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IFSAR Formatting Legend

Color	Description
Black	Original Westinghouse AP1000 DCD Revision 19 content (part of plant-specific DCD)
Purple	Departures from AP1000 DCD Revision 19 content (part of plant-specific DCD)
Blue	Standard FSAR content
Green	Site-specific FSAR content
Red	Linked cross-references (chapters, appendices, sections, subsections, tables, figures, and references)

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Chapter 15 Accident Analyses

15.0.1 Classification of Plant Conditions

The ANSI 18.2 (Reference 1) classification divides plant conditions into four categories according to anticipated frequency of occurrence and potential radiological consequences to the public. The four categories are as follows:

- Condition I: Normal operation and operational transients
- Condition II: Faults of moderate frequency
- Condition III: Infrequent faults
- Condition IV: Limiting faults

The basic principle applied in relating design requirements to each of the conditions is that the most probable occurrences should yield the least radiological risk, and those extreme situations having the potential for the greatest risk should be those least likely to occur. Where applicable, reactor trip and engineered safeguards functioning are assumed to the extent allowed by considerations such as the single failure criterion in fulfilling this principle.

15.0.1.1 Condition I: Normal Operation and Operational Transients

Condition I occurrences are those that are expected to occur frequently or regularly in the course of power operation, refueling, maintenance, or maneuvering of the plant. As such, Condition I occurrences are accommodated with margin between a plant parameter and the value of that parameter requiring either automatic or manual protective action.

Because Condition I events occur frequently, they must be considered from the point of view of their effect on the consequences of fault conditions (Conditions II, III, and IV). In this regard, analysis of each fault condition described is generally based on a conservative set of initial conditions corresponding to adverse conditions that can occur during Condition I operation.

A typical list of Condition I events follows.

Steady-State and Shutdown Operations

See Table 1.1-1 of Chapter 16.

Operation with Permissible Deviations

Various deviations that occur during continued operation as permitted by the plant Technical Specifications are considered in conjunction with other operational modes. These deviations include the following:

- Operation with components or systems out of service (such as an inoperable rod cluster control assembly [RCCA])

- Leakage from fuel with limited cladding defects

- Excessive radioactivity in the reactor coolant:

  - Fission products
  - Corrosion products
  - Tritium

- Operation with steam generator tube leaks

- Testing

Operational Transients

- Plant heatup and cooldown
- Step load changes (up to ±10 percent)
- Ramp load changes (up to 5 percent/minute)
- Load rejection up to and including design full-load rejection transient

15.0.1.2 Condition II: Faults of Moderate Frequency

These faults, at worst, result in a reactor trip with the plant being capable of returning to operation. By definition, these faults (or events) do not propagate to cause a more serious fault (Condition III or IV events). In addition, Condition II events are not expected to result in fuel rod failures, reactor coolant system failures, or secondary system overpressurization. The following faults are included in this category:

- Feedwater system malfunctions that result in a decrease in feedwater temperature (see Subsection 15.1.1)

- Feedwater system malfunctions that result in an increase in feedwater flow (see Subsection 15.1.2)

- Excessive increase in secondary steam flow (see Subsection 15.1.3)

- Inadvertent opening of a steam generator relief or safety valve (see Subsection 15.1.4)

- Inadvertent operation of the passive residual heat removal heat exchanger (see Subsection 15.1.6)

- Loss of external electrical load (see Subsection 15.2.2)

- Turbine trip (see Subsection 15.2.3)

- Inadvertent closure of main steam isolation valves (see Subsection 15.2.4)

- Loss of condenser vacuum and other events resulting in turbine trip (see Subsection 15.2.5)

- Loss of ac power to the station auxiliaries (see Subsection 15.2.6)

- Loss of normal feedwater flow (see Subsection 15.2.7)

- Partial loss of forced reactor coolant flow (see Subsection 15.3.1)

- Uncontrolled RCCA bank withdrawal from a subcritical or low-power startup condition (see Subsection 15.4.1)

- Uncontrolled RCCA bank withdrawal at power (see Subsection 15.4.2)

- RCCA misalignment (dropped full-length assembly, dropped full-length assembly bank, or statically misaligned assembly) (see Subsection 15.4.3)

- Startup of an inactive reactor coolant pump at an incorrect temperature (see Subsection 15.4.4)

- Chemical and volume control system malfunction that results in a decrease in the boron concentration in the reactor coolant (see Subsection 15.4.6)
- Inadvertent operation of the passive core cooling system during power operation (see Subsection 15.5.1)
- Chemical and volume control system malfunction that increased reactor coolant inventory (see Subsection 15.5.2)
- Inadvertent opening of a pressurizer safety valve (see Subsection 15.6.1)
- Break in instrument line or other lines from the reactor coolant pressure boundary that penetrate containment (see Subsection 15.6.2)

15.0.1.3 Condition III: Infrequent Faults

Condition III events are faults that may occur infrequently during the life of the plant. They may result in the failure of only a small fraction of the fuel rods. The release of radioactivity is not sufficient to interrupt or restrict public use of those areas beyond the exclusion area boundary, in accordance with the guidelines of 10 CFR 50.34. By definition, a Condition III event alone does not generate a Condition IV event or result in a consequential loss of function of the reactor coolant system or containment barriers. The following faults are included in this category:

- Steam system piping failure (minor) (see Subsection 15.1.5)
- Complete loss of forced reactor coolant flow (see Subsection 15.3.2)
- RCCA misalignment (single RCCA withdrawal at full power) (see Subsection 15.4.3)
- Inadvertent loading and operation of a fuel assembly in an improper position (see Subsection 15.4.7)
- Inadvertent operation of automatic depressurization system (see Subsection 15.6.1)
- Loss-of-coolant accidents (LOCAs) resulting from a spectrum of postulated piping breaks within the reactor coolant pressure boundary (small break) (see Subsection 15.6.5)
- Gas waste management system leak or failure (see Subsection 15.7.1)
- Liquid waste management system leak or failure (see Subsection 15.7.2)
- Release of radioactivity to the environment due to a liquid tank failure (see Subsection 15.7.3)
- Spent fuel cask drop accidents (see Subsection 15.7.5)

15.0.1.4 Condition IV: Limiting Faults

Condition IV events are faults that are not expected to take place, but are postulated because their consequences include the potential of the release of significant amounts of radioactive material. They are the faults that must be designed against, and they represent limiting design cases. Condition IV faults are not to cause a fission product release to the environment resulting in doses in excess of the guideline values of 10 CFR 50.34. A single Condition IV event is not to cause a consequential loss of required functions of systems needed to cope with the fault, including those of the emergency core cooling system and the containment. The following faults are classified in this category:

- Steam system piping failure (major) (see Subsection 15.1.5)
- Feedwater system pipe break (see Subsection 15.2.8)
- Reactor coolant pump shaft seizure (locked rotor) (see Subsection 15.3.3)
- Reactor coolant pump shaft break (see Subsection 15.3.4)
- Spectrum of RCCA ejection accidents (see Subsection 15.4.8)
- Steam generator tube rupture (see Subsection 15.6.3)
- LOCAs resulting from a spectrum of postulated piping breaks within the reactor coolant pressure boundary (large break) (see Subsection 15.6.5)
- Design basis fuel handling accidents (see Subsection 15.7.4)

15.0.2 Optimization of Control Systems

A control system setpoint study is performed prior to plant operation to simulate performance of the primary plant control systems and overall plant performance. In this study, emphasis is placed on the development of the overall plant control systems that automatically maintain conditions in the plant within the allowed operating window and with optimum control system response and stability over the entire range of anticipated plant operating conditions. The control system setpoints are developed using the nominal protection and safety monitoring system setpoints implemented in the plant. Where appropriate (such as in margin to reactor trip analyses), instrumentation errors are considered and are applied in an adverse direction with respect to maintaining system stability and transient performance. The accident analysis and plant control system setpoint study in combination show that the plant can be operated and meet both safety and operability requirements throughout the core life and for various levels of power operation.

The plant control system setpoint study is comprised of analyses of the following control systems: plant control, axial offset control, rapid power reduction, steam dump (turbine bypass), steam generator level, pressurizer pressure, and pressurizer level.

15.0.3 Plant Characteristics and Initial Conditions Assumed in the Accident Analyses

15.0.3.1 Design Plant Conditions

Table 15.0-1 lists the principal power rating values assumed in the analyses performed. The thermal power output includes the effective thermal power generated by the reactor coolant pumps. Selected AP1000 loop layout elevations are shown in Figure 15.0.3-2 to aid in interpreting plots shown in other Chapter 15 subsections.

The values of other pertinent plant parameters used in the accident analyses are given in Table 15.0-3.

15.0.3.2 Initial Conditions

For most accidents that are departure from nucleate boiling (DNB) limited, nominal values of initial conditions are assumed. The allowances on power, temperature, and pressure are determined on a statistical basis and are included in the departure from nucleate boiling ratio (DNBR) design limit values (see Section 4.4), as described in WCAP-11397-P-A (Reference 2). This procedure is known as the Revised Thermal Design Procedure (RTDP) and is discussed more fully in Section 4.4.

For most accidents that are not DNB limited, or for which the revised thermal design procedure is not used, the initial conditions are obtained by adding the maximum steady-state errors to rated values. The following conservative steady-state errors are assumed in the analysis:

Core power	$\pm$ 2 percent allowance for calorimetric error. The main feedwater flow measurement supports a 1-percent power uncertainty; use of a 2-percent power uncertainty is conservative.
Average reactor coolant system temperature	+6.5 or -7.0°F allowance for controller deadband and measurement errors
Pressurizer pressure	$\pm$ 50 psi allowance for steady-state fluctuations and measurement errors

Initial values for core power, average reactor coolant system temperature, and pressurizer pressure are selected to minimize the initial DNBR unless otherwise stated in the sections describing the specific accidents. Table 15.0-2 summarizes the initial conditions and computer codes used in the accident analyses.

The plant operating instrumentation selected for feedwater flow measurement is a Caldon [Cameron] LEFM CheckPlus System (Reference 201), which will be calibrated (in a certified laboratory using a piping configuration representative of the plant piping design) prior to installation and will be tested after installation in the plant in accordance with the LEFM CheckPlus commissioning procedure. This selected plant operating instrumentation has documented instrumentation uncertainties to calculate a power calorimetric uncertainty that confirms the 1% uncertainty assumed for the initial reactor power in the safety analysis bounds the calculated calorimetric power uncertainty values. The calculated calorimetric is done in accordance with a previously accepted Westinghouse methodology (Reference 202). Administrative controls implement maintenance and contingency activities related to the power calorimetric instrumentation.

15.0.3.3 Power Distribution

The transient response of the reactor system is dependent on the initial power distribution. The nuclear design of the reactor core minimizes adverse power distribution through the placement of fuel assemblies and control rods. Power distribution may be characterized by the nuclear enthalpy rise hot channel factor ($F_{\Delta H}$) and the total peaking factor (F_q). Unless specifically noted otherwise, the peaking factors used in the accident analyses are those presented in Chapter 4.

For transients that may be DNB limited, the radial peaking factor is important. The radial peaking factor increases with decreasing power level due to control rod insertion. This increase in $F_{\Delta H}$ is included in the core limits illustrated in Figure 15.0.3-1. Transients that may be departure from nucleate boiling limited are assumed to begin with an $F_{\Delta H}$, consistent with the initial power level defined in the Technical Specifications.

The axial power shape used in the DNB calculation is a chopped cosine, as discussed in Section 4.4, for transients analyzed at full power and the most limiting power shape calculated or allowed for accidents initiated at nonfull power or asymmetric RCCA conditions.

The radial and axial power distributions just described are input to the VIPRE-01 code as described in Section 4.4.

For transients that may be overpower-limited, the total peaking factor (F_q) is important. Transients that may be overpower-limited are assumed to begin with plant conditions, including power distributions, which are consistent with reactor operation as defined in the Technical Specifications.

For overpower transients that are slow with respect to the fuel rod thermal time constant (for example, the chemical and volume control system malfunction that results in a slow decrease in the boron concentration in the reactor coolant system as well as an excessive increase in secondary steam flow) and that may reach equilibrium without causing a reactor trip, the fuel rod thermal evaluations are performed as discussed in Section 4.4.

For overpower transients that are fast with respect to the fuel rod thermal time constant (for example, the uncontrolled RCCA bank withdrawal from subcritical or lower power startup and RCCA ejection incident, both of which result in a large power rise over a few seconds), a detailed fuel transient heat transfer calculation is performed.

15.0.4 Reactivity Coefficients Assumed in the Accident Analysis

The transient response of the reactor system is dependent on reactivity feedback effects, in particular, the moderator temperature coefficient and the Doppler power coefficient. These reactivity coefficients are discussed in Subsection 4.3.2.3.

In the analysis of certain events, conservatism requires the use of large reactivity coefficient values. The values used are given in Figure 15.0.4-1, which shows the upper and lower bound Doppler power coefficients as a function of power, used in the transient analysis. The justification for use of conservatively large versus small reactivity coefficient values is treated on an event-by-event basis. In some cases, conservative combinations of parameters are used to bound the effects of core life, although these combinations may not represent possible realistic situations.

15.0.5 Rod Cluster Control Assembly Insertion Characteristics

The negative reactivity insertion following a reactor trip is a function of the acceleration of the RCCAs as a function of time and the variation in rod worth as a function of rod position. For accident analyses, the critical parameter is the time of insertion up to the dashpot entry, or approximately 85 percent of the rod cluster travel. In analyses where all of the reactor coolant pumps are coasting down prior to, or simultaneous, with RCCA insertion, a time of 2.09 seconds is used for insertion time to dashpot entry.

In Figure 15.0.5-1, the curve labeled “complete loss of flow transients” shows the RCCA position versus time normalized to 2.09 seconds assumed in accident analyses where all reactor coolant pumps are coasting down. In analyses where some or all of the reactor coolant pumps are running, the RCCA insertion time to dashpot is conservatively taken as 2.47 seconds. The RCCA position versus time normalized to 2.47 seconds is also shown in Figure 15.0.5-1.

The use of such a long insertion time provides conservative results for accidents and is intended to apply to all types of RCCAs, which may be used throughout plant life. Drop time testing requirements are specified in the Technical Specifications.

Figure 15.0.5-2 shows the fraction of total negative reactivity insertion versus normalized rod position for a core where the axial distribution is skewed to the lower region of the core. An axial distribution skewed to the lower region of the core can arise from an unbalanced xenon distribution. This curve is used to compute the negative reactivity insertion versus time following a reactor trip, which is input to the point kinetics core models used in transient analyses. The bottom-skewed power distribution itself is not an input into the point kinetics core model.

There is inherent conservatism in the use of Figure 15.0.5-2 in that it is based on a skewed flux distribution, which would exist relatively infrequently. For cases other than those associated with unbalanced xenon distributions, significantly more negative reactivity is inserted than that shown in the curve, due to the more favorable axial distribution existing prior to trip.

The normalized RCCA negative reactivity insertion versus time is shown in Figure 15.0.5-3. The curves shown in this figure were obtained from Figures 15.0.5-1 and 15.0.5-2. A total negative reactivity insertion following a trip of 4 percent Δk is assumed in the transient analyses except where specifically noted otherwise. This assumption is conservative with respect to the calculated trip reactivity worth available as shown in Table 4.3-3.

The normalized RCCA negative reactivity insertion versus time curve for an axial power distribution skewed to the bottom (Figure 15.0.5-3) is used in those transient analyses for which a point kinetics core model is used. Where special analyses require use of three-dimensional or axial one-dimensional core models, the negative reactivity insertion resulting from the reactor trip is calculated directly by the reactor kinetics code and is not separable from the other reactivity feedback effects. In this case, the RCCA position versus time of Figure 15.0.5-1 is used as code input.

15.0.6 Protection and Safety Monitoring System Setpoints and Time Delays to Trip Assumed in Accident Analyses

A reactor trip signal acts to open two trip breaker sets connected in series, feeding power to the control rod drive mechanisms. The loss of power to the mechanism coils causes the mechanisms to release the RCCAs, which then fall by gravity into the core. There are various instrumentation delays associated with each trip function including delays in signal actuation, in opening the trip breakers, and in the release of the rods by the mechanisms. The total delay to trip is defined as the time delay from the time that trip conditions are reached to the time the rods are free and begin to fall. Limiting trip setpoints assumed in accident analyses and the time delay assumed for each trip function are given in Table 15.0-4a. Reference is made in that table to overtemperature and overpower ΔT trip shown in Figure 15.0.3-1.

Table 15.0-4a also summarizes the setpoints and the instrumentation delay for engineered safety features (ESF) functions used in accident analyses. Time delays associated with equipment actuated (such as valve stroke times) by ESF functions are summarized in Table 15.0-4b.

The difference between the limiting setpoint assumed for the analysis and the nominal setpoint represents an allowance for instrumentation channel error and setpoint error. Nominal setpoints are specified in the plant Technical Specifications. During plant startup tests, it is demonstrated that actual instrument time delays are equal to or less than the assumed values. Additionally, protection system channels are calibrated and instrument response times are determined periodically in accordance with the plant Technical Specifications.

15.0.7 Instrumentation Drift and Calorimetric Errors, Power Range Neutron Flux

Examples of the instrumentation uncertainties and calorimetric uncertainties used in establishing the power range high neutron flux setpoint are presented in Table 15.0-5.

The calorimetric uncertainty is the uncertainty assumed in the determination of core thermal power as obtained from secondary plant measurements. The total ion chamber current (sum of the top and bottom sections) is calibrated (set equal) to this measured power on a daily basis.

The secondary power is obtained from measurement of feedwater flow, feedwater inlet temperature to the steam generators, and steam pressure. Installed plant instrumentation is used for these measurements.

15.0.8 Plant Systems and Components Available for Mitigation of Accident Effects

The plant is designed to afford proper protection against the possible effects of natural phenomena, postulated environmental conditions, and dynamic effects of the postulated accidents. In addition, the design incorporates features that minimize the probability and effects of fires and explosions.

Chapter 17 discusses the quality assurance program that is implemented to provide confidence that the plant systems satisfactorily perform their assigned safety functions. The incorporation of these features in the plant, coupled with the reliability of the design, provides confidence that the normally operating systems and components listed in Table 15.0-6 are available for mitigation of the events discussed in Chapter 15.

In determining which systems are necessary to mitigate the effects of these postulated events, the classification system of ANSI N18.2-1973 (Reference 1) is used. The design of safety-related systems (including protection systems) is consistent with IEEE Standard 379-2000 and Regulatory Guide 1.53 in the application of the single-failure criterion. Conformance to Regulatory Guide 1.53 is summarized in Subsection 1.9.1.

Table 15.0-8 summarizes the nonsafety-related systems assumed in the analyses to mitigate the consequences of events. Except for the cases listed in Table 15.0-8, control system action is not used for mitigation of accidents.

15.0.9 Fission Product Inventories

The sources of radioactivity for release are dependent on the specific accident. Activity may be released from the primary coolant, from the secondary coolant, and from the reactor core if the accident involves fuel damage. The radiological consequences analyses use the conservative design basis source terms identified in Appendix 15A.

15.0.10 Residual Decay Heat

15.0.10.1 Total Residual Heat

Residual heat in a subcritical core is calculated for the LOCA according to the requirements of 10 CFR 50.46, as described in WCAP-10054-P-A and WCAP-12945-P (References 3 and 4). The large-break LOCA methodology considers uncertainty in the decay power level. The small-break LOCA events and post-LOCA long-term cooling analyses use 10 CFR 50, Appendix K, decay heat, which assumes infinite irradiation time before the core goes subcritical to determine fission product decay energy. For all other accidents, the same models are used, except that fission product decay energy is based on core average exposure at the end of an equilibrium cycle.

15.0.10.2 Distribution of Decay Heat Following a Loss-of-Coolant Accident

During a LOCA, the core is rapidly shut down by void formation, RCCA insertion, or both, and a large fraction of the heat generation considered comes from fission product decay gamma rays. This heat is not distributed in the same manner as steady-state fission power. Local peaking effects, which are important for the neutron-dependent part of the heat generation, do not apply to the gamma ray contribution. The steady-state factor, which represents the fraction of heat generated within the cladding and pellet, drops to 95 percent or less for the hot rod in a LOCA.

For example, consider the transient resulting from the postulated double-ended break of the largest reactor coolant system pipe; one-half second after the rupture, about 30 percent of the heat generated in the fuel rods is from gamma ray absorption. The gamma power shape is less peaked than the steady-state fission power shape, reducing the energy deposited in the hot rod at the expense of adjacent colder rods. A conservative estimate of this effect on the hot rod is a reduction of 10 percent of the gamma ray contribution or 3 percent of the total heat. Because the water density is considerably reduced at this time, an average of 98 percent of the available heat is deposited in the fuel rods; the remaining 2 percent is absorbed by water, thimbles, sleeves, and grids. Combining the 3 percent total heat reduction from gamma redistribution with this 2 percent absorption produce as the net effect a factor of 0.95, which exceeds the actual heat production in the hot rod. The actual hot rod heat generation is computed during the AP1000 large-break LOCA transient as a function of core fluid conditions.

15.0.11 Computer Codes Used

Summaries of some of the principal computer codes used in transient analyses are given as follows. Other codes – in particular, specialized codes in which the modeling has been developed to simulate one given accident, such as those used in the analysis of the reactor coolant system pipe rupture (see Subsection 15.6.5) – are summarized in their respective accident analyses sections. The codes used in the analyses of each transient are listed in Table 15.0-2. WCAP-15644 (Reference 11) provides the basis for use of analysis codes.

15.0.11.1 FACTRAN Computer Code

FACTRAN (Reference 5) calculates the transient temperature distribution in a cross section of a metal-clad UO_2 fuel rod and the transient heat flux at the surface of the cladding using as input the nuclear power and the time-dependent coolant parameters (pressure, flow, temperature, and density). The code uses a fuel model which simultaneously exhibits the following features:

- A sufficiently large number of radial space increments to handle fast transients
- Material properties which are functions of temperature and a sophisticated fuel-to-clad gap heat transfer calculation
- The necessary calculations to handle post-DNB transients: film boiling heat transfer correlations, zircaloy-water reaction, and partial melting of the materials

FACTRAN is further discussed in WCAP-7908-A (Reference 5).

15.0.11.2 LOFTRAN Computer Code

The LOFTRAN (Reference 6) program is used for studies of transient response of a pressurized water reactor system to specified perturbations in process parameters. LOFTRAN simulates a multiloop system by a model containing reactor vessel, hot and cold leg piping, steam generator (tube and shell sides), and pressurizer. The pressurizer heaters, spray, and safety valves are also considered in the program. Point model neutron kinetics, and reactivity effects of the moderator, fuel, boron, and rods are included. The secondary side of the steam generator uses a homogeneous, saturated mixture for the thermal transients and a water level correlation for indication and control. The protection and safety monitoring system is simulated to include reactor trips on high neutron flux, overtemperature ΔT, high and low pressure, low flow, and high pressurizer level. Control systems are also simulated, including rod control, steam dump, feedwater control, and pressurizer level and pressure control. The emergency core cooling system, including the accumulators, is also modeled.

LOFTRAN is a versatile program suited to both accident evaluation and control studies as well as parameter sizing.

LOFTRAN also has the capability of calculating the transient value of DNBR based on the input from the core limits illustrated in Figure 15.0.3-1. The core limits represent the minimum value of DNBR as calculated for typical or thimble cell.

The LOFTRAN code is modified to allow the simulation of the passive residual heat removal (PRHR) heat exchanger, core makeup tanks, and associated protection and safety monitoring system actuation logic. A discussion of these models and additional validation is presented in WCAP-14234 (Reference 10).

LOFTTR2 (Reference 8) is a modified version of LOFTRAN with a more realistic break flow model, a two-region steam generator secondary side, and an improved capability to simulate operator actions during a steam generator tube rupture (SGTR) event.

The LOFTTR2 code is modified to allow the simulation of the PRHR heat exchanger, core makeup tanks, and associated protection system actuation logic. The modifications are identical to those made to the LOFTRAN code. A discussion of these models is presented in WCAP-14234 (Reference 10).

15.0.11.3 TWINKLE Computer Code

The TWINKLE (Reference 7) program is a multidimensional spatial neutron kinetics code, which is patterned after steady-state codes currently used for reactor core design. The code uses an implicit finite-difference method to solve the two-group transient neutron diffusion equations in one, two, and three dimensions. The code uses six delayed neutron groups and contains a detailed multiregion fuel-clad-coolant heat transfer model for calculating pointwise Doppler and moderator feedback effects. The code handles up to 2000 spatial points and performs its own steady-state initialization. Aside from basic cross-section data and thermal-hydraulic parameters, the code accepts as input basic driving functions, such as inlet temperature, pressure, flow, boron concentration, control rod motion, and others. Various edits are provided (for example, channelwise power, axial offset, enthalpy, volumetric surge, point-wise power, and fuel temperatures).

The TWINKLE code is used to predict the kinetic behavior of a reactor for transients that cause a major perturbation in the spatial neutron flux distribution.

15.0.11.4 VIPRE-01 Computer Code

The VIPRE-01 code is described in Subsection 4.4.4.5.2.

15.0.11.5 COAST Computer Program

The COAST computer program is used to calculate the reactor coolant flow coastdown transient for any combination of active and inactive pumps and forward or reverse flow in the hot or cold legs. The program is described in Reference 13 and was referenced in Reference 12. The program was approved in Reference 14.

The equations of conservation of momentum are written for each of the flow paths of the COAST model assuming unsteady one-dimensional flow of an incompressible fluid. The equation of conservation of mass is written for the appropriate nodal points. Pressure losses due to friction, and geometric losses are assumed proportional to the flow velocity squared. Pump dynamics are modeled using a head-flow curve for a pump at full speed and using four-quadrant curves, which are parametric diagrams of pump head and torque on coordinates of speed versus flow, for a pump at other than full speed.

15.0.11.6 ANC Computer Code

The ANC computer code is used to solve the two-group neutron diffusion equation in three spatial dimensions. ANC can also solve the three-dimensional kinetics equations for six delayed neutron groups.

15.0.12 Component Failures

15.0.12.1 Active Failures

SECY-77-439 (Reference 9) provides a description of active failures. An active failure results in the inability of a component to perform its intended function.

An active failure is defined differently for different components. For valves, an active failure is the failure of a component to mechanically complete the movement required to perform its function. This includes the failure of a remotely operated valve to change position on demand. The spurious, unintended movement of the valve is also considered as an active failure. Failure of a manual valve to change position under local operator action is included.

Spring-loaded safety or relief valves that are designed for and operate under single-phase fluid conditions are not considered for active failures to close when pressure is reduced below the valve set point. However, when valves designed for single-phase flow are challenged with two-phase flow, such as a steam generator or pressurizer safety valve, the failure to reseat is considered as an active failure.

For other active equipment – such as pumps, fans, and rotating mechanical components – an active failure is the failure of the component to start or to remain operating.

For electrical equipment, the loss of power, such as the loss of offsite power or the loss of a diesel generator, is considered as a single failure. In addition, the failure to generate an actuation signal, either for a single component actuation or for a system-level actuation, is also considered as an active failure.

Spurious actuation of an active component is considered as an active failure for active components in safety-related passive systems. An exception is made for active components if specific design features or operating restrictions are provided that can preclude such failures (such as power lockout, confirmatory open signals, or continuous position alarms).

A single incorrect or omitted operator action in response to an initiating event is also considered as an active failure; the error is limited to manipulation of safety-related equipment and does not include thought-process errors or similar errors that could potentially lead to common cause or multiple errors.

15.0.12.2 Passive Failures

SECY-77-439 also provides a description of passive failures. A passive failure is the structural failure of a static component that limits the effectiveness of the component in carrying out its design function. A passive failure is applied to fluid systems and consists of a breach in the fluid system boundary. Examples include cracking of pipes, sprung flanges, or valve packing leaks.

Passive failures are not assumed to occur until 24 hours after the start of the event. Consequential effects of a pipe leak – such as flooding, jet impingement, and failure of a valve with a packing leak – must be considered.

Where piping is significantly overdesigned or installed in a system where the pressure and temperature conditions are relatively low, passive leakage is not considered a credible failure mechanism. Line blockage is also not considered as a passive failure mechanism.

15.0.12.3 Limiting Single Failures

The most limiting single active failure (where one exists), as described in Section 3.1, of safety-related equipment, is identified in each analysis description. The consequences of this failure are described therein. In some instances, because of redundancy in protection equipment, no single failure that could adversely affect the consequences of the transient is identified. The failure assumed in each analysis is listed in Table 15.0-7.

15.0.13 Operator Actions

For events where the PRHR heat exchanger is actuated, the plant automatically cools down to a safe, stable shutdown condition. Where a stabilized condition is reached automatically following a reactor trip, it is expected that the operator may, following event recognition, take manual control and proceed with orderly shutdown of the reactor in accordance with the normal, abnormal, or emergency operating procedures. The exact actions taken and the time at which these actions occur depend on what systems are available and the plans for further plant operation.

However, for these events, operator actions are not required to maintain the plant in a safe and stable condition for at least 72 hours. Operator actions typical of normal operation are credited for the inadvertent actuations of equipment in response to a Condition II event.

15.0.14 Loss of Offsite ac Power

As required in GDC 17 of 10 CFR Part 50, Appendix A, anticipated operational occurrences and postulated accidents are analyzed assuming a loss of offsite ac power. The loss of offsite power is not considered as a single failure, and the analysis is performed without changing the event category. In the analyses, the loss of offsite ac power is considered to be a potential consequence of the event.

A loss of offsite ac power will be considered a consequence of an event due to disruption of the grid following a turbine trip during the event. Event analyses that do not result in a possible consequential disruption of offsite ac power do not assume offsite power is lost.

For those events where offsite ac power is lost, an appropriate time delay between turbine trip and the postulated loss of offsite ac power is assumed in the analyses. A time delay of 3 seconds is used. This time delay is based on the inherent stability of the offsite power grid as discussed in Section 8.2. Following the time delay, the effect of the loss of offsite ac power on plant auxiliary equipment – such as reactor coolant pumps, main feedwater pumps, condenser, startup feedwater pumps, and RCCAs – is considered in the analyses. Turbine trip occurs 5 seconds following a reactor trip condition being reached. This delay is part of the AP1000 reactor trip system.

Design basis LOCA analyses are governed by the GDC-17 requirement to consider the loss of offsite power. For the AP1000 design, in which all the safety-related systems are passive, the availability of offsite power is significant only regarding reactor coolant pump operation for LOCA events. A sensitivity study for AP1000 has shown that for large-break LOCAs, assuming the loss of offsite power coincident with the inception of the LOCA event is nonlimiting relative to assuming continued reactor coolant pump operation until the automatic reactor coolant pump trip occurs following an “S” signal less than 10 seconds into the transient. For small-break LOCA events, the AP1000 automatic reactor coolant pump trip feature prevents continued operation of the reactor coolant pumps from mixing the liquid and vapor present within a two-phase reactor coolant system inventory to increase the liquid break flow and deplete the reactor coolant system mass inventory rapidly. The automatic reactor coolant pump trip occurs early enough during AP1000 small-break LOCA transients that emergency core cooling system performance is not affected by the loss of offsite power assumption because the total break flow is approximately equivalent for reactor coolant pump trip occurring either at time zero or as a result of the “S” signal. Whether a loss of offsite power is postulated at the inception of the LOCA event or occurs automatically later on is unimportant in the Subsection 15.6.5.4C long-term cooling analyses because with either assumption, the reactor coolant pumps are tripped long before the long-term cooling timeframe.

The AP1000 protection and safety monitoring system and passive safeguards systems are not dependent on offsite power or on any backup diesel generators. Following a loss of ac power, the protection and safety monitoring system and passive safeguards are able to perform the safety functions and there are no additional time delays for these functions to be completed.

15.0.15 Combined License Information

15.0.15.1 Following selection of the actual plant operating instrumentation, calculation of the primary power calorimetric uncertainty is addressed in Subsection 15.0.3.2.

15.0.16 References

1. American National Standards Institute N18.2, “Nuclear Safety Criteria for the Design of Stationary PWR Plants,” 1973.

2. Friedland, A. J., and Ray, S., “Revised Thermal Design Procedure,” WCAP-11397-P-A (Proprietary) and WCAP-11397-A (Non-Proprietary), April 1989.

3. Lee, N., “Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code,” WCAP-10054-P-A (Proprietary) and WCAP-10081 (Non-Proprietary), August 1985.

4. Bajorek, S. M., et al., “Code Qualification Document for Best-Estimate LOCA Analysis,” WCAP-12945-P-A, Volume 1, Revision 2, and Volumes 2 through 5, Revision 1, (Proprietary) and WCAP-14747 (Non-Proprietary), 1998.

5. Hargrove, H. G., “FACTRAN - A FORTRAN-IV Code for Thermal Transients in a UO_2 Fuel Rod,” WCAP-7908-A, December 1989.

6. Burnett, T. W. T., “LOFTRAN Code Description,” WCAP-7907-P-A (Proprietary) and WCAP-7907-A (Non-Proprietary), April 1984.

7. Risher, D. H., Jr., and Barry, R. F., “TWINKLE - A Multi-Dimensional Neutron Kinetics Computer Code,” WCAP-7979-P-A (Proprietary) and WCAP-8028-A (Non-Proprietary), January 1975.

8. Lewis, R. N., “SGTR Analysis Methodology to Determine the Margin to Steam Generator Overfill,” WCAP-10698-P-A (Proprietary) and WCAP-10750-A (Non-Proprietary), August 1987.

9. Case, E. G., “Single Failure Criterion,” SECY-77-439, August 17, 1977.

10. Bachrach, U., Carlin, E. L., “LOFTRAN and LOFTTR2 AP600 Code Applicability Document,” WCAP-14234, Revision 1 (Proprietary) and WCAP-14235, Revision 1 (Non-Proprietary), August 1997.

11. "AP1000 Code Applicability Report," WCAP-15644-P (Proprietary) and WCAP-15644-NP (Non-Proprietary), Revision 2, March 2004.

12. “Combustion Engineering Standard Safety Analysis Report,” CESSAR Docket No. STN-50-470, December 1975.

13. “COAST Code Description,” CENPD-98-A, April 1973, Proprietary Information.

14. CENPD-98-A, “COAST Code Description,” April 1973 (NRC Approval Letter dated December 4, 1974).

201. Final Safety Evaluation for Cameron Measurement Systems Engineering Report ER-157P, Revision 8, “Caldon Ultrasonics Engineering Report ER-157P, ‘Supplement to Topical Report ER-80P: Basis for a Power Uprate with the LEFM Check or Checkplus™ System’,” (TAC No. ME1321). August 16, 2010. ADAMS Accession No. ML102160694.

202. Final Safety Evaluation for Beaver Valley Power Station, Unit Nos. 1 and 2 (BVPS-1 and 2) – Issuance of Amendment re: 1.4-Percent Power Uprate and Revised BVPS-2 Heatup and Cooldown Curves. September 24, 2001, ADAMS Accession No. ML012490569.

Table 15.0-1
Nuclear Steam Supply System Power Ratings

Thermal power output (MWt)	3415
Effective thermal power generated by the reactor coolant pumps (MWt)	15
Core thermal power (MWt)	3400

Table 15.0-2 (Sheet 1 of 5)
Summary of Initial Conditions and Computer Codes Used

Section	Faults	Computer Codes Used	Reactivity Coefficients Assumed			Initial Thermal Power Output Assumed (MWt)
			Moderator Density ($\Delta k/gm/cm^3$)	Moderator Temperature (pcm/°F)	Doppler	
15.1	Increase in heat removal from the primary system					
	Feedwater system malfunctions causing a reduction in feedwater temperature	Bounded by excessive increase in secondary steam flow	–	–	–	–
	Feedwater system malfunctions that result in an increase in feedwater flow	LOFTRAN, FACTRAN, VIPRE-01	0.470	–	Upper curve of Figure 15.0.4-1	0 and 3415
	Excessive increase in secondary steam flow	LOFTRAN, FACTRAN, VIPRE-01	0.0 and 0.470	–	Upper and lower curves of Figure 15.0.4-1	3415
	Inadvertent opening of a steam generator relief or safety valve	LOFTRAN, VIPRE-01	Function of moderator density (see Figure 15.1.4-1)	–	See Subsection 15.1.4.	0 (subcritical)
	Steam system piping failure	LOFTRAN, VIPRE-01	Function of moderator density (see Figure 15.1.4-1)	–	See Subsection 15.1.5	0 (subcritical)
	Inadvertent operation of the PRHR heat exchanger	N/A	N/A	–	N/A	3415

Table 15.0-2 (Sheet 2 of 5)
Summary of Initial Conditions and Computer Codes Used

Section	Faults	Computer Codes Used	Reactivity Coefficients Assumed			Initial Thermal Power Output Assumed (MWt)
			Moderator Density (Δk/gm/cm^3)	Moderator Temperature (pcm/°F)	Doppler	
15.2	Decrease in heat removal by the secondary system					
	Loss of external electrical load and/or turbine trip	LOFTRAN, FACTRAN, VIPRE-01	0.0 and 0.470	–	Lower and upper curves of Figure 15.0.4-1	3415 and 3483.3 (a)
	Inadvertent closure of main steam isolation valves	Bounded by turbine trip event	–	–	–	–
	Loss of condenser vacuum and other events resulting in turbine trip	Bounded by turbine trip event	–	–	–	–
	Loss of nonemergency ac power to the plant auxiliaries	LOFTRAN	0.0	–	Lower curve of Figure 15.0.4-1	3483.3 (a)
	Loss of normal feedwater flow	LOFTRAN	0.0	–	Lower curve of Figure 15.0.4-1	3483.3 (a)
	Feedwater system pipe break	LOFTRAN	0.0	–	Lower curve of Figure 15.0.4-1	3483.3 (a)
15.3	Decrease in reactor coolant system flow rate					

Table 15.0-2 (Sheet 3 of 5)
Summary of Initial Conditions and Computer Codes Used

Section	Faults	Computer Codes Used	Reactivity Coefficients Assumed: Moderator Density ($\Delta k/gm/cm^3$)	Reactivity Coefficients Assumed: Moderator Temperature (pcm/°F)	Reactivity Coefficients Assumed: Doppler	Initial Thermal Power Output Assumed (MWt)
15.3	Partial and complete loss of forced reactor coolant flow	LOFTRAN, FACTRAN, COAST, VIPRE-01	0.0	–	Lower curve of Figure 15.0.4-1	3415
	Reactor coolant pump shaft seizure (locked rotor) and reactor coolant pump shaft break	LOFTRAN, FACTRAN, COAST, VIPRE-01	0.0	–	Lower curve of Figure 15.0.4-1	3483.3 (a)
15.4	Reactivity and power distribution anomalies					
	Uncontrolled RCCA bank withdrawal from a subcritical or low power startup condition	TWINKLE, FACTRAN, VIPRE-01	–	0.0	Coefficient is consistent with a Doppler defect of -0.67%Δk	0
	Uncontrolled RCCA bank withdrawal at power	LOFTRAN, FACTRAN, VIPRE-01	0.0 and 0.470	–	Upper and lower curves of Figure 15.0.4-1	10%, 60%, and 100% of 3415
	RCCA misalignment	LOFTRAN, VIPRE-01	NA	–	NA	3415
	Startup of an inactive reactor coolant pump at an incorrect temperature	NA	NA	–	NA	NA

Table 15.0-2 (Sheet 4 of 5)
Summary of Initial Conditions and Computer Codes Used

Section	Faults	Computer Codes Used	Reactivity Coefficients Assumed			Initial Thermal Power Output Assumed (MWt)
			Moderator Density ($\Delta k/gm/cm^3$)	Moderator Temperature (pcm/°F)	Doppler	
15.4	Chemical and volume control system malfunction that results in a decrease in the boron concentration in the reactor coolant	NA	NA	–	NA	0 and 3415
	Inadvertent loading and operation of a fuel assembly in an improper position	ANC	NA	–	NA	3415
	Spectrum of RCCA ejection accidents	ANC, VIPRE	Refer to Subsection 15.4.8	Refer to Subsection 15.4.8	Refer to Subsection 15.4.8	Refer to Subsection 15.4.8
15.5	Increase in reactor coolant inventory					
	Inadvertent operation of the emergency core cooling system during power operation	LOFTRAN	0.0	–	Upper curve of Figure 15.0.4-1	3483.3 (a)
	Chemical and volume control system malfunction that increases reactor coolant inventory	LOFTRAN	0.0	–	Upper curve of Figure 15.0.4-1	3483.3 (a)

Table 15.0-2 (Sheet 5 of 5)
Summary of Initial Conditions and Computer Codes Used

Section	Faults	Computer Codes Used	Reactivity Coefficients Assumed: Moderator Density (Δk/gm/cm^3)	Reactivity Coefficients Assumed: Moderator Temperature (pcm/°F)	Reactivity Coefficients Assumed: Doppler	Initial Thermal Power Output Assumed (MWt)
15.6	Decrease in reactor coolant inventory					
	Inadvertent opening of a pressurizer safety valve and inadvertent operation of ADS	LOFTRAN, FACTRAN, VIPRE-01	0.0	–	Lower curve of Figure 15.0.4-1	3415
	Steam generator tube failure	LOFTTR2	0.0	–	Lower curve of Figure 15.0.4-1	3483.3 (a)
	A break in an instrument line or other lines from the reactor coolant pressure boundary that penetrate containment	NA	NA	–	NA	NA
	LOCAs resulting from the spectrum of postulated piping breaks within the reactor coolant pressure boundary	NOTRUMP WCOBRA/TRAC HOTSPOT	See Subsection 15.6.5 references	–	See Subsection 15.6.5 references	3468.0 (SBLOCA) 3434.0 (LBLOCA)

Notes:

a. 102% of rated thermal power – The main feedwater flow measurement supports a 1-percent power uncertainty; use of a 2-percent power uncertainty is conservative.

b. BOC – Beginning of core cycle
EOC – End of core cycle

Table 15.0-3
Nominal Values of Pertinent Plant Parameters Used in Accident Analyses

	RTDP With 10% Steam Generator Tube Plugging	Without RTDP[(a)]	
		Without Steam Generator Tube Plugging	With 10% Steam Generator Tube Plugging
Thermal output of NSSS (MWt)	3415	3415	3415
Core inlet temperature (°F)	535.8	535.5	535.0
Vessel average temperature (°F)	573.6	573.6	573.6
Reactor coolant system pressure (psia)	2250.0	2250.0	2250.0
Reactor coolant flow per loop (gpm)	15.08 E+04	14.99 E+04	14.8 E+04
Steam flow from NSSS (lbm/hr)	14.96 E+06	14.96 E+06	14.95 E+06
Steam pressure at steam generator outlet (psia)	802.2	814.0	796.0
Assumed feedwater temperature at steam generator inlet (°F)	440.0	440.0	440.0
Average core heat flux (Btu/-hr-ft^2)	1.99 E+05	1.99 E+05	1.99 E+05

Note:

a. Steady-state errors discussed in Subsection 15.0.3 are added to these values to obtain initial conditions for most transient analyses.

Table 15.0-4a (Sheet 1 of 2)
Protection and Safety Monitoring System
Setpoints and Time Delay Assumed in Accident Analyses

Function	Limiting Setpoint Assumed in Analyses	Time Delays (seconds)
Reactor trip on power range high neutron flux, high setting	118%	0.9
Reactor trip on power range high neutron flux, low setting	35%	0.9
Reactor trip on source range neutron flux reactor trip	Not applicable	0.9
Overtemperature ΔT	Variable (see Figure 15.0.3-1)	2.0
Overpower ΔT	Variable (see Figure 15.0.3-1)	2.0
Reactor trip on high pressurizer pressure	2460 psia	2.0
Reactor trip on low pressurizer pressure	1800 psia	2.0
Reactor trip on low reactor coolant flow in either hot leg	87% loop flow	1.45
Reactor trip on reactor coolant pump under speed	90%	0.767
Reactor trip on low steam generator narrow range level	95,000 lbm	2.0
High steam generator narrow range level coincident with reactor trip (P-4)	85% of narrow range level span	2.0 (startup feedwater isolation) 2.0 (chemical and volume control system makeup isolation)
High-2 steam generator level	95% of narrow range level span	2.0 (reactor trip) 0.0 (turbine trip) 2.0 (feedwater isolation)
Reactor trip on high-3 pressurizer water level	76% of span	2.0
PRHR actuation on low steam generator wide range level	55,000 lbm	2.0
"S" signal and steam line isolation on low T_{cold}	500°F	2.0

Table 15.0-4a (Sheet 2 of 2)
Protection and Safety Monitoring System
Setpoints and Time Delay Assumed in Accident Analyses

Function	Limiting Setpoint Assumed in Analyses	Time Delays (seconds)
“S” signal and steam line isolation on low steam line pressure	405 psia (with an adverse environment assumed) 535 psia (without an adverse environment assumed)	2.0
“S” signal on low pressurizer pressure	1700 psia	2.0
Reactor trip on PRHR discharge valves not closed	Valve not closed	1.25
“S” signal on high-2 containment pressure	8 psig	2.0 2.2 (LBLOCA)
Reactor coolant pump trip following “S”	–	15.0 4.0 (LBLOCA)
PRHR actuation of high-3 pressurizer water level	76% of span	2.0 (plus 15.0-second timer delay)
Chemical and volume control system isolation on high-2 pressurizer water level	63% of span	2.0
Chemical and volume control system isolation on high-1 pressurizer water level coincident with “S” signal	28% of span	2.0
Boron dilution block on source range flux doubling	3 over 50 minutes	80.0
ADS Stage 1 actuation on core makeup tank low level signal(1)	67.5% of tank volume	32.0 seconds for control valve to begin to open
ADS Stage 4 actuation on core makeup tank low-low level signal(1)	20% of tank volume	2.0 seconds for squib valve to begin to open
CMT actuation on pressurizer low-2 water level	0% of span	2.0

Note:
1. The delay times reflect the design basis of the AP1000. The applicable Chapter 15 accidents were evaluated for the design basis delay times. The results of this evaluation have shown that there is a small impact on the analysis and the conclusions remain valid. The output provided for the analyses is representative of the transient phenomenon.

Table 15.0-4b
Limiting Delay Times for
Equipment Assumed in Accident Analyses

Component	Time Delays (seconds)
Feedwater isolation valve closure, feedwater control valve closure, or feedwater pump trip	10 (maximum value for non-LOCA) 5 (maximum value for mass/energy)
Steam line isolation valve closure	5
Core makeup tank discharge valve opening time	15 (maximum) 10 (nominal value for best-estimate LOCA)
Chemical and volume control system isolation valve closure(1)	30
PRHR discharge valve opening time	15 (maximum) 10 (nominal value for best-estimate LOCA) 1.0 second (small-break LOCA value: follows a 15-second interval of no valve movement)
Demineralized water transfer and storage system isolation valve closure time	20
Steam generator power-operated relief valve block valve closure	44
Automatic depressurization system (ADS) valve opening times(1)	See Table 15.6.5-10.

Note:

1. The valve stroke times reflect the design basis of the AP1000. The applicable Chapter 15 accidents were evaluated for the design basis valve stroke times. The results of this evaluation have shown that there is a small impact on the analysis and the conclusions remain valid. The output provided for the analyses is representative of the transient phenomenon.

Table 15.0-5
Determination of Maximum Power Range
Neutron Flux Channel Trip Setpoint, Based on Nominal Setpoint
and Inherent Typical Instrumentation Uncertainties

Nominal setpoint (% of rated power)		109
Calorimetric errors in the measurement of secondary system thermal power:		
Variable	**Accuracy of Measurement of Variable**	**Effect on Thermal Power Determination (% of Rated Power)**
Feedwater temperature	±3°F	
Steam pressure (small correction on enthalpy)	±6 psi	
Feedwater flow	±0.5% ΔP instrument span (two channels per steam generator)	
Assumed calorimetric error		2.0 (a)* The main feedwater flow measurement supports a 1% power uncertainty; use of a 2% power uncertainty is conservative.
Radial power distribution effects on total ion chamber current		7.8 (b)*
Allowed mismatch between power range neutron flux channel and calorimetric measurement		2.0 (c)*
Instrumentation channel drift and setpoint reproducibility	0.4% of instrument span (120% power span)	0.84(d)*
Instrumentation channel temperature effects		0.48(e)*
*Total assumed error in setpoint (% of rated power): $[(a)^2 + (b)^2 + (c)^2 + (d)^2 + (e)^2]^{1/2}$		±8.4
Maximum power range neutron flux trip setpoint assuming a statistical combination of individual uncertainties (% of rated power)		118

Table 15.0-6 (Sheet 1 of 5)
Plant Systems and Equipment
Available for Transient and Accident Conditions

Incident	Reactor Trip Functions	ESF Actuation Functions	ESF and Other Equipment
Section 15.1			
Increase in heat removal from the primary system			
Feedwater system malfunctions that result in an increase in feedwater flow	High-2 Steam Generator Level, Power range high flux, overtemperature	High-2 steam generator level produced feedwater isolation and turbine trip	Feedwater isolation valves
Excessive increase in secondary steam flow	Power range high flux, overtemperature ΔT, overpower ΔT, manual	–	–
Inadvertent opening of a steam generator safety valve	Power range high flux, overtemperature ΔT, overpower ΔT, Low pressurizer pressure, “S”, manual	Low pressurizer pressure, low compensated steam line pressure, low T_{cold}, low-2 pressurizer level	Core makeup tank, feedwater isolation valves, main steam isolation valves (MSIVs), startup feedwater isolation, accumulators
Steam system piping failure	Power range high flux, overtemperature ΔT, overpower ΔT, Low pressurizer pressure, “S”, manual	Low pressurizer pressure, low compensated steam line pressure, high-2 containment pressure, low T_{cold}, manual	Core makeup tank, feedwater isolation valves, main steam line isolation valves (MSIVs), accumulators, startup feedwater isolation
Inadvertent operation of the PRHR	PRHR discharge valve position	Low pressurizer pressure, low T_{cold}, low-2 pressurizer level	Core makeup tank

Table 15.0-6 (Sheet 2 of 5)
Plant Systems and Equipment
Available for Transient and Accident Conditions

Incident	Reactor Trip Functions	ESF Actuation Functions	ESF and Other Equipment
Section 15.2			
Decrease in heat removal by the secondary system			
Loss of external load/turbine trip	High pressurizer pressure, high pressurizer water level, overtemperature ΔT, overpower ΔT, Steam generator low narrow range level, low RCP speed, manual	–	Pressurizer safety valves, steam generator safety valves
Loss of nonemergency ac power to the station auxiliaries	Steam generator low narrow range level, high pressurizer pressure, high pressurizer level, manual	Steam generator low narrow range level coincident with low startup water flow, steam generator low wide range level	PRHR, steam generator safety valves, pressurizer safety valves
Loss of normal feedwater flow	Steam generator low narrow range level, high pressurizer pressure, high pressurizer level, manual	Steam generator low narrow range level coincident with low startup water flow, steam generator low wide range level	PRHR, steam generator safety valves, pressurizer safety valves
Feedwater system pipe break	Steam generator low narrow range level, high pressurizer pressure, high pressurizer level, manual	Steam generator low narrow range level coincident with low startup feedwater flow, Steam generator low wide range level, low steam line pressure, high-2 containment pressure	PRHR, core makeup tank, MSIVs, feedline isolation, pressurizer safety valves, steam generator safety valves

Table 15.0-6 (Sheet 3 of 5)
Plant Systems and Equipment
Available for Transient and Accident Conditions

Incident	Reactor Trip Functions	ESF Actuation Functions	ESF and Other Equipment
Section 15.3			
Decrease in reactor coolant system flow rate			
Partial and complete loss of forced reactor coolant flow	Low flow, underspeed, manual	–	Steam generator safety valves, pressurizer safety valves
Reactor coolant pump shaft seizure (locked rotor)	Low flow, high pressurizer pressure, manual	–	Pressurizer safety valves, steam generator safety valves
Section 15.4			
Reactivity and power distribution anomalies			
Uncontrolled RCCA bank withdrawal from a subcritical or low power startup condition	Source range high neutron flux, intermediate range high neutron flux, power range high neutron flux (low setting), power range high neutron flux (high setting), high nuclear flux rate, manual	–	–
Uncontrolled RCCA bank withdrawal at power	Power range high neutron flux, high power range positive neutron flux rate, overtemperature ΔT, over-power ΔT, high pressurizer pressure, high pressurizer water level, manual	–	Pressurizer safety valves, steam generator safety valves
RCCA misalignment	Overtemperature ΔT, manual	–	–
Startup of an inactive reactor coolant pump at an incorrect temperature	Power range high flux, low flow (P-10 interlock), manual	–	–

Table 15.0-6 (Sheet 4 of 5)
Plant Systems and Equipment
Available for Transient and Accident Conditions

Incident	Reactor Trip Functions	ESF Actuation Functions	ESF and Other Equipment
Section 15.4 (Cont.)			
Chemical and volume control system malfunction that results in a decrease in boron concentration in the reactor coolant	Source range high flux, power range high flux, overtemperature ΔT, manual	Source range flux doubling	CVS to RCS isolation valves, makeup pump suction isolation valves, from the demineralized water transfer and storage system
Spectrum of RCCA ejection accidents	Power range high flux, high positive flux rate, manual	–	Pressurizer safety valves
Section 15.5			
Increase in reactor coolant inventory			
Inadvertent operation of the CMT during power operation	High pressurizer pressure, manual, “safeguards” trip, high pressurizer level	High pressurizer level, low T_{cold}	Core makeup tank, pressurizer safety valves, chemical and volume control system isolation, PRHR, steam generator safety valves
Chemical and volume control system malfunction that increases reactor coolant inventory	High pressurizer pressure, “safeguards” trip, high pressurizer level, manual	High pressurizer level, low T_{cold}, low steam line pressure	Core makeup tank, pressurizer safety valves, chemical and volume control system isolation, PRHR
Section 15.6			
Decrease in reactor coolant inventory			
Inadvertent opening of a pressurizer safety valve or ADS path	Low pressurizer pressure, overtemperature ΔT, manual	Low pressurizer pressure	Core makeup tank, ADS, accumulator
Failure of small lines carrying primary coolant outside containment	–	Manual isolation of the Sample System or CVS discharge lines	Sample System isolation valves, Chemical and volume control system discharge line isolation valves

Table 15.0-6 (Sheet 5 of 5)
Plant Systems and Equipment
Available for Transient and Accident Conditions

Incident	Reactor Trip Functions	ESF Actuation Functions	ESF and Other Equipment
Section 15.6 (Cont.)			
Steam generator tube rupture	Low pressurizer pressure, overtemperature ΔT, safeguards (“S”), manual	Low pressurizer pressure, high-2 steam generator water level, high steam generator level coincident with reactor trip (P-4), low steam line pressure, low pressurizer level	Core makeup tank, PRHR, steam generator safety and/or relief valves, MSIVs, radiation monitors (air removal, steam line, and steam generator blowdown), startup feedwater isolation, chemical and volume control system pump isolation, pressurizer heater isolation, steam generator power-operated relief valve isolation
LOCAs resulting from the spectrum of postulated piping breaks within the reactor coolant pressure boundary	Low pressurizer pressure, safeguards (“S”), manual	High-2 containment pressure, low pressurizer pressure	Core makeup tank, accumulator, ADS, steam generator safety and/or relief valves, PRHR, in-containment water storage tank (IRWST)

Table 15.0-7 (Sheet 1 of 2)
Single Failures Assumed in Accident Analyses

Event Description	Failure
Feedwater temperature reduction[(a)]	–
Excessive feedwater flow	One protection division
Excessive steam flow	One protection division
Inadvertent secondary depressurization	One core makeup tank discharge valve
Steam system piping failure	One core makeup tank discharge valve
Inadvertent operation of the PRHR	One protection division
Steam pressure regulator malfunction[(b)]	–
Loss of external load	One protection division
Turbine trip	One protection division
Inadvertent closure of main steam isolation valve	One protection division
Loss of condenser vacuum	One protection division
Loss of ac power	One PRHR discharge valve
Loss of normal feedwater	One PRHR discharge valve
Feedwater system pipe break	One PRHR discharge valve
Partial loss of forced reactor coolant flow	One protection division
Complete loss of forced reactor coolant flow	One protection division
Reactor coolant pump locked rotor	One protection division
Reactor coolant pump shaft break	One protection division
RCCA bank withdrawal from subcritical	One protection division
RCCA bank withdrawal at power	One protection division
Dropped RCCA, dropped RCCA bank	One protection division
Statically misaligned RCCA[(c)]	–
Single RCCA withdrawal	One protection division

Notes:
a. No protection action required
b. Not applicable to AP1000
c. No transient analysis

Table 15.0-7 (Sheet 2 of 2)
Single Failures Assumed in Accident Analyses

Event Description	Failure
Flow controller malfunction[b]	–
Uncontrolled boron dilution	One protection division
Improper fuel loading[c]	–
RCCA ejection	One protection division
Inadvertent CMT operation at power	One PRHR discharge valve
Increase in reactor coolant system inventory	One PRHR discharge valve
Inadvertent reactor coolant system depressurization	One protection division
Failure of small lines carrying primary coolant outside containment[c]	–
Steam generator tube rupture	Faulted steam generator power-operated relief valve fails open
Spectrum of LOCA Small breaks Large breaks	 One ADS Stage 4 valve One CMT valve
Long-term cooling	One ADS Stage 4 valve

Notes:
a. No protection action required
b. Not applicable to AP1000
c. No transient analysis

Table 15.0-8
Nonsafety-Related System and Equipment Used for Mitigation of Accidents

Event		Nonsafety-related System and Equipment
15.1.5	Feedwater system malfunctions that result in an increase in feedwater flow	Main feedwater pump trip
15.1.4	Inadvertent opening of a steam generator relief or safety valve	MSIV backup valves[1] Main steam branch isolation valves
15.1.5	Steam system piping failure	MSIV backup valves[1] Main steam branch isolation valves
15.2.7	Loss of normal feedwater	Pressurizer heater block
15.5.1	Inadvertent operation of the core makeup tanks during power operation	Pressurizer heater block
15.5.2	Chemical and volume control system malfunction that increases reactor coolant inventory	Pressurizer heater block
15.6.2	Failure of small lines carrying primary coolant outside containment	Sample line isolation valves
15.6.3	Steam generator tube rupture	Pressurizer heater block MSIV backup valves(1) Main steam branch isolation valves
15.6.5	Small-break LOCA	Pressurizer heater block

Note:
1. These include the turbine stop or control valves, the turbine bypass valves, and the moisture separator reheater 2nd stage steam isolation valves.

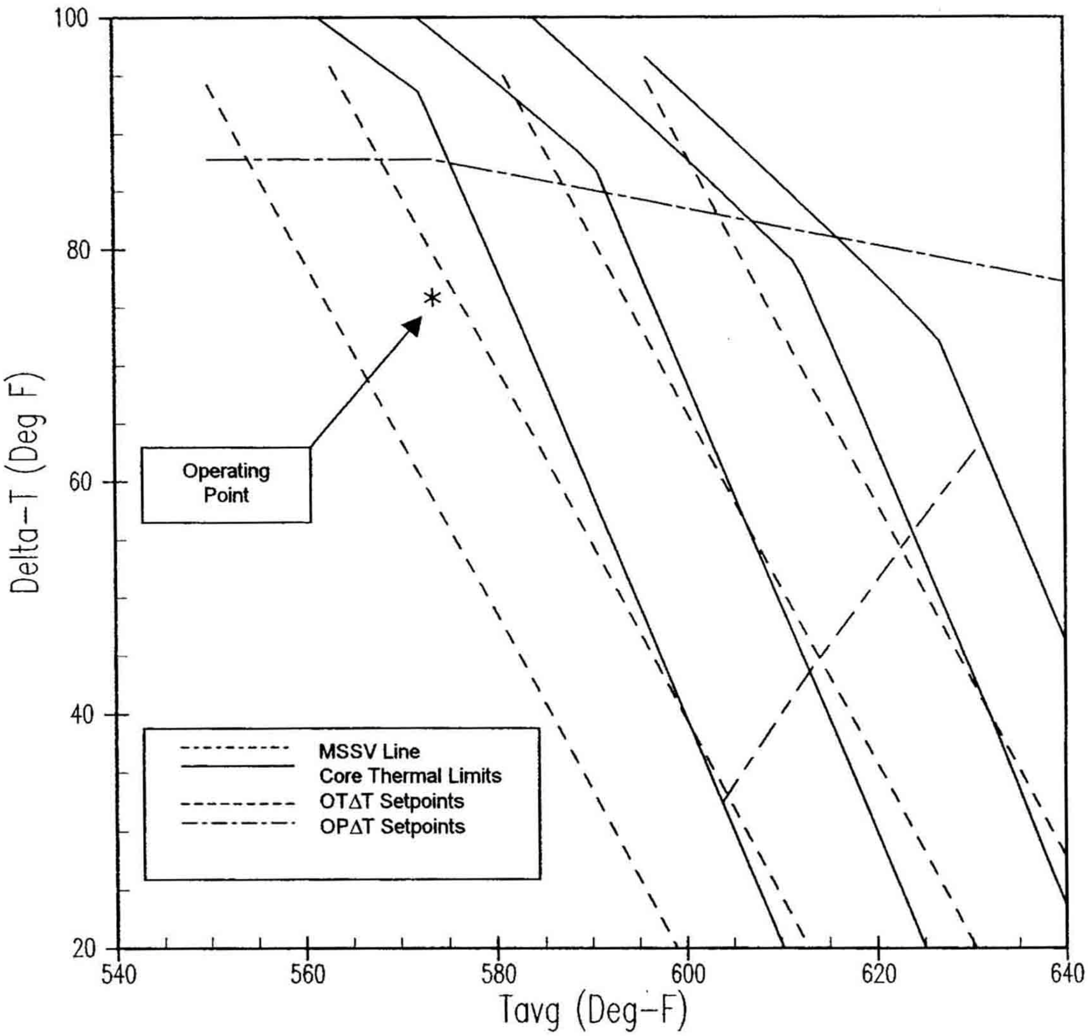
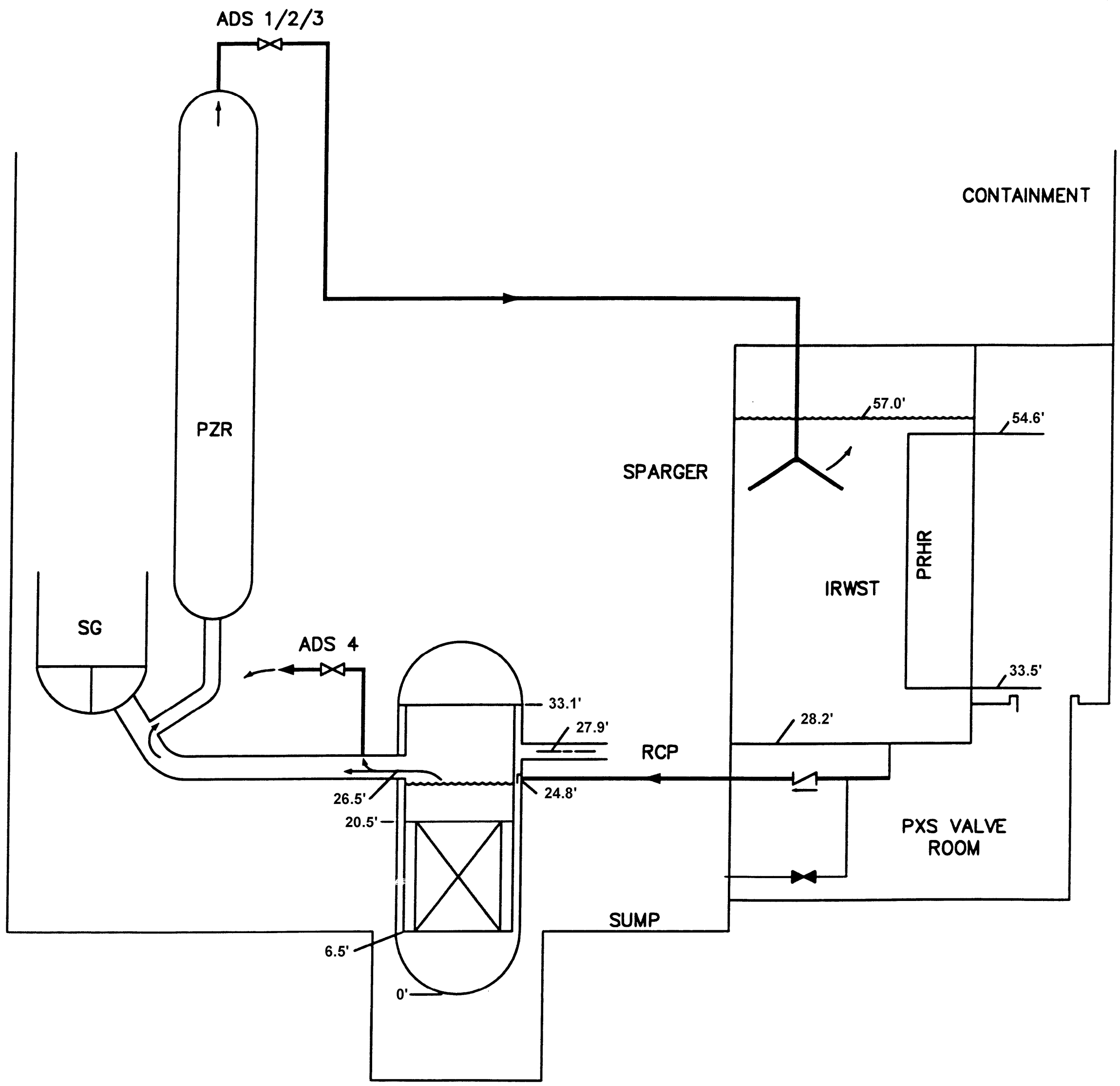


Figure 15.0.3-1
Overpower and Overtemperature ΔT Protection



Note: All elevations are relative to the bottom inside surface of the reactor vessel

Figure 15.0.3-2
AP1000 Loop Layout

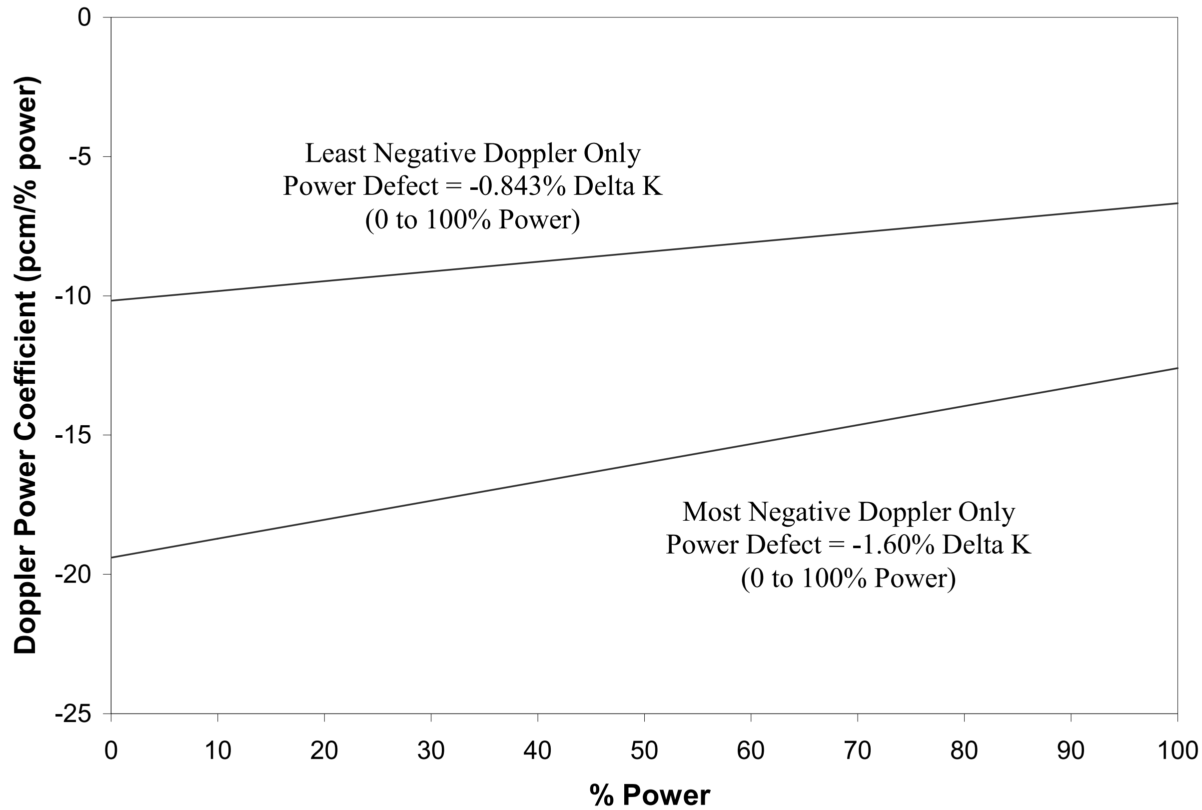


Figure 15.0.4-1
Doppler Power Coefficient used in Accident Analysis

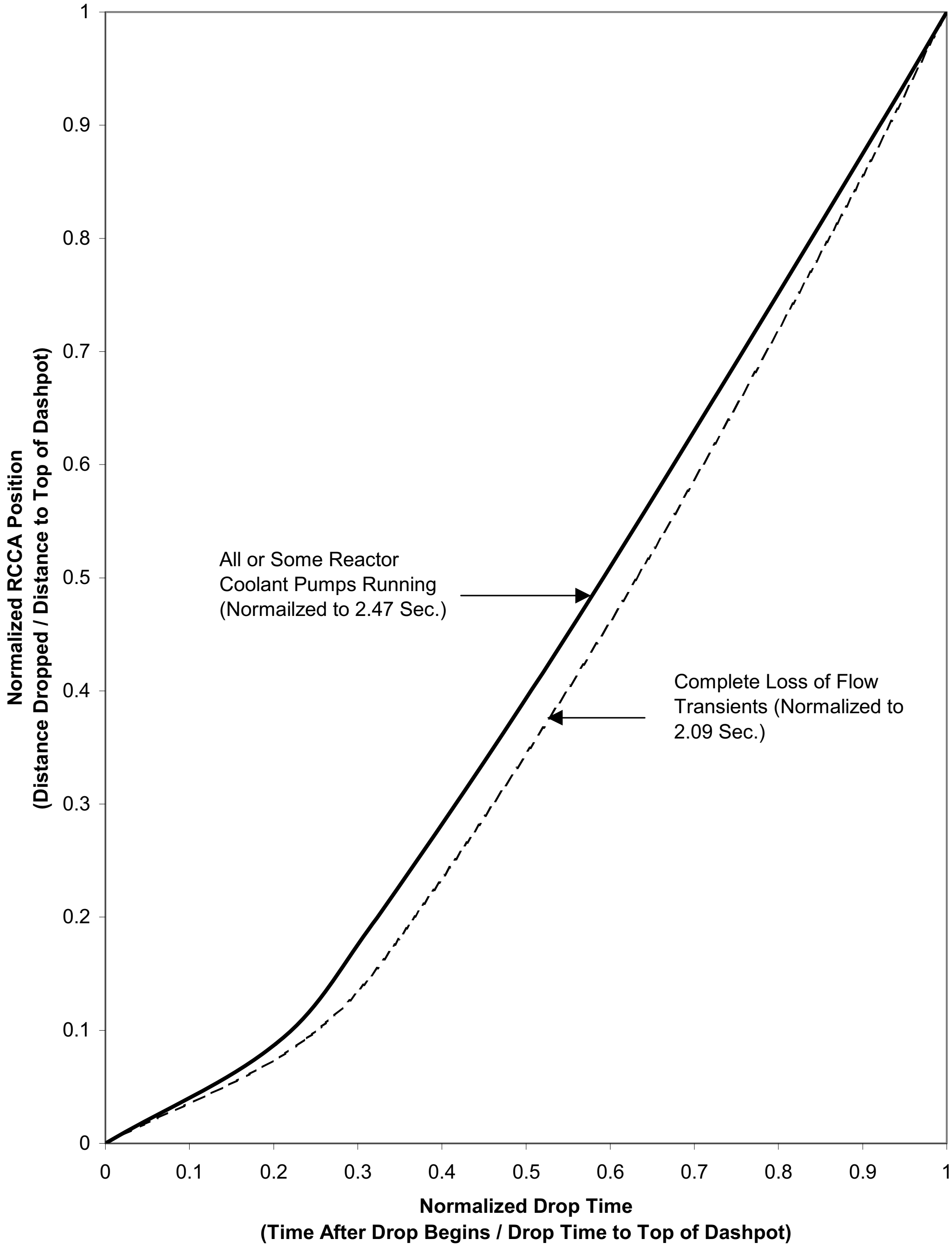


Figure 15.0.5-1
RCCA Position Versus Time to Dashpot

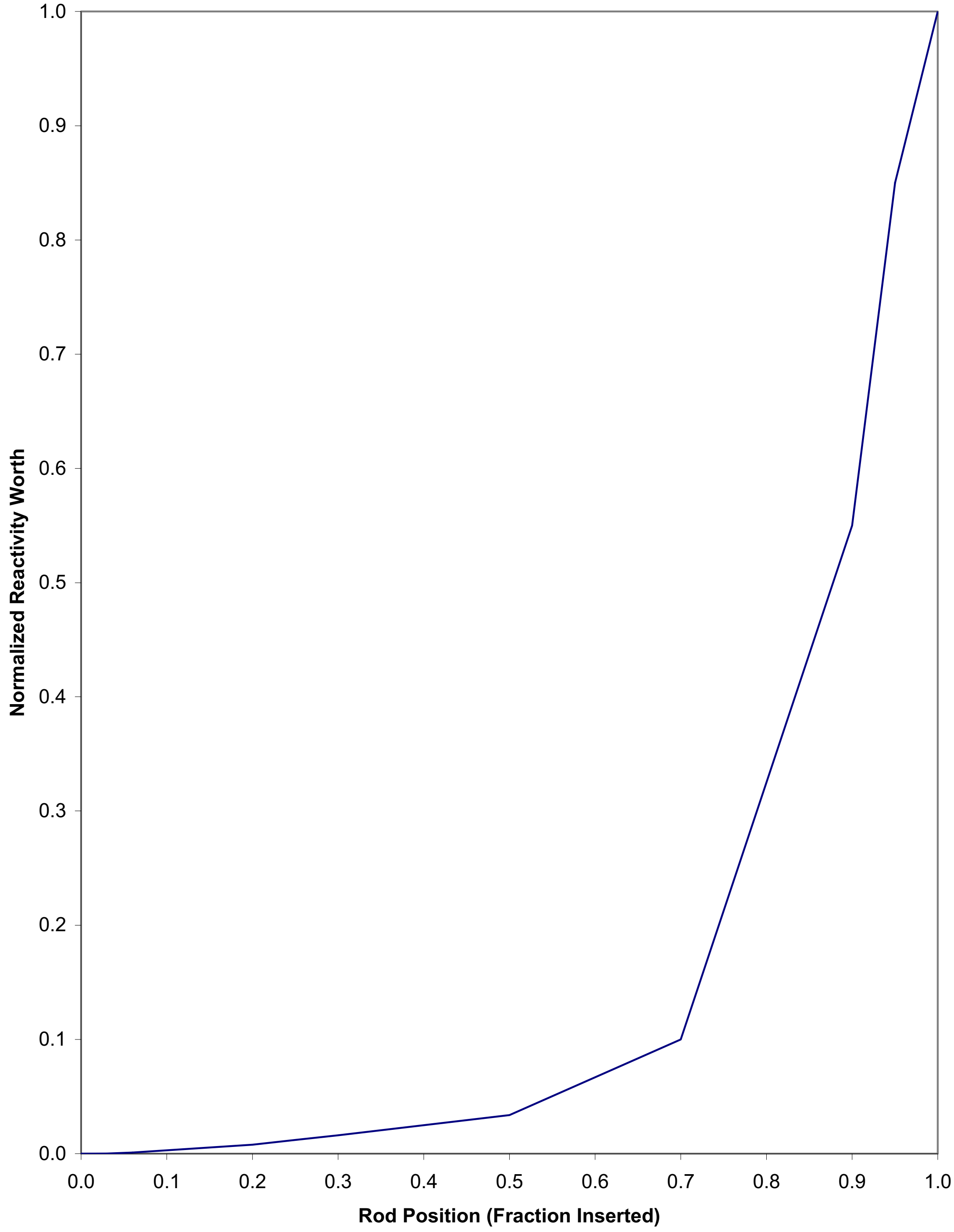


Figure 15.0.5-2
Normalized Rod Worth Versus Position

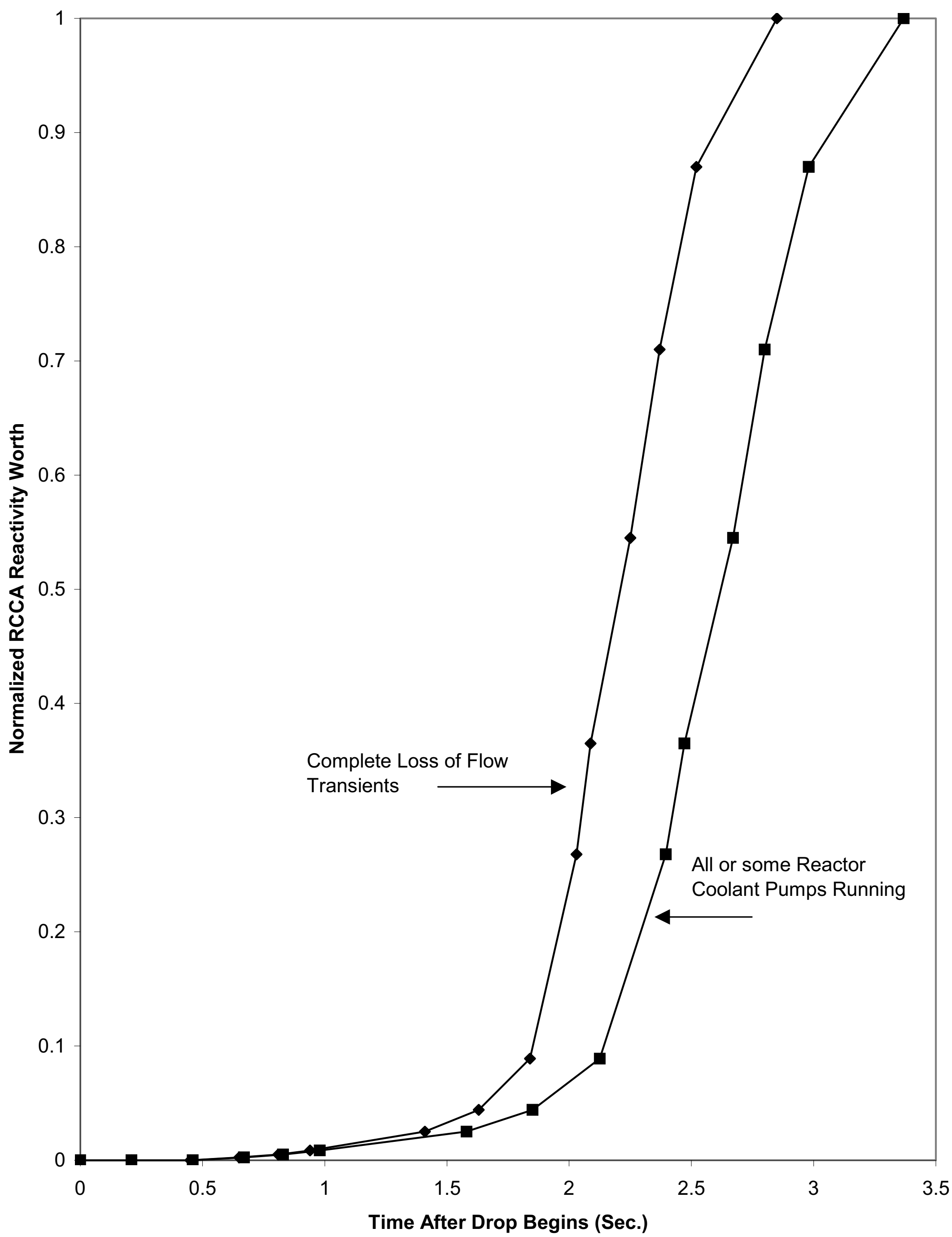


Figure 15.0.5-3
Normalized RCCA Bank
Reactivity Worth Versus Drop Time

15.1 Increase in Heat Removal From the Primary System

A number of events that could result in an increase in heat removal from the reactor coolant system are postulated. Detailed analyses are presented for the events that have been identified as limiting cases.

Discussions of the following reactor coolant system cooldown events are presented in this section:

- Feedwater system malfunctions causing a reduction in feedwater temperature
- Feedwater system malfunctions causing an increase in feedwater flow
- Excessive increase in secondary steam flow
- Inadvertent opening of a steam generator relief or safety valve
- Steam system piping failure
- Inadvertent operation of the passive residual heat removal (PRHR) heat exchanger

The preceding events are Condition II events, with the exception of small steam system piping failures, which are considered to be Condition III, and large steam system piping failure Condition IV events. Subsection 15.0.1 contains a discussion of classifications and applicable criteria.

The accidents in this section are analyzed. The most severe radiological consequences result from the main steam line break accident discussed in Subsection 15.1.5. The radiological consequences are reported only for that limiting case.

15.1.1 Feedwater System Malfunctions that Result in a Decrease in Feedwater Temperature

15.1.1.1 Identification of Causes and Accident Description

Reductions in feedwater temperature cause an increase in core power by decreasing reactor coolant temperature. Such transients are attenuated by the thermal capacity of the secondary plant and of the reactor coolant system. The overpower/overtemperature protection (neutron overpower, overtemperature, and overpower ΔT trips) prevents a power increase that could lead to a departure from nucleate boiling ratio (DNBR) that is less than the design limit values.

A reduction in feedwater temperature may be caused by a low-pressure heater train or a high-pressure heater train out of service or bypassed. At power, this increased subcooling creates an increased load demand on the reactor coolant system.

With the plant at no-load conditions, the addition of cold feedwater may cause a decrease in reactor coolant system temperature and a reactivity insertion due to the effects of the negative moderator coefficient of reactivity. However, the rate of energy change is reduced as load and feedwater flows decrease, so the no-load transient is less severe than the full-power case. The net effect on the reactor coolant system due to a reduction in feedwater temperature is similar to the effect of increasing secondary steam flow; that is, the reactor reaches a new equilibrium condition at a power level corresponding to the new steam generator ΔT.

A decrease in normal feedwater temperature is classified as a Condition II event, an incident of moderate frequency.

The protection available to mitigate the consequences of a decrease in feedwater temperature is the same as that for an excessive steam flow increase, as discussed in Subsection 15.0.8 and listed in Table 15.0-6.

15.1.1.2 Analysis of Effects and Consequences

15.1.1.2.1 Method of Analysis

This transient is analyzed by calculating conditions at the feedwater pump inlet following the removal of a low-pressure feedwater heater train from service. These feedwater conditions are then used to recalculate a heat balance through the high-pressure heaters. This heat balance gives the new feedwater conditions at the steam generator inlet.

The following assumptions are made:

- Initial plant power level corresponding to 100-percent nuclear steam supply system thermal output.
- The worst single failure in the pre-heating section of the Main Feedwater System, resulting in the maximum reduction in feedwater temperature, occurs.

Plant characteristics and initial conditions are further discussed in Subsection 15.0.3.

15.1.1.2.2 Results

A fault in the feedwater heaters section of the Feedwater System causes a reduction in feedwater temperature that increases the thermal load on the primary system. The maximum reduction in feedwater temperature, due to a single failure in the feedwater system, is lower than 71.5°F. This reduction results in an increase in heat load on the primary system of less than 10-percent full power.

15.1.1.3 Conclusions

The decrease in feedwater temperature transient is less severe than the increase in feedwater flow event or the increase in secondary steam flow event (see Subsections 15.1.2 and 15.1.3). Based on the results presented in Subsections 15.1.2 and 15.1.3, the applicable Standard Review Plan Subsection 15.1.1 evaluation criteria for the decrease in feedwater temperature event are met.

15.1.2 Feedwater System Malfunctions that Result in an Increase in Feedwater Flow

15.1.2.1 Identification of Causes and Accident Description

Addition of excessive feedwater causes an increase in core power by decreasing reactor coolant temperature. Such transients are attenuated by the thermal capacity of the secondary plant and the reactor coolant system. The overpower/overtemperature protection (neutron overpower, overtemperature, and overpower ΔT trips) prevents a power increase that leads to a DNBR less than the safety analysis limit value.

An example of excessive feedwater flow is a full opening of a feedwater control valve due to a feedwater control system malfunction or an operator error. At power, this excess flow causes an increased load demand on the reactor coolant system due to increased subcooling in the steam generator.

With the plant at no-load conditions, the addition of cold feedwater may cause a decrease in reactor coolant system temperature and a reactivity insertion due to the effects of the negative moderator coefficient of reactivity.

Continuous addition of excessive feedwater is prevented by the steam generator high-2 water level signal trip, which closes the feedwater isolation valves and feedwater control valves and trips the turbine, main feedwater pumps, and reactor.

An increase in normal feedwater flow is classified as a Condition II event, fault of moderate frequency.

Plant systems and equipment available to mitigate the effects of the accident are discussed in Subsection 15.0.8 and listed in Table 15.0-6.

In meeting the requirements of GDC 17 of 10 CFR Part 50, Appendix A, a loss of offsite power is assumed to occur as a consequence of the turbine trip for the excessive feedwater flow case initiated from full-power conditions. As discussed in Subsection 15.0.14, an excessive feedwater flow transient initiated with the plant at no-load conditions need not consider a consequential loss of offsite power. With the plant initially at zero-load, the turbine would not have been connected to the grid, so any subsequent reactor or turbine trip would not disrupt the grid and produce a consequential loss of offsite ac power.

15.1.2.2 Analysis of Effects and Consequences

15.1.2.2.1 Method of Analysis

The excessive heat removal due to a feedwater system malfunction transient primarily is analyzed by using the LOFTRAN computer code (Reference 1). LOFTRAN simulates a multiloop system, neutron kinetics, pressurizer, pressurizer safety valves, pressurizer spray, steam generator, and steam generator safety valves. The code computes pertinent plant variables, including temperatures, pressures, and power level.

For that portion of the feedwater malfunction transient that includes a primary coolant flow coastdown caused by the consequential loss of offsite power, a combination of three computer codes is used to perform the DNBR analysis. First the LOFTRAN code is used to predict the nuclear power transient, the flow coastdown, the primary system pressure transient, and the primary coolant temperature transient. The FACTRAN code (Reference 5) is then used to calculate the heat flux based on the nuclear power and flow from LOFTRAN. Finally, the VIPRE-01 code (see Section 4.4) is used to calculate the DNBR during the transient, using the heat flux from FACTRAN and the flow from LOFTRAN.

The transient is analyzed to demonstrate plant behavior if excessive feedwater addition occurs because of system malfunction or operator error that allows a feedwater control valve to open fully. The following two cases are analyzed assuming a conservatively large negative moderator temperature coefficient:

- Accidental opening of one feedwater control valve with the reactor just critical at zero load conditions.

- Accidental opening of one feedwater control valve with the reactor in automatic control at full power.

The reactivity insertion rate following a feedwater system malfunction is calculated with the following assumptions:

- For the feedwater control valve accident at full power, one feedwater control valve is assumed to malfunction resulting in a step increase to 120 percent of nominal feedwater flow to one steam generator.

- For the feedwater control valve accident at zero-load condition, a feedwater control valve malfunction occurs, which results in a step increase in flow to one steam generator from 0 in 120 percent of the nominal full-load value for one steam generator.
- For the zero-load condition, feedwater temperature is at a conservatively low value of 40°F.
- No credit is taken for the heat capacity of the reactor coolant system and steam generator thick metal in attenuating the resulting plant cooldown.
- The feedwater flow resulting from a fully open control valve is terminated by a steam generator high-2 level trip signal, which closes feedwater control and isolation valves and trips the main feedwater pumps, the turbine, and the reactor.

Plant characteristics and initial conditions are further discussed in Subsection 15.0.3.

Normal reactor control systems are not required to function. The protection and safety monitoring system may function to trip the reactor because of overpower or high-2 steam generator water level conditions. No single active failure prevents operation of the protection and safety monitoring system. A discussion of anticipated transients without trip considerations is presented in Section 15.8.

The analysis assumes that the turbine trip during the case initiated from full power results in a consequential loss of offsite power that produces the coastdown of the reactor coolant pumps. As described in Subsection 15.0.14, the loss of offsite power is modeled to occur 3.0 seconds after the turbine trip. The excessive feedwater flow analysis conservatively delays the start of rod insertion until 2.0 seconds after the reactor trip signal is generated, while assuming that the turbine trip occurs with a zero time delay following the generation of the turbine trip signal. The interaction of these assumptions produces maximum core power with minimum core coolant flow during the period of reactor coolant pump coastdown and thereby minimizes the predicted DNBRs.

15.1.2.2.2 Results

In the case of an accidental full opening of one feedwater control valve with the reactor at zero power and the preceding assumptions, the maximum reactivity insertion rate is less than the maximum reactivity insertion rate analyzed in Subsection 15.4.1 for an uncontrolled rod cluster control assembly (RCCA) bank withdrawal from a subcritical or low-power startup condition. Therefore, the results of the analysis are not presented here. If the incident occurs with the unit just critical at no-load, the reactor may be tripped by the power range high neutron flux trip (low setting) set at approximately 25-percent nominal full power.

The full-power case (maximum reactivity feedback coefficients, automatic rod control) results in the greatest power increase. Assuming the rod control system to be in the manual control mode results in a slightly less severe transient.

When the steam generator water level in the faulted loop reaches the high-2 level setpoint, the feedwater control valves and feedwater isolation valves are automatically closed and the main feedwater pumps are tripped. This prevents continuous addition of the feedwater. In addition, a turbine trip and a reactor trip are initiated.

Transient results show the increase in nuclear power and ΔT associated with the increased thermal load on the reactor (see Figures 15.1.2-1 and 15.1.2-2). A new equilibrium condition is reached and all the plant parameters, except for the SG water level, remain almost constant. Following the turbine trip, the consequential loss of offsite power produces the reactor coolant system flow coastdown shown in Figure 15.1.2-3. The minimum DNBR is predicted to occur before the reactor trip and the reactor coolant pump coastdown caused by the loss of offsite power. The minimum DNBR predicted is 2.14 using the WRB-2 equation, which is well above the design limit described in Section 4.4. Following the reactor trip, the plant approaches a stabilized and safe condition; standard plant shutdown procedures may then be followed to further cool down the plant.

Because the power level rises by a maximum of about 12 percent above nominal during the excessive feedwater flow incident, the fuel temperature also rises until after reactor trip occurs. The core heat flux lags behind the neutron flux response because of the fuel rod thermal time constant. Therefore, the peak value does not exceed 118 percent of its nominal value (the assumed high neutron flux trip setpoint). The peak fuel temperature thus remains well below the fuel melting temperature.

The transient results show that departure from nucleate boiling (DNB) does not occur at any time during the excessive feedwater flow incident. Thus, the capability of the primary coolant to remove heat from the fuel rods is not reduced and the fuel cladding temperature does not rise significantly above its initial value during the transient.

The calculated sequence of events for this accident is shown in Table 15.1.2-1.

15.1.2.3 Conclusions

The results of the analysis show that the minimum DNBR encountered for an excessive feedwater addition at power is above the design limit value. The DNBR design basis is described in Section 4.4.

Additionally, the reactivity insertion rate that occurs at no-load conditions following excessive feedwater addition is less than the maximum value considered in the analysis of the rod withdrawal from subcritical condition analysis (see Subsection 15.4.1).

15.1.3 Excessive Increase in Secondary Steam Flow

15.1.3.1 Identification of Causes and Accident Description

An excessive increase in secondary system steam flow (excessive load increase incident) results in a power mismatch between the reactor core power and the steam generator load demand. The plant control system is designed to accommodate a 10-percent step load increase or a 5-percent-per-minute ramp load increase in the range of 25- to 100-percent full power. Any loading rate in excess of these values may cause a reactor trip actuated by the protection and safety monitoring system. Steam flow increases greater than 10 percent are analyzed in Subsections 15.1.4 and 15.1.5.

This accident could result from either an administrative violation such as excessive loading by the operator or an equipment malfunction in the steam dump control or turbine speed control.

During power operation, turbine bypass to the condenser is controlled by reactor coolant condition signals. A high reactor coolant temperature indicates a need for turbine bypass. A single controller malfunction does not cause turbine bypass. An interlock blocks the opening of the valves unless a large turbine load decrease or a turbine trip has occurred.

Protection against an excessive load increase accident is provided by the following protection and safety monitoring system signals:

- Overpower ΔT
- Overtemperature ΔT
- Power range high neutron flux

An excessive load increase incident is considered to be a Condition II event, as described in Subsection 15.0.1.

In meeting the requirements of GDC 17 of 10 CFR Part 50, Appendix A, an analysis has been performed to evaluate the effects produced by a possible consequential loss of offsite power during the excessive load increase event. As discussed in Subsection 15.0.14, the loss of offsite power need be considered only as a direct consequence of a turbine trip occurring while the plant is operating at power. For the four excessive load increase cases presented, reactor and turbine trips are not predicted to occur. However, to address the loss of offsite power issue, analysis has been performed that conservatively assumes a reactor trip and an associated turbine trip occur at the time of peak power. Consistent with the discussion in Subsection 15.0.14, the analysis then models a loss of offsite power occurring 3.0 seconds after the turbine trip. The primary effect of the loss of offsite power is to cause the reactor coolant pumps to coast down.

15.1.3.2 Analysis of Effects and Consequences

15.1.3.2.1 Method of Analysis

This accident is primarily analyzed using the LOFTRAN computer code (Reference 1). LOFTRAN simulates the neutron kinetics, reactor coolant system, pressurizer, pressurizer safety valves, pressurizer spray, steam generator, steam generator safety valves, and feedwater system. The code computes pertinent plant variables including temperatures, pressures, and power level.

For the excessive load increase analysis that includes a primary coolant flow coastdown caused by the consequential loss of offsite power, a combination of three computer codes is used to perform the DNBR analysis. First the LOFTRAN code is used to predict the nuclear power transient, the flow coastdown, the primary system pressure transient, and the primary coolant temperature transient. The FACTRAN code (Reference 5) is then used to calculate the heat flux based on the nuclear power and flow from LOFTRAN. Finally, the VIPRE-01 code (see Section 4.4) is used to calculate the DNBR during the transient, using the heat flux from FACTRAN and the flow from LOFTRAN.

Four cases are analyzed to demonstrate plant behavior following a 10-percent step load increase from rated load. These cases are as follows:

- Reactor control in manual with minimum moderator reactivity feedback
- Reactor control in manual with maximum moderator reactivity feedback
- Reactor control in automatic with minimum moderator reactivity feedback
- Reactor control in automatic with maximum moderator reactivity feedback

For the minimum moderator feedback cases, the core has the least negative moderator temperature coefficient of reactivity; therefore, reductions in coolant temperature have the least impact on core power. For the maximum moderator feedback cases, the moderator temperature coefficient of reactivity has its highest absolute value. This results in the largest amount of reactivity feedback due to changes in coolant temperature. For all the cases analyzed both with and without automatic rod control, no credit is taken for ΔT trips on overtemperature or overpower in order to demonstrate the inherent transient capability of the plant. Under actual operating conditions, such a trip may occur, after which the plant quickly stabilizes.

A 10-percent step increase in steam demand is assumed, and each case is analyzed without credit being taken for pressurizer heaters. At initial reactor power, reactor coolant system pressure and temperature are assumed to be at their full power values. Uncertainties in initial conditions are included in the limit DNBR as described in WCAP-11397-P-A (Reference 2). Plant characteristics and initial conditions are further discussed in Subsection 15.0.3.

In addressing the consequential loss of offsite power, limiting cases are analyzed that model a reactor trip and an associated turbine trip occurring at the time of peak power during the limiting excessive load increase transient. The analysis has been performed conservatively assuming a reactor trip with a coincident turbine trip followed by a loss of offsite power 3.0 seconds later, as discussed in Subsection 15.0.14. The primary effect of the loss of offsite power is to cause the reactor coolant pumps to coast down.

Normal reactor control systems and engineered safety systems are not required to function.

15.1.3.2.2 Results

Figures 15.1.3-1 through 15.1.3-10 show the transient with the reactor in the manual control mode and no reactor trip signals occur. For the minimum moderator feedback case, there is a slight power increase and the average core temperature shows a large decrease. This results in a DNBR that increases above its initial value. For the maximum moderator feedback manually controlled case, there is a much faster increase in reactor power due to the moderator feedback. A reduction in the DNBR occurs, but the DNBR remains above the design limit (see Section 4.4).

Figures 15.1.3-11 through 15.1.3-20 show the transient assuming the reactor is in the automatic control mode. A reactor trip signal setpoint is reached but, conservatively, reactor trip is not credited. Both the minimum and maximum moderator feedback cases show that core power increases and thereby reduces the rate of decrease in coolant average temperature and pressurizer pressure. For both of these cases, the minimum DNBR remains above the design limit (see Section 4.4).

For the cases with no reactor trip signal, the plant power stabilizes at an increased power level. Normal plant operating procedures are followed to reduce power. Because of the measurement errors assumed in the setpoints, it is possible that reactor trip could actually occur for the automatic control and maximum feedback cases. The plant reaches a stabilized condition following the trip.

For the analysis performed modeling a loss of offsite power and the subsequent reactor coolant pump coastdown, the results show that the minimum DNBRs predicted during the excessive load increase cases occur prior to the time the flow coastdown begins. Therefore, the DNB ratio results provided in Figures 15.1.3-5, 15.1.3-10, 15.1.3-15, and 15.1.3-20 are bounding, and the minimum DNBR during the flow coastdown remains well above the design limit defined in Section 4.4. Since the loss of offsite power is delayed for 3.0 seconds after the turbine trip, the RCCAs are inserted well into the core before the reactor coolant system flow coastdown begins. The resulting power reduction compensates for the reduced flow encountered once power to the reactor coolant pumps is lost.

The excessive load increase incident is an overpower transient for which the fuel temperature rises. Reactor trip may not occur for some of the cases analyzed, and the plant reaches a new equilibrium condition at a higher power level corresponding to the increase in steam flow.

Because DNB does not occur during the excessive load increase transients, the capability of the primary coolant to remove heat from the fuel rod is not reduced. Thus, the fuel cladding temperature does not rise significantly above its initial value during the transient.

The calculated sequence of events for the excessive load increase cases with no reactor trip are shown in Table 15.1.2-1.

15.1.3.3 Conclusions

The analysis presented in this subsection demonstrates that for a 10-percent step load increase, the DNBR remains above the design limit. The design basis for DNB is described in Section 4.4. The plant rapidly reaches a stabilized condition following the load increase.

15.1.4 Inadvertent Opening of a Steam Generator Relief or Safety Valve

15.1.4.1 Identification of Causes and Accident Description

The most severe core conditions resulting from an accidental depressurization of the main steam system are associated with an inadvertent opening of a single steam dump, relief, or safety valve. The analyses performed assuming a rupture of a main steam line are given in Subsection 15.1.5.

The steam release, as a consequence of this accident, results in an initial increase in steam flow which decreases during the accident as the steam pressure falls. The energy removal from the reactor coolant system causes a reduction of coolant temperature and pressure. In the presence of a negative moderator temperature coefficient, the cooldown results in an insertion of positive reactivity.

The analysis is performed to demonstrate that the following Standard Review Plan Subsection 15.1.4 evaluation criterion is satisfied.

Assuming the most reactive stuck RCCA, with offsite power available, and assuming a single failure in the engineered safety features system, there will be no consequential damage to the fuel or reactor coolant system after reactor trip for a steam release equivalent to the spurious opening, with failure to close, of the largest of any single steam dump, relief, or safety valve. This criterion is met by showing the DNB design basis is not exceeded.

Accidental depressurization of the secondary system is classified as a Condition II event as described in Section 15.1.

The following systems provide the necessary protection against an accidental depressurization of the main steam system:

- Core makeup tank actuation from one of the following signals:
  - Safeguards ("S") signal
    - Two out of four low pressurizer pressure signals
    - Two out of four high-2 containment pressure signals
    - Two out of four low T_{cold} signals in any one loop
    - Two out of four low steam line pressure signals in any one loop
  - Two out of four low-2 pressurizer level signals
- The overpower reactor trips (neutron flux and ΔT) and the reactor trip occurring in conjunction with receipt of the "S" signal
- Redundant isolation of the main feedwater lines

  Sustained high feedwater flow causes additional cooldown. Therefore, in addition to the normal control action that closes the main feedwater valves following reactor trip, an "S" signal rapidly closes the feedwater control valves and feedwater isolation valves, and trips the main feedwater pumps.
- Redundant isolation of the startup feedwater system

Sustained high startup feedwater flow causes additional cooldown. Therefore, the low T_{cold} signal closes the startup feedwater control and isolation valves.

- Trip of the fast-acting main steam line isolation valves (assumed to close in less than 10 seconds) on one of the following signals:

  - Two out of four low steam line pressure signals in any one loop (above permissive P-11)
  - Two out of four high negative steam pressure rates in any loop (below permissive P-11)
  - Two out of four low T_{cold} signals in any one loop
  - Two out of four high-2 containment pressure signals

Plant systems and equipment available to mitigate the effects of the accident are discussed in Subsection 15.0.8 and listed in Table 15.0-6.

15.1.4.2 Analysis of Effects and Consequences

15.1.4.2.1 Method of Analysis

The following analyses of a secondary system steam release are performed:

- A full plant digital computer simulation using the LOFTRAN code (Reference 1) to determine reactor coolant system temperature and pressure during cooldown, and the effect of core makeup tank injection

- Analyses to determine that there is no damage to the fuel or reactor coolant system

The following conditions are assumed to exist at the time of a secondary steam system release:

- End-of-life shutdown margin at no-load, equilibrium xenon conditions, and with the most reactive RCCA stuck in its fully withdrawn position. Operation of RCCA mechanical shim and axial offset banks during core burnup is restricted by the insertion limits so that shutdown margin requirements are satisfied.

- The most negative moderator coefficient corresponding to the end-of-life rodded core with the most reactive RCCA in the fully withdrawn position. The variation of the coefficient with temperature is included. The k_{eff} (considering moderator temperature and density effects) versus temperature corresponding to the negative moderator temperature coefficient used is shown in Figure 15.1.4-1. The core power is modeled as a function of core mass flow, core boron concentration, and core inlet temperature.

- Minimum capability for injection of boric acid solution corresponding to the most restrictive single failure in the passive core cooling system. Low-concentration boric acid must be swept from the core makeup tank lines downstream of isolation valves before delivery of boric acid (3400 ppm) to the reactor coolant loops. This effect has been accounted for in the analysis.

- The case studied is a steam flow of 520 pounds per second at 1200 psia with offsite power available. This conservatively models the maximum capacity of any single steam dump, relief, or safety valve. Initial hot shutdown conditions at time zero are assumed because this represents the most conservative initial conditions.

Should the reactor be just critical or operating at power at the time of a steam release, the reactor is tripped by the normal overpower protection when power level reaches a trip point. Following a trip at power, the reactor coolant system contains more stored energy than at no-load. This is because the average coolant temperature is higher than at no-load, and there is appreciable energy stored in the fuel. The additional stored energy is removed via the cooldown caused by the steam release before the no-load conditions of the reactor coolant system temperature and shutdown margin assumed in the analyses are reached.

After the additional stored energy is removed, the cooldown and reactivity insertions proceed in the same manner as in the analysis, which assumes no-load condition at time zero. However, because the initial steam generator water inventory is greatest at no-load, the magnitude and duration of the reactor coolant system cooldown are less for steam line release occurring at power:

- In computing the steam flow, the Moody Curve (Reference 3) for f(L/D) = 0 is used.
- Perfect moisture separation occurs in the steam generator.
- Offsite power is available, because this maximizes the cooldown.
- Maximum cold startup feedwater flow is assumed.
- Four reactor coolant pumps are initially operating.
- Manual actuation of the PRHR system at time zero is conservatively assumed to maximize the cooldown.

15.1.4.2.2 Results

The results presented conservatively indicate the events that would occur assuming a secondary system steam release because it is postulated that the conditions just described occur simultaneously.

Figures 15.1.4-2 through 15.1.4-12 show the transient results for a steam flow of 520 pounds per second at 1200 psia.

The assumed steam release is typical of the capacity of any single steam dump, relief, or safety valve. Core makeup tank injection and the associated tripping of the reactor coolant pumps are initiated automatically by the low T_{cold} “S” signal. Boron solution at 3400 ppm enters the reactor coolant system, providing enough negative reactivity to prevent a significant return to power and core damage. Later in the transient, as the reactor coolant pressure continues to fall, the accumulators actuate and inject boron solution at 2600 ppm.

The transient is conservative with respect to cooldown, because no credit is taken for the energy stored in the system metal other than that of the fuel elements and steam generator tubes, and the PRHR system is assumed to be actuated at time zero. Because the limiting portion of the transient occurs over a period of about 5 minutes, the neglected stored energy is likely to have a significant effect in slowing the cooldown.

The calculated time sequence of events for this accident is listed in Table 15.1.2-1.

15.1.4.3 Margin to Critical Heat Flux

The analysis demonstrates that the DNB design basis, as described in Section 4.4, is met for the inadvertent opening of a steam generator relief or safety valve. As shown in Figure 15.1.4-2, no significant return to power occurs and, therefore, DNB does not occur. The minimum DNBR is conservatively calculated and is above the 95/95 limit.

15.1.4.4 Conclusions

The analysis shows that the criterion stated in this subsection is satisfied. For an inadvertent opening of any single steam dump or a steam generator relief or safety valve, the DNB design basis is met.

15.1.5 Steam System Piping Failure

15.1.5.1 Identification of Causes and Accident Description

The steam release arising from a rupture of a main steam line results in an initial increase in steam flow, which decreases during the accident as the steam pressure falls. The energy removal from the reactor coolant system causes a reduction of coolant temperature and pressure. In the presence of a negative moderator temperature coefficient, the cooldown results in an insertion of positive reactivity.

If the most reactive RCCA is assumed stuck in its fully withdrawn position after reactor trip, there is an increased possibility that the core becomes critical and returns to power. A return to power following a steam line rupture is a potential problem mainly because of the existing high-power peaking factors, assuming the most reactive RCCA to be stuck in its fully withdrawn position. The core is ultimately shut down by the boric acid solution delivered by the passive core cooling system.

The analysis of a main steam line rupture is performed to demonstrate that the following Standard Review Plan Subsection 15.1.5 evaluation criterion is satisfied.

Assuming the most reactive stuck RCCA with or without offsite power and assuming a single failure in the engineered safety features system, the core cooling capability is maintained. As shown in Subsection 15.1.5.4, radiation doses are within the guidelines.

DNB and possible cladding perforation following a steam pipe rupture are not necessarily unacceptable. The following analysis shows that the DNB design basis is not exceeded for any steamline rupture, assuming the most reactive assembly stuck in its fully withdrawn position.

A major steam line rupture is classified as a Condition IV event.

Effects of minor secondary system pipe breaks are bounded by the analysis presented in this section. Minor secondary system pipe breaks are classified as Condition III events, as described in Subsection 15.0.1.3.

The major rupture of a steam line is the most limiting cooldown transient and is analyzed at zero power with no decay heat. Decay heat retards the cooldown and thereby reduces the likelihood that the reactor returns to power. A detailed analysis of this transient with the most limiting break size, a double-ended rupture, is presented here.

Certain assumptions used in this analysis are discussed in WCAP-9226 (Reference 4). WCAP-9226 also contains a discussion of the spectrum of break sizes and power levels analyzed.

The following functions provide the protection for a steam line rupture (see Subsection 7.2.1.1.2):

- Core makeup tank actuation from any of the following:
  - Two out of four low pressurizer pressure signals
  - Two out of four high-2 containment pressure signals
  - Two out of four low steam line pressure signals in any loop
  - Two out of four low T_{cold} signals in any one loop
  - Two out of four low-2 pressurizer level signals

- The overpower reactor trips (neutron flux and ΔT) and the reactor trip occurring in conjunction with receipt of the "S" signal

- Redundant isolation of the main feedwater lines

  Sustained high feedwater flow causes additional cooldown. Therefore, in addition to the normal control action that closes the main feedwater control valves, the "S" signal rapidly closes the feedwater control valves and feedwater isolation valves, and trips the main feedwater pumps.

- Redundant isolation of the startup feedwater system

  Sustained high startup feedwater flow causes additional cooldown. Therefore, the low T_{cold} signal closes the startup feedwater control and isolation valves.

- Fast-acting main steam line isolation valves (assumed to close in less than 10 seconds) on any of the following:

  – Two out of four high-2 containment pressure

  – Two out of the four low steam line pressure signals in any one loop (above permissive P-11)

  – Two out of four high negative steam pressure rates in any one loop (below permissive P-11)

  – Two out of four low T_{cold} signals in any one loop

A fast-acting main steam isolation valve is provided in each steam line. These valves are assumed to fully close within 10 seconds of actuation following a large break in the steam line. For breaks downstream of the main steam line isolation valves, closure of at least one valve in each line terminates the blowdown.

For any break in any location, no more than one steam generator would experience an uncontrolled blowdown even if one of the main steam line isolation valves fails to close. A description of steam line isolation is included in Chapter 10.

Flow restrictors are installed in the steam generator outlet nozzle, as an integral part of the steam generator. The effective throat area of the nozzles is 1.4 ft^2, which is considerably less than the main steam pipe area; thus, the flow restrictors serve to limit the maximum steam flow for a break at any location.

Design criteria and methods of protection of safety-related equipment from the dynamic effects of postulated piping ruptures are provided in Section 3.6.

15.1.5.2 Analysis of Effects and Consequences

15.1.5.2.1 Method of Analysis

The analysis of the steam pipe rupture is performed to determine the following:

- The core heat flux and reactor coolant system temperature and pressure resulting from the cooldown following the steam line break. The LOFTRAN code (Reference 1) is used to model the system transient.

- The thermal-hydraulic behavior of the core following a steam line break. A detailed thermal-hydraulic digital computer code, VIPRE-01, is used to determine if DNB occurs for the core transient conditions computed by the LOFTRAN code.

The following conditions are assumed to exist at the time of a main steam line break accident:

- End-of-cycle shutdown margin at no-load, equilibrium xenon conditions, and the most reactive rod control assembly stuck in its fully withdrawn position. Operation of the control rod mechanical shim and axial offset banks during core burnup is restricted by the insertion limits so that shutdown margin requirements are satisfied.

- A negative moderator coefficient corresponding to the end-of-life rodded core with the most reactive RCCA in the fully withdrawn position. The variation of the coefficient with temperature is included. The k_{eff} (considering moderator temperature and density effects) versus temperature corresponding to the negative moderator temperature coefficient used is shown in Figure 15.1.4-1. The core power is modeled as a function of core mass flow, core boron concentration, and core inlet temperature.

The core properties used in the LOFTRAN mode for feedback calculations are generated by combining those in the sector nearest the affected steam generator with those associated with the remaining sector. The resultant properties reflect a combination process that accounts for inlet plenum fluid mixing and a conservative weighing of the fluid properties from the coldest core sector.

In verifying the conservatism of this method, the power predictions of the LOFTRAN modeling are confirmed by comparison with detailed core analysis for the limiting conditions of the cases considered. This core analysis conservatively models the hypothetical core configuration (that is, stuck RCCA, nonuniform inlet temperatures, pressure, flow, and boron concentration) and directly evaluates the total reactivity feedback including power, boron, and density redistribution in an integral fashion. The effect of void formation is also included.

Comparison of the results from the detailed core analysis with the LOFTRAN predictions verify the overall conservatism of the methodology. That is, the specific power, temperature, and flow conditions used to perform the DNB analysis are conservative.

- The core makeup tanks and the accumulators are the portions of the passive core cooling system used in mitigating a steam line rupture. There are no single failures that prevent core makeup tank injection. In modeling the core makeup tanks and the accumulators, conservative assumptions are used that minimize the capability to add borated water. Specifically, the core makeup tank injection line characteristics modeled reflect the failure of one core makeup tank discharge valve.

- The maximum overall fuel-to-coolant heat transfer coefficient is used to maximize the rate of cooldown.

- Because the steam generators are provided with integral flow restrictors with a 1.4-ft^2 throat area, any rupture in a steam line with a break area greater than 1.4 ft^2, regardless of location, has the same effect on the primary plant as the 1.4-ft^2 double-ended rupture. The limiting case considered in determining the core power and reactor coolant system transient is the complete severance of a pipe, with the plant initially at no-load conditions and full reactor coolant flow with offsite power available. The results of this case bound the loss of offsite power case for the following reasons:

- Loss of offsite power results in an immediate reactor coolant pump coastdown at the initiation of the transient. This reduces the severity of the reactor coolant system cooldown by reducing primary-to-secondary heat transfer. The lessening of the cooldown, in turn, reduces the magnitude of the return to power.

- Following actuation, the core makeup tank provides borated water that injects into the reactor coolant system. Flow from the core makeup tank increases if the reactor coolant pumps have coasted down. Therefore, the analysis performed with offsite power and continued reactor coolant pump operation reduces the rate of boron injection into the core and is conservative.

- The protection system automatically provides a safety-related signal that initiates the coastdown of the reactor coolant pumps in parallel with core makeup tank actuation. Because this reactor coolant pump trip function is actuated early during the steam line break event (right after core makeup tank actuation), there is very little difference in the predicted DNBR between cases with and without offsite power.

- Because of the passive nature of the safety injection system, the loss of offsite power does not delay the actuation of the safety injection system.

- Power peaking factors corresponding to one stuck RCCA are determined at the end of core life. The coldest core inlet temperatures are assumed to occur in the sector with the stuck rod. The power peaking factors account for the effect of the local void in the region of the stuck RCCA during the return to power phase following the steam line break. This void in conjunction with the large negative moderator coefficient partially offsets the effect of the stuck assembly. The power peaking factors depend upon the core power, temperature, pressure, and flow and, therefore, may differ for each case studied.

The analysis assumes initial hot standby conditions at time zero in order to present a representative case which will yield limiting post-trip DNBR results for this transient. If the reactor is just critical or operating at power at the time of a steam line break, the reactor is tripped by the overpower protection system when power level reaches a trip point.

Following a trip at power, the reactor coolant system contains more stored energy than at no-load because the average coolant temperature is higher than at no-load, and there is energy stored in the fuel. The additional stored energy reduces the cooldown caused by the steam line break before the no-load conditions of reactor coolant system temperature and shutdown margin assumed in the analyses are reached.

After the additional stored energy has been removed, the cooldown and reactivity insertions proceed in the same manner as in the analysis that assumes a no-load condition at time zero.

- In computing the steam flow during a steam line break, the Moody Curve (Reference 3) for f(L/D) = 0 is used.

- Perfect moisture separation occurs in the steam generator.

- Maximum cold startup feedwater flow plus nominal 100 percent main feedwater flow is assumed.

- Four reactor coolant pumps are initially operating.

- Manual actuation of the PRHR system at time zero is conservatively assumed to maximize the cooldown.

15.1.5.2.2 Results

The calculated sequence of events for the analyzed case is shown in Table 15.1.2-1. The results presented conservatively indicate the events that would occur assuming a steam line rupture because it is postulated that the conditions described occur simultaneously.

15.1.5.2.3 Core Power and Reactor Coolant System Transient

Figures 15.1.5-1 through 15.1.3-13 show the reactor coolant system transient and core heat flux following a main steam line rupture (complete severance of a pipe) at initial no-load condition.

Offsite power is assumed available so that, initially, full reactor coolant flow exists. During the course of the event, the reactor protection system initiates a trip of the reactor coolant pumps in conjunction with actuation of the core makeup tanks. The transient shown assumes an uncontrolled steam release from only one steam generator. Steam release from more than one steam generator is prevented by automatic trip of the main steam isolation valves in the steam lines by high containment pressure signals or by low steam line pressure signals. Even with the failure of one valve, release is limited to approximately 10 seconds for the other steam generator while the one generator blows down. The main steam isolation valves fully close in less than 10 seconds from receipt of a closure signal.

As shown in Figure 15.1.5-1, the core attains criticality with the RCCAs inserted (with the design shutdown assuming the most reactive RCCA stuck) before boron solution at 3400 ppm (from core makeup tanks) or 2600 ppm (from accumulators) enters the reactor coolant system. A peak core power significantly lower than the nominal full-power value is attained.

The calculation assumes that the boric acid is mixed with and diluted by the water flowing in the reactor coolant system before entering the reactor core. The concentration after mixing depends upon the relative flow rates in the reactor coolant system and from the core makeup tanks or accumulators (or both). The variation of mass flow rate in the reactor coolant system due to water density changes is included in the calculation. The variation of flow rate from the core makeup tanks or accumulators (or both) due to changes in the reactor coolant system pressure and temperature and the pressurizer level is also included. The reactor coolant system and passive injection flow calculations include line losses.

At no time during the analyzed steam line break event does the core makeup tank level approach the setpoint for actuation of the automatic depressurization system. During non-LOCA events, the core makeup tanks remain filled with water. The volume of injection flow leaving the core makeup tank is offset by an equal volume of recirculation flow that enters the core makeup tanks via the reactor coolant system cold leg balance lines.

The PRHR system provides a passive, long-term means of removing the core decay and stored heat by transferring the energy via the PRHR heat exchanger to the in-containment refueling water storage tank (IRWST). The PRHR heat exchanger is normally actuated automatically when the steam generator level falls below the low wide-range level. For the main steam line rupture case analyzed, the PRHR exchanger is conservatively actuated at time zero to maximize the cooldown.

15.1.5.2.4 Margin to Critical Heat Flux

The case presented in Subsection 15.1.5.2.2 conservatively models the expected behavior of the plant during a steam system piping failure. This includes the tripping of the reactor coolant pumps coincident with core makeup tank actuation. A DNB analysis is performed using limiting assumptions that bound those of Subsection 15.1.5.2.2.

Under the low flow (natural circulation) conditions present in the AP1000 transient, the return to power is severely limited by the large negative feedback due to flow and power. The minimum DNBR is conservatively calculated and is above the 95/95 limit.

15.1.5.3 Conclusions

The analysis shows that the DNB design basis is met for the steam system piping failure event. DNB and possible cladding perforation following a steam pipe rupture are not precluded by the criteria. The preceding analysis shows that no DNB occurs for the main steam line rupture assuming the most reactive RCCA stuck in its fully withdrawn position.

15.1.5.4 Radiological Consequences

The evaluation of the radiological consequences of a postulated main steam line break outside containment assumes that the reactor has been operating with the design basis fuel defect level (0.25 percent of power produced by fuel rods containing cladding defects) and that leaking steam generator tubes have resulted in a buildup of activity in the secondary coolant.

Following the rupture, startup feedwater to the faulted loop is isolated and the steam generator is allowed to steam dry. Any radioiodines carried from the primary coolant into the faulted steam generator via leaking tubes are assumed to be released directly to the environment. It is conservatively assumed that the reactor is cooled by steaming from the intact loop.

15.1.5.4.1 Source Term

The only significant radionuclide releases due to the main steam line break are the iodines and alkali metals that become airborne and are released to the environment as a result of the accident. Noble gases are also released to the environment. Their impact is secondary because any noble gases entering the secondary side during normal operation are rapidly released to the environment.

The analysis considers two different reactor coolant iodine source terms, both of which consider the iodine spiking phenomenon. In one case, the initial iodine concentrations are assumed to be those associated with equilibrium operating limits for primary coolant iodine activity. The iodine spike is assumed to be initiated by the accident with the spike causing an increasing level of iodine in the reactor coolant.

The second case assumes that the iodine spike occurs prior to the accident and that the maximum resulting reactor coolant iodine concentration exists at the time the accident occurs.

The reactor coolant noble gas concentrations are assumed to be those associated with equilibrium operating limits for primary coolant noble gas activity. The reactor coolant alkali metal concentrations are based on those associated with the design basis fuel defect level.

The secondary coolant is assumed to have an iodine source term of 0.01 μCi/g dose equivalent I-131. This is 1 percent of the maximum primary coolant activity at equilibrium operating conditions. The secondary coolant alkali metal concentration is also assumed to be 1 percent of the primary concentration.

15.1.5.4.2 Release Pathways

There are three components to the accident releases:

- The secondary coolant in the steam generator of the faulted loop is assumed to be released out the break as steam. Any iodine and alkali metal activity contained in the coolant is assumed to be released.
- The reactor coolant leaking into the steam generator of the faulted loop is assumed to be released to the environment without any credit for partitioning or plateout onto the interior of the steam generator.
- The reactor coolant leaking into the steam generator of the intact loop would mix with the secondary coolant and thus raise the activity concentrations in the secondary water. While the steam release from the intact loop would have partitioning of non-gaseous activity, this analysis conservatively assumes that any activity entering the secondary side is released.

Credit is taken for decay of radionuclides until release to the environment. After release to the environment, no consideration is given to radioactive decay or to cloud depletion by ground deposition during transport offsite.

15.1.5.4.3 Dose Calculation Models

The models used to calculate doses are provided in Appendix 15A.

15.1.5.4.4 Analytical Assumptions and Parameters

The assumptions and parameters used in the analysis are listed in Table 15.1.5-1.

15.1.5.4.5 Identification of Conservatisms

The assumptions and parameters used in the analysis contain a number of significant conservatisms:

- The reactor coolant activities are based on a fuel defect level of 0.25 percent. The expected fuel defect level is far less than this (see Section 11.1).
- The assumed leakage of 150 gallons of reactor coolant per day into each steam generator is conservative. The leakage is expected to be a small fraction of this during normal operation.
- The conservatively selected meteorological conditions are present only rarely.

15.1.5.4.6 Doses

Using the assumptions from Table 15.1.5-1, the calculated total effective dose equivalent (TEDE) doses for the case with accident-initiated iodine spike are determined to be less than 0.6 rem at the site boundary for the limiting 2-hour interval (4.8 to 6.8 hours) and 1.1 rem at the low population zone outer boundary. These doses are small fractions of the dose guideline of 25 rem TEDE identified in 10 CFR Part 50.34. A “small fraction” is defined, consistent with the Standard Review Plan, as being 10 percent or less. The TEDE doses for the case with pre-existing iodine spike are determined to be less than 0.5 rem at the site boundary for the limiting 2-hour interval (0 to 2 hours) and 0.4 rem at the low population zone outer boundary. These doses are within the dose guidelines of 10 CFR Part 50.34.

At the time the main steam line break occurs, the potential exists for a coincident loss of spent fuel pool cooling with the result that the pool could reach boiling and a portion of the radioactive iodine in the spent fuel pool could be released to the environment. The loss of spent fuel pool cooling has been evaluated for a duration of 30 days. The 30-day contribution to the dose at the site boundary and the low population zone boundary is less than 0.01 rem TEDE. When this is added to the dose calculated for the main steam line break, the resulting total dose remains less than the values reported above.

15.1.6 Inadvertent Operation of the PRHR Heat Exchanger

15.1.6.1 Identification of Causes and Accident Description

The inadvertent actuation of the PRHR heat exchanger causes an injection of relatively cold water into the reactor coolant system. This produces a reactivity insertion in the presence of a negative moderator temperature coefficient. To prevent this reactivity increase from causing reactor power increase, a reactor trip is initiated when either PRHR discharge valve comes off of its fully shut seat.

The inadvertent actuation of the PRHR heat exchanger could be caused by operator error or a false actuation signal, or by malfunction of a discharge valve. Actuation of the PRHR heat exchanger involves opening one of the isolation valves, which establishes a flow path from one reactor coolant system hot leg, through the PRHR heat exchanger, and back into its associated steam generator cold leg plenum.

The PRHR heat exchanger is located above the core to promote natural circulation flow when the reactor coolant pumps are not operating. With the reactor coolant pumps in operation, flow through the PRHR heat exchanger is enhanced. The heat sink for the PRHR heat exchanger is provided by the IRWST, in which the PRHR heat exchanger is submerged. Because the fluid in the heat exchanger is in thermal equilibrium with water in the tank, the initial flow out of the PRHR heat exchanger is significantly colder than the reactor coolant system fluid. Following this initial insurge, the reduction in cold leg temperature is limited by the cooling capability of the PRHR heat exchanger. Because the PRHR heat exchanger is connected to only one reactor coolant system loop, the cooldown resulting from its actuation is asymmetric with respect to the core.

The response of the plant to an inadvertent PRHR heat exchanger actuation with the plant at no-load conditions is bounded by the analyses performed for the inadvertent opening of a steam generator relief or safety valve event (Subsection 15.1.4) and the steam system piping failure event (Subsection 15.1.5). Both of these events are conservatively analyzed assuming PRHR heat exchanger actuation coincident with the steam line depressurization. Therefore, only the response of the plant to an inadvertent PRHR initiation with the core at power is considered.

In meeting the requirements of GDC 17 of 10 CFR Part 50, Appendix A, the effects of a possible consequential loss of ac power during an inadvertent PRHR heat exchanger actuation event have been evaluated to not adversely impact the analysis results. This conclusion is based on a review of the time sequence associated with a consequential loss of ac power in comparison to the reactor shutdown time for an inadvertent PRHR heat exchanger actuation event. The primary effect of the loss of ac power is to cause the reactor coolant pumps (RCPs) to coast down. The protection and safety monitoring system includes a 5-second minimum delay between the reactor trip and the turbine trip. In addition, a 3-second delay between the turbine trip and the loss of offsite ac power is assumed, consistent with Section 15.1.3 of NUREG-1793. Considering these delays between the time of the reactor trip and RCP coastdown due to the loss of ac power, it is clear that the plant shutdown sequence will have passed the critical point and the control rods will have been completely inserted before the RCPs begin to coast down. Therefore, the consequential loss of ac power does not adversely impact this inadvertent PRHR heat exchanger actuation analysis because the plant will be shut down well before the RCPs begin to coast down.

The inadvertent actuation of the PRHR heat exchanger event is a Condition II event, a fault of moderate frequency. Plant systems and equipment available to mitigate the effects of the accident are discussed in Subsection 15.0.8 and listed in Table 15.0-6. The following reactor protection system functions are available to provide protection in the event of an inadvertent PRHR heat exchanger actuation:

- PRHR discharge valve not closed
- Overpower/overtemperature reactor trips (neutron flux and ΔT)
- Two out of four low pressurizer pressure signals

Due to the potential consequences as a result of the reactivity excursion, a reactor trip has been designed so that upon an inadvertent PRHR actuation, a reactor trip will occur. This reactor trip is generated when either of the discharge valves is not closed. This ensures that the reactor will be tripped prior to a power increase due to the cold water injection.

15.1.6.2 Analysis of Effects and Consequences

Since a reactor trip is initiated as soon as the PRHR discharge valves are not fully closed, this event is essentially a reactor trip from the initial condition and requires no separate transient analysis.

15.1.6.3 Conclusions

Inadvertent actuation of the PRHR does not result in violation of the core thermal design limits (DNB and linear power generation) or RCS overpressure.

15.1.7 Combined License Information

This section contained no requirement for additional information.

15.1.8 References

1. Burnett, T. W. T., et al., “LOFTRAN Code Description,” WCAP-7907-P-A (Proprietary), and WCAP-7907-A (Nonproprietary), April 1984.

2. Friedland, A. J., and Ray, S., “Revised Thermal Design Procedure,” WCAP-11397-P-A (Proprietary) and WCAP-11397-A (Nonproprietary), April 1989.

3. Moody, F. S., “Transactions of the ASME, Journal of Heat Transfer,” Figure 3, page 134, February 1965.

4. Wood, D. C., and Hollingsworth, S. D., “Reactor Core Response to Excessive Secondary Steam Releases,” WCAP-9226 (Proprietary) and WCAP-9227 (Nonproprietary), January 1978.

5. Hargrove, H. G., “FACTRAN - A FORTRAN-IV Code for Thermal Transients in a UO_2 Fuel Rod,” WCAP-7908-A, December 1989.

6. “AP1000 Code Applicability Report,” WCAP-15644-P (Proprietary) and WCAP-15644-NP (Nonproprietary), Revision 2, March 2004.

Table 15.1.2-1 (Sheet 1 of 2)
Time Sequence of Events for Incidents That Result in an Increase in Heat Removal From the Primary System

Accident	Event	Time (seconds)
Excessive increase in secondary steam flow		
– Manual reactor control (minimum moderator feedback)	10-percent step load increase Equilibrium conditions reached (approximate time only)	0.0 250.0
– Manual reactor control (maximum moderator feedback)	10-percent step load increase Equilibrium conditions reached (approximate time only)	0.0 70.0
– Automatic reactor control (minimum moderator feedback)	10-percent step load increase Equilibrium conditions reached (approximate time only)	0.0 125.0
– Automatic reactor control (maximum moderator feedback)	10-percent step load increase Equilibrium conditions reached (approximate time only)	0.0 50.0
Feedwater system malfunctions that result in an increase in feedwater flow	One main feedwater control valve fails fully open Turbine trip/feedwater isolation and reactor trip on high steam generator level Rod motion begins Loss of offsite power occurs Minimum DNBR occurs	0.0 201.9 203.9 204.9 205.8
Inadvertent operation of the PRHR	PRHR discharge valves go fully open Reactor trip setpoint reached Rod motion begins Rods fully inserted	0.0 0.0 1.25 3.95

Table 15.1.2-1 (Sheet 2 of 2)
Time Sequence of Events for Incidents That
Result in an Increase in Heat Removal From the Primary System

Accident	Event	Time (seconds)
Inadvertent opening of a steam generator relief or safety valve	Inadvertent opening of one main steam safety or relief valve	0.0
	“S” actuation signal on safeguards low T_{cold}	120.3
	Core makeup tank actuation	137.3
	Boron reaches core	152.6
Steam system piping failure	Steam line ruptures	0.0
	“S” actuation signal on safeguards low steam line pressure	1.4
	Criticality attained	28.0
	Boron reaches core	33.2
	Pressurizer empty	58.2

Table 15.1.5-1
Parameters Used in Evaluating the Radiological Consequences of a Main Steam Line Break

Reactor coolant iodine activity	
– Accident-initiated spike	Initial activity equal to the equilibrium operating limit for reactor coolant activity of 1.0 µCi/g dose equivalent I-131 with an assumed iodine spike that increases the rate of iodine release from fuel into the coolant by a factor of 500 (see Appendix 15A). Duration of spike is 5 hours.
– Preaccident spike	An assumed iodine spike that has resulted in an increase in the reactor coolant activity to 60 µCi/g of dose equivalent I-131 (see Appendix 15A)
Reactor coolant noble gas activity	Equal to the operating limit for reactor coolant activity of 280 µCi/g dose equivalent Xe-133
Reactor coolant alkali metal activity	Design basis activity (see Table 11.1-2)
Secondary coolant initial iodine and alkali metal activity	1% of reactor coolant concentrations at maximum equilibrium conditions
Duration of accident (hr)	72
Atmospheric dispersion (χ/Q) factors	See Table 15A-5 in Appendix 15A
Steam generator in faulted loop	
– Initial water mass (lb)	3.32 E+05
– Primary to secondary leak rate (lb/hr)	52.25[a]
– Iodine partition coefficient	1.0
– Steam released (lb) 0 - 2 hr 2 - 72 hr	 3.321E+05 3.66 E+03
Steam generator in intact loop	
– Primary to secondary leak rate (lb/hr)	52.25[a]
– Iodine partition coefficient	1.0
– Steam released (lb) 0 - 2 hr 2 - 72 hr	 3.321E+05 3.66 E+03
Nuclide data	See Table 15A-4

Note:
a. Equivalent to 150 gpd cooled liquid at 62.4 lb/ft^3.

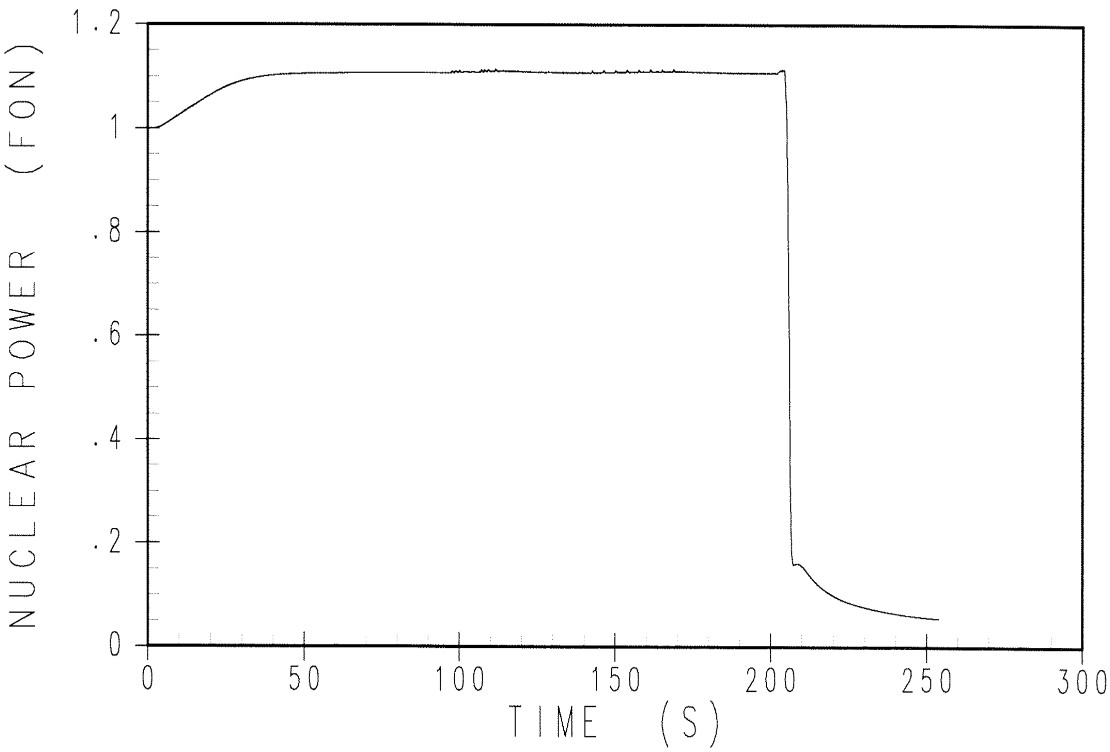


Figure 15.1.2-1
Feedwater Control Valve Malfunction Nuclear Power

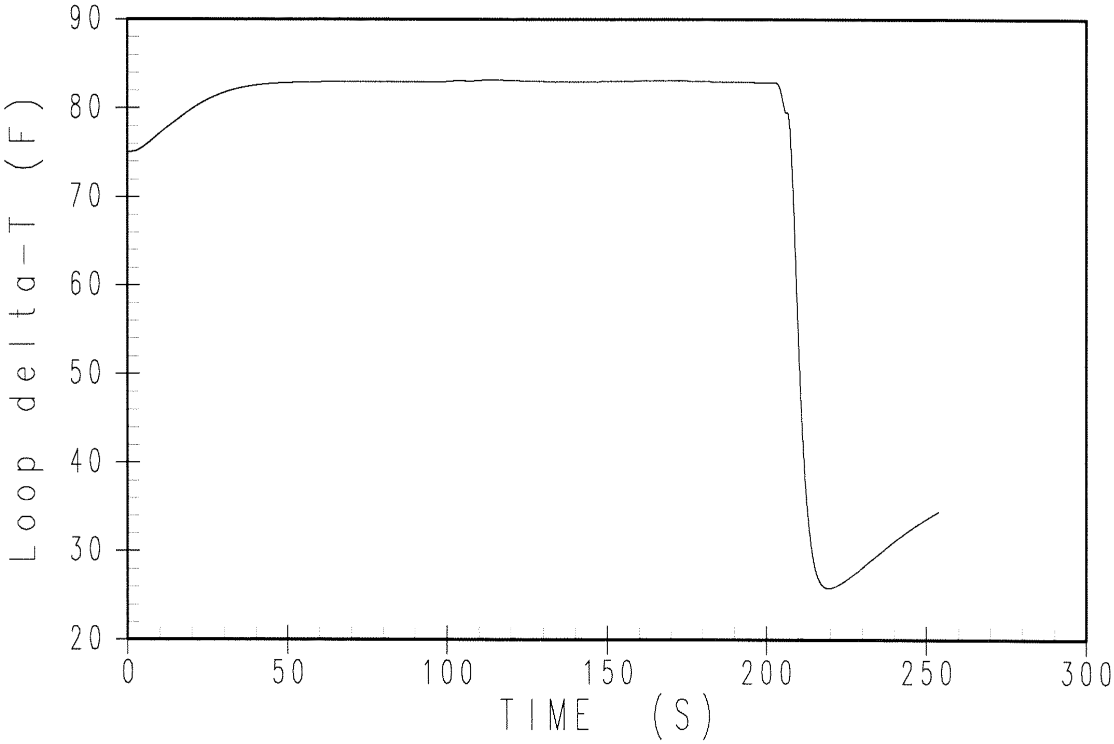


Figure 15.1.2-2
Feedwater Control Valve Malfunction Loop ΔT

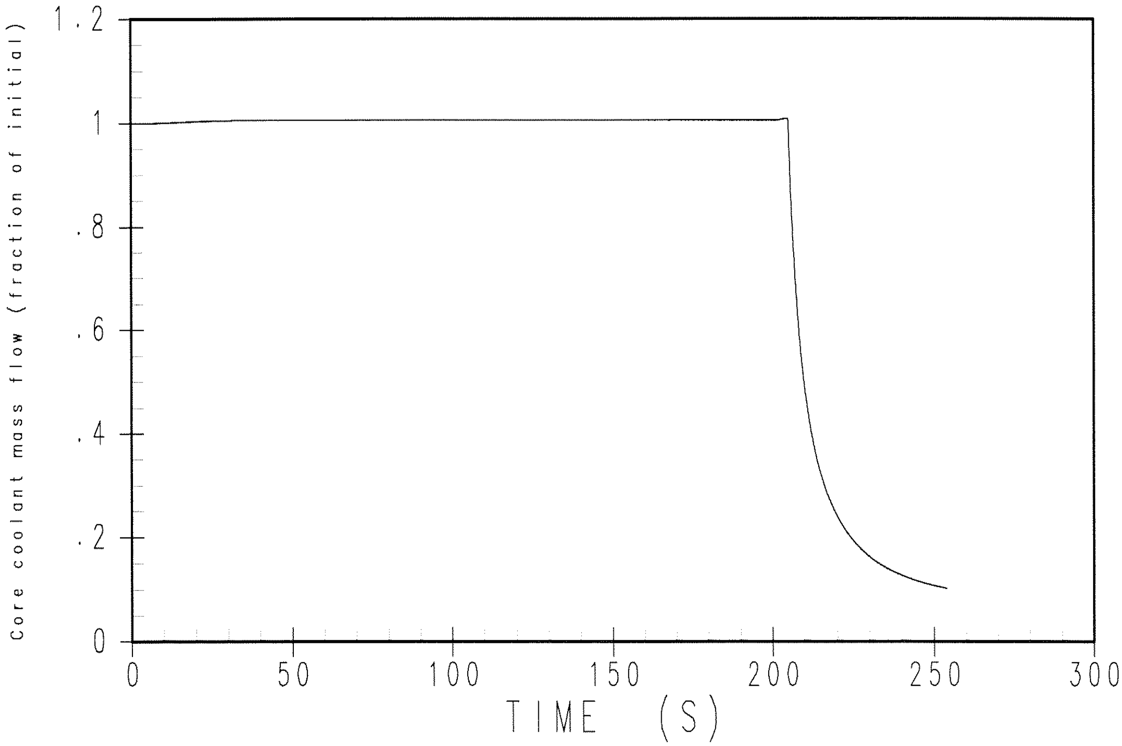


Figure 15.1.2-3
Feedwater Control Valve Malfunction Core Coolant Mass Flow

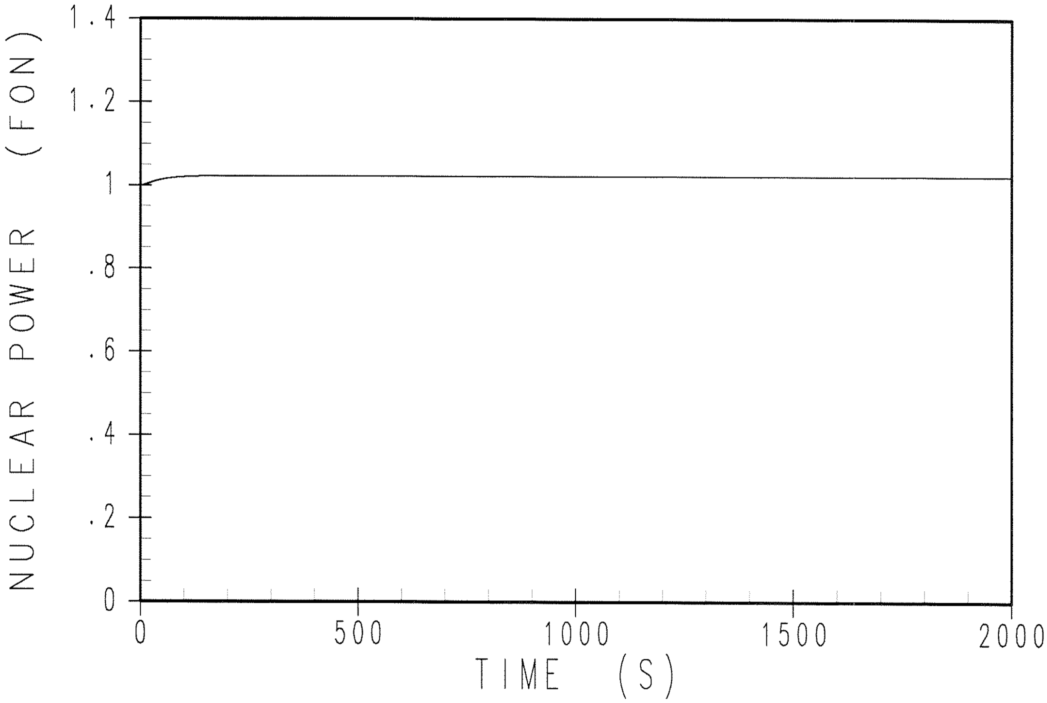


Figure 15.1.3-1
Nuclear Power (Fraction of Nominal) Versus Time for 10-percent Step Load Increase, Manual Control and Minimum Moderator Feedback

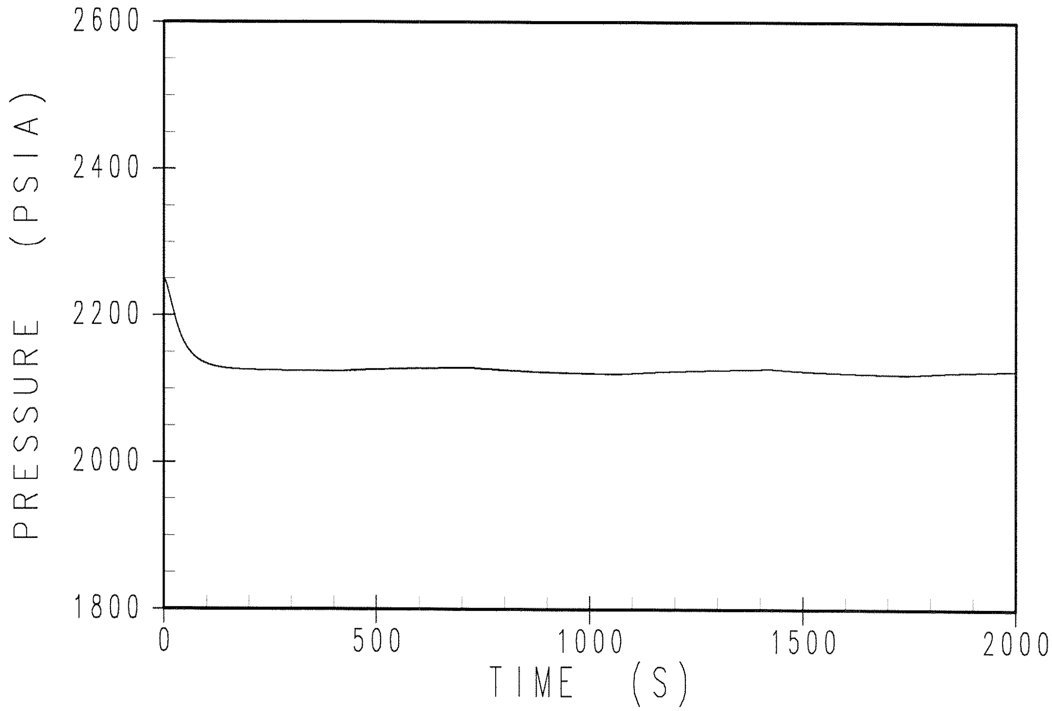


Figure 15.1.3-2
Pressurizer Pressure (psia) Versus Time for 10-percent Step Load Increase, Manual Control and Minimum Moderator Feedback

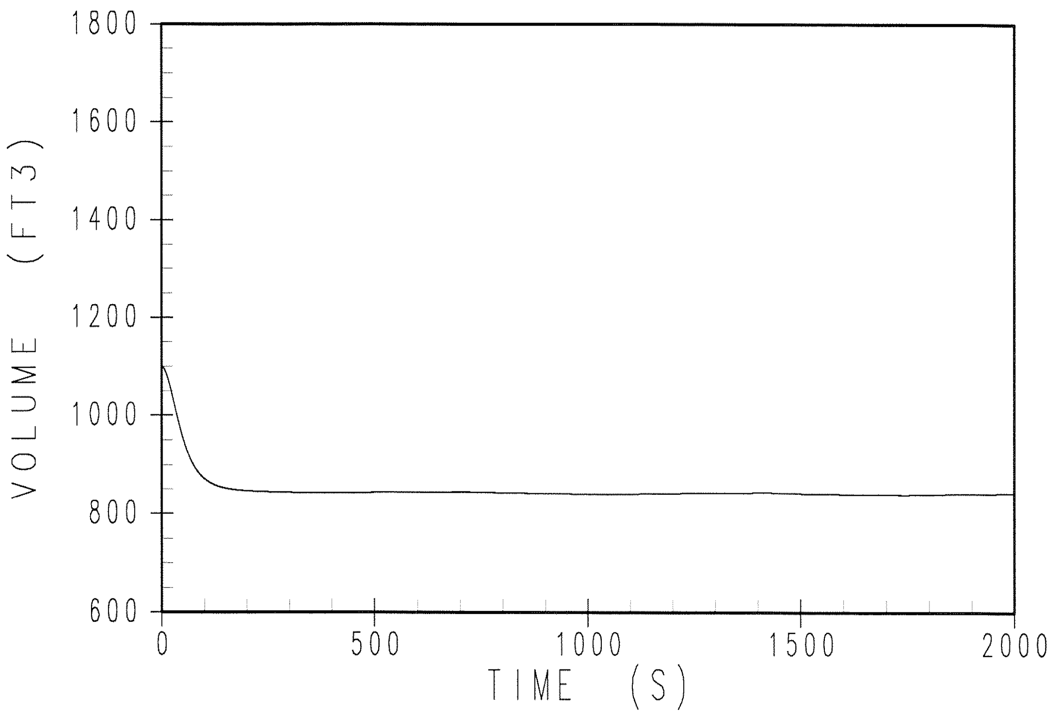


Figure 15.1.3-3
Pressurizer Water Volume (ft^3) Versus Time for 10-percent Step Load Increase, Manual Control and Minimum Moderator Feedback

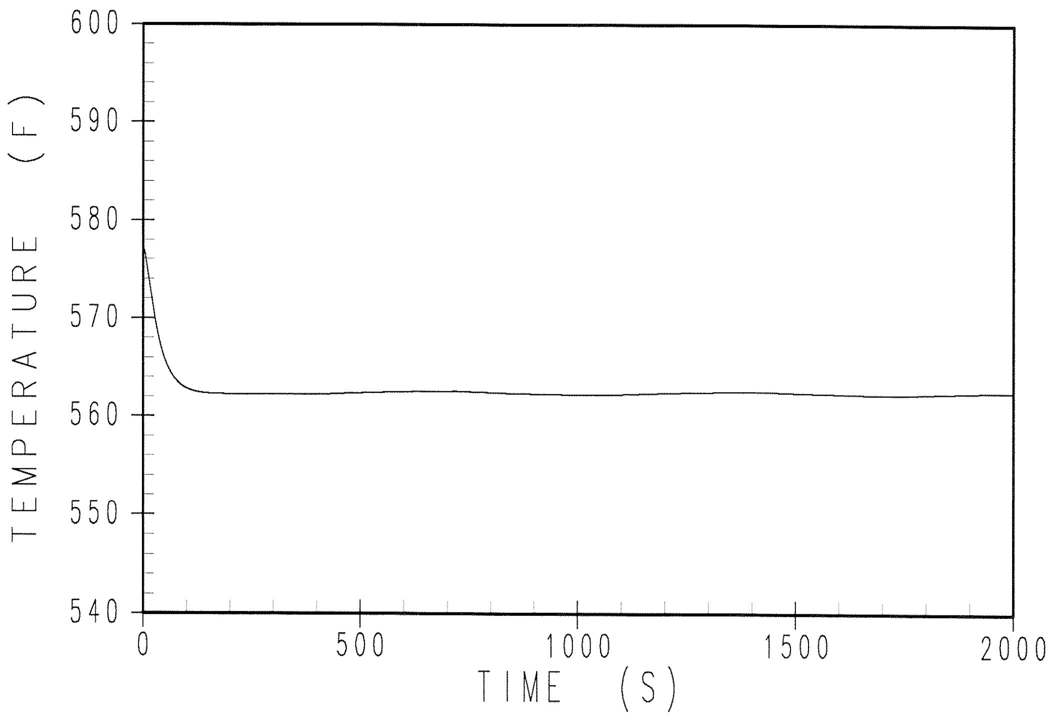


Figure 15.1.3-4
Core Average Temperature (°F) Versus Time for 10-percent Step Load Increase, Manual Control and Minimum Moderator Feedback

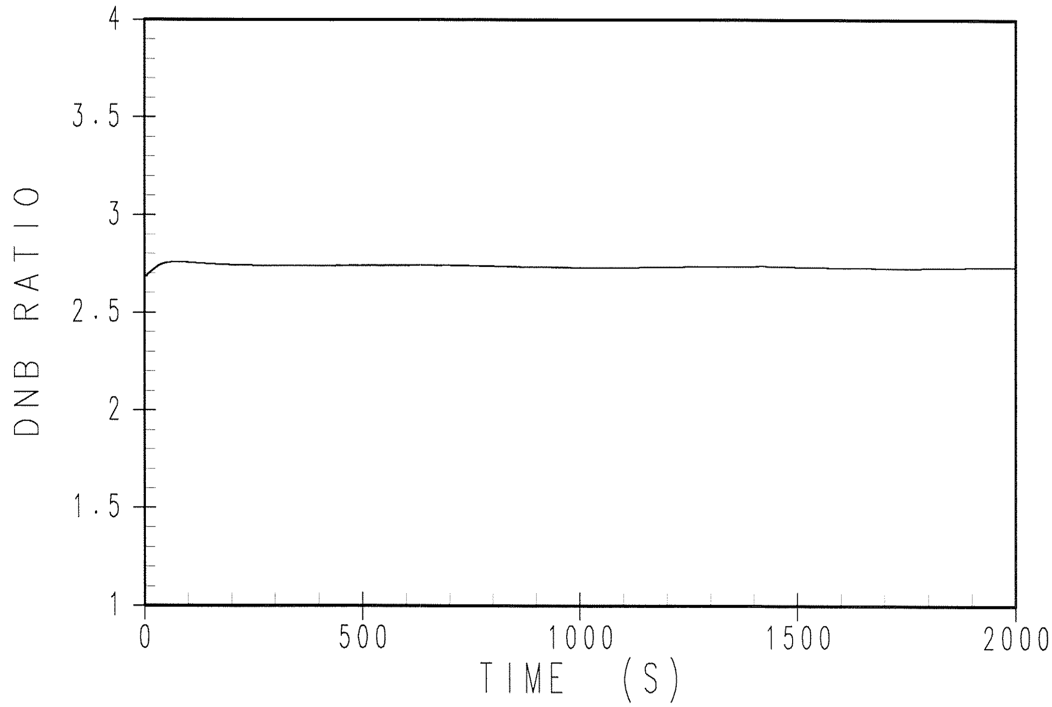


Figure 15.1.3-5
DNBR Versus Time for 10-percent Step Load Increase, Manual Control and Minimum Moderator Feedback

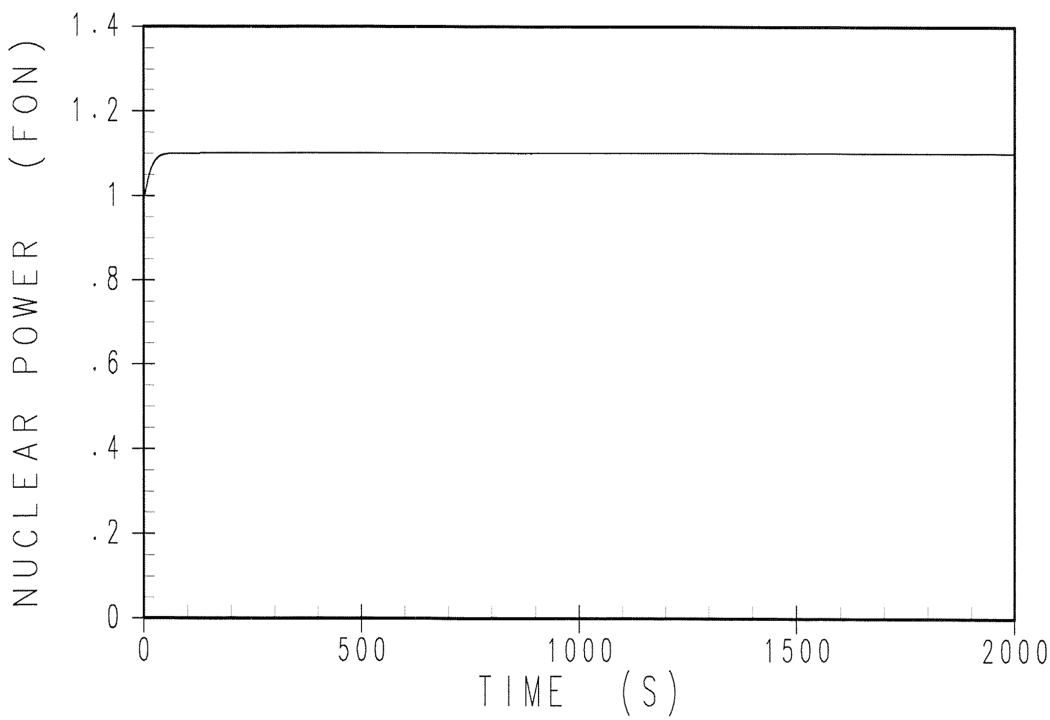


Figure 15.1.3-6
Nuclear Power (Fraction of Nominal) Versus Time for 10-percent Step Load Increase, Manual Control and Maximum Moderator Feedback

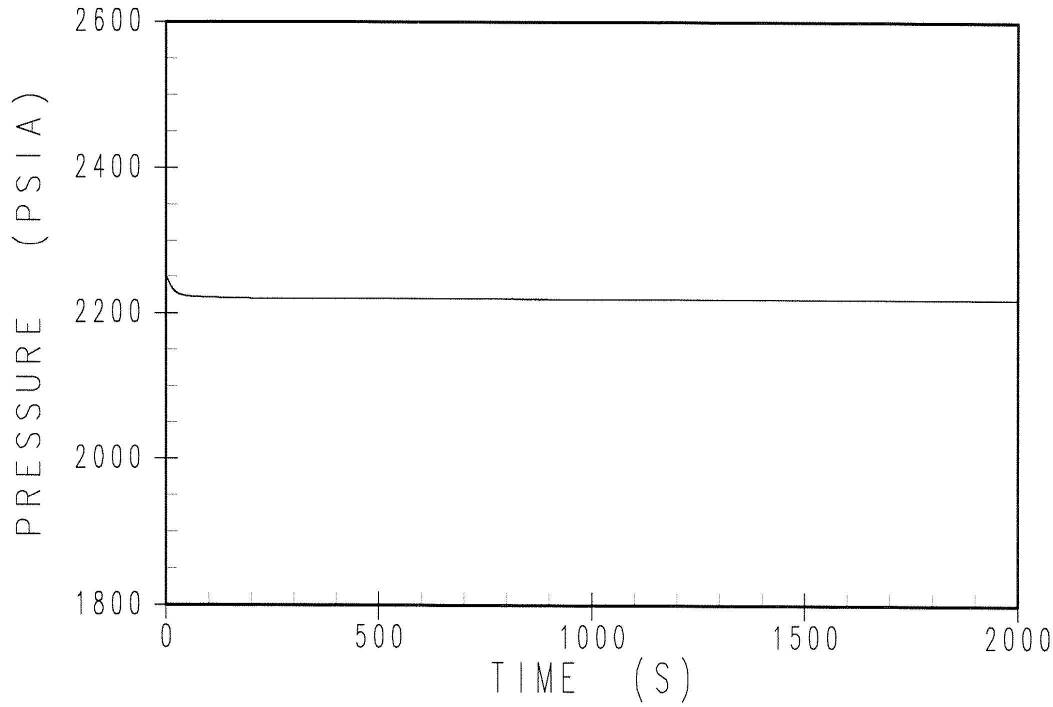


Figure 15.1.3-7
Pressurizer Pressure (psia) Versus Time for 10-percent Step Load Increase, Manual Control and Maximum Moderator Feedback

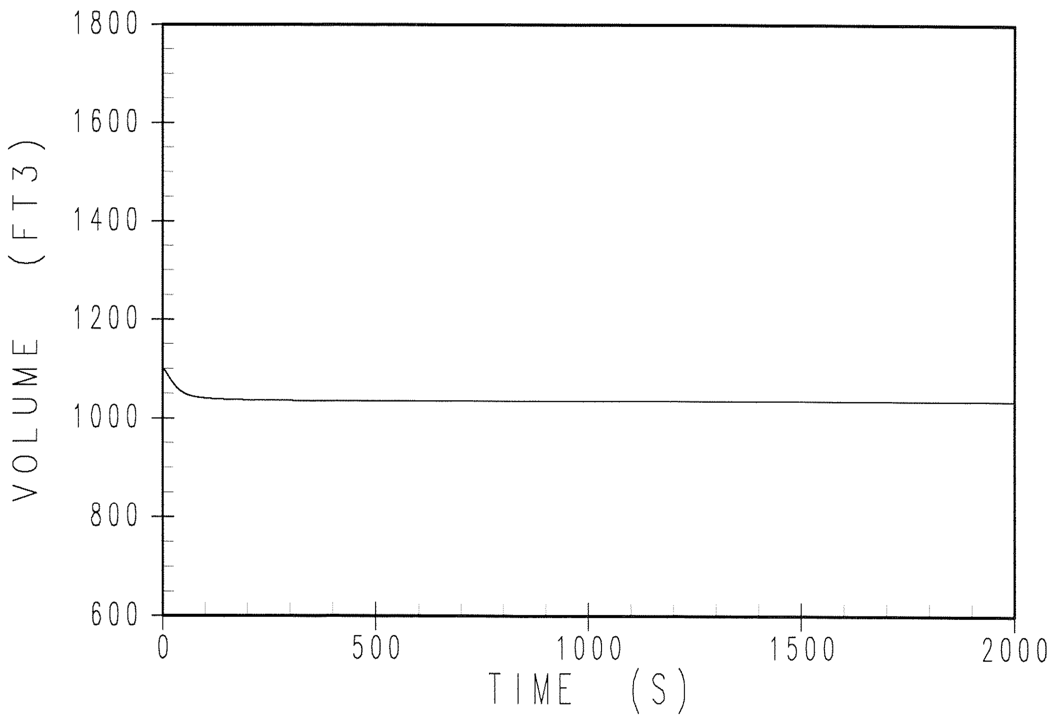


Figure 15.1.3-8
Pressurizer Water Volume (ft^3) Versus Time for 10-percent Step Load Increase, Manual Control and Maximum Moderator Feedback

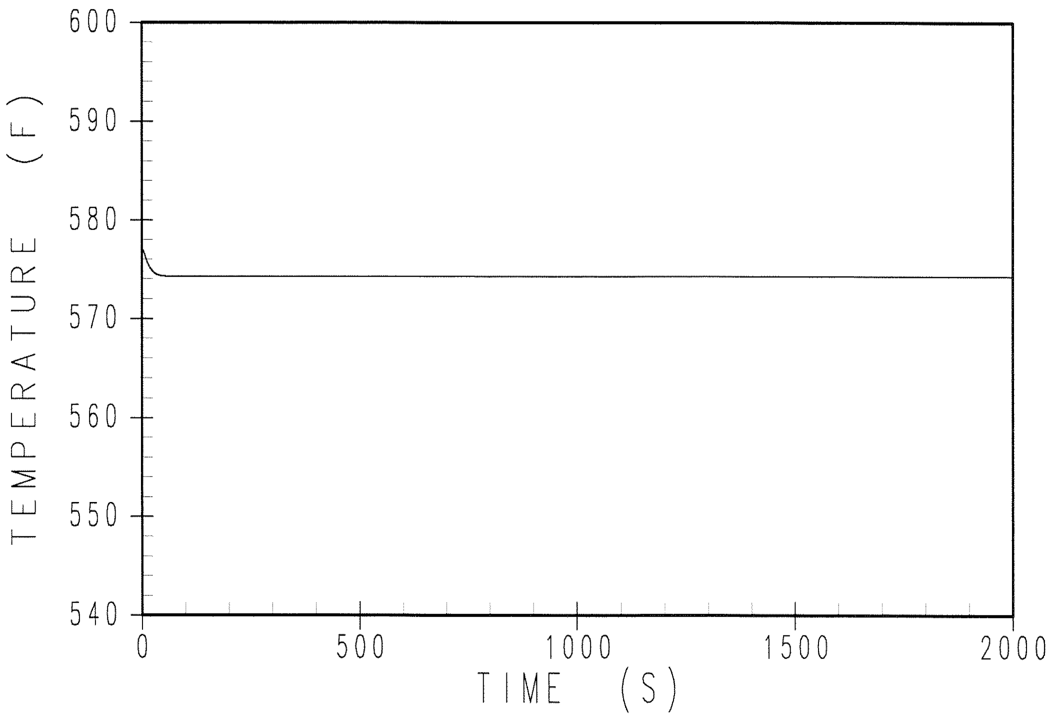


Figure 15.1.3-9
Core Average Temperature (°F) Versus Time for 10-percent Step Load Increase, Manual Control and Maximum Moderator Feedback

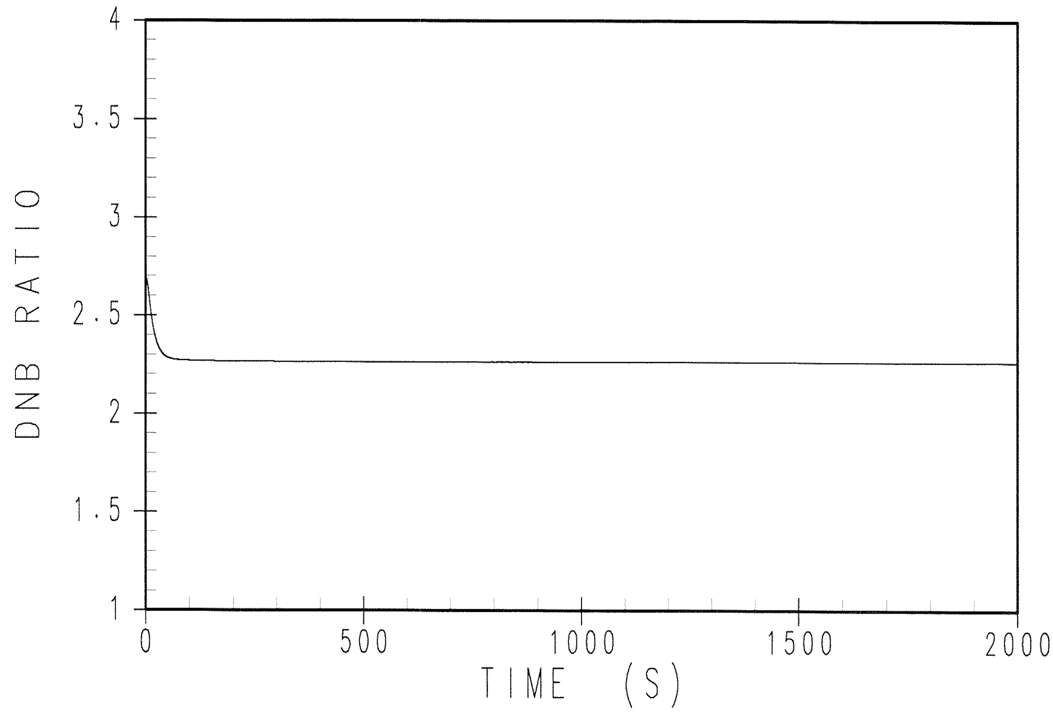


Figure 15.1.3-10
DNBR Versus Time for 10-percent Step Load Increase, Manual Control and Maximum Moderator Feedback

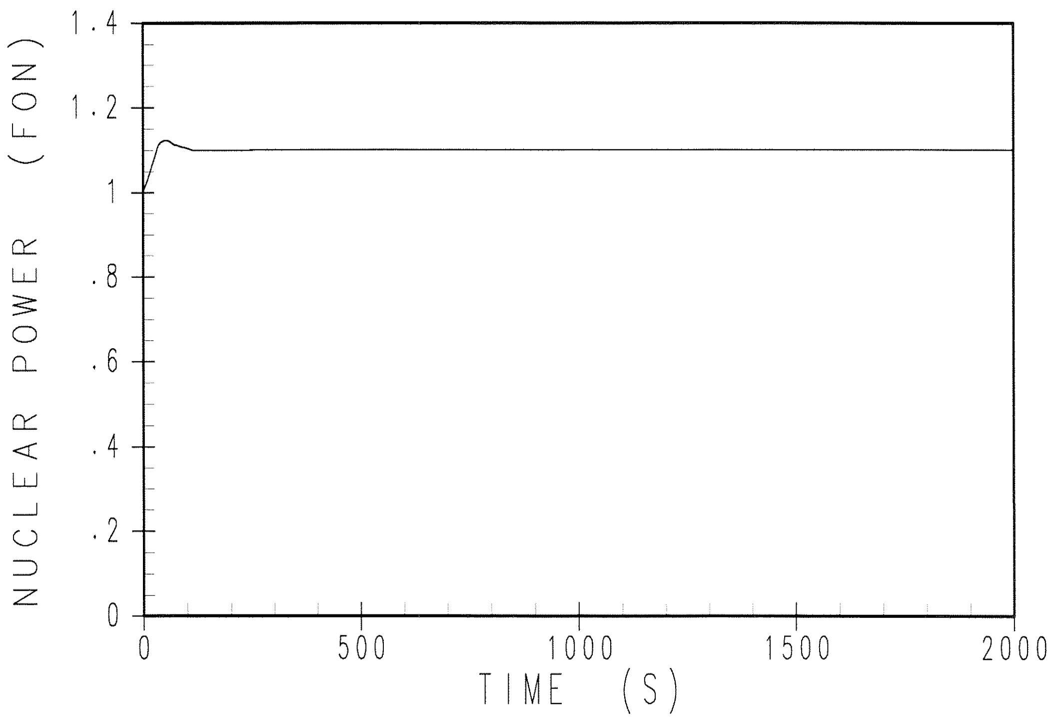


Figure 15.1.3-11
Nuclear Power (Fraction of Nominal) Versus Time for 10-percent Step Load Increase, Automatic Control and Minimum Moderator Feedback

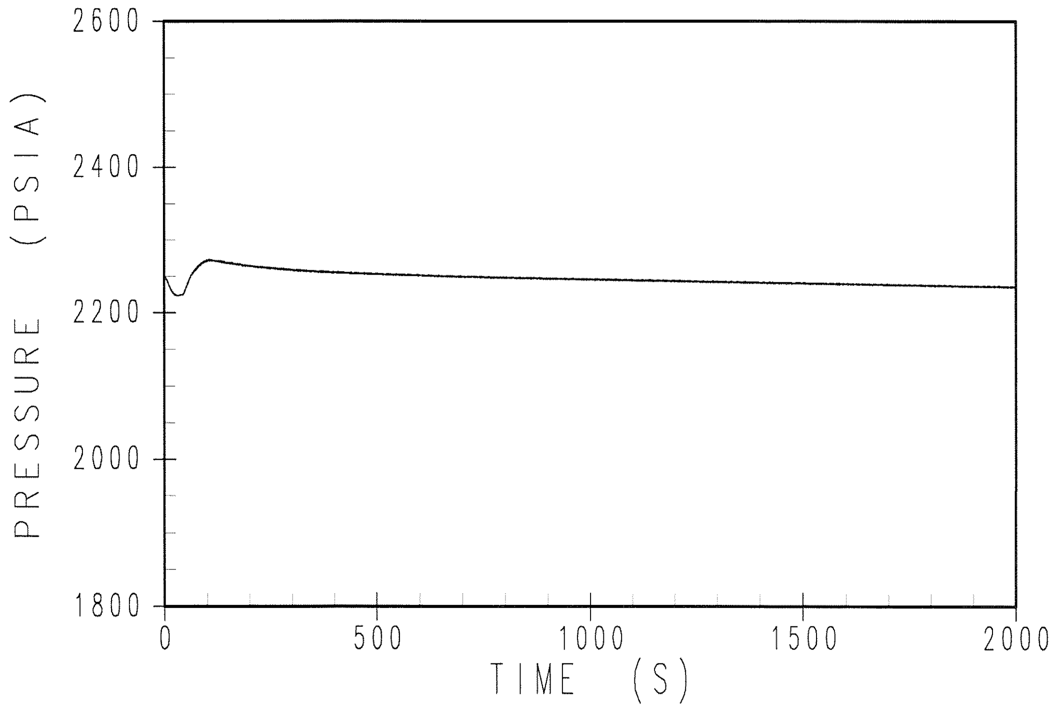


Figure 15.1.3-12
Pressurizer Pressure (psia) Versus Time for 10-percent Step Load Increase, Automatic Control and Minimum Moderator Feedback

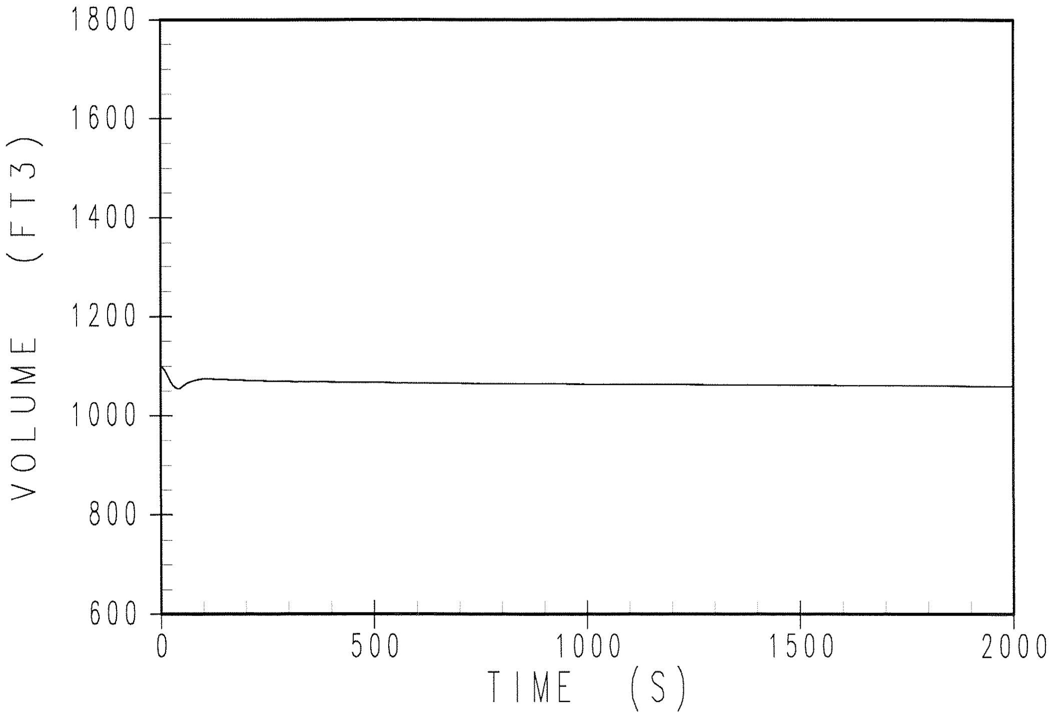


Figure 15.1.3-13
Pressurizer Water Volume (ft^3) Versus Time for 10-percent Step Load Increase, Automatic Control and Minimum Moderator Feedback

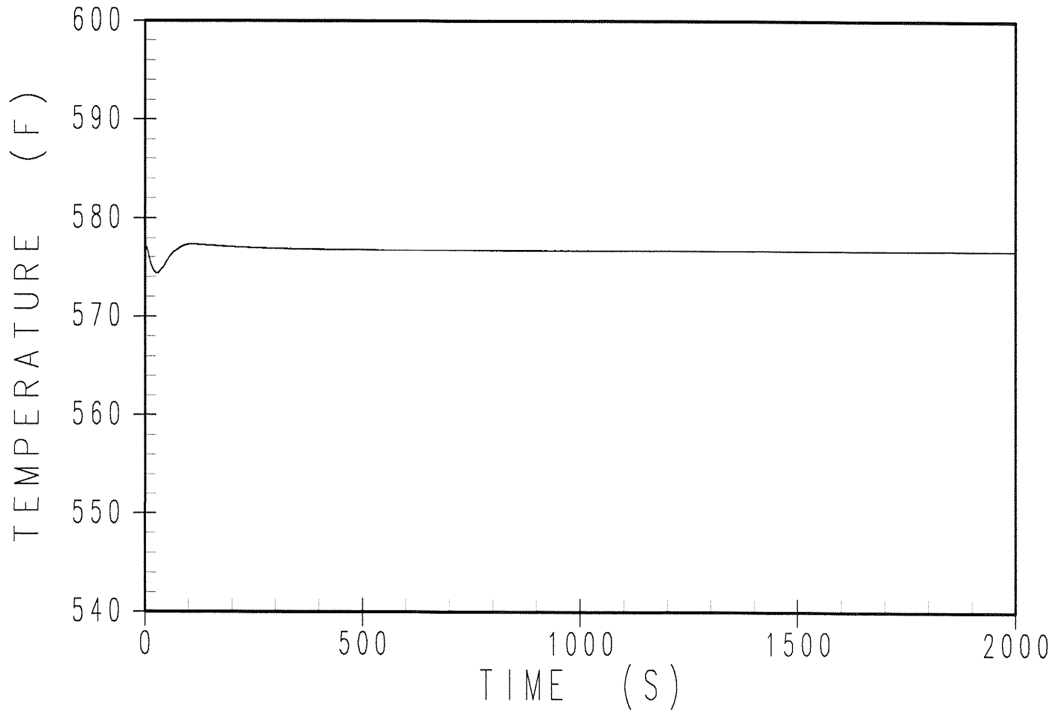


Figure 15.1.3-14
Core Average Temperature (°F) Versus Time for 10-percent Step Load Increase, Automatic Control and Minimum Moderator Feedback

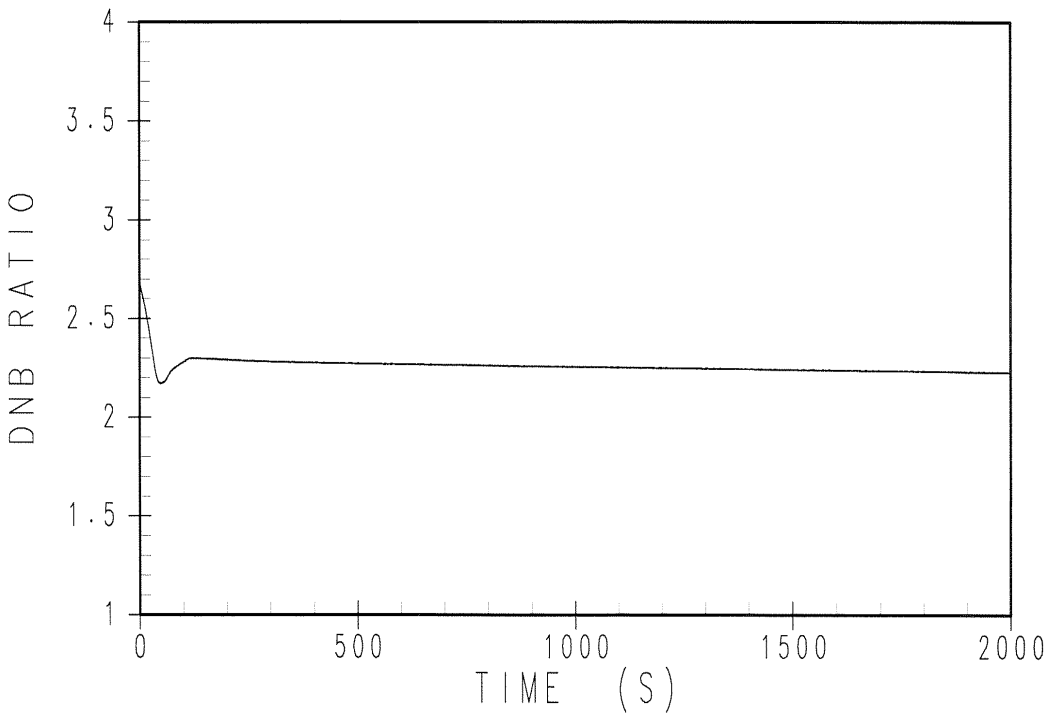


Figure 15.1.3-15
DNBR Versus Time for 10-percent Step Load Increase, Automatic Control and Minimum Moderator Feedback

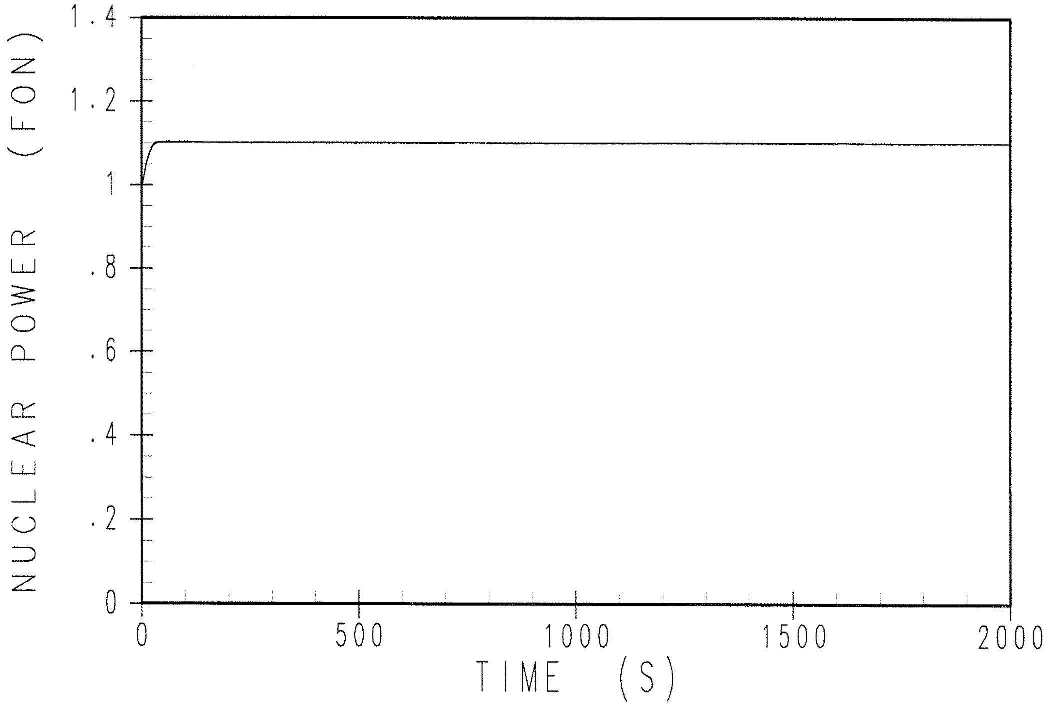


Figure 15.1.3-16
Nuclear Power (Fraction of Nominal) Versus Time for 10-percent Step Load Increase, Automatic Control and Maximum Moderator Feedback

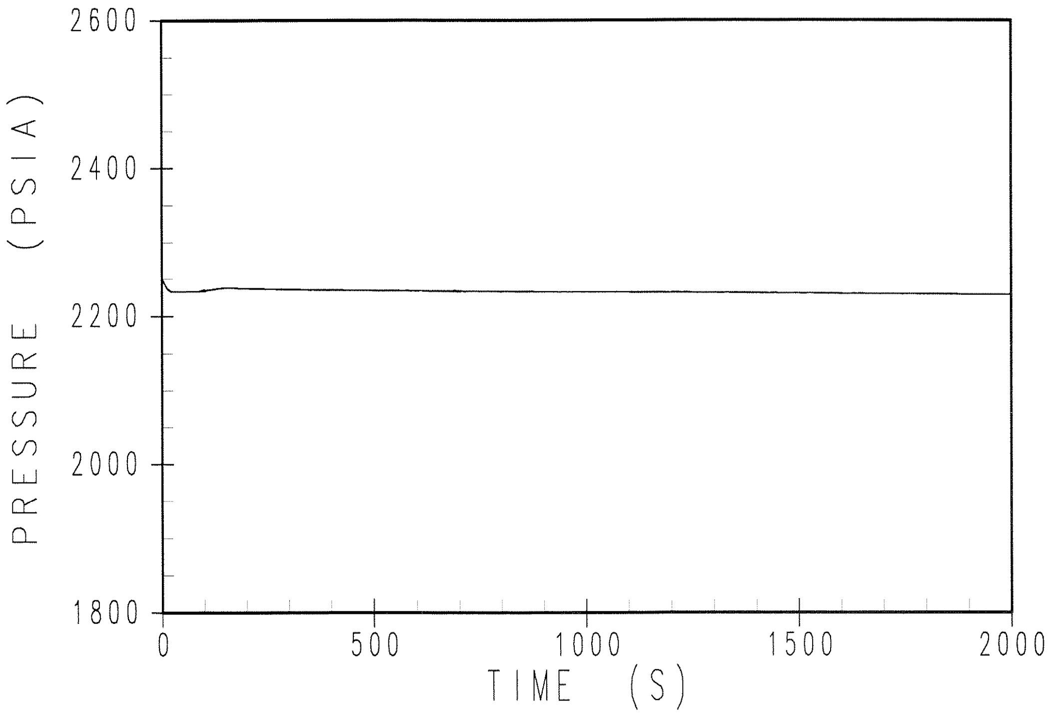


Figure 15.1.3-17
Pressurizer Pressure (psia) Versus Time for 10-percent Step Load Increase, Automatic Control and Maximum Moderator Feedback

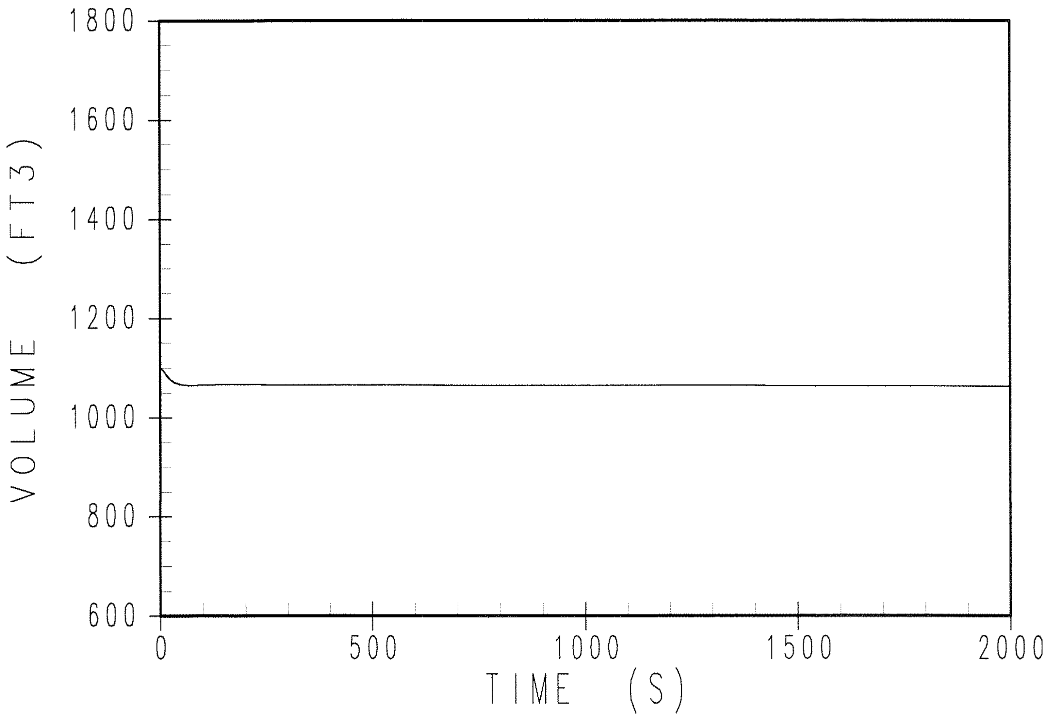


Figure 15.1.3-18
Pressurizer Water Volume (ft^3) Versus Time for 10-percent Step Load Increase, Automatic Control and Maximum Moderator Feedback

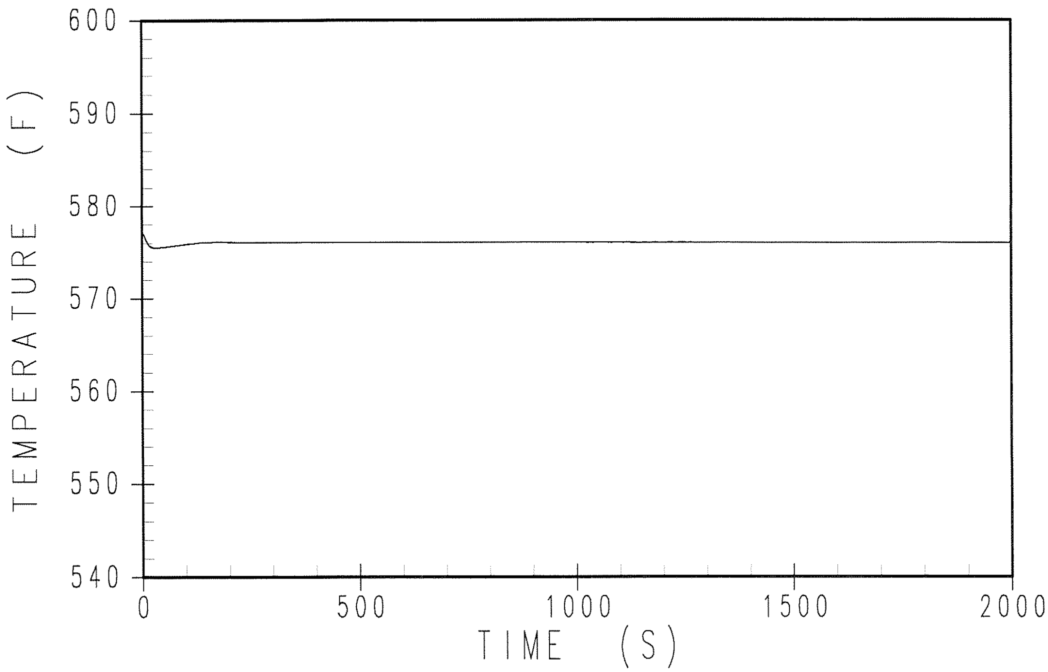


Figure 15.1.3-19
Core Average Temperature (°F) Versus Time for 10-percent Step Load Increase, Automatic Control and Maximum Moderator Feedback

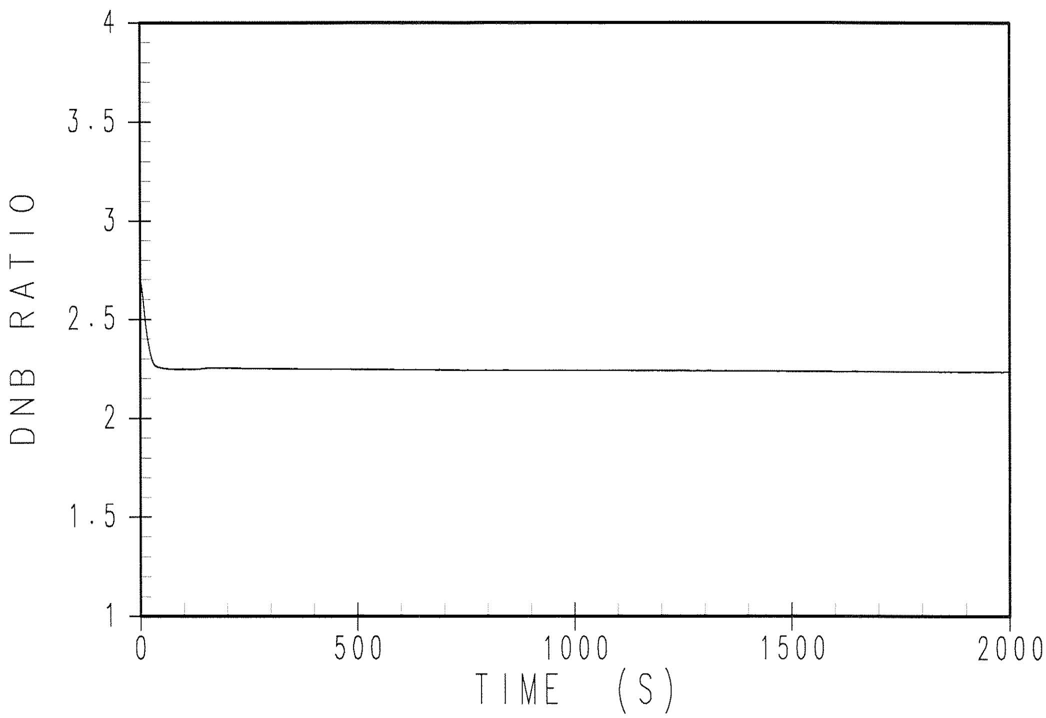


Figure 15.1.3-20
DNBR Versus Time for 10-percent Step Load Increase, Automatic Control and Maximum Moderator Feedback

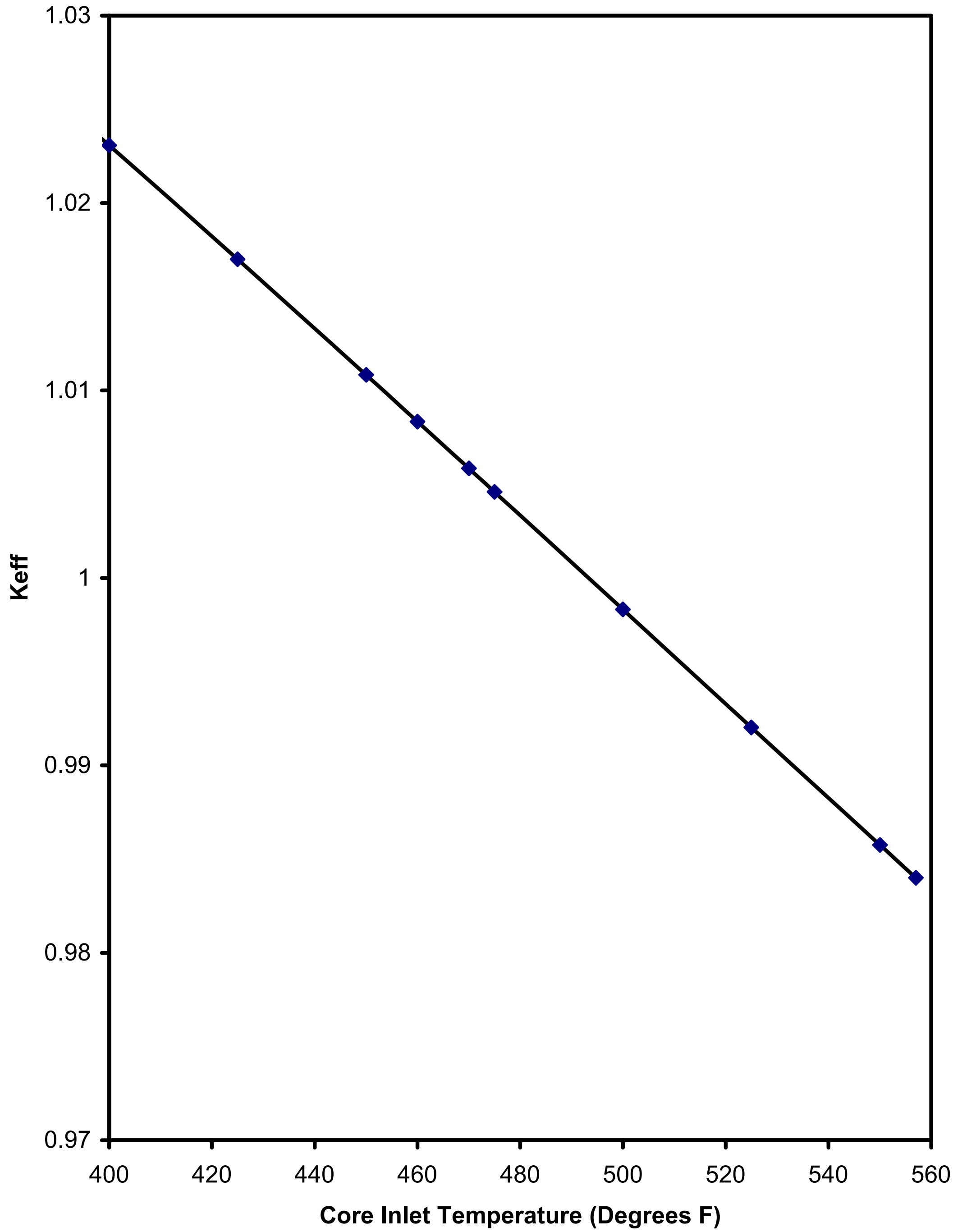


Figure 15.1.4-1
K_{eff} Versus Core Inlet Temperature
Steam Line Break Events

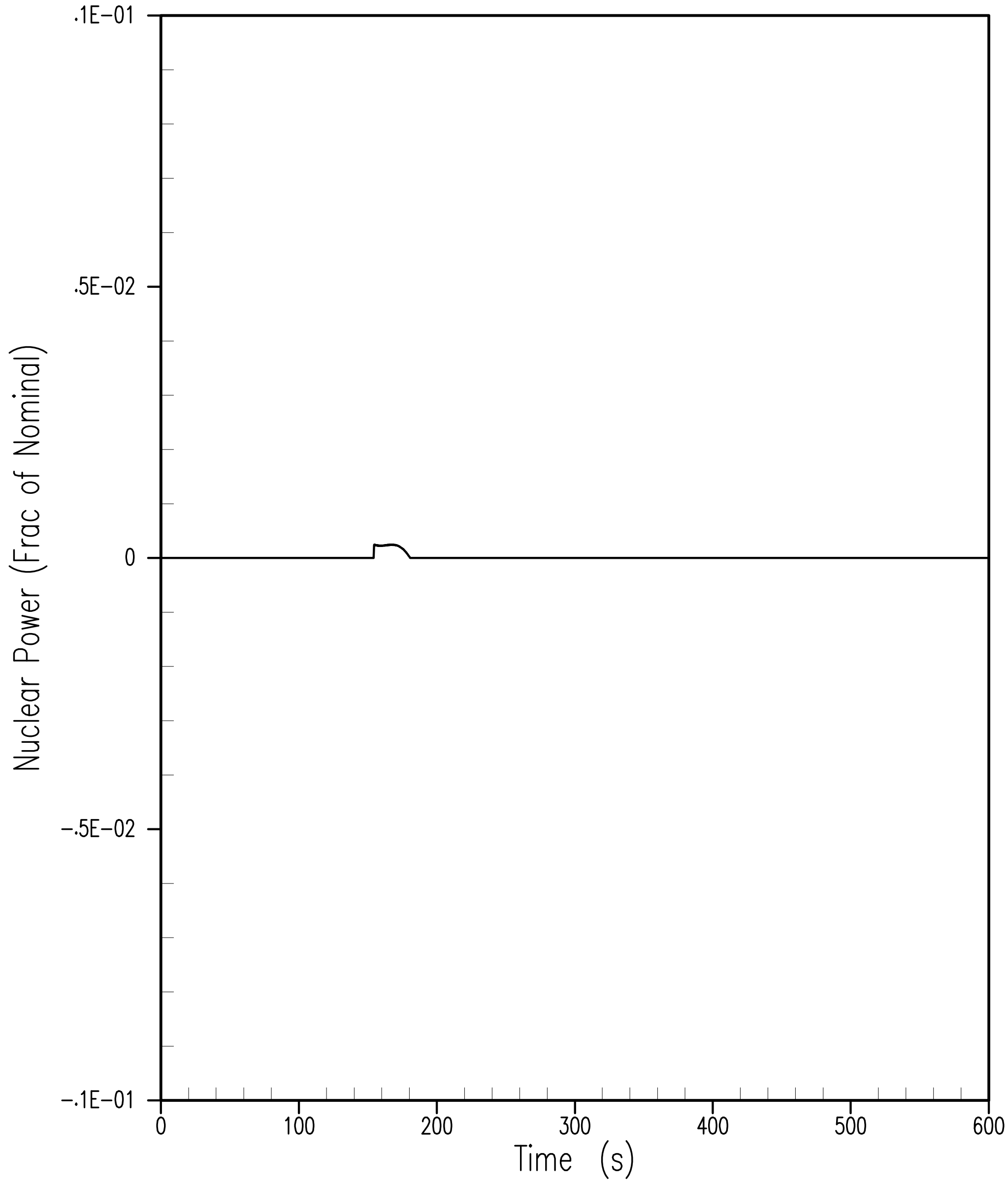


Figure 15.1.4-2
Nuclear Power Transient
Inadvertent Opening of a Steam Generator Relief or Safety Valve

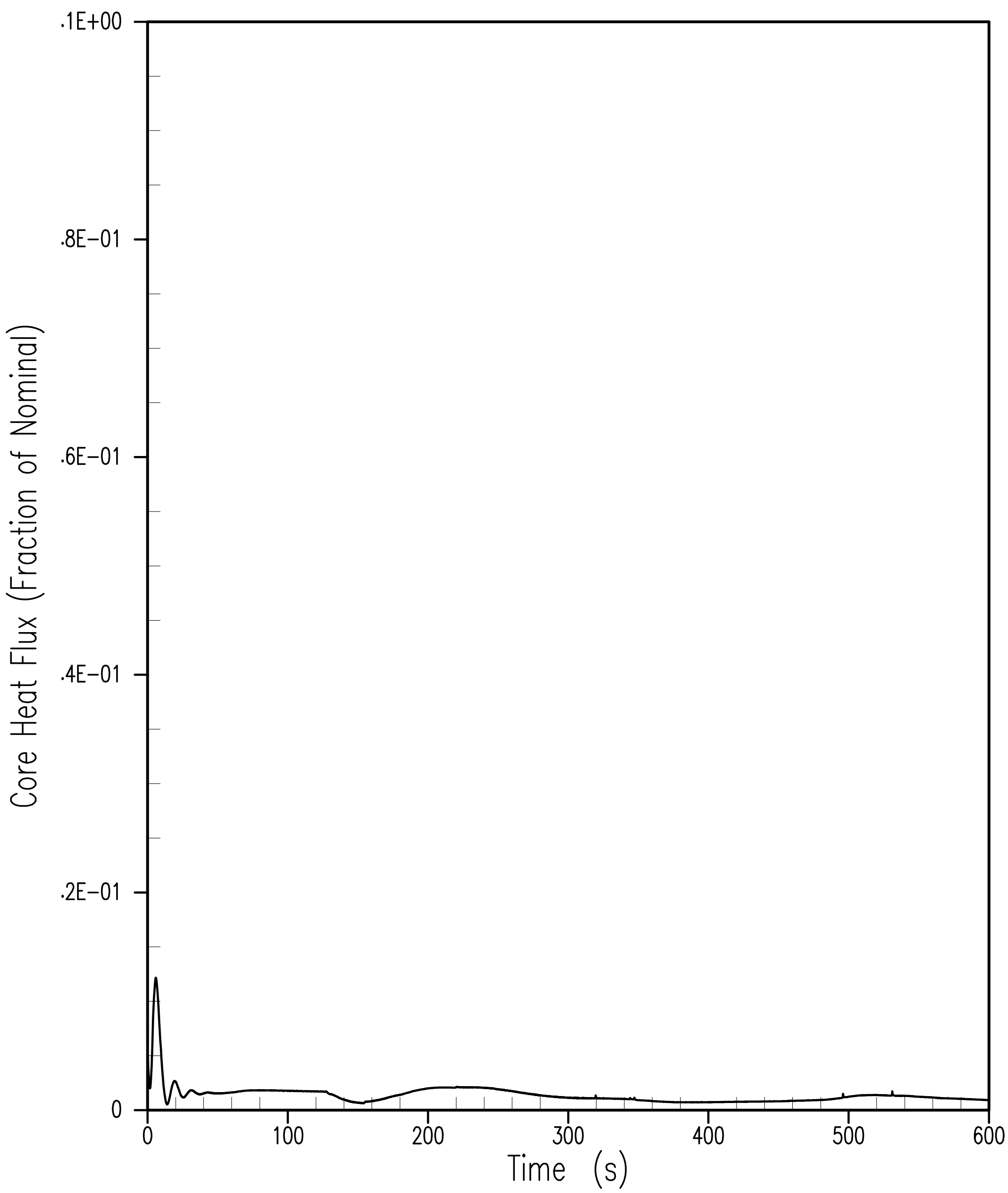


Figure 15.1.4-3
Core Heat Flux Transient
Inadvertent Opening of a Steam Generator Relief or Safety Valve

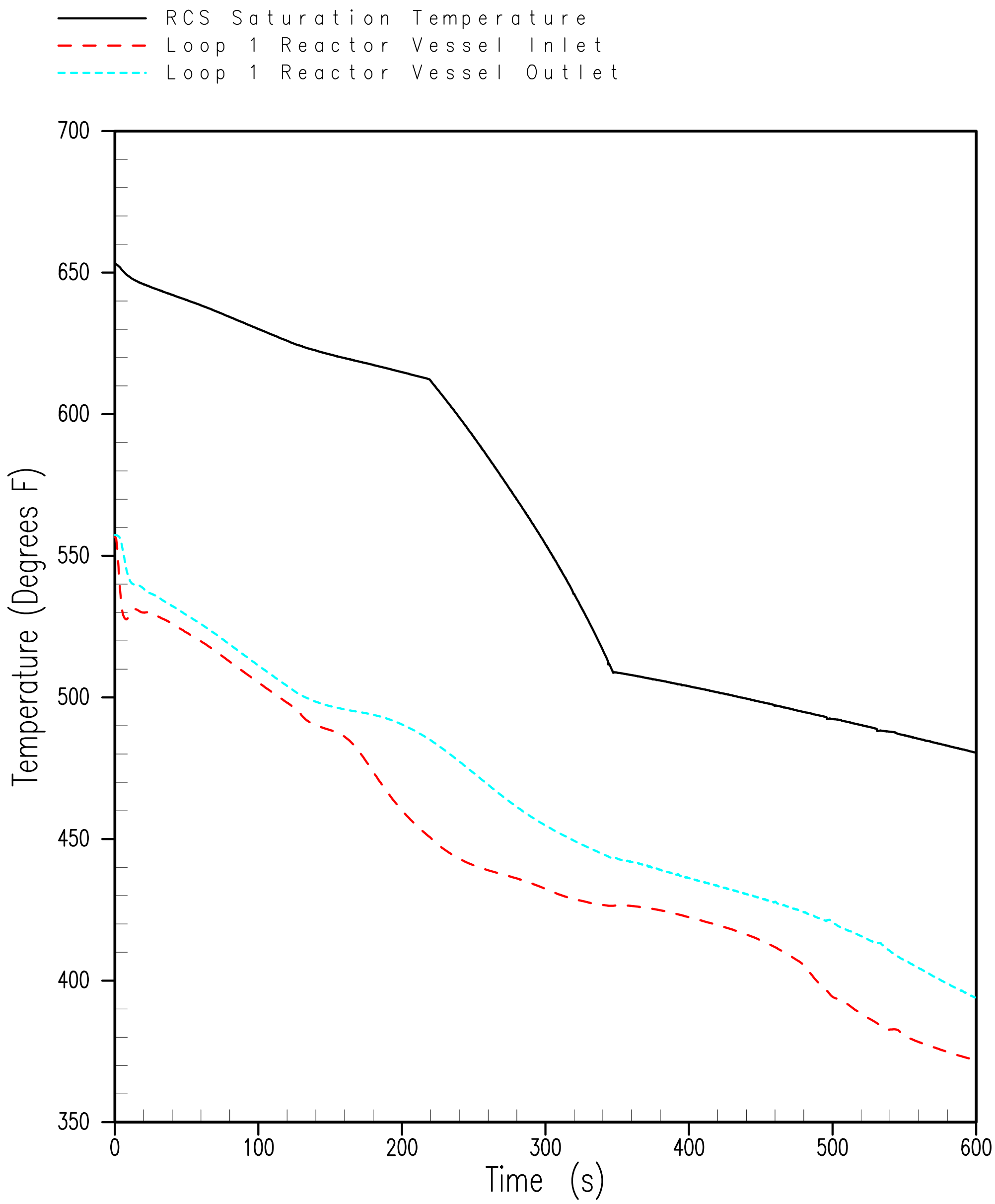


Figure 15.1.4-4
Loop 1 Reactor Coolant Temperatures
Inadvertent Opening of a Steam Generator Relief or Safety Valve

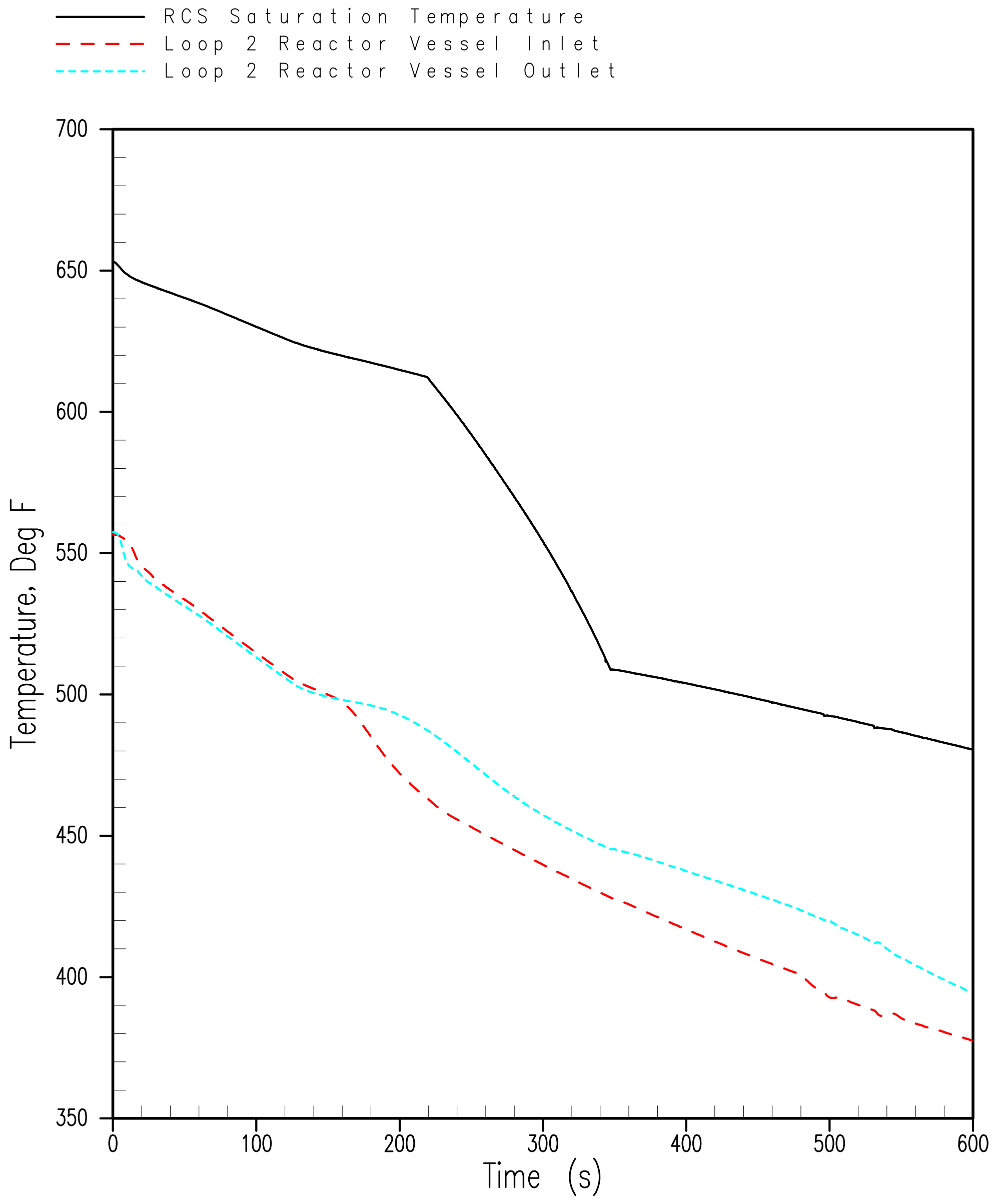


Figure 15.1.4-5
Loop 2 (Faulted Loop) Reactor Coolant Temperatures
Inadvertent Opening of a Steam Generator Relief or Safety Valve

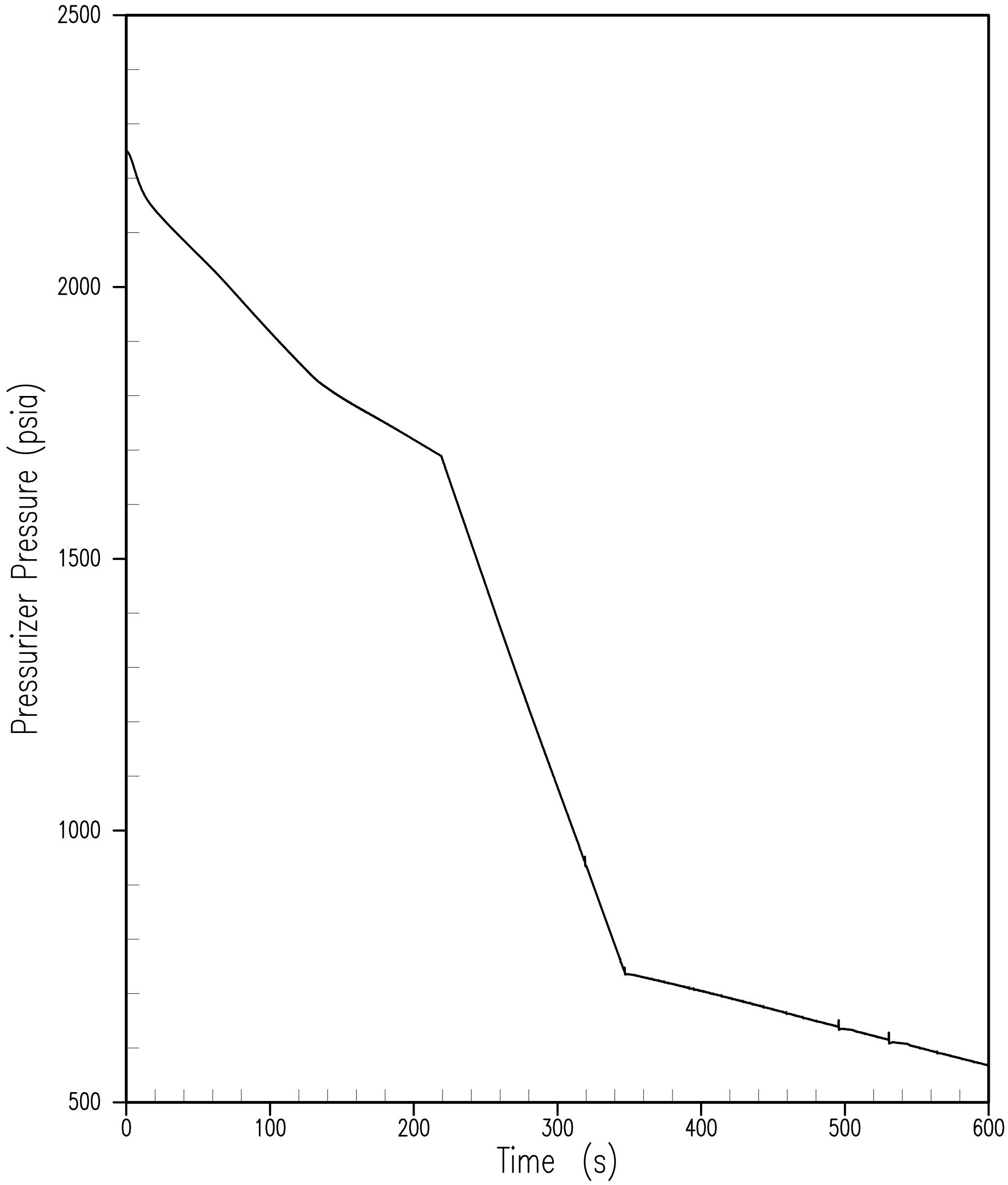


Figure 15.1.4-6
Reactor Coolant System Pressure Transient
Inadvertent Opening of a Steam Generator Relief or Safety Valve

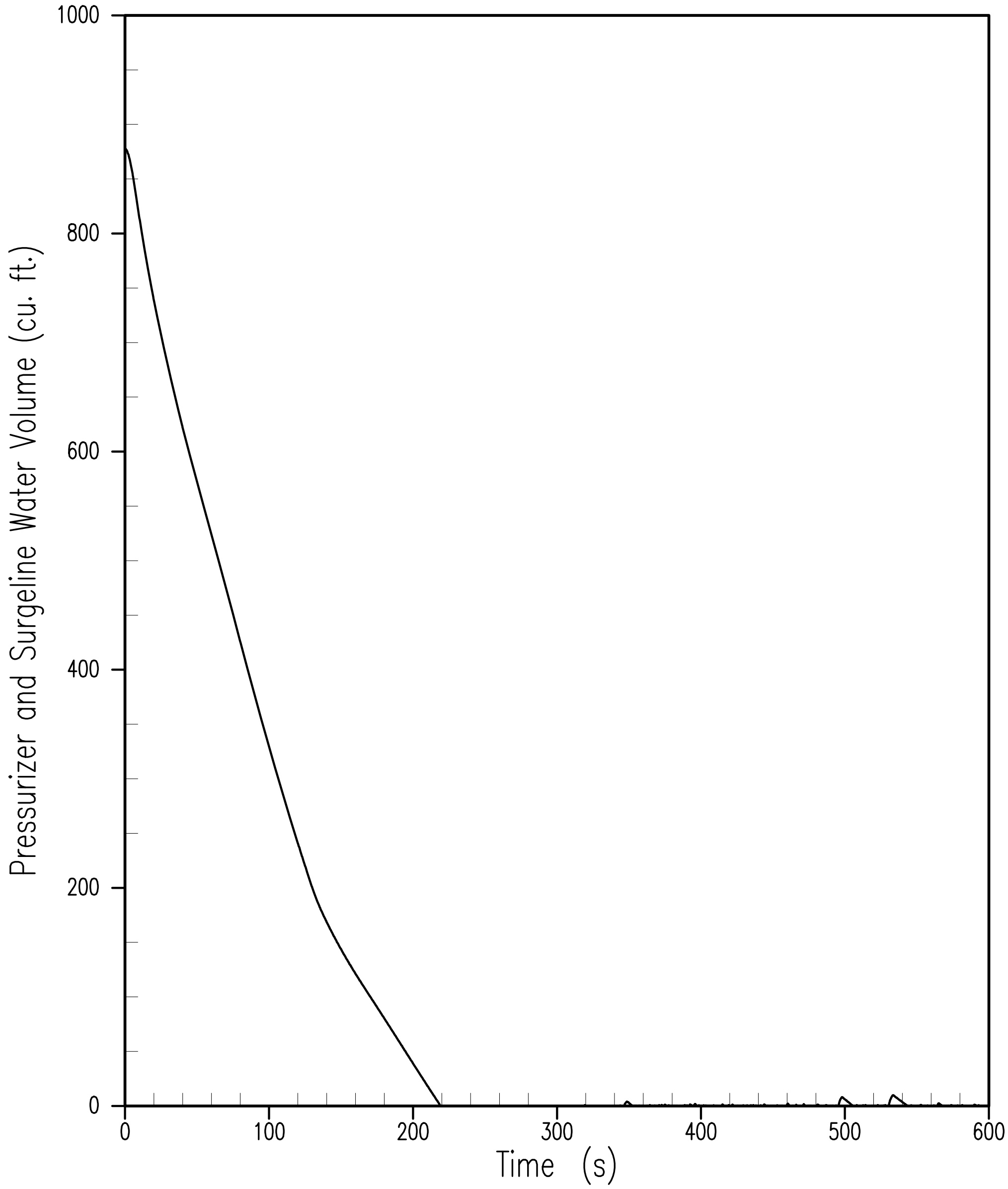


Figure 15.1.4-7
Pressurizer Water Volume Transient
Inadvertent Opening of a Steam Generator Relief or Safety Valve

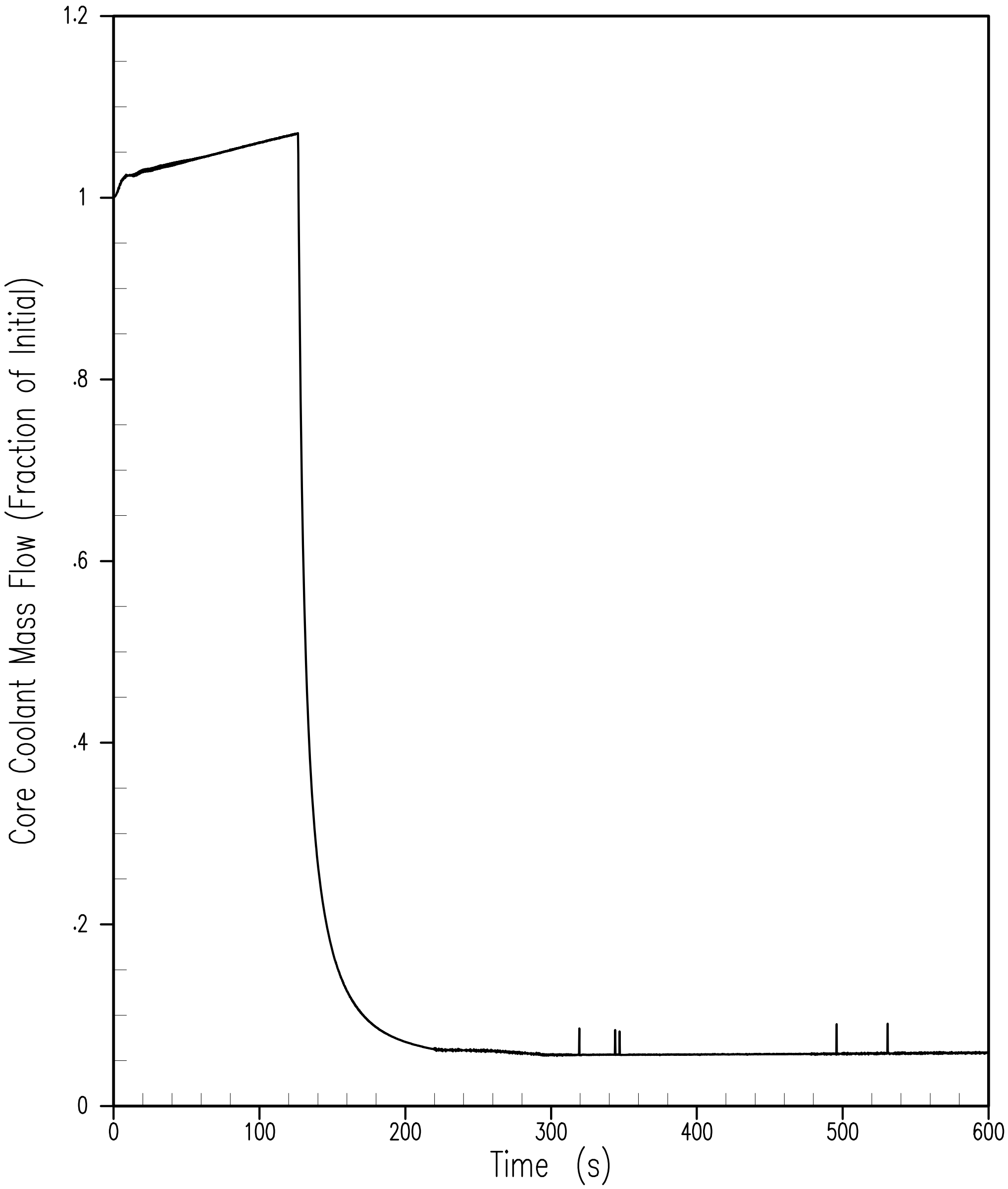


Figure 15.1.4-8
Core Flow Transient
Inadvertent Opening of a Steam Generator Relief or Safety Valve

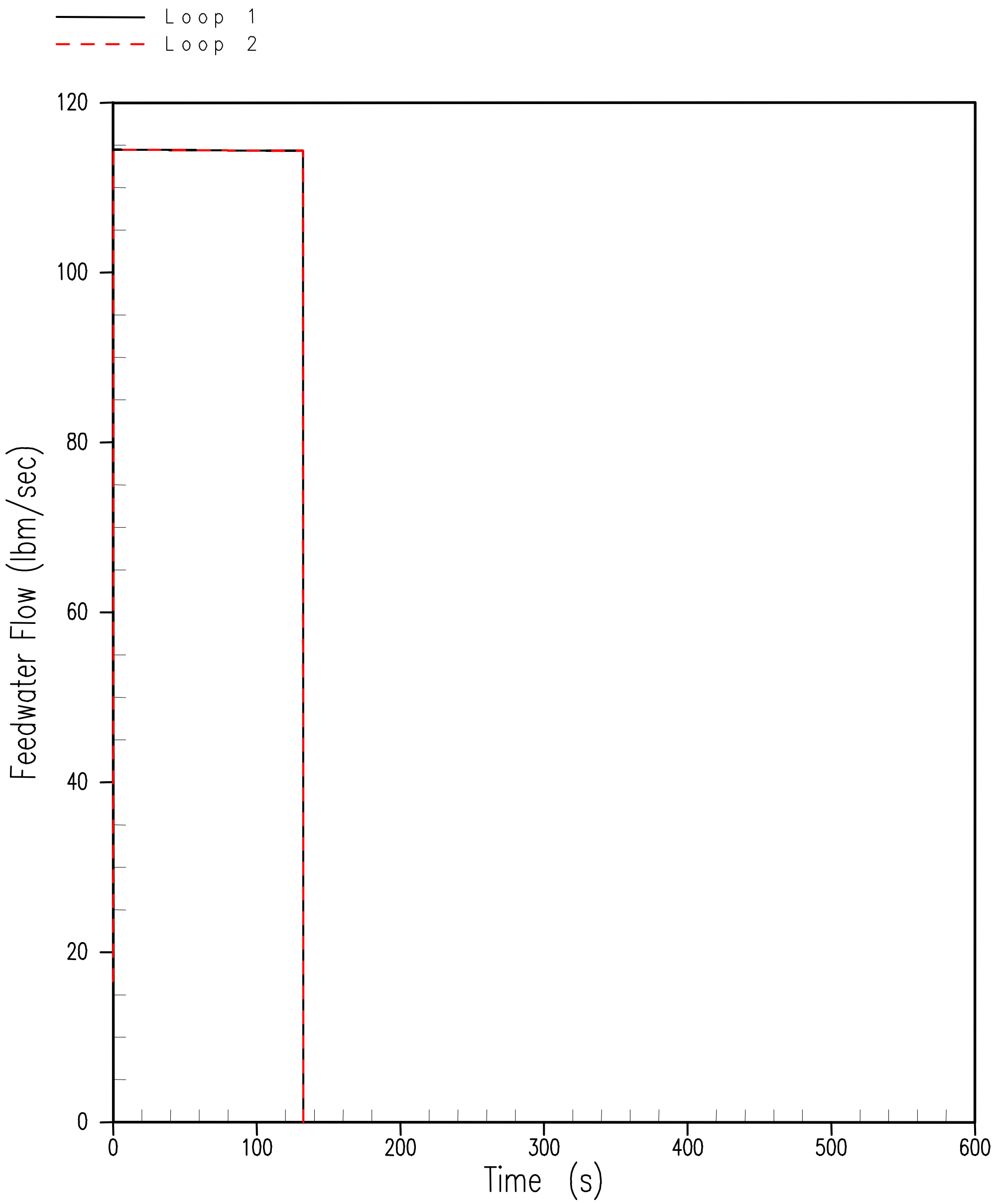


Figure 15.1.4-9
Feedwater Flow Transient
Inadvertent Opening of a Steam Generator Relief or Safety Valve

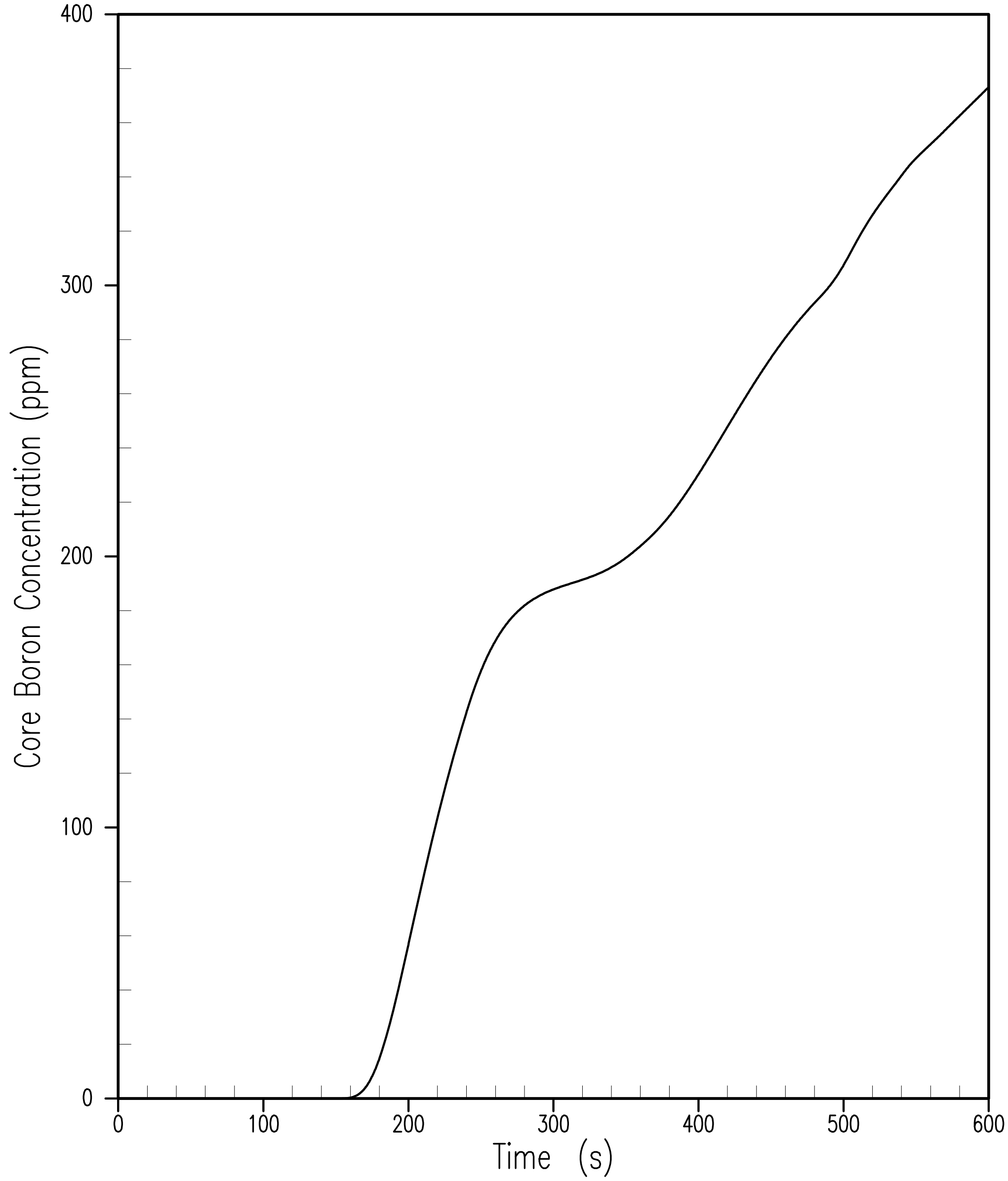


Figure 15.1.4-10
Core Boron Transient
Inadvertent Opening of a Steam Generator Relief or Safety Valve

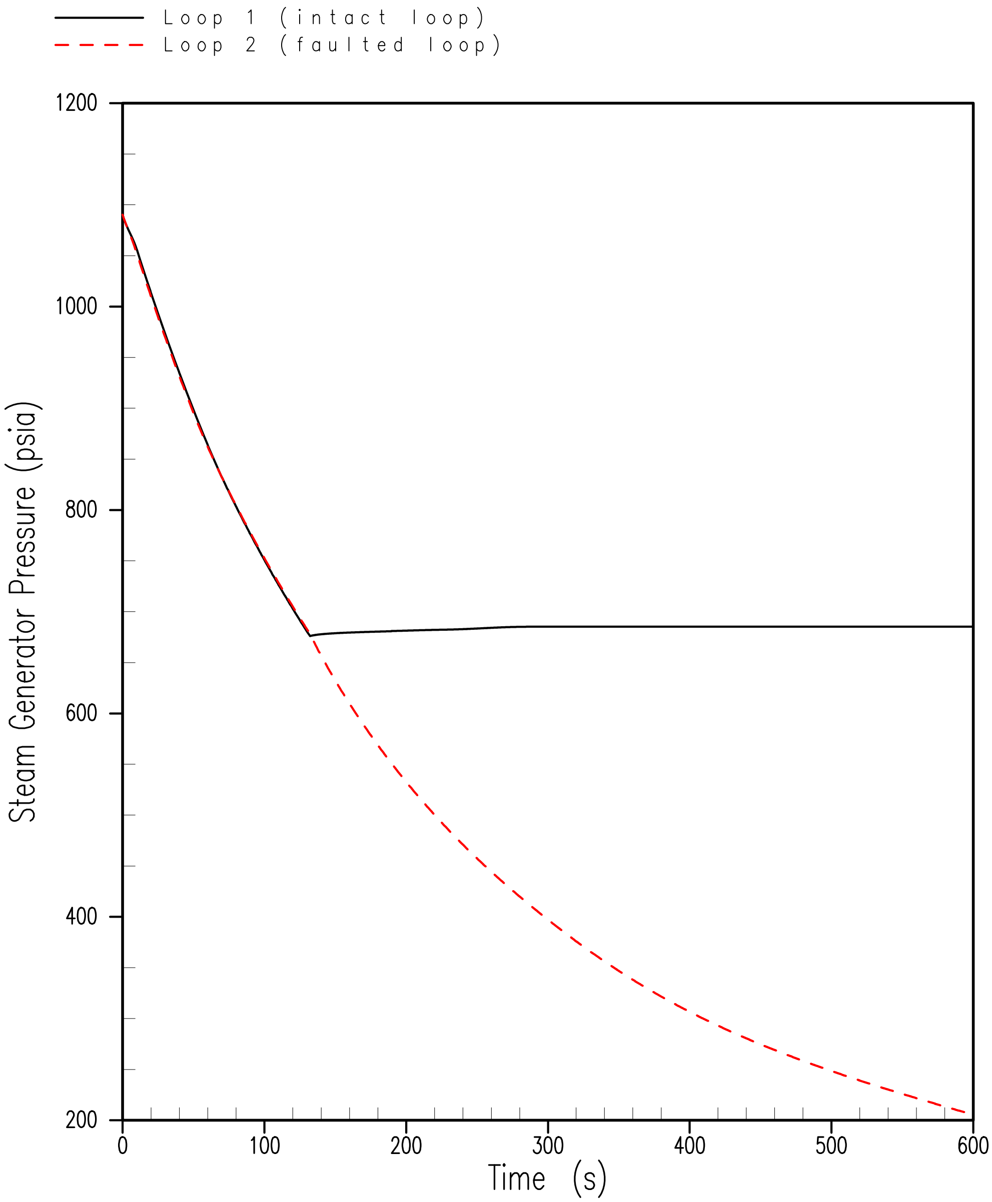


Figure 15.1.4-11
Steam Pressure Transient
Inadvertent Opening of a Steam Generator Relief or Safety Valve

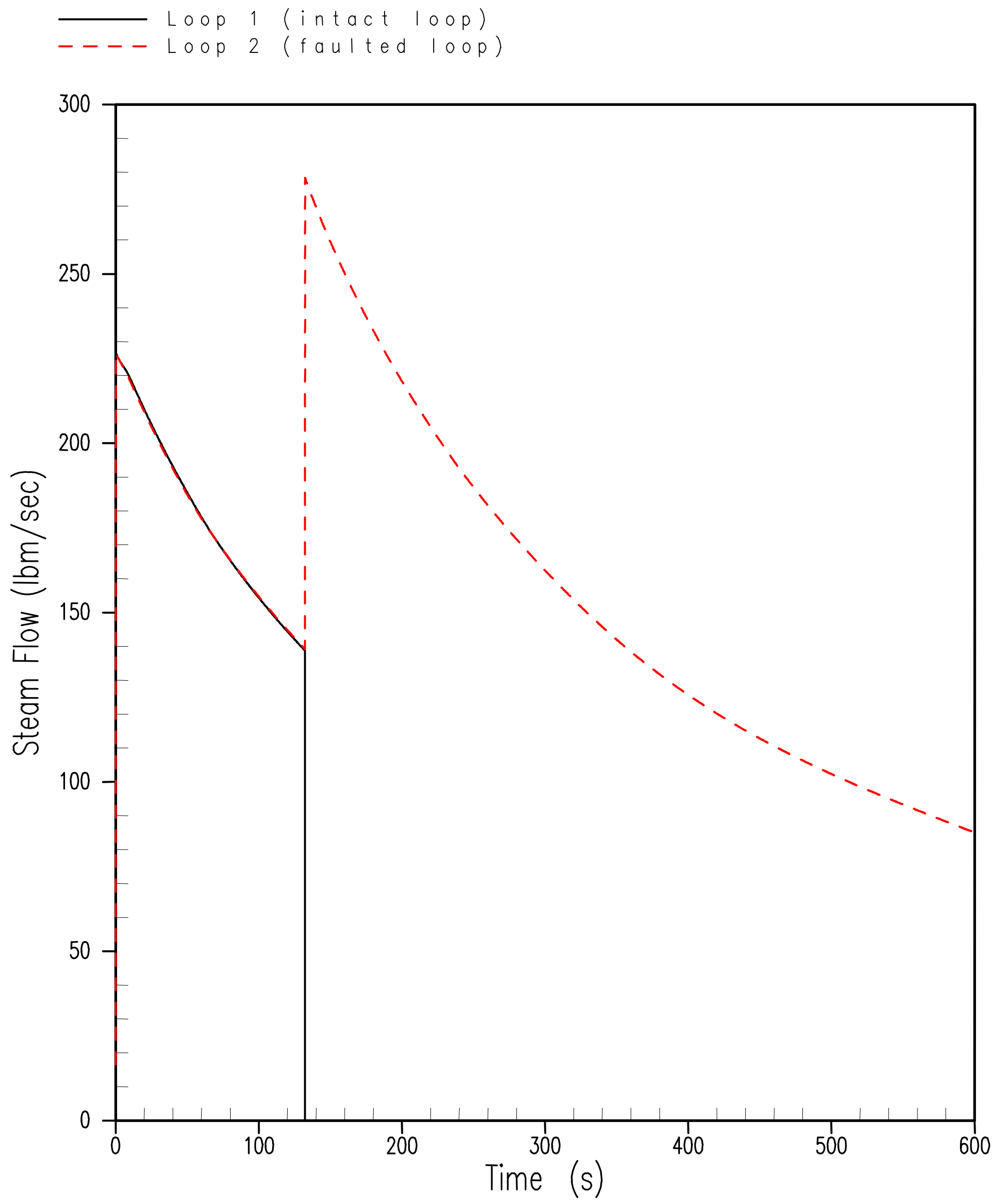


Figure 15.1.4-12
Steam Flow Transient
Inadvertent Opening of a Steam Generator Relief or Safety Valve

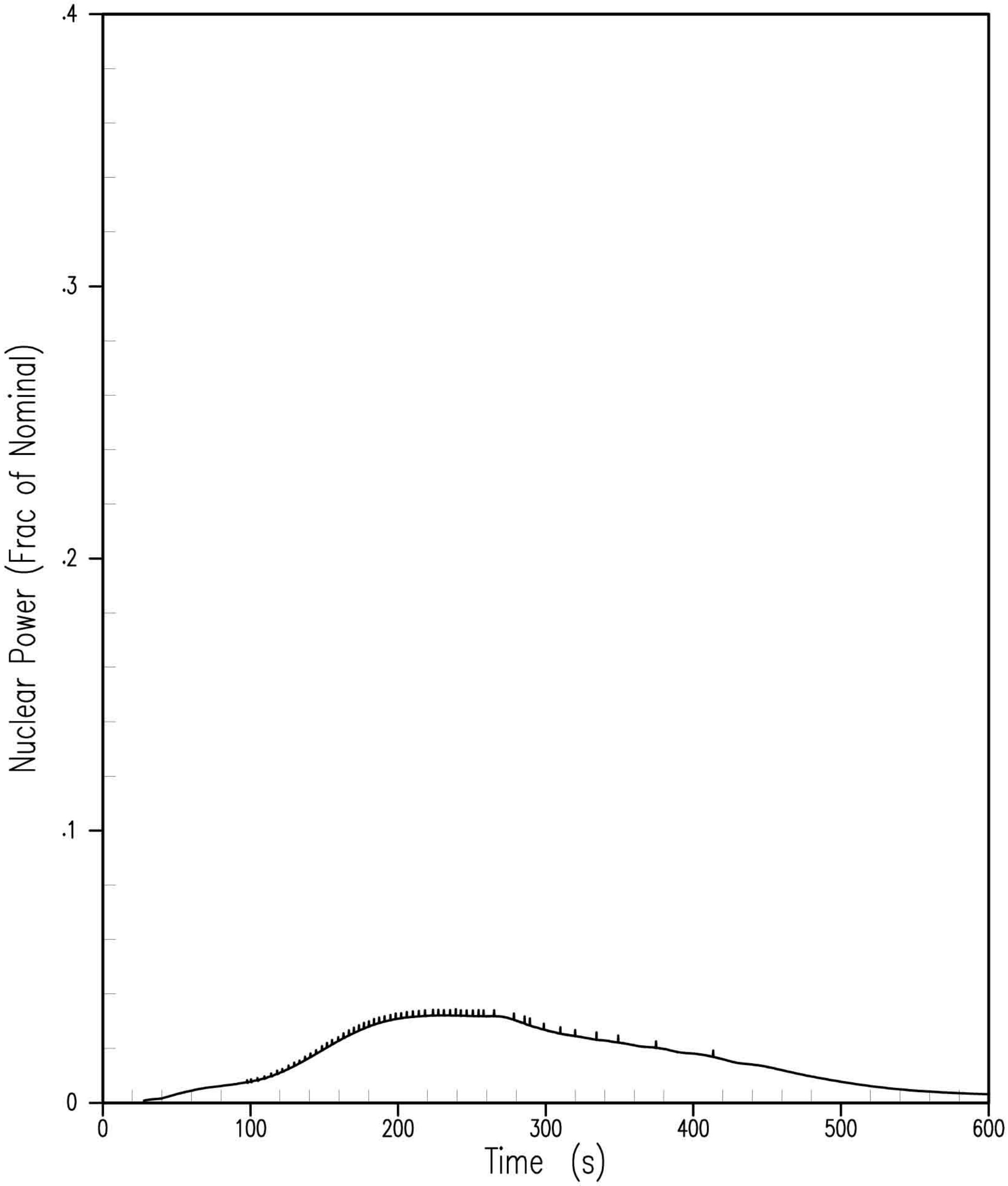


Figure 15.1.5-1
Nuclear Power Transient Steam System Piping Feature

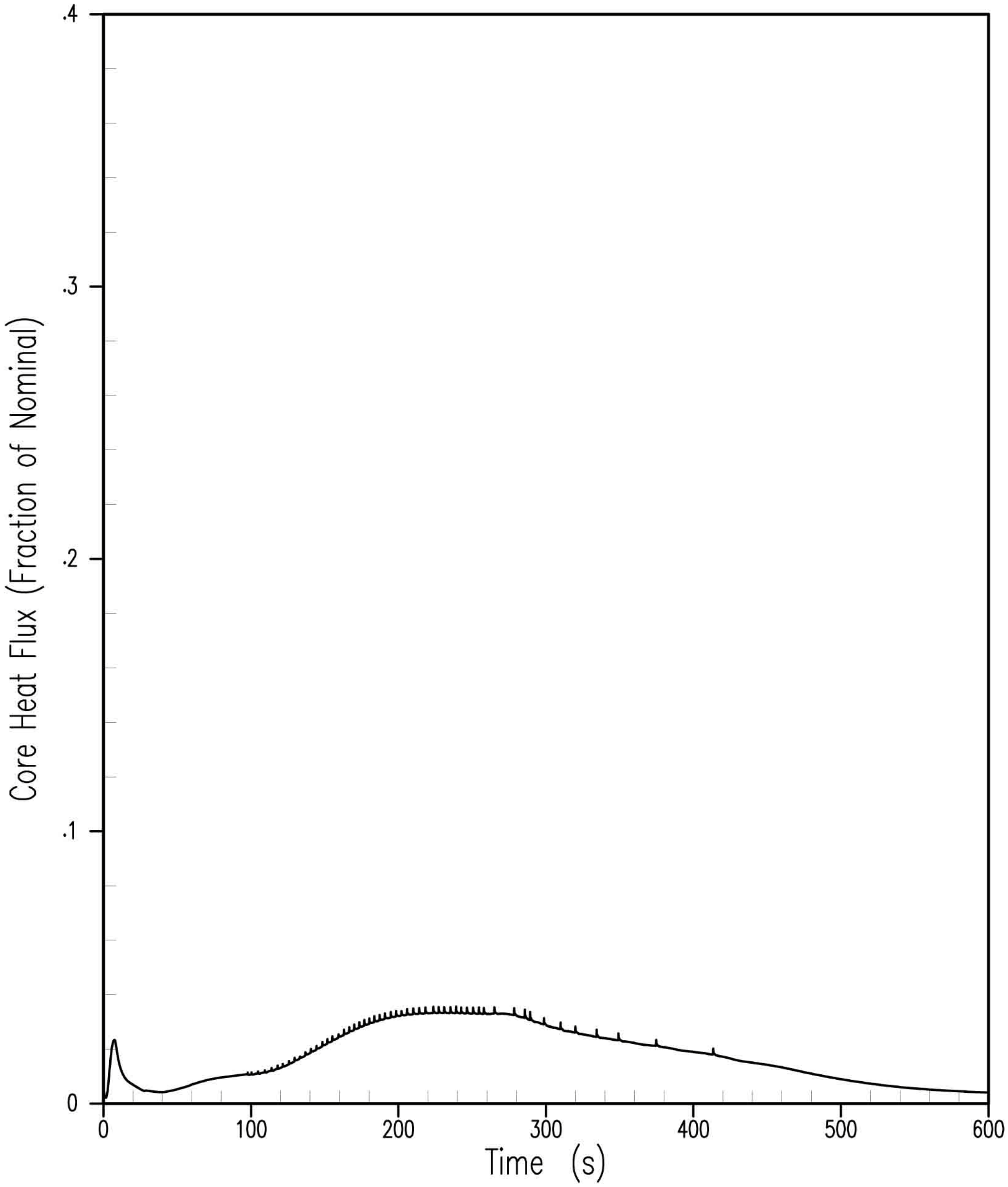


Figure 15.1.5-2
Core Heat Flux Transient Steam System Piping Failure

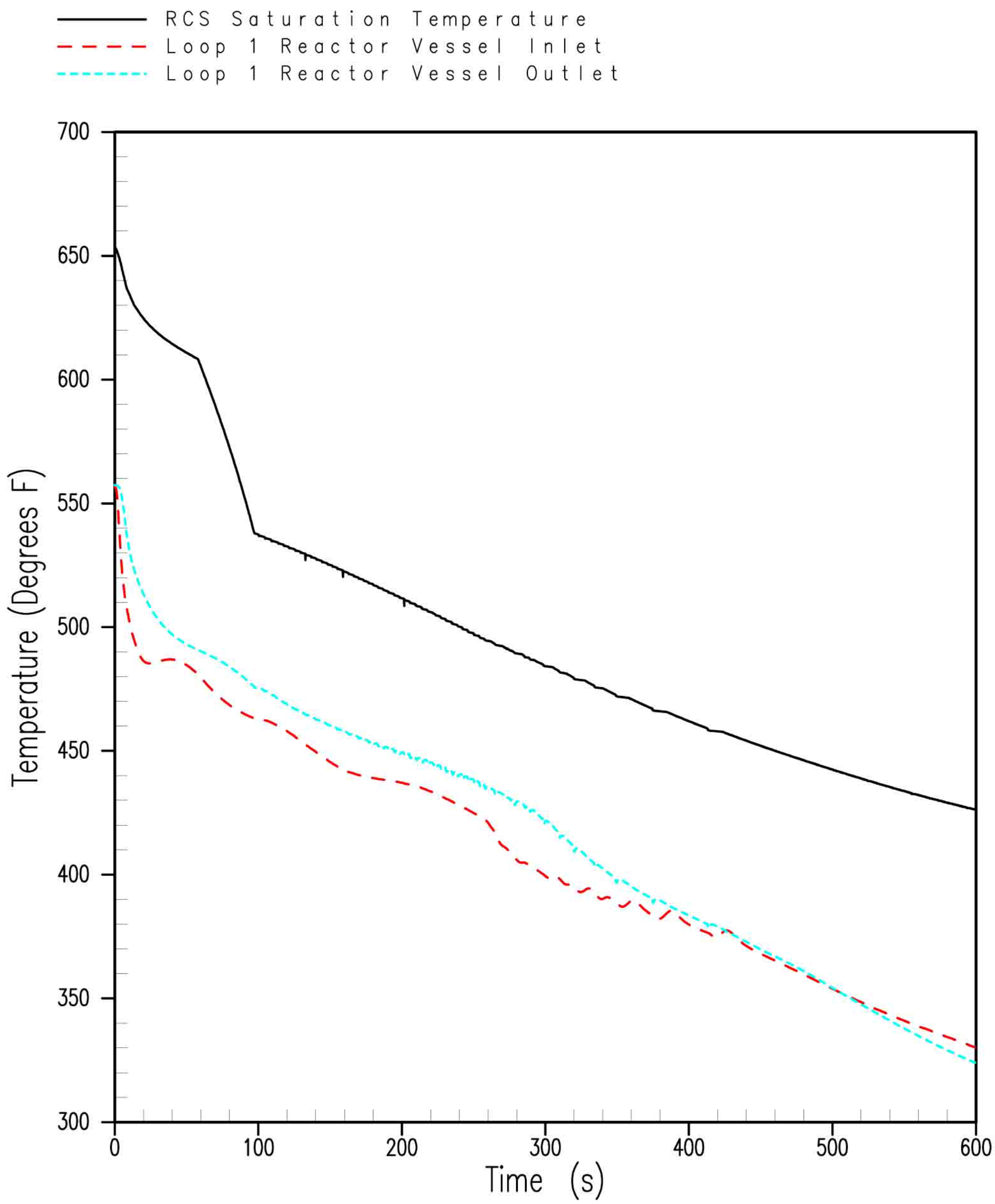


Figure 15.1.5-3
Loop 1 Reactor Coolant Temperatures
Steam System Piping Failure

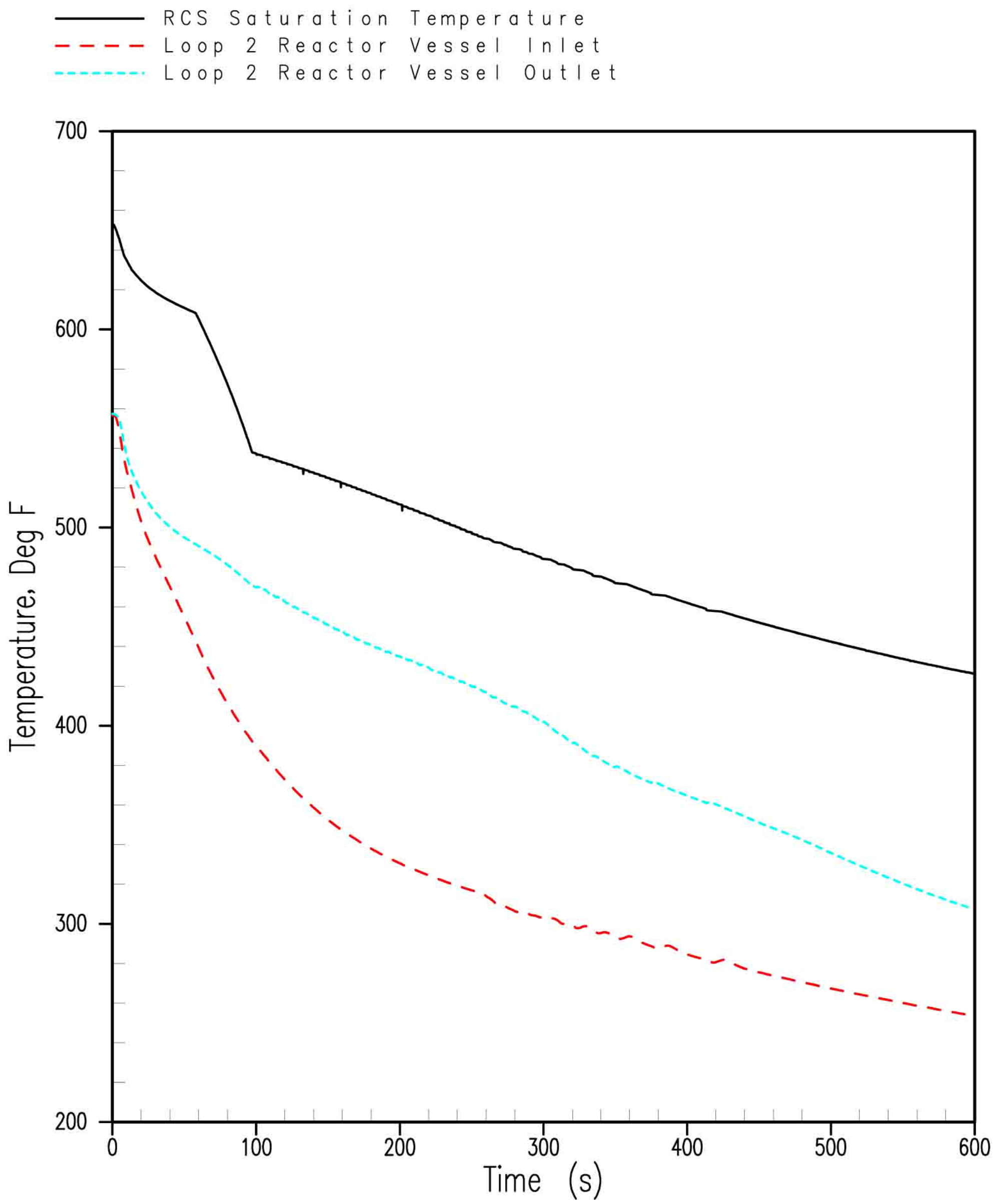


Figure 15.1.5-4
Loop 2 Reactor Coolant Temperatures
Steam System Piping Failure

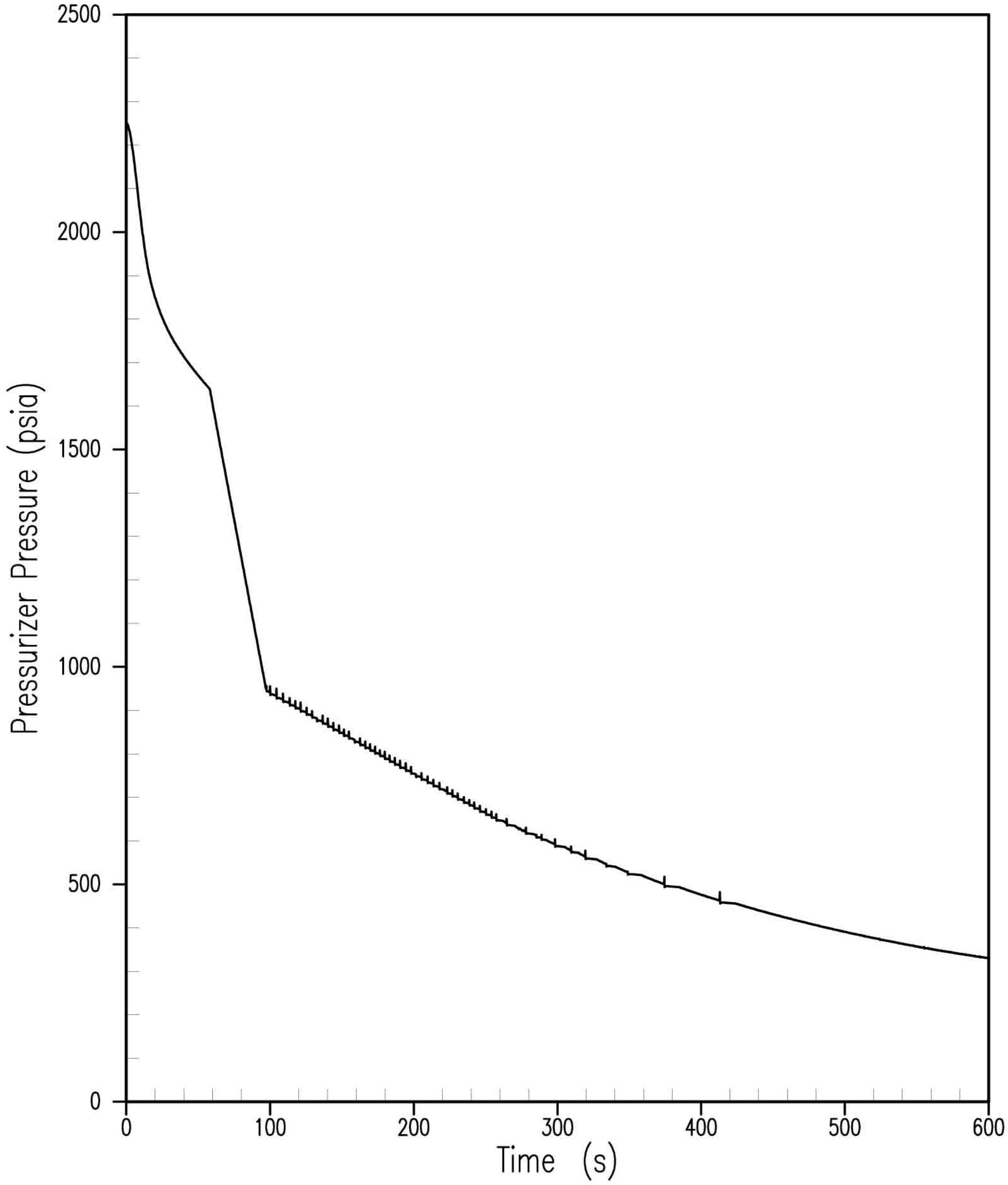


Figure 15.1.5-5
Reactor Coolant System Pressure Transient
Steam System Piping Failure

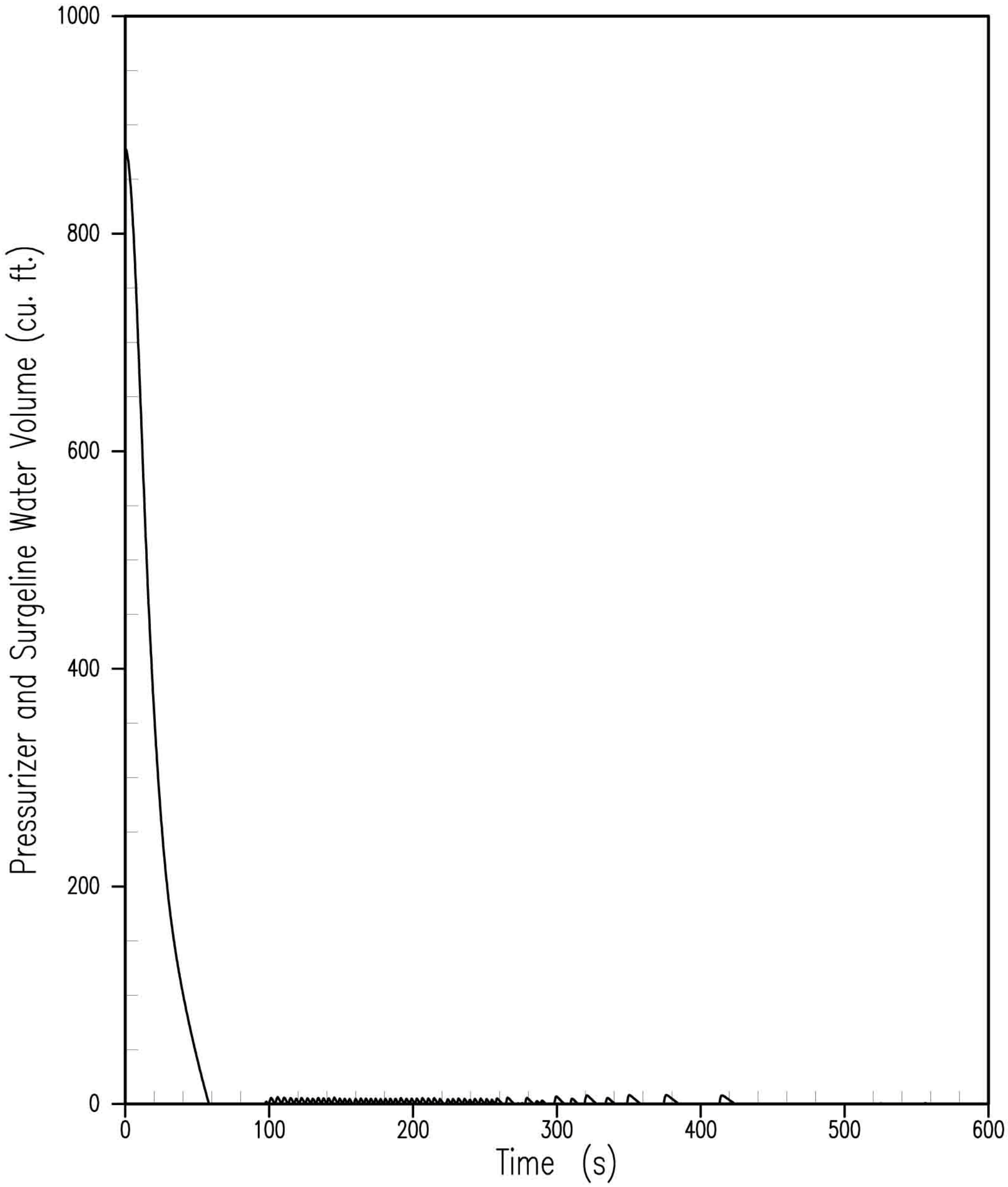


Figure 15.1.5-6
Pressurizer Water Volume Transient
Steam System Piping Failure

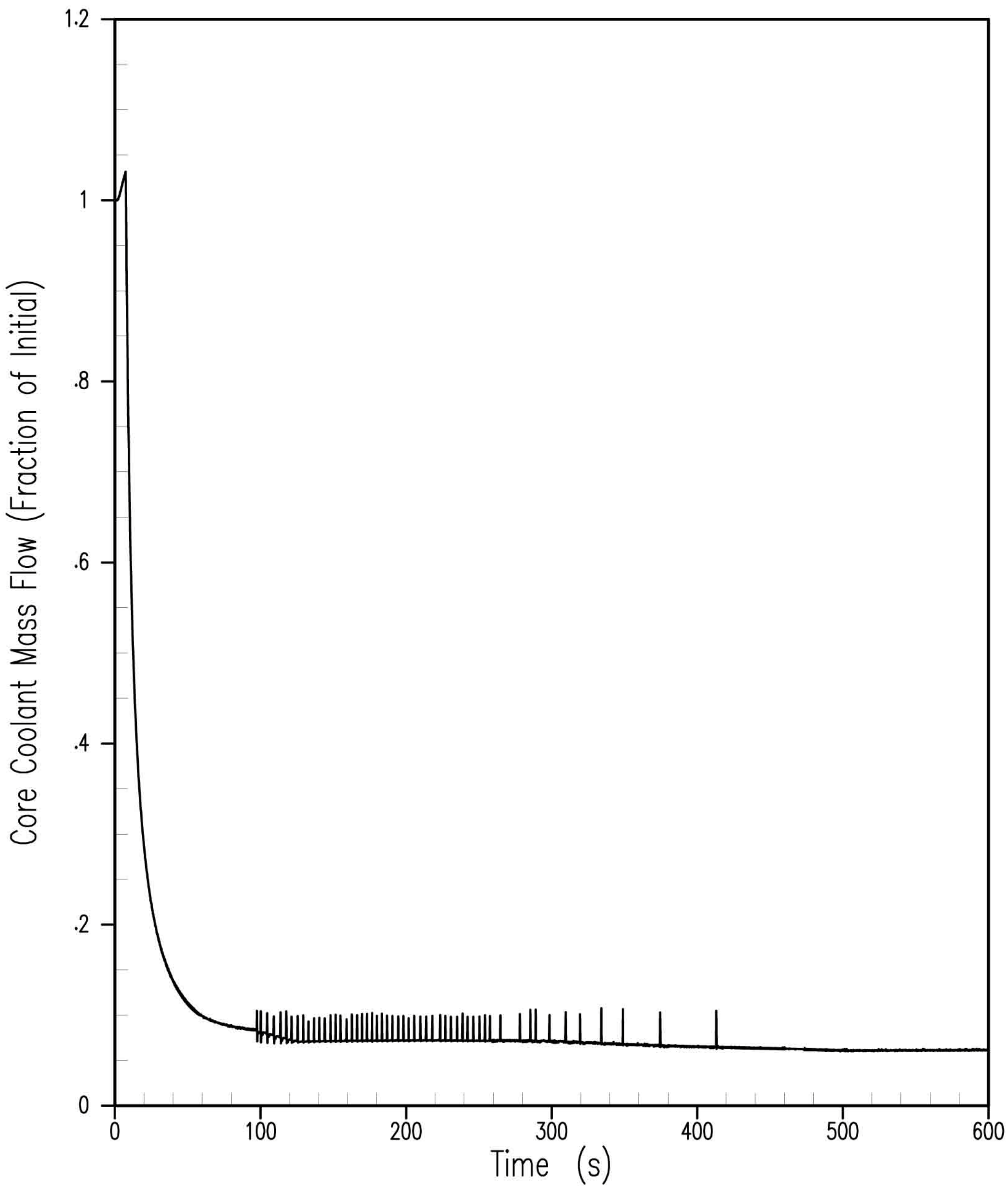


Figure 15.1.5-7
Core Flow Transient Steam System Piping Failure

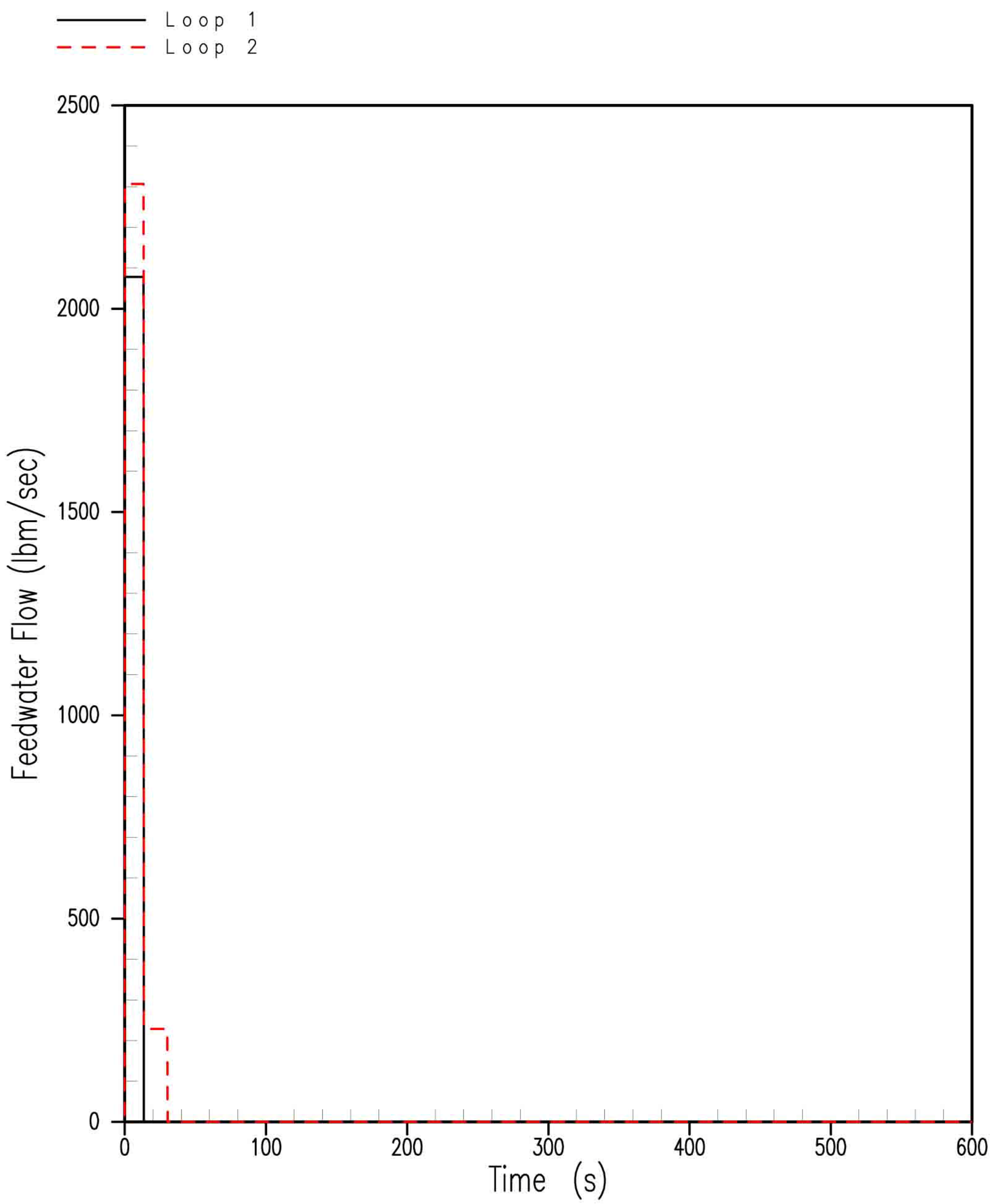


Figure 15.1.5-8
Feedwater Flow Transient Steam System Piping Failure

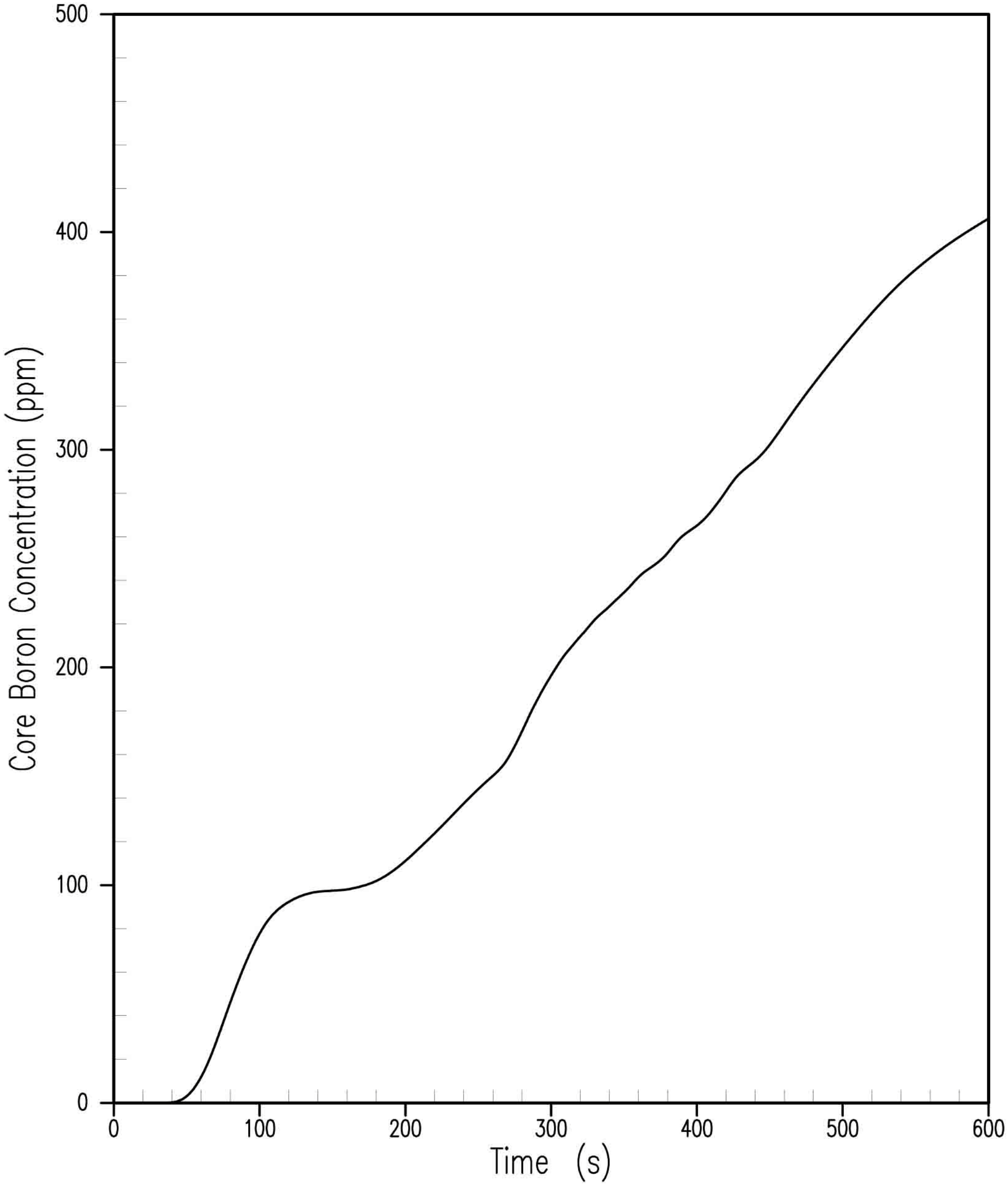


Figure 15.1.5-9
Core Boron Transient Steam System Piping Failure

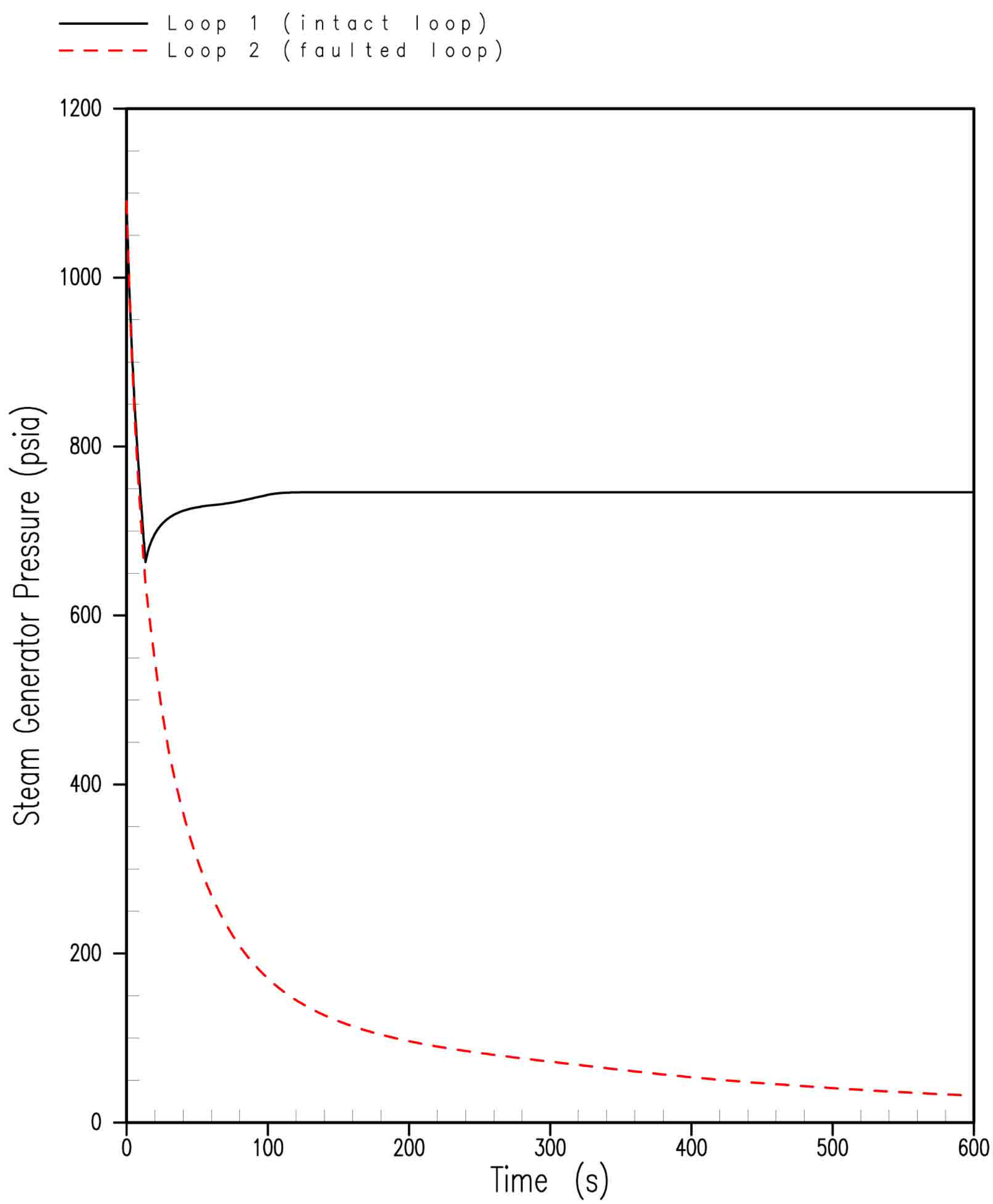


Figure 15.1.5-10
Steam Pressure Transient Steam System Piping Failure

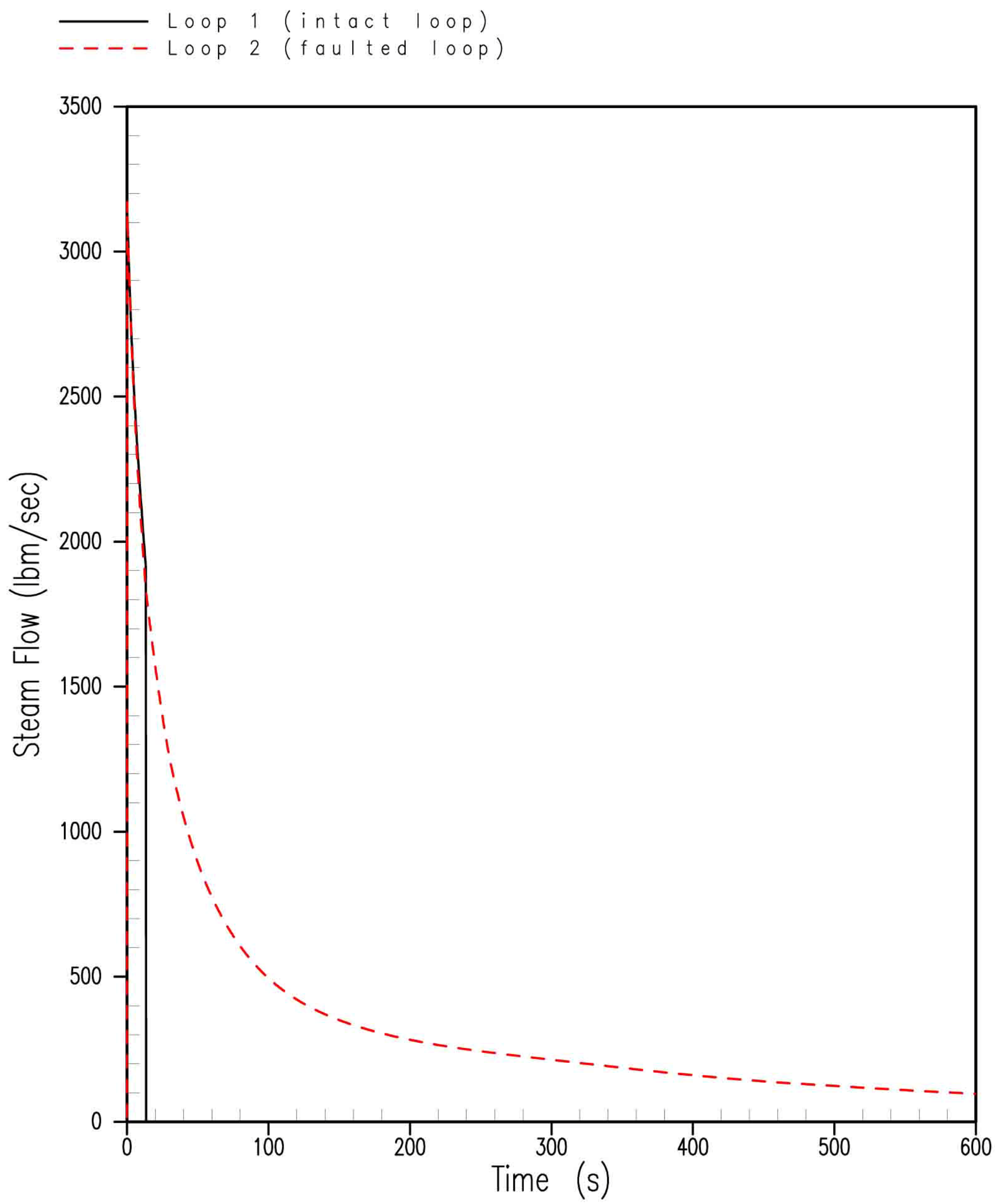


Figure 15.1.5-11
Steam Flow Transient Steam System Piping Failure

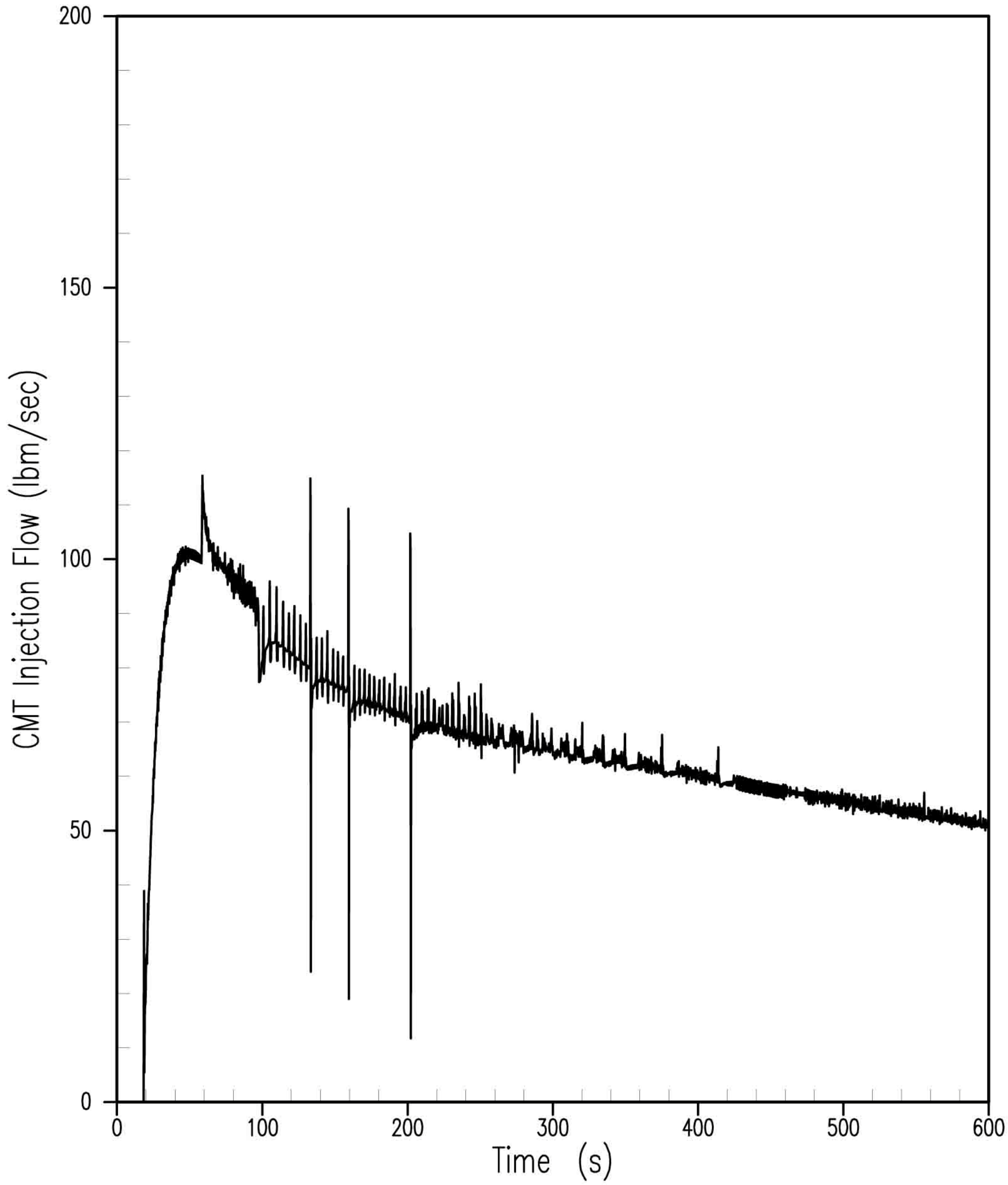


Figure 15.1.5-12
Core Makeup Tank Injection Flow
Steam System Piping Failure

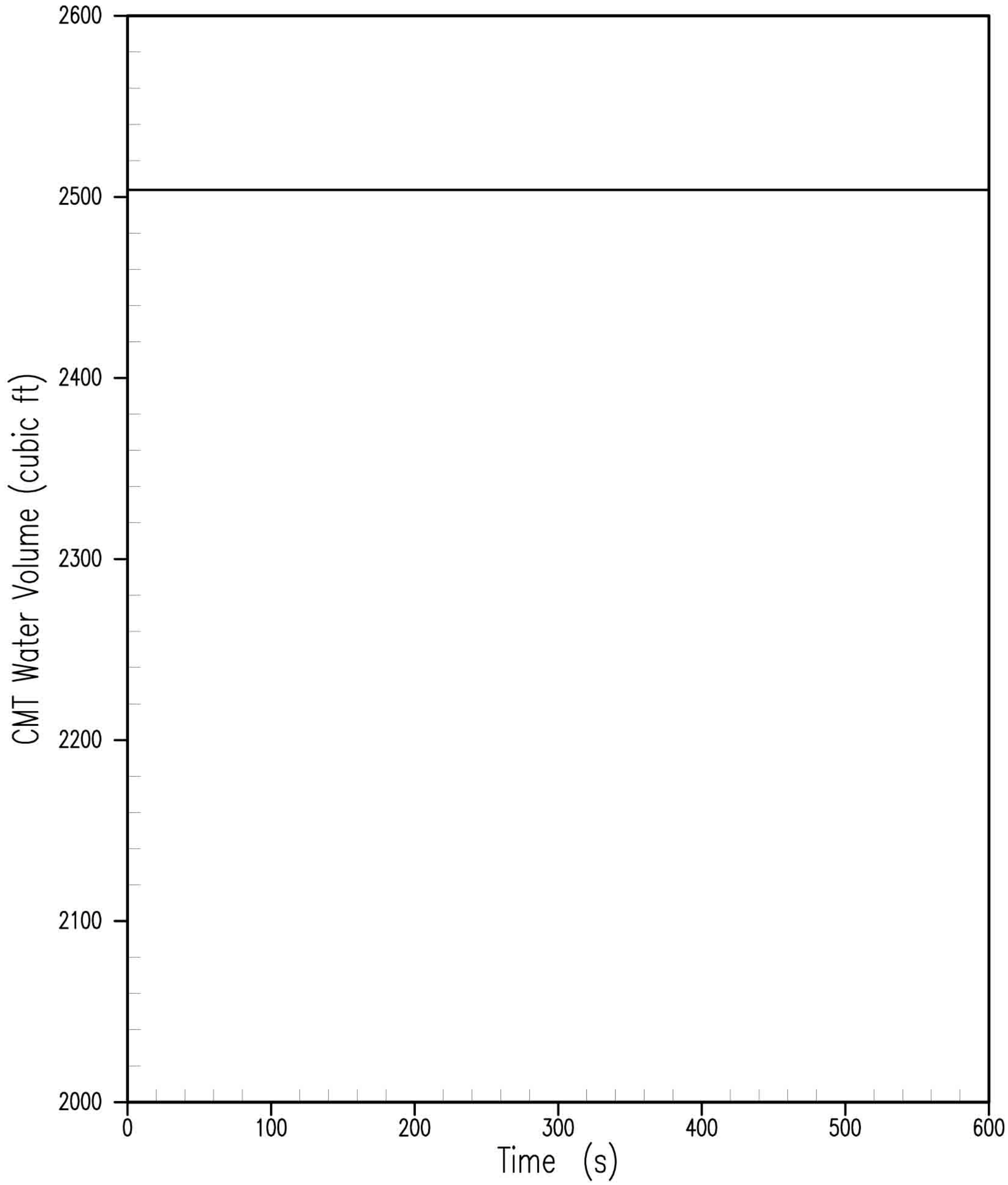


Figure 15.1.5-13
Core Makeup Tank Water Volume Steam System Piping Failure

Figures 15.1.6-1–15.1.6-8 Not Used

15.2 Decrease in Heat Removal by the Secondary System

A number of transients and accidents that could result in a reduction of the capacity of the secondary system to remove heat generated in the reactor coolant system are postulated. Analyses are presented in this section for the following events that are identified as more limiting than the others:

- Steam pressure regulator malfunction or failure that results in decreasing steam flow
- Loss of external electrical load
- Turbine trip
- Inadvertent closure of main steam isolation valves
- Loss of condenser vacuum and other events resulting in turbine trip
- Loss of ac power to the station auxiliaries
- Loss of normal feedwater flow
- Feedwater system pipe break

The above items are considered to be Condition II events, with the exception of a feedwater system pipe break, which is considered to be a Condition IV event.

For events in this section where PRHR HX actuation occurs, transients are presented until the PRHR HX heat removal matches decay heat generation. After that point in time, PRHR HX performance is driven by the performance of the passive containment cooling systems to control containment pressure and the ability of the condensate collection features to return condensate to the in-containment refueling water storage tank. The performance of these systems, for extended decay heat removal, is described in Subsection 6.3.1.1.1.

The radiological consequences of the accidents in this section are bounded by the radiological consequences of a main steam line break (see Subsection 15.1.5).

15.2.1 Steam Pressure Regulator Malfunction or Failure that Results in Decreasing Steam Flow

There are no steam pressure regulators in the AP1000 whose failure or malfunction causes a steam flow transient.

15.2.2 Loss of External Electrical Load

15.2.2.1 Identification of Causes and Accident Description

A major load loss on the plant can result from loss of electrical load due to an electrical system disturbance. The ac power remains available to operate plant components such as the reactor coolant pumps; as a result, the standby onsite diesel generators do not function for this event. Following the loss of generator load, an immediate fast closure of the turbine control valves occurs. The automatic turbine bypass system accommodates the excess steam generation. Reactor coolant temperatures and pressure do not significantly increase if the turbine bypass system and pressurizer pressure control system function properly. If the condenser is not available, the excess steam generation is relieved to the atmosphere. Additionally, main feedwater flow is lost if the condenser is not available. For this transient, feedwater flow is maintained by the startup feedwater system.

For a loss of electrical load without subsequent turbine trip, no direct reactor trip signal is generated. The plant trips from the protection and safety monitoring system if a safety limit is approached. A continued steam load of approximately 5 percent exists after total loss of external electrical load because of the steam demand of plant auxiliaries.

If a safety limit is approached, protection is provided by high pressurizer pressure, high pressurizer water level, and overtemperature ΔT trips. Voltage and frequency relays associated with the reactor coolant pump provide no additional safety function for this event. Following a complete loss of external electrical load, the maximum turbine overspeed is not expected to affect the voltage and frequency sensors. Any increased frequency to the reactor coolant pump motors results in a slightly increased flow rate and subsequent additional margin to safety limits. For postulated loss of load and subsequent turbine-generator overspeed, an overfrequency condition is not seen by the protection and safety monitoring system equipment or other safety-related loads. Safety-related loads and the protection and safety monitoring system equipment are supplied from the 120-Vac instrument power supply system, which in turn is supplied from the inverters. The inverters are supplied from a dc bus energized from batteries or by a regulated ac voltage.

If the steam dump valves fail to open following a large loss of load, the steam generator safety valves may lift and the reactor may be tripped by the high pressurizer pressure signal, the high pressurizer water level signal, or the overtemperature ΔT signal. This would cause steam generator shell side pressure and reactor coolant temperature to increase rapidly. However, the pressurizer safety valves and steam generator safety valves are sized to protect the reactor coolant system and steam generator against overpressure for load losses, without assuming the operation of the turbine bypass system, pressurizer spray, or automatic rod cluster control assembly control.

The steam generator safety valve capacity is sized to remove the steam flow at the nuclear steam supply system thermal rating from the steam generator, without exceeding 110 percent of the steam system design pressure. The pressurizer safety valve capacity is sized to accommodate a complete loss of heat sink, with the plant initially operating at the maximum turbine load, along with operation of the steam generator safety valves. The pressurizer safety valves can then relieve sufficient steam to maintain the reactor coolant system pressure within 110 percent of the reactor coolant system design pressure.

A discussion of overpressure protection can be found in WCAP-7769, Revision 1 (Reference 1) and WCAP-16779 (Reference 8).

A loss-of-external-load event is classified as a Condition II event, fault of moderate frequency.

A loss-of-external-load event results in a plant transient that is bounded by the turbine trip event analyzed in Subsection 15.2.3. Therefore, a detailed transient analysis is not presented for the loss-of-external-load event.

The primary side transient is caused by a decrease in heat transfer capability, from primary to secondary, due to a rapid termination of steam flow to the turbine, accompanied by an automatic reduction of feedwater flow (should feedwater flow not be reduced, a larger heat sink is available and the transient is less severe). Reduction of steam flow to the turbine following a loss-of-external load event occurs due to automatic fast closure of the turbine control valves. Following a turbine trip event, termination of steam flow occurs via turbine stop valve closure, which occurs in approximately 0.15 seconds. The transient in primary pressure, temperature, and water volume is less severe for the loss-of-external-load event than for the turbine trip due to a slightly slower loss of heat transfer capability.

The protection available to mitigate the consequences of a loss-of-external-load event is the same as that for a turbine trip, as listed in Table 15.0-6.

15.2.2.2 Analysis of Effects and Consequences

Refer to Subsection 15.2.3.2 for the method used to analyze the limiting transient (turbine trip) in this grouping of events. The results of the turbine trip event analysis bound those expected for the loss-of-external-load event, as discussed in Subsection 15.2.2.1.

Plant systems and equipment that may be required to function in order to mitigate the effects of a complete loss of load are discussed in Subsection 15.0.8 and listed in Table 15.0-6.

The protection and safety monitoring system may be required to terminate core heat input and to prevent departure from nucleate boiling (DNB). Depending on the magnitude of the load loss, pressurizer safety valves and/or steam generator safety valves may open to maintain system pressures below allowable limits. No single active failure prevents operation of any system required to function. Normal plant control systems and engineered safety systems are not required to function. The passive residual heat removal (PRHR) system may be automatically actuated following a loss of main feedwater, further mitigating the effects of the transient.

15.2.2.3 Conclusions

Based on results obtained for the turbine trip event and considerations described in Subsection 15.2.2.1, the applicable Standard Review Plan, Subsection 15.2.1, evaluation criteria for a loss-of-external-load event, are met (see Subsection 15.2.3).

15.2.3 Turbine Trip

15.2.3.1 Identification of Causes and Accident Description

The turbine stop valves close rapidly (about 0.15 seconds) on loss of trip fluid pressure actuated by one of a number of possible turbine trip signals. Turbine trip initiation signals include:

- Generator trip
- Low condenser vacuum
- Loss of lubricating oil
- Turbine thrust bearing failure
- Turbine overspeed
- Manual trip
- Reactor trip

Upon initiation of stop valve closure, steam flow to the turbine stops abruptly. Sensors on the stop valves detect the turbine trip and initiate turbine bypass. The loss of steam flow results in a rapid increase in secondary system temperature and pressure, with a resultant primary system transient, described in Subsection 15.2.2.1, for the loss-of-external-load event. A slightly more severe transient occurs for the turbine trip event due to the rapid loss of steam flow caused by the abrupt valve closure.

The automatic turbine bypass system accommodates up to 40 percent of rated steam flow. Reactor coolant temperatures and pressure do not increase significantly if the turbine bypass system and pressurizer pressure control system are functioning properly. If the condenser is not available, the excess steam generation is relieved to the atmosphere and main feedwater flow is lost. For this situation, feedwater flow is maintained by the startup feedwater system to provide adequate residual and decay heat removal capability. Should the turbine bypass system fail to operate, the steam generator safety valves may lift to provide pressure control. See Subsection 15.2.2.1 for a further discussion of the transient.

A turbine trip is classified as a Condition II event, fault of moderate frequency.

A turbine trip is a more limiting than a loss-of-external-load event, loss of condenser vacuum, and other events which result in a turbine trip. As such, this event is analyzed and presented in Subsection 15.2.3.2.

15.2.3.2 Analysis of Effects and Consequences

15.2.3.2.1 Method of Analysis

In this analysis, the behavior of the unit is evaluated for a complete loss of steam load from 100 percent of full power, without rapid power reduction, primarily to show the adequacy of the pressure-relieving devices, and to demonstrate core protection margins. The turbine is assumed to trip without actuating the rapid power reduction system. This assumption delays reactor trip until conditions in the reactor coolant system result in a trip due to other signals. Thus, the analysis assumes a bounding transient. In addition, no credit is taken for the turbine bypass system. Main feedwater flow is terminated at the time of turbine trip, with no credit taken for startup feedwater or the PRHR heat exchanger (except for long-term recovery) to mitigate the consequences of the transient.

In meeting the requirements of GDC 17 of 10 CFR Part 50, Appendix A, analyses are performed to evaluate the effects produced by a possible consequential loss of offsite power during a complete loss of steam load. As discussed in Subsection 15.0.14, the loss of offsite power is considered as a direct consequence of a turbine trip occurring while the plant is operating at power. The primary effect of the loss of offsite power is to cause the reactor coolant pumps to coast down.

The turbine trip transients are analyzed by using the computer program LOFTRAN (Reference 2). The program simulates the neutron kinetics, reactor coolant system, pressurizer, pressurizer safety valves, pressurizer spray, steam generator, and steam generator safety valves. The program computes pertinent plant variables, including temperatures, pressures, and power level.

In the turbine trip analyses, that include a primary coolant flow coastdown caused by a consequential loss of offsite power, a combination of three computer codes is used to perform the departure from nucleate boiling ratio (DNBR) analyses. First, the LOFTRAN code (References 2 and 6) is used to calculate the plant system transient. The FACTRAN code (Reference 7) is then used to calculate the core heat flux based on nuclear power and reactor coolant flow from LOFTRAN. Finally, the VIPRE-01 code (see Section 4.4) is used to calculate the DNBR using heat flux from FACTRAN and flow from LOFTRAN.

The major assumptions used in the analysis are summarized below.

Initial Operating Conditions

Two sets of initial operating conditions are used. Cases performed to evaluate the minimum DNBR obtained are analyzed using the revised thermal design procedure. Initial core power, reactor coolant temperature, and pressure are assumed to be at their nominal values consistent with steady-state full-power operation. Uncertainties in initial conditions are included in the DNBR limit as described in WCAP-11397-P-A (Reference 5).

Cases performed to evaluate the maximum calculated RCS pressure include uncertainties on the initial conditions. Initial core power, reactor coolant temperature, and pressure are assumed to be at the nominal full-power values plus or minus uncertainties. The direction of the uncertainties is chosen to maximize the RCS pressure.

Reactivity Coefficients

Two cases are analyzed:

- Minimum reactivity feedback – A least-negative moderator temperature coefficient and a least-negative Doppler-only power coefficient are assumed (see Figure 15.0.4-1).
- Maximum reactivity feedback – A conservatively large negative moderator temperature coefficient and a most-negative Doppler-only power coefficient are assumed (see Figure 15.0.4-1).

Reactor Control

From the standpoint of the maximum pressures attained, it is conservative to assume that the reactor is in manual control. If the reactor is in automatic control, the control rod banks move prior to trip and reduce the severity of the transient.

Steam Release

No credit is taken for the operation of the turbine bypass system or steam generator power-operated relief valves. The steam generator pressure rises to the safety valve setpoint where steam release through safety valves limits secondary steam pressure at the setpoint value.

Pressurizer Spray

Two cases for both the minimum and maximum reactivity feedback cases are analyzed:

- Full credit is taken for the effect of pressurizer spray in reducing or limiting the coolant pressure. Safety valves are also available.
- No credit is taken for the effect of pressurizer spray in reducing or limiting the coolant pressure. Safety valves are operable.

Feedwater Flow

Main feedwater flow to the steam generators is assumed to be lost at the time of turbine trip. No credit is taken for startup feedwater flow or the PRHR heat exchanger, because a stabilized plant condition is reached before initiation of the startup feedwater or the PRHR heat exchanger is normally assumed to occur. The startup feedwater flow or PRHR heat exchanger remove core decay heat following plant stabilization.

Reactor Trip

Reactor trip is actuated by the first reactor trip setpoint reached, with no credit taken for the rapid power reduction on the turbine trip. Trip signals are expected due to high pressurizer pressure, overtemperature ΔT, low RCP speed, high pressurizer water level, and low steam generator water level.

Plant characteristics and initial conditions are further discussed in Subsection 15.0.3. Plant systems and equipment that may be required to function in order to mitigate the effects of a turbine trip event are discussed in Subsection 15.0.8 and listed in Table 15.0-6.

The protection and safety monitoring system may be required to function following a turbine trip. Pressurizer safety valves and/or steam generator safety valves may be required to open to maintain system pressures below allowable limits. No single active failure prevents operation of systems required to function. Cases are analyzed, both with and without the operation of pressurizer spray, to determine the worst case for presentation.

Availability of Offsite Power

Each case is analyzed with and without offsite power available. As discussed in Subsection 15.0.14, the loss of offsite power is considered to be a consequence of an event due to disruption of the electrical grid following a turbine trip during the event. The grid is assumed to remain stable for 3 seconds following the turbine trip. In the analysis for the complete loss of steam load, the event is initiated by a turbine trip. Therefore, offsite power is assumed to be lost 3 seconds after the start of the event. For the loss of steam load analysis, the primary impact of the loss of offsite power is a coastdown of the reactor coolant pumps.

15.2.3.2.2 Results

The transient responses for a turbine trip from 100 percent of full-power operation are shown for eight cases. The eight analysis cases are performed assuming minimum and maximum reactivity feedback, with and without credit for pressurizer spray, and with and without offsite power available. The results of the analyses are shown in Figures 15.2.3-1 through 15.2.3-26. The calculated sequence of events for the accident is shown in Table 15.2-1.

Minimum Reactivity Feedback, Without Pressurizer Spray, With and Without Offsite Power Available

The results for these cases are shown in Figures 15.2.3-15 through 15.2.3-20. In the case with offsite power available, the reactor is tripped by the high pressurizer pressure trip function. The pressure safety valves are actuated in this case and maintain the reactor coolant system pressure below 110 percent of the design value. The DNB design basis defined in Section 4.4 is met for this case.

If offsite power is lost, the reactor is tripped by the low reactor coolant pump speed reactor trip function. Offsite power is assumed to be lost 3 seconds after turbine trip. This causes a reduction in reactor coolant system flow, which is illustrated in Figure 15.2.3-20. The DNB transient is similar to, and bounded by, the minimum reactivity feedback case with pressurizer spray and without offsite power. The DNB design basis defined in Section 4.4 is met for this case. The pressurizer safety valves actuate in this case and maintain the reactor coolant system pressure below 110 percent of the design value. Pressurizer pressure for this case is shown in Figure 15.2.3-16. Note that the with and without offsite power cases have different assumptions regarding initial pressure. The initial pressure assumptions were based upon sensitivities that were run. With respect to maximum reactor coolant system pressure, this case is the most limiting for complete loss of steam load cases.

Minimum Reactivity Feedback, With Pressurizer Spray, With and Without Offsite Power Available

Figures 15.2.3-1 through 15.2.3-7 show the transient responses with and without offsite power available. In the case with offsite power available, the reactor is tripped by the high pressurizer pressure trip function. Pressurizer pressure is shown in Figure 15.2.3-2, and the pressure within the reactor coolant system is maintained below 110 percent of the design value. The DNBR for the case with offsite power is shown in Figure 15.2.3-6, and the DNB design basis defined in Section 4.4 is met.

The case without offsite power is tripped by the low reactor coolant pump speed trip function. The DNB design basis defined in Section 4.4 is met. This case is the most limiting case with respect to DNB margin of the loss of steam load cases. The pressurizer pressure is shown in Figure 15.2.3-2, and the pressure within the reactor coolant system is maintained below 110 percent of the design value.

Maximum Reactivity Feedback, With Pressurizer Spray, With and Without Offsite Power Available

Figures 15.2.3-8 through 15.2.3-14 show the transient responses with and without offsite power available. In the case with offsite power available, the reactor is tripped by the high pressurizer pressure trip function. The pressure safety valves are actuated in this case and maintain the reactor coolant system pressure below 110 percent of the design value. Pressurizer pressure is shown in Figure 15.2.3-9. The transient DNBR for the case with offsite power available is shown in Figure 15.2.3-13. The DNB design basis defined in Section 4.4 is met for this case.

The case without offsite power is tripped by the low reactor coolant pump speed trip function. The DNB transient is similar to, and bounded by, the minimum feedback case with pressurizer spray and without offsite power. The DNB design basis defined in Section 4.4 is met. The pressurizer pressure is shown in Figure 15.2.3-9, and the pressure within the reactor coolant system is maintained below 110 percent of the design value.

Maximum Reactivity Feedback, Without Pressurizer Spray, With and Without Offsite Power Available

Figures 15.2.3-21 through 15.2.3-26 show the transient responses with and without offsite power available. In the case with offsite power available, the reactor is tripped by the high pressurizer pressure function.

Pressurizer pressure is shown in Figure 15.2.3-22, and the pressure within the reactor coolant system is maintained below 110 percent of the design value. Note that the with and without power cases have different assumptions regarding initial pressure. The initial pressure assumptions were based upon sensitivities that were run. The DNB design basis defined in Section 4.4 is met for this case.

The case without offsite power is tripped by the low reactor coolant pump speed trip function. The DNB transient is similar to, and bounded by, the minimum feedback case with pressurizer spray and without offsite power. The DNB design basis defined in Section 4.4 is met. The pressurizer pressure is shown in Figure 15.2.3-22, and the pressure within the reactor coolant system is maintained below 110 percent of the design value.

15.2.3.3 Conclusions

Results of the analyses show that a turbine trip presents no challenge to the integrity of the reactor coolant system or the main steam system. Pressure-relieving devices incorporated in the two systems are adequate to limit the maximum pressures to within the design limits.

The analyses show that the predicted DNBR is greater than the design limit at any time during the transient. Thus, the departure from nucleate boiling design basis, as described in Section 4.4, is met.

15.2.4 Inadvertent Closure of Main Steam Isolation Valves

Inadvertent closure of the main steam isolation valves results in a turbine trip with no credit taken for the turbine bypass system. Turbine trips are discussed in Subsection 15.2.3.

15.2.5 Loss of Condenser Vacuum and Other Events Resulting in Turbine Trip

Loss of condenser vacuum is one of the events that can cause a turbine trip. Turbine trip initiating events are described in Subsection 15.2.3. A loss of condenser vacuum prevents the use of steam dump to the condenser. Because steam dump is assumed to be unavailable in the turbine trip analysis, no additional adverse effects result if the turbine trip is caused by loss of condenser vacuum. Therefore, the analysis results and conclusions contained in Subsection 15.2.3 apply to the loss of the condenser vacuum. In addition, analyses for the other possible causes of a turbine trip, listed in Subsection 15.2.3.1, are covered by Subsection 15.2.3. Possible overfrequency effects, due to a turbine overspeed condition, are discussed in Subsection 15.2.2.1 and are not a concern for this type of event.

15.2.6 Loss of ac Power to the Plant Auxiliaries

15.2.6.1 Identification of Causes and Accident Description

The loss of power to the plant auxiliaries is caused by a complete loss of the offsite grid accompanied by a turbine-generator trip. The onsite standby ac power system remains available but is not credited to mitigate the accident.

From the decay heat removal point of view, in the long term this transient is more severe than the turbine trip event analyzed in Subsection 15.2.3 because, for this case, the decrease in heat removal by the secondary system is accompanied by a reactor coolant flow coastdown, which further reduces the capacity of the primary coolant to remove heat from the core. The reactor will trip:

- Upon reaching one of the trip setpoints in the primary or secondary systems as a result of the flow coastdown and decrease in secondary heat removal.
- Due to the loss of power to the control rod drive mechanisms as a result of the loss of power to the plant.

Following a loss of ac power with turbine and reactor trips, the sequence described below occurs:

- Plant vital instruments are supplied from the Class 1E and uninterruptable power supply.
- As the steam system pressure rises following the trip, the steam generator power-operated relief valves may be automatically opened to the atmosphere. The condenser is assumed not to be available for turbine bypass. If the steam flow rate through the power-operated relief valves is not available, the steam generator safety valves may lift to dissipate the sensible heat of the fuel and coolant plus the residual decay heat produced in the reactor.
- The onsite standby power system, if available, supplies ac power to the selected plant non-safety loads.
- As the no-load temperature is approached, the steam generator power-operated relief valves (or safety valves, if the power-operated relief valves are not available) are used to dissipate the residual decay heat and to maintain the plant at the hot shutdown condition if the startup feedwater is available to supply water to the steam generators.
- If startup feedwater is not available, the PRHR heat exchanger is actuated.

During a plant transient, core decay heat removal is normally accomplished by the startup feedwater system if available, which is started automatically when low levels occur in either steam generator. If that system is not available, emergency core decay heat removal is provided by the PRHR heat exchanger. The PRHR heat exchanger is a C-tube heat exchanger connected, through inlet and outlet headers, to the reactor coolant system. The inlet to the heat exchanger is from the reactor coolant system hot leg, and the return is to the steam generator outlet plenum. The heat exchanger is located above the core to provide natural circulation flow when the reactor coolant pumps are not operating. The IRWST provides the heat sink for the heat exchanger. The PRHR heat exchanger, in conjunction with the passive containment cooling system, provides core cooling and maintains reactor coolant system conditions to satisfy the evaluation criteria. After the IRWST water reaches saturation (in about two and half hours), steam starts to vent to the containment atmosphere. The condensation that collects on the containment steel shell (cooled by the passive containment cooling system) returns to the IRWST, maintaining fluid level for the PRHR heat exchanger heat sink. The analysis shows that the natural circulation flow in the reactor coolant system following a loss of ac power event is sufficient to remove residual heat from the core.

Upon the loss of power to the reactor coolant pumps, coolant flow necessary for core cooling and the removal of residual heat is maintained by natural circulation in the reactor coolant and PRHR loops.

A loss of ac power to the plant auxiliaries is a Condition II event, a fault of moderate frequency. This event is more limiting with respect to long-term heat removal than the turbine trip initiated decrease in secondary heat removal without loss of ac power, which is discussed in Subsection 15.2.3. A loss of offsite power to the plant auxiliaries will also result in a loss of normal feedwater.

The plant systems and equipment available to mitigate the consequences of a loss of ac power event are discussed in Subsection 15.0.8 and listed in Table 15.0-6.

15.2.6.2 Analysis of Effects and Consequences

15.2.6.2.1 Method of Analysis

The analysis is performed to demonstrate the adequacy of the protection and safety monitoring system, the PRHR heat exchanger, and the reactor coolant system natural circulation capability in removing long-term (approximately 36,000 seconds) decay heat. This analysis also demonstrates the adequacy of these systems in preventing excessive heatup of the reactor coolant system with possible reactor coolant system overpressurization or loss of reactor coolant system water.

A modified version of the LOFTRAN code (Reference 2), described in WCAP-15644 (Reference 6), is used to simulate the system transient following a plant loss of offsite power. The simulation describes the plant neutron kinetics and reactor coolant system, including the natural circulation, pressurizer, and steam generator system responses. The digital program computes pertinent variables, including the steam generator level, pressurizer water level, and reactor coolant average temperature.

The assumptions used in this analysis minimize the energy removal capability of the PRHR heat exchanger and maximize the coolant system expansion.

The transient response of the plant following a loss of ac power to plant auxiliaries is similar to the loss of normal feedwater flow accident (see Subsection 15.2.7), except that power is assumed to be lost to the reactor coolant pumps at the time of the reactor trip.

The assumptions used in the analysis are as follows:

- The plant is initially operating at 102 percent of the design power rating with initial reactor coolant temperature 7°F below the nominal value and the pressurizer pressure 50 psi above the nominal value. The main feedwater flow measurement supports a 1-percent power uncertainty; use of a 2-percent power uncertainty is conservative.

- Core residual heat generation is based on ANSI 5.1 (Reference 3). ANSI 5.1 is a conservative representation of the decay energy release rates.

- Reactor trip occurs on steam generator low level (narrow range). Offsite power is assumed to be lost at the time of reactor trip. This is more conservative than the case in which offsite power is lost at time zero because of the lower steam generator water mass at the time of the reactor trip.

- A heat transfer coefficient is assumed in the steam generator associated with reactor coolant system natural circulation flow conditions following the reactor coolant pump coastdown.

- The PRHR heat exchanger is actuated by the low steam generator water level (narrow range coincident with low start up feed water flow).

- For the loss of ac power to the station auxiliaries, the only safety function required is core decay heat removal. That is accomplished by the PRHR heat exchanger. One of two parallel valves in the PRHR outlet line is assumed to fail to open. This is the worst single failure.
- Secondary system steam relief is achieved through the steam generator safety valves.

- The pressurizer safety valves are assumed to function.

Plant characteristics and initial conditions are further discussed in Subsection 15.0.3.

Plant systems and equipment necessary to mitigate the effects of a loss of ac power to the station auxiliaries are discussed in Subsection 15.0.8 and listed in Table 15.0-6. Normal reactor control systems are not required to function. The protection and safety monitoring system is required to function following a loss of ac power. The PRHR heat exchanger is required to function with a minimum heat transfer capability. No single active failure prevents operation of any system required to function.

The DNB analysis is not specifically addressed for this event since, from the point of view of DNBR transient, the loss of ac power to auxiliaries is similar and bounded by the Turbine Trip event analyzed in Subsection 15.2.3. In fact, the Turbine Trip is analyzed assuming that, following the turbine trip, a loss of ac power occurs with three seconds delay. This results in the coastdown of reactor coolant pumps, but, in the analysis, reactor trip on the loss of power is not assumed. The reactor trip is assumed to occur on an RCP Underspeed set point and rods begin to drop with more than one second delay from the pumps coastdown.

If a loss of ac power occurs as an initiating event, the first result would be the immediate reactor trip and the concomitant coastdown of the reactor coolant pumps. The calculated DNBR for such an event would be the same or higher than predicted for the Complete Loss of Reactor Coolant System flow as presented in 15.3.2.

15.2.6.2.2 Results

The transient response of the reactor coolant system following a loss of ac power to the plant auxiliaries is shown in Figures 15.2.6-1 through 15.2.6-12. The calculated sequence of events for this event is listed in Table 15.2-1.

The LOFTRAN code results show that the natural circulation flow and the PRHR system are sufficient to provide adequate core decay heat removal following reactor trip and reactor coolant pump coastdown.

Immediately following the reactor trip, the heat transfer capability of the PRHR heat exchanger and the steam generator heat extraction rate are sufficient to slowly cool down the plant. The cooldown continues until a low T_{cold} "S" signal is reached. The "S" signal actuates the core makeup tanks. During this transient, the core makeup tanks operate in water recirculation mode. The cold borated water injected by the core makeup tanks accelerates the cooldown of the plant. The core makeup tank flow slowly decreases as the core makeup tank fluid temperature increases due to water recirculation.

As the plant cools down, the heat removal capacity of the PRHR heat exchanger is lowered. When the heat removal rate from the reactor coolant system, due to the core makeup tank injection and the PRHR heat exchanger, decreases below the core decay heat produced, the reactor coolant system begins heating up again. As the reactor coolant system temperature is elevated, the heat removal capacity of the PRHR heat exchanger increases. The reactor coolant system temperature slowly increases until the heat removal rate of the PRHR heat exchanger matches the core decay heat produced.

Pressurizer safety valves open to discharge steam to containment and reclose later in the transient when the heat removal rate of the PRHR heat exchanger exceeds the decay heat production rate.

The capacity of the PRHR heat exchanger is sufficient to avoid water relief through the pressurizer safety valves.

The calculated sequence of events for this accident is listed in Table 15.2-1. As shown in Figures 15.2.6-5 and 15.2.6-6, in the long-term the plant starts a slow cooldown driven by the PRHR heat exchanger. Plant procedures may be followed to further cool down the plant.

15.2.6.3 Conclusions

Results of the analysis show that for the loss of ac power to plant auxiliaries event, all safety criteria are met. PRHR heat exchanger capacity is sufficient to prevent water relief through the pressurizer safety valves.

The analysis demonstrates that sufficient long-term reactor coolant system heat removal capability exists, via natural circulation and the PRHR heat exchanger, following reactor coolant pump coastdown to prevent fuel or cladding damage and reactor coolant system overpressure.

15.2.7 Loss of Normal Feedwater Flow

15.2.7.1 Identification of Causes and Accident Description

A loss of normal feedwater (from pump failures, valve malfunctions, or loss of ac power sources) results in a reduction in the capability of the secondary system to remove the heat generated in the reactor core. If startup feedwater is not available, the safety-related PRHR heat exchanger is automatically aligned by the protection and safety monitoring system to remove decay heat.

A small secondary system break can affect normal feedwater flow control, causing low steam generator levels prior to protective actions for the break. This scenario is addressed by the assumptions made for the feedwater system pipe break (see Subsection 15.2.8).

The following occurs upon loss of normal feedwater (assuming main feedwater pump fails or valve malfunctions):

- The steam generator water inventory decreases as a consequence of the continuous steam supply to the turbine. The mismatch between the steam flow to the turbine and the feedwater flow leads to the reactor trip on a low steam generator water level signal. The same signal also actuates the startup feedwater system (see Subsection 15.2.6.1).

- As the steam system pressure rises following the trip, the steam generator power-operated relief valves are automatically opened to the atmosphere. The condenser is assumed to be unavailable for turbine bypass. If the steam flow path through the power-operated relief valves is not available, the steam generator safety valves may lift to dissipate the sensible heat of the fuel and coolant plus the residual decay heat produced in the reactor.

- As the no-load temperature is approached, the steam generator power-operated relief valves (or safety valves, if the power-operated relief valves are not available) are used to dissipate the decay heat and to maintain the plant at the hot shutdown condition, if the startup feedwater is used to supply water to the steam generator.

- If startup feedwater is not available, the PRHR heat exchanger is actuated on either a low steam generator water level (narrow range), coincident with a low startup feedwater flow rate signal or a low steam generator water level (wide range) signal. The PRHR heat exchanger transfers the core decay heat and sensible heat to the IRWST so that core heat removal is uninterrupted following a loss of normal and startup feedwater (see Subsection 15.2.6).

A loss-of-normal-feedwater event is classified as a Condition II event, a fault of moderate frequency.

15.2.7.2 Analysis of Effects and Consequences

An analysis of the system transient is presented below to show that, following a loss of normal feedwater, the PRHR heat exchanger is capable of removing the stored and decay heat to prevent either overpressurization of the reactor coolant system or loss of water from the reactor coolant system.

15.2.7.2.1 Method of Analysis

An analysis using a modified version of the LOFTRAN code (Reference 2), described in WCAP-15644 (Reference 6), is performed to obtain the plant transient following a loss of normal feedwater. The simulation describes the neutron kinetics, reactor coolant system (including the natural circulation), pressurizer, and steam generators. The program computes pertinent variables, including the steam generator level, pressurizer water level, and reactor coolant average temperature.

The assumptions used in the analysis are as follows:

- The plant is initially operating at 102 percent of the design power rating. The main feedwater flow measurement supports a 1-percent power uncertainty; use of a 2-percent power uncertainty is conservative.

- Reactor trip occurs on steam generator low (narrow range) level.

- The only safety function required is the core decay heat removal that is carried by the PRHR heat exchanger; therefore, the worst single failure is assumed to occur in the PRHR heat exchanger. The actuation of the PRHR heat exchanger requires the opening of one of the two fail-open valves arranged in parallel at the PRHR heat exchanger discharge. Because no single failure can be assumed that impairs the opening of both valves, the failure of a single valve is assumed.

  The PRHR heat exchanger is actuated by the low steam generator water level narrow range signal, coincident with low start up feedwater flow.

- Secondary system steam relief is achieved through the steam generator safety valves.

- The initial reactor coolant average temperature is 7°F lower than the nominal value, and initial pressurizer pressure is 50 psi lower than nominal.

The loss of normal feedwater analysis is performed to demonstrate the adequacy of the protection and safety monitoring system and the PRHR heat exchanger in removing long-term decay heat and preventing excessive heatup of the reactor coolant system with possible resultant reactor coolant system overpressurization or loss of reactor coolant system water. The assumptions used in this analysis minimize the energy removal capability of the system, and maximize the coolant system expansion.

For the loss of normal feedwater transient, the reactor coolant volumetric flow remains at its normal value and the reactor trips via the low steam generator narrow range level trip. The reactor coolant pumps continue to run until automatically tripped when the core makeup tanks are actuated.

Plant characteristics and initial conditions are further discussed in Subsection 15.0.3.

Plant systems and equipment necessary to mitigate the effects of a loss of normal feedwater accident are discussed in Subsection 15.0.8 and listed in Table 15.0-6. Normal reactor control systems are not required to function. The protection and safety monitoring system is required to function following a loss of normal feedwater. The PRHR heat exchanger is required to function with a minimum heat transfer capability. No single active failure prevents operation of any system to perform its required function. A discussion of anticipated transients without scram considerations is presented in Section 15.8.

15.2.7.2.2 Results

Figures 15.2.7-1 through 15.2.7-10 show the significant plant parameters following a loss of normal feedwater.

Prior to reactor trip and the insertion of the rods into the core, the loss of normal feedwater transient is the same as the transient response presented in Subsection 15.2.6 for the loss of ac power to plant auxiliaries. The DNB results, presented in Figure 15.2.6-12 for the loss of ac power to plant auxiliaries, are also applicable for a loss of normal feedwater and demonstrate that the DNB design basis is met.

Following the reactor and turbine trip from full load, the water level in the steam generators falls due to the reduction of steam generator void fraction. Steam flow through the safety valves continues to dissipate the stored and core decay heat.

The capacity of the PRHR heat exchanger, when the reactor coolant pumps are operating, is much larger than the decay heat, and in the first part of the transient, the reactor coolant system is cooled down and the pressure decreases.

The cooldown continues until a low T_{cold} “S” signal is eventually reached. The “S” signal actuates the core makeup tanks. During this transient, the core makeup tanks operate in water recirculation mode. The cold borated water injected by the core makeup tanks accelerates the cooldown of the plant. The core makeup tank flow slowly decreases as the core makeup tank fluid temperature increases due to water recirculation.

As the plant cools down, the heat removal capacity of the passive residual heat exchanger is lowered. The heat removal rate from the reactor coolant system, due to the core makeup tank injection and the PRHR heat exchanger, then decreases below the core decay heat produced. The reactor coolant system then begins heating up again. As the reactor coolant system temperature is elevated, the heat removal capacity of the PRHR heat exchanger increases again. The reactor coolant system temperature slowly increases until the heat removal rate of the PRHR heat exchanger matches the core decay heat produced.

The capacity of the PRHR heat exchanger is sufficient to avoid water relief through the pressurizer safety valves.

The calculated sequence of events for this accident is listed in Table 15.2-1. As shown in Figures 15.2.7-3 and 15.2.7-4, the plant starts a slow cooldown driven by the PRHR heat exchanger. Plant procedures may be followed to further cool down the plant.

15.2.7.3 Conclusions

Results of the analysis show that a loss of normal feedwater does not adversely affect the core, the reactor coolant system, or the steam system. The heat removal capacity of the PRHR heat exchanger is such that reactor coolant water is not relieved from the pressurizer safety valves. DNBR always remains above the design limit values, and reactor coolant system and steam generator pressures remain below 110 percent of their design values.

15.2.8 Feedwater System Pipe Break

15.2.8.1 Identification of Causes and Accident Description

A major feedwater line rupture is a break in a feedwater line large enough to prevent the addition of sufficient feedwater to the steam generators in order to maintain shell-side fluid inventory in the steam generators. If the break is postulated in a feedwater line between the check valve and the steam generator, fluid from the steam generator may also be discharged through the break. (A break upstream of the feedwater line check valve would affect the plant only as a loss of feedwater. This case is covered by the evaluation in Subsections 15.2.6 and 15.2.7.)

Depending upon the size of the break and the plant operating conditions at the time of the break, the break could cause either a reactor coolant system cooldown (by excessive energy discharge through the break) or a reactor coolant system heatup. Potential reactor coolant system cooldown resulting from a secondary pipe rupture is evaluated in Subsection 15.1.5. Therefore, only the reactor coolant system heatup effects are evaluated for a feedwater line rupture in this subsection.

The feedwater line rupture reduces the ability to remove heat generated by the core from the reactor coolant system for the following reasons:

- Feedwater flow to the steam generators is reduced. Because feedwater is subcooled, its loss may cause reactor coolant temperatures to increase prior to reactor trip.
- Fluid in the steam generator may be discharged through the break and would not be available for decay heat removal after trip.
- The break may be large enough to prevent the addition of main feedwater after trip.

A major feedwater line rupture is classified as a Condition IV event.

The severity of the feedwater line rupture transient depends on a number of system parameters, including the break size, initial reactor power, and the functioning of various control and safety-related systems. Sensitivity studies presented in WCAP-9230 (Reference 4) illustrate that the most limiting feedwater line rupture is a double-ended rupture of the largest feedwater line. At the beginning of the transient, the main feedwater control system is assumed to malfunction due to an adverse environment. Interactions between the break and the main feedwater control system result in no feedwater flow being injected or lost through the steam generator feedwater nozzles. This assumption causes the water levels in both steam generators to decrease equally until the low steam generator level (narrow range) reactor trip setpoint is reached. After reactor trip, a full double-ended rupture of the feedwater line is assumed such that the faulted steam generator blows down through the break and no main feedwater is delivered to the intact steam generator. These assumptions conservatively bound the most limiting feedwater line rupture that can occur. Analysis is performed at full power assuming the loss of offsite power at the time of the reactor trip. This is more conservative than the case where power is lost at the initiation of the event. The case with offsite power available is not presented because, due to the fast core makeup tanks actuation (on an “S” signal generated by the low steam line pressure), the reactor coolant pumps are tripped by the protection and safety monitoring system a few seconds after the reactor trip. The only difference between the cases with and without offsite power available is the operating status of the reactor coolant pumps.

The following provides the protection for a main feedwater line rupture:

- A reactor trip on any of the following four conditions:
  - High pressurizer pressure
  - Overtemperature ΔT
  - High-3 pressurizer water level
  - Low steam generator water level in either steam generator
  - “S” signals from either of the following:
    - Two out of four low steam line pressure in either steam generator
    - Two out of four high containment pressure (high-2)

Refer to Sections 7.1 and 7.2 for a description of the actuation system.

The PRHR heat exchanger functions to:

- Provide a passive method for decay heat removal. The heat exchanger is a C-tube type, located inside the IRWST. The heat exchanger is above the reactor coolant system to provide natural circulation of the reactor coolant. Operation of the PRHR heat exchanger is initiated by the opening of one of the two parallel power-operated valves at the PRHR heat exchanger cold leg.

- Prevent substantial overpressurization of the reactor coolant system (less than 110 percent of design pressures).

- Maintain sufficient liquid in the reactor coolant system so that the core remains in place, and geometrically intact, with no loss of core cooling capability.

Refer to Subsection 6.3.2.2.5 for a description of the PRHR heat exchanger.

15.2.8.2 Analysis of Effects and Consequences

15.2.8.2.1 Method of Analysis

An analysis using a modified version, described in WCAP-15644 (Reference 6), of the LOFTRAN code (Reference 2) is performed to determine the plant transient following a feedwater line rupture. The code describes the reactor thermal kinetics, reactor coolant system (including natural circulation), pressurizer, steam generators, and feedwater system responses and computes pertinent variables, including the pressurizer pressure, pressurizer water level, and reactor coolant average temperature.

The case analyzed assumes a double-ended rupture of the largest feedwater pipe at full power. Major assumptions used in the analysis are as follows:

- The plant is initially operating at 102 percent of the design plant rating. The main feedwater flow measurement supports a 1-percent power uncertainty; use of a 2-percent power uncertainty is conservative.
- Initial reactor coolant average temperature is 6.5°F above the nominal value, and the initial pressurizer pressure is 50 psi below its nominal value.
- The pressurizer spray is turned on.
- Initial pressurizer level is at a conservative maximum value and a conservative initial steam generator water level is assumed in both steam generators.
- No credit is taken for the high pressurizer pressure reactor trip.
- At the start of the transient, interaction between the break in the feedline and the main feedwater control system is assumed to result in a complete loss of feedwater flow to both steam generators. No feedwater flow is delivered to or lost through the steam generator nozzles.
- Reactor trip is assumed to be initiated when the low steam generator narrow range level setpoint is reached on the ruptured steam generator.
- After reactor trip, the faulted steam generator blows down through a double-ended break area of 1.755 ft^2. A saturated liquid discharge is assumed until all the water inventory is discharged from the faulted steam generator. This minimizes the heat removal capability of the faulted steam generator and maximizes the resultant heatup of the reactor coolant. No feedwater flow is assumed to be delivered to the intact steam generator.
- The PRHR heat exchanger is actuated by the low steam generator water level (wide range) signal. A 15-second delay is assumed following the low level signal to allow time for the alignment of PRHR heat exchanger valves.
- Credit is taken for heat energy deposited in reactor coolant system metal during the reactor coolant system heatup.
- No credit is taken for charging or letdown.
- Pressurizer safety valve setpoint is assumed to be at its minimum value.
- Steam generator heat transfer area is assumed to decrease as the shell-side liquid inventory decreases. The heat transfer remains approximately 100 percent in the faulted steam generator until the liquid mass reaches about 11 percent. The heat transfer is then reduced to 0 percent with the liquid inventory.
- Conservative core residual heat generation is assumed based upon long-term operation at the initial power level preceding the trip (Reference 3).
- No credit is taken for the following four protection and safety monitoring system reactor trip signals to mitigate the consequences of the accident:
  - High pressurizer pressure

– Overtemperature ΔT
– High pressurizer level
– High containment pressure

The PRHR heat exchanger is initiated if the steam generator water level drops to the low steam generator level (wide range). Similarly, receipt of a low steam line pressure signal in at least one steam line initiates a steam line isolation signal that closes all main steam line and feed line isolation valves. This signal also gives an "S" signal that initiates flow of cold borated water from the core makeup tanks to the reactor coolant system.

Plant characteristics and initial conditions are further discussed in Subsection 15.0.3.

The plant control system is not assumed to function in order to mitigate the consequences of the event. The protection and safety monitoring system is required to function following a feedwater line rupture as analyzed here. No single active failure prevents operation of this system.

The engineered safety features assumed to function are the PRHR heat exchanger, core makeup tank, and steam line isolation valves. The single failure assumed is the failure of one of the two parallel discharge valves in the PRHR outlet line (see Table 15.0-7).

For the case without offsite power, there is a flow coastdown until flow in the loops reaches the natural circulation value. The natural circulation capability of the reactor coolant system is shown (see Subsection 15.2.6) to be sufficient to remove core decay heat following reactor trip for the loss of ac power transient. Pump coastdown characteristics are demonstrated in Subsections 15.3.1 and 15.3.2 for single and multiple reactor coolant pump trips, respectively.

A description and analysis of the core makeup tank is provided in Subsection 6.3.2.2.1. The PRHR heat exchanger is described in Subsection 6.3.2.2.5.

15.2.8.2.2 Results

Calculated plant parameters following a major feedwater line rupture are shown in Figures 15.2.8-1 through 15.2.8-10. The calculated sequence of events for the case analyzed is listed in Table 15.2-1.

The results presented in Figures 15.2.8-5 and 15.2.8-7 show that pressures in the reactor coolant system and main steam system remain below 110 percent of the respective design pressure. Pressurizer pressure decreases after reactor trip on the low steam generator water level (70.3 seconds) due to the loss of heat input.

In the first part of the transient, due to the conservative analysis assumptions, the system response following the feedwater line rupture is similar to the loss of ac power to the station auxiliaries (Subsection 15.2.6). The DNB results, presented in Figure 15.2.6-12 for the loss of ac power to plant auxiliaries, are also applicable to a feedwater system pipe break and demonstrate that the DNB design basis is met.

After the trip, the core makeup tanks are actuated (95 seconds) on low steam line pressure in the ruptured loop while the PRHR heat exchanger is actuated on a low steam generator water level wide range (90.1 seconds).

The addition of the PRHR heat exchanger and the core makeup tanks flow rates helps to cool down the primary system and to provide sufficient fluid to keep the core covered with water.

Pressurizer safety valves open due to the mismatch between decay heat and the heat transfer capability of the PRHR heat exchanger. In the first part of the transient, there is a cooling effect due to the core makeup tanks that inject cold water into the reactor coolant system and receive hot water from the cold leg. This effect decreases due to the heatup of the core makeup tanks from recirculation flow. Also, the injection driving head is lowered as the core makeup tanks heat up.

Reactor coolant system temperatures are low (approximately 510°F at about 2,500 seconds) and, in this condition, the PRHR heat exchanger cannot remove the entire decay heat load. Reactor coolant system temperatures increase until an equilibrium between decay heat power and heat absorbed by the PRHR heat exchanger is reached. After about 11,300 seconds, the heat transfer capability of the PRHR heat exchanger exceeds the decay heat power and the reactor coolant system temperatures, pressure, and pressurizer water volumes start to steadily decrease. Core cooling capability is maintained throughout the transient because reactor coolant system inventory is increasing due to core makeup tank injection.

15.2.8.3 Conclusions

Results of the analyses show that for the postulated feedwater line rupture, the capacity of the PRHR heat exchanger is adequate to remove decay heat, to prevent overpressurizing the reactor coolant system, and to maintain the core cooling capability. Radioactivity doses from ruptures of the postulated feedwater lines are less than those presented for the postulated main steam line break. The Standard Review Plan, Subsection 15.2.8, evaluation criteria are therefore met.

15.2.9 Combined License Information

This section contained no requirement for additional information.

15.2.10 References

1. Cooper, L., Miselis, V., and Starek, R. M., “Overpressure Protection for Westinghouse Pressurized Water Reactors,” WCAP-7769, Revision 1, June 1972. (Also letter NS-CE-622, C. Eicheldinger (Westinghouse) to D. B. Vassallo (NRC), additional information on WCAP-7769, Revision 1, April 16, 1975).

2. Burnett, T. W. T., et al., “LOFTRAN Code Description,” WCAP-7907-P-A (Proprietary) and WCAP-7907-A (Nonproprietary), April 1984.

3. “American National Standard for Decay Heat Power in Light Water Reactors,” ANSI/ANS-5.1-1979, August 1979.

4. Lang, G. E., and Cunningham, J. P., “Report on the Consequences of a Postulated Main Feedline Rupture,” WCAP-9230 (Proprietary) and WCAP-9231 (Nonproprietary), January 1978.

5. Friedland, A. J., and Ray, S., “Revised Thermal Design Procedure,” WCAP-11397-P-A (Proprietary) and WCAP-11397-A (Nonproprietary), April 1989.

6. “AP1000 Code Applicability Report,” WCAP-15644-P (Proprietary) and WCAP-15644-NP (Nonproprietary), Revision 2, March 2004.

7. Hargrove, H. G., “FACTRAN – A FORTRAN-TV Code for Thermal Transients in a UO_2 Fuel Rod,” WCAP-7908-A, December 1989.

8. Matthys, C., “Overpressure Protection Report for AP1000 Nuclear Power Plant,” WCAP-16779-NP, April 2007.

Table 15.2-1 (Sheet 1 of 7)
Time Sequence of Events for Incidents Which
Result in a Decrease in Heat Removal By the Secondary System

Accident	Event	Time (seconds)
I. Turbine trip		
A.1. With pressurizer control, minimum reactivity feedback, with offsite power available	Turbine trip; loss of main feedwater	0.0
	Minimum DNBR occurs	0.0
	High pressurizer pressure reactor trip point reached	6.2
	Rods begin to drop	8.2
	Peak RCS pressure occurs	10.0
	Initiation of steam release from steam generator safety valves	12.4
A.2. With pressurizer control, minimum reactivity feedback, without offsite power available	Turbine trip; loss of main feedwater	0.0
	Offsite power lost, reactor coolant pumps begin coasting down	3.0
	Low reactor coolant pump speed reactor trip setpoint reached	3.47
	Rods begin to drop	4.24
	Minimum DNBR (1.57) occurs	6.0
	Peak RCS pressure occurs	6.3
	Initiation of steam release from steam generator safety valves	18.7

Table 15.2-1 (Sheet 2 of 7)
Time Sequence of Events for Incidents Which Result in a Decrease in Heat Removal By the Secondary System

Accident	Event	Time (seconds)
B.1. With pressurizer control, maximum reactivity feedback, with offsite power available	Turbine trip; loss of main feedwater flow	0.0
	Minimum DNBR occurs	0.0
	High pressurizer pressure reactor trip setpoint reached	6.6
	Rods begin to drop	8.6
	Peak RCS pressure occurs	9.6
	Initiation of steam release from steam generator safety valves	13.0
B.2. With pressurizer control, maximum reactivity feedback, without offsite power available	Turbine trip; loss of main feedwater	0.0
	Offsite power lost, reactor coolant pumps begin coasting down	3.0
	Low reactor coolant pump speed reactor trip setpoint reached	3.47
	Rods begin to drop	4.24
	Minimum DNBR (2.44) occurs	4.4
	Peak RCS pressure occurs	7.7
	Initiation of steam release from steam generator safety valves	24.9

Table 15.2-1 (Sheet 3 of 7)
Time Sequence of Events for Incidents Which Result in a Decrease in Heat Removal By the Secondary System

Accident	Event	Time (seconds)
C.1. Without pressurizer control, minimum reactivity feedback, with offsite power available	Turbine trip; loss of main feedwater flow	0.0
	High pressurizer pressure reactor trip point reached	5.9
	Rods begin to drop	7.9
	Peak RCS pressure occurs	9.5
	Initiation of steam release from steam generator safety valves	10.5
C.2. Without pressurizer control, minimum reactivity feedback, without offsite power available	Turbine trip; loss of main feedwater	0.0
	Offsite power lost, reactor coolant pumps begin coasting down	3.0
	Low reactor coolant pump speed reactor trip setpoint reached	3.47
	Rods begin to drop	4.24
	Peak RCS pressure occurs	6.3
	Initiation of steam release from steam generator safety valves	14.0

Table 15.2-1 (Sheet 4 of 7)
Time Sequence of Events for Incidents Which Result in a Decrease in Heat Removal By the Secondary System

Accident	Event	Time (seconds)
D.1. Without pressurizer control, maximum reactivity feedback, with offsite power available	Turbine trip; loss of main feedwater flow	0.0
	High pressurizer pressure reactor trip	6.0
	Rods begin to drop	8.0
	Peak RCS pressure occurs	8.4
	Initiation of steam release from steam generator safety valves	10.7
D.2. Without pressurizer control, maximum reactivity feedback, without offsite power available	Turbine trip; loss of main feedwater	0.0
	Offsite power lost, reactor coolant pumps begin coasting down	3.0
	Low reactor coolant pump speed reactor trip setpoint reached	3.47
	Rods begin to drop	4.24
	Peak RCS pressure occurs	5.9
	Initiation of steam release from steam generator safety valves	15.6

Table 15.2-1 (Sheet 5 of 7)
Time Sequence of Events for Incidents Which Result in a Decrease in Heat Removal By the Secondary System

Accident	Event	Time (seconds)
II.A. Loss of ac power to the plant auxiliaries	Feedwater is lost	10.0
	Low steam generator water level reactor trip set point is reached	70.4
	Rods begin to drop, ac power is lost, reactor coolant pumps start to coastdown	72.4
	Pressurizer safety valves open	76.5
	Maximum pressurizer pressure reached	77.0
	Steam generator safety valves open	87.0
	PRHR heat exchanger actuation on low steam generator water level (narrow range coincident with low start up flow rate)	132.4
	Maximum pressurizer water volume reached	139.0
	Pressurizer safety valves reclose	142.0
	Steam generator 1 safety valves close	2,326
	Core makeup tank actuation on low T_{cold} “S” signal	4,753
	Steam line isolation on low T_{cold} “S” signal	4,765
	Steam generator 2 safety valves close	7,006
	Pressurizer safety valves open	8,056
	Pressurizer safety valves reclose	16,944
	PRHR heat exchanger extracted heat matches decay heat	~ 19,100
	Second pressurizer water volume peak is reached	22,152

Table 15.2-1 (Sheet 6 of 7)
Time Sequence of Events for Incidents Which Result in a Decrease in Heat Removal By the Secondary System

Accident	Event	Time (seconds)
III. Loss of normal feedwater flow	Feedwater is lost	10.0
	Low steam generator water level (narrow range) reactor trip reached	70.4
	Rods begin to drop	72.4
	Steam generator safety valves open	80.0
	PRHR heat exchanger actuation on low steam generator water level (narrow range coincident with low start up feeedwater flow rate)	132.4
	Steam generator safety valves reclose	144
	Cold leg temperature reaches low T_{cold} setpoint	1,154.6
	Reactor coolant pump trip on low T_{cold} “S” signal	1,160.6
	Steam line isolation on low T_{cold} “S” signal	1,166.6
	Core makeup tank actuation on low T_{cold} “S” signal	1,171.6
	Pressurizer safety valves open	3,500
	Pressurizer safety valves reclose	17,702
	Passive residual heat removal heat exchanger extracted heat matches decay heat	~ 17,620
	Maximum pressurizer water volume reached	19,548

Table 15.2-1 (Sheet 7 of 7)
Time Sequence of Events for Incidents Which Result in a Decrease in Heat Removal By the Secondary System

Accident	Event	Time (seconds)
IV. Feedwater system pipe break	Main feedwater flow to both steam generators stops due to interaction between the break and the main feedwater control system	10.0
	Low steam generator water level (narrow range) setpoint reached	70.3
	Reverse flow from the faulted steam generator through a full double-ended rupture starts	70.3
	Rods begin to drop	72.3
	Loss of offsite power occurs	72.3
	Low steam generator water level (wide range) set point reached	73.1
	Pressurizer safety valves open	74.5
	Low steam line pressure set point reached	78.0
	Pressurizer safety valves close	80.0
	All steam and feedline isolation valves close	90.0
	PRHR heat exchanger actuation on low steam generator water level (wide range)	90.1
	Core makeup tank valves fully opened	95.0
	Faulted steam generator empties	100.0
	Intact steam generator safety valves open	180
	Intact steam generator safety valves close	425
	Pressurizer safety valves open	1,848
	PRHR heat exchanger extracted heat matches decay heat	~ 11,300
	Pressurizer safety valves close	~ 11,300

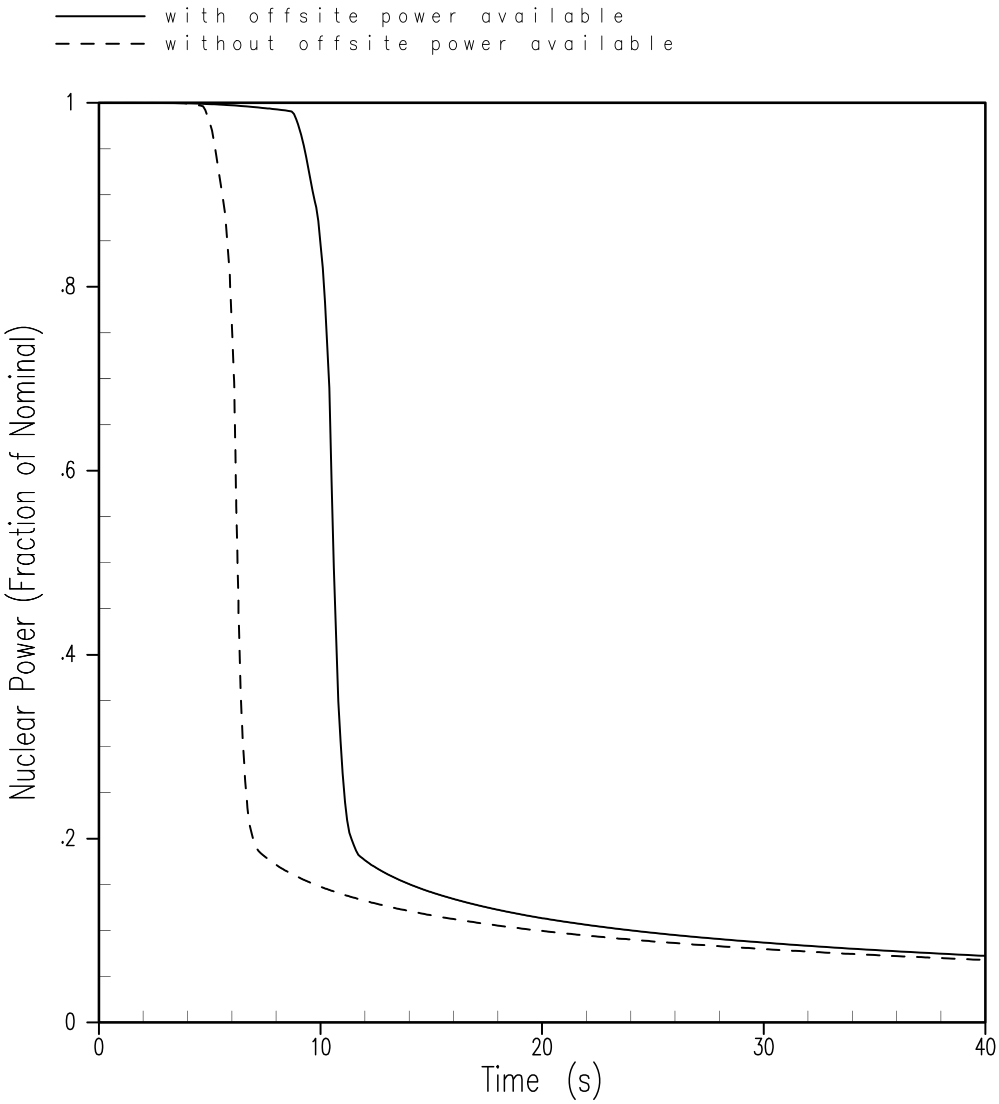


Figure 15.2.3-1
Nuclear Power (Fraction of Nominal) versus Time for Turbine Trip Accident with Pressurizer Spray and Minimum Moderator Feedback

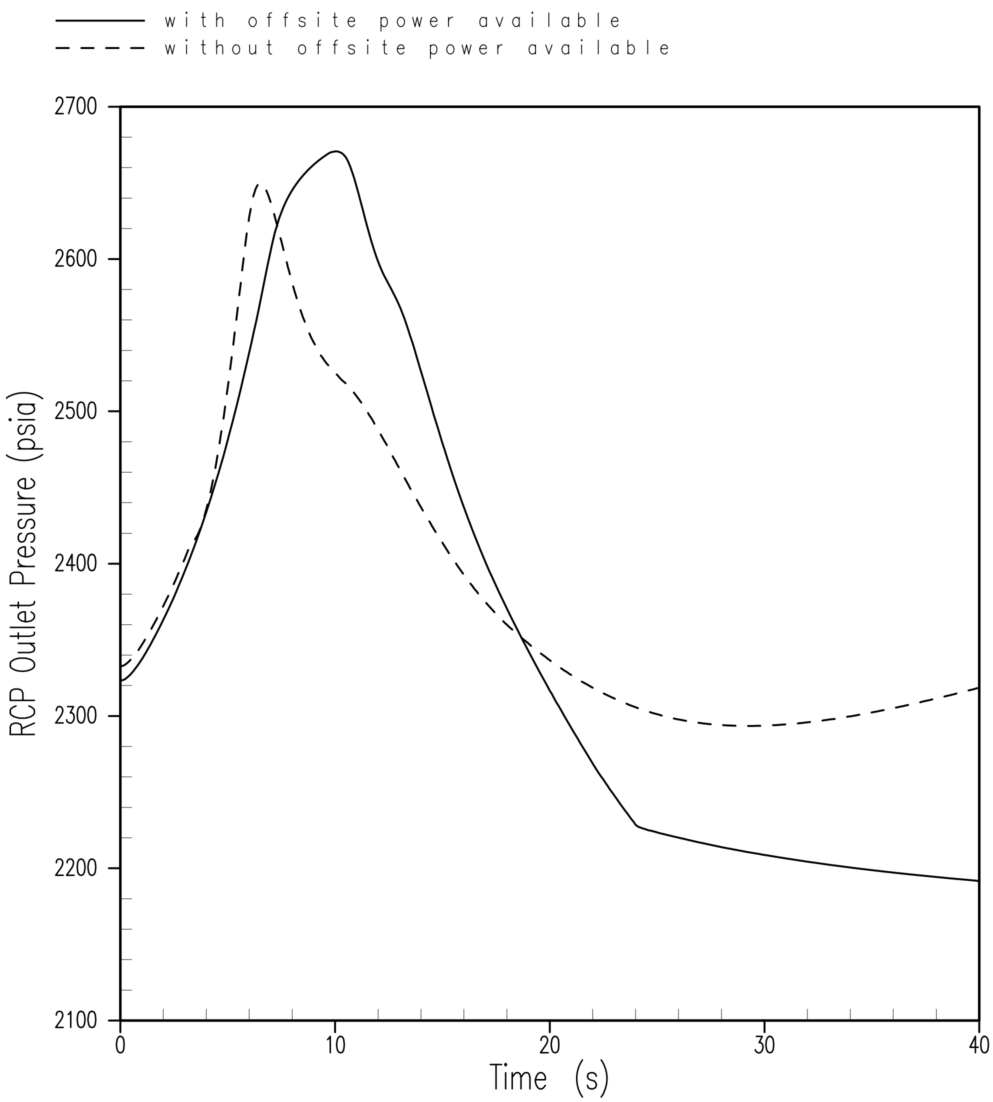


Figure 15.2.3-2
Pressurizer Pressure (psia) versus Time for Turbine Trip Accident with Pressurizer Spray and Minimum Moderator Feedback

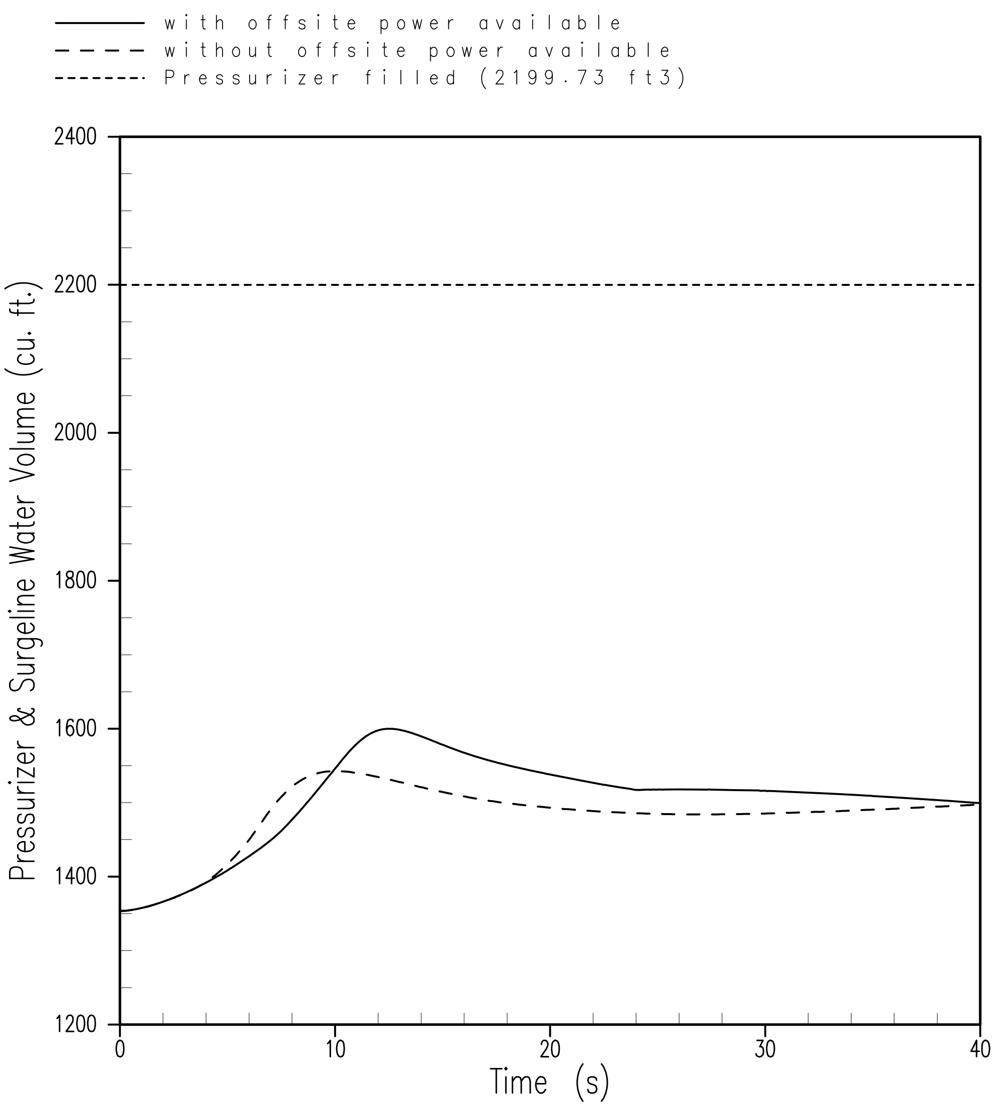


Figure 15.2.3-3
Pressurizer Water Volume (ft^3) versus Time for Turbine Trip Accident with Pressurizer Spray and Minimum Moderator Feedback

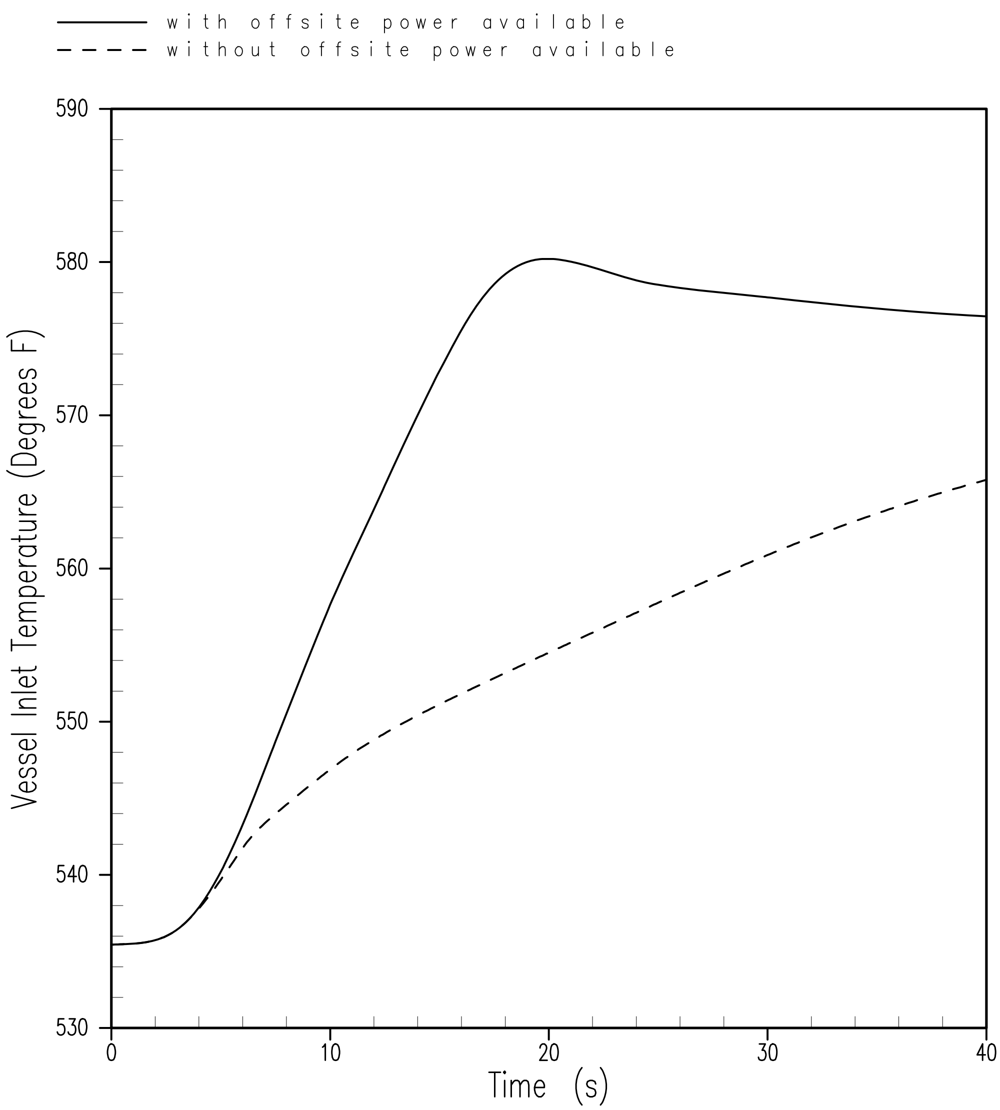


Figure 15.2.3-4
Vessel Inlet Temperature (°F) versus Time for Turbine Trip Accident with Pressurizer Spray and Minimum Moderator Feedback

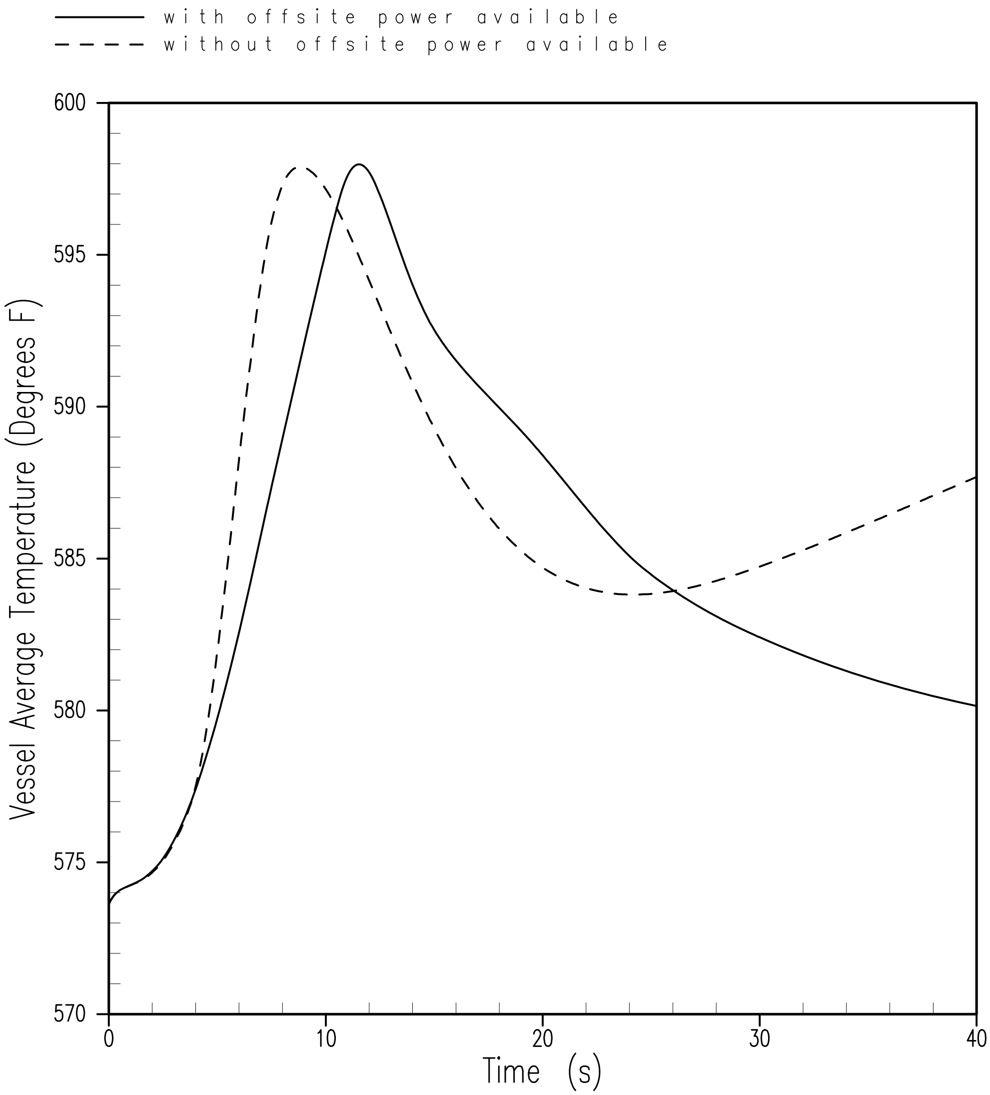


Figure 15.2.3-5
Vessel Average Temperature (°F) versus Time for Turbine Trip Accident with Pressurizer Spray and Minimum Moderator Feedback

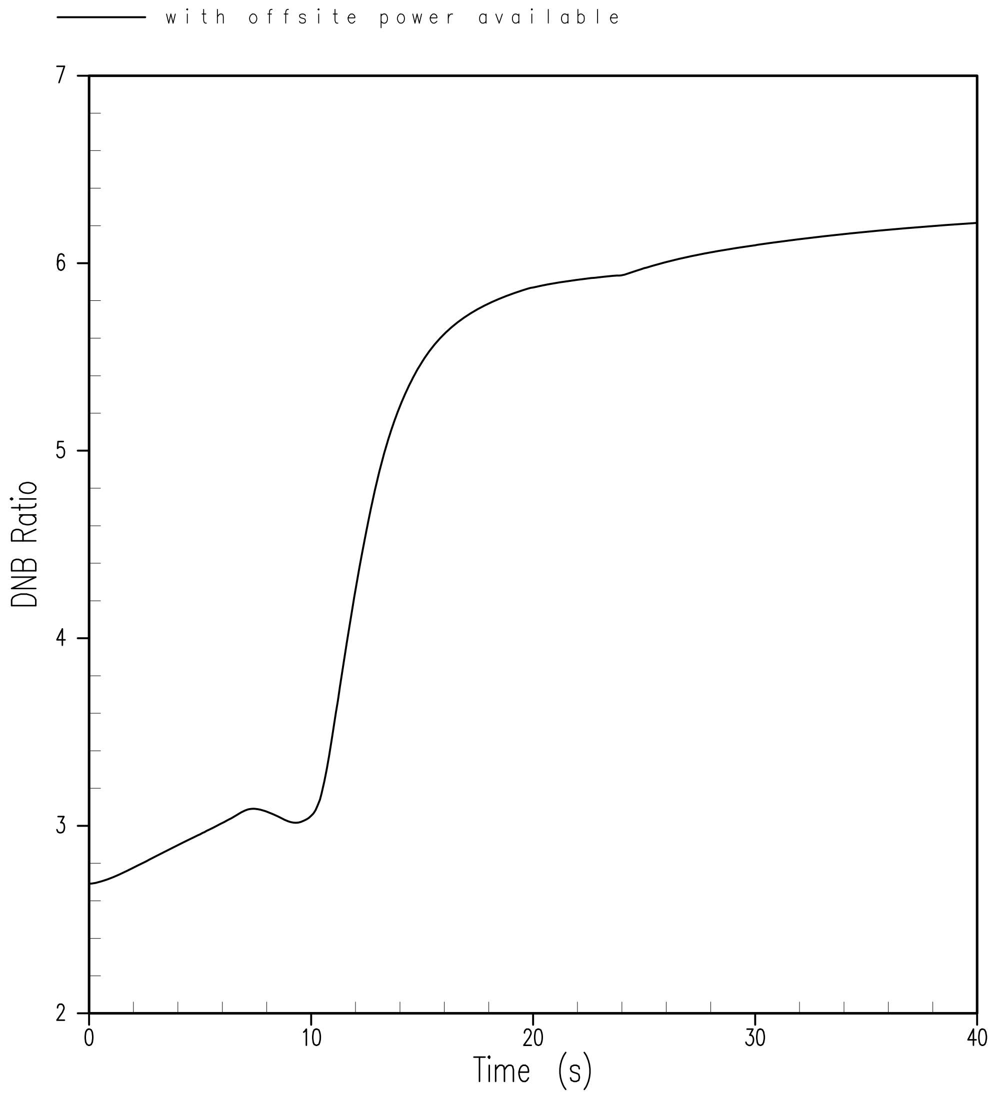


Figure 15.2.3-6
DNBR versus Time for Turbine Trip Accident
with Pressurizer Spray and Minimum Moderator Feedback

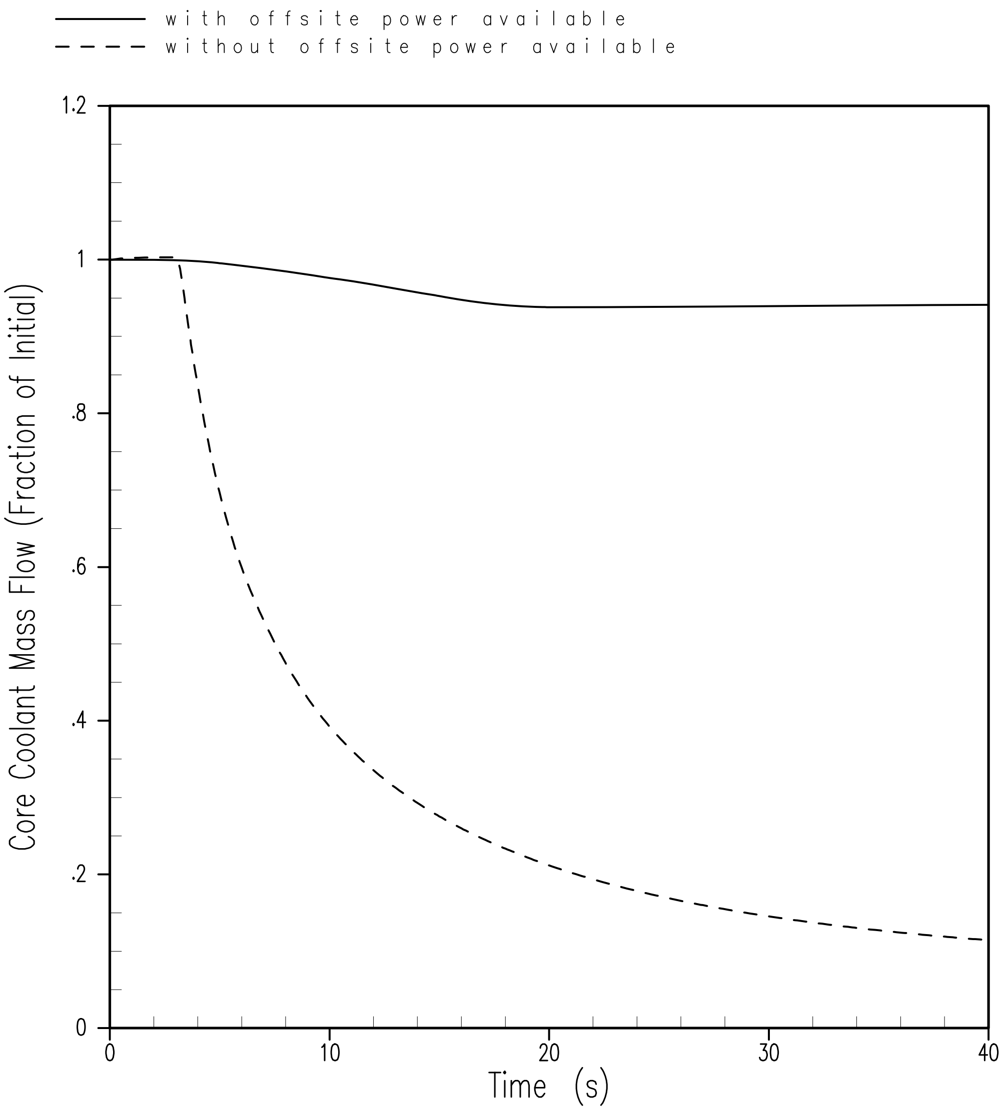


Figure 15.2.3-7
Core Mass Flow Rate (Fraction of Initial) versus Time for Turbine Trip Accident with Pressurizer Spray and Minimum Moderator Feedback

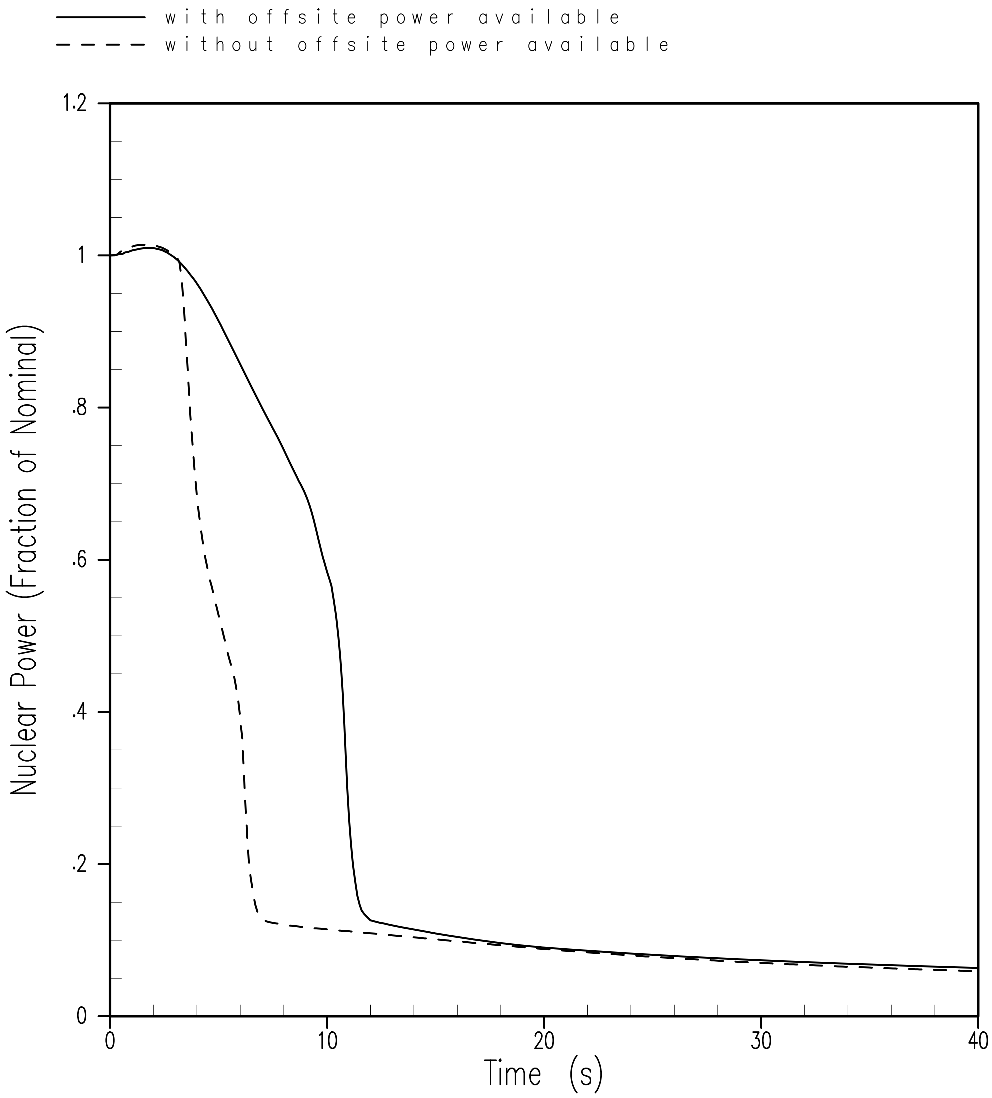


Figure 15.2.3-8
Nuclear Power (Fraction of Nominal) versus Time for Turbine Trip Accident with Pressurizer Spray and Maximum Moderator Feedback

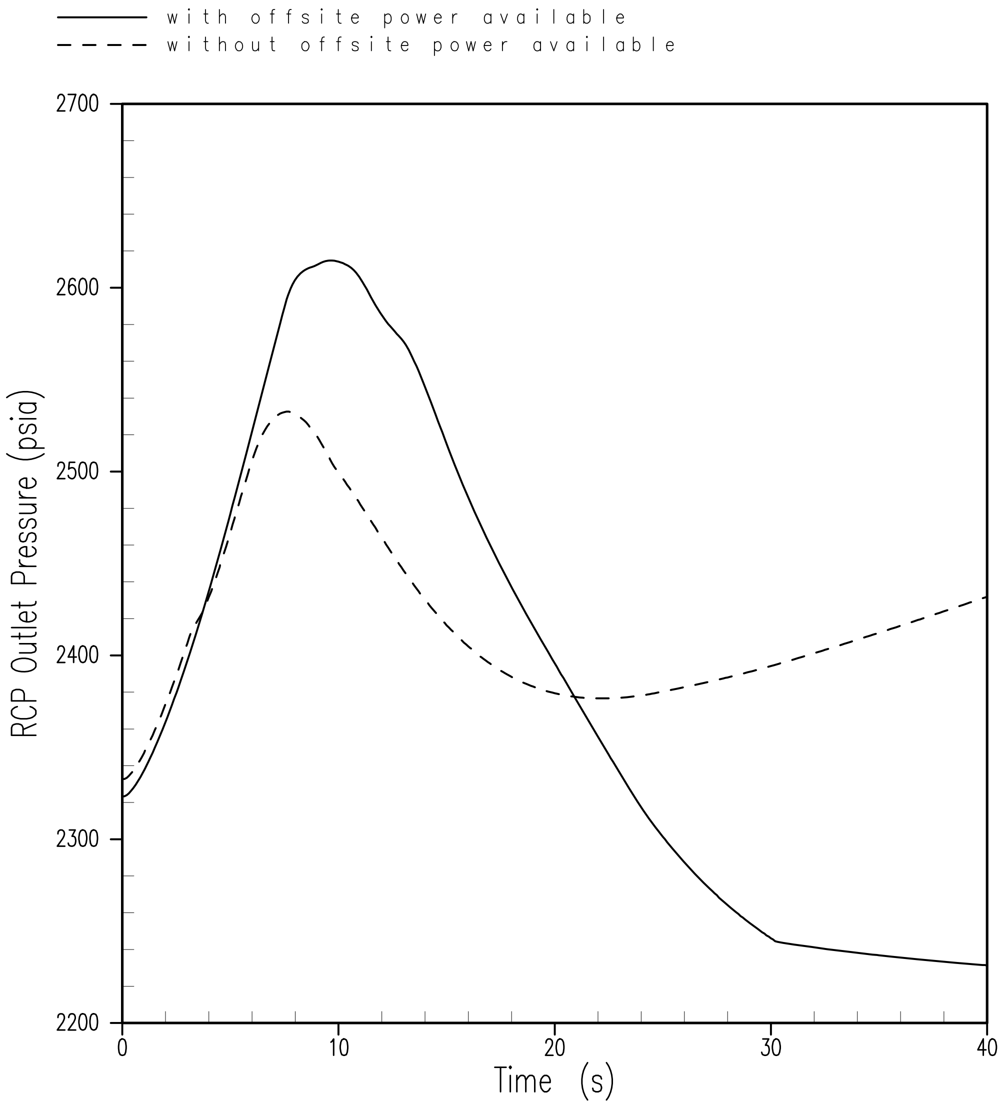


Figure 15.2.3-9
Pressurizer Pressure (psia) versus Time for Turbine Trip
Accident with Pressurizer Spray and Maximum Moderator Feedback

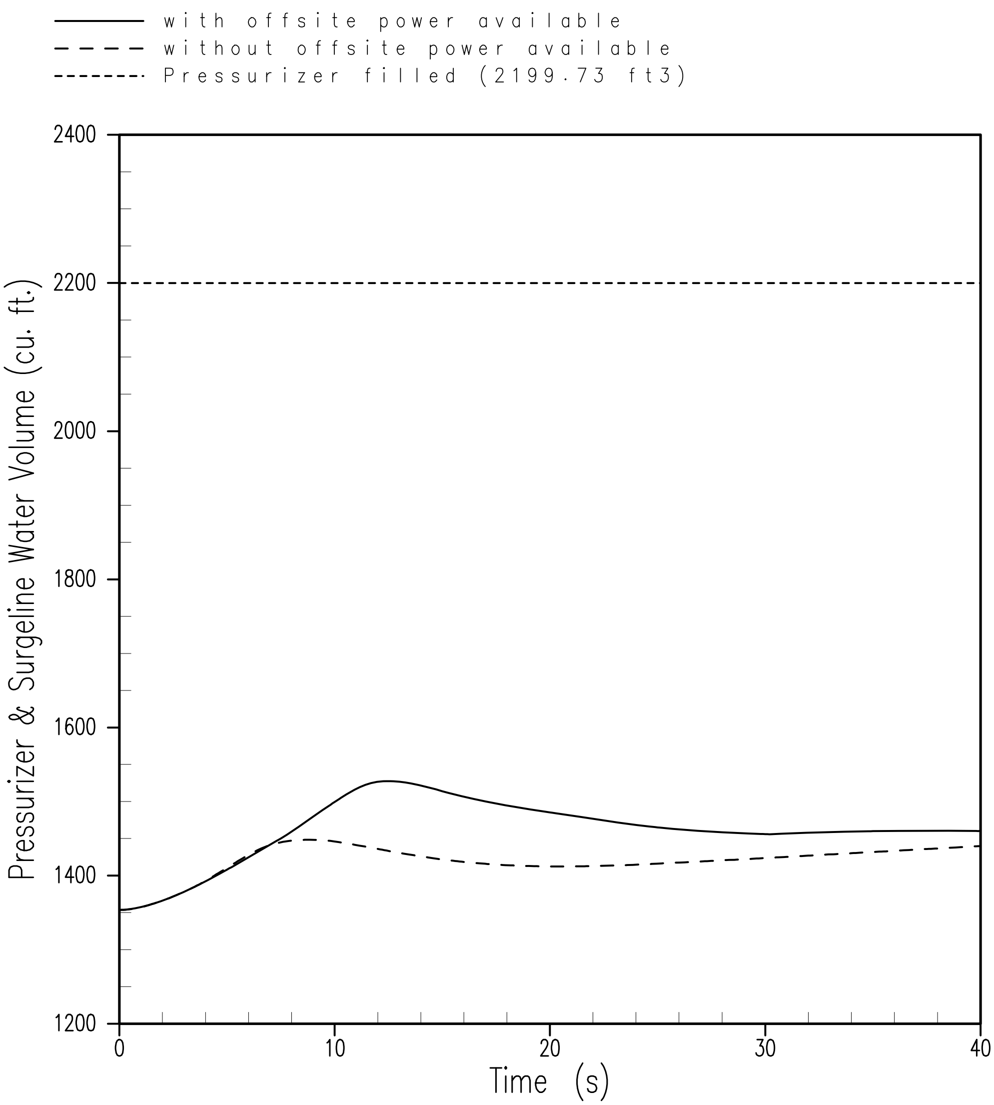


Figure 15.2.3-10
Pressurizer Water Volume (ft^3) versus Time for Turbine Trip Accident with Pressurizer Spray and Maximum Moderator Feedback

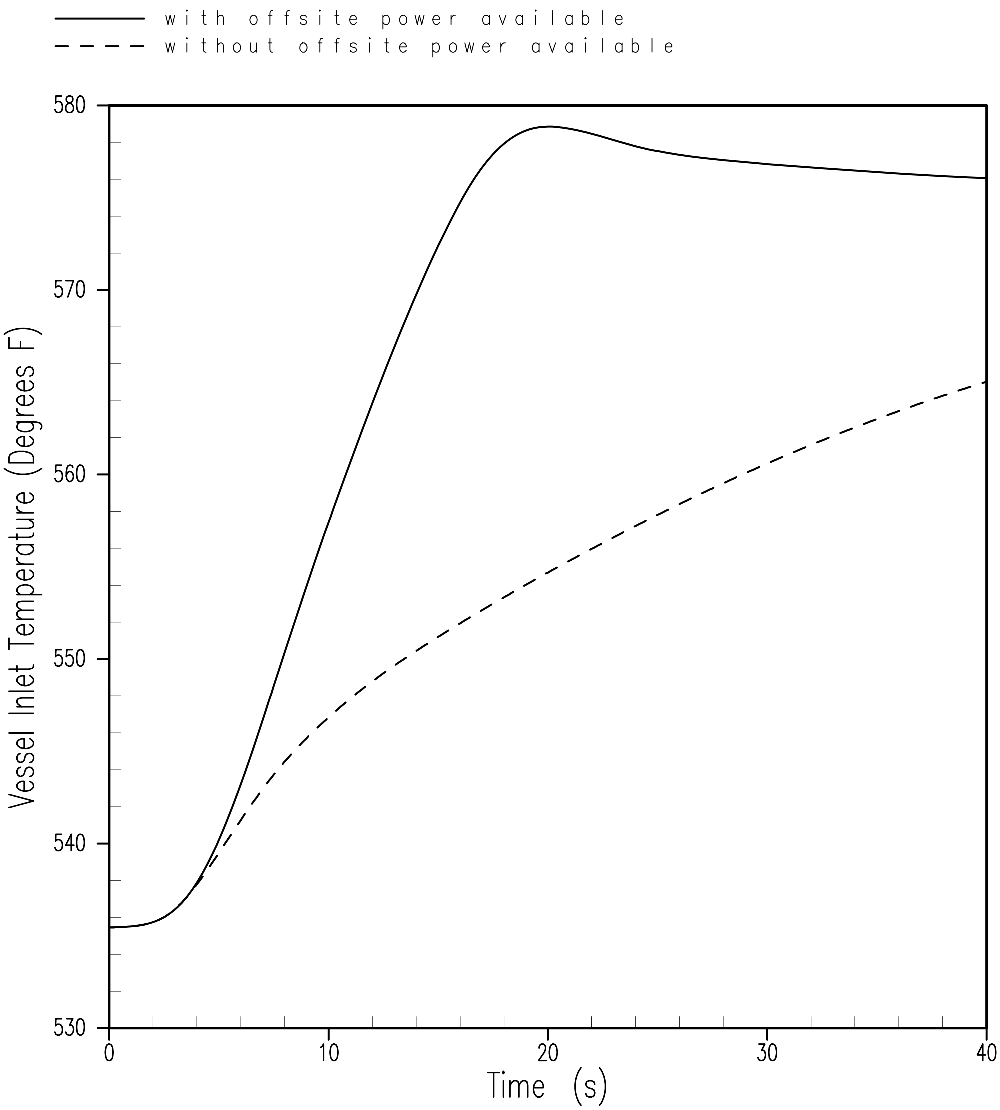


Figure 15.2.3-11
Vessel Inlet Temperature (°F) versus Time for Turbine Trip Accident with Pressurizer Spray and Maximum Moderator Feedback

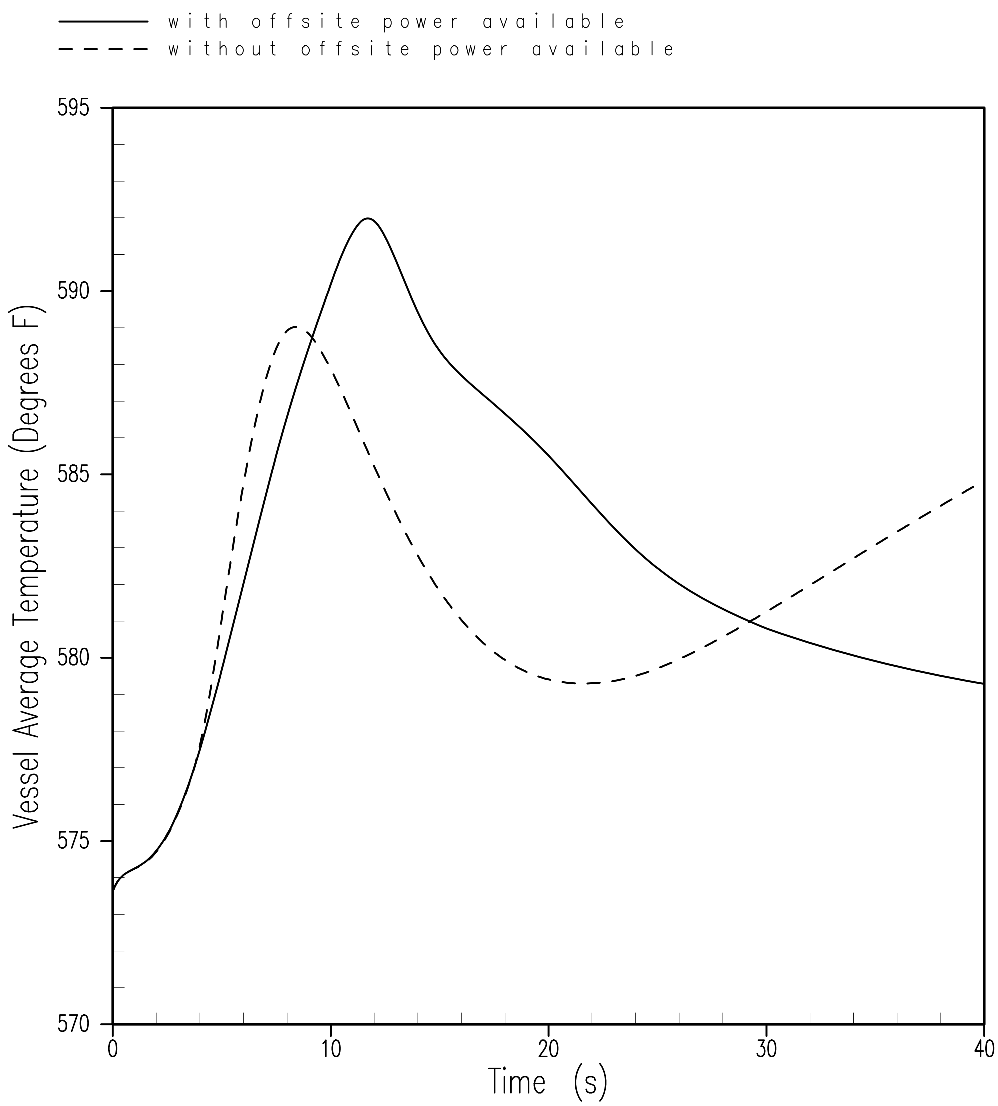


Figure 15.2.3-12
Vessel Average Temperature (°F) versus Time for Turbine Trip Accident with Pressurizer Spray and Maximum Moderator Feedback

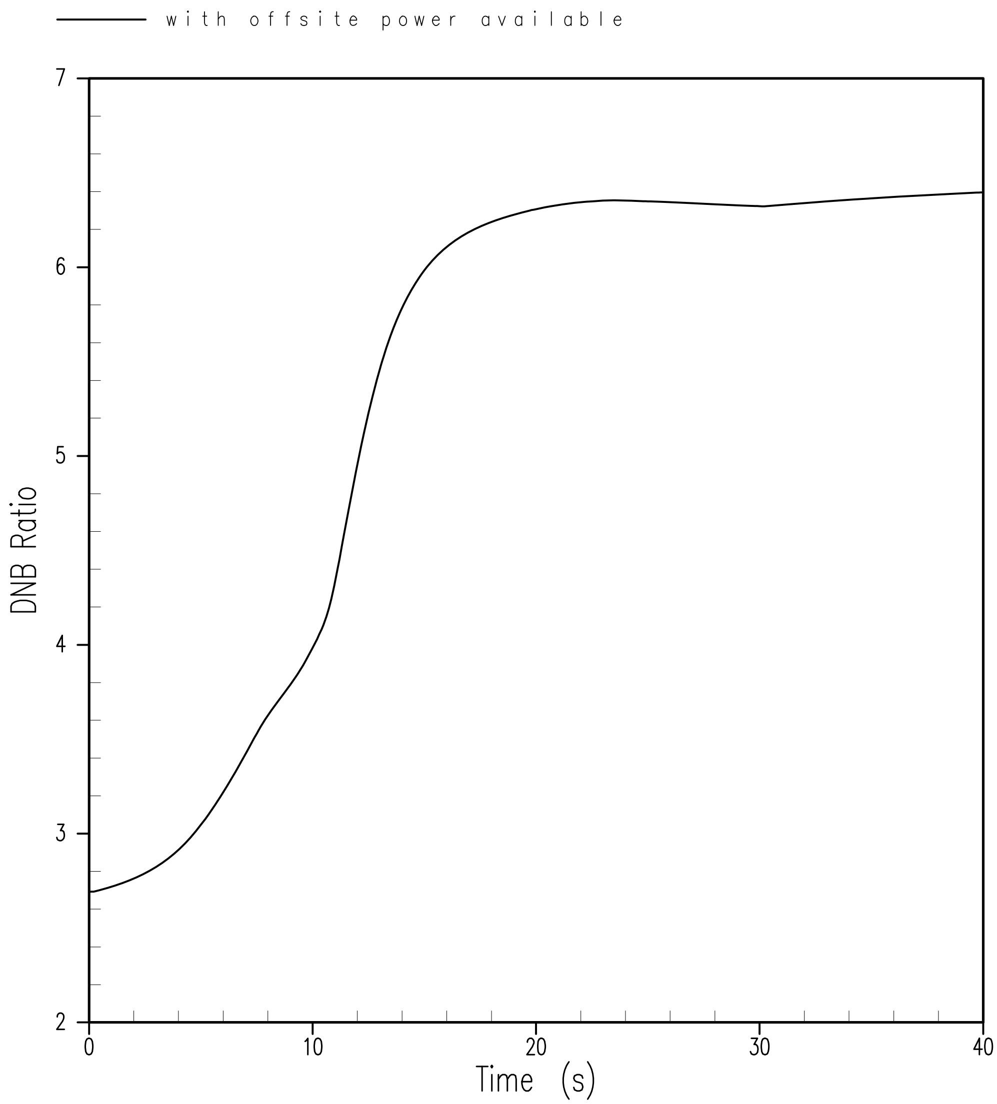


Figure 15.2.3-13
DNBR versus Time for Turbine Trip Accident
with Pressurizer Spray and Maximum Moderator Feedback

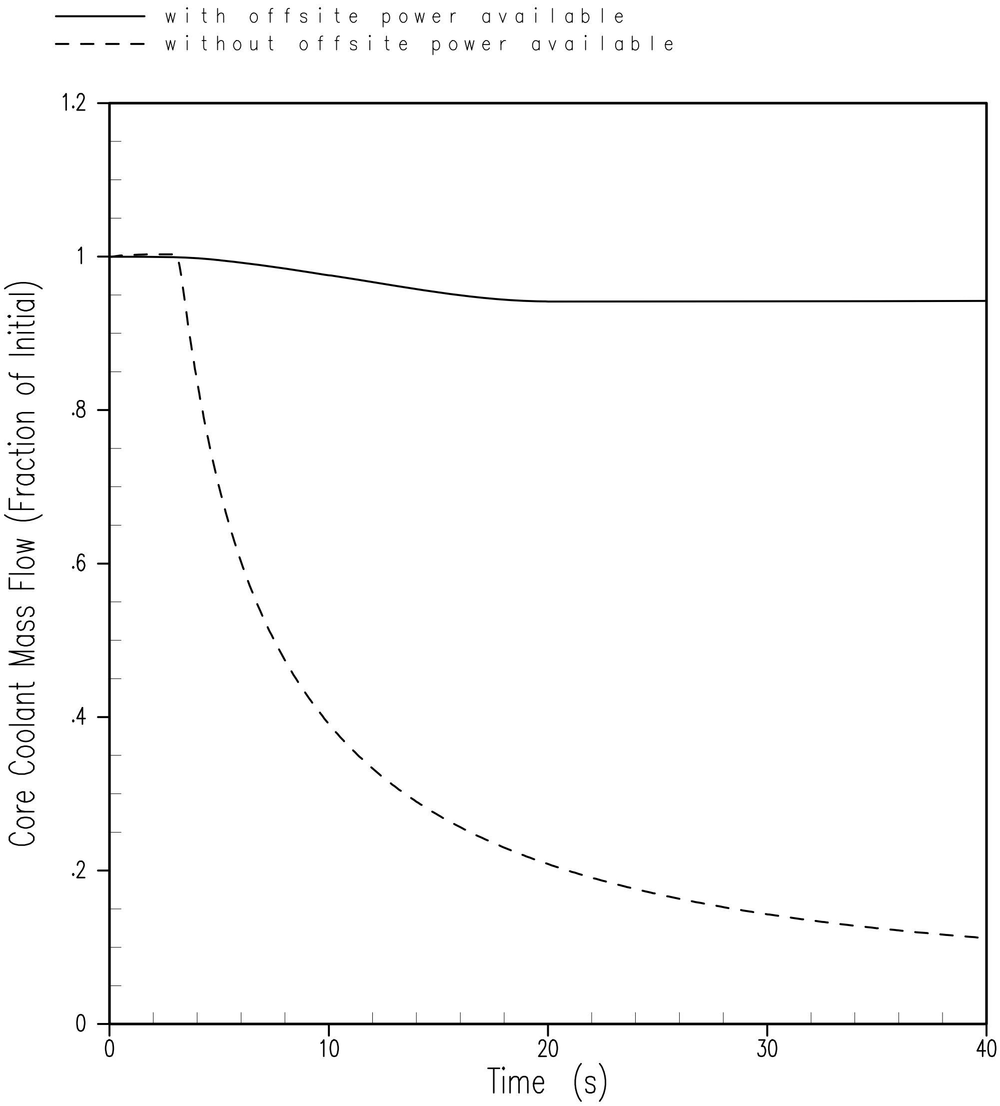


Figure 15.2.3-14
Core Mass Flow Rate (Fraction of Initial) versus Time for Turbine Trip Accident with Pressurizer Spray and Maximum Moderator Feedback

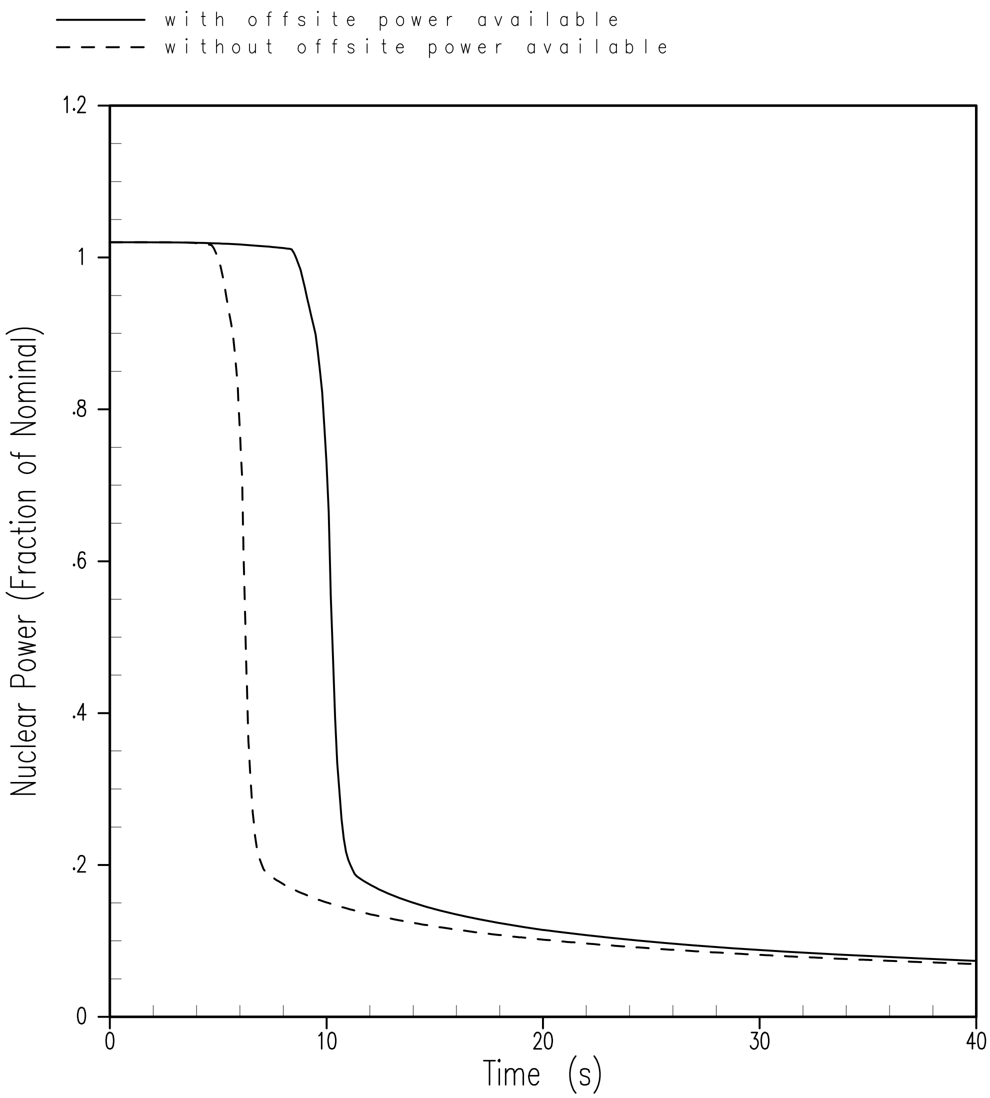


Figure 15.2.3-15
Nuclear Power (Fraction of Nominal) versus Time for Turbine Trip Accident Without Pressurizer Spray and Minimum Moderator Feedback

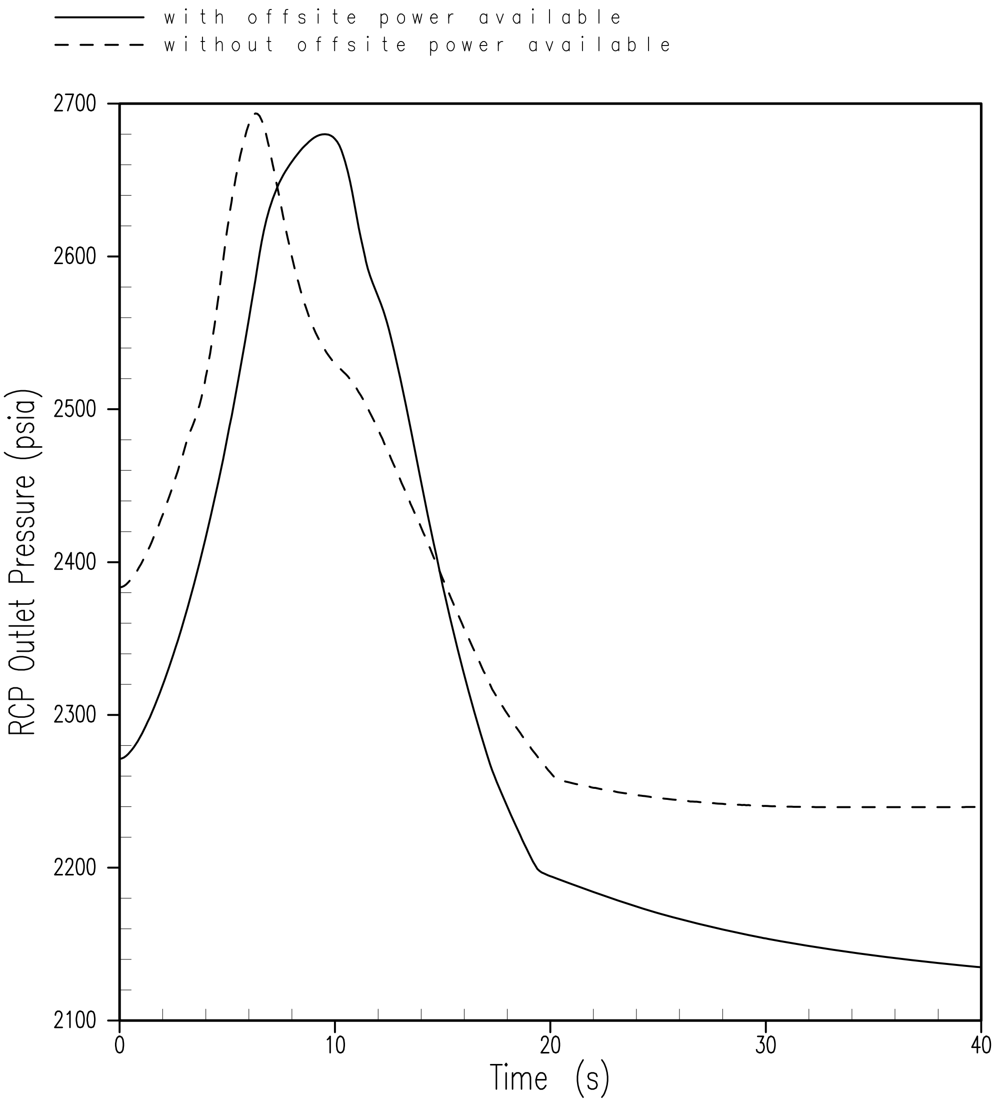


Figure 15.2.3-16
Pressurizer Pressure (psia) versus Time for Turbine Trip Accident Without Pressurizer Spray and Minimum Moderator Feedback

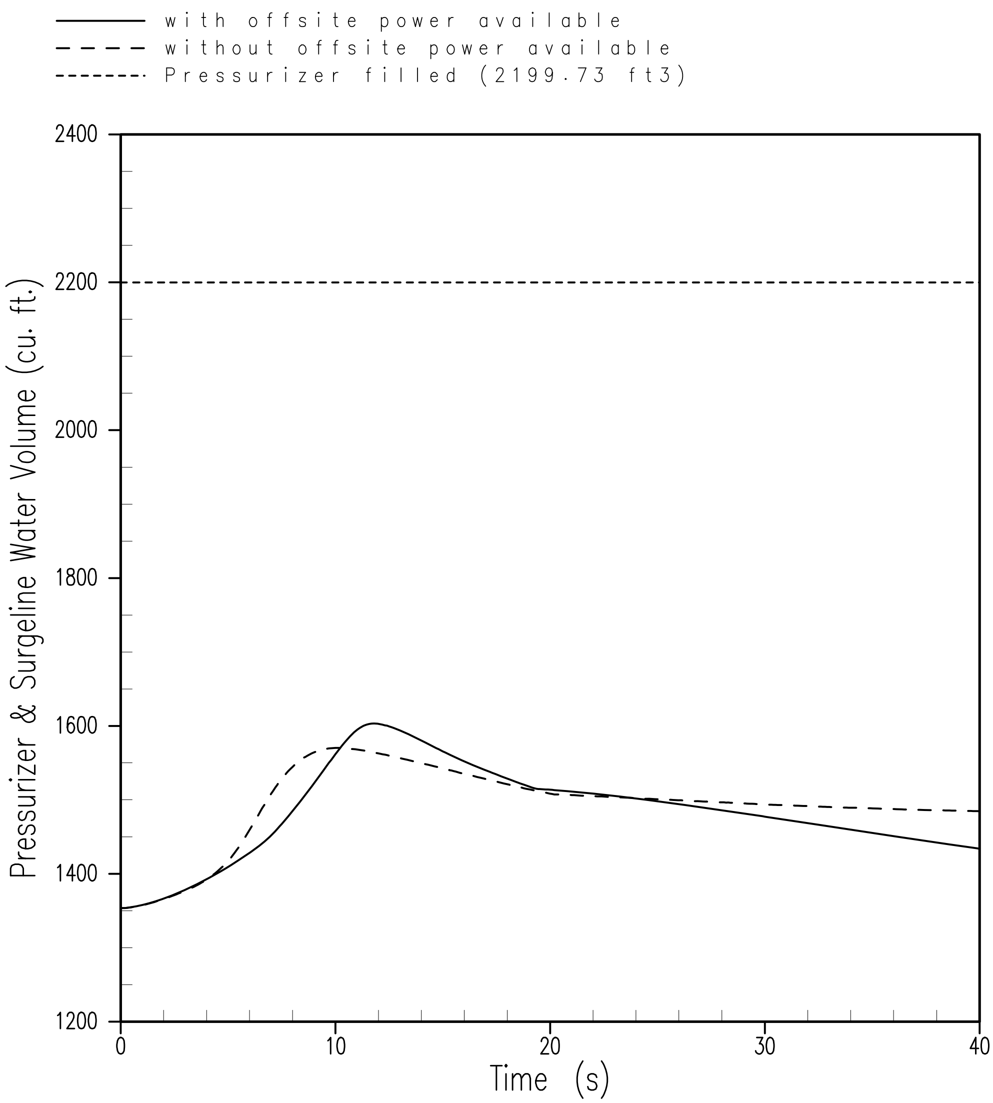


Figure 15.2.3-17
Pressurizer Water Volume (ft^3) versus Time for Turbine Trip Accident Without Pressurizer Spray and Minimum Moderator Feedback

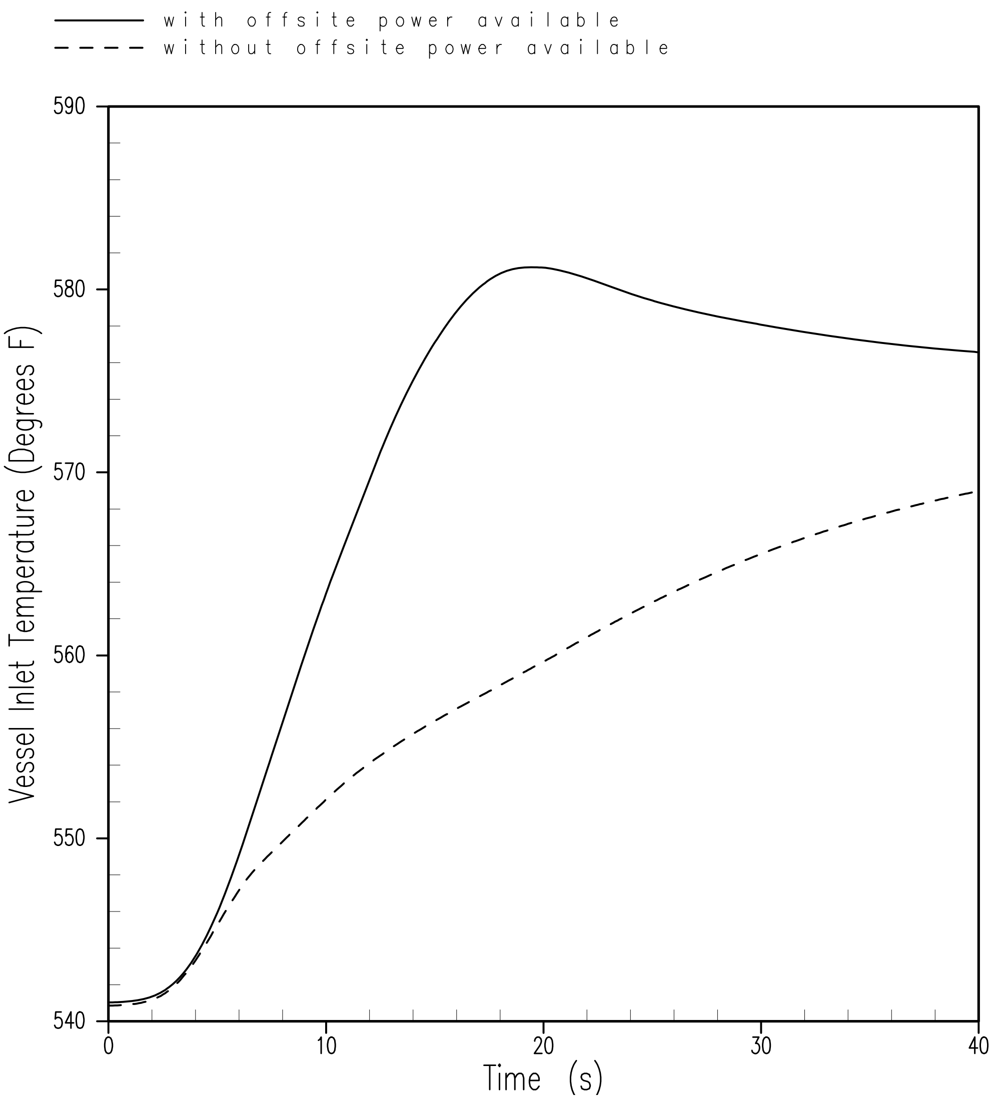


Figure 15.2.3-18
Vessel Inlet Temperature (°F) versus Time for Turbine Trip Accident Without Pressurizer Spray and Minimum Moderator Feedback

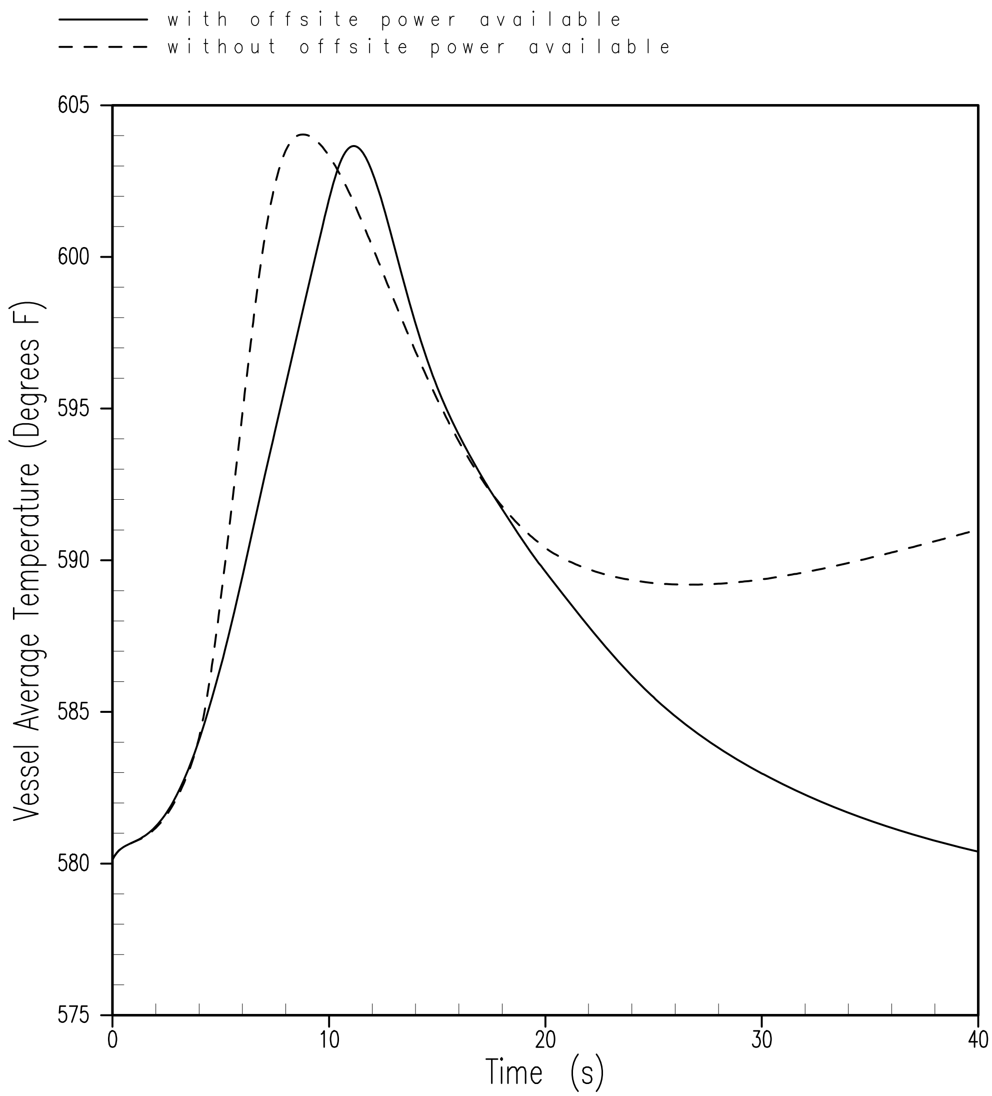


Figure 15.2.3-19
Vessel Average Temperature (°F) versus Time for Turbine Trip Accident Without Pressurizer Spray and Minimum Moderator Feedback

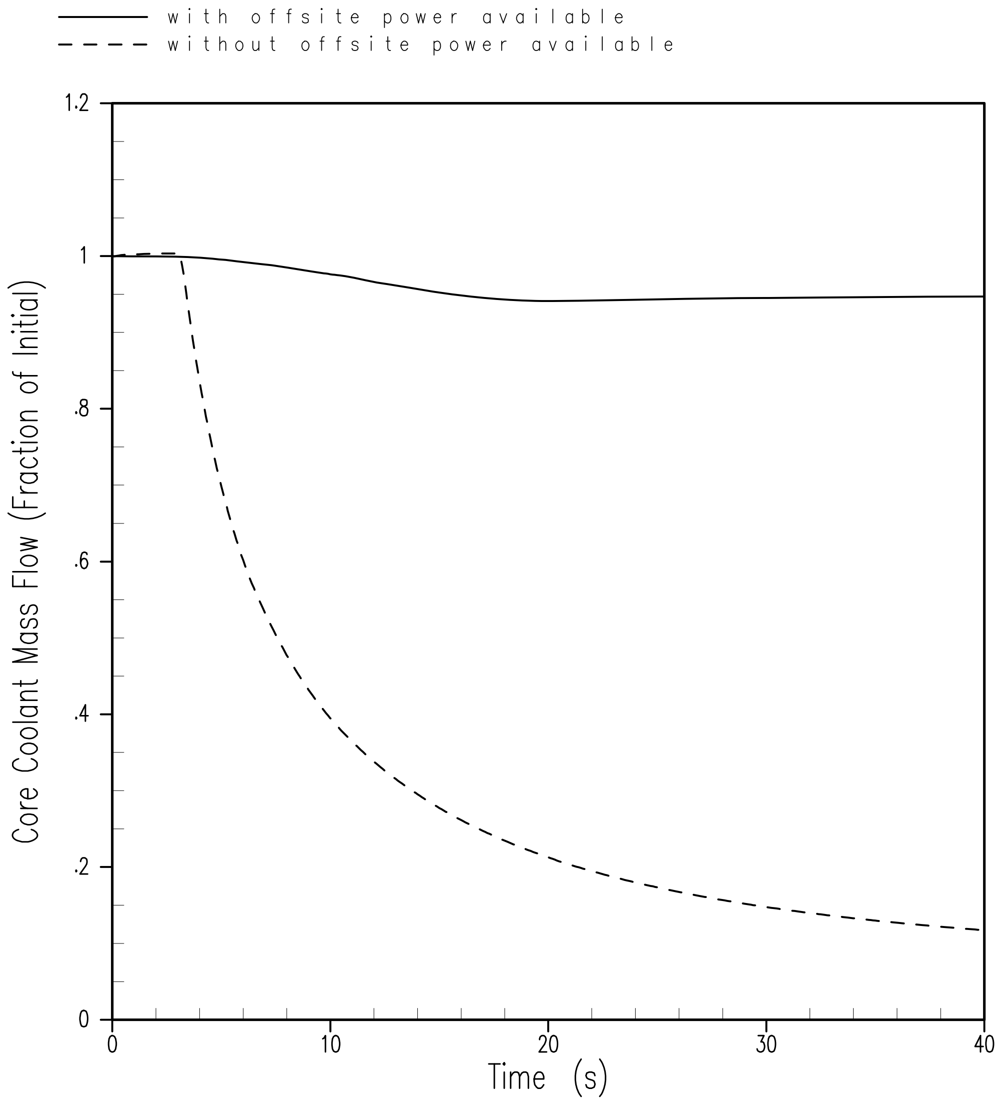


Figure 15.2.3-20
Core Mass Flow Rate (Fraction of Initial) versus Time for Turbine Trip Accident Without Pressurizer Spray and Minimum Moderator Feedback

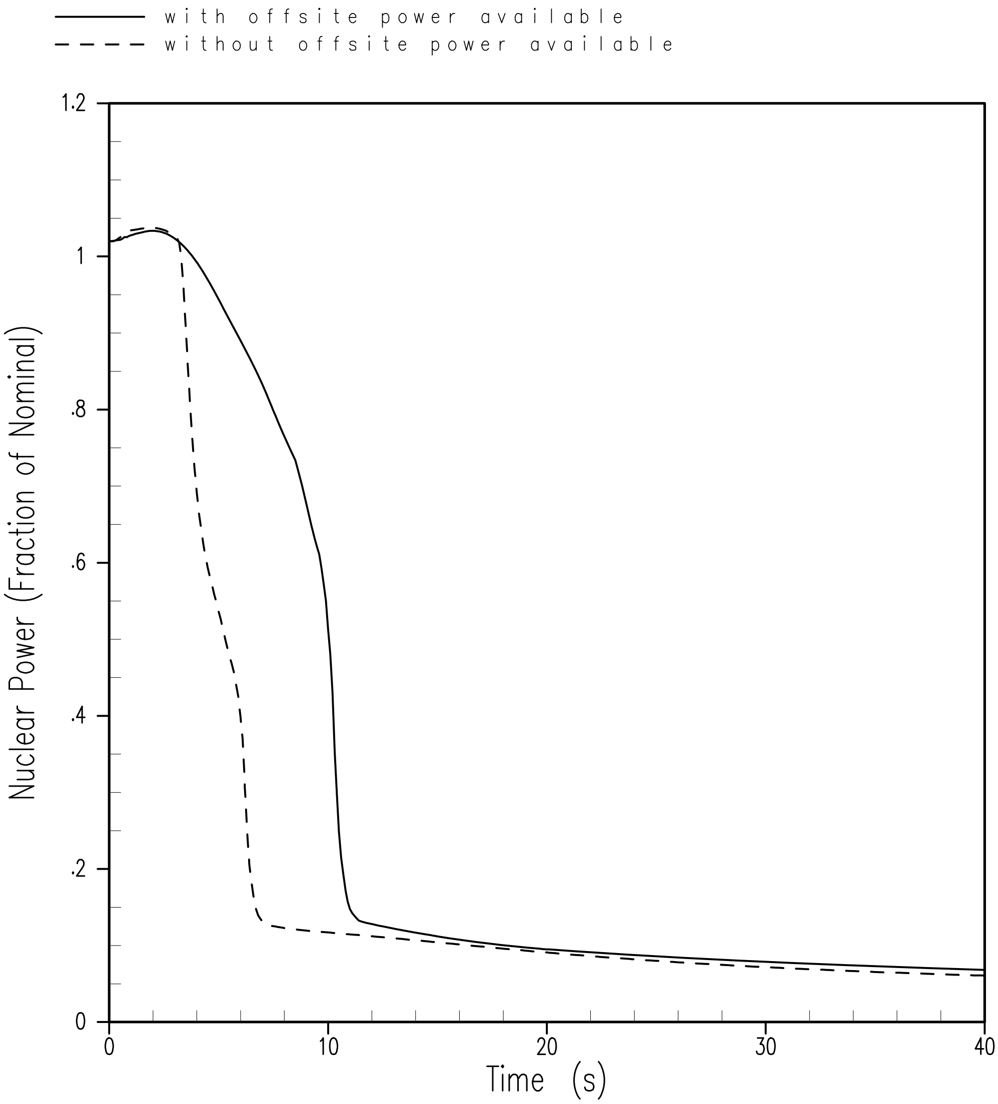


Figure 15.2.3-21
Nuclear Power (Fraction of Nominal) versus Time for Turbine Trip Accident Without Pressurizer Spray and Maximum Moderator Feedback

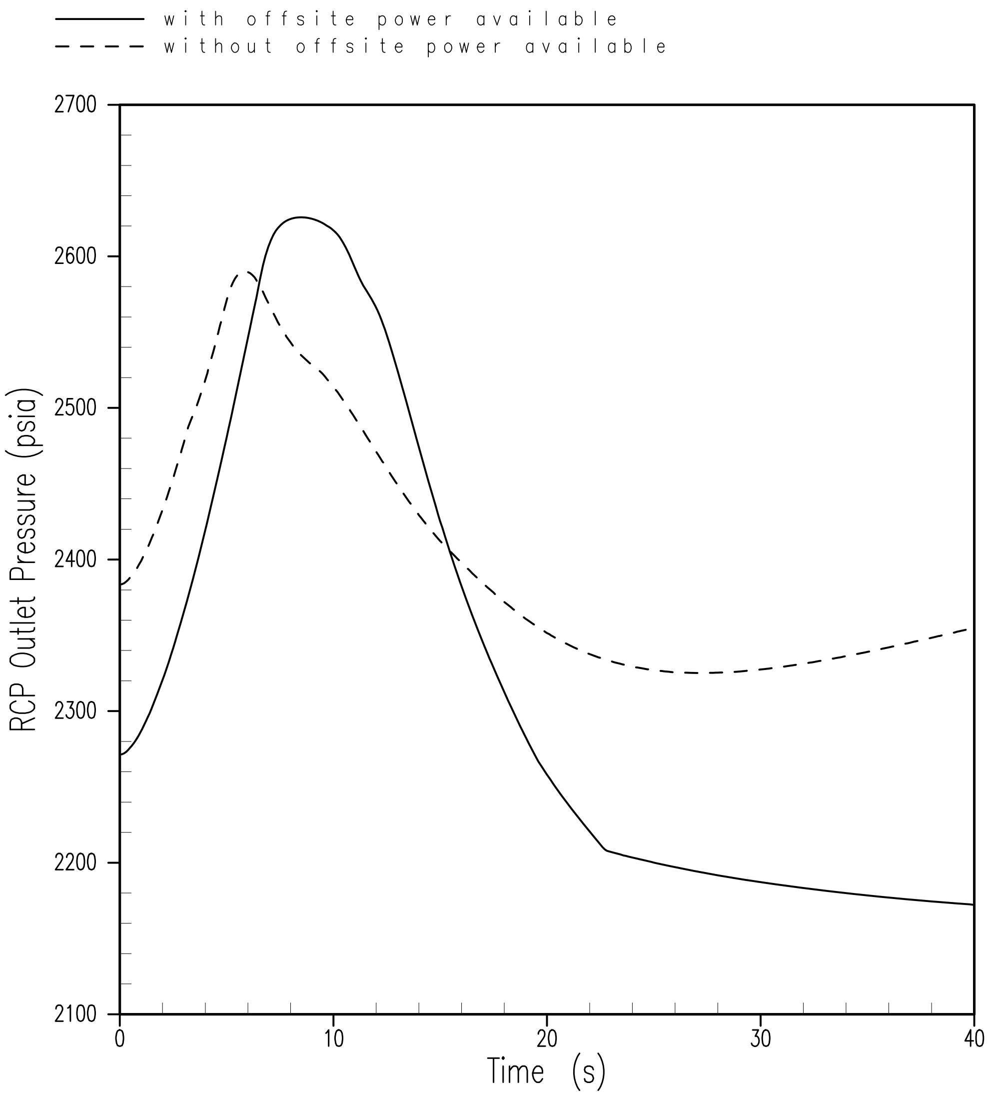


Figure 15.2.3-22
Pressurizer Pressure (psia) versus Time for Turbine Trip Accident Without Pressurizer Spray and Maximum Moderator Feedback

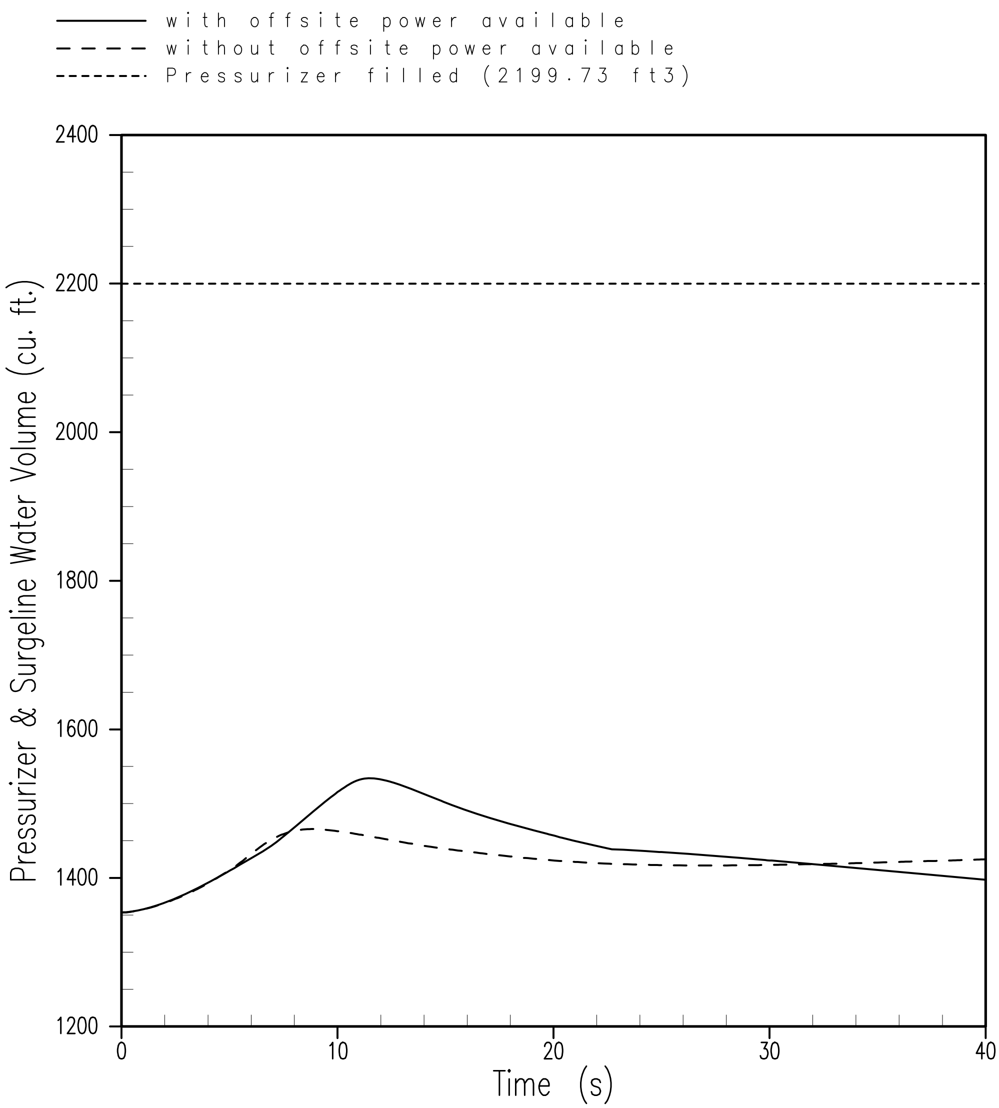


Figure 15.2.3-23
Pressurizer Water Volume (ft^3) versus Time for Turbine Trip Accident Without Pressurizer Spray and Maximum Moderator Feedback

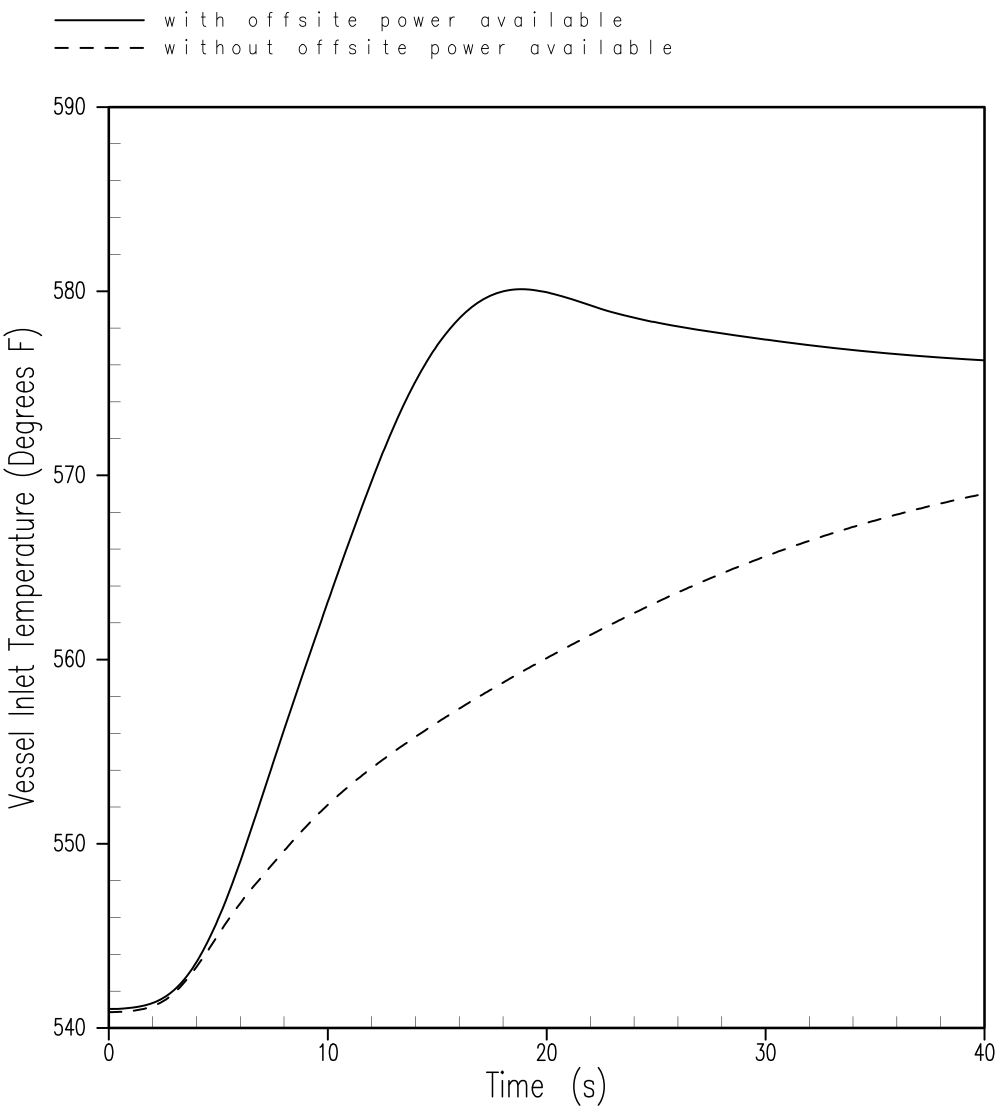


Figure 15.2.3-24
Vessel Inlet Temperature (°F) versus Time for Turbine Trip Accident Without Pressurizer Spray and Maximum Moderator Feedback

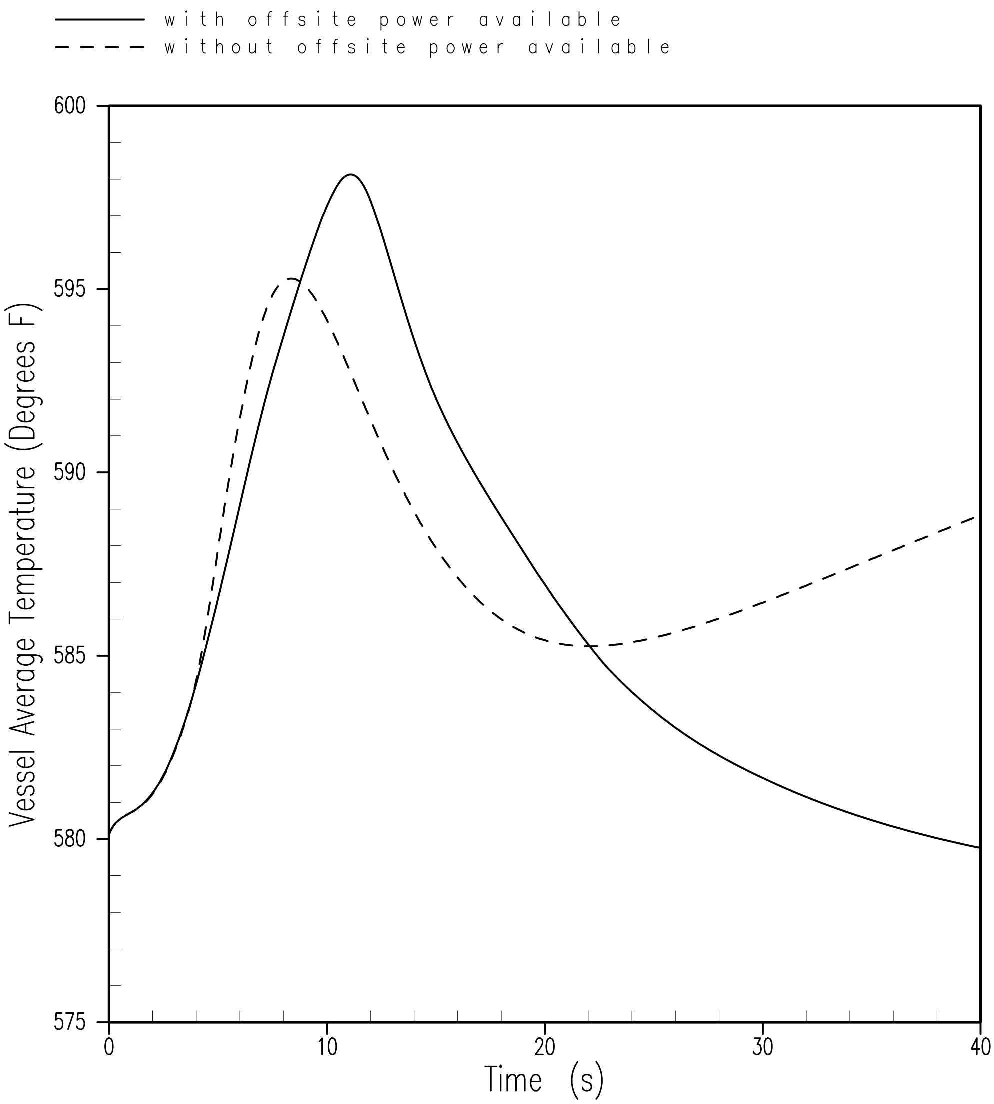


Figure 15.2.3-25
Vessel Average Temperature (°F) versus Time for Turbine Trip Accident Without Pressurizer Spray and Maximum Moderator Feedback

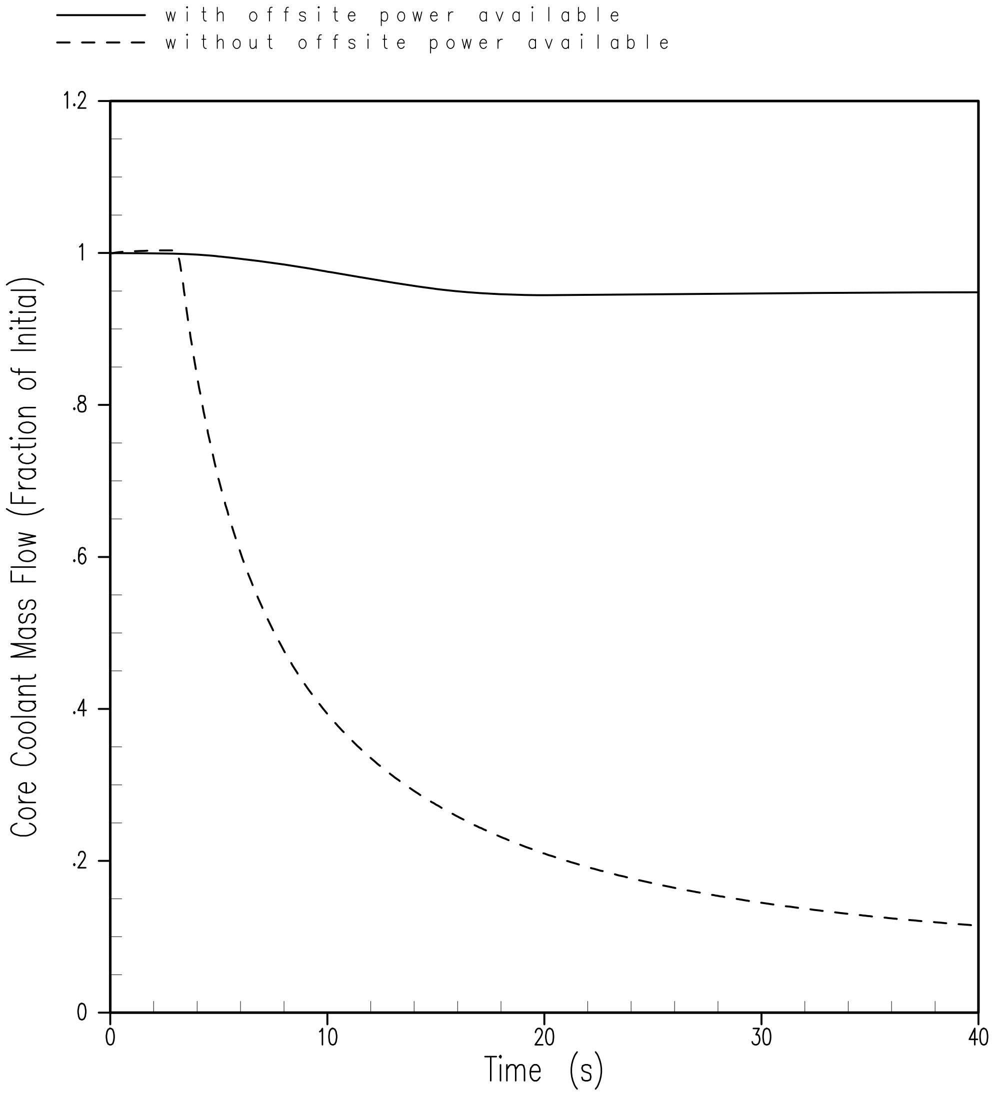


Figure 15.2.3-26
Core Mass Flow Rate (Fraction of Initial) versus Time for Turbine Trip Accident Without Pressurizer Spray and Maximum Moderator Feedback

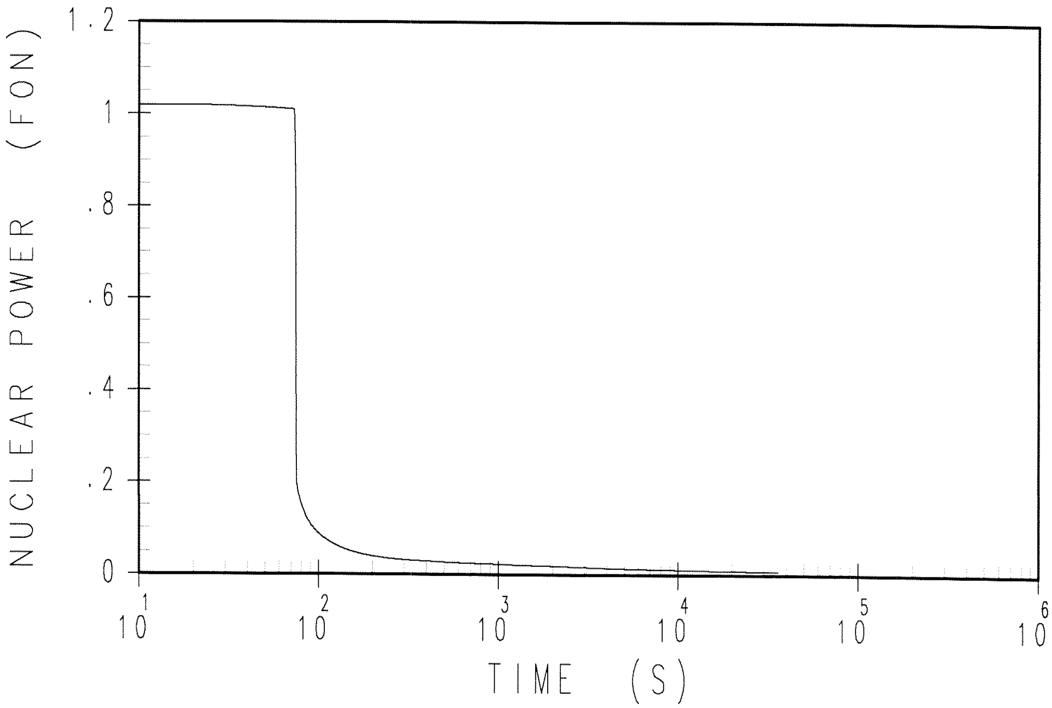


Figure 15.2.6-1
Nuclear Power Transient for Loss of ac Power to the Plant Auxiliaries

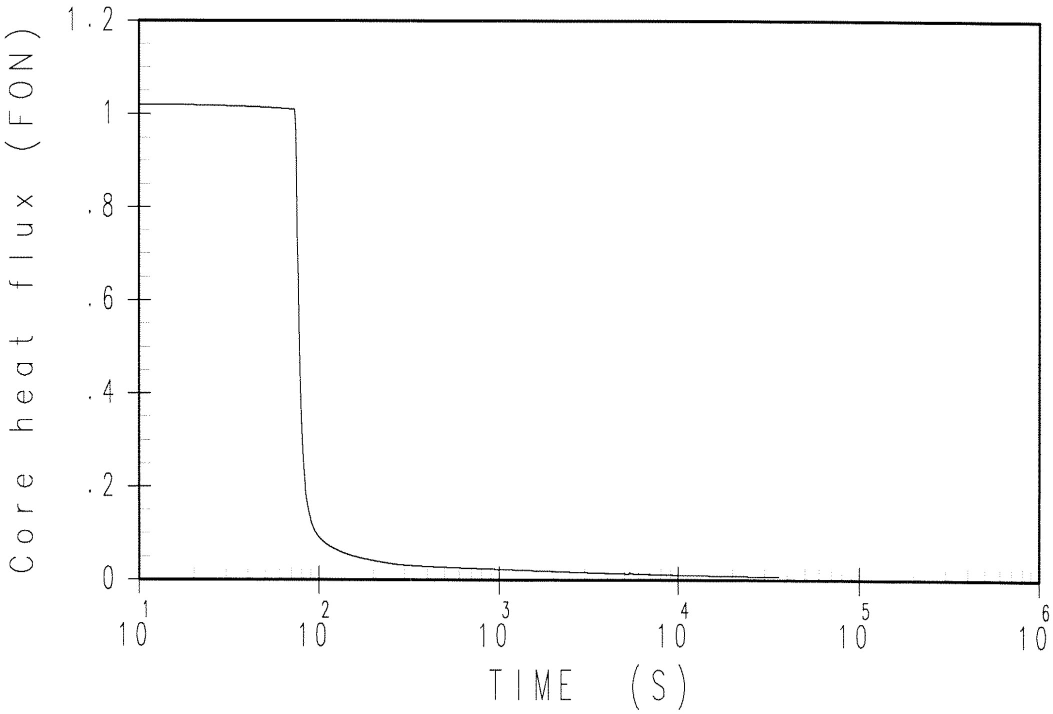


Figure 15.2.6-2
Core Heat Flux Transient for Loss of ac Power to the Plant Auxiliaries

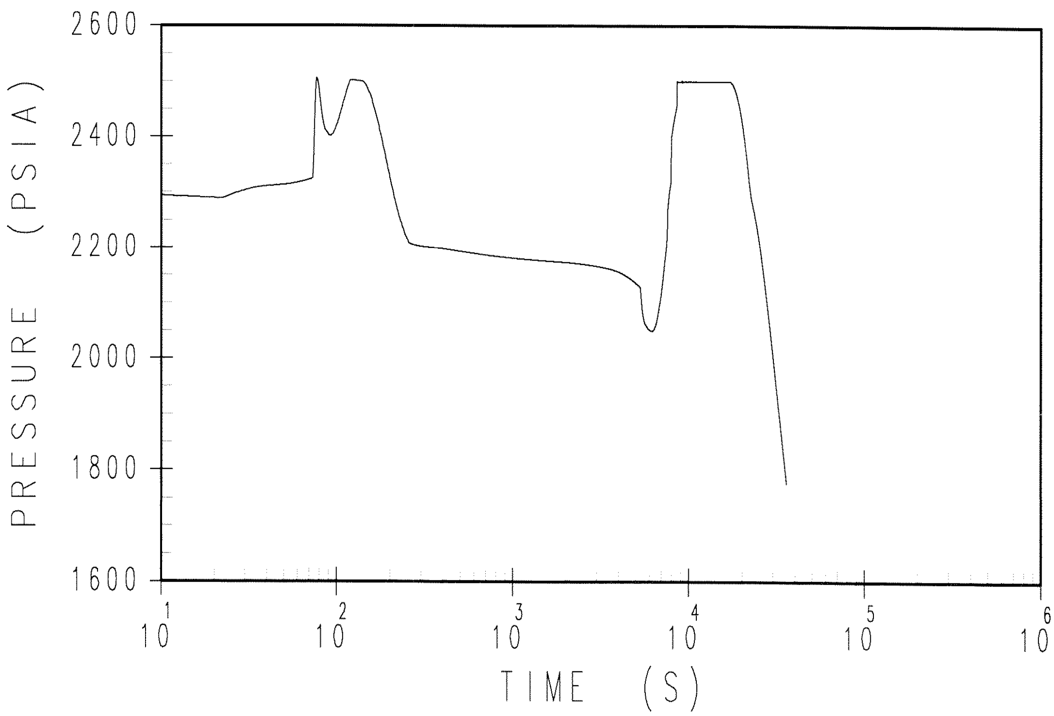


Figure 15.2.6-3
Pressurizer Pressure Transient for Loss of ac Power to the Plant Auxiliaries

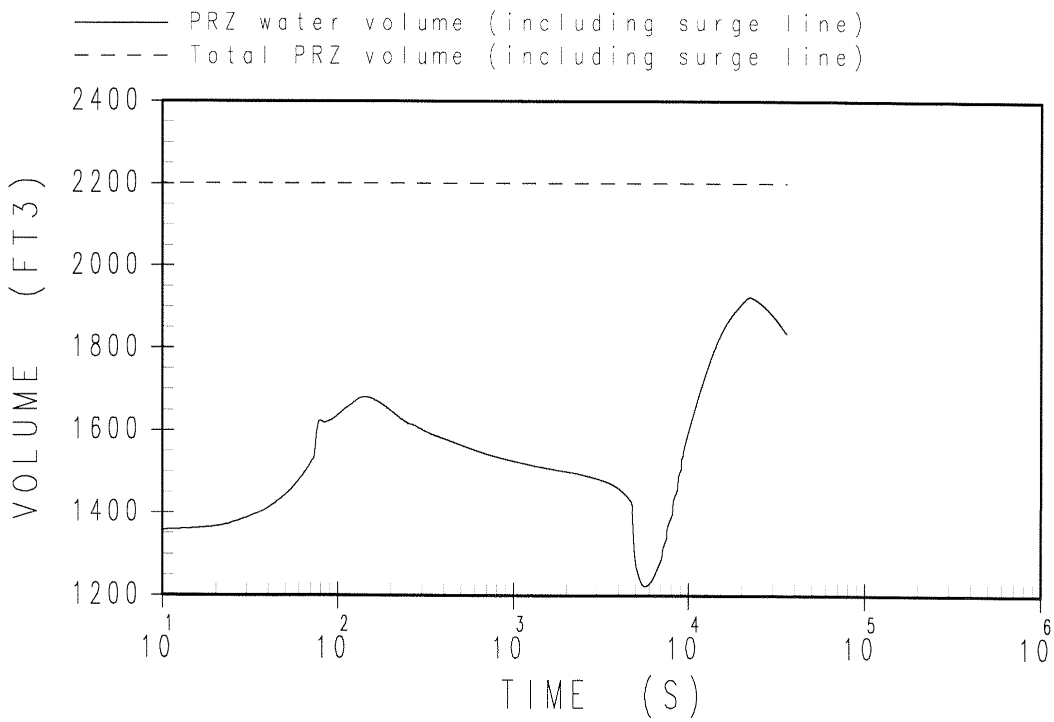


Figure 15.2.6-4
Pressurizer Water Volume Transient for Loss of ac Power to the Plant Auxiliaries

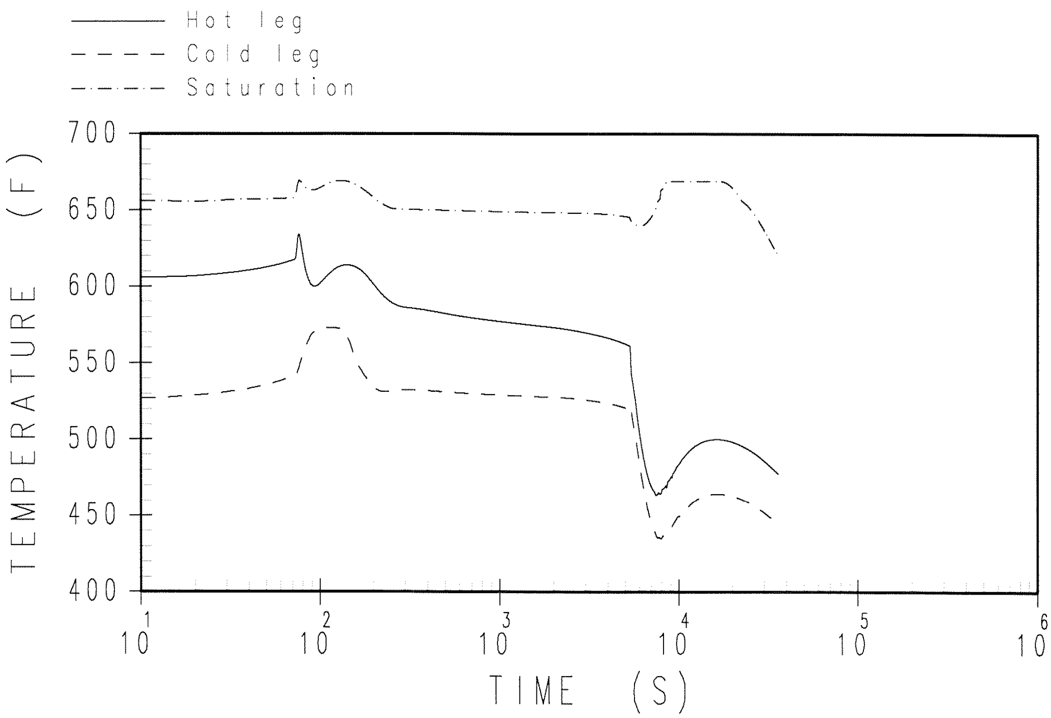


Figure 15.2.6-5
Reactor Coolant System Temperature Transients in Loop Containing the PRHR for Loss of ac Power to the Plant Auxiliaries

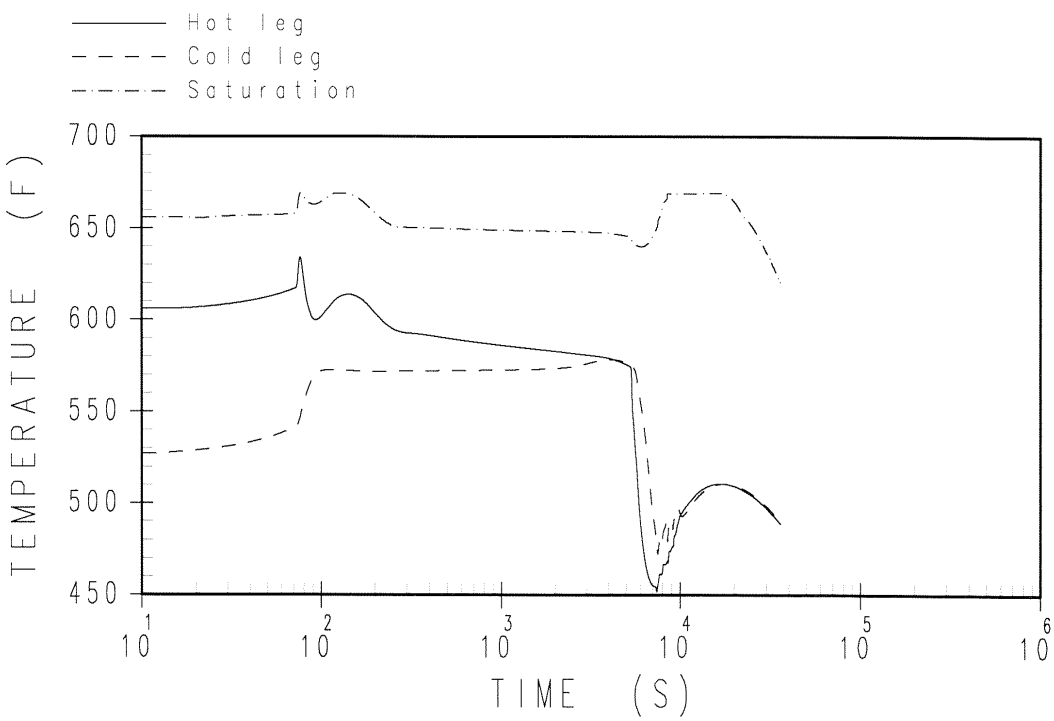


Figure 15.2.6-6
Reactor Coolant System Temperature Transients in Loop Not Containing the PRHR for Loss of ac Power to the Plant Auxiliaries

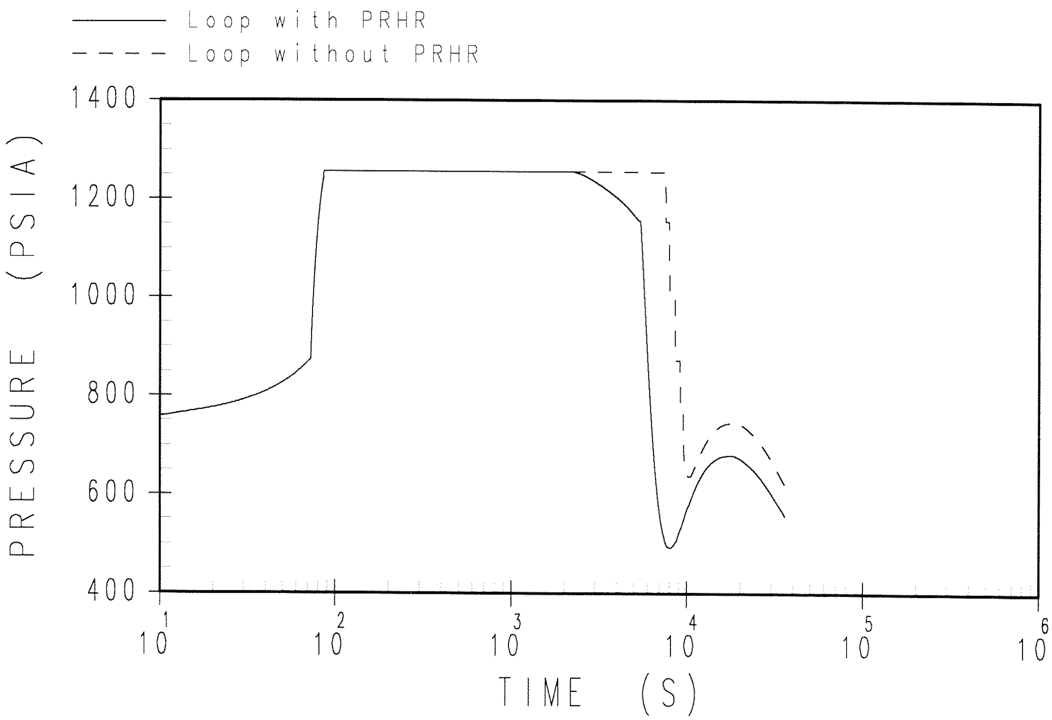


Figure 15.2.6-7
Steam Generator Pressure Transient
for Loss of ac Power to the Plant Auxiliaries

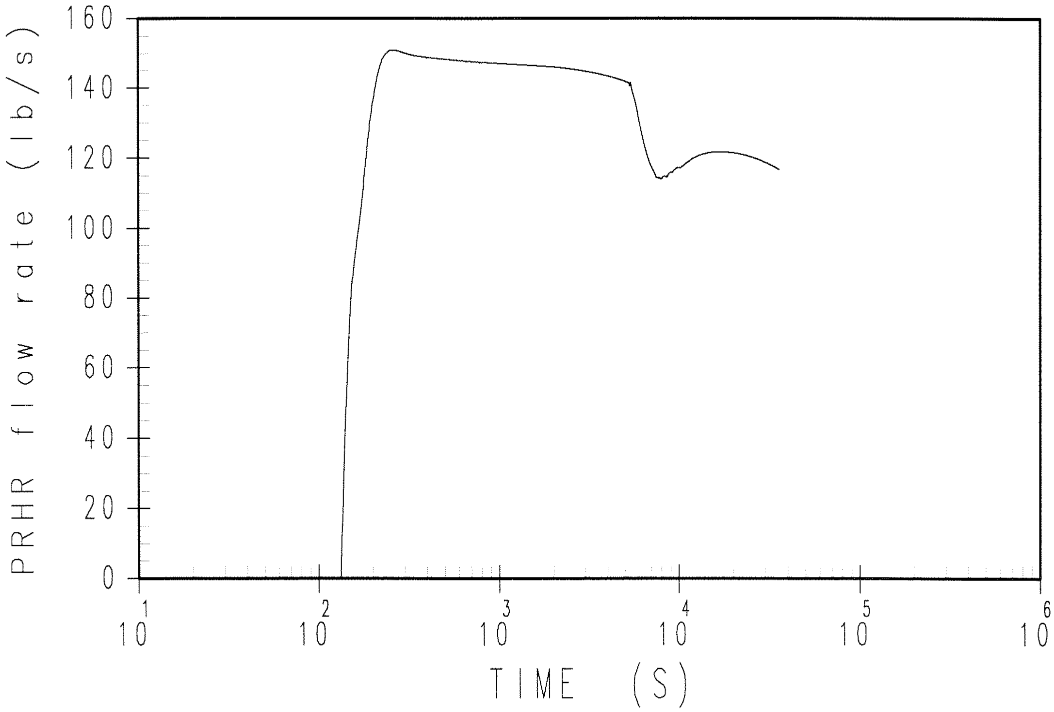


Figure 15.2.6-8
PRHR Flow Rate Transient
for Loss of ac Power to the Plant Auxiliaries

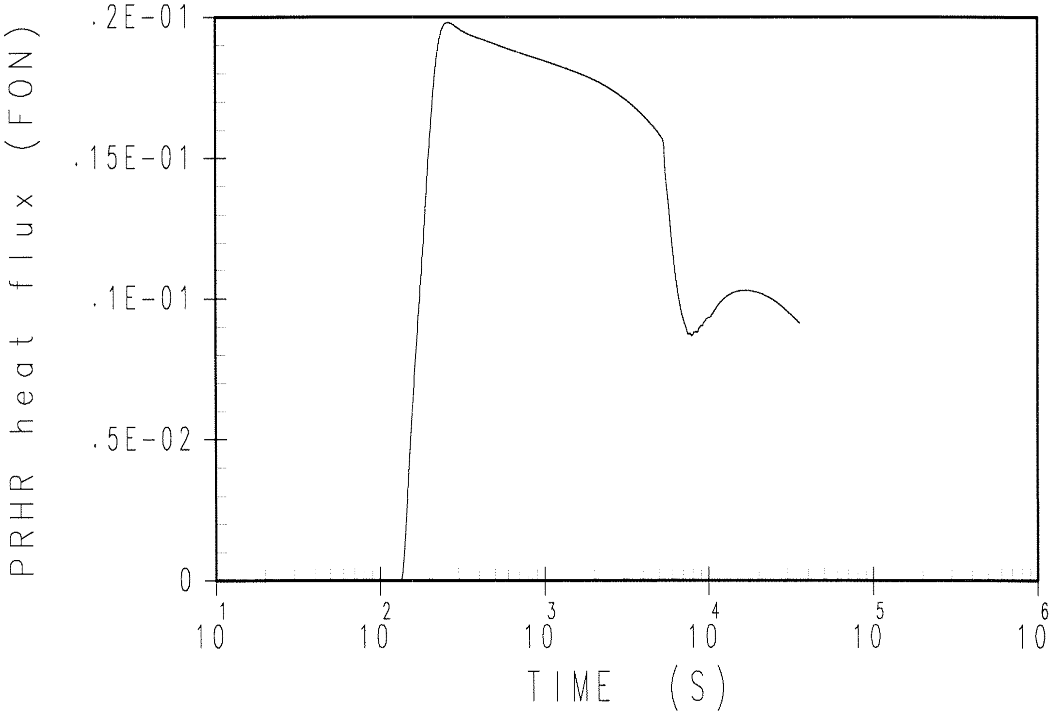


Figure 15.2.6-9
PRHR Heat Flux Transient
for Loss of ac Power to the Plant Auxiliaries

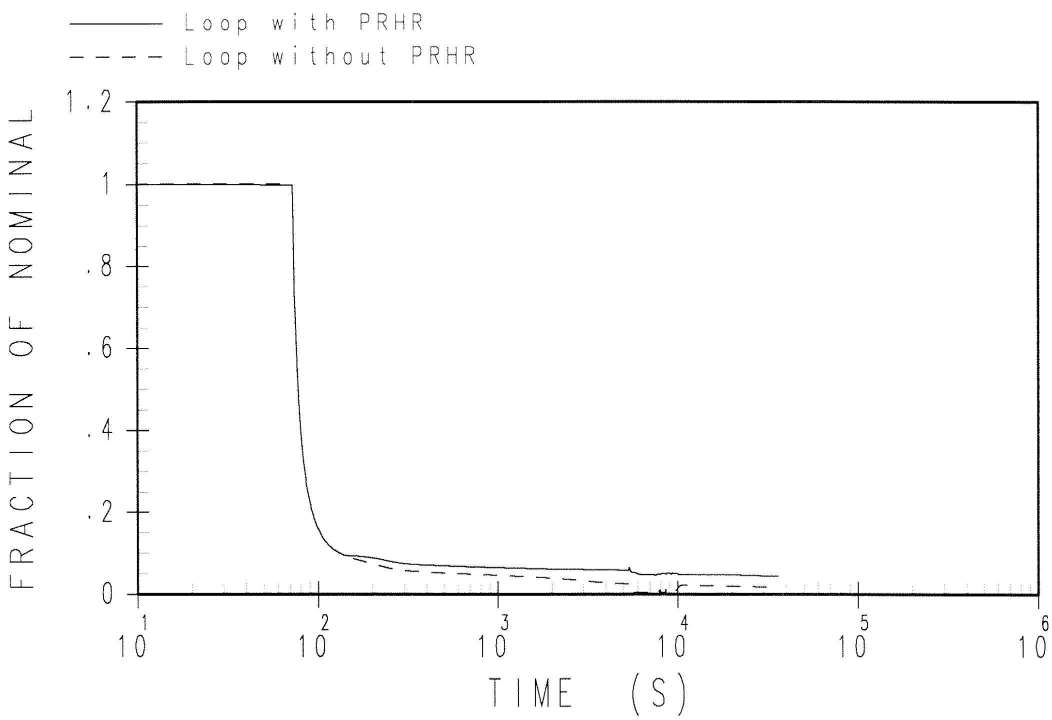


Figure 15.2.6-10
Reactor Coolant Volumetric Flow Rate
Transient for Loss of ac Power to the Plant Auxiliaries

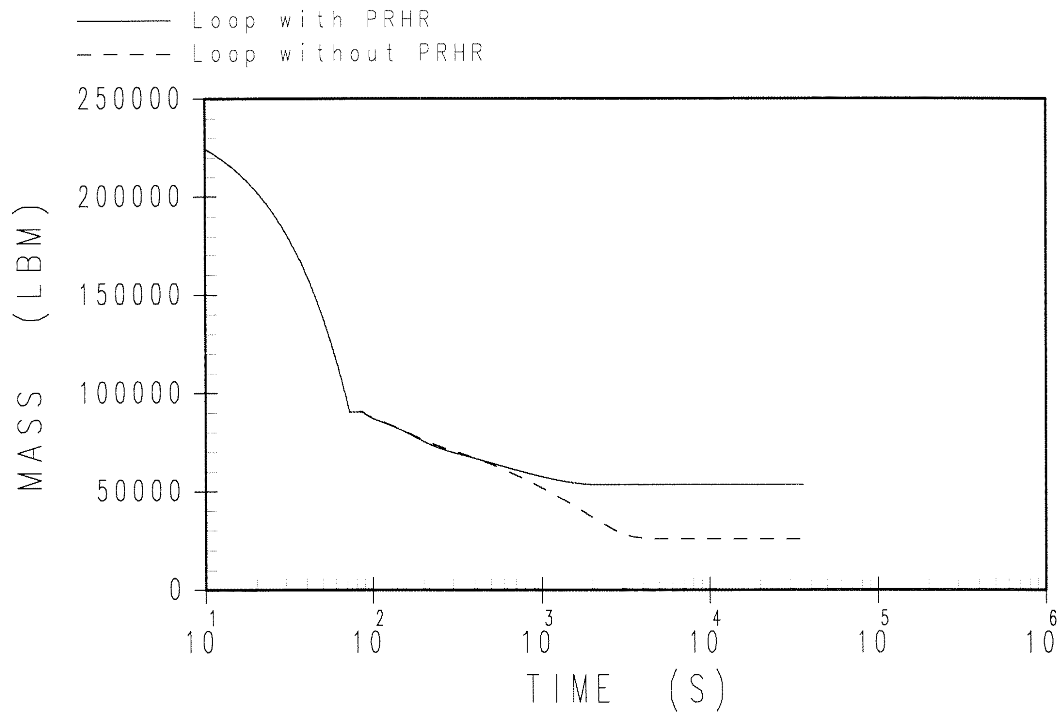


Figure 15.2.6-11
Steam Generator Inventory Transient
for Loss of ac Power to the Plant Auxiliaries

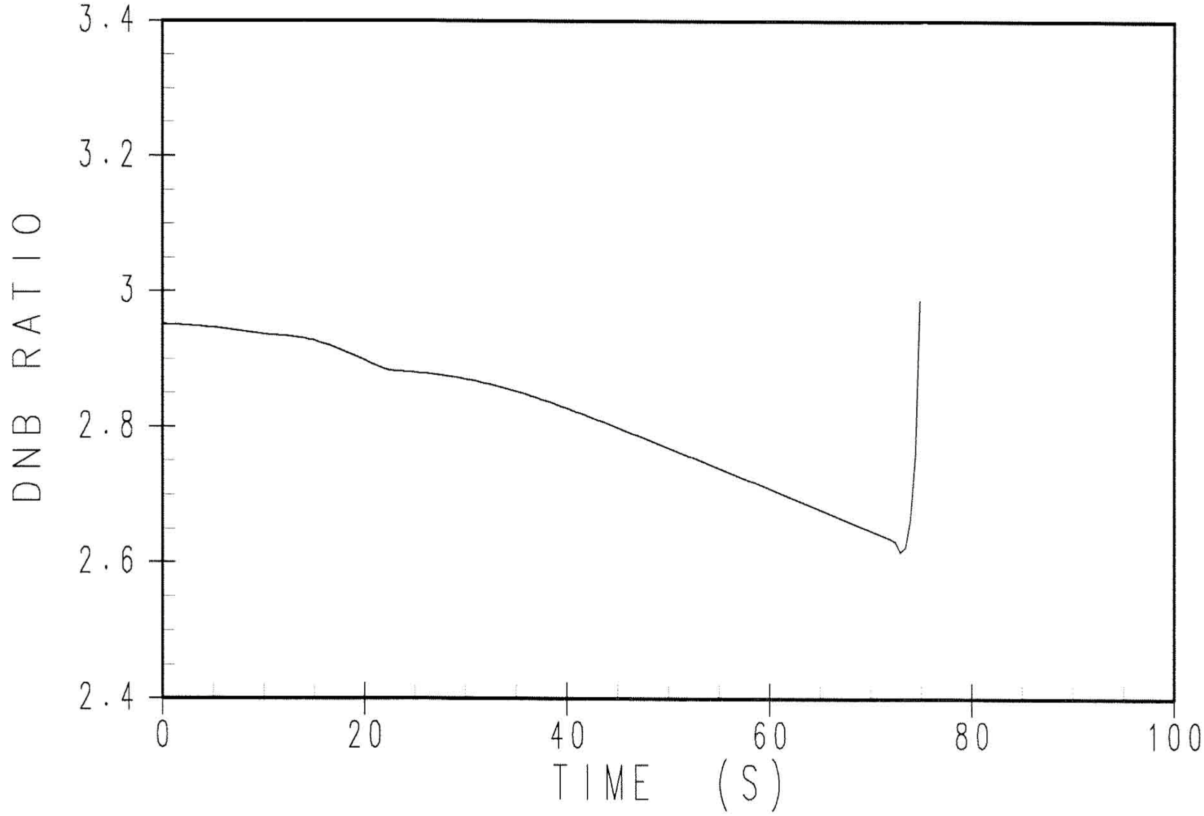


Figure 15.2.6-12
DNB Ratio Transient
for Loss of ac Power to the Plant Auxiliaries

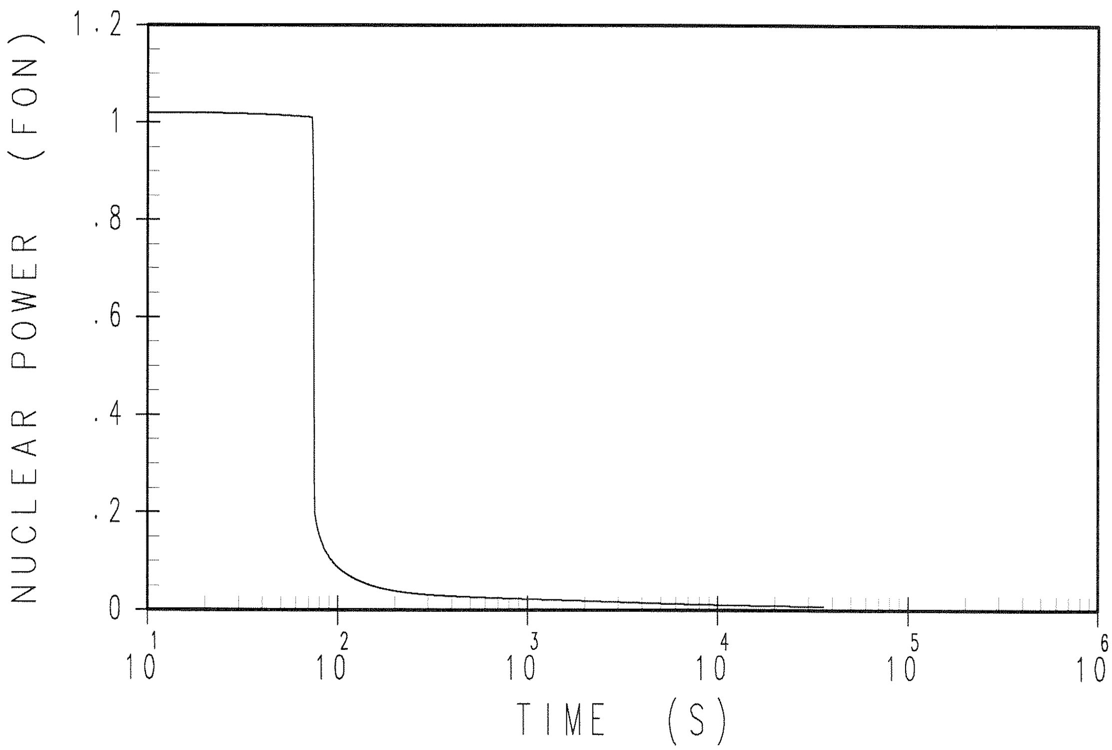


Figure 15.2.7-1
Nuclear Power Transient for Loss of Normal Feedwater Flow

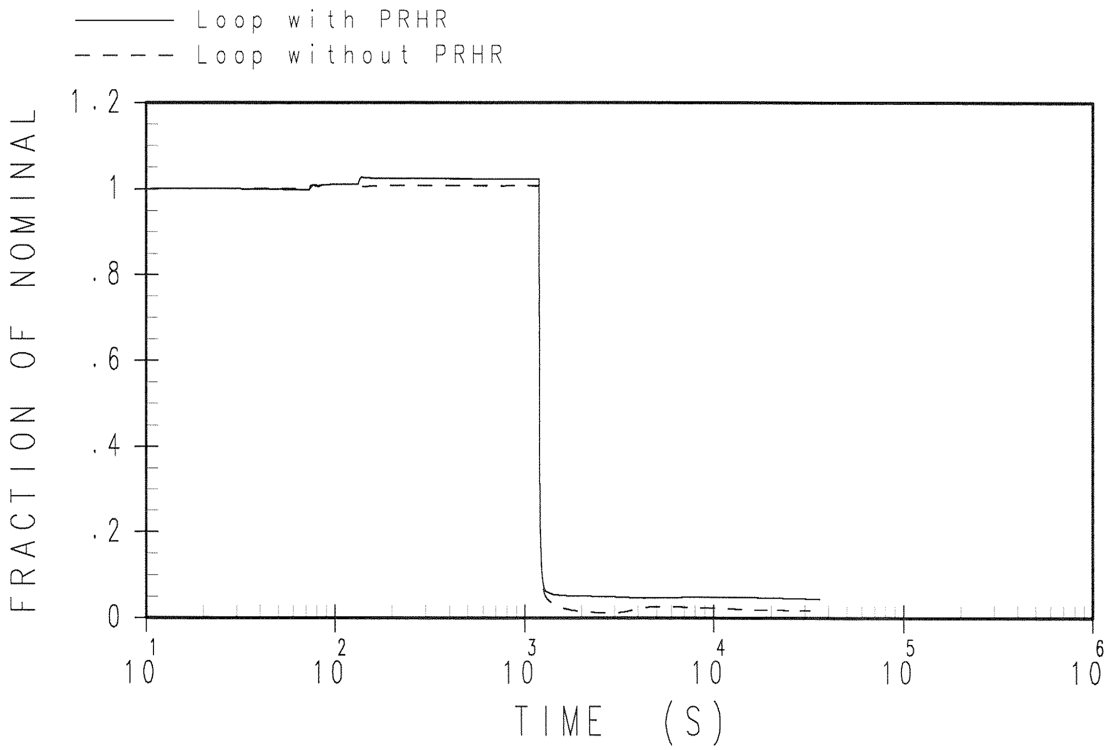


Figure 15.2.7-2
Reactor Coolant System Volumetric Flow
Transient for Loss of Normal Feedwater Flow

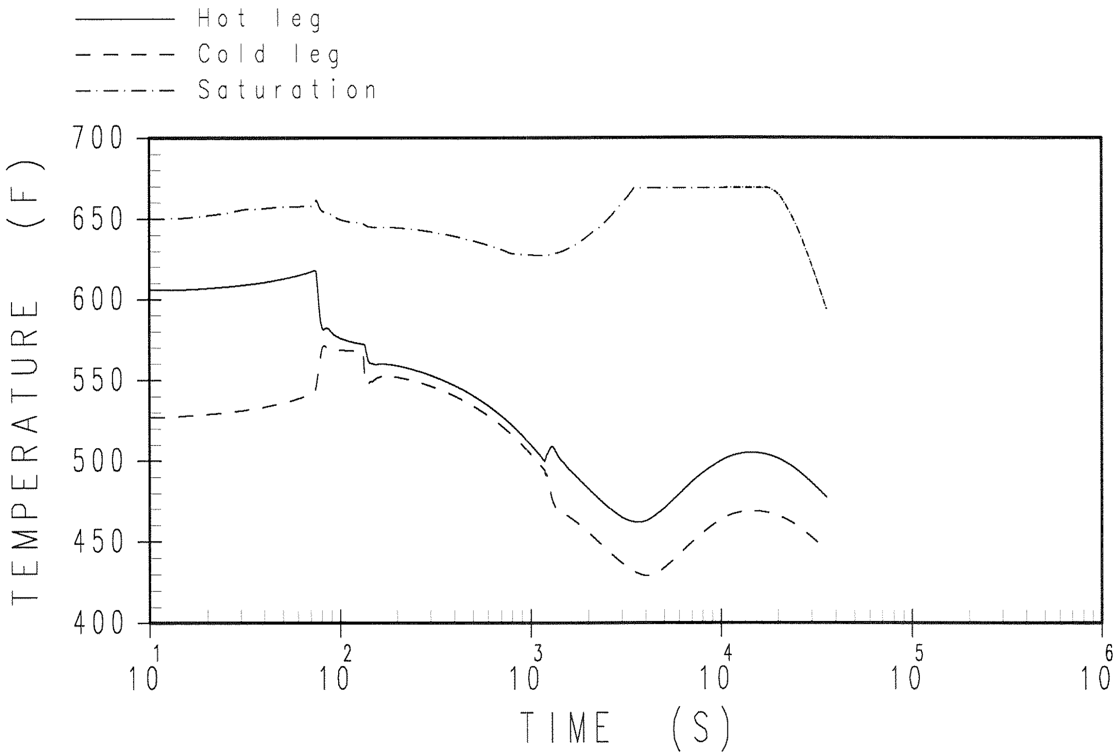


Figure 15.2.7-3
Reactor Coolant System Temperature Transients in Loop Containing the PRHR for Loss Normal Feedwater Flow

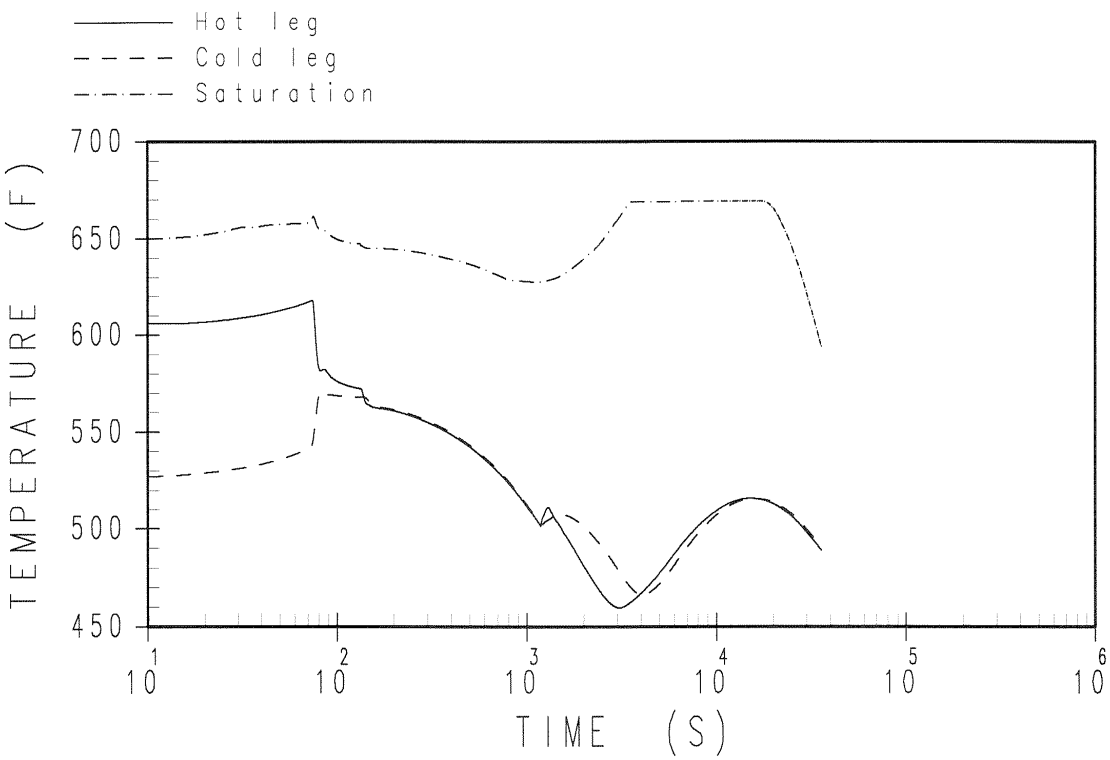


Figure 15.2.7-4
Reactor Coolant System Temperature Transients in Loop Not Containing the PRHR for Loss of Normal Feedwater Flow

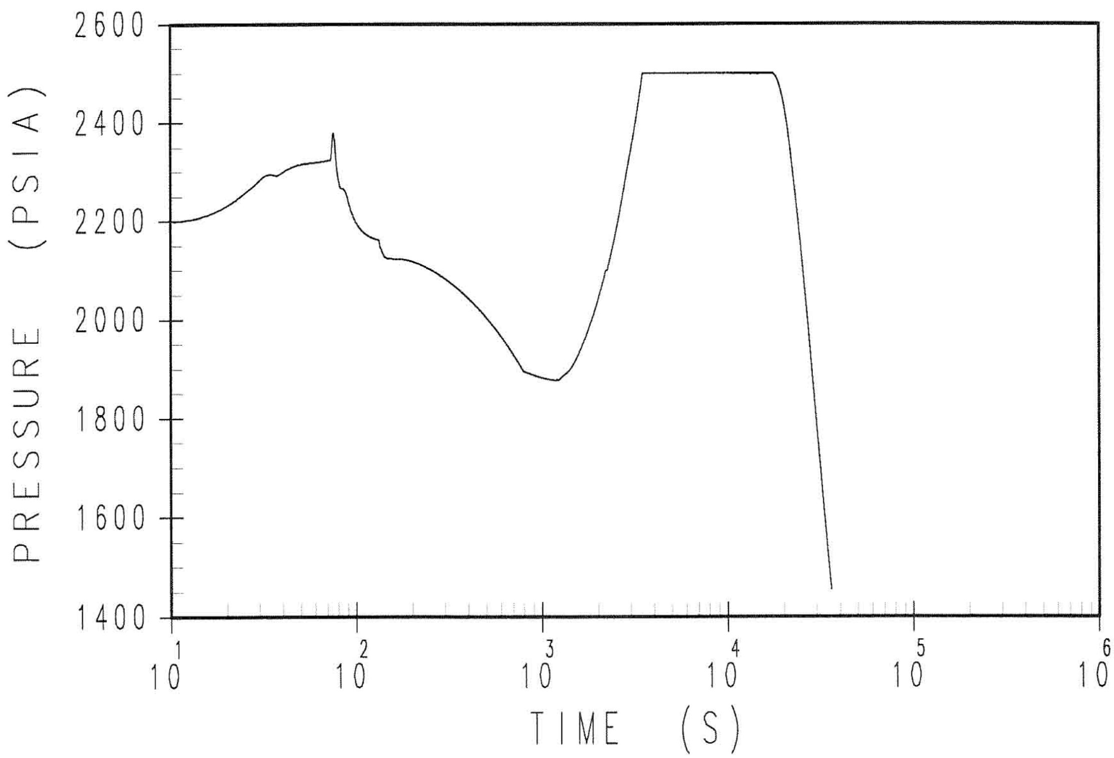


Figure 15.2.7-5
Pressurizer Pressure Transient for Loss of Normal Feedwater Flow

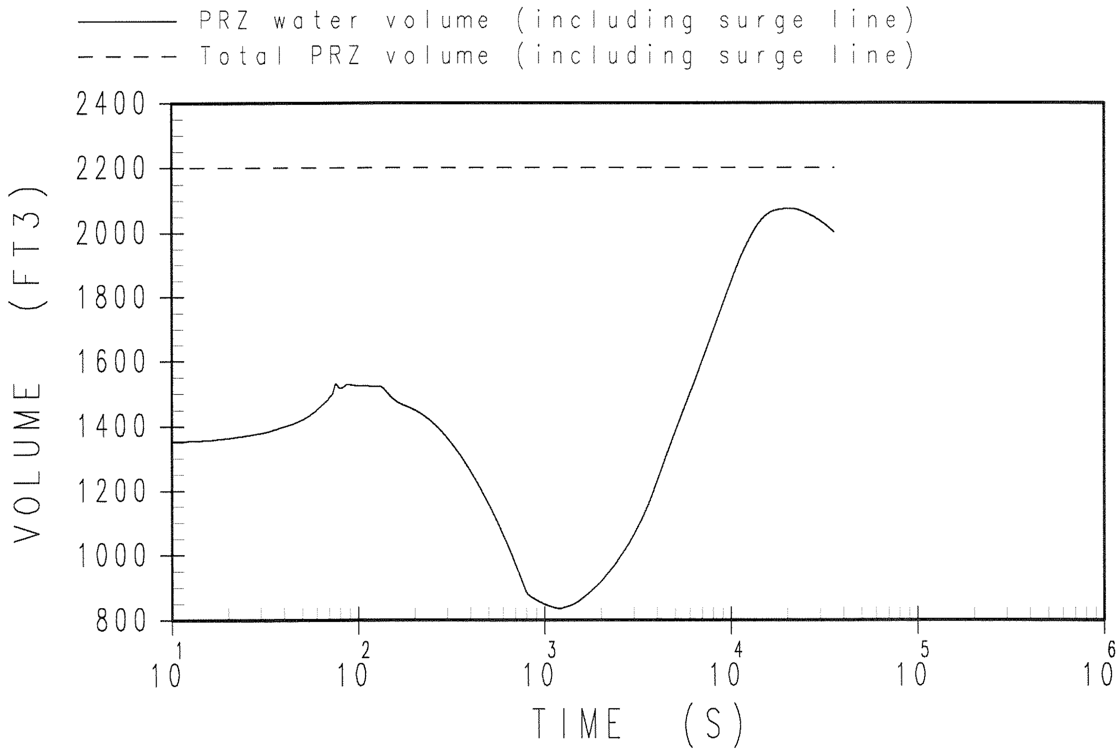


Figure 15.2.7-6
Pressurizer Water Volume Transient for Loss of Normal Feedwater Flow

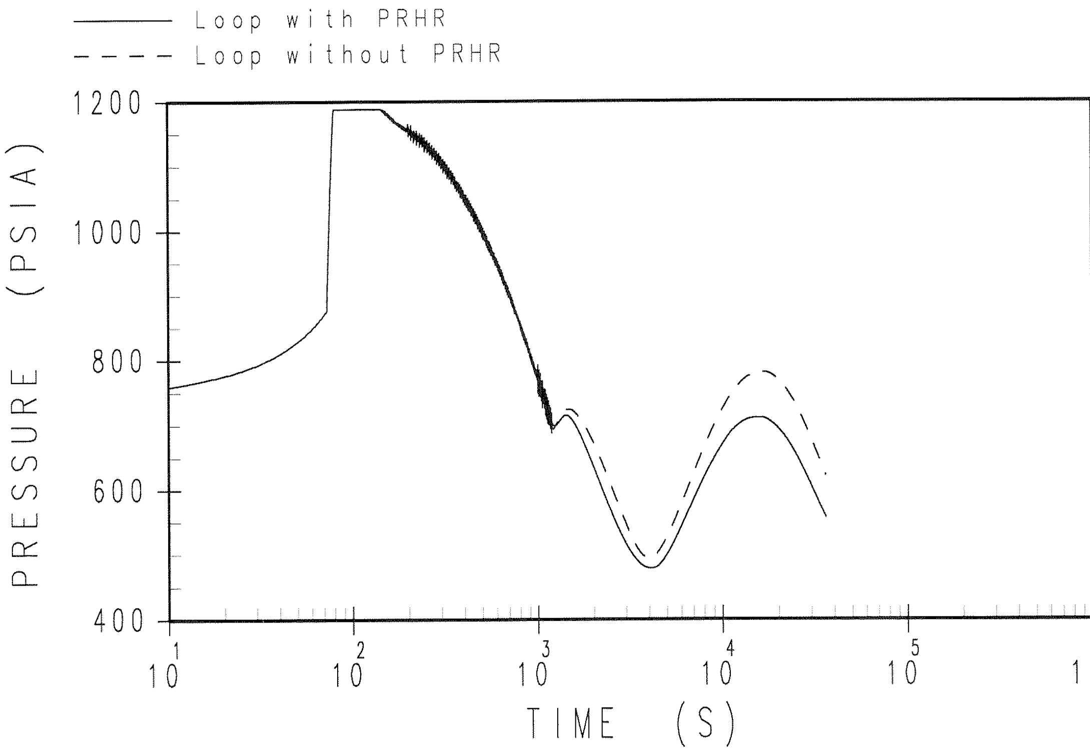


Figure 15.2.7-7
Steam Generator Pressure Transient
for Loss of Normal Feedwater Flow

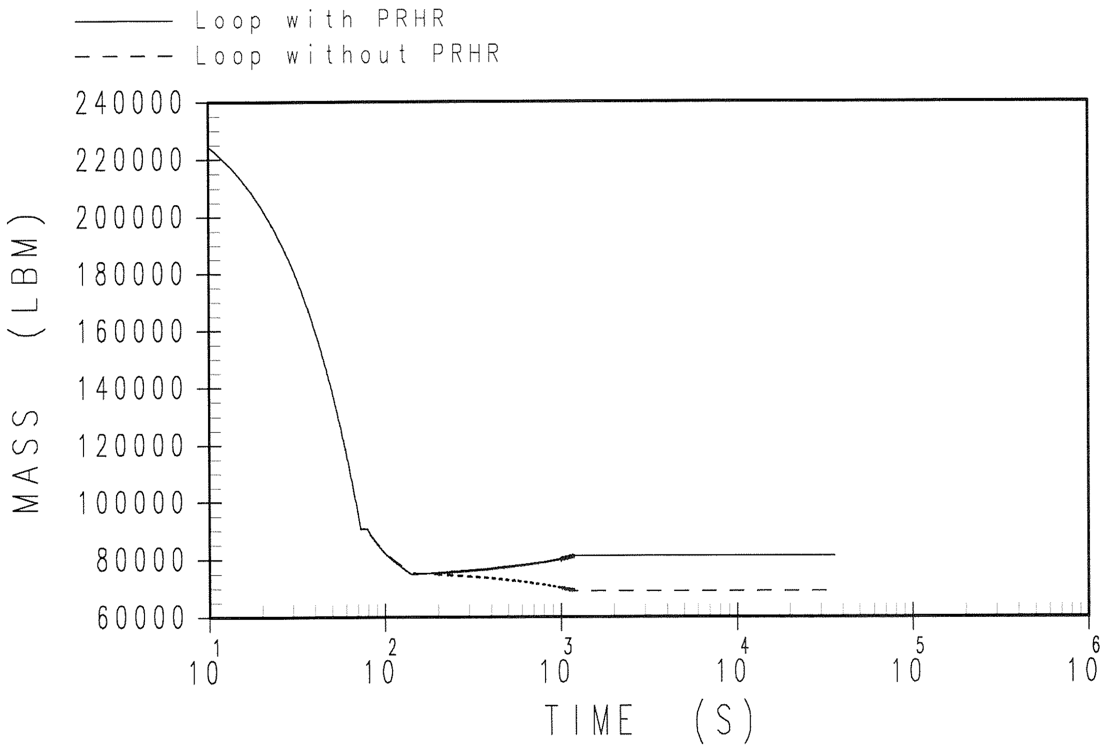


Figure 15.2.7-8
Steam Generator Inventory Transient for Loss of Normal Feedwater Flow

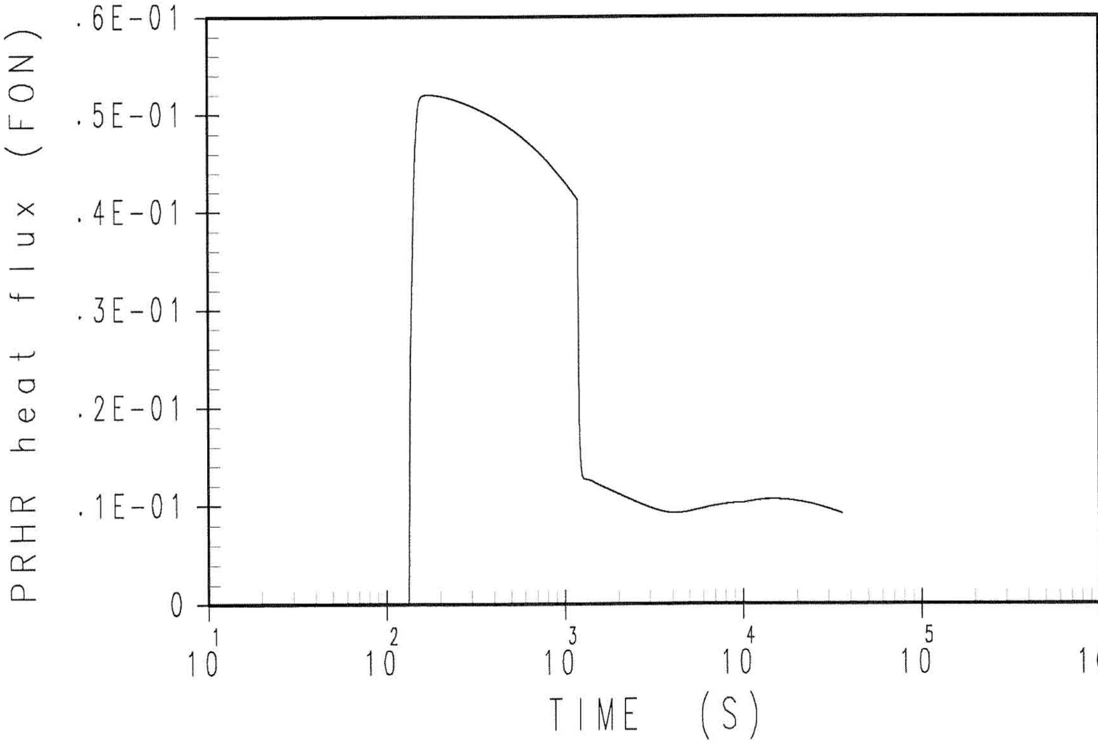


Figure 15.2.7-9
PRHR Heat Flux Transient
for Loss of Normal Feedwater Flow

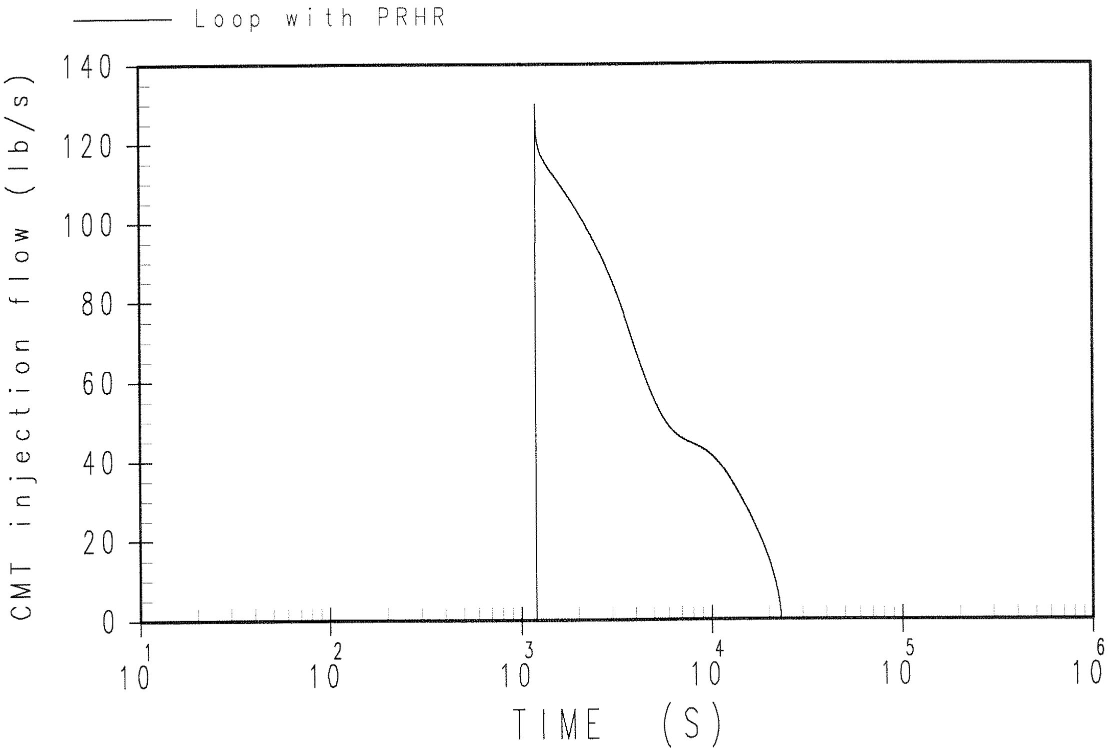


Figure 15.2.7-10
CMT Injection Flow Rate Transient
for Loss of Normal Feedwater Flow

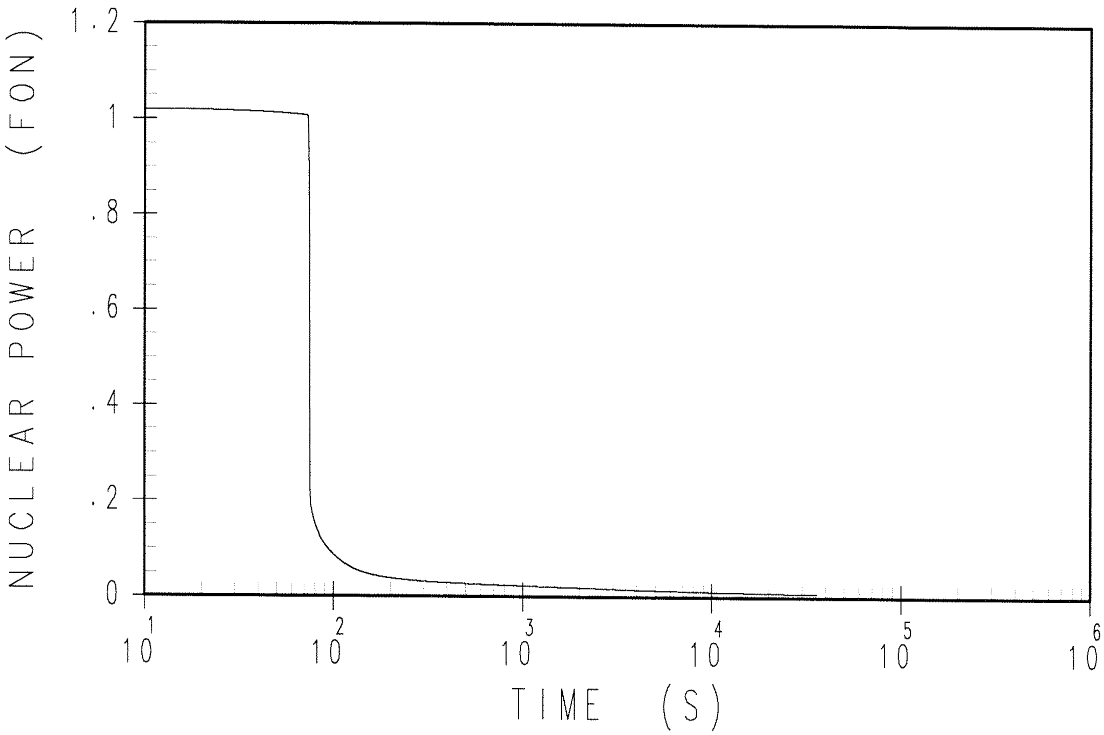


Figure 15.2.8-1
Nuclear Power Transient for
Main Feedwater Line Rupture

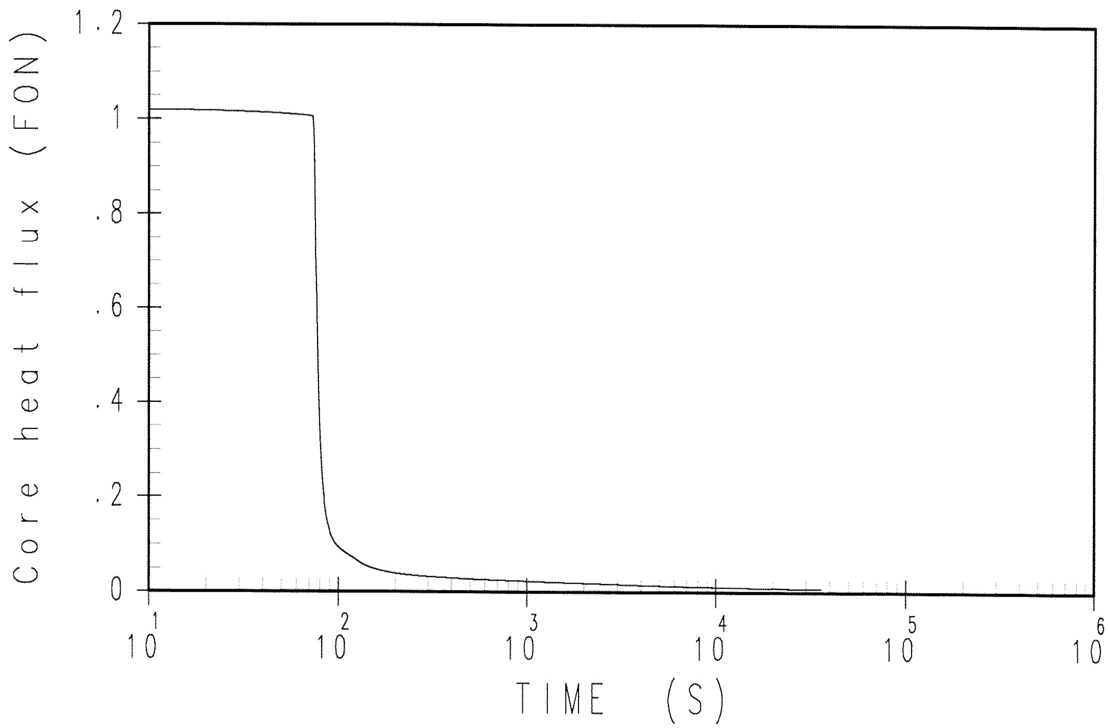


Figure 15.2.8-2
Core Heat Flux Transient for
Main Feedwater Line Rupture

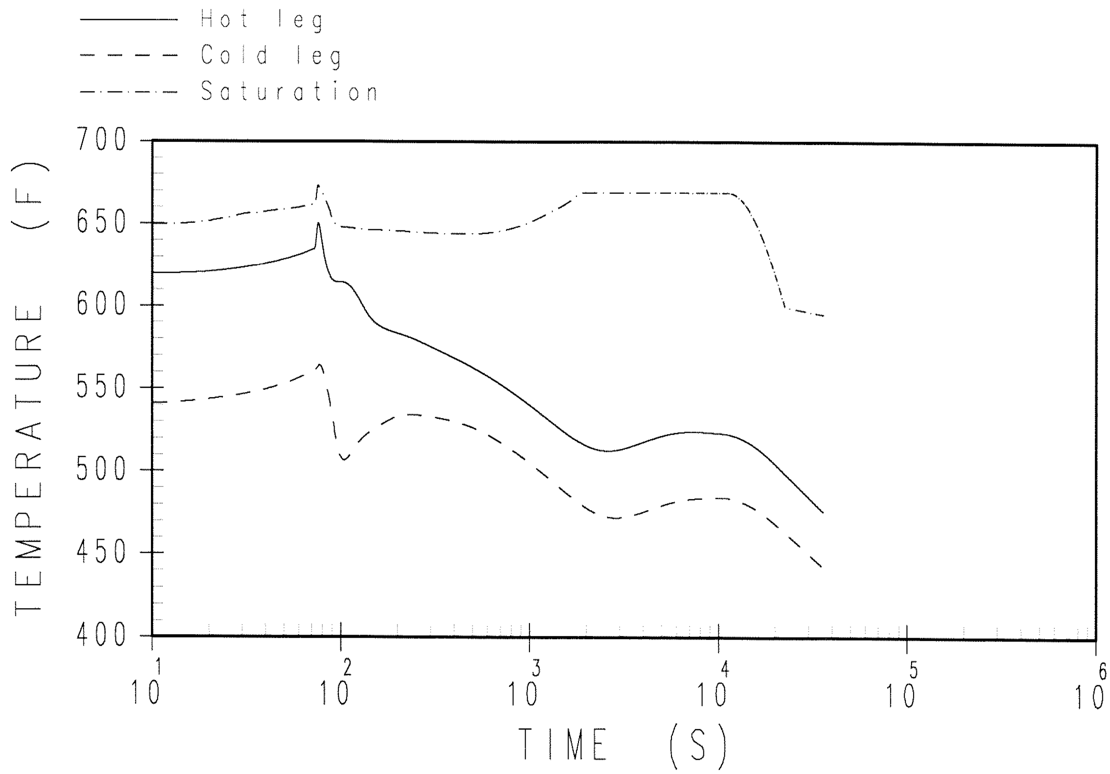


Figure 15.2.8-3
Faulted Loop Reactor Coolant System
Temperature Transients for Main Feedwater Line Rupture

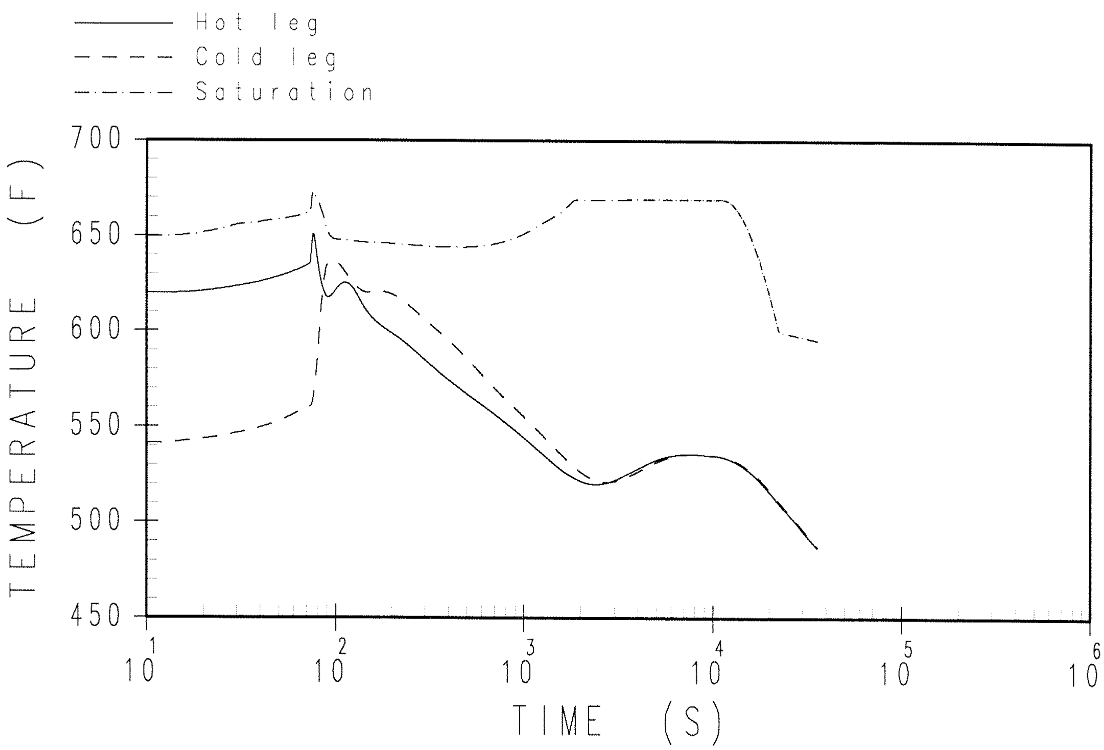


Figure 15.2.8-4
Intact Loop Reactor Coolant System
Temperature Transients for Main Feedwater Line Rupture

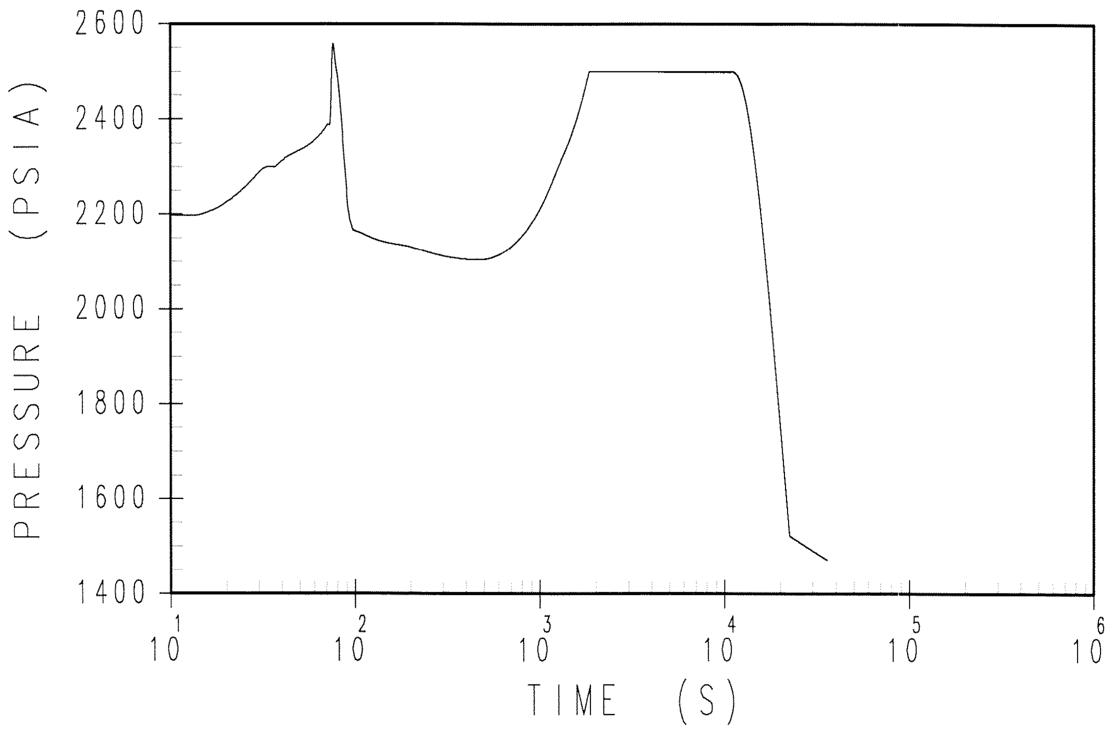


Figure 15.2.8-5
Pressurizer Pressure Transient for Main Feedwater Line Rupture

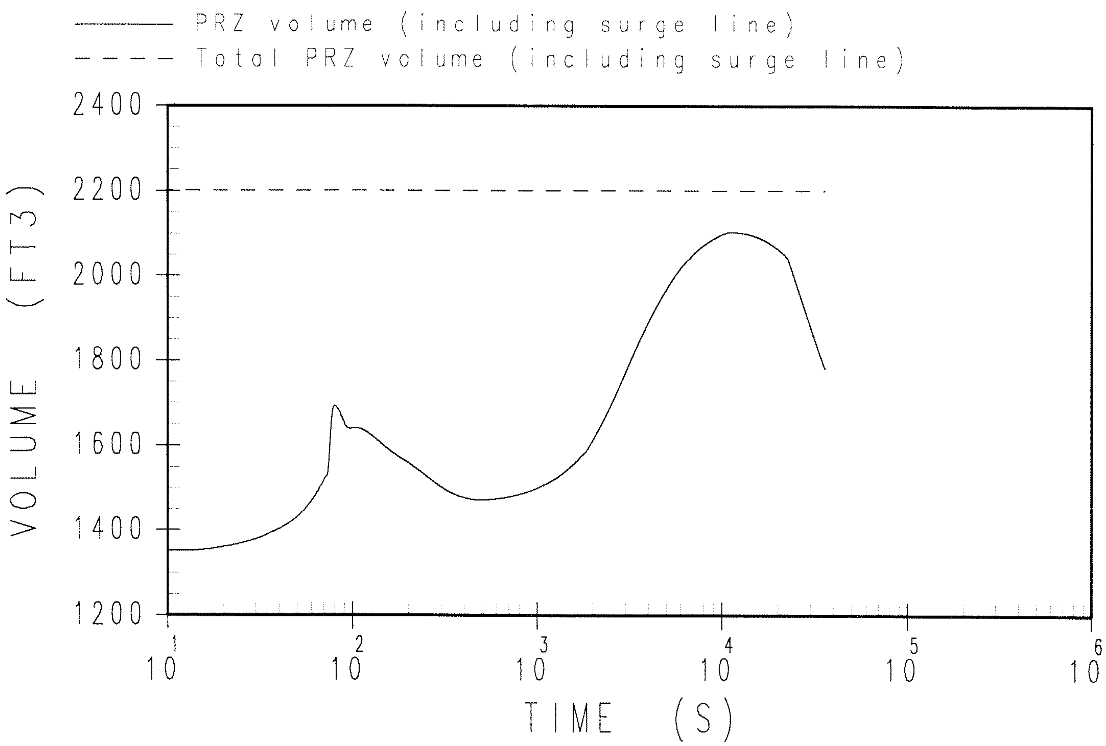


Figure 15.2.8-6
Pressurizer Water Volume Transient for Main Feedwater Line Rupture

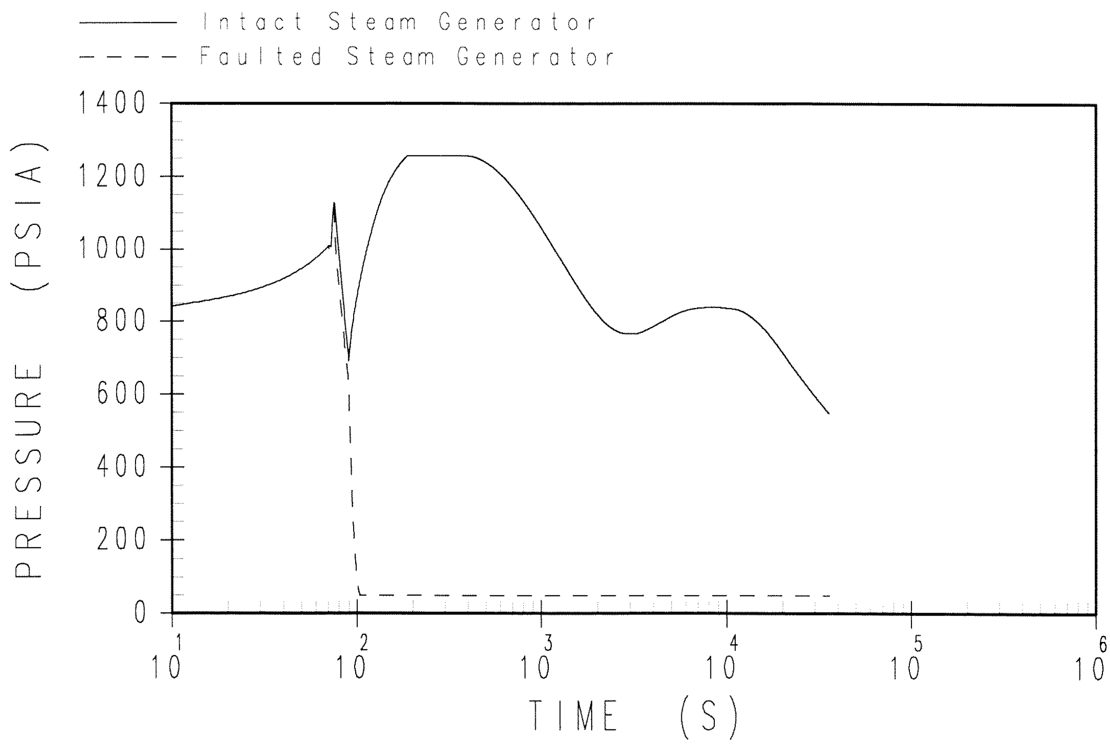


Figure 15.2.8-7
Steam Generator Pressure Transient for Main Feedwater Line Rupture

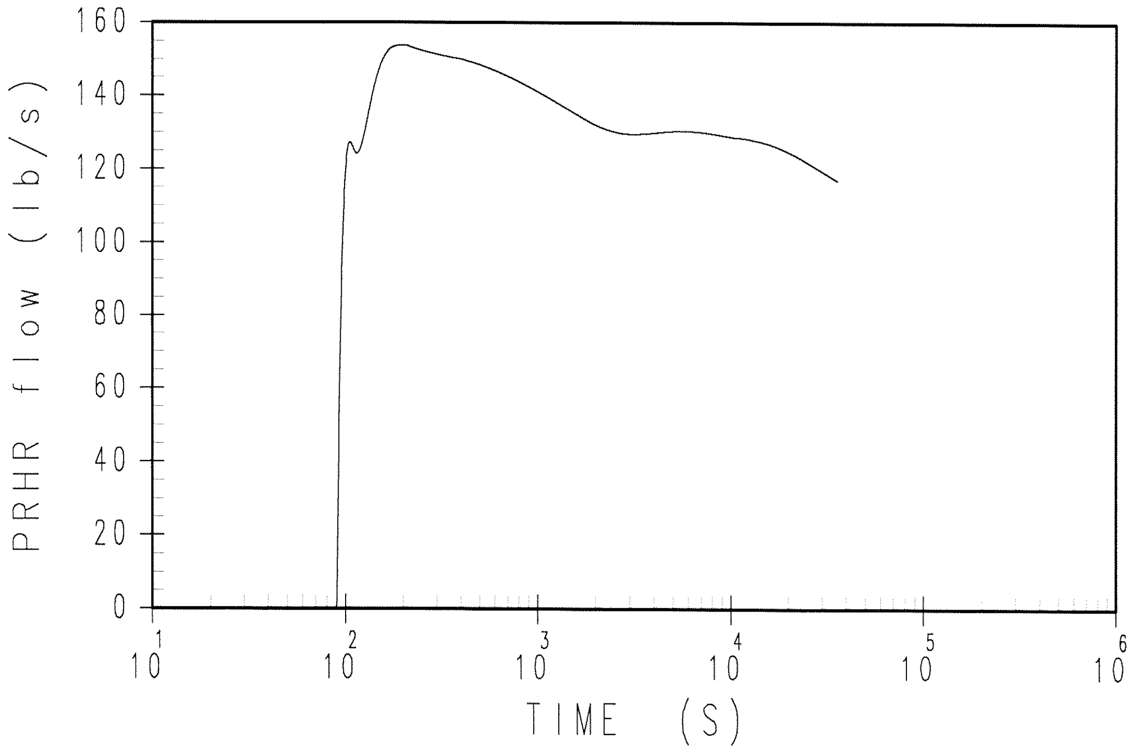


Figure 15.2.8-8
PRHR Flow Rate Transient for
Main Feedwater Line Rupture

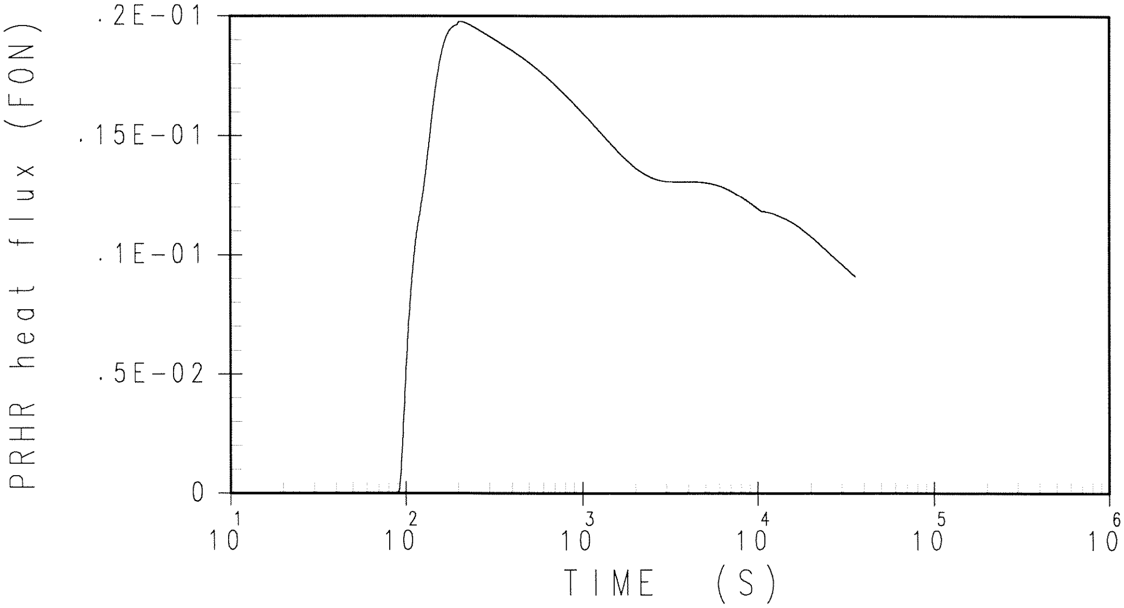


Figure 15.2.8-9
PRHR Heat Flux Transient for
Main Feedwater Line Rupture

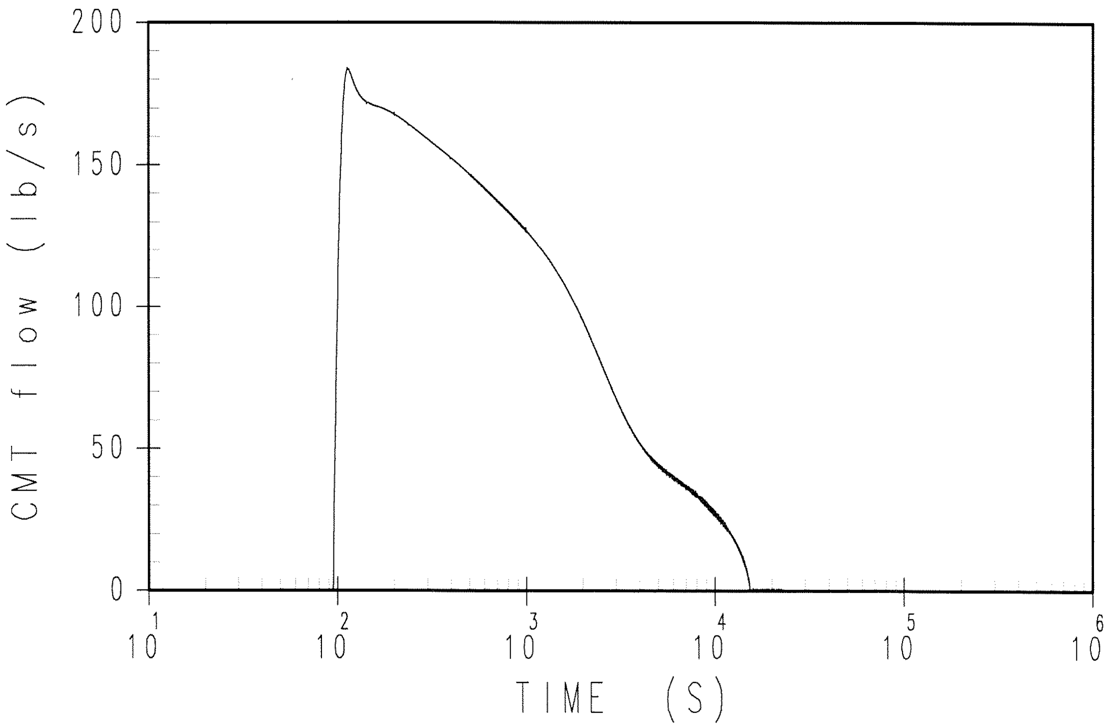


Figure 15.2.8-10
CMT Injection Flow Rate Transient for
Main Feedwater Line Rupture

15.3 Decrease in Reactor Coolant System Flow Rate

A number of faults that could result in a decrease in the reactor coolant system flow rate are postulated. These events are discussed in this section. Detailed analyses are presented for the most limiting of the following reactor coolant system flow decrease events:

- Partial loss of forced reactor coolant flow
- Complete loss of forced reactor coolant flow
- Reactor coolant pump shaft seizure (locked rotor)
- Reactor coolant pump shaft break

The first event is a Condition II event, the second is a Condition III event, and the last two are Condition IV events.

The four limiting flow rate decrease events described above are analyzed in this section. The most severe radiological consequences result from the reactor coolant pump shaft seizure accident discussed in Subsection 15.3.3. Doses are reported only for that case.

15.3.1 Partial Loss of Forced Reactor Coolant Flow

15.3.1.1 Identification of Causes and Accident Description

A partial loss of coolant flow accident can result from a mechanical or an electrical failure of a reactor coolant pump or from a fault in the power supply to the pump or pumps. If the reactor is at power at the time of the event, the immediate effect of the loss of coolant flow is a rapid increase in the coolant temperature.

Normal power for the pumps is supplied through four buses connected to the generator. When a generator trip occurs, the buses are supplied from offsite power. The pumps continue to operate.

A partial loss of coolant flow is classified as a Condition II incident (a fault of moderate frequency), as defined in Subsection 15.0.1.

Protection against this event is provided by the low primary coolant flow reactor trip signal, which is actuated by two-out-of-four low-flow signals. Above permissive P10, low flow in either hot leg actuates a reactor trip (see Section 7.2).

As specified in GDC 17 of 10 CFR Part 50, Appendix A, the effects of a loss of offsite power are considered in evaluating partial loss of forced reactor coolant flow transients. As discussed in Subsection 15.0.14, the loss of offsite power is considered to be a potential consequence of the event due to disruption of the electrical grid following a turbine trip during the event. A delay of 3 seconds is assumed between the turbine trip and the loss of offsite power. In addition, turbine trip occurs 5.0 seconds following a reactor trip condition being reached. This delay on turbine trip is a feature of the AP1000 reactor trip system. The primary effect of the loss of offsite power is to cause the remaining operating reactor coolant pumps to coast down.

15.3.1.2 Analysis of Effects and Consequences

15.3.1.2.1 Method of Analysis

This transient is analyzed using three computer codes. First, the LOFTRAN code (Reference 1) is used to calculate the core flow during the transient based on the input loop flows, the nuclear power transient, and the primary system pressure and temperature transients as predicted from the loss of two reactor coolant pumps. The FACTRAN code (Reference 2) is then used to calculate the heat flux transient based on the nuclear power and flow from LOFTRAN. Finally, the VIPRE-01 code (see Section 4.4) is used to calculate the departure from nucleate boiling ratio (DNBR) during the transient, based on the heat flux from FACTRAN and the flow from LOFTRAN. The DNBR transients presented represent the minimum of the typical cell or the thimble cell.

15.3.1.2.2 Initial Conditions

Initial reactor power, pressure, and reactor coolant system temperature are assumed to be at their nominal values. Uncertainties in initial conditions are included in the DNBR limit, as described in WCAP-11397-P-A (Reference 5).

Plant characteristics and initial conditions assumed in this analysis are further discussed in Subsection 15.0.3.

15.3.1.2.3 Reactivity Coefficients

A conservatively large absolute value of the Doppler-only power coefficient is used (see Figure 15.0.4-1). This is equivalent to a total integrated Doppler reactivity from 0- to 100-percent power of 0.0160 Δk.

The least-negative moderator temperature coefficient is assumed because this results in the maximum core power during the initial part of the transient, when the minimum DNBR is reached.

For these analyses, a curve of trip reactivity versus time based on a 2.5-second rod cluster control assembly insertion time to the dashpot is used (see Subsection 15.0.5).

15.3.1.2.4 Flow Coastdowns

Conservative flow coastdowns are used to simulate the transient. The flow coastdowns are calculated externally to the LOFTRAN code using the COAST computer code which is described in Subsection 15.0.11.

Plant systems and equipment necessary to mitigate the effects of the accident are discussed in Subsection 15.0.8 and listed in Table 15.0-6. No single active failure in any of these systems or equipment adversely affects the consequences of the accident.

15.3.1.2.5 Results

Figures 15.3.1-1 through 15.3.1-6 show the transient response for the loss of two reactor coolant pumps with offsite power available. Figure 15.3.1-6 shows the DNBR to be always greater than the design limit value as defined in Section 4.4.

The plant is tripped by the low-flow trip rapidly enough so that the capability of the reactor coolant to remove heat from the fuel rods is not greatly reduced. The average fuel and cladding temperatures do not increase significantly above their initial values.

The calculated sequence of events for the case analyzed is shown in Table 15.3-1. The affected reactor coolant pumps continue to coast down, and the core flow reaches a new equilibrium value.

With the reactor tripped, a stable plant condition is attained. Normal plant shutdown may then proceed.

In the event that a loss of offsite power occurs as a consequence of a turbine trip during a partial loss of reactor coolant flow, the DNB design basis continues to be met. The loss of offsite power causes the remaining two operating reactor coolant pumps to coast down.

At the time when the remaining two operating reactor coolant pumps start coasting down, reactor trip has already been initiated, core heat flux has started decreasing, and DNBR is increasing. DNBR continues to increase as the remaining two reactor coolant pumps coast down because the core heat flux has decreased and is continuing to decrease rapidly. The minimum DNB ratio occurs at the same time for cases with and without offsite power available.

15.3.1.3 Conclusions

The analysis shows that, for the partial loss of reactor coolant flow, the DNBR does not decrease below the design basis value at any time during the transient. The DNBR design basis is described in Section 4.4. The applicable Standard Review Plan, Subsection 15.3.1 (Reference 4), evaluation criteria are met.

15.3.2 Complete Loss of Forced Reactor Coolant Flow

15.3.2.1 Identification of Causes and Accident Description

A complete loss of flow accident may result from a simultaneous loss of electrical supplies to the reactor coolant pumps. If the reactor is at power at the time of the accident, the immediate effect of a loss of coolant flow is a rapid increase in the coolant temperature.

Electric power for the reactor coolant pumps is supplied through buses, connected to the generator through the unit auxiliary transformers. When a generator trip occurs, the buses receive power from external power lines and the pumps continue to supply coolant flow to the core.

A complete loss of flow accident is a Condition III event (an infrequent fault), as defined in Subsection 15.0.1. The following signals provide protection against this event:

- Reactor coolant pump underspeed
- Low reactor coolant loop flow

The reactor trip on reactor coolant pump underspeed protects against conditions that can cause a loss of voltage to the reactor coolant pumps. This function is blocked below approximately 10-percent power (permissive P10).

The reactor trip on reactor coolant pump underspeed is also provided to trip the reactor for an underfrequency condition resulting from frequency disturbances on the power grid. If the maximum grid frequency decay rate is less than approximately 5 hertz per second, this trip protects the core from underfrequency events. WCAP-8424, Revision 1 (Reference 3), provides analyses of grid frequency disturbances and the resulting protection requirements that are applicable to the AP1000.

The reactor trip on low primary coolant loop flow is provided to protect against loss of flow conditions that affect only one or two reactor coolant loop cold legs. This function is generated by two-out-of-four low-flow signals per reactor coolant loop hot leg. Above permissive P10, low flow in either hot leg actuates a reactor trip. If the maximum grid frequency decay rate is less than approximately 2.5 hertz per second, this trip function also protects the core from this underfrequency event. This effect is described in WCAP-8424, Revision 1 (Reference 3).

15.3.2.2 Analysis of Effects and Consequences

15.3.2.2.1 Method of Analysis

The complete loss of flow transient is analyzed for a loss of power to four reactor coolant pumps.

For the case analyzed with a complete loss of voltage, followed by the reactor coolant pumps coasting down, the method of analysis and the assumptions made regarding initial operating conditions and reactivity coefficients are identical to those discussed in Subsection 15.3.1, with one exception. Following the loss of power supply to all pumps at power, a reactor trip is actuated by the reactor coolant pump underspeed trip.

A loss of forced primary coolant flow can result from a reduction in the reactor coolant pump motor supply frequency. The results of the complete loss of voltage, followed by the reactor coolant pump coasting down, bound the complete loss of flow initiated by a frequency decay of up to 5 hertz per second. Therefore, only the results of the complete loss of voltage case are presented in Subsection 15.3.2.2.2.

15.3.2.2.2 Results

Figures 15.3.2-1 through 15.3.2-6 show the transient response for the complete loss of voltage to all four reactor coolant pumps. The reactor is assumed to trip on the reactor coolant pump underspeed signal. Figure 15.3.2-6 shows that the DNBR is always greater than the design limit value defined in Section 4.4.

The calculated sequences of events for the cases analyzed are shown in Table 15.3-1. The reactor coolant pumps continue to coast down, and natural circulation flow is established, as demonstrated in Subsection 15.2.6. With the reactor tripped, a stable plant condition is attained. Normal plant shutdown may then proceed.

15.3.2.3 Conclusions

The analysis demonstrates that, for the complete loss of forced reactor coolant flow, the DNBR does not decrease below the design basis limit value at any time during the transient. The design basis for the DNBR is described in Section 4.4. The applicable Standard Review Plan, Subsection 15.3.1 (Reference 4), evaluation criteria are met.

15.3.3 Reactor Coolant Pump Shaft Seizure (Locked Rotor)

15.3.3.1 Identification of Causes and Accident Description

The accident postulated is an instantaneous seizure of a reactor coolant pump rotor, as discussed in Section 5.4. Flow through the affected reactor coolant loop is rapidly reduced, leading to a reactor trip on a low-flow signal.

Following the reactor trip, heat stored in the fuel rods continues to be transferred to the coolant, causing the coolant temperature to increase and expand. At the same time, heat transfer to the shell side of the steam generator in the faulted loop is reduced because: 1) the reduced flow results in a decreased tube-side film coefficient, and 2) the reactor coolant in the tubes cools down while the shell-side temperature increases. (Turbine steam flow is reduced to 0 upon plant trip.) The rapid expansion of the coolant in the reactor core, combined with reduced heat transfer in the steam generators, causes an insurge into the pressurizer and a pressure increase throughout the reactor coolant system. The insurge into the pressurizer compresses the steam volume, actuates the automatic spray system, and opens the pressurizer safety valves, in that sequence. For conservatism, the pressure-reducing effect of the spray is not included in the analysis.

This event is classified as a Condition IV incident (a limiting fault), as defined in Subsection 15.0.1.

15.3.3.2 Analysis of Effects and Consequences

15.3.3.2.1 Method of Analysis

Two digital computer codes are used to analyze this transient. The LOFTRAN code (Reference 1) calculates the resulting core flow transient following the pump seizure and the nuclear power following reactor trip. This code is also used to determine the peak pressure. The thermal behavior of the fuel located at the core hot spot is investigated by using the FACTRAN code (Reference 2). This code uses the core flow and the nuclear power calculated by LOFTRAN. The FACTRAN code includes a film-boiling heat transfer coefficient.

At the beginning of the postulated locked rotor accident (at the time the shaft in one of the reactor coolant pumps is assumed to seize), the plant is assumed to be in operation under the most adverse steady-state operating conditions, that is, maximum steady-state thermal power, maximum steady-state pressure, and maximum steady-state coolant average temperature. Plant characteristics and initial conditions are further discussed in Subsection 15.0.3. The accident is evaluated for both cases with and without offsite power available. For the case without offsite power available, power is lost to the unaffected pumps at 3.0 seconds following turbine/generator trip. Turbine trip occurs 5.0 seconds following a reactor trip condition being reached. This delay on turbine trip is a feature of the AP1000 reactor trip system.

For the peak pressure evaluation, the initial pressure is conservatively estimated as 50 psi above nominal pressure (2250 psia), which allows for errors in the pressurizer pressure measurement and control channels. This is done to obtain the highest possible rise in the coolant pressure during the transient. To obtain the maximum pressure in the primary side, conservatively high loop pressure drops are added to the calculated pressurizer pressure.

15.3.3.2.2 Evaluation of the Pressure Transient

After pump seizure, the neutron flux is rapidly reduced by control rod insertion. Rod motion is assumed to begin 1.45 seconds after the flow in the affected loop reaches the reactor trip setpoint. No credit is taken for the pressure-reducing effect of the pressurizer spray, steam dump, or controlled feedwater flow after plant trip. Although these operations are expected to result in a lower peak reactor coolant system pressure, an additional conservatism is provided by ignoring their effect.

The pressurizer safety valves are fully open at 2575 psia. Their capacity for steam relief is described in Section 5.4.

15.3.3.2.3 Evaluation of Departure from Nucleate Boiling in the Core During the Accident

For this accident, an evaluation of the consequences with respect to fuel rod thermal transients is performed. Results obtained from analysis of this “hot spot” condition represent the upper limit with respect to cladding temperature and zirconium-water reaction.

In the evaluation, the rod power at the hot spot is conservatively assumed to be 2.6 times the average rod power (that is, $F_Q = 2.6$) at the initial core power level.

15.3.3.2.4 Film-Boiling Coefficient

The film-boiling coefficient is calculated in the FACTRAN code (Reference 2) using the Bishop-Sandberg-Tong film-boiling correlation. The fluid properties are evaluated at film temperature (average between wall and bulk temperatures). The program calculates the film coefficient at every time step, based upon the actual heat transfer conditions at the time. The nuclear power, system pressure, bulk density, and mass flow rate as a function of time are used as program input.

For this analysis, the initial values of the pressure and the bulk density are used throughout the transient because they are the most conservative with respect to cladding temperature response. For conservatism, DNB is assumed to start at the beginning of the accident.

15.3.3.2.5 Fuel Cladding Gap Coefficient

The magnitude and time dependence of the heat transfer coefficient between fuel and cladding (gap coefficient) have a pronounced influence on the thermal results. The larger the value of the gap coefficient, the more heat is transferred between the pellet and the cladding. Based on investigations on the effect of the gap coefficient upon the maximum cladding temperature during the transient, the gap coefficient is assumed to increase from a steady-state value consistent with initial fuel temperature to 10,000 Btu/h-ft^2-°F at the initiation of the transient. Thus, the large amount of energy stored in the fuel because of the small initial value of the gap coefficient is released to the cladding at the initiation of the transient.

15.3.3.2.6 Zirconium-Steam Reaction

The zirconium-steam reaction can become significant above a cladding temperature of 1800°F. The Baker-Just parabolic rate equation is used to define the rate of the zirconium-steam reaction:

$$\frac{d\,(w^2)}{dt} = 33.3 \times 10^6 \exp\left(-\frac{45{,}500}{1.986\,T} \right)$$

where:

w = amount reacted (mg/cm^2)

t = time (s)

T = temperature (Kelvin)

The reaction heat is 1510 cal/g.

The effect of the zirconium-steam reaction is included in the calculation of the hot spot cladding temperature transient.

Plant systems and equipment available to mitigate the effects of the accident are discussed in Subsection 15.0.8 and listed in Table 15.0-6. No single active failure in any of these systems or equipment adversely affects the consequences of the accident.

15.3.3.2.7 Results

Figures 15.3.3-1 through 15.3.3-7 show the transient results for one locked rotor with four reactor coolant pumps in operation with and without offsite power available. The without-offsite-power case bounds the results for the case with offsite power. The results of these calculations are also summarized in Table 15.3-2. The peak reactor coolant system pressure reached during the transient is less than that which causes stresses to exceed the faulted condition stress limits of the ASME Code, Section III. Also, the peak cladding surface temperature is considerably less than 2700°F. The cladding temperature is conservatively calculated, assuming that DNB occurs at the initiation of the transient. These results represent the most limiting conditions with respect to the locked rotor event or the pump shaft break.

The calculated sequence of events for the case analyzed is shown in Table 15.3-1. With the reactor tripped, a stable plant condition is eventually attained. Normal plant shutdown may then proceed.

15.3.3.3 Radiological Consequences

The evaluation of the radiological consequences of a postulated locked reactor coolant pump rotor accident assumes that the reactor has been operating with the design basis fuel defect level (0.25 percent of power produced by fuel rods containing cladding defects) and that leaking steam generator tubes have resulted in a buildup of activity in the secondary coolant.

As a result of the accident, it is determined that no fuel rods are damaged such that the activity contained in the fuel-cladding gap is released to the reactor coolant. However, a conservative analysis has been performed assuming 10 percent of the rods are damaged. Activity carried over to the secondary side because of primary-to-secondary leakage is available for release to the environment via the steam line safety valves or the power-operated relief valves.

15.3.3.3.1 Source Term

The significant radionuclide releases due to the locked rotor accident are the iodines, alkali metals (cesiums, rubidiums) and noble gases. The reactor coolant iodine source term assumes a pre-existing iodine spike. The initial reactor coolant noble gas and alkali metal concentrations are assumed to be those associated with the design basis fuel defect level. These initial reactor coolant activities are of secondary importance compared to the release of the gap inventory of fission products from the portion of the core assumed to fail because of the accident.

Based on NUREG-1465 (Reference 6), the fission product gap fraction is 3 percent of fuel inventory. For this analysis, the gap fraction is increased to 8 percent of the inventory for I-131, 10 percent for Kr-85, 5 percent for other iodines and noble gases and 12 percent for alkali metals. Also, to address the fact that the failed fuel rods may have been operating at power levels above the core average, the source term is increased by the lead rod radial peaking factor.

The initial secondary coolant activity is assumed to be 1 percent of the maximum equilibrium primary coolant activity for iodines and alkali metals.

15.3.3.3.2 Release Pathways

There are two components to the accident releases:

- The activity initially in the secondary coolant is available for release as long as steam releases continue.
- The reactor coolant leaking into the steam generators is assumed to mix with the secondary coolant. The activity from the primary coolant mixes with the secondary coolant. As steam is released, a portion of the iodine and alkali metal activity in the coolant is released. The fraction of activity released is defined by the assumed flashing fraction and the partition

coefficient assumed for the steam generator. The noble gas activity entering the secondary side is released to the environment. These releases are terminated when the steam releases stop.

Credit is taken for the decay of radionuclides until release to the environment. After release to the environment, no consideration is given to radioactive decay or to cloud depletion by ground deposition during transport offsite.

15.3.3.3.3 Dose Calculation Models

The models used to calculate offsite doses are provided in Appendix 15A.

15.3.3.3.4 Analytical Assumptions and Parameters

The assumptions and parameters used in the analysis are listed in Table 15.3-3.

Two separate accident scenarios are addressed. In the first scenario, it is assumed that the non-safety grade startup feedwater system is not available to provide feedwater to the steam generators. In this event, the water level in the steam generators drops, resulting in tube uncovery and there is flashing of a portion of the primary coolant assumed to be leaking into the secondary side of the steam generators. Also, the period of steaming is terminated at 1.5 hours when the capacity of the passive residual heat removal system exceeds the decay heat generation rate.

In the second scenario, it is assumed that the startup feedwater system is available to maintain water level in the steam generators such that the tubes remain covered. In this scenario, direct release of flashed primary coolant is not considered. Also, the passive residual heat removal system does not actuate, resulting in a longer period of steaming releases.

15.3.3.3.5 Identification of Conservatisms

The assumptions used in the analysis contain a number of significant conservatisms:

- Although fuel damage is assumed to occur as a result of the accident, no fuel damage is anticipated.

- The reactor coolant activities are based on a fuel defect level of 0.25 percent; whereas, the expected fuel defect level is far less than this (see Section 11.1).

- The leakage of reactor coolant into the secondary system, at 300 gallons per day, is conservative. The leakage is normally a small fraction of this.

- It is unlikely that the conservatively selected meteorological conditions are present at the time of the accident.

15.3.3.3.6 Doses

Using the assumptions from Table 15.3-3, the calculated total effective dose equivalent (TEDE) doses are determined to be less than 0.5 rem at the exclusion area boundary for the limiting 2-hour interval (0 to 2 hours) and less than 0.2 rem at the low population zone outer boundary for the scenario in which there is no feedwater available to maintain water level in the steam generators. The doses for the scenario in which it is assumed that water level in the steam generators is maintained are 0.4 rem at the exclusion area boundary for the limiting 2-hour interval of 6 to 8 hours and 0.4 rem at the low population zone outer boundary. These doses are a small fraction of the dose guideline of

25 rem TEDE identified in 10 CFR Part 50.34. A "small fraction" is identified as 10 percent or less consistent with the Standard Review Plan (Reference 4).

At the time the locked reactor coolant pump rotor event occurs, the potential exists for a coincident loss of spent fuel pool cooling with the result that the pool could reach boiling and a portion of the radioactive iodine in the spent fuel pool could be released to the environment. The loss of spent fuel pool cooling has been evaluated for a duration of 30 days. There is no contribution to the 2-hour site boundary dose because the pool boiling would not occur until after the first 2 hours. The 30-day contribution to the dose at the low population zone boundary is less than 0.01 rem TEDE, and when this is added to the dose calculated for the locked rotor event, the resulting total dose remains less than the value reported above.

15.3.4 Reactor Coolant Pump Shaft Break

15.3.4.1 Identification of Causes and Accident Description

The accident is postulated as an instantaneous failure of a reactor coolant pump shaft. Flow through the affected reactor coolant loop is rapidly reduced, though the initial rate of reduction of coolant flow is greater for the reactor coolant pump rotor seizure event. Reactor trip occurs on a low-flow signal in the affected loop.

Following the reactor trip, heat stored in the fuel rods continues to be transferred to the coolant, causing the coolant to expand. At the same time, heat transfer to the shell side of the steam generator in the faulted loop is reduced because: 1) the reduced flow results in a decreased tube-side film coefficient, and 2) the reactor coolant in the tubes cools down while the shell-side temperature increases. (Turbine steam flow is reduced to 0 upon plant trip.) The rapid expansion of the coolant in the reactor core, combined with reduced heat transfer in the steam generators, causes an insurge into the pressurizer and a pressure increase throughout the reactor coolant system. The insurge into the pressurizer compresses the steam volume, actuates the automatic spray system, and opens the pressurizer safety valves, in that sequence. For conservatism, the pressure-reducing effect of the spray is not included in the analysis.

This event is classified as a Condition IV incident (limiting fault), as defined in Subsection 15.0.1.

15.3.4.2 Conclusion

With a failed shaft, the impeller could be free to spin in a reverse direction as opposed to being fixed in position as is the case when a locked rotor occurs. This results in a decrease in the end point (steady-state) core flow. For both the shaft break and locked rotor incidents, reactor trip occurs very early in the transient. In addition, the locked rotor analysis conservatively assumes that DNB occurs at the beginning of the transient. The calculated results presented for the locked rotor analysis bound the reactor coolant pump shaft break event.

15.3.5 Combined License Information

This section contained no requirement for additional information.

15.3.6 References

1. Burnett, T. W. T., et al., "LOFTRAN Code Description," WCAP-7907-P-A (Proprietary) and WCAP-7907-A (Nonproprietary), April 1984.

2. Hargrove, H. G., "FACTRAN - A FORTRAN-IV Code for Thermal Transients in a UO_2 Fuel Rod," WCAP-7908-A, December 1989.

3. Baldwin, M. S., et al., “An Evaluation of Loss of Flow Accidents Caused by Power System Frequency Transients in Westinghouse PWRs,” WCAP-8424, Revision 1, May 1975.

4. NUREG-0800, “Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants,” U.S. Nuclear Regulatory Commission, July 1981.

5. Friedland, A. J., and Ray, S., “Revised Thermal Design Procedure,” WCAP-11397-P-A (Proprietary) and WCAP-11397-A (Nonproprietary), April 1989.

6. Soffer, L., et al., “Accident Source Terms for Light-Water Nuclear Power Plants,” NUREG-1465, February 1995.

Table 15.3-1
Time Sequence of Events for Incidents
That Result in a Decrease In Reactor Coolant System Flow Rate

Accident	Event	Time (seconds)
Partial loss of forced reactor coolant flow		
– Loss of two pumps with four pumps running	Coastdown begins Low-flow reactor trip Rods begin to drop Minimum DNBR occurs	0.00 1.61 3.06 4.90
Complete loss of forced reactor coolant		
– Loss of four pumps with four pumps running	Operating pumps lose power and begin coasting down Reactor coolant pump underspeed trip point reached Rods begin to drop Minimum DNBR occurs	0.00 0.47 1.24 3.0
Reactor coolant pump shaft seizure (locked rotor)		
– One locked rotor with four pumps running with offsite power available	Rotor on one pump locks Low-flow trip point reached Rods begin to drop Maximum reactor coolant system pressure occurs Maximum cladding temperature occurs	0.00 0.10 1.55 2.30 3.90
– One locked rotor with four pumps running without offsite power available	Rotor on one pump locks Low-flow trip point reached Rods begin to drop Maximum reactor coolant system pressure occurs Maximum cladding temperature occurs	0.00 0.10 1.55 2.30 3.90

Table 15.3-2
Summary of Results for Locked Rotor Transients
(Four Reactor Coolant Pumps Operating Initially)

	Without Offsite Power Available
Maximum reactor coolant system pressure (psia)	2703
Maximum cladding temperature, core hot spot (°F)	1819
Zr-H_2O reaction, core hot spot (percentage by weight)	0.30

Table 15.3-3 (Sheet 1 of 2)
Parameters Used in Evaluating the Radiological Consequences of a Locked Rotor Accident

Initial reactor coolant iodine activity	An assumed iodine spike that has resulted in an increase in the reactor coolant activity to 60 µCi/gm of dose equivalent I-131 (see Appendix 15A)[a]
Reactor coolant noble gas activity	Equal to the operating limit for reactor coolant activity of 280 µCi/gm dose equivalent Xe-133
Reactor coolant alkali metal activity	Design basis activity (see Table 11.1-2)
Secondary coolant initial iodine and alkali metal activity	1% of design basis reactor coolant concentrations at maximum equilibrium conditions
Fraction of fuel rods assumed to fail	0.10
Core activity	See Table 15A-3
Radial peaking factor (for determination of activity in failed fuel rods)	1.75
Fission product gap fractions I-131 Kr-85 Other iodines and noble gases Alkali metals	 0.08 0.10 0.05 0.12
Reactor coolant mass (lb)	3.7 E+05
Secondary coolant mass (lb)	6.04 E+05
Condenser	Not available
Atmospheric dispersion factors	See Table 15A-5
Primary to secondary leak rate (lb/hr)	104.5[b]
Partition coefficient in steam generators iodine alkali metals	 0.01 0.0035
Accident scenario in which startup feedwater is not available Duration of accident (hr) Steam released (lb) 0-1.5 hours[c] Leak flashing fraction[d] 0-60 minutes > 60 minutes	 1.5 hr 6.48 E+05 0.04 0

Table 15.3-3 (Sheet 2 of 2)
Parameters Used in Evaluating the Radiological Consequences of a Locked Rotor Accident

Accident scenario in which startup feedwater is available	
Duration of accident (hr)	8.0 hr
Steam release rate (lb/sec)	60
Leak flashing fraction	Not applicable

Notes:

a. The assumption of a pre-existing iodine spike is a conservative assumption for the initial reactor coolant activity. However, compared to the activity released to the coolant from the assumed fuel failures, it is not significant.

b. Equivalent to 300 gpd cooled liquid at 62.4 lb/ft^3.

c. Heat removal is achieved by steaming and by passive core cooling system operation in the limiting case where the startup feedwater system is not available. When heat removal by the passive core cooling system exceeds the decay heat load, steam releases are terminated.

d. No credit for iodine partitioning is taken for flashed leakage. Credit is taken for a partition coefficient of 0.10 for alkali metals. Flashing is terminated by the passive core cooling system operation reducing the RCS below the saturation temperature of the secondary.

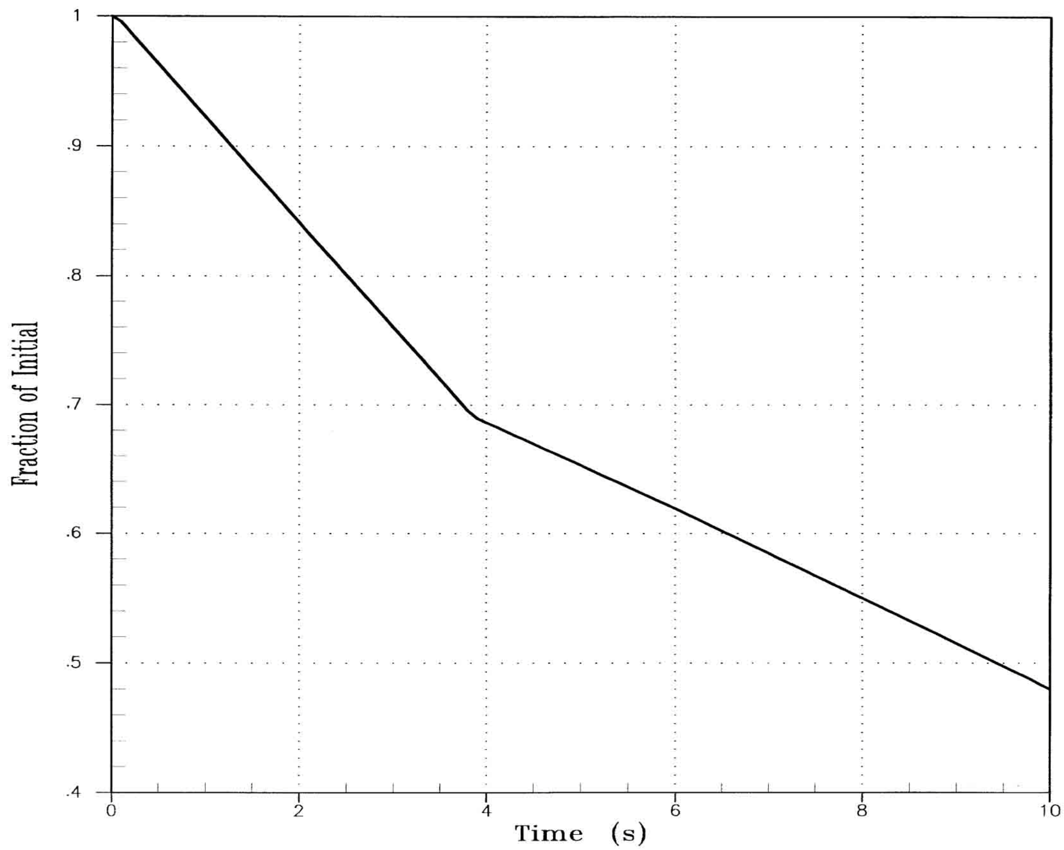


Figure 15.3.1-1
Core Mass Flow Transient for Four Cold Legs in Operation, Two Pumps Coasting Down

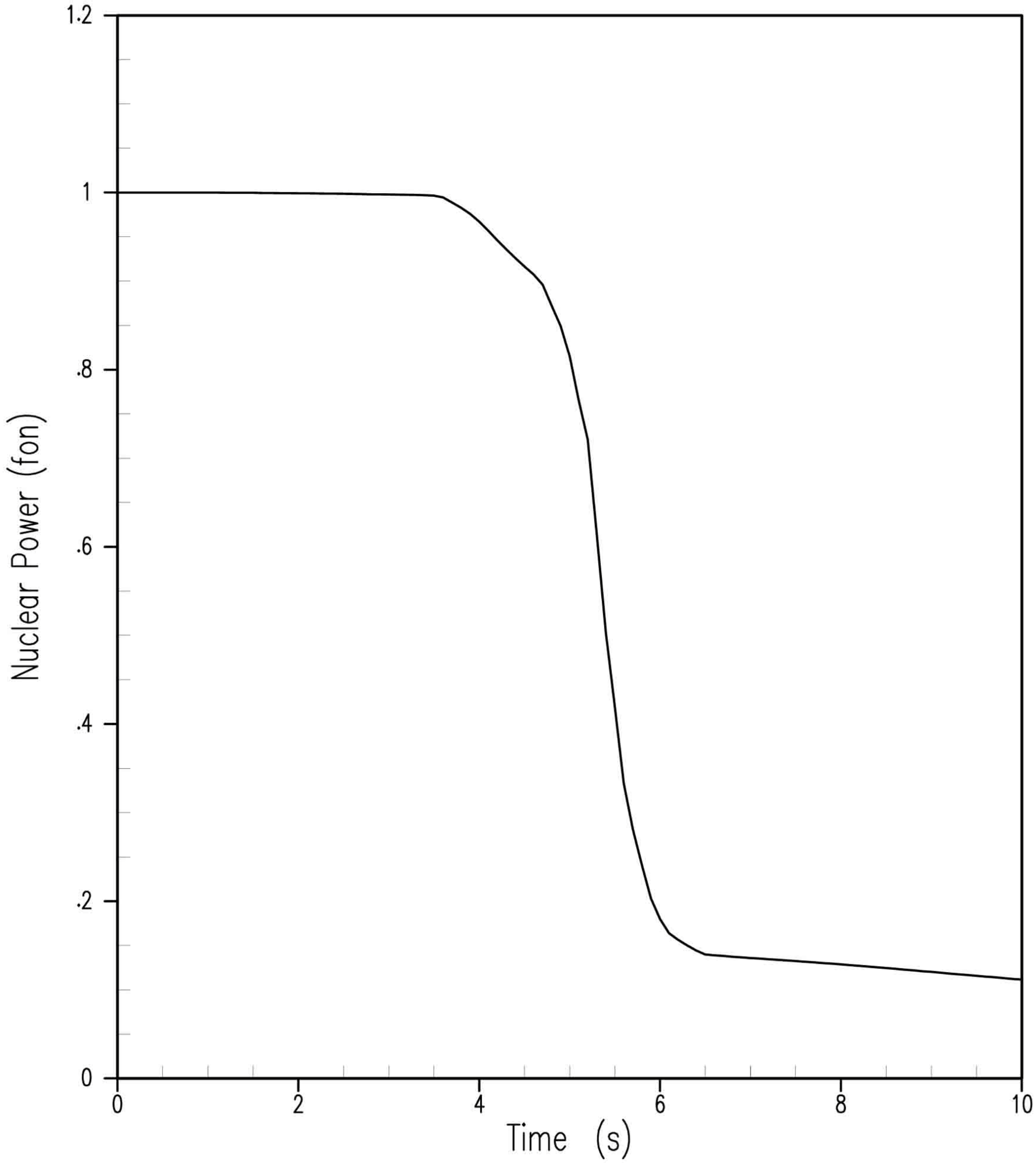


Figure 15.3.1-2
Nuclear Power Transient for Four Cold Legs in Operation, Two Pumps Coasting Down

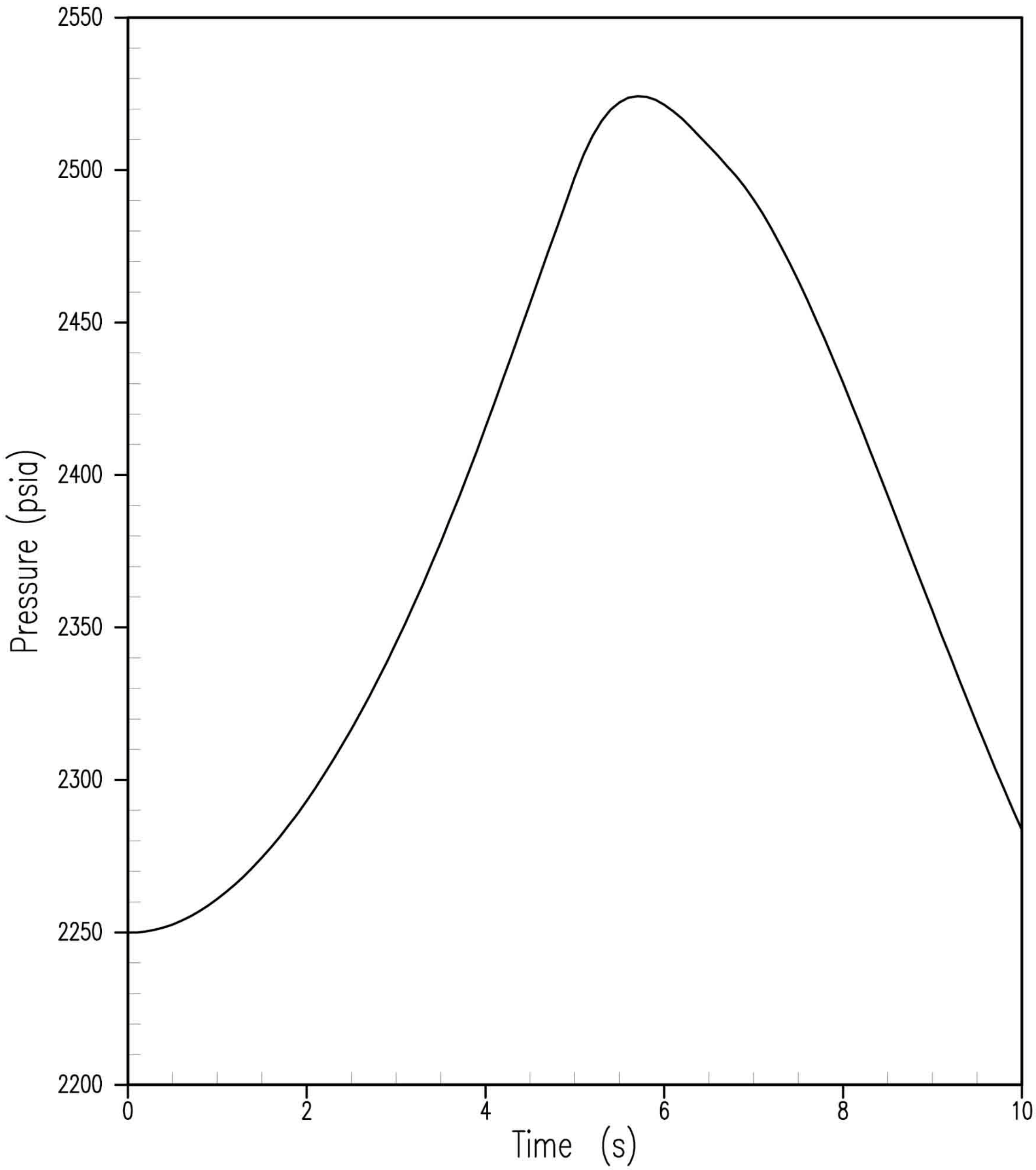


Figure 15.3.1-3
Pressurizer Pressure Transient for Four Cold Legs in Operation, Two Pumps Coasting Down

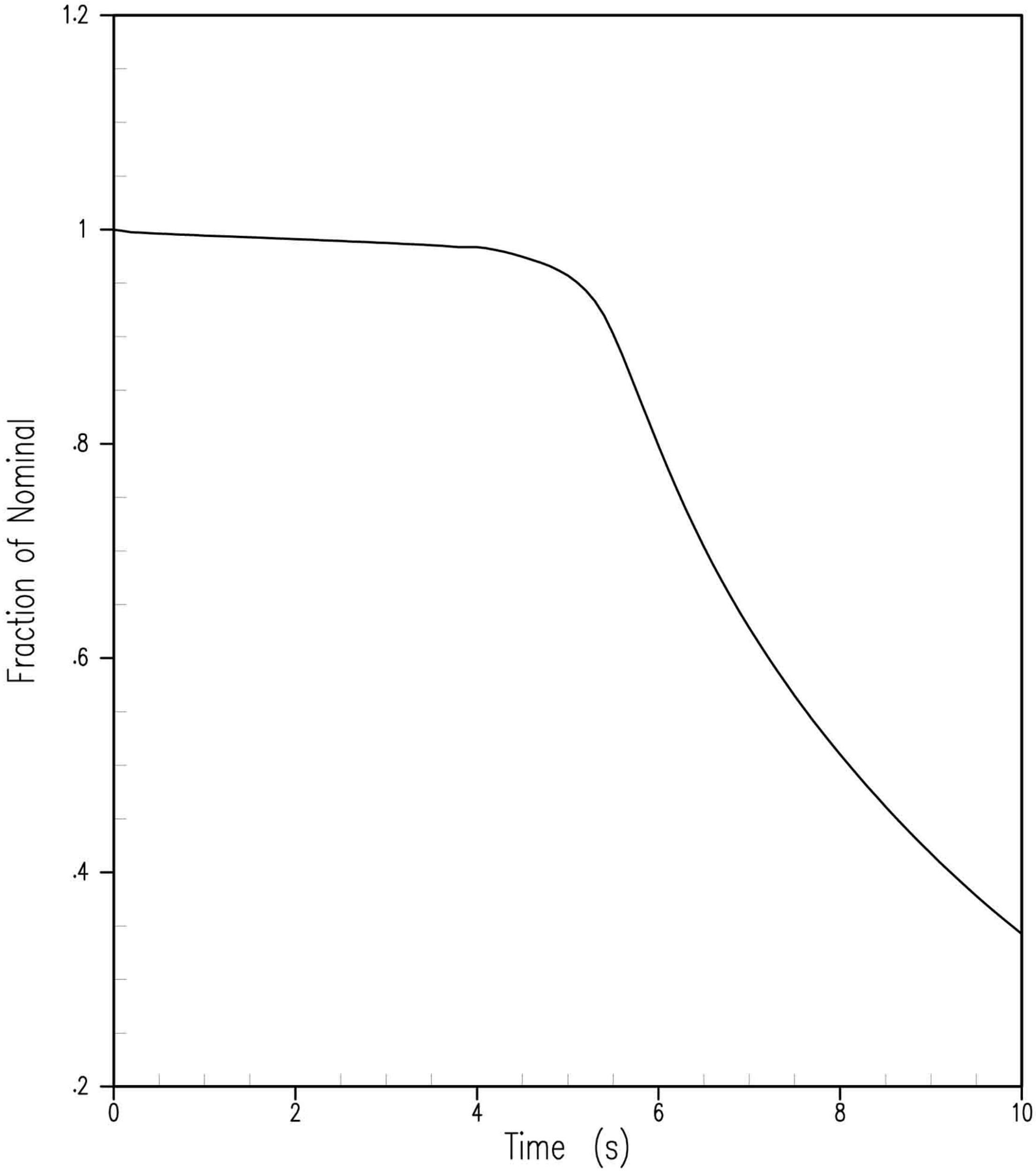


Figure 15.3.1-4
Average Channel Heat Flux Transient for Four Cold Legs in Operation, Two Pumps Coasting Down

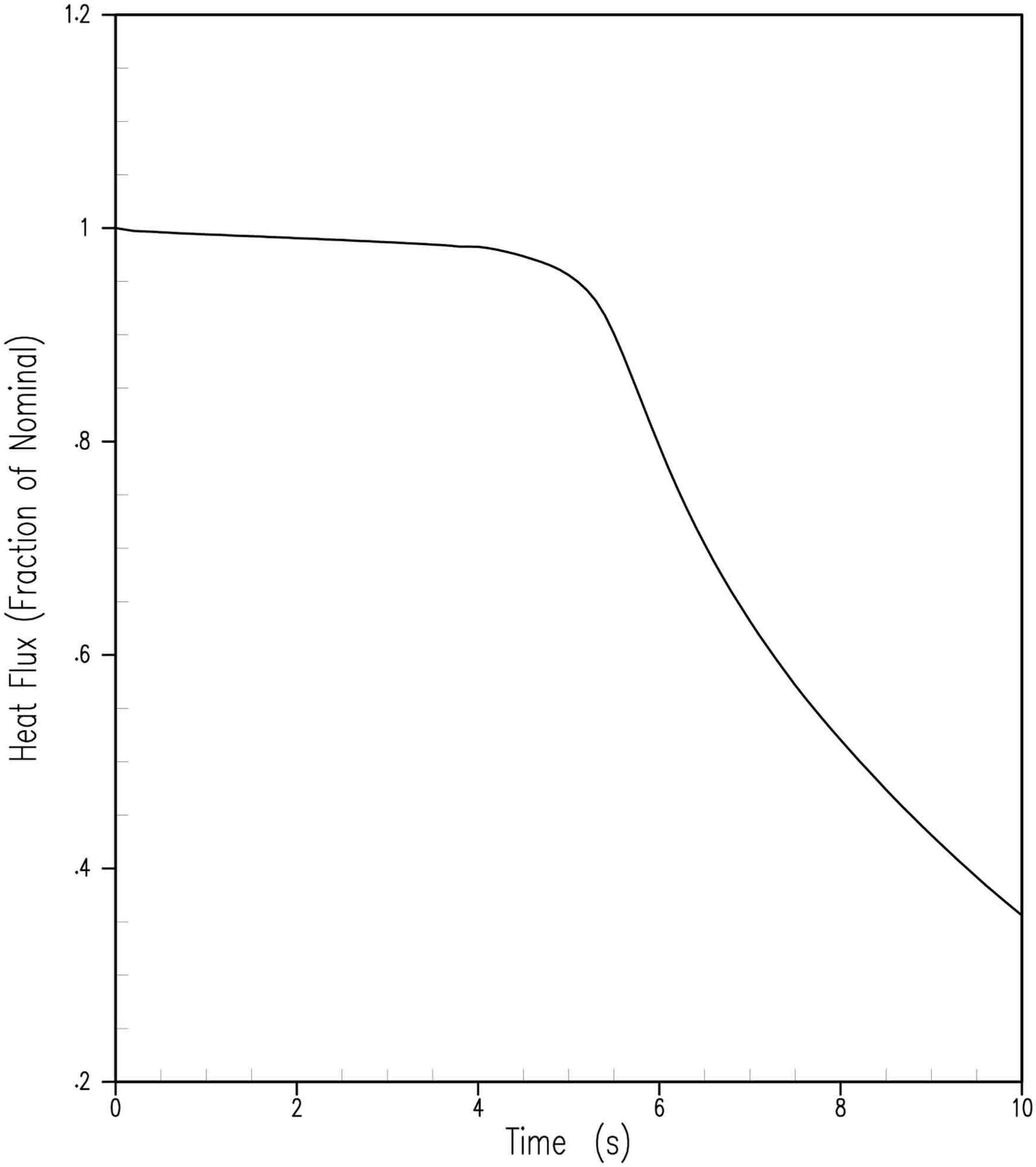


Figure 15.3.1-5
Hot Channel Heat Flux Transient for Four Cold Legs in Operation, Two Pumps Coasting Down

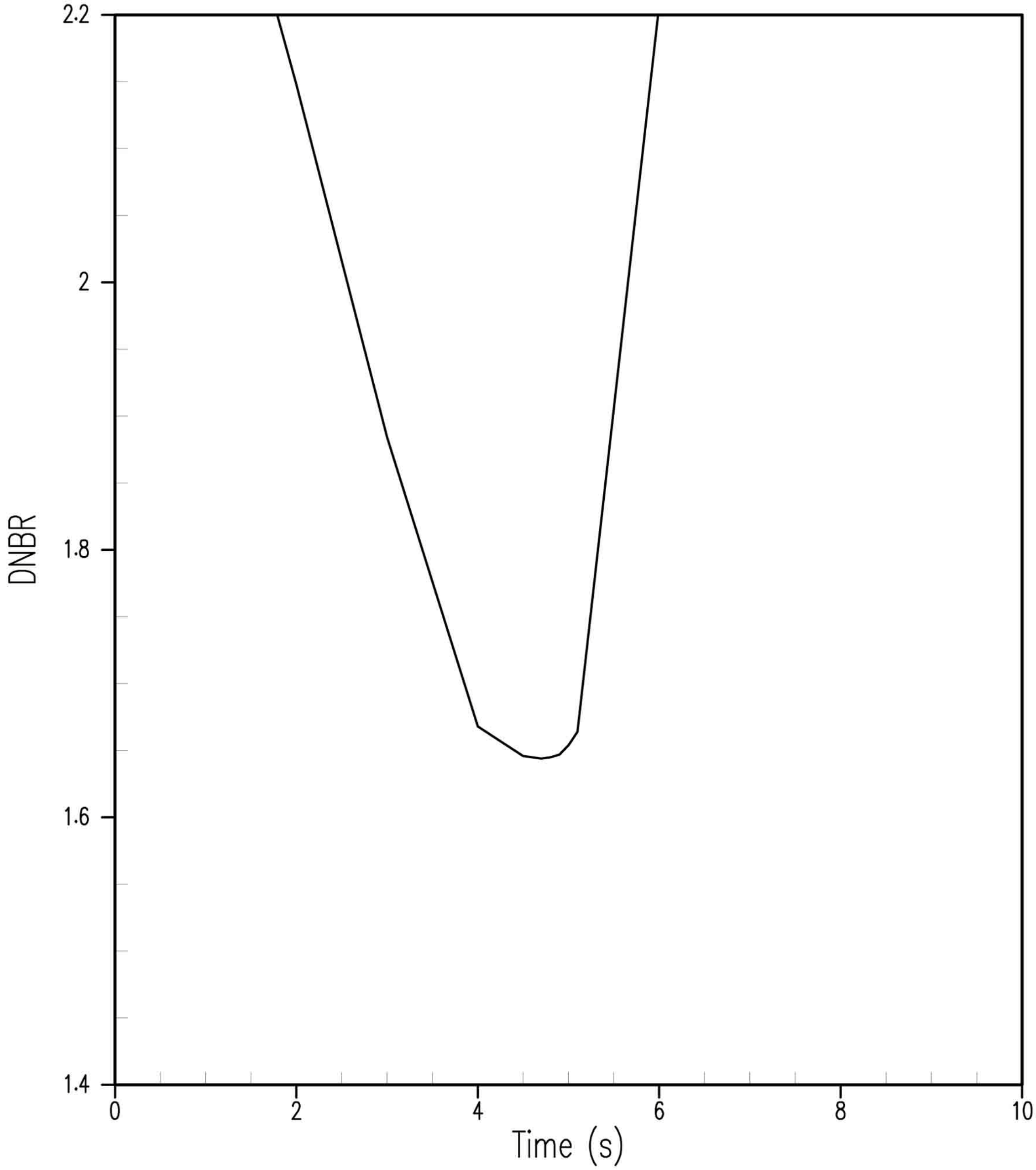


Figure 15.3.1-6
DNB Transient for Four Cold Legs in Operation, Two Pumps Coasting Down

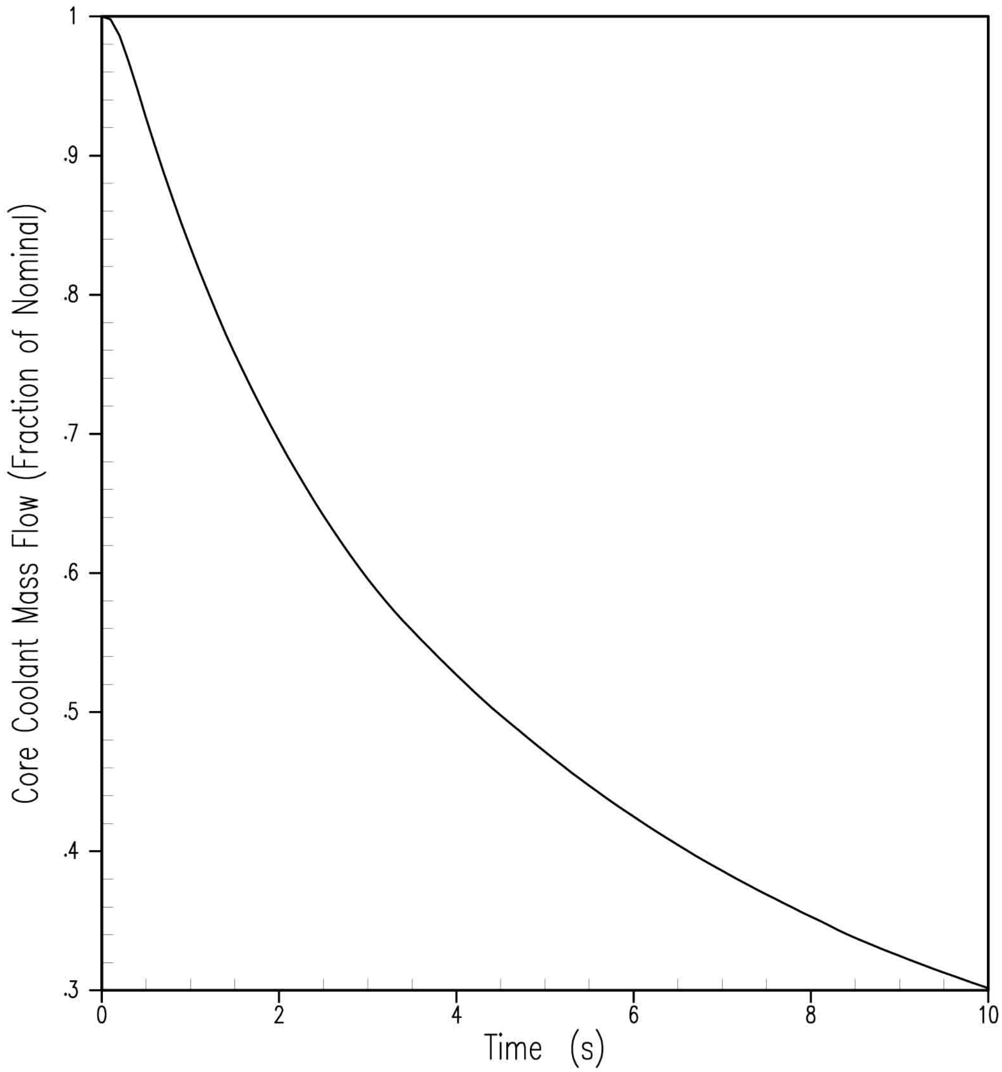


Figure 15.3.2-1
Flow Transient for Four Cold Legs in Operation, Four Pumps Coasting Down

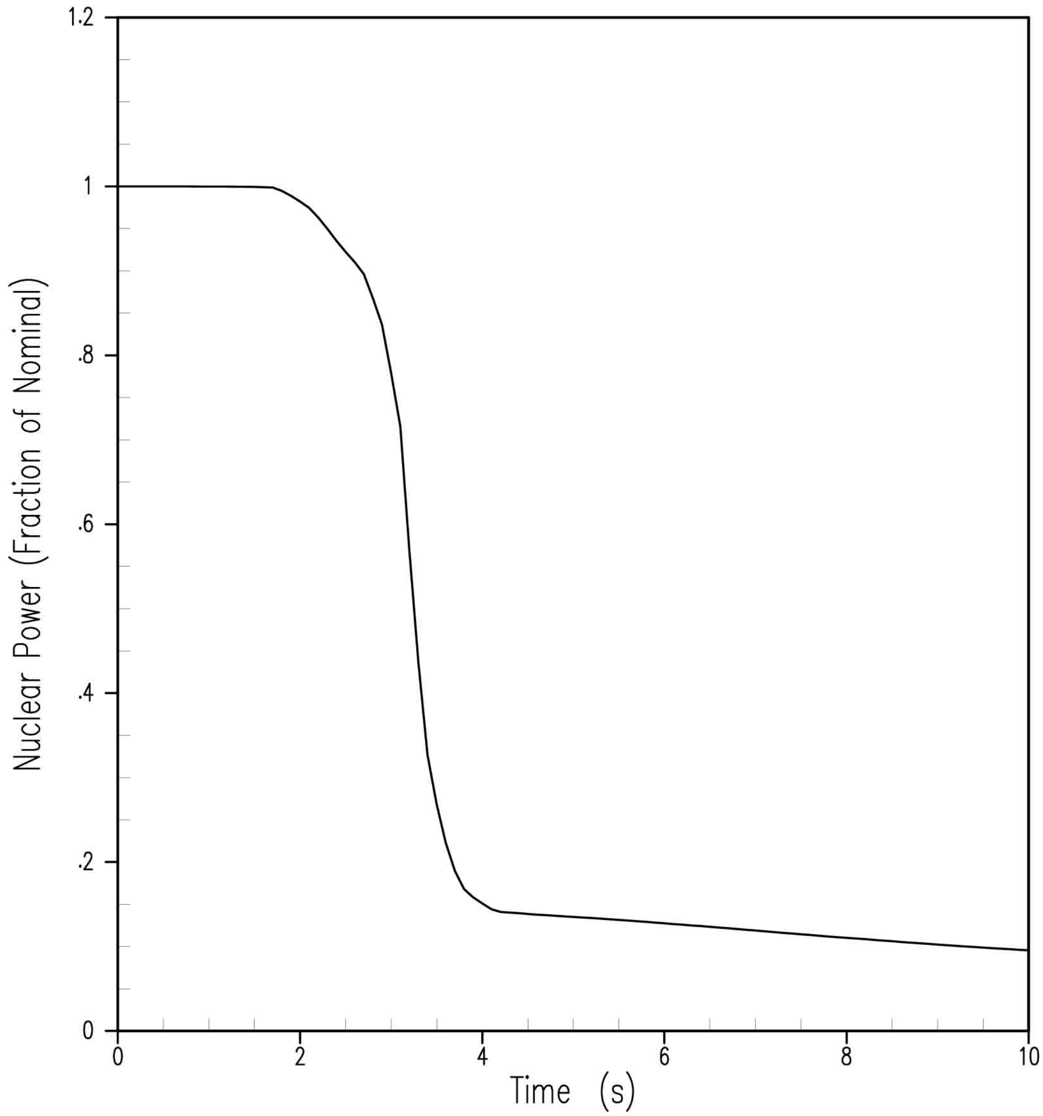


Figure 15.3.2-2
Nuclear Power Transient for Four Cold Legs in Operation, Four Pumps Coasting Down

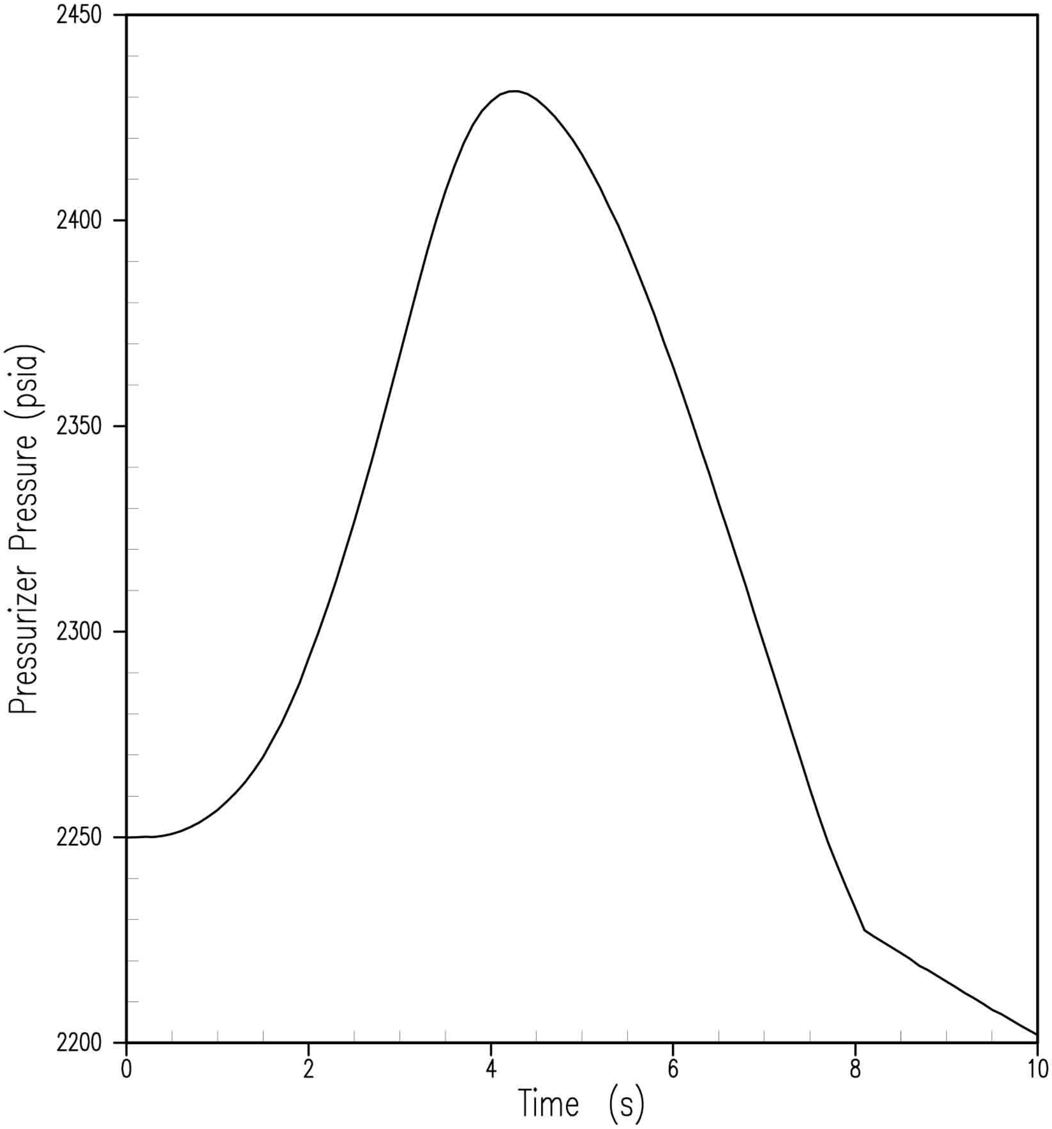


Figure 15.3.2-3
Pressurizer Pressure Transient for Four Cold Legs in Operation, Four Pumps Coasting Down

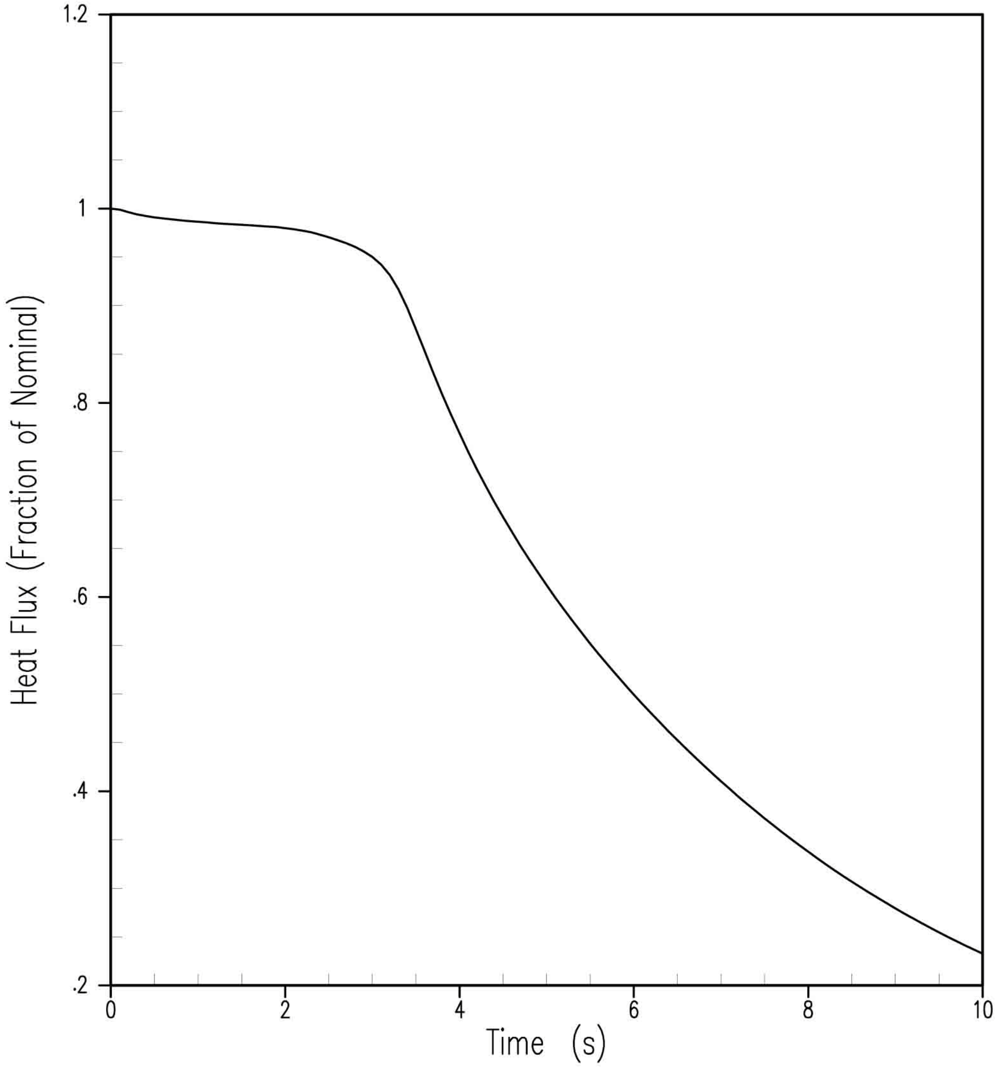


Figure 15.3.2-4
Average Channel Heat Flux Transient for
Four Cold Legs in Operation, Four Pumps Coasting Down

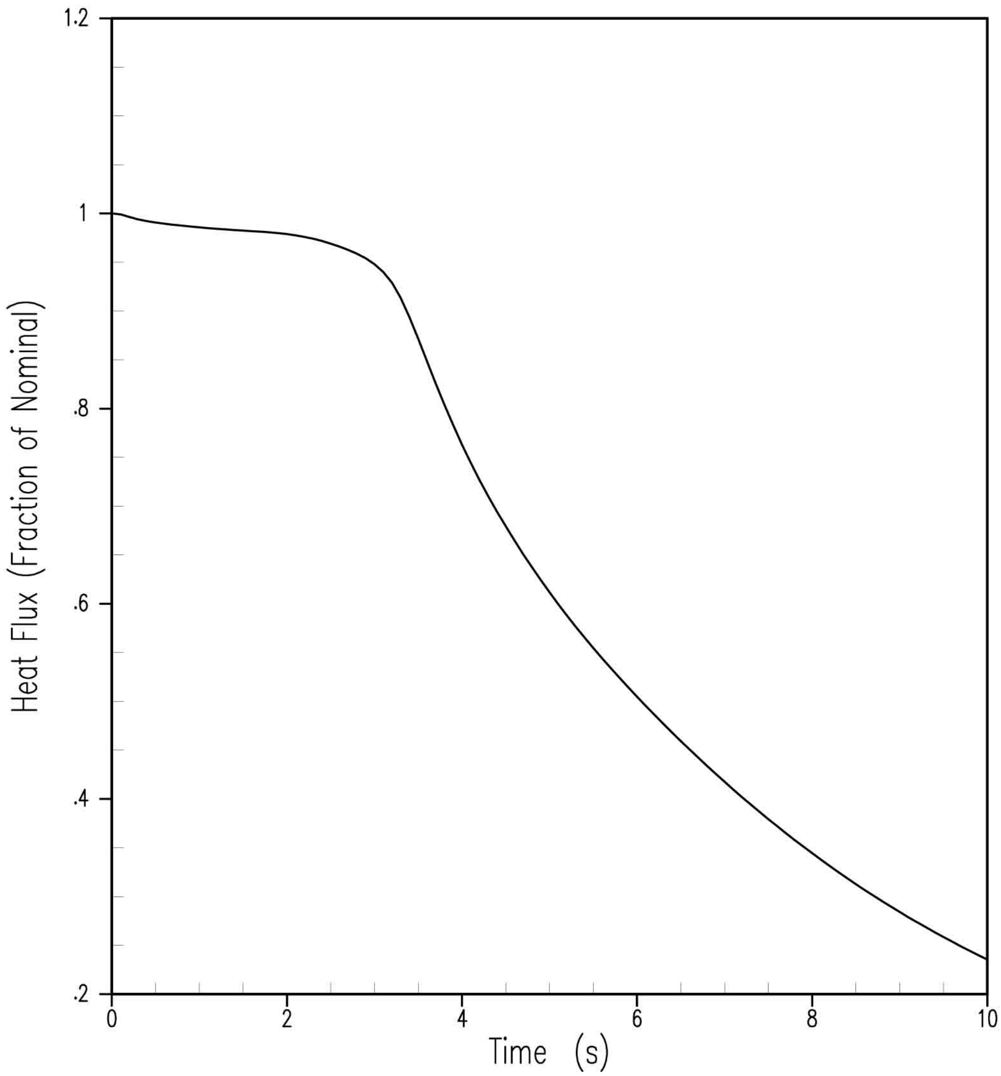


Figure 15.3.2-5
Hot Channel Heat Flux Transient for
Four Cold Legs in Operation, Four Pumps Coasting Down

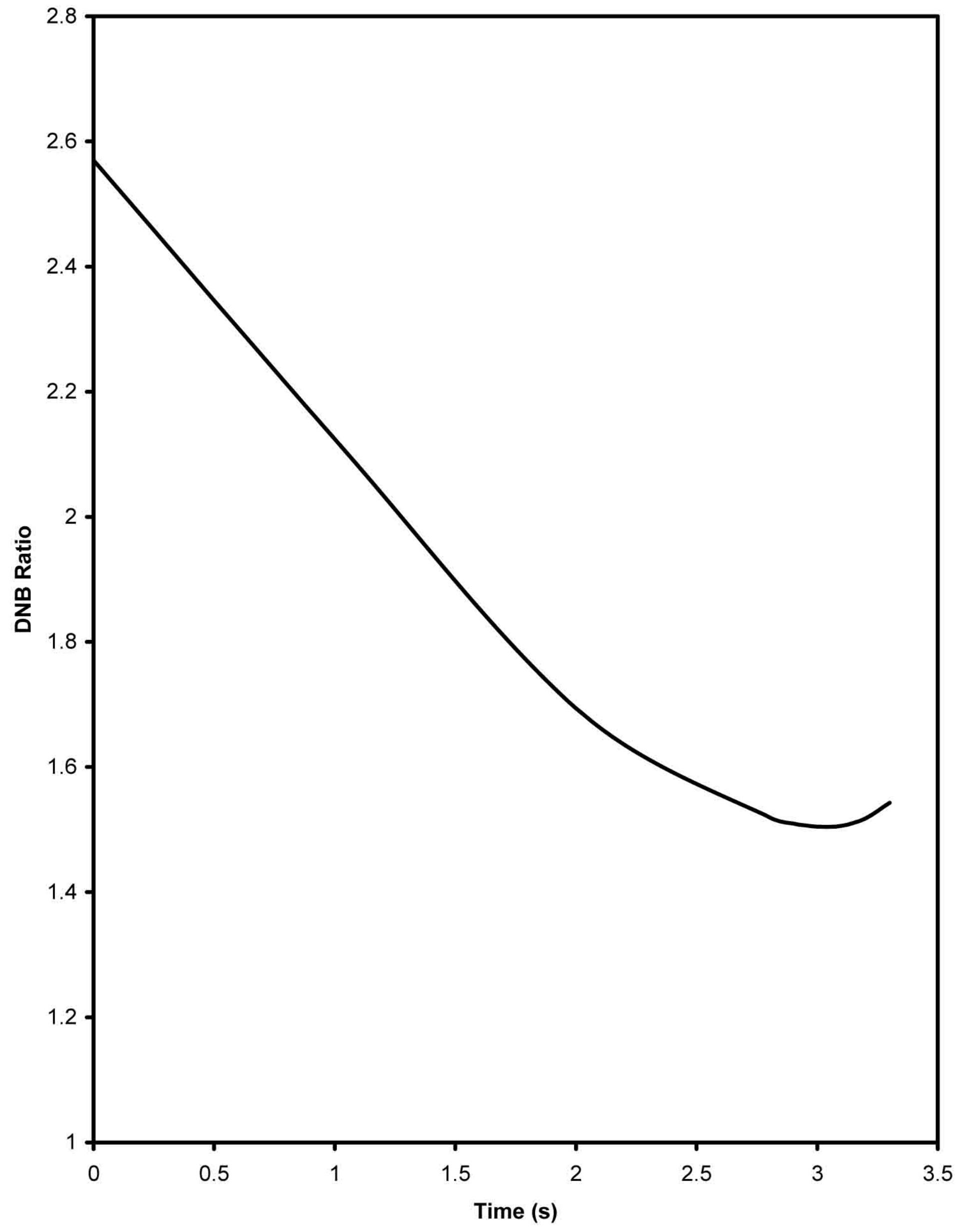


Figure 15.3.2-6
DNBR Transient for Four Cold Legs
in Operation, Four Pumps Coasting Down

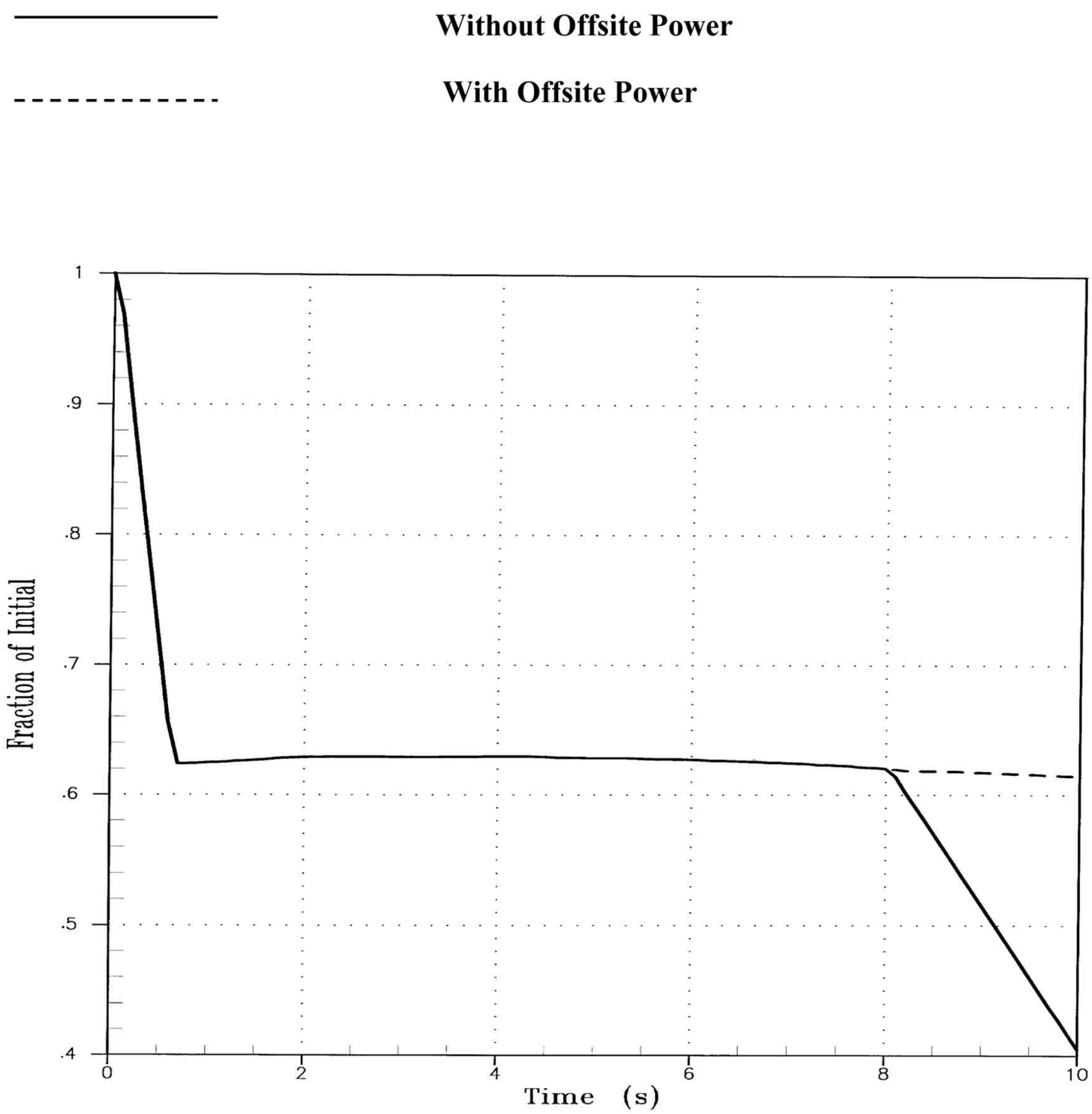


Figure 15.3.3-1
Core Mass Flow Transient for
Four Cold Legs in Operation, One Locked Rotor

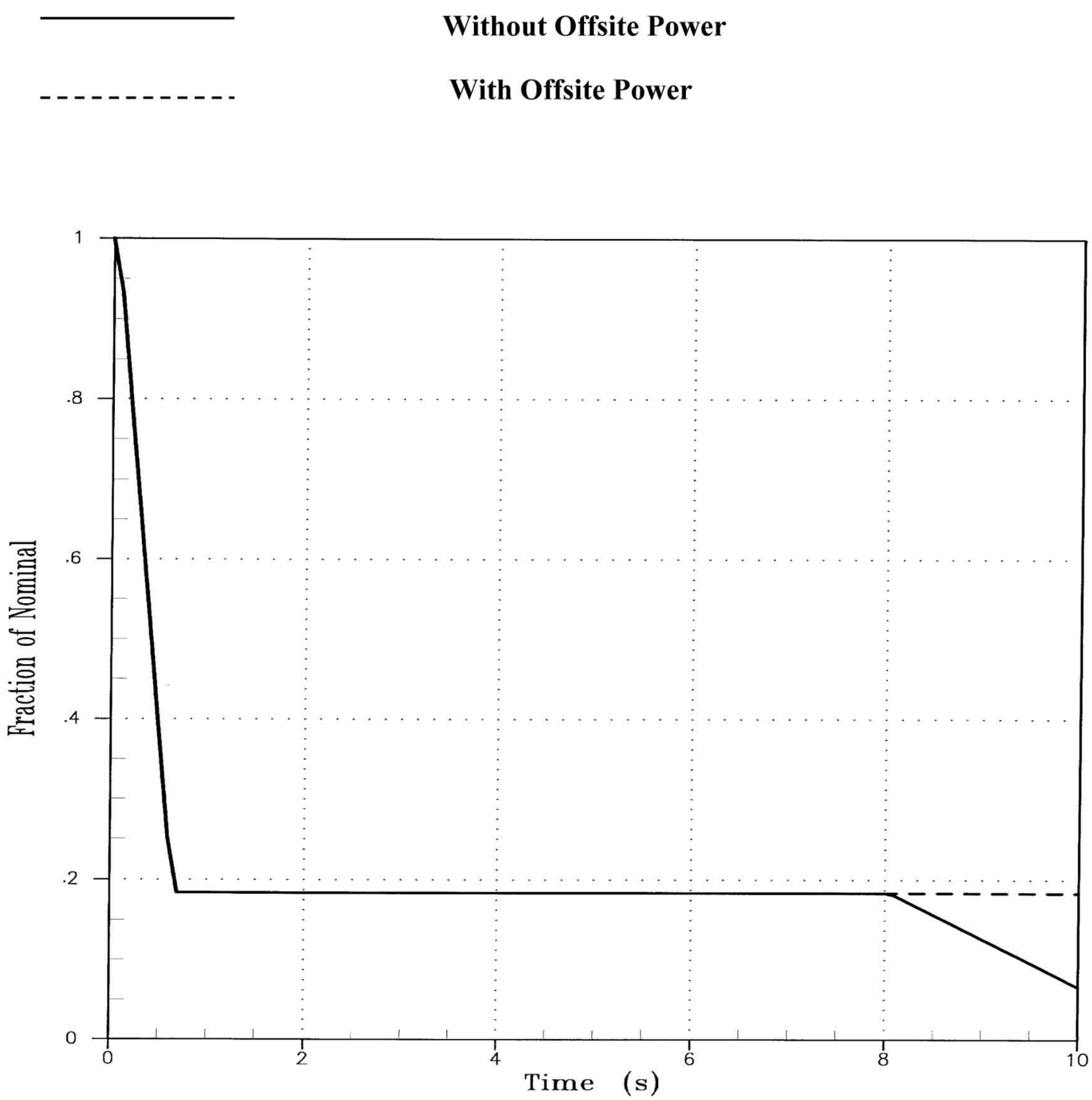


Figure 15.3.3-2
Faulted Loop Volumetric Flow Transient for Four Cold Legs in Operation, One Locked Rotor

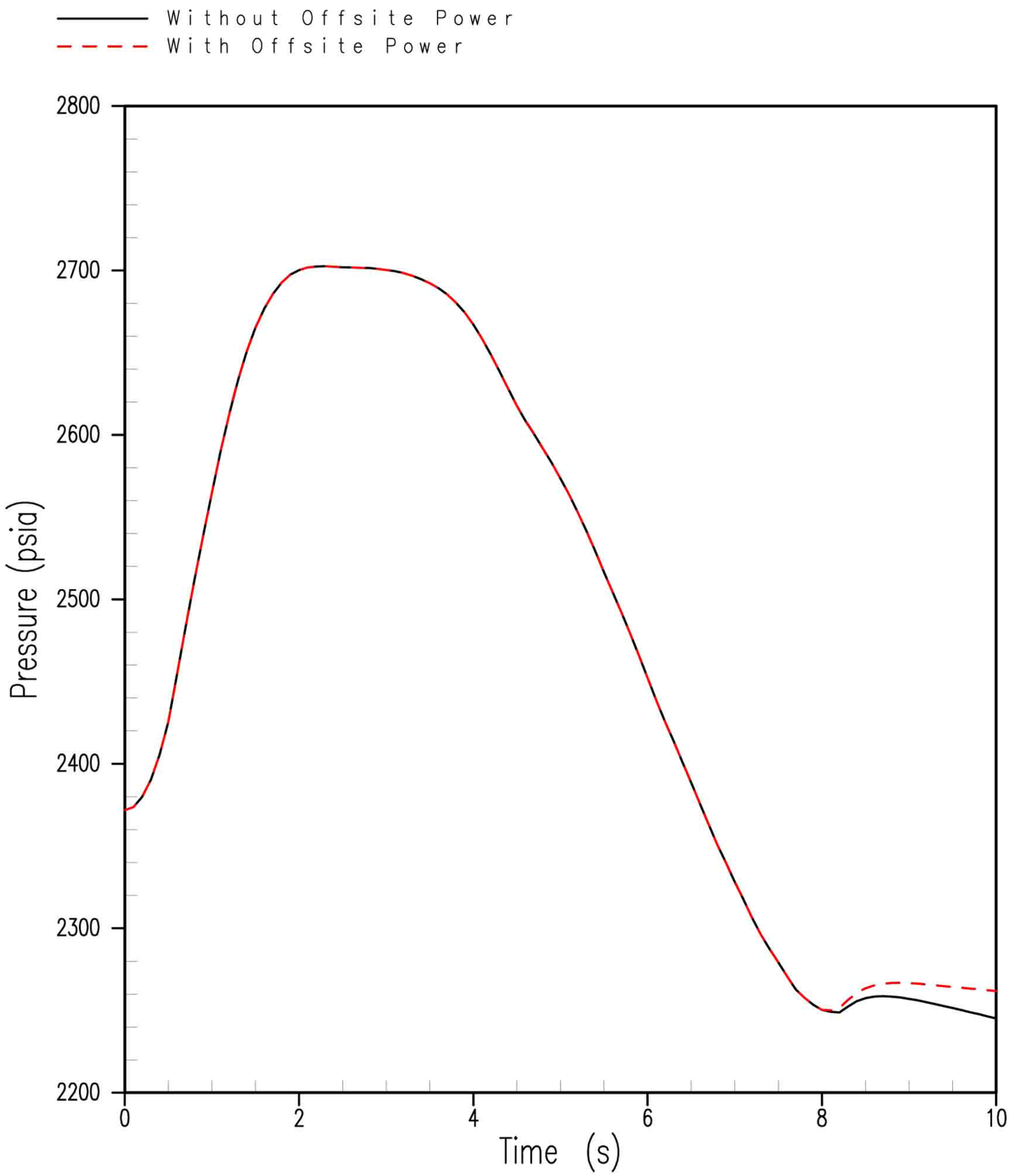


Figure 15.3.3-3
Peak Reactor Coolant Pressure for
Four Cold Legs in Operation, One Locked Rotor

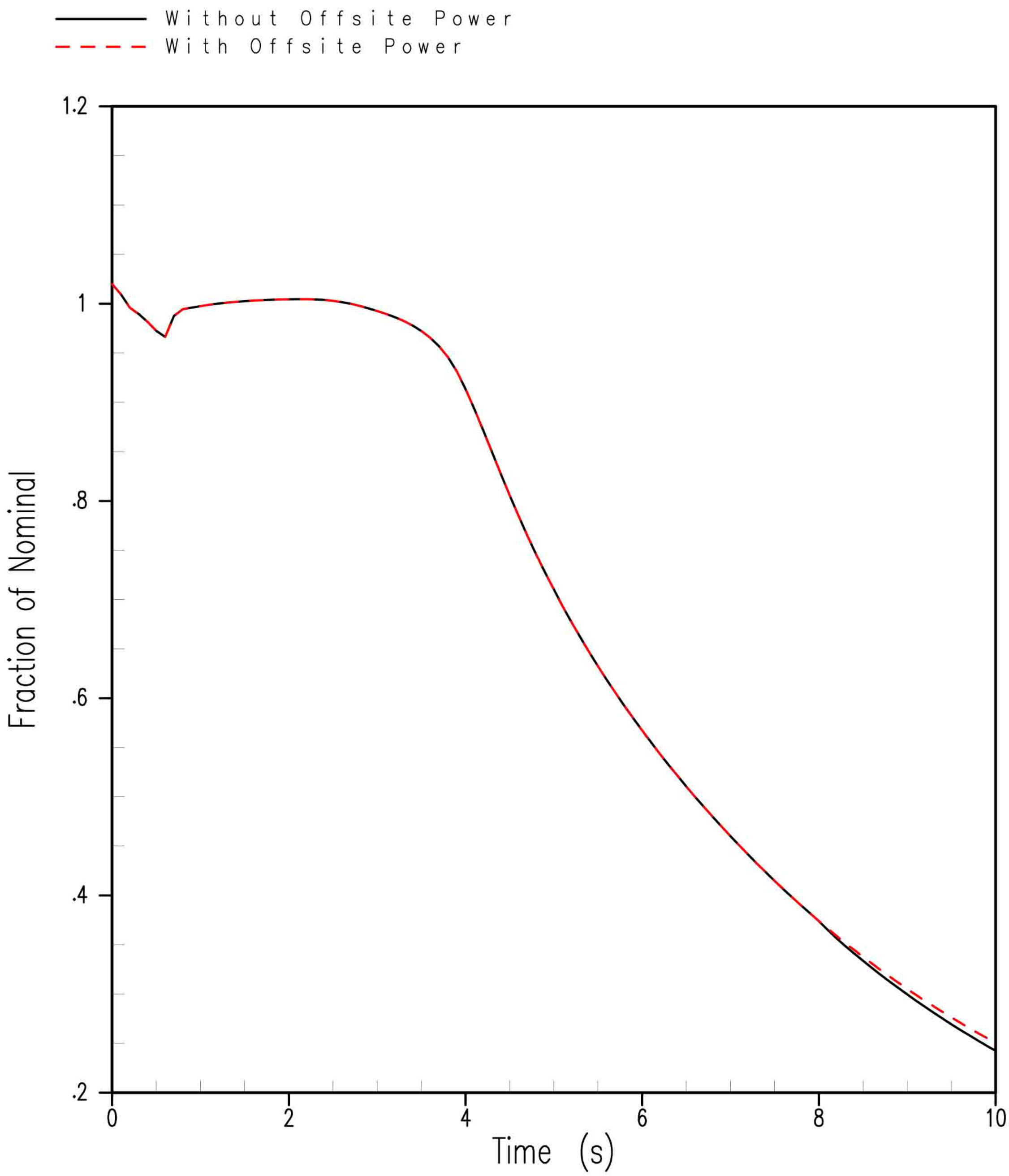


Figure 15.3.3-4
Average Channel Heat Flux Transient for
Four Cold Legs in Operation, One Locked Rotor

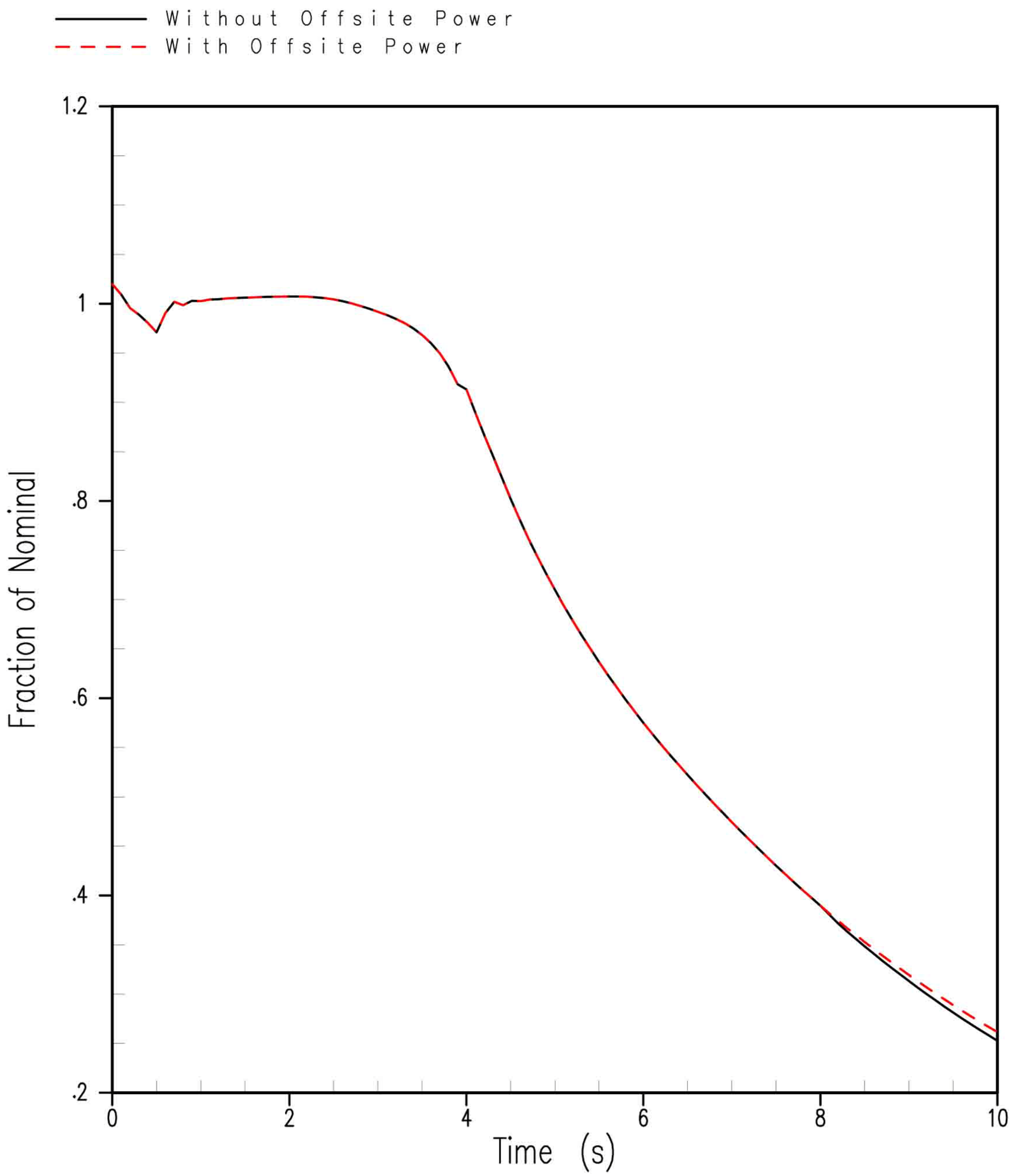


Figure 15.3.3-5
Hot Channel Heat Flux Transient for
Four Cold Legs in Operation, One Locked Rotor

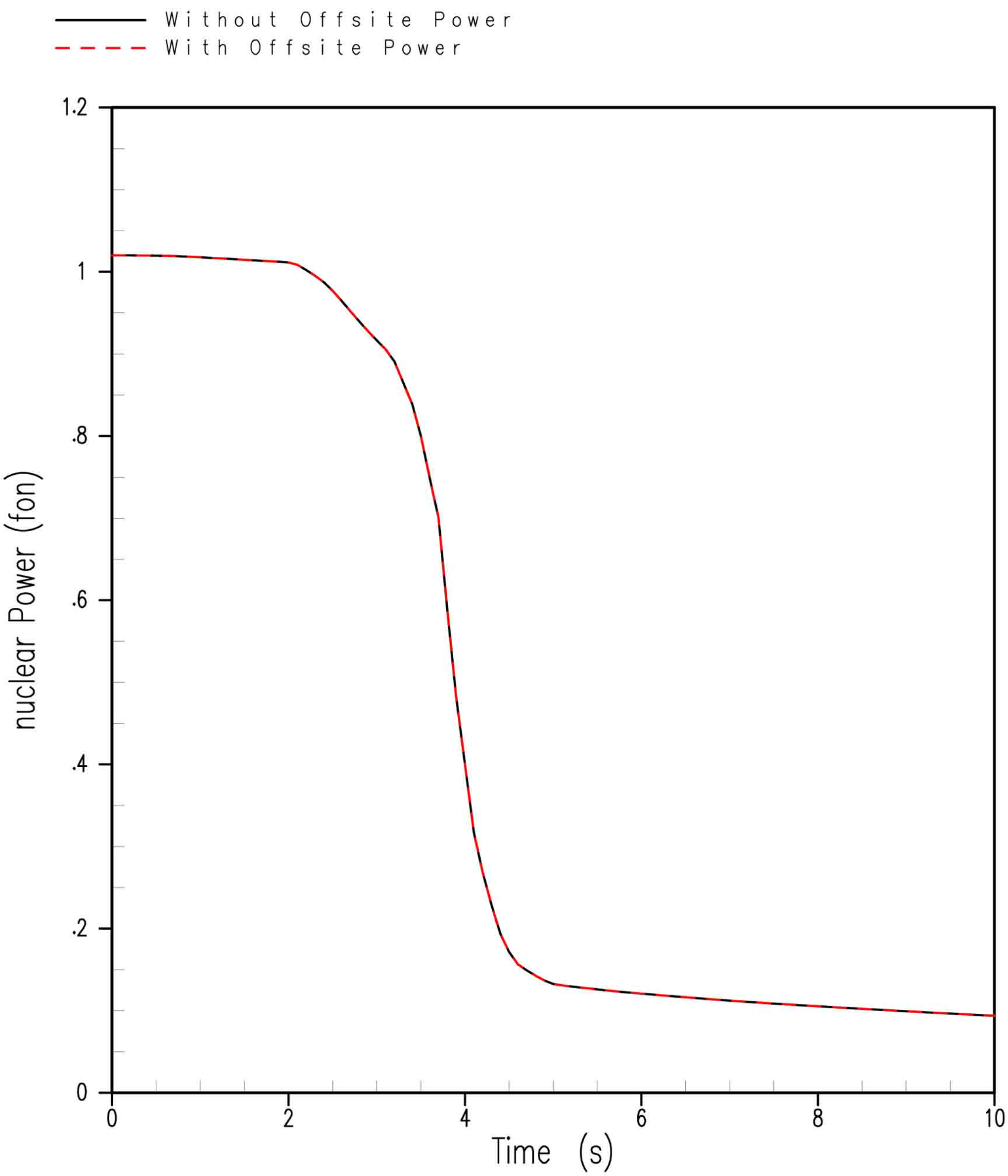


Figure 15.3.3-6
Nuclear Power Transient for
Four Cold Legs in Operation, One Locked Rotor

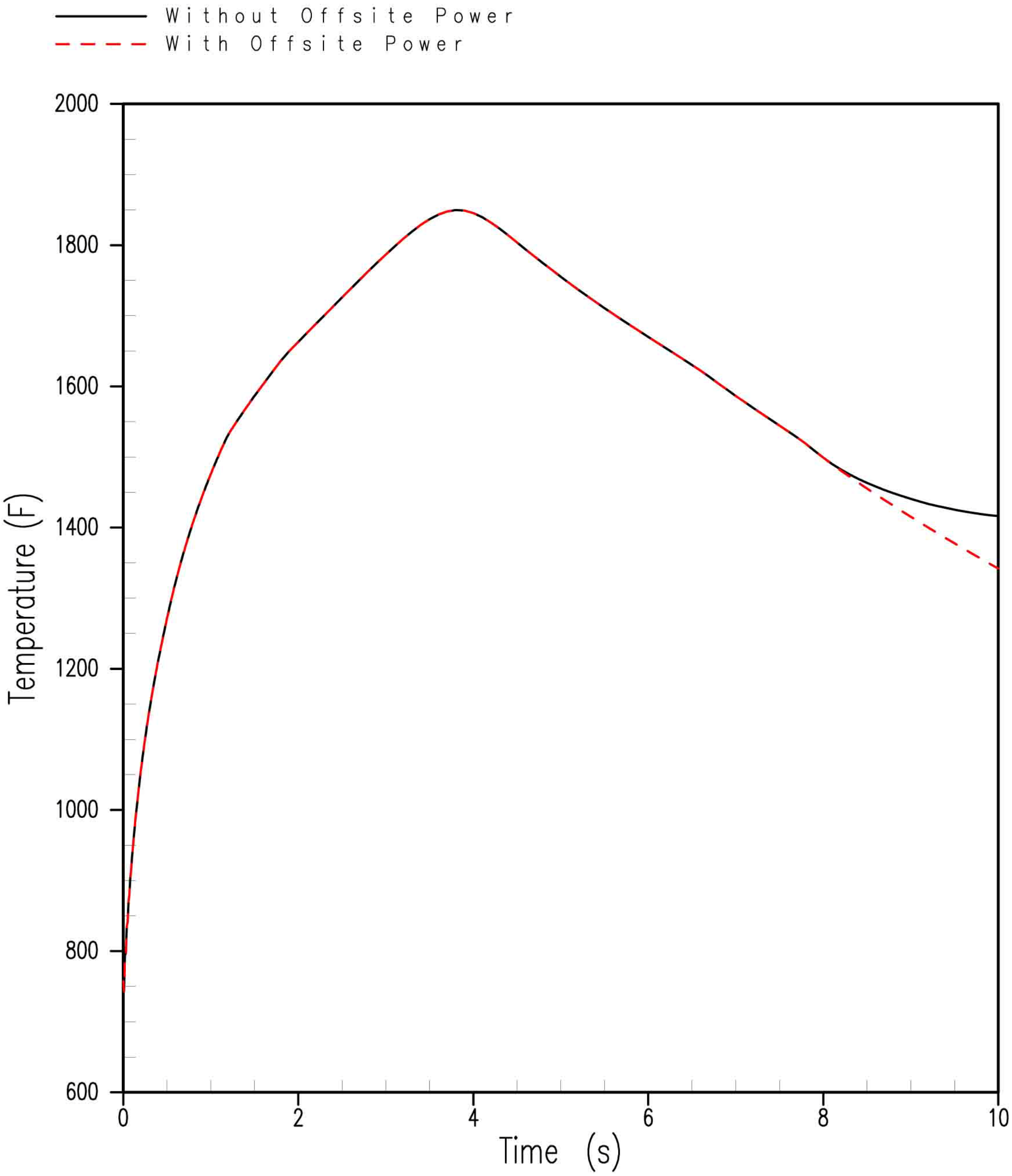


Figure 15.3.3-7
Cladding Inside Temperature Transient for
Four Cold Legs in Operation, One Locked Rotor

15.4 Reactivity and Power Distribution Anomalies

A number of faults are postulated that result in reactivity and power distribution anomalies. Reactivity changes could be caused by control rod motion or ejection, boron concentration changes, or addition of cold water to the reactor coolant system. Power distribution changes could be caused by control rod motion, misalignment, or ejection, or by static means such as fuel assembly mislocation. These events are discussed in this section. Analyses are presented for the most limiting of these events.

The following incidents are discussed in this section:

A. Uncontrolled rod cluster control assembly (RCCA) bank withdrawal from a subcritical or low-power startup condition

B. Uncontrolled RCCA bank withdrawal at power

C. RCCA misalignment

D. Startup of an inactive reactor coolant pump at an incorrect temperature

E. A malfunction or failure of the flow controller in a boiling water reactor recirculation loop that results in an increased reactor coolant flow rate (not applicable to AP1000)

F. Chemical and volume control system malfunction that results in a decrease in the boron concentration in the reactor coolant

G. Inadvertent loading and operation of a fuel assembly in an improper position

H. Spectrum of RCCA ejection accidents

Items A, B, D, and F above are Condition II events, item G is a Condition III event, and item H is a Condition IV event. Item C includes both Conditions II and III events.

The applicable transients in this section have been analyzed. It has been determined that the most severe radiological consequences result from the complete rupture of a control rod drive mechanism housing as discussed in Subsection 15.4.8.

Radiological consequences are reported only for the limiting case.

15.4.1 Uncontrolled Rod Cluster Control Assembly Bank Withdrawal from a Subcritical or Low-Power Startup Condition

15.4.1.1 Identification of Causes and Accident Description

An RCCA withdrawal accident is an uncontrolled addition of reactivity to the reactor core caused by the withdrawal of RCCAs which results in a power excursion. Such a transient can be caused by a malfunction of the reactor control or rod control systems. This can occur with the reactor subcritical, at hot zero power, or at power. The at-power case is discussed in Subsection 15.4.2.

Although the reactor is normally brought to power from a subcritical condition by RCCA withdrawal, initial startup procedures with a clean core use boron dilution. The maximum rate of reactivity increase in the case of boron dilution is less than that assumed in this analysis (see Subsection 15.4.6).

The RCCA drive mechanisms are grouped into preselected bank configurations. These groups prevent the RCCAs from being automatically withdrawn in other than their respective banks. Power supplied to the banks is controlled such that no more than two banks are withdrawn at the same time and in their proper withdrawal sequence. The RCCA drive mechanisms are the magnetic latch type, and coil actuation is sequenced to provide variable speed travel. The maximum reactivity insertion rate analyzed is that occurring with the simultaneous withdrawal of the combination of two sequential RCCA banks having the maximum combined worth at maximum speed.

This event is a Condition II event (a fault of moderate frequency) as defined in Subsection 15.0.1.

The neutron flux response to a continuous reactivity insertion is characterized by a fast rise terminated by the reactivity feedback effect of the negative Doppler coefficient. This self-limitation of the power excursion limits the power during the delay time for protective action. Should a continuous RCCA withdrawal accident occur, the transient is terminated by the following automatic features of the protection and safety monitoring system:

- Source range high neutron flux reactor trip

  This trip function is actuated when two out of four independent source range channels indicate a neutron flux level above a preselected, manually adjustable setpoint. It may be manually bypassed only after an intermediate range flux channel indicates a flux level above a specified level. It is automatically reinstated when the coincident two out of four intermediate range channels indicate a flux level below a specified level.

- Intermediate range high neutron flux reactor trip

  This trip function is actuated when two out of four independent, intermediate range channels indicate a flux level above a preselected, manually adjustable setpoint. It may be manually bypassed only after two out of four power range channels are reading above approximately 10 percent of full power. It is automatically reinstated when the coincident two out of four channels indicate a power level below this value.

- Power range high neutron flux reactor trip (low setting)

  This trip function is actuated when two out of four power range channels indicate a power level above approximately 25 percent of full power. It may be manually bypassed when two out of four power range channels indicate a power level above approximately 10 percent of full power. It is automatically reinstated when the coincident two out of four channels indicate a power level below this value.

- Power range high neutron flux reactor trip (high setting)

  This trip function is actuated when two out of four power range channels indicate a power level above a preset setpoint. It is always active.

- High nuclear flux rate reactor trip

  This trip function is actuated when the positive rate of change of neutron flux on two out of four nuclear power range channels indicate a rate above a preset setpoint.

  In addition, control rod stops on high intermediate range flux level (one out of two) and high power range flux level (one out of four) serve to discontinue rod withdrawal and prevent the need to actuate the intermediate range flux level trip and the power range flux level trip, respectively.

15.4.1.2 Analysis of Effects and Consequences

15.4.1.2.1 Method of Analysis

The analysis of the uncontrolled RCCA bank withdrawal from subcritical accident is performed in three stages: first, an average core nuclear power transient calculation; then, an average core heat transfer calculation; and finally, the departure from nucleate boiling ratio (DNBR) calculation. In the first stage, the average core nuclear calculation is performed using spatial neutron kinetics methods, using the code TWINKLE (Reference 1), to determine the average power generation with time, including the various total core feedback effects (Doppler reactivity and moderator reactivity).

In the second stage, the average heat flux and temperature transients are determined by performing a fuel rod transient heat transfer calculation in FACTRAN (Reference 2). In the final stage, the average heat flux is used in VIPRE-01 (described in Section 4.4) for the transient DNBR calculation.

Plant characteristics and initial conditions are discussed in Subsection 15.0.3. The following assumptions are made to give conservative results for a startup accident:

- Because the magnitude of the power peak reached during the initial part of the transient for any given rate of reactivity insertion is strongly dependent on the Doppler coefficient, conservatively low values, as a function of power, are used (see Table 15.0-2).

- Contribution of the moderator reactivity coefficient is negligible during the initial part of the transient because the heat transfer time between the fuel and the moderator is much longer than the neutron flux response time. After the initial neutron flux peak, the succeeding rate of power increase is affected by the moderator reactivity coefficient. A conservative value is used in the analysis to yield the maximum peak heat flux (see Table 15.0-2).

- The reactor is assumed to be at hot zero power. This assumption is more conservative than that of a lower initial system temperature. The higher initial system temperature yields a larger fuel-water heat transfer coefficient, larger specific heats, and a less negative (smaller absolute magnitude) Doppler coefficient, all of which tend to reduce the Doppler feedback effect and thereby increase the neutron flux peak. The initial effective multiplication factor (k_{eff}) is assumed to be 1.0 because this results in the worst nuclear power transient.

- Reactor trip is assumed to be initiated by the power range high neutron flux (low setting). The most adverse combination of instrument and setpoint errors, as well as delays for trip signal actuation and RCCA release, is taken into account. A 10-percent uncertainty increase is assumed for the power range flux trip setpoint, raising it to 35 percent from the nominal value of 25 percent.

  Because the rise in the neutron flux is so rapid, the effect of errors in the trip setpoint on the actual time at which the rods are released is negligible. In addition, the reactor trip insertion characteristic is based on the assumption that the highest worth RCCA is stuck in its fully withdrawn position. See Subsection 15.0.5 for RCCA insertion characteristics.

- The maximum positive reactivity insertion rate assumed is greater than that for the simultaneous withdrawal of the combination of the two sequential RCCA banks having the greatest combined worth at maximum speed (45 inches per minute). Control rod drive mechanism design is discussed in Section 4.6.

- The most limiting axial and radial power shapes, associated with having the two highest combined worth banks in their high-worth position, are assumed in the departure from nucleate boiling (DNB) analysis.

- The initial power level is assumed to be below the power level expected for any shutdown condition (10^{-9} of nominal power). The combination of highest reactivity insertion rate and lowest initial power produces the highest peak heat flux.
- Four reactor coolant pumps are assumed to be in operation.
- Pressurizer pressure is assumed to be 50 psi below nominal for steady-state fluctuations and measurement uncertainties.

Plant systems and equipment available to mitigate the effects of the accident are discussed in Subsection 15.0.8 and listed in Table 15.0-6. No single active failure in any of these systems or components adversely affects the consequences of the accident. A loss of offsite power as a consequence of a turbine trip disrupting the grid is not considered because the accident is initiated from a subcritical condition where the plant is not providing power to the grid.

15.4.1.2.2 Results

Figures 15.4.1-1 through 15.4.1-3 show the transient behavior for the uncontrolled RCCA bank withdrawal from subcritical incident. The accident is terminated by reactor trip at 35 percent of nominal power. The reactivity insertion rate used is greater than that calculated for the two highest-worth sequential rod cluster control banks, both assumed to be in their highest incremental worth region.

Figure 15.4.1-1 shows the average neutron flux transient. The energy release and the fuel temperature increases are relatively small. The heat flux response (of interest for DNB considerations) is also shown in Figure 15.4.1-2. The beneficial effect of the inherent thermal lag in the fuel is evidenced by a peak heat flux much less than the full-power nominal value. There is margin to DNB during the transient because the rod surface heat flux remains below the critical heat flux value, and there is a high degree of subcooling at all times in the core. Figure 15.4.1-3 shows the response of the average fuel and cladding temperatures. The minimum DNBR at all times remains above the design limit value (see Section 4.4).

The calculated sequence of events for this accident is shown in Table 15.4-1. With the reactor tripped, the plant returns to a stable condition. Subsequently, the plant may be cooled down further by following normal plant shutdown procedures.

15.4.1.3 Conclusions

In the event of an RCCA withdrawal accident from the subcritical condition, the core and the reactor coolant system are not adversely affected because the combination of thermal power and the coolant temperature results in a DNBR greater than the safety analysis limit value. Thus, no fuel or cladding damage is predicted as a result of DNB.

15.4.2 Uncontrolled Rod Cluster Control Assembly Bank Withdrawal at Power

15.4.2.1 Identification of Causes and Accident Description

Uncontrolled RCCA bank withdrawal at power results in an increase in the core heat flux. Because the heat extraction from the steam generator lags behind the core power generation until the steam generator pressure reaches the relief or safety valve setpoint, there is a net increase in the reactor coolant temperature. Unless terminated by manual or automatic action, the power mismatch and resultant coolant temperature rise could eventually result in DNB. Therefore, to avert damage to the fuel cladding, the protection and safety monitoring system (PMS) is designed to terminate any such transient before the DNBR falls below the design limit (see Section 4.4).

This event is a Condition II incident (a fault of moderate frequency) as defined in Subsection 15.0.1.

The automatic features of the PMS that prevent core damage following the postulated accident include the following:

- Power range neutron flux instrumentation actuates a reactor trip if two out of four divisions exceed an overpower setpoint. In particular, the power range neutron flux instrumentation provides the following reactor trip functions:

  1. Reactor trip on high power range neutron flux (high setpoint)

  2. Reactor trip on high power range positive neutron flux rate

  The latter trip protects the core when a sudden abnormal increase in power is detected in the power range neutron flux channel in two out of four PMS divisions. It provides protection against reactivity insertion rates accidents at mid and low power, and it is always active.

- Reactor trip is actuated if any two out of four ΔT power divisions exceed an overtemperature ΔT setpoint. This setpoint is automatically varied with axial power imbalance, coolant temperature, and pressure to protect against DNB.

- Reactor trip is actuated if any two out of four ΔT power divisions exceed an overpower ΔT setpoint. This setpoint is automatically varied with axial power imbalance to prevent the allowable linear heat generation rate (kW/ft) from being exceeded.

- A high pressurizer pressure reactor trip is actuated from any two out of four pressure divisions when a set pressure is exceeded. This set pressure is less than the set pressure for the pressurizer safety valves.

- A high pressurizer water level reactor trip is actuated from any two out of four level divisions that exceed the setpoint when the reactor power is above approximately 10 percent (permissive-P10).

In addition to the preceding reactor trips, there are the following RCCA withdrawal blocks:

- High neutron flux (two out of four power range)
- Overpower ΔT (two out of four)
- Overtemperature ΔT (two out of four)

The manner in which the combination of overpower and overtemperature ΔT trips provide protection over the full range of reactor coolant system conditions is described in Chapter 7 and Reference 13.

Figure 15.0.3-1 presents allowable reactor coolant loop average temperature and ΔT for the design power distribution and flow as a function of primary coolant pressure. The boundaries of operation defined by the overpower ΔT trip and the overtemperature ΔT trip are represented as “protection lines” on this diagram. The protection lines are drawn to include adverse instrumentation and setpoint uncertainties so that under nominal conditions, a trip occurs well within the area bounded by these lines.

The area of permissible operation (power, pressure, and temperature) is bounded by the combination of reactor trips:

- High neutron flux (fixed setpoint)
- High pressurizer pressure (fixed setpoint)

- Low pressurizer pressure (fixed setpoint)
- Overpower and overtemperature ΔT (variable setpoints)

In meeting the requirements of GDC 17 of 10 CFR Part 50, Appendix A, the effects of a possible consequential loss of offsite power during the RCCA bank withdrawal at-power event have been evaluated to not adversely impact the analysis results. This conclusion is based on a review of the time sequence associated with a consequential loss of offsite power in comparison to the reactor shutdown time for an uncontrolled RCCA bank withdrawal at-power event. The primary effect of the loss of offsite power is to cause the reactor coolant pumps (RCPs) to coast down. The PMS includes a 5.0 second minimum delay between the reactor trip and the turbine trip. In addition, a 3.0 second delay between the turbine trip and the loss of offsite power is assumed, consistent with Section 15.1.3 of NUREG-1793. Considering these delays between the time of the reactor trip and RCP coastdown due to the loss of offsite power, it is clear that the plant shutdown sequence will have passed the critical point and the control rods will have been completely inserted before the RCPs begin to coast down. Therefore, the consequential loss of offsite power does not adversely impact this uncontrolled RCCA bank withdrawal at-power analysis because the plant will be shut down well before the RCPs begin to coast down.

15.4.2.2 Analysis of Effects and Consequences

15.4.2.2.1 Method of Analysis

This transient is analyzed using the LOFTRAN (References 3 and 11) code. This code simulates the neutron kinetics, reactor coolant system, pressurizer, pressurizer safety valves, pressurizer spray, steam generators, and steam generator safety valves. The code computes pertinent plant variables including temperatures, pressures, and power level. The core limits as illustrated in Figure 15.0.3-1 are used to define the inputs to LOFTRAN that determine the minimum DNBR during the transient.

Plant characteristics and initial conditions are discussed in Subsection 15.0.3. In performing a conservative analysis for an uncontrolled RCCA bank withdrawal at-power accident, the following assumptions are made:

- The nominal initial conditions are assumed in accordance with the revised thermal design procedure. Uncertainties in the initial conditions are included in the DNBR limit as described in WCAP-11397-P-A (Reference 9).

- Two sets of reactivity coefficients are considered:

  Minimum reactivity feedback — A least-negative moderator temperature coefficient of reactivity is assumed, corresponding to the beginning of core life. A variable Doppler power coefficient with core power is used in the analysis. A conservatively small (in absolute magnitude) value is assumed (see Figure 15.0.4-1).

  Maximum reactivity feedback — A conservatively large positive moderator density coefficient and a large (in absolute magnitude) negative Doppler power coefficient are assumed (see Figure 15.0.4-1).

- The reactor trip on high neutron flux is assumed to be actuated at a conservative value of 118 percent of nominal full power. The ΔT trips include adverse instrumentation and setpoint uncertainties; the delays for trip actuation are assumed to be the maximum values.

- The RCCA trip insertion characteristic is based on the assumption that the highest-worth assembly is stuck in its fully withdrawn position.

- A range of reactivity insertion rates is examined. The maximum positive reactivity insertion rate is greater than that for the simultaneous withdrawal of the combination of the two control banks having the maximum combined worth at maximum speed.

The effect of RCCA movement on the axial core power distribution is accounted for by causing a decrease in overtemperature ΔT trip setpoint proportional to a decrease in margin to DNB.

Plant systems and equipment available to mitigate the effects of the accident are discussed in Subsection 15.0.8 and listed in Table 15.0-6. No single active failure in these systems or equipment adversely affects the consequences of the accident. A discussion of anticipated transients without scram considerations is presented in Section 15.8.

15.4.2.2.2 Results

Figures 15.4.2-1 through 15.4.2-6 show the transient response for a representative rapid RCCA withdrawal incident starting from full power with offsite power lost as a consequence of turbine trip. Reactor trip on high neutron flux occurs shortly after the start of the accident. Because this is rapid with respect to the thermal time constants of the plant, small changes in temperature and pressure result, and the DNB design basis described in Section 4.4 is met.

The transient response for a representative slow RCCA withdrawal from full power, with offsite power lost as a consequence of turbine trip, is shown in Figures 15.4.2-7 through 15.4.2-12. Reactor trip on overtemperature ΔT occurs after a longer period. The rise in temperature and pressure is consequently larger than for rapid RCCA withdrawal. The DNB design basis described in Section 4.4 is met.

Figure 15.4.2-13 shows the minimum DNBR as a function of reactivity insertion rate from initial full-power operation for minimum and maximum reactivity feedback. Minimum DNBR, occurs immediately after rod motion. Two reactor trip functions provide protection over the whole range of reactivity insertion rates. These are the high neutron flux and overtemperature ΔT functions. The DNB design basis described in Section 4.4 is met.

Figures 15.4.2-14 and 15.4.2-15 show the minimum DNBR as a function of reactivity insertion rate for RCCA bank withdrawal incidents for minimum and maximum reactivity feedback, starting at 60-percent and 10-percent power, respectively. Minimum DNBR occurs immediately after rod motion and before the loss of offsite power. The results are similar to the 100-percent power case, except as the initial power is decreased, the range over which the overtemperature ΔT trip is effective is increased and for the maximum feedback cases the transient is always terminated by the overtemperature ΔT reactor trip. The DNB design basis described in Section 4.4 is met.

The shape of the curves of minimum DNBR versus reactivity insertion rate in the referenced figures is due both to reactor core and coolant system transient response and to PMS action in initiating a reactor trip.

Referring to Figure 15.4.2-14, for example, it is noted that for transients initiated from 60-percent power:

A. For high reactivity insertion rates above 14 pcm/s, reactor trip is initiated by the high neutron flux trip for the minimum reactivity feedback cases. Reactor trip is initiated by overtemperature ΔT for the whole range of reactivity insertion rates for the maximum reactivity feedback cases. For minimum reactivity feedback cases, the neutron flux level in the core rises rapidly for the higher reactivity insertion rates while core heat flux and coolant system temperature lag behind due to the thermal capacity of the fuel and coolant system fluid. Thus, the reactor is tripped prior to a significant increase in heat flux or water temperature with resultant high minimum DNBRs during the transient. As reactivity insertion rate decreases, core heat flux and coolant temperatures remain more nearly in equilibrium with the neutron flux. Thus, minimum DNBR during the transient decreases with decreasing insertion rate.

B. The overtemperature ΔT reactor trip circuit initiates a reactor trip when two out of four ΔT power divisions exceed an overtemperature ΔT setpoint. This trip circuit is described in Chapter 7 and Reference 13. The T_{COLD} and T_{HOT} signals, which are inputs to the overtemperature ΔT setpoint calculation, are lead-lag compensated to account for the inherent thermal and transport delays in the reactor coolant system in response to power increases.

C. For reactivity insertion rates less than approximately 40 pcm/s for the minimum feedback cases, the rise in reactor coolant system pressure is sufficiently high that the pressurizer safety valve setpoint is reached prior to reactor trip. Opening of this valve limits the rise in reactor coolant pressure as the temperature continues to rise. Because the overtemperature ΔT reactor trip setpoint is based on both temperature and pressure, limiting the reactor coolant pressure by opening the pressurizer safety valve brings about the overtemperature ΔT earlier than if the valve remains closed. For this reason, the overtemperature ΔT setpoint initiates reactor trip at reactivity insertion rates of approximately 14 pcm/s and below for the minimum feedback cases. For the maximum feedback case, the pressurizer safety valves open prior to reactor trip for reactivity insertion rates as high as 110 pcm/s.

D. For the minimum feedback case, at reactivity insertion rates less than approximately 14 pcm/s the overtemperature ΔT trip predominates and the effectiveness of the overtemperature ΔT trip increases (in terms of increased minimum DNB) because for these lower reactivity insertion rates, the power increase is slower, the rate of rise of average coolant temperature is slower, and the system lags and delays become less significant.

E. For reactivity insertion rates less than approximately 3 pcm/s for the minimum feedback cases and less than approximately 70 pcm/s for maximum feedback cases, the rise in the reactor coolant temperature is sufficiently high so that the steam generator safety valve setpoint is reached prior to trip. Opening of these valves, which act as an additional heat load on the reactor coolant system, sharply decreases the rate of increase of reactor coolant system average temperature. This decrease in the rate of increase of the average coolant system temperature during the transient is accentuated by the lead-lag compensation. This causes the overtemperature ΔT setpoint to be reached later, with resulting lower minimum DNBRs.

As described in item D above, at lower reactivity insertion rates the overtemperature ΔT trip predominates and the effectiveness of the overtemperature ΔT trip increases (in terms of increased minimum DNBR) because for these lower reactivity insertion rates, the power increase is slower, the rate of rise of average coolant temperature is slower, and the system lags and delays become less significant.

Steam generator safety valves never open before the reactor trip for transients initiated at full power. So there are not the competing effects due to the opening of the pressurizer safety valve and steam generator safety valves described in items C and E. Hence, for both the minimum and maximum feedback cases, the local minimum in the DNBR curve due to the steam generator safety valves opening is not present.

Figures 15.4.2-13, 15.4.2-14, and 15.4.2-15 illustrate minimum DNBR calculated for minimum and maximum reactivity feedback.

Because the RCCA bank withdrawal at-power incident is an overpower transient, the fuel temperatures rise during the transient until after reactor trip occurs. For high reactivity insertion rates, the overpower transient is fast with respect to the fuel rod thermal time constant and the core heat flux lags behind the neutron flux response. Taking into account the effect of the RCCA withdrawal on the axial core power distribution, the peak fuel temperature still remains below the fuel melting temperature.

For slow reactivity insertion rates, the core heat flux remains more nearly in equilibrium with the neutron flux. The overpower transient is terminated by the overtemperature ΔT reactor trip before DNB occurs. Taking into account the effect of the RCCA withdrawal on the axial core power distribution, the peak centerline temperature remains below the fuel melting temperature.

The reactor is tripped fast enough during the RCCA bank withdrawal at-power transient that the ability of the primary coolant to remove heat from the fuel rods is not reduced. Thus, the fuel cladding temperature does not rise significantly above its initial value during the transient.

The calculated sequence of events for this accident, with offsite power available, is shown in Table 15.4-1. With the reactor tripped, the plant returns to a stable condition. The plant may be cooled down further by following normal plant shutdown procedures.

As discussed previously in Subsection 15.4.2.1, even if a consequential loss of offsite power and the subsequent RCP coastdown were to be explicitly modeled, the minimum DNBR would be predicted to occur during the time period of the RCCA bank withdrawal at-power event prior to the time the flow coastdown begins. Therefore, the minimum DNBRs calculated in the analysis are bounding.

15.4.2.3 Conclusions

The power range neutron flux instrumentation and overtemperature ΔT trip functions provide adequate protection over the entire range of possible reactivity insertion rates. The DNB design basis, as defined in Section 4.4, is met for all cases.

15.4.3 Rod Cluster Control Assembly Misalignment (System Malfunction or Operator Error)

15.4.3.1 Identification of Causes and Accident Description

RCCA misoperation accidents include:

- One or more dropped RCCAs within the same group
- Statically misaligned RCCA
- Withdrawal of a single RCCA

Each RCCA has a position indicator channel which displays the position of the assembly. The displays of assembly positions are grouped for the operator’s convenience. Fully inserted assemblies are further indicated by a rod-at-bottom signal, which actuates a local alarm and a main control room annunciator. Group demand position is also indicated.

RCCAs are moved in preselected banks, and the banks are moved in a preselected sequence. Each bank of RCCAs is divided into one or two groups of four or five RCCAs each. The rods comprising a group operate in parallel. The two groups in a bank move sequentially such that the first group is always within one step of the second group in the bank. A definite schedule of actuation (or deactuation) of the stationary gripper, movable gripper, and lift coils of a mechanism is required to withdraw the RCCA attached to the mechanism. Because the stationary gripper, movable gripper, and lift coils associated with the RCCAs of a rod group are driven in parallel, any single failure which causes rod withdrawal affects the entire group. A single electrical or mechanical failure in the plant control system could, at most, result in dropping one or more RCCAs within the same group. Mechanical failures can cause either RCCA insertion or immobility, but not RCCA withdrawal.

The dropped RCCAs, dropped RCCA bank, and statically misaligned RCCA events are Condition II incidents (incidents of moderate frequency) as defined in Subsection 15.0.1. The single RCCA withdrawal event is a Condition III incident, as discussed below.

No single electrical or mechanical failure in the rod control system could cause the accidental withdrawal of a single RCCA from the inserted bank at full-power operation. The operator could withdraw a single RCCA in the control bank because this feature is necessary to retrieve an assembly should one be accidentally dropped. The event analyzed results from multiple wiring failures or multiple significant operator errors and subsequent and repeated operator disregard of event indication. The probability of such a combination of conditions is considered low such that the limiting consequences may include slight fuel damage.

The event is classified as a Condition III incident consistent with the philosophy and format of American National Standards Institute, ANSI N18.2. By definition, “Condition III occurrences include incidents, any one of which may occur during the lifetime of a particular plant,” and “shall not cause more than a small fraction of fuel elements in the reactor to be damaged . . .” (Reference 10).

This selection of criterion is in accordance with General Design Criterion 25, which states, “The protection system shall be designed to assure that specified acceptable fuel design limits are not exceeded for any <u>single</u> malfunction of the reactivity control systems, such as accidental withdrawal (not ejection or dropout) of control <u>rods</u>.” (Emphases have been added.) It has been shown that single failures resulting in RCCA bank withdrawals do not violate specified fuel design limits. Moreover, no single malfunction can result in the withdrawal of a single RCCA. Thus, it is concluded that criterion established for the single rod withdrawal at power is appropriate and in accordance with General Design Criterion 25.

A dropped RCCA or RCCA bank may be detected by one or more of the following:

- Sudden drop in the core power level as seen by the nuclear instrumentation system
- Asymmetric power distribution as seen by the incore or excore neutron detectors or core exit thermocouples, through online core monitoring
- Rod at bottom signal
- Rod deviation alarm
- Rod position indication

Misaligned RCCAs are detected by one or more of the following:

- Asymmetric power distribution as seen by the incore or excore neutron detectors or core exit thermocouples, through online core monitoring
- Rod deviation alarm
- Rod position indicators

The resolution of the rod position indicator channel is $\pm$5 percent span ($\pm$7.5 inches). A deviation of any RCCA from its group by twice this distance (10 percent of span or 15 inches) does not cause power distributions worse than the design limits. The deviation alarm alerts the operator to rod deviation with respect to the group position in excess of 5 percent of span.

If one or more of the rod position indicator channels is out of service, operating instructions are followed to verify the alignment of the nonindicated RCCAs. The operator also takes action as required by the Technical Specifications.

In the extremely unlikely event of multiple electrical failures that result in single RCCA withdrawal, rod deviation and rod control urgent failure are both displayed to the operator, and the rod position indicators indicate the relative positions of the assemblies in the bank. The urgent failure alarm also inhibits automatic rod motion in the group in which it occurs. Withdrawal of a single RCCA by operator action, whether deliberate or by a combination of errors, results in activation of the same alarm and the same visual indication. Withdrawal of a single RCCA results in both positive reactivity insertion tending to increase core power and an increase in local power density in the core area associated with the RCCA. Automatic protection for this event is provided by the overtemperature ΔT reactor trip. The Condition III Standard Review Plan Section 15.4.3 evaluation criteria are met; however, due to the increase in local power density, the limits in Figure 15.0.3-1 may be exceeded.

Plant systems and equipment available to mitigate the effects of the various control rod misoperations are discussed in Subsection 15.0.8 and listed in Table 15.0-6. No single active failure in any of these systems or equipment adversely affects the consequences of the accident.

15.4.3.2 Analysis of Effects and Consequences

15.4.3.2.1 Dropped RCCAs, Dropped RCCA Bank, and Statically Misaligned RCCA

15.4.3.2.1.1 Method of Analysis

- One or more dropped RCCAs from the same group

  A drop of one or more RCCAs from the same group results in an initial reduction in the core power and a perturbation in the core radial power distribution. Depending on the worth and position of the dropped rods, this may cause the allowable design power peaking factors to be exceeded. Following the drop, the reduced core power and continued steam demand to the turbine causes the reactor coolant temperature to decrease. In the manual control mode, the plant will establish a new equilibrium condition. The new equilibrium condition is reached through reactivity feedback. In the presence of a negative moderator temperature coefficient, the reactor power rises monotonically back to the initial power level at a reduced inlet temperature with no power overshoot. The absence of any power overshoot establishes the automatic operating mode as a limiting case. If the reactor coolant system temperature reduction is very large, the turbine power may not be able to be maintained due to the reduction in the secondary-side steam pressure and the volumetric flow limit of the turbine system. In this case, the equilibrium power level is less than the initial power. In the automatic control mode, the plant control system detects the drop in core power and initiates withdrawal of a control bank. Power overshoot may occur, after which the control system will insert the control bank and return the plant to the initial power level. The magnitude of the power overshoot is a function of the plant control system characteristics, core reactivity coefficients, the dropped rod worth, and the available control bank worth.

  For evaluation of the dropped RCCA event, the transient system response is calculated using the LOFTRAN code (References 3 and 11). The code simulates the neutron kinetics, reactor coolant system, pressurizer, pressurizer safety valves, pressurizer spray, steam generator and steam generator safety valves. The code computes pertinent plant variables, including temperatures, pressures and power level.

Steady-state nuclear models using the computer codes described in Table 4.1-2 are used to obtain a hot channel factor consistent with the primary system transient conditions and reactor power. By combining the transient primary conditions with the hot channel factor from the nuclear analysis, the departure from nucleate boiling design basis is shown to be met using the VIPRE-01 code.

- Statically misaligned RCCA

  Steady-state power distributions are analyzed using the computer codes as described in Table 4.1-2. The peaking factors are then used as input to the VIPRE-01 code to calculate the DNBR.

15.4.3.2.1.2 Results

- One or more dropped RCCAs

  Figures 15.4.3-1 through 15.4.3-4 show the transient response of the reactor to a dropped rod (or rods) in automatic control. The nuclear power and heat flux drop to a minimum value and recover under the influence of both rod withdrawal and thermal feedback. The prompt decrease in power is governed by the dropped rod worth because the plant control system does not respond during the short rod drop time period. The plant control system detects the reduction in core power and initiates control bank withdrawal to restore the primary side power. Power overshoot occurs after which the core power is restored to the initial power level.

  The primary system conditions are combined with the hot channel factors from the nuclear analysis for the DNB evaluation. Uncertainties in the initial conditions are included in the DNB evaluation as discussed in Subsection 15.0.3.2. The calculated minimum DNBR for the limiting case for any single or multiple rod drop from the same group is greater than the design limit value described in Section 4.4. The sequence of events for a representative case is shown in Table 15.4-1.

  The analysis described previously includes consideration of drops of the RCCA groups which can be selected for insertion as part of the rapid power reduction system. This system is provided to allow the reactor to ride out a complete loss of load from full power without a reactor trip and is described in Subsection 7.7.1.10. If these RCCAs are inadvertently dropped (in the absence of a loss-of-load signal), the transient behavior is the same as for the RCCA drop described. The evaluation showed that the DNBR remains above the design limit value as a result of the inadvertent actuation of the rapid power reduction system.

  The consequential loss of offsite power described in Subsection 15.0.14 is not limiting for the dropped RCCA event. Due to the delay from reactor trip until turbine trip and the rapid power reduction produced by the reactor trip, the minimum DNBR occurs before the reactor coolant pumps begin to coast down.

- Statically misaligned RCCA

  The most severe misalignment situations with respect to DNBR arise from cases in which one RCCA is fully inserted, or where the mechanical shim or axial offset rod banks are inserted up to their insertion limit with one RCCA fully withdrawn while the reactor is at full power. Multiple independent alarms, including a bank insertion limit or rod deviation alarm, alert the operator well before the postulated conditions are approached.

  For RCCA misalignments in which the mechanical shim or axial offset banks are inserted to their respective insertion limits, with any one RCCA fully withdrawn, the DNBR remains above the safety analysis limit value. This case is analyzed assuming the initial reactor power, pressure, and reactor coolant system temperature are at their nominal values, but with the increased radial peaking factor associated with the misaligned RCCA. Uncertainties in the initial conditions are included in the DNB evaluation as described in Subsection 15.0.3.2.

DNB does not occur for the RCCA misalignment incident, and thus the ability of the primary coolant to remove heat from the fuel rod is not reduced. The peak fuel temperature is that corresponding to a linear heat generation rate based on the radial peaking factor penalty associated with the misaligned RCCA and the design axial power distribution. The resulting linear heat generation is well below that which causes fuel melting.

Following the identification of an RCCA group misalignment condition by the operator, the operator takes action as required by the plant Technical Specifications and operating instructions.

15.4.3.2.2 Single Rod Cluster Control Assembly Withdrawal

15.4.3.2.2.1 Method of Analysis

Power distributions within the core are calculated using the computer codes described in Table 4.1-2. The peaking factors are then used by VIPRE-01 to calculate the DNBR for the event. The case of the worst rod withdrawn from the mechanical shim or axial offset bank inserted at the insertion limit, with the reactor initially at full power, is analyzed. This incident is assumed to occur at beginning of life because this results in the minimum value of moderator temperature coefficient. This assumption maximizes the power rise and minimizes the tendency of increased moderator temperature to flatten the power distribution.

15.4.3.2.2.2 Results

For the single rod withdrawal event, two cases are considered as follows:

A. If the reactor is in the manual control mode, continuous withdrawal of a single RCCA results in both an increase in core power and coolant temperature and an increase in the local hot channel factor in the area of the withdrawing RCCA. In the overall system response, this case is similar to those presented in Subsection 15.4.2. The increased local power peaking in the area of the withdrawn RCCA results in lower minimum DNBRs than for the withdrawn bank cases. Depending on initial bank insertion and location of the withdrawn RCCA, automatic reactor trip may not occur sufficiently fast to prevent the minimum DNBR from falling below the safety analysis limit value. Evaluation of this case at the power and coolant conditions at which the overtemperature ΔT trip is expected to trip the plant shows that an upper limit for the number of rods with a DNBR less than the safety analysis limit value is 5 percent.

B. If the reactor is in the automatic control mode, the multiple failures that result in the withdrawal of a single RCCA result in the immobility of the other RCCAs in the controlling bank. The transient then proceeds in the same manner as case A.

   For such cases, a reactor trip ultimately occurs although not sufficiently fast in all cases to prevent a minimum DNBR in the core of less than the safety analysis limit value. Following reactor trip, normal shutdown procedures are followed.

   The consequential loss of offsite power described in Subsection 15.0.14 is not limiting for the single RCCA withdrawal event. Due to the delay from reactor trip until turbine trip and the rapid power reduction produced by the reactor trip, the minimum DNBR, for rods where the DNBR did not fall below the design limit value (see Section 4.4) in the cases described, occurs before the reactor coolant pumps begin to coast down.

15.4.3.3 Conclusions

For cases of dropped RCCAs or dropped banks, including inadvertent drops of the RCCAs in those groups selected to be inserted as part of the rapid power reduction system, it is shown that the DNBR remains greater than the safety analysis limit value and, therefore, the DNB design basis is met.

For cases of any one RCCA fully inserted, or the mechanical shim or axial offset banks inserted to their rod insertion limits with any single RCCA in one of those banks fully withdrawn (static misalignment), the DNBR remains greater than the safety analysis limit value (see Section 4.4).

For the case of the accidental withdrawal of a single RCCA, with the reactor in the automatic or manual control mode and initially operating at full power with the mechanical shim or axial offset banks at their insertion limits, an upper bound of the number of fuel rods experiencing DNB is 5 percent of the total fuel rods in the core.

15.4.4 Startup of an Inactive Reactor Coolant Pump at an Incorrect Temperature

The Technical Specifications (3.4.4) require all RCPs to be operating while in Modes 1 and 2. The maximum initial core power level for the startup of an inactive loop transient is approximately zero MWt. Furthermore, the reactor will initially be subcritical by the Technical Specification requirement. There will be no increase in core power, and no automatic or manual protective action is required.

15.4.5 A Malfunction or Failure of the Flow Controller in a Boiling Water Reactor Loop that Results in an Increased Reactor Coolant Flow Rate

This subsection is not applicable to the AP1000.

15.4.6 Chemical and Volume Control System Malfunction that Results in a Decrease in the Boron Concentration in the Reactor Coolant

15.4.6.1 Identification of Causes and Accident Description

Other than control rod withdrawal, the principal means of positive reactivity insertion to the core is the addition of unborated, primary-grade water from the demineralized water transfer and storage system into the reactor coolant system through the reactor makeup portion of the chemical and volume control system. Normal boron dilution with these systems is manually initiated under strict administrative controls requiring close operator surveillance. Procedures limit the rate and duration of the dilution. A boric acid blend system is available to allow the operator to match the makeup water boron concentration to that of the reactor coolant system during normal charging.

An inadvertent boron dilution is caused by the failure of the demineralized water transfer and storage system or chemical and volume control system, either by controller, operator or mechanical failure. The chemical and volume control system and demineralized water transfer and storage system are designed to limit, even under various postulated failure modes, the potential rate of dilution to values that, with indication by alarms and instrumentation, allowing sufficient time for automatic or operator response to terminate the dilution.

An inadvertent dilution from the demineralized water transfer and storage system through the chemical and volume control system may be terminated by isolating the makeup flow to the reactor coolant system, by isolating the makeup pump suction line to the demineralized water transfer and storage system storage tank, or by tripping the makeup pumps. Lost shutdown margin may be regained by adding borated water (greater than 4000 ppm) to the reactor coolant system from the boric acid tank.

Generally, to dilute, the operator performs two actions:

- Switch control of the makeup from the automatic makeup mode to the dilute mode.
- Start the chemical and volume control system makeup pumps.

Failure to carry out either of those actions prevents initiation of dilution. Because the AP1000 chemical and volume control system makeup pumps do not run continuously (they are expected to be operated once per day to make up for reactor coolant system leakage), a makeup pump is started when the volume control system is placed into dilute mode.

The status of the reactor coolant system makeup is available to the operator by the following:

- Indication of the boric acid and blended flow rates
- Chemical and volume control system makeup pumps status
- Deviation alarms, if the boric acid or blended flow rates deviate by more than the specified tolerance from the preset values
- When reactor is subcritical
  - High flux at shutdown alarm
  - Indicated source range neutron flux count rates
  - Audible source range neutron flux count rate
  - Source range neutron flux-multiplication alarm
- When the reactor is critical
  - Axial flux difference alarm (reactor power ≥ 50 percent rated thermal power)
  - Control rod insertion limit low and low-low alarms
  - Overtemperature ΔT alarm (at power)
  - Overtemperature ΔT reactor trip
  - Power range neutron flux-high, both high and low setpoint reactor trips.

This event is a Condition II incident (a fault of moderate frequency), as defined in Subsection 15.0.1.

15.4.6.2 Analysis of Effects and Consequences

Boron dilutions during refueling, cold shutdown, hot shutdown, hot standby, startup, and power modes of operation are considered in this analysis. Conservative values for necessary parameters are used (high reactor coolant system critical boron concentrations, high boron worths, minimum shutdown margins, and lower-than-actual reactor coolant system volumes). These assumptions (see Table 15.4-2) result in conservative determinations of the time available for operator or automatic system response after detection of a dilution transient in progress.

In meeting the requirements of GDC 17 of 10 CFR Part 50, Appendix A, a loss of offsite power is considered for the boron dilution case initiated from the power mode of operation (Mode 1) with the reactor in manual control. This is the analyzed Mode 1 boron dilution case that produces a reactor and turbine trip (Subsection 15.4.6.2.6). The loss of offsite power is assumed to occur as a direct result of a turbine trip that would disrupt the grid and produce a consequential loss of offsite ac power. As discussed in Subsection 15.0.14, that scenario can occur only with the plant at power and connected to the grid. Therefore, only a boron dilution case initiated from full power will address the consequential loss of offsite power.

15.4.6.2.1 Dilution During Refueling (Mode 6)

An uncontrolled boron dilution transient cannot occur during this mode of operation. Inadvertent dilution is prevented by administrative controls, which isolate the reactor coolant system from the potential source of unborated water by locking closed specified valves in the chemical and volume control system during refueling operations. These valves block the flow paths that allow unborated makeup water to reach the reactor coolant system. Makeup which is required during refueling uses water supplied from the boric acid tank (which contains borated water).

15.4.6.2.2 Dilution During Cold Shutdown (Mode 5)

The following conditions are assumed for inadvertent boron dilution while in this operating mode:

- A dilution flow of 175 gpm of unborated water exists.
- A volume of 2592.2 ft^3 is a conservative estimate of the minimum active reactor coolant system volume corresponding to the water level at mid-loop in the vessel while on normal residual heat removal. The assumed active volume does not include the volume of the reactor vessel upper head region.
- Control rods are fully inserted, which is the normal condition in cold shutdown and a critical boron concentration of 1483 ppm. This is a conservative boron concentration with control rods inserted and allows for the most reactive rod to be stuck in the fully withdrawn position.
- The shutdown margin is equal to 1.6-percent $\Delta k/k$, the minimum value identified by the core operating limit report (COLR) for the cold shutdown mode. Combined with the preceding, this gives a shutdown boron concentration of 1675 ppm.
- At least one reactor coolant pump will be normally operating during plant operation in Mode 5. It may be possible under some conditions, however, to operate the plant in Mode 5 with no reactor coolant pumps operating. For this reason, the mixing volume assumed for the analysis in Mode 5 will include the reactor coolant loop and normal residual heat removal system volumes that are being actively mixed by the residual heat removal system pumps.

In the event of an inadvertent boron dilution transient during cold shutdown, the source range nuclear instrumentation detects an increase in the neutron flux by comparing the current source range flux to that of about 50 minutes earlier. Upon detecting a sufficiently large flux increase, an alarm is sounded for the operator, and valves are actuated to terminate the dilution automatically.

Upon the actuation of a source range flux doubling signal, the makeup flow to the reactor coolant system and the makeup pump suction line to the demineralized water transfer and storage system storage tank are isolated. This thereby terminates the dilution. In addition, the makeup pumps are tripped for equipment protection only. This function is not credited in the safety analysis.

The automatic protective actions initiate about 11 minutes after the start of dilution. These automatic actions minimize the approach to criticality and maintain the plant in a subcritical condition. After the automatic protection functions take place, the operator may take action to restore the Technical Specification shutdown margin.

15.4.6.2.3 Dilution During Safe Shutdown (Mode 4)

The following conditions are assumed for an inadvertent boron dilution while in this mode:

- A dilution flow of 175 gpm of unborated water exists.

- Reactor coolant system water volume is 7539.8 ft^3. This is a conservative estimate of the minimum active volume of the reactor coolant system while on normal residual heat removal.

- All control rods are fully inserted, except the most reactive rod which is assumed stuck in the fully withdrawn position, and a conservative critical boron concentration of 1449 ppm.

- The shutdown margin is equal to 1.6-percent k/k, the minimum value required by the core operating limit report (COLR) for the hot shutdown mode. This gives a shutdown boron concentration of 1649 ppm.

- The reactor coolant system dilution volume is considered well-mixed. The Technical Specifications require that when in Mode 4, at least one reactor coolant pump shall be operable, which provides sufficient flow through the system to maintain the system well-mixed. If a reactor coolant pump is not operating, the demineralized water isolation valves are closed and an uncontrolled boron dilution transient cannot occur, as discussed in Subsection 15.4.6.2.1.

In the event of an inadvertent boron dilution transient during safe shutdown, the source range nuclear instrumentation detects a sufficiently large increase in the neutron flux, automatically initiates valve movement to terminate the dilution, and sounds an alarm.

Upon the actuation of a source range flux doubling signal, the makeup flow to the reactor coolant system and the makeup pump suction line to the demineralized water transfer and storage system storage tank are isolated. This thereby terminates the dilution. In addition, the makeup pumps are tripped for equipment protection only. This function is not credited in the safety analysis.

The protective actions initiate about 28 minutes after the start of the dilution. No operator action is required to terminate this transient.

15.4.6.2.4 Dilution During Hot Standby (Mode 3)

The following conditions are assumed for an inadvertent boron dilution while in this mode:

- A dilution flow of 175 gpm of unborated water exists.

- The reactor coolant system volume is 7539.8 ft^3. This is a conservative estimate of the minimum active volume of the reactor coolant system with the reactor coolant system filled and vented and one reactor coolant pump running.

- Critical boron concentration is 1281 ppm. This is a conservative boron concentration assuming control rods are fully inserted minus the most reactive rod, which is assumed stuck in the fully withdrawn position.

- The shutdown margin is equal to 1.6-percent k/k, the minimum value required by the core operating limit report (COLR) for the hot standby mode. This gives a shutdown boron concentration of 1509 ppm.

- The reactor coolant system dilution volume is considered well-mixed. The Technical Specifications require that when in Mode 3, at least one reactor coolant pump shall be operable, which provides sufficient flow through the system to maintain the system well mixed. If a reactor coolant pump is not operating, the demineralized water isolation valves are closed and an uncontrolled boron dilution transient cannot occur, as discussed in Subsection 15.4.6.2.1.

In the event of an inadvertent boron dilution transient in hot standby, the source range nuclear instrumentation detects a sufficiently large increase in the neutron flux, automatically initiates valve movement to terminate the dilution, and sounds an alarm. Upon the actuation of a source range flux doubling signal, the makeup flow to the reactor coolant system and the makeup pump suction line to the demineralized water transfer and storage system storage tank are isolated. This thereby terminates the dilution. In addition, the makeup pumps are tripped for equipment protection only. This function is not credited in the safety analysis.

Protective actions initiate about 32 minutes after start of dilution. No operator action is required to terminate this transient.

15.4.6.2.5 Dilution During Startup (Mode 2)

The plant is in the startup mode only for startup testing at the beginning of each cycle. During this mode of operation, rod control is in manual. Normal actions taken to change power level, either up or down, require operator actuation. The Technical Specifications require an available shutdown margin of 1.6-percent Δk/k and four reactor coolant pumps operating. Other conditions assumed are the following:

- There is a dilution flow of 200 gpm of unborated water.
- Minimum reactor coolant system water volume is 8126 ft^3. This is a very conservative estimate of the active reactor coolant system volume, minus the pressurizer volume.
- An initial maximum critical boron concentration, corresponding to the rods inserted to the insertion limits, is 1327 ppm. The minimum change in boron concentration from this initial condition to a hot zero power critical condition with all rods inserted is 1088 ppm. Full rod insertion, minus the most reactive stuck rod, occurs because of reactor trip.

This mode of operation is a transitory operational mode in which the operator intentionally dilutes and withdraws control rods to take the plant critical. During this mode, the plant is in manual control. For a normal approach to criticality, the operator manually withdraws control rods and dilutes the reactor coolant with unborated water at controlled rates until criticality is achieved. Once critical, the power escalation is slow enough to allow the operator to manually block the source range reactor trip after receiving the P-6 permissive signal from the intermediate range detectors (nominally at 10^5 cps). Too fast a power escalation (due to an unknown dilution) would result in reaching P-6 unexpectedly, leaving insufficient time to manually block the source range reactor trip. Failure to perform this manual action results in a reactor trip and immediate shutdown of the reactor.

Upon any reactor trip signal, or low input voltage to the Class 1E dc and uninterruptible power supply system battery chargers, a safety-related function automatically isolates the potentially unborated water from the demineralized water transfer and storage system and thereby terminates the dilution. Additionally, the suction lines for the chemical and volume control system pumps are automatically realigned to draw borated water from the chemical and volume control system boric acid tank.

After reactor trip, the dilution would have to continue for approximately 383 minutes to overcome the available shutdown margin. Even assuming that the non-safety-related boration operation does not occur, the unborated water that may remain in the purge volume of the chemical and volume control system is not sufficient to return the reactor to criticality. Therefore, the automatic termination of the dilution flow from the demineralized water transfer and storage system prevents a post-trip return to criticality.

15.4.6.2.6 Dilution During Full Power Operation (Mode 1)

The plant may be operated at power two ways: automatic T_{avg}/rod control and under operator control. The COLR and Technical Specifications require an available shutdown margin of 1.6-percent Δk/k and four reactor coolant pumps operating. With the plant at power and the reactor coolant system at pressure, the dilution rate is limited by the capacity of the chemical and volume control system makeup pumps. The analysis is performed assuming two chemical and volume control system pumps are in operation, even though normal operation is with one pump. Conditions assumed for a dilution in this mode are the following:

- There is a dilution flow of 200 gpm of unborated water.
- Minimum reactor coolant system water volume is 8126 ft^3. This is a very conservative estimate of the active reactor coolant system volume, minus the pressurizer volume.
- An initial maximum critical boron concentration, corresponding to the rods inserted to the insertion limits, is 1080 ppm. The minimum change in boron concentration from this initial condition to a hot zero power critical condition with all rods inserted is 841 ppm. Full rod insertion, minus the most reactive stuck rod, occurs due to reactor trip.

With the reactor in manual control and no operator action taken to terminate the transient, the power and temperature rise causes the reactor to reach the overtemperature ΔT trip setpoint resulting in a reactor trip. Upon any reactor trip signal, a safety-related function automatically isolates the unborated water from the demineralized water transfer and storage system and thereby terminates the dilution. Additionally, the suction lines for the chemical and volume control system pumps are automatically realigned to draw borated water from the chemical and volume control system boric acid tank.

Because the realignment of the suction for the chemical and volume control system pumps to the boric acid tank is a non-safety-related operation, the only consideration given to the reboration phase of the event in the safety analysis is the unborated purge volume.

After reactor trip, the dilution would have to continue for at least 325 minutes to overcome the available shutdown margin. The unborated water that may remain in the purge volume of the chemical and volume control system does not return the reactor to criticality. Therefore, the automatic termination of the dilution flow from the demineralized water transfer and storage system precludes a post-trip return to criticality.

Should a consequential loss of offsite power occur after reactor and turbine trip, it does not alter the fact that the dilution event has been terminated by automatic protection features. As indicated previously, the reactor trip signal that occurs in parallel with the turbine trip will actuate a safety-related function that automatically isolates the unborated water from the demineralized water system and thereby terminates the dilution. A subsequent loss of offsite power will cause the chemical and volume control system pumps to shut down. Should power and chemical and volume control system flow be restored, the unborated water that may remain in the purge volume of the chemical and volume control system will still not return the reactor to criticality.

The boron dilution transient in this case is essentially the equivalent to an uncontrolled rod withdrawal at power (see Subsection 15.4.2). The maximum reactivity insertion rate for a boron dilution transient is conservatively estimated to be in the range of 0.5 to 0.8 pcm per second and is within the range of insertion rates analyzed for uncontrolled rod withdrawal at power. Before reaching the overtemperature ΔT reactor trip, the operator receives an alarm on overtemperature ΔT and an overtemperature ΔT turbine runback.

With the reactor in automatic rod control, the pressurizer level controller limits the dilution flow rate to the maximum letdown rate. If a dilution rate in excess of the letdown rate is present, the pressurizer level controller throttles charging flow down to match letdown rate. For the safety analysis, a conservative dilution flow rate of 200 gpm is assumed. With the reactor in automatic rod control, a boron dilution results in a power and temperature increase in such a way that the rod controller attempts to compensate by slow insertion of the control rods. This action by the controller results in at least three alarms to the operator:

A. Rod insertion limit - low level alarm
B. Rod insertion limit - low-low level alarm if insertion continues
C. Axial flux difference alarm (ΔI outside of the target band)

Given the many alarms, indications, and the inherent slow process of dilution at power, the operator has sufficient time for action. The operator has at least 328 minutes from the rod insertion limit low-low alarm until shutdown margin is lost at beginning of cycle. The time is significantly longer at end of cycle because of the low initial boron concentration.

Because the analysis for the boron dilution event with the reactor in automatic rod control does not predict a reactor and turbine trip, considering the consequential loss of offsite power for this case is not needed.

The preceding results demonstrate that in all modes of operation, an inadvertent boron dilution is prevented or responded to by automatic functions, or sufficient time is available for operator action to terminate the transient. Following termination of the dilution flow and initiation of boration, the reactor is in a stable condition.

15.4.6.3 Conclusions

Inadvertent boron dilution events are prevented during refueling and automatically terminated during cold shutdown, safe shutdown, and hot standby modes. Inadvertent boron dilution events during startup or power operation, if not detected and terminated by the operators, result in an automatic reactor trip. Following reactor trip, automatic termination of the dilution occurs and post-trip return to criticality is prevented.

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

15.4.7.1 Identification of Causes and Accident Description

Fuel and core loading errors can inadvertently occur, such as those arising from the inadvertent loading of one or more fuel assemblies into improper positions, having a fuel rod with one or more pellets of the wrong enrichment, or having a full fuel assembly with pellets of the wrong enrichment. This leads to increased heat fluxes if the error results in placing fuel in core positions calling for fuel of lesser enrichment. Also included among possible core-loading errors is the inadvertent loading of one or more fuel assemblies requiring burnable poison rods into a new core without burnable poison rods.

An error in enrichment, beyond the normal manufacturing tolerances, can cause power shapes more peaked than those calculated with the correct enrichments. A 5-percent uncertainty margin is included in the design value of power peaking factor assumed in the analysis of Condition I and Condition II transients. The online core monitoring system is used to verify power shapes at the start of life and is capable of revealing fuel assembly enrichment errors or loading errors that cause power shapes to be peaked in excess of the design value. Power-distribution-related measurements are incorporated into the evaluation of calculated power distribution information using the incore instrumentation processing algorithms contained within the online monitoring system. The processing algorithms contained within the online monitoring system are functionally identical to those historically used for the evaluation of power distributions measurements in Westinghouse pressurized water reactors.

Each fuel assembly is marked with an identification number and loaded in accordance with a core-loading diagram to reduce the probability of core loading errors. During core loading, the identification number is checked before each assembly is moved into the core. Serial numbers read during fuel movement are subsequently recorded on the loading diagram as a further check on proper placement after the loading is completed.

The power distortion due to a combination of misplaced fuel assemblies could significantly increase peaking factors and is readily observable with the online core monitoring system. The fixed incore instrumentation within the instrumented fuel assembly locations is augmented with core exit thermocouples. There is a high probability that these thermocouples would also indicate any abnormally high coolant temperature rise. Incore flux measurements are taken during the startup subsequent to every refueling operation.

This event is a Condition III incident (an infrequent fault) as defined in Subsection 15.0.1.

15.4.7.2 Analysis of Effects and Consequences

15.4.7.2.1 Method of Analysis

Steady-state power distributions in the x-y plane of the core are calculated at 30-percent rated thermal power using the three-dimensional nodal code ANC (Reference 7). Representative power distributions in the x-y plane for a correctly loaded core are described in Chapter 4.

For each core loading error case analyzed, the percent deviations from detector readings for a normally loaded core are shown in the incore detector locations. (See Figures 15.4.7-1 through 15.4.7-4.)

15.4.7.2.2 Results

The following core loading error cases are analyzed:

Case A:

Case in which a Region 1 assembly is interchanged with a Region 3 assembly. The particular case considered is the interchange of two assemblies near the periphery of the core (see Figure 15.4.7-1).

Case B:

Case in which a Region 1 assembly is interchanged with a neighboring Region 2 fuel assembly. For the particular case considered, the interchange is assumed to take place close to the core center and with burnable poison rods located in the correct Region 2 position, but in a Region 1 assembly mistakenly loaded in the Region 2 position (see Figure 15.4.7-2).

Case C:

Enrichment error – Case in which a Region 2 fuel assembly is loaded in the core central position (see Figure 15.4.7-3).

Case D:

Case in which a Region 2 fuel assembly instead of a Region 1 assembly is loaded near the core periphery (see Figure 15.4.7-4).

15.4.7.3 Conclusions

Fuel assembly enrichment errors are prevented by administrative procedures implemented in fabrication.

In the event that a single pin or pellet has a higher enrichment than the nominal value, the consequences in terms of reduced DNBR and increased fuel and cladding temperatures are limited to the incorrectly loaded pin or pins and perhaps the immediately adjacent pins.

Fuel assembly loading errors are prevented by administrative procedures implemented during core loading. In the unlikely event that a loading error occurs, analyses in this section confirm that resulting power distribution effects are either readily detected by the online core monitoring system or cause a sufficiently small perturbation to be acceptable within the uncertainties allowed between nominal and design power shapes.

15.4.8 Spectrum of Rod Cluster Control Assembly Ejection Accidents

15.4.8.1 Identification of Causes and Accident Description

This accident is defined as the mechanical failure of a control rod mechanism pressure housing, resulting in the ejection of an RCCA and drive shaft. The consequence of this mechanical failure is a rapid positive reactivity insertion together with an adverse core power distribution, possibly leading to localized fuel rod damage.

15.4.8.1.1 Design Precautions and Protection

15.4.8.1.1.1 Mechanical Design

The mechanical design is discussed in Section 4.6. Mechanical design and quality control procedures intended to prevent the possibility of an RCCA drive mechanism housing failure are listed below:

- Each control rod drive mechanism housing is completely assembled and shop tested at 4100 psi.
- The mechanism housings are individually hydrotested after they are attached to the head adapters in the reactor vessel head. The housings are checked during the hydrotest of the completed reactor coolant system.
- Stress levels in the mechanism are not affected by anticipated system transients at power or by the thermal movement of the coolant loops. Moments induced by the safe shutdown earthquake can be accepted within the allowable primary working stress range specified by the ASME Code, Section III, for Class 1 components.
- The latch mechanism housing and rod travel housing are each a single length of forged stainless steel. This material exhibits excellent notch toughness at temperatures that are encountered.

A significant margin of strength in the elastic range together with the large energy absorption capability in the plastic range gives additional confidence that gross failure of the housing does not occur. The joints between the latch mechanism housing and head adapter, and between the latch mechanism housing and rod travel housing, are threaded joints reinforced by canopy-type rod welds, which are subject to periodic inspections.

15.4.8.1.1.2 Nuclear Design

If a rupture of an RCCA drive mechanism housing is postulated, the operation using chemical shim is such that the severity of an ejected RCCA is inherently limited. In general, the reactor is operated with the power control (or mechanical shim) RCCAs inserted only far enough to permit load follow. The axial offset RCCAs are positioned so that the targeted axial offset can be met throughout core life. Reactivity changes caused by core depletion and xenon transients are normally compensated for by boron changes and the mechanical shim banks, respectively. Further, the location and grouping of the power control and axial offset RCCAs are selected with consideration for an RCCA ejection accident. Therefore, should an RCCA be ejected from its normal position during full-power operation, a less severe reactivity excursion than analyzed is expected.

It may occasionally be desirable to operate with larger than normal insertions. For this reason, a power control and axial offset rod insertion limit is defined as a function of power level. Operation with the RCCAs above this limit provides adequate shutdown capability and an acceptable power distribution. The position of the RCCAs is continuously indicated in the main control room. An alarm occurs if a bank of RCCAs approaches its insertion limit or if one RCCA deviates from its bank. Operating instructions require boration at the low level alarm and emergency boration at the low-low level alarm.

15.4.8.1.1.3 Reactor Protection

The reactor protection in the event of a rod ejection accident is described in WCAP-15806-P-A (Reference 4). The protection for this accident is provided by the high neutron flux trip (high and low setting) and the high rate of neutron flux increase trip. These protection functions are described in Section 7.2.

15.4.8.1.1.4 Effects on Adjacent Housings

Failures of an RCCA mechanism housing, due to either longitudinal or circumferential cracking, does not cause damage to adjacent housings. The control rod drive mechanism is described in Subsection 3.9.4.1.1.

15.4.8.1.1.5 Not Used

15.4.8.1.1.6 Not Used

15.4.8.1.1.7 Consequences

The probability of damage to an adjacent housing is considered remote. If damage is postulated, it is not expected to lead to a more severe transient because RCCAs are inserted in the core in symmetric patterns and control rods immediately adjacent to worst ejected rods are not in the core when the reactor is critical. Damage to an adjacent housing could, at worst, cause that RCCA not to fall on receiving a trip signal. This is already taken into account in the analysis by assuming a stuck rod adjacent to the ejected rod.

15.4.8.1.1.8 Summary

Failure of a control rod housing does not cause damage to adjacent housings that increase the severity of the initial accident.

15.4.8.1.2 Limiting Criteria

This event is a Condition IV incident (ANSI N18.2). See Subsection 15.0.1 for a discussion of ANS classification. Because of the extremely low probability of an RCCA ejection accident, some fuel damage is considered an acceptable consequence.

NUREG-0800 Standard Review Plan (SRP) 4.2, Revision 3 (Reference 24), interim criteria applicable to new plant design certification are applied to provide confidence that there is little or no possibility of fuel dispersal in the coolant, gross lattice distortion, or severe shock waves. These criteria are the following:

- The pellet clad mechanical interaction (PCMI) failure criteria is a change in radial average fuel enthalpy greater than the corrosion-dependent limit depicted in Figure B-1 of SRP 4.2, Revision 3, Appendix B.
- The high cladding temperature failure criteria for zero-power conditions is a peak radial average fuel enthalpy greater than 170 cal/g for fuel rods with an internal rod pressure at or below system pressure and 150 cal/g for fuel rods with an internal rod pressure exceeding system pressure.
- For intermediate (greater than 5-percent rated thermal power) and full-power conditions, fuel cladding is presumed to fail if local heat flux exceeds thermal design limits (e.g., DNBR).
- For core coolability, it is conservatively assumed that the average fuel pellet enthalpy at the hot spot remains below 200 cal/g (360 Btu/lb) for irradiated fuel. This bounds non-irradiated fuel, which has a slightly higher enthalpy limit.
- For core coolability, the peak fuel temperature must remain below incipient fuel melting conditions.
- Mechanical energy generated as a result of (1) non-molten fuel-to-coolant interaction and (2) fuel rod burst that must be addressed with respect to reactor pressure boundary, reactor internals, and fuel assembly structural integrity.
- No loss of coolable geometry due to (1) fuel pellet and cladding fragmentation and dispersal and (2) fuel rod ballooning.
- Peak reactor coolant system pressure is less than that which could cause stresses to exceed the “Service Limit C” as defined in the ASME code.

15.4.8.2 Analysis of Effects and Consequences

Method of Analysis

The calculation of the RCCA ejection transients is performed in two stages: first, an average core calculation and then, a hot rod calculation. The average core calculation is performed using spatial neutron kinetics methods to determine the average power generation with time, including the various total core feedback effects (Doppler reactivity and moderator reactivity). Enthalpy, fuel temperature, and DNB transients are then determined by performing a conservative fuel rod transient heat transfer calculation.

A discussion of the method of analysis appears in WCAP-15806-P-A (Reference 4).

Average Core Analysis

The three-dimensional nodal code ANC (References 14, 15, 16, 17, 21, 22, and 27) is used for the average core transient analysis. This code solves the two-group neutron diffusion theory kinetic equation in three spatial dimensions (rectangular coordinates) for six delayed neutron groups. The core moderator and fuel temperature feedbacks are based on the NRC approved Westinghouse version of the VIPRE-01 code and methods (References 18 and 19).

Hot Rod Analysis

The hot fuel rod models are based on the Westinghouse VIPRE models described in WCAP-15806-P-A (Reference 4). The hot rod model represents the hottest fuel rod from any channel in the core. VIPRE performs the hot rod transients for fuel enthalpy, temperature, and DNBR using as input the time dependent nuclear core power and power distribution from the core average analysis. A description of the VIPRE code is provided in Reference 18.

System Overpressure Analysis

If the fuel coolability limits are not exceeded, the fuel dispersal into the coolant or a sudden pressure increase from thermal to kinetic energy conversion is not needed to be considered in the overpressure analysis. Therefore, the overpressure condition may be calculated on the basis of conventional fuel rod to coolant heat transfer and the prompt heat generation in the coolant. The system overpressure analysis is conducted by first performing the core power response analysis to obtain the nuclear power transient (versus time) data. The nuclear power data is then used as input to a plant transient computer code to calculate the peak reactor coolant system pressure.

This code calculates the pressure transient, taking into account fluid transport in the reactor coolant system and heat transfer to the steam generators. For conservatism, no credit is taken for the possible pressure reduction caused by the assumed failure of the control rod pressure housing.

15.4.8.2.1 Calculation of Basic Parameters

Input parameters for the analysis are conservatively selected as described in Reference 4. Table 15.4-3 is deleted and not used.

15.4.8.2.1.1 Ejected Rod Worths and Hot Channel Factors

The values for ejected rod worths and hot channel factors are calculated using three-dimensional static methods. Standard nuclear design codes are used in the analysis. The calculation is performed for the maximum allowed bank insertion at a given power level, as determined by the rod insertion limits. Adverse xenon distributions are considered in the calculation.

Appropriate safety analysis allowances are added to the ejected rod worth and hot channel factors to account for calculational uncertainties, including an allowance for nuclear peaking due to densification as discussed in Reference 4.

15.4.8.2.1.2 Not Used

15.4.8.2.1.3 Moderator and Doppler Coefficients

The critical boron concentration is adjusted in the nuclear code to obtain a moderator temperature coefficient that is conservative compared to actual design conditions for the plant consistent with Reference 4. The fuel temperature feedback in the neutronics code is reduced consistent with Reference 4 requirements.

15.4.8.2.1.4 Delayed Neutron Fraction, β_{eff}

Calculations of the effective delayed neutron fraction (β_{eff}) typically yield values no less than 0.50 percent at end of cycle. The accident is sensitive to β_{eff} if the ejected rod worth is equal to or greater than β_{eff}. To allow for future cycles, a pessimistic estimate of β_{eff} of 0.44 percent is used in the analysis.

15.4.8.2.1.5 Trip Reactivity Insertion

The trip reactivity insertion accounts for the effect of the ejected rod and one adjacent stuck rod. The trip reactivity is simulated by dropping a limited set of rods of the required worth into the core. The start of rod motion occurs 0.9 second after the high neutron flux trip setpoint is reached. This delay is assumed to consist of 0.583 second for the instrument channel to produce a signal, 0.167 second for the trip breakers to open, and 0.15 second for the coil to release the rods. A curve of trip rod insertion versus time is used, which assumes that insertion to the dashpot does not occur until 2.47 seconds after the start of fall. The choice of such a conservative insertion rate means that there is over 1 second after the trip setpoint is reached before significant shutdown reactivity is inserted into the core. This conservatism is important for the hot full power accidents.

The minimum design shutdown margin available at hot zero power may be reached only at end of life in the equilibrium cycle. This value includes an allowance for the worst stuck rod, adverse xenon distribution, conservative Doppler and moderator defects, and an allowance for calculational uncertainties. Calculations show that the effect of two stuck RCCAs (one of which is the worst ejected rod) is to reduce the shutdown by about an additional 1-percent Δk. Therefore, following a reactor trip resulting from an RCCA ejection accident, the reactor is subcritical when the core returns to hot zero power.

15.4.8.2.1.6 Reactor Protection

As discussed in Subsection 15.4.8.1.1.3, reactor protection for a rod ejection is provided by the high neutron flux trip (high and low setting) and the high rate of neutron flux increase trip. These protection functions are part of the protection and safety monitoring system. No single failure of the protection and safety monitoring system negates the protection functions required for the rod ejection accident or adversely affects the consequences of the accident.

15.4.8.2.1.7 Results

For all cases, the core is preconditioned by assuming a fuel cycle depletion with control rod insertion that is conservative relative to expected baseload operation. All cases assume that the mechanical shim and axial offset control RCCAs are inserted to their insertion limits before the event and xenon is skewed to yield a conservative initial axial power shape. The limiting RCCA ejection cases for a typical cycle are summarized following the criteria outlined in Subsection 15.4.8.1.2.

- PCMI and high cladding temperature (hot zero power)

  The resulting maximum fuel average enthalpy rise and maximum fuel average enthalpy are less than the criteria given in Subsection 15.4.8.1.2.

- High cladding temperature (≥ 5% rated thermal power)

  The fraction of the core calculated to have a DNBR less than the safety analysis limit is less than the amount of failed fuel assumed in the dose analysis described in Subsection 15.4.8.3.

- Core coolability

  The resulting maximum fuel average enthalpy is less than the criterion given in Subsection 15.4.8.1.2. Fuel melting is not predicted to occur at the hot spot.

There are no fuel failures due to the fuel enthalpy deposition, i.e., both fuel and cladding enthalpy limits were met. Additionally, the coolability criteria for peak fuel enthalpy and the fuel melting criteria were met. Therefore, the fuel dispersal into the coolant, a sudden pressure increase from thermal to kinetic energy conversion, gross lattice distortion, or severe shock waves are precluded.

The nuclear power and fuel transients for the limiting cases are presented in Figures 15.4.8-1 through 15.4.8-3.

The calculated sequence of events for the limiting cases is presented in Table 15.4-1. Reactor trip occurs early in the transients, after which the nuclear power excursion is terminated.

The ejection of an RCCA constitutes a break in the reactor coolant system, located in the reactor pressure vessel head. The effects and consequences of loss-of-coolant accidents (LOCAs) are discussed in Subsection 15.6.5. Following the RCCA ejection, the plant response is the same as a LOCA.

The consequential loss of offsite power described in Subsection 15.0.14 is not limiting for the enthalpy and temperature transients resulting from an RCCA ejection accident. Due to the delay from reactor trip until turbine trip and the rapid power reduction produced by the reactor trip, the peak fuel and cladding temperatures occur before the reactor coolant pumps begin to coast down.

15.4.8.2.1.8 Fission Product Release

It is assumed that fission products are released from the gaps of all rods entering DNB. In the cases considered, less than 10 percent of the rods are assumed to enter DNB based on a detailed three-dimensional kinetics and hot rod analysis. The maximum fuel average enthalpy rise of rods predicted to enter DNB will be less than 60 cal/g. Fuel melting does not occur at the hot spot.

The consequential loss of offsite power described in Subsection 15.0.14 is not limiting for the calculation of the number of rods assumed to enter DNB for the RCCA ejection accident. Due to the delay from reactor trip until turbine trip and the rapid power reduction produced by the reactor trip, the minimum DNBR, for rods where the DNBR did not fall below the design limit (see Section 4.4) in the cases described, occurs before the reactor coolant pumps begin to coast down.

15.4.8.2.1.9 Peak Reactor Coolant System Pressure

Calculations of the peak reactor coolant system pressure demonstrate that the peak pressure does not exceed that which would cause the stress to exceed the Service Level C Limit as described in the ASME Code, Section III. Therefore, the accident for this plant does not result in an excessive pressure rise or further damage to the reactor coolant system.

The consequential loss of offsite power described in Subsection 15.0.14 is not limiting for the pressure surge transient resulting from an RCCA ejection accident. Due to the delay from reactor trip until turbine trip and the rapid power reduction produced by the reactor trip, the peak system pressure occurs before the reactor coolant pumps begin to coast down.

15.4.8.2.1.10 Lattice Deformations

A large temperature gradient exists in the region of the hot spot. Because the fuel rods are free to move in the vertical direction, differential expansion between separate rods cannot produce distortion. However, the temperature gradients across individual rods may produce a differential expansion, tending to bow the midpoint of the rods toward the hotter side of the rod.

Calculations indicate that this bowing results in a negative reactivity effect at the hot spot because the core is undermoderated, and bowing tends to increase the undermoderation at the hot spot. In practice, no significant bowing is anticipated because the structural rigidity of the core is sufficient to withstand the forces produced.

Boiling in the hot spot region would produce a net flow away from that region. However, the heat from the fuel is released to the water relatively slowly, and it is considered inconceivable that crossflow is sufficient to produce lattice deformation. Even if massive and rapid boiling, sufficient to distort the lattices, is hypothetically postulated, the large void fraction in the hot spot region produces a reduction in the total core moderator to fuel ratio and a large reduction in this ratio at the hot spot. The net effect is therefore a negative feedback.

In conclusion, no credible mechanism exists for a net positive feedback resulting from lattice deformation. In fact, a small negative feedback may result. The effect is conservatively ignored in the analysis.

15.4.8.3 Radiological Consequences

The evaluation of the radiological consequences of a postulated rod ejection accident assumes that the reactor is operating with a limited number of fuel rods containing cladding defects and that leaking steam generator tubes result in a buildup of activity in the secondary coolant. See Subsection 15.4.8.3.1 and Table 15.4-4.

As a result of the accident, 10 percent of the fuel rods are assumed to be damaged (see Subsection 15.4.8.2.1.8) such that the activity contained in the fuel cladding gap is released to the reactor coolant. No fuel melt is calculated to occur as a result of the rod ejection (see Subsection 15.4.8.2.1.8).

Activity released to the containment via the spill from the reactor vessel head is assumed to be available for release to the environment because of containment leakage. Activity carried over to the secondary side due to primary-to-secondary leakage is available for release to the environment through the steam line safety or power-operated relief valves.

15.4.8.3.1 Source Term

The significant radionuclide releases due to the rod ejection accident are the iodines, alkali metals, and noble gases. The reactor coolant iodine source term assumes a pre-existing iodine spike. The reactor coolant noble gas concentrations are assumed to be those associated with equilibrium operating limits for primary coolant noble gas activity. The initial reactor coolant alkali metal concentrations are assumed to be those associated with the design fuel defect level. These initial reactor coolant activities are of secondary importance compared to the release of fission products from the portion of the core assumed to fail.

Based on NUREG-1465 (Reference 12), the fission product gap fraction is 3 percent of fuel inventory. For this analysis, the gap fractions are modified following the guidance of DG-1199 (Reference 25), which incorporates the effects of enthalpy rise in the fuel following the reactivity insertion, consistent with Appendix B of SRP 4.2, Revision 3 (Reference 24). DG-1199 included expanded guidance for determining nuclide gap fractions available for release following a rod ejection. Reference 26 was issued as a clarification to the gap fraction guidance in DG-1199. An enthalpy rise of 60 cal/gm is used to calculate the gap fractions (see Subsection 15.4.8.2.1.8). Also, to address the fact that the failed fuel rods may have been operating at power levels above the core average, the source term is increased by the lead rod radial peaking factor. No fuel melt is calculated to occur as a result of the rod ejection (see Subsection 15.4.8.2.1.8).

15.4.8.3.2 Release Pathways

There are three components to the accident releases:

- The activity initially in the secondary coolant is available for release as long as steam releases continue.

- The reactor coolant leaking into the steam generators is assumed to mix with the secondary coolant. The activity from the primary coolant mixes with the secondary coolant and, as steam is released, a portion of the iodine and alkali metal in the coolant is released. The fraction of activity released is defined by the assumed flashing fraction and the partition coefficient assumed for the steam generator. The noble gas activity entering the secondary side is released to the environment. These releases are terminated when the steam releases stop.

- The activity from the reactor coolant system and the core is released to the containment atmosphere and is available for leakage to the environment through the assumed design basis containment leakage.

Credit is taken for decay of radionuclides until release to the environment. After release to the environment, no consideration is given to radioactive decay or to cloud depletion by ground deposition during transport offsite.

15.4.8.3.3 Dose Calculation Models

The models used to calculate doses are provided in Appendix 15A.

15.4.8.3.4 Analytical Assumptions and Parameters

The assumptions and parameters used in the analysis are listed in Table 15.4-4.

15.4.8.3.5 Identification of Conservatisms

The assumptions used in the analysis contain a number of conservatisms:

- Although fuel damage is assumed to occur as a result of the accident, no fuel damage is anticipated.

- The reactor coolant activities are based on conservative assumptions (refer to Table 15.4-4); whereas, the activities based on the expected fuel defect level are far less (see Section 11.1).

- The leakage of reactor coolant into the secondary system, at 300 gallons per day, is conservative. The leakage is normally a small fraction of this.

- It is unlikely that the conservatively selected meteorological conditions are present at the time of the accident.

- The leakage from containment is assumed to continue for a full 30 days. It is expected that containment pressure is reduced to the point that leakage is negligible before this time.

15.4.8.3.6 Doses

Using the assumptions from Table 15.4-4, the calculated total effective dose equivalent (TEDE) doses are determined to be 4.0 rem at the site boundary for the limiting 2-hour interval (0 to 2 hours) and 5.9 rem at the low population zone outer boundary. These doses are well within the dose guideline of 25 rem TEDE identified in 10 CFR Part 50.34. The phrase “well within” is taken as being 25 percent or less.

At the time the rod ejection accident occurs, the potential exists for a coincident loss of spent fuel pool cooling with the result that the pool could reach boiling and a portion of the radioactive iodine in the spent fuel pool could be released to the environment. The loss of spent fuel pool cooling has been evaluated for a duration of 30 days. There is no contribution to the 2-hour site boundary dose because the pool boiling would not occur until after the first 2 hours. The 30-day contribution to the dose at the low population zone boundary is less than 0.01 rem TEDE, and when this is added to the dose calculated for the rod ejection accident, the resulting total dose remains less than the value reported above.

15.4.9 Combined License Information

This section contained no requirement for additional information.

15.4.10 References

1. Barry, R. F., and Risher, D. H., Jr., “TWINKLE--A Multi-Dimensional Neutron Kinetics Computer Code,” WCAP-7979-P-A (Proprietary) and WCAP-8028-A (Nonproprietary), January 1975.

2. Hargrove, H. G., “FACTRAN--A FORTRAN-IV Code for Thermal Transients in a UO_2 Fuel Rod,” WCAP-7908-A, December 1989.

3. Burnett. T. W. T., et al., “LOFTRAN Code Description,” WCAP-7907-P-A (Proprietary) and WCAP-7907-A (Nonproprietary), April 1984.

4. Beard, C. L. et al., “Westinghouse Control Rod Ejection Accident Analysis Methodology Using Multi-Dimensional Kinetics,” WCAP-15806-P-A (Proprietary) and WCAP-15807-NP-A (Nonproprietary), November 2003.

5. Taxelius, T. G., ed, “Annual Report-SPERT Project, October 1968, September 1969,” Idaho Nuclear Corporation, IN-1370, June 1970.

6. Liimataninen, R. C., and Testa, F. J., “Studies in TREAT of Zircaloy-2-Clad, UO_2-Core Simulated Fuel Elements,” ANL-7225, January-June 1966, p 177, November 1966.

7. Liu, Y. S., et al., “ANC – A Westinghouse Advanced Nodal Computer Code,” WCAP-10965-P-A (Proprietary) and WCAP-10966-A (Nonproprietary), September 1986.

8. Not Used.

9. Friedland, A. J., and Ray, S., “Revised Thermal Design Procedure,” WCAP-11397-P-A (Proprietary) and WCAP-11397-A (Nonproprietary), April 1989.

10. American National Standards Institute N18.2, “Nuclear Safety Criteria for the Design of Stationary PWR Plants,” 1973.

11. “AP1000 Code Applicability Report,” WCAP-15644-P (Proprietary) and WCAP-15644-NP (Nonproprietary), Revision 2, March 2004.

12. Soffer, L. et al., “Accident Source Terms for Light-Water Nuclear Power Plants,” NUREG-1465, February 1995.

13. Not Used.

14. Nguyen, T. Q., et al., “Qualifications of the PHOENIX-P/ANC Nuclear Design System for Pressurized Water Reactor Cores,” WCAP-11596-P-A (Proprietary) and WCAP-11597-A (Nonproprietary), June 1988.

15. Ouisloumen, M., et al., “Qualification of the Two-Dimensional Transport Code PARAGON,” WCAP-16045-P-A (Proprietary) and WCAP-16045-NP-A (Nonproprietary), August 2004.

16. Liu, Y. S., “ANC – A Westinghouse Advanced Nodal Computer Code; Enhancements to ANC Rod Power Recovery,” WCAP-10965-P-A, Addendum 1 (Proprietary) and WCAP-10966-A Addendum 1 (Nonproprietary), April 1989.

17. Letter from Liparulo, N.J. (Westinghouse) to Jones, R. C., (NRC), “Notification to the NRC Regarding Improvements to the Nodal Expansion Method Used in the Westinghouse Advanced Nodal Code (ANC),” NTD-NRC-95-4533, August 22, 1995.

18. Sung, Y. X., Schueren, P., and Meliksetian, A., “VIPRE-01 Modeling and Qualification for Pressurized Water Reactor Non-LOCA Thermal-Hydraulic Safety Analysis,” WCAP-14565-P-A (Proprietary) and WCAP-15306-NP-A (Nonproprietary), October 1999.

19. Stewart, C. W., et al., “VIPRE-01: A Thermal/Hydraulic Code for Reactor Cores,” Volumes 1, 2, 3 (Revision 3, August 1989), and Volume 4 (April 1987), NP-2511-CCM-A, Electric Power Research Institute, Palo Alto, California.

20. Foster, J. P. and Sidener, S., “Westinghouse Improved Performance Analysis and Design Model (PAD 4.0),” WCAP-15063-P-A, Revision 1 with Errata (Proprietary) and WCAP-15064-NP-A (Nonproprietary), July 2000.

21. Zhang, B., et al., “Qualification of the NEXUS Nuclear Data Methodology,” WCAP-16045-P-A, Addendum 1-A (Proprietary) and WCAP-16045-NP-A, Addendum 1-A (Nonproprietary), August 2007.

22. Zhang, B., et al., “Qualification of the New Pin Power Recovery Methodology,” WCAP-10965-P-A, Addendum 2-A (Proprietary), September 2010.

23. Smith, L. D, et al., “Modified WRB-2 Correlation, WRB-2M, for Predicting Critical Heat Flux in 17x17 Rod Bundles with Modified LPD Mixing Vane Grids,” WCAP-15025-P-A (Proprietary) and WCAP-15026-NP-A (Nonproprietary), April 1999.

24. NUREG-0800, Standard Review Plan, Section 4.2, Revision 3, “Fuel System Design,” Appendix B, “Interim Acceptance Criteria and Guidance for the Reactivity Initiated Accidents,” March 2007.

25. Draft Regulatory Guide DG-1199, “Proposed Revision 1 of Regulatory Guide 1.183; Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors,” October 2009. NRC ADAMS Accession Number: ML090960464.

26. NRC Memorandum from Anthony Mendiola to Travis Tate, “Technical Basis for Revised Regulatory Guide 1.183 (DG-1199) Fission Product Fuel-to-Cladding Gap Inventory,” July 2011. NRC ADAMS Accession Number: ML111890397.

27. Letter from Liparulo, N. J. (Westinghouse) to Jones, R. C. (NRC), “Process Improvement to the Westinghouse Neutronics Code System,” NSD-NRC-96-4679, March 29, 1996.

Table 15.4-1 (Sheet 1 of 3)
Time Sequence of Events for Incidents Which Result in Reactivity and Power Distribution Anomalies

Accident	Event	Time (seconds)
Uncontrolled RCCA bank withdrawal from a subcritical or low-power startup condition	Initiation of uncontrolled rod withdrawal from 10^{-9} of nominal power	0.0
	Power range high neutron flux (low setting) setpoint reached	10.4
	Peak nuclear power occurs	10.6
	Rods begin to fall into core	11.3
	Peak heat flux occurs	12.7
	Minimum DNBR occurs	12.7
	Peak average clad temperature occurs	13.3
	Peak average fuel temperature occurs	13.4
One or more dropped RCCAs	Rods drop	0.0
	Control system initiates control bank withdrawal	0.4
	Peak nuclear power occurs	21.7
	Peak core heat flux occurs	24.2
Uncontrolled RCCA bank withdrawal at power		
1. Case A	Initiation of uncontrolled RCCA withdrawal at a high-reactivity insertion rate (75 pcm/s)	0.0
	Power range high neutron flux high trip point reached	6.6
	Rods begin to fall into core	7.5
	Minimum DNBR occurs	7.7
	Loss of ac power occurs	15.2
2. Case B	Initiation of uncontrolled RCCA withdrawal at a small reactivity insertion rate (3 pcm/s)	0.0
	Overtemperature ΔT setpoint reached	524.4
	Rods begin to fall into core	526.4
	Minimum DNBR occurs	526.7
	Loss of ac power occurs	534.1

Table 15.4-1 (Sheet 2 of 3)
Time Sequence of Events for Incidents Which Result in Reactivity and Power Distribution Anomalies

Accident	Event	Time (seconds)
Chemical and volume control system malfunction that results in a decrease in the boron concentration in the rector coolant		
1. Dilution during startup	Power range – low setpoint reactor trip due to dilution	0.0
	Dilution automatically terminated by demineralized water transfer and storage system isolation	215.0
2. Dilution during full-power Operation		
a. Automatic reactor control	Operator receives low-low rod insertion limit alarm due to dilution	0.0
	Shutdown margin lost	19,680
b. Manual reactor control	Initiate dilution	0.0
	Reactor trip on overtemperature ΔT due to dilution	180.0
	Dilution automatically terminated by demineralized water transfer and storage system isolation	395.0
RCCA ejection accident		
1. PCMI limiting event	Initiation of rod ejection	0.00
	Peak nuclear power occurs	0.14
	Reactor trip setpoint reached	<0.30
	Peak cladding temperature occurs	0.36
	Peak enthalpy deposition occurs	0.44
	Rods begin to fall into core	1.20

Table 15.4-1 (Sheet 3 of 3)
Time Sequence of Events for Incidents Which Result in Reactivity and Power Distribution Anomalies

Accident	Event	Time (seconds)
2. Peak cladding temperature limiting event	Initiation of rod ejection	0.00
	Peak nuclear power occurs	0.08
	Minimum DNBR occurs	0.11
	Peak cladding temperature occurs	0.11
	Reactor trip setpoint reached	<0.30
	Rods begin to fall into core	1.20
3. Peak enthalpy/peak fuel centerline temperature event	Initiation of rod ejection	0.00
	Peak nuclear power occurs	0.06
	Reactor trip setpoint reached	<0.30
	Rods begin to fall into core	1.20
	Peak fuel center temperature occurs	2.50
	Peak cladding temperature occurs	2.80

Table 15.4-2
Parameters

Assumed Dilution Flow Rates		
Mode	**Flow Rate (gal/min)**	
3 through 5	175	
1 through 2	200	
Volume		
Mode	**Volume (ft^3)**	**Volume (gal)**
1 and 2	8126	60,786
3	7539.8	56,401
4	7539.8	56,401
5	2592.2	19,391

Table 15.4-3
Not Used

Table 15.4-4 (Sheet 1 of 2)
Parameters Used in Evaluating the Radiological Consequences of a Rod Ejection Accident

Initial reactor coolant iodine activity	An assumed iodine spike that has resulted in an increase in the reactor coolant activity to 60 μCi/g (2.22E+06 Bq/g) of dose equivalent I-131 (see Appendix 15A)[(a)]
Reactor coolant noble gas activity	Equal to the operating limit for reactor coolant activity of 280 μCi/g (1.036E+07 Bq/g) dose equivalent Xe-133
Reactor coolant alkali metal activity	Design basis activity (see Table 11.1-2)
Secondary coolant initial iodine and alkali metal activity	1% of reactor coolant concentrations at maximum equilibrium conditions
Radial peaking factor (for determination of activity in failed/melted fuel)	1.75
Fuel cladding failure	
– Fraction of fuel rods assumed to fail	0.1
– Fuel enthalpy increase (cal/g)	60
– Fission product gap fractions	
Iodine 131	0.1238
Iodine 132	0.1338
Krypton 85	0.5120
Other noble gases	0.1238
Other halogens	0.0938
Alkali metals	0.6860
Iodine chemical form (%)	
– Elemental	4.85
– Organic	0.15
– Particulate	95.0
Core activity	See Table 15A-3 in Appendix 15A
Nuclide data	See Table 15A-4 in Appendix 15A
Reactor coolant mass (lb)	3.7E+05 (1.68E+05 kg)

Note:
a. The assumption of a pre-existing iodine spike is a conservative assumption for the initial reactor coolant activity. However, compared to the activity assumed to be released from damaged fuel, it is not significant.

Table 15.4-4 (Sheet 2 of 2)
Parameters Used in Evaluating the Radiological Consequences of a Rod Ejection Accident

Condenser	Not available
Duration of accident (days)	30
Atmospheric dispersion (χ/Q) factors	See Table 15A-5
Secondary system release path	
– Primary to secondary leak rate (lb/hr)	104.5[a] (47.4 kg/hr)
– Leak flashing fraction	0.04[b]
– Secondary coolant mass (lb)	6.06 E+05 (2.75E+05 kg)
– Duration of steam release from secondary system (sec)	1800
– Steam released from secondary system (lb)	1.08 E+05 (4.90E+04 kg)
– Partition coefficient in steam generators	
• Iodine	0.01
• Alkali metals	0.0035
Containment leakage release path	
– Containment leak rate (% per day)	
• 0-24 hr	0.10
• >24 hr	0.05
– Airborne activity removal coefficients (hr^{-1})	
• Elemental iodine	1.9[c]
• Organic iodine	0
• Particulate iodine or alkali metals	0.1
– Decontamination factor limit for elemental iodine removal	200
– Time to reach the decontamination factor limit for elemental iodine (hr)	2.78

Notes:

a. Equivalent to 300 gpd (1.14 m^3/day) cooled liquid at 62.4 lb/ft^3 (999.6 kg/m^3).
b. No credit for iodine partitioning is taken for flashed leakage.
c. From Appendix 15B.

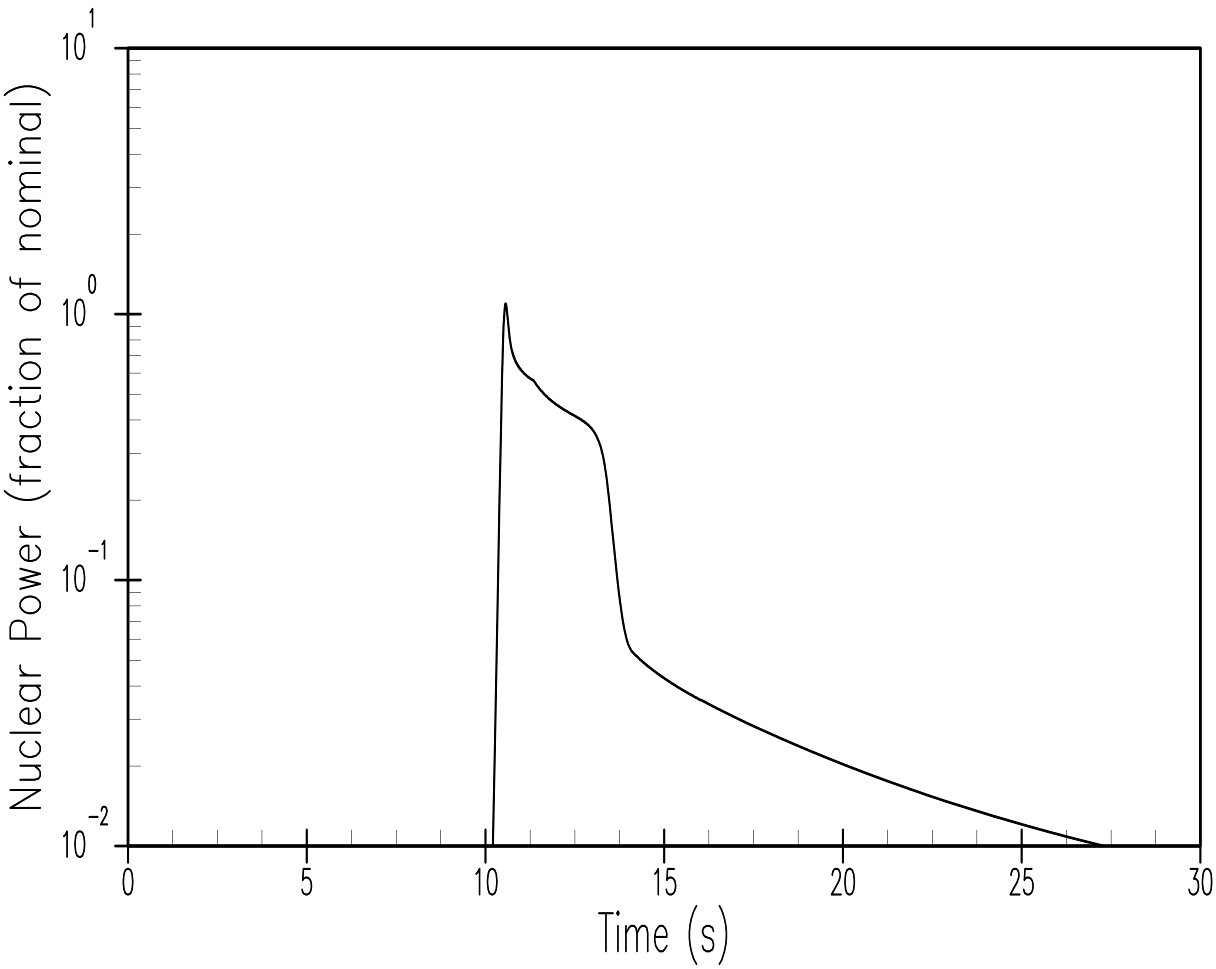


Figure 15.4.1-1
RCCA Withdrawal from Subcritical Nuclear Power

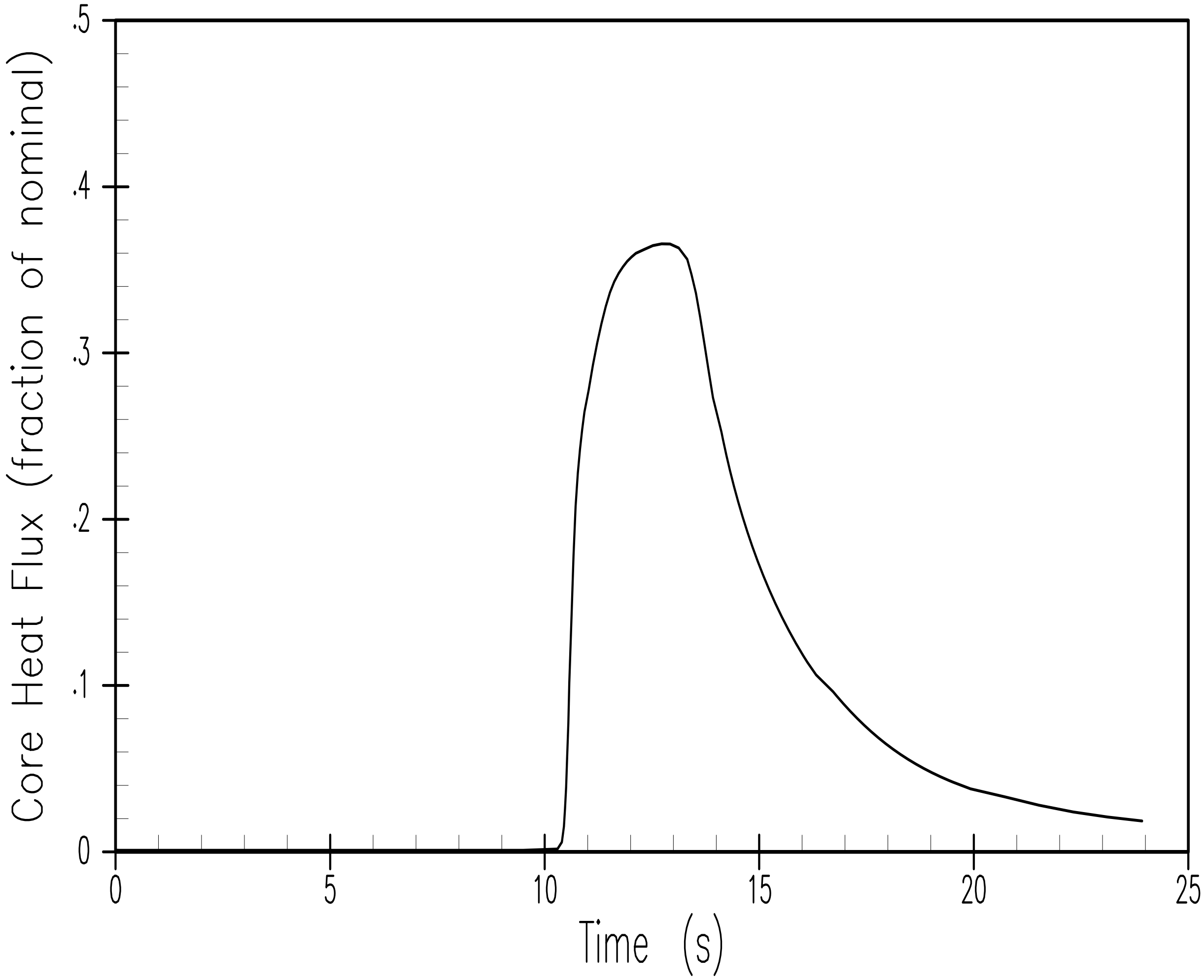


Figure 15.4.1-2
RCCA Withdrawal from Subcritical
Average Channel Core Heat Flux

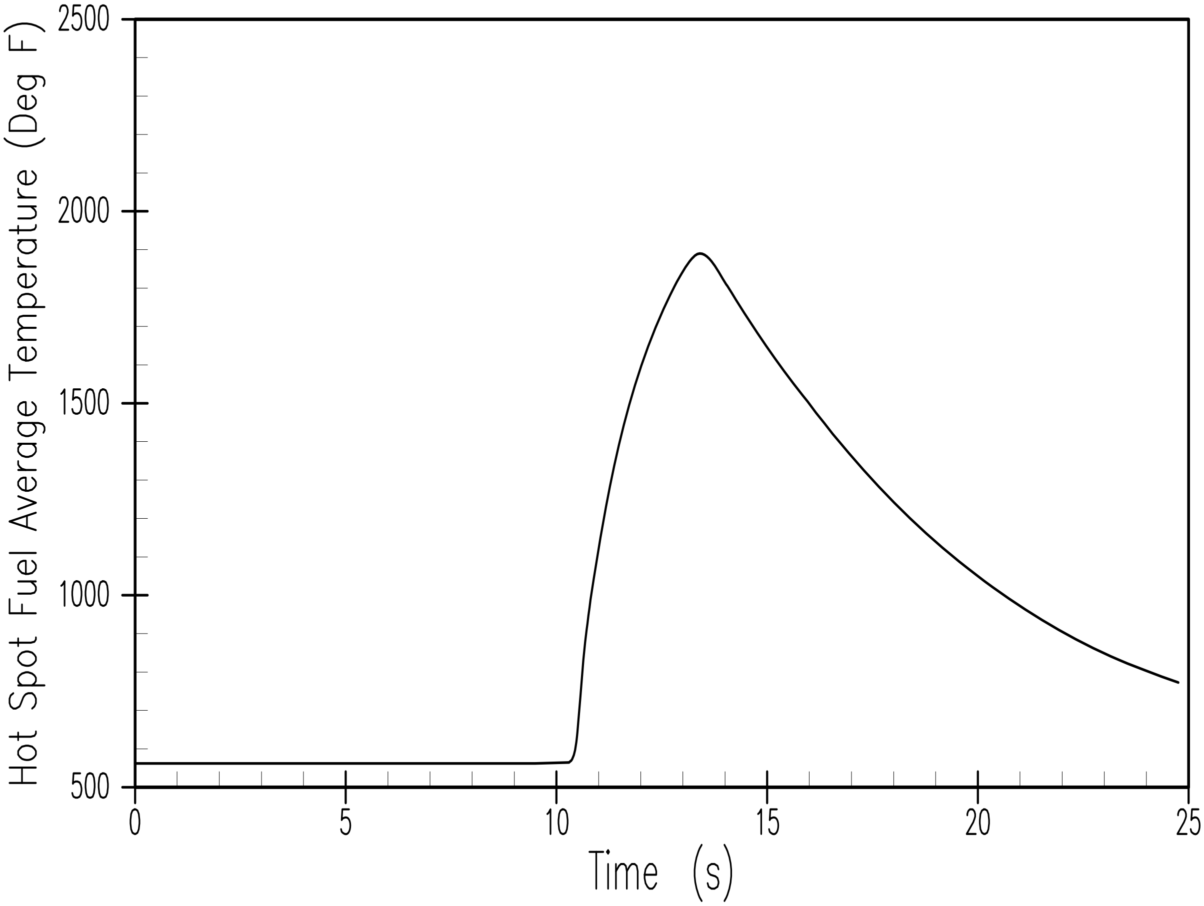


Figure 15.4.1-3 (Sheet 1 of 2)
RCCA Withdrawal from Subcritical
Hot Spot Fuel Average Temperature

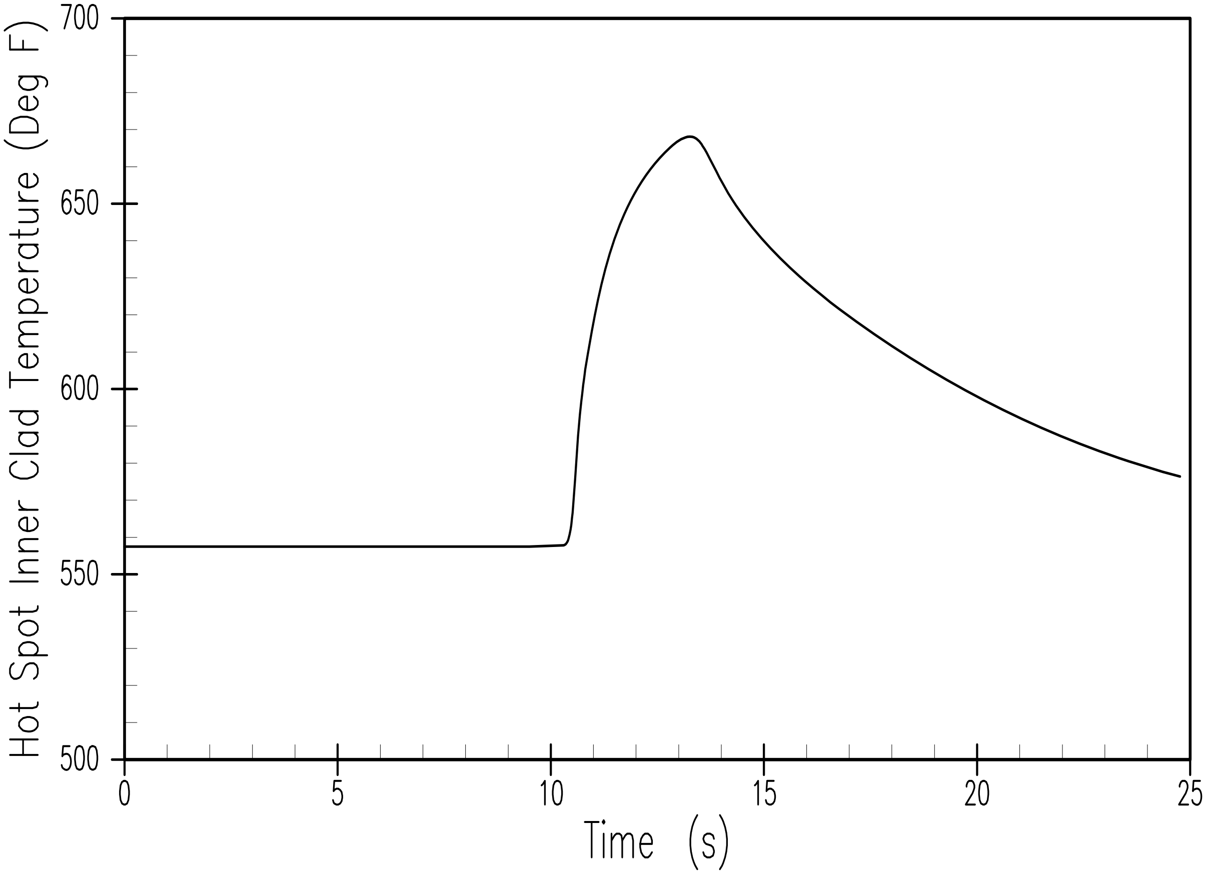


Figure 15.4.1-3 (Sheet 2 of 2)
RCCA Withdrawal from Subcritical
Hot Spot Cladding Inner Temperature

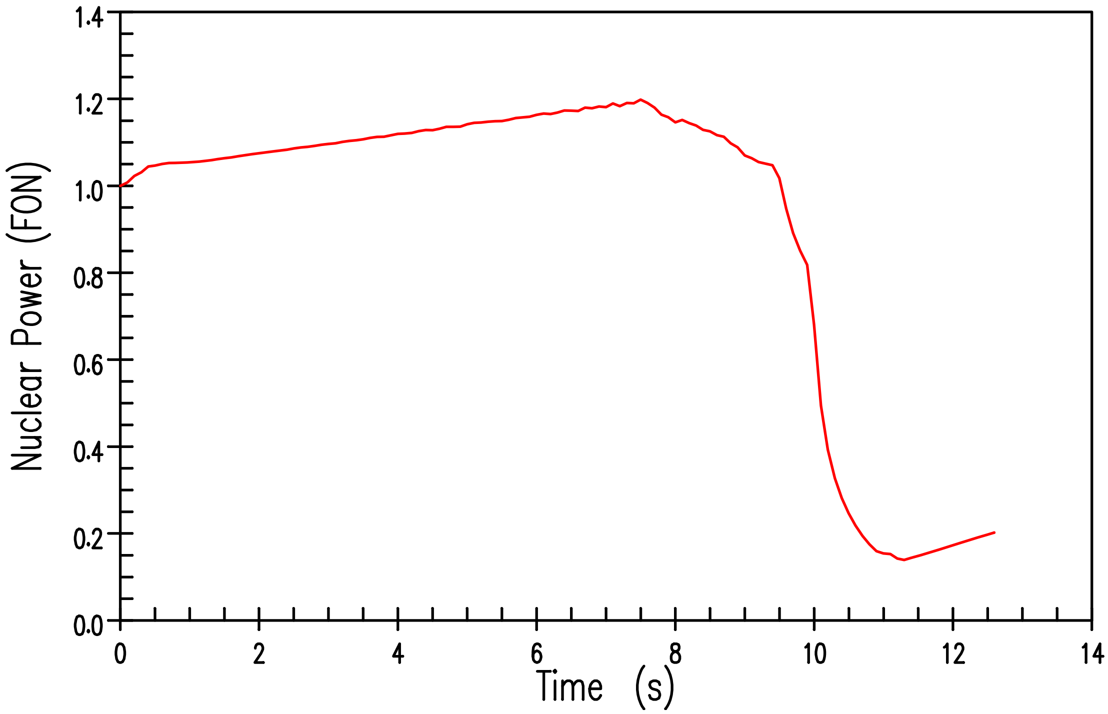


Figure 15.4.2-1
Nuclear Power Transient for an
Uncontrolled RCCA Bank Withdrawal from Full Power
With Maximum Reactivity Feedback (75 pcm/s)

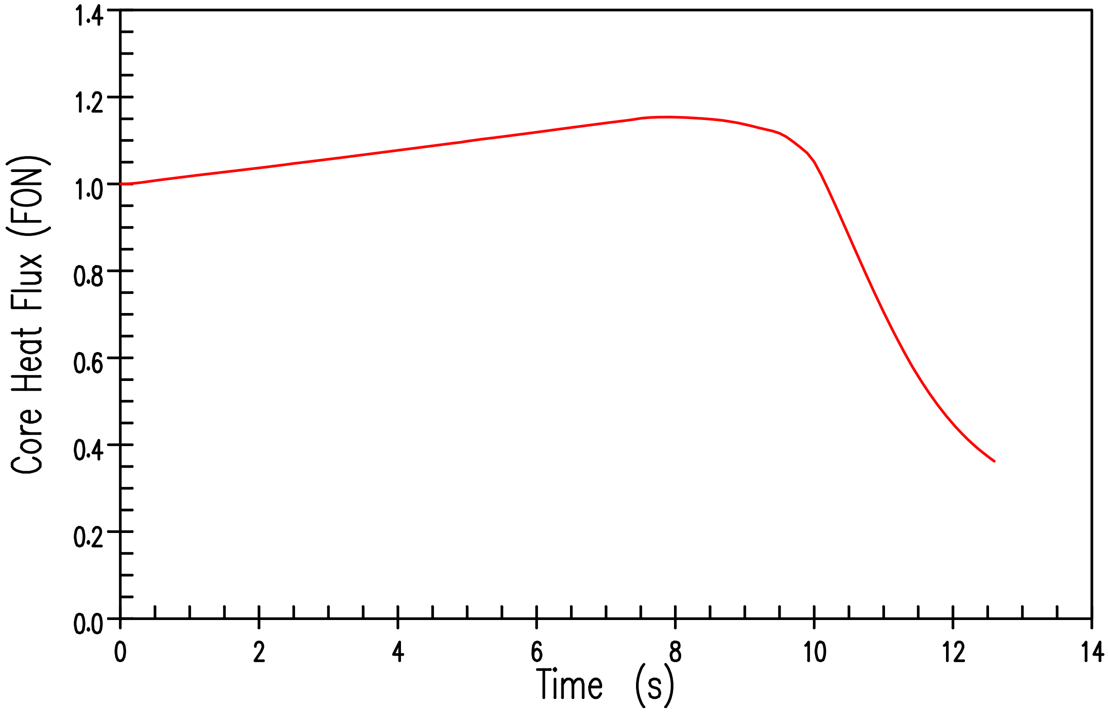


Figure 15.4.2-2
Thermal Flux Transient for an
Uncontrolled RCCA Bank Withdrawal from Full Power
With Maximum Reactivity Feedback (75 pcm/s)

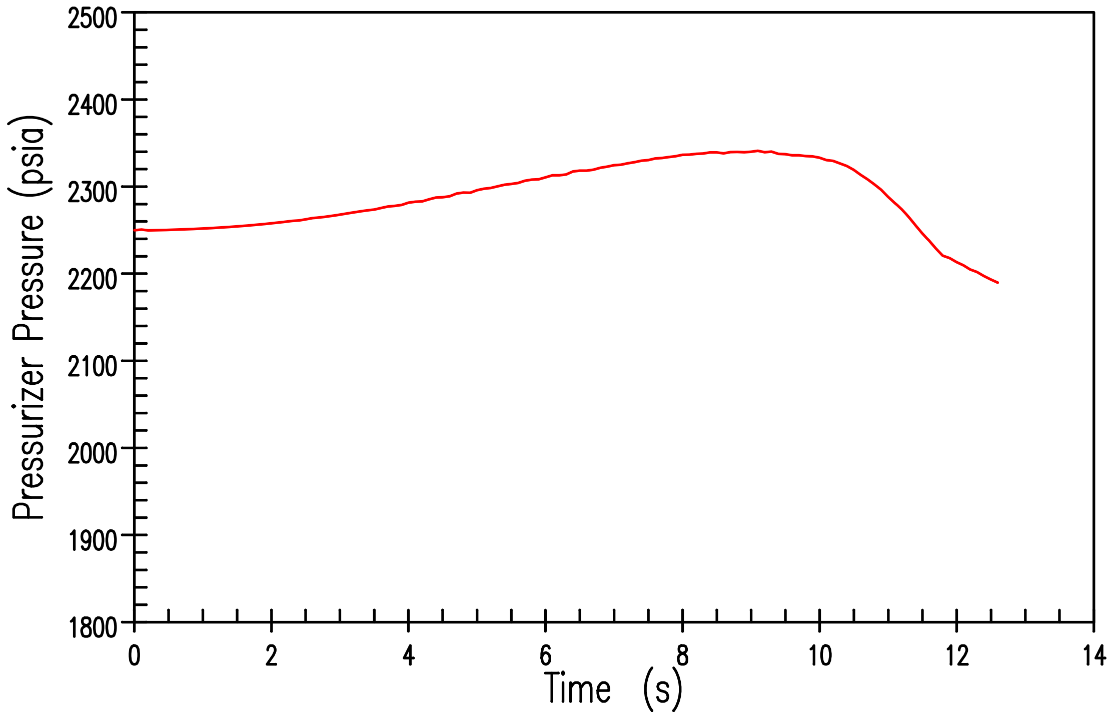


Figure 15.4.2-3
Pressurizer Pressure Transient for an
Uncontrolled RCCA Bank Withdrawal from Full Power
With Maximum Reactivity Feedback (75 pcm/s)

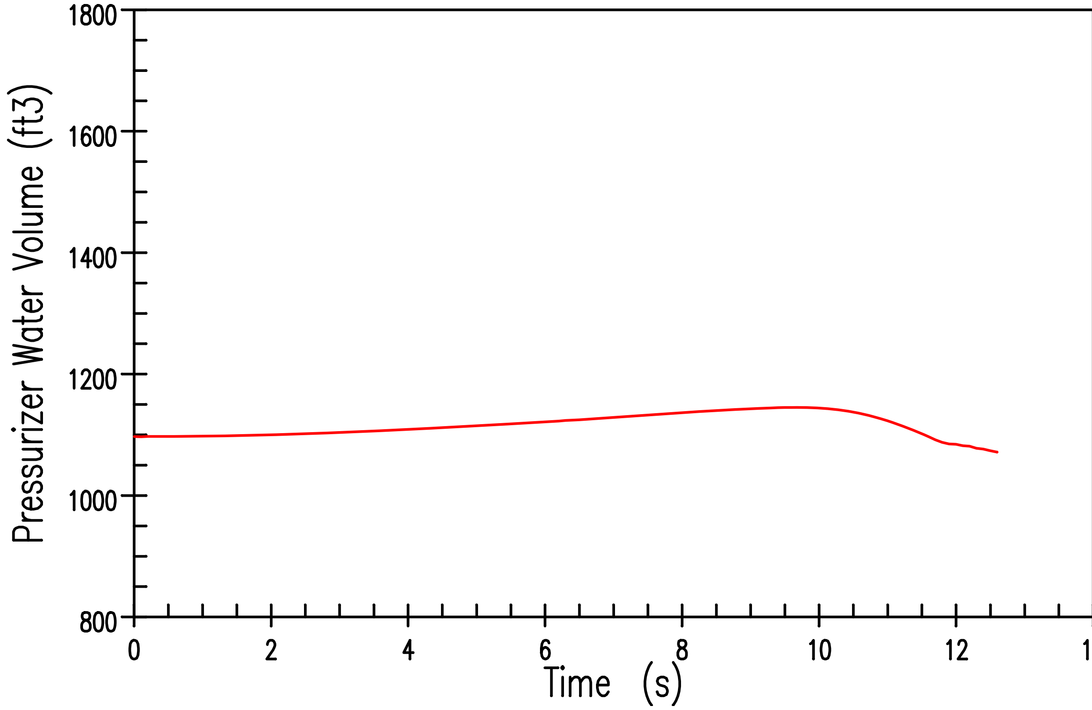


Figure 15.4.2-4
Pressurizer Water Volume Transient for an Uncontrolled RCCA Bank Withdrawal from Full Power With Maximum Reactivity Feedback (75 pcm/s)

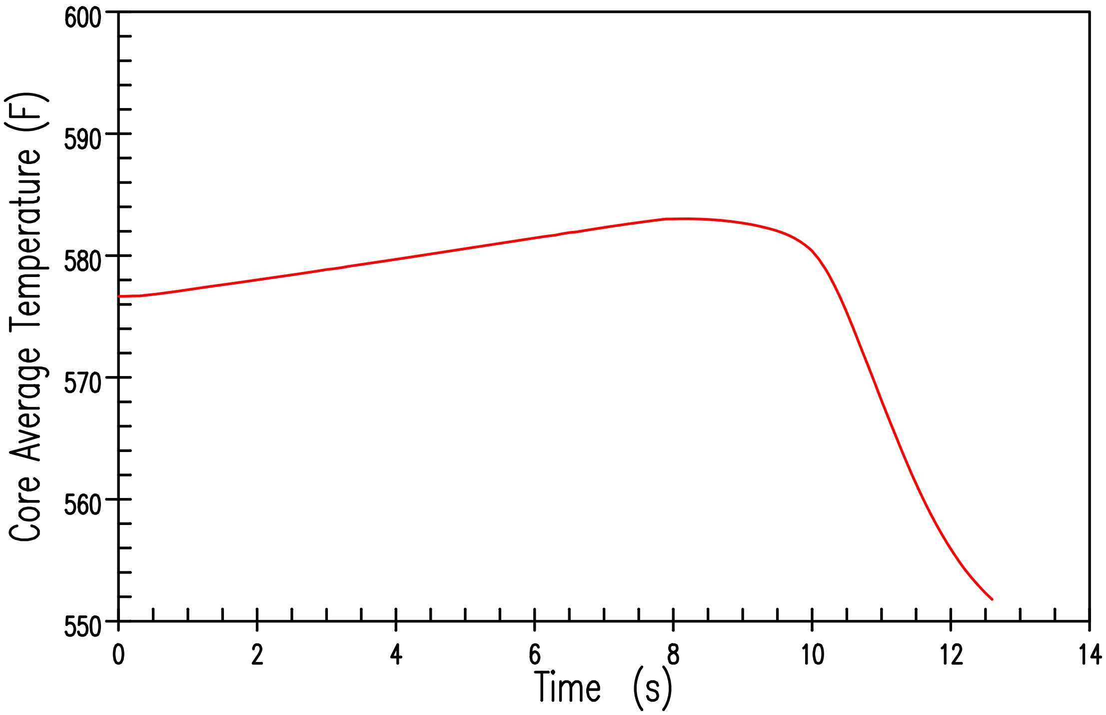


Figure 15.4.2-5
Core Coolant Average Temperature Transient for an Uncontrolled RCCA Bank Withdrawal from Full Power With Maximum Reactivity Feedback (75 pcm/s)

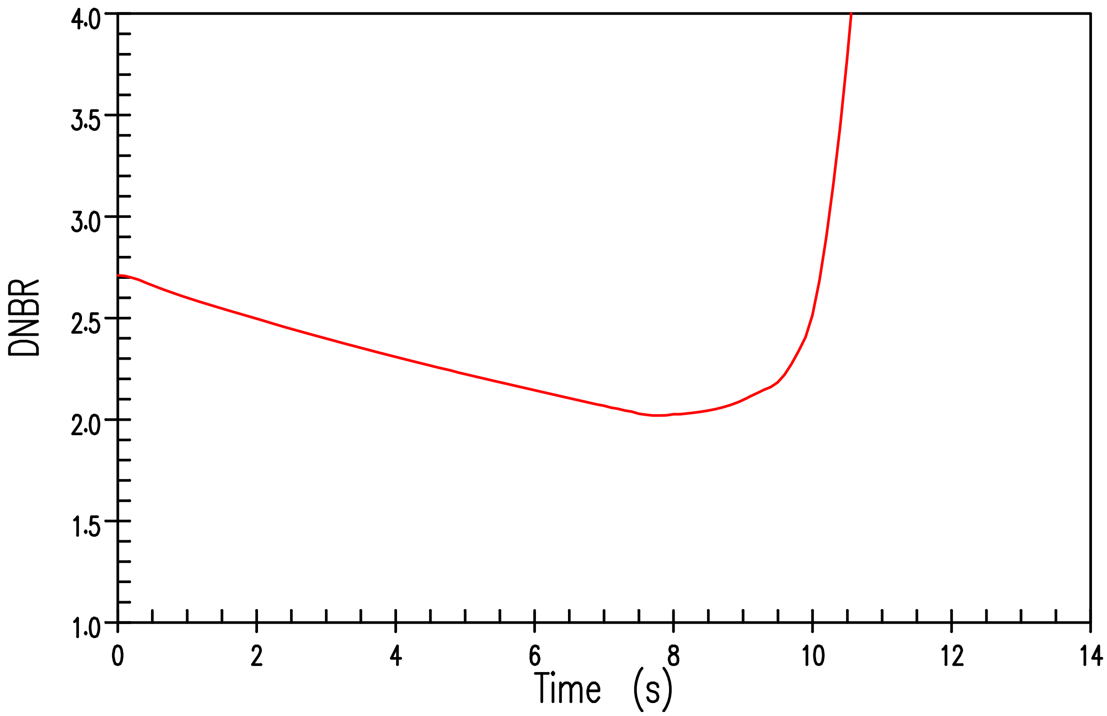


Figure 15.4.2-6
DNBR Transient for an
Uncontrolled RCCA Bank Withdrawal from Full Power
With Maximum Reactivity Feedback (75 pcm/s)

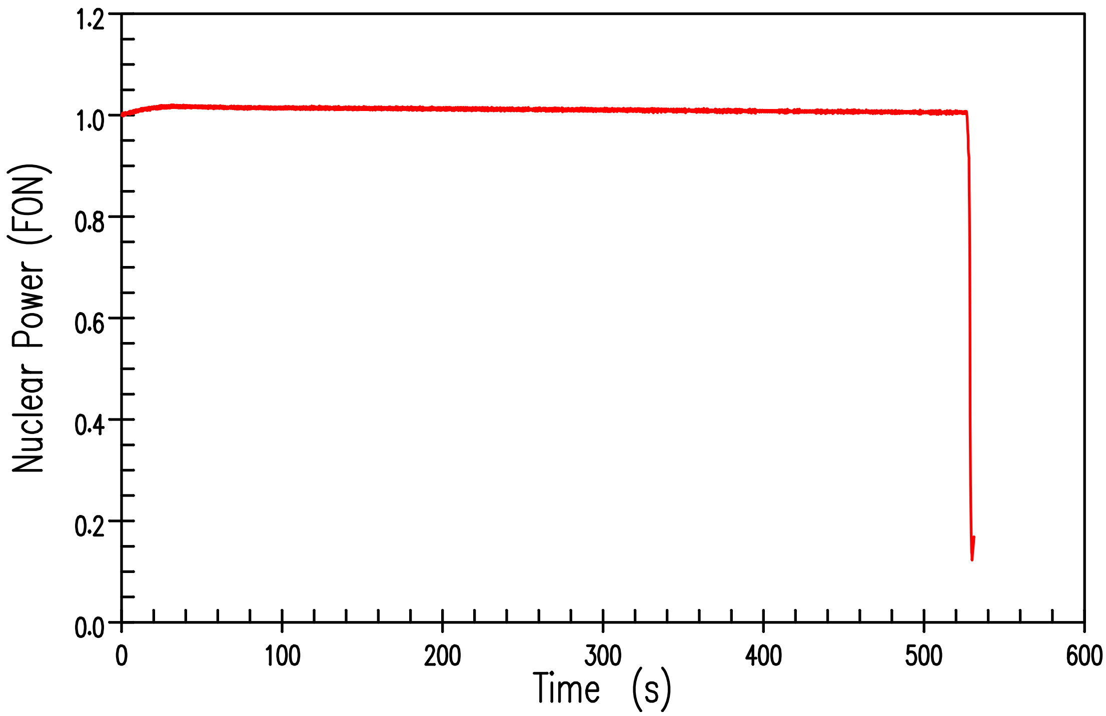


Figure 15.4.2-7
Nuclear Power Transient for an
Uncontrolled RCCA Bank Withdrawal from Full Power
With Maximum Reactivity Feedback (3 pcm/s)

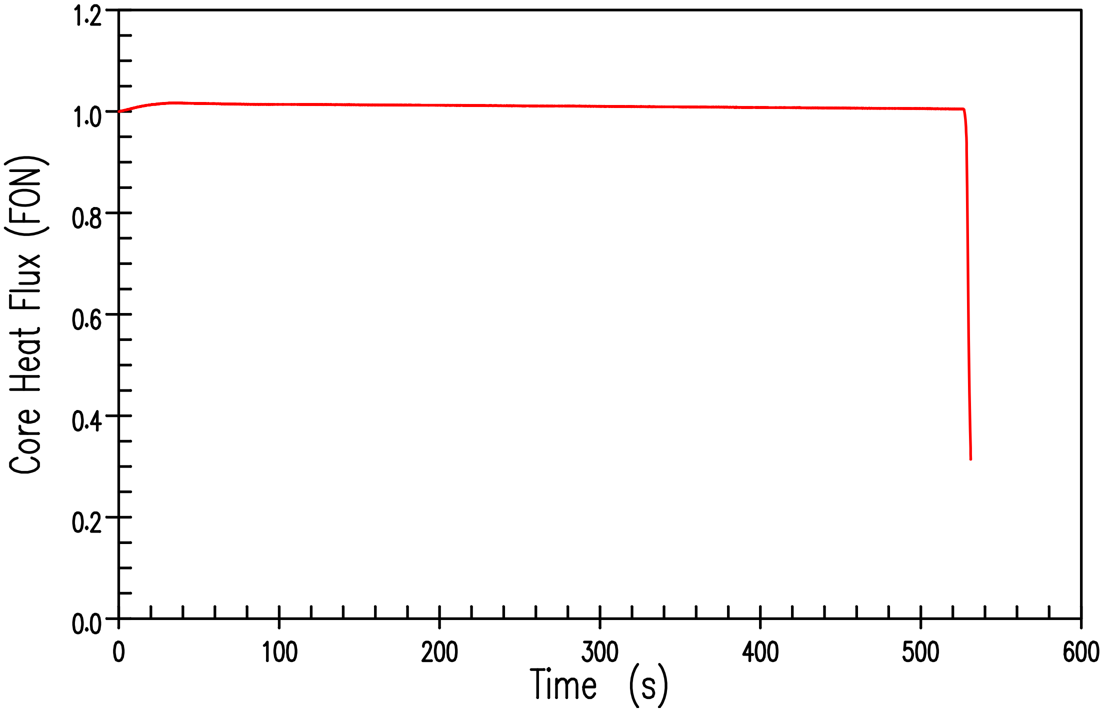


Figure 15.4.2-8
Thermal Flux Transient for an
Uncontrolled RCCA Bank Withdrawal from Full Power
With Maximum Reactivity Feedback (3 pcm/s)

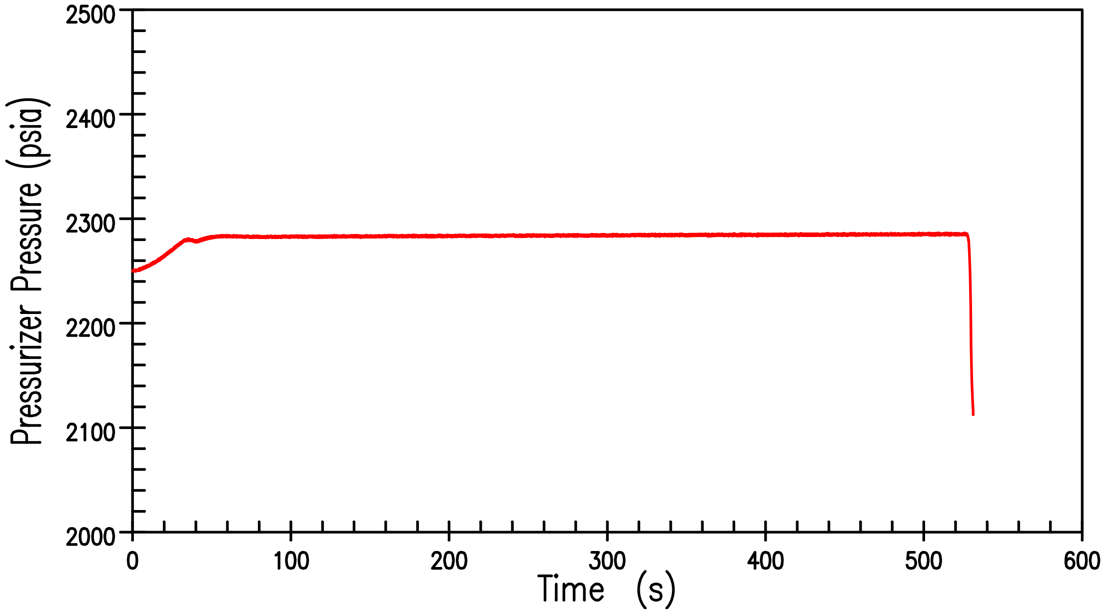


Figure 15.4.2-9
Pressurizer Pressure Transient for an Uncontrolled RCCA Bank Withdrawal from Full Power With Maximum Reactivity Feedback (3 pcm/s)

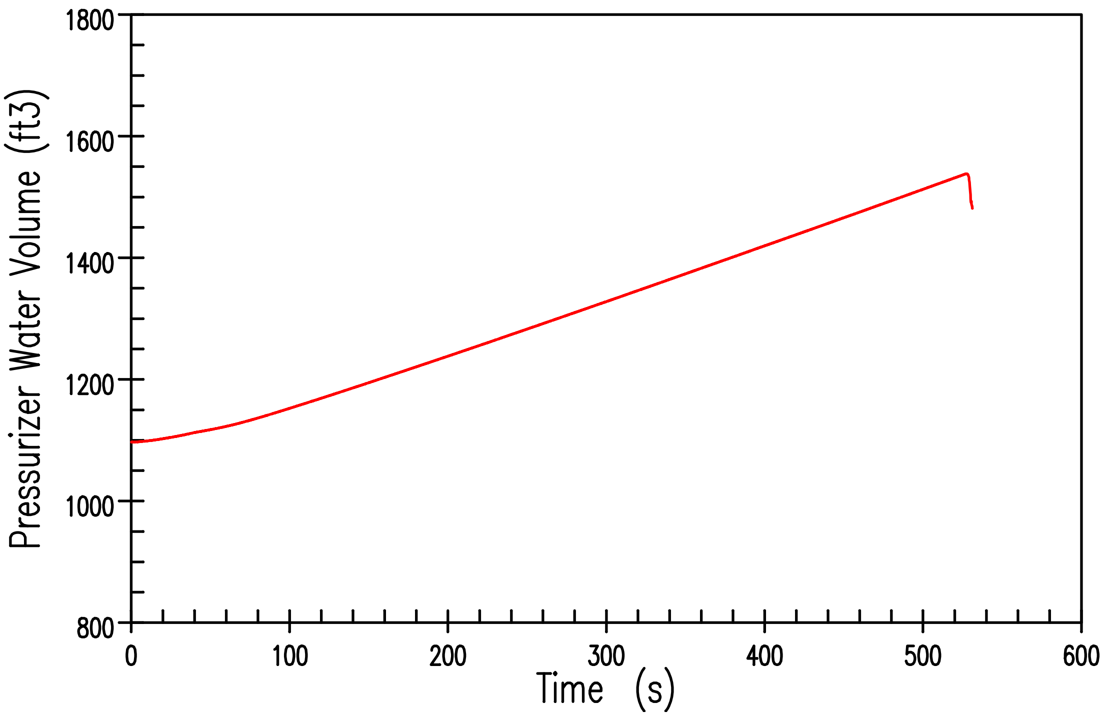


Figure 15.4.2-10
Pressurizer Water Volume Transient for an Uncontrolled RCCA Bank Withdrawal from Full Power With Maximum Reactivity Feedback (3 pcm/s)

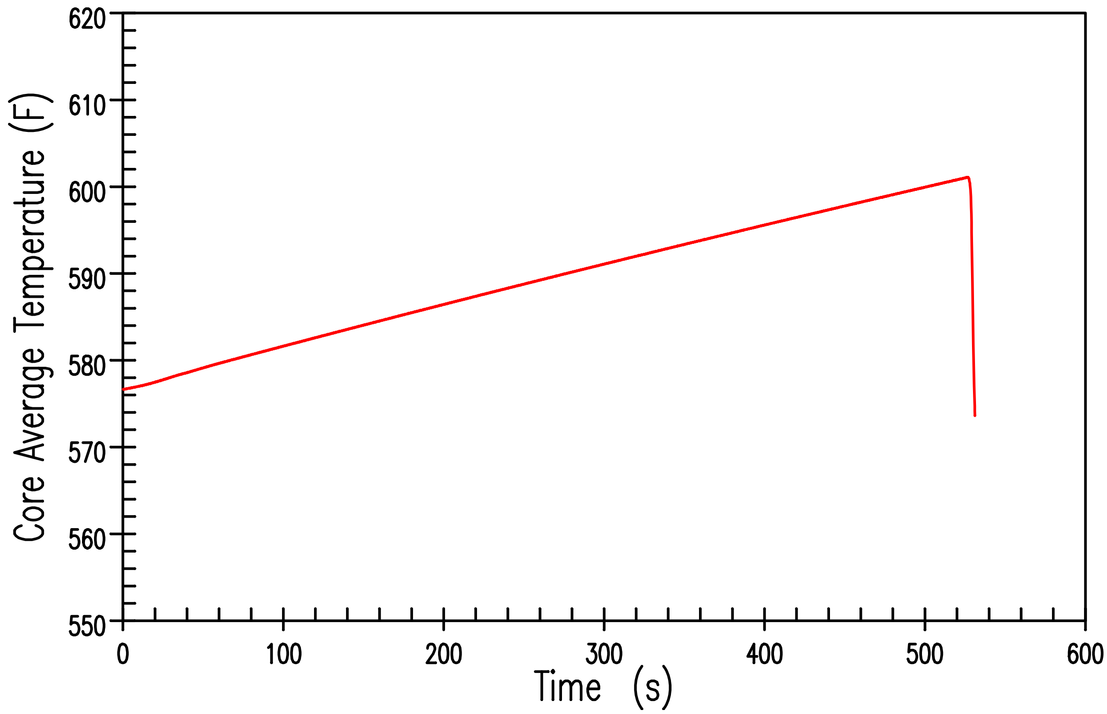


Figure 15.4.2-11
Core Coolant Average Temperature Transient for an Uncontrolled RCCA Bank Withdrawal from Full Power With Maximum Reactivity Feedback (3 pcm/s)

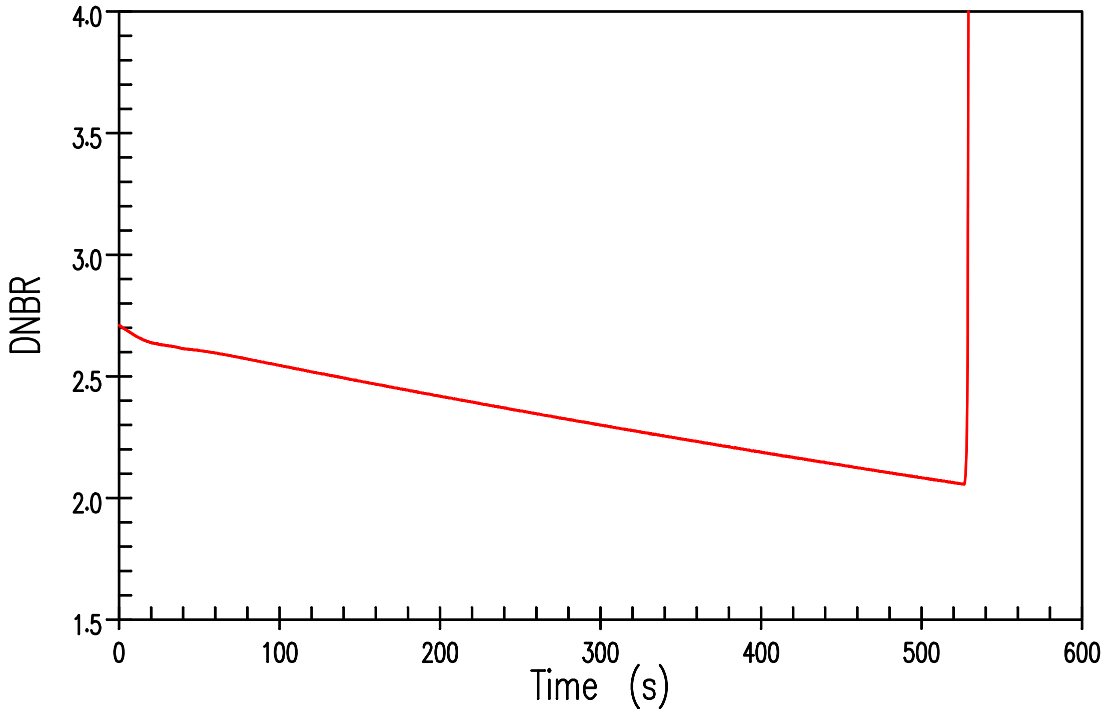


Figure 15.4.2-12
DNBR Transient for an
Uncontrolled RCCA Bank Withdrawal from Full Power
With Maximum Reactivity Feedback (3 pcm/s)

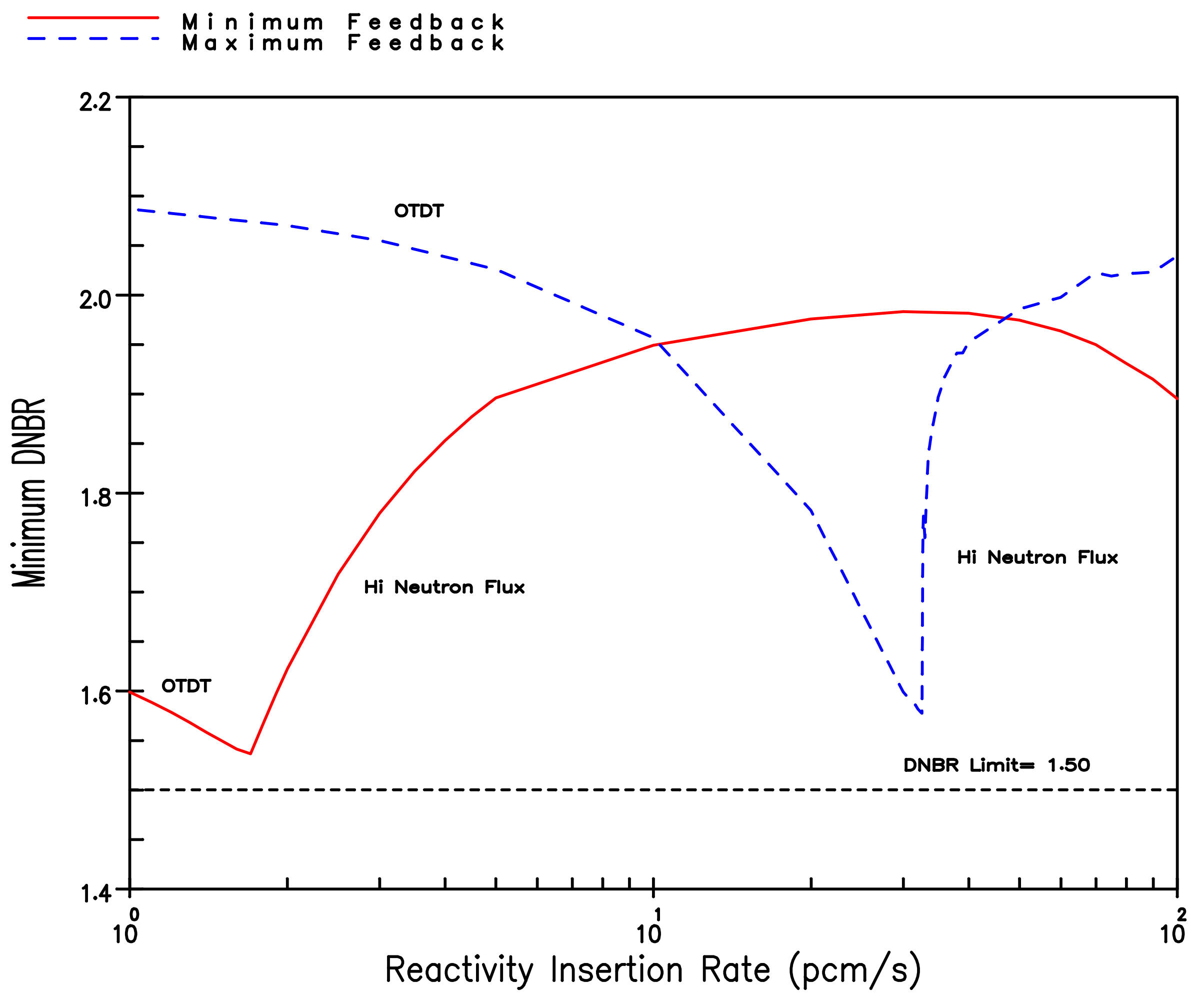


Figure 15.4.2-13
Minimum DNBR Versus Reactivity Insertion Rate for Rod Withdrawal at 100-percent Power

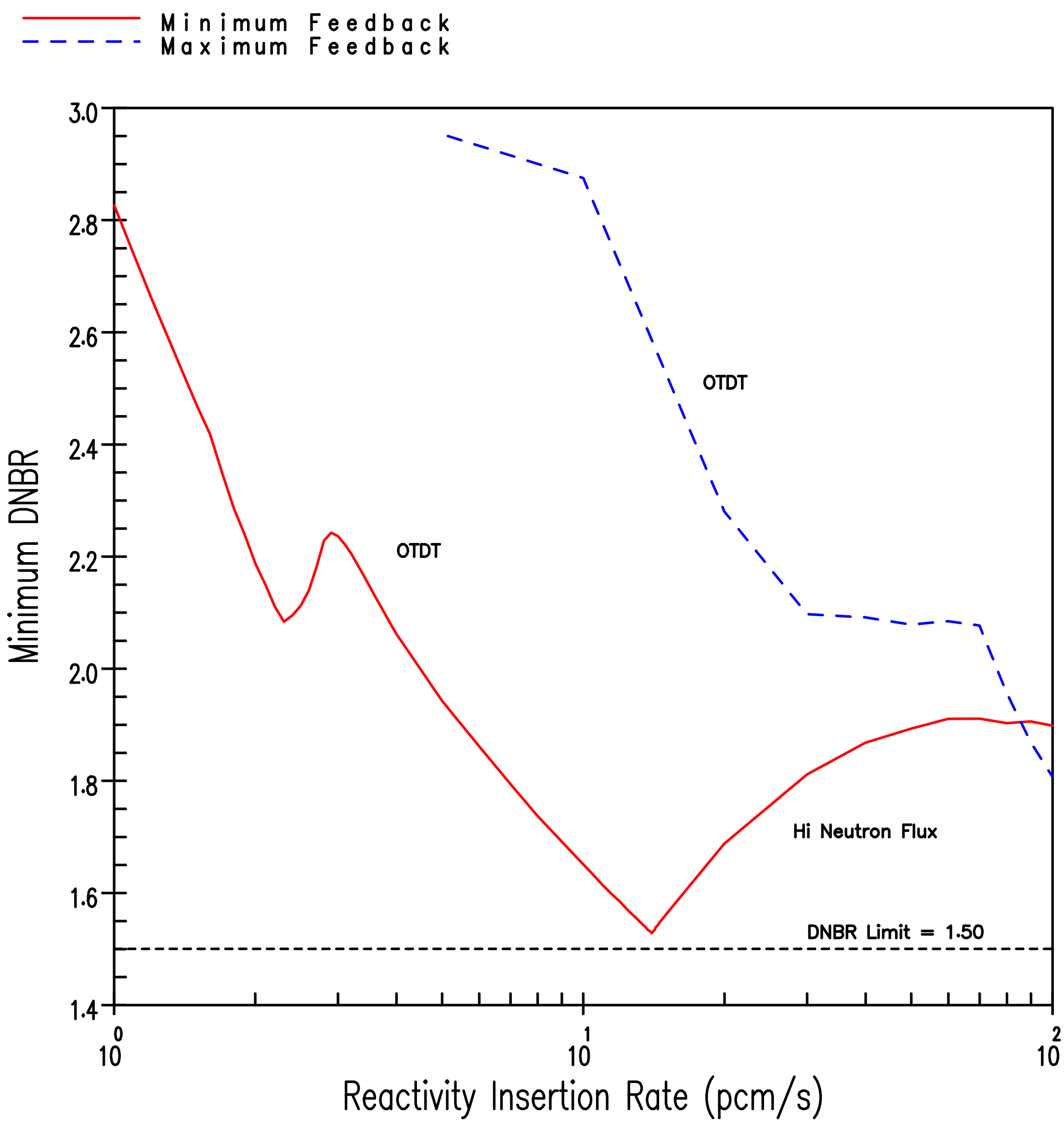


Figure 15.4.2-14
Minimum DNBR Versus Reactivity Insertion Rate for Rod Withdrawal at 60-percent Power

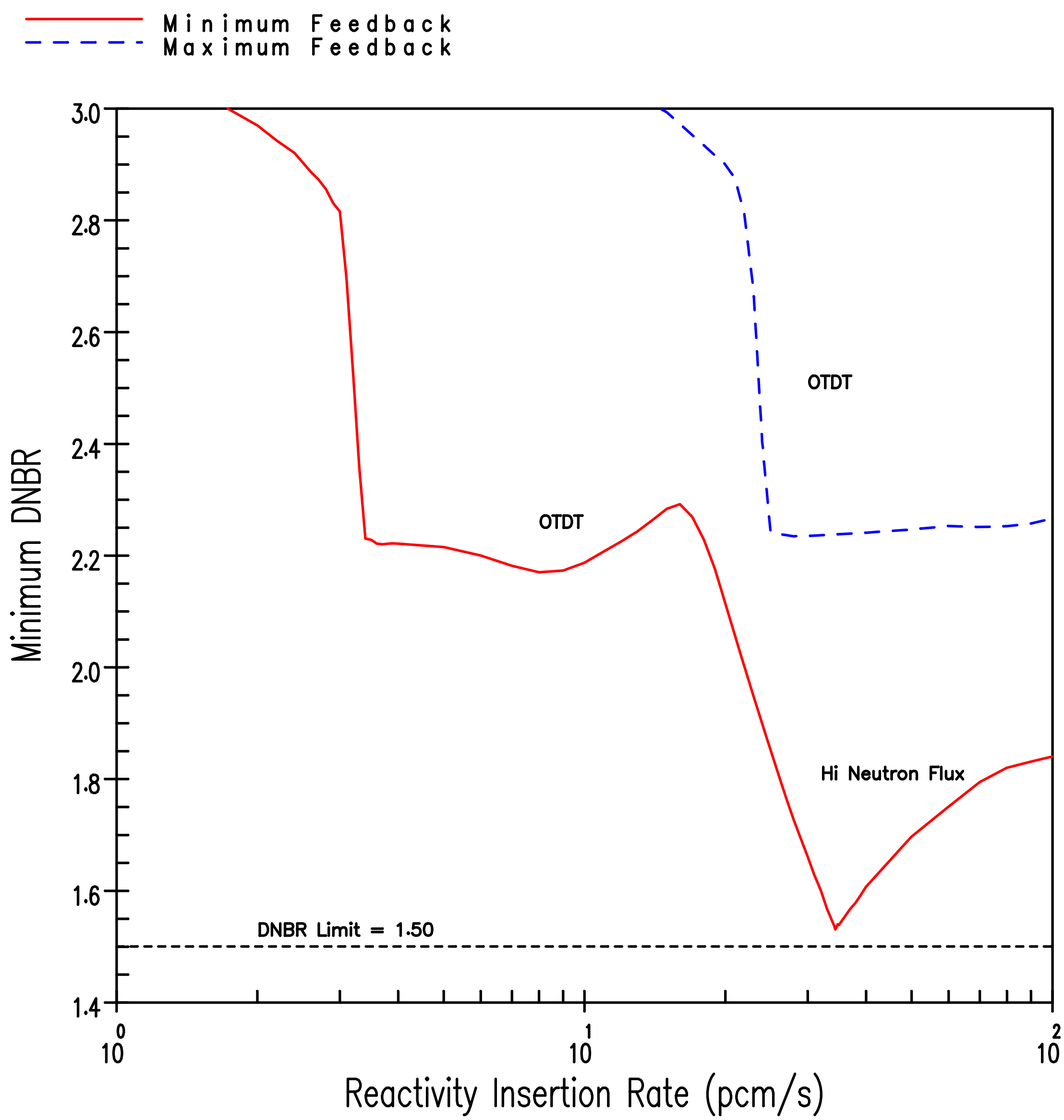


Figure 15.4.2-15
Minimum DNBR Versus Reactivity Insertion Rate for Rod Withdrawal at 10-percent Power

Figures 15.4.2-16–15.4.2-17 Not Used

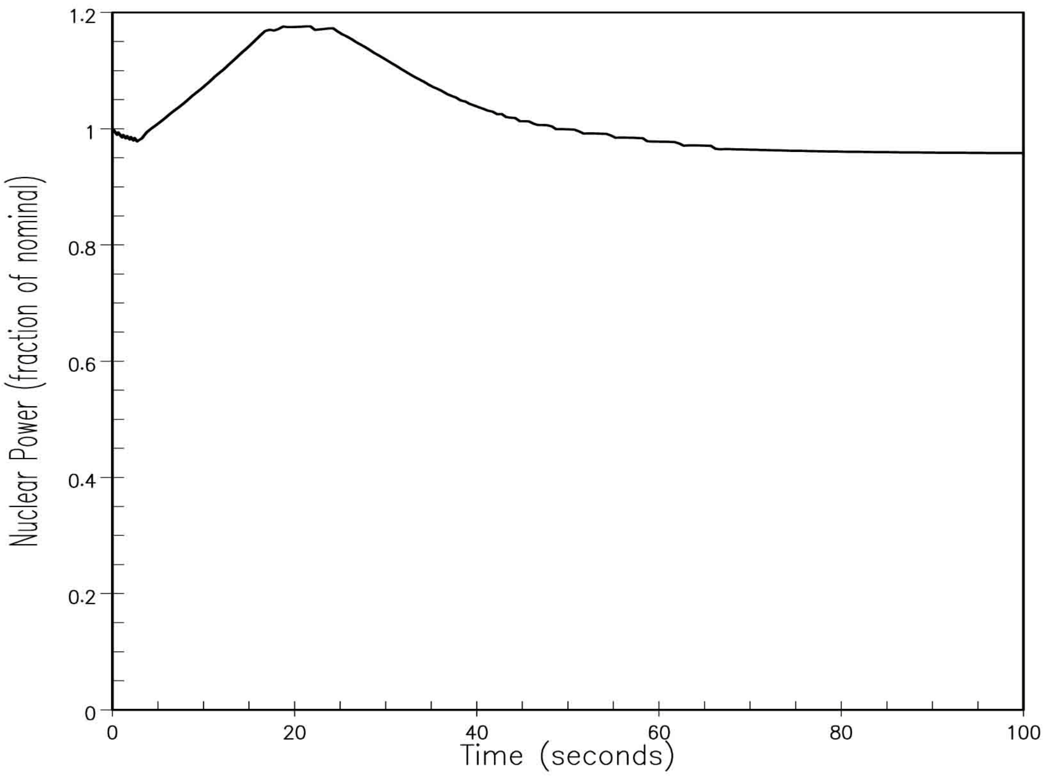


Figure 15.4.3-1
Nuclear Power Transient for Dropped RCCA

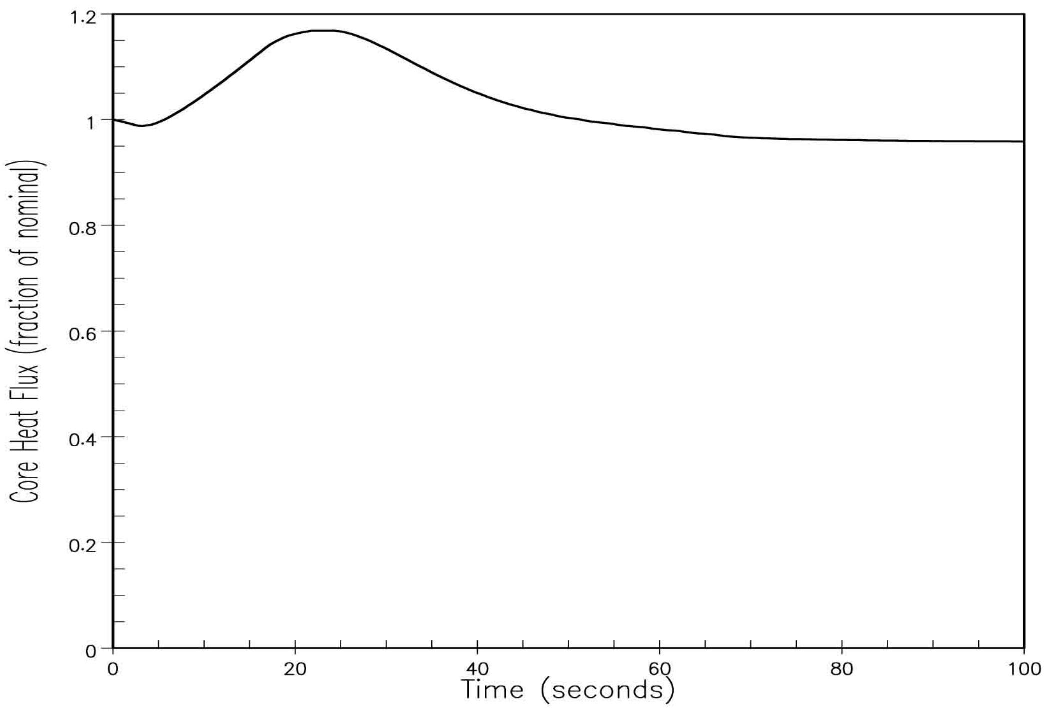


Figure 15.4.3-2
Core Heat Flux Transient for Dropped RCCA

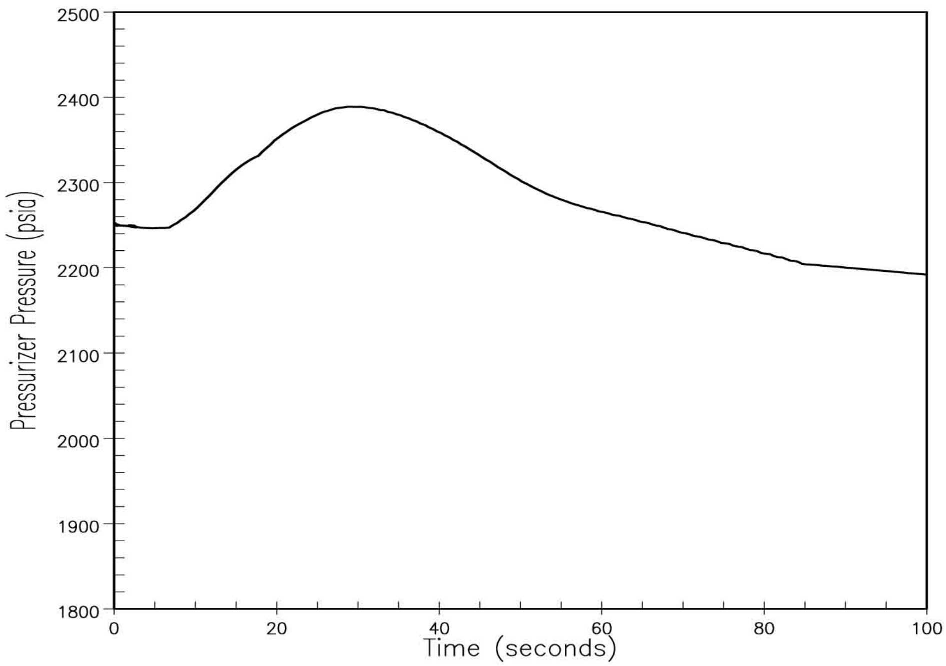


Figure 15.4.3-3
Pressurizer Pressure Transient for Dropped RCCA

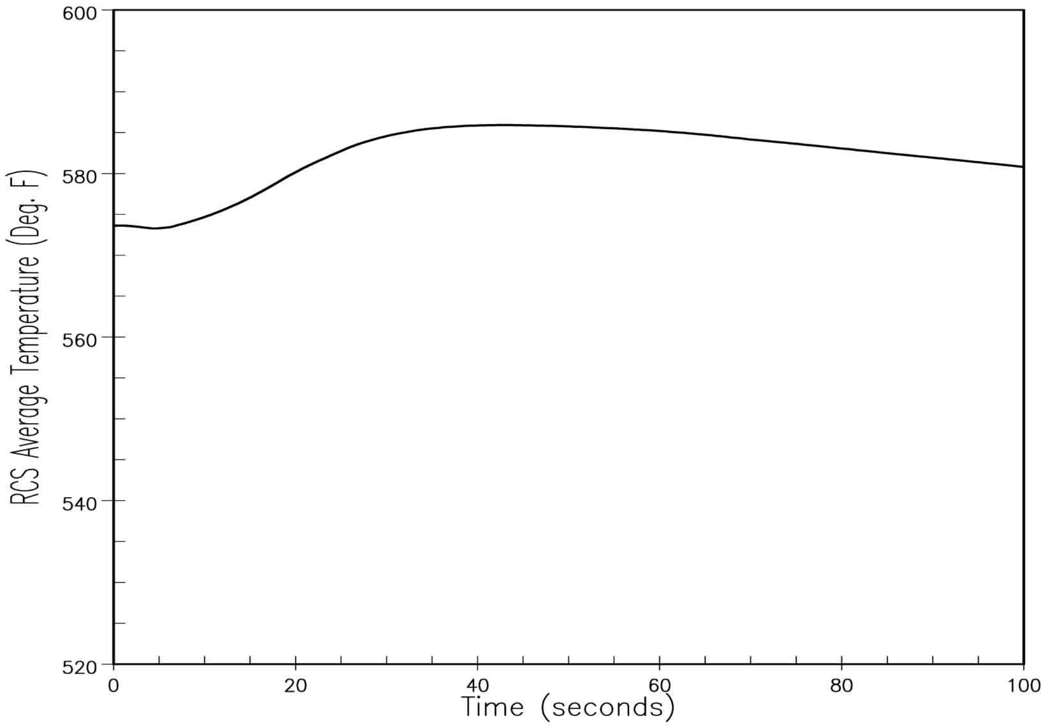


Figure 15.4.3-4
RCS Average Temperature Transient for Dropped RCCA

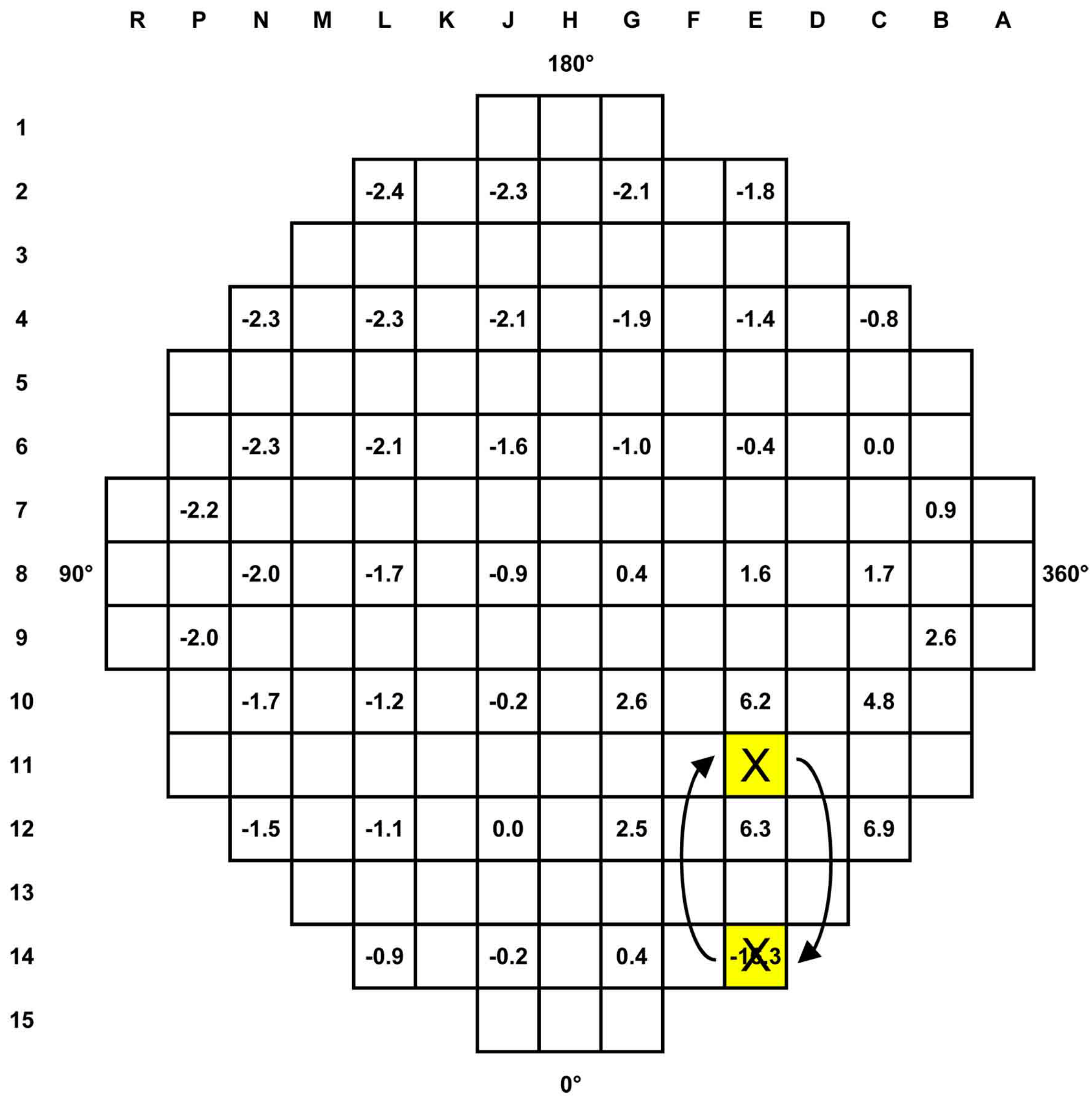


Figure 15.4.7-1
Representative Percent Change in Local Assembly Average Power
for Interchange Between Region 1 and Region 3 Assembly

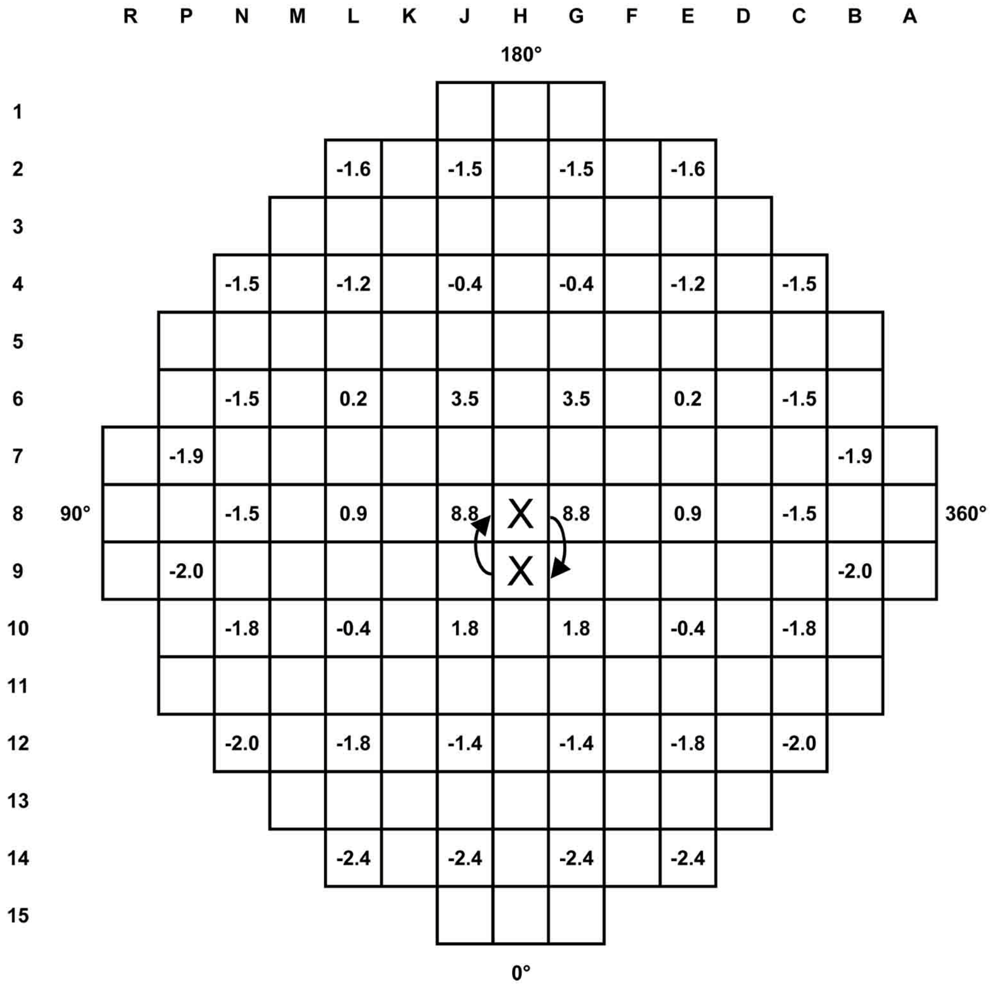


Figure 15.4.7-2
Representative Percent Change in Local Assembly Average Power for Interchange Between Region 1 and Region 2 Assembly with the BP Rods Transferred to Region 1 Assembly

180°

	R	P	N	M	L	K	J	H	G	F	E	D	C	B	A
1															
2					-2.2		-2.2		-2.2		-2.2				
3															
4			-2.1		-1.7		-1.1		-1.1		-1.7		-2.1		
5															
6			-2.0		-0.3		3.2		3.2		-0.3		-2.0		
7		-2.2												-2.2	
8			-1.9		0.8		9.9	X	9.9		0.8		-1.9		
9		-2.2												-2.2	
10			-2.0		-0.3		3.2		3.2		-0.3		-2.0		
11															
12			-2.1		-1.7		-1.1		-1.1		-1.7		-2.1		
13															
14					-2.2		-2.2		-2.2		-2.2				
15															

90° (left, row 8) 360° (right, row 8)

0°

Figure 15.4.7-3
Representative Percent Change in Local Assembly Average Power for Enrichment Error (Region 2 Assembly Loaded into Core Central Position)

180°

		R	P	N	M	L	K	J	H	G	F	E	D	C	B	A	
1																	
2						-6.9		-6.9		-6.6		-6.0					
3																	
4				-6.7		-6.7		-6.5		-6.0		-5.0		-3.6			
5																	
6				-6.4		-6.2		-5.5		-4.2		-2.8		-1.4			
7			-6.1												1.1		
8	90°			-5.8		-5.2		-3.7		-1.3		1.6		3.2			360°
9			-5.6												6.9		
10				-5.0		-3.8		-1.4		3.3		11.1		13.8			
11																	
12				-4.1		-2.4		0.6		7.4		26.3	X	39.7			
13																	
14						-1.1		2.0		7.5		20.5					
15																	

0°

Figure 15.4.7-4
Representative Percent Change in Local Assembly Average Power for Loading Region 2 Assembly into Region 1 Position Near Core Periphery

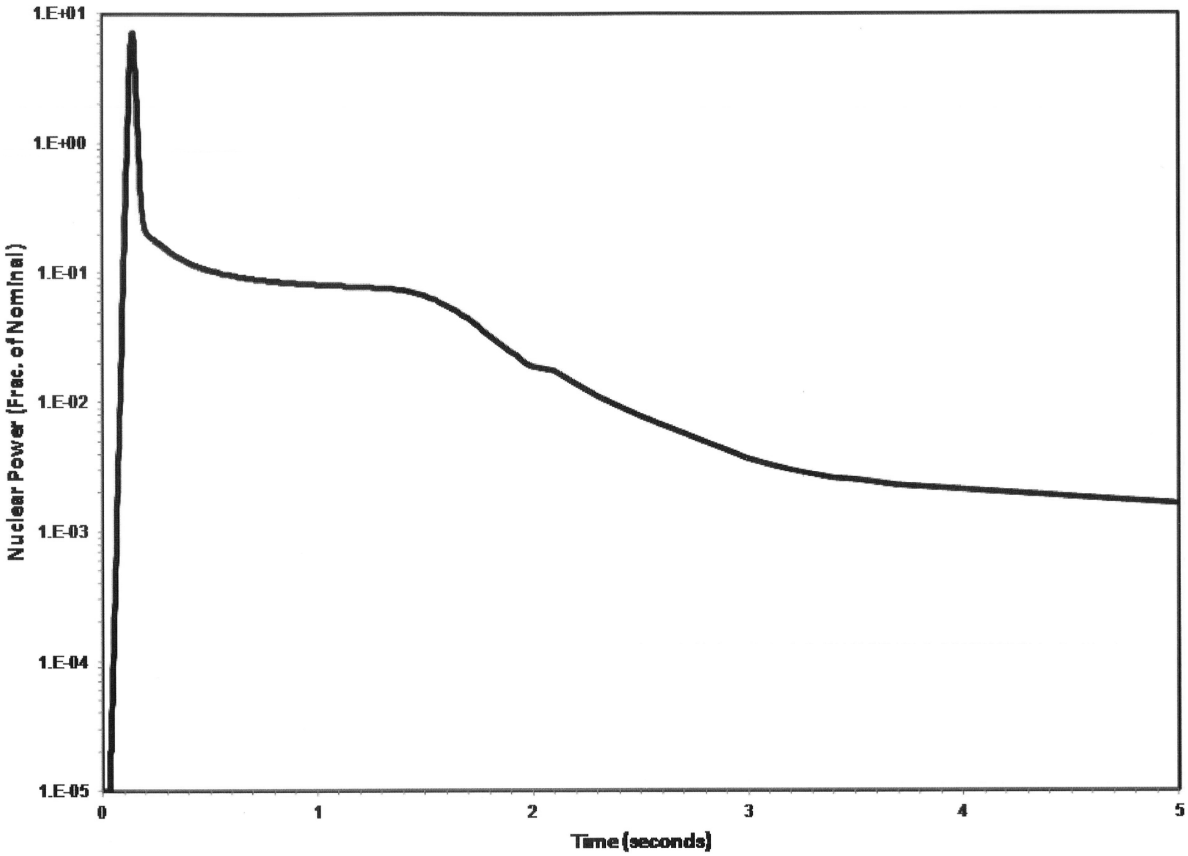


Figure 15.4.8-1
Nuclear Power Transient Versus Time for the PCMI Rod Ejection Accident

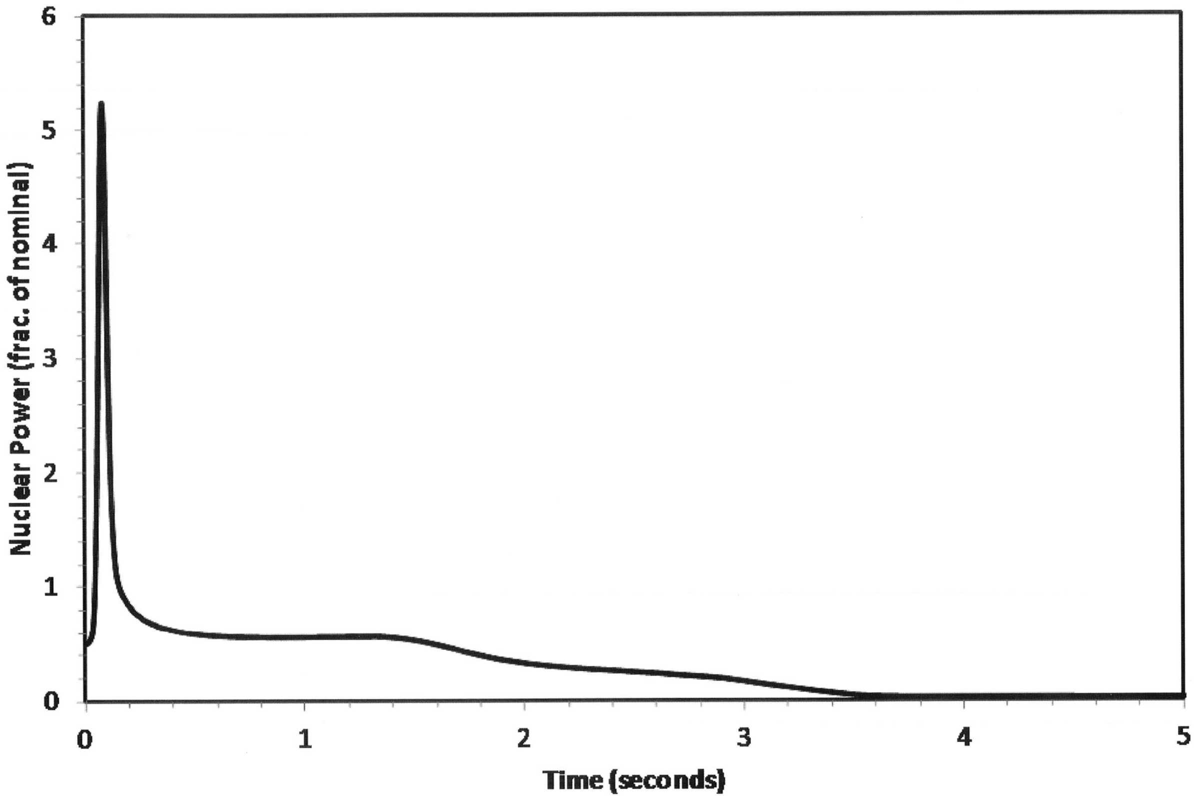


Figure 15.4.8-2
Nuclear Power Transient Versus Time for the High Cladding Temperature Rod Ejection Accident

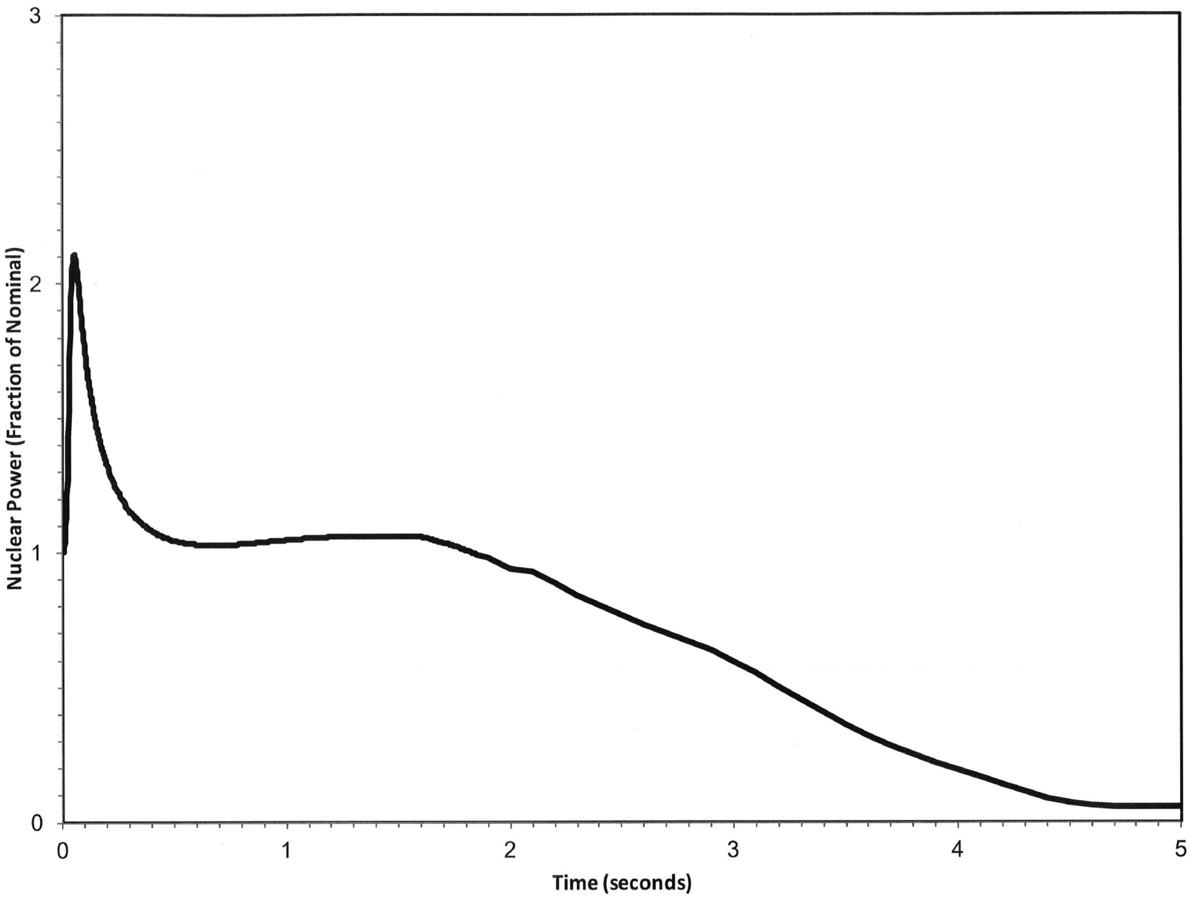


Figure 15.4.8-3
Nuclear Power Transient Versus Time for the Peak Enthalpy and Fuel Centerline Temperature Rod Ejection Accident

Figure 15.4.8-4 Not Used

15.5 Increase in Reactor Coolant Inventory

This section presents a discussion and analysis of the following events:

- Inadvertent operation of the core makeup tanks during power operation
- Chemical and volume control system malfunction that increases reactor coolant inventory

These Condition II events cause an increase in reactor coolant inventory.

15.5.1 Inadvertent Operation of the Core Makeup Tanks During Power Operation

15.5.1.1 Identification of the Causes and Accident Description

Spurious core makeup tank operation at power could be caused by an operator error, a false electrical actuation signal, or a valve malfunction. A spurious signal may originate from any of the safeguards (“S”) actuation channels as described in Section 7.3. The AP1000 protection logic is such that a single failure cannot actuate both core makeup tanks without also actuating the passive residual heat removal (PRHR) heat exchanger. A scenario such as this is the spurious “S” signal event. However, if one core makeup tank is inadvertently actuated by a single failure, the event may progress with the plant at power until a reactor trip is reached. For the plant under automatic rod control, a reactor trip on high-3 pressurizer water level reactor trip is expected to occur followed by the PRHR actuation and eventually by an “S” signal, which would then actuate the second core makeup tank. When a consequential loss of offsite power is assumed, this event is more conservative than the spurious “S” signal event.

The inadvertent opening of the core makeup tank discharge valves, due to operator error or valve failure, results in significant core makeup tank injection flow leading to a boration similar to that resulting from a chemical and volume control system malfunction event. If the automatic rod control system is operable, it will begin to withdraw rods from the core to counteract the reactivity effects of the boration. As a result, the core makeup tank will continue injection and slowly raise the pressurizer level until the high-3 pressurizer level trip setpoint is reached. In meeting the requirements of GDC 17 of 10 CFR Part 50, Appendix A, a loss of offsite power is assumed to occur as a consequence of reactor trip. The primary effect of this assumption is the coastdown of the reactor coolant pumps. The core makeup tank injection will increase as the steam generator outlet temperature increases resulting in a lower density in the CMT balance line. This event will then proceed similarly to a spurious “S” signal or chemical and volume control system malfunction event. However, this event is more limiting primarily due to the higher pressurizer level at the time of reactor trip and to the significant heat up of the injected fluid during the pre-trip phase of the accident. Thus, the inadvertent core makeup tank actuation event with a consequential loss of offsite power is analyzed here.

Upon receipt of the high-3 pressurizer level reactor trip signal, the reactor is tripped; then the turbine is immediately tripped, and after a 3-second delay, a consequential loss of offsite power is assumed. The basis for the 3-second delay is described in Subsection 15.0.14. The high-3 pressurizer level signal also actuates the PRHR heat exchanger and blocks the pressurizer heaters, but a 15-second delay is built in to prevent unnecessary actuation of the PRHR heat exchanger if offsite power is maintained.

Following reactor trip, the reactor power drops and the average reactor coolant system temperature decreases with subsequent coolant shrinkage. However, due to the assumed loss of offsite power, the reactor coolant cold leg temperature, in the loop without PRHR, increases and the core makeup tank starts injecting cold water into the reactor coolant system at a much higher rate. The primary coolant system shrinkage is counteracted by the core makeup tank injection, and the pressurizer water volume starts to increase because of the heatup of the cold injected fluid by the decay heat.

The high-3 pressurizer level setpoint is once again reached, and after a 15-second delay, the signal is sent to actuate the PRHR heat exchanger and block the pressurizer heaters.

Eventually, the core makeup tank heats up and the gravity-driven recirculation is significantly reduced. The PRHR heat exchanger continues to extract heat from the reactor coolant system, and the pressurizer water volume starts to decrease. Ultimately, the core makeup tank stops recirculating, the PRHR heat removal matches decay heat and the reactor coolant system cooldown begins eventually leading to a “S” signal on a Low T_{cold} setpoint.

The cold injection flow from the second CMT initially results in a fast decrease in temperature and shrinkage of the reactor coolant. However, as the temperature decreases, the PRHR heat removal capability diminishes and a moderate heat up occurs followed by the increase of pressurizer water level. The second CMT injection rate is much lower than that experienced during the first part of the transient from the first CMT. Due to the colder cold leg temperatures, the density in balance line is much higher than during the first part of the transient, resulting in a reduction of the total buoyancy driving head. Ultimately, the PRHR heat removal once again matches the decay heat and the final reactor coolant system cooldown begins.

This event is a Condition II incident (a fault of moderate frequency) as defined in Subsection 15.0.1.

15.5.1.2 Analysis of Effects and Consequences

The plant response to an inadvertent core makeup tank actuation is analyzed by using a modified version of the computer program LOFTRAN described in Subsection 15.0.11.2. The code simulates the neutron kinetics, reactor coolant system, pressurizer, pressurizer safety valves, pressurizer spray, steam generator, steam generator safety valves, PRHR heat exchanger, and core makeup tank. The program computes pertinent plant variables, including temperatures, pressures, and power level.

Reactor power and average temperature drop immediately following the trip, and the operating conditions never approach the core limits. The PRHR heat exchanger removes the long-term decay heat and prevents possible reactor coolant system overpressurization or loss of reactor coolant system water.

Core makeup tank and PRHR system performance is conservatively simulated. Core makeup tank enthalpies have been maximized. This is conservative because it minimizes the cooling provided by the core makeup tanks as flow recirculates and thereby increases the peak pressurizer water volume during the transient. Core makeup tank injection and balance lines pressure drop is minimized. This maximizes the core makeup tank flow injected in the primary system. During this event, the core makeup tanks remain filled with water. The volume of injection flow leaving the core makeup tanks is offset by an equal volume of recirculation flow that enters the core makeup tanks via the balance lines. PRHR heat transfer capability has been minimized.

Plant characteristics and initial conditions are further discussed in Subsection 15.0.3.

The limiting case presented here bounds cases that model explicit operator action 60 minutes after reactor trip. The assumptions for this case are as follows:

- Initial operating conditions

  The initial reactor power is assumed to be 102 percent of nominal. The main feedwater flow measurement supports a 1-percent power uncertainty; use of a 2-percent power uncertainty is conservative. The initial pressurizer pressure is assumed to be 50 psi below nominal. The initial reactor coolant system average temperature is assumed to be 7°F below nominal.

- Control systems

  The pressurizer spray system and automatic rod control system are conservatively assumed to operate. The pressurizer heaters are automatically blocked on a high-3 pressurizer level signal, so they cannot add heat to the system during the period of thermal expansion that produces the peak pressurizer water volume. Thus, the pressurizer heaters are assumed to be inoperable during this event. Other control systems are conservatively not assumed to function during the transient. Cases with the turbine bypass (steam dump) and feedwater control systems working result in lower secondary and primary temperatures and in greater margin to overfilling.

- Moderator and Doppler coefficients of reactivity

  A least-negative moderator temperature coefficient, a Low (absolute value) Doppler power coefficient, and a maximum boron worth are assumed. With these minimum feedback parameters and the operability of the pressurizer spray system and automatic rod control system assumed, the reactivity effects of the boron injection from the core makeup tanks is counteracted. As a result, the high-3 pressurizer signal is the first reactor trip signal generated during the transient.

- Boron injection

  The transient is initiated by an inadvertent opening of the discharge valves of one of the two core makeup tanks. The core makeup tank injects 3400 ppm borated water.

- Protection and safety monitoring system actuations

  Reactor trip is initiated by the high-3 pressurizer level signal.

  The core decay heat is removed by the PRHR heat exchanger. The worst single failure is assumed to occur in the outlet line of the PRHR heat exchanger. One of the two parallel isolation valves is assumed to fail to open.

Plant systems and equipment available to mitigate the effect of the accident are discussed in Subsection 15.0.8 and listed in Table 15.0-6.

15.5.1.3 Results

Figures 15.5.1-1 through 15.5.1-11 show the transient response to the inadvertent operation of one of the two core makeup tanks during power operation. The inadvertent opening of the core makeup tank discharge valves occurs at 10 seconds. As the core makeup tank continues to add inventory to the primary system, the pressurizer level begins to increase until the high-3 pressurizer level reactor trip setpoint is reached at about 520.7 seconds. After a 2-second delay, the neutron flux starts decreasing due to the reactor trip, which is immediately followed by the turbine trip. Following reactor trip, the reactor power drops and the average reactor coolant system temperature decreases with subsequent coolant shrinkage. However, due to the assumed loss of offsite power, the reactor coolant pumps trip at about 525.4 seconds. The cold leg temperature increases and the core makeup tank starts injecting cold water into the reactor coolant system at a much higher rate due to the increased driving head resulting from the density decreases in balance line. The primary coolant system shrinkage is counteracted by the core makeup tank injection, and the pressurizer water volume starts to increase because of the heatup of the cold injected fluid by the decay heat. The high-3 pressurizer level setpoint is once again reached at about 541.9 seconds, and after a 15-second delay, the signal is sent to actuate the PRHR heat exchanger and block the pressurizer heaters. Following a conservative 17-second delay, the valves are assumed to open to actuate the PRHR heat exchanger at about 573.9 seconds.

After reactor trip, the pressure in the primary and secondary systems increases initially due to the assumed unavailability of the non-safety-related control systems. The primary and secondary system pressures eventually decrease as the PRHR system removes decay heat. The core makeup tank works in recirculation mode, meaning it is always filled with water because cold borated water injected through the injection line is replaced by hot water coming from the cold leg (balance lines). At approximately 5,000 seconds, the PRHR heat flux matches the core decay heat. However, the pressurizer level continues to slowly increase until the core makeup tank recirculation is decreased sufficiently to significantly limit the mass addition to the RCS.

At 5,880 seconds, the pressurizer safety valves close. At about 6,600 seconds, the pressurizer water volume stops increasing. At about 12,354 seconds, the Low T_{cold} "S" setpoint is reached and the second CMT is actuated. The pressurizer level initially shrinks due to the addition of cold borated water. As the core makeup tank continues to add inventory to the primary system, the pressurizer level begins to increase. At approximately 13,300 seconds, the first core makeup tank essentially stops recirculating. The PRHR heat flux decreases below decay heat and a moderate heat up is experienced by the plant. Finally, at 21,800 seconds, the PRHR heat transfer matches the decay heat and the final cooldown commences.

Figure 15.5.1-6 shows the departure from nucleate boiling ratio (DNBR) until the time of reactor coolant trip and subsequent flow coastdown due to the loss of offsite power. At this time, core power and heat flux have diminished sufficiently, due to the reactor trip, that DNBR is well above the design limit value defined in Section 4.4.

The calculated sequence of events is shown in Table 15.5-1.

The limiting case presented here bounds all cases that model explicit operator action 60 minutes after reactor trip. For such events, the operator would take action to reduce the increase in coolant inventory. As the pressurizer water level would increase above the high pressurizer water level that normally isolates chemical and volume control system makeup, the normal letdown line could be placed into service to reduce the increase in coolant inventory. If letdown could not be placed into service, the operator could use the safety related reactor vessel head vent valves to reduce the increase in coolant inventory. For these events, following the procedures outlined in the Emergency Response Guidelines AFR-I.1, there is sufficient time for the operator to mitigate the consequences of this event, and the results of such an event have a greater margin to pressurizer overfill than that presented in this analysis.

15.5.1.4 Conclusions

The results of this analysis show that inadvertent operation of the core makeup tanks during power operation does not adversely affect the core, the reactor coolant system, or the steam system. The PRHR heat removal capacity is such that reactor coolant water is not relieved from the pressurizer safety valves. DNBR always remains above the design limit values, and reactor coolant system and steam generator pressures remain below 110 percent of their design values.

15.5.2 Chemical and Volume Control System Malfunction That Increases Reactor Coolant Inventory

15.5.2.1 Identification of Causes and Accident Description

An increase of reactor coolant inventory, which results from addition of cold unborated water to the reactor coolant system, is analyzed in Subsection 15.4.6.

In this Subsection 15.5.2, the increase of reactor coolant system inventory due to the addition of borated water is analyzed.

The increase of reactor coolant system coolant inventory may be due to the spurious operation of one or both of the chemical and volume control system pumps or by the closure of the letdown path. If the chemical and volume control system is injecting highly borated water into the reactor coolant system, the reactor experiences a negative reactivity excursion due to the injected boron, causing a decrease in reactor power and subsequent coolant shrinkage. The load decreases due to the effect of reduced steam pressure after the turbine control valve fully opens.

At high chemical and volume control system boron concentration, low reactivity feedback conditions, and reactor in manual rod control, an “S” signal will be generated by either the low T_{cold} or low steam line pressure setpoints before the chemical and volume control system can inject a significant amount of water into the reactor coolant system. In this case, the chemical and volume control system malfunction event proceeds similarly to, and is only slightly more limiting than, a spurious “S” signal event. If the automatic rod control is modeled and the pressurizer spray functions properly to prevent a high pressure reactor trip signal, no “S” signals are generated and this specific event is terminated by automatic isolation of the chemical and volume control system on the safety-related high-2 pressurizer level setpoint.

Under typical operating conditions for the AP1000, the boron concentration of the injected chemical and volume control system water is equal to that of the reactor coolant system. If the chemical and volume control system is functioning in this manner and the pressurizer spray system functions properly to prevent a high pressure reactor trip signal, no “S” signals are generated and this specific event is also terminated by automatic isolation of the chemical and volume control system on the safety-related high-2 pressurizer level setpoint.

While these scenarios are the most probable outcomes of a chemical and volume control system malfunction, several combinations of boron concentration, feedback conditions, and plant system interactions have been identified which can result in more limiting scenarios with respect to pressurizer overfill. The key factors that make this event more limiting than a spurious “S” signal event are that the reactor coolant system is at a lower average temperature, higher pressure, and a higher pressurizer level at the time an “S” signal is generated. These factors produce a greater volume of higher density water and, thus, a larger reactor coolant system mass at the time of the “S” signal. In addition, at lower reactor coolant system average temperature, the PRHR is less effective in removing decay heat, which results in greater expansion of the cold water injected by the core makeup tanks.

The limiting analysis scenario minimizes reactor coolant system average temperature, maximizes reactor coolant system mass, and maximizes pressurizer water volume at the time of an “S” signal. This scenario is as follows:

- Both of the chemical and volume control system pumps spuriously begin delivering flow at a boron concentration slightly higher than that of the reactor coolant system. (Assuming that a chemical and volume control system malfunction results in both chemical and volume control system pumps delivering flow is a conservative assumption. One chemical and volume control system pump is automatically controlled and one is manually controlled.)

- The non-safety-related pressurizer spray is assumed to be available, so that a high pressurizer pressure reactor trip is prevented.

Due to the boron addition in the core, the plant cools down until an “S” signal is generated on low cold leg temperature. On the “S” signal, the reactor is tripped, the reactor coolant pumps are tripped, the pressurizer heaters are blocked, and the main feedwater lines, steam lines, and chemical and volume control system are isolated. After a conservative 17-second delay, the PRHR heat exchanger is actuated and the core makeup tank discharge valves are opened.

Normally, the reactor coolant pumps would be tripped 15 seconds after the receipt of the “S” signal. However, to meet the requirements of GDC 17 of 10 CFR Part 50, Appendix A, a loss of offsite power is assumed to occur as a consequence of reactor trip. The primary effect of this assumption is the coastdown of the reactor coolant pumps. Immediately following reactor trip, the turbine is tripped, and after a 3-second delay, a consequential loss of offsite power is assumed. The basis for the 3-second delay is described in Subsection 15.0.14. As a result, the reactor coolant pumps are conservatively assumed to trip about 10 seconds before they would otherwise trip due to the “S” signal.

This event is a Condition II incident (a fault of moderate frequency) as defined in Subsection 15.0.1.

15.5.2.2 Analysis of Effects and Consequences

The malfunction of the chemical and volume control system is analyzed by using a modified version of the computer program LOFTRAN (Reference 1). The code simulates the neutron kinetics, reactor coolant system, pressurizer, pressurizer safety valves, pressurizer spray, steam generator, steam generator safety valves, PRHR heat exchanger, and core makeup tank. The program computes pertinent plant variables including temperatures, pressures, and power level.

Because of the power and temperature reduction during the transient, operating conditions do not approach the core limits. The PRHR heat exchanger removes the long-term decay heat to prevent possible reactor coolant system overpressurization or loss of reactor coolant system water.

Using an iterative analysis process, the boron concentration is chosen such that this limiting case bounds the cases that model explicit operator action 30 minutes after the reactor trip.

The assumptions are as follows:

- Initial operating conditions

  The initial reactor power is assumed to be 102 percent of nominal. The main feedwater flow measurement supports a 1-percent power uncertainty; use of a 2-percent power uncertainty is conservative. The initial pressurizer pressure is assumed to be 50 psi above nominal. The initial reactor coolant system average temperature is assumed to be 6.5°F above nominal.

- Moderator and Doppler coefficients of reactivity

  A least-negative moderator temperature coefficient, a low (absolute value) Doppler power coefficient, and a maximum boron worth are assumed. For a different set of reactivity feedback parameters, a different chemical and volume control system boron concentration can result in an identical transient.

- Reactor control

  Rod control is not modeled.

- Pressurizer heaters

  The pressurizer heaters are automatically blocked on an “S” signal, and do not add heat to the system during the period of fluid thermal expansion that produces the peak pressurizer water volume. Thus, the pressurizer heaters are assumed to be inoperable during this event.

- Pressurizer spray

The spray system controls the pressurizer pressure so that a high pressurizer pressure reactor trip is prevented.

- Boron injection

  After 10 seconds at steady state, the chemical and volume control system pumps start injecting borated water, which is slightly above the reactor coolant system boron concentration. Upon receipt of an “S” signal, the chemical and volume control system pumps are isolated and the core makeup tanks begin injecting 3400 ppm borated water.

- Turbine load

  The turbine load is assumed constant until the turbine D-EHC drives the control valve wide open. Then the turbine load drops as steam pressure drops.

- Protection and safety monitoring system actuations

  If the automatic rod control system is modeled and the pressurizer spray system functions properly, no reactor trip signal is expected to occur. Instead, the event is terminated by automatic isolation of the chemical and volume control system on the safety grade high-2 pressurizer level setpoint. If the automatic rod control system is not active and the pressurizer spray system is assumed to be available, reactor trip may be initiated on either low T_{cold} “S” or a low steam line pressure “S” signal.

  The core decay heat is removed by the PRHR heat exchanger. The worst single failure is assumed to occur in the outlet line of the PRHR heat exchanger. One of the two parallel isolation valves is assumed to fail to open.

  Plant systems and equipment available to mitigate the effect of the accident are discussed in Subsection 15.0.8 and listed in Table 15.0-6.

15.5.2.3 Results

Figures 15.5.2-1 through 15.5.2-11 show the transient response to a chemical and volume control system malfunction that results in an increase of reactor coolant system inventory. Neutron flux slowly decreases due to boron injection, but steam flow does not decrease until later in the transient when the turbine control valves are wide open.

As the chemical and volume control system injection flow increases reactor coolant system inventory, pressurizer water volume begins increasing while the primary system is cooling down. At about 1,090 seconds, the low T_{cold} setpoint is reached, the reactor trips, and the control rods start moving into the core.

Immediately following reactor trip, the turbine is tripped and after a 3-second delay, a consequential loss of offsite power is assumed and the reactor coolant pumps trip. The basis for the 3-second delay is described in Subsection 15.0.14. Soon after reactor trip, the pressurizer heaters are blocked and the main feedwater lines, steam lines, and chemical and volume control system are isolated. After a conservative 17-second delay, the PRHR heat exchanger is actuated and the core makeup tank discharge valves are opened. The core makeup tanks work in recirculation mode, meaning they are always filled with water because cold borated water injected through the injection lines is replaced by hot water coming from the cold leg balance lines.

The operation of the PRHR heat exchanger and the core makeup tanks cools down the plant. Due to the swelling of the core makeup tank water, the pressurizer level is still increasing. As the reactor coolant system average temperature goes below 490°F, the cooling effect due to the core makeup tanks is decreasing. In this condition, the PRHR heat exchanger cannot remove the entire decay heat. Reactor coolant system temperature tends to increase until an equilibrium between decay heat power and heat absorbed by the PRHR heat exchanger is reached.

When the PRHR heat flux matches the core decay heat, the pressurizer water volume stops increasing, and the pressurizer safety valves close. Then the core makeup tanks essentially stop injecting.

Figure 15.5.2-6 shows the DNBR until the time of reactor coolant pump trip and subsequent flow coastdown due to the loss of offsite power. At this time, core power and heat flux have diminished sufficiently, due to the reactor trip, that DNBR is well above the design limit value defined in Section 4.4.

The calculated sequence of events is shown in Table 15.5-1.

The limiting case presented here bounds all cases that model explicit operator action 30 minutes after reactor trip. For such events, the operator could take action to reduce the increase in coolant inventory. As the pressurizer water level would increase above the high pressurizer water level that normally isolates chemical and volume control system makeup, the normal letdown line could be placed into service to reduce the increase in coolant inventory. If letdown could not be placed into service, the operator would use the safety-related reactor vessel head vent valves to reduce the increase in coolant inventory. For these events, following the procedures outlined in the AP1000 Emergency Response Guidelines AFR-I.1, there is sufficient time for the operator to mitigate the consequences of this event, and the results of such an event have a greater margin to pressurizer overfill than that presented in this analysis.

15.5.2.4 Conclusions

The results of this analysis show that a chemical and volume control system malfunction does not adversely affect the core, the reactor coolant system, or the steam system. The PRHR heat removal capacity is such that reactor coolant water is not relieved from the pressurizer safety valves. DNBR remains above the design limit values, and reactor coolant system and steam generator pressures remain below 110 percent of their design values.

If the automatic rod control system and the pressurizer spray systems are assumed to function, no reactor trip signal is expected to occur. Instead, the event is terminated by automatic isolation of the chemical and volume control system on the safety grade high pressurizer level setpoint. If manual rod control is assumed and the pressurizer spray system is assumed to be unavailable, reactor trip may be initiated on either a high pressurizer pressure, low T_{cold} “S”, or a low steam line pressure “S” signal.

15.5.3 Boiling Water Reactor Transients

This subsection is not applicable to the AP1000.

15.5.4 Combined License Information

This subsection contained no requirement for additional information.

15.5.5 References

1. Burnett, T. W. T., et al., “LOFTRAN Code Description,” WCAP-7907-P-A (Proprietary) and WCAP-7907-A (Nonproprietary), April 1984.

Table 15.5-1 (Sheet 1 of 2)
Time Sequence of Events For Incidents Which Result in an Increase in Reactor Coolant Inventory

Accident	Event	Time (seconds)
Inadvertent operation of the core makeup tanks during power operation	Core makeup tank discharge valves open	10
	High-3 pressurizer level setpoint reached	520.7
	Rod motion begins	522.7
	Loss of offsite power	525.4
	Reactor coolant pumps trip	525.4
	High-3 pressurizer level setpoint reached	541.9
	PRHR heat exchanger actuated	573.9
	Pressurizer safety valves open	574.0
	Pressurizer safety valves close	594.0
	Pressurizer safety valves open	1,312
	Pressurizer safety valves close	5,880
	Low T_{cold} “S” setpoint is reached	12,354
	Second CMT starts recirculating	12,361
	First Core makeup tank stops recirculating	13,300
	Main steam and feed lines are isolated	12,366
	Pressurizer safety valves open	14,960
	Pressurizer safety valves close	20,140
	Peak pressurizer water volume occurs	20,480
	PRHR matches decay heat	21,800
	Second Core makeup tank stops recirculating	30,900

Table 15.5-1 (Sheet 2 of 2)
Time Sequence of Events For Incidents Which Result in an Increase in Reactor Coolant Inventory

Accident	Event	Time (seconds)
Chemical and volume control system malfunction that increases reactor coolant inventory	Chemical and volume control system charging pumps start	10
	Low T_{cold} "S" signal is reached	1,088
	Rod motion begins	1,090
	Loss of offsite power	1,093
	Reactor coolant pumps trip	1,093
	Main steam and feed lines are isolated	1,100
	Chemical and volume control system charging pumps are isolated	1,100
	Core makeup tank discharge valves open	1,100
	PRHR heat exchanger actuated	1,105
	Pressurizer safety valves open	1,424
	PRHR matches decay heat	14,720
	Pressurizer safety valves close	15,088
	Peak pressurizer water volume occurs	15,262
	Core makeup tanks stop recirculating	20,200

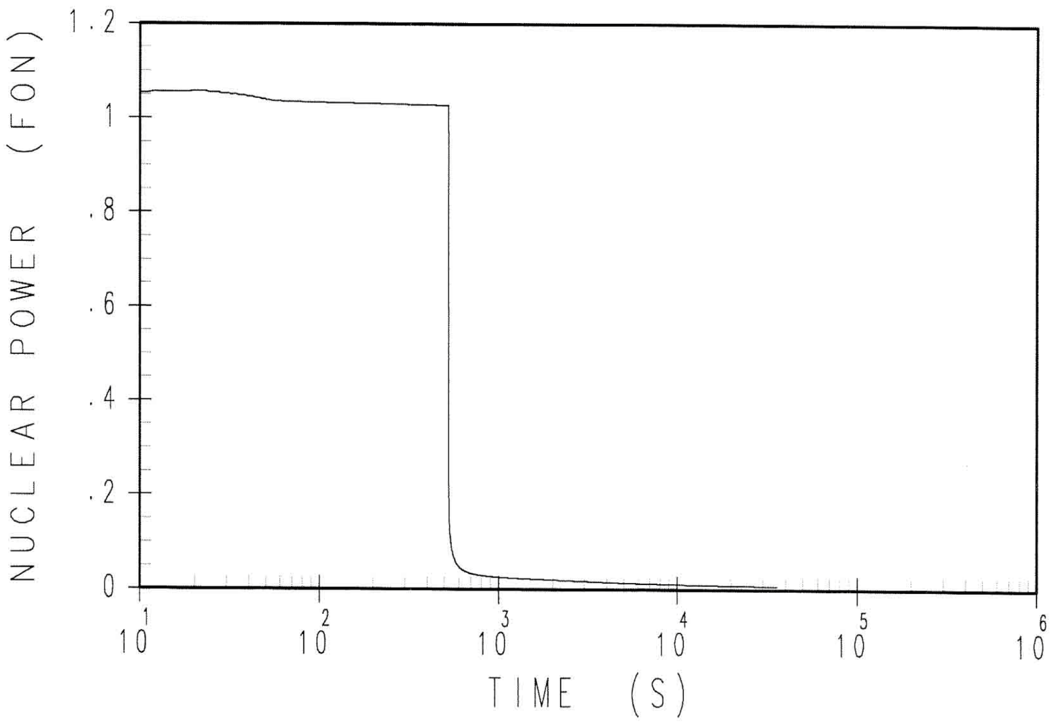


Figure 15.5.1-1
Core Nuclear Power Transient for Inadvertent Operation of the Emergency Core Cooling System Due to a Spurious Opening of the Core Makeup Tank Discharge Valves

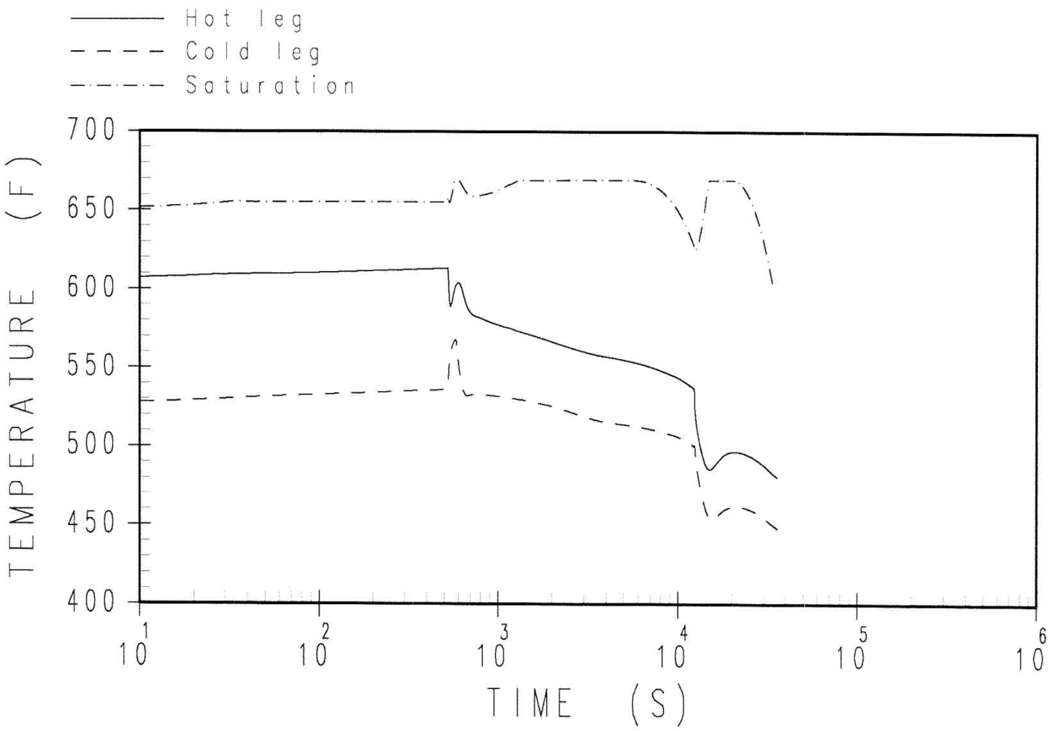


Figure 15.5.1-2
RCS Temperature Transient in Loop Containing the PRHR
for Inadvertent Operation of the Emergency Core Cooling System
Due to a Spurious Opening of the Core Makeup Tank Discharge Valves

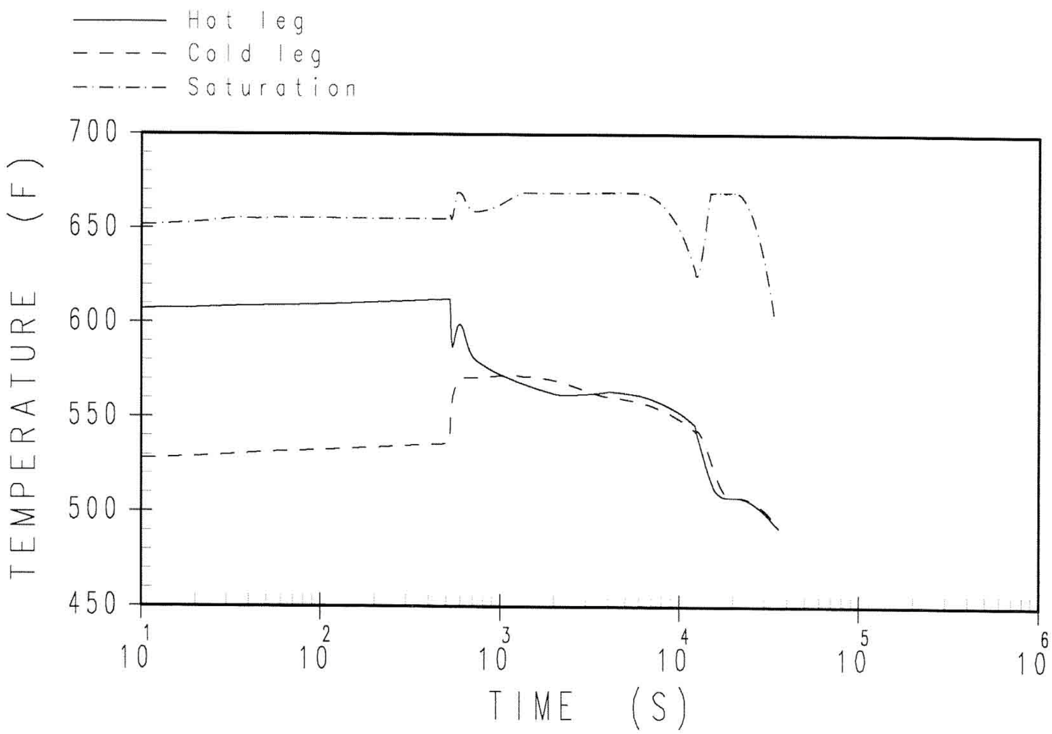


Figure 15.5.1-3
RCS Temperature Transient in Loop Not Containing the PRHR for Inadvertent Operation of the Emergency Core Cooling System Due to a Spurious Opening of the Core Makeup Tank Discharge Valves

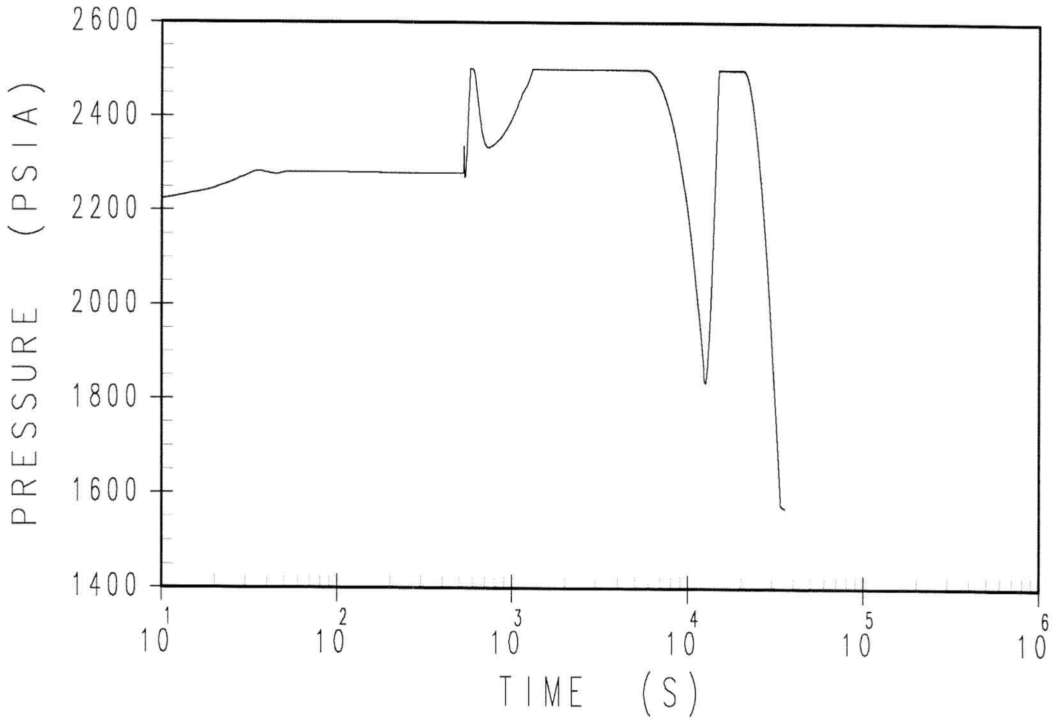


Figure 15.5.1-4
Pressurizer Pressure Transient for Inadvertent Operation of the Emergency Core Cooling System Due to a Spurious Opening of the Core Makeup Tank Discharge Valves

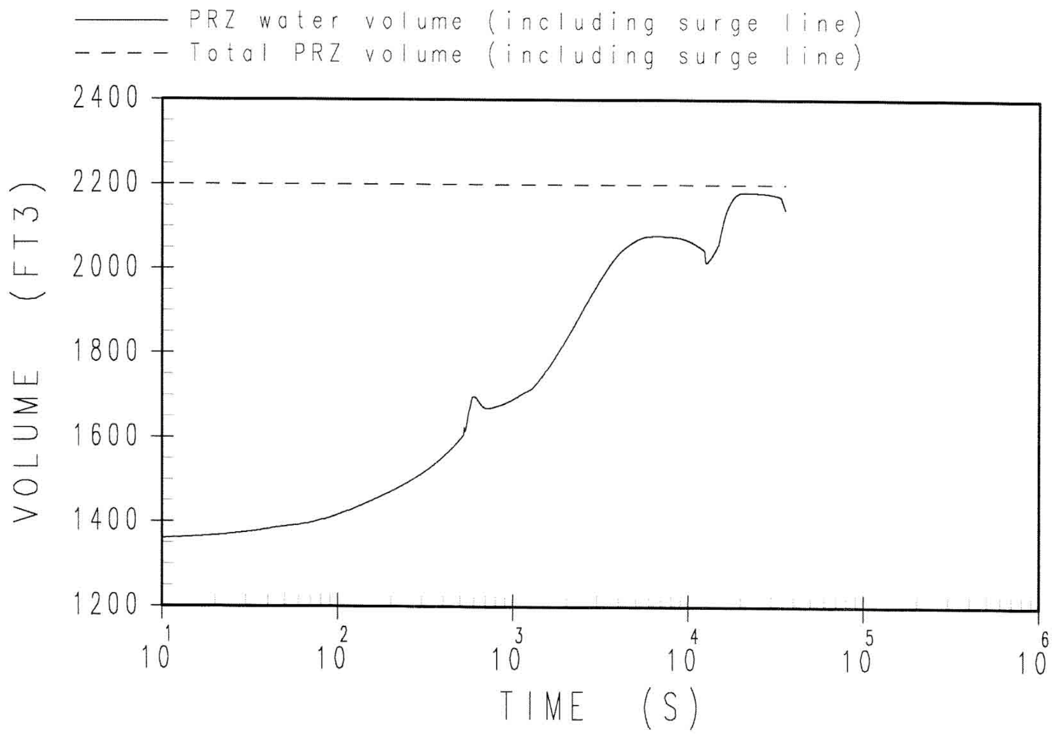


Figure 15.5.1-5
Pressurizer Water Volume Transient for Inadvertent Operation of the Emergency Core Cooling System Due to a Spurious Opening of the Core Makeup Tank Discharge Valves

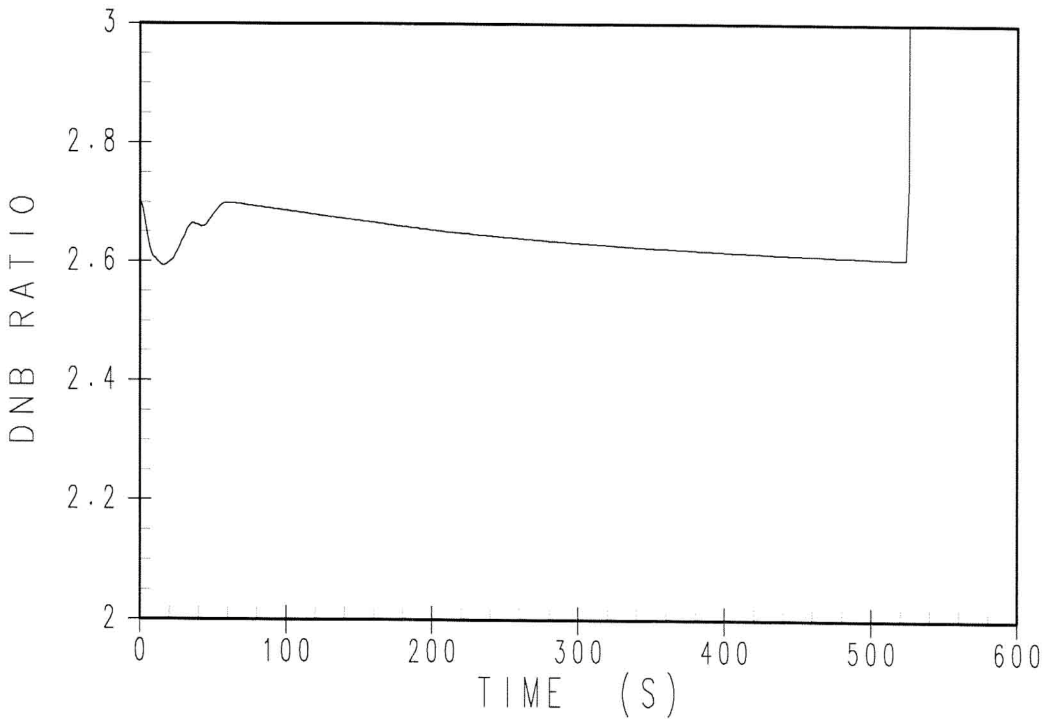


Figure 15.5.1-6
DNBR Transient for Inadvertent Operation of the Emergency Core Cooling System Due to a Spurious Opening of the Core Makeup Tank Discharge Valves

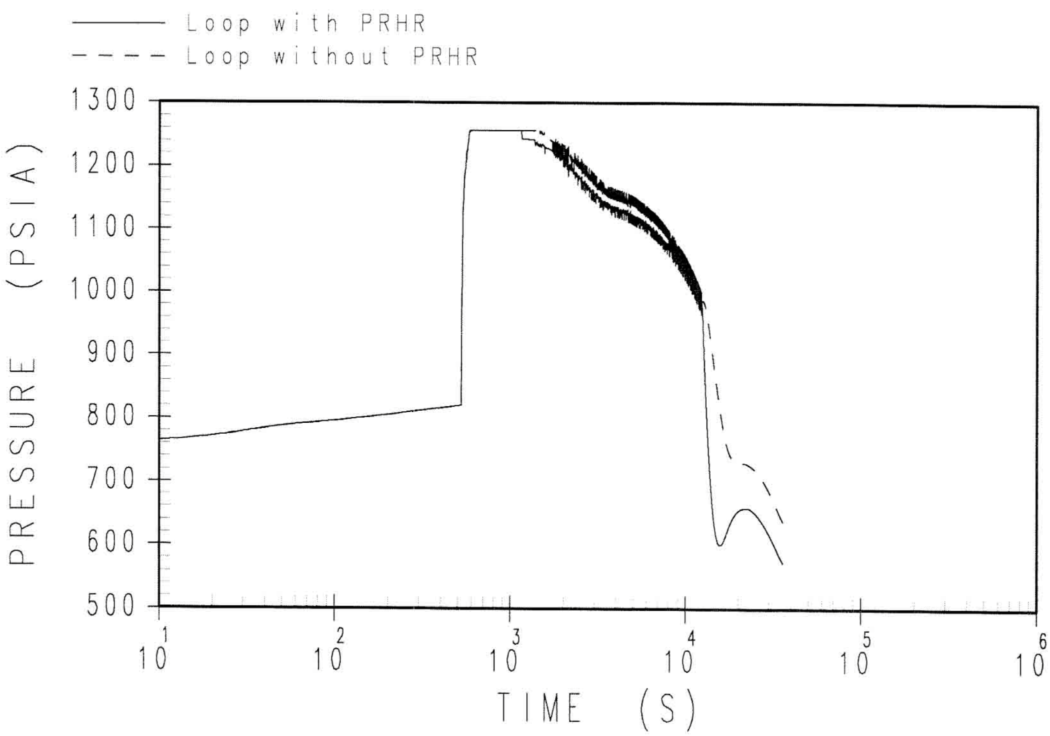


Figure 15.5.1-7
Steam Generator Pressure Transient for Inadvertent Operation of the Emergency Core Cooling System Due to a Spurious Opening of the Core Makeup Tank Discharge Valves

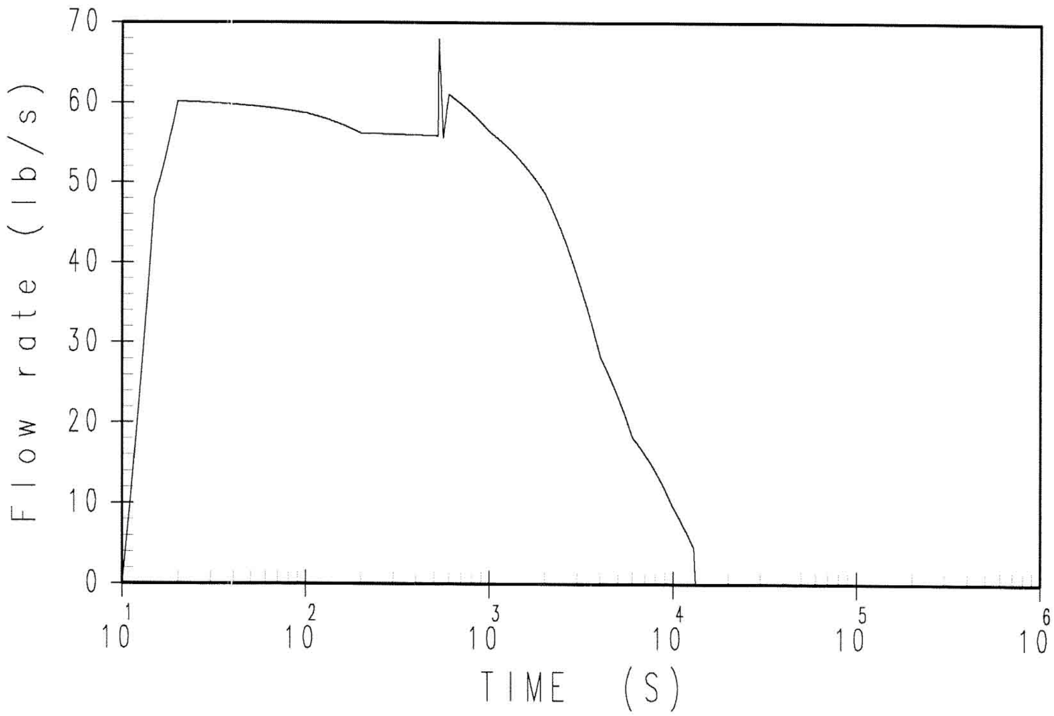


Figure 15.5.1-8
Inadvertent Actuated CMT Flow Rate Transient
for Inadvertent Operation of the Emergency Core Cooling System
Due to a Spurious Opening of the Core Makeup Tank Discharge Valves

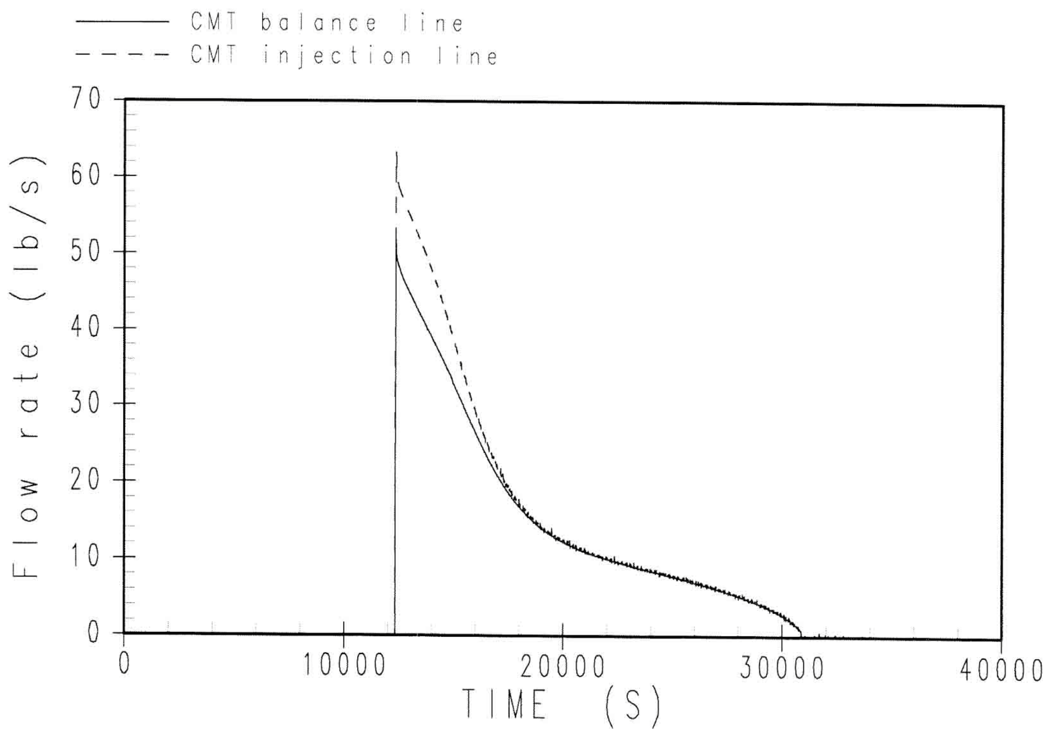


Figure 15.5.1-9
Intact CMT Flow Rate Transient
for Inadvertent Operation of the Emergency Core Cooling System
Due to a Spurious Opening of the Core Makeup Tank Discharge Valves

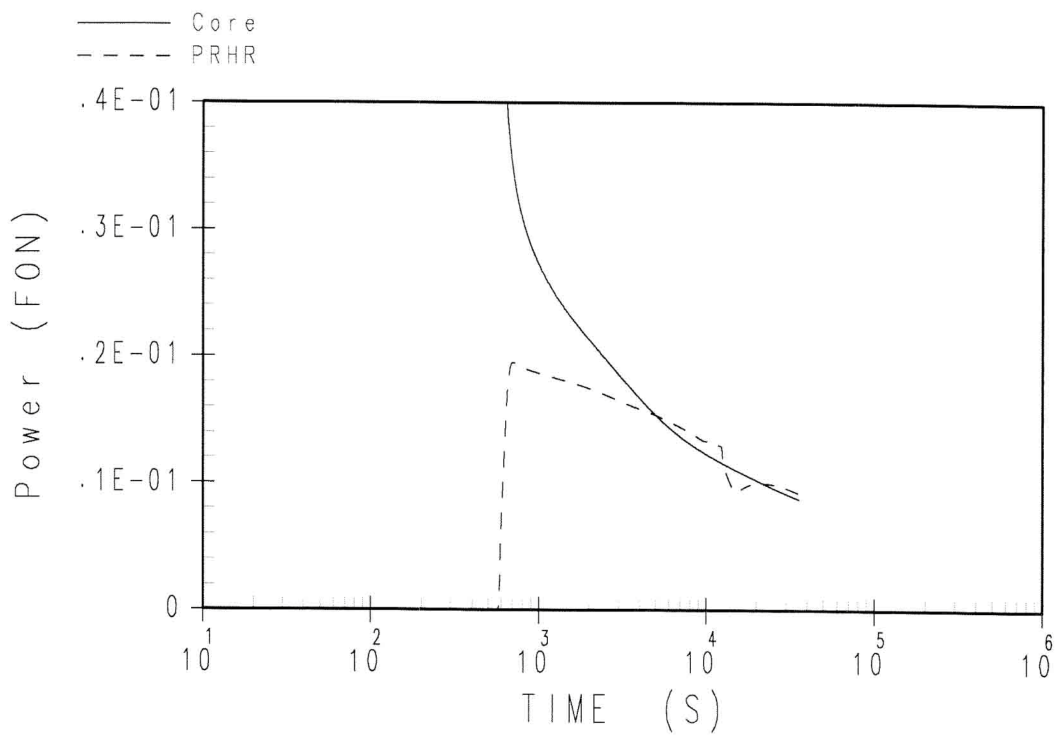


Figure 15.5.1-10
PRHR and Core Heat Flux Transient for Inadvertent Operation of the Emergency Core Cooling System Due to a Spurious Opening of the Core Makeup Tank Discharge Valves

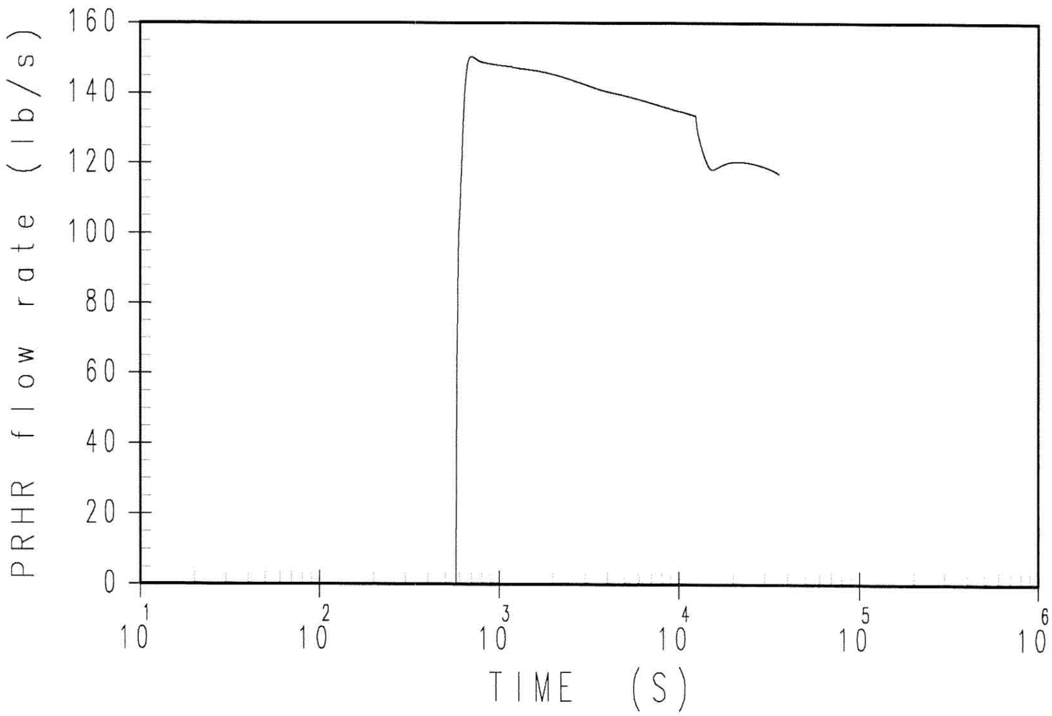


Figure 15.5.1-11
PRHR Flow Rate Transient for Inadvertent Operation of the Emergency Core Cooling System Due to a Spurious Opening of the Core Makeup Tank Discharge Valves

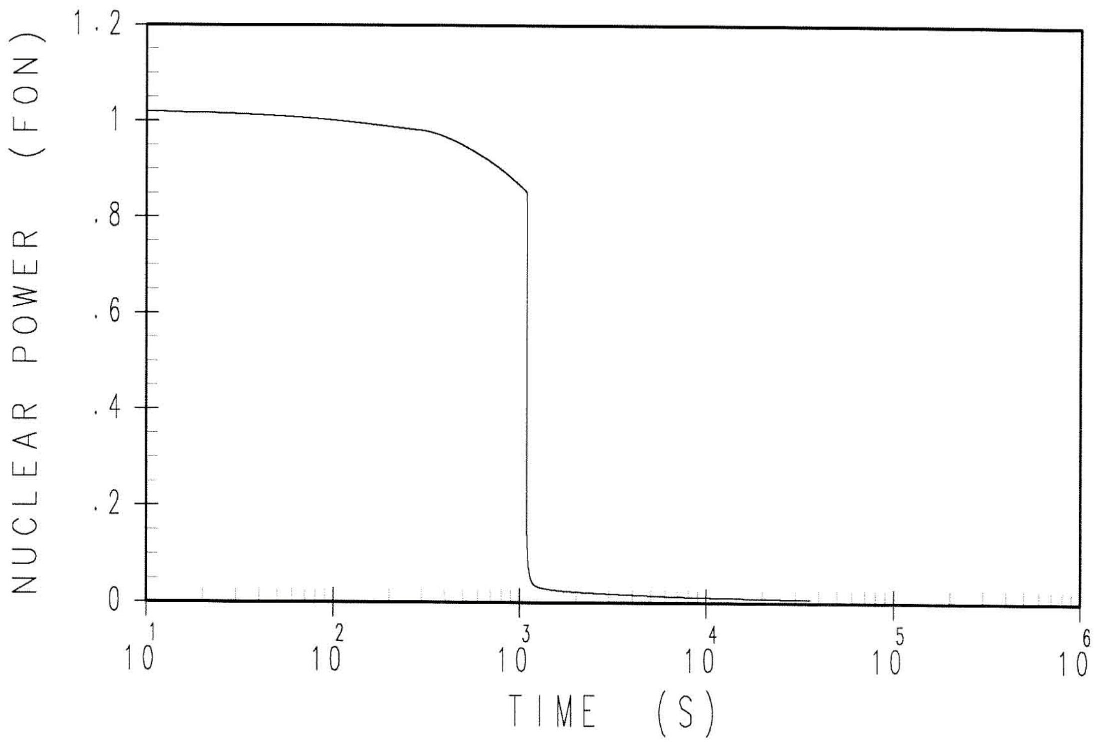


Figure 15.5.2-1
Core Nuclear Power Transient for Chemical and Volume Control System Malfunction

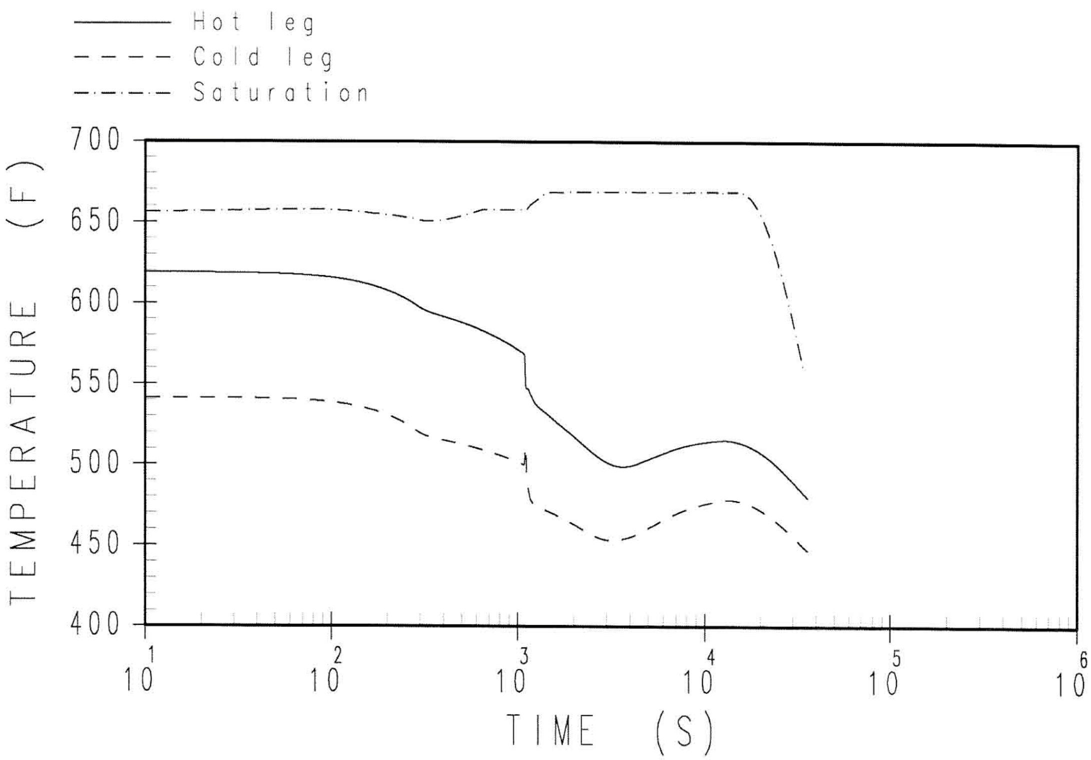


Figure 15.5.2-2
RCS Temperature Transient in Loop Containing the PRHR
for Chemical and Volume Control System Malfunction

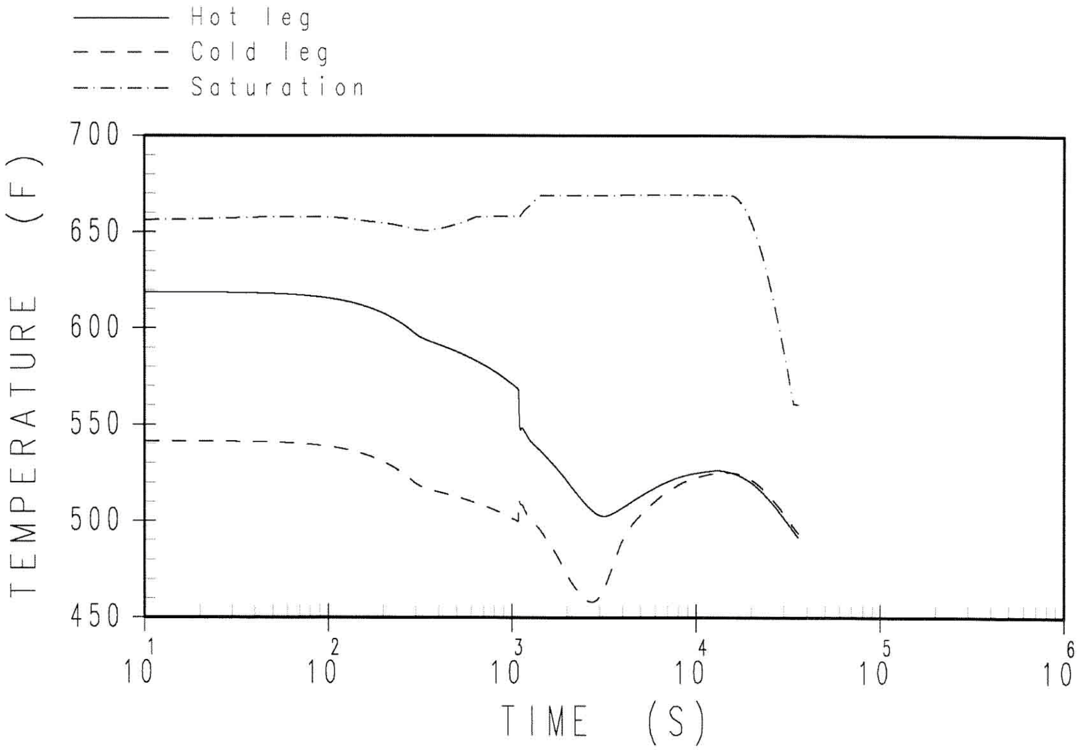


Figure 15.5.2-3
RCS Temperature Transient in Loop Not Containing the PRHR
for Chemical and Volume Control System Malfunction

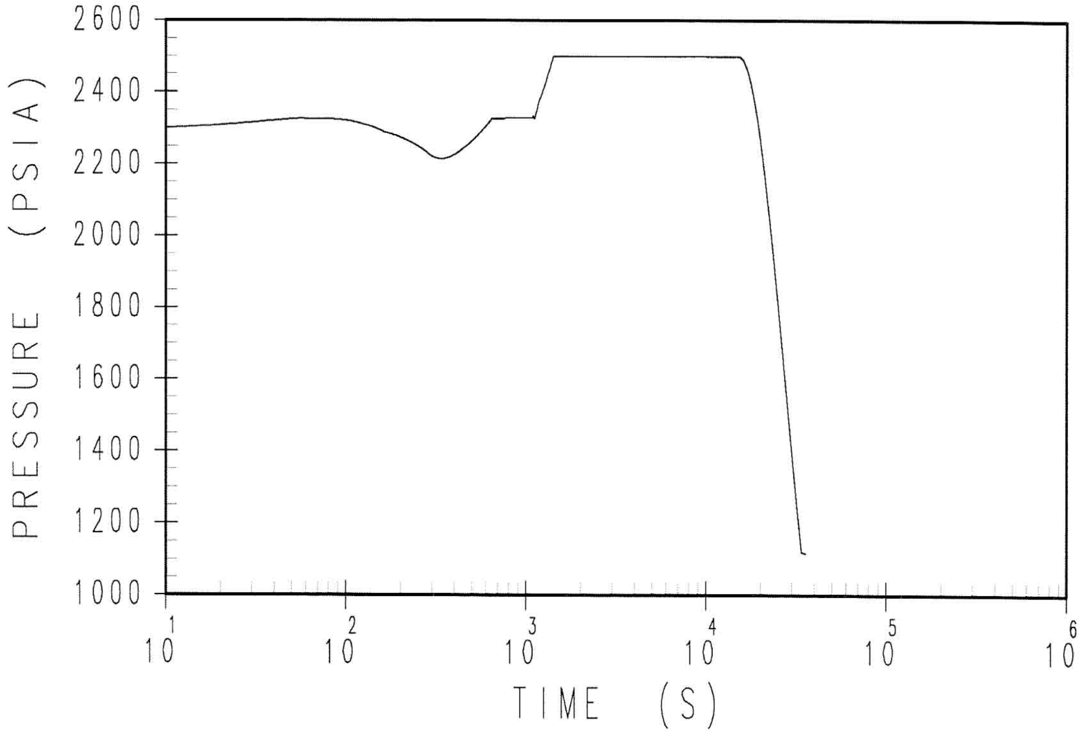


Figure 15.5.2-4
Pressurizer Pressure Transient
for Chemical and Volume Control System Malfunction

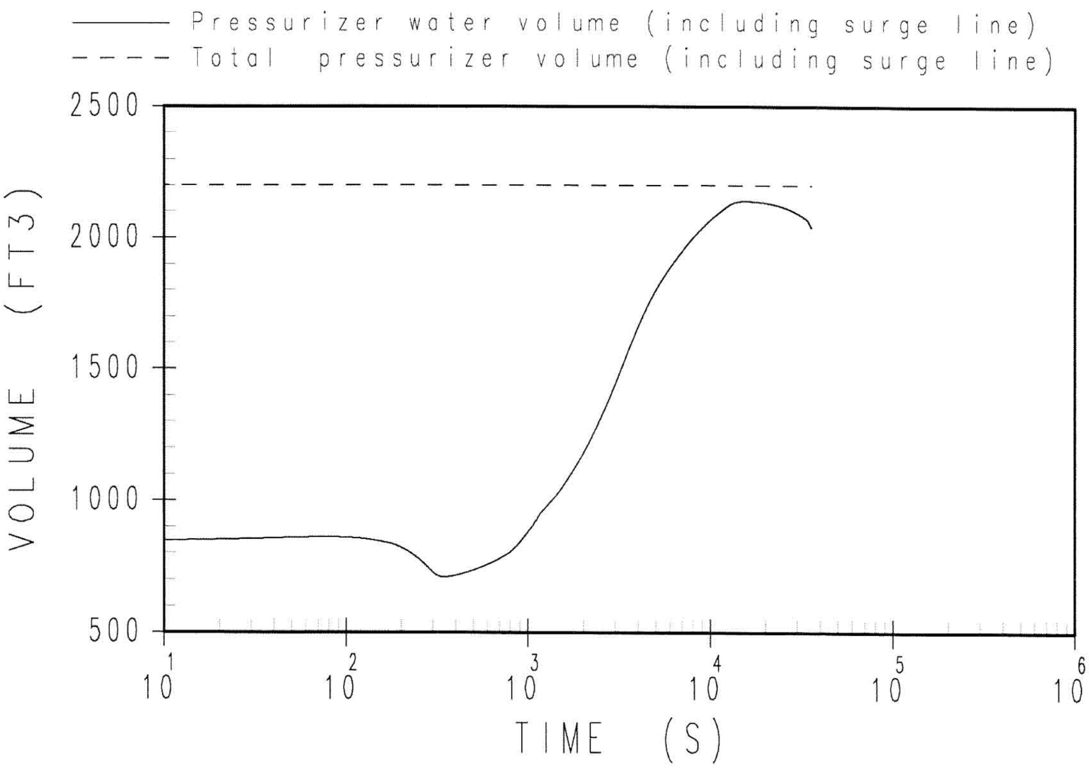


Figure 15.5.2-5
Pressurizer Water Volume Transient
for Chemical and Volume Control System Malfunction

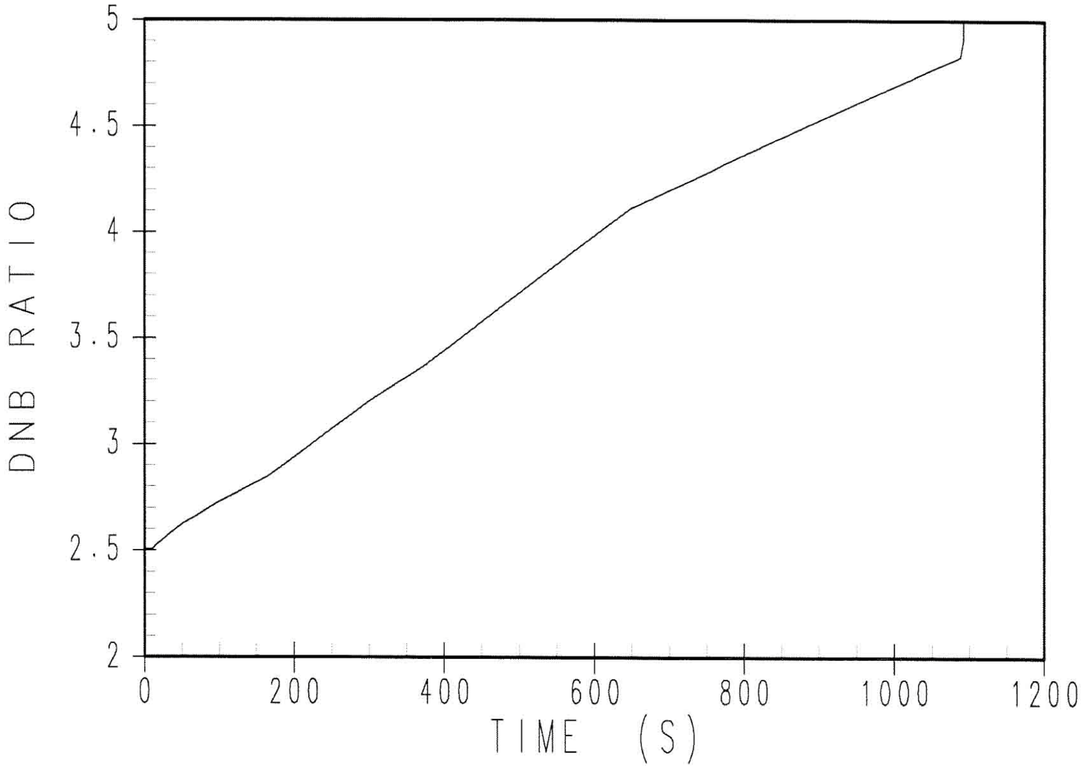


Figure 15.5.2-6
DNBR Transient for Chemical and Volume Control System Malfunction

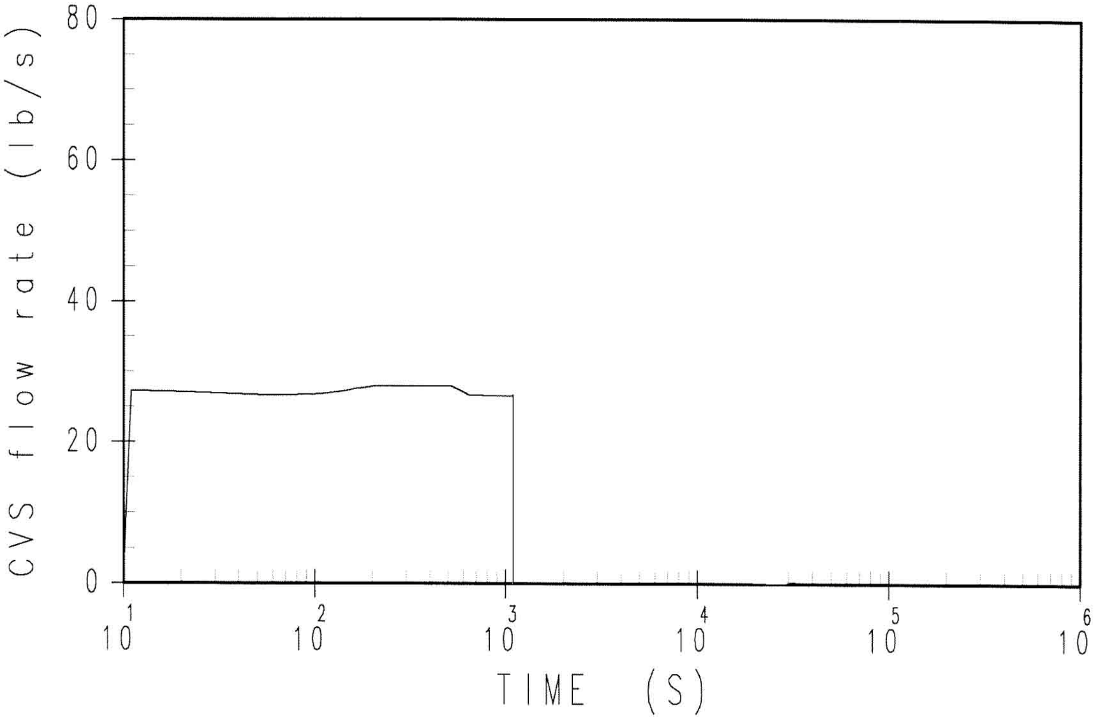


Figure 15.5.2-7
CVS Flow Rate Transient
for Chemical and Volume Control System Malfunction

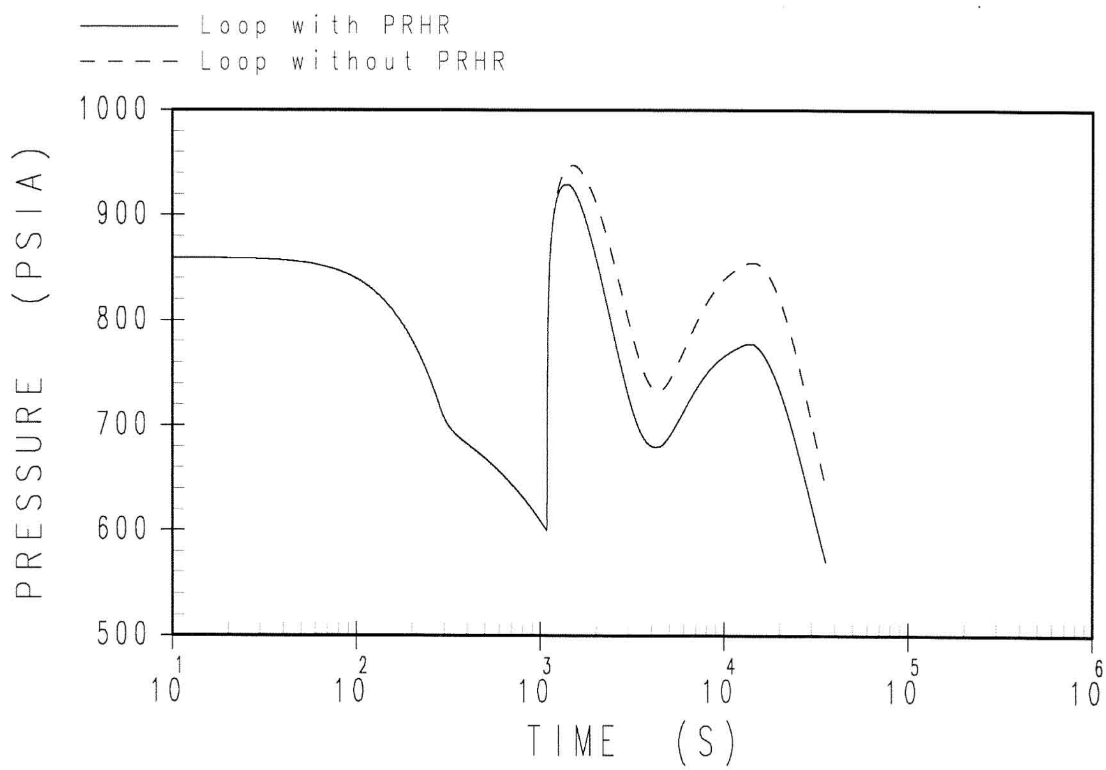


Figure 15.5.2-8
Steam Generator Pressure Transient
for Chemical and Volume Control System Malfunction

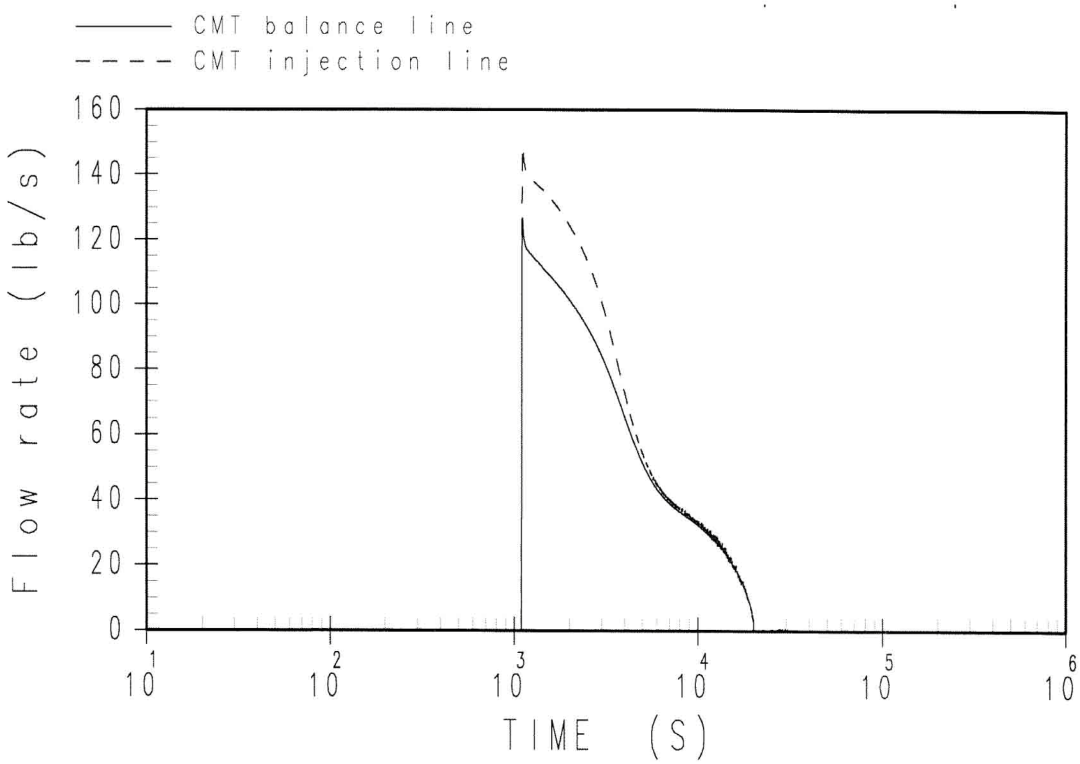


Figure 15.5.2-9
CMT Injection Line and Balance Line Flow Transient
for Chemical and Volume Control System Malfunction

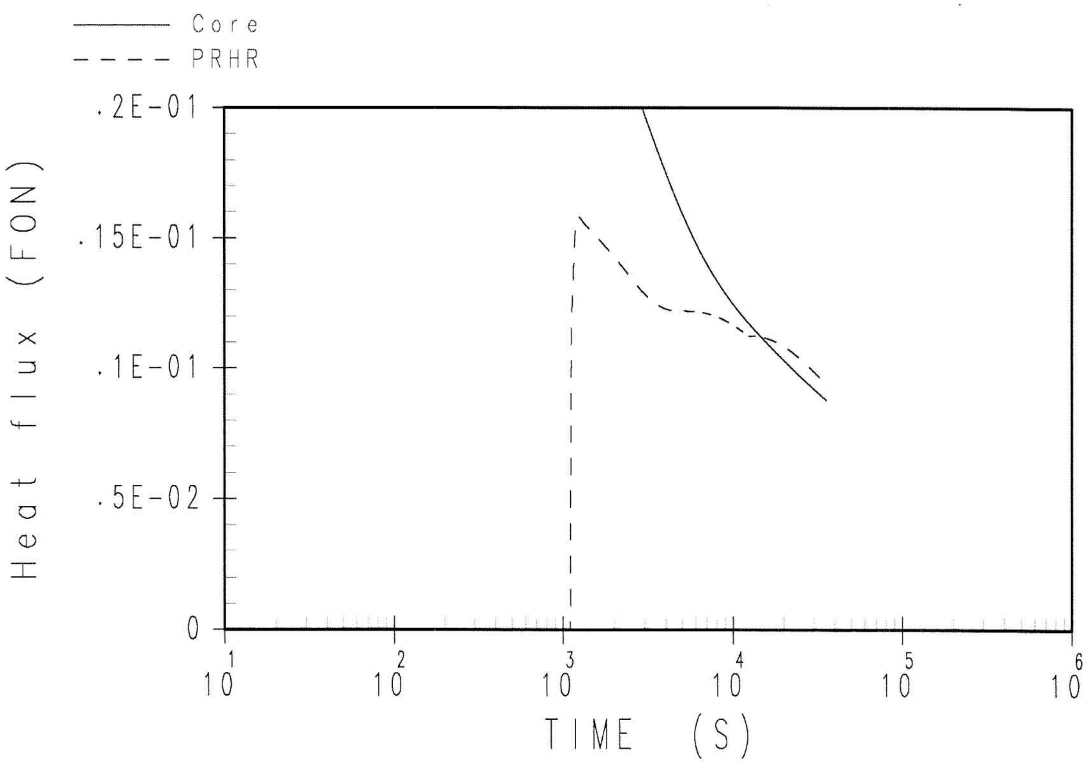


Figure 15.5.2-10
PRHR and Core Heat Flux Transient
for Chemical and Volume Control System Malfunction

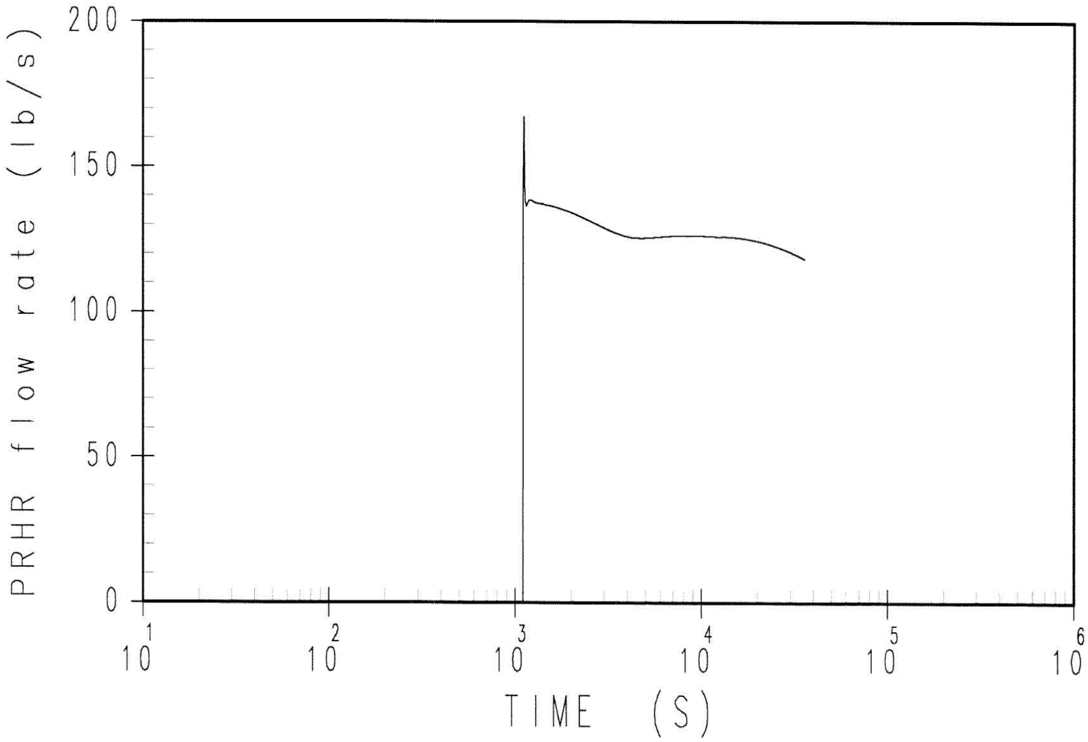


Figure 15.5.2-11
PRHR Flow Rate Transient
for Chemical and Volume Control System Malfunction

15.6 Decrease in Reactor Coolant Inventory

This section discusses the following events that result in a decrease in reactor coolant inventory:

- An inadvertent opening of a pressurizer safety valve or inadvertent operation of the automatic depressurization system (ADS)
- A break in an instrument line or other lines from the reactor coolant pressure boundary that penetrate the containment
- A steam generator tube failure
- A loss-of-coolant accident (LOCA) resulting from a spectrum of postulated piping breaks within the reactor coolant pressure boundary

The applicable accidents in this category have been analyzed. It has been determined that the most severe radiological consequences result from the major LOCA described in Subsection 15.6.5. The LOCA, chemical and volume control system letdown line break outside the containment and the steam generator tube rupture (SGTR) accident are analyzed for radiological consequences. Other accidents described in this section are bounded by these accidents.

15.6.1 Inadvertent Opening of a Pressurizer Safety Valve or Inadvertent Operation of the ADS

15.6.1.1 Identification of Causes and Accident Description

Two types of inadvertent depressurization are discussed in this section. One covers all inadvertent operation of ADS valves. The other covers inadvertent opening of a pressurizer safety valve.

An inadvertent depressurization of the reactor coolant system can occur as a result of an inadvertent opening of a pressurizer safety valve or ADS valves. Initially, the event results in a rapidly decreasing reactor coolant system pressure. The pressure decrease causes a decrease in power via the moderator density feedback. The average coolant temperature decreases slowly, but the pressurizer level increases until reactor trip.

The reactor may be tripped by the following reactor protection system signals:

- Overtemperature ΔT
- Pressurizer low pressure

The ADS is designed such that inadvertent operation of the ADS is classified as a Condition III event, an infrequent fault.

An inadvertent opening of a pressurizer safety valve is a Condition II event, a fault of moderate frequency.

The ADS system consists of four stages of depressurization valves. The ADS stages are interlocked; for example, Stage 1 is initiated first and subsequent stages are not actuated until previous stages have been actuated. Each stage includes two redundant parallel valve paths such that no single failure prevents operation of the ADS stage when it is called upon to actuate and the spurious opening of a single ADS valve does not initiate ADS flow. To actuate the ADS manually from the main control room, the operators actuate two separate controls positioned at some distance apart on the main control board. Therefore, one unintended operator action does not cause ADS actuation.

ADS Stage 1 has a design opening time of 40 seconds and an effective flow area of 7 in^2 (maximum). ADS Stages 2 and 3 have design opening times of 100 seconds and an effective flow area of 26 in^2 (maximum).

The valve stroke times shown in Chapter 15 tables (input/assumptions) reflect the design basis of the AP1000. The accidents addressed in this section were evaluated for these design basis valve stroke times. The results of this evaluation have shown that there is a small impact on the analysis and the conclusions remain valid. The output provided in this section for the analyses is representative of the transient phenomenon.

In each ADS path are two valves in series such that no mechanical failure could result in an inadvertent operation of an ADS stage. The ADS Stage 4 squib valves cannot be opened while the reactor coolant system is at nominal operating pressure.

For this analysis, multiple failures and or errors are assumed which actuate both Stage 1 ADS paths. Although ADS Stages 2 and 3 have larger depressurization valves, the opening time of the Stage 1 depressurization valves is faster. This results in the most severe reactor coolant system depressurization due to ADS operation with the reactor at power.

Inadvertent opening of a pressurizer safety valve can only be postulated due to a mechanical failure. Although a pressurizer safety valve is smaller than the combined two Stage 1 ADS valves, the pressurizer safety valve is postulated to open in a short time.

Therefore, analyses are presented in this section for the inadvertent opening of a pressurizer safety valve and the inadvertent opening of two paths of Stage 1 of the ADS. These analyses are performed to demonstrate that the departure from nucleate boiling ratio (DNBR) does not decrease below the design limit values (see Section 4.4) while the reactor is at power.

In meeting the requirements of GDC 17 of 10 CFR Part 50, Appendix A, analyses have been performed to evaluate the effects produced by a possible consequential loss of offsite power during inadvertent reactor coolant system depressurization events. As discussed in Subsection 15.0.14, the loss of offsite power is considered as a direct consequence of a turbine trip occurring while the plant is operating at power. The primary effect of the loss of offsite power is to cause the reactor coolant pumps to coast down.

15.6.1.2 Analysis of Effects and Consequences

15.6.1.2.1 Method of Analysis

The accidental depressurization transient is analyzed by using the computer code LOFTRAN (References 14 and 15). The code simulates the neutron kinetics, reactor coolant system, pressurizer, pressurizer safety valves, main steam isolation valves, pressurizer spray, steam generator, and steam generator safety valves. The code computes pertinent plant variables including temperatures, pressures, and power level.

For reactor coolant system depressurization analyses that include a primary coolant flow coastdown caused by a consequential loss of offsite power, a combination of three computer codes is used to perform the DNBR analyses. First the LOFTRAN code is used to perform the plant system transient. The FACTRAN code (Reference 18) is then used to calculate the core heat flux based on nuclear power and reactor coolant flow from LOFTRAN. Finally, the VIPRE-01 code (see Section 4.4) is used to calculate the DNBR using heat flux from FACTRAN and flow from LOFTRAN.

Plant characteristics and initial conditions are discussed in Subsection 15.0.3. The following assumptions are made to give conservative results in calculating the DNBR during the transient:

- Initial conditions are discussed in Subsection 15.0.3. Uncertainties in initial conditions are included in the DNBR limit as discussed in WCAP-11397-P-A (Reference 16).
- A least negative moderator temperature coefficient is assumed. The spatial effect of voids resulting from local or subcooled boiling is not considered in the analysis with respect to reactivity feedback or core power shape.
- A large (absolute value) Doppler coefficient of reactivity is used such that the resulting amount of positive feedback is conservatively high to retard any power decrease.

Plant systems and equipment necessary to mitigate the effects of reactor coolant system depressurization are discussed in Subsection 15.0.8 and are listed in Table 15.0-6.

Normal reactor control systems are not required to function. The rod control system is assumed to be in the automatic mode to maintain the core at full power until the reactor trip protection function is reached. This is a worst case assumption. The reactor protection system functions to trip the reactor on the appropriate signal. No single active failure prevents the reactor protection system from functioning properly.

15.6.1.2.2 Results

The system response to an inadvertent opening of a pressurizer safety valve is shown in Figures 15.6.1-1 through 15.6.1-5. The figures show the results for cases with and without offsite power available. The calculated sequence of events for both inadvertent opening of a pressurizer safety valve scenarios are shown in Table 15.6.1-1.

A pressurizer safety valve is assumed to step open at the start of the event. The reactor coolant system then depressurizes until the overtemperature ΔT reactor trip setpoint is reached. Figure 15.6.1-3 shows the pressurizer pressure transient.

In the case where offsite power is lost, ac power is assumed to be lost 3 seconds after a turbine trip signal occurs. At this time, the reactor coolant pumps are assumed to start coasting down and reactor coolant system flow begins decreasing (Figure 15.6.1-5). The availability of offsite power has minimal impact on the pressure transient during the period of interest.

Prior to tripping of the reactor, the core power remains relatively constant (Figure 15.6.1-1). The minimum DNBR during the event occurs shortly after the rods begin to be inserted into the core (Figure 15.6.1-2). In the case where offsite power is lost, reactor trip has already been initiated and core heat flux has started decreasing when the reactor coolant system flow reduction starts. The DNBR continues to increase when reactor coolant system flow begins to decrease due to the loss of offsite power. Therefore, the minimum DNBR occurs at the same time for cases with and without offsite power available. The DNBR remains above the design limit values as discussed in Section 4.4 throughout the transient.

The system response for inadvertent operation of the ADS is shown in Figures 15.6.1-6 through 15.6.1-10. The figures show the results for cases with and without offsite power available. The sequences of events are provided in Table 15.6.1-1. The responses for inadvertent operation of the ADS are very similar to those obtained for inadvertent opening of a pressurizer safety valve.

15.6.1.3 Conclusion

The results of the analysis show that the overtemperature ΔT reactor protection system signal provides adequate protection against the reactor coolant system depressurization events. The calculated DNBR remains above the design limit defined in Section 4.4. The long-term plant responses due to a stuck-open ADS valve or pressurizer safety valve, which cannot be isolated, is bounded by the small-break LOCA analysis.

15.6.2 Failure of Small Lines Carrying Primary Coolant Outside Containment

The small lines carrying primary coolant outside containment are the reactor coolant system sample line and the discharge line from the chemical and volume control system to the liquid radwaste system. These lines are used only periodically. No instrument lines carry primary coolant outside the containment.

When excess primary coolant is generated because of boron dilution operations, the chemical and volume control system purification flow is diverted out of containment to the liquid radwaste system. Before passing outside containment, the flow stream passes through the chemical and volume control system heat exchangers and mixed bed demineralizer. The flow leaving the containment is at a temperature of less than 140°F and has been cleaned by the demineralizer. The flow out a postulated break in this line is limited to the chemical and volume control system purification flow rate of 100 gpm. Considering the low temperature of the flow and the reduced iodine activity because of demineralization, this event is not analyzed. The postulated sample line break is more limiting.

The sample line isolation valves inside and outside containment are open only when sampling. The failure of the sample line is postulated to occur between the isolation valve outside the containment and the sample panel. Because the isolation valves are open only when sampling, the loss of sample flow provides indication of the break to plant personnel. In addition, a break in a sample line results in activity release and a resulting actuation of area and air radiation monitors. The loss of coolant reduces the pressurizer level and creates a demand for makeup to the reactor coolant system. Upon indication of a sample line break, the operator would take action to isolate the break.

The sample line includes a flow restrictor at the point of sample to limit the break flow to less than 130 gpm. The liquid sampling lines are 1/4 inch tubing which further restricts the break flow of a sampling line outside containment. Offsite doses are based on a conservative break flow of 130 gpm with isolation after 30 minutes.

15.6.2.1 Source Term

The only significant radionuclide releases are the iodines and the noble gases. The analysis assumes that the reactor coolant iodine is at the maximum Technical Specification level for continuous operation. In addition, it is assumed that an iodine spike occurs at the time of the accident. The reactor coolant noble gas activities are assumed to be those associated with the design basis fuel defect level.

15.6.2.2 Release Pathway

The reactor coolant that is spilled from the break is assumed to be at high temperature and pressure. A large portion of the flow flashes to steam, and the iodine in the flashed liquid is assumed to become airborne.

The iodine and noble gases are assumed to be released directly to the environment with no credit for depletion, although a large fraction of the airborne iodine is expected to deposit on building surfaces. No credit is assumed for radioactive decay after release.

15.6.2.3 Dose Calculation Models

The models used to calculate doses are provided in Appendix 15A.

15.6.2.4 Analytical Assumptions and Parameters

The assumptions and parameters used in the analysis are listed in Table 15.6.2-1.

15.6.2.5 Identification of Conservatisms

The assumptions used contain the following significant conservatisms:

- The reactor coolant activities are based on a fuel defect level of 0.25 percent; whereas, the expected fuel defect level is far less than this (see Section 11.1).
- It is unlikely that the conservatively selected meteorological conditions would be present at the time of the accident.

15.6.2.6 Doses

Using the assumptions from Table 15.6.2-1, the calculated total effective dose equivalent (TEDE) doses are determined to be 1.3 rem at the exclusion area boundary and 0.6 rem at the low population zone outer boundary. These doses are a small fraction of the dose guideline of 25 rem TEDE identified in 10 CFR Part 50.34. The phrase “a small fraction” is taken as being ten percent or less.

At the time the accident occurs, there is the potential for a coincident loss of spent fuel pool cooling with the result that the pool could reach boiling and a portion of the radioactive iodine in the spent fuel pool could be released to the environment. The loss of spent fuel pool cooling has been evaluated for a duration of 30 days. There is no contribution to the 2-hour site boundary dose because pool boiling would not occur until after 2 hours. The 30-day contribution to the dose at the low population zone boundary is less than 0.01 rem TEDE and, when this is added to the dose calculated for the small line break outside containment, the resulting total dose remains less than the value reported above.

15.6.3 Steam Generator Tube Rupture

15.6.3.1 Identification of Cause and Accident Description

15.6.3.1.1 Introduction

The accident examined is the complete severance of a single steam generator tube. The accident is assumed to take place at power with the reactor coolant contaminated with fission products corresponding to continuous operation with a limited number of defective fuel rods within the allowance of the Technical Specifications. The accident leads to an increase in contamination of the secondary system due to leakage of radioactive coolant from the reactor coolant system. In the event of a coincident loss of offsite power, or a failure of the condenser steam dump, discharge of radioactivity to the atmosphere takes place via the steam generator power-operated relief valves or the safety valves.

The assumption of a complete tube severance is conservative because the steam generator tube material (Alloy 690) is a corrosion-resistant and ductile material. The more probable mode of tube failure is one or more smaller leaks of undetermined origin. Activity in the secondary side is subject to continual surveillance, and an accumulation of such leaks, which exceeds the limits established in the Technical Specifications, is not permitted during operation.

The AP1000 design provides automatic protective actions to mitigate the consequences of an SGTR. The automatic actions include reactor trip, actuation of the passive residual heat removal (PRHR) heat exchanger, initiation of core makeup tank flow, termination of pressurizer heater operation, and isolation of chemical and volume control system flow and startup feedwater flow on high-2 steam generator level or high steam generator level coincident with reactor trip (P-4). These protective actions result in automatic cooldown and depressurization of the reactor coolant system, termination of the break flow and release of steam to the atmosphere, and long-term maintenance of stable conditions in the reactor coolant system. These protection systems serve to prevent steam generator overfill (see discussion in Subsections 15.6.3.1.2 and 15.6.3.1.3) and to maintain offsite radiation doses within the allowable guideline values for a design basis SGTR. The operator may take actions that would provide a more rapid mitigation of the consequences of an SGTR.

Because of the series of alarms described next, the operator can readily determine when an SGTR occurs, identify and isolate the ruptured steam generator, and complete the required recovery actions to stabilize the plant and terminate the primary-to-secondary break flow. The recovery procedures are completed on a time scale that terminates break flow to the secondary system before steam generator overfill occurs and limits the offsite doses to acceptable levels without actuation of the ADS. Indications and controls are provided to enable the operator to carry out these functions.

15.6.3.1.2 Sequence of Events for a Steam Generator Tube Rupture

The following sequence of events occur following an SGTR:

- Pressurizer low pressure and low level alarms are actuated and chemical and volume control system makeup flow and pressurizer heater heat addition starts or increases in an attempt to maintain pressurizer level and pressure. On the secondary side, main feedwater flow to the affected steam generator is reduced because the primary-to-secondary break flow increases steam generator level.

- The condenser air removal discharge radiation monitor, steam generator blowdown radiation monitor, and/or main steam line radiation monitor alarm indicate an increase in radioactivity in the secondary system.

- Continued loss of reactor coolant inventory leads to a reactor trip generated by a low pressurizer pressure or over-temperature ΔT signal. Following reactor trip, the SGTR leads to a decrease in reactor coolant pressure and pressurizer level, counteracted by chemical and volume control system flow and pressurizer heater operation. A safeguards (“S”) signal that provides core makeup tank and PRHR heat exchanger actuation is initiated by low pressurizer pressure or low-2 pressurizer level. The “S” signal automatically terminates the normal feedwater supply and trips the reactor coolant pumps. The power to the pressurizer heaters is also terminated. Startup feedwater flow is initiated on a low steam generator narrow range level signal and controls the steam generator levels to the programmed level.

- The reactor trip automatically trips the turbine, and if offsite power is available, the steam dump valves open permitting steam dump to the condenser. In the event of a loss of offsite power or loss of the condenser, the steam dump valves automatically close to protect the condenser. The steam generator pressure rapidly increases resulting in steam discharge to the atmosphere through the steam generator power-operated relief valves and/or the safety valves.

- Following reactor trip and core makeup tank and PRHR actuation, the PRHR heat exchanger operation – combined with startup feedwater flow, borated core makeup tank flow, and chemical and volume control system flow – provides a heat sink that absorbs the decay heat. This reduces the amount of steam generated in the steam generators and steam bypass to the condenser. In the case of loss of offsite power, this reduces steam relief to the atmosphere.

- Injection of the chemical and volume control system and core makeup tank flow stabilizes reactor coolant system pressure and pressurizer water level, and the reactor coolant system pressure trends toward an equilibrium value, where the total injected flow rate equals the break flow rate.

15.6.3.1.3 Steam Generator Tube Rupture Automatic Recovery Actions

The AP1000 incorporates several protection system and passive design features that automatically terminate a steam generator tube leak and stabilize the reactor coolant system, in the highly unlikely event that the operators do not perform recovery actions. Following an SGTR, the injecting chemical and volume control system flow (and pressurizer heater heat addition if the pressure control system is operating) maintains the primary-to-secondary break flow and the ruptured steam generator secondary level increases as break flow accumulates in the steam generator. Eventually, the ruptured steam generator secondary level reaches the high and high-2 steam generator narrow range level setpoint, which is near the top of the narrow range level span.

The AP1000 protection system automatically provides several safety-related actions to cool down and depressurize the reactor coolant system, terminate the break flow and steam release to the atmosphere, and stabilize the reactor coolant system in a safe condition. The safety-related actions include initiation of the PRHR system heat exchanger, isolation of the chemical and volume control system pumps and pressurizer heaters, and isolation of the startup feedwater pumps. In addition, the protection and safety monitoring system provides a safety-related signal to trip the redundant, nonsafety related pressurizer heater breakers.

Actuating the PRHR heat exchanger transfers core decay heat to the in-containment reactor water storage tank (IRWST) and initiates a cooldown (and a consequential depressurization) of the reactor coolant system.

Isolation of the chemical and volume control system pumps and pressurizer heaters minimizes the repressurization of the primary system. This allows primary pressure to equilibrate with the secondary pressure, which effectively terminates the primary-to-secondary break flow. Because the core makeup tank continues to inject when needed to provide boration following isolation of the chemical and volume control system pumps, isolating the chemical and volume control system pumps does not present a safety concern.

Isolation of the startup feedwater provides protection against a failure of the startup feedwater control system, which could potentially result in the ruptured steam generator being overfilled.

With decay heat removal by the PRHR heat exchanger, steam generator steaming through the power-operated relief valves ceases and steam generator secondary level is maintained.

15.6.3.1.4 Steam Generator Tube Rupture Assuming Operator Recovery Actions

In the event of an SGTR, the operators can diagnose the accident and perform recovery actions to stabilize the plant, terminate the primary-to-secondary leakage, and proceed with orderly shutdown of the reactor before actuation of the automatic protection systems. The operator actions for SGTR recovery are provided in the plant emergency operating procedures. The major operator actions include the following:

- Identify the ruptured steam generator – The ruptured steam generator can be identified by an unexpected increase in steam generator narrow range level or a high radiation indication from any main steam line monitor, steam generator blowdown line monitor, or steam generator sample.

- Isolate the ruptured steam generator – Once the steam generator with the ruptured tube is identified, recovery actions begin by isolating steam flow from and stopping feedwater flow to the ruptured steam generator.

- Cooldown of the reactor coolant system using the intact steam generator or the PRHR system – After isolation of the ruptured steam generator, the reactor coolant system is cooled as rapidly as possible to less than the saturation temperature corresponding to the ruptured steam generator pressure. This provides adequate subcooling in the reactor coolant system after depressurization of the reactor coolant system to the ruptured steam generator pressure in subsequent actions.

- Depressurize the reactor coolant system to restore reactor coolant inventory – When the cooldown is completed, the chemical and volume control system and core makeup tank injection flow increases the reactor coolant system pressure until break flow matches the total injection flow. Consequently, these flows must be terminated or controlled to stop primary-to-secondary leakage. However, adequate reactor coolant inventory must first be provided. This includes both sufficient reactor coolant subcooling and pressurizer inventory to maintain a reliable pressurizer level indication after the injection flow is stopped.

Because leakage from the primary side continues after the injection flow is stopped, until reactor coolant system and ruptured steam generator pressures equalize, the reactor coolant system is depressurized to provide sufficient inventory to verify that the pressurizer level remains on span after the pressures equalize.

- Termination of the injection flow to stop primary to secondary leakage – The previous actions establish adequate reactor coolant system subcooling, a secondary side heat sink, and sufficient reactor coolant inventory to verify that injection flow is no longer needed. When these actions are completed, core makeup tank and chemical and volume control system flow is stopped to terminate primary-to-secondary leakage. Primary-to-secondary leakage continues after the injection flow is stopped until the reactor coolant system and ruptured steam generator pressures equalize. Chemical and volume control system makeup flow, letdown, pressurizer heaters, and decay heat removal via the intact steam generator or the PRHR heat exchanger are then controlled to prevent repressurization of the reactor coolant system and reinitiation of leakage into the ruptured steam generator.

Following the injection flow termination, the plant conditions stabilize and the primary-to-secondary break flow terminates. At this time, a series of operator actions is performed to prepare the plant for cooldown to cold shutdown conditions. The actions taken depend on the available plant systems and the plan for further plant repair and operation.

15.6.3.2 Analysis of Effects and Consequences

An SGTR results in the leakage of contaminated reactor coolant into the secondary system and subsequent release of a portion of the activity to the atmosphere. An analysis is performed to demonstrate that the offsite radiological consequences resulting from an SGTR are within the allowable guidelines.

One of the concerns for an SGTR is the possibility of steam generator overfill because this can potentially result in a significant increase in the offsite radiological consequences. Automatic protection and passive design features are incorporated into the AP1000 design to automatically terminate the break flow to prevent overfill during an SGTR. These features include actuation of the PRHR system, isolation of chemical and volume control system flow, and isolation of startup feedwater.

An analysis is performed, without modeling expected operator actions to isolate the ruptured steam generator and cool down and depressurize the reactor coolant system, to demonstrate the role that the AP1000 design features have in preventing steam generator overfill. The limiting single failure for the overfill analysis is assumed to be the failure of the startup feedwater control valve to throttle flow when nominal steam generator level is reached. Other conservative assumptions that maximize steam generator secondary volume (such as high initial steam generator level, minimum initial reactor coolant system pressure, loss of offsite power, maximum chemical and volume control system injection flow, maximum pressurizer heater addition, maximum startup feedwater flow, and minimum startup feedwater delay time) are also assumed.

The results of this analysis demonstrate the effectiveness of the AP1000 protection system and passive system design features and support the conclusion that an SGTR event would not result in steam generator overfill.

For determining the offsite radiological consequences, an SGTR analysis is performed assuming the limiting single failure and limiting initial conditions relative to offsite doses. Because steam generator overfill is prevented for the AP1000, the results of this analysis represent the limiting radiological consequences for an SGTR.

A thermal-hydraulic analysis is performed to determine the plant response for a design basis SGTR, the integrated primary-to-secondary break flow, and the mass releases from the ruptured and intact steam generators to the condenser and to the atmosphere. This information is then used to calculate the radioactivity release to the environment and the resulting radiological consequences.

15.6.3.2.1 Method of Analysis

15.6.3.2.1.1 Computer Program

The plant response following an SGTR until the primary-to-secondary break flow is terminated is analyzed with the LOFTTR2 program (Reference 21). The LOFTTR2 program is modified to model the PRHR system, core makeup tanks, and protection system actions appropriate for the AP1000. These modifications to LOFTTR2 are described in WCAP-14234, Revision 1 (Reference 14).

15.6.3.2.1.2 Analysis Assumptions

The accident modeled is a double-ended break of one steam generator tube located at the top of the tube sheet on the outlet (cold leg) side of the steam generator. The location of the break on the cold leg side of the steam generator results in higher initial primary-to-secondary leakage than a break on the hot side of the steam generator.

The reactor is assumed to be operating at full power at the time of the accident, and the initial secondary mass is assumed to correspond to operation at nominal steam generator mass minus an allowance for uncertainties. Offsite power is assumed to be lost and the rods are assumed to be inserted at the start of the event because continued operation of the reactor coolant pumps has been determined to reduce flashing of primary-to-secondary break flow and, consequently, lower offsite radiological doses. Maximum chemical and volume control system flows and pressurizer heater heat addition are assumed immediately (even though offsite power is not available) to conservatively maximize primary-to-secondary leakage. The steam dump system is assumed to be inoperable, consistent with the loss of offsite power assumption, because this results in steam release from the steam generator power-operated relief valves to the atmosphere following reactor trip. The chemical and volume control system and pressurizer heater modeling is conservatively chosen to delay the low pressurizer pressure “S” and the low-2 pressurizer level signal and associated protection system actions.

The limiting single failure is assumed to be the failure of the ruptured steam generator power-operated relief valve. Failure of this valve in the open position causes an uncontrolled depressurization of the ruptured steam generator, which increases primary-to-secondary leakage and the mass release to the atmosphere.

It is assumed that the ruptured steam generator power-operated relief valve fails open when the low-2 pressurizer level signal is generated. This results in the maximum integrated flashed primary-to-secondary break flow.

The valve is subsequently isolated when the associated block valve is automatically closed on a low steam line pressure protection system signal.

No operator actions are modeled in this limiting analysis, and the plant protection system provides the protection for the plant. Not modeling operator actions is conservative because the operators are expected to have sufficient time to recover from the accident and supplement the automatic protection system. In particular, the operator would take action to reduce the primary pressure before the high steam generator level coincident with reactor trip (P-4) chemical and volume control and startup feedwater system shutoff signals are generated. It is also expected that the operator can close the block valve to the ruptured steam generator power-operated relief valve in much shorter time than the automatic protection signal. The operators can quickly diagnose a power-operated relief valve failure based on the rapid depressurization of the steam generator and increase in steam flow. They can then close the block valve from the control panel.

Consistent with the assumed loss of offsite power, the main feedwater pumps coast down and no startup feedwater is assumed to conservatively minimize steam generator secondary inventory and thus maximize secondary activity concentration and steam release.

15.6.3.2.1.3 Results

The sequence of events for this transient is presented in Table 15.6.3-1. The system responses to the SGTR accident are shown in Figures 15.6.3-1 to 15.6.3-10.

Offsite power is lost concurrent with the rupture of the tube. The reactor trips due to the loss of offsite power. The main feedwater pumps are assumed to coast down following reactor trip. The startup feedwater pumps are conservatively assumed not to start. Following the tube rupture, reactor coolant flows from the primary into the secondary side of the ruptured steam generator. In response to this loss of reactor coolant, pressurizer level and reactor coolant system pressure decreases as shown in Figures 15.6.3-1 and 15.6.3-2. As a result of the decreasing pressurizer level and pressure, two chemical and volume control system pumps are automatically initiated to provide makeup flow and the pressurizer heaters turn on.

After reactor trip, core power rapidly decreases to decay heat levels and the core inlet to outlet temperature differential decreases. The turbine stop valves close, and steam flow to the turbine is terminated. The steam dump system is conservatively assumed to be inoperable. The secondary side pressure increases rapidly after reactor trip until the steam generator power-operated relief valves (and safety valves, if their setpoints are reached) lift to dissipate the energy, as shown in Figure 15.6.3-3.

Maximum heat addition to the pressurizer from the pressurizer heaters increases the primary pressure.

As the leak flow continues to deplete primary inventory, low pressurizer level “S” and core makeup tank and PRHR actuation signals are reached. Power to the pressurizer heaters is shut off so that they will not provide additional heat to the primary should the pressurizer level return. The ruptured steam generator power-operated relief valve is assumed to fail open at this time.

The failure causes the intact and ruptured steam generators to rapidly depressurize (Figure 15.6.3-3). This results in an initial increase in primary-to-secondary leakage and a decrease in the reactor coolant system temperatures. Both the intact and ruptured steam generators depressurize because the steam generators communicate through the open steam line isolation valves.

The decrease in the reactor coolant system temperature results in a decrease in the pressurizer level and reactor coolant system pressure (Figures 15.6.3-1 and 15.6.3-2). Depressurization of the primary and secondary systems continues until the low steam line pressure setpoint is reached. As a result, the steam line isolation valves and intact and ruptured steam generator power-operated relief block valves are closed.

Following closure of the block valves, the primary and secondary pressures and the ruptured steam generator secondary water volume and mass increase as break flow accumulates. This increase continues until the steam generator secondary level reaches the high narrow range level when the chemical and volume control and startup feedwater systems are isolated.

With continued reactor coolant system cooldown, depressurization provided by the PRHR heat exchanger, and with the chemical and volume control system isolated, primary system pressure eventually falls to match the secondary pressure. The break flow terminates as shown in Figure 15.6.3-5, and the system is stabilized in a safe condition. As shown in Figure 15.6.3-8, steam release through the intact loop, unfaulted power-operated relief valve does not occur following PRHR initiation because the PRHR is capable of removing the core decay heat.

As shown in Figure 15.6.3-9, the core makeup tank flow trends toward zero because the gravity head diminishes as the core makeup tank temperature approaches the reactor coolant system temperature due to the continued balance line flow. The core makeup tank remains full, and ADS actuation does not occur.

The ruptured steam generator water volume is shown in Figure 15.6.3-6. The water volume in the ruptured steam generator when the break flow is terminated is significantly less than the total steam generator volume of greater than 8868 ft^3.

The design basis SGTR event does not result in fuel failures. In the event of an SGTR, the reactor coolant system depressurizes due to the primary-to-secondary leakage through the ruptured steam generator tube. This depressurization reduces the calculated DNBR. The depressurization prior to reactor trip for the SGTR has been compared to the depressurization for the reactor coolant system depressurization accidents analyzed in Subsection 15.6.1. The rate of depressurization is much slower for the SGTR than for the reactor coolant system depressurization accidents. Following reactor trip, the DNBR increases rapidly. Thus, the conclusion of Subsection 15.6.1, that the calculated DNBR remains above the limit, is extended to the SGTR analysis, justifying the assumption of no failed fuel.

15.6.3.2.1.4 Mass Releases

The mass release of an SGTR event is determined for use in evaluating the exclusion area boundary and low population zone radiation exposure. The steam releases from the ruptured and intact steam generators and the primary-to-secondary leakage into the ruptured steam generator are determined from the LOFTTR2 results for the period from the initiation of the accident until the leakage is terminated.

Following reactor trip, the releases to the atmosphere are through the steam generator power-operated relief valves (and steam generator safety valves for a short period). Steam relief through the power-operated relief valves continues until RNS conditions are met. The mass releases for the SGTR event are presented in Table 15.6.3-2.

15.6.3.3 Radiological Consequences

The evaluation of the radiological consequences of the postulated SGTR assumes that the reactor is operating with the design basis fuel defect level (0.25 percent of power produced by fuel rods containing cladding defects) and that leaking steam generator tubes result in a buildup of activity in the secondary coolant.

Following the rupture, any noble gases carried from the primary coolant into the ruptured steam generator via the break flow are released directly to the environment. The iodine and alkali metal activity entering the secondary side is also available for release, with the amount of release dependent on the flashing fraction of the reactor coolant and on the partition coefficient in the steam generator. In addition to the activity released through the ruptured loop, there is also a small amount of activity released through the intact loop.

15.6.3.3.1 Source Term

The significant radionuclide releases from the SGTR are the noble gases, alkali metals and the iodines that become airborne and are released to the environment as a result of the accident.

The analysis considers two different reactor coolant iodine source terms, both of which consider the iodine spiking phenomenon. In one case, the initial iodine concentrations are assumed to be those associated with the equilibrium operating limits for primary coolant iodine activity. The iodine spike is assumed to be initiated by the accident with the spike causing an increasing level of iodine in the reactor coolant.

The second case assumes that the iodine spike occurs before the accident and that the maximum reactor coolant iodine concentration exists at the time the accident occurs.

The reactor coolant noble gas and alkali metal concentrations are assumed to be those associated with the design fuel defect level.

The secondary coolant iodine and alkali metal activity is assumed to be 1 percent of the maximum equilibrium primary coolant activity.

15.6.3.3.2 Release Pathways

The noble gas activity contained in the reactor coolant that leaks into the intact steam generator and enters the ruptured steam generator through the break is assumed to be released immediately as long as a pathway to the environment exists. There are three components to the modeling of iodine and alkali metal releases:

- Intact loop steaming, with credit for partitioning of iodines and alkali metals (includes continued primary-to-secondary leakage at the maximum rate allowable by the Technical Specifications)
- Ruptured loop steaming, with credit for partitioning of iodines and alkali metals (includes modeling of increasing activity in the secondary coolant due to the break flow)

- Release of flashed reactor coolant through the ruptured loop, with no credit for scrubbing (this conservatively assumes that break location is at the top of the tube bundle)

Credit is taken for decay of radionuclides until release to the environment. After release to the environment, no consideration is given to radioactive decay or to cloud depletion of iodines by ground deposition during transport offsite.

15.6.3.3.3 Dose Calculation Models

The models used to calculate doses are provided in Appendix 15A.

15.6.3.3.4 Analytical Assumptions and Parameters

The assumptions and parameters used in the analysis are listed in Table 15.6.3-3.

15.6.3.3.5 Identification of Conservatisms

The assumptions used in the analysis contain a number of significant conservatisms, such as:

- The reactor coolant activities are based on a fuel defect level of 0.25 percent; whereas, the expected fuel defect level is far less (see Section 11.1).

- It is unlikely that the conservatively selected meteorological conditions are present at the time of the accident.

15.6.3.3.6 Doses

Using the assumptions from Table 15.6.3-3, the calculated TEDE doses for the case in which the iodine spike is assumed to be initiated by the accident are determined to be 0.7 rem at the exclusion area boundary for the limiting 2-hour interval (0-2 hours) and 0.5 rem at the low population zone outer boundary. These doses are a small fraction of the dose guideline of 25 rem TEDE identified in 10 CFR Part 50.34. A “small fraction” is defined, consistent with the Standard Review Plan, as being ten percent or less.

For the case in which the SGTR is assumed to occur coincident with a pre-existing iodine spike, the TEDE doses are determined to be 1.4 rem at the exclusion area boundary for the limiting 2-hour interval (0 to 2 hours) and 0.7 rem at the low population zone outer boundary. These doses are within the dose guideline of 25 rem TEDE identified in 10 CFR Part 50.34.

At the time the accident occurs, there is the potential for a coincident loss of spent fuel pool cooling with the result that the pool could reach boiling and a portion of the radioactive iodine in the spent fuel pool could be released to the environment. The loss of spent fuel pool cooling has been evaluated for a duration of 30 days. There is no contribution to the 2-hour exclusion area boundary dose because pool boiling would not occur until after 2.0 hours. The 30-day contribution to the dose at the low population zone boundary is less than 0.01 rem TEDE and, when this is added to the doses calculated for the steam generator tube rupture, the resulting total doses remain as reported above.

15.6.3.4 Conclusions

The results of the SGTR analysis show that the overfill protection logic and the passive system design features provide protection to prevent steam generator overfill. Following an SGTR accident, the operators can identify and isolate the ruptured steam generator and complete the required actions to terminate the primary-to-secondary break flow before steam generator overfill or ADS actuation occurs.

Even when no operator actions are assumed, the AP1000 protection system and passive design features initiate automatic actions that can terminate a steam generator tube leak and stabilize the reactor coolant system in a safe condition while preventing steam generator overfill and ADS actuation.

The resulting offsite radiological doses for the limiting case analyzed are within the dose acceptance limits.

15.6.4 Spectrum of Boiling Water Reactor Steam System Piping Failures Outside of Containment

This section is not applicable to the AP1000.

15.6.5 Loss-of-Coolant Accidents Resulting from a Spectrum of Postulated Piping Breaks Within the Reactor Coolant Pressure Boundary

15.6.5.1 Identification of Causes and Frequency Classification

A LOCA is the result of a pipe rupture of the reactor coolant system pressure boundary. For the analyses reported here, a major pipe break (large break) is defined as a rupture with a total cross-sectional area equal to or greater than 1.0 ft^2. This event is considered a Condition IV event (a limiting fault) because it is not expected to occur during the lifetime of the plant but is postulated as a conservative design basis (see Subsection 15.0.1).

A minor pipe break (small break), as considered in this subsection, is defined as a rupture of the reactor coolant pressure boundary (Section 5.2) with a total cross-sectional area less than 1.0 ft^2 in which the normally operating charging system flow is not sufficient to sustain pressurizer level and pressure. This is considered a Condition III event because it is an infrequent fault that may occur during the life of the plant.

The acceptance criteria for the LOCA are described in 10 CFR 50.46 (Reference 1) as follows:

- The calculated maximum fuel element cladding temperature shall not exceed 2200°F.
- Localized cladding oxidation shall not exceed 17 percent of the total cladding thickness before oxidation.
- The amount of hydrogen generated from fuel element cladding reacting chemically with water or steam shall not exceed 1 percent of the total amount if all metal cladding were to react.
- The core remains amenable to cooling for any calculated change in core geometry.
- The core temperature is maintained at a low value, and decay heat is removed for the extended period of time required by the long-lived radioactivity remaining in the core.

These criteria are established to provide significant margin in emergency core cooling system performance following a LOCA.

For the AP1000, the small breaks (less than 1.0 ft^2) yield results with more margin than large breaks.

15.6.5.2 Basis and Methodology for LOCA Analyses

Should a major break occur, depressurization of the reactor coolant system results in a pressure decrease in the pressurizer. The reactor trip signal subsequently occurs when the pressurizer low-pressure trip setpoint is reached. A safeguards actuation ("S") signal is generated when the appropriate setpoint is reached. These measures limit the consequences of the accident in two ways:

- Reactor trip and borated water injection complement void formation in causing rapid reduction of power to a residual level corresponding to fission product decay heat. Insertion of control rods to shut down the reactor is neglected in the large-break analysis.

- Injection of borated water provides core cooling and prevents excessive cladding temperatures.

The acceptability of the computer codes approved for AP600 LOCA analyses for the AP1000 application is documented in Reference 24. The acceptability of additional computer codes for the AP1000 Best-Estimate Large-Break LOCA analysis is documented in Reference 34.

15.6.5.2.1 Description of Large-Break LOCA Transient

Before the break occurs, the unit is in an equilibrium condition in which the heat generated in the core is being removed via the secondary system. During blowdown, heat from fission product decay stored energy in the fuel, hot internals, and vessel continues to be transferred to the reactor coolant. At the beginning of the blowdown phase, the entire reactor coolant system contains subcooled liquid, which transfers heat from the core by forced convection with some fully developed nucleate boiling. After the break, the core heat transfer is based upon local fluid conditions. Transition boiling and dispersed flow film boiling are the major heat transfer mechanisms.

The heat transfer between the reactor coolant system and the secondary system may be in either direction, depending upon the relative temperatures. In the case of continued heat addition to the secondary system, secondary system pressure increases and the main steam safety valves may lift to limit the pressure. The safety injection signal actuates a feedwater isolation signal, which isolates normal feedwater flow by closing the main feedwater isolation valves.

The reactor coolant pumps trip automatically during the accident following an "S" signal. The effects of pump coastdown are included in the blowdown. The blowdown phase of the transient ends when the reactor coolant system pressure (initially assumed at 2250 psia) falls to a value approaching that of the containment atmosphere.

When the "S" signal occurs, the core makeup tank isolation valves are opened. The core makeup tank begins to inject subcooled borated water into the reactor vessel through the direct vessel injection lines.

Subsection 15.6.5.4C presents calculations that show the effective post-LOCA long-term cooling of the AP1000 by passive means.

15.6.5.2.2 Description of Small-Break LOCA Transient

The AP1000 includes passive safety features to prevent or minimize core uncovery during small-break LOCAs. The passive safety design approach of the AP1000 is to depressurize the reactor coolant system if the break or leak is greater than the makeup capability of the charging system. By depressurizing the reactor system, large volumes of borated water in the accumulators and in the IRWST become available for cooling the core. This analysis demonstrates that, with a single failure, the passive systems are capable of depressurizing the reactor coolant system while maintaining acceptable core conditions and establishing stable delivery of cooling water from the IRWST.

During a small-break LOCA, the AP1000 reactor coolant system depressurizes to the pressurizer low-pressure setpoint, actuating a reactor trip signal. The passive core cooling system is aligned for delivery following the generation of an “S” signal when the pressurizer low-pressure setpoint is reached. The passive core cooling system includes two core makeup tanks, two accumulators, a large IRWST, and the PRHR heat exchanger.

The core makeup tanks operate at reactor coolant system pressure. They provide high-pressure safety injection in the event of a small-break LOCA. The core makeup tanks share a common discharge line with the accumulators and IRWST; they are filled with borated water to provide core shutdown margin. The injection of the core makeup tanks is provided by gravity head of the colder water in the core makeup tanks. The core makeup tanks are located above the reactor coolant loops, and each is equipped with a pressure balancing line from a cold leg to the top of the tank.

The pressurized accumulators provide additional borated water to the reactor coolant system in the event of a LOCA. Nominally, these 2000-ft^3 tanks are filled with 1700 ft^3 of water and 300 ft^3 of nitrogen at an initial pressure of 700 psig. Once sufficient reactor coolant system depressurization occurs, either as a result of a LOCA or the actuation of the ADS, accumulator injection commences.

The IRWST provides an additional source of water for long-term core cooling. To attain injection from the IRWST, the reactor coolant system pressure must be lowered to approximately 13 psi above containment pressure. For this pressure to be achieved during a small-break LOCA, the ADS system is initiated.

The ADS consists of a series of valves, connected to the pressurizer and hot legs, which provide a phased depressurization of the reactor coolant system. As the reactor system loses inventory through the break, the core makeup tanks provide flow to the reactor vessel. When the level in the core makeup tank drops to the 67.5-percent level, the ADS valves open to accelerate the reactor coolant system depressurization rate. The ADS Stage 1 4-inch valves open at the 67.5-percent level; the 8-inch Stage 2 and the 8-inch Stage 3 valves open in a timed sequence thereafter. The flow from the first three stages of the ADS is discharged into the IRWST through a sparger system. The fourth stages of the ADS are connected to the reactor coolant system hot legs and discharge to containment atmosphere. The ADS Stage 4 valves are activated when the core makeup tank level reaches the 20-percent level.

As the reactor coolant system depressurizes and mass is lost out the break, mass is added to the reactor vessel from the core makeup tanks and the accumulators. When the system is depressurized below the IRWST delivery pressure, flow from the IRWST continues to maintain the core in a coolable state. Calculations described in Subsection 15.6.5.4B indicate that acceptable core cooling is provided for the small-break LOCA transients. Subsection 15.6.5.4C calculations show that effective post-LOCA core cooling is provided in the long term by passive means.

15.6.5.3 Radiological Consequences

Although the analysis of the core response during a LOCA (see Subsection 15.6.5.4) shows that core integrity is maintained, for the evaluation of the radiological consequences of the accident, it is assumed that major core degradation and melting occur.

The dose calculations take into account the release of activity by way of the containment purge line prior to its isolation near the beginning of the accident and the release of activity resulting from containment leakage. Purge of the containment for hydrogen control is not an intended mode of operation and is not considered in the dose analysis. While the normal residual heat removal system is capable of post-LOCA cooling, it is not a safety-related system and may not be available following the accident. If it is operable, it would be used only if the source term is not far above the normal shutdown primary coolant source term. It is assumed that core cooling is accomplished by the passive core cooling system, which does not pass coolant outside of containment. Thus, there is no recirculation leakage release path to be modeled.

15.6.5.3.1 Source Term

The release of activity to the containment consists of two parts. The initial release is the activity contained in the reactor coolant system. This is followed by the release of core activity.

15.6.5.3.1.1 Primary Coolant Release

The reactor coolant is assumed to have activity levels consistent with operation at the Technical Specification limits of 280 μCi/gm dose equivalent Xe-133 and 1.0 μCi/gm dose equivalent I-131.

Based on NUREG-1465 (Reference 19), for a plant using leak-before-break methodology, the release of coolant into the containment can be assumed to last for 10 minutes. The AP1000 is a leak-before-break plant, and the water in the reactor coolant system is assumed to blow down into the containment over a period of 10 minutes. The flow rate is assumed to be constant over the 10-minute period. As the reactor coolant enters the containment, the noble gases and half of the iodine activity are assumed to be released into the containment atmosphere.

15.6.5.3.1.2 Core Release

The release of activity from the fuel takes place in two stages as summarized in Table 15.6.5-1. First is the gap release which is assumed to occur at the end of the primary coolant release phase (i.e., at ten minutes into the accident) and continue over a period of half an hour. The second stage is that of the in-vessel core melt in which the bulk of the activity releases associated with the accident occur. The source term model is based on NUREG-1465 and Regulatory Guide 1.183 (Reference 20).

The core fission product inventory at the time of the accident is based on operation near the end of a fuel cycle at 102-percent power and is provided in Table 15A-3 of Appendix 15A. The main feedwater flow measurement supports a 1-percent power uncertainty; use of a 2-percent power uncertainty is conservative. Consistent with NUREG-1465, there are three groups of nuclides considered in the gap activity releases: noble gases, iodines, and alkali metals (cesium and rubidium). For the core melt phase, there are five additional nuclide groups for a total of eight. The five additional nuclide groups are the tellurium group, the noble metals group, the cerium group, the lanthanide group, and barium and strontium. The specific nuclides included in the source term are as shown in Table 15A-3.

Gap Activity Release

Consistent with NUREG-1465 guidance for a plant using leak-before-break methodology, the gap release phase begins after the primary coolant release phase ends at ten minutes and has a duration of 0.5 hour.

In-vessel Core Release

After the gap activity release phase, there is an in-vessel release phase which lasts for 1.3 hours and which releases activity to the containment due to core melting. The fractions of the core activity released to the containment atmosphere during this phase are from NUREG-1465:

Noble gases	0.95
Iodines	0.35
Alkali metals	0.25

Tellurium group	0.05
Noble metals	0.0025
Ba and Sr	0.02
Cerium group	0.0005
Lanthanide group	0.0002

Consistent with NUREG-1465, the releases are assumed to occur at a constant rate over the 1.3-hour phase duration.

15.6.5.3.1.3 Iodine Form

The iodine form is consistent with the NUREG-1465 model. The model shows the iodine to be predominantly in the form of nonvolatile cesium iodide with a small fraction existing as elemental iodine. Additionally, the model assumes that a portion of the elemental iodine reacts with organic materials in the containment to form organic iodine compounds. The resulting iodine species split is as follows:

- Particulate 0.95
- Elemental 0.0485
- Organic 0.0015

If the post-LOCA cooling solution has a pH of less than 6.0, part of the cesium iodide may be converted to the elemental iodine form. The passive core cooling system provides sufficient trisodium phosphate to the post-LOCA cooling solution to maintain the solution pH at 7.0 or greater following a LOCA (see Subsection 6.3.2.1.4).

15.6.5.3.2 In-containment Activity Removal Processes

The AP1000 does not include active systems for the removal of activity from the containment atmosphere. The containment atmosphere is depleted of elemental iodine and of particulates as a result of natural processes within the containment.

Elemental iodine is removed by deposition onto surfaces. Particulates are removed by sedimentation, diffusiophoresis (deposition driven by steam condensation), and thermophoresis (deposition driven by heat transfer). No removal of organic iodine is assumed. Appendix 15B provides a discussion of the models and assumptions used in calculating the removal coefficients.

Particulates removed from the containment atmosphere to the containment shell are assumed to be washed off the shell by the flow of water resulting from condensing steam (i.e. condensate flow). The particulates may be either washed into the sump, which is controlled to a pH ≥7 post-accident or into the IRWST, which is not pH controlled post-accident. Due to the conditions in the IRWST, a portion of the particulate iodine washed into the IRWST may chemically convert to an elemental form and re-evolve, subject to partitioning, as airborne. A water-steam partition factor of 10 for elemental iodine is applied. This value bounds the time-dependent partition factors calculated using the NUREG/CR-5950 (Reference 35) models and the calculated IRWST water temperature and pH as a function of time.

The IRWST is a closed tank with weighted louvers, and without boiling, there would be no motive force for the release of re-evolved gaseous iodine from the IRWST gas space to the containment. Thus, the assumption of boiling in the IRWST liquid is imposed to force the release of the re-evolved iodine to the containment atmosphere. A portion (3%) of the re-evolved elemental iodine is assumed to convert to an organic form upon its release to containment.

15.6.5.3.3 Release Pathways

The release pathways are the containment purge line and containment leakage. The activity releases are assumed to be ground level releases.

During the initial part of the accident, before the containment is isolated, it is assumed that containment purge is in operation and that activity is released through this pathway until the purge valves are closed. No credit is taken for the filters in the purge exhaust line.

The majority of the releases due to the LOCA are the result of containment leakage. The containment is assumed to leak at its design leak rate for the first 24 hours and at half that rate for the remainder of the analysis period.

15.6.5.3.4 Offsite Dose Calculation Models

The offsite dose calculation models are provided in Appendix 15A. The models address the determination of the TEDE doses from the combined acute doses and the committed effective dose equivalent doses.

The exclusion area boundary dose is calculated for the 2-hour period over which the highest doses would be accrued by an individual located at the exclusion area boundary. Because of the delays associated with the core damage for this accident, the first 2 hours of the accident are not the worst 2-hour interval for accumulating a dose.

The low population zone boundary dose is calculated for the nominal 30-day duration of the accident.

For both the exclusion area boundary and low population zone dose determinations, the calculated doses are compared to the dose guideline of 25 rem TEDE from 10 CFR Part 50.34.

15.6.5.3.5 Main Control Room Dose Model

There are two approaches used for modeling the activity entering the main control room. If power is available, the normal heating, ventilation, and air-conditioning (HVAC) system will switch over to a supplemental filtration mode (Section 9.4). The normal HVAC system is not a safety-class system but provides defense in depth.

Alternatively, if the normal HVAC is inoperable or, if operable, the supplemental filtration train does not function properly resulting in increasing levels of airborne iodine in the main control room, the emergency habitability system (Section 6.4) would be actuated when High-2 iodine or particulate activity is detected. The emergency habitability system provides passive pressurization of the main control room from a bottled air supply to prevent inleakage of contaminated air to the main control room. The bottled air also induces flow through the passive air filtration system which filters contaminated air in the main control room. There is a 72-hour supply of air in the emergency habitability system. After this time, the main control room is assumed to be opened and unfiltered air is drawn into the main control room by way of an ancillary fan. After 7 days, offsite support is assumed to be available to reestablish operability of the control room habitability system by replenishing the compressed air supply. As a defense-in-depth measure, the nonsafety-related normal control room HVAC would be brought back into operation with the supplemental filtration train if power is available.

The main control room is accessed by a vestibule entrance, which restricts the volume of contaminated air that can enter the main control room from ingress and egress. The design of the emergency habitability system (VES) provides 65 scfm ±5 scfm to the control room and maintains it in a pressurized state. The path for the purge flow out of the main control room is through the vestibule entrance and this should result in a dilution of the activity in the vestibule and a reduction in the amount of activity that might enter the main control room. However, no additional credit is taken for dilution of the vestibule via the purge. The projected inleakage into the main control room through ingress/egress is 5 cfm. An additional 10 cfm of unfiltered inleakage is conservatively assumed from other sources.

Activity entering the main control room is assumed to be uniformly dispersed. With the VES in operation, airborne activity is removed from the main control room atmosphere via the passive recirculation filtration portion of the VES.

The main control room dose calculation models are provided in Appendix 15A for the determination of doses resulting from activity which enters the main control room envelope.

15.6.5.3.6 Analytical Assumptions and Parameters

The analytical assumptions and parameters used in the radiological consequences analysis are listed in Table 15.6.5-2.

15.6.5.3.7 Identification of Conservatisms

The LOCA radiological consequences analysis assumptions include a number of conservatisms. Some of these conservatisms are discussed in the following subsections.

15.6.5.3.7.1 Primary Coolant Source Term

The source term is based on operation with the design fuel defect level of 0.25 percent; whereas, the expected fuel defect level is far less.

15.6.5.3.7.2 Core Release Source Term

The assumed core melt is a major conservatism associated with the analysis. In the event of a postulated LOCA, no major core damage is expected. Release of activity from the core is limited to a fraction of the core gap activity.

15.6.5.3.7.3 Atmospheric Dispersion Factors

The atmospheric dispersion factors assumed to be present during the course of the accident are conservatively selected. Actual meteorological conditions are expected to result in significantly higher dispersion of the released activity.

Site-specific X/Q (atmospheric dilution factor) values provided in Subsection 2.3.4 are bounded by the values given in Tables 15A-5 and 15A-6.

15.6.5.3.8 LOCA Doses

15.6.5.3.8.1 Offsite Doses

The doses calculated for the exclusion area boundary and the low population zone boundary are listed in Table 15.6.5-3. The doses are within the 10 CFR 50.34 dose guideline of 25 rem TEDE.

The reported exclusion area boundary doses are for the time period of 1.3 to 3.3 hours. This is the 2-hour interval that has the highest calculated doses. The dose that would be incurred over the first 2 hours of the accident is well below the reported dose.

At the time the LOCA occurs, there is the potential for a coincident loss of spent fuel pool cooling with the result that the pool could reach boiling and a portion of the radioactive iodine in the spent fuel pool could be released to the environment. The loss of spent fuel pool cooling has been evaluated for a duration of 30 days. There is no contribution to the 2-hour site boundary dose because pool boiling would not occur until after the limiting 2 hours. The 30-day contribution to the dose at the low population zone boundary is less than 0.01 rem TEDE and, when this is added to the dose calculated for the LOCA, the resulting total dose remains less than that reported in Table 15.6.5-3.

15.6.5.3.8.2 Doses to Operators in the Main Control Room

The doses calculated for the main control room personnel due to airborne activity entering the main control room are listed in Table 15.6.5-3. Also listed on Table 15.6.5-3 are the doses due to direct shine from the activity in the adjacent buildings, shine from radioactivity accumulated on the VES or VBS filters, and sky-shine from the radiation that streams out the top of the containment shield building and is reflected back down by air-scattering. The total of these dose paths is within the dose criteria of 5 rem TEDE as defined in GDC-19.

As discussed above for the offsite doses, there is the potential for a dose to the operators in the main control room due to iodine releases from postulated spent fuel boiling. The calculated dose from this source is less than 0.01 rem TEDE and is reported in Table 15.6.5-3.

15.6.5.4 Core and System Performance

Subsection 15.6.5.4A describes the large-break LOCA analysis methodology and results. Subsections 15.6.5.4B.1.0 through 15.6.5.4B.4.0 describe the small-break LOCA analysis methodology and results.

15.6.5.4A Large-Break LOCA Analysis Methodology and Results

Westinghouse applies the WCOBRA/TRAC computer code to perform best-estimate large-break LOCA analyses in compliance with 10 CFR 50 (Reference 5). WCOBRA/TRAC is a thermal-hydraulic computer code that calculates realistic fluid conditions in a PWR during the blowdown and reflood of a postulated large-break LOCA. The methodology used for the AP1000 analysis is documented in WCAP-12945-P-A, WCAP-14171, Revision 2, and WCAP-16009-P-A (References 10, 11, and 32).

The NRC staff has reviewed and approved the best-estimate LOCA methodology (ASTRUM methodology), as documented in the SER attached in front of Reference 32, for estimating the 95th percentile PCT for two-loop, three-loop and four-loop Westinghouse PWRs and the AP600. In Reference 3, the NRC staff has reviewed and approved a best-estimate LOCA methodology, as documented in Reference 11, for estimating the 95th percentile PCT for the AP600. In the Reference 32 and Reference 11 methodologies, the WCOBRA/TRAC code is used to calculate the effects of initial conditions, power distributions, and global models, and the HOTSPOT code is used to calculate the effects of local models.

In the ASTRUM uncertainty methodology (Reference 32), as used in the AP1000 LB LOCA analysis, global models and initial-condition, power-distribution, and local uncertainties are sampled independently for each of 124 runs over the same ranges of uncertainty and distributions as in References 10, 32, and 33, as described in Reference 34. The sampled global models, initial conditions, and power-distribution uncertainties become inputs to each of the 124 WCOBRA/ TRAC calculations. The thermal-hydraulic boundary conditions for the hot rod are input to the local uncertainties calculation performed by the HOTSPOT code.

Results from the 124 calculations are ranked by PCT from highest to lowest. A similar procedure is repeated for maximum local oxidation (MLO) and core wide oxidation (CWO). In order statistics as applied in the ASTRUM methodology, the limiting case for a parameter, such as peak cladding temperature (PCT), is a conservative estimate of the 95th percentile with 95 percent confidence. The limiting PCT, limiting MLO, and CWO may come from the same case or as many as three different cases because each parameter is assumed to be independent of the other two. The assumption of independence of the calculated licensing parameters is a conservative assumption because there is a dependence of MLO and CWO on cladding temperature.

For the AP1000 large-break LOCA analysis, the best-estimate LOCA analysis methodology is applied as described in Reference 34. The best-estimate large-break LOCA analysis complies with the stipulated applicability limits in the Reference 32 approval.

The post-LOCA long-term core cooling and core boron concentration analyses discussed in Subsection 15.6.5.4C are applicable to the large-break LOCA transient.

15.6.5.4A.1 General Description of WCOBRA/TRAC Modeling

WCOBRA/TRAC is the best-estimate thermal-hydraulic computer code used to calculate realistic fluid conditions in the PWR during blowdown and reflood of a postulated large-break LOCA.

The WCOBRA/TRAC Code Qualification Document (Reference 10) contains a complete description of the code models and justifies their applicability to PWR large-break LOCA analysis.

Table 15.6.5-4 lists the AP1000-specific parameters identified for use in the large-break LOCA analysis. WCOBRA/TRAC studies were performed for AP1000 to establish sensitivities to parameter variations. These studies included effects of ranging steam generator tube plugging, ranging the relative power in the low-power assemblies, loss of offsite power coincident with the break initiation, and break location. The calculated results were used to identify bounding conditions, which are then used in the AP1000 uncertainty calculations.

The WCOBRA/TRAC vessel nodalization is developed from plant design drawings to divide the vessel into 10 vertical sections. The bottom of section 1 is the inside vessel bottom, and the top of section 10 is the inside top of the vessel upper head. In addition to the major downcomer and core flow paths, the modeled bypass flow paths are the upper head cooling spray, guide thimbles, and core bypass. After defining the elevations for each section, a noding scheme is defined for the WCOBRA/TRAC model as shown in Reference 34. WCOBRA/TRAC assumes a vertical flow path for vertically stacked channels, unless specified otherwise in the input. Positive flow for the vertically connected channels (and cells) is upward. Several of the 10 sections are divided vertically into 2 or more levels; these levels are referred to as cells within a channel.

The WCOBRA/TRAC loop model represents the major primary, secondary, and passive safety systems components. Both loops are explicitly modeled, including the hot leg, the steam generator, and the two cold legs and associated pumps. The loop designated “1” has the pressurizer and the PRHR system connections, and loop “2” cold legs have the core makeup tank pressure balance line connections. The reactor coolant pump models contain the AP1000 homologous curves together with appropriate two-phase head and torque multipliers and degradation data. AP1000 values for pump coastdown characteristics are also applied. The passive safety features are modeled using design data for elevations, liquid volumes, and line losses. Because the ADS is not actuated until long after the time of PCT in large-break LOCA events, it is not modeled in detail.

15.6.5.4A.2 Steady-State Calculation

A WCOBRA/TRAC LOCA calculation is initiated from a point at which the flows, temperatures, powers, and pressures are at their approximate steady-state values before the postulated break occurs. Steady-state WCOBRA/TRAC calculations are run for a brief time period to verify that the calculated conditions are steady and that the desired reactor conditions are achieved.

The values used to set the steady-state plant conditions reflect the AP1000 parameters for reactor coolant pump flows, core power, and steam generator tube plugging levels. The fuel parameters provide the steady-state fuel temperatures, pressures, and gap conductances as a function of fuel burnup and linear power. The calculated fuel temperatures from WCOBRA/TRAC are adjusted to match the specified fuel data by adjusting the gap heat transfer coefficient between the pellet and the cladding. Once the vessel fluid temperatures, flows, pressures, loop pressure drop, and core parameters are in agreement with the desired values and are steady, a suitable initial condition is achieved.

15.6.5.4A.3 Signal Logic for Large-Break LOCA

The reactor trip signal occurs due to compensated pressurizer pressure within the first second of the large-break transient. Because control rod insertion is not modeled in WCOBRA/TRAC, no effects on reactivity ensue. A safeguards “S” signal occurs due to containment high pressure at 2.2 seconds of large-break LOCA transients.

As a consequence of this signal, after appropriate delays, the PRHR and core makeup tank isolation valves open and containment isolation occurs. The rapid depressurization of the primary system during a large-break LOCA leads to the initiation of accumulator injection early in the large-break transient. The accumulator flow diminishes core makeup tank delivery to such an extent that the core makeup tank level does not approach the ADS Stage 1 valve actuation point until after the accumulator tank is empty. The accumulator empties long after the blowdown portion of the large-break LOCA transient is complete. Actuation of the ADS on CMT water level does not occur until long after the AP1000 PCT is calculated to occur.

15.6.5.4A.4 Transient Calculation

Once the steady-state calculation is found to be acceptable, the transient calculation is initiated. The semi-implicit pipe break model is added to the desired break location. Cold-leg breaks are analyzed because the hot-leg break location is nonlimiting in the large-break LOCA best-estimate methodology. The break size and type are sampled consistent with the WCAP-16009-P-A (Reference 32) methodology. The containment backpressure is specified consistent with WCAP-16009-P-A (Reference 32) methodology. The steady-state calculation is restarted with the above changes to begin the transient.

The calculation is continued until the fuel rods are quenched.

Table 15.6.5-5 shows a general sequence of events following a large cold-leg break LOCA and the relationship of these events to the blowdown and reflood portion of the transient.

15.6.5.4A.5 Large-Break LOCA Analysis Results

For the AP1000 large-break LOCA analysis, the best-estimate LOCA analysis methodology documented in Reference 34 is applied. The AP1000 large-break LOCA analysis complies with the restrictions in Reference 32. AP1000 sensitivity calculations evaluated the sensitivity to the modeling of the CMT and PRHR relative to the reference transient configuration. A case in which the CMT was isolated from the rest of the AP1000 was analyzed, and the calculated PCT was lower than the PCT of the reference transient configuration. Also, a case in which the PRHR was isolated from the rest of the AP1000 was analyzed, and the calculated PCT was 2°F higher than the reference transient configuration. The ASTRUM methodology samples the parameters ranged in the global model matrix of calculations, and the final 95 percent uncertainty calculations have been performed for AP1000. Local and core-wide cladding oxidation values have been determined using the methodology approved in Reference 32.

In the AP1000 ASTRUM analysis, the same uncertainty calculation was the limiting PCT and maximum local oxidation (MLO) case. The limiting PCT/MLO case in the AP1000 ASTRUM analysis was a split break. Figures 15.6.5.4A-1 through 15.6.5.4A-12 present the parameters of principal interest for the limiting PCT/MLO case. Values of the following parameters are presented:

- Highest calculated cladding temperature at any elevation for the five fuel rods modeled
- Hot rod cladding temperature transient at the limiting elevation for PCT
- Core fluid mass flows at the top of the core for the fuel assemblies modeled in WCOBRA/TRAC
- Pressurizer pressure
- Break flow rates
- Core and downcomer collapsed liquid levels
- Accumulator water flow rates
- Core makeup tank flow rates

15.6.5.4A.6 Description of AP1000 Large-Break LOCA Transient

A description of the limiting PCT/MLO case from the AP1000 ASTRUM analysis follows. The limiting PCT/MLO case is a split break. The sequence of events is presented in Table 15.6.5-6. The break was modeled to occur in one of the cold legs in the loop containing the core makeup tanks. After the break opens, the vessel rapidly depressurizes and the core flow quickly reverses. The hot assembly fuel rods dry out and begin to heat up (Figures 15.6.5.4A-1 and 15.6.5.4A-2) after the initial flow reversal (Figure 15.6.5.4A-3).

In Figure 15.6.5.4A-1, “Hot Rod” refers to the hot rod at the maximum linear heat rate for the run, “Hot Assembly” refers to the average rod in the hot assembly that contains the hot rod, “Support Column/Open Hole” refers to the support column/open hole assembly average rod, “Guide Tubes” refers to the guide tube assembly average rod, and “Low Power” refers to the peripheral fuel assembly rod.

The steam generator secondaries are assumed to be isolated immediately at the inception of the break to maximize their stored energy. The massive size of the break causes an immediate, rapid pressurization of the containment. At 2.2 seconds, credit is taken for receipt of an “S” signal due to High-2 containment pressure. Applying the pertinent signal processing delay means that the valves isolating the core makeup tanks from the direct vessel injection line and the PRHR begin to open at 4.2 seconds into the transient. The reactor coolant pumps automatically trip after a 4 second delay from the actuation of the core makeup tank isolation valves at 8.2 seconds into the transient. Core shutdown occurs due to voiding; no credit is taken for the control rod insertion effect.

The system depressurizes rapidly (Figure 15.6.5.4A-4) as the initial mass inventory is depleted due to break flow. The pressurizer drains completely approximately 30 seconds into the transient, and accumulator injection commences 18 seconds into the transient (Figure 15.6.5.4A-5). Accumulator actuation shuts off core makeup tank flow (Figure 15.6.5.4A-6), which has been occurring since the isolation valve opened. The CMT liquid level remains well above the ADS Stage 1 actuation setpoint throughout the AP1000 LBLOCA cladding temperature excursion, even though CMT injection begins again around 150 seconds.

The dynamics of the 95th percentile estimator PCT/MLO case are shown in terms of the flow rates of liquid, vapor, and entrained liquid at the top of the core (Figures 15.6.5.4A-7 through 15.6.5.4A-9) for the peripheral, open hole/support column average power interior, and guide tube average power interior assemblies (the corresponding figure for the hot assembly is Figure 15.6.5.4A-3).

Figures 15.6.5.4A-8 and 15.6.5.4A-9 illustrate the impact of upper head drain through the guide tubes and upper core plate holes, respectively, on the core flow. While liquid remains in the upper head above the top of the guide tubes, the guide tubes (Figure 15.6.5.4A-8) are the preferred path for draining liquid into the upper plenum. Once the upper head begins to flash, liquid drains directly down the guide tubes and that fraction that is able to penetrate into the core does so, at a maximum flow rate exceeding 1000 lbm/sec of total liquid flow between 11 and 24 seconds. At that point, the flow entering the guide tubes in the upper head is largely steam; residual liquid is supplied to the guide tube fuel assemblies at a constant or decreasing rate out to 30 seconds.

Figure 15.6.5.4A-9 presents the open hole/support column assembly top of core flow behavior. The timing of the initial downflow into the open hole/support column assemblies is similar to that of the downflow into the guide tube fuel assemblies, beginning at 13 seconds. Between 19 and 24 seconds, the combined flow of continuous and entrained liquid is 300 to 1000 lbm/sec; the entrained liquid flow continues to be significant until 40 seconds.

Liquid downflow is delayed into the hot assembly. By 19 seconds into the transient, liquid that has built up in the global region above the hot assembly begins to flow into the hot assembly (Figure 15.6.5.4A-3). Significant flow of continuous liquid into the hot assembly exists between 19 to 24 seconds. The liquid flow is not enough to quench the hot rod and hot assembly rod at all elevations (Figure 15.6.5.4A-1) although effective cooling is achieved.

Figure 15.6.5.4A-7 demonstrates that liquid downflow exists through the top of the peripheral core assemblies from 8 seconds to 13 seconds and again from 16 seconds to 21 seconds in the 95th percentile estimator PCT/MLO case. The power of the fuel in this region is significantly lower than that of the fuel in the open hole and guide tube locations (Table 15.6.5-4), so liquid downflow occurs earlier than in the average power assemblies.

After 18 seconds into the transient, the accumulator begins to inject water into the upper downcomer region, most of which is initially bypassed to the break. The break flow rate diminishes as the transient progresses (Figure 15.6.5.4A-10). At 34.5 seconds, the accumulator injection begins to refill the lower plenum. At approximately 54.0 seconds, the lower plenum fills to the point that water begins to reflood the core from below. The void fraction at the core bottom begins to decrease, and as time passes, core cooling increases substantially. Figure 15.6.5.4A-11 presents the collapsed liquid levels in the core; Figure 15.6.5.4A-12 presents the collapsed liquid levels in the downcomer. The cladding temperature begins to decrease once the core water level has risen high enough in the core.

15.6.5.4A.7 Global Model Sensitivity Studies and Uncertainty Evaluation

Subsection 15.6.5.4A discusses the treatment of the global model parameters and the uncertainty evaluation in the ASTRUM methodology.

15.6.5.4A.8 Large-Break LOCA Conclusions

In accordance with 10 CFR 50.46, the conclusions of the best-estimate large-break LOCA analysis are that there is a high level probability that the following criteria are met.

1. The calculated maximum fuel element cladding temperature (i.e., peak cladding temperature (PCT)) will not exceed 2200°F.

2. The calculated total oxidation of the cladding (i.e., maximum cladding oxidation) will nowhere exceed 0.17 times the total cladding thickness before oxidation.

3. The calculated total amount of hydrogen generated from the chemical reaction of the cladding with water or steam (i.e., maximum hydrogen generation) will not exceed 0.01 times the hypothetical amount that would be generated if all of the metal in the cladding cylinders surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react.

4. The calculated changes in core geometry are such that the core remains amenable to cooling.

   Note that criterion 4 has historically been satisfied by adherence to criteria 1 and 2, and by assuring that fuel deformation due to combined LOCA and seismic loads is specifically addressed. Criteria 1 and 2 are satisfied for best-estimate large-break LOCA applications. The approved methodology specifies that effects of LOCA and seismic loads on core geometry do not need to be considered unless grid crushing extends beyond the assemblies in the low power channel as defined in the WCOBRA/TRAC model. This situation has not been calculated to occur for the AP1000. Therefore, acceptance criterion 4 is satisfied.

5. After successful initial operation of the emergency core cooling system (ECCS), the core temperature will be maintained at an acceptably low value and decay heat will be removed for the extended period of time required by the long-lived radioactivity remaining in the core.

   Criterion 5 is satisfied if a coolable core geometry is maintained and the core is cooled continuously following the LOCA. The AP1000 passive core cooling system provides effective core cooling following a large-break LOCA event, even assuming the limiting single failure of a core makeup tank delivery line isolation valve. The large-break LOCA transient has been extended beyond fuel rod quench until 1400 seconds, a time at which the CMT liquid level has decreased to the low-2 setpoint that actuates the fourth-stage ADS valves and IRWST injection. A significant increase in safety injection flow rate occurs when the IRWST becomes active. The analysis performed demonstrates that CMT injection is sufficient to maintain the mass inventory in the core and downcomer, from the period of fuel rod quench until IRWST injection. The AP1000 passive core cooling system provides effective post-LOCA long-term core cooling.

Table 15.6.5-8 presents the calculated 95th percentile PCT, maximum cladding oxidation, maximum hydrogen generation, and core cooling results.

Based on the analysis, the Westinghouse Best-Estimate Large-Break LOCA methodology has shown that the acceptance criteria of 10 CFR 50.46 are satisfied for AP1000.

15.6.5.4B Small-Break LOCA Analyses

Should a small break LOCA occur, depressurization of the reactor coolant system results in a pressure decrease in the pressurizer. The reactor trip signal occurs when the pressurizer low-pressure trip setpoint is reached. An "S" signal is generated when the appropriate setpoint is reached. These measures limit the consequences of the accident in two ways:

- Reactor trip leads to a rapid reduction of power to a residual level corresponding to fission product decay heat by the insertion of control rods to shut down the reactor.

- Injection of borated water provides core cooling and prevents excessive cladding temperatures.

15.6.5.4B.1 Description of Small-Break LOCA Transient

The AP1000 plant design includes passive safety features to prevent or minimize core uncovery during small-break LOCAs. The passive safety design approach of the AP1000 is to depressurize the reactor coolant system if the break or leak is greater than the capability of the makeup system or if the nonsafety makeup system fails to perform. By depressurizing the reactor system, large volumes of borated water in the accumulators and in the IRWST become available for cooling the core. This analysis demonstrates that, with a single failure, the passive systems are capable of depressurizing the reactor coolant system while maintaining acceptable core conditions and establishing stable delivery of cooling water from the IRWST.

During a small-break LOCA, the AP1000 reactor coolant system depressurizes to the pressurizer low-pressure setpoint, actuating a reactor trip signal. The passive core cooling system is aligned for delivery following the generation of an "S" signal when the pressurizer low-pressure setpoint is reached. The passive core cooling system includes two core makeup tanks, two accumulators, a large IRWST, and the PRHR heat exchanger.

The core makeup tanks operate at reactor coolant system pressure. They provide high-pressure safety injection in the event of a small-break LOCA. The core makeup tanks share a common discharge line with the accumulators and IRWST; they are filled with borated water to provide core shutdown margin. Gravity head of the colder water in the core makeup tanks provides the injection of the core makeup tanks. The core makeup tanks are located above the reactor coolant loops, and each is equipped with a pressure balancing line from a cold leg to the top of the tank.

The pressurized accumulators provide additional borated water to the reactor coolant system in the event of a LOCA. Nominally, these 2000-ft^3 tanks are filled with 1700 ft^3 of water and 300 ft^3 of nitrogen at an initial pressure of 700 psig. Once sufficient reactor coolant system depressurization occurs, either as a result of a LOCA or the actuation of the ADS, accumulator injection begins.

The IRWST at a minimum provides an additional 73,900 ft^3 of water for long-term core cooling. To attain injection from the IRWST, the reactor coolant system pressure must be lowered to approximately 13 psi above containment pressure. For this pressure to be achieved during a small-break LOCA, the actuation of the ADS valves is required.

The ADS consists of a series of valves, connected to the pressurizer and hot legs, which provide a phased depressurization of the reactor coolant system. As the reactor system loses inventory through the break, the core makeup tanks provide flow to the reactor vessel. When the level in the core makeup tank drops to the 67.5-percent level, the ADS valves open to accelerate the reactor coolant system depressurization rate. The ADS Stage 1 4-inch valves open at the 67.5-percent level; the 8-inch Stage 2 and the 8-inch Stage 3 valves open in a timed sequence thereafter. The flow from the first three stages of the ADS is discharged into the IRWST through a sparger system. The fourth stages of the ADS are connected to the reactor coolant system hot legs and discharge to containment atmosphere. The ADS Stage 4 valves are activated when the core makeup tank level reaches the 20-percent level.

As the reactor system depressurizes and mass is lost out the break, mass is added to the reactor vessel from the core makeup tanks and the accumulators. When the system is depressurized below the IRWST delivery pressure, flow from the IRWST continues to maintain the core in a coolable state. Calculations described in this section indicate that acceptable core cooling is provided for the small-break LOCA transients.

15.6.5.4B.2 Small-Break LOCA Analysis Methodology

Small-break LOCA response is evaluated for AP1000 with an evaluation model that conforms to 10 CFR 50 Appendix K. The elements of the AP1000 small-break LOCA evaluation model are the following:

- NOTRUMP computer code
- NOTRUMP homogeneous sensitivity model
- Critical heat flux assessment during accumulator injection

15.6.5.4B.2.1 NOTRUMP Computer Code

The NOTRUMP computer code is used in the analysis of LOCAs due to small-breaks in the reactor coolant system. The NOTRUMP computer code is a one-dimensional, general network code, which includes a number of advanced features. Among these features are the calculation of thermal non-equilibrium in all fluid volumes, flow regime-dependent drift flux calculations with counter-current flooding limitations, mixture level tracking logic in multiple-stacked fluid nodes, and regime-dependent heat transfer correlations. The version of NOTRUMP used in AP1000 small-break LOCA calculations has been validated against applicable passive plant test data (Reference 22). The code has limited capability in modeling upper plenum and hot leg entrainment and did not predict the core collapsed level during the accumulator injection phase adequately. The NOTRUMP homogeneous sensitivity model (discussed in Subsection 15.6.5.4B.2.2) and the critical heat flux assessment during the accumulator injection phase (discussed in Subsection 15.6.5.4B.2.3) supplement the base NOTRUMP analysis to demonstrate the adequacy of the design.

In NOTRUMP, the reactor coolant system is nodalized into volumes interconnected by flow paths. The transient behavior of the system is determined from the governing conservation equations of mass, energy, and momentum applied throughout the system. A description of NOTRUMP is given in References 12 and 13. The AP600 modeling approach, described in Reference 17, is also used to develop the AP1000 model; NOTRUMP's applicability to AP1000 is documented in Reference 24.

The use of NOTRUMP in the analysis involves the representation of the reactor core as heated control volumes with an associated bubble rise model to permit a transient mixture height calculation. The multi-node capability of the program enables an explicit and detailed spatial representation of various system components. Table 15.6.5-9 lists important input parameters and initial conditions of the analysis.

A steady-state input deck for the AP1000 was set up to comply, where appropriate, with the standard small-break LOCA Evaluation Model methodology. Major features of the modeling of the AP1000 follow:

- Accumulators are modeled at an initial pressure of 715 psia.

- The flow through the ADS links is modeled using the Henry-Fauske, the homogeneous equilibrium (HEM), and the Murdock/Baumann critical flow models. The Henry-Fauske correlation is used for low-quality two-phrase flow, and the HEM model, for high-quality flow, with a transition between the two beginning at 10-percent static quality. The Murdock-Bauman model is used if the ADS flow path is venting superheated steam.

- Isolation and check valves used in the passive safety systems are modeled.

- The IRWST is modeled as two connected fluid nodes. The lower node is connected to the direct vessel injection line and is the source of injection water to the DVI lines driven by gravity head. The upper node acts as a sink for the ADS flow from the pressurizer and as a heat sink for the PRHR heat exchanger. These nodes are modeled as having an initial temperature of 120°F, a pressure of 14.7 psia, and the nominal full-power operation level of 28.8 feet. Therefore, the minimum head for IRWST injection is assumed. For the DEDVI simulations, a conservative 20 psia containment pressure was used based on containment pressurization calculations performed with the WGOTHIC containment model.

- The PRHR system is modeled in accordance with the guidance provided in References 22 and 24. The PRHR isolation valve is modeled as opening with the maximum delay after the generation of an "S" signal to conservatively deny the cooling capability of the heat exchanger to the reactor coolant system for an extended period.

- The core power is initially set to 102 percent of the nominal core power. The main feedwater flow measurement supports a 1-percent power uncertainty; use of a 2-percent power uncertainty is conservative. The reactor trip signal occurs when the pressurizer pressure falls below 1800 psia. A conservative delay time is modeled between the reactor trip signal and reactor trip. Decay heat is modeled according to the ANS-1971 (Reference 2) standard, with 20-percent uncertainty added.

- The "S" signal is generated when the pressurizer pressure falls below 1700 psia. The isolation valves on the core makeup tank injection lines begin to open after the signal setpoint is reached; the valves are then assumed to open linearly. The main feedwater isolation valves are ramped closed between 2 and 7 seconds after the "S" signal. The reactor coolant pumps are tripped 6.0 seconds after the "S" signal.

- The ADS actuation signals are generated on low core makeup tank levels and the ADS timer delays. A list of the ADS parameters is given in Table 15.6.5-10 for AP1000. ADS Stages 1, 2, and 3 are modeled as discharging through spargers submerged in the IRWST at the appropriate depth.

- The pressure in the boundary node modeling of the containment is 14.7 psia in all NOTRUMP cases except the DEDVI line break, which used 20.0 psia.

- The steam generator secondary is isolated 6 seconds after the reactor trip signal, due to closure of the turbine stop valves. The main steam safety valves actuate and remove energy from the steam generator secondary when pressure reaches 1235 psia.

Active single failures of the passive safeguards systems are considered. The limiting failure is judged to be one out of four ADS Stage 4 valves failing to open on demand, the failure that most severely impacts depressurization capability. The safety design approach of the AP1000 is to depressurize the reactor coolant system to the containment pressure in an orderly fashion such that the large reservoir of water stored in the IRWST is available for core cooling. The mass inventory plots provided for the breaks show the minimum inventory condition generally occurs at the start of IRWST injection.

Penalizing the depressurization is the most conservative approach in postulating the single failure for such breaks.

The small-break LOCA spectrum analyzed for AP1000 includes a break that exhibits a minimum reactor vessel inventory early in the transient, before the accumulators become active: the 10-inch cold leg break. In this transient, the early mass inventory decrease is terminated by injection flow from the accumulators, and depressurization through the break enables accumulator injection to begin with no contribution from the actuation of ADS Stages 1, 2, and 3. For consistency, the conservative failure of one of the ADS Stage 4 valves located off the PRHR inlet pipe, which adversely affects the depressurization necessary to achieve IRWST injection in small-break LOCAs, is assumed in all cases. Sensitivity analysis shows that assuming failure of one ADS Stage 4 valve on the non-PRHR loop does not significantly impact core cooling.

15.6.5.4B.2.1.1 AP1000 Model-Detailed Noding

Refer to Reference 17 for details of the AP600 NOTRUMP modeling. The AP1000 model was developed in the same fashion with modifications to the AP600 model introduced as follows. A modification performed for AP1000 was the addition of two core nodes one foot each in length to reflect the added active fuel length of this design. The ADS-4 flow path resistances were increased to accommodate shortcomings in NOTRUMP identified during the integral test facility simulations, namely, the lack of a detailed momentum flux model in the ADS-4 discharge paths. A detailed calculation of the energy and momentum equations is performed for the ADS-4 piping over a range of flow and pressure conditions to provide a benchmark for the NOTRUMP ADS-4 flow path resistance. The methodology used to determine the resistance increase is described in Reference 24. By increasing the ADS-4 resistances, the onset of IRWST injection is more appropriately calculated. This methodology directly addresses the effect of momentum flux in ADS-4. The ADS-4 resistance increase utilized is computed for the NOTRUMP analyses in this section to be a 70 percent ADS-4 flow path resistance increase.

15.6.5.4B.2.1.2 Plant Initial Conditions/Steady-State

A steady-state calculation is performed prior to initiating the transient portion of the calculation.

Table 15.6.5-9 contains the most important initial conditions for the transient calculations. The behaviors of the primary pressure and pressurizer level, steam generator pressures, and the core flow rate are stable at the end of the 100-second steady-state calculation.

15.6.5.4B.2.2 NOTRUMP Homogeneous Sensitivity Model

In order to address the uncertainties associated with entrainment in the upper plenum and hot leg following ADS-4 operation, a sensitivity study is performed with the limiting break with respect to these phenomena, effectively maximizing the amount of entrainment downstream of the core. This methodology is described and the results are presented for the double-ended direct vessel injection (DEDVI) line break in detail in Reference 24.

[*In order to maximize the entrainment downstream of the core for the limiting break with respect to entrainment, NOTRUMP is run with the regions of the upper plenum, hot leg, and ADS-4 lines in a homogeneous fluid condition, with slip = 1, to demonstrate that even with maximum entrainment, the 10 CFR 50.46 criteria are met.*]*

15.6.5.4B.2.3 Critical Heat Flux Assessment During Accumulator Injection

[*An assessment is performed of the peak core heat flux with respect to the critical heat flux during the later ADS depressurization time period for a double-ended rupture of the direct vessel injection line.*

*NRC Staff approval is required prior to implementing a change in this information.

This time period corresponds to the accumulator injection phase of the transient. The predicted average mass flux at the core inlet and the reactor pressure from the NOTRUMP computer code base model analysis are used as input parameters to critical heat flux correlation as described in Reference 30. The requirements of 10 CFR 50.46 are met provided the maximum heat flux is less than the critical heat flux calculated by the correlation.]* NOTRUMP has been shown (Reference 24) to adequately predict mass flux and pressure for integral systems tests.

The predicted mass flux at the core inlet is on the average constant and corresponds to 7.2 lbm ft^{-2} s^{-1} (~35 kg m^{-2} s^{-1}). The key thermal-hydraulic parameters at different times during the ADS depressurization time period are summarized in following table.

Time (sec)	UP Pressure (kPa)	UP Pressure (psia)	Mass Flux (kg/m^2s)	Average Heat Flux (kW/m^2)
400	1310	190	35	20.2
500	655	95	35	19.1
570	345	50	35	18.5
600	276	40	35	18.2

For the critical heat flux assessment, the peak core heat flux is applied to simulate the hot assembly condition in a conservative manner. No credit is taken for increased flow in the hot assembly that is known to occur in rod bundles.

The correlation applied for this assessment is from vertical tube data (Reference 30) and recognizes two regimes depending on the mass flux. The main difference between the two is the mass flux dependence. They are as follows:

$$q^*_{CL} = q^*_{CF} + 0.01351\left(D^*\right)^{-0.473}(L/D)^{-0.533}\left|G^*\right|^{1.45} \quad \text{for low } G^*$$

and,

$$q^*_{CH} = q^*_{CF} + 0.05664\left(D^*\right)^{-0.247}(L/D)^{-0.501}\left|G^*\right|^{0.77} \quad \text{for high } G^*$$

The first term of above correlations is,

$$q^*_{CF} = 1.61\left(\frac{A}{Ah}\right)\frac{\left(D^*\right)^{0.5}}{\left[1+\left(\frac{\rho_g}{\rho_l}\right)^{0.25}\right]^2}$$

where A is the flow area and Ah is the heated area.

The dimensionless CHF is calculated as,

$$q^*_{CHF} = \min(q^*_{CL}, q^*_{CH})$$

Dimensionless CHF, G, and D are defined as,

$$q^*_{CHF} = \frac{q''_{CHF}}{h_{fg}\sqrt{\lambda \rho_g g \Delta\rho}}$$

$$G^* = \frac{G}{\sqrt{\lambda \rho_g g \Delta\rho}}$$

$$D^* = \frac{D}{\lambda}$$

where λ is the length scale of the Taylor instability:

$$\lambda = \sqrt{\frac{\sigma}{g\Delta\rho}}$$

Conservative application of this correlation with the AP1000 parameters indicates that the peak AP1000 heat flux during this period is at least 40 percent below the predicted critical heat flux.

This CHF assessment addresses core cooling during a time period where the NOTRUMP computer code may not conservatively predict the core average void fraction. The requirements of 10 CFR 50.46 are met during this period since this CHF assessment indicates peak core heat flux is less than critical heat flux. Cladding temperatures will remain near the coolant saturation temperature, well below the 10 CFR 50.46 peak cladding temperature limit.

15.6.5.4B.3 Small-Break LOCA Analysis Results

Several small-break LOCA transients are analyzed using NOTRUMP, and the results of these calculations are presented. The results demonstrate that the minimum reactor coolant system mass inventory condition occurs for the relatively large system pipe breaks. Smaller breaks exhibit a greater margin-to-core uncovery.

15.6.5.4B.3.1 Introduction

The small-break LOCA safety design approach for AP1000 is to provide for a controlled depressurization of the primary system if the break cannot be terminated, or if the nonsafety-related charging system is postulated to be lost or cannot maintain acceptable plant conditions. Nonsafety-related systems are not modeled in this design basis analysis; the testing conducted in the SPES-2 facility has indicated that the mass inventory condition during small LOCAs is significantly improved when these nonsafety-related systems operate. The core makeup tank level activates primary system depressurization. The core makeup tank provides makeup to help compensate for the postulated break in the reactor coolant system. As the core makeup tank level drops, Stages 1 through 4 of the ADS valves are ramped open in sequence. The ADS valve descriptions for the AP1000 plant design are presented in Table 15.6.5-10. The reactor coolant system depressurizes due to the break and the ADS valves, while subcooled water from the core makeup tanks and accumulators enters the reactor vessel downcomer to maintain system inventory and keep the core covered. Design basis maximum values of passive core cooling system resistances are applied to obtain a conservative prediction of system behavior during the small LOCA events.

During controlled depressurization via the ADS, the accumulators and core makeup tanks maintain system inventory for small-break LOCAs. Once the reactor coolant system depressurizes, injection from the IRWST maintains long-term core cooling. For continued injection from the IRWST, the reactor coolant system must remain depressurized. To conservatively model this condition, design maximum resistance values are specified for the IRWST delivery lines.

A series of small-break LOCA calculations are performed to assess the AP1000 passive safety system design performance. In these calculations, the decay heat used is the ANS-1971 (Reference 2) plus 20 percent for uncertainty as specified in 10 CFR 50, Appendix K (Reference 1). This maximizes the core steam generation to be vented. The breaks analyzed in this document include the following:

Inadvertent ADS Actuation

A “no-break” small-break LOCA calculation that uses an inadvertent opening of the 4-inch nominal size ADS Stage 1 valves is a situation that minimizes the venting capability of the reactor coolant system. Only the ADS valve vent area is available; no additional vent area exists due to a break. This case examines whether sufficient vent area is available to completely depressurize the reactor coolant system and achieve injection from the IRWST without core uncovery. The worst single failure for this situation is a failure of one of four ADS Stage 4 valves connected to either of the two hot legs. The ADS Stage 4 valve is the largest ADS valve, and it vents directly to the containment with no additional backpressure from the spargers being submerged in the IRWST.

2-inch Break in a Cold Leg with Core Makeup Tank Balance Line Connections

The small size of the break leads to a long period of recirculatory flow from the cold leg into the core makeup tank. This delays the formation of a vapor space in the core makeup tank and therefore the actuation of the ADS.

Double-Ended Rupture of the Direct Vessel Injection Line

The injection line break evaluates the ability of the plant to recover from a moderately sized break with only half of the total emergency core cooling system capacity available. The vessel side of the break of the DEDVI line break is 4 inches in equivalent diameter. The double-ended nature of this break means that there are effectively two breaks modeled:

- Downcomer to containment. The direct vessel injection nozzle includes a venturi, which limits the available break area.

- Direct vessel injection line into containment from the cold leg balance line and the broken loop core makeup tank.

The containment pressure was conservatively assumed to pressurize to 20 psia. This pressure was selected based on iterative execution of the NOTRUMP and WGOTHIC codes. The NOTRUMP code provides the mass and energy releases from the AP1000 DEDVI break to the AP1000 WGOTHIC containment model while the WGOTHIC code calculates the containment pressure response. The containment pressure assumed in the NOTRUMP simulations was conservatively selected from the generated pressure history curves obtained from the WGOTHIC runs.

The peak core heat flux during the accumulator injection period is assessed relative to the predicted critical heat flux as discussed in Subsection 15.6.5.4B.2.3.

An additional injection line break case is analyzed assuming containment pressure is at 14.7 psia.

Double-Ended Rupture of the Direct Vessel Injection Line Entrainment Sensitivity

The sensitivity case is performed to assess the effect of higher than expected entrainment in the upper plenum and hot legs on the overall system response and core cooling.

10-inch Cold Leg Break

This break models a break size that approaches the upper limit size for small-break LOCAs.

15.6.5.4B.3.2 Transient Results

The transient results are presented in tables and figures for the key AP1000 parameters of interest in the following sections.

15.6.5.4B.3.3 Inadvertent Actuation of Automatic Depressurization System

An inadvertent ADS signal is spuriously generated and the 4-inch ADS valves open. The plant, which is operating at 102-percent power, is depressurized via the ADS alone. The main feedwater flow measurement supports a 1-percent power uncertainty; use of a 2-percent power uncertainty is conservative. Only safety-related systems are assumed to operate in this and other small-break LOCA cases. Additional ADS valves open; after a 70-second delay, the ADS Stage 2 8-inch valves open, and after an additional 120 seconds, the ADS Stage 3 valves open. At the 20-percent core makeup tank level, the ADS Stage 4A valve, which is connected to the hot leg, receives a signal to open. After a 60-second delay, both Stage 4B valves (one connected to the hot leg and the other connected to the PRHR inlet pipe) open. The path that fails to open as the assumed single active failure is the Stage 4A valve off the PRHR inlet pipe. The reactor steady-state initial conditions assumed can be found in Table 15.6.5-9. The sequence of events for the transient is given in Table 15.6.5-11.

The transient is initiated by the opening of the two ADS Stage 1 paths. Reactor trip, reactor coolant pump trip, and safety injection signals are generated via pressurizer low-pressure signals with appropriate delays. After generation of the reactor trip signal, the turbine stop valves begin to close. The main feedwater isolation valves begin to close 2 seconds after the “S” signal pressure setpoint is reached. The opening of the ADS valves and the reduction in core power due to reactor trip causes the primary pressure to fall rapidly (Figure 15.6.5.4B-1). Flow of fluid toward the open ADS paths causes the pressurizer to fill rapidly (Figure 15.6.5.4B-2), and the ADS flow becomes two-phase (Figures 15.6.5.4B-3 and 15.6.5.4B-4). The safety injection signal opens the valves isolating the core makeup tanks and circulation of cold water begins (Figures 15.6.5.4B-5 and 15.6.5.4B-6). The mixture level (Figures 15.6.5.4B-7 and 15.6.5.4B-8) in the core makeup tanks is relatively constant until the accumulators inject (Figures 15.6.5.4B-10 and 15.6.5.4B-11). The reactor coolant pumps begin to coast down due to an automatic trip signal following a 6.0-second delay.

Continued mass flow through the ADS Stage 1, 2, and 3 valves drains the upper parts of the circuit. The steam generator tube cold leg sides start to drain, followed by the drop in mixture levels in the hot leg sides. As the ADS Stage 2 and 3 paths begin to open, increased ADS flow causes the primary pressure to fall rapidly (Figure 15.6.5.4B-1). Following the emptying of the steam generator tube cold leg sides, the cold legs have drained and a mixture level forms in the downcomer (Figure 15.6.5.4B-9).

The primary pressure falls below the pressure in the accumulators thus causing the accumulator check valves to open and accumulator delivery to begin (Figures 15.6.5.4B-10 and Figures 15.6.5.4B-11). The accumulators, and then the core makeup tanks inject until they empty. The ADS flow falls off as the primary pressure decreases. The flow from the accumulators raise the mixture levels in the upper plenum and downcomer (Figures 15.6.5.4B-16 and 15.6.5.4B-9).

As the levels in the core makeup tanks reach the ADS Stage 4 setpoint, one out of two paths is opened from the top of the hot leg (loop 1) and begin discharging fluid. After 30 seconds, the second path in loop one opens, as does a loop 2 Stage 4 path. Activating the Stage 4 paths leads to reduced flow through ADS Stages 1, 2, and 3. The reduced flow allows the pressurizer level to fall, and these stages begin to discharge only steam. Once the core makeup tanks are empty, delivery ceases (Figures 15.6.5.4B-7 and 15.6.5.4B-8). Once the reactor coolant system pressure has fallen sufficiently due to the ADS Stage 4 discharge, (Figure 15.6.5.4B-12) gravity drain from the IRWST begins (Figures 15.6.5.4B-13 and 15.6.5.4B-14). At 5000 seconds, the calculation is considered complete; IRWST delivery exceeds the ADS flows (which are removing the decay heat), and the reactor coolant system inventory is slowly rising (Figure 15.6.5.4B-15). Core uncovery does not occur and the upper plenum mixture level remains well above the core elevation throughout (Figure 15.6.5.4B-16).

The inadvertent opening of the ADS Stage 1 transient confirms the minimum venting area capability to depressurize the reactor coolant system to the IRWST pressure. The analysis indicates that the ADS sizing is sufficient to depressurize the reactor coolant system assuming the worst single failure as the failure of a Stage 4 ADS path to open and decay heat equal to the 10 CFR 50 Appendix K (Reference 1) value of the ANS-1971 Standard (Reference 2) plus 20 percent, which over estimates the core steam generation rate. Even under these limiting conditions, IRWST injection is obtained, and the core remains covered such that no cladding heatup occurs.

15.6.5.4B.3.4 2-inch Cold Leg Break in the Core Makeup Tank Loop

This case models a 2-inch break occurring in the bottom of cold leg connected to the balance line of CMT-1. The reactor steady-state initial conditions assumed for this transient can be found in Table 15.6.5-9. The event times for this transient are given in Table 15.6.5-12.

The break opens at time zero, and the pressurizer pressure begins to fall as shown in Figure 15.6.5.4B-17 as mass is lost out the break. The pressurizer mixture level initially decreases as given in Figure 15.6.5.4B-18. The break fluid flow is shown in Figures 15.6.5.4B-32 and 15.6.5.4B-33. The pressurizer pressure falls below the reactor trip set point, causing the reactor to trip (after the appropriate time delay) and causing isolation of the steam generator steam lines. The core makeup tank isolation valves on both delivery lines and the PRHR delivery line isolation valve open after an “S” signal occurs (with appropriate delays); the reactor coolant pumps trip after an “S” with a 6.0-second delay. The reactor coolant system is cooled by natural circulation with the steam generators removing the energy through their safety valves (as well as by the break) and via the PRHR. Once the core makeup tank isolation valves open, the core makeup tanks begin to inject borated water into the reactor coolant system as shown in Figures 15.6.5.4B-22 and 15.6.5.4B-23.

As time proceeds, the loops drain to the reactor vessel. The mixture level in the downcomer begins to drop as seen in Figure 15.6.5.4B-30, and the core remains completely covered. The core makeup tank reaches the 67.5-percent level, and after an appropriate delay, the ADS Stage 1 valves open. When the ADS is actuated, the mixture level increases in the pressurizer (Figure 15.6.5.4B-18) because an opening has been created at the top of the pressurizer. After these valves open, a more rapid depressurization occurs as seen in Figure 15.6.5.4B-17; the accumulator setpoint is reached and the accumulators begin to inject. The injection flow from the core makeup tanks are shown in Figures 15.6.5.4B-22 and 15.6.5.4B-23, and from the accumulators, in Figures 15.6.5.4B-24 and 15.6.5.4B-25.

As Figures 15.6.5.4B-22 and 15.6.5.4B-23 indicate, when the accumulators begin to inject, the flow from both core makeup tanks is reduced, and the flow is temporarily stopped due to the pressurization of the core makeup tanks injection lines by the accumulators.

The ADS Stage 2 valves, maintaining the depressurization rate as shown in Figure 15.6.5.4B-17. ADS Stage 3 valves open, thereby increasing the system venting capability. The ADS Stage 4 valves open when the core makeup tank water level is reduced to 20 percent. Figures 15.6.5.4B-28 and 15.6.5.4B-31 indicate the instantaneous liquid and integrated total mass discharged from the ADS Stage 4 valves. After the ADS Stage 4 path opens, the pressurizer begins to drain mixture into the hot legs as seen in Figure 15.6.5.4B-18. The Figure 15.6.5.4B-29 mass inventory plot considers the primary inventory to be the reactor coolant system proper, including the pressurizer; the mass present in the passive safety system components is not included at time zero. Once the downcomer pressure drops below the IRWST injection pressure, flow enters the reactor vessel from the IRWST. The mixture level in the reactor vessel is approximately at the hot leg elevation as shown in Figure 15.6.5.4B-30 throughout this transient; the core never uncovers, and the peak cladding temperature occurs for this transient at the inception of the event. The 2-inch break cases exhibit large margin-to-core uncovery.

15.6.5.4B.3.5 Direct Vessel Injection Line Break

This case models the double-ended rupture of the DVI line at the nozzle into the downcomer. The broken loop injection system (consisting of an accumulator, a core makeup tank, and an IRWST delivery line) is modeled to spill completely out the DVI side of the break. The steady-state reactor coolant system conditions for this transient are shown in Table 15.6.5-9. Design maximum resistances are applied to the inlet and outlet lines of that core makeup tank to conservatively minimize intact loop core makeup tank delivery through the time of minimum reactor coolant system mass inventory. Minimum resistances are applied to the broken loop IRWST injection line to maximize the spill to containment, thus minimizing the reactor coolant system mass inventory. This case uses a containment backpressure defined to be a constant 20 psia. While not exactly reflecting the containment pressure history that occurs as a result of the DVI line break, it represents a conservatively low estimate of the expected containment pressure response during a DEDVI transient. The containment pressurizes for a DEDVI break as a result of the break mass and energy releases in addition to the ADS-4 discharge paths that vent directly to the containment atmosphere.

The containment pressurization was calculated using the mass and energy releases from the NOTRUMP small-break LOCA code in the WGOTHIC containment model. Mass and energy releases from both sides of the DVI break (both vessel side and DVI side) and ADS-4 valve discharges were provided in a tabular form to the WGOTHIC AP1000 model used to compute containment pressurization for the long-term cooling analysis.

The event times for this transient are shown in Table 15.6.5-13. The break is assumed to open instantaneously at 0 seconds. The accumulator on the broken loop starts to discharge via the DVI line to the containment. Figure 15.6.5.4B-36 shows the subcooled discharge from the downcomer nozzle, which causes a rapid reactor coolant system (RCS) depressurization (Figure 15.6.5.4B-38). A reactor trip signal is generated, followed by generation of the “S” signal. Following a delay, the isolation valves on the core makeup tank and PRHR delivery lines begin to open. The “S” signal also causes closure of the main feedwater isolation valves after a 2-second delay and trips the reactor coolant pumps after a 6-second delay. The opening of the core makeup tank isolation valves allows the broken loop core makeup tank to discharge directly to the containment (Figure 15.6.5.4B-39), and a small circulatory flow develops through the intact loop core makeup tank (Figure 15.6.5.4B-40).

As the pressure falls, the reactor coolant system fluid saturates, and a mixture level forms in the upper plenum and then falls to the hot leg elevation (Figure 15.6.5.4B-41). The upper parts of the reactor coolant system start to drain, and a mixture level forms in the downcomer (Figure 15.6.5.4B-42) and falls below the elevation of the break. Two-phase discharge, then vapor flow occurs from the downcomer side of the break (Figure 15.6.5.4B-37).

In the core makeup tank connected to the broken loop, a level forms and starts to fall. The ADS Stage 1 setpoint is reached, and the ADS Stage 1 valves open after the signal delay time elapses. The ensuing steam discharge from the top of the pressurizer (Figure 15.6.5.4B-43) increases the reactor coolant system depressurization rate. The depressurization rate is also increased due to the steam discharge from the downcomer to the containment (Figure 15.6.5.4B-37) as the downcomer mixture level falls below the DVI nozzle (Figure 15.6.5.4B-42).

During the initial portion of the DEDVI break, only liquid flows out the top of the core (Figure 15.6.5.4B-45). Soon, steam flows out also (Figure 15.6.5.4B-46) because the void fraction in the core increases (Figure 15.6.5.4B-44). The break in the downcomer draws fluid from the bottom of the core (Figure 15.6.5.4B-47) and insufficient liquid remains in the core and upper plenum to sustain the mixture level. The mixture level therefore starts to decrease (Figure 15.6.5.4B-41). The mixture level falls to a minimum and then starts to recover, as flow re-enters the core from the downcomer (Figure 15.6.5.4B-41 compared to 15.6.5.4B-47).

The ADS Stage 2 valves open after the appropriate time delay between the actuation of the first two stages of the ADS. The intact loop accumulator starts to inject into the downcomer (Figure 15.6.5.4B-50) causing the mixture level in the downcomer to slowly rise (Figure 15.6.5.4B-42). The mixture level also increases within the upper plenum.

The ADS Stage 3 valves open upon completion of the time delay of 120 seconds between the actuation of Stages 2 and 3 of the ADS. The broken loop core makeup tank level reaches the ADS Stage 4 setpoint, but the ADS Stage 4 valves do not open until the minimum time delay between the actuation of ADS Stages 3 and 4 occurs. Two-phase discharge ensues through three of the four Stage 4 paths (Figures 15.6.5.4B-48 and 15.6.5.4B-49). The broken loop core makeup tank and accumulator empty rapidly.

The fluid level at the top of the intact loop core makeup tank starts to decrease slowly (Figure 15.6.5.4B-52) because injection from the tank has begun (Figure 15.6.5.4B-40). The intact loop accumulator has emptied (Figure 15.6.5.4B-50) and the reduced pressure in the injection line allows the core makeup tank to inject continuously.

During the period of accumulator injection, the downcomer mixture level rises slowly (Figure 15.6.5.4B-42). Figure 15.6.5.4B-53 presents the RCS mass inventory. With only intact loop core makeup tank injection available for a period of time, the downcomer level once again falls and core boil-off increases the rate of reactor coolant system inventory depletion until sufficient CMT/IRWST injection flow can be introduced. However, the level in the upper plenum is maintained near the hot leg elevation (Figure 15.6.5.4B-41) throughout the remainder of the transient.

Once the pressure in the broken DVI line falls below that in the IRWST, the water from the tank is spilled to the containment.

Stable, but decreasing, injection continues from the intact loop core makeup tank as the reactor coolant system pressure declines slowly. The reactor coolant system pressure continues to fall until it drops below that of the IRWST and injection begins (Figure 15.6.5.4B-51). With the reduced initial RCS inventory recovery from the accumulators and only a single intact injection path available for the DEDVI line break, the minimum inventory occurs near the initiation of IRWST injection flow. After injection flow greater than the sum of the break and ADS flows exists, a slow rise in the reactor coolant system inventory (Figure 15.6.5.4B-53) occurs. Since no core uncovery is predicted for this scenario, no cladding heatup occurs.

The critical heat flux assessment described in Subsection 15.6.5.4B.2.3 addresses core cooling during a time period where the NOTRUMP computer code may not conservatively predict the core average void fraction. The requirements of 10 CFR 50.46 are met during this period since this CHF assessment indicates peak core heat flux is less than critical heat flux. Cladding temperatures will remain near the coolant saturation temperature, well below the 10 CFR 50.46 peak cladding temperature limit.

Another DEDVI line break analysis is performed that is the same as the case discussed above except that containment pressure is assumed to be at 14.7 psia. Table 15.6.5-13A provides the time sequence of events for this analysis. Figures 15.6.5.4B-36A through 15.6.5.4B-55A provide the transient results for this analysis. The transient is like the case at 20 psia except that IRWST injection occurs somewhat later due to the lower containment pressure.

15.6.5.4B.3.6 10-inch Cold Leg Break

This case models a 10-inch break occurring in the bottom of a cold leg connected to the balance line of CMT-1. The reactor steady-state initial conditions assumed for this transient are found in Table 15.6.5-9. The event times for this transient are given in Table 15.6.5-14.

The break opens at time zero, and the pressurizer pressure begins to fall, as shown in Figure 15.6.5.4B-56, as mass is lost out the break. The pressurizer mixture level initially decreases as given in Figure 15.6.5.4B-57. The break fluid flow is shown in Figures 15.6.5.4B-75 and 15.6.5.4B-76 for the liquid and vapor components respectively. The pressurizer pressure falls below the reactor trip set point. This causes the reactor to trip (after the appropriate time delay) and isolation of the steam generator steam lines. The core makeup tank isolation valves on both delivery lines and the PRHR delivery line isolation valve open after an “S” signal occurs (with appropriate delays); the reactor coolant pumps trip after an “S” with a 6.0-second delay. The reactor coolant system is cooled by natural circulation with energy being removed by the steam generator safety valves, the core makeup tanks, and the PRHR heat exchanger. Once the core makeup tank isolation valves open, the core makeup tanks begin to inject borated water into the reactor coolant system as shown in Figures 15.6.5.4B-61 and 15.6.5.4B-62.

As time proceeds, the loops drain to the reactor vessel. The mixture level in the downcomer begins to drop as seen in Figure 15.6.5.4B-60, and the core remains completely covered. Due to the size and location of the break involved, the accumulator setpoint is reached prior to the core makeup tanks transitioning from recirculation to injection mode. The flows from the core makeup tanks are shown in Figures 15.6.5.4B-61 and 15.6.5.4B-62, and from the accumulators, in Figures 15.6.5.4B-63 and 15.6.5.4B-64. The response of core makeup tank 1 is offset compared to that of core makeup tank 2 as a result of the break size/location being modeled. Core makeup tank 2 reaches the 67.5-percent level first, and after an appropriate delay, the ADS Stage 1 valves open. When the ADS is actuated, the mixture level increases in the pressurizer (Figure 15.6.5.4B-57) because an opening has been created at the top of the pressurizer. After these valves open, a more rapid depressurization occurs as seen in Figure 15.6.5.4B-56.

During the initial portion of the 10-inch break, both liquid and steam flow out the top of the core (Figures 15.6.5.4B-71 and 15.6.5.4B-72) as the void fraction in the core increases (Figure 15.6.5.4B-73). The break in the cold leg draws fluid from the bottom of the core, and insufficient liquid remains in the core and upper plenum to sustain the mixture level. The mixture level, therefore, starts to decrease (Figure 15.6.5.4B-69). The mixture level falls to a minimum and then starts to recover as accumulator flows enter the downcomer (Figures 15.6.5.4B-63 and 15.6.5.4B-64). During this time period (~75-125 seconds), a portion of the core exhibits the potential for core dryout to occur without the prediction of a traditional core uncovery period (for example, core two-phase mixture level dropping into the active fuel region). To conservatively account for this potential core dryout period, a composite core mixture level was created which collapses to the minimum of the actual core/upper plenum two-phase mixture level and the bottom of the lowest core node that exceeds the core dryout onset conditions. A 90-percent quality limit was chosen as the indicator of the onset of core dryout indicative of the critical heat flux (as predicted by Griffith's modification of the Zuber equation, in References 28 and 29); dryout is assumed at core qualities above this value. The resulting composite core mixture level resulting from this approach can be seen in Figure 15.6.5.4B-70. To conservatively estimate the effects of this dryout period, an adiabatic heat-up calculation was performed, and the resulting peak cladding temperature is determined to be approximately 1370°F. Even under these conservative adiabatic heat-up assumptions, the AP1000 plant design exhibits large margins to the 10 CFR 50.46 Appendix-K limits for the 10-inch break.

As Figures 15.6.5.4B-61 and 15.6.5.4B-62 indicate, when the accumulators begin to inject, the flow from both core makeup tanks is reduced and the flow is temporarily stopped due to the pressurization of the injection lines of the core makeup tanks by the accumulators. The opening of ADS Stage 2 valves maintains the depressurization rate as shown in Figure 15.6.5.4B-56. ADS Stage 3 valves subsequently open. This increases the system venting capability. The ADS Stage 4 valves open when the core makeup tank water level is reduced to 20 percent. Figures 15.6.5.4B-67 and 15.6.5.4B-74 indicate the instantaneous liquid and integrated total mass discharged from the ADS Stage 4 valves. After the ADS Stage 4 path opens, the pressurizer begins to drain mixture into the hot legs as seen in Figure 15.6.5.4B-57. The Figure 15.6.5.4B-68 mass inventory plot considers the primary inventory to be the reactor coolant system proper, including the pressurizer; the mass present in the passive safety system components is not included. Once the downcomer pressure drops below the IRWST injection pressure, flow enters the reactor vessel from the IRWST. The mixture level in the reactor vessel is approximately at the hot leg elevation as shown in Figure 15.6.5.4B-69 throughout this transient; the core never uncovers, even though the period of potential core dryout was predicted to occur during the initial blowdown period. Even when the core dryout is conservatively accounted for, large margins to the 10 CFR 50.46 Appendix-K limits of 2200°F exist.

15.6.5.4B.3.7 Direct Vessel Injection Line Break (Entrainment Sensitivity)

In order to assess the potential impact of higher than expected entrainment in the upper plenum and hot legs on the overall system response and core cooling, an AP1000 plant sensitivity run was performed. The sensitivity case was performed with the DEDVI line break simulation as described in the following. The simulation utilizes the same initial conditions as the base DEDVI line simulation presented in Subsection 15.6.5.4B.3.5. The transient response is essentially identical until ADS-4 actuation, at which time the higher than expected entrainment is included in the analysis by assuming homogenous conditions in the regions downstream of the core. In addition, since homogenous treatment of these regions will eliminate the pressure drop effect of the accumulated mass stored in the upper plenum, the NOTRUMP model was conservatively adjusted to account for this effect following the transition of the ADS-4 flow paths to noncritical conditions.

Figure 15.6.5.4B-79 presents a comparison of the upper downcomer pressure between the base and sensitivity cases. The sensitivity case results in higher upper downcomer pressure and subsequently results in delayed IRWST injection (Figure 15.6.5.4B-80). This can also be observed in the intact DVI line flow, which comprises all intact injection flow components (that is, accumulator, CMT, and IRWST) per Figure 15.6.5.4B-81, and the pressurizer mixture level response (Figure 15.6.5.4B-90), which follows the change in pressure response. As expected, the initial ADS-4 liquid discharge is much higher (Figure 15.6.5.4B-82) until the inventory, which resided in the upper plenum and hot leg regions, depletes (Figure 15.6.5.4B-83). The net effect is a decrease in the ADS-4 vapor discharge rate (Figure 15.6.5.4B-84) and subsequently higher RCS pressures.

Due to the elimination of the inventory stored in the upper plenum, the downcomer mass is also reduced (Figure 15.6.5.4B-85). Since the static head that existed in the upper plenum is eliminated when the model is made homogenous, the downcomer mixture is subsequently driven into the core as the static heads equilibrate. This results in the core region mass increasing initially due to the introduction of cold downcomer fluid to the core region (Figure 15.6.5.4B-86). The net effect of the sensitivity case is that the vessel inventory is substantially decreased over the base model simulation (Figure 15.6.5.4B-87); however, this inventory is sufficient to provide adequate core cooling because the ADS-4 continually draws liquid flow through the core (Figure 15.6.5.4B-82). Even though there is no liquid storage in the upper plenum for the homogenous case (Figure 15.6.5.4B-88), the core collapsed liquid level (Figure 15.6.5.4B-89) is not impacted significantly.

This sensitivity demonstrates that the AP1000 plant response is relatively insensitive to upper plenum and hot leg entrainment. Even with the assumption of homogenous fluid nodes above the core, adequate core cooling is demonstrated. No significant core uncovery/heatup is predicted for this scenario.

15.6.5.4B.4 Conclusions

The small-break LOCA analyses performed show that the performance of the AP1000 plant design to small-break LOCA scenarios is excellent and that the passive safeguards systems in the AP1000 are sufficient to mitigate LOCAs. Specifically, it is concluded that:

- The primary side can be depressurized by the ADS to allow stable injection into the core.
- Injection from the core makeup tanks, accumulators, and IRWST prevents excessive cladding heatup for small-break LOCAs analyzed, including double-ended ruptures in the passive safeguards system lines. The peak AP1000 heat flux during the accumulator injection period is below the predicted critical heat flux.
- The effect of increasing upper plenum/hot leg entrainment does not significantly affect plant safety margins.

The analyses performed demonstrate that the 10 CFR 50.46 Acceptance Criteria are met by the AP1000. Summarizing the small-break LOCA spectrum:

Break Location/Diameter	AP1000 Minimum RCS Inventory (lbm)	Peak Cladding Temperature
Inadvertent ADS	105,800	(1)
2-inch cold leg break	106,620	(1)
10-inch cold leg break	78,160	<1370°F
DEDVI	113,710	(1)
DEDVI (Entrainment Study)	~82,000	(1)

(1) There is no core heatup as a result of this transient. PCT occurs at transient initiation

The 10-inch cold leg break exhibits the limiting minimum inventory condition that occurs during the initial blowdown period and is terminated by accumulator injection. The AP1000 design is such that the minimum inventory occurs just prior to IRWST injection for all breaks except the 10-inch cold leg break. All breaks simulated in the break spectrum produce results that demonstrate significant margin to peak cladding temperature regulatory limits.

15.6.5.4C Post-LOCA Long-Term Cooling

15.6.5.4C.1 Long-Term Cooling Analysis Methodology

The AP1000 safety-related systems are designed to provide adequate cooling of the reactor indefinitely. Initially, this is achieved by discharging water from the IRWST into the vessel. When the low-3 level setpoint is reached in the IRWST, the containment recirculation subsystem isolation valves open and water from the containment reactor coolant system (RCS) compartment can flow into the vessel through the PXS piping. The water in containment rises in temperature toward the saturation temperature. Long-term heat removal from the reactor and containment is by heat transfer through the containment shell to atmosphere.

The purpose of the long-term cooling analysis is to demonstrate that the passive systems provide adequate emergency core cooling system performance during the IRWST injection/containment recirculation time scale. The long-term cooling analysis is performed using the WCOBRA/TRAC computer code to verify that the passive injection system is providing sufficient flow to the reactor vessel to cool the core and to preclude boron precipitation.

The AP1000 long-term cooling analysis is supported by the series of tests at the Oregon State University AP600 APEX Test Facility. This test facility is designed to represent the AP600 reactor safety-related systems and nonsafety-related systems at quarter-scale during long-term cooling. The data obtained during testing at this facility has been shown to apply to the AP1000 (Reference 25). These tests were modeled using WCOBRA/TRAC with an equivalent noding scheme to that used for AP600 (Reference 17) in order to validate the code for long-term cooling analysis.

Reference 24 provides details of the AP1000 WCOBRA/TRAC modeling. The coarse reactor vessel modeling used for AP600 has been replaced with a detailed noding like that applied in the large-break LOCA analyses described in Subsection 15.6.5.4A. The reactor vessel noding used in the AP1000 long-term cooling analyses in core and upper plenum regions is equivalent to that used in full-scale test simulations (see Reference 24).

A DEDVI line break is analyzed because it is the most limiting long-term cooling case in the relationship between decay power and available liquid driving head. Because the IRWST spills directly onto the containment floor in a DEDVI break, this event has the highest core decay power when the transfer to sump injection is initiated. In postulated DEDVI break cases, the compartment water level exceeds the elevation at which the DVI line enters the reactor vessel, so water can flow from the containment into the reactor vessel through the broken DVI line; this in-flow of water through the broken DVI line assists in the heat removal from the core. The steam produced by boiling in the core vents to the containment through the ADS valves and condenses on the inner surface of the steel containment vessel. The condensate is collected and drains to the IRWST to become available for injection into the reactor coolant system. The WCOBRA/TRAC analysis presented analyzes the DEDVI small-break LOCA event from a time (3000 seconds) at which IRWST injection is fully established to beyond the time of containment recirculation. During this time, the head of water to drive the flow into the vessel for IRWST injection decreases from the initial level to its lowest value at the containment recirculation switchover time. PXS Room B is the location of the break in the DVI line. At this break location, liquid level in containment at the time of recirculation is a minimum.

A continuous analysis of the post-LOCA long term cooling is provided from the time of stable IRWST injection through the time of sump recirculation for the DEDVI break. Maximum design resistances are applied in WCOBRA/TRAC for both the ADS Stage 4 flow paths and the IRWST injection and containment recirculation flow paths.

The break modeled is a double-ended guillotine rupture of one of the direct vessel injection lines. The long-term cooling phase begins after the simultaneous opening of the isolation valves in the IRWST

DVI lines and the opening of ADS Stage 4 squib valves, when flow injection from the IRWST has been fully established. Initial conditions are taken from the NOTRUMP DEDVI case at 20 psia containment pressure reported in Subsection 15.6.5.4B.

15.6.5.4C.2 DEDVI Line Break with ADS Stage 4 Single Failure, Passive Core Cooling System Only Case; Continuous Case

This subsection presents the results of a DEDVI line break analysis during IRWST injection phase continuing into sump recirculation. Initial conditions at the start of the case are prescribed based on the NOTRUMP DEDVI break results to allow a calculation to begin shortly after IRWST injection begins in the small break long-term cooling transient. The WCOBRA/TRAC calculation is then allowed to proceed until a quasi-steady-state is achieved. At this time, the predicted results are independent of the assumed initial conditions. This calculation uses boundary conditions taken from a WGOTHIC analysis of this event. During the calculation, which is carried out for 10,000 seconds until a quasi-steady-state sump recirculation condition has been established, the IRWST water level is decreased continuously until the sump recirculation setpoint is reached.

In the analysis, one of the two ADS Stage 4 valves in the PRHR loop is assumed to have failed. The initial reactor coolant system liquid inventory and temperatures are determined from the NOTRUMP calculation. The core makeup tanks do not contribute to the DVI injection during this phase of the transient. Steam generator secondary side conditions are taken from the NOTRUMP calculation (at the beginning of long-term cooling). The reactor coolant pumps are tripped and not rotating.

The levels and temperatures of the liquid in the containment sump and the containment pressure are based on WGOTHIC calculations of the conservative minimum pressure during this long-term cooling transient, including operation of the containment fan coolers. Small changes in the RCS compartment level do not have a major effect on the predicted core collapsed liquid level or on the predicted flow rate through the core. The minimum compartment floodup level for this break scenario is 107.8 feet or greater.

In this transient, the IRWST provides a hydraulic head sufficient to drive water into the downcomer through the intact DVI nozzle. Also, water flows into the downcomer from the broken DVI line once the liquid level in the compartment with the broken line is adequate to support flow. The water flows down the downcomer and up through the core, into the upper plenum. Steam produced in the core and liquid flow out of the reactor coolant system via the ADS Stage 4 valves. There is little flow out of ADS Stages 1, 2, and 3 even when the IRWST liquid level falls below the sparger elevation, so they are not modeled in this calculation. The venting provided by the ADS-4 paths enables the liquid flow through the core to maintain core cooling.

Approximately 500 seconds of WCOBRA/TRAC calculation are required to establish the quasi-steady-state condition associated with IRWST injection at the start of long-term cooling and so are ignored in the following discussion. The hot leg levels are such that during the IRWST injection phase the quality of the ADS Stage 4 mass flows varies as water is carried out of the hot legs. This periodically increases the pressure drop across the ADS Stage 4 valves and the upper plenum pressure. The higher pressure in the upper plenum reduces the injection flow. This cycle of pressure variations due to changing void fractions in the flow through ADS Stage 4 is consistent with test observations and is expected to recur often during long-term cooling.

The head of water in the IRWST causes a flow of subcooled water into the downcomer at an approximate rate of 170 lbm/s through the intact DVI nozzle at the start of long-term cooling. The downcomer level at the end of the code initiation (the start of long-term cooling) is about 18.0 feet (Figure 15.6.5.4C-1). Note that the time scale of this and other figures in Subsection 15.6.5.4C.2 is offset by 2500 seconds; that is, a time of 500 seconds on the Figure 15.6.5.4C-1 axis equals 3000 seconds transient time for the DEDVI break. All of the injection water flows down the downcomer and up through the core. The accumulators have been fully discharged before the start of the time window and do not contribute to the DVI flow.

Boiling in the core produces steam and a two-phase mixture, which flows into the upper plenum. The core is 14 feet high, and the core average collapsed liquid level (Figure 15.6.5.4C-2) is shown from the start of long-term cooling. The boiling process causes a variable rate of steam production and resulting pressure changes, which in turn causes oscillations in the liquid flow rate at the bottom of the core and also variations in the core collapsed level and the flow rates of liquid and vapor out of the top of the core. In the WCOBRA/TRAC noding, the core is divided both axially and radially as described in Reference 24. The void fractions in the top two cells of the hot assembly are shown as Figures 15.6.5.4C-3 and 15.6.5.4C-4. The average void fraction of these upper core cells is about 0.8 during long-term cooling, during IRWST injection, and into the containment recirculation period. There is a continuous flow of two-phase fluid into the hot legs, and mainly vapor flow toward the ADS Stage 4 valve occurs at the top of the pipe. The collapsed liquid level in the hot leg varies between 0.8 feet to 1.6 feet (Figure 15.6.5.4C-5). The hot legs on average are more than 50-percent full. Vapor and liquid flows at the top of the core are shown in Figures 15.6.5.4C-6 and 15.6.5.4C-7, the upper plenum collapsed liquid level in Figure 15.6.5.4C-8. Figures 15.6.5.4C-9 and 15.6.5.4C-10 are ADS stage 4 mass flowrates.

The pressure in the upper plenum is shown in Figure 15.6.5.4C-11. The upper plenum pressure fluctuation that occurs is due to the ADS Stage 4 water discharge. The PCT of the hot rod follows saturation temperature (Figure 15.6.5.4C-12), which demonstrates that no uncovery and no cladding temperature excursion occurs. A small pressure drop is calculated across the reactor vessel, and injection rates through the DVI lines into the vessel are presented in Figures 15.6.5.4C-13 and 15.6.5.4C-14. Figure 15.6.5.4C-14 shows the flow is outward through the broken DVI line at the start of the long-term cooling period, and it increases to a maximum average value of about 52 lbm/s after the compartment water level has increased above the nozzle elevation to permit liquid injection into the reactor vessel. In contrast, the intact DVI line flow falls from 170 lbm/s with a full IRWST to about 65 lbm/s flow from the containment at the end of the calculation. The recirculation core liquid throughput is more than adequate to preclude any boron buildup on the fuel.

Figures 15.6.5.4C-1A through 15.6.5.4C-14A present the sensitivity of long-term cooling performance to a bounding containment pressure of 14.7 psia. The DEDVI break in the PXS “B” Room case is restarted at 6500 seconds to assess in a window mode calculation the effect of this reduced containment pressure at the most limiting time in the transient, the switchover to containment recirculation. The initial 700 seconds of the window establish the reactor vessel pressure condition that is consistent with the 14.7 psia containment pressure. After 7200 seconds, the WCOBRA/TRAC calculation provides the transient behavior of the AP1000 at the reduced containment pressure.

15.6.5.4C.3 DEDVI Break and Wall-to-Wall Floodup; Containment Recirculation

This subsection presents a DEDVI line break analysis with wall-to-wall flooding due to leakage between compartments, using the window mode methodology. All containment free volume beneath the level of the liquid is assumed filled in this calculation to generate the minimum water level condition during containment recirculation. The time identified for this calculation is 14 days into the event, and the core power is calculated accordingly. The initial conditions at the start of the window are consistent with the analysis described in Subsection 15.6.5.4C.2. Containment recirculation is simulated during the time window. The calculation is carried out over a time period long enough to establish a quasi-steady-state solution; after 400 seconds of problem time, the flow dynamics are quasi-steady-state and the predicted results are independent of the assumed initial conditions. The liquid level is simulated constant at 28.2 feet above the bottom inside surface of the reactor vessel (refer to Figure 15.0.3-2 for AP1000 reference plant elevations) during the time window, and the liquid temperatures in the containment sump and the PXS “B” room are 196°F and 182°F, respectively. The containment pressure is conservatively assumed to be 14.7 psia. The single failure of an ADS Stage 4 flow path is assumed as in the Subsection 15.6.5.4C.2 case.

Focusing on the post 400-second time interval of this case, the containment liquid provides a hydraulic head sufficient to drive water into the downcomer through the DVI nozzles. The water introduced into the downcomer flows down the downcomer and up through the core, into the upper plenum. Steam produced in the core entrains liquid and flows out of the reactor coolant system via the ADS Stage 4 valves. The DVI flow and the venting provided by the ADS paths provide a liquid flow through the core that enables the core to remain cool.

The downcomer collapsed liquid level (Figure 15.6.5.4C-15) varies between 23 and 25 feet during the analysis. Pressure spikes produced by boiling in the core can cause the mass flow of the DVI flow rates shown in Figures 15.6.5.4C-27 and 15.6.5.4C-28 into the vessel to fluctuate upward and downward.

Boiling in the core produces steam and a two-phase mixture, which flows out of the core into the upper plenum. The core is 14 feet high, and the core collapsed liquid level (Figure 15.6.5.4C-16) maintains a mean level close to the top of the core. The boiling process causes pressure variations, which in turn, cause variations in the core collapsed level and the flow rates of liquid and vapor out of the top of the core. In the WCOBRA/TRAC analysis, the core is nodalized as described in Reference 24. The void fraction in the top cell is shown in Figure 15.6.5.4C-17 for the core hot assembly, and Figure 15.6.5.4C-18 shows the void fraction that exists one cell further down in the hot assembly. The PCT does not rise appreciably above the saturation temperature (Figure 15.6.5.4C.3-26). The flow through the core and out of the reactor coolant system is more than sufficient to provide adequate flushing to preclude concentration of the boric acid solution. Liquid collects above the upper core plate in the upper plenum, where the average collapsed liquid level is about 3.6 feet (Figure 15.6.5.4C-22). There is no significant flow through the cold legs into either the broken or the intact loops, and there is no significant quantity of liquid residing in any of the cold legs.

The pressure in the upper plenum is shown in Figure 15.6.5.4C-25. The upper plenum pressurization, which occurs periodically, is due to the ADS Stage 4 water discharge. The collapsed liquid level in the hot leg of the pressurizer loop varies between 1.0 feet and 2.1 feet, as shown in Figure 15.6.5.4C-19. Injection rates through the DVI lines into the vessel are presented in Figures 15.6.5.4C-27 and 15.6.5.4C-28.

15.6.5.4C.4 Post Accident Core Boron Concentration

An evaluation has been performed of the potential for the boron concentration to build up in the core following a cold leg LOCA. The evaluation methodology, simplified calculations, and their results are discussed in Reference 24. This evaluation considers both short-term operations, before ADS is actuated, and long-term operations, after ADS is actuated. These evaluations and their results are discussed in the follow paragraphs.

Short-term – Prior to ADS actuation, it is not likely for boron to build up significantly in the core. Normally, water circulation mixes boron in the RCS and prevents buildup in the core. In order for boron to start to build up in the core region, water circulation through the steam generators and PRHR HX has to stop. In addition, significant injection of borated water is needed from the CMTs and the CVS. For this situation to happen, the hot legs need to void sufficiently to allow the steam generator tubes to drain. Once the steam generator tubes void, the cold legs will also void since they are located higher than the hot legs. When the top of the cold legs void, the CMTs will begin to drain. When the CMTs drain to the ADS stage 1 setpoint, ADS is actuated.

Short-term Results – As shown in Subsection 15.6.5.4B.3.4, a 2-inch LOCA requires less than 16 minutes from the time that the hot legs void significantly until ADS is actuated. For larger LOCAs, this time difference is shorter, as seen for the 10-inch cold leg LOCA (Subsection 15.6.5.4B.3.6). The core boron concentration will not build up significantly in this short time. If the break is smaller than 2 inches, voiding of the hot legs will occur at a later time. With maximum operation of CVS makeup, it takes more than 3 hours for the core boron concentration to build up significantly. In addition, the volume of the boric acid tank limits the maximum buildup of boron in the core.

Following a small LOCA where ADS is not actuated, the operators are guided to sample the RCS boron concentration and to initiate a post-LOCA cooldown and depressurization. The cooldown and depressurization of the RCS reduces the leak rate and facilitates recovery of the pressurizer level. Recovery of the pressurizer level allows for re-establishment of water flow through the RCS loops, which mixes the boron. The operators are guided to take an RCS boron sample within 3 hours of the accident and several more during the plant cooldown. The purpose of the boron samples is to assess that there is adequate shutdown margin and that the RCS boron concentration has not built up to excessive levels. The maximum calculated core boron concentration 3 hours after a LOCA without ADS actuation is less than 16,000 ppm. Operator action within 3 hours maintains the maximum core boron concentration well below the boron solubility limit for the core inlet temperatures during the cooldown.

Long-term – Once ADS is actuated, water carryover out the ADS Stage 4 lines limits the potential core boron concentration buildup following a cold leg LOCA. The design of the AP1000 facilitates water discharge from the hot legs as follows:

- PXS recirculation flow capability tends to fill the hot legs and bring the water level up to the ADS Stage 4 inlet.
- ADS Stage 4 lines discharge at an elevation 3 to 4 feet above the containment water level.

With water carried out ADS Stage 4, the core boron concentration increases until the boron added to the core in the safety injection flow equals the boron removed in the water leaving the RCS through the ADS Stage 4 flow. The lower the ADS Stage 4 vent quality, the lower the core boron concentration buildup.

Long-term Results – Analyses have been performed (Reference 24) to bound the maximum core boron concentration buildup. These analyses demonstrate that highest ADS Stage 4 vent qualities result from the following:

- Highest decay heat levels
- Lowest PXS injection/ADS 4 vent flows, including high line resistances and low containment water levels

The long-term cooling analysis discussed in Subsection 15.6.5.4C.2 is consistent with these assumptions. The ADS Stage 4 vent quality resulting from this analysis is less than 40 percent at the beginning of IRWST injection and reaches a maximum of less than 50 percent around the initiation of recirculation. It decreases after this peak, dropping to a value less than 8 percent at 14 days.

With the maximum ADS Stage 4 vent qualities, the maximum core boron concentration peaks at a value of about 7400 ppm at the time of recirculation initiation. After this time, the core boron concentration decreases as the ADS Stage 4 vent quality decreases, reaching 5000 ppm about 9 hours after the accident. The core boron solubility temperature reaches a maximum of 58°F (at 7400 ppm) and quickly drops to 40°F (at 5000 ppm). With these low core boron solubility temperatures, there is no concern with cold PXS injection water causing boron precipitation in the core. With the IRWST located inside containment, its water temperature is normally expected to be above these solubility temperatures. The minimum core inlet temperature is greater than the solubility temperature considering heatup of the injection by steam condensation in the downcomer and pickup of sensible heat from the reactor vessel, core barrel, and lower support plate.

The boron concentration water in the containment is initially about 2980 ppm. As the core boron concentration increases, the containment concentration decreases slightly. The minimum boron concentration in containment is greater than 2950 ppm. The solubility temperature of the containment water at its maximum boron concentration is 32°F.

With high decay heat values, the ADS Stage 4 vent flows and velocities are high. These high vent velocities result in flow regimes that are annular for more than 30 days. The annular flow regime moves water up and out the ADS Stage 4 lines. This flow regime is based on the Taitel-Dukler vertical flow regime map. Lower decay heat levels can be postulated later in time or just after a refueling outage. Significantly lower decay heat levels result in lower ADS Stage 4 vent qualities. They also result in ADS Stage 4 vent flows/velocities that are lower. Even with low ADS Stage 4 vent flow velocities, the AP1000 plant will move water out the ADS Stage 4 operating as a manometer. Small amounts of steam generated in the core reduce the density of the steam/water mixture in the core, upper plenum, and ADS Stage 4 line as it bubbles up through the water. As a result, the injection head is sufficient to push the less dense, bubbly steam/water mix out the ADS Stage 4 line.

At the time recirculation begins, the containment level will be about 109.3 feet (for a non-DVI LOCA) and will be about 108.0 feet (for a DVI LOCA). Over a period of weeks after a LOCA, water may slowly leak from the flooded areas in containment to other areas inside containment that did not initially flood. As a result, the minimum containment water could decrease to 103.5 feet. During recirculation operation following a LOCA and ADS actuation, the operators are guided to maintain the containment water level above the 107-foot elevation by adding borated water to the containment. In addition, if the plant continues to operate in the recirculation mode, the operators are guided to increase the level to 109 feet within 30 days of the accident. These actions provide additional margin in water flow through the ADS Stage 4 line. The operators are also guided to sample the hot leg boron concentration prior to initiating recovery actions that might introduce low temperature water to the reactor.

15.6.5.4C.5 Conclusions

Calculations of AP1000 long-term cooling performance have been performed using the WCOBRA/TRAC model developed for AP1000 and described in Reference 24. The DEDVI case was chosen because it reaches sump recirculation at the earliest time (and highest decay heat). A window mode case at the minimum containment water level postulated to occur 2 weeks into long-term cooling was also performed.

The DEDVI small-break LOCA exhibits no core uncovery due to its adequate reactor coolant system mass inventory condition during the long-term cooling phase from initiation into containment recirculation. Adequate flow through the core is provided to maintain a low cladding temperature and to prevent any buildup of boric acid on the fuel rods. The wall-to-wall floodup case using the window mode technique demonstrates that effective core cooling is also provided at the minimum containment water level. The results of these cases demonstrate the capability of the AP1000 passive systems to provide long-term cooling for a limiting LOCA event.

15.6.6 References

1. 10 CFR 50.46, “Acceptance Criteria for Emergency Core Cooling Systems for Light Water Cooled Nuclear Power Reactors,” and Appendix K to 10 CFR 50, “ECCS Evaluation Models.”

2. American Nuclear Society Proposed Standard, ANS 5.1 “Decay Energy Release Rates Following Shutdown of Uranium-Cooled Thermal Reactors,” October (1971), Revised October (1973).

3. “Final Safety Evaluation Report Related to Certification of the AP600 Standard Design,” NUREG-1512, September 1998.

4. Not used.

5. “Emergency Core Cooling Systems; Revision to Acceptance Criteria,” Federal Register, Vol. 53, No. 180, September 16, 1988.

6. Not used.

7. “AP600 Design Control Document,” Revision 3, December 1999.

8. Letter from R. C. Jones, Jr., (USNRC), to N. J. Liparulo, (W), Subject: “Acceptance for Referencing of the Topical Report, WCAP-12945 (P), Westinghouse CQD for Best Estimate LOCA Analysis,” June 28, 1996.

9. Not used.

10. Bajorek, S. M., et al., “Code Qualification Document for Best-Estimate LOCA Analysis,” WCAP-12945-P-A, Volume 1, Revision 2, and Volumes 2 through 5, Revision 1, and WCAP-14747 (Non-Proprietary), 1998.

11. Hochreiter, L. E., et al., “WCOBRA/TRAC Applicability to AP600 Large-Break Loss-of-Coolant Accident,” WCAP-14171, Revision 2 (Proprietary) and WCAP-14172, Revision 2 (Nonproprietary), March 1998.

12. Meyer, P. E., “NOTRUMP - A Nodal Transient Small-Break and General Network Code,” WCAP-10079-P-A (Proprietary) and WCAP-10080-A (Nonproprietary), August 1985.

13. Lee, N., Rupprecht, S. D., Schwarz, W. R., and Tauche, W. D., “Westinghouse Small-Break ECCS Evaluation Model Using the NOTRUMP Code,” WCAP-10054-P-A (Proprietary) and WCAP-10081-A (Nonproprietary), August 1985.

14. Carlin, E. L., Bachrach, U., “LOFTRAN & LOFTTR2 AP600 Code Applicability Document,” WCAP-14234, Revision 1 (Proprietary) and WCAP-14235, Revision 1 (Nonproprietary), August 1997.

15. Burnett, T. W. T., et al., “LOFTRAN Code Description,” WCAP-7907-P-A (Proprietary) and WCAP-7907-A (Nonproprietary), April 1984.

16. Friedland, A. J., Ray S., “Revised Thermal Design Procedure,” WCAP-11397-P-A (Proprietary) and WCAP-11397-A (Nonproprietary), April 1989.

17. Kemper, R. M., “AP600 Accident Analyses B Evaluation Models,” WCAP-14601, Revision 2 (Proprietary) and WCAP-15062, Revision 2 (Nonproprietary), May 1998.

18. Hargrove, H. G., “FACTRAN - A FORTRAN-IV Code for Thermal Transients in a UO2 Fuel Rod,” WCAP-7908-A, December 1989.

19. Soffer, L., et al., NUREG-1465, “Accident source Terms for Light-Water Nuclear Power Plants,” February 1995.

20. Regulatory Guide 1.183, “Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors,” July 2000.

21. Lewis, R. N., et al., “SGTR Analysis Methodology to Determine the Margin to Steam Generator Overfill,” WCAP-10698-P-A (Proprietary) and WCAP-10750-A (Nonproprietary), August 1987.

22. “NOTRUMP Final Validation Report for AP600,” WCAP-14807, Revision 5 (Proprietary) and WCAP-14808, Revision 2 (Nonproprietary), August 1998.

23. Garner, D. C., et al., “WCOBRA/TRAC OSU Long-Term Cooling Final Validation Report,” WCAP-14776, Revision 4 (Proprietary) and WCAP-14777, Revision 4 (Nonproprietary), April 1998.

24. “AP1000 Code Applicability Report,” WCAP-15644-P (Proprietary) and WCAP-15644-NP (Nonproprietary), Revision 2, March 2004.

25. “AP1000 PIRT and Scaling Assessment,” WCAP-15613 (Proprietary) and WCAP-15706 (Nonproprietary), February 2001.

26. Kemper, R. M., “Applicability of the NOTRUMP Computer Code to AP600 SSAR Small-Break LOCA Analyses,” WCAP-14206 (Proprietary) and WCAP-14207 (Nonproprietary), November 1994.

27. Not used.

28. Zuber, et al., “The Hydrodynamic Crisis in Pool Boiling of Saturated and Subcooled Liquids,” Part II, No. 27, International Developments in Heat Transfer, 1961.

29. Griffith, et al., “PWR Blowdown Heat Transfer,” Thermal and Hydraulic Aspects of Nuclear Reactor Safety, ASME, New York, Volume 1, 1977.

30. Chang, S. H. et al. “A study of critical heat flux for low flow of water in vertical round tubes under low pressure,” Nuclear Engineering and Design, 132, 225-237, 1991.

31. Not used.

32. Nissley, M. E., et al., 2005, “Realistic Large Break LOCA Evaluation Methodology Using the Automated Statistical Treatment of Uncertainty Method (ASTRUM),” WCAP-16009-P-A and WCAP-16009-NP-A (Non-proprietary).

33. Dederer, S. I., et al., 1999, “Application of Best Estimate Large Break LOCA Methodology to Westinghouse PWRs with Upper Plenum Injection,” WCAP-14449-P-A, Revision 1 and WCAP-14450 (Non-proprietary).

34. APP-GW-GLE-026, “Change to ASTRUM Methodology for Best Estimate Large Break Loss of Coolant Accident Analysis,” Westinghouse Electric Company LLC.

35. Beahm, E. C. et al., NUREG/CR-5950, “Iodine Evolution and pH Control,” December 1992.

Table 15.6.1-1
Time Sequence of Events for Incidents That Cause a Decrease in Reactor Coolant Inventory

Accident	Event	Time (seconds)
Inadvertent opening of a pressurizer safety valve with offsite power available	Pressurizer safety valve opens fully	0.0
	Overtemperature ΔT reactor trip setpoint reached	18.55
	Rods begin to drop	20.55
	Minimum DNBR occurs	21.3
Inadvertent opening of a pressurizer safety valve without offsite power available	Pressurizer safety valve opens fully	0.0
	Overtemperature ΔT reactor trip setpoint reached	18.55
	Turbine trip signal	20.23
	Rods begin to drop	20.55
	Minimum DNBR occurs	21.3
	ac power lost, reactor coolant pumps begin coasting down	23.23
Inadvertent opening of two ADS Stage 1 trains with offsite power available	ADS valves begin to open	0.0
	Overtemperature ΔT reactor trip setpoint reached	18.40
	Rods begin to drop	20.40
	Minimum DNBR occurs	21.30
	ADS valves fully open	25.0
Inadvertent opening of two ADS Stage 1 trains without offsite power available	ADS valves begin to open	0.0
	Overtemperature ΔT reactor trip setpoint reached	18.40
	Turbine trip signal	20.1
	Rods begin to drop	20.40
	Minimum DNBR occurs	21.3
	ac power lost, reactor coolant pumps begin coasting down	23.1
	ADS valves fully open	25.0

Table 15.6.2-1
Parameters Used in Evaluating the Radiological Consequences of a Small Line Break Outside Containment

Reactor coolant iodine activity	Initial activity equal to the design basis reactor coolant activity of 1.0 μCi/g dose equivalent I-131 with an assumed iodine spike that increases the rate of iodine release from fuel into the coolant by a factor of 500 (see Table 15A-2 in Appendix 15A)[a]
Reactor coolant noble gas activity	280 μCi/g dose equivalent Xe-133
Break flow rate (gpm)	130[b]
Fraction of reactor coolant flashing	0.47
Duration of accident (hr)	0.5
Atmospheric dispersion (χ/Q) factors	See Table 15A-5
Nuclide data	See Table 15A-4

Notes:
a. Use of accident-initiated iodine spike is consistent with the guidance in the Standard Review Plan.
b. At density of 62.4 lb/ft^3.

Table 15.6.3-1
Steam Generator Tube Rupture Sequence of Events

Events	Time (seconds)
Double-ended steam generator tube rupture	0
Loss of offsite power	0
Reactor trip	0
Reactor coolant pumps and main feedwater pumps assumed to trip and begin to coastdown	0
Two chemical and volume control pumps actuated and pressurizer heaters turned on	0
Low-2 pressurizer level signal generated	2,498
Ruptured steam generator power-operated relief valve fails open	2,498
Core makeup tank injection and PRHR operation begins (following maximum delay)	2,515
Ruptured steam generator power-operated relief valve block valve closes on low steam line pressure signal	2,979
Chemical and volume control system isolated on high-2 steam generator narrow range level setpoint	12,541
Break flow terminated	24,100

Table 15.6.3-2
Steam Generator Tube Rupture Mass Release Results

Total Mass Flow from Initiation of Event to Cooldown to RNS[1] Conditions

	Start of Event to Break Flow Termination (Pounds Mass)	Break Flow Termination to Cut-in of RHR (Pounds Mass)
Ruptured steam generator – Atmosphere	238,600	93,200
Intact steam generator – Atmosphere	183,400	1,234,900
Break flow	385,000	0

Note:
1. RNS = normal residual heat removal

Table 15.6.3-3
Parameters Used in Evaluating the Radiological Consequences of a Steam Generator Tube Rupture

Reactor coolant iodine activity	
– Accident initiated spike	Initial activity equal to the equilibrium operating limit for reactor coolant activity of 1.0 µCi/g dose equivalent I-131 with an assumed iodine spike that increases the rate of iodine release from fuel into the coolant by a factor of 335 (see Appendix 15A). Duration of spike is 8.0 hours.
– Preaccident spike	An assumed iodine spike that results in an increase in the reactor coolant activity to 60 µCi/g of dose equivalent I-131 (see Appendix 15A)
Reactor coolant noble gas activity	280 µCi/g dose equivalent Xe-133
Reactor coolant alkali metal activity	Design basis activity (see Table 11.1-2)
Secondary coolant initial iodine and alkali metal	1% of reactor coolant concentrations at maximum equilibrium conditions
Reactor coolant mass (lb)	3.7 E+05
Offsite power	Lost on reactor trip
Condenser	Lost on reactor trip
Time of reactor trip	Beginning of the accident
Duration of steam releases (hr)	15.94
Atmospheric dispersion factors	See Appendix 15A
Nuclide data	See Appendix 15A
Steam generator in ruptured loop	
– Initial secondary coolant mass (lb)	1.16 E+05
– Primary-to-secondary break flow	See Figure 15.6.3-5
– Integrated flashed break flow (lb)	See Figure 15.6.3-10
– Steam released (lb)	See Table 15.6.3-2
– Iodine partition coefficient	1.0 E-02[a]
– Alkali metals partition coefficient	3.5 E-03[a]
Steam generator in intact loop	
– Initial secondary coolant mass (lb)	2.30 E+04
– Primary-to-secondary leak rate (lb/hr)	52.16[b]
– Steam released (lb)	See Table 15.6.3-2
– Iodine partition coefficient	1.0 E-02[a]
– Alkali metals partition coefficient	3.5 E-03[a]

Notes:

a. Partition coefficient does not apply to flashed break flow.

b. Equivalent to 150 gpd at psia cooled liquid at 62.4 lb/ft^3.

Table 15.6.5-1
Core Activity Releases to the Containment Atmosphere

Nuclide	Gap Release Released over 0.5 hr. (0.167 - 0.667 hr)[(1)]	Core Melt In-vessel Release (0.667 - 1.967 hr)[(1)]
Noble gases	0.05	0.95
Iodines	0.05	0.35
Alkali metals	0.05	0.25
Tellurium group	–	0.05
Strontium and barium	–	0.02
Noble metals group	–	0.0025
Cerium group	–	0.0005
Lanthanide group	–	0.0002

Notes:
1. Releases are stated as fractions of the original core fission product inventory.
2. Dash (–) indicates not applicable.

Table 15.6.5-2 (Sheet 1 of 3)
Assumptions and Parameters Used in Calculating Radiological Consequences of a Loss-of-Coolant Accident

Primary coolant source data	
– Noble gas concentration	280 µCi/g dose equivalent Xe-133
– Iodine concentration	1.0 µCi/g dose equivalent I-131
– Primary coolant mass (lb)	4.39 E+05
Containment purge release data	
– Containment purge flow rate (cfm)	16000
– Time to isolate purge line (seconds)	30
– Time to blow down the primary coolant system (minutes)	10
– Fraction of primary coolant iodine that becomes airborne	1.0
Core source data	
– Core activity at shutdown	See Table 15A-3
– Release of core activity to containment atmosphere (timing and fractions)	See Table 15.6.5-1
– Iodine species distribution (%)	
• Elemental	4.85
• Organic	0.15
• Particulate	95
Containment leakage release data	
– Containment volume (ft^3)	2.06 E+06
– Containment leak rate, 0-24 hr (% per day)	0.10
– Containment leak rate, > 24 hr (% per day)	0.05
– Elemental iodine deposition removal coefficient (hr^{-1})	1.9
– Decontamination factor limit for elemental iodine removal	200
– Removal coefficient for particulates (hr^{-1})	See Appendix 15B
Main control room model	
– Main control room volume (ft^3)	3.89 E+04
– Volume of HVAC, including main control room and control support area (ft^3)	1.2 E+05
– Normal HVAC operation (prior to switchover to an emergency mode)	
• Air intake flow (cfm)	1650
• Filter efficiency	Not applicable
– Atmospheric dispersion factors (sec/m^3)	See Table 15A-6

Table 15.6.5-2 (Sheet 2 of 3)
Assumptions and Parameters Used in Calculating Radiological Consequences of a Loss-of-Coolant Accident

Main control room model (cont.)	
– Occupancy	
• 0 –24 hr	1.0
• 24–96 hr	0.6
• 96–720 hr	0.4
– Breathing rate (m^3/sec)	3.5 E-04
Control room with emergency habitability system credited (VES Credited)	
– Main control room activity level at which the emergency habitability system actuation is actuated (Ci/m^3 of dose equivalent I-131)	2.0 E-07
– Response time to actuate VES based on radiation monitor response time and VBS isolation (sec)	200
– Interval with operation of the emergency habitability system	
• Flow from compressed air bottles of the emergency habitability system (cfm)	60
• Unfiltered inleakage via ingress/egress (scfm)	5
• Unfiltered inleakage from other sources (scfm)	10
• Recirculation flow through filters (scfm)	600
• Filter efficiency (%)	
• Elemental iodine	90
• Organic iodine	90
• Particulates	99
– Time at which the compressed air supply of the emergency habitability system is depleted (hr)	72
– After depletion of emergency habitability system bottled air supply (>72 hr)	
• Air intake flow (cfm)	1900
• Intake flow filter efficiency (%)	Not applicable
• Recirculation flow (cfm)	Not applicable
– Time at which the compressed air supply is restored and emergency habitability system returns to operation (hr)	168

Table 15.6.5-2 (Sheet 3 of 3)
Assumptions and Parameters Used in Calculating Radiological Consequences of a Loss-of-Coolant Accident

Control room with credit for continued operation of HVAC (VBS Supplemental Filtration Mode Credited)	
– Time to switch from normal operation to the supplemental air filtration mode (sec)	265
– Unfiltered air inleakage (cfm)	25
– Filtered air intake flow (cfm)	860
– Filtered air recirculation flow (cfm)	2740
– Filter efficiency (%)	
• Elemental iodine	90
• Organic iodine	90
• Particulates	99
Miscellaneous assumptions and parameters	
– Offsite power	Not applicable
– Atmospheric dispersion factors (offsite)	See Table 15A-5
– Nuclide dose conversion factors	See Table 15A-4
– Nuclide decay constants	See Table 15A-4
– Offsite breathing rate (m^3/sec)	
0–8 hr	3.5 E-04
8–24 hr	1.8 E-04
24–720 hr	2.3 E-04

Table 15.6.5-3
Radiological Consequences of a
Loss-of-Coolant Accident With Core Melt

	TEDE Dose (rem)
Exclusion zone boundary dose (1.3 - 3.3 hr)[1]	23.5
Low population zone boundary dose (0 - 30 days)	22.2
Main control room dose (emergency habitability system in operation)	
– Airborne activity entering the main control room	3.70
– Direct radiation from adjacent structures, including sky shine	0.30
– Filter shine	0.32
– Spent fuel pooling boiling	0.01
– Total	4.33
Main control room dose (normal HVAC operating in the supplemental filtration mode)	
– Airborne activity entering the main control room	4.50
– Direct radiation from adjacent structures, including sky shine	0.30
– Filter shine	0.03
– Spent fuel pooling boiling	0.01
– Total	4.84

Note:
1. This is the 2-hour period having the highest dose.

Table 15.6.5-4
Major Plant Parameter Assumptions
Used in the Best-Estimate Large-Break LOCA Analysis

Parameter	Value
Plant Physical Configuration	
Steam generator tube plugging level	≤ 10% (10% tube plugging bounds 0%)
Hot assembly location	Under support column (Bounds under open hole or guide tube)
Pressurizer location	In intact loop (Bounds location in broken loop)
Initial Operating Conditions	
Reactor power	Core power < 1.01*3400 MWt
Peak linear heat rate	$F_Q \leq 2.6$
Hot rod assembly power	$F_{\Delta H} \leq 1.75$
Hot assembly power	$P_{HA} \leq 1.683$
Axial power distribution[(1)]	See Figure 15.6.5.4A-13
Peripheral assembly power	$0.2 \leq P_{LOW} \leq 0.8$
Fluid Conditions	
Reactor coolant system average temperature	573.6 – 7.5°F ≤ T_{AVG} ≤ 573.6 + 7.5°F
Pressurizer pressure	2250 ± 50 psia
Pressurizer level (water volume)	1000 ft^3 (nominal)
Accumulator temperature	50°F ≤ T_{ACC} ≤ 120°F
Accumulator pressure	651.7 psia ≤ P_{ACC} ≤ 783.7 psia
Accumulator water volume	1667 ft^3 ≤ V_{ACC} ≤ 1732 ft^3
Reactor Coolant System Boundary Conditions	
Single failure assumption	Failure of one CMT isolation valve to open
Offsite power availability	Available (Bounds loss of offsite power at time zero)
Reactor coolant pump automatic trip delay time after receiving S-signal	4 s
Containment pressure	Bounded (minimum)

Note:
1. Treatment of axial power distribution consistent with WCAP-16009-P-A (Reference 32) methodology.

Table 15.6.5-5
AP1000 LOCA Chronology

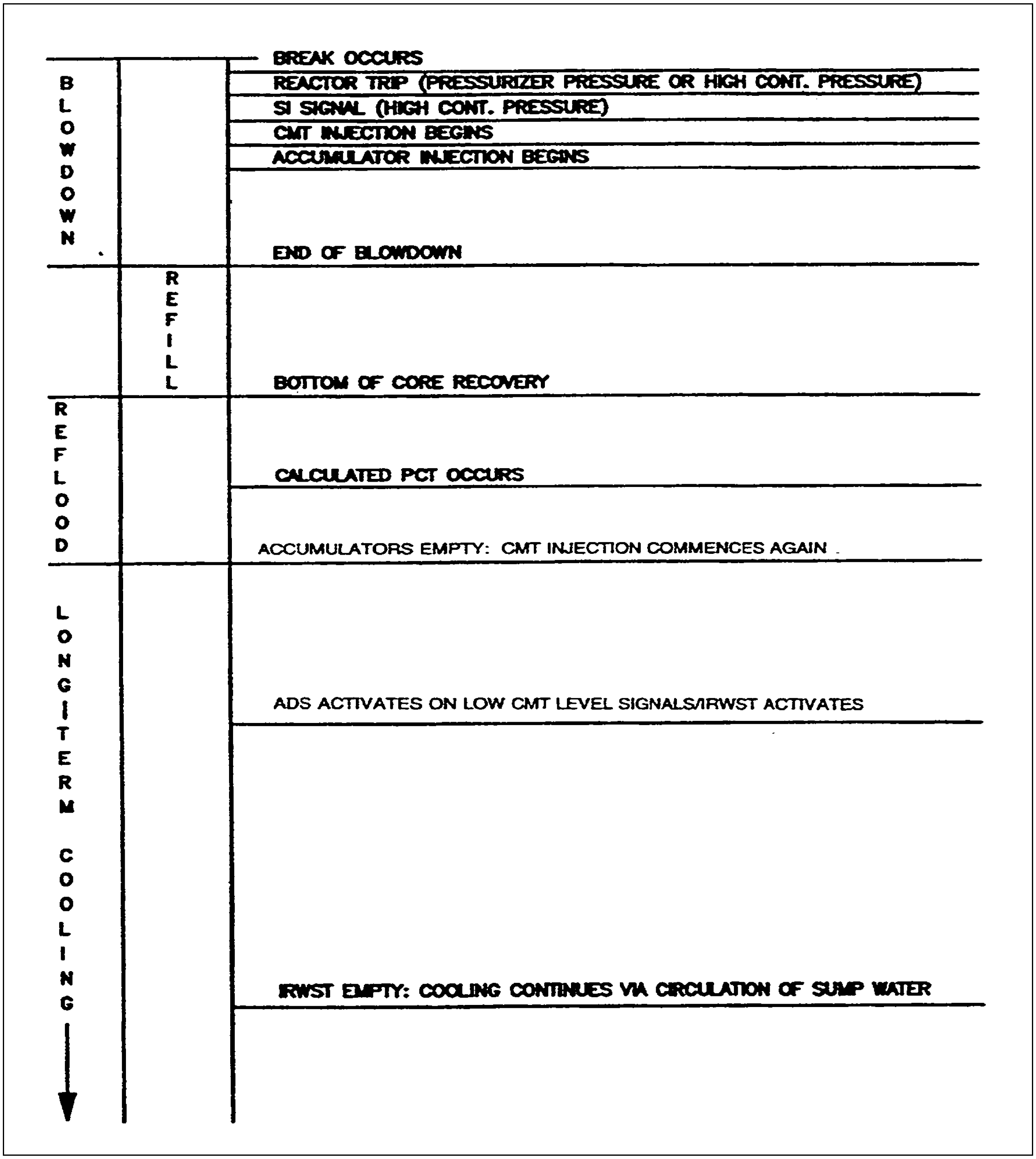


Table 15.6.5-6
Best-Estimate Large-Break Sequence of Events for the Limiting PCT/MLO Case

Event	Time (seconds)
Break initiation	0.0
Safeguards signal	2.2
CMT isolation valves begin to open	4.2
Reactor coolant pumps trip	8.2
Accumulator injection begins	18
End of blowdown	34.5
Bottom of core recovery	54.0
Calculated PCT occurs	~65
Core quench occurs	~115
CMT injection resumes	~150
End of transient	231

Table 15.6.5-7 Not Used

Table 15.6.5-8
Best-Estimate Large-Break LOCA Results

10 CFR 50.46 Requirement	Value	Criteria
Calculated 95th percentile PCT (°F)	1837	≤ 2200
Maximum local cladding oxidation (%)	2.25	≤ 17
Maximum core-wide cladding oxidation (%)	0.2	≤ 1
Coolable geometry	Core remains coolable	Core remains coolable
Long-term cooling	Core remains cool in long term	Core remains cool in long term

Table 15.6.5-9
Initial Conditions for AP1000 Small-Break LOCA Analysis

Condition	Calculation	Nominal Steady-state
Pressurizer pressure (psia)	2303.1	2300
Vessel inlet temperature (°F)	534.03	534.3
Vessel outlet temperature (°F)	612.83	612.9
Vessel flow rate (lb/sec)	31086	31089
Steam generator pressure (psia)	806.5	788.5

Table 15.6.5-10
AP1000 ADS Parameters(1)

Actuation Signal (percentage of core makeup tank level)		**Actuation Time (seconds)**	**Minimum Valve Flow Area (for each path, in^2)**	**Number of Paths**	**Valve Opening Time (seconds)**
Stage 1 — Control Low 1	67.5	32 after CMT-Low 1	4.6	2 out of 2	≤ 40
Stage 2 — Control		48 after Stage 1	21	2 out of 2	≤ 100
Stage 3 — Control		120 after Stage 2	21	2 out of 2	≤ 100
Stage 4A	20	128 after Stage 3(2)	67	1 out of 2	≤ 4(3)
Stage 4B		60 after Stage 4A	67	2 out of 2	≤ 4(3)

Notes:
1. The valve stroke times reflect the design basis of the AP1000. The applicable Chapter 15 accidents were evaluated for the design basis valve stroke times. The results of this evaluation have shown that there is a small impact on the analysis and the conclusions remain valid. The output provided for the analyses is representative of the transient phenomenon.
2. The interlock requires coincidence of CMT Low-2 level as well as 128 seconds after the Stage 3 actuation signal is generated.
3. This includes “arm-fire” processing delay and the assumed valve opening time.

Table 15.6.5-11
Inadvertent ADS Depressurization Sequence of Events

Event	AP1000 Time (seconds)
Inadvertent opening of ADS valves	0.0
Reactor trip signal	37.8
Steam turbine stop valves close	43.8
“S” signal	44.1
Main feed isolation valves begin to close	49.1
Reactor coolant pumps start to coast down	50.1
ADS Stage 2	70.0
ADS Stage 3	190.0
Accumulator injection starts	268
Accumulator empties	693
ADS Stage 4	1746
Core makeup tank empty	2112
IRWST injection starts	2663

Table 15.6.5-12
2-Inch Cold Leg Break in CLBL Line Sequence of Events

Event	AP1000 Time (seconds)
Break opens	0.0
Reactor trip signal	54.7
Steam turbine stop valves close	60.7
"S" signal	61.9
Main feed isolation valves begin to close	63.9
Reactor coolant pumps start to coast down	67.9
ADS Stage 1	1334.1
ADS Stage 2	1404.1
Accumulator injection starts	1405
ADS Stage 3	1524.1
Accumulator empties	1940.2
ADS Stage 4	2418.6
Core makeup tank empty	2895
IRWST injection starts	3280

Table 15.6.5-13
Double-Ended Injection Line Break Sequence of Events – 20 psi

Event	AP1000 Time (seconds)
Break opens	0.0
Reactor trip signal	13.1
Steam turbine stop valves close	19.1
“S” signal	18.6
Main feed isolation valves begin to close	20.6
Reactor coolant pumps start to coast down	24.6
ADS Stage 1	182.5
ADS Stage 2	252.5
Intact accumulator injection starts	254
ADS Stage 3	372.5
ADS Stage 4	492.5
Intact accumulator empties	600.0
Intact loop IRWST injection starts*	1470
Intact loop core makeup tank empties	2123

Note:
* Continuous injection period

Table 15.6.5-13A
Double-Ended Injection Line Break Sequence of Events – 14.7 psi

Event	AP1000 Time (seconds)
Break opens	0.0
Reactor trip signal	13.1
Steam turbine stop valves close	19.1
“S” signal	18.5
Main feed isolation valves begin to close	20.5
Reactor coolant pumps start to coast down	24.5
ADS Stage 1	182.7
Intact accumulator injection starts	251
ADS Stage 2	252.7
ADS Stage 3	372.7
ADS Stage 4	492.7
Intact accumulator empties	598.4
Intact loop core makeup tank empties	2006
Intact loop IRWST injection starts*	2076

Note:
* Continuous injection period

Table 15.6.5-14
10-inch Cold Leg Break in Sequence of Events

Event	AP1000 Time (seconds)
Break opens	0.0
Reactor trip signal	5.2
“S” signal	6.4
Main feed isolation valves begin to close	8.4
Steam turbine stop valves close	11.2
Reactor coolant pumps start to coast down	12.4
Accumulator injection starts	85.
Accumulator 1 empties	418.2
Accumulator 2 empties	425.5
ADS Stage 1	750.0
ADS Stage 2	820.
ADS Stage 3	940.
ADS Stage 4	1491.
Core makeup tank 2 empty	1800.*
IRWST injection starts	~1800
Core makeup tank 1 empty	1900.*

Note:
* The CMTs never truly empty although they cease to discharge at these times.

Table 15.6.5-15
Double-Ended Injection Line Break Sequence of Events
(Entrainment Sensitivity)

Event	AP1000 Time (seconds)
Break opens	0.0
Reactor trip signal	13.1
Steam turbine stop valves close	19.1
“S” signal	18.6
Main feed isolation valves begin to close	20.6
Reactor coolant pumps start to coast down	24.6
ADS Stage 1	182.8
ADS Stage 2	252.8
Intact accumulator injection starts	255
ADS Stage 3	372.8
ADS Stage 4	492.8
Intact accumulator empties	608.9
Intact loop IRWST injection starts*	1711
Intact loop core makeup tank empties	2095

Note:
* Continuous injection period

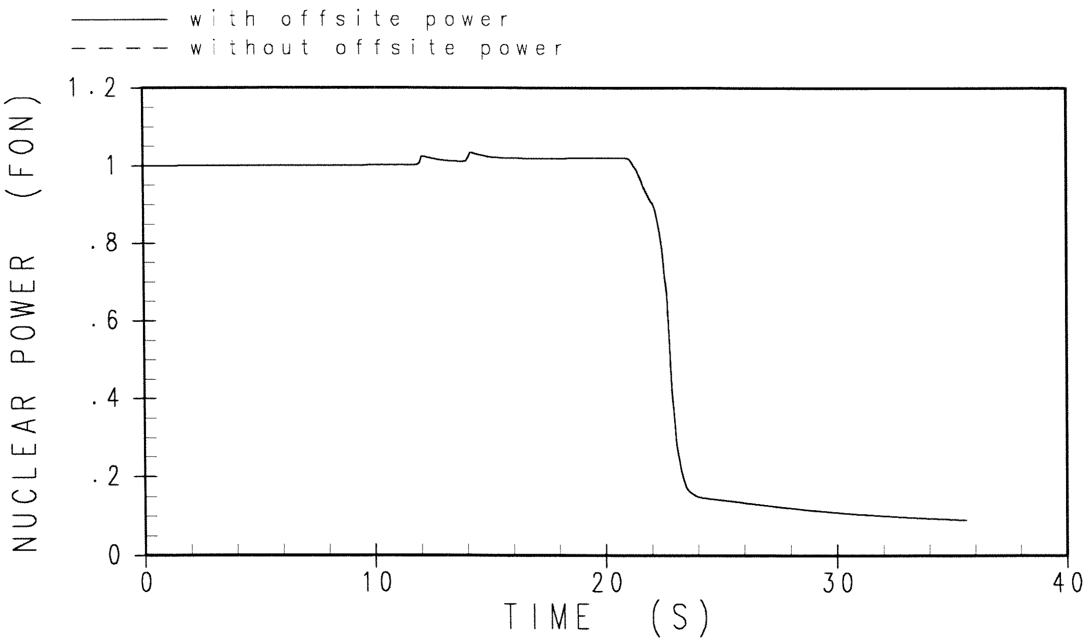


Figure 15.6.1-1
Nuclear Power Transient
Inadvertent Opening of a Pressurizer Safety Valve

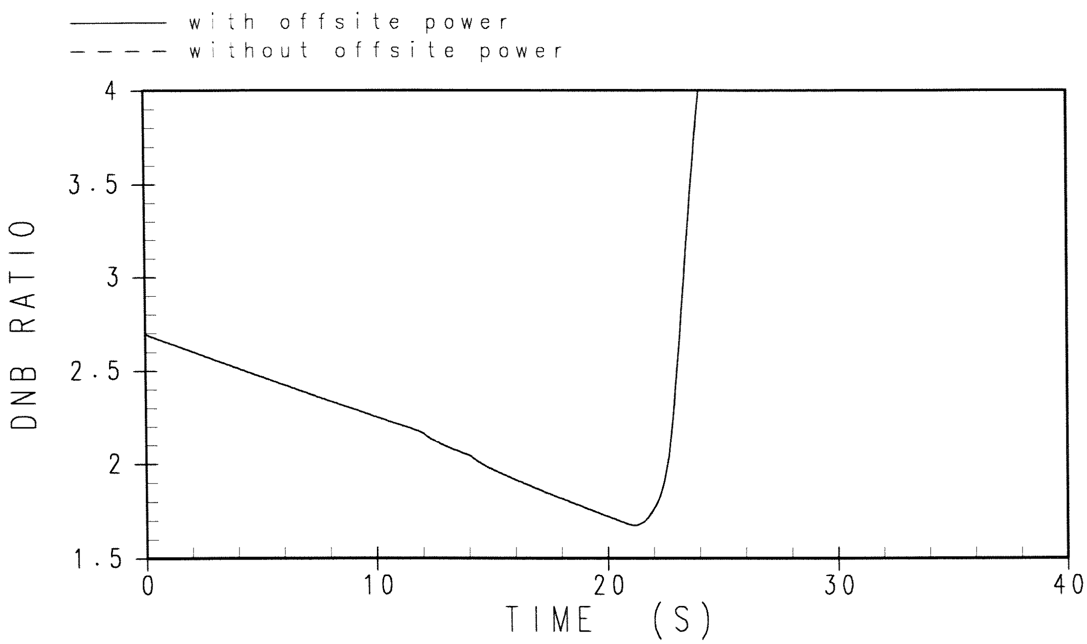


Figure 15.6.1-2
DNBR Transient
Inadvertent Opening of a Pressurizer Safety Valve

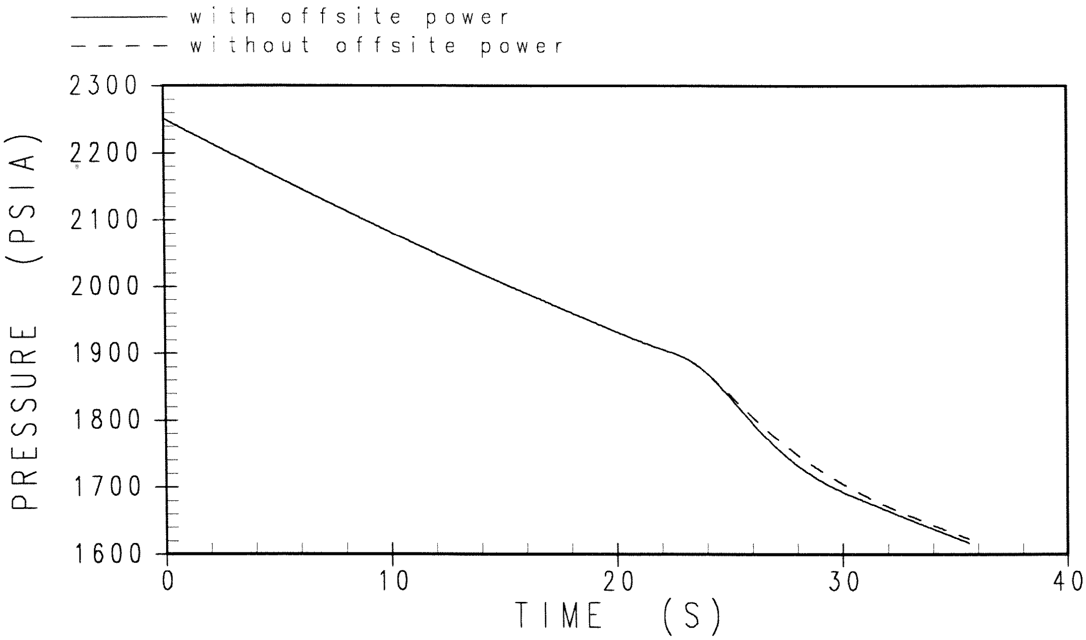


Figure 15.6.1-3
Pressurizer Pressure Transient
Inadvertent Opening of a Pressurizer Safety Valve

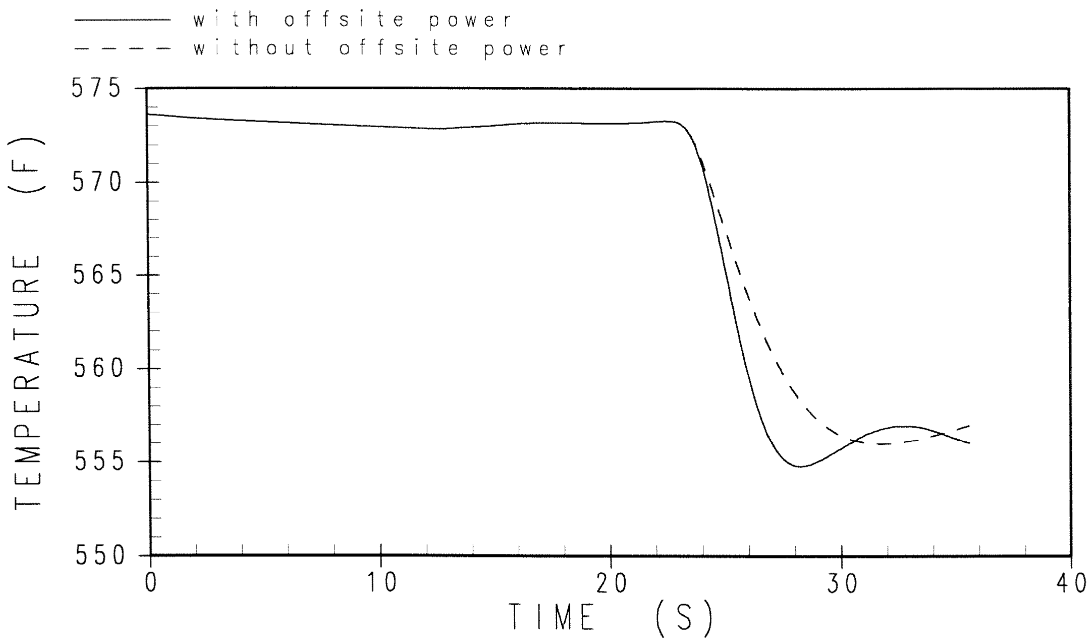


Figure 15.6.1-4
Vessel Average Temperature
Inadvertent Opening of a Pressurizer Safety Valve

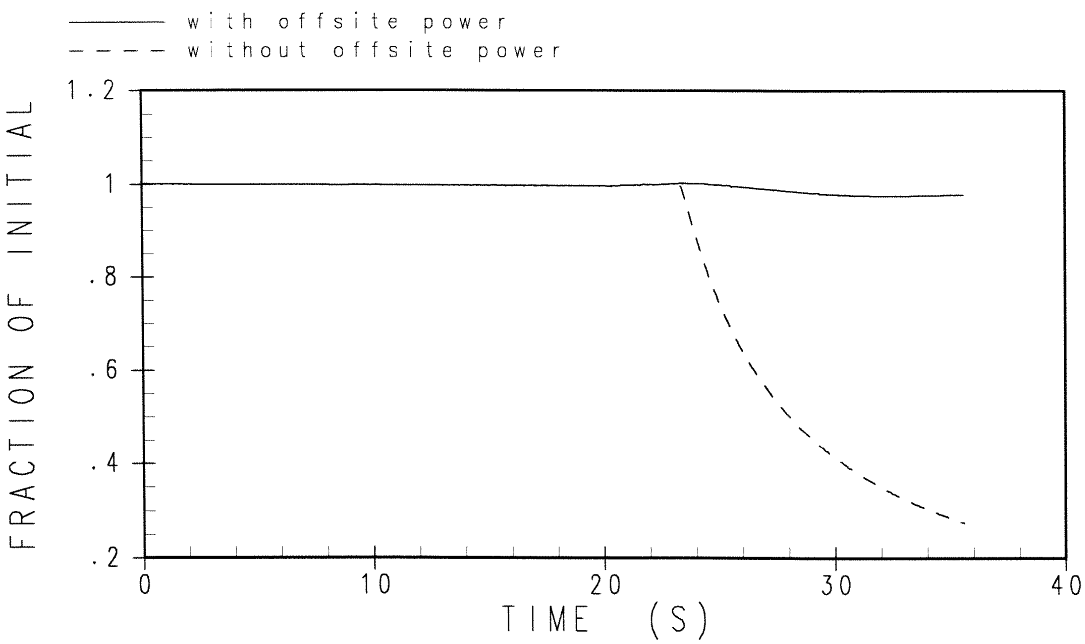


Figure 15.6.1-5
Core Mass Flow Rate
Inadvertent Opening of a Pressurizer Safety Valve

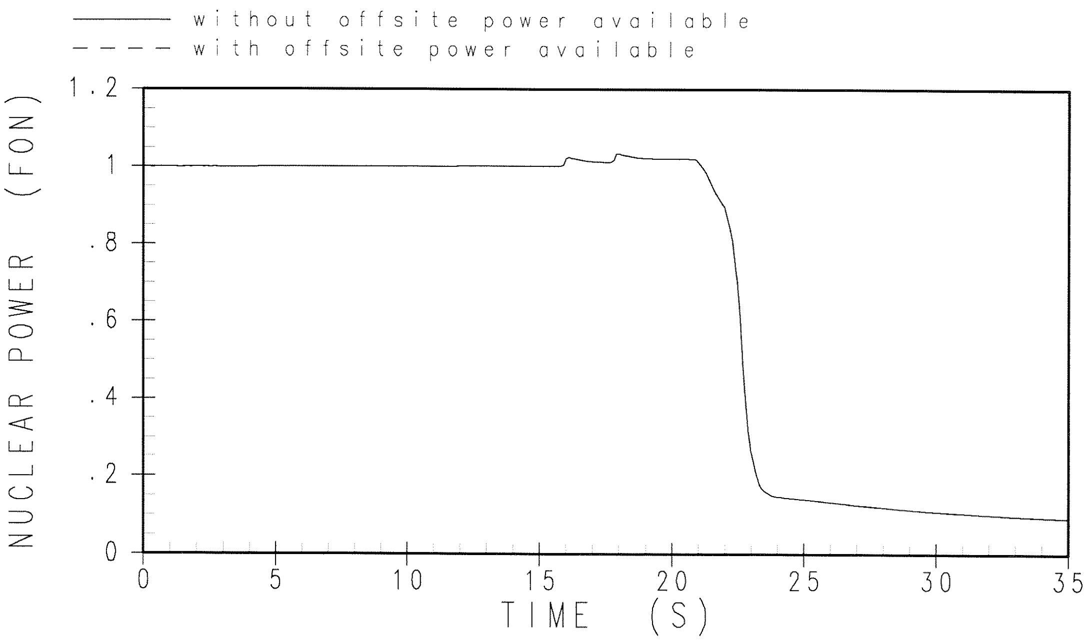


Figure 15.6.1-6
Nuclear Power Transient
Inadvertent Opening of Two ADS Stage 1 Trains

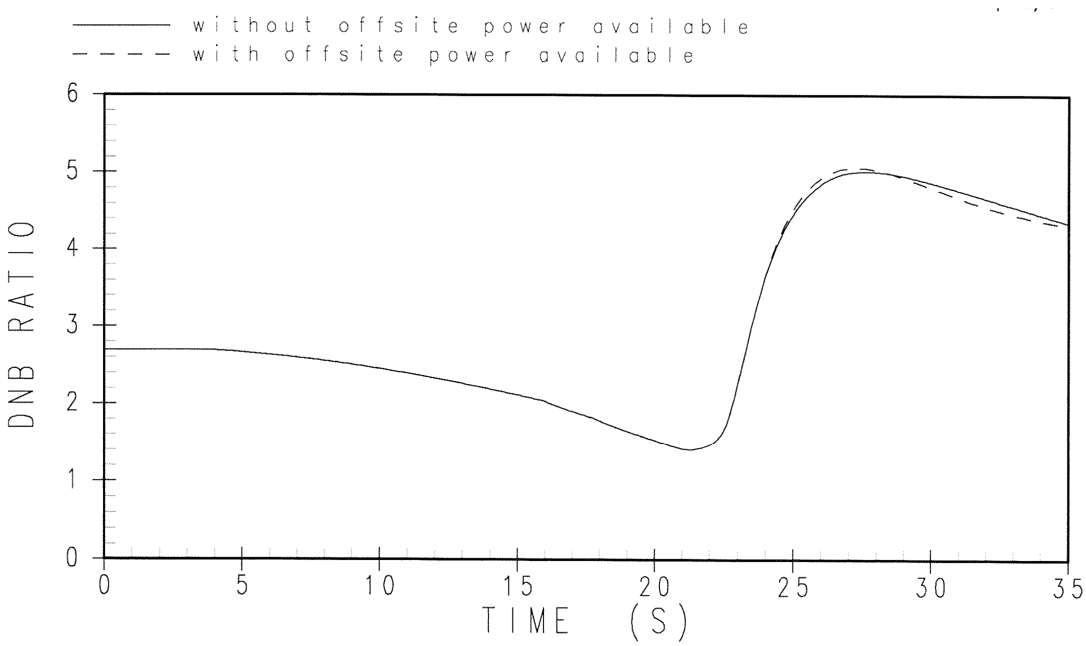


Figure 15.6.1-7
DNBR Transient
Inadvertent Opening of Two ADS Stage 1 Trains

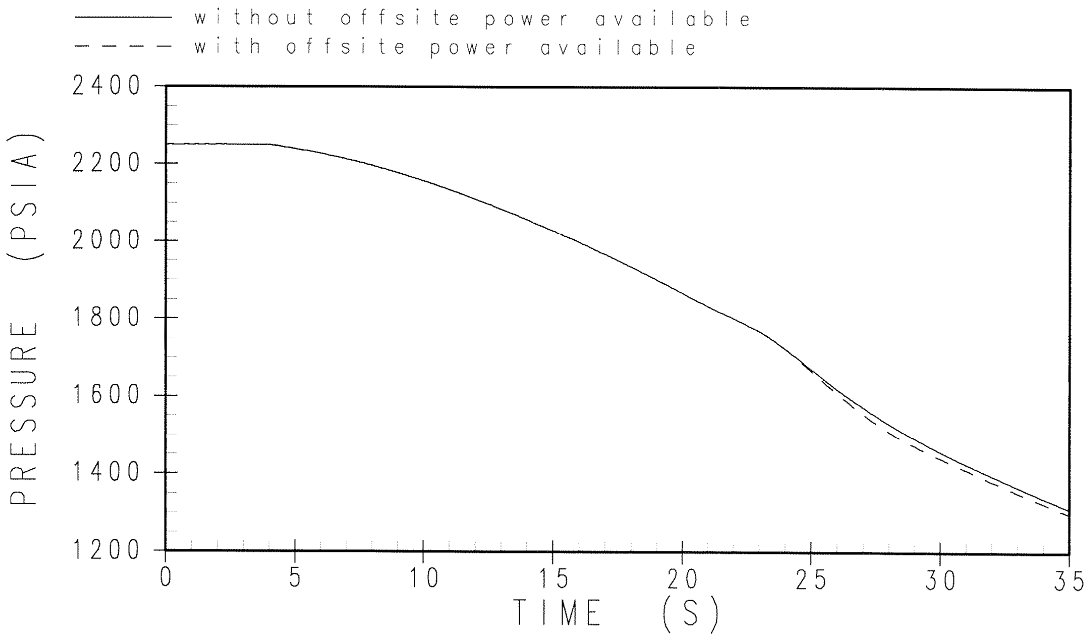


Figure 15.6.1-8
Nuclear Power Transient
Inadvertent Opening of Two ADS Stage 1 Trains

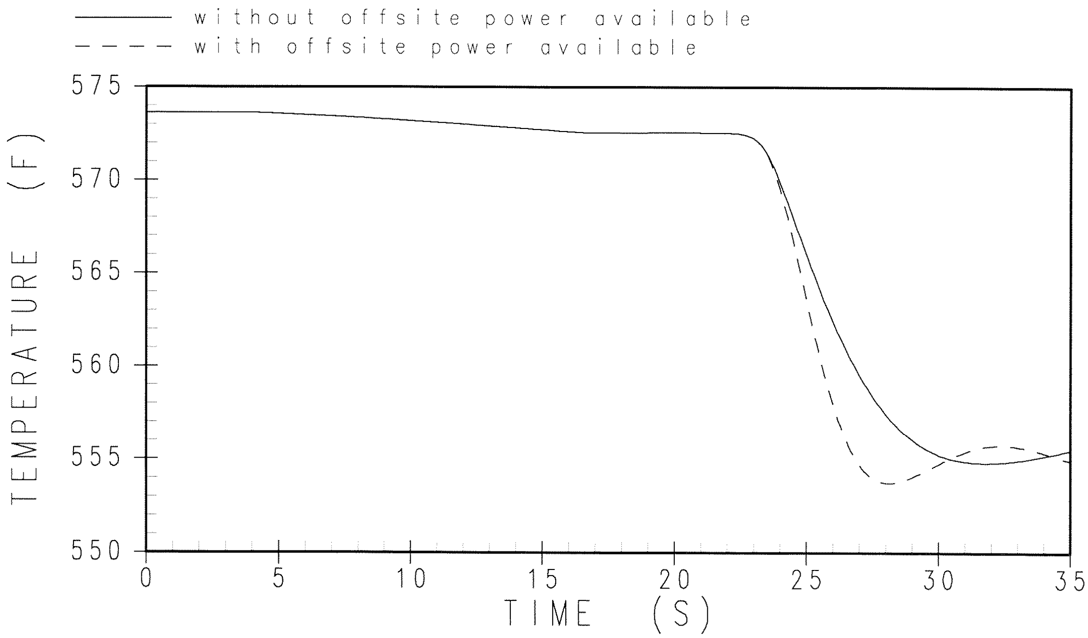


Figure 15.6.1-9
Nuclear Power Transient
Inadvertent Opening of Two ADS Stage 1 Trains

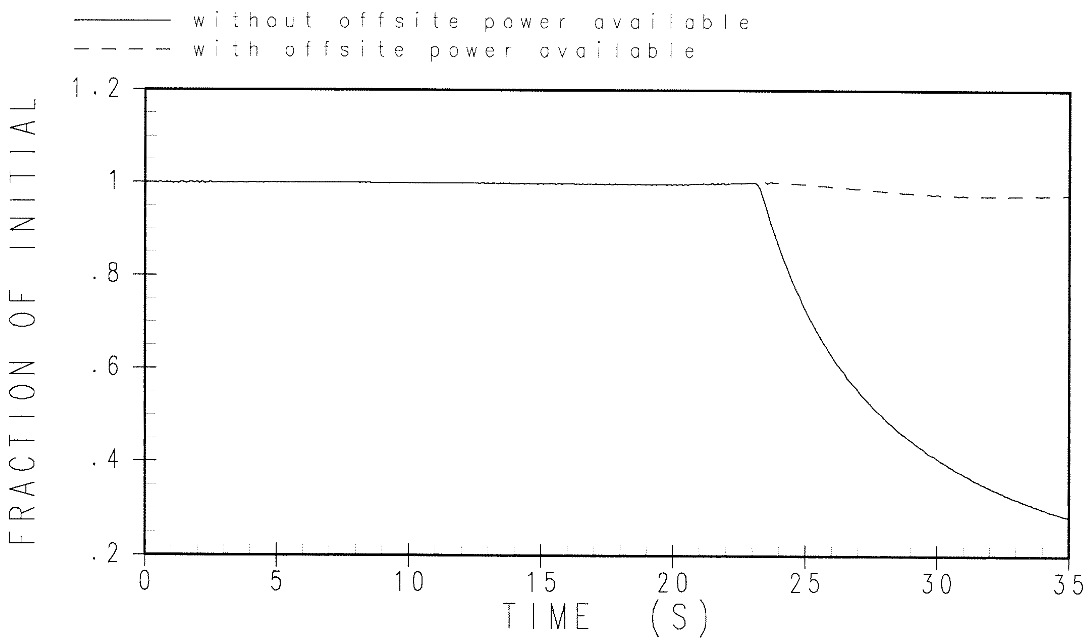


Figure 15.6.1-10
Core Mass Flow Rate
Inadvertent Opening of Two ADS Stage 1 Trains

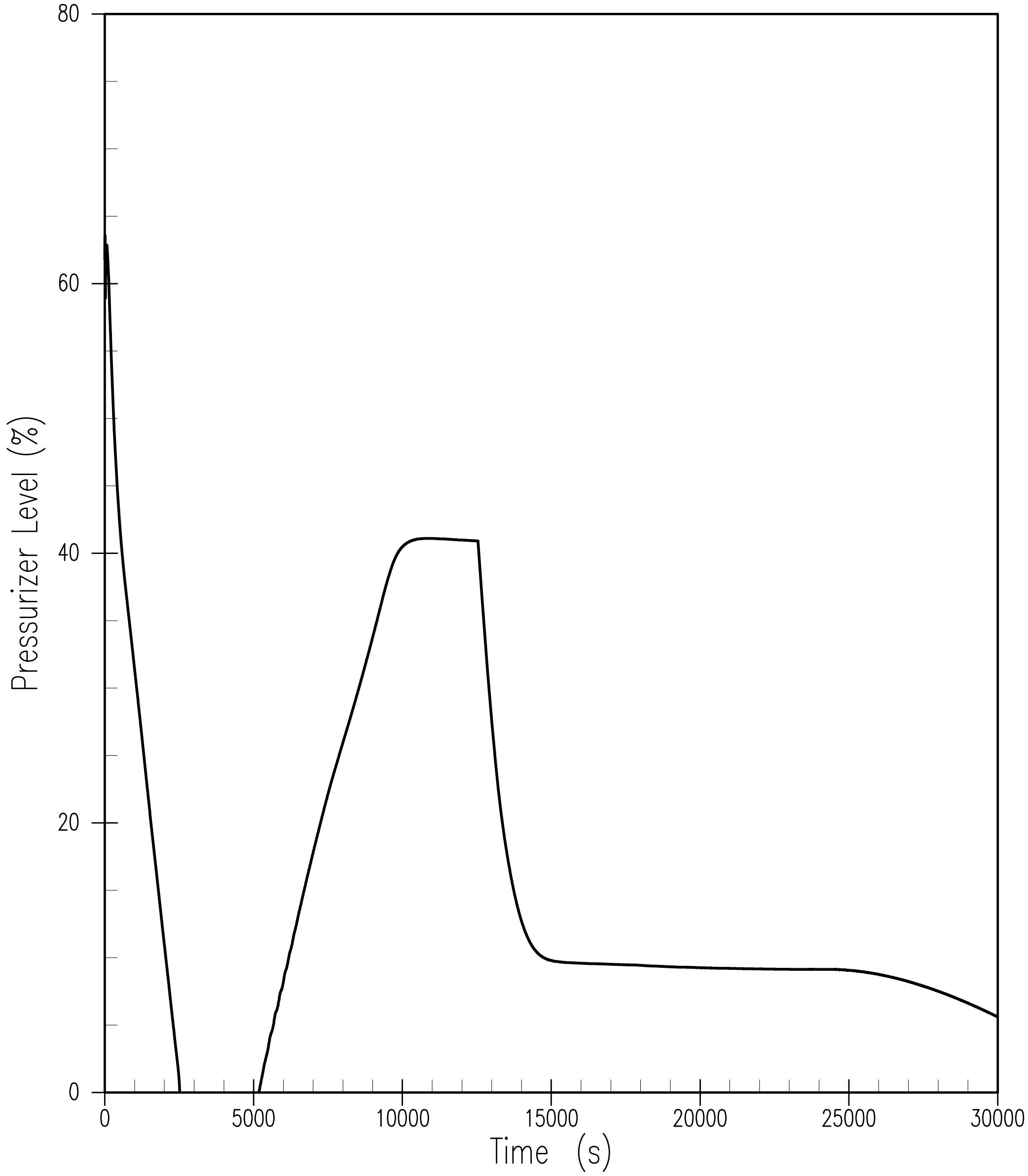


Figure 15.6.3-1
Pressurizer Level for SGTR

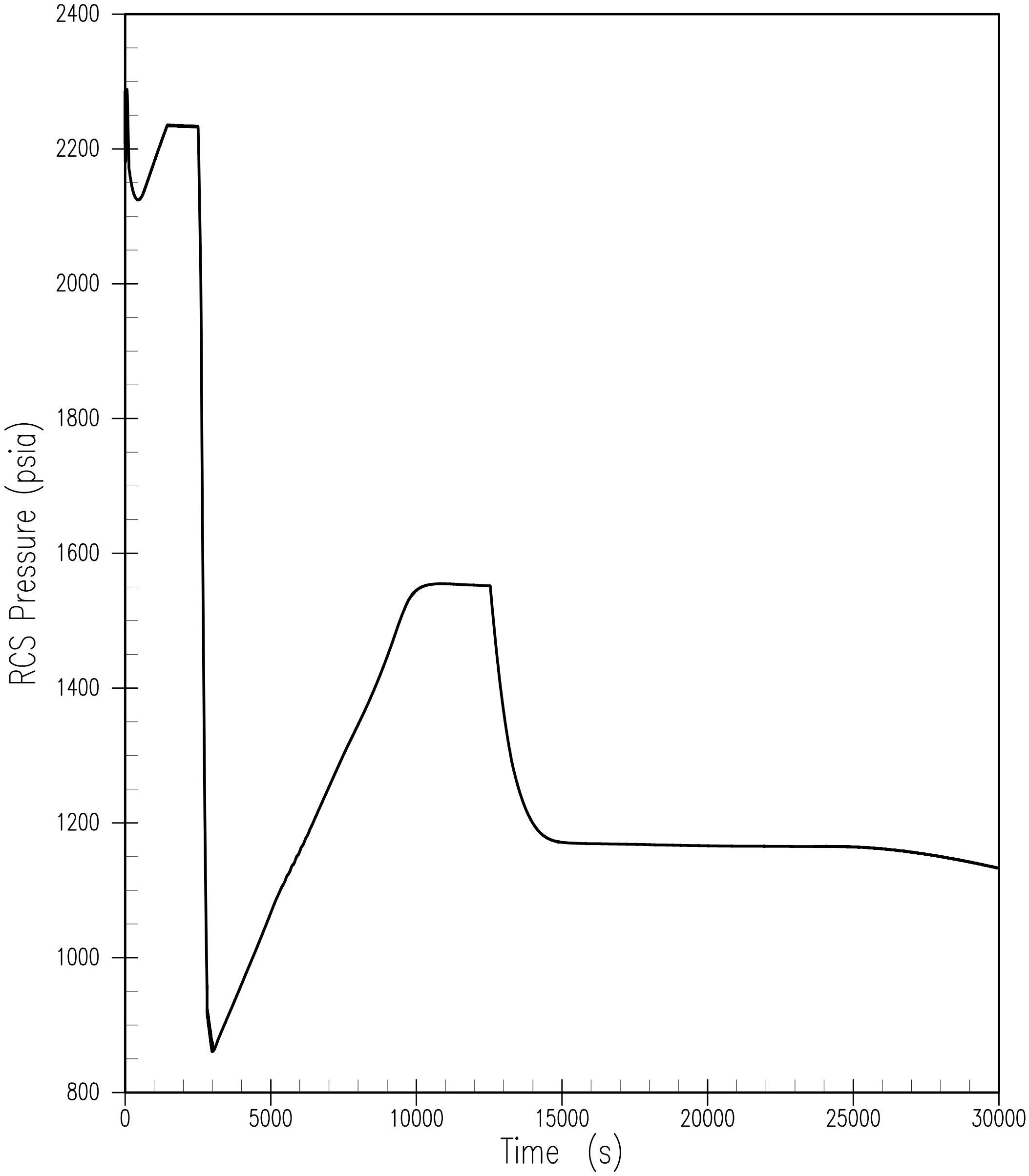


Figure 15.6.3-2
Reactor Coolant System Pressure for SGTR

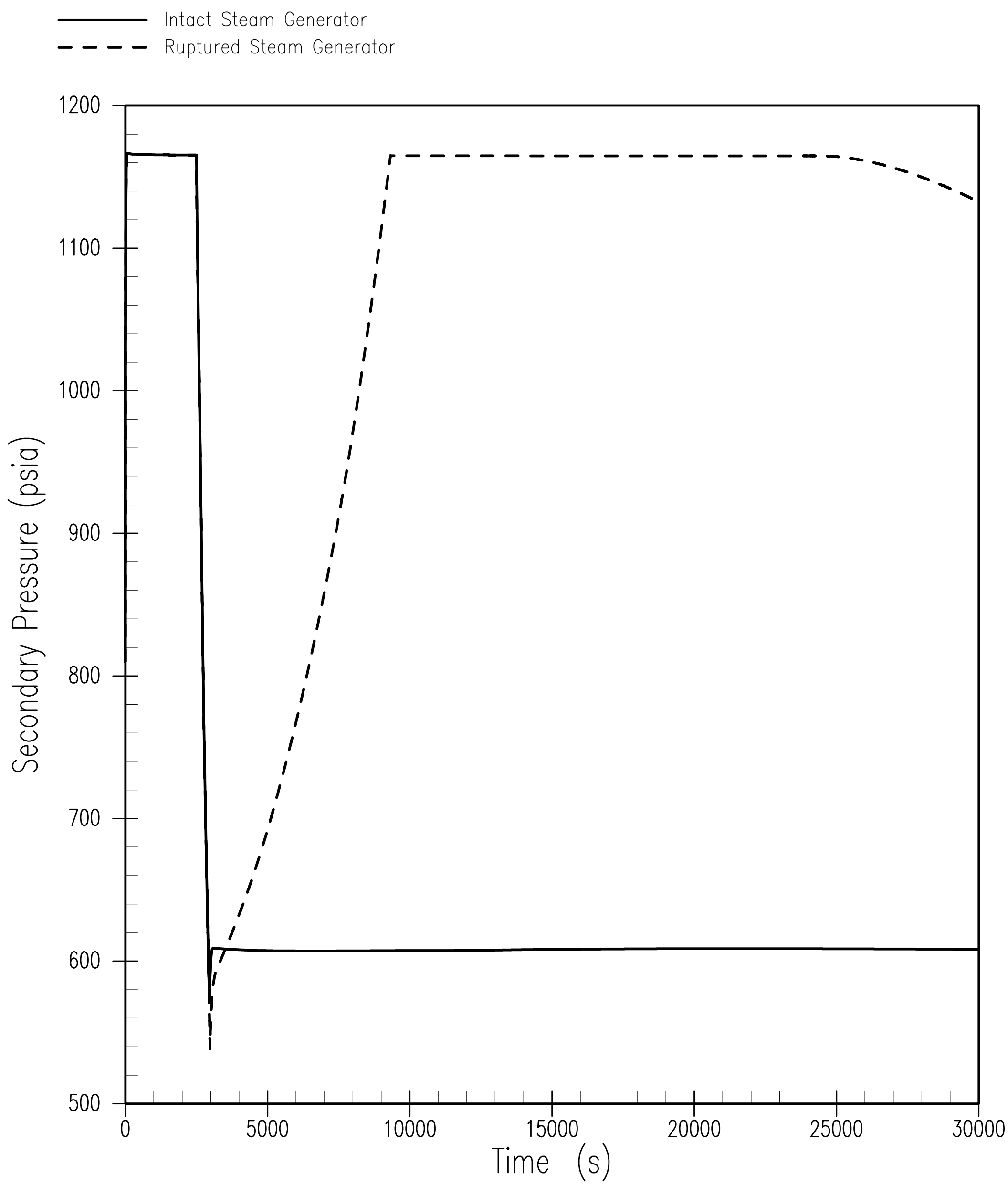


Figure 15.6.3-3
Secondary Pressure for SGTR

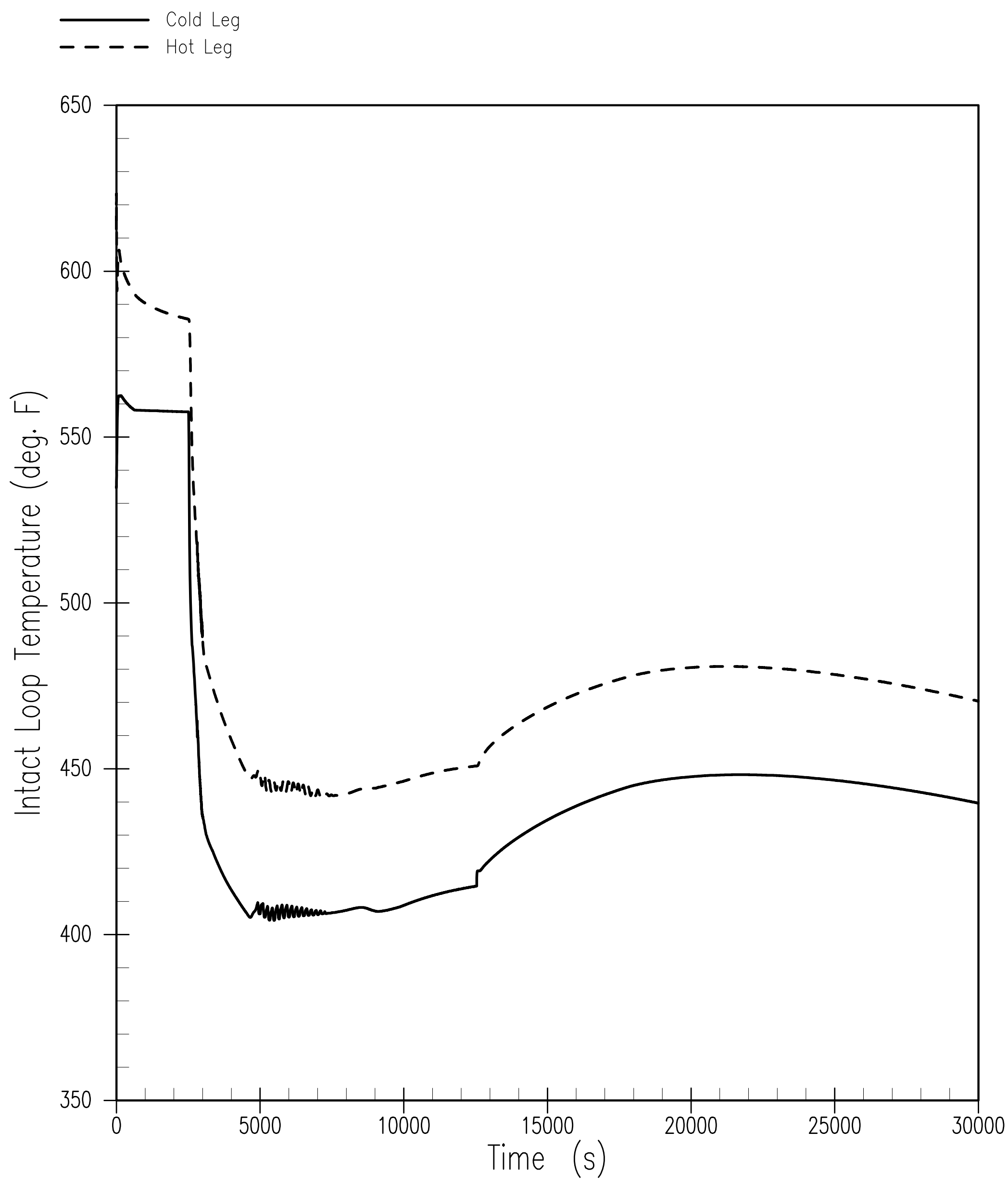


Figure 15.6.3-4
Intact Loop Hot and Cold Leg
Reactor Coolant System Temperature for SGTR

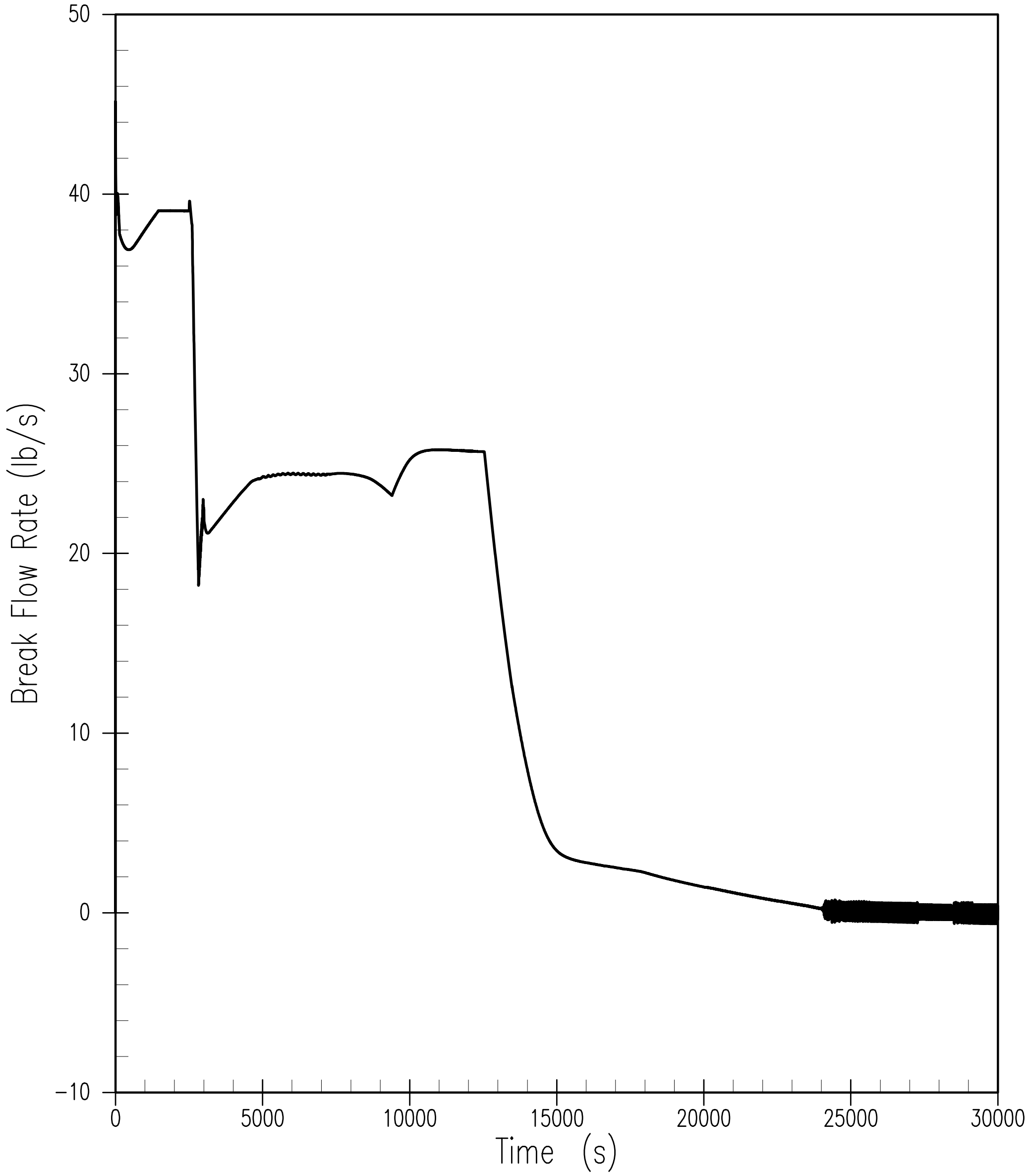


Figure 15.6.3-5
Primary-to-Secondary Break Flow Rate for SGTR

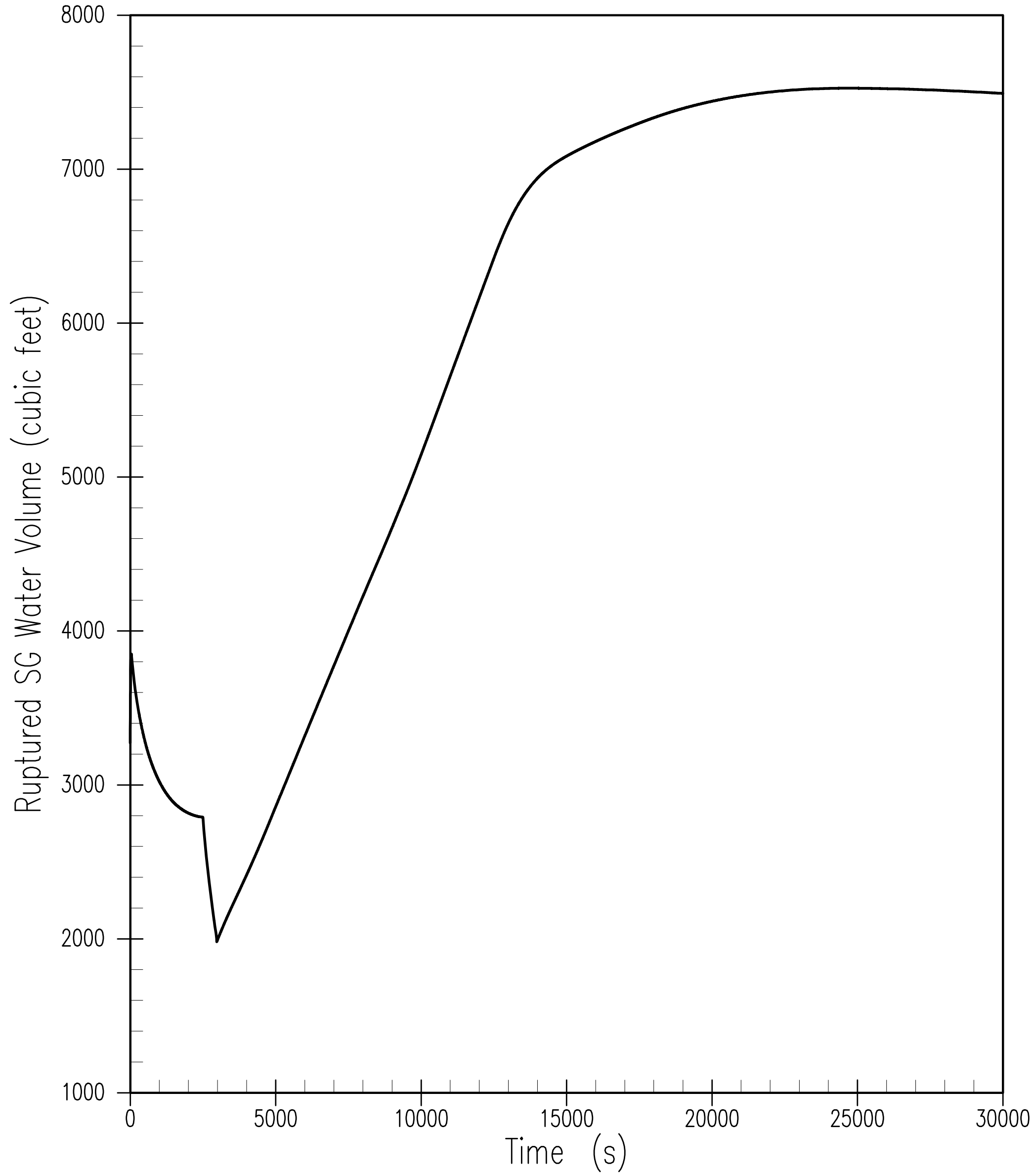


Figure 15.6.3-6
Ruptured Steam Generator Water Volume for SGTR

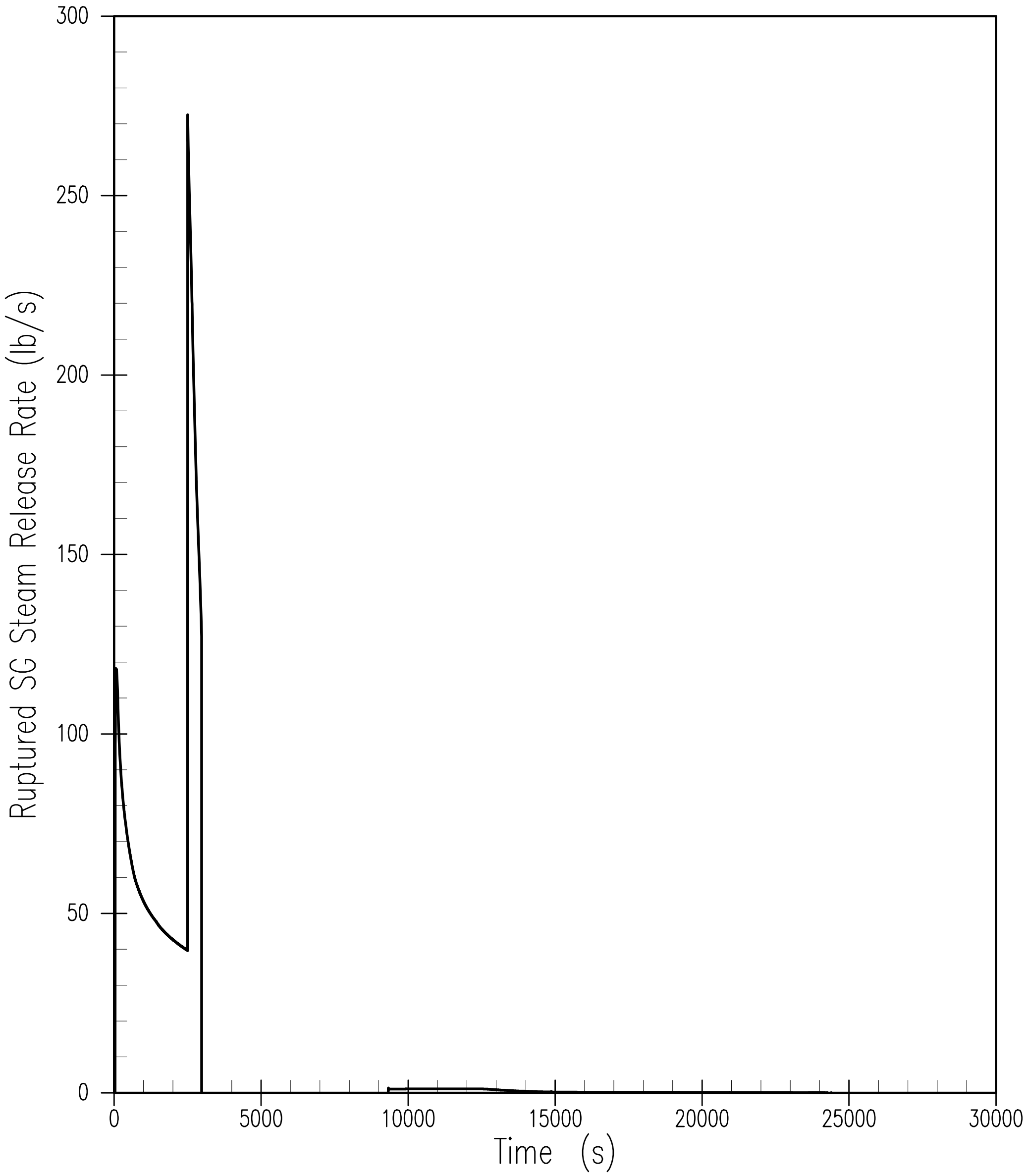


Figure 15.6.3-7
Ruptured Steam Generator Mass
Release Rate to the Atmosphere for SGTR

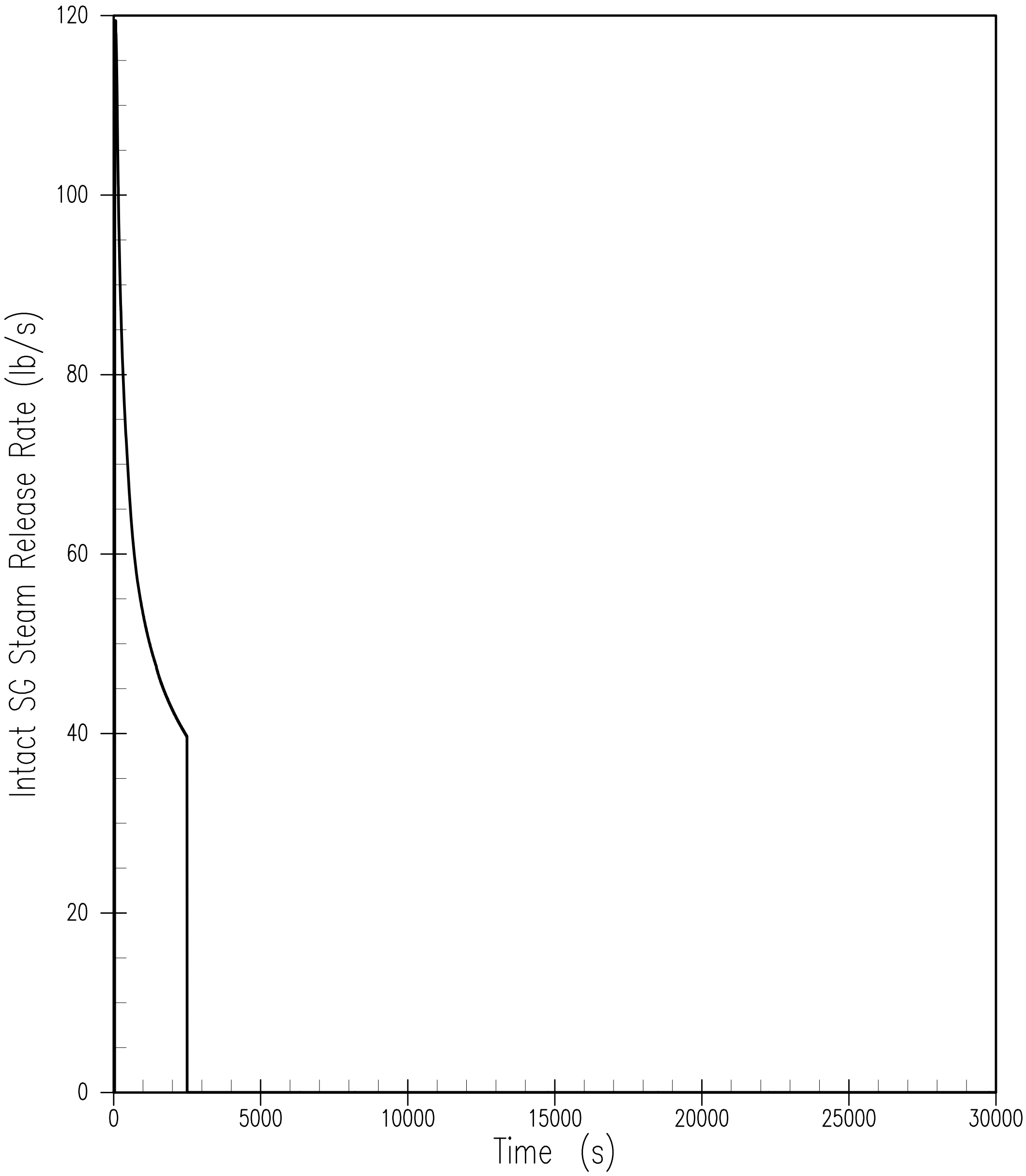


Figure 15.6.3-8
Intact Steam Generator Mass
Release Rate to the Atmosphere for SGTR

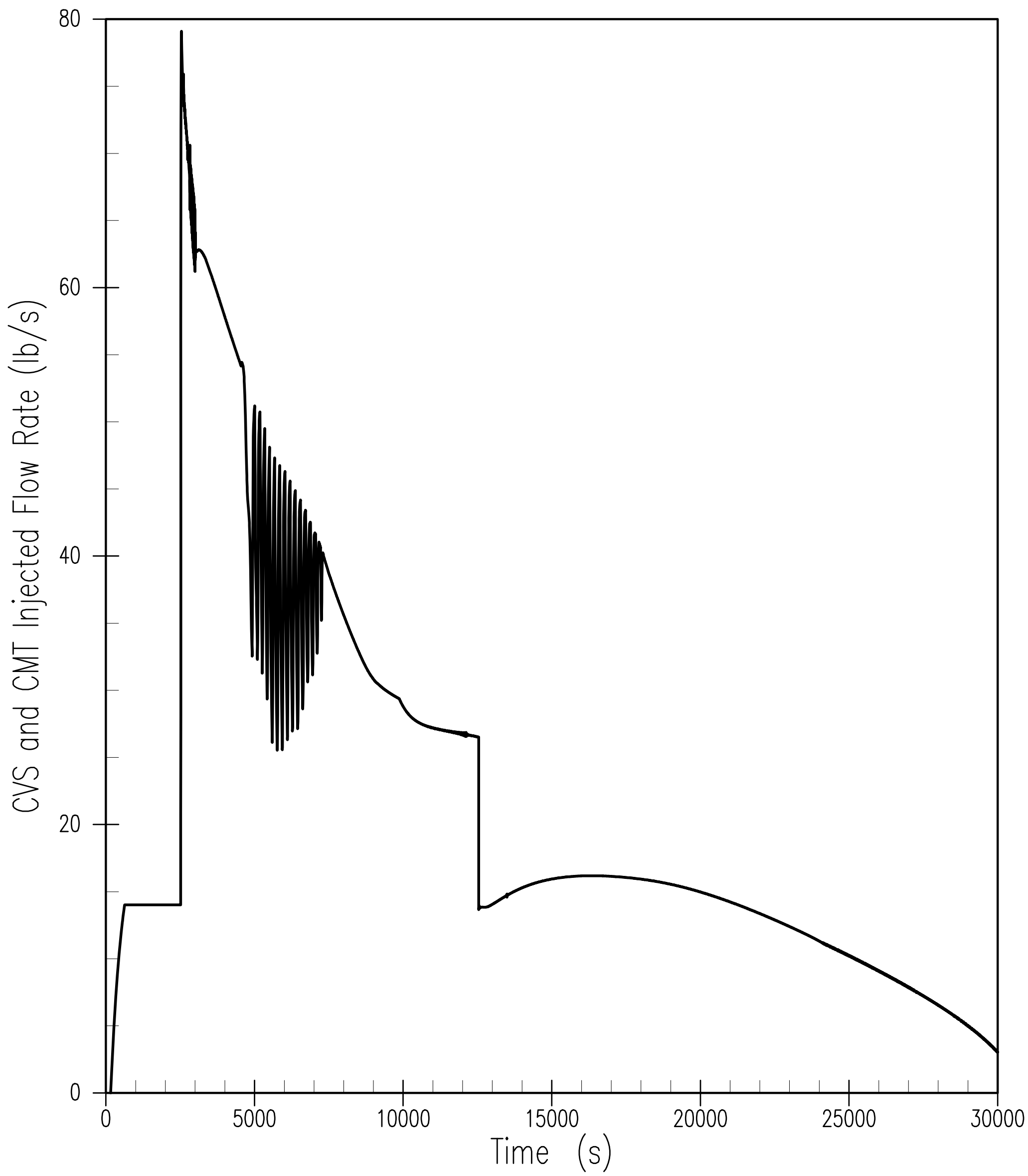


Figure 15.6.3-9
Ruptured Loop Chemical and Volume Control System and Core Makeup Tank Injection Flow for SGTR

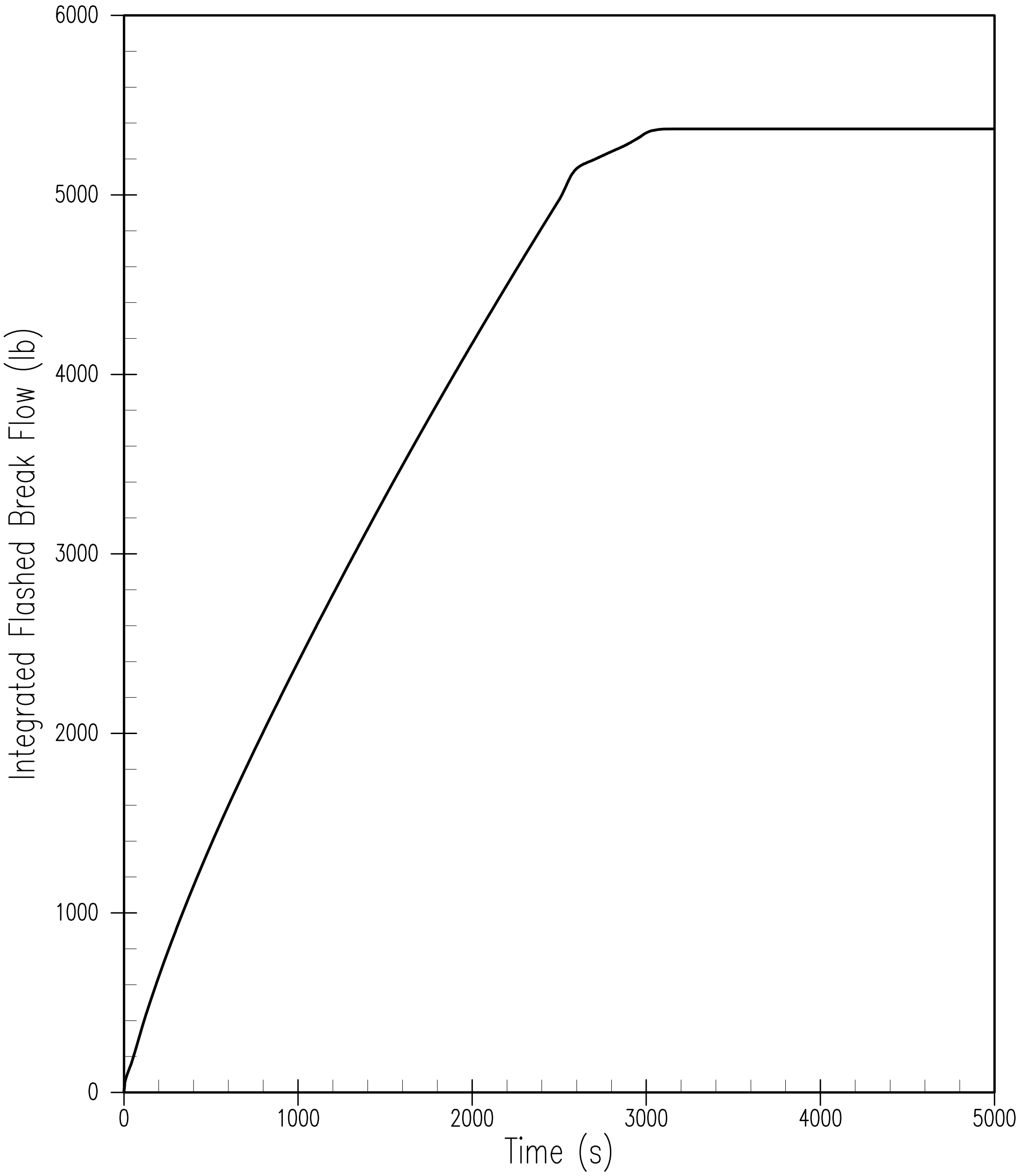


Figure 15.6.3-10
Integrated Flashed Break Flow for SGTR

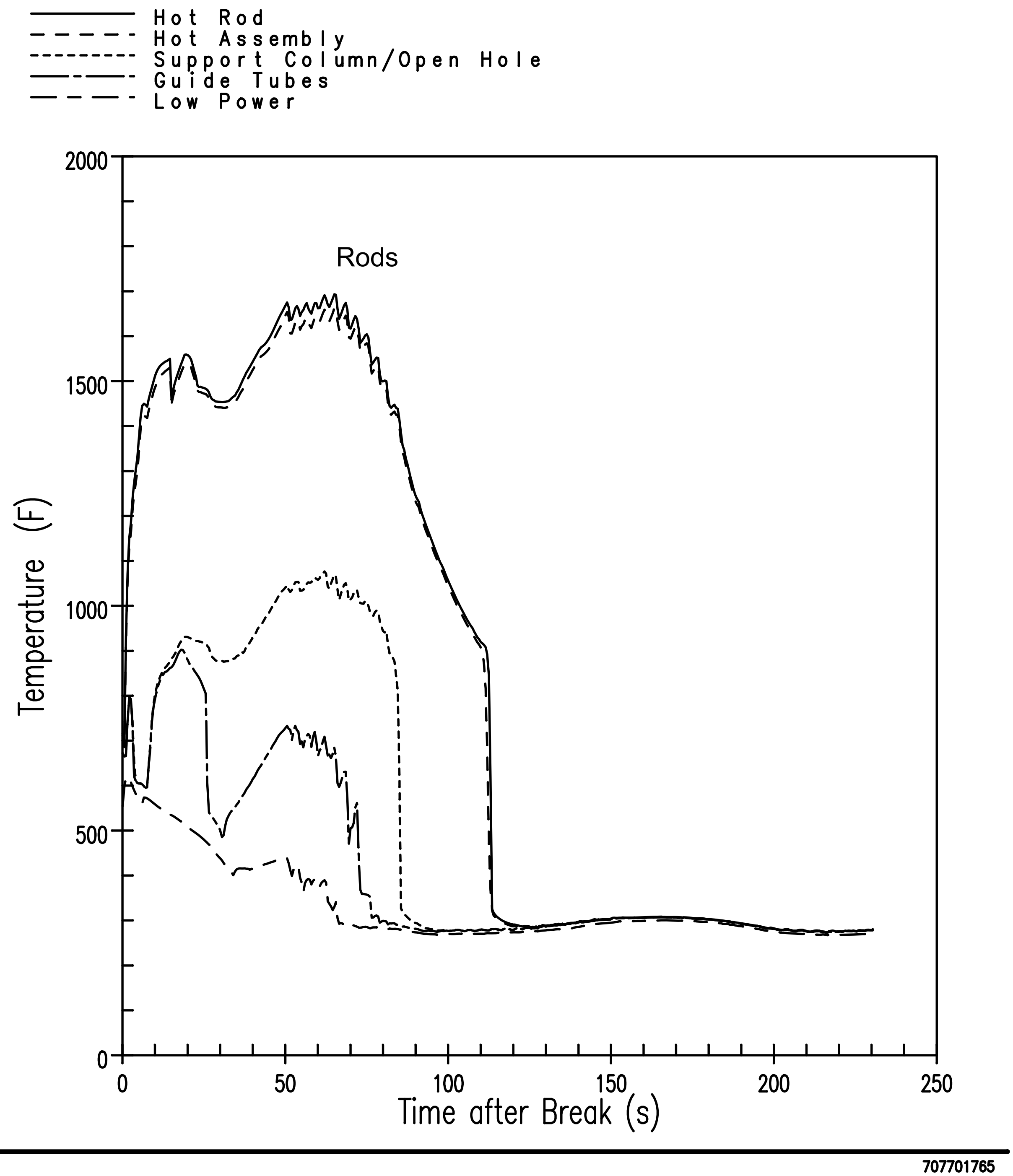


Figure 15.6.5.4A-1
<u>W</u>COBRA/TRAC Peak Cladding Temperature for
All Five Rod Groups for 95th Percentile Estimator PCT/MLO Case

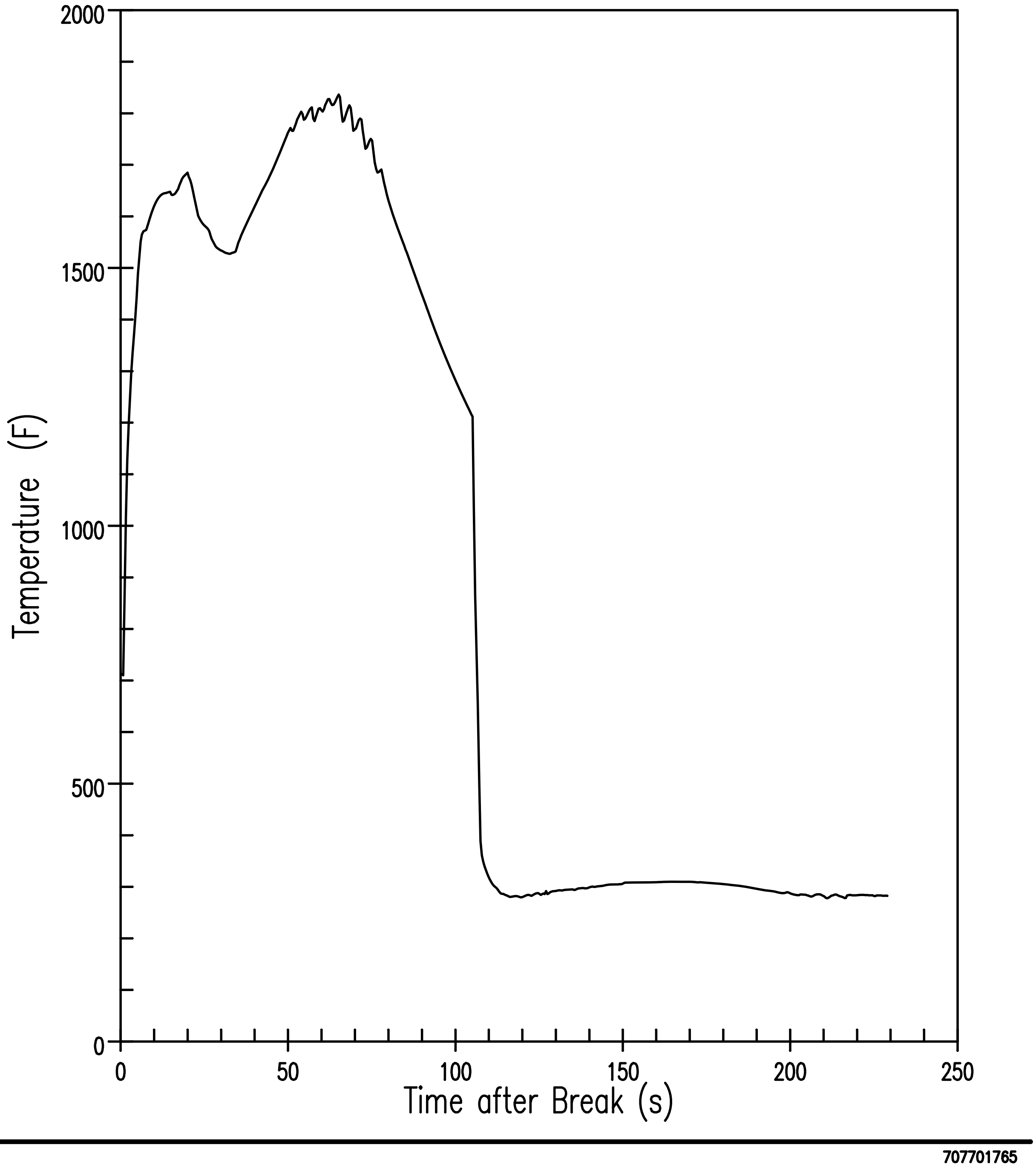


Figure 15.6.5.4A-2
HOTSPOT Cladding Temperature Transient at
Limiting Elevation for 95th Percentile Estimator PCT/MLO Case

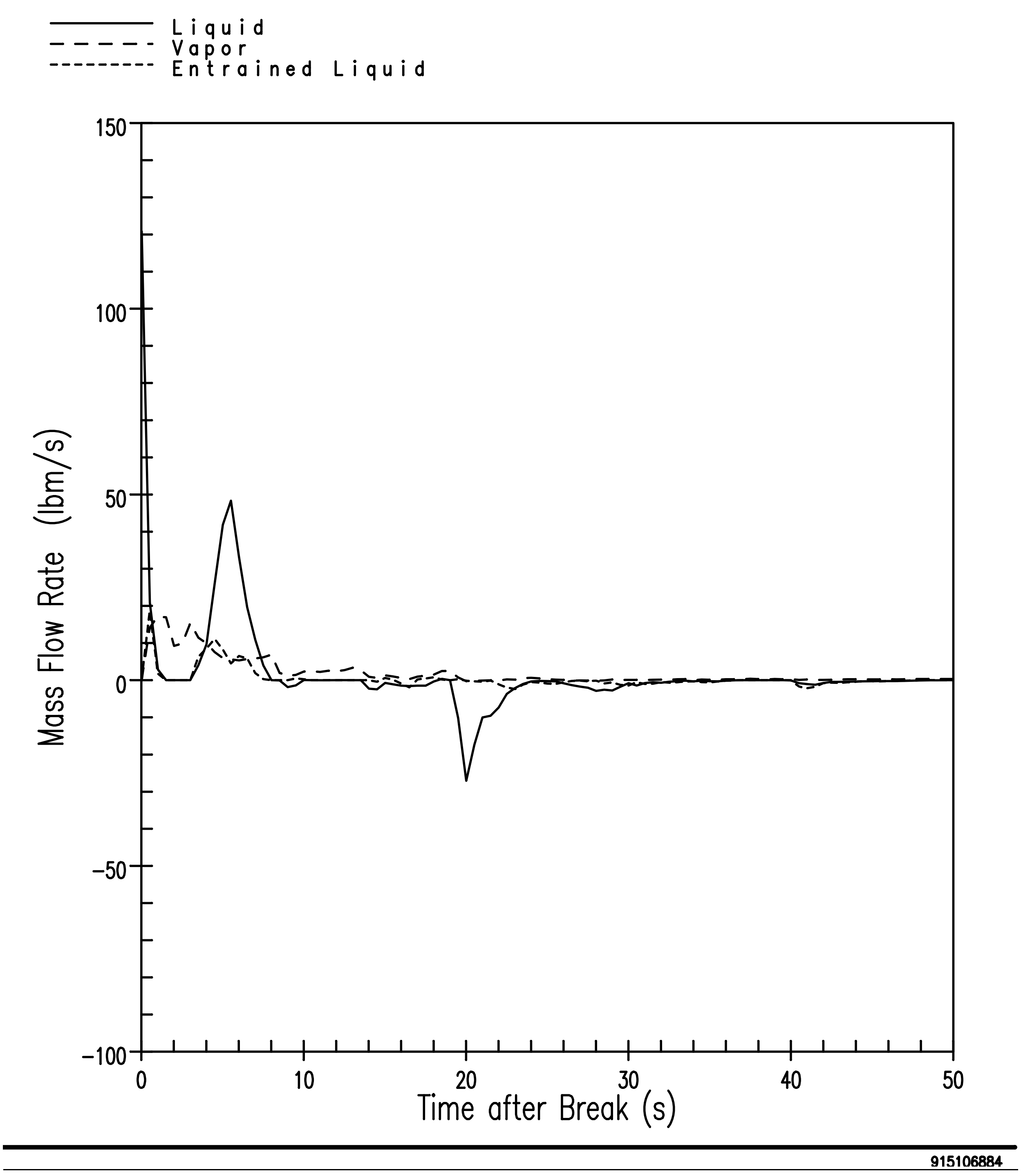
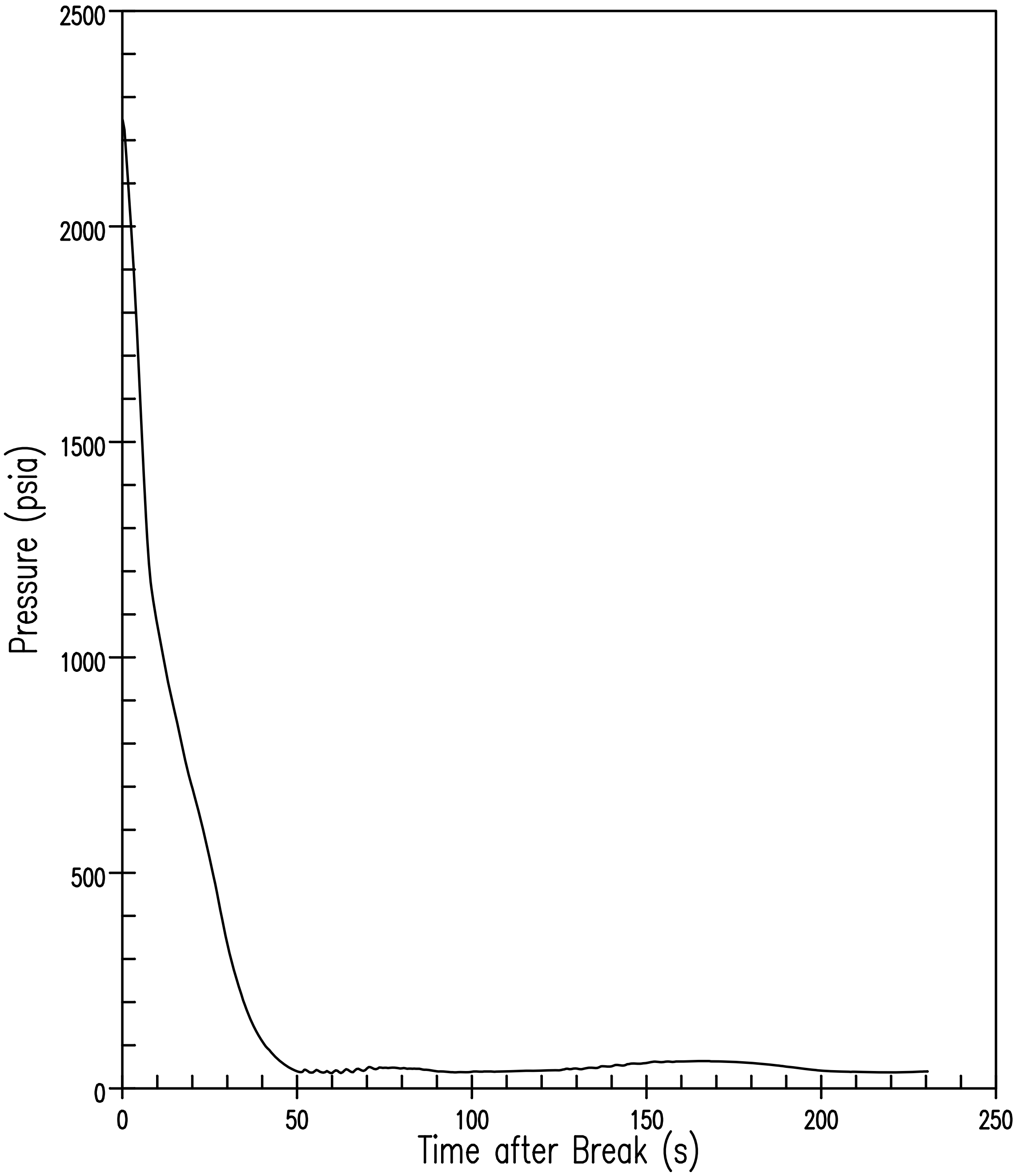


Figure 15.6.5.4A-3
Total Mass Flow at Top of Hot Assembly Channel for 95th Percentile Estimator PCT/MLO Case



707701765

Figure 15.6.5.4A-4
Pressurizer Pressure for
95th Percentile Estimator PCT/MLO Case

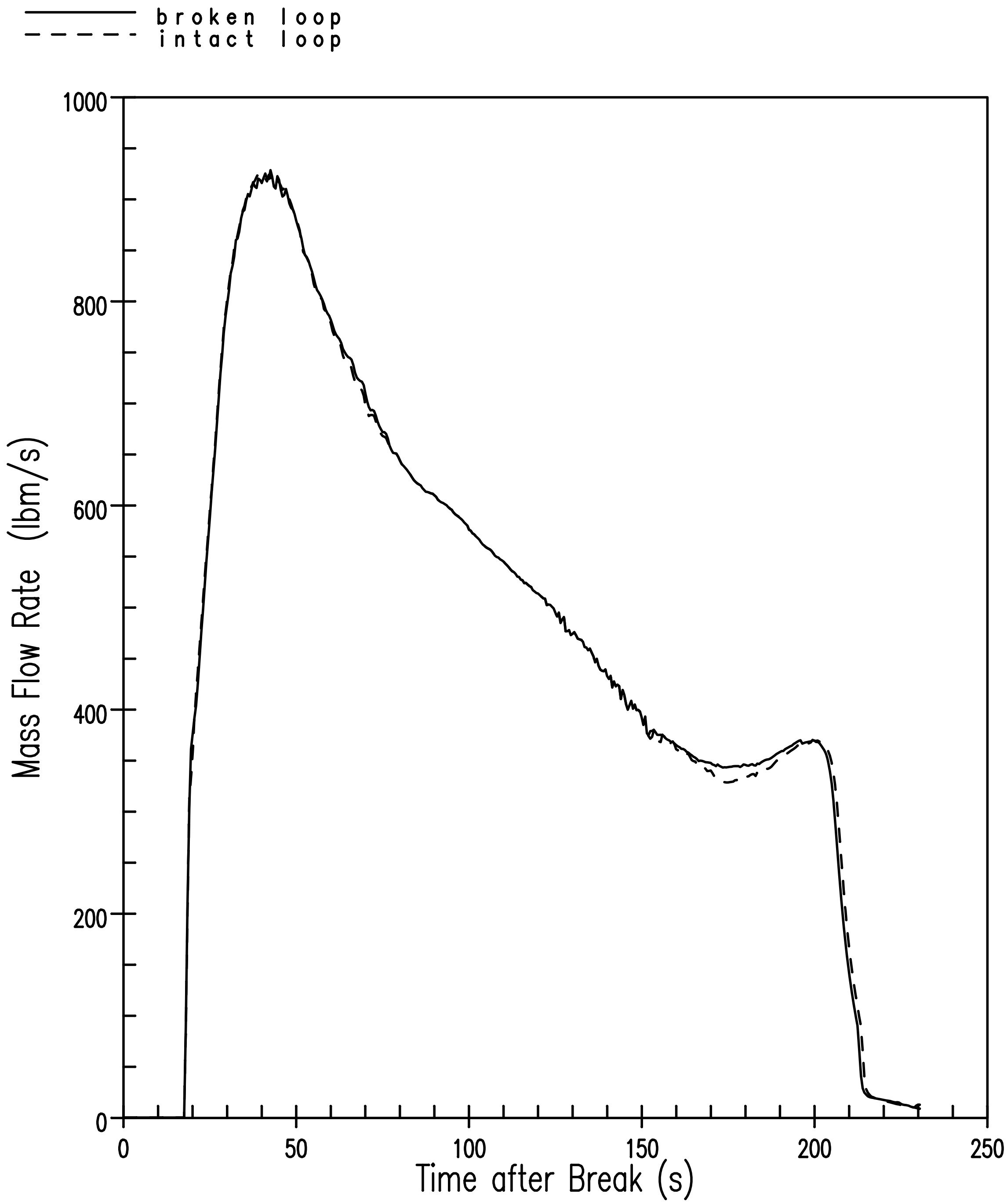


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Figure 15.6.5.4A-5
Accumulator Injection Flow for
95th Percentile Estimator PCT/MLO Case

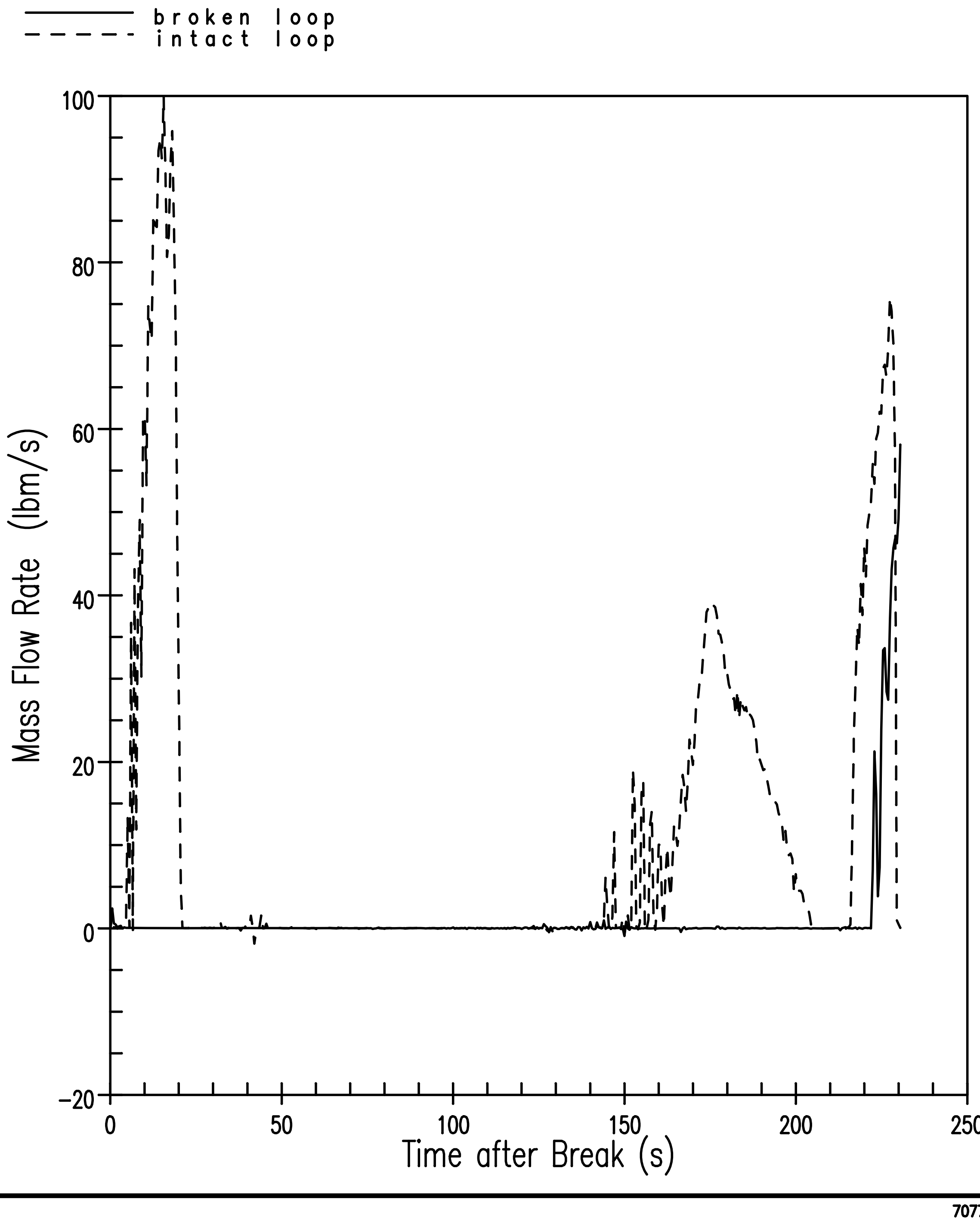

Figure 15.6.5.4A-6
Core Makeup Tank Injection Flow
for 95th Percentile Estimator PCT/MLO Case

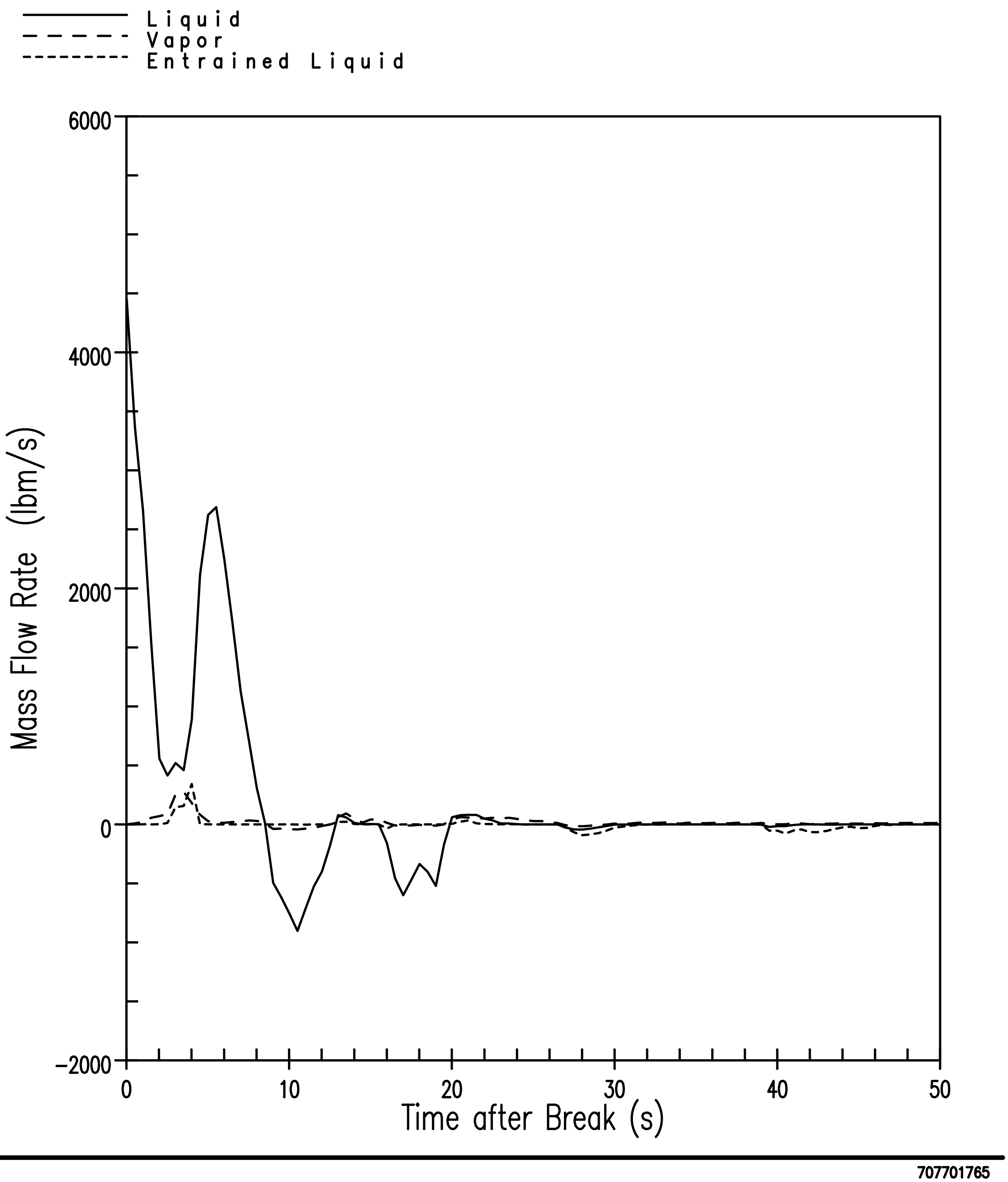

Figure 15.6.5.4A-7
Total Mass Flow at Top of Peripheral Assemblies Channel
for 95th Percentile Estimator PCT/MLO Case

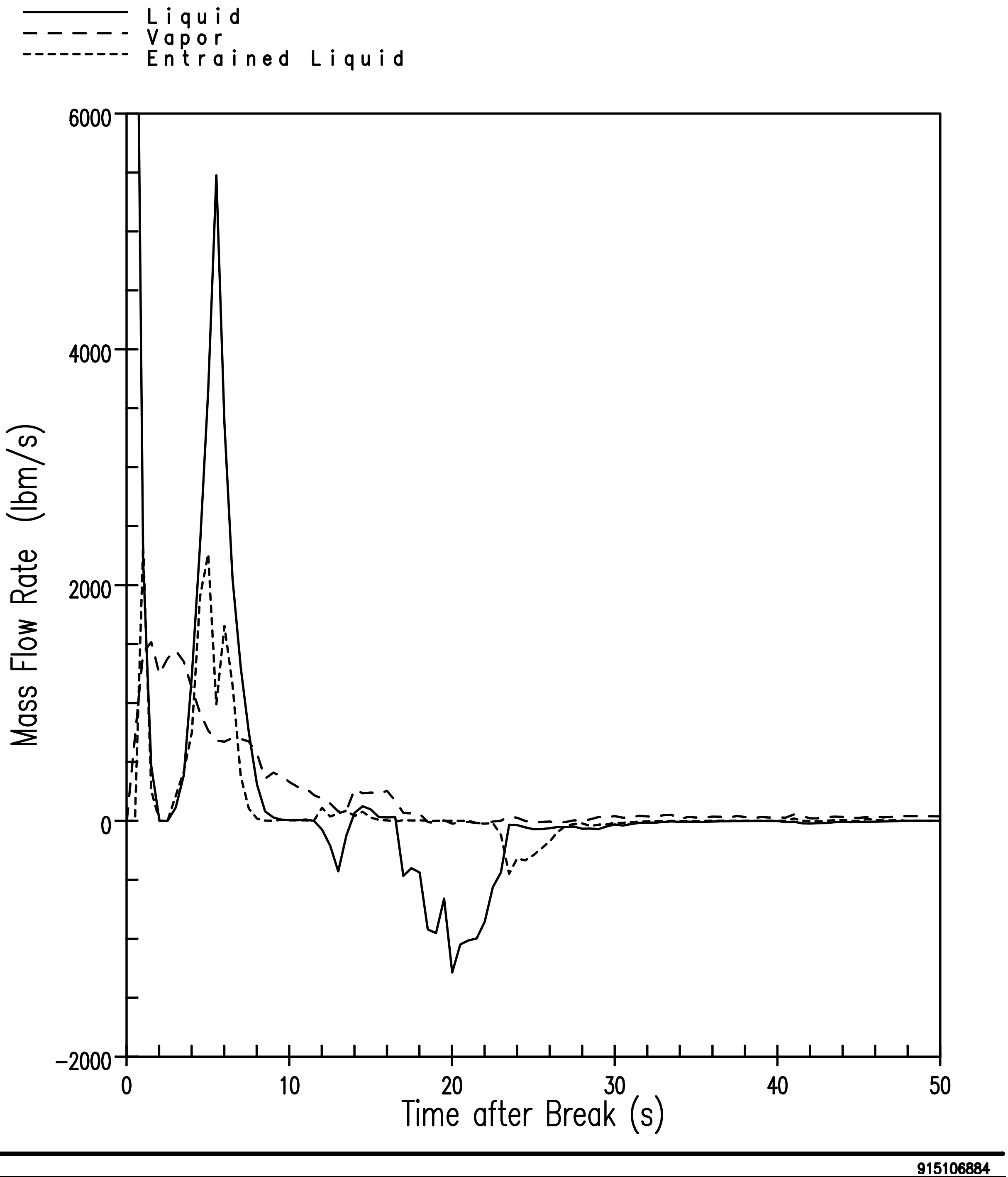

Figure 15.6.5.4A-8
Total Mass Flow at Top of Guide Tube Assemblies Channel for 95th Percentile Estimator PCT/MLO Case

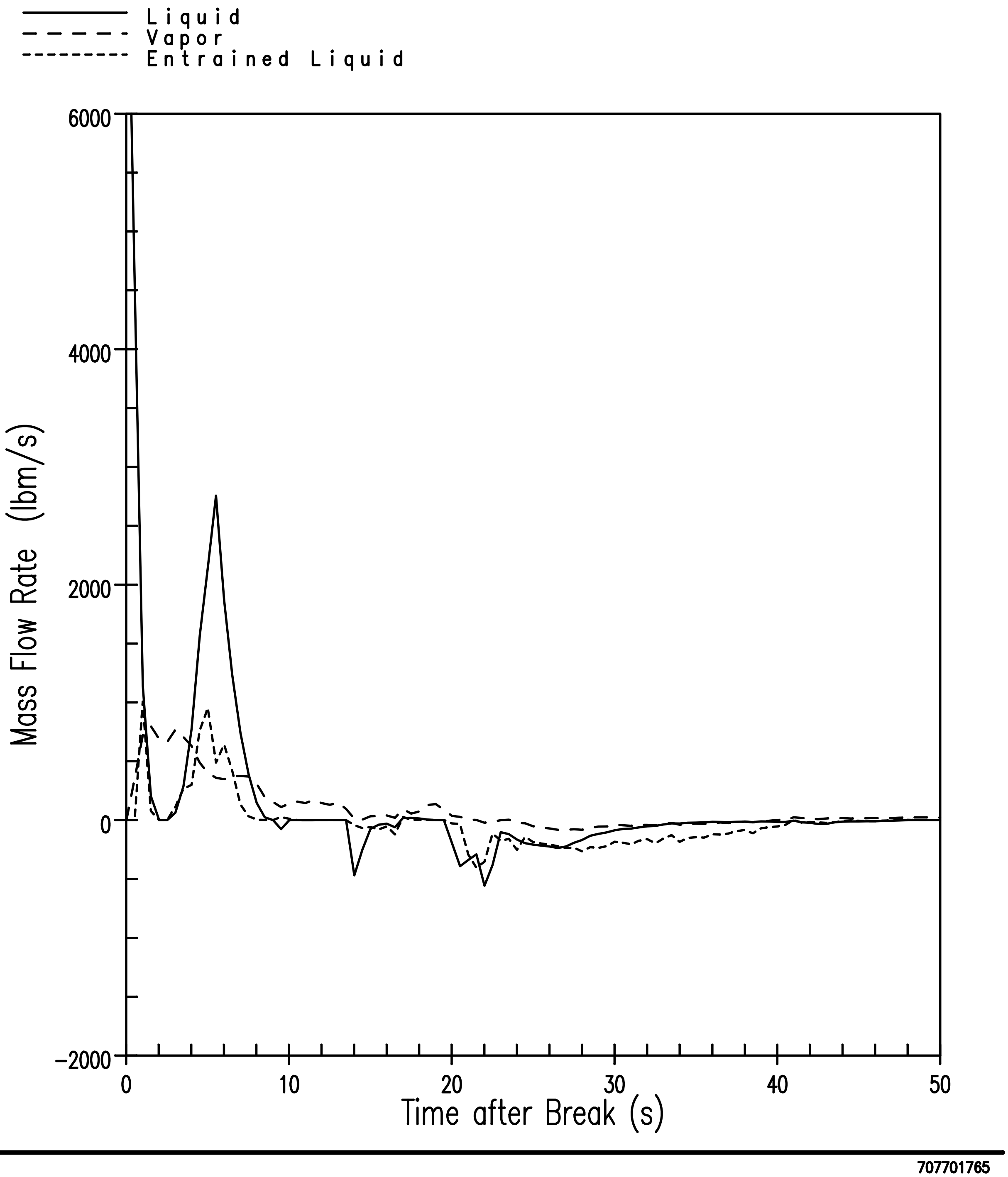

Figure 15.6.5.4A-9
Total Mass Flow at Top of Support Column/Open Hole Assemblies Channel for 95th Percentile Estimator PCT/MLO Case

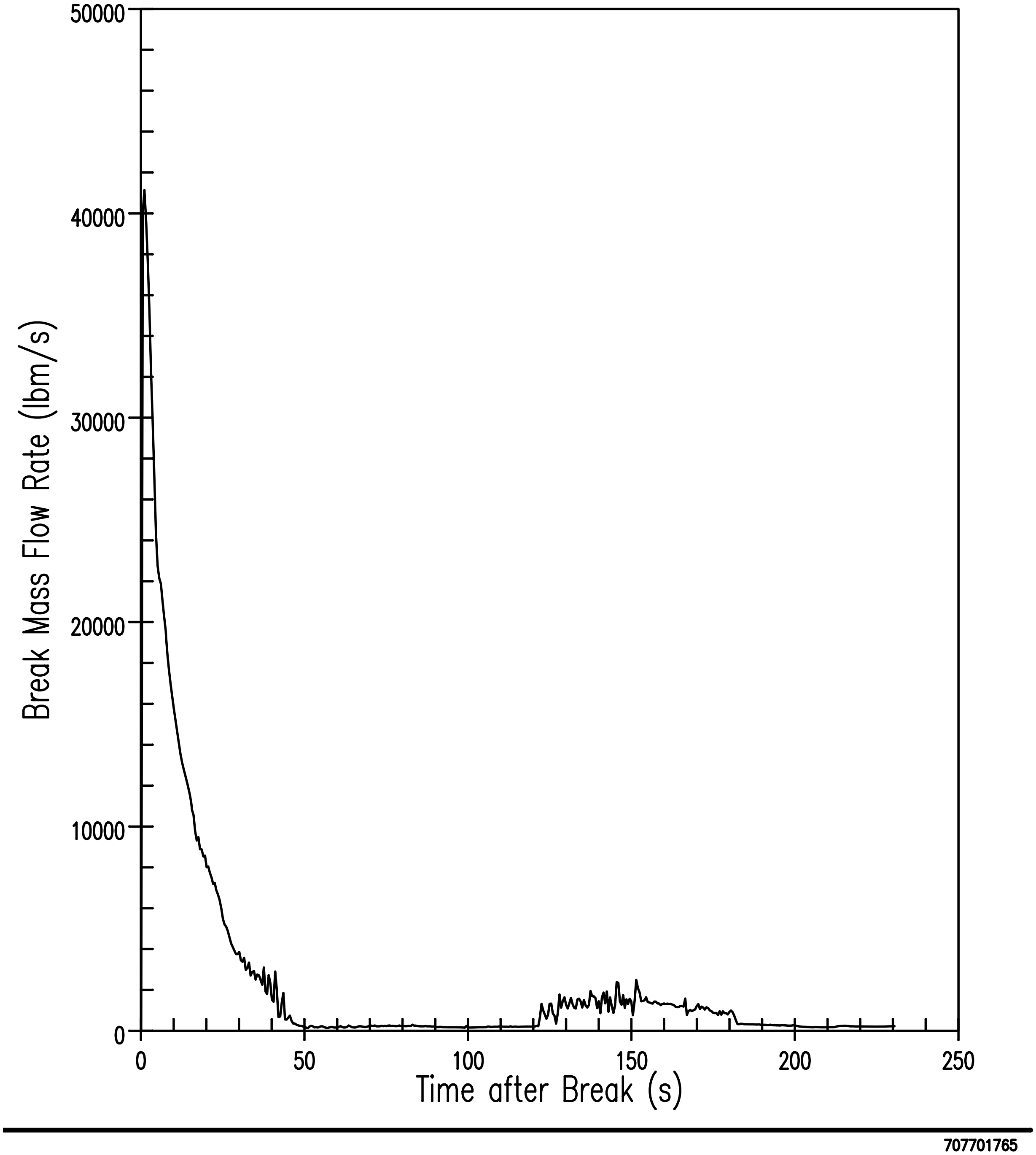

Figure 15.6.5.4A-10
Break Mass Flow for
95th Percentile Estimator PCT/MLO Case

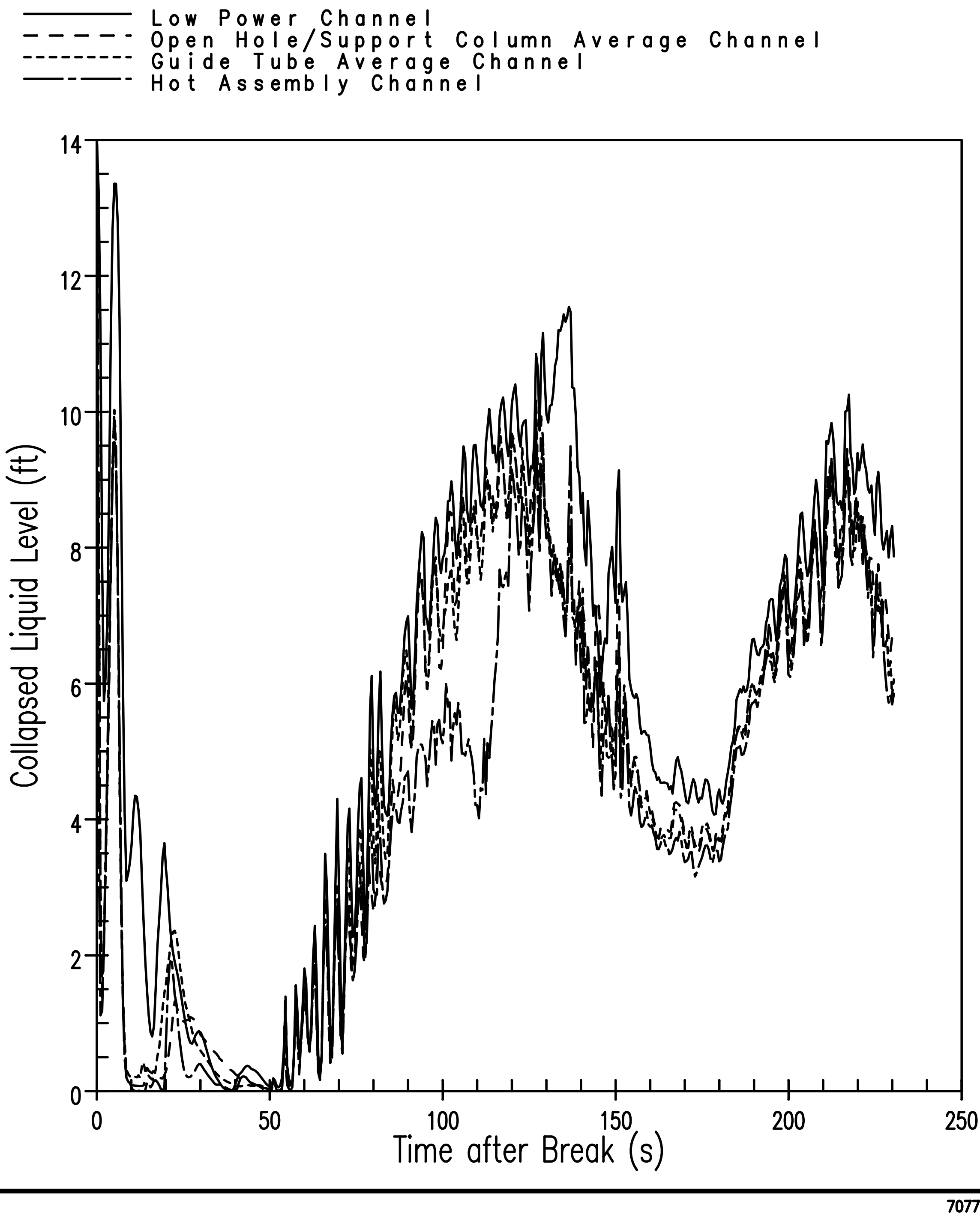

Figure 15.6.5.4A-11
Core Channel Collapsed Liquid Levels
for 95th Percentile Estimator PCT/MLO Case

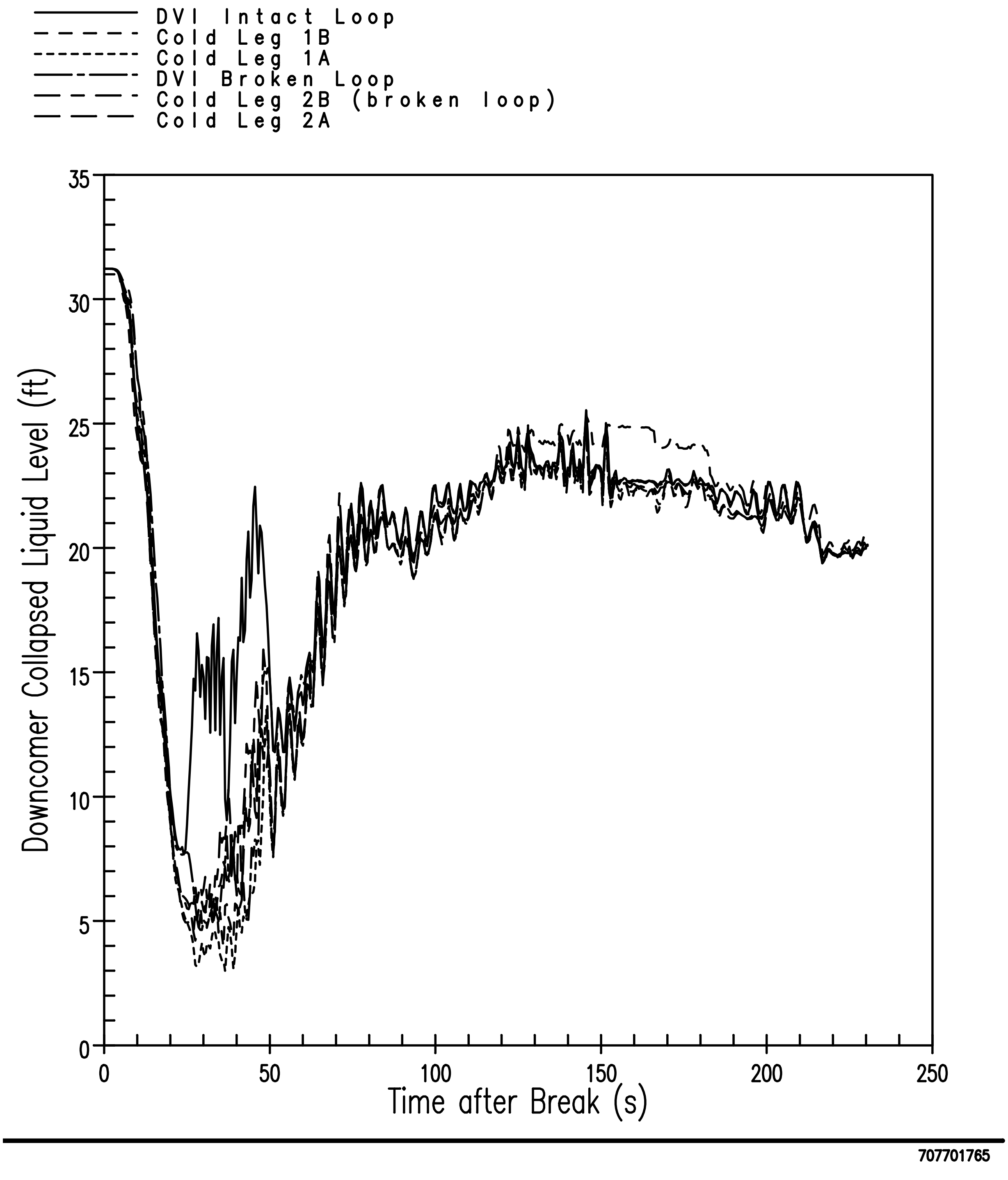

Figure 15.6.5.4A-12
Downcomer Channel Collapsed Liquid Levels
for 95th Percentile Estimator PCT/MLO Case

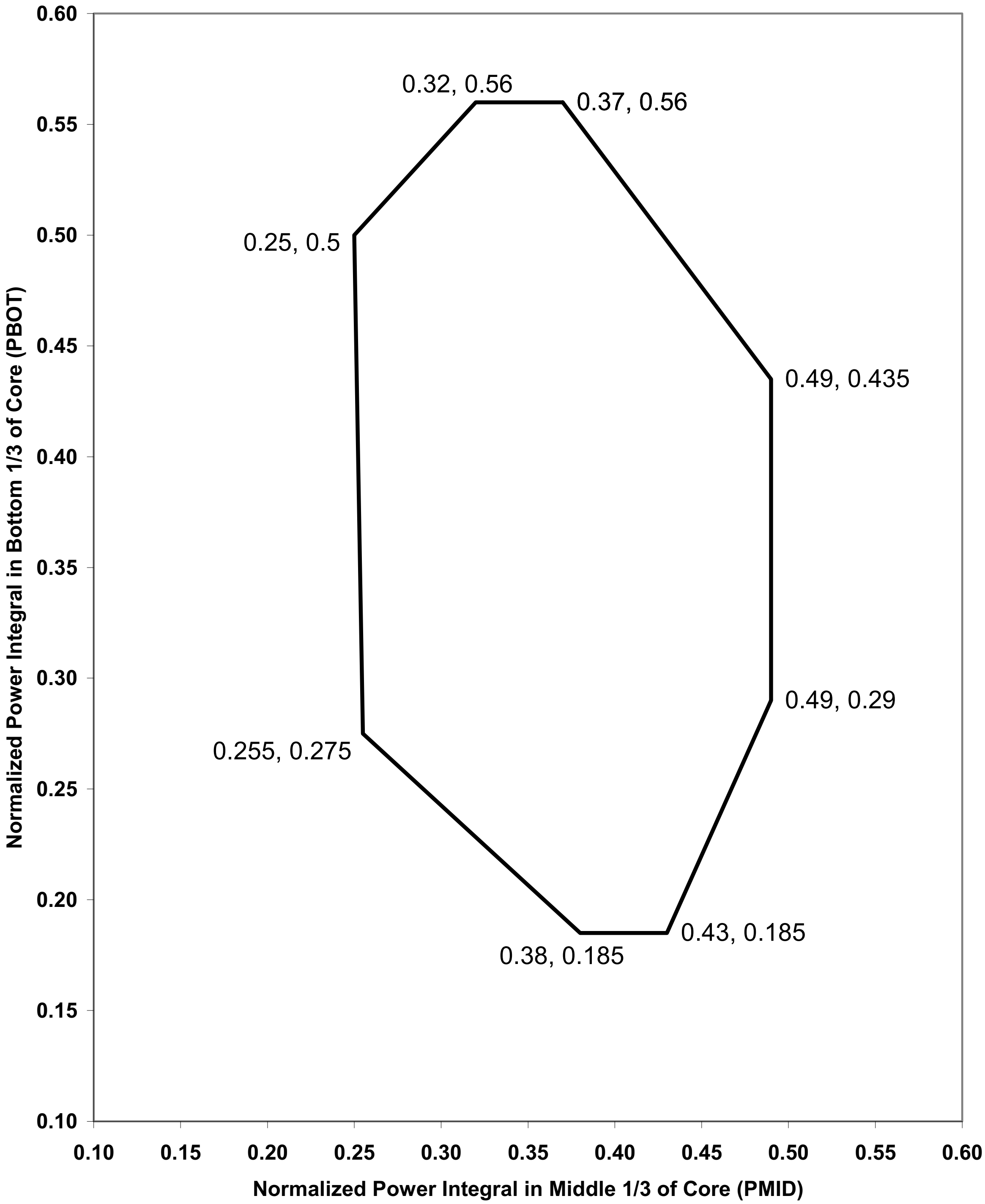

Figure 15.6.5.4A-13
PBOT/PMID Box Supported by
AP1000 ASTRUM Analysis

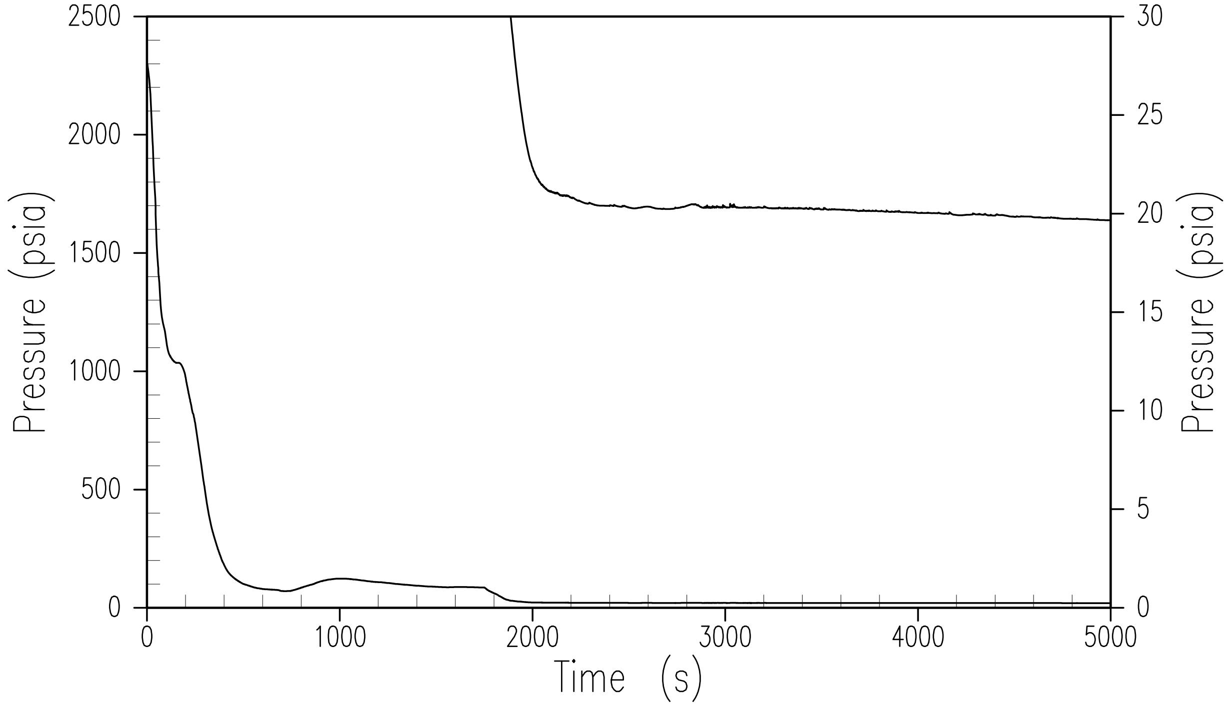

Figure 15.6.5.4B-1
Inadvertent ADS – RCS Pressure

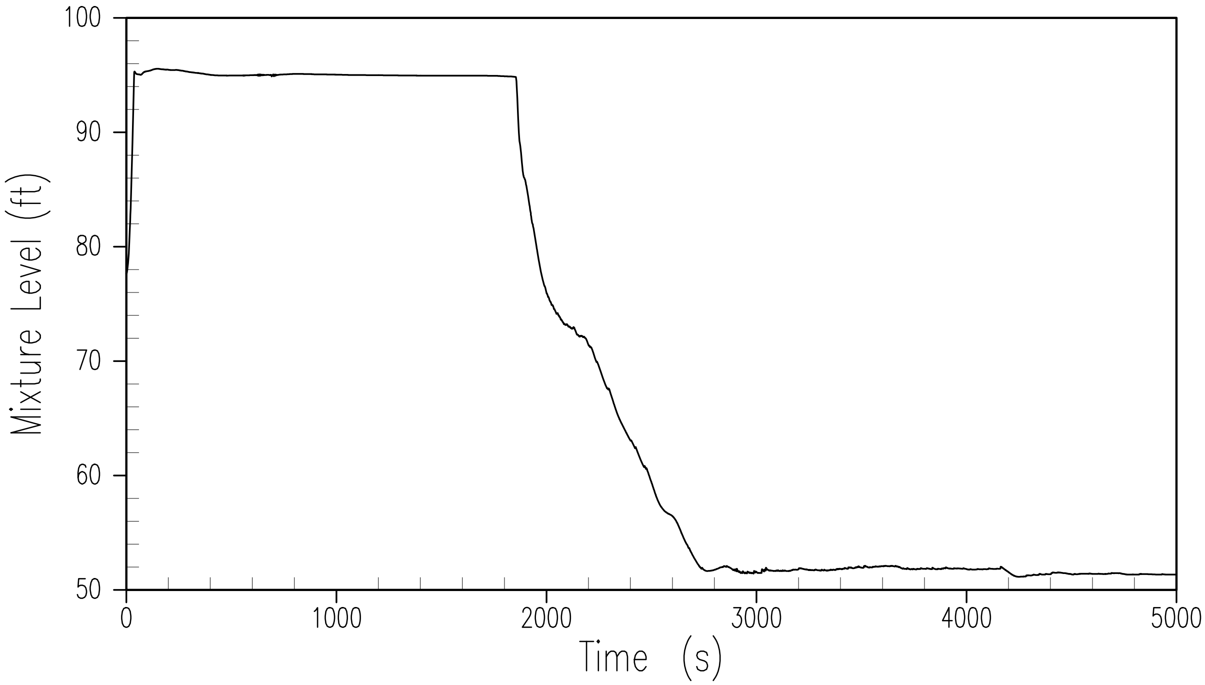

Figure 15.6.5.4B-2
Inadvertent ADS – Pressurizer Mixture Level

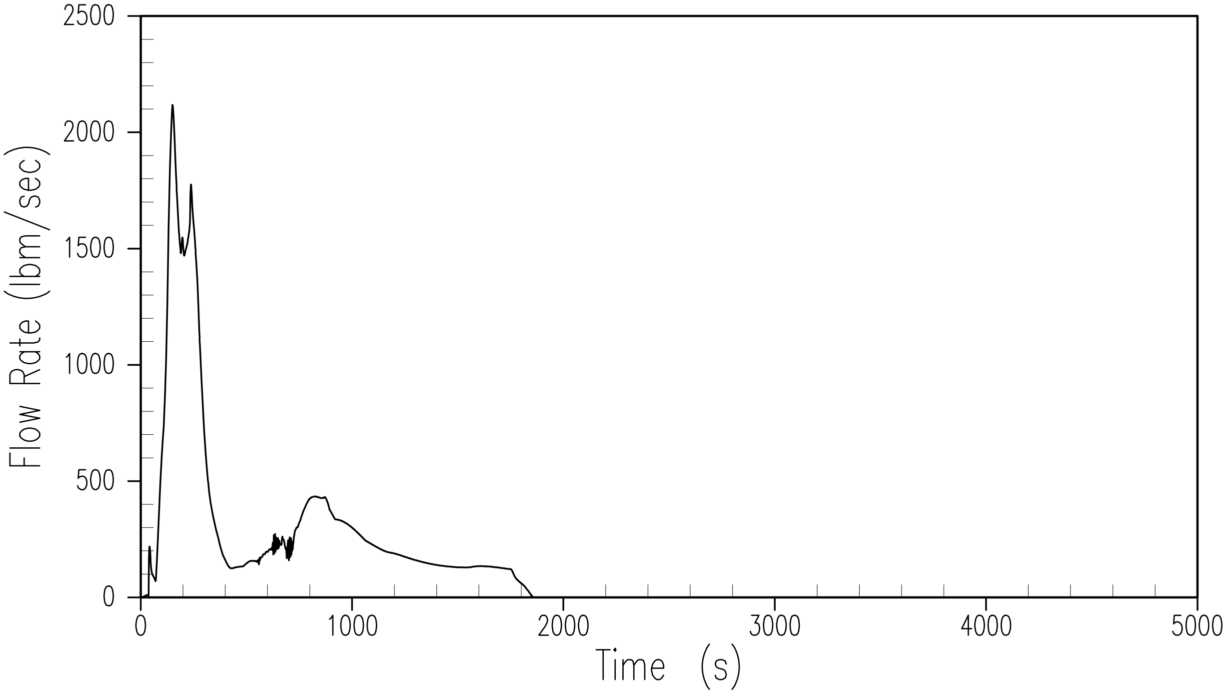

Figure 15.6.5.4B-3
Inadvertent ADS – ADS 1-3 Liquid Discharge

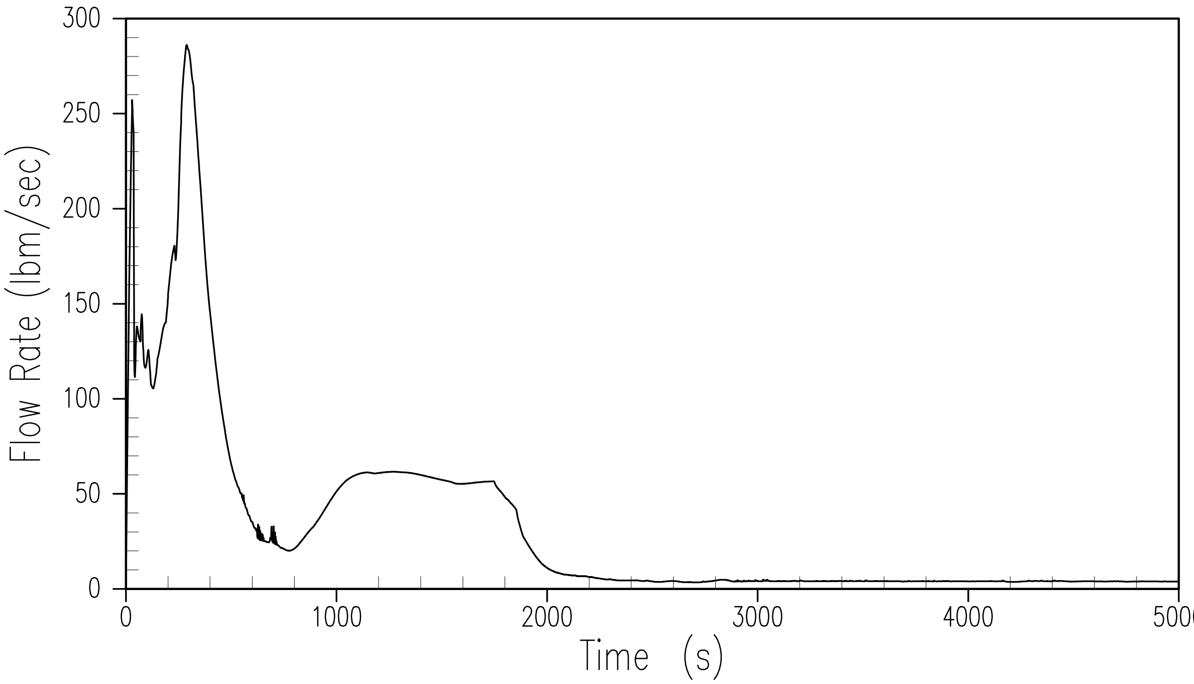

Figure 15.6.5.4B-4
Inadvertent ADS – ADS 1-3 Vapor Discharge

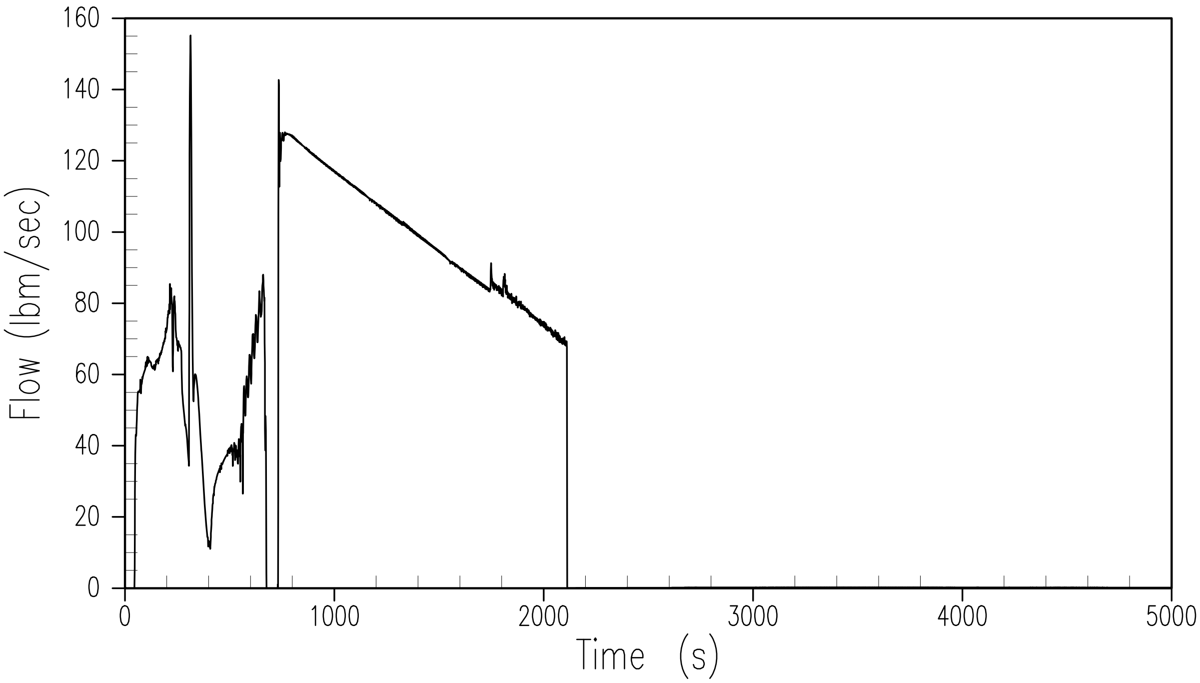

Figure 15.6.5.4B-5
Inadvertent ADS – CMT-1 Injection Rate

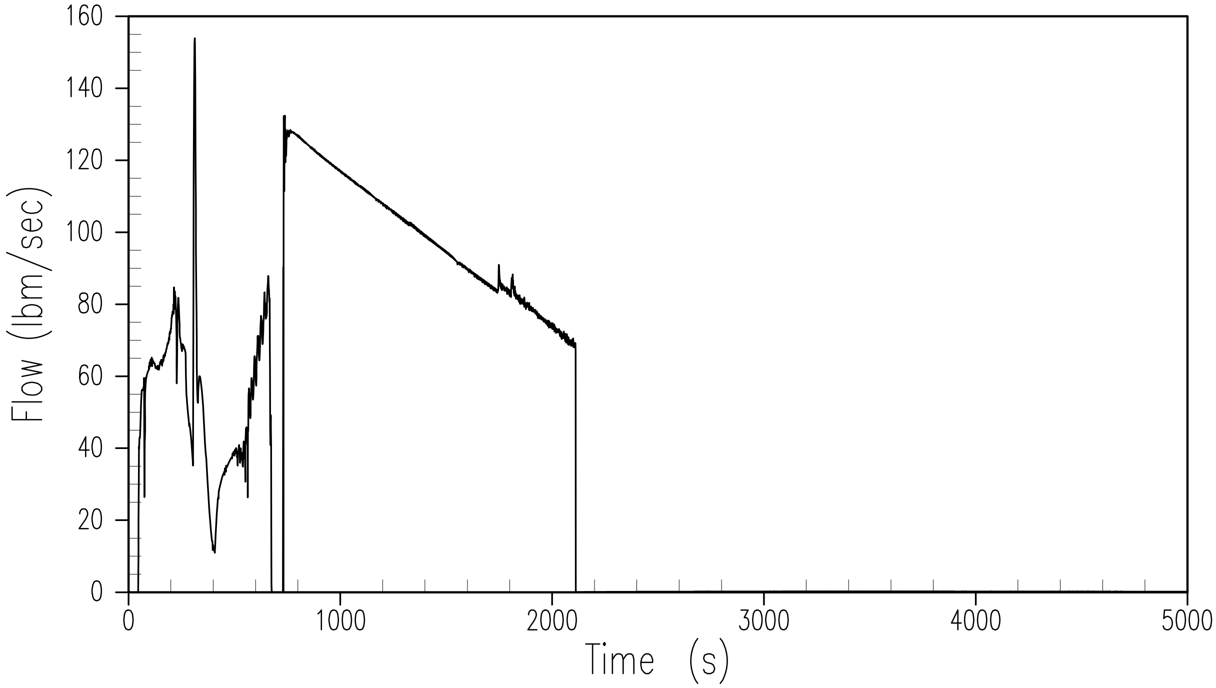

Figure 15.6.5.4B-6
Inadvertent ADS – CMT-2 Injection Rate

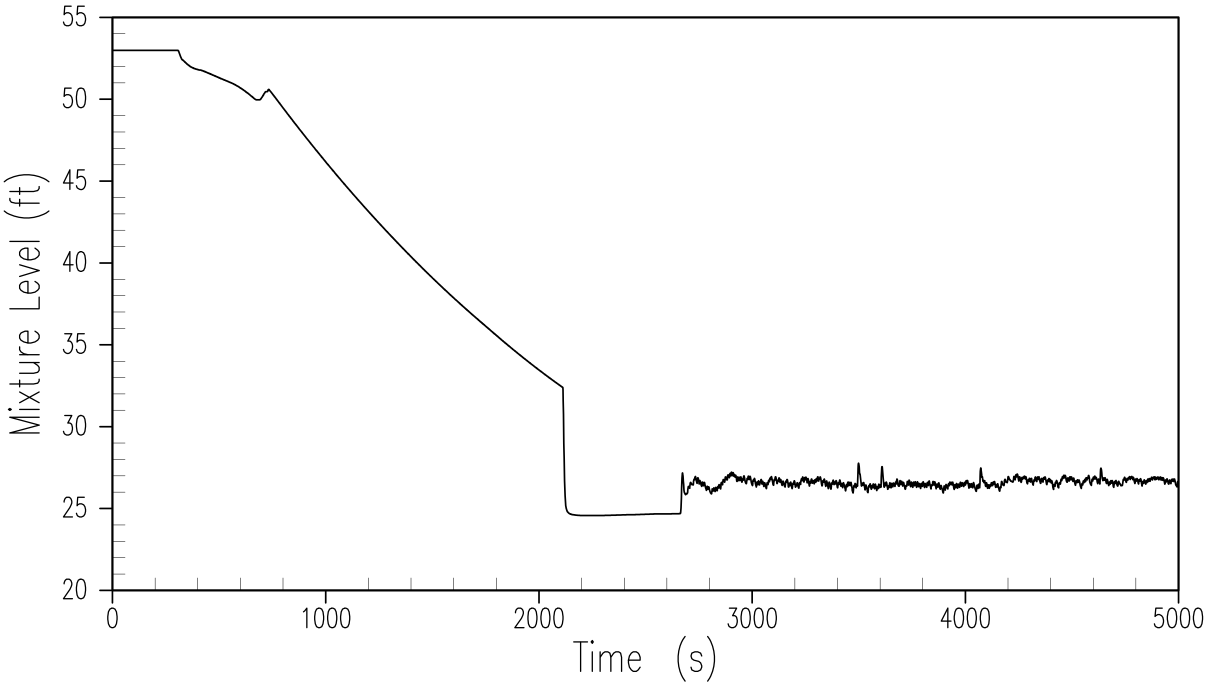

Figure 15.6.5.4B-7
Inadvertent ADS – CMT-1 Mixture Level

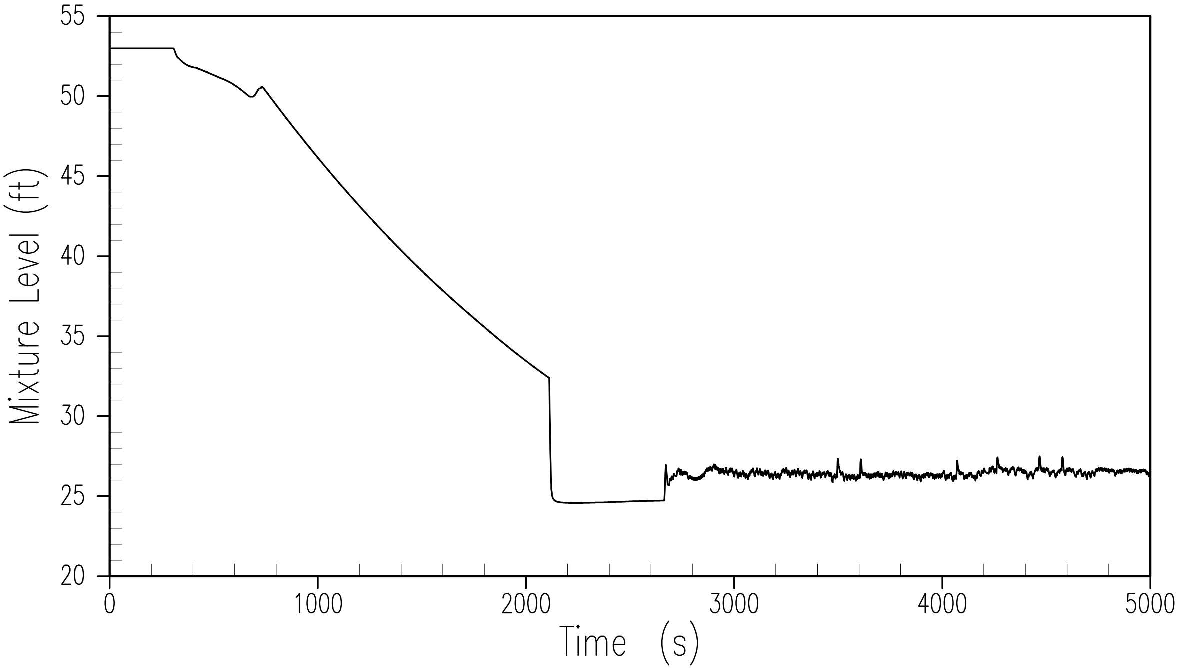

Figure 15.6.5.4B-8
Inadvertent ADS – CMT-2 Mixture Level

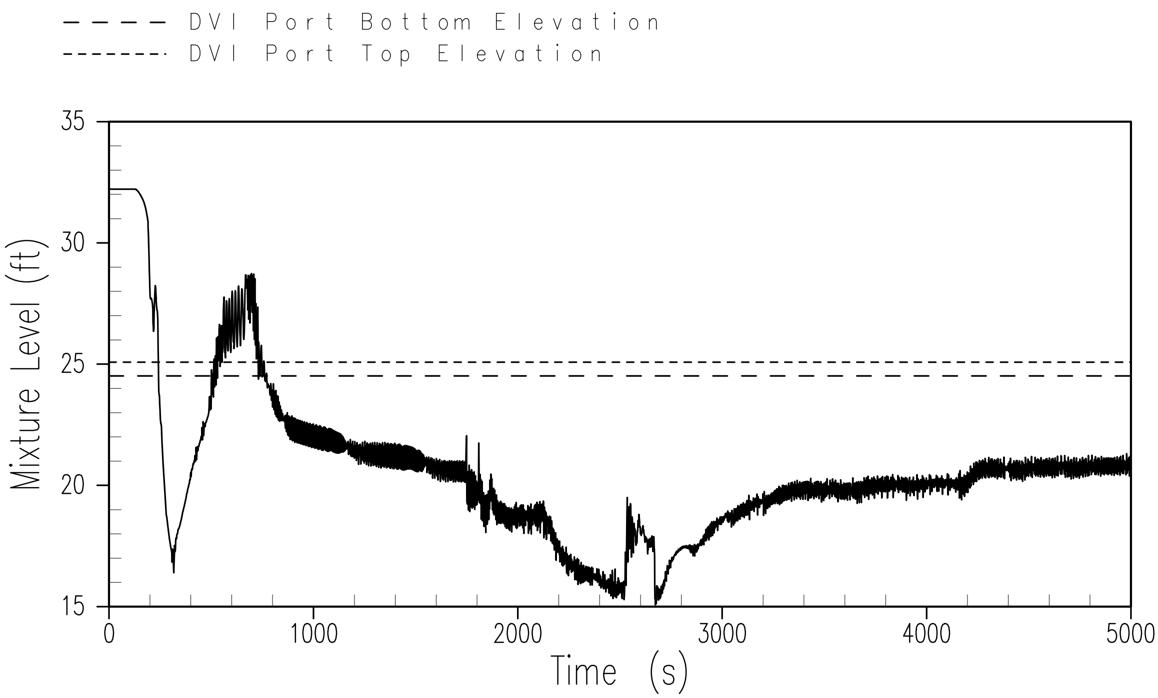

Figure 15.6.5.4B-9
Inadvertent ADS – Downcomer Mixture Level

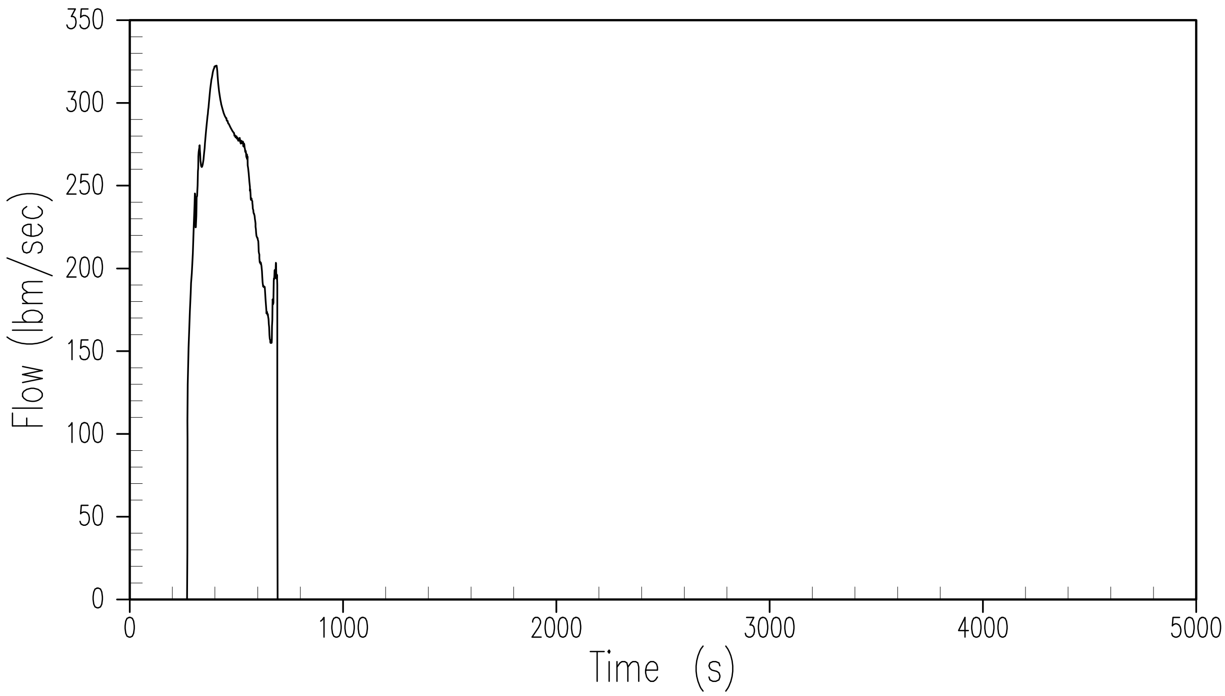

Figure 15.6.5.4B-10
Inadvertent ADS – Accumulator-1 Injection Rate

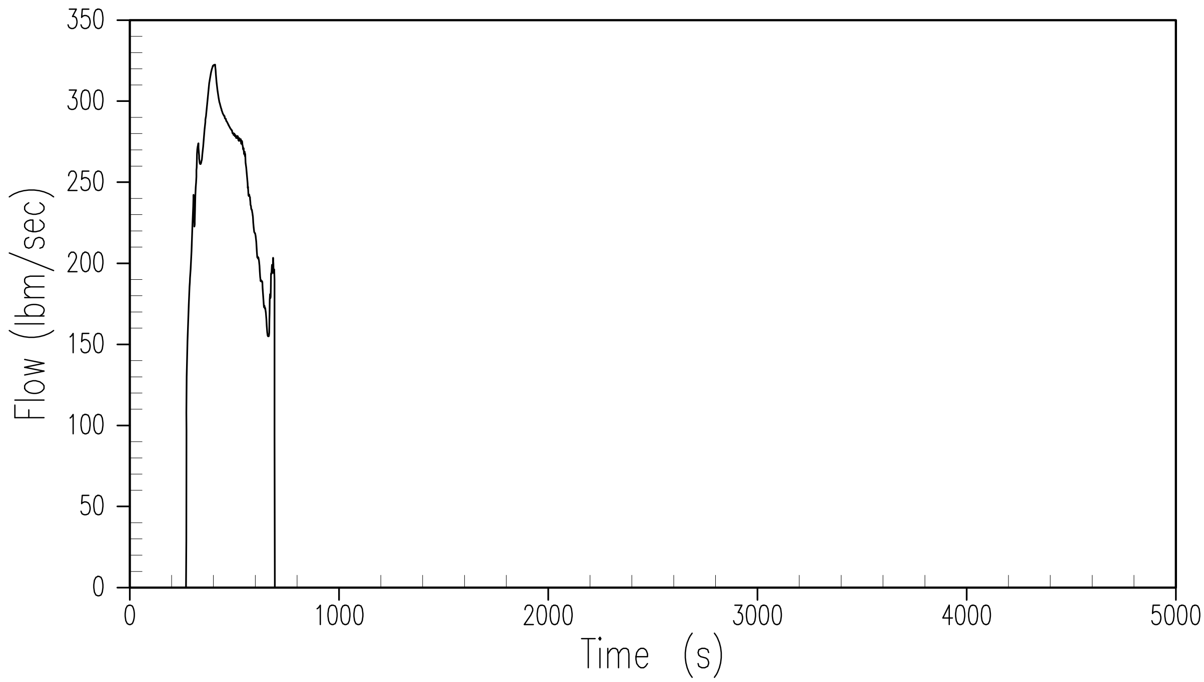

Figure 15.6.5.4B-11
Inadvertent ADS – Accumulator-2 Injection Rate

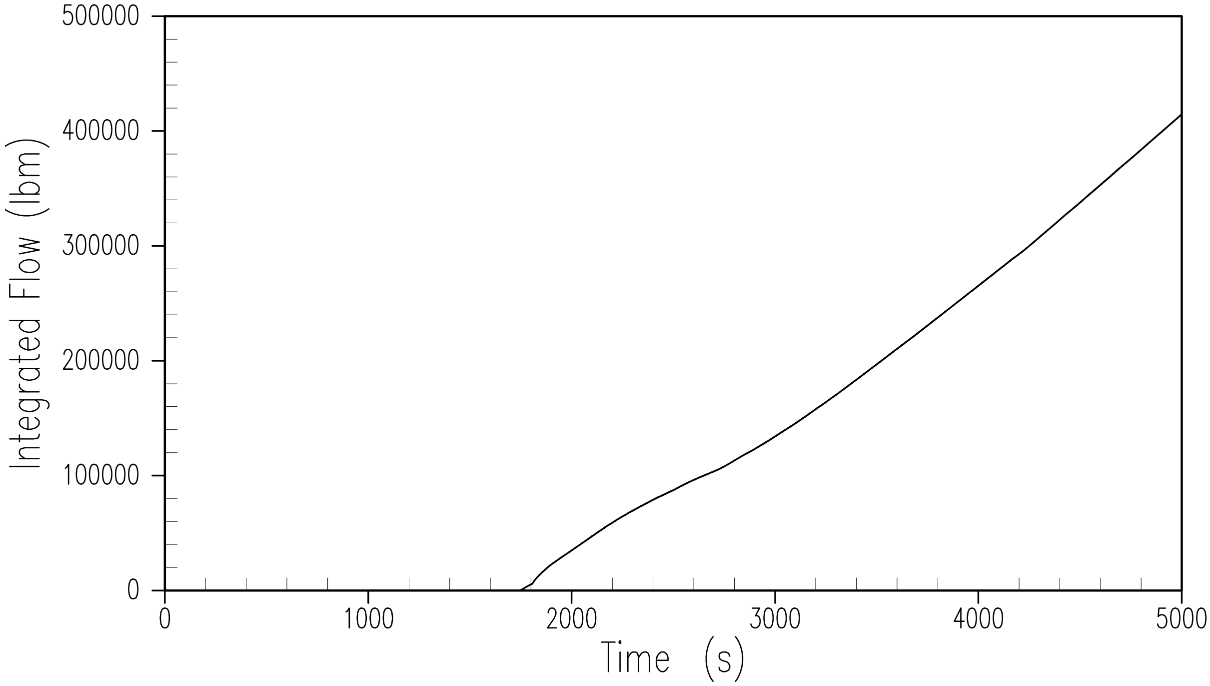

Figure 15.6.5.4B-12
Inadvertent ADS – ADS-4 Integrated Discharge

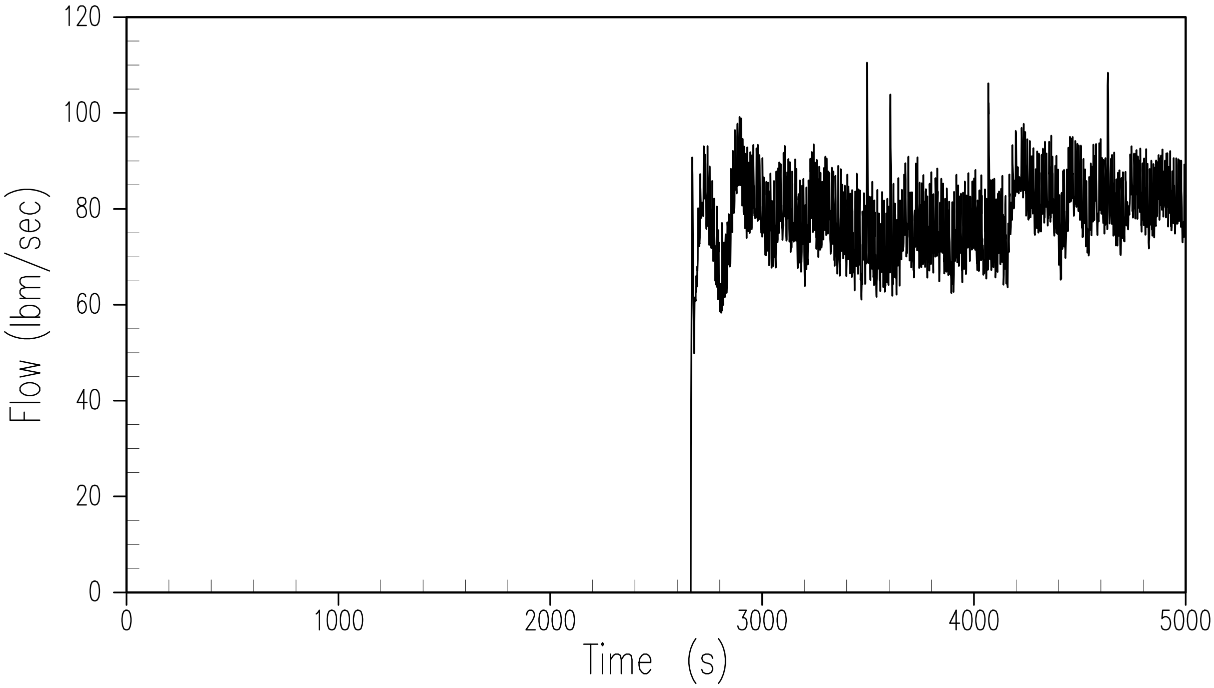

Figure 15.6.5.4B-13
Inadvertent ADS – IRWST-1 Injection Rate

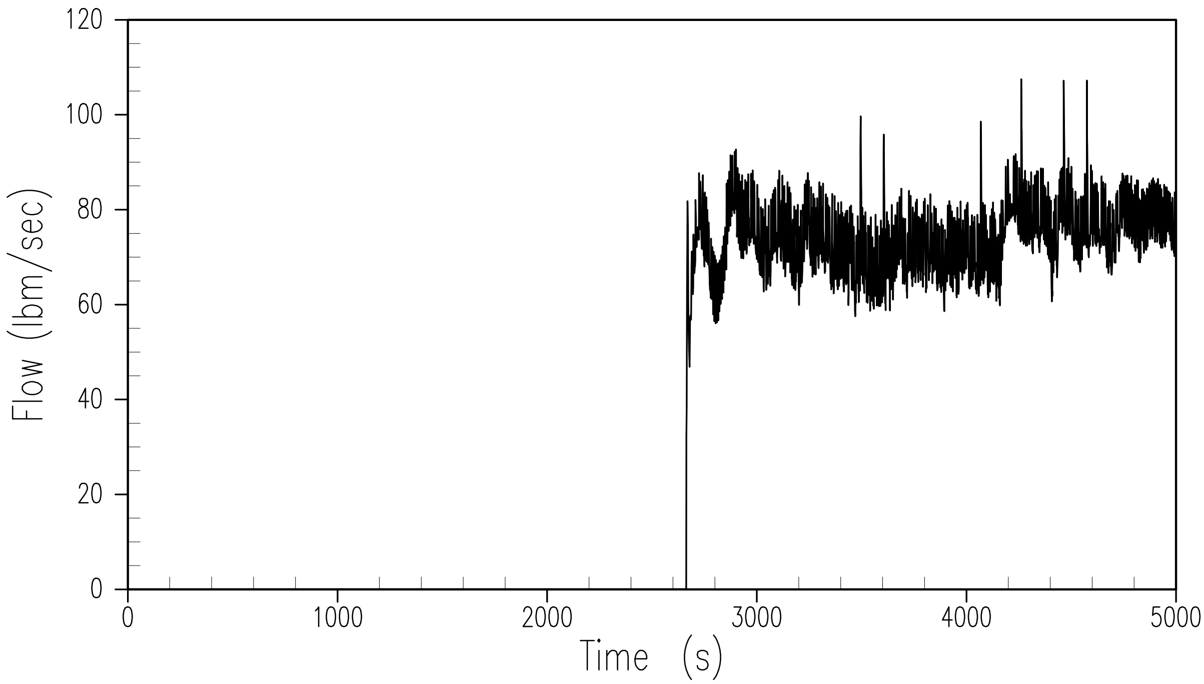

Figure 15.6.5.4B-14
Inadvertent ADS – IRWST-2 Injection Rate

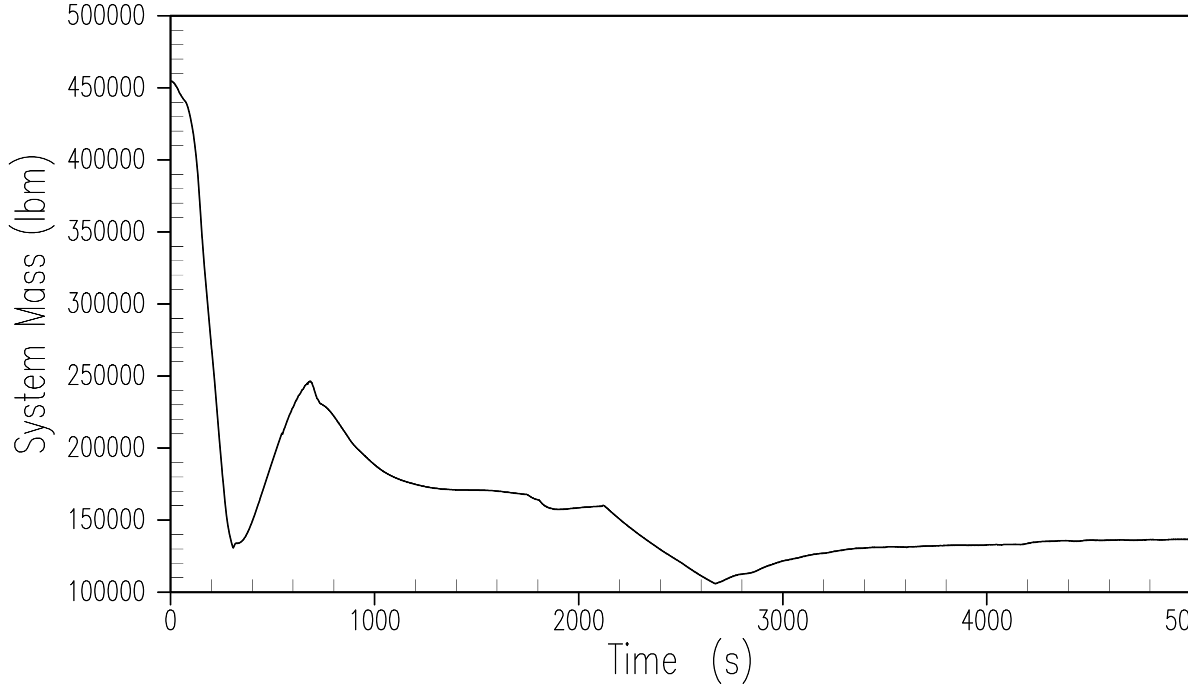

Figure 15.6.5.4B-15
Inadvertent ADS – RCS System Inventory

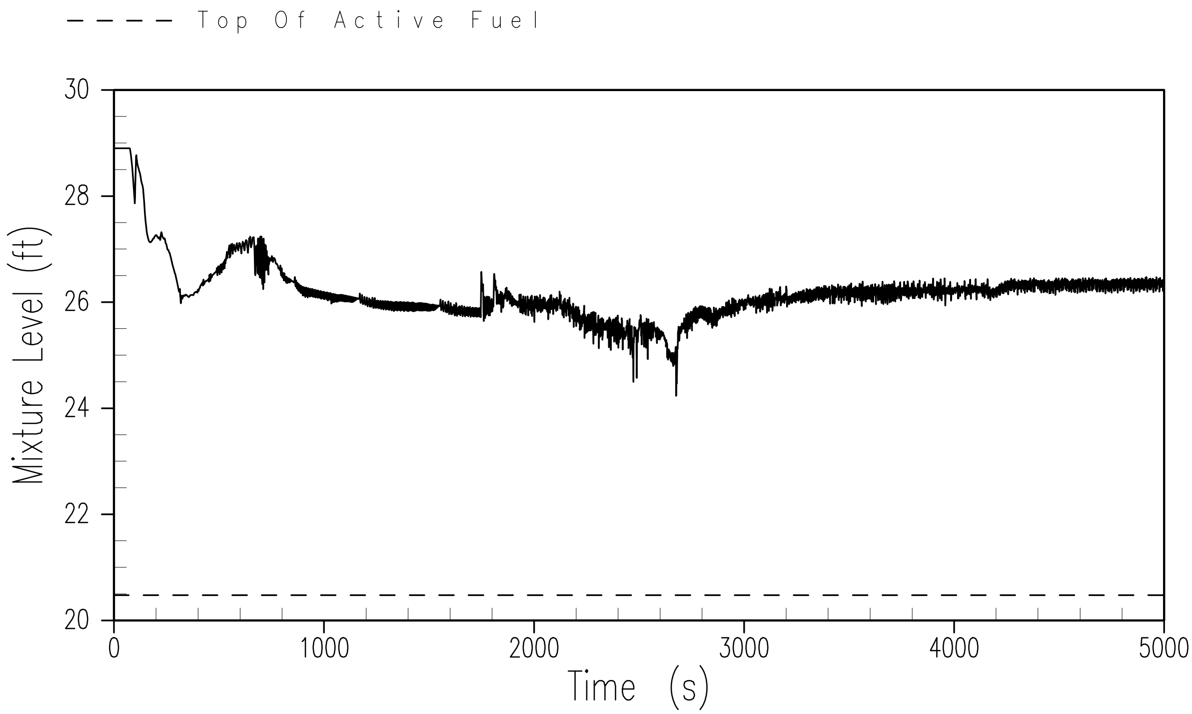

Figure 15.6.5.4B-16
Inadvertent ADS – Core/Upper Plenum Mixture Level

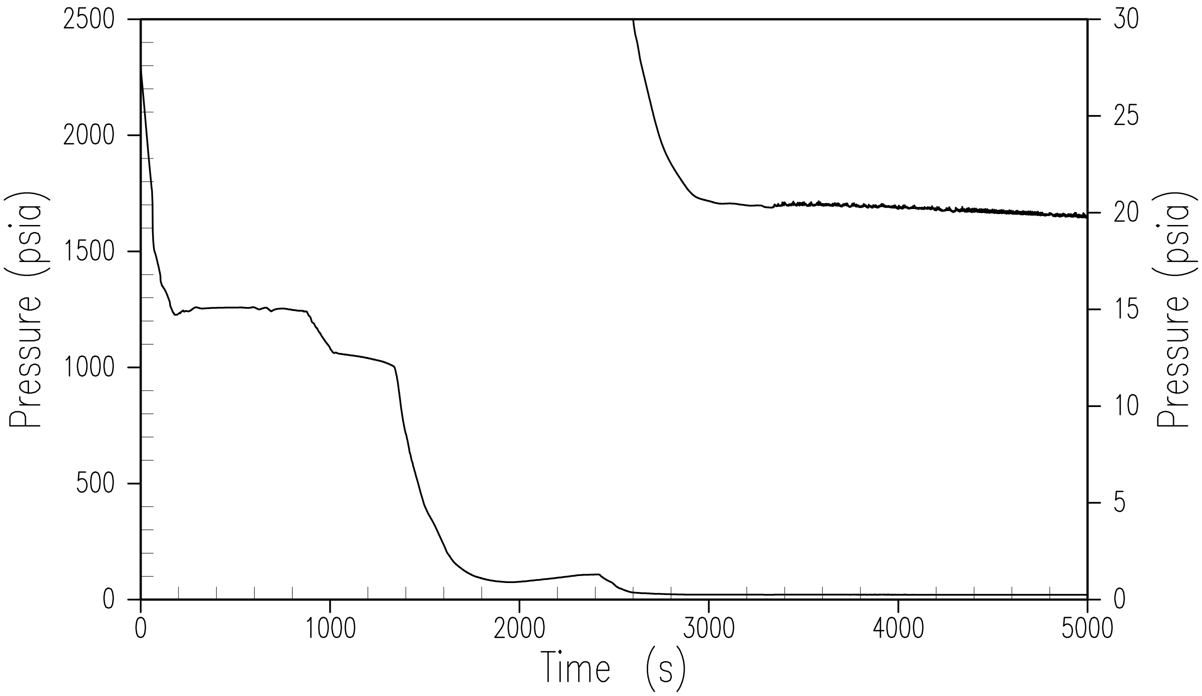

Figure 15.6.5.4B-17
2-inch Cold Leg Break – RCS Pressure

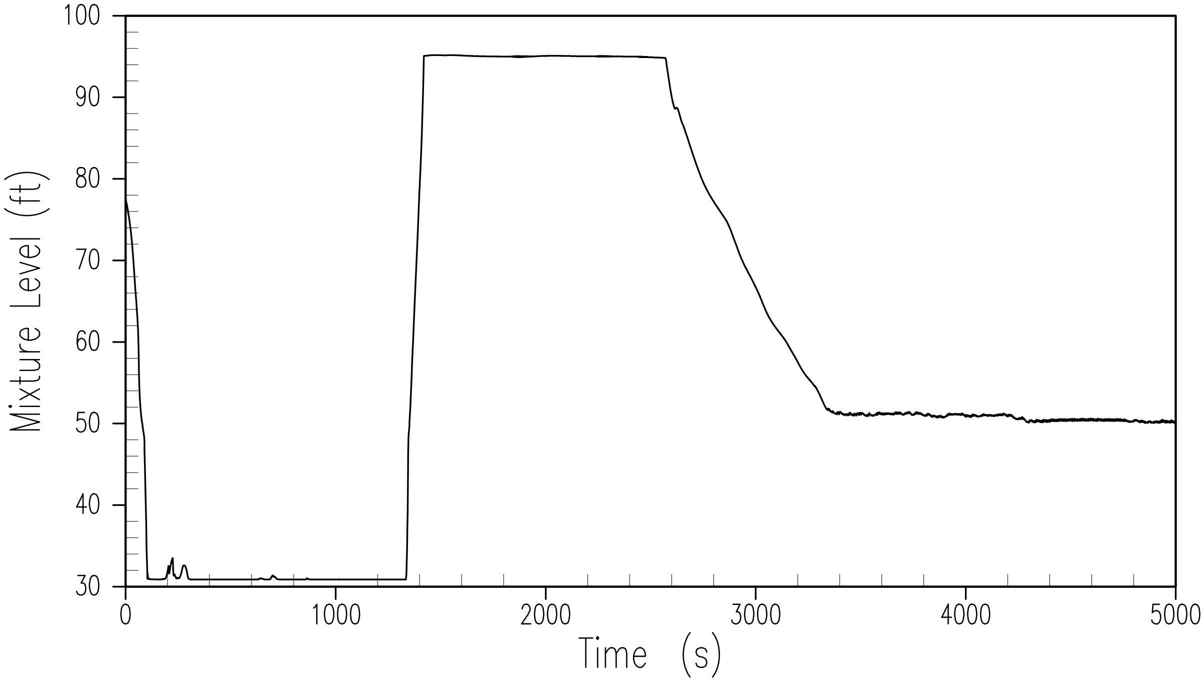

Figure 15.6.5.4B-18
2-inch Cold Leg Break – Pressurizer Mixture Level

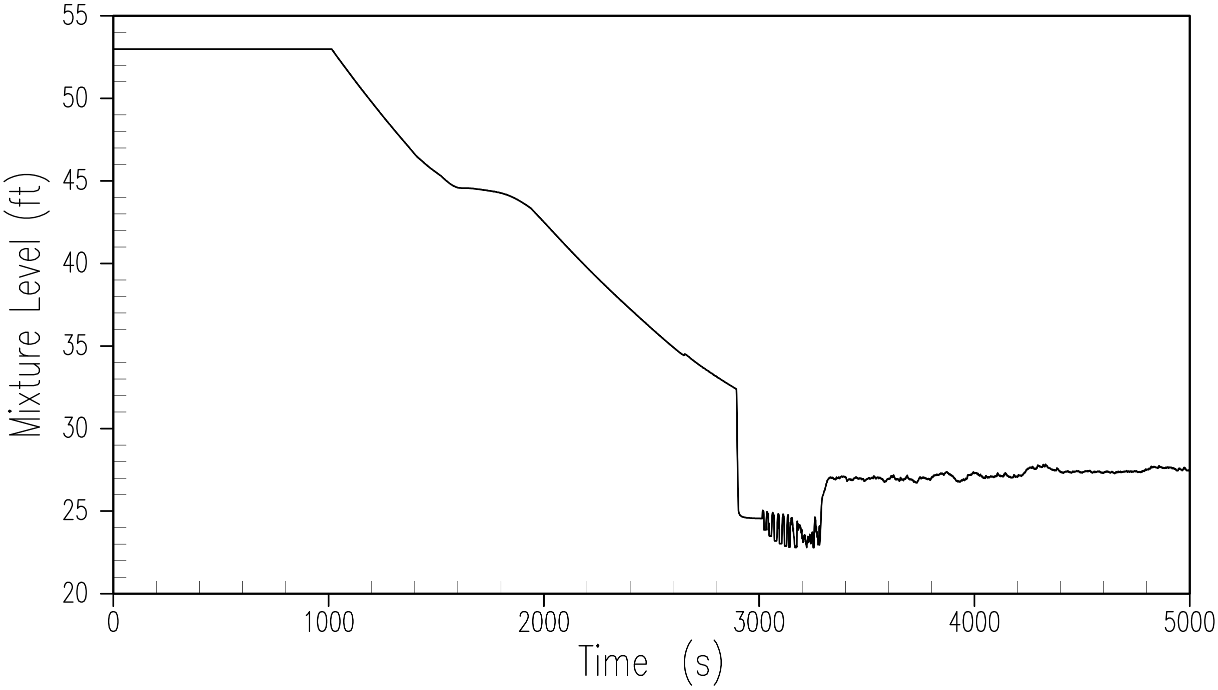

Figure 15.6.5.4B-19
2-inch Cold Leg Break – CMT-1 Mixture Level

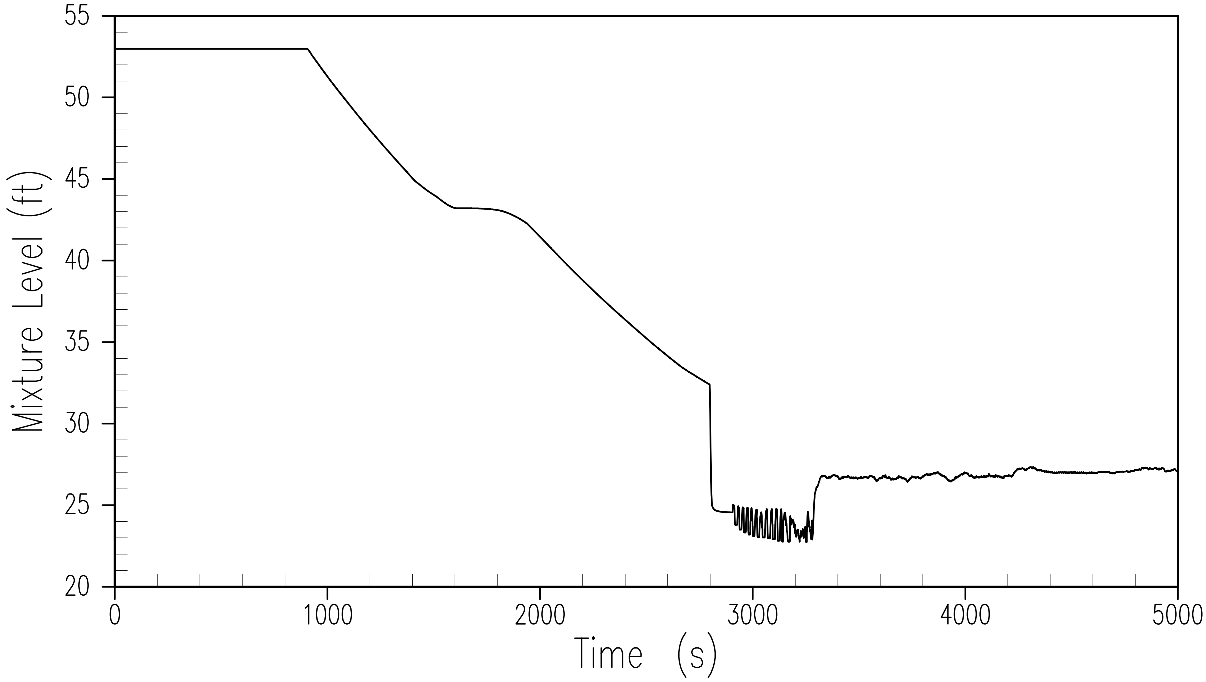

Figure 15.6.5.4B-20
2-inch Cold Leg Break – CMT-2 Mixture Level

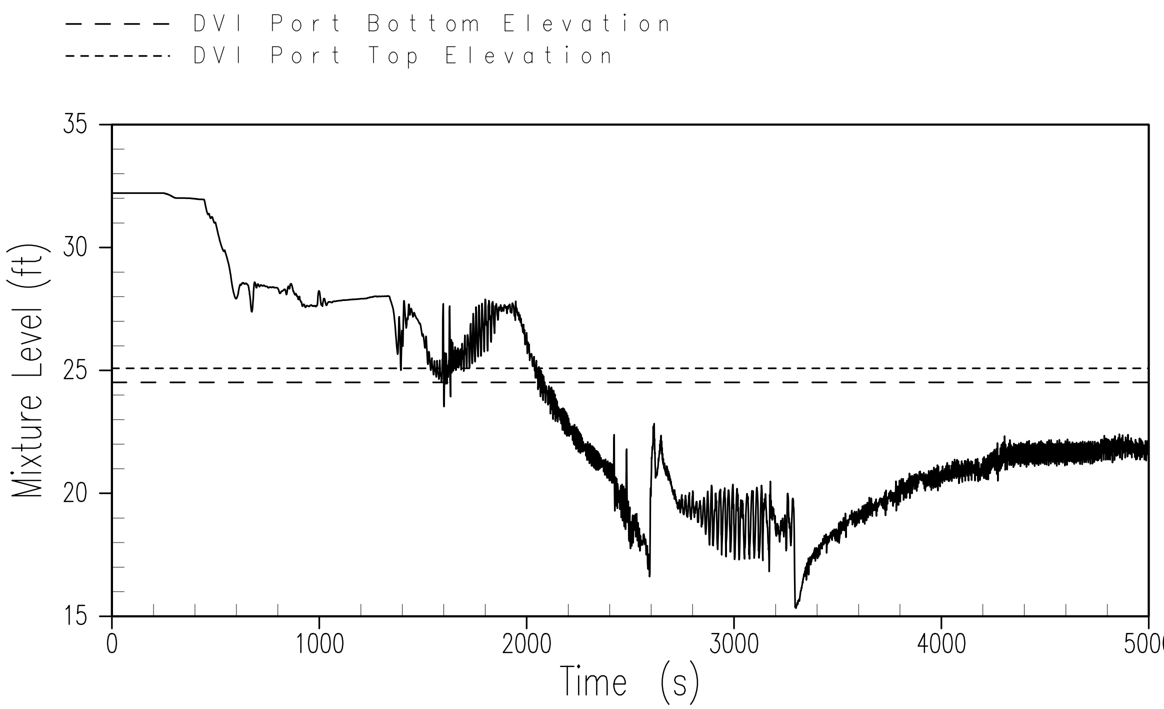

Figure 15.6.5.4B-21
2-inch Cold Leg Break – Downcomer Mixture Level

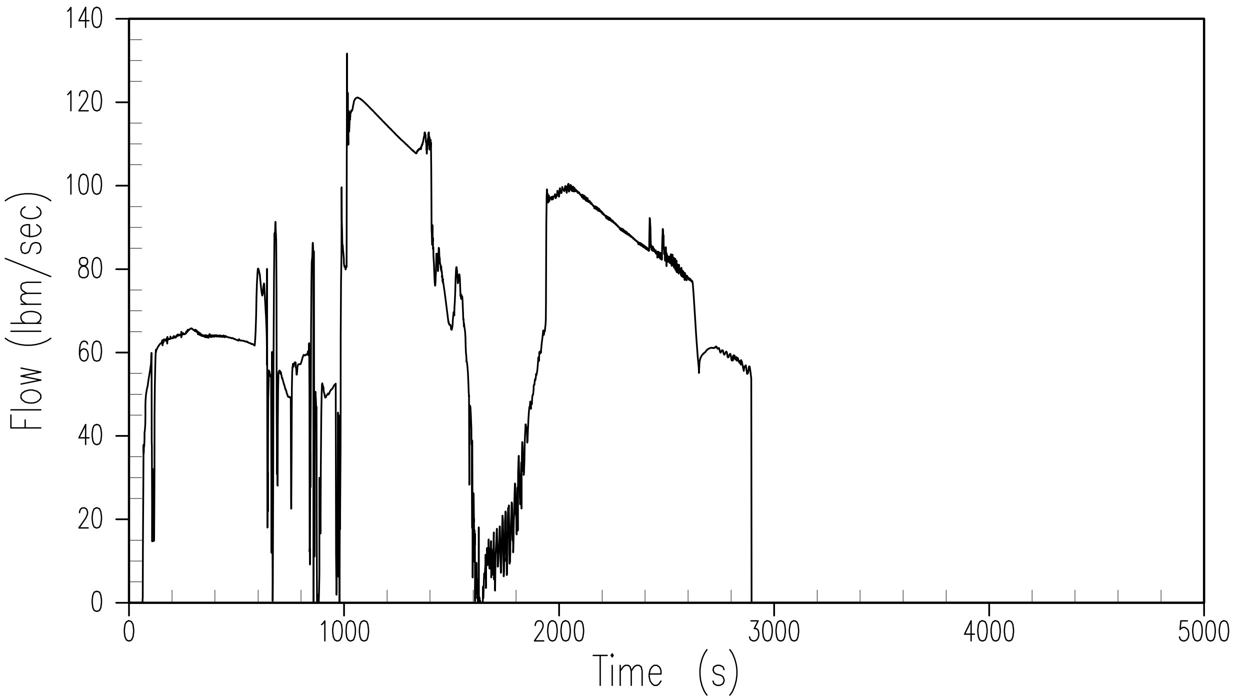

Figure 15.6.5.4B-22
2-inch Cold Leg Break – CMT-1 Injection Rate

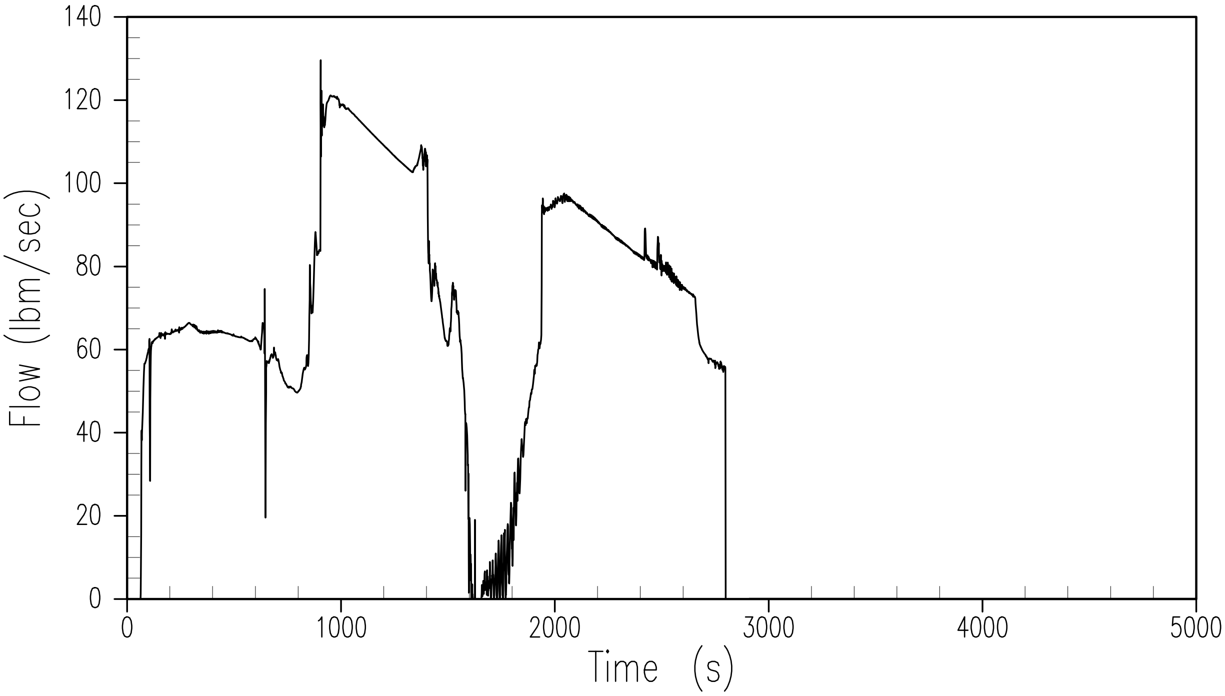

Figure 15.6.5.4B-23
2-inch Cold Leg Break – CMT-2 Injection Rate

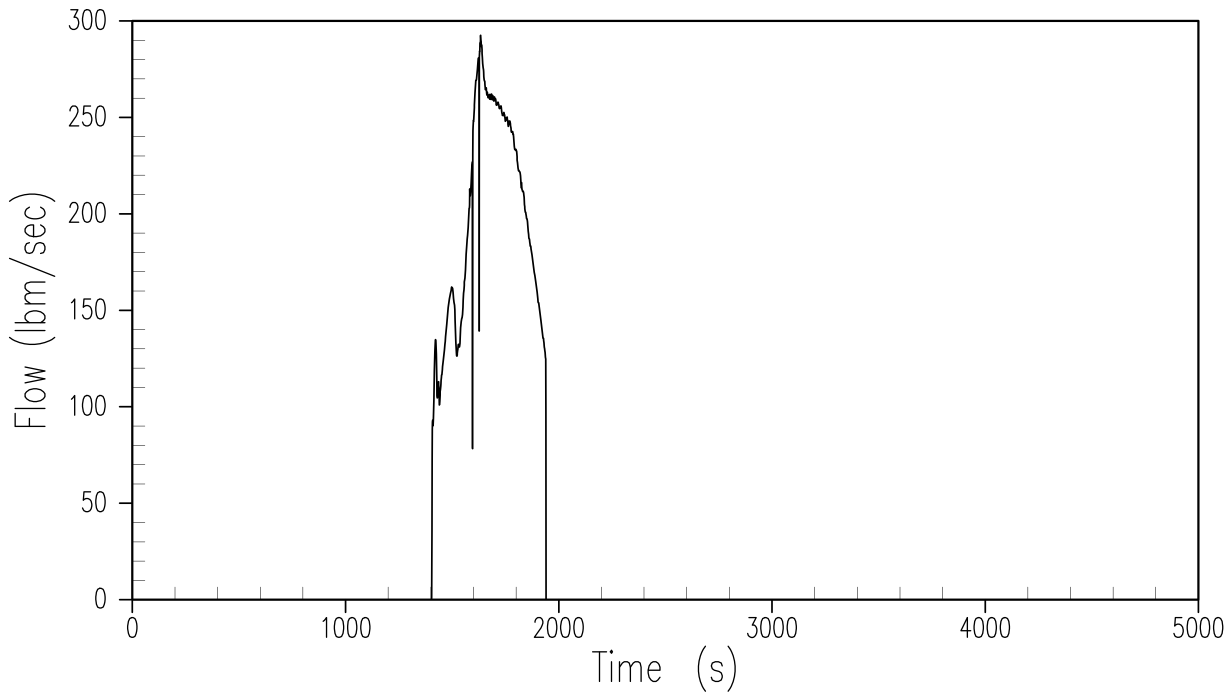

Figure 15.6.5.4B-24
2-inch Cold Leg Break – Accumulator-1 Injection Rate

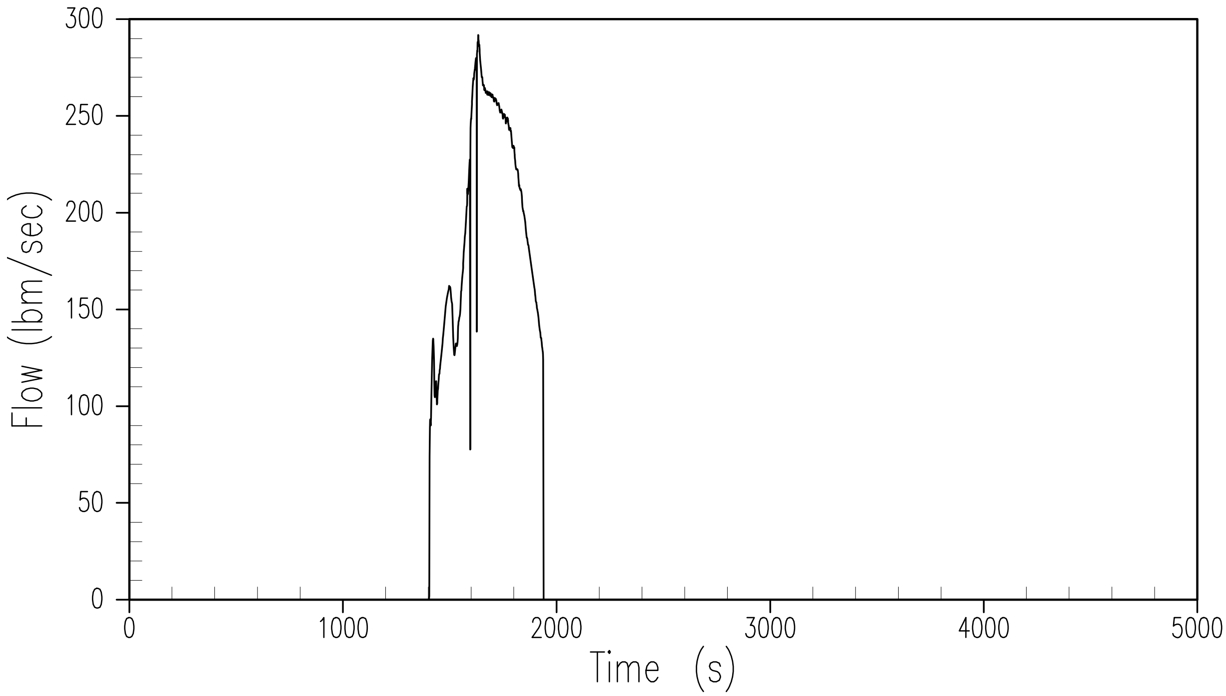

Figure 15.6.5.4B-25
2-inch Cold Leg Break – Accumulator-2 Injection Rate

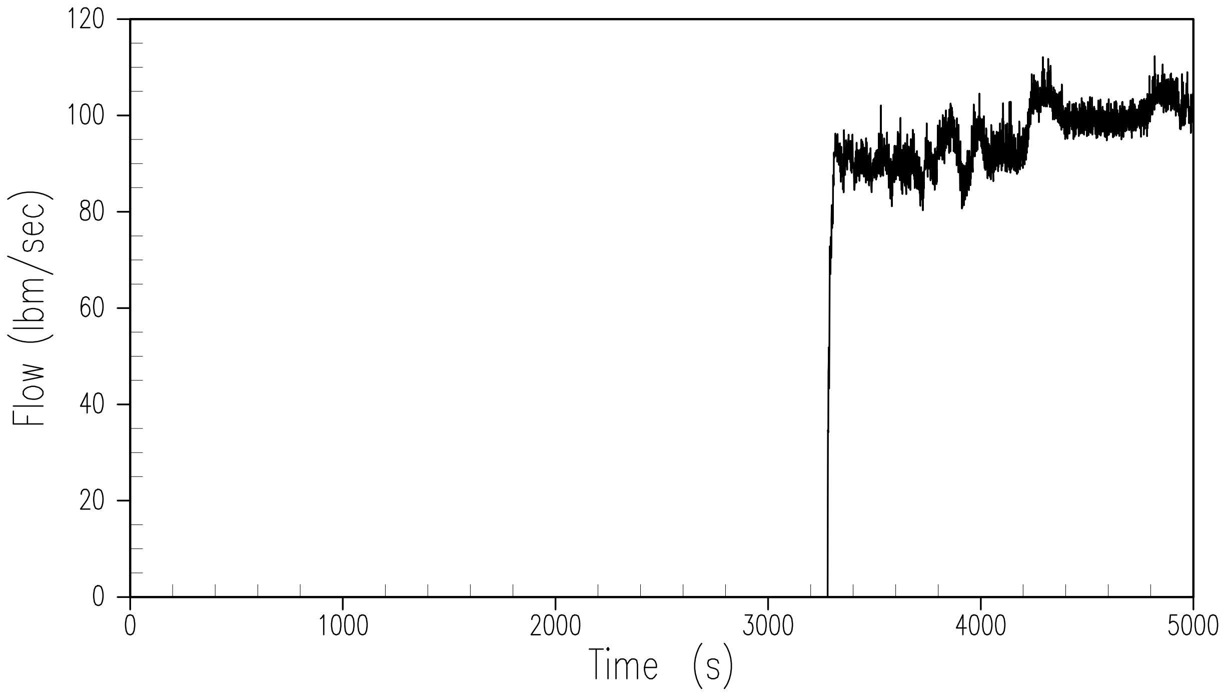

Figure 15.6.5.4B-26
2-inch Cold Leg Break – IRWST-1 Injection Rate

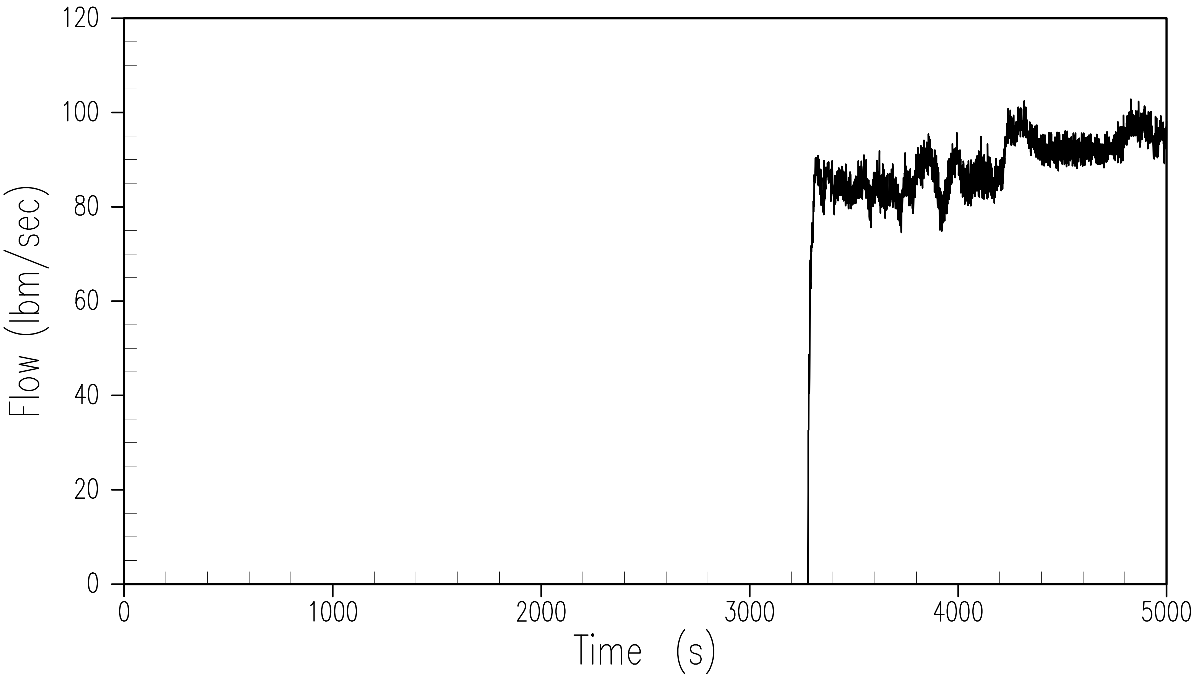

Figure 15.6.5.4B-27
2-inch Cold Leg Break – IRWST-2 Injection Rate

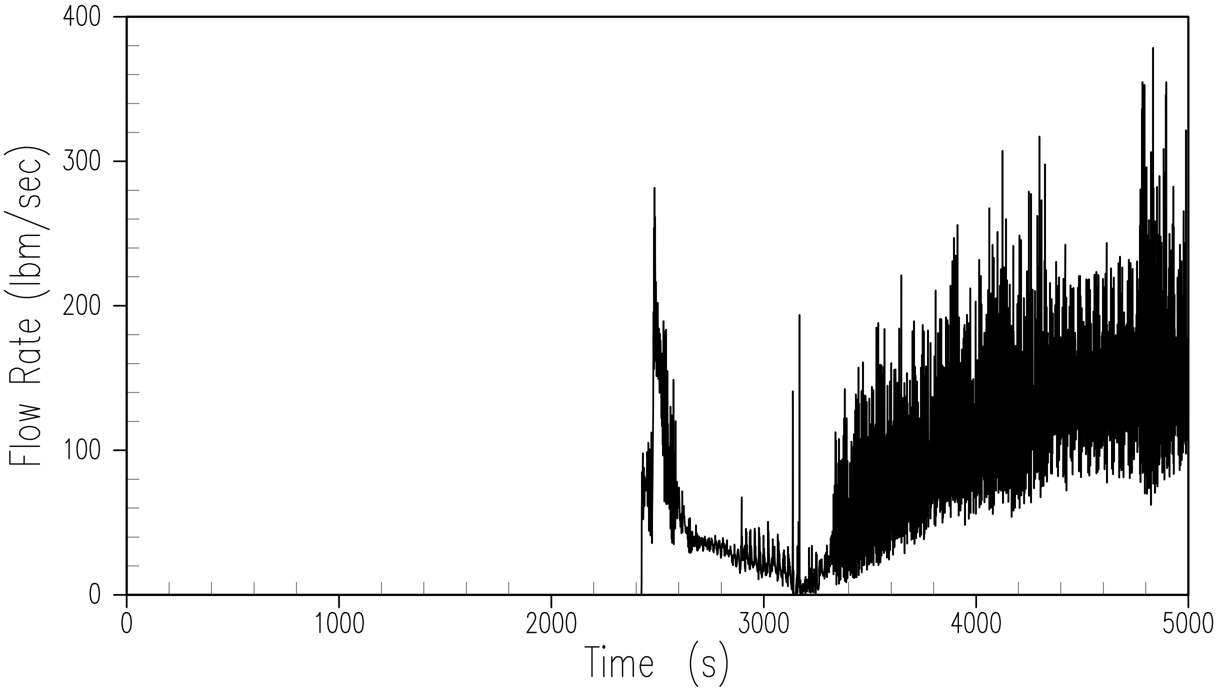

Figure 15.6.5.4B-28
2-inch Cold Leg Break – ADS-4 Liquid Discharge

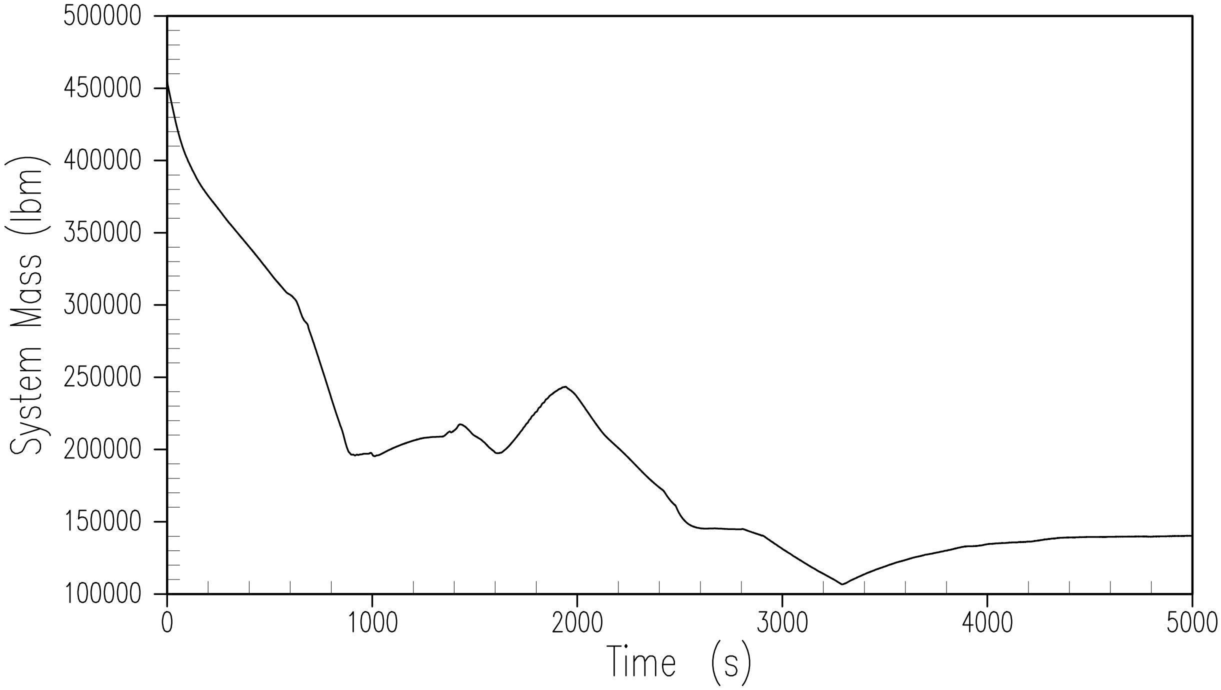

Figure 15.6.5.4B-29
2-inch Cold Leg Break – RCS System Inventory

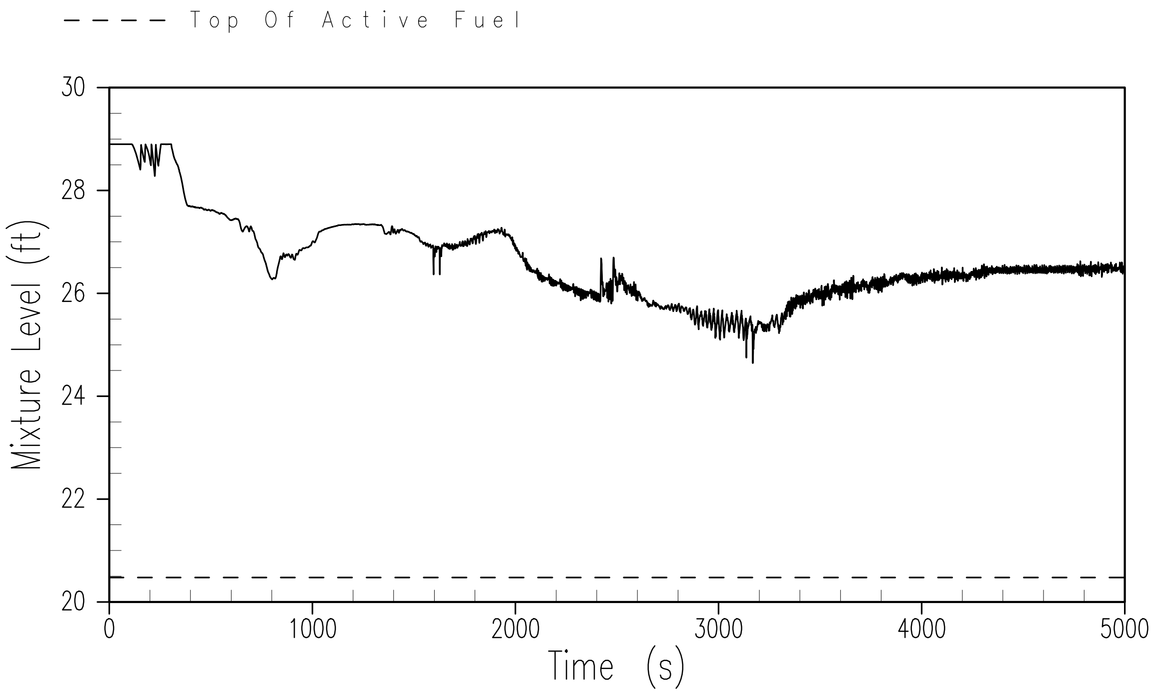

Figure 15.6.5.4B-30
2-inch Cold Leg Break – Core/Upper Plenum Mixture Level

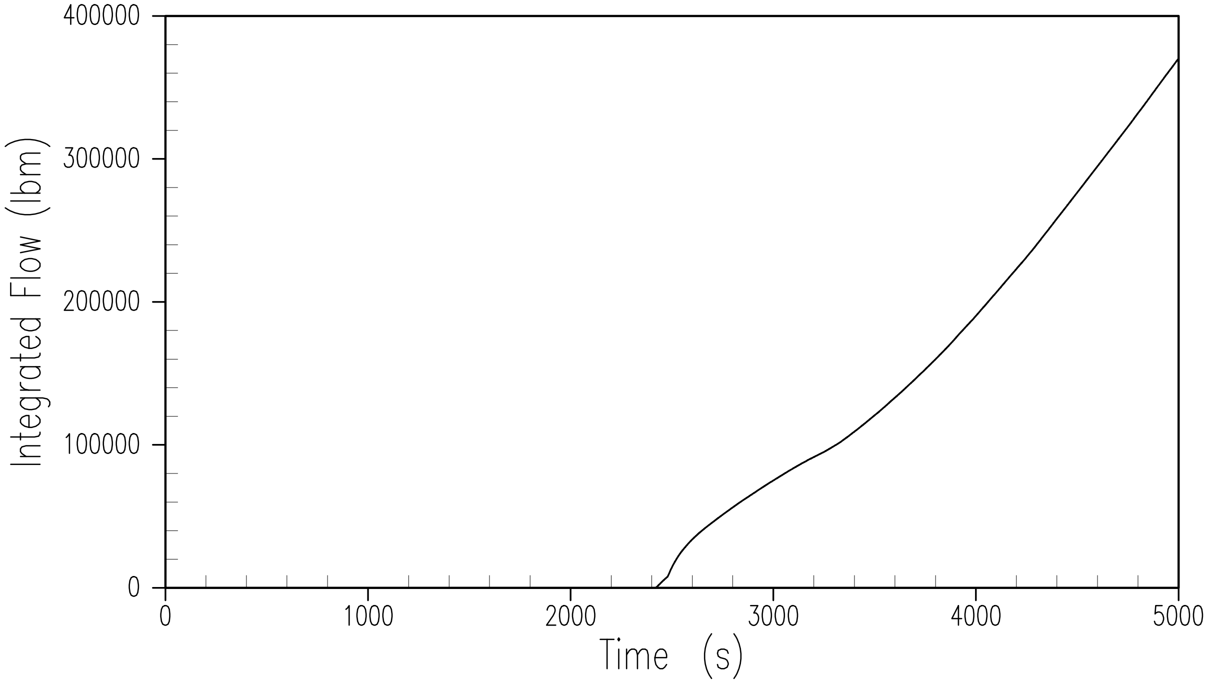

Figure 15.6.5.4B-31
2-inch Cold Leg Break – ADS-4 Integrated Discharge

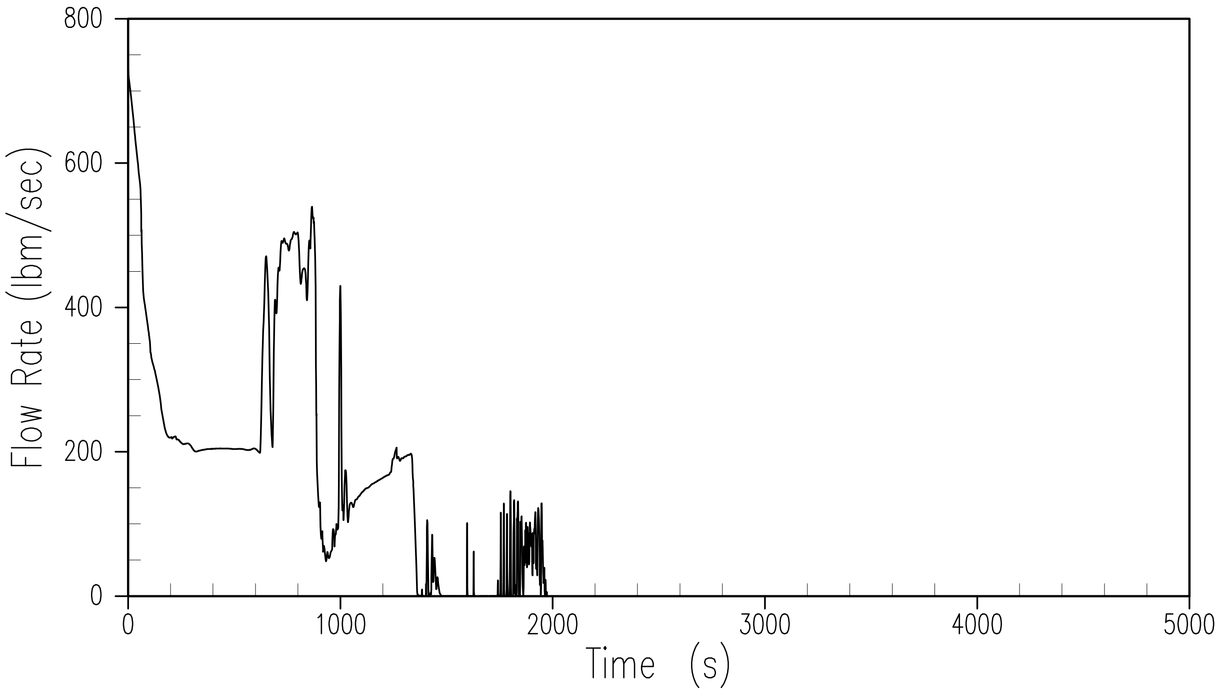

Figure 15.6.5.4B-32
2-inch Cold Leg Break – Liquid Break Discharge

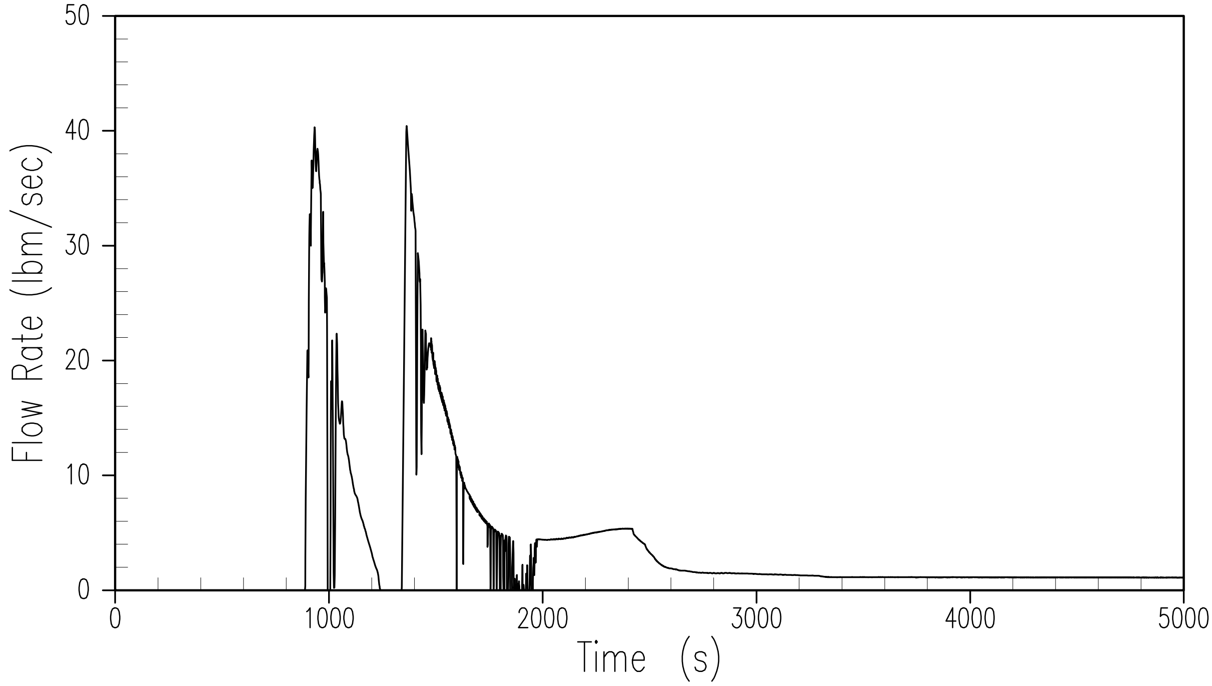

Figure 15.6.5.4B-33
2-inch Cold Leg Break – Vapor Break Discharge

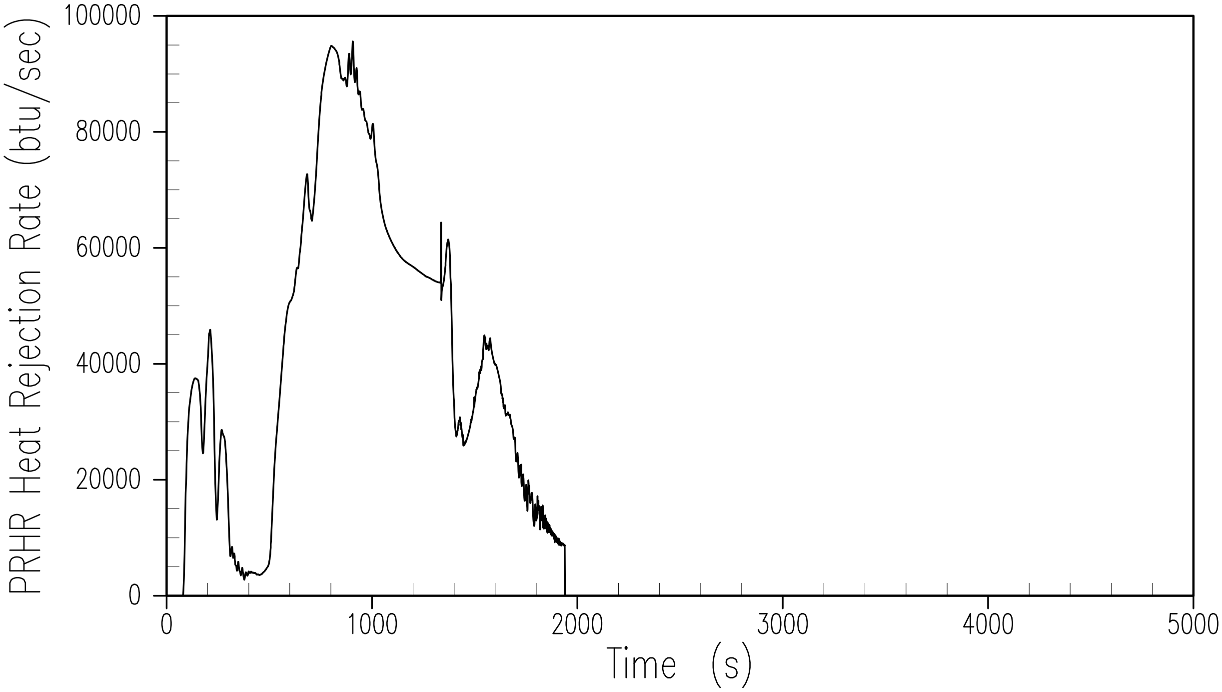

Figure 15.6.5.4B-34
2-inch Cold Leg Break – PRHR Heat Removal Rate

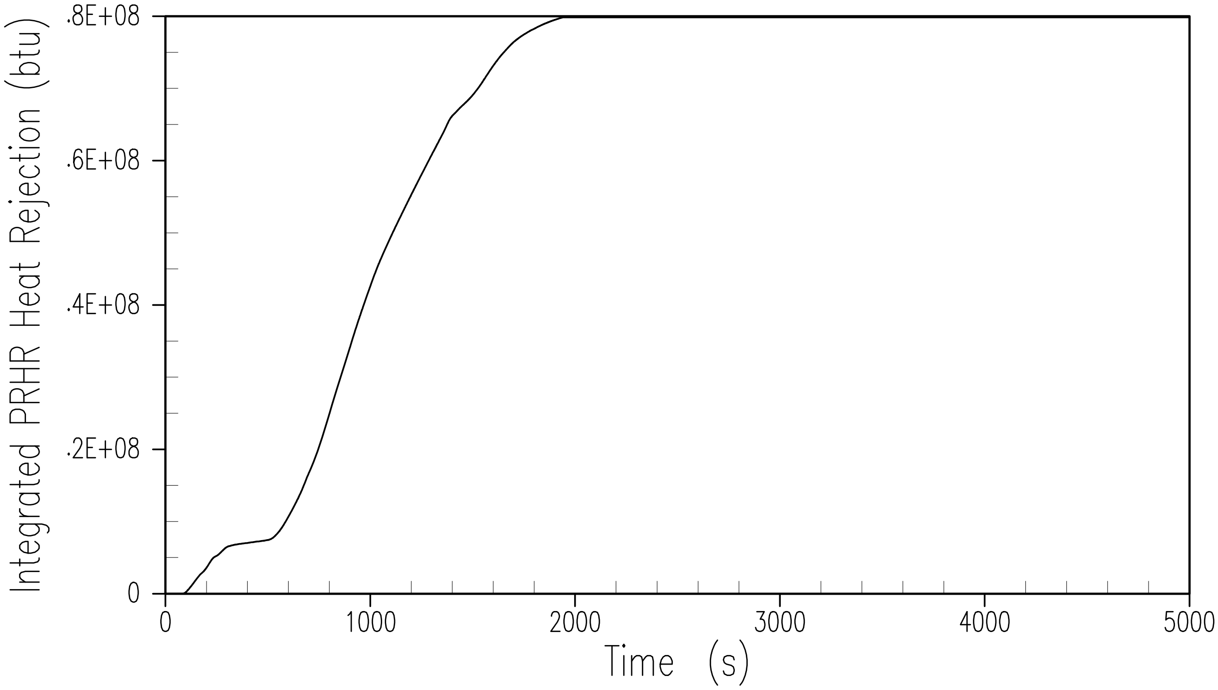

Figure 15.6.5.4B-35
2-inch Cold Leg Break – Integrated PRHR Heat Removal

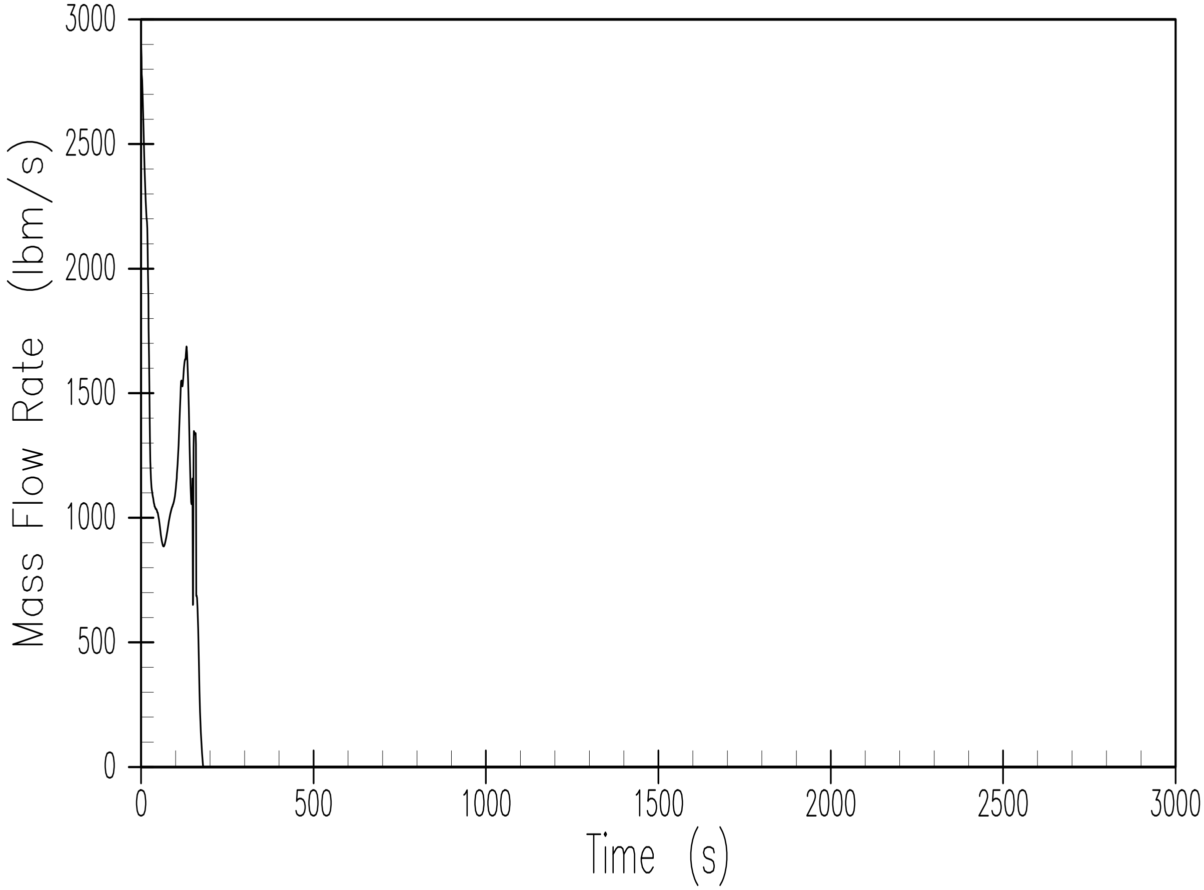

Figure 15.6.5.4B-36
DEDVI – Vessel Side Liquid Break Discharge – 20 psi

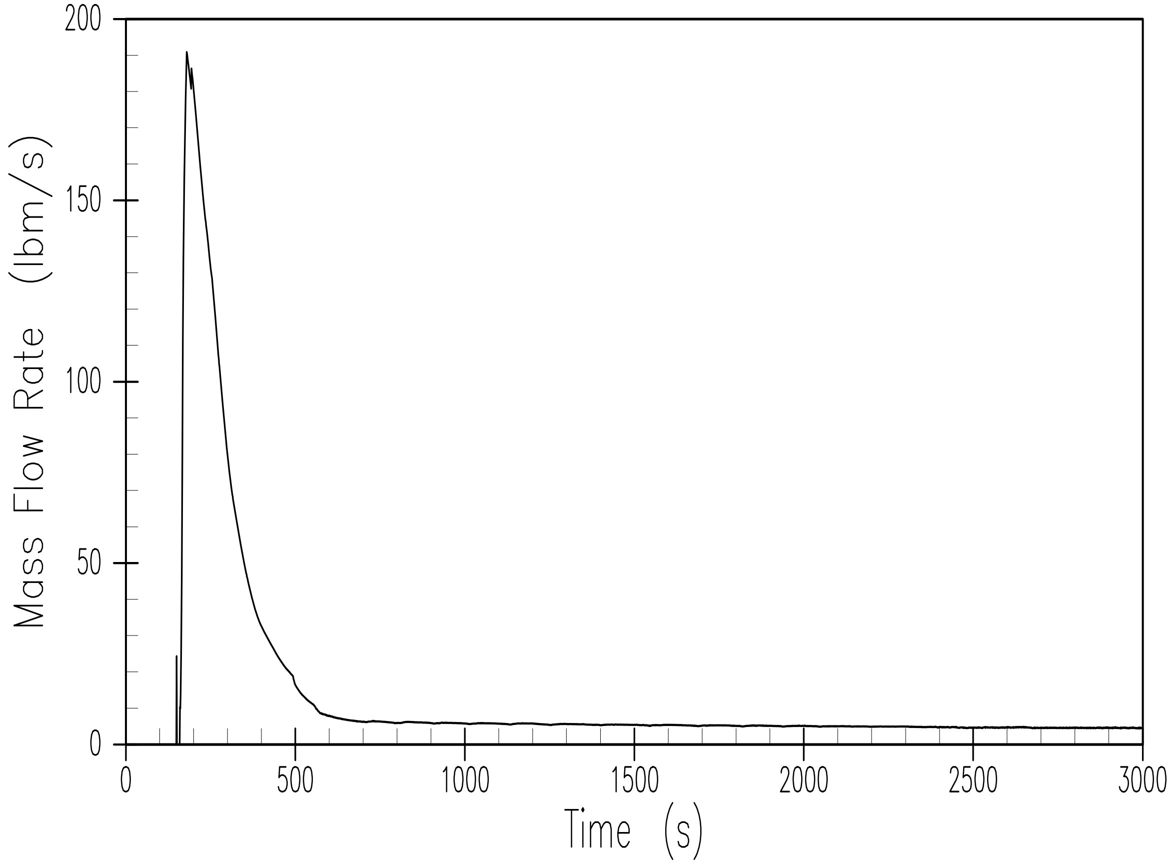

Figure 15.6.5.4B-37
DEDVI – Vessel Side Vapor Break Discharge – 20 psi

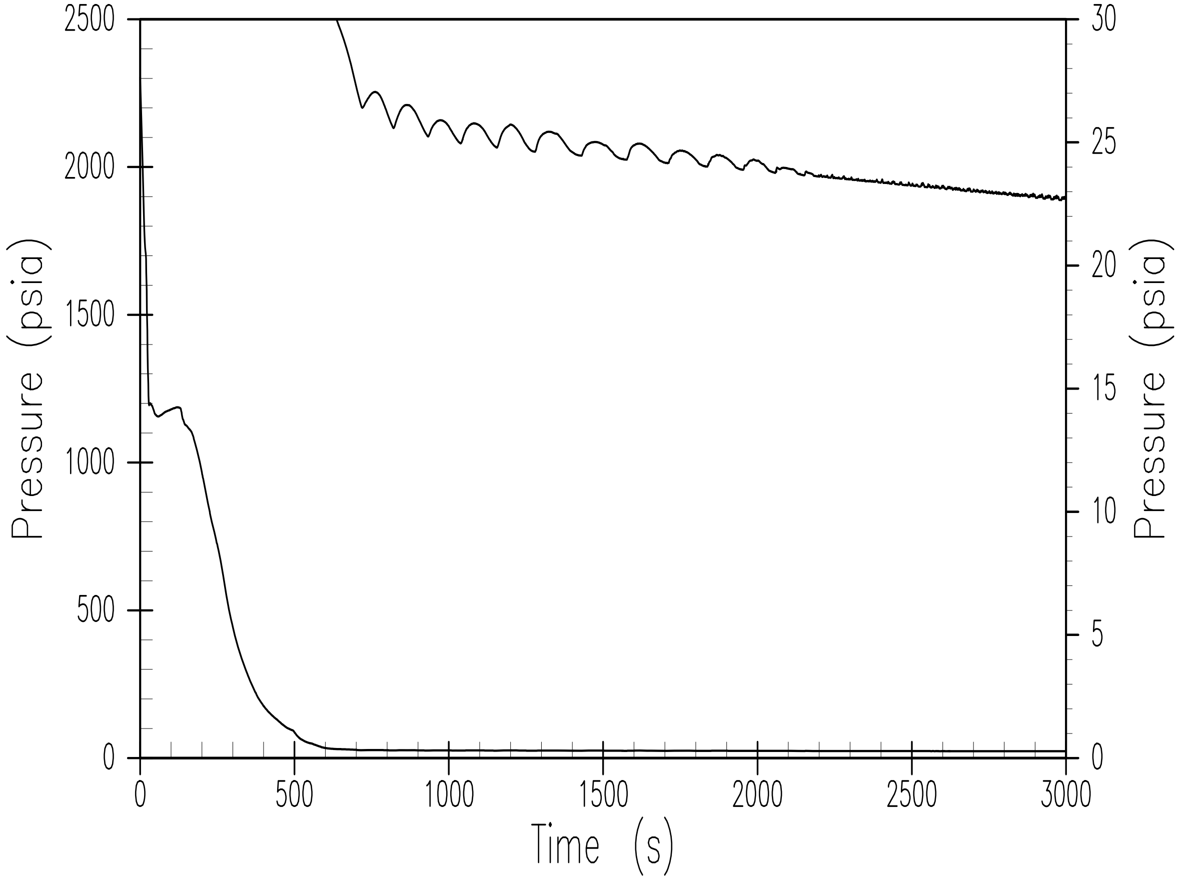

Figure 15.6.5.4B-38
DEDVI – RCS Pressure – 20 psi

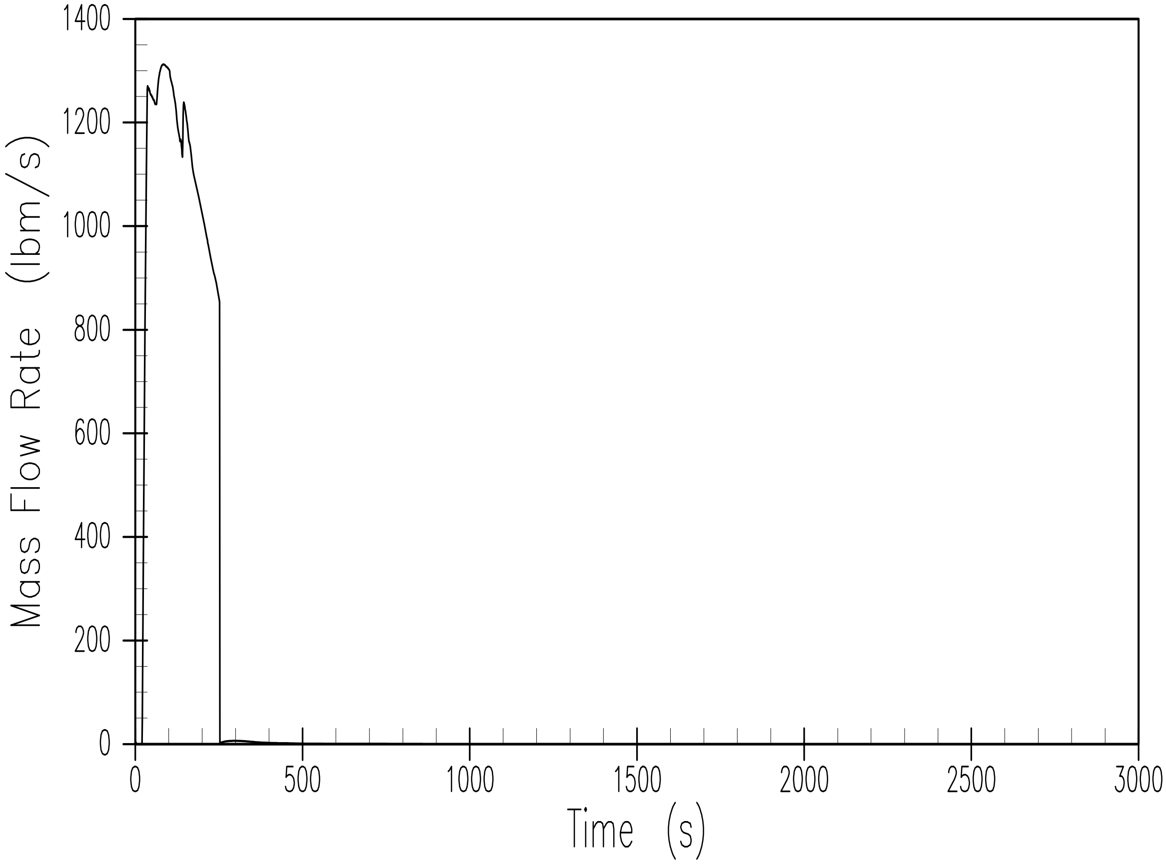

Figure 15.6.5.4B-39
DEDVI – Broken CMT Injection Rate – 20 psi

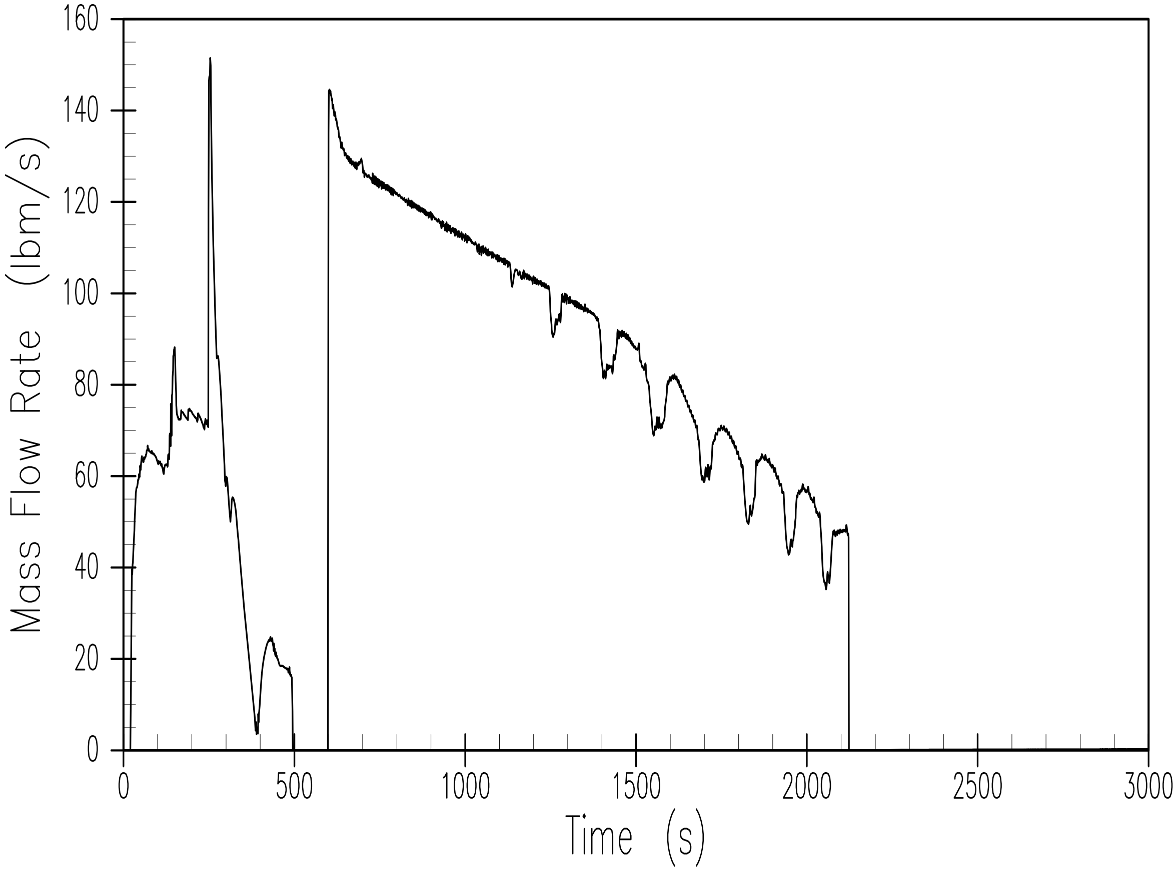

Figure 15.6.5.4B-40
DEDVI – Intact CMT Injection Rate – 20 psi

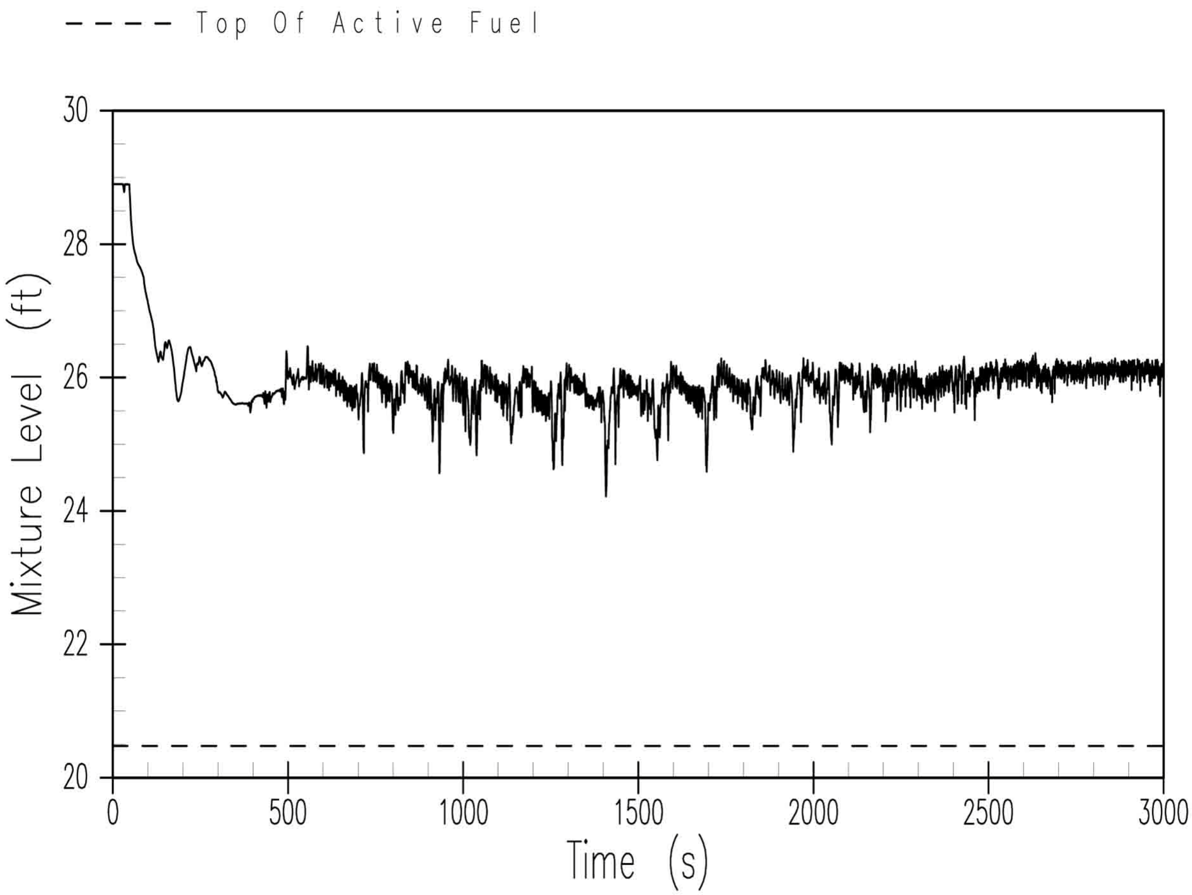

Figure 15.6.5.4B-41
DEDVI – Core/Upper Plenum Mixture Level – 20 psi

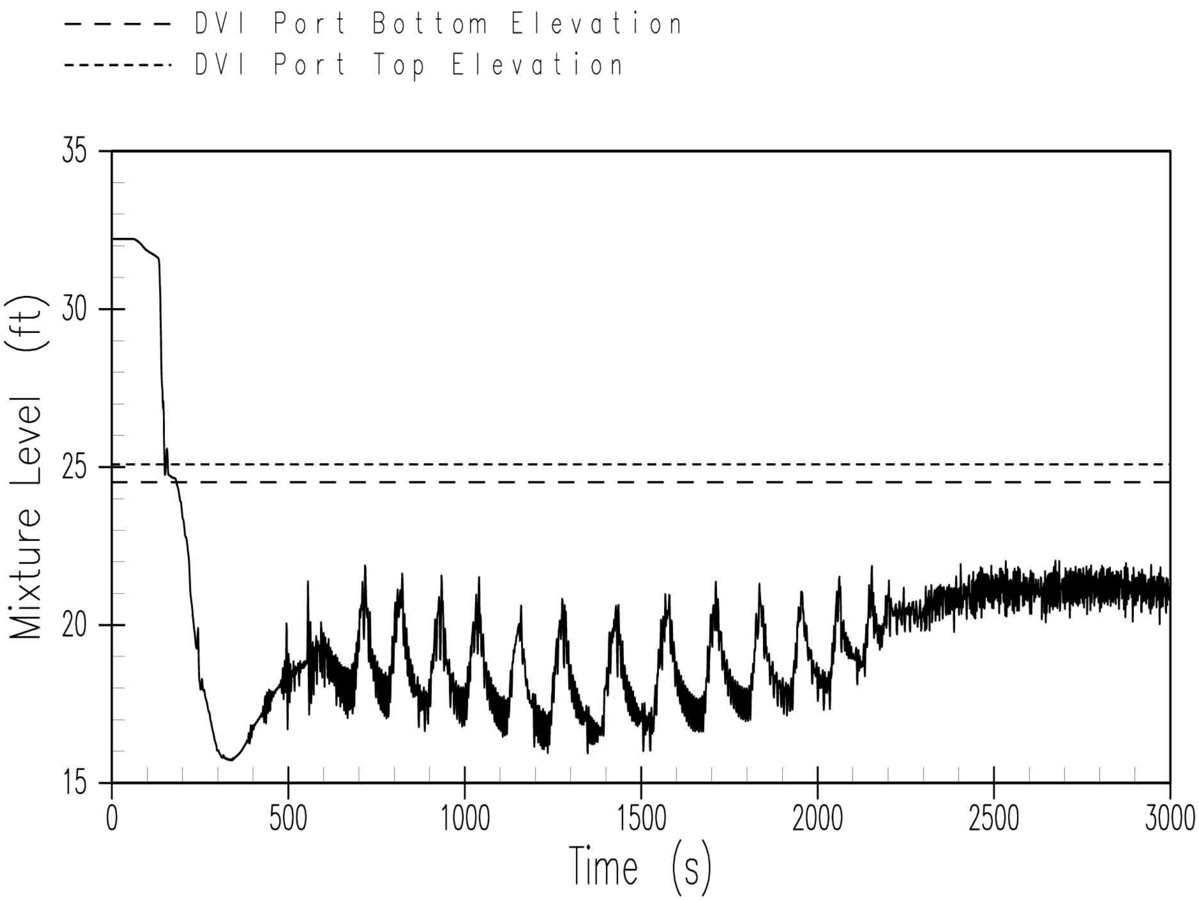

Figure 15.6.5.4B-42
DEDVI – Downcomer Mixture Level – 20 psi

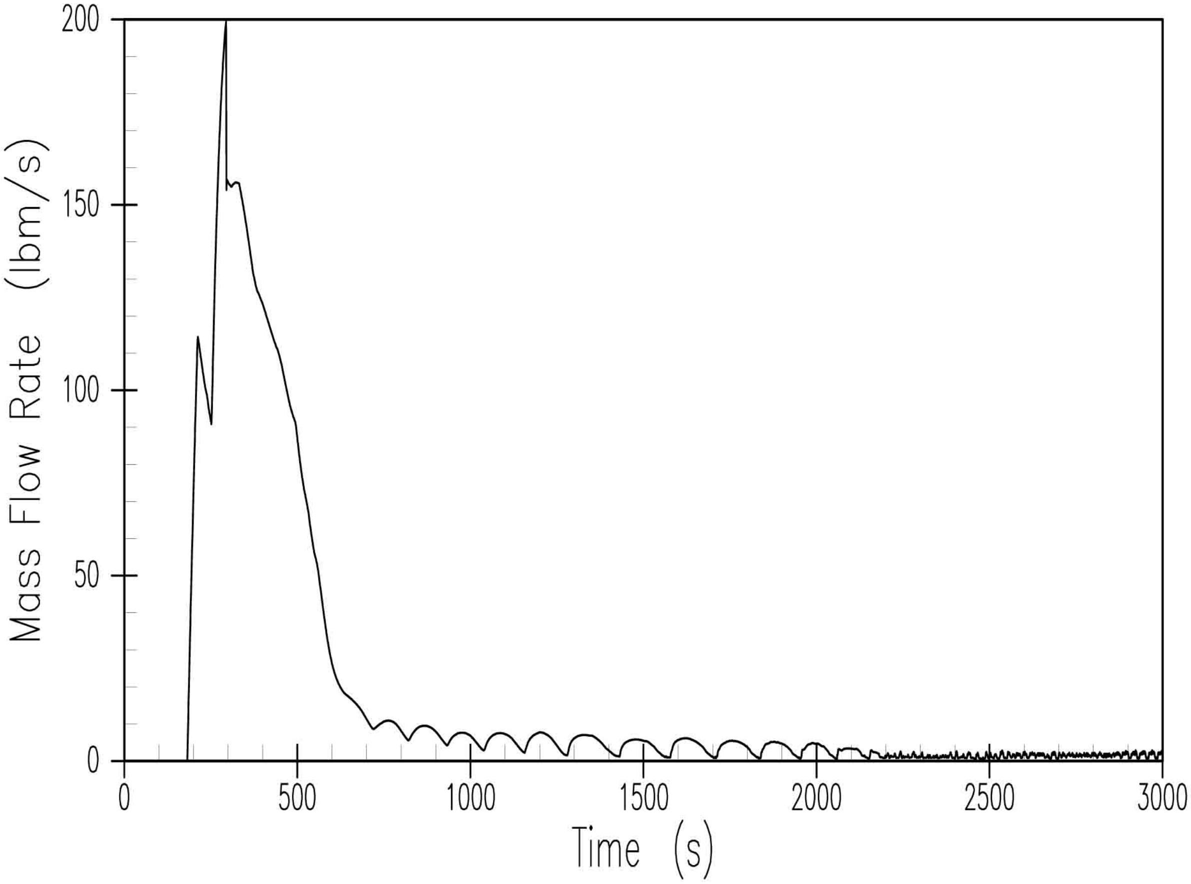

Figure 15.6.5.4B-43
DEDVI – ADS 1-3 Vapor Discharge – 20 psi

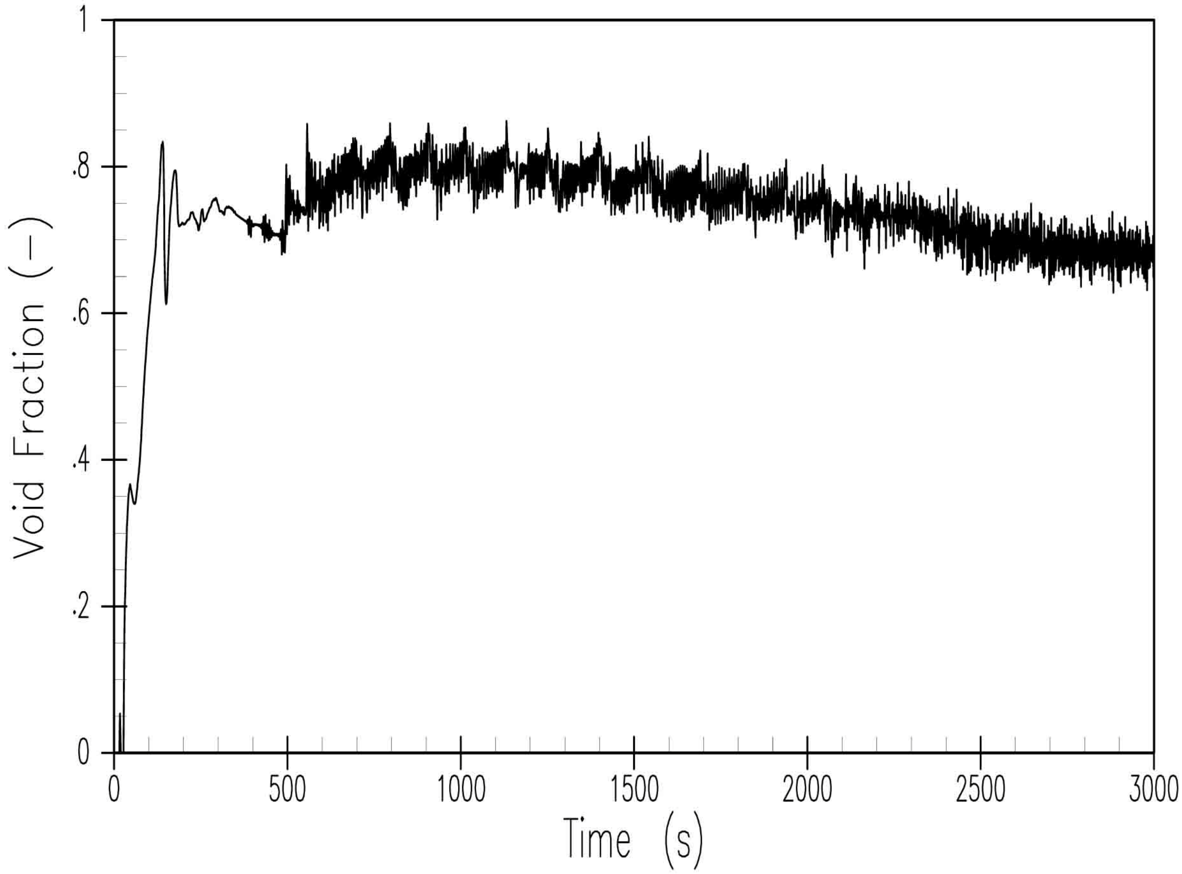

Figure 15.6.5.4B-44
DEDVI – Core Exit Void Fraction – 20 psi

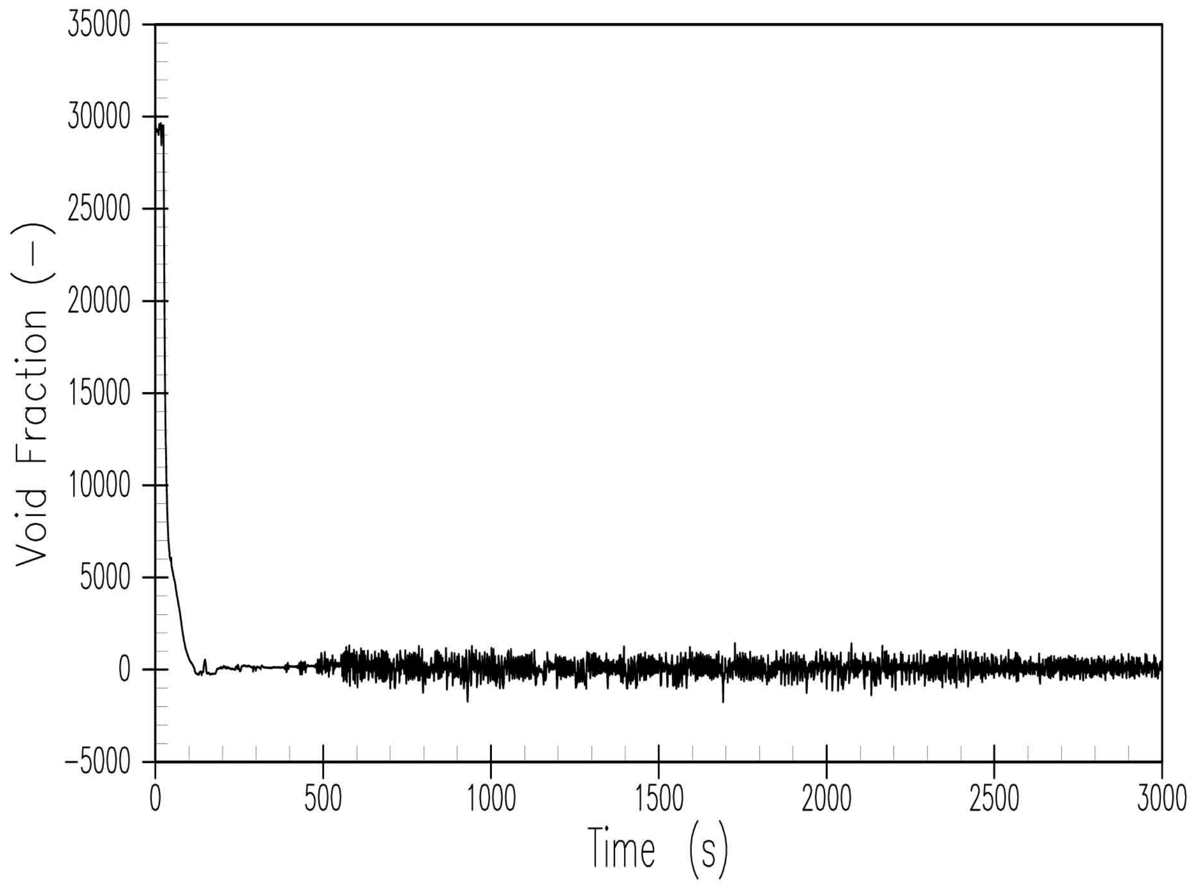

Figure 15.6.5.4B-45
DEDVI – Core Exit Liquid Flow Rate – 20 psi

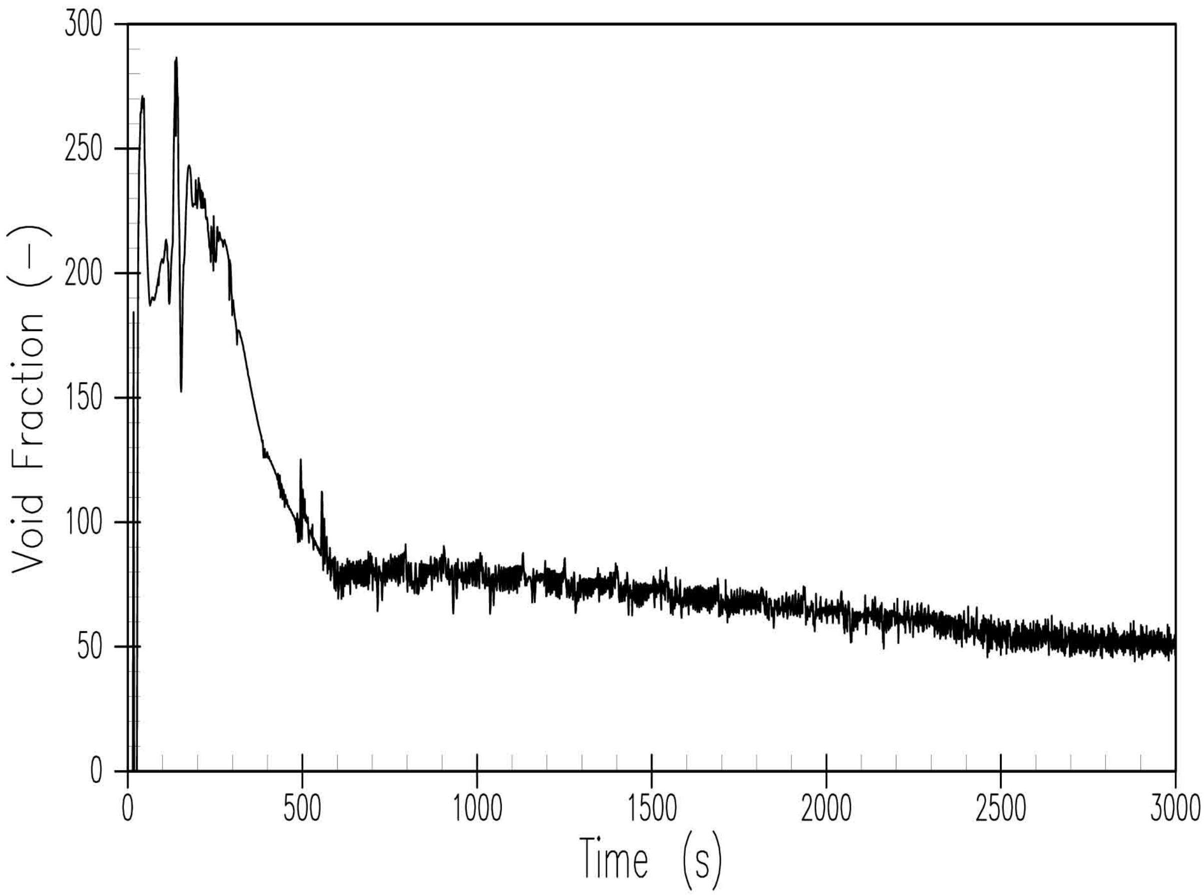

Figure 15.6.5.4B-46
DEDVI – Core Exit Vapor Flow Rate – 20 psi

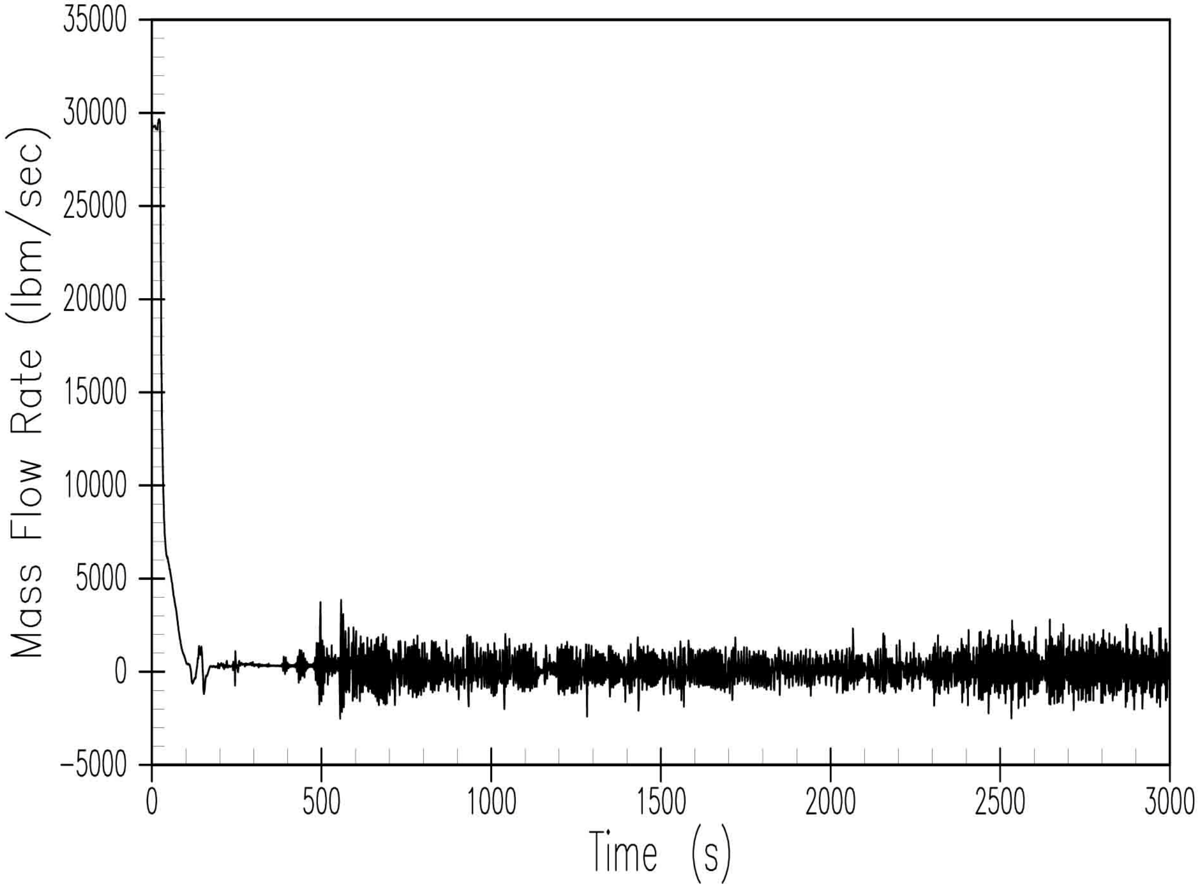

Figure 15.6.5.4B-47
DEDVI – Lower Plenum to Core Flow Rate – 20 psi

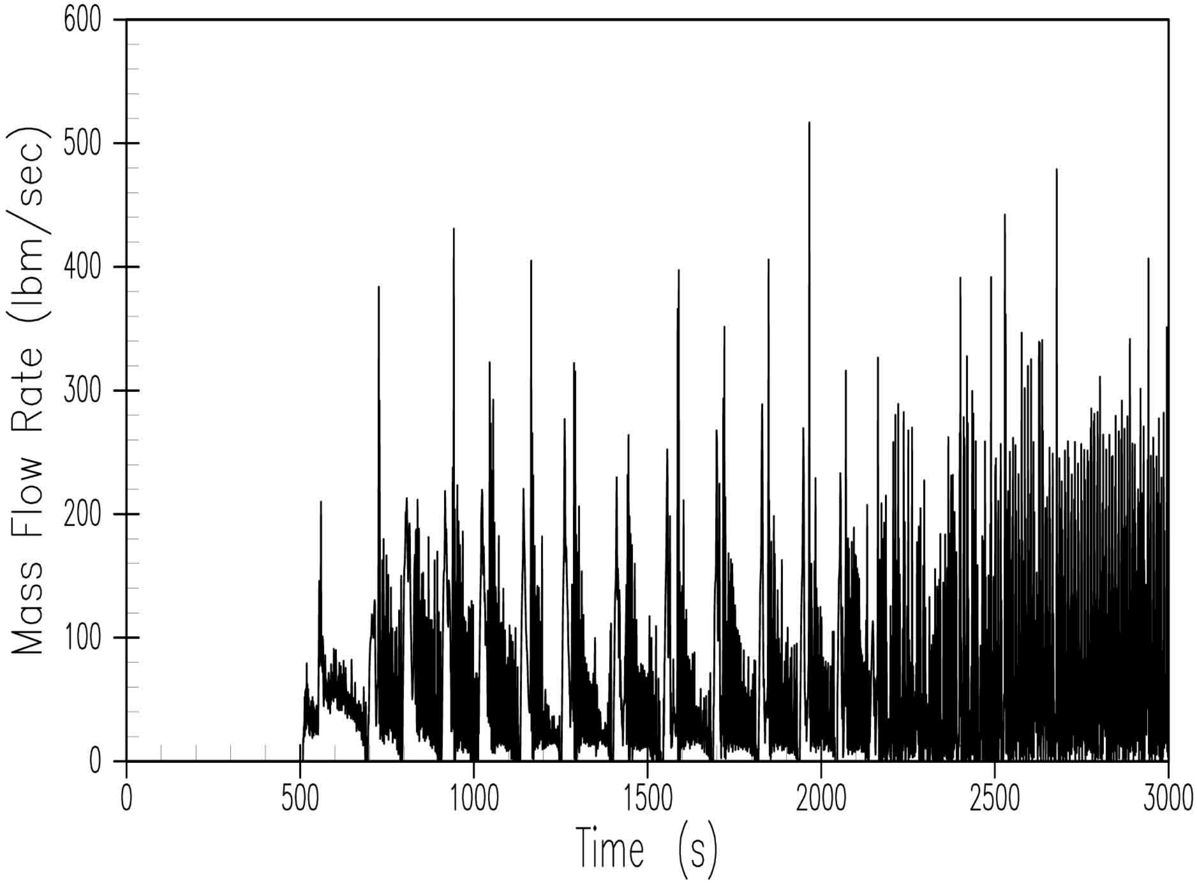

Figure 15.6.5.4B-48
DEDVI – ADS-4 Liquid Discharge – 20 psi

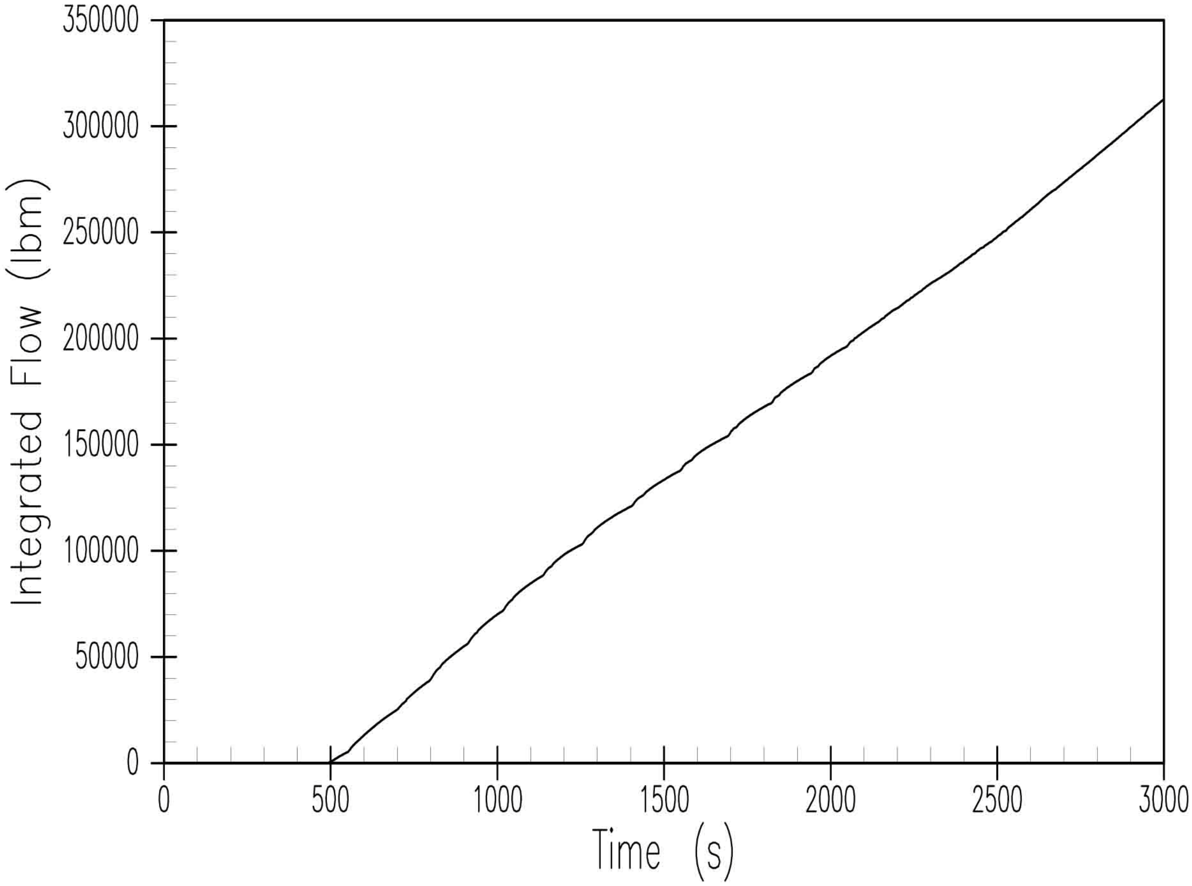

Figure 15.6.5.4B-49
DEDVI – ADS-4 Integrated Discharge – 20 psi

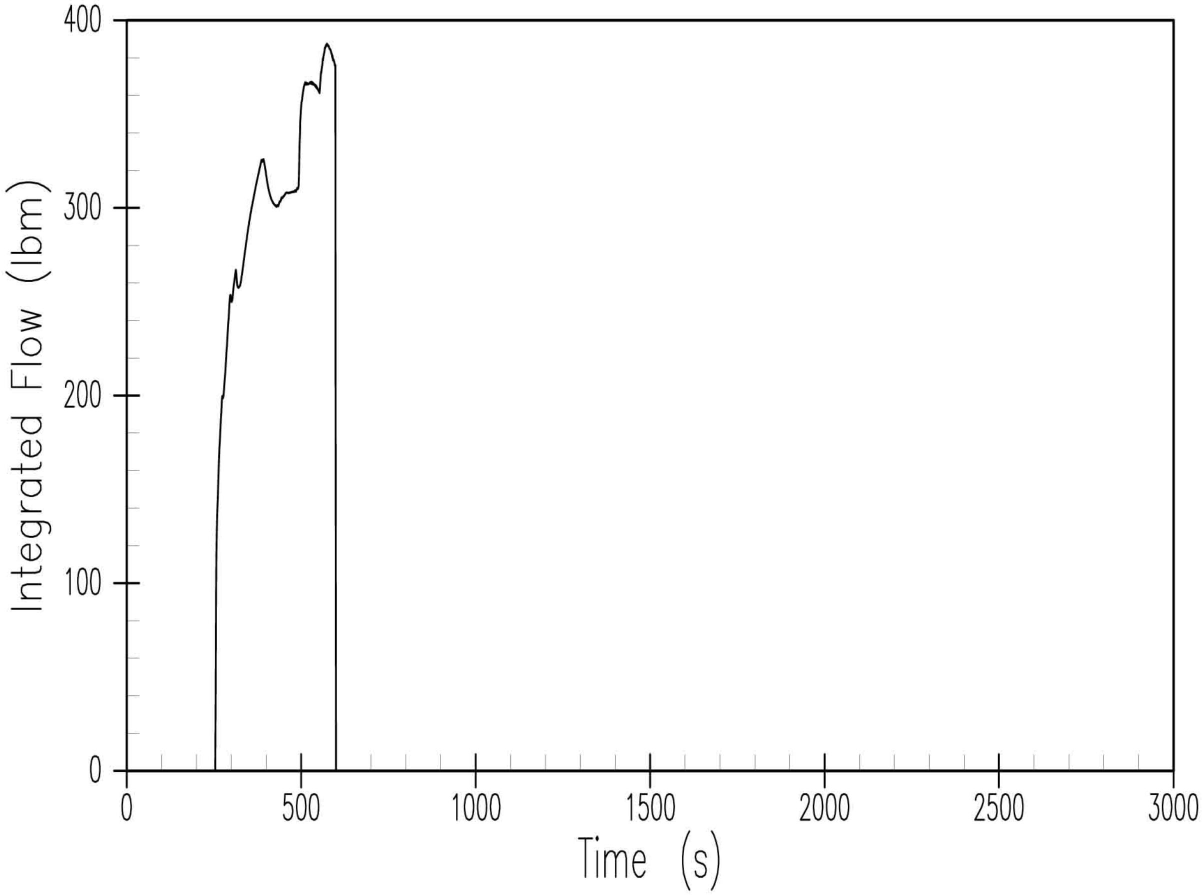

Figure 15.6.5.4B-50
DEDVI – Intact Accumulator Flow Rate – 20 psi

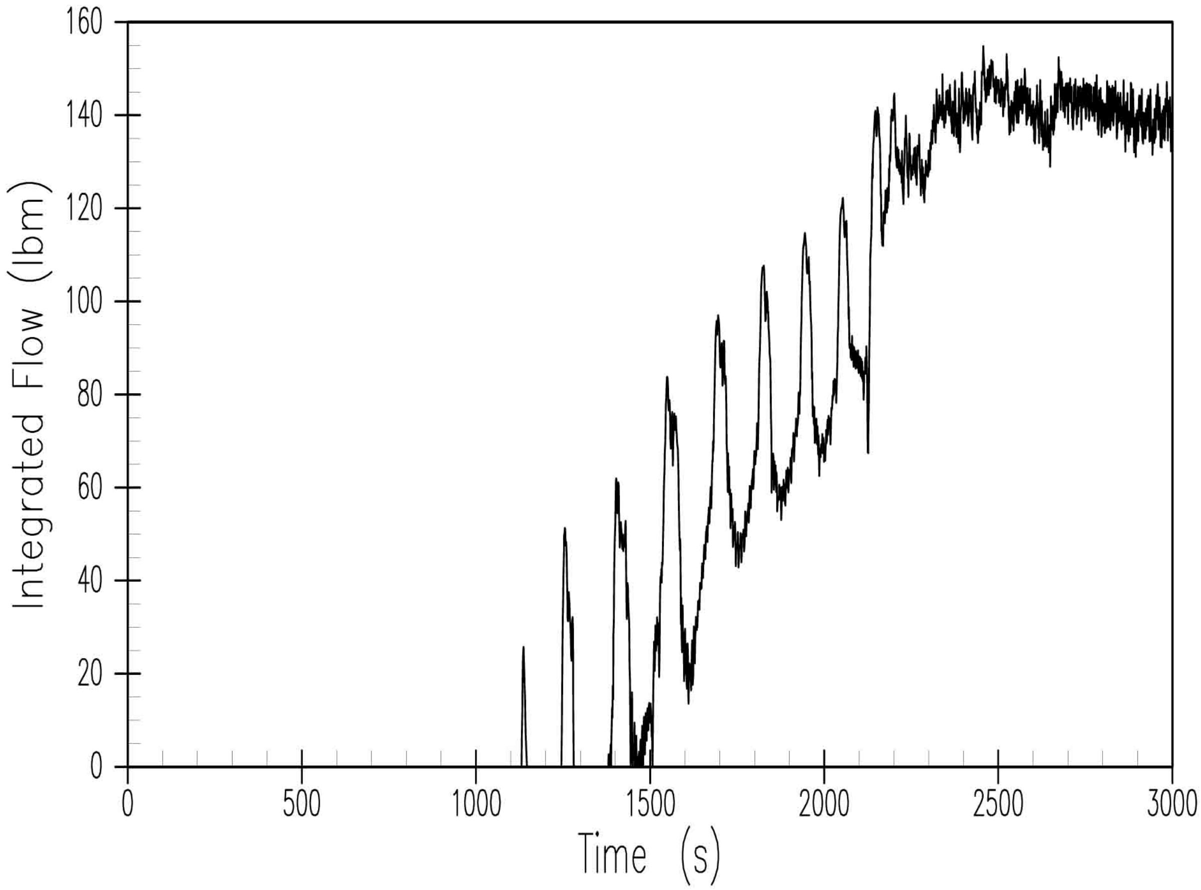

Figure 15.6.5.4B-51
DEDVI – Intact IRWST Injection Rate – 20 psi

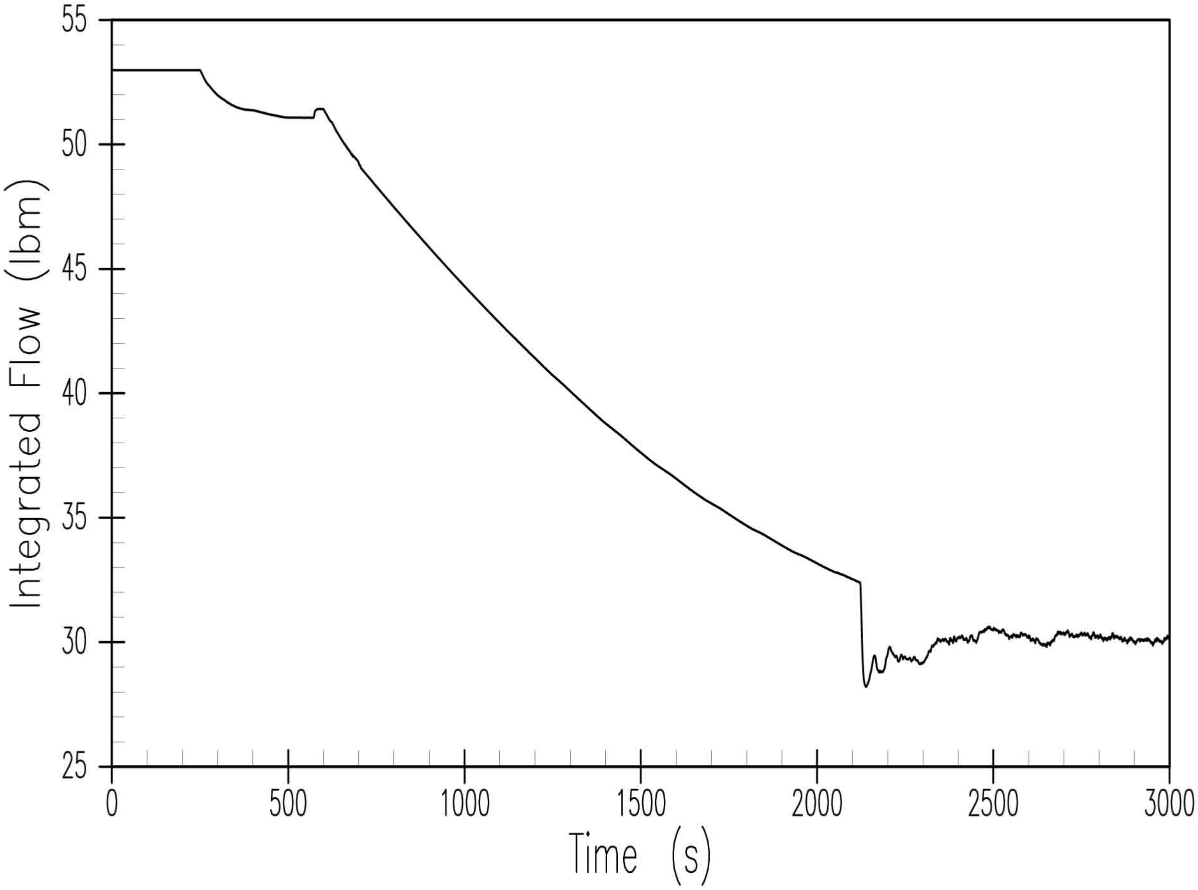

Figure 15.6.5.4B-52
DEDVI – Intact CMT Mixture Level – 20 psi

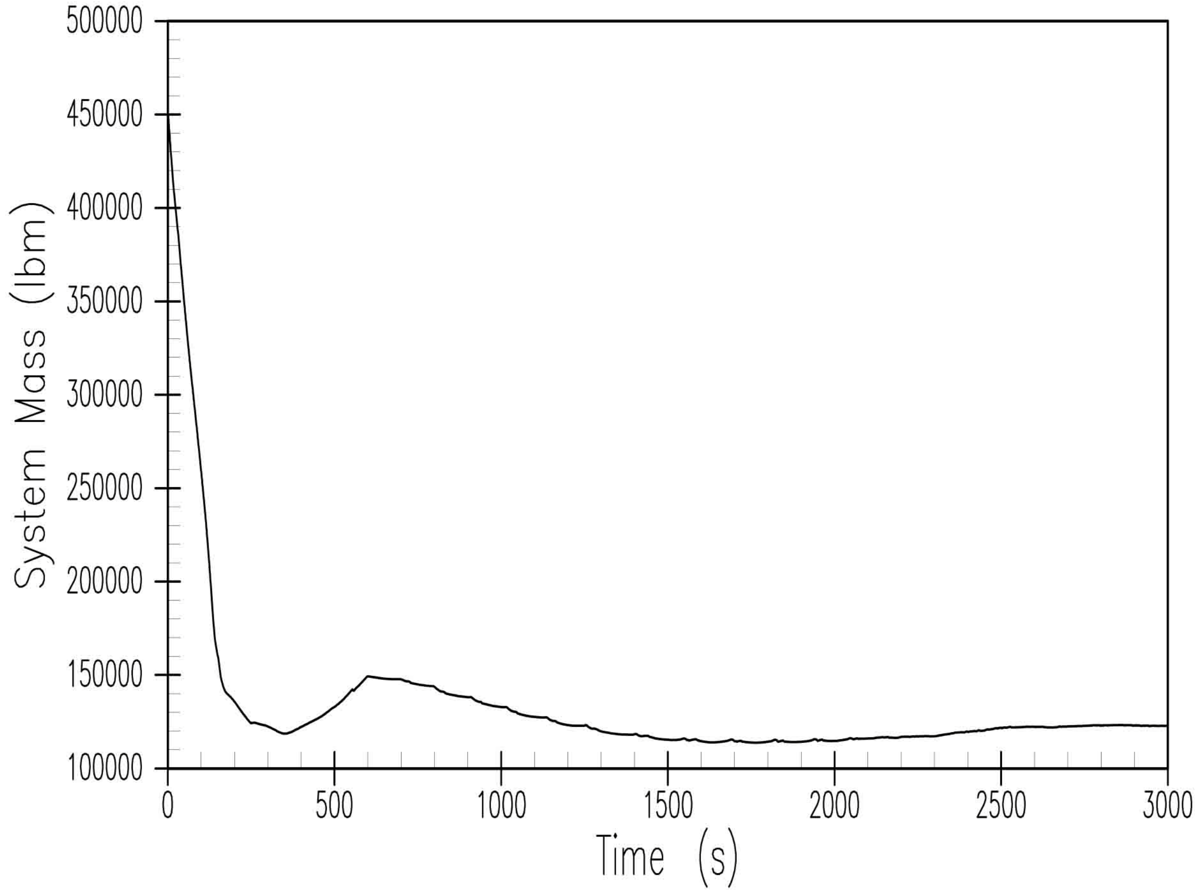

Figure 15.6.5.4B-53
DEDVI – RCS System Inventory – 20 psi

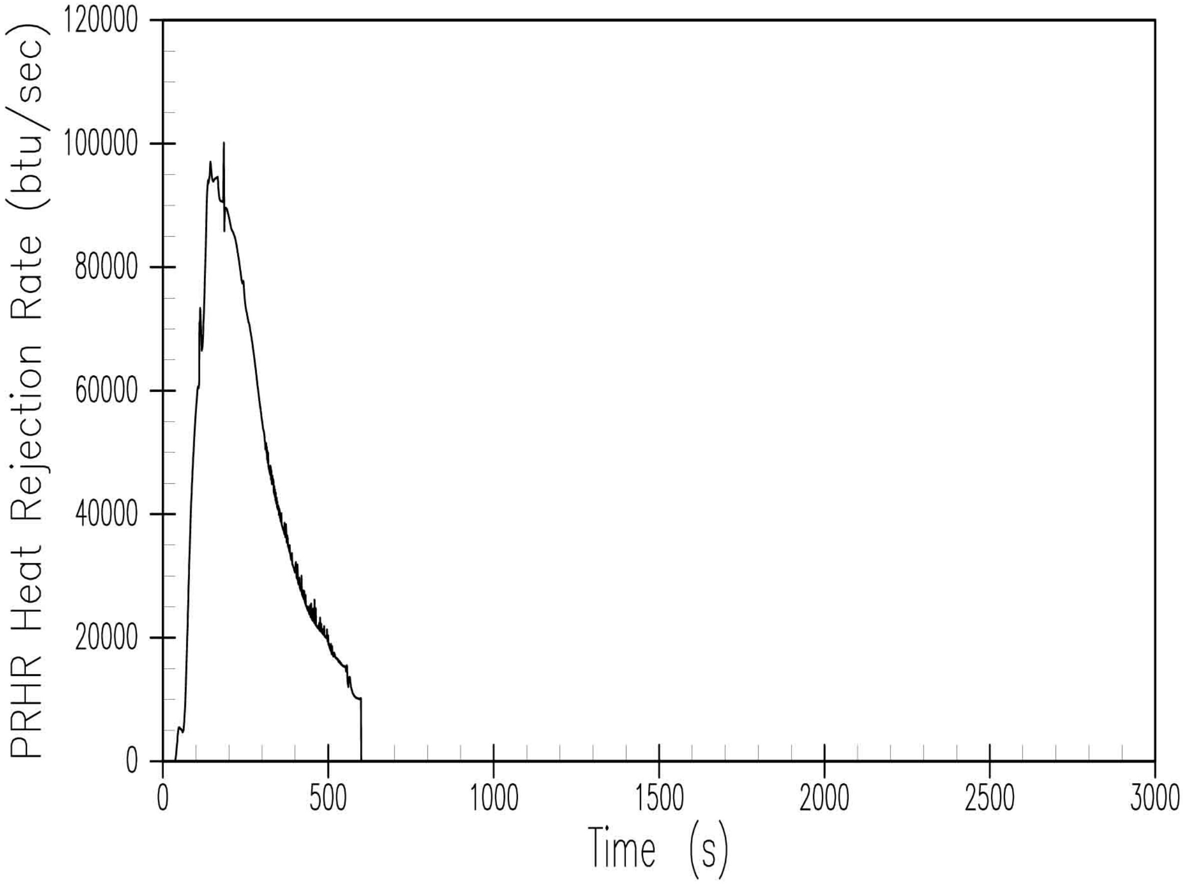

Figure 15.6.5.4B-54
DEDVI – PRHR Heat Removal Rate – 20 psi

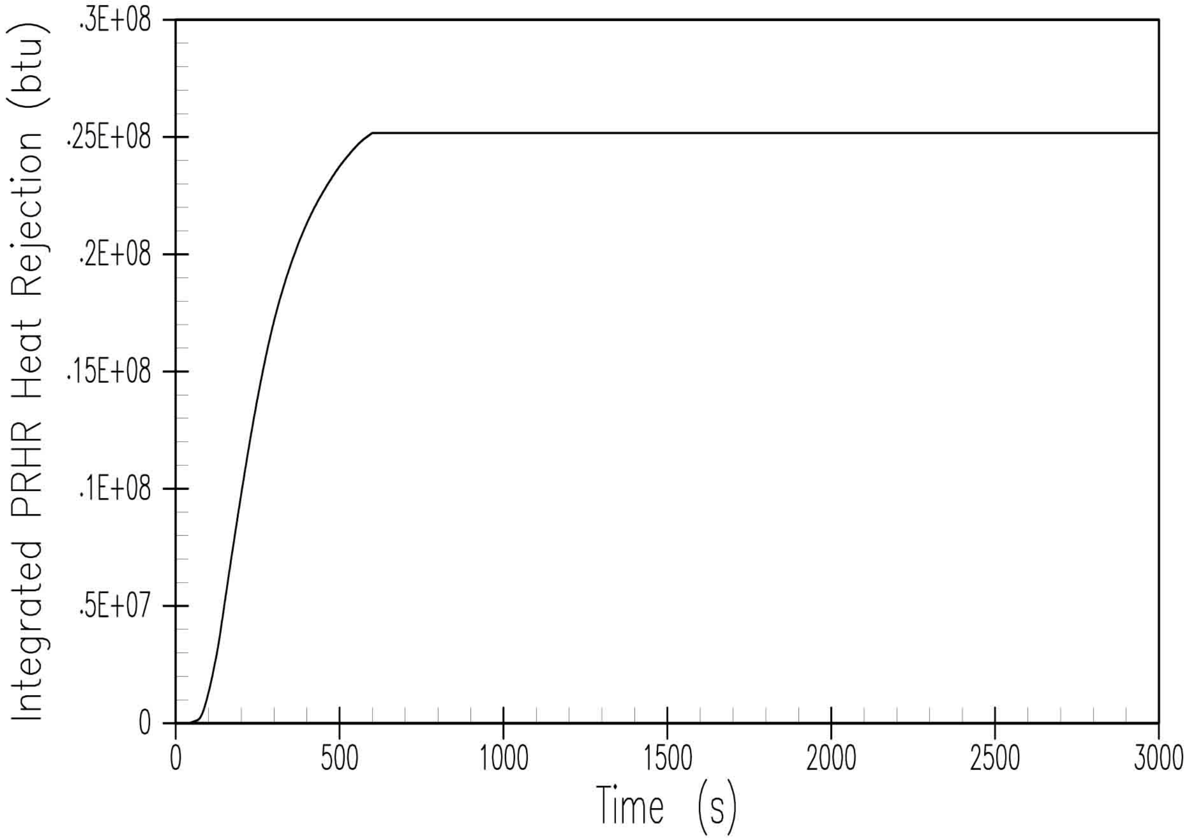

Figure 15.6.5.4B-55
DEDVI – Integrated PRHR Heat Removal – 20 psi

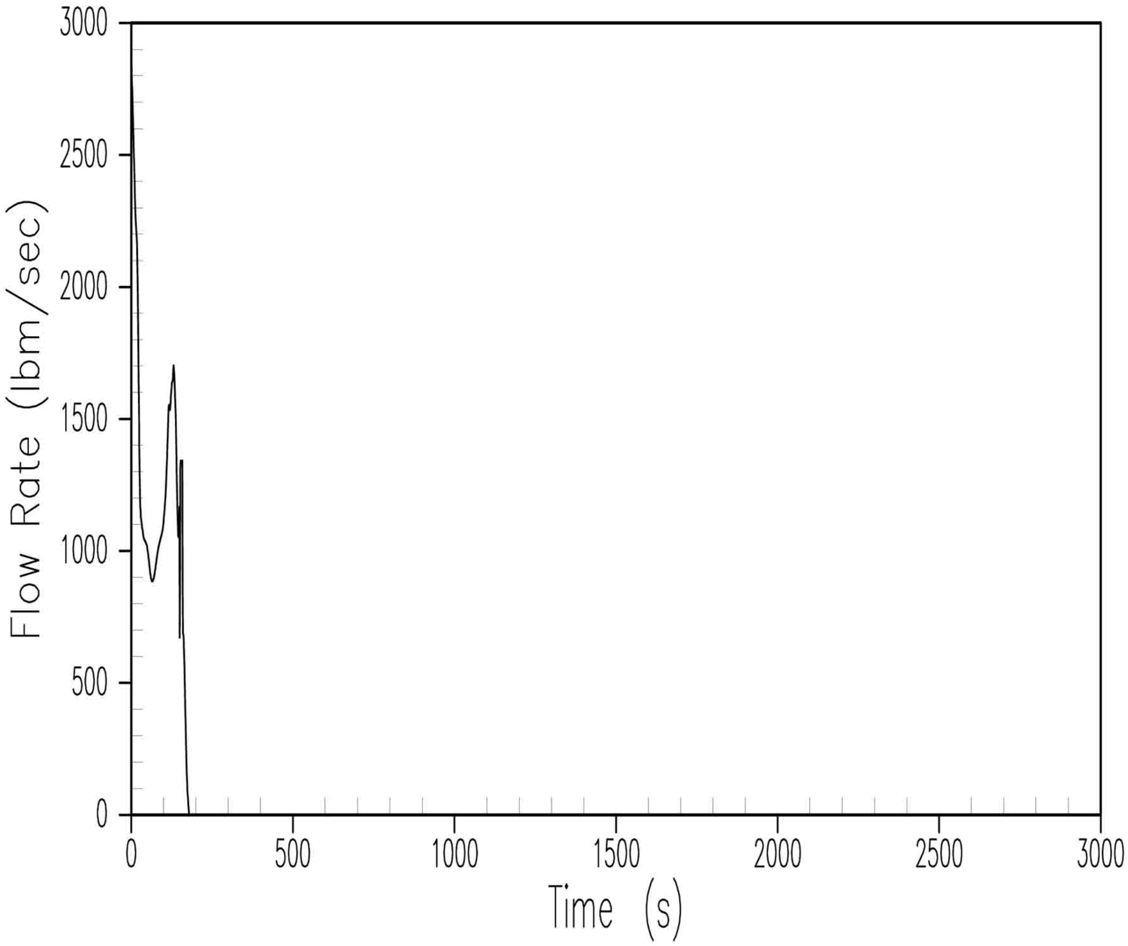

Figure 15.6.5.4B-36A
DEDVI – Vessel Side Liquid Break Discharge – 14.7 psi

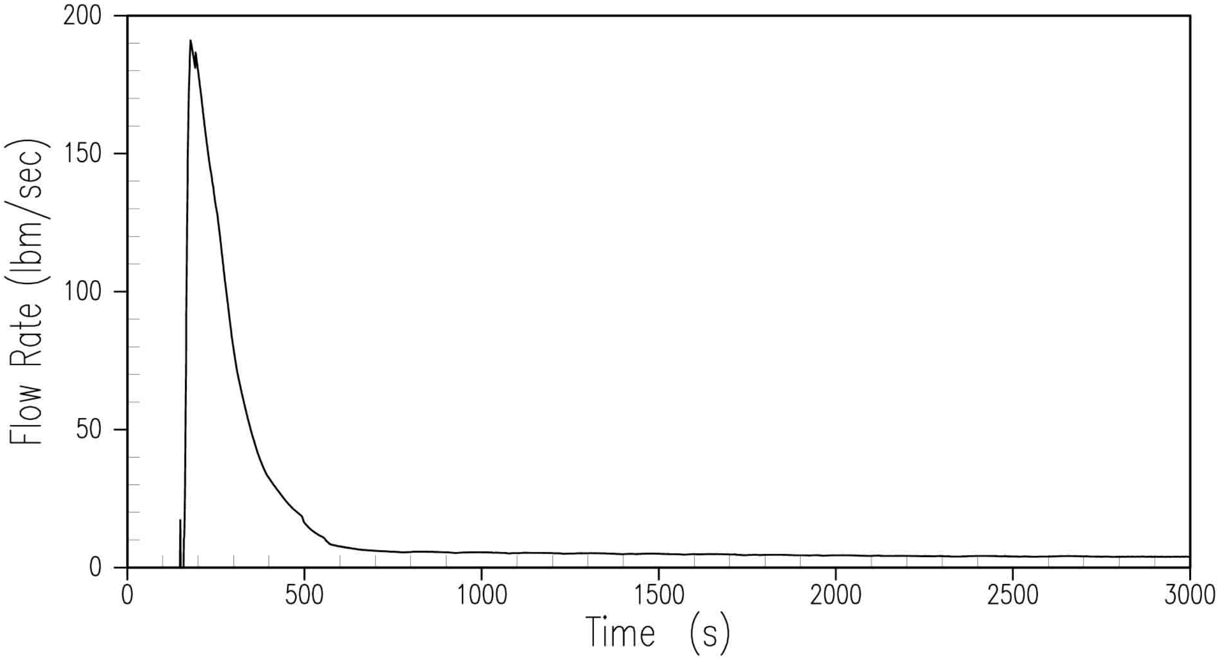

Figure 15.6.5.4B-37A
DEDVI – Vessel Side Vapor Break Discharge – 14.7 psi

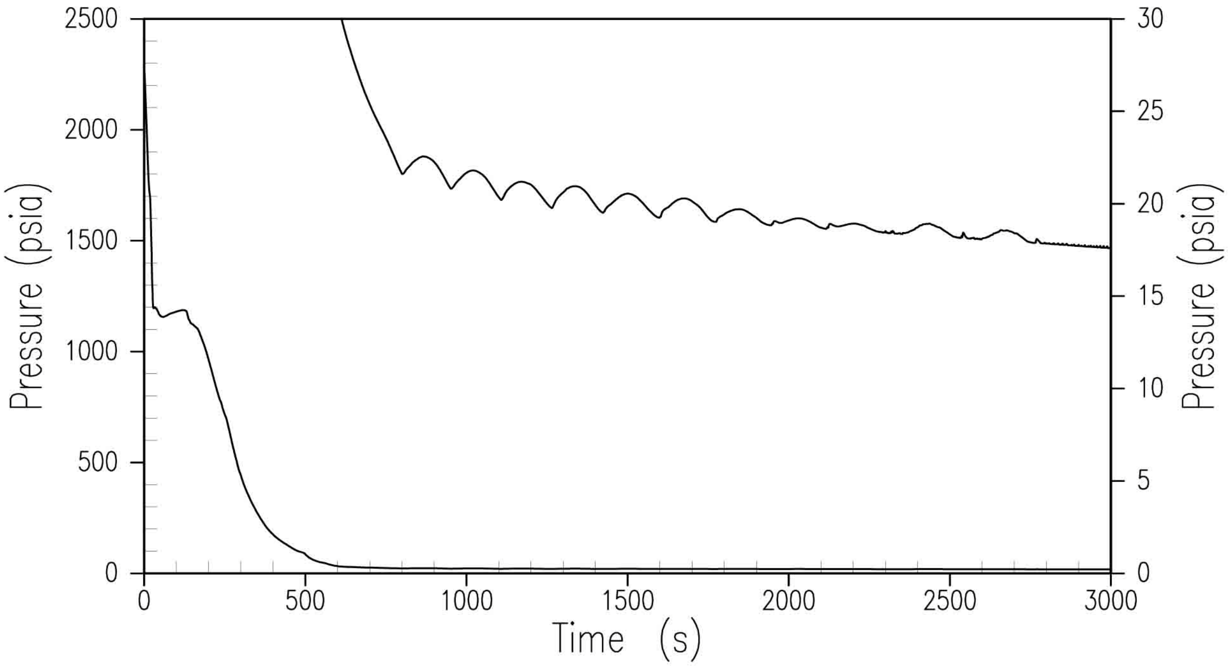

Figure 15.6.5.4B-38A
DEDVI – RCS Pressure – 14.7 psi

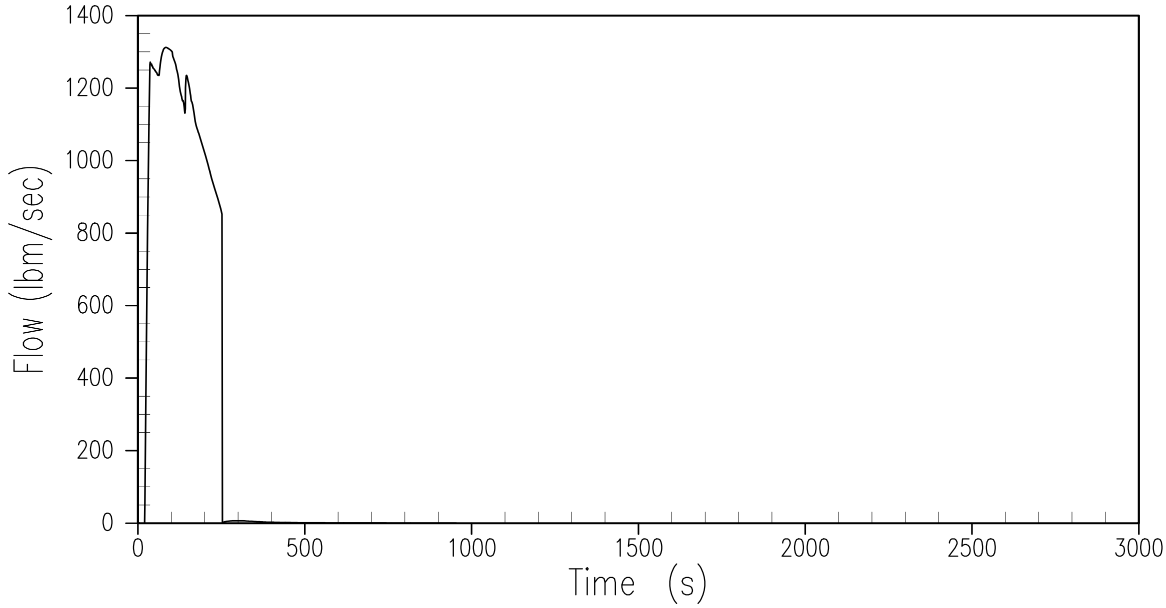

Figure 15.6.5.4B-39A
DEDVI – Broken CMT Injection Rate – 14.7 psi

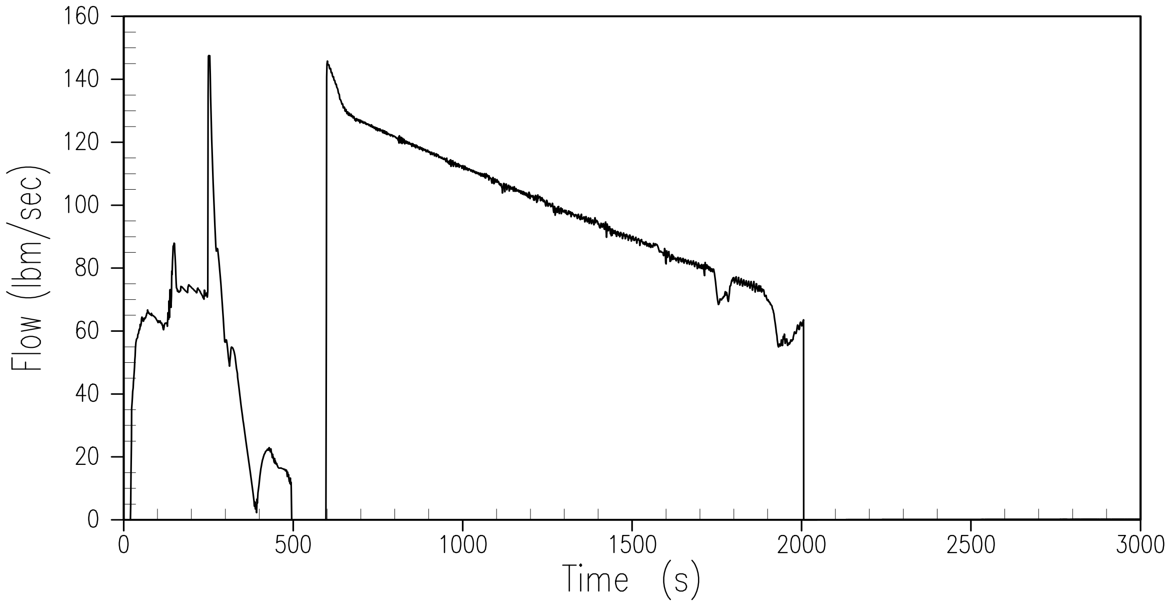

Figure 15.6.5.4B-40A
DEDVI – Intact CMT Injection Rate – 14.7 psi

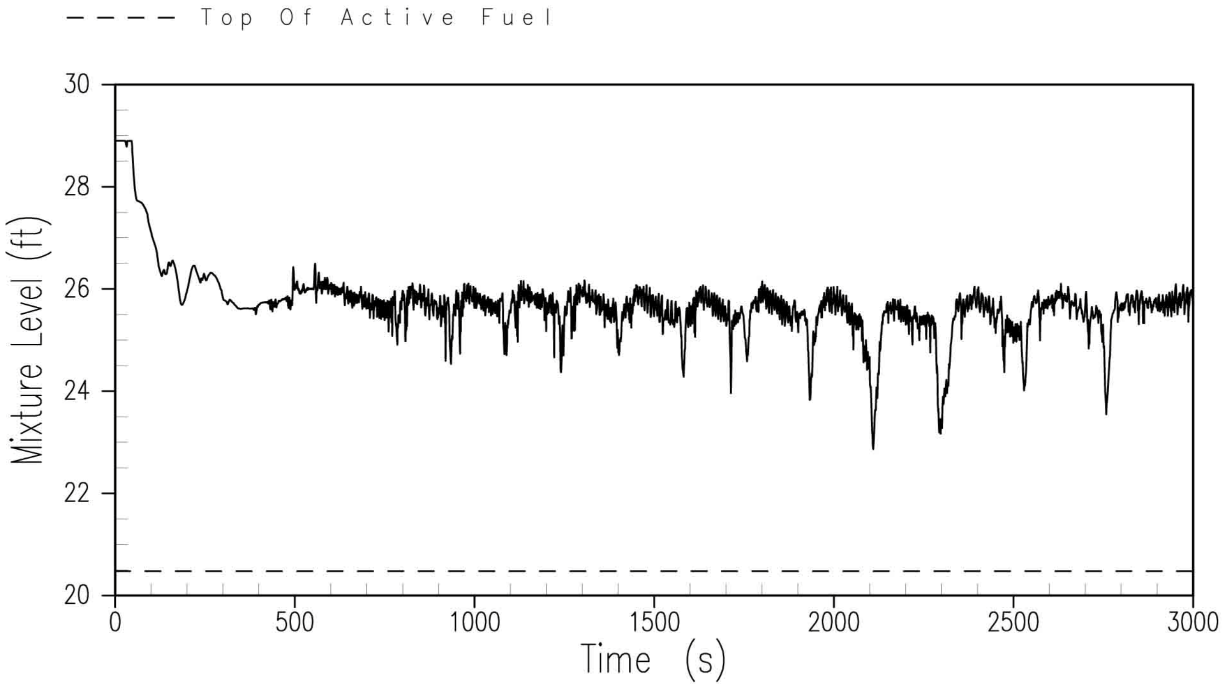

Figure 15.6.5.4B-41A
DEDVI – Core/Upper Plenum Mixture Level – 14.7 psi

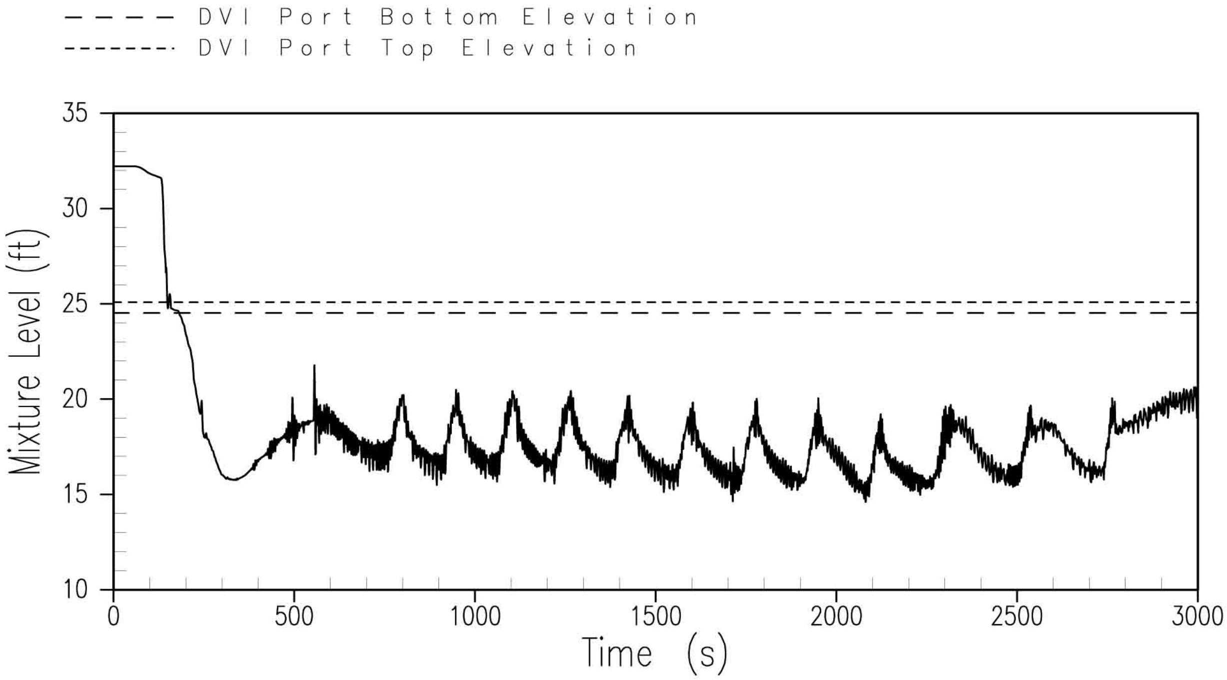

Figure 15.6.5.4B-42A
DEDVI – Downcomer Mixture Level – 14.7 psi

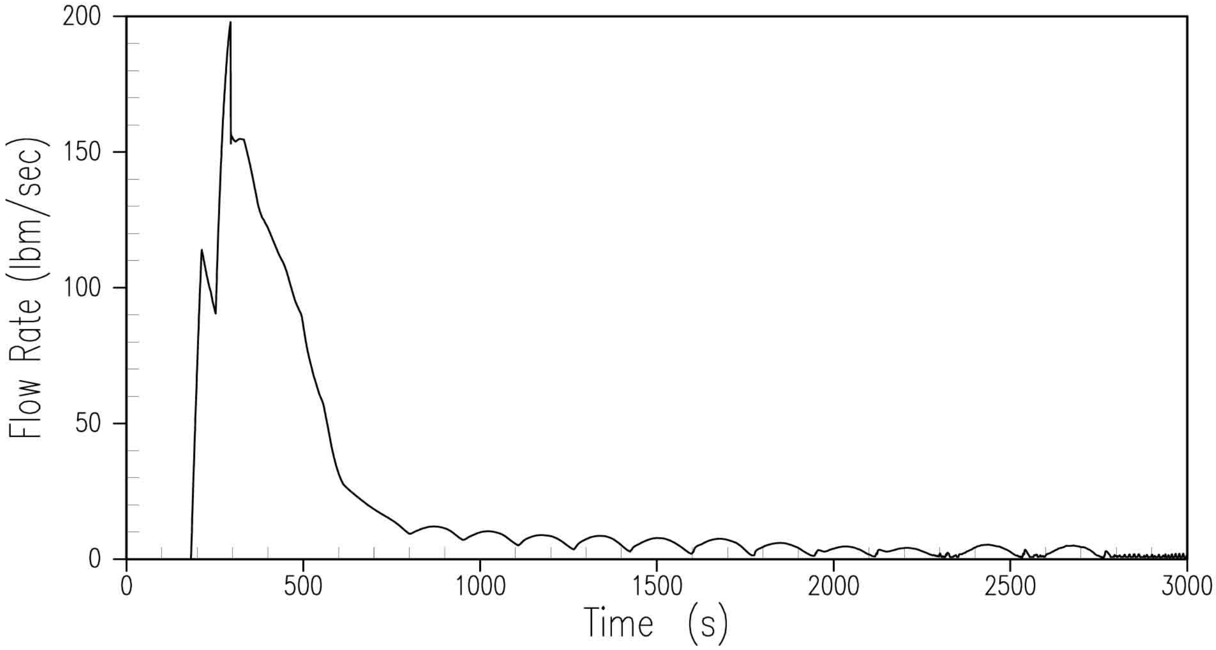

Figure 15.6.5.4B-43A
DEDVI – ADS 1-3 Vapor Discharge – 14.7 psi

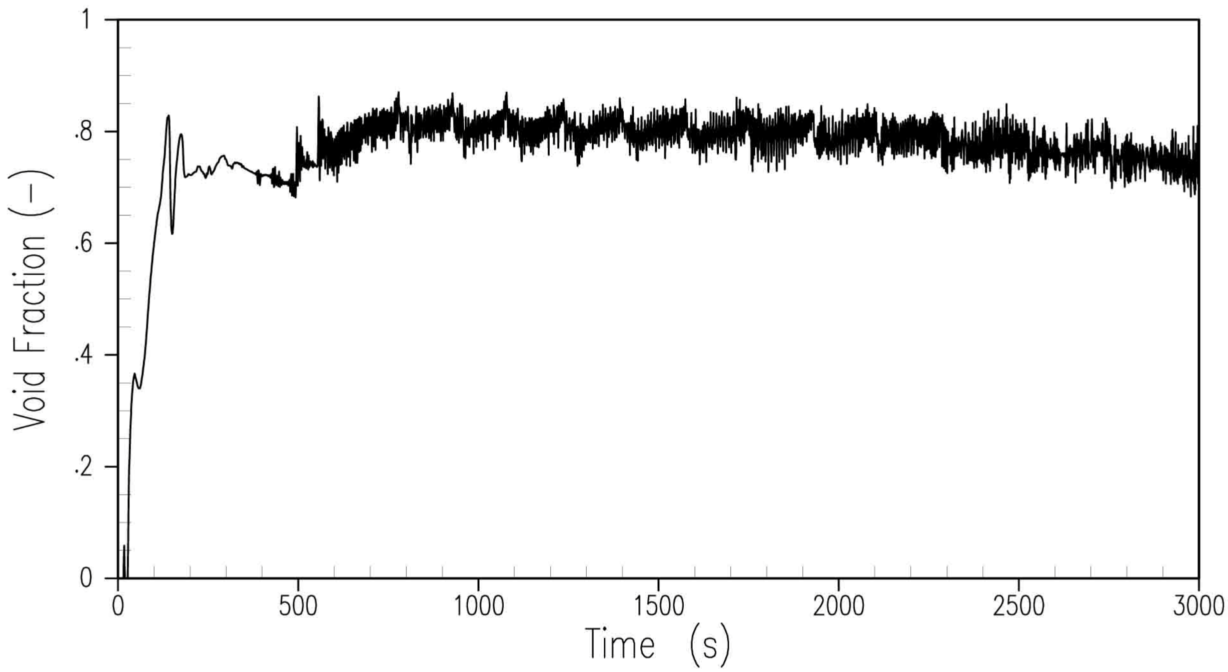

Figure 15.6.5.4B-44A
DEDVI – Core Exit Void Fraction – 14.7 psi

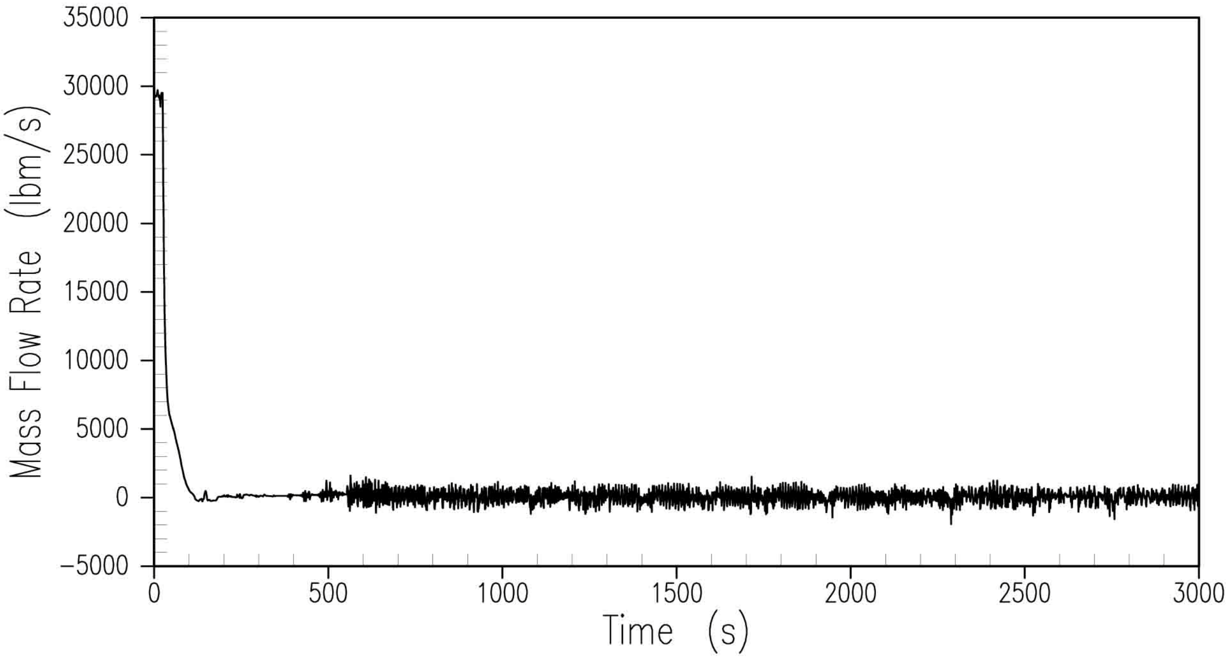

Figure 15.6.5.4B-45A
DEDVI – Core Exit Liquid Flow Rate – 14.7 psi

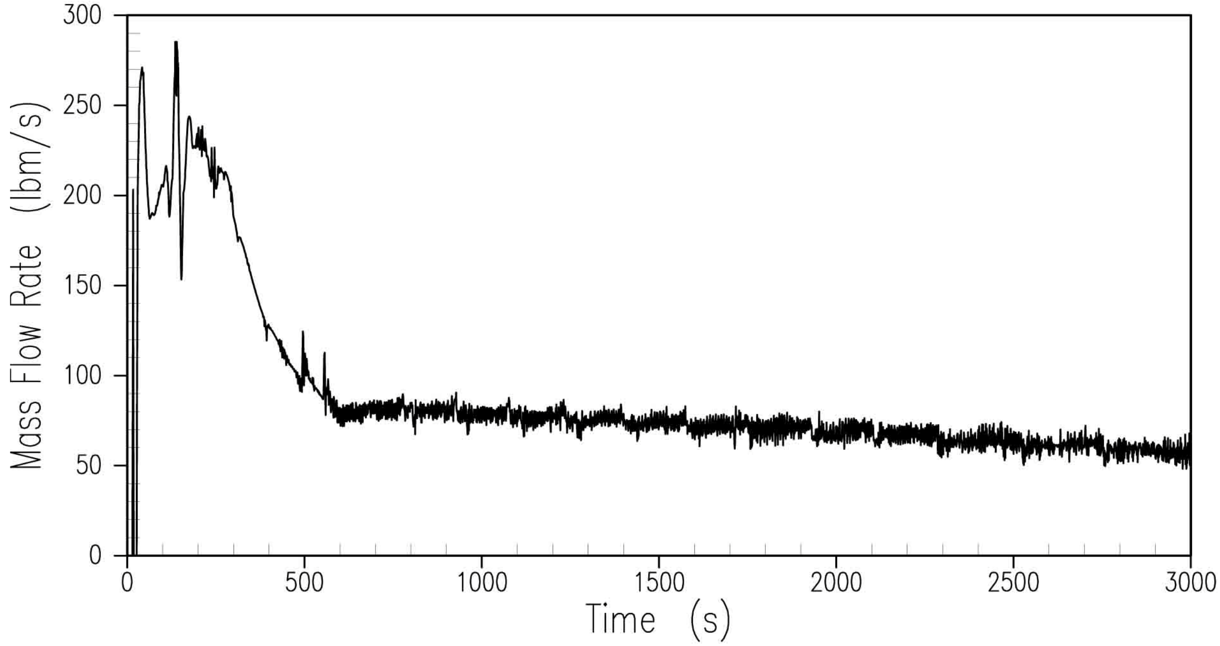

Figure 15.6.5.4B-46A
DEDVI – Core Exit Vapor Flow Rate – 14.7 psi

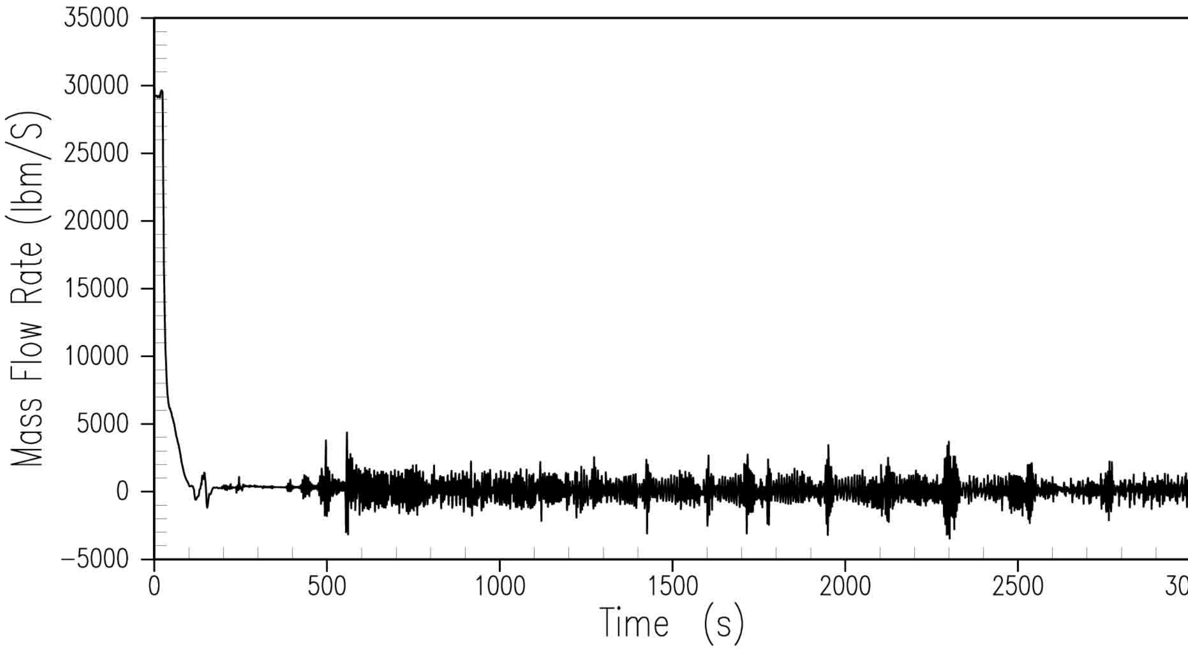

Figure 15.6.5.4B-47A
DEDVI – Lower Plenum to Core Flow Rate – 14.7 psi

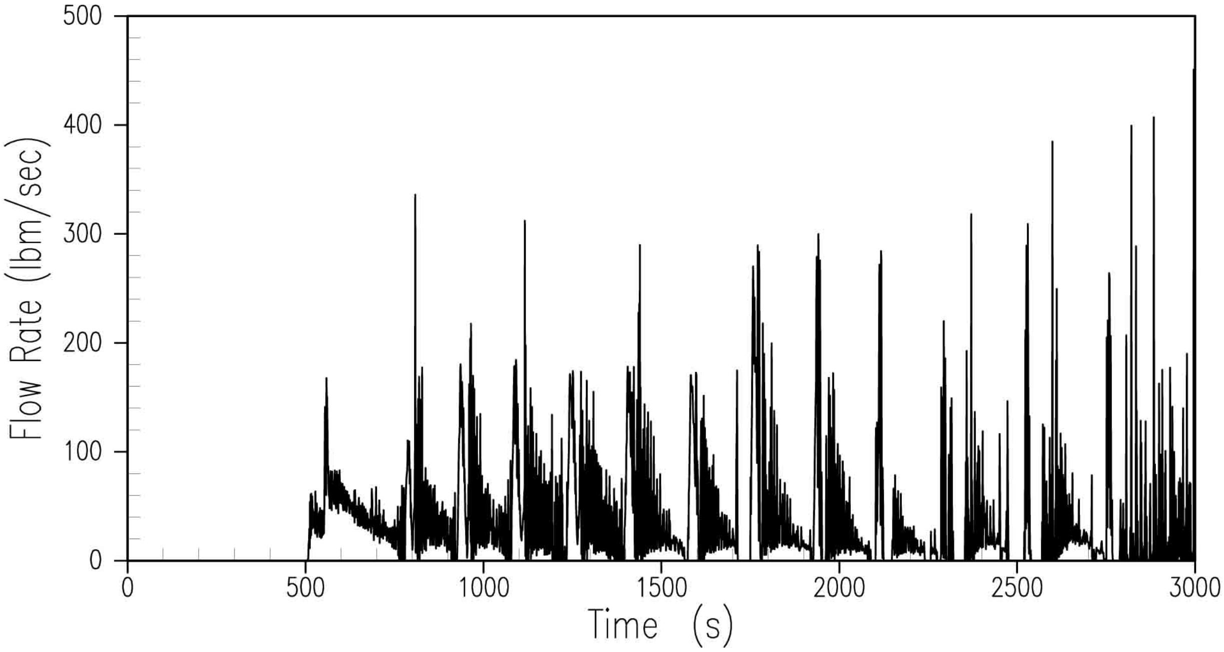

Figure 15.6.5.4B-48A
DEDVI – ADS-4 Liquid Discharge – 14.7 psi

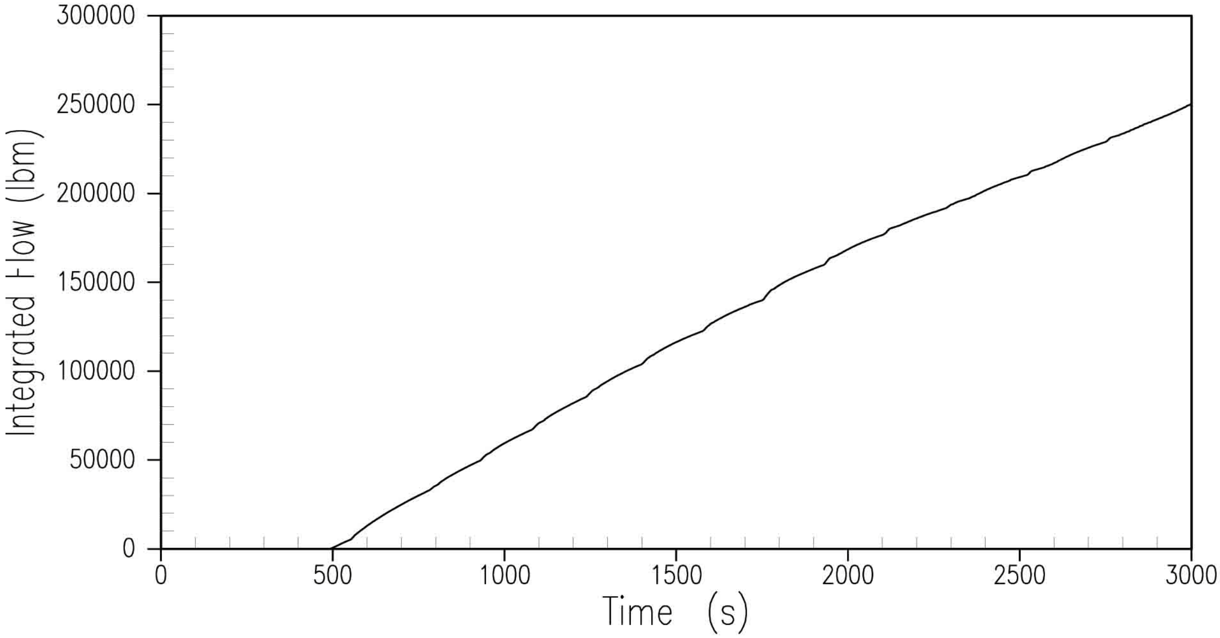

Figure 15.6.5.4B-49A
DEDVI – ADS-4 Integrated Discharge – 14.7 psi

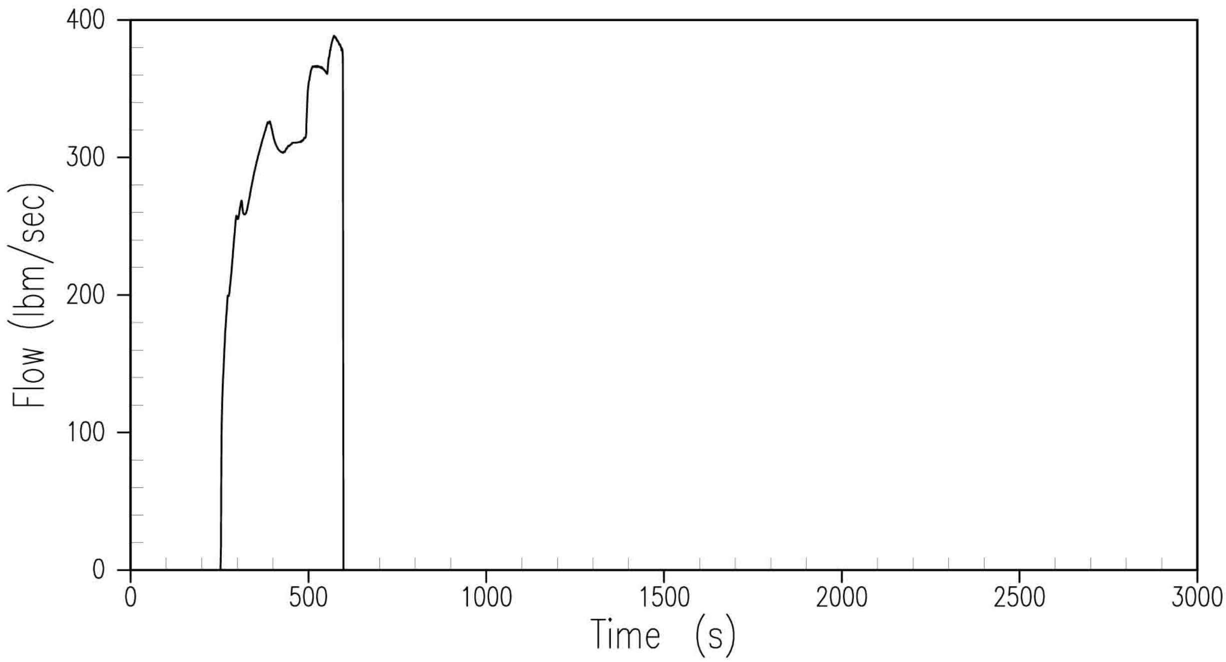

Figure 15.6.5.4B-50A
DEDVI – Intact Accumulator Flow Rate – 14.7 psi

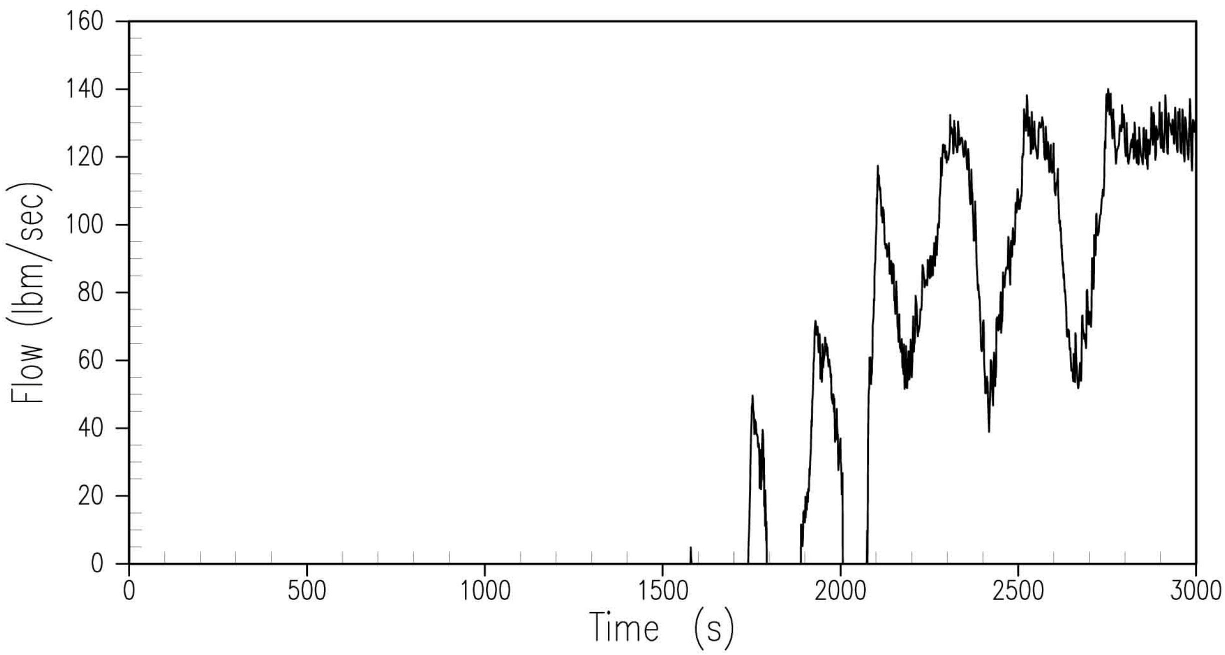

Figure 15.6.5.4B-51A
DEDVI – Intact IRWST Injection Rate – 14.7 psi

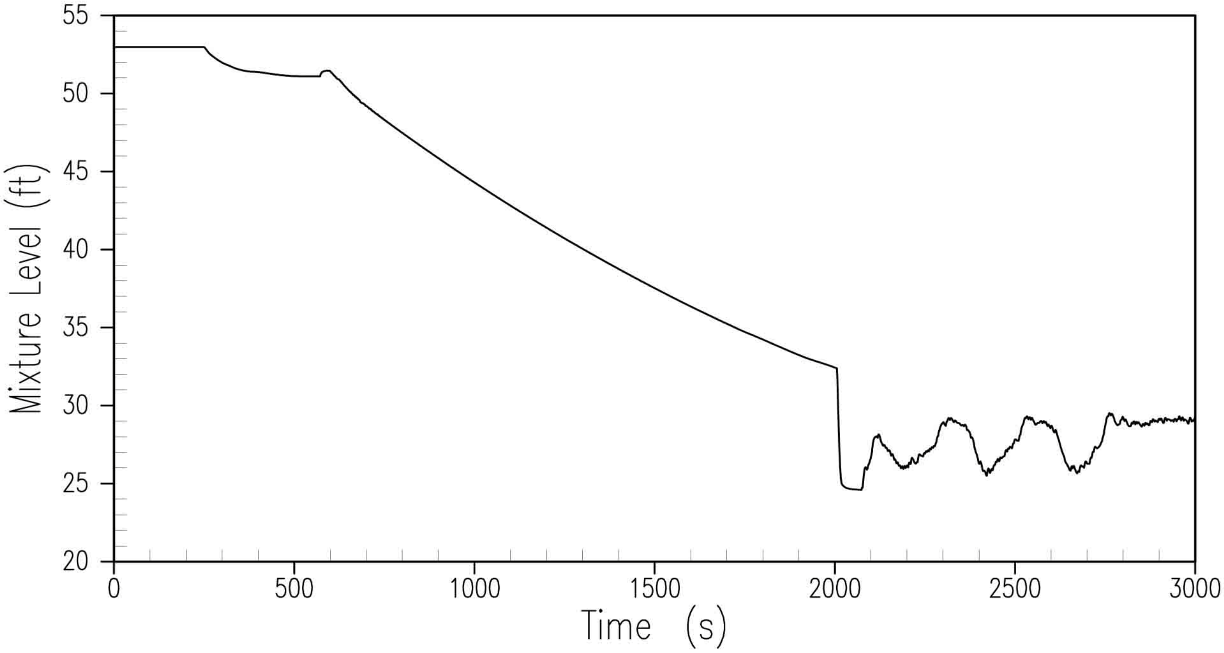

Figure 15.6.5.4B-52A
DEDVI – Intact CMT Mixture Level – 14.7 psi

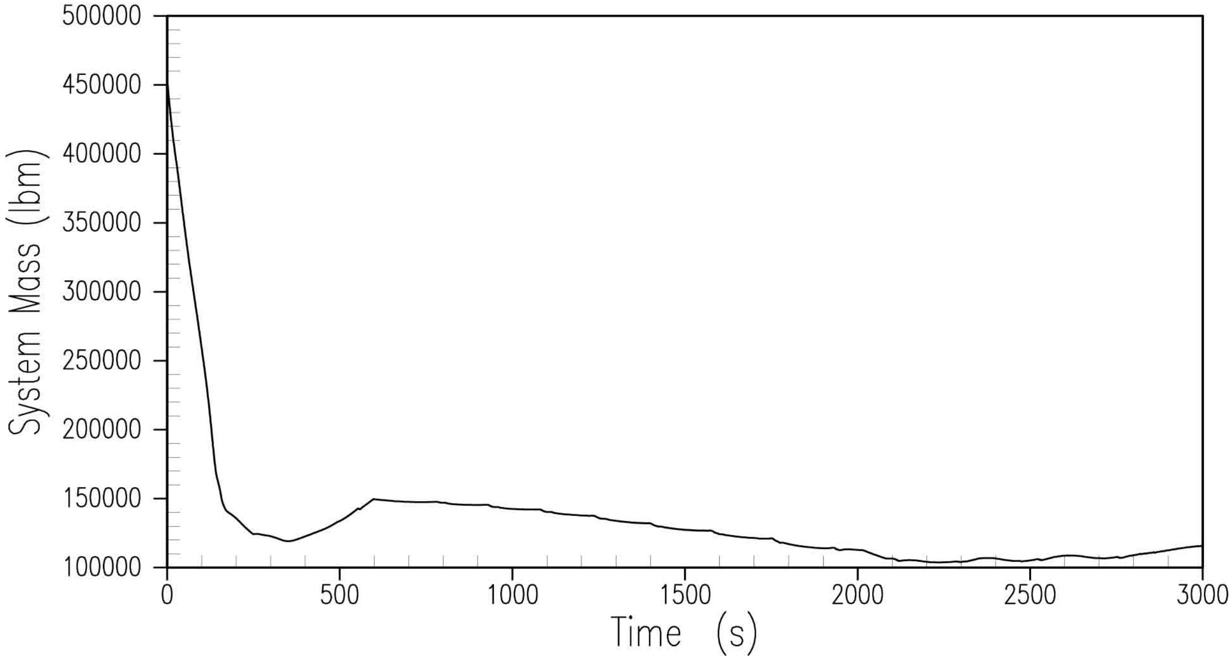

Figure 15.6.5.4B-53A
DEDVI – RCS System Inventory – 14.7 psi

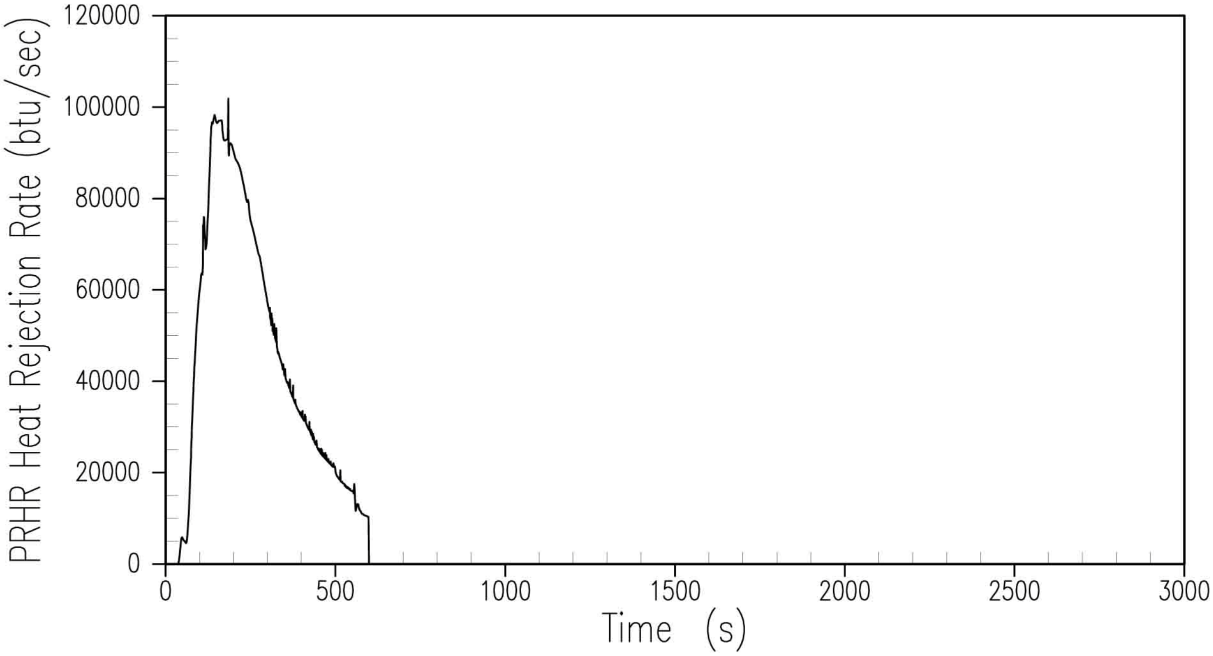

Figure 15.6.5.4B-54A
DEDVI – PRHR Heat Removal Rate – 14.7 psi

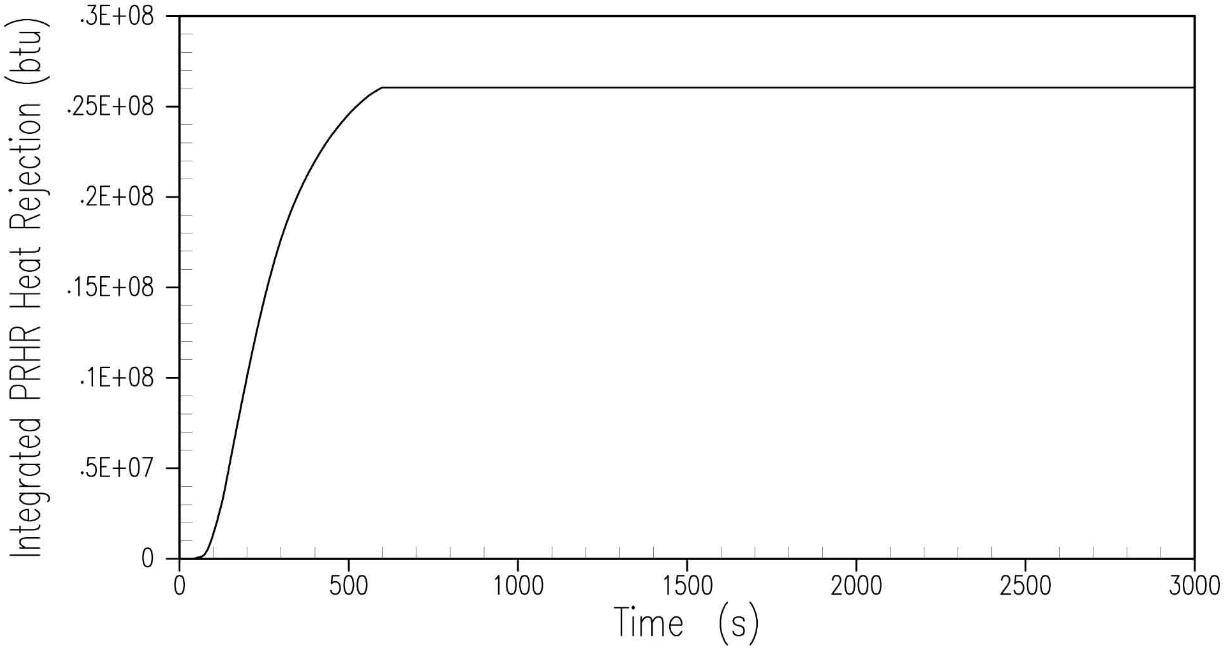

Figure 15.6.5.4B-55A
DEDVI – Integrated PRHR Heat Removal – 14.7 psi

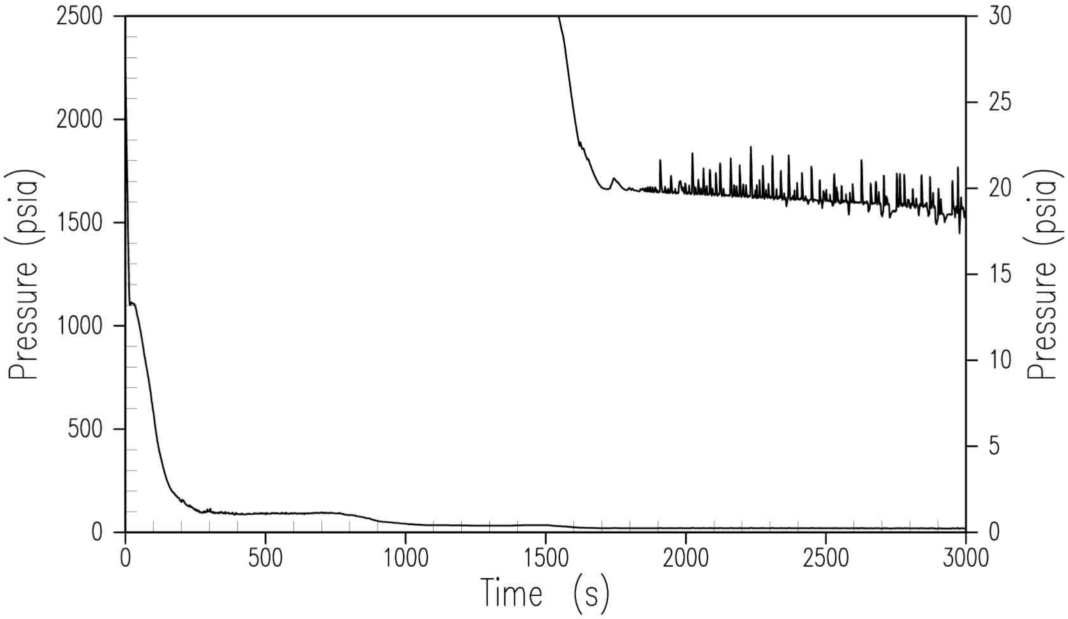

Figure 15.6.5.4B-56
10-inch Cold Leg Break – RCS Pressure

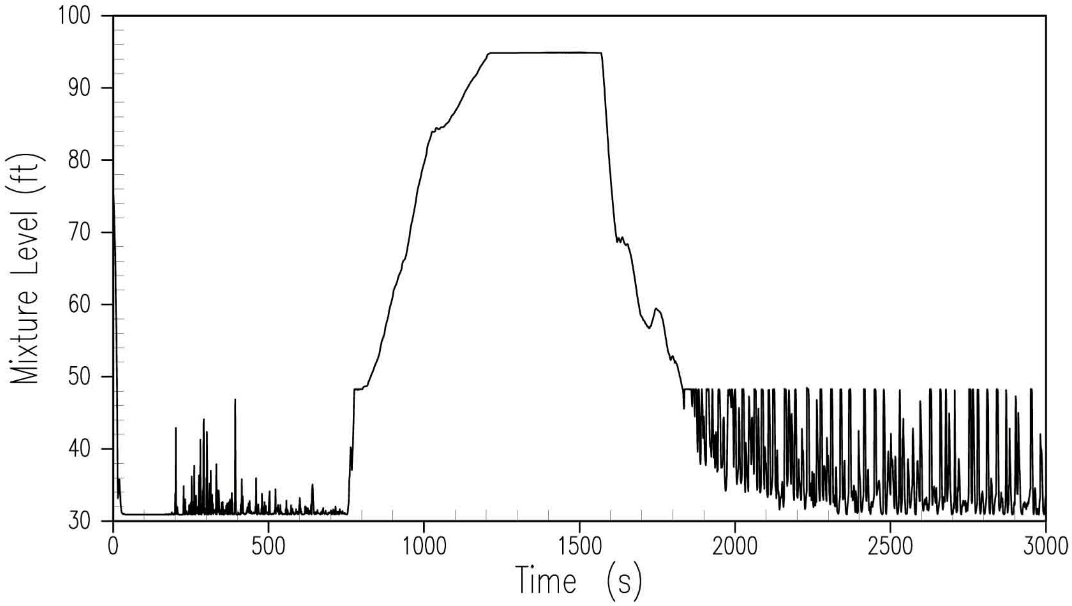

Figure 15.6.5.4B-57
10-inch Cold Leg Break – Pressurizer Mixture Level

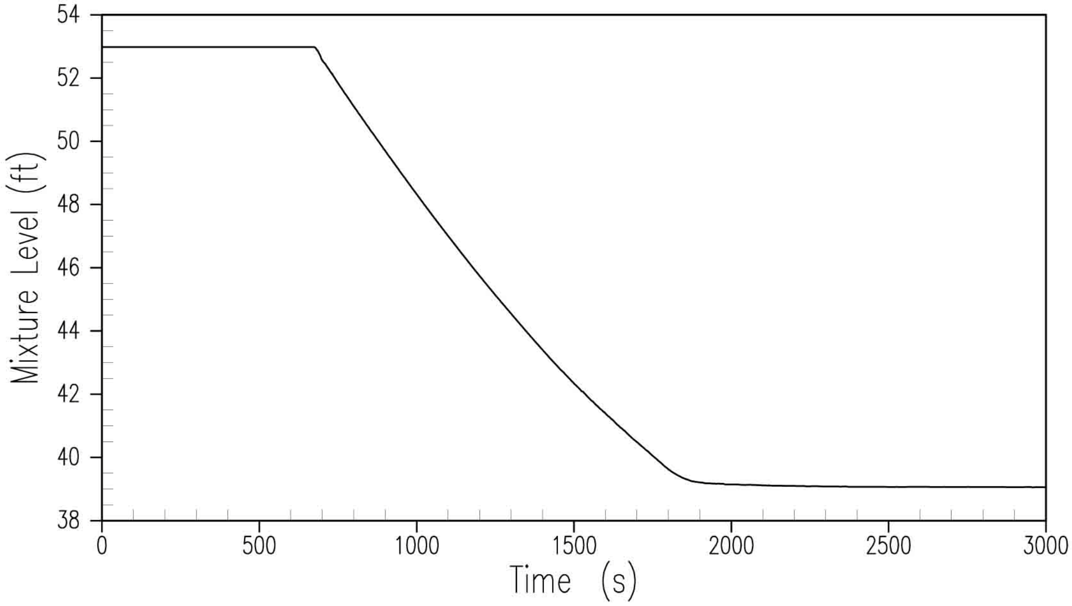

Figure 15.6.5.4B-58
10-inch Cold Leg Break – CMT-1 Mixture Level

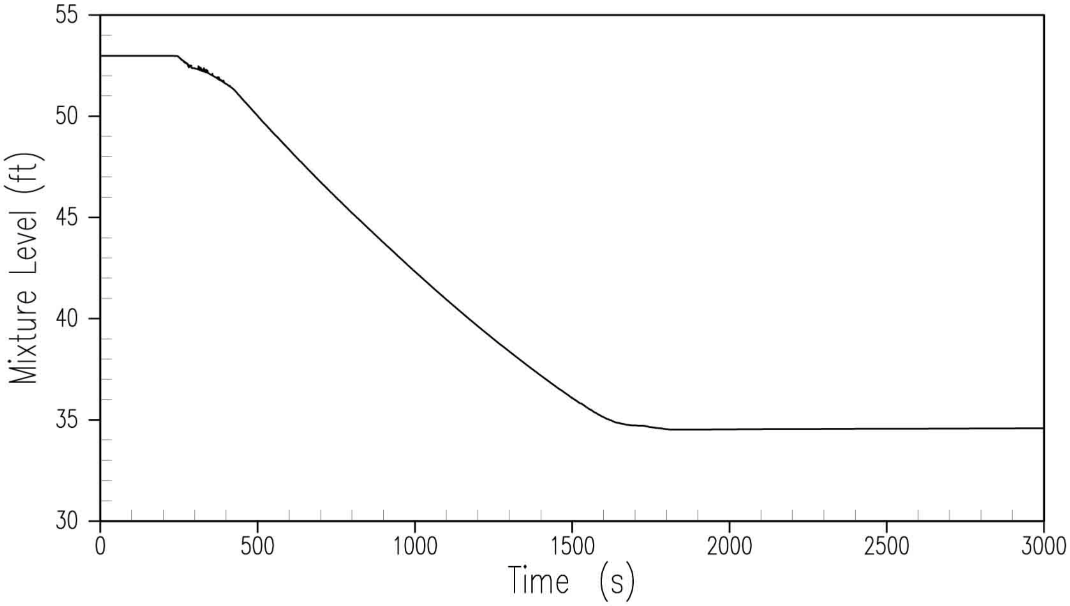

Figure 15.6.5.4B-59
10-inch Cold Leg Break – CMT-2 Mixture Level

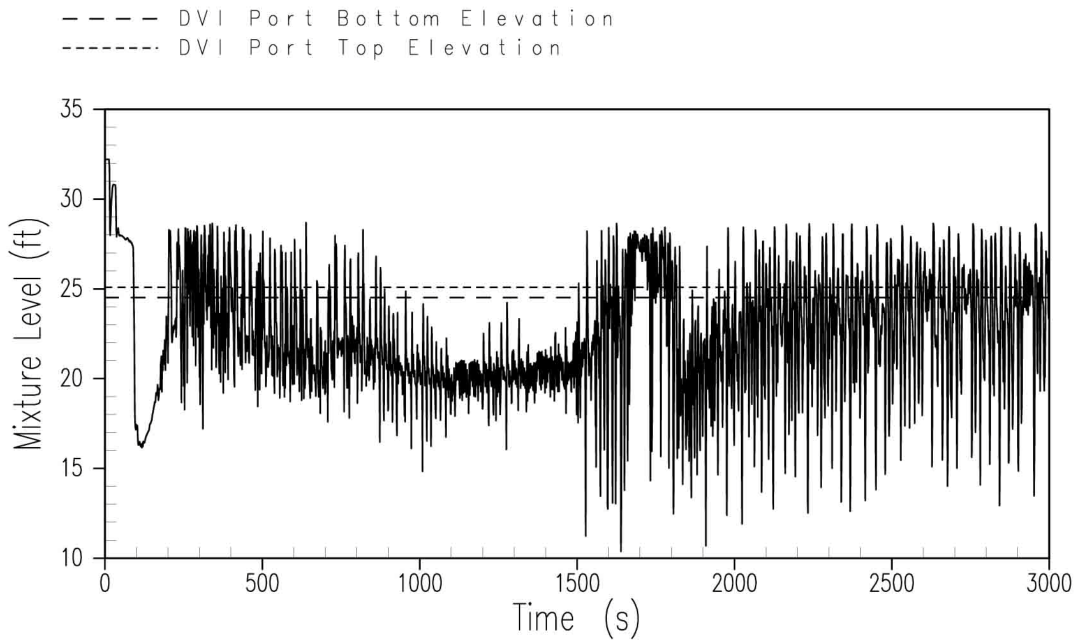

Figure 15.6.5.4B-60
10-inch Cold Leg Break – Downcomer Mixture Level

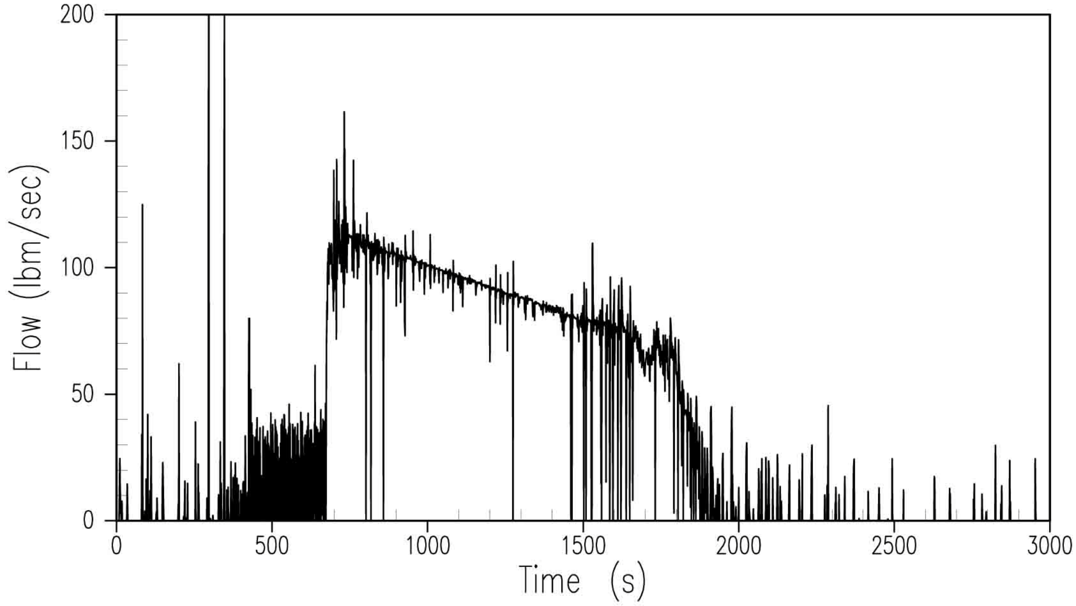

Figure 15.6.5.4B-61
10-inch Cold Leg Break – CMT-1 Injection Rate

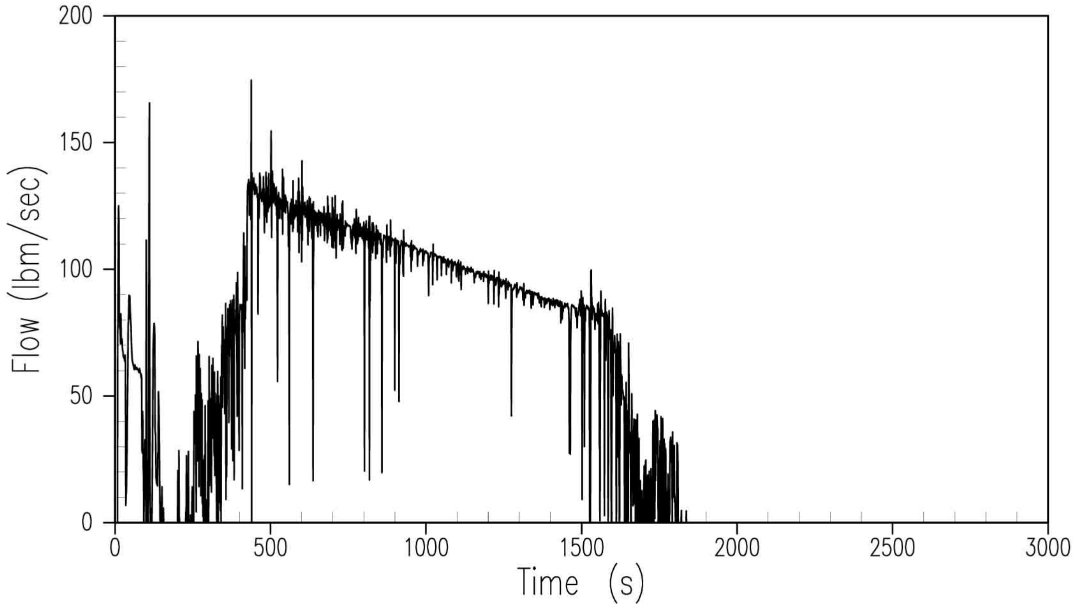

Figure 15.6.5.4B-62
10-inch Cold Leg Break – CMT-2 Injection Rate

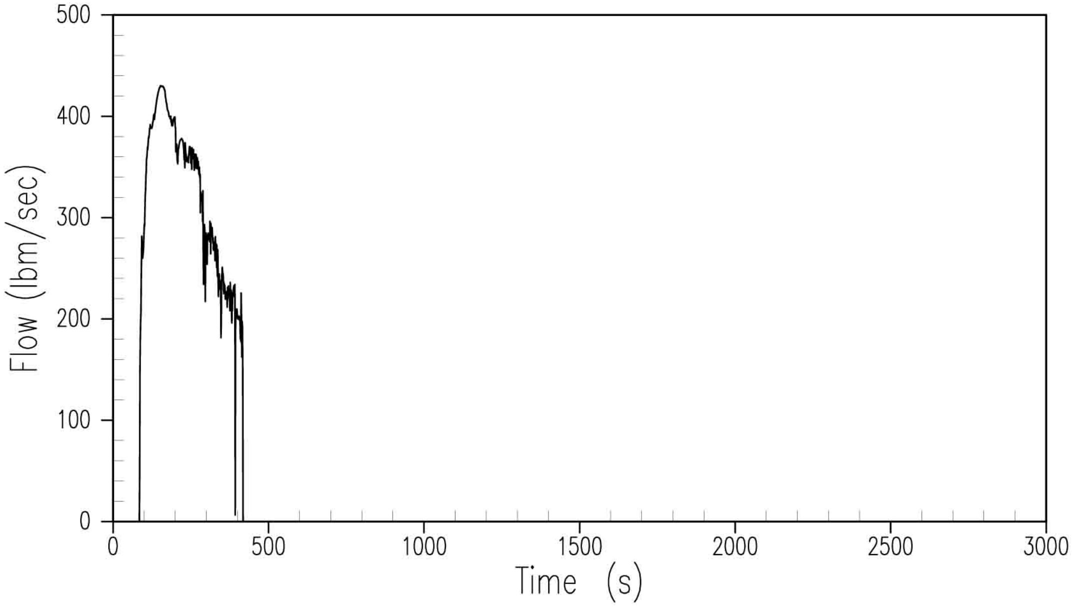

Figure 15.6.5.4B-63
10-inch Cold Leg Break – Accumulator-1 Injection Rate

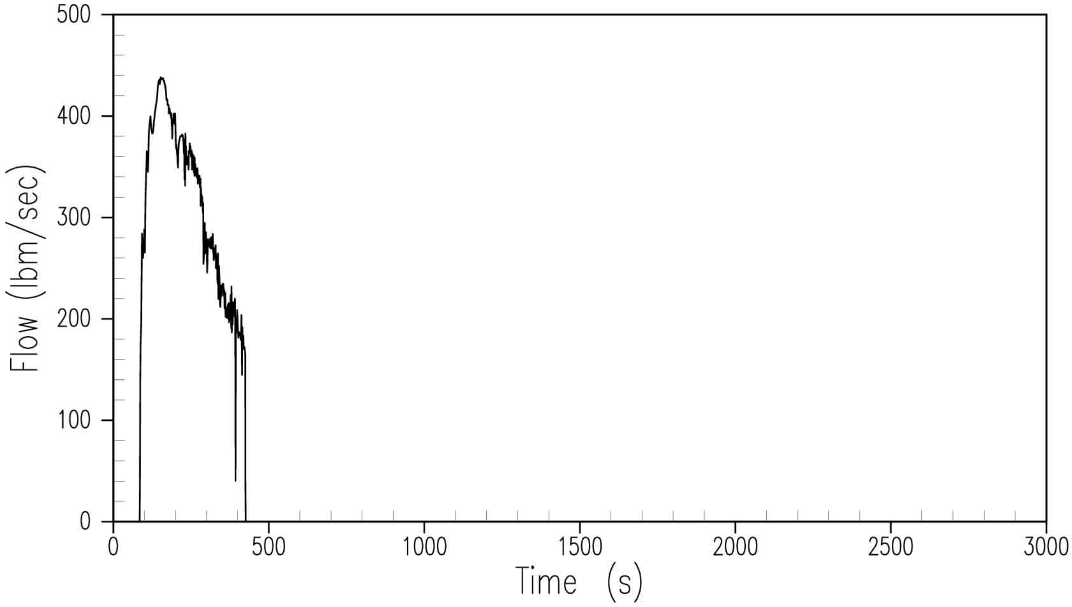

Figure 15.6.5.4B-64
10-inch Cold Leg Break – Accumulator-2 Injection Rate

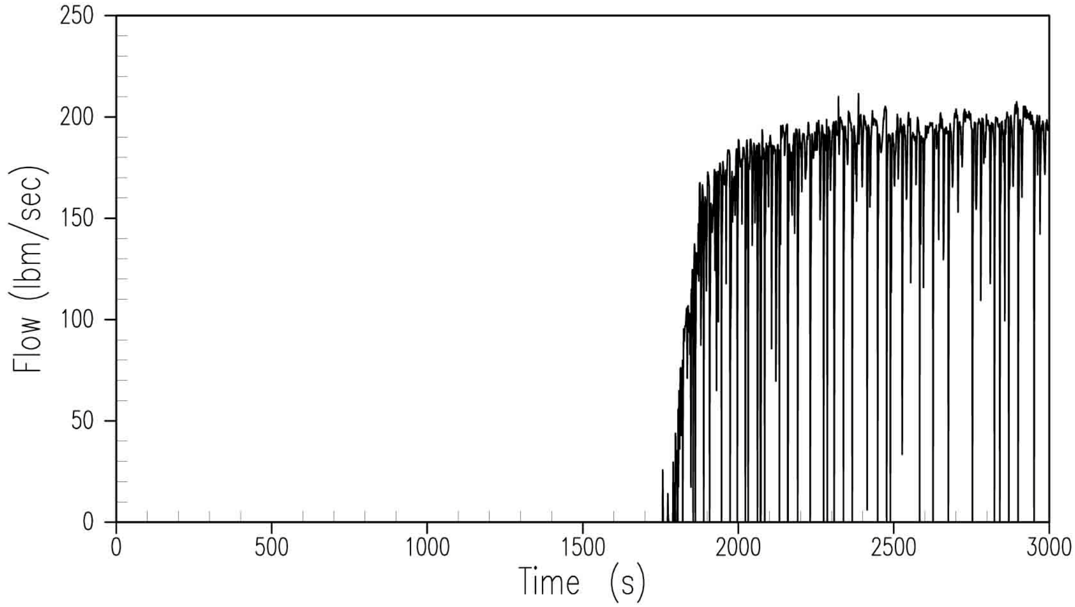

Figure 15.6.5.4B-65
10-inch Cold Leg Break – IRWST-1 Injection Rate

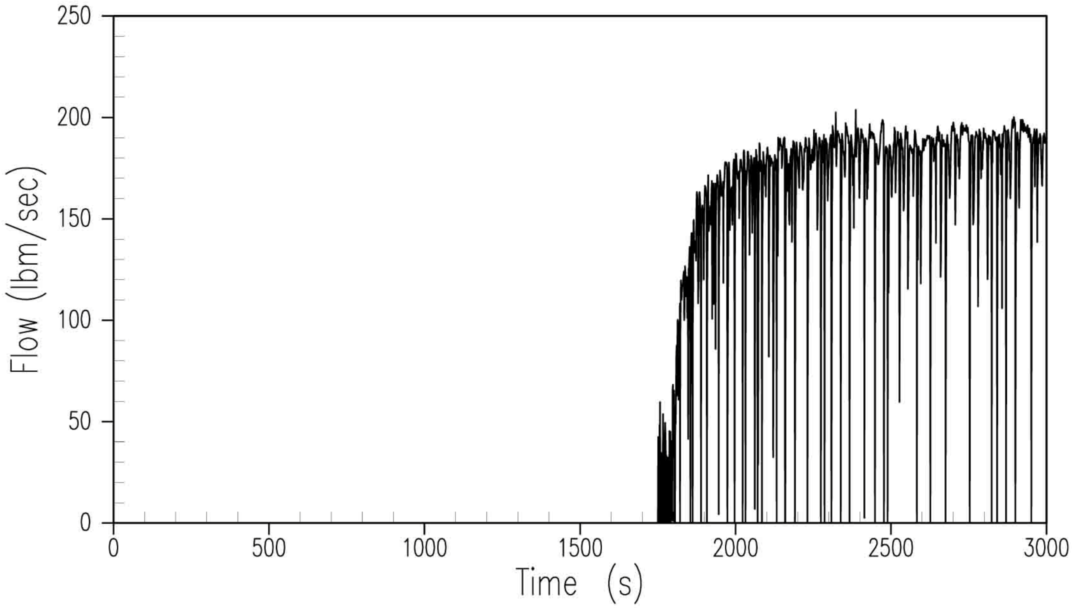

Figure 15.6.5.4B-66
10-inch Cold Leg Break – IRWST-2 Injection Rate

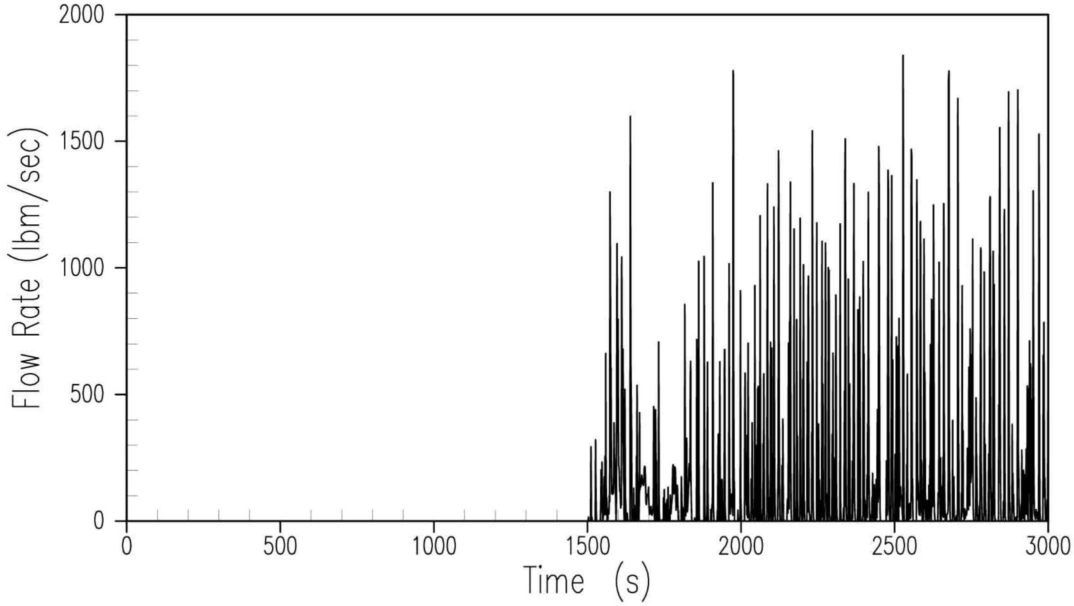

Figure 15.6.5.4B-67
10-inch Cold Leg Break – ADS-4 Liquid Discharge

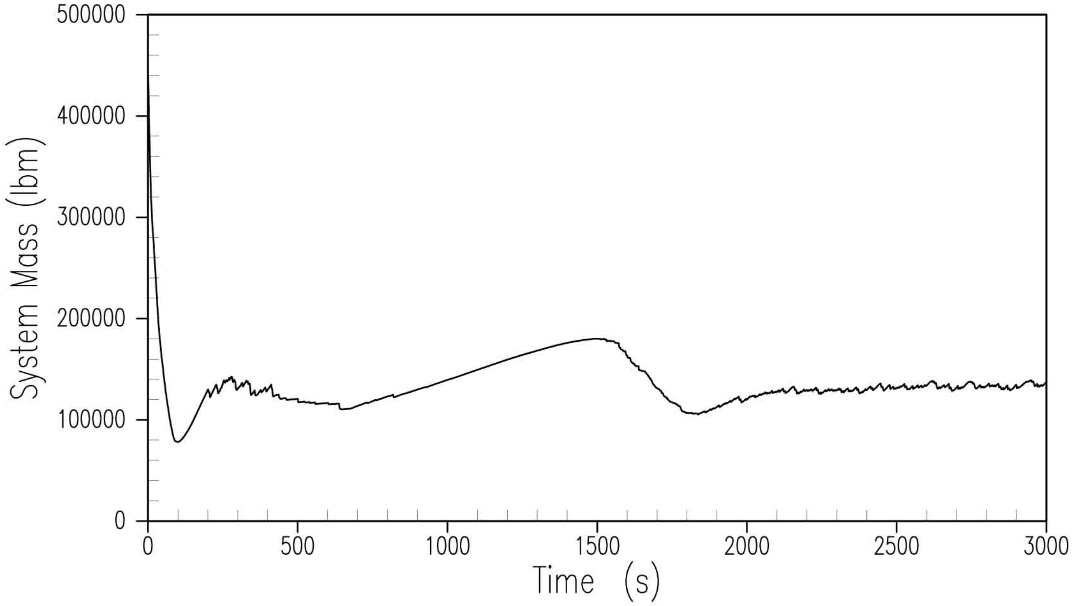

Figure 15.6.5.4B-68
10-inch Cold Leg Break – RCS System Inventory

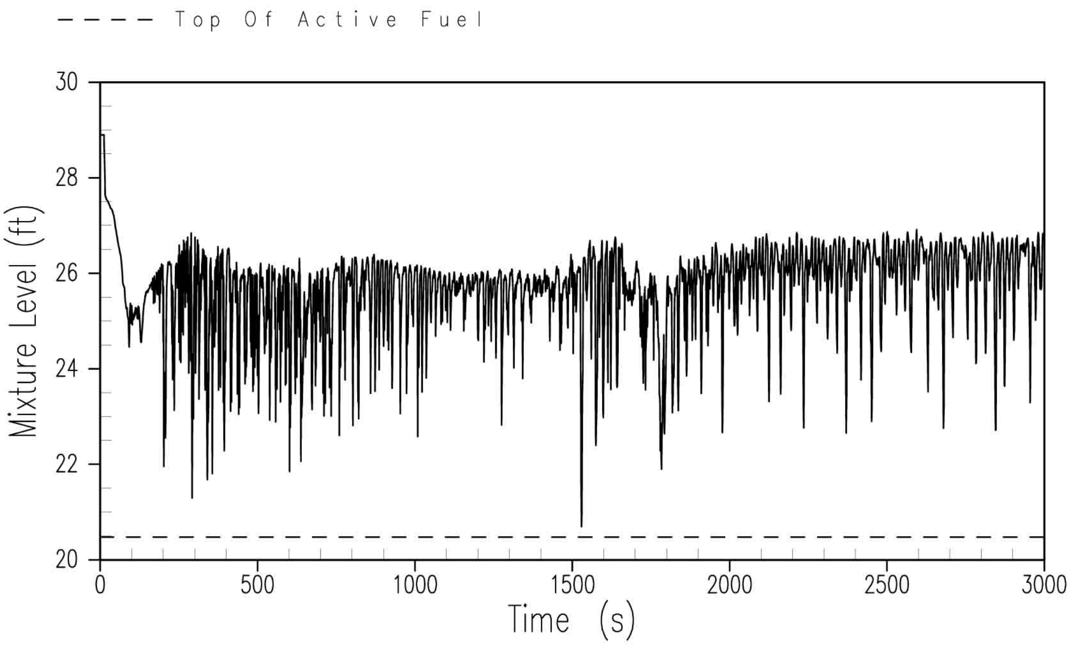

Figure 15.6.5.4B-69
10-inch Cold Leg Break – Core/Upper Plenum Mixture Level

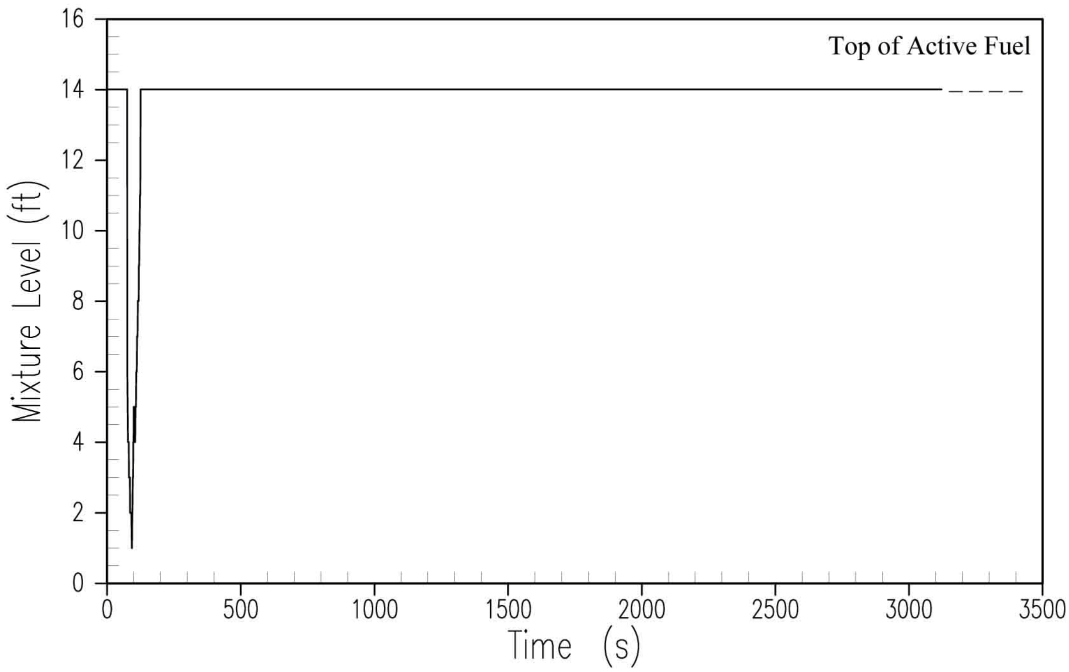

Figure 15.6.5.4B-70
10-inch Cold Leg Break – Composite Core Mixture Level

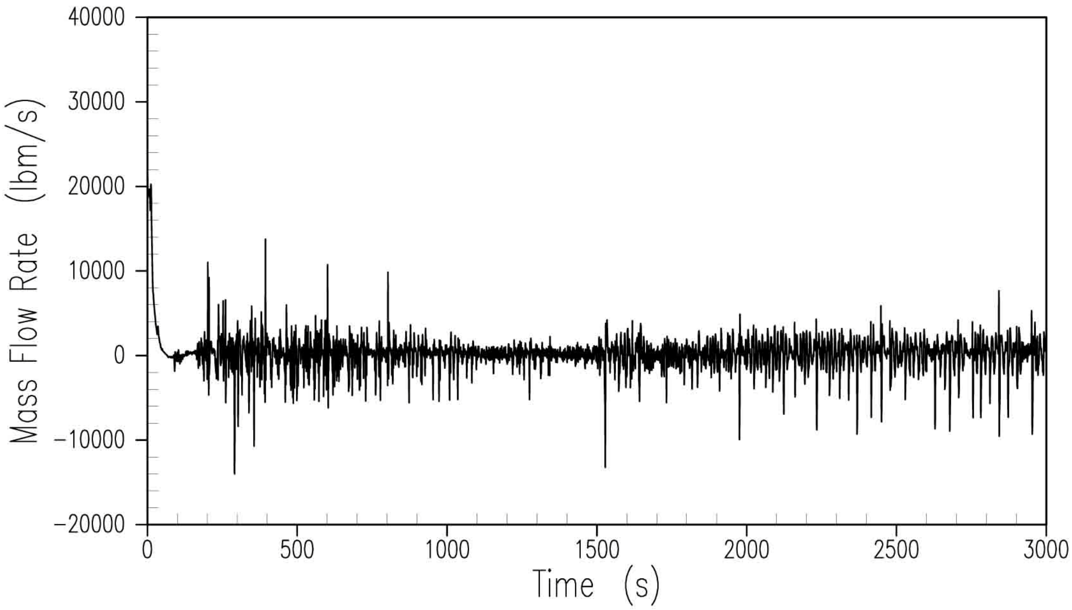

Figure 15.6.5.4B-71
10-inch Cold Leg Break – Core Exit Liquid Flow

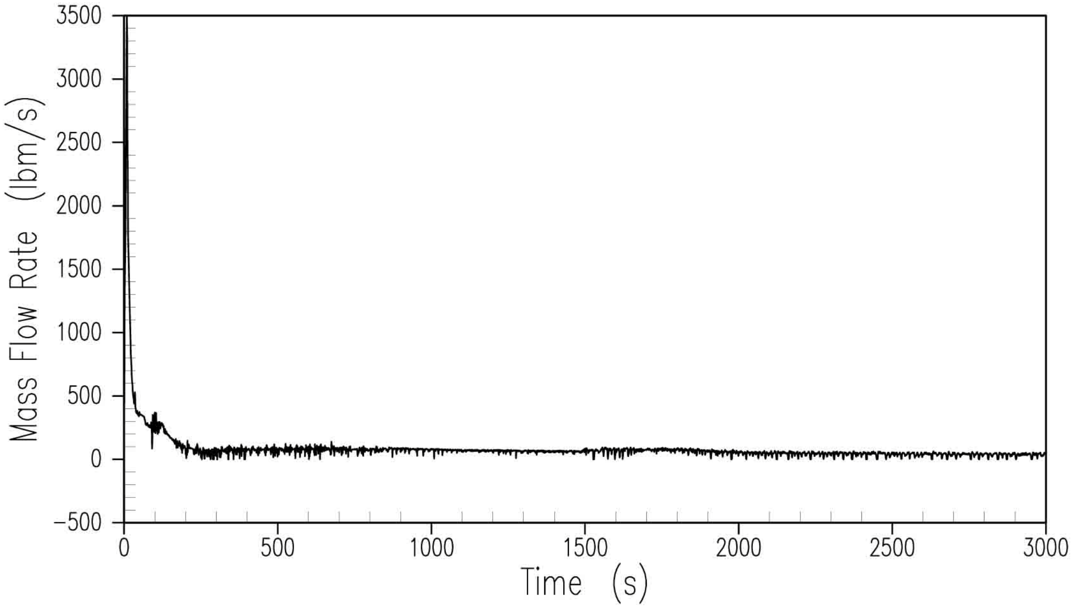

Figure 15.6.5.4B-72
10-inch Cold Leg Break – Core Exit Vapor Flow

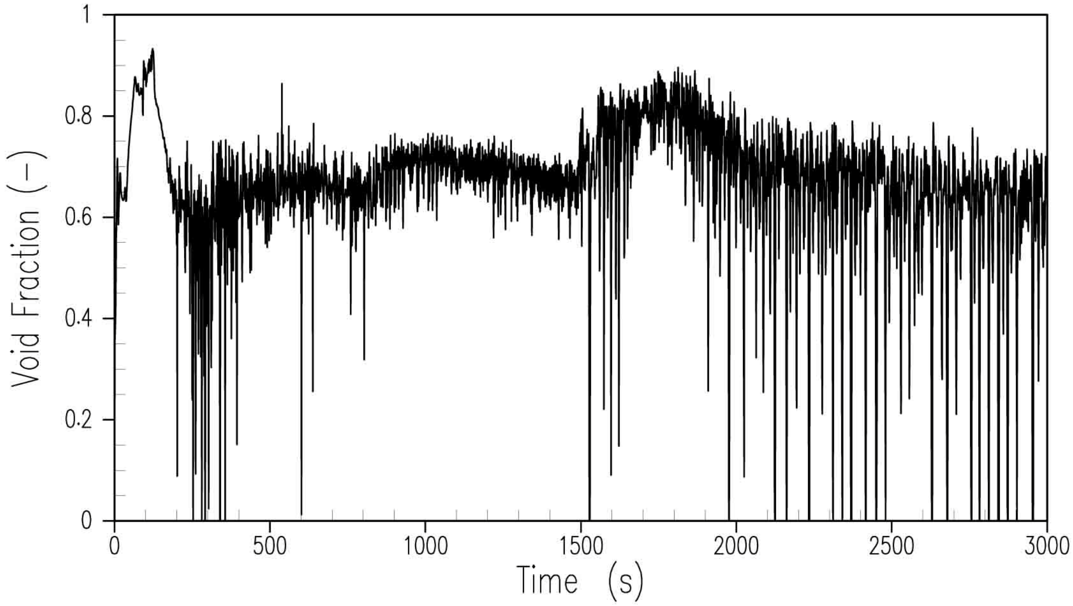

Figure 15.6.5.4B-73
10-inch Cold Leg Break – Core Exit Void Fraction

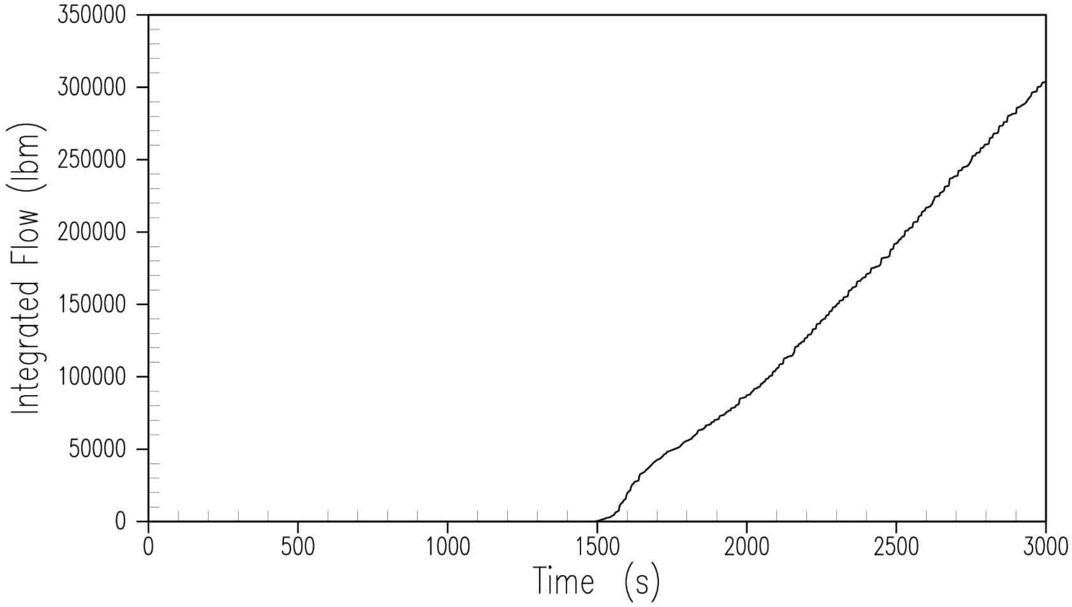

Figure 15.6.5.4B-74
10-inch Cold Leg Break – ADS-4 Integrated Discharge

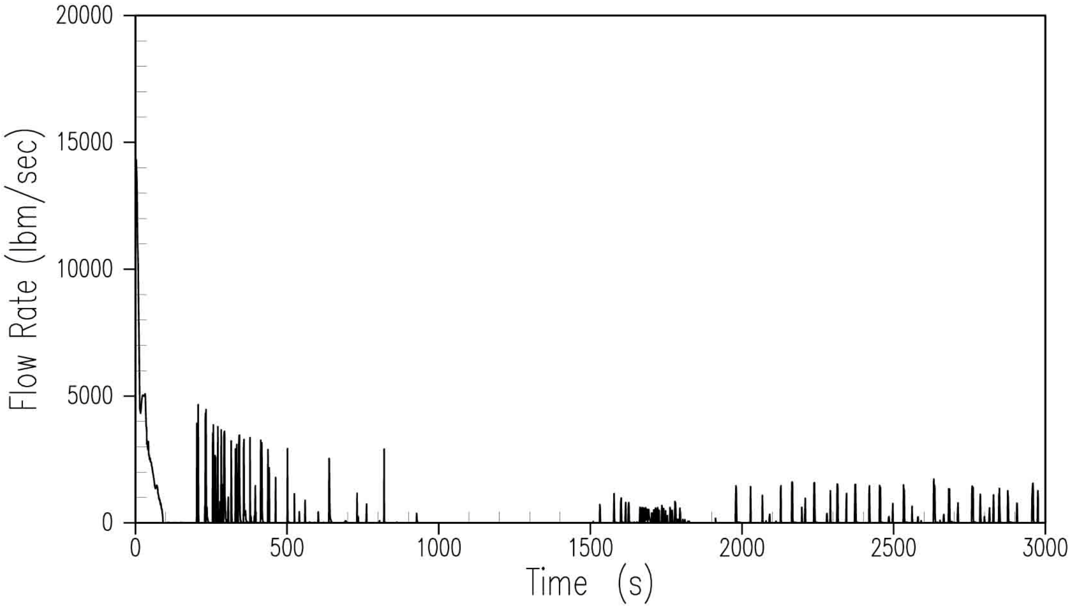

Figure 15.6.5.4B-75
10-inch Cold Leg Break – Liquid Break Discharge

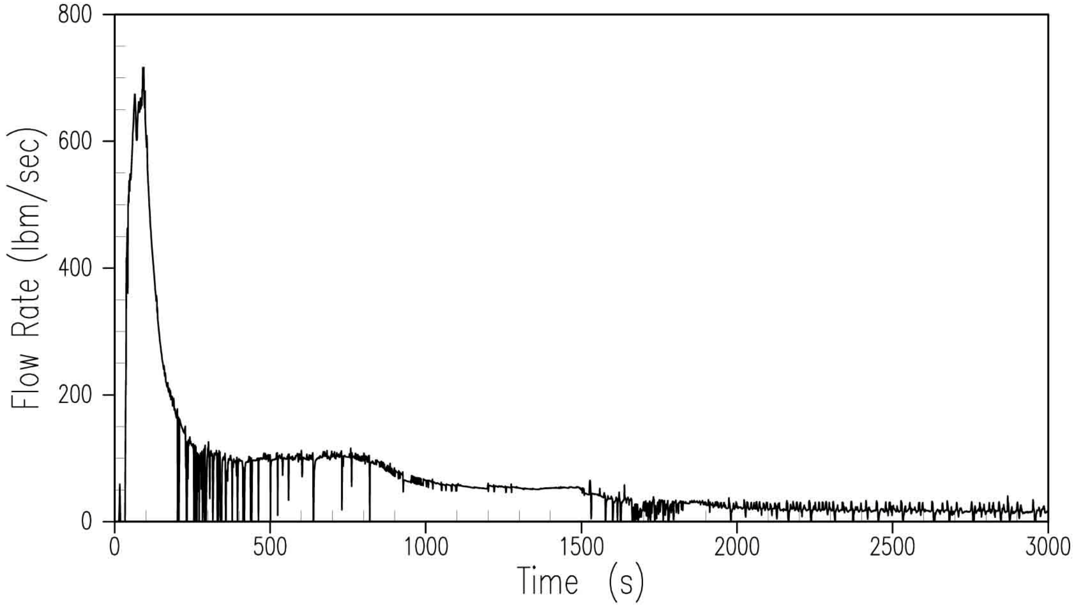

Figure 15.6.5.4B-76
10-inch Cold Leg Break – Vapor Break Discharge

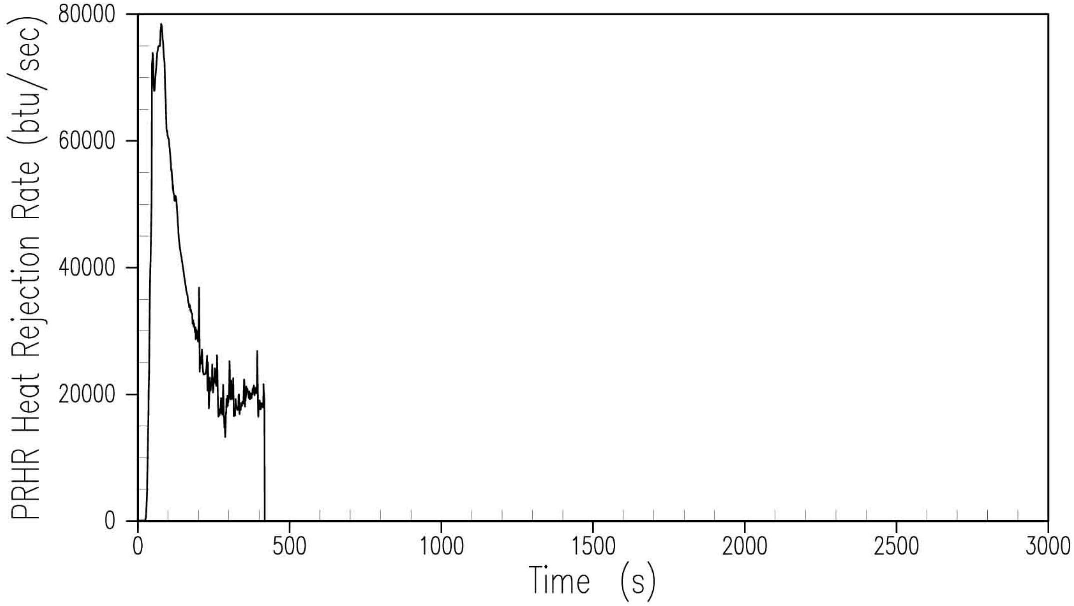

Figure 15.6.5.4B-77
10-inch Cold Leg Break – PRHR Heat Removal Rate

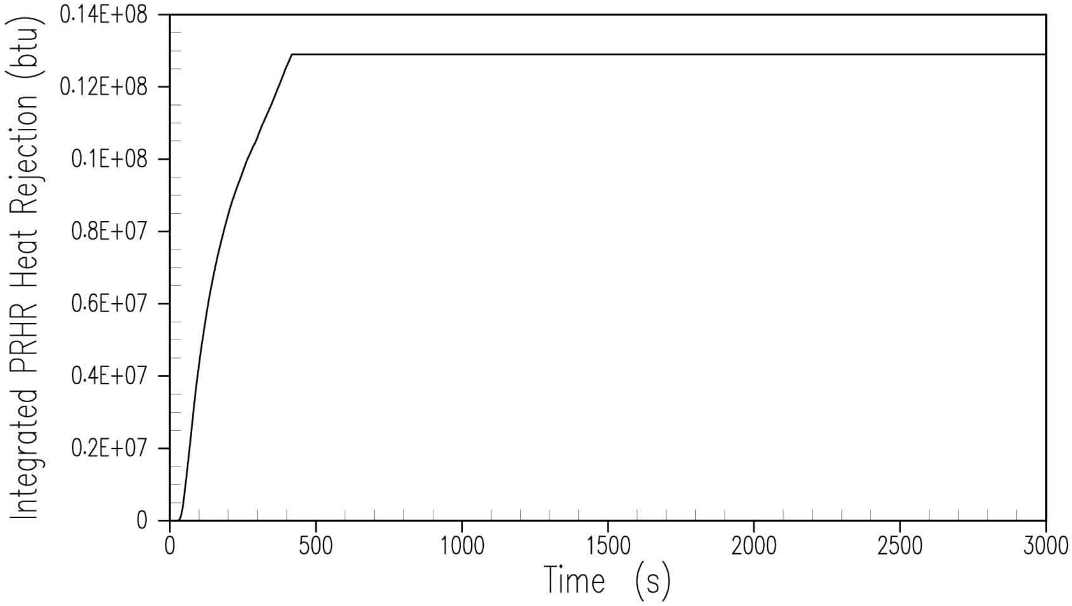

Figure 15.6.5.4B-78
10-inch Cold Leg Break – Integrated PRHR Heat Removal

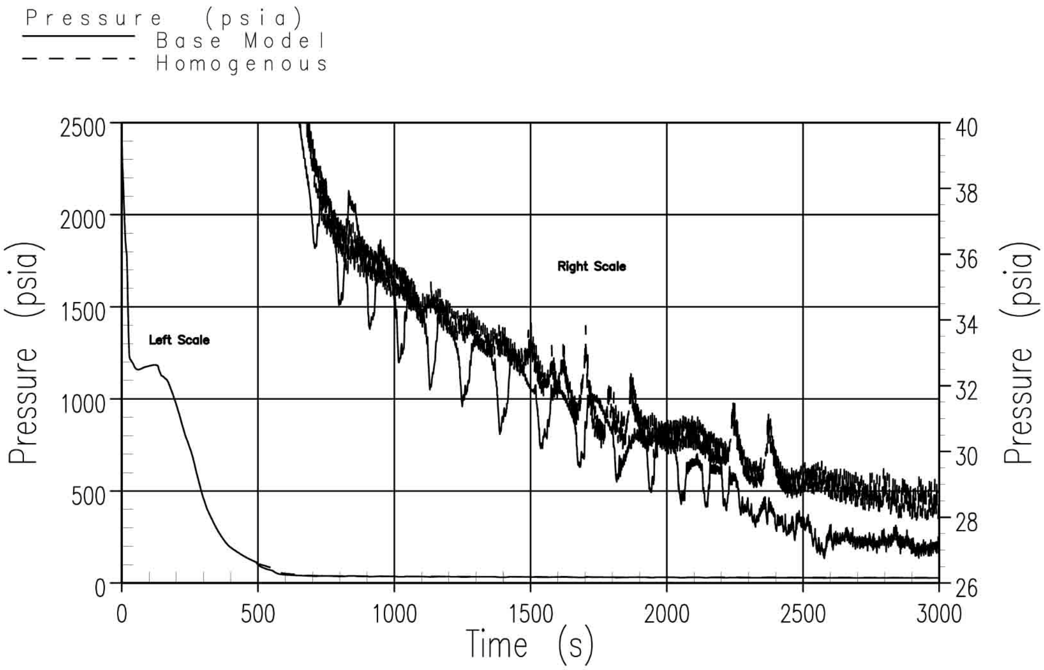

Figure 15.6.5.4B-79
DEDVI – Downcomer Pressure Comparison

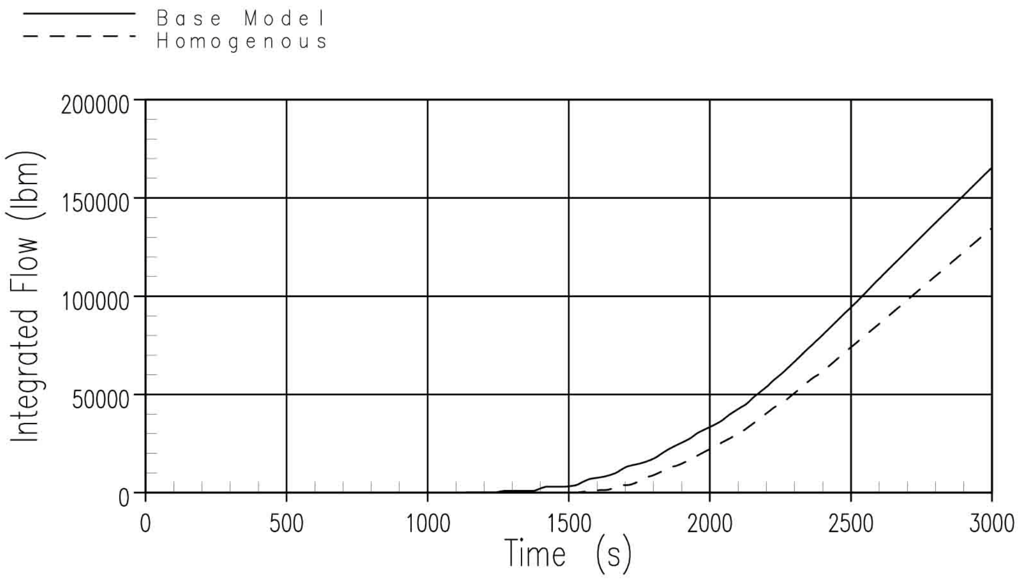

Figure 15.6.5.4B-80
DEDVI – Intact IRWST Injection Flow

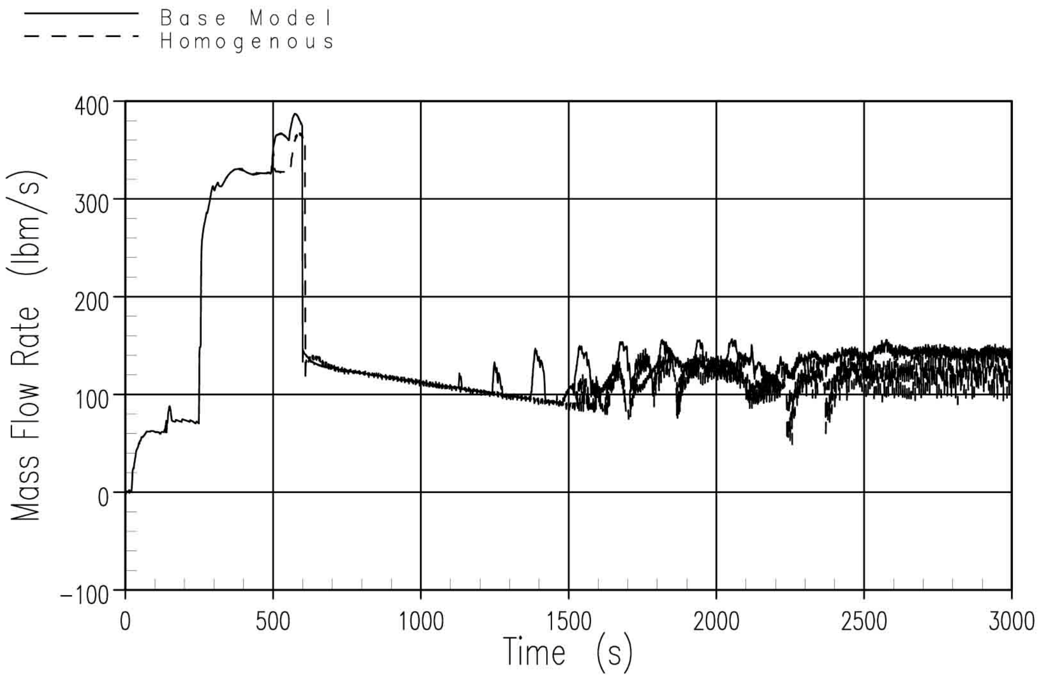

Figure 15.6.5.4B-81
DEDVI – Intact DVI Line Injection Flow

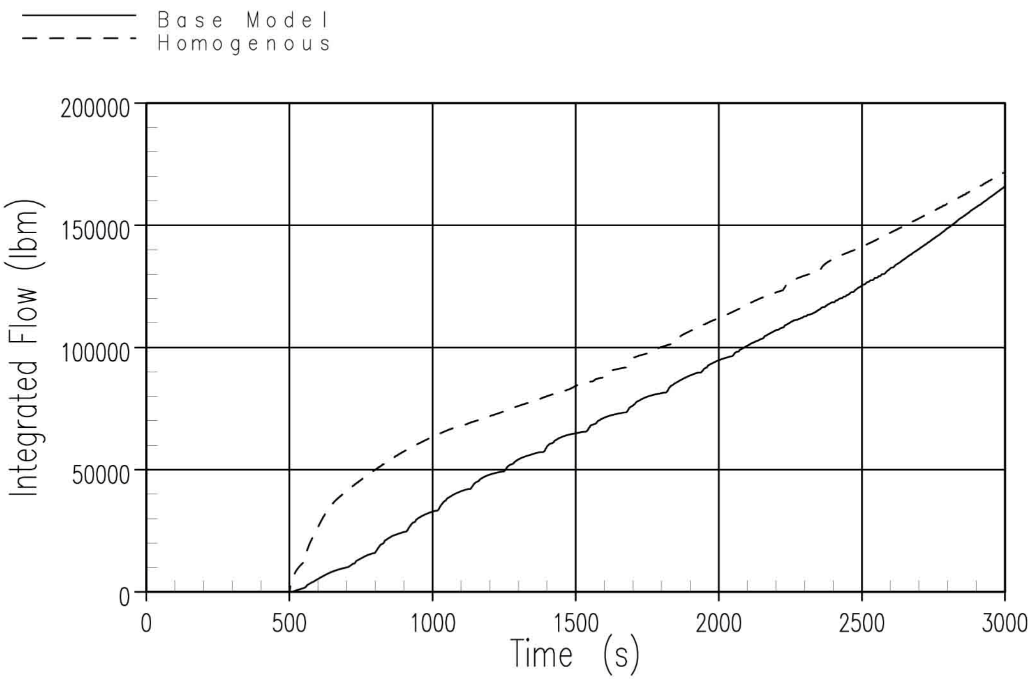

Figure 15.6.5.4B-82
DEDVI – ADS-4 Integrated Liquid Discharge Comparison

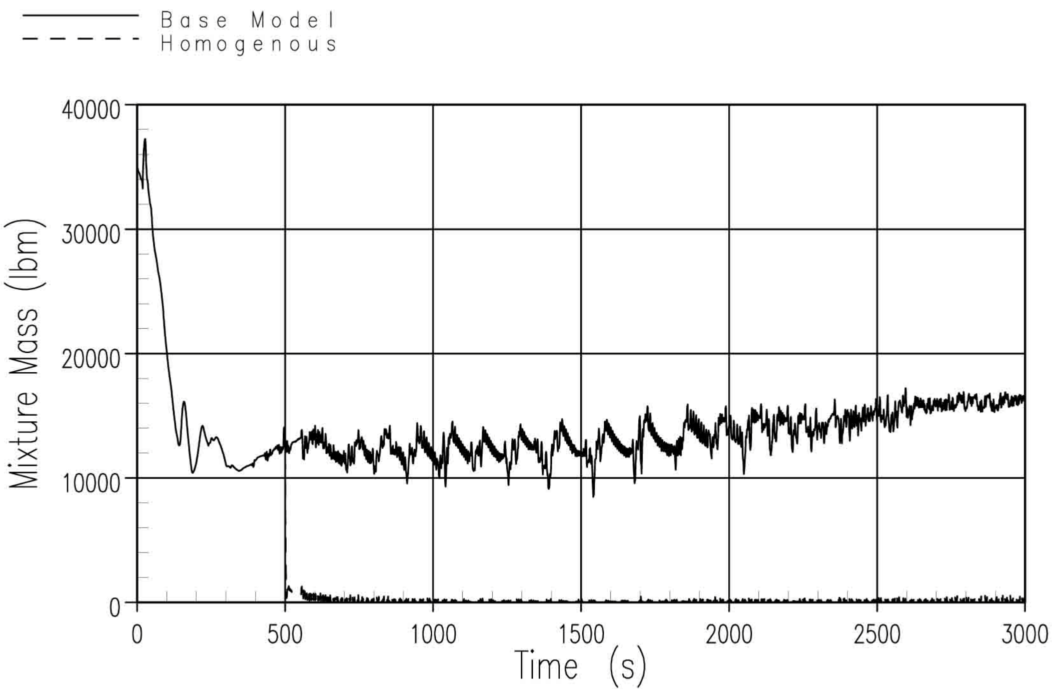

Figure 15.6.5.4B-83
DEDVI – Upper Plenum Mixture Mass Comparison

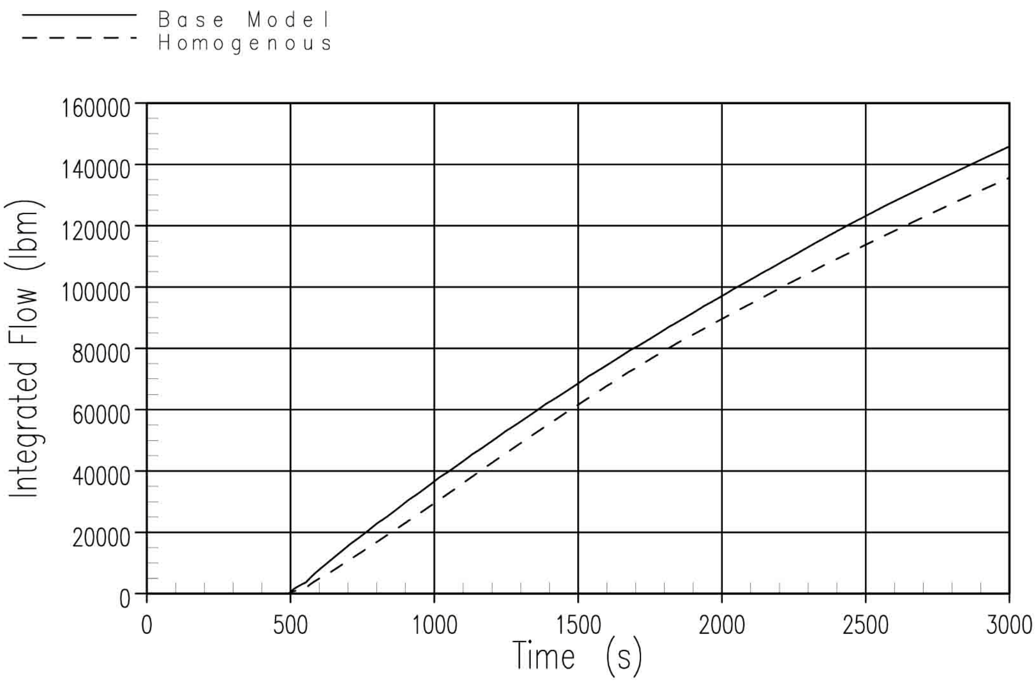

Figure 15.6.5.4B-84
DEDVI – ADS-4 Integrated Vapor Discharge Comparison

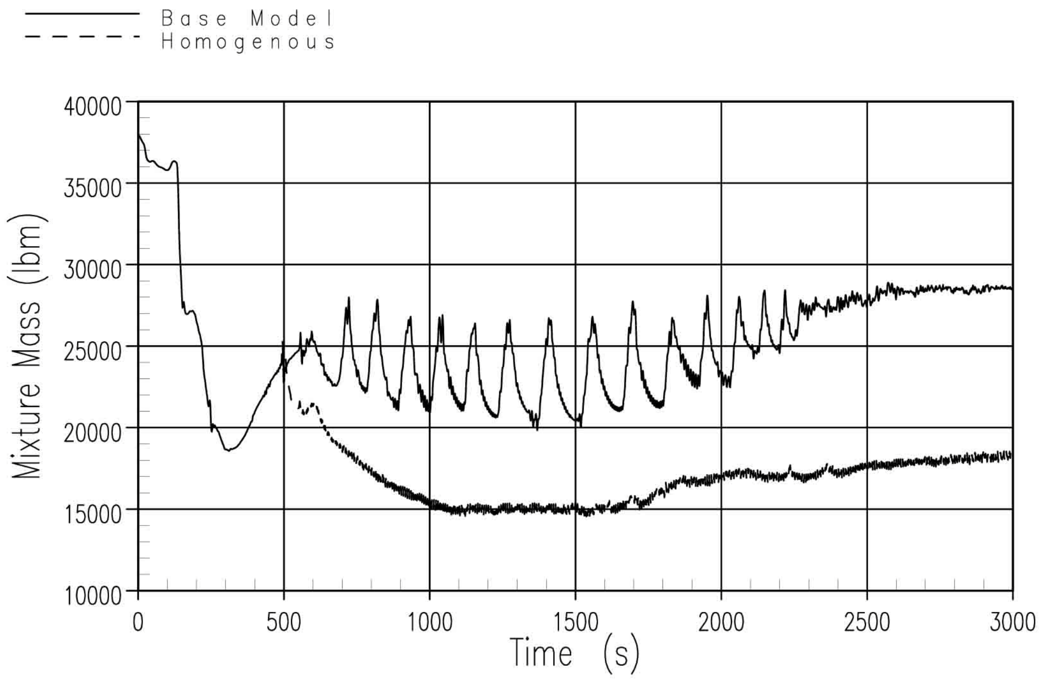

Figure 15.6.5.4B-85
DEDVI – Downcomer Region Mass Comparison

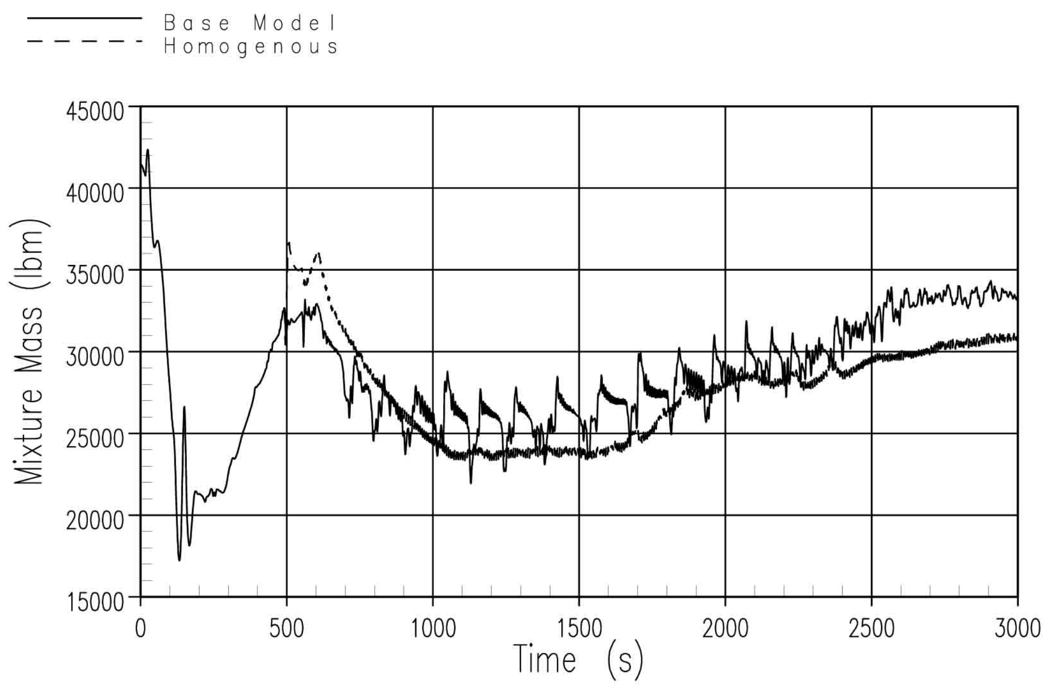

Figure 15.6.5.4B-86
DEDVI – Core Region Mass Comparison

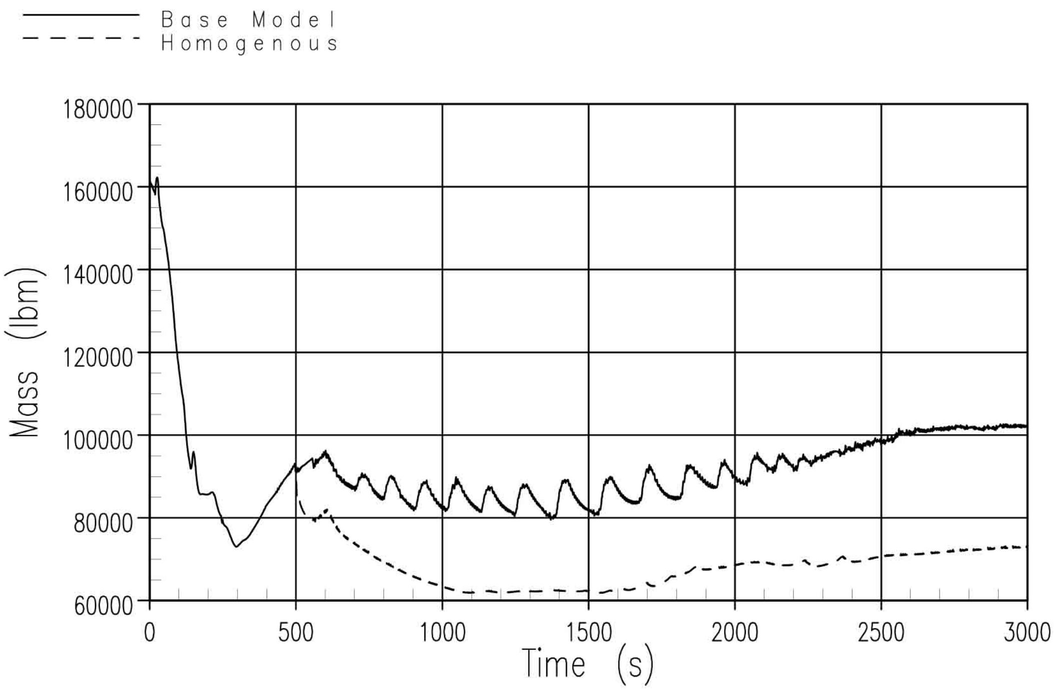

Figure 15.6.5.4B-87
DEDVI – Vessel Mixture Mass Comparison

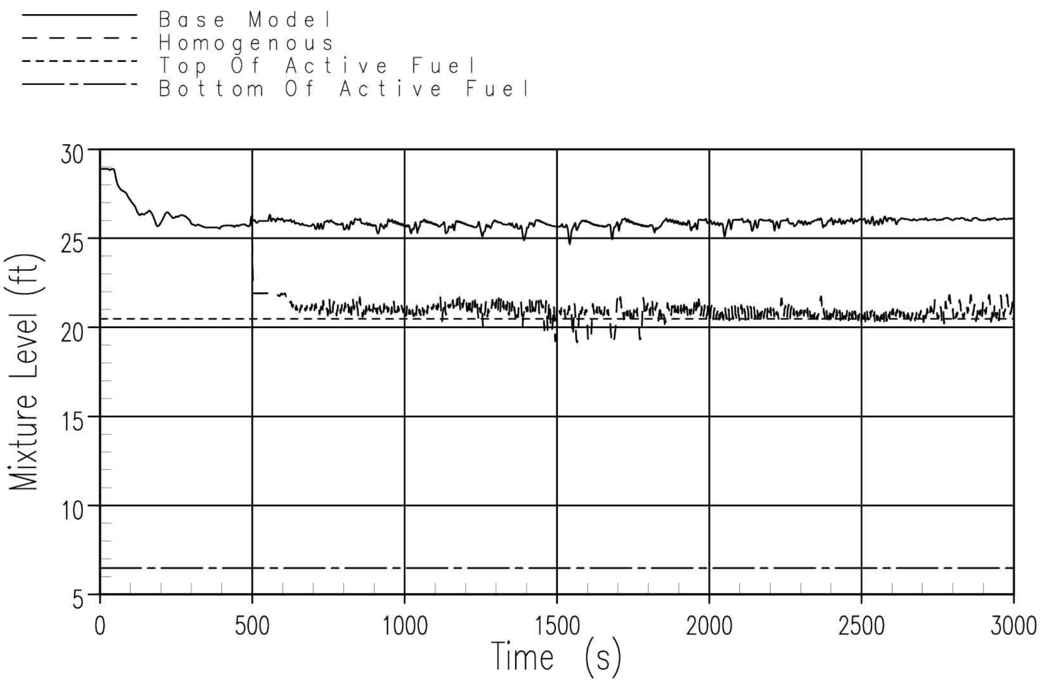

Figure 15.6.5.4B-88
DEDVI – Core/Upper Plenum Mixture Level Comparison

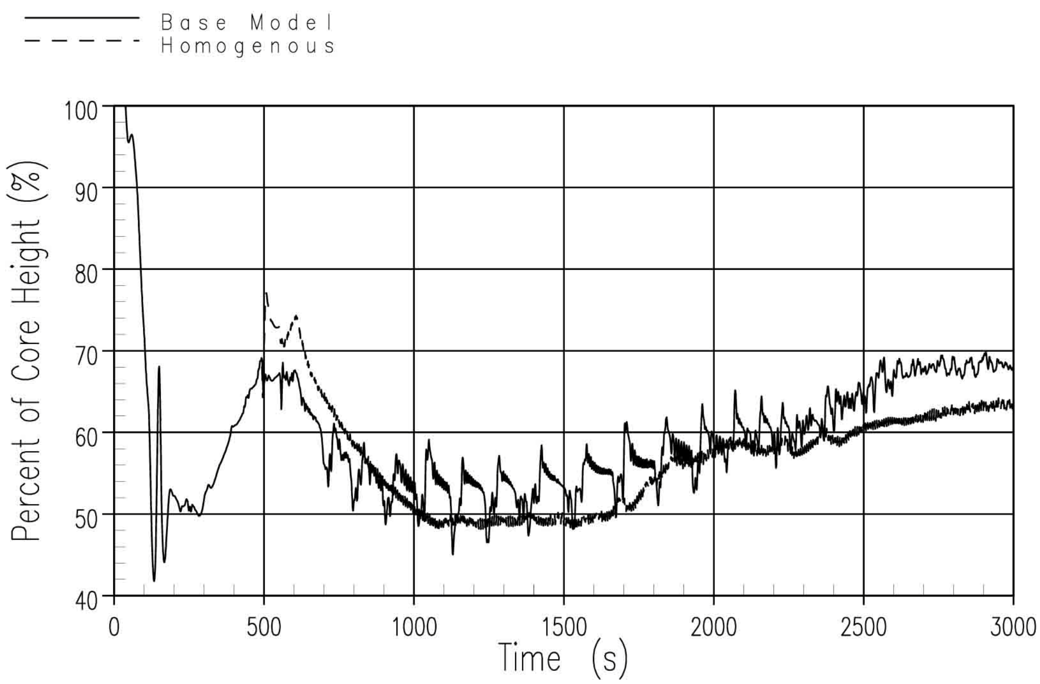

Figure 15.6.5.4B-89
DEDVI – Core Collapsed Liquid Level Comparison

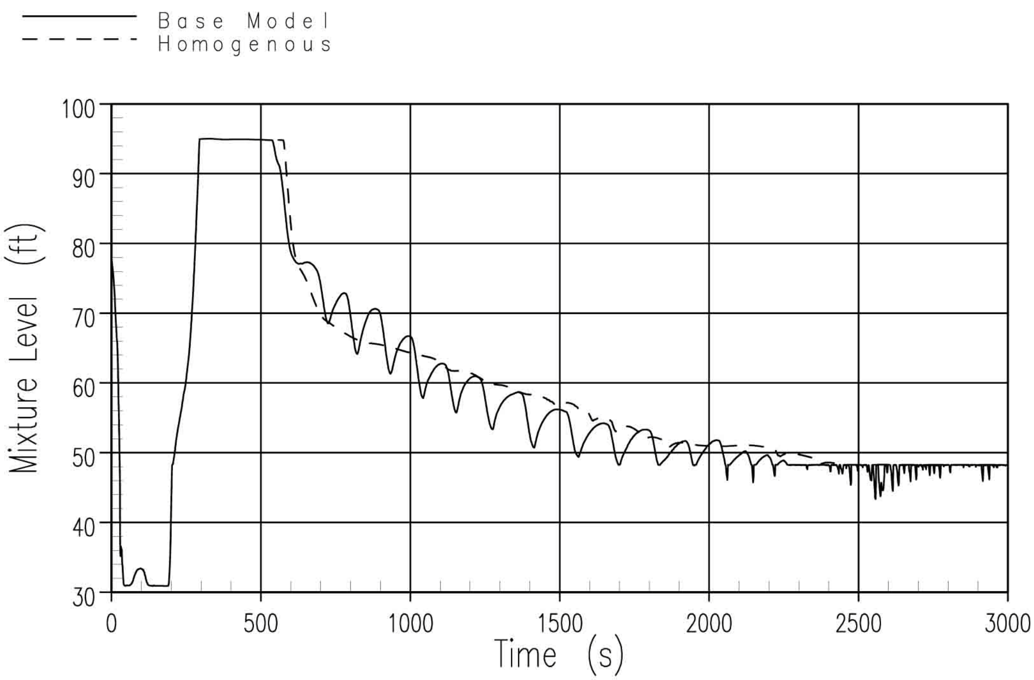

Figure 15.6.5.4B-90
DEDVI – Pressurizer Mixture Level Comparison

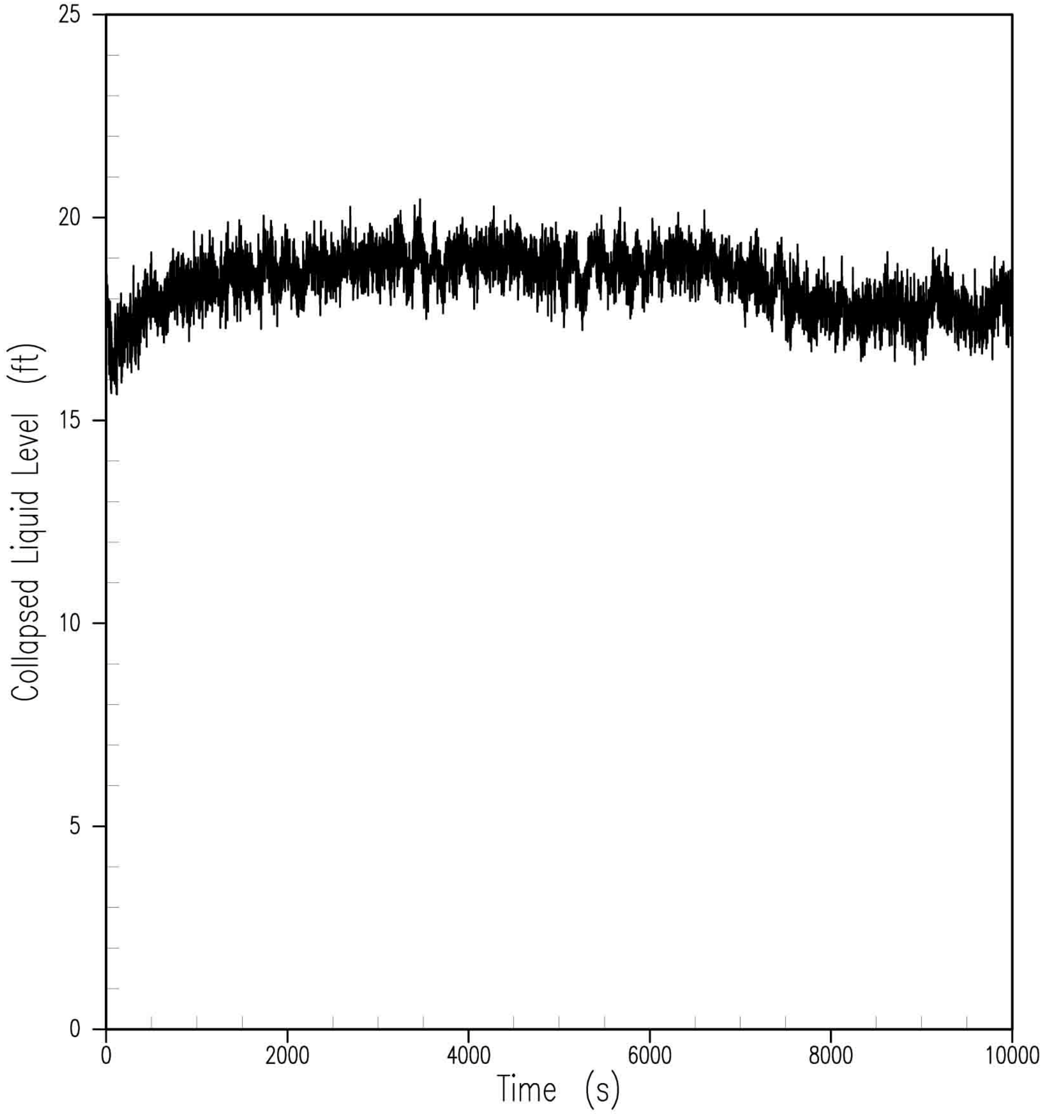

Figure 15.6.5.4C-1
Collapsed Level of Liquid in the Downcomer
(DEDVI Case)

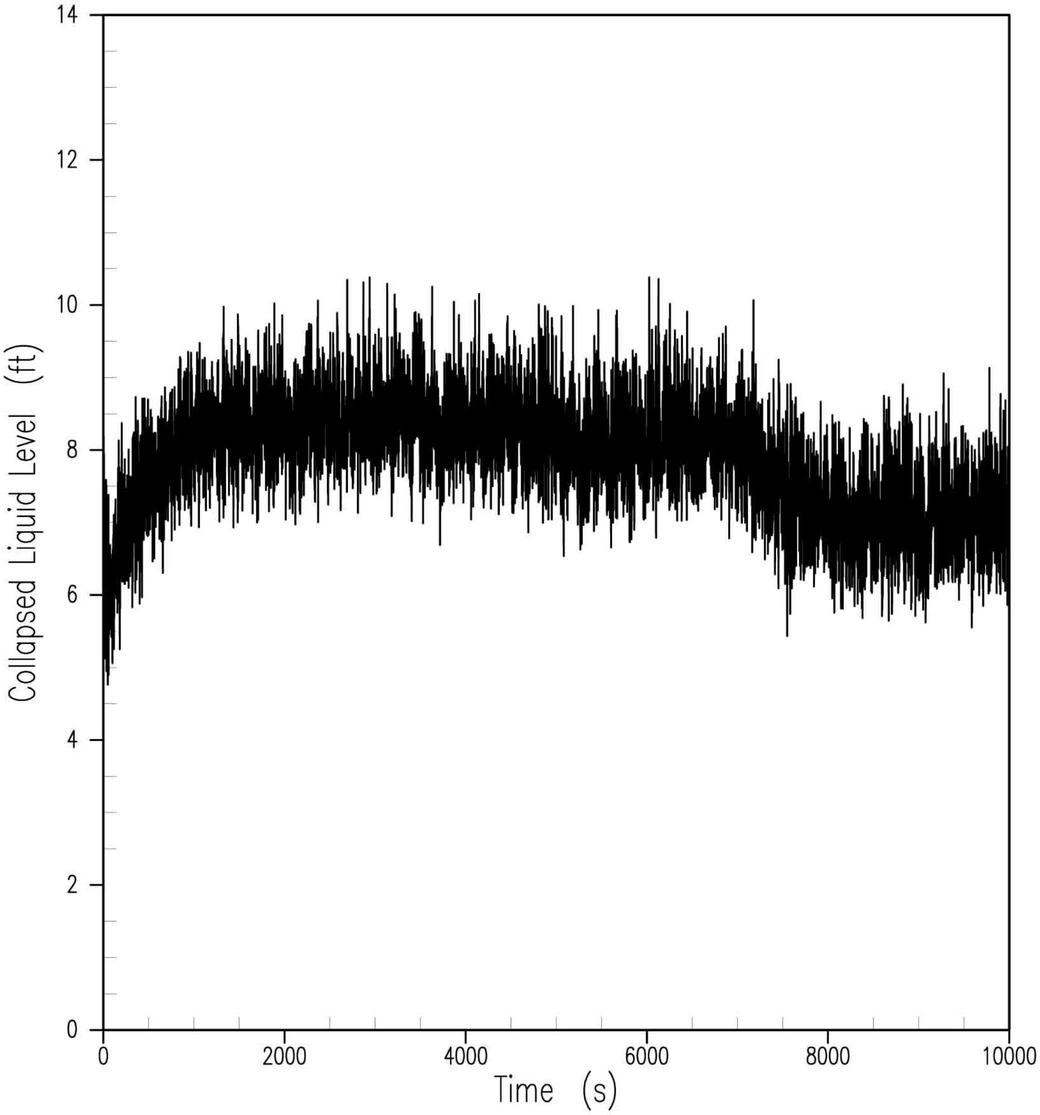

Figure 15.6.5.4C-2
Collapsed Level of Liquid over the Heated Length of the Fuel (DEDVI Case)

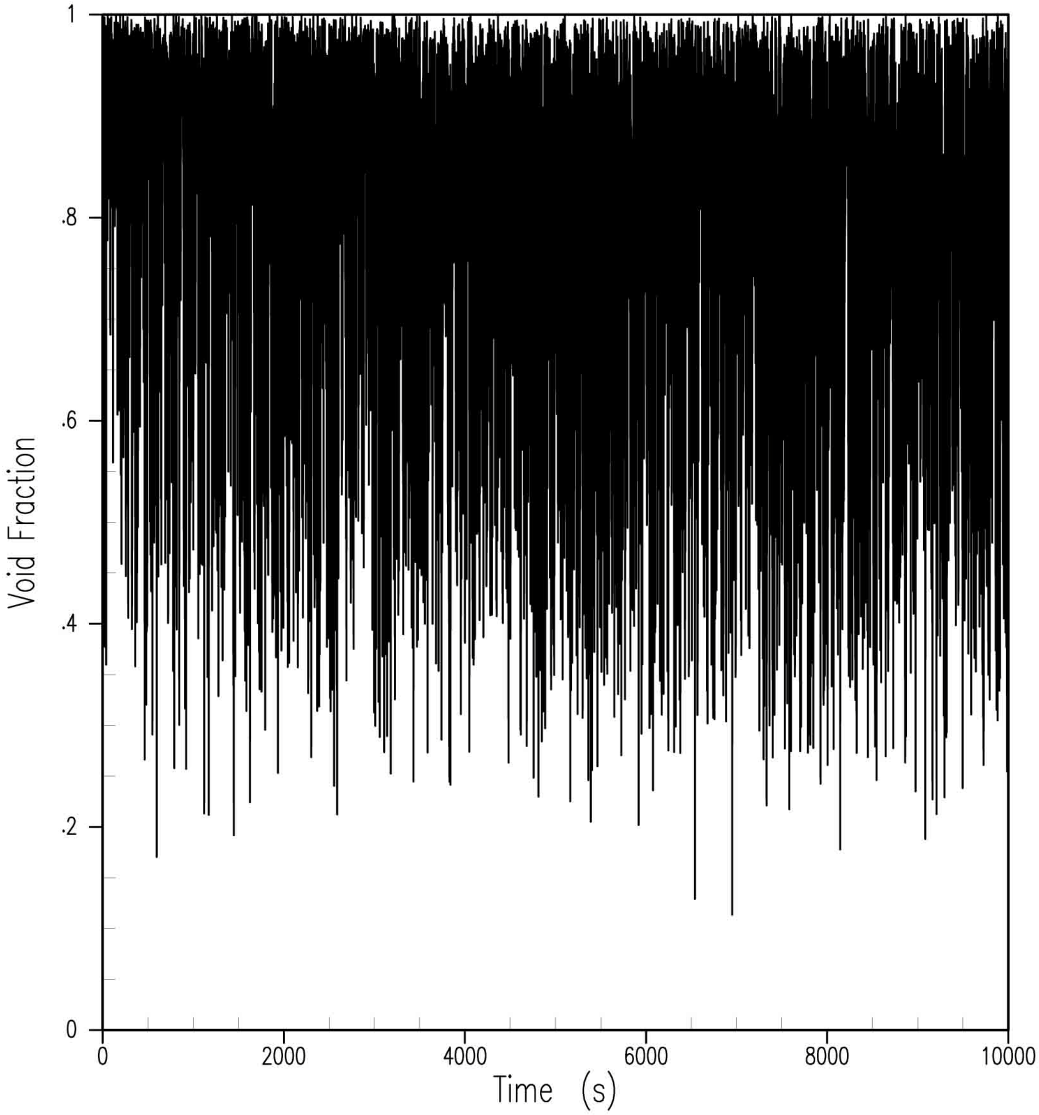

Figure 15.6.5.4C-3
Void Fraction in Core Hot Assembly Top Cell
(DEDVI Case)

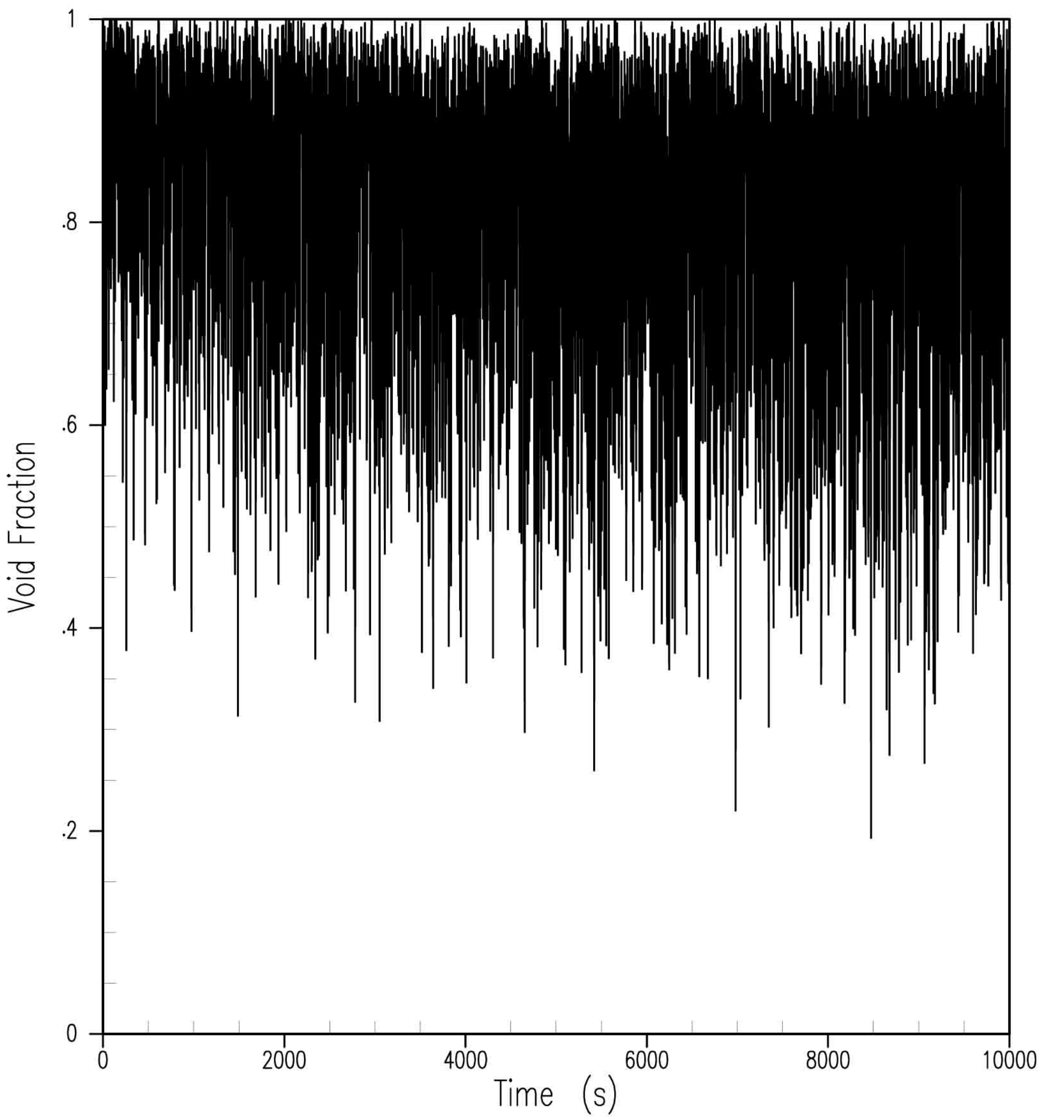

Figure 15.6.5.4C-4
Void Fraction in Core Hot Assembly Second from Top Cell
(DEDVI Case)

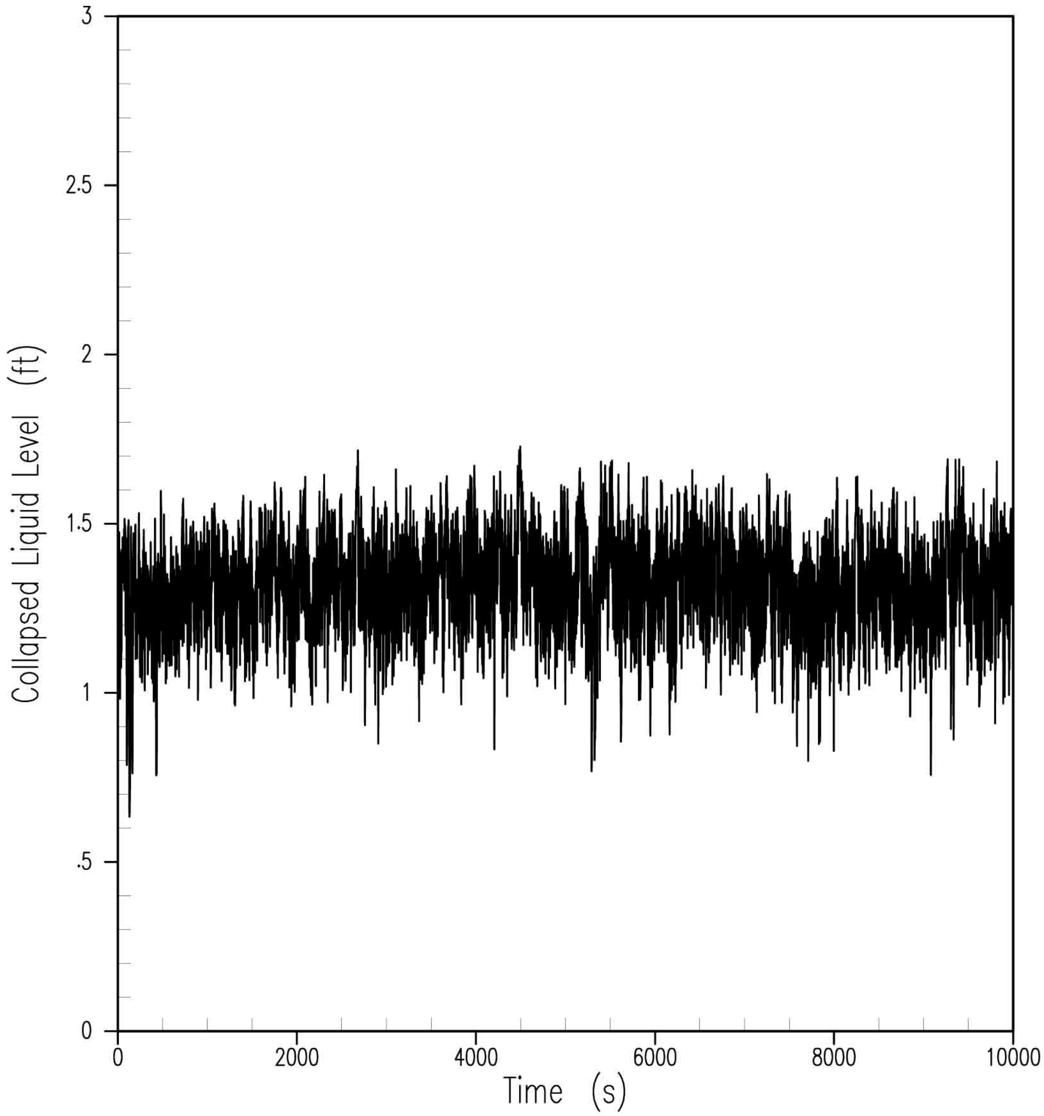

Figure 15.6.5.4C-5
Collapsed Liquid Level in the Hot Leg of Pressurizer Loop (DEDVI Case)

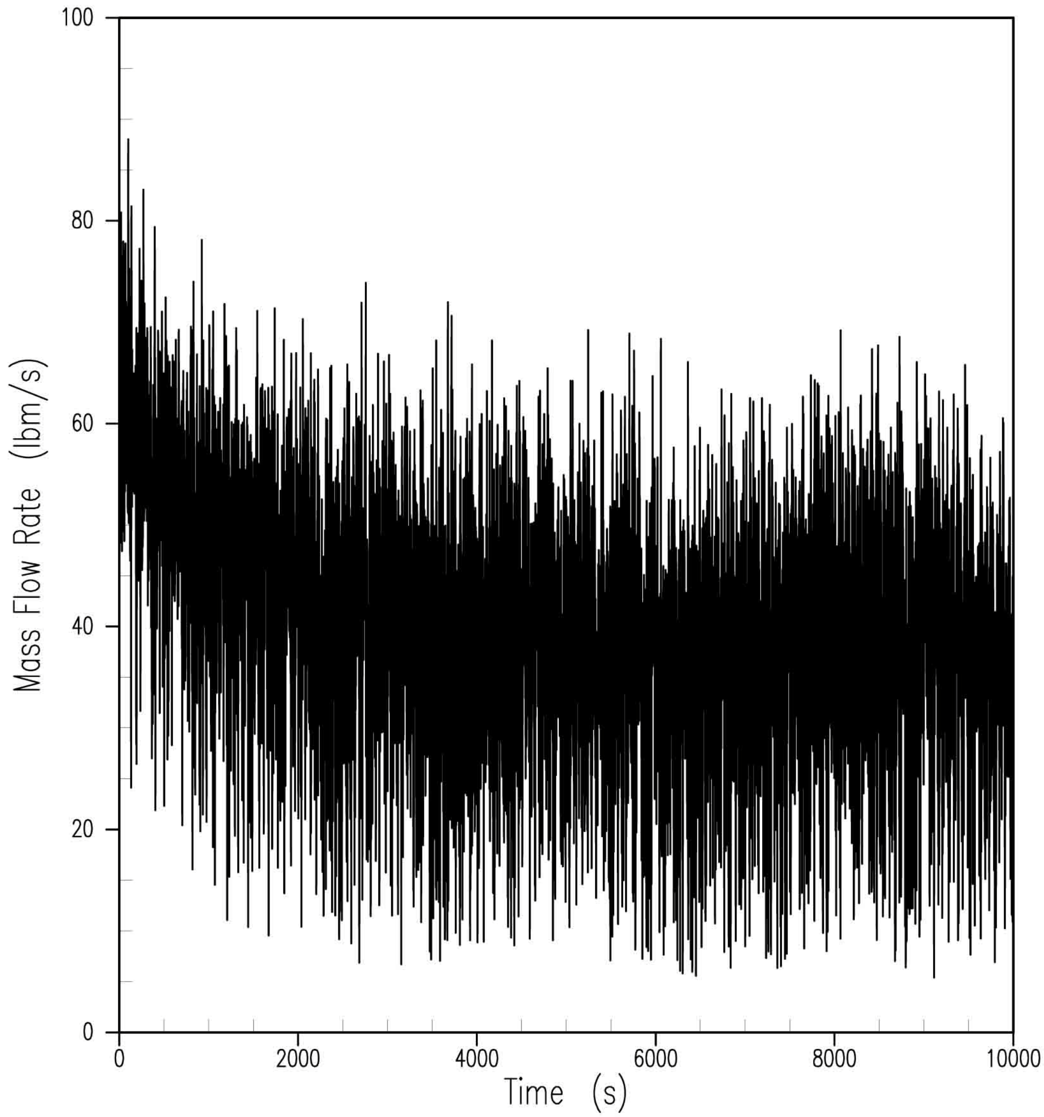

Figure 15.6.5.4C-6
Vapor Rate out of the Core
(DEDVI Case)

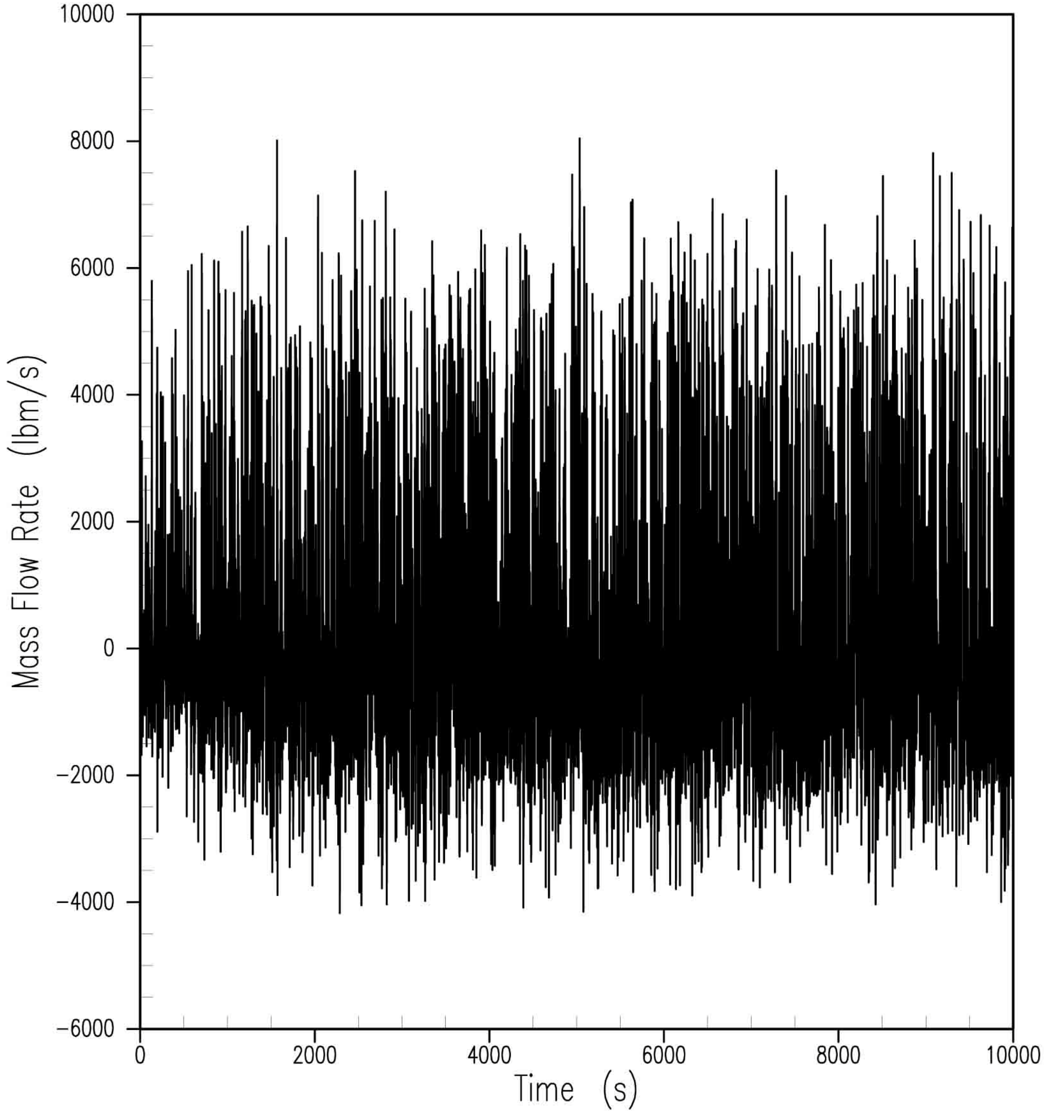

Figure 15.6.5.4C-7
Liquid Flow Rate out of the Core
(DEDVI Case)

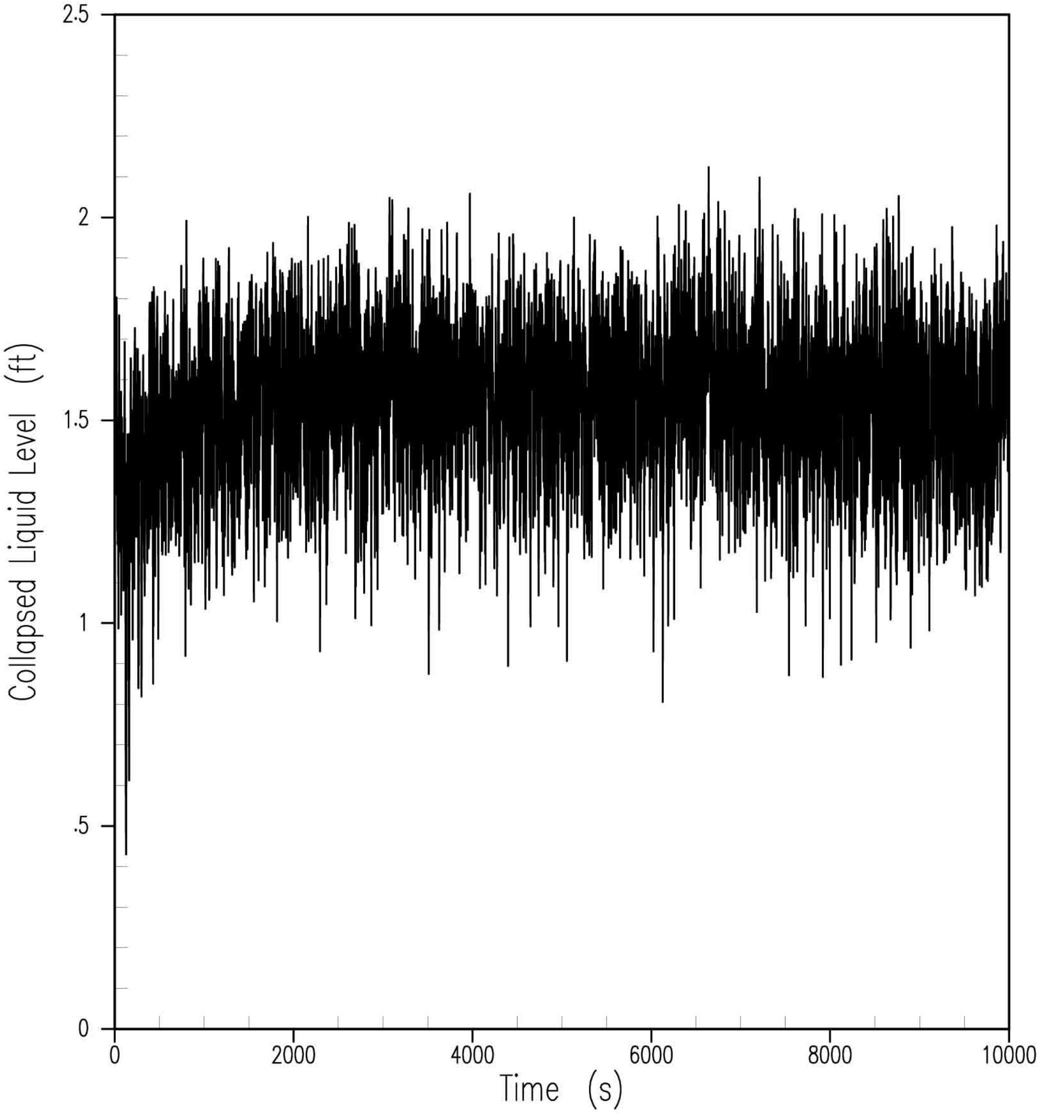

Figure 15.6.5.4C-8
Collapsed Liquid Level in the Upper Plenum
(DEDVI Case)

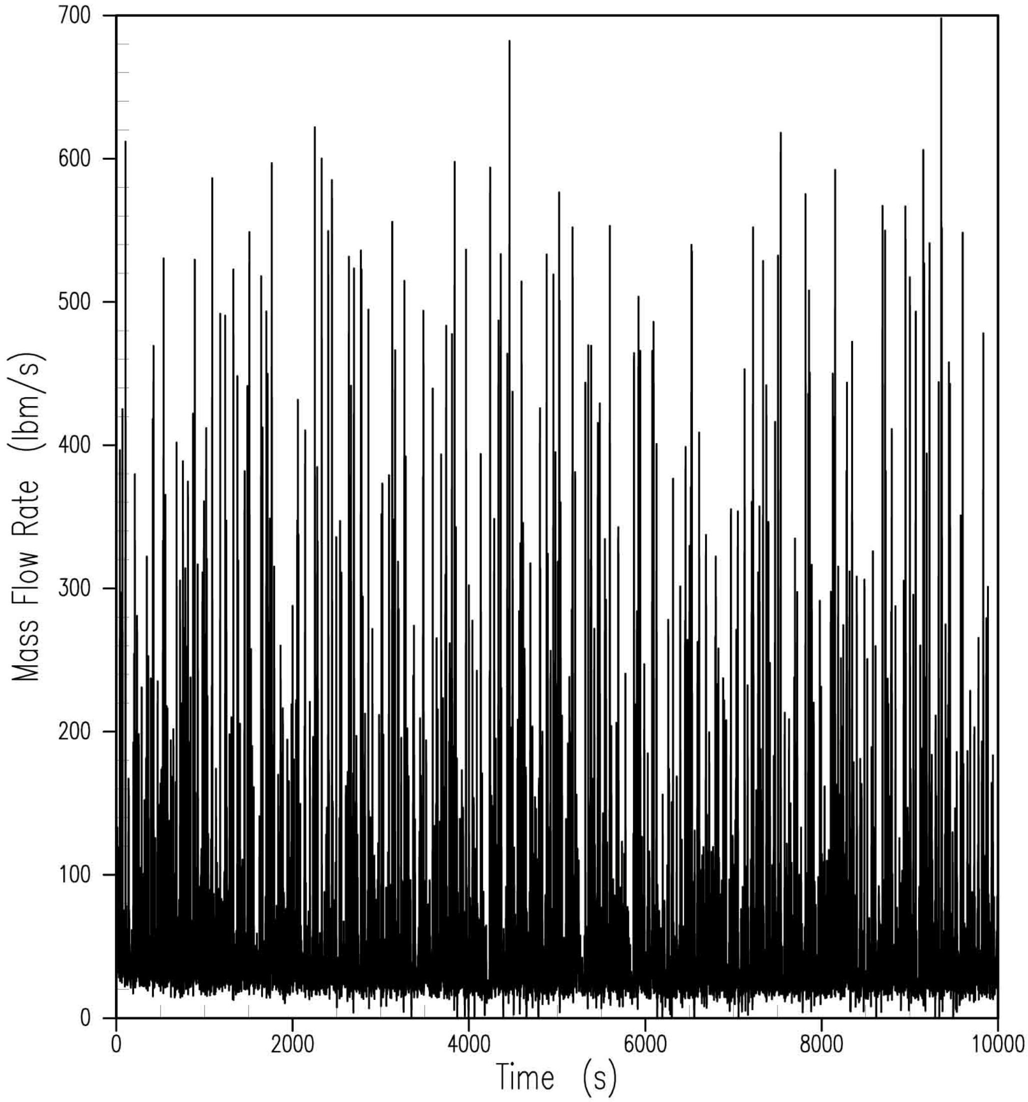

Figure 15.6.5.4C-9
Mixture Flow Rate Through ADS Stage 4A Valves
(DEDVI Case)

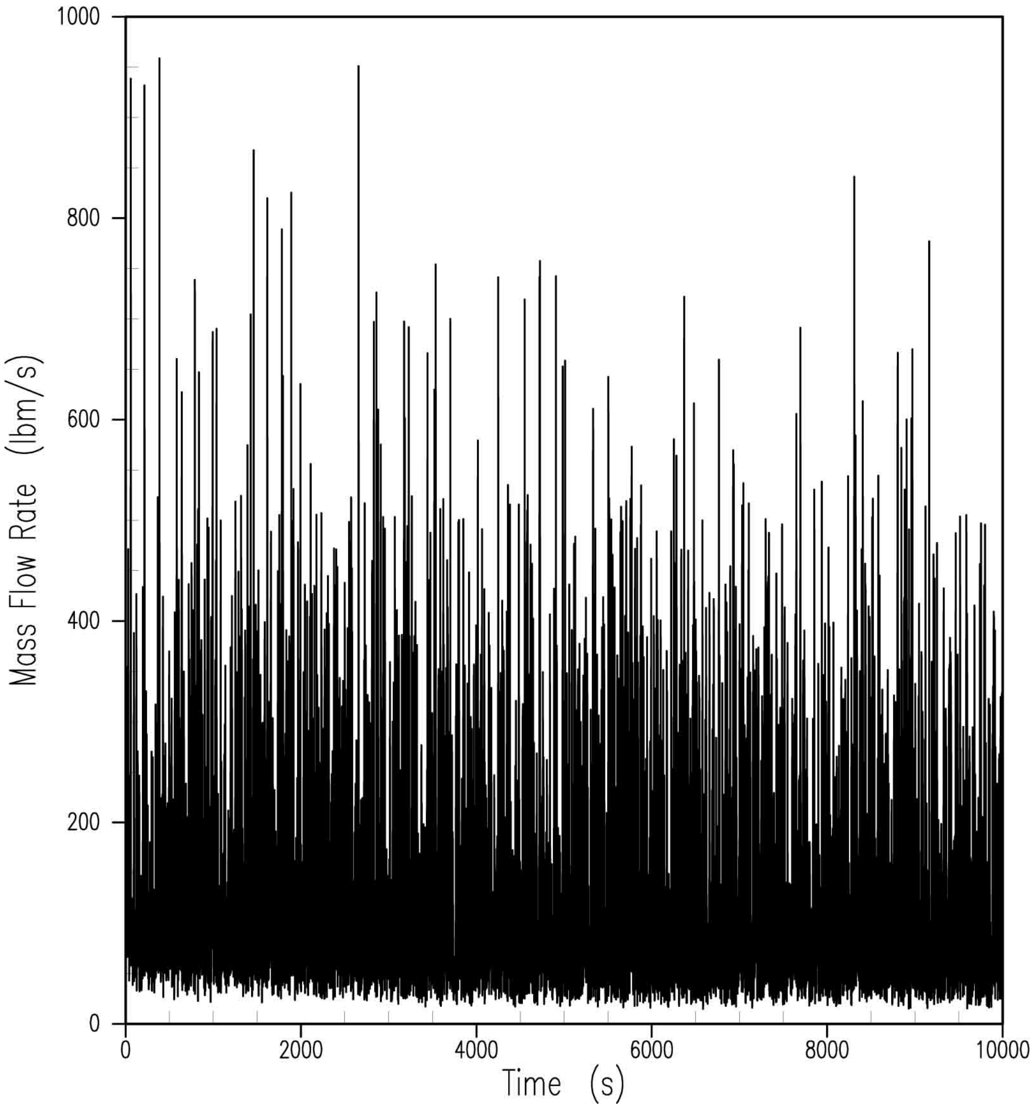

Figure 15.6.5.4C-10
Mixture Flow Rate Through ADS Stage 4B Valves
(DEDVI Case)

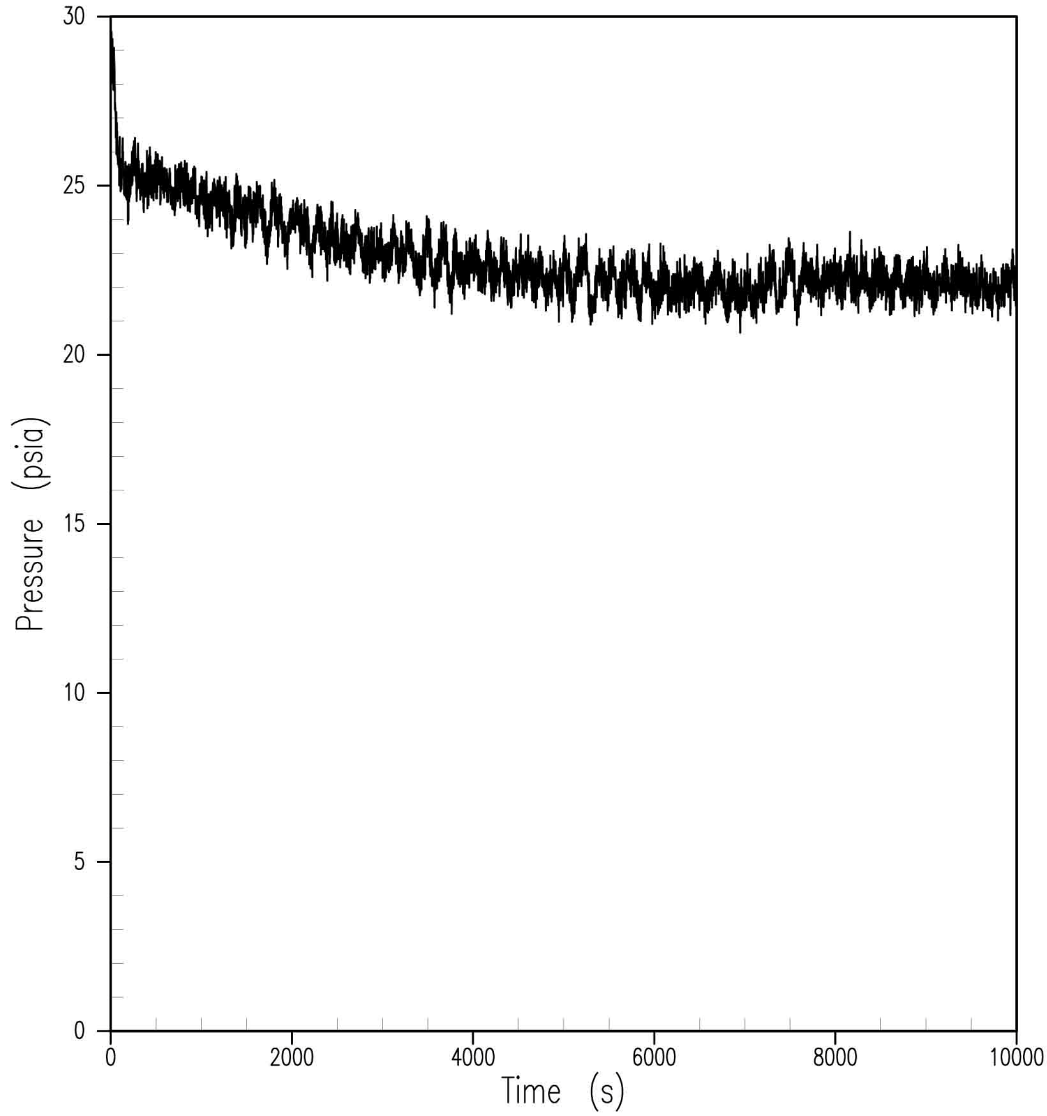

Figure 15.6.5.4C-11
Upper Plenum Pressure
(DEDVI Case)

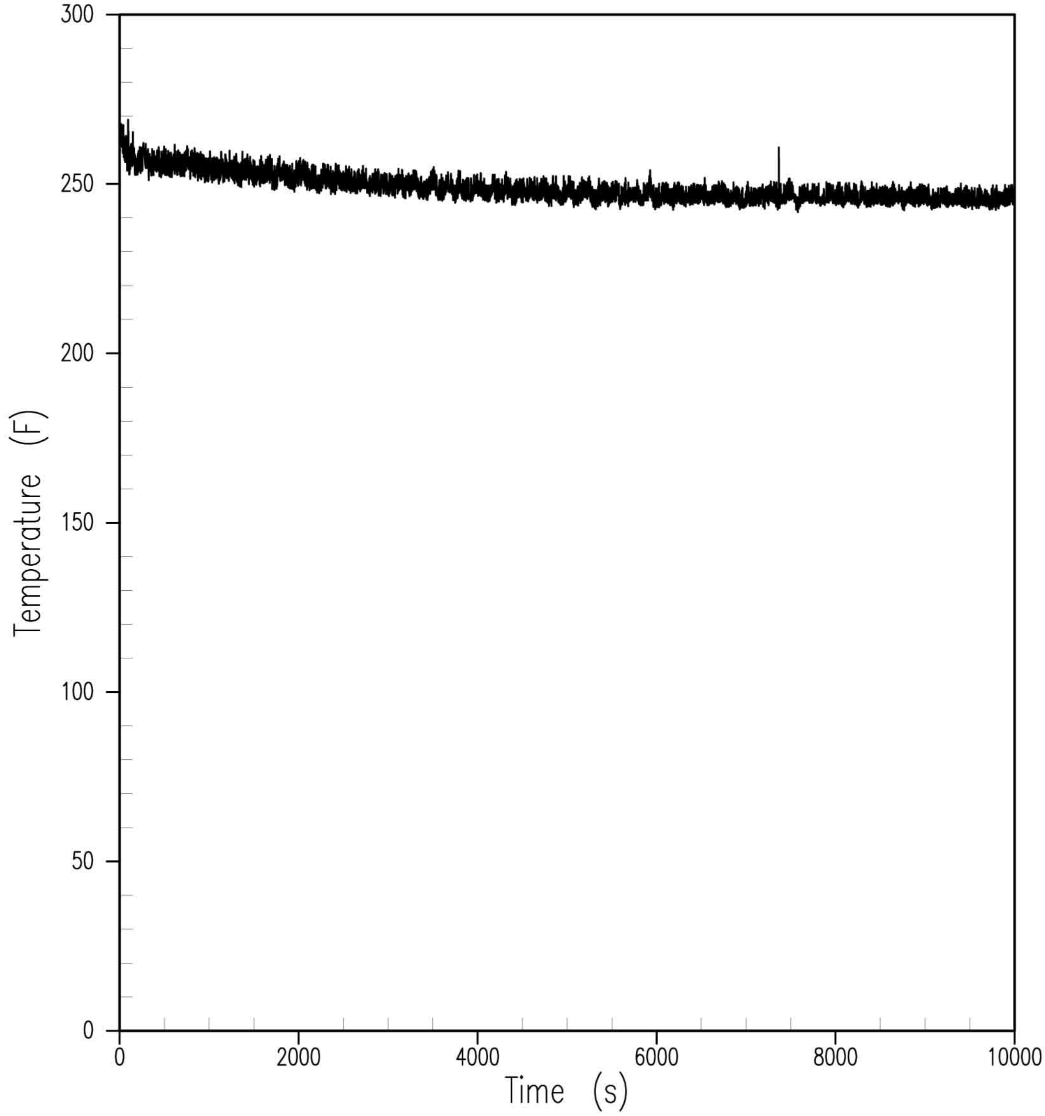

Figure 15.6.5.4C-12
Peak Cladding Temperature
(DEDVI Case)

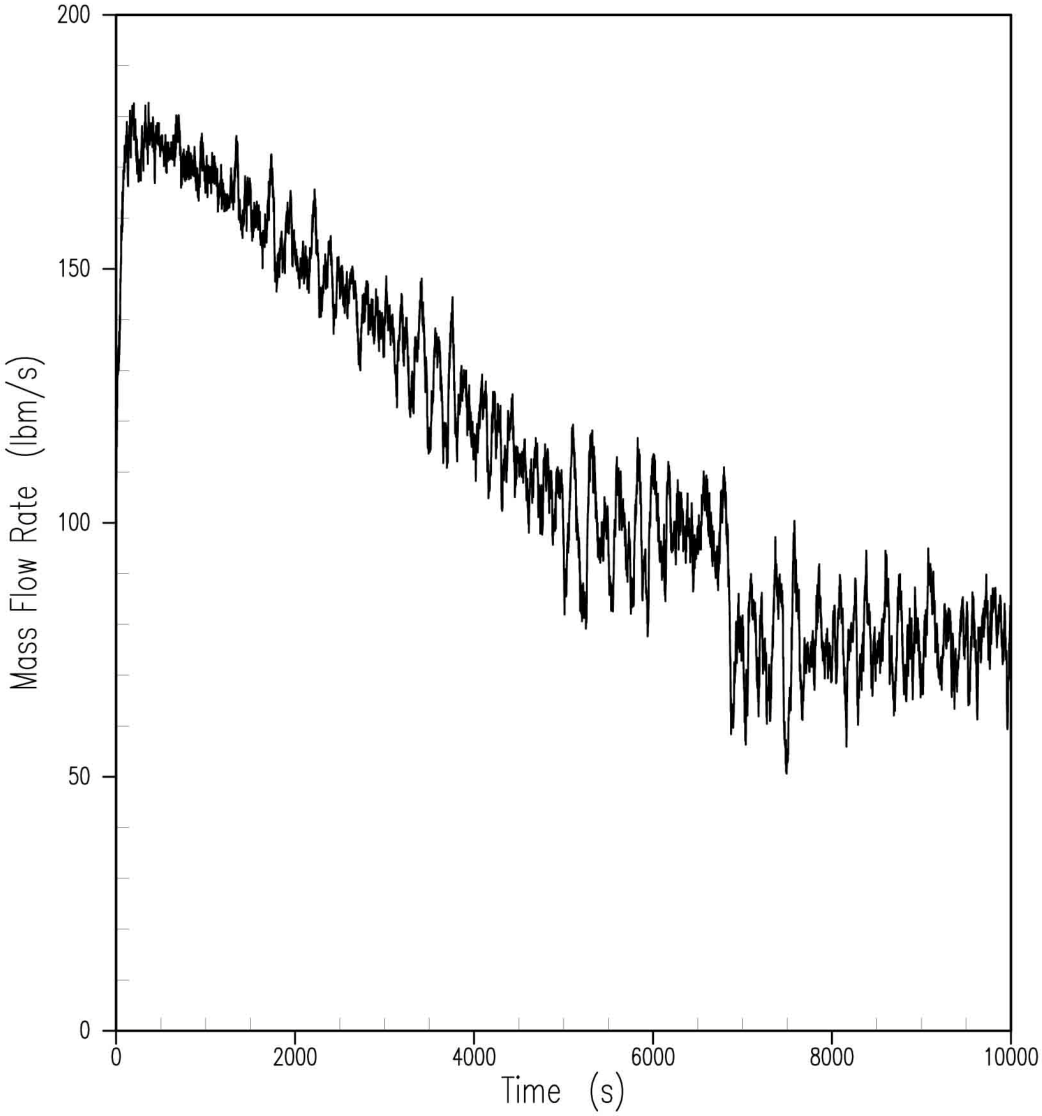

Figure 15.6.5.4C-13
DVI–A Mixture Flow Rate
(DEDVI Case)

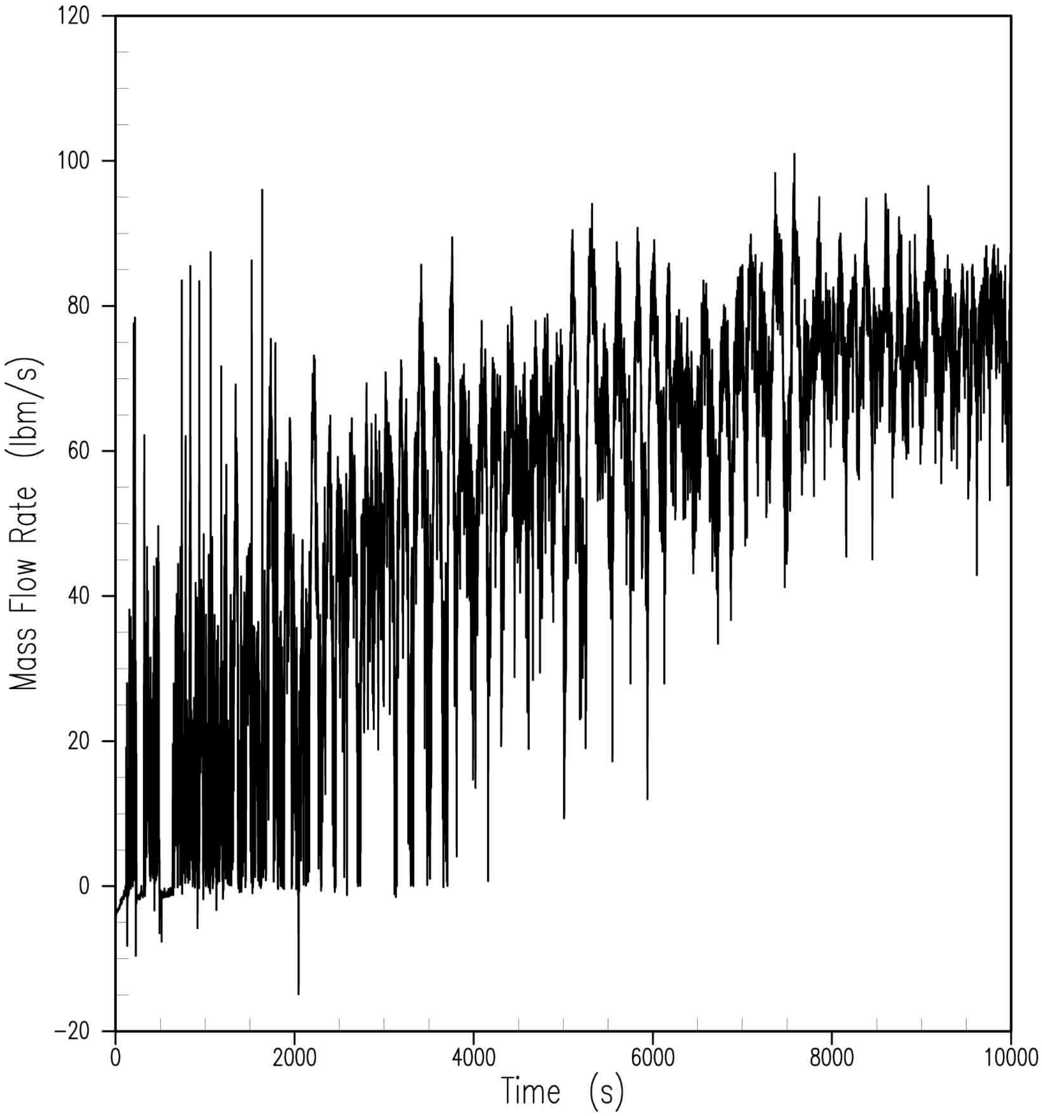

Figure 15.6.5.4C-14
DVI–B Mixture Flow Rate
(DEDVI Case)

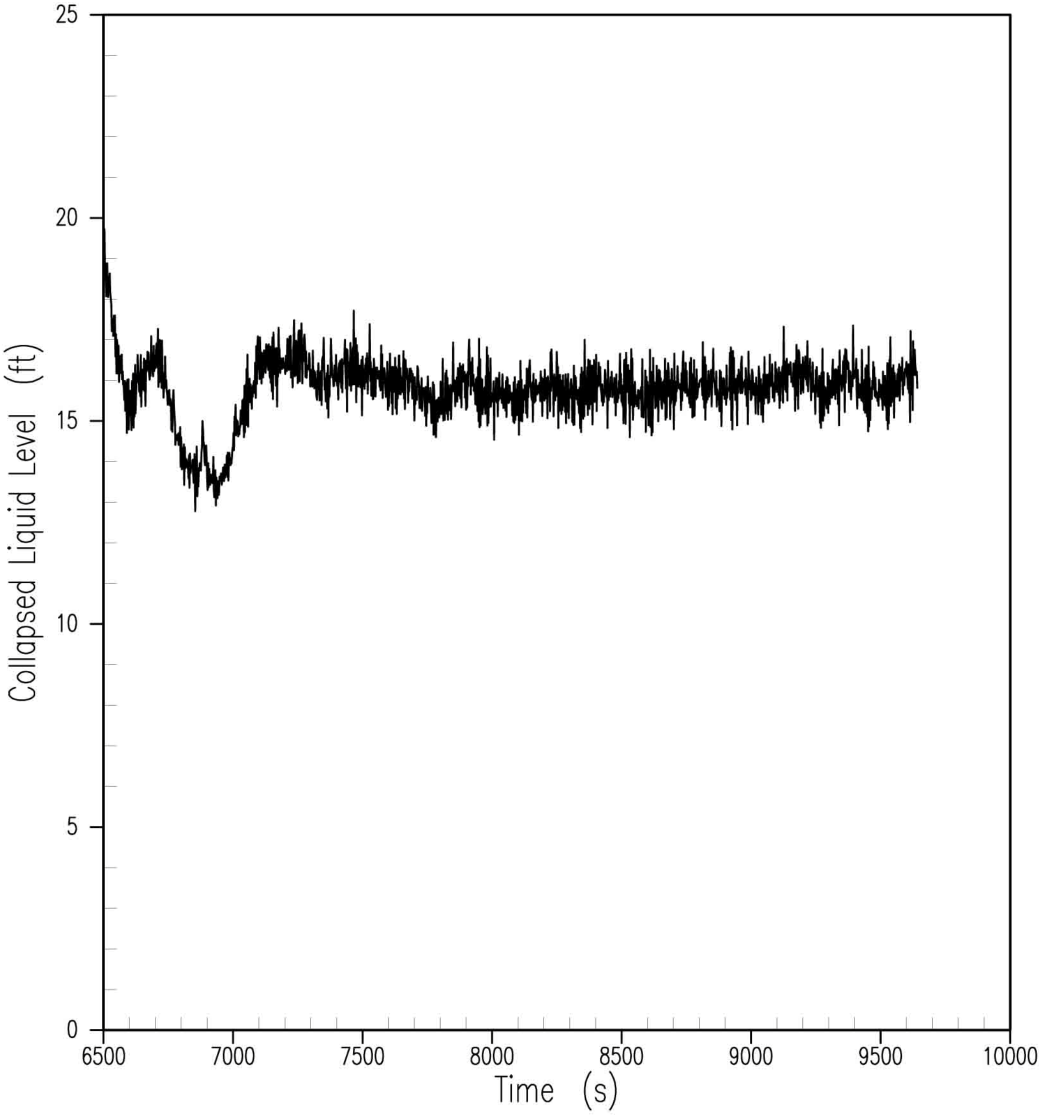

Figure 15.6.5.4C-1A
Collapsed Level of Liquid in the Downcomer
(DEDVI Case) – 14.7 psi

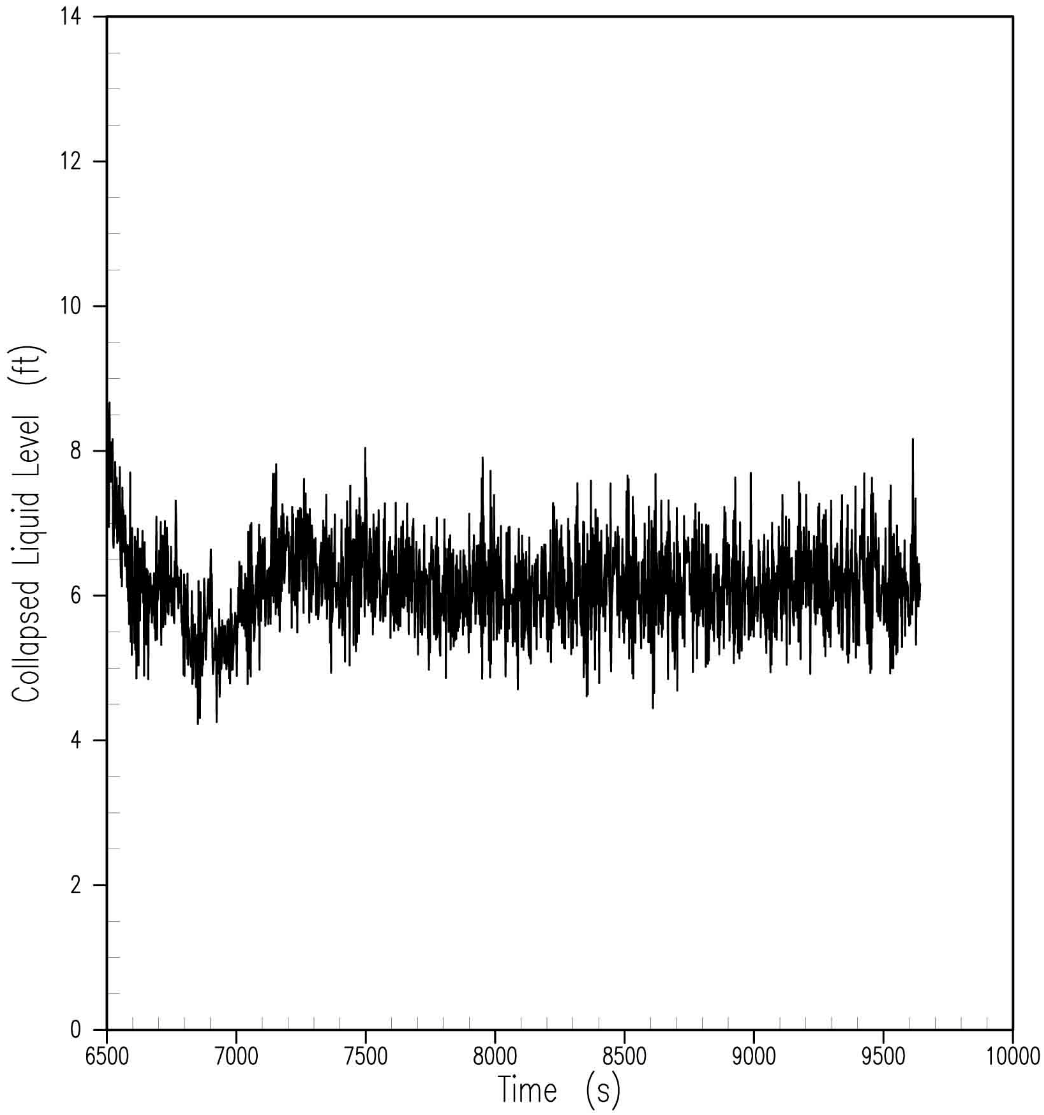

Figure 15.6.5.4C-2A
Collapsed Level of Liquid over the Heated Length of the Fuel
(DEDVI Case) – 14.7 psi

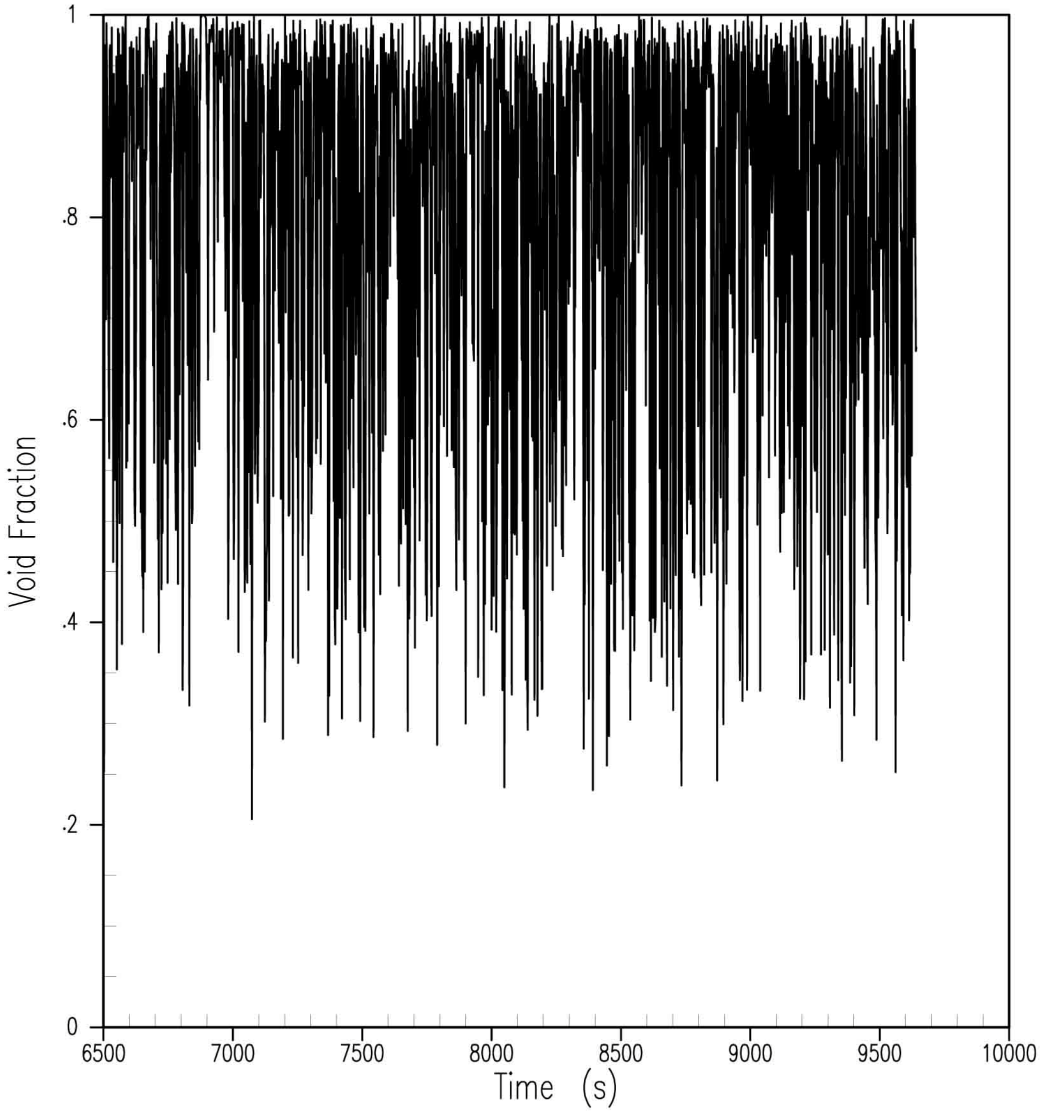

Figure 15.6.5.4C-3A
Void Fraction in Core Hot Assembly Top Cell
(DEDVI Case) – 14.7 psi

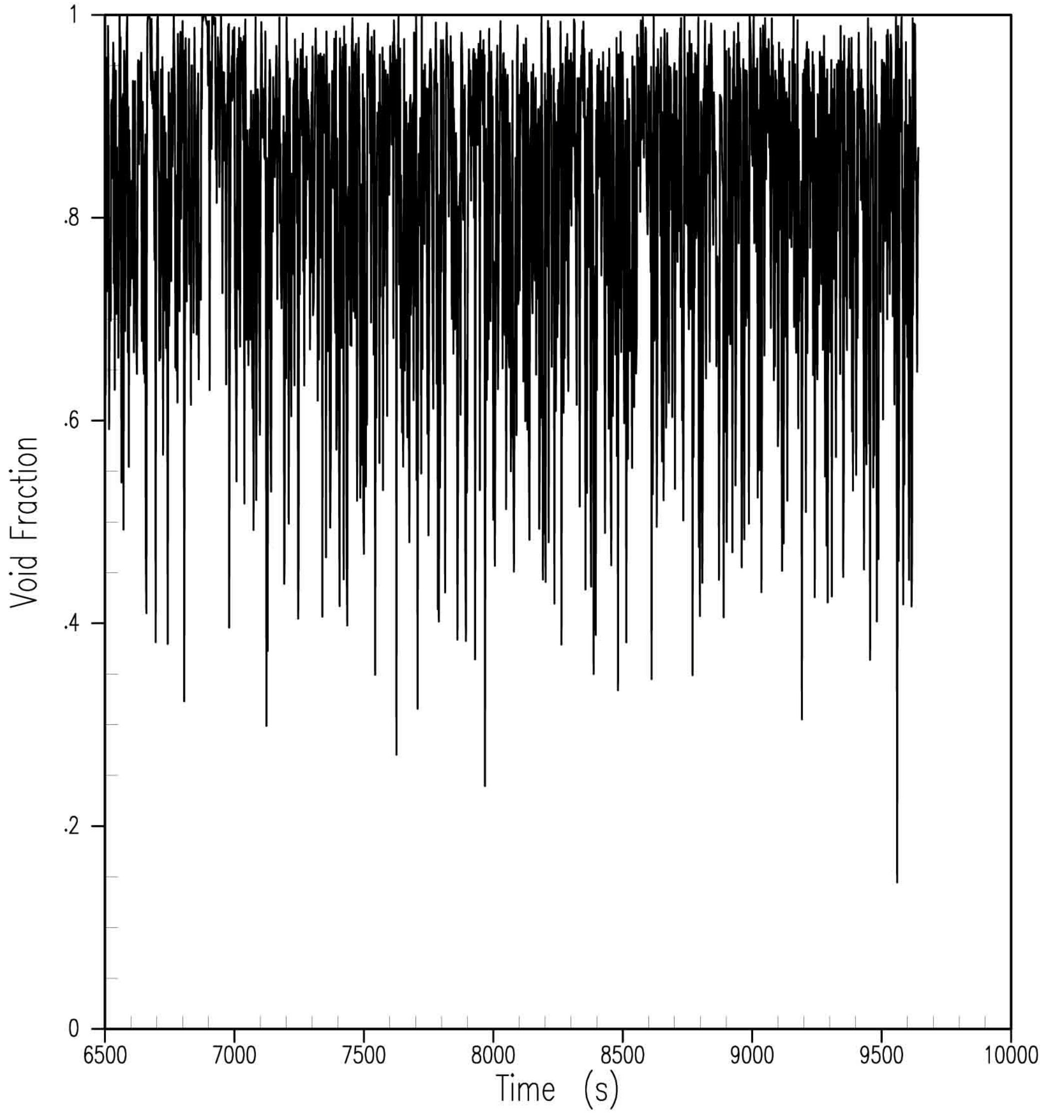

Figure 15.6.5.4C-4A
Void Fraction in Core Hot Assembly Second from Top Cell
(DEDVI Case) – 14.7 psi

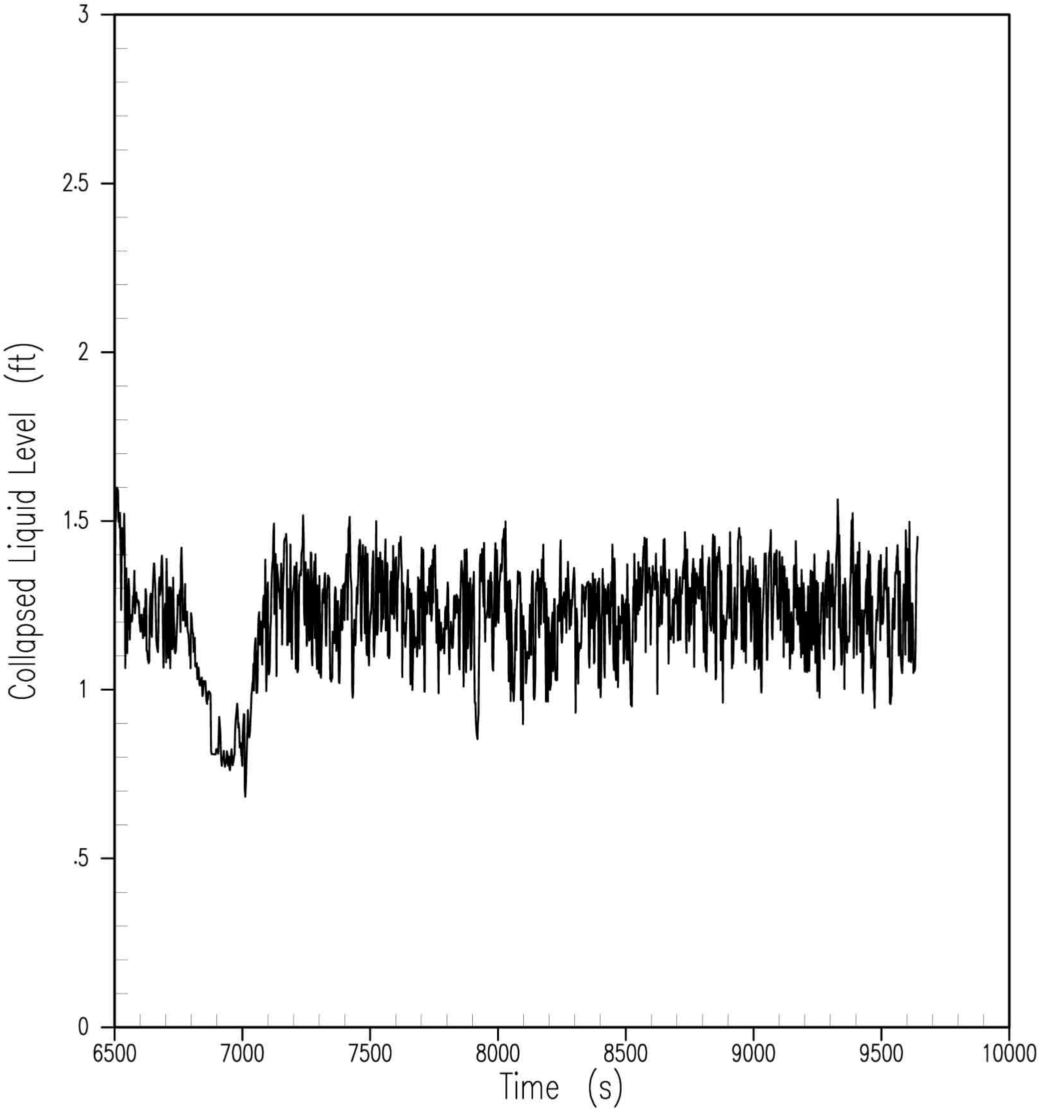

Figure 15.6.5.4C-5A
Collapsed Liquid Level in the Hot Leg
of Pressurizer Loop (DEDVI Case) – 14.7 psi

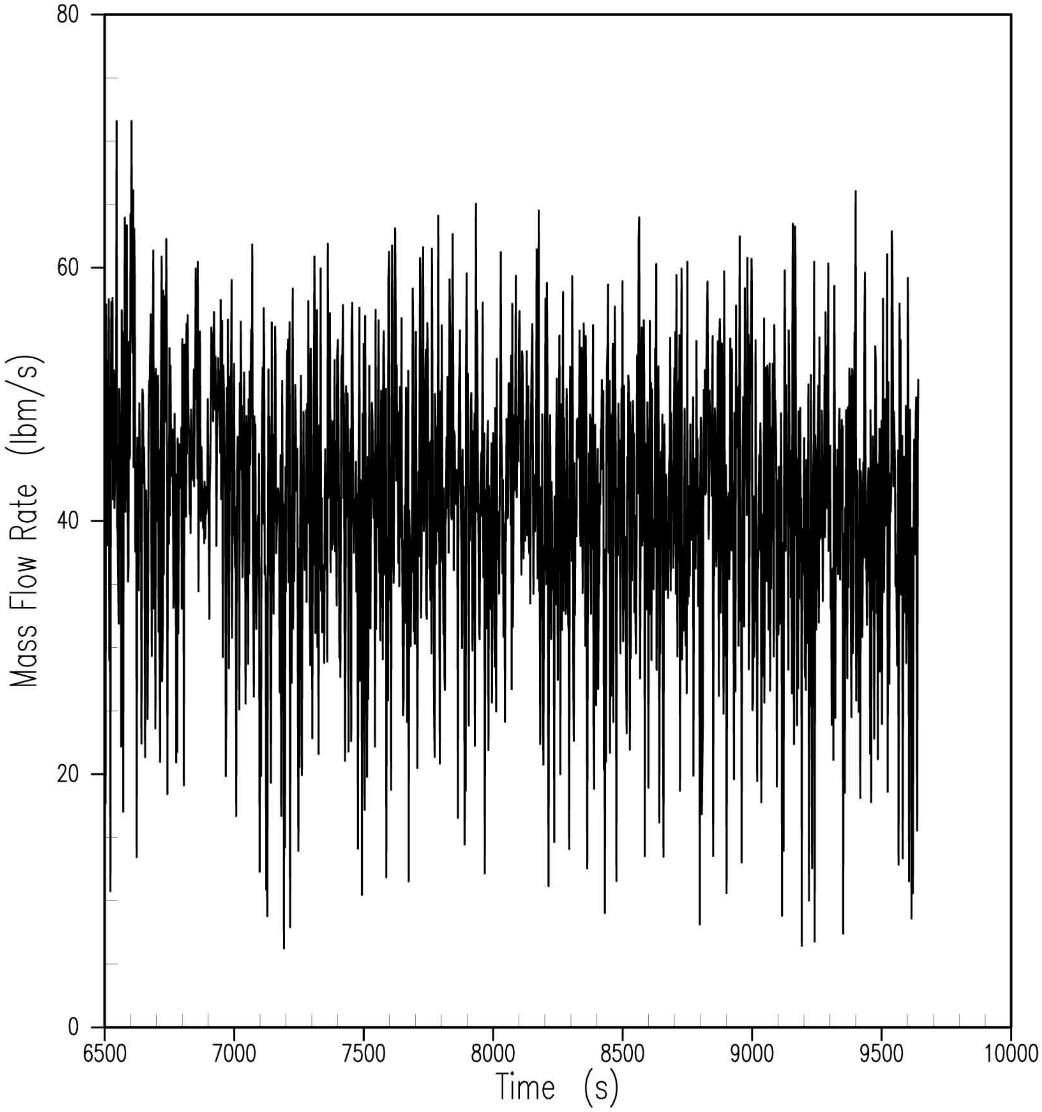

Figure 15.6.5.4C-6A
Vapor Rate out of the Core
(DEDVI Case) – 14.7 psi

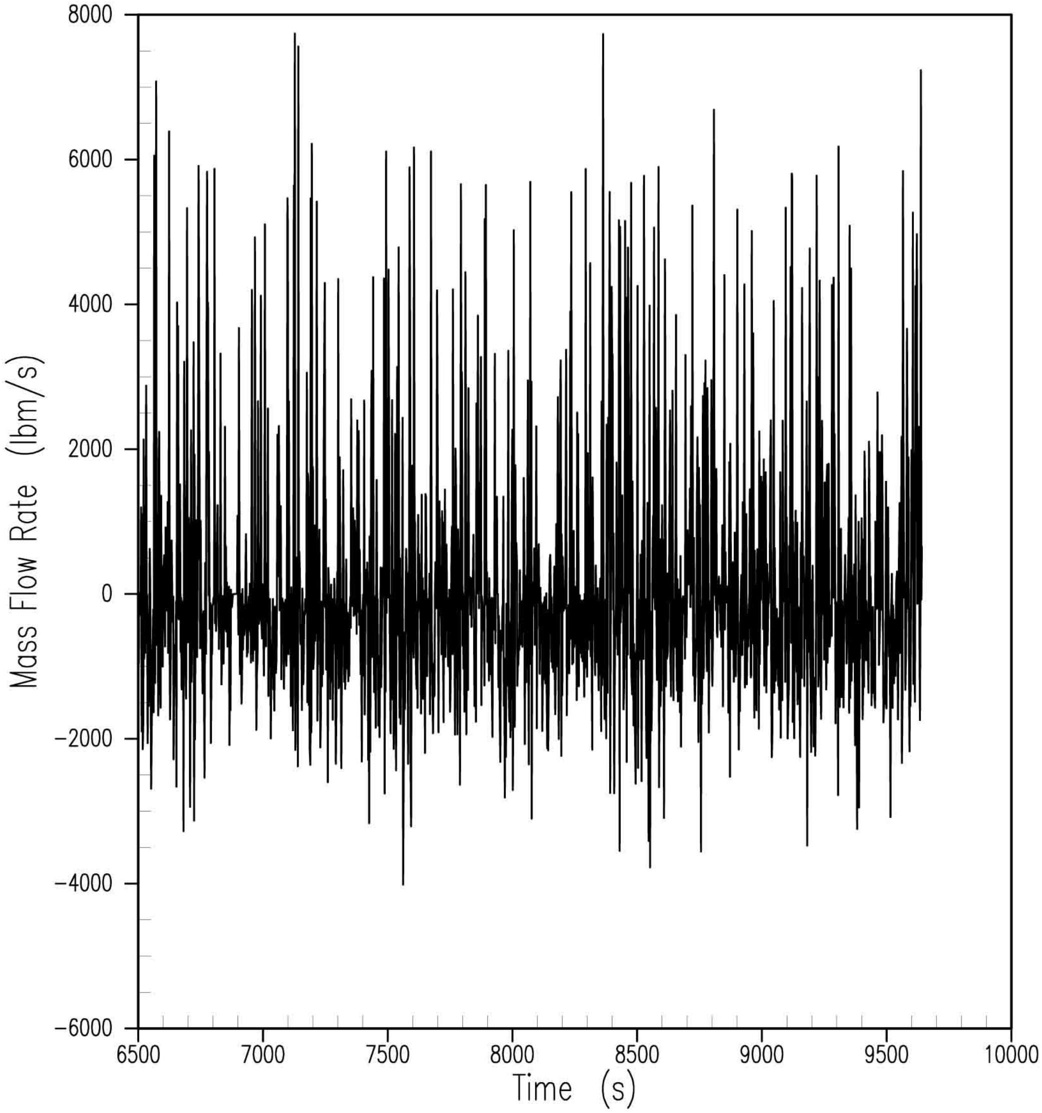

Figure 15.6.5.4C-7A
Liquid Flow Rate out of the Core
(DEDVI Case) – 14.7 psi

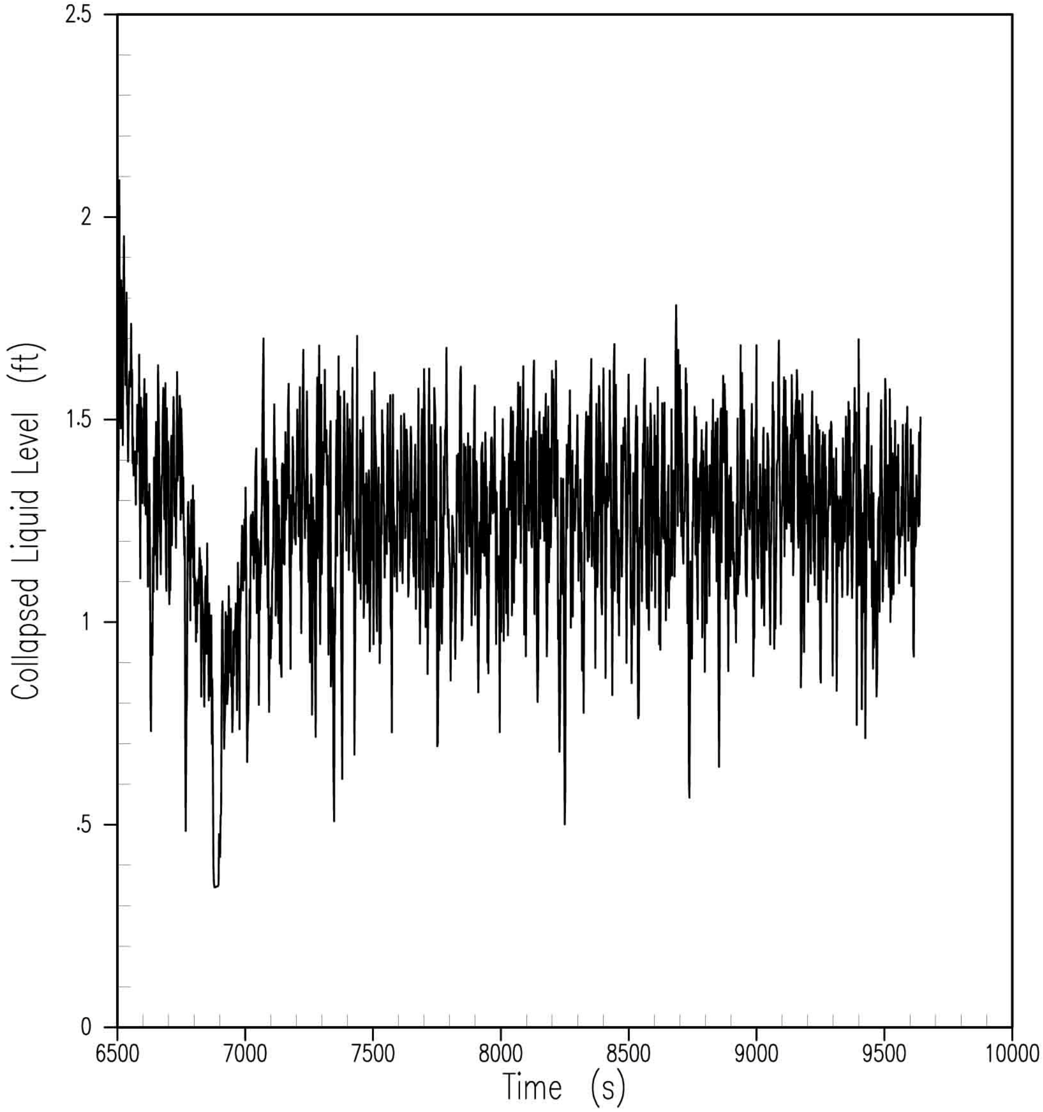

Figure 15.6.5.4C-8A
Collapsed Liquid Level in the Upper Plenum
(DEDVI Case) – 14.7 psi

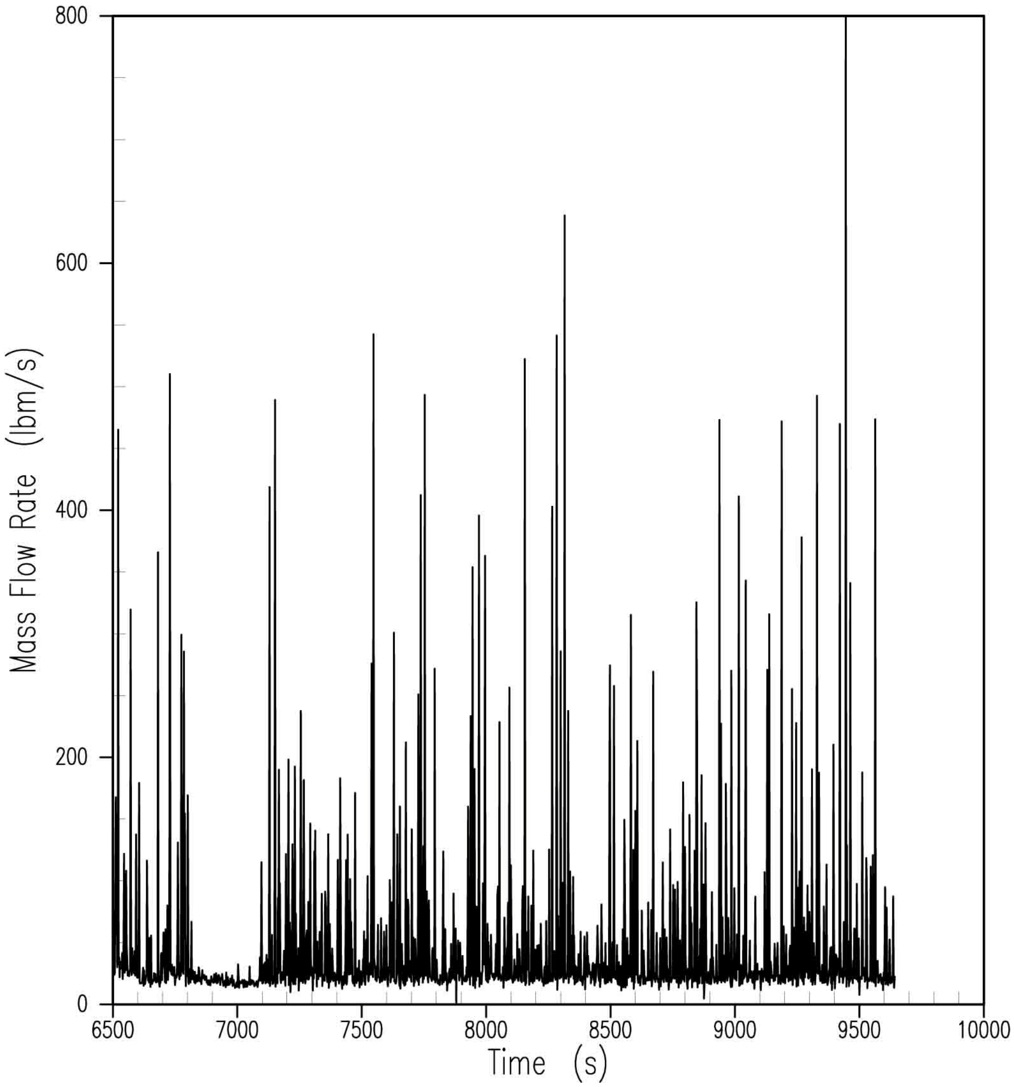

Figure 15.6.5.4C-9A
Mixture Flow Rate Through ADS Stage 4A Valves
(DEDVI Case) – 14.7 psi

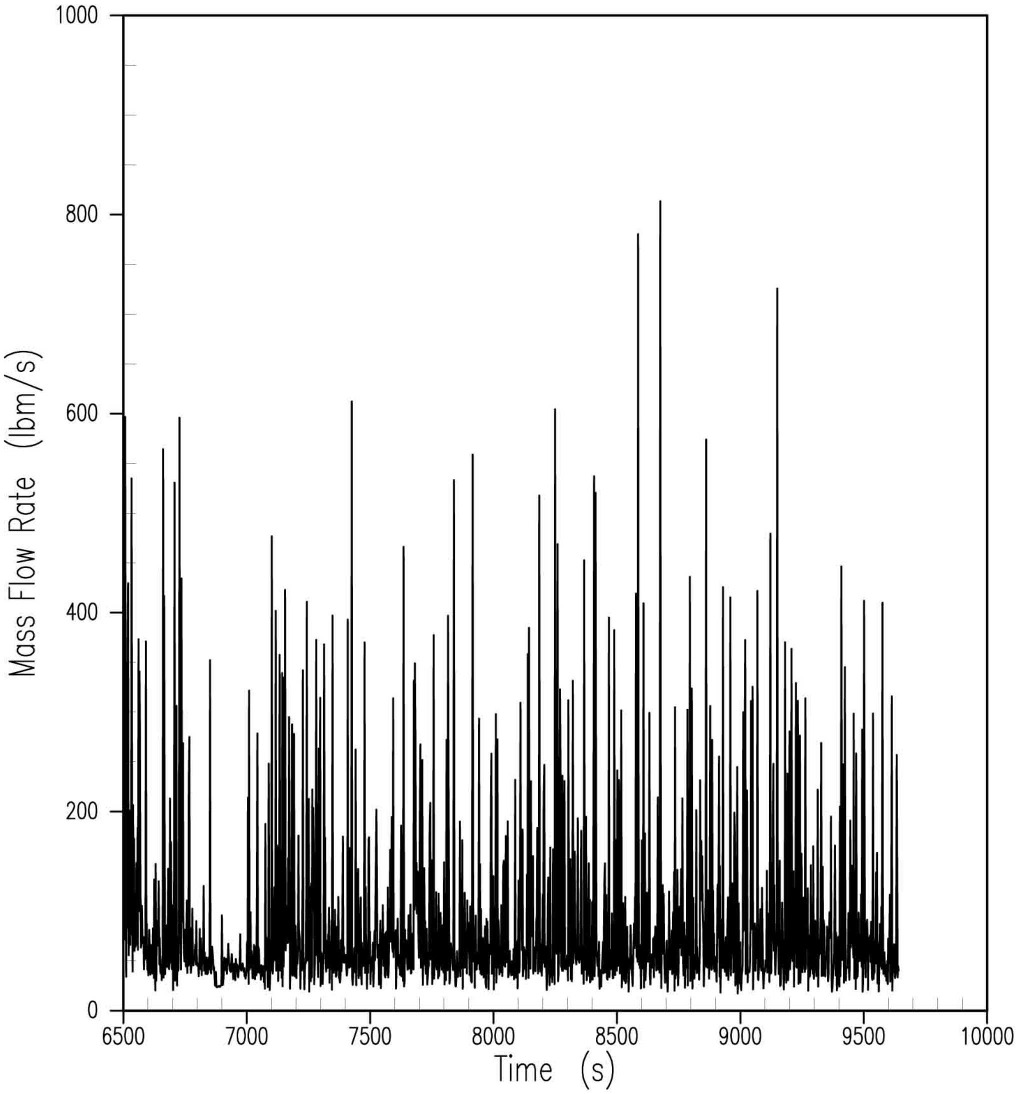

Figure 15.6.5.4C-10A
Mixture Flow Rate Through ADS Stage 4B Valves
(DEDVI Case) – 14.7 psi

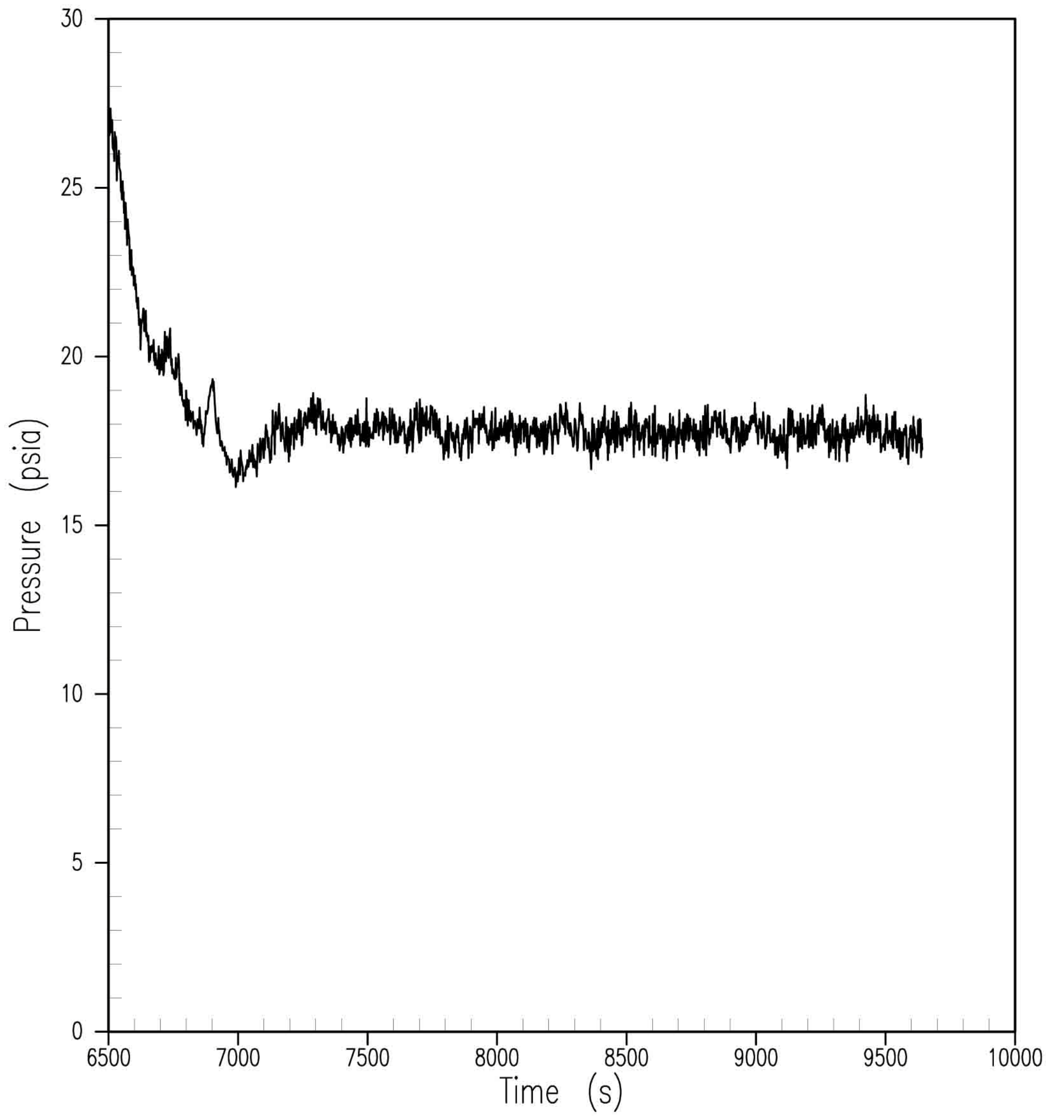

Figure 15.6.5.4C-11A
Upper Plenum Pressure
(DEDVI Case) – 14.7 psi

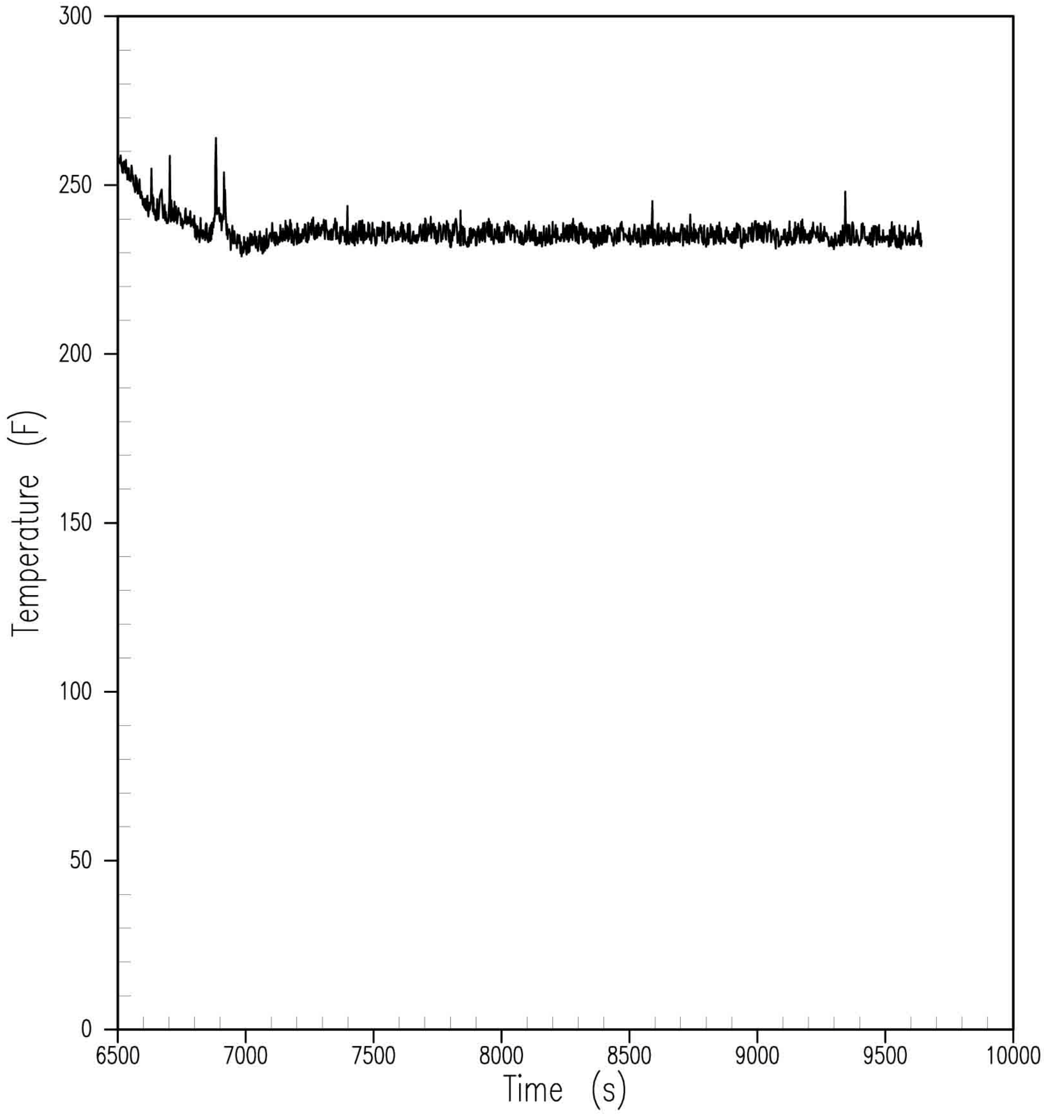

Figure 15.6.5.4C-12A
Peak Cladding Temperature
(DEDVI Case) – 14.7 psi

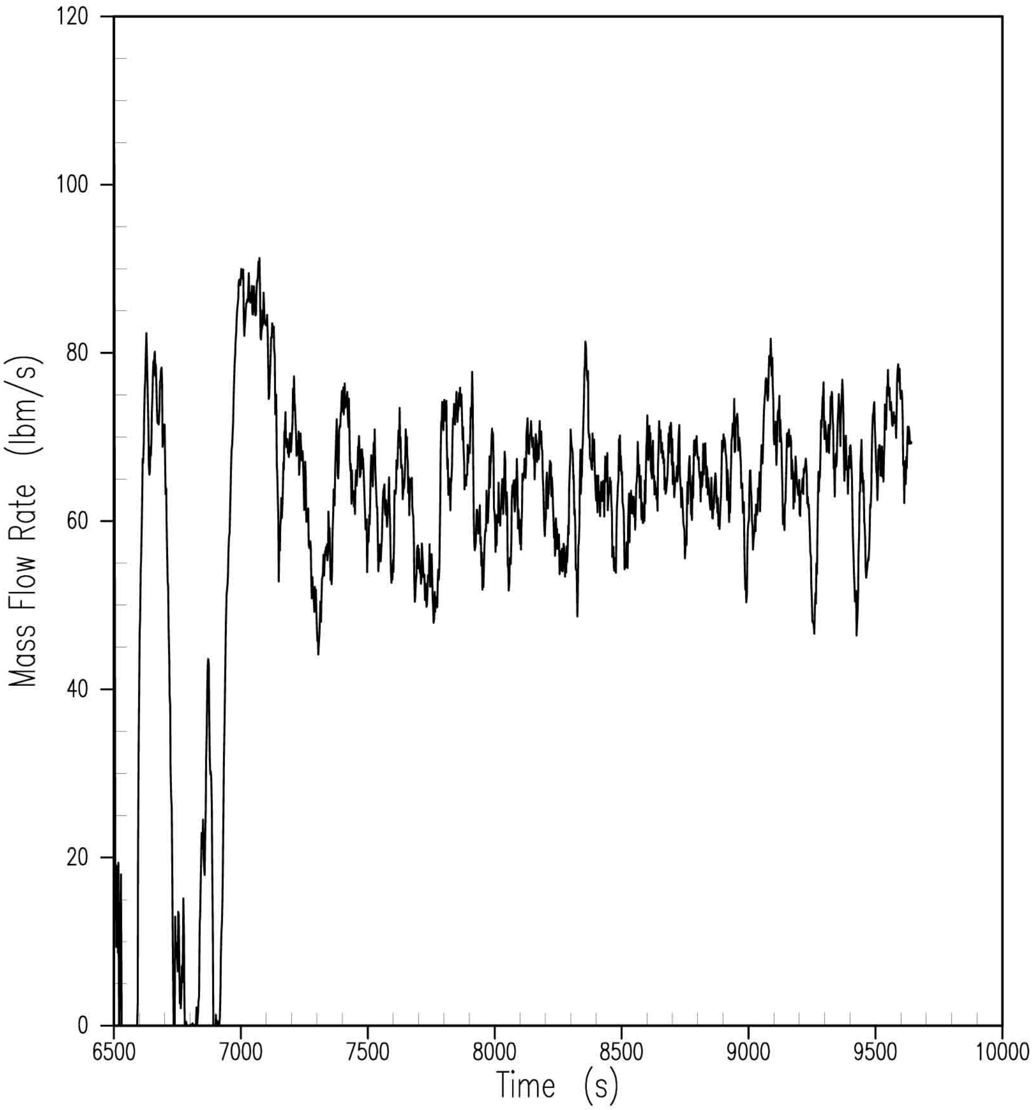

Figure 15.6.5.4C-13A
DVI–A Mixture Flow Rate
(DEDVI Case) – 14.7 psi

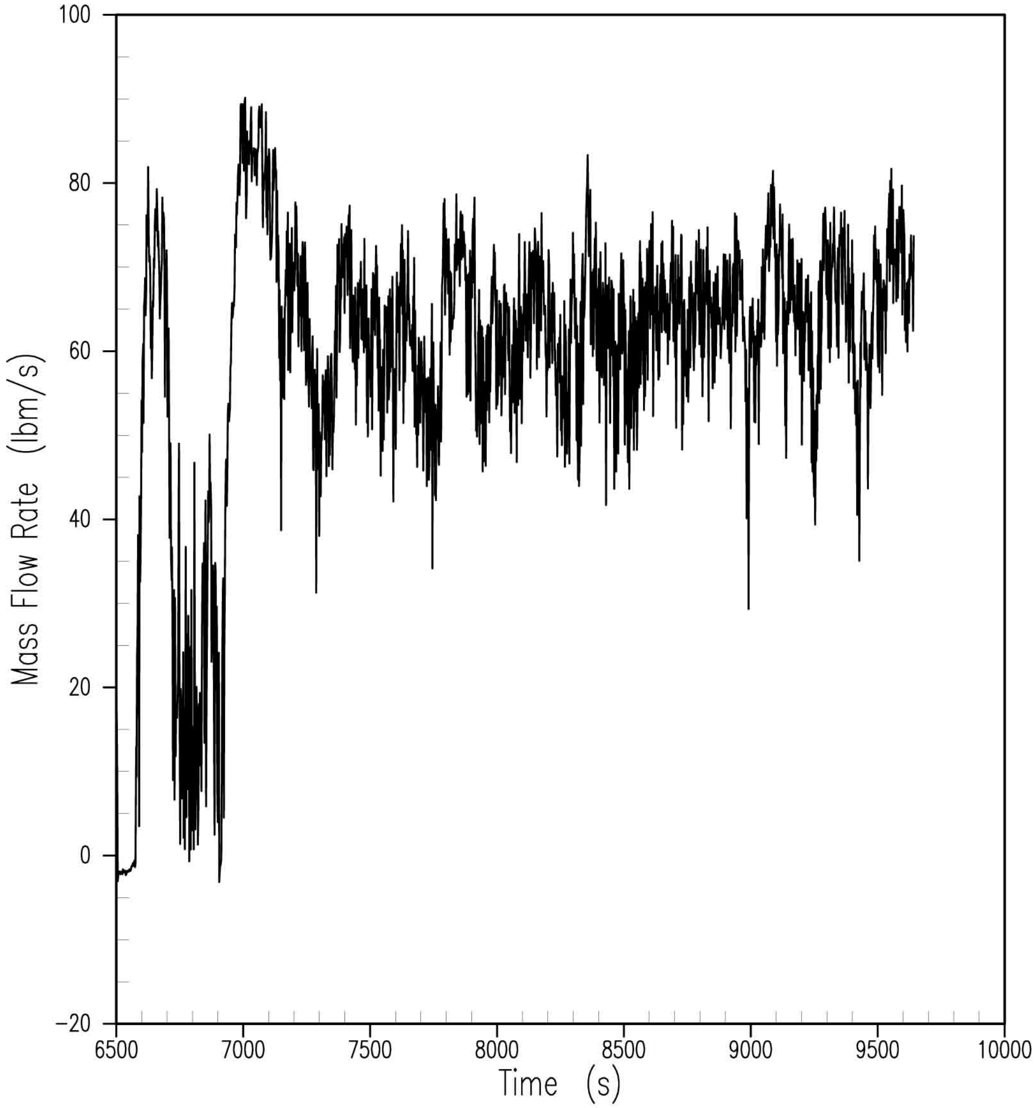

Figure 15.6.5.4C-14A
DVI–B Mixture Flow Rate
(DEDVI Case) – 14.7 psi

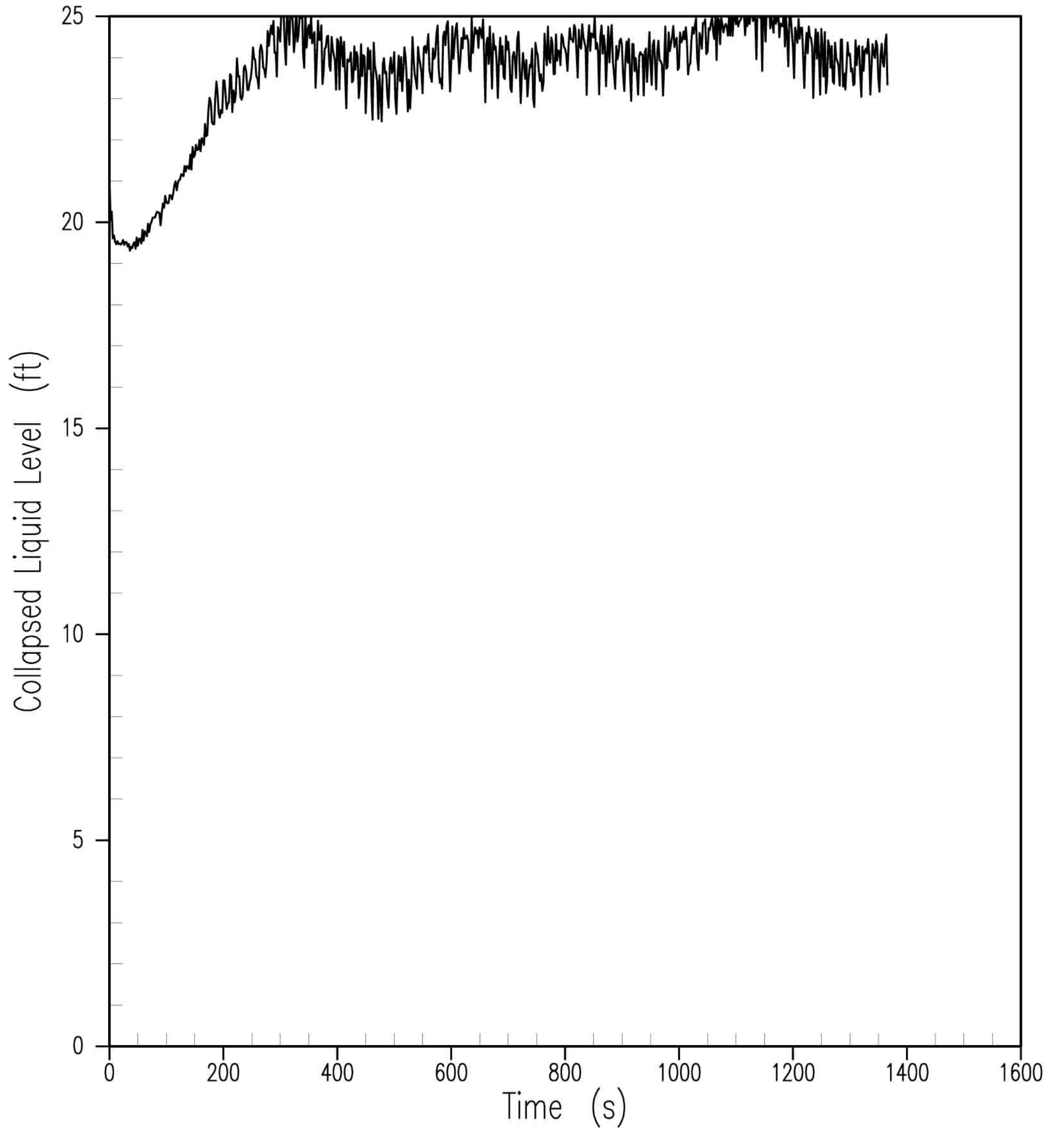

Figure 15.6.5.4C-15
Collapsed Level of Liquid in the Downcomer
(Wall-to-Wall Floodup Case) – 14.7 psi

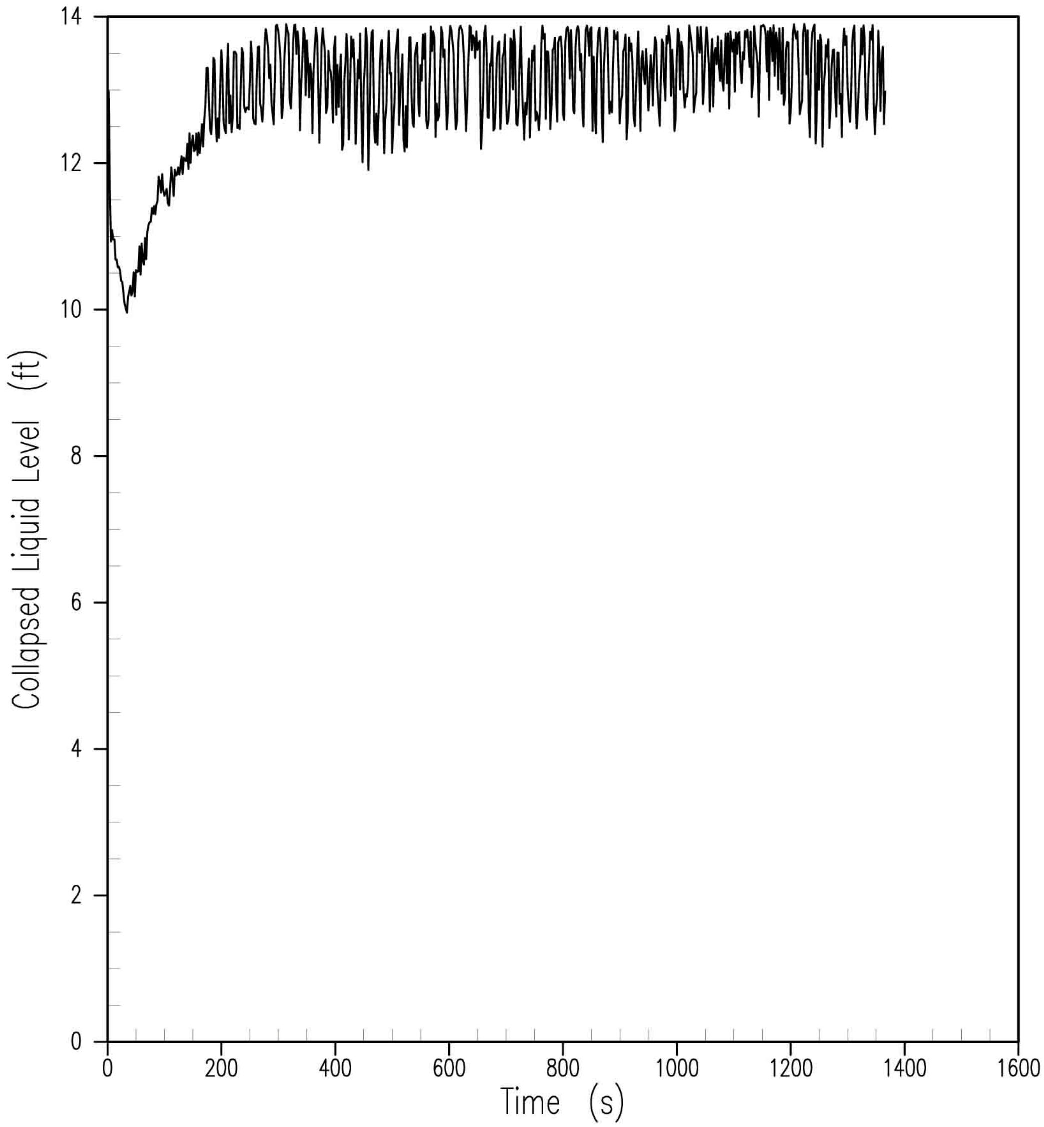

Figure 15.6.5.4C-16
Collapsed Level of Liquid Over the Heated Length of the Fuel
(Wall-to-Wall Floodup Case) – 14.7 psi

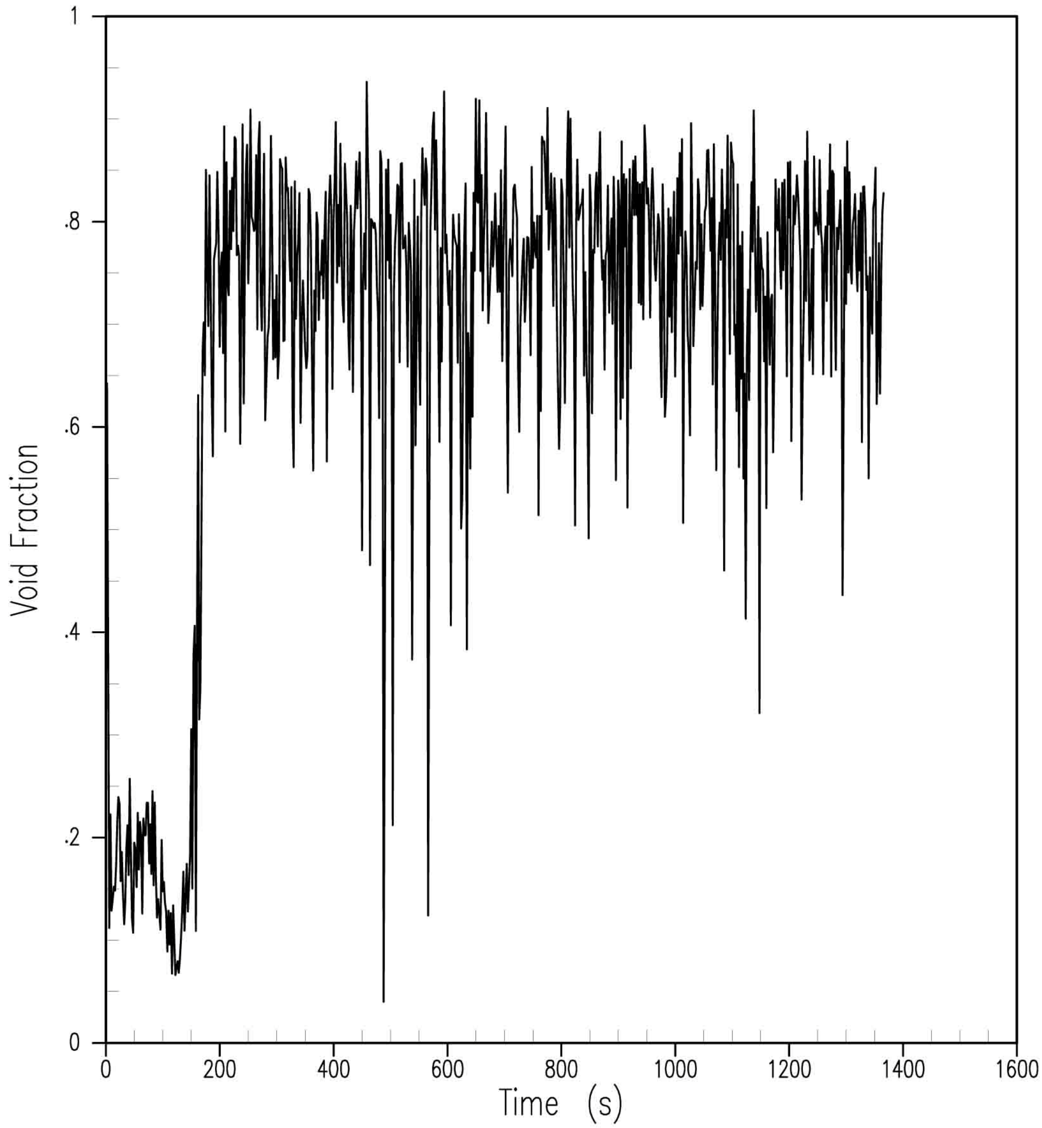

Figure 15.6.5.4C-17
Void Fraction in Core Hot Assembly Top Cell
(Wall-to-Wall Floodup Case) – 14.7 psi

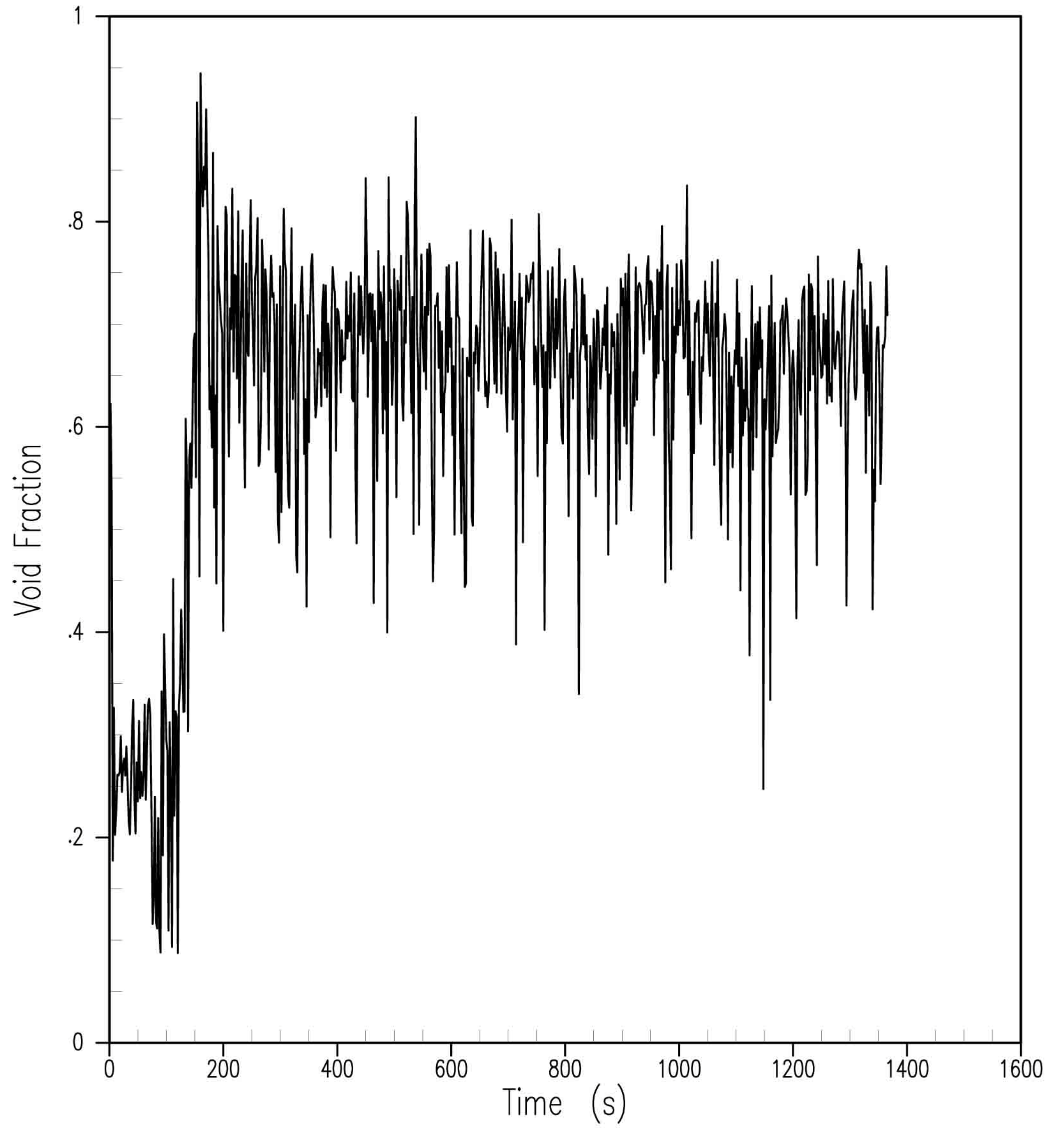

Figure 15.6.5.4C-18
Void Fraction in Core Hot Assembly Second from Top Cell
(Wall-to-Wall Floodup Case) – 14.7 psi

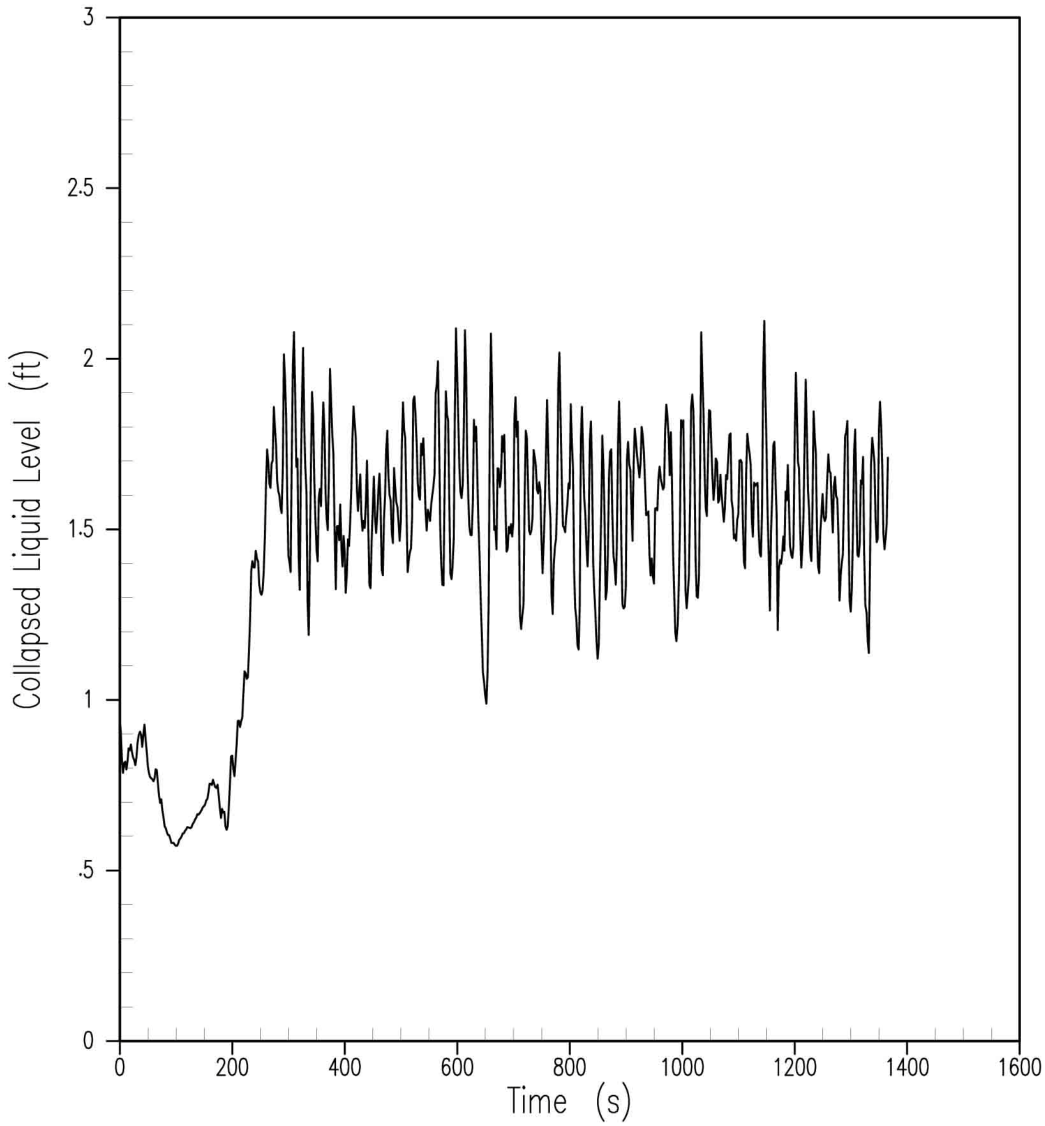

Figure 15.6.5.4C-19
Collapsed Liquid Level in the Hot Leg of Pressurizer Loop
(Wall-to-Wall Floodup Case) – 14.7 psi

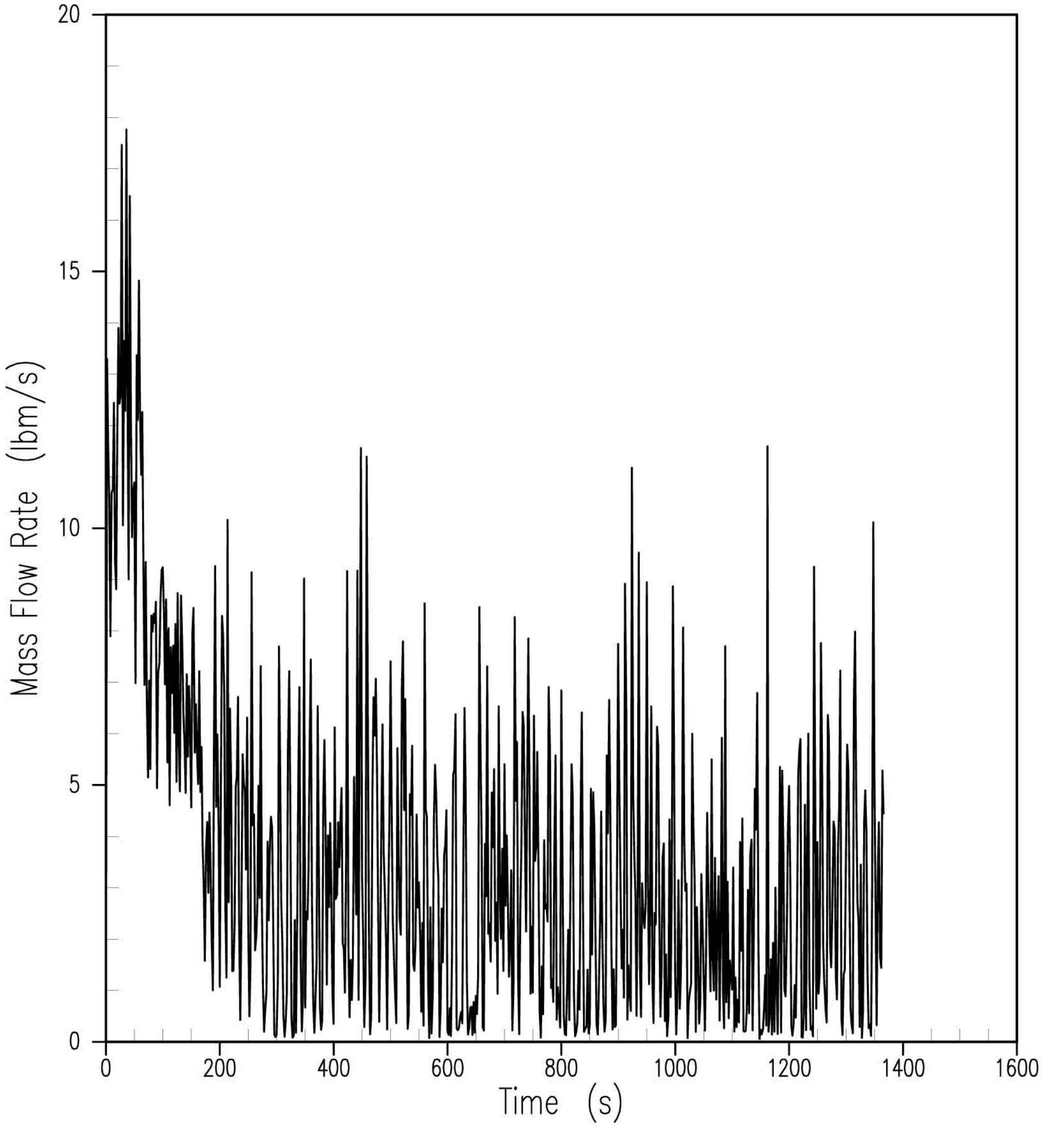

Figure 15.6.5.4C-20
Vapor Rate out of the Core
(Wall-to-Wall Floodup Case) – 14.7 psi

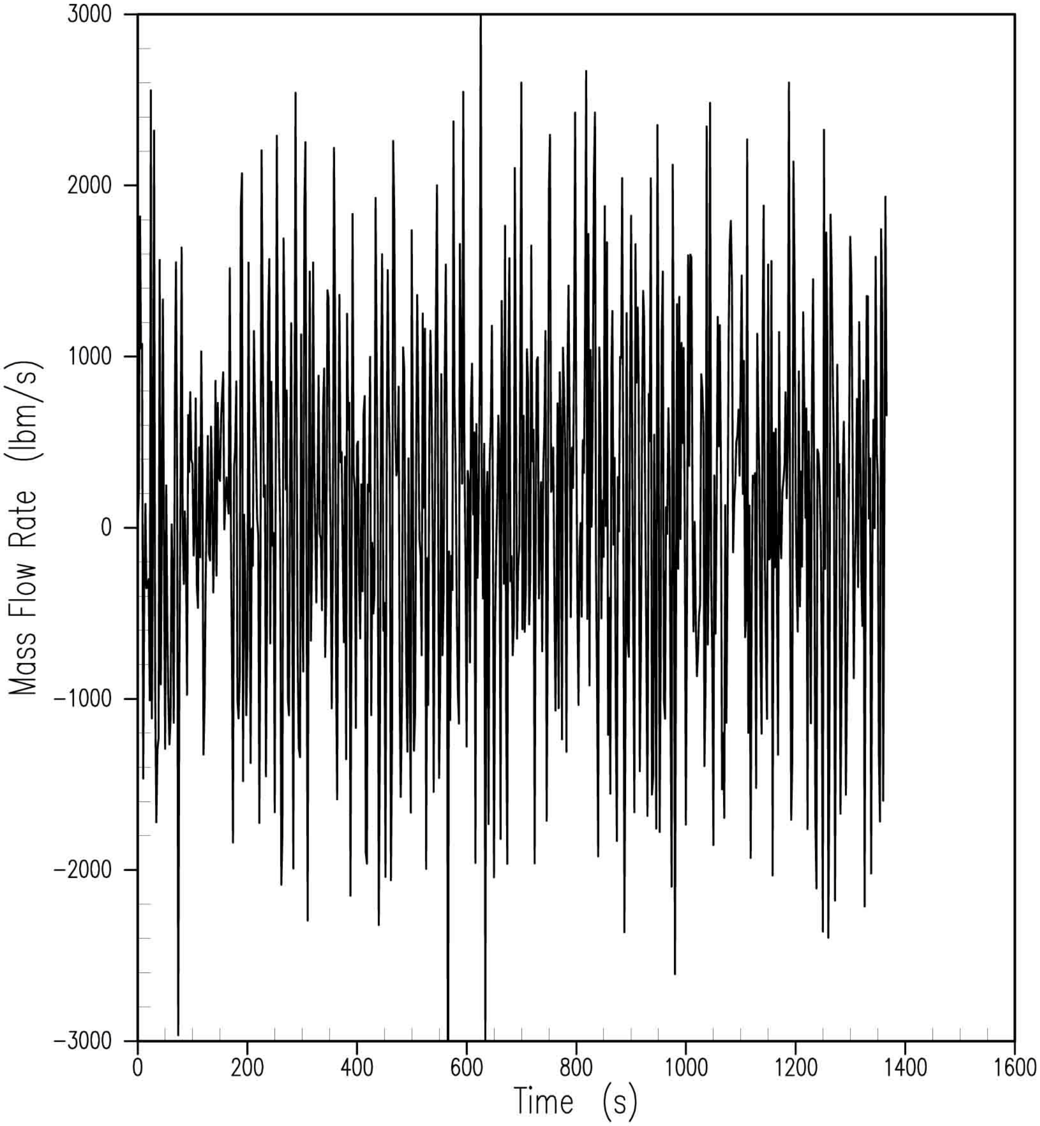

Figure 15.6.5.4C-21
Liquid Flow Rate out of the Core
(Wall-to-Wall Floodup Case) – 14.7 psi

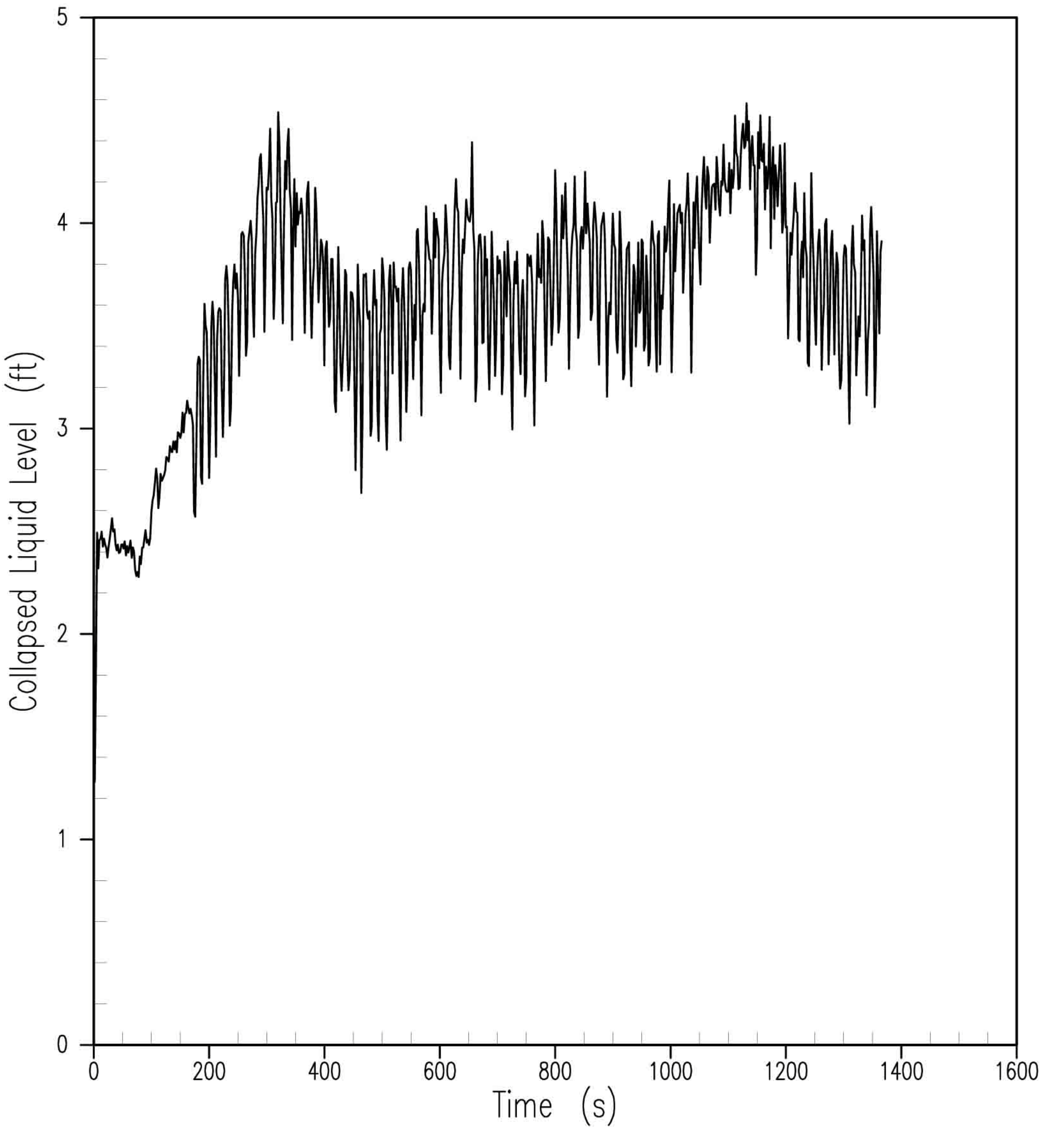

Figure 15.6.5.4C-22
Collapsed Liquid Level in the Upper Plenum
(Wall-to-Wall Floodup Case) – 14.7 psi

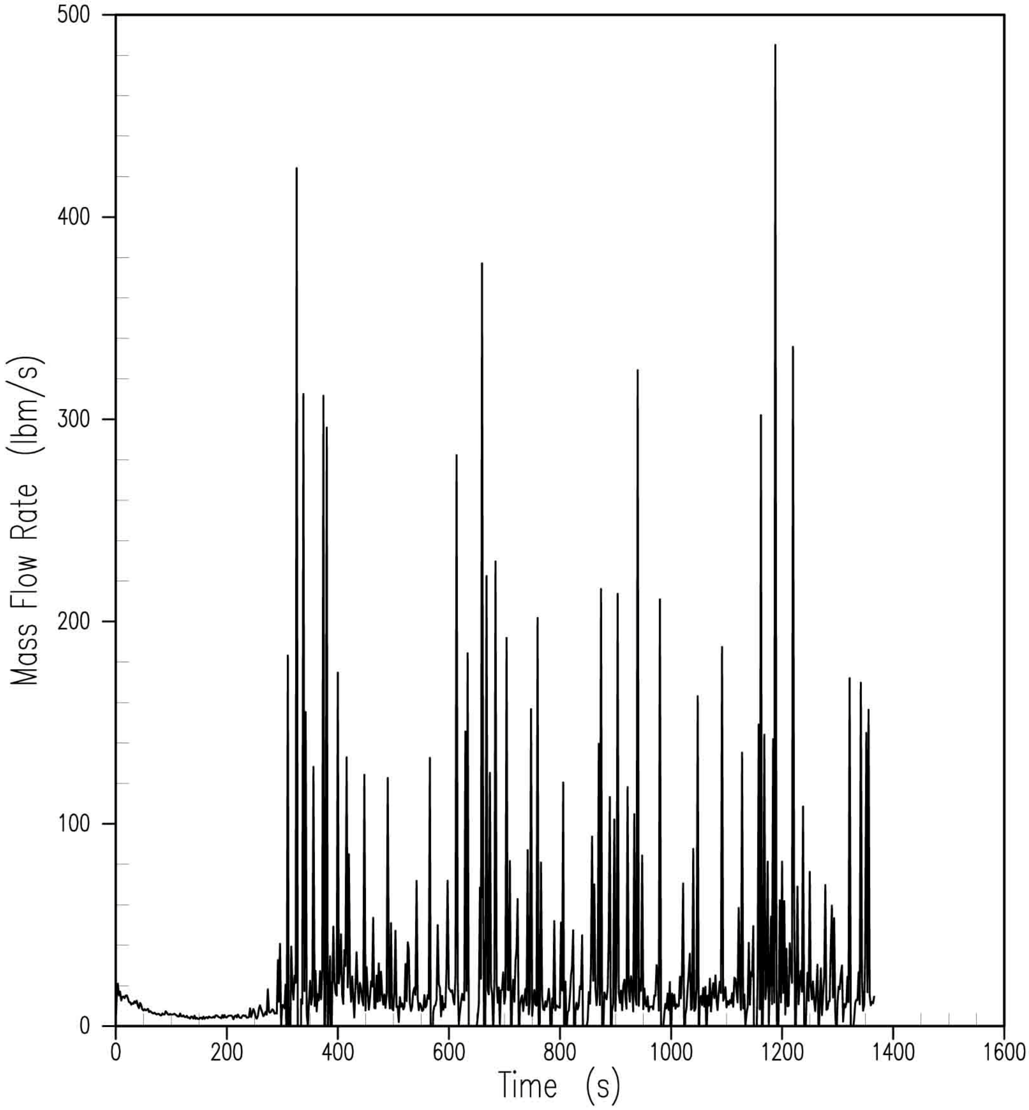

Figure 15.6.5.4C-23
Mixture Flow Rate Through ADS Stage 4A Valves
(Wall-to-Wall Floodup Case) – 14.7 psi

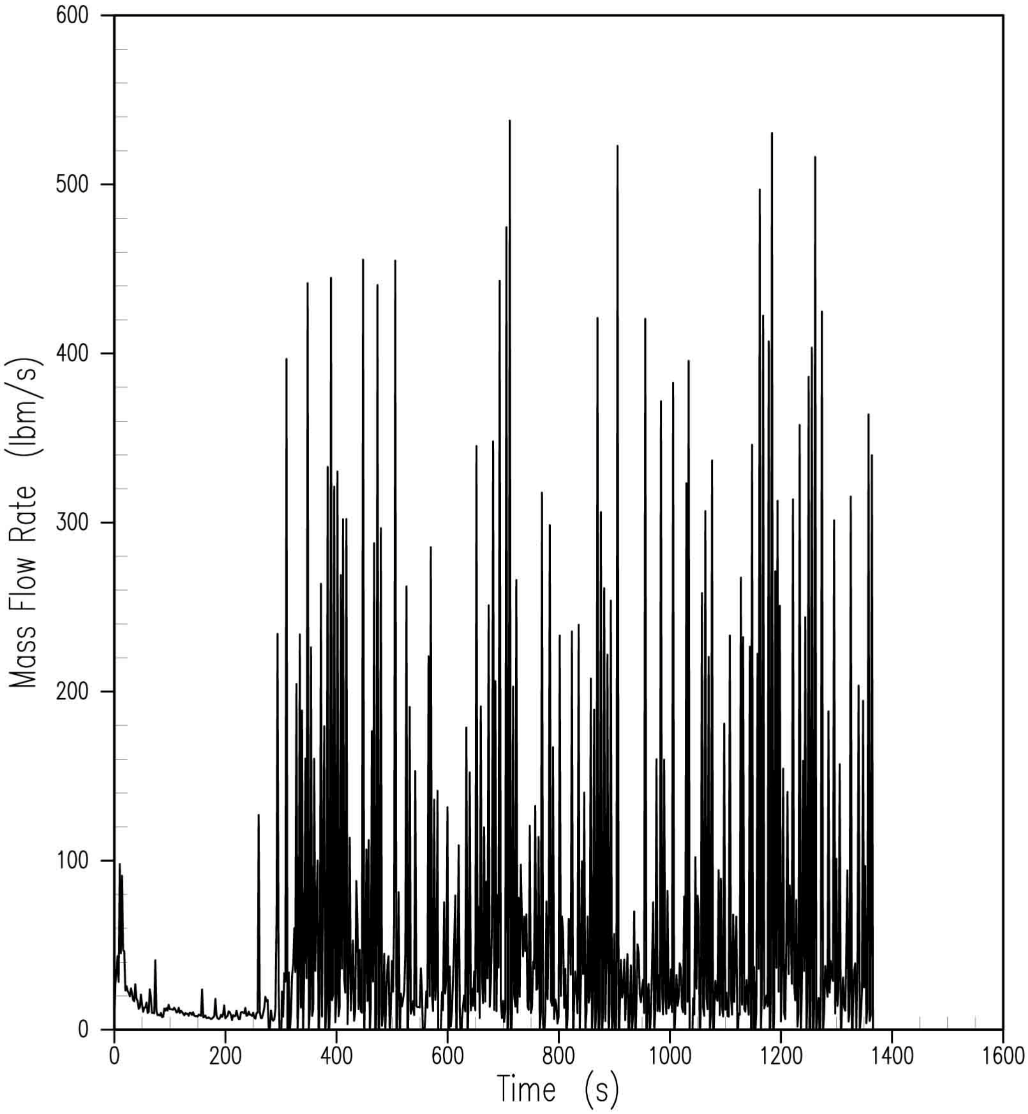

Figure 15.6.5.4C-24
Mixture Flow Rate Through ADS Stage 4B Valves
(Wall-to-Wall Floodup Case) – 14.7 psi

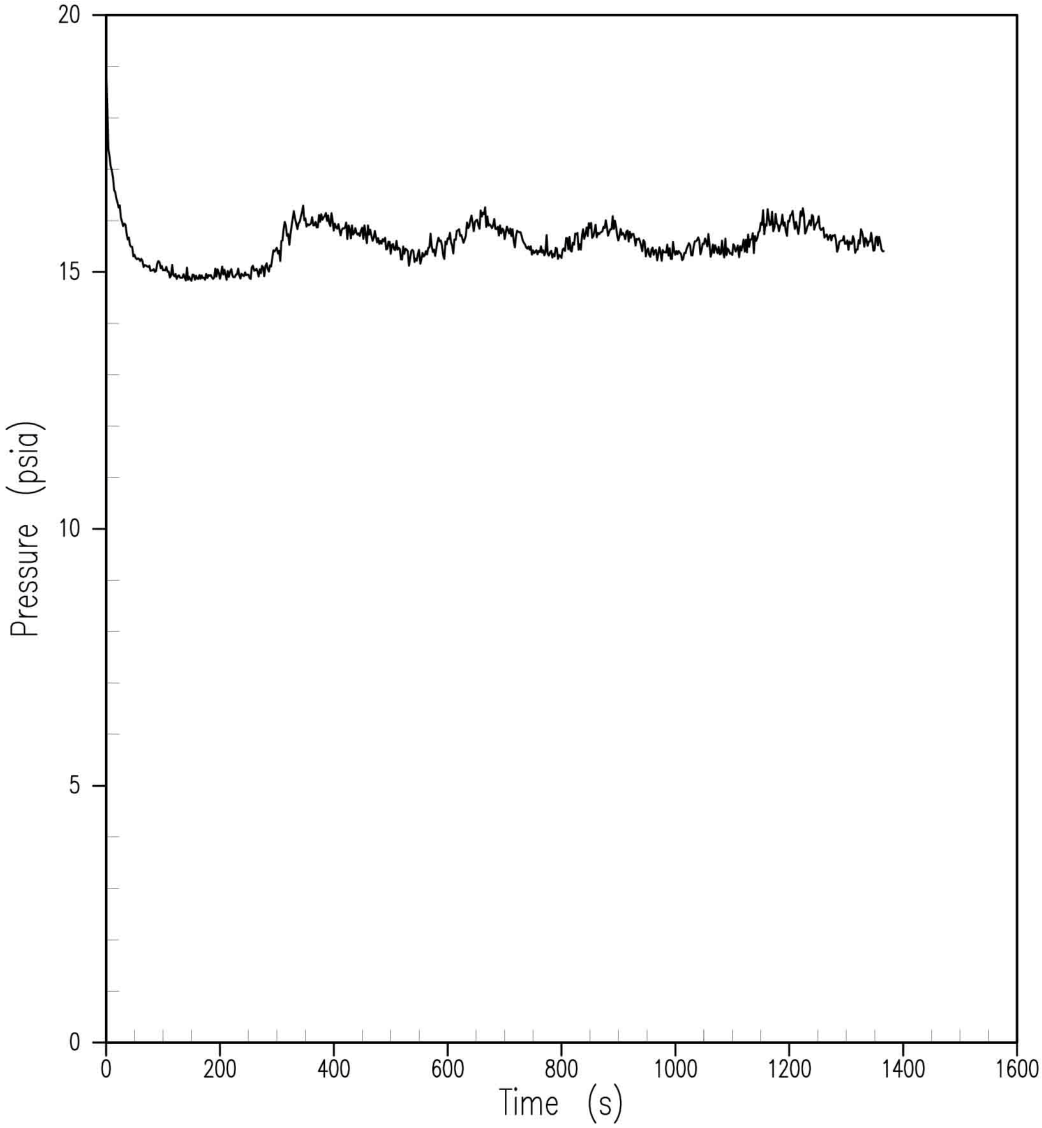

Figure 15.6.5.4C-25
Upper Plenum Pressure
(Wall-to-Wall Floodup Case) – 14.7 psi

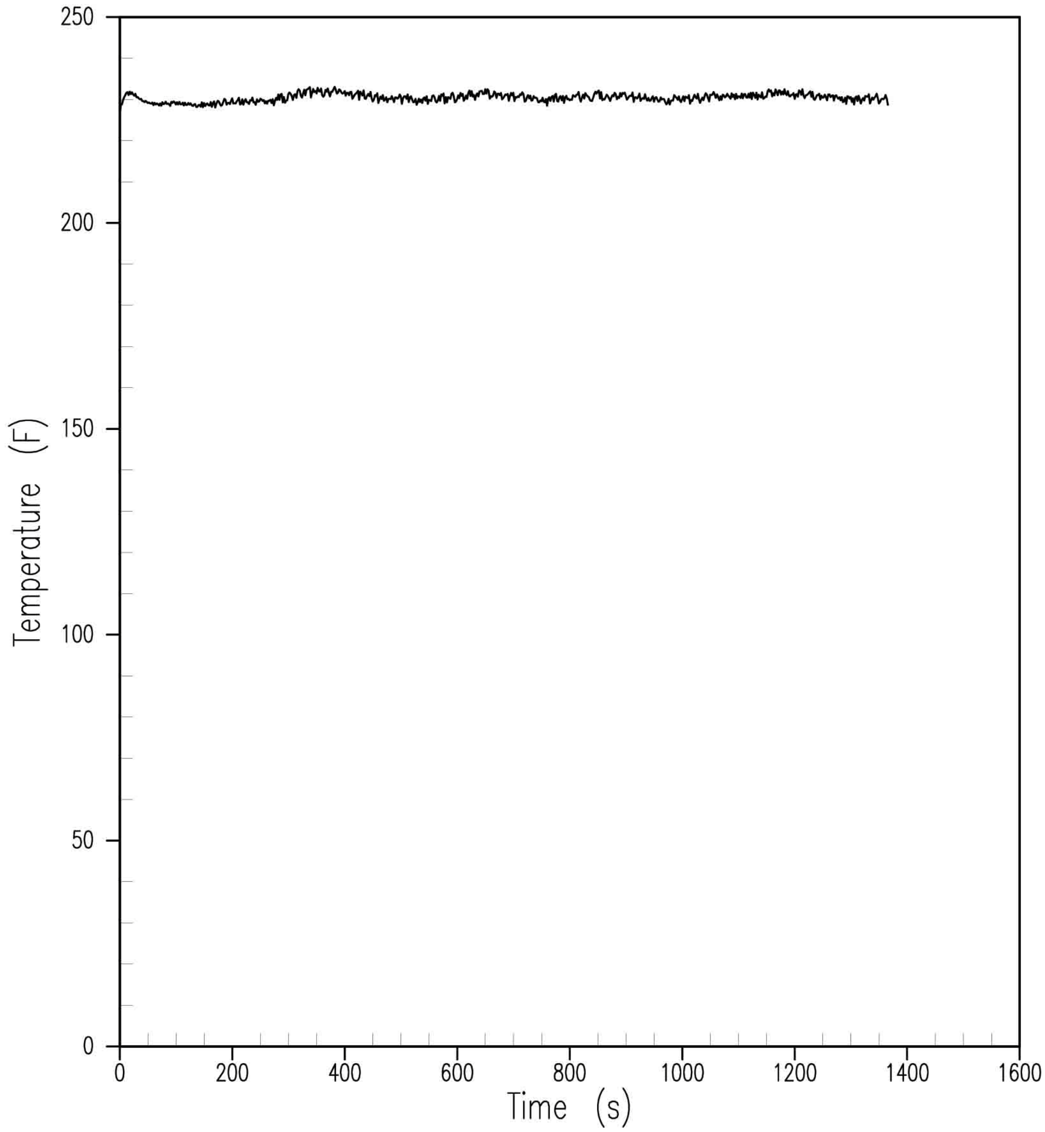

Figure 15.6.5.4C-26
Hot Rod Cladding Temperature Near Top of Core
(Wall-to-Wall Floodup Case) – 14.7 psi

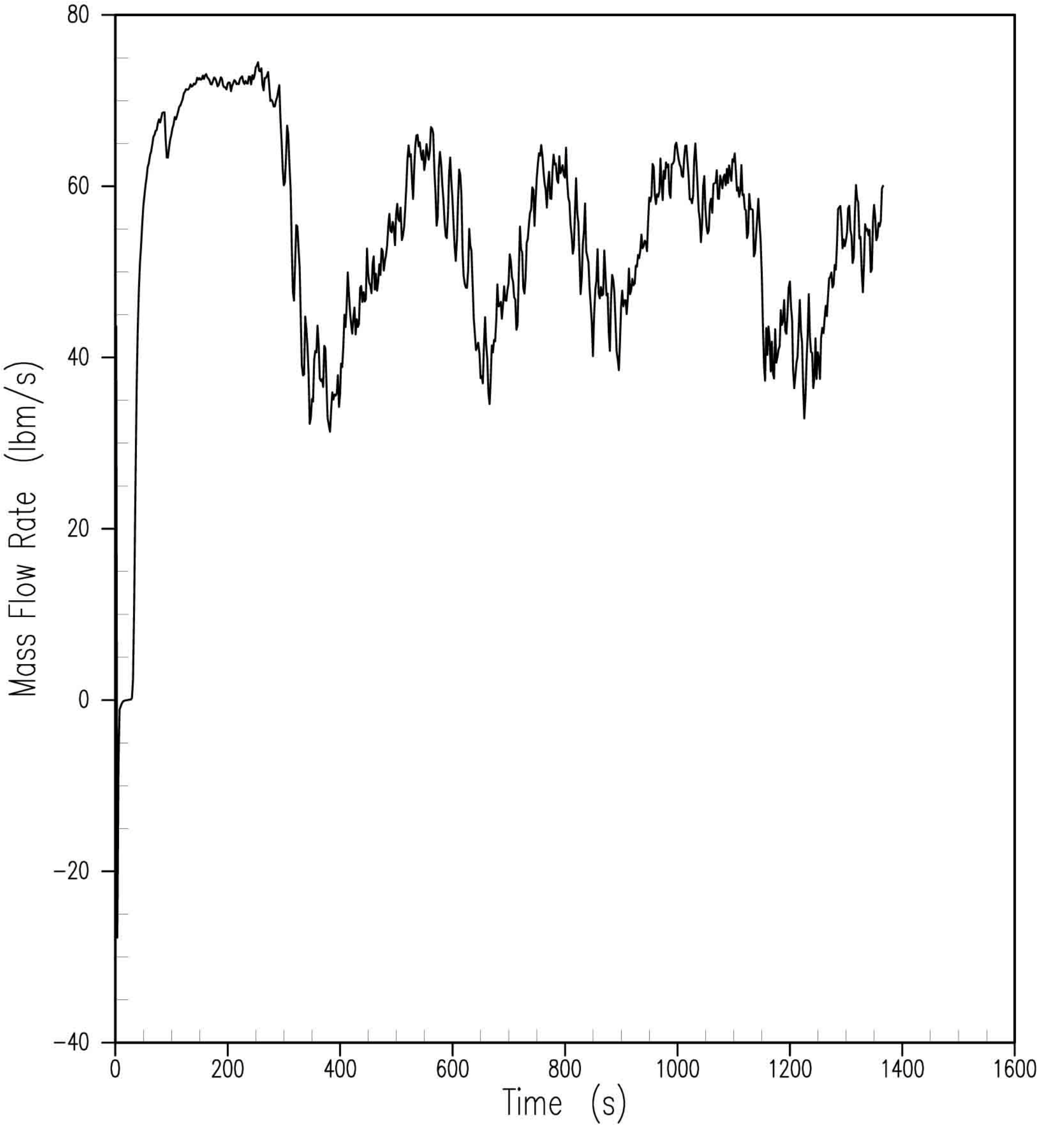

Figure 15.6.5.4C-27
DVI-A Mixture Flow Rate
(Wall-to-Wall Floodup Case) – 14.7 psi

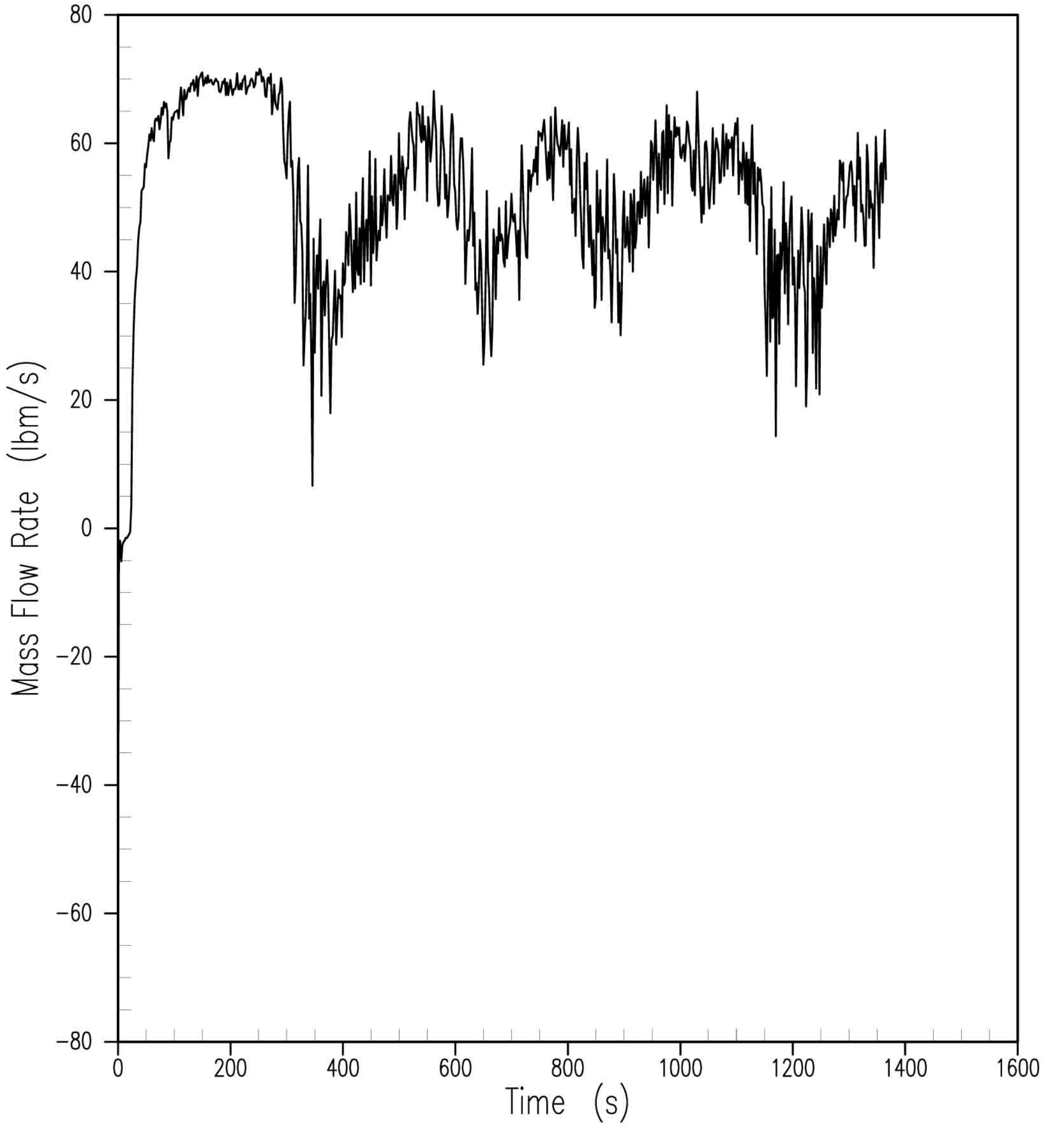

Figure 15.6.5.4C-28
DVI-B Mixture Flow Rate
(Wall-to-Wall Floodup Case) – 14.7 psi

15.7 Radioactive Release from a Subsystem or Component

This group of events includes the following:

- Gas waste management system leak or failure
- Liquid waste management system leak or failure (atmospheric release)
- Release of radioactivity to the environment via liquid pathways
- Fuel handling accident
- Spent fuel cask drop accident

15.7.1 Gas Waste Management System Leak or Failure

The AP1000 gaseous radwaste system is a low-pressure, low-flow charcoal delay process. Failure of the gaseous radwaste system results in a minor release of activity that is not significant. The Standard Review Plan no longer includes this event as part of the review. Therefore, no analysis is provided.

15.7.2 Liquid Waste Management System Leak or Failure (Atmospheric Release)

The AP1000 liquid radwaste system tanks do not contain significant levels of gaseous activity because liquids expected to contain gaseous radioactivity are processed by a gas stripper before being directed to storage. The tanks are open to the atmosphere so that any evolution of gaseous activity is continually released through the monitored plant vent. The Standard Review Plan no longer includes this event as part of the review. Therefore, no analysis is provided.

15.7.3 Release of Radioactivity to the Environment Due to a Liquid Tank Failure

Tanks containing radioactive fluids are located inside plant structures.

In the event of a tank failure, the liquid would be drained by the floor drains to the auxiliary building sump. From the sump, the water would be directed to the waste holdup tank. The basemat of the auxiliary building is 6-feet thick, the exterior walls are 3-feet thick, and the building is seismic Category I. The exterior walls are sealed to prevent leakage. Thus, it is assumed that there is no release of the spilled liquid waste to the environment. However, the Standard Review Plan states that credit cannot be taken for liquid retention by unlined building foundations. Analysis of the impact of this event is performed as discussed in Subsection 2.4.13. This analysis includes consideration of tank liquid level, processing and decay of tank contents, potential paths of spilled waste to the environment, as well as other pertinent factors.

15.7.4 Fuel Handling Accident

A fuel handling accident can be postulated to occur either inside the containment or in the fuel handling area inside the auxiliary building. The fuel handling accident is defined as the dropping of a spent fuel assembly such that every rod in the dropped assembly has its cladding breached so that the activity in the fuel/cladding gap is released.

The possibility of a fuel handling accident is remote because of the many administrative controls and the equipment operating limits that are incorporated in the fuel handling operations (see Subsection 9.1.4). Only one spent fuel assembly is lifted at a time, and the fuel is moved at low speeds, exercising caution that the fuel assembly not strike anything during movement. The containment, auxiliary building, refueling pool, and spent fuel pool are designed to seismic Category I requirements to thus provide their integrity in the event of a safe shutdown earthquake. The spent fuel storage racks are located to prevent a credible external missile from reaching the stored fuel assemblies. The fuel handling equipment is designed to prevent the handling equipment from falling onto the fuel in the reactor vessel or that stored in the spent fuel pool. The facility is designed so that heavy objects, such as the spent fuel shipping cask, cannot be carried over or tipped into the spent fuel pool.

15.7.4.1 Source Term

The inventory of fission products available for release at the time of the accident is dependent on a number of factors, such as the power history of the fuel assembly, the time delay between reactor shutdown and the beginning of fuel handling operations, and the volatility of the nuclides.

The fuel handling accident source term is derived from the core source term detailed in Appendix 15A by taking into account the factors below. The assumptions used to define the fuel handling accident initial airborne release source term are provided in Table 15.7-1 along with the derived source term.

15.7.4.1.1 Fission Product Gap Fraction

During power operation, a portion of the fission products generated in the fuel pellet matrix diffuses into the fuel/cladding gap. The fraction of the assembly fission products found in the gap depends on the rate of diffusion for the nuclide in question as well as the rate of radioactive decay. In the event of a fuel handling accident, the gaseous and volatile radionuclides contained in the fuel/cladding gap are free to escape from the fuel assembly. The radionuclides of concern are the noble gases (kryptons and xenons) and iodines. Based on NUREG-1465 (Reference 1), the fission product gap fraction is 3-percent of fuel inventory. For this analysis, the gap fractions are increased to be consistent with the guidance of Regulatory Guide 1.183 (Reference 2). The gap fractions are listed in Table 15.7-1.

15.7.4.1.2 Iodine Chemical Form

Consistent with NUREG-1465 guidance, the iodine released from the damaged fuel rods is assumed to be 95-percent cesium iodide, 4.85-percent elemental iodine, and 0.15-percent organic iodine.

Cesium iodide is nonvolatile, and the iodine in this form dissolves in water but does not readily become airborne. However, consistent with the guidance in Regulatory Guide 1.183, it is assumed that the cesium iodide is instantaneously converted to the elemental form when released from the fuel into the low pH water pool.

15.7.4.1.3 Assembly Power Level

All fuel assemblies are assumed to be handled inside the containment during the core shuffle so a peak power assembly is considered for the accident. Any fuel assembly can be transferred to the spent fuel pool; during a core off-load, all fuel assemblies are discharged to the spent fuel pool. To obtain a bounding condition for the fuel handling accident analysis, it is assumed that the accident involves a fuel assembly that operated at the maximum rated fuel rod peaking factor. This is conservative because the entire fuel assembly does not operate at this level.

15.7.4.1.4 Radiological Decay

The fission product decay time experienced prior to the fuel handling accident is at least 48 hours.

15.7.4.2 Release Pathways

The spent fuel handling operations take place underwater. Thus, activity releases are first scrubbed by the column of water 23 feet in depth. This has no effect on the releases of noble gases or organic iodine but there is a significant removal of elemental iodine. Consistent with the guidance in Regulatory Guide 1.183, the overall pool scrubbing decontamination factor for iodine is assumed to be 200.

After the activity escapes from the water pool, it is assumed that it is released directly to the environment within a 2-hour period without credit for any additional iodine removal process.

If the fuel handling accident occurs in the containment, the release of activity can be terminated by closure of the containment purge lines on detection of high radioactivity. No credit is taken for this in the analysis. Additionally, no credit is taken for removal of airborne iodine by the filters in the containment purge lines.

For the fuel handling accident postulated to occur in the spent fuel pool, there is assumed to be no filtration in the release pathway. Activity released from the pool is assumed to pass directly to the environment with no credit for holdup or delay of release in the building.

15.7.4.3 Dose Calculation Models

The models used to calculate doses are provided in Appendix 15A.

Table 15.7-1 lists the assumptions used in the analysis. The guidance of Regulatory Guide 1.183 is reflected in the analysis assumptions.

15.7.4.4 Identification of Conservatisms

The fuel handling accident dose analysis assumptions contain a number of conservatisms. Some of these conservatisms are described in the following subsections.

15.7.4.4.1 Fuel Assembly Power Level

The source term is based on the assumption that all of the fuel rods in the damaged assembly have been operating at the maximum fuel rod radial peaking factor. In actuality, this is true for only a small fraction of the fuel rods in any assembly. The overall assembly power level is less than the maximum radial peaking factor.

15.7.4.4.2 Fission Product Gap Fraction

The assumption of Regulatory Guide 1.183 gap fractions for the short-lived nuclides is conservative by a factor of 2 or more, depending on the nuclide.

15.7.4.4.3 Amount of Fuel Damage

It is assumed that all fuel rods in a fuel assembly are damaged so as to release the fission product inventory in the fuel/cladding gap. In an actual fuel handling accident, it is expected that there would be few rods damaged to this extent.

15.7.4.4.4 Iodine Plateout on Fuel Cladding

Although it is expected that virtually all elemental iodine plates out on the fuel cladding and is unavailable for atmospheric release, no credit is taken for plateout.

15.7.4.4.5 Presence of Organic Iodine

Although 0.15% of the iodine is assumed to be in the organic form (and thus not subject to scrubbing removal in the water pool), there would be no organic iodine in the fuel rods. Any formation of organic iodine would occur gradually and would not contribute to early releases of activity.

15.7.4.4.6 Conversion of Cesium Iodide to Form Elemental Iodine

The analysis assumes that all of the cesium iodide converts immediately to the elemental iodine form after release to the water pool and is treated in the same manner as the iodine initially in the elemental form. While the low pH solution does support conversion to the elemental form, the conversion would not occur unless the cesium iodide was dissolved in the water. The elemental iodine that is formed would thus be in the water solution and not in the bubbles of gas released from the damaged fuel. Additionally, conversion of cesium iodide would occur slowly and the elemental iodine formed would not be immediately available for release.

15.7.4.4.7 Meteorology

It is unlikely that the conservatively selected meteorological conditions are present at the time of the accident.

15.7.4.4.8 Time Available for Radioactive Decay

The dose analysis assumes that the fuel handling accident involves one of the first fuel assemblies handled. If it were one of the later fuel handling operations, there is additional decay and a reduction in the source term.

The dose evaluation was performed assuming 48 hours decay.

15.7.4.5 Offsite Doses

Using the assumptions from Table 15.7-1, the calculated doses from the initial releases are determined to be 2.8 rem TEDE at the site boundary and 1.2 rem TEDE at the low population zone outer boundary. These doses are well within the dose guideline of 25 rem TEDE identified in 10 CFR Part 50.34. The phrase “well within” is taken as meaning 25 percent or less.

15.7.5 Spent Fuel Cask Drop Accident

The spent fuel cask handling crane is prevented from travelling over the spent fuel. No radiological consequences analysis is necessary for the dropped cask event.

15.7.6 Combined License Information

An analysis of the consequences of potential release of radioactivity to the environment due to a liquid tank failure is addressed in Subsection 2.4.13.

15.7.7 References

1. Sofer, L., et al., "Accident Source Terms for Light-Water Nuclear Power Plants," NUREG-1465, February 1995.

2. U. S. NRC Regulatory Guide 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors, " July 2000.

Table 15.7-1
Assumptions Used to Determine
Fuel Handling Accident Radiological Consequences

Source term assumptions – Core power (MWt) – Decay time (hr)	 3434[(1)] 48
Core source term after 48 hours decay (Ci) I-130 I-131 I-132 I-133 I-135 Kr-85m Kr-85 Kr-88 Xe-131m Xe-133m Xe-133 Xe-135m Xe-135	 1.28 E+05 8.18 E+07 9.10 E+07 4.06 E+07 1.17 E+06 1.52 E+04 1.07 E+06 5.45 E+02 1.02 E+06 4.47 E+06 1.70 E+08 1.91 E+05 1.04 E+07
Number of fuel assemblies in core	157
Amount of fuel damage	One assembly
Maximum rod radial peaking factor	1.75
Percentage of fission products in gap I-131 Other iodines Kr-85 Other noble gases	 8 5 10 5
Pool decontamination factor for iodine	200
Activity release period (hr)	2
Atmospheric dispersion factors	See Table 15A-5 in Appendix 15A
Breathing rates (m^3/sec)	3.5 E-4
Nuclide data	See Appendix 15A

Note:
1. The main feedwater flow measurement supports a 1-percent power uncertainty.

15.8 Anticipated Transients Without Scram

15.8.1 General Background

An anticipated transient without scram (ATWS) is an anticipated operational occurrence during which an automatic reactor scram is required but fails to occur due to a common mode fault in the reactor protection system. Under certain circumstances, failure to execute a required scram during an anticipated operational occurrence could transform a relatively minor transient into a more severe accident. ATWS events are not considered to be in the design basis for Westinghouse plants.

15.8.2 Anticipated Transients Without Scram in the AP1000

For Westinghouse plants, the ATWS rule (10 CFR 50.62) requires the installation of equipment that is diverse from the reactor protection system to automatically trip the turbine and initiate decay heat removal. This equipment must be designed to perform its function in a reliable manner and be independent from sensor output to final actuation device from the existing reactor protection system.

The basis for the ATWS rule requirements, as outlined in SECY-83-293 (Reference 1), is to reduce the risk of core damage because of ATWS to less than 10^{-5} per reactor year.

The AP1000 includes a diverse actuation system, which provides the AMSAC protection features mandated for Westinghouse plants by 10 CFR 50.62, plus a diverse reactor scram (see Section 7.7). Thus, the ATWS rule is met.

15.8.3 Conclusion

The AP1000 is equipped with a diverse actuation system, which provides the functions required by the ATWS rule (10 CFR 50.62). The ATWS core damage frequency for the AP1000 is well below the SECY-83-293 goal of 10^{-5} per reactor year. The AP1000 ATWS core damage frequency is discussed in Chapter 33 of the Probabilistic Risk Assessment (PRA). The AP1000 design meets the ATWS rule (10 CFR 50.62) and its ATWS core damage frequency safety goal basis.

15.8.4 Combined License Information

This section contained no requirement for additional information.

15.8.5 References

1. Dircks, W. J., "Amendments to 10 CFR 50 Related to Anticipated Transients Without Scram (ATWS) Events," SECY-83-293, U.S. NRC, July 19, 1983.

Appendix 15A Evaluation Models and Parameters for Analysis of Radiological Consequences of Accidents

This appendix contains the parameters and models that form the basis of the radiological consequences analyses for the various postulated accidents.

15A.1 Offsite Dose Calculation Models

Radiological consequences analyses are performed to determine the total effective dose equivalent (TEDE) doses associated with the postulated accident. The determination of TEDE doses takes into account the committed effective dose equivalent (CEDE) dose resulting from the inhalation of airborne activity (that is, the long-term dose accumulation in the various organs) as well as the effective dose equivalent (EDE) dose resulting from immersion in the cloud of activity.

15A.1.1 Immersion Dose (Effective Dose Equivalent)

Assuming a semi-infinite cloud, the immersion doses are calculated using the equation:

$$D_{im} = \sum_i DCF_i \sum_j R_{ij} (\chi / Q)_j$$

where:

D_{im} = Immersion (EDE) dose (rem)

DCF_i = EDE dose conversion factor for isotope i (rem-m^3/Ci-s)

R_{ij} = Amount of isotope i released during time period j (Ci)

$(\chi/Q)_j$ = Atmospheric dispersion factor during time period j (s/m^3)

15A.1.2 Inhalation Dose (Committed Effective Dose Equivalent)

The CEDE doses are calculated using the equation:

$$D_{CEDE} = \sum_i DCF_i \sum_j R_{ij} (BR)_j (\chi / Q)_j$$

where:

D_{CEDE} = CEDE dose (rem)

DCF_i = CEDE dose conversion factor (rem per curie inhaled) for isotope i

R_{ij} = Amount of isotope i released during time period j (Ci)

$(BR)_j$ = Breathing rate during time period j (m^3/s)

$(\chi/Q)_j$ = Atmospheric dispersion factor during time period j (s/m^3)

15A.1.3 Total Dose (Total Effective Dose Equivalent)

The TEDE doses are the sum of the EDE and the CEDE doses.

15A.2 Main Control Room Dose Models

Radiological consequences analyses are performed to determine the TEDE doses associated with the postulated accident. The determination of TEDE doses takes into account the CEDE dose resulting from the inhalation of airborne activity (that is, the long-term dose accumulation in the various organs) as well as the EDE dose resulting from immersion in the cloud of activity.

15A.2.1 Immersion Dose Models

Due to the finite volume of air contained in the main control room, the immersion dose for an operator occupying the main control room is substantially less than it is for the case in which a semi-infinite cloud is assumed. The finite cloud doses are calculated using the geometry correction factor from Murphy and Campe (Reference 1).

The equation is:

$$D_{im} = \frac{1}{GF} \sum_i DCF_i \sum_j (IAR)_{ij} O_j$$

where:

D_{im} = Immersion (EDE) dose (rem)

GF = Main control room geometry factor =1173/$V^{0.338}$

V = Volume of the main control room (ft^3)

DCF_i = EDE dose conversion factor for isotope i (rem-m^3/Ci-s)

$(IAR)_{ij}$ = Integrated activity for isotope i in the main control room during time period j (Ci-s/m^3)

O_j = Fraction of time period j that the operator is assumed to be present

15A.2.2 Inhalation Dose

The CEDE doses are calculated using the equation:

$$D_{CEDE} = \sum_i DCF_i \sum_j (IAR)_{ij} (BR)_j O_j$$

where:

D_{CEDE} = CEDE dose (rem)

DCF_i = CEDE dose conversion factor (rem per curie inhaled) for isotope i

$(IAR)_{ij}$ = Integrated activity for isotope i in the main control room during time period j (Ci-s/m^3)

$(BR)_j$ = Breathing rate during time period j (m^3/s)

O_j = Fraction of time period j that the operator is assumed to be present

15A.2.3 Total Dose (Total Effective Dose Equivalent)

The TEDE doses are the sum of the EDE and the CEDE doses.

15A.3 General Analysis Parameters

15A.3.1 Source Terms

The sources of radioactivity for release are dependent on the specific accident. Activity may be released from the primary coolant, from the secondary coolant, and from the core if the accident involves fuel failures. The radiological consequences analyses use conservative design basis source terms.

15A.3.1.1 Primary Coolant Source Term

The design basis primary coolant source terms are listed in Table 11.1-2. These source terms are based on continuous plant operation with 0.25-percent fuel defects. The remaining assumptions used in determining the primary coolant source terms are listed in Table 11.1-1.

The accident dose analyses take into account increases in the primary coolant source terms for iodines and noble gases above those listed in Table 11.1-2, consistent with the Tech Spec limits of 1.0 µCi/g dose equivalent I-131 for the iodines and 280 µCi/g dose equivalent Xe-133 for the noble gases.

The radiological consequences analyses for certain accidents also take into account the phenomenon of iodine spiking, which causes the concentration of radioactive iodines in the primary coolant to increase significantly. Table 15A-1 lists the concentrations of iodine isotopes associated with a pre-existing iodine spike. This is an iodine spike that occurs prior to the accident and for which the peak primary coolant activity is reached at the time the accident is assumed to occur. These isotopic concentrations are also defined as 60 µCi/g dose equivalent I-131. The probability of this adverse timing of the iodine spike and accident is small.

Although it is unlikely for an accident to occur at the same time that an iodine spike is at its maximum reactor coolant concentration, for many accidents it is expected that an iodine spike would be initiated by the accident or by the reactor trip associated with the accident. Table 15A-2 lists the iodine appearance rates (rates at which the various iodine isotopes are transferred from the core to the primary coolant by way of the assumed cladding defects) for normal operation. The iodine spike appearance rates are assumed to be as much as 500 times the normal appearance rates.

15A.3.1.2 Secondary Coolant Source Term

The secondary coolant source term used in the radiological consequences analyses is conservatively assumed to be 1 percent of the primary coolant equilibrium source term. This is more conservative than using the design basis secondary coolant source terms listed in Table 11.1-5.

Because the iodine spiking phenomenon is short-lived and there is a high level of conservatism for the assumed secondary coolant iodine concentrations, the effect of iodine spiking on the secondary coolant iodine source terms is not modeled.

There is assumed to be no secondary coolant noble gas source term because the noble gases entering the secondary side due to primary-to-secondary leakage enter the steam phase and are discharged via the condenser air removal system.

15A.3.1.3 Core Source Term

Table 15A-3 lists the core source terms at shutdown for an assumed three-region equilibrium cycle at end of life after continuous operation at 2 percent above full core thermal power. The main feedwater flow measurement supports a 1-percent power uncertainty; use of a 2-percent power uncertainty is conservative. In addition to iodines and noble gases, the source terms listed include nuclides that are identified as potentially significant dose contributors in the event of a degraded core accident. The design basis loss-of-coolant accident analysis is not expected to result in significant core damage, but the radiological consequences analysis assumes severe core degradation.

15A.3.2 Nuclide Parameters

The radiological consequence analyses consider radioactive decay of the subject nuclides prior to their release, but no additional decay is assumed after the activity is released to the environment. Table 15A-4 lists the decay constants for the nuclides of concern.

Table 15A-4 also lists the dose conversion factors for calculation of the CEDE doses due to inhalation of iodines and other nuclides and EDE dose conversion factors for calculation of the dose due to immersion in a cloud of activity. The CEDE dose conversion factors are from EPA Federal Guidance Report No. 11 (Reference 2) and the EDE dose conversion factors are from EPA Federal Guidance Report No. 12 (Reference 3).

15A.3.3 Atmospheric Dispersion Factors

Subsection 2.3.4 lists the off-site short-term atmospheric dispersion factors (χ/Q) for the reference site. Table 15A-5 (Sheet 1 of 2) reiterates these χ/Q values.

The atmospheric dispersion factors (χ/Q) to be applied to air entering the main control room following a design basis accident are specified at the HVAC intake and at the annex building entrance (which would be the air pathway to the main control room due to ingress/egress). A set of χ/Q values is identified for each potential activity release location that has been identified and the two control room receptor locations. These χ/Q values are listed in Table 15A-6 and are provided in Table 2.0-201.

Site-specific X/Q values provided in Subsection 2.3.4 are bounded by the values given in Table 15A-5 and Table 15A-6.

Table 15A-7 identifies the AP1000 source and receptor data to be used when determining the site-specific control room χ/Q values using the ARCON96 code (References 4 and 5).

The main control room χ/Q values do not incorporate occupancy factors.

The locations of the potential release points and their relationship to the main control room air intake and the annex building access door are shown in Figure 15A-1.

15A.4 References

1. Murphy, K. G., Campe, K. M., "Nuclear Power Plant Control Room Ventilation System Design for Meeting General Criterion 19," paper presented at the 13th AEC Air Cleaning Conference.

2. EPA Federal Guidance Report No. 11, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion," EPA-520/1-88-020, September 1988.

3. EPA Federal Guidance Report No. 12, "External Exposure to Radionuclides in Air, Water, and Soil," EPA 402-R-93-081, September 1993.

4. NUREG/CR-6331, Ramsdell, J. V. and Simonen, C. A., "Atmospheric Relative Concentrations in Building Wakes," Revision 1, May 1997.

5. Regulatory Guide 1.194, Atmospheric Relative Concentrations for Control Room Radiological Habitability Assessments at Nuclear Power Plants, June 2003.

Table 15A-1
Reactor Coolant Iodine Concentrations for
Maximum Iodine Spike of 60 µCi/g Dose Equivalent I-131

Nuclide	µCi/g
I-130	0.66
I-131	43.4
I-132	57.5
I-133	78.8
I-134	13.4
I-135	47.8

Table 15A-2
Iodine Appearance Rates in the Reactor Coolant

Nuclide	Equilibrium Appearance Rate (Ci/min)
I-130	7.03×10^{-3}
I-131	3.39×10^{-1}
I-132	1.38
I-133	7.42×10^{-1}
I-134	6.81×10^{-1}
I-135	6.37×10^{-1}

Table 15A-3 (Sheet 1 of 2)
Reactor Core Source Term(1)

	Nuclide	Inventory (Ci)		Nuclide	Inventory (Ci)
Iodines	I-130	3.66×10^6	Noble Gases	Kr-85m	2.63×10^7
	I-131	9.63×10^7		Kr-85	1.06×10^6
	I-132	1.40×10^8		Kr-87	5.07×10^7
	I-133	1.99×10^8		Kr-88	7.14×10^7
	I-134	2.18×10^8		Xe-131m	1.06×10^6
	I-135	1.86×10^8		Xe-133m	5.84×10^6
Cs Group	Cs-134	1.94×10^7		Xe-133	1.90×10^8
	Cs-136	5.53×10^6		Xe-135m	3.87×10^7
	Cs-137	1.13×10^7		Xe-135	4.84×10^7
	Cs-138	1.82×10^8		Xe-138	1.65×10^8
	Rb-86	2.29×10^5	Sr & Ba	Sr-89	9.66×10^7
Te Group	Te-127m	1.32×10^6		Sr-90	8.31×10^6
	Te-127	1.02×10^7		Sr-91	1.20×10^8
	Te-129m	4.50×10^6		Sr-92	1.29×10^8
	Te-129	3.04×10^7		Ba-139	1.78×10^8
	Te-131m	1.40×10^7		Ba-140	1.71×10^8
	Te-132	1.38×10^8	Ce Group	Ce-141	1.63×10^8
	Sb-127	1.03×10^7		Ce-143	1.52×10^8
	Sb-129	3.10×10^7		Ce-144	1.23×10^8
Ru Group	Ru-103	1.45×10^8		Pu-238	3.83×10^5
	Ru-105	9.83×10^7		Pu-239	3.37×10^4
	Ru-106	4.77×10^7		Pu-240	4.94×10^4
	Rh-105	9.00×10^7		Pu-241	1.11×10^7
	Mo-99	1.84×10^8		Np-239	1.93×10^9
	Tc-99m	1.61×10^8			

Note:

1. The following assumptions apply:
   - Core thermal power of 3468 MWt (2 percent above the design core power of 3400 MWt). The main feedwater flow measurement supports a 1-percent power uncertainty; use of a 2-percent power uncertainty is conservative.
   - Three-region equilibrium cycle core at end of life

Table 15A-3 (Sheet 2 of 2)
Reactor Core Source Term[1]

	Nuclide	Inventory (Ci)
La Group	Y-90	8.66×10^{6}
	Y-91	1.24×10^{8}
	Y-92	1.30×10^{8}
	Y-93	1.49×10^{8}
	Nb-95	1.67×10^{8}
	Zr-95	1.66×10^{8}
	Zr-97	1.64×10^{8}
	La-140	1.82×10^{8}
	La-141	1.62×10^{8}
	La-142	1.57×10^{8}
	Pr-143	1.46×10^{8}
	Nd-147	6.48×10^{7}
	Am-241	1.25×10^{4}
	Cm-242	2.95×10^{6}
	Cm-244	3.62×10^{5}

Note:

1. The following assumptions apply:
   - Core thermal power of 3468 MWt (2 percent above the design core power of 3400 MWt). The main feedwater flow measurement supports a 1-percent power uncertainty; use of a 2-percent power uncertainty is conservative.
   - Three-region equilibrium cycle core at end of life

Table 15A-4 (Sheet 1 of 4)
Nuclide Parameters

A. HALOGENS			
Isotope	**Decay Constant (hr^{-1})**	**EDE Dose Conversion Factor ($Sv\text{-}m^3/Bq\text{-}s$)**	**CEDE Dose Conversion Factor (Sv/Bq)**
I-130	5.61×10^{-2}	1.04×10^{-13}	7.14×10^{-10}
I-131	3.59×10^{-3}	1.82×10^{-14}	8.89×10^{-9}
I-132	3.01×10^{-1}	1.12×10^{-13}	1.03×10^{-10}
I-133	3.33×10^{-2}	2.94×10^{-14}	1.58×10^{-9}
I-134	7.91×10^{-1}	1.30×10^{-13}	3.55×10^{-11}
I-135	1.05×10^{-1}	7.98×10^{-14}	3.32×10^{-10}
B. NOBLE GASES			
Isotope	**Decay Constant (hr^{-1})**	**EDE Dose Conversion Factor ($Sv\text{-}m^3/Bq\text{-}s$)**	
Kr-85m	1.55×10^{-1}	7.48×10^{-15}	
Kr-85	7.38×10^{-6}	1.19×10^{-16}	
Kr-87	5.45×10^{-1}	4.12×10^{-14}	
Kr-88	2.44×10^{-1}	1.02×10^{-13}	
Xe-131m	2.43×10^{-3}	3.89×10^{-16}	
Xe-133m	1.32×10^{-2}	1.37×10^{-15}	
Xe-133	5.51×10^{-3}	1.56×10^{-15}	
Xe-135m	2.72	2.04×10^{-14}	
Xe-135	7.63×10^{-2}	1.19×10^{-14}	
Xe-138	2.93	5.77×10^{-14}	

Table 15A-4 (Sheet 2 of 4)
Nuclide Parameters

C. ALKALI METALS			
Nuclide	**Decay Constant (hr^{-1})**	**EDE Dose Conversion Factor ($Sv\text{-}m^3/Bq\text{-}s$)**	**CEDE Dose Conversion Factor (Sv/Bq)**
Cs-134	3.84×10^{-5}	7.57×10^{-14}	1.25×10^{-8}
Cs-136	2.2×10^{-3}	1.06×10^{-13}	1.98×10^{-9}
Cs-137(1)	2.64×10^{-6}	2.88×10^{-14}	8.63×10^{-9}
Cs-138	1.29	1.21×10^{-13}	2.74×10^{-11}
Rb-86	1.55×10^{-3}	4.81×10^{-15}	1.79×10^{-9}
D. TELLURIUM GROUP			
Nuclide	**Decay Constant (hr^{-1})**	**EDE Dose Conversion Factor ($Sv\text{-}m^3/Bq\text{-}s$)**	**CEDE Dose Conversion Factor (Sv/Bq)**
Te-127m	2.65×10^{-4}	1.47×10^{-16}	5.81×10^{-9}
Te-127	7.41×10^{-2}	2.42×10^{-16}	8.60×10^{-11}
Te-129m	8.6×10^{-4}	1.55×10^{-15}	6.47×10^{-9}
Te-129	5.98×10^{-1}	2.75×10^{-15}	2.42×10^{-11}
Te-131m	2.31×10^{-2}	7.01×10^{-14}	1.73×10^{-9}
Te-132	8.86×10^{-3}	1.03×10^{-14}	2.55×10^{-9}
Sb-127	7.5×10^{-3}	3.33×10^{-14}	1.63×10^{-9}
Sb-129	1.6×10^{-1}	7.14×10^{-14}	1.74×10^{-10}
E. STRONTIUM AND BARIUM			
Nuclide	**Decay Constant (hr^{-1})**	**EDE Dose Conversion Factor ($Sv\text{-}m^3/Bq\text{-}s$)**	**CEDE Dose Conversion Factor (Sv/Bq)**
Sr-89	5.72×10^{-4}	7.73×10^{-17}	1.12×10^{-8}
Sr-90	2.72×10^{-6}	7.53×10^{-18}	3.51×10^{-7}
Sr-91	7.3×10^{-2}	3.45×10^{-14}	4.49×10^{-10}
Sr-92	2.56×10^{-1}	6.79×10^{-14}	2.18×10^{-10}
Ba-139	5.02×10^{-1}	2.17×10^{-15}	4.64×10^{-11}
Ba-140	2.27×10^{-3}	8.58×10^{-15}	1.01×10^{-9}

Note:

1. The listed average gamma disintegration energy for Cs-137 is due to the production and decay of Ba-137m.

Table 15A-4 (Sheet 3 of 4)
Nuclide Parameters

F. NOBLE METALS			
Nuclide	**Decay Constant (hr^{-1})**	**EDE Dose Conversion Factor ($Sv\text{-}m^3/Bq\text{-}s$)**	**CEDE Dose Conversion Factor (Sv/Bq)**
Ru-103	$7.35x10^{-4}$	$2.25x10^{-14}$	$2.42x10^{-9}$
Ru-105	$1.56x10^{-1}$	$3.81x10^{-14}$	$1.23x10^{-10}$
Ru-106	$7.84x10^{-5}$	0.0	$1.29x10^{-7}$
Rh-105	$1.96x10^{-2}$	$3.72x10^{-15}$	$2.58x10^{-10}$
Mo-99	$1.05x10^{-2}$	$7.28x10^{-15}$	$1.07x10^{-9}$
Tc-99m	$1.15x10^{-1}$	$5.89x10^{-15}$	$8.80x10^{-12}$
G. CERIUM GROUP			
Nuclide	**Decay Constant (hr^{-1})**	**EDE Dose Conversion Factor ($Sv\text{-}m^3/Bq\text{-}s$)**	**CEDE Dose Conversion Factor (Sv/Bq)**
Ce-141	$8.89x10^{-4}$	$3.43x10^{-15}$	$2.42x10^{-9}$
Ce-143	$2.1x10^{-2}$	$1.29x10^{-14}$	$9.16x10^{-10}$
Ce-144	$1.02x10^{-4}$	$8.53x10^{-16}$	$1.01x10^{-7}$
Pu-238	$9.02x10^{-7}$	$4.88x10^{-18}$	$1.06x10^{-4}$
Pu-239	$3.29x10^{-9}$	$4.24x10^{-18}$	$1.16x10^{-4}$
Pu-240	$1.21x10^{-8}$	$4.75x10^{-18}$	$1.16x10^{-4}$
Pu-241	$5.5x10^{-6}$	$7.25x10^{-20}$	$2.23x10^{-6}$
Np-239	$1.23x10^{-2}$	$7.69x10^{-15}$	$6.78x10^{-10}$

Table 15A-4 (Sheet 4 of 4)
Nuclide Parameters

H. LANTHANIDE GROUP			
Nuclide	**Decay Constant (hr^{-1})**	**EDE Dose Conversion Factor ($Sv\text{-}m^3/Bq\text{-}s$)**	**CEDE Dose Conversion Factor (Sv/Bq)**
Y-90	$1.08x10^{-2}$	$1.90x10^{-16}$	$2.28x10^{-9}$
Y-91	$4.94x10^{-4}$	$2.60x10^{-16}$	$1.32x10^{-8}$
Y-92	$1.96x10^{-1}$	$1.30x10^{-14}$	$2.11x10^{-10}$
Y-93	$6.86x10^{-2}$	$4.80x10^{-15}$	$5.82x10^{-10}$
Nb-95	$8.22x10^{-4}$	$3.74x10^{-14}$	$1.57x10^{-9}$
Zr-95	$4.51x10^{-4}$	$3.60x10^{-14}$	$6.39x10^{-9}$
Zr-97	$4.1x10^{-2}$	$9.02x10^{-15}$	$1.17x10^{-9}$
La-140	$1.72x10^{-2}$	$1.17x10^{-13}$	$1.31x10^{-9}$
La-141	$1.76x10^{-1}$	$2.39x10^{-15}$	$1.57x10^{-10}$
La-142	$4.5x10^{-1}$	$1.44x10^{-13}$	$6.84x10^{-11}$
Nd-147	$2.63x10^{-3}$	$6.19x10^{-15}$	$1.85x10^{-9}$
Pr-143	$2.13x10^{-3}$	$2.10x10^{-17}$	$2.19x10^{-9}$
Am-241	$1.83x10^{-7}$	$8.18x10^{-16}$	$1.20x10^{-4}$
Cm-242	$1.77x10^{-4}$	$5.69x10^{-18}$	$4.67x10^{-6}$
Cm-244	$4.37x10^{-6}$	$4.91x10^{-18}$	$6.70x10^{-5}$

Table 15A-5
Offsite Atmospheric Dispersion Factors (χ/Q)
for Accident Dose Analysis

Site boundary χ/Q (s/m^3)	
0–2 hours[(1)]	5.1×10^{-4}
Low population zone χ/Q (s/m^3)	
0–8 hours	2.2×10^{-4}
8–24 hours	1.6×10^{-4}
24–96 hours	1.0×10^{-4}
96–720 hours	8.0×10^{-5}

Notes:

1. Nominally defined as the 0- to 2-hour interval, but is applied to the 2-hour interval having the highest activity releases in order to address 10 CFR Part 50.34 requirements

Table 15A-6
Control Room Atmospheric Dispersion Factors (χ/Q) for Accident Dose Analysis

χ/Q (s/m^3) at HVAC Intake for the Identified Release Points[(1)]						
	Plant Vent or PCS Air Diffuser[(3)]	**Ground Level Containment Release Points[(4)]**	**PORV and Safety Valve Releases[(5)]**	**Steam Line Break Releases**	**Fuel Handling Area[(6)]**	**Condenser Air Removal Stack[(7)]**
0 – 2 hours	3.0E-3	6.0E-3	2.0E-2	2.4E-2	6.0E-3	6.0E-3
2 – 8 hours	2.5E-3	3.6E-3	1.8E-2	2.0E-2	4.0E-3	4.0E-3
8 – 24 hours	1.0E-3	1.4E-3	7.0E-3	7.5E-3	2.0E-3	2.0E-3
1 – 4 days	8.0E-4	1.8E-3	5.0E-3	5.5E-3	1.5E-3	1.5E-3
4 – 30 days	6.0E-4	1.5E-3	4.5E-3	5.0E-3	1.0E-3	1.0E-3
χ/Q (s/m^3) at Annex Building Door for the Identified Release Points[(2)]						
	Plant Vent or PCS Air Diffuser[(3)]	**Ground Level Containment Release Points[(4)]**	**PORV and Safety Valve Releases[(5)]**	**Steam Line Break Releases**	**Fuel Handling Area[(6)]**	**Condenser Air Removal Stack[(7)]**
0 – 2 hours	1.0E-3	1.0E-3	4.0E-3	4.0E-3	6.0E-3	2.0E-2
2 – 8 hours	7.5E-4	7.5E-4	3.2E-3	3.2E-3	4.0E-3	1.8E-2
8 – 24 hours	3.5E-4	3.5E-4	1.2E-3	1.2E-3	2.0E-3	7.0E-3
1 – 4 days	2.8E-4	2.8E-4	1.0E-3	1.0E-3	1.5E-3	5.0E-3
4 – 30 days	2.5E-4	2.5E-4	8.0E-4	8.0E-4	1.0E-3	4.5E-3

Notes:

1. These dispersion factors are to be used 1) for the time period preceding the isolation of the main control room and actuation of the emergency habitability system, 2) for the time after 72 hours when the compressed air supply in the emergency habitability system would be exhausted and outside air would be drawn into the main control room, and 3) for the determination of control room doses when the non-safety ventilation system is assumed to remain operable such that the emergency habitability system is not actuated.
2. These dispersion factors are to be used when the emergency habitability system is in operation and the only path for outside air to enter the main control room is that due to ingress/egress.
3. These dispersion factors are used for analysis of the doses due to a postulated small line break outside of containment. The plant vent and PCS air diffuser are potential release paths for other postulated events (loss-of-coolant accident, rod ejection accident, and fuel handling accident inside the containment); however, the values are bounded by the dispersion factors for ground level releases.
4. The listed values represent modeling the containment shell as a diffuse area source, and are used for evaluating the doses in the main control room for a loss-of-coolant accident, for the containment leakage of activity following a rod ejection accident, and for a fuel handling accident occurring inside the containment.
5. The listed values bound the dispersion factors for releases from the steam line safety & power-operated relief valves. These dispersion factors would be used for evaluating the doses in the main control room for a steam generator tube rupture, a main steam line break, a locked reactor coolant pump rotor, and for the secondary side release from a rod ejection accident.
6. The listed values bound the dispersion factors for releases from the fuel storage and handling area. The listed values also bound the dispersion factors for releases from the fuel storage area in the event that spent fuel boiling occurs and the fuel building relief panel opens on high temperature. These dispersion factors are used for the fuel handling accident occurring outside containment and for evaluating the impact of releases associated with spent fuel pool boiling.
7. This release point is included for information only as a potential activity release point. None of the design basis accident radiological consequences analyses model release from this point.

Table 15A-7
Control Room Source/Receptor Data for Determination Of Atmospheric Dispersion Factors

		Horizontal Straight-Line Distance To Receptor		
Source Description	**Release Elevation Note 1 (m)**	**Control Room HVAC Intake (Elevation 19.9 m) (Δ1)**	**Annex Building Access (Elevation 1.5 m) (Δ2)**	**Comment**
Plant Vent (⬡1)	55.7	147.2 ft (44.9 m)	379.3 ft (115.6 m)	
PCS Air Diffuser (⬡2)	69.8	118.1 ft (36.0 m)	343.2 ft (104.6 m)	
Fuel Building Blowout Panel (⬡3)	17.4	203.2 ft (61.9 m)	427.4 ft (130.3 m)	Note 3
Radwaste Building Truck Staging Area Door (⬡4)	1.5	218.5 ft (66.6 m)	433.5 ft (132.1 m)	Note 3
Steam Vent (⬡5)	17.1	61.5 ft (18.8 m)	261.6 ft (79.7 m)	
PORV/Safety Valves (⬡6)	19.2	66.9 ft (20.4 m)	255.4 ft (77.8 m)	
Condenser Air Removal Stack (⬡7)	38.4	198.3 ft (60.4 m)	58.3 ft (17.8 m)	Note 3
Containment Shell (Diffuse Area Source) (⬡8)	Same as Receptor Elevation (19.9 m or 1.5 m)	42.0 ft (12.8 m)	272.3 ft (83.0 m)	Note 2

Notes:

1. All elevations relative to grade at 0.0 m.
2. For calculating distance, the source is defined as the point on the containment shell closest to receptor.
3. Vertical distance traveled is conservatively neglected.
4. ⬡ – Refer to Symbols on Figure 15A-1.
5. Δ – Refer to Symbols on Figure 15A-1.

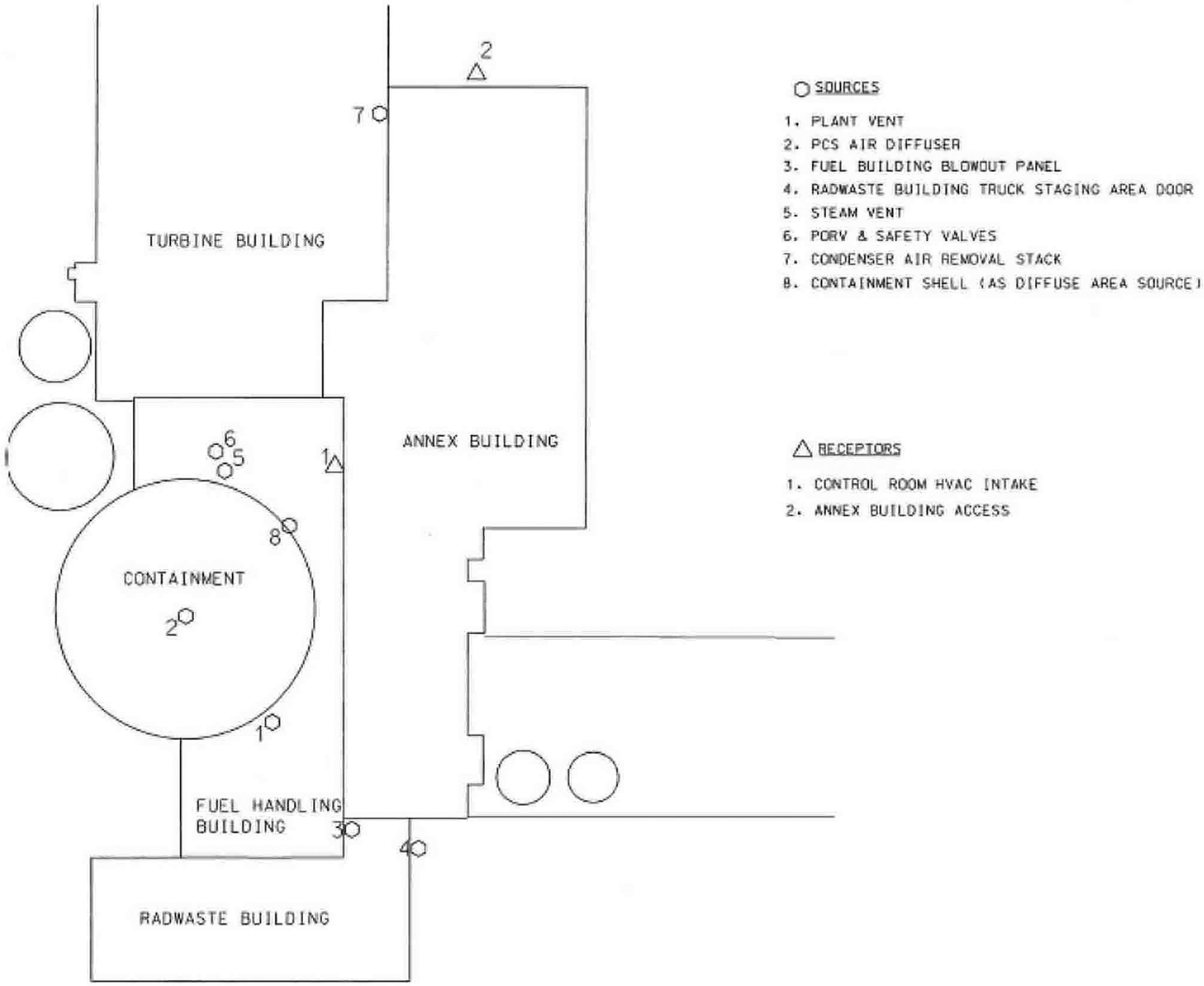

Figure 15A-1
Site Plan with Release and Intake Locations

Appendix 15B Removal of Airborne Activity from the Containment Atmosphere Following a LOCA

The AP1000 design does not depend on active systems to remove airborne particulates or elemental iodine from the containment atmosphere following a postulated loss-of-coolant accident (LOCA) with core melt. Naturally occurring passive removal processes provide significant removal capability such that airborne elemental iodine is reduced to very low levels within a few hours and the airborne particulates are reduced to extremely low levels within 12 hours.

15B.1 Elemental Iodine Removal

Elemental iodine is removed by deposition onto the structural surfaces inside the containment. The removal of elemental iodine is modeled using the equation from the Standard Review Plan (Reference 1):

$$\lambda_d = \frac{K_w A}{V}$$

where:

λ_d = first order removal coefficient by surface deposition

K_w = mass transfer coefficient (specified in Reference 1 as 4.9 m/hr)

A = surface area available for deposition

V = containment building volume

The available deposition surface is 251,000 ft^2, and the containment building net free volume is 2.06 x 10^6 ft^3. From these inputs, the elemental iodine removal coefficient is 1.9 hr^{-1}.

Consistent with the guidance of Reference 1, credit for elemental iodine removal is assumed to continue until a decontamination factor (DF) of 200 is reached in the containment atmosphere. Because the source term for the LOCA (defined in Subsection 15.6.5.3) is modeled as a gradual release of activity into the containment, the determination of the time at which the DF of 200 is reached needs to be based on the amount of elemental iodine that enters the containment atmosphere over the duration of core activity release.

15B.2 Aerosol Removal

The deposition removal of aerosols from the containment atmosphere is accomplished by a number of processes including sedimentation, diffusiophoresis, and thermophoresis. All three of the deposition processes are significant contributors to the overall removal process in the AP1000. The large contributions from diffusiophoresis and thermophoresis to the total removal are a direct consequence of the high heat transfer rates from the containment atmosphere to the containment wall that characterize the passive containment cooling system.

Because of the AP1000 passive containment cooling system design, there are high sensible heat transfer rates (resulting in higher thermophoretic removal of aerosols) when condensational heat transfer is low (and the aerosol removal by diffusiophoresis is also low). The reverse is also true. Thus, there is an appreciable deposition removal throughout the accident from either diffusiophoresis or thermophoresis, in addition to the removal by sedimentation.

15B.2.1 Mathematical Models

The models used for the three aerosol removal processes are discussed as follows.

15B.2.1.1 Sedimentation

Gravitational sedimentation is a major mechanism of aerosol removal in a containment. A standard model (Stokes equation with the Cunningham slip correction factor) for this process is used. The Stokes equation (Reference 2) is:

$$\upsilon_s = \frac{2\rho_p g r^2 Cn}{9\mu}$$

where:

υ_s = settling velocity of an aerosol particle

ρ_p = material density of the particle

g = gravitational acceleration

r = particle radius

μ = gas viscosity

Cn = Cunningham slip correction factor, a function of the Knudsen number (Kn) which is the gas molecular mean free path divided by the particle radius

However, the Stokes equation makes the simplifying assumption that the particles are spherical. The particles are expected to be nonspherical, and it is conventional to address this by introducing a "dynamic shape factor" (Reference 2) in the denominator of the Stokes equation, such that the settling velocity for the nonspherical particle is the same as for a spherical particle of equal volume. The value of the dynamic shape factor (ϕ) thus depends on the shape of the particle and, in general, must be experimentally determined.

The concept of dynamic shape factor can also be applied to a spherical particle consisting of two components, one of which has the density of the particle material, while the other component has a different density (Reference 9). In this manner, the impact of the void fraction in the particle can be modeled. Thus, the revised Stokes equation is:

$$\upsilon_s = \frac{2\rho_p g r^2 Cn}{9\mu\phi}$$

The derivation of ϕ follows.

The two-component particle is considered to have a density ρ_{av} and an effective radius of r_e. Assuming that the second component of the particle is the void volume and letting the void fraction be ε, then the average density of the particle is:

ρ_{av} = the average density of the particle = ρ_p (1-ε) + $\rho_v\varepsilon$

where:

ρ_v = density of the void material (0.0 for gas filled, 1.0 for water filled)

ε = void fraction

ρ_p = material density (solid particle with no voids)

The definition of ϕ is obtained from the Stokes equation and the equation for mass of a sphere:

$$\frac{2\rho_p g r^2 Cn}{9\mu\phi} = \frac{2\rho_{av} g r_e^2 Cn}{9\mu}$$

which reduces to:

$$\rho_p r^2 = \phi\rho_{av} r_e^2$$

and

$$\frac{4\rho_p \pi r_0^3}{3} = \frac{4\rho_{av} \pi r_e^3}{3}$$

which reduces to:

$$\rho_p r_0^3 = \rho_{av} r_e^3$$

Then:

$$\phi = \frac{\rho_p r^2}{\rho_{av} r_e^2}$$

and

$$r_e = r\left(\frac{\rho_{av}}{\rho_p}\right)^{-1/3}$$

From these two relationships, the dynamic shape factor is given by:

$$\phi = \left(\frac{\rho_{av}}{\rho_p} \right)^{-1/3}$$

15B.2.1.2 Diffusiophoresis

Diffusiophoresis is the process whereby particles are swept to a surface (for example, containment wall) by the flow set up by a condensing vapor (Stefan flow). The deposition rate is independent of the particle size and is proportional to the steam condensation rate on the surface. The standard equation for this phenomenon is due to Waldmann and Schmitt (Reference 3):

$$\upsilon_d = \frac{\sqrt{M_v}}{\sqrt{M_v} + \chi_{a/v}\sqrt{M_a}} \frac{W}{\rho_v}$$

where:

υ_d = diffusiophoretic deposition velocity

$\chi_{a/v}$ = ratio of mole fraction of air to mole fraction of steam in the containment atmosphere

M_v = molecular weight of steam

M_a = molecular weight of air

W = steam condensation rate on the wall

ρ_v = mass density of steam in the containment atmosphere

Because of the design of the passive containment cooling system, steam condensation rates are high at certain times in the design basis LOCA; thus at these times, diffusiophoretic deposition rates are significant.

15B.2.1.3 Thermophoresis

Thermophoresis is the process whereby particles drift toward a surface (for example, the containment wall) under the influence of a temperature gradient in the containment atmosphere at the surface. The effect arises because the gas molecules on the hot side of the particles undergo more collisions with the particle than do those on the cold side. Therefore, there is a net momentum transfer to the particle in the hot-to-cold direction. There are several models in the literature for this effect; the one used is the Brock equation in a form due to Talbot et al. (Reference 4). As indicated below, this model is in agreement with experimental data. The thermophoretic deposition rate is somewhat dependent on particle size and is proportional to the temperature gradient at the wall, or equivalently, the sensible heat transfer rate to the wall. The Talbot equation is:

$$\upsilon_{th} = \frac{2\,C_s\,Cn\,(\mu_g/\rho_g)[\alpha + C_T\,Kn]\,dT}{[1 + 2(\alpha + C_T\,Kn)][1 + 3\,C_M\,Kn]}\left(\frac{1}{T}\right)\frac{dT}{dy}$$

where:

υ_{th} = thermophoretic deposition velocity

α = k_g/k_p which is the ratio of the thermal conductivities of the gas (evaluated at the gas temperature at each time step) and the aerosol particle (k_p is set equal to the thermal conductivity of water – the results are not sensitive to k_p or α.)

Kn = Knudsen number (equal to the gas molecular mean free path divided by the particle radius)

Cn = Cunningham slip correction factor, a function of the Knudsen number

μ_g = gas viscosity

ρ_g = gas density

C_S = slip accommodation coefficient (Reference 4 gives the best value as 1.17.)

C_T = thermal accommodation coefficient (Reference 4 gives the best value as 2.18.)

C_M = momentum accommodation coefficient (Reference 4 gives the best value as 1.14.)

The temperature gradient at the wall, dT/dy, can be evaluated as

$$\frac{dT}{dy} = \frac{\phi_s}{k_g}$$

where ϕ_s is the sensible heat flux to the wall, and k_g is the thermal conductivity of the gas. The sensible heat flux used in the analysis is the convective heat transfer calculated as discussed in Subsection 15B.2.4.7.

15B.2.2 Other Removal Mechanisms

In addition to the above mechanisms, there are others that were not considered, including turbulent diffusion and turbulent agglomeration. The neglect of these mechanisms adds further conservatism to the calculation.

15B.2.3 Validation of Removal Mechanisms

The aerosol processes are well established and have been confirmed in many separate effects experiments, which are discussed in standard references (References 2 through 4). The Stokes formula for sedimentation velocity has been well confirmed for particles whose diameters are less than about 50 μm. In the present calculations, these make up basically all of the aerosol.

There are some separate effects validations of the diffusiophoretic effect, but the best confirmation comes from integral experiments such as the LACE tests (Reference 5). Calculations of these and other integral tests accurately predict the integrated mass of plated aerosol material only if diffusiophoresis is taken into account. If it is neglected, the predicted plated mass is about two orders of magnitude too small, compared to the observed plated mass.

The Talbot equation for the thermophoretic effect has been experimentally confirmed to within about 20 to 50 percent over a wide range of particle sizes (Reference 4). The temperature gradient at the wall, which drives this phenomenon, can be approximated by the temperature difference between the bulk gas and the wall divided by an appropriate length scale obtained from heat transfer correlations. Alternatively, because sensible heat transfer rates to the wall are available, it is easier and more accurate to use these rates directly to infer the temperature gradient.

15B.2.4 Parameters and Assumptions for Calculating Aerosol Removal Coefficients

The parameters and assumptions were selected to conservatively model the environment that would be expected to exist as a result of a LOCA with concurrent core melt.

15B.2.4.1 Containment Geometry

The containment is assumed to be a cylinder with a volume of 55,481 m^3 (1.959×10^6 ft^3). This volume includes those portions of the containment volume that would be participating in the aerosol transport and mixing; this excludes dead-ended volumes and flooded compartments. The horizontal surface area available for aerosol deposition by sedimentation is 2900 m^2 (31,200 ft^2). This includes projecting areas such as decks in addition to the floor area and excludes areas in dead-ended volumes and areas that would be flooded post-LOCA. The surface area for Brownian diffusive plateout of aerosols is 8008 m^2 (86,166 ft^2).

15B.2.4.2 Source Size Distribution

The aerosol source size distribution is assumed to be lognormal, with a geometric mean radius of 0.22 μm and a geometric standard deviation equal to 1.81. These values are derived from an evaluation of a large number of aerosol distributions measured in a variety of degraded-fuel tests and experiments. The sensitivity of aerosol removal coefficient calculations to these values is small.

15B.2.4.3 Aerosol Void Fraction

Review of scanning electron microscope photographs of deposited aerosol particles from actual core melt and fission product vaporization and aerosolization experiments (the Argonne STEP-4 test and the INEL Power Burst Facility SFD 1-4 test) indicates that the deposited particles are relatively dense, supporting a void fraction of 0.2.

The above-mentioned test results indicate that a void fraction of 0.2 is appropriate for modeling the aerosols resulting from a core melt. As part of the sensitivity study that was performed for the AP600 project, a case was run with a void fraction of 0.9. That analysis showed that the high void fraction resulted in an integrated release of aerosols over a 24-hour period that was less than 14 percent greater than that calculated when using the void fraction of 0.2. Thus, it is clear that the removal of aerosols from the containment atmosphere is not highly sensitive to the value selected for the void fraction. This is largely due to the fact that, while the selected value for void fraction has a significant impact on the calculated sedimentation removal, the impact on thermophoresis and diffusiophoresis removal is slight or none. The impact for AP1000 of using the higher value for void fraction would be less than was determined for the AP600 since sedimentation removal comprises a smaller fraction of the total removal calculated for the AP1000.

For additional conservatism, the AP1000 aerosol removal analysis uses a void fraction of 0.4 and assumes the voids are filled with air.

15B.2.4.4 Fission Product Release Fractions

Core inventories of fission products are from ORIGEN calculations for the AP1000 at end of the fuel cycle. Fractional releases to the containment of the fission products are those specified in Subsection 15.6.5.3.

15B.2.4.5 Inert Aerosol Species

The inert species include SnO_2, UO_2, Cd, Ag, and Zr. These act as surrogates for all inert materials forming aerosols. The ratio of the total mass of inert species to fission product species was assumed to be 1.5:1. This value and the partitioning of the total inert mass among its constituents are consistent with results from degraded fuel experiments (Reference 6).

15B.2.4.6 Aerosol Release Timing and Rates

Aerosol release timing is in accordance with the source term defined in Subsection 15.6.5.3. Aerosol release takes place in two main phases: a gap release lasting for 0.5 hour, followed by an early in-vessel release of 1.3 hours duration. During each phase, the aerosols are assumed to be released at a constant rate. These rates were obtained for each species by combining its core inventory, release fraction, and times of release.

Only cesium and iodine are released during the gap release phase. During the in-vessel release phase, the other fission product and inert species are released as well.

15B.2.4.7 Containment Thermal-Hydraulic Data

The thermal-hydraulic parameters used in the aerosol removal calculation are the containment gas temperature, the containment pressure, the steam condensation rate on the wall, the steam mole fraction, and the convective heat transfer rate, all as functions of time. The AP1000-specific parameters were obtained using MAAP4 (Reference 7) for the 3BE-1 severe accident sequence (medium LOCA with failure to inject water from the refueling water storage tank into the reactor vessel). The thermal-hydraulic data are thus consistent with a core melt sequence.

15B.2.5 Aerosol Removal Coefficients

The aerosol removal coefficients are provided in Table 15B-1 starting at the onset of core damage through 24 hours. The removal coefficients for times beyond 24 hours are not of concern because there would be so little aerosol remaining airborne at that time. The values range between 0.29 hr^{-1} and 1.1 hr^{-1} during the time between the onset of core damage (0.167 hour) and 24 hours.

These removal coefficients conservatively neglect steam condensation on the airborne particles, turbulent diffusion, and turbulent agglomeration. Additionally, the assumed source aerosol size is conservatively small being at the low end of the mass mean aerosol size range of 1.5 to 5.5 μm used in NUREG/CR-5966 (Reference 8). Selection of smaller aerosol size would underestimate sedimentation.

Unlike the case for the elemental iodine removal, there is no limit assumed on the removal of aerosols from the containment atmosphere.

15B.3 References

1. NUREG-0800, Section 6.5.2, Revision 2, "Containment Spray as a Fission Product Cleanup System."

2. Fuchs, N. A., The Mechanics of Aerosols, Pergamon Press, Oxford, 1964.

3. Waldmann, L., and Schmitt, K. H., "Thermophoresis and Diffusiophoresis of Aerosols," Aerosol Science, C. N. Davies, ed., Academic Press, 1966.

4. Talbot, L., Chang, R. K., Schefer, R. W., and Willis, D. R., "Thermophoresis of Particles in a Heated Boundary Layer," J. Fluid Mech. 101, 737-758 (1980).

5. Rahn, F. J., "The LWR Aerosol Containment Experiments (LACE) Project," Summary Report, EPRI-NP-6094D, Electric Power Research Institute, Palo Alto, Nov. 1988.

6. Petti, D. A., Hobbins, R. R., and Hagrman, D. L., "The Composition of Aerosols Generated during a Severe Reactor Accident: Experimental Results from the Power Burst Facility Severe Fuel Damage Test 1-4," Nucl. Tech. 105, p.334 (1994).

7. MAAP4 - Modular Accident Analysis Program for LWR Power Plants, Computer Code Manual, May 1994.

8. Powers D. A., and Burson, S. B., "A Simplified Model of Aerosol Removal by Containment Sprays," NUREG/CR-5966, June 1993.

9. Powers, D. A., "Monte Carlo Uncertainty Analysis of Aerosol Behavior in the AP600 Reactor Containment under Conditions of a Specific Design-Basis Accident, Part 1," Technical Evaluation Report, Sandia National Laboratories, June 1995.

Table 15B-1 (Sheet 1 of 3)
Aerosol Removal Coefficients in the AP1000 Containment Following a Design Basis LOCA With Core Melt

Time Interval (hours)	Removal Coefficient (hr^{-1})
0.167 – 0.179	1.141
0.179 – 0.200	1.013
0.200 – 0.251	0.944
0.251 – 0.292	0.882
0.292 – 0.433	0.842
0.433 – 0.631	0.901
0.631 – 0.684	0.821
0.684 – 0.801	0.781
0.801 – 0.893	0.735
0.893 – 1.033	0.699
1.033 – 1.171	0.662
1.171 – 1.233	0.627
1.233 – 1.331	0.594
1.331 – 1.395	0.562
1.395 – 1.429	0.551
1.429 – 1.475	0.576
1.475 – 1.519	0.537
1.519 – 1.579	0.510
1.579 – 1.653	0.483
1.653 – 1.776	0.458
1.776 – 1.903	0.430
1.903 – 1.991	0.462
1.991 – 2.067	0.429
2.067 – 2.176	0.396
2.176 – 2.371	0.380
2.371 – 2.621	0.337

Table 15B-1 (Sheet 2 of 3)
Aerosol Removal Coefficients in the AP1000 Containment Following a Design Basis LOCA With Core Melt

Time Interval (hours)	Removal Coefficient (hr^{-1})
2.621 – 2.822	0.320
2.822 – 2.872	0.357
2.872 – 2.973	0.327
2.973 – 3.176	0.302
3.176 – 3.684	0.287
3.684 – 3.737	0.328
3.737 – 3.839	0.304
3.839 – 3.990	0.298
3.990 – 4.090	0.317
4.090 – 4.438	0.346
4.438 – 4.684	0.369
4.684 – 4.880	0.396
4.880 – 4.928	0.449
4.928 – 5.362	0.435
5.362 – 5.460	0.459
5.460 – 5.511	0.518
5.511 – 5.608	0.487
5.608 – 6.040	0.479
6.040 – 6.090	0.537
6.090 – 6.615	0.506
6.615 – 6.753	0.567
6.753 – 7.194	0.513
7.194 – 7.285	0.594
7.285 – 7.814	0.518
7.814 – 7.904	0.581

Table 15B-1 (Sheet 3 of 3)
Aerosol Removal Coefficients in the AP1000 Containment Following a Design Basis LOCA With Core Melt

Time Interval (hours)	Removal Coefficient (hr^{-1})
7.904 – 8.431	0.528
8.431 – 8.521	0.589
8.521 – 9.387	0.529
9.387 – 9.553	0.568
9.553 – 11.189	0.530
11.189 – 14.937	0.516
14.937 – 17.610	0.506
17.610 – 24	0.492