AECL EACL

Licensing Submission

PIRT for Critical Inlet Header Break LOCA in ACR-700

ACR USA

108US-03500-LS-001
Revision 0

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This report represents the first US-style Phenomena Identification and Ranking Table (PIRT) activity undertaken by AECL. By its nature, the PIRT process is a group activity (i.e., resource intensive). The group meetings were held multi-site with the aid of computer communication; a quite workable innovation for a PIRT effort.

The facilitator, Brent Boyack, ably guided this PIRT effort and participated as a Panel member. Other PIRT Panel members, providing wide-ranging expertise, included Ben Rouben (physics), Brock Sanderson (fuel, fuel channel), Darryl Dormuth (thermal-hydraulics), Dave Wren (ACR™* R&D), Glen McGee (thermal-hydraulics), Harve Sills (fuel, fuel channel), Lawrence Dickson (fuel, fission products), Nik Popov (thermal-hydraulics), Paul Ingham (experimental thermal-hydraulics), Rick Jones (physics) and Yanfei Rao (subchannel thermal-hydraulics). These people participated in a total of seven meeting days spread over three separate meetings.

Dave Wren and Ken Hau (ACR Processes) provided a detailed event sequence table tailored to the PIRT activity requirements. A detailed scenario description and plot package that was provided by Dave Wright and Shyam Ramachandran (safety analysts) has been incorporated into this document.

The contributions of all the above are gratefully acknowledged. Their cooperative efforts led to the comprehensive PIRT effort described in this document.

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* ACR™ (Advanced CANDU Reactor™) is a trademark of Atomic Energy of Canada Limited (AECL).

EXECUTIVE SUMMARY

ACR-700 CRITICAL BREAK LARGE LOSS-OF-COOLANT ACCIDENT PHENOMENA IDENTIFICATION AND RANKING TABLE

The ACR-700 design is based on the use of modular horizontal fuel channels surrounded by a heavy water moderator, the same as with all CANDU reactors. The major innovation in ACR is the use of slightly enriched uranium fuel and light water as coolant. This results in a more compact reactor design and a reduction of heavy water inventory, both contributing to a significant decrease in cost compared to CANDU reactors, which employ natural uranium as fuel and heavy water as coolant.

This document provides the results of a PIRT effort for the ACR-700 reactor undergoing a 25% reactor inlet header (RIH) break with subsequent loss of Class IV power. This break is a critical header break with respect to channel flow stagnation, i.e., produces the highest sheath temperatures for LBLOCA events. The most important systems, components, and processes / phenomena occurring during each phase of this LBLOCA are identified and tabulated. Rationales are provided to support the assigned importance-rank.

The primary Figure of Merit (FOM) for evaluating ECCS performance, for this PIRT activity, is sheath temperature. Sheath maximum temperature is limited to the US-licensing limit of 2200°F (1204°C) during this LBLOCA event.

A secondary FOM for evaluating ECCS performance is the integrity of the pressure tube (PT), which is the in-core pressure boundary. With adequate cooling during the heat transport system (HTS) depressurization, PT high-temperature strain is limited (e.g., <2%) and PT integrity is assured. Sheath temperature and PT high-temperature strain are closely correlated parameters permitting two FOMs to be considered.

An important effort in demonstrating the adequacy of the safety analysis code suite is the preparation of a Phenomena Identification and Ranking Table (PIRT). The PIRT is nuclear power plant (NPP) and scenario specific. Plant systems and components that may have an impact on the evaluation criteria are identified, as are the influencing phenomena. The identified phenomena are then ranked relative to their influence on the FOM. Once the PIRT is complete, the analysis code suite used to simulate the transient would then be reviewed to ensure that the component models are adequate. A high degree of adequacy is required for those models of the system/component behavior that most strongly impact the course of the transient as identified by the PIRT process.

The reactor is assumed to be operating at full power with equilibrium poison levels at the time of the LOCA. Design values are assumed. The scenario divides into three distinct phases:

- Early blowdown cooling preceding initiation of emergency coolant injection (ECI) (i.e., rupture disc open);
- Late blowdown cooling / ECI / refill; and
- Long-term cooling.

The following three tables summarize the PIRT findings during each phase for the high importance-ranked processes/phenomena. Note that components are assumed to be available and to function as per design.

Early Blowdown Cooling (0 – 53 seconds)

System	Component	Processes/Phenomena
Heat Transport System	Break, Inlet Header, Outlet Header, Feeder, Pump Suction Piping/SG Outlet Plenum, HTS Pump	Critical flow, flow regime, flow – pressure driven, stored energy release, flashing, pressure drop (1-phase, 2-phase), flow reversal, pump performance / characteristics, coastdown / rundown
Fuel Channel	End Fittings, Pressure Tube	Flashing, flow – pressure driven, pressure drop (1-phase, 2-phase), flow reversal, convective heat transfer, radiant heat transfer, deformation
Fuel Bundle	Fuel Element (general)	Stored energy release, conduction, radiant heat transfer, gap conductance, boiling – film, Critical Heat Flux (CHF), deformation, fission heating, end power peaking, flow regime, void generation from heat transfer, entrainment, de-entrainment, flashing, level - swelling, multi-dimensional flow, flow – pressure driven, pressure drop (1-phase, 2-phase), flow reversal, flow – stalled (stagnation)
Shutdown Systems	SDS1 (shut off rods), SDS2 (poison injection)	Reactivity effect of firing SDS1, Reactivity effect of firing SDS2
Secondary and Feedwater System	Main Steam Line/Header, Main Steam Safety Valve	Flow – pressure driven, change in path/state (open/close)

Sheath maximum temperature occurs mid-way through this first phase.

Late Blowdown Cooling / ECI / Refill (53 seconds – 250 seconds)

System	Component	Processes/Phenomena
Emergency Cooling Injection	Injection Water Storage Tank, ECI Injection Valve, ECI Piping, Large Header Interconnect, Rupture Disc	Pressure, level, flow – pressure driven, change in path/state (open/close), pressure drop (1-phase, 2-phase), refill
Heat Transport System	Break, Inlet Header, Outlet Header, Feeder	Critical flow, flow – pressure driven, flow regime, pressure drop (1-phase, 2-phase), distribution (multiple channels), flow – counter current
Fuel Channel	End Fittings, Pressure Tube	Flow – pressure driven, flow – counter current, pressure drop (1-phase, 2-phase), deformation (includes failure)
Fuel Bundle	Fuel Element (general)	Forced convection to liquid, boiling – nucleate, boiling – film, decay heating, flow regime, void generation from heat transfer, entrainment, de-entrainment, level - swelling, multi-dimensional flow, flow – pressure driven, pressure drop (1-phase, 2-phase), flow reversal

Long-Term Cooling (> 250 seconds)

System	Component	Processes/Phenomena
Long-Term Cooling	Sump, Debris Screen, Recovery Pump and Piping, LTC valves	Level, flow – gravity driven (draining), flow – pressure driven, pressure drop (1-phase, 2-phase), change in path/state (open/close)
Reserve Water System	Reserve Water Tank	Level
Heat Transport System	Break, Inlet Header, Outlet Header, Feeder	Flow – pressure driven, flow regime, pressure drop (1-phase, 2-phase)
Fuel Channel	End Fittings	Flow – pressure driven, flow regime, pressure drop (1-phase, 2-phase)
Fuel Bundle	Fuel Element (general)	Forced convection to liquid, boiling – nucleate, decay heating, flow – pressure driven, pressure drop (1-phase, 2-phase), flow reversal
Containment System	Containment	Flow – gravity driven (draining)

ACRONYMS

ACR	Advanced CANDU Reactor
AECL	Atomic Energy of Canada Limited
ASDV	Atmospheric Steam Discharge Valve
CANDU	CANada Deuterium Uranium
CANFLEX®*	CANDU FLEXible fuelling
CHF	Critical Heat Flux
CNSC	Canadian Nuclear Safety Commission
CSDV	Condenser Steam Dump Valve
CT	Calandria Tube
ECCS	Emergency Core Cooling System
ECI	Emergency Coolant Injection system
FOM	Figure Of Merit
HTS	Heat Transport System
LBLOCA	Large Break Loss-Of-Coolant Accident
LOCA	Loss-Of-Coolant Accident
LTC	Long-Term Cooling system
LWR	Light Water Reactor
MSIV	Main Steam Isolation Valve
MSSV	Main Steam Safety Valve
NPP	Nuclear Power Plant
NRC	(US) Nuclear Regulatory Commission
P&IC	Pressure and Inventory Control
PCT	Peak Clad Temperature
PDO	Post-DryOut
PHWR	Pressurized Heavy Water Reactor
PIRT	Phenomena Identification and Ranking Table
PT	Pressure Tube
PTR	Pressure Tube Reactor
RCW	Recirculated Cooling Water
RIH	Reactor Inlet Header
RRS	Reactor Regulating System
RWS	Reserve Water System
SDS	ShutDown System
SDS1	ShutDown System 1 (shut off rods)
SDS2	ShutDown System 2 (poison injection)

* CANFLEX® is a registered trademark of AECL and the Korea Atomic Energy Research Institute (KAERI).

SG	Steam Generator
TBD	Technical Basis Document
VM	Validation Matrix

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APPENDICES

1. INTRODUCTION

Over the last 30 years, AECL has established a successful and internationally recognized line of CANDU pressure tube reactors (PTR) with heavy water moderator, in particular, the medium-sized CANDU 6 reactor[1]. The accumulated experience is being used in the development of the Advanced CANDU Reactor (ACR).

Atomic Energy of Canada Limited (AECL) developed the ACR-700 (Reference [1]) to meet customer needs for reduced capital cost, shorter construction schedule, higher capacity factor, lower operating cost, increased operating life, simpler component replacement and enhanced safety features. Currently, the ACR-700 is undergoing pre-application review by the United States Nuclear Regulatory Commission (NRC) and a pre-licensing review by the Canadian Nuclear Safety Commission (CNSC).

USNRC developed and demonstrated the Code Scaling, Applicability, and Uncertainty (CSAU) methodology (Reference [2]) by quantifying the uncertainty associated with the best-estimate calculation of key safety-related parameters for an accident in a nuclear power plant. Subsequently, the Code of Federal Regulations was revised and the NRC issued guidance that applicants could license the ECCS of future plants using either the 10 CFR 50 Appendix K methodology, or they could use a best-estimate methodology, provided the uncertainty in key licensing parameters (e.g., the peak clad temperature (PCT)) was quantified.

An essential, early step in the code adequacy demonstration is the completion of a Phenomena Identification and Ranking Table (PIRT) effort. The PIRT is plant and scenario specific. AECL has completed a Technical Basis Document (TBD) for the CANDU 6 design and an ACR-specific TBD that reflects the ACR-700 specific design features and accident scenarios (Reference [3]). Both TBD documents address high-level phenomena identification and ranking that is used in the Canadian computer code development and validation process (Reference [4]). The PIRT effort documented in this activity is aimed at performing a confirmatory phenomena identification and ranking process that follows the methodology developed in the US (Reference [5]) and is used to confirm completeness of the phenomena identification and ranking documented in the TBD.

AECL has analyzed a spectrum of LBLOCAs for the ACR-700 design. The 25% Reactor Inlet Header (RIH) is the critical RIH break that results in near-stagnation flow along the channels downstream of the broken inlet header. Therefore the 25% RIH break is selected for this report as the basis for the PIRT effort.

The general PIRT process and its specific application to the 25% RIH LOCA PIRT effort are documented in Section 2. A description of the ACR-700 plant and the key systems and components that respond to the selected scenario are described in Section 3. The 25% RIH LOCA scenario for the ACR-700 is described in Section 4. The three phases of the accident for which PIRT assessments were made are described. The three phases are the blowdown period, late blowdown/ECI/refill period, and the long-term cooling period. Summary PIRT results are presented in Section 5. Details of the PIRT results are presented in the Appendices. Phenomena definitions are provided in Appendix A.

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[1] The latest CANDU reactor design was completed at the Qinshan CANDU Phase III site in People's Republic of China. Unit No. 1 went in-service in 2002 December, while Unit No. 2 went in-service in 2003 July.

2. METHODOLOGY

2.1 Need for Identification and Ranking

The physical processes and phenomena that occur in nuclear reactors can be both complex and highly coupled. The ability to predict the behavior of nuclear reactors during normal operation as well as their response to accident conditions is of paramount importance. With predictability comes understanding. Both are required to ensure safe reactor operation.

Several fundamental elements form the basis for a safe design. The design itself is also of great importance. An important recent trend in reactor design is the reliance on simplified, passive and/or inherent safety features to reduce the reliance on both active, complex hardware and systems, and operator interventions. The ability to accurately predict the behavior of the design under operational and accident conditions using qualified analytic methods is essential.

Predictability, including an understanding of safety margins, is based upon fundamental data from experiments, experiments in both scaled component and integral facilities, and calculations using analytic tools. However, it is not feasible to build a full-scale test reactor and then expose that reactor to the aggressive conditions of all design basis accidents. Therefore, analyses based upon qualified analytic methods have become essential to confirming the safety basis for nuclear reactors. The development and qualification of transient and accident analysis methods is central to both designing and demonstrating the safety of a reactor design.

Recently, the NRC has issued a draft regulatory guide for “Transient and Accident Analysis Methods” (Reference [6]). This regulatory guide, DG-1120, articulates six basic principles of evaluation model development and assessment. The first two principles are of particular concern to this report. The first is to “determine the requirements for the evaluation model.” Central to this step is “identification of the . . . components, phenomena, physical processes, and parameters (hereafter collected under the general designation of ‘phenomena’) needed to evaluate event behavior relative to the figures of merit described in the Standard Review Plan and derived from the General Design Criteria in Appendix A to 10 CFR 50.” This identification step is an essential element of the Phenomenon Identification and Ranking Table (PIRT) process. The second essential element is ranking each plausible phenomenon relative to a figure of merit (FOM), also called an evaluation criterion. The ranking step is based upon the reality that all processes and phenomena that occur during a transient or accident do not equally influence plant behavior. The PIRT process reduces candidate phenomena to a manageable set by identifying and ranking the phenomena with respect to their influence on the figure of merit.

As stated in Reference [6], the principal product of the process outlined above is a Phenomenon Identification and Ranking Table (PIRT). Evaluation Model development and assessment should be based upon a credible PIRT. The PIRT should be used to determine the requirements for physical model development, scalability, validation, and sensitivities studies. Given these importance statements, it is necessary to recognize that “the PIRT is not an end in itself”, but is rather a “tool” to be used to guide and focus subsequent efforts.

2.2 PIRT Process

The PIRT process has evolved from its initial development and application (References [2], [7], and [8]) to its description as a generalized process (Reference [5]). A PIRT can be used to support several important decision-making processes. For example, the information can be used to support either the definition of requirements for related experiments and analytic tools or the adequacy and applicability of existing experiments and analytic tools. This information is important because it is neither cost effective nor is it required to assess each feature of an experiment or analytic tool in a uniform fashion. The PIRT methodology brings into focus the phenomena that dominate, while identifying all plausible effects to demonstrate completeness.

A simplified description of the generalized PIRT process, as applied to the development of a PIRT for thermal-hydraulic transients and accidents in nuclear reactors, is illustrated in Figure 1 and described as follows:

Step 1: Define the issue that is driving the need, e.g., licensing, operational, or programmatic. The definition may evolve as a hierarchy starting with federal regulations and/or design and safety goals and descending to a consideration of key physical processes.

Step 2: Define the specific objectives of the PIRT. The PIRT objectives are usually specified by the sponsoring agency. A clear statement of PIRT objectives is important because it defines the focus, content, and intended applications of the PIRT product. The PIRT objectives should include a description of the final products to be prepared.

Step 3: Define the hardware, equipment and scenario for which the PIRT is to be prepared. Generally, a specific hardware configuration and specific scenario are defined. Usually, but not always, the scenario is divided into phases. This is done because the importance of a phenomenon often varies during the course of a scenario. In addition, some system components may not be activated throughout the scenario.

Experience obtained from previous PIRT efforts indicates that any consideration of multiple hardware configurations or scenarios impedes PIRT development. After the baseline PIRT is completed for the specified hardware and scenario, the applicability of the PIRT to related hardware configurations and scenarios can be assessed.

Step 4: Compile and review the contents of a database that captures the relevant experimental and analytic knowledge relative to the physical processes and hardware for which the PIRT is being developed. Each Panel member should review and become familiar with the information in the database.

Step 5: Define the Figure of Merit (FOM). The FOM is the primary evaluation criterion used to judge the relative importance of each phenomenon. Therefore, it must be identified before proceeding with the ranking portion of the PIRT effort. It is extremely important that all PIRT Panel members come to a common and clear understanding of the FOM and how it will be used in the ranking effort. The characteristics of a well-defined FOM is that it is: (1) directly related to the issue(s) being addressed; (2) directly related to the phenomena expected to occur during the scenario; (3) easily comprehended, (4) explicit; and (5) measurable. For design basis accident scenarios, the FOM is usually associated with regulatory requirements.

Step 6: Identify all plausible phenomena i.e., PIRT elements. A primary objective of this step is completeness. In addition to preparing the list of phenomena, precise definitions of each phenomenon should be developed and made available to the PIRT Panel to ensure that Panel members have a common understanding of each phenomenon. Within the context of this PIRT, the term “phenomenon” encompasses phenomena, processes, conditions, characteristics, and state variables. In each PIRT effort, there is a phenomenological hierarchy beginning at the system level and proceeding in turn through the component level, local level, microscopic level, atomic levels and so on. Each PIRT Panel must determine the appropriate phenomenological levels to include in its list of identified phenomena. Insights into the levels to be included can often be derived by considering the data needs for analytic methods and the level at which experimental data is collected. Usually, there is no need to proceed further down the phenomenological hierarchy than (a) the level at which physical processes modeled with analytic methods or (b) the level at which data, either direct or indirect, are acquired.

Step 7: Develop the importance-ranking and rationale for each phenomenon. Importance is ranked relative to the FOM adopted in Step 5. Several ranking scales have been used in the past. However, consistent application of the scale is of equal importance as the specifics of the scale. A word-based scale, e.g., High, Medium, Low or Inactive / Insignificant importance, has proven useful. Numerical scales, e.g., 1-5, have also been used. Outcomes are closely associated with the ranking process and the members of the PIRT Panel should understand the outcomes as they embark on the ranking effort. For example, a phenomenon assigned an importance-rank of High must be simulated with a high degree of accuracy in both experiments and analysis tools while a phenomenon with an importance rank of Low requires significantly less accuracy in both experimental and analytic simulations.

Step 8: Assess the level of knowledge regarding each phenomenon. This is a relatively new step in the evolving PIRT process. It was not included, for example, in a recent generalized description of the PIRT process (Reference [5]). As with importance ranking, several scales have been used in the past. Again, a consistent application of the scale is of equal importance as the specifics of the scale. A numerical scale, e.g., 1-4, which includes in its definitions a statement on uncertainty, has been used. A word-based scale, e.g., Known, Partially Known or Unknown, has also been used. By explicitly addressing uncertainty due to a lack of knowledge, an observed defect of earlier PIRT efforts has been addressed, namely, the tendency of PIRT Panel members to assign high importance to a phenomenon for which Panel members concluded that there was significantly less than full knowledge and understanding.

A consistent outcome of PIRT efforts has been that phenomena found to be highly important relative to the FOM, but for which the knowledge level is insufficient, are carefully examined to determine if additional experiments or analytic efforts are warranted.

Step 9: Document the PIRT results. The primary objective of this step is to provide sufficient coverage and depth that a knowledgeable reader can understand what was done (process) and the outcomes (results). The essential results to be documented are the phenomena considered and their associated definitions, the importance of each phenomena and associated rationale for the judgment of importance, the level of knowledge or uncertainty regarding each phenomenon and associated rationale, and the results and rationales for any assessments of extended applicability for the baseline PIRT. Other information may be included as determined by the Panel or requested by the Sponsor.

As presented in Figure 1, the PIRT process proceeds from start to end without iteration. In reality, however, the option to revisit any step is available and is often exercised during the PIRT development process.

2.3 PIRT Process Application ACR-700 25% Reactor Inlet Header (RIH) Loss of Coolant Accident (LOCA) With Subsequent Loss of Class IV Power

Although the PIRT process has been generalized, there are numerous details that must be addressed for each PIRT application. The initial pioneering PIRT application (References [2] and [7]) considered the response of a specific pressurized water reactor (PWR) design to a large break LOCA. Such plants have been built and operated for a number of years and both the experimental and analytic databases for PWR designs are large. The current PIRT application focuses on the Advanced CANDU Reactor (ACR-700) undergoing a 25% RIH LOCA with subsequent loss of Class IV power (hereafter called the 25% RIH LOCA).

Numerous specific decisions were made during the development of the 25% RIH LOCA PIRT. These are summarized in the following for each PIRT process step described in Section 2.2.

Step 1 – Issue: The United States (US) Nuclear Regulatory Commission (NRC) is conducting a licensing review of the ACR-700. The NRC employs the PIRT process in both its work philosophy and licensing approach. An example of the latter is the extensive use of PIRT results and PIRT-related guidance provided to licensees regarding transient and accident analysis methods used to analyze the transient and accident behavior that are within the design basis of a plant (Reference [6]). Atomic Energy of Canada Limited (AECL) has made extensive use of PIRT-like results in the validation of its computer programs used for safety analysis of the CANDU series and now the ACR design (Reference [3]). This PIRT will provide a basis for confirming the previously developed CANDU-specific phenomena identification and ranking results for the ACR (Reference [3]).

Step 2 - Objectives: The objectives of the ACR-700 25% RIH LOCA PIRT effort were to: (1) develop a PIRT consistent with US practices, (2) develop and document the PIRT to meet the intent of DG-1120 guidance (Reference [6]) for qualifying AECL transient and accident analysis methods for the selected accident scenario, (3) prepare AECL staff to effectively interact with the experts engaged by the NRC to independently develop PIRTs for the ACR-700, and (4) provide a basis for evaluating the previously developed phenomena identification and ranking results for the ACR (Reference [3]).

Step 3 – Hardware and Scenario: This PIRT was developed for the ACR-700 design. A detailed description of the ACR-700 is provided in Reference [1]. A description of the ACR-700 design is also presented in Section 3, including a description of the key features of the ACR-700 design as they relate to the response of the design to a critical break large LOCA.

This PIRT was developed for a 25% RIH LOCA, a critical header break that results in near-flow-stagnation conditions in multiple channels across the core. The scenario was divided into three phases, the blowdown period beginning at time zero and ending at 53 seconds, the late blowdown / Emergency Coolant Injection (ECI) / refill period between 53 and 250 seconds, and the long-term cooling period for times greater than 250 seconds. A detailed scenario description is provided in Section 4.

Step 4 – Figure of Merit: Each phenomenon was assessed relative to a two-fold Figure of Merit. The primary FOM is associated with the temperature response of the fuel sheath (cladding in the US). Specifically, each phenomenon was assessed for its impact on the sheath temperature during each phase of the accident scenario. During the phase in which the sheath maximum temperature is attained (the blowdown period for this accident), each phenomenon was assessed for its impact on the approach of the sheath maximum temperature 1200°C. The secondary FOM is associated with the performance of the pressure tube (PT) during the accident scenario. Specifically, each phenomenon was assessed for its impact on PT high-temperature strain leading to maximum high-temperature plastic strains <2%.

Step 5 – Database: Because AECL staff developed the PIRT documented herein, the experience base of the PIRT Panel was ideal. Specific information about the plant design and response of the design to accident conditions, applicable integral, component, and separate-effect experimental facilities and data, and analytic tools and results were available throughout the course of the PIRT development. The Panel had access to large LOCA design assist documentation, as well as graphic results of a CATHENA (Reference [9]) simulation for the 25% RIH LOCA.

Step 6 – Identify Phenomena: The general description for Step 6 in Section 2.2 stated that the objective was to identify a complete set of plausible phenomena and provide definitions for each phenomenon listed. As applied in this PIRT effort, it was stated that the term "phenomenon" encompasses phenomena, processes, conditions, characteristics, and state variables.

To support the phenomenon identification effort, a generalized list of PIRT phenomena was compiled by considering three potential elements of a component in a thermal-hydraulic system. These three elements are the structure, volume, and flow path associated with each component. Each element was further characterized by its physical characteristics, initial state at the start of the accident, and phenomena that could occur during the scenario. A fourth element, Equipment, was added to address several phenomena that could not easily be introduced into the structure, volume, and flow path construct.

The generalized PIRT phenomena list was assembled using phenomenon lists from three sources. The first source was ACR-700 specific phenomena organized by discipline, several of which were reactor physics, fuel channel and system thermal-hydraulics, fuel and fuel channel thermal-mechanical effects (Reference [3]). These phenomena are further identified in Table 1 with their unique identifier from Reference [3]. The identifier is shown in parenthesis and follows the phenomena entry. The second source was a consolidated list of phenomena of high importance from US PWR and BWR PIRTs (Reference [10]). The third source consisted of several large LOCA PIRTs conducted in the US. These PIRTs included complete lists of phenomena for the given plant and scenario. The PIRT for the AP600 large LOCA is an example (Reference [11]). The resultant consolidated PIRT phenomenon list is presented in Table 1.

The PIRT Panel developed a list of ACR-700 systems. Within each system, the components were listed. Finally, facilitators for this PIRT effort compiled a draft PIRT phenomena list by populating each component with plausible phenomena from the generalized PIRT phenomena list presented in Table 1. This draft PIRT phenomena list and accompanying definitions were then reviewed by the PIRT Panel and revised as needed. The final PIRT phenomena list considered by the Panel is presented in Section 5. The definitions for each phenomenon considered are presented in Appendix A.

Step 7 – Importance Ranking: The Panel ranked each phenomenon in the table relative to the two-fold FOM as described in Step 4 of this section. A summary of the importance-ranking assigned to each phenomenon is provided in Section 5.

Each phenomenon was assigned an importance-rank of "High," "Medium," "Low" or "Inactive/Insignificant." The definitions associated with each of these importance-ranks are shown in the Table 2. The results of a PIRT effort are frequently used to focus attention on the areas in experimental facilities and analysis methods that are most important in accurately representing plant behavior. Thus, Table 2 also presents the "Application Outcomes" associated with each of the importance-ranks as an additional set of information to be used by Panel members as they conduct their ranking efforts.

The Panel employed a hierarchical ranking process. During the first step of the importance-ranking effort, an importance-rank of H, M, L, or I was assigned for each system for each phase of the accident. Next, the importance of each component was evaluated for each phase. However, an important constraint[2] was applied to the component rankings for each phase, namely, that a component could not have a higher importance-rank than assigned to the associated system for the corresponding phase. Again, a constraint[2] was applied to the phenomenon rankings for each phase, namely, that a phenomenon could not have a higher importance-rank than assigned to the associated component for the corresponding phase.

The hierarchical ranking process was employed because this approach has the benefit of limiting the number of phenomena identified as being of "High" importance. The relationship of each phenomenon to power, flow, and the power/flow ratio was

[2] The rationale may indicate that the higher-level object(s) should have the assigned rank either increased or reduced.

considered. These factors are highly significant to the two FOMs, each element of which is directly impacted by these three factors.

Step 8 – Knowledge Level: Panel members assessed the current knowledge level for each phenomenon in the PIRT table. The knowledge level for each phenomenon was assessed relative to the impact of each phenomenon on the FOM. Numbers between 1 and 4 were assigned to reflect the knowledge level with the associated definitions shown in Table 3.

Step 9 – Documentation: This document represents the realization of the documentation step. An introduction to this PIRT effort is provided in Section 1. The general PIRT process and its specific application to the 25% RIH LOCA PIRT effort are documented in Section 2. A description of the ACR-700 plant and the key systems and components that respond to the selected scenario are described in Section 3. The 25% RIH LOCA scenario for the ACR-700 is described in Section 4. The three phases of the accident for which PIRT assessments were made are described. The three phases are the blowdown period, late blowdown/ECI/refill period, and the long-term cooling period. Summary PIRT results are presented in Section 5. Details of the PIRT results are presented in the Appendices. Phenomena definitions are provided in Appendix A.

3. PLANT DESCRIPTION

The Phenomena Identification and Ranking Table (PIRT) is plant specific. Therefore, a brief description of the ACR-700 system is provided as a necessary element in this documentation of the PIRT evaluation (Reference[1]). Technical data for an ACR-700 reactor unit is provided in Table 4. In addition, a brief review of those systems and components having the greatest impact on the course of the LBLOCA during the blowdown, ECCS injection, and long-term cooling periods of the transient is provided.

3.1 ACR-700 Description Overview

Atomic Energy of Canada Limited (AECL) has developed the ACR-700 (Advanced CANDU Reactor-700 to meet customer needs for reduced capital cost, shorter construction schedule, high capacity factor, low operating cost, increased operating life, simple component replacement, and enhanced safety features.

The ACR design is based on the use of modular horizontal fuel channels surrounded by a heavy water moderator, the same feature as in all CANDU reactors. The major innovation in ACR is the use of slightly enriched uranium fuel, and light water as the coolant, which circulates in the fuel channels. This results in a more compact reactor design and a reduction of heavy water inventory, both contributing to a significant decrease in cost compared to CANDU reactors that employ natural uranium as fuel and heavy water as coolant.

The design also features higher pressures and temperatures of reactor coolant and main steam, thus providing an improved thermal efficiency with respect to the existing CANDU plants. These thermal-hydraulic characteristics further emphasize the ACR drive towards improved economics.

The above changes and other evolutionary design improvements are well supported by the existing knowledge base and build on the traditional characteristics of the CANDU system, including: proven, simple and economical fuel bundle design; on-power fuelling; separate cool, low-pressure moderator with back-up heat sink capability; and low neutron absorption for good fuel utilization.

The safety enhancements made in ACR encompass safety margins, performance and reliability of safety related systems. In particular, the use of the CANFLEX fuel bundle, with lower linear rating and higher critical heat flux, permits increased operating and safety margins of the reactor. Passive safety features draw from those of the existing CANDU plants (e.g., the two independent shutdown systems), and other passive features are added to strengthen the safety of the plant (e.g., a gravity supply of emergency feedwater to the steam generators).

The reactor consists of a set of 284 horizontally aligned fuel channels arranged on a square pitch. The fuel channels contain the fuel and the high-pressure light water coolant. They are mounted in a calandria vessel containing the heavy water moderator. Individual calandria tubes surround each individual fuel channel.

The calandria vessel is enclosed by end shields, which support each end of the calandria. They are filled with steel balls and water to provide shielding. The fuel channels are located by adjustable restraints on the two end shields and are connected by individual feeder pipes to the Heat Transport System (HTS).

The calandria vessel is enclosed in a larger shield tank. The shield tank has a reactivity mechanisms deck mounted on its top and has horizontal penetrations for more reactivity systems. A series of thimbles are used to connect various reactivity and shutdown systems through to the calandria vessel.

3.2 Reactor Heat Transport System

The major components of the heat transport system (HTS) are the 284 reactor fuel channels, two vertical steam generators, four motor driven pumps, two reactor inlet headers (RIH), two reactor outlet headers (ROH), and inter-connecting piping. The HTS is arranged in one closed circuit. A simplified heat transport circuit schematic is shown in Figure 2.

The fuel channels are horizontal and allow access to both ends by the fuelling machines. The headers, steam generators and pumps are located above the reactor to provide thermosyphoning if power is lost to the heat transport pumps. The general layout of the heat transport system in the reactor building is illustrated in Figure 3.

The fuel channel assemblies support and locate the fuel within the reactor core. They allow for flow of the heat transport coolant without leakage, and they also provide for shielding. Each fuel channel assembly consists of a pressure tube, two end fittings and associated hardware. Feeder pipes connect the inlet and outlet end fittings to the reactor inlet header and reactor outlet header, respectively, to complete the heat transport system loop. Each feeder pipe is connected to an end fitting sideport by a welded connection.

Light water coolant flows into the inlet end fitting, through the holes in the liner tube into the central circular section, then into the concentrically aligned shield plug in the end fitting and is directed into the pressure tube. Each end fitting incorporates a shield plug to provide shielding.

The outboard end face of each end fitting makes a sealed connection with the fuelling machine to perform on-power fuel insertion and removal. The channel closure is removed and stored by the fuelling machine during refueling and is re-installed in the end fitting before the fuelling machine comes off the channel.

3.3 Steam Generators

Two identical steam generators with integral preheaters transfer heat from the reactor coolant on the steam generator primary side to raise the temperature of, and boil, feedwater on the steam generator secondary side. The steam generator consists of an inverted vertical U-tube bundle installed in a shell. Steam-separating equipment is housed in the upper portion of the shell.

3.4 Heat Transport Pumps

The four heat transport pumps are vertical, single stage centrifugal pumps with single suction and double discharge. When maintenance of the shaft seals or the pump internals is required, the coolant level in the HTS can be lowered to a level below the pumps. The Long-Term Cooling system cools the HTS after a reactor shutdown to a temperature suitable for maintenance, maintains that temperature, and provides a means of draining, refilling and level control of the HTS to allow for this maintenance.

A gland seal external circuit supplies cooled and filtered water for lubricating and cooling the mechanical seals. A leakage recovery cavity takes the seal leakage to the leakage collection system. Each pump is driven by a vertical, totally enclosed, air and water-cooled squirrel cage induction motor. The motor has built-in inertia to prolong pump rundown on loss of power.

3.5 Feeders

The feeders at each end of the reactor run from the fuel channels horizontally or vertically up the face of the reactor and then horizontally across and above the fuelling machine area to the reactor headers. The feeders are sized to provide the necessary flow to each channel and to fit in the space between the end fittings. They are welded to the feeder connection provided on each end fitting. Both the inlet and outlet lower feeders are stainless steel pipes to prevent flow-assisted corrosion. The headers and a portion of the upper feeders are carbon steel with 0.3% Cr.

3.6 Fuel Channel Assembly

The fuel channel assembly comprises a pressure tube, a calandria tube, two end fittings (one at each end of the pressure tube), and various internal components. Each pressure tube is thermally insulated from the low temperature, low pressure moderator by a carbon dioxide (CO_2) filled gas annulus between the pressure tube and calandria tube. Spacers, positioned along the length of the pressure tube, maintain the annular space and prevent contact between the two tubes.

The zirconium alloy (Zr-2.5% Nb) pressure tube is attached to the inboard end of the stainless steel end fitting by a roll-expanded joint.

The fuel channel is designed to achieve a target 30-year operating life with a 90 percent capacity factor. The fuel channel design will accommodate the predicted sag, axial and diametral growth due to irradiation of the pressure tube over this operating life. The fuel channel assemblies are designed to allow for replacement.

3.7 Fuel Bundle

Each fuel channel is fuelled with twelve CANFLEX fuel bundles fabricated from Zircaloy-4 components and sintered UO_2 fuel pellets (Reference [12]). The fuel bundles are short (approximately 50 cm long) and 4 cm in diameter. Each fuel bundle consists of 43 fuel elements arranged in concentric circles of 7, 14 and 21 fuel elements about a central fuel element. The fuel elements are welded at each end to thin flexible end plates. Spacer pads are fixed to the fuel elements to maintain fuel element separation. Bearing pads are attached to the outer fuel elements to centre the bundle in the fuel channel and avoid fuel element contact with the pressure tube. Flow turbulence buttons are also fixed to the sheath surface to improve heat transfer.

The fuel sheath is thin and creep collapses under coolant pressure onto the fuel pellet stack early in irradiation. Good fuel/sheath contact, low fission product gas release and the use of a helium fill gas all promote lower fuel operating temperatures.

3.8 Headers

There are two reactor outlet headers, one at each end of the reactor. Each of the reactor outlet headers receives the flow from the outlet feeders on one reactor face and conducts the flow to two steam generator inlet lines, which lead to a single steam generator.

There are two reactor inlet headers, one at each end of the reactor. Each of the reactor inlet headers receives the flow from two heat transport pumps through four discharge lines and channels the flow to the inlet feeders on one reactor face.

Design margin is provided in the steam generator to cater for fouling and tube plugging if required as the plant matures. The ACR-700 reactors are designed for reactor inlet header operating temperature value of about 278.5°C, expected with steam generator fouling corresponding to ageing maturity at mid-life of the plant. For a new plant, the RIH operating temperature is lower than the design value.

3.9 Pressurizer

The heat transport system is a pressurized light water closed loop. The heat transport pressure and inventory control (P&IC) system is designed to provide a means of pressure and inventory control for this closed loop as well as to provide adequate overpressure protection. The control of pressure and inventory is achieved using a distributed control system (DCS). Overpressure protection is independent of the DCS.

The pressurizer is a major component of the Pressure and Inventory Control System. The pressure vessel is partly full of coolant in liquid phase with the remainder being saturated vapour in equilibrium with the liquid. During normal operating condition, the pressurizer is connected to the HTS. This is called ‘normal mode’ operation of the HTS pressure control system.

Controlling the pressure of the vapour space in the upper region of the pressurizer controls the reactor outlet header pressure at the setpoint. The pressure is increased by activating the heaters in the vessel and reduced by a controlled spray flow supplied by a line connected to a reactor inlet header. When the HTS is stable under normal conditions, one variable heater is used to compensate for pressurizer heat loss. During an upward maneuvering condition, the pressurizer spray is used to control increasing pressure.

The pressurizer level setpoint increases with reactor power and the water level is automatically controlled at the setpoint by the feed and bleeds circuit under the control of the unit computers. In ‘solid mode’, the pressurizer is isolated from the HTS circuit. In this case, the pressure control of the HTS is achieved by feed and bleed. Duplicated and instrumented steam relief valves connecting to the bleed condenser provide overpressure protection for the pressurizer. During a shutdown, the isolated pressurizer normally remains pressurized. Swell and shrinkage in the heat transport system, during warmup, startup, shutdown and cooldown, are accommodated in the pressurizer and the coolant storage tank, and are compensated for by bleeding from the system via the bleed circuit or feeding into the system via the feed circuit.

3.10 Feed and Bleed System

The feed and bleed circuit is provided for the inventory control of the HTS. The feed and bleed circuit is designed to handle the shrinkage and swell rates which take place during warmup and cooldown. Two high-pressure multi-stage pressurizing pumps are provided, one of which is normally operating with the other on standby.

The bleed flow is discharged from the HTS into the bleed condenser and cooled by a reflux tube bundle with a flow from the discharge of HTS pressurizing pumps. By passing through the bleed cooler, the coolant is further cooled down to 66°C (150°F). The water is routed to the heat transport purification system for filtering and purification.

The normal feed flow is taken from the HTS purification circuit and the coolant storage tank. During some abnormal situations (for example, design basis events such as relief valve failure, small loss of coolant accident, etc.), the heat transport system pressure falls and the HTS pressurizing pump flow subsequently increases. In order to avoid the pressurizing pump from tripping due to low suction pressure, override of the feed valves is provided such that the feed valves begin to throttle to reduce the flow.

During normal operation, the pressurizing pump also supplies purified coolant flow to the fuelling machines via a booster pump for the refueling operation.

3.11 Main Steam Safety Valves (MSSVs)

A total of eight spring loaded and pneumatic operated safety valves are provided, four per steam generator. The combined capacity of three out of four MSSVs provides a capacity of 120% of the steam flow from each steam generator. The steam relief capacity of 120% has been chosen because the reactor power could, under a slow loss of regulation, go as high as 120% before a reactor is shut down.

3.12 Main Steam Isolation Valves (MSIVs)

The MSIVs, installed downstream of the atmospheric steam discharge valves (ASDVs), are motorized and are remote, manually operated from the main control room. Appropriate MSIVs will only be closed after reactor shutdown when the long-term cooling system is placed in service and the heat transport system is depressurized, following leakage from the primary side of the steam generator to the secondary side. To avoid steam hammer, the MSIV closing time is approximately two minutes.

3.13 Atmospheric Steam Discharge Valves (ASDVs)

A total of four globe type control valves are provided. These valves have a total capacity of 10% of the nominal steam flow and are used as a heat sink when the main condenser is either unavailable or is inadequate. The valves are actuated in response to the steam generator pressure control program demands. The ASDVs are used during normal plant operation, plant warmup, under loss of Class IV power, loss of condenser, turbine trip, or loss of the line when MSIVs are available and open.

3.14 Condenser Steam Discharge Valves (CSDVs)

The main function of these valves is to discharge live steam from the main steam balance header to the condenser. They are used to discharge steam during severe transients, such as turbine trip, to avoid activating the safety valves.

Their operating characteristics are as follows:

- During normal operation, they are on pressure control with an offset to bias them closed.
- During poison prevent, their steady-state opening is proportional to the power mismatch between poison prevent level and actual turbine steam consumption.

- Their opening is conditioned by the condenser protective system. The valves may be automatically tripped by the condenser protective system; in this event, a manual reset is required before they can be reopened.
- On a turbine trip, a signal is applied to open quickly. They will revert to the pressure control mode after they have opened fully.
- Provision is made to allow the operator to open them via computer.

3.15 Moderator Systems

The moderator system consists of a closed heavy water recirculating loop, which serves to cool and circulate the heavy water moderator through the calandria. The high purity D_2O moderator, used to slow fission neutrons to sustain criticality, is circulated through the main moderator circuit during normal operation. Heat generated within the moderator is removed in the moderator circuit to maintain a constant moderator temperature. The moderator circuit also acts as a medium for dispersion of reactivity control agents. During a loss of coolant accident coincident with a loss of the long-term cooling system, the moderator system acts as a heat sink, with heat being removed by the moderator heat exchangers.

3.16 Calandria Rupture Disc

The calandria rupture disc assemblies are not the principal pressure relief devices required by the ASME Code for the calandria vessel, the moderator system, or the moderator cover gas system.

They are backup devices to the normal means of pressure relief, which limit the magnitude of the peak pressure in the calandria for “emergency” conditions. Each rupture disc assembly consists of a weld neck mounting flange (suitable for welding to the relief duct), an inlet flange, a rupture disc, an outlet flange/knife blade holder, studs or bolts, and nuts. The rupture disc is clamped between the inlet and outlet flanges. The rupture disc is required to burst if the internal pressure of the moderator cover gas system reaches a specified level.

3.17 Reactor Containment System

The containment system is an envelope around the nuclear components of the heat transport system where failure of these components could result in the release of a significant amount of radioactivity to the public. Because of the large amounts of energy stored in the heat transport system, the envelope must withstand a pressure rise. The criterion for determining the effectiveness of the envelope is the integrated leak rate for the period of the pressure excursion.

To meet the design leakage requirements two diverse principles are used. The first involves the detailed design of the envelope to minimize the leak rate. The envelope comprises a primary containment, and a system to automatically isolate or “button up” the reactor building after a loss-of-coolant accident. The second method involves a system that will absorb the energy released to the envelope, thus reducing the peak pressure and the duration of the pressure excursion. The building local air coolers do this.

3.18 Emergency Core Cooling System

Following a loss-of-coolant accident, the reactor shutdown and emergency core cooling systems acting together must, as a design target, prevent excessive fuel damage. In the event of a major break in the heat transport system, the water escapes through the break, depressurizing the system (the blowdown phase). The reactor is tripped automatically. The combination of increase in pressure differential across the fuel sheath caused by the gaseous fission products and the increase in sheath temperature is a factor affecting the sheath failure threshold during blowdown. If the threshold is exceeded, the sheath can swell and could result in sheath rupture. However, during blowdown the sheath temperature increase is limited and excessive sheath failures are prevented. The need to remove residual heat in the fuel at the end of blowdown, and decay heat produced thereafter, leads to the requirement for an emergency core cooling system.

The emergency core cooling system is designed to supply emergency coolant to the reactor in two stages. During the high pressure stage, water is injected into the reactor core via the ECI system on a LOCA signal. To enhance the effectiveness of this high pressure injection, the main steam safety valves are also opened on a LOCA signal to provide a rapid cooldown of the steam generators and depressurization of the heat transport system. When the HTS pressure drops below the rupture pressure of the one-way rupture discs, the rupture discs burst, thereby enabling ECI coolant injection to the reactor inlet headers. In addition, valves on the ECI interconnect line between the reactor outlet headers, open on a LOCA signal to assist in establishing a cooling flow path.

The long-term cooling (LTC) system for long-term recirculation/recovery after LOCA is the second phase. For a LOCA, the LTC system is initiated following operation of the ECI system. On the LOCA signal, water is automatically introduced into the containment sumps and the LTC pumps start automatically. When the water accumulators are nearly empty, the ECI accumulator isolation valves close and the recovery stage begins by pumping water from the sumps into the HTS via the LTC heat exchangers and thus the LTC is initiated. The LTC delivers flow to the reactor inlet headers, thereby utilizing the cooling path already established by the high pressure ECI system. The LTC system is also used for long-term cooling of the reactor after shutdown following other accidents and transients.

3.19 Reserve Water System

The ACR design includes a reserve water system (RWS) with a reserve water tank, which is located at a high elevation in the reactor building and provides an emergency source of water to the containment sumps for recovery by the long-term cooling system in the event of a LOCA. In addition, the tank provides emergency water by gravity to the steam generators (emergency feedwater), moderator system, shield cooling system and the heat transport system if required.

3.20 Long-Term Cooling System

The LTC system provides fuel cooling in the long-term (recovery stage) of a loss of coolant accident (LOCA) following ECI operation, and serves to remove decay heat in the long-term following transients and accidents with the HTS pressure boundary intact, or following a normal reactor shutdown.

The LTC system can remove decay heat either from an intact HTS during steam generator or HTS pump maintenance, or following a LOCA. The process equipment of this system is designed to handle the highest loads for all required modes of operation.

4. LBLOCA SCENARIO DESCRIPTION

The large break Loss-of-Coolant Accident (LBLOCA) has been the most extensively analyzed design basis event for nuclear power plants around the world as it provides a severe analytic test for assessing safety system performance.

A qualitative description of a LBLOCA event sequence is as follows:

- A large break is postulated to occur in a large diameter pipe of the Heat Transport System (HTS), discharging coolant into containment.
- The pressure, temperature and humidity of the containment atmosphere increase.
- The HTS depressurization causes coolant voiding in the core and a decrease in core reactivity.
- The reactor shuts down on a process trip (e.g., low HTS pressure, low HTS flow) or on Reactor Building high pressure depending on the break size and the initial reactor power.
- Containment isolation is automatically initiated on a high reactor building pressure signal. The high reactor building pressure signal also conditions the ECI signal and both of the two steam generator crash cooldown signals. The second SG crash cool signal is generated with the same set of parameters as the ECI signal but with an independent set of instrumentation. Both signals open the RWT valves to the inlet headers.
- The HTS loses inventory and depressurizes at a rate depending on the break size and location.
- Following reactor trip, the turbine trips and runs down. The condenser steam dump valves (CSDVs) open to by-pass steam to the condenser. The atmospheric steam discharge valves (ASDVs) open and close to maintain system pressure.
- The main feedwater system feeds the steam generators from the condenser hotwell until Class IV power is lost. Subsequently, water is supplied by auxiliary feedwater pumps, which run on Class III power.
- The HTS flow decreases faster in the core pass downstream of the break. If the break is large enough, the flow will reverse in that pass. For some break sizes, the flow momentarily falls very low as the break upstream of the core pass balances the pumps. Some channels may become steam-filled and others may experience stratified two-phase flow, exposing some fuel elements to steam cooling. Fuel temperatures rise. A rise in fuel temperatures increases the internal fuel element gas pressures, whereas a rise in sheath temperatures reduces the sheath strength. Increased internal fuel element gas pressure along with the decreased coolant pressure increases fuel sheath stresses. If the fuel temperature becomes high enough, sheath failure can occur.
- The pressurizer discharges its inventory into the HTS. The decreasing pressurizer level causes the light water bleed valves to close, and feed valves to open up, adding light water makeup to the HTS.
- Following reactor trip, the average fuel temperature decreases as the heat generation rate decreases and the temperature profile in the fuel pin flattens out. The sheath temperature increases depending on the heat transfer from the sheath to the coolant, and subsequently decreases due to the reduced heat generation rate and blowdown cooling flow.

- When the HTS pressure falls below a specified setpoint, the ECI signal, which is conditioned by the high reactor building pressure signal, is generated. This signal results in the following events:
    - The high-pressure Emergency Coolant Injection (ECI) System is initiated by the ECI signal. The isolation valves in the lines from the high-pressure ECI water storage tank to the injection points into the HTS reactor inlet headers are opened. The one-way rupture discs burst open at a pressure differential of 0.52 MPa (0.52 MPa is the highest burst pressure observed in experimental results). The ECI piping downstream of the one-way rupture discs is pressurized to the heat transport system pressure. Thus, the high-pressure ECI flow will begin when the pressure in the HTS is about 0.52 MPa less than the high-pressure ECI pressure from the ECI water storage tank. High-pressure ECI continues until the associated high-pressure ECI water storage tank is nearly empty.
    - Valves on the ECI large interconnect line between the reactor outlet headers open on the ECI signal to assist in establishing a cooling flow path.
    - Steam generator crash cooldown is initiated 30 seconds after the ECI signal through the automatic opening of the main steam safety valves (MSSVs). This assists the ECI by further depressurizing the HTS.
    - On the ECI signal, water is automatically introduced into the containment sumps from the Reserve Water System (RWS). The long term cooling (LTC) pumps start automatically on a high reactor-building sump level signal. When the ECI water storage tanks are nearly empty, the storage tank isolation valves close and the LTC stage begins by pumping water from the reactor-building sump. The LTC delivers flow to the reactor inlet headers, thereby utilizing the cooling flow path already established by the high-pressure ECI system.
- Soon after ECI and SG crash cooldown begins, emergency coolant water begins to refill the core pass.
- The ECI refills both core passes and a quasi-steady-state flow pattern is established.
- Long-term cooling is maintained by the flow of LTC system coolant through the circuit, with decay heat removal by the LTC heat exchangers and through the break.

The phases of a LBLOCA accident are defined according to the major time periods during the accident progression for which characteristic system behaviors are exhibited. The ACR Technical Basis Document (TBD) (Reference [3]) identified the phases for each of the disciplines involved in large break LOCA analysis. The containment behavior, fission product release and transport, and radiation physics disciplines are not relevant to the current Figure of Merit (sheath peak temperature or pressure tube high-temperature strain).

The identified phases of the ACR LBLOCA are,

1. Early blowdown cooling: The period during which the reactor is being shut down and the HTS blowdown continues prior to ECI initiation (i.e., opening of the ECI rupture disc). The dominant system behavior during this period is a result of heat transport system depressurization, reactor shutdown, blowdown cooling, fuel and sheath heat-up, pressure tube heat-up, possible fuel failure and consequent fission-product release.

2. Late blowdown cooling / ECI / Refill: The period of ongoing heat transport system blowdown with ECI inventory entering into the heat transport system. The dominant system behavior during this period again is due to heat transport system depressurization, blowdown cooling, ECI delivery, fuel heat-up, pressure-tube heat-up and fission product release. During the later stages, refill of channels in the core proceeds and a quasi-steady-state is attained. The dominant system behavior during this period is determined by ECI delivery, heat transport system refill, fuel cooling and possible fission product release.
3. Long-term cooling: The period in which the long-term cooling (LTC) provides recovery flow to maintain core cooling.

The duration of each of these phases depends on the break size with longer phases associated with smaller breaks.

The PIRT activity described in this document is specific to the ACR-700 reactor for the case of a 25% RIH LBLOCA with subsequent loss of Class IV power[3]. At the time of the transient, the reactor is operating at full power with equilibrium poison levels. Design values are used for plant parameters.

The events occurring during the 25% RIH LBLOCA scenario are described in detailed point-form in Table 5. The description is based on a CATHENA (Reference [9]) simulation of this accident (Figure 6 to Figure 206).

The CATHENA circuit model consists of a one-loop Heat Transport System (HTS), Steam and Feedwater System, and Emergency Core Cooling System (ECCS). Figure 4 shows the CATHENA nodalization of the HTS. There are two inlet headers (RIH1 and RIH2), two outlet headers (ROH1 and ROH2), two steam generators, and four HTS pumps. There are two core passes. Each core pass is represented by an average channel that models 142 fuel channels. ECC is injected into RIH1 and RIH2. The break is specified to occur in RIH2. The downstream pass (i.e., the channel immediately downstream of the break) is labeled CHAN2-1 in Figure 4. The sheath temperatures of primary interest arise in the downstream pass. The second channel, labeled CHAN1-1 in the CATHENA circuit model, is called the upstream pass. This terminology is used in this scenario description.

The CATHENA model nodalization of the ECCS is represented in Figure 5.

The simulation employs some conservative assumptions, both as to the initial conditions and to the modeling of a limited number of phenomena. For example, the variation of reactor power with time is an input rather than calculated; no credit is taken for negative reactivity insertion due to void generation. However, the calculated results do capture the key processes and phenomena expected to occur during the LOCA scenario.

The phase overview descriptions that follow emphasize the sheath temperature behavior (i.e., the Figure of Merit) and the processes/phenomena that cause the predicted behavior. While discussing the sheath temperature response, the system and component interactions leading to the sheath temperature are discussed. Clearly, the processes/phenomena occurring in some components have a greater impact than those occurring in other components. Thus, the contributions of all components to the sheath temperature response are not covered in the same

[3] With respect to Figure 4-1, the break occurs in the header, RIH2.

detail. The components above the headers, for example, receive less attention as they contribute little to the sheath temperature response during Phases 2 and 3. Exceptions, when they occur, are discussed.

4.1 Blowdown Period (Phase 1: 0 – 53 seconds)

Phase 1 is initiated by a break in the reactor inlet header (RIH2). Reactor coolant is discharged at a high rate through the break, flow immediately reverses in the downstream pass, and the void fraction in the channel rapidly increases. In this environment, the sheath temperatures rapidly increase, due both to the power generated in the fuel and the redistribution of stored heat to the sheath.

The sheath temperatures experience two peaks during Phase 1 (Figure 6). The first peak occurs at 7 seconds after break initiation. The rise in sheath temperature is terminated by a reduction in power following reactor trip, and a reduction in break flow as voiding starts in RIH2 and the break begins to pass two-phase flow, which results in a brief period of positive flow from RIH2 into the downstream pass. The sheath temperature continues to decrease as forward flow supplied by the RCS pumps continues into the downstream pass. The valves in the large header interconnect line open and flow passes through the large header interconnect line into ROH1. However, this flow does not immediately cool the downstream pass. The cooling of the sheath deteriorates as the RCS pump flow decreases due to pump suction voiding, causing the flow in the downstream pass to decrease and reverse. A second sheath temperature peak occurs at 18.5 seconds. This period of sheath heating is concluded as the decay power continues to decrease and the reverse flow in the downstream pass increases in magnitude. The reverse flow is maintained as the large header interconnect flow continues, albeit at a reduced level, and the inventory in the hot leg (or up leg) of steam generator #1 drains into ROH1 whence it proceeds through the feeder pipe to the downstream pass channel. The sheath temperature continues to decrease until the end of Phase 1 at 53 seconds.

A detailed description of the processes and phenomena occurring during Phase 1 follows.

- The accident scenario is initiated by a break in RIH2. Taking a 100% double-ended guillotine break of the cylindrical cross section of the RIH as twice the cross-sectional area of the RIH, the area of a 25% break is 50% of the cross-sectional area of the RIH.
- For the first two seconds after break initiation and by assumption, the reactor continues at full power (Figure 7). The downstream pass channel, i.e., the channel immediately downstream of the break, rapidly voids (Figure 67) and the sheath temperatures increase at a near adiabatic rate (Figure 6).
- The stored energy in the fuel bundles is redistributed from the fuel to the sheath, contributing to the early rise in sheath temperature. One consequence of the early voiding in the downstream pass channel is that little of the stored energy is rejected to the coolant compared to the stored energy release in the upstream pass channel (Figure 73). As the energy is not transferred to the coolant, it contributes to the sheath temperature rise as the temperature gradient through the fuel elements flattens.
- At 2.0 seconds, Shutdown Systems 1 (SDS1) and 2 (SDS2) activate and reactor power rapidly decreases to approximately 11% of full power at 3.5 seconds (Figure 7). The rate of sheath temperature increase slows in response to the reduced power level. The rate of sheath

temperature increase approaches zero at 5.5 seconds but a small calculated decrease in channel flow causes the sheath temperature to resume its increase to a temperature of 1066ºC at approximately 7 seconds (Figure 6).

- At 7 seconds, the sheath temperature begins to rapidly decrease from its maximum, reaching a temperature of 937ºC at 11.5 seconds and then decreasing. There are multiple processes that produce this result.
  - The reactor power has been reduced to decay heat levels.
  - RIH2, the inlet header in which the break occurs, remains liquid filled until approximately 5 seconds, at which time the header begins to void (Figure 31 and Figure 32).
  - With the onset of voiding in RIH2, the break flow begins to rapidly decrease from 5900 kg/s to approximately 2450 kg/s at 9 seconds (Figure 10 and Figure 11).
  - While the break flow is rapidly decreasing, the reactor coolant pump is still delivering two-phase fluid to RIH2 (Figure 82 and Figure 83), although at a rapidly decreasing rate. The flow delivered to RIH2 is 2000 kg/s at 7 seconds and approximately 1350 kg/s at 10 seconds. The reduced flow is a result of voiding at the pump suction (Figure 112 and Figure 113).
  - At 7 seconds, the combination of reduced break flow and continued delivery of two-phase fluid from the pump to RIH2 produces a small coolant flow from RIH2 through the feeders (Figure 148 and Figure 149) to the downstream pass channel. This flow terminates the sheath temperature increase and the sheath temperatures begin to decrease (Figure 6).
  - The large header interconnect valve begins to open at approximately 7 seconds and coolant flows from ROH2, through the large interconnect line (Figure 76) into ROH1.
  - A portion of the large interconnect line flow is contributed by the pressurizer, which discharges into ROH2 (Figure 145).
- At approximately 11.5 seconds, the sheath temperatures begin to once again increase (Figure 6). There are several processes that contribute to this outcome.
  - The reduction in sheath temperature at about 7 seconds was the combined result of restored positive flow through the feeder to the downstream pass channel (Figure 148 and Figure 149).
  - Beginning at approximately 11.5 seconds, the flow characteristics are altered.
  - The inlet feeder flow rapidly decreases and then reverses (Figure 148 and Figure 149) under the influence of a reversal of the RIH2 to ROH1 pressure difference (Figure 28).
  - The interconnect flow, which reached a maximum for this phase at 11.5 seconds, begins to decrease (Figure 76).
  - The HTS pump head and flow continue to decrease during this period due to pump suction voiding (and also pump trip), while the break flow remains almost constant.

    - The combination of reduced flow into the downstream pass channel combined with a nearly constant decay power results in a resumption of sheath heating (Figure 6) that continues until approximately 18.5 seconds.
- The next major inflection point in sheath temperature begins at approximately 18.5 seconds when cooling is restored to the downstream pass channel and the sheath temperature begins to decrease. With the exception of a minor and short-lived temperature increase beginning at 22 seconds, the sheath temperature continues to decrease until it reaches 350ºC at 53 seconds, the end of the first phase. There are several processes that contribute to cooling of the sheath during the remainder of Phase 1.
    - The decay power decreases from 7 percent to 5 percent of full power between 10 and 50 seconds.
    - A reverse flow is established passing through ROH1 (Figure 46 and Figure 47), the outlet feeder (Figure 148 and Figure 149), the downstream pass channel, the inlet feeder, and into RIH2 (Figure 55). The reverse flow through the downstream pass channel is small at 18.5 seconds but increases with time. At 30 seconds, a reverse flow of 350 kg/s has been established. At this time the large interconnect line flow has decreased to 135 kg/s (Figure 76). The remainder of the flow passing through ROH1 is liquid draining downward from steam generator #1 (Figure 118 and Figure 119).
    - The rate of sheath temperature decrease slows at 42 seconds (Figure 6), which is caused by a reduction in the downstream pass channel flow rate (Figure 55). The reverse downstream pass channel flow derives from flows into ROH1. These come from two sources, the large interconnect line flow and the drainage flow from steam generator #1. The large interconnect line flow is negligible at this time (Figure 76). Also, the drainage flow from steam generator #1 approaches zero ((Figure 118 and Figure 119)). However, there is sufficient inventory remaining in ROH1 and the outlet feeder to continue the cooling trend by supplying flow through the outlet feeder (Figure 148 and Figure 149) to the downstream pass channel, although the cooling continues at a reduced rate (Figure 6). The coolant moves under the negative pressure gradient (relative to steady-state operation) between RIH2 and ROH1 (Figure 28).

4.2 Late Blowdown / ECI / Refill (Phase 2: 53 – 250 seconds)

Phase 2 is initiated by firing of the high-pressure ECI system. Two nitrogen-pressurized ECI accumulators inject coolant into inlet headers RIH1 and RIH2. As the break occurs in RIH2, little or no effective cooling is obtained from the ECI accumulator discharging into RIH2.

The behavior of the maximum sheath temperature during Phase 2 is most directly related to the sequential process of refilling the headers and components below the headers in the upstream pass and then the headers and components below the headers associated with the downstream pass.

As ECI begins, the HTS is highly voided. The ECI flow into RIH2 condenses steam and much of the coolant remaining in the system is pulled through the downstream pass and flows out the break. The ECI flow into RIH1 then begins to refill what is, at that time, a highly voided system.

The refilling begins with RIH1, moves through the upstream pass, fills ROH2 and re-establishes flow through the large header interconnect. Filling of components connected to the downstream pass then commences. As filling of ROH1 proceeds, two-phase flow passes into the downstream pass channel and terminates the temperature increase that began shortly after the start of Phase 2. Shortly before the end of Phase 2, the downstream pass channel is refilled.

A detailed description of the processes and phenomena occurring during Phase 2 follows.

- At the transition from Phase 1 to Phase 2 at 53 seconds, the maximum sheath temperature is decreasing as residual inventory in ROH1 and the outlet feeder for the downstream pass moves through the downstream pass to the inlet feeder and RIH2. However, there is little flow into ROH1 from either the large header interconnect or draining from steam generator #1 at the end of Phase 1.
- Cooling of the sheath continues until approximately 61 seconds, after which the maximum sheath temperature begins to increase (Figure 205).
- ECI flow into RIH2 begins at about 53 seconds; ECI flow into RIH1 begins shortly thereafter (Figure 202). With the ECI flow into RIH2, the break flow rate increases immediately to 2000 kg/s (Figure 12 and Figure 13). However, the ECI flow into RIH2 shortly after the start of injection is between 900 and 1050 kg/s.
- At the start of ECI flow into RIH2, the header is almost fully voided (Figure 33 and Figure 34). The steam in RIH2 begins to condense, rapidly decreasing the RIH2 pressure and increasing the negative pressure difference from ROH1 to RIH2 (Figure 29). With the increased pressure difference, a two-phase mixture is pulled through the downstream pass and into RIH2 to be condensed by the ECI flow (Figure 150 and Figure 151). The flow pulled through the downstream pass under the influence of the reduced RIH2 pressure due to condensation, combined with ECI flow, constitutes the break flow immediately after the start of ECI.
- Although ECI flow into RIH1 begins at 55 seconds, there is a time delay before a flow rate approaching the ECI flow into RIH1 reaches the downstream pass through the large header interconnect (Figure 77). During the interval between 55 and 88 seconds, the maximum large header interconnect flow is 60 kg/s. At 88 seconds, the large header interconnect flow rapidly increases and maintains an average flow rate of approximately 400 kg/s until the end of Phase 2 at 250 seconds.
- The interval between 55 and 88 seconds is a period of refilling volumes. RIH1 is almost fully voided as the ECI flow begins and is the first component to refill at 62 seconds (Figure 33 and Figure 34). The inlet feeder for the upstream pass is the next to refill at 65 seconds. The outlet feeder for the upstream pass fills at 87 seconds (Figure 162 and Figure 163) and a rapid increase in flow through the large header interconnect follows immediately (Figure 77).
- The filling of components in the downstream pass path by the flow through the large header interconnects then begins. ROH1 fills at 155 seconds (Figure 33 and Figure 34). The outlet feeder for the downstream pass channel fills at approximately 215 seconds (Figure 162 and Figure 163). The inlet feeder for the downstream pass channel first refills at 215 seconds and permanently refills at 225 seconds (Figure 162 and Figure 163).

- The downstream pass channel begins to refill at 105 seconds (Figure 68). At 180 seconds, the downstream pass channel first refills. It remains liquid filled for 10 seconds, has a recurrence of voiding to 50% for 15 seconds and permanently refills at 205 seconds (Figure 68).
- The behavior of the maximum sheath temperature (Figure 205) during Phase 2 is the direct outcome of the refilling process just explained. Cooling of the sheath continues for a brief interval following the start of ECI as the condensation occurring in RIH2 induces flow through the downstream pass. However, with little coolant being supplied to ROH1 through either the large header interconnect or by continued draining of the steam generator #1 inventory into RIH2, at 61 seconds, the maximum sheath temperature begins to increase.
- The sheath temperature increase continues until refilling of the components between RIH1 and ROH2 is accomplished and flow is re-established through the large header interconnect to ROH1. With the interconnect flow re-established, filling of the components between ROH1 and RIH2, including the downstream pass, proceeds.
- Although refilling of the downstream pass starts at 105 seconds, sufficient coolant to terminate the sheath temperature increase is not present in the downstream pass channel until approximately 130 seconds. The downstream pass channel integrated void at 130 seconds is 0.75.
- There is a brief overlap between the termination of ECI flow and the start of long term cooling (LTC) injection. ECI flow continues to 256 seconds; LTC injection begins at 250 seconds, which is defined as the start of Phase 3.

4.3 Long-Term Cooling (Phase 3: > 250 seconds)

Phase 3 is initiated by injection of coolant flow in the headers by the LTC system. The LTC system draws its coolant supply from the reactor building sump and delivers coolant to RIH1 and RIH2. There is a 75 second interval between the termination of ECI flow and achieving full LTC flow, causing the sheath temperature to increase. After full LTC flow is established, the maximum single channel sheath temperature decreases and the core undergoes an initial core quench. The sheath temperatures remain near the coolant saturation temperature thereafter.

A detailed description of the processes and phenomena occurring during Phase 3 follows.

- There is an interval of approximately 75 seconds between the termination of ECI flow at 256 seconds and establishing full LTC flow at about 330 seconds.
- During this interval, the sheath temperature increases by approximately 50ºC (Figure 206).
- At full flow, the LTC delivers approximately 250 kg/s coolant to each inlet header (Figure 203). The flow delivered to RIH1 first passes through the upstream pass channel, through the large header interconnect, and then through the downstream pass channel and into RIH2 (Figure 57) and thence out the break.
- The total break flow of 500 kg/s (Figure 14 and Figure 15) is equal to the LTC flow of the same amount (Figure 203). The 250 kg/s delivered to RIH2 passes directly out the break while the 250 kg/s delivered to RIH1 passes through both the upstream and downstream pass channels before going out the break.

- At 390 seconds, approximately 60 seconds after full LTC flow is established, the sheath temperature peaks and begins to decrease.
- Core cooling at full LTC flow continues until the end of the calculated transient. The LTC flow results in an initial core quench at 740 seconds.

5. PIRT RESULTS

As indicated in the acknowledgements, this Phenomena Identification and Ranking Table (PIRT) effort involved a number of domain experts. For each phase of the large break LOCA (LBLOCA), a hierarchy of importance-ranking evaluations was developed by consensus[4]. The importance-rank was assigned on the basis of the potential impact of associated, credible phenomena on the Figure of Merit (FOM). The top-down assessment logically proceeds from system, to components, and finally processes/phenomena. A bottom-up re-evaluation follows to ensure consistency for the assigned importance-ranks.

This section summarizes the results of the ACR-700 LBLOCA PIRT for a 25% RIH break with subsequent loss of Class IV power. The reactor is assumed to be operating at full power with equilibrium poison levels. Design values are assumed for plant parameters. All components are assumed to be available and to function as per design.

The following summarizes the four importance-ranks used in this exercise.

High (**H**)	The phenomenon has a controlling impact on the primary Figure of Merit
Medium (**M**)	The phenomenon has a moderate impact on the primary Figure of Merit
Low (**L**)	The phenomenon has a minimal impact on the primary Figure of Merit
Inactive (**I**)	The phenomenon has no impact on or is insignificant with respect to the primary Figure of Merit.

The highest potential importance-rank assigned to a component during a particular phase of a LBLOCA is limited to the highest importance-rank of the associated system during that same phase. A similar logic is applied to processes/phenomena with respect to their associated component. A definition of the PIRT term for a process/phenomenon[5] is provided in Appendix A.

It is understood that a phenomenon ranked High (**H**), Medium (**M**) or Low (**L**) must, at a minimum, be modeled within the Evaluation Model (Reference [6]). The accuracy required of a given analytic model is directly related to the importance rank assigned to the phenomenon being modeled.

A detailed scenario description is provided in Section 4 of this report. Much of the description is based upon a code-calculated sequence (e.g., Table 5). Total reliance on the code-calculated results is not prudent. However, a number of additional sources of information were available and used to inform this PIRT effort. These include the outcomes from applicable experiments in scaled integral, component, and separate effect facilities as well as insights from operation of a considerable fleet of CANDU reactors.

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4 Achieving consensus had at least two advantages: 1) ensuring side-effects/phenomena interactions were properly noted; and 2) a shared appreciation of overall system response.

5 It has proven difficult to clearly differentiate between processes and phenomena. The word "phenomenon", as used in this report, is intended to include phenomena, processes and, in some cases, key output parameters that would be used in Evaluation Model validation (Reference 7).

The PIRT exercise is only a portion of an iterative process to ensure that the Evaluation Model is applicable to the accident scenario in question. Deficiencies identified during the PIRT process would lead to Change Control as required by software Quality Assurance (References [13] and [14]). Significant changes to the analysis code(s) would require a re-assessment of the PIRT.

5.1 ACR-700 LBLOCA PIRT

The 25% RIH break scenario with subsequent loss of Class IV power is described in Section 4 of this report. This scenario represents a critical header break with respect to channel flow stagnation. Table 6 provides a PIRT Summary recording the PIRT Panel's importance-rank evaluations. Note that the term, Availability, indicating the importance of the component functioning as designed, is used only for completeness and is entered in the "phenomenon" column of the table.

5.1.1 Blowdown Period

The early blowdown period covers the period from break initiation until the start of emergency core coolant injection into the headers (i.e., when the ECI rupture disc bursts at 53 seconds). During this period, the following systems have a controlling impact (***High***) on the primary Evaluation Criterion:

- Heat Transport System (HTS)
- Fuel Channel
- Fuel Bundle
- Shutdown Systems (SDS)
- Secondary and Feedwater System

5.1.1.1 Heat Transport System (HTS)

The Heat Transport System (HTS), as defined within this document, consists of the **break**[6], **inlet headers**, **outlet headers**, **feeders**, permanent header interconnect, SG inlet piping/plenum, SG – Primary Side, **pump suction piping/SG outlet plenum**, **HTS pump** and pump discharge piping.

It is assumed that all components between the break and the pressure tube have the same importance, as all intermediate components are required to provide the connecting pathway for the flow (i.e., a series argument). The HTS is required for fuel cooling during this phase and must maintain the integrity of coolant flow to the fuel channels.

The break represents the large loss mechanism for HTS coolant inventory. Maximum discharge flow (7115 kg/s) occurs quite early (0.1 s). By 10s, the discharge flow has dropped to ~50% of the maximum flow. At the same time, the specific enthalpy is increasing. By the end of this phase, the break discharge flow has fallen to ~1000 kg/s (i.e., 14% of maximum flow).

5.1.1.2 Fuel Channel

The fuel channel, as defined within this document, consists of the **inlet/outlet end fittings**, **pressure tube**, calandria tube and fixed spacers.

[6] Within Section 5.1, components with a high importance-rank are shown in bold font.

The end fittings contain significant stored heat that is given up to the coolant during its flow through the complex flow channels within this component.

Pressure tube strain is the secondary FOM. However, significant deformation of the in-core pressure boundary for an accident of Event Class 2 (Reference [15]) should be a rare occurrence. In this scenario, pressure tube peak temperatures (450°C) occur with 30 seconds of the RIH break. For temperatures below 600°C, negligible pressure tube high-temperature strain occurs.

5.1.1.3 Fuel Bundle

The fuel bundle, as defined within this document, consists of the **fuel elements (general)**[7], outer fuel elements, the center elements containing dysprosium, and the end plates.

Sheath maximum temperature is the primary FOM. The maximum temperature (~1160°C) occurs approximately 7 seconds after the RIH break. The fuel bundle remains the principle source of heat during this phase of the scenario.

5.1.1.4 Shutdown Systems (SDS)

The shutdown systems, as defined in this document, consist of SDS1 (shut off rods) and SDS2 (poison injection). Either SDS is capable of individually shutting down the reactor. Both can insert approximately –50 mk within 2 seconds of the break.

5.1.1.5 Secondary and Feedwater System

The secondary and feedwater system, as defined in this document, consists of the SG – secondary side, **main steam line/header**, turbine stop valve, turbine, steam reheater, condenser, SG feedwater, **main steam safety valves** (MSSVs), main steam isolation valves (MSIVs), atmospheric steam discharge valves (ASDVs) and condenser steam dump valves (CSDVs).

The steam generator (SG) is a heat sink during this phase of the scenario. The signal to crash cool the SGs is also initiated during this phase. The MSSVs/ADSVs operate to keep the SG pressure below setpoint until the MMSVs, located on the main steam line, open fully to crash cool the SGs.

5.1.2 Late Blowdown / ECI / Refill

The late blowdown / ECI / refill period covers the period from the initiation of ECI injection until the refill of the core (i.e., at approximately 200 seconds). During this period, the following systems have a controlling impact (***High***) on the principle Evaluation Criterion:

- Heat Transport System (HTS)
- Fuel Channel
- Fuel Bundle
- Emergency Coolant Injection (ECI)

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[7] Captures the phenomenological aspects of CANLUB, Fuel Pellet, Fuel Sheath, End Cap, Spacer Pad, Filling Gas and CHF Enhancers.

5.1.2.1 Heat Transport System (HTS)

The HTS components consist of the **break**, **inlet headers**, **outlet headers**, **feeders**, permanent header interconnect, SG inlet piping/plenum, SG – primary side, pump suction/piping, HTS pumps, and pump discharge piping.

It is assumed that all components between the break and the pressure tube have the same importance, as all intermediate components are required to provide the connecting pathway for the flow (i.e., a series argument). However, the HTS only refills to the headers and components above the headers are not in the flow path. The indicated high importance-ranked components are required for fuel cooling and must maintain the integrity of coolant flow to the fuel channels.

5.1.2.2 Fuel Channel

The fuel channel components consist of the **inlet/outlet end fittings**, **pressure tube**, calandria tube, and fixed spacers.

Pressure tube strain is the secondary FOM. Pressure tube temperatures remain too low during this phase to accumulate any pressure tube deformation.

5.1.2.3 Fuel Bundle

The fuel bundle components consist of the **fuel element (general)**, outer fuel element, center fuel element, and end plates.

Sheath maximum temperature, which, prior to rewet, initially increases to approximately 480°C during this phase, is the primary FOM. The fuel bundle remains a significant source of decay heat and radiant heat during this phase of the scenario.

5.1.2.4 Emergency Coolant Injection (ECI)

The emergency coolant injection system ECI, as defined in this document, consists of the nitrogen charging system, **injection water storage tank**, **ECI injection valve**, **ECI piping**, **large header interconnect**, **rupture disc** and floating ball seal.

During the previous phase, the ECI LOCA signal is conditioned (~0.5 s), the ECI tank recirculation and nitrogen supply valves close (7 s), the ECI injector valves / ECI header interconnect valve are fully open by 27 s. These actions must all occur for the ECI system to be fully effective. When the rupture disc bursts at 53 s, emergency coolant injection into the headers begins.

5.1.3 Long-Term Cooling System

The long-term cooling period covers the period from LTC injection into a header and continues for the 90-day mission period specified for the long-term cooling (LTC) system. During this period, the following systems have a controlling impact (***High***) on the primary Evaluation Criterion:

- Heat Transport System (HTS)
- Fuel Channel
- Fuel Bundle

- Long-Term Cooling System (LTC)
- Containment

5.1.3.1 Heat Transport System (HTS)

The HTS components consist of the **break**, **inlet headers**, **outlet headers**, **feeders**, permanent header interconnect, SG inlet piping/plenum, SG – primary side, pump suction/piping, HTS pumps, and pump discharge piping.

It is assumed that all components between the break and the pressure tube have the same importance, as all intermediate components are required to provide the connecting pathway for the flow (i.e., a series argument). However, the HTS only refills to the headers and components above the headers are not in the flow path. The high importance-ranked components are required for fuel cooling and must maintain the integrity of coolant flow to the fuel channels.

5.1.3.2 Fuel Channel

The fuel channel components consist of the **inlet/outlet end fittings**, pressure tube, calandria tube and fixed spacers.

Pressure tube strain is the secondary FOM. However, pressure tube temperatures are so low during this phase that negligible pressure tube high-temperature strain occurs.

Coolant flow through the end-fittings, a complex flow path, is important in maintaining adequate cooling to the fuel channels.

5.1.3.3 Fuel Bundle

The fuel bundle components consist of the **fuel element (general)**, outer fuel element, center fuel element with dysprosium, and the end plates.

Sheath maximum temperature is the primary Figure of Merit. During this phase, all fuel channels are refilled. However, the fuel bundle remains a significant source of decay heat during this phase of the scenario.

5.1.3.4 Long-Term Cooling (LTC) System

The long-term cooling system, as defined in this document, consists of the reactor building **sump**, **debris screen**, **recovery pump and piping**, heat exchanger and **LTC valves**.

Although the long-term cooling system is operating in re-circulation mode before this phase of the scenario, long-term cooling water is not supplied to the reactor headers until 250 seconds into the scenario when the ECI water storage tanks become isolated on low level and LTC injecting into the header begins.

5.1.3.5 Containment

The important components of the containment system consist of the **containment structure** (i.e., in particular the steel liner) and the local air coolers as they both provide condensate to the LTC sump. During this phase, the condensate represents a small fraction of the available cooling inventory.

The containment represents the structure that defines and contains the flow paths for the long-term recovery water.

5.2 Knowledge Assessment

Step 8 of the PIRT process described in Section 2.2 addresses the level of knowledge regarding each ranked phenomenon. The importance-rank is related to the sensitivity of the FOM to changes in the phenomenon whereas the knowledge level is related to the uncertainty associated with modeling the phenomenon. Phenomenon found to be highly important relative to the FOM, but for which the knowledge level is low, should be carefully examined to determine if additional experiments or analytic efforts are required.

The Validation Matrix (VM) documents provide further information on the state of knowledge and uncertainties. Validation Matrix documents, specific to the ACR, are currently being developed based on corresponding CANDU-specific validation matrix documents. Each phenomenon synopsis in the VMs provides references to support the information given as well as sources of additional detail. Data sets that can be used to validate each phenomenon are also identified and described.

6. SUMMARY

Atomic Energy of Canada Limited (AECL) has made extensive use of PIRT-like results in the validation of its computer programs used for safety analysis of the CANDU reactor series and now the ACR design. However, the AECL PIRT practices and documentation differ in some important aspects from the PIRT results and documentation with which the NRC is familiar.

The ACR LLOCA PIRT provides a basis for confirming the previously developed CANDU-specific phenomena identification and ranking results for applicability to the ACR.

This document provides the results and a detailed assessment of a US-style PIRT effort for the ACR-700 reactor. The most important systems, components, and processes/phenomena occurring during each phase of a LBLOCA are identified and tabulated. This PIRT effort was undertaken for a 25% RIH LOCA with subsequent loss of Class IV power; a critical header break that results in a period of near-flow-stagnation in the affected channels. The scenario was divided into three phases, the blowdown period beginning at time zero and ending at 53 seconds, the late blowdown / Emergency Coolant Injection (ECI) / refill period between 53 and 250 seconds, and the long-term cooling period for times greater than 250 seconds.

Each phenomenon was assessed relative to a two-fold Figure of Merit. The primary FOM is associated with the temperature response of the fuel sheath (cladding in the US). Specifically, each phenomenon was assessed for its impact on the sheath maximum temperature during each phase of the accident scenario. During the phase in which the sheath maximum temperature is attained (the blowdown period for this accident), each phenomenon was assessed for its impact on the approach of the sheath maximum temperature 1200°C. The secondary FOM is associated with the performance of the pressure tube (PT) during the accident scenario. Specifically, each phenomenon was assessed for its impact on PT high-temperature strain leading to maximum high-temperature plastic strains <2%.

The Technical Basis Document (TBD) for the ACR-700 reactor treats each accident scenario in a generic fashion using several FOMs simultaneously. In contrast, the PIRT activity is addressing phenomena at the component / phase level and uses a single FOM[8]. The agreement of the high importance-ranked phenomena between the TBD and the PIRT activity is excellent.

Table 7 provides a summary of the high importance-ranked phenomena for each component during each phase of the ACR-700 LBLOCA scenario. These PIRT results can be used to support several important decision-making processes in the code validation and qualification process.

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[8] The two-fold FOM used in this PIRT activity involved two highly correlated FOMs.

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[15] X. Changming and N. Chunlei, “Safety Basis for ACR”, Atomic Energy of Canada Limited report 108-03600-AB-003, Rev. 0, 2003 July 22.

Table 1
Generalized PIRT Phenomena List

(Codes in parentheses refer to phenomena codes as per the TBD document [3])

STRUCTURE

Physical Properties (examples)

- Density
- Heat capacity
- Thermal conductivity
- Hardness
- Yield strength

Initial State (examples)

- Thickness
- Surface area
- Orientation
- Temperature
- Stored energy
- Reactivity

Phenomena

- Stored energy release (FC2)(TH6)
- Conduction (TH6)
- Radiant heat transfer (TH11)(RC21)
- Contact heat transfer
- Conductance (gap)
- Embrittlement
- Fragmentation
- Debris generation
- Fuel-coolant interaction (FCI)
- Ablation/erosion
- Post dryout heat transfer (TH12)
- Rewet (TH12)
- Critical heat flux (CHF) (TH9)
- Departure from nucleate boiling (DNB)
- Temperature distribution
- Condensation
- Oxidation (FC9)(FC10)(FC20)
- Reduction (FC10)
- Thermal-chemical reaction (Zircaloy/water) (TH13)
- Jet impingement loading (TH22)
- Deformation (FC6)(FC18)(FC19)(TH19)
- Mechanical interaction (TH19)
- Relocation (FC11)
- Relocated melt surface area
- Failure (FC7)
- Melting (FC11)
- Gas release (FC5)
- Decay heating (FC1)
- Fission heating (FC1)
- Reactivity
  - Fuel temperature change (PH7)
  - Coolant temperature change (PH2)(PH4)
  - Void (PH1)
  - Density / void (PH1)(PH3)
  - Purity (PH6)(PH16)
  - Poison concentration (PH5)
  - Isotopic concentration change (PH8)
  - Lattice distortion (PH15)

VOLUME
• **Physical Properties (examples)**
• Density
• Heat capacity
• Thermal conductivity
• Viscosity
• Chemical reactivity
 	
• **Initial State (example)**
• Phase (liquid, vapour, two-phase)
• Temperature
• Pressure
• Concentration (poison, non-condensable gas)
• Level
• Velocity
 	
• **Phenomena**
• Pressure change
	• Phase separation / stratification (TH3)
	• Level change
	• Conduction (TH6)(MH43)
	• Convection (TH7)(MH44)
	  o Forced to liquid
	  o Forced to vapour
	• Radiant heat transfer (TH11)(MH45)
	• Boiling
	  o Subcooled (TH2)
	  o Nucleate (TH8)(TH2)
	  o Transition (TH2)
	  o Film (TH2)
	• Entrainment
	• De-entrainment
	• Flashing (TH2)(FC23)
	• Void generation from heat transfer
	• Evaporation (TH2)

• Condensation (TH3)
  • Surface
  • Inter-phase
• Interfacial shear
• Swelling (MH42)(TH4)
• Natural circulation (TH17)
• Multi-dimensional flow
• Non-condensable gas effect (TH23)
• Degassing (MH46)
• Cavitation (MH9)
• Oscillation (TH16)
• Turbulence (MH11)
	• Steam binding
	• Criticality
	• Poison
	  • Transport (MH15)
	  • Mixing
	  • Diffusion
	• Deflagration (MH34)
	• Burning
	• Waterhammer
	  • Steam condensation induced (PH11)
	  • Mechanically induced (PH12)

FLOW PATH
Physical properties (examples)
• Resistance (form loss)
• Surface roughness
• Length
• Diameter
• Blockage
• Roughness
Initial State (examples)
• Path/state (open/close)
• Volume
• Phase (liquid, vapour, two-phase)
• Temperature
• Pressure
• Concentration
• Level
• Velocity
	Phenomena
	• Pressure drop
	• Change in path/state (open/close)
	• Flow
	  • Blockage
	  • Bypass
	  • Distribution (multiple channels)
	  • Counter-current (TH15)
	  • Gravity driven (draining)
	  • Gravity driven (natural circulation)
	  • Interrupted (valve action)
	  • Pressure driven
	  • Flow regime at break
	  • Critical (TH1)
	  • Stalled (stagnation)
	  • Reverse
	  • Transport
	  • Debris
	  • Poison
	  • Non-condensable gas

EQUIPMENT
Phenomena
• Break orientation
• Pump coastdown
• Pump performance (single and two-phase) (TH5)
• Power
  • 3D distribution
  • 3D kinetics
  • Local peaking
  • End power peaking
  • Radial distribution
	• Flux distribution (PH14)
	• Power: core physics response to moderator level (PH16)
	• Asymmetries (loop-to-loop)
	• Device-movement induced reactivity (PH11)
	• Class I, II, III or IV power-availability

Table 2
Importance-Ranks and Definitions

Rank	Definition	Application Outcomes
High (H)	Phenomenon has a controlling impact on the Figure of Merit	Simulation of experiments and/or analytic modeling with a high degree of accuracy is critical
Medium (M)	Phenomenon has a moderate impact on the Figure of Merit	Simulation of experiments and/or analytic modeling with a moderate degree of accuracy needed
Low (L)	Phenomenon has a minimal impact on the Figure of Merit	Simple models may needed to preserve functional dependencies
Inactive or Insignificant (I)	Phenomenon has either an insignificant or no impact on the Figure of Merit	Simulation of experiments or analytic modeling is not required

Table 3
Knowledge Levels and Definitions

Rank	Definition
4	Fully known, small uncertainty
3	Known, moderate uncertainty
2	Partially known, large uncertainty
1	Very limited knowledge, uncertainty cannot be characterized

Table 4
Technical Data for Each ACR-700 Unit

Reactor	
Type	Pressure Tube Reactor (PTR)
Thermal Output to Steam Generators [MWth]	1982
Coolant	Pressurized Light Water
Moderator	Heavy Water
Calandria Vessel Diameter [m]	5.2
Fuel Channel	Horizontal, concentric pressure tube/calandria tube with 403 SS end fittings
Number of fuel channels	284
Lattice Pitch [mm]	220
Reflector Thickness [mm]	510
Fuel	
Fuel	Sintered pellets of slightly enriched UO_2 and natural UO_2 in central element; Zircaloy-4 sheaths
Enrichment Level	2.1 wt% ^{235}U in 42 pins, and central pin NU with 7.5 wt% Dysprosium
Fuel Burnup [MWd/te U]	21,000
Fuel Bundle Assembly	43 element CANFLEX
Length of Bundle [mm]	495.3
Outside Diameter (maximum) [mm]	103
Bundle Weight [kg]	22.7 (includes 19.2 kg U)
Bundles per Fuel Channel	12
Bundle Power (maximum time-averaged) [kW]	870
Channel Power (maximum time-averaged) [MW]	7.5
Fuel Channel	
Pressure Tube Outer Radius [mm]	58.2
Pressure Tube Inner Radius [mm]	51.7
Pressure Tube Material	Zircaloy - 2.5 wt% Nb
Calandria Tube Outer Radius [mm]	78.0
Calandria Tube Inner Radius [mm]	75.5
Calandria Tube Material	Zircaloy-2
Heavy Water	
Moderator System [Mg D_2O]	129
Reserve [Mg D_2O]	2
Heat Transport System	
Reactor Outlet Header Pressure [MPa(g)]	11.9
Reactor Outlet Header Temperature [°C]	325
Reactor Inlet Header Pressure [MPa(g)]	13.1
Reactor Inlet Header Temperature [°C]	278.5
Reactor core coolant flow (total) [Mg/s]	6.9
Single Channel Flow (Maximum) [kg/s]	26
Steam Generators	
Number	2
Type	Vertical U-tube with integral preheater
Steam Temperature (Nominal) [°C]	281
Steam Quality	0.999
Steam Pressure [MPa(g)]	6.4

Heat Transport Pumps	
Number	4
Pump Type	Vertical, centrifugal, single suction, double discharge
Motor Type	AC, vertical, squirrel cage induction
Rated Flow [L/s]	2250
Rated Head [m]	230
Motor Rating [MWe]	6.9
Containment	
Type	Pre-stressed concrete with steel liner
Inside Diameter [m]	39.5
Height (top of base slab to inside of dome) [m]	59
Design Pressure [kPa(g)]	250
Turbine Generator	
Steam Turbine Type	Impulse type, tandem compound double exhaust flow, reheat condensing turbine with a last-stage blade length of 132 cm (52 inches).
Steam Turbine Composition	One single flow, high-pressure cylinder, two external moisture separators/reheaters and two double flow, low pressure cylinders.
Net Heat to Turbine [MWth]	1980
Gross/Net Electrical Output* (nominal) [MWe]	731/680*
Gross Turbine Generator Efficiency	36.9%
Steam Temperature at Main Stop Valve [°C]	279
Steam Pressure at Main Stop Valve [MPa(g)]	6.2
Final Feedwater Temperature [°C]	218
Condenser Vacuum [kPa(a)]	4.9*
* Gross electrical output is dependent on cooling water temperature, the turbine generator and condenser design, and the grid frequency.	

Table 5
Event Sequence for 25% RIH Break with Consequent Loss of Class IV

Elapsed Time (s)	Event Description
Early Blowdown (0 – 53 seconds)	
0	• Break occurs • All other systems operating normally
0.1	• Maximum discharge flow occurs (7115 kg/s). Discharge from pump flow (forward direction) and inlet feeder (reverse direction) • Net coolant flow in downstream pass (i.e., RIH2 → ROH1) starts to decrease • Flashing of coolant into containment commences
0.4	• Low RCS flow trip signal (1st trip signal[9]) • Coolant discharge from pump flow (forward) and header / feeders / upstream pass fuel channels (reverse) • Reactor regulating system (RRS) attempts to maintain power
~0.5	• Containment pressure rises and exceeds containment high pressure trip setpoint of 3.45 kPa(g) o ECI signal conditioned o Closure of containment isolation valves initiated o Trip miscellaneous local air coolers (LACs), switch safety-related LACs to low speed o Reactor trip signal
2.0	• Low RCS pressure trip signal (2nd trip signal). • SDS1/SDS2 shutdown initiated. • Mechanical control absorbers (MCAs) drive in • Zone control absorbers frozen • Coolant inventory in upstream pass decreasing • Flow initiated in permanent header interconnect from ROH2 to ROH1 • Water flows from the pressurizer into ROH2; pressurizer pressure dropping; pressurizer level dropping o condenser spray valves close (normal control) o bleed valves close (normal control) o feed valves open (normal control)
3.0	• Reactor power at 36% of full reactor power • Fuel temperatures begin to decrease
3.6	• SDS1 fully inserted (-51 mk) • SDS2 poison spreading in moderator • Fuel heat generation due to decay heat and delayed neutron fission • Average fuel temperature decreasing; radial fuel temperature gradient flattening
5.0	• Reactor power at 11% of full reactor power • Heat transfer to SG decreasing rapidly; SG pressure dropping

9 To meet Regulatory requirements, the first trip signal is generally not credited.

Elapsed Time (s)	Event Description
7.0	• ECI signal actuated on low RCS pressure o ECI valves start to open o ECI recirculation circuit isolation valves and the nitrogen charging system gas supply valves close o ECI large header interconnect valves start to open o signal to initiate SG crash cooldown (with delay) o signal to start Class III diesels (180 s until available). o Reserve Water System (RWS) dump valves start to open o RWS injection valves to RIHs start to open o LTC pumps start (2 x 330 L/s) o feed flow from the P&IC system is terminated. o temperature control valves on LTC heat exchangers open fully • Coolant flow in downstream pass starts to reverse; flow in some channels becomes 2-phase stratified or steam filled • Fuel cladding temperature depends upon local cooling o fuel in downstream pass experiences reduced coolant flow o cladding temperatures rise on fuel with reduced flow o fuel element internal gas pressures rise with increasing element temperature o cladding experiences increased strain • Water draining from pressurizer into RCS
9.1	• Turbine unloading begins • Loss of Class IV power (associated with load switching failures during turbine unloading) • RCS pumps trip and begin to rundown • Pressurizer heater turned off • Local air cooler fans stop, and restart when Class II power is available • RCW pumps, main feed water pumps and RCS feed pumps stop
10	• Break discharge flow dropping rapidly (~30% of peak flow); specific enthalpy increasing • Reactor header pressure ~7 MPa in all headers • Flow rate in downstream pass channels close to zero • Forward flow rate in upstream pass channels ~.50% of peak flow rate • SG still acting as heat sink for RCS • Flow starts through large header interconnect from ROH2 to ROH1 • Flashing at break filling containment with steam; containment pressure at ~200 kPa(a) and rising; steam condensing on “cold” surfaces and starting to flow down walls • Cladding temperatures rising rapidly to 1000°C in downstream pass • PT temperatures at 400°C in downstream pass and rising slowly
20	• Voiding in downstream pass • Cladding temperature remains at 1000°C • PT temperature rises to 450°C

Elapsed Time (s)	Event Description
27	• Reactor power at ~6% • ECI valves fully open • RWS dump valves fully open; LTC sump starts to fill with water (1000 m^3); sump level exceeds LTC pump head requirement before LTC injection valves open • Large header interconnect valves fully open
30	• Cladding temperature decreasing (~800°C) • PT temperature peaks at ~500°C
37	• MSSVs open for crash cooldown • SG secondary side depressurizing • Flow through SG1 decreases to near zero because of voiding; flow through SG2 is ~500 kg/s
53	• Reverse flow continues in downstream pass • Cladding temperature <400°C • Fuel channel rewet; PT temperature <300°C
Late Blowdown / ECI / Refill (53 – 250 seconds)	
53	• ECI rupture disc on RIH2 opens (RIH1 pressure = 4.4 MPa) • Break discharge flow ~1000 kg/s • Cold (41°C upper limit) injection flow from the ECI water storage tank (at least 105 m^3 per ECI tank is available for injection) starts into RIH2 • Break discharge rises briefly with disc rupture as cold water mostly directed to containment through the break
55	• ECI rupture disc on RIH1 opens (RIH1 pressure = 4.4 MPa) • Cold injection flow from the ECI water storage tank starts into RIH1 o water flows through upstream pass; water cannot flow through SG (steam filled), so flow directed to large outlet header interconnect • SGs largely bypassed but act as heat source to limited steam flow
~100	• Flow through SG2 drops to low level (~100 kg/s) • Flow through upstream pass drops briefly to zero and then rises again (always forward) • Flow through downstream pass continues reversed

187	• All Class III powered loads have been sequenced
	• LTC pumps running on Class III power in recirculation mode
	• Local air cooler fans running on Class III power
	  o Condensation flow in containment established
	• Service water available to air coolers and heat exchangers
	• Auxiliary feed water pumps start
	  o cooling to SGs eliminates them as a heat source

Long-Term Cooling (250 – 2000 seconds; end of calculated accident)	
200	• Downstream pass rewet complete (upstream pass always cooled) • Reactor power at ~3.5% • Flow through SG2 drops to zero • Larger header interconnect flow ~200 kg/s from ROH2 to ROH1 o flow experiences transient pulses and oscillations

Elapsed Time (s)	Event Description
222	• ECI flow to RIH2 starts to terminate; ECI injection valves start to close • RIH2 pressure ~1.5 MPa • LTC isolation valves to RIH2 start to open (on high sump level) • Flow through downstream pass experiences large oscillations
236	• ECI flow to RIH1 starts to terminate; ECI injection valves start to close • RIH1 pressure ~1.5 MPa • LTC isolation valves to RIH1 start to open (on low ECI water storage tank level) • No water is flowing from the RWS to headers as header pressure is too high
250	• LTC flow to RIH2 begins
~500	• Pumped flow from RCS water storage tank stops on low tank level • P&IC feed pumps trip • Large header interconnect flow oscillates about a steady-state level of ~250 kg/s
~2000	• System in quasi-steady-state • No major oscillations in large header interconnect flow • Channel void is zero • Flow through SG is zero • Channel flow: upstream pass = 250 kg/s forward; downstream pass = 200 kg/s reverse

Table 6
PIRT Summary for ACR-700 LBLOCA: 25% RIH Break with Consequent Loss of Class IV Power

<table>
<tr><td colspan="3">Figure of Merit (FOM)[10]
Primary: Sheath Temperature (maximum temperature < 1200°C)
Secondary: Pressure Tube (PT) ballooning (maximum high-temperature plastic strain < 2%)</td></tr>
<tr><td>Rank
High (H)</td><td colspan="2">
Phenomenon has a controlling impact on the primary Figure of Merit. Simulation of experiments and analytic modeling with a high degree of accuracy is critical (major/minor trends reasonably within range of data (includes scaling)).</td></tr>
<tr><td>Medium (M)</td><td colspan="2">Phenomenon has a moderate impact on the primary Figure of Merit. Simulation of experiments and/or analytic modeling with a moderate degree of accuracy (major trends generally within range of data) is required.</td></tr>
<tr><td>Low (L)</td><td colspan="2">Phenomenon has a minimal impact on the primary Figure of Merit. Modeling must be present to preserve functional dependencies.</td></tr>
<tr><td>Inactive (I)</td><td colspan="2">Phenomenon has no impact on or is insignificant with respect to the primary Figure of Merit. Modeling must be present only if the functional dependencies are required.</td></tr>
<tr><td>Time Phase
1</td><td>
0 – 53 s</td><td>
Early blowdown cooling (until ECI rupture disc opens)</td></tr>
<tr><td>2</td><td>53 – 250 s</td><td>Late blowdown cooling, EC injection and core refill</td></tr>
<tr><td>3</td><td>>250 s</td><td>Long-term cooling operation (LTC coolant injection into headers)</td></tr>
</table>

[10] The Figures of Merit (FOM) are correlated parameters (~80%). If pressure tube temperatures do not exceed 600°C, the pressure tube high-temperature strain will be negligible and only the primary FOM can be used.

System	Rank by Time Phase			Component	Rank by Time Phase			Process/Phenomenon	Rank by Time Phase		
	1	2	3		1	2	3		1	2	3
Reactor System[11]											
				Shield Tank	L	L	L	Availability	L	L	L
				End Shield[12]	L	L	L	Availability	L	L	L
Moderator System	L	L	L								
				Calandria	L	L	L	Availability	L	L	L
				Moderator Piping / Header	L	L	L	Availability	L	L	L
				Moderator Fluid	L	L	L				
				Volume				Reactivity – moderator purity	L	L	L
								Reactivity – moderator temperature	L	L	L
								Reactivity – moderator poison	L	L	L
				Moderator Pump	L	L	L	Availability	L	L	L
				Moderator Heat Exchanger	L	L	L	Availability	L	L	L
				Moderator Rupture Discs	I	I	I	Availability	I	I	I
				Ion Exchanges (inactive)	I	I	I	Availability	I	I	I
Emergency Cooling Injection (ECI)											
	M	**H**	M	**Nitrogen Charging System**	I	I	I	Availability	I	I	I
				Injection Water Storage Tank	L	**H**	L				
				Structure				Stored energy release	L	L	L
				Volume				Temperature	L	L	L
								Pressure	L	**H**	L
								Level	L	**H**	L
				Flow Path				Flow – pressure driven	L	**H**	L
				ECI Injection Valve	L	**H**	L				
				Flow Path				Change in path/state (open/close)	L	I	L
								Flow – pressure driven	L	**H**	L
								Pressure drop (1-phase, 2-phase)	L	**H**	L

11 The bulk of the reactor systems are covered under subsequent sections. As negligible pressure tube deformation occurs, the lattice geometry is as per design.

12 Includes the lattice tubes.

System	Rank by Time Phase			Component	Rank by Time Phase			Process/Phenomenon	Rank by Time Phase		
	1	2	3		1	2	3		1	2	3
				ECI Piping	L	**H**	L				
				Structure				Stored energy release	L	L	L
				Flow Path				Flow – pressure driven	I	**H**	L
								Pressure drop (1-phase, 2-phase)	I	**H**	L
				Large Header Interconnect	M	**H**	M				
				Structure				Stored energy release	L	L	L
				Volume				Flashing	M	L	L
								Refill	L	**H**	L
				Flow Path				Flow – pressure driven	M	**H**	M
								Pressure drop (1-phase, 2-phase)	M	**H**	M
				Rupture Disc	I	**H**	L				
				Flow Path				Change in path/state (open/close)	I	**H**	I
								Flow – pressure driven	I	**H**	L
								Pressure drop (1-phase, 2-phase)	I	**H**	L
				Floating Ball Seal	I	I	I				
				Volume				Waterhammer	I	I	I
				Flow Path				Change in path/state (open/close)	I	I	I
Long-Term Cooling (LTC)	L	L	**H**								
				Sump	L	L	**H**				
				Structure				Stored energy release	L	L	L
				Volume				Temperature	L	L	L
								Level	L	L	**H**
				Flow Path				Flow – gravity driven (draining)	L	L	**H**
								Flow blockage	L	L	L
								Debris transport	L	L	L
				Debris Screen	L	L	**H**				
				Flow Path				Flow – gravity driven (draining)	L	L	**H**
								Pressure drop (1-phase, 2-phase)	L	L	**H**
								Flow blockage	L	L	L
				Recovery Pump and Piping	L	L	**H**				
				Structure				Stored energy release	L	L	L
				Flow Path				Pump performance / characteristics	L	L	M

System	Rank by Time Phase			Component	Rank by Time Phase			Process/Phenomenon	Rank by Time Phase		
	1	2	3		1	2	3		1	2	3
								Flow – pressure driven	L	L	**H**
								Pressure drop (1-phase, 2-phase)	L	L	**H**
				Heat Exchanger	L	L	L				
				Structure				Stored energy release	L	L	L
								Conduction	L	L	L
				Volume				Forced convection to liquid (primary)	L	L	L
								Forced convection to liquid (secondary)	L	L	L
				Flow Path				Flow – pressure driven	L	L	L
								Pressure drop (1-phase, 2-phase)	L	L	L
				LTC Valves	L	L	**H**				
				Flow path				Change in path/state (open/close)	L	L	**H**
								Pressure drop (1-phase, 2-phase)	L	L	**H**
Reserve Water System (RWS)	L	M	**H**								
				Reserve Water Tank	L	L	**H**				
				Volume				Level[13]	L	L	**H**
				Reserve Water Piping	L	M	L				
				Flow Path				Flow – gravity driven (draining)	L	M	L
								Pressure drop (1-phase, 2-phase)	L	L	L
				Reserve Water Valve	L	M	L				
				Flow Path				Change in path/state (open/close)	L	M	L
								Flow – gravity driven (draining)	L	M	L
								Pressure drop (1-phase, 2-phase)	L	L	L
Electrical Supply	**H**	**H**	**H**								
				Class IV Power Supply	**H**	I	I	Availability	**H**	I	I
				Class III Power Supply	I	**H**	**H**	Availability	I	**H**	**H**
				Class II Power Supply	These components are assumed to be available with a very high level of reliability.						
				Class I Power Supply							

[13] There are several standpipes within the Reserve Water Tank. The first 1000 m^3 are dumped into the reactor building. The remaining inventory is available for delivery to several injection points.

System	Rank by Time Phase			Component	Rank by Time Phase			Process/Phenomenon	Rank by Time Phase		
	1	2	3		1	2	3		1	2	3
Reactor Regulating System (RRS)	L	L	L								
				Zone Control Rod	L	L	L				
								Reactivity – device movement	L	L	L
				Mechanical Control Absorber (MCA)	L	L	L				
								Reactivity – device movement	L	L	L
Heat Transport System (HTS)	**H**	**H**	**H**								
				HTS Circuit	L	L	M				
				Flow path				Flow – natural circulation	L	L	M
				Break	**H**	**H**	**H**				
				Flow Path				Critical flow	**H**	**H**	I
								Flow – pressure driven	I	**H**	**H**
								Break orientation (top, side, bottom or offset)	L	M	L
				Inlet Header	**H**	**H**	**H**				
				Structure				Stored energy release	L	L	L
								Pipe thrust	L	I	I
				Volume				Flashing	M	L	L
								Condensation (inter-phase)	L	M	L
								Waterhammer-condensation induced	L	L	L
								Oscillations	M	M	L
								Flow regime	**H**	**H**	**H**
								Entrainment	M	M	L
								De-entrainment	L	L	L
								Multi-dimensional flow	M	M	L
								Mixing – multiple fluid streams	M	M	L
				Flow Path				Flow – pressure driven	**H**	**H**	**H**
								Flow – gravity driven (draining)	L	L	L
								Distribution (multiple channels)	L	**H**	M
								Pressure drop (1-phase, 2-phase)	L	L	L

System	Rank by Time Phase			Component	Rank by Time Phase			Process/Phenomenon	Rank by Time Phase		
	1	2	3		1	2	3		1	2	3
								Flow reversal	**H**	L	L
				Outlet Header	**H**	**H**	**H**				
				Structure				Stored energy release	L	L	L
				Volume				Flashing	M	L	L
								Condensation (inter-phase)	L	M	L
								Waterhammer-condensation induced	L	L	L
								Oscillations	L	M	L
								Flow regime	M	M	L
								Entrainment	L	L	L
								De-entrainment	L	L	L
								Multi-dimensional flow	M	M	L
								Mixing – multiple fluid streams	M	M	L
				Flow Path				Flow – pressure driven	**H**	**H**	**H**
								Flow – gravity driven (draining)	L	L	L
								Distribution (multiple channels)	L	L	M
								Pressure drop (1-phase, 2-phase)	L	L	L
								Flow reversal	**H**	L	L
				Feeder	**H**	**H**	**H**				
				Structure				Stored energy release	**H**	L	L
				Volume				Flashing	**H**	L	L
								Condensation (inter-phase)	L	L	L
								Waterhammer-condensation induced	L	L	L
								Oscillations	L	M	L
								Flow regime	**H**	**H**	M
				Flow Path				Flow – pressure driven	**H**	**H**	**H**
								Flow – gravity driven (draining)	L	L	L
								Pressure drop (1-phase, 2-phase)	**H**	**H**	**H**
								Flow reversal	**H**	L	L
								Flow – counter current	L	**H**	L
				Permanent Header Interconnect	L	L	L				
				Flow Path				Flow – pressure driven	L	L	L

System	Rank by Time Phase			Component	Rank by Time Phase			Process/Phenomenon	Rank by Time Phase		
	1	2	3		1	2	3		1	2	3
				SG Inlet Piping/Plenum	M	M	L				
				Structure				Stored energy release	L	L	L
				Volume				Flashing	M	L	L
								Reflux condensation	L	L	L
								Flow regime	L	L	L
								Multi-dimensional flow	M	L	L
								Entrainment	L	L	L
								De-entrainment	L	L	L
				Flow path				Flow – pressure driven	M	L	L
								Flow – gravity driven (draining)	L	M	L
								Flow – counter current	L	L	L
								Pressure drop (1-phase, 2-phase)	M	M	L
								Flow reversal	M	L	L
				SG - Primary Side	M	M	M				
				Structure				Stored energy release	L	L	L
								Forced convection to liquid	M	L	L
								Forced convection to vapour	M	L	M
				Volume				Flashing	M	L	L
								Level	L	L	M
								Condensation (inter-phase)	L	L	L
								Reflux condensation	L	L	L
				Flow Path				Flow – pressure driven	M	L	L
								Flow – gravity driven (draining)	L	M	L
								Flow – counter current	L	L	L
								Pressure drop (1-phase, 2-phase)	M	M	L
								Flow reversal	M	M	L
				Pump Suction Piping/SG Outlet Plenum	**H**	M	L				
				Structure				Stored energy release	L	L	L
				Volume				Flashing	**H**	L	L
								Flow regime	L	L	L
								Multi-dimensional flow	L	L	L

System	Rank by Time Phase			Component	Rank by Time Phase			Process/Phenomenon	Rank by Time Phase		
	1	2	3		1	2	3		1	2	3
								Entrainment	L	L	L
								De-entrainment	L	L	L
				Flow Path				Flow – pressure driven	M	L	L
								Flow – gravity driven (draining)	L	M	L
								Flow – counter current	L	L	L
								Pressure drop (1-phase, 2-phase)	M	M	L
								Flow reversal	M	M	L
				HTS Pump	**H**	M	L				
				Structure				Stored energy release	**H**	L	L
				Volume				Flashing	**H**	L	L
								Pump performance / characteristics	**H**	L	L
				Flow Path				Coastdown/rundown	**H**	M	L
								Pressure drop (1-phase, 2-phase)	L	M	L
				Pump Discharge Piping	M	M	L				
				Structure				Stored energy release	L	L	L
				Volume				Flashing	M	L	L
								Reflux condensation	L	L	L
								Flow regime	L	L	L
								Entrainment	L	L	L
								De-entrainment	L	L	L
				Flow path				Flow – pressure driven	M	L	L
								Flow – gravity driven (draining)	L	L	L
								Flow – counter current	L	L	L
								Pressure drop (1-phase, 2-phase)	M	M	L
								Flow reversal	M	M	L
Fuel Channel	**H**	**H**	**H**								
				End Fittings (inlet & outlet)	**H**	**H**	**H**				
				Structure				Stored Energy Release	L	L	L
				Volume				Flashing	**H**	L	L
								Flow regime	M	M	L
								Condensation (inter-phase)	L	L	L
								Entrainment	M	M	L

System	Rank by Time Phase			Component	Rank by Time Phase			Process/Phenomenon	Rank by Time Phase		
	1	2	3		1	2	3		1	2	3
								De-entrainment	L	L	L
				Flow Path				Flow – pressure driven	**H**	**H**	**H**
								Flow – counter current	L	**H**	L
								Flow – gravity driven (draining)	L	L	L
								Pressure drop (1-phase, 2-phase)	**H**	**H**	**H**
								Flow reversal	**H**	L	L
				Pressure Tube	**H**	M	L				
				Structure				Stored energy release	M	M	L
								Convective heat transfer	**H**	M	L
								Conduction	M	L	L
								Radiant heat transfer	**H**	M	I
								Solid-to-Solid (fuel element to pressure tube) contact heat transfer	L	L	I
								Deformation (includes failure)	**H**	**H**	L
				Calandria Tube	M	L	L				
				Structure				Stored energy release	I	I	I
								Convective heat transfer	M	L	L
								Conduction	I	I	I
								Radiant heat transfer	M	L	L
				Fixed Spacer (garter spring)	L	L	L				
				Structure				Conduction	L	L	L
Fuel Bundle[14]	**H**	**H**	**H**								
				Fuel Element (general)	**H**	**H**	**H**				
				Structure				Stored energy release	**H**	M	L
								Conduction	**H**	M	M
								Radiant heat transfer	**H**	L	L
								Gap conductance	**H**	M	M
								Forced convection to liquid	M	**H**	**H**
								Boiling - nucleate	M	**H**	**H**

14 Will capture the phenomenological aspects of CANLUB, Fuel Pellet, Fuel Sheath, End Cap, Bearing Pad, Spacer Pad, Filling Gas, CHF Enhancers

System	Rank by Time Phase			Component	Rank by Time Phase			Process/Phenomenon	Rank by Time Phase		
	1	2	3		1	2	3		1	2	3
								Boiling - film	**H**	**H**	L
								Rewet	L	M	L
								Critical Heat Flux (CHF)	**H**	I	I
								Oxidation	M	L	L
								Embrittlement	L	L	L
								Reactivity - fuel string relocation	L	L	L
								Deformation – fuel string relocation	L	L	L
								Deformation	**H**	M	M
								Constrained axial expansion	I	I	I
								Failure	M	L	L
								Melting	L	I	I
								Fragmentation	L	L	L
								Decay heating	L	**H**	**H**
								Fission heating	**H**	M	L
								Reactivity - fuel temperature change	M	L	L
								Reactivity - coolant temperature change	L	L	L
								Swelling – fission product	L	L	L
								Reactivity - density/void	M	L	L
								Radial power distribution	M	L	L
								End power peaking	**H**	L	L
				Volume				Flow regime	**H**	**H**	L
								Void generation from heat transfer	**H**	**H**	L
								Entrainment	**H**	**H**	L
								De-entrainment	**H**	**H**	L
								Flashing	**H**	L	L
								Level - swelling	**H**	**H**	L
								Multi-dimensional flow	**H**	**H**	M
				Flow Path				Flow – pressure driven	**H**	**H**	**H**
								Pressure drop (1-phase, 2-phase)	**H**	**H**	**H**
								Flow reversal	**H**	**H**	**H**
								Flow - stalled (stagnation)	**H**	L	L

System	Rank by Time Phase			Component	Rank by Time Phase			Process/Phenomenon	Rank by Time Phase		
	1	2	3		1	2	3		1	2	3
				Outer Fuel Element	M	L	L				
								Contact heat transfer – bearing pad-to-pressure tube	M	L	L
				Center Element (dysprosium)	M	L	L				
								Reactivity – isotopic change	M	L	L
				End Plate	L	L	L				
				Structure				Deformation	L	L	L
								Oxidation	L	L	L
Reactor Protection System	I	I	I	**Flux Detectors**	I	I	I				
				Structure				Flux detector response	I	I	I
Shutdown Systems (SDS)	**H**	L	L								
				SDS1 (Shut Off Rods)	**H**	L	L	Reactivity effect of firing SDS1	**H**	L	L
				SDS2 (Poison Injection)	**H**	L	L	Reactivity effect of firing SDS2	**H**	L	L
Pressure and Inventory Control System (P&IC)	M	L	L								
				Liquid Relief Valve (LRV)	I	I	I				
				Flow path				Change in path/state (open/close)	I	I	I
								Flow – pressure driven	I	I	I
								Pressure drop (1-phase, 2-phase)	I	I	I
				Pressurizer	M	L	L				
				Structure				Stored energy release	L	I	I
				Volume				Flashing	M	L	L
								Level	M	L	L
								Level - swelling	L	L	L
								Non-condensable gas generation	L	L	L
				Flow path				Flow – pressure driven	M	L	L
								Pressure drop (1-phase, 2-phase)	L	L	L
				Feed and Bleed System	M	L	L				
								Flow – pressure driven	M	L	L
Secondary and Feedwater System	**H**	M	L								
				SG - Secondary Side	M	M	L				

System	Rank by Time Phase			Component	Rank by Time Phase			Process/Phenomenon	Rank by Time Phase		
	1	2	3		1	2	3		1	2	3
				Structure				Forced convection to liquid	L	L	L
								Forced convection to vapour	M	L	L
				Volume				Level	M	L	L
								Level - swelling	M	M	L
								Boiling - nucleate	M	L	L
								Oscillations	L	L	L
				Main Steam Line/Header	**H**	L	L				
				Flow Path				Flow – pressure driven	**H**	L	L
				Turbine Stop Valve	L	I	I				
				Flow Path				Change in path/state (open/close)	L	I	I
				Turbine	L	I	I	Availability	L	I	I
				Steam Reheater	L	I	I	Availability	L	I	I
				Condenser	L	I	I	Availability	L	I	I
				SG Feedwater	M	L	L				
				Flow path				Flow – pressure driven	M	L	L
				Main Steam Safety Valve (MSSV)	**H**	I	I				
				Flow path				Change in path sate (open/close)	**H**	I	I
				Main Steam Isolation Valve (MSIV)	I	I	I				
				Flow path				Change in path sate (open/close)	I	I	I
				Atmospheric Steam Discharge Valve (ASDV)	M	I	I				
				Flow path				Change in path sate (open/close)	M	I	I
				Condenser Steam Dump Valve (CSDV)	I	I	I				
				Flow path				Change in path sate (open/close)	I	I	I
Containment System	L	L	**H**								
				Containment	L	L	**H**				
				Structure				Stored energy release	I	I	I
								Condensation – fluid-to-surface	L	L	L
				Volume				Pressure	L	L	L

System	Rank by Time Phase			Component	Rank by Time Phase			Process/Phenomenon	Rank by Time Phase		
	1	2	3		1	2	3		1	2	3
								Condensation (inter-phase)	L	L	L
								Temperature	L	L	L
								Jet impingement (debris generation)	L	I	I
				Flow Path				Flow – gravity driven (draining)	L	L	**H**
								Debris transport	L	L	L
				Local Air Cooler	L	L	M				
				Structure				Condensation – fluid-to-surface	L	L	M

Table 7
Summary of High Importance-Ranked Systems, Components and Phenomena

System	Component	Process / Phenomenon
Early Blowdown Phase (0 s – 53 s)		
Heat Transport System (HTS)	Break	Critical flow
	Inlet Header	Flow regime Flow – pressure driven Flow reversal
	Outlet Header	Flow – pressure driven Flow reversal
	Feeder	Stored energy release Flashing Flow regime Flow – pressure driven Pressure drop (1-phase, 2-phase) Flow reversal
	Pump Suction Piping / SG Outlet Plenum	Flashing
	HTS Pump	Stored energy release Flashing Pump performance / characteristics Coastdown / rundown
Fuel Channel	End Fittings (inlet & outlet)	Flashing Flow – pressure driven Pressure drop (1-phase, 2-phase) Flow reversal
	Pressure Tube	Convective heat transfer Radiant heat transfer Deformation (includes failure)
Fuel Bundle	Fuel Element (general)	Stored energy release Conduction Radiant heat transfer Gap conductance Boiling – film Critical Heat Flux (CHF) Deformation

System	Component	Process / Phenomenon
		Fission heating End power peaking Flow regime Void generation from heat transfer Entrainment De-entrainment Flashing Level – swelling Multi-dimensional flow Flow – pressure driven Pressure drop (1-phase, 2-phase) Flow reversal Flow – stalled (stagnation)
Shutdown Systems (SDS)	SDS1 (shut off rods)	Reactivity effect of firing SDS1
	SDS2 (poison injection)	Reactivity effect of firing SDS2
Secondary and Feedwater System	Main Steam Line / Header	Flow – pressure driven
	Main Steam Safety Valve (MSSV)	Change in path/state (open/close)
Late Blowdown / ECI / Refill (53 s – 250 s)		
Emergency Cooling Injection (ECI)	Injection Water Storage Tank	Pressure Level Flow – pressure driven
	ECI Injection Valve	Flow – pressure driven Pressure drop (1-phase, 2-phase)
	ECI Piping	Flow – pressure driven Pressure drop (1-phase, 2-phase)
	Large Header Interconnect	Refill Flow – pressure driven Pressure drop (1-phase, 2-phase)
	Rupture Disc	Change in path/state (open/close) Flow – pressure driven Pressure drop (1-phase, 2-phase)

System	Component	Process / Phenomenon
Heat Transport System (HTS)	Break	Critical flow Flow – pressure driven
	Inlet Header	Flow regime Flow – pressure driven Distribution (multiple channels)
	Outlet Header	Flow – pressure driven
	Feeder	Flow regime Flow – pressure driven Pressure drop (1-phase, 2-phase) Flow – counter current
Fuel Channel	End Fittings (inlet & outlet)	Flow – pressure driven Pressure drop (1-phase, 2-phase) Flow – counter current
	Pressure Tube	Deformation (includes failure)
Fuel Bundle	Fuel Element (general)	Forced convection to liquid Boiling - nucleate Boiling – film Decay heating Flow regime Void generation from heat transfer Entrainment De-entrainment Level – swelling Multi-dimensional flow Flow – pressure driven Pressure drop (1-phase, 2-phase) Flow – reversal
Long-Term Cooling Phase (> 250 s)		
Long-Term Cooling (LTC)	Sump	Level Flow – gravity driven (draining)
	Debris Screen	Flow – gravity driven (draining) Pressure drop (1-phase, 2-phase)

System	Component	Process / Phenomenon
	Recovery Pump and Piping	Flow – pressure driven Pressure drop (1-phase, 2-phase)
	LTC Valves	Change in path/state (open/close) Pressure drop (1-phase, 2-phase)
Reserve Water System (RWS)	Reserve Water Tank	Level
Heat Transport System (HTS)	Break	Flow – pressure driven
	Inlet Header	Flow regime Flow – pressure driven
	Outlet Header	Flow – gravity driven (draining)
	Feeder	Flow – gravity driven (draining) Pressure drop (1-phase, 2-phase)
Fuel Channel	End Fittings (inlet & outlet)	Flow – pressure driven Pressure drop (1-phase, 2-phase)
Fuel Bundle	Fuel Element (general)	Forced convection to liquid Boiling – nucleate Decay heating Flow – pressure driven Pressure drop (1-phase, 2-phase) Flow reversal
Containment System	Containment	Flow – gravity driven (draining)

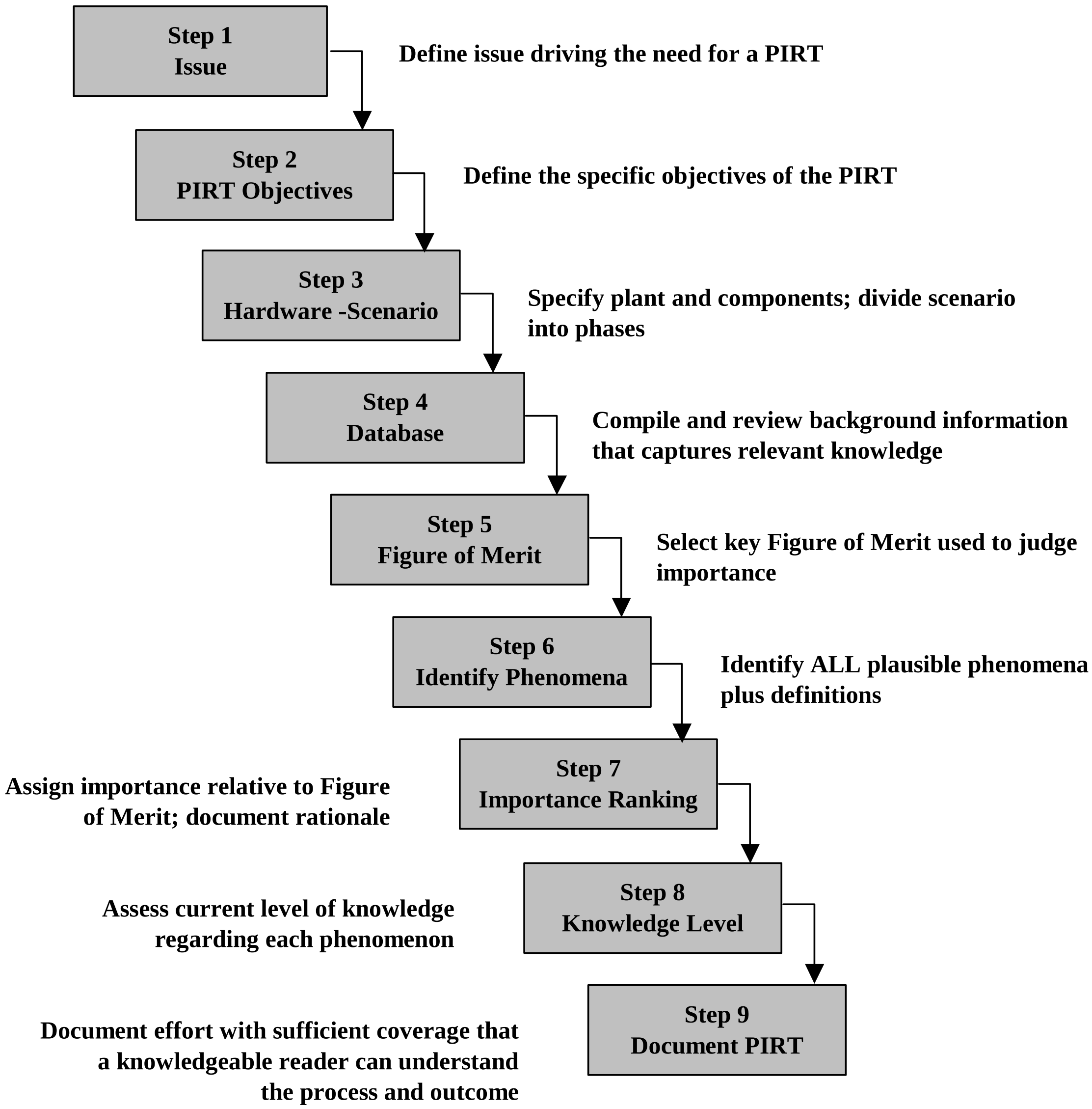


Figure 1 General PIRT Process

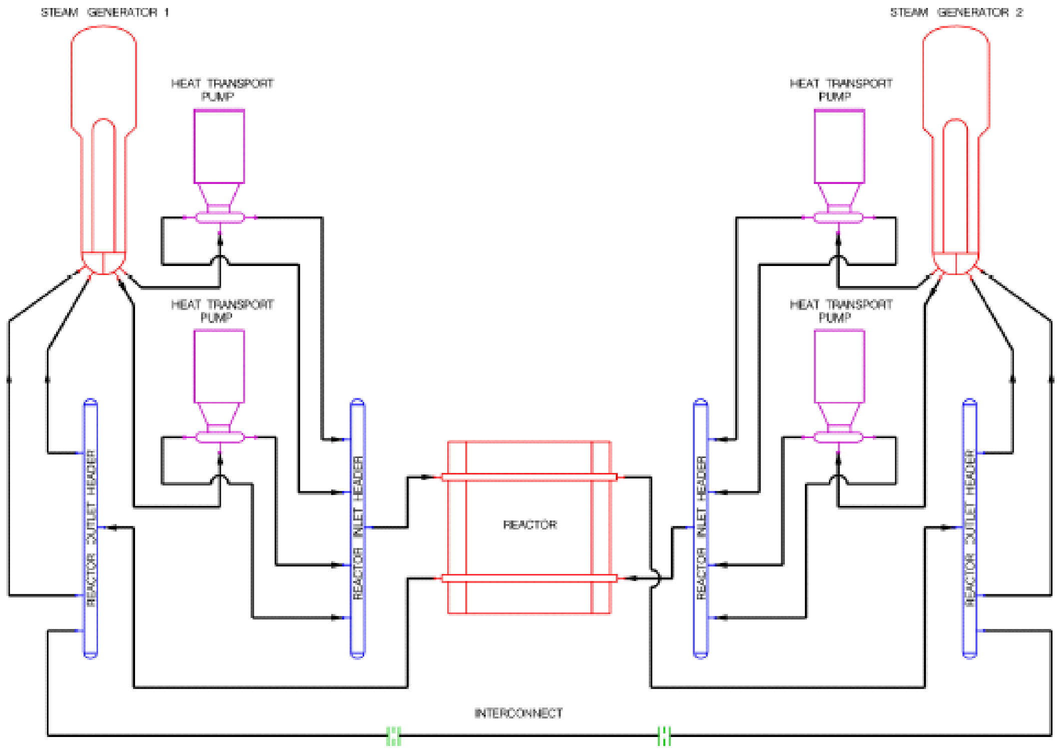
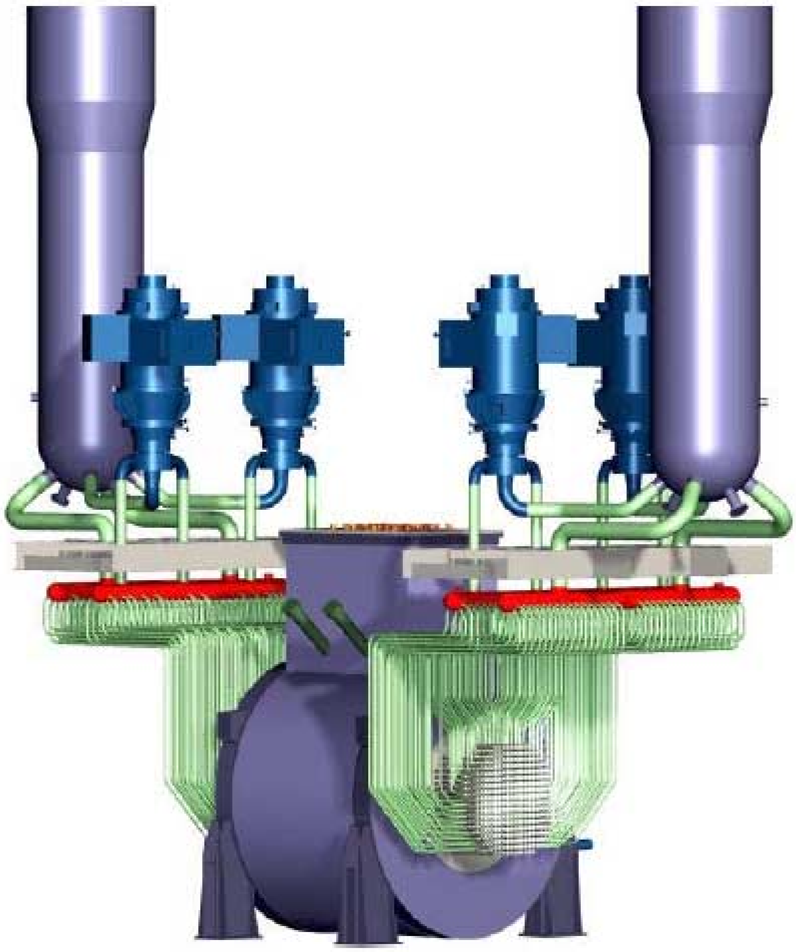


Figure 2 ACR-700 Simplified Heat Transport Circuit Schematic



ACR-700 Heat Transport System Layout

Figure 3 ACR-700 Heat Transport System Layout

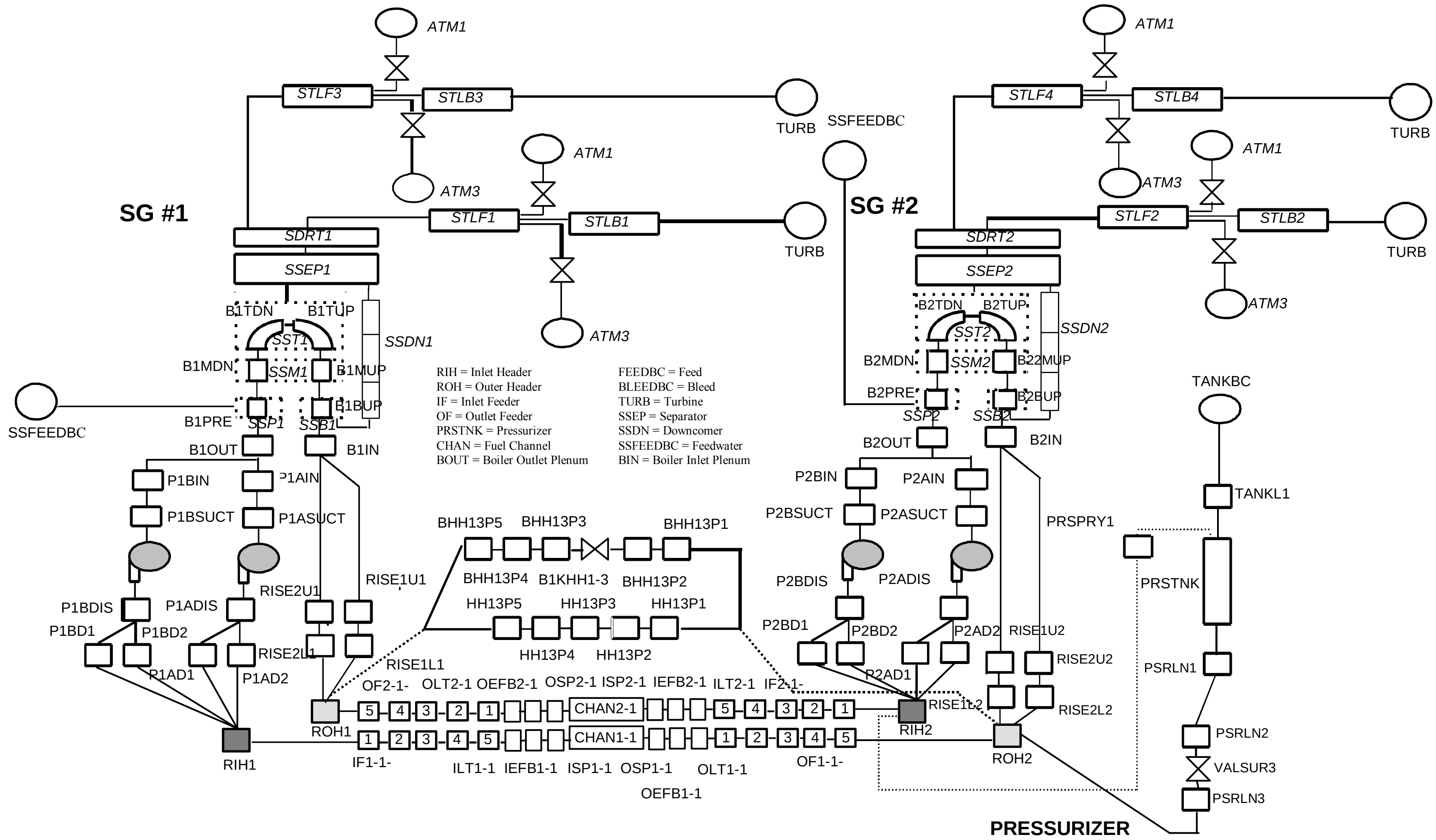


Figure 4 Nodalization Diagram of Primary Heat Transport System

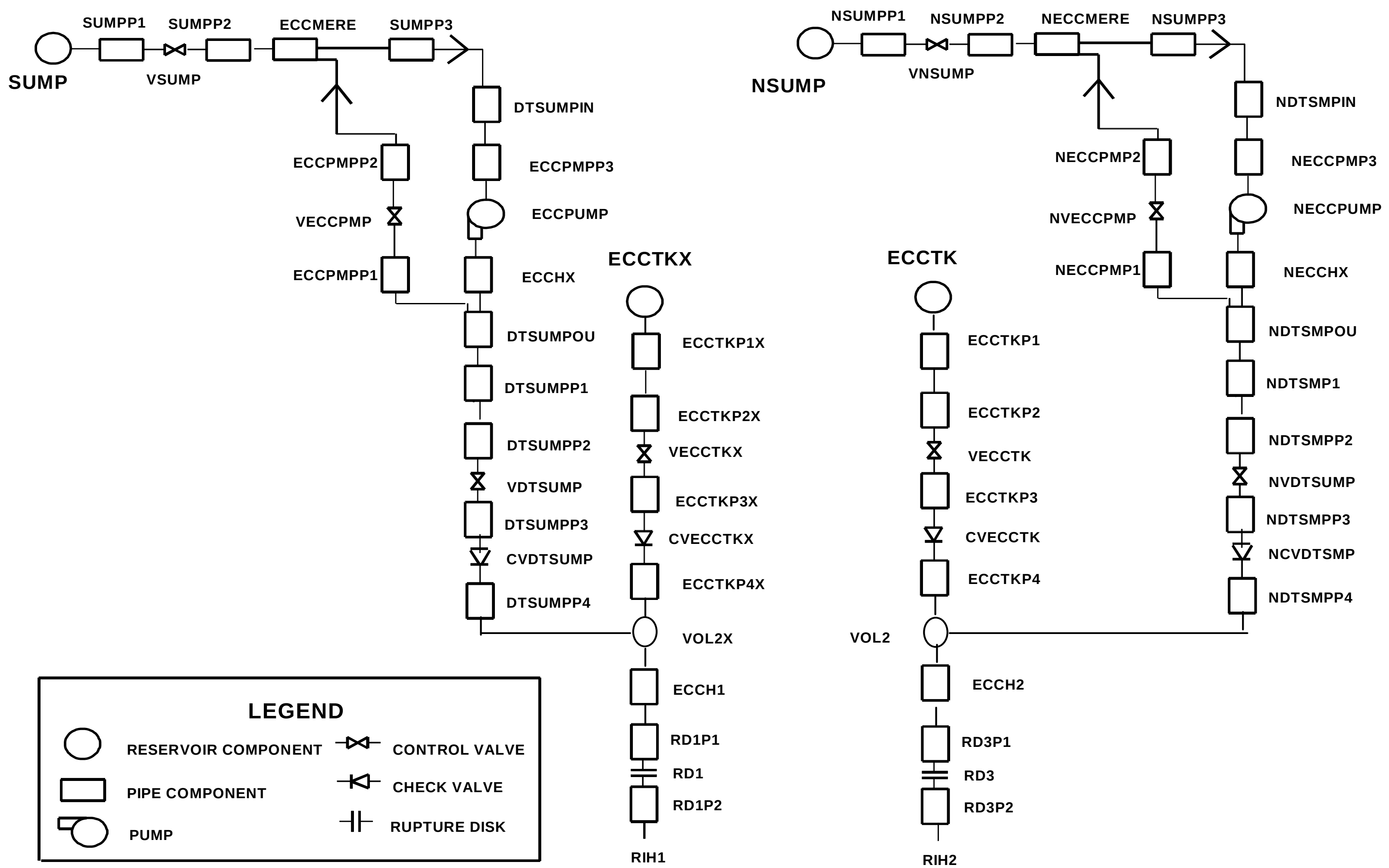


Figure 5 Nodalization Diagram of ECC System

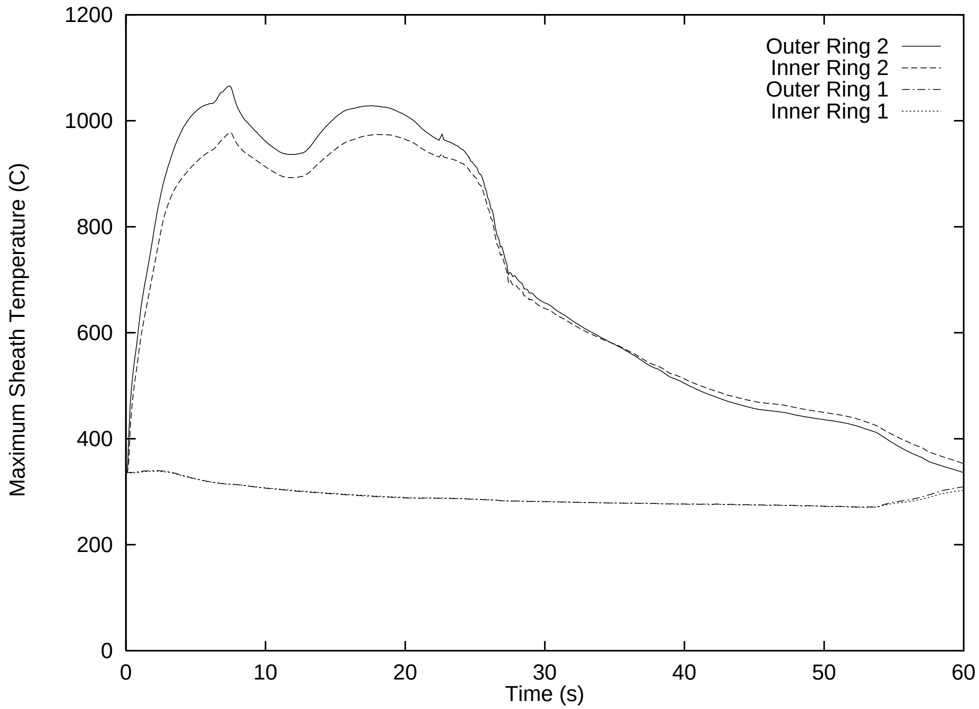


Figure 6 Maximum Sheath Temperature along Channel for 25% RIH Break with Subsequent Loss of Class IV Power

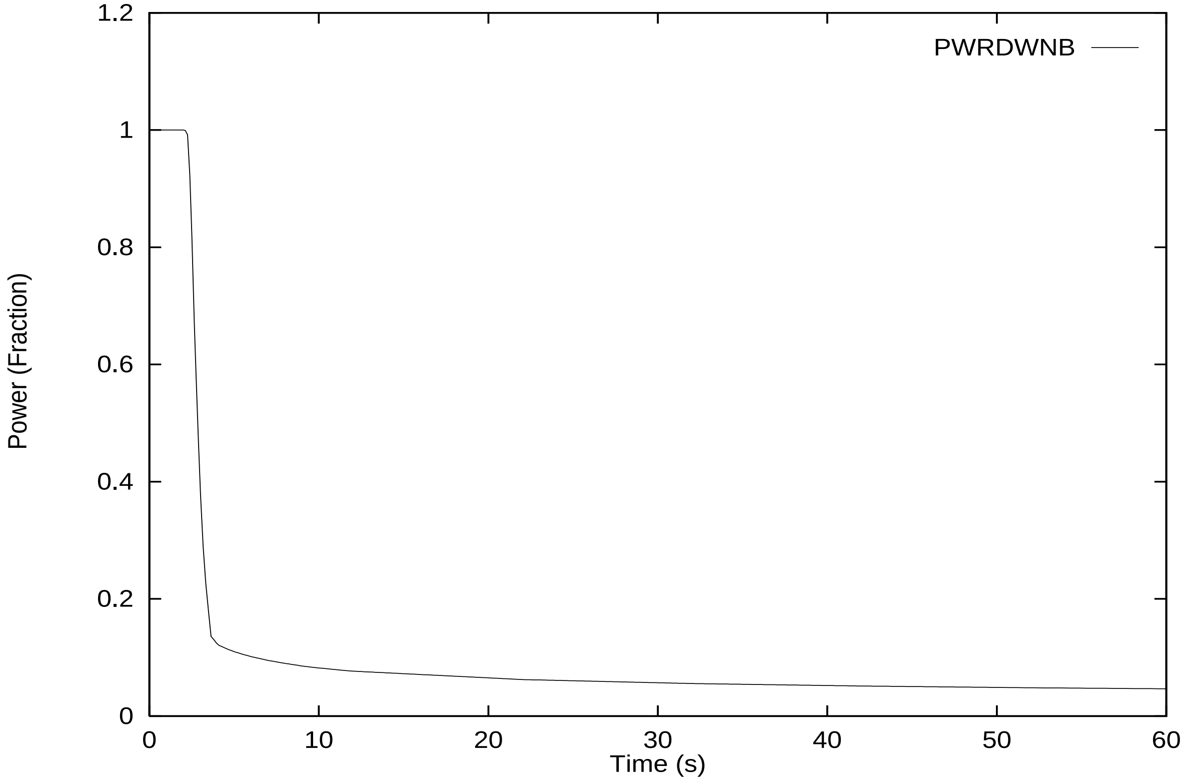


Figure 7 Reactor Power for 25% RIH Break with Subsequent Loss of Class IV Power

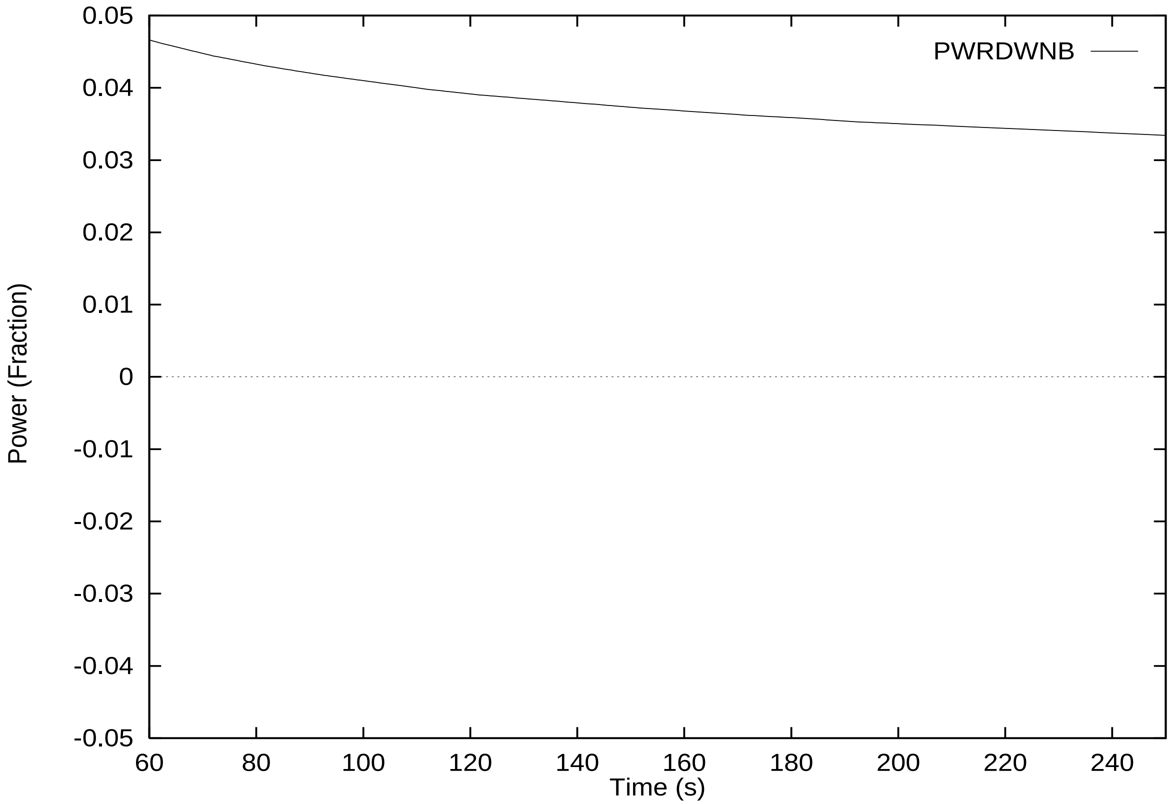


Figure 8 Reactor Power for 25% RIH Break with Subsequent Loss of Class IV Power

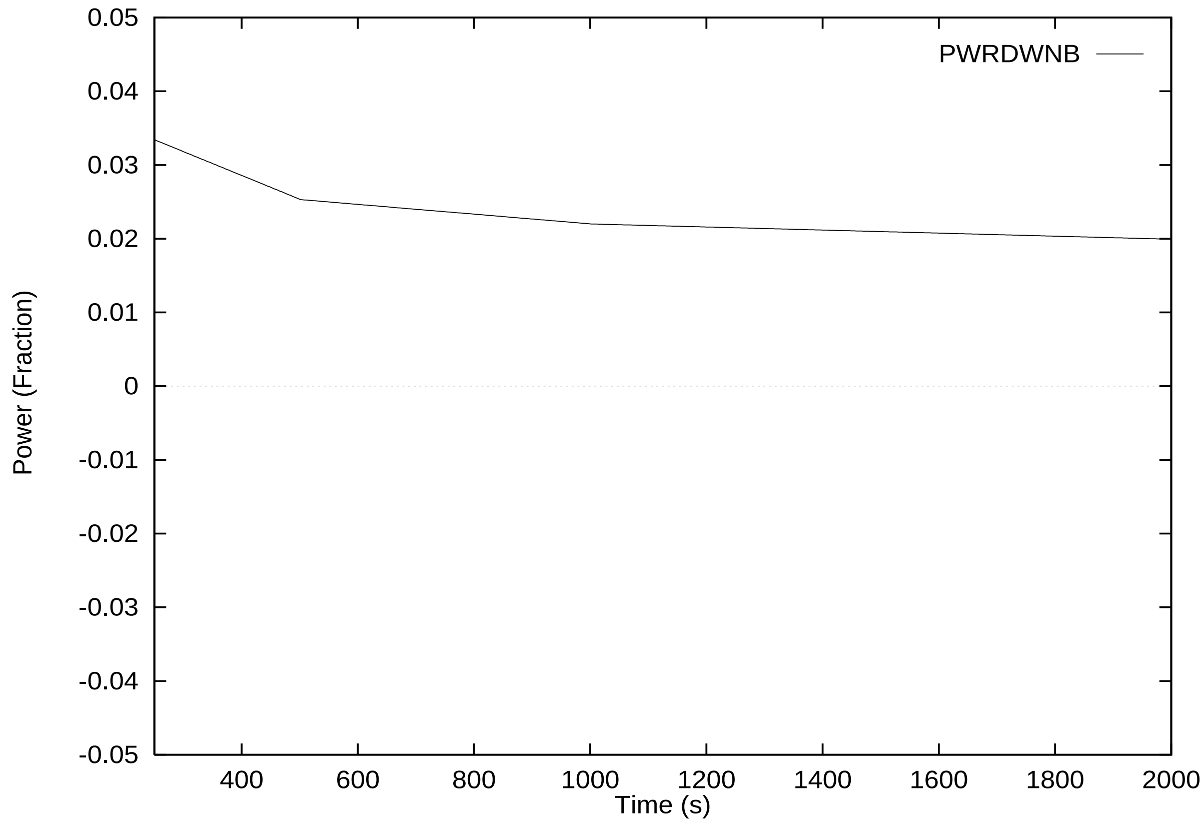


Figure 9 Reactor Power for 25% RIH Break with Subsequent Loss of Class IV Power

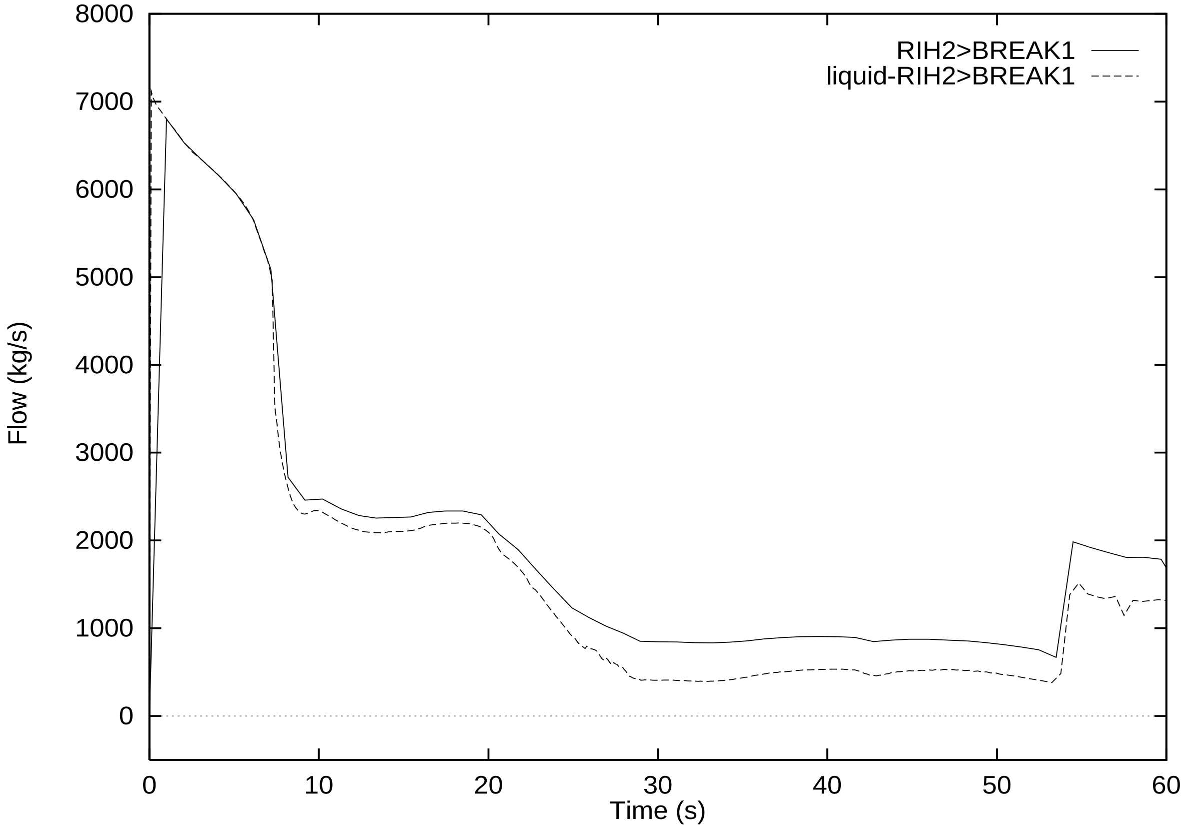


Figure 10 Break Discharge Flow for 25% RIH Break with Subsequent Loss of Class IV Power

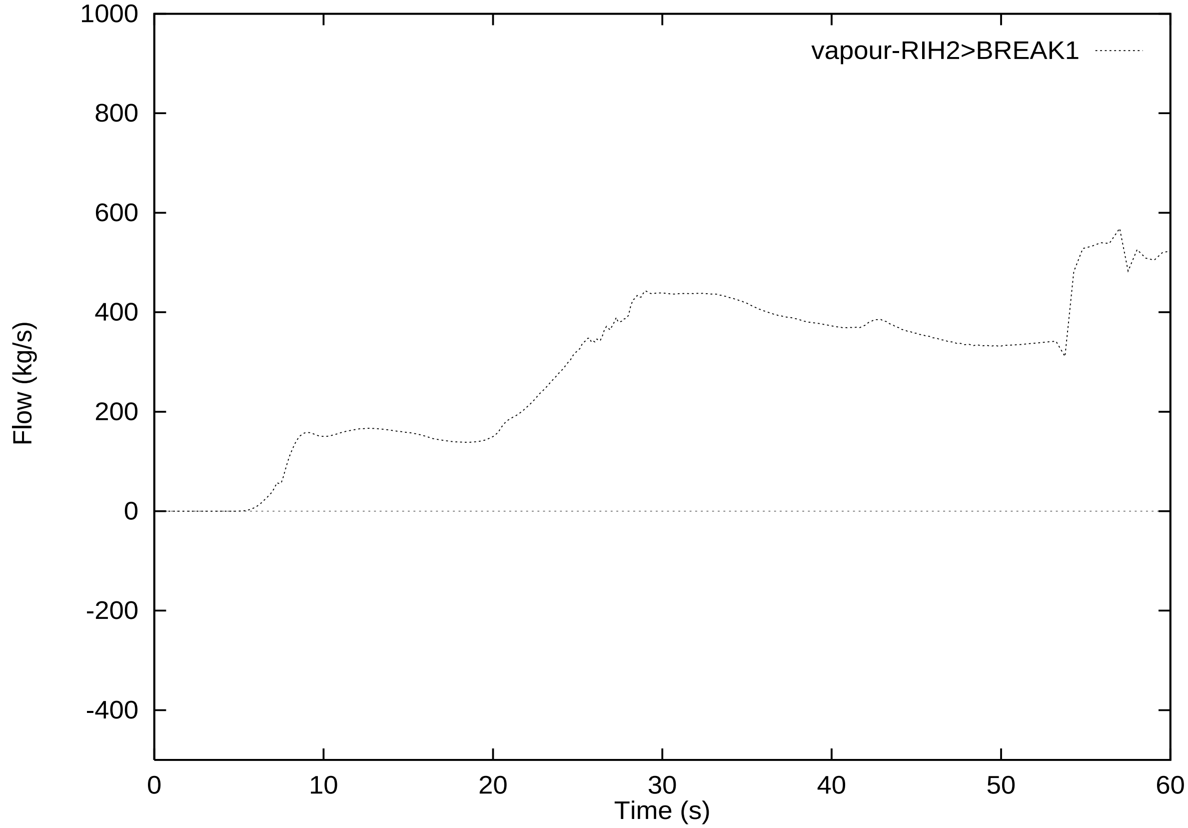


Figure 11 Break Discharge Flow for 25% RIH Break with Subsequent Loss of Class IV Power

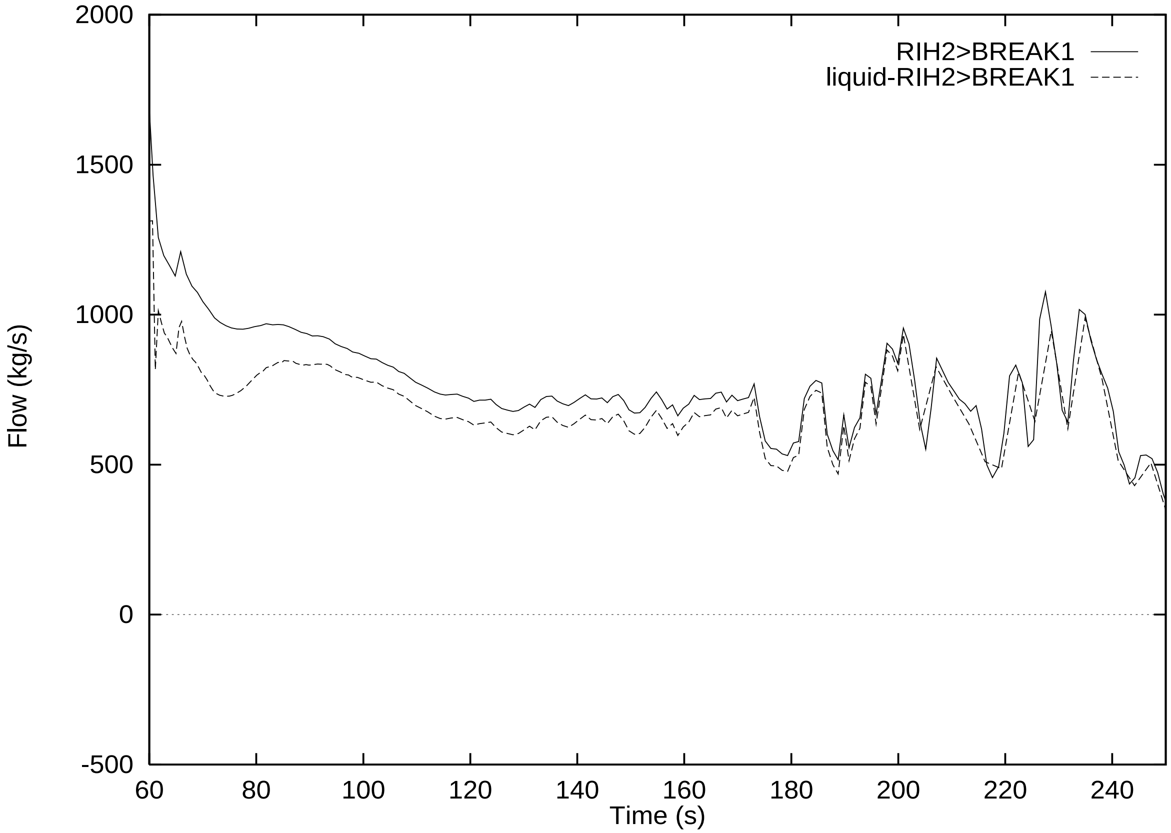


Figure 12 Break Discharge Flow for 25% RIH Break with Subsequent Loss of Class IV Power

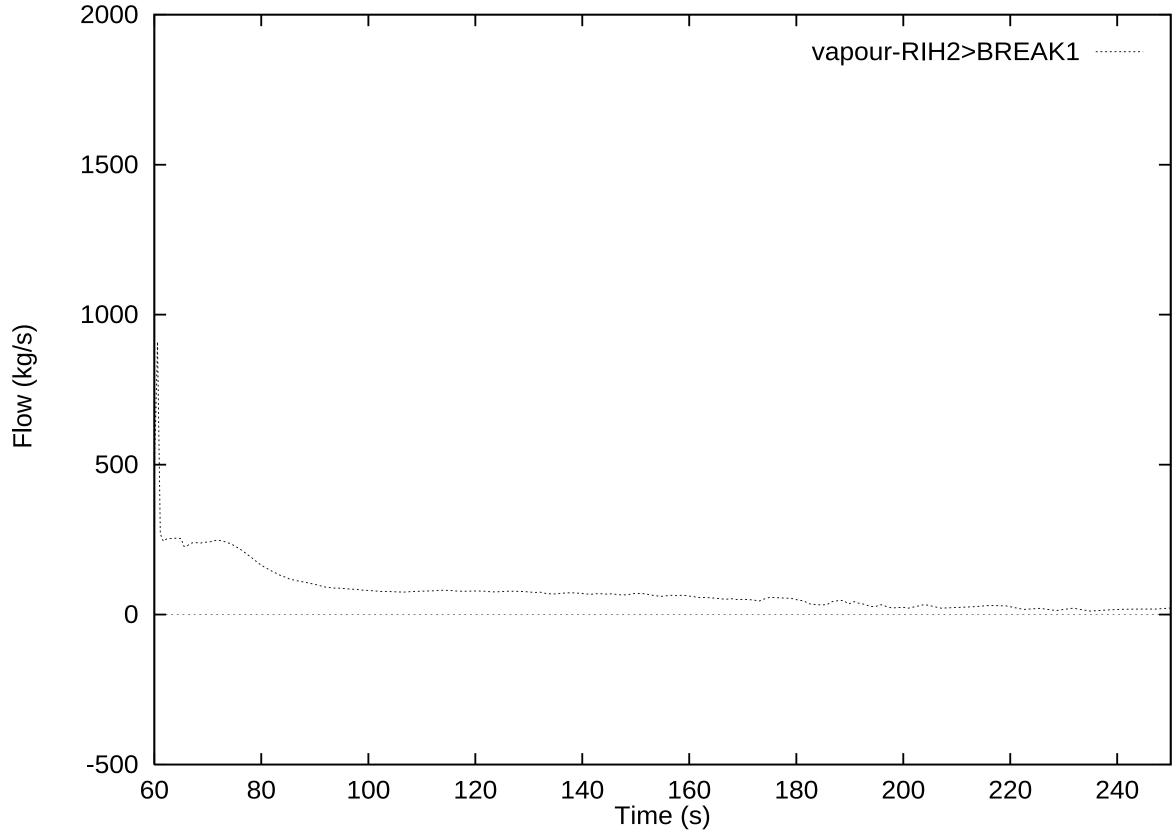


Figure 13 Break Discharge Flow for 25% RIH Break with Subsequent Loss of Class IV Power

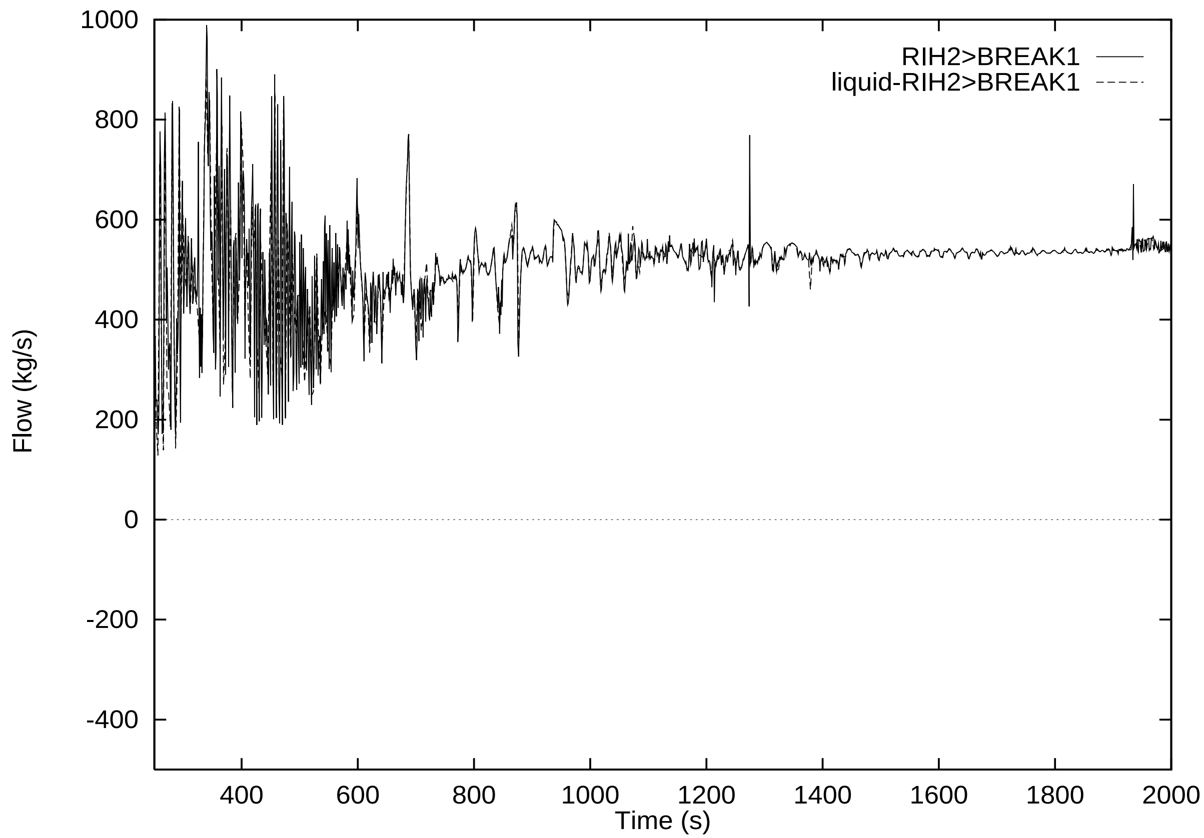


Figure 14 Break Discharge Flow for 25% RIH Break with Subsequent Loss of Class IV Power

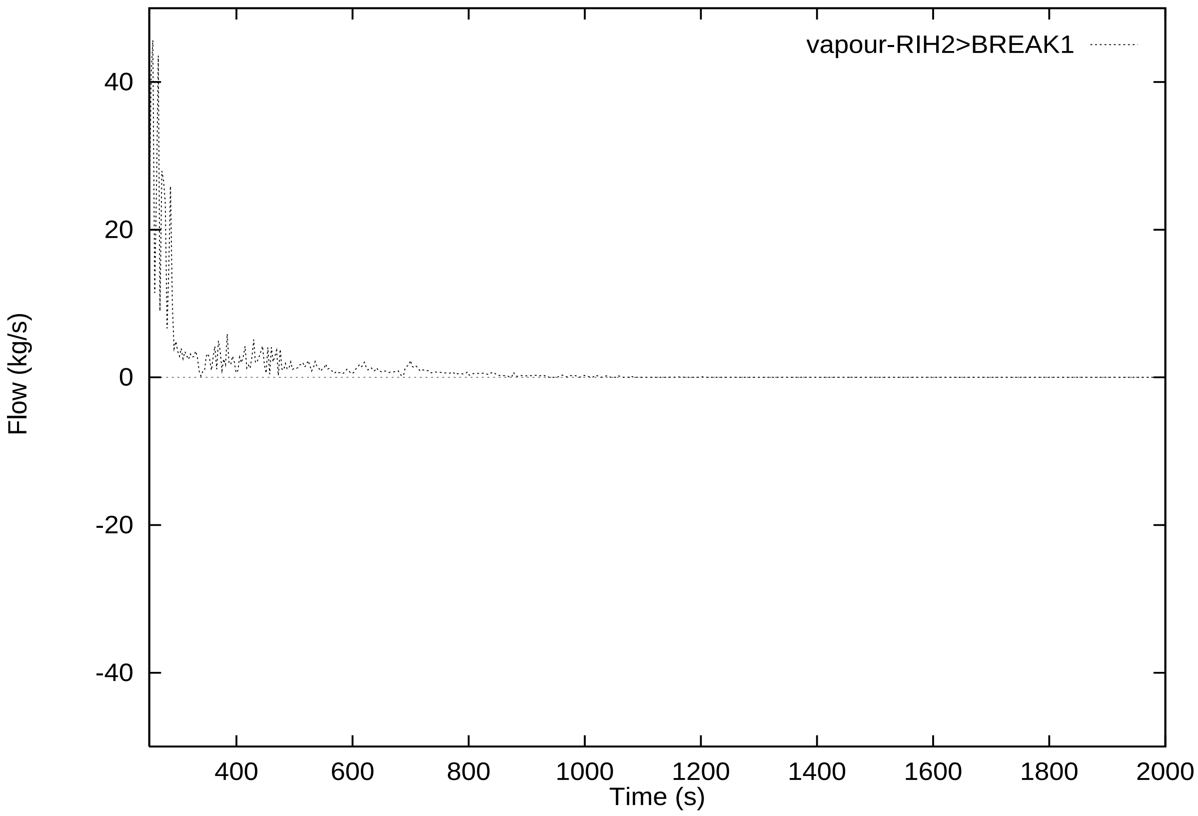


Figure 15 Break Discharge Flow for 25% RIH Break with Subsequent Loss of Class IV Power

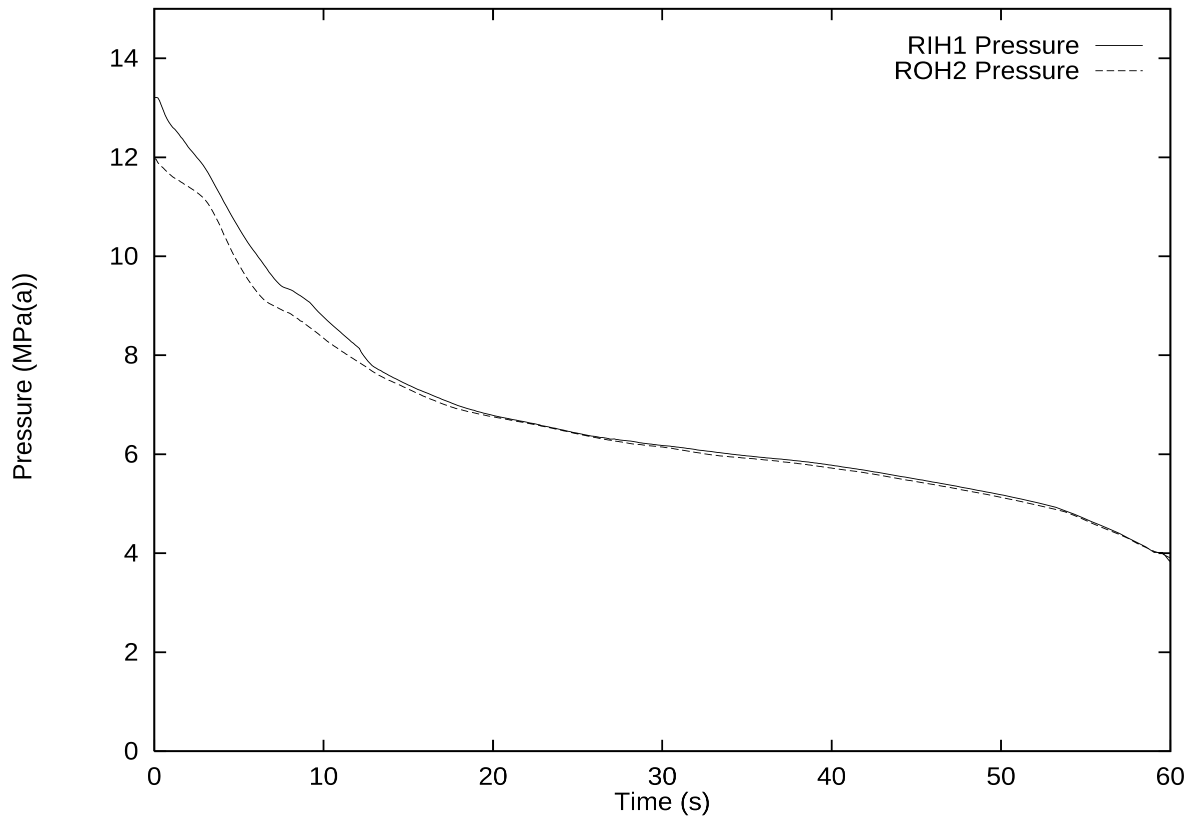


Figure 16 Reactor Header Pressures for 25% RIH Break with Subsequent Loss of Class IV Power

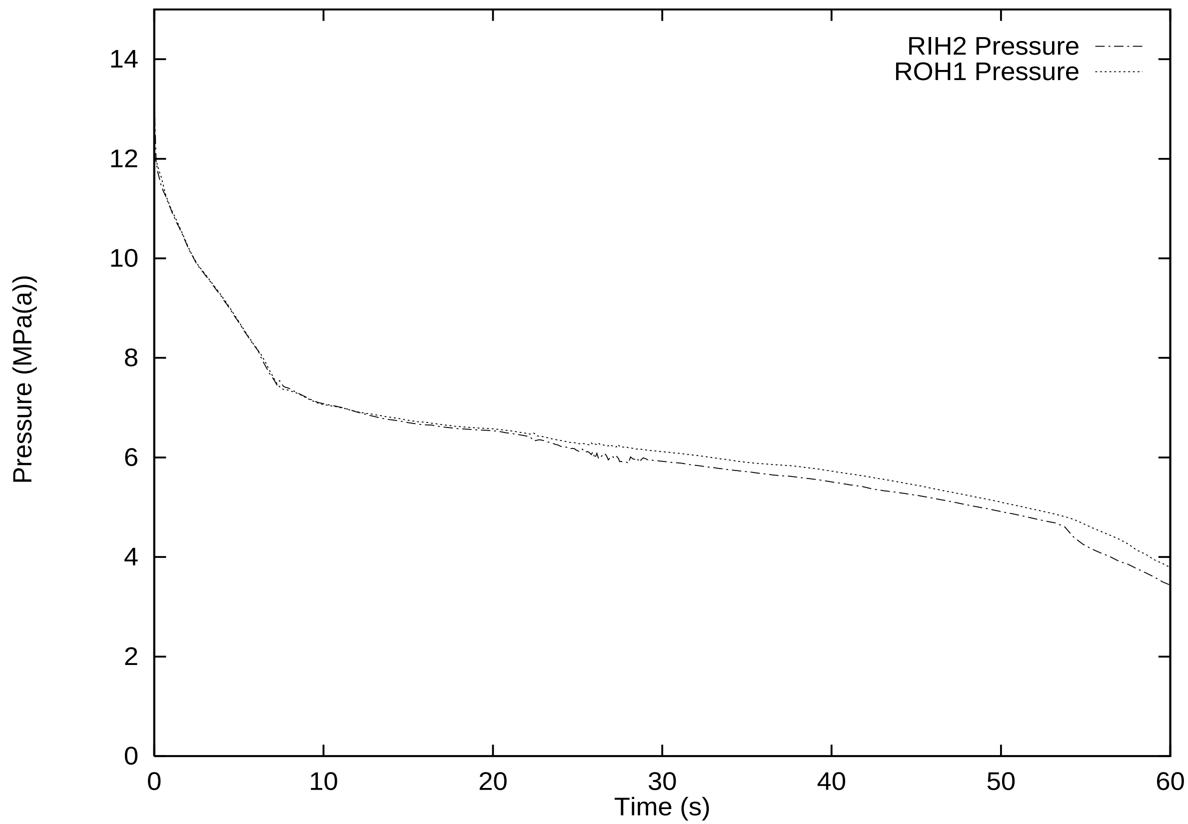


Figure 17 Reactor Header Pressures for 25% RIH Break with Subsequent Loss of Class IV Power

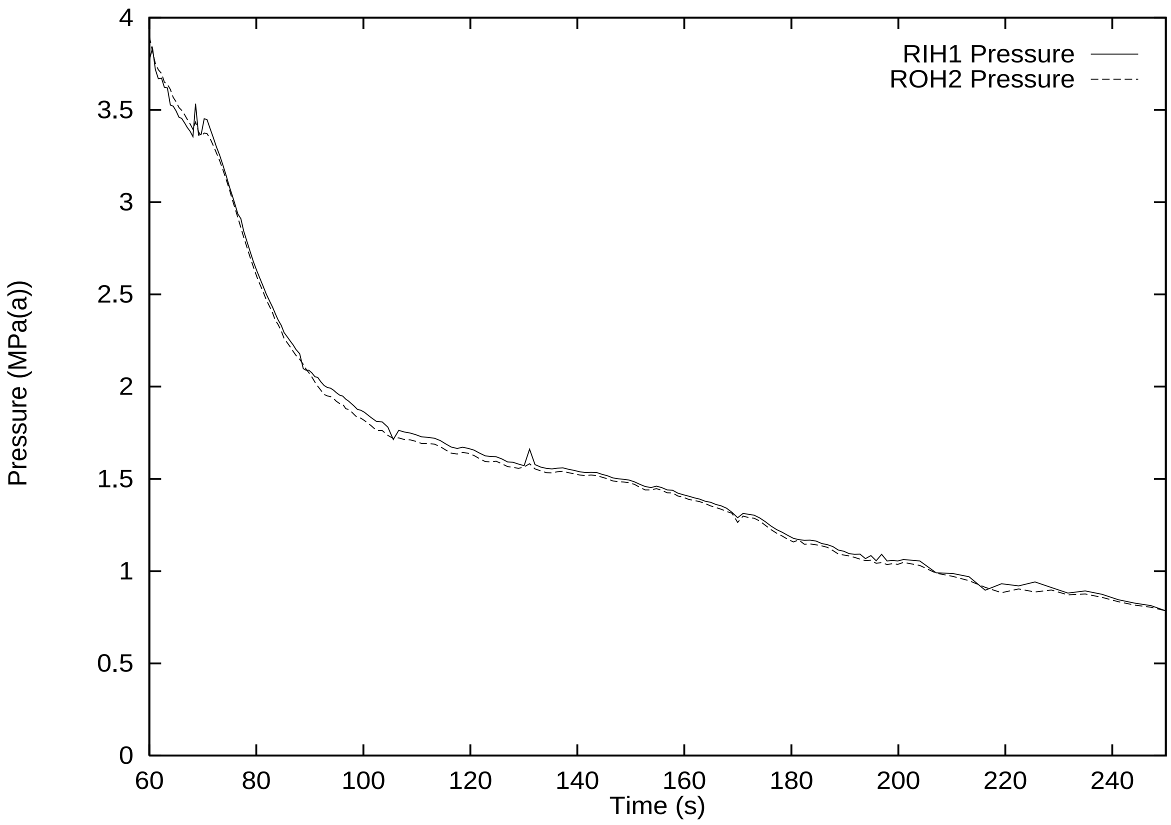


Figure 18 Reactor Header Pressures for 25% RIH Break with Subsequent Loss of Class IV Power

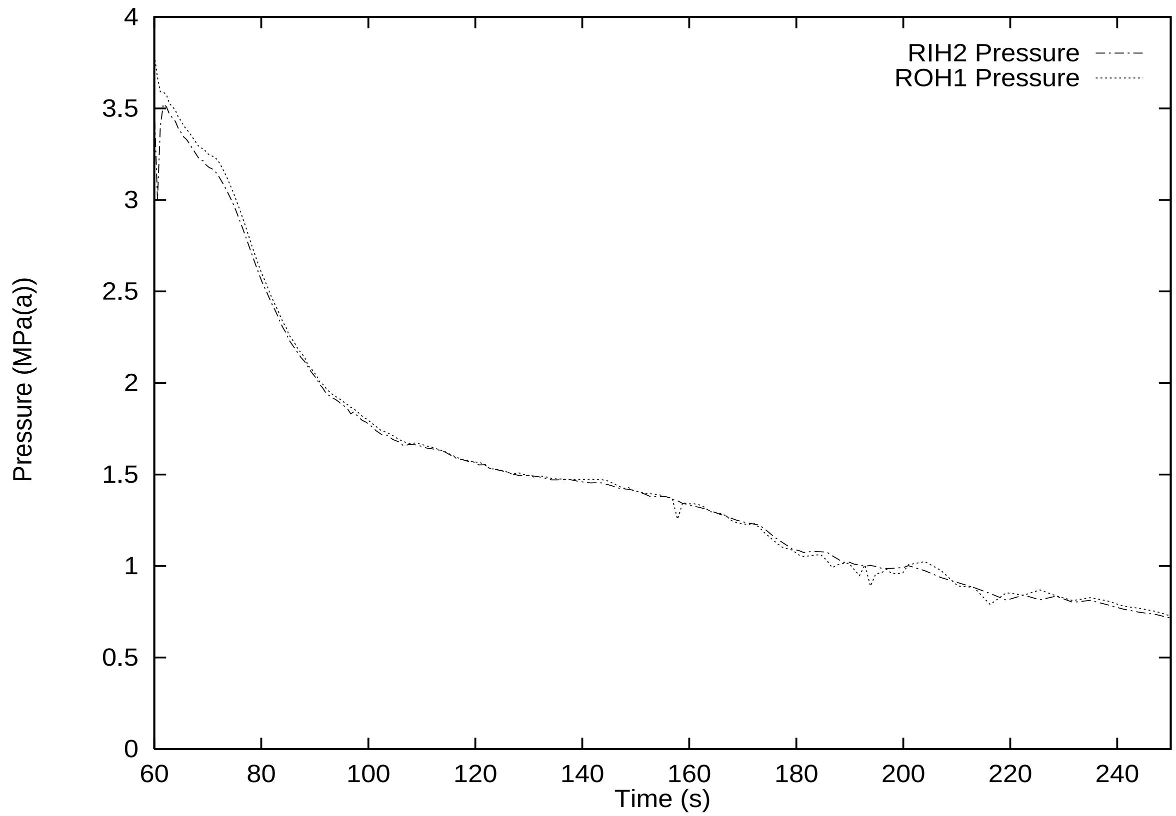


Figure 19 Reactor Header Pressures for 25% RIH Break with Subsequent Loss of Class IV Power

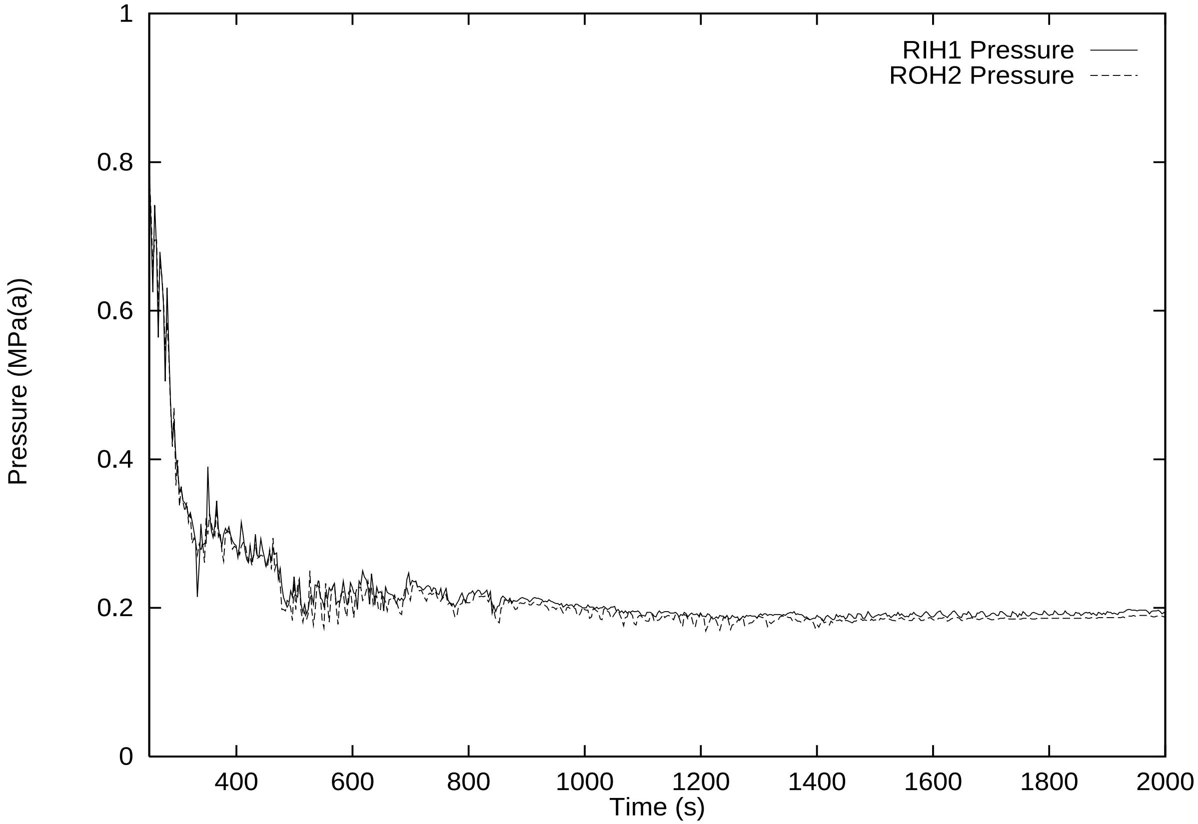


Figure 20 Reactor Header Pressures for 25% RIH Break with Subsequent Loss of Class IV Power

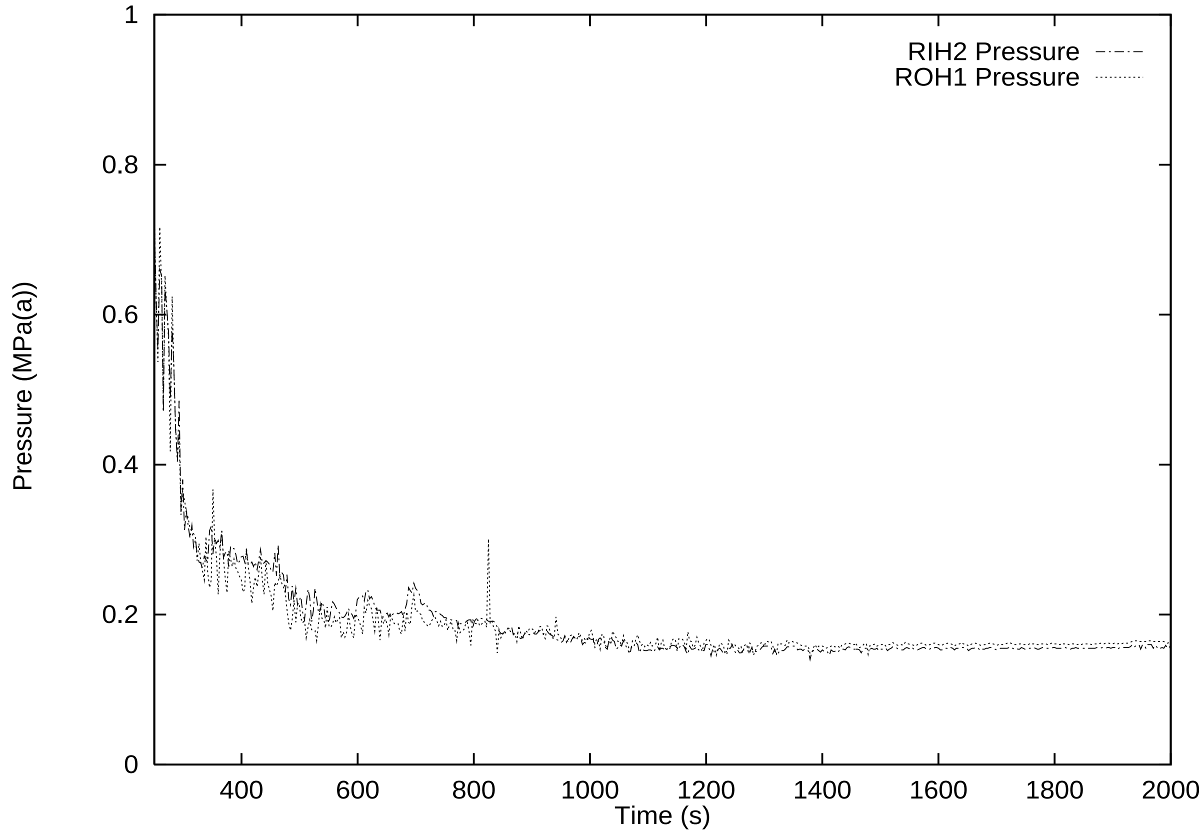


Figure 21 Reactor Header Pressures for 25% RIH Break with Subsequent Loss of Class IV Power

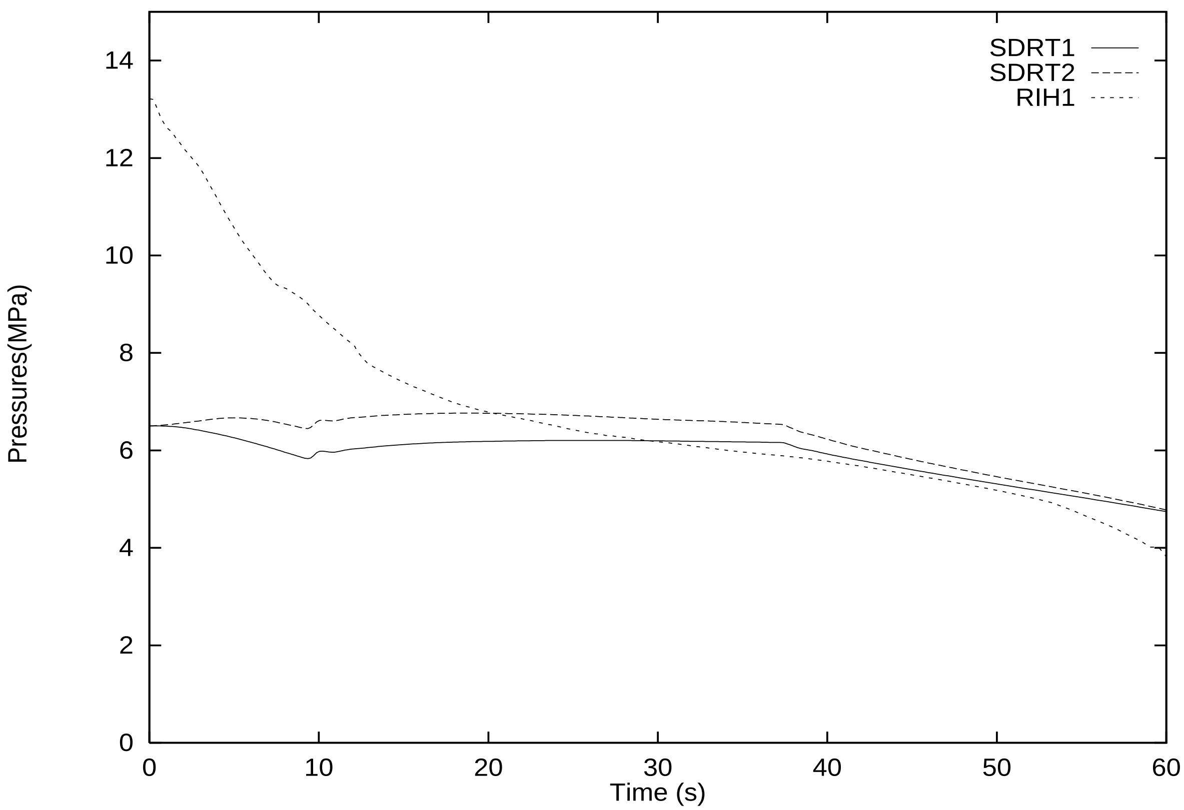


Figure 22 Primary and Secondary Side Pressures for 25% RIH Break with Subsequent Loss of Class IV Power

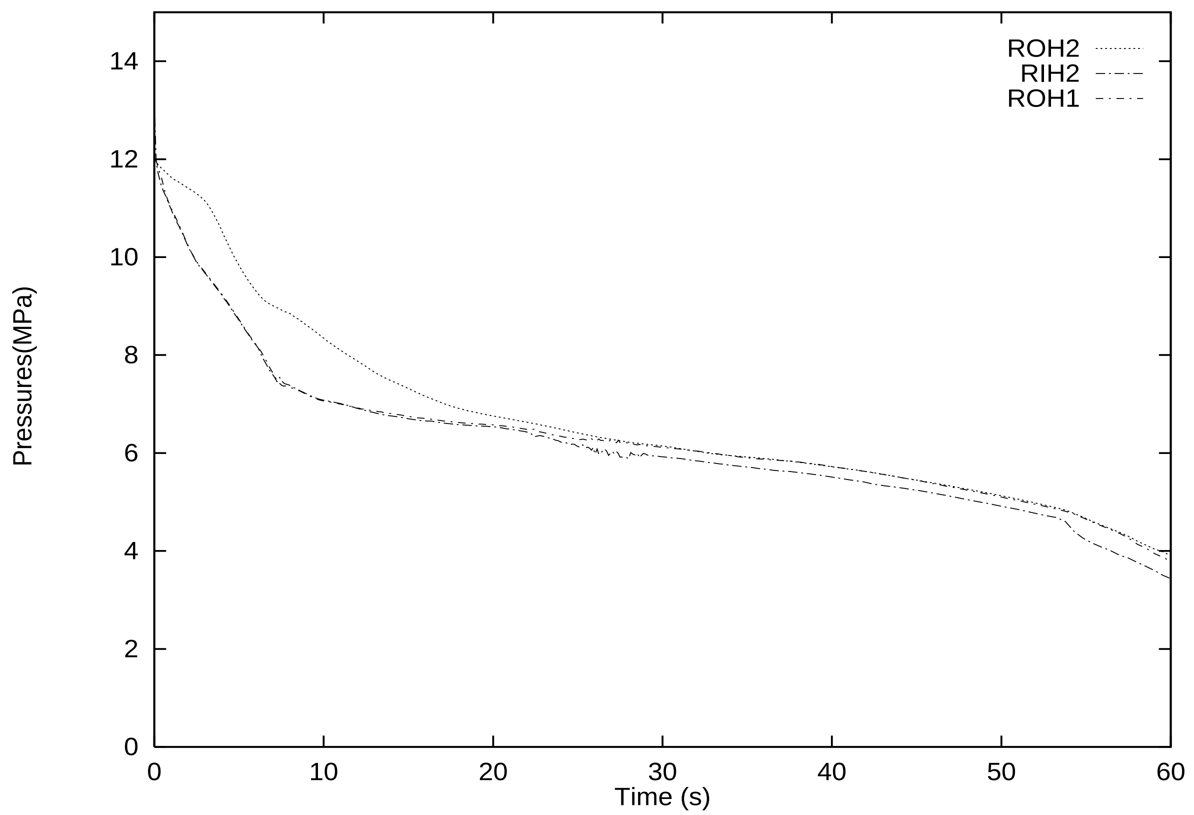


Figure 23 Primary and Secondary Side Pressures for 25% RIH Break with Subsequent Loss of Class IV Power

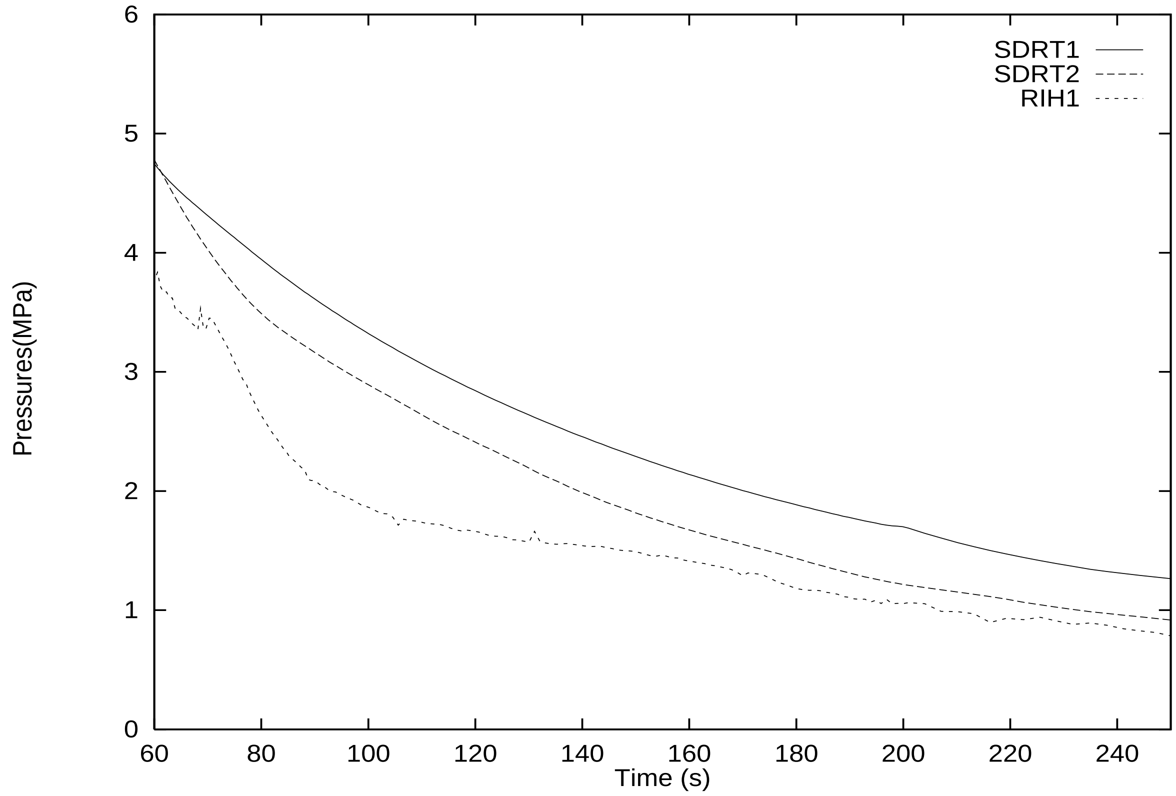


Figure 24 Primary and Secondary Side Pressures for 25% RIH Break with Subsequent Loss of Class IV Power

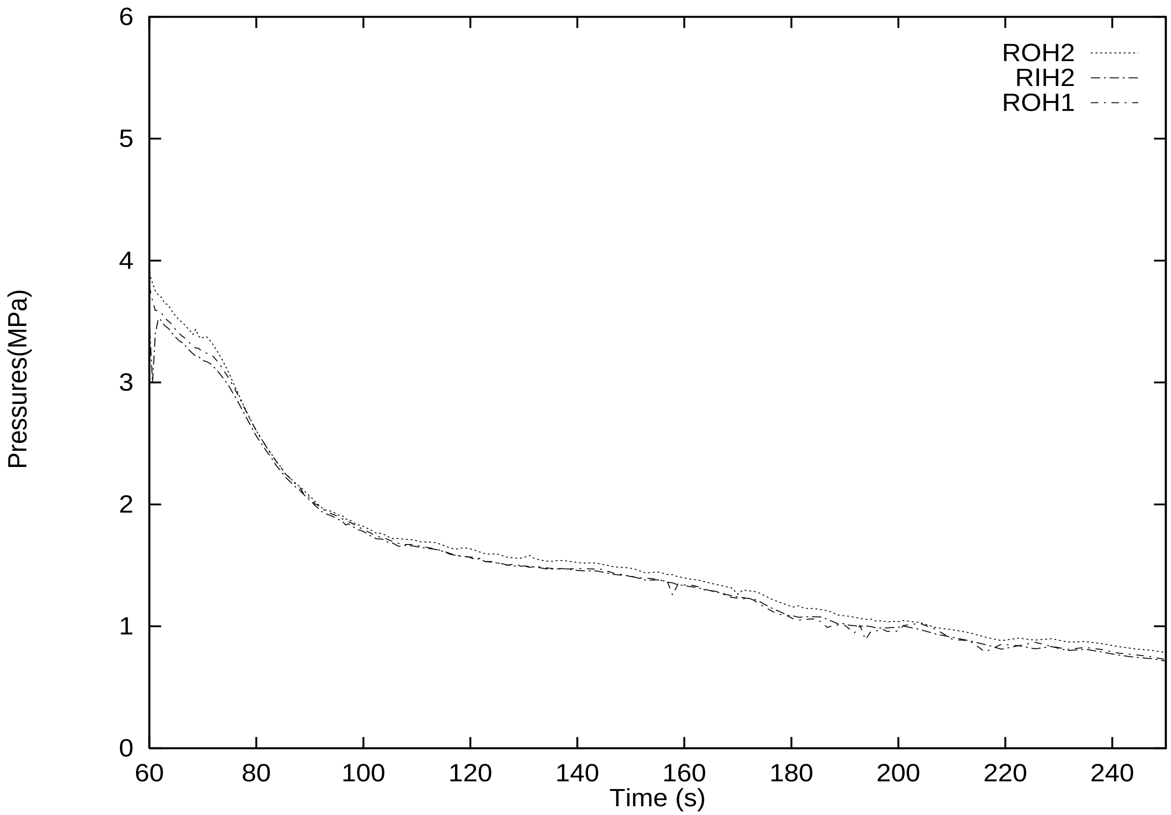


Figure 25 Primary and Secondary Side Pressures for 25% RIH Break with Subsequent Loss of Class IV Power

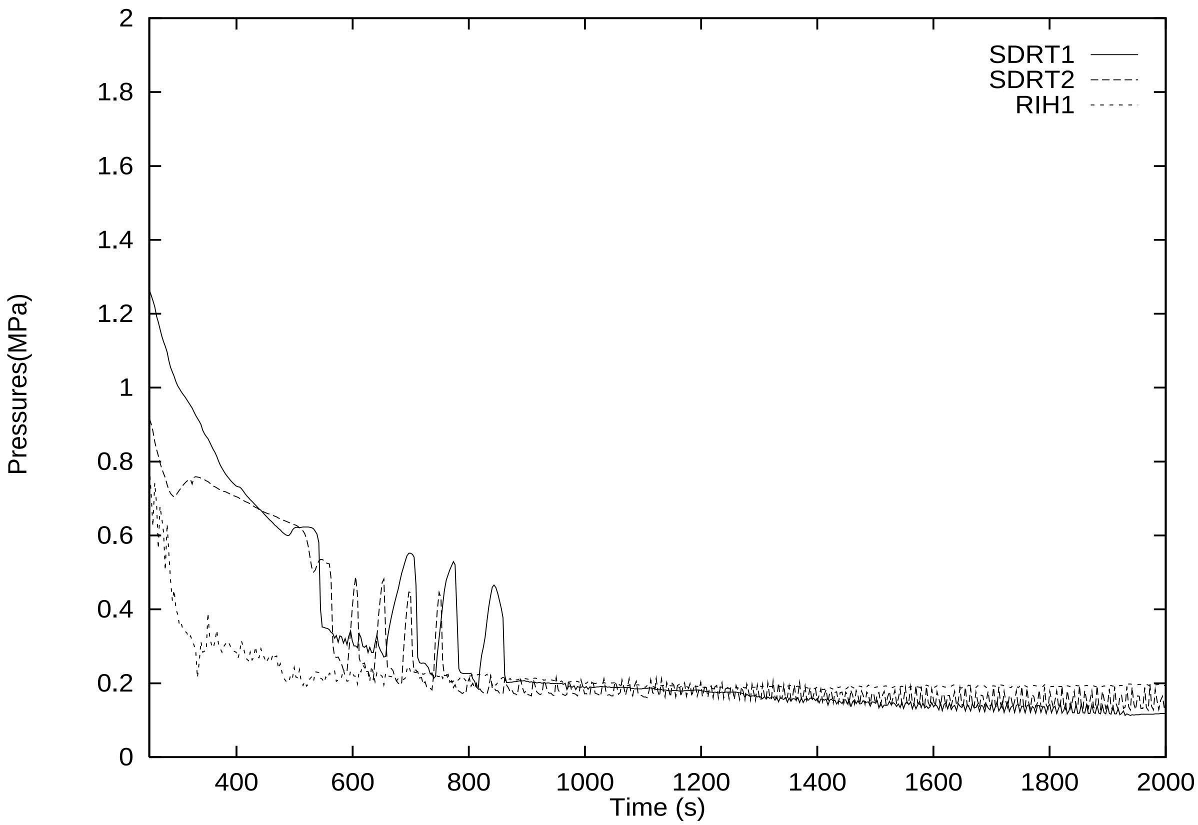


Figure 26 Primary and Secondary Side Pressures for 25% RIH Break with Subsequent Loss of Class IV Power

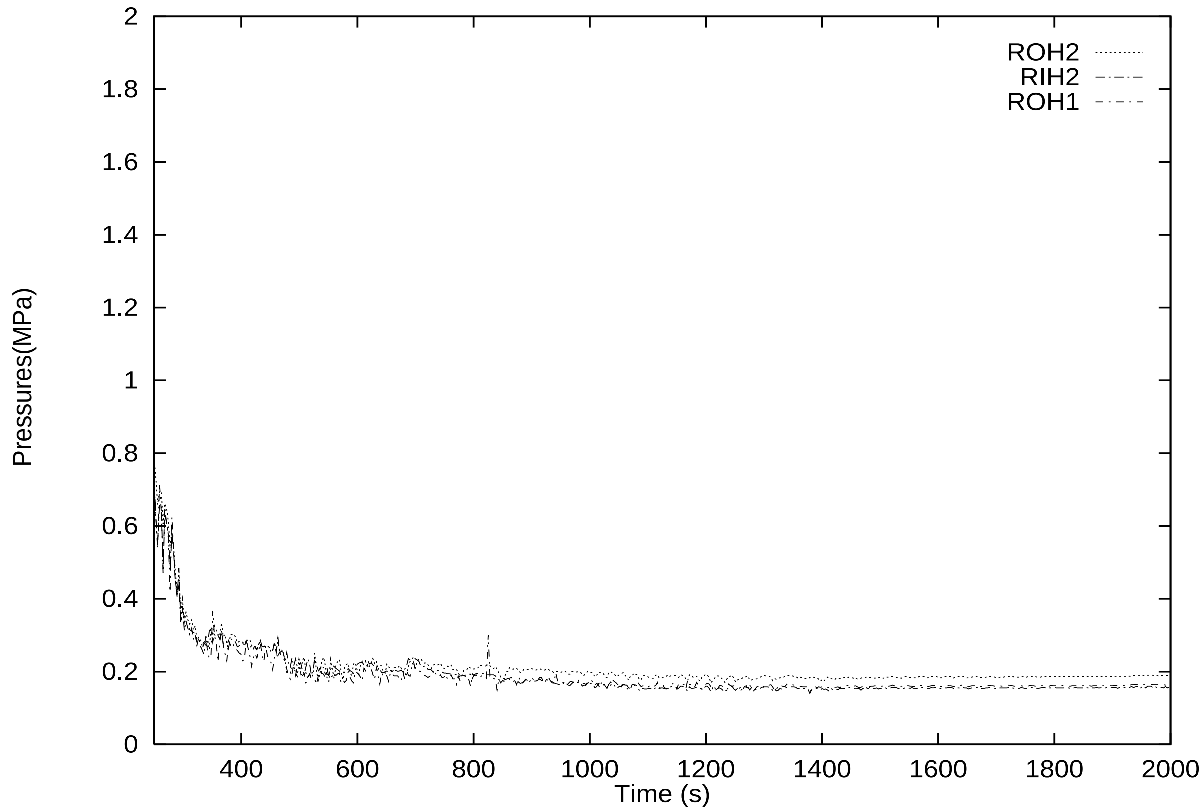


Figure 27 Primary and Secondary Side Pressures for 25% RIH Break with Subsequent Loss of Class IV Power

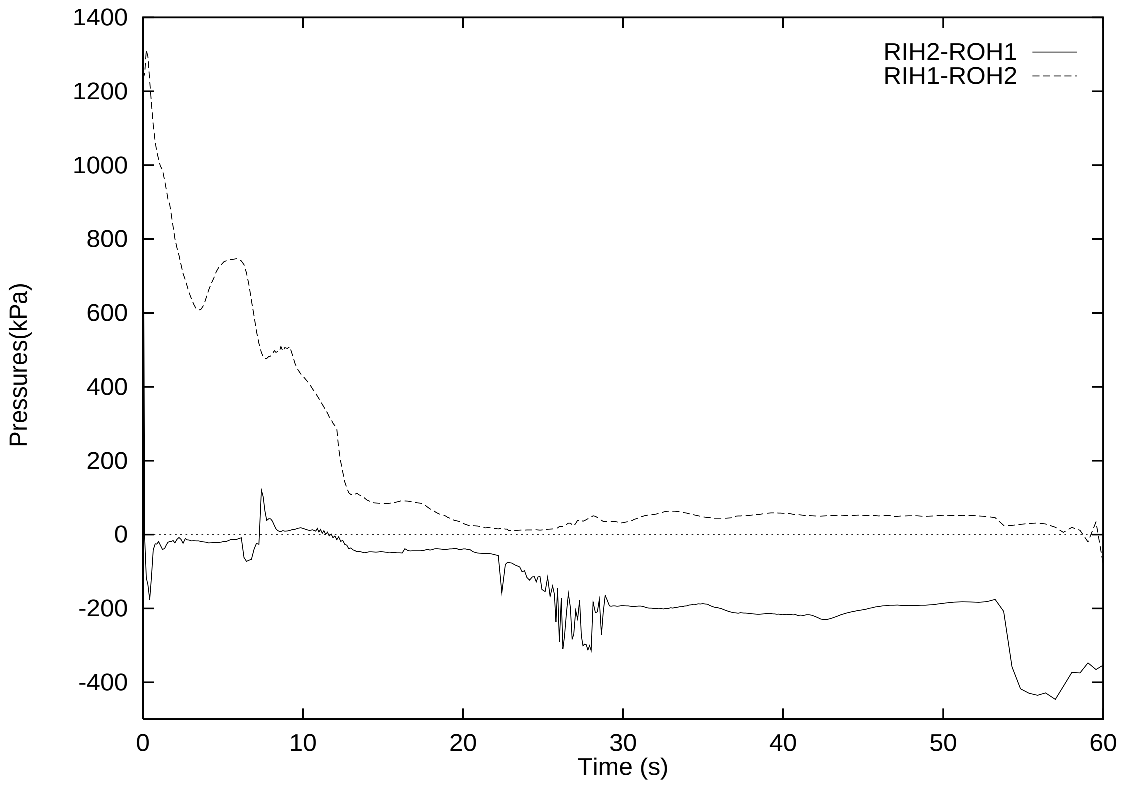


Figure 28 Header Pressure Differences for 25% RIH Break with Subsequent Loss of Class IV Power

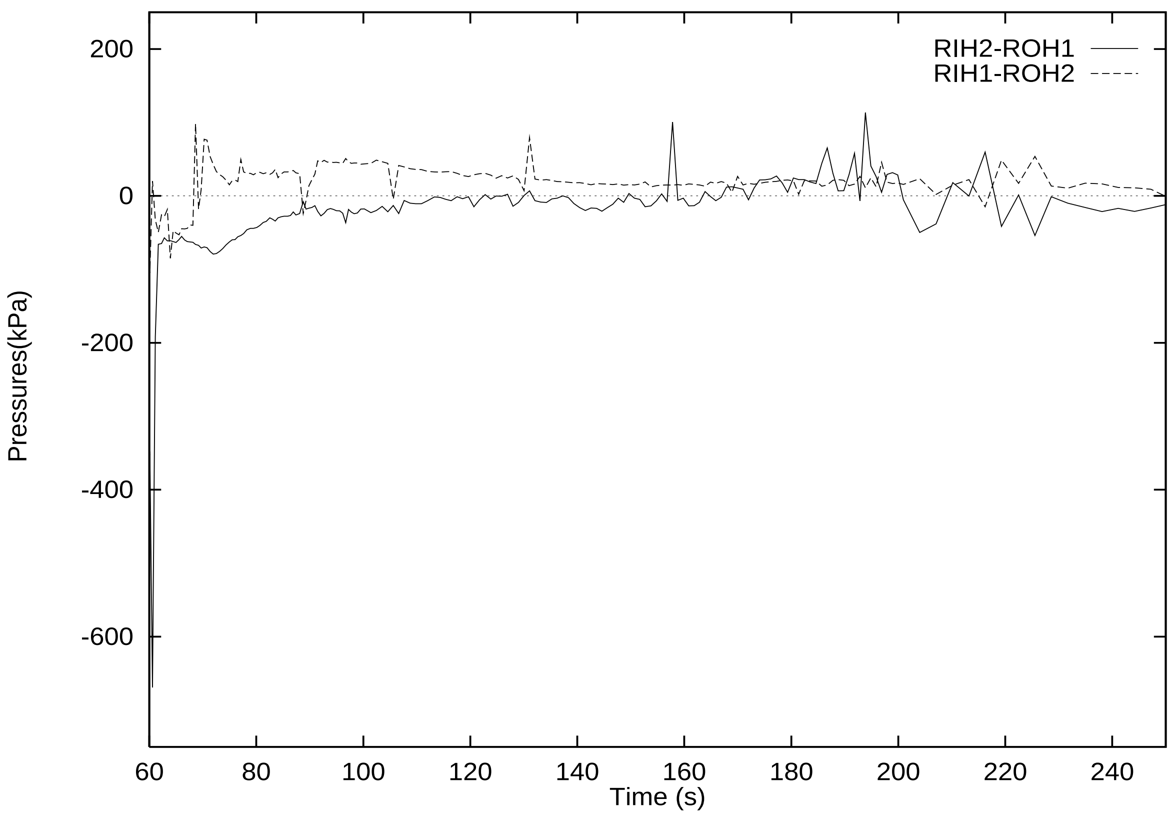


Figure 29 Header Pressure Differences for 25% RIH Break with Subsequent Loss of Class IV Power

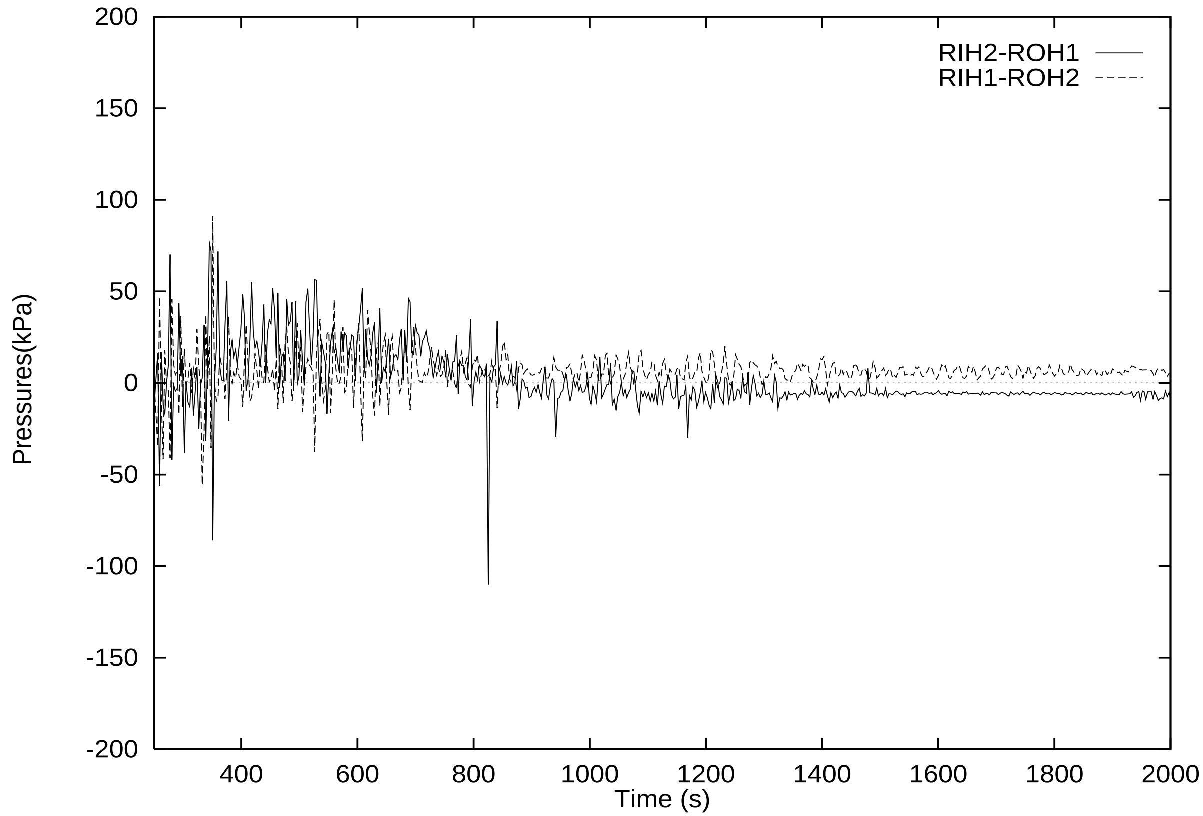


Figure 30 Header Pressure Differences for 25% RIH Break with Subsequent Loss of Class IV Power

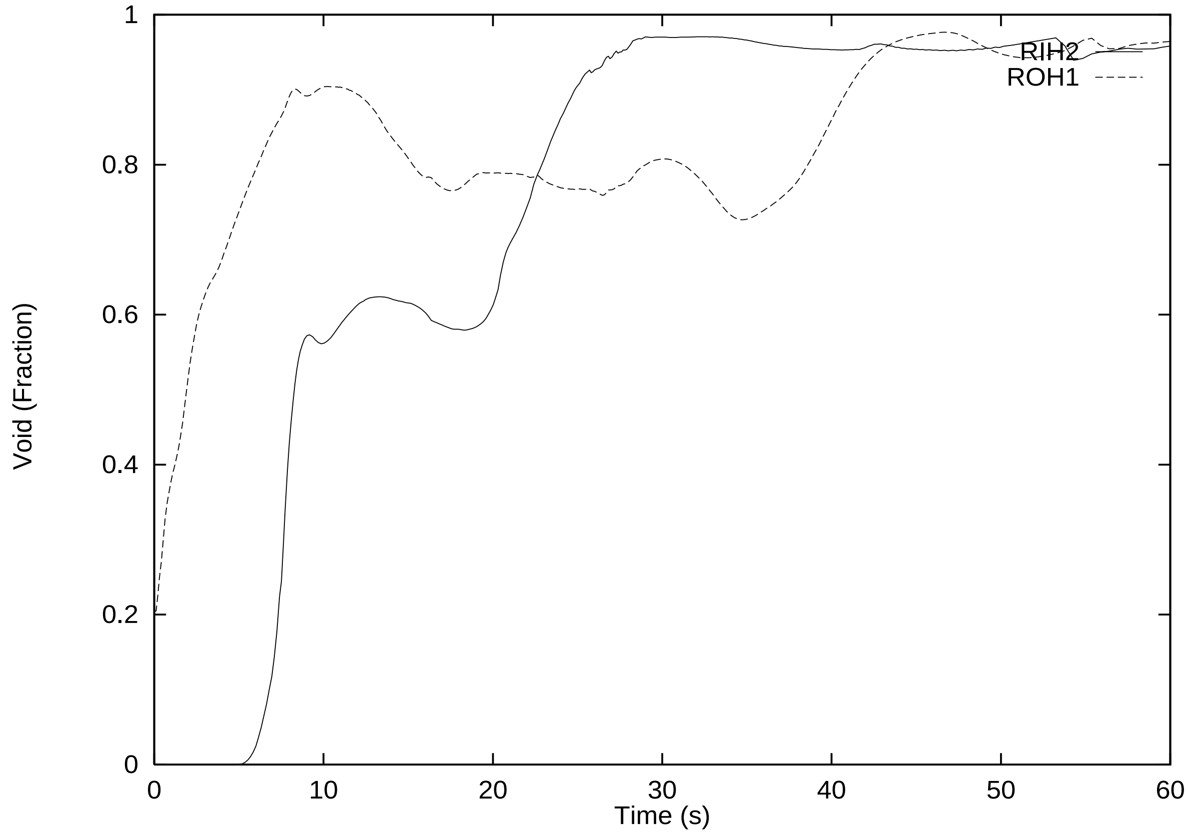


Figure 31 Header Voids for 25% RIH Break with Subsequent Loss of Class IV Power

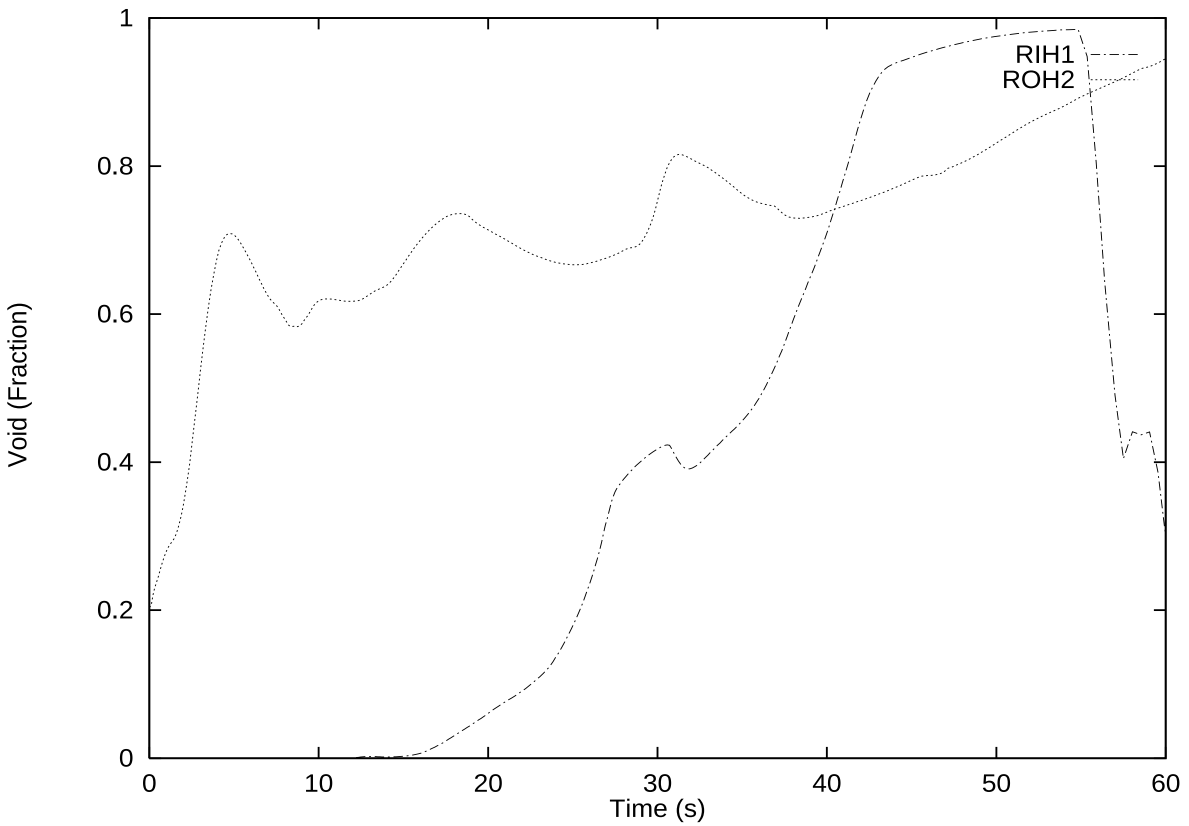


Figure 32 Header Voids for 25% RIH Break with Subsequent Loss of Class IV Power

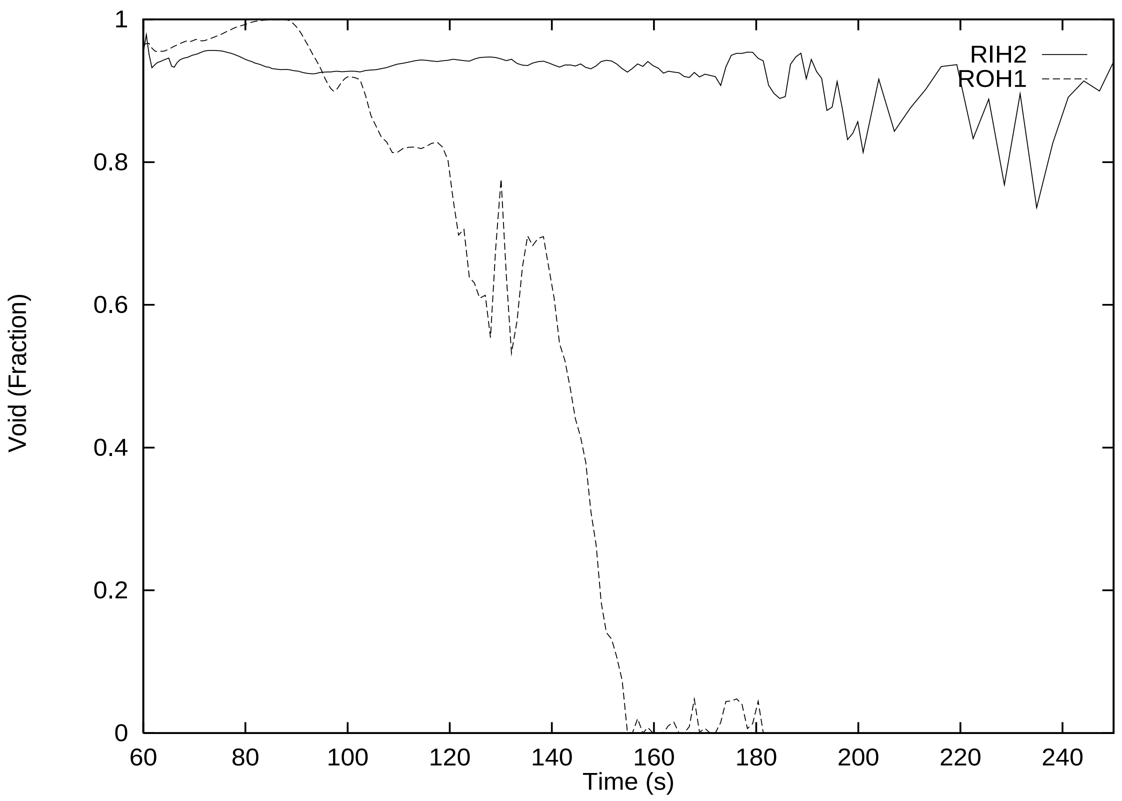


Figure 33 Header Voids for 25% RIH Break with Subsequent Loss of Class IV Power

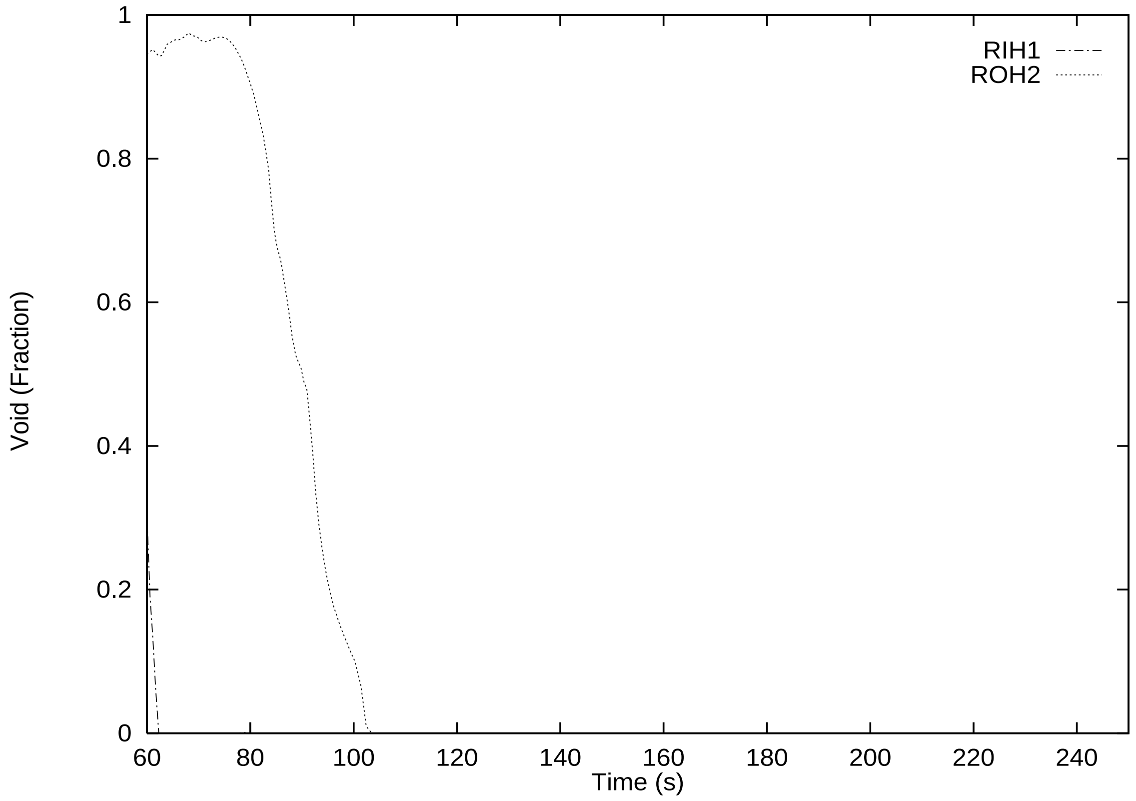


Figure 34 Header Voids for 25% RIH Break with Subsequent Loss of Class IV Power

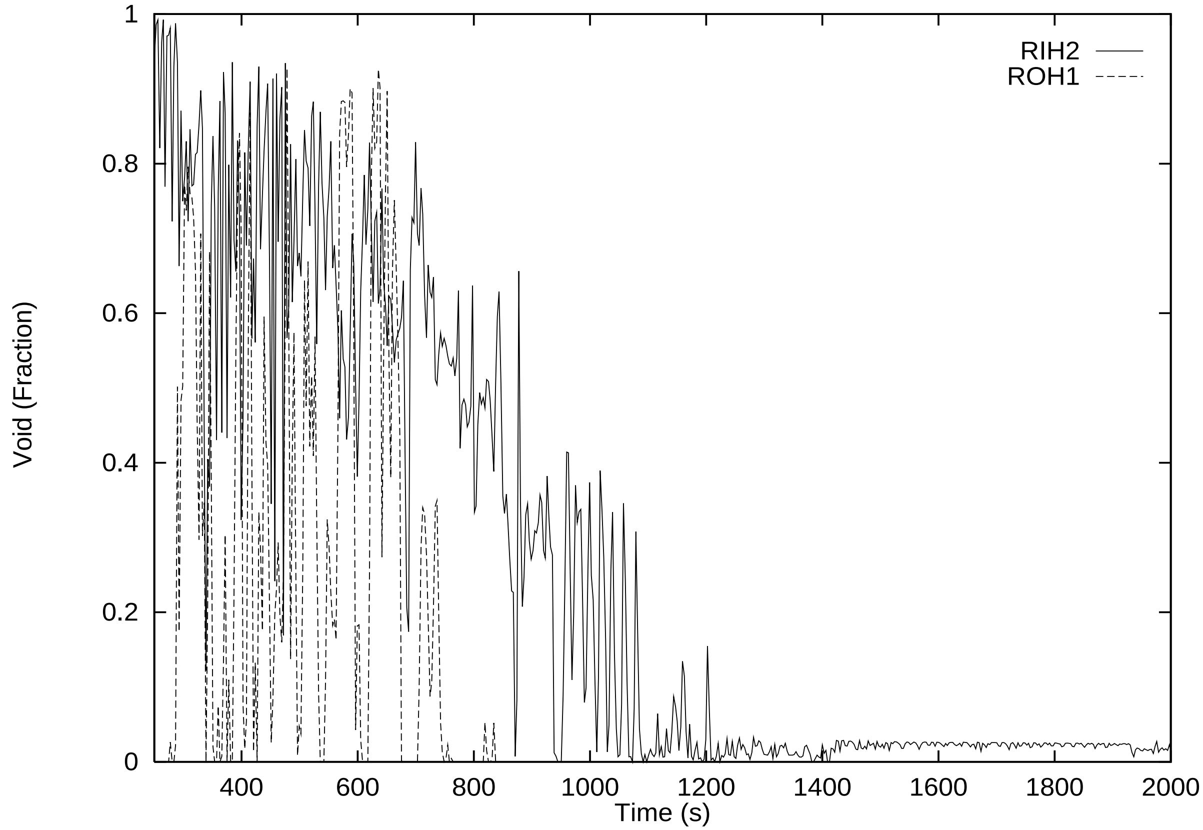


Figure 35 Header Voids for 25% RIH Break with Subsequent Loss of Class IV Power

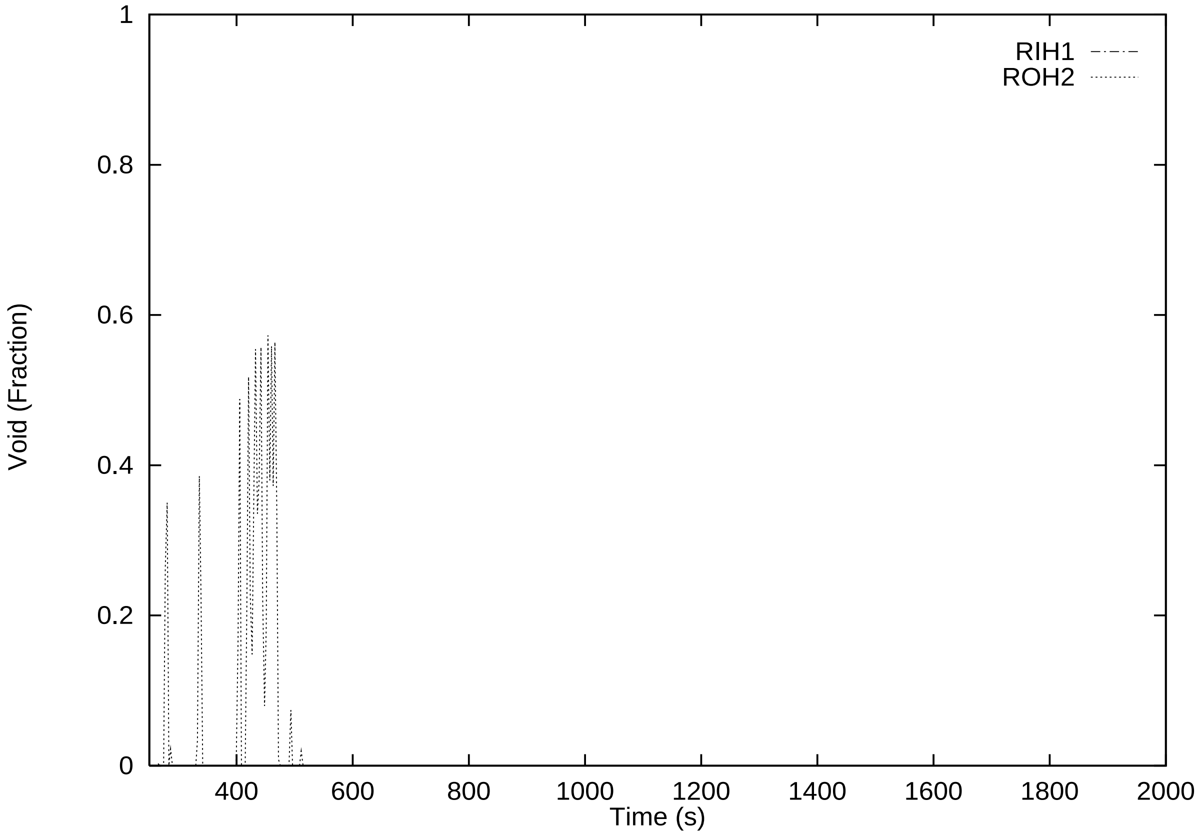


Figure 36 Header Voids for 25% RIH Break with Subsequent Loss of Class IV Power

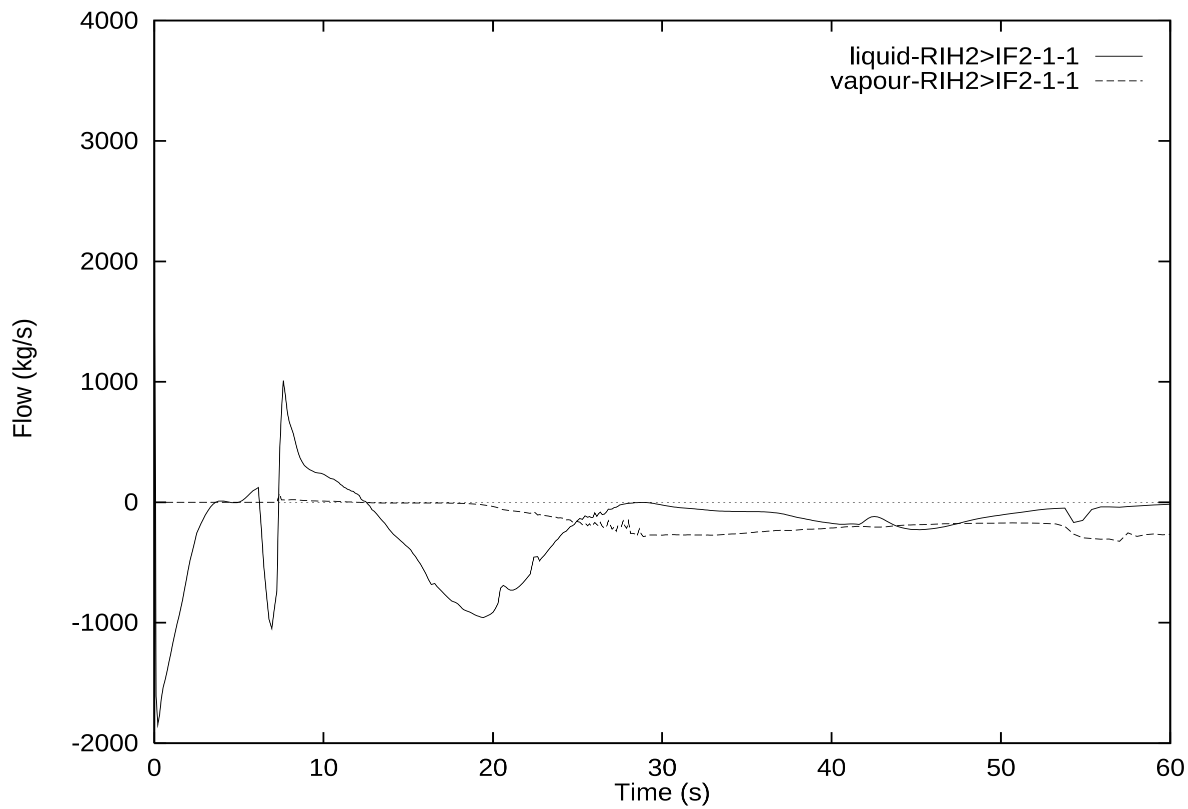


Figure 37 Inlet Header Flows for 25% RIH Break with Subsequent Loss of Class IV Power

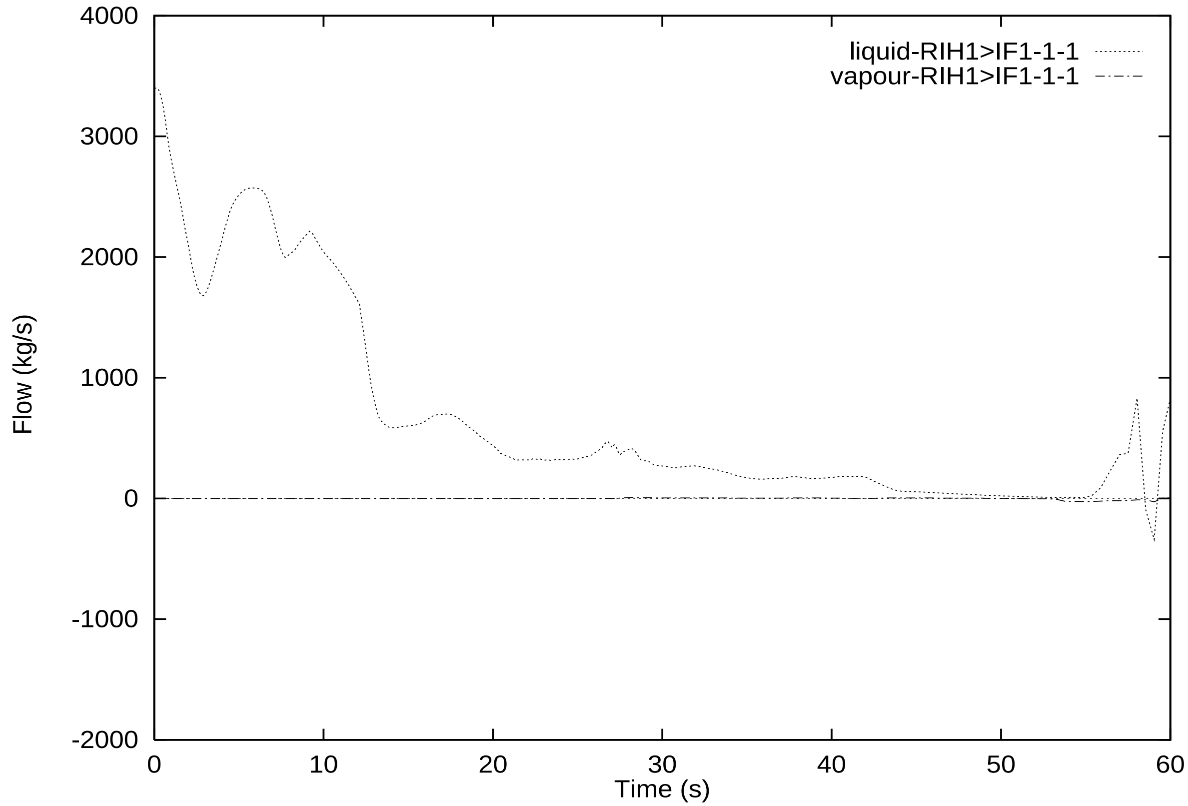


Figure 38 Inlet Header Flows for 25% RIH Break with Subsequent Loss of Class IV Power

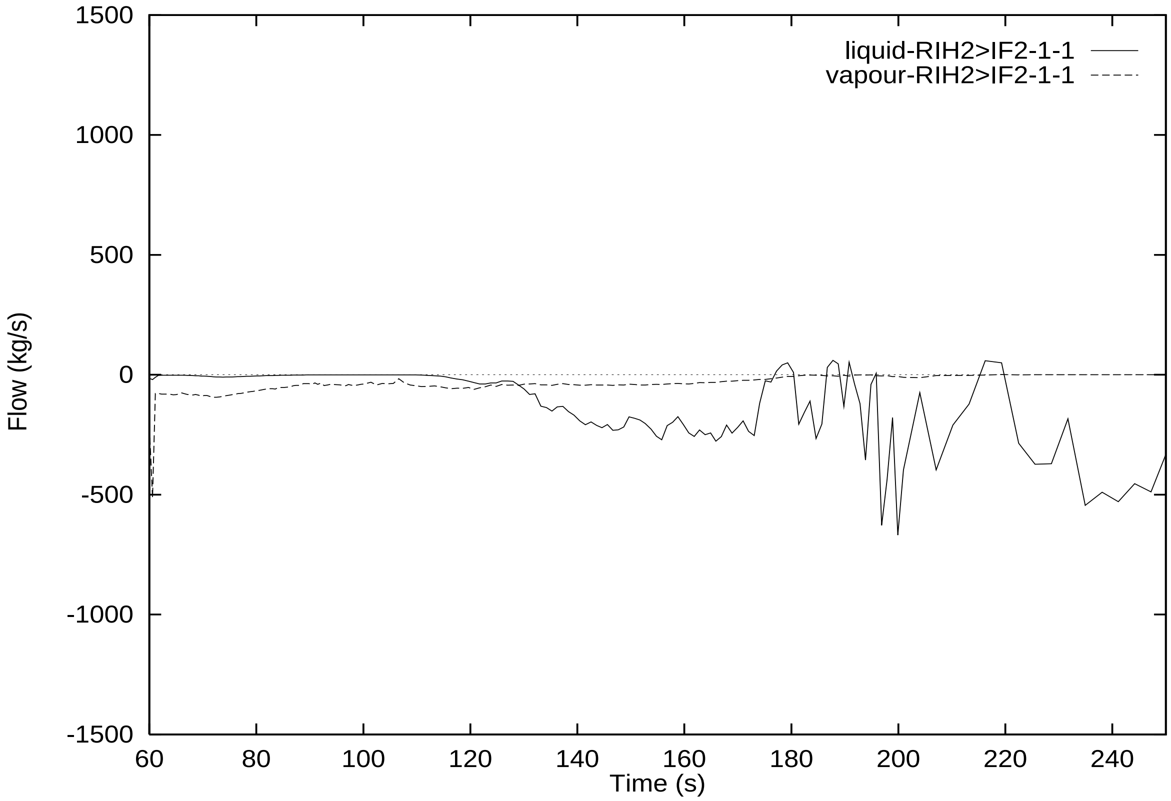


Figure 39 Inlet Header Flows for 25% RIH Break with Subsequent Loss of Class IV Power

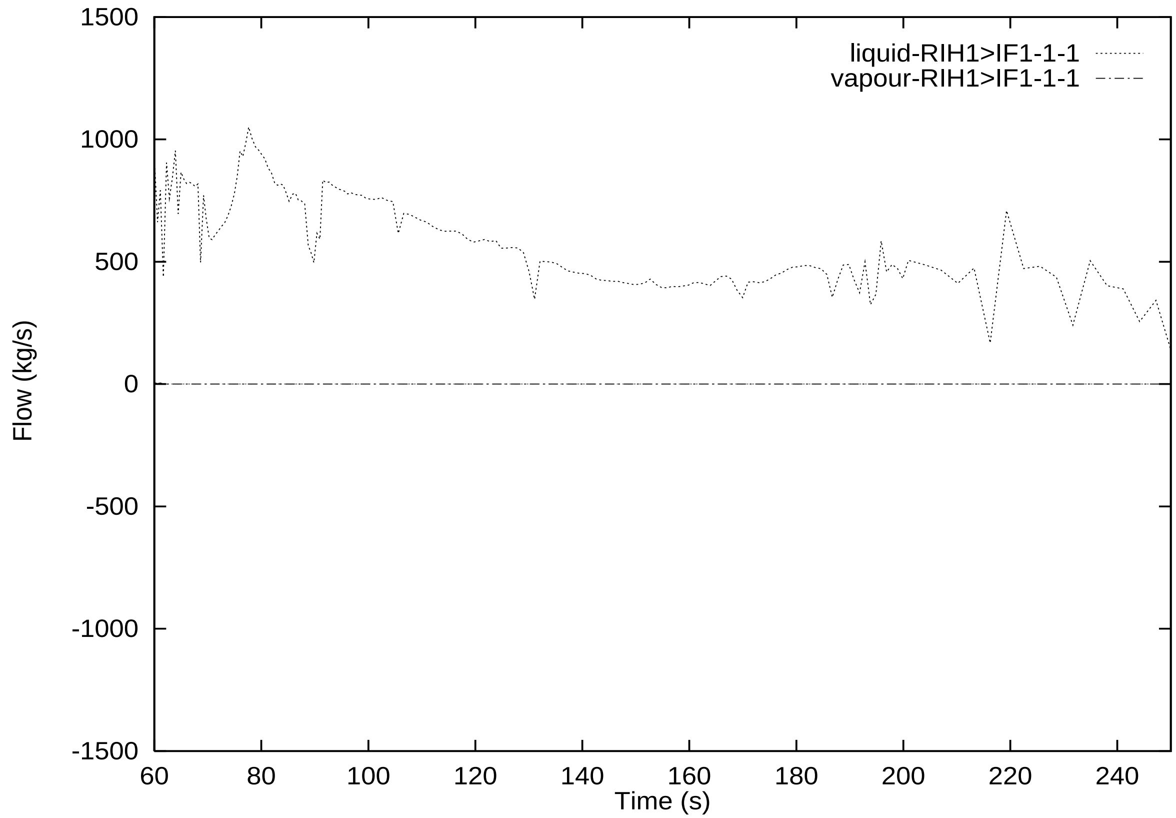


Figure 40 Inlet Header Flows for 25% RIH Break with Subsequent Loss of Class IV Power

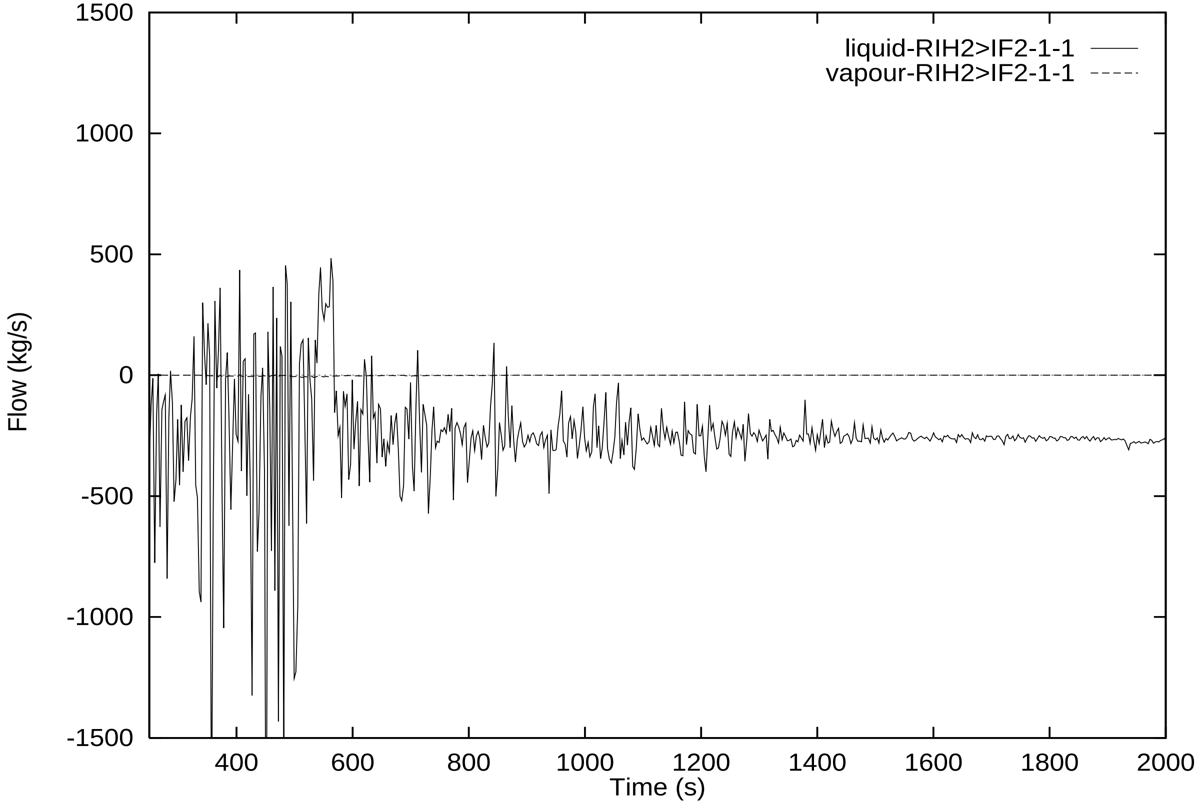


Figure 41 Inlet Header Flows for 25% RIH Break with Subsequent Loss of Class IV Power

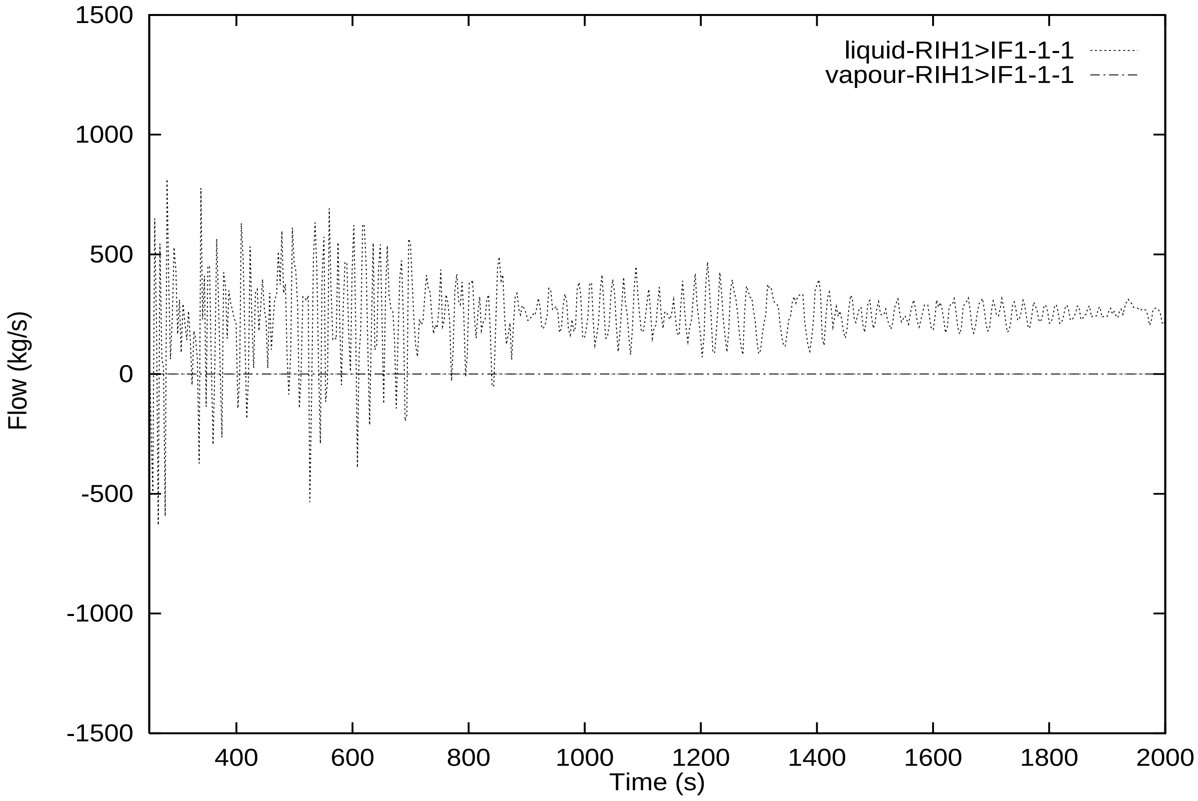


Figure 42 Inlet Header Flows for 25% RIH Break with Subsequent Loss of Class IV Power

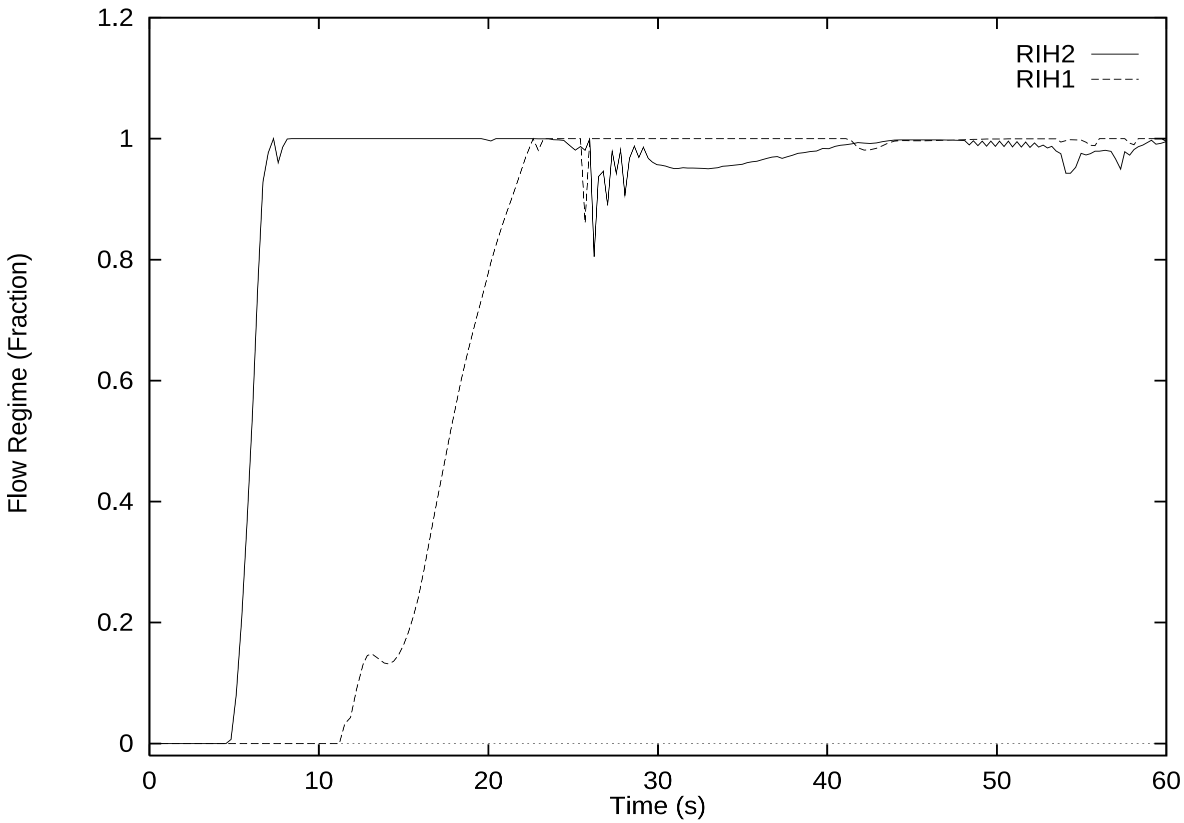


Figure 43 Inlet Header Flow Regime for 25% RIH Break with Subsequent Loss of Class IV Power

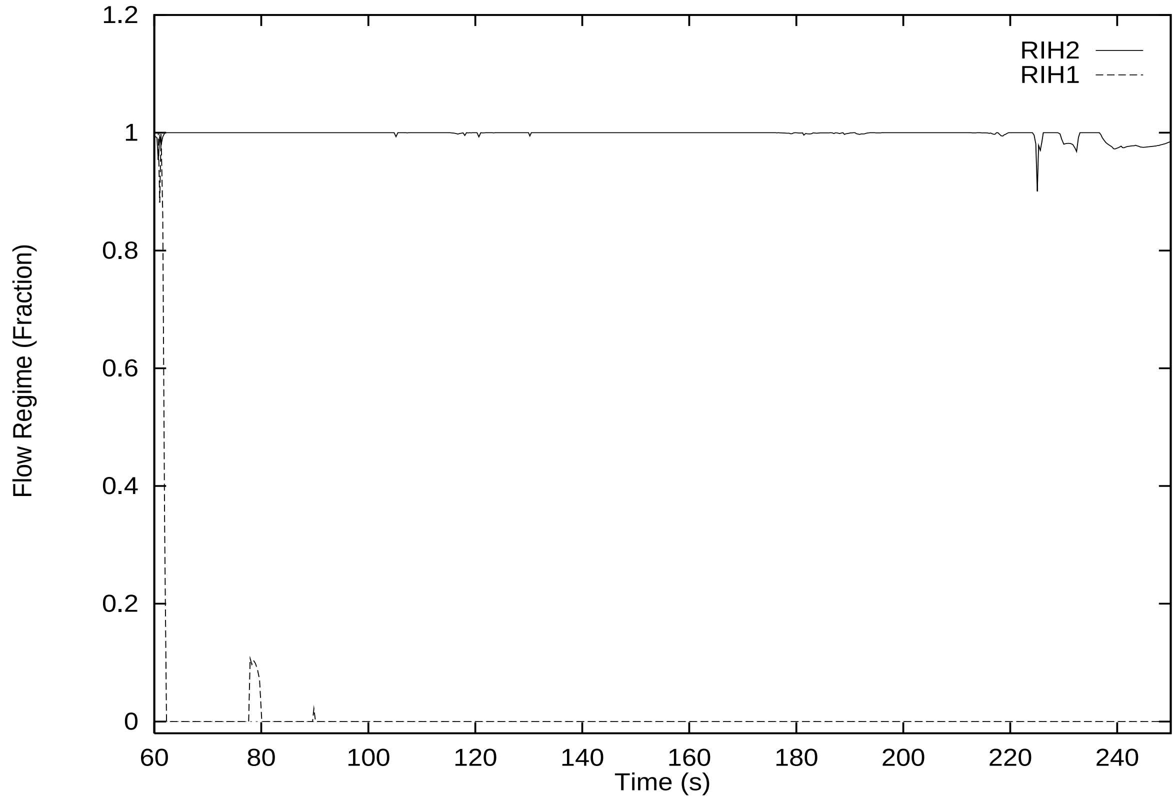


Figure 44 Inlet Header Flow Regime for 25% RIH Break with Subsequent Loss of Class IV Power

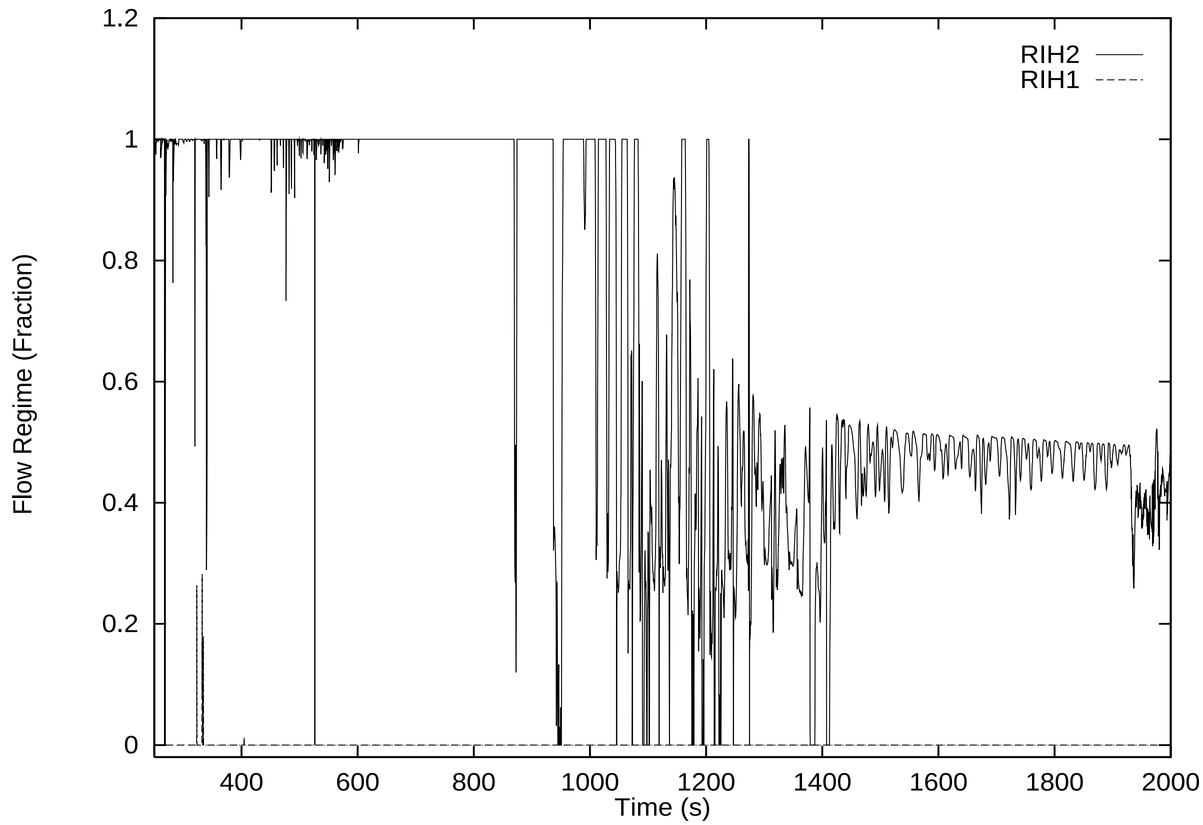


Figure 45 Inlet Header Flow Regime for 25% RIH Break with Subsequent Loss of Class IV Power

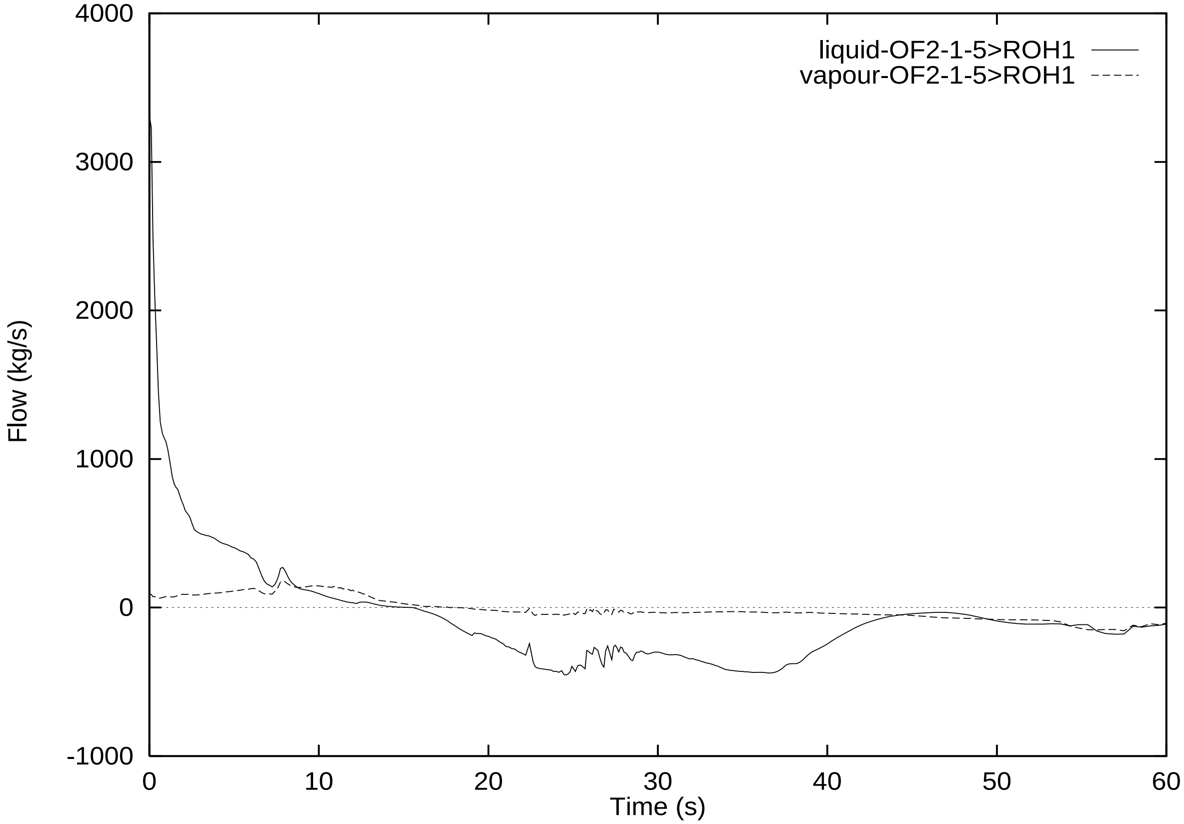


Figure 46 Outlet Header Flows for 25% RIH Break with Subsequent Loss of Class IV Power

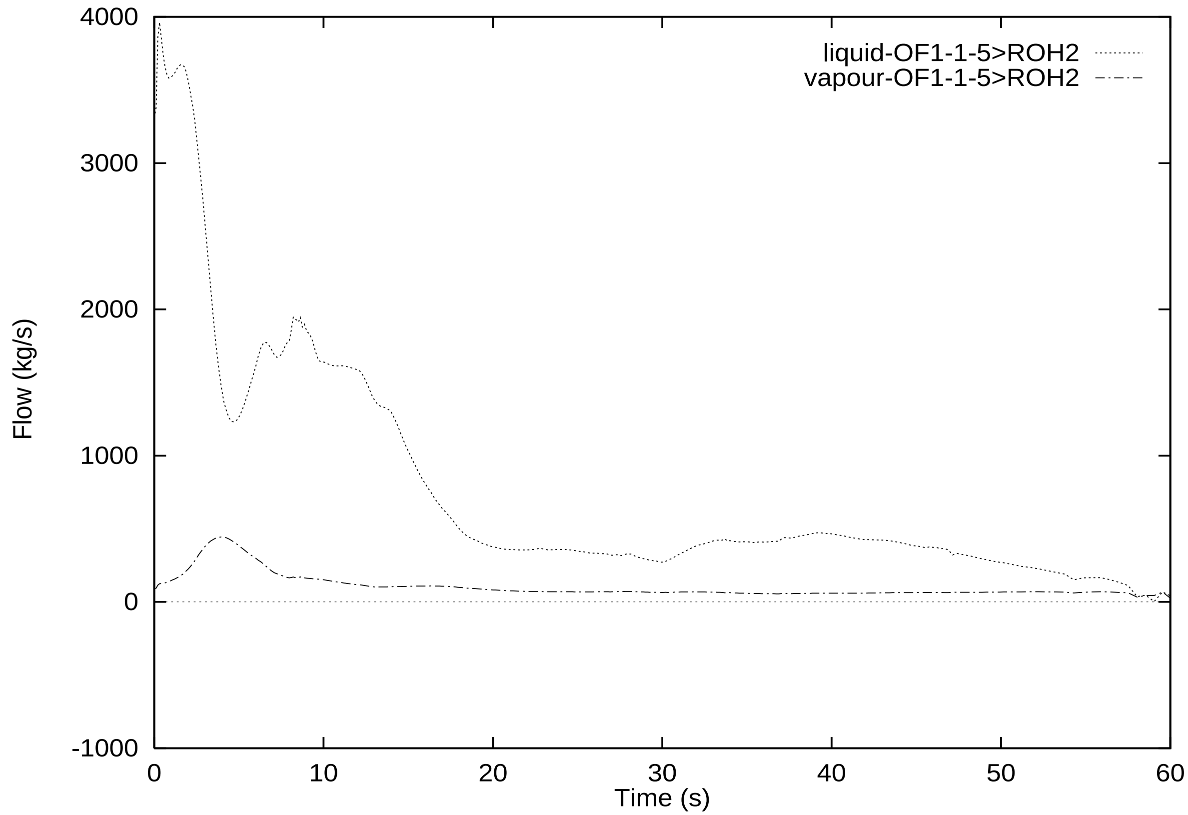


Figure 47 Outlet Header Flows for 25% RIH Break with Subsequent Loss of Class IV Power

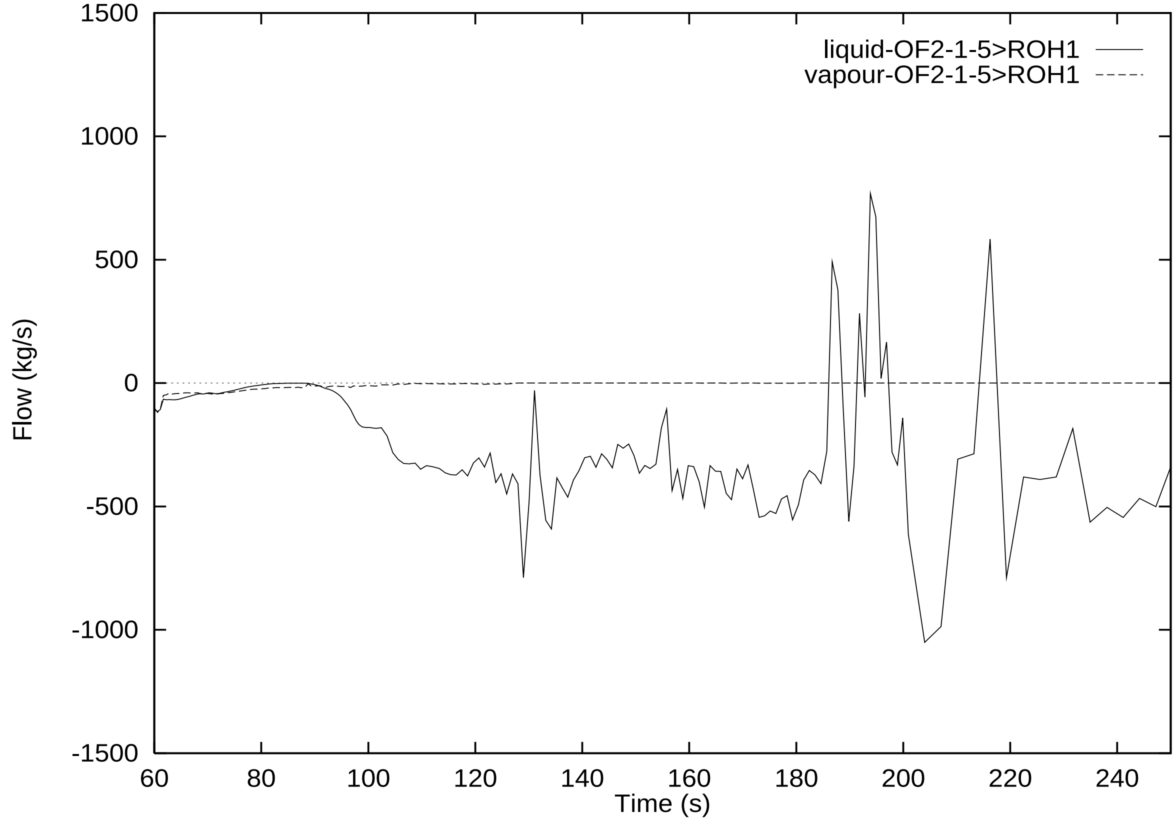


Figure 48 Outlet Header Flows for 25% RIH Break with Subsequent Loss of Class IV Power

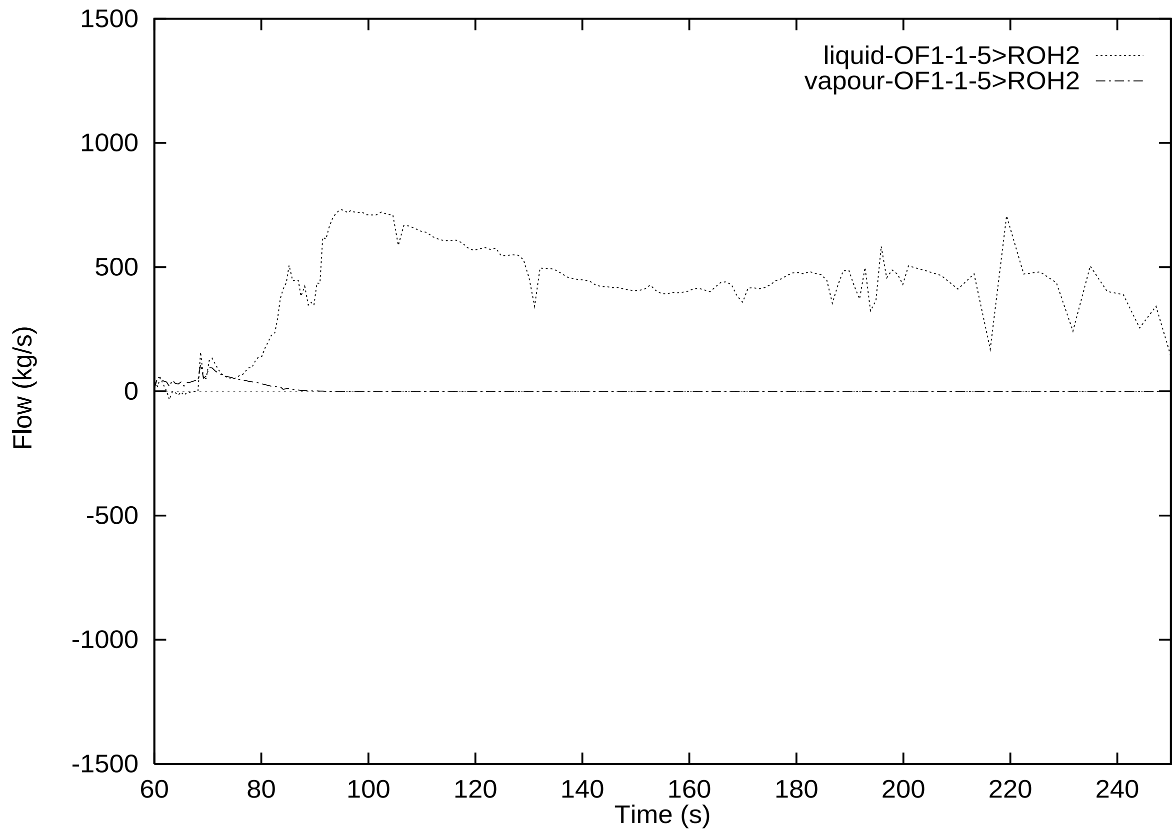


Figure 49 Outlet Header Flows for 25% RIH Break with Subsequent Loss of Class IV Power

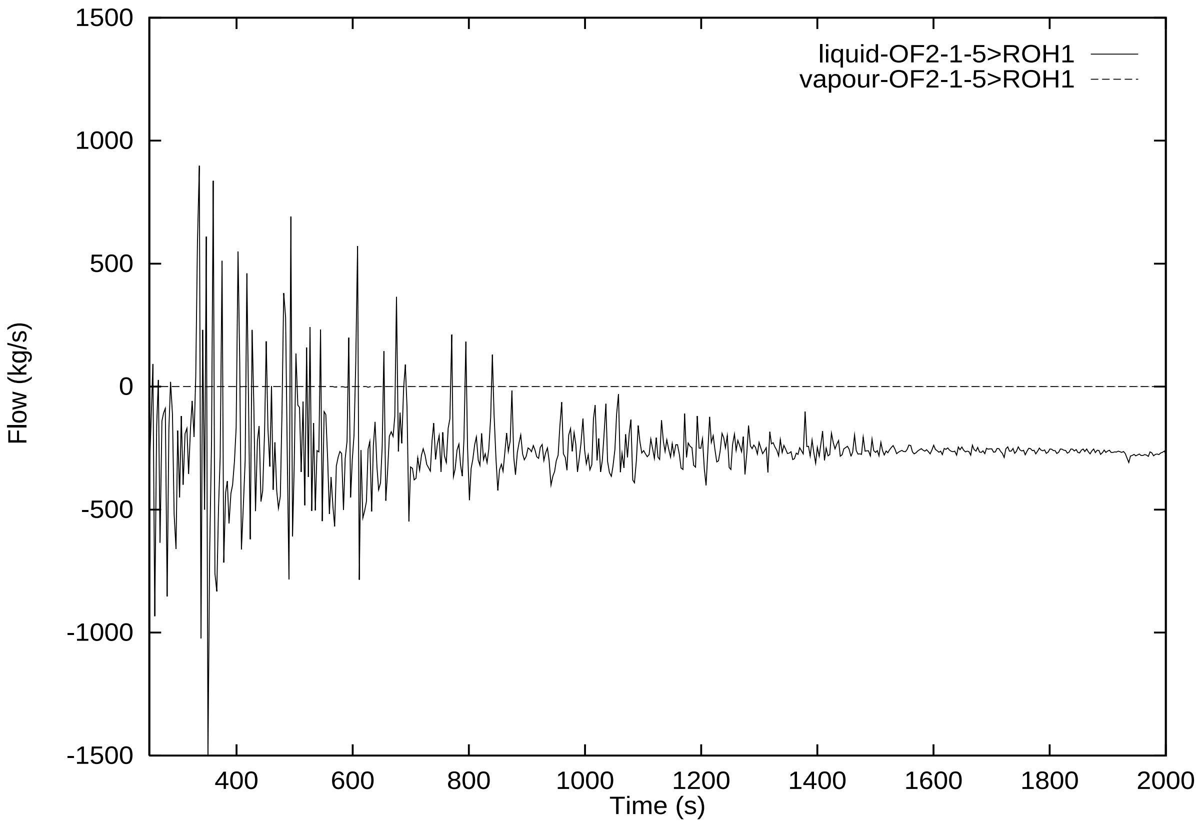


Figure 50 Outlet Header Flows for 25% RIH Break with Subsequent Loss of Class IV Power

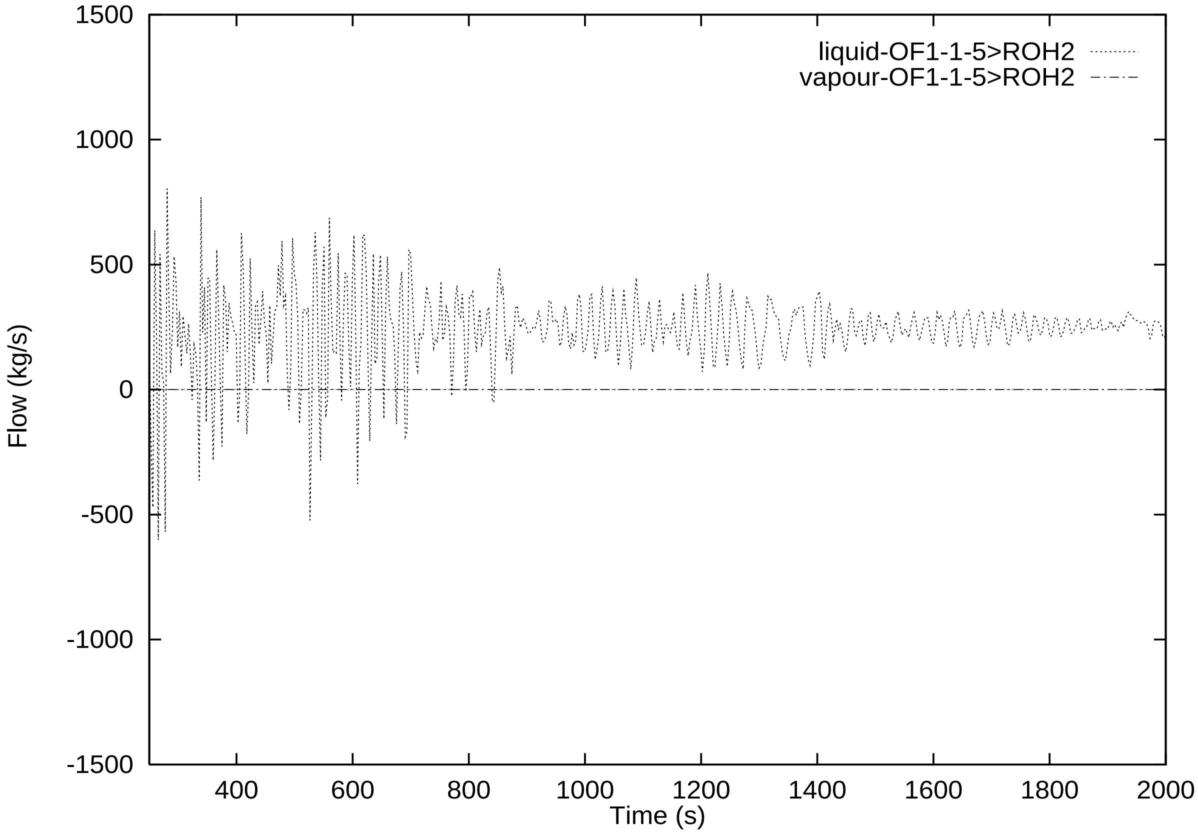


Figure 51 Outlet Header Flows for 25% RIH Break with Subsequent Loss of Class IV Power

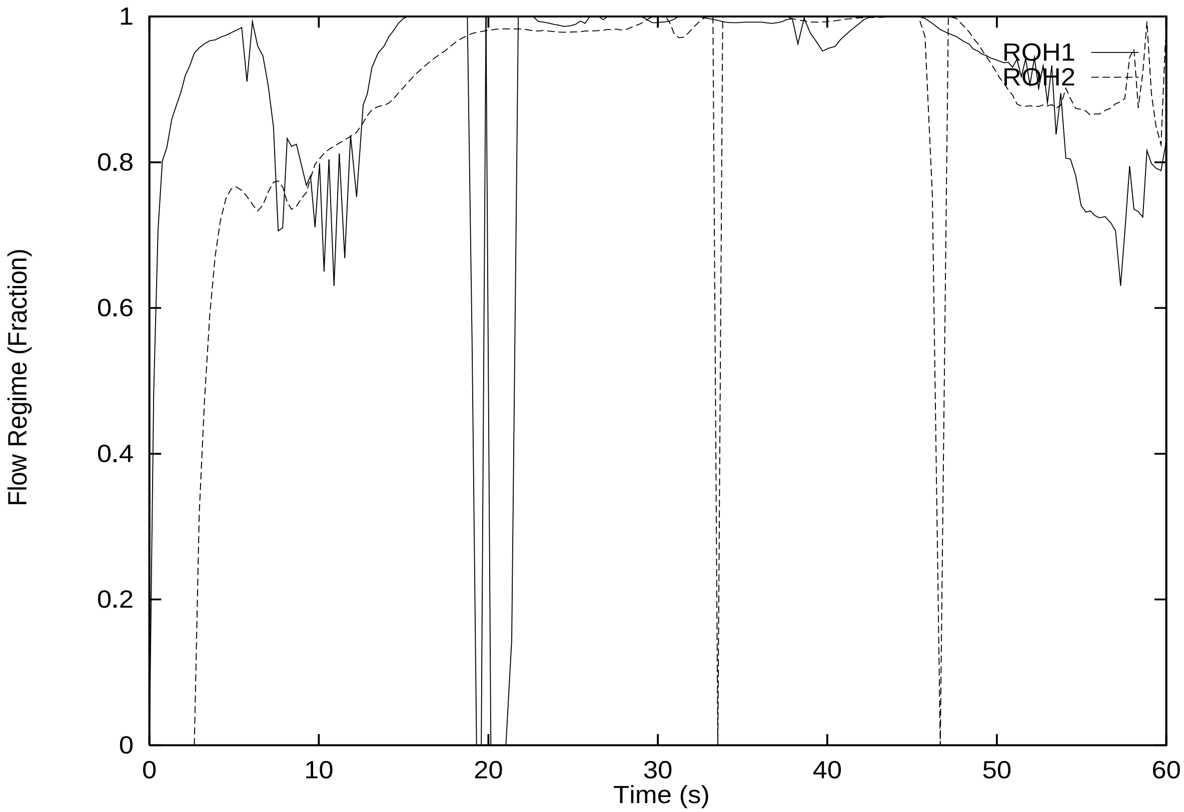


Figure 52 Outlet Header Flow Regime for 25% RIH Break with Subsequent Loss of Class IV Power

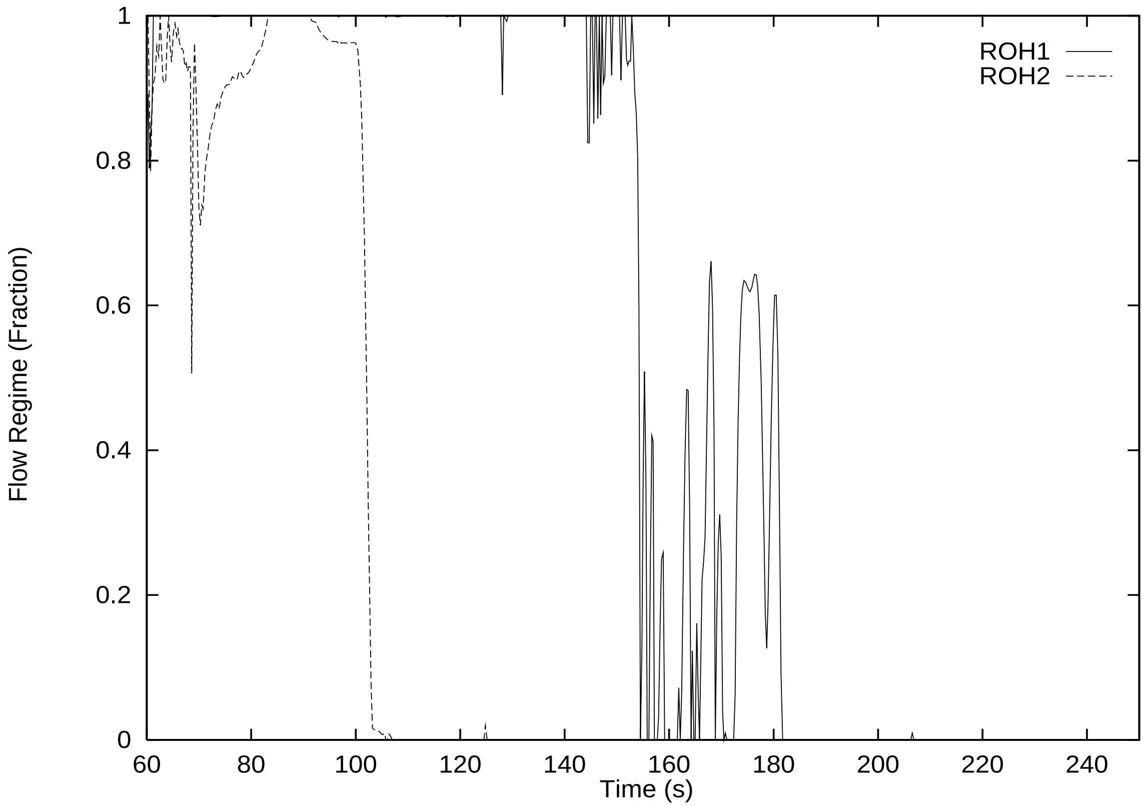


Figure 53 Outlet Header Flow Regime for 25% RIH Break with Subsequent Loss of Class IV Power

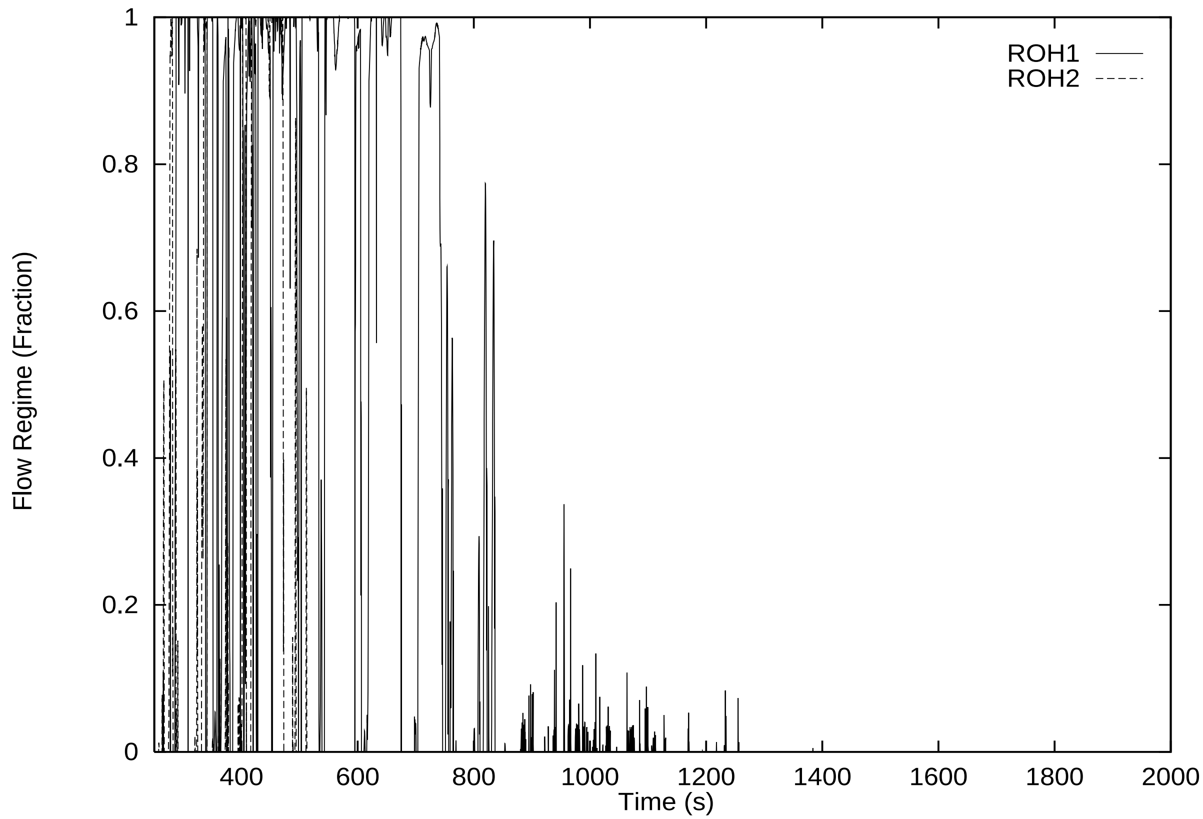


Figure 54 Outlet Header Flow Regime for 25% RIH Break with Subsequent Loss of Class IV Power

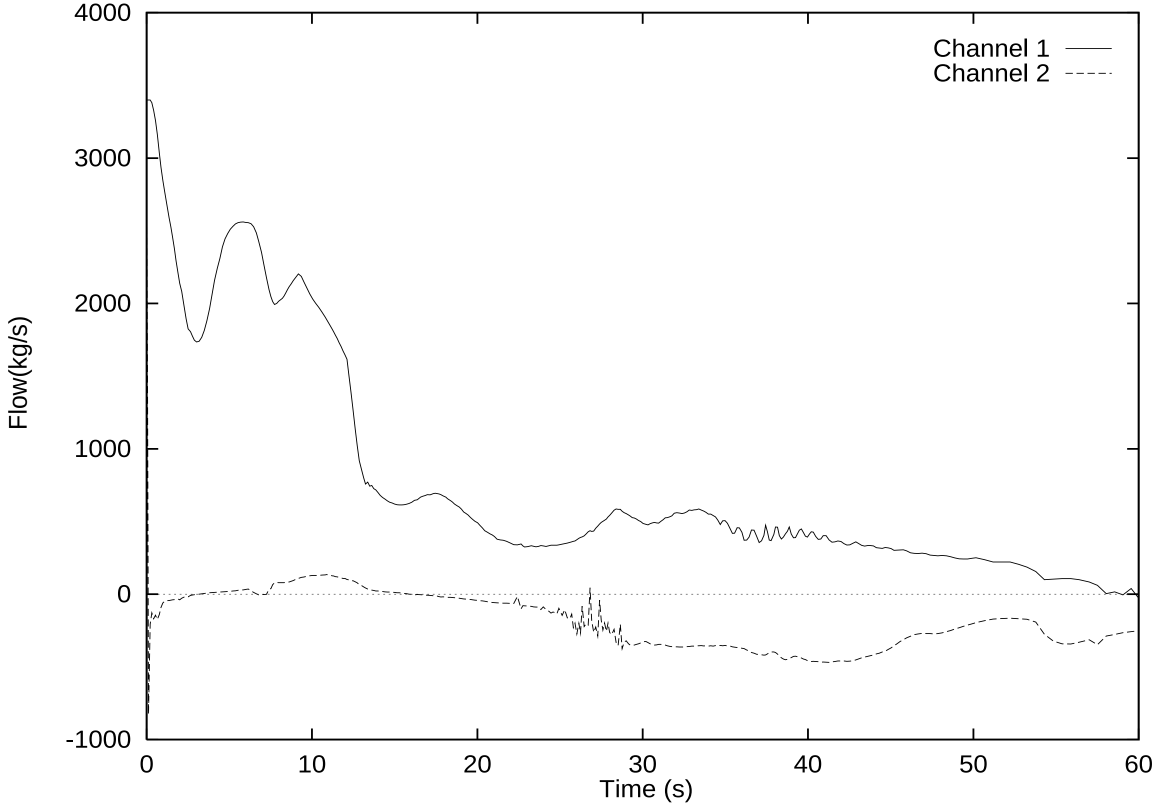


Figure 55 Fuel Channel Flows for 25% RIH Break with Subsequent Loss of Class IV Power

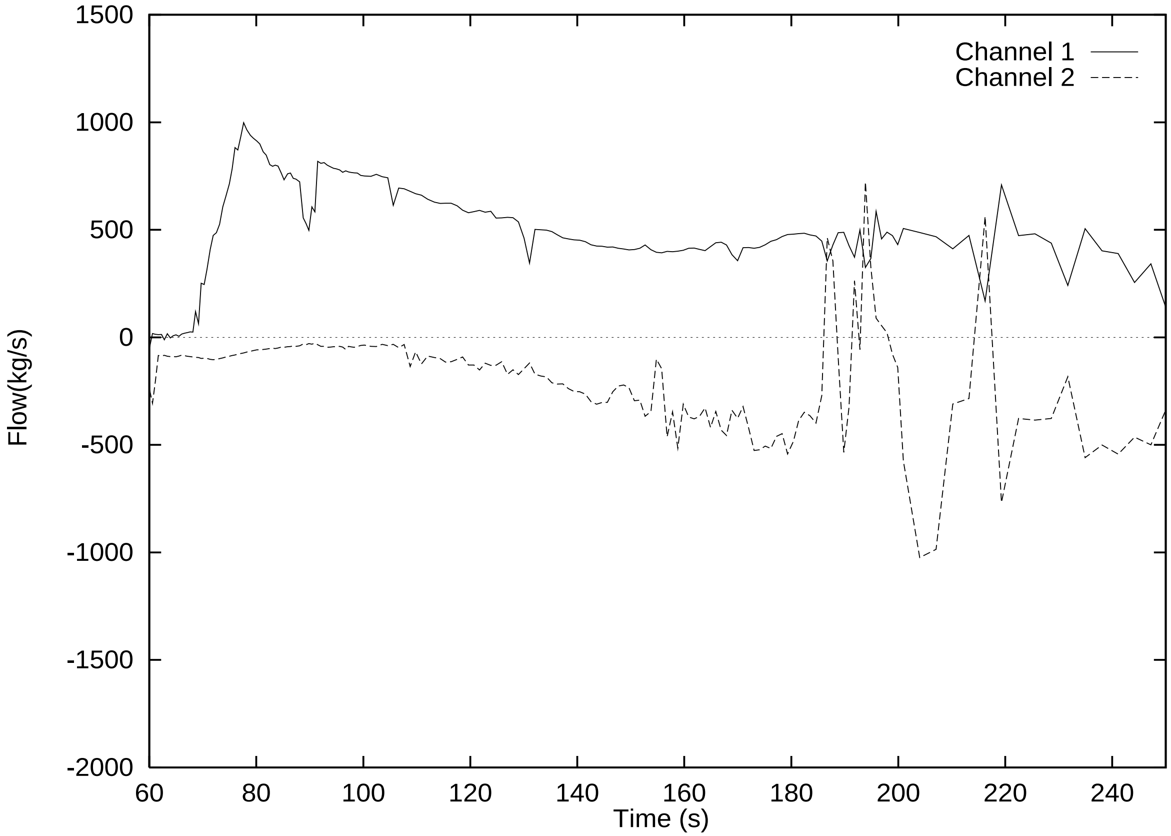


Figure 56 Fuel Channel Flows for 25% RIH Break with Subsequent Loss of Class IV Power

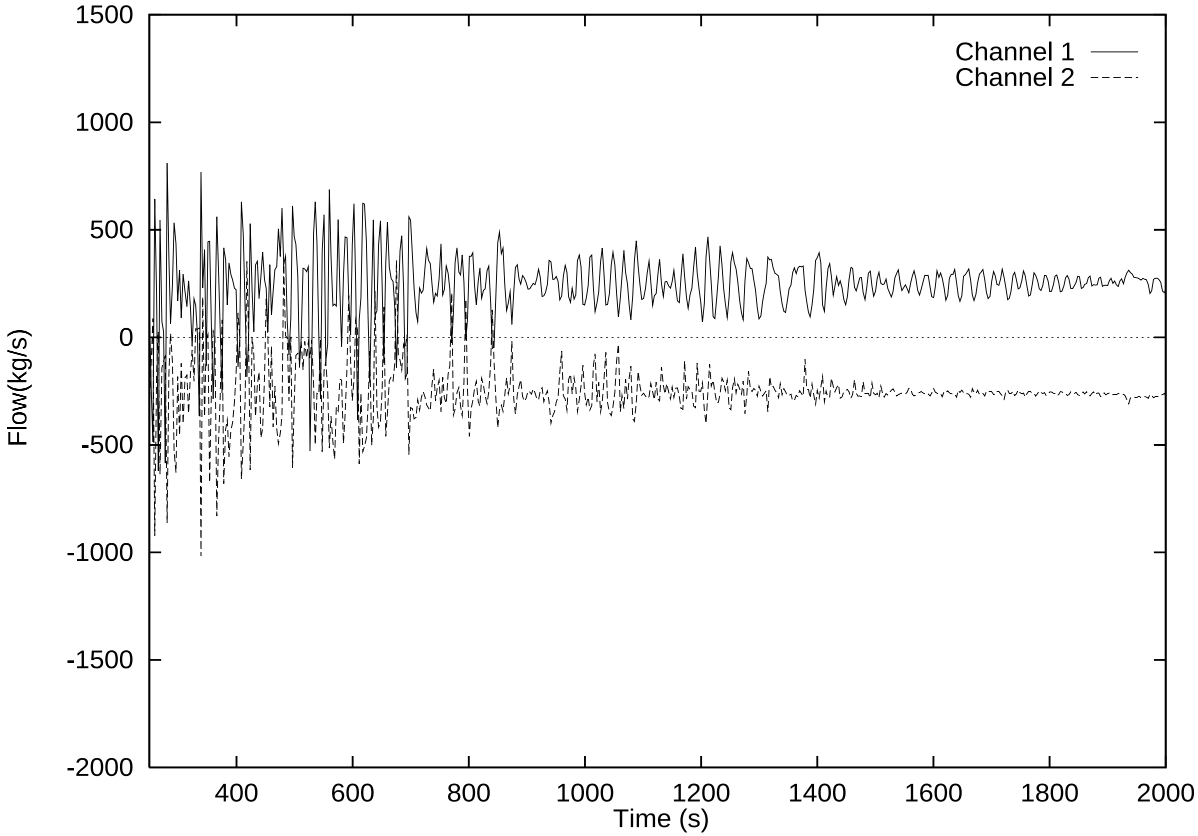


Figure 57 Fuel Channel Flows for 25% RIH Break with Subsequent Loss of Class IV Power

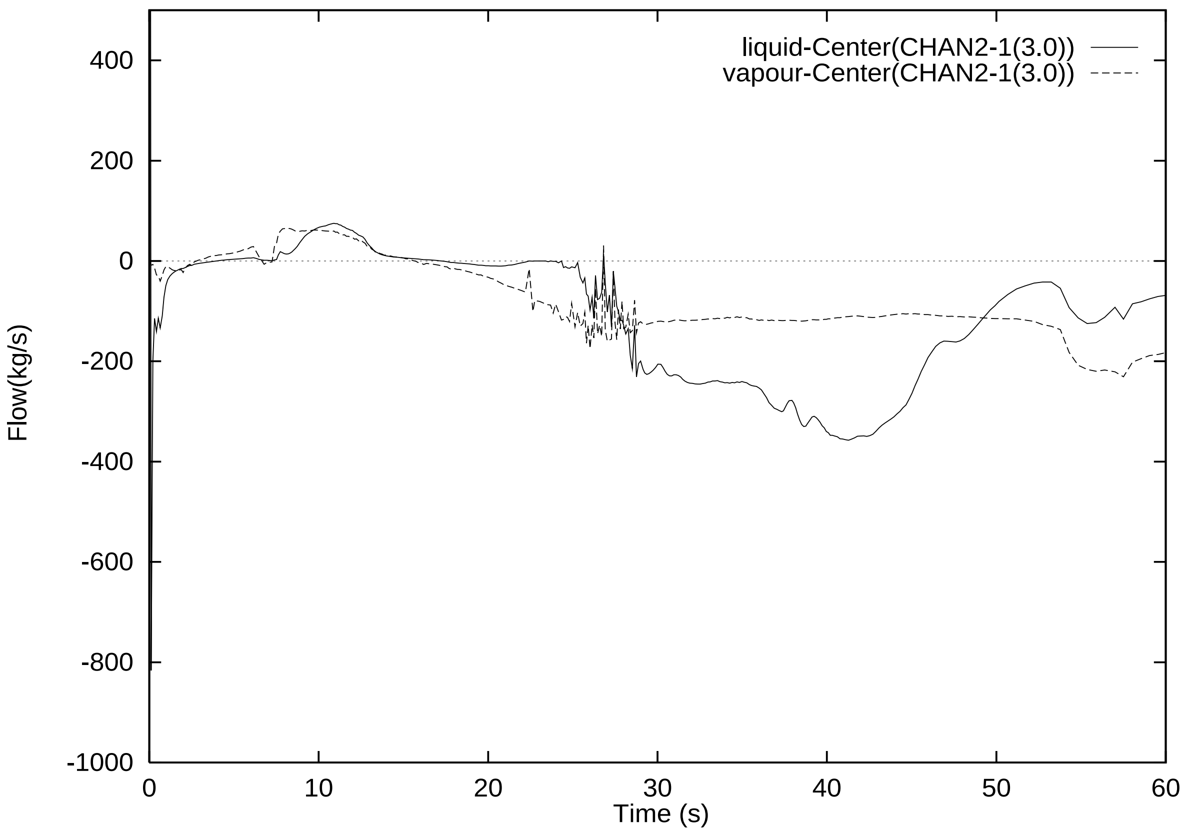


Figure 58 Fuel Channel 2 Centre Flow for 25% RIH Break with Subsequent Loss of Class IV Power

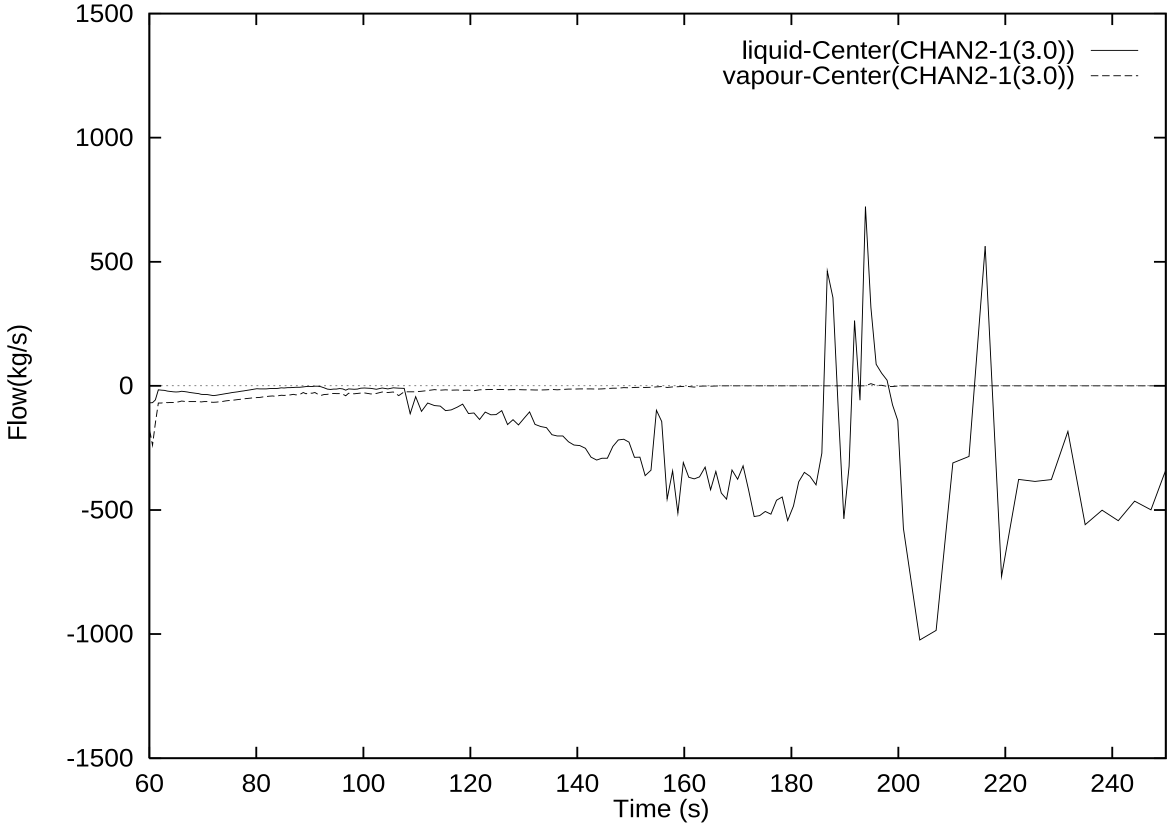


Figure 59 Fuel Channel 2 Centre Flow for 25% RIH Break with Subsequent Loss of Class IV Power

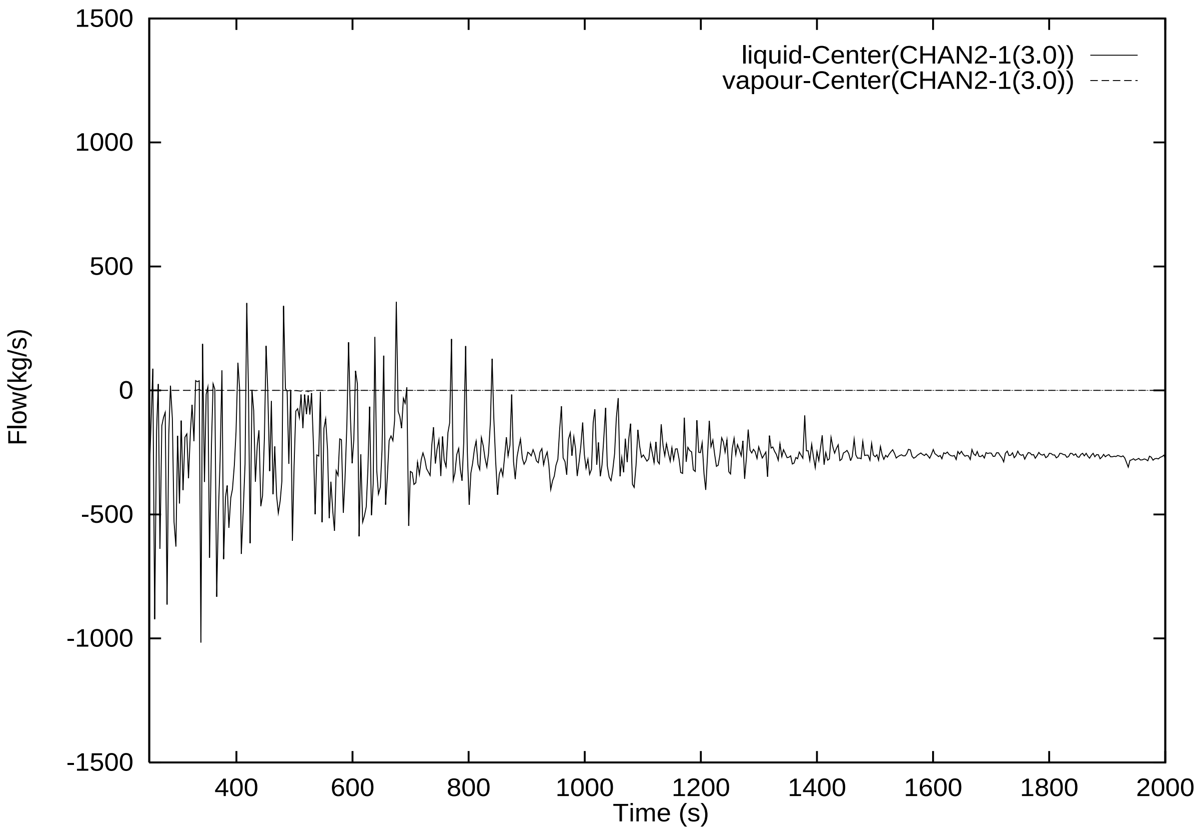


Figure 60 Fuel Channel 2 Centre Flow for 25% RIH Break with Subsequent Loss of Class IV Power

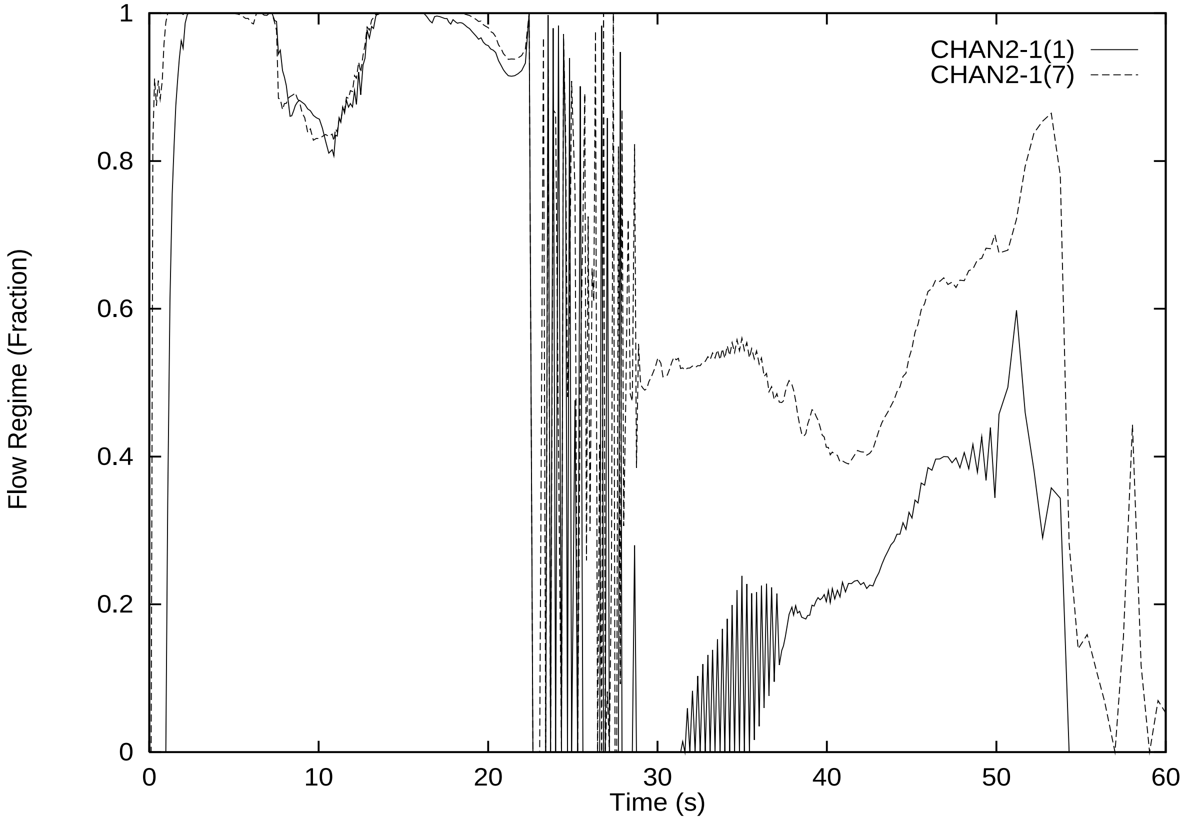


Figure 61 Fuel Channel 2 Flow Regime for 25% RIH Break with Subsequent Loss of Class IV Power

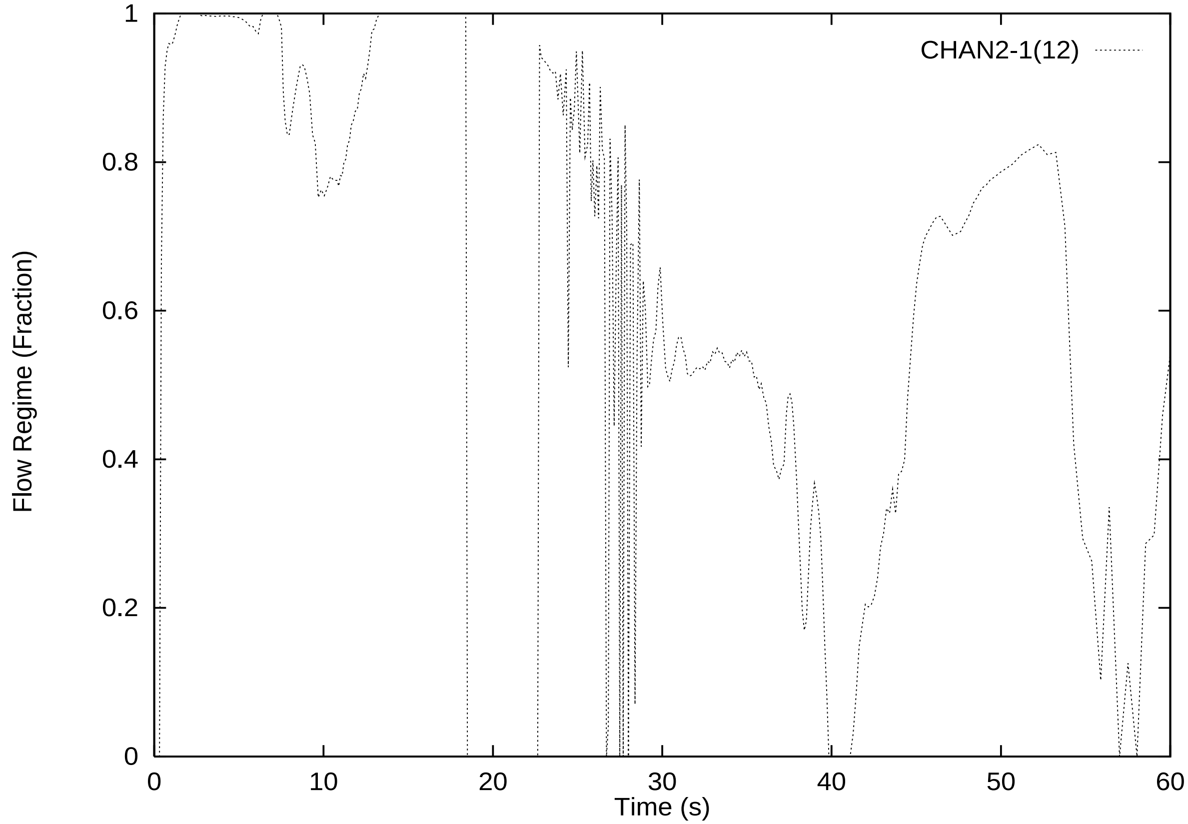


Figure 62 Fuel Channel 2 Flow Regime for 25% RIH Break with Subsequent Loss of Class IV Power

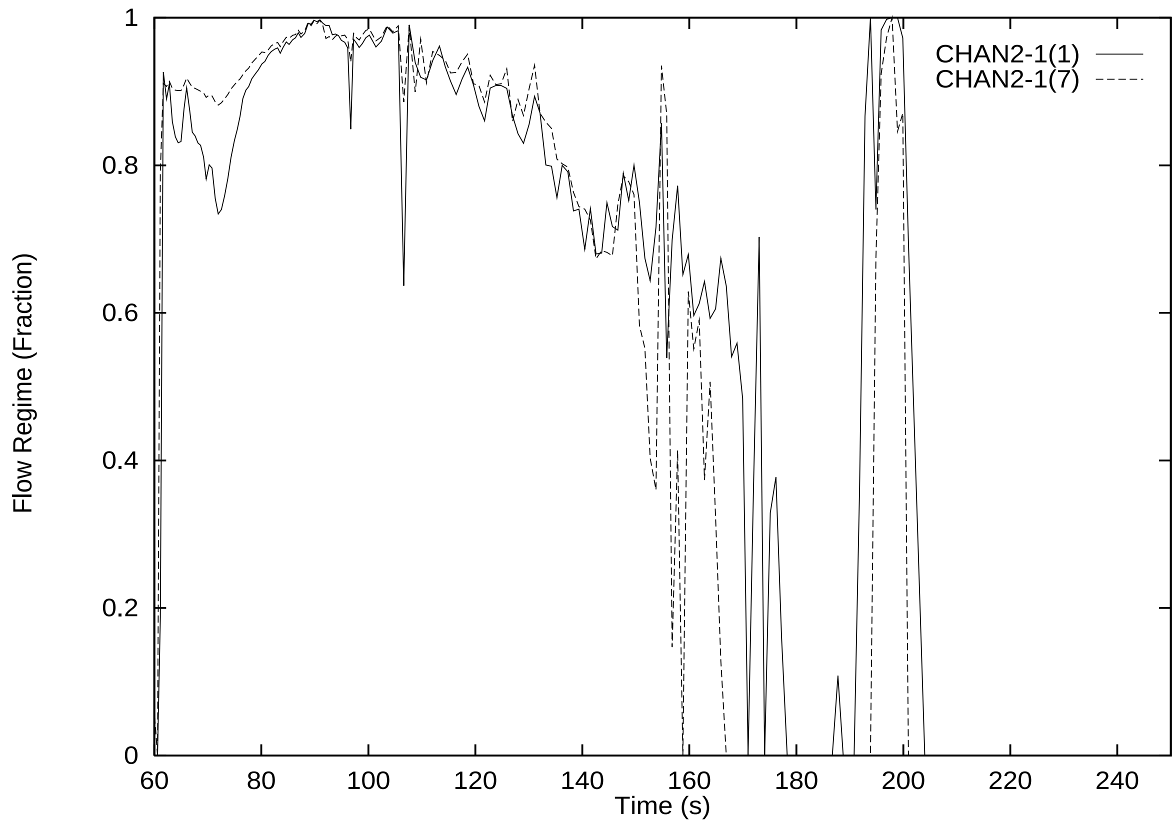


Figure 63 Fuel Channel 2 Flow Regime for 25% RIH Break with Subsequent Loss of Class IV Power

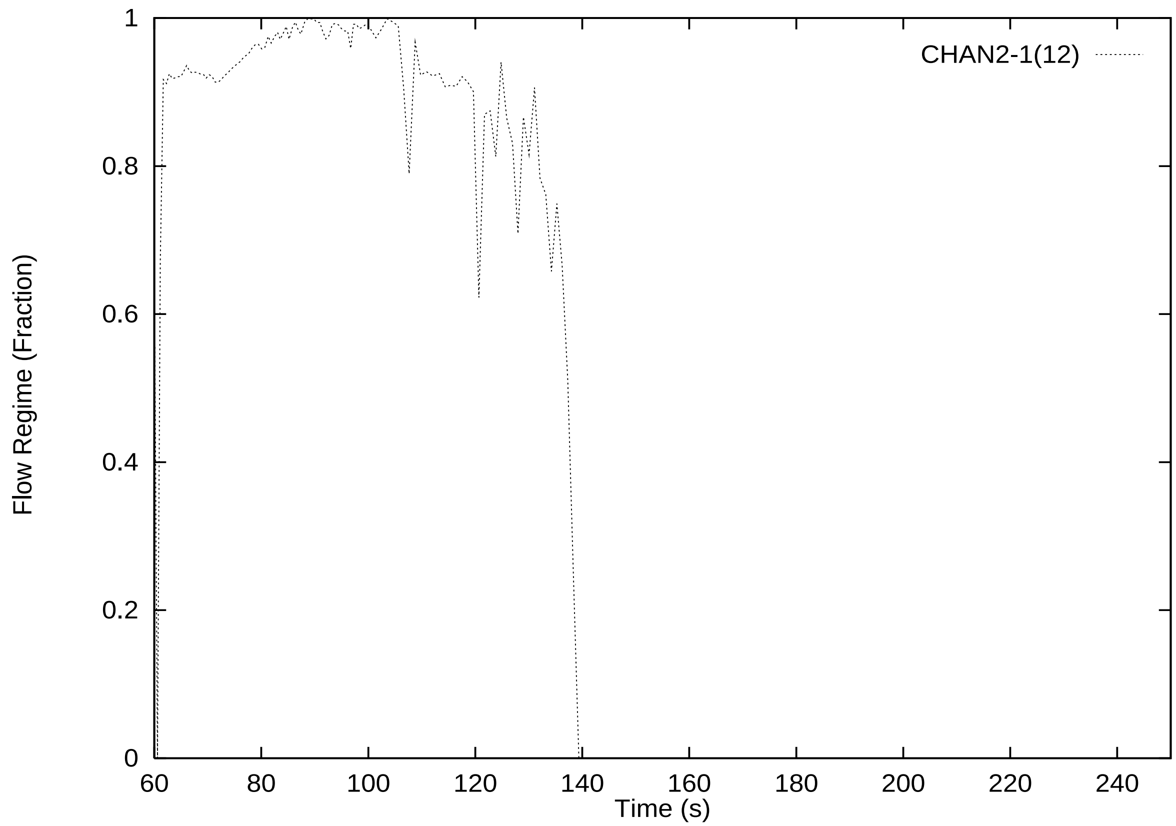


Figure 64 Fuel Channel 2 Flow Regime for 25% RIH Break with Subsequent Loss of Class IV Power

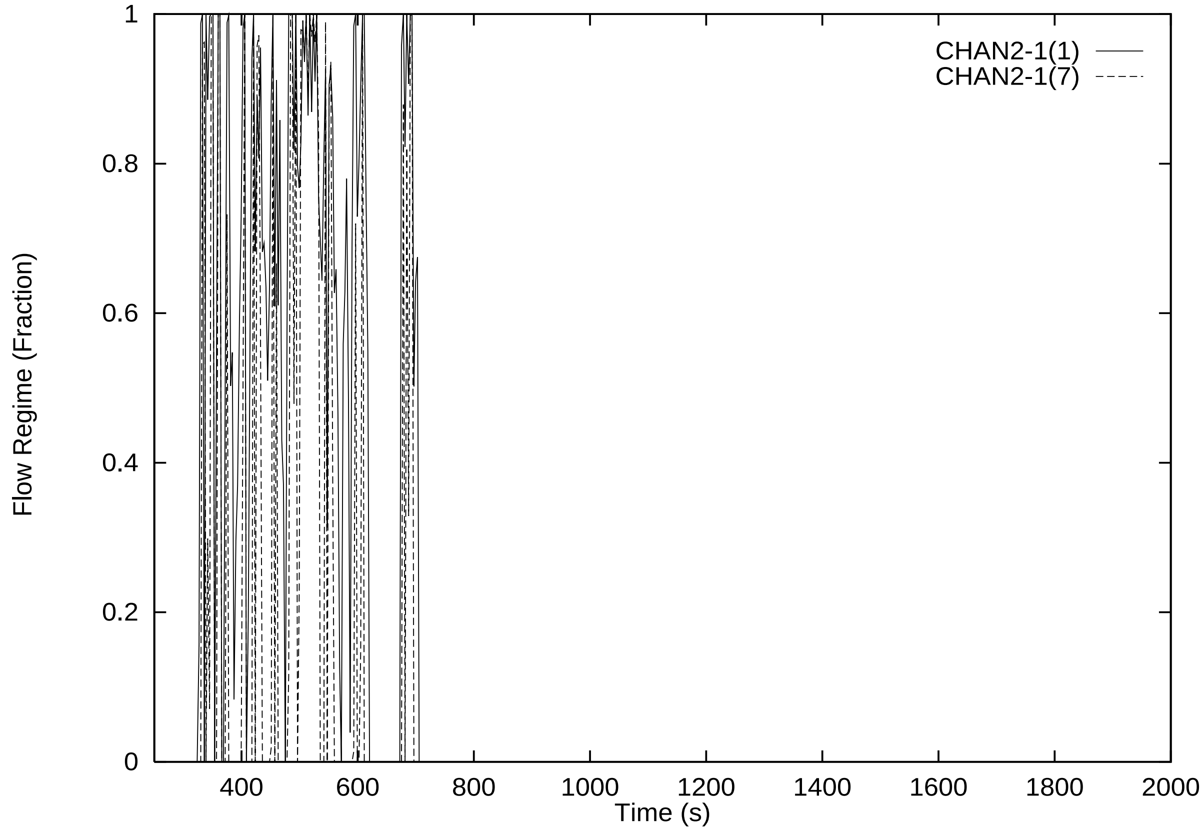


Figure 65 Fuel Channel 2 Flow Regime for 25% RIH Break with Subsequent Loss of Class IV Power

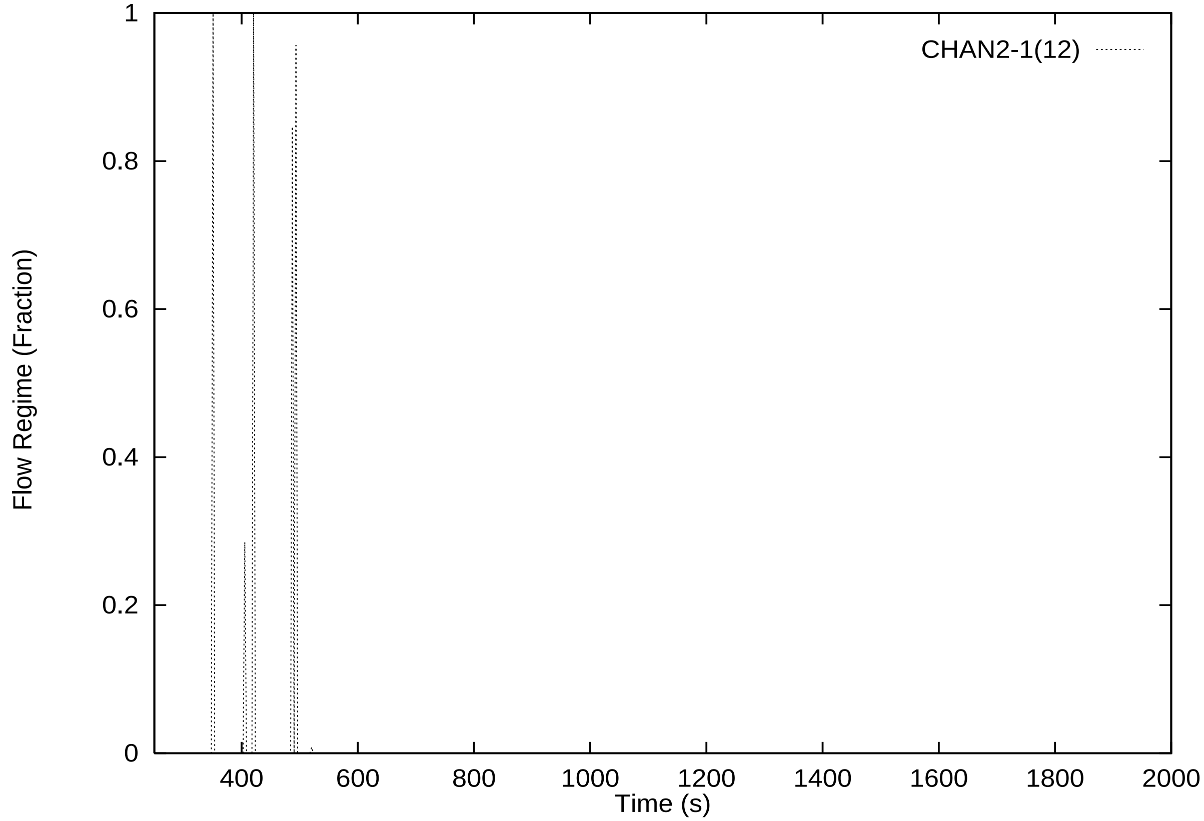


Figure 66 Fuel Channel 2 Flow Regime for 25% RIH Break with Subsequent Loss of Class IV Power

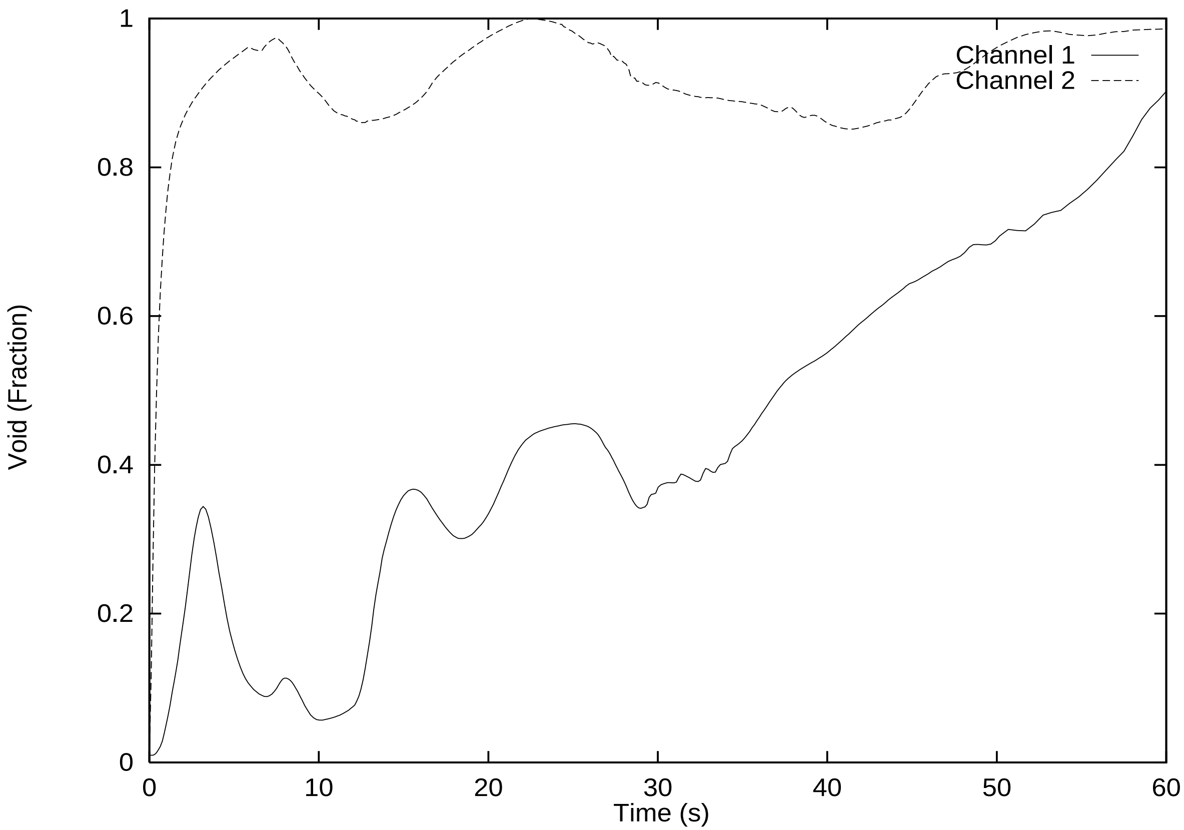


Figure 67 Fuel Channel Integrated Voids for 25% RIH Break with Subsequent Loss of Class IV Power

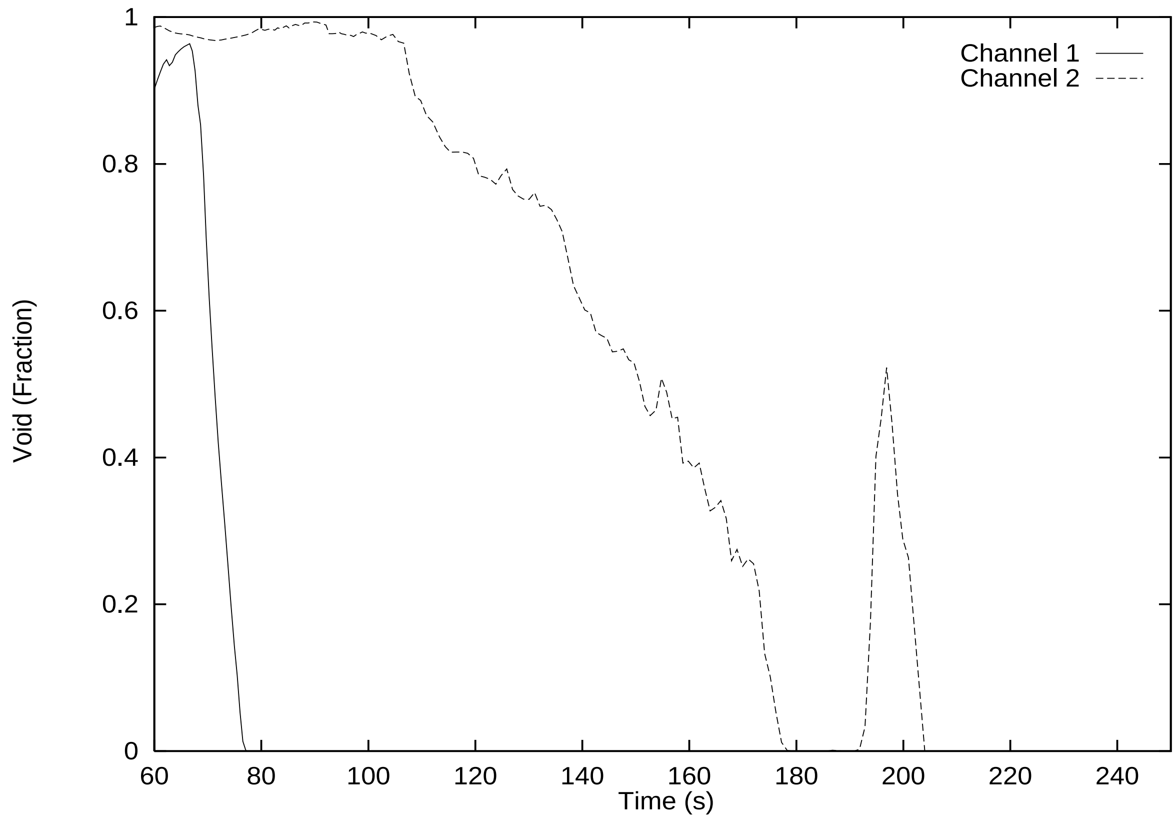


Figure 68 Fuel Channel Integrated Voids for 25% RIH Break with Subsequent Loss of Class IV Power

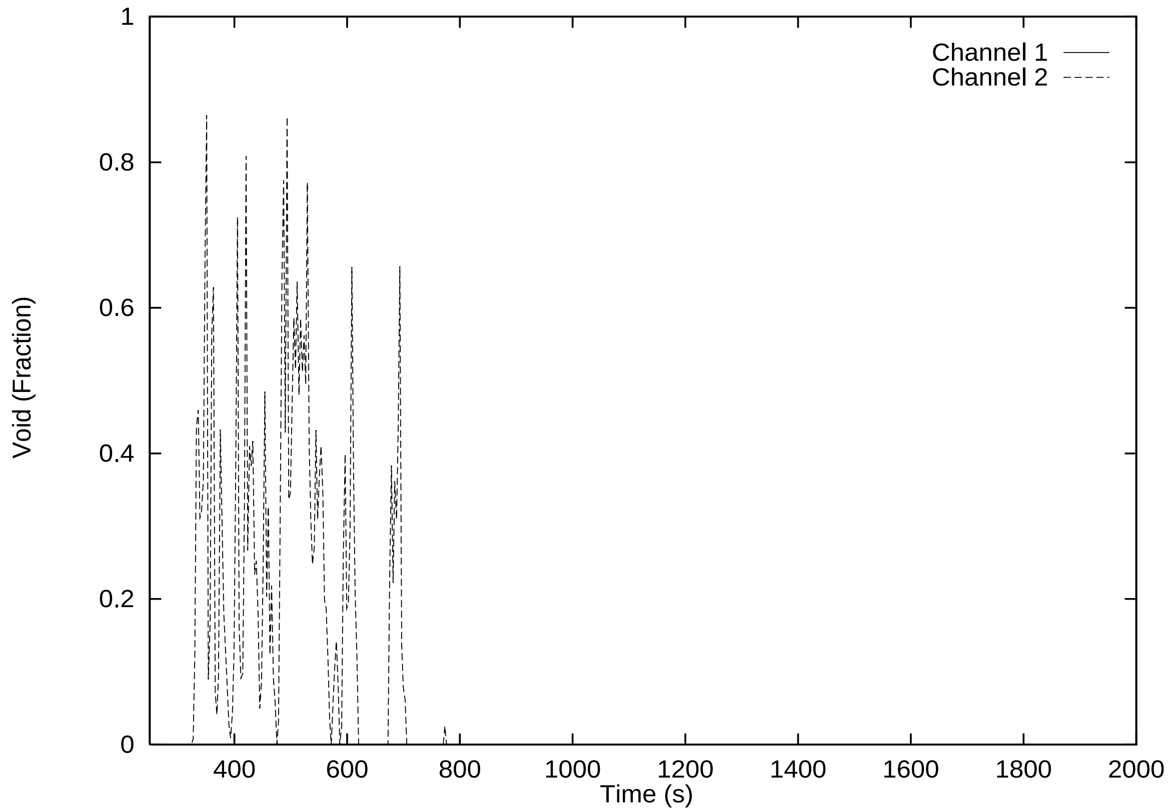


Figure 69 Fuel Channel Integrated Voids for 25% RIH Break with Subsequent Loss of Class IV Power

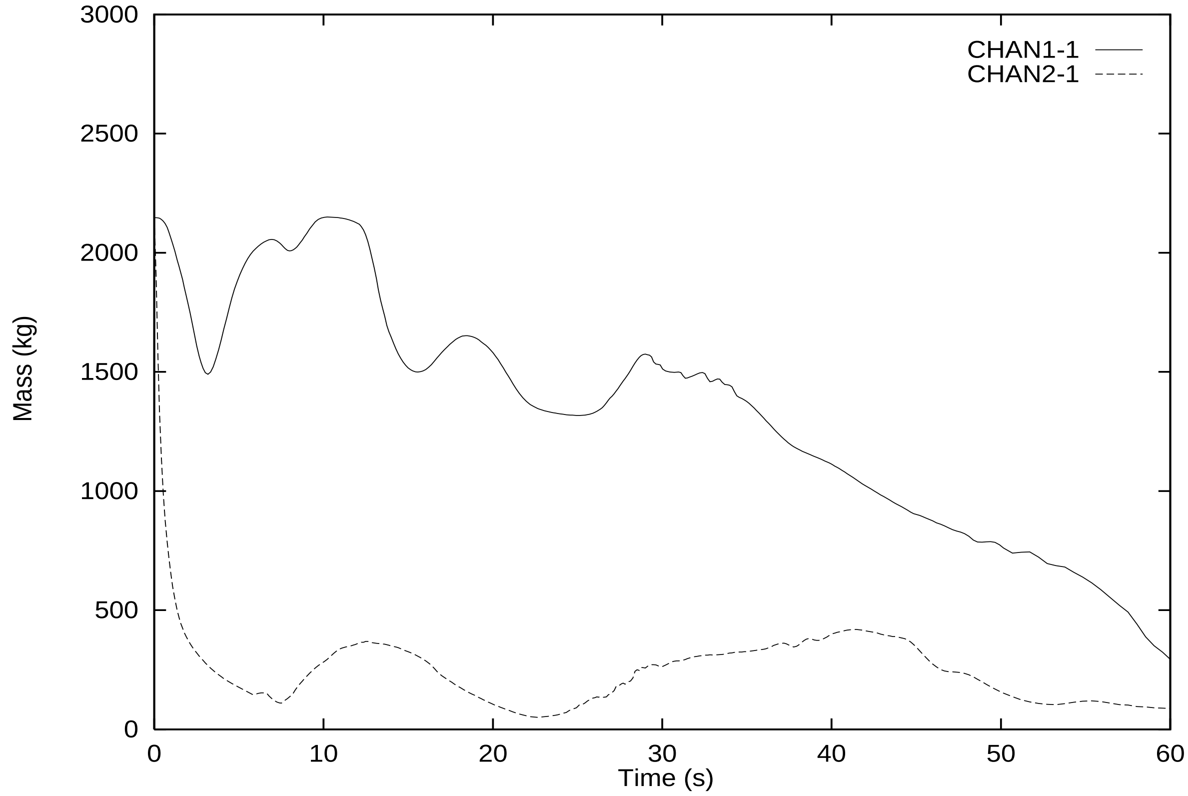


Figure 70 Fuel Channel Integrated Mass for 25% RIH Break with Subsequent Loss of Class IV Power

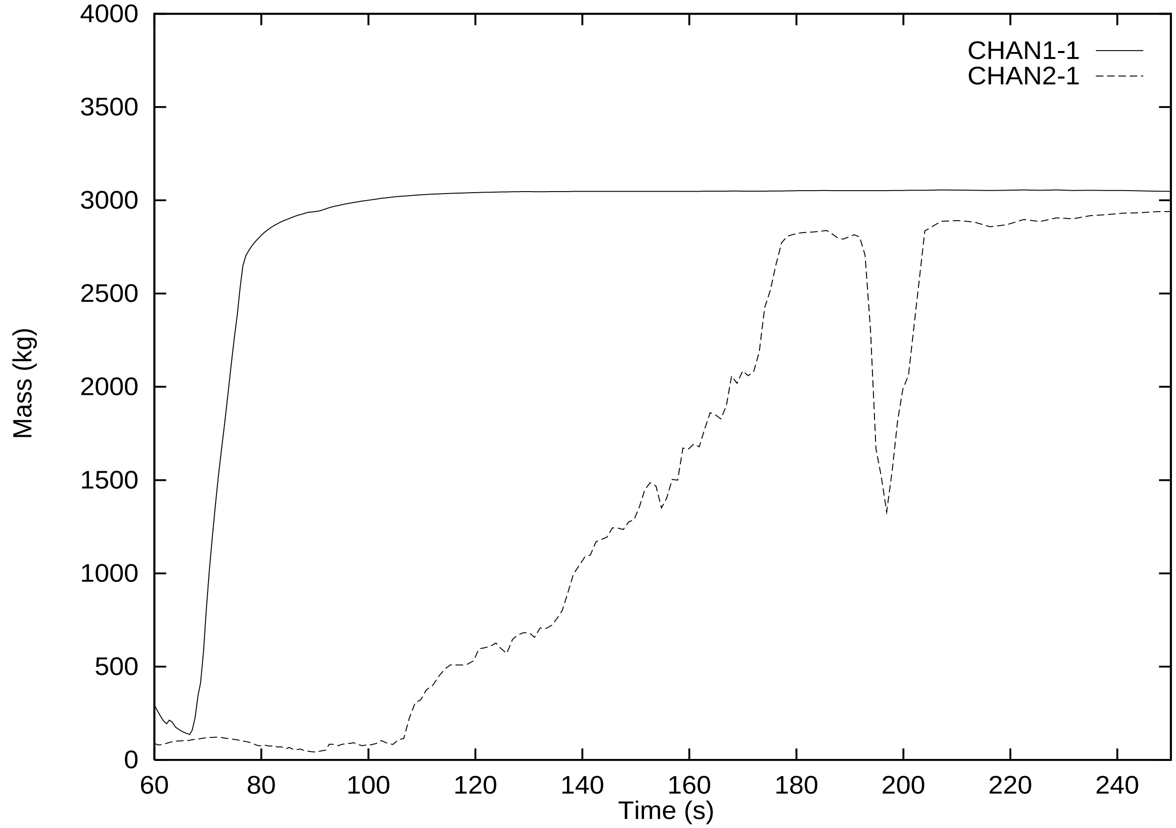


Figure 71 Fuel Channel Integrated Mass for 25% RIH Break with Subsequent Loss of Class IV Power

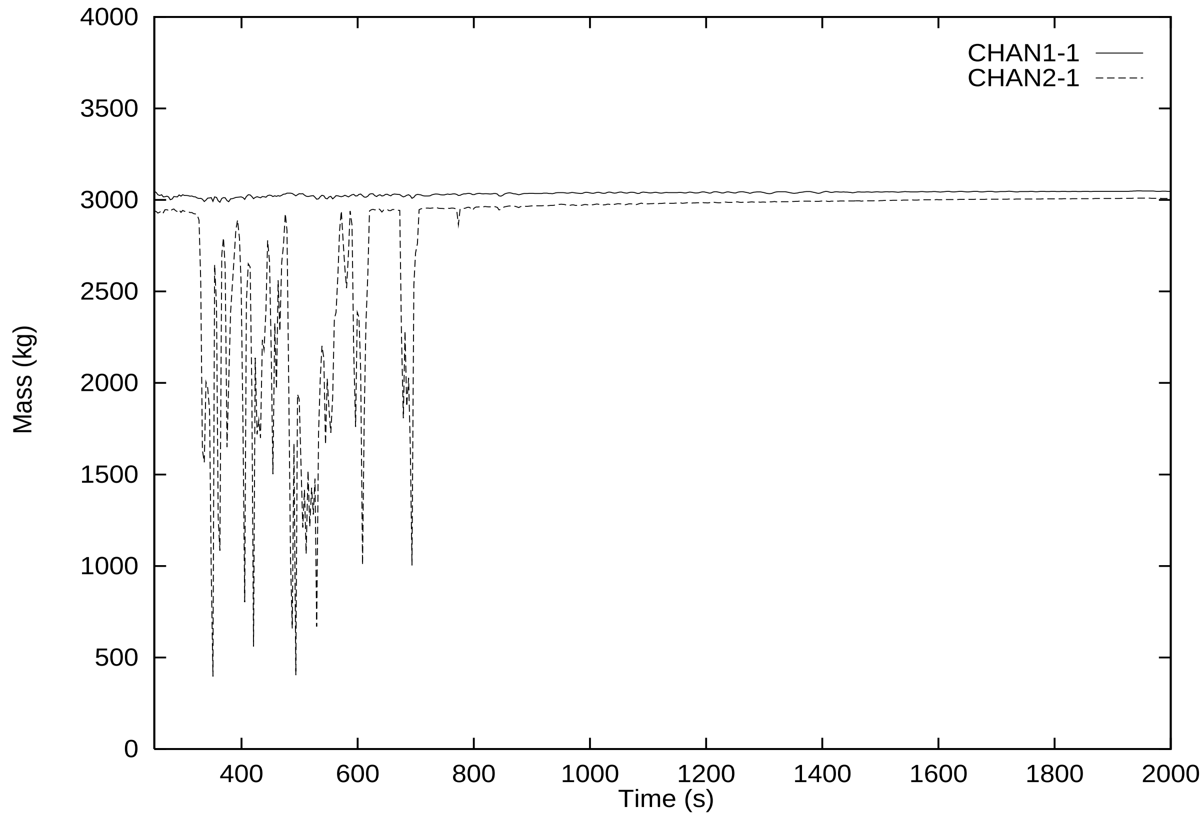


Figure 72 Fuel Channel Integrated Mass for 25% RIH Break with Subsequent Loss of Class IV Power

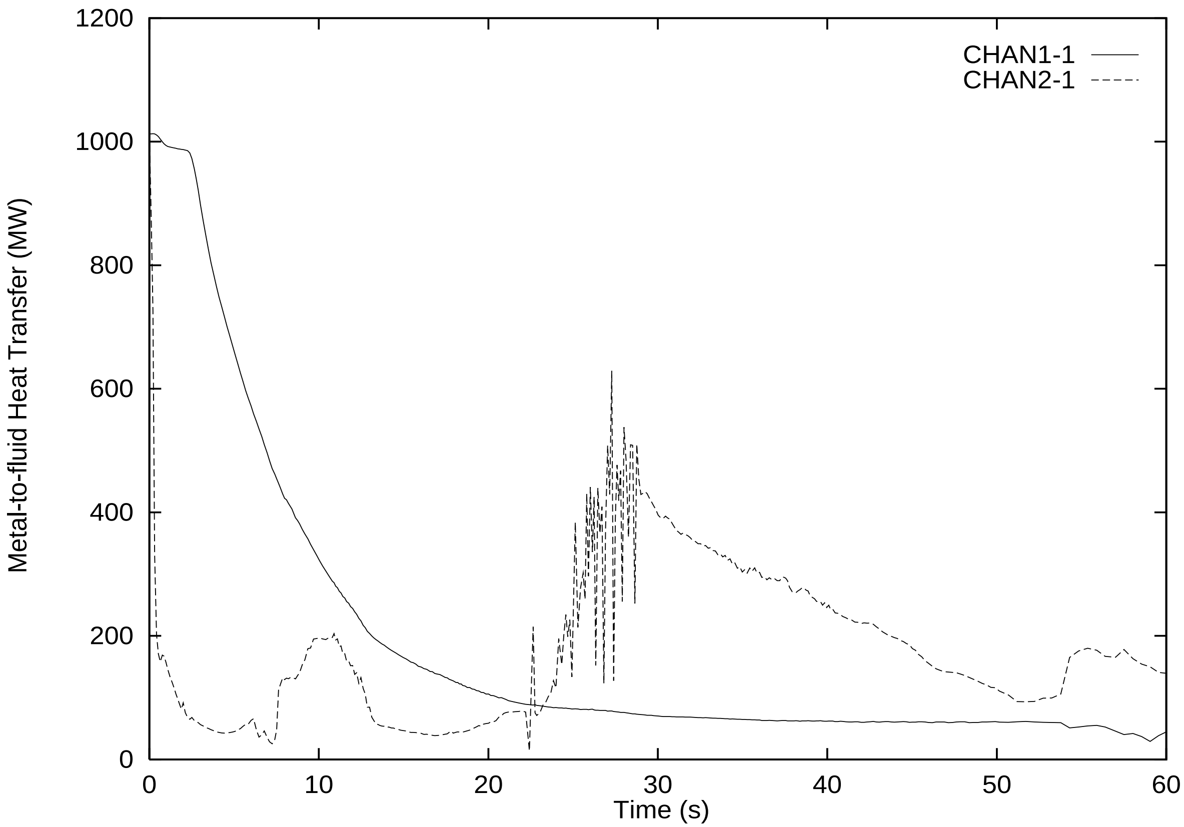


Figure 73 Stored Energy Release in Fuel Channels for 25% RIH Break with Subsequent Loss of Class IV Power

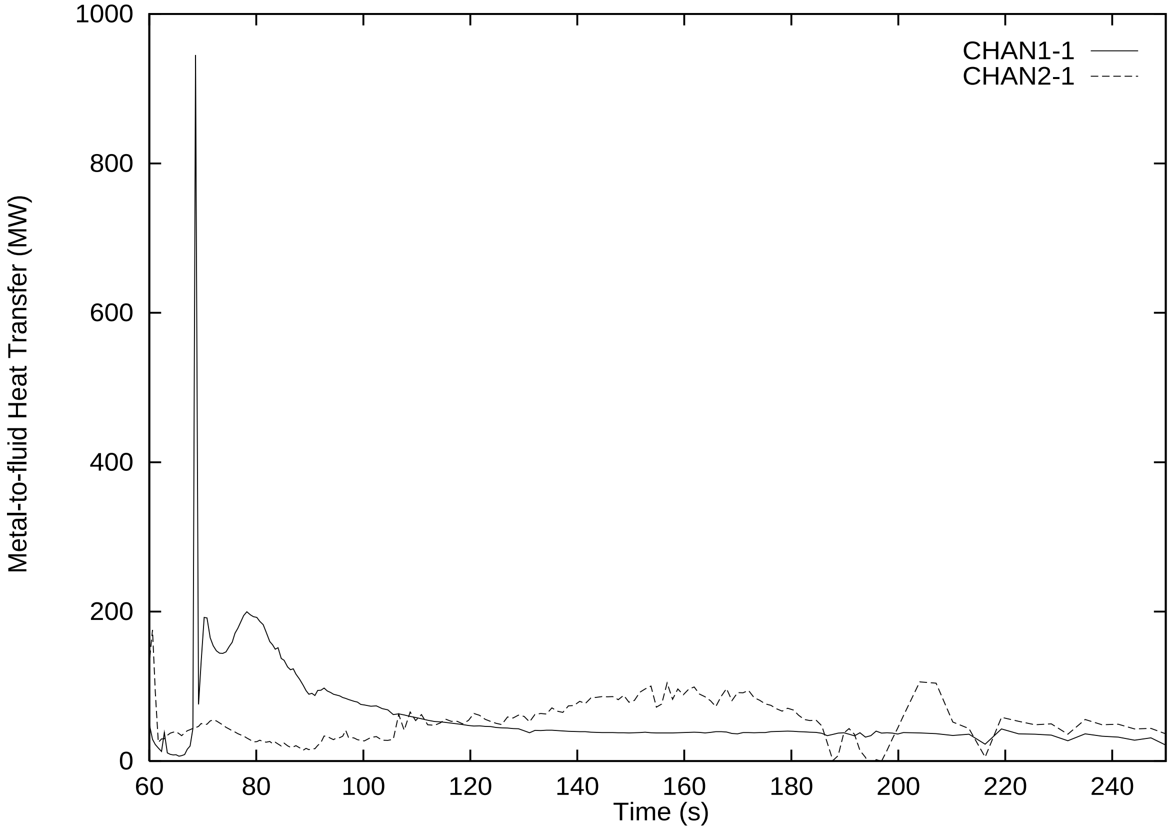


Figure 74 Stored Energy Release in Fuel Channels for 25% RIH Break with Subsequent Loss of Class IV Power

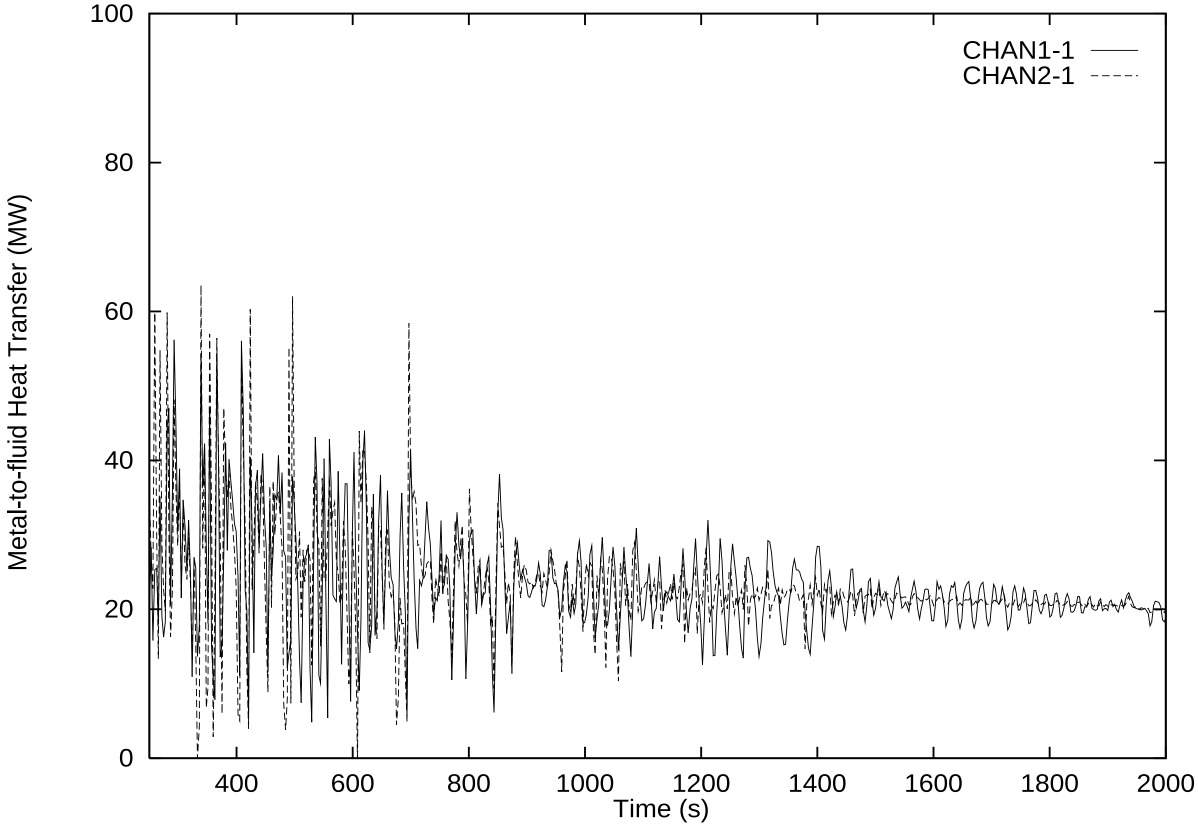


Figure 75 Stored Energy Release in Fuel Channels for 25% RIH Break with Subsequent Loss of Class IV Power

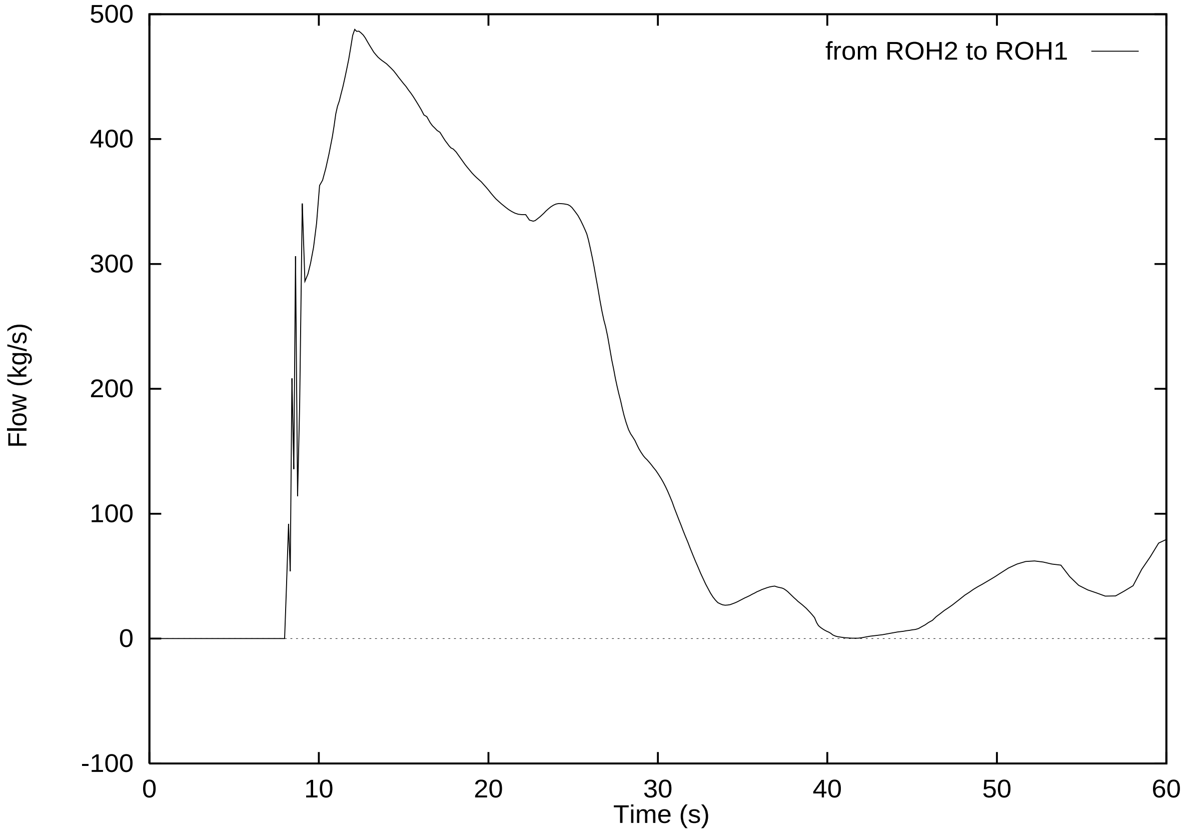


Figure 76 Large Header Interconnect Flow for 25% RIH Break with Subsequent Loss of Class IV Power

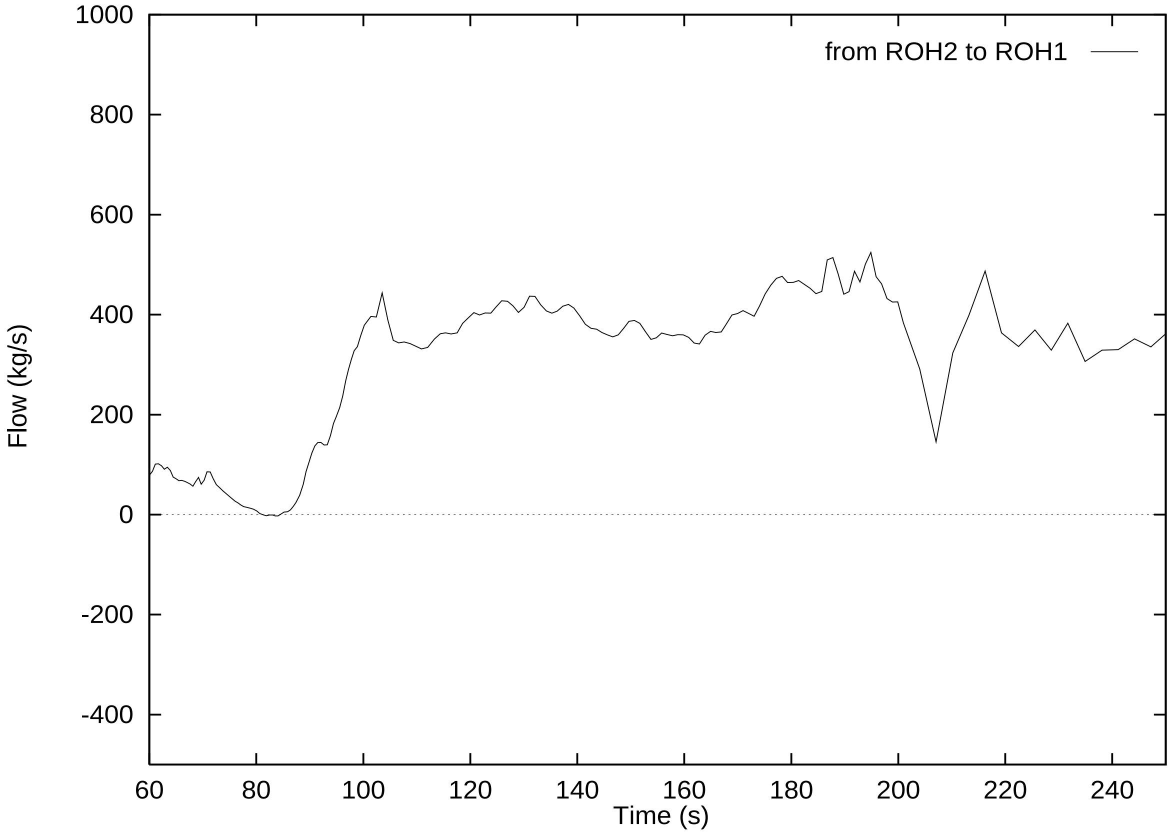


Figure 77 Large Outlet Header Interconnect Flow for 25% RIH Break with Subsequent Loss of Class IV Power

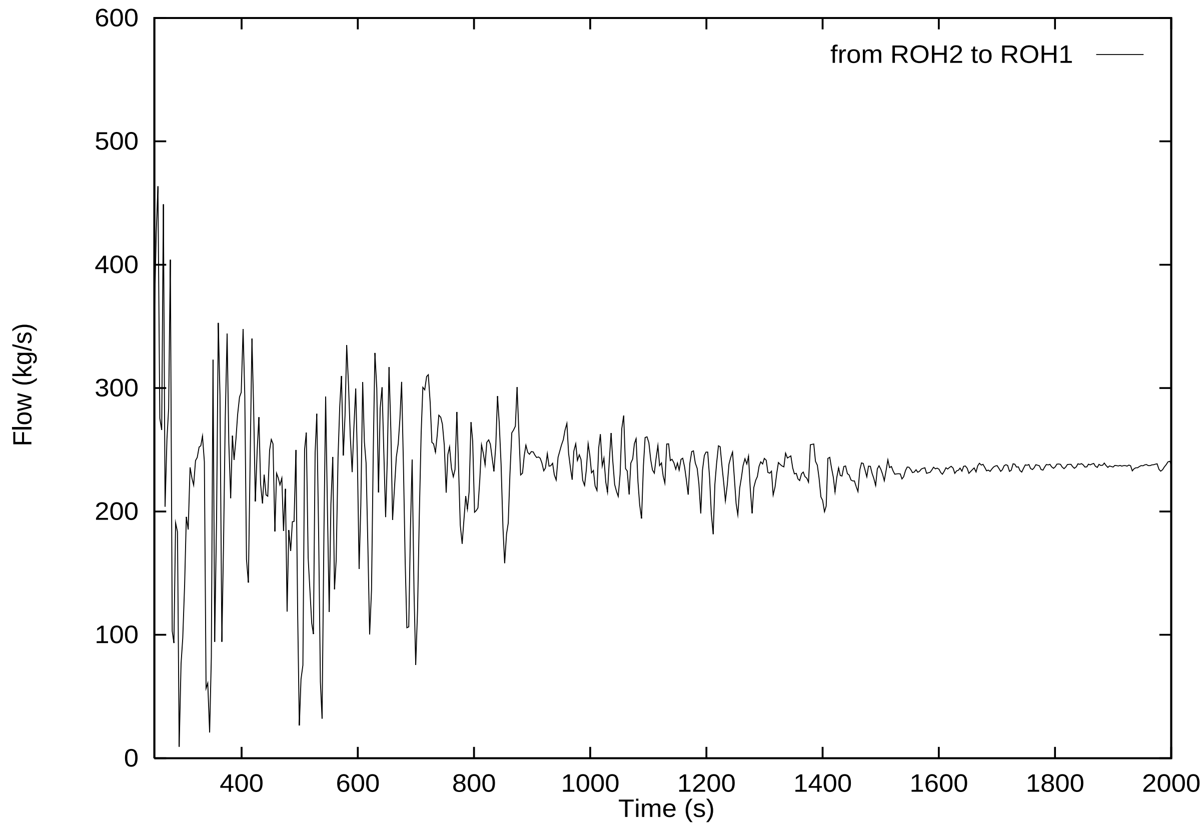


Figure 78 Large Outlet Header Interconnect Flow for 25% RIH Break with Subsequent Loss of Class IV Power

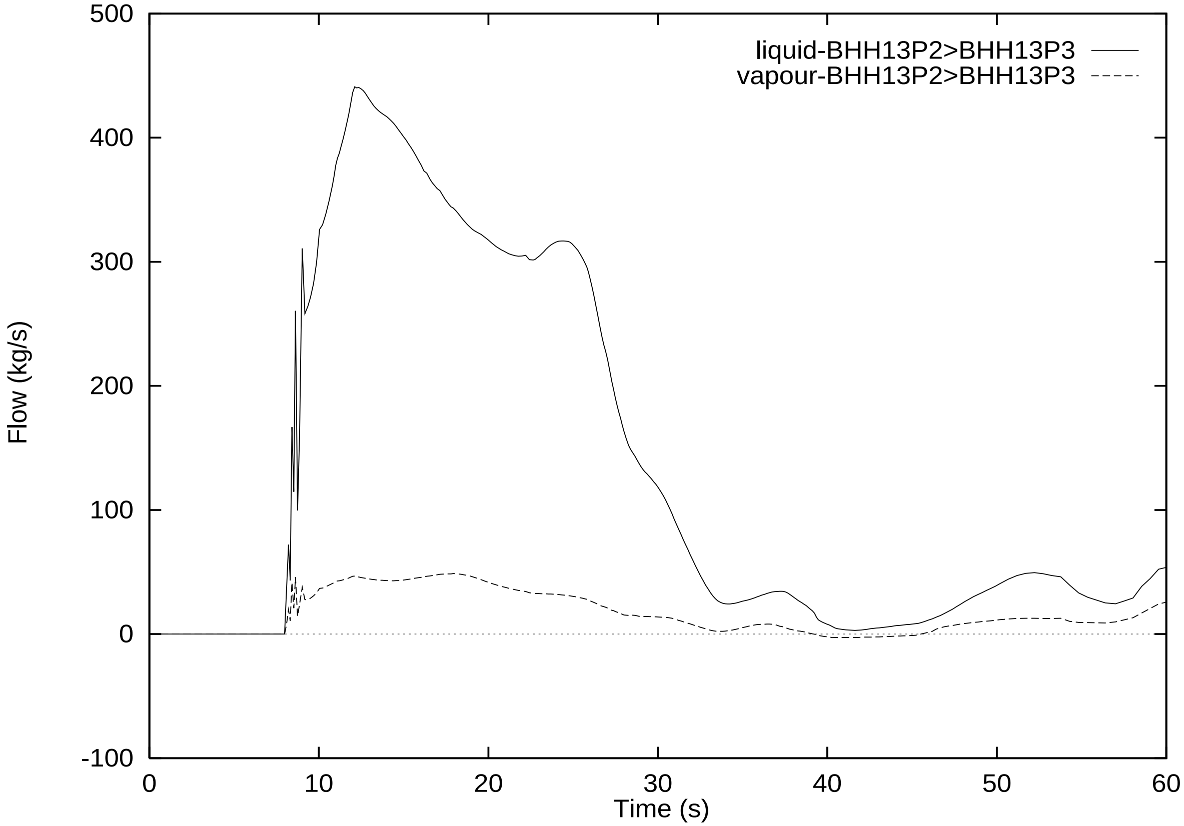


Figure 79 Large Outlet Header Interconnect Flow for 25% RIH Break with Subsequent Loss of Class IV Power

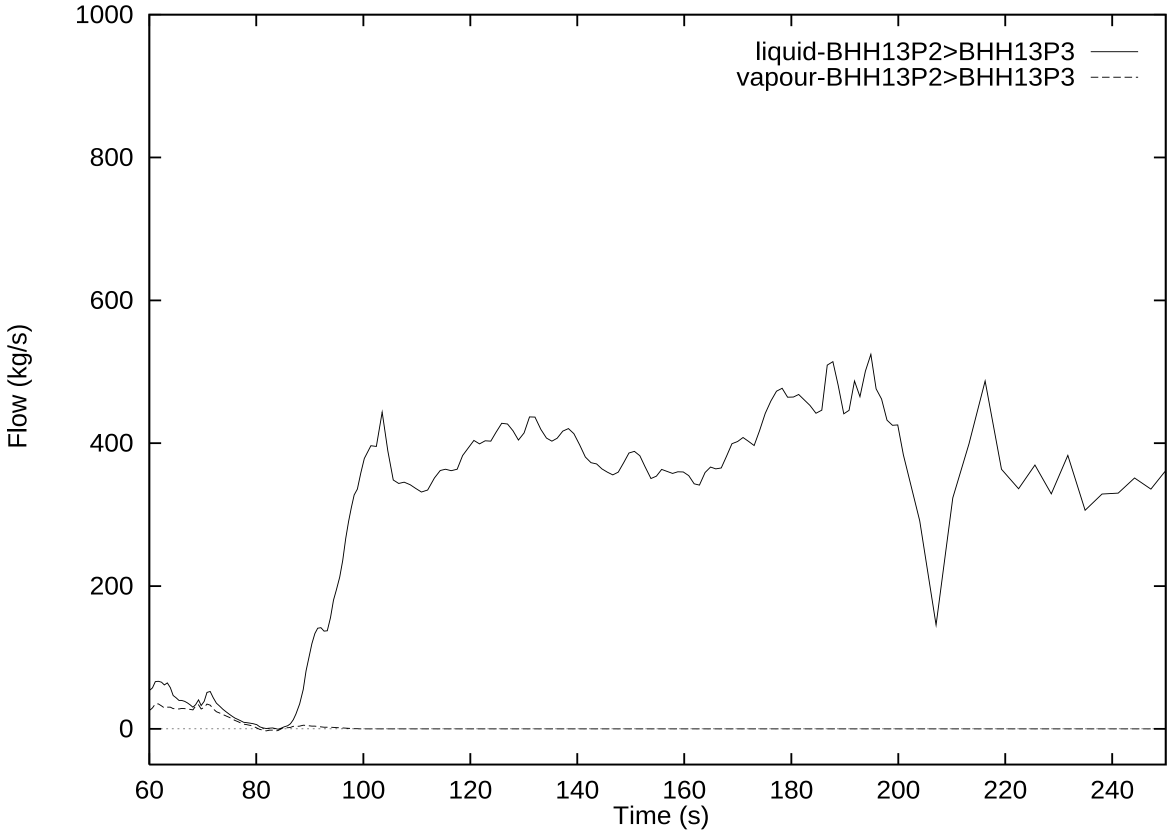


Figure 80 Large Outlet Header Interconnect Flow for 25% RIH Break with Subsequent Loss of Class IV Power

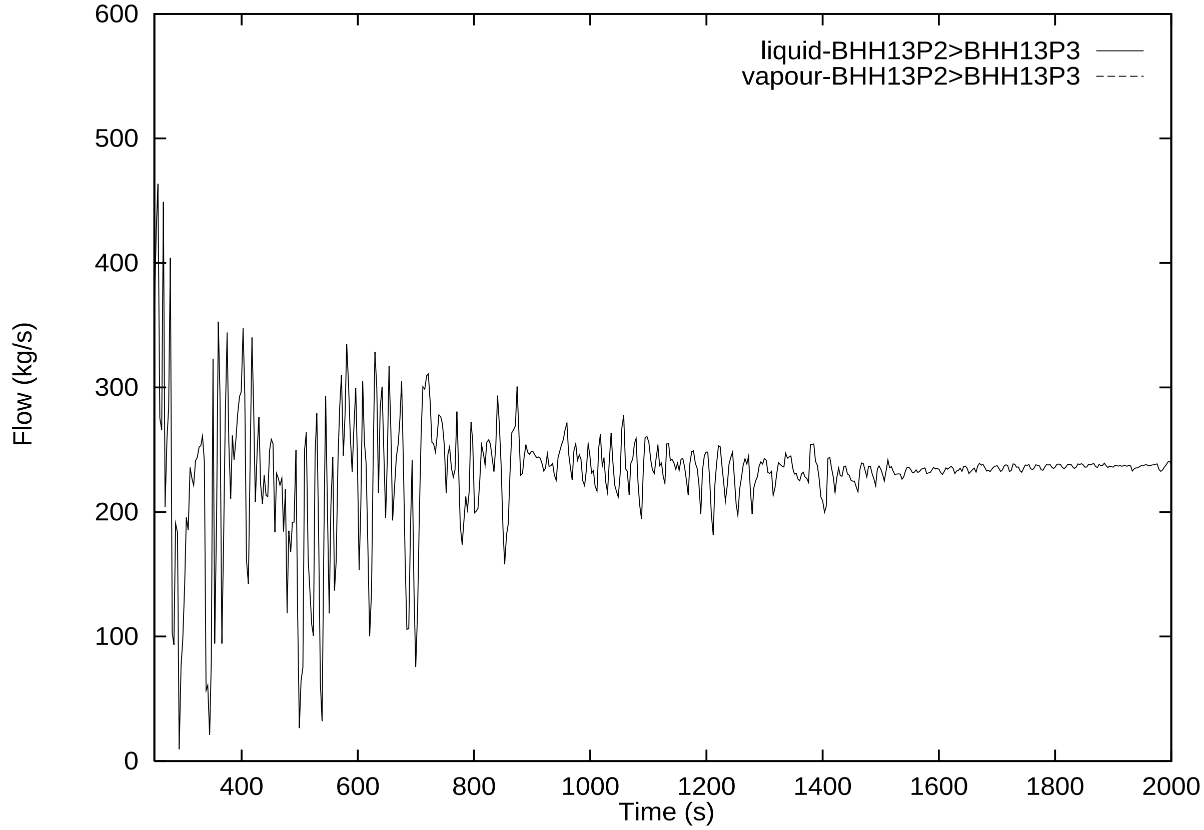


Figure 81 Large Outlet Header Interconnect Flow for 25% RIH Break with Subsequent Loss of Class IV Power

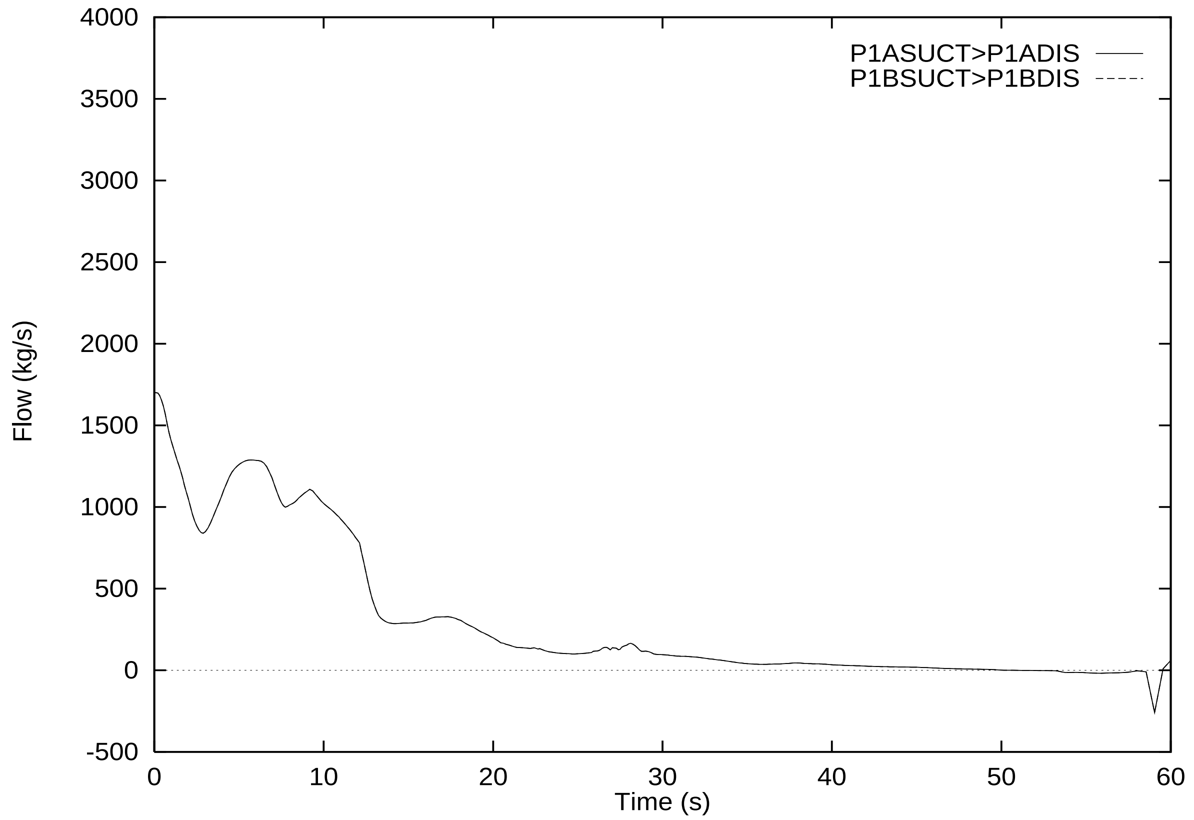


Figure 82 Pump Flows for 25% RIH Break with Subsequent Loss of Class IV Power

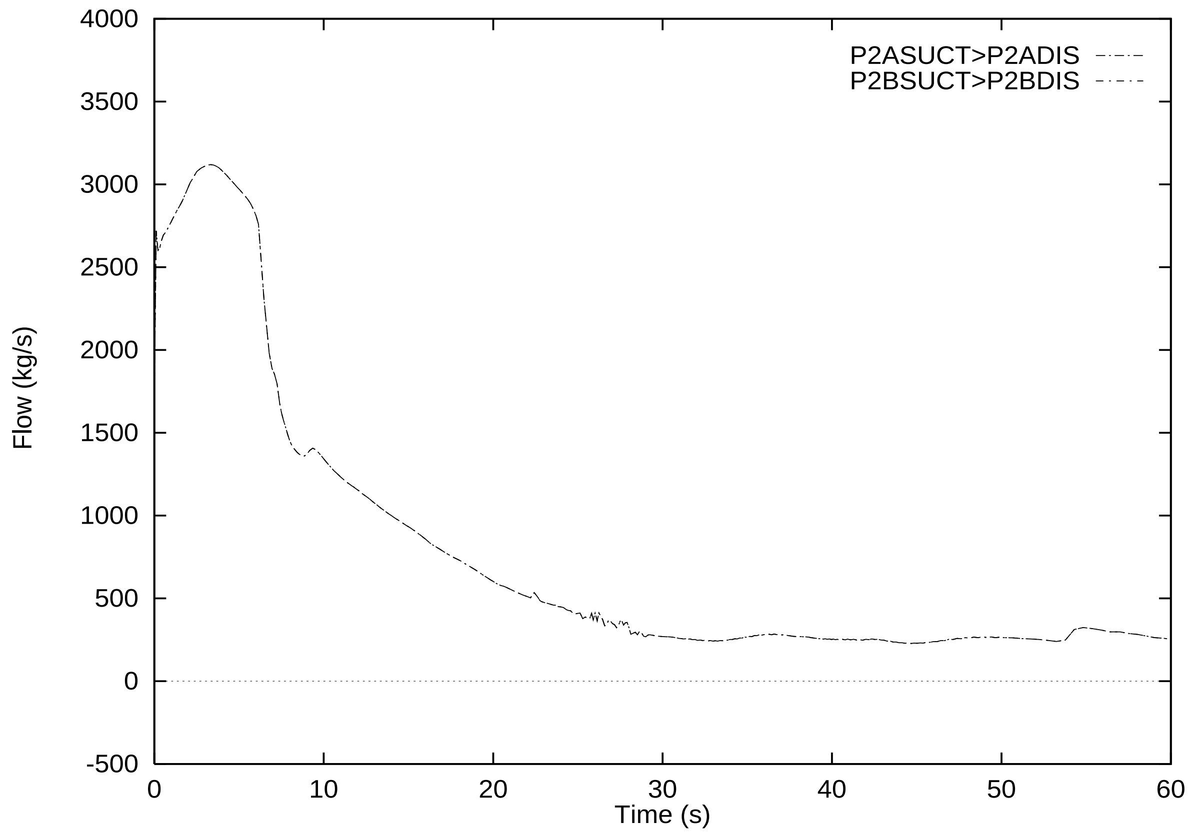


Figure 83 Pump Flows for 25% RIH Break with Subsequent Loss of Class IV Power

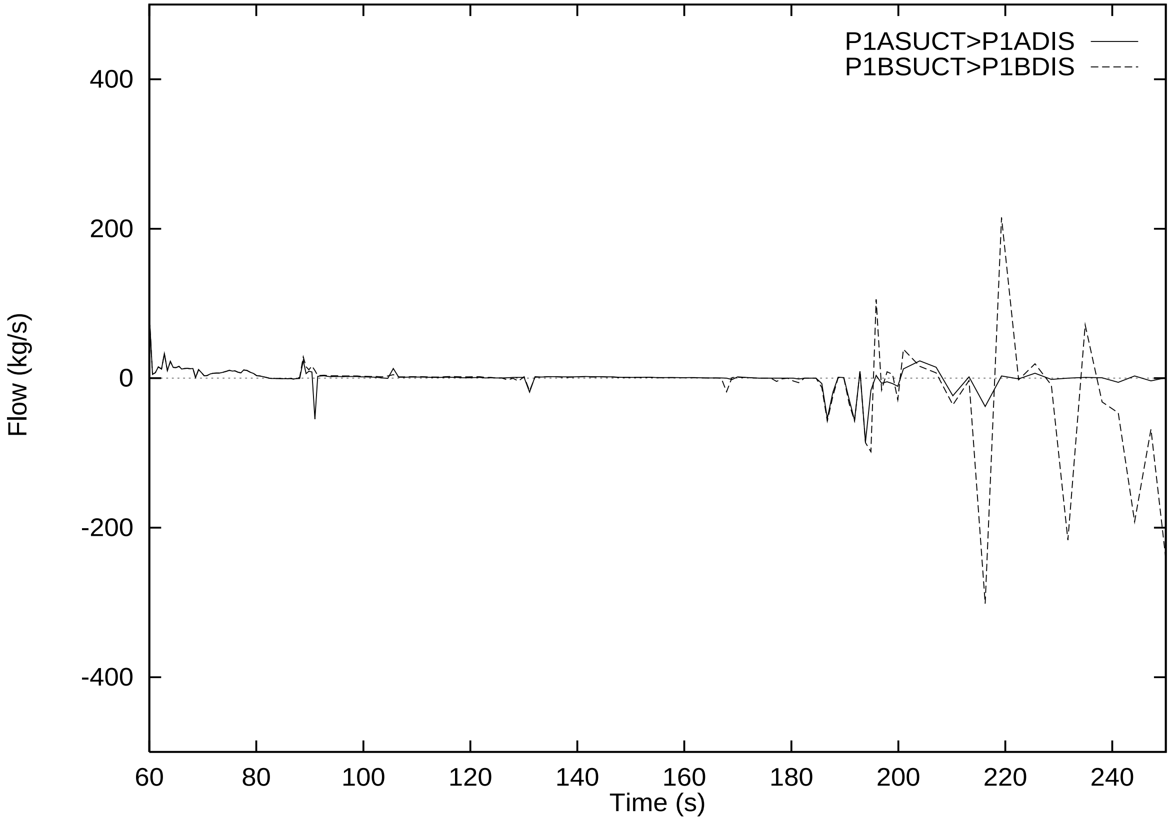


Figure 84 Pump Flows for 25% RIH Break with Subsequent Loss of Class IV Power

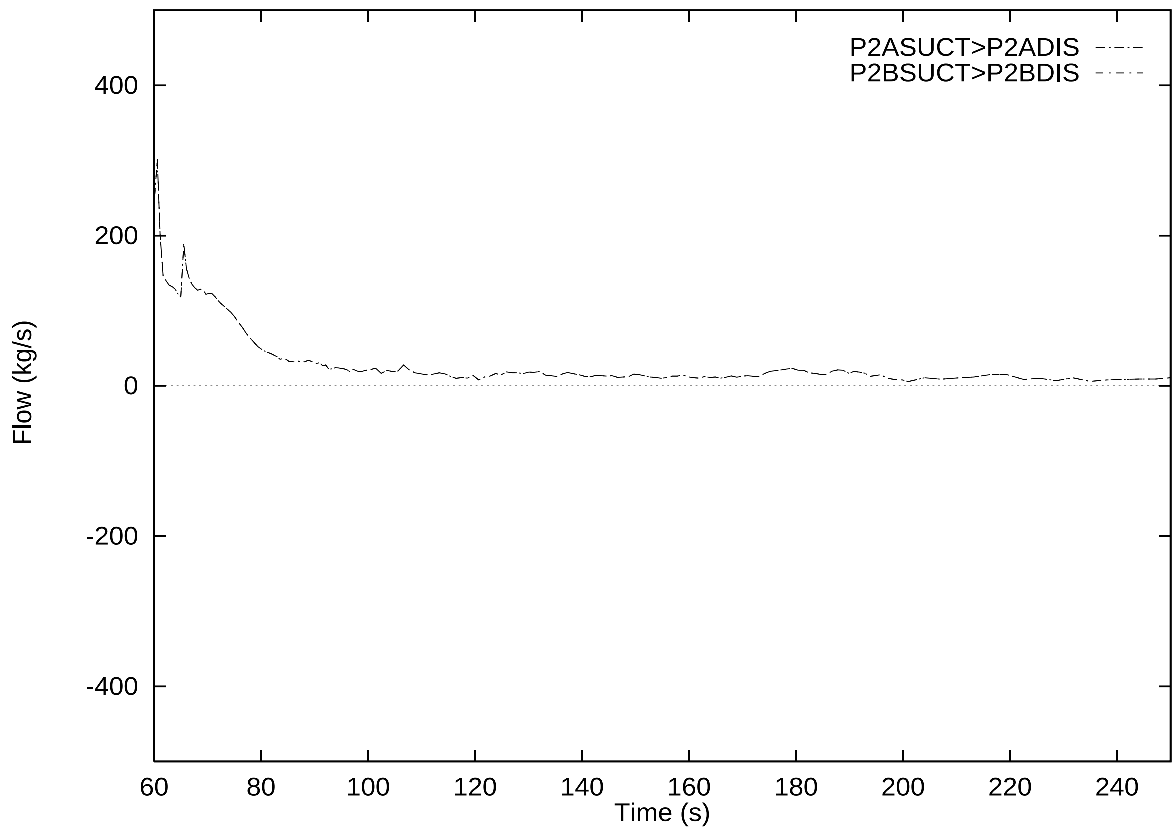


Figure 85 Pump Flows for 25% RIH Break with Subsequent Loss of Class IV Power

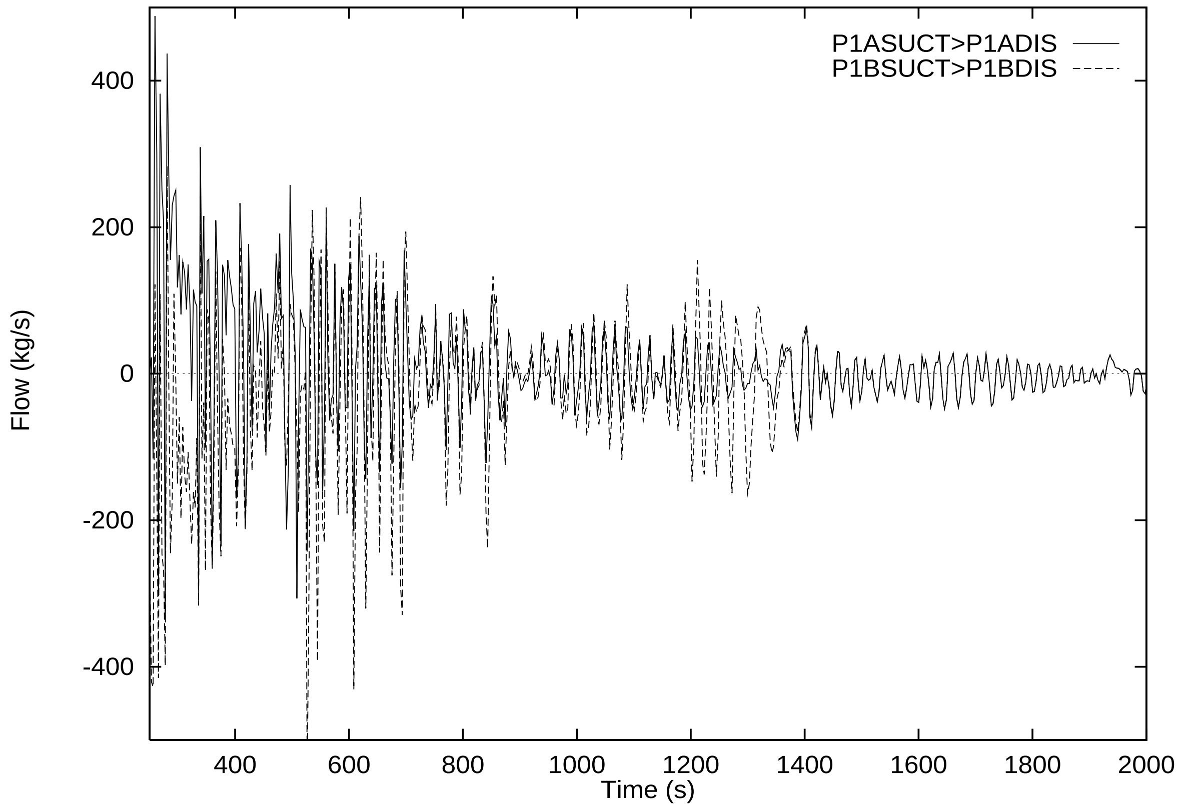


Figure 86 Pump Flows for 25% RIH Break with Subsequent Loss of Class IV Power

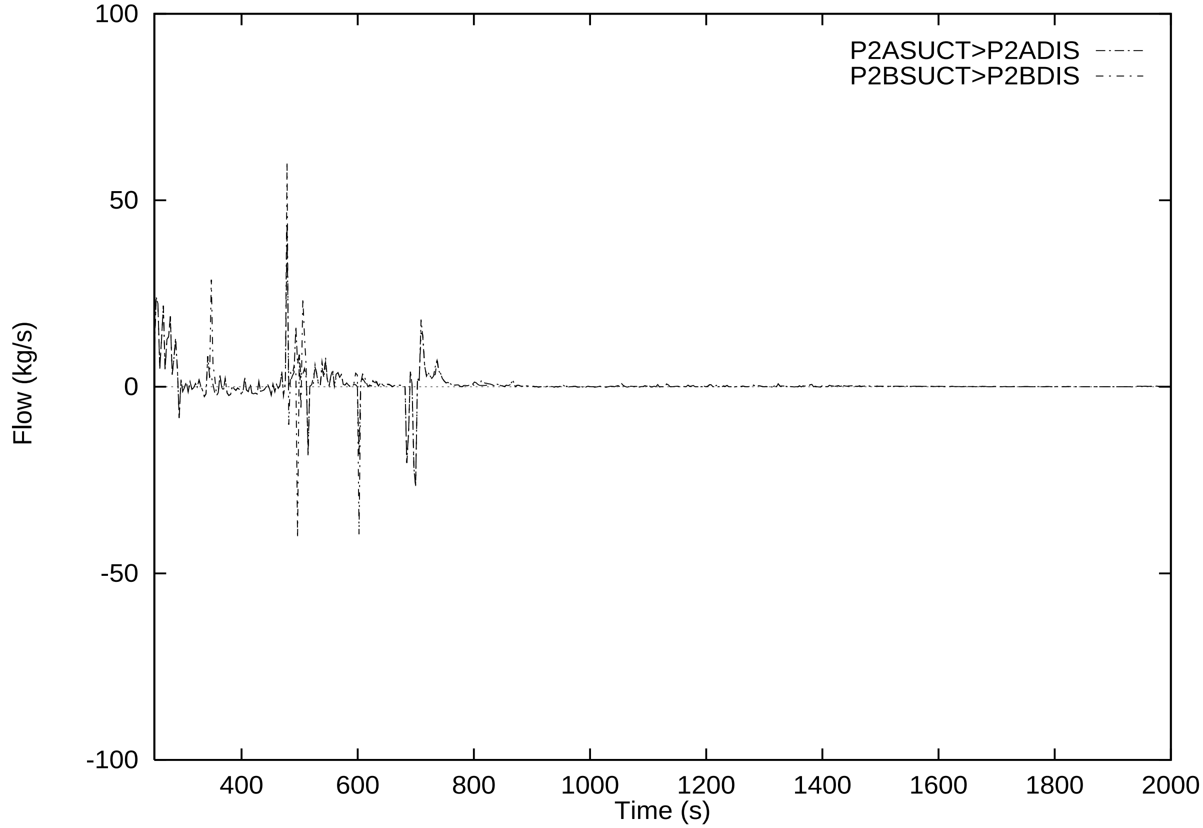


Figure 87 Pump Flows for 25% RIH Break with Subsequent Loss of Class IV Power

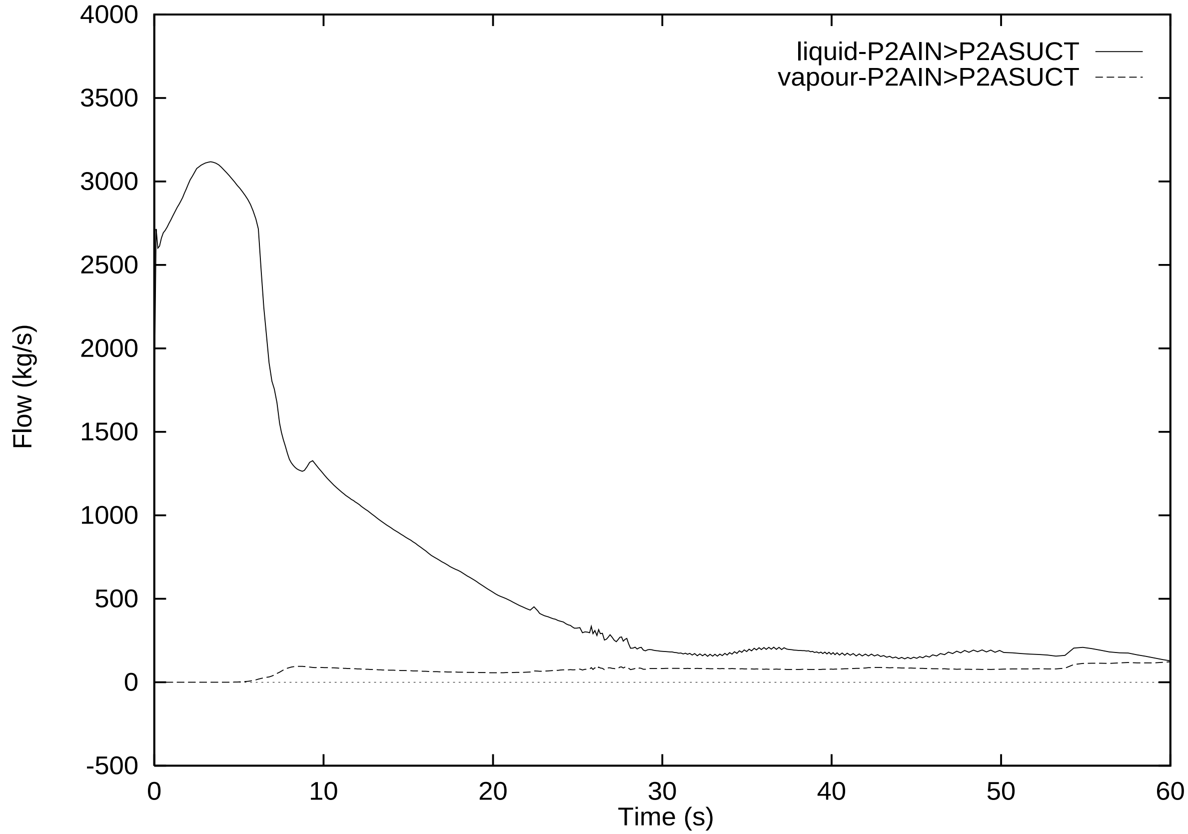


Figure 88 Pump 2A & 2B Suction Piping Flows for 25% RIH Break with Subsequent Loss of Class IV Power

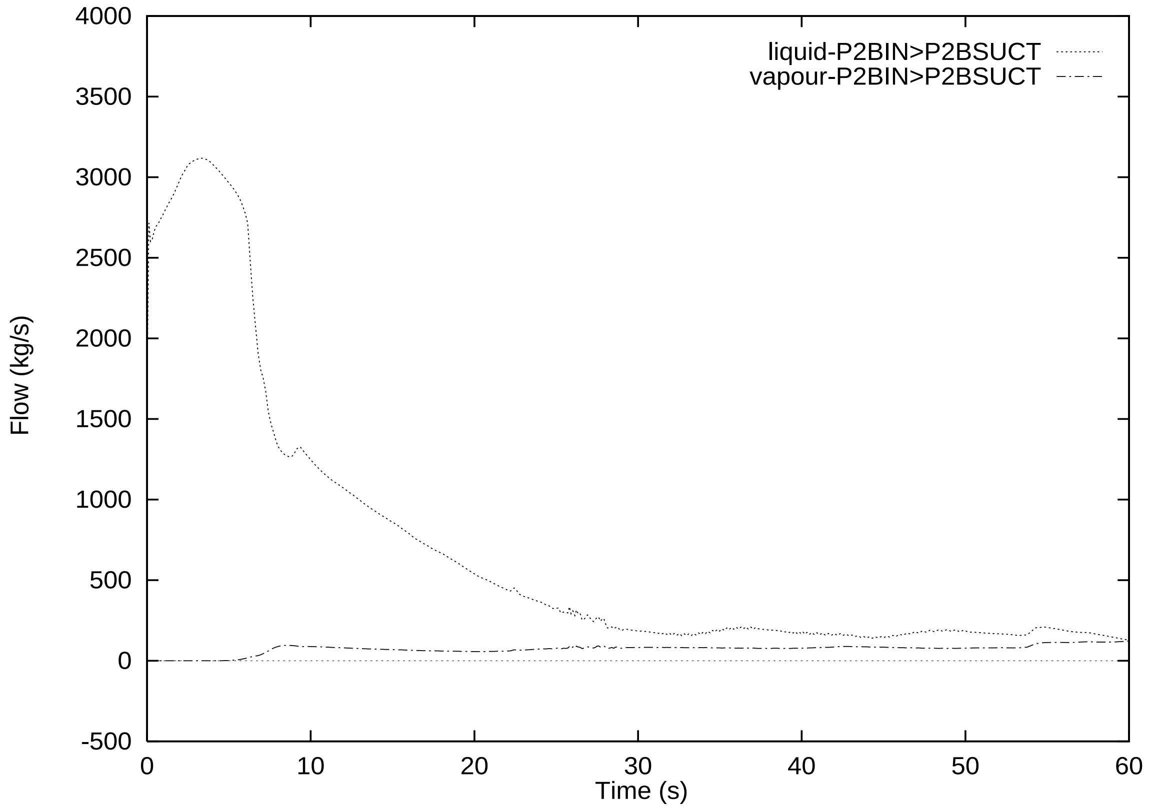


Figure 89 Pump 2A & 2B Suction Piping Flows for 25% RIH Break with Subsequent Loss of Class IV Power

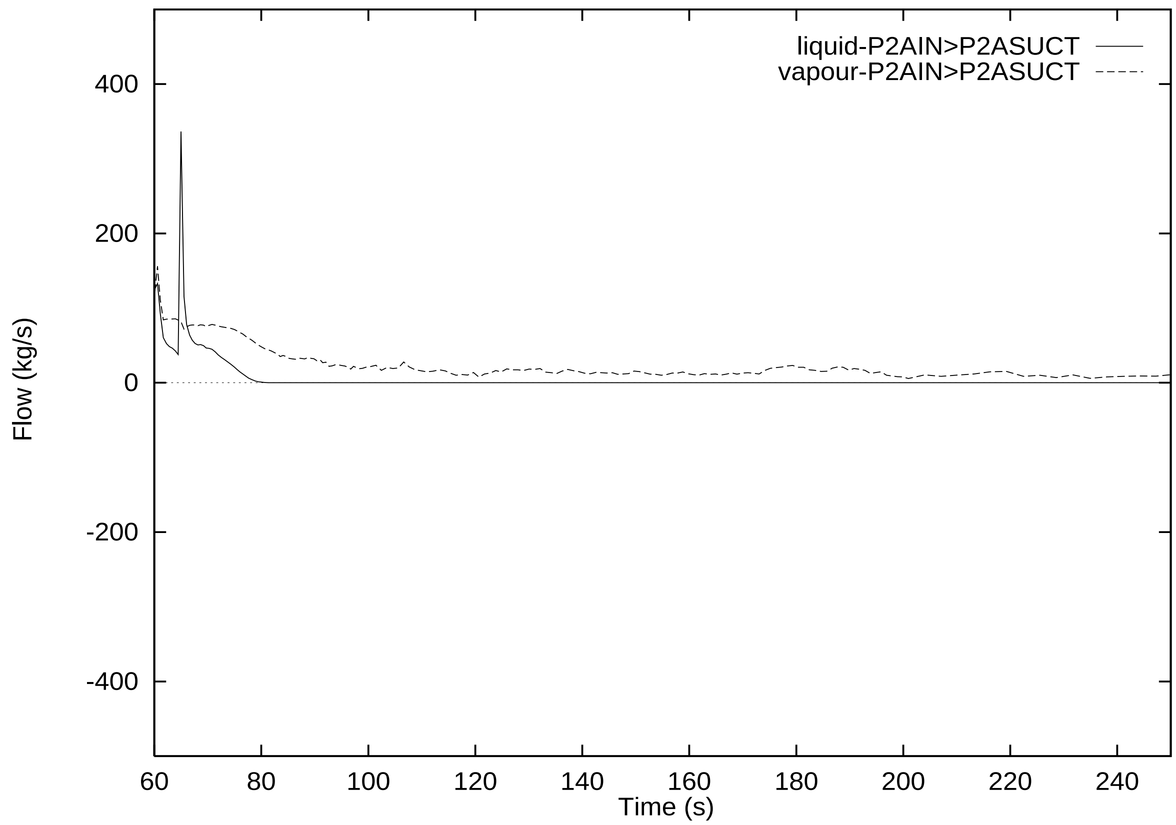


Figure 90 Pump 2A & 2B Suction Piping Flows for 25% RIH Break with Subsequent Loss of Class IV Power

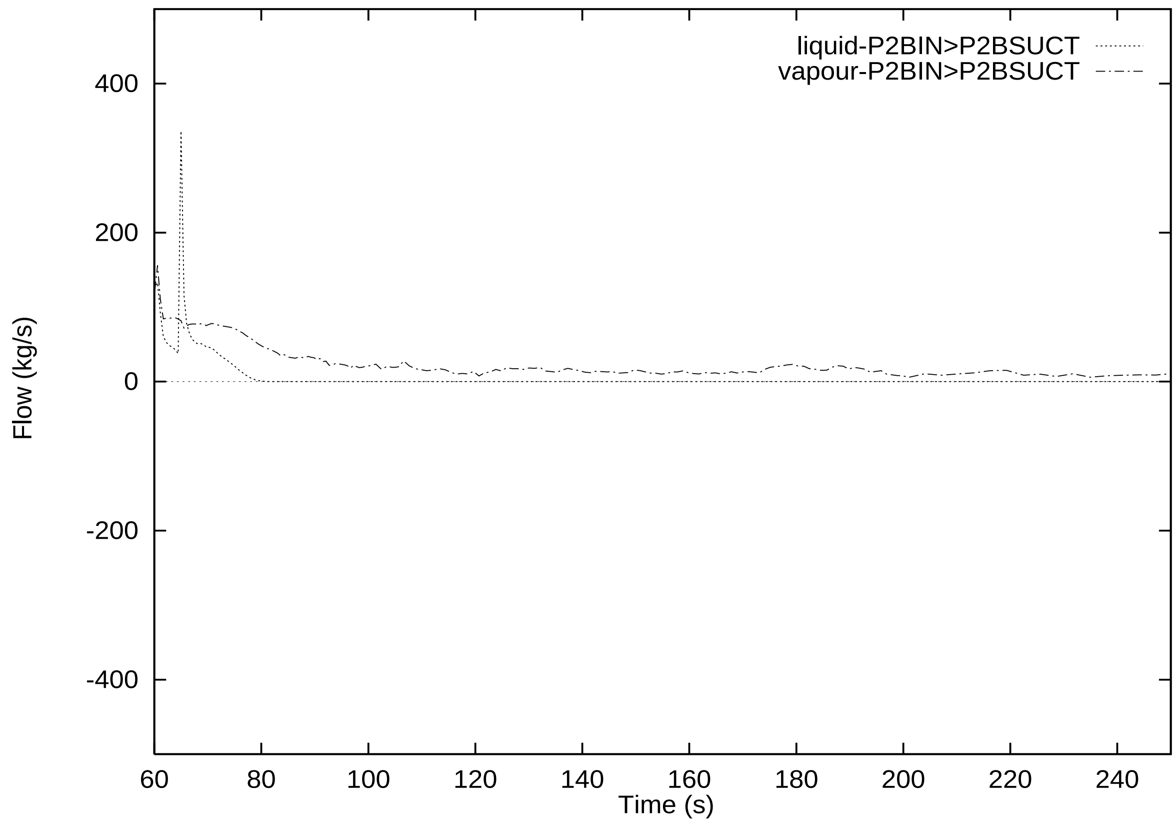


Figure 91 Pump 2A & 2B Suction Piping Flows for 25% RIH Break with Subsequent Loss of Class IV Power

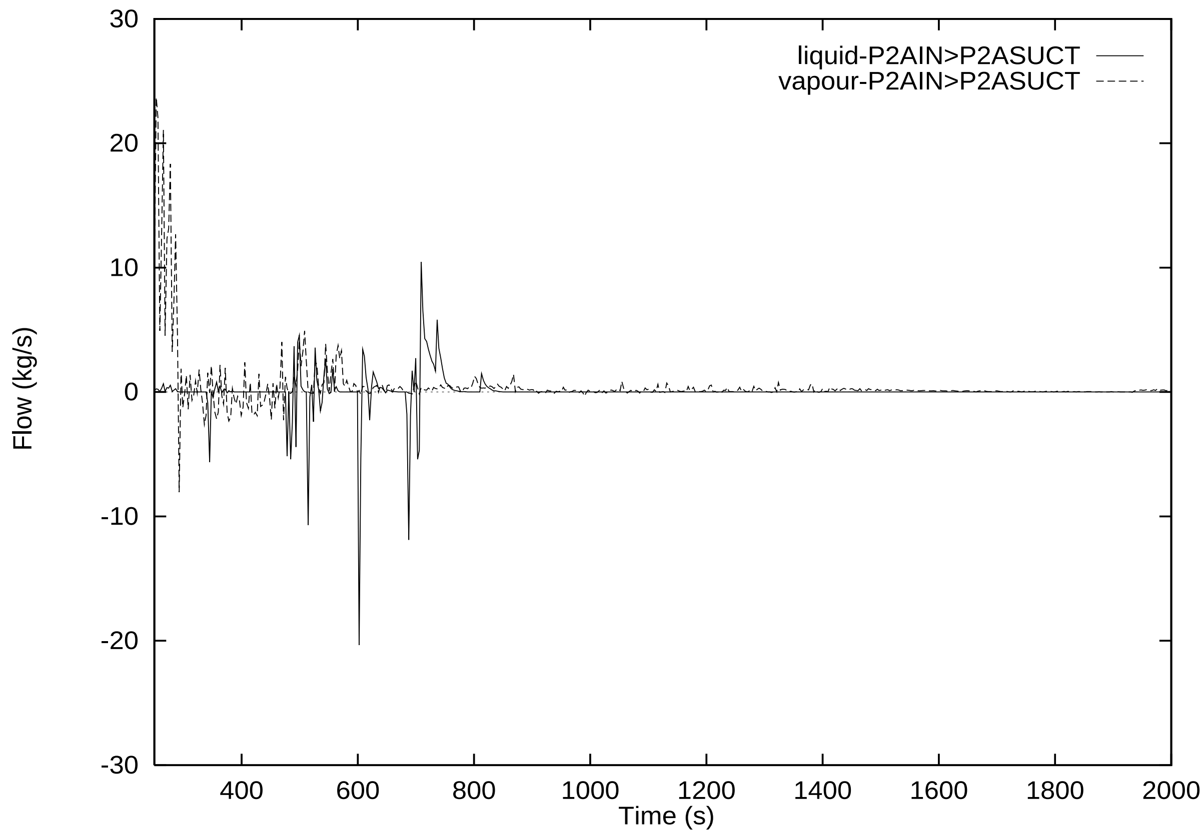


Figure 92 Pump 2A & 2B Suction Piping Flows for 25% RIH Break with Subsequent Loss of Class IV Power

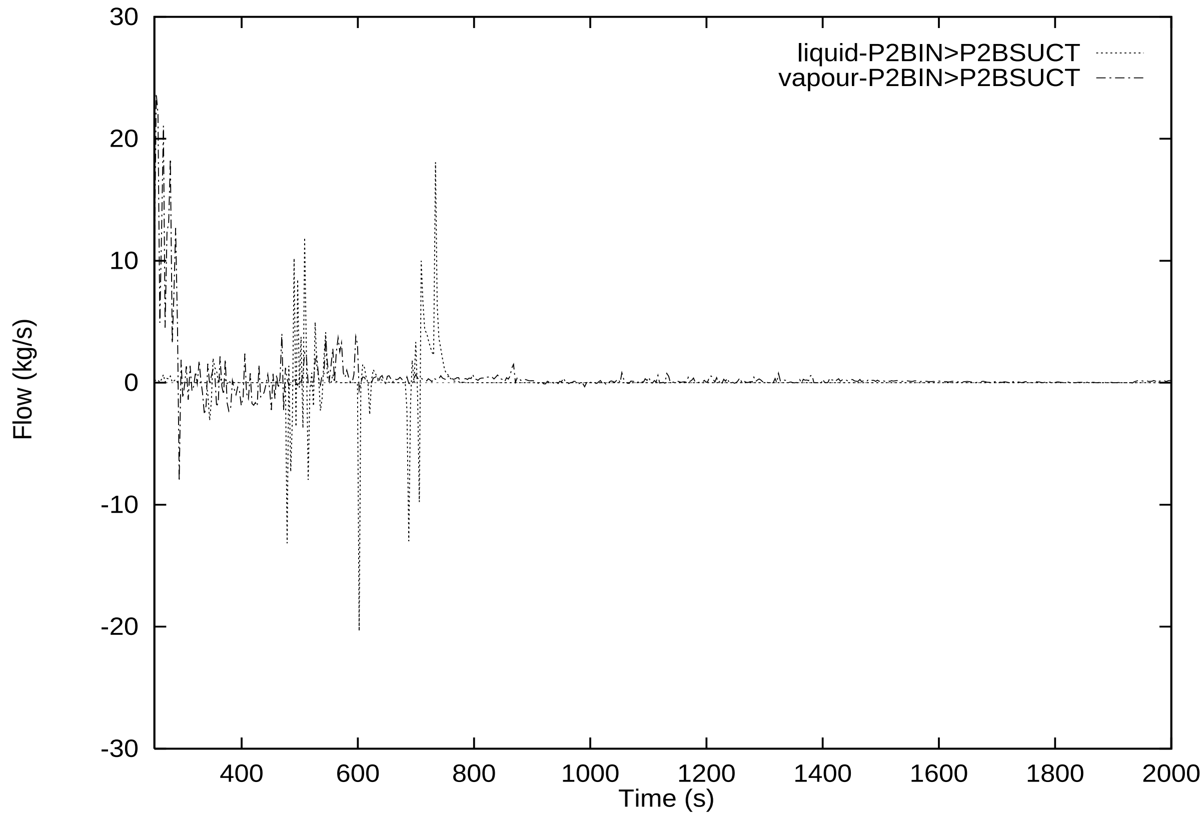


Figure 93 Pump 2A & 2B Suction Piping Flows for 25% RIH Break with Subsequent Loss of Class IV Power

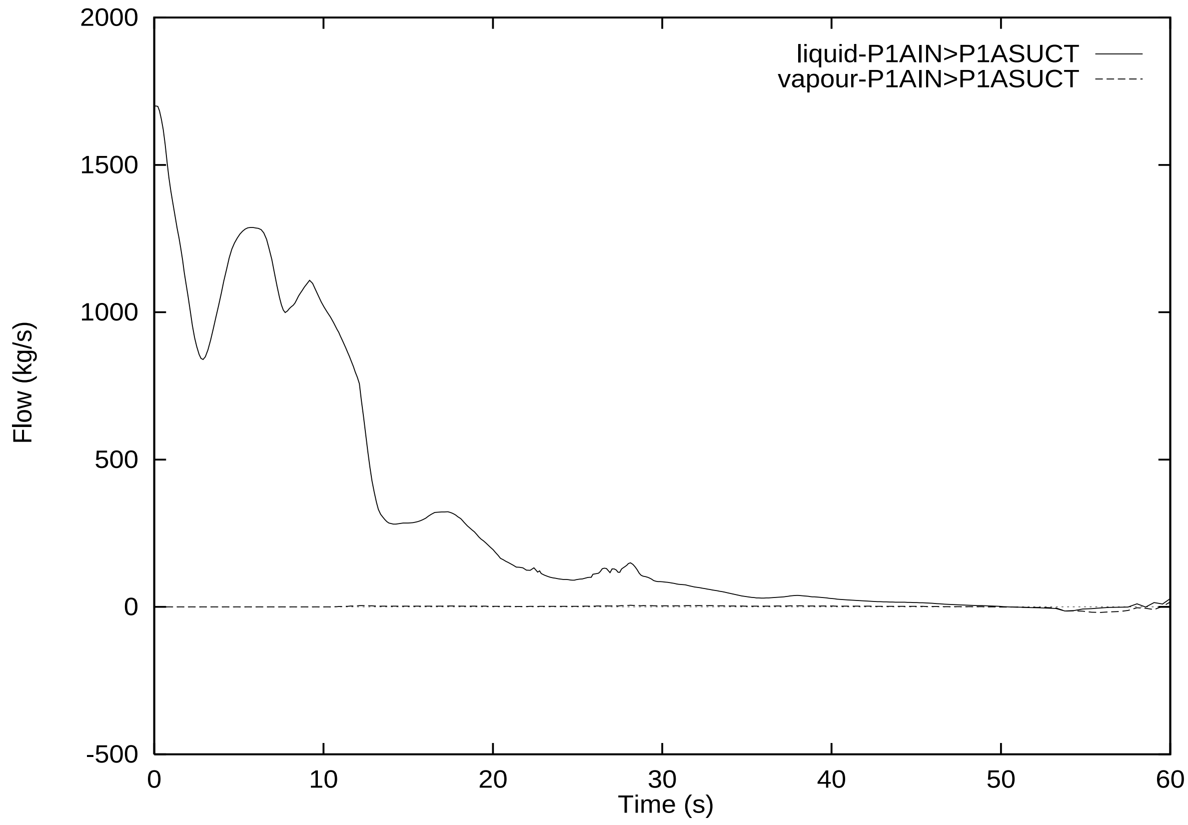


Figure 94 Pump 1A & 1B Suction Piping Flows for 25% RIH Break with Subsequent Loss of Class IV Power

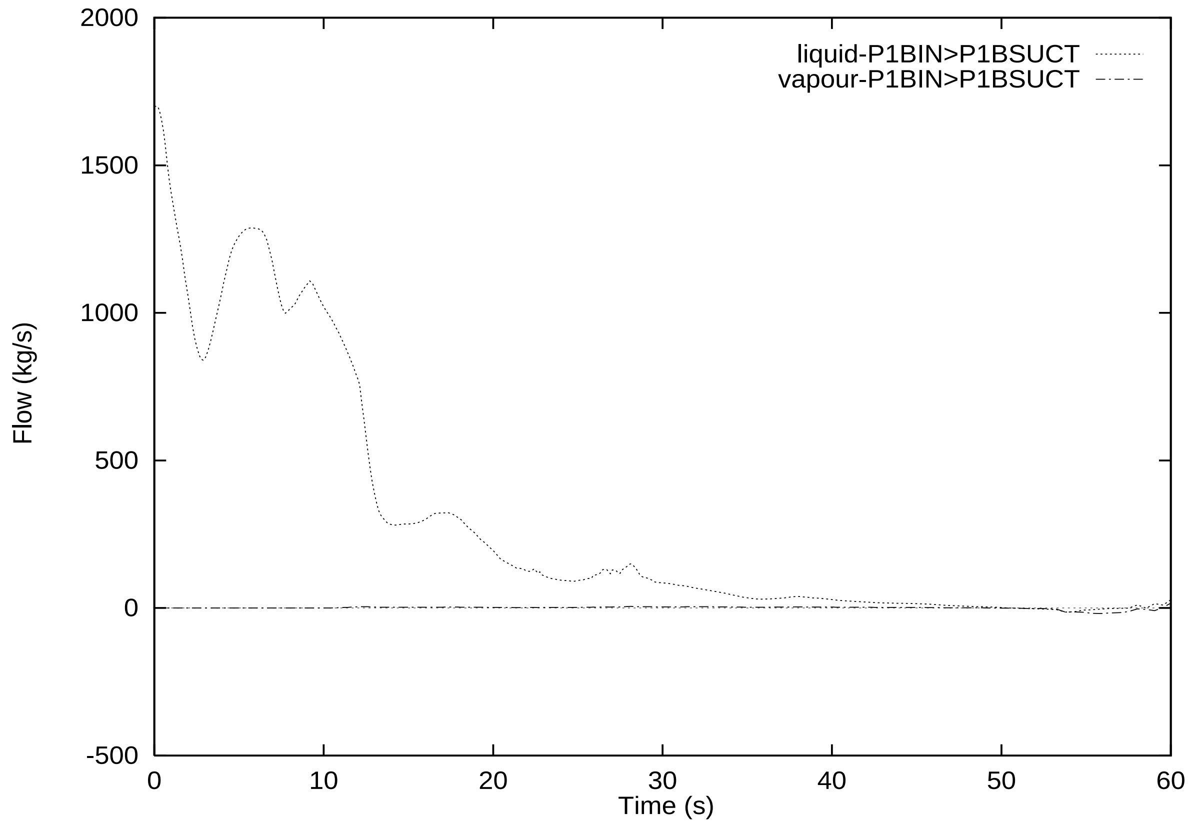


Figure 95 Pump 1A & 1B Suction Piping Flows for 25% RIH Break with Subsequent Loss of Class IV Power

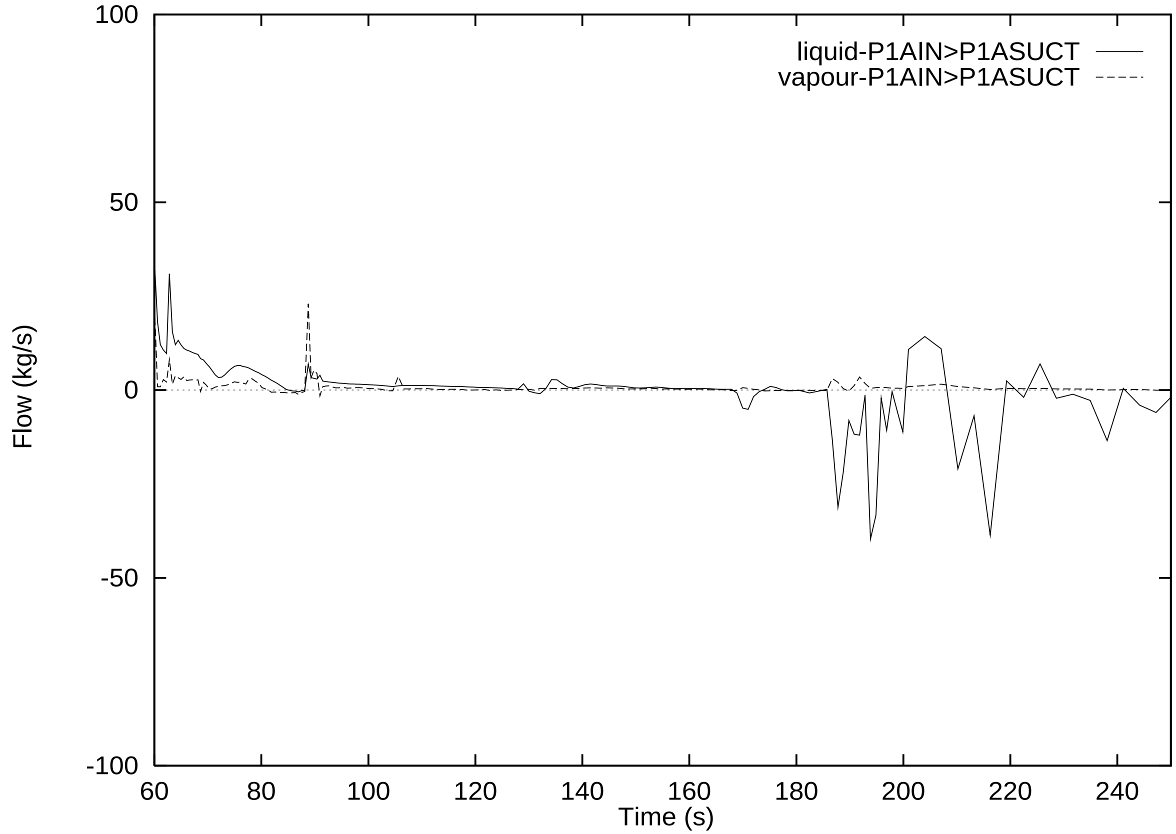


Figure 96 Pump 1A & 1B Suction Piping Flows for 25% RIH Break with Subsequent Loss of Class IV Power

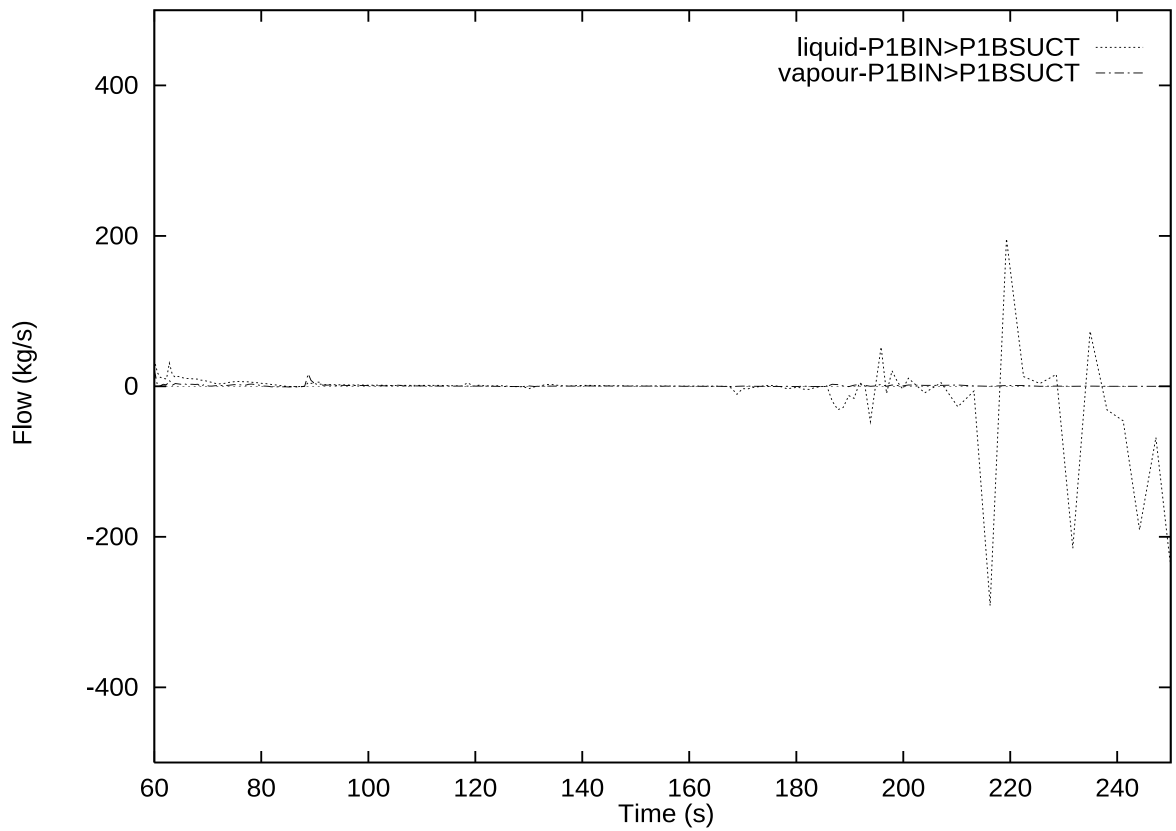


Figure 97 Pump 1A & 1B Suction Piping Flows for 25% RIH Break with Subsequent Loss of Class IV Power

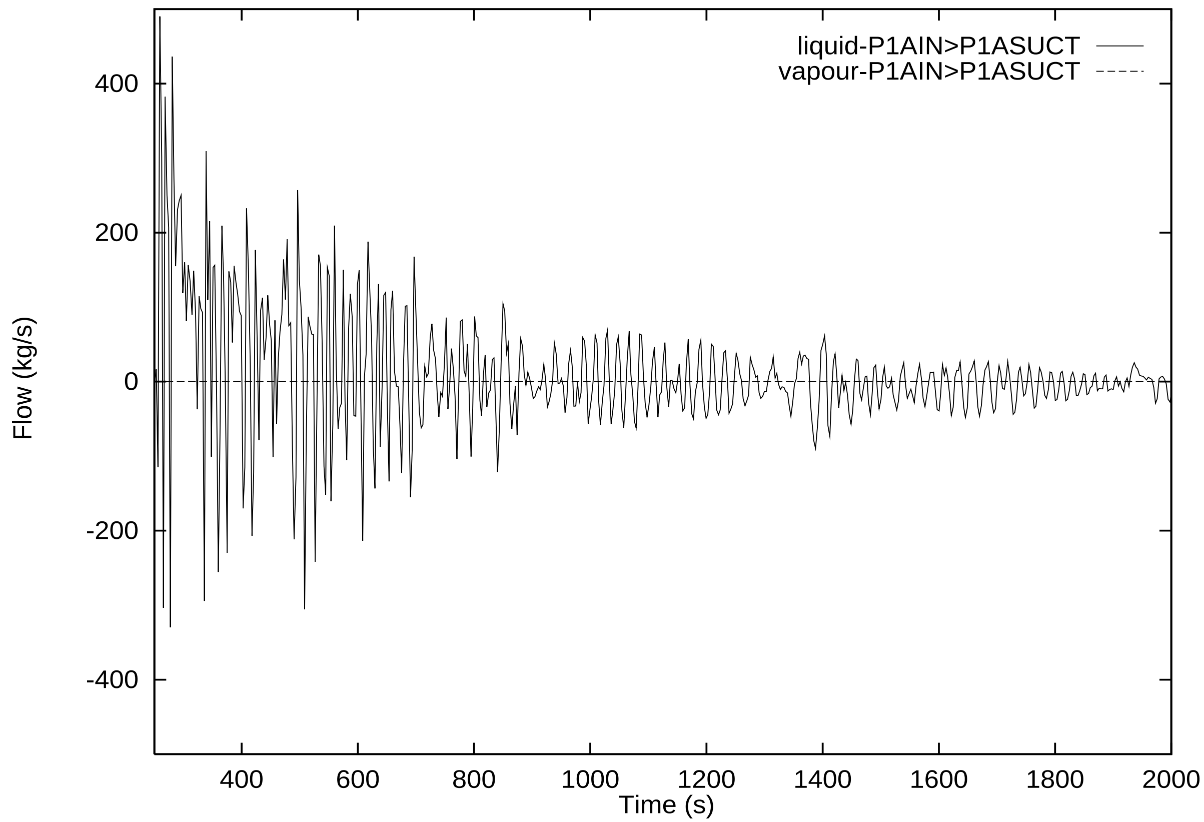


Figure 98 Pump 1A & 1B Suction Piping Flows for 25% RIH Break with Subsequent Loss of Class IV Power

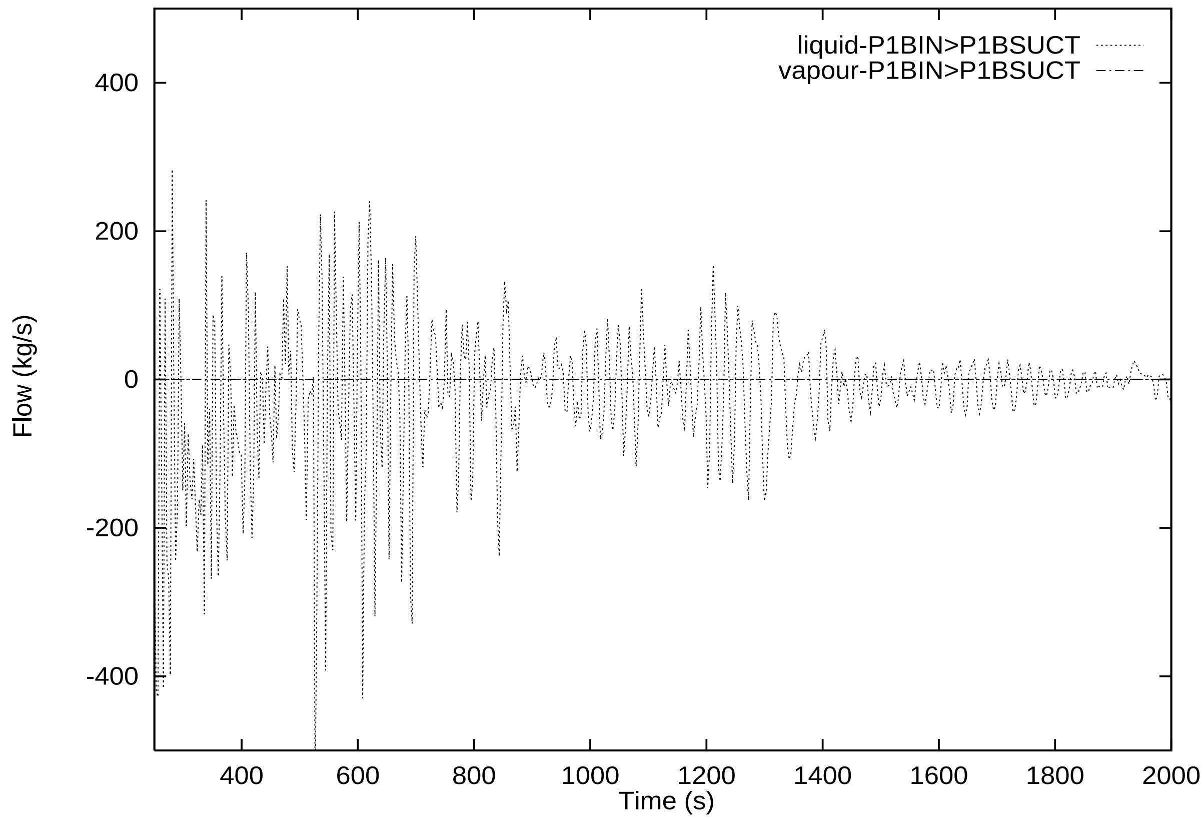


Figure 99 Pump 1A & 1B Suction Piping Flows for 25% RIH Break with Subsequent Loss of Class IV Power

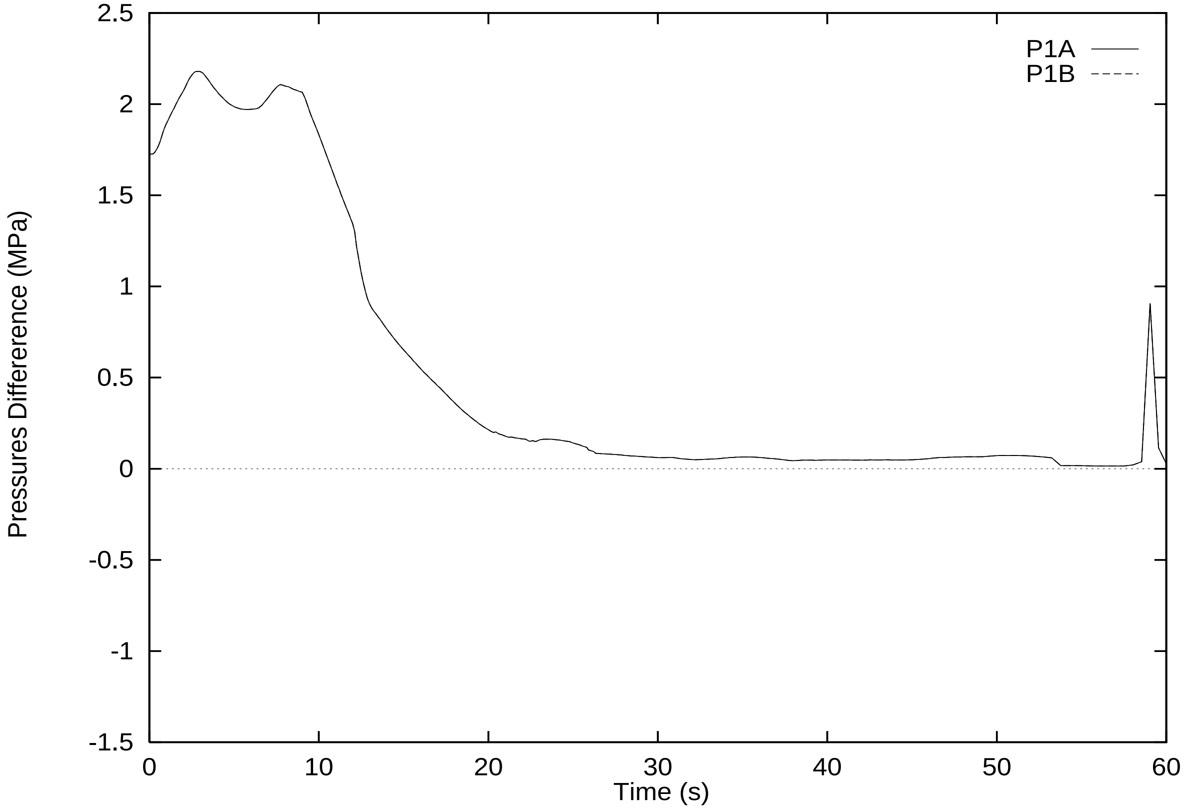


Figure 100 Pump Heads for 25% RIH Break with Subsequent Loss of Class IV Power

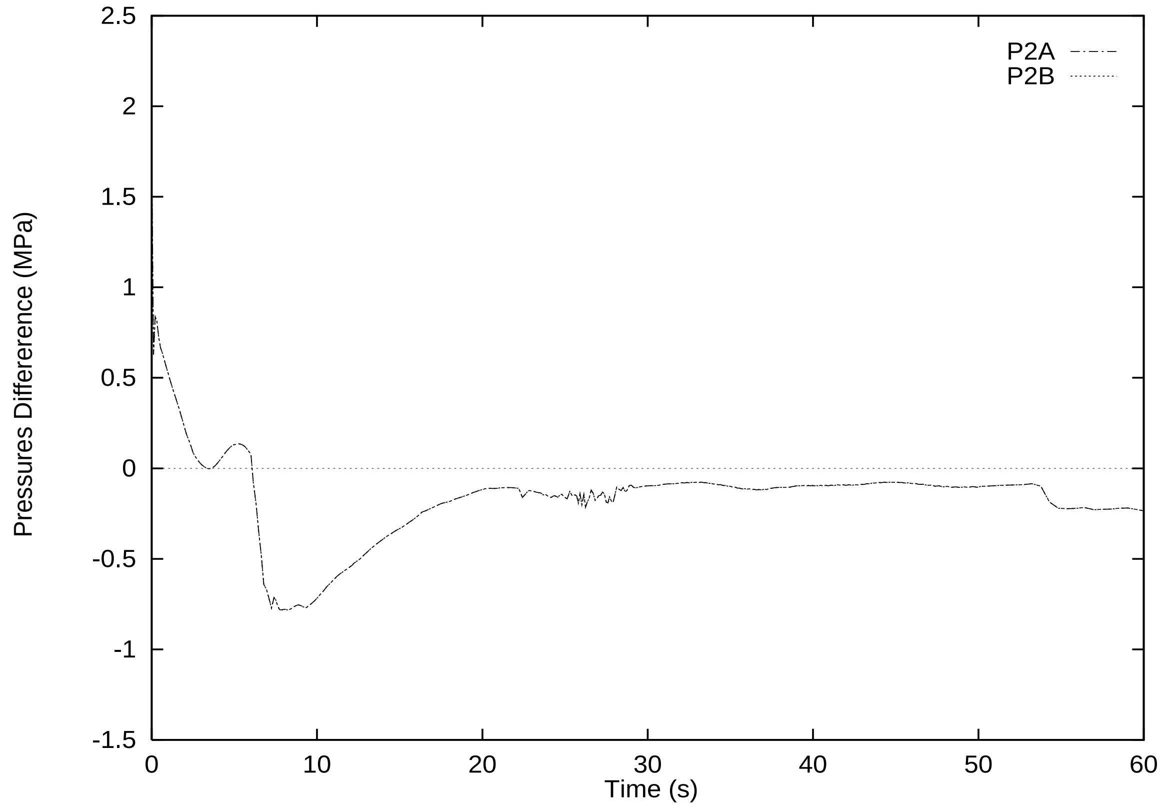


Figure 101 Pump Heads for 25% RIH Break with Subsequent Loss of Class IV Power

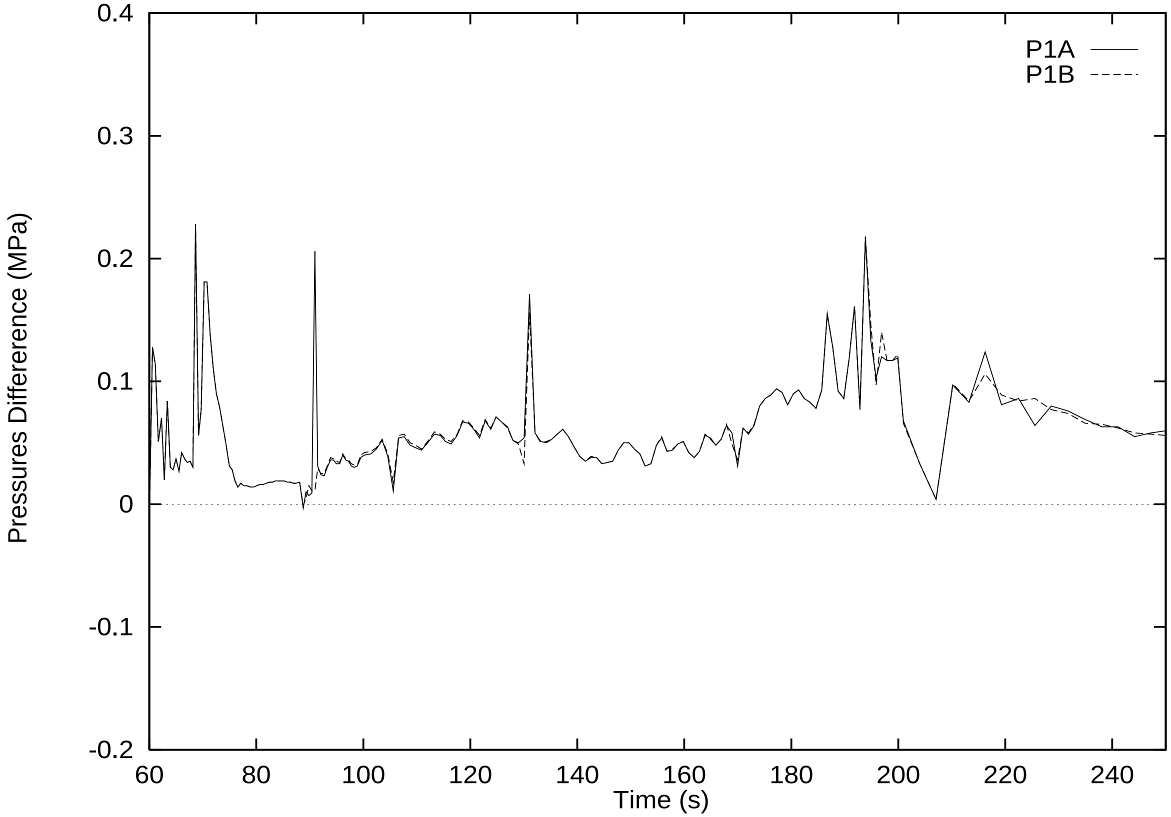


Figure 102 Pump Heads for 25% RIH Break with Subsequent Loss of Class IV Power

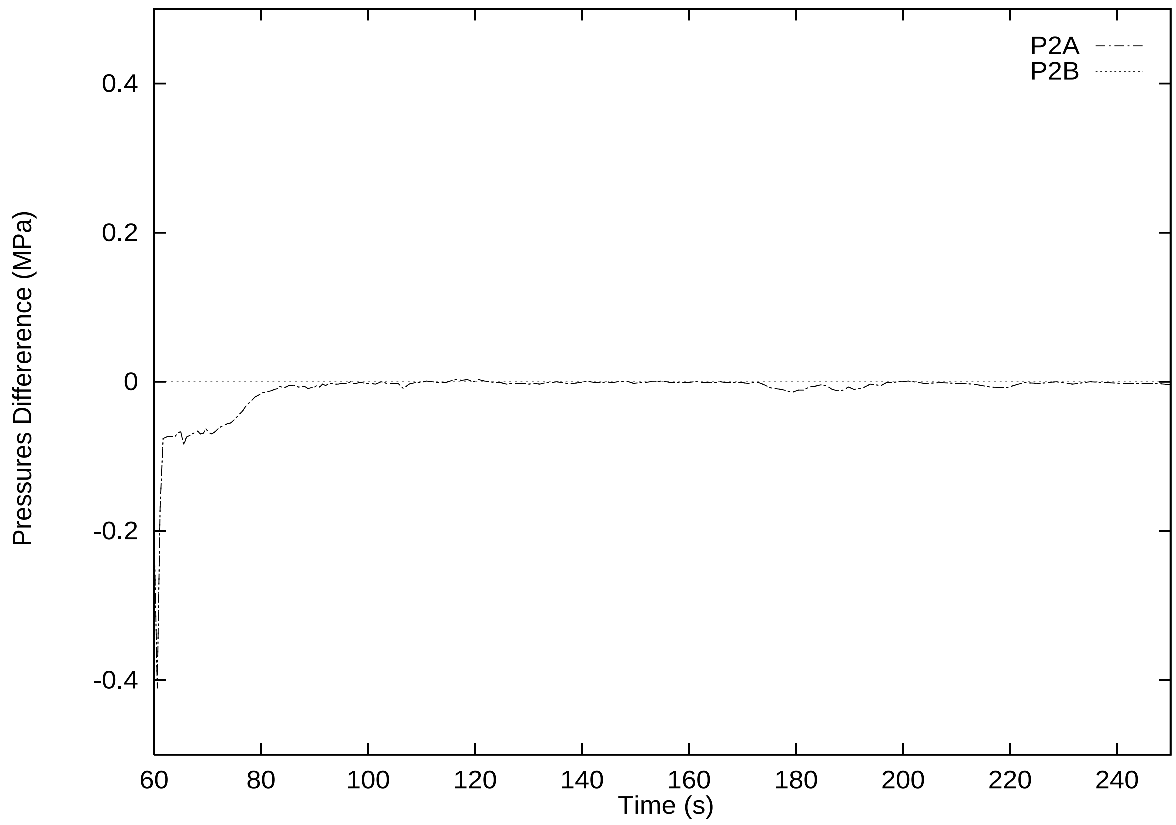


Figure 103 Pump Heads for 25% RIH Break with Subsequent Loss of Class IV Power

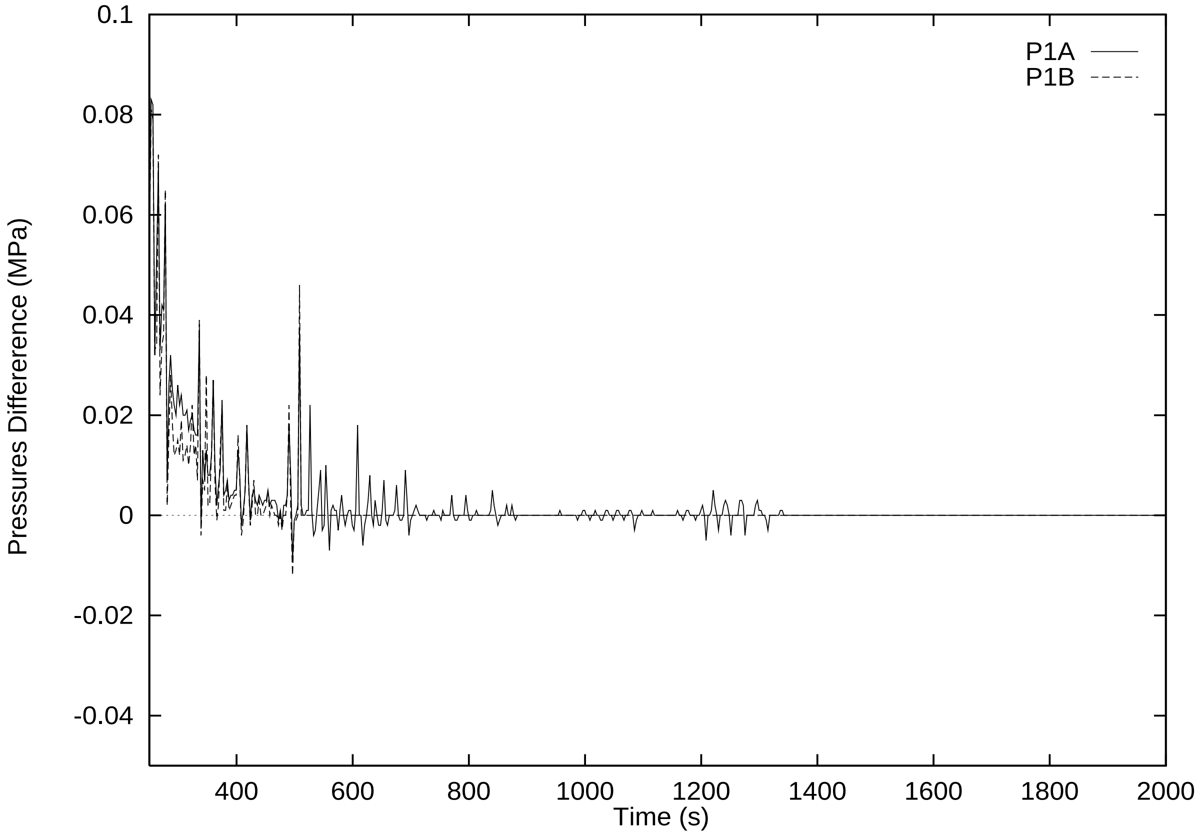


Figure 104 Pump Heads for 25% RIH Break with Subsequent Loss of Class IV Power

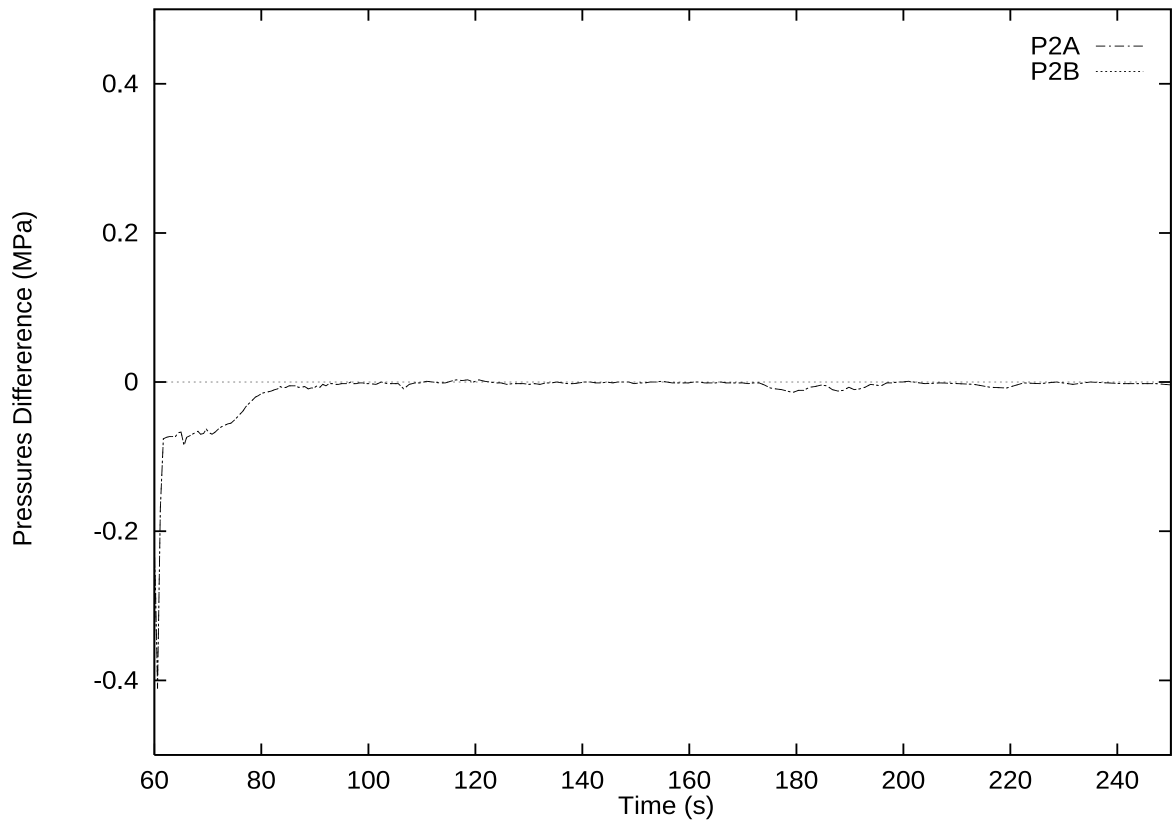


Figure 105 Pump Heads for 25% RIH Break with Subsequent Loss of Class IV Power

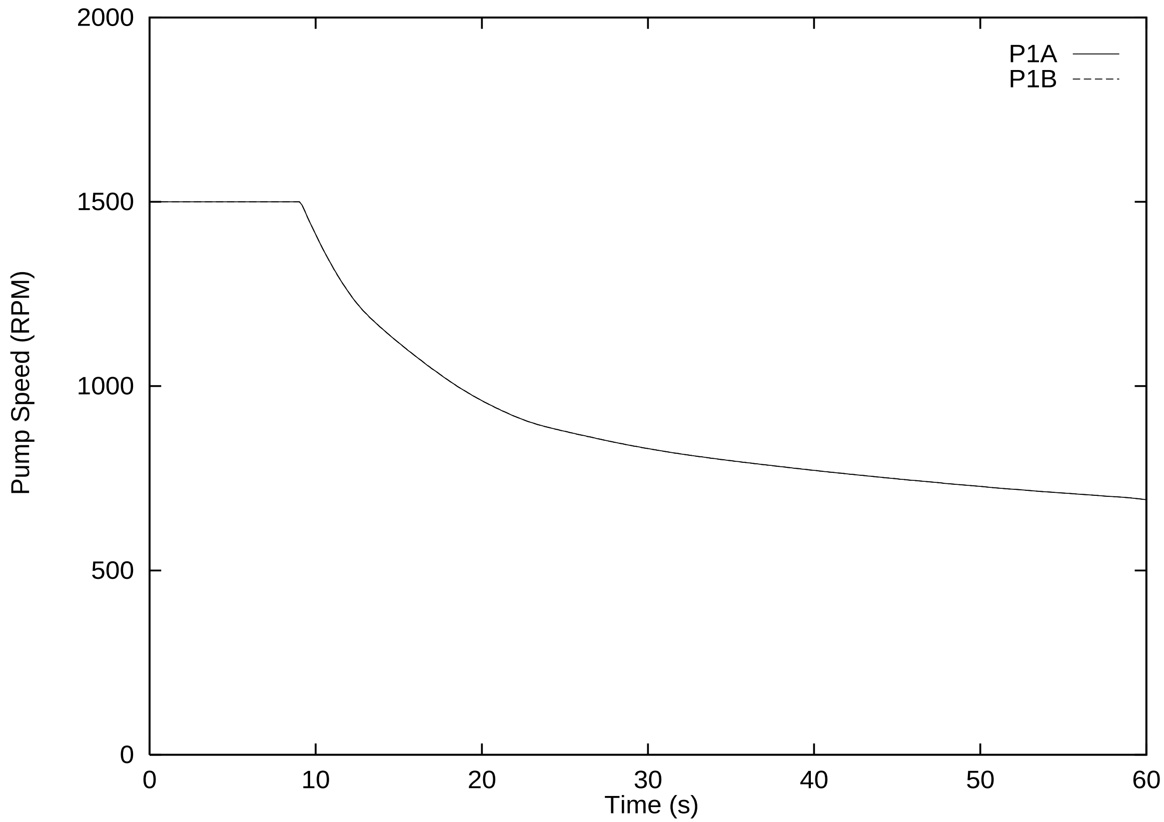


Figure 106 HTS Pump Speeds for 25% RIH Break with Subsequent Loss of Class IV Power

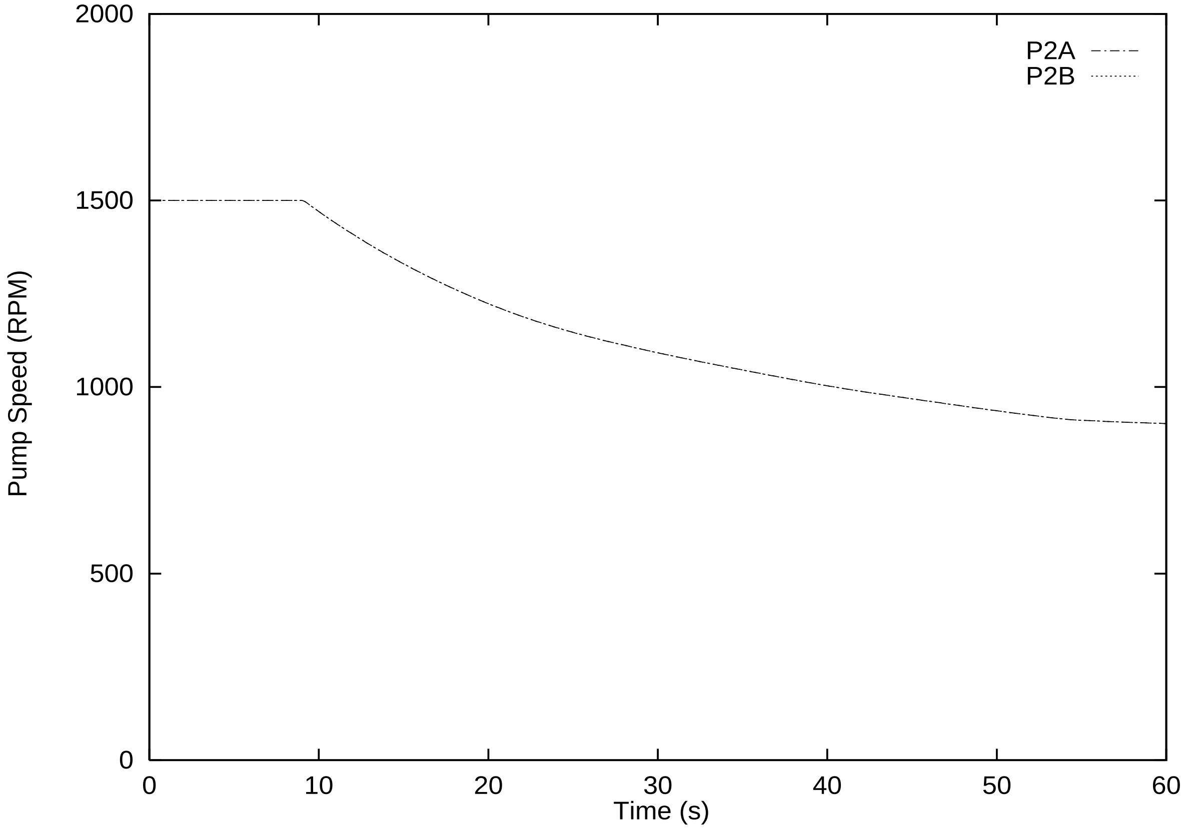


Figure 107 HTS Pump Speeds for 25% RIH Break with Subsequent Loss of Class IV Power

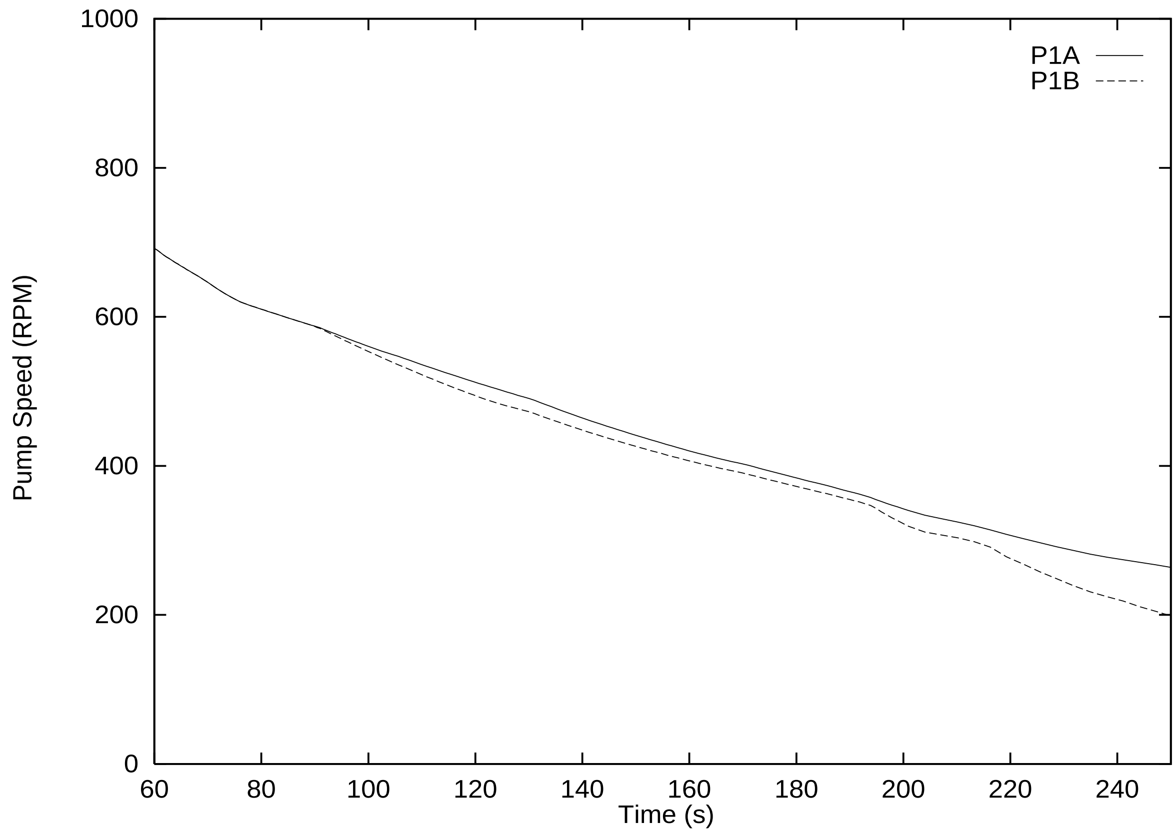


Figure 108 HTS Pump Speeds for 25% RIH Break with Subsequent Loss of Class IV Power

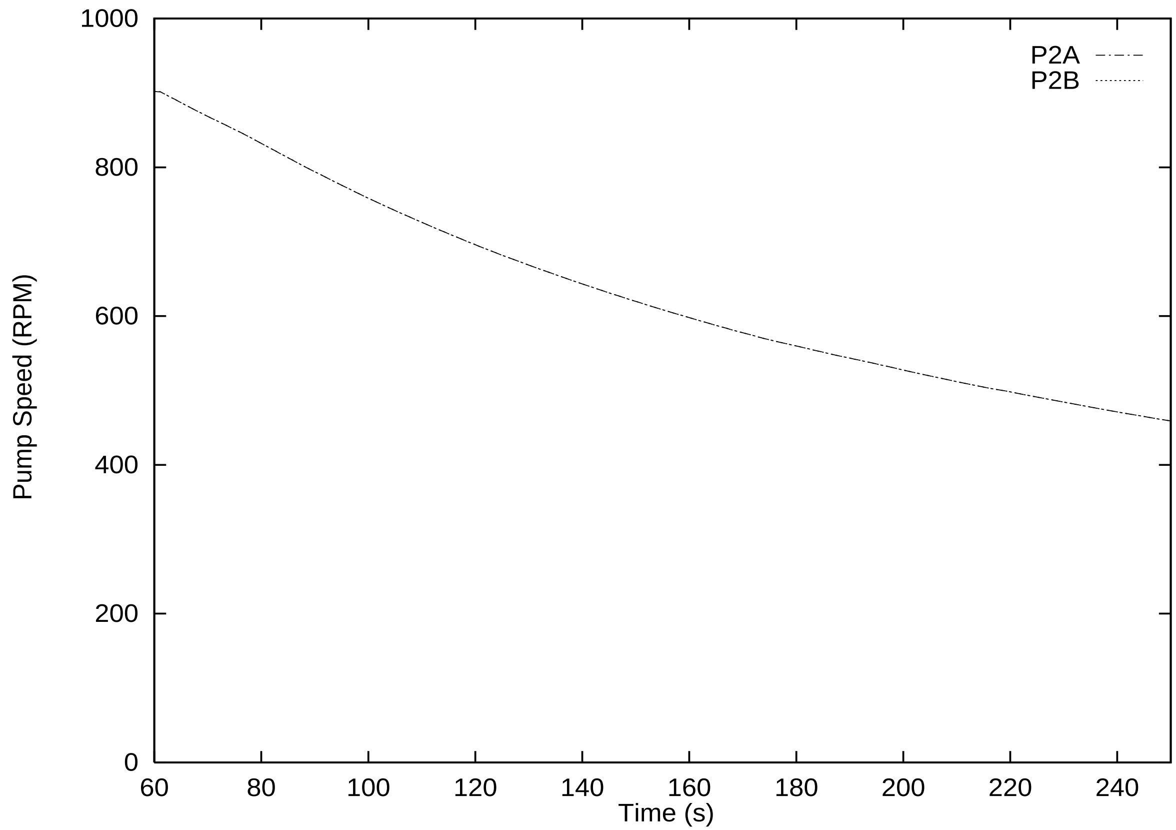


Figure 109 HTS Pump Speeds for 25% RIH Break with Subsequent Loss of Class IV Power

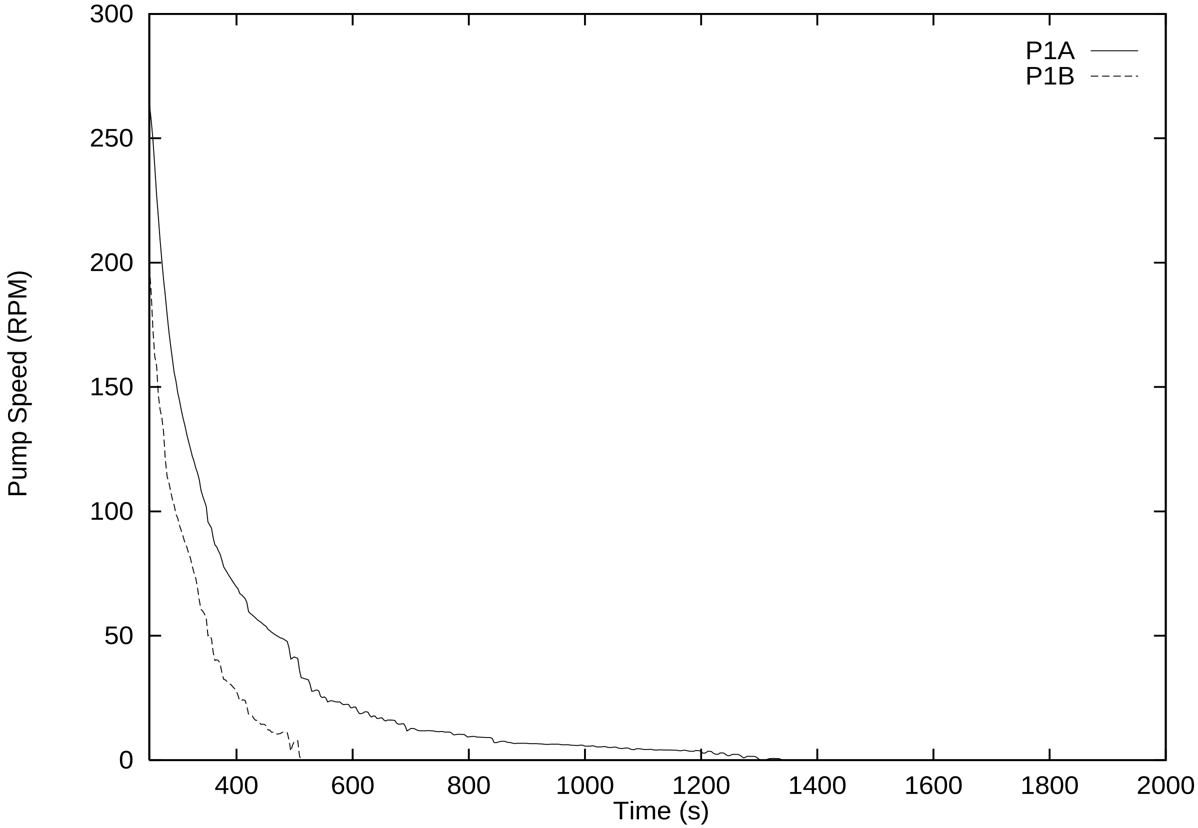


Figure 110 HTS Pump Speeds for 25% RIH Break with Subsequent Loss of Class IV Power

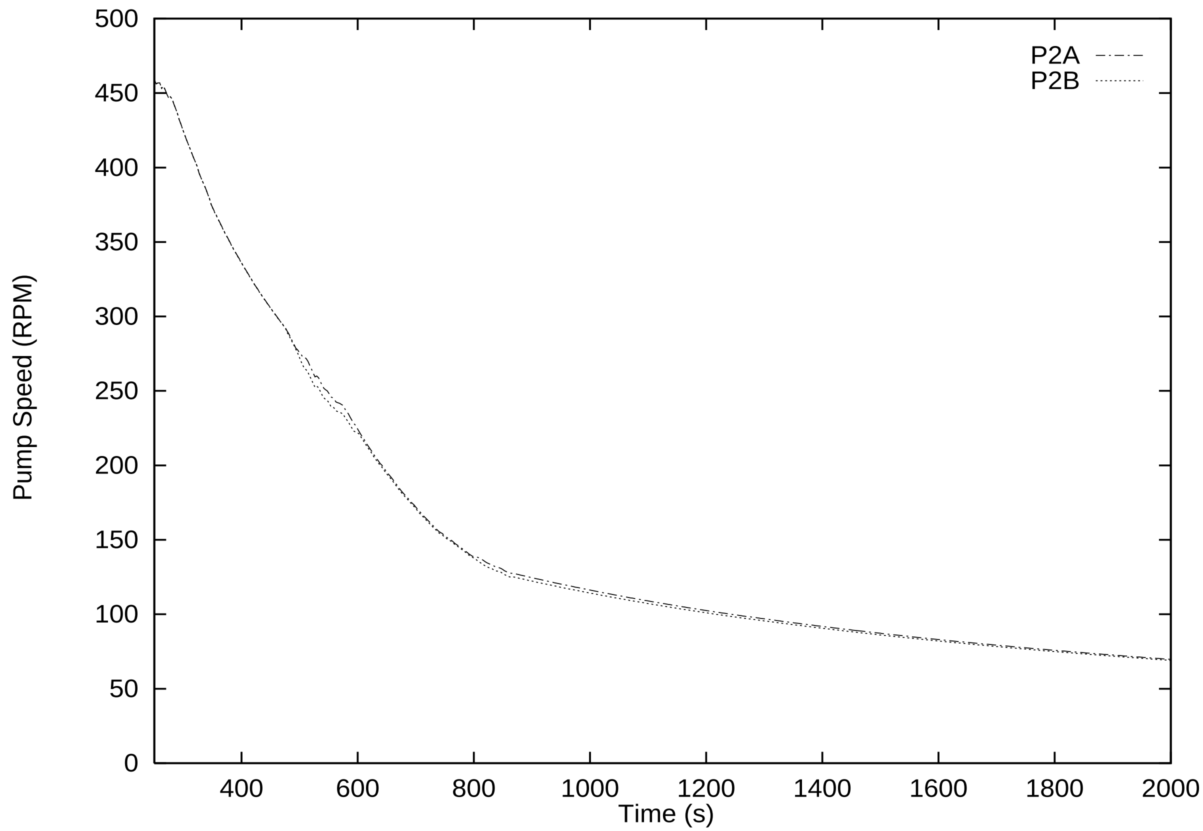


Figure 111 HTS Pump Speeds for 25% RIH Break with Subsequent Loss of Class IV Power

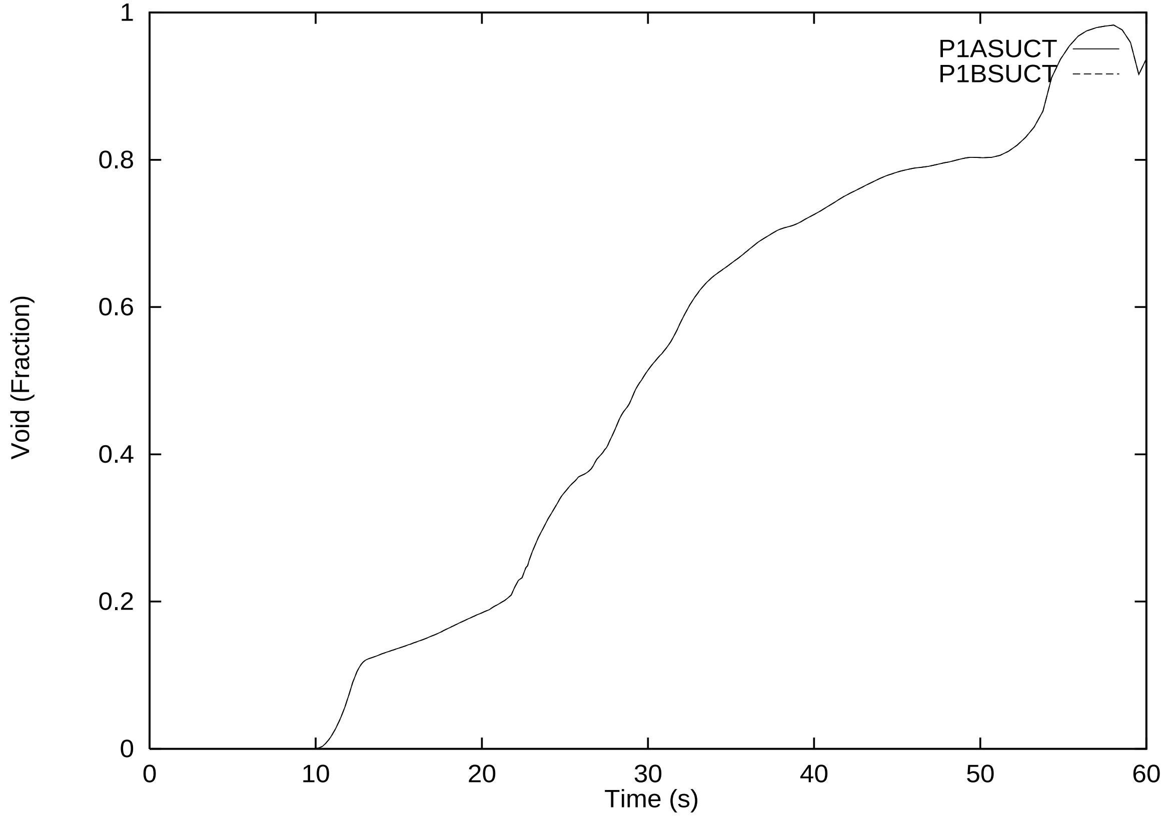


Figure 112 Pump Suction Voids for 25% RIH Break with Subsequent Loss of Class IV Power

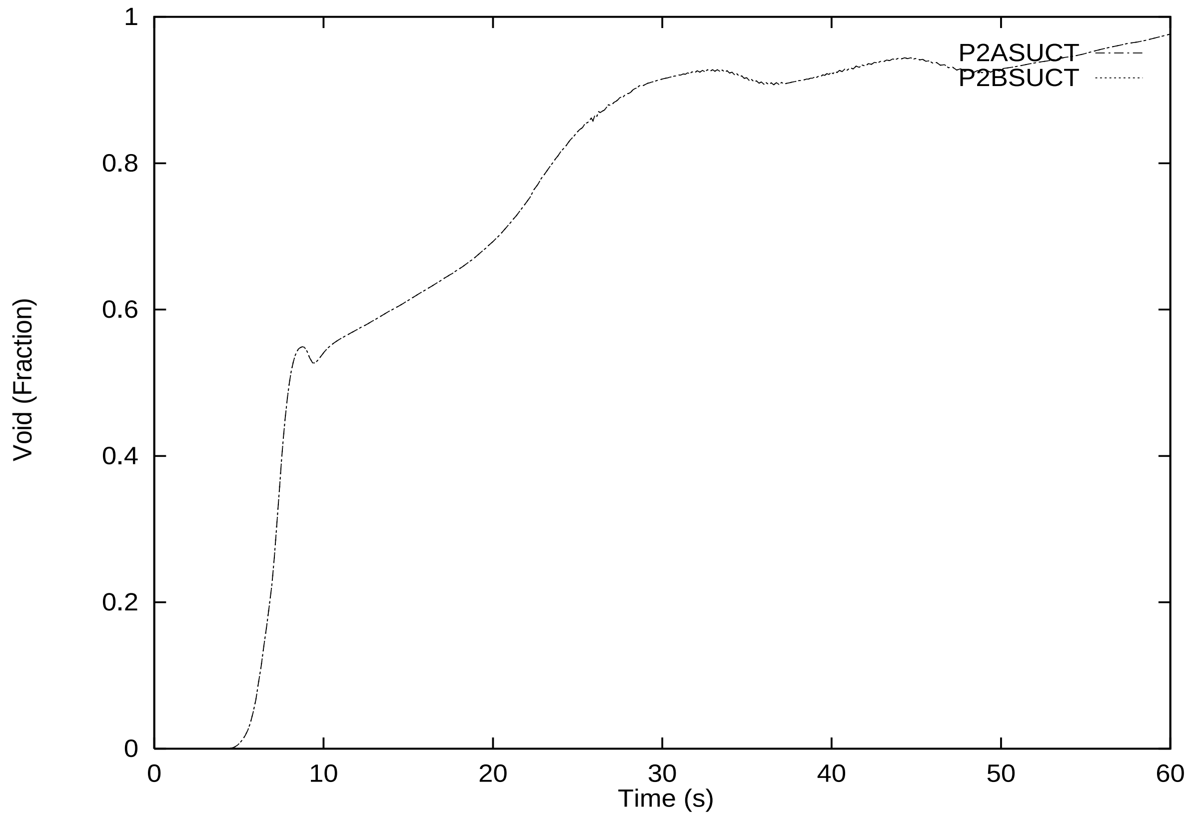


Figure 113 Pump Suction Voids for 25% RIH Break with Subsequent Loss of Class IV Power

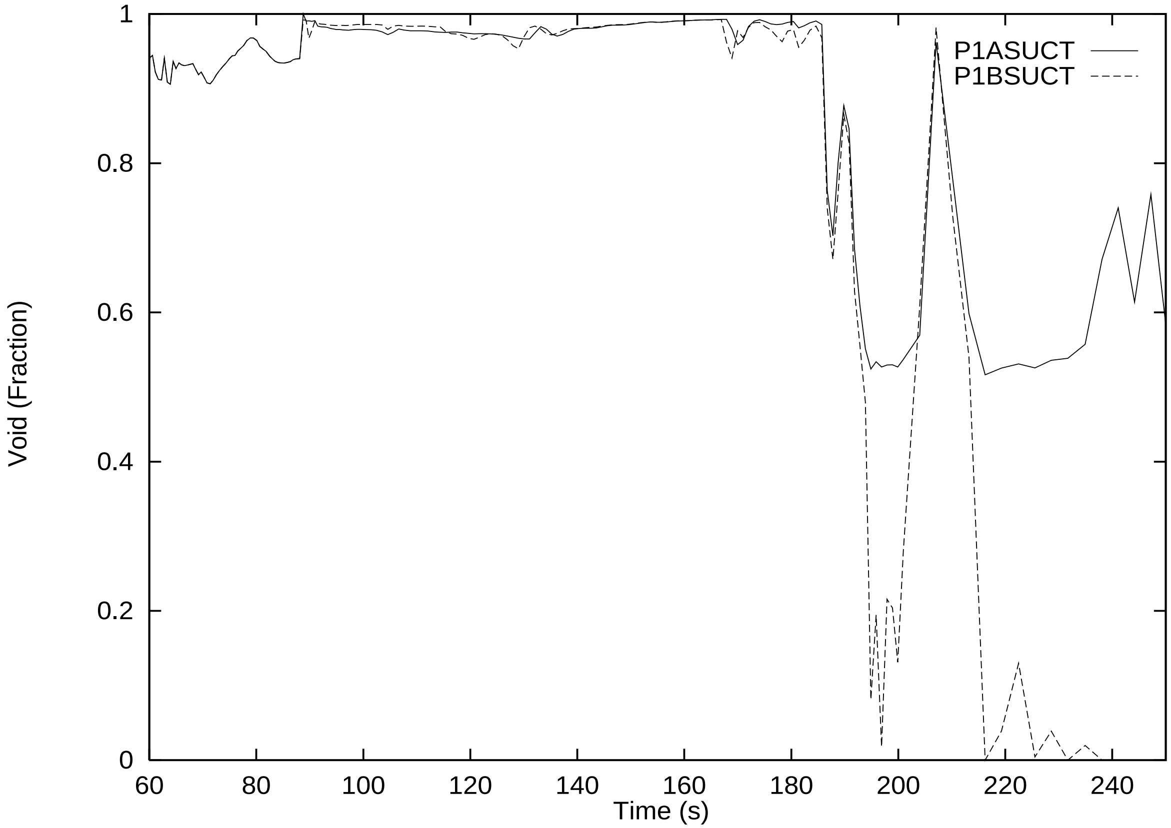


Figure 114 Pump Suction Voids for 25% RIH Break with Subsequent Loss of Class IV Power

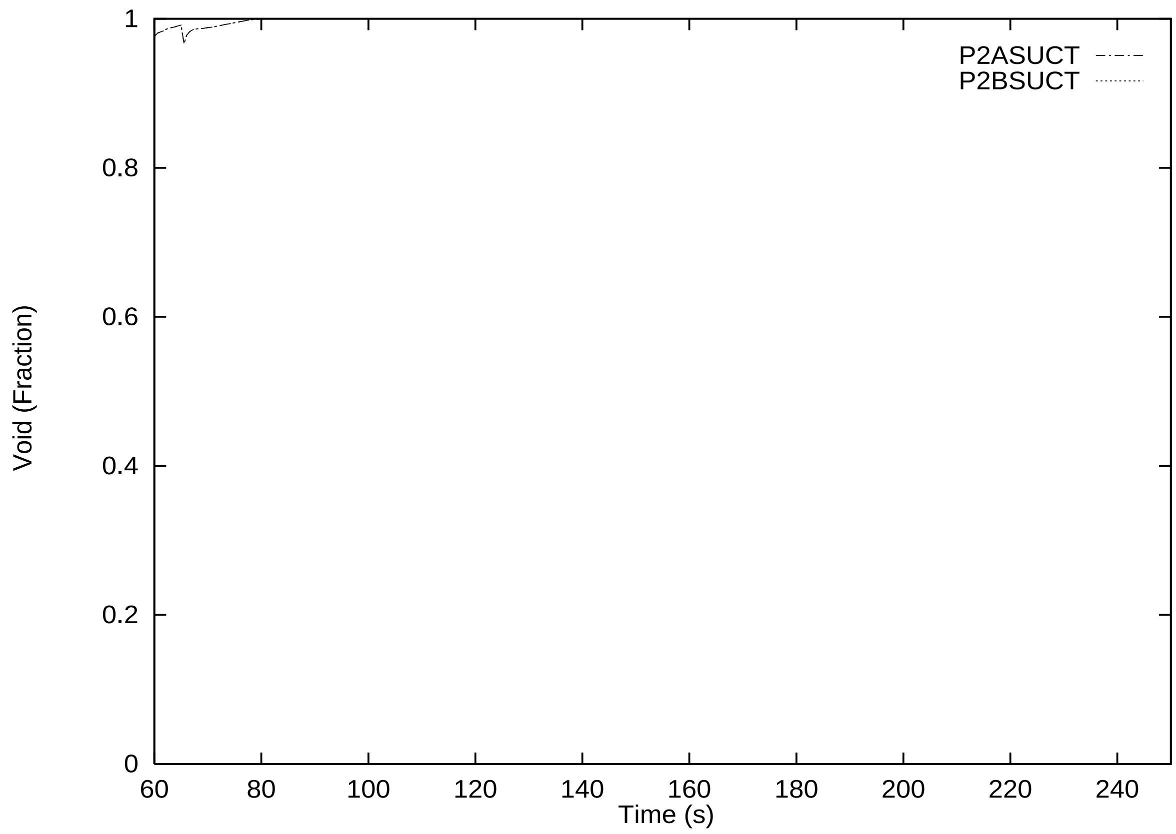


Figure 115 Pump Suction Voids for 25% RIH Break with Subsequent Loss of Class IV Power

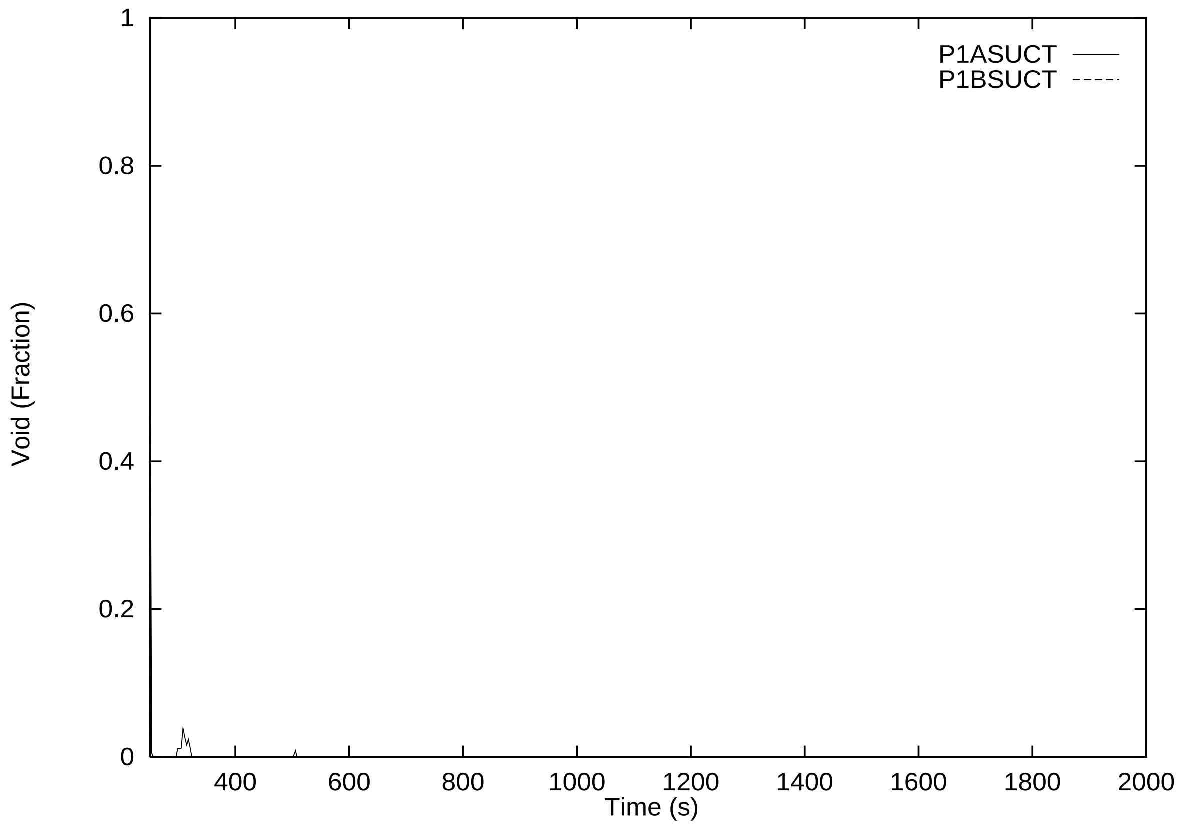


Figure 116 Pump Suction Voids for 25% RIH Break with Subsequent Loss of Class IV Power

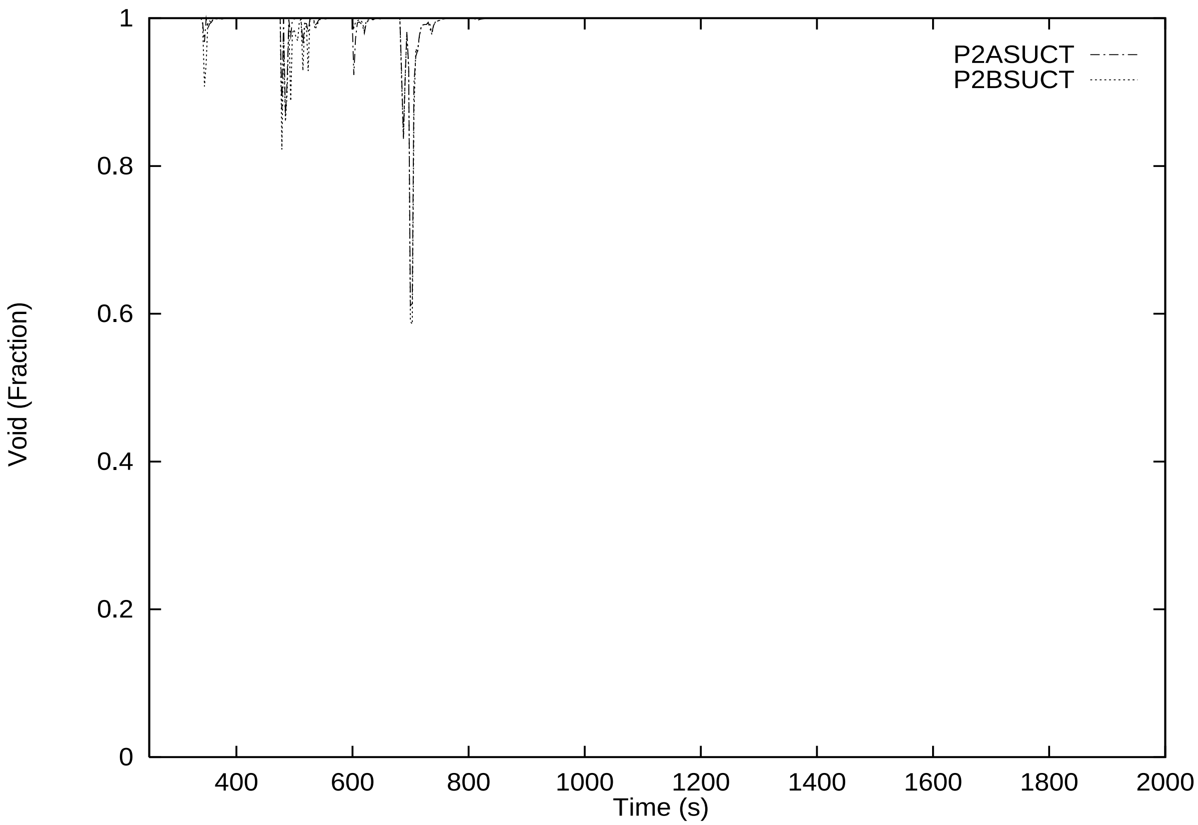


Figure 117 Pump Suction Voids for 25% RIH Break with Subsequent Loss of Class IV Power

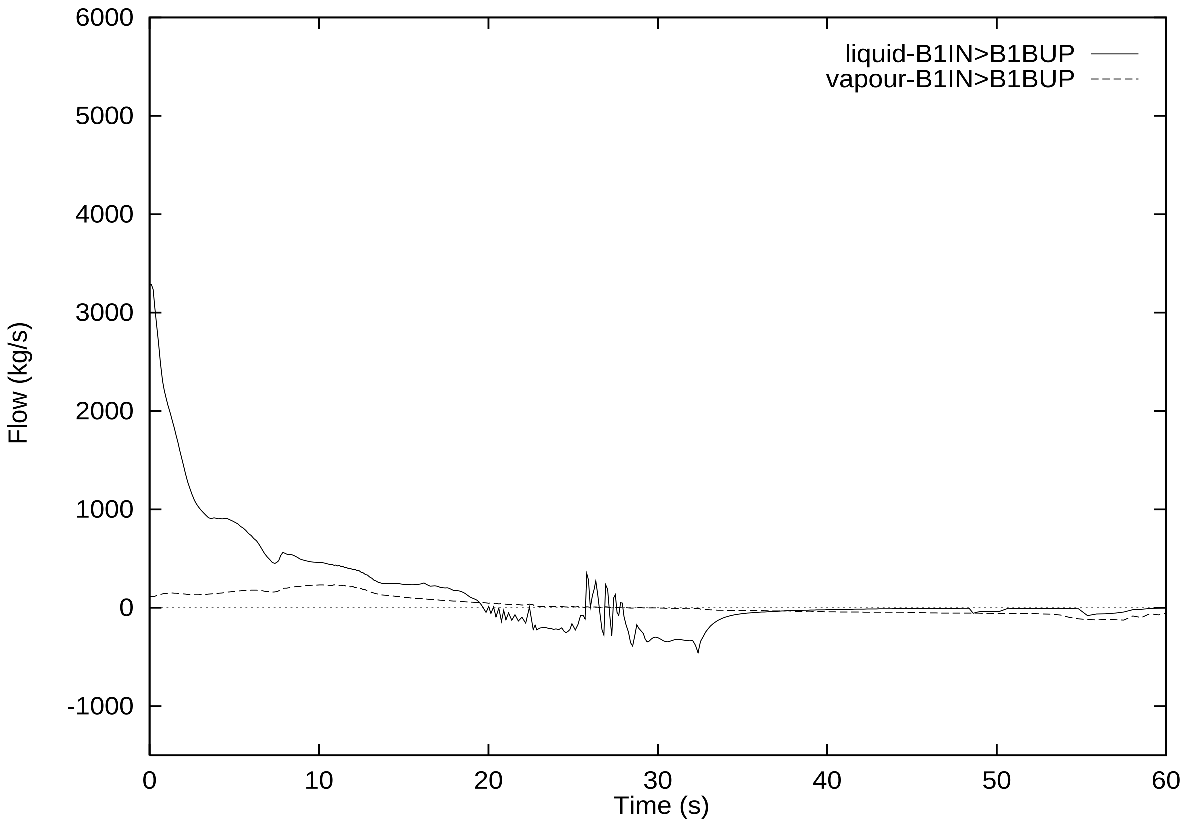


Figure 118 SG Inlet Plenum Flows for 25% RIH Break with Subsequent Loss of Class IV Power

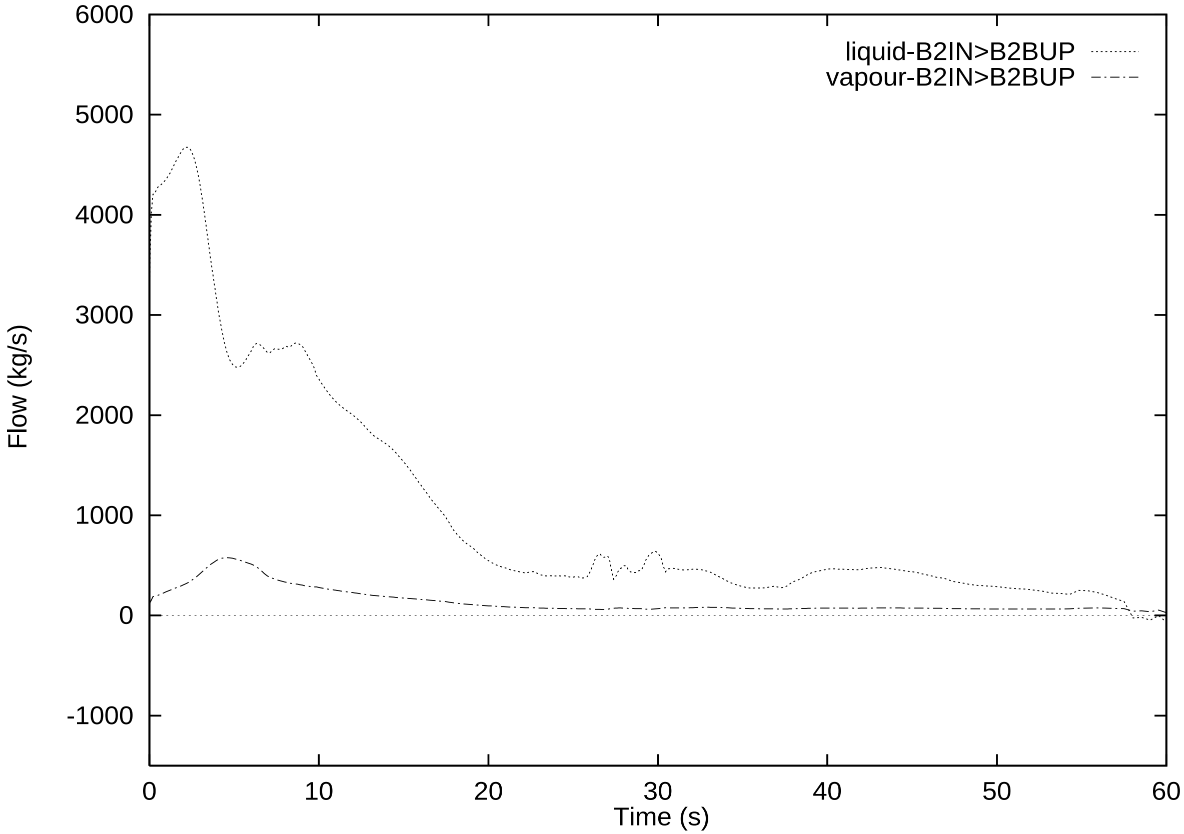


Figure 119 SG Inlet Plenum Flows for 25% RIH Break with Subsequent Loss of Class IV Power

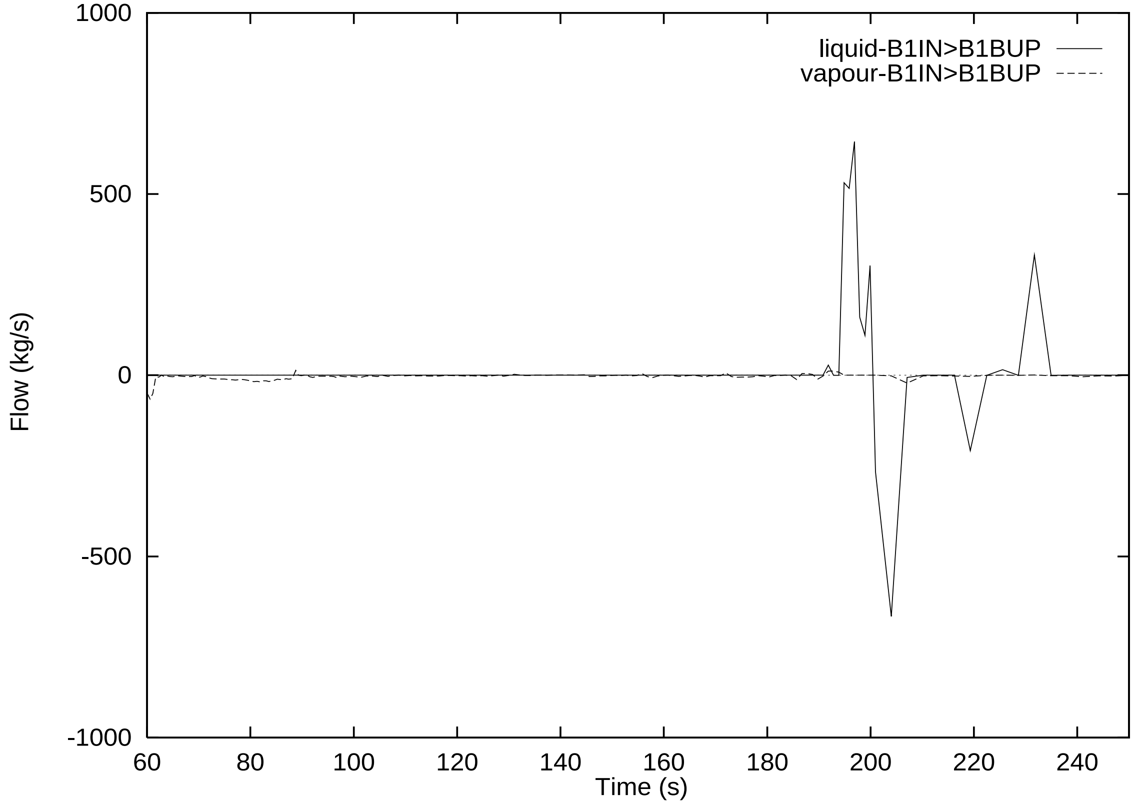


Figure 120 SG Inlet Plenum Flows for 25% RIH Break with Subsequent Loss of Class IV Power

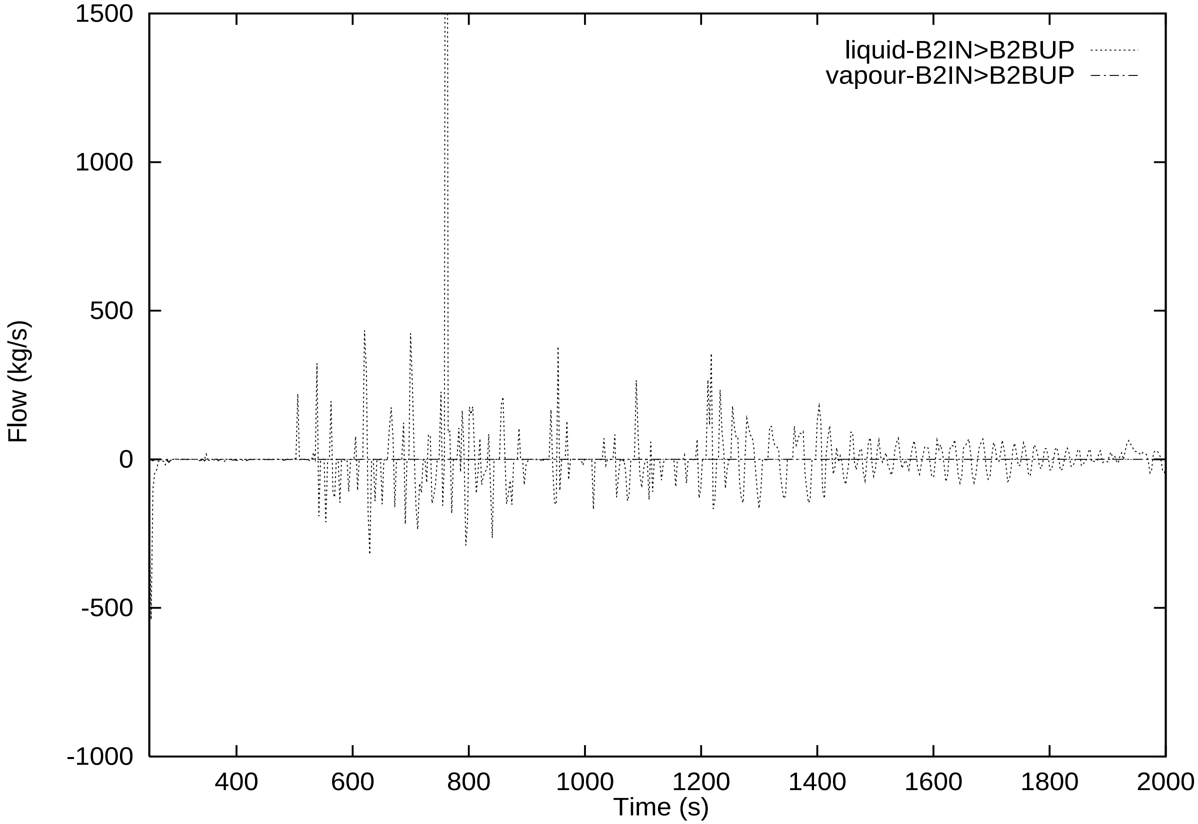


Figure 121 SG Inlet Plenum Flows for 25% RIH Break with Subsequent Loss of Class IV Power

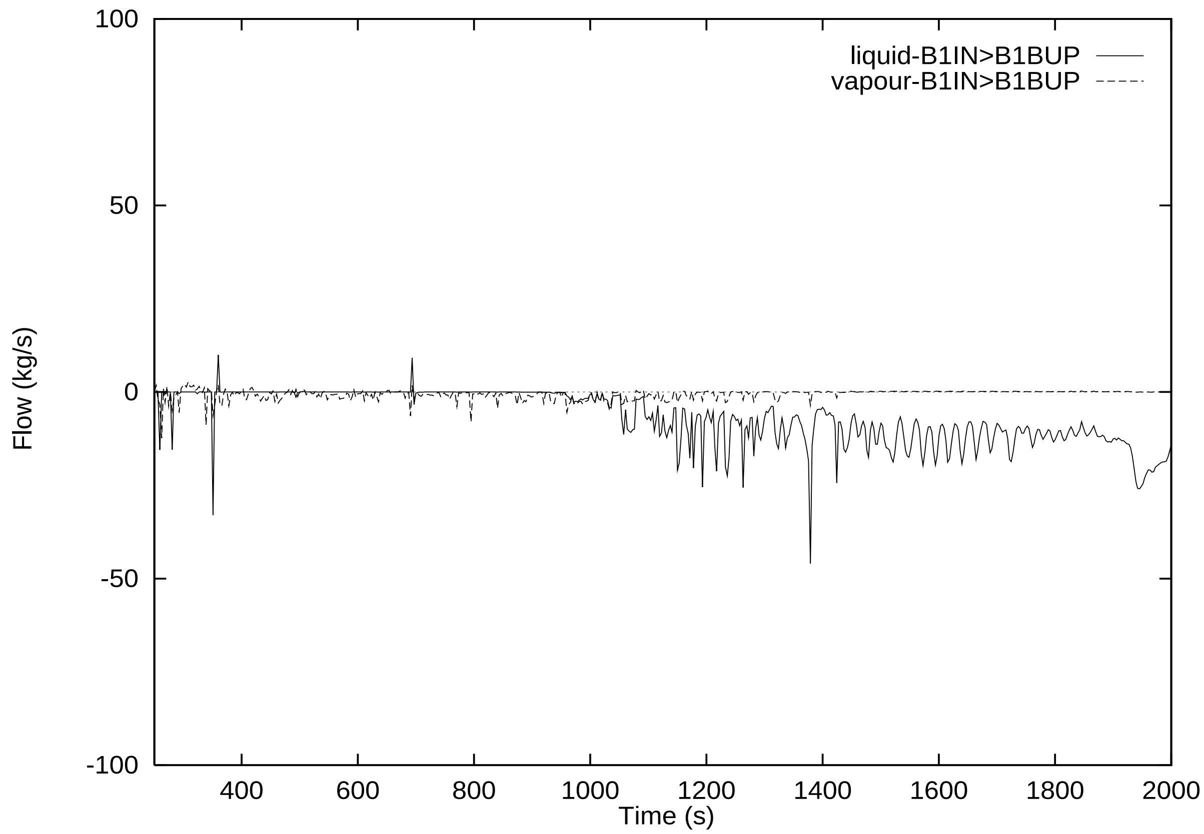


Figure 122 SG Inlet Plenum Flows for 25% RIH Break with Subsequent Loss of Class IV Power

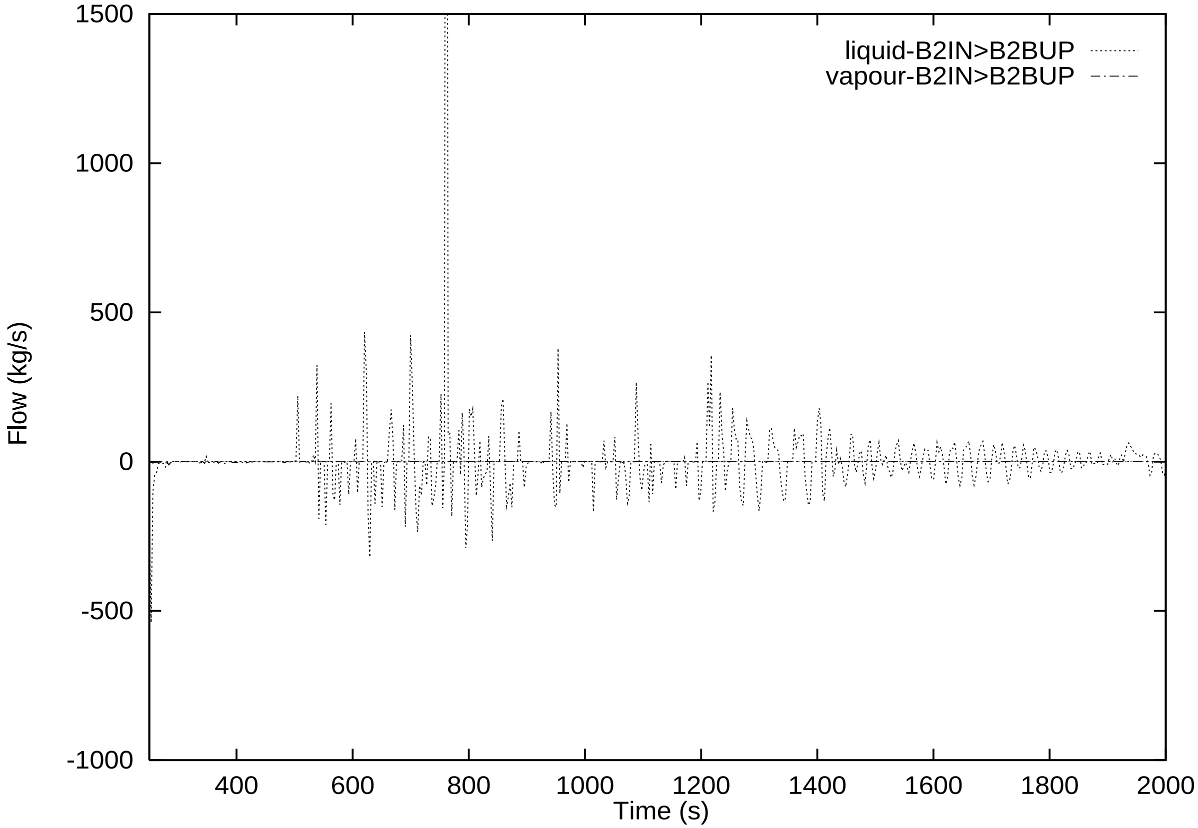


Figure 123 SG Inlet Plenum Flows for 25% RIH Break with Subsequent Loss of Class IV Power

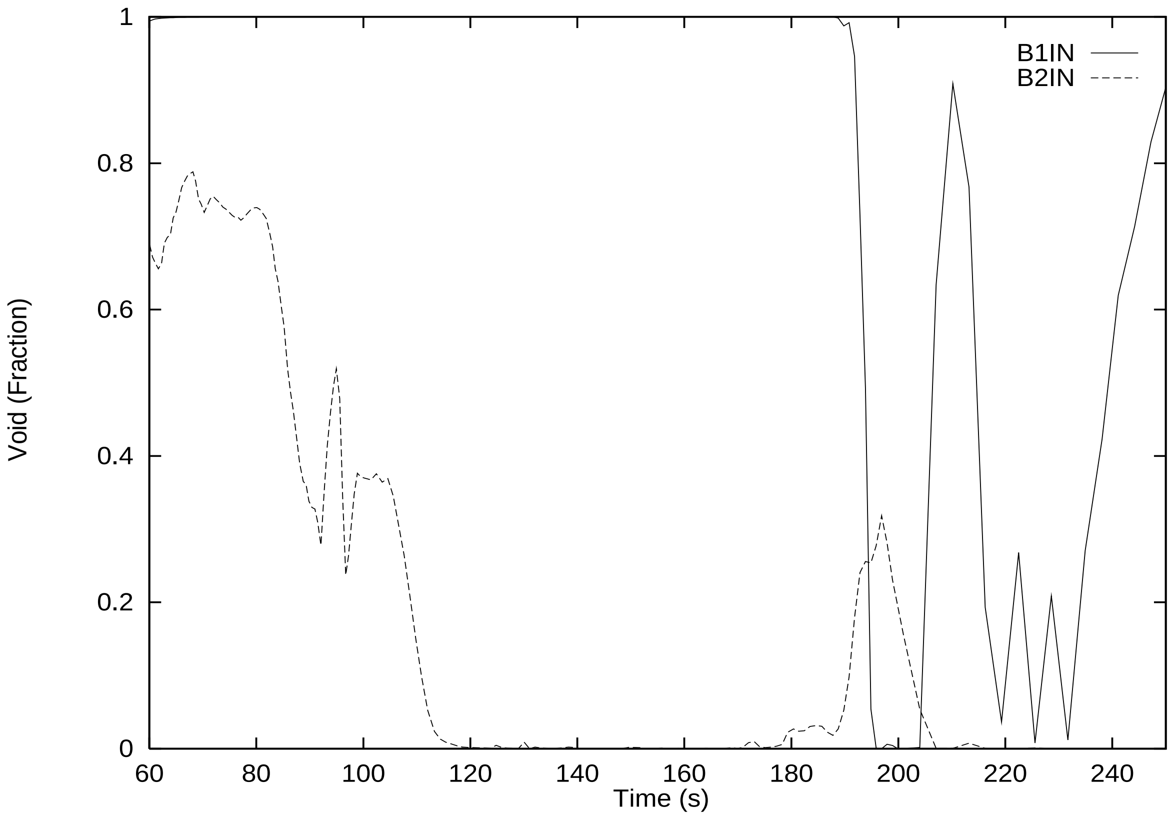


Figure 124 SG Inlet Plenum Voids for 25% RIH Break with Subsequent Loss of Class IV Power

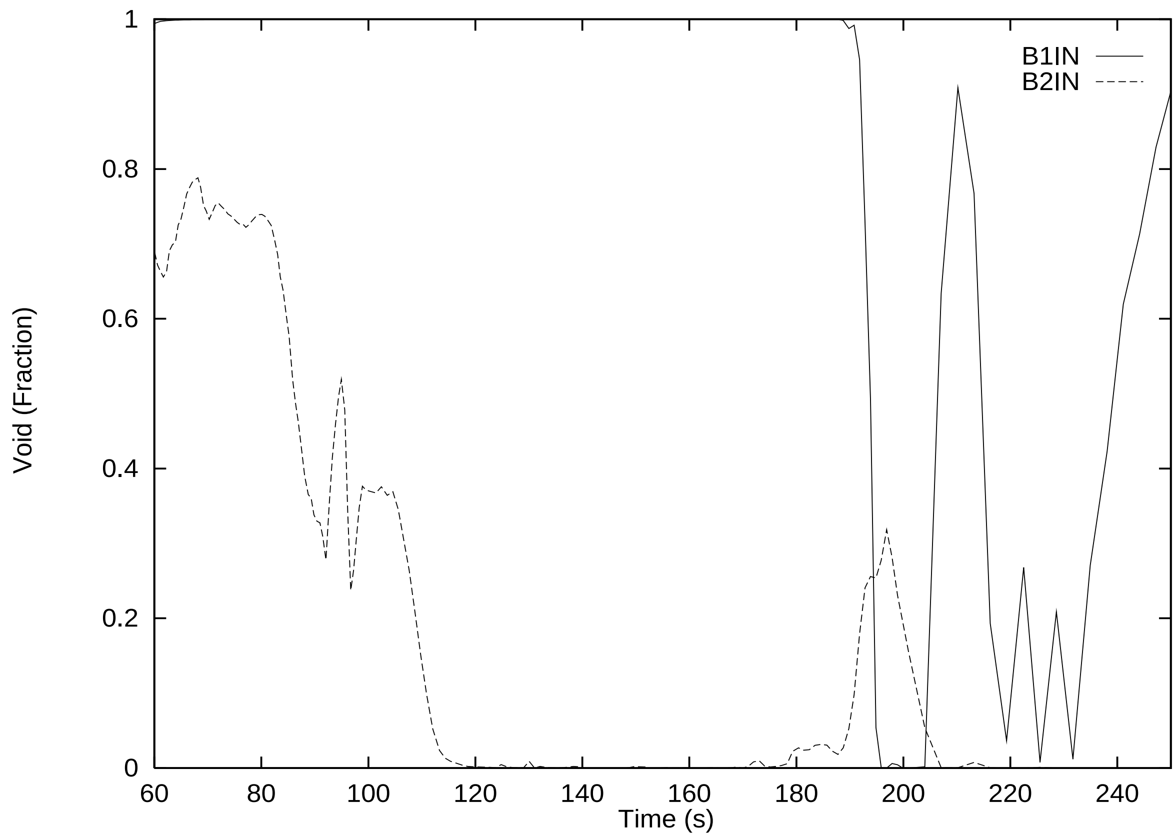


Figure 125 SG Inlet Plenum Voids for 25% RIH Break with Subsequent Loss of Class IV Power

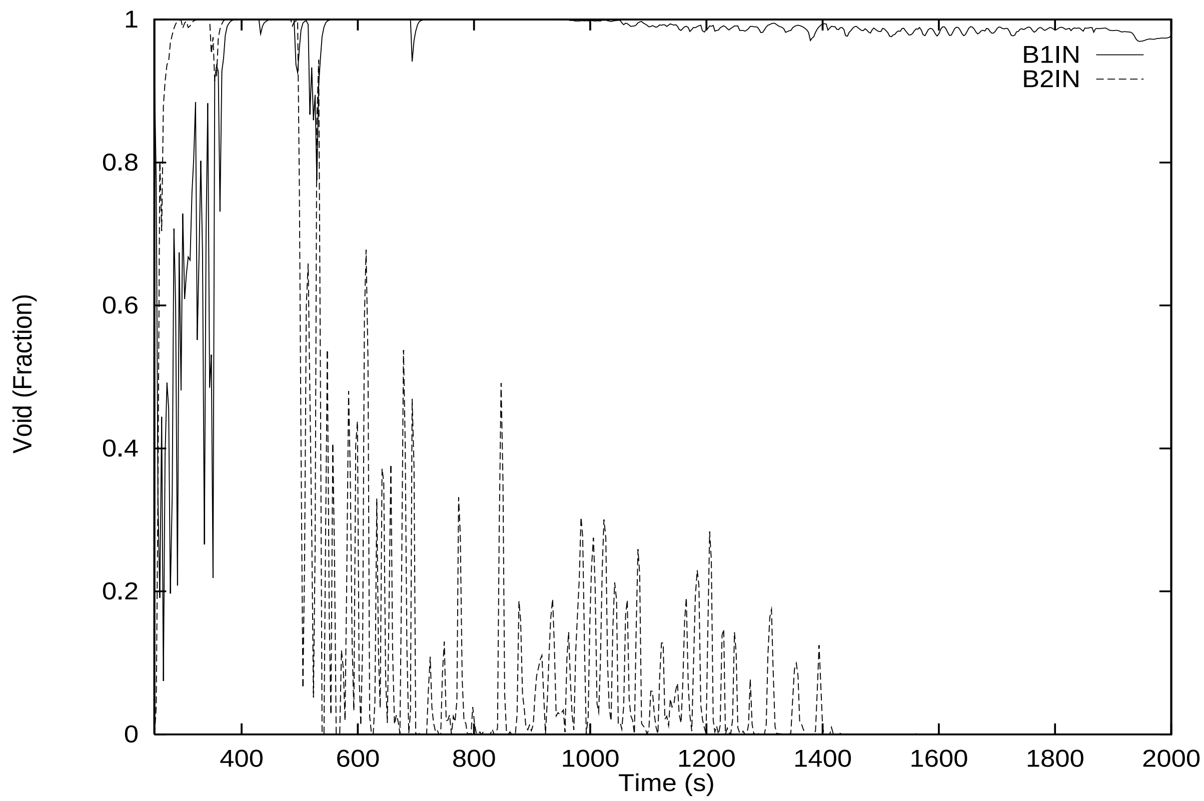


Figure 126 SG Inlet Plenum Voids for 25% RIH Break with Subsequent Loss of Class IV Power

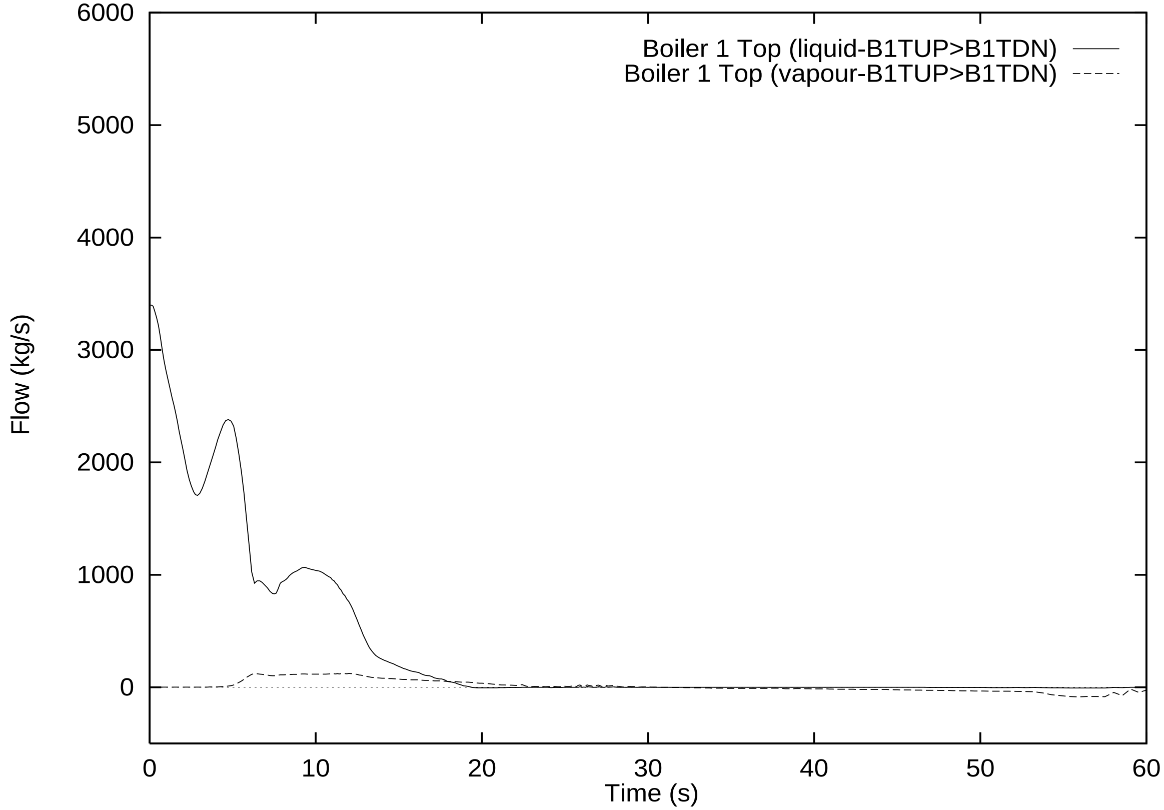


Figure 127 SG Primary Side Flows for 25% RIH Break with Subsequent Loss of Class IV Power

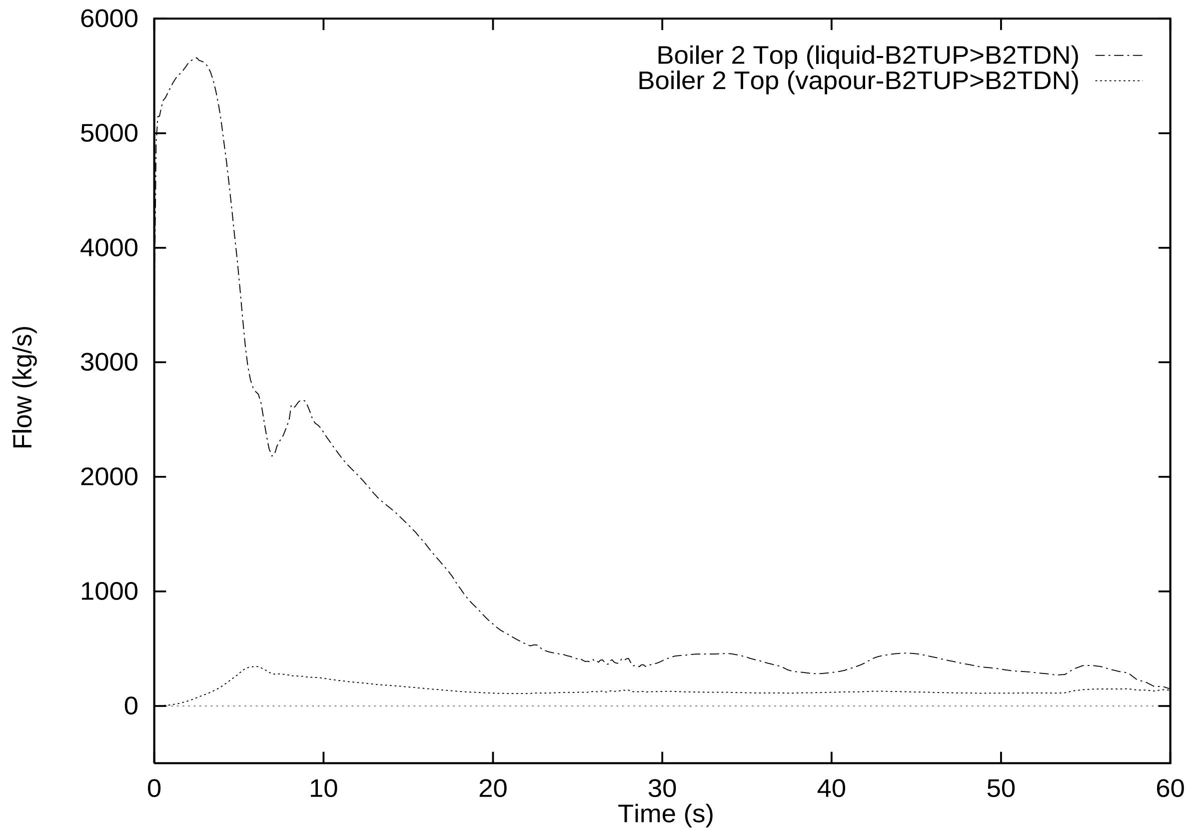


Figure 128 SG Primary Side Flows for 25% RIH Break with Subsequent Loss of Class IV Power

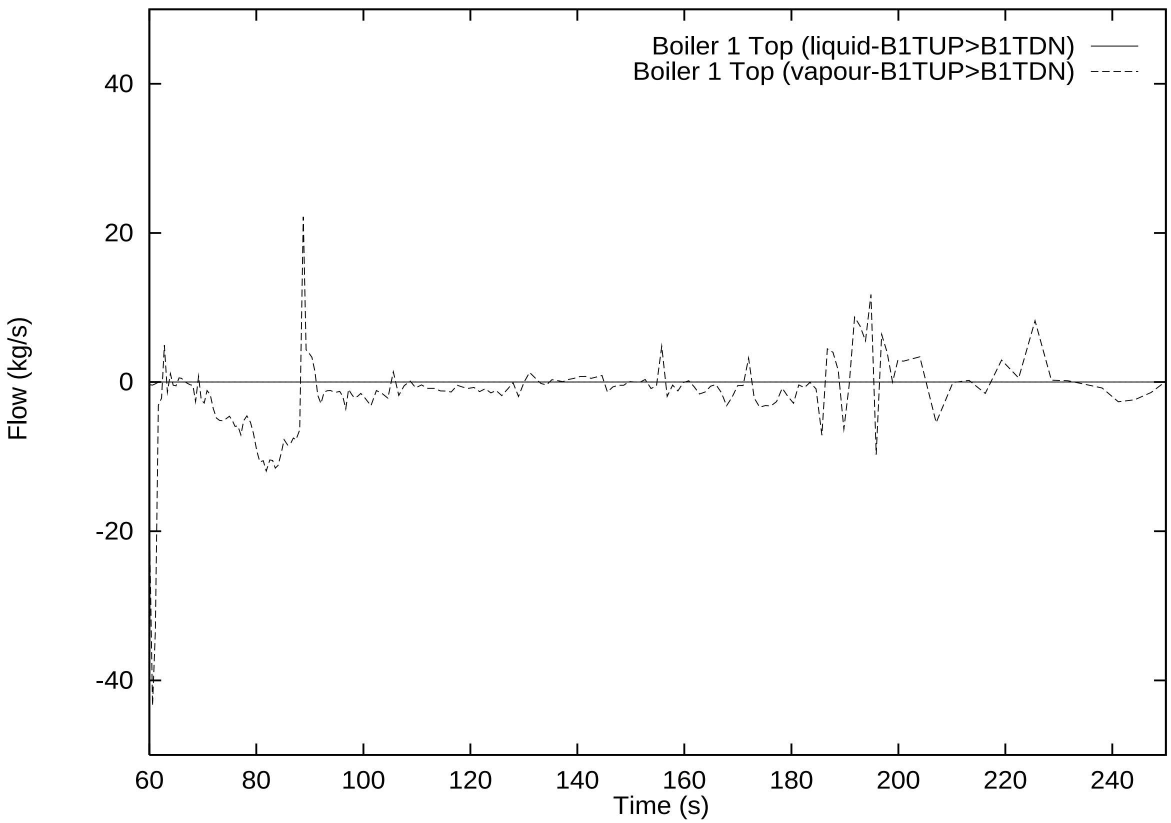


Figure 129 SG Primary Side Flows for 25% RIH Break with Subsequent Loss of Class IV Power

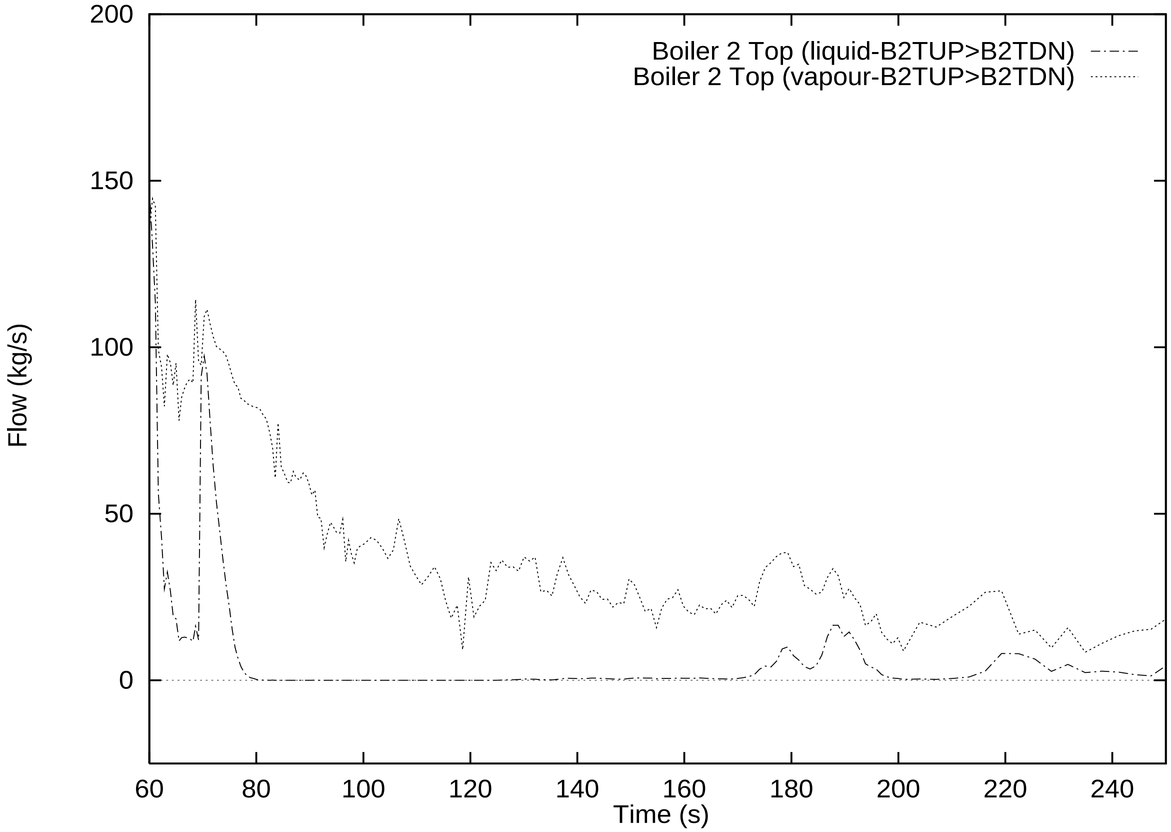


Figure 130 SG Primary Side Flows for 25% RIH Break with Subsequent Loss of Class IV Power

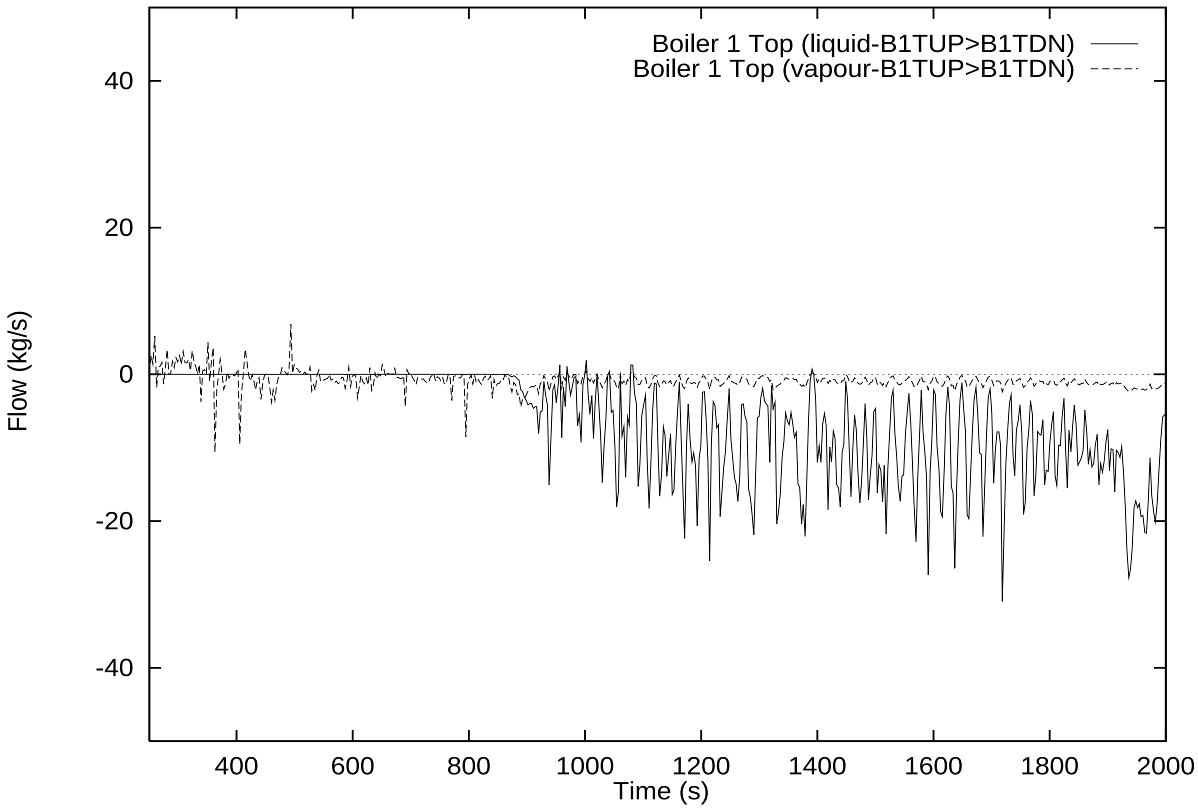


Figure 131 SG Primary Side Flows for 25% RIH Break with Subsequent Loss of Class IV Power

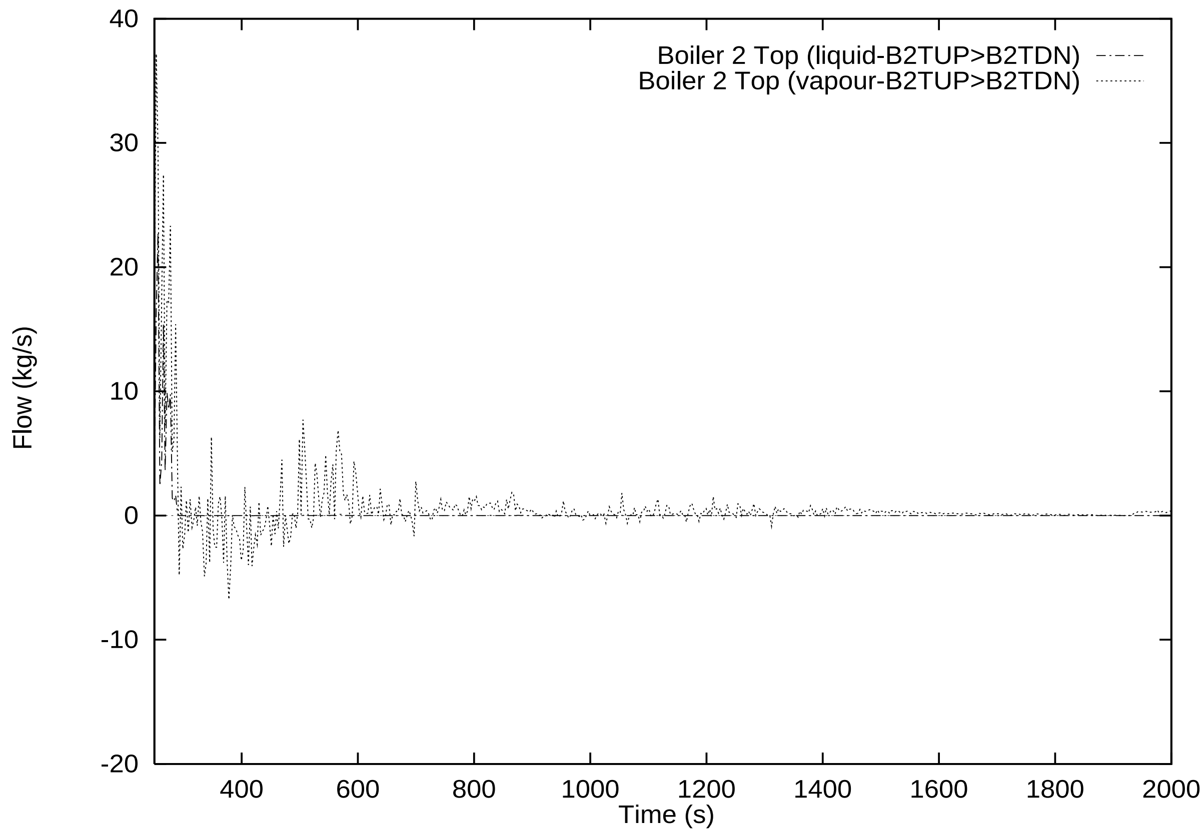


Figure 132 SG Primary Side Flows for 25% RIH Break with Subsequent Loss of Class IV Power

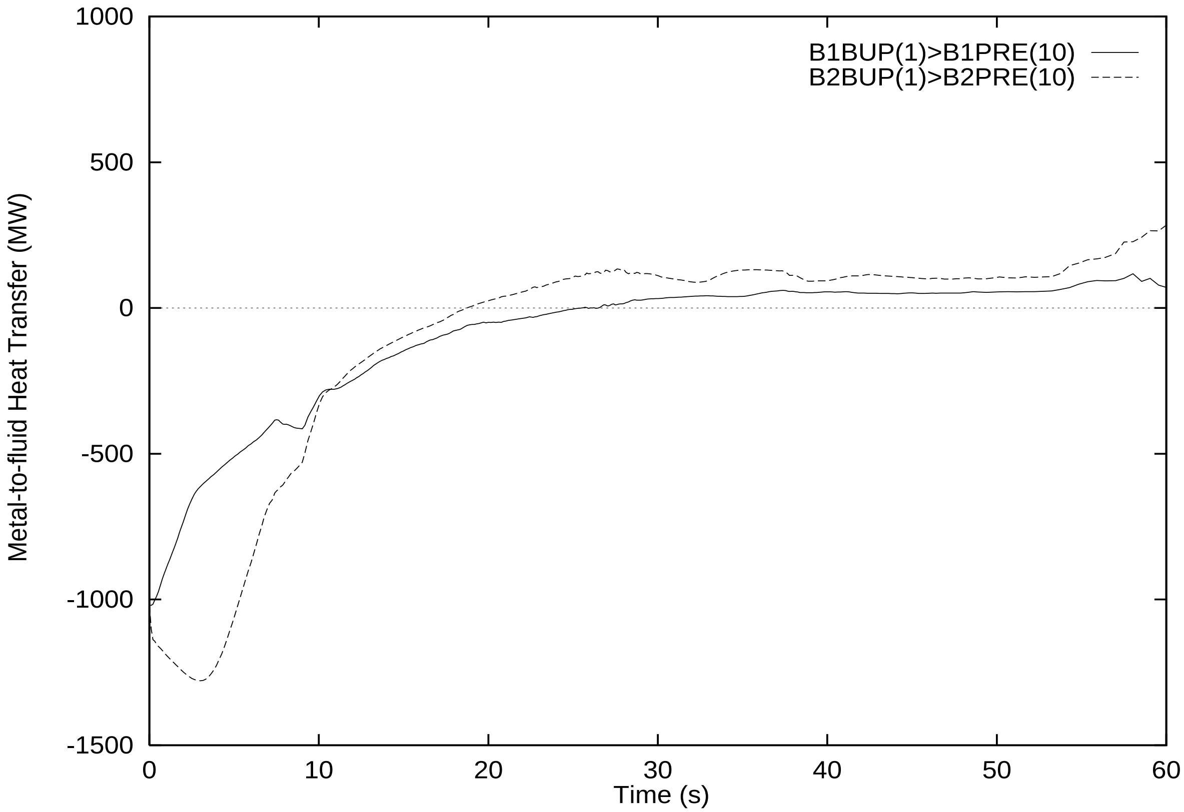


Figure 133 SG Primary Side Metal-to-fluid Heat Transfer for 25% RIH Break with Subsequent Loss of Class IV Power

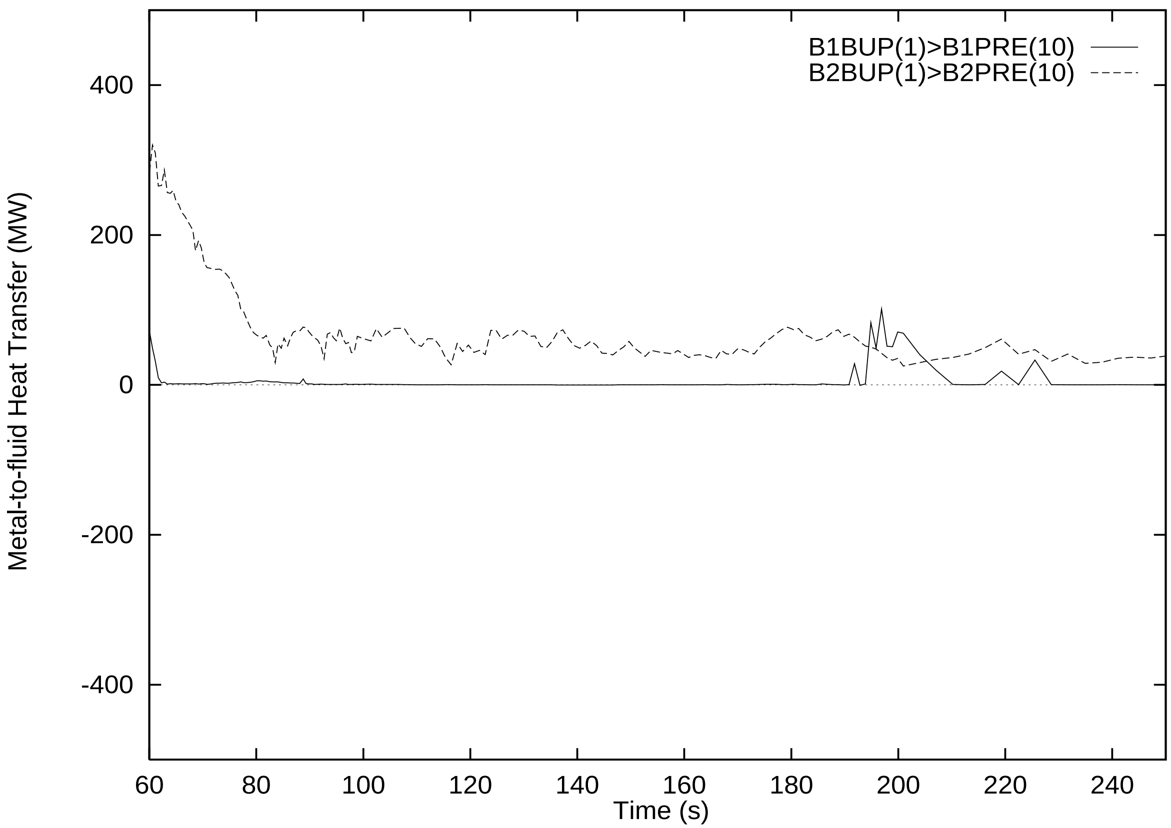


Figure 134 SG Primary Side Metal-to-fluid Heat Transfer for 25% RIH Break with Subsequent Loss of Class IV Power

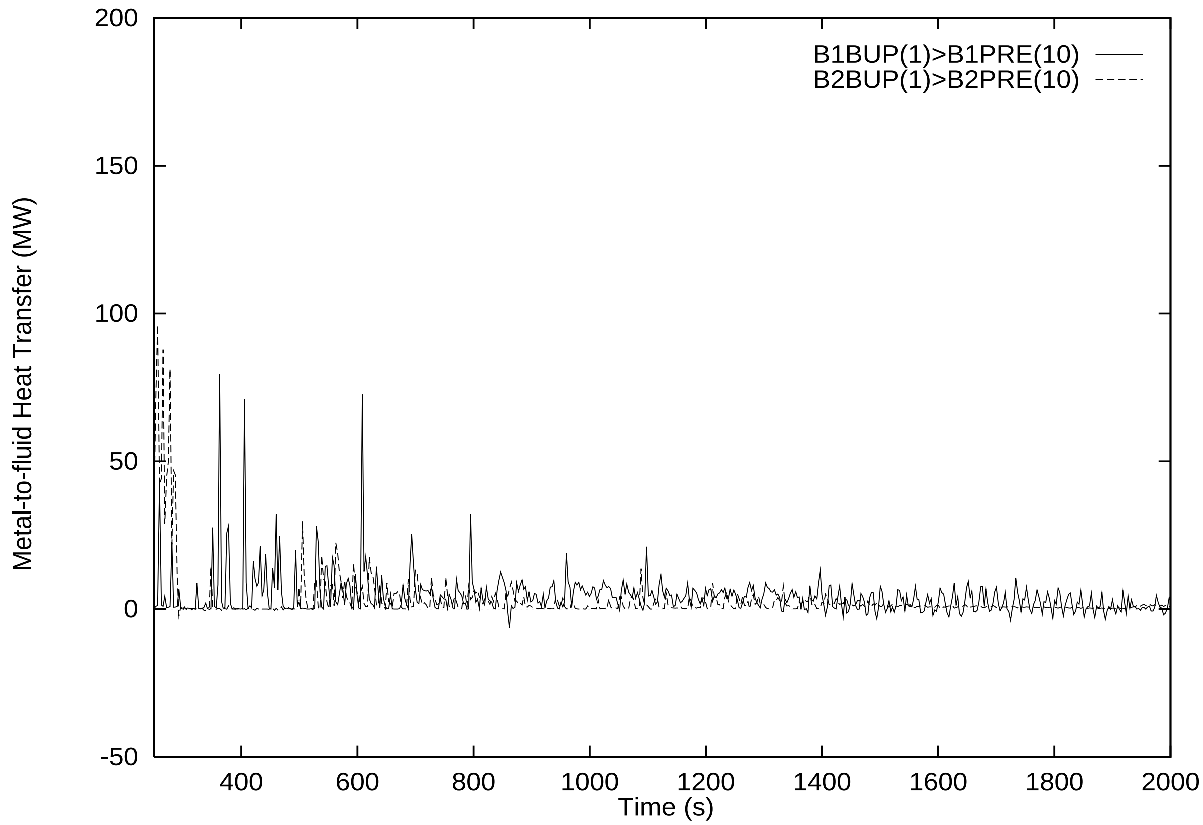


Figure 135 SG Primary Side Metal-to-fluid Heat Transfer for 25% RIH Break with Subsequent Loss of Class IV Power

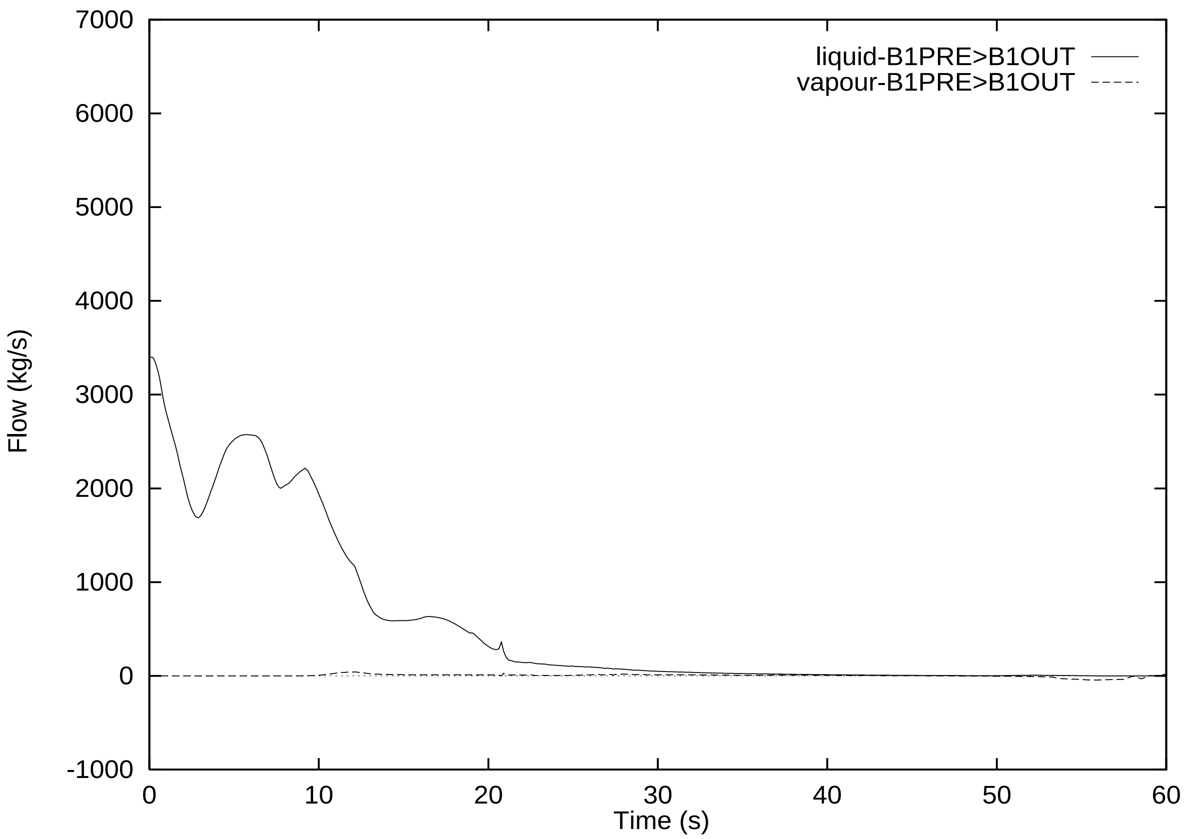


Figure 136 SG Outlet Plenum Flows for 25% RIH Break with Subsequent Loss of Class IV Power

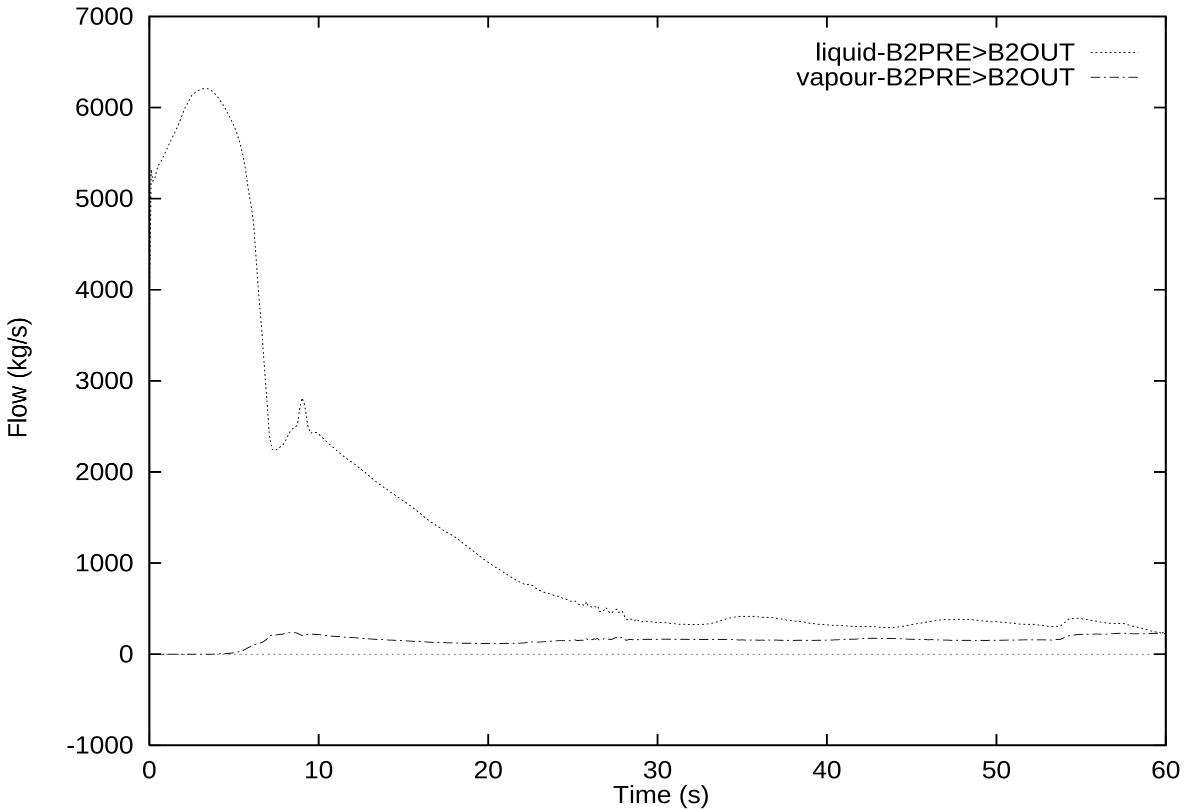


Figure 137 SG Outlet Plenum Flows for 25% RIH Break with Subsequent Loss of Class IV Power

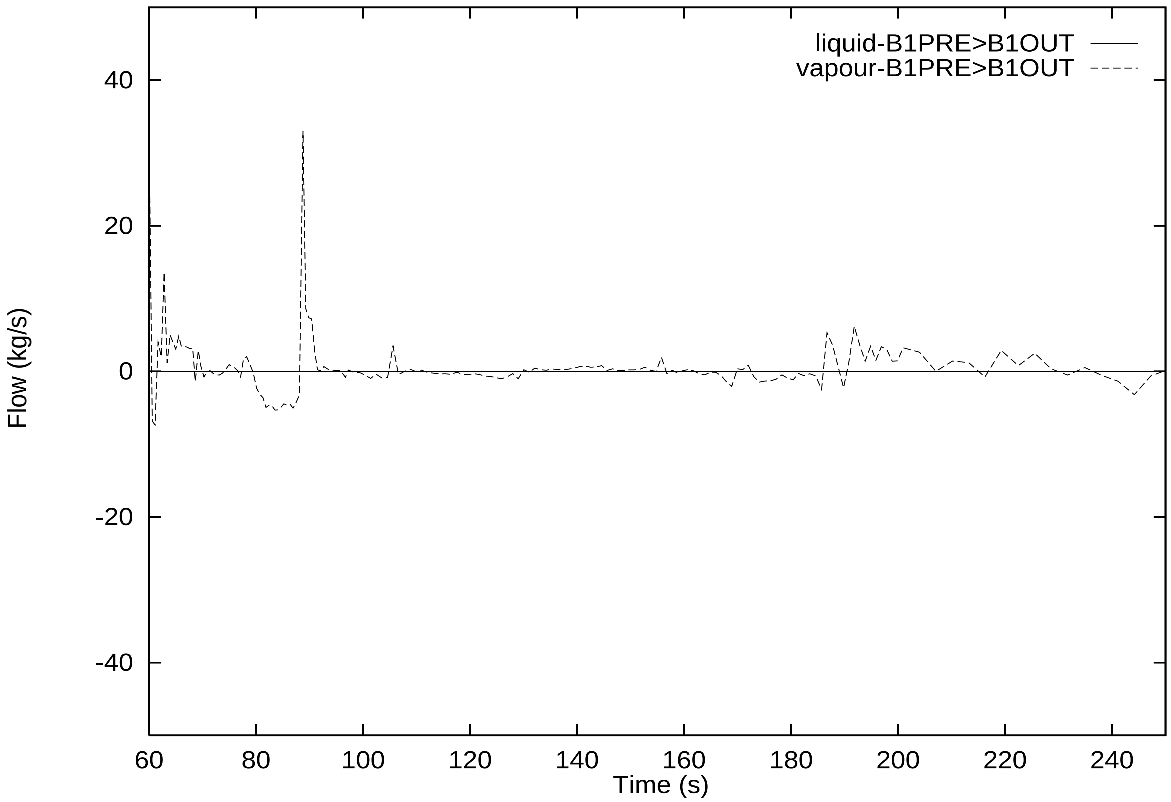


Figure 138 SG Outlet Plenum Flows for 25% RIH Break with Subsequent Loss of Class IV Power

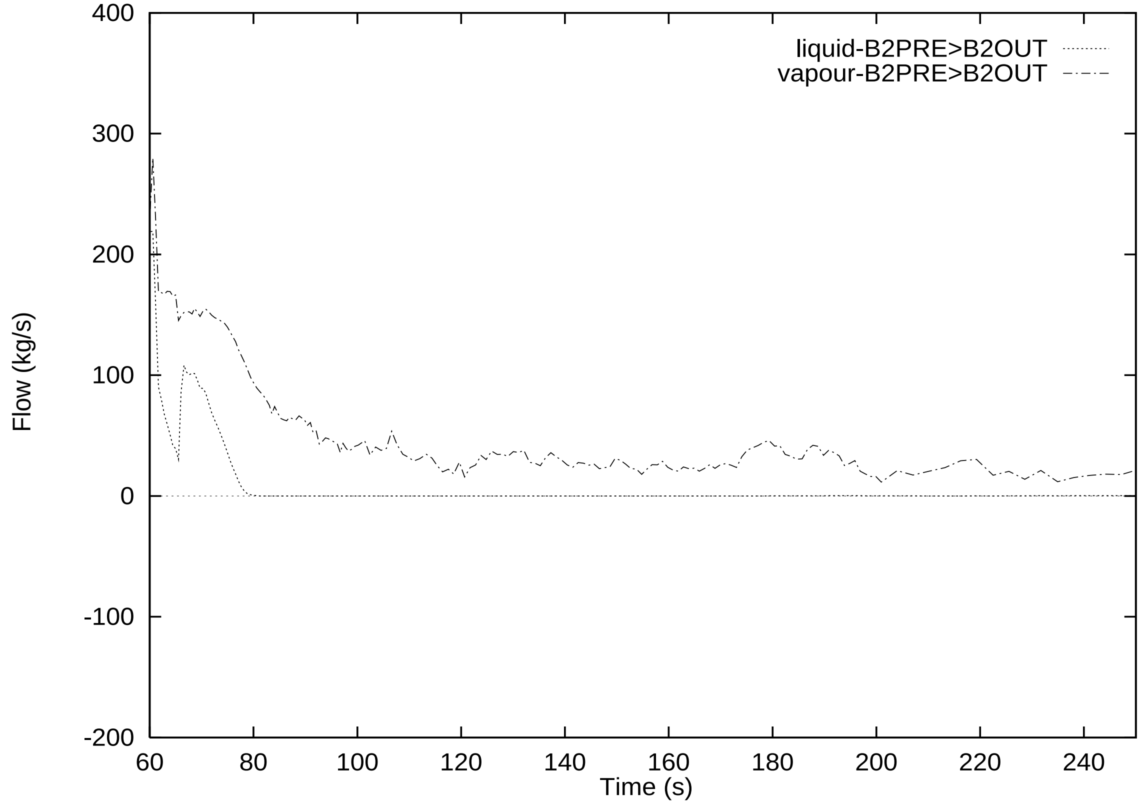


Figure 139 SG Outlet Plenum Flows for 25% RIH Break with Subsequent Loss of Class IV Power

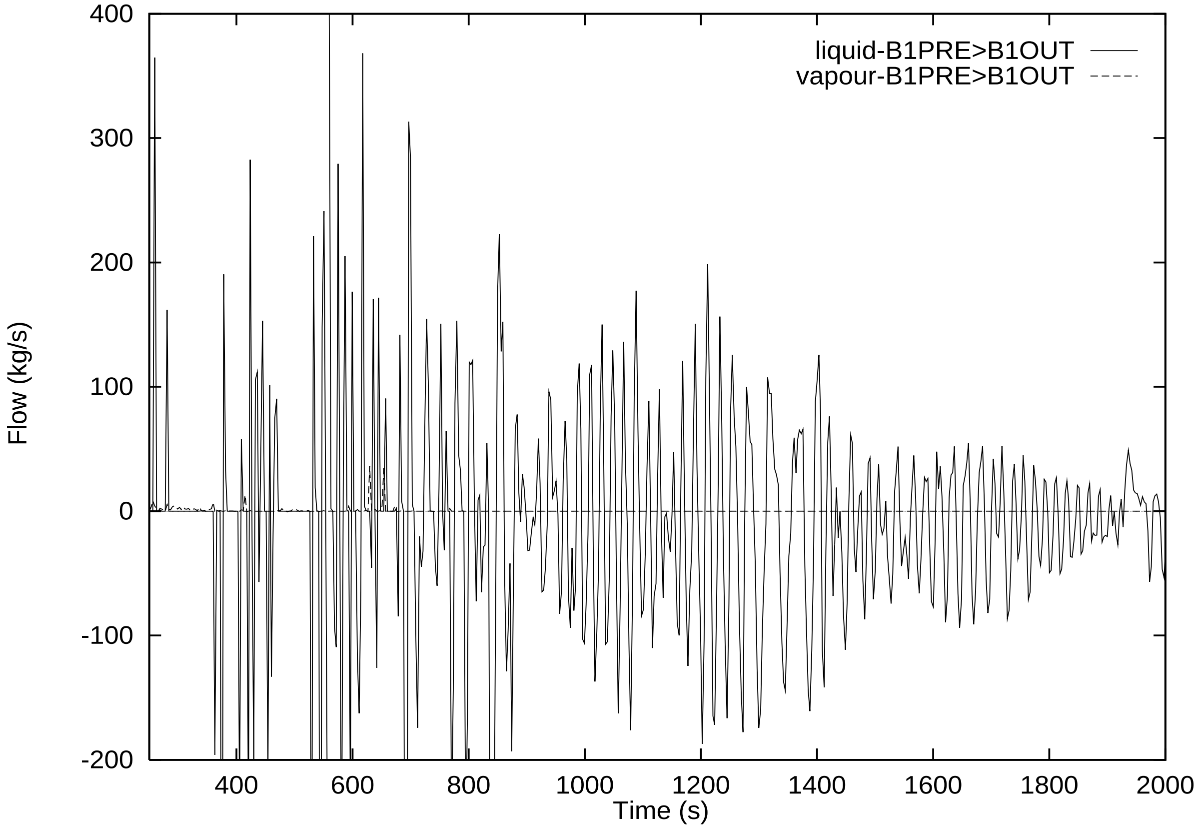


Figure 140 SG Outlet Plenum Flows for 25% RIH Break with Subsequent Loss of Class IV Power

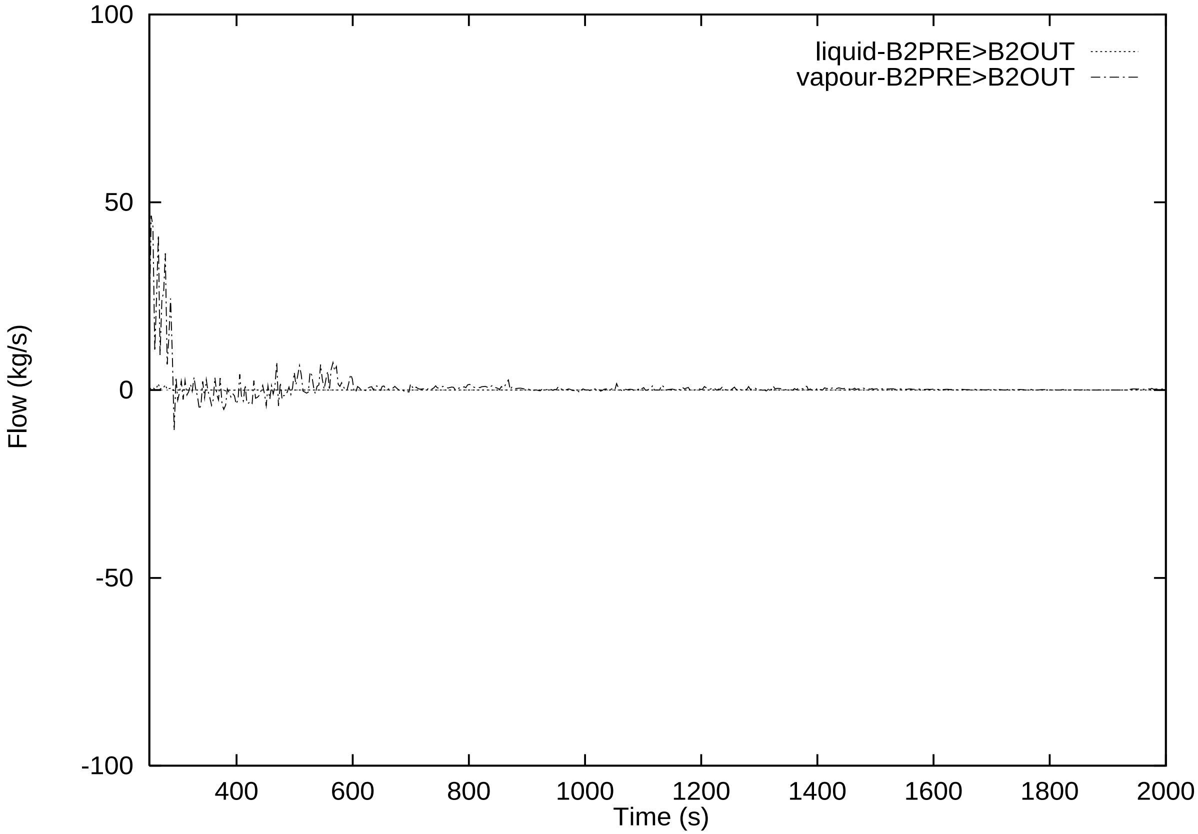


Figure 141 SG Outlet Plenum Flows for 25% RIH Break with Subsequent Loss of Class IV Power

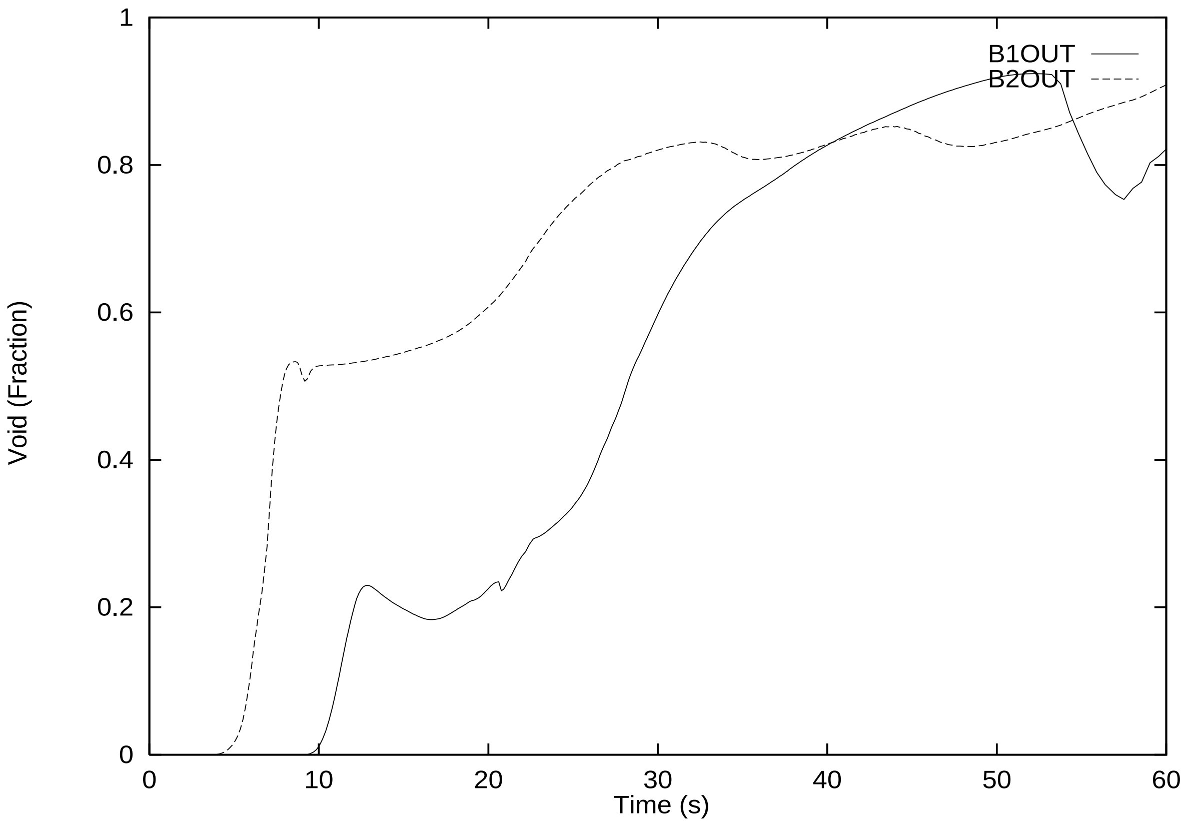


Figure 142 SG Outlet Plenum Voids for 25% RIH Break with Subsequent Loss of Class IV Power

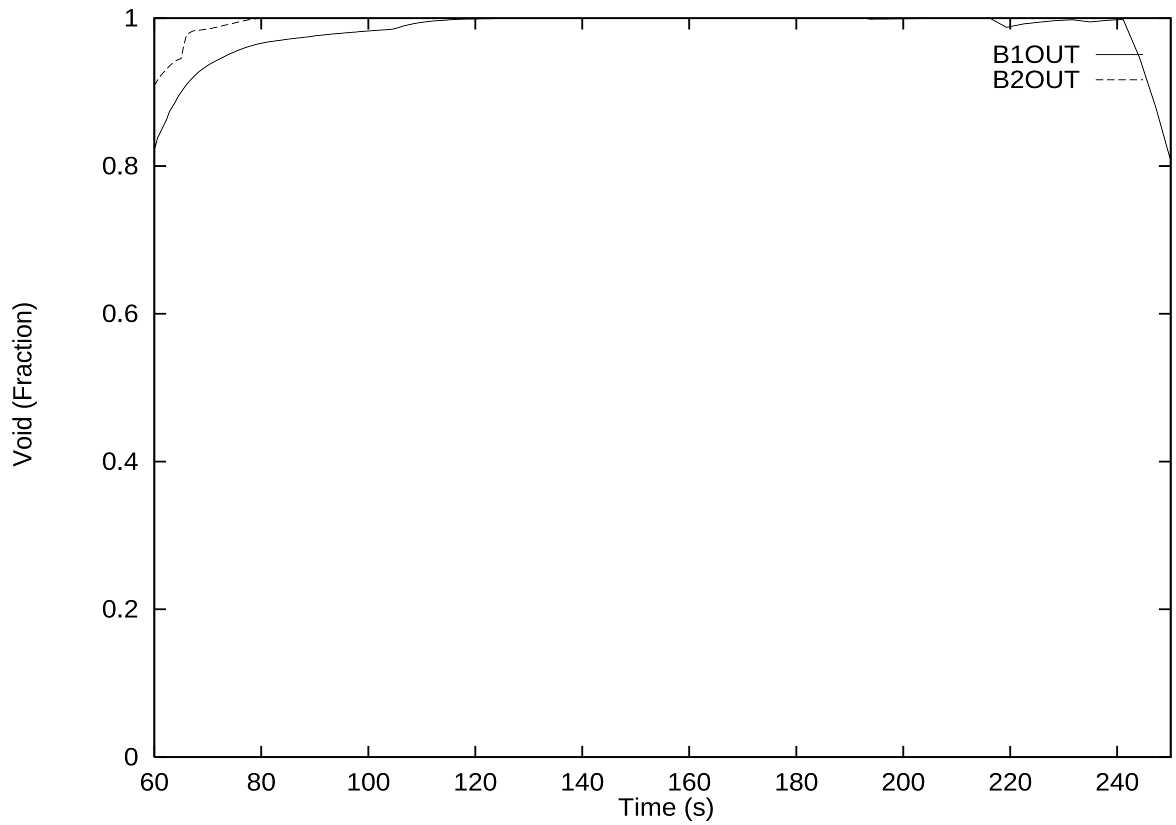


Figure 143 SG Outlet Plenum Voids for 25% RIH Break with Subsequent Loss of Class IV Power

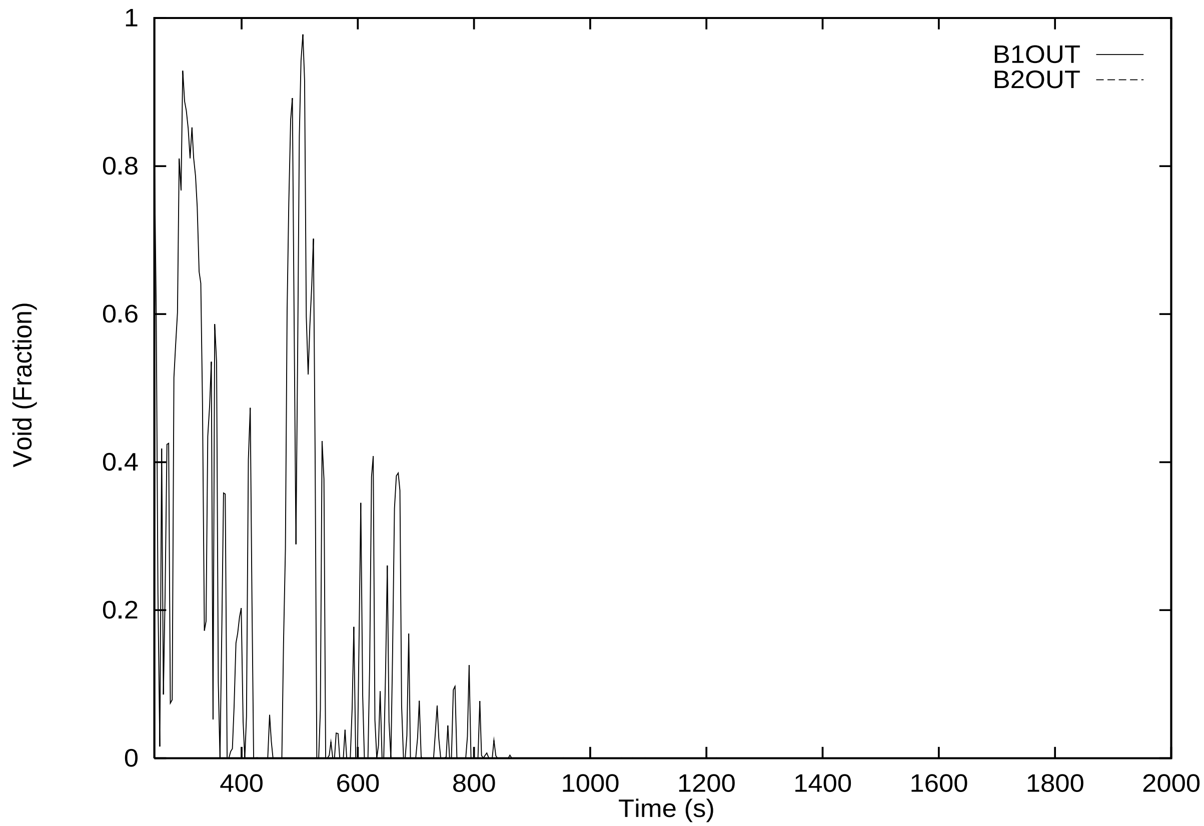


Figure 144 SG Outlet Plenum Voids for 25% RIH Break with Subsequent Loss of Class IV Power

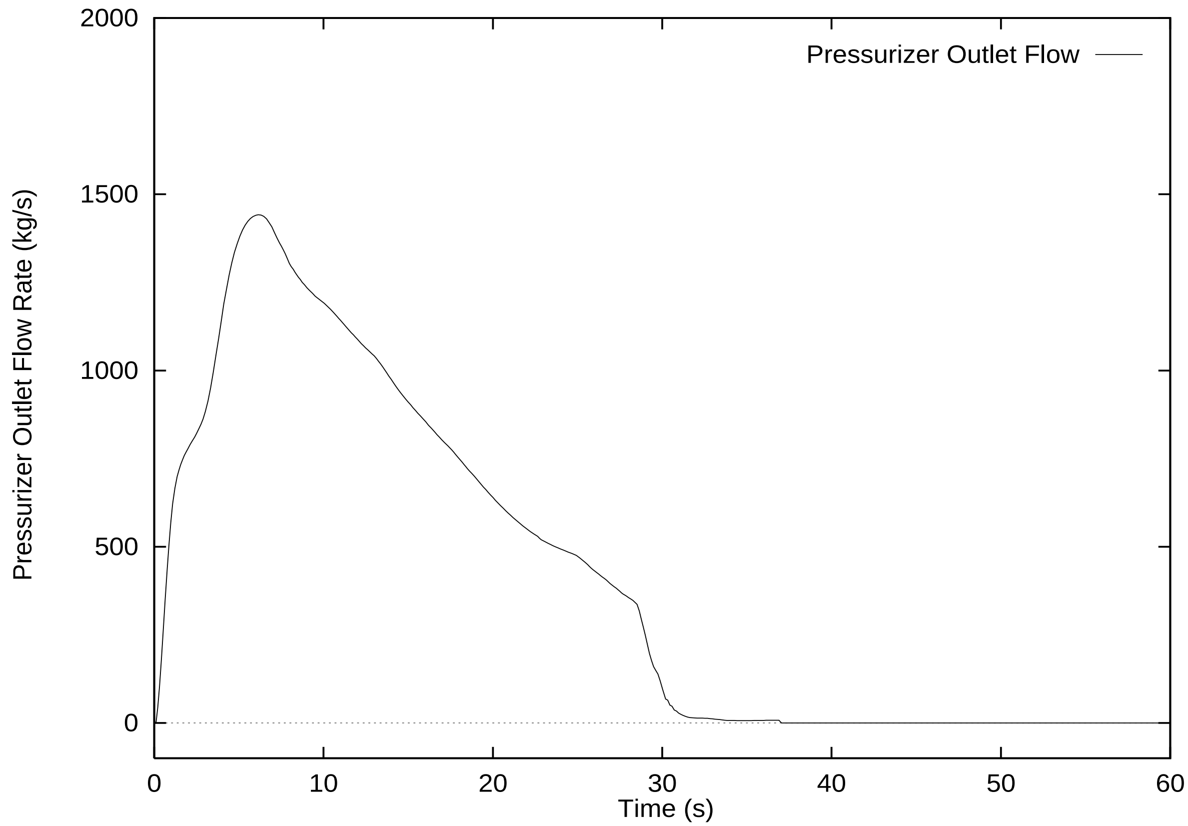


Figure 145 Pressurizer Outlet Flow Rate for 25% RIH Break with a Subsequent Loss of Class IV Power

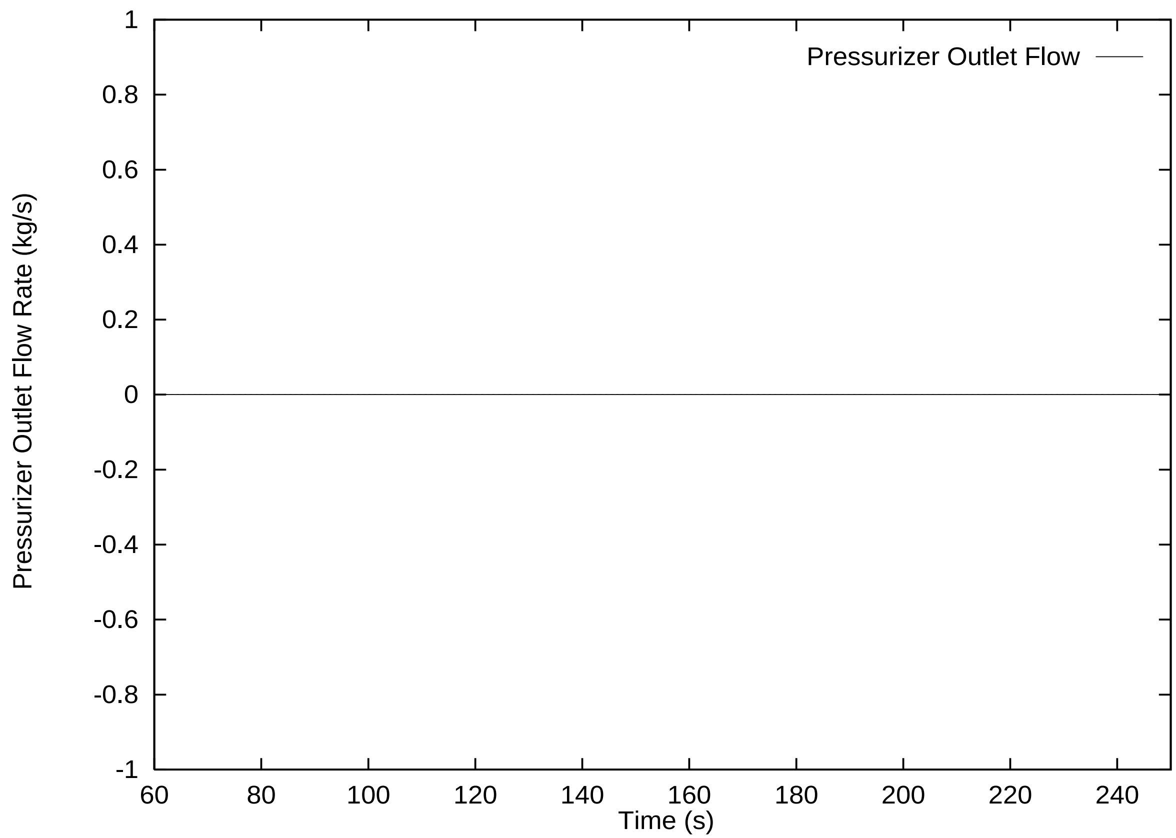


Figure 146 Pressurizer Outlet Flow Rate for 25% RIH Break with a Subsequent Loss of Class IV Power

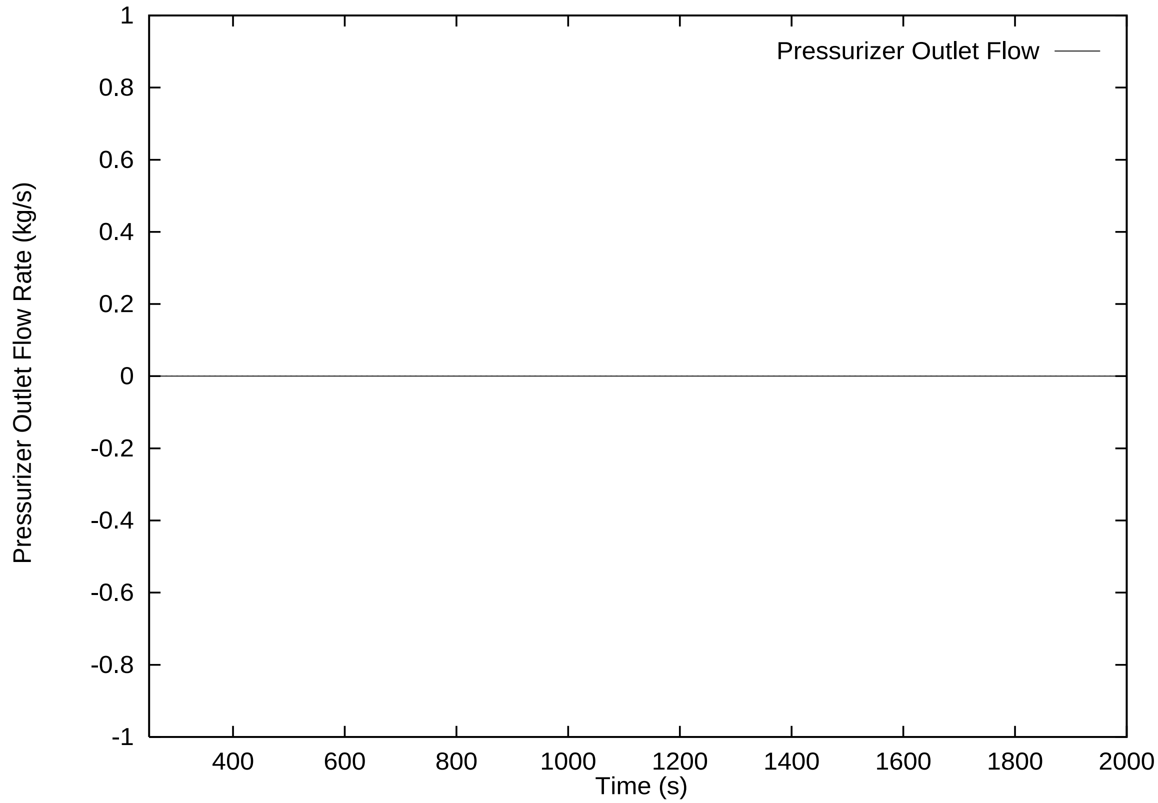


Figure 147 Pressurizer Outlet Flow Rate for 25% RIH Break with a Subsequent Loss of Class IV Power

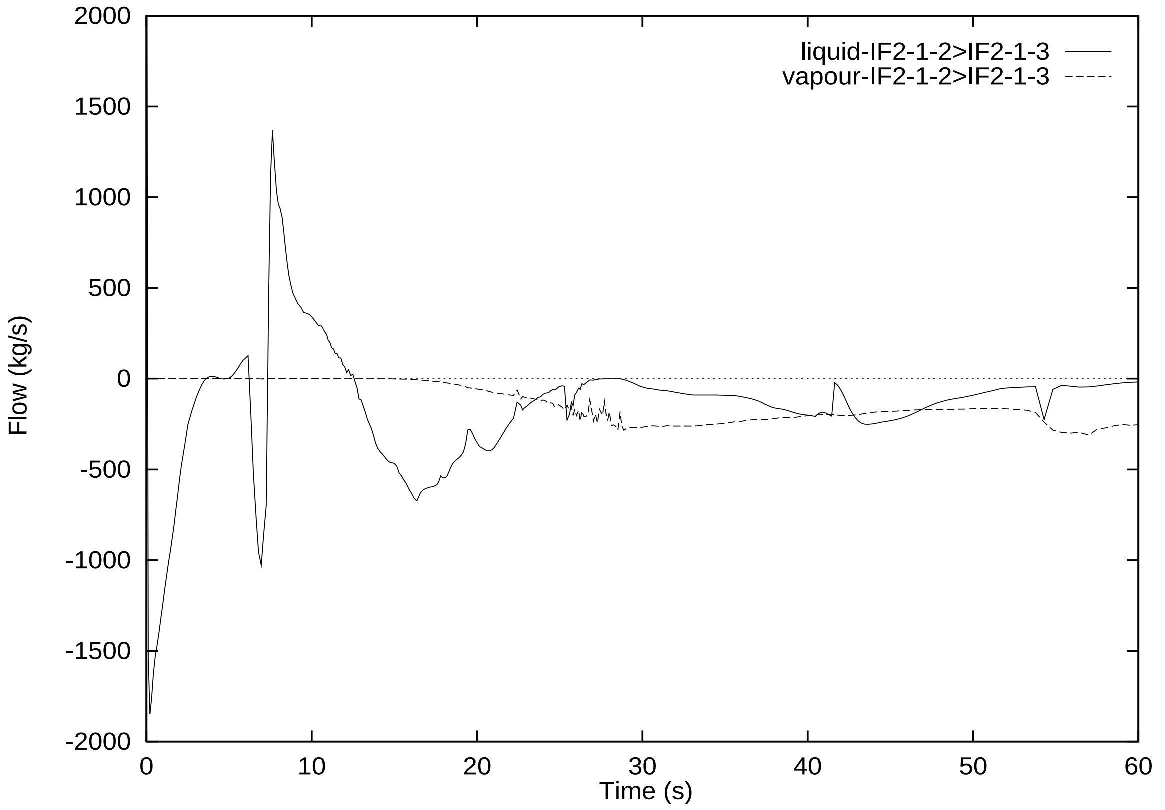


Figure 148 Feeder 2 Flows for 25% RIH Break with Subsequent Loss of Class IV Power

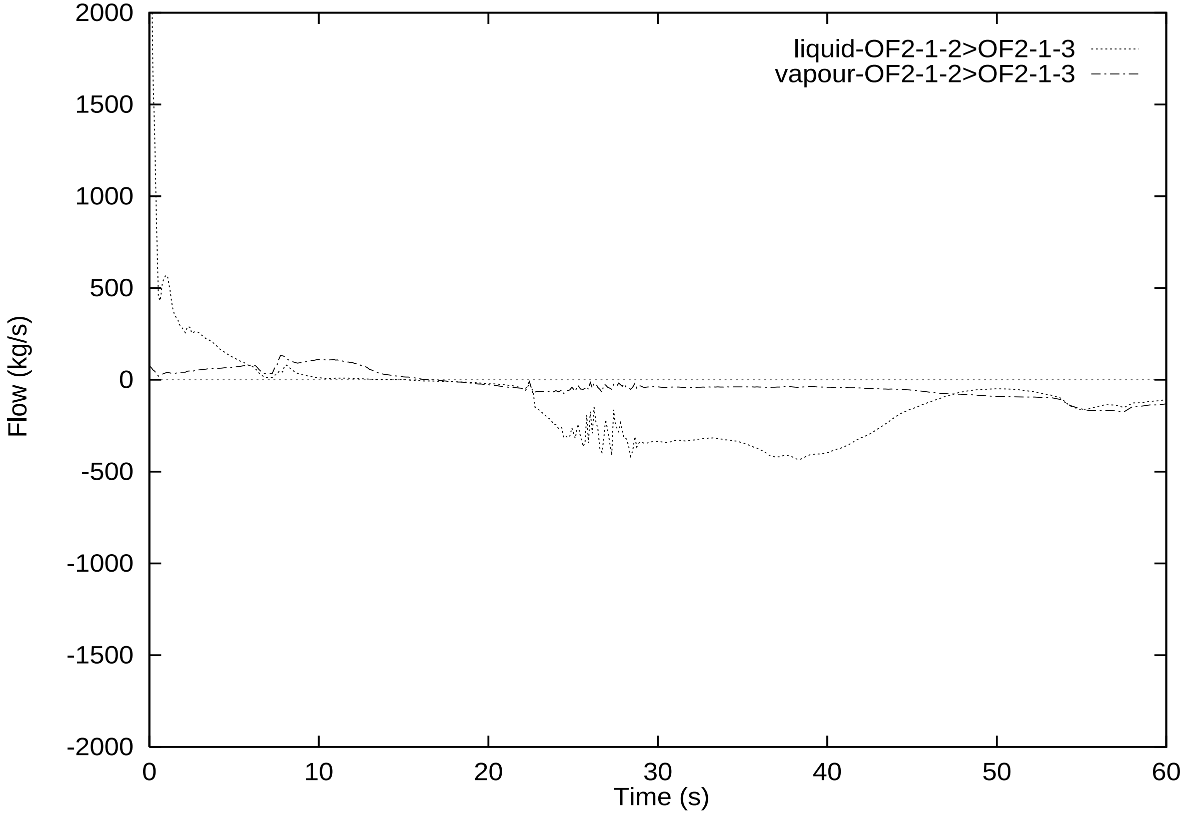


Figure 149 Feeder 2 Flows for 25% RIH Break with Subsequent Loss of Class IV Power

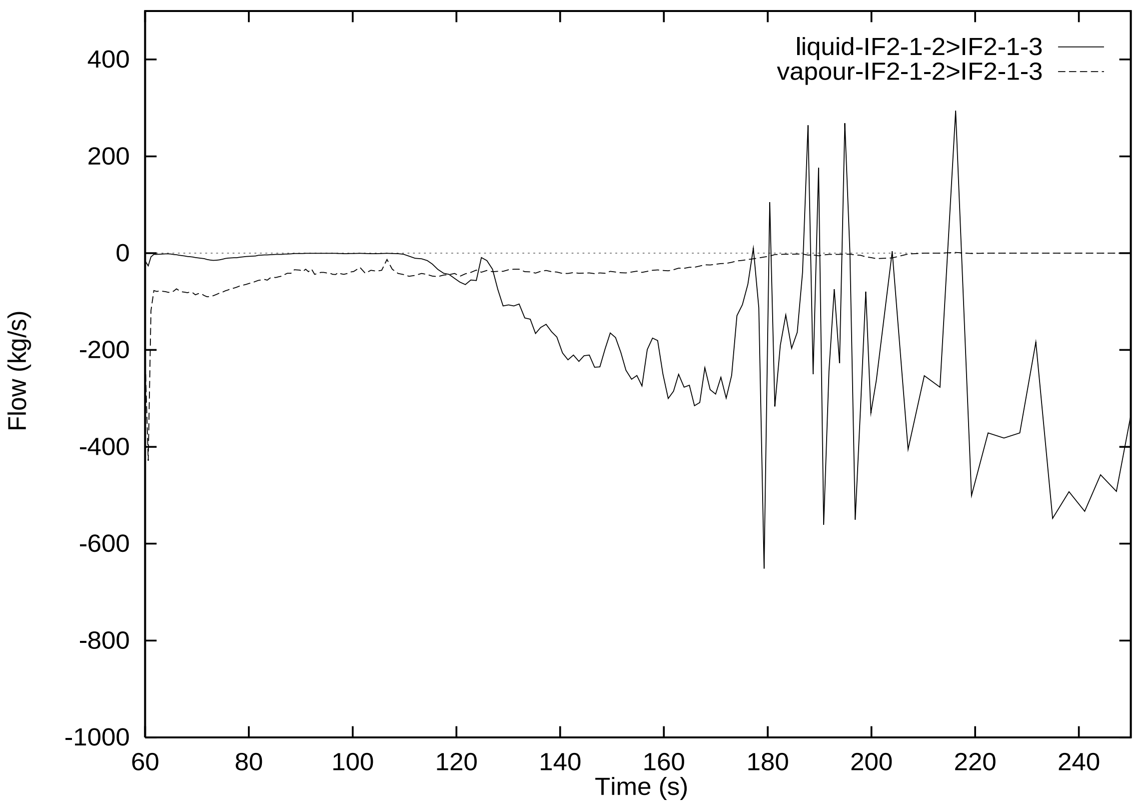


Figure 150 Feeder 2 Flows for 25% RIH Break with Subsequent Loss of Class IV Power

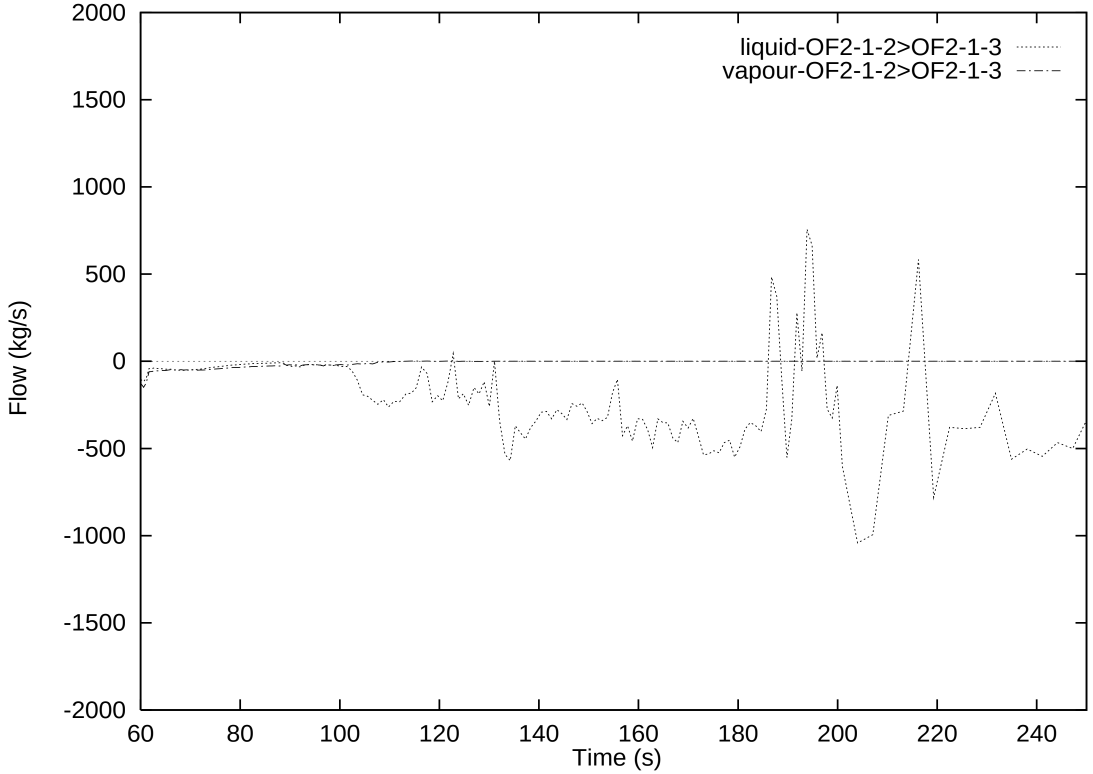


Figure 151 Feeder 2 Flows for 25% RIH Break with Subsequent Loss of Class IV Power

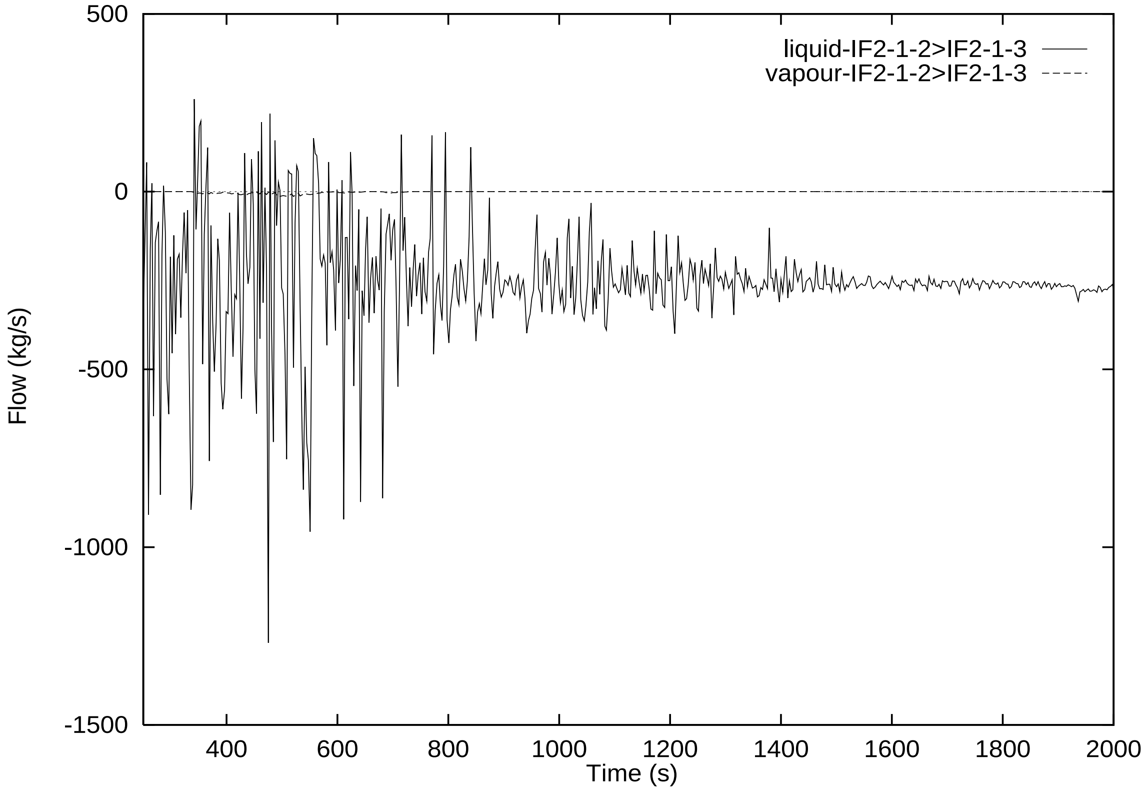


Figure 152 Feeder 2 Flows for 25% RIH Break with Subsequent Loss of Class IV Power

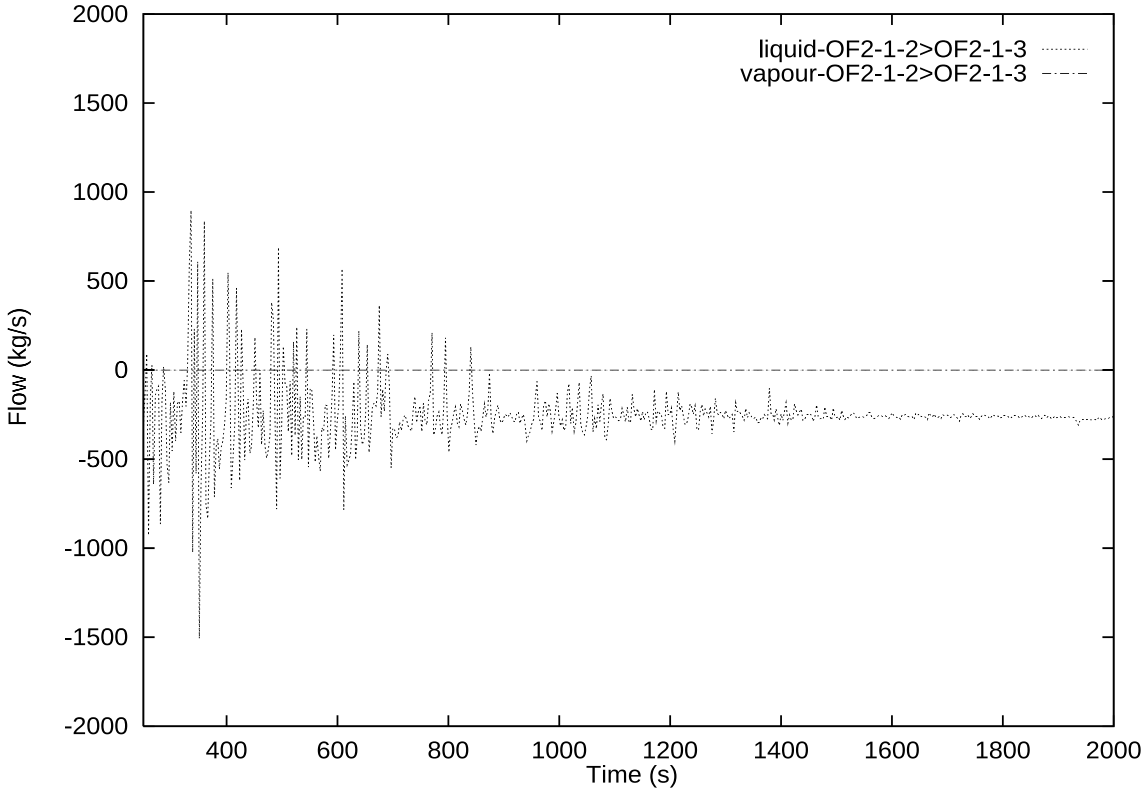


Figure 153 Feeder 2 Flows for 25% RIH Break with Subsequent Loss of Class IV Power

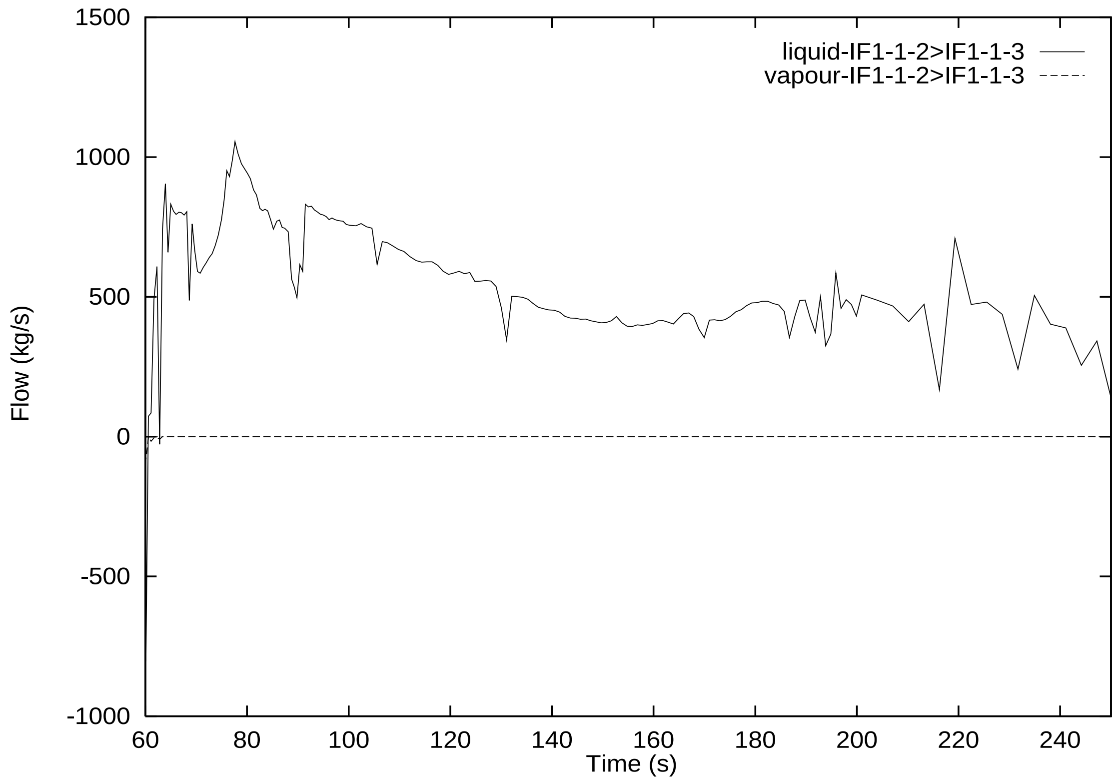


Figure 154 Feeder 1 Flows for 25% RIH Break with Subsequent Loss of Class IV Power

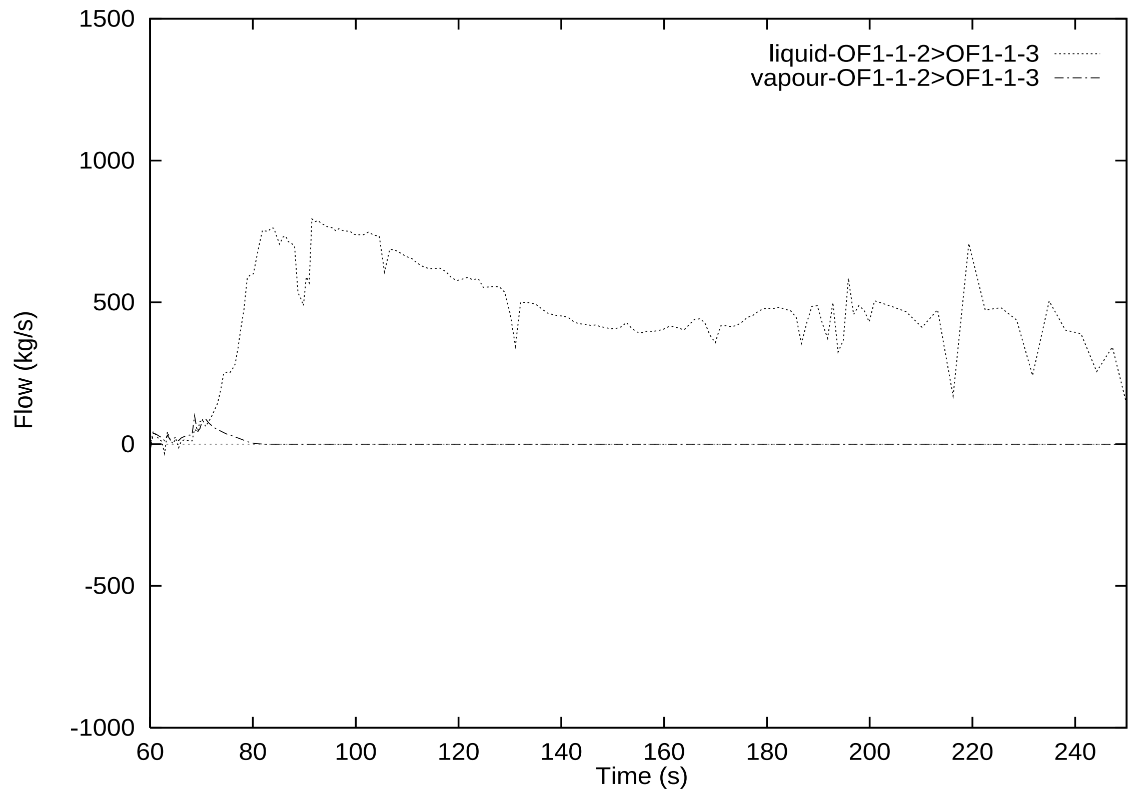


Figure 155 Feeder 1 Flows for 25% RIH Break with Subsequent Loss of Class IV Power

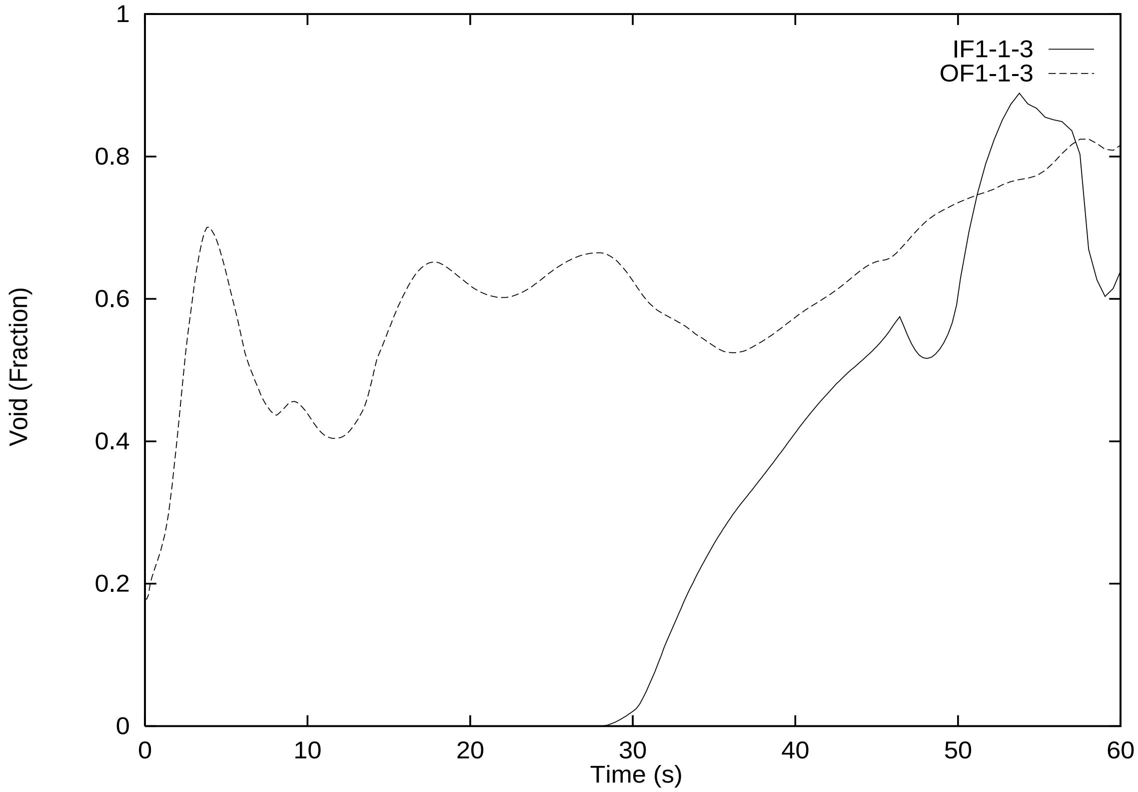


Figure 156 Feeder 1 Flows for 25% RIH Break with Subsequent Loss of Class IV Power

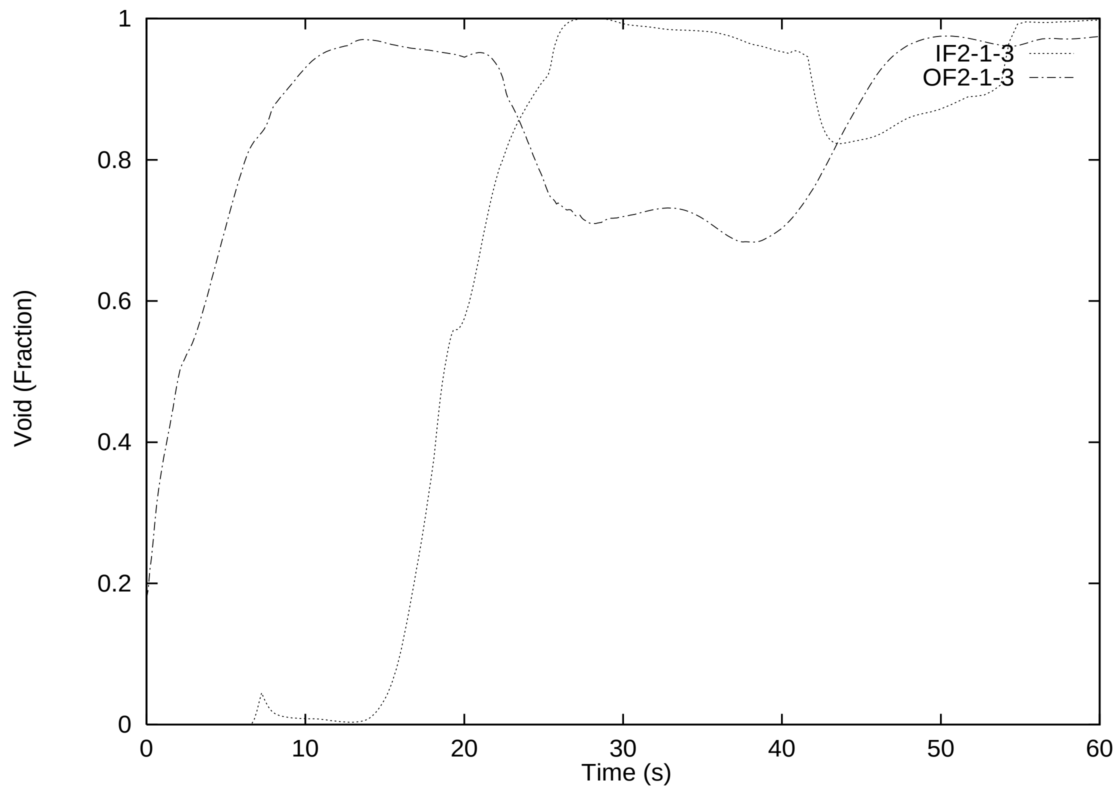


Figure 157 Feeder 1 Flows for 25% RIH Break with Subsequent Loss of Class IV Power

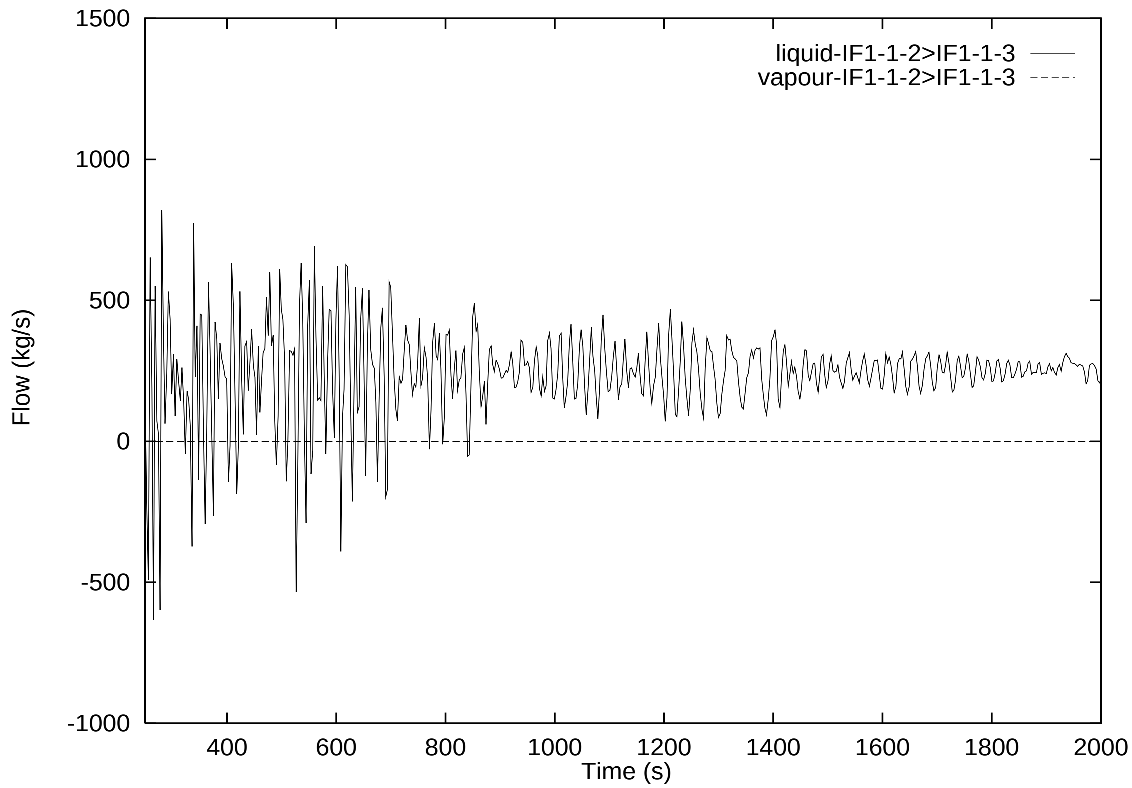


Figure 158 Feeder 1 Flows for 25% RIH Break with Subsequent Loss of Class IV Power

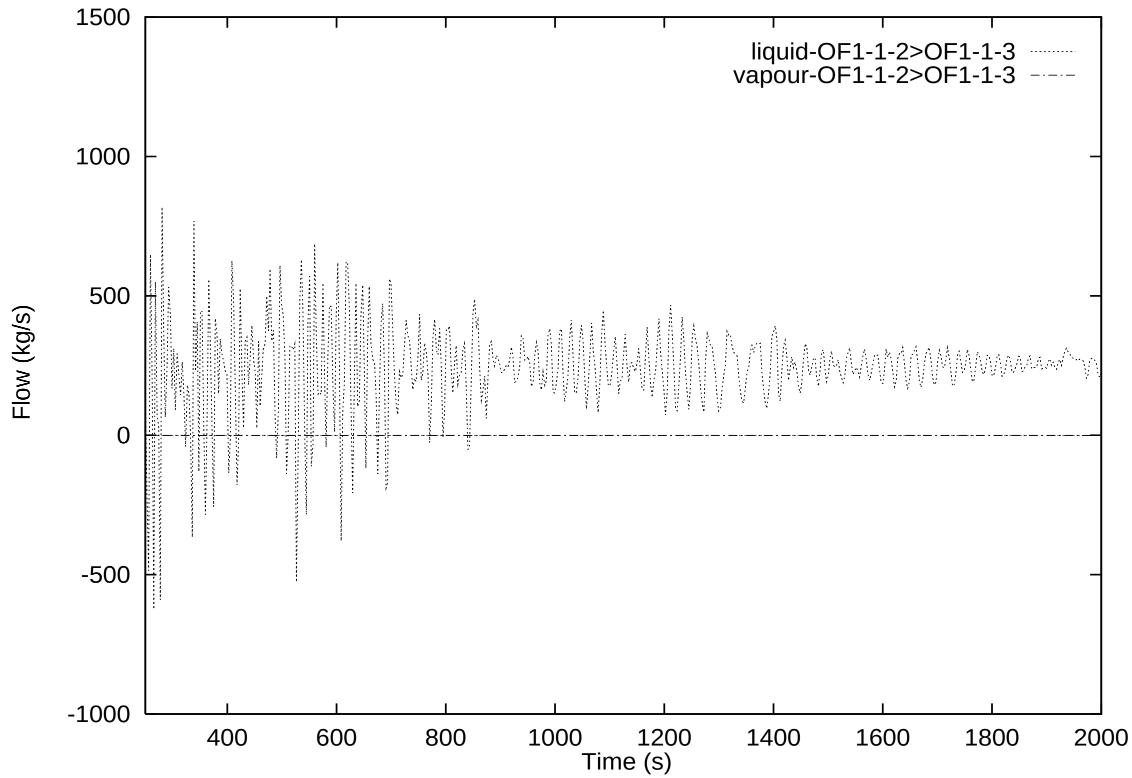


Figure 159 Feeder 1 Flows for 25% RIH Break with Subsequent Loss of Class IV Power

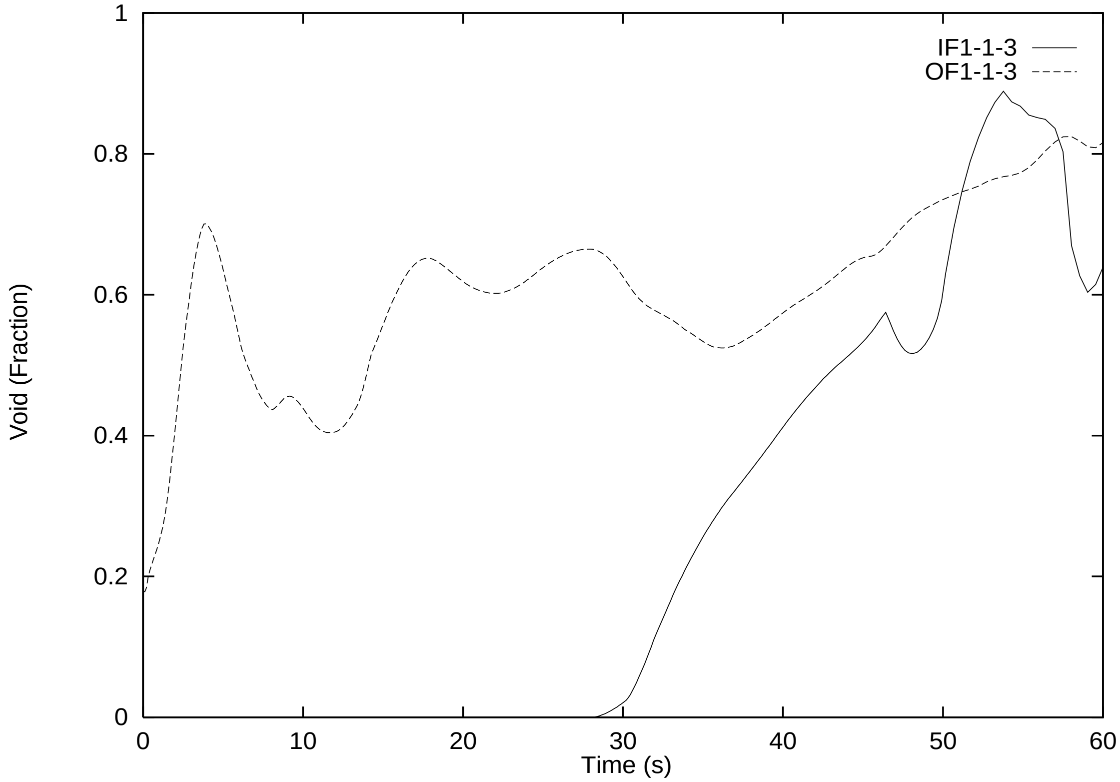


Figure 160 Feeder Voids for 25% RIH Break with Subsequent Loss of Class IV Power

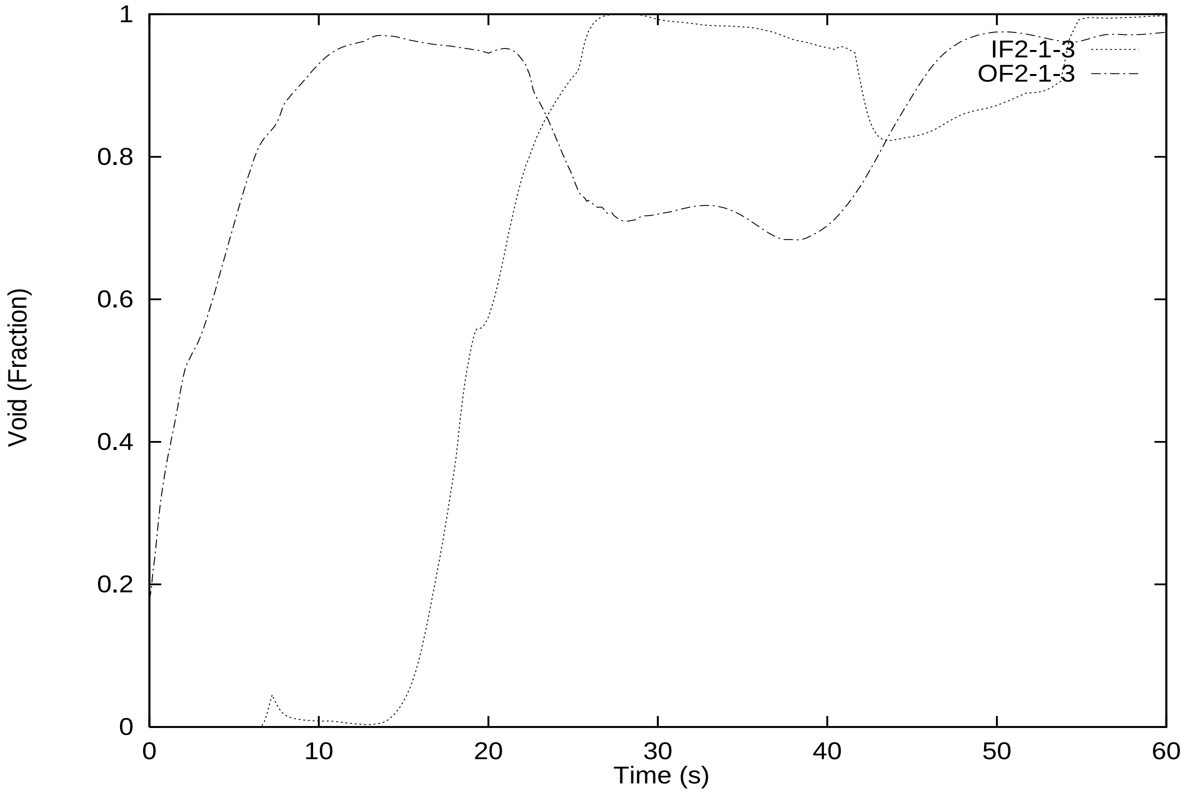


Figure 161 Feeder Voids for 25% RIH Break with Subsequent Loss of Class IV Power

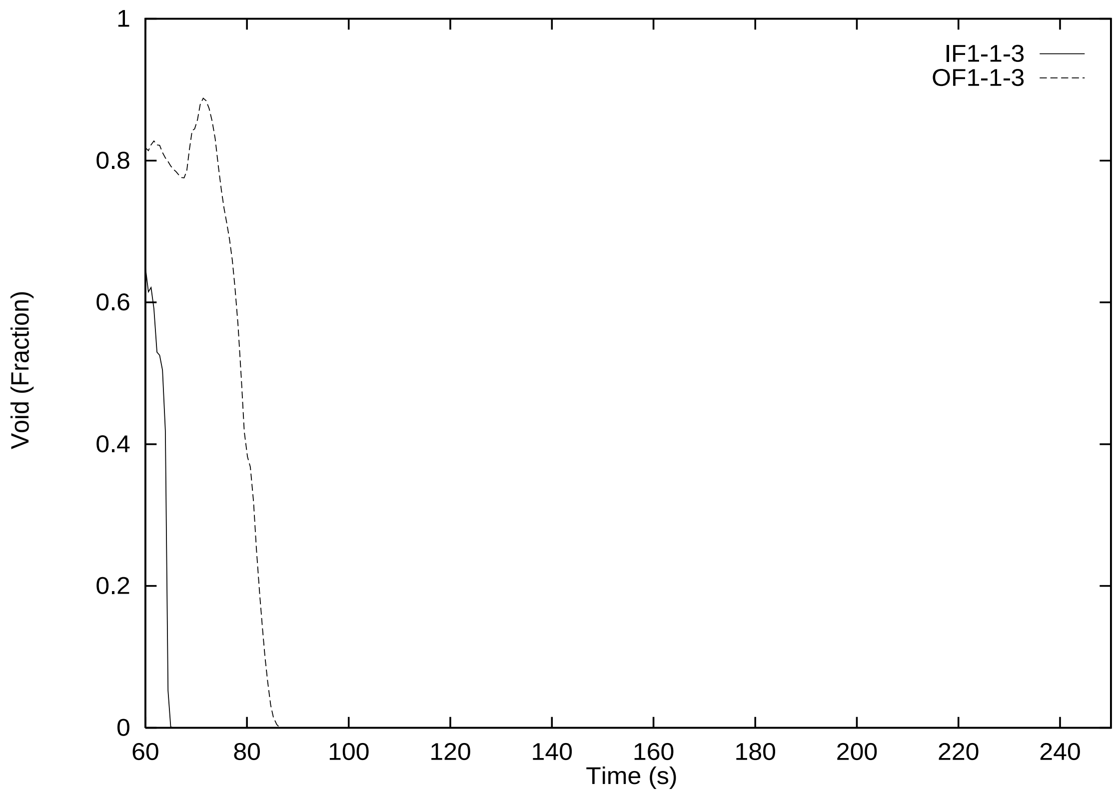


Figure 162 Feeder Voids for 25% RIH Break with Subsequent Loss of Class IV Power

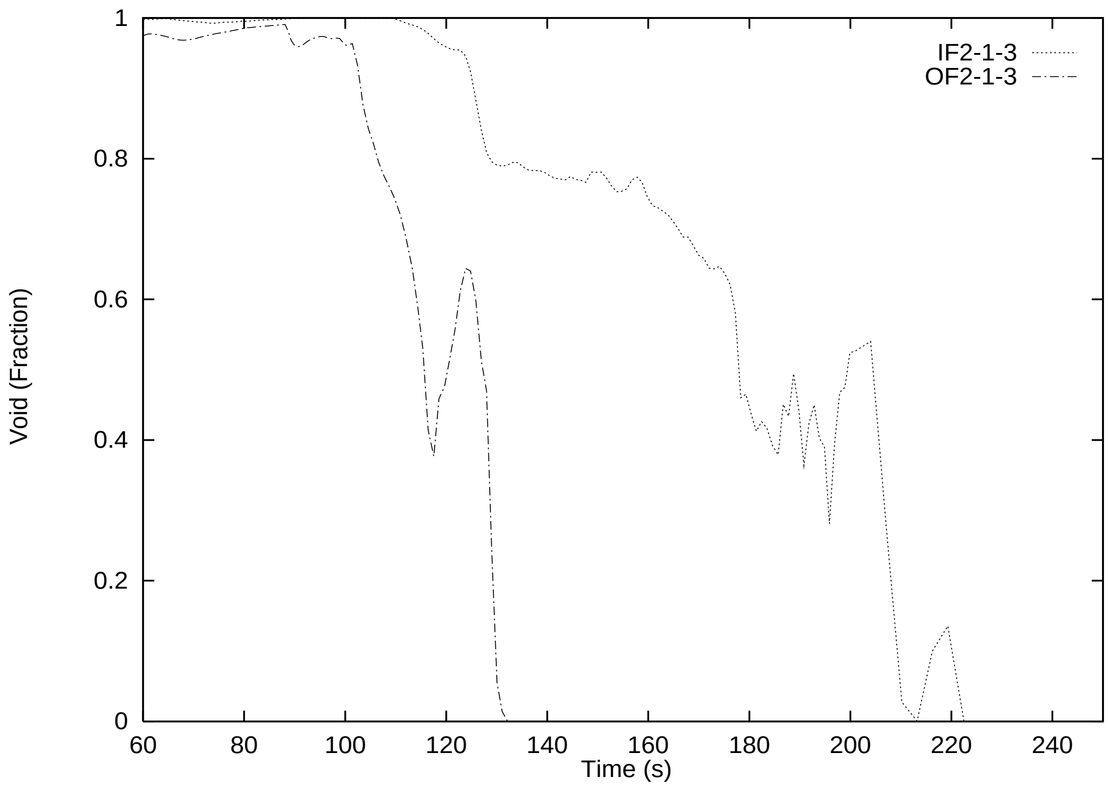


Figure 163 Feeder Voids for 25% RIH Break with Subsequent Loss of Class IV Power

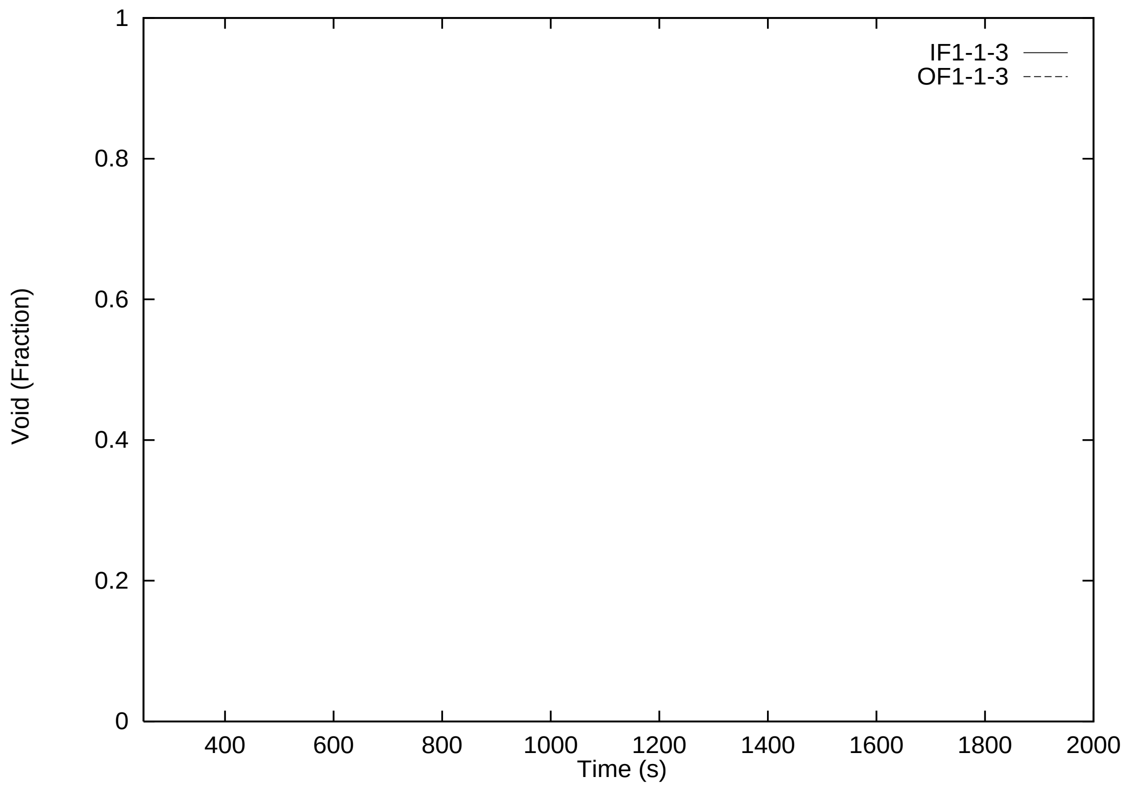


Figure 164 Feeder Voids for 25% RIH Break with Subsequent Loss of Class IV Power

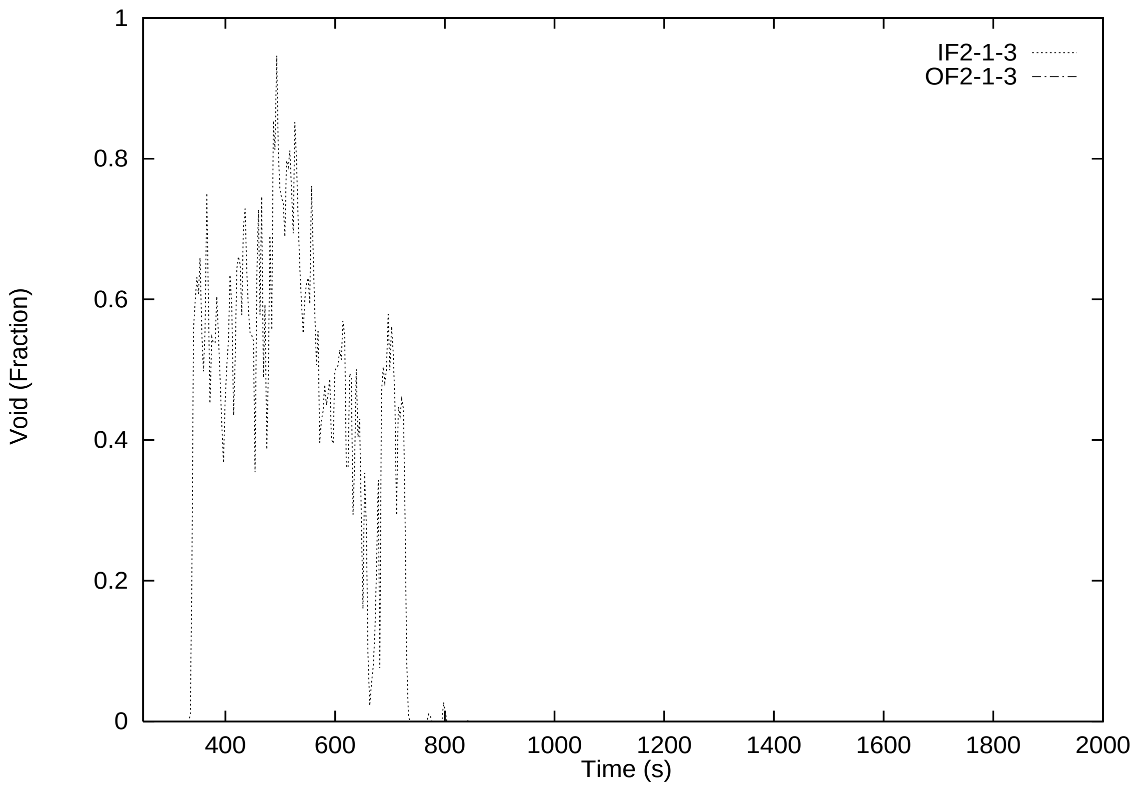


Figure 165 Feeder Voids for 25% RIH Break with Subsequent Loss of Class IV Power

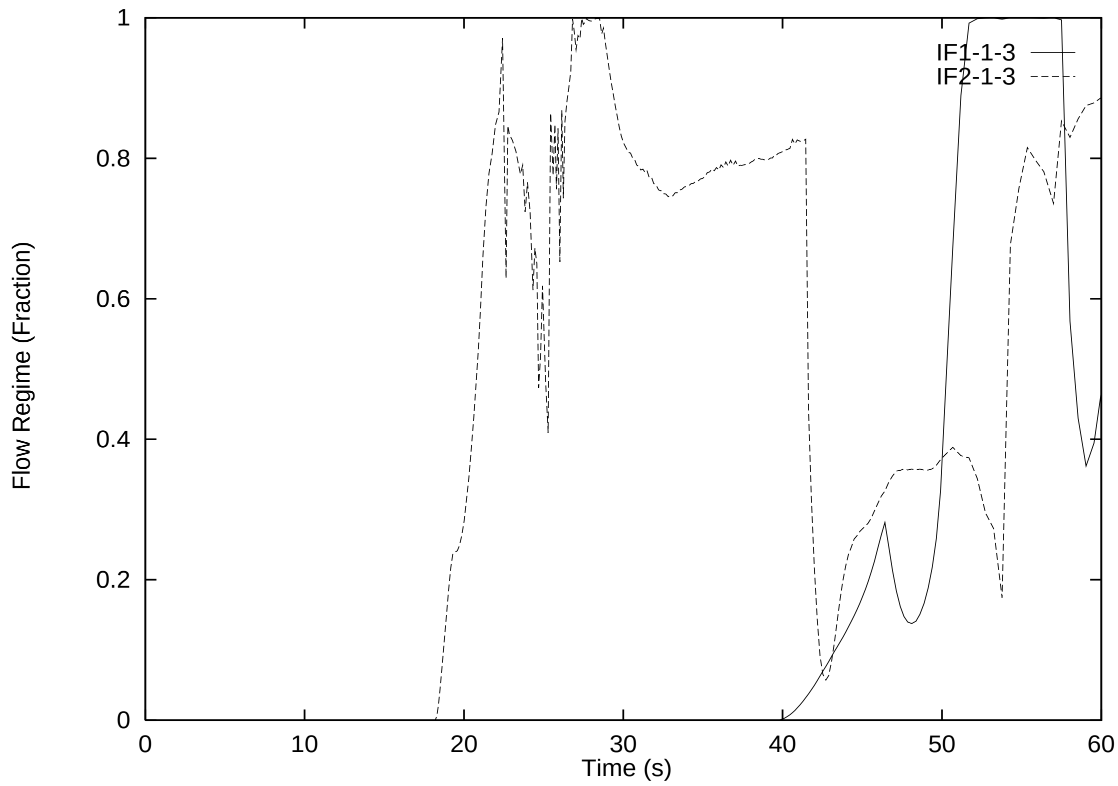


Figure 166 Feeder Flow Regime for 25% RIH Break with Subsequent Loss of Class IV Power

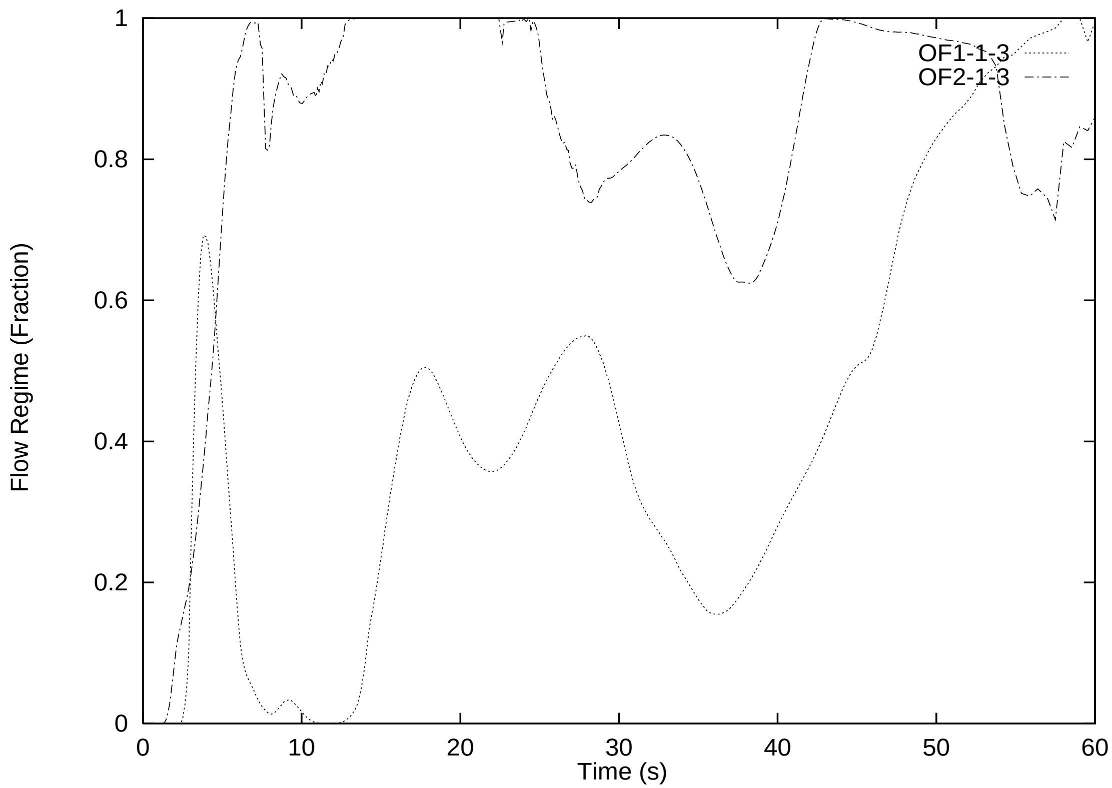


Figure 167 Feeder Flow Regime for 25% RIH Break with Subsequent Loss of Class IV Power

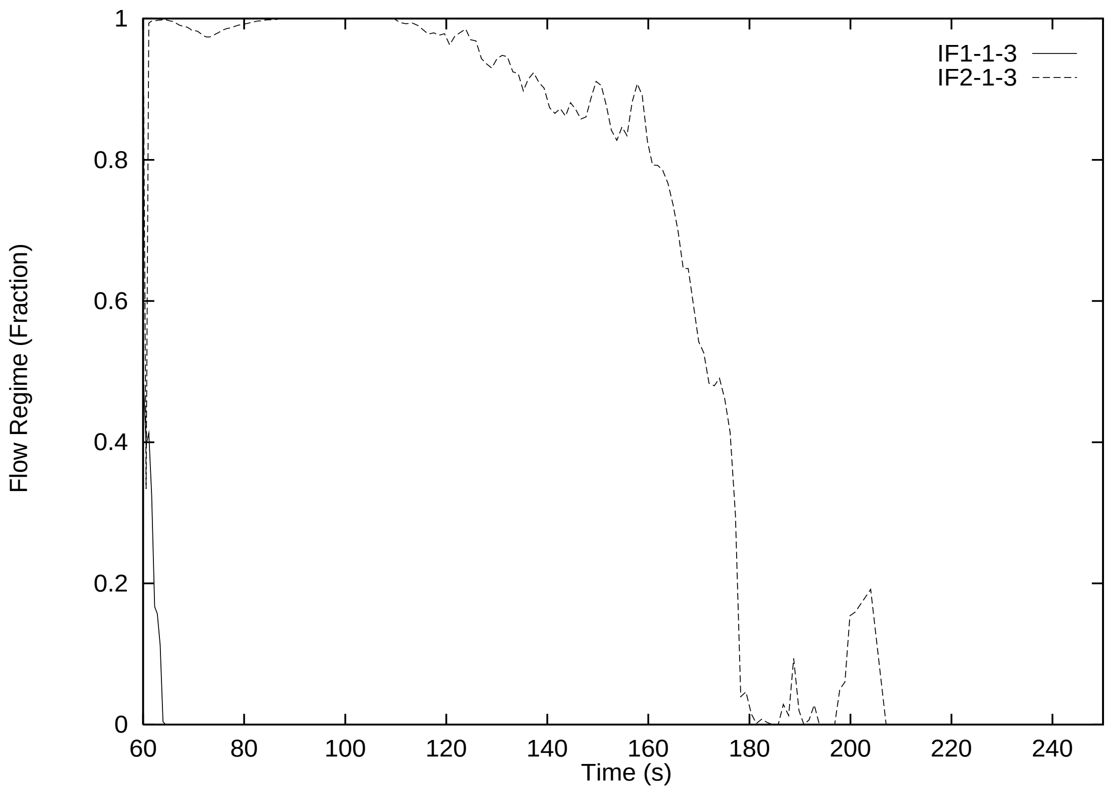


Figure 168 Feeder Flow Regime for 25% RIH Break with Subsequent Loss of Class IV Power

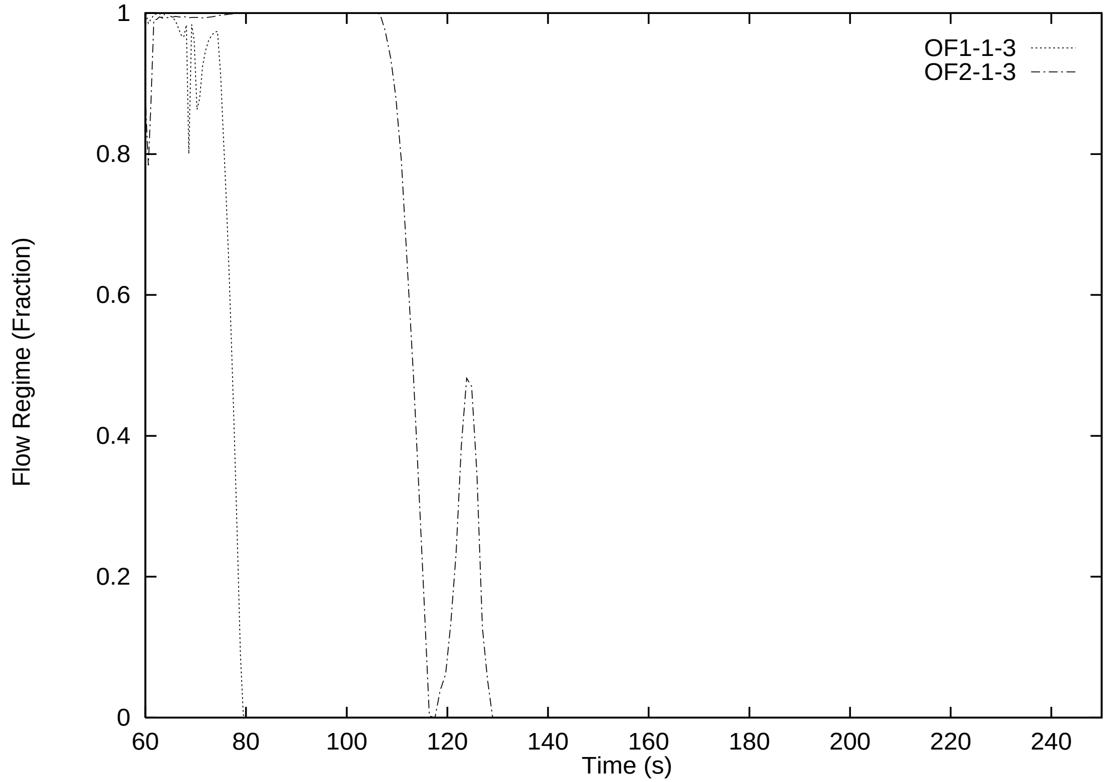


Figure 169 Feeder Flow Regime for 25% RIH Break with Subsequent Loss of Class IV Power

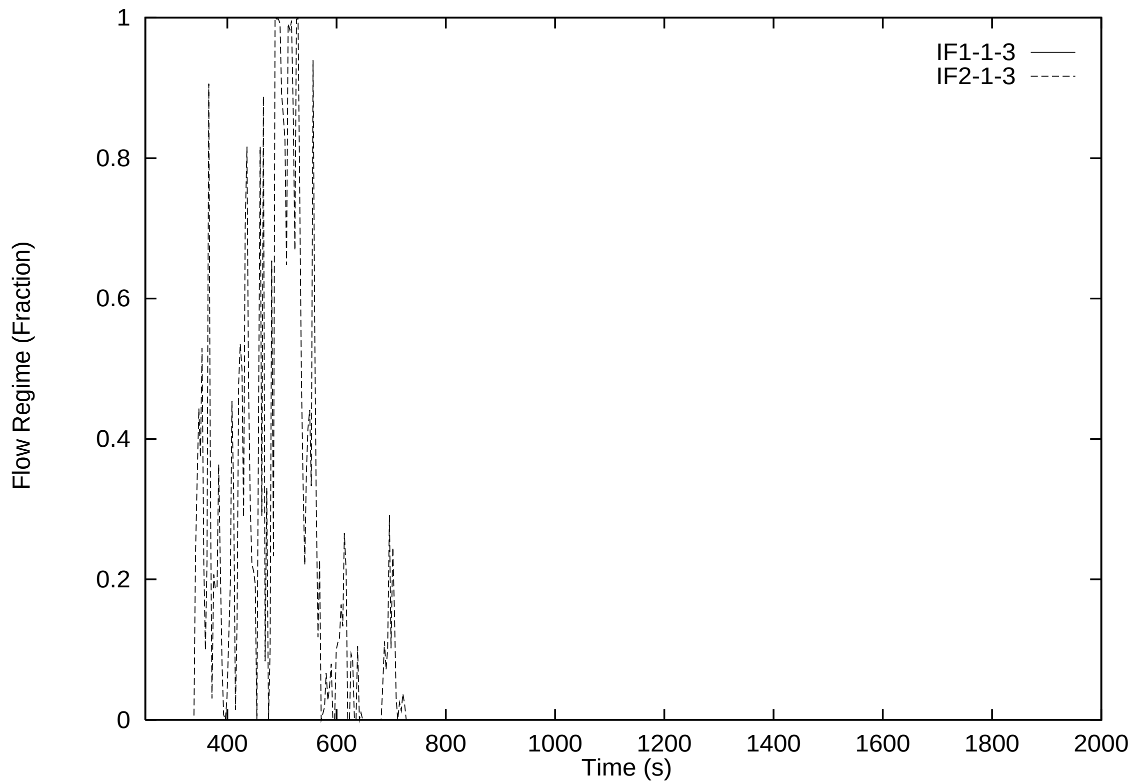


Figure 170 Feeder Flow Regime for 25% RIH Break with Subsequent Loss of Class IV Power

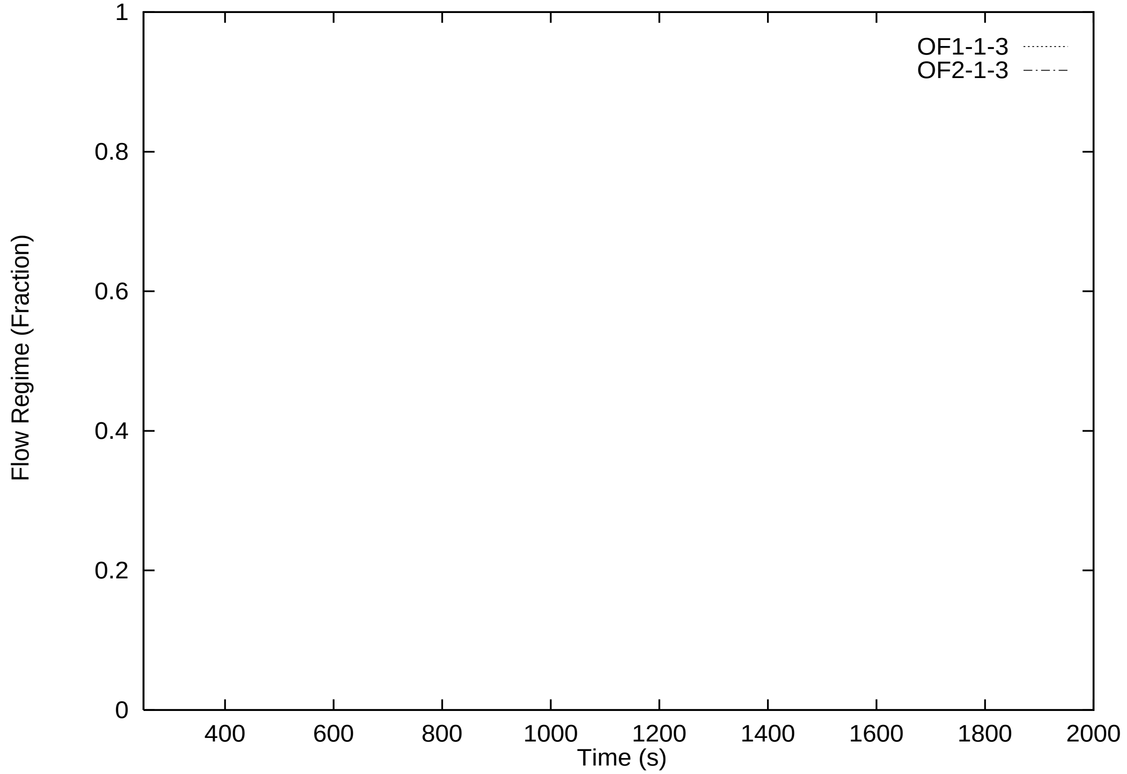


Figure 171 Feeder Flow Regime for 25% RIH Break with Subsequent Loss of Class IV Power

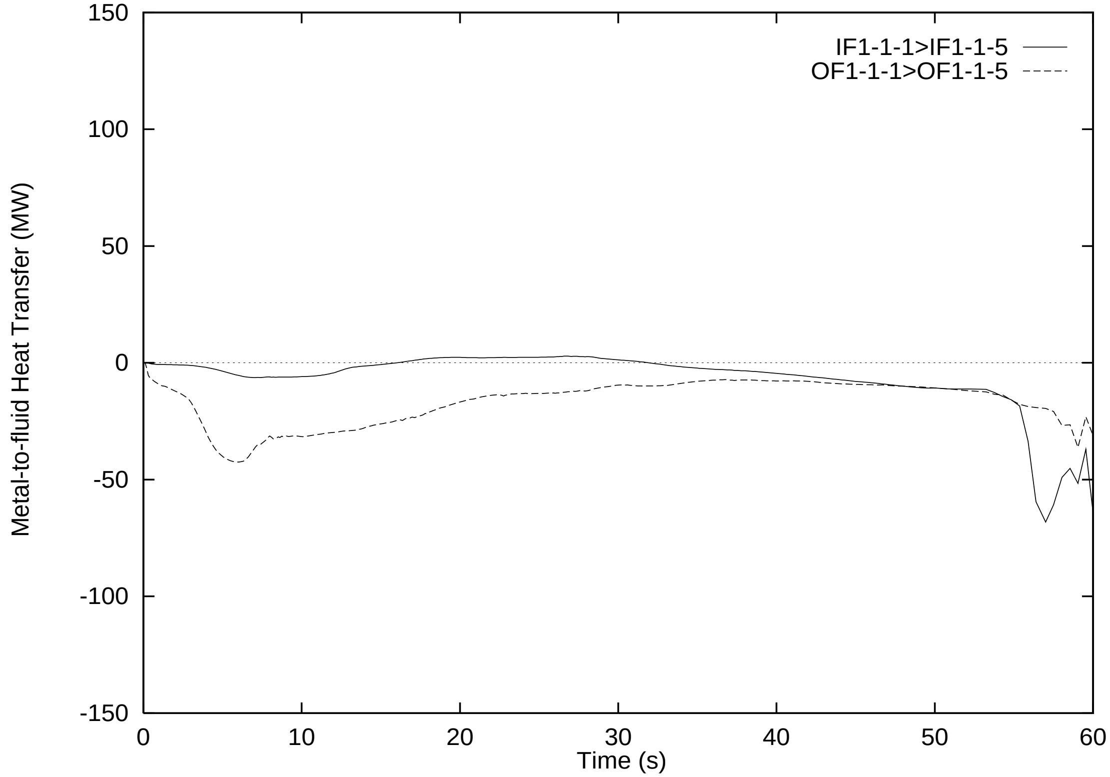


Figure 172 Stored Energy Release in Feeder for 25% RIH Break with Subsequent Loss of Class IV Power

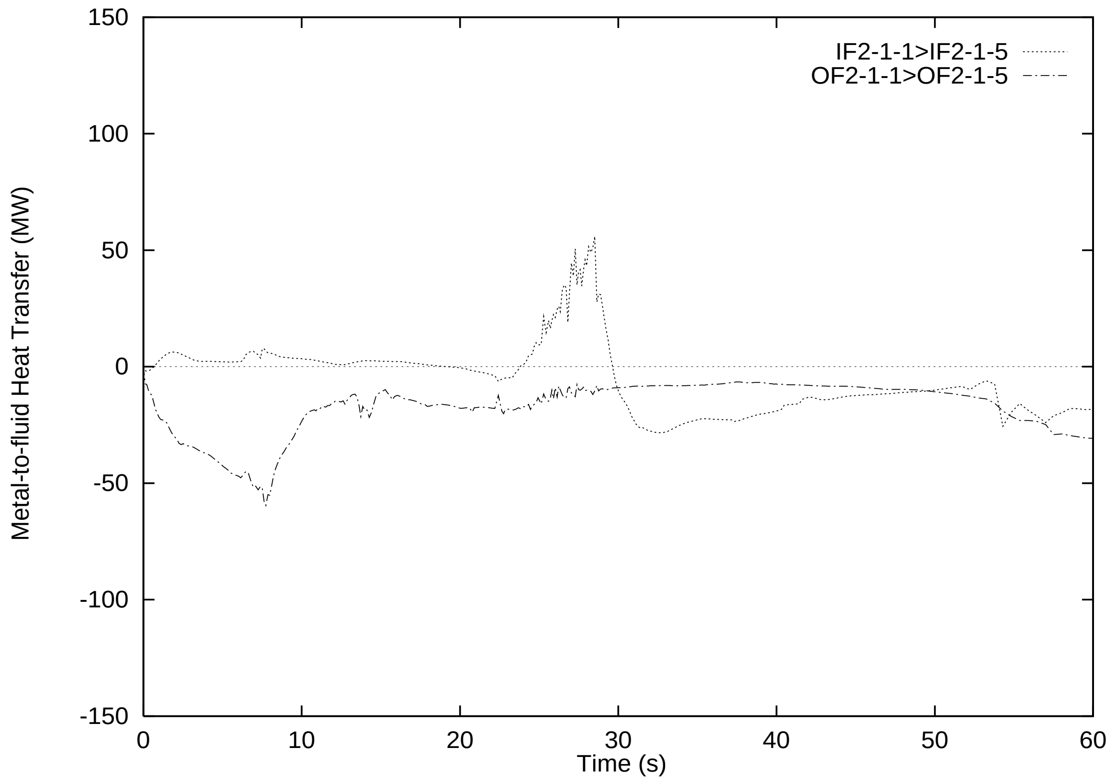


Figure 173 Stored Energy Release in Feeder for 25% RIH Break with Subsequent Loss of Class IV Power

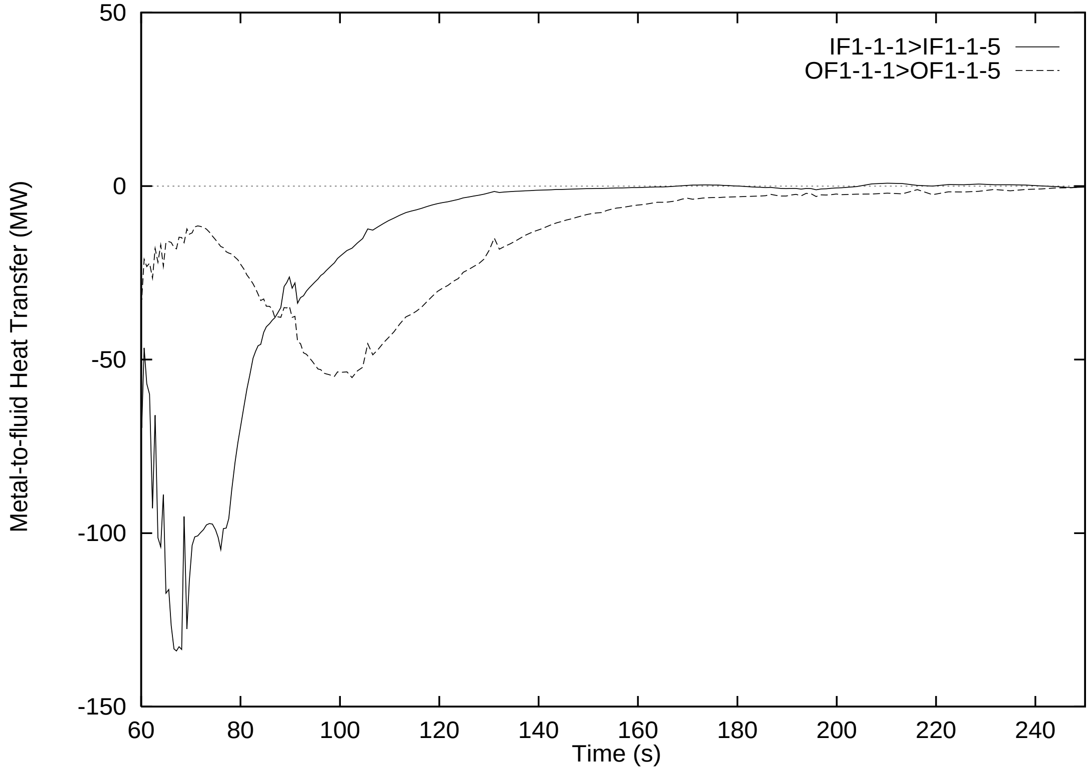


Figure 174 Stored Energy Release in Feeder for 25% RIH Break with Subsequent Loss of Class IV Power

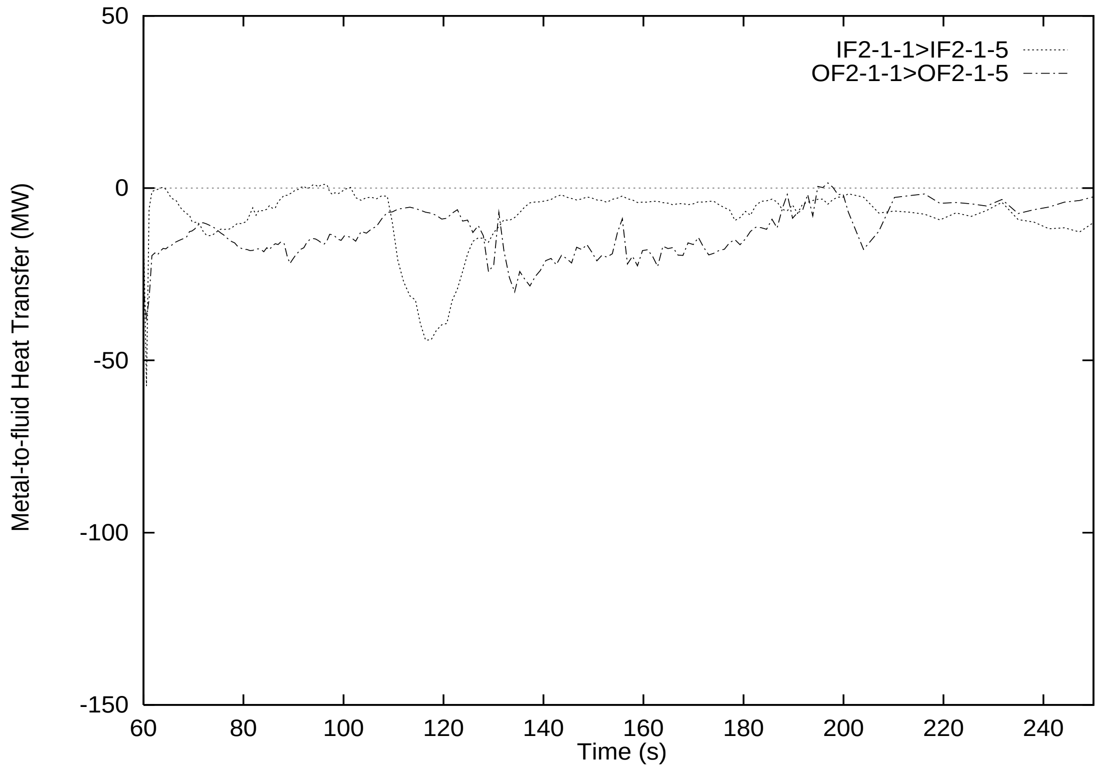


Figure 175 Stored Energy Release in Feeder for 25% RIH Break with Subsequent Loss of Class IV Power

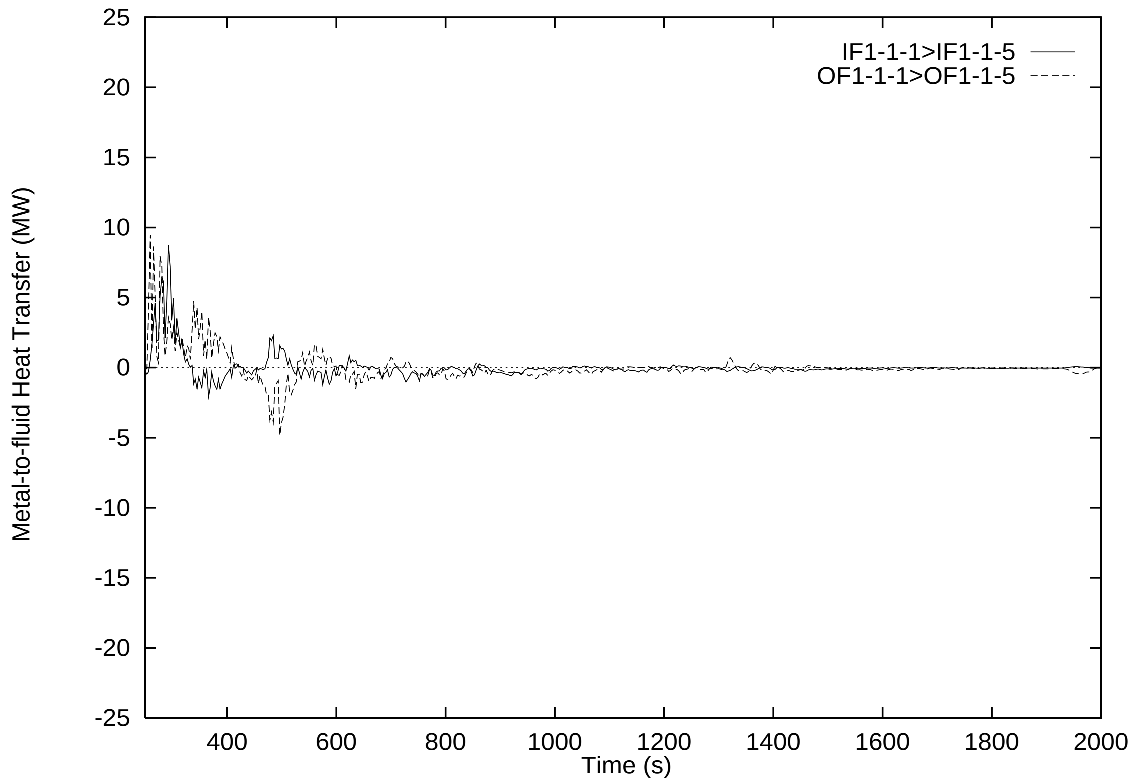


Figure 176 Stored Energy Release in Feeder for 25% RIH Break with Subsequent Loss of Class IV Power

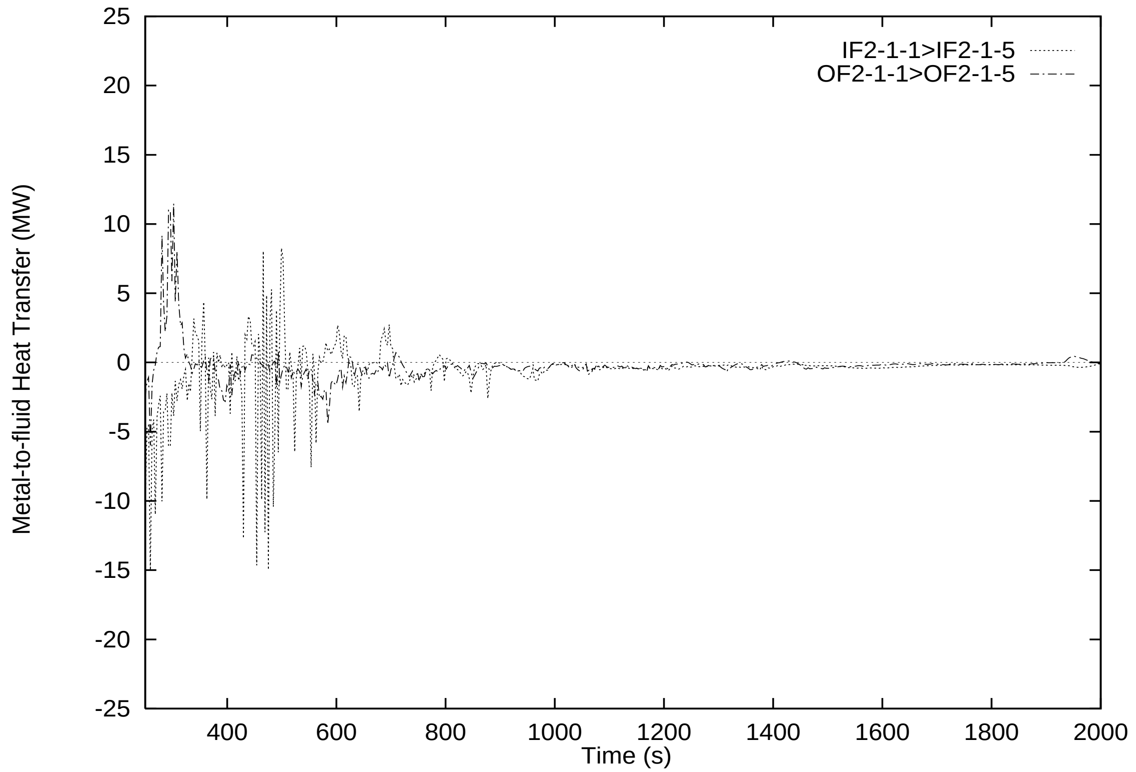


Figure 177 Stored Energy Release in Feeder for 25% RIH Break with Subsequent Loss of Class IV Power

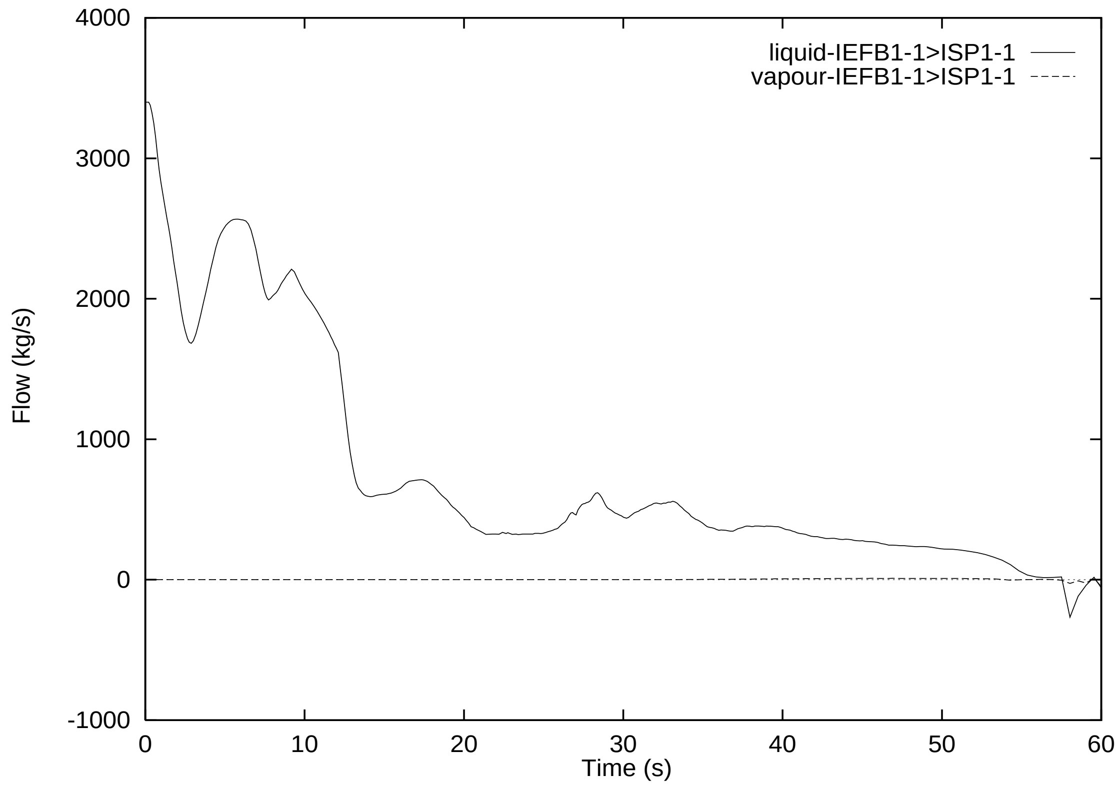


Figure 178 End Fitting 1 Flows for 25% RIH Break with Subsequent Loss of Class IV Power

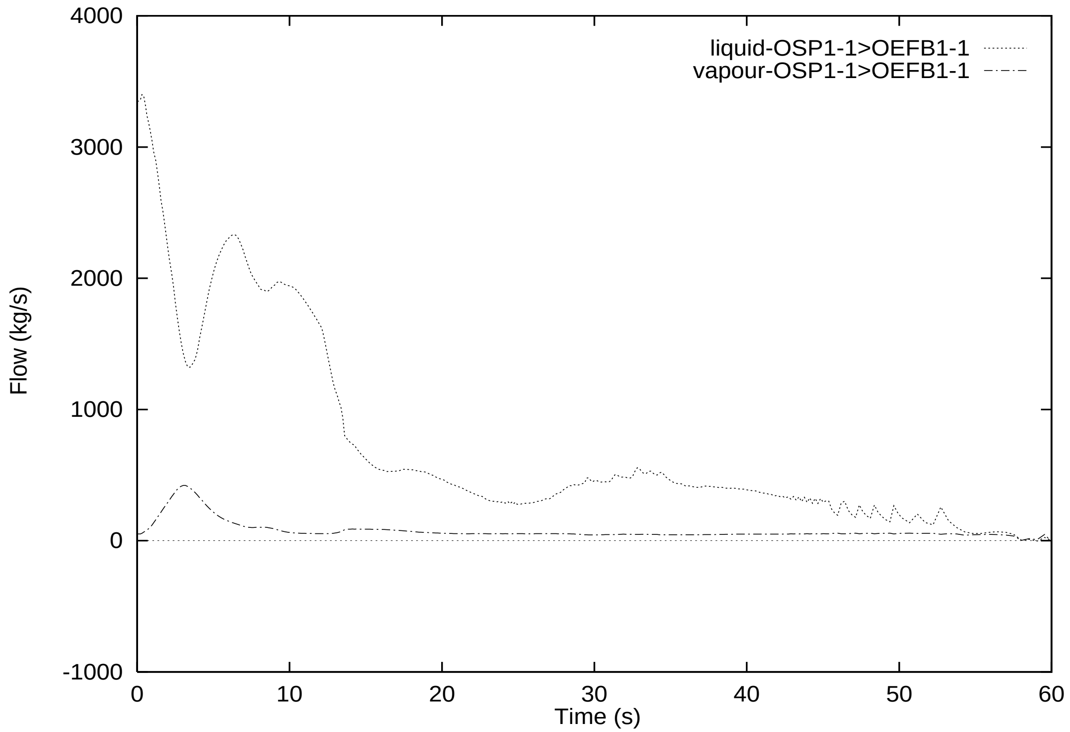


Figure 179 End Fitting 1 Flows for 25% RIH Break with Subsequent Loss of Class IV Power

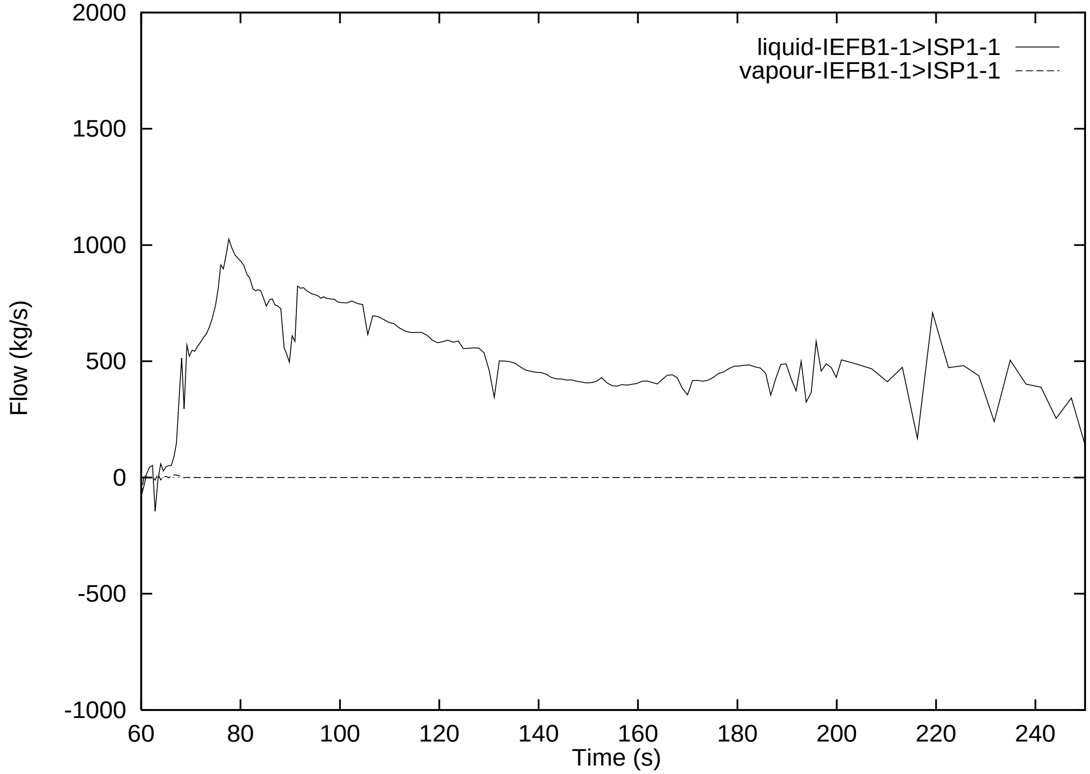


Figure 180 End Fitting 1 Flows for 25% RIH Break with Subsequent Loss of Class IV Power

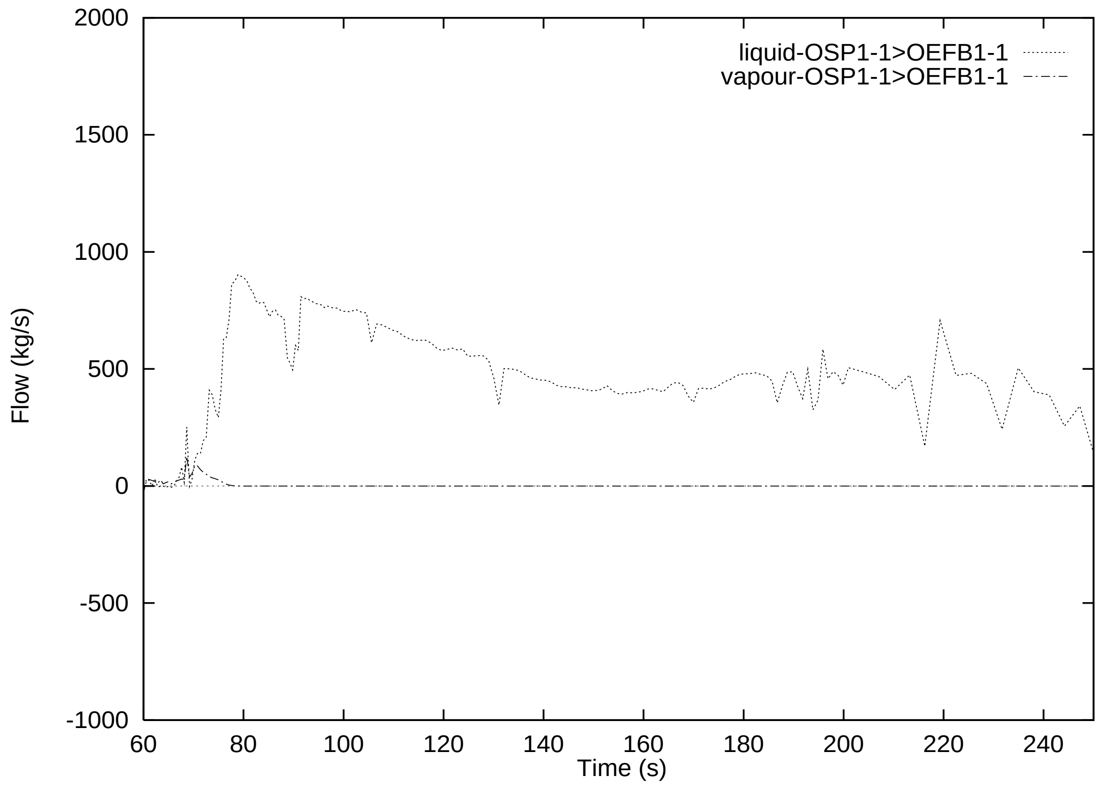


Figure 181 End Fitting 1 Flows for 25% RIH Break with Subsequent Loss of Class IV Power

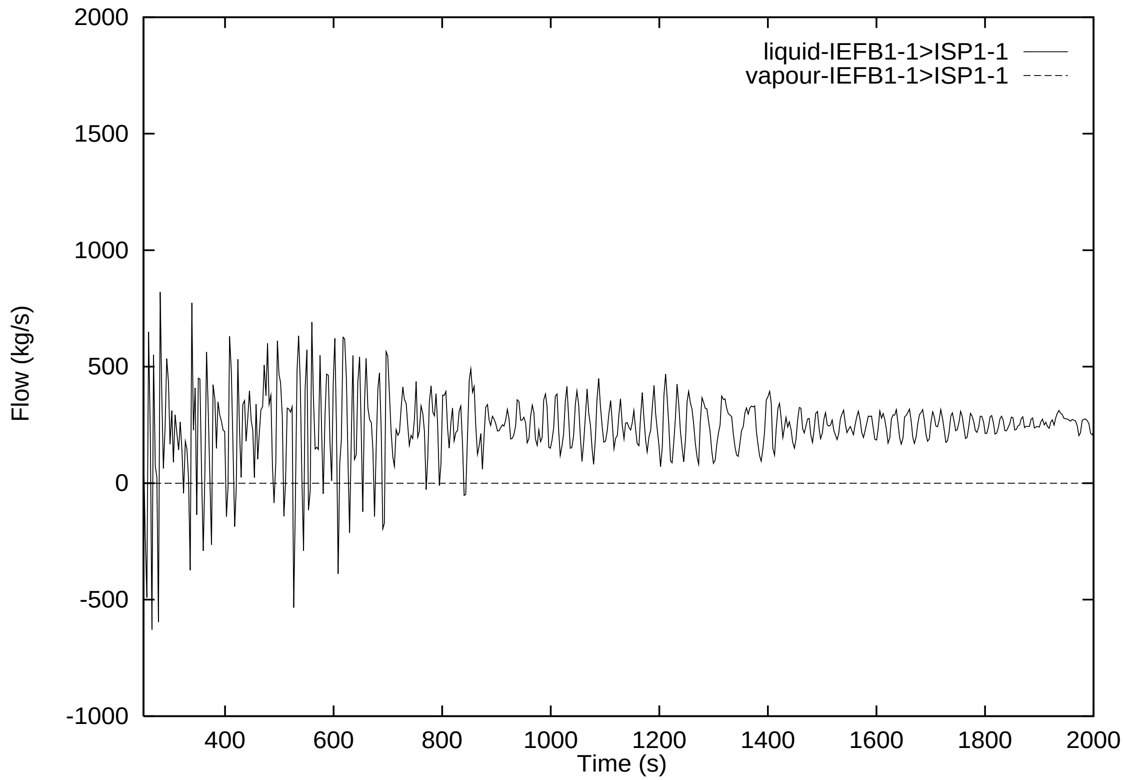


Figure 182 End Fitting 1 Flows for 25% RIH Break with Subsequent Loss of Class IV Power

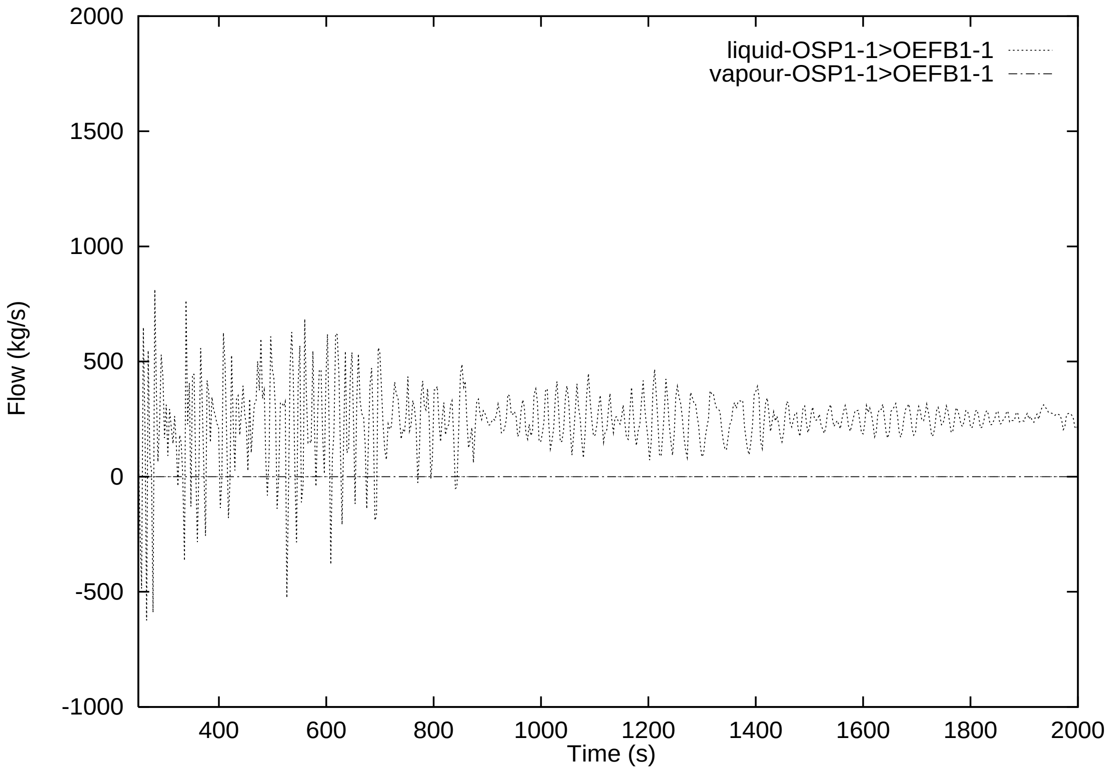


Figure 183 End Fitting 1 Flows for 25% RIH Break with Subsequent Loss of Class IV Power

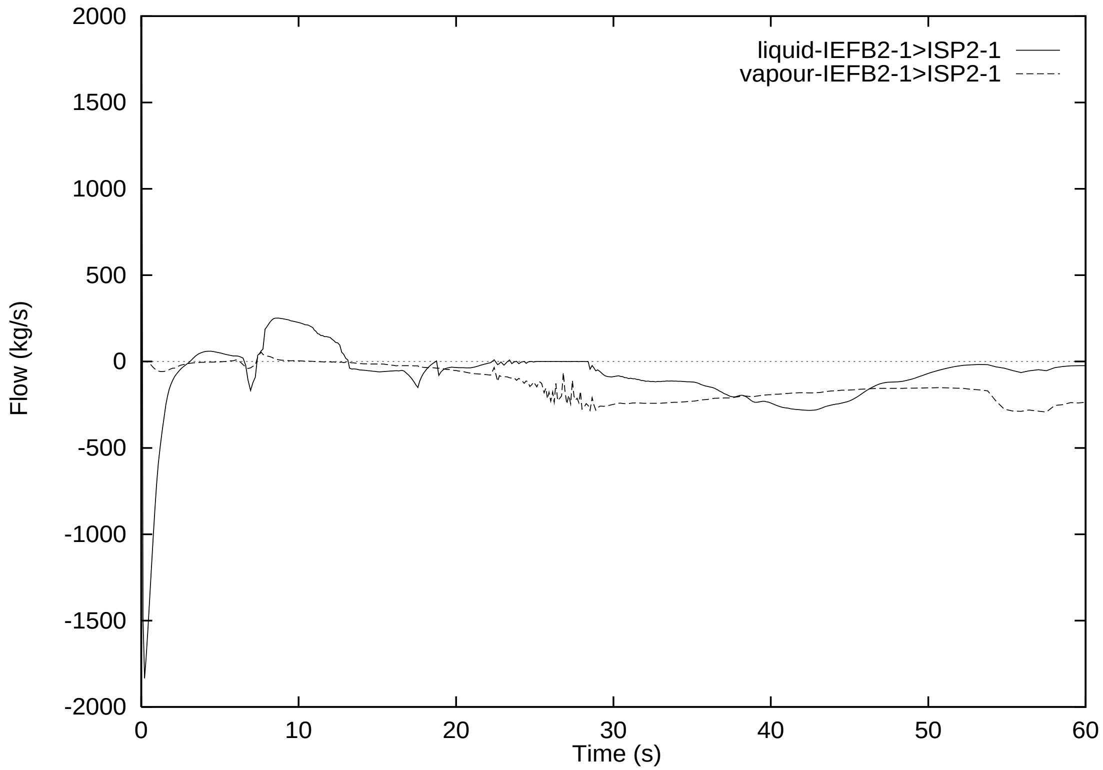


Figure 184 End Fitting 2 Flows for 25% RIH Break with Subsequent Loss of Class IV Power

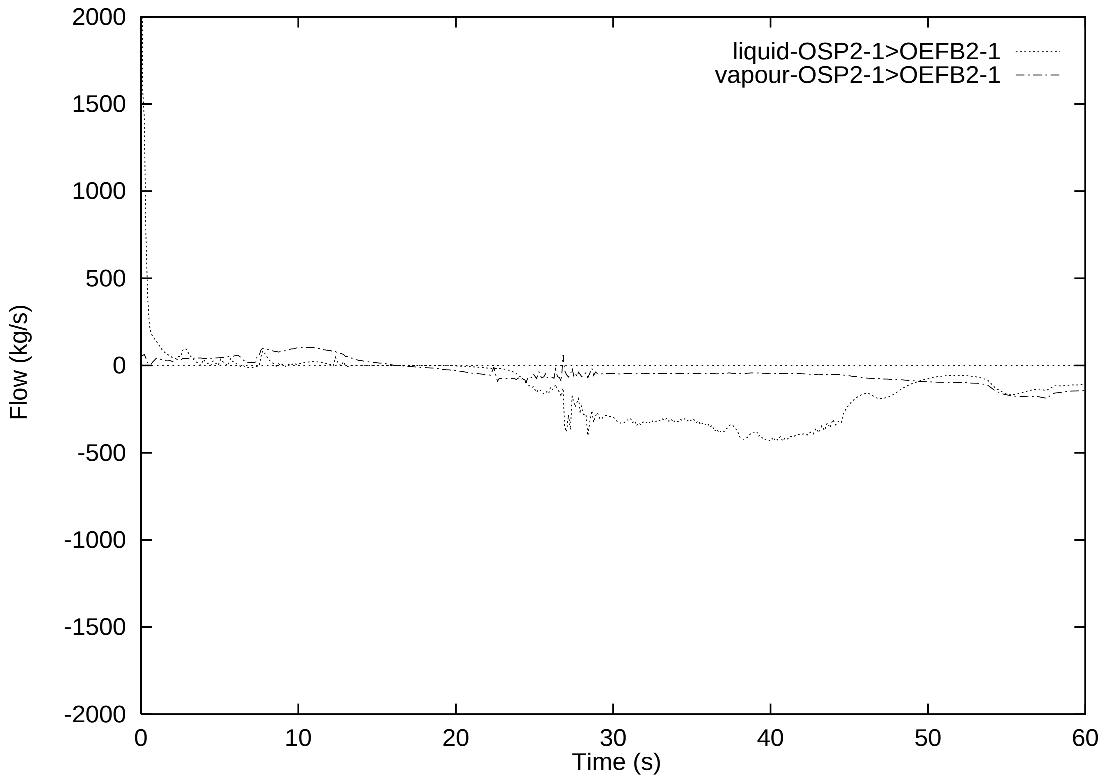


Figure 185 End Fitting 2 Flows for 25% RIH Break with Subsequent Loss of Class IV Power

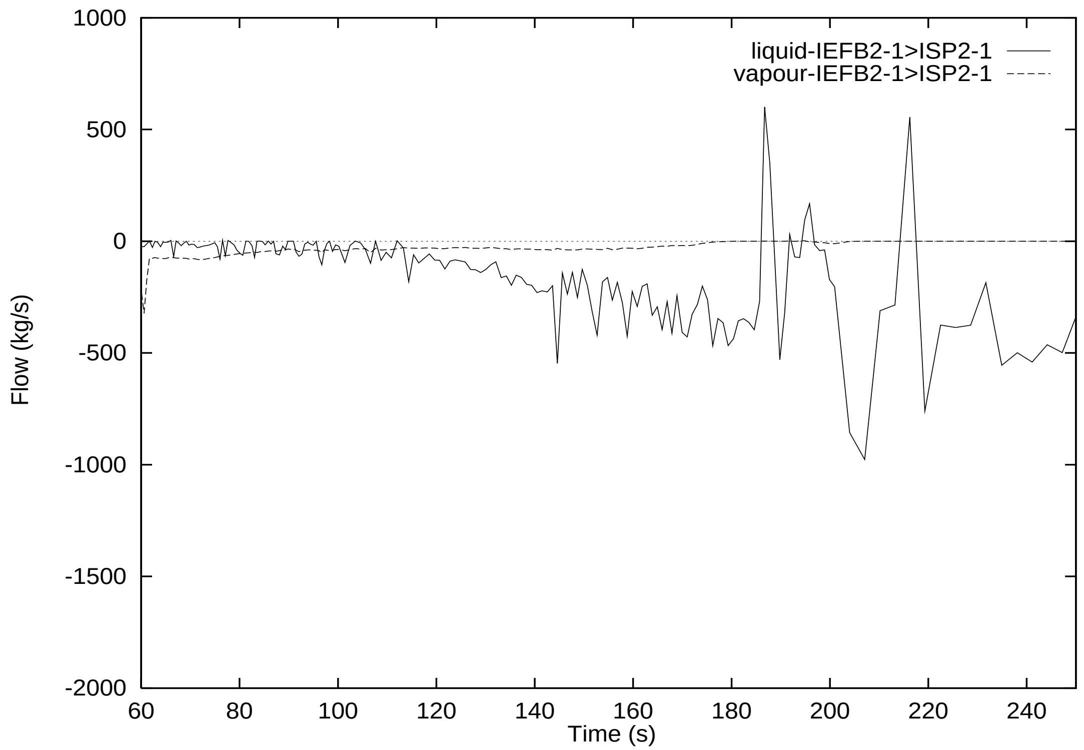


Figure 186 End Fitting 2 Flows for 25% RIH Break with Subsequent Loss of Class IV Power

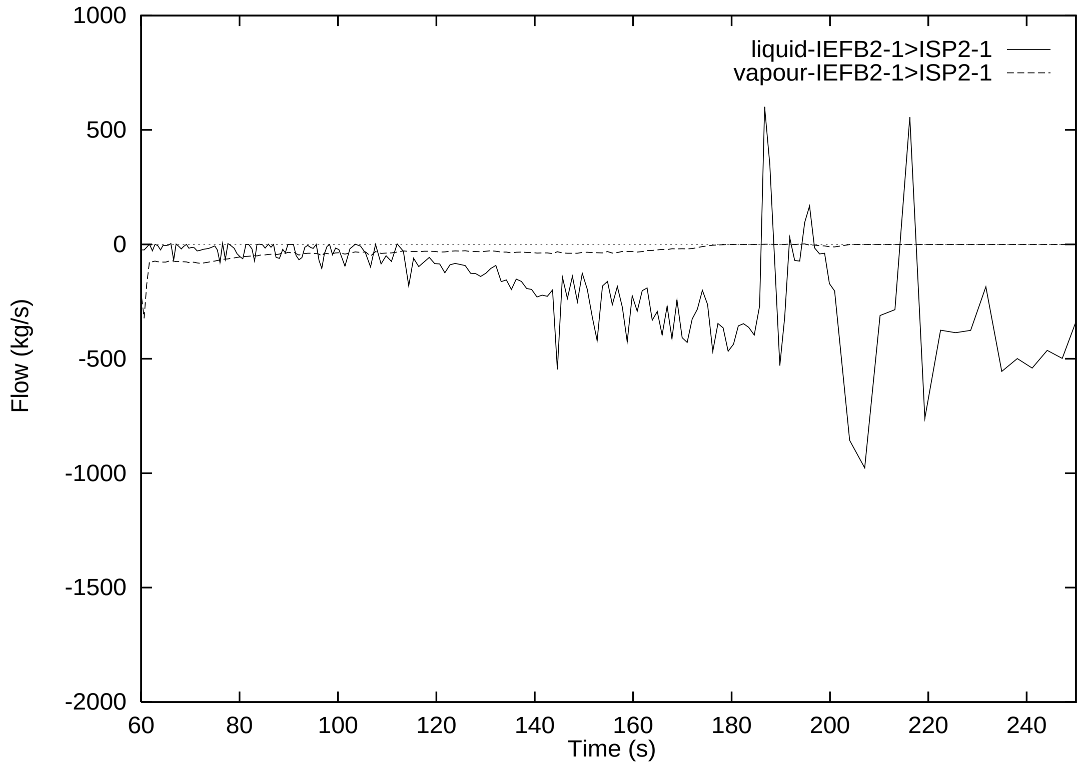


Figure 187 End Fitting 2 Flows for 25% RIH Break with Subsequent Loss of Class IV Power

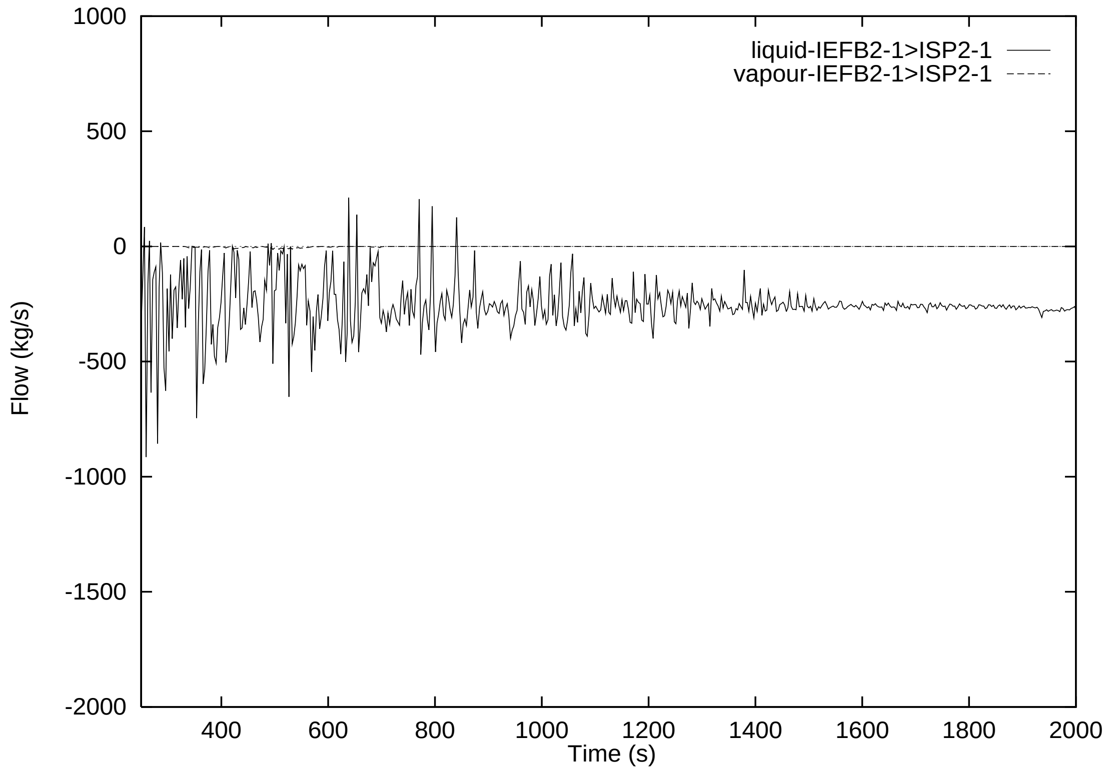


Figure 188 End Fitting 2 Flows for 25% RIH Break with Subsequent Loss of Class IV Power

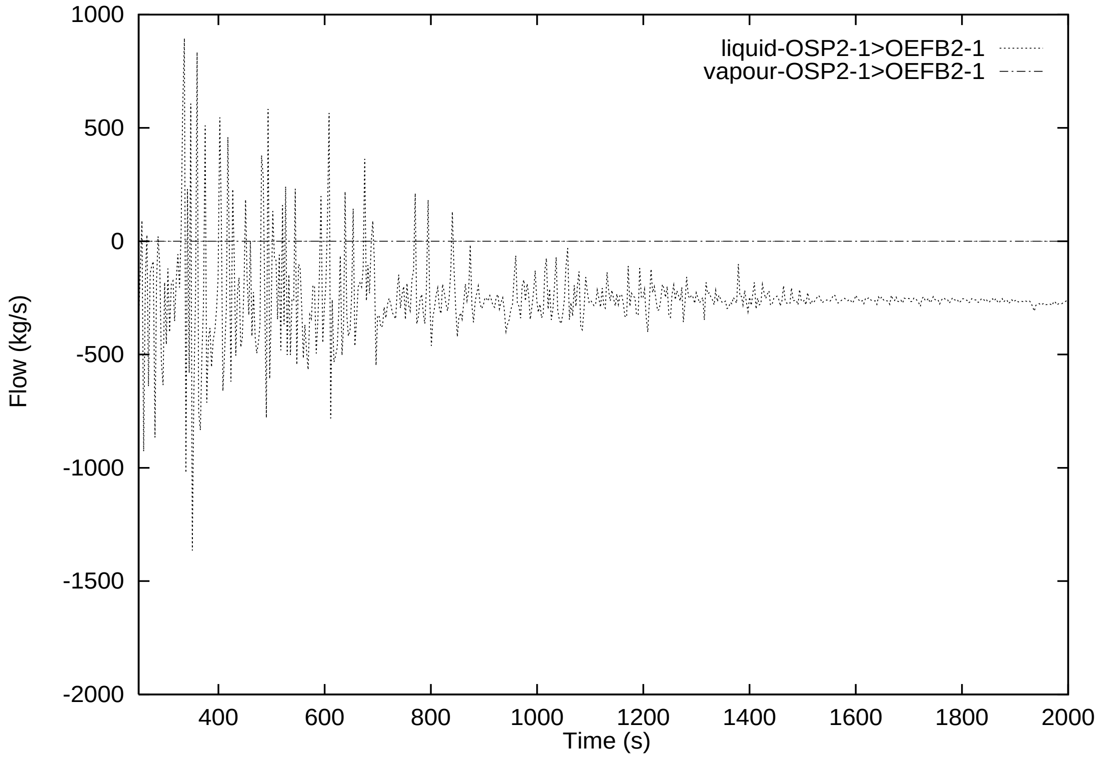


Figure 189 End Fitting 2 Flows for 25% RIH Break with Subsequent Loss of Class IV Power

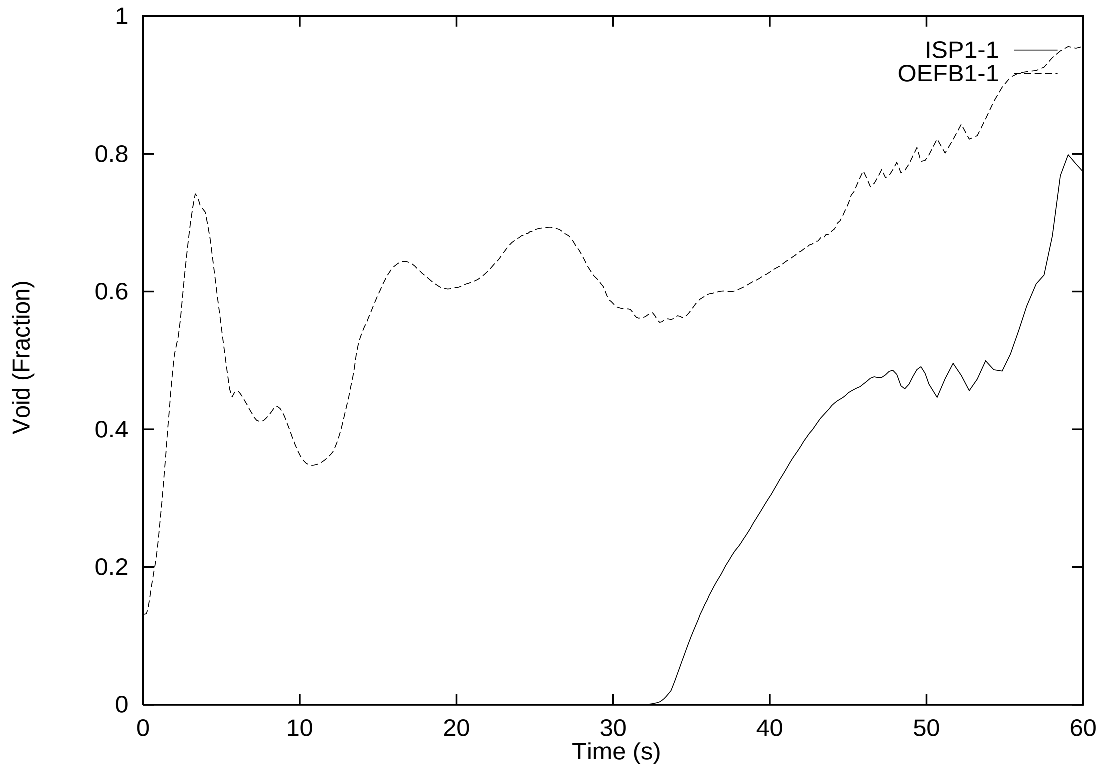


Figure 190 End Fitting Voids for 25% RIH Break with Subsequent Loss of Class IV Power

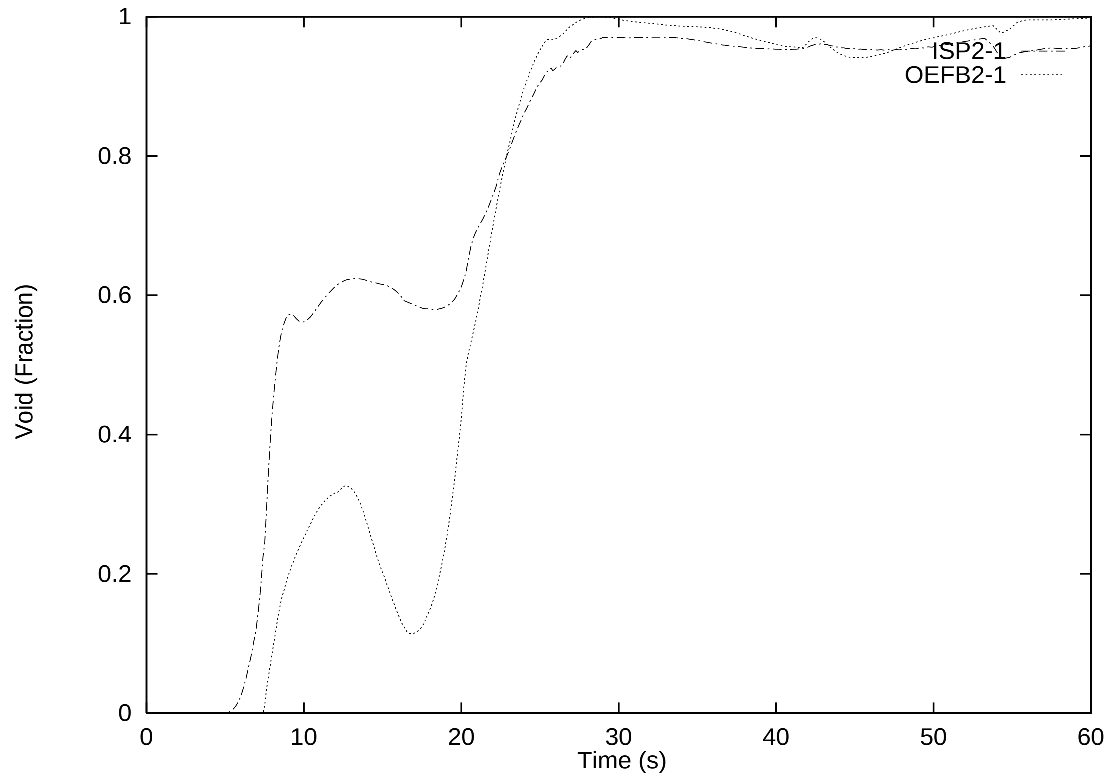


Figure 191 End Fitting Voids for 25% RIH Break with Subsequent Loss of Class IV Power

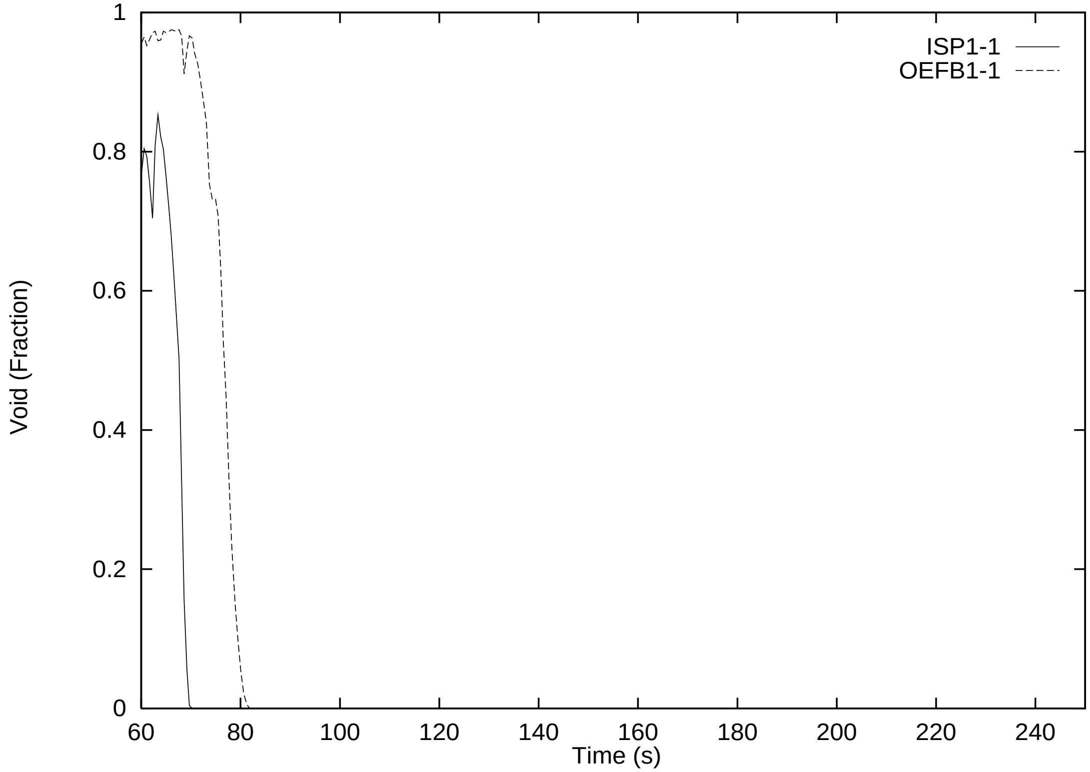


Figure 192 End Fitting Voids for 25% RIH Break with Subsequent Loss of Class IV Power

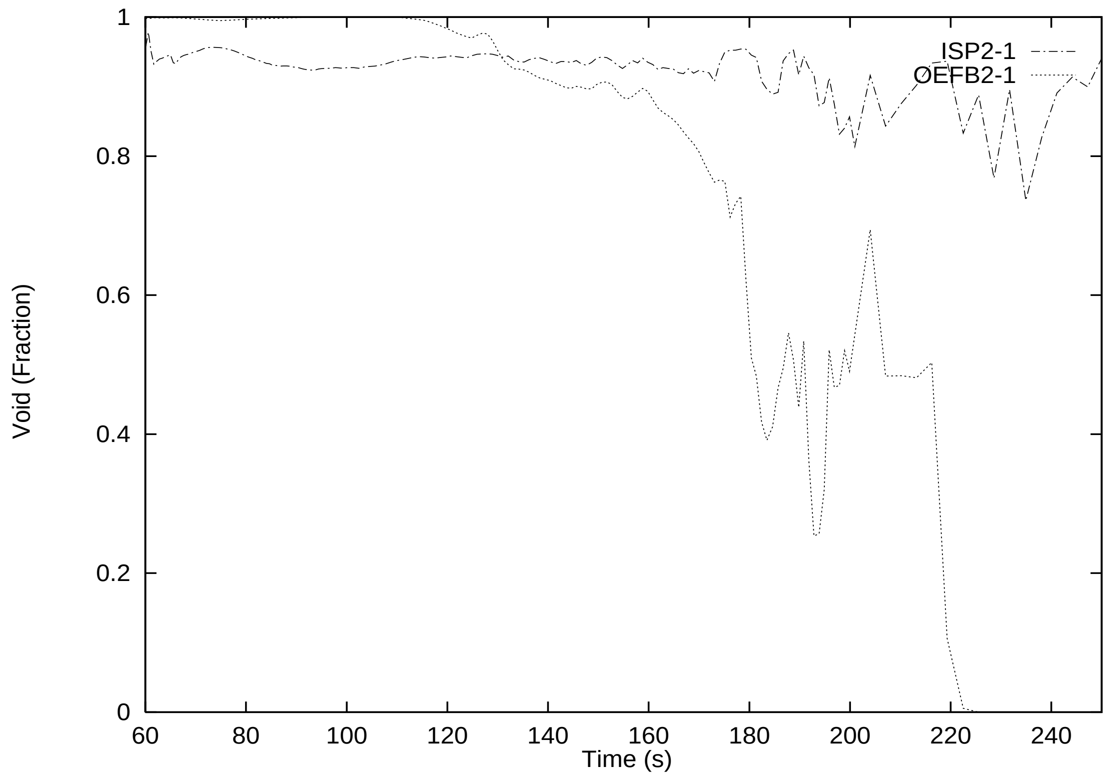


Figure 193 End Fitting Voids for 25% RIH Break with Subsequent Loss of Class IV Power

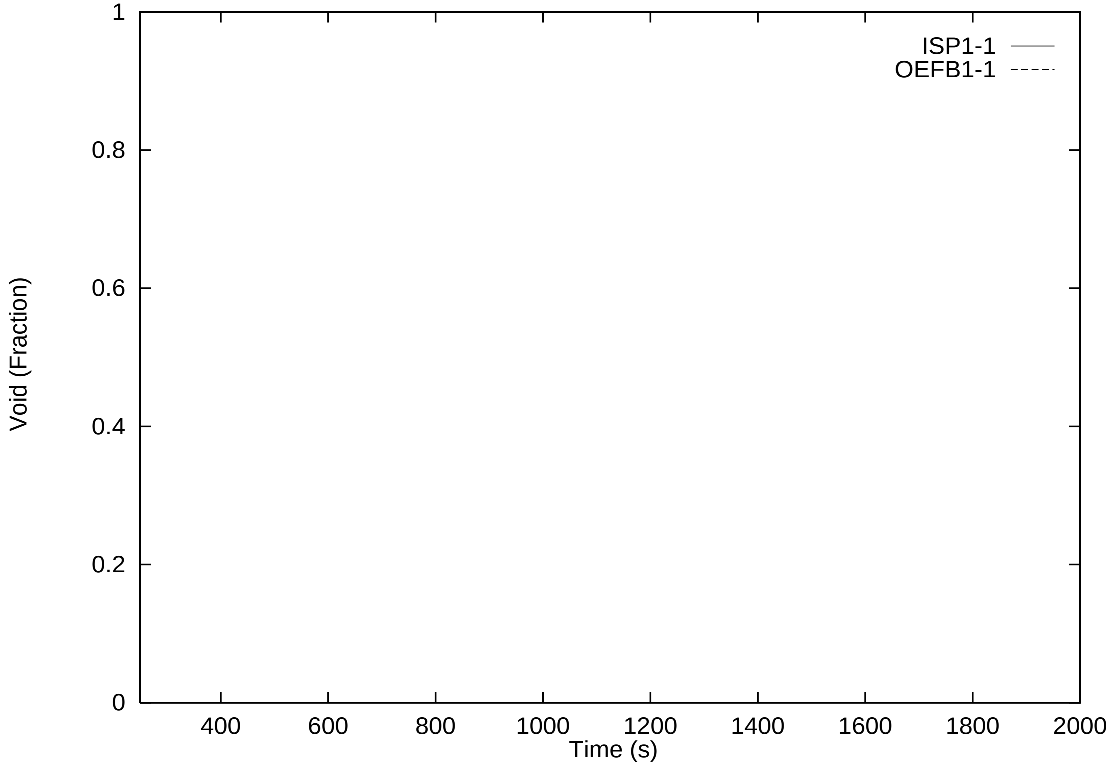


Figure 194 End Fitting Voids for 25% RIH Break with Subsequent Loss of Class IV Power

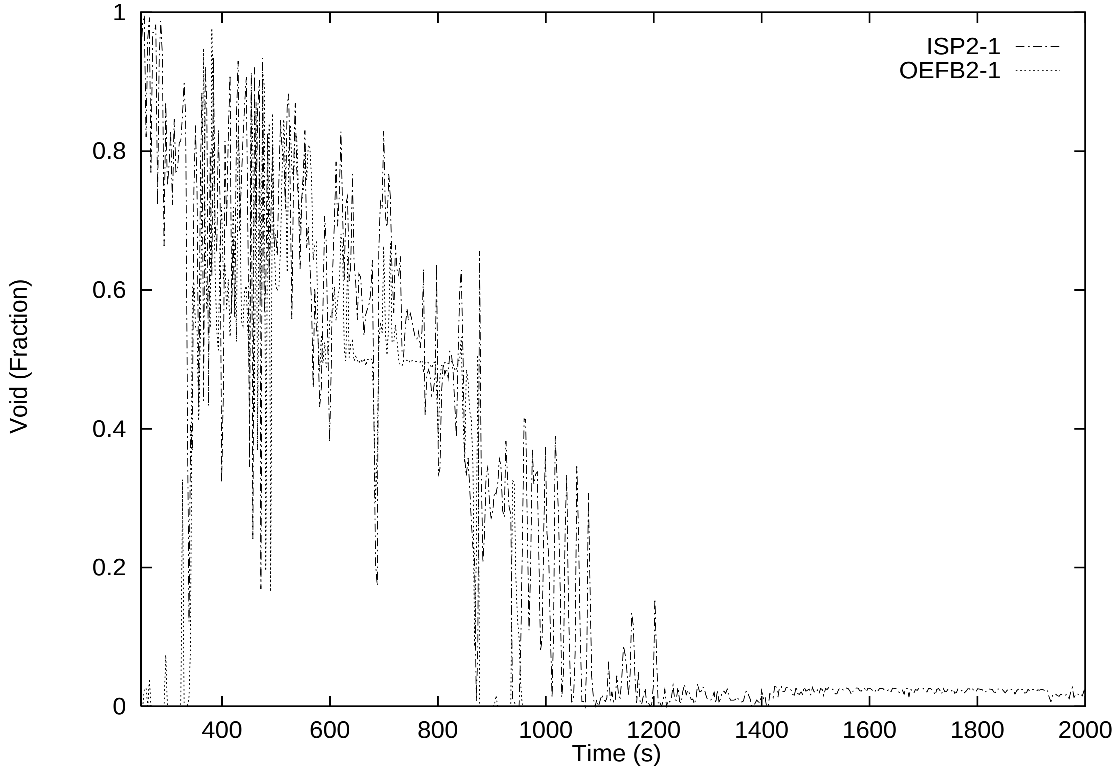


Figure 195 End Fitting Voids for 25% RIH Break with Subsequent Loss of Class IV Power

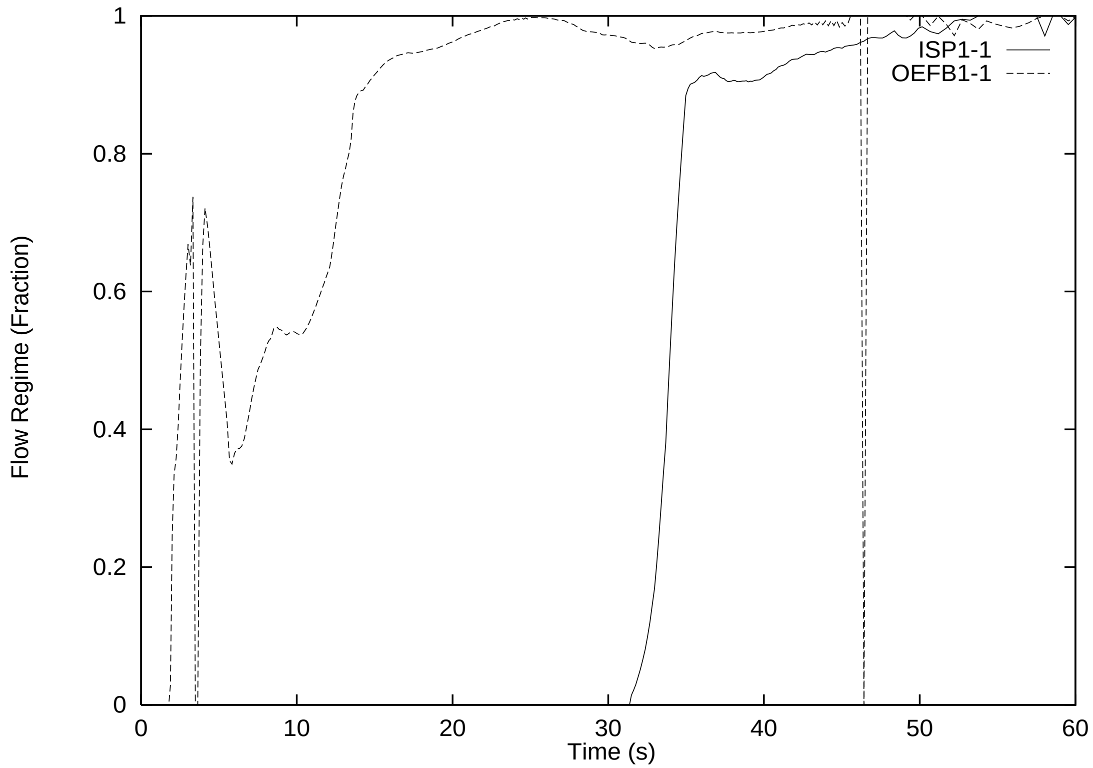


Figure 196 End Fitting Flow Regime for 25% RIH Break with Subsequent Loss of Class IV Power

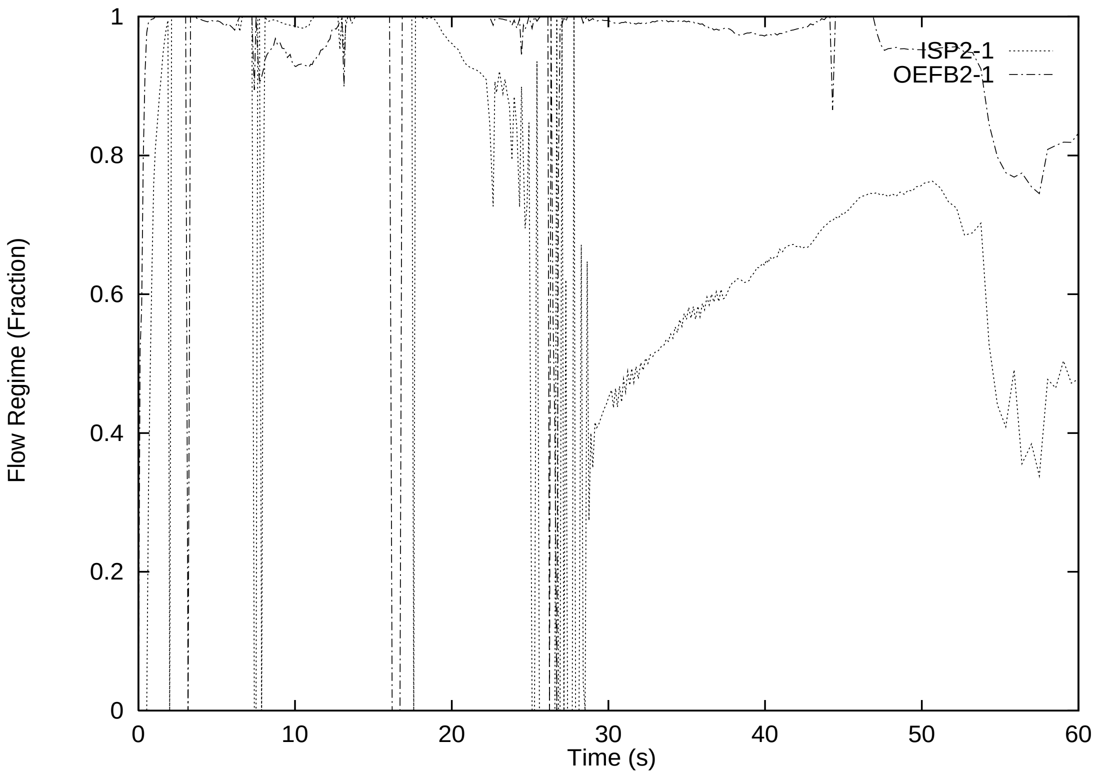


Figure 197 End Fitting Flow Regime for 25% RIH Break with Subsequent Loss of Class IV Power

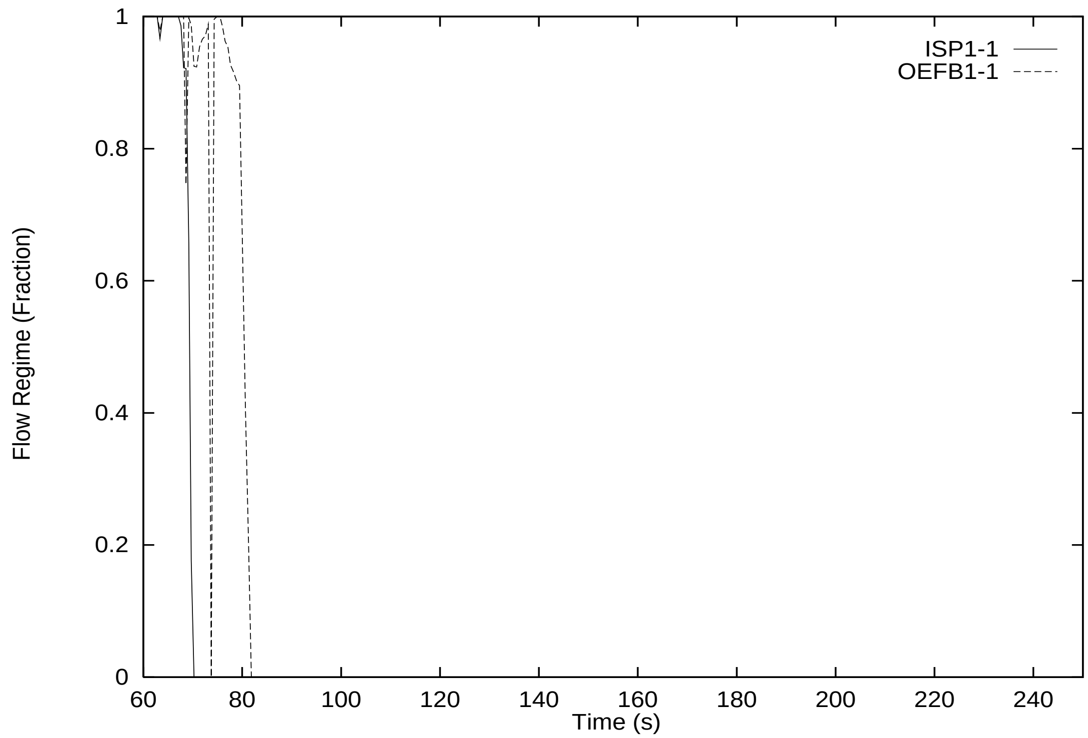


Figure 198 End Fitting Flow Regime for 25% RIH Break with Subsequent Loss of Class IV Power

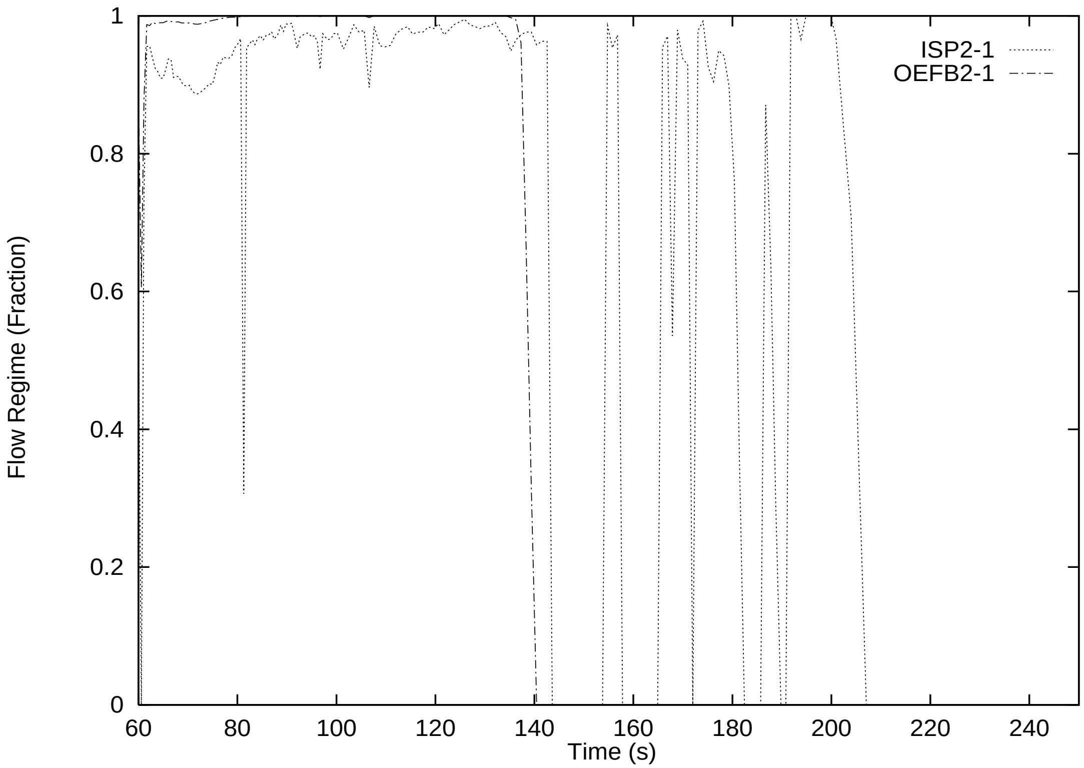


Figure 199 End Fitting Flow Regime for 25% RIH Break with Subsequent Loss of Class IV Power

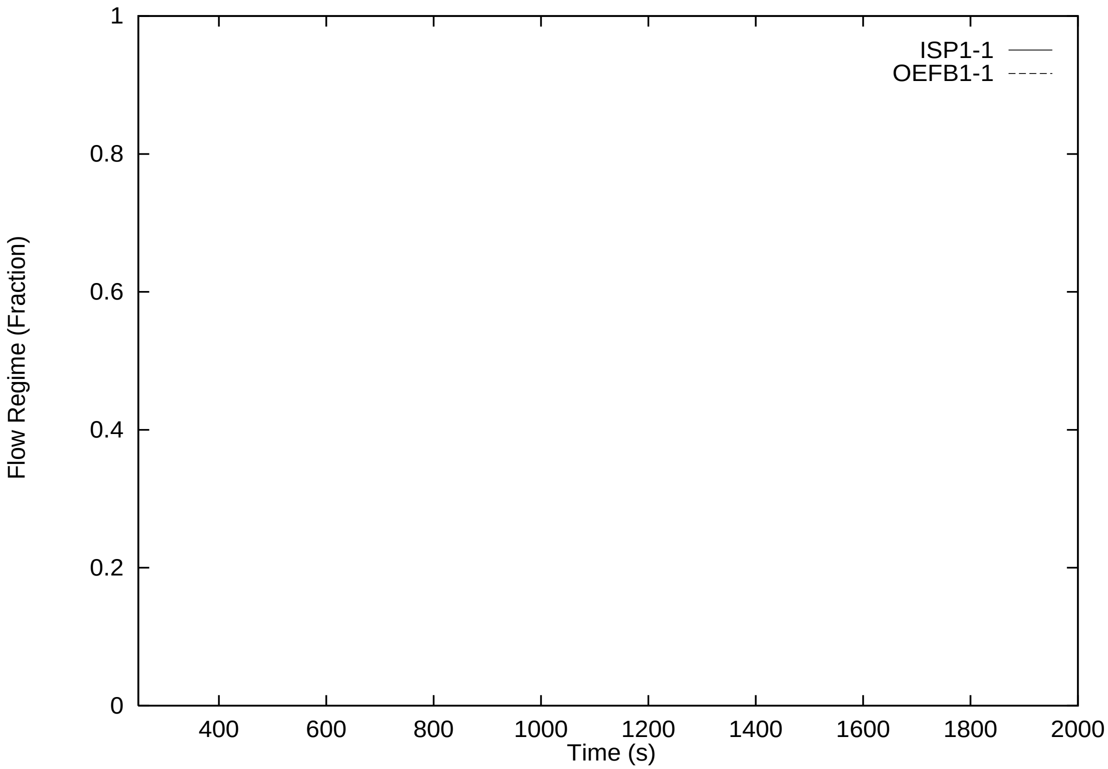


Figure 200 End Fitting Flow Regime for 25% RIH Break with Subsequent Loss of Class IV Power

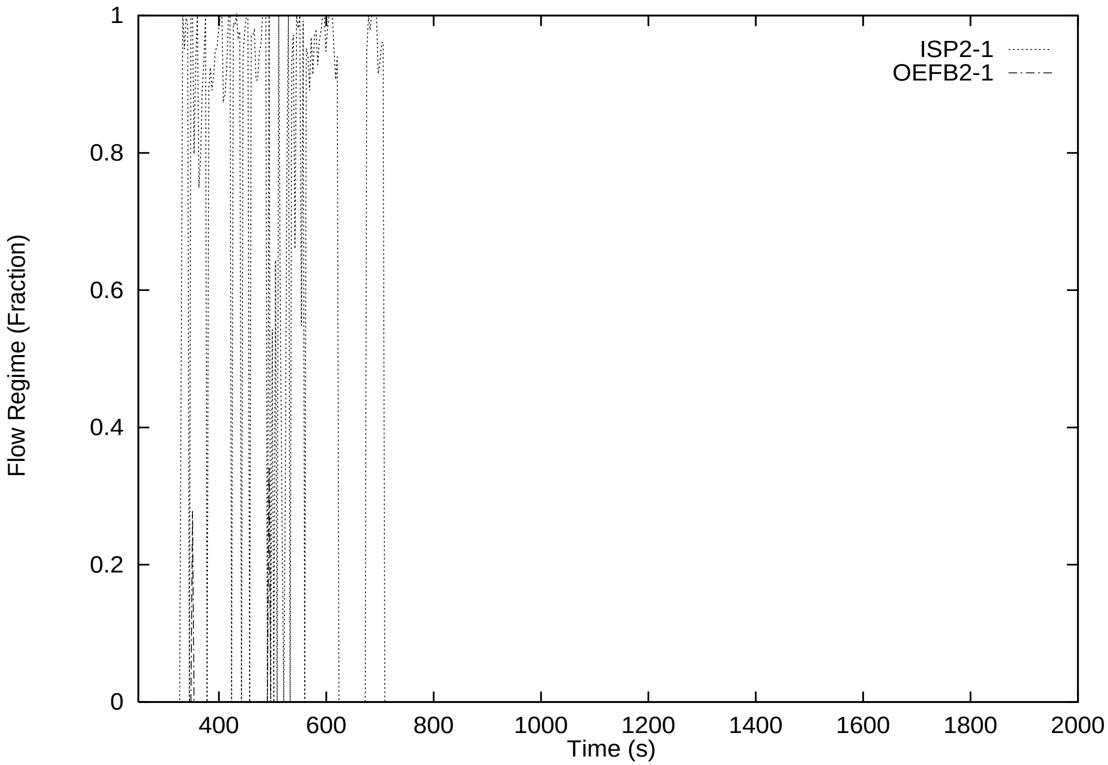


Figure 201 End Fitting Flow Regime for 25% RIH Break with Subsequent Loss of Class IV Power

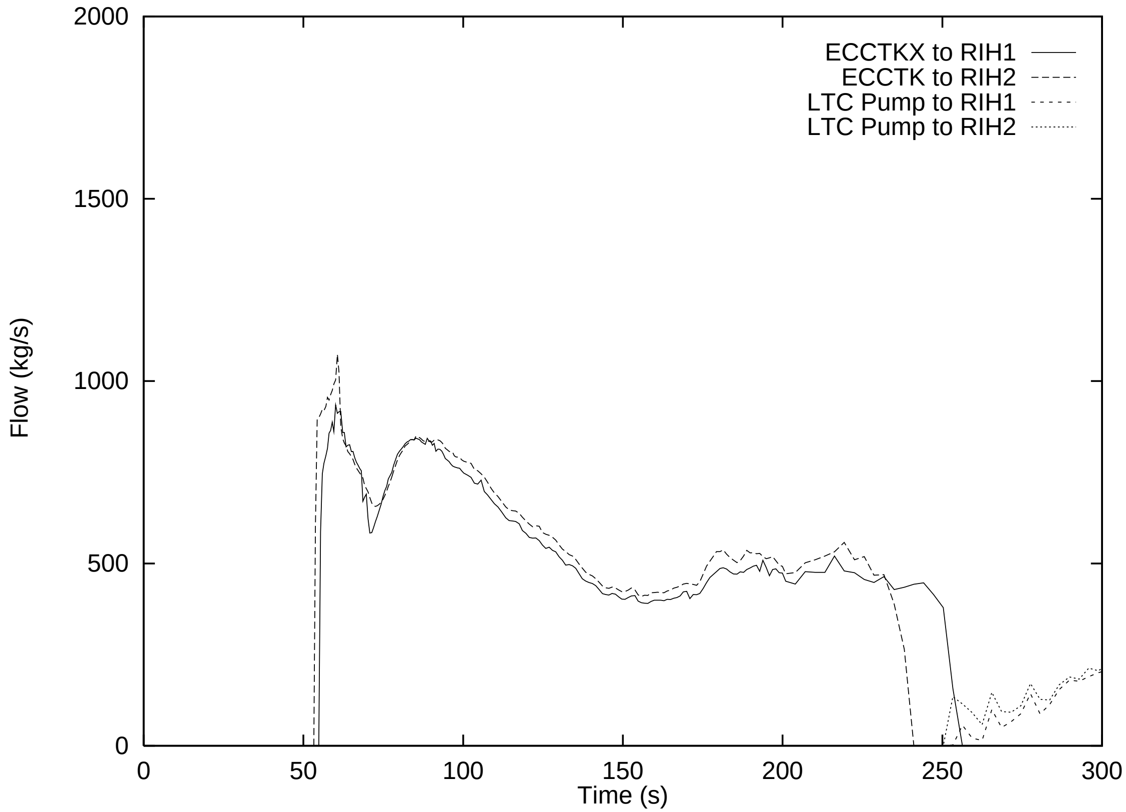


Figure 202 ECI Flows for 25% RIH Break with Subsequent Loss of Class IV Power

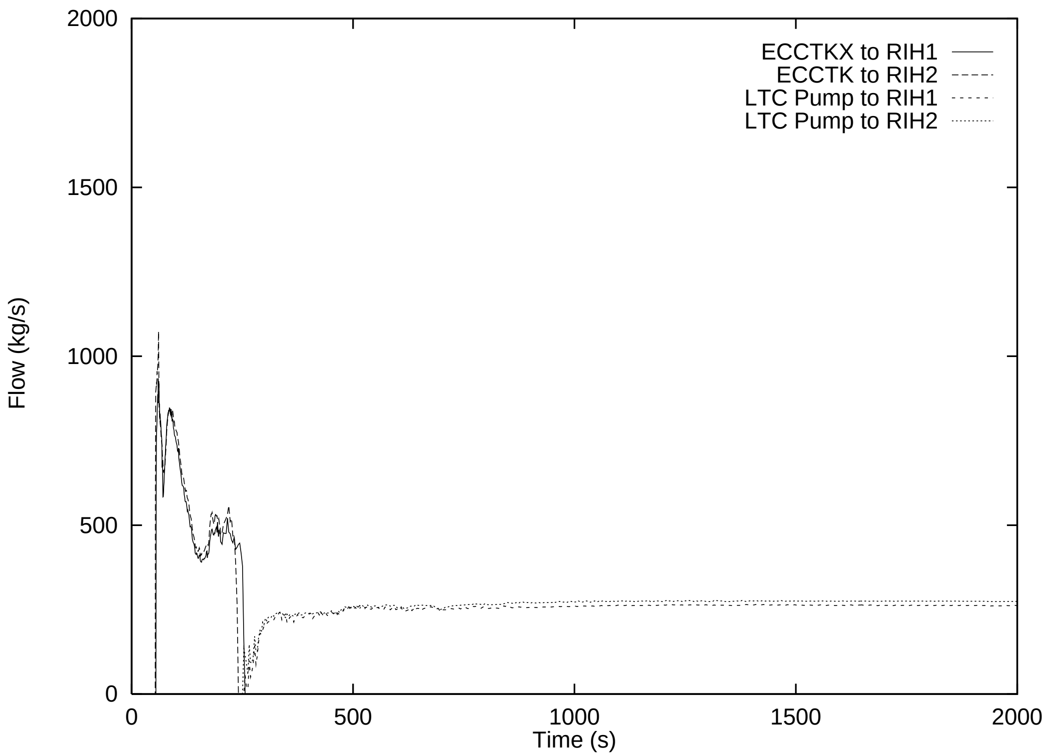


Figure 203 ECC Flows for 25% RIH Break with Subsequent Loss of Class IV Power

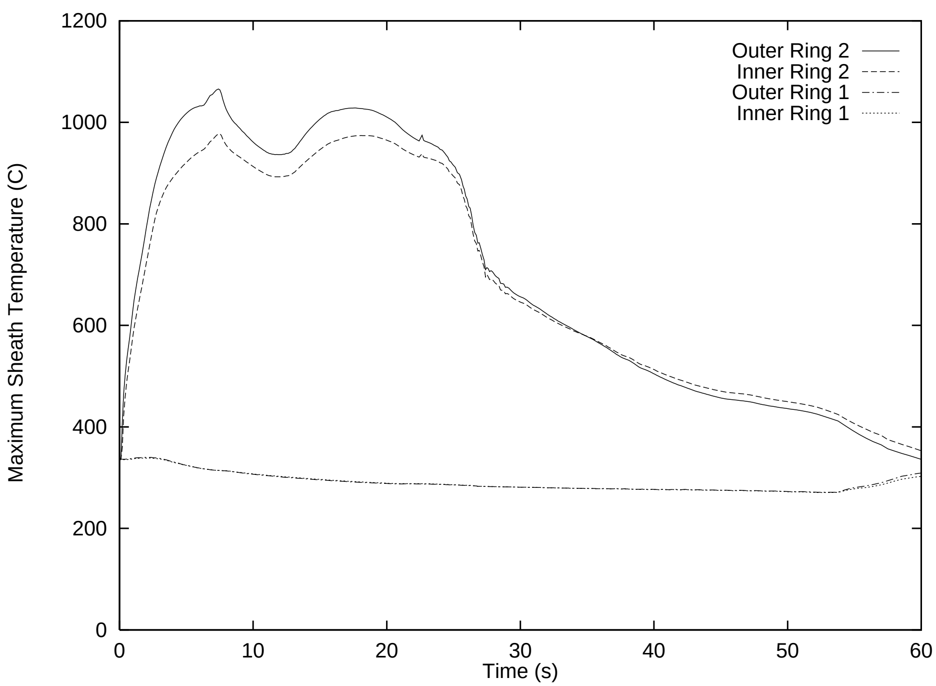


Figure 204 Maximum Sheath Temperature for 25% RIH Break with Subsequent Loss of Class IV Power

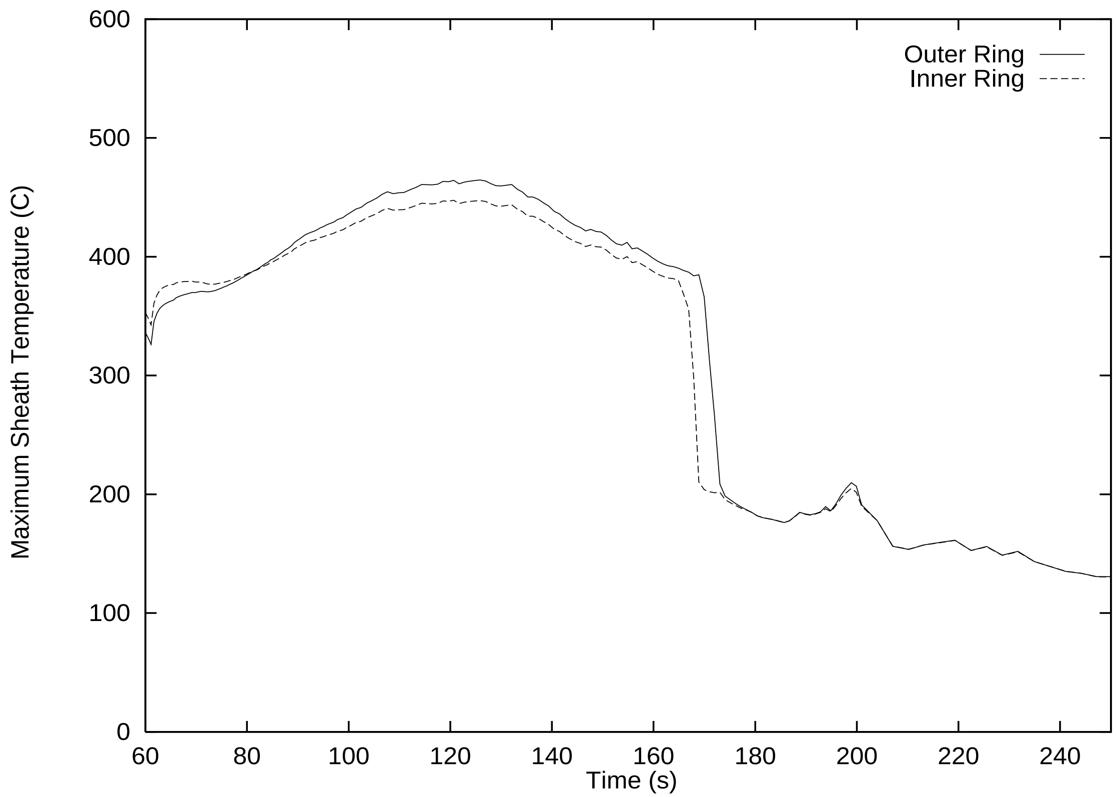


Figure 205 Maximum Sheath Temperature for 25% RIH Break with Subsequent Loss of Class IV Power

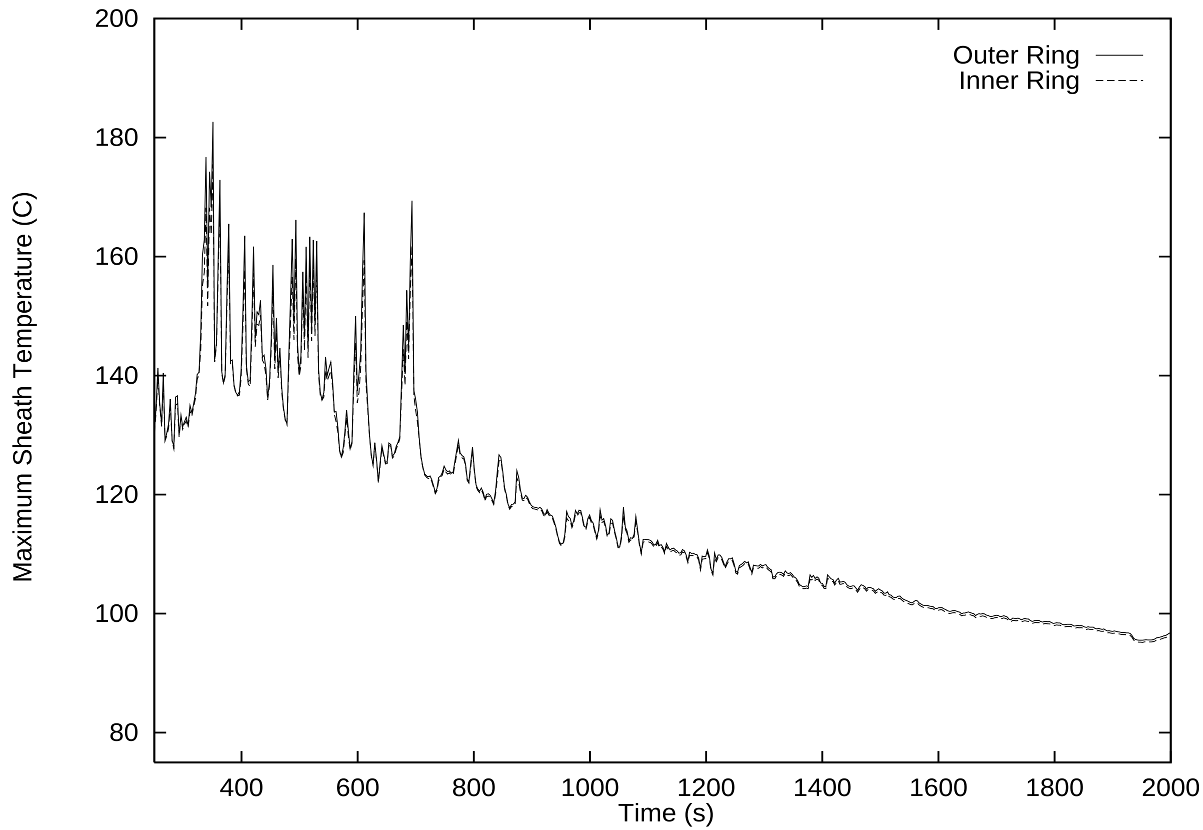


Figure 206 Maximum Sheath Temperature for 25% RIH Break with Subsequent Loss of Class IV Power

Appendix A

Phenomenon Definitions

The phenomenon definitions used in this Phenomena Identification and Ranking Table (PIRT) activity are shown in the following table.

PIRT Term	Definition
Availability	The component functions as per design.
Boiling - film	A boiling regime in which vapour blankets all or an appreciable portion of the heating surface.
Boiling - nucleate	A boiling regime in which bubble formation is at the liquid-solid interface that results in slow surface temperature increases for a relatively large increase in surface heat flux.
Boiling - subcooled	A boiling regime beginning with the onset of nucleate boiling and continuing to the onset of saturated boiling. The boundary between the latter two regimes occurring when the bulk liquid temperature approaches saturation at the given pressure.
Boiling - transition	A boiling regime that spans the boiling surface between critical heat flux and minimum film boiling (i.e., Liedenfrost point).
Break orientation	The azimuthal orientation of the break.
Change in path/state (open/close)	The flow path within the component is opened or closed during the phase.
Coastdown/rundown	The reduction in time of pressure driven flow following pump trip.
Condensation – fluid-to-surface	The process whereby steam is cooled due to contact with a colder surface, resulting in a change of phase from vapor to liquid at the surface.
Condensation – inter-phase	The process whereby steam is cooled due to contact with a colder liquid, resulting in a change of phase from vapor to liquid at the interface between the two phases.
Conduction	Heat conduction is the process by which heat flows from a region of higher temperature to a region of lower temperature within a medium or between different mediums in direct physical contact.
Constrained axial expansion	The axial compression of the fuel string when the fuel string axial deformation exceeds the sum of the axial thermal expansion of the pressure tube and the residual fuel string-to-shield plug axial gap.
Contact heat transfer – bearing pad-to-pressure tube	Heat transfer from a bearing pad to its pressure tube. It is a combination of conduction, convection and radiant heat transfer, all of which are lumped together for an effective heat transfer.

PIRT Term	Definition
Contact heat transfer – fuel element-to-pressure tube	Heat transfer from a fuel element to its pressure tube. It is a combination of conduction, convection and radiant heat transfer, all of which are lumped together for an effective heat transfer.
Convective heat transfer	The process of energy transport to a fluid be the combined action of heat conduction, energy storage, and mixing motion.
Critical flow	The maximum possible flow through a flow-constricting item of hardware, usually a nozzle, orifice, or break in a pipe.
Critical Heat Flux (CHF)	The maximum heat flux that defines the boundary between nucleate and transition boiling regimes. It is also known as dryout, burnout or boiling crisis.
Debris generation	The formation of debris by jet impingement on reactor structures and the sweeping of containment debris by water flow.
Debris transport	The movement of debris towards the reactor-building sump by water flows within containment.
Decay heating	The heat from delayed emission of beta and gamma rays.
De-entrainment	The process whereby liquid is mechanically removed by impingement (de-entrained) from a steam flow.
Deformation	A change in a component's geometry while under stress or the movement of one component towards another. For the current application it can apply to pressure-tube deformation, fuel-element deformation or calandria-tube deformation.
Deformation – fuel string relocation	A change in the fuel string/fuel bundle geometry owing to the axial movement and impact of a fuel string by hydraulic forces following a large break in the Heat Transport System.
Deformation (includes failure)	Movement of one component towards another. For the current application it applies to pressure-tube deformation (both ballooning and sagging) and, if deformation is significant, failure.
Departure from Nucleate Boiling (DNB)	(see Boiling – transition).
Distribution – multiple channels	Multiple flow paths from the headers.
Embrittlement	A marked reduction in fuel sheath ductility due to oxygen or hydrogen uptake.
End power peaking	The enhanced power at the axial ends of fuel string of pellets in a fuel bundle, due to reduced neutron absorption beyond the fuel string.

PIRT Term	Definition
Entrainment	The process whereby liquid is captured (entrained) by a high-velocity steam flow.
Failure	The breaching of the fuel sheath.
Fission heating	Heat promptly generated in the fuel by fissions.
Flashing	Void formation without heat addition.
Flow – counter current	The process whereby liquid flows opposite (counter) to the gas flow direction.
Flow – gravity driven	The downward flow of fluid under the influence of gravity.
Flow – gravity driven (draining)	The downward flow of fluid on a surface under the influence of gravity.
Flow – natural circulation	Flow driven by density differences.
Flow – pressure driven	Flow leaving a component under the influence of an upstream forcing function.
Flow - stalled (stagnation)	The stagnation of flow in a fuel channel caused by an inlet header break where the hydraulic losses out the break balance the hydraulic forces in a feeder / fuel channel arrangements, causing the flow to stagnate.
Flow blockage	A severe restriction of flow by debris relocation.
Flow regime	The distribution of phases (liquid/vapour) within a pipe, e.g., bubbly flow, slug, stratified, annular. Also the friction between phases caused by velocity differences at the vapour/liquid interface.
Flow reversal	Change in established flow direction to the reverse direction.
Forced convection to liquid	The process of energy transport to a fluid by the combined action of heat conduction, energy storage, and mixing motion, where the fluid is single-phase liquid.
Forced convection to vapour	The process of energy transport to a fluid by the combined action of heat conduction, energy storage, and mixing motion, where the fluid is single-phase vapour.
Fragmentation	The breakup of a fuel element due to large internal or external forces.
Gap conductance	The overall thermal resistance to the flow of heat between two opposing surfaces.
Jet impingement (debris generation)	The creation of debris from the impact of the discharge flow from the break on nearby structure.
Level	The vertical height of a column of single or two-phase fluid.
Level - swelling	The increase in the level of a lower (primarily) liquid region due to vapour generation.
Mechanical interaction	The physical interaction between two contacting surfaces.

PIRT Term	Definition
Melting	The change in state of a material from a solid to a liquid.
Mixing – multiple fluid streams	The combining of two or more fluid streams into a single stream.
Multi-dimensional flow	Flow with two or more dominant velocity vectors.
Non-condensable gas generation	The impact of the presence of non-condensable gases upon heat transfer or any other phenomenon such as flow, condensation, flashing, and vapor volume expansion.
Oscillations	The periodic variation of any given hydraulic characteristic between two values.
Oxidation	A chemical reaction that increases the oxidation content of a material. Of specific interest is cladding oxidation, which occurs at elevated temperatures, which can occur only under accident conditions.
Post-dryout heat transfer	Wall-to-coolant heat transfer where the heated wall is no longer covered by liquid in a sustainable way when the heat flux through the wall exceeds CHF. Post-dryout (PDO) heat transfer includes transition boiling and film boiling. (see also Boiling – transition and Boiling –film)
Pipe thrust	The reaction force to the break discharge flow.
Pressure	The force (per unit area) applied by a fluid on its surroundings.
Pressure drop (1-phase, 2-phase)	The reduction in pressure with distance
Pump performance / characteristics	The behavior of a pump under all normal and off-normal conditions.
Radial power distribution	Decrease in power density in the fuel due to decrease of neutron flux in the fuel bundle interior.
Radiant heat transfer	The transfer of energy from a higher temperature body to a lower temperature body without relying on the intervening medium, i.e., the transfer can take place in a vacuum.
Reactivity – coolant temperature change	The change in core reactivity due to the effect of change in coolant temperature on coolant density and neutron spectrum.
Reactivity – density / void	The change in core reactivity due to an increase or decrease in the amount of void in the cooling fluid.
Reactivity – device movement	The change in core reactivity due to the firing of either shutdown system, movement of mechanical control absorbers, or movement of zonal control rods.
Reactivity – fuel string relocation	A change in core reactivity due to fuel string relocation from flow reversal in the fuel channel.
Reactivity – fuel temperature change	The change in core reactivity due to the effect of a change in fuel temperature.

PIRT Term	Definition
Reactivity – isotope change	The change in core reactivity due to an increase or decrease in isotopic concentration of materials in the nuclear lattice.
Reactivity – moderator poison	The change in core reactivity due to a change in moderator poison concentration.
Reactivity – moderator purity	A change in core reactivity due to a change in moderator purity.
Reactivity – moderator temperature	The change in neutron spectrum resulting from a change in moderator temperature (commonly called the moderator-temperature reactivity coefficient).
Reactivity effect of firing SDS1	Rapid core-reactivity reduction due to increased absorption of neutrons in inserted rods.
Reactivity effect of firing SDS2	Rapid core-reactivity reduction due to increased absorption of neutrons in injected poison.
Refill	The point at which a pipe is completely filled with liquid. Refill is indicated experimentally by a decrease in surface temperature below the local saturation temperature.
Reflux condensation	Condensation of upward-flowing steam in a vertical, cooled pipe, and interaction with the liquid condensate film, which may drain downwards, or be dragged upwards by the steam.
Rewet	The post-dryout process in which liquid once again resumes intimate contact with a heated surface.
Solid-to-solid (fuel element-to-pressure tube) contact heat transfer	Heat transfer from a fuel element to its pressure tube. It is a combination of conduction, convection and radiation heat transfer, all of which are lumped together for an effective heat transfer.
Stored energy release	The process by which the energy within a solid structure is released to a lower energy state through one or more heat transfer processes, e.g., conduction and convection.
Swelling – fission product	The increase in fuel volume due to the formation of pressurized, gas filled pores/bubbles and inclusion of solid fission products in the fuel matrix.
Temperature	A measure of the sensible heat energy content per unit volume.
Void generation from heat transfer	The generation of vapour (boiling) due to heat transfer with a wall or evaporation at the two-phase interface.
Waterhammer	Pressure pulse(s) created as a result of a rapid change in fluid velocity.
Waterhammer – condensation induced	Pressure pulse(s) created as a result of a rapid change in fluid velocity, where the acceleration of the fluid is driven by condensation of a trapped vapour volume.