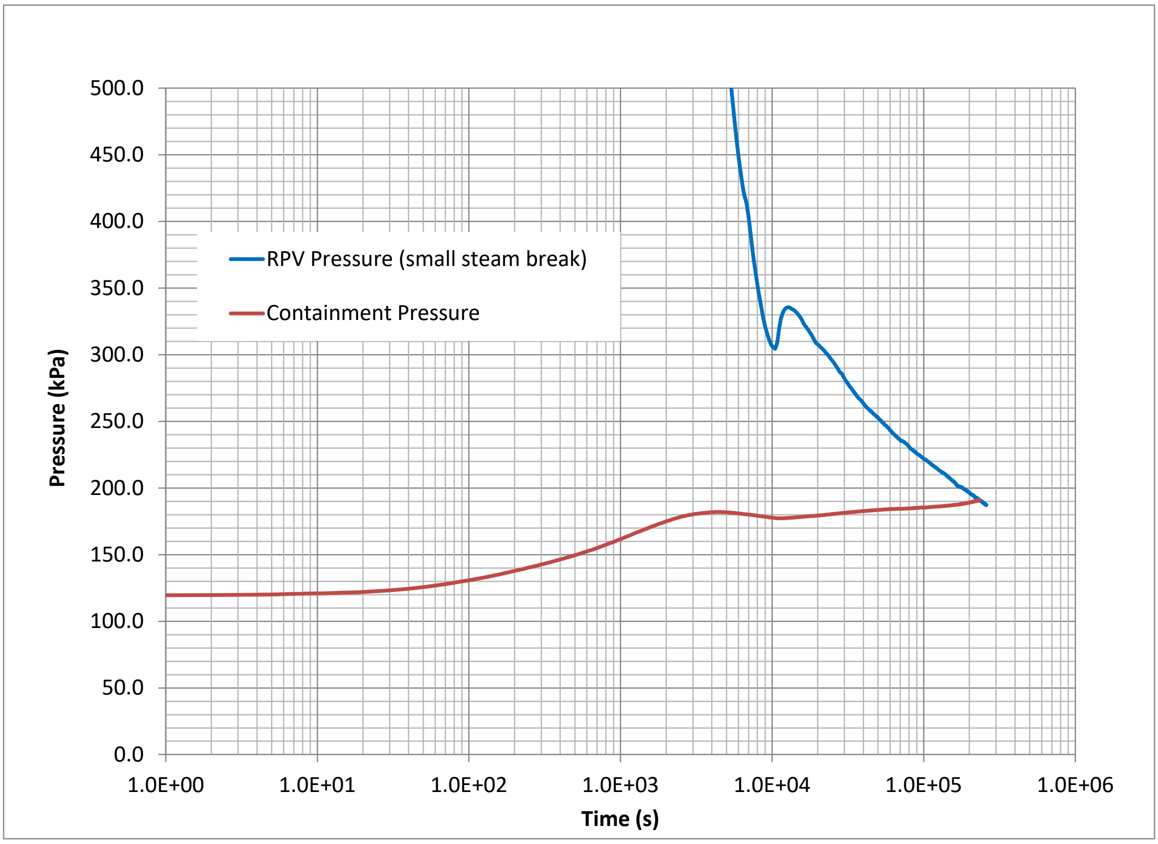

Figure 15.5-112: Containment Pressure, Small Steam Pipe Break with LOPP 2 ICS Trains, Conservative Case

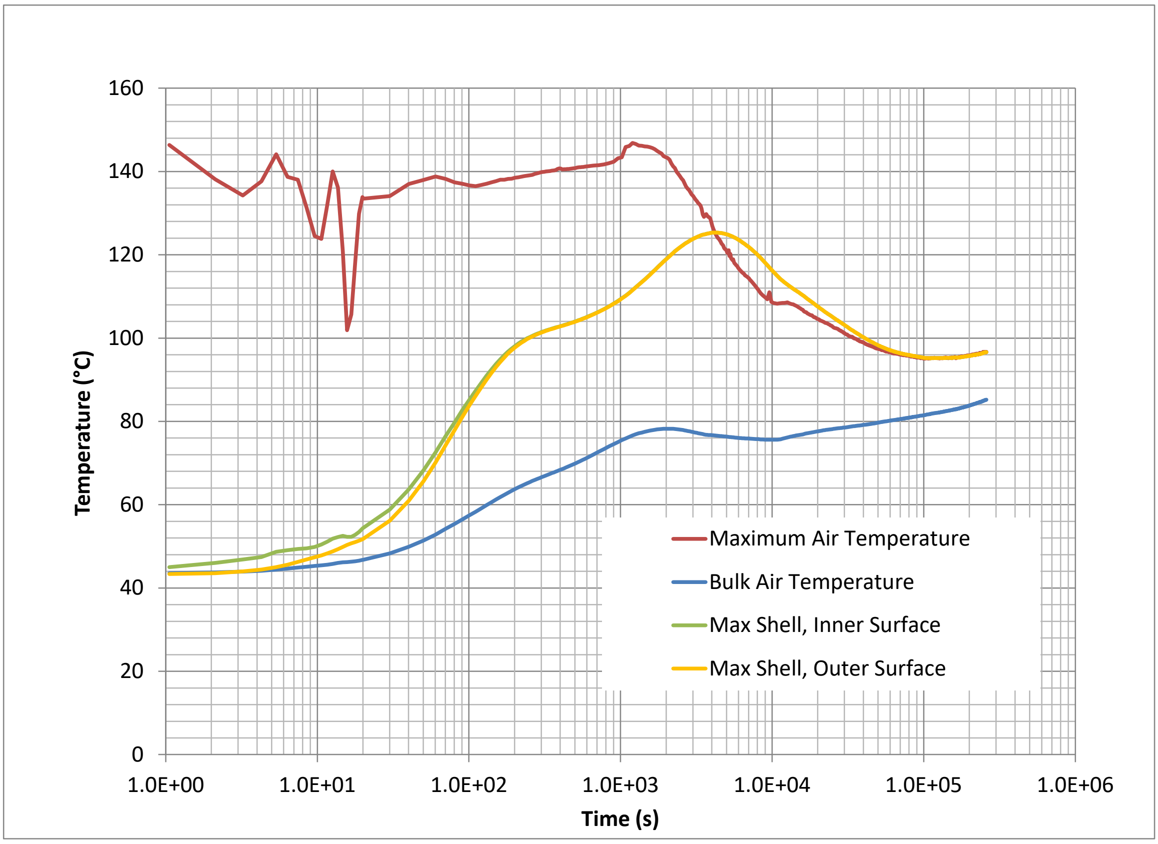

Figure 15.5-113: Containment Temperature, Small Steam Pipe Break, Conservative Case

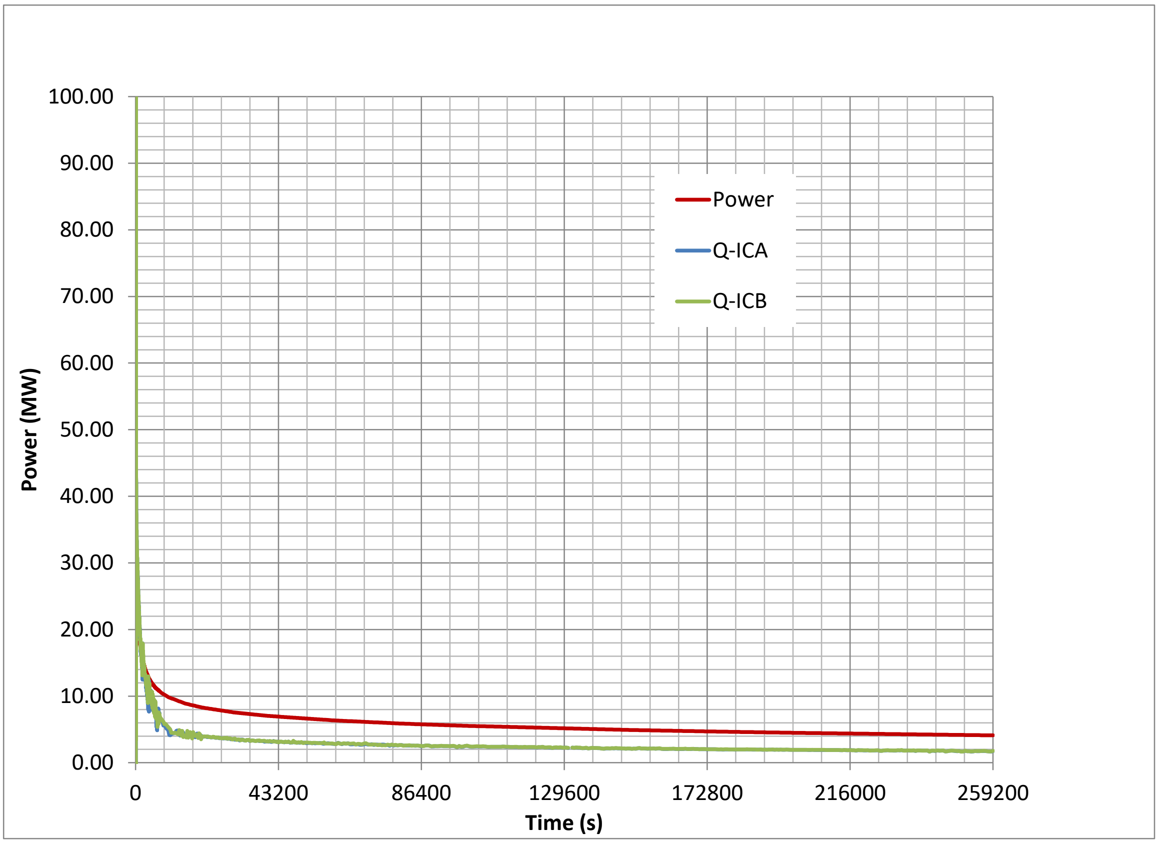

Figure 15.5-114: Reactor Power, Small Liquid Pipe Break, Conservative Case

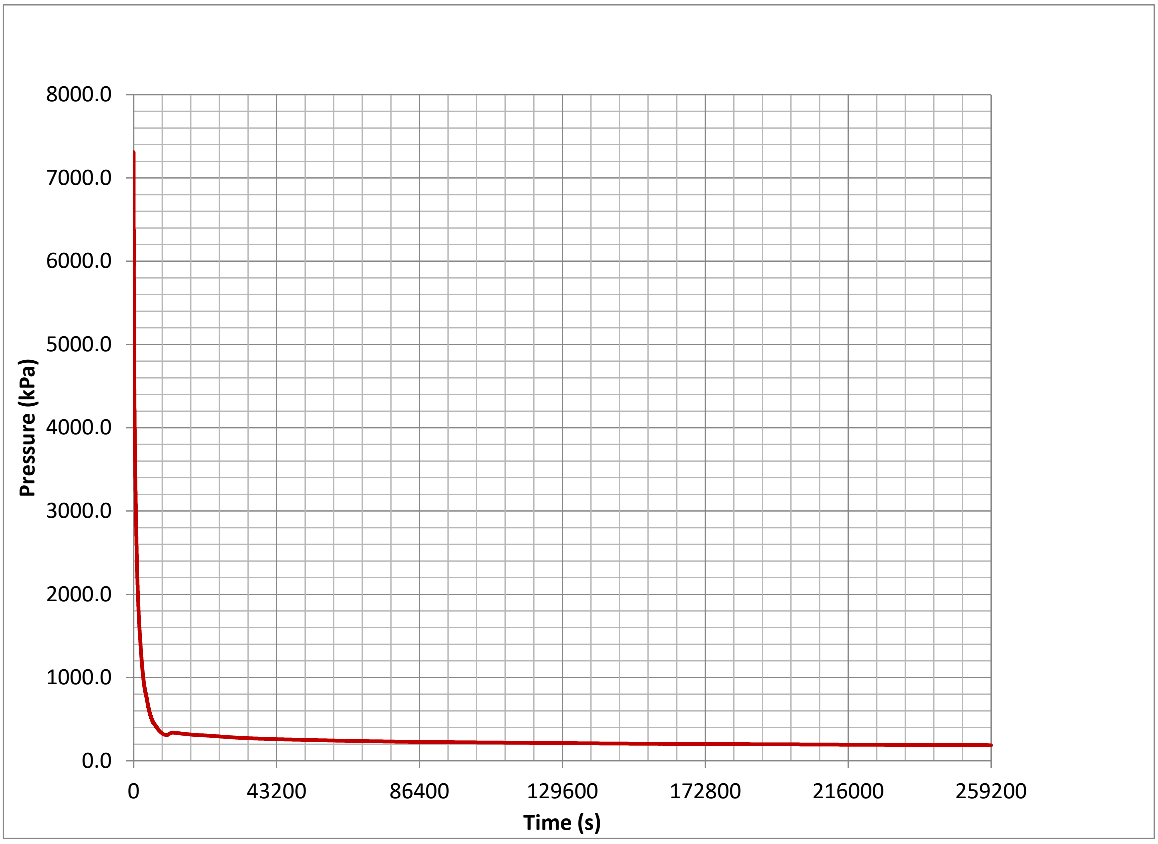

Figure 15.5-115: Reactor Pressure, Small Liquid Pipe Break, Conservative Case

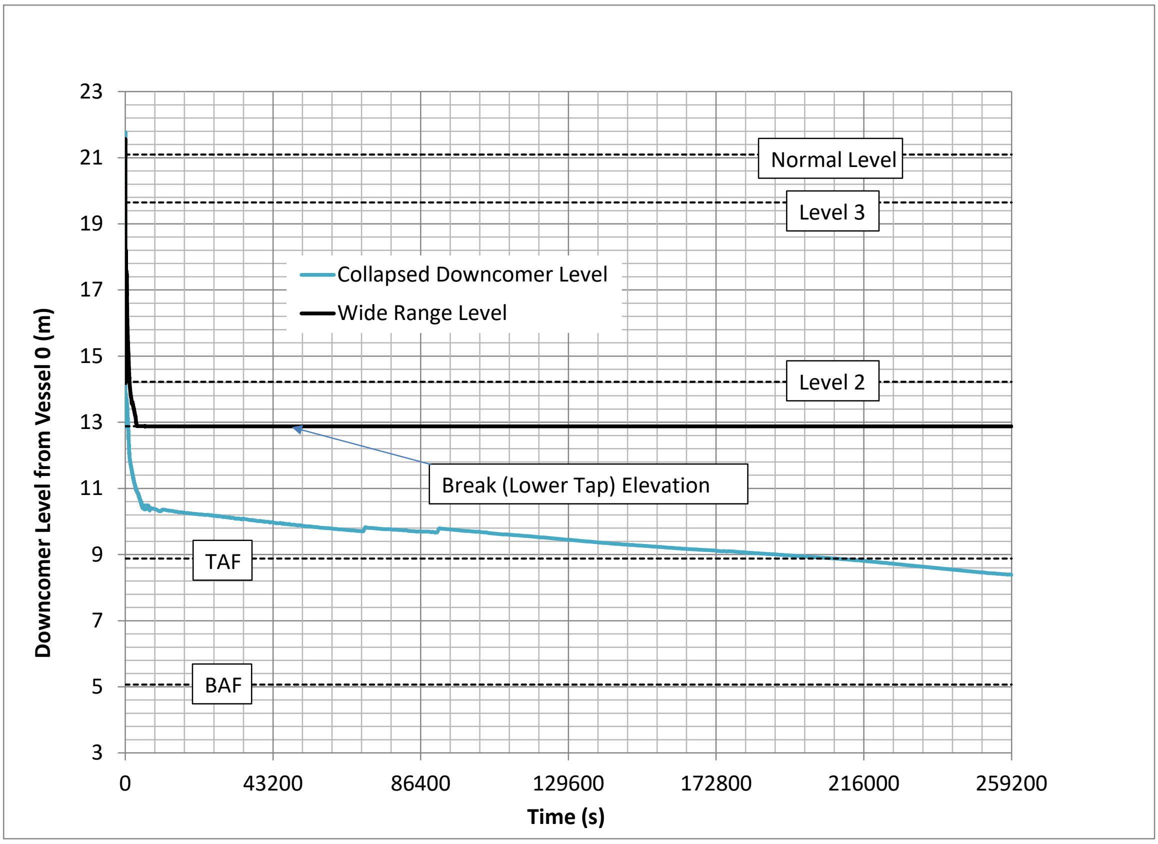

Figure 15.5-116: Reactor Water Level, Small Liquid Break, Conservative Case

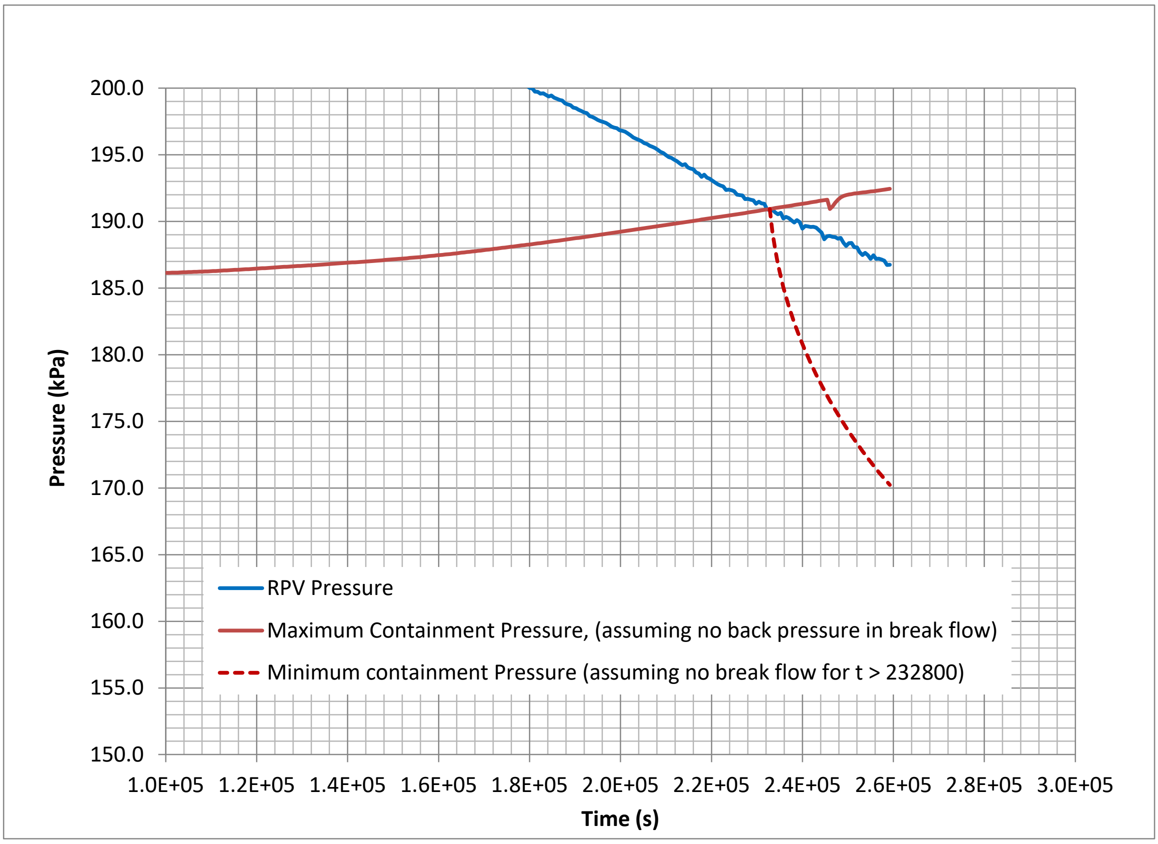

Figure 15.5-117: Containment Pressure After RPV Depressurisation, Small Liquid Pipe Break, Conservative Case

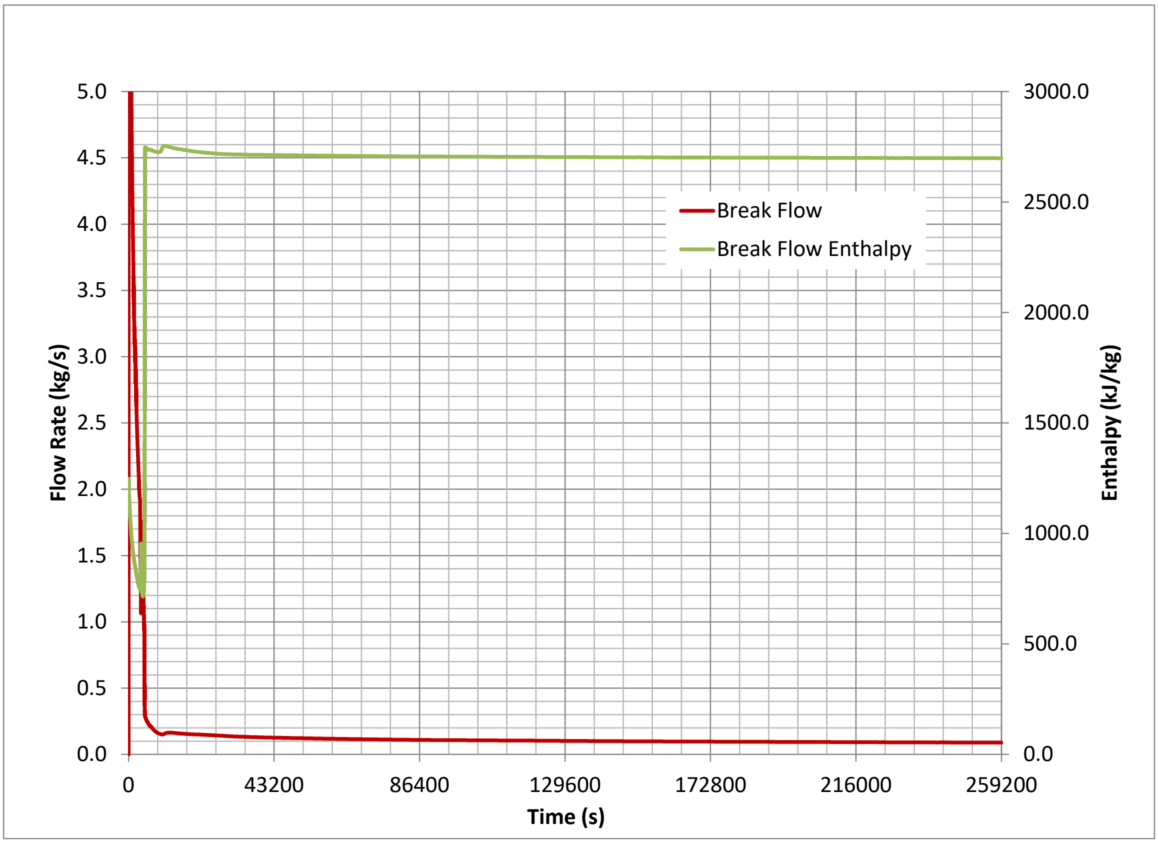

Figure 15.5-118: Break Flow Rate and Enthalpy, Small Liquid Pipe Break, Conservative Case

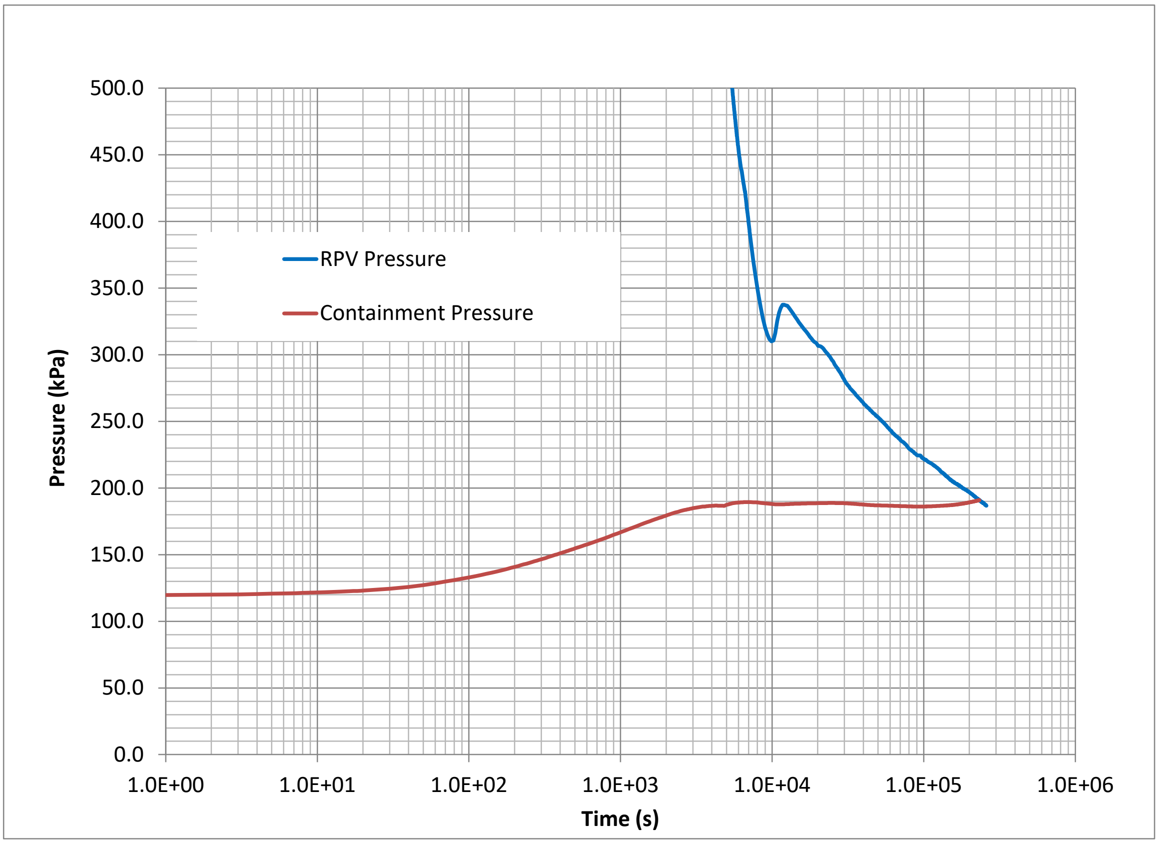

Figure 15.5-119: Containment Pressure, Small Liquid Pipe Break, Conservative Case

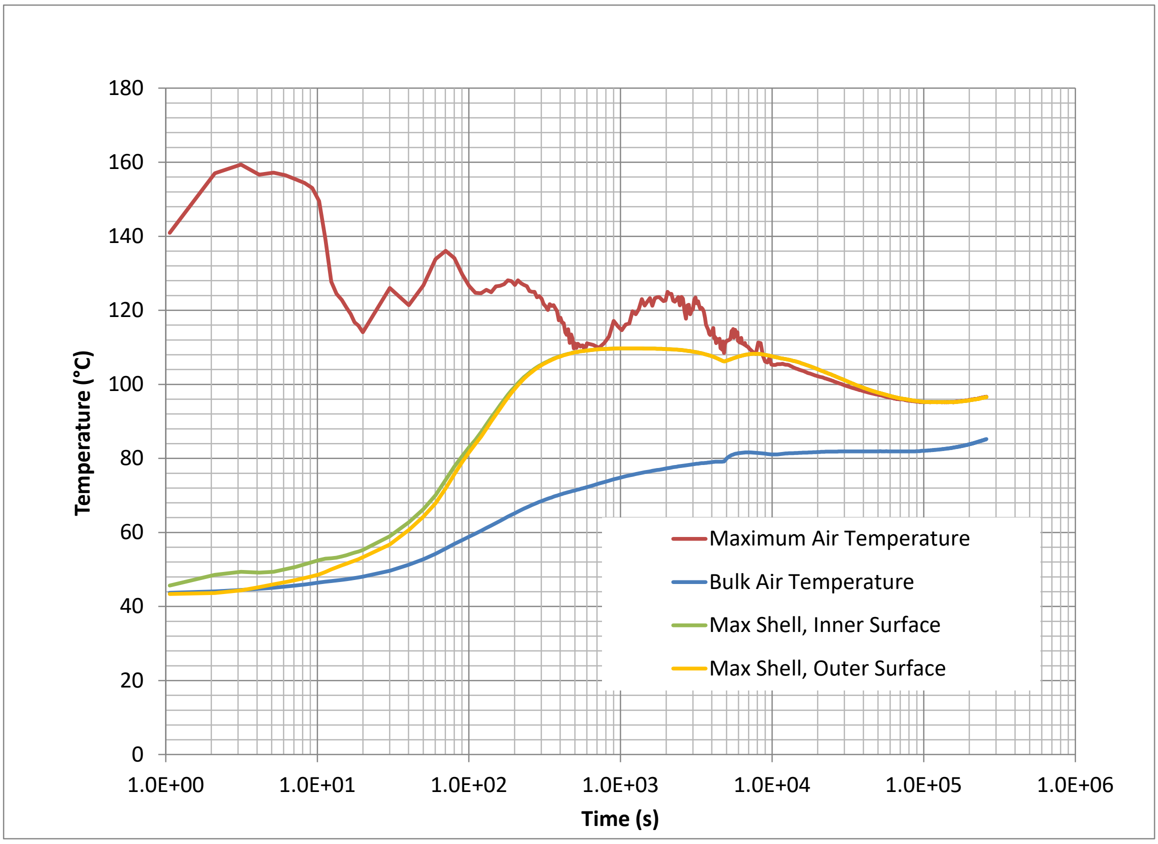

Figure 15.5-120: Containment Temperature, Small Liquid Pipe Break, Conservative Case

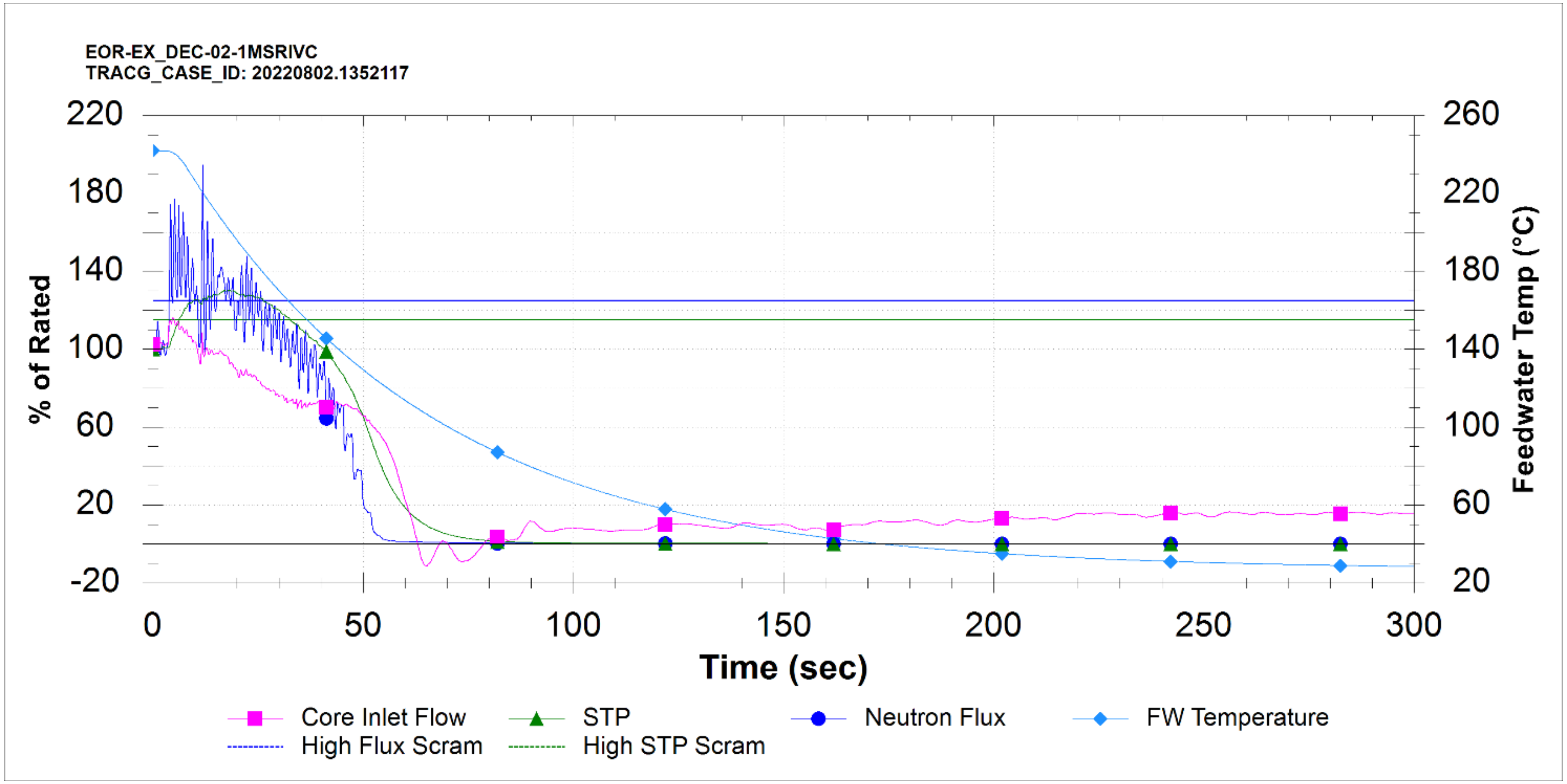

Figure 15.5-121: Closure of One Main Steam Reactor Isolation Valve with Failure to Scram (DEC)

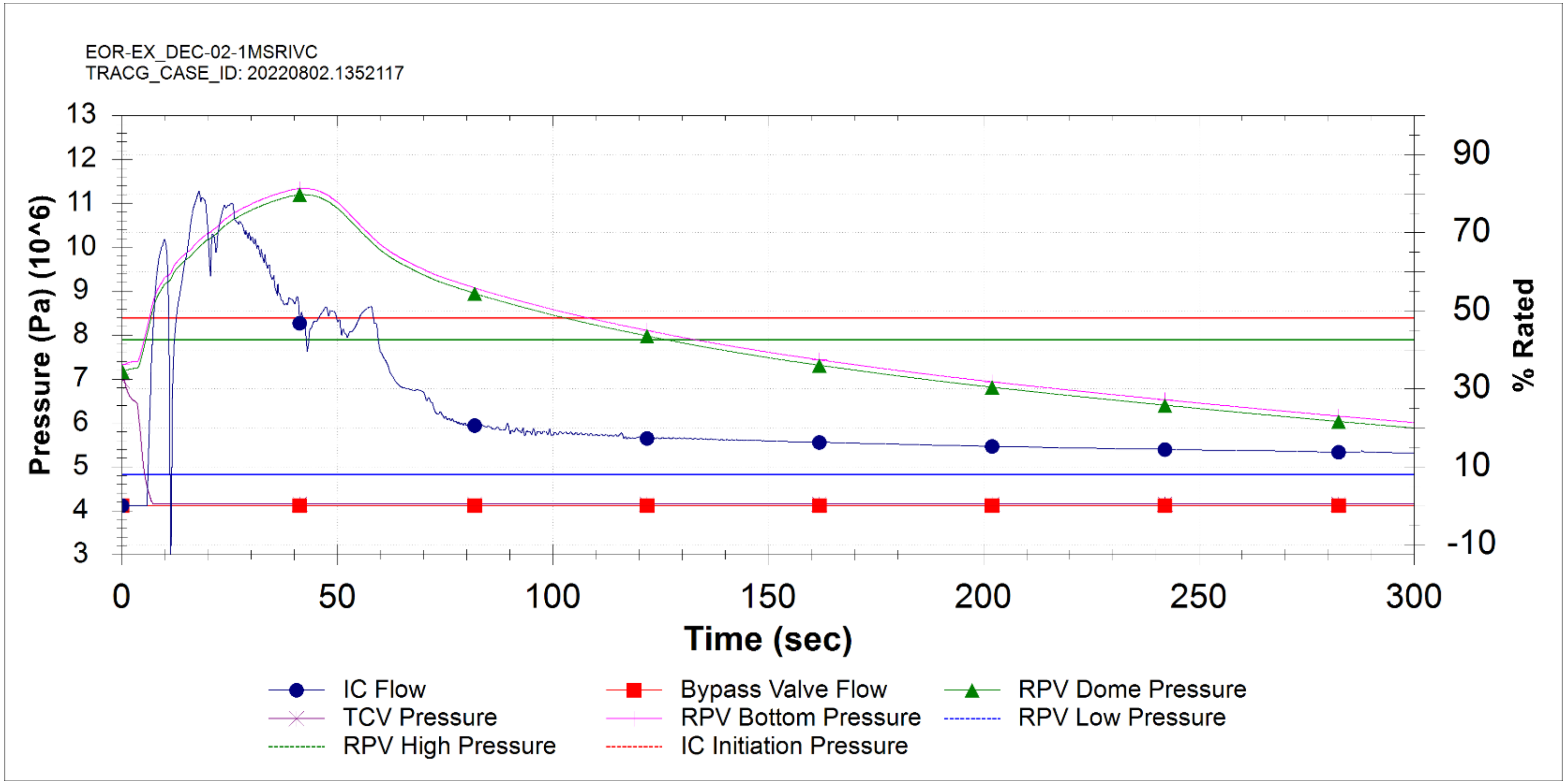

Figure 15.5-122: Closure of One Main Steam Reactor Isolation Valve with Failure to Scram (DEC)

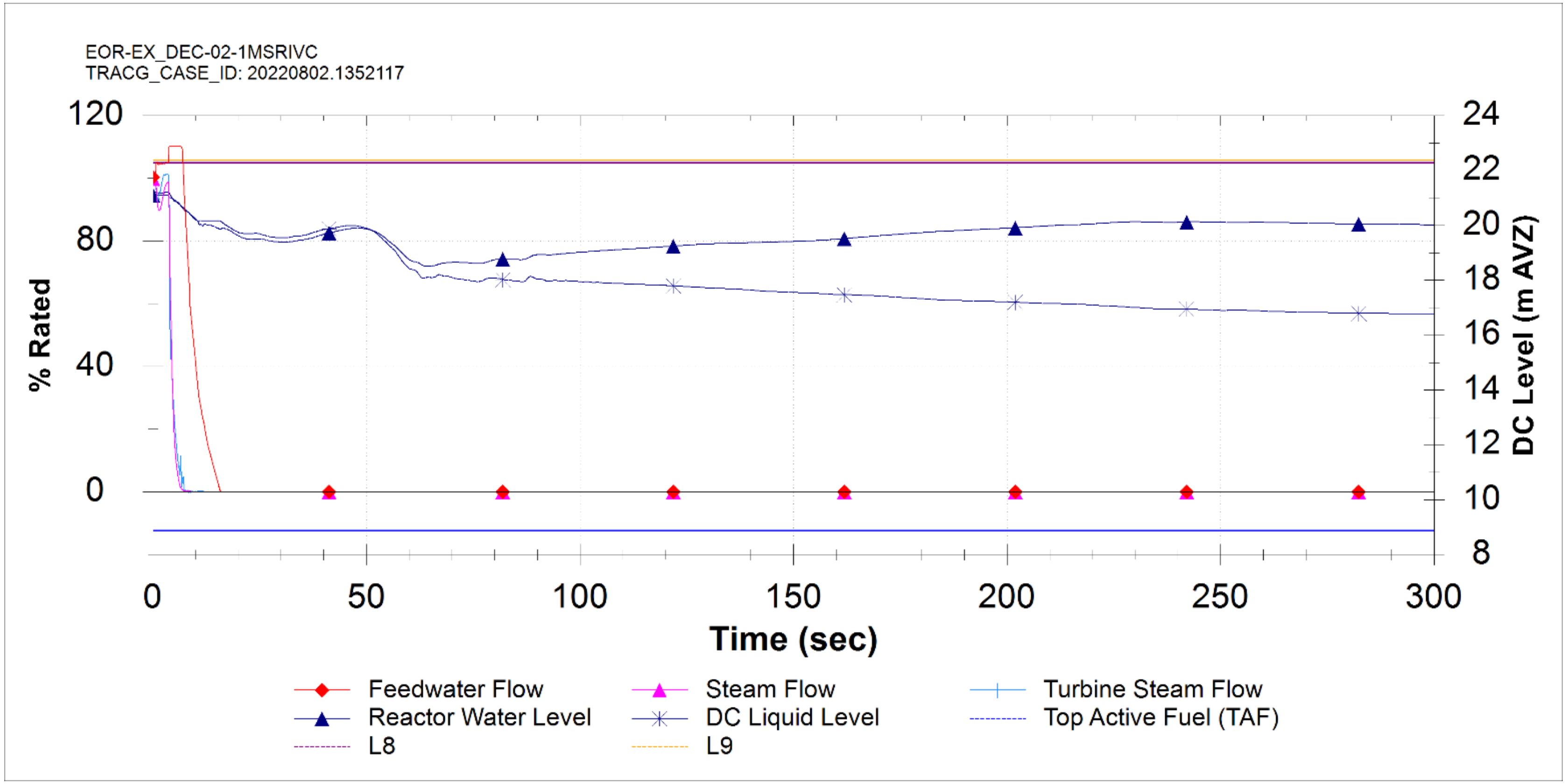

Figure 15.5-123: Closure of One Main Steam Reactor Isolation Valve with Failure to Scram (DEC)

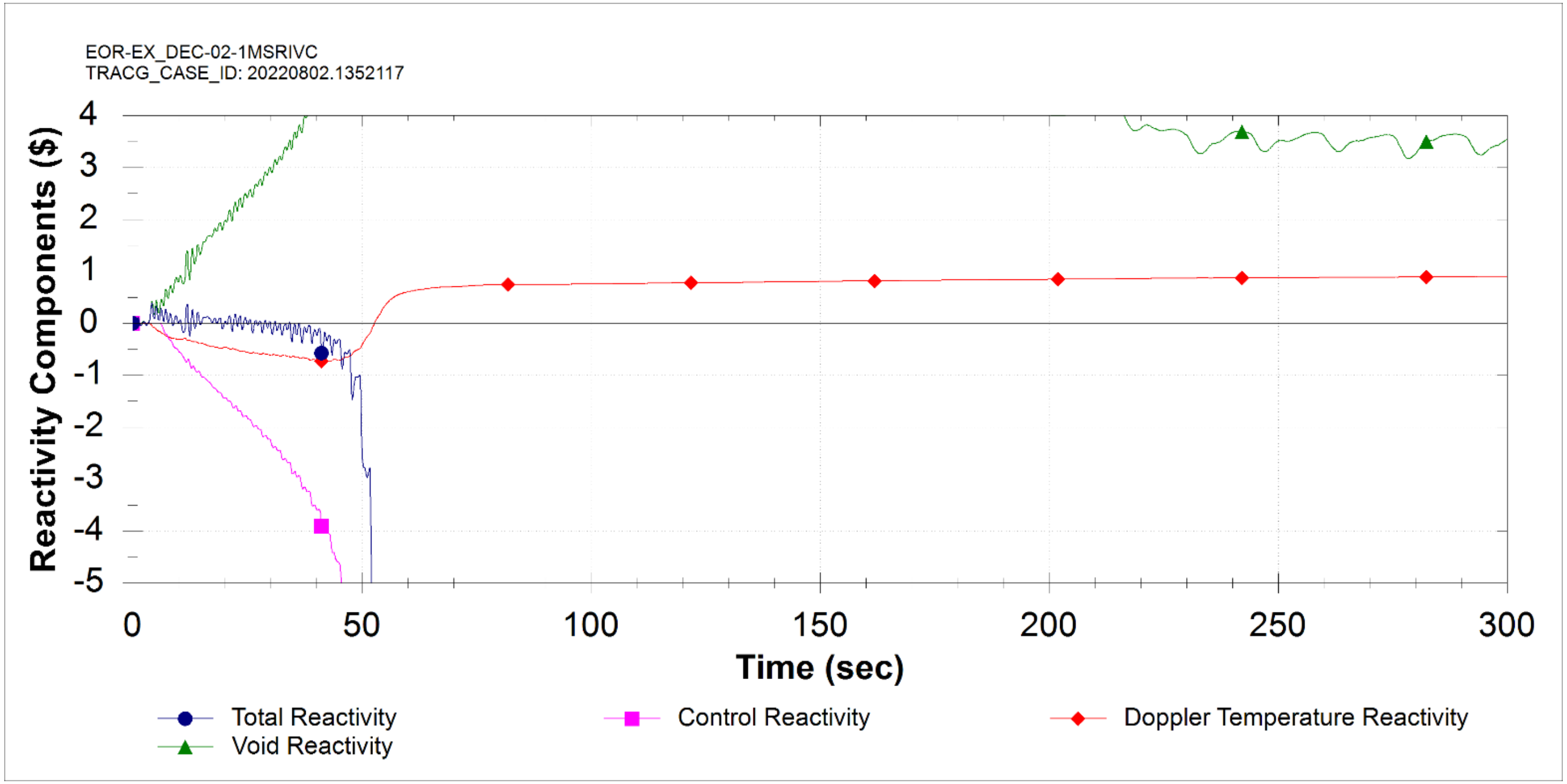

Figure 15.5-124: Closure of One Main Steam Reactor Isolation Valve with Failure to Scram (DEC)

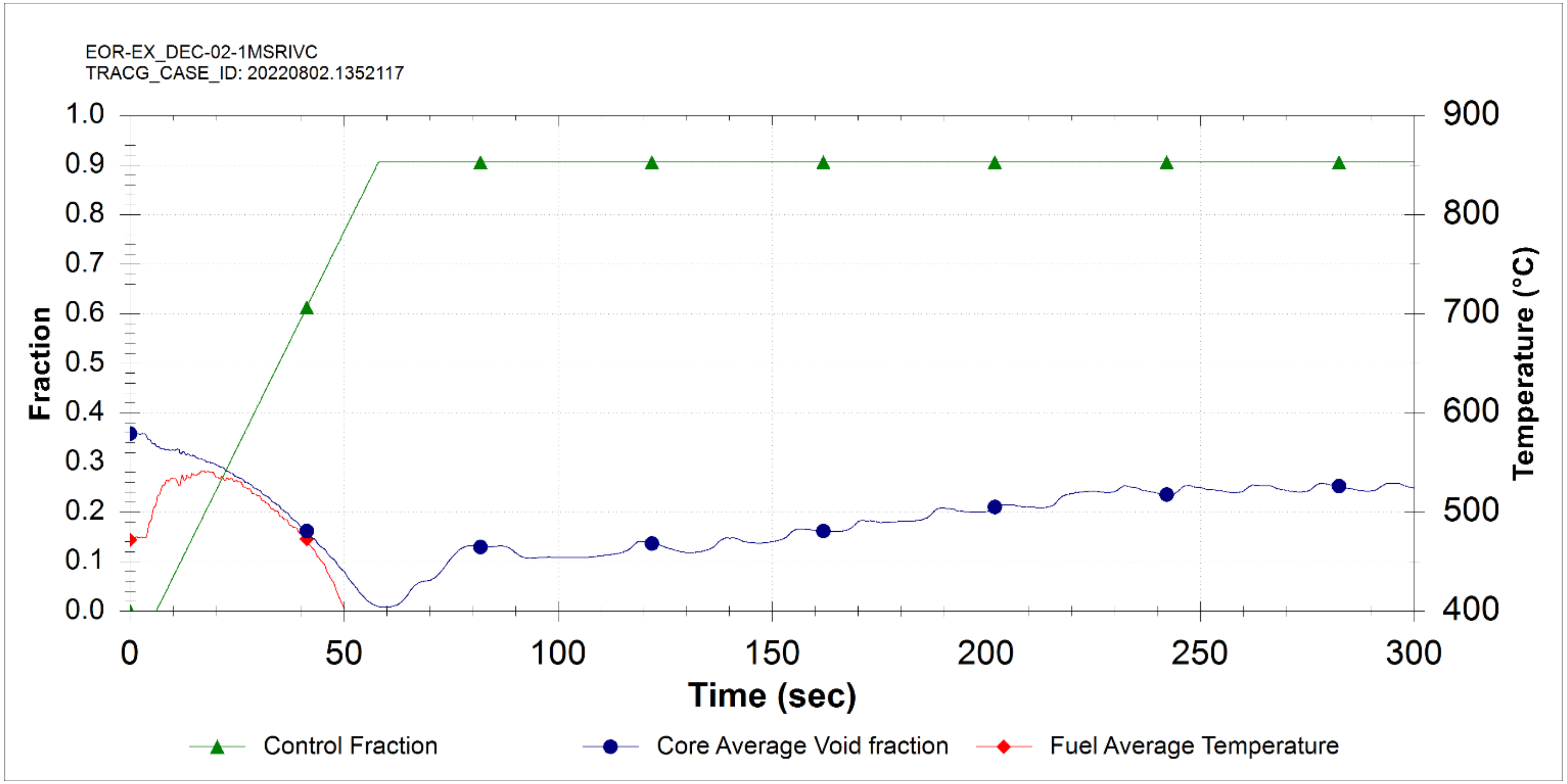

Figure 15.5-125: Closure of One Main Steam Reactor Isolation Valve with Failure to Scram (DEC)

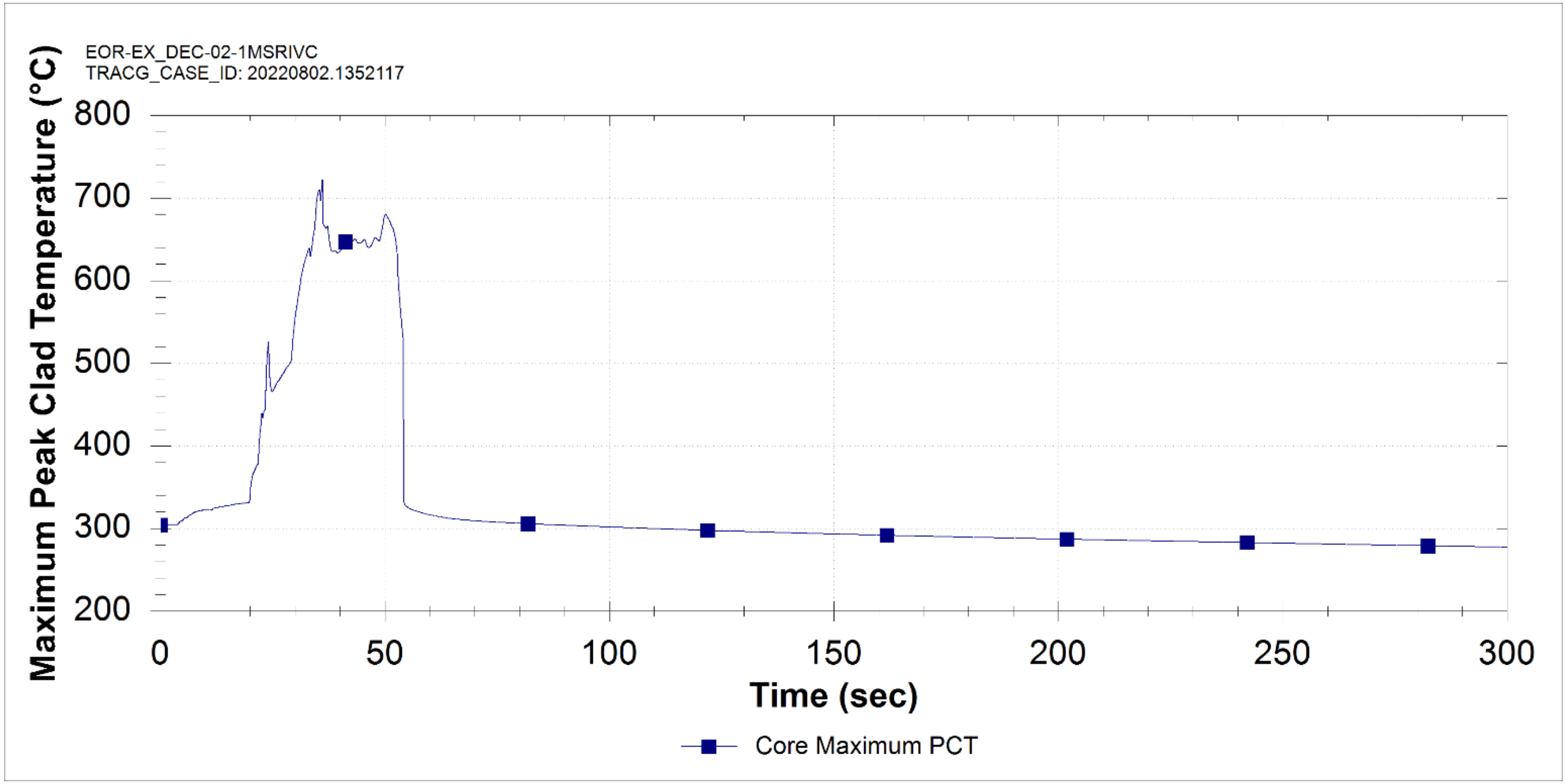

Figure 15.5-126: Closure of One Main Steam Reactor Isolation Valve with Failure to Scram (DEC)

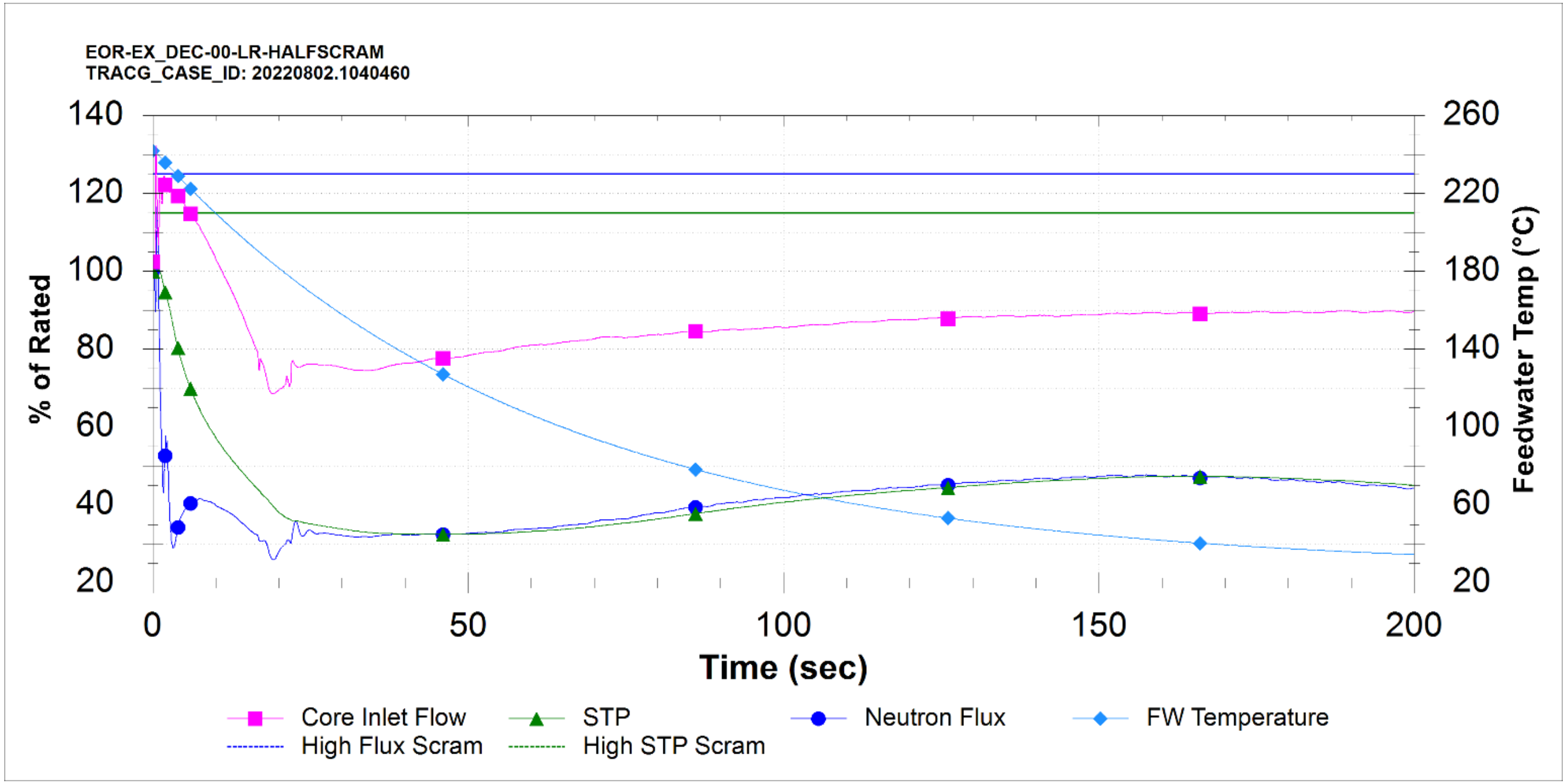

Figure 15.5-127: Complex Sequence Generator Load Rejection (DEC)

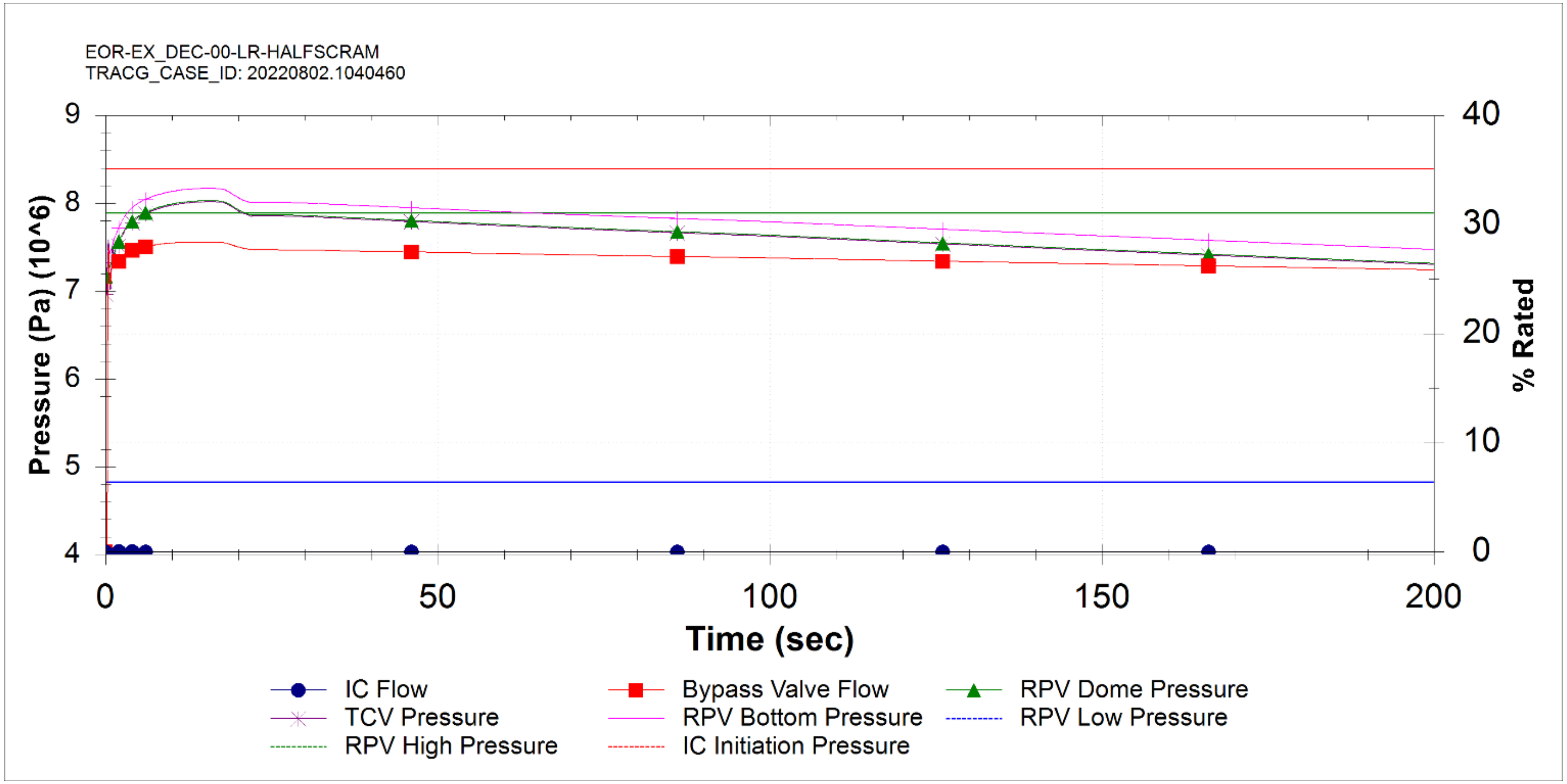

Figure 15.5-128: Complex Sequence Generator Load Rejection (DEC)

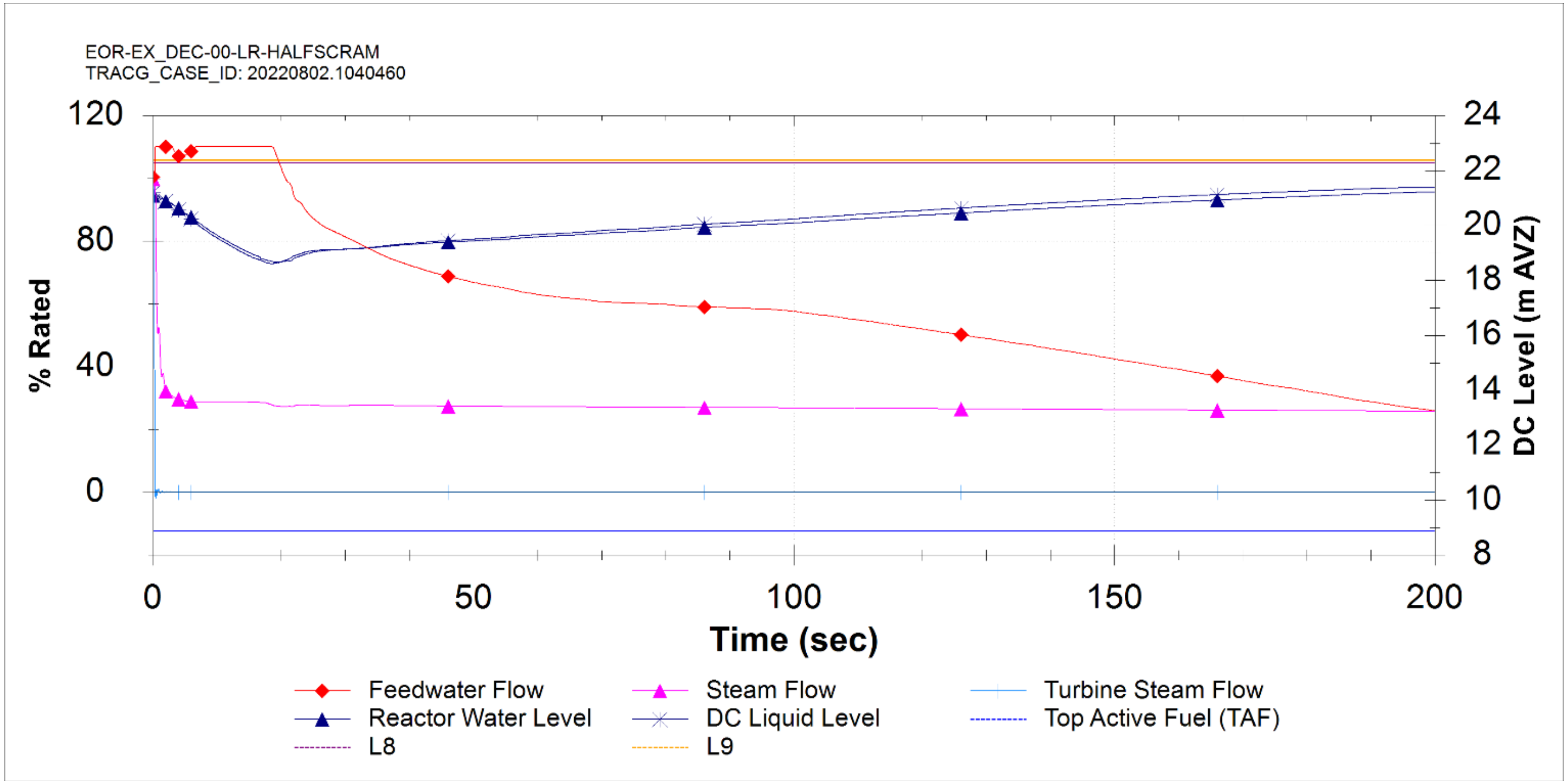

Figure 15.5-129: Complex Sequence Generator Load Rejection (DEC)

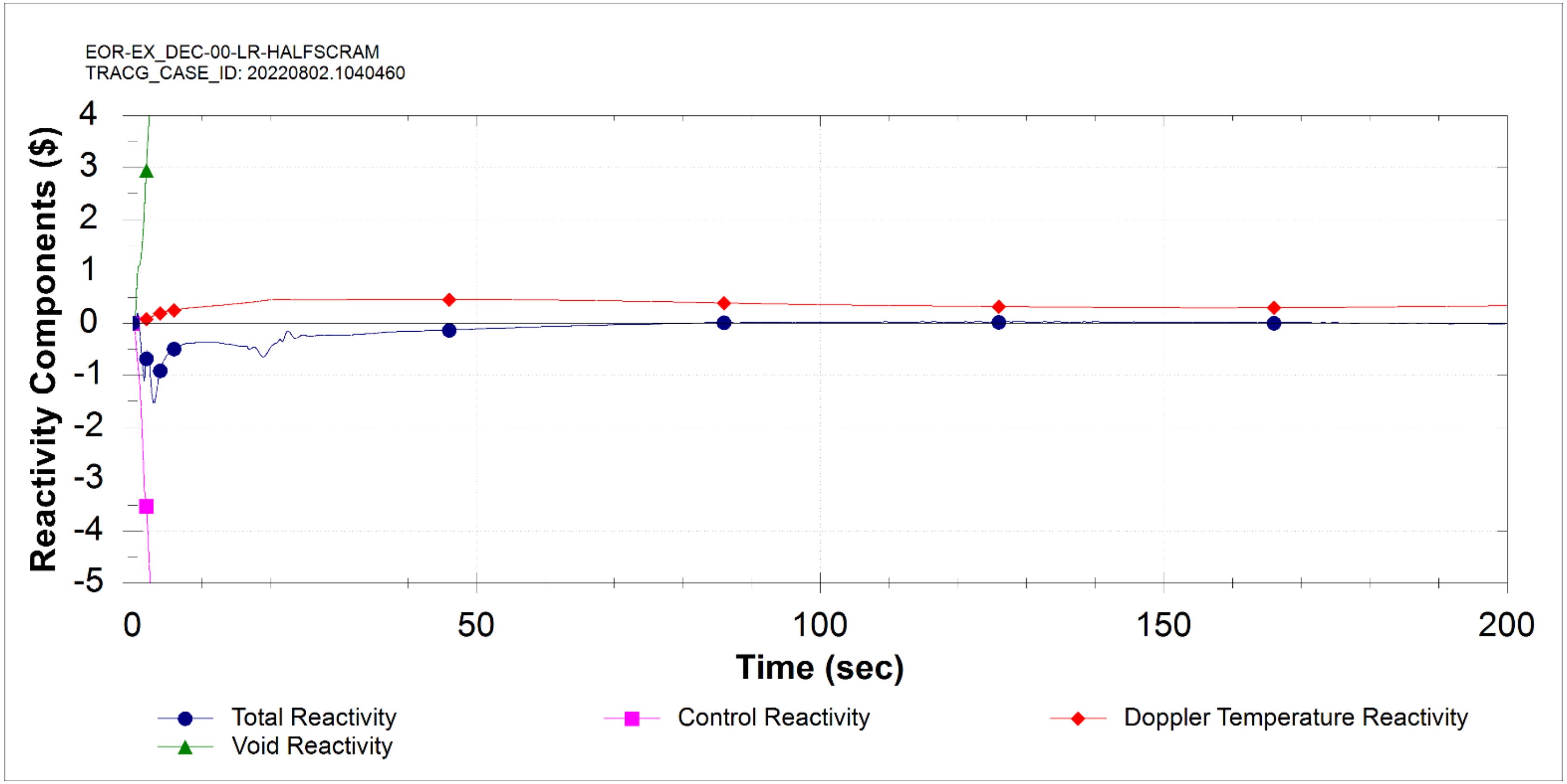

Figure 15.5-130: Complex Sequence Generator Load Rejection (DEC)

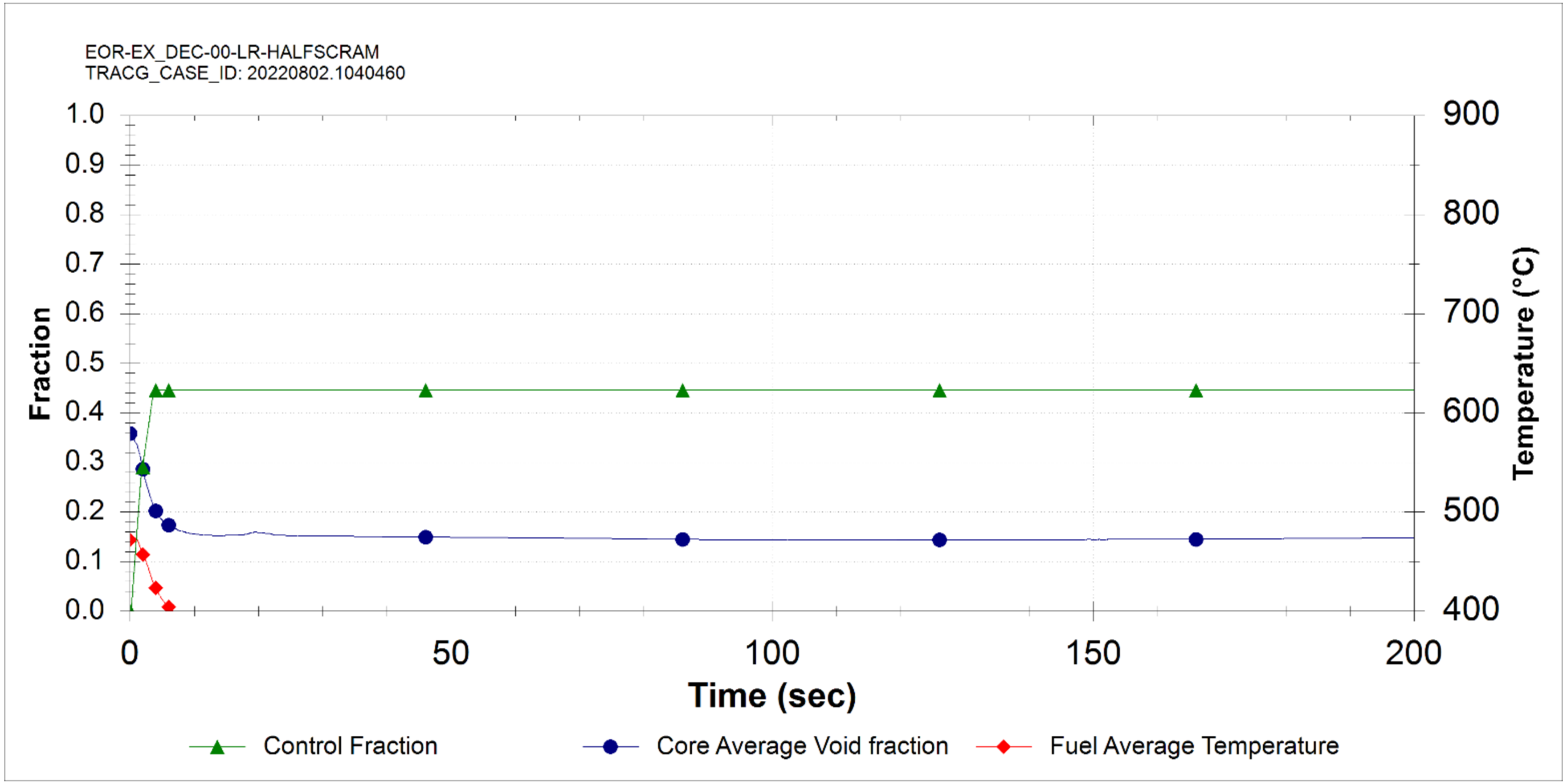

Figure 15.5-131: Complex Sequence Generator Load Rejection (DEC)

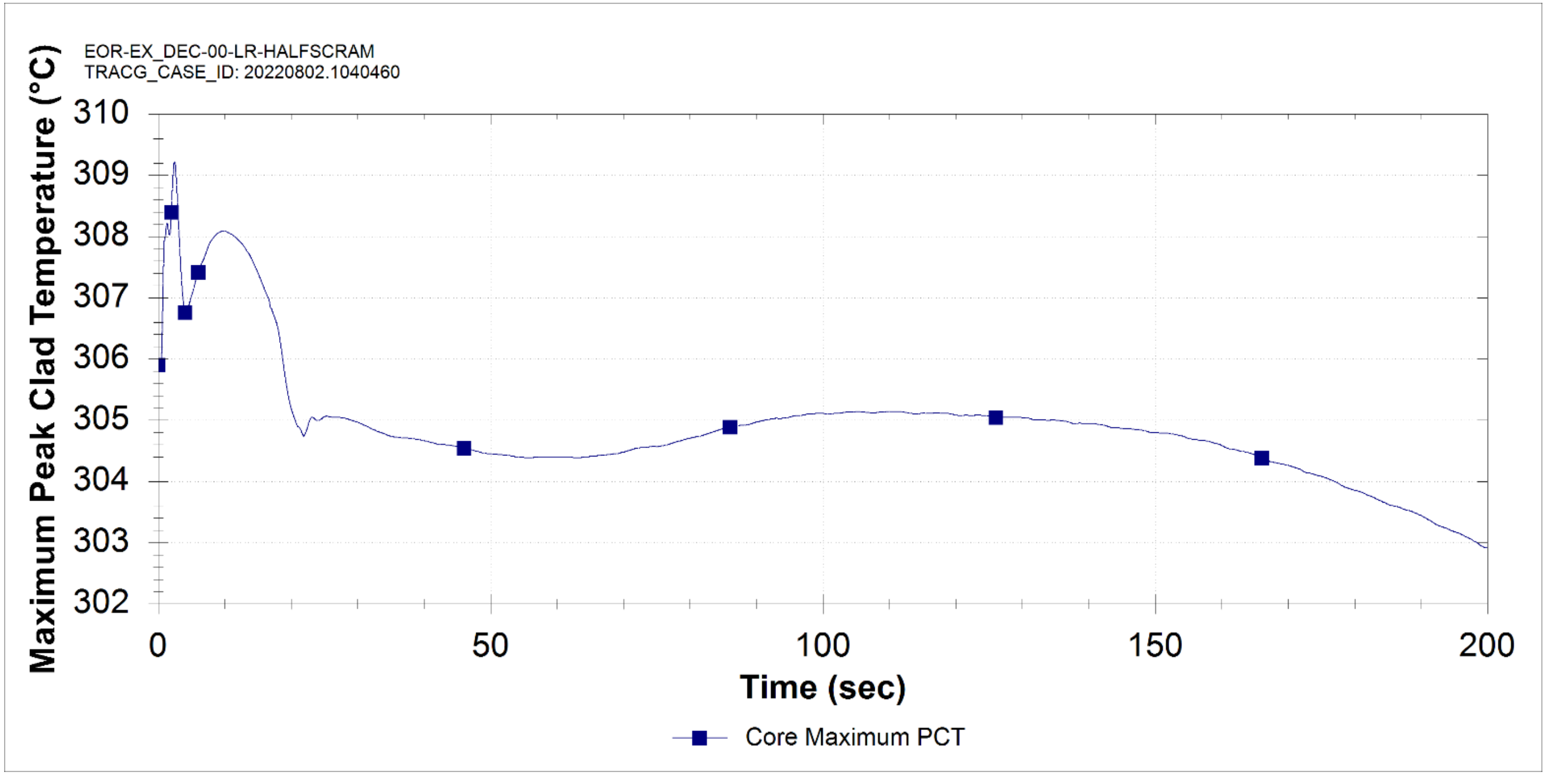

Figure 15.5-132: Complex Sequence Generator Load Rejection (DEC)

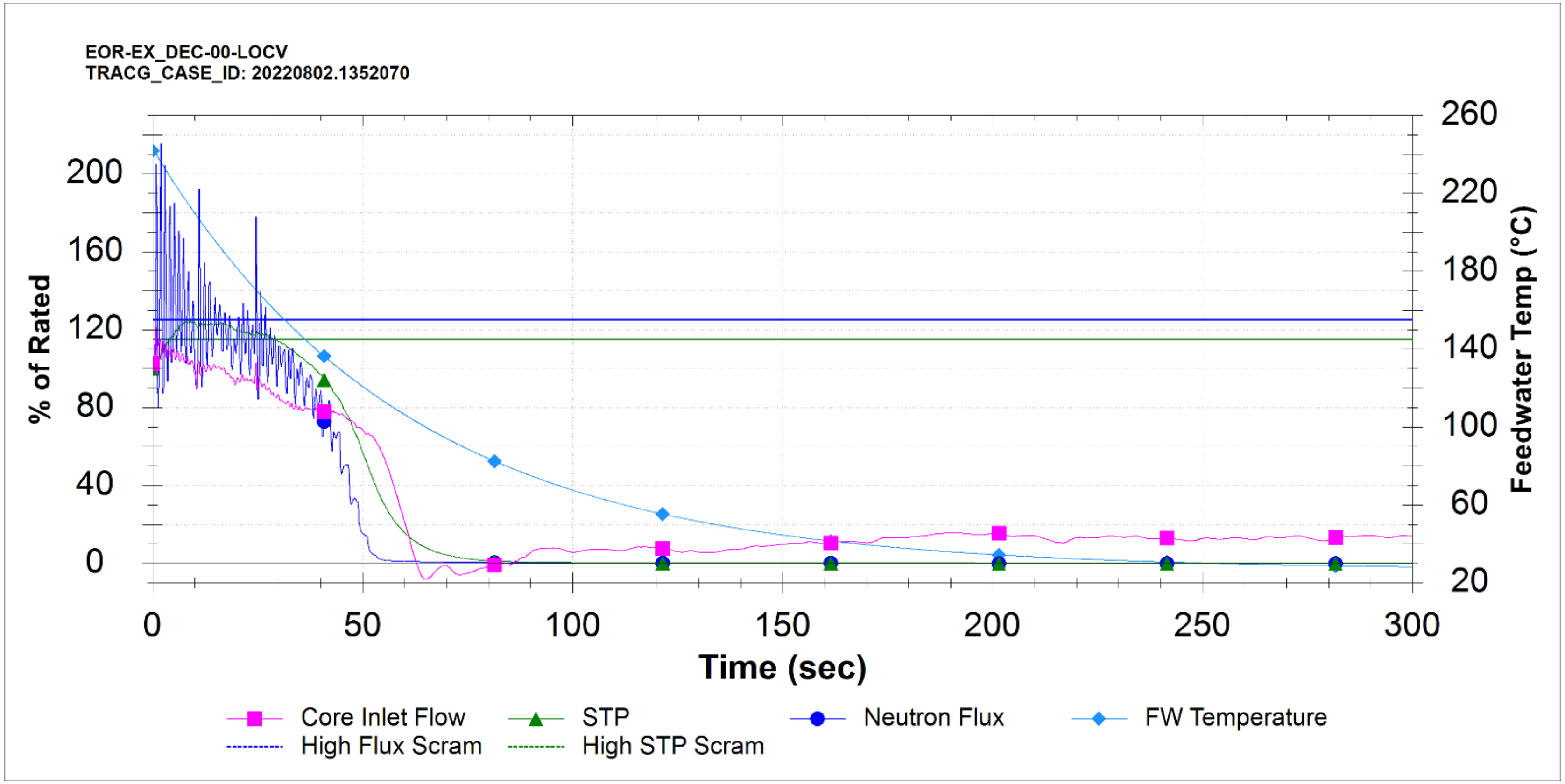

Figure 15.5-133: Loss of Condenser Vacuum (DEC)

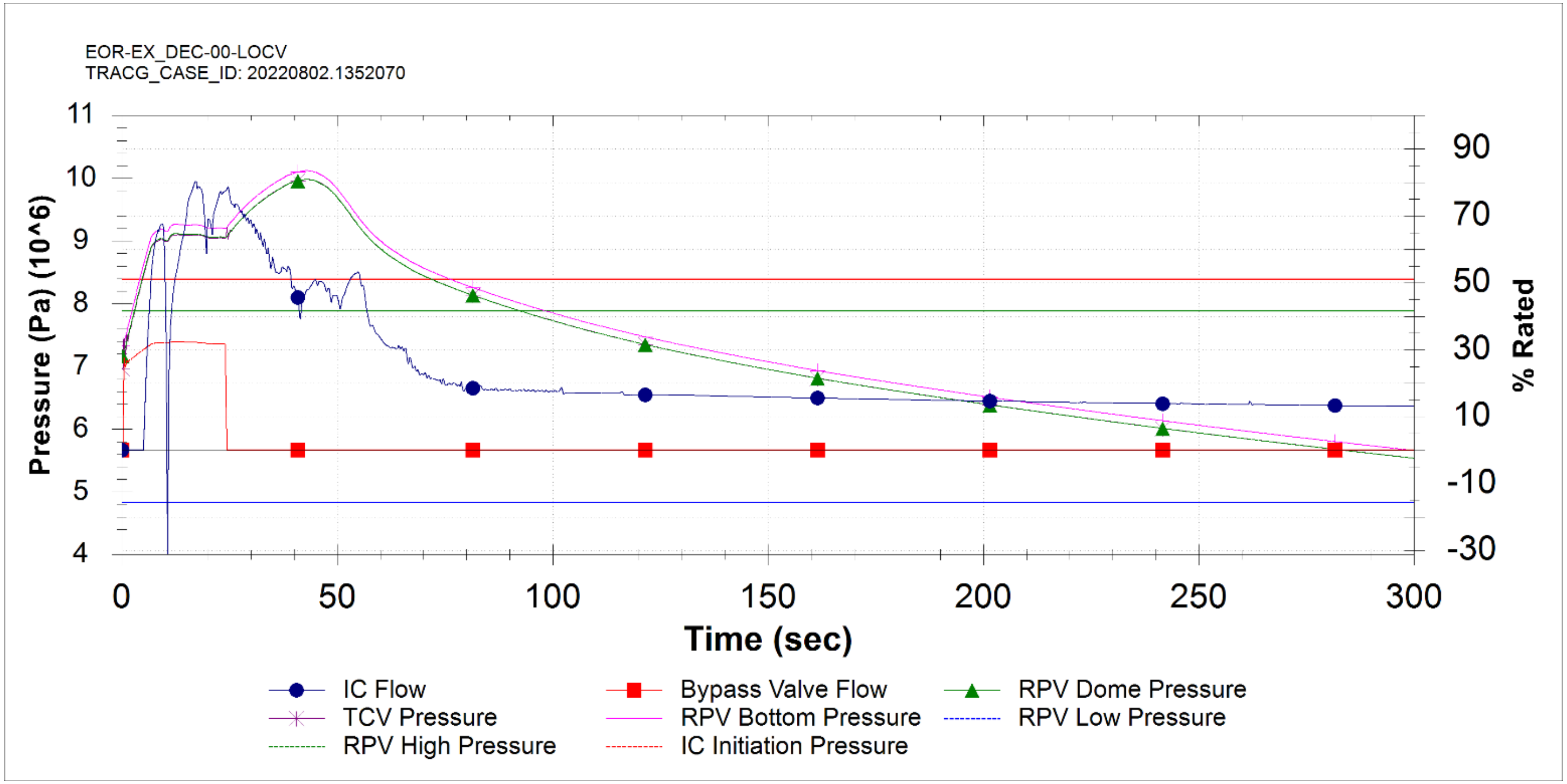

Figure 15.5-134: Loss of Condenser Vacuum (DEC)

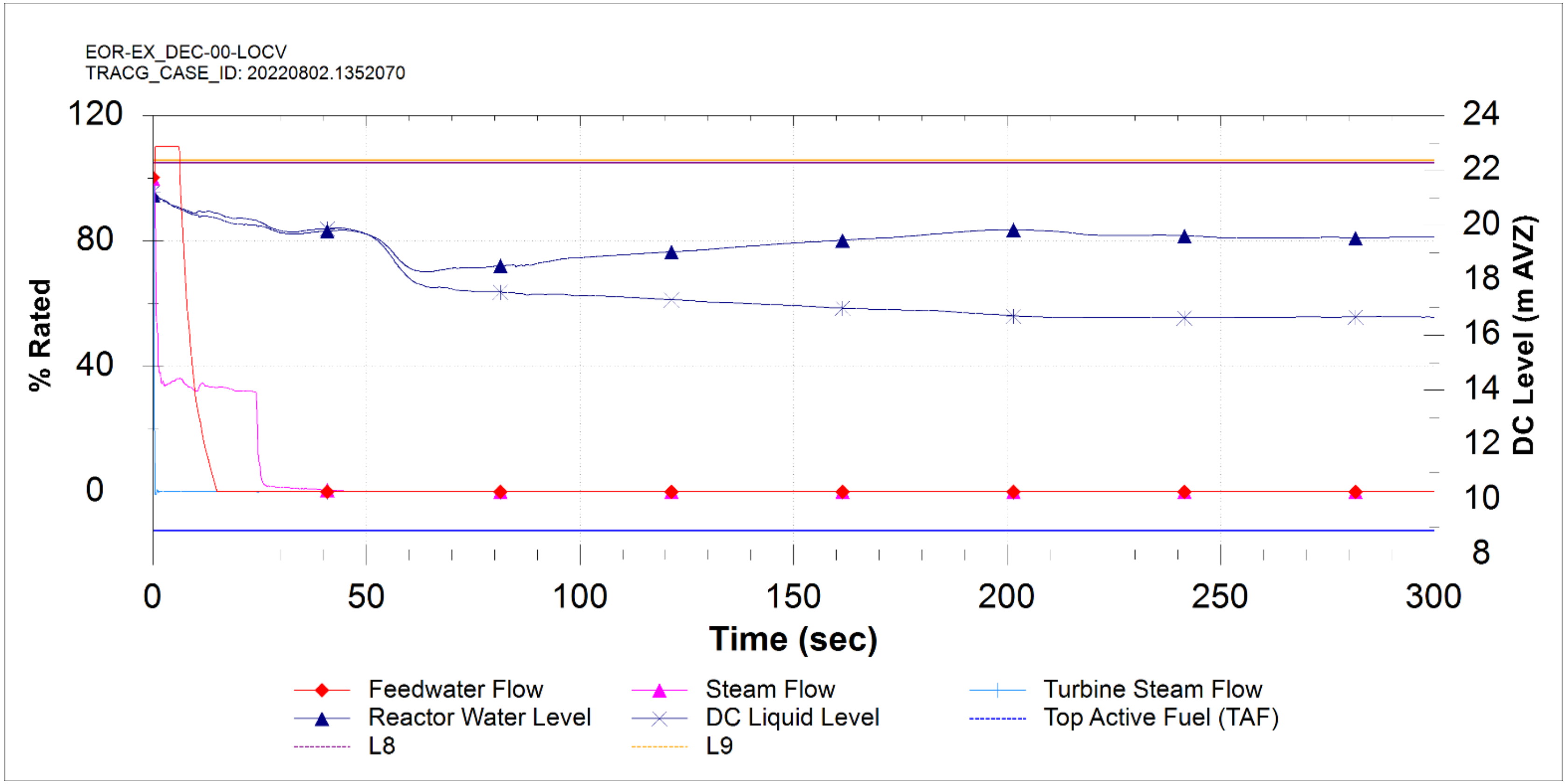

Figure 15.5-135: Loss of Condenser Vacuum (DEC)

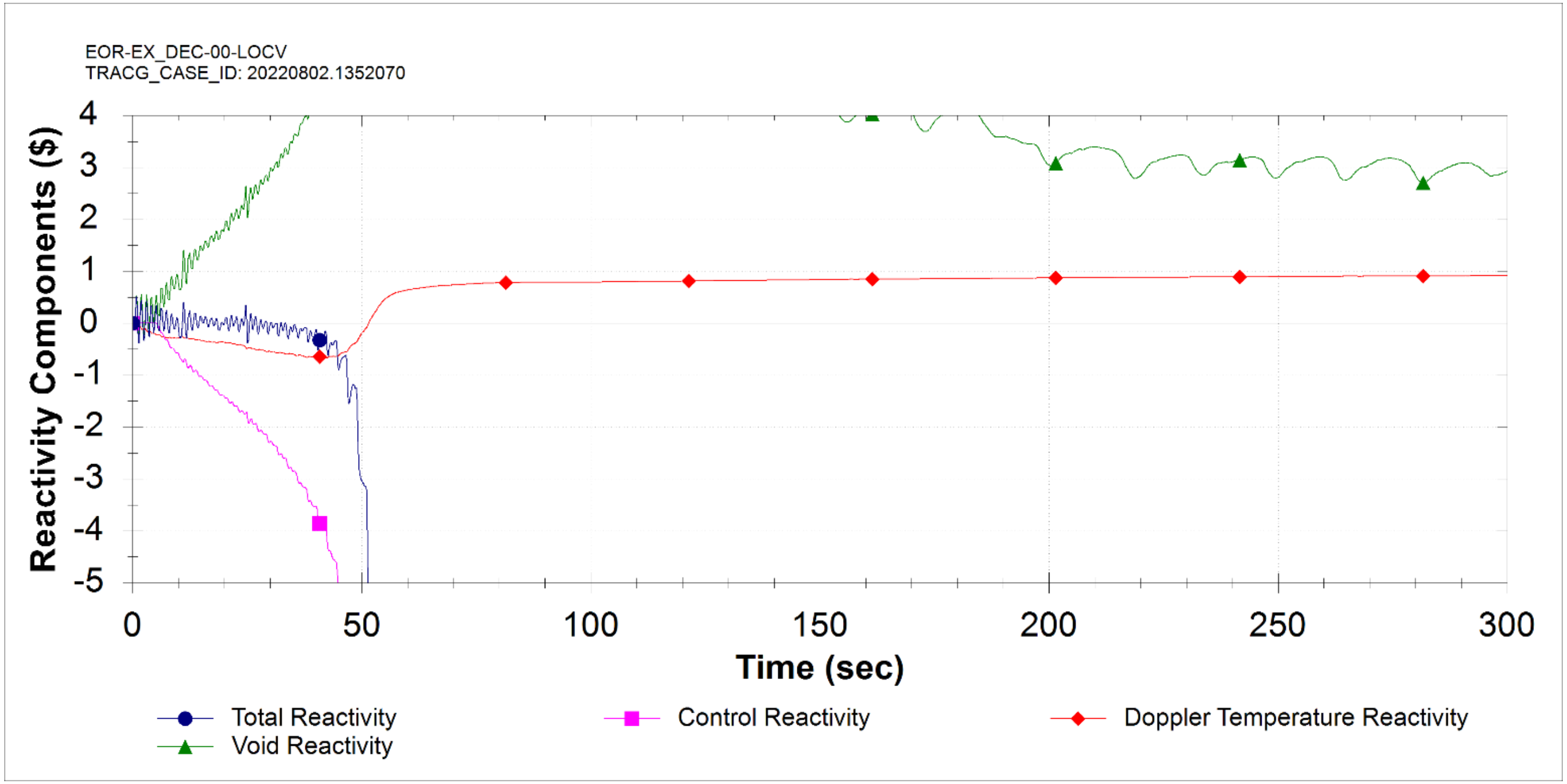

Figure 15.5-136: Loss of Condenser Vacuum (DEC)

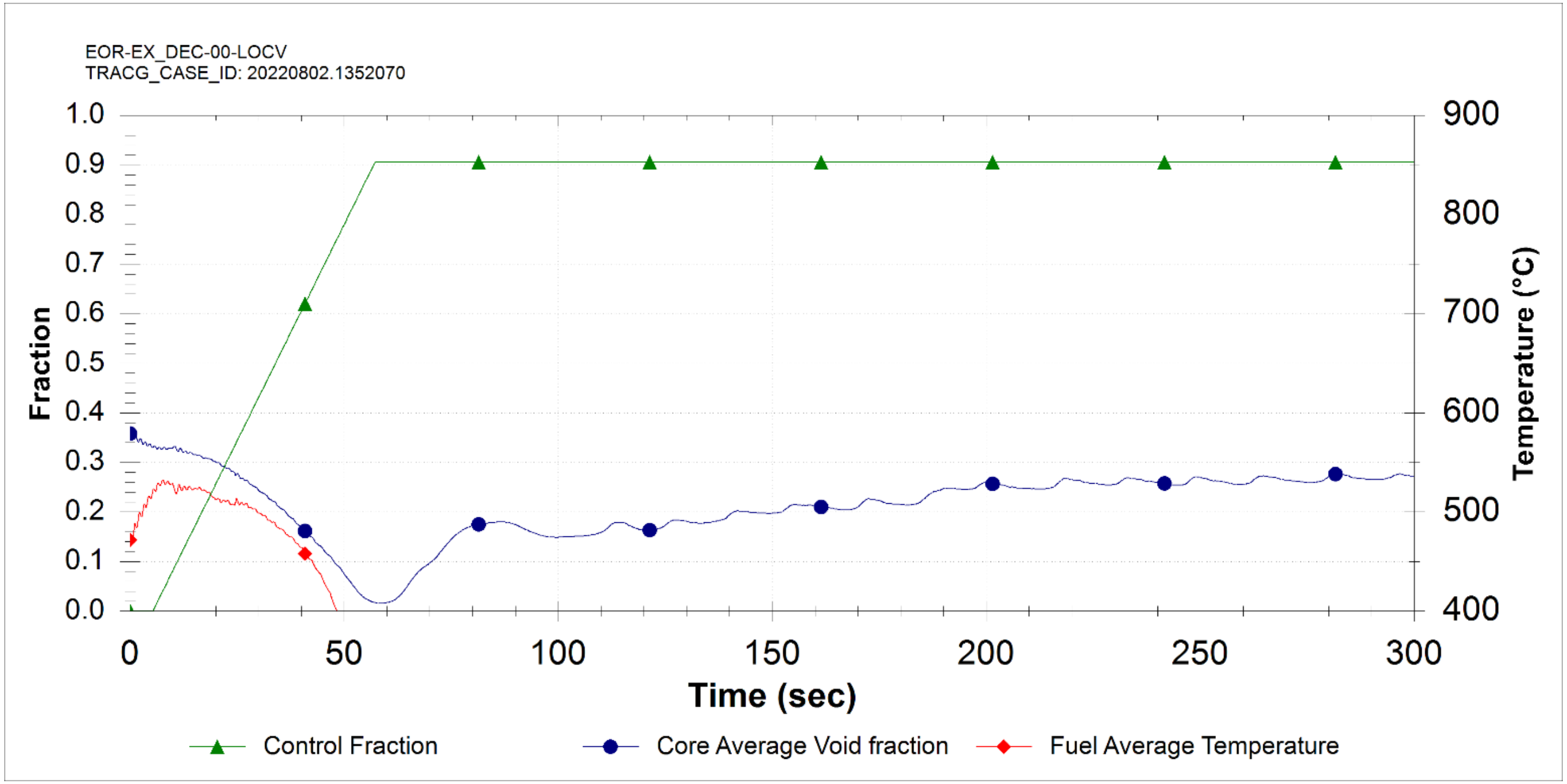

Figure 15.5-137: Loss of Condenser Vacuum (DEC)

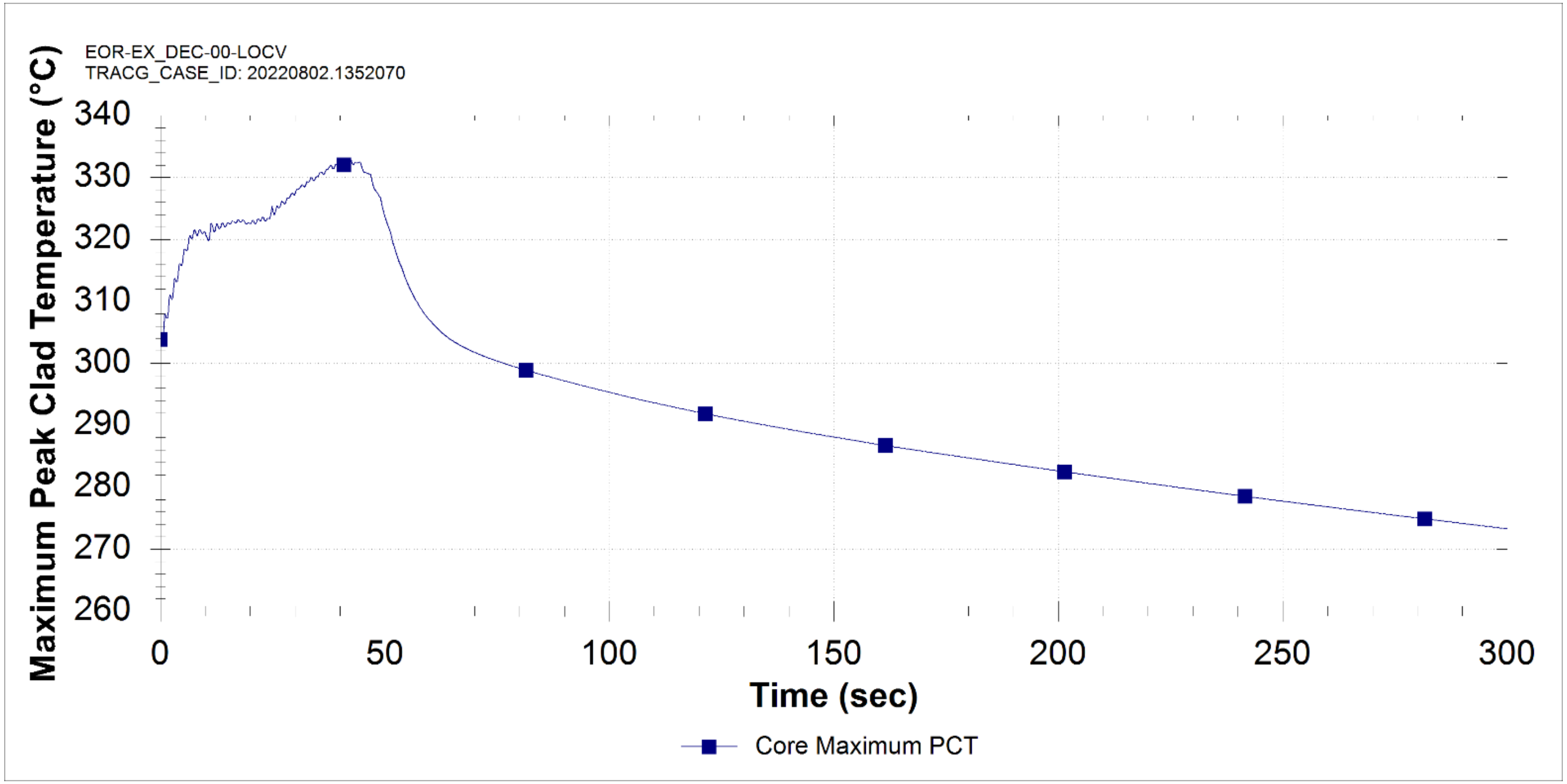

Figure 15.5-138: Loss of Condenser Vacuum (DEC)

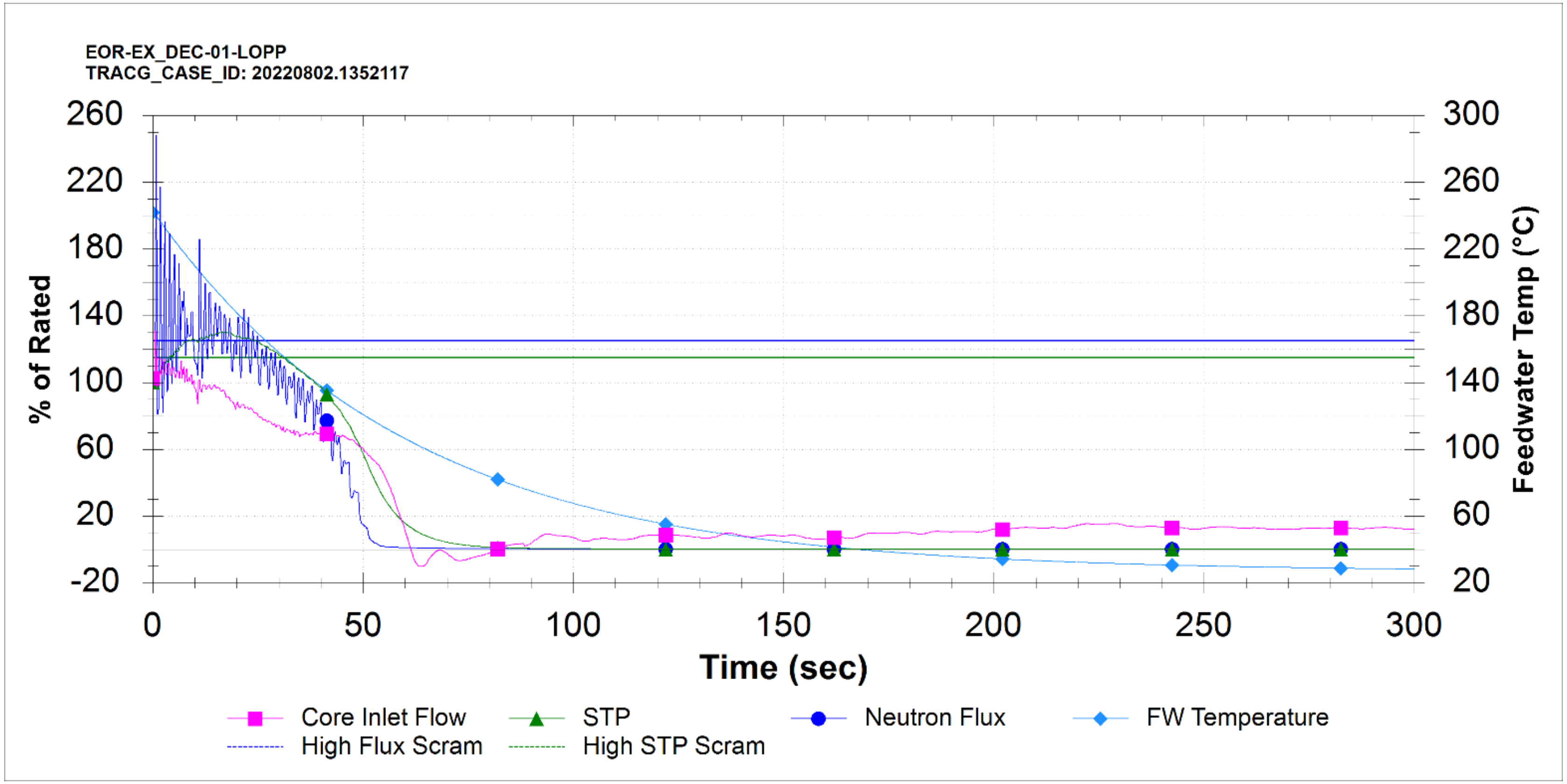

Figure 15.5-139: Loss of Preferred Power (DEC)

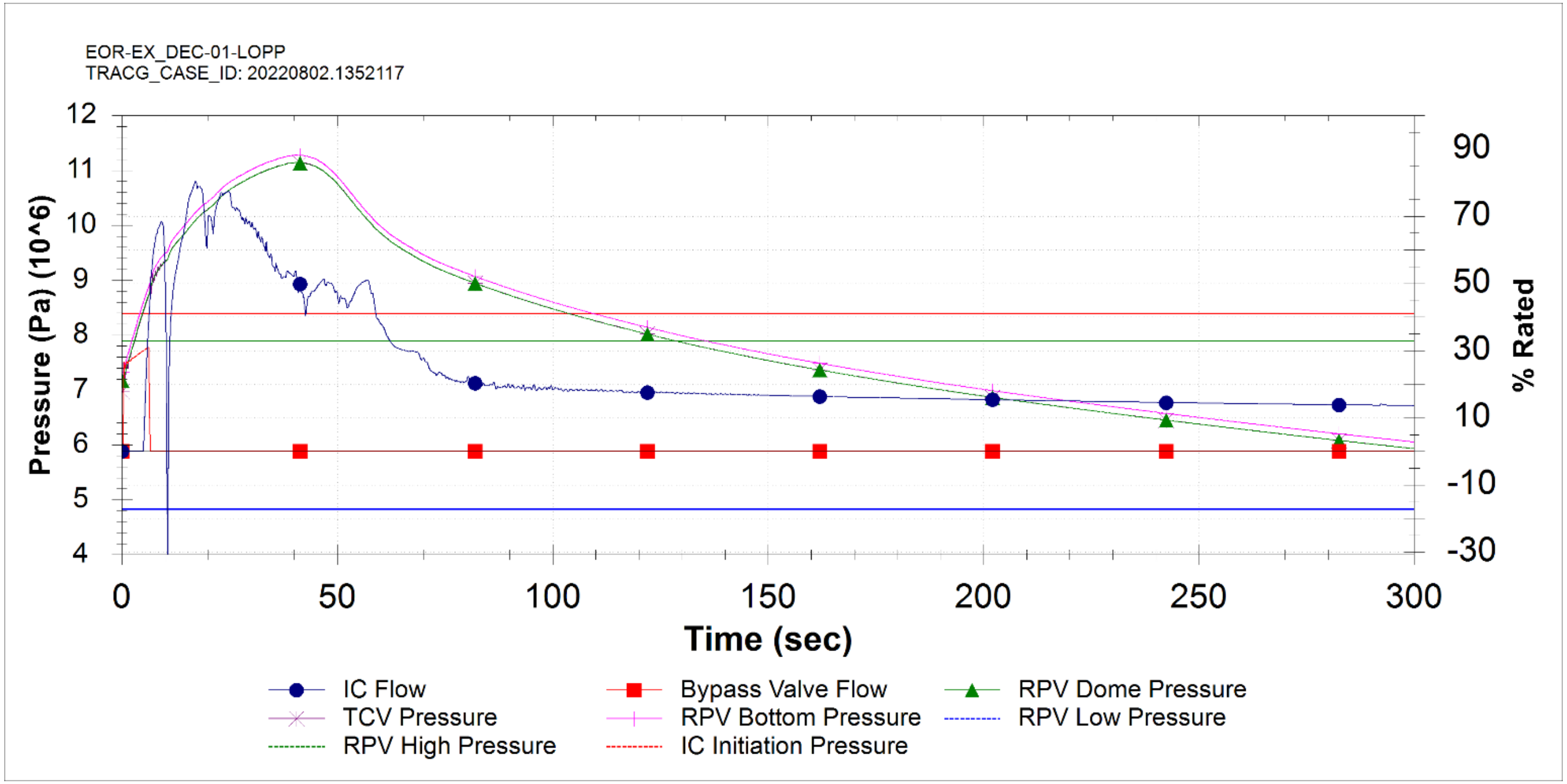

Figure 15.5-140: Loss of Preferred Power (DEC)

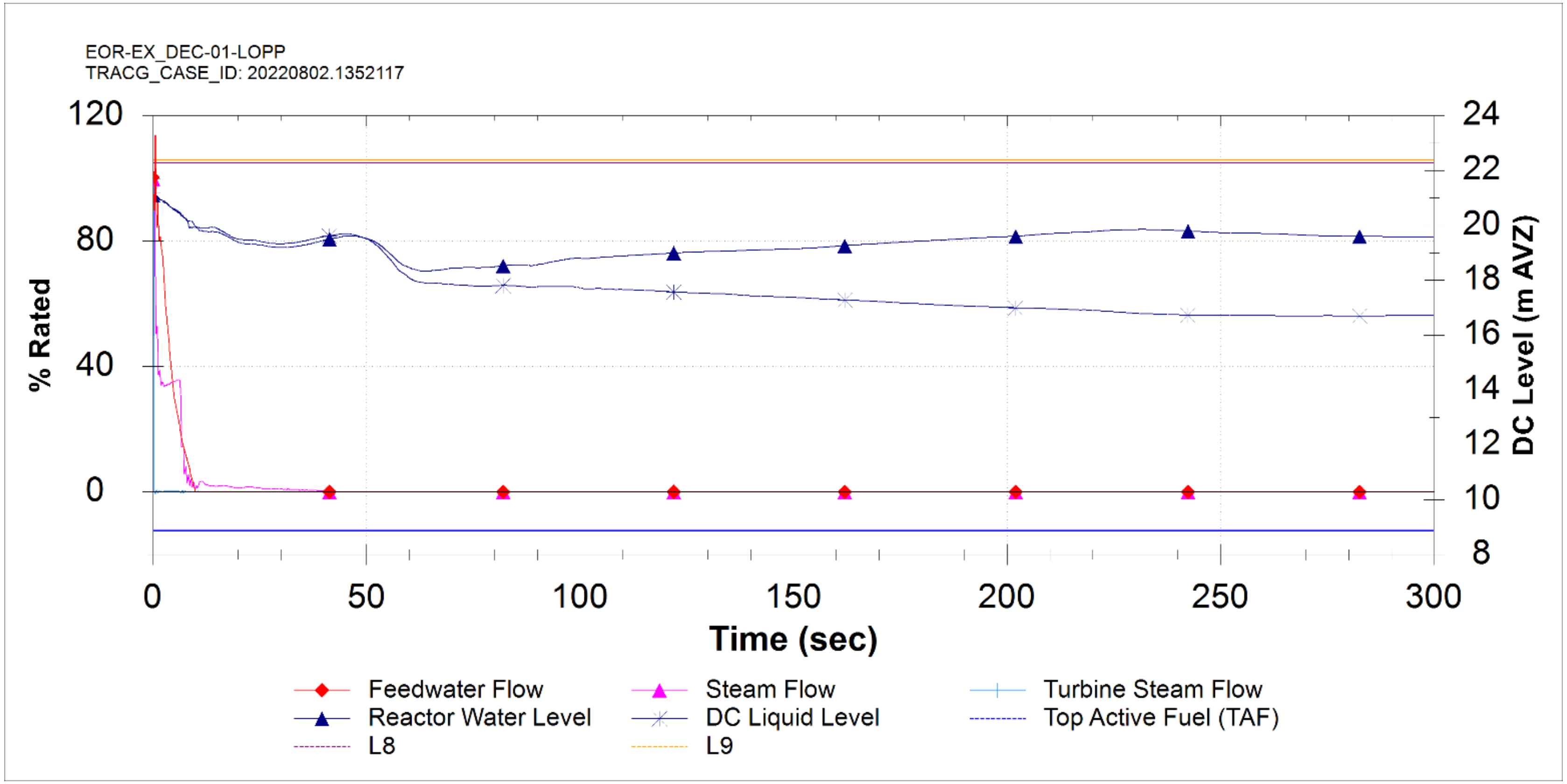

Figure 15.5-141: Loss of Preferred Power (DEC)

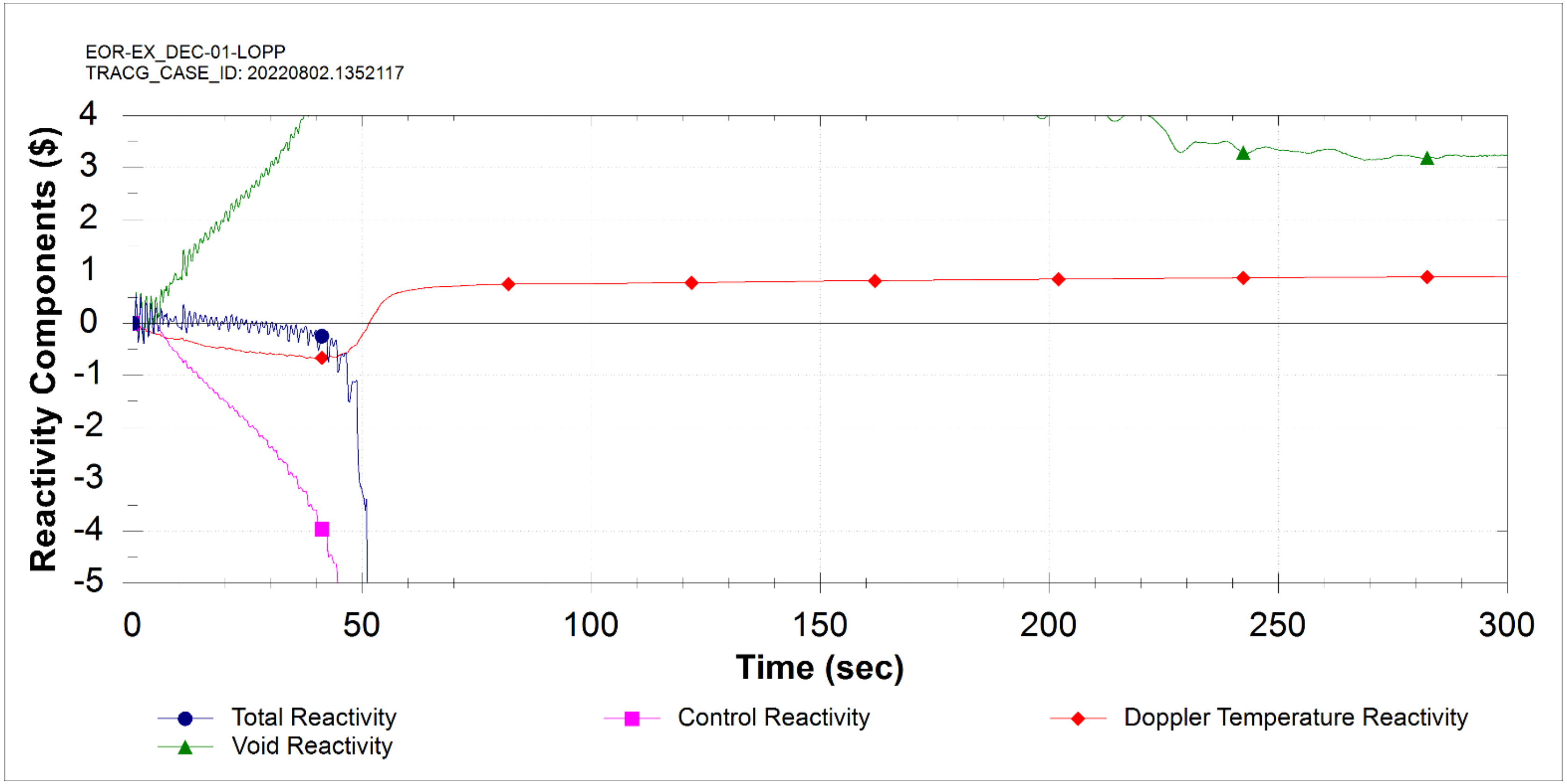

Figure 15.5-142: Loss of Preferred Power (DEC)

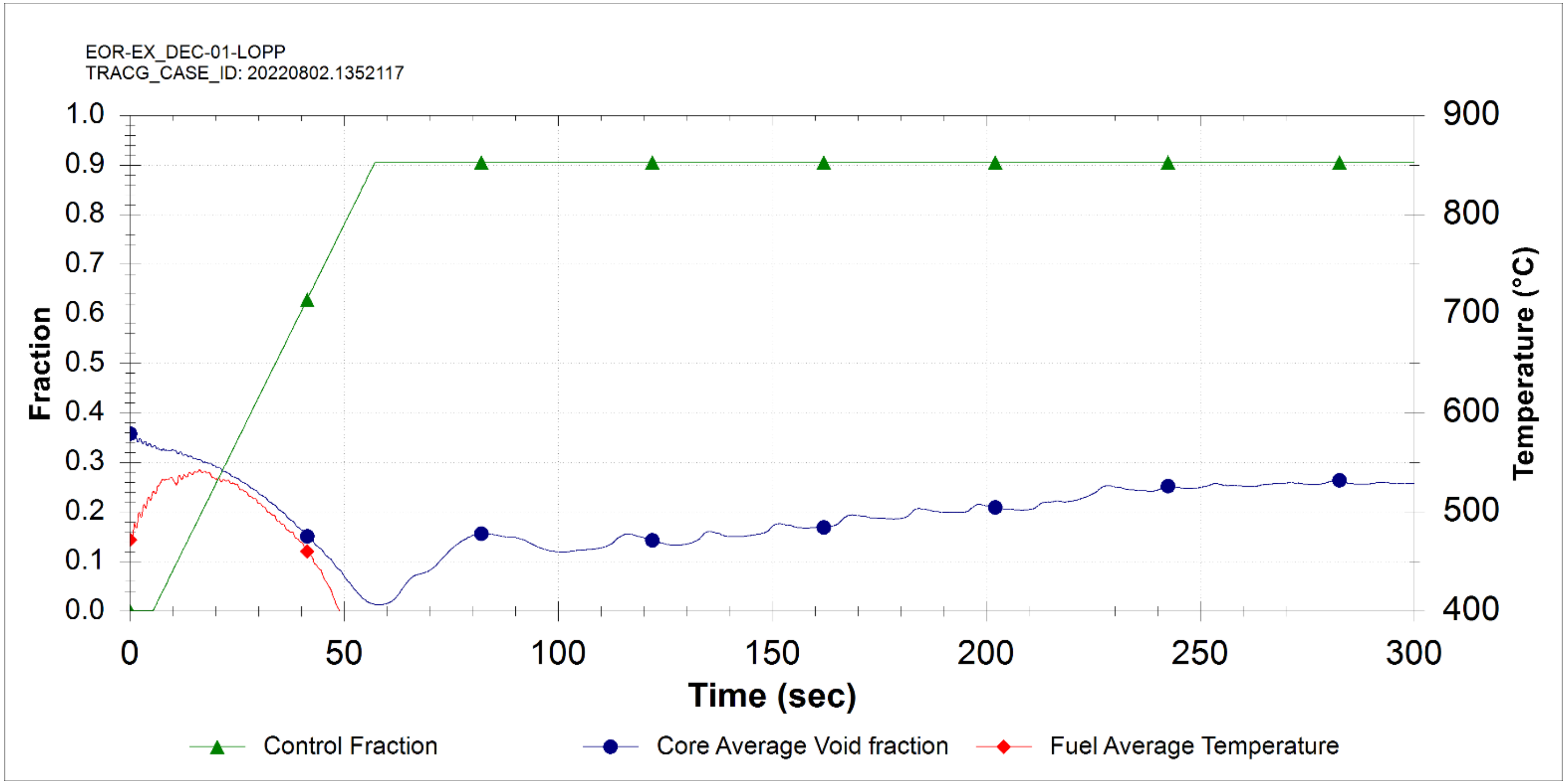

Figure 15.5-143: Loss of Preferred Power (DEC)

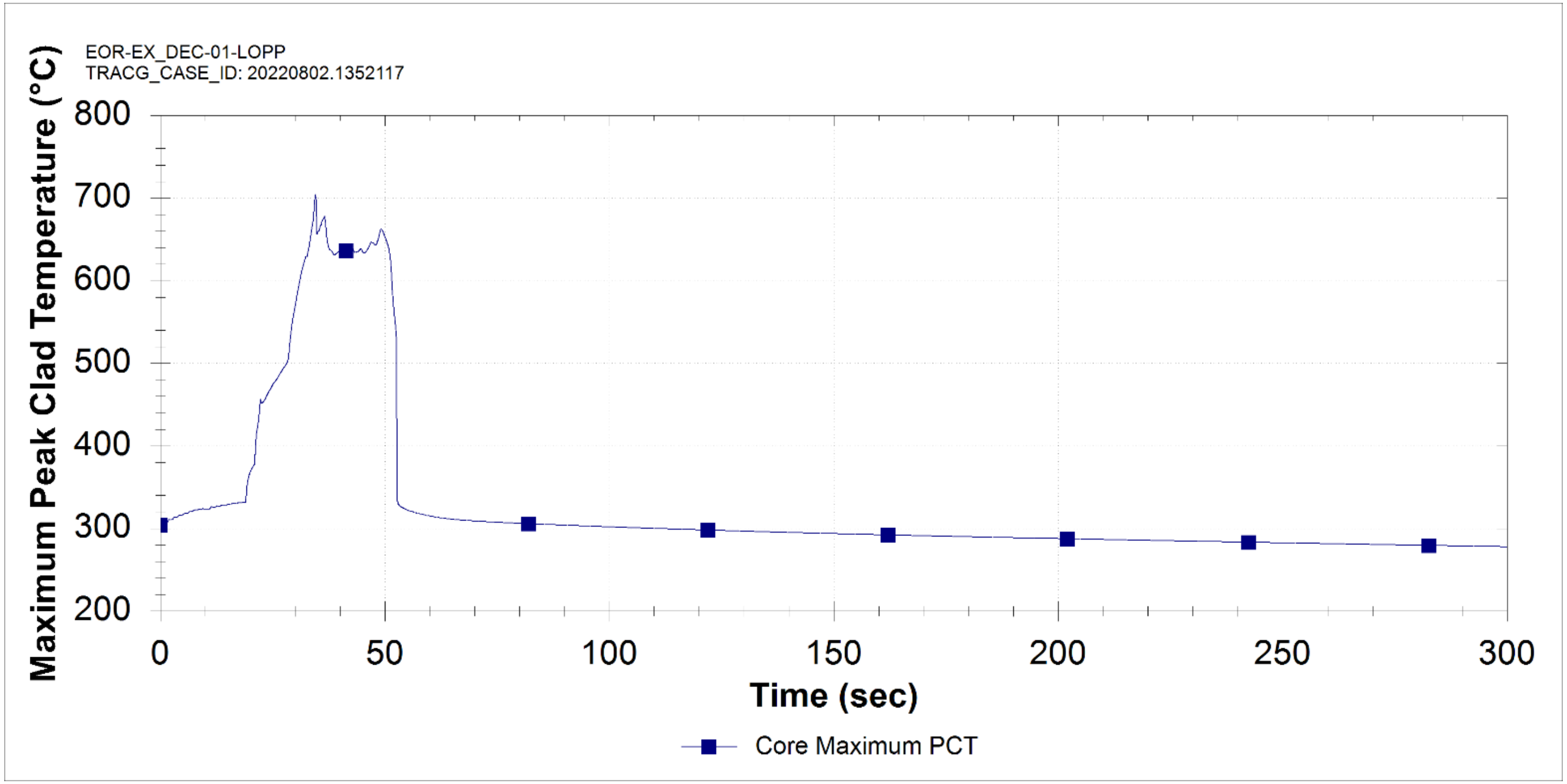

Figure 15.5-144: Loss of Preferred Power (DEC)

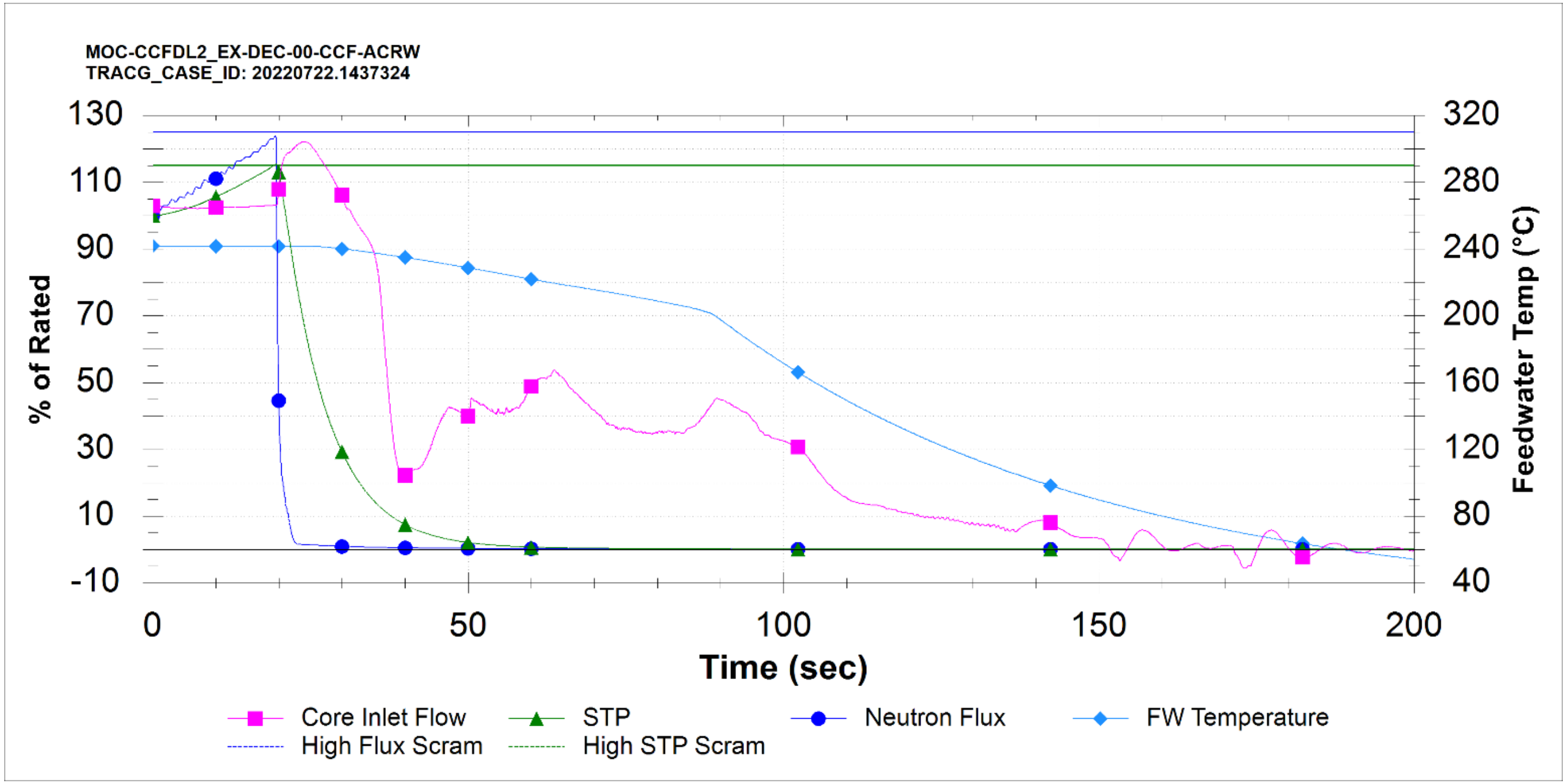

Figure 15.5-145: All Control Rod Withdrawal at Power (ACRW)

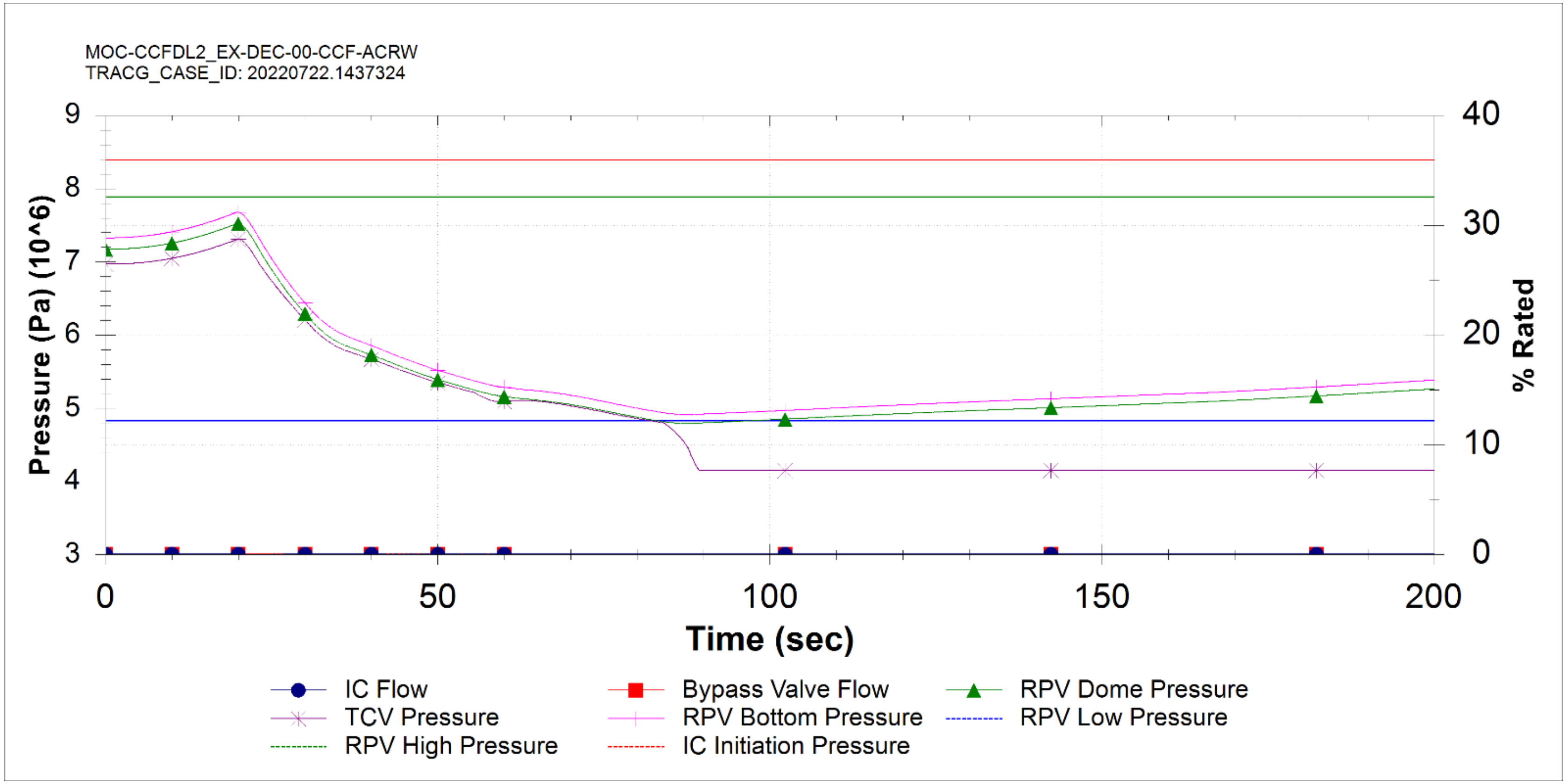

Figure 15.5-146: All Control Rod Withdrawal at Power (ACRW)

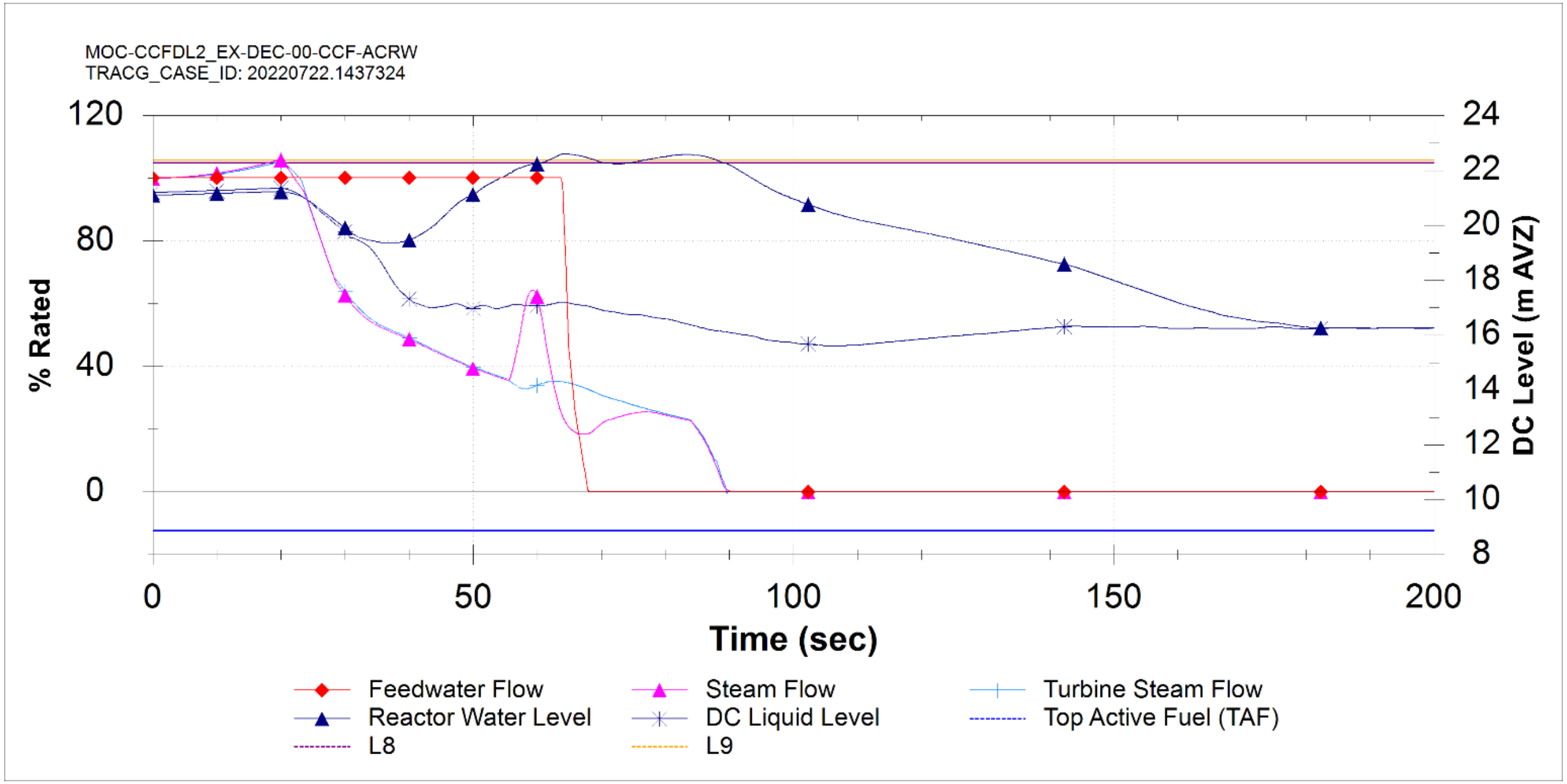

Figure 15.5-147: All Control Rod Withdrawal at Power (ACRW)

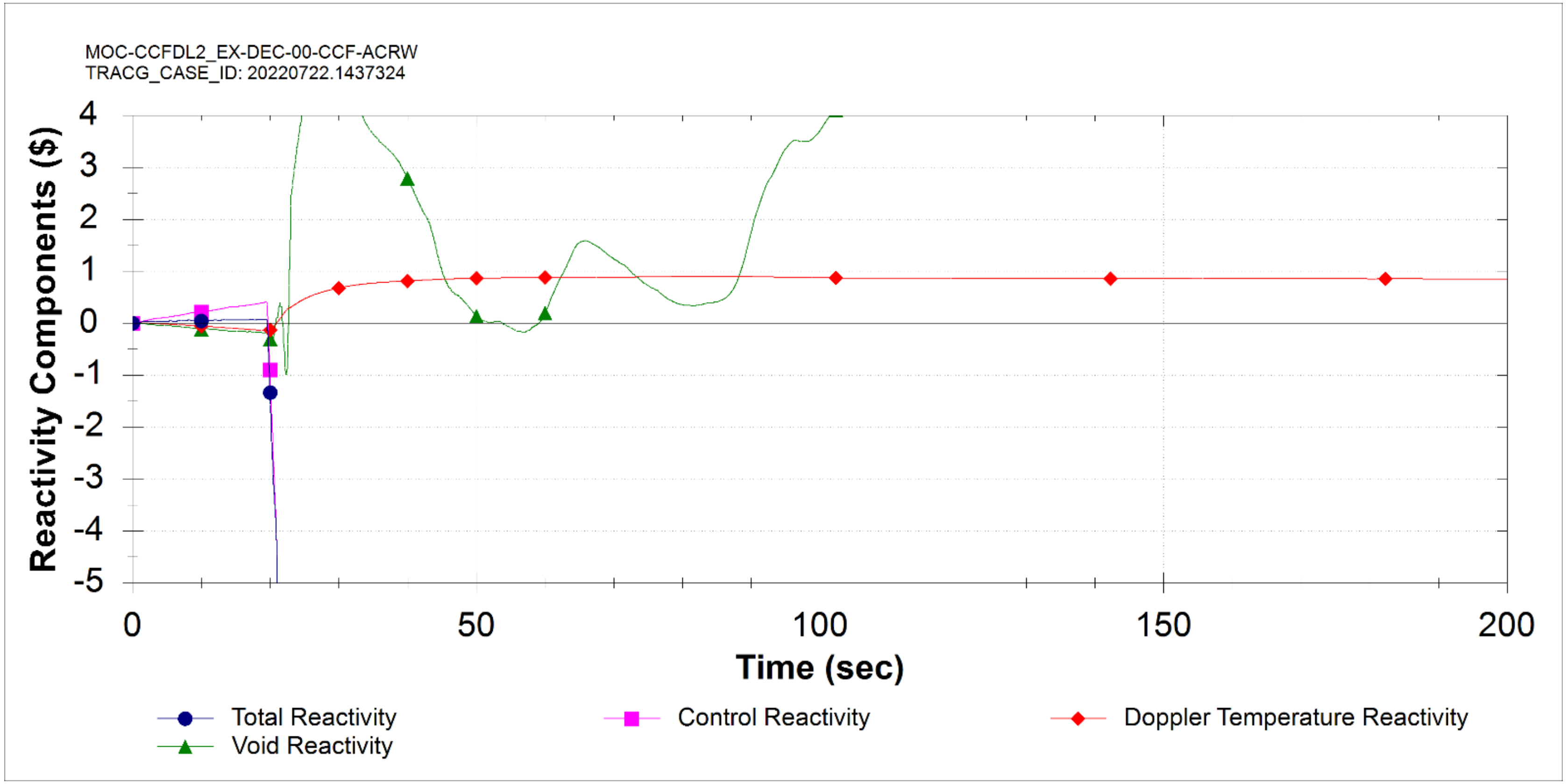

Figure 15.5-148: All Control Rod Withdrawal at Power (ACRW)

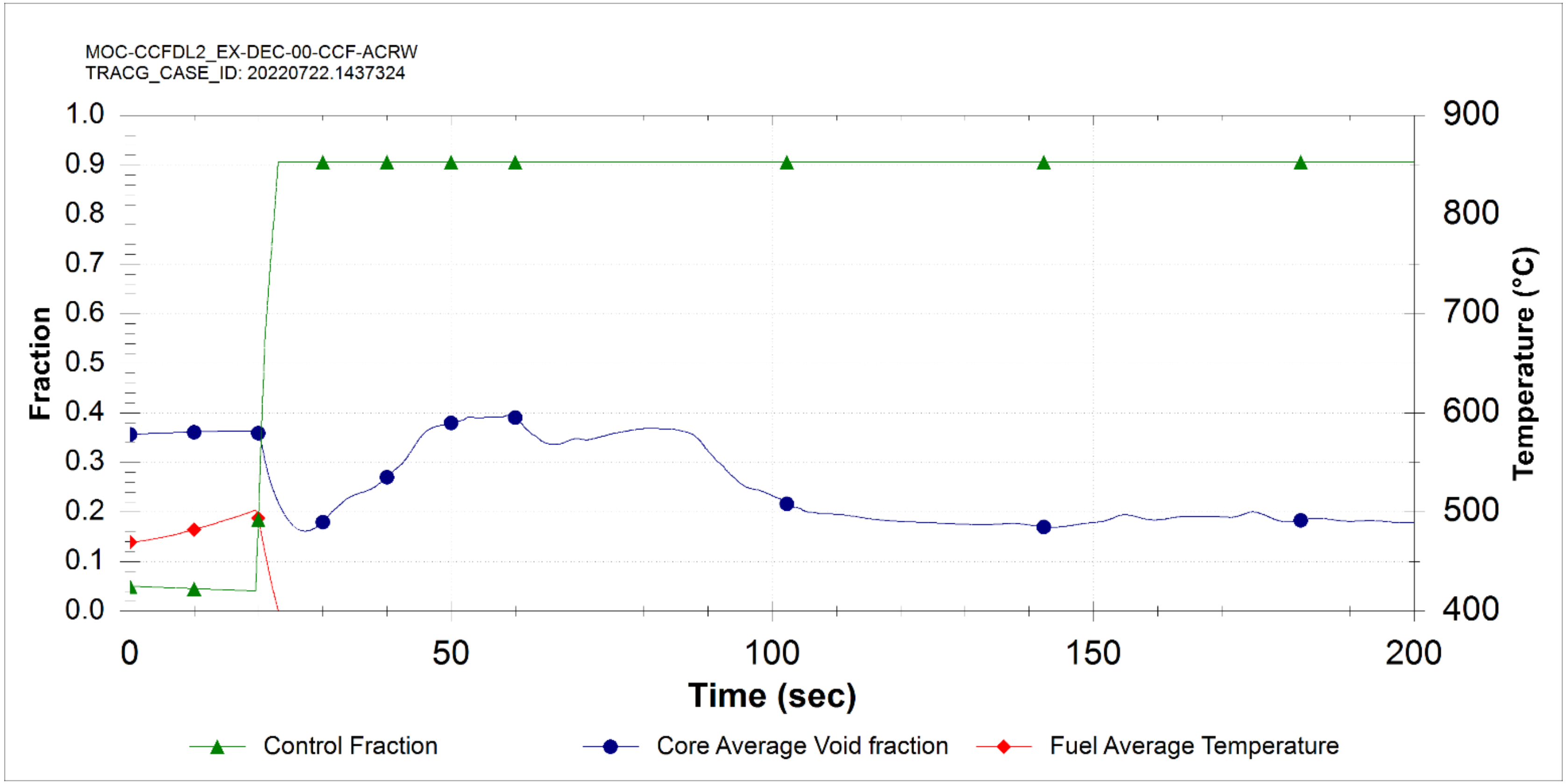

Figure 15.5-149: All Control Rod Withdrawal at Power (ACRW)

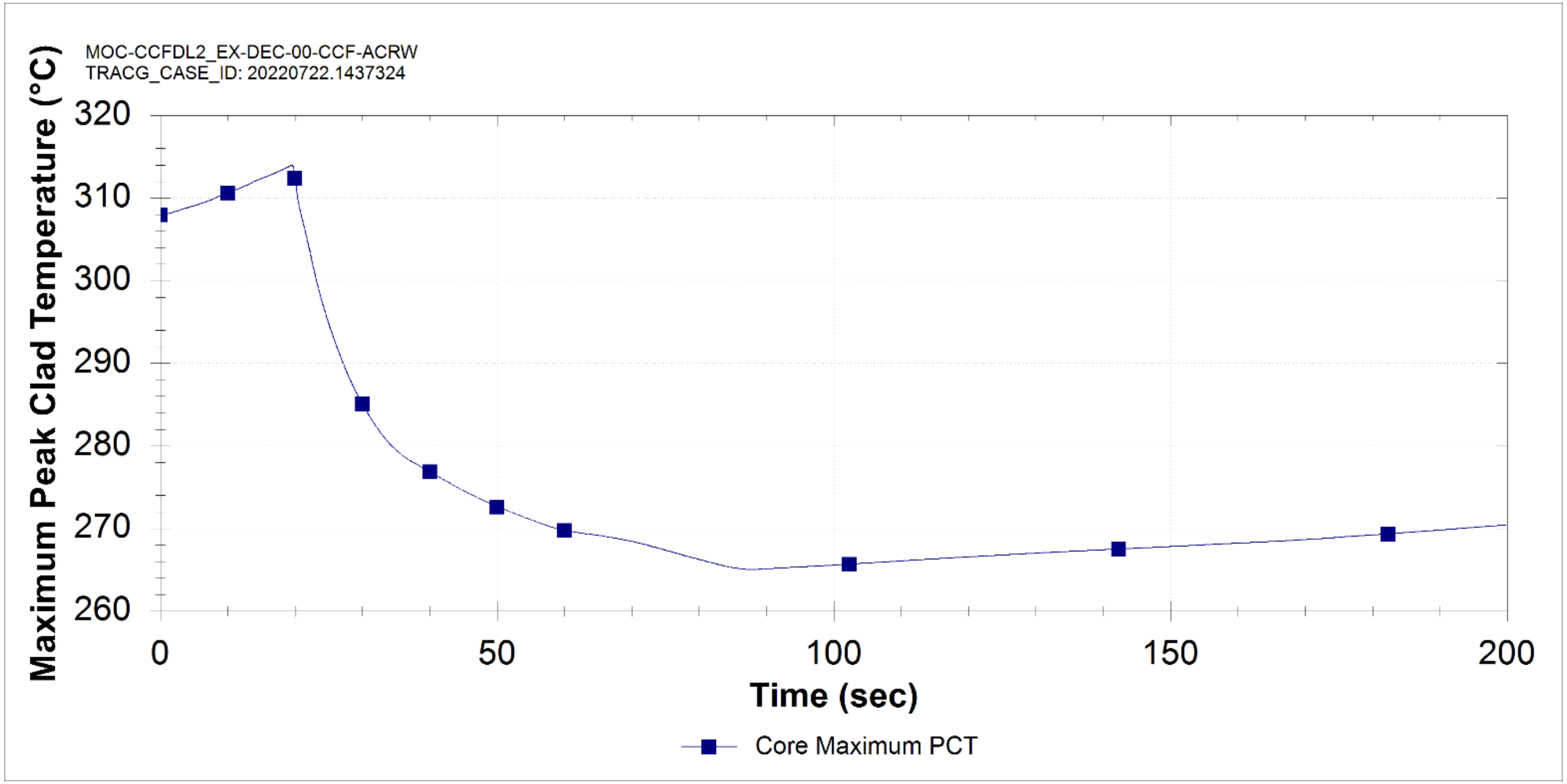

Figure 15.5-150: All Control Rod Withdrawal at Power (ACRW)

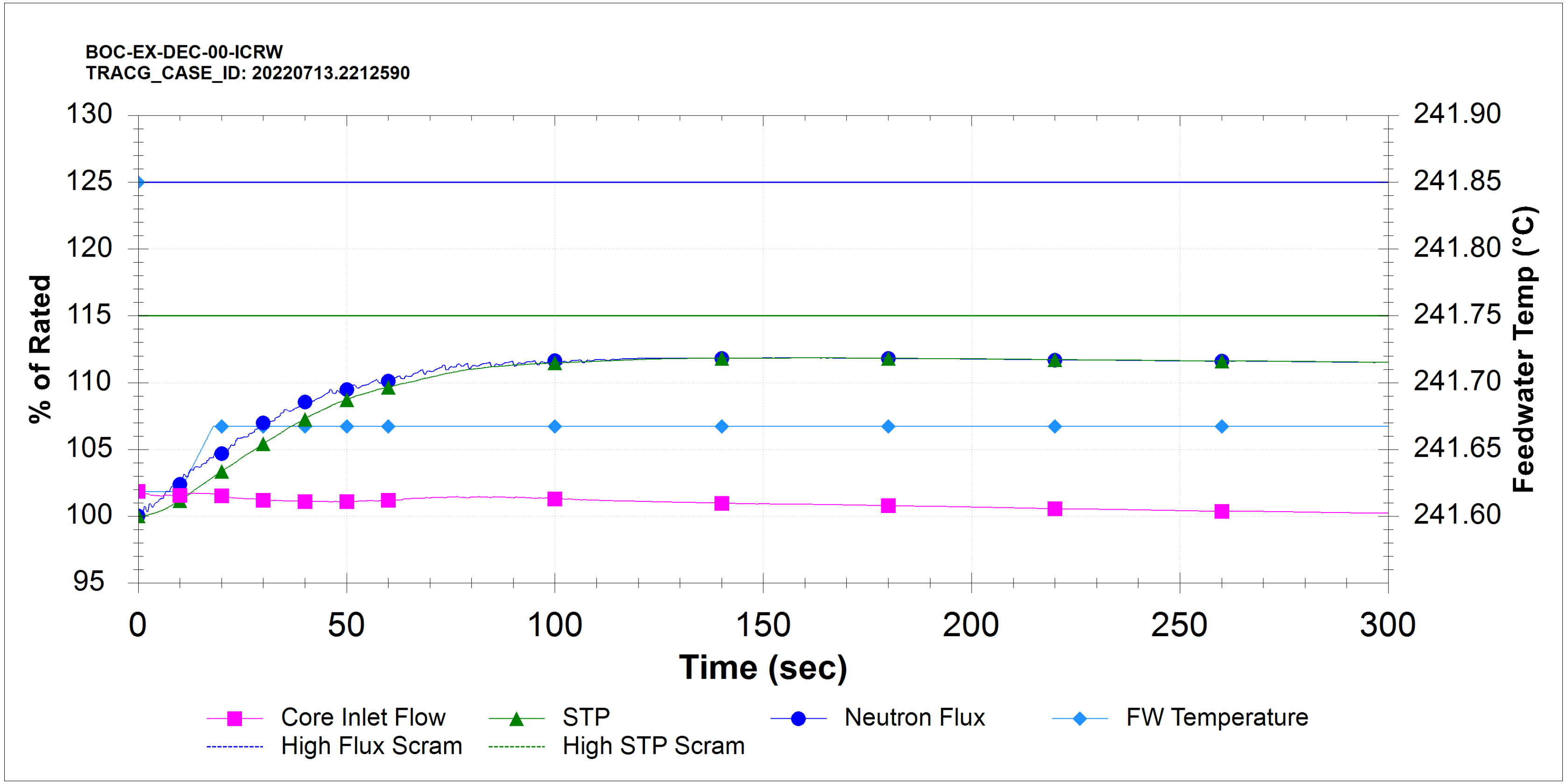

Figure 15.5-151: Inadvertent Control Rod Withdrawal at Power – Single Rod (ICRW)

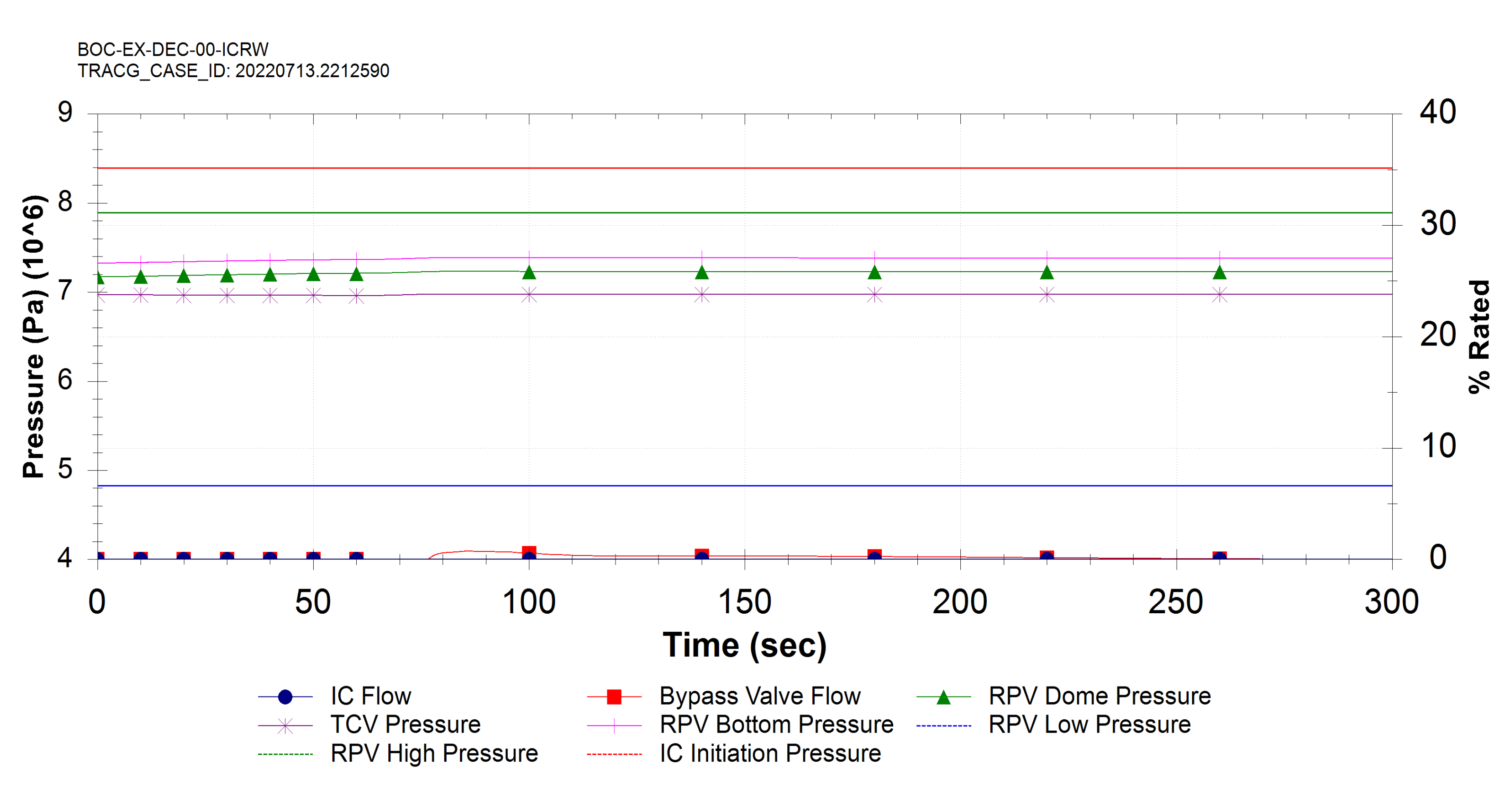

Figure 15.5-152: Inadvertent Control Rod Withdrawal at Power – Single Rod (ICRW)

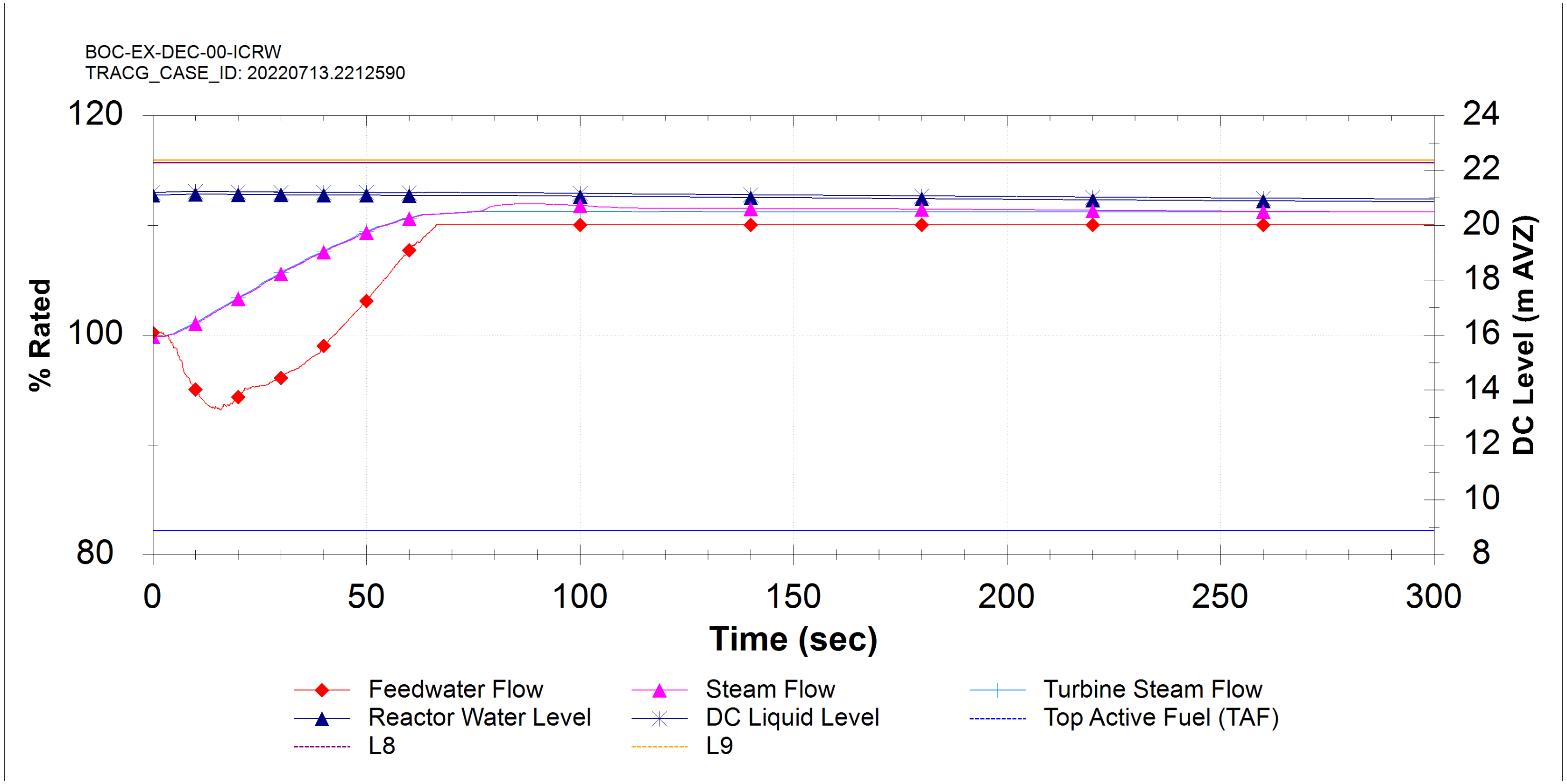

Figure 15.5-153: Inadvertent Control Rod Withdrawal at Power – Single Rod (ICRW)

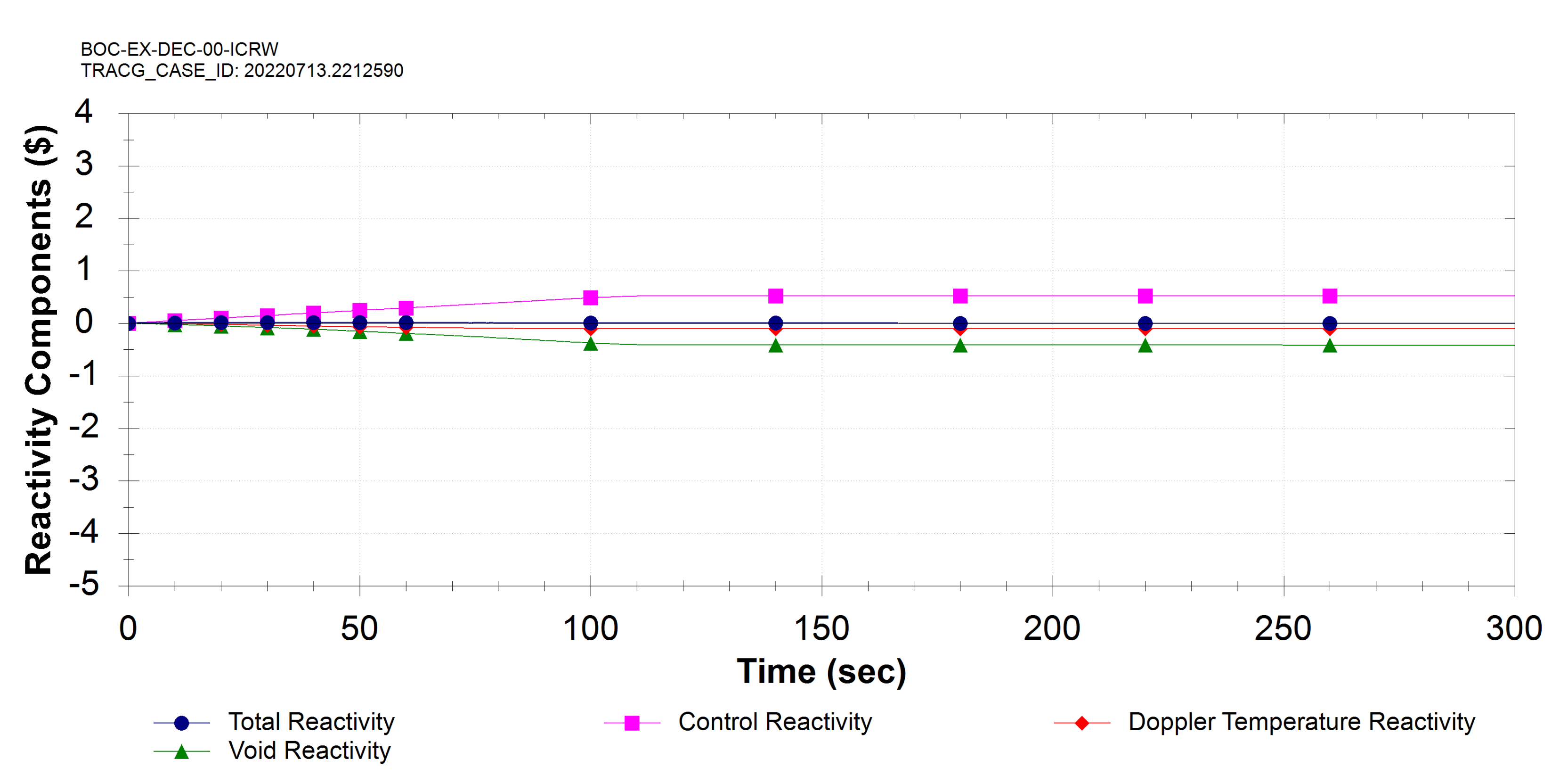

Figure 15.5-154: Inadvertent Control Rod Withdrawal at Power – Single Rod (ICRW)

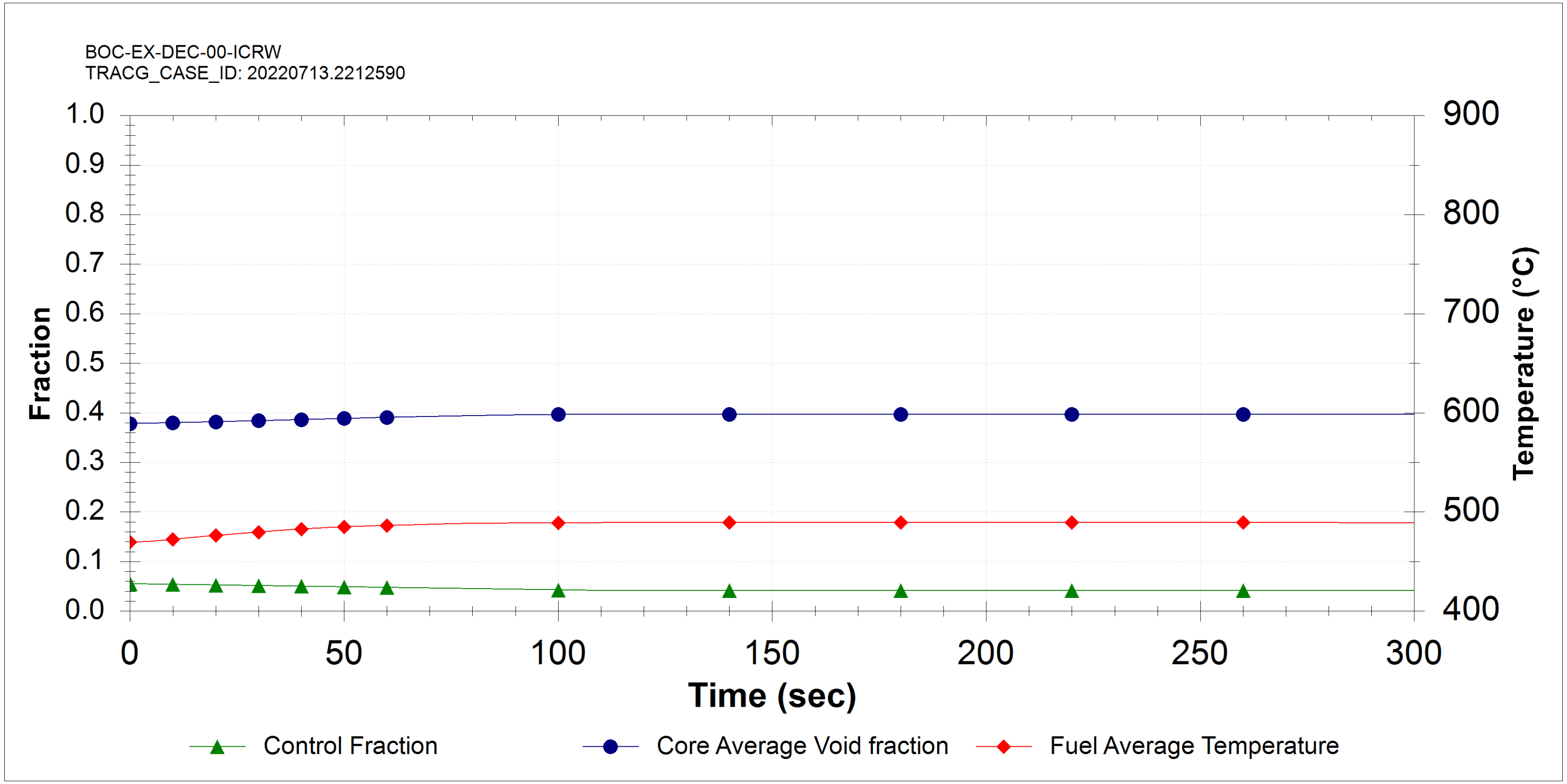

Figure 15.5-155: Inadvertent Control Rod Withdrawal at Power – Single Rod (ICRW)

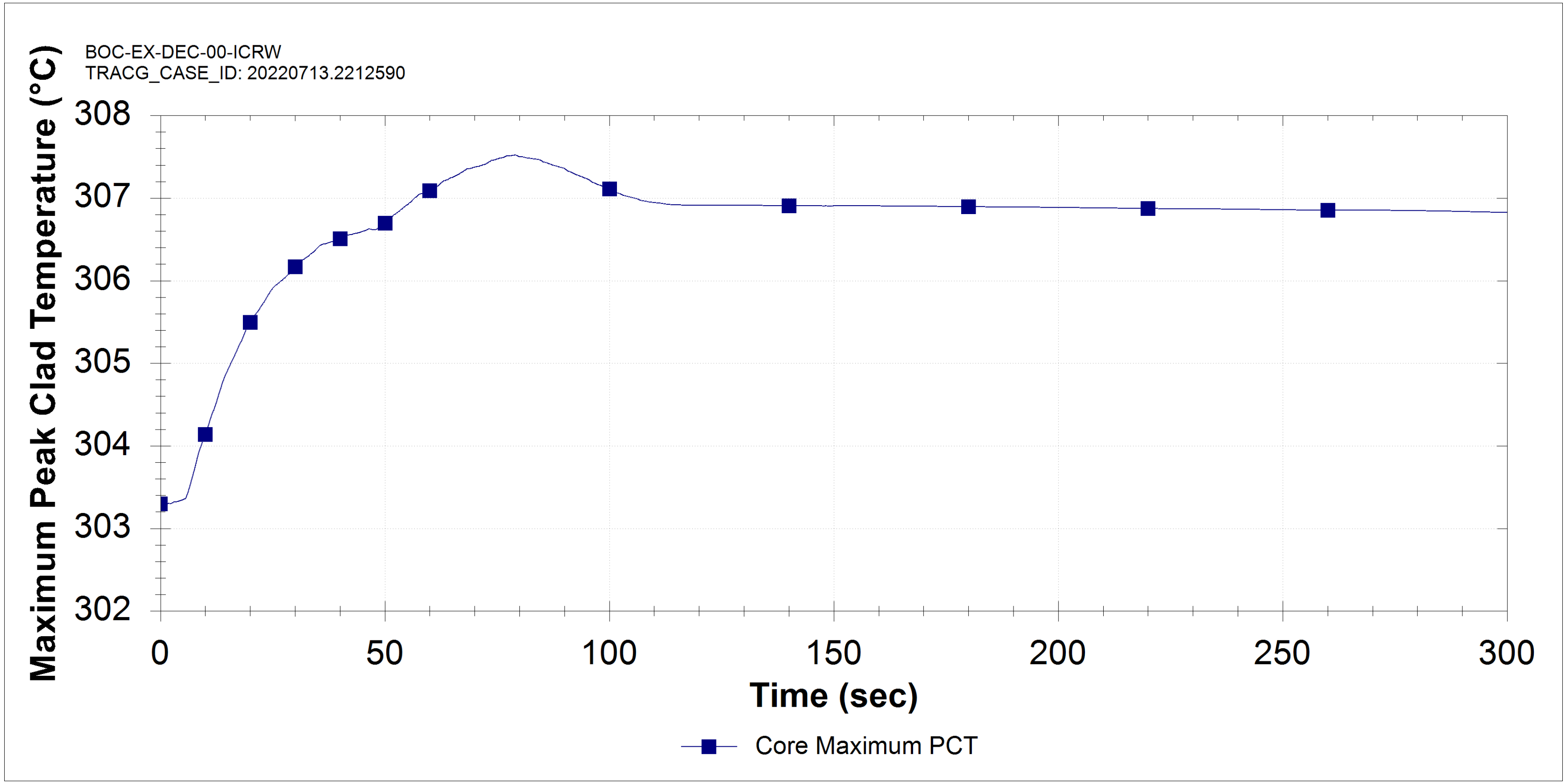

Figure 15.5-156: Inadvertent Control Rod Withdrawal at Power – Single Rod (ICRW)

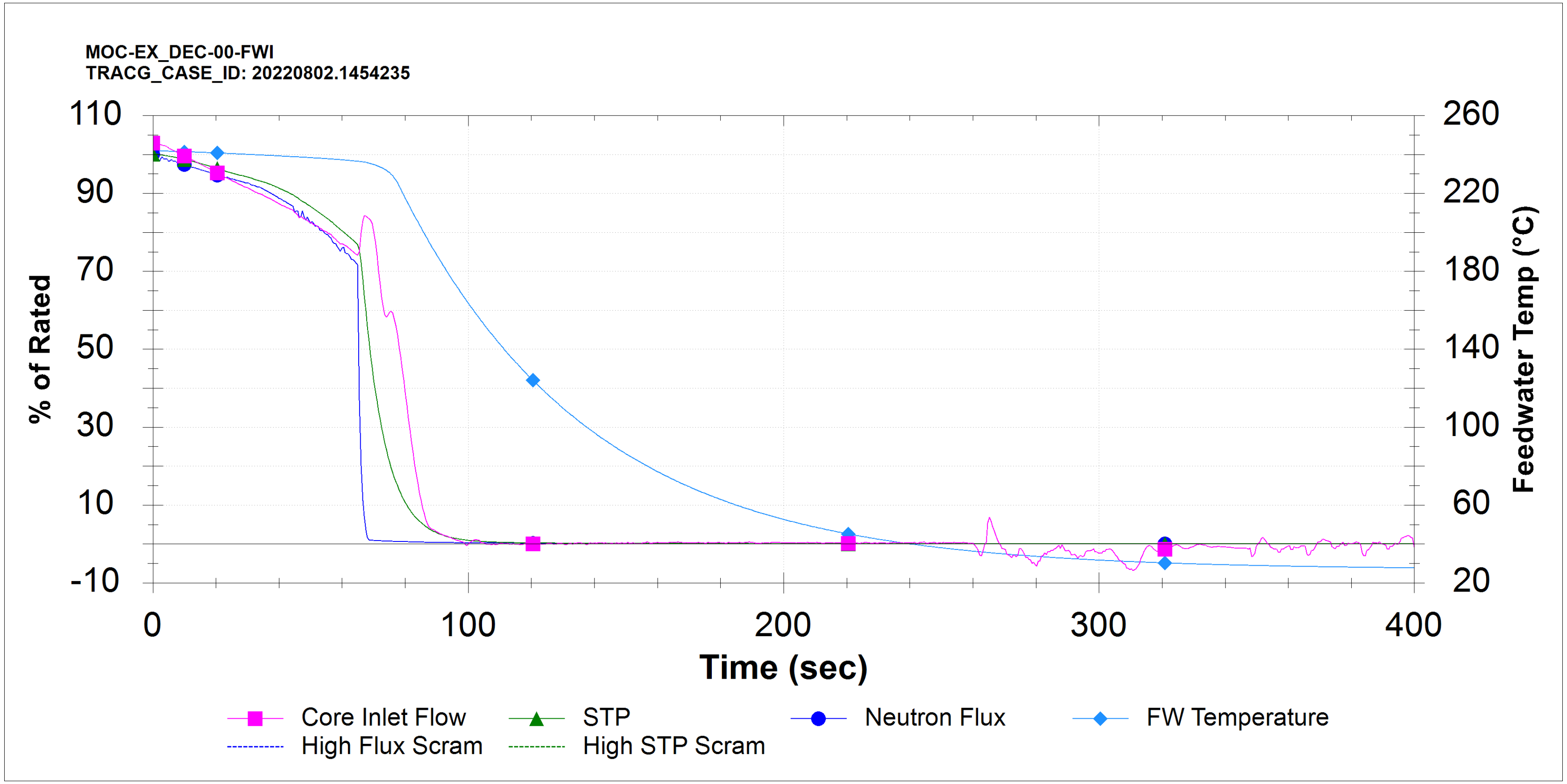

Figure 15.5-157: Feedwater Isolation (DEC)

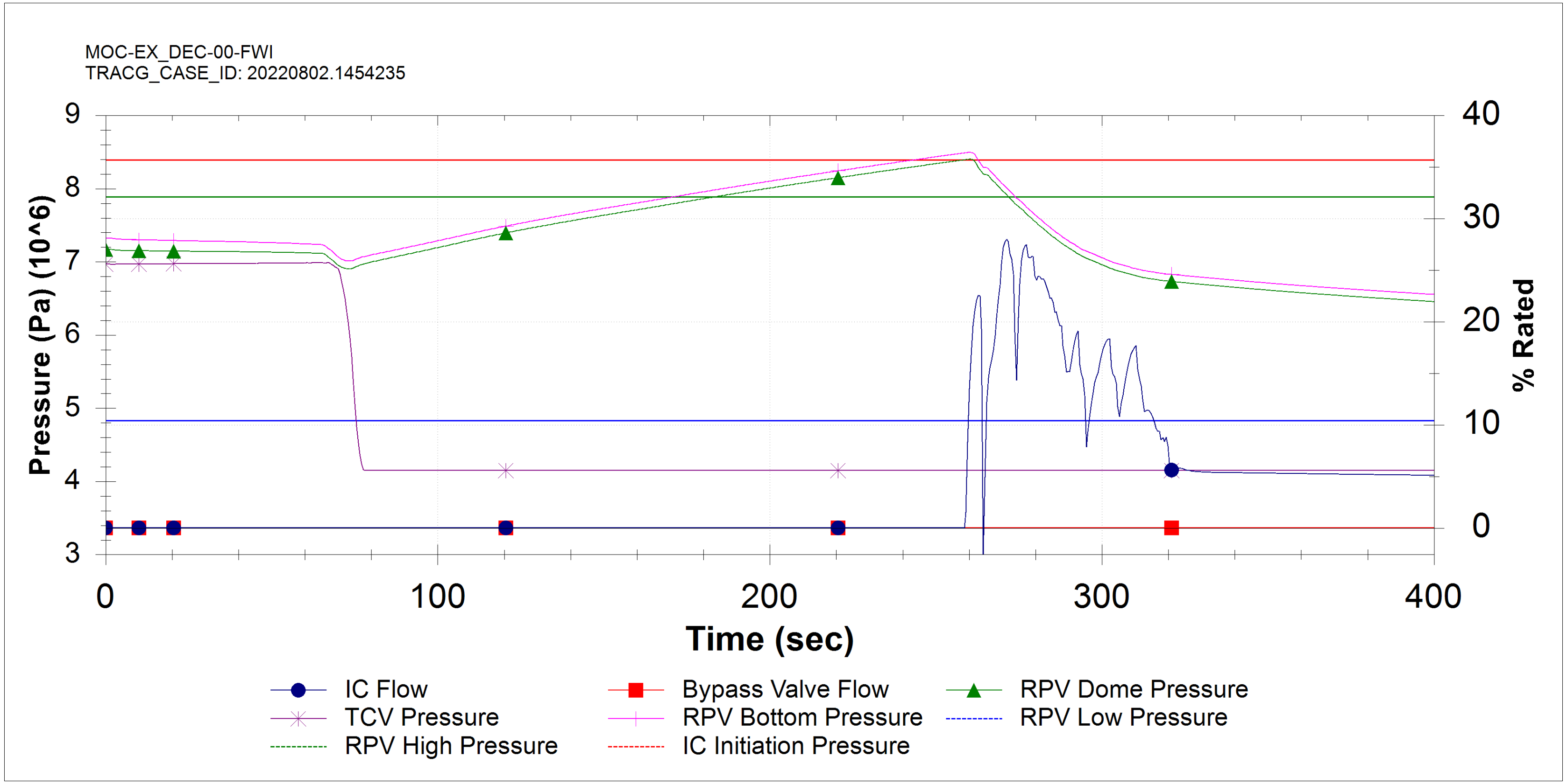

Figure 15.5-158: Feedwater Isolation (DEC)

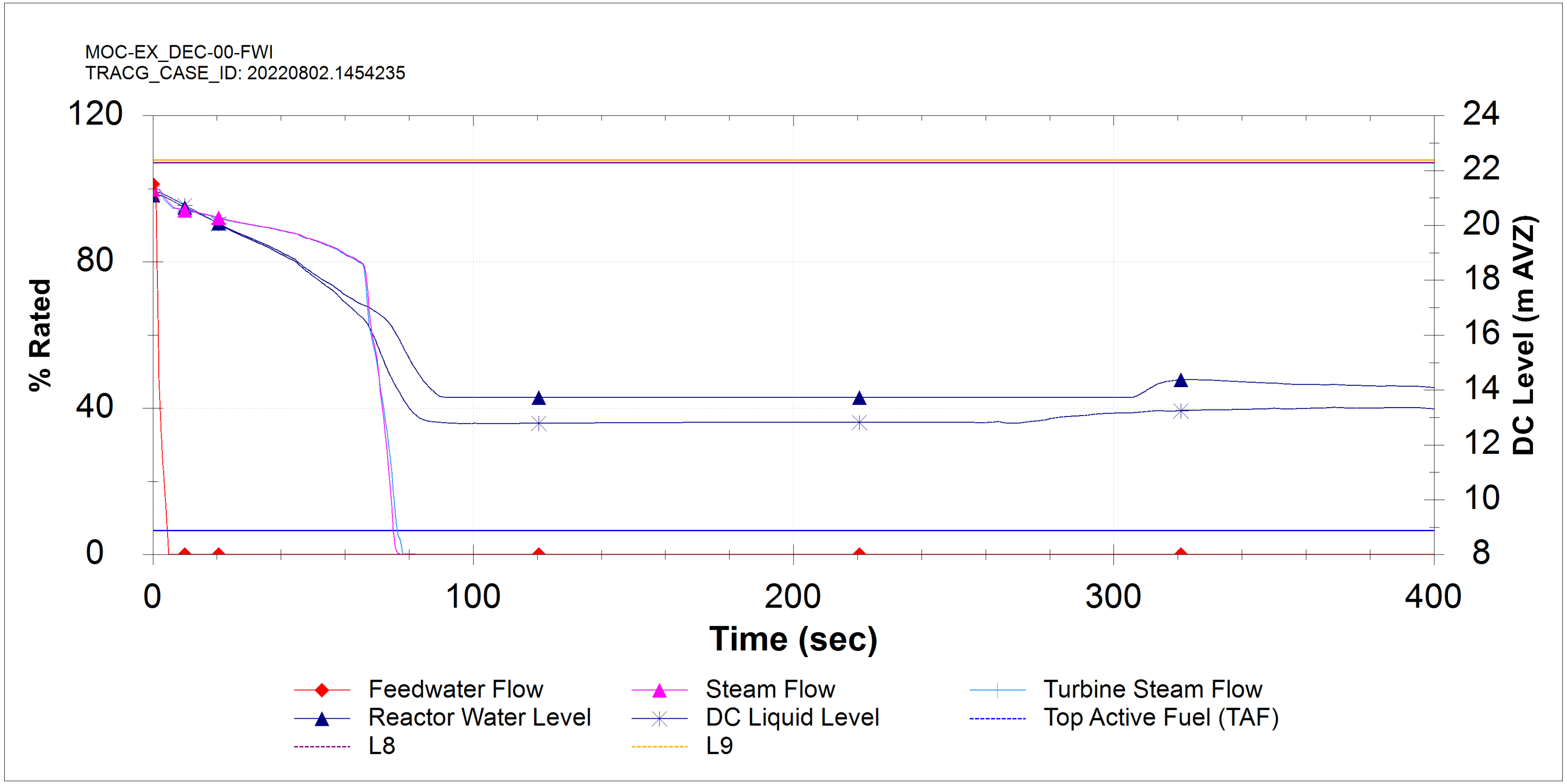

Figure 15.5-159: Feedwater Isolation (DEC)

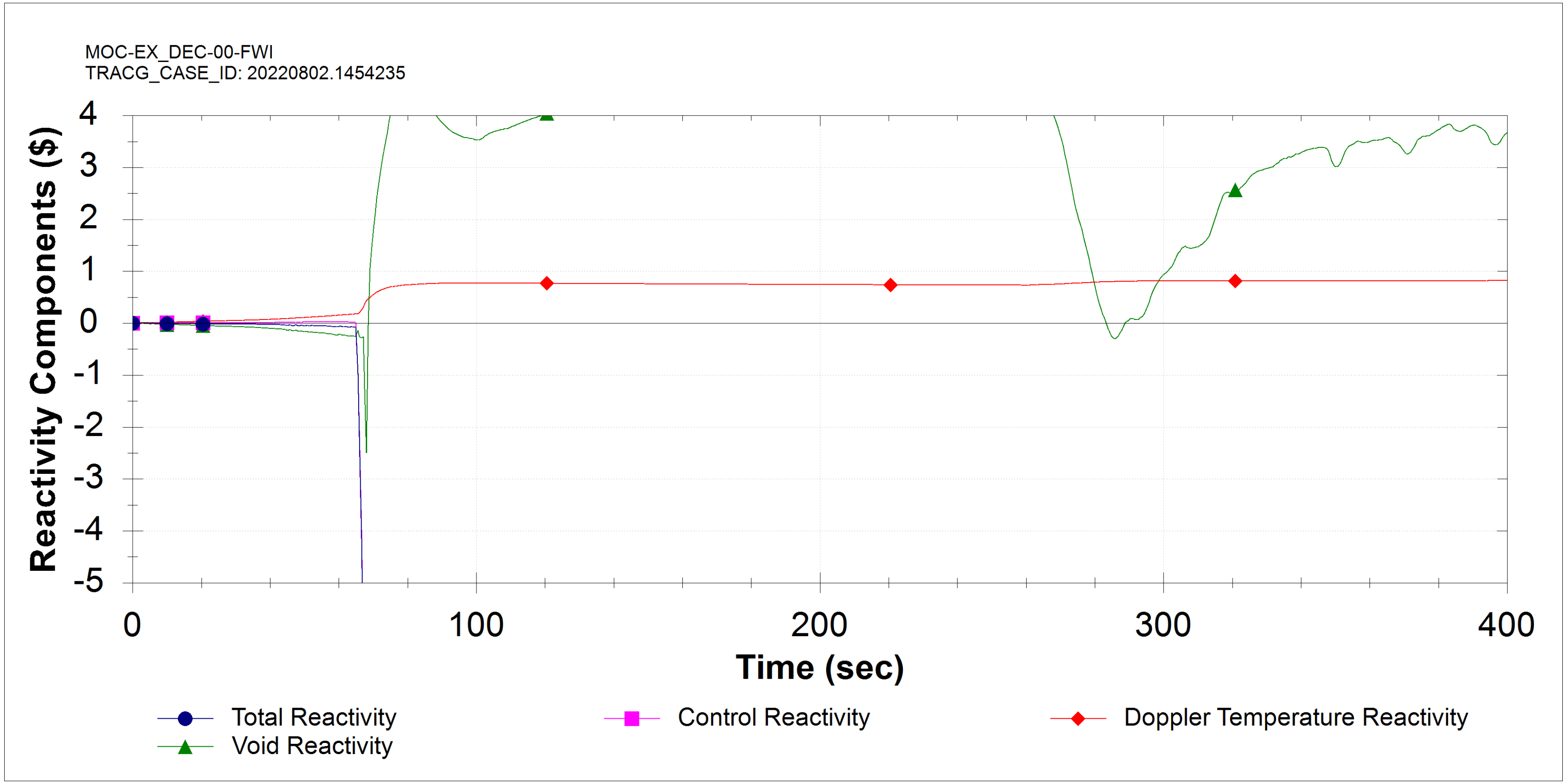

Figure 15.5-160: Feedwater Isolation (DEC)

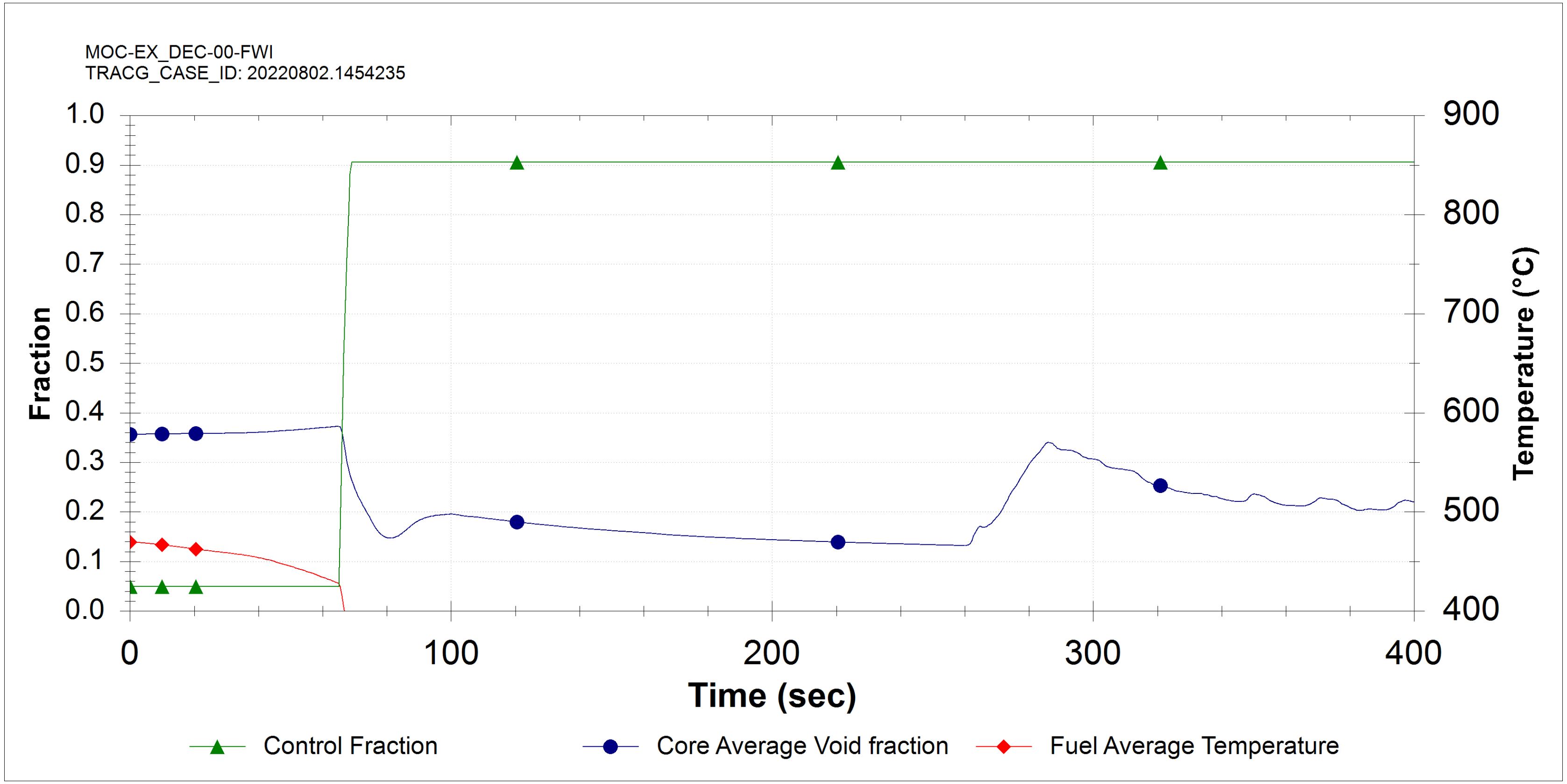

Figure 15.5-161: Feedwater Isolation (DEC)

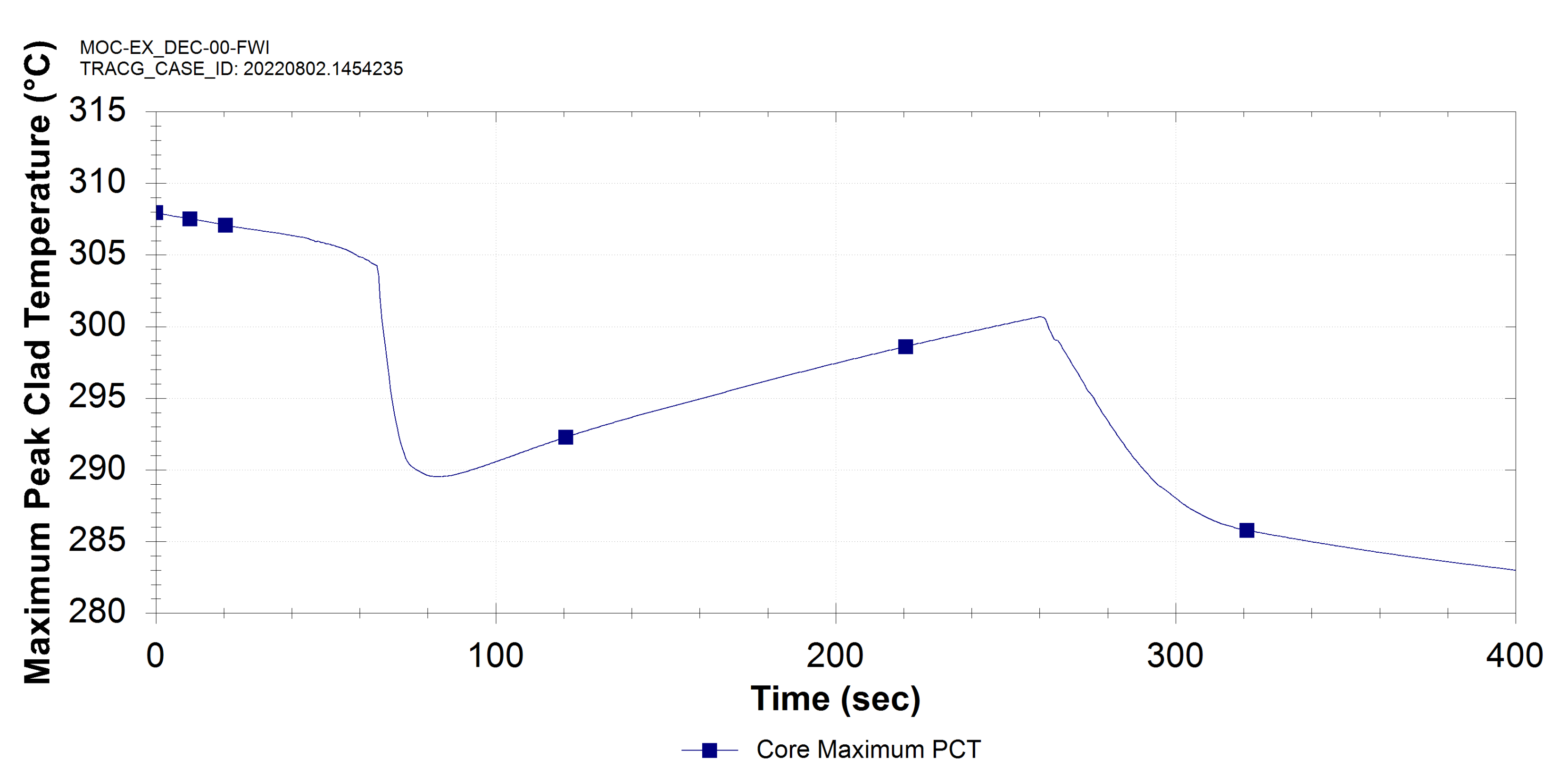

Figure 15.5-162: Feedwater Isolation (DEC)

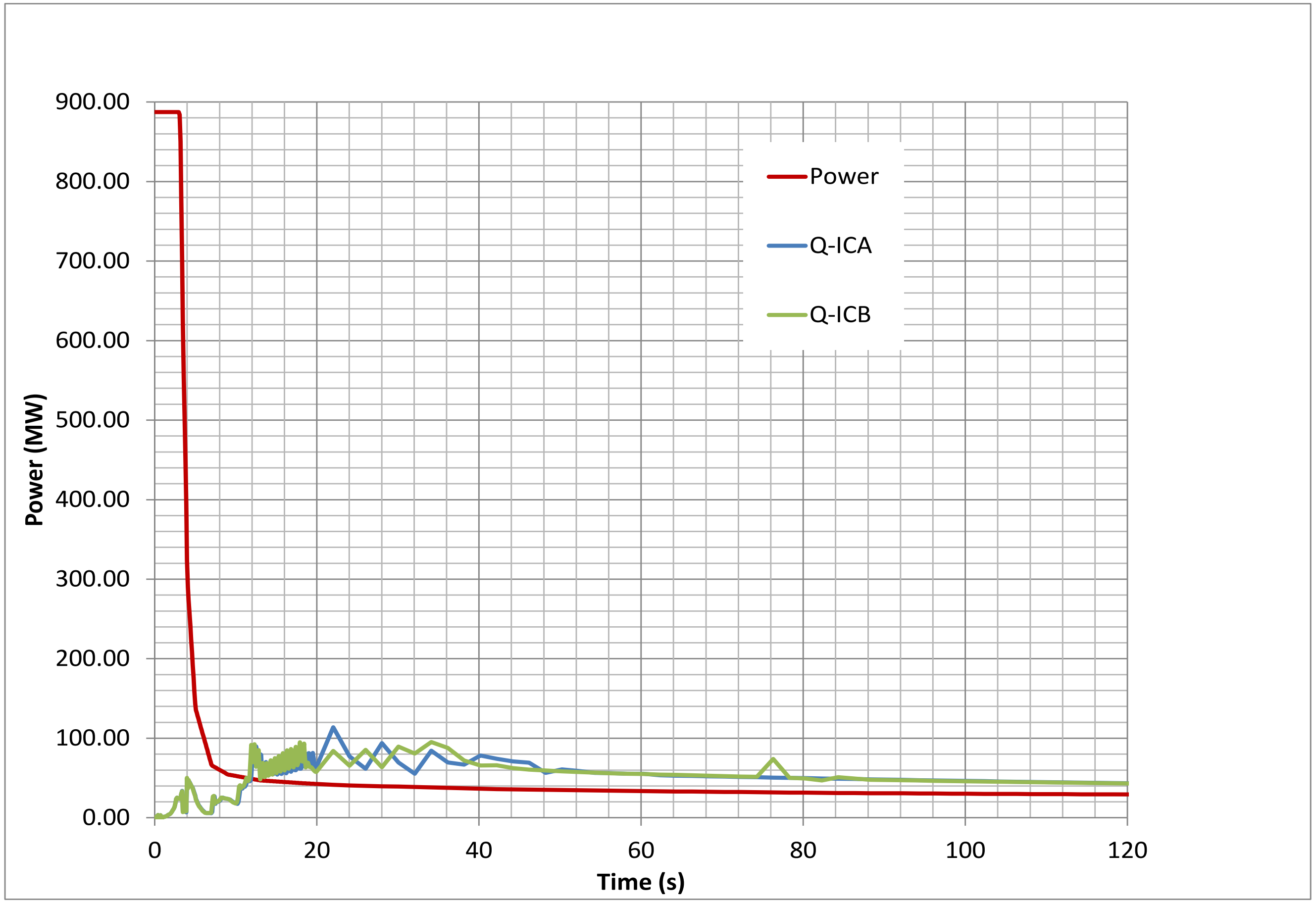

Figure 15.5-164: Reactor Power, Large FW Pipe Break, Conservative Case

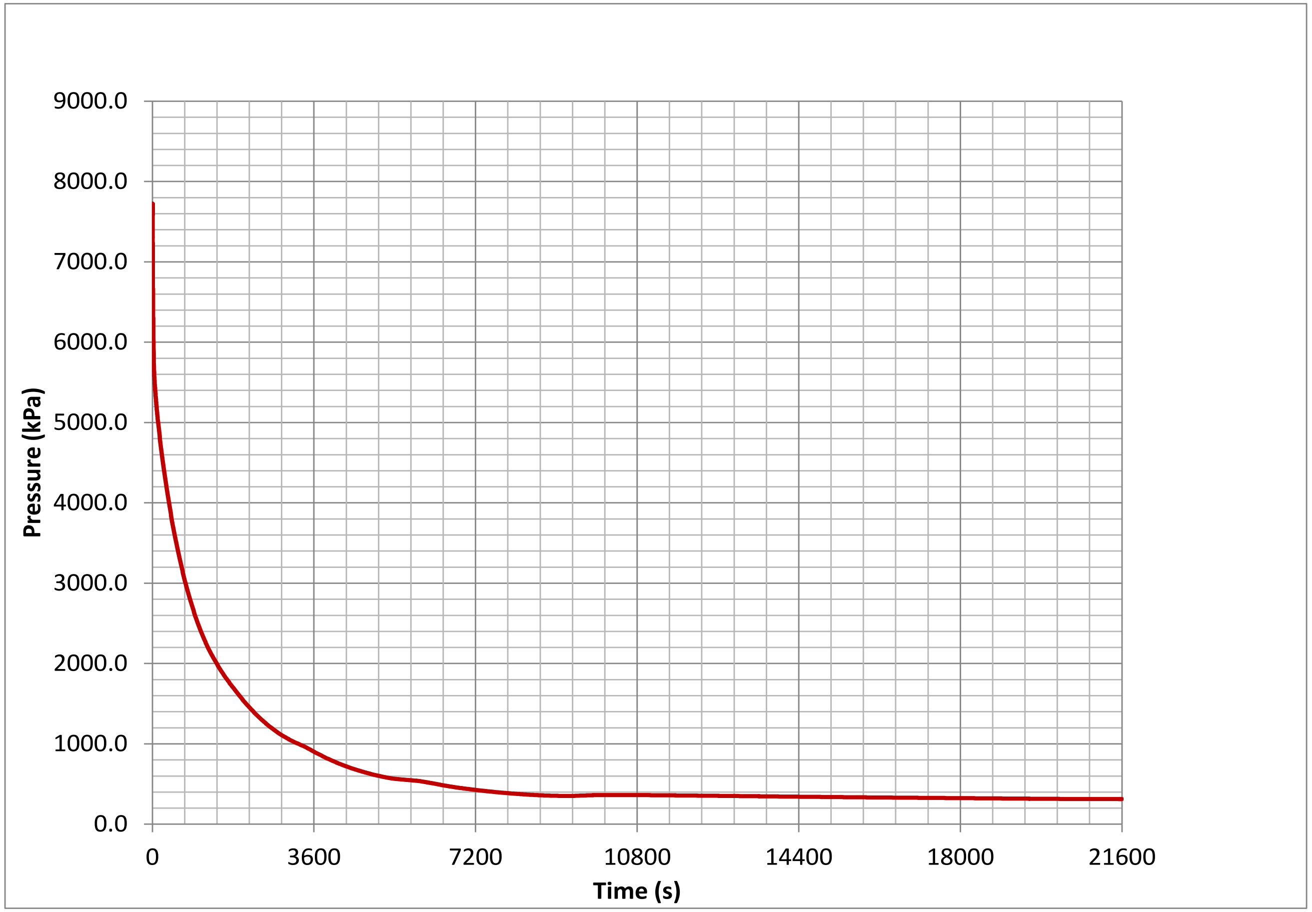

Figure 15.5-165: Reactor Pressure, Large FW Pipe Break, Conservative Case

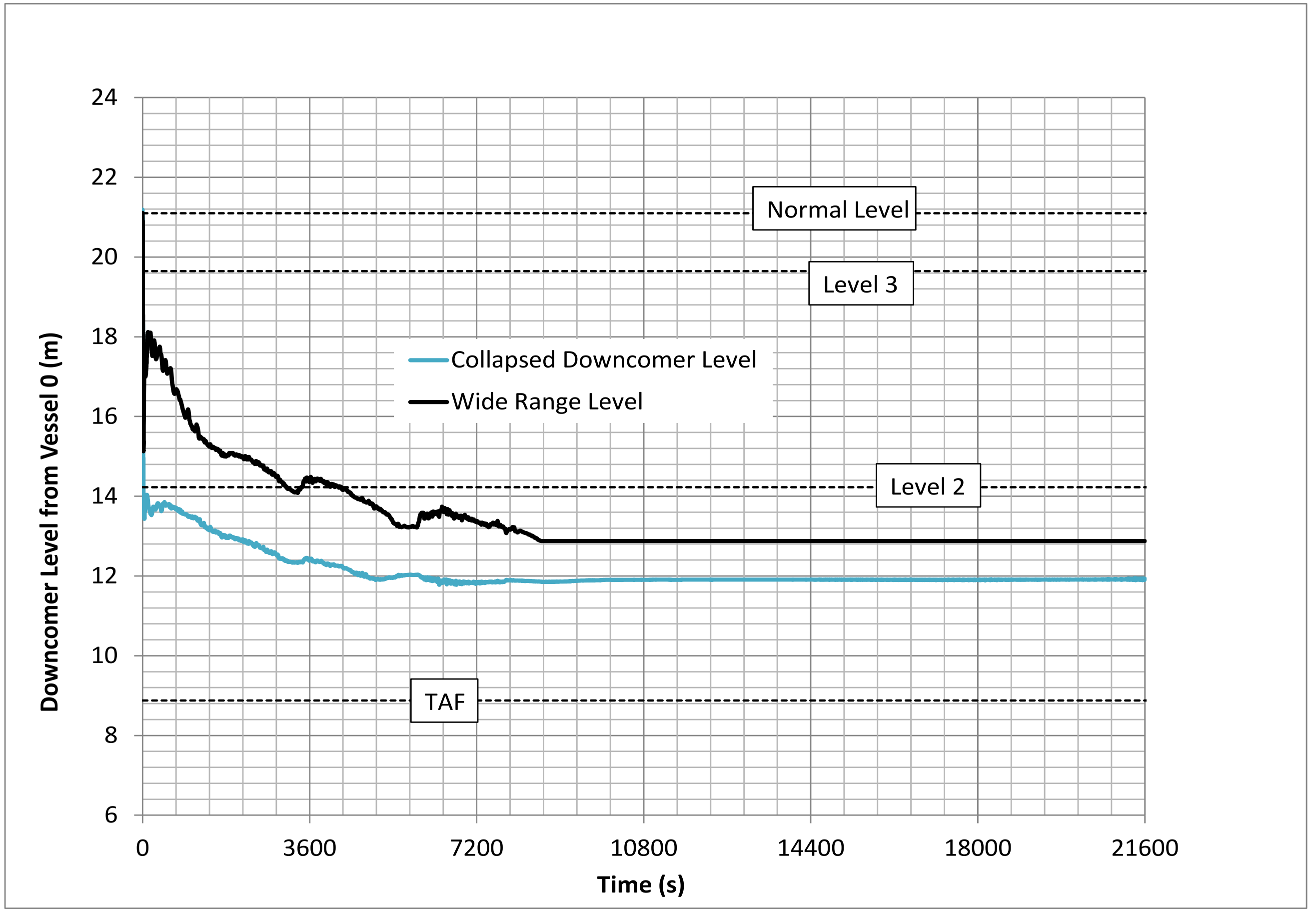

Figure 15.5-166: Reactor Water Level, Large FW Pipe Break, Conservative Case

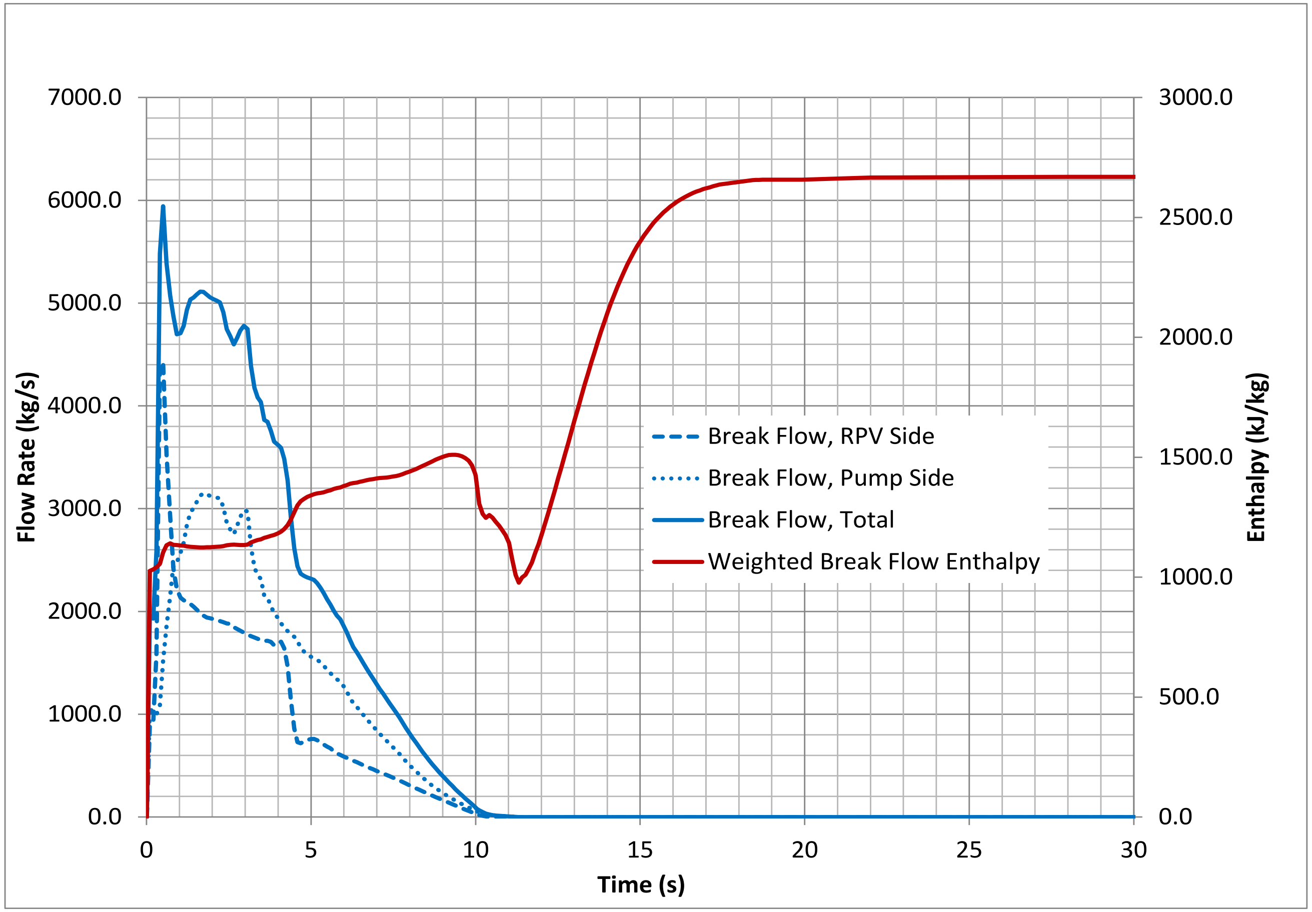

Figure 15.5-167: Break Flow Rate and Enthalpy, Large FW Pipe Break, Conservative Case

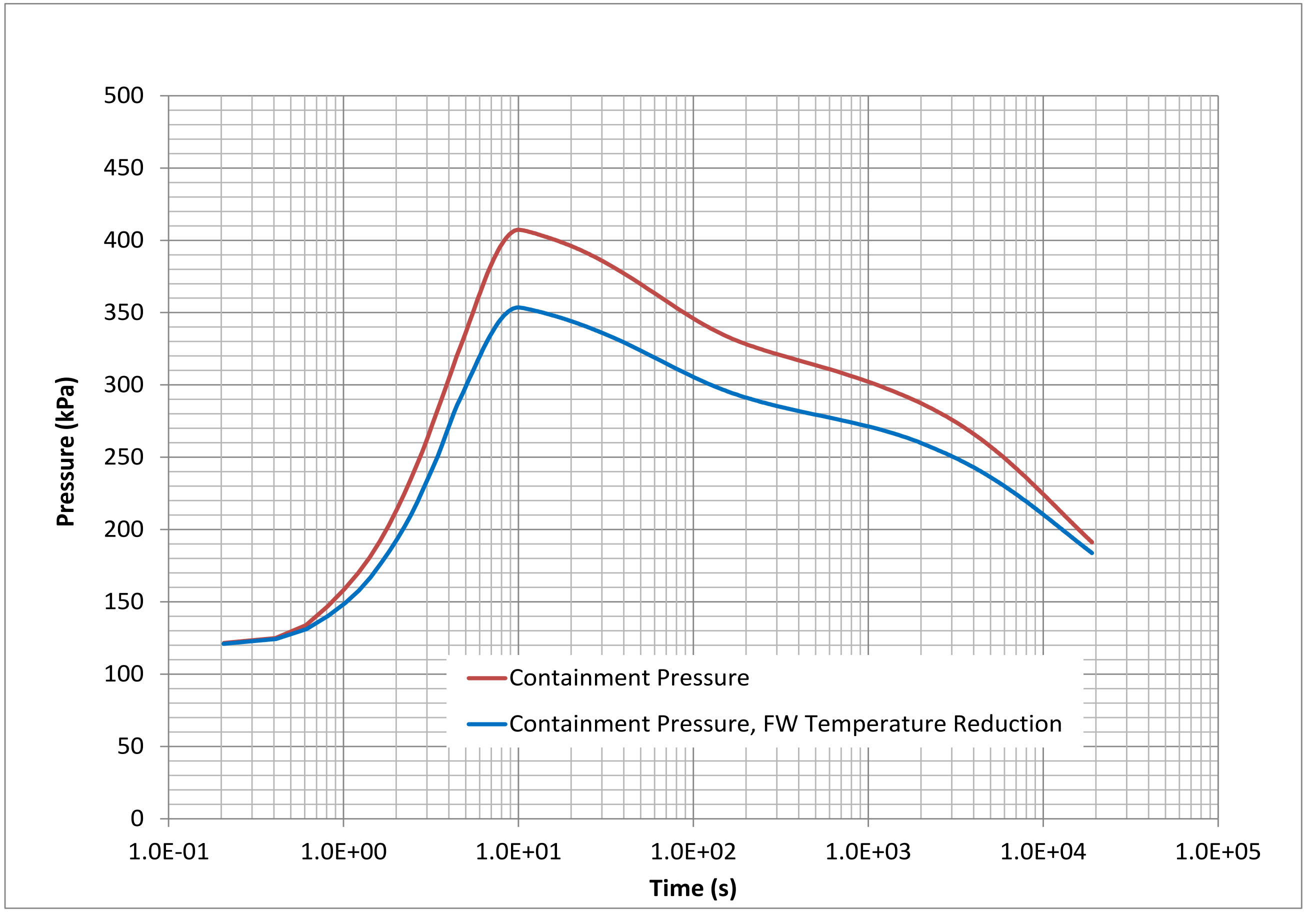

Figure 15.5-168: Containment Pressure, Large FW Pipe Break, Conservative Case

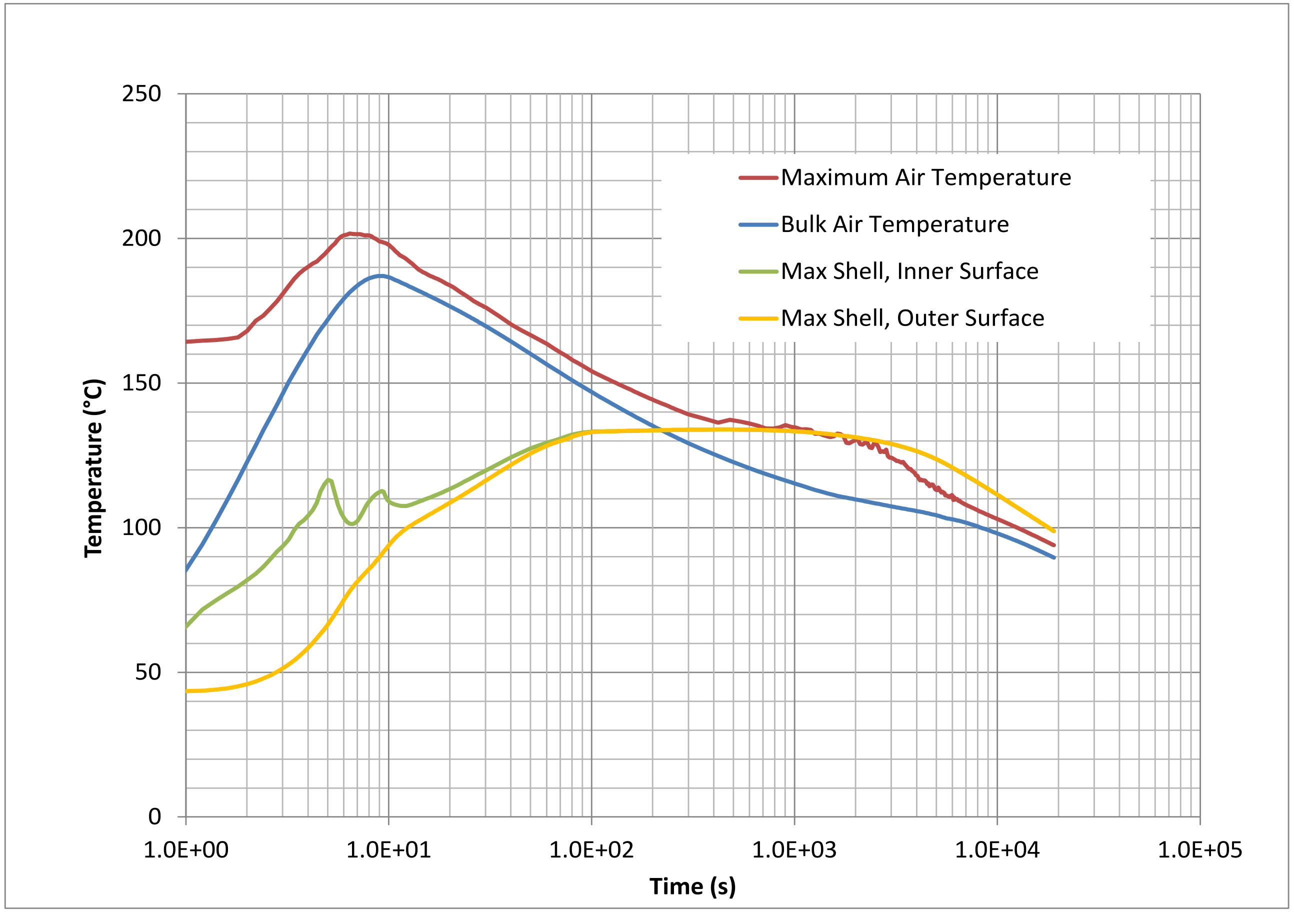

Figure 15.5-169: Containment Temperature, Large FW Pipe Break, Conservative Case

15.5.12 References

15.5-1 NEDC-34179P, “BWRX-300 UK GDA Ch. 15.1: Safety Analysis - General Considerations,” Rev A, GE-Hitachi Nuclear Energy, Americas, LLC.

15.5-2 NEDC-34180P, “BWRX-300 UK GDA Ch. 15.2: Safety Analysis - Identification, Categorisation, and Grouping of Postulated Initiating Events and Accident Scenarios,” Rev A, GE-Hitachi Nuclear Energy, Americas, LLC.

15.5-3 NEDC-34181P, “BWRX-300 UK GDA Ch. 15.3: Safety Analysis - Objectives and Acceptance Criteria,” Rev A, GE-Hitachi Nuclear Energy, Americas, LLC.

15.5-4 NEDC-34182P, “BWRX-300 UK GDA Ch. 15.4: Safety Analysis - Human Actions,” Rev A, GE-Hitachi Nuclear Energy, Americas, LLC.

15.5-5 NEDC-34183P, “BWRX-300 UK GDA Ch. 15.5: Deterministic Safety Analyses,” Rev A, GE-Hitachi Nuclear Energy, Americas, LLC.

15.5-6 NEDC-34184P, “BWRX-300 UK GDA Ch. 15.6: Probabilistic Safety Assessment,” Rev A, GE-Hitachi Nuclear Energy, Americas, LLC.

15.5-7 NEDC-34185P, “BWRX-300 UK GDA Ch. 15.7: Deterministic Safety Analyses – Analysis of Internal Hazards,” Rev A, GE-Hitachi Nuclear Energy, Americas, LLC.

15.5-8 NEDC-34186P, “BWRX-300 UK GDA Ch. 15.8: Safety Analysis - External Hazards, Rev A, GE-Hitachi Nuclear Energy, Americas, LLC.

15.5-9 NEDC-34187P, “BWRX-300 UK GDA Ch. 15.9: Summary of Results of the Safety Analyses,” Rev A, GE-Hitachi Nuclear Energy, Americas, LLC.

15.5-10 NEDC-34165P, “BWRX-300 UK GDA Ch. 3: Safety Objectives and Design Rules for SSCs.” Rev A, GE-Hitachi Nuclear Energy, Americas, LLC.

15.5-11 NEDC-34165P, “BWRX-300 UK GDA Ch.4: Reactor,” Rev A, GE-Hitachi Nuclear Energy, Americas, LLC.

15.5-12 NEDC-34167P, “BWRX-300 UK GDA Ch. 5: Reactor Coolant System and Associated Systems,” Rev A, GE-Hitachi Nuclear Energy, Americas, LLC.

15.5-13 NEDC-34168P, “BWRX-300 UK GDA Ch. 6: Engineered Safety Features,” Rev A, GE-Hitachi Nuclear Energy, Americas, LLC.

15.5-14 NEDC-34169P, “BWRX-300 UK GDA Ch. 7: Instrumentation and Control,” Rev A, GE-Hitachi Nuclear Energy, Americas, LLC.

15.5-15 NEDC-34170P, “BWRX-300 UK GDA Ch. 8: Electrical Power,” Rev A, GE-Hitachi Nuclear Energy, Americas, LLC.

15.5-16 NEDC-34171P, “BWRX-300 UK GDA Ch. 9A: Auxiliary Systems,” Rev A, GE-Hitachi Nuclear Energy, Americas, LLC.

15.5-17 NEDC-34173P, “BWRX-300 UK GDA Ch. 10: Steam and Power Conversion Systems,” Rev A, GE-Hitachi Nuclear Energy, Americas, LLC.

15.5-18 NEDC-34174P, “BWRX-300 UK GDA Ch. 11: Management of Radioactive Waste,” Rev A, GE-Hitachi Nuclear Energy, Americas, LLC.

15.5-19 NEDC-34175P, “BWRX-300 UK GDA Ch. 12: Radiation Protection,” Rev A, GE-Hitachi Nuclear Energy, Americas, LLC.

15.5-20 NEDC-34188P, “BWRX-300 UK GDA Ch. 16: Operational Limits and Conditions of Safe Operation,” Rev A, GE-Hitachi Nuclear Energy, Americas, LLC.

15.5-21 NEDC-34189P, “BWRX-300 UK GDA Ch. 17: Management for Safety and Quality Assurance,” Rev A, GE-Hitachi Nuclear Energy, Americas, LLC.

15.5-22 NEDC-34190P, “BWRX-300 UK GDA Ch. 18: Human Factors Engineering,” Rev A, GE-Hitachi Nuclear Energy, Americas, LLC.

15.5-23 NEDC-34191P, “BWRX-300 UK GDA Ch. 19: Emergency Preparedness and Response,” Rev A, GE-Hitachi Nuclear Energy, Americas, LLC.

15.5-24 NEDC-34199P, “BWRX-300 UK GDA Ch. 27: ALARP Evaluation,” Rev A, GE-Hitachi Nuclear Energy, Americas, LLC.

15.5-25 NEDC-34043P, Revision 0, “BWRX-300 TRACG Application,” GE Hitachi Nuclear Energy, May 2024.

15.5-26 NUREG/CR-5249, “Quantifying Reactor Safety Margins: Application of Code Scaling, Applicability, and Uncertainty Evaluation Methodology to a Large-Break, Loss-of-Coolant Accident,” Idaho National Engineering Laboratory, December 1989.

15.5-27 USNRC. Regulatory Guide 1.203, Revision 0, “Transient and Accident Analysis Methods,” U.S. Nuclear Regulatory Commission, December 2005.

15.5-28 NEDC-33922P-A, Revision 3, “Licensing Topical Report BWRX-300 Containment Evaluation Method,” GE Hitachi Nuclear Energy, June 2022.

15.5-29 NEDO-32177, Class I (Non-proprietary), “TRACG Qualification,” Revision 3, GE Hitachi Nuclear Energy, Revision 3, August 2007.

15.5-30 GOTHIC Thermal Analysis Package Qualification Report, Version 8.3 (QA), November 2018.

15.5-31 NEDC-32082, Revision 0, “BWR Steady-State Thermal Hydraulic Methodology (ISCOR),” GE Hitachi Nuclear Energy, August 1992.

15.5-32 NEDC-33939, Revision 1, “Steady State Nuclear Methods TGBLA06/PANAC11 Application Methodology for BWRX-300,” GE Hitachi Nuclear Energy, August 2023.

15.5-33 ANSI/ANS-18.1-2020, “Radioactive Source Term for Normal Operation of Light Water Reactors,” American National Standards Institute/American Nuclear Society, July 2020.

15.5-34 NUREG/CR-2858, “PAVAN: An Atmospheric-Dispersion Program for Evaluating Design-Basis Accidental Releases of Radioactive Materials from Nuclear Power Stations”, U.S. Nuclear Regulatory Commission, November 1982.

15.5-35 NUREG/CR-6604, “RADTRAD: A Simplified Model for Radionuclide Transport and Removal and Dose Estimation,” U.S. Nuclear Regulatory Commission, April 1998.

15.5-36 Radiation Protection Computer Code Analysis and Maintenance Program, ARCON, NUREG/CR-6331, Revision 1, “Atmospheric Relative Concentrations in Building Wakes,” U.S. Nuclear Regulatory Commission, June 2003.

15.5-37 ORIGEN2, Version 2.1, “Isotope Generation and Depletion Code – Matrix Exponential Method,” Oak Ridge National Laboratory (ORNL), August 1991.

15.5-38 Regulatory Guide 1.183, Revision 0, “Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors,” U.S. Nuclear Regulatory Commission, July 2000.

15.5-39 NEDC-33912P-A, Revision 1, “BWRX-300 Reactivity Control,” GE Hitachi Nuclear Energy, February 2021.

15.5-40 "Alternative Fuel Handling Accident Transport Methodology," ADAMS Accession Number ML19248C668, U.S. Nuclear Regulatory Commission.

15.5-41 NEDC-33911P-A, Revision 3, "BWRX-300 Containment Performance," GE Hitachi Nuclear Energy, January 2022.

15.5-42 United States Environmental Protection Agency, Federal Guidance Report No. 11, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion," September 1988.

15.5-43 United States Environmental Protection Agency, Federal Guidance Report No.12, "External Time in cycle to Radionuclides in Air, Water, and Soil," September 1993.

15.5-44 United States Environmental Protection Agency, EPA-402-R-93-002, Federal Guidance Report No. 15, "External Time in cycle to Radionuclides in Air, Water, and Soil," August 2019.

15.5-45 NEDO-32708, Revision 1, "Radiological Accident Evaluation: The CONAC04A Code," GE Hitachi Nuclear Energy, August 1997.

15.5-46 USNRC. 10 CFR 50, Appendix B. 14 June 2021. Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants.

15.5-47 USNRC. Regulatory Guide 1.145, Revision 1. February 1983. Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants.

15.5-48 USNRC. Regulatory Guide 1.194, Revision 0. June 2003. Atmospheric Relative Concentrations for Control Room Radiological Habitability Assessments at Nuclear Power Plants.

15.5-49 EPRI MAAP5 – Modular Accident Analysis Program for LWR Power Plants, Transmittal Document for MAAP5 Code Revision MAAP 5.06, Electric Power Research Institute, 2021.

15.5-50 BWRX-300 Safety Strategy, GEH, 006N5064, Revision 6, January 2024.

15.5-51 005N3558 Revision 3, June 21, 2024. BWRX-300 Fault Evaluation.

15.5-52 IAEA. SSG-61. September 2021. Format and Content of the Safety Analysis Report for Nuclear Power Plants.

15.5-53 Safety Assessment Principles for Nuclear Facilities, Office for Nuclear Regulation, 2014 editions, Revision 1 (January 2020).

15.5-54 BWRX-300 UK GDA Safety Case Development Strategy, GE-Hitachi Nuclear Energy Americas, LLC, NEDC-34140P.

15.5-55 IAEA. SSG-2. Deterministic Safety Analysis for Nuclear Power Plants.

15.5-56 IAEA. SSR-2/1. Safety of Nuclear Power Plants: Design.

15.5-57 GNF LTR NEDE-33885P-A, Revision 1 Control Rod Drop Accident Methodology.

15.5-58 Thermal Hydraulics Summary Report for UK BWRX-300, NEDC-34269P, Revision 0, August 2024.

15.5-59 NUREG/CR-6928. "Industry-Average Performance for Components and Initiating Events at U.S. Commercial Nuclear Power Plants," U.S. Nuclear Regulatory Commission, Idaho National Laboratory, February 2007.

15.5-60 NEDC-34039P, Revision 0, "BWRX-300 GNF2 Steady State Nuclear Methods TGBLA06/PANAC11 Application Methodology," GE Hitachi Nuclear Energy, April 2024.

APPENDIX A CLAIMS, ARGUMENTS AND EVIDENCE

A.1 Claims, Argument, Evidence (CAE)

The ONR "Safety Assessment Principles for Nuclear Facilities," (Reference 15.5-53) identify ONR's expectation that a safety case should clearly set out the trail from safety claims, through arguments to evidence. The CAE approach can be explained as follows:

1. Claims (assertions) are statements that indicate why a facility is safe,
2. Arguments (reasoning) explain the approaches to satisfying the claims,
3. Evidence (facts) supports and forms the basis (justification) of the arguments.

The GDA CAE structure is defined within NEDC-34140P, "BWRX-300 UK GDA Safety Case Development Strategy," (SCDS) (Reference 15.5-54) and is a logical breakdown of an overall claim that:

> *The BWRX-300 is capable of being constructed, operated, and decommissioned in accordance with the standards of environmental, safety, security and safeguard protection required in the UK.*

This overall claim is broken down into Level 1 claims relating to environment, safety, security, and safeguards, which are then broken down again into Level 2 area related sub-claims and then finally into Level 3 (chapter level) sub-claims.

The relevant Level 1 claim is:

> *2 – The safety risks to workers and the public during the construction, commissioning, operation and decommissioning of a BWRX-300 have been reduced As Low As Reasonably Practicable (ALARP)*

The relevant Level 2 claim is:

> *2.3 – A suitable and sufficient safety analysis has been undertaken which presents a comprehensive fault and hazard analysis that specifies the requirements on the safety measures and informs emergency arrangements.*

The Level 2 claim on ALARP is discussed below.

The Level 3 sub-claims that this chapter demonstrates compliance against are identified within the SCDS NEDC-34140P (Reference 15.5-54) and are as follows:

> *2.3.1 – All initiating events with the potential to lead to significant radiation time in cycle or release of radioactive material, including the effects of internal and external hazards, have been identified and appropriately assessed.*
>
> *2.3.2 – Design basis events have been appropriately assessed to specify requirements on safety functions and on safety measures and assess their effectiveness*
>
> *2.3.3 – Beyond Design Basis and Severe Accidents have been appropriately assessed to identify further risk reducing measures and inform emergency arrangements.*
>
> *2.3.4 – Probabilistic Safety Assessment is carried out to reflect the BWRX-300 design and evaluate risk levels.*
>
> *2.3.5 – Human Factors assessments have been appropriately integrated into the design, safety assessments and management arrangements, to meet the relevant safety requirements.*

In order to facilitate compliance, demonstration against the above Level 3 sub-claims, this PSR chapter has derived a suite of arguments that comprehensively explain how their applicable Level 3 sub-claims are met.

It is not the intention to generate a comprehensive suite of evidence to support the derived arguments, as this is beyond the scope of GDA Step 2. However, where evidence sources are available, examples are provided.

A.2 Risk Reduction As Low As Reasonably Practicable (ALARP)

The relevant Level 2 claim is:

2.4 – Safety risks have been reduced As Low As Reasonably Practicable (ALARP).

It is important to note that nuclear safety risks cannot be demonstrated to have been reduced ALARP within the scope of a 2-Step GDA. It is considered that the most that can be realistically achieved is to provide a reasoned justification that the BWRX-300 SMR design aspects will effectively contribute to the development of a future ALARP statement.

The Level 3 sub-claims are:

2.4.1 – Relevant Good Practice (RGP) has been taken into account across all disciplines.

2.4.2 – Operational Experience (OPEX) and Learning from Experience (LfE) has been taken into account across all disciplines.

2.4.3 – Optioneering (all reasonably practicable measures have been implemented to reduce risk).

2.4.4 – Probabilistic ALARP (i.e., balanced design and numerical target shortfalls assessed.

In this respect, this chapter contributes to the overall future ALARP case by demonstrating that the chapter-specific arguments derived may be supported by existing and future planned evidence sources covering sub-claims 2.4.1, 2.4.2, and 2.4.3.

In particular, where there are shortfalls against deterministic rules or design criteria, explicit optioneering will be undertaken to determine what, if any, reasonably practicable changes to the design [including the manner in which it is operated] could be implemented to close [or partially close] or to mitigate the shortfall.

Probabilistic safety aspects of the ALARP argument, claim 2.4.4, are addressed within PSR Ch. 15.6.

Table A-1: Deterministic Safety Claims and Arguments

Chapter Claim	Chapter 15 Argument	Sections and/or Reports that Evidence the Arguments
2.3 – A suitable and sufficient safety analysis has been undertaken which presents a comprehensive fault and hazard analysis that specifies the requirements on the safety measures and informs emergency arrangements.		
2.3.1 – All initiating events with the potential to lead to significant radiation time in cycle or release of radioactive material, including the effects of internal and external hazards, have been identified and appropriately assessed.	A systematic, iterative approach to the identification of faults and hazards is being used to identify initiating events.	PSR Ch. 15 FAP PSR15.528, FAP PSR15.5-30
	The identified initiating events have been placed into categories and assessed accordingly.	PSR Ch. 15.2 Chapters 15.5, 15.6, 15.7, and 15.8
2.3.2 – Design basis events have been appropriately assessed to specify requirements on safety functions and on safety measures and assess their effectiveness.	The appropriate subset of all identified initiating events has been identified for Design Basis Analysis (DBA). Initiating event frequencies have been determined on a best-estimate basis, apart from natural hazards which have been assessed on a conservative basis.	PSR Ch. 15.1 PSR Ch. 15.2 PSR Ch. 15.5 FAP PSR15.5-30, FAP PSR15.5-32
	The relevant design basis fault sequences have been identified. These include, where appropriate, single failures, consequential failures, and common cause failures. The most onerous initial operating state and plant configuration permitted by the operating rules are assumed. The correct performance of safety-related and non-safety equipment is not assumed where it would mitigate the consequences of the fault.	PSR Ch. 15.1 PSR Ch. 15.2 PSR Ch. 15.5 Tables 15.5-1, 15.5-3, 15.5-6 to 15.5-33, 15.5-43 to 15.5-45
	The analysis of design basis fault sequences has been performed so that the margin to the best estimate is always sufficient to cover [on the safe side] all credible uncertainties and reflect the overall significance of the estimate to the safety case. It is therefore conservative.	Section 15.5.1.1 006N5420 Revision 1, TRACG Application for BWRX-300, Sections 3.4, 3.5, and 3.6.

Chapter Claim	Chapter 15 Argument	Sections and/or Reports that Evidence the Arguments
2.3 – A suitable and sufficient safety analysis has been undertaken which presents a comprehensive fault and hazard analysis that specifies the requirements on the safety measures and informs emergency arrangements.		
	Appropriately conservative assumptions have been employed in the radiological calculations.	Sections 15.5.8, 15.5.9 Tables 15.5-1, 15.5-3 FAP PSR15.5-31, FAP PSR15.5-33
	The analytical models have been subject to appropriate verification and validation.	Subsection 15.5.1.2 NEDC-34043P, Revision 0, “BWRX-300 TRACG Application” NEDC-33922P-A, Revision 3, “Licensing Topical Report BWRX-300 Containment Evaluation Method,” NEDO-32177, Revision 3, “TRACG Qualification” GOTHIC Thermal Analysis Package Qualification Report, Version 8.3 (QA) FAP PSR15.5-36
	Appropriate acceptance criteria and targets, including a defined stable, safe state, have been employed to judge the effectiveness of the designated safety measures.	PSR Ch. 15.2 PSR Ch. 15.3 FAPPSR15.5-32, FAP PSR15.5-38
	Sensitivity studies have been employed to demonstrate that small changes in DBA parameters do not lead to cliff-edge increases in radiological consequences.	Subsections 15.5.3.1.1; 15.5.4.3.2; 15.5.5.2.1; 15.5.5.3.2; and Sections 15.5.8 and 15.5.9.
	There is a clear, and auditable linkage between the identified initiating events, fault sequences, and designated safety measures.	Sections 15.5.4 to 15.5.9 Tables 15.5-5; 15.5-6 to 15.5-15; 15.5-18 to 15.5-21; 15.5-26 to 15.5-32; 15.5-33.

Chapter Claim	Chapter 15 Argument	Sections and/or Reports that Evidence the Arguments
2.3 – A suitable and sufficient safety analysis has been undertaken which presents a comprehensive fault and hazard analysis that specifies the requirements on the safety measures and informs emergency arrangements.		
		005N3558 BWRX-300 Fault Evaluation (Revision 3) and Fault List Attachment 1. FAP PSR 15.5-28, FAP PSR15.5-29
	Operating limits and conditions can be identified to ensure the plant is operated safely at all times.	PSR Ch. 3 Safety Objectives and Design Basis Rules for Structures, Systems and Components 005N9461, Revision 4. BWRX-300 Structures, Systems, and Components Safety Classification FAP PSR15.5-37
2.3.3 – Beyond Design Basis and Severe Accidents have been appropriately assessed to identify further risk reducing measures and inform emergency arrangements.	A systematic approach has been used to analyse beyond design basis states, focussing on how the accident state or scenario will be controlled and/or mitigated.	15.5.5 – Analysis of Design Extension Conditions Without Significant Fuel Degradation 15.5.6 – Analysis of Design Extension Conditions With Core Melting BWRX-300 Safety Strategy (Revision 6): Sections 3.1.4.2, 3.1.4.4, 3.1.7.3, and 3.1.7.5 FAP-DSA-001, FAP-DSA-003, FAP-DSA-008, FAP-DSA-009
	The severe accident analysis has been used to assist in the identification of further reasonably practicable measures and form the basis for accident management strategies and procedures and support the production of emergency plans.	15.5.5 – Analysis of Design Extension Conditions Without Significant Fuel Degradation 15.5.6 – Analysis of Design Extension Conditions With Core Melting FAP PSR15.5-29, FAP PSR15.5-30, FAP PSR15.5-36

APPENDIX B FORWARD ACTIONS

Table B-1: Deterministic Safety Analysis Forward Actions

Action ID	Finding	Forward Actions	Lead Discipline	Delivery Phase
PSR15.5-28	**Development of Fault Schedule** The provision of a fault schedule, a tabular summary of the essential parts of a nuclear facility's safety case, is Relevant Good Practice (RGP) in the UK. Although the current Fault List has many features in common with a fault schedule, it does not meet all of the expectations for one. The associated Safety Assessment Principles (SAPs) are ESS.11, and FA.8.	Develop a format for a fault schedule which will meet UK fault schedule expectations whilst reflecting the BWRX-300 engineering and operational philosophy, and the safety case. Utilise the developing fault schedule during the ongoing design development work. The following aspects should be considered in particular • Bounding faults. • Initiating fault frequencies. • Unmitigated consequences. • Claimed safety measures. Consider use of the UK ABWR fault schedule as a starting point.	Fault Studies	PCSR / Detailed Design
PSR15.5-29	**Diversity for Frequent Faults** It is long established Relevant Good Practice (RGP) in the UK to provide a diverse safety system, qualified to an appropriate standard, for each Fundamental Safety Function (FSF), to protect against Frequent Fault (i.e., those having a frequency greater than 1 x 10^{-3} per year). One aspect of this UK specific expectation, the need to show that there is an alternate means to trip the reactor for certain events, is common international practice. The PSA (PSR Ch. 15.6) considers such	Provide a demonstration that there is diversity in the designated protection for all Frequent Faults (ie those having a frequency of greater than 1 x 10^{-3} per year). Confirm that status of the Boron Injection System (BIS). Provide a demonstration of its functional capability and reliability if the system needs to be claimed in the Deterministic Safety Analysis (DSA).	Fault Studies	PCSR / Detailed Design

Action ID	Finding	Forward Actions	Lead Discipline	Delivery Phase
	Anticipated Transients Without Scram (ATWSs) but there is no Deterministic Safety Analysis (DSA) for them. Although the BWRX-300 incorporates two means of inserting the Control Rods (motor run in and hydraulic) this may not meet ONR expectations for diversity. The associated Safety Assessment Principles (SAPs) are: EDR.3, FA.6, and ERC.2.			
PSR15.5-30	**Completeness of the Fault List** The list of faults considered in PSR Ch. 15.5 are principally bounding reactor faults during power operation. UK RGP is for the list of faults to be identified, and justified, in the safety case. The associated Safety Assessment Principles (SAPs) are: FA.2, FA.5, FA.6, EMC.3, EHA.1, and Numerical Target 4.	A systematic and auditable process [eg FMEA, HAZOP] should be undertaken to produce a comprehensive list of faults. This should cover: • Faults in all operational modes • Non-reactor faults [e.g., those associated with the Spent Fuel Pool (SFP); fuel route; radioactive waste facilities] • Faults associated with essential support systems • Faults involving an initiating event and failure of one or more safety measures • Faults involving partial failures as well as total failures • Faults arising from internal and external hazards The resulting list should be cross-checked with the faults considered in the PSA. New Design Basis Analysis (DBA) should be provided for newly identified faults that cannot be demonstrated to be bounded by existing analysis. This FAP item is connected with item DSA-005 on	Fault Studies	PCSR / Detailed Design

Action ID	Finding	Forward Actions	Lead Discipline	Delivery Phase
		numerical target development.		
PSR15.5-31	**Generic Site Specification** The offsite atmospheric dispersion factors (χ/Q) employed in the deterministic safety analysis [mainly sections 15.5.8 and 15.5.9] need to be assessed for a Great Britain (GB) Nuclear Power Plant (NPP) site. The relevant Safety Assessment Principle (SAP) is: ST.3.	Assumptions used in dose assessment calculations that will be broadly consistent with those that could reasonably be expected for a NPP site in GB should be identified. These will include distance of the reactor and other buildings with radiological inventory from the site boundary; and expected weather conditions. These should be compared with the assumptions currently used in the Deterministic Safety Analysis (DSA). If they are not bounded by the existing assumptions, the radiological calculations should be repeated using the new Generic Site Envelope (GSE) assumptions.	Fault Studies	PCSR / Detailed Design
PSR15.5-32	**Numerical Target Development** Various sets of acceptance criteria used throughout PSR Ch. 15.5, such as: • 10 CFR 50 Appendix A General Design Criteria • NUREG-0800 • 10 CFR 50.34 No radiological criteria are identified for Design Extension Conditions (DECs). Much of the Design Basis Analysis (DBA) employs decoupling criteria to demonstrate the physical barriers to fission product release are maintained and which therefore, modulo activity in the coolant, meet all radiological criteria. In the UK, there is no prescription of such criteria, and so it is for the safety case to	Develop a radiological criterion suitable for defining the set of faults subject to Design Basis Analysis (DBA) and judging the effectiveness of the safety measures designated in the DBA. The criterion does not need to be identical to Numerical Target 4 of the SAPs but should be broadly comparable to it. This FAP item is connected with item DSA-006 on radiological consequence calculation methods.	Fault Studies	PCSR / Detailed Design

Action ID	Finding	Forward Actions	Lead Discipline	Delivery Phase
	identify and justify them. However, ONR will make use of its own numerical targets whilst making its regulatory judgements. The associated Safety Assessment Principles (SAPs) are: Numerical Target 4, FA.5, and FA.7.			
PSR15.5-33	**Dose Calculation Assumptions** The approach to dose calculation in the Deterministic Safety Analysis (DSA) employs US assumptions and approaches such as: • Dose Conversion Factors (DCFs) • Breathing rates. • Occupancy factors. • ANSI ANS-18.1-2020. In the UK, there is no prescription of the approach to dose calculations, and so it is for the safety case to identify and justify them. The associated Safety Assessment Principles (SAPs) are FA.3, FA.7, and paragraphs 619 and 729.	Compare the assumptions and methods used in dose consequence calculations with UK practice, making appropriate changes if necessary. This FAP item is connected with item DSA-005 on numerical target development.	Fault Studies	PCSR / Detailed Design
PSR15.5-36	**Verification and Validation** The details of the Verification, Validation and Uncertainty Quantification (VVUQ) of the two principal analysis methods, employing the TRACG and GOTHIC codes, presented in the Preliminary Safety Report (PSR) are at a relatively high level. The associated Safety Assessment Principles (SAPs) are AV.1, AV.2, AV.3, AV.4, and AV.5.	Provide additional information on the overall approach to Verification, Validation, and Uncertainty Quantification (VVUQ) for the analysis methods employed in the Deterministic Safety Analysis (DSA), in particular those involving the TRACG and GOTHIC codes.	Fault Studies	PCSR / Detailed Design

Action ID	Finding	Forward Actions	Lead Discipline	Delivery Phase
PSR15.5-37	**Implementable Requirements** The details of when and how the Deterministic Safety Analysis (DSA) has informed implementable requirements such as: • Classification of Systems, Structures, and Components (SSCs) • Operating Rules. presented in the Preliminary Safety Report (PSR) are not clear. The associated Safety Assessment Principles (SAP) is FA.9.	Provide additional information on the use of the Deterministic Safety Analysis (DSA) in the classification of Systems, Structures, and Components (SSCs) and the derivation of Operations Rules.	Fault Studies	PCSR / Detailed Design
PSR15.5-38	**Stable Safe State** The presentation of the Deterministic Safety Analysis (DSA) in the Preliminary Safety Report (PSR) is appropriately focused on the initial phase and the achievement of a controlled state. However, Relevant Good Practice (RGP) for Design Basis Analysis (DBA) is for the design basis safety case to demonstrate that safety measures are designated and capable to bringing the facility to a stable, safe state. The associated Safety Assessment Principle (SAP) is FA.8.	Provide a systematic demonstration of how to take the reactor from a controlled state to a stable, safe state for each fault. The safety systems and any operator actions required should be identified, and appropriate transient analysis provided to demonstrate the claimed measures can deliver the necessary functions. The stable, safe state should be defined and justified.	Fault Studies / Human Factors	PCSR / Detailed Design