4 NSSS SYSTEMS

This chapter describes the results of the efforts performed in the nuclear steam supply system (NSSS) area to support the uprating. Evaluations and analyses were performed to confirm that the NSSSs continue to perform their intended functions under the replacement steam generator (RSG) conditions. The systems addressed in this chapter are as follows:

Fluid systems:

- Reactor coolant system (RCS)
- Chemical and volume control system (CVCS)
- Residual heat removal system (RHRS)
- Safety injection system (SIS)
- Boron thermal regeneration system (BTRS)
- Boron recycle system (BRS)
- Liquid waste processing system (LWPS)
- Gaseous waste processing system (GWPS)
- Sampling system

NSSS/balance-of-plant (BOP) interface systems:

- Main steam system (MSS)
- Steam dump system
- Condensate and feedwater system (C&FS)
- Auxiliary feedwater system (AFWS)
- Steam generator blowdown system (SGBS)

NSSS Control Systems:

- Plant Operability Margins
- Pressure Control Component Sizing
- P-9 Permissive Setpoint
- Turbine Trip Transient
- Low-Temperature Overpressure System (LTOPS)

4.1 NSSS FLUID SYSTEMS

Westinghouse has reviewed the impact of the RSGs on the nuclear steam supply system (NSSS) fluid systems originally supplied by Westinghouse for the Callaway plant. The analyses and/or evaluations for the NSSS fluid systems have been defined to address the proposed NSSS design parameter changes associated with the RSGs. The definition of the base scope analyses and/or evaluations assumes that there are no changes to the plant design configuration as currently reflected in the Westinghouse analyses of record for the NSSS fluid systems. The applicable system functional requirements and performance criteria were reviewed relative to the NSSS design parameters to show that each system remains capable of performing its design-basis functions at the revised conditions.

Reactor Coolant System

The revised RCS conditions were assessed from a systems operation perspective with the overall conclusion that the RSGs will not affect the ability of the RCS to perform its design-basis functions and no system changes are required beyond the pressurizer spray line low-temperature alarm setpoint and the pressurizer relief tank (PRT) level setpoint. Based on the revised RCS conditions, PRT setpoints for 38 percent (low Tavg) and 60 percent (high Tavg) were calculated and forwarded to AmerenUE. Also the recommendation was made for AmerenUE to review the current pressurizer spray line low-temperature setpoint for the low Tavg temperature.

Chemical and Volume Control System

The CVCS was evaluated to assess the impact of the RSGs on the CVCS operation. Since the CVCS is normally connected to the RCS, changes in the RCS operating conditions (as defined by the NSSS design parameters) were reviewed against the system requirements. Based on the evaluation, the RSG does not impact the CVCS.

Residual Heat Removal System

The RHRS will continue to be capable of performing its intended decay heat removal function at the RSG revised RCS operating conditions.

Safety Injection System

The revised RCS operating conditions for the RSG have no direct effect on the overall performance capability of the SIS. This system will continue to deliver a selected range of calculated flow performance (minimum and maximum) as determined by interfacing system/structure operating conditions (such as, RCS pressure or containment pressure). The SIS performance is justified by acceptable plant safety analysis results, as presented in Section 6.

Boron Thermal Regeneration System

The RSG Program will not result in significant changes in core boron requirements. Therefore, the capability of the BTRS to process the stored boron will not be impacted. A review of the RSG conditions indicates that the steam generator primary-side outlet temperature remains below the 560°F design inlet temperature for the regenerative heat exchanger. This means that the heat loads defined for the letdown heat exchanger and all downstream equipment, such as the boron thermal regenerative chiller unit, will continue to conservatively envelope the actual heat loads experienced during plant operation. The RSG Program will not result in significant changes to CVCS operating pressures and temperatures. The overall conclusion from this assessment is that no system changes will be necessary for the BTRS to perform its design-basis functions with the RSG conditions.

Boron Recycle System

The RSG Program will not result in significant changes in core boron requirements. Therefore, the capability of the BRS to process and store effluents will not be impacted. This will be assessed as part of the approved reload process. The Westinghouse BRS is a standard design and the same system is utilized in many different plant designs. The BRS is designed to accommodate activity levels associated with fuel defects in cores operated at higher power levels than at the RSG conditions. The RSG will not result in any changes to CVCS operating pressures and temperatures. The overall conclusion from this assessment is that no system changes are necessary for the BRS to perform its design-basis functions with the RSG conditions, and a detailed system level evaluation is not necessary.

Liquid Waste Processing System

The primary-side steam generator outlet temperature is below 600°F at the RSG conditions. The temperature of the PRT will not be impacted by the RSG Program. The excess letdown heat exchanger is designed to cool the excess letdown flow from 560°F down to 165°F. The fluid then enters the seal water return heat exchanger or the reactor coolant drain tank (RCDT) heat exchanger (via the RCDT). Since the steam generator primary side remains less than 560°F for both the thermal design flow and the best-estimate flow conditions with the RSG, the heat loads and cooling water requirements will continue to conservatively envelope the heat loads experienced during actual plant operations. Therefore, the LWPS is not impacted by the RSG Program.

Gaseous Waste Processing System

The potential impact of the RSG operating conditions on the GWPS functions is that changes in the activity level could result in reduced capability of the GWPS to process and store waste gas effluents over the plant life. The GWPS is designed to accommodate activity levels associated with fuel defects in cores operated at higher power levels than the RSG conditions. The overall conclusion from this assessment is that no system changes are necessary for the GWPS to perform its design-basis functions with the RSG conditions.

Sampling System

The sampling system extracts samples from the RCS hot legs as well as the pressurizer. The hot leg temperatures do not increase for the RSG from the evaluation for the current condition. The heat load and cooling water requirements for the sample heat exchanger are based on an inlet temperature of 653°F, which is consistent with the pressurizer saturation temperature. The nominal pressurizer conditions remain unchanged by the RSG. Therefore, there is no impact on the sampling system due to the RSG program.

4.1.1 Conclusion

The evaluations showed that the NSSS fluid systems can perform their design functions at the RSG conditions without any system changes required beyond the pressurizer spray line low-temperature alarm setpoint and the PRT level setpoint.

4.2 NSSS/BALANCE-OF-PLANT INTERFACE SYSTEMS

As part of the Callaway RSG Program, the following BOP fluid systems were reviewed to assess compliance with Westinghouse NSSS/BOP interface guidelines (Reference 1):

- Main steam system
- Steam dump system
- Condensate and feedwater system
- Auxiliary feedwater system
- Steam generator blowdown system

The review was performed based on the range of NSSS design parameters developed to support Framatome Model 73/19T RSGs (Table 2-1).

4.2.1 Input Parameters and Assumptions

A comparison of the RSG design parameters given in Table 2-1 with the current design parameters previously evaluated for systems and components indicates differences that could impact the performance of the BOP systems. For example, the Framatome Model 173/19T RSGs and the lower limit on Tavg (570.7°F) results in about a 0.3-percent decrease in steam/feedwater mass flow rates. Additionally, the average steam generator tube plugging level of 5 percent, in combination with the upper limit on Tavg (588.4°F), results in about a 6.9-percent increase in full-load steam pressure.

4.2.2 Description of Analyses and Results

Evaluations of the above BOP systems relative to compliance with Westinghouse NSSS/BOP interface guidelines (Reference 1) were performed to address the NSSS design parameters for RSGs which include ranges for parameters such as Tavg (570.7 to 588.4°F) and steam generator tube plugging (0 to 5 percent). These ranges on design parameters result in ranges on BOP parameters such as steam generator outlet pressure (867 to 1,033 psia). In addition, the RSG parameters include a feedwater temperature range of 390° to 446°F. The NSSS/BOP interface evaluations were performed to address these NSSS and BOP design parameters. The results of the NSSS/BOP interface evaluations are described in the following subsections.

Main Steam System

Operation at reduced full-load steam pressures and corresponding higher steam line pressure drops will have a negative effect on plant heat rate. It is recommended that the plant be operated at the highest achievable full-load steam pressure to minimize plant heat rate. The capacity of the main steam safety valves (MSSVs) meets the Westinghouse sizing criteria for the range NSSS design parameters approved for the RSGs. An evaluation of the capacity of the atmospheric relief valves (ARVs) concluded that the original design basis in terms of cooldown capability could still be achieved over the full range of NSSS design parameters approved for the RSGs. The NSSS/BOP interface systems requirements (fast closure time) imposed on the design main steam isolation valves (MSIVs) and associated pipe loads, are not impacted by the RSGs.

Steam Dump System

The Westinghouse sizing criterion recommends 40 percent of rated steam flow to permit the NSSS to withstand an external load reduction up to 50 percent of plant rated electrical load. NSSS operation within the range of design parameters approved for the RSGs will result in either reduced or increased steam dump capability (39 to 47 percent of rated steam flow) relative to the Westinghouse sizing criterion. The NSSS control systems margin to trip analysis provides an evaluation of the adequacy of steam dump capability in conjunction with the control system setpoints.

Condensate and Feedwater System

The C&FS hydraulic evaluation indicates that the change in full-load main feedwater control valves (FCVs) pressure drop will be ≤ 3 psi and full-load FCV lift will be ≤ 1.0 percent for the range of NSSS design parameters approved for the RSGs. This results in the FCV lift of ≤ 90 percent. Accordingly, the range of FCV operating conditions at full load is judged to be acceptable for steady-state and transient feedwater control. The NSSS/BOP interface systems requirements (fast closure time) imposed on the design feedwater isolation valves (FWIVs) and FCVs, and associated pipe loads, are not impacted by the RSGs.

Auxiliary Feedwater System

The minimum flow requirements of the AFWS are dictated by safety analyses, and the results of the revised safety analyses for the RSGs confirmed that the current AFWS performance is acceptable. The plant licensing bases require that, in the event of a loss-of-all-ac-power, sufficient condensate storage tank (CST) usable inventory must be available to bring the unit from full power to an RCS temperature of 406°F during a 4-hour coping period. For the range of NSSS operating conditions approved for the RSGs, analyses indicate a minimum usable inventory of 150,000 gallons is required to meet the loss-of-ac-power licensing basis. Therefore, no change is required to the plant Technical Specifications, since the plant Technical Specifications (Technical Specification B3.7.6) currently ensure a minimum usable volume of 158,000 gals.

Steam Generator Blowdown System

The blowdown required to control secondary chemistry and steam generator solids will not be impacted by the RSGs. Based on the revised range of NSSS design parameters approved for the RSGs, the no-load steam pressure (1,106 psia) remains the same. However, the minimum full-load steam pressure could be as low as 867 psia or 41 psi lower than the current minimum full-load pressure (908 psia). This decrease in blowdown system inlet pressure will impact the required maximum lift of the blowdown flow control valves. Therefore, the design of the blowdown system control valves must be reviewed by AmerenUE to determine if blowdown flow control is adequate for the range of NSSS design parameters approved for RSGs.

4.2.3 Conclusion

The evaluations showed that the NSSS/BOP interface systems are acceptable at the RSG conditions without any system changes. However, the design of the blowdown system control valves must be reviewed by AmerenUE to determine if blowdown flow control is adequate for the range of NSSS design parameters approved for RSGs.

4.2.4 References

1. WCAP-7451, Rev. 2, Westinghouse Steam Systems Design Manual, August 1973.

4.3 NSSS CONTROL SYSTEMS

The RSG Program included analyses and evaluations in the following areas in order to assess the NSSS control systems performance at the revised conditions:

- Plant operability margins
- Pressure control component sizing
  - Power-operated relief valves (PORVs)
  - Pressurizer spray valves
  - Steam dump valves
  - Pressurizer heaters
- The P-9 permissive setpoint
- Turbine trip transient
- Low-temperature overpressure system

4.3.1 Plant Operability Margins

The operating margin to the various reactor trip and engineered safety features (ESF) actuation setpoints during and following normal (Condition I) operating transients at RSG conditions was evaluated. When there is a change in the plant operating conditions, reactor core kinetics, or the control system setpoints, the effect of the changes on the plant operating margins needs to be evaluated. The RSG Program includes a full-power Tavg window of 570.7° to 588.4°F. In addition, the steam dump and the pressurizer level control systems setpoints were revised for the RSG Program. Because of these changes, the relevant reactor trip and ESF setpoints and time constants were evaluated to assess the plant operability margin during and following the Condition I operating transients at RSG conditions. The margins to the reactor trip and ESF setpoints were shown to be adequate for RSG conditions.

4.3.2 Pressure Control Component Sizing

The installed capacities of the following NSSS pressure control components were evaluated for the RSG conditions. The adequacy of the installed pressurizer and steam generator safety valves with the RSGs is confirmed by the acceptable results of the appropriate non-loss-of-coolant accident safety analyses, as presented in Section 6.

Pressurizer PORVs

The installed pressurizer PORVs capacity should limit the peak pressurizer pressure to a value below the high pressurizer pressure reactor trip setpoint on a 50-percent load rejection with steam dump transient. This sizing criterion for pressurizer PORVs is conservatively met if the total installed capacity is greater than or equal to the peak pressurizer insurge flow rate during and following this transient. The analysis results showed there were no automatic reactor trips on a 50-percent load rejection transient and that a maximum total relief capacity of 417,488 lb/hr of saturated steam at 2,350 psia is required at the RSG conditions. The installed total capacity of the 2 PORVs is 420,000 lb/hr of saturated steam at 2,350 psia, which is greater than the required capacity. Therefore, the installed pressurizer PORVs are adequate at the RSG conditions.

Pressurizer Spray Valves

The sizing basis for the pressurizer spray valves is to prevent challenging the pressurizer PORVs for a design-basis 10-percent step load decrease transient. For a load decrease up to 10-percent power, the spray valves are the sole means of providing pressure control without actuating the pressurizer PORVs in the automatic mode of pressure control. The analysis results showed a maximum pressurizer pressure of 2,342 psia at RSG conditions. The peak pressure remains below the pressurizer PORV opening set pressure of 2,350 psia. Therefore, the pressurizer PORVs will not be challenged for a design-basis 10-percent step load decrease transient. The installed total pressurizer spray valve capacity of no less than 900 gpm is adequate at the RSG conditions.

Steam Dump Valves

The steam dump valves are designed to function as an artificial heat sink during large-load rejections (50-percent) and to provide a means of relieving the stored energy and decay heat after a reactor trip. The main requirement for their capacity is that they be able to relieve sufficient steam to prevent an automatic reactor trip following a large-load rejection. Other secondary requirements are to avoid steam generator safety valve lifting following either a large-load rejection or a reactor trip from full-power transient. The installed steam dump valve capacity is adequate at the RSG conditions, provided that the full-load Tavg is no lower than 573°F. The results showed no automatic reactor trip and no lifting of the steam generator safety valves following a design-basis 50-percent load rejection transient and a reactor trip from full-power transient.

Pressurizer Heaters

The pressurizer heaters total installed capacity is proportional to the pressurizer volume. The required installed capacity for pressurizer heaters is 1 kW/cubic foot of pressurizer total volume. With a nominal 1,800 cubic foot pressurizer at Callaway, this equates to a required pressurizer heaters total installed capacity of 1,800 kW. The actual total installed capacity is 1,800 kW at Callaway, therefore, it meets the Westinghouse standard requirement of 1 kW/cubic foot of pressurizer total volume. The adequacy of the pressurizer heaters total installed capacity with the RSGs was evaluated as part of the plant operability margin to trip analysis which was shown to be acceptable.

4.3.2.1 Pressure Control Conclusion

The installed capacities of the NSSS pressure control components were evaluated and were shown to be acceptable at the RSG conditions. The analysis resulted in a restriction on the Tavg range. The installed steam dump valve capacity is adequate at the RSG conditions, provided that the full-load Tavg is no lower than 573°F.

4.3.3 P-9 Permissive Setpoint

The Callaway plant includes an interlock system that eliminates direct reactor trips on turbine trips below a certain power level. This interlock is designated as the P-9 permissive interlock. The current setpoint for the P-9 interlock is 50 percent of the rated thermal power (RTP). The RSG conditions have the potential to adversely affect the current P-9 setpoint. The analysis demonstrated compliance with the NRC NUREG-0737 requirement with the maximum allowable P-9 permissive setpoint (50 percent power).

4.3.4 Turbine Trip Transient

The steam generator level undergoes level changes during turbine trip transients. This level change is designated as "shrink/swell." The control system must be able to respond to these level changes and smoothly return the steam generator level back to its setpoint. The level deviation must be controlled so that a reactor trip on low-low level or a turbine trip/feedwater isolation on high-high level does not occur. Besides the control of the steam generator level, turbine trip resulting in reactor trips (that is, turbine trips above the P-9 setpoint) will cause a large level shrink as the steam voids are collapsed in the tube bundle region. An analysis was performed to assess the steam generator level shrink/swell and associated control system responses during two key transients:

- Turbine trip without reactor trip from just below the P-9 setpoint
- Turbine trip with reactor trip from above the P-9 setpoint (limiting point is 100-percent power)

All control system responses were smooth; no control system oscillatory or diverging responses were noted.

4.3.5 Low-Temperature Overpressure System

The LTOPS, also known as the cold overpressure mitigation system (COMS), provides RCS pressure relief capability during relatively low-temperature operation (that is, RCS temperature less than about 350°F). Two pressurizer PORVs are used to provide the automatic relief capability during the design-basis mass input (MI) and the design-basis heat injection (HI) transients to automatically prevent the RCS pressure from exceeding the pressure and temperature limits of 10 CFR 50, Appendix G. The design-basis MI and HI transients for Callaway are defined in the Technical Specifications. There are differences between the Model F and the RSG that would affect the current design-basis LTOPS HI transient. The HI transient was, therefore, analyzed with the Framatome Model 73/19T steam generator for the RSG Program. The design-basis MI transient for Callaway was not re-analyzed. An evaluation concluded that the current MI analysis remain applicable for the RSG Program. Using the HI transient re-analysis results and the current MI results, revised LTOPS setpoints were developed.

The revised LTOPS PORV setpoints are applicable for the Callaway RSG. The wide-range pressure uncertainty for LTOPS has increased from the current 85 to 93.3 psig for the RSG Program but the revised LTOPS setpoints are still slightly higher (about 8 psig) for the RSG Program than the current setpoints. The MI transient continues to be limiting for RCS temperatures less than about 180°F. The current LTOPS arming temperature of 275°F remains applicable for the RSG Program.

4.3.6 NSSS Control Systems Conclusion

The evaluations showed that the all NSSS control systems responses were smooth and no control system oscillatory or diverging responses resulted. There is adequate plant operating margins at the RSG conditions. Also the revised setpoints for the LTOPS were acceptable.

5 NUCLEAR STEAM SUPPLY SYSTEM COMPONENTS

Evaluations were performed to determine the effects of the replacement steam generator (RSG) parameters on the nuclear steam supply system (NSSS) components. In general, the RSG-related input used for these evaluations are the NSSS design parameters (Section 2) and the NSSS design transients. Additional input parameters specific to particular components (for example, NSSS auxiliary equipment design transients for the auxiliary equipment evaluations) were considered. The purpose of the evaluations performed for the NSSS components was to confirm that they continue to satisfy the applicable codes, standards, and regulatory guides under the RSG conditions.

Evaluations were performed in the following areas, and are described within the remainder of this section:

- Reactor vessel structural evaluation
- Reactor pressure vessel (RPV) system
- Fuel assemblies
- Control rod drive mechanisms (CRDMs)
- Reactor coolant loop (RCL) piping and supports
- Reactor coolant pumps (RCPs) and motors
- Pressurizer
- NSSS auxiliary equipment

5.1 REACTOR VESSEL STRUCTURAL

5.1.1 Introduction

Evaluations were performed for the various regions of the Callaway reactor vessel to determine the stress and fatigue usage effects of operation with Framatome Model 73/19T RSGs and the resulting revised NSSS operating conditions.

5.1.2 Input Parameters and Assumptions

The evaluations assess the effects of the revised operating parameters, design transients, and design loads, described in Sections 2.0, 3.1, and 5.2, on the most limiting locations (with regard to ranges of stress intensity and fatigue usage factors in each of the regions as identified in the reactor vessel stress reports and addenda (References 1 through 4). The evaluations consider a worst-case set of operating parameters and design transients from among conditions resulting from operation with the RSGs, and the current design bases.

5.1.3 Description of Analyses and Evaluations

All of the parameters and design transients appropriate for the operating conditions associated with the RSGs are fully addressed by the evaluations. Also, reactor vessel operation (in accordance with the Callaway RSG Program) for the remainder of the current operating license is justified. In addition, reactor vessel operation from plant startup until installation of the RSGs, and any future operation in accordance with the original design bases, is fully addressed by the stress and fatigue analyses in the reactor vessel stress reports. Where appropriate, revised maximum ranges of stress intensity and maximum usage factors were calculated to reflect the conditions of the RSG programs or to incorporate the cold overpressure mitigation system (COMS) transient. In other cases, the original design-basis stress analyses remain conservative so that no new calculations were necessary, and the maximum ranges of stress intensity and fatigue usage factors reported in the reactor vessel stress reports for Callaway are unchanged.

5.1.4 Acceptance Criteria and Results

The maximum range of primary-plus-secondary stress intensity resulting from normal and upset condition design transient mechanical and thermal loads should not exceed 3 S_m at operating temperature (Reference 5, Paragraph NB-3222.2 of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (B&PV) Code).

The maximum cumulative usage factor resulting from the peak stress intensities due to the normal and upset condition design transient mechanical and thermal loads should not exceed 1.0 in accordance with the procedure outlined in Paragraph NB-3222.4 of Reference 5.

The new conditions and revised design transients associated with the RSGs do not affect the maximum ranges of stress intensity reported in the Callaway reactor vessel stress reports. All of the results remain below the 3 S_m limit for the various locations, with the exception of the outlet nozzle safe end and the bottom-mounted instrumentation tubes. The maximum range of stress intensity at these locations is justified in exceeding the 3 S_m limit with a simplified elastic-plastic analysis in accordance with the ASME code (Reference 5) in the original Callaway stress report (References 1 through 4). Some of the maximum cumulative fatigue usage factors at the various limiting locations in the vessels also increase due to operation with the RSGs and the COMS transient. All of the cumulative usage factors, however, remain under the limit of 1.0. A separate section addressing the head adapter plugs (not found in the original design stress report) was added to the evaluation and the results summary. Table 5.1-1 provides the results.

No faulted conditions are changed as a result of the steam generator replacement. Therefore, there are no changes to the faulted stress results.

5.1.5 Conclusions

Based on the satisfactory results of the evaluations described in Table 5.1-1, the Callaway reactor vessel is acceptable for plant operation with the Framatome Model 73/19T RSGs. Considering any combination of the current design bases and the revised design transients associated with operation with the RSGs for the specified numbers of occurrences, the reactor vessel stress and fatigue analyses and evaluations justify operation with a range of vessel outlet temperatures (Thot) from 603.2°F to 620.0°F, and a range of vessel inlet temperatures (Tcold) from 538.2°F to 556.8°F. Therefore, this reactor vessel evaluation, in conjunction with the reactor vessel stress reports, addresses reactor operation within the expanded operating temperature ranges, as described above. Such operation is shown to be acceptable in accordance with the 1971 Edition of Section III of the ASME B&PV Code, with addenda through the Winter of 1972 Addenda (Reference 5) used in the original design of the Callaway reactor vessel.

5.1.6 References

1. Combustion Engineering, Inc. Report CENC-1509, "Addendum 2 to Analytical Report for Union Electric Company Callaway Nuclear Power Plant Unit No. 1 Reactor Vessel," B. H. Boyd and A. S. Hofses, March 1982.

2. Combustion Engineering, Inc. Report CENC-1303, "Analytical Report for Callaway Nuclear Power Plant Unit No. 1, Union Electric Company (SNUPPS)," B. H. Boyd and A. S. Hofses, October 1977.

3. Combustion Engineering, Inc. Report CENC-1376, "Addendum 2 to Analytical Report for Callaway Nuclear Power Plant Unit No. 1, Union Electric Company (SNUPPS)," B. H. Boyd, Jr., May 1979.

4. Westinghouse Report MED-PCE-1928, "SCP-5789 Addendum to Analytical Report for Union Electric Company Callaway Nuclear Power Plant Unit No. 1 Reactor Vessel (Feedwater Heaters Out-of-Service Transient Evaluation), S. L. Abbott, August 1984.

5. ASME Boiler and Pressure Vessel Code, Section III, "Nuclear Power Plant Components," 1971 Edition up to and including the Winter 1972 Addenda.

Table 5.1-1 Reactor Vessel Structural – Tabulation of Results		
Location	**Maximum Range of Stress Intensity**	**Cumulative Fatigue Usage Factor**
Outlet Nozzles and Supports	Outlet Nozzle Safe End*: [ ]^{a,c} ksi > 3 S_m = 53.7 ksi Nozzle: [ ]^{a,c} ksi < 3 S_m = 80.1 ksi Outlet Nozzle Support Pad [ ]^{a,c} ksi < 3 S_m = 80.1 ksi	Outlet Nozzle [ ]^{a,c} < 1.0 Support Pad [ ]^{a,c} < 1.0
Inlet Nozzles and Supports	Inlet Nozzle Safe End: [ ]^{a,c} ksi < 3 S_m = 51.0 ksi Nozzle: [ ]^{a,c} ksi < 3 S_m = 80.1 ksi Inlet Nozzle Support Pad [ ]^{a,c} ksi < 3 S_m = 80.1 ksi	Inlet Nozzle [ ]^{a,c} < 1.0 Support Pad [ ]^{a,c} < 1.0
Closure Head Flange, Vessel Flange, and Closure Studs	Closure Head Flange [ ]^{a,c} ksi < 3 S_m = 80.1 ksi Vessel Flange [ ]^{a,c} ksi < 3 S_m = 80.1 ksi Closure Studs [ ]^{a,c} ksi < 3 S_m = 123.6 ksi	Head Flange [ ]^{a,c} < 1.0 Vessel Flange [ ]^{a,c} < 1.0 Closure Studs [ ]^{a,c} < 1.0
CRDM Housings	[ ]^{a,c} ksi < 3 S_m = 69.9 ksi	[ ]^{a,c} < 1.0
Bottom Head to Shell Juncture	[ ]^{a,c} ksi < 3 S_m = 80.1 ksi	[ ]^{a,c} < 1.0
Bottom-Mounted Instrumentation Tubes	Location 1* [ ]^{a,c} ksi > 3 S_m = 69.9 ksi Location 2 [ ]^{a,c} ksi < 3 S_m = 69.9 ksi	Location 1 [ ]^{a,c} < 1.0 Location 2 [ ]^{a,c} < 1.0
Vessel Wall Transition	[ ]^{a,c} ksi < 3 S_m = 80.1 ksi	[ ]^{a,c} < 1.0
Core Support Pads	[ ]^{a,c} ksi < 3 S_m = 80.1 ksi	[ ]^{a,c} < 1.0
Head Adapter Plugs	[ ]^{a,c} ksi < 3 S_m = 80.1 ksi	[ ]^{a,c} < 1.0

Notes:

* Stress intensities are qualified in the original Callaway stress report with a simplified elastic-plastic analysis per Subsection NB of the ASME B&PV Code.

Bracketed [ ]^{a,c} information designates data that is Westinghouse proprietary, as discussed in Section 1.6 of this report.

5.2 REACTOR PRESSURE VESSEL SYSTEM

5.2.1 Reactor Internals Heat Generation Rates

5.2.1.1 Introduction

The presence of radiation-induced heat generation in reactor internals components, in conjunction with the various reactor coolant fluid temperatures, results in thermal gradients within and between the components. These thermal gradients cause thermal stress and thermal growth, which must be considered in the design and analysis of the various components. The primary design considerations are to ensure that thermal growth is consistent with the functional requirements of the components, and to ensure that the applicable ASME Code requirements are satisfied as part of the components evaluation that is described in subsection 5.2.2 of this report. In order to satisfy these requirements, the reactor internals must be analyzed with respect to fatigue and maximum allowable stress considerations.

The reactor internals components subjected to significant radiation-induced heat generation are the upper core plate (UCP), lower core plates (LCP), lower core support, core baffle plates, former plates, core barrel, thermal shield, baffle-former bolts, and barrel-former bolts. However, due to relatively low heat generation rates in the lower core support and the thermal shield, these components experience little, if any, temperature rise relative to the surrounding reactor coolant.

This section provides a description of the methodology that was used to determine the radiation-induced heat generation rates for the axial core components, that is, the upper and LCPs, and selected radial reactor internals components (that is, the core baffle, core barrel, and neutron pad) due to the RSG Program. Although design-basis neutron exposure data for the reactor internals components are documented in WCAP-9620, Revision 1 (Reference 1), key core power distribution, fuel product, and methodology differences presently exist such that the axial component data reported in WCAP-9620-R1 are non-conservative. However, as demonstrated in the Callaway cycle-specific analysis performed to support the RSG Program, the radial component data from WCAP-9620-R1 remains conservative. Key axial components for the Callaway RSG Program were addressed using previously developed baseline upper and LCP heating rates applicable to Callaway (that is, 4-loop design with a 3.00-inch UCP and a 2.00-inch LCP).

5.2.1.2 Input Parameters and Assumptions

For the core plates, baseline gamma heating rates were determined for both long- and short-term conditions since the WCAP-9620-R1 radial power distribution data were no longer deemed bounding for the reactor internals design calculations of these components. Long-term heat generation rates intended to represent time-averaged behavior are used in component fatigue analyses, whereas the short-term results are intended to provide conservative values for use in calculating maximum temperatures and thermal stresses of components. For the long-term heat generation rate evaluation of the core plates, a reactor power level of 3,950 MWt was utilized in conjunction with a flat axial core power distribution, since these parameters significantly influence the core plate gamma heating rates and the aforementioned conditions conservatively bound the Callaway RSG Program. For the short-term heat generation rate evaluation of the UCP, the reactor power of 3,950 MWt continued to be assumed and a conservative design-basis top-peaked axial power distribution from Reference 1 was utilized. Analogous conditions were applied in the short-term heating rate evaluation of the LCP. However, in this case, the design-basis bottom-peaked axial power distribution from Reference 1 was employed for conservatism.

For the radial reactor internals components, only a long-term analysis was performed. This is because it was anticipated that the current Callaway based gamma heating rates would be bounded by the corresponding data reported in WCAP-9620-R1. (This scenario was hypothesized since Callaway has transitioned to low-leakage loading patterns, whereas an out-in loading pattern was assumed in WCAP-9620-R1. Therefore, the long-term case was examined to provide confirmation that the WCAP results remained conservative for the radial components.) Since the long-term radial case of WCAP-9620-R1 was shown to be bounding, the short-term radial case of WCAP-9620-R1 would also remain bounding and, therefore, was not calculated. The long-term heat generation rate evaluation of the baffle, barrel, and thermal shield was based on the maximum Callaway reactor power level reported in Table 5.2.1-1 (3,565 MWt), and a composite radial power distribution derived from cycle-specific fuel reload designs (that is, Cycles 11 and 12 for Callaway). Although the core former region was not explicitly analyzed as part of this set of calculations, conclusions about gamma heating rates in this region of the core were inferred, since the formers are located between the baffle and barrel regions.

Design-basis heat generation rates applicable to the Callaway radial internals were obtained from Appendix E of Reference 1. The core power distributions upon which those calculations were based were derived from statistical studies of 23 independent fuel cycles from 10 four-loop reactors. These power distributions represented an upper tolerance limit for beginning-of-cycle (BOC) and end-of-cycle (EOC) power in the peripheral fuel assemblies, based on a 95-percent probability with a 95-percent confidence level. Most of the evaluated fuel cycles were based on an out-in fuel loading strategy (fresh fuel on the periphery) which, when combined with the statistical processing of the data, resulted in a design-basis core power distribution that tended to be biased high on the periphery. This high bias on the core periphery was desired by the reactor internals analysts to ensure conservative but realistic design calculations for the critical baffle-barrel region of the reactor internals. The high bias also explains why the WCAP-9620-R1 radial component heating rate results were expected to bound the corresponding Callaway values.

5.2.1.3 Description of Analysis/Evaluation and Results

The heat generation rate analyses were carried out using the DORT (DOORS 3.1 code package, Reference 2) 2-dimensional discrete ordinates transport code in the forward mode and the BUGLE-96 cross-section library (Reference 3). This suite of codes has been used to support numerous pressure vessel fluence and heat generation rate evaluations and are generally accepted by the Nuclear Regulatory Commission (NRC) for deterministic particle transport calculations (such as, neutron exposure and gamma-ray heating rate evaluations).

Two different coordinate systems were used in the 2-dimensional heating rate analyses to precisely model the components undergoing evaluation. The core baffle plates were analyzed using an x,y coordinate system, and the core barrel and thermal shield heating rates were determined using an r,θ geometric model.

5.2.1.4 Acceptance Criteria and Results

There are no specific acceptance criteria since this is an input to the reactor internals evaluation that is described in subsection 5.2.2 of this report. The results of the radiation-induced heat generation rate calculations were provided as inputs for the reactor internals evaluations described in subsection 5.2.2. The volume-averaged heat generation rates for the core plates and radial reactor internals components that were evaluated as part of this study are summarized in Table 5.2.1-1. In accordance with WCAP-9620-R1, this table also segregates the core plate heating rates into 2 distinct regions. Region A refers to the cylindrical portion of the core plates that are axially adjacent to the active fuel, and Region B refers to the annular portion of the plates that are located radially outboard of the active fuel.

As expected, the revised zone average gamma heating rates for the core plates tended to be much higher than the corresponding WCAP-9620-R1 data. As a result, the spatial distributions of long-term and short-term heating rates for the upper and LCPs that are presented in Tables 5.2.1-2 through 5.2.1-5, respectively, were also identified for consideration as part of the component evaluation that is described in subsection 5.2.2 of this report.

Table 5.2.1-1 also shows that the current Callaway zone average gamma heating rates for the core baffle, core barrel, and thermal shield continue to remain bounded by the conservative radial component heating rates that are reported in WCAP-9620-R1. Similar results may also be inferred for the core former region, since this is located between the baffle and barrel.

5.2.1.5 Conclusions

The component gamma heating rates that are presented in Tables 5.2.1-1 through 5.2.1-5 were provided as inputs to the reactor internals evaluation described in subsection 5.2.2.

5.2.1.6 References

1. WCAP-9620, Rev. 1, "Reactor Internals Heat Generation Rates and Neutron Fluences," A. H. Fero, December 1983.

2. RSICC Computer Code Collection CCC-650, "DOORS 3.1, One-, Two-, and Three-Dimensional Discrete Ordinates Neutron/Photon Transport Code System," August 1996.

3. RSIC Data Library Collection DLC-185, "BUGLE-96, Coupled 47 Neutron, 20 Gamma-Ray Group Cross Section Library Derived from ENDF/B-VI for LWR Shielding and Pressure Vessel Dosimetry Applications," March 1996.

Table 5.2.1-1 Reactor Internals Zone Average Gamma Heating Rates		
	Region Average Long-Term Heating Rates (Btu/hr-lbm)	
Location	WCAP-9620-R1 Analysis (Ref. 1, Appendix E)	Callaway Analysis*
Baffle Plate 18	945	503
Baffle Plate 19	1070	662
Baffle Plate 20	996	610
Baffle Plate 21	802	506
Core Barrel	186	103
Thermal Shield	37.8	26.8
	Upper and Lower Core Plates Heating Rates (Btu/hr-lbm)	
	WCAP-9620-R1 Analysis (Ref. 1, Appendix E)	Current Baseline Analysis (Ref. 2)
Long-Term Heating Rates		
Upper Core Plate A	23	205
Upper Core Plate B	6.37	23
Lower Core Plate A	249	903
Lower Core Plate B	52.4	88
Short-Term Heating Rates		
Upper Core Plate A	54	220
Upper Core Plate B	17.9	26
Lower Core Plate A	822	1480
Lower Core Plate B	201	167

*Average (from Cycles 11 and 12) results from the Callaway specific analysis are reported.

Table 5.2.1-2 Spatial Distribution of Long-Term Gamma Heating Rates (Btu/hr-lbm) in the UCP for Callaway Nuclear Unit 1

Radial Mesh Midpoint (inches)	Bottom Surface 0.00	Distance Through Plate (inches) 0.25	0.75	1.25	1.75	2.25	2.75	Top Surface 3.00
0.98	479	432	339	270	218	178	148	133
2.95	479	432	338	269	217	178	148	133
4.92	478	431	337	268	216	177	147	132
6.89	477	430	335	267	215	176	147	132
8.86	475	428	334	266	215	176	146	132
10.83	473	427	333	265	214	175	146	131
12.80	472	425	332	264	213	175	145	131
14.76	471	424	331	263	213	174	145	130
16.73	469	423	330	263	212	174	145	130
18.70	468	422	330	262	212	173	144	130
20.67	467	421	329	262	211	173	144	130
22.64	467	421	329	262	211	173	144	130
24.61	467	421	329	262	211	173	144	130
26.57	467	421	329	262	211	173	144	130
28.54	467	421	329	262	211	173	144	130
30.51	467	421	329	262	211	173	144	130
32.48	467	421	329	262	211	173	144	129
34.45	466	420	328	262	211	173	144	129
36.42	464	418	327	261	210	172	143	129
38.39	461	416	325	259	209	171	142	128
40.35	457	412	322	257	207	169	141	127
42.32	451	406	318	253	204	167	139	125
44.29	442	399	312	248	200	164	136	122
46.26	431	388	304	242	195	160	133	119
48.23	416	375	293	234	188	154	128	115
50.20	396	358	280	223	179	147	122	109
52.17	372	336	263	209	169	138	114	103
54.13	344	310	243	193	156	127	106	95
56.10	312	281	220	175	141	116	96	86
58.07	278	251	196	156	126	103	86	77
60.04	243	219	171	137	110	90	75	67
62.01	207	187	146	116	94	77	64	57
63.78	175	157	123	98	79	65	54	48
64.96	152	137	107	85	68	56	46	41
65.65	136	123	96	76	61	50	41	37
66.15	123	111	86	68	55	45	37	33
66.64	91	83	67	53	43	35	29	26
67.20	64	59	48	39	31	26	22	20
67.89	55	49	36	29	23	19	16	15
68.70	54	46	31	23	18	14	13	12
69.52	53	45	29	20	14	12	10	10
70.33	51	43	27	18	13	10	9	8
71.15	48	41	25	17	12	9	7	7
71.96	44	37	23	15	10	8	7	6
72.78	40	34	21	14	9	7	6	5
73.59	35	29	18	12	8	6	5	5
74.00	33	27	17	11	7	5	5	4

Table 5.2.1-3 Spatial Distribution of Short-Term Gamma Heating Rates (Btu/hr-lbm) in the UCP for Callaway Nuclear Unit 1

Radial Mesh Midpoint (inches)	Bottom Surface 0.00	Distance Through Plate (inches) 0.25	0.75	1.25	1.75	2.25	2.75	Top Surface 3.00
0.98	516	466	365	290	234	192	160	144
2.95	516	465	363	289	233	191	159	143
4.92	515	464	362	288	232	191	159	143
6.89	514	463	361	287	232	190	158	143
8.86	512	462	360	286	231	189	158	142
10.83	511	460	359	285	230	189	157	142
12.80	510	459	358	284	229	188	157	141
14.76	508	458	357	284	229	188	157	141
16.73	507	456	356	283	228	187	156	141
18.70	506	455	355	282	228	187	156	140
20.67	505	455	355	282	227	187	156	140
22.64	504	454	354	282	227	186	156	140
24.61	504	454	354	282	227	186	156	140
26.57	504	454	354	282	227	187	156	140
28.54	504	454	354	282	227	187	156	140
30.51	504	454	354	282	227	186	156	140
32.48	503	453	354	281	227	186	155	140
34.45	502	452	353	281	227	186	155	140
36.42	500	450	352	280	226	185	154	139
38.39	496	447	349	278	224	184	153	138
40.35	491	443	346	275	222	182	152	136
42.32	484	437	341	271	219	179	149	134
44.29	475	428	334	266	215	176	146	132
46.26	462	417	326	259	209	171	142	128
48.23	446	402	314	250	201	165	137	123
50.20	425	383	299	238	192	157	131	117
52.17	399	360	281	223	180	147	123	110
54.13	369	332	259	206	166	136	113	102
56.10	334	301	236	188	151	124	103	93
58.07	298	269	210	167	135	111	92	83
60.04	261	235	184	146	118	97	80	72
62.01	222	201	157	125	101	82	69	62
63.78	187	169	132	105	85	69	58	52
64.96	163	147	115	91	73	60	50	45
65.65	145	131	102	81	65	53	44	40
66.15	131	118	92	73	59	48	40	36
66.64	98	89	72	57	46	38	32	29
67.20	70	64	53	43	34	28	24	22
67.89	62	55	41	32	25	21	18	17
68.70	61	53	35	26	20	16	14	13
69.52	61	52	33	23	17	13	12	11
70.33	59	50	32	21	15	12	10	9
71.15	56	47	30	20	14	10	9	8
71.96	52	44	28	18	13	9	8	7
72.78	47	40	25	16	11	8	7	6
73.59	42	35	22	14	10	7	6	5
74.00	39	33	20	13	9	7	6	5

Table 5.2.1-4 Spatial Distribution of Long-Term Gamma Heating Rates (Btu/hr-lbm) in the Lower Core Plate for Callaway Nuclear Unit 1

Radial Mesh Midpoint (inches)	Bottom Surface 0.00	Distance Through Plate (inches) 0.25	0.75	1.25	1.75	Top Surface 2.00
0.98	694	782	958	1196	1518	1679
2.95	693	780	956	1196	1522	1684
4.92	693	781	956	1197	1524	1687
6.89	690	778	953	1193	1519	1683
8.86	686	773	946	1185	1507	1668
10.83	680	766	939	1174	1493	1652
12.80	676	761	932	1165	1482	1641
14.76	672	757	927	1159	1474	1631
16.73	670	755	924	1156	1470	1628
18.70	669	753	922	1153	1467	1624
20.67	667	751	919	1150	1463	1619
22.64	665	749	916	1146	1458	1613
24.61	665	748	915	1144	1455	1611
26.57	667	750	918	1148	1460	1616
28.54	670	755	924	1157	1471	1628
30.51	675	760	932	1166	1484	1642
32.48	677	764	936	1173	1492	1651
34.45	678	765	937	1174	1493	1653
36.42	678	764	936	1172	1491	1650
38.39	677	763	935	1171	1490	1649
40.35	678	764	937	1172	1492	1652
42.32	679	766	941	1178	1500	1660
44.29	681	769	945	1185	1508	1670
46.26	679	768	946	1187	1511	1674
48.23	670	759	937	1177	1499	1660
50.20	650	737	912	1146	1460	1617
52.17	616	700	866	1090	1388	1537
54.13	567	644	798	1004	1279	1417
56.10	505	573	708	890	1134	1256
58.07	434	491	604	758	965	1068
60.04	359	405	496	621	788	872
62.01	286	321	391	488	618	683
63.78	224	251	304	377	476	525
64.96	186	207	249	308	386	425
65.65	163	180	216	266	331	363
66.15	144	160	191	236	292	320
66.64	120	133	158	197	253	280
67.20	98	107	127	161	216	244
67.89	79	87	103	134	185	211
68.70	64	71	84	111	157	180
69.52	54	59	70	93	135	156
70.33	45	49	58	79	117	136
71.15	38	42	49	67	102	120
71.96	32	35	42	58	89	105
72.78	26	29	35	49	76	90
73.59	22	24	28	40	64	76
74.00	19	21	25	36	58	69

Table 5.2.1-5 Spatial Distribution of Short-Term Gamma Heating Rates (Btu/hr-lbm) in the Lower Core Plate for Callaway Nuclear Unit 1

Radial Mesh Midpoint (inches)	Bottom Surface 0.00	Distance Through Plate (inches) 0.25	0.75	1.25	1.75	Top Surface 2.00
0.98	1178	1313	1584	1956	2457	2708
2.95	1174	1310	1581	1957	2464	2717
4.92	1175	1310	1581	1958	2465	2719
6.89	1171	1306	1576	1951	2458	2711
8.86	1163	1297	1565	1938	2440	2690
10.83	1154	1287	1553	1922	2418	2666
12.80	1148	1279	1542	1908	2401	2648
14.76	1142	1273	1534	1898	2388	2633
16.73	1138	1268	1530	1893	2382	2627
18.70	1135	1265	1526	1888	2376	2620
20.67	1132	1262	1522	1883	2370	2613
22.64	1130	1259	1517	1877	2362	2605
24.61	1129	1258	1516	1875	2360	2602
26.57	1133	1262	1521	1881	2367	2610
28.54	1138	1269	1531	1894	2383	2628
30.51	1145	1277	1541	1908	2402	2649
32.48	1149	1282	1549	1918	2414	2662
34.45	1150	1284	1550	1920	2417	2665
36.42	1150	1283	1549	1918	2415	2663
38.39	1149	1282	1548	1917	2414	2662
40.35	1149	1283	1550	1919	2417	2666
42.32	1151	1286	1555	1927	2428	2678
44.29	1151	1288	1561	1935	2439	2690
46.26	1145	1283	1559	1935	2439	2691
48.23	1128	1265	1541	1915	2414	2664
50.20	1092	1227	1497	1863	2348	2591
52.17	1035	1163	1421	1769	2230	2461
54.13	953	1071	1308	1629	2054	2267
56.10	850	954	1162	1446	1824	2012
58.07	732	820	995	1235	1557	1717
60.04	609	680	822	1016	1277	1408
62.01	488	543	652	805	1009	1111
63.78	387	428	511	627	783	861
64.96	324	357	423	517	641	703
65.65	285	313	370	450	554	606
66.15	253	279	330	403	493	539
66.64	214	235	277	342	432	478
67.20	177	193	226	285	377	423
67.89	146	160	188	243	332	376
68.70	122	134	158	208	290	331
69.52	104	114	136	181	257	295
70.33	89	98	117	158	229	265
71.15	76	84	101	138	204	236
71.96	64	72	87	119	179	209
72.78	54	61	73	102	155	181
73.59	45	50	60	83	129	152
74.00	40	44	53	74	116	138

5.2.2 Reactor Vessel Internals

5.2.2.1 Introduction

The RPV system consists of the reactor vessel, reactor internals, fuel, and CRDMs. The reactor internals function to support and orient the reactor core fuel assemblies and control rod assemblies, absorb control rod assembly dynamic loads, and transmit these and other loads to the reactor vessel. The reactor vessel internal components also function to direct coolant flow through the fuel assemblies (core), to provide adequate cooling flow to the various internals structures, and to support in-core instrumentation. They are designed to withstand forces due to structure deadweight, preload of fuel assemblies, control rod assembly dynamic loads, vibratory loads, earthquake accelerations, and loss-of-coolant accident (LOCA) loads.

Operating a plant at conditions other than those considered in the original design requires that the reactor vessel/internals/fuel system interface be addressed in order to ensure compatibility and that the structural integrity of the reactor vessel/internals/fuel system is not adversely affected. In addition, thermal-hydraulic analyses are required to determine plant-specific input to the LOCA and non-LOCA safety analyses as well as NSSS performance evaluations.

Generally, the areas of concern most affected by changes in system operating conditions are:

- Reactor internals system thermal/hydraulic performance
- Rod control cluster assembly (RCCA) scram performance
- Flow-induced vibration (FIV)
- Reactor internals system structural response and integrity

Figure 5.2.2-1 illustrates the various components and features of the reactor internals system for the Callaway plant. The lower core support assembly consists of the lower support plate, lower support columns, and LCP, and supports the fuel assemblies on the sides and at the bottom. The hold-down spring rests on top of the flange of the lower core support assembly. The upper core support assembly consists of the upper support plate, upper support columns, and UCP, and rests on top of the hold-down spring. The vessel upper head compresses the hold-down spring providing joint preload.

The core barrel, which is part of the lower core support assembly, provides a flow boundary for the reactor coolant. When the primary coolant enters the reactor vessel, it impinges on the side of the core barrel and is directed downward through the annulus formed by the gap between the outside diameter of the core barrel and the inside diameter of the vessel. The flow then enters the lower plenum area between the bottom of the lower support plate and the vessel bottom head and is redirected upward through the core. After passing through the core, the coolant enters the upper core support region and then proceeds radially outward through the reactor vessel outlet nozzles. Another portion of the primary coolant bypasses the fuel and cools the upper head region. The perforations in the various components, such as the lower support plate, control and meter the flow through the core.

The purpose of this section of the report is to summarize the work performed to assess the effect on the RPV/internals system due to replacement steam generators at Callaway.

5.2.2.2 Input Parameters

The principal input parameters utilized in the reactor internal components and RPV system analyses for the Callaway RSG Program are:

1. Reactor Coolant System (RCS) Conditions

   Table 5.2.2-1 presents a summary of the RCS conditions utilized in the RPV/internals calculations.

2. NSSS Primary-Side Design Transients

   The normal and upset transients for the primary loop were used.

3. Internal Heat Generation Rates

   The thermal loads on a number of reactor internal components are affected by the combination of internal heat generation due to gamma heating in the structures in conjunction with the change in primary-side design transients. See subsection 5.2.1 for a detailed discussion on heat generation rates.

4. Fuel Design

   Geometric, hydraulic, and dynamic parameters for a full core of Westinghouse 17x17 VANTAGE + (V+) fuel were utilized in the evaluations.

These parameters were included in the various analyses described in the following subsections.

5.2.2.3 Thermal/Hydraulic System Evaluations

System Pressure Losses

The principal RCS flow route through the RPV system at Callaway begins at the four inlet nozzles. At this point, flow turns downward through the reactor vessel core barrel annulus. After passing through this downcomer region, the flow enters the lower reactor vessel dome region. This region is occupied by the internals energy absorber structure, lower support columns, bottom-mounted instrumentation columns, and supporting tie plates. From this region, flow passes upward through the LCP, and into the core region. After passing up through the core, the coolant flows into the upper plenum, turns, and exits the reactor vessel through the four outlet nozzles. Note that the upper plenum region is occupied by support columns and RCCA guide columns.

A key area in evaluation of core performance is the determination of hydraulic behavior of coolant flow within the reactor internals system, that is, vessel pressure drops, core bypass flows, RPV fluid temperatures and hydraulic lift forces. The pressure loss data is a necessary input to the LOCA and non-LOCA safety analyses and to overall NSSS performance calculations. The hydraulic forces are critical in the assessment of the structural integrity of the reactor internals, core clamping loads generated by the internals hold-down spring, and the stresses in the reactor vessel closure studs. The analysis determined the distribution of pressure and flow within the reactor vessel, internals, and the reactor core.

Bypass Flow Analysis

Bypass flow is the total amount of reactor coolant flow bypassing the core region and is not considered effective in the core heat transfer process. Since variations in the size of some of the bypass flow paths (such as, gaps at the outlet nozzles and the core barrel) occur during manufacturing or change due to different fuel assembly designs or changes in the RCS conditions, plant-specific as-built dimensions are used in order to demonstrate that the bypass flow limits are not violated. Therefore, analyses are performed to determine core bypass flow values to either show that the design bypass flow limit for the plant will not be exceeded or to determine a revised design core bypass flow.

The present design core bypass flow limit is 8.6 percent (with thimble plugs removed) of the total reactor vessel flow. The purpose of this evaluation is to ensure that the design value of 8.6 percent can be maintained at the RCS conditions. The principal core bypass flow paths are:

1. Baffle-barrel region
2. Vessel head cooling spray nozzles
3. Core barrel – reactor vessel outlet nozzle gap
4. Fuel assembly – baffle plate cavity gap
5. Fuel assembly thimble tubes

Fuel assembly hydraulic characteristics and system parameters (such as, inlet temperature, reactor coolant pressure and flow), were used to determine the impact of the RSG conditions on the total core bypass flow. The total core bypass flow value was determined to be 7.26 percent. Therefore, the design core bypass flow value of 8.6 percent of the total vessel flow is maintained.

Hydraulic Lift Forces

The reactor internals hold-down spring is essentially a large-diameter Belleville-type spring of rectangular cross-section. The purpose of this spring is to maintain a net clamping force between the reactor vessel head flange and upper internals flange and the reactor vessel shell flange and the core barrel flange of the internals. An evaluation was performed to determine hydraulic lift forces on the various reactor internal components to ensure that the reactor internals assembly would remain seated and stable for all conditions. The result shows that, at the RSG RCS conditions, the Callaway reactor internals assembly would remain seated and stable.

Momentum Flux and Fuel Rod Stability and Baffle-Plate Pressure-Relief Hole Velocity

Baffle jetting is a hydraulically induced instability or vibration of fuel rods caused by a high-velocity jet of water. This jet is created by high-pressure water being forced through gaps between the baffle plates, which surround the core. The baffle jetting could lead to cladding failure and the disbursement of the uranium into the coolant. At Callaway, the potential for baffle-jetting-related failures are significantly reduced with the upflow design in the baffle-barrel region.

The result of the evaluation indicated that baffle jetting related fuel damage is not excepted to occur during normal reactor operation and that the and baffle plate pressure relief hole velocity was acceptable.

5.2.2.4 Rod Cluster Control Assembly Scram Performance Evaluation

The RCCAs represent perhaps the most critical interface between the fuel assemblies and the other internals components. It is imperative to show that the new RCS conditions will not adversely impact the operation of the control rods, either during accident conditions or normal operation.

The evaluation results indicated that the maximum drop time-to-dashpot entry of 2.7 seconds for the Callaway plant remains applicable for accident analyses.

5.2.2.5 Mechanical System Evaluations

This section addresses the impact of the RSG Program on the RPV dynamic analysis (LOCA and seismic). Changes in plant operating conditions impact the performance of the RPV and its internals under all modes of operation. It is, therefore, important that the mechanical response of the RPV and its internals be evaluated.

5.2.2.5.1 Loss-of-Coolant-Accident Analysis

Analyses have been performed to determine the impact of a LOCA on the Callaway reactor vessel and internal components. The analysis considered 3 separate break locations:

- An accumulator line break (cold leg)
- A pressurizer surge line break (hot leg)
- A residual heat removal line break (hot leg)

LOCA Results Summary

Core plate motions and reactor vessel motions were generated and transmitted to the cognizant engineering group for use in the fuel grid and reactor coolant loop piping analysis for the RSG.

Impact loads between various vessel-to-internal and internal-to-internal component interfaces were generated and were compared to loads listed in the generic 4-loop stress report WNEP-7702 (Reference 1). The reactor vessel internal components qualified in Reference 1 are similar to the Callaway reactor vessel internal components. Therefore, because the calculated loads for Callaway are less than the generic stress report loads, the Callaway internal components remain qualified.

5.2.2.5.2 Seismic Analysis

Analyses have been performed to determine the impact of a seismic event on the Callaway reactor vessel and internal components. The seismic event analyzed corresponds to a safe shutdown earthquake (SSE) event.

Seismic Analysis Results Summary

Impact loads between various vessel-to-internal and internal-to-internal component interfaces were generated. Results of the analysis were compared to loads listed in the generic 4-loop stress report WNEP-7702 (Reference 1). The reactor vessel internal components qualified in Reference 1 are similar to the Callaway reactor vessel internal components. Therefore, because the calculated loads for Callaway are less than the generic stress report loads, the Callaway internal components remain qualified.

5.2.2.5.3 Flow-Induced Vibrations

Flow-induced vibrations of pressurized water reactor (PWR) internals have been studied at Westinghouse for a number of years. The objective of these studies was to demonstrate the structural integrity and reliability of reactor internal components. These efforts have included in-plant tests, scale model tests, as well as tests in fabricators' shop and bench tests of components along with various analytical investigations. The results of these scale model and in-plant tests indicate that the vibrational behavior of 2-, 3-, and 4-loop plants is essentially similar; and the results obtained from each of the tests compliment one another and make possible a better understanding of the FIV phenomena.

The purpose of this section is to show whether or not the vibration characteristics of the Callaway reactor internals are significantly affected by the increase in mechanical design flow and decrease in the vessel/core inlet temperature and that the structural integrity of the Callaway reactor internals is not impaired with regard to FIVs.

The design parameters that could potentially influence the FIV response of the reactor internals include the inlet nozzle flow velocities, vessel/core inlet temperatures, and the vessel outlet temperatures. An evaluation was performed. The lower temperatures were found to have a negligible effect on the FIV loads.

The higher mechanical design flow rate at Callaway has been shown to be acceptable with regards to FIV loading of the reactor internal components.

5.2.2.6 Structural Evaluation of Reactor Internal Components

In addition to supporting the core, a secondary function of the reactor vessel internals assembly is to direct coolant flows within the vessel. While directing the primary flow through the core, the internals assembly also establishes secondary flow paths for cooling the upper regions of the reactor vessel and for cooling the internals structural components. Some of the parameters influencing the mechanical design of the internals lower assembly are the pressure and temperature differentials across its component parts and the flow rate required to remove the heat generated within the structural components due to radiation (such as, gamma heating). The configuration of the internals provides for adequate cooling capability. Also, the thermal gradients, resulting from gamma heating and core coolant temperature changes, are maintained below acceptable limits within and between the various structural components.

Structural evaluations are required to demonstrate that the structural integrity of the reactor components is not adversely affected directly by the change in RCS conditions and transients and/or by secondary effects of the change on reactor thermal hydraulic or structural performance. The presence of heat generated in reactor internal components, along with the various fluid temperatures, results in thermal gradients within and between components. These thermal gradients result in thermal stresses and thermal growth that must be accounted for in the design and analysis of the various components.

The Callaway reactor internals were designed to Subsection NG of the ASME B&PV Code Section III (Reference 2). The structural integrity of the Callaway reactor internals design has been shown by analyses performed on both generic and plant-specific bases. These analyses were used as the basis for the evaluation of the Callaway reactor internal components for the revised thermal transients and gamma heating.

5.2.2.6.1 Lower Core Plate Evaluation

Introduction

The LCP for the Callaway plant is a perforated circular plate that supports and positions the fuel assemblies. The plate contains numerous holes to allow fluid flow through the plate. The fluid flow is provided to each fuel assembly and the baffle-barrel region flow exits through the holes. The plate is bolted at the periphery to a ring welded to the inside diameter of the core barrel. The center span of the plate is supported by the lower support columns, which are attached at the lower end to the lower support plate.

For the Callaway RSG Program, this report documents thermal and structural analyses on the LCP at steady-state and transient conditions. Reactor power (heat generation rates), flow rates, fluid temperatures, and thermal transients affect thermal loads on the LCP. New gamma heating rates, along with changes to the existing thermal design transients based on new operating conditions, have altered the thermal loads on the LCP. Therefore, it is required that the stresses on the plate be re-evaluated and compared to the allowable stress values outlined in the ASME Code Subsection NG (Reference 2).

In the LCP evaluation, the new heat generation rates were applied at the appropriate steady-state and transient conditions at which high-temperature gradients exist within the LCP. Analyses were conducted for four conditions, including steady-state full-power, full-flow, steady-state 118-percent overpower, transient up envelope, and transient down envelope. It was determined that the steady-state conditions will create maximum temperature gradients and concurrently maximum stress intensities within the LCP.

Input Data

The significant inputs to the analyses include the following:

1. Thermal properties of SA 240 Type 304 stainless steel from ASME B&PV Code, 1974 edition (Reference 3).

2. Heat generation rates

3. New thermal transients

4. Existing thermal transients from generic stress report WNEP-7602 (Reference 4)

5. NSSS design parameter conditions for Cases 1 and 2 from Table 5.2.2-1. Cases 1 and 2 were examined because these conditions maximized the thermal stresses on the LCP.

Acceptance Criteria

The acceptance criterion for this calculation is the ASME B&PV, Section III, Division 1, 1974 edition (References 2 and 3). This document provided the LCP material properties used in the analysis and was the criteria for the primary-plus-secondary stress and fatigue usage limits.

Results

The LCP was evaluated to the intent of the ASME Code, Subsection NG (Reference 2).

The stresses determined at steady-state 100-percent power conservatively envelope all normal and upset transients, except overpower. Therefore, the power range of 100 percent is applied to all unit loading and unloading transients (13,200 cycles). The long-term heating rates apply to these conditions. The minimum stress case for all unit loading and unloading transients is approximately a zero-stress state, associated with the plant in a cold shutdown or hot standby (0-percent power) condition. Moreover, no other conditions occur which can cause a significant stress reversal at the maximum stress locations. Therefore, the zero stress state is the minimum condition and steady-state 100-percent power stress state is the maximum condition.

The short-term heating rates are determined based on nuclear transient conditions. The short-term heat generation rates are based on Condition II transients (Reference 5). These conditions are generally noted in a transient document as upset or in the Code as Level B. In the RSG evaluation, the short-term heating rates will be applied at two types of conditions, the applicable upset transients and the overpower events. For the upset transients, only those that could potentially impact core reactivity at power need to be considered including Reactor Trip, Loss of Flow, Loss of Load, and Loss of Power.

An analysis on the LCP for the emergency and faulted conditions was not required and the results from the existing stress report (Reference 1) are still applicable. This is because a change in the thermal loading will not affect a stress analysis for emergency and faulted conditions. Moreover, the LOCA loads are most limiting for faulted conditions, and the existing LOCA evaluation for a 1 ft^2 LOCA break bounds the analysis with leak-before-break LOCA. From Reference 1, the mechanical stresses for faulted conditions bound emergency conditions. Test conditions do not need to be evaluated separately because they are already included in the normal conditions.

A summary of applicable combinations of critical associated loads is listed in Table 5.2.2-2. The critical load parameter is the heat generation rate applied to the transients. The power is the percent power associated with the condition.

Note that the list of cycles for the transients is based on a 40-year life of the plant. Therefore, the use of these amounts of cycles is conservative for the RSG Program evaluation.

The seismic cycles are evaluated as 20 events, 20 cycles each, which equates to 400 cycles. These events will be combined with the conditions that cause the most severe stresses in the LCP. Therefore, 40 of these cycles can be combined with the overpower thermal events, while the remaining 360 cycles will be combined with 100-percent power Level 2 (short-term) events to be conservative.

The normal plant heatup and cooldown transients (200 cycles) are not included in the fatigue evaluation, since they cause an insignificant stress in the LCP compared to the steady-state 100-percent power condition.

Secondary Stress Calculations

The thermal stresses for each of the following transient conditions were calculated:

1. Unit loading/unloading 0–100-percent power, short-term heat rates
2. Unit loading/unloading 0–100-percent power, long-term heat rates
3. Overpower events, short-term heat rates

Primary + Secondary Stress Evaluations According to the ASME Code

The allowable stress for $P_m + P_b + Q$ is $3 \cdot S_m$. Conservatively, S_m was taken at 650°F. This criterion is applied to the most limiting thermal stress case, overpower events, short-term heat rates. This value was combined with the maximum stress intensity due to operating basis earthquake (OBE) + mechanical loads for the evaluation.

The resulting margin of safety is acceptable according to the ASME Code as indicated in Table 5.2.2-3. Cases 1 and 2, which have stress values less than the overpower condition, are met as well. Although the margin of safety is small, it should be noted that conservatisms exist in the ASME Code allowables.

Fatigue Evaluation According to the ASME Code

The fatigue analysis involves determining the alternating stress (Sa) from the peak stress (F), per NG-3222.4(e), and obtaining an allowable number of stress cycles from the design fatigue curve I-9.2 of Reference 3. From NG-3222.4(e)(4), the alternating stress must be corrected for the effects of the elastic modulus. This is done by multiplying half the peak stress by the ratio of the modulus used in the model in Figure I-9.2 (E_c = 26.0e6 psi) to the modulus used in the analysis at 650°F ($E_{@650}$ 25.1e6 psi). Note that a stress concentration factor is already considered in the model, and therefore an adjustment to the peak stress is not needed.

The peak primary stress intensity from OBE + mechanical loads was determined and the resultant alternating stress, Sa, was calculated as follows.

The peak secondary stresses were determined at 100-percent power as shown in Table 5.2.2-4. The peak stress for 118-percent overpower condition is calculated by scaling up the 100-percent power short-term peak stress by 18 percent. A summary of the peak and alternating stresses is shown in Table 5.2.2-4.

To be conservative, the OBE + mechanical load cycles were combined with the most severe secondary-stress conditions until the 400 cycles are exhausted. Therefore, 40 of the 400 cycles are combined with the overpower events, short-term condition, while the remaining 360 cycles will combine with 360 cycles from the unit loading/unloading 0-100-percent power, short-term condition. The resulting alternating stress values are used to determine the allowable cycles from the design fatigue curve, Figure I-9.2 of Reference 3. Because of the available margin, the cycles will be conservatively estimated from the curve. The incremental usage factor for each combination is combined based on the cumulative damage criteria and summarized in Table 5.2.2-5. The LCP fatigue usage factor is below the fatigue usage limit of 1.0.

Conclusion

Based on the results presented in Table 5.2.2-3, it can be concluded that the stress and fatigue values for LCP are within ASME Code requirements for the Callaway RSG Program.

5.2.2.6.2 Upper Core Plate Evaluations

The UCP for Callaway is a perforated circular plate that is located between the fuel assemblies and the upper guide tubes and upper support columns. The plate contains numerous holes to allow fluid flow through the plate. The plate is exposed to fluid flow from each fuel assembly and from the baffle-barrel region. The center span of the plate is supported by the upper support columns that are attached at the lower end of the upper support plate. The plate is pinned along the periphery to the core barrel to eliminate rotation.

For the Callaway RSG Program, this report documents thermal and structural analyses on the UCP at steady-state and transient conditions. Reactor power (heat generation rates), flow rates, fluid temperatures, and thermal transients affect thermal loads on the LCP. New gamma heating rates, along with changes to the existing thermal design transients based on new operating conditions, have altered the thermal loads on the UCP. Therefore, it is required that the stresses on the plate be reevaluated and compared to the allowable stress values outlined in the ASME Code Subsection NG (Reference 2).

In the UCP evaluation, the new heat generation rates are applied at the appropriate steady-state and transient conditions at which high-temperature gradients exist within the UCP. Analyses are conducted for four conditions, including steady-state full-power, full-flow, steady-state 118-percent overpower, and the worst transient condition, loss of load, low Tavg. It is determined that the steady-state conditions will create maximum temperature gradients and concurrently maximum stress intensities within the UCP.

Input Data

The significant inputs to the analyses include the following:

1. Thermal properties of SA 240 Type 304 stainless steel from ASME B&PV Code, 1974 edition (Reference 3)

2. Heat generation rates

3. New thermal transients

4. Existing thermal transients from generic stress report WNEP-7602 (Reference 4)

5. NSSS design parameter conditions from Table 5.2.2-1.

Cases 1 through 4 were examined and Cases 3 and 4 resulted in the more limiting results.

Acceptance Criteria

The acceptance criterion for this calculation is the ASME B&PV Code, Section III, Division 1, 1974 edition (References 2 and 3). This document provided the UCP material properties used in the analysis and was the criteria for the primary-plus-secondary stress and fatigue usage limits.

Results

The UCP was evaluated to the intent of the ASME Code, subsection NG (Reference 2). Cases 3 and 4 were shown to be the limiting cases and are presented.

The stresses determined at steady-state 100-percent power conservatively envelope all normal and upset transients, except overpower. Therefore, the power range of 100 percent is applied to all unit loading and unloading transients (13,200 cycles). The long-term heating rates apply to these conditions. The minimum stress case for all unit loading and unloading transients is approximately a zero-stress state, associated with the plant in a cold shutdown or hot standby (0-percent power) condition. Moreover, no other conditions occur which can cause a significant stress reversal at the maximum stress locations. Therefore, the zero-stress state is the minimum condition and steady-state 100-percent power stress state is the maximum condition.

The short-term heating rates are determined based on nuclear transient conditions. The short-term heat generation rates are based on Condition II transients. These conditions are generally noted in a transient document as upset or in the Code as Level B. In the current evaluation, the short-term heating rates will be applied at two types of conditions, the applicable upset transients and the overpower events. For the upset transients, only those that could potentially impact core reactivity at power need to be considered including reactor trip, loss of flow, loss of load, and loss of power.

An analysis on the UCP for the emergency and faulted conditions was not required and the results from the existing stress report (Reference 1) are still applicable. This is because a change in the thermal loading will not affect a stress analysis for emergency and faulted conditions. Moreover, the LOCA loads are most limiting for faulted conditions, and the existing LOCA evaluation for a 1 ft^2 LOCA break, bounds the analysis with leak-before-break LOCA. From Reference 1, the mechanical stresses for faulted conditions bound emergency conditions. Test conditions do not need to be evaluated separately because they are already included in the normal conditions.

A summary of applicable combinations of critical associated loads is listed in Table 5.2.2-6. The critical load parameter is the heat generation rate applied to the transients. Another critical load parameter is the power range. The power is the percent power associated with the condition.

Note that the list of cycles for the transients is based on a 40-year life of the plant. Therefore, the use of these amounts of cycles is conservative for the RSG Program evaluation.

The seismic cycles are evaluated as 20 events, 20 cycles each, which equates to 400 cycles. These events will be combined with the conditions that cause the most severe stresses in the UCP. Therefore, 40 of these cycles can be combined with the overpower thermal events, while the remaining 360 cycles will be combined with 100-percent power Level 2 (short-term) events to be conservative.

The normal plant heatup and cooldown transient (200 cycles) is not included in the fatigue evaluation because it causes an insignificant stress in the LCP compared to the steady-state 100-percent power condition.

Secondary Stress Calculations

The thermal stresses for each of the following transient conditions were calculated:

1. Unit loading/unloading 0–100-percent power, short-term heat rates
2. Unit loading/unloading 0–100-percent power, long-term heat rates
3. Overpower events, short-term heat rates

Stress Evaluations According to the ASME Code

The allowable stress is $3 \cdot S_m$. Conservatively, S_m was taken at 650°F. This criterion will be applied to the most limiting thermal stress case, overpower events, short-term heat rates. This value was combined with the primary stresses due to OBE + mechanical loads for the evaluation.

The temperature difference between the core exit fluid and baffle-barrel exit fluid for operating Cases 3 and 4 was determined to be 43.8°F, while the same temperature difference for Cases 1 and 2 was calculated to be 42.5°F. From these temperature differences, a ratio is determined to be used to conservatively evaluate the stresses for Cases 3 and 4 from the stresses determined for Cases 1 and 2.

Applying a ratio to the peak stresses yields the adjusted peak stresses in Table 5.2-7.

The resulting margin of safety is acceptable according to the ASME Code as indicated in Table 5.2.2-8. Cases 1 and 2, which had stress values less than the overpower condition even with the ratio being accounted for, are met as well. Although the margin of safety is small, it should be noted that multiple conservative assumptions were used in the calculation.

Fatigue Evaluation According to the ASME Code

The fatigue analysis involves determining the alternating stress (S_a) from the peak stress (F), per NG-3222.4(e), and obtaining an allowable number of stress cycles from the design fatigue curve I-9.2 of Reference 3. From NG-3222.4(e)(4), the alternating stress must be corrected for the effects of the elastic modulus. This is done by multiplying half the peak stress by the ratio of the modulus used in the model in Figure I-9.2 (E_c=26.0e6 psi) to the modulus used in the analysis at 650°F ($E_{@650}$=25.1e6 psi). Note that a stress concentration factor is already considered in the model, and therefore an adjustment to the peak stress is not needed.

The peak primary stress intensity from OBE + Mechanical loads was determined. The resultant alternating stress, S_a, was calculated.

The peak secondary stresses have been determined at 100-percent power. The peak stress for 118-percent overpower condition is calculated by scaling up the 100-percent power short-term peak stress by 18-percent. A summary of the peak and alternating stresses are shown in Table 5.2.2-10.

To be conservative, the OBE + mechanical load cycles were combined with the most severe secondary stress conditions until the 400 cycles are exhausted. Therefore, 40 of the 400 cycles will be combined with the overpower events, short-term condition, while the remaining 360 cycles will combine with 360 cycles from the unit loading/unloading 0-100-percent power, short-term condition. The resulting alternating stress values will be used to determine the allowable cycles from the design fatigue curve, Figure I-9.2 of Reference 3. Because of the available margin, the cycles will be conservatively estimated from the curve. The incremental usage factor for each combination is combined based on the cumulative damage criteria and summarized in Table 5.2.2-9. The UCP fatigue usage factor is below the fatigue usage limit of 1.0.

Conclusions

Based on the results presented in Table 5.2.2-8, it can be concluded that the stress and fatigue values for the UCP are within the ASME Code requirements for the Callaway RSG Program.

5.2.2.6.3 Structural Evaluation of the Other Reactor Internal Components

Introduction

This analysis evaluates the effects of the RSG Program on the other Callaway reactor internals. The effects of parameters that are changing from the current operation for the RSG Program were analyzed. This includes new NSSS design transients as well as a new mechanical design flow. For the new RSG NSSS design transients, thermal and stress/fatigue analyses need to be determined. In order to eliminate the need to re-analyze the reactor internals thermally and structurally, the new Callaway RSG NSSS design transients were compared with design transients for a similar plant. In this case, the Callaway RSG design transients were compared to the design transients for the Wolf Creek Rerating Program. This is acceptable because the two plants have the same designs and operating conditions for each program are the same, with the exception mechanical design flow.

The mechanical design flow increased from 104,200 gpm/loop to 109,200 gpm/loop for the Callaway RSG Program. The effects of the new mechanical design flow was evaluated for 2 of the more limiting internal components, that is, the guide tubes and upper support columns.

Input Data

The new RSG NSSS design transients and the NSSS design operating parameters for the Callaway RSG Program were used as the basis for the evaluation. The applicable Wolf Creek Rerating design transients and the Systems Standard Design Criteria (SSDC) 1.3F design transients (Reference 6) were also used.

Acceptance Criteria

The acceptance criterion for the core support structures of the Callaway reactor internals is Section III of the ASME B&PV Code, 1974 Edition (Reference 2).

Method of Analysis

The SSDC 1.3F document (Reference 6) provides NSSS design transients that are applicable to both Callaway and Wolf Creek under their original operating conditions. These original NSSS design transients for Wolf Creek were then revised for a Rerating Program. Design transient curves for the Rerating Program that were not revised for the program can be found in the SSDC 1.3F design specification. The revised Wolf Creek design transients and the SSDC 1.3F design transients are used in this evaluation to compare with the new Callaway RSG NSSS design transients. If the design transients for the Wolf Creek Rerating Program, along with the applicable SSDC 1.3F design transients, envelope those for the Callaway RSG then the thermal and stress/fatigue analyses for the Wolf Creek Rerating are applicable to the Callaway RSG Program. In other words, the evaluation of the effects of the transients on the reactor internals for the Wolf Creek Rerating can be applied to the Callaway RSG Program. The comparison of the Callaway RSG Program and the Wolf Creek Rerating Program is acceptable because the plants have same reactor internals design and the NSSS design operating conditions are the same, with the exception of mechanical design flow.

The mechanical design flow for the Callaway RSG Program was increased from 104,200 gpm/loop to 109,200 gpm/loop. As a result, the hydraulic loads on the guide tubes and upper support columns will increase.

Results

The design transients provided for the Callaway RSG Program were compared to the existing design transients for the Wolf Creek Rerating Program and SSDC 1.3F. The comparison of the Callaway RSG Program to the Wolf Creek Rerating is acceptable because the plants have identical reactor internal designs and NSSS design operating conditions. It was concluded that all of the Callaway RSG design transients are consistent with the revised Wolf Creek Rerating design transients. Therefore, the thermal and stress/fatigue analyses on the reactor internals that were completed for the Wolf Creek Rerating are applicable to the Callaway RSG Program. The acceptance criteria of the ASME Code continue to be met for the Callaway RSG Program

Based on a new mechanical design flow for the Callaway RSG Program, the flow loads on the guide tubes and the upper support columns were determined. The maximum flow load on a guide tube is within the design load of 1,400 lb. The maximum flow load on an upper support column is under the acceptable load of 712 lb. Both of these loads are measured at mechanical design flow.

5.2.2.7 References

1. Westinghouse WNEP-7702 (Proprietary), "Generic Stress Report of 4 Loop Standard Reactor Core Support Structures – Structural and Fatigue Analysis," June 1977.

2. ASME Boiler and Pressure Vessel Code, Section III, Division 1, Subsection NG, 1974 Edition.

3. ASME Boiler and Pressure Vessel Code, Section III, Division 1, Appendix I, "Nuclear Power Plant Components," 1974 Edition.

4. Westinghouse WNEP-7602 (Proprietary), "Generic Stress Report of 4 Loop Standard Reactor Core Support Structures – Thermal Analysis," June 1977.

5. Westinghouse WCAP-9620 (Proprietary), "Reactor Internals Heat Generation Rates and Neutron Fluences," October 1979.

6. Westinghouse Rev. 0, "Systems Standard 1.3F Nuclear Steam Supply System Reactor Coolant System Design Transients for Standard Plants with Model F Steam Generators," March 1978.

Table 5.2.2-1 NSSS Design Parameters for Callaway Replacement Steam Generator Program				
Thermal Design Parameter	**Case 1**	**Case 2**	**Case 3**	**Case 4**
Reactor Power, MWt	3,565	3,565	3,565	3,565
Thermal Design Flow, Loop gpm	93,600	93,600	93,600	93,600
Reactor Coolant Pressure, psia	2,250	2,250	2,250	2,250
Core Bypass, %	8.6	8.6	8.6	8.6
Vessel/Core Inlet Temperature, °F	556.8	556.8	538.2	538.2
Steam Generator Tube Plugging, %	0	5	0	5
Mechanical Design Flow, gpm	109,200	109,200	109,200	109,200

Table 5.2.2-2 Summary of Applicable Thermal Design Transients			
Case	**Description**	**Cycles**	**Power (%)**
1.	Unit Loading/Unloading 0–100-Percent Power, Short-Term Heat Generation Rates	600	100
2.	Unit Loading/Unloading 0–100-Percent Power, Long-Term Heat Generation Rates	13,200	100
3.	Overpower Events – 118-Percent Power, Short-Term Rates	40	118

Table 5.2.2-3 Summary of Results for Lower Core Plate			
Category	**Max Stress Value**	**Allowable Stress Value**	**Margin of Safety**
$P_m + P_b + Q$	[ ]^{a,c}	48.6	[ ]^{a,c}
Fatigue	–	–	U_{cum} = [ ]^{a,c}
Bracketed [ ]^{a,c} information designates data that is Westinghouse proprietary, as discussed in Section 1.6 of this report.			

Table 5.2.2-4 Summary of Peak and Alternating Stress Values (ksi) in Lower Core Plate				
Stress Type	Condition	Peak Stress	Alternating Stress	No. of Cycles
Primary	OBE + Mechanical Load	[ ]^{a,c}	[ ]^{a,c}	400
Secondary	Unit Loading/Unloading 0–100-Percent Power, Short Term	[ ]^{a,c}	[ ]^{a,c}	600
Secondary	Unit Loading/Unloading 0–100-Percent Power, Long Term	[ ]^{a,c}	[ ]^{a,c}	13,200
Secondary	Overpower Events, Short Term	[ ]^{a,c}	[ ]^{a,c}	40
Bracketed [ ]^{a,c} information designates data that is Westinghouse proprietary, as discussed in Section 1.6 of this report.				

Table 5.2.2-5 Summary of Usage Factors for Lower Core Plate				
Condition	Alternating Stress (ksi)	Cycles	Allowable Cycles	Usage Factor
Overpower Events, Short Term + OBE + Mechanical Load	[ ]^{a,c}	40	7,000	[ ]^{a,c}
Unit Loading/Unloading 0–100-Percent, Short Term + OBE + Mechanical Load	[ ]^{a,c}	360	13,000	[ ]^{a,c}
Unit Loading/Unloading 0–100-Percent, Short Term	[ ]^{a,c}	240	30,000	[ ]^{a,c}
Unit Loading/Unloading 0–100-Percent, Long Term	[ ]^{a,c}	13,200	400,000	[ ]^{a,c}
			Cumulative Usage Factor =	[ ]^{a,c}
Bracketed [ ]^{a,c} information designates data that is Westinghouse proprietary, as discussed in Section 1.6 of this report.				

Table 5.2.2-6 Summary of Applicable Thermal Design Transients			
Case	Description	Cycles	Power Range (%)
1.	Unit Loading/Unloading 0–100-Percent Power, Short-Term Heat Generation Rates	600	100
2.	Unit Loading/Unloading 0–100-Percent Power, Long-Term Heat Generation Rates	13,200	100
3.	Overpower Events – 118-Percent Power, Short-Term Rates	40	118

Table 5.2.2-7 Cases 3 & 4 - Summary of Peak Stresses (ksi) in Upper Core Plate			
Location	Power	Maximum Peak Stress	
		Long Term	Short Term
Notch Fillet (Notch Submodel)	100%	[ ]^{a,c}	[ ]^{a,c}
Notch Fillet	118%	[ ]^{a,c}	[ ]^{a,c}
Bracketed [ ]^{a,c} information designates data that is Westinghouse proprietary, as discussed in Section 1.6 of this report.			

Table 5.2.2-8 Summary of Results for Upper Core Plate			
Category	Max Stress Value	Allowable Stress Value	Margin of Safety
$P_m + P_b + Q$	[ ]^{a,c}	48.6 ksi	[ ]^{a,c}
Fatigue	-	-	U_{cum} = [ ]^{a,c}
Bracketed [ ]^{a,c} information designates data that is Westinghouse proprietary, as discussed in Section 1.6 of this report.			

Table 5.2.2-9 Summary of Adjusted Usage Factors for Upper Core Plate				
Condition	Alternating Stress (ksi)	Cycles	Allowable Cycles	Usage Factor
Overpower Events, Short Term + OBE + Mechanical Load	[ ]^{a,c}	40	15,500	[ ]^{a,c}
Unit Loading/Unloading 0-100%, Short Term + OBE + Mechanical Load	[ ]^{a,c}	360	30,000	[ ]^{a,c}
Unit Loading/Unloading 0-100%, Short Term	[ ]^{a,c}	240	70,000	[ ]^{a,c}
Unit Loading/Unloading 0-100%, Long Term	[ ]^{a,c}	13,200	80,000	[ ]^{a,c}
			Cumulative Usage Factor =	[ ]^{a,c}
Bracketed [ ]^{a,c} information designates data that is Westinghouse proprietary, as discussed in Section 1.6 of this report.				

Table 5.2.2-10 Summary of Adjusted Peak and Alternating Stress Values (ksi) in Upper Core Plate				
Stress Type	**Condition**	**Peak Stress**	**Alternating Stress**	**# of Cycles**
Primary	OBE + Mechanical Load	[ ]^{a,c}	[ ]^{a,c}	400
Secondary	Unit Loading/Unloading 0-100% Power, Short Term	[ ]^{a,c}	[ ]^{a,c}	600
Secondary	Unit Loading/Unloading 0-100% Power, Long Term	[ ]^{a,c}	[ ]^{a,c}	13,200
Secondary	Overpower Events, Short Term	[ ]^{a,c}	[ ]^{a,c}	40
Bracketed [ ]^{a,c} information designates data that is Westinghouse proprietary, as discussed in Section 1.6 of this report.				

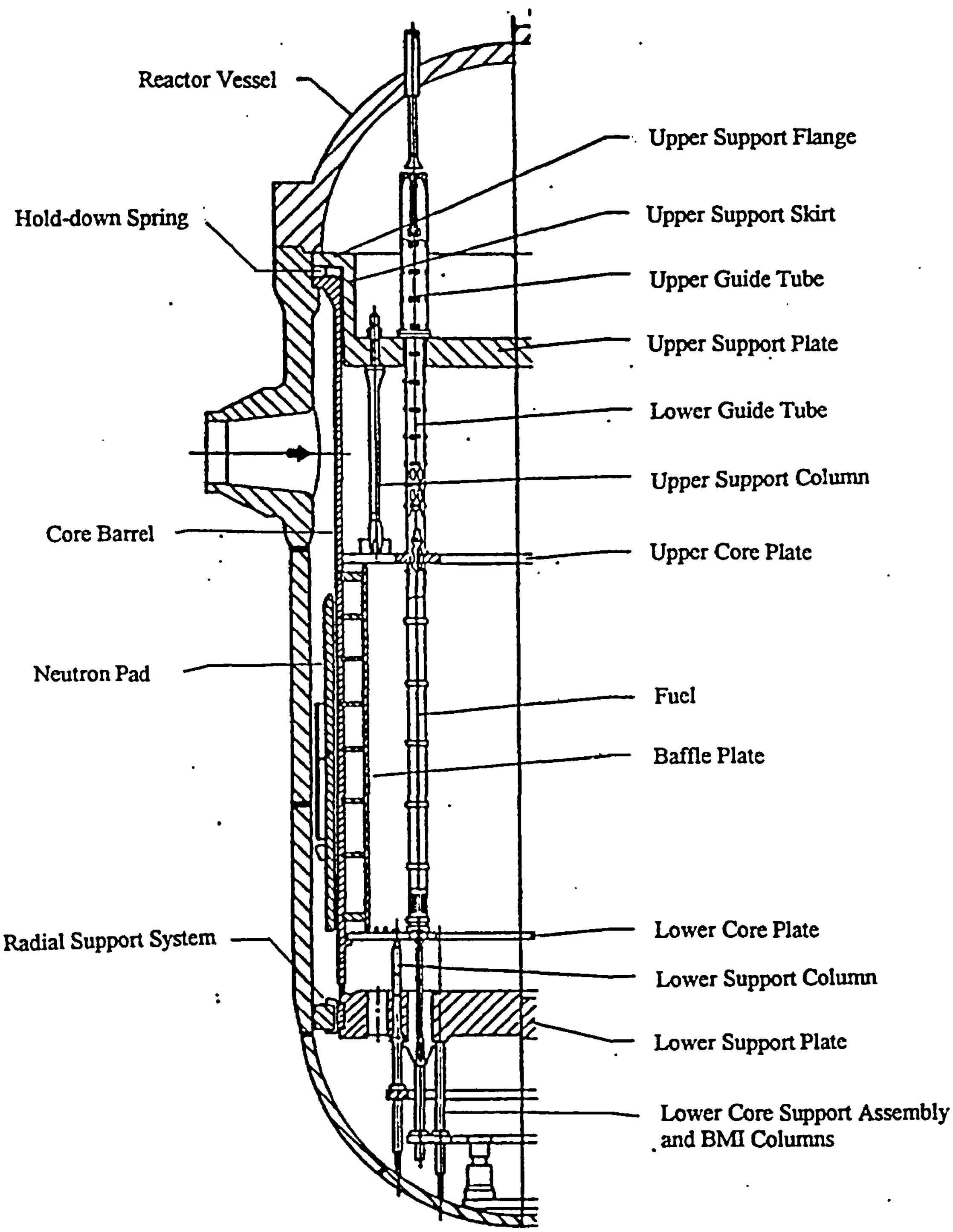

Figure 5.2.2-1 Callaway Reactor Vessel Internal Components

5.3 FUEL ASSEMBLIES

5.3.1 Introduction and Background

Fuel assemblies are designed to perform satisfactorily throughout their lifetime. The combined effects of the design-basis loads are considered in evaluating the capability of the fuel assemblies to maintain structural integrity.

As part of the Callaway RSG Program, the 17x17 VANTAGE 5 with intermediate flow mixers (IFMs) fuel assembly has been analyzed for LOCA and seismic loading conditions. The time histories representing the earthquake motions and the pipe rupture transient were obtained from the reactor vessel and internals system model. In addition, the top nozzle hold-down spring forces were also evaluated.

5.3.2 Input Parameters and Assumptions

The NSSS design parameters shown in Table 2-1 and the fuel assembly design and loss coefficients were used as the basis for the evaluation. The evaluation of the 17x17 VANTAGE 5 fuel assembly was performed in accordance with the NRC requirements as given in Standard Review Plan (SRP) 4.2, Appendix A.

5.3.3 Description of Analyses and Evaluations

5.3.3.1 Seismic and LOCA Evaluations

The maximum grid loads and assembly deflections for LOCA and seismic conditions were determined for the Callaway RSG conditions. The seismic inputs were based on the OBE and SSE. The LOCA inputs were based on pipe breaks of the accumulator line (ACC), pressurizer surge line (PZR), and residual heat removal line (RHR). The 17x17 VANTAGE 5 fuel assembly and reactor core models for seismic and LOCA are the same.

5.3.3.2 Spring Hold-Down Force Evaluation

There are four sets of top nozzle hold-down springs for each fuel assembly. Each set of hold-down springs consists of three leaves fabricated to form a cantilever leaf spring set. The fixed end of the spring set is held in place. The top nozzle springs are designed to retract within the top nozzle enclosure.

The lift forces for Callaway for the RSG Program have been evaluated for the impact on the fuel assembly hold-down spring capability. The spring evaluation was performed considering a last pump startup (LPS) temperature of 140°F.

5.3.4 Acceptance Criteria and Results

The results of the evaluation must support the following:

- The fuel assembly must be able to maintain structural integrity (coolable geometry) under combined LOCA and SSE loading conditions.

- The fuel assembly hold-down springs must be able to maintain margin to fuel assembly lift-off at the RSG conditions.

5.3.4.1 Lateral Impact Results

The grid impact forces resulting from OBE conditions are well below the allowable grid strengths for 17x17 VANTAGE 5 (w/IFMs) homogeneous core. The results of the seismic OBE analysis indicate that the maximum impact forces show adequate grid load margin and that the core coolable geometry and control rod insertion requirements are met for OBE load conditions.

The grid loads resulting from the LOCA and seismic events and the combined SSE and a limiting LOCA loading condition (accumulator line break) by using the square root of the sum of the squares (SRSS) method are well below the allowable grid strengths for homogeneous core. The evaluation of the 17x17 VANTAGE 5 fuel assembly in accordance with NRC requirements as given in SRP 4.2, Appendix A, shows that the 17x17 VANTAGE 5 fuel is structurally acceptable for the Callaway RSG. The results of the seismic and LOCA analyses indicate that the maximum impact forces show adequate grid load margin and that the core coolable geometry and control rod insertion requirements are met.

5.3.4.2 Fuel Assembly Stress Results

The maximum fuel assembly deflections and vertical impact forces were used to assess the fuel assembly structural integrity. The fuel assembly stress analyses were considered for OBE condition and the combined SSE and LOCA conditions.

The results of the OBE condition indicate that all stresses are below the allowable limits for the thimble tube and fuel rod.

For the combined SSE and LOCA conditions, the stress analysis of 17x17 V5 with IFMs for a representative plant bounds the 17x17 VANTAGE 5 of the Callaway RSG condition. The analysis results indicate adequate margin for both fuel rod and thimble tube. The fragmentation of the 17x17 VANTAGE 5 fuel rods and thimble tubes will not occur.

5.3.4.3 Spring Hold-Down Force Results

Both the current reconstitutable top nozzle with standard springs and the Westinghouse integral nozzle (WIN) top nozzle with WIN hold-down springs were evaluated. The most limiting results are with the standard hold-down springs. The functional requirement of no fuel assembly lift-off is still met.

5.3.5 Conclusions

Based on the results presented above, the 17x17 VANTAGE 5 w/IFMs fuel assembly design was determined to be structurally acceptable for the Callaway RSG Program.

The top nozzle hold-down force analysis results show that the hold-down spring force requirements are satisfied for both transition and homogenous cores and the design basis evaluation is still bounding. This is valid with both the reconstitutable top nozzle and the WIN top nozzle.

5.4 CONTROL ROD DRIVE MECHANISMS AND CAPPED LATCH HOUSINGS

5.4.1 Introduction and Background

This section addresses the ASME Code structural considerations for the pressure boundary components of the full length Model L-106A1 CRDMs and capped latch housings (CLHs). The CRDMs and CLHs were evaluated for the Callaway RSG Program parameters and the associated NSSS design transients.

This evaluation provides verification of continued structural suitability of the pressure boundary components of the existing CRDMs and CLHs for the RSG Program.

5.4.2 Input Parameters and Assumptions

The Callaway Model L-106A1 CRDMs and CLHs were originally designed and analyzed to the equipment specifications, Reference 1, and the ASME Code. The ASME Code Editions for CRDMs and CLHs are shown in Table 5.4-1.

The input parameters that were used to perform the analyses and evaluations for the RSG include the original NSSS design parameters and NSSS design transients, the RSG parameters (Section 2.0) along with the RSG NSSS design transients (Section 3.1), and the current design-basis evaluations for the CRDMs and CLHs. No additional input was required for the CRDM and CLH analyses and evaluations for the RSG.

There were no other changes considered to the pressure or thermal design parameters for the RSG. Seismic analyses and non-pressure boundary component evaluations are unaffected.

5.4.3 Description of Analyses and Evaluations

The ASME Code structural and fatigue limits and criteria of the generic CRDM and CLH reports are used to define the basis of the adequacy in the current evaluations of Callaway. The ASME Code year and addenda for each of the reports and the RCS normal operating temperature considered therein are shown in Table 5.4-1.

The CRDMs and CLHs are installed in the reactor vessel upper head (hot head CRDMs) and are affected by the reactor coolant pressure, vessel outlet temperature, and the hot leg NSSS design transients. The reactor coolant pressure is the same for both the RSG and for the current basis, which did not change from the original analysis. Since the reactor coolant pressure remains the same as originally specified, the CRDMs and CLHs remain bounded for the reactor coolant pressure condition for the RSG.

The highest vessel outlet temperature for the RSG is 620°F. Since most of the previous analyses used material allowables based on the design temperature of 650°F, the revised temperatures defined for the RSG are, in most cases, enveloped by the previous analyses. The only evaluations that were not enveloped by prior work are the evaluations for the Loss of Flow Transient. These evaluations were addressed for the new transients by multiplying the existing stresses by a ratio of the new transients to the old transients. This approach is conservative; rigorous analysis would provide even better margins of safety. After this was performed, it was shown that all cases were within the allowable values.

5.4.3.1 Control Rod Drive Mechanism Evaluation

The Callaway CRDMs are designed to the requirements of Reference 1. The generic analysis of the Type 106A1 CRDMs is given in the Westinghouse Electro-Mechanical Division Engineering Memorandum 5303, Rev. 0 (Reference 2), which contains (1) the stress analysis and Code comparison, (2) the fatigue analysis, (3) the seismic analysis, and (4) the heat transfer analysis. The main body of Reference 2 performs the calculations and evaluations to Section III of the ASME Code, the 1974 Edition with Addenda through the Winter of 1974.

For Callaway, the stress limits in Reference 2 are based on either the design temperature of 650°F or the local component temperature obtained in the analyses. Most of the stress limits are based on the design temperature of 650°F, which is unchanged by the RSG parameters. Where stress limits based upon local temperatures have been used, RSG component local temperatures will be determined and used as the basis for RSG allowable stresses.

5.4.3.2 NSSS Design Pressure Transients, Seismic, and COMS Evaluation

The COMS transient was evaluated for Callaway in Reference 3, and the definition of this transient has not changed for the RSG. Therefore, no additional analysis is necessary to re-evaluate the COMS transient for Callaway.

5.4.3.3 CLH Evaluation

The CLHs were originally fabricated in accordance with Reference 2. The generic stress report covering the CLHs is Reference 2, which was used to verify the pressure boundary structural integrity of the Callaway CLHs. The site-specific Pressure Boundary Summary Report (PBSR) for Callaway CLHs is Reference 4. The portion of Reference 4 that addresses the CLHs was developed from Reference 2.

5.4.4 Acceptance Criteria and Results

The acceptance criteria and results of the RSG analyses and evaluations for the CRDMs and CLHs with the new transients are summarized in Table 5.4-2.

As stated in Reference 2, Section 8.0, the detailed fatigue evaluation, as per the ASME Code, paragraph NB3222.4 (e), is not required. This is because all of the components of the joints conform to the waiver of fatigue requirements, ASME Code paragraph NB3222.4 (d).

5.4.5 Conclusions

The Callaway CRDMs and CLHs were evaluated for the RSG parameters and the associated NSSS design transients.

Based on the previous analysis and the analyses and evaluations performed for the RSG, the Callaway CRDM and CLH pressure boundary components are acceptable in accordance with the ASME Code for the RSG Program.

5.4.6 References

1. Westinghouse Equipment Specification 677470, Rev. 5, "General, Control Rod Drive Mechanism (CRDM), Capped Latch Housing (CLH) Assembly (Optional), Seismic Sleeve Assembly (Integrated Head Package (IHP) Plan Only), Model L-106A, L-106A-1, and L-106S, General 115, Reactor Coolant System," May 14, 1982.

2. Westinghouse Engineering Memorandum 5303, Rev. 0, "L106A-1 CRDM Generic Design Report Stress and Thermal Analysis," S. K. Ganguly, and J. R. Raymond, June 10, 1980.

3. Westinghouse Engineering Memorandum 6568, Rev. 0, "Models L-106A1 and L-106A CRDM/CLH Generic Pressure Boundary Summary Report, COMS Evaluation, S.O. 8L586," R. R. King, October 12, 1992.

4. Westinghouse Engineering Memorandum 5536, Rev. 1, "Union Electric Company, Callaway Nuclear Station No. 1, L106A-1 Full-Length Control Rod Drive Mechanism and Capped Latch Housing Assembly, Pressure Boundary Component Summary Report," S. K. Ganguly, August 4, 1982.

5. Westinghouse Engineering Memorandum 5303 Addendum 1, "L106A-1 CRDM Generic Design Report Stress and Thermal Analysis," S. K. Ganguly, October 28, 1985.

6. Westinghouse Engineering Memorandum 5303, Addendum 2, "L106A-1 CRDM Generic Design Report Stress and Thermal Analysis," J. D. Price, January 17, 1986.

Table 5.4-1 ASME Code Edition and RCS Normal Operating Temperatures of CRDM and CLH Component Evaluation Reports Compared to Replacement Steam Generator Reactor Coolant Temperature			
Report	**ASME Code Edition**	**Report RCS Normal Operating Evaluation Temperature**	**Highest Case Reactor Coolant Temperature for the RSG**
Generic Reports			
EM 5303, Rev. 0, Generic CRDM Analysis (Ref. 2)	Winter 1974	550°F (analysis of hot leg transients based on 650°F material allowable	620°F
EM 5303, Add. 1, Middle Canopy (Ref. 5)	Winter 1974	550°F (analysis of hot leg transients based on 650°F material allowable	620°F
EM 5303, Add. 2, Upper Canopy (Ref. 6)	Winter 1974	550°F (analysis of hot leg transients based on 650°F material allowable	620°F
EM 6568, Generic CRDM/CLH Analysis for COMS Transient (Ref. 3)	1983 Edition with Addenda through the 1986 Edition	560°F	620°F
Site-Specific Pressure Boundary Summary Reports			
EM 5536, CRDM/ CLH Analysis for Callaway (Ref. 4)	Winter 1974	550°F	620°F
EM 5536, Add. 1, Spare CRDM	Winter 1974	550°F	620°F
EM 5536, Add. 2, Design Documentation Change Notice 1	Winter 1974	550°F	620°F

Table 5.4-2 CRDM Evaluation Results

Component	ASME Code III Req.	Upper Joint		Middle Joint		Lower Joint		Capped Latch Housing	
		Calc	Allowed	Calc	Allowed	Calc	Allowed	Calc	Allowed
Cap	$P_m + P_b + Q$	[ ]^{a,c}	48,300					[ ]^{a,c}	48,300
	$\sigma_1 + \sigma_2 + \sigma_3$	[ ]^{a,c}	64,400					[ ]^{a,c}	64,400
Rod Travel Housing	$P_m + P_b + Q$	[ ]^{a,c}	48,300	[ ]^{a,c}	48,300				
	$\sigma_1 + \sigma_2 + \sigma_3$	[ ]^{a,c}	64,400	[ ]^{a,c}	64,400				
Latch Housing	$P_m + P_b + Q$			[ ]^{a,c}	48,300	[ ]^{a,c}	48,300	[ ]^{a,c}	48,300
	$\sigma_1 + \sigma_2 + \sigma_3$			[ ]^{a,c}	64,400	[ ]^{a,c}	64,400	[ ]^{a,c}	64,400
Head Adapter	$P_m + P_b + Q$					[ ]^{a,c}	48,300		
	$\sigma_1 + \sigma_2 + \sigma_3$					[ ]^{a,c}	64,400		
Canopy	$P_m + P_b + Q$	[ ]^{a,c}	48,300	[ ]^{a,c}	48,300	[ ]^{a,c}	48,300	[ ]^{a,c} (1)	48,300
	$\sigma_1 + \sigma_2 + \sigma_3$	[ ]^{a,c}	64,400	[ ]^{a,c}	80,000	[ ]^{a,c}	64,400	[ ]^{a,c}	64,400
Threaded Area	P_m	[ ]^{a,c}	9,660	[ ]^{a,c}	9,660	[ ]^{a,c}	9,660	[ ]^{a,c}	9,660
	Stress Intensity Due to Bell Mouth	[ ]^{a,c}	17,900	[ ]^{a,c}	17,900	[ ]^{a,c}	20,700	[ ]^{a,c}	17,900

1. This stress exceeds the allowable by 125 psi. Per page 67 of the original analysis, exceeding the allowable by less than 0.2 percent is considered insignificant in relation to other conservatism in the analysis. In particular, it is noted that the operating temperature at this location is 550°F. The Code allowable stress intensity at 550°F is 16.9 ksi opposed to 16.1 ksi at 650°F, resulting in a 3 S_m value of 50.7 ksi, which is higher than the calculated stress.

Bracketed [ ]^{a,c} information designates data that is Westinghouse proprietary, as discussed in Section 1.6 of this report.

5.5 REACTOR COOLANT LOOP PIPING, SUPPORTS, AND LEAK BEFORE BREAK

5.5.1 Reactor Coolant Loop Piping and Supports

5.5.1.1 Introduction

The reactor coolant loop (RCL) piping and support system includes a detailed model of the steam generator component. With the RSG, the structural design of the steam generator has changed by a significant enough amount that a re-analysis of the RCL piping and support system was required. This analysis was performed to qualify the RCL piping and primary equipment supports, and to re-demonstrate leak-before-break (LBB) for the piping. The following activities relative to the Callaway piping and support system were completed as part of this effort:

- An RCL piping stress analysis was performed for all load combinations and higher stressed locations. A re-analysis for thermal, seismic, and LOCA conditions was performed to show the design-basis requirements continue to be met.
- Primary equipment nozzle loads were determined to qualify all the primary equipment nozzles.
- Equipment support loads and displacements were reviewed for all primary component supports to demonstrate compliance with the design-basis allowable limits. The primary component supports extend from the primary component (steam generator, reactor vessel, RCP) to the architect-engineer (A/E) supplied building structure.
- Piping analyses of all Class 1 auxiliary lines attached to the RCL were completed to include the thermal, seismic, LOCA, main steam (MS), and feedwater (FW) pipe break, and fatigue effects from the RSG.
- The MS and FW piping systems were requalified with the RSG for the current thermal, seismic, and LOCA analyses.
- Redemonstration of LBB at all applicable locations. Westinghouse has previously demonstrated analytically that LBB can be applied to Callaway for the RCL, pressurizer surge line, accumulator line, and residual heat removal (RHR) line. At this time, LBB approval for each of these lines except the surge line has been granted by the NRC. A requalification was performed to demonstrate that LBB still applies with the RSG changes.

5.5.1.2 RCL Support Configuration

The Callaway NSSS consists of four RCLs attached to a common RPV. Each RCL includes the primary loop piping, one steam generator, and one RCP. The primary equipment support systems allow virtually unrestrained horizontal movement of the RCL during plant normal operation, but provide restraint for all upset, emergency, and faulted conditions. The primary equipment supports for the steam generators consist of pinned end columns and compression bumpers. The steam generator hydraulic snubbers were eliminated following the analysis performed by Westinghouse in 1994. The supports for the RCPs consist of pinned end columns and tension-only tie rods. The reactor vessel supports consist of a sliding shoe arrangement with side bumpers located under each nozzle.

The support configuration for the RCL with the RSG will remain unchanged from the existing analysis, except for the whip restraints. The hot leg and crossover leg whip restraints will be deactivated during RSG outage.

5.5.1.3 Program Approach

The existing design-basis RCL model was used previously in the RCL structural analysis discussed in References 1 and 2. The WECAN and WESTDYN models from References 1 and 2 were modified to include the RSGs.

The time-history seismic analyses and LOCA, main steam, feedwater break, thermal, and deadweight analyses were performed with the RSGs. The RCL hot leg and crossover leg whip restraints will be deactivated during the RSG outage. The cross-over leg pipe whip restraints are assumed to be inactive for Callaway during loop overtemperature transients and dynamic events such as earthquake and pipe rupture.

Using the results of deadweight, thermal, time-history seismic, LOCA, main steam, and feedwater break analyses, all loads are combined and reviewed with respect to the RSG design.

The results of the above analysis were then used to demonstrate that the effect of the RSG on the RCL piping, supports, and LBB is acceptable, with no other hardware modifications and without any significant impact on the remainder of the plant.

5.5.1.4 Stress Criteria

5.5.1.4.1 Piping Stress Criteria

The piping stress criteria used in the evaluation are based upon the type of operating condition. These stress criteria are the same as those used previously as the design basis in References 1 and 3. The definition of the type of operating condition depends upon the type of loading or combination of loading assumed to exist during that condition. The operating conditions and the stress criteria associated with each of them are summarized in Table 5.5-1.

5.5.1.4.2 Reactor Coolant Loop Support Criteria

The types of loadings considered for this plant consist of deadweight, thermal expansion, operating pressure, seismic events, LOCA, and pipe ruptures. The stress criteria used for evaluating the supports are based upon the type of operating condition. These stress criteria are the same as those used previously as the design basis in Reference 2.

The definition of the type of operating condition depends upon the type of loading or combination of loadings assumed to act during that condition. The 4 operating conditions (normal, upset, emergency and faulted) and the stress criteria associated with each are described below.

The normal condition considers loadings that exist during the normal operation of the plant. Normal loadings result from thermal expansion, operating pressure, and weight of the primary components, piping, and contained water. Stresses in all support system elements must not exceed allowable values contained in Subsection NF of the ASME Code (Reference 4) and the limits specified in Design Specification 955117, Revision 1 (Reference 5). For the reactor vessel supports, which are classified as plate- and shell-type supports in Subsection NF, the primary membrane stress intensities (P_m) are limited to 1.0 times the tabulated S_m value, and the primary membrane plus bending stress intensities ($P_m + P_b$) are maintained below 1.5 times the tabulated S_m value. Also, plate and shell-type supports must meet the additional requirements of Design Specification 955117, Revision 1 (Reference 5).

The upset condition assumes that the loads resulting from the OBE are added to the normal loadings of weight, operating pressure, and thermal expansion. As a result of these concurrent loadings, stresses in all support elements must not exceed the stress limits for the normal condition.

The emergency condition postulates infrequent incidents such as a small LOCA (less than 1 square inch) or small steam line break in addition to the normal loadings of weight, operating pressure, and thermal expansion. Stresses in linear-type members cannot exceed 1.3 times the allowable normal and upset stresses. The limit for primary membrane plus primary bending stress intensity for plate- and shell-type supports is 1.8 times tabulated S_m value. All emergency conditions have a negligible effect on primary equipment supports. They will not be evaluated.

Loadings due to weight, operating pressure, SSE, and pipe rupture are included in the plant faulted condition. Stresses from these loadings generally must not exceed the limits specified by Subsection NF and Appendix F of the ASME Code, Section III. Paragraph F-1370 of Appendix F (As amended by Regulatory Guide 1.124 - Reference 6) specifies that the normal condition allowable stresses for linear – type supports may be increased by the smaller factor of 2.0 or 1.167 $\frac{S_u}{S_y}$ if $S_u \geq 1.2S_y$ or 1.4 if $S_u < 1.2S_y$ where S_y represents material tensile strength. Both are taken as specified minimum values.

Member compressive axial loads are limited to 0.67 times the critical buckling load of the member or 0.9 times the critical buckling load of the member when justified by alternate analysis. Primary membrane stress intensities cannot exceed the greater of 1.2 S_y and 1.5 S_m or 0.7 S_u, whichever is smaller. Primary membrane plus bending stress intensities are limited to the greater of 1.8 S_y and 2.25 S_m, or 1.05 S_u, whichever is smaller.

Table 5.5-2 summarizes the loading combinations and stress limits for the primary equipment supports.

The results of the RCL analyses for the RSG are used to determine the loads acting on the equipment supports.

Two evaluation methods are used in the support evaluation:

- The "umbrella" load basis
- The design criteria basis

The umbrella load basis is used in cases when the loads from the RCL piping analysis for the RSG are bounded by the existing analysis loads. The umbrella loads are based on the previous plant analyses, documented in WCAP-9728, Volume 2 (Reference 2). Upon completion of the system analysis, conformance is demonstrated between the actual plant loads and the loads used in the existing analyses of the components.

Any deviations, where the actual loads are larger than the umbrella loads, are handled by individualized analysis using the design basis criteria for support qualification as summarized in Table 4-1 of WCAP-9728, Volume 2 (Reference 2).

To verify the structural integrity of the RCL support system, loads including thermal, weight, pressure, SSE, and pipe rupture are applied and stresses are compared to the previously calculated stresses as defined in Tables E-1 though E-5 of WCAP-9728, Volume 2 (Reference 2) or the allowable stresses summarized in Table 4-1 of Reference 2.

The primary equipment support components were evaluated for all loading conditions considering the RSG. The analytical models, loading criteria, and evaluation methods as described above were used in the support evaluation for the RSG Program. These components include the reactor vessel horizontal and vertical restraint; RCP columns and tie rods; and the steam generator columns, lower horizontal, and upper horizontal supports, including the compression bumpers.

5.5.1.5 Evaluation Results

The evaluation results of the RCL piping, primary components, and component supports are discussed in this section. The evaluations address the impacts due to the RSG change. The evaluation efforts summarized in this section are:

- Loop piping component evaluation (subsection 5.5.1.5.1)
- RCL support evaluation (subsection 5.5.1.5.2)
- RCL equipment nozzle load evaluation (subsection 5.5.1.5.3)
- Auxiliary line piping and supports evaluation (subsection 5.5.1.5.4)
- MS and FW piping evaluation (subsection 5.5.1.5.5)
- LBB redemonstration (subsection 5.5.1.5.6)

5.5.1.5.1 Loop Piping Components

The maximum design condition combined stress due to OBE, weight, and pressure in the RCL was less than the Code allowable stress value of 28.35 ksi. The maximum faulted condition combined stress in the RCL piping due to pressure, weight, SSE, and either a LOCA or a main steam system (MSS) or a FW line break was less than the code allowable stress value of 56.7 ksi. The maximum versus allowable stresses are shown in Tables 5.5-3 and 5.5-4. A complete re-analysis for the LOCA, MS break, and FW breaks was performed.

Based on this evaluation, the RCL piping components were qualified in accordance with the stress criteria described in subsection 5.5.1.4.1. Detailed qualification of all the RCL piping components was performed.

The fatigue evaluation of the Callaway RCL piping was performed considering the effects of design transient revisions for Tavg reduction, feedwater temperature reduction, and RSG. The RCL piping stresses and fatigue usage factors were found to be in conformance with the requirements of the ASME Code (Reference 7) for the fatigue evaluation performed under all normal, upset, and test conditions. Therefore, the piping system is acceptable.

Class 1 RCL branch nozzles for the RSG Program were evaluated. The RCL branch nozzle stresses and usage factors were found to be within the requirements of the Design Specification (Reference 3) and ASME Code (Reference 7). Therefore, the Class1 RCL branch nozzles are acceptable and will maintain their structural integrity.

5.5.1.5.2 Reactor Coolant Loop Supports

The results of the stress evaluation of the primary equipment supports for Callaway are summarized in this section. The manner in which the stresses are calculated is described, and maximum member stresses for each of the loading combinations are tabulated.

For the normal condition, the thermal, weight, and pressure forces (obtained from the RCL analysis) acting on the support structures are combined algebraically. The combined load component vector is used in the determination of the stress of each member in the support system.

The maximum stresses in members of the steam generator and the RCP supports, were calculated for each of the operating condition loadings. The stresses in all pins, compression collar, remaining snubber bodies, bolts, and connections that are a part of the steam generator and RCP supports are well within allowable values.

The reaction of the reactor vessel support structures to an applied force resulting from all loading conditions was analyzed. Dynamic forces applied to these structures are the combination of forces obtained from the RCL analysis, reactor cavity pressure, and the reactor vessel internals analysis. The maximum stress intensity and allowable stress intensities of the elements of the reactor vessel supports are given in Table 5.5-5.

The data of loadings applied to the building structures were documented and provided to AmerenUE.

5.5.1.5.3 RCL Equipment Nozzle Loads

The equipment nozzle loads were qualified. The actual deadweight, thermal, OBE, SSE, LOCA, main steam and feedwater break loads from the RCL piping analyses for the RSG Program were compared to the allowable nozzle loads provided in the equipment design specifications. Results of the primary equipment nozzle loads evaluation are found to be acceptable.

5.5.1.5.4 Auxiliary Line Piping and Supports

For the RSG, the structural design of the steam generator has changed and re-analyses of the Class 1 Auxiliary piping and support system, attached to the RCL piping, was required for the lines.

The effect of the previously implemented Class 1 piping snubber eliminations and the steam generator snubber elimination were incorporated into the Class 1 auxiliary lines piping analysis. In addition, the piping analyses included modifications done on the accumulator lines, excess letdown line loop 4 and RHR line loop 1.

The Class 1 primary stresses were evaluated in accordance with the Design specification (Reference 3) and the ASME Code requirements (Reference 7), and they were found to be acceptable. Refer to Tables 5.5-3 and 5.5-4.

Fatigue evaluation of the Class 1 auxiliary piping for the RSG Program was performed. The Class 1 auxiliary piping stresses and fatigue usage factors were found to be in conformance with the requirements of the ASME Code (Reference 8) for the fatigue evaluation performed under all normal, upset, and test conditions.

All equipment nozzles, containment penetration, flange nozzles and RCL branch nozzles, are found to be acceptable.

The Class 1 auxiliary piping and non-class 1 extension have been shown to be adequate and will maintain its structural integrity and meet the safety-related requirements under the specified conditions of the Design Specification (Reference 3).

Class 1 auxiliary line support loads were evaluated in accordance with the Bechtel Design Specification (Reference 9) and the applicable ASME Code (Reference 4) requirements and found to be acceptable.

5.5.1.5.5 Main Steam and Feedwater Piping

New piping analyses for the main steam and feedwater piping were performed for the RSG.

In the seismic analyses, multiple response spectra were used for the main steam lines and N411 damping spectra were used for the main feedwater lines.

The seismic analyses were performed with the new spectra and anchor motions for the RSG.

Thermal and LOCA analyses were reconciled with the new displacements from the RCL piping analysis for the RSG.

The stresses were evaluated and found to be acceptable. The results of the main steam and feedwater piping stress evaluation are summarized in Tables 5.5-6 and 5.5-7.

In addition, steam generator and containment penetration nozzles, valves, and support lugs for the RSG Program were found to be acceptable.

Piping support loads for the RSG were evaluated and found to be acceptable.

5.5.1.5.6 LBB Redemonstration

Subsection 5.5.2 provides a discussion of the LBB redemonstration with the RSG. The conclusion is that LBB for the RCL piping, pressurizer surge line, accumulator line, and RHR line was demonstrated to be acceptable with the RSGs.

5.5.1.6 Conclusions

Deadweight, thermal, time-history seismic OBE and SSE, LOCA, MSS line, and FW line break analyses were performed for Callaway to demonstrate the acceptability of the steam generator replacement. The results are summarized below:

- The new stresses in the RCL piping are less than the allowable stresses.
- The reactor coolant piping secondary stresses and fatigue usage factors are in conformance with the requirements of the Code for the fatigue damage evaluation performed under all normal, upset, and test conditions. Therefore, the piping system is adequate for all design transient conditions described in the design specification (Reference 3).
- The Class 1 RCL branch nozzles stresses and usage factors are within the allowable limits. Therefore, the Class1 RCL branch nozzles are acceptable and will maintain their structural integrity.
- The new RCP, reactor vessel, and RSG primary-side equipment nozzle loads are less than the allowable nozzle loads.
- Stresses in all the support members of the RPV, RCP, and steam generator are within the allowable values for normal, upset, and faulted conditions.
- The auxiliary lines piping and supports are shown to be acceptable. The piping system results remain within the ASME Code allowable.
- MS and FW piping and supports are shown to be acceptable. The analysis results for the MS and FW piping remain within the ASME Code allowable.
- LBB for the RCL piping, pressurizer surge line, accumulator line, and RHR line was demonstrated (see subsection 5.5.2 for details).

5.5.1.7 References

1. WCAP-9728-V1(Proprietary), Rev. 3, "Structural Analysis of the Reactor Coolant Loop for Standard Nuclear Unit Power Plant," August 1994.

2. WCAP-9728-V2 (Proprietary), Rev. 4, "Structural Analysis of the Reactor Coolant Loop for Standard Nuclear Unit Power Plant System," August 1994.

3. Piping Design Specification 955238, Rev. 2, "Piping Design Specification, ANS Safety Class 1, SAP/SCP-137," December 8, 1995.

4. Nuclear Power Plant Components, ASME Boiler and Pressure Vessel Code Section III, ASME, New York, the 1974 Edition with Addenda up to and including Winter 1974.

5. Westinghouse Equipment Specification 955117, Rev. 1.

6. U.S. Nuclear Regulatory Commission, Office of Standards Development, Regulatory Guides 1.124 and 1.130, Rev. 1.

7. Nuclear Power Plant Components, ASME Boiler and Pressure Vessel Code Section III, ASME, New York, the 1974 Edition ASME Code with Addenda up to and including Winter 1975.

8. Nuclear Power Plant Components, ASME Boiler and Pressure Vessel Code Section III, ASME, New York, the ASME Code, Summer 1979 Addenda (for stress analysis purposes).

9. Bechtel Design Specification 10466-M-217(Q), Revision 7.

Table 5.5-1 Piping Stress Criteria		
Condition/ Requirement	**Loading Combinations**	**Stress Limits**
Design	Design Pressure Weight + OBE	$\sigma^{(1)} < 1.5S_m^{(2)}$
Fatigue (Normal/ Upset/Test)	Normal transients Upset transients Test transients Weight + OBE	Equations 10, 11, 12, 13, and 14 of Code (Ref. 6)
Emergency	Emergency transient Pressure Weight	$\sigma^{(1)} < 2.25S_m^{(2)}$ $p_o$max<150 percent of design pressure
Faulted	Faulted transient Pressure Weight	$\sigma^{(1)} < 3.0S_m^{(2)}$ $P_o$max < 200 percent of design pressure
	Operating pressure Weight + SSE	$\sigma^{(1)} < 3.0S_m^{(2)}$
	Faulted transient Pressure associated with pipe rupture Weight + SSE + pipe rupture	$\sigma^{(1)} < 3.0S_m^{(2)}$ $P_o$max < 200 percent of design pressure

Notes:

σ = Total primary stress in the system

S_m = Allowable stress from the ASME Code (Reference 7)

Table 5.5-2 Support Structure Loading Combinations and Stress Criteria for Normal, Upset, and Faulted Conditions

Operating Condition	Loading Combination	Stress Limit	
		Linear-Type Supports	Plate and Shell Supports
Normal	Thermal expansion Weight Operating pressure	Within working limits	$P_m \leq 1.0\ S_m$ $P_m + P_b \leq 1.5\ S_m$
Upset	Thermal expansion Weight Operating pressure OBE Pipe support attachment	Within working limits	$P_m \leq 1.0\ S_m$ $P_m + Pb \leq 1.5\ S_m$
Faulted(3)	Operating pressure Weight LOCA SSE Jet impingement Pipe support attachment	1.167 Su/Sy if Su ≥ 1.2 Sy or 1.4 if Su < 1.2 Sy times working limits	$P_m \leq$ 1.5 S_m or 1.2 Sy (greater)(1) $P_m + P_b \leq$ 2.25 S_m or 1.8 Sy (greater)(2)

Notes:

(1) Not to exceed 0.7 S_u

(2) Not to exceed 1.05 S_u

(3) Faulted = Deadweight + Pressure ± $\sqrt{(SSE)^2 + (LOCA)^2}$

Table 5.5-3 Class 1 RCL Piping ASME Stress Actual versus Allowable—Equation 9[1, 2]				
	Eq. 9 Design Maximum (ksi)	Eq. 9 Design Allowable[2] (ksi)	Eq. 9[1] Faulted Maximum (ksi)	Eq. 9[1] Faulted Allowable[2] (ksi)
Hot Leg	[ ]^{a,c}	28.35	[ ]^{a,c}	56.7
Crossover Leg	[ ]^{a,c}	28.35	[ ]^{a,c}	56.7
Cold Leg	[ ]^{a,c}	28.35	[ ]^{a,c}	56.7

Notes:

(1) Equation 9 from ASME Code, Section 3, NB-3650.

(2) Allowable stress based on S_m = 18.9 ksi for SA351 CF8A at 650°F.

Bracketed [ ]^{a,c} information designates data that is Westinghouse proprietary, as discussed in Section 1.6 of this report.

Table 5.5-4 Class 1 RCL Piping ASME Stress Actual versus Allowable—Equations 12 and 13 and Fatigue Usage Factor[1, 2]					
	Eq. 12 Maximum (ksi)	Eq. 12 Allowable (ksi)	Eq. 13 Maximum (ksi)	Eq. 13 Allowable (ksi)	Fatigue Usage Factor
Hot Leg	[ ]^{a,c}	56.7	[ ]^{a,c}	56.7	[ ]^{a,c}
Crossover Leg	[ ]^{a,c}	56.7	[ ]^{a,c}	56.7	[ ]^{a,c}
Cold Leg	[ ]^{a,c}	56.7	[ ]^{a,c}	56.7	[ ]^{a,c}

Notes:

(1) Equations 12 and 13 from ASME Code, Section 3, NB-3650.

(2) Equation 12 stresses and fatigue usage factors, and Equation 13 stresses address the effects of T_{hot} reduction.

Bracketed [ ]^{a,c} information designates data that is Westinghouse proprietary, as discussed in Section 1.6 of this report.

Table 5.5-5 Reactor Vessel Support Structure Maximum and Allowable Stresses for Normal, Upset, and Faulted Conditions		
Support Component	Maximum Stress Intensity (ksi)	Allowable Stress Intensity (ksi)
Normal	P_m = [ ]^{a,c} $P_m + P_b$ = [ ]^{a,c}	S_m = 21.7 1.5 S_m = 32.55
Upset	P_m = [ ]^{a,c} $P_m + P_b$ = [ ]^{a,c}	S_m = 21.7 1.5 S_m = 32.55
Faulted	P_m = [ ]^{a,c} $P_m + P_b$ = [ ]^{a,c}	0.7 S_u = 45.5 1.05 S_u = 68.25

Bracketed [ ]^{a,c} information designates data that is Westinghouse proprietary, as discussed in Section 1.6 of this report.

Table 5.5-6 Main Feedwater Lines ASME Stress Actual versus Allowable for Equations 8, 9, 11 and Pipe Break Exclusion						
Load Combination	**Equation**	**FW Loop 1 Maximum Stress (ksi)**	**FW Loop 2 Maximum Stress (ksi)**	**FW Loop 3 Maximum Stress (ksi)**	**FW Loop 4 Maximum Stress (ksi)**	**Allowable Limits (ksi)**
P+DW	8	[ ]^{a,c}	[ ]^{a,c}	[ ]^{a,c}	[ ]^{a,c}	15.0
P+DW+OBE	9D	[ ]^{a,c}	[ ]^{a,c}	[ ]^{a,c}	[ ]^{a,c}	18.0
P+DW+SRSS (SSE, LOCA)	9F	[ ]^{a,c}	[ ]^{a,c}	[ ]^{a,c}	[ ]^{a,c}	36.0
P+DW+CVS	9F	[ ]^{a,c}	[ ]^{a,c}	[ ]^{a,c}	[ ]^{a,c}	36.0
P+DW+TH+SAM	11	[ ]^{a,c}	[ ]^{a,c}	[ ]^{a,c}	[ ]^{a,c}	37.5
SAM in M_b	Pipe Break Exclusion	[ ]^{a,c}	[ ]^{a,c}	[ ]^{a,c}	[ ]^{a,c}	32.4

CVS = Check valve slam
DW = Piping deadweight
LOCA = Loss-of-coolant accident (pipe rupture loads)
OBE = Operating basis earthquake
P = Pressure
SAM = Anchor displacement of OBE
SSE = Safe shutdown earthquake
TH = Thermal

Bracketed [ ]^{a,c} information designates data that is Westinghouse proprietary, as discussed in Section 1.6 of this report.

Table 5.5-7 Main Steam Lines ASME Stress Actual versus Allowable for Equations 8, 9, 11 and Pipe Break Exclusion						
Load Combination	**Equation**	**MS Loop 1 Maximum Stress (ksi)**	**MS Loop 2 Maximum Stress (ksi)**	**MS Loop 3 Maximum Stress (ksi)**	**MS Loop 4 Maximum Stress (ksi)**	**Allowable Limits (ksi)**
P+DW Operating	8	[ ]^{a,c}	[ ]^{a,c}	[ ]^{a,c}	[ ]^{a,c}	17.5
P+DW Test	9D	[ ]^{a,c}	[ ]^{a,c}	[ ]^{a,c}	[ ]^{a,c}	21.0
P+DW+OBE	9D	[ ]^{a,c}	[ ]^{a,c}	[ ]^{a,c}	[ ]^{a,c}	21.0
P+DW+FVC	9D	[ ]^{a,c}	[ ]^{a,c}	[ ]^{a,c}	[ ]^{a,c}	21.0
P+DW+SRSS (SSE, LOCA)	9F	[ ]^{a,c}	[ ]^{a,c}	[ ]^{a,c}	[ ]^{a,c}	42.0
P+DW+SRSS (SSE, FWB)	9F	[ ]^{a,c}	[ ]^{a,c}	[ ]^{a,c}	[ ]^{a,c}	42.0
P+DW+TH	11	[ ]^{a,c}	[ ]^{a,c}	[ ]^{a,c}	[ ]^{a,c}	43.8
SAM in M_b	Pipe Break Exclusion	[ ]^{a,c}	[ ]^{a,c}	[ ]^{a,c}	[ ]^{a,c}	37.8

DW = Piping deadweight
FVC = Fast valve closure
FWB = Feedwater break
LOCA = Loss-of-coolant accident (pipe rupture loads)
OBE = Operating basis earthquake
P = Pressure
SAM = Anchor displacement of OBE
SSE = Safe shutdown earthquake
TH = Thermal

Bracketed [ ]^{a,c} information designates data that is Westinghouse proprietary, as discussed in Section 1.6 of this report.

5.5.2 Leak-Before-Break Redemonstration

5.5.2.1 Reactor Coolant System Primary Loop Piping

5.5.2.1.1 Introduction

The original structural design basis of the Callaway Nuclear Power Plant required that the dynamic effects resulting from pipe breaks be considered and that protective measure for such breaks be incorporated into the design. Subsequent to the original Callaway design, an additional concern of asymmetric blowdown loads was raised as described in Unresolved Safety Issue A-2 (Asymmetric Blowdown Loads on the Reactor Coolant System) and Generic Letter 84-04 (Reference 1). However, research by the NRC and industry (coupled with operating experience) determined that safety could be negatively impacted by the placement of pipe whip restraints on certain systems. As a result, NRC and industry initiatives resulted in demonstrating that LBB criteria can be applied to RCS piping based on fracture mechanics technology and material toughness.

The Callaway plant primary loop piping analysis for the application of LBB was documented in Reference 2 and approved by the NRC (Reference 3). Reference 4 documented the LBB results for the Callaway plant primary loop piping after elimination of steam generator snubbers. In order to justify the elimination of RCS primary loop pipe breaks from the structural design basis, the piping analysis demonstrated that the following objectives were met:

- Demonstrate that adequate margin exists between the "critical" crack size and a postulated crack that yields a detectable leak rate
- Demonstrate that there is sufficient margin between the leakage through a postulated crack and the leak detection capability of the plant
- Demonstrate margin on applied load
- Demonstrate that fatigue crack growth is negligible

The flaw stability criteria performed for the analysis examined both global and local stability (as applicable) for a postulated through-wall circumferential flaw. The global analysis was carried out using the plastic instability method, based on traditional plastic limit load concepts, but accounting for strain hardening and the presence of a flaw. The local stability (as applicable) criteria were based on the J integral-tearing modulus (J-T) approach. The results of the analysis were documented in References 2 and 4 for the Callaway plant and satisfied the required objectives.

To support the RSGs at Callaway, the current LBB analysis was updated to address the RSG conditions. The RSG evaluation and results are described in the following subsections.

5.5.2.1.2 Input Parameters and Assumptions

The parameters important in the evaluation were the geometry, loadings, piping forces, moments, normal operating temperature, and normal operating pressure. The normal operating temperature and normal operating pressure for the RSG are presented in Table 2-1.

5.5.2.1.3 Description of Analyses and Evaluations

The recommendations and criteria proposed in Reference 5 were used in this evaluation. The primary loop piping deadweight, thermal (normal 100-percent power expansion), SSE, and pressure loads due to the RSG Program were used. The normal operating temperature and pressure at the RSG conditions were also used in the evaluation. The evaluation showed that all the LBB recommended margins were satisfied for the RSG condition. The margins from Reference 5 are also described in the following subsection.

5.5.2.1.4 Acceptance Criteria and Results

The LBB acceptance criteria are based on SRP Section 3.6.3 (Reference 5). The recommended margins are as follows:

- Margin of 10 on leak rate
- Margin of 2.0 on flaw size
- Margin on loads (using faulted load combinations by the absolute summation method

The evaluation results showed the following at all the critical locations:

- Leak rate – A margin of 10 exists between the calculated leak rate from the leakage flaw and the leak detection capability of 1 gpm.

- Flaw size – A margin of 2.0 or more exists between the critical flaw and the flaw having a leak rate of 10 gpm (the leakage flaw).

- Loads – A margin on loads exists.

The evaluation results show that the LBB conclusions provided in References 2 and 4 for the Callaway plant remain unchanged for the RSG conditions.

5.5.2.1.5 Conclusions

The LBB acceptance criteria are satisfied for the primary loop piping at the RSG conditions. All the recommended margins are satisfied. Therefore, it is concluded that the dynamic effects of RCS primary loop pipe breaks need not be considered in the structural design basis of the Callaway plant at the RSG conditions.

5.5.2.2 Pressurizer Surge Line, Accumulator Lines, and Residual Heat Removal Lines

5.5.2.2.1 Summary

The Callaway plant pressurizer surge line, accumulator lines, and RHR lines analyses for the application of LBB were documented in WCAP-15983-P (Reference 6), WCAP-16019-P (Reference 7), and WCAP-16020-P (Reference 8). Evaluations were performed to determine the impact of the loadings and other parameters on the LBB analyses due to the RSG conditions.

The results of the evaluations show that all the LBB acceptance criteria and recommended margins are satisfied at the RSG conditions.

The Callaway plant evaluation results showed the following at all the critical locations consistent with the previous LBB analyses for the pressurizer surge line, accumulator lines, and RHR lines:

- Leak rate – A margin of 10 exists between the calculated leak rate from the leakage flaw and the leak detection capability of 1 gpm.
- Flaw size – A margin of 2.0 or more exists between the critical flaw and the flaw having a leak rate of 10 gpm (the leakage flaw).
- Loads – A margin on loads exists.

5.5.2.2.2 Conclusions

It was determined that the conclusions of the previous LBB analyses shown in References 6 through 8 for Callaway plant for the pressurizer surge line, accumulator lines, and RHR lines remain valid. Therefore, it is concluded that the dynamic effects of pressurizer surge line, accumulator lines, and RHR lines pipe breaks need not be considered in the structural design basis of the Callaway plant at the RSG conditions.

5.5.2.3 References

1. USNRC Generic Letter 84-04, Subject "Safety Evaluation of Westinghouse Topical Reports Dealing with Elimination of Postulated Pipe Breaks in PWR Primary Main Loops," February 1, 1984.

2. WCAP-10691, "Technical Bases for Eliminating Large Primary Loop Pipe Rupture as a Structural Design Basis for Callaway and Wolf Creek Plants," October 1984.

3. NUREG-0881 Supplement No. 5, "Safety Evaluation Report Related to the Operation of Wolf Creek Generating Station, Unit No. 1, Docket No. STN 50-482, Kansas Gas and Electric Company, et al.," March 1985.

4. WCAP-14059, "Technical Justification for Eliminating Large Primary Loop Pipe Rupture as the Structural Design Basis for the Callaway and Wolf Creek Plants after Elimination of SG Snubbers," August 1994.

5. NRC Standard Review Plan: Public Comments Solicited; 3.6.3 Leak-Before-Break Evaluation Procedures; Federal Register, Vol. 52, No. 167, Friday, August 28, 1987, Notices, pp. 32626-32633.

6. WCAP-15983-P, "Technical Justification for Eliminating Pressurizer Surge Line Rupture as the Structural Design Basis for the Callaway Nuclear Power Plant," February 2003.

7. WCAP-16019-P, "Technical Justification for Eliminating 10" Accumulator Lines Rupture as the Structural Design Basis for the Callaway Nuclear Power Plant," February 2003.

8. WCAP-16020-P, "Technical Justification for Eliminating 12" Residual Heat Removal (RHR) Lines Rupture as the Structural Design Basis for Callaway Nuclear Power Plant," February 2003.

5.6 REACTOR COOLANT PUMPS AND MOTORS

Evaluations were performed to assess the impact of the Callaway RSG Program on the RCPs and motors.

5.6.1 Introduction and Background

Each Callaway reactor coolant loop contains a Model 93A-1 single-stage shaft-seal pump driven by a 7,000 HP (nameplate rating) motor. The pump is a vertical assembly consisting of (from top to bottom) the motor, the motor support stand, the seal assembly, and the hydraulic unit.

The RCPs and RCP motors were evaluated to determine the impact of the RSG Program conditions. For the RCPs, the intent is to demonstrate that the RCP structural integrity is not adversely impacted by the RSG Program NSSS design parameters, and that this pressure boundary component continues to comply with industry codes and standards. The Code of Record for the Callaway RCPs is the ASME B&PV Code, Section III, 1971 Edition with Addenda through the Summer 1973 Addenda (Reference 1).

For the RCP motors, the intent is to show that the motors still comply with the operability requirements given in the equipment specifications for the motors.

5.6.2 Input Parameters and Assumptions

The structural evaluation of the RCP pressure boundary uses as inputs the cold leg pressure and temperature from the NSSS design parameters, and the NSSS cold leg design transients.

The RCS best-estimate flows are used, along with the cold leg temperatures, for the evaluation of the RCP motors.

The RCP motor evaluation is also dependent on the specific pump impellers being evaluated. The evaluation performed is based on the use of the RCP impellers identified in Table 5.6-1.

5.6.3 Description of Analyses and Evaluations

5.6.3.1 Structural

The purpose of this evaluation is to demonstrate the continued applicability of the equipment specifications (References 2 and 3) and the plant-specific pressure boundary evaluations (References 4 through 9) in order to demonstrate compliance with applicable requirements of industry codes and standards. The critical components that are covered by this evaluation are the pump casing and feet, the bolting ring and bolts, the thermal barrier heat exchanger assembly, and the seal housing and bolts. These components are evaluated for both steady-state and transient conditions. Compliance with ASME Code requirements is used to demonstrate acceptability. The Code of Record for the Callaway RCPs is the ASME B&PV Code, Section III, 1971 Edition with Addenda through the Summer 1973 Addenda (Reference 1).

5.6.3.2 Electrical

The RCP motor performance is evaluated based on the performance characteristics of the currently installed impellers (Table 5.6-1). A hydraulic evaluation is performed considering the RSG Program flow rates and the reactor coolant temperatures, using the performance characteristics of the identified impellers to determine the power requirements for hot and cold reactor coolant conditions and the corresponding hydraulic axial thrusts and torque coefficients.

The outputs of the hydraulic evaluation are used to evaluate the capability of the RCP motor. The initial evaluation is to determine if the motor remains within its hot and cold nameplate ratings. If the motor is being forced to operate at conditions outside its nameplate ratings, then additional motor evaluations are performed. The stator temperature rise under hot and cold operating conditions is determined. A conservative all-heat-stored analysis is performed to determine the temperature rise of the rotor cage winding under worst-case pump starting conditions, which are with cold reactor coolant, reverse flow from the other RCPs running, and 80 percent of the normal voltage. The loads on the motor thrust bearings, considering the revised hydraulic thrusts from the hydraulic evaluation, are also evaluated for continued acceptability.

5.6.4 Acceptance Criteria and Results

5.6.4.1 Reactor Coolant Pump

The RCP is located in the cold leg, downstream of the steam generator outlet. The pump normal operating pressure per the equipment specification (Reference 2) is 2,250 psia and the normal operating temperature is 556.6°F at the pump inlet. The original RCP analysis (Reference 4) was based on an operating pressure of 2,250 psia and an inlet temperature of 558.6°F. The design parameters of Section 2.1 indicate that the RCS operating pressure is 2,250 psia and that the RCP temperature (reactor pressure vessel inlet) is between 538.2°F and 556.8°F. The temperature of 556.8°F at the reactor pressure vessel inlet corresponds to a temperature of 556.6°F at the pump inlet. The additional 0.2°F is due to heat input from the RCP. Both the temperature and the pressure for the RSG Program are the same or less than the equipment specification values and the values considered in the original analysis.

The NSSS design parameters applicable to the Callaway RSG Program are given in Section 2.1. The parameters of interest to the RCP evaluation are the vessel inlet temperature and the reactor coolant pressure. These are given in Table 5.6-2 for the current program, and from the original PBSR (Reference 4).

As Table 5.6-2 shows, the maximum vessel inlet temperature is slightly lower than the temperature originally considered in the PBSR. The reactor coolant pressure has not changed. Due to lower associated allowable design stress limits, higher temperatures are more limiting for the RCP structural design qualification. Therefore, the previous analyses are bounding and applicable to the RSG Program operating temperatures and pressure listed in Section 2.1.

Transient Evaluation

The RCP structural evaluation also considers the NSSS design transients that are defined for the RSG Program. Transients that have been considered in previous analyses that are applicable to the Callaway RCPs need not be considered further. Only those transients that are changing for the RSG Program need to be evaluated for their effect on the qualification of the RCPs.

Since only the Tcold and RCS pressure transients affect the RCPs, only the transient variations of those parameters need to be considered when comparing the transients applicable to the RSG Program to those previously evaluated. When this comparison is performed, the only NSSS design transients for the RSG Program that are different from those previously considered are the large step decrease with steam dump, the COMS operation, and the small LOCA transients.

The small LOCA transient is an emergency condition transient. The change to this transient consists of the addition of a figure giving Tcold temperature variation for the low Tavg operating condition. The figure is similar to the figure specified in the design specifications in shape, time scale, and final temperature, but has a larger ΔT because the starting temperature of the transient is lower. Since the ASME Code (NB-3224) does not require evaluation of secondary or peak stresses or fatigue for emergency conditions, this change in the small LOCA transient has no effect on the evaluation of the RCP.

For the original Callaway analysis (Reference 4) for the normal, upset, and testing conditions, the transients were grouped and bounding transient conditions were considered for each group. For the transient group that included the large step load decrease transient, a bounding temperature transient consisting of a step change of 25°F was considered. The revised large step load decrease cold leg transient is defined as a temperature change of 23.4°F. The bounding transient for the group consisting of step changes from 550°F to 525°F to 550°F, therefore, continues to bound the large step load decrease transient. Since the previous analyses continue to bound the large step load decrease transient, no further consideration of this transient is required.

The COMS operation transients were not considered in the previous RCP evaluations applicable to Callaway. For the purposes of the RCP evaluation, this transient can be considered to consist of several different occurrences. These are the bulk temperature transient, the local temperature transient, and the pressure transients. The pressure transients considered for the Code pressure boundary evaluations are 6,000 cycles of $\Delta P = 605$ psi.

The bulk temperature transient is a temperature rise of 38°F occurring over a 10-second time period, applicable to the RCS, which includes the RCP. Ten cycles of this transient are postulated. The temperature of the RCS at the initiation of this transient may be anywhere between 70°F and 350°F. This aspect of the COMS transient is considered by adding 10 additional cycles to the transient group that included the loss of load, loss of power, partial loss of flow, reactor trip without cooldown, and reactor trip with cooldown but no safety injection, among others.

The local temperature transient is described in the transient definitions as follows:

"In the loop with pump start and forward flow, the RCP, the reactor vessel downcomer/inlet plenum, and connecting RCS coolant pipe will experience a very brief initial temperature transient from T_{RCS} to T_{LS} to T_{RCS} ...T_{RCS} 160°F or less and T_{LS} is between 40°F and 115°F." In the above description of the local temperature transient, T_{RCS} is the maximum temperature in the RCS at the time the RCP is put into operation, and T_{LS} is the minimum loop seal (crossover leg) temperature.

This local temperature transient is based on transporting 300 gallons of cold water postulated to have accumulated in the crossover leg through the RCP and then on into the reactor vessel. To determine how much this local transient affects the temperature of the pump casing, a thermal analysis was performed. The result is that the temperature at the surface of the pump casing changes by 58°F for the local transient. When the bulk temperature transient is considered along with this local transient, the inner surface of the casing will see a ΔT of –58° and +38°F from the initial conditions, or a total temperature change of 96°F.

It is noted that the range of ΔT for the local plus the bulk transient is 96°F. This is less overall temperature change than the previously analyzed inadvertent RCS depressurization (ΔT = 160°F) and excessive feedwater flow transients (ΔT = 132°F peak, 105°F sustained) (Reference 4), so that the local effects of the COMS transient may also be handled merely by increasing the number of cycles considered for the existing transient groups.

The qualification of the pump is based on using a fatigue waiver, as defined in Section NB-3222.4(d) of the ASME Code, to address fatigue for the Code pressure boundary parts of the pump. The addition of the COMS transient to the NSSS design transients requires that the fatigue waiver calculations be re-performed. The previous fatigue waiver calculations are contained in Reference 4. Re-performing the fatigue waiver calculations shows that the pump casing, the thermal barrier, and the seal housing still qualify for the fatigue waiver. The addition of the 6,000 new pressure cycles associated with the COMS transient means that the bolting ring no longer qualifies for a fatigue waiver.

Since the bolting ring no longer qualifies for a fatigue waiver, a calculation of the fatigue cumulative usage factor for the bolting ring is required. Because the bolting ring is separated from the reactor coolant by the thermal barrier, the bolting ring does not see significant thermal stresses associated with the NSSS design transients other than heatup and cooldown. The bolting ring does see stress changes associated with the pressure transients, though. When the stress changes associated with heatup and cooldown are considered, along with the stress changes associated with the pressure transients, a cumulative usage factor remained below the Code allowable of 1.0. Refer to Table 5.6-3.

Since the bolting ring no longer qualifies for the fatigue waiver, the main closure bolts must also have a fatigue evaluation performed. These bolts are tightened to a preload sufficient to withstand the pressures in the RCP without leakage, and to provide metal-to-metal contact of the joint, where the gaskets are contained in grooves. As such, the stress in the bolts will not change significantly with pressure, and the stress changes contributing significantly to fatigue are associated with large temperature changes like those experienced during plant heatup and cooldown. A fatigue analysis of these bolts shows that the calculated cumulative usage factor for the bolts, due to the 200 specified cycles of heatup and cooldown remains below the Code allowable of 1.0.

Even though the fatigue waiver conditions are satisfied for the other components of the pump, there are 2 other areas where cumulative usage factors are calculated as part of simplified elastic-plastic analyses. These analyses are performed per Section NB-3228.3 of the ASME Code when the 3 S_m limit on the range of local primary-plus-secondary stress intensity is exceeded, where S_m is the allowable stress intensity defined in the ASME Code for the material at operating temperature. The pump components where these analyses were performed are the weir and the casing nozzles.

For the weir, the stresses associated with the excessive feedwater flow transient caused the stress intensity range to exceed 3 S_m. To calculate the cumulative usage factor, all of the transients except the 5-percent/minute unit loading and unloading and the boron concentration equalization transients were grouped and assumed to have a stress range equal to the worst case stress range for the excessive feedwater flow transient. A separate stress range was calculated for the 5-percent/minute unit loading and unloading and the boron concentration equalization transients, since they combine large numbers of cycles with low stress intensities for the weir. The stress associated with the COMS pressure transients is low enough that the 6,000 pressure cycles can be considered along with the 5-percent/minute unit loading and unloading and the boron concentration equalization transients. The 10 temperature transients associated with the COMS transient were added to the transient group that used the excessive feedwater flow transient stress range for calculating usage. The result of this calculation is that the cumulative usage factor for the weir increased from the previous calculation for the RSG Program conditions but remains below the ASME Code allowable value of 1.0.

For the casing nozzles, the most severe location for the cumulative usage factor calculated as a result of the simplified elastic-plastic analysis is the casing/discharge nozzle juncture. As for the weir, the revised cumulative usage factor will be calculated by simply increasing the number of cycles considered. The previous analysis of this area conservatively used the stress intensity range associated with the inadvertent depressurization transient as the basis of the usage factor calculation, even though the other transient considered, the excessive feedwater flow transient, could have been evaluated with a lower stress intensity range. The current evaluation will continue in this conservative fashion, utilizing the stress intensity range associated with the inadvertent depressurization transient as the basis for the evaluation of all three transients (inadvertent depressurization, excessive feedwater flow, and COMS) that are considered. On this basis, the cumulative usage factor for this area increases but remains below the ASME Code allowable value of 1.0.

The evaluation of the RCPs for the Callaway RSG Program did not require the calculation of new stresses for comparison to Code allowables. New values of cumulative usage factor were calculated for 3 areas of the pumps. These are all below the Code allowable value of 1.0. These new or revised usage factors are summarized in Table 5.6-3.

The RCP code pressure boundary, therefore, is shown to continue to comply with the requirements of the ASME Code for the NSSS design parameters and NSSS design transients defined for the RSG Program.

5.6.4.2 Reactor Coolant Pump Motor

From the hydraulic evaluation, impeller serial number 1232 was identified as the impeller with the highest power requirements. The worst condition was identified as a reactor coolant flow of 103,500 gpm/loop with a steam generator outlet temperature of 538°F. The maximum brake horsepower (BHP) requirements were identified as 7,013 BHP for the hot reactor coolant condition and 8,999 BHP for the cold reactor coolant condition. The associated hydraulic thrusts were calculated as 52,317 pounds for the hot reactor coolant condition and 71,228 pounds for the cold reactor coolant condition. The Callaway RCP motors have a nameplate rating of 7,000 HP for the hot reactor coolant condition and a nameplate rating of 8,750 HP for the cold reactor coolant condition. The hot reactor coolant condition power requirement is only 0.2 percent above the motor's nameplate rating, but the cold reactor coolant condition power requirement exceeds the cold nameplate rating by 2.8 percent. Therefore, additional evaluations were performed to demonstrate the acceptability of the RCP motors.

Per the Equipment Specifications, References 10 and 11, the motor is required to drive the pump continuously under hot reactor coolant conditions without exceeding a stator winding temperature rise of 70°C. This corresponds to the National Electrical Manufacturer's Association (NEMA) Class B temperature rise limit in a 50°C ambient temperature. The motors during test have shown a stator winding temperature rise no greater than 44.75°C at the rated hot condition load of 7,000 HP. Therefore, adequate margin exists for continuous operation at loads in excess of the 7,000 HP nameplate rating.

The maximum hot reactor coolant load under the revised operating conditions is 7,013 HP, which is less than a 0.2-percent increase over the nameplate rating of the motor. The stator temperature rise at this new load is estimated to be no greater than 44.85°C, which is a negligible change over the temperature rise at the nameplate rating. Therefore, continuous operation at the revised load is acceptable.

Per the Equipment Specification, Reference 10, the motor is required to drive the pump for up to 50 hours (continuous) and 3,000 hours maximum over the 40-year design life of the plant under cold reactor coolant conditions without exceeding a stator winding temperature rise of 95°C. This corresponds to the NEMA-guaranteed limit for a Class F winding in a 50°C ambient temperature. By test, the motors are shown to have a stator temperature rise of 65.5°C at the cold condition nameplate rating of 8,750 HP. Analysis indicates that the cold reactor coolant condition temperature rise of the stator at the revised load of 8,999 HP will be approximately 68.0°C, which is well below the NEMA limit. Therefore, cold loop operation at the revised load is acceptable.

Per the Equipment Specification, Reference 11, the motor is required to start across the line with a minimum 80-percent starting voltage, against the reverse flow of the other pumps running at full speed, under cold loop conditions. The limiting component for this type of duty is the rotor cage winding. A conservative all-heat-stored analysis is used to determine if the rotor cage winding temperature rise exceeds the design limits, which are 300°C on the bars and 50°C on the resistance rings. The results of this analysis show a bar temperature rise of 165.6°C and a ring temperature rise of 16.8°C. These temperatures do not exceed the design limits. Therefore, the motor can safely accelerate the load under worst-case conditions.

Performance of the thrust bearings in an RCP motor can be adversely affected by excessive or inadequate loading. Per the Equipment Specification, Reference 11, and the hydraulic evaluation in the axial down thrust for the revised parameters increased from 46,000 pounds to 52,317 pounds for hot reactor coolant operation and increased from 66,000 pounds to 71,228 pounds for cold reactor coolant operation. The thrust bearings are designed for loads exceeding 109,000 pounds, so the changes in the bearing loads are considered acceptable for the revised loads.

5.6.5 Conclusions

Based on the evaluation, the design of the Model 93A-1 RCP meets the applicable ASME Code requirements for structural integrity at the revised RCS conditions associated with the RSG Program for the Callaway plant. The RCP motors are also shown to be acceptable for the loads calculated for the revised RCS conditions.

5.6.6 References

1. ASME Boiler and Pressure Vessel (B&PV) Code, Section III, Division 1, Nuclear Power Plant Components, Subsection NB, Class 1 Components, 1971 Edition up to and including Summer 1973 Addenda.

2. Westinghouse Design Specification 952385, Rev. 10, "Standardized Nuclear Unit Power Plant System, Reactor Coolant Pump Model 93A-1," June 6, 1997.

3. Westinghouse Design Specification G-953115, Rev. 1, "General, Reactor Coolant Pump Model (93A-1)," December, 1984.

4. Westinghouse Engineering Memorandum 5556, Rev. 0, "Union Electric Company, Callaway Plant Unit No. 1, Pressure Boundary Summary Report, SCP – S.O. U416," E. Danfelt, Westinghouse Electric Corporation, Electro-Mechanical Division, June 8, 1981.

5. Westinghouse Engineering Memorandum 5556, Addendum 1, Rev. 0, "Union Electric Company, Callaway Plant Unit No. 1, Pressure Boundary Summary Report, Addendum 1 – Analysis of Spare Reactor Coolant Pump Assembly, SCP S.O. U481," R. M. Perlman, May 6, 1983.

6. Westinghouse Engineering Memorandum 5556, Addendum 2, Rev. 0, "Union Electric Company, Callaway Plant Unit No. 1, Pressure Boundary Summary Report, Addendum 2 – 93A-1 Reactor Coolant Pump Redesigned No. 1 Seal Housing Bolt Stress Analysis," J. Casamassa, January 22, 1997.

7. Westinghouse Engineering Memorandum 5556, Addendum 3, Rev. 0, "Union Electric Company, Callaway Plant Unit No. 1, Pressure Boundary Summary Report, Addendum 3 – Evaluation of a Spare Thermal Barrier Assembly, S.O. 2Q37D," D. E. Dietrich, July 14, 1997.

8. Westinghouse Engineering Memorandum 5556, Addendum 4, Rev. 0, "Union Electric Company, Callaway Plant Unit No. 1, Pressure Boundary Summary Report, Addendum 4 – Pump Interchangeability Evaluation, S.O. 2Q66E (SCP)," D. E. Dietrich, August 16, 1999.

9. Westinghouse Engineering Memorandum 5556, Addendum 5, Rev. 0, "Union Electric Company Callaway Plant Unit No. 1, Pressure Boundary Summary Report, Evaluation of a Spare Thermal Barrier Assembly, S. O. 2Q17F," R. M. Perlman, February 22, 2000.

10. Westinghouse Specification 569700, Rev. L, "General Specification for Induction Motor for Shaft Seal Type Pump," September 15, 1977.

11. Westinghouse Specification 569719, Rev. H, "Supplementary Ordering Information for Shaft Seal Type Pump Motor, Union Electric Company Supplement to E-569700, SCP," January 17, 1977.

Table 5.6-1 Pump Impellers Considered in Evaluation of Callaway Reactor Coolant Pump Motors		
Impeller Serial No.	**Pump Serial Number**	**Originally Built For**
Impeller S/N 1232	RCP 2-9744D36G01	Wolf Creek
Impeller S/N 1231	RCP 4-9744D36G01	Wolf Creek
Impeller S/N 1492	RCP 5-9744D35G01	Vogtle
Impeller S/N 1443	RCP 9-9750D67G01	Callaway Spare

Table 5.6-2 Operating Temperatures and Pressures Used for Analysis of Callaway Reactor Coolant Pumps		
Letter or Report	**Vessel Inlet Temperature**	**Reactor Coolant Pressure**
Current RSG Program	538.2°F to 556.8°F	2250 psia
EM-5556 (Original PBSR)	558.6°F (pump inlet)	2250 psia

Table 5.6-3 Cumulative Usage Factors Recalculated for the Reactor Coolant Pumps for the Callaway RSG Program			
Component/Location	**Previous Cumulative Usage Factor**	**Recalculated Cumulative Usage Factor**	**Allowable**
Bolting Ring	None(1)	[ ]^{a,c}	1.0
Main Closure Bolts	None(1)	[ ]^{a,c}	1.0
Weir	[ ]^{a,c}	[ ]^{a,c}	1.0
Casing/Discharge Nozzle Juncture	[ ]^{a,c}	[ ]^{a,c}	1.0

(1) The previous calculations showed that the bolting ring and main closure bolts qualified for a fatigue waiver, but a calculation of the cumulative usage was performed for the closure components anyway since this aspect of the design was new.

Bracketed [ ]^{a,c} information designates data that is Westinghouse proprietary, as discussed in Section 1.6 of this report.

5.7 PRESSURIZER EVALUATION

5.7.1 Introduction and Background

An evaluation was performed to support the Callaway RSG Program to address the impact on the pressurizer. This evaluation is based on the range of NSSS operating parameters to support an NSSS power level of 3,579 MWt.

The functions of the pressurizer are to absorb any expansion or contraction of the primary reactor coolant due to changes in temperature and/or pressure and, in conjunction with the pressure control system components, to keep the RCS at the desired pressure. The first function is accomplished by keeping the pressurizer approximately half full of water and half full of steam at normal conditions, connecting the pressurizer to the RCS at the hot leg of one of the reactor coolant loops and allowing inflow to or outflow from the pressurizer as required. The second function is accomplished by keeping the temperature in the pressurizer at the water saturation temperature (Tsat) corresponding to the desired pressure. The temperature of the water and steam in the pressurizer can be raised by operating electric heaters at the bottom of the pressurizer and can be lowered by introducing relatively cool spray water into the steam space at the top of the pressurizer.

The components in the lower end of the pressurizer (such as the surge nozzle, lower head/heater well and support skirt) are affected by pressure and surges through the surge nozzle. The components in the upper end of the pressurizer (such as the spray nozzle, safety and relief nozzle, upper head/upper shell, manway and instrument nozzle) are affected by pressure, spray flow through the spray nozzle, and steam temperature differences.

The limiting operating conditions of the pressurizer occur when the RCS pressure is high and the RCS hot leg (Thot) and cold leg (Tcold) temperatures are low. This maximizes the ΔT that is experienced by the pressurizer. Due to flow out of and into the pressurizer during various transients, the surge nozzle alternately sees water at the pressurizer temperature (Tsat) and water from the RCS hot leg at Thot. If the RCS pressure is high (which means, correspondingly, that Tsat is high) and Thot is low, then the surge nozzle will see maximum thermal gradients; and, thus experience the maximum thermal stress. Likewise, the spray nozzle and upper shell temperatures alternate between steam at Tsat and spray water, which for many transients, is at Tcold. Thus, if RCS pressure is high (Tsat is high) and Tcold is low, then the spray nozzle and upper shell will also experience the maximum thermal gradients and thermal stresses.

5.7.2 Input Parameters and Assumptions

The NSSS design parameters provided in Section 2.0 and the NSSS design transients discussed in Section 3.1 provide the operating and transient conditions for those operating and transient conditions that differ from those addressed in References 1 through 5, to which the Callaway plant pressurizers were already designed and analyzed.

The reactor vessel outlet (Thot) and reactor vessel inlet (Tcold) temperatures from the NSSS design parameters in Section 2.0 define the normal operating temperatures for the surge and spray lines to the pressurizer. The reactor coolant pressure from Section 2.0 defines the pressurizer normal operating pressure (2,250 psia) and saturated temperature (653 °F). The minimum values of Thot and Tcold from all cases in Table 2-1 are used in the evaluation of the pressurizer.

The pressurizer temperature and pressure variations, surge and spray line flow rates, and reactor vessel Thot and Tcold for the NSSS design transients are applicable to the pressurizer.

Seismic analyses and non-pressure boundary component evaluations are unaffected by the RSG Program.

5.7.3 Description of Analyses and Evaluations

The analysis was performed by modifying results from the original Callaway plant pressurizer stress reports (References 4 and 5), which were performed to the requirements of the ASME B&PV Code, Section III, 1974 Edition, Summer 1974 Addendum (Reference 6). Analytical models of various sections of the pressurizer were subjected to pressure loads, external loads (such as piping loads), and thermal transients.

The input parameters associated with the Callaway RSG Program were reviewed and compared to the design inputs considered in the current pressurizer stress reports. In cases where revised input parameters are not obviously bounded, pressurizer structural analyses and evaluations were performed. Any impacts to the existing design-basis analysis were performed through a comparative analysis of the changes. This method involves a simplified engineering approach, using the existing analyses as the basis of the evaluation. Revised cumulative usage factors were calculated, as applicable, and compared to previous licensed results. The evaluation results were then compared with the ASME Code (References 6 and 7) to confirm that the allowable limits are maintained.

5.7.4 Acceptance Criteria and Results

The initial set of acceptance criteria for evaluating design inputs affecting the pressurizer stress reports by comparison with the design inputs considered in References 1 through 5 is the following:

1. Hot and cold leg temperatures remain within the ranges of the operating temperatures that have previously been considered and justified in the pressurizer stress reports.

2. The NSSS design transients are less than or equal to the design transients previously considered in the pressurizer stress reports with regard to both severity and numbers of occurrences. Additionally, no new NSSS design transients that have not previously been considered are identified. (The pressurizer temperature and pressure variations for each transient were considered in this comparison review to determine the relative severity of the revised design transients compared to the existing design transients.)

3. Design loads are less than or equal in magnitude to the loads that were previously considered in the pressurizer stress reports with no changes to the load application points and numbers of occurrences.

If comparison of the design inputs for the Callaway RSG Program with the design inputs considered in References 1 through 5 reveal hot and/or cold leg temperatures, NSSS design transients, or design loads that do not comply with the above criteria, pressurizer structural analyses and evaluations were performed, as necessary to incorporate the revised design inputs. The acceptance criterion is that the Callaway plant pressurizer components meet the stress/fatigue analysis requirements of the ASME Code, Section III (References 6 and 7) for the plant operation in accordance with the RSG Program.

The critical pressurizer components are the spray nozzle, upper shell, surge nozzle, lower shell, skirt, flange, heater well, safety and relief nozzles, instrument nozzle, and immersion heater. The results of the evaluation are valid for the RSG Program to 3,579 MWt. The fatigue usage summary for the Callaway plant pressurizer is presented in Table 5.7–1.

Summary stress results are given in Reference 4 for the original design transients. The RSG Program transients were compared to the original transients in References 1 through 3 and found to be bounded by the original design transients, except that adjustments to account for transients that had not been included in the original evaluations were needed. Those transients were the feedwater heater out of service and bank of feedwater heaters out of service. The fatigue usage was recalculated at the most critical location of each component. All ASME Code stress limits are satisfied. The components evaluated were:

- Spray nozzle
- Upper head
- Surge nozzle
- Safety and relief nozzle
- Support skirt and flange
- Lower head
- Heater well
- Seismic support lug
- Shell at support lug
- Trunnion buildup
- Instrument nozzle
- Manway bolt
- Manway pad
- Valve support bracket
- Immersion heater

5.7.5 Conclusions

The analysis performed here shows that the Callaway RSG Program transients will have a minimal effect on the pressurizer components. Table 5.7-1 compares the fatigue usages calculated here with those reported in the original stress reports (References 4 and 5).

All critical components of the Callaway plant pressurizer were evaluated for operation in the RSG Program. It was determined that all ASME Code stress limits remain satisfied for all components, for all proposed operating conditions.

5.7.6 References

1. Westinghouse Rev. 0, "Systems Standard 1.3F Nuclear Steam Supply System Reactor Coolant System Design Transients for Standard Plants with Model F Steam Generators," March, 1978.

2. Westinghouse Design Specification 955285, Rev. 0, "General Pressurizer Series 84F," May 1, 1981.

3. Westinghouse Design Specification 952575, Rev. 5, "Addendum to Design Specification 955285, Rev. 0, Standardized Nuclear Unit Power Plant System (SNUPPS)," November 14, 1988.

4. Westinghouse WNET-138 Volumes 2–16, Model F Series 84 Pressurizer Generic Stress Report, 1980.

5. Westinghouse WNEP-8825, "Cold Overpressure Mitigation System Evaluation for the Series 84 Model F Pressurizer," May, 1988.

6. ASME Boiler and Pressure Vessel Code, Section III, 1974 Edition with Addenda through Summer 1974.

7. ASME Boiler and Pressure Vessel Code, Section III, 1989 Edition.

Table 5.7-1 Callaway Plant Fatigue Usage Comparisons	
Component	**Fatigue Usage**
Spray Nozzle	[ ]^{a,c}
Upper Head	[ ]^{a,c}
Surge Nozzle	[ ]^{a,c}
Safety and Relief Nozzle	[ ]^{a,c}
Support Skirt and Flange	[ ]^{a,c}
Lower Head	[ ]^{a,c}
Heater Well	[ ]^{a,c}
Seismic Support Lug	[ ]^{a,c}
Shell at Support Lug	[ ]^{a,c}
Trunnion Buildup	[ ]^{a,c}
Instrument Nozzle	[ ]^{a,c}
Manway Bolt	[ ]^{a,c}
Manway Pad	[ ]^{a,c}
Valve Support Bracket	[ ]^{a,c}
Immersion Heater	[ ]^{a,c}

Bracketed [ ]^{a,c} information designates data that is Westinghouse proprietary, as discussed in Section 1.6 of this report.

5.8 AUXILIARY EQUIPMENT

5.8.1 Introduction and Background

This analysis evaluates the Callaway plant auxiliary heat exchangers, tanks, pumps, and valves on a system basis, provided by Westinghouse for impact by the thermal transients and maximum operating temperatures, pressures, and flow rates resulting from the RSG conditions. The systems affected by the RSG conditions for which Westinghouse supplied equipment are the RCS, chemical and volume control system (CVCS), safety injection system (SIS), RHR system , and the component cooling water (CCW) system. The evaluation consists of a structural fatigue review and flow capacity review of the component pressure boundaries. The review does not include a structural evaluation or a performance/controllability evaluation of the subcomponents for any of the components listed in this report (such as, valve actuators, controllers, electronics, or pump motors), for the RSG, unless specifically noted.

5.8.2 Input Parameters and Assumptions

The auxiliary system heat exchangers, tanks, auxiliary system pumps and valves that Westinghouse provided to Callaway were reviewed. The component design information is contained in the design documents and was used as input to the evaluation. This information contained in these documents includes pressure and temperature design conditions as well as design transients applicable to each individual identified component. The NSSS design parameters provided in Table 2-1 document the impact of the RSG Program on the NSSS operating temperatures and pressures. This information was applied where applicable for evaluation of the auxiliary equipment maximum operating temperatures and pressures. Section 3.0 defines the impact on the auxiliary heat exchangers, tanks, pumps, and valves subject to the auxiliary system transients and the impact on the NSSS transients on the auxiliary system valves that are subjected to these transients.

It is assumed that the original equipment provided by Westinghouse represent the actual hardware in the plant, and that any changes made to the auxiliary heat exchangers, tanks, pumps, and valves or operation of these components have been made by the plant in accordance with the original technical and quality assurance requirements.

5.8.3 Description of Analyses and Evaluations

The equipment design parameters were reviewed for the auxiliary heat exchanger, tanks, pumps, and valves. The specific criteria included design temperature, pressure, thermal transients, and flow rates. These parameters were compared to those used in the RSG Program to determine if the design parameters still enveloped those for the program.

5.8.3.1 Auxiliary System Heat Exchangers

The NSSS auxiliary heat exchangers were evaluated for the RSG Program conditions. The design data used in the manufacture of each heat exchanger as well as the system(s) in which each heat exchanger is located were reviewed. The specifications identified the applicable design transients, and the data sheets identified the design temperature and pressure.

Based on the NSSS design parameters for the RSG Program, there is no impact on the auxiliary systems heat exchangers. The operating temperature and pressure ranges for these vessels remain bounded by the original design parameters. The heat exchangers identified as having transients in the original design specifications are the regenerative, letdown, excess letdown, letdown reheat, and RHR heat exchangers. All of these transients remain bounded by the original design conditions.

5.8.3.2 Auxiliary System Tanks

The only tank for which transients are identified is the safety injection accumulators. The operating temperatures and pressures for all auxiliary tanks remain within the design basis, and the safety injection accumulators remain bounded by the original design transients. As a result, none of the auxiliary tanks are impacted by the RSG Program conditions.

5.8.3.3 Auxiliary System Pumps

The NSSS auxiliary pumps were evaluated for the RSG Program conditions. The design data used in the manufacture of each pump was reviewed. The operating temperature and pressure ranges for these pumps remain bounded by the original design parameters. The original design transients for the auxiliary equipment bound the transients associated with the RSG Program. There is no impact on the auxiliary system pumps as a result of the RSG Program.

5.8.3.4 Auxiliary System Valves

The NSSS auxiliary system valves were evaluated for the RSG Program conditions. The design data used in the manufacture of auxiliary valves as well as the system(s) in which each valve is located were reviewed. The specifications identified the applicable design transients, and the data sheets identified the design temperature and pressures.

There is no impact upon the auxiliary system valves as a result of the RSG Program. The operating temperature and pressure ranges for the valves remain bounded by the original design parameters. The original design transients for the auxiliary equipment remain bounded for the transients associated with the RSG Program.

5.8.4 Acceptance Criteria and Results

In order to qualify the equipment, it must be demonstrated that the maximum system operating temperatures, pressures, and flow rates for the RSG Program must be bounded by or equal to the original system design conditions. If this is the case, no further effort is required to qualify the auxiliary tanks, heat exchangers, pumps, and valves. Any values in excess of the design values will be addressed in this report.

In addition, the original design transients must bound the revised auxiliary tanks, heat exchangers, pumps, and valve transients, with fatigue usage factors being less than 1.0, for the RSG Program. If this is the case, no further effort is required to qualify the equipment for this aspect of the RSG. Should the revised transients not be bounded by the original equipment design, then each affected piece of equipment will need to be re-qualified for the new transient conditions on a case-by-case basis.

A comparison of the RSG conditions shows that all maximum operating temperatures and pressures for systems evaluated are bounded by the existing design basis. Since all tanks, heat exchangers, pumps, and valves were designed and manufactured consistent with the system design and applicable codes and standards, all of the NSSS tanks, heat exchangers, pumps, and valves are acceptable for the maximum system operating temperatures and pressures resulting from the RSG Program. The auxiliary equipment and NSSS thermal transients resulting from the RSG are bounded by the original Callaway design parameters. Therefore, the auxiliary tanks, heat exchangers, pumps, and valves remain acceptable for the thermal transients resulting from the RSG Program. This is also applicable to any equipment for the above identified systems that may have been changed or replaced in accordance with the original Westinghouse technical and quality assurance requirements.

5.8.5 Conclusions

The Callaway auxiliary tanks, heat exchangers, pumps, and valves are acceptable for the RSG Program conditions since there is no change to the auxiliary systems operating conditions identified as a consequence of the RSG.

The results for the RSG Program are consistent with, and continue to comply with, the current Callaway licensing basis/acceptance requirements.

6 NUCLEAR STEAM SUPPLY SYSTEM ACCIDENT ANALYSES

This section provides the results of the analyses and/or evaluations that were performed for the nuclear steam supply system (NSSS) accident analyses in support of the Replacement Steam Generator (RSG) Program. The accident analysis areas addressed in this section include:

- Initial condition uncertainties
- Loss-of-coolant accidents (LOCAs)
- Non-LOCA events, including anticipated transients without scram (ATWS)
- Steam generator tube rupture (SGTR)
- LOCA mass and energy releases
- Main steam line break (MSLB) mass and energy releases
- Steam tunnel (Area 5)
- LOCA hydraulic forces
- Reactor trip/engineered safety feature actuation system setpoints
- Rod ejection releases for dose

The detailed results and conclusions of each analysis are presented within each subsection.

6.1 INITIAL CONDITION UNCERTAINTIES

6.1.1 Introduction and Background

Initial condition uncertainties are conservative steady-state instrumentation measurement uncertainties that are applied to nominal parameter values in order to obtain conservative initial conditions for use in safety (accident) analyses. The initial condition uncertainties documented in Table 6.1-1 were calculated at replacement steam generator (RSG) conditions for use in the RSG analyses and/or evaluations that were performed to assess the acceptability of the safety analyses. The initial condition uncertainties were then provided as input to the various analysis groups.

6.1.2 Input Parameters and Assumptions

The uncertainty calculations for the Callaway Plant were performed based on the plant-specific instrumentation, plant calibration procedures, and customer-issued letters.

6.1.3 Description of Analyses and Evaluations

The uncertainty analysis uses the square-root-sum-of-the-squares (SRSS) technique to combine the uncertainty components of an instrument channel in an appropriate combination of those components, or groups of components, that are statistically independent. Those uncertainties that are not independent are arithmetically summed to produce groups that are independent of each other, which can then be statistically combined.

Initial condition uncertainties were calculated for the following parameters that were known to be affected by the Callaway RSG conditions or are explicitly modeled in the Callaway RSG safety analyses:

- Reactor coolant system (RCS) Tavg control
- Steam generator water level control

The initial condition uncertainties accounted for issues identified in References 1 and 2.

6.1.4 Acceptance Criteria and Results

The acceptance criterion for the Tavg control and steam generator water level control uncertainty is that the calculated initial condition uncertainties must be less than or equal to the initial condition uncertainty values used in the safety analyses. This criterion has been satisfied.

6.1.5 Conclusions

The results of this analysis are summarized in Table 6.1-1 and used in the accident analysis.

6.1.6 References

1. Westinghouse NSAL-02-05, Rev. 1, "Steam Generator Water Level Control System Uncertainty Issue," April 2002.

2. Westinghouse NSAL-03-09 "Steam Generator Water Level Uncertainties," September 2003.

Table 6.1-1 Callaway Plant Summary of Initial Condition Uncertainties

Parameter	Previously Assumed Initial Condition Uncertainty(1)	Calculated Final Initial Condition Uncertainties(1)
RCS Tavg Control	[ ]^{a,c} (random) [ ]^{a,c} (bias)	[ ]^{a,c} (random) [ ]^{a,c} (bias)
Steam Generator Water Level Control @ 514" = 51.35% span (cold)(2)	[ ]^{a,c} span	[ ]^{a,c} span [ ]^{a,c} span

Notes:

1. The sign convention for the uncertainties is:
   "+" means instrumentation reads higher than the actual parameter
   "-" means instrumentation reads lower than the actual parameter
2. The calculated final initial condition uncertainty includes issues defined in Reference 2.

Bracketed [ ]^{a,c} information designates data that is Westinghouse proprietary, as discussed in Section 1.6 of this report.

6.2 LOCA TRANSIENTS

6.2.1 Large-Break LOCA

6.2.1.1 Introduction

The large-break loss-of-coolant-accident (LOCA) (LBLOCA) analysis of record for Callaway was completed using the 1981 Westinghouse LBLOCA Evaluation Model with BASH (BASH-EM) (Reference 1). The LBLOCA analysis was re-performed for the Replacement Steam Generator (RSG) Program to incorporate the RSG design, a full-power vessel average temperature (Tavg) range from 570.7°F to 588.4°F, and other changes identified through discussions between AmerenUE and Westinghouse. The analysis also modeled the appropriate plant-specific design features of Callaway such as an inverted top hat upper support plate, an upflow barrel-baffle region, and an upper head temperature equal to the cold leg temperature (Tcold). All prior 10 CFR 50.46 assessments were incorporated into the analysis, primarily through the use of corrected code versions and selection of input values.

The following sections provide an overview of the LBLOCA analysis methodology, assumptions and initial conditions, acceptance criteria, and results.

6.2.1.2 Analysis Methodology, Assumptions, and Initial Conditions

The LBLOCA methodology using BASH-EM was developed in accordance with the requirements of 10 CFR 50 Appendix K. This regulation was designed to produce a conservative prediction of the analysis results and includes various conservative modeling requirements such as the decay heat model (1971 ANS Infinite + 20%), the zirconium-water reaction model (Baker-Just), and the treatment of fuel rod burst and blockage. Additional input assumptions and initial conditions for the LBLOCA analysis are found in Tables 6.2.1-1 through 6.2.1-4.

The main codes comprising BASH-EM are described in Table 1-2 and include SATAN-VI, which calculates the blowdown thermal-hydraulic transient; BASH, which calculates the refill and reflood thermal-hydraulic transients; COCO, which calculates the minimum containment pressure transient; and LOCBART, which calculates the hot rod cladding temperature and oxidation transients for all three phases of the LBLOCA analysis. The LOCBART calculations were extended beyond the onset of downcomer boiling in BASH by reducing the core inlet flooding rate, to ensure adequate termination of the fuel rod cladding temperature and oxidation transients. Extension of the LOCBART calculations had no effect on peak cladding temperature (PCT) and no more than a minor effect on maximum local oxidation, with significant margin to the regulatory limits for each of these parameters.

6.2.1.3 Acceptance Criteria

The acceptance criteria for the LBLOCA analysis are specified in 10 CFR 50.46, as follows:

1. The calculated maximum fuel element cladding temperature shall not exceed 2,200°F.

2. The calculated total oxidation of the cladding shall 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 shall 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. Calculated changes in core geometry shall be such that the core remains amenable to cooling.

5. After any calculated successful initial operation of the emergency core cooling system (ECCS), the calculated core temperature shall be maintained at an acceptably low value and decay heat shall be removed for the extended period of time required by the long-lived radioactivity remaining in the core. (Note that this criterion is not addressed as part of the short-term LBLOCA analysis.)

6.2.1.4 Results

Calculations were completed varying the break discharge coefficient (CD), vessel average temperature, pumped injection flow rates (minimum/maximum), axial power distribution, and fuel rod type (integral fuel burnable absorber (IFBA)/non-IFBA) to determine the limiting conditions for the analysis. Table 6.2.1-5 provides the limiting results for each break discharge coefficient, and Table 6.2.1-6 provides the time sequence of events for the case that produced the maximum PCT. The PCT is 1,938°F; the maximum local oxidation is 6.4 percent; the core-wide hydrogen generation is less than 0.54 percent; and the core geometry remains amenable to cooling. The transient results for the limiting analysis case are provided in Figures 6.2.1-1 to 6.2.1-11.

6.2.1.5 Conclusions

The LBLOCA analysis results meet the pertinent acceptance criteria of 10 CFR 50.46. The PCT is less than 2,200°F; the maximum local oxidation is less than 17 percent; the core-wide hydrogen generation is less than 1 percent; and the core geometry remains amenable to cooling.

6.2.1.6 References

1. WCAP-10266-P-A, Revision 2, "The 1981 Version of the Westinghouse ECCS Evaluation Model Using the BASH Code," March 1987.

Table 6.2.1-1 Input Assumptions and Initial Conditions		
1.	**Core Parameters**	
	Licensed Core Power	3,565 MWt
	Calorimetric Uncertainty	2%
	Total Core Peaking Factor, F_Q	2.50
	Hot Channel Enthalpy Rise Factor, $F_{\Delta H}$	1.65
	Axial Offset (AO)	-20% ≤ AO ≤ +15%
	K(z) Limit	1.0 from 0 to 6 ft., 1.0 to 0.925 from 6 to 12 ft.
2.	**Reactor Coolant System**	
	Thermal Design Flow	93,600 gpm/loop
	Nominal Vessel Average Temperature	570.7/588.4°F
	Vessel Average Temperature Uncertainty	-3.0/+4.3°F
	Pressurizer Pressure	2,250 psia
	Pressurizer Pressure Uncertainty	±30 psi
3.	**Reactor Protection System (Reactor Trip)**	
	Reactor Trip Setpoint	1,860 psia
4.	**Containment**	
	Containment Net Free Volume	2,700,000 ft^3
	Minimum Refueling Water Storage Tank (RWST) Spray Temperature	37°F
	Number of Containment Spray Pumps Operating	2
	Minimum Time for Containment Spray Flow Delivery Relative to Accident Initiation	15 s
	Total Maximum Containment Spray Flow Rate	7,754 gpm
	Maximum Number of Containment Fan Coolers Operating	4
	Minimum Time to Initiate Fan Coolers Post-LOCA	35 s
	Fan Cooler Heat Removal Rate	Table 6.2.1-2
5.	**Steam Generators**	
	Steam Generator Model	Framatome 73/19T
	Maximum Steam Generator Tube Plugging	5%

Table 6.2.1-1 Input Assumptions and Initial Conditions (cont.)		
6.	**Safety Injection (SI) (Pumped SI)**	
	SI Configuration	Loss of one train of SI
	SI Water Temperature Range	37-100°F
	High Containment Pressure SI Setpoint	20.7 psia
	Low Pressurizer Pressure SI Setpoint	1,715 psia
	SI Actuation Delay Time	29 s
	Pump Curve Degradation Used in Minimum Injected Flow Calculations	7%
	Minimum SI Flows	Table 6.2.1-3
	Maximum SI Flows	Table 6.2.1-4
7.	**Safety Injection (Accumulators)**	
	Water/Gas Temperature	120°F
	Range of Accumulator Water Volume	6,061 – 6,655 gal
	Minimum Cover Gas Pressure	602 psia

Table 6.2.1-2 Fan Cooler Heat Removal Rate

Containment Temperature (°F)	Capacity (Btu/hr)
120	3.75E+07
150	6.75E+07
200	12.8E+07
225	16.0E+07
250	19.0E+07

Table 6.2.1-3 Minimum SI Flows Versus RCS Pressure			
PRCS (psig)	LHSI (lbm/s)	IHSI (lbm/s)	HHSI (lbm/s)
0	393.4	63.3	39.3
20	318.5	62.8	39.1
40	241.0	62.2	38.8
60	165.2	61.6	38.6
80	114.2	61.1	38.4
100	51.9	60.5	38.1
120	0.0	59.9	37.9
140	0.0	59.4	37.7
160	0.0	58.8	37.4
180	0.0	58.2	37.2
200	0.0	57.6	37.0
600	0.0	44.3	32.2

Table 6.2.1-4 Maximum SI Flows Versus RCS Pressure			
PRCS (psig)	LHSI (lbm/s)	IHSI (lbm/s)	HHSI (lbm/s)
0	1,056.5	85.8	91.4
20	969.4	85.2	91.0
40	878.0	84.6	90.6
60	780.6	84.0	90.2
80	675.0	83.4	89.8
100	556.4	82.8	89.4
120	446.0	82.2	89.0
140	361.3	81.6	88.6
160	258.9	80.9	88.2
180	117.4	80.3	87.8
200	0	79.7	87.4
600	0	66.3	78.7

Table 6.2.1-5 Large-Break LOCA Results							
Case	PCT (°F)	PCT Time(s)	PCT Elev. (ft.)	Hot Rod Burst Time(s)	Hot Rod Burst Elev.(ft.)	Hot Rod Max $Zr\text{-}O_2$ Reaction (%)	Hot Rod Max $Zr\text{-}O_2$ Reaction Elevation (ft)
CD=0.4	1594.8	109.0	7.0	93.81	7.25	1.84%	7.25
CD=0.6(1)	1937.2	146.2	7.25	54.59	6.25	2.89%	7.25
CD=0.6(2)	1877.2	167.8	9.0	81.01	9.0	6.37%	9.0
CD=0.8	1764.1	135.2	7.25	46.82	6.25	1.53%	7.25
CD=1.0	1644.7	8.6	6.25	67.11	7.25	2.01%	7.25

Notes:

1. This C_D=0.6 case resulted in the highest peak cladding temperature.
2. This C_D=0.6 case resulted in the highest maximum local oxidation.

Table 6.2.1-6 Large-Break LOCA Time Sequence of Events	
Start (s)	0.0
Reactor Trip Signal (s)	0.67
Safety Injection Signal (s)	1.6
Accumulator Injection (s)	18.4
End of Blowdown (s)	33.8
Start of Safety Injection (s)	30.6
Bottom of Core Recovery (s)	47.3
Accumulator Empty (s)	54.3

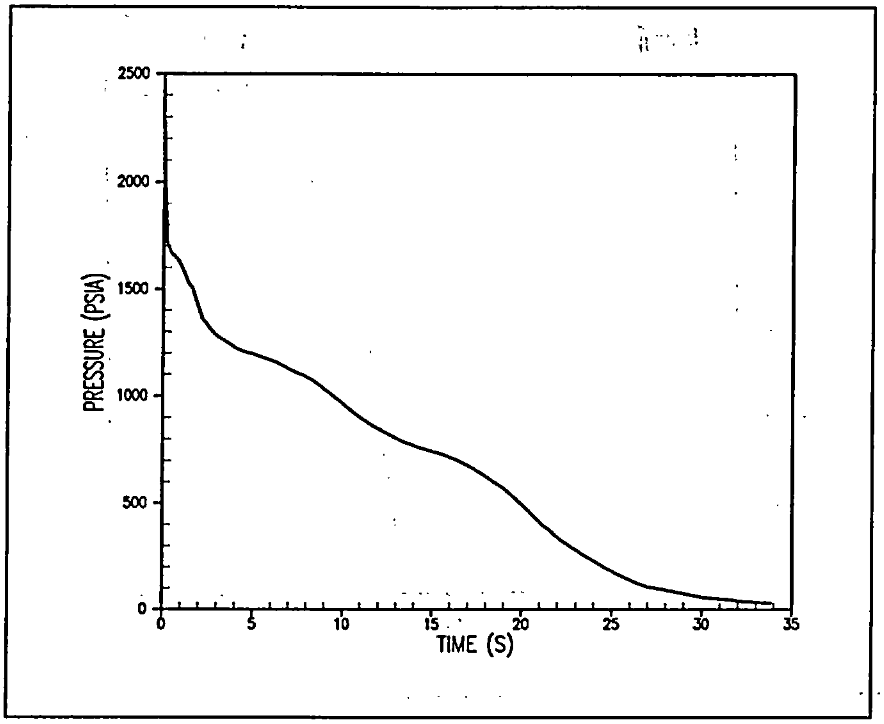

Figure 6.2.1-1 Core Pressure During Blowdown

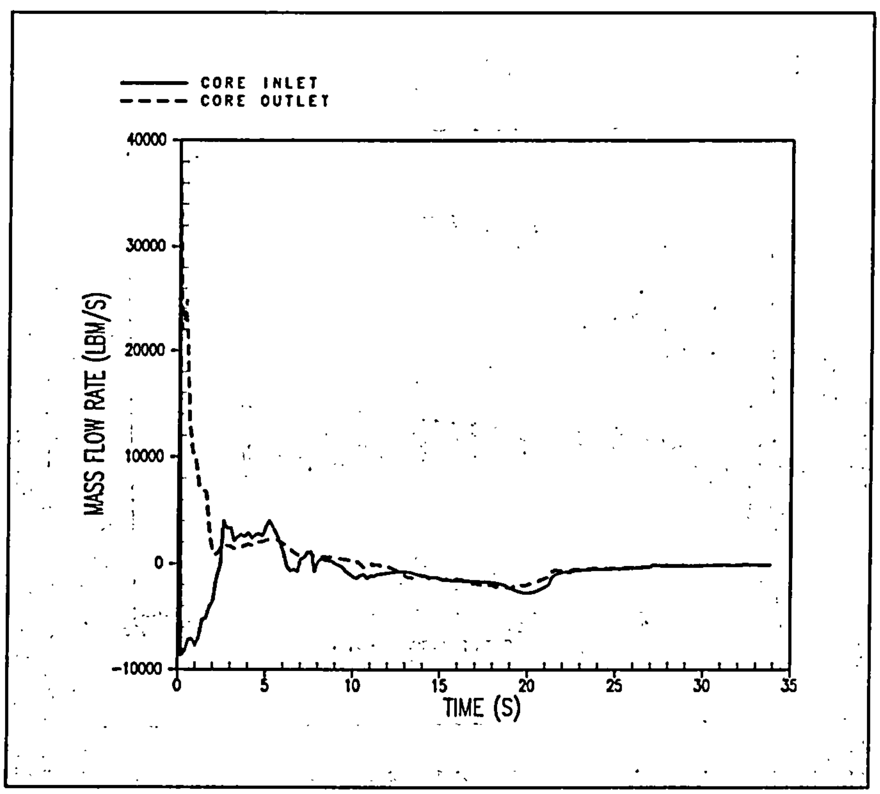

Figure 6.2.1-2 Core Inlet and Outlet Mass Flow Rate During Blowdown

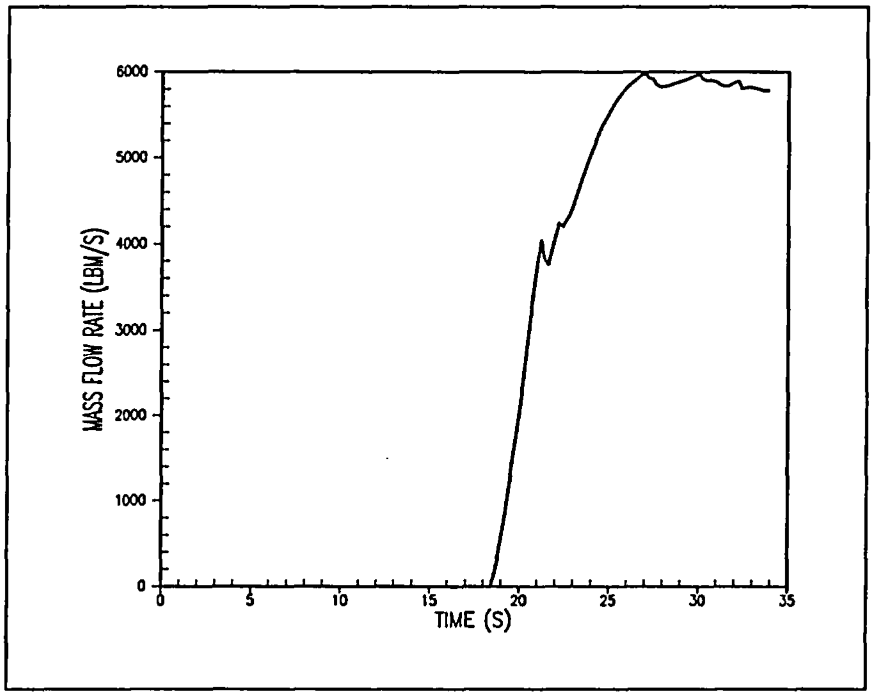

Figure 6.2.1-3 Intact Loop Accumulator Mass Flow Rate During Blowdown

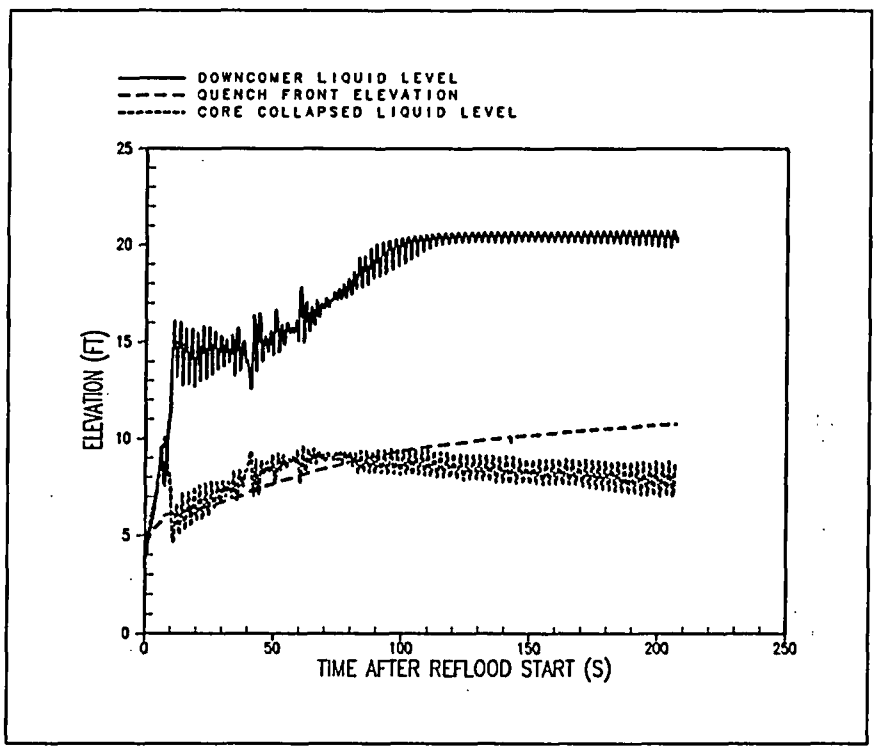

Figure 6.2.1-4 Vessel Liquid Levels During Reflood

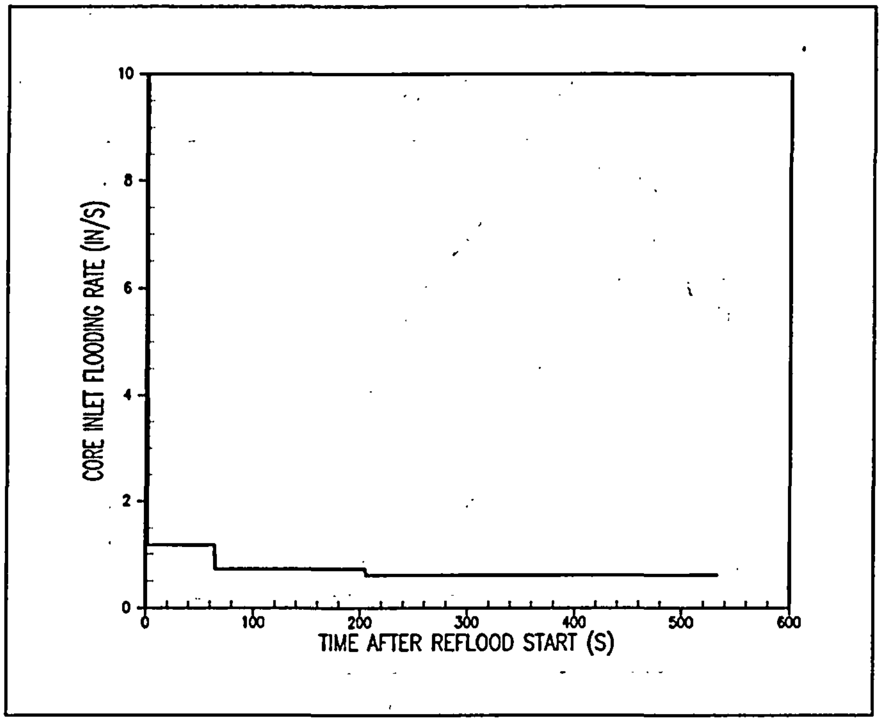

Figure 6.2.1-5 Core Inlet Flooding Rate During Reflood

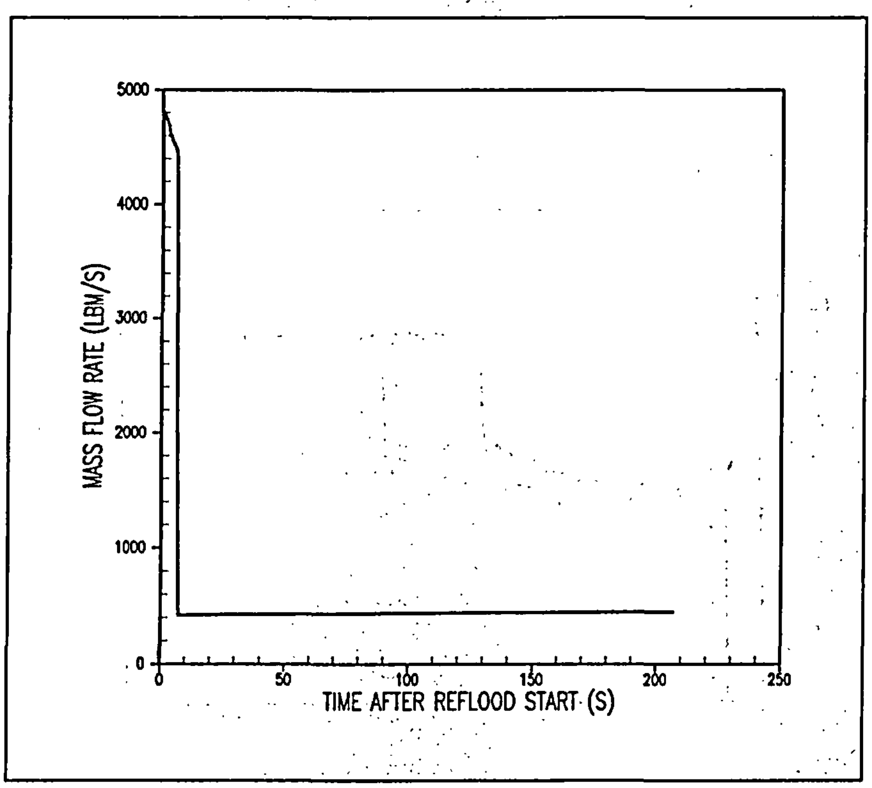

Figure 6.2.1-6 Intact Leg Accumulator and SI Mass Flow Rate During Reflood

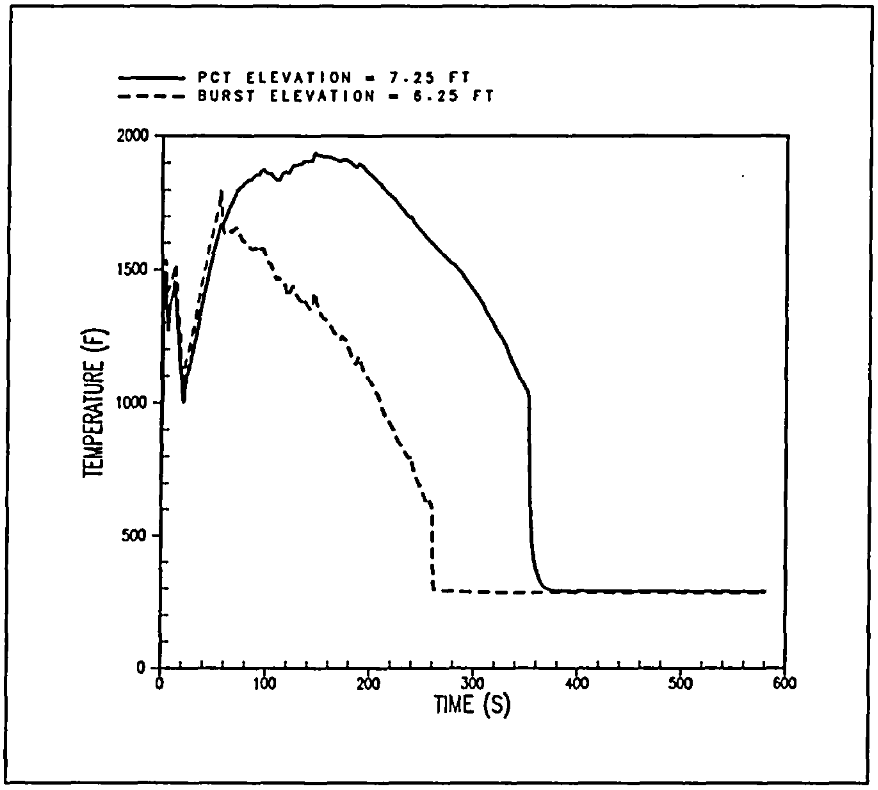

Figure 6.2.1-7 Cladding Temperature at PCT and Burst Elevations

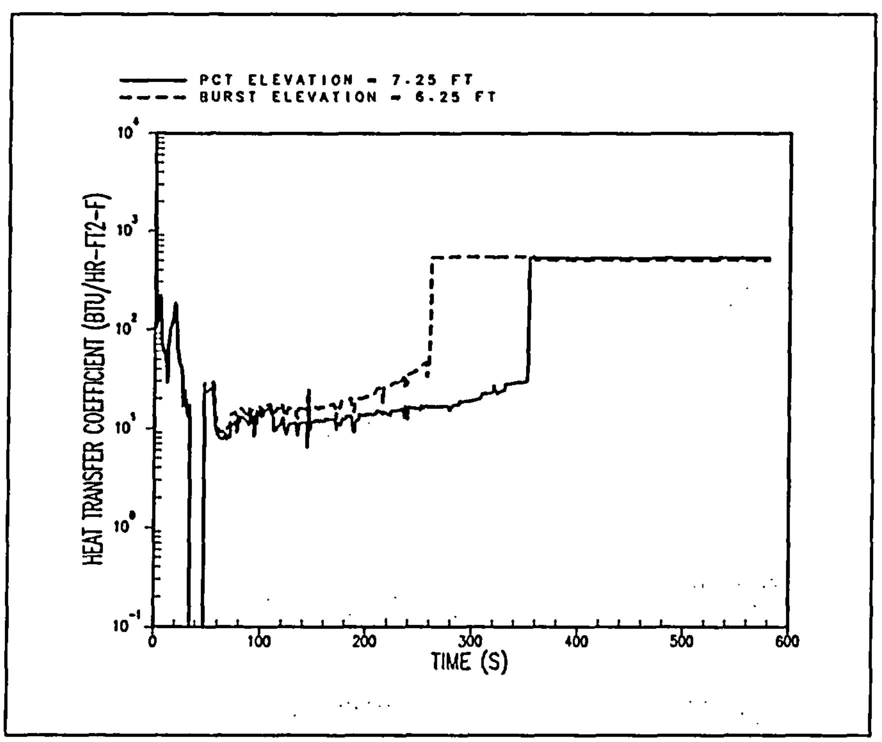

Figure 6.2.1-8 Cladding Surface Heat Transfer Coefficient at PCT and Burst Elevations

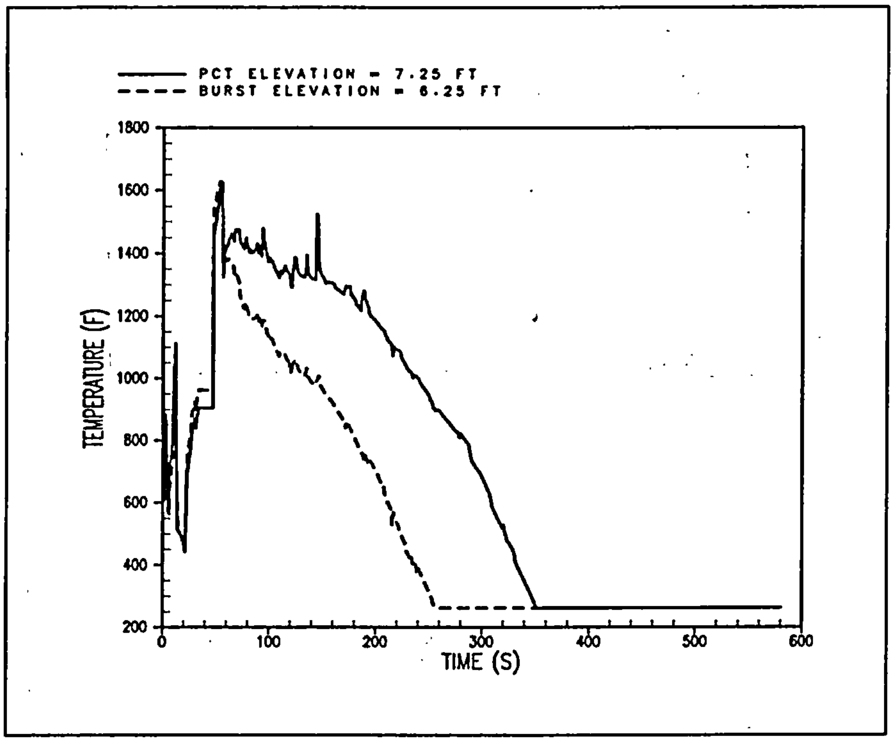

Figure 6.2.1-9 Vapor Temperature at PCT and Burst Elevations

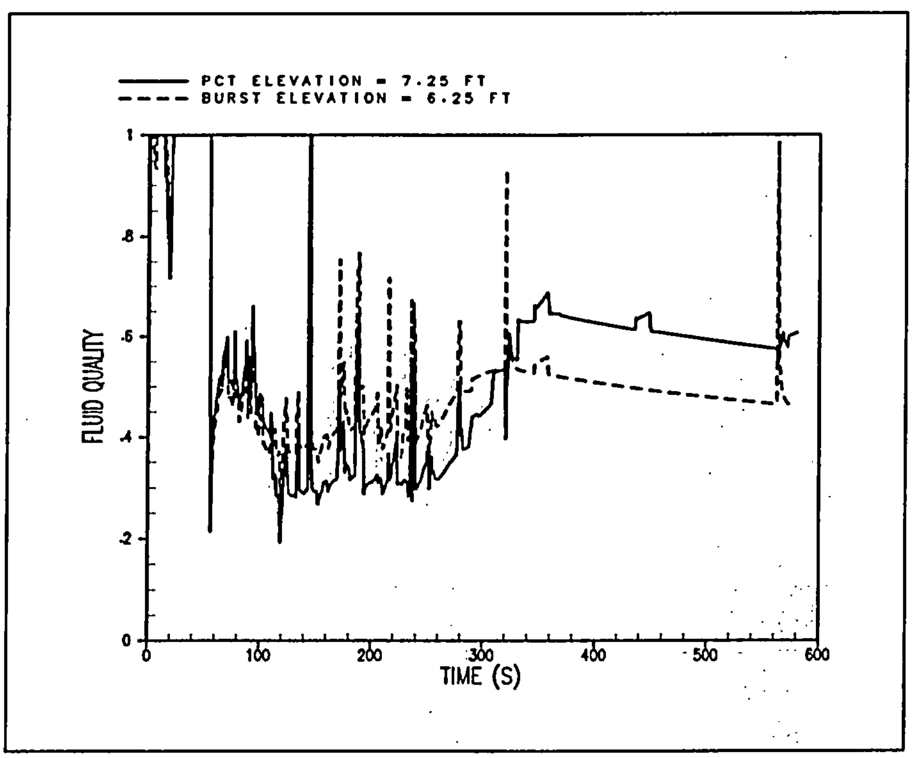

Figure 6.2.1-10. Fluid Quality at PCT and Burst Elevations

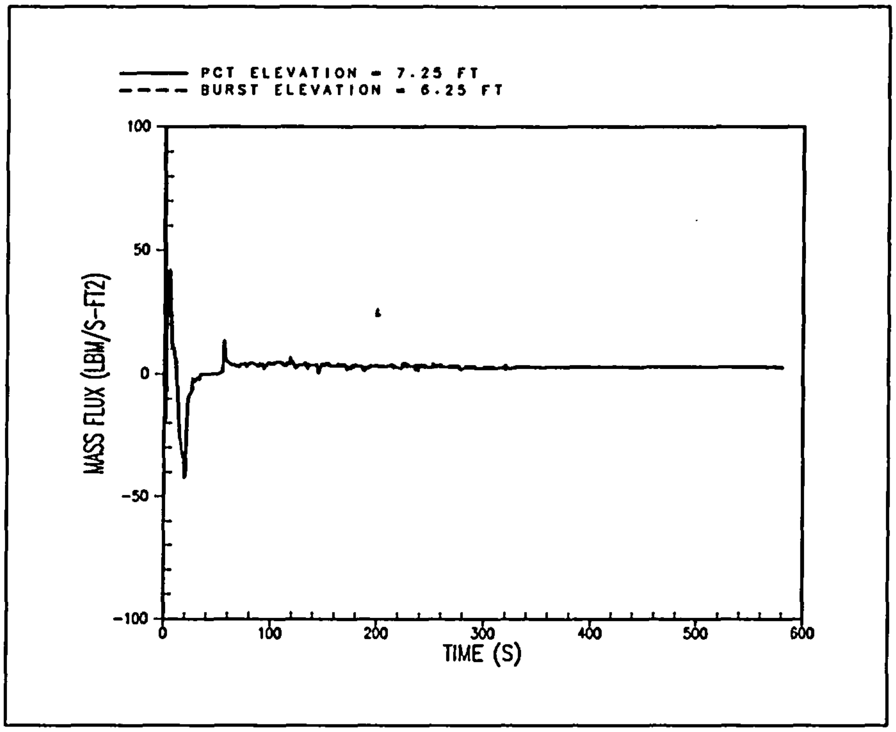

Figure 6.2.1-11 Fluid Mass Velocity at PCT and Burst Elevations

6.2.2 Small-Break LOCA

6.2.2.1 Introduction

The small-break LOCA (SBLOCA) analysis of record for Callaway was completed using the 1985 Westinghouse SBLOCA Evaluation Model with NOTRUMP (NOTRUMP-EM) (References 1 and 2), including changes to the methodology in Reference 3. The SBLOCA analysis was re-performed for the RSG Program to incorporate the RSG design, a full-power vessel average temperature (Tavg) range from 570.7°F to 588.4°F, and other changes identified through discussions between AmerenUE and Westinghouse. The analysis also modeled the appropriate plant-specific design features of Callaway such as an inverted top hat upper support plate, an upflow barrel-baffle region, and an upper head temperature equal to the cold leg temperature (Tcold). All prior 10 CFR 50.46 assessments were incorporated into the analysis, primarily through the use of corrected code versions and selection of input values.

The following sections provide an overview of the SBLOCA analysis methodology, assumptions and initial conditions, acceptance criteria, and results.

6.2.2.2 Analysis Methodology, Assumptions, and Initial Conditions

The SBLOCA methodology using NOTRUMP-EM was developed in accordance with the requirements of 10 CFR 50 Appendix K. This regulation was designed to produce a conservative prediction of the analysis results and includes various conservative modeling requirements such as the decay heat model (1971 ANS Infinite + 20%) and the zirconium-water reaction model (Baker-Just). Additional input assumptions and initial conditions for the SBLOCA analysis are found in Tables 6.2.2-1 through 6.2.2-3.

The main codes comprising NOTRUMP-EM are described in Table 1-2. The codes include NOTRUMP, which calculates the thermal-hydraulic transient; and SBLOCTA, which calculates the hot rod cladding temperature and oxidation transients for the SBLOCA analysis.

6.2.2.3 Acceptance Criteria

The acceptance criteria for the SBLOCA analysis are specified in 10 CFR 50.46, as follows:

1. The calculated maximum fuel element cladding temperature shall not exceed 2,200°F.

2. The calculated total oxidation of the cladding shall 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 shall 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. Calculated changes in core geometry shall be such that the core remains amenable to cooling.

5. After any calculated successful initial operation of the ECCS, the calculated core temperature shall be maintained at an acceptably low value and decay heat shall be removed for the extended period of time required by the long-lived radioactivity remaining in the core. (Note that this criterion is not addressed as part of the short-term SBLOCA analysis.)

6.2.2.4 Results

Tables 6.2.2-4 and 6.2.2-5 provide the NOTRUMP and SBLOCTA results for the SBLOCA analysis, respectively. The PCT is 1,043°F, and the maximum local oxidation is 0.02 percent. The core-wide hydrogen generation remains well below the 10 CFR 50.46 acceptance limit of 1 percent, and the core geometry remains amenable to cooling. The transient results for the limiting analysis case are provided in Figures 6.2.2-1 to 6.2.2-5.

6.2.2.5 Conclusions

The SBLOCA analysis results meet the pertinent acceptance criteria of 10 CFR 50.46. The PCT is less than 2,200°F; the maximum local oxidation is less than 17 percent; the core-wide hydrogen generation is less than 1 percent; and, the core geometry remains amenable to cooling.

6.2.2.6 References

1. WCAP-10079-P-A, "NOTRUMP - A Nodal Transient Small Break and General Network Code," P. E. Meyer, August 1985.

2. WCAP-10054-P-A, "Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code," N. Lee, et al., August 1985.

3. WCAP-10054-P-A, Addendum 2, Revision 1, "Addendum to the Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code: Safety Injection into the Broken Loop and COSI Condensation Model," C. M. Thompson, et al., July 1997.

Table 6.2.2-1 Input Assumptions and Initial Conditions		
A.	**Core Parameters**	
	Licensed Core Power	3565 MWt
	Calorimetric Uncertainty	2%
	Total Core Peaking Factor, F_Q	2.50
	Channel Enthalpy Rise Factor, $F_{\Delta H}$	1.65
	Axial Offset	+20%
	K(z) Limit	1.0 from 0 to 6 ft, 1.0 to 0.925 from 6 to 12 ft
B.	**Reactor Coolant System**	
	Thermal Design Flow	93,600 gpm/loop
	Nominal Vessel Average Temperature Range	570.7 / 588.4°F
	Pressurizer Pressure	2,250 psia
	Pressurizer Pressure Uncertainty	30 psi
C.	**Reactor Protection System**	
	Reactor Trip Setpoint	1860 psia
	Reactor Trip Signal Processing Time (Includes Rod Drop Time)	4.7 seconds
D.	**Emergency Feedwater (EFW) System**	
	Maximum EFW Temperature	120°F
	Minimum EFW Flow Rate	235 gpm/steam generator
	Initiation Signal	ESFAS
	EFW Delivery Delay Time	60 seconds
E.	**Steam Generators**	
	Steam Generator Tube Plugging	5%
	Main Feedwater (MFW) Isolation Signal	Low pressurizer pressure SI signal
	MFW Isolation Delay Time	2.0 seconds
	MFW Flow Coastdown Time	15.0 seconds
	Feedwater Temperature	390 – 446 °F
	Steam Generator Safety Valve Flow Rates	Table 6.2.2-2

Table 6.2.2-1 Input Assumptions and Initial Conditions (cont.)		
F.	**Safety Injection**	
	Limiting Single Failure	1 ESF Emergency Bus
	Maximum SI Water Temperature	100°F
	Low-Low Pressurizer Pressure Signal	1705 psia
	SI Delay Time	29 seconds
	Safety Injection Flow Rates	Table 6.2.2-3
G.	**Accumulators**	
	Water/Gas Temperature	120°F
	Range of Accumulator Water Volume	6,061 – 6,655 gal
	Minimum Cover Gas Pressure	602 psia
H.	**RWST Draindown Input**	
	Maximum Containment Spray Flow	6,330 gpm
	Minimum Usable RWST Volume	227,758 gal
	Maximum SI Water Temperature After Switchover to Cold Leg Recirculation Signal is Generated	186°F

Table 6.2.2-2 Steam Generator Safety Valve Flows Per Steam Generator

MSSV	Set Pressure (psig)	Uncertainty	Accumulation	Rated Flow at Full Open Pressure (lbm/hr)
1	1185	3.6	3	803,844
2	1197	3.6	3	803,844
3	1210	3.6	3	803,844
4	1222	3.6	3	803,844
5	1234	3.6	3	803,844

Table 6.2.2-3 Safety Injection Flows vs. Pressure Minimum Safeguards, Spill to RCS Pressure		
RCS Pressure (psig)	Injected Flow (gpm)	Spilled Flow (gpm)
0	738.2	253.6
100	716.3	246.2
200	694.2	238.5
300	670.2	230.2
400	645.8	221.9
500	620.1	213.2
600	594.0	204.1
700	567.0	194.9
800	539.2	185.3
900	509.1	175.0
1000	477.3	164.1
1100	443.6	152.6
1200	389.1	133.8
1300	343.5	118.2
1400	278.7	96.0
1500	160.8	55.7
1600	151.1	52.3
1700	141.3	48.9
1800	131.4	45.5
1900	121.2	42.0
2000	109.1	37.8
2100	96.4	33.4
2200	82.2	28.4
2300	65.7	22.8
2400	31.5	11.0
2500	0.0	0.0

Table 6.2.2-4 NOTRUMP Transient Results				
Event Time (sec)	**2 Inch**	**3 Inch**	**4 Inch**	**6 Inch**
Break Initiation	0	0	0	0
Reactor Trip Signal	85.8	21.3	12.3	7.42
S-Signal	97.0	32.0	22.1	14.8
Safety Injection Begins	126.0	61.0	51.1	43.8
Loop Seal Clearing[1]	1387	651	330	165
Core Uncovery	N/A	813	720	435
Accumulator Injection Begins	N/A	N/A	970	395
Core Recovery	N/A	>2332	>1364	>457
1. Loop seal clearing is defined as break vapor flow > 1 lb/s				

Table 6.2.2-5 Beginning of Life (BOL) Rod Heatup Results			
	3 Inch	**4 Inch**	**6 Inch**
Time in Life	BOL	BOL	BOL
PCT (°F)	949	1043	772
PCT Time (s)	1050.1	1048.8	168.4
PCT Elevation (ft)	11.00	11.25	10.75
Hot Rod Burst Time (s)	N/A	N/A	N/A
Hot Rod Burst Elevation (ft)	N/A	N/A	N/A
Max. Local ZrO_2 (%)	0.02	0.02	<0.01
Max. Local ZrO_2 Elev (ft)	11.00	11.25	11.00
Note: Since the 2-inch break case resulted in no core uncovery, there are no SBLOCTA rod heatup results for the 2-inch break case.			

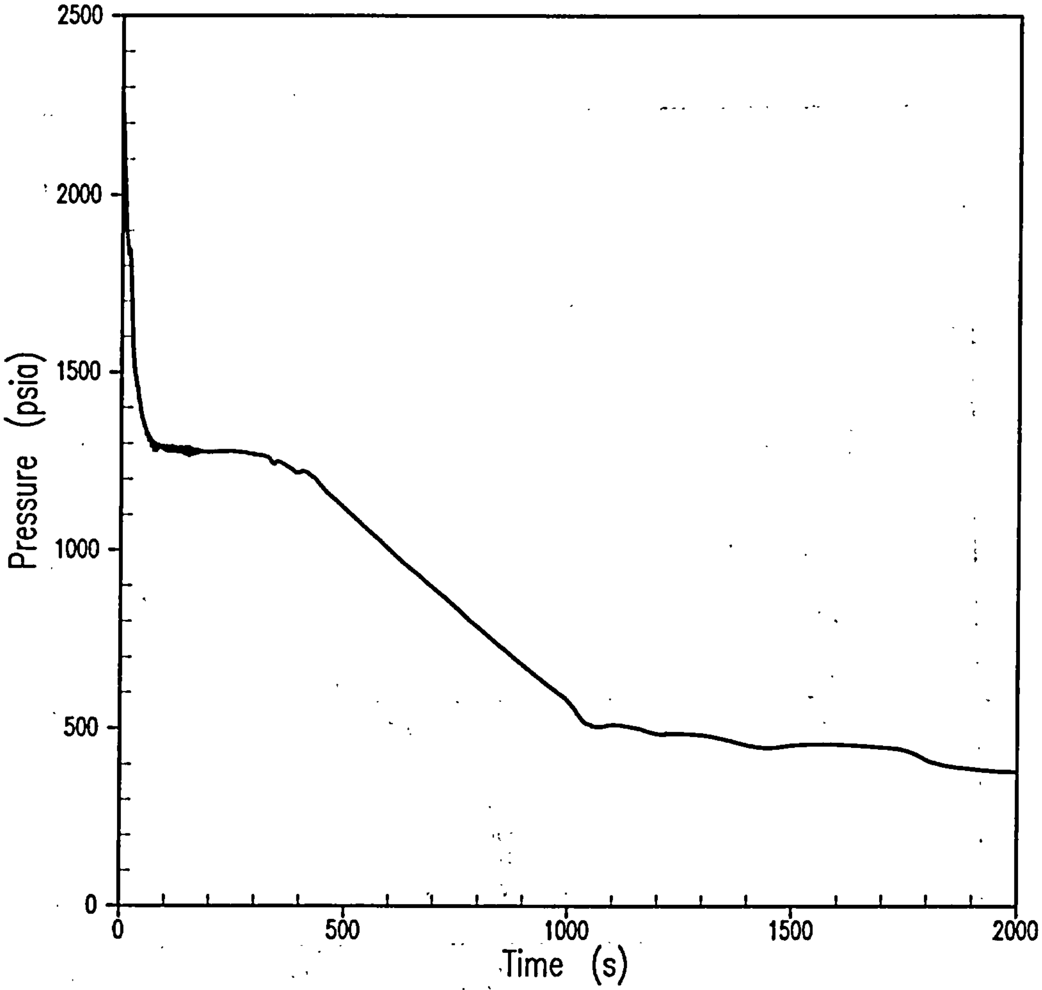

Figure 6.2.2-1 Pressurizer Pressure 4-Inch Break

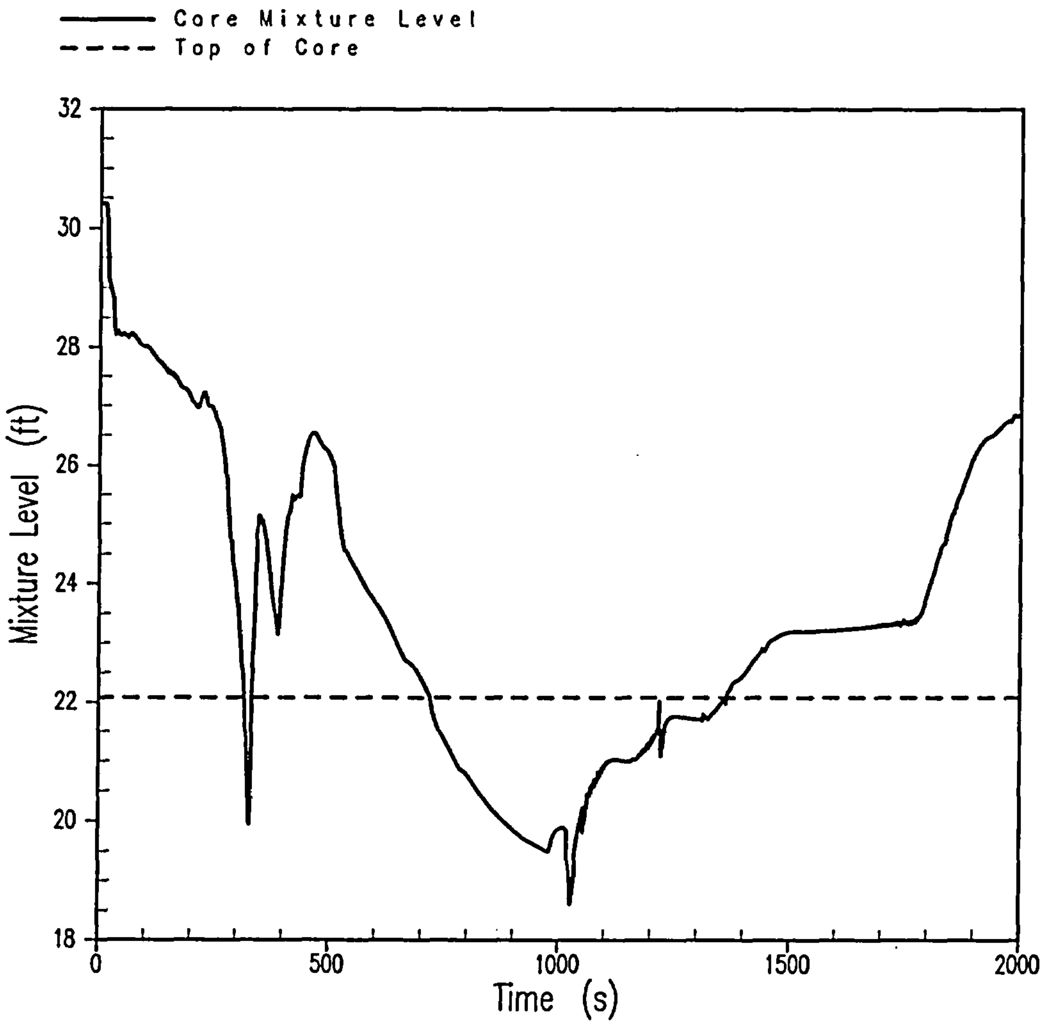

Figure 6.2.2-2 Core Mixture Level 4-Inch Break

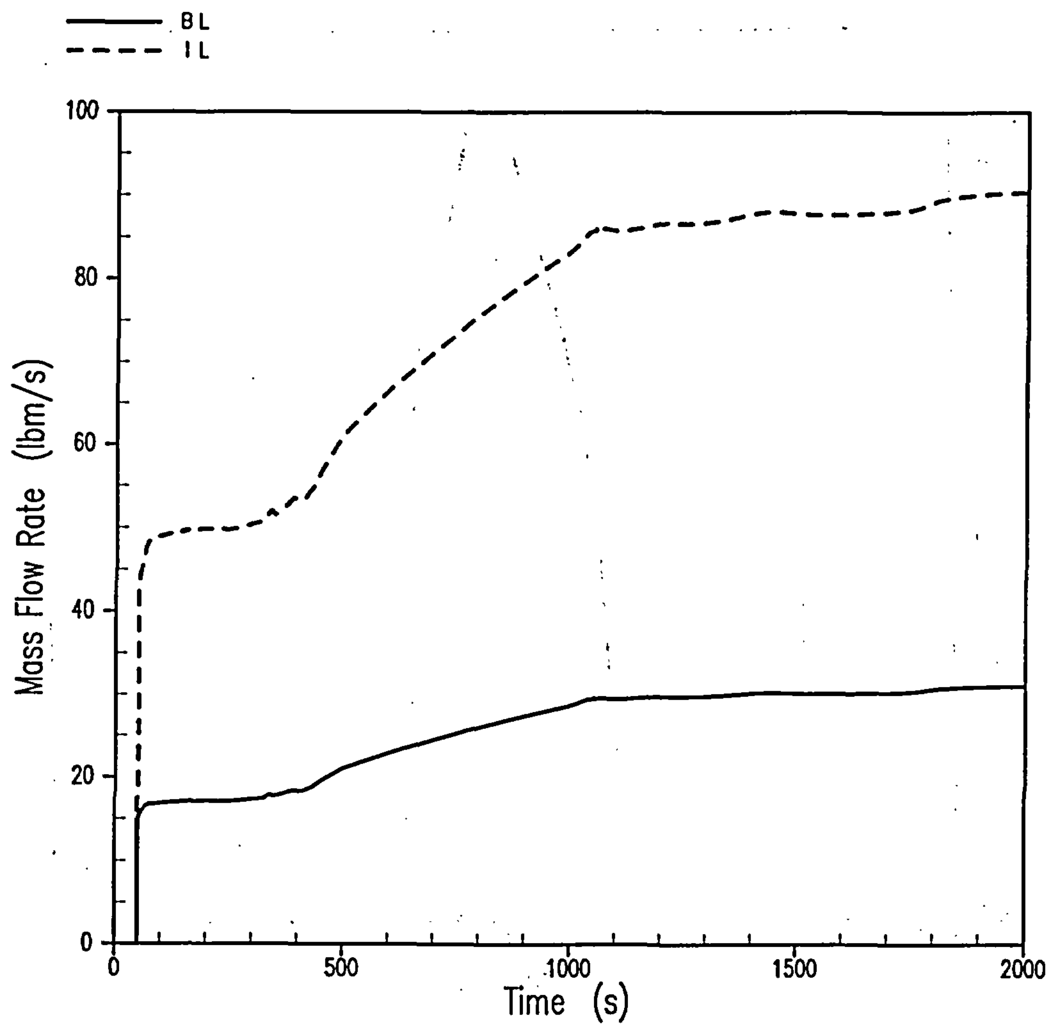

Figure 6.2.2-3 Broken Loop and Intact Loop Pumped SI Flow Rate 4-Inch Break

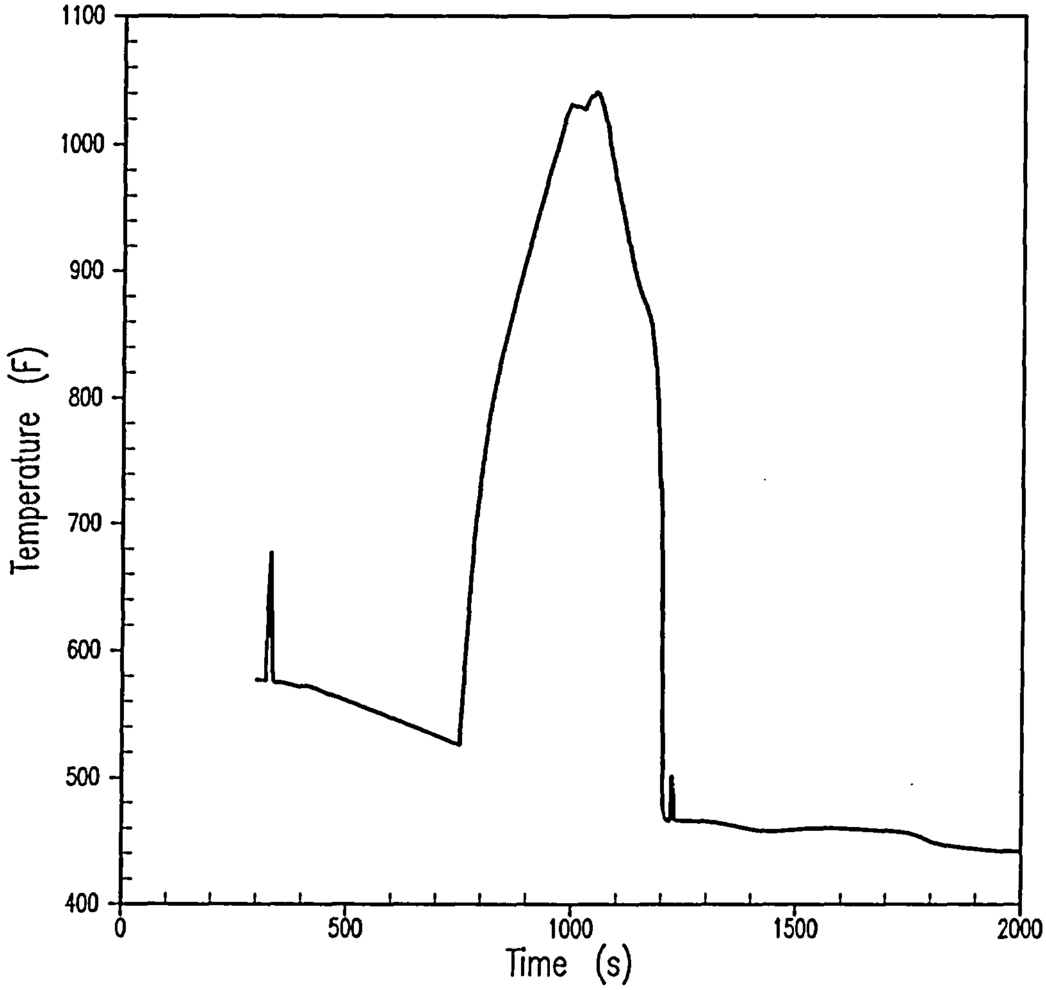

Figure 6.2.2-4 Peak Cladding Temperature at PCT Elevation 4-Inch Break

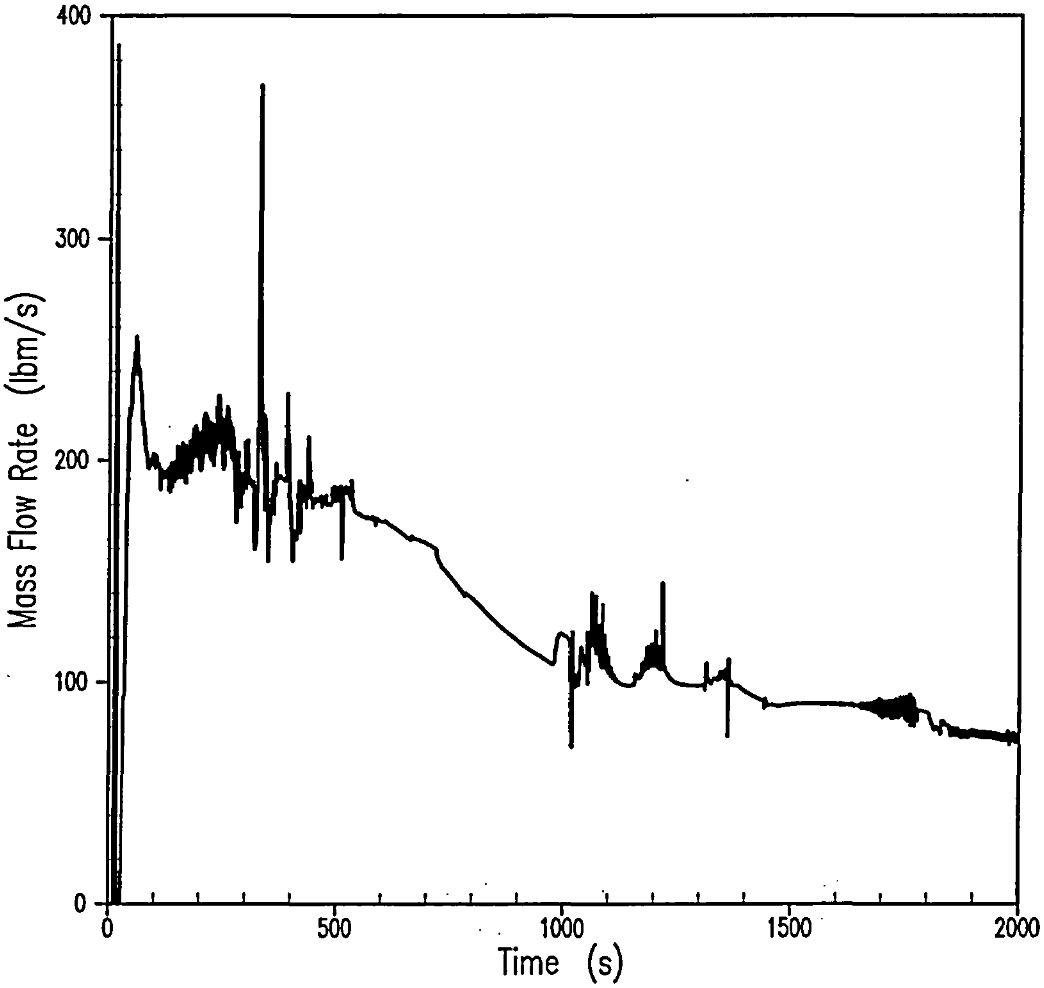

Figure 6.2.2-5 Core Exit Vapor Flow 4-Inch Break

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6.3 NON-LOCA ACCIDENT SAFETY ANALYSIS

The non-loss-of-coolant accident (non-LOCA) safety analyses discussed herein support the Callaway Replacement Steam Generator (RSG) Program and the transition to the Westinghouse RETRAN-based non-LOCA analysis methodology (Reference 1). The RETRAN code is used to perform transient calculations of select Final Safety Analysis Report (FSAR) Chapter 15 non-LOCA transients, and is coupled with the VIPRE code (Reference 5) for detailed departure from nucleate boiling ratio (DNBR) calculations. Currently, Callaway's non-LOCA analyses of record are based mostly on the LOFTRAN code, with the detailed DNBR calculations being performed with the THINC code. In addition, the Revised Thermal Design Procedure (RTDP) (Reference 2) departure from nucleate boiling (DNB) methodology has been applied as a replacement for the Improved Thermal Design Procedure (ITDP).

The key features of the Callaway RSG Program are the following:

- Framatome steam generators (Model 73/19T) – installation planned for Cycle 15 operation
- Full-power Tavg window between 588.4°F and 570.7°F
- Two full-power feedwater temperature values: 446°F and 390°F
- Maximum steam generator tube plugging level of 5 percent
- Although Callaway currently has the core thimble plugs installed, the RSG analyses cover conditions with either thimble plugs installed or removed
- Minimum measured flow (MMF) of 382,630 gpm, consistent with the Callaway Technical Specifications
- The trip time delay (TTD) logic will not be used after RSG implementation.
- An auctioneered-high Tavg signal is used for Tavg control in the rod control system.
- Other key assumptions remain unchanged from that assumed in the current analyses:
  - Core/NSSS power (3565 MWt/3579 MWt)
  - Thermal design flow (TDF) of 374,400 gpm
  - Fuel type (VANTAGE+ with IFMs, or simply, VANTAGE+)

6.3.0.1 Events Evaluated or Analyzed

Table 6.3-1 presents a list of all the non-LOCA events that have been either analyzed or evaluated as part of the Callaway RSG implementation and transition to the RETRAN code. The results of these transient evaluations and analyses demonstrate that all applicable safety analysis acceptance criteria continue to be met for the Callaway plant. Table 6.3-2 summarizes the results obtained for each of the non-LOCA events that were re-analyzed.

The analysis of the Startup of an Inactive Reactor Coolant Pump at an Incorrect Temperature (FSAR 15.4.4), historically performed for Westinghouse-designed plants, postulates the inadvertent start of an inactive reactor coolant pump (RCP) at a high power level (typically between 65-percent and 75-percent rated thermal power). Such a scenario results in the injection of cold water into the core, which would cause a reactivity insertion and subsequent power increase due to the moderator reactivity feedback effect. The Callaway Technical Specifications preclude operation of the plant in Modes 1 and 2 unless all 4 RCPs are in operation. The maximum initial core power level for this event, therefore, is 0 MWt. Under these conditions, there can be no significant reactivity insertion because the reactor coolant system (RCS) is initially at a near uniform temperature. Furthermore, the reactor will initially be subcritical by Technical Specification requirement. Therefore, there will be no increase in core power, and no automatic or manual protective action is required. Based on this, it has been determined that no analysis is required to show that the applicable acceptance criteria are met for this event. As such, this event has not been re-analyzed in support of the Callaway RSG Program. No further discussion pertaining to this event is presented in this report.

An updated Loss of Load/Turbine Trip analysis covering operation with inoperable main steam safety valves (MSSVs) was performed. A similar analysis had been performed for the Callaway Plant in 1999. The updated results obtained demonstrated that operation with inoperable MSSVs is acceptable provided the power is appropriately reduced. The maximum power levels as a function of operable MSSVs per loop are presented in Technical Specification 3.7.1. The current Technical Specification values were shown to remain valid, with the exception of the case with only 3 operable MSSVs per loop. In this scenario, part-power operation with the high neutron flux setpoint set at 45-percent rated thermal power (RTP) was found to be acceptable, instead of the previous value of 49-percent RTP.

6.3.0.2 Analysis Methodology

The transient-specific analysis methodologies applied in the Callaway RSG Program analyses are equivalent to those used in the LOFTRAN-based analyses discussed in the plant's FSAR, with the RETRAN code replacing LOFTRAN. Comparisons between LOFTRAN and RETRAN are discussed in detail in Reference 1. For additional information on the RETRAN code, refer to Section 6.3.0.6. Other minor differences in approach for less limiting events (for example, the excessive steam load increase transient) are discussed in the event-specific discussions found in subsections of Section 6.3 of this report (for example, Section 6.3.2, for the excessive steam load increase transient).

6.3.0.3 Fuel Design Mechanical Features

The fuel currently in use at Callaway is the Westinghouse 17×17 VANTAGE+ fuel. No changes in fuel features are assumed in the RSG Program. With respect to the non-LOCA transient analyses, the effects of fuel design mechanical features are accounted for in fuel-related input assumptions such as fuel and cladding dimensions, cladding material, fuel temperatures, and core bypass flow.

6.3.0.4 Peaking Factors, Kinetics Parameters

The power distribution is characterized by a nuclear enthalpy rise hot channel factor (radial peaking, $F^N_{\Delta H}$) of 1.59 (RTDP)/1.65 (non-RTDP) and a heat flux hot channel factor (total peaking, F_Q) of 2.50 (assumed rated thermal power F_Q) for the VANTAGE+ fuel. The $F^N_{\Delta H}$ is important for transients that are analyzed for DNB concerns (Table 6.3-2 identifies those events analyzed as part of the Callaway RSG Program that are analyzed for DNB concerns, as well as the DNB methodology used: RTDP or non-RTDP). As $F^N_{\Delta H}$ increases with decreasing power level, due to rod insertion, all transients analyzed for DNB concerns are assumed to begin with an $F^N_{\Delta H}$ consistent with the $F^N_{\Delta H}$ defined in the Technical Specifications Core Operating Limits Report (COLR) for the assumed nominal power level. The F_Q is important for transients that are analyzed for overpower concerns. Although the F_Q can be greater than the rated thermal power F_Q limit at part-power conditions (per the COLR limits), the intent of the F_Q limits is to ensure that the full-power hot-spot heat flux is not exceeded ($F_Q \times$ power equals the hot-spot heat flux). Consequently, an initial hot full-power (HFP) F_Q of 2.50 is supported by all non-LOCA transients that are limiting with respect to overpower, such as, rod cluster control assembly (RCCA) ejection and HFP steam line break.

The minimum shutdown margin at hot zero-power (HZP) conditions, with the most reactive RCCA fully withdrawn, is 1.3-percent $\Delta k/k$. This amount of shutdown margin is also assumed in calculating the initial boron concentrations for the inadvertent boron dilution event (not explicitly re-analyzed for the RSG Program, but discussed in Section 6.3.12) in Modes 3 and 4. The value assumed for this event for Mode 5 operation is 1.0-percent $\Delta k/k$.

6.3.0.5 Other Major Assumptions

For most transients that are analyzed for DNB concerns, the RTDP methodology (Reference 2) is employed. With this methodology, nominal values are assumed for the initial conditions of power, temperature, pressure, and flow, and the corresponding uncertainty allowances are accounted for statistically in defining the departure from DNBR safety analysis limit. Note that the *nominal* RCS flow assumed in RTDP transient analyses is the MMF of 382,630 gpm, and the difference between TDF and MMF is the flow uncertainty.

More specifically regarding the RTDP methodology, uncertainties in plant operating parameters, nuclear and thermal parameters, fuel fabrication parameters, computer codes, and DNB correlation predictions are combined statistically to obtain the overall DNB uncertainty factor. This factor is used to define the design limit DNB ratio (DNBR) (1.21 and 1.22 for thimble and typical cells, respectively). In other words, the design limit DNBR is a DNBR value that is greater than the DNBR correlation limit by an amount that accounts for the RTDP uncertainties. To provide DNBR margin to offset various penalties such as those due to rod bow and instrument bias, and to provide flexibility in design and operation of the plant, the design limit DNBR is conservatively increased to a value designated as the safety analysis limit DNBR, to which transient-specific DNBR values are compared. Using the WRB-2 DNB correlation, the DNBR safety analysis limits applicable to Callaway were determined to be 1.55 and 1.59 for the thimble and typical cells, respectively.

For transient analyses that are not DNB-limited, or for which RTDP is not employed, the initial conditions are obtained by applying the maximum, steady-state uncertainties to the nominal values in the most conservative direction. This is known as the Standard Thermal Design Procedure (STDP) or non-RTDP. In these analyses, the RCS flow is assumed to be equal to the TDF, and the following steady-state initial condition uncertainties are applied:

- The nuclear steam supply system (NSSS) power allowance for calorimetric uncertainty is ± 2 percent.
- The Tavg allowance for deadband and system measurement uncertainties is +4.3°F / -3.0°F.
- The pressurizer pressure allowance for steady-state fluctuations and measurement uncertainties is ± 30 psi.

In addition to the initial conditions uncertainties listed above, the following uncertainties are also modeled in cases where they yield more limiting results, regardless of whether RTDP or STDP DNB methodology is being employed:

- The pressurizer water level allowance for steady-state fluctuations and measurement uncertainties is ±5 percent of span.
- The steam generator water level allowance for steady-state fluctuations and measurement uncertainties is +6.2 percent / -7.9 percent of narrow range span (NRS).

Table 6.3-3 lists the non-LOCA initial condition assumptions used. Aside from those already listed above, other major assumptions considered in the non-LOCA transient analyses include:

a. Staggered lift setpoints are modeled for the main steam safety valves (MSSVs) using plant-specific Technical Specification setpoints. A +3-percent setpoint tolerance and a 5-psi ramp to account for accumulation to full-open conditions have been assumed in the MSSV model. Also considered in determining the MSSVs opening setpoints is the modeling of a loss coefficient that conservatively accounts for the pressure drop from the steam generator exit to the MSSV inlet; the value used yields a pressure drop of approximately 10 psi (generic, bounding value) under nominal operating conditions. Finally, an additional 15-psi pressure drop is assumed at full-open and full-flow conditions.

b. The pressurizer safety valves (PSVs) are modeled assuming a ±2-percent setpoint tolerance on a nominal setpoint of 2,475 psia. Additionally, when it is conservative to do so (that is, for peak RCS pressure concerns), the effects of the PSV loop seals are explicitly modeled. This includes the modeling of a 1.15-second delay to account for the purging of the loop seal volume, and an additional +1-percent setpoint uncertainty to account for the setpoint shift; both of these effects are discussed in detail in Reference 4.

c. Consistent with the Technical Specifications (COLR), a +5 pcm/°F moderator temperature coefficient (MTC) is assumed up to 70-percent power. This ramps to a value of zero MTC at 100-percent power.

d. The fission product contribution to decay heat assumed in the non-LOCA analyses is consistent with the American National Standards Institute / American Nuclear Society standard ANSI/ANS-5.1-1979 for decay heat power in light water reactors (Reference 7), including 2 standard deviations of conservatism.

6.3.0.6 Computer Codes Utilized

Summary descriptions of the computer codes used in the non-LOCA transient analyses performed in support of the Callaway RSG Program are provided in this subsection. Table 6.3-4 lists the computer codes used in each of these non-LOCA analyses.

RETRAN

RETRAN is used for studies of transient response of a pressurized water reactor (PWR) system to specified perturbations in process parameters. This code simulates a multi-loop system by a lumped parameter model containing the reactor vessel, hot and cold leg piping, RCPs, steam generators (tube and shell sides), main steam lines, and the pressurizer. The pressurizer heaters, spray, relief valves, and safety valves may also be modeled. RETRAN includes a point neutron kinetics model and reactivity effects of the moderator, fuel, boron, and control rods. The secondary side of the steam generator uses a detailed nodalization for the thermal transients. The reactor protection system (RPS) simulated in the code includes reactor trips on high neutron flux, overtemperature ΔT (OTΔT) and overpower ΔT (OPΔT), low RCS flow, high and low pressurizer pressure, high pressurizer level, and low-low steam generator water level. Control systems are also simulated including rod control and pressurizer pressure control. Parts of the safety injection system (SIS), including the accumulators, may also be modeled. RETRAN approximates the transient value of DNBR based on input from the core thermal safety limits.

The RETRAN licensing topical report, WCAP-14882-P-A (Reference 1), was approved by the Nuclear Regulatory Commission (NRC) via a Safety Evaluation Report (SER) from F. Akstulewicz (NRC) to H. Sepp (Westinghouse), dated February 11, 1999. The RETRAN SER identifies 3 conditions of acceptance, which are summarized below along with justifications for application to the Callaway RSG Program.

1. ***"The transients and accidents that Westinghouse proposes to analyze with RETRAN are listed in this SER (Table 1) and the NRC staff review of RETRAN usage by Westinghouse was limited to this set. Use of the code for other analytical purposes will require additional justification."***

   <u>Justification</u>

   The transients listed in Table 1 of the SER are:

   - Feedwater system malfunctions
   - Excessive increase in steam flow
   - Inadvertent opening of a steam generator relief or safety valve
   - Steam line break
   - Loss of external load/turbine trip

- Loss of offsite power
- Loss of normal feedwater flow
- Feedwater line rupture
- Loss of forced reactor coolant flow
- Locked RCP rotor/sheared shaft
- Control rod cluster withdrawal at power
- Dropped control rod cluster/dropped control bank
- Inadvertent increase in coolant inventory
- Inadvertent opening of a pressurizer relief or safety valve
- Steam generator tube rupture

The transients analyzed for Callaway using RETRAN are:

- Excessive heat removal due to feedwater system malfunctions (FSAR 15.1.1 and 15.1.2)

- HZP steam line break (FSAR 15.1.5)

- HFP steam line break

- Loss of external electrical load/turbine trip (FSAR 15.2.2 - 15.2.5)

- Loss of AC power to the station auxiliaries (FSAR 15.2.6)

- Loss of normal feedwater (FSAR 15.2.7)

- Feedwater system pipe break (FSAR 15.2.8)

- Partial and complete loss of forced reactor coolant flow (FSAR 15.3.1 and 15.3.2)

- RCP shaft seizure or break (FSAR 15.3.3 and 15.3.4)

- Uncontrolled RCCA withdrawal at power (FSAR 15.4.2)

- Inadvertent operation of the emergency core cooling system (ECCS) during power operation (FSAR 15.5.1)

- Inadvertent opening of a pressurizer safety or relief valve (FSAR 15.6.1)

As each transient analyzed for Callaway using RETRAN matches one of the transients listed in Table 1 of the SER, additional justification is not required.

2. ***"WCAP-14882 describes modeling of Westinghouse designed 4-, 3-, and 2-loop plants of the type that are currently operating. Use of the code to analyze other designs, including the Westinghouse AP600, will require additional justification."***

Justification

The Callaway Nuclear Plant consists of a single 4-loop Westinghouse-designed unit that was "currently operating" at the time the SER was written (February 11, 1999). Therefore, additional justification is not required.

3. *"Conservative safety analyses using RETRAN are dependent on the selection of conservative input. Acceptable methodology for developing plant-specific input is discussed in WCAP-14882 and in Reference 14 [WCAP-9272-P-A]* [which is Reference 11 in this Licensing Report]. *Licensing applications using RETRAN should include the source of and justification for the input data used in the analysis."*

Justification

The input data used in the RETRAN analyses performed by Westinghouse came from both AmerenUE and Westinghouse sources. For the most part, the input used is consistent with that used in the current LOFTRAN-based analyses of record. Assurance that the RETRAN input data is conservative for Callaway is provided via Westinghouse's use of transient-specific analysis guidance documents. Each analysis guidance document provides a description of the subject transient, a discussion of the plant protection systems that are expected to function, a list of the applicable event acceptance criteria, a list of the analysis input assumptions (such as, directions of conservatism for initial condition values), a detailed description of the transient model development method, and a discussion of the expected transient analysis results. Based on the analysis guidance documents, conservative, plant-specific input values were requested and collected from the responsible AmerenUE and Westinghouse sources. Consistent with the Westinghouse Reload Evaluation Methodology described in WCAP-9272-P-A (Reference 11), the fuel-related safety analysis input values used in the Callaway analyses were selected to conservatively bound the values expected in subsequent operating cycles.

LOFTRAN

Transient response studies of a PWR to specified perturbations in process parameters use the LOFTRAN computer code. This code simulates a multi-loop system by a model containing the reactor vessel, hot and cold leg piping, steam generators (tube and shell sides), the pressurizer and the pressurizer heaters, spray, relief valves, and safety valves. LOFTRAN also includes a point neutron kinetics model and reactivity effects of the moderator, fuel, boron, and rods. The secondary side of the steam generator uses a homogeneous, saturated mixture for the thermal transients. The code simulates the RPS, which includes reactor trips on high neutron flux, OTΔT and OPΔT, high and low pressurizer pressure, low RCS flow, low-low steam generator water level, and high pressurizer level. Control systems are also simulated including rod control, steam dump, and pressurizer pressure control. The SIS, including the accumulators, is also modeled. LOFTRAN can also approximate the transient value of DNBR based on input from the core thermal safety limits.

The LOFTRAN licensing topical report, WCAP-7907-P-A (Reference 12), was approved by the NRC via an SER from C. O. Thomas (NRC) to E. P. Rahe (Westinghouse), dated July 29, 1983. LOFTRAN is the primary code used in the current analyses of record for Callaway. LOFTRAN remains the system transient code for the analyses of the dropped rod transient (FSAR 15.4.3) and the anticipated transients without scram (ATWS, FSAR 15.8).

ANC

ANC (Reference 13) is an advanced nodal code capable of 2-dimensional and 3-dimensional neutronics calculations. ANC is the reference model for certain safety analysis calculations, power distributions, peaking factors, critical boron concentrations, control rod worths, reactivity coefficients, etc. In addition, 3-dimensional ANC validates 1-dimensional and 2-dimensional results and provides information about radial (x-y) peaking factors as a function of axial position. It can calculate discrete pin powers from nodal information as well.

VIPRE

The VIPRE computer program (Reference 5) performs thermal-hydraulic calculations. This code calculates coolant density, mass velocity, enthalpy, void fractions, static pressure, and DNBR distributions along flow channels within a reactor core.

6.3.0.7 Overtemperature- and Overpower-ΔT Reactor Trip Setpoints

The current OTΔT and OPΔT reactor trip setpoints have been reconfirmed using the methodology described in WCAP-8745-P-A (Reference 10). Conservative core thermal limits, developed using the RTDP methodology, are assumed. The core limits are applicable to the VANTAGE+ fuel, assuming the nominal core power of 3,565 MWt and nominal RCS pressure of 2,250 psia. The core thermal limits used to calculate the OTΔT/OPΔT setpoints are provided in Figure 6.3-1. The OTΔT and OPΔT trip setpoints are illustrated in Figure 6.3-2 and presented in Table 6.3-5. The adequacy of these setpoints is confirmed by showing that the DNB design basis is met in the analyses of those events that credit these functions for accident mitigation.

The boundaries of operation defined by the OTΔT and OPΔT trips are represented as "protection lines" in Figure 6.3-2. The protection lines are drawn to include all adverse instrumentation and setpoint errors so that under nominal conditions, a trip would occur well within the area bounded by these lines. These protection lines are based upon the safety analysis limit OTΔT and OPΔT setpoint values, which are essentially the Technical Specification nominal values with allowances for the adverse instrumentation behavior, setpoint errors, and acceptable drift between instrument calibrations. The utility of this diagram is in the fact that the limit imposed by any given DNBR can be represented as a line (ΔT versus T_{avg}). The DNB lines represent the locus of conditions for which the DNBR equals the safety analysis limit value (1.55 and 1.59 for the thimble and typical cells, respectively). All points below and to the left of a DNB line for a given pressure have a DNBR greater than the safety analysis limit DNBR value.

The area of permissible operation (power, temperature, and pressure) is bounded by the combination of the high neutron flux (fixed setpoint), high and low pressurizer pressure (fixed setpoints), OTΔT and OPΔT (variable setpoints) reactor trips, and the opening of the MSSVs (modeled as a single valve with an opening setpoint equal to the lowest MSSV setpoint plus 3-percent tolerance, a 20-psi pressure drop between the steam generators and the MSSVs (generic value used bounds plant-specific value, 15 psi), and a 5-psi ramp to allow the valve to reach its full relief capacity), which limits the maximum RCS average temperature. The adequacy of the OTΔT and OPΔT setpoints has been confirmed by demonstrating that the DNB design basis is met for those transients analyzed for DNB concerns.

As a result of the revalidation of the existing OTΔT and OPΔT setpoint constants, it has also been confirmed that the temperature ranges associated with the resistance temperature detector (RTD) instrumentation continue to be acceptable as they bound the wide range of expected transient conditions under which the OTΔT and OPΔT functions may be required to operate. Those temperature ranges are:

- Tcold: 510°F – 630°F
- Thot: 530°F – 650°F
- Tavg: 530°F – 630°F

6.3.0.8 RPS and ESFAS Functions Assumed in Analyses

Table 6.3-6 contains a list of the different RPS and engineered safeguards features actuation system (ESFAS) functions explicitly credited in the non-LOCA transient analyses that were re-analyzed in support of the Callaway RSG Program. The safety analysis setpoints, as well as the time delays associated with each of these functions, are also presented in Table 6.3-6.

Table 6.3-1 Transients Evaluated or Analyzed for the Callaway RSG Program			
Transient	**Licensing Report Section**	**FSAR Section**	**Notes**
Feedwater System Malfunctions that Result in a Decrease in Feedwater Temperature	6.3.1	15.1.1	A
Feedwater System Malfunctions that Result in an Increase in Feedwater Flow	6.3.1	15.1.2	A
Excessive Increase in Secondary Steam Flow	6.3.2	15.1.3	A
Inadvertent Opening of a Steam Generator Relief or Safety Valve	6.3.3	15.1.4	E
Steam System Piping Failure (HZP)	6.3.3	15.1.5	A
Steam System Piping Failure (HFP)	6.3.3A	--	A
Loss of External Electrical Load / Turbine Trip	6.3.4	15.2.2 - 15.2.5	A
Loss of Non-Emergency AC Power to the Station Auxiliaries	6.3.5	15.2.6	A
Loss of Normal Feedwater Flow	6.3.5	15.2.7	A
Feedwater System Pipe Break	6.3.6	15.2.8	A
Partial Loss of Forced Reactor Coolant Flow	6.3.7	15.3.1	A
Complete Loss of Forced Reactor Coolant Flow (Undervoltage and Underfrequency)	6.3.7	15.3.2	A
Reactor Coolant Pump Shaft Seizure (Locked Rotor) / Reactor Coolant Pump Shaft Break	6.3.8	15.3.3 and 15.3.4	A
Uncontrolled Rod Cluster Control Assembly Bank Withdrawal from a Subcritical or Low Power Startup Condition	6.3.9	15.4.1	E
Uncontrolled Rod Cluster Control Assembly Bank Withdrawal at Power	6.3.10	15.4.2	A
Rod Cluster Control Assembly Misoperation	6.3.11	15.4.3	E
Chemical and Volume Control System Malfunction that Results in a Decrease in the Boron Concentration in the Reactor Coolant	6.3.12	15.4.6	E
Spectrum of Rod Cluster Control Assembly Ejection Accidents	6.3.13	15.4.8	E
Inadvertent Operation of the ECCS During Power Operation	6.3.14	15.5.1	A
Chemical and Volume Control System Malfunction that Increases Reactor Coolant Inventory	6.3.15	15.5.2	E
Inadvertent Opening of a Pressurizer Safety or Relief Valve	6.3.16	15.6.1	A
Anticipated Transients Without Scram	6.3.17	15.8	A

Notes:
A = Complete Analysis
E = Evaluation

Table 6.3-2 Non-LOCA Analysis Limits and Results for Revised RSG Analyses				
			Analysis Result	
FSAR Section	Event Description	Result Parameter	Analysis Limit	Limiting Case
15.1.1	Feedwater System Malfunctions that Result in a Decrease in Feedwater Temperature	Minimum DNBR (RTDP, WRB-2)	1.55	1.733
15.1.2	Feedwater System Malfunctions that Result in an Increase in Feedwater Flow	Minimum DNBR (RTDP, WRB-2)	1.55	1.927
15.1.3	Excessive Increase in Secondary Steam Flow	Minimum DNBR (RTDP, WRB-2)	1.55	>1.55
15.1.5	Steam System Piping Failure (HZP)	Minimum DNBR (non-RTDP, W-3, thm/typ)	1.50	1.908 / 1.898
	Steam System Piping Failure (HFP)	Minimum DNBR (RTDP, WRB-2, thm/typ)	1.55 / 1.59	1.80 / 1.83
		Peak fuel centerline linear power (kW/ft)	22.46	21.19
15.2.2 - 15.2.5	Loss of External Electrical Load / Turbine Trip	Minimum DNBR (RTDP, WRB-2)	1.55	1.900
		Peak RCS pressure, psia	2748.5	2731.6
		Peak main steam system pressure, psia	1318.5	1294.0
15.2.6	Loss of Non-Emergency AC Power to the Station Auxiliaries	Peak pressurizer mixture volume (ft^3)	1800	1425
15.2.7	Loss of Normal Feedwater Flow	Peak pressurizer mixture volume (ft^3)	1800	1231
15.2.8	Feedwater System Pipe Break	Minimum margin to hot leg saturation (°F)	>0	41.6

Table 6.3-2 Non-LOCA Analysis Limits and Results for Revised RSG Analyses (cont.)

FSAR Section	Event Description	Result Parameter	Analysis Result	
			Analysis Limit	Limiting Case
15.3.1	Partial Loss of Forced Reactor Coolant Flow	Minimum DNBR (RTDP, WRB-2, thm/typ)	1.55 / 1.59	1.90 / 1.94
15.3.2	Complete Loss of Forced Reactor Coolant Flow - Undervoltage	Minimum DNBR (RTDP, WRB-2, thm/typ)	1.55 / 1.59	1.76 / 1.79
	Complete Loss of Forced Reactor Coolant Flow - Underfrequency	Minimum DNBR (RTDP, WRB-2, thm/typ)	1.55 / 1.59	1.78 / 1.80
15.3.3 and 15.3.4	Reactor Coolant Pump Shaft Seizure (Locked Rotor) / Reactor Coolant Pump Shaft Break	Peak RCS pressure, psia	2748.5	2559.5
		Peak cladding temperature (°F)	2700	1790
		Peak Zirconium-Water reaction (%)	16	0.30
		Rods-in-DNB (%)	5	< 5
15.4.2	Uncontrolled Rod Cluster Control Assembly Bank Withdrawal at Power	Minimum DNBR (RTDP, WRB-2)	1.55	1.572
		Maximum core average heat flux, % of nominal	118.52	117.12
		Peak main steam system pressure, psia	1318.5	1283.8
15.5.1	Inadvertent Operation of the ECCS During Power Operation	Minimum DNBR (RTDP, WRB-2)	1.55	2.335
		Water relief through the pressurizer safety valves	None	None
15.6.1	Inadvertent Opening of a Pressurizer Safety or Relief Valve	Minimum DNBR (RTDP, WRB-2)	1.55	1.859
15.8	Anticipated Transients Without Scram	Peak RCS pressure, psia	3215	3177

Table 6.3-3 Non-LOCA Plant Initial Condition Assumptions			
Parameter	**RTDP**	**Non-RTDP**	**Notes**
NSSS Power (MWt)	3579	3579 * 1.02	–
Nominal Total RCP Heat (MWt)	14	14 or 20	1
Maximum Full-Power Vessel Tavg (°F)	588.4	588.4 + 4.3, or 588.4 - 3.0	2
Minimum Full-Power Vessel Tavg (°F)	570.7	570.7 + 4.3, or 570.7 - 3.0	2
No-Load RCS Temperature (°F)	557.0	557.0	3
RCS Flow (gpm)	382,630 (MMF)	374,400 (TDF)	–
Pressurizer Pressure (psia)	2250	2250 ± 30	–
Steam Flow (lbm/hr)	see Note 4	see Note 4	–
Steam Pressure (psia)	see Note 4	see Note 4	–
Feedwater Temperature (°F)	390°F, or 446°F	390°F, or 446°F	5
Pressurizer Water Level (% span)	see Note 6	see Note 6	–
Steam Generator Water Level (% NRS)	51.3, 51.3 + 6.2, or 51.3 - 7.9	51.3, 51.3 + 6.2, or 51.3 - 7.9	7

Notes:

1. For cases where the long-term operation of the RCPs is conservative, a maximum RCP heat input (20 MWt) is modeled. In all cases, the nominal NSSS power of 3579 MWt is maintained.
2. A full-power RCS Tavg window between 570.7°F and 588.4°F is supported. Some analyses only use one full-power Tavg value either because a clear direction of conservatism exists, or because the full-power Tavg does not have a significant effect on the analysis results.
3. All analyses assume a programmed no-load RCS Tavg of 557°F. For the events initiated from a no-load condition (HZP steam line break), the use of the no-load temperature as the initial temperature bounds the case of startup operations being performed at a HZP temperature lower than 557°F. This is because the DNBR calculations are more limiting at the higher RCS temperature.
4. The initial steam flow rate and steam pressure depend on various other initial conditions. See Table 2-1 for samples values for these parameters. Note that the initial steam flow rate is not an input to the RETRAN model, but rather a calculated value within the code.
5. Full-power feedwater temperatures of 390°F and 446°F are supported. In cases where the feedwater temperature was judged to have an impact on the results, a sensitivity to this parameter was performed and the limiting case was presented.
6. The nominal pressurizer water level varies as a function of RCS full-power Tavg. For example, the nominal pressurizer water level program ramps linearly from 25% of span at the no-load RCS Tavg of 557°F (starting point of the level program is common for all full-power Tavg values) to 60% of span at the high nominal RCS Tavg value of 588.4°F. Similarly, it ramps from 25% of span at the no-load RCS temperature to 38% of span for the low nominal RCS full-power Tavg of 570.7°F. In general, the high limit of the level program is given by:

   High Limit = 1.2429 * FLTAVG - 671.3 (% span)

   where FLTAVG is equal to the measured auctioneered high full-power Tavg; 570.7°F ≤FLTAVG ≤588.4°F
7. The nominal steam generator water level modeled in the analyses performed in support of the Callaway RSG Program is a constant 51.3% NRS, regardless of the power level.

Table 6.3-4 Summary of Initial Conditions and Computer Codes Used in Revised RSG Analyses							
Accident	**Computer Codes Used**	**DNB Correlation**	**RTDP**	**Initial Power, %**	**Reactor Coolant Flow, gpm**	**Vessel Average Coolant Temp, °F**	**RCS Pressure, psia**
Feedwater System Malfunctions that Result in a Decrease in Feedwater Temperature	RETRAN	WRB-2	Yes	100	382,630	588.4	2250
Feedwater System Malfunctions that Result in an Increase in Feedwater Flow	RETRAN	WRB-2	Yes	100	382,630	588.4	2250
Steam System Piping Failure (HZP)	RETRAN ANC VIPRE	W-3	No	0	374,400	557	2250
Steam System Piping Failure (HFP)	RETRAN VIPRE	WRB-2	Yes	100	382,630	588.4	2250
Loss of External Electrical Load / Turbine Trip	RETRAN	N/A (pressure) WRB-2 (DNBR)	N/A (pressure) Yes (DNBR)	102 (pressure) 100 (DNBR)	374,400 (pressure) 382,630 (DNBR)	588.4 (DNB) 585.4 (pressure)	2250 (DNB) 2220 (pressure)
Loss of Non-Emergency AC Power to the Station Auxiliaries	RETRAN	N/A	N/A	102	374,400	567.7	2220
Loss of Normal Feedwater Flow	RETRAN	N/A	N/A	102	374,400	567.7	2220
Feedwater System Pipe Break	RETRAN	N/A	N/A	102	374,400	592.7	2220

Table 6.3-4 Summary of Initial Conditions and Computer Codes Used in Revised RSG Analyses (cont.)

Accident	Computer Codes Used	DNB Correlation	RTDP	Initial Power, %	Reactor Coolant Flow, gpm	Vessel Average Coolant Temp, °F	RCS Pressure, psia
Partial Loss of Forced Reactor Coolant Flow	RETRAN VIPRE	WRB-2	Yes	100	382,630	588.4	2250
Complete Loss of Forced Reactor Coolant Flow-Undervoltage	RETRAN VIPRE	WRB-2	Yes	100	382,630	588.4	2250
Complete Loss of Reactor Coolant Flow-Underfrequency	RETRAN VIPRE	WRB-2	Yes	100	382,630	588.4	2250
Reactor Coolant Pump Shaft Seizure (Locked Rotor) / Reactor Coolant Pump Shaft Break	RETRAN VIPRE	WRB-2 (DNBR) N/A (pressure & hot spot)	Yes (DNBR) N/A (pressure & hot spot)	100 (DNBR) 102 (pressure & hot spot)	382,630 (DNBR) 374,400 (pressure & hot spot)	588.4 (DNBR) 592.7 (pressure & hot spot)	2250 (DNBR) 2280 (pressure & hot spot)
Uncontrolled Rod Cluster Control Assembly Bank Withdrawal at Power	RETRAN	WRB-2	Yes	100 60 10	382,630	588.4 575.84 560.14	2250
Inadvertent Operation of the ECCS During Power Operation	RETRAN	WRB-2 (DNBR) N/A (pzr fill)	Yes (DNBR) N/A (pzr fill)	100 (DNBR) 102 (pzr fill)	382,630 (DNBR) 374,400 (prz fill)	588.4 (DNBR) 567.7 (pzr fill)	2250 (DNBR) 2220 (pzr fill)
Inadvertent Opening of a Pressurizer Safety or Relief Valve	RETRAN	WRB-2	Yes	100	382,630	588.4	2250

Table 6.3-5 Overtemperature and Overpower ΔT Setpoints	
Allowable Tavg range	570.7°F to 588.4°F
K_1 (safety analysis value)	1.29
K_1 (Technical Specification value)	AmerenUE Scope
K_2	0.0251/°F
K_3	0.00116/psi
K_4 (safety analysis value)	1.165
K_4 (Technical Specification value)	AmerenUE Scope
K_6	0.0015/°F[1]
T' and T"	570.7°F to 588.4°F[2]
P'	2250 psia
f(ΔI) Deadband	-23% ΔI to +10% ΔI[3]
f(ΔI) Negative Gain	- 3.25 %/%ΔI
f(ΔI) Positive Gain	+ 2.973 %/%ΔI
High Pressurizer Pressure Reactor Trip Setpoint	2435 psia
Low Pressurizer Pressure Reactor Trip Setpoint	1860 psia

(1) The value presented is for Tavg > T'; for Tavg ≤ T', K6 is equal to 0.0/°F.

(2) Value to be set equal to less than or equal to the full-power loop-specific indicated Tavg.

(3) The safety analysis deadband values above support an AmerenUE setpoint calculation that uses available DNBR margin to squeeze the deadband in by 2% RTP on both the positive and negative wings. See Table 7.1-3.

Table 6.3-6	Summary of RPS and ESFAS Functions Actuated in Revised RSG Analyses			
FSAR Section	**Event Description**	**RPS or ESFAS Signal(s) Actuated**	**Analysis Setpoint**	**Delay (sec)**
15.1.1	Feedwater System Malfunctions that Result in a Decrease in Feedwater Temperature	Reactor trip on OPΔT [1]	Variable (see Table 6.3-3)	2.0
15.1.2	Feedwater System Malfunctions that Result in an Increase in Feedwater Flow	Feedwater isolation on high-high steam generator water level Reactor trip on low-low steam generator water level	100% NRS 0% NRS	17.0 2.0
15.1.5	Steam System Piping Failure (HZP)	SI on compensated low steam line pressure Feedwater and steamline isolation on SI	473 psia (for all)	17.0 (for all)
	Steam System Piping Failure (HFP)	SI and reactor trip on compensated low steam line pressure Reactor trip on OPΔT [1]	473 psia Variable (see Table 6.3-3)	2.0 2.0
15.2.2 - 15.2.5	Loss of External Electrical Load / Turbine Trip	Reactor trip on OTΔT [1] Reactor trip on high pressurizer pressure	Variable (see Table 6.3-3) 2435 psia	2.0 1.0
15.2.6	Loss of Non-Emergency AC Power to the Station Auxiliaries	Reactor trip on low-low steam generator water level Feedwater isolation on low-low steam generator water level Auxiliary feedwater initiated on low-low steam generator water level	0% NRS (for all)	2.0 17.0 60.0
15.2.7	Loss of Normal Feedwater Flow	Reactor trip on low-low steam generator water level Feedwater isolation on low-low steam generator water level Auxiliary feedwater initiated on low-low steam generator water level	0% NRS (for all)	2.0 17.0 60.0

Table 6.3-6 Summary of RPS and ESFAS Functions Actuated in Revised RSG Analyses (cont.)

FSAR Section	Event Description	RPS or ESFAS Signal(s) Actuated	Analysis Setpoint	Delay (sec)
15.2.8	Feedwater System Pipe Break	SI on compensated low steam line pressure [2] Reactor trip on low-low steam generator water level Feedwater isolation on low-low steam generator water level Auxiliary feedwater initiated on low-low steam generator water level	473 psia 0% NRS 0% NRS 0% NRS	27.0 [3] 2.0 17.0 60.0
15.3.1	Partial Loss of Forced Reactor Coolant Flow	Reactor trip on low RCS flow	87% loop flow	1.0
15.3.2	Complete Loss of Forced Reactor Coolant Flow - Undervoltage	Reactor trip on RCP undervoltage	N/A [4]	1.5 [4]
	Complete Loss of Forced Reactor Coolant Flow - Underfrequency	Reactor trip on RCP underfrequency	57 Hz [5]	0.6 [5]
15.3.3 and 15.3.4	Reactor Coolant Pump Shaft Seizure (Locked Rotor) / Reactor Coolant Pump Shaft Break	Reactor trip on low RCS flow	87% loop flow	1.0
15.4.2	Uncontrolled Rod Cluster Control Assembly Bank Withdrawal at Power	Power-range high neutron flux reactor trip (high setting) Reactor trip on OTΔT [1]	118% RTP Variable (see Table 6.3-3)	0.5 2.0
15.5.1	Inadvertent Operation of the ECCS During Power Operation	Reactor trip on low pressurizer pressure	1860 psia	2.0
15.6.1	Inadvertent Opening of a Pressurizer Safety or Relief Valve	Reactor trip on OTΔT [1]	Variable (see Table 6.3-3)	2.0

Table 6.3-6 Summary of RPS and ESFAS Functions Actuated in Revised RSG Analyses (cont.)

Notes:

(1) The modeling of the OTΔT and OPΔT reactor trips include a time constant (first order lag) of 6.0 seconds for the measurement of the vessel Tavg and ΔT. This lag accounts for the response of the RTDs, the RTD electronic filter (if any), the RTD bypass piping fluid transport delay, and the RTD bypass piping heatup thermal lag. In addition, a straight delay of 2.0 seconds is assumed which accounts for electronics delay, reactor trip breakers opening, and RCCA gripper release.

(2) The analysis of Section 6.3.6 models the SI system as described above, however, assumes no SI flow for added conservatism.

(3) The value presented is for cases where offsite power is available. The total delay for cases where offsite power is lost is 39 seconds.

(4) The analysis does not explicitly model a specific setpoint for this RPS function. Instead, it sets the time of rod motion at 1.5 seconds after the RCPs begin to coast down. The Technical Specification undervoltage setpoint is 68% of nominal, or 9384 V_{ac}. The basis for the undervoltage trip setpoint is as follows: Pump pullout for a standard Westinghouse pump at nominal flow conditions occurs at approximately 65% of nominal voltage. The undervoltage trip setpoint is set slightly above the pullout voltage to trip the reactor before pump pullout occurs. The major concern for avoiding pullout lies in not knowing how the voltage decays, and therefore, not being able to predict the pump speed if the pump does pull out. The setpoint is typically not set higher than 70% of nominal to preclude spurious trips due to voltage dips that result from normal operation.

(5) Frequency decay assumed to begin at time zero. The underfrequency decay trip setpoint is reached in 0.6 seconds based on a nominal grid frequency of 60 Hz, a trip setpoint of 57 Hz, and a decay rate of 5 Hz/second [(60-57)/5 = 0.6 seconds]. In addition, a 0.6-second delay time is applied after the underfrequency trip setpoint is reached. The underfrequency delay time accounts for underfrequency trip circuitry, trip breaker opening time, and RCCA gripper release time.

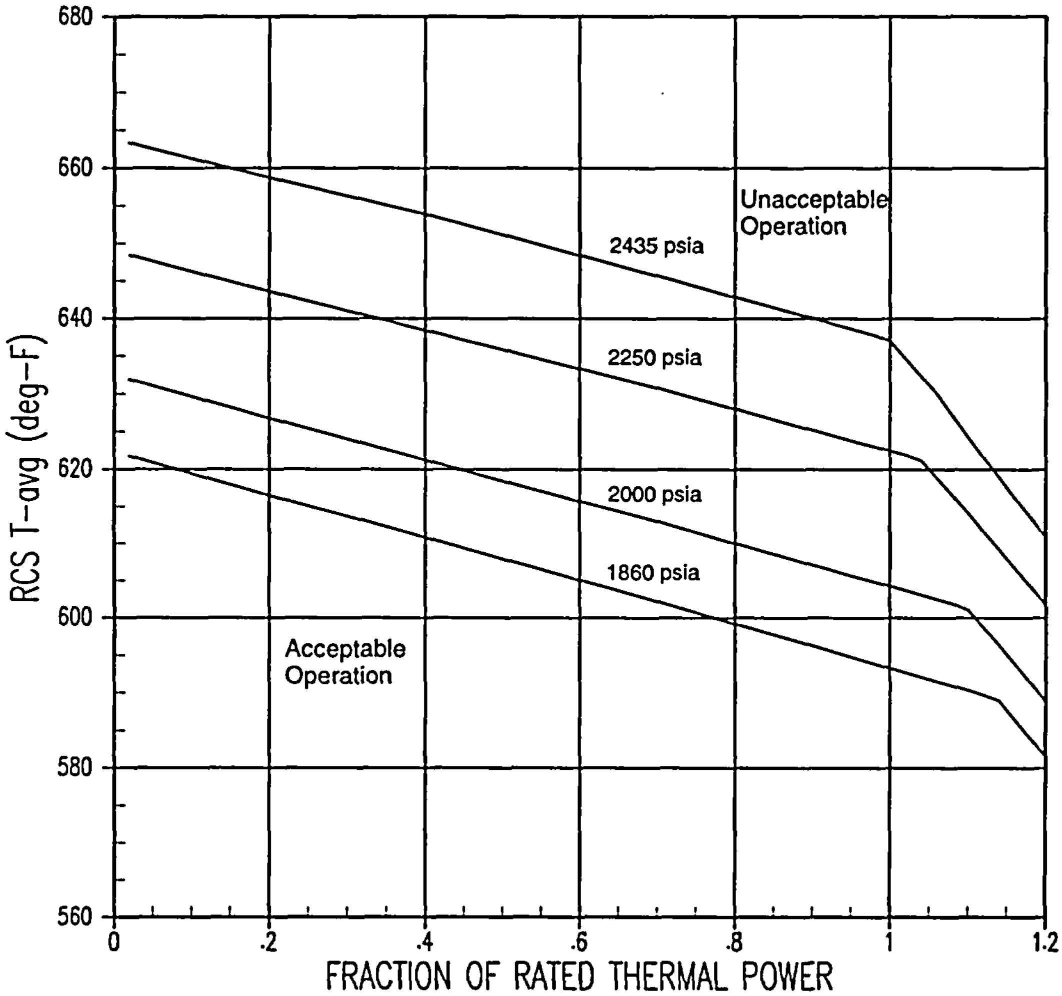


Figure 6.3-1 Reactor Core Safety Limits

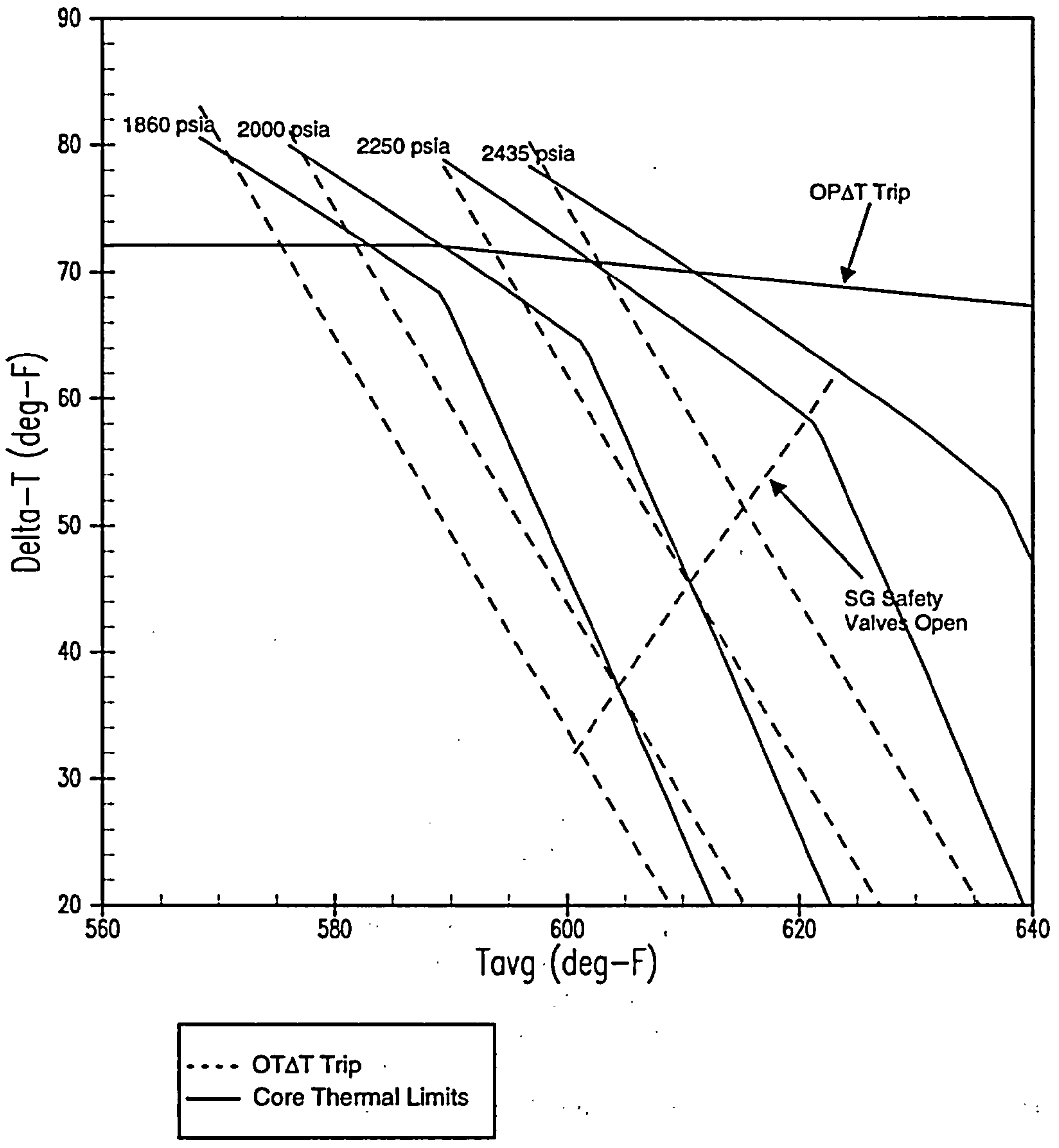


Figure 6.3-2 Illustration of Overtemperature and Overpower ΔT Protection

6.3.1 Excessive Heat Removal Due to Feedwater System Malfunctions (FSAR Sections 15.1.1 and 15.1.2)

A change in steam generator feedwater conditions that results in an increase in feedwater flow or a decrease in feedwater temperature could result in excessive heat removal from the plant primary coolant system. Such changes in feedwater flow or feedwater temperature are a result of a failure of a feedwater control valve or feedwater bypass valve, failure in the feedwater control system, or operator error.

The occurrence of these failures that result in an excessive heat removal from the plant primary coolant system cause the primary-side temperature and pressure to decrease significantly. The existence of a negative moderator and fuel temperature reactivity coefficients, and the actions initiated by the reactor rod control system can cause core reactivity to rise, as the primary-side temperature decreases. In the absence of the reactor protection system (RPS) reactor trip or other protective action, this increase in core power, coupled with the decrease in primary-side pressure, can challenge the core thermal limits.

6.3.1.1 Accident Description

Feedwater Temperature Reduction

An extreme example of excessive heat removal from the reactor coolant system (RCS) is the transient associated with the accidental opening of the feedwater bypass valve, which diverts flow around the low-pressure feedwater heaters. The function of this valve is to maintain net positive suction head on the main feedwater pump in the event that the heater drain pump flow is lost (such as, following a large-load reduction). In the event of an accidental opening of the feedwater bypass valve, there is a sudden reduction in feedwater inlet temperature to the steam generators. This increased subcooling would create a greater load demand on the RCS due to the increased heat transfer in the steam generators.

With the plant at no-load conditions, the addition of cold feedwater may cause a decrease in RCS temperature and thus a reactivity insertion due to the effects of the negative moderator temperature coefficient. However, the rate of energy change is reduced as load and feedwater flow decrease, so that the transient is less severe than the full-power case.

The net effect on the RCS due to a reduction in feedwater temperature is similar to the effect of increasing secondary steam flow; that is, the reactor will reach a new equilibrium condition at a power level corresponding to the new steam generator ΔT. The overpower/overtemperature protection (high neutron flux, OTΔT, and OPΔT trips) prevent any power increase that could lead to a departure from nucleate boiling ratio (DNBR) lower than the safety analysis limit value.

Feedwater Flow Increase

Another example of excessive heat removal from the RCS is a common-mode failure in the feedwater control system that leads to the accidental opening of the feedwater regulating valves to one steam generator.

Accidental opening of the feedwater regulating valves results in an increase of feedwater flow to one steam generator, causing excessive heat removal from the RCS. At power, excess feedwater flow causes a greater load demand on the primary side due to increased subcooling in the steam generator. With the plant at zero-power conditions, the addition of relatively cold feedwater may cause a decrease in primary-side temperature, and, therefore, a reactivity insertion due to the effects of the negative moderator temperature coefficient. The resultant decrease in the average temperature of the core causes an increase in core power due to moderator and control system feedback. However, for a feedwater flow increase of less than 150 percent, the hot-zero-power (HZP) case is considered to be less limiting than that caused by the occurrence of a feedwater malfunction event from full-power condition. As such, only the more limiting hot-full-power (HFP) scenario is explicitly analyzed.

Continuous addition of cold feedwater after a reactor trip is prevented since the reduction of RCS temperature, pressure, and pressurizer level leads to the actuation of safety injection (SI) on low pressurizer pressure. The SI signal trips the main feedwater pumps, closes the feedwater pump discharge valves, and closes the main feedwater control valves.

6.3.1.2 Method of Analysis

The feedwater malfunction analysis is performed to demonstrate that the departure from nucleate boiling (DNB) design basis is satisfied. This is accomplished by showing that the calculated minimum DNBR is greater than the safety analysis limit DNBR. The overall analysis process is described as follows.

The feedwater system malfunction transient is analyzed using the RETRAN code. The RETRAN computer code is described in detail in Section 6.3.0.6 of this report. The results from the RETRAN computer code are used to determine if the DNBR safety analysis limits for the excessive heat removal due to feedwater malfunction event are met.

Feedwater Temperature Reduction

This transient is initiated by a feedwater temperature reduction at the steam generators' inlets following spurious opening of the low-pressure heater bypass valve (more limiting than a failure of the high-pressure feedwater heater bypass valve). This transient is analyzed at full-power conditions (with auto and manual rod control). The following assumptions are made:

1. The plant is operating at full-power conditions with the initial reactor power, pressure, and RCS average temperatures assumed to be at the nominal values.

2. Uncertainties in initial conditions are included in the DNBR limit calculated using the Revised Thermal Design Procedure (RTDP) methodology (Reference 2), where applicable.

3. Feedwater temperature control is assumed to malfunction resulting in a step decrease to 280°F from the nominal high feedwater temperature value (446°F).

4. The heat capacity of the RCS metal and steam generator shell are ignored, thereby maximizing the temperature reduction of the RCS coolant.

5. The feedwater flow into the steam generators is terminated by an SI signal, following a low-pressurizer pressure signal, which closes all feedwater isolation valves. The main feedwater pumps would also be tripped and their discharge valves would be closed.

Normal reactor control systems are not required to function. The OPΔT protection function will trip the reactor. No single active failure will prevent operation of the RPS.

Feedwater Flow Increase

Feedwater system failures including the accidental opening of the feedwater regulating valves have the potential of allowing increased feedwater flow to one steam generator that will result in excessive heat removal from the RCS. Therefore, it is assumed that the feedwater control valves fail in the fully open position allowing the maximum feedwater flow to both steam generators. Cases with and without automatic rod control initiated at HFP conditions were considered. The HZP conditions are bounded by conditions analyzed at HFP.

The following assumptions are made for the analysis of the feedwater malfunction event involving the accidental opening of the feedwater regulating valves:

1. The plant is operating at full-power conditions with the initial reactor power, pressure, and RCS average temperatures assumed to be at the nominal values.

2. Uncertainties in initial conditions are included in the DNBR limit calculated using the RTDP methodology (Reference 2), where applicable.

3. The feedwater temperature window of 390° to 446°F is consistent with normal plant conditions. Cases with the lowest and the highest temperature have been considered.

4. The excessive feedwater flow event assumes accidental opening of the feedwater control valves with the reactor at full power with automatic and manual rod. The feedwater flow malfunction results in a step increase to 190 percent of the nominal full-power feedwater flow to one steam generator.

5. Maximum (end-of-life) reactivity feedback conditions with a minimum Doppler-only power defect is conservatively assumed.

6. The heat capacity of the RCS metal and steam generator shell are ignored, thereby maximizing the temperature reduction of the RCS coolant.

7. The feedwater flow resulting from a fully open control valve leads to the high-high steam generator water level signal that closes all main feedwater control and feedwater control-bypass valves, trips the main feedwater pumps and closes all feedwater pump discharge valves. No credit is taken for turbine trip on high-high steam generator water level signal. A reactor trip on low-low steam generator water level (following feedwater isolation (FWI) on high-high steam generator water level) will terminate the transient. In order to delay the low-low steam generator

water level setpoint from being reached, it is conservative to model maximum auxiliary feedwater (AFW) as soon as the high-high signal is generated.

Since turbine trip (and subsequent reactor trip) on a high-high steam generator water level signal is not credited, succeeding RPS functions following the high-high steam generator water level signal (that is, OTΔT, OPΔT, high neutron flux, or low-low steam generator water level), will initiate a reactor trip followed by a turbine trip.

6.3.1.3 Results

Feedwater Temperature Reduction

The results of the analyses demonstrate that a feedwater temperature reduction meets the applicable DNBR acceptance criterion; the minimum DNBR is maintained above the safety analysis limit value of 1.55.

The most limiting case is the temperature reduction from a full-power initial condition with manual rod control. This case gives the largest reactivity feedback and results in the greatest power increase. The reactor is tripped by an OPΔT signal. This causes a turbine trip 2 seconds after reactor trip. Minimum DNBR occurs at about this time. The addition of the cooler feedwater is terminated once the SI (following a low-pressurizer pressure) causes all feedwater isolation valves to be automatically closed.

Table 6.3.1-1 shows the time sequence of events for the HFP temperature reduction transient analyzed at full-power initial conditions assuming manual rod control. Figures 6.3.1-1 through 6.3.1-5 show transient responses for various system parameters during a temperature reduction initiated from HFP conditions assuming manual rod control..

Feedwater Flow Increase

The results of the analyses demonstrate that a feedwater flow increase meet the applicable DNBR acceptance criterion; the minimum DNBR is maintained above the safety analysis limit value of 1.55.

The most limiting case is the excessive feedwater flow from a full-power initial condition at the lowest feedwater temperature with automatic rod control. This case gives the largest reactivity feedback and results in the greatest power increase. A feedwater isolation occurs when the steam generator water level in either steam generator reaches the high-high water level setpoint. A reactor trip is actuated when the steam generator water level reaches the low-low level. Assuming the reactor to be in manual rod control results in a slightly less severe transient. The rod control system is not required to function for this event. However, assuming that the rod control system is operable yields a slightly more limiting transient.

For each excessive feedwater flow case, continuous addition of cold feedwater is prevented by automatic closure of all feedwater control valves, closure of all feedwater bypass valves, a trip of the feedwater pumps, and a turbine trip on high-high steam generator water level. In addition, the feedwater discharge isolation valves will automatically close upon receipt of the feedwater pump trip signal.

Table 6.3.1-2 shows the time sequence of events for the HFP feedwater malfunction transients analyzed at full-power initial conditions at the minimum FW temperature and assuming automatic rod control. Figures 6.3.1-6 through 6.3.1-10 show transient responses for various system parameters during a feedwater system malfunction initiated from HFP conditions assuming the lowest feedwater temperature with automatic rod control.

6.3.1.4 Conclusions

Feedwater system malfunction transients involving a reduction in feedwater temperature or an increase in feedwater flow rate have been analyzed. These transients show an increase in reactor power due to the excessive heat removal in the steam generators. Based on results presented on Tables 6.3.1-1 and 6.3.1-2, the applicable acceptance criteria for the feedwater malfunction transients have been met. Analyses of the accidental opening of the feedwater regulating valves were performed from a full-power initial condition with and without automatic rod control, and for feedwater temperature covering the feedwater temperature window (390° to 446°F). It has been demonstrated that considerable margin to the safety analysis acceptance criteria exists throughout the transient. Therefore, the DNB design basis is satisfied. Hence, no fuel damage is predicted.

Table 6.3.1-1 Time Sequence of Events for Temperature Reduction Event at Full Power	
Event	**Time (Seconds)**
Injection of Colder Water Starts	0.0
OPΔT Signal Reached	30.7
Reactor Trip on OPΔT Occurs	32.7
Minimum DNBR Reached	33.1
Turbine Trip on Reactor Trip Occurs	34.7
SI Setpoint on Low Pressurizer Pressure Reached	79.2
Feedwater Isolation Valves Fully Closed on SI	96.2
Results	
Minimum DNBR	1.733
Safety Analysis Limit DNBR	1.55

Table 6.3.1-2 Time Sequence of Events for Feedwater Flow Increase Event at Full Power	
Event	**Time (Seconds)**
Main Feedwater Control Valves Fail Full Open	0.0
High-High Steam Generator Water Level Trip Setpoint is Reached	44.7
AFW Actuation on High-High Steam Generator Water Level	44.7
Feedwater Isolation Valves Fully Closed on High-High Steam Generator Water Level	61.7
Minimum DNBR Occurs	66.4
Low-Low Steam Generator Water Level Setpoint is Reached	106.5
Reactor Trip on Low-Low Steam Generator Water Level Occurs	108.5
Turbine Trip on Reactor Trip Occurs	110.5
Results	
Minimum DNBR	1.927
Safety Analysis Limit DNBR	1.55

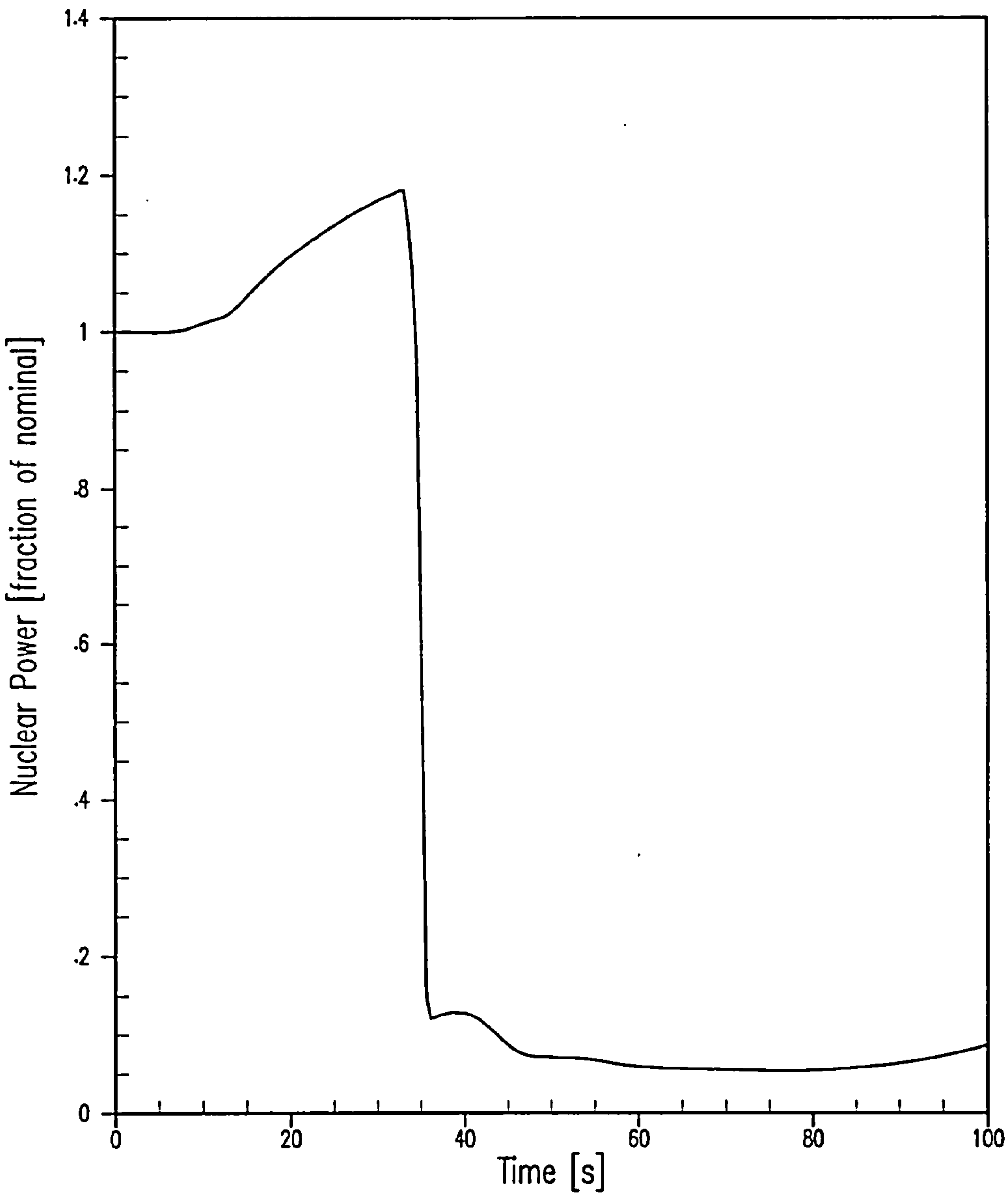


Figure 6.3.1-1 **Feedwater Temperature Reduction at Full Power – Reactor Power versus Time**

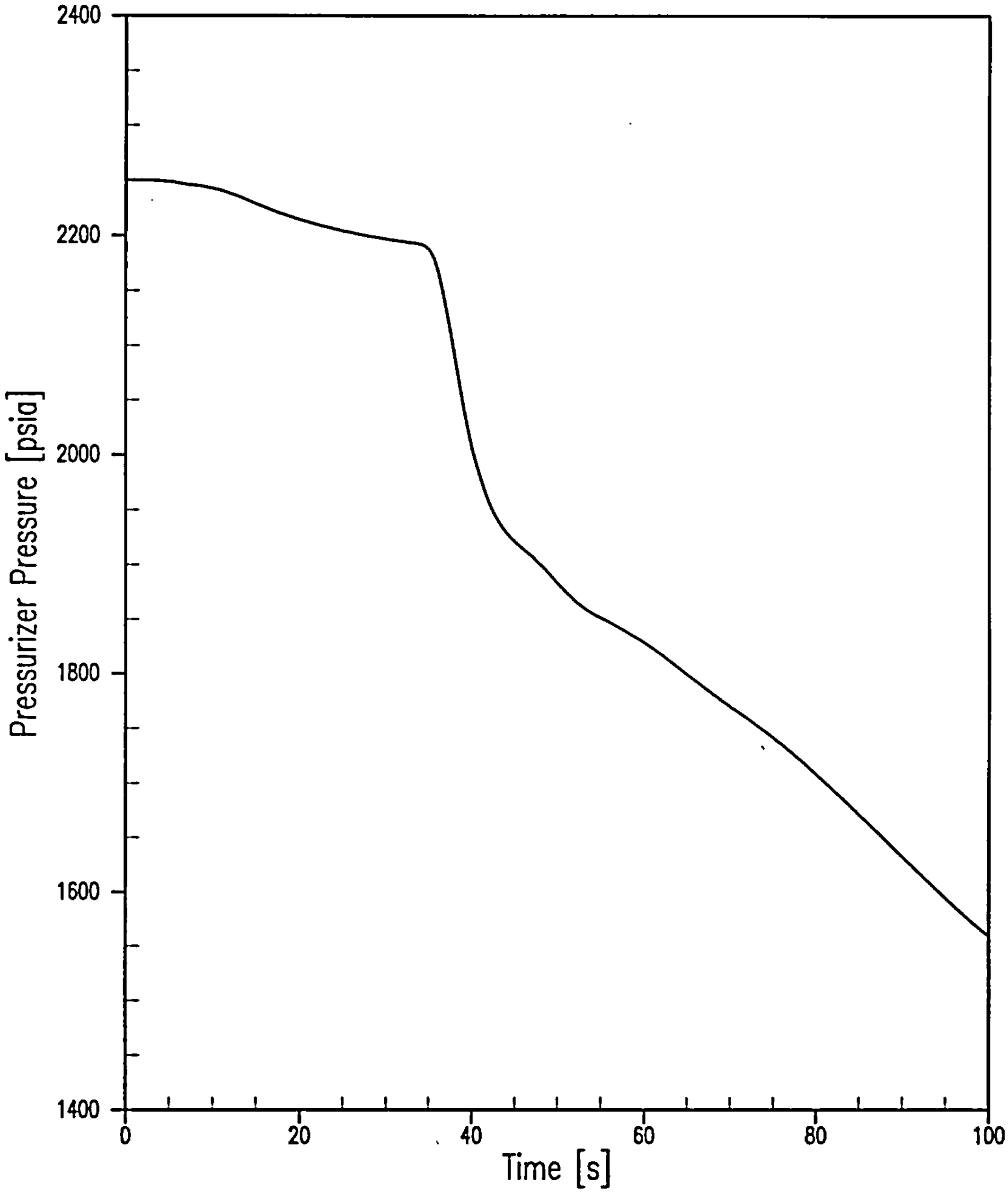


Figure 6.3.1-2 **Feedwater Temperature Reduction at Full Power – Pressurizer Pressure versus Time**

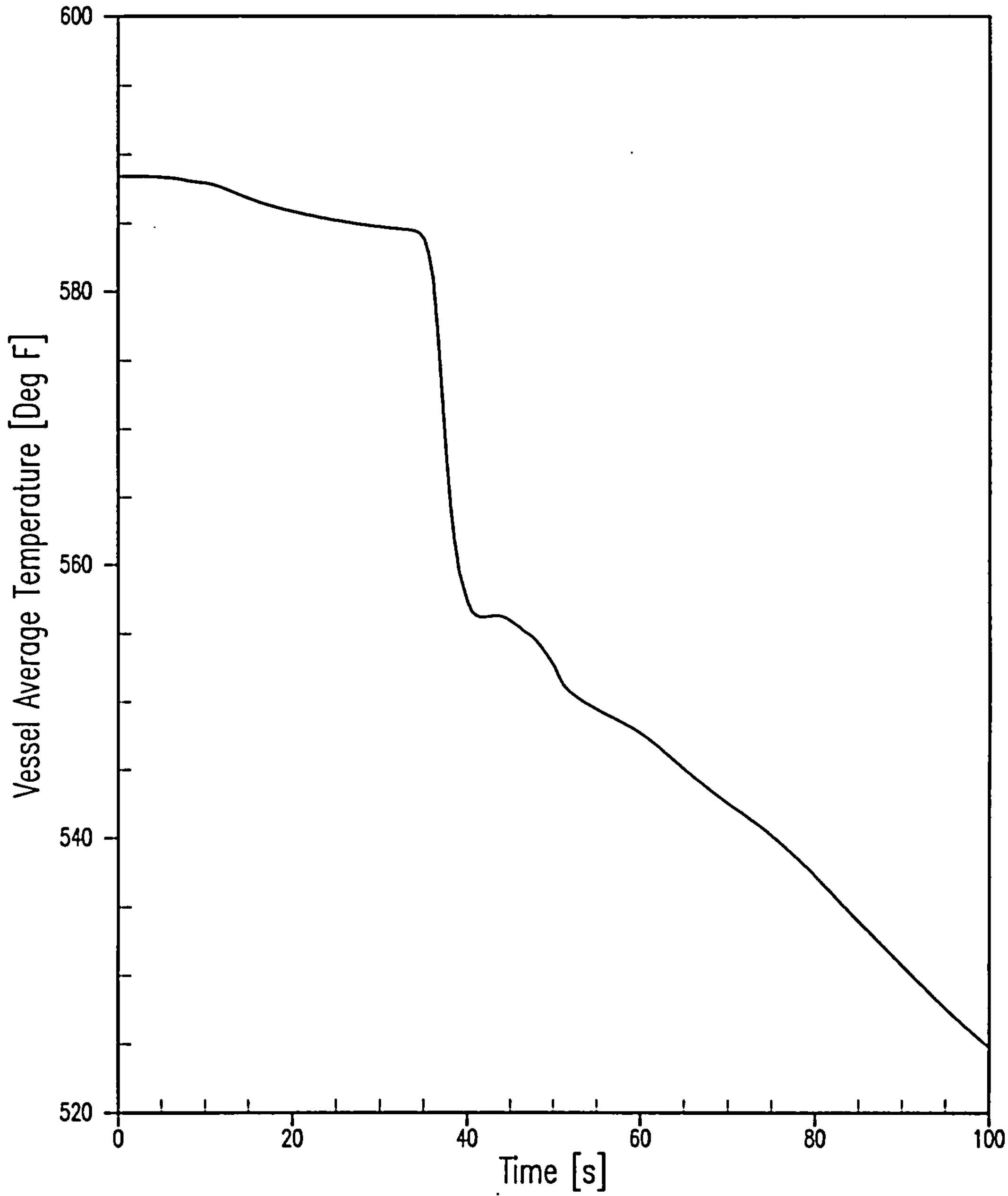


Figure 6.3.1-3 **Feedwater Temperature Reduction at Full Power – Vessel Average Temperature versus Time**

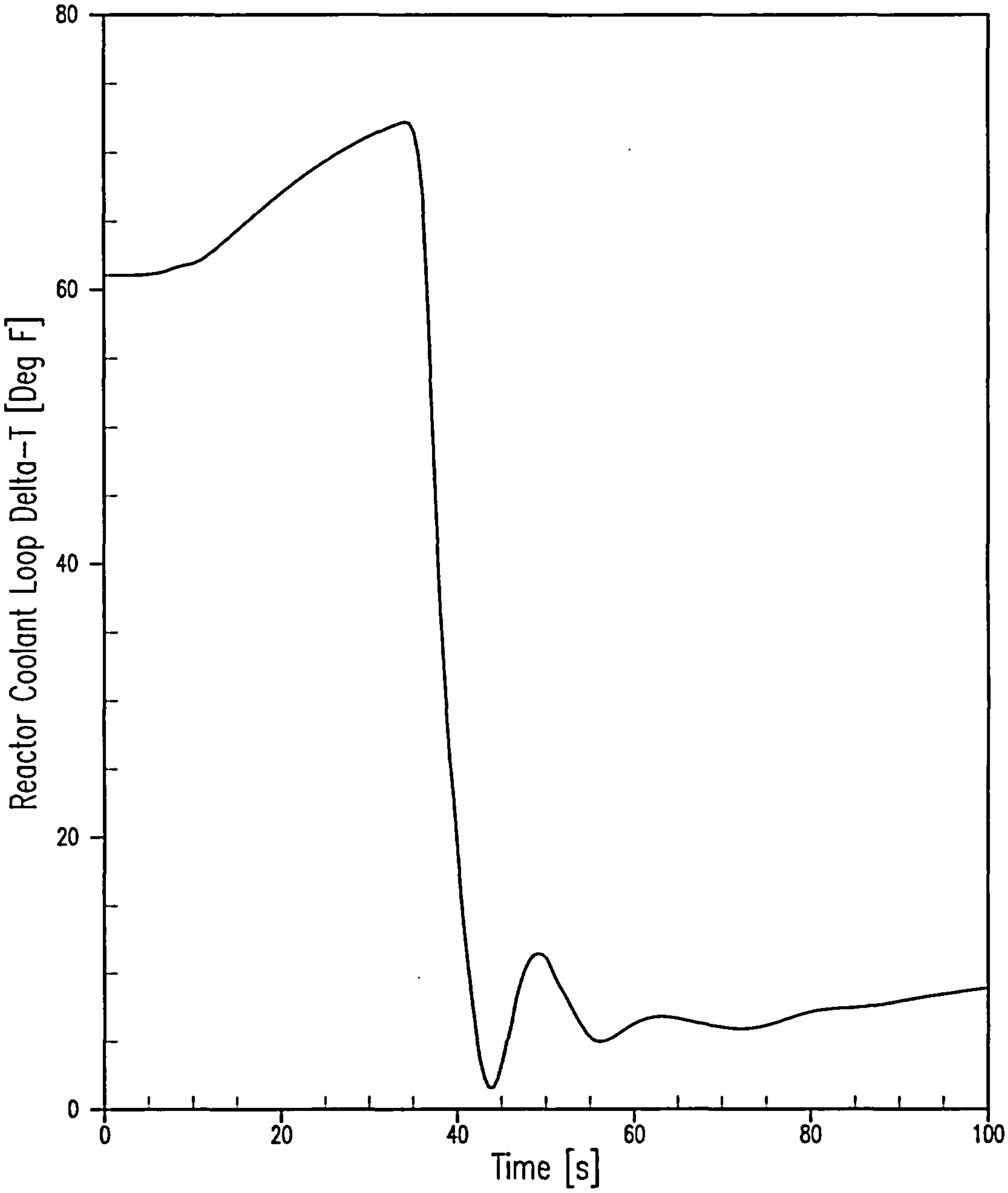


Figure 6.3.1-4 **Feedwater Temperature Reduction at Full Power – Reactor Coolant Delta-T versus Time**

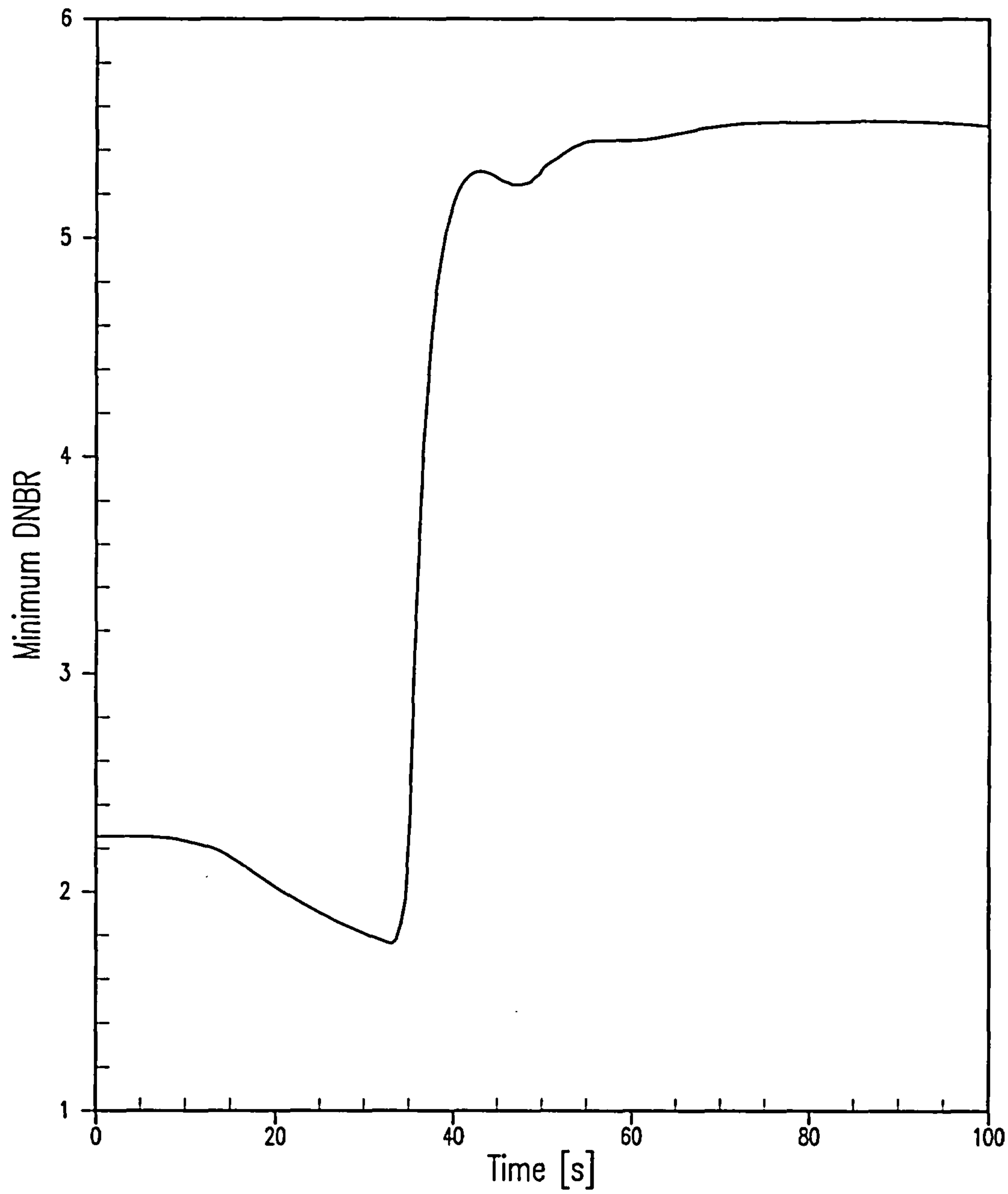


Figure 6.3.1-5 Feedwater Temperature Reduction at Full Power – Minimum DNBR versus Time

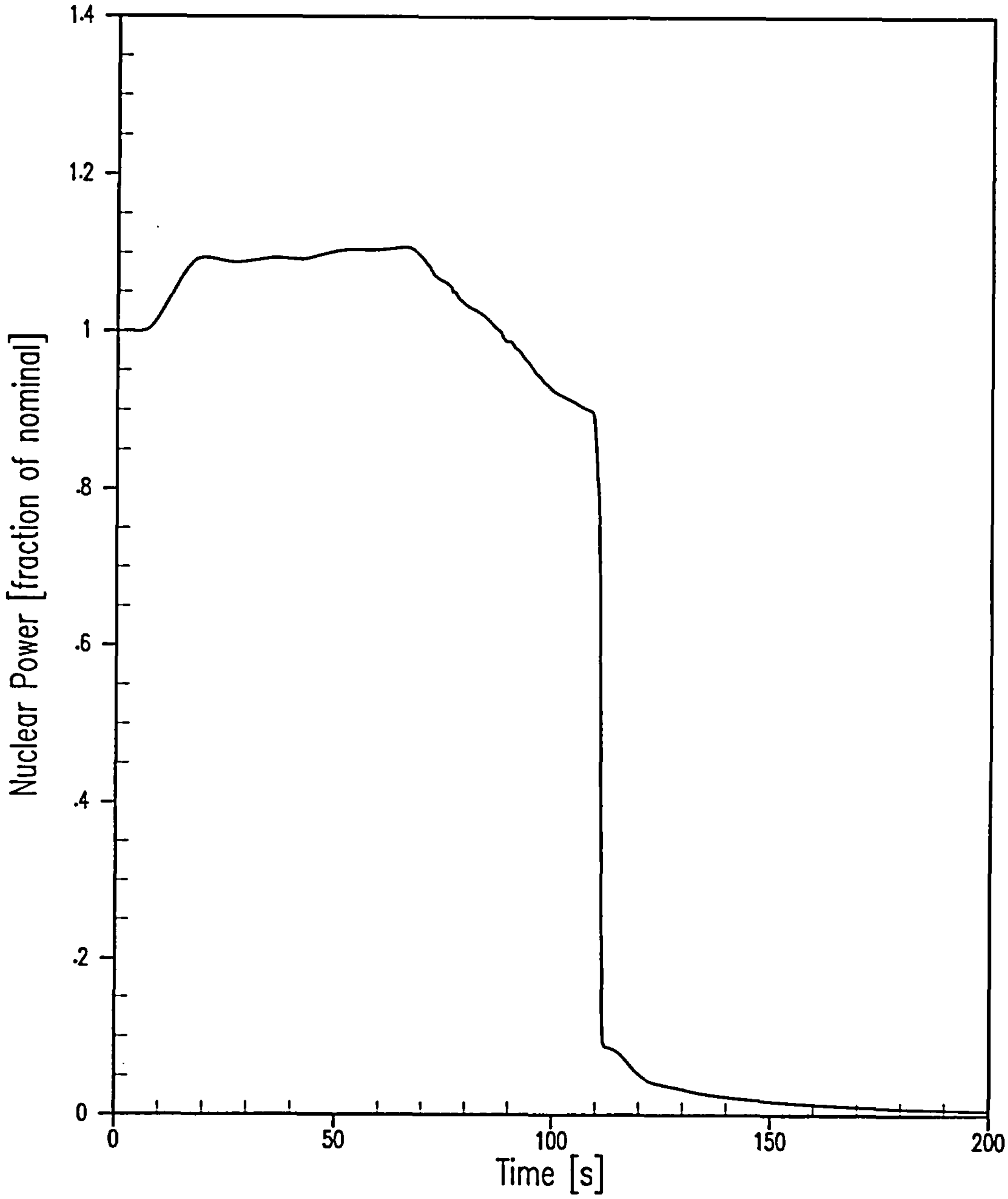


Figure 6.3.1-6 Feedwater Flow Increase Event at Full Power – Reactor Power versus Time

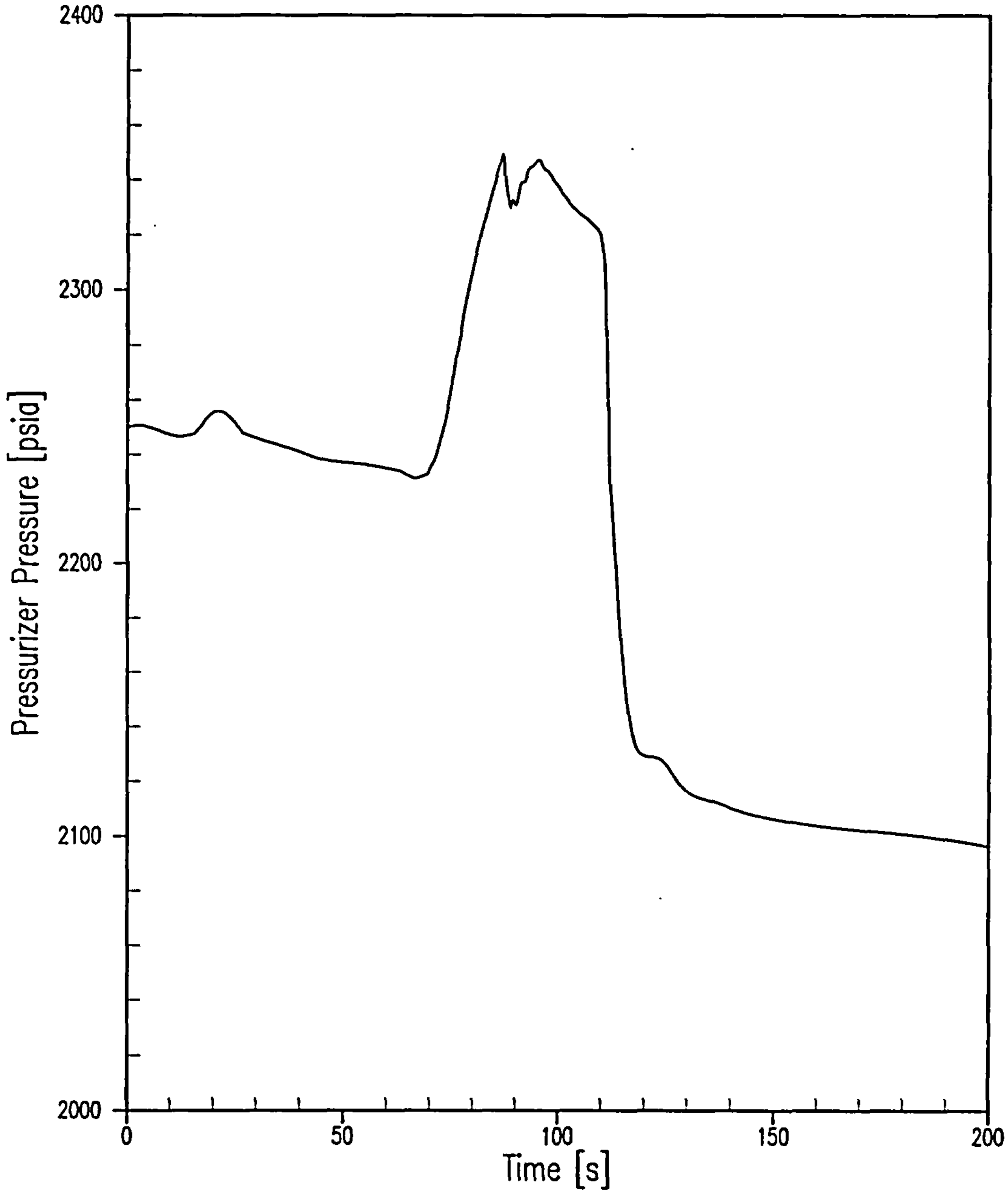


Figure 6.3.1-7 **Feedwater Flow Increase Event at Full Power – Pressurizer Pressure versus Time**

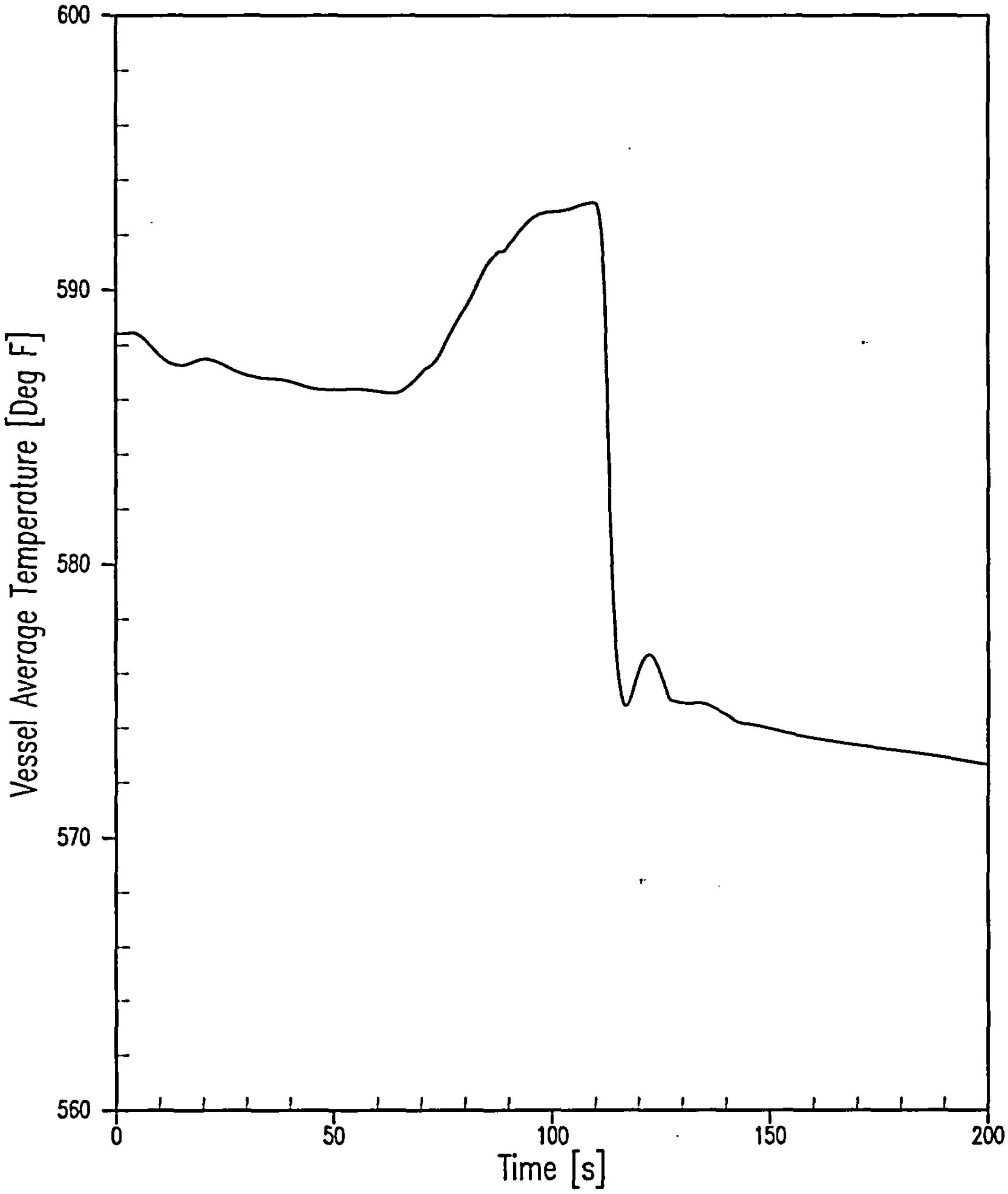


Figure 6.3.1-8 **Feedwater Flow Increase Event at Full Power – Vessel Average Temperature versus Time**

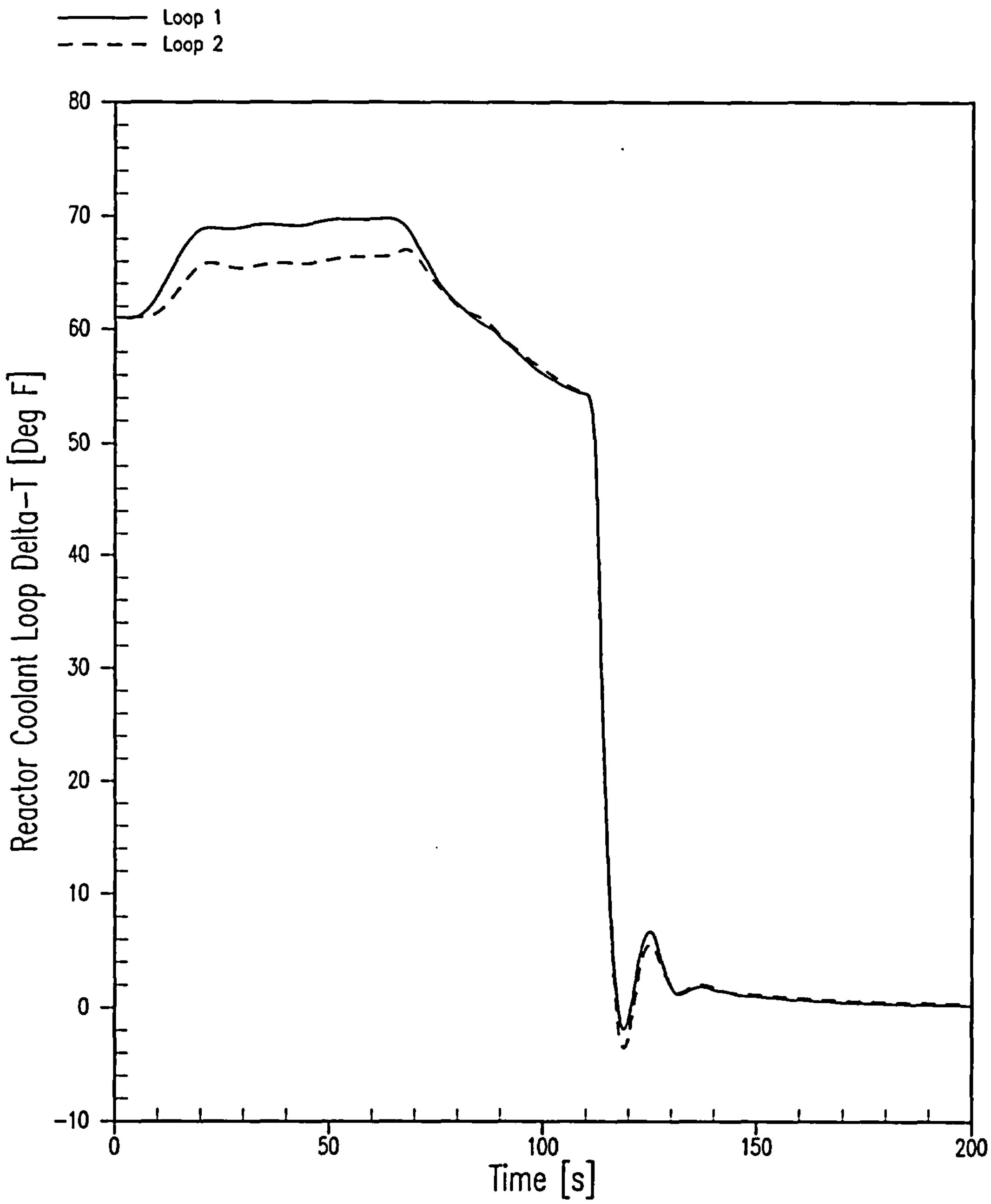


Figure 6.3.1-9 **Feedwater Flow Increase Event at Full Power – Reactor Coolant Delta-T versus Time**

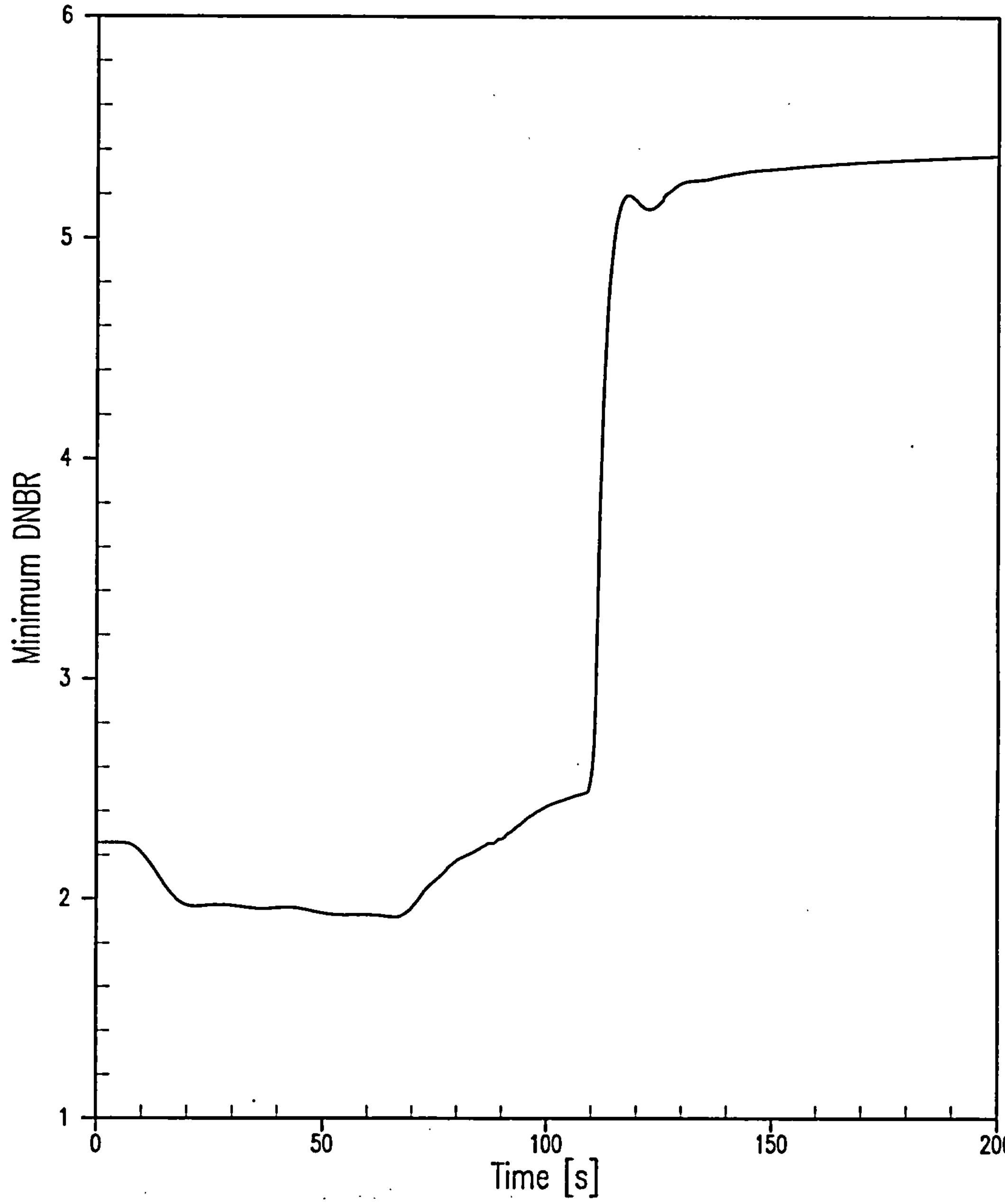


Figure 6.3.1-10 Feedwater Flow Increase Event at Full Power – Minimum DNBR versus Time

6.3.2 Excessive Increase in Secondary Steam Flow (FSAR Section 15.1.3)

6.3.2.1 Accident Description

An excessive load increase incident is defined as an event resulting in a rapid increase in the steam generator steam flow that causes a power mismatch between the reactor core power and the steam generator load demand. The reactor control system is designed to accommodate a 10-percent step load increase or a 5-percent per minute ramp load increase (without a reactor trip) in the range of 15 to 95 percent of full power. Any loading rate in excess of these values may cause a reactor trip actuated by the reactor protection system (RPS). 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, steam dump to the condenser is controlled by reactor coolant condition signals; that is, a high reactor coolant temperature indicates a need for steam dump. A single controller malfunction does not cause steam dump; an interlock is provided that blocks the opening of the valves unless a large turbine load decrease or turbine trip has occurred.

The possible consequence of this accident (assuming no protective functions) is a departure from nucleate boiling (DNB) with subsequent fuel damage. Note that the accident is typically characterized by an approach of parameter values to the protection setpoints without the setpoints actually being reached.

6.3.2.2 Method of Analysis

The excessive load increase incident is analyzed to show that:

- The integrity of the core is maintained.
- The peak reactor coolant system (RCS) and main steam system (MSS) pressures remain below 110 percent of the design values.
- The pressurizer does not become water-solid.

Of these, the primary concern is core integrity, which is maintained by ensuring that the minimum DNB ratio (DNBR) remains above the safety analysis limit value.

However, this transient does not typically result in the actuation of any RPS function (that is, no reactor trip). The effect of this transient on the DNBR is analyzed by applying conservatively large deviations on the initial conditions for power, average coolant temperature, and pressurizer pressure at the normal full-power operating conditions, in order to generate a limiting set of statepoints. Each of these deviations bounds the expected variation that could occur as a result of an excessive load increase incident and is only applied in the direction that has the most adverse impact on DNBR (increased power and coolant temperature, and decreased pressure). The reactor condition statepoints (power, temperature, and pressure) are then compared to the conditions corresponding to operation at the DNBR safety analysis limit (shown previously in Figure 6.3-1).

The results of the analysis performed to support the Callaway Replacement Steam Generator (RSG) Program show that the minimum DNBR remains above the safety analysis limit value. Based on this simplified statepoint analysis, a more detailed analysis using the RETRAN code (Reference 1) is not necessary to support implementation of the RSG Program. The Final Safety Analysis Report (FSAR) has been updated to state that a representative excessive load increase transient is presented as opposed to one specific to the RSG Program.

6.3.2.3 Conclusions

In the event of an excessive load increase incident, (that is, a 10-percent step load increase), the minimum DNBR remains above the safety analysis limit value, thereby precluding fuel or cladding damage. Peak RCS and MSS pressures do not challenge the applicable pressure limits.

6.3.3 Steam System Piping Failure (FSAR Sections 15.1.4 and 15.1.5)

The double-ended rupture of a main steam line at zero-power event was analyzed for the Replacement Steam Generator (RSG) Program using the RETRAN computer code (Reference 1). A detailed description of the analysis is provided in this section. The analysis performed models a large double-ended rupture (also referred to as a hypothetical steam line break) that bounds the inadvertent opening of a steam generator power-operated relief valve (PORV) or dump valve discussed in Final Safety Analysis Report (FSAR) Section 15.1.4 (also referred to as a credible break). No explicit analysis of the credible break is performed as it is always the less limiting of the two cases. Section 6.3.3A of this report contains a description of the analysis performed to address main steam line rupture from full-power conditions.

6.3.3.1 Accident Description

The steam release arising from a rupture of a main steam line will result in an initial increase in steam flow that decreases during the accident as the steam pressure falls. The energy removal from the reactor coolant system (RCS) causes a reduction of coolant temperature and pressure. In the presence of a negative moderator temperature coefficient, the cooldown results in a positive reactivity insertion and subsequent reduction in core shutdown margin. If the most-reactive rod cluster control assembly (RCCA) is assumed stuck in its fully withdrawn position after reactor trip, there is an increased possibility that the core will become critical and return to power. A return to power following a steam pipe rupture is a potential problem, mainly because of the high power peaking factors that would exist assuming the most-reactive RCCA to be stuck in its fully withdrawn position. The core is ultimately shut down by boric acid injection delivered by the emergency core cooling system (ECCS) and accumulators.

The rupture of a major steam line is the most-limiting cooldown transient. It is analyzed at zero power with no decay heat since decay heat would retard the cooldown, thus reducing the return to power. A detailed discussion of this transient with the most-limiting break size (a double-ended rupture) is presented below.

The following functions provide the necessary protection against a steam pipe rupture:

1. Safety injection system (SIS) actuation from any of the following:
    a. Two-out-of-four low-pressurizer pressure signals
    b. Two-out-of-three low main steam line pressure signals in any one loop
    c. Two-out-of-three high-1 containment pressure signals (for main steam system (MSS) depressurization events inside containment).
2. The overpower reactor trips (neutron flux and ΔT) and the reactor trip occurring in conjunction with receipt of the SI signal.
3. Redundant isolation of the main feedwater lines to prevent sustained high feedwater flow, which would cause additional cooldown. Therefore, an SI signal will rapidly close all feedwater control

valves, trip the main feedwater pumps, and indirectly close the feedwater isolation valves that backup the control valves. In addition, trip of the main feedwater pumps results in automatic closure of the respective pump discharge isolation valve.

4. Trip of the fast-acting main steam isolation valves (MSIVs) (assumed to close with a 15-second stroke time) after receipt of an ECCS or main steam line isolation signal on:

   a. SI actuation derived from two-out-of-three low steam line pressure signals in any one loop (above Permissive P-11)

   b. Two-out-of-three high negative steam pressure rate signals in any loop (used only during cooldown and heatup operations: below Permissive P-11)

   c. Two-out-of-three high-high (High-2) containment pressure signals (for main steam system depressurization events inside containment).

For breaks downstream of the isolation valves, closure of all valves will completely terminate the blowdown. For any break, in any location, no more that one steam generator would experience an uncontrolled blowdown even if one of the isolation valves fails to close. Circuit design assures that the main steam isolation blowdown valves are automatically closed whenever the MSIVs are automatically closed.

Following a steam line break, only one steam generator can blow down completely. Each main steam line is provided with an isolation valve located outside of the containment immediately downstream of the steam line safety valves. The isolation valves are signal-actuated valves that close to prevent flow in the normal (forward) flow direction. The valves on all four steam lines will be driven closed to isolate the respective steam generators. Thus, only one steam generator can blow down, minimizing the potential steam release and resultant RCS cooldown. Redundant isolation of the main feedwater lines is provided by: (1) control actions that close the main feedwater valves following reactor trip and (2) trip of the main feedwater pumps and closure of pump discharge and feedwater control valves following receipt of an SI signal. The remaining three steam generators will still be available for dissipation of any decay heat after the initial transient is over. In the case of loss of offsite power, this heat is removed to the atmosphere via the atmospheric relief valves, which have been sized to handle this situation.

Steam flow is measured by monitoring the pressure difference between pressure taps located on the steam generator drum and downstream of the integral flow restrictor nozzles. The effective throat diameter of the flow restrictor nozzles of 6.0315 inches is considerably smaller than the diameter of the main steam pipe. These restrictors are located in the outlet nozzles of the steam generators and serve to limit the maximum steam flow for any break at any location.

6.3.3.2 Method of Analysis

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

1. End-of-life (EOL) shutdown margin at no-load, equilibrium xenon conditions, and the most-reactive assembly stuck in its fully withdrawn position. Operation of the control rod banks

during core burnup is restricted in such a way (that is, technical specification rod insertion limits) that addition of positive reactivity in a steam line break accident will not lead to a more adverse condition than the case analyzed.

2. A negative moderator temperature coefficient (MTC) corresponding to the EOL rodded core with the most-reactive rod in the fully withdrawn position. The coefficient assumption was revised for the RSG analysis to improve the core physics prediction of the point kinetics core model. The variation of the coefficient with temperature and pressure has been included. The k_{eff} versus coolant average temperature at 1,140 psia corresponding to the negative MTC plus the Doppler temperature effect used is shown in the FSAR Figure 15.1-11. All reactivity physics parameters are weighted toward the core sector exposed to the greatest cooldown from the faulted loop.

3. Minimum capability for injection of high concentration boric acid solution corresponding to the most-restrictive single failure in the ECCS. The 2,350 ppm boron solution corresponds to the minimum boron concentration in the refueling water storage tank (RWST). No credit has been taken for the low concentration of boric acid that must be purged from the ECCS lines downstream of the RWST isolation valves prior to the delivery of the concentrated boric acid from the RWST to the reactor coolant loops.

   The SI flow corresponds to that delivered by one charging pump delivering full flow to the cold leg header. The modeling of the ECCS in the Westinghouse pressurized water reactor (PWR) RETRAN model is described in Reference 1.

   The boric acid solution from the ECCS is assumed to be uniformly delivered to the four reactor coolant loops. The boron in the loops is then delivered to the inlet plenum where the coolant (and boron) from each loop is mixed and delivered to the core. The calculation assumes the boric acid is mixed with and diluted by the water flowing in the RCS prior to entering the core. The concentration after mixing depends on the relative flow rates of the RCS and the ECCS. The stuck RCCA is assumed to be conservatively located in the core sector near the faulted steam generator.

   For the case where offsite power is assumed, the sequence of events in the ECCS is the following. After the generation of the SI signal (appropriate delays for instrumentation, logic, and signal processing included), the appropriate valves begin to operate and the charging pump starts. In 27 seconds, the valves are assumed to be in the final position and the pump is assumed to be at full speed and to be drawing suction from the RWST. The 27 seconds includes 2 seconds for electronic delay, 15 seconds for the RWST valve(s) to open, and 10 seconds for the volume control tank (VCT) valve(s) to close. (The charging pump start, normal charging isolation, and high-head injection header alignment occur in conjunction with the RWST valve alignment.)

   In cases where offsite power is not available, an additional 12-second delay is assumed to start the diesels and to re-energize the engineered safety features (ESF) electrical buses. That is, after a total of 39 seconds following the time an SI setpoint is reached at the sensor, the ECCS is assumed to be capable of delivering flow to the RCS.

The SIS piping contains low concentration (0 ppm assumed) borated water, which delays the injection of the 2,350 ppm borated RWST water from reaching the RCS. This delay in 2,350 ppm boron solution reaching the RCS is inherently included in the RETRAN modeling.

4. To maximize the primary-to-secondary heat transfer rate, 0-percent steam generator tube plugging (SGTP) is assumed.

5. Since the steam generators are provided with integral flow restrictors with a 1.39 ft^2 throat area, any rupture with a break greater than 1.39 ft^2, regardless of the location, would have the same effect on the nuclear steam supply system (NSSS) as the 1.39 ft^2 break. The following 2 cases have been considered in determining the core power and RCS transients.

   a. Complete severance of a pipe, with the plant initially at no-load conditions, and full reactor coolant flow (thermal design flow (TDF)) with offsite power available.

   b. Complete severance of a pipe with the plant initially at no-load conditions with offsite power unavailable; loss of offsite power results in reactor coolant pump coastdown.

6. Power peaking factors corresponding to one stuck RCCA and non-uniform core inlet coolant temperatures are determined at EOL. 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 control assembly during the return-to-power phase following the steam line break. This void, in conjunction with the large negative MTC, partially offsets the effect of the stuck assembly. The power peaking factors depend on the core power, operating history, temperature, pressure, and flow, and thus are different for each case studied.

7. In computing the steam flow during a steam line break, the Moody Curve (Reference 3) for f(L/D) = 0 is used. The Moody multiplier is 1 with a discharge at dry saturated steam conditions.

8. Perfect moisture separation in the steam generator is assumed unless the mixture level reaches the top of the steam generator. The assumption leads to conservative results since, in fact, considerable water would be discharged. Water entrainment would reduce the magnitude of the temperature decrease in the core.

9. All main and auxiliary feedwater pumps are assumed to be operating at full capacity when the rupture occurs. This assumption increases the cooldown in accidents such as steam line rupture. Main feedwater is isolated following the SI signal; however, auxiliary feedwater continues for the duration of the transient.

10. The effect of heat transferred from thick metal in the RCS and the steam generators is not included in the cases analyzed. The heat transferred from these sources is a net benefit in departure from nucleate boiling (DNB) when the effect of the extra heat on reactivity and peak power is considered.

6.3.3.3 Description of Analysis

A detailed analysis using the RETRAN computer code is performed to determine the plant transient conditions following a main steam line break. The code computes pertinent variables, including the core heat flux, RCS temperature, and pressure. A detailed core analysis is then performed using the ANC code to determine if the RETRAN-predicted reactivity feedback model is conservative. Statepoints consisting of core heat flux, RCS temperature and pressure are then used as input to the detailed thermal and hydraulic digital computer code, VIPRE, to determine if DNB occurs. Details of the Westinghouse PWR RETRAN model are documented in Reference 1. The RETRAN, ANC, and VIPRE computer codes are described in detail in Section 6.3.0.6 of this report.

A major break in a pipeline is classified as an American Nuclear Society (ANS) Condition IV event. Minor secondary-system pipe breaks are classified as ANS Condition III events. All of these events are analyzed to meet Condition II criteria. The only criterion that may be challenged during this event is the one that states that the critical heat flux should not be exceeded. The evaluation shows that this criterion is met by ensuring that the minimum DNB ratio (DNBR) does not go below the limit value at any time during the transient.

6.3.3.4 Results

The time sequence of events for postulated steam line rupture accidents with and without offsite power are presented in Table 6.3.3-1.

Figures 6.3.3-1 through 6.3.3-6 show the plant response following a main steam pipe rupture. Offsite power is assumed to be available such that full reactor coolant flow exists. The transient shown assumes an uncontrolled steam release from only one steam generator.

As can be seen, the core attains criticality with RCCAs inserted (with the design minimum shutdown margin and assuming one stuck RCCA), but is quickly returned to a subcritical condition as a boric acid solution at 2,350 ppm (from the RWST) enters the RCS. The delay time consists of the time to receive and actuate the SI signal, to start the charging pumps, and to completely align valve trains in the ECCS lines, including VCT isolation. The charging pumps are then ready to deliver flow. At this stage, a further delay is incurred before 2,350 ppm boron solution can be injected to the RCS due to the low concentration solution being swept from the SI lines. Should a partial loss of offsite power occur such that power is lost to the ESF functions, an additional SI delay of 12 seconds would occur while the diesel generators start up and re-energize the ESF buses. Allowing for these delays, a peak core power well below the nominal full-power value is attained.

Should the core be critical at near zero power when the rupture occurs, the initiation of the SI signal by low steam line pressure or high containment pressure will trip the reactor. Steam release from more than one steam generator will be prevented by automatic closure of the isolation valves in the steam lines by an SI signal derived from two-out-of-three low steam line pressure signals in any one loop, two-out-of-three high negative steam pressure rate signals in any loop, or two-out-of-three High-2 containment pressure signals. The MSIVs are assumed to be fully closed in 15 seconds (stroke time) after receipt of a closure signal. This analysis conservatively assumed 2 seconds to account for signal processing.

Figures 6.3.3-7 through 6.3.3-12 show the plant response for the case discussed above with a total loss of offsite power. This assumption results in a coastdown of the reactor coolant pumps. In this case, the core power increases at a slower rate and reaches a lower peak value than in the cases in which offsite power is available to the reactor coolant pumps. The ability of the emptying steam generator to extract heat from the RCS is reduced by the decreased flow in the RCS.

It should be noted that following a steam line break, only one steam generator blows down completely. Thus, the remaining steam generators are still available for dissipation of decay heat after the initial transient is over. In the case with a loss of offsite power, this heat is removed to the atmosphere via the steam line safety valves.

Following blowdown of the faulted steam generator, the plant can be brought to a stabilized hot-standby condition through control of the auxiliary feedwater (AFW) flow and SI flow, as described by plant operating procedures. The operating procedures call for operator action to limit RCS pressure and pressurizer level by terminating SI flow and to control steam generator level and RCS coolant temperature using the auxiliary feedwater system. However, these actions are not modeled or credited in the safety analysis discussed in this section.

6.3.3.5 Conclusions

A DNB analysis was performed for the steam line break cases described above. The analysis demonstrated that the minimum DNBR remains above the limit value and, thus, concludes that the DNB design basis is met for the steam line break event initiated from zero power with the RSGs.

6.3.3A Hot-Full-Power Steam System Piping Failure

The analysis of the steam line rupture core response transient typically found in Safety Analysis Reports assumes zero-power (Mode 2) conditions. The greatest cooldown, and therefore the greatest reactivity excursion, would occur from a Mode 2 condition, where the decay heat level is low and the steam generator shell-side inventory and pressure are high. Section 6.3.3 of this report presents the results obtained for the zero-power steam line rupture event for Callaway.

For a number of years, this was the only steam line rupture - core response event analyzed by Westinghouse. In the mid-1970s, Westinghouse issued WCAP-9226 (Reference 6) for the steam line rupture event that examined the effects of power level, break size, plant variations, and various single failures. As a result of this study, the WCAP concludes that "... the largest double-ended steam line rupture at end of life, hot shutdown conditions with the most reactive RCCA in the fully withdrawn position is a limiting and sufficiently conservative licensing basis to demonstrate that the Westinghouse PWR is in compliance with 10CFR100 criteria for Condition II, III and IV steam line break transients." However, to ensure that certain plant modifications that may have been implemented over time (for example, changes to the overpower and overtemperature ΔT reactor trip function setpoints and associated dynamic compensation parameters, etc.) do not invalidate the assumptions that went into WCAP-9226, a specific analysis at full-power conditions is also performed.

Method of Analysis

The steam line rupture – full-power core response event is analyzed with a Westinghouse version of the RETRAN code (Reference 1). The RETRAN computer code is described in detail in Section 6.3.0.6 of this report.

The purpose of the analysis is to demonstrate that a reactor trip occurs in adequate time to ensure fuel cladding integrity. Breaks of various sizes are postulated to occur in the steam line before the MSIV. For the RSG design, each steam line has an effective flow area of 4.6 ft^2, with a flow restrictor having an effective flow area of 1.39 ft^2. This flow restrictor is an integral part of the steam generator.

A range of break sizes are analyzed, ranging from breaks smaller than or equivalent to the inadvertent opening of a steam system valve to the hypothetical double-ended rupture of a main steam line. The larger break sizes generate reactor trips on the low steam line pressure SIS reactor trip function while smaller breaks trip on the overpower ΔT reactor trip function. The most limiting break size is the largest break case that results in a reactor trip on the overpower ΔT reactor trip function.

Results

The transient response for the steam line rupture – full-power core response analysis is shown in Figures 6.3.3A-1 through 6.3.3A-11. Table 6.3.3A-1 presents the time sequence of events and results for the limiting break size (0.88 ft^2).

Conclusions

The analysis shows that the acceptance criteria applicable to this event are satisfied. The DNBR safety analysis limit is met, and there is no melting of the fuel centerline.

Table 6.3.3-1 Time Sequence of Events for Rupture of a Main Steam Line	
Event	**Time (Seconds)**
With Offsite Power:	
Steam Line Ruptures	0.0
Low Steam Line Pressure Setpoint Reached in Two Loops	2.059
Steam Line Isolation Occurs	18
Criticality Attained	23
SI Begins	28
Borated Water from the RWST Reaches the Core	100
Minimum DNBR Occurs	252
Peak Core Thermal Power Occurs	267
Without Offsite Power:	
Steam Line Ruptures	0.0
Low Steam Line Pressure Setpoint Reached in Two Loops	2.059
Loss of AC Power and RCPs Begin Coastdown	3.0
Steam Line Isolation Occurs	18
Criticality Attained	30
SI Begins	40
Borated Water from the RWST Reaches the Core	112
Peak Core Thermal Power Occurs	414

Table 6.3.3A-1 Time Sequence of Events for Full-Power Steam Line Rupture		
Event		**Time (Seconds)**
Break Initiation with Reactor at Full Power		0.0
OPΔT Setpoint Reached in Loop 4		12.4
OPΔT Setpoint Reached in Loops 1, 2, and 3		14.9
Rods Start to Move on OPΔT Reactor Trip Signal		16.9
Minimum DNBR Reached		17.6
Maximum Core Heat Flux Reached		17.6
Turbine Trip Following Reactor Trip		18.9
Results	**Calculated Values**	**Limit**
Peak Fuel Centerline Linear Power (kw/ft)	21.19	22.46
Minimum DNBR (thm/typ)	1.80/1.83	1.55 / 1.59

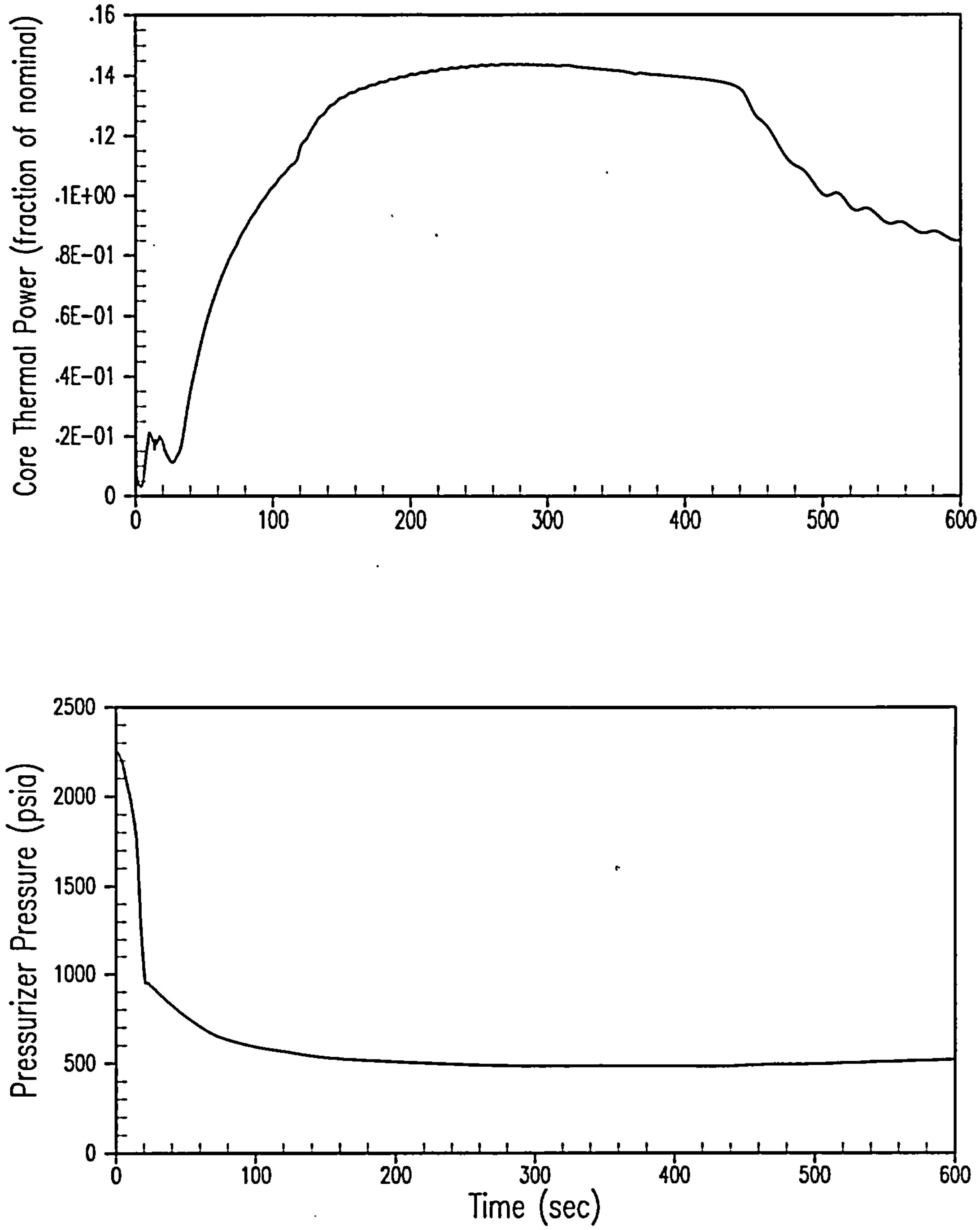
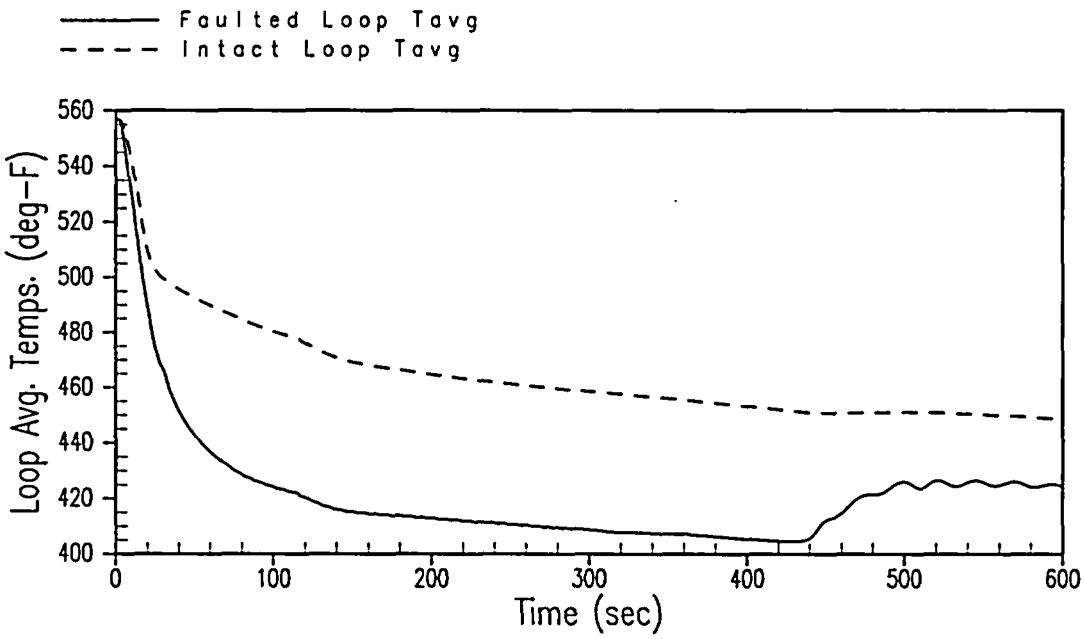


Figure 6.3.3-1 HZP Steam Line Break Transient with Offsite Power Double-Ended Rupture – Core Thermal Power and Pressurizer Pressure versus Time



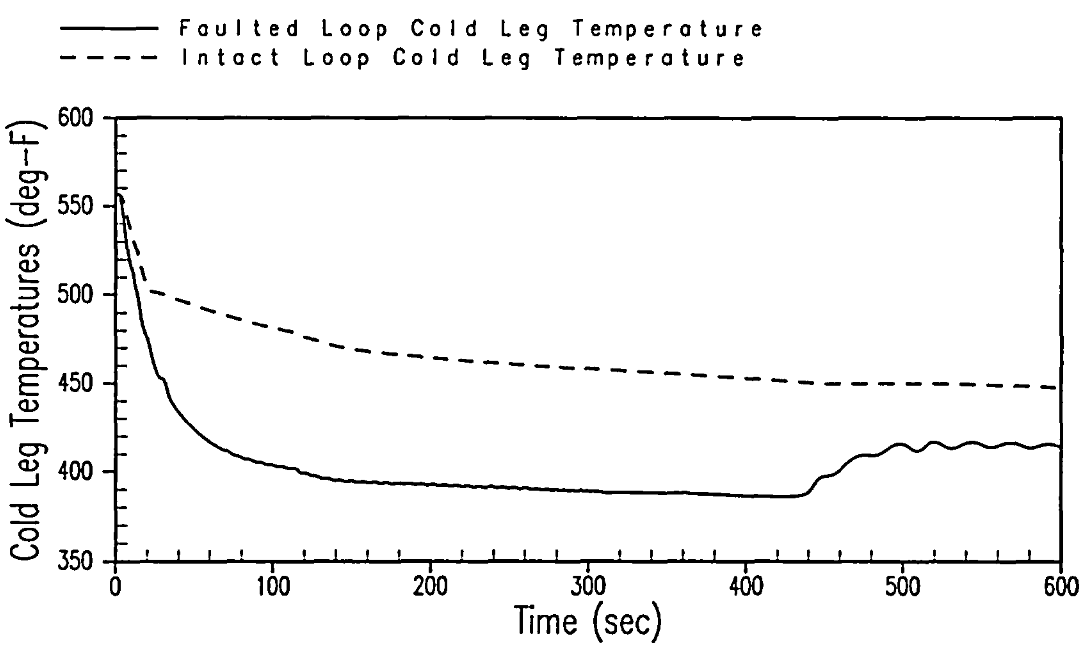
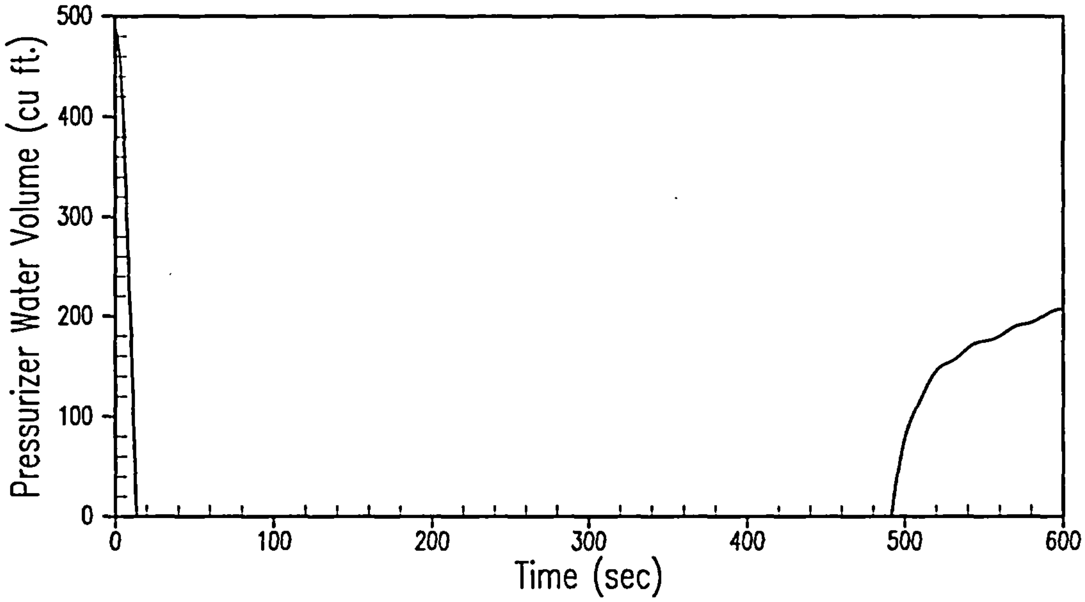


Figure 6.3.3-2 HZP Steam Line Break Transient with Offsite Power Double-Ended Rupture – RCS Loop Average and Cold Leg Temperatures versus Time



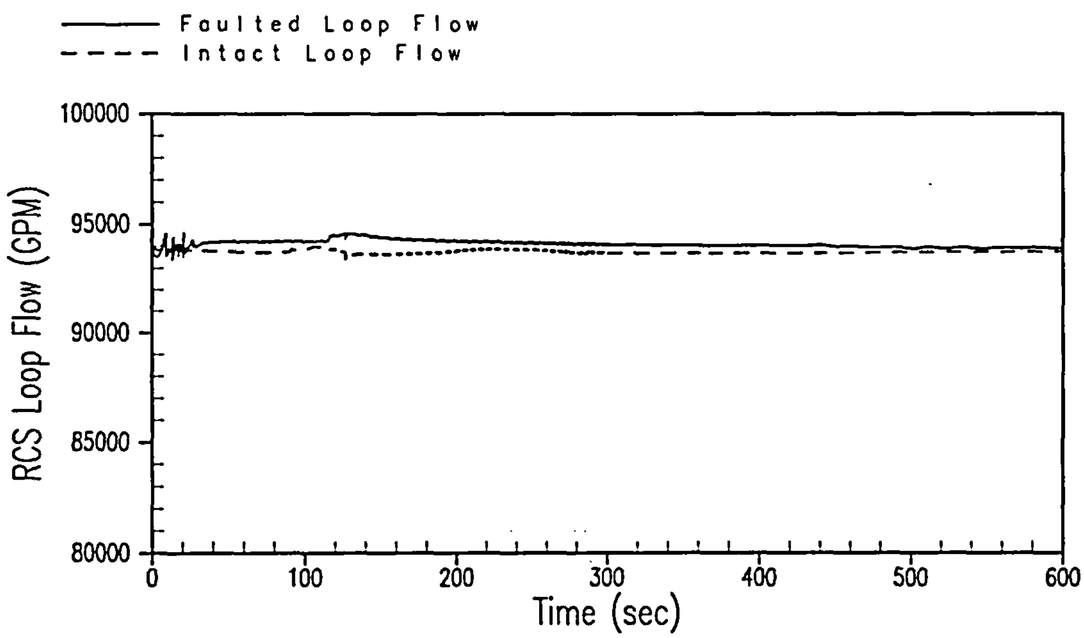


Figure 6.3.3-3 **HZP Steam Line Break Transient with Offsite Power Double-Ended Rupture – Pressurizer Water Volume and RCS Loop Flow versus Time**

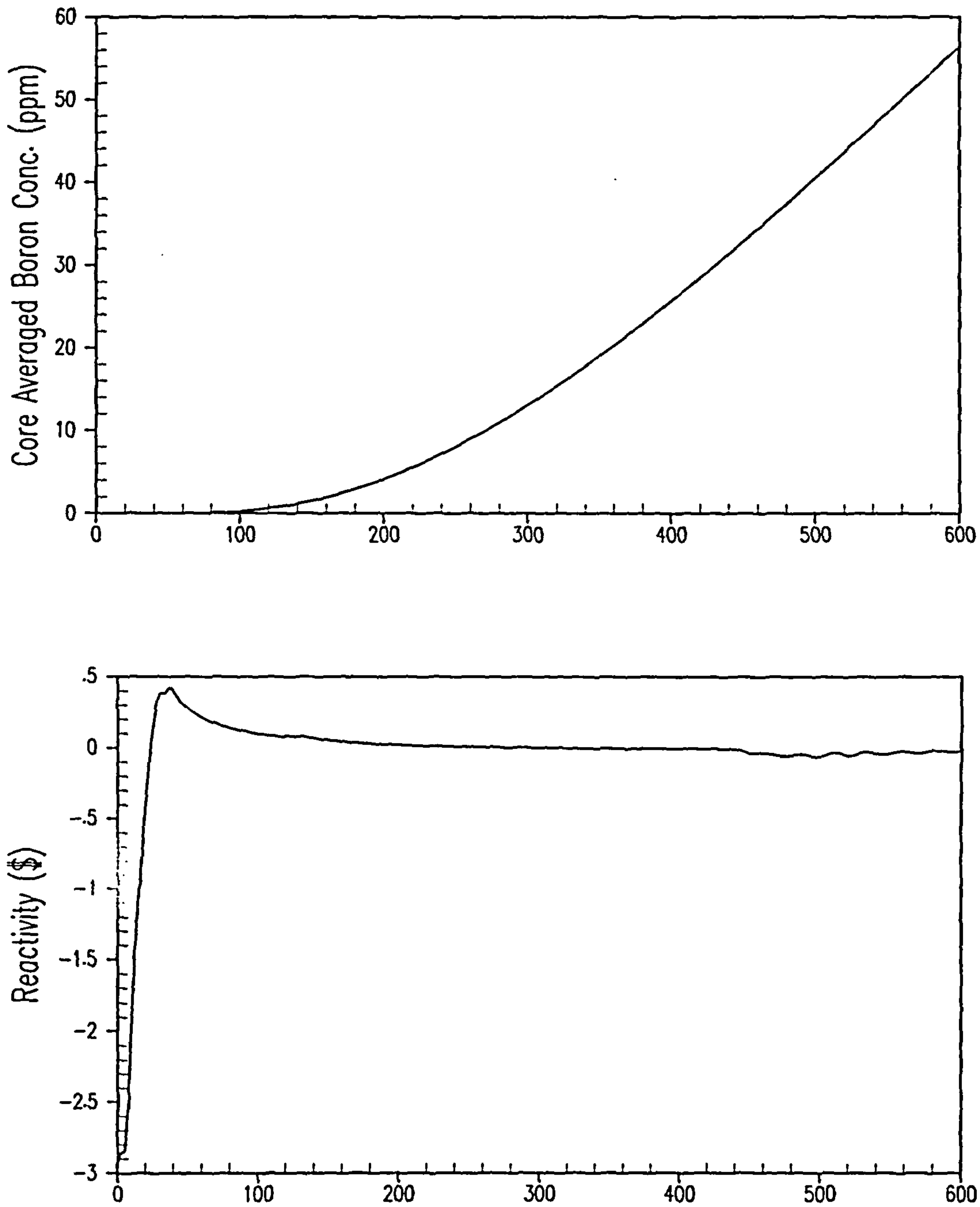
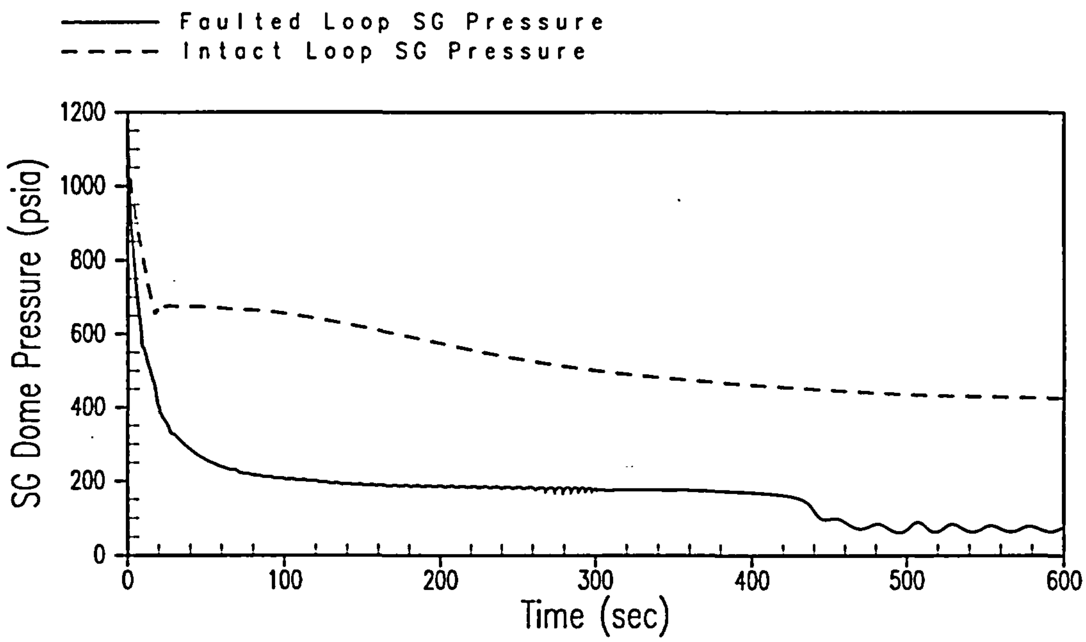


Figure 6.3.3-4 **HZP Steam Line Break Transient with Offsite Power Double-Ended Rupture – Core Boron Concentration and Reactivity versus Time**



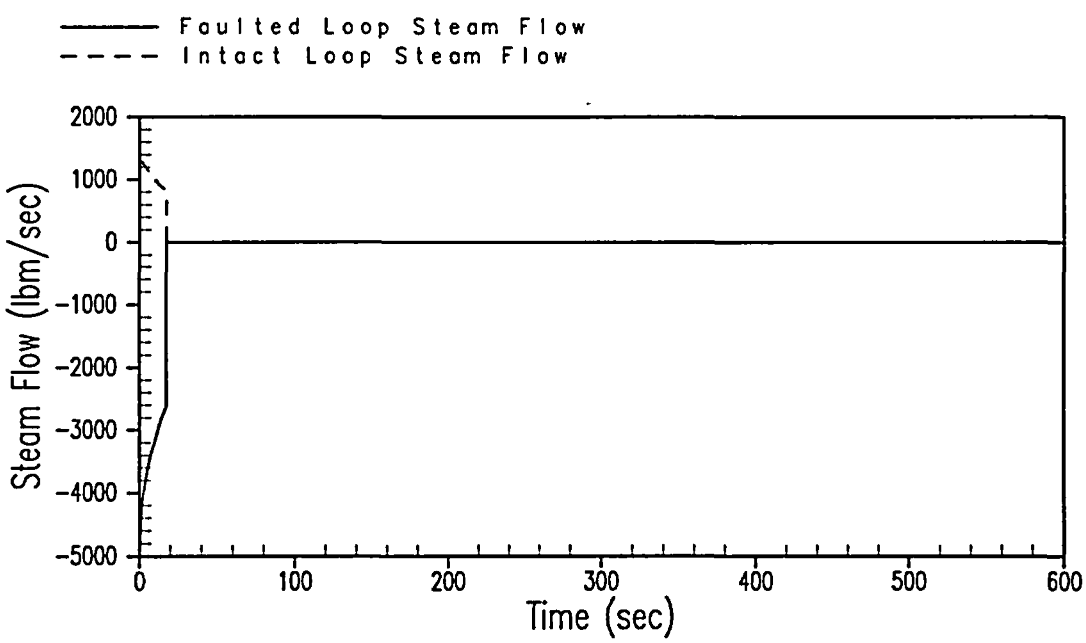


Figure 6.3.3-5 HZP Steam Line Break Transient with Offsite Power Double-Ended Rupture – Steam Pressure and Steam Flow versus Time

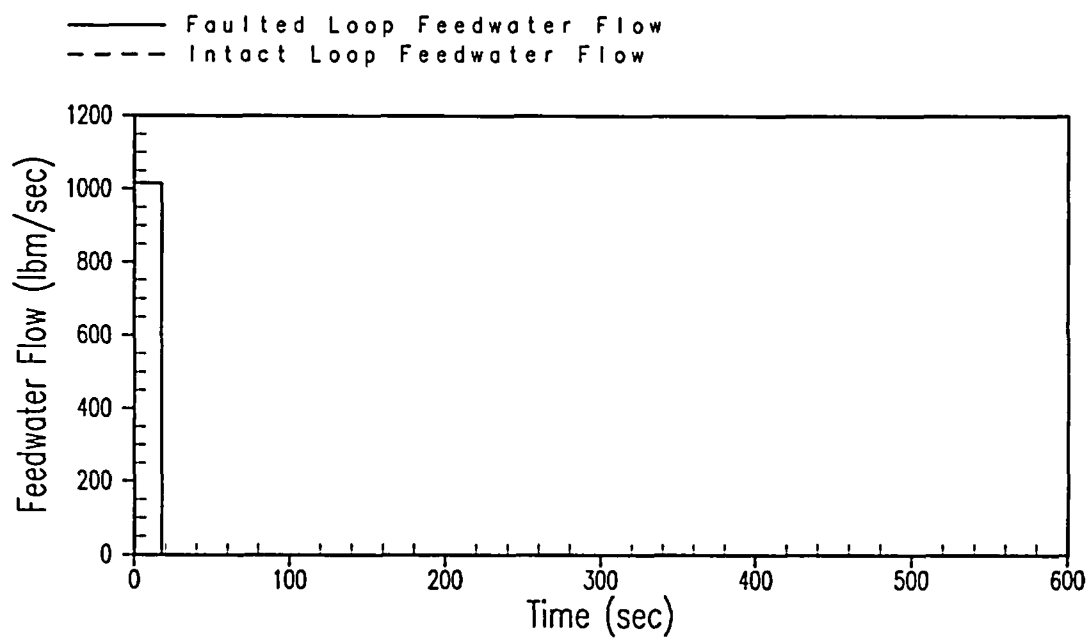


Figure 6.3.3-6 HZP Steam Line Break Transient with Offsite Power Double-Ended Rupture – Feedwater Flow versus Time

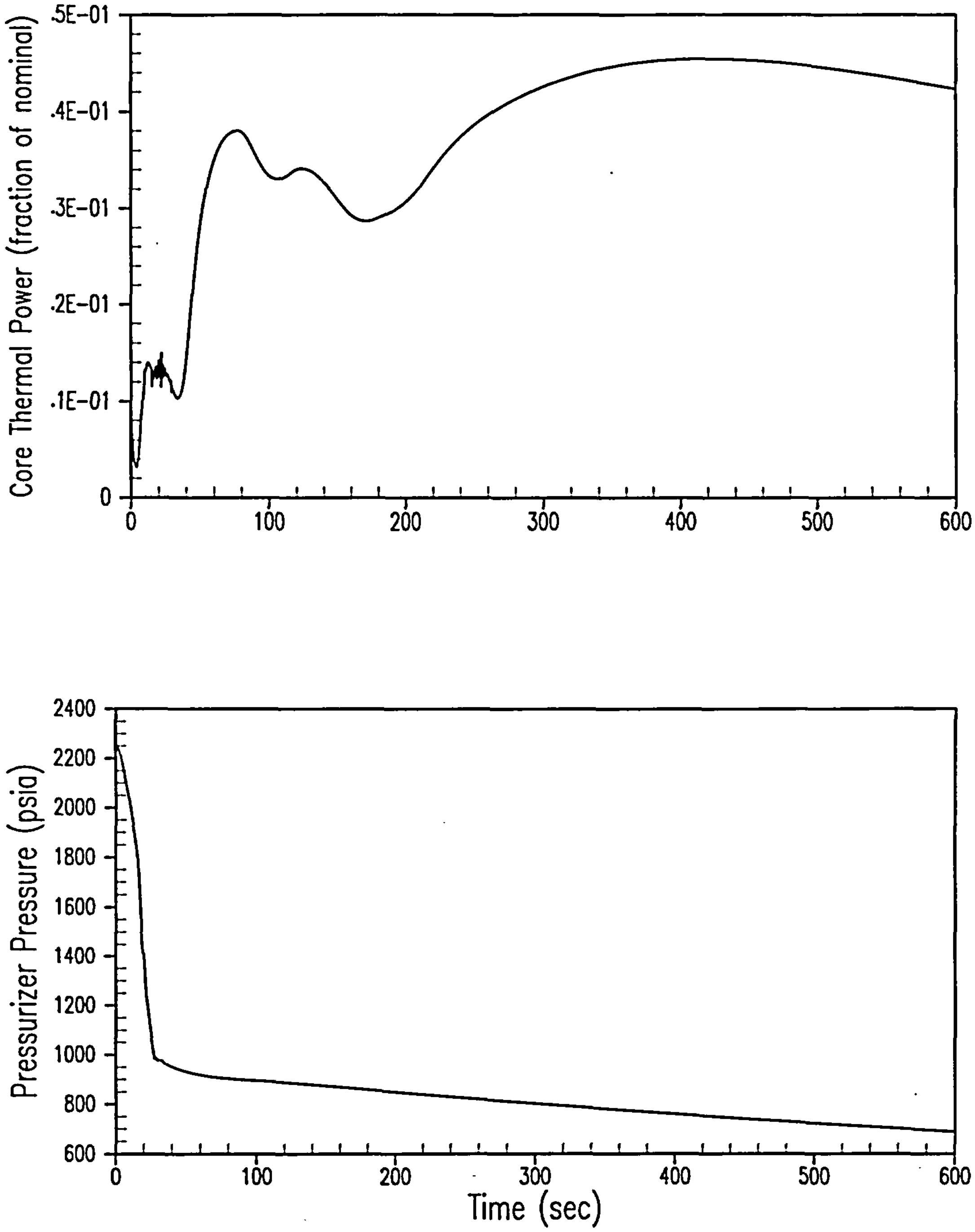
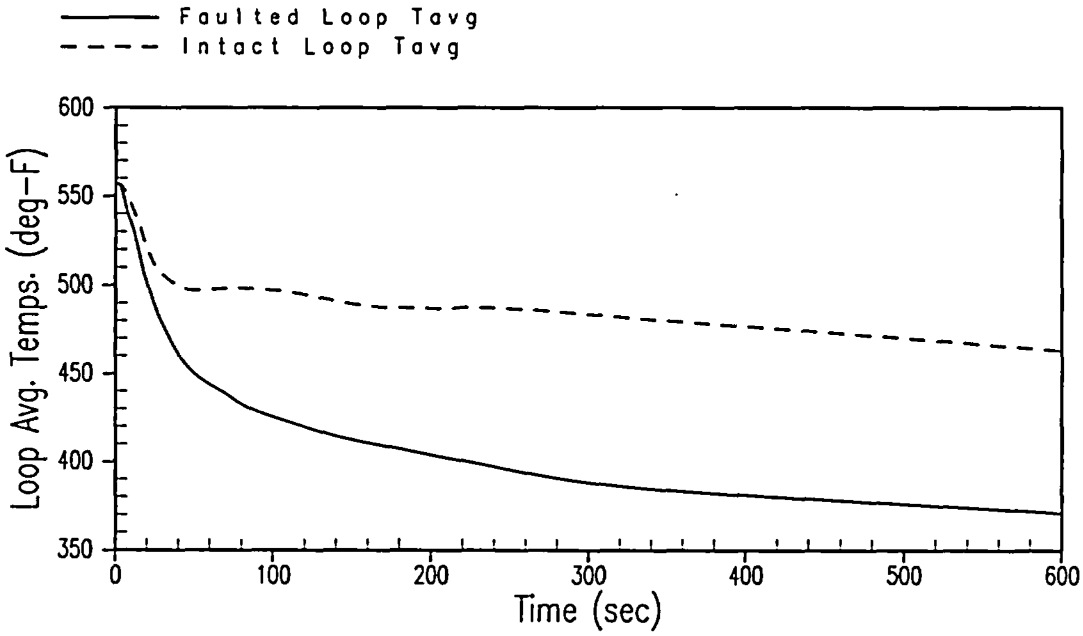


Figure 6.3.3-7 HZP Steam Line Break Transient without Offsite Power Double-Ended Rupture – Core Thermal Power and Pressurizer Pressure versus Time



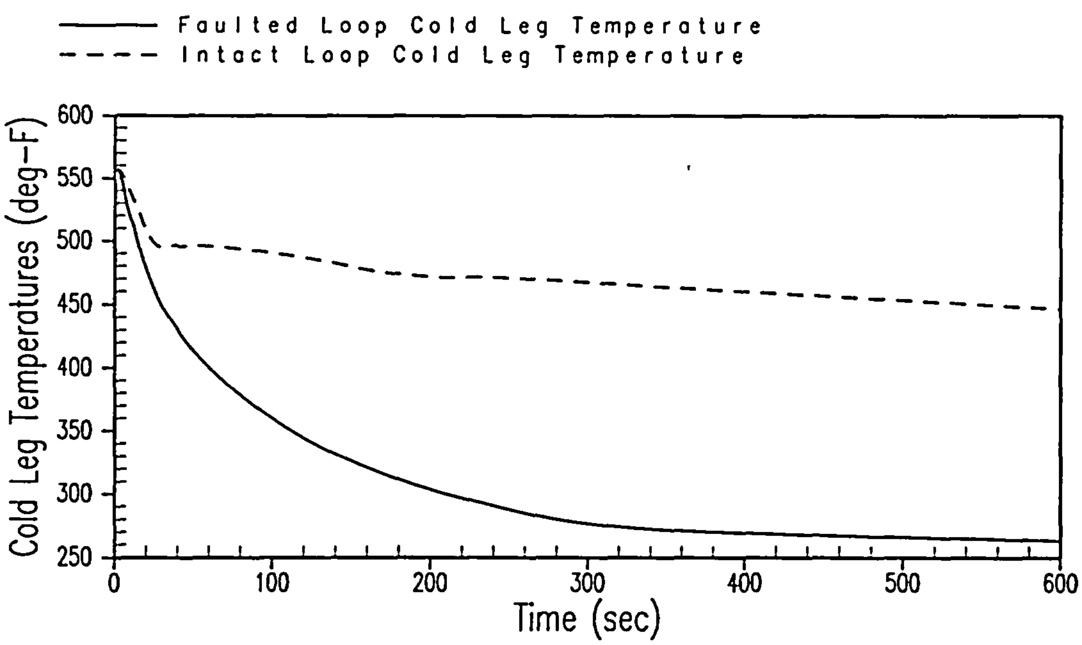
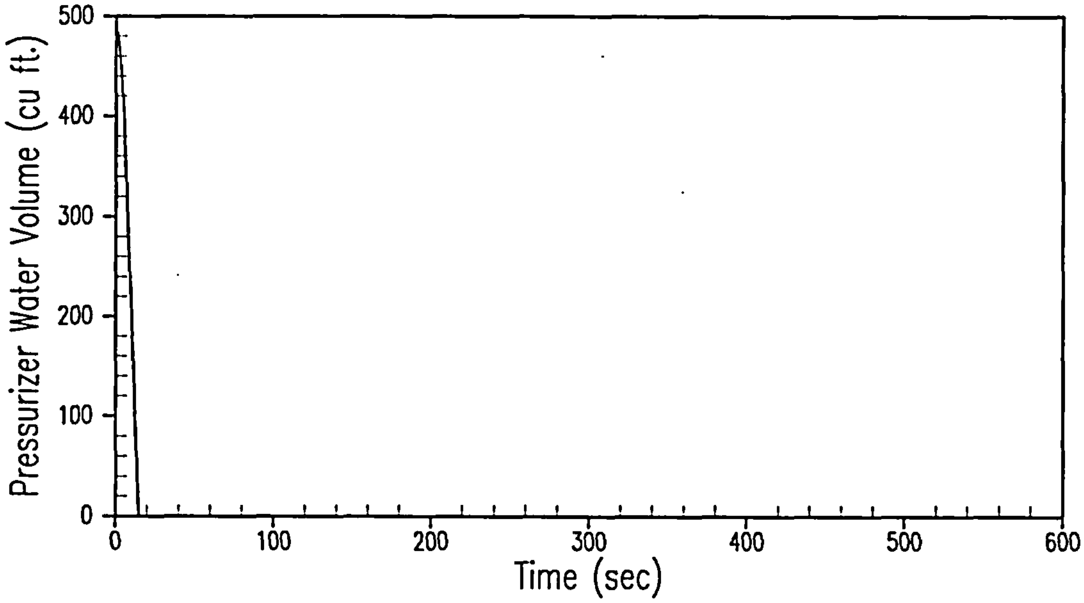


Figure 6.3.3-8 **HZP Steam Line Break Transient without Offsite Power Double-Ended Rupture – RCS Loop Average and Cold Leg Temperatures versus Time**



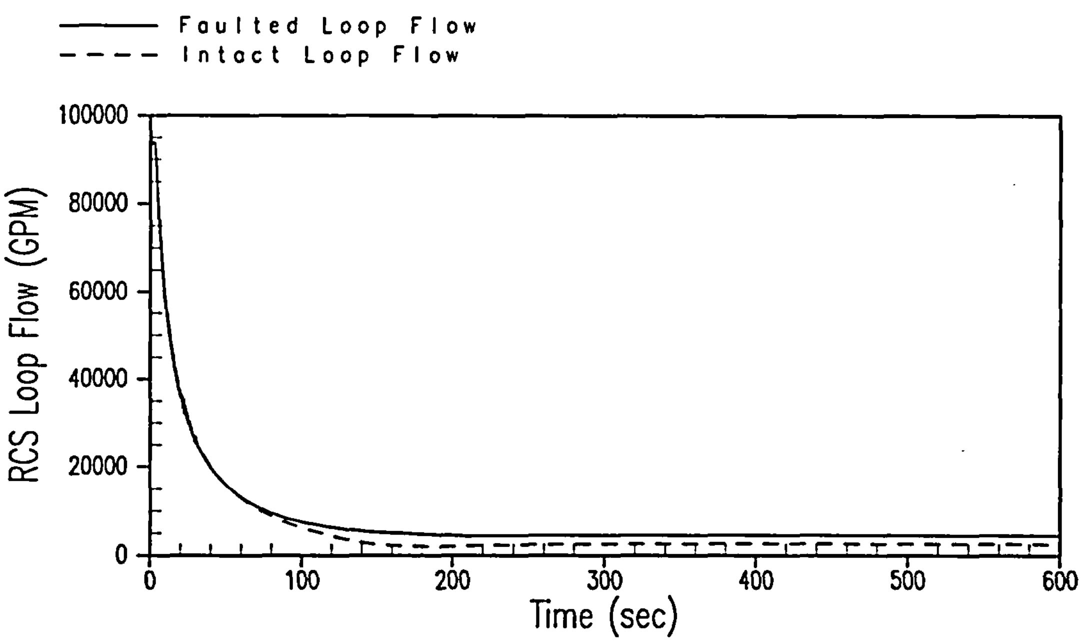


Figure 6.3.3-9 HZP Steam Line Break Transient without Offsite Power Double-Ended Rupture – Pressurizer Water Volume and RCS Loop Flow versus Time

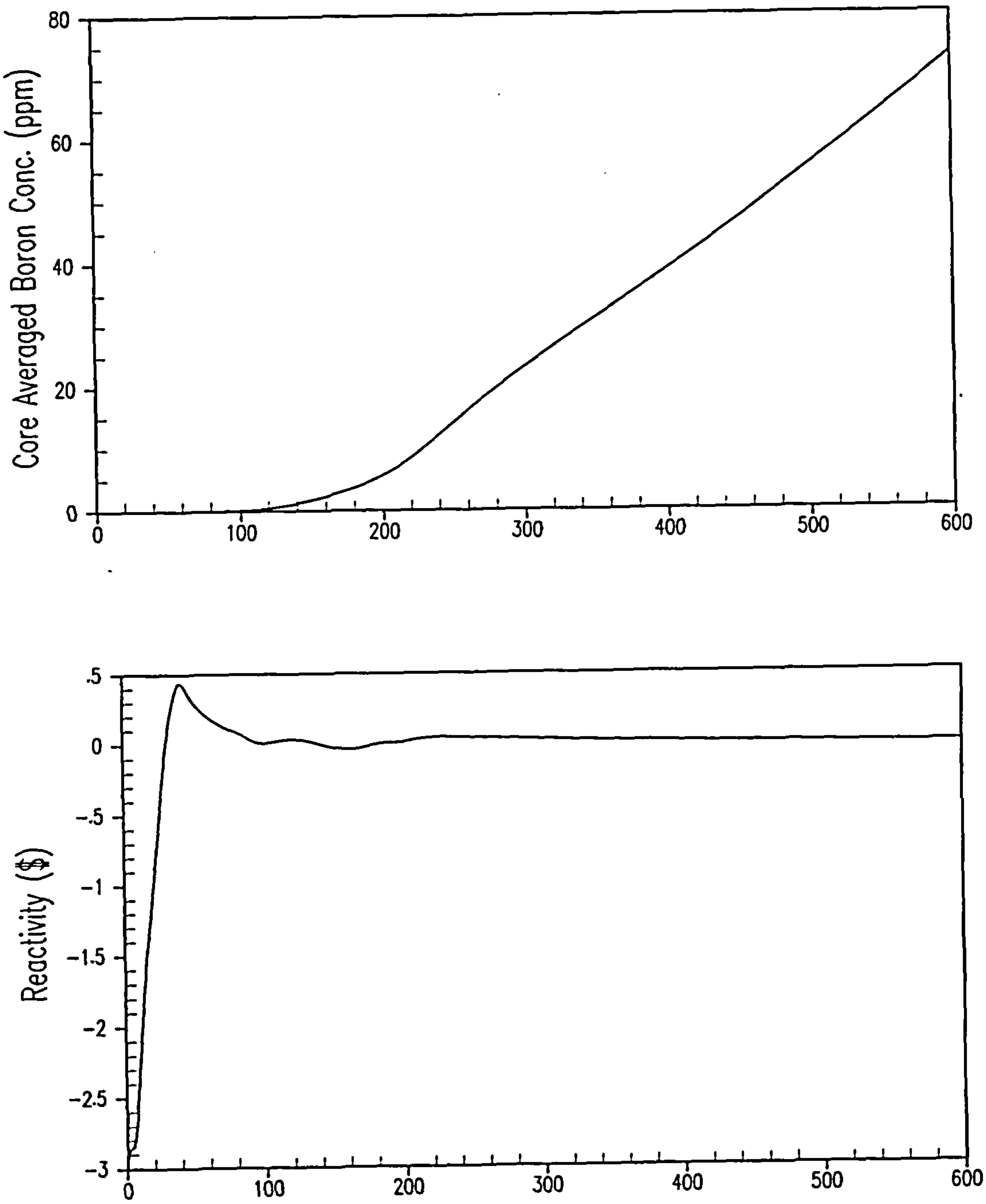


Figure 6.3.3-10 HZP Steam Line Break Transient without Offsite Power Double-Ended Rupture – Core Boron Concentration and Reactivity versus Time

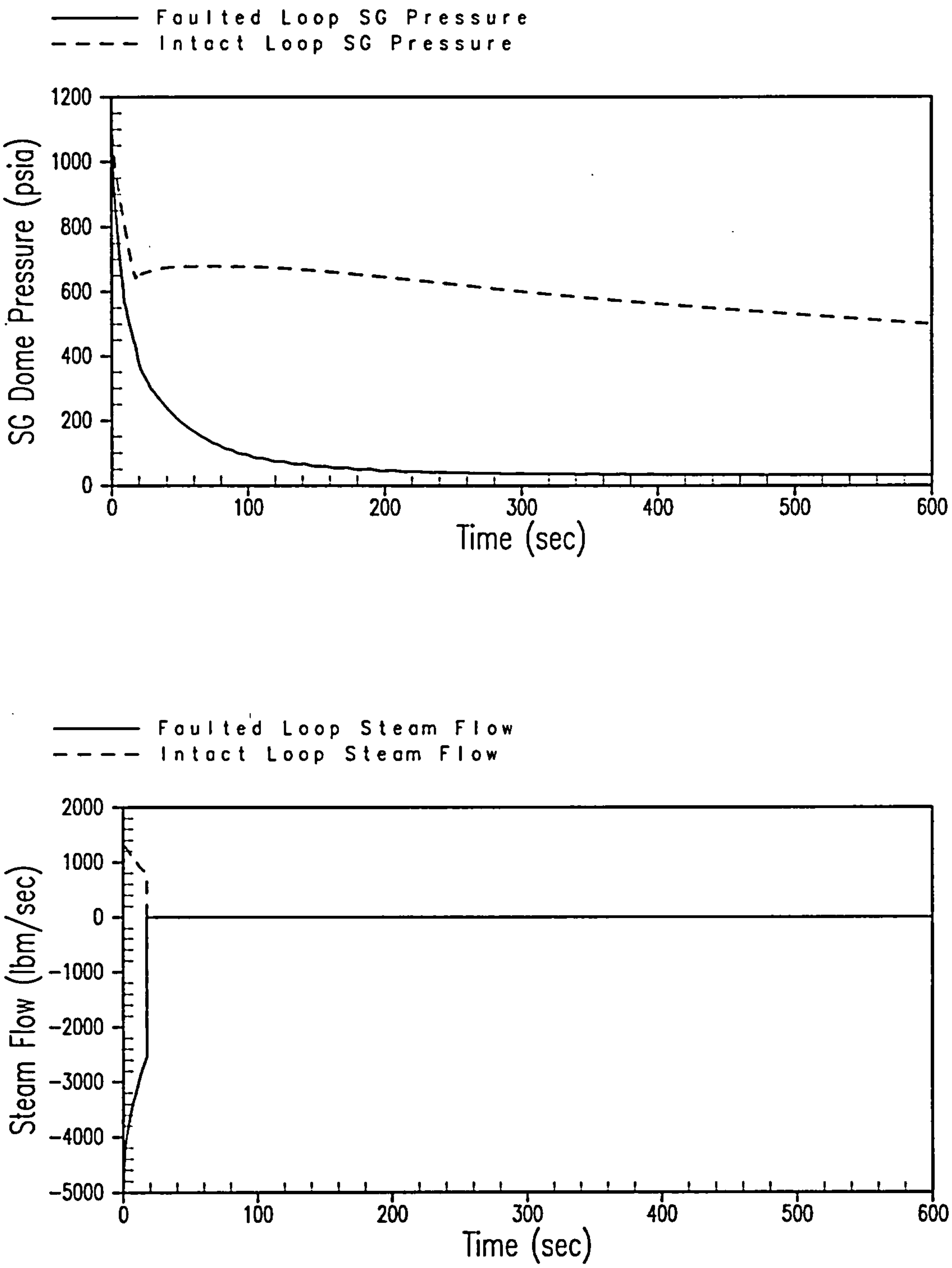


Figure 6.3.3-11 HZP Steam Line Break Transient without Offsite Power Double-Ended Rupture – Steam Pressure and Steam Flow versus Time

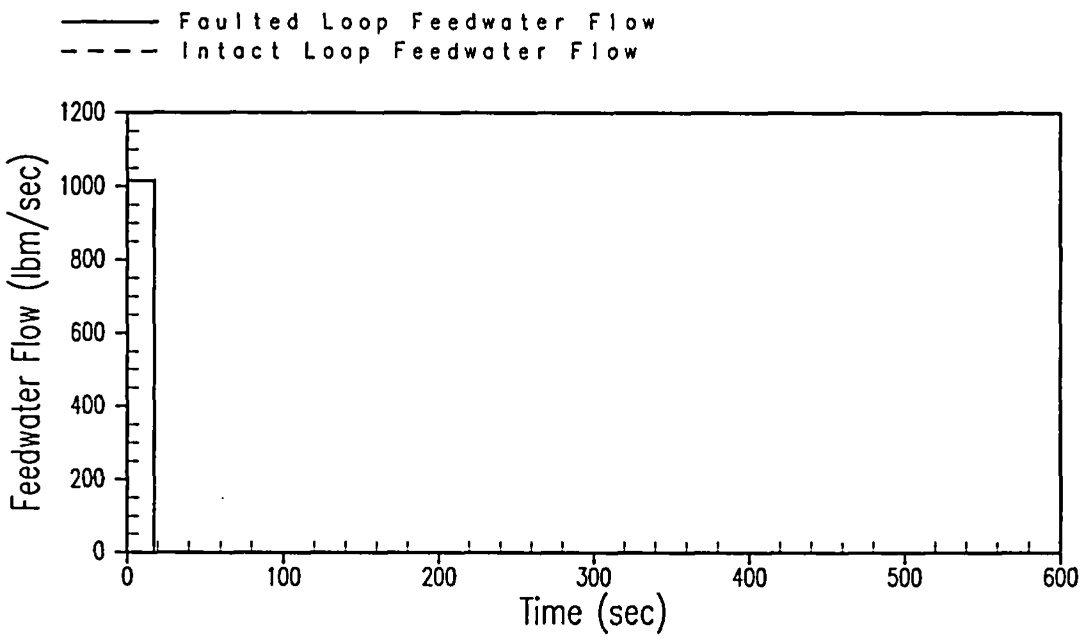


Figure 6.3.3-12 HZP Steam Line Break Transient without Offsite Power Double-Ended Rupture – Feedwater Flow versus Time

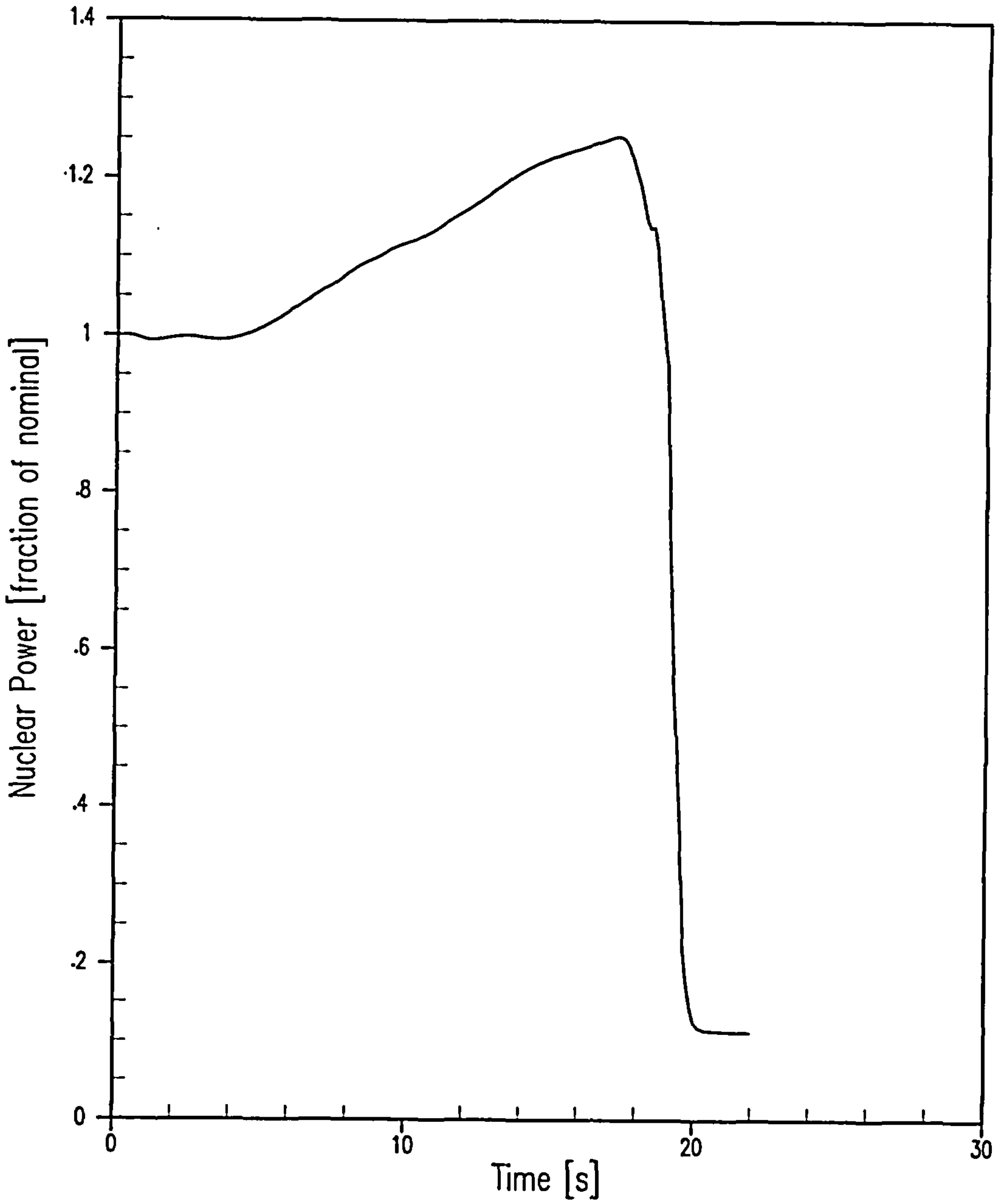


Figure 6.3.3A-1 Steam Line Rupture (Full-Power Core Response) – Nuclear Power versus Time

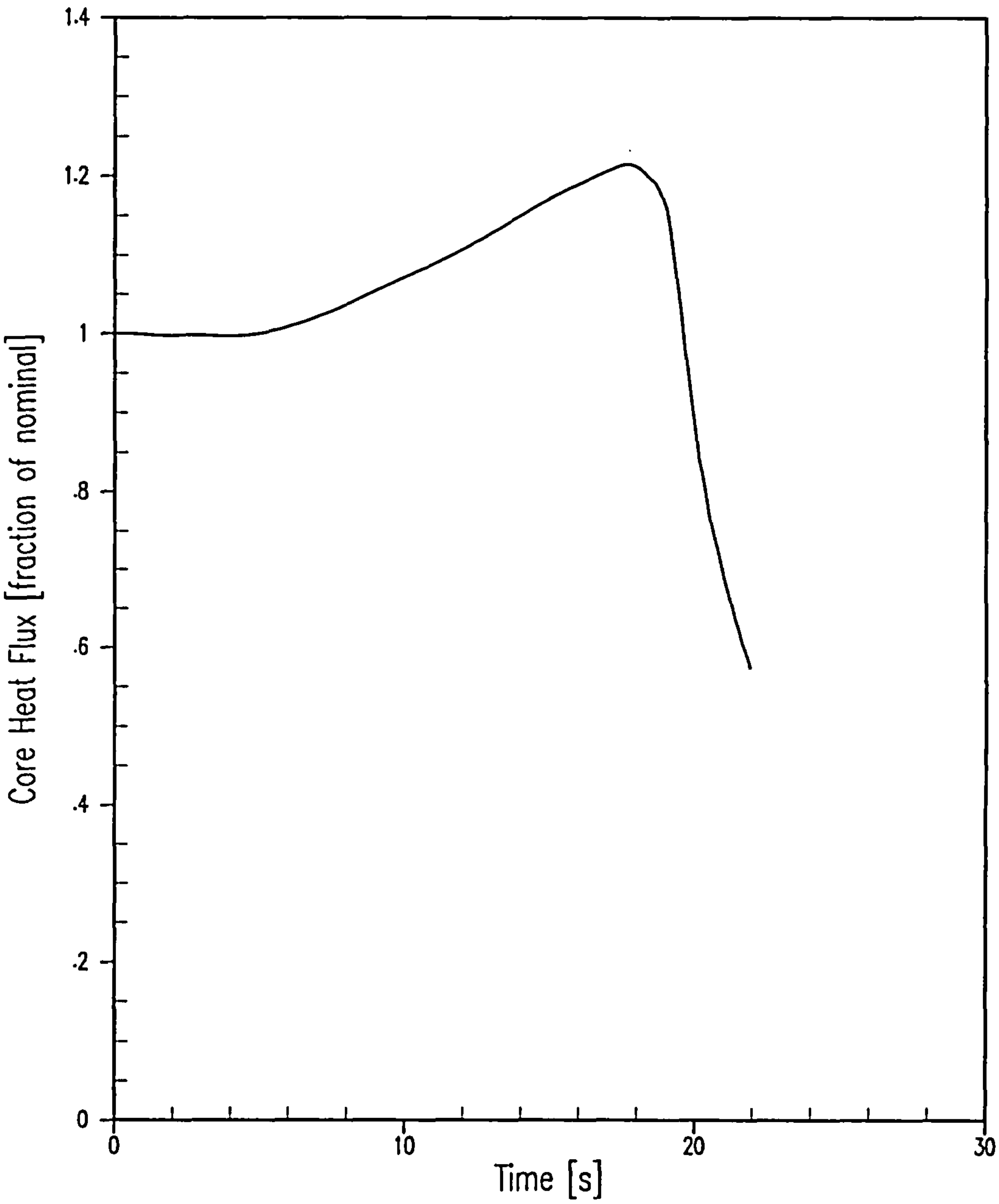


Figure 6.3.3A-2 Steam Line Rupture (Full-Power Core Response) – Core Heat Flux versus Time

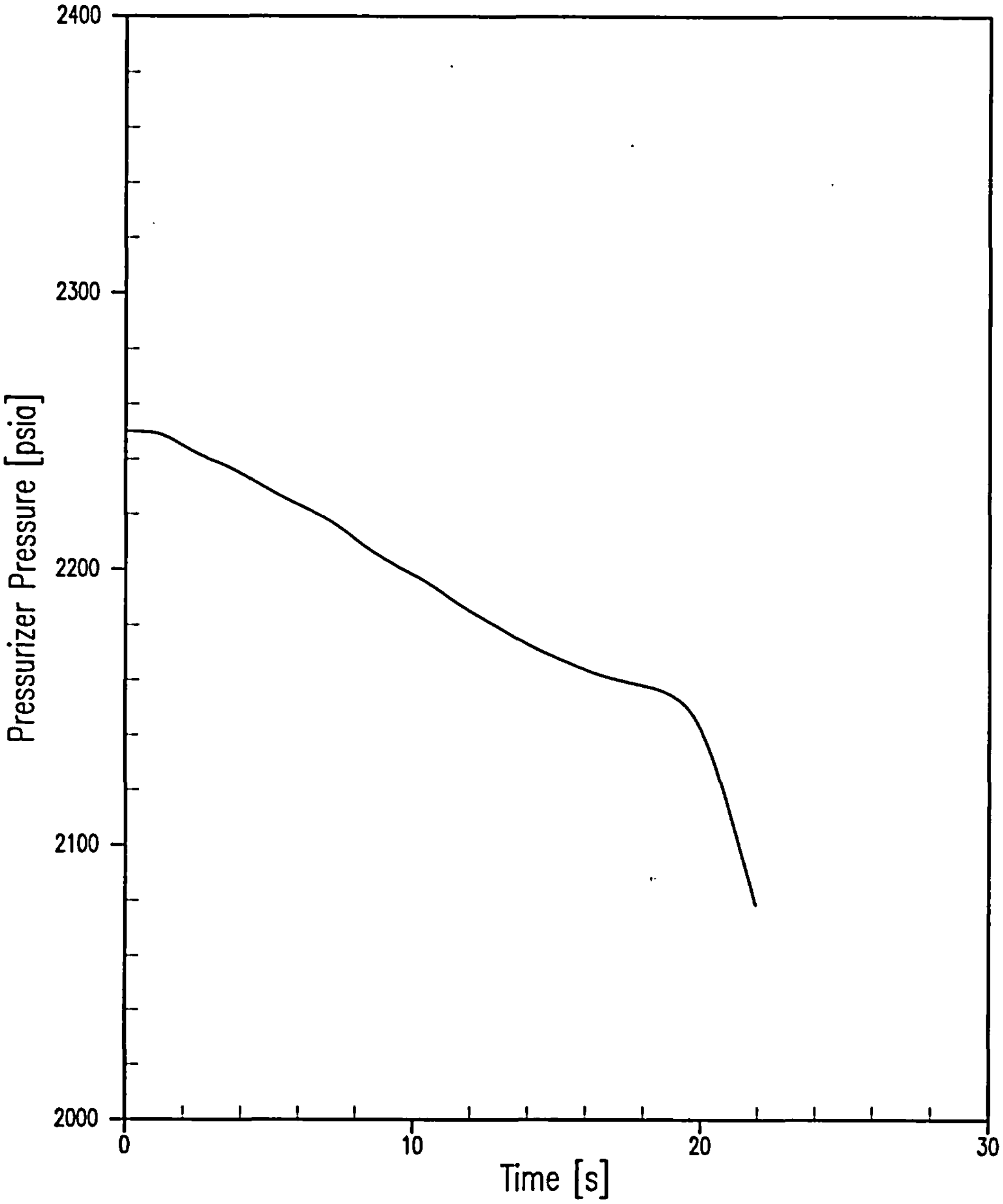


Figure 6.3.3A-3 Steam Line Rupture (Full-Power Core Response) – Pressurizer Pressure versus Time

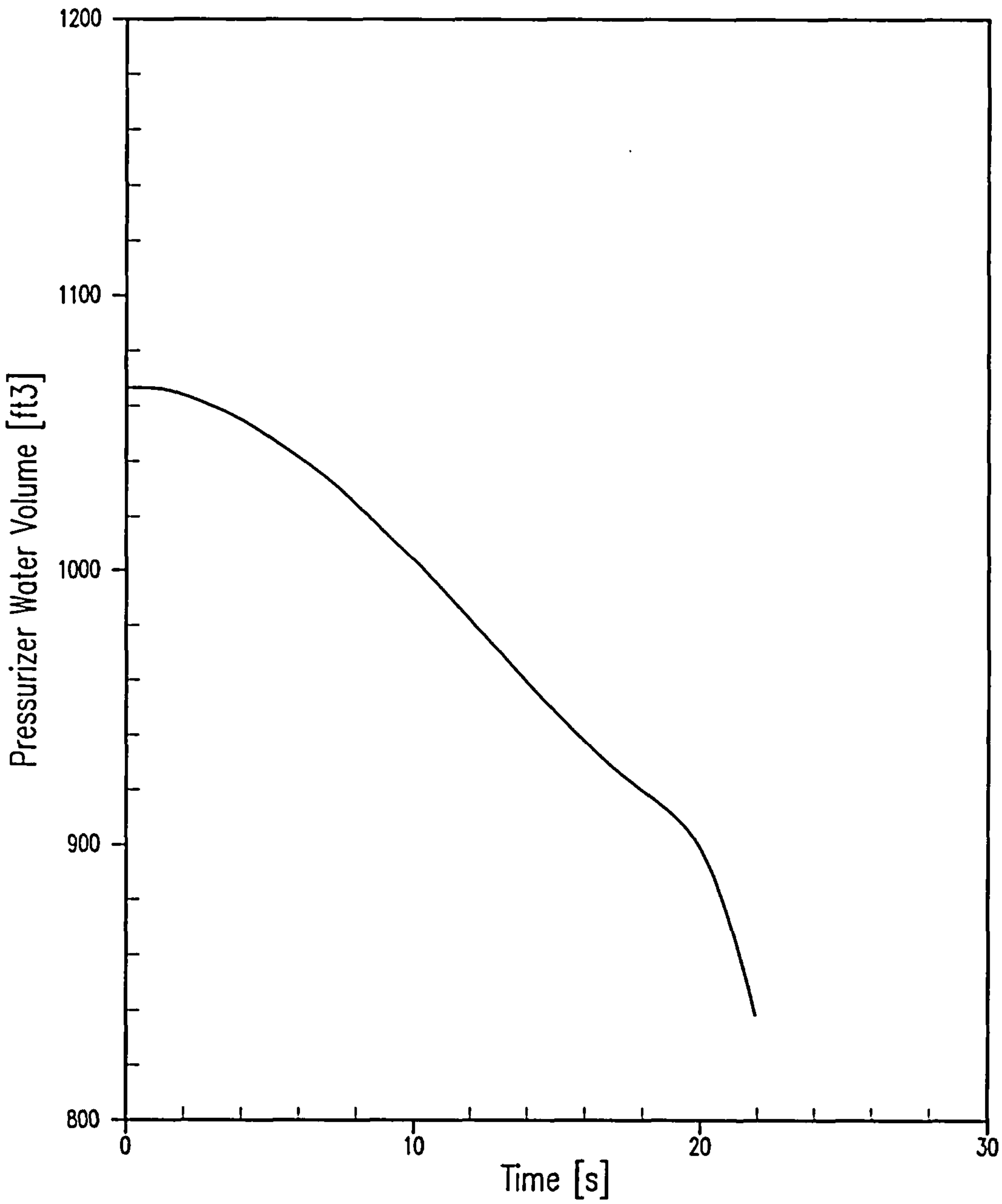


Figure 6.3.3A-4 Steam Line Rupture (Full-Power Core Response) – Pressurizer Water Volume versus Time

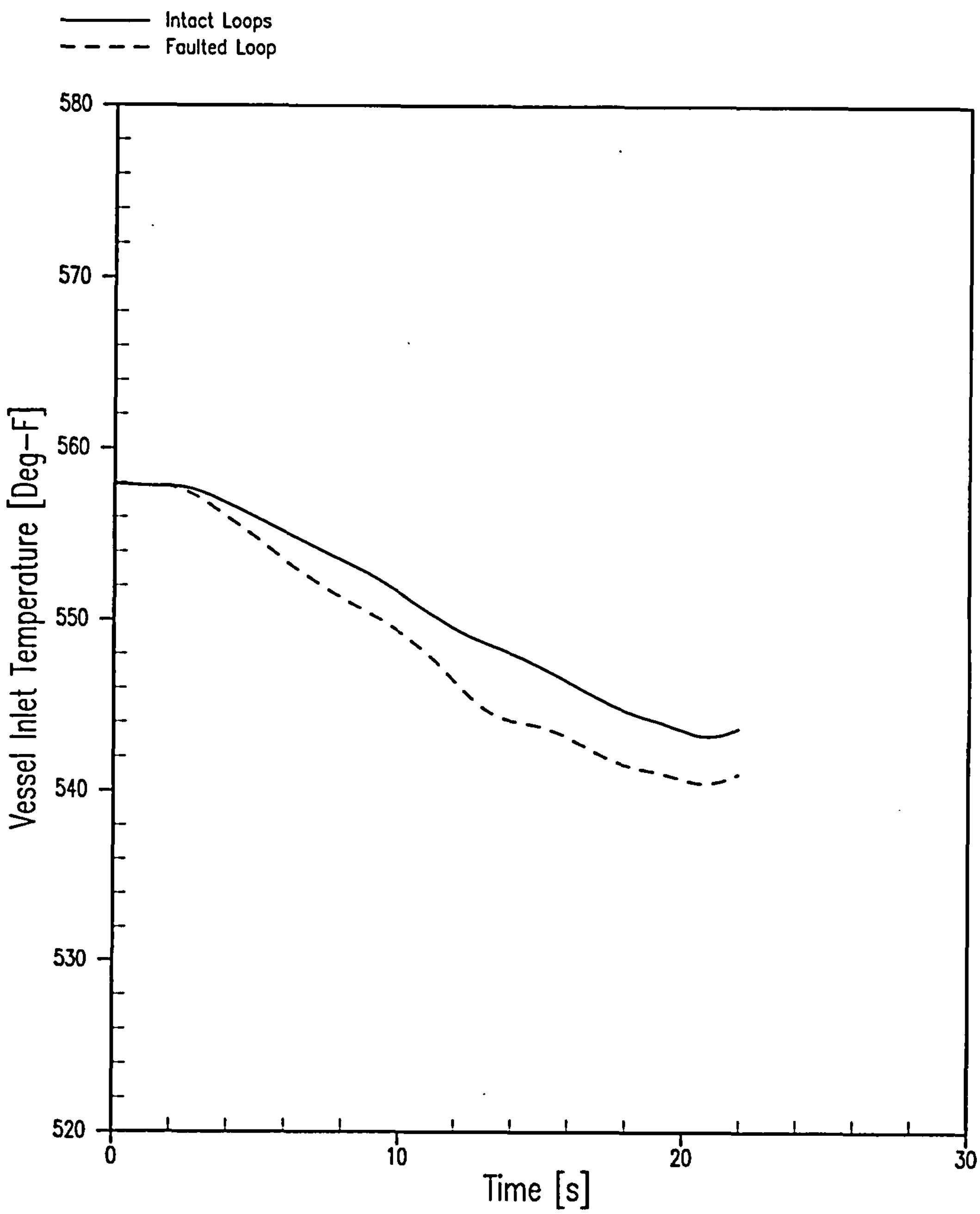


Figure 6.3.3A-5 Steam Line Rupture (Full-Power Core Response) – Reactor Vessel Inlet Temperatures versus Time

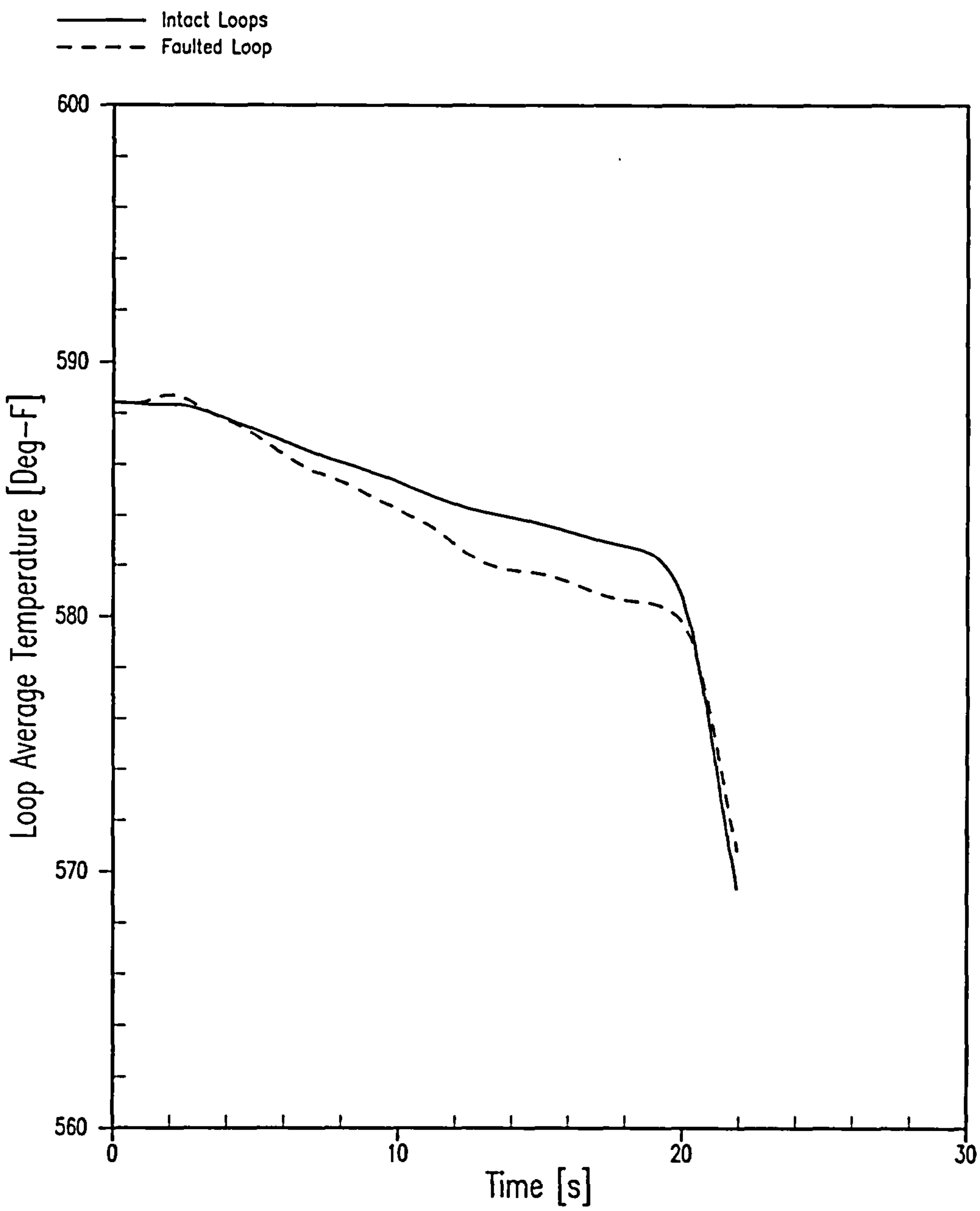


Figure 6.3.3A-6 Steam Line Rupture (Full-Power Core Response) – Loop Average Temperatures versus Time

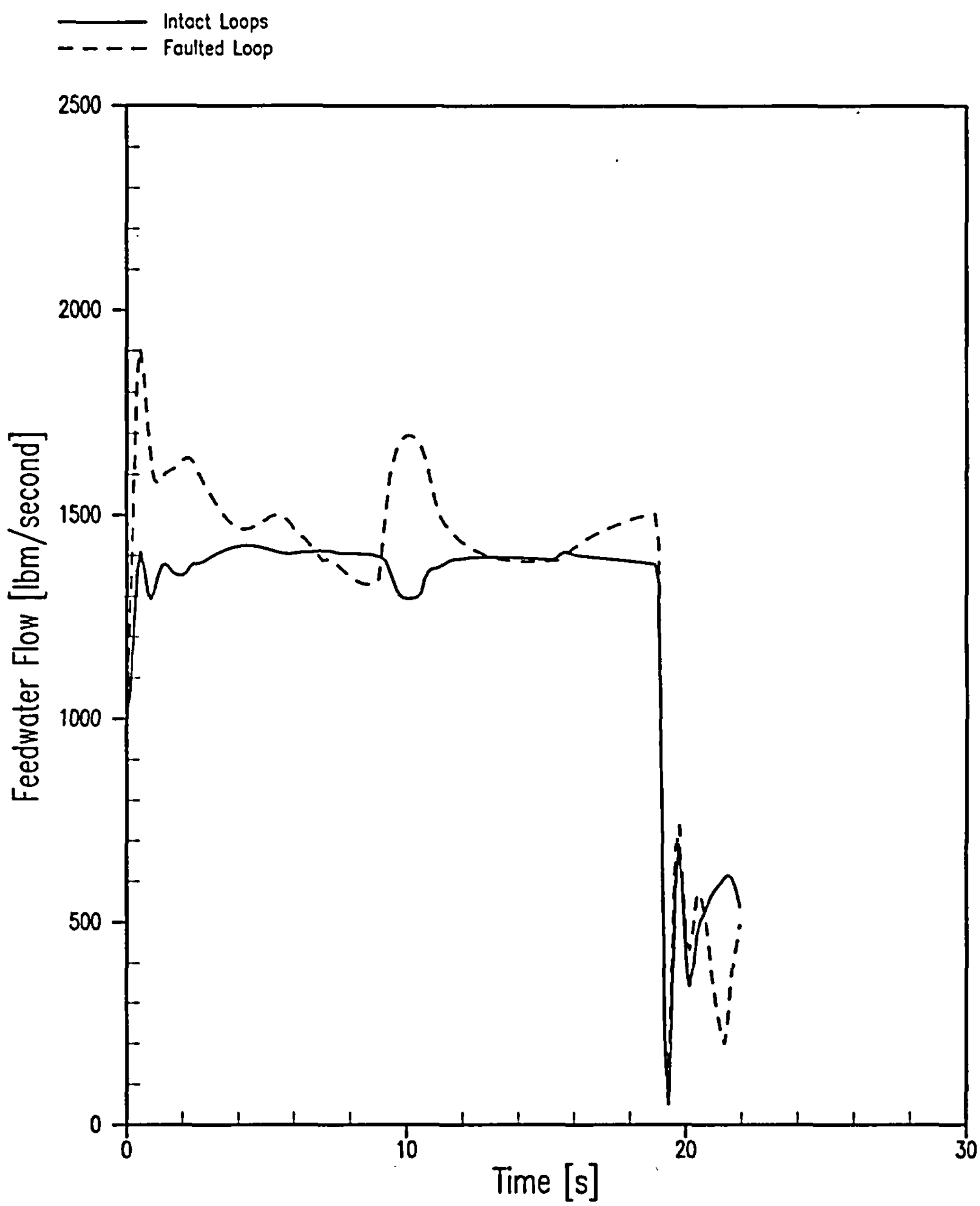


Figure 6.3.3A-7 Steam Line Rupture (Full-Power Core Response) – Feedwater Flow versus Time

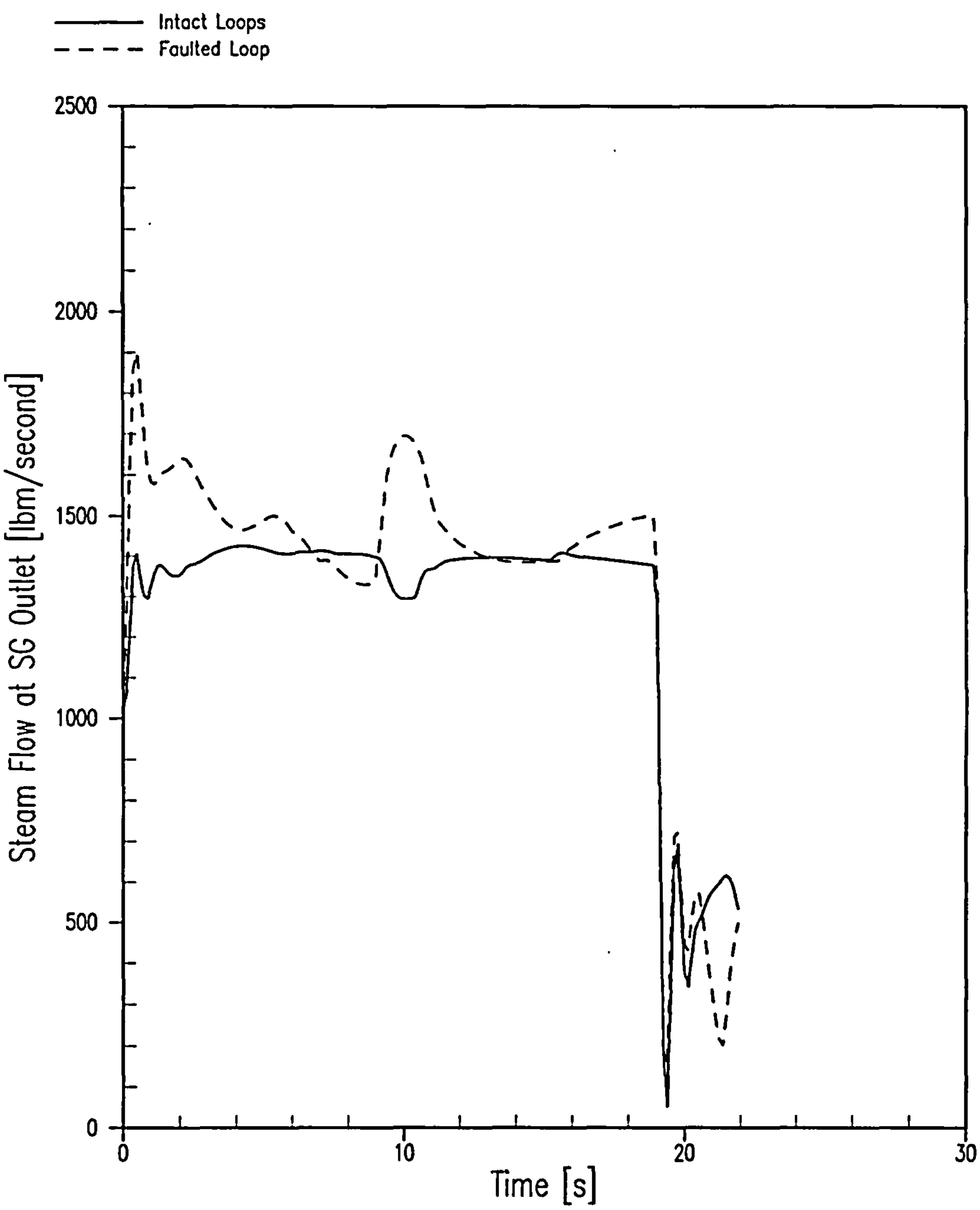


Figure 6.3.3A-8 Steam Line Rupture (Full-Power Core Response) – Steam Generator Outlet Steam Flow versus Time

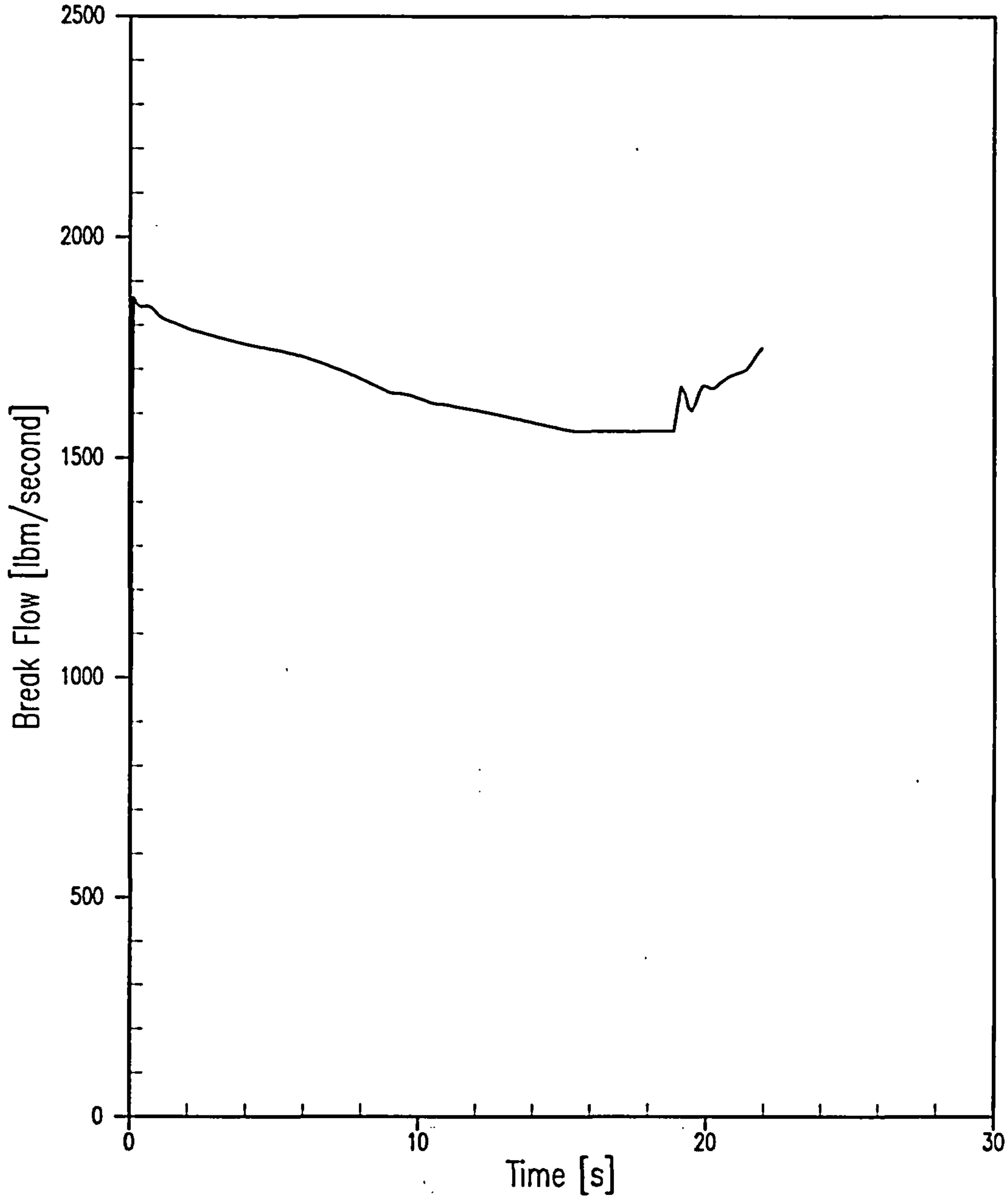


Figure 6.3.3A-9 **Steam Line Rupture (Full-Power Core Response) – Break Flow versus Time**

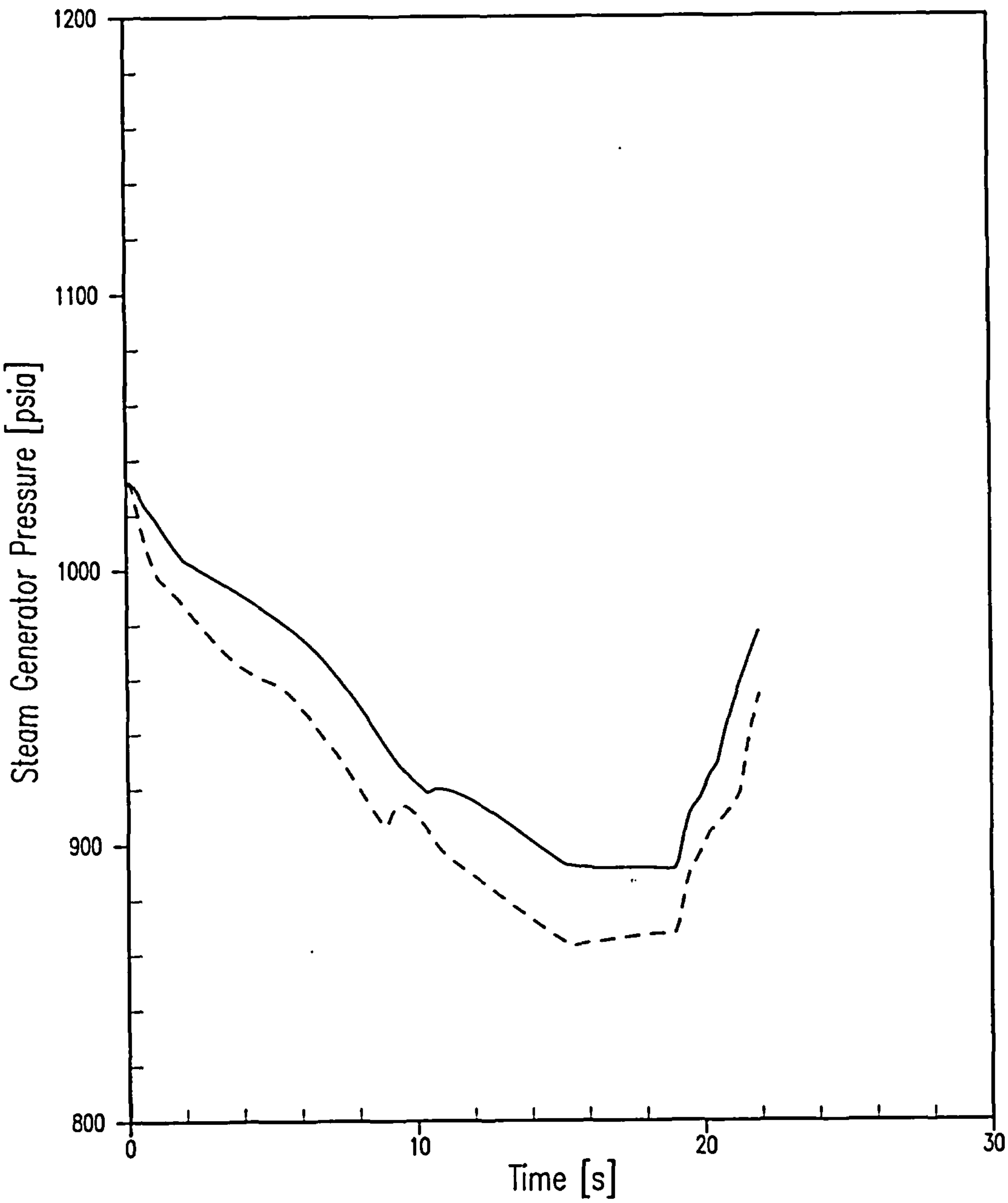


Figure 6.3.3A-10 Steam Line Rupture (Full-Power Core Response) – Steam Generator Pressure versus Time

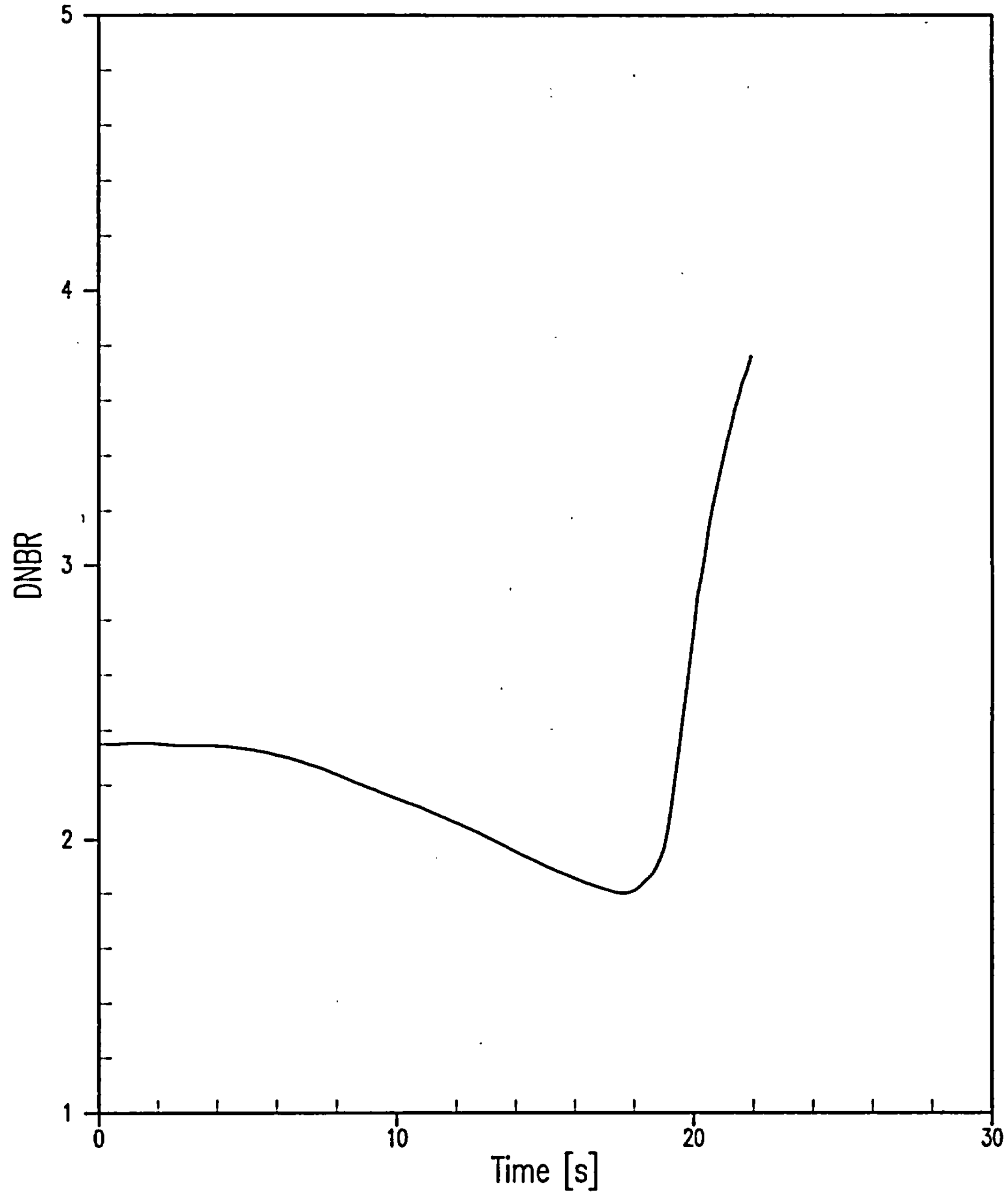


Figure 6.3.3A-11 Steam Line Rupture (Full-Power Core Response) – DNBR versus Time

6.3.4 Loss of External Electrical Load / Turbine Trip (FSAR Sections 15.2.2 - 15.2.5)

6.3.4.1 Accident Description

The loss of external electrical load event is defined as a complete loss of steam load or a turbine trip from full power without a direct reactor trip. This anticipated transient is analyzed as a turbine trip from full power because it bounds both events, the loss of external electrical load and turbine trip. The turbine trip event is more severe than the total loss of external electrical load event since it results in a more rapid reduction in steam flow.

For a turbine trip, the reactor would be tripped directly (unless below approximately 10-percent power) from a signal derived from either the turbine auto-stop oil pressure or a closure of the turbine stop valves. The automatic steam dump system accommodates the excess steam generation. Reactor coolant temperatures and pressures do not significantly increase if the steam dump system and pressurizer pressure control system are functioning properly. If the turbine condenser were not available, the excess steam generation would be dumped to the atmosphere. Additionally, main feedwater flow would be lost if the turbine condenser were not available. For this situation, steam generator level would be maintained by the auxiliary feedwater system (AFWS).

For a loss of external electrical load without subsequent turbine trip, no direct reactor trip signal would be generated. The plant would be expected to trip from the reactor protection system (RPS). A continued steam load of approximately 5 percent would exist after a total loss of external electrical load because of the steam demand of plant auxiliaries.

In the event of a large loss of load in which the steam dump valves fail to open or a complete loss of load with the steam dump operating, the main steam safety valves (MSSVs) may lift and the reactor may be tripped by any of the following signals: high pressurizer pressure, high pressurizer water level, overtemperature ΔT (OTΔT) and overpower ΔT (OPΔT), or low-low steam generator water level. The steam generator shell-side pressure and reactor coolant temperatures will increase rapidly. However, the pressurizer safety valves (PSVs) and MSSVs are sized to protect the reactor coolant system (RCS) and steam generators against overpressure for all load losses without assuming the operation of the steam dump system. The steam dump valves will not be opened for load reductions of 10 percent or less, but may open for larger load reductions. The RCS and main steam system (MSS) steam relieving capacities were designed to ensure safety of the unit without requiring automatic rod control, pressurizer pressure control, steam bypass control systems, or a reactor trip on turbine trip.

6.3.4.2 Method of Analysis

The loss of load transient is analyzed using the RETRAN computer code (Reference 1). The code is described in detail in Section 6.3.0.6 of this report.

The loss of load accident is analyzed for the following:

- To confirm that the PSVs and MSSVs are adequately sized to prevent overpressurization of the RCS and MSS, respectively

- To ensure that the increase in RCS temperature does not result in departure from nucleate boiling (DNB) in the core

The RPS is designed to automatically terminate any such transient before the DNB ratio (DNBR) falls below the limit value.

In this analysis, the behavior of the unit is evaluated for a complete loss of steam load from full power with no credit taken for a direct reactor trip on turbine trip. This assumption will delay reactor trip until conditions in the RCS cause a trip on some other signal. Therefore, the analysis assumes a worst-case transient and demonstrates the adequacy of the pressure-relieving devices and plant-specific RPS setpoints assumed in the analysis for this event.

Of the two cases analyzed, one is performed to address DNB concerns, and one ensures that the peak RCS pressure remains below the design limit (2,748.5 psia). Peak MSS pressures are reached in the DNBR case. The major assumptions for these cases are summarized as follows:

1. For the case analyzed to demonstrate that the core thermal limits are adequately protected (beginning-of-cycle (BOC) reactivity feedback conditions with automatic pressurizer pressure control), the loss of load accident is analyzed using the Revised Thermal Design Procedure (RTDP) (Reference 2). For this case, initial core power, reactor coolant temperature, and reactor coolant pressure are assumed to be at the nominal values consistent with steady-state full-power operation. Uncertainties in initial conditions are included in determining the DNBR limit value (Reference 2). For the case analyzed to demonstrate the adequacy of the primary pressure-relieving devices (BOC reactivity feedback conditions without automatic pressurizer pressure control), the loss of load accident is analyzed using the Standard Thermal Design Procedure (STDP). For this case, initial core power is assumed at the maximum value consistent with steady-state full-power operation, including allowances for calibration and instrument errors. Initial pressurizer pressure is assumed at the minimum value for this case, since it delays reactor trip on high pressurizer pressure and results in more severe primary-side temperature and pressure transients. This results in the maximum power difference for the loss of load. Callaway-specific calculations have shown that a reactor coolant temperature at the nominal high Tavg value consistent with steady-state full-power operation minus uncertainties results in conservative peak RCS pressure calculations.

2. The loss of load event results in a primary-system heatup and, therefore, is conservatively analyzed assuming minimum reactivity feedback consistent with BOC conditions. This includes assuming an a moderator temperature coefficient (MTC) value consistent with BOC hot-full-power (HFP) conditions (that is, zero MTC) and a least negative Doppler power coefficient (DPC). Maximum feedback (end-of-cycle – EOC) cases that were previously considered in the Final Safety Analysis Report (FSAR) are no longer analyzed since they have been determined (as part of the Westinghouse methodology for the analysis of this event) to be non-limiting with respect to the minimum DNBR, peak RCS pressure, and peak MSS pressure.

3. It is conservative to assume that the reactor is in manual control. If the reactor were in automatic control, the control rod banks would move prior to trip and reduce the severity of the transient.

4. No credit is taken for the operation of the steam dump system or steam generator power-operated relief valves (PORVs). The steam generator pressure rises to the safety valve setpoints, where steam release through the MSSVs limits the secondary-side steam pressure to the setpoint values. The MSSVs are explicitly modeled in the loss of load licensing basis analysis assuming a +3.0-percent tolerance with a 5 psi accumulation to full open. The MSSV model also assumes a 10 psi pressure drop from the steam generator exit to the MSSV inlet in determining the opening setpoints and an additional 15 psi pressure drop at full-open and full-flow conditions. Note that by maximizing the pressure transient in the MSS, the saturation temperature in the steam generators is maximized, resulting in limiting pressure and temperature conditions in the RCS.

5. The modeling of the pressurizer pressure control is as follows:

   a. For the case analyzed for DNB, automatic pressurizer pressure control is assumed. Therefore, full credit is taken for the effect of the pressurizer spray and PORVs in reducing or limiting the primary coolant pressure. Safety valves are also available and are modeled assuming a –2-percent setpoint tolerance.

   b. For the case analyzed for RCS overpressure concerns, it is assumed that automatic pressurizer pressure control is not available. Therefore, no credit is taken for the effect of the pressurizer spray or PORVs in reducing or limiting the primary coolant pressure. Safety valves are assumed operable, but are modeled assuming a +2-percent setpoint tolerance. The effects of the PSV loop seals (+1-percent set pressure shift and loop seal purgetime) are also conservatively modeled in the analysis, as described in Reference 4.

6. Main feedwater flow to the steam generators is assumed to be lost at the time of turbine trip. No credit is taken for AFW flow since a stabilized plant condition will be reached before AFW initiation is normally assumed to occur for full-power cases. However, the AFW pumps would be expected to start on a trip of the main feedwater pumps. The AFW flow would remove core decay heat following plant stabilization.

7. The analysis is performed assuming a maximum steam generator tube plugging (SGTP) level of 5 percent.

6.3.4.3 Results

The transient responses for a total loss of load from full-power operation are shown in Figures 6.3.4-1 through 6.3.4-11 for the 2 cases assuming BOC reactivity feedback conditions with and without automatic pressurizer pressure control (pressurizer spray and PORVs).

Figures 6.3.4-1 through 6.3.4-6 show the transient responses for the total loss of steam load at BOC (minimum feedback reactivity coefficients) assuming full credit for the pressurizer spray and PORVs to calculate the transient DNBR response. Following event initiation, the pressurizer pressure and average RCS temperature increase due to the rapidly reduced steam flow and heat removal capacity of the secondary side. The peak pressurizer pressure and water volume and RCS average temperature are reached shortly after the reactor is tripped by the OTΔT trip function. The DNBR initially increases slightly, then decreases until the reactor is tripped. Finally, following reactor trip, it increases rapidly.

The minimum DNBR remains well above the safety analysis limit value. The MSSVs actuate to limit the MSS pressure below 110 percent of the steam generator shell design pressure. Table 6.3.4-1 summarizes the sequence of events and limiting conditions for this case.

The total loss of load event is also analyzed assuming the plant to be initially operating at full power at BOC with no credit taken for the pressurizer spray or PORVs to maximize the RCS pressure response. Figures 6.3.4-7 through 6.3.4-11 show the transients for this case. The nuclear power remains relatively constant prior to reactor trip, while pressurizer pressure, pressurizer water volume, and RCS average temperature increase due to the sudden reduction in primary to secondary heat transfer. The reactor is tripped on the high pressurizer pressure trip signal. In this case, the PSVs are actuated and maintain the primary side pressure below 110 percent of the design value. The MSSVs actuate to limit the MSS pressure below 110 percent of the steam generator shell design pressure. Table 6.3.4-2 summarizes the sequence of events and limiting conditions for this case.

6.3.4.4 Conclusions

The results of the analyses show that the plant design is such that a total loss of external electrical load without a direct or immediate reactor trip presents no hazard to the integrity of the RCS or MSS. Pressure-relieving devices that have been incorporated into the plant design are adequate to limit the maximum pressures to within the safety analysis limits (that is, 2,748.5 psia for the RCS and 1,318.5 psia for the MSS).

The integrity of the core is maintained by operation of the RPS (that is, the minimum DNBR is maintained above the safety analysis limit value of 1.55). Therefore, no core safety limit will be violated as a result of implementing the RSG Program.

Table 6.3.4-1 Time Sequence of Events for Loss of External Electrical Load Event with Pressurizer Pressure Control (for Minimum DNBR)	
Event	**Time (Seconds)**
Turbine Trip	0.0
Reactor Trip on OTΔT	11.8
Rod Motion Begins	13.8
Time of Minimum DNBR	14.0
Time of Peak MSS Pressure	17.0
Results	
Minimum DNBR Value	1.900
DNBR Limit	1.55
Peak MSS Pressure	1,294.0 psia
MSS Pressure Limit	1,318.5 psia

Table 6.3.4-2 Time Sequence of Events for Loss of External Electrical Load Event without Pressurizer Pressure Control (for RCS Overpressure)	
Event	**Time (Seconds)**
Turbine Trip	0.0
Reactor Trip on High Pressurizer Pressure	5.6
Rod Motion Begins	6.6
Time of Peak RCS Pressure	8.7
Time of Peak MSS Pressure	13.2
Results	
Peak RCS Pressure	2,731.6 psia
RCS Pressure Limit	2,748.5 psia
Peak MSS Pressure	1,270.5 psia
MSS Pressure Limit	1,318.5 psia

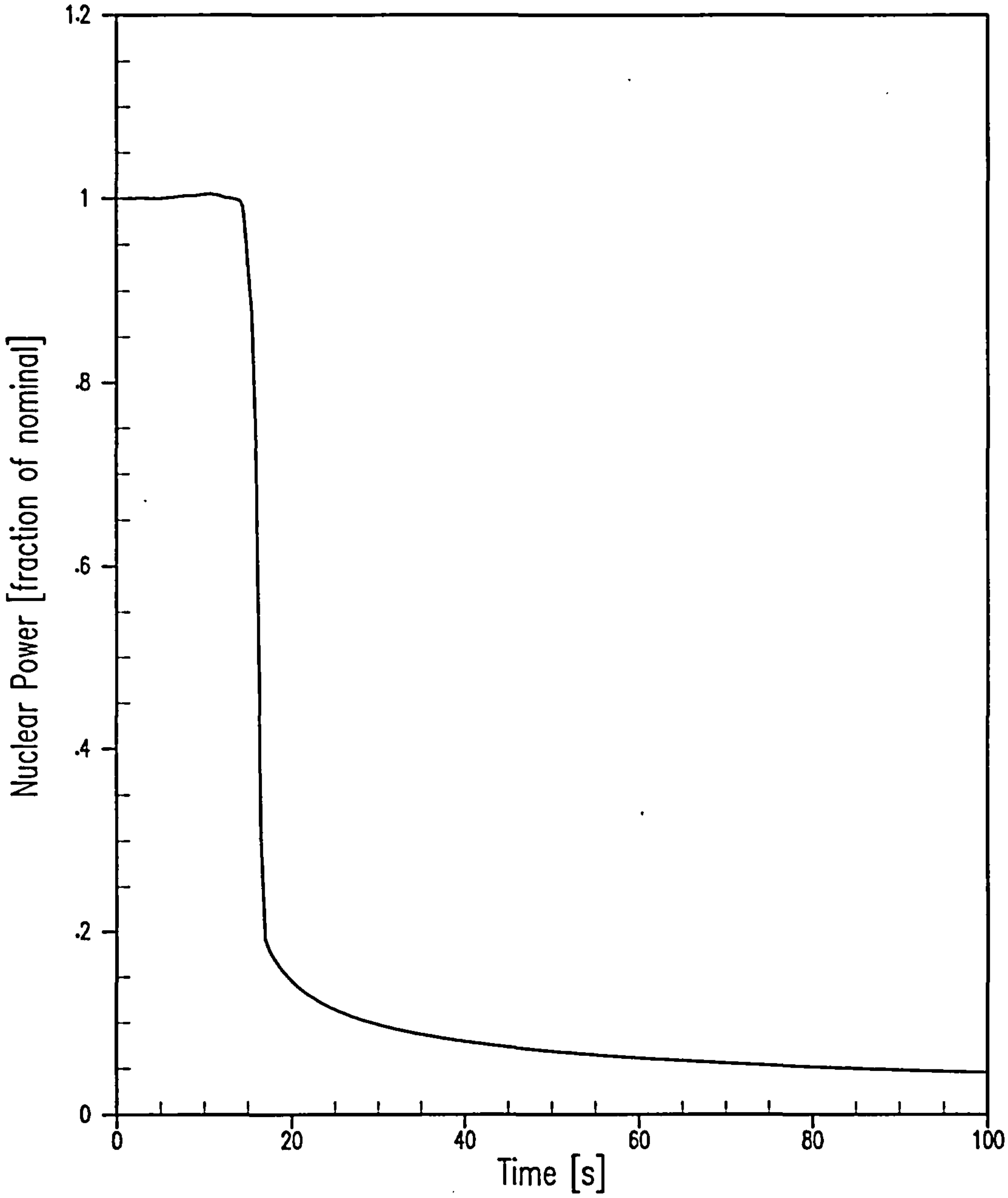


Figure 6.3.4-1 **Loss of External Electrical Load with Automatic Pressure Control (DNB Case) – Nuclear Power versus Time**

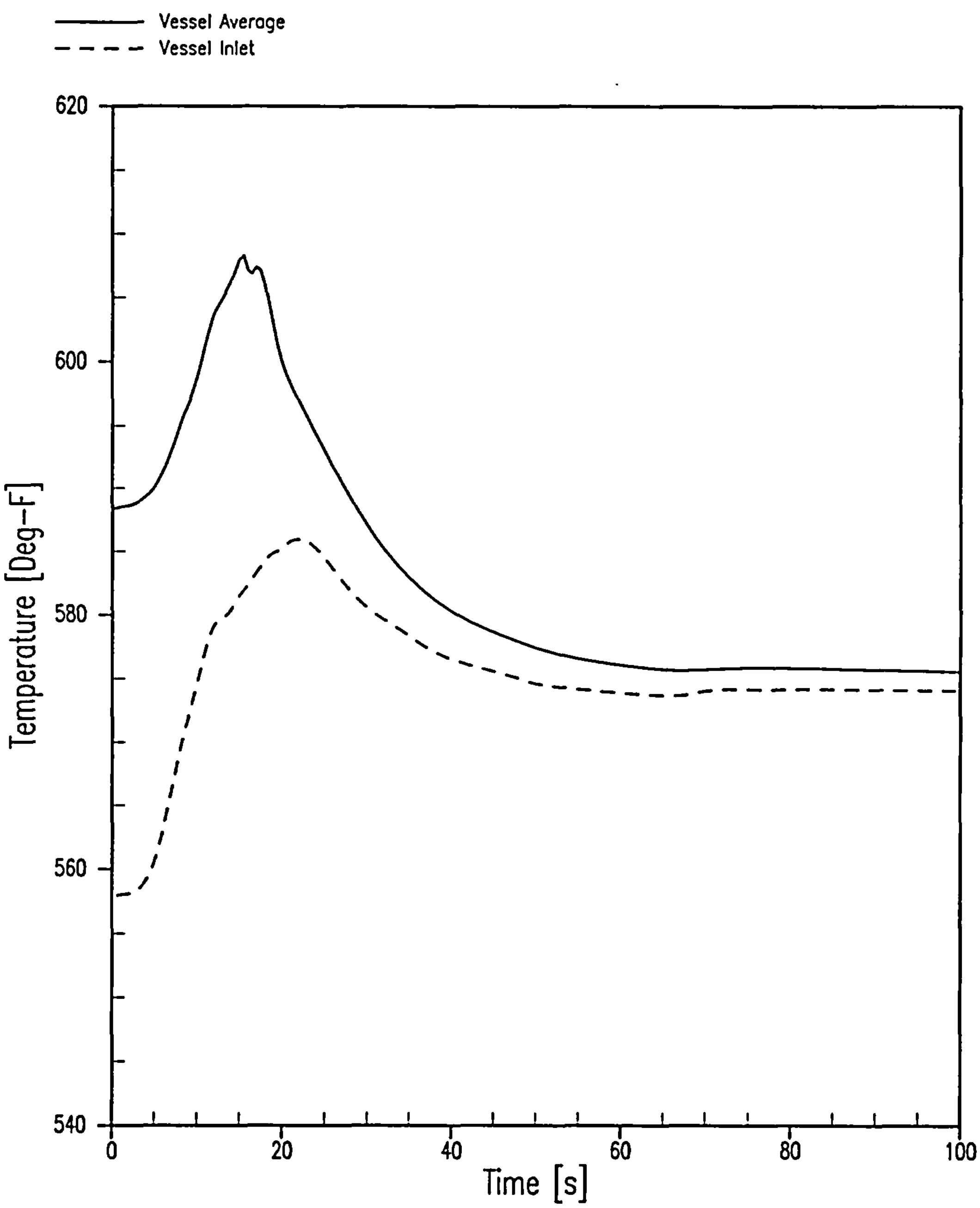


Figure 6.3.4-2 Loss of External Electrical Load with Automatic Pressure Control (DNB Case) – Vessel Average and Vessel Inlet Temperatures versus Time

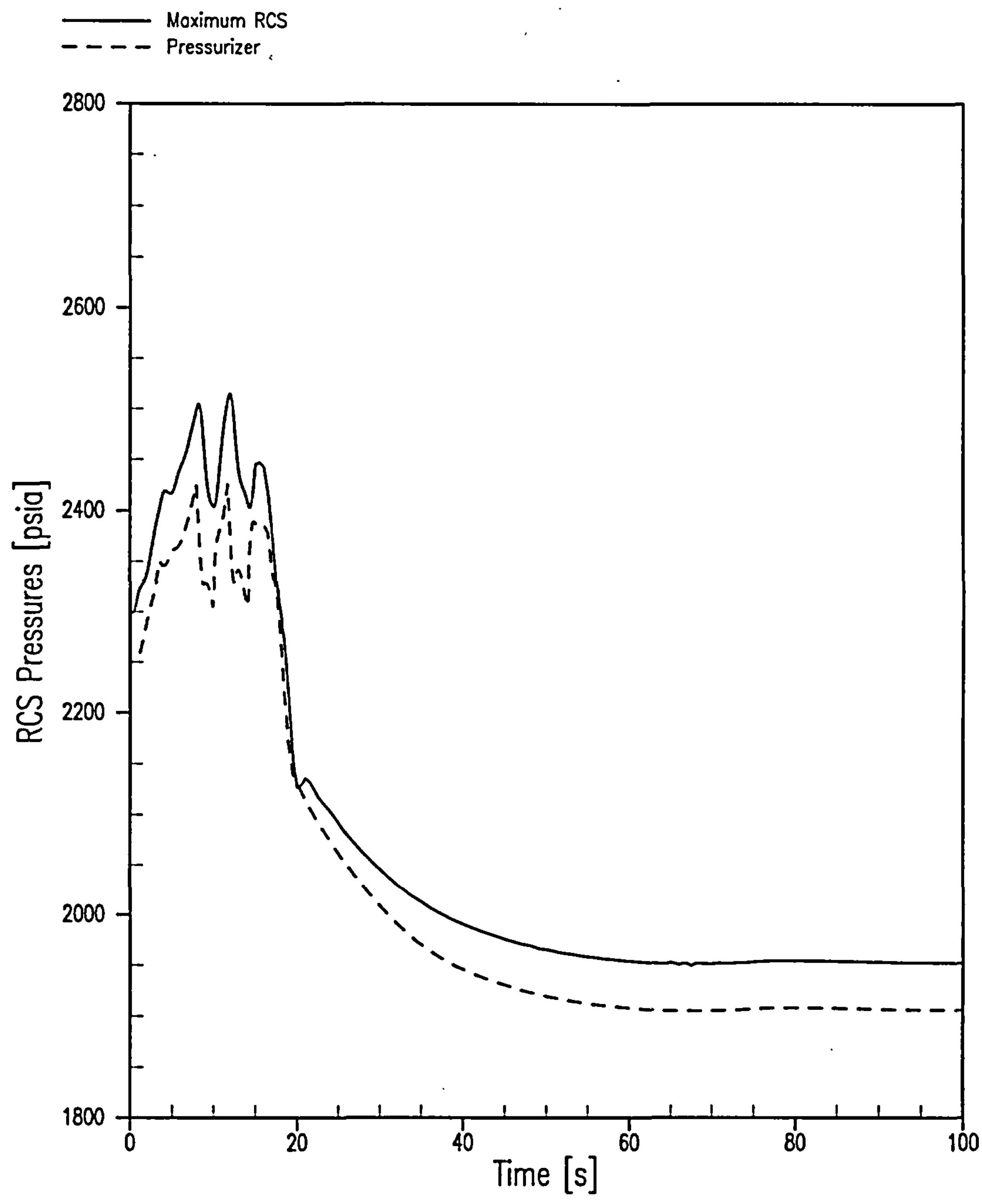


Figure 6.3.4-3 **Loss of External Electrical Load with Automatic Pressure Control (DNB Case) – Maximum RCS and Pressurizer Pressures versus Time**

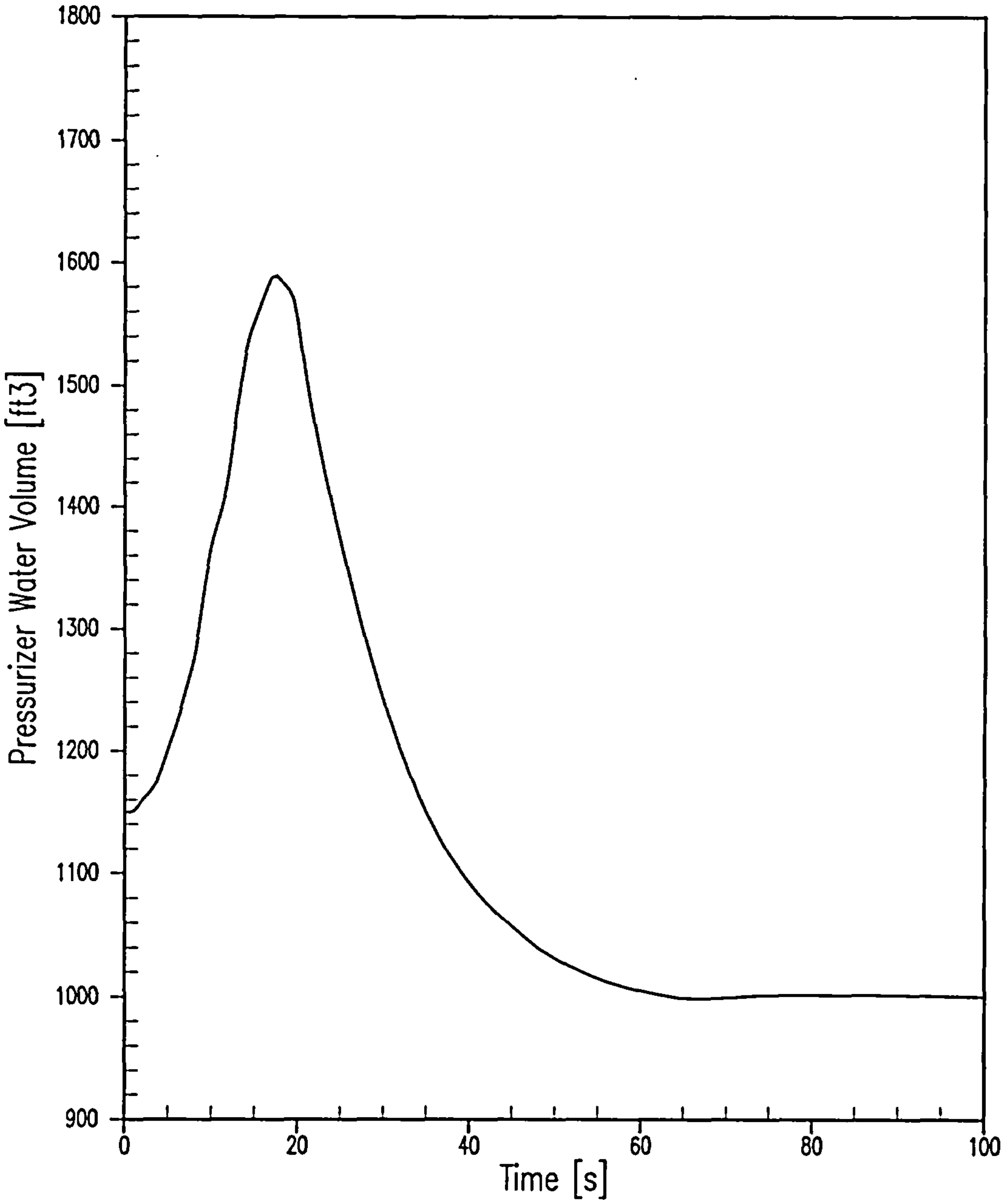


Figure 6.3.4-4 **Loss of External Electrical Load with Automatic Pressure Control (DNB Case) – Pressurizer Water Volume versus Time**

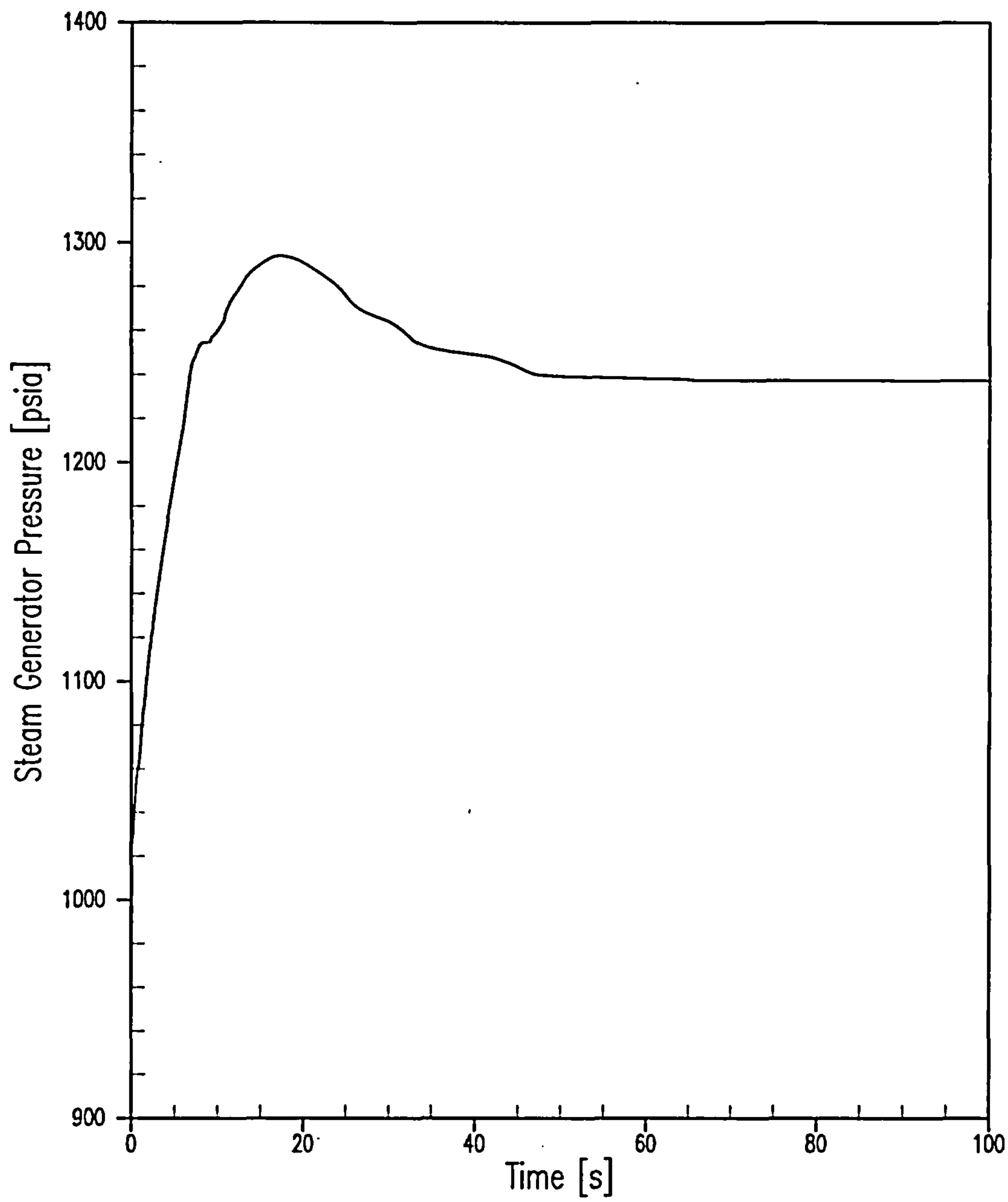


Figure 6.3.4-5 **Loss of External Electrical Load with Automatic Pressure Control (DNB Case) – Steam Generator Pressure versus Time**

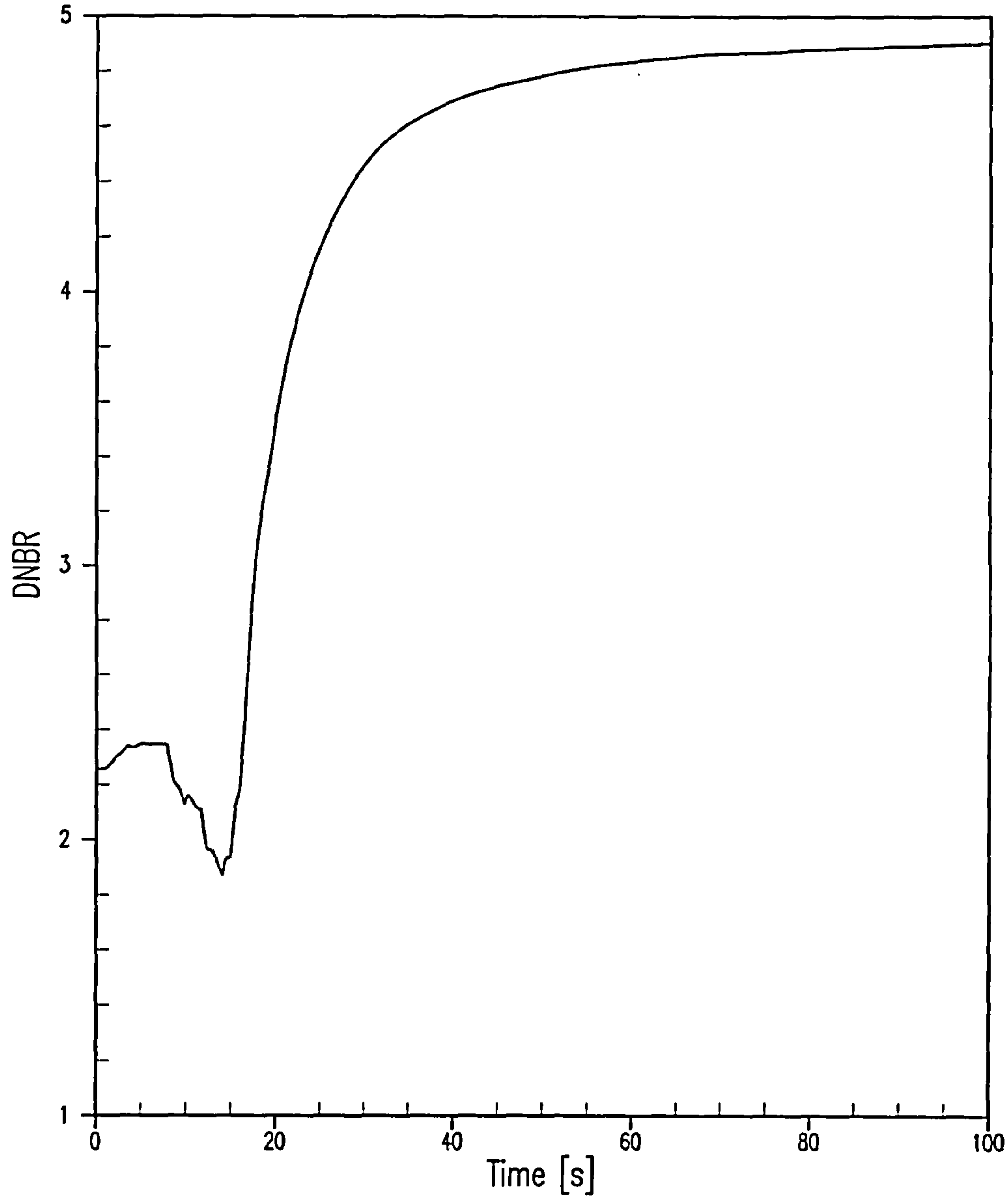


Figure 6.3.4-6 **Loss of External Electrical Load with Automatic Pressure Control (DNB Case) – DNBR versus Time**

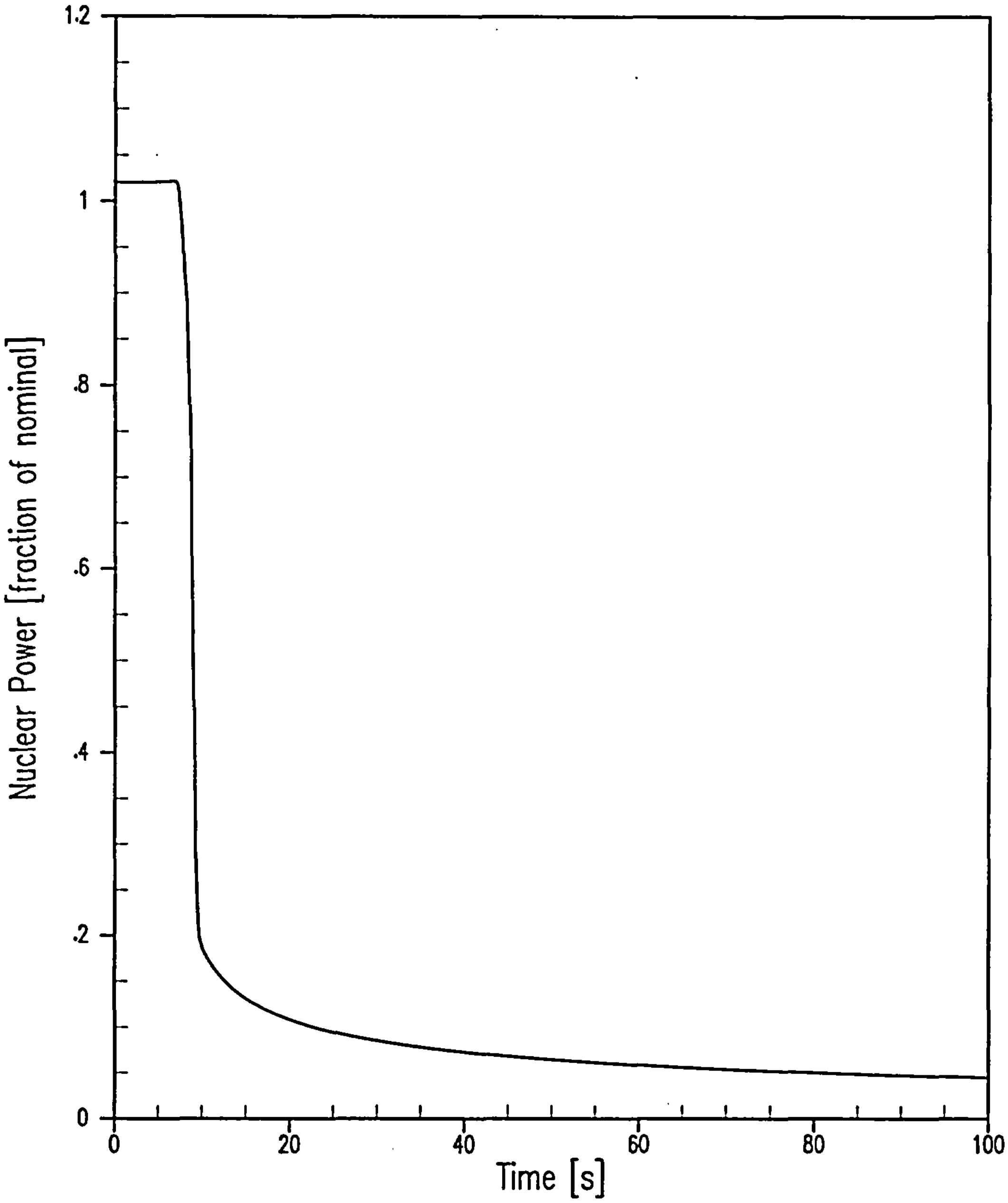


Figure 6.3.4-7 **Loss of External Electrical Load Without Automatic Pressure Control (RCS Overpressure Case) – Nuclear Power versus Time**

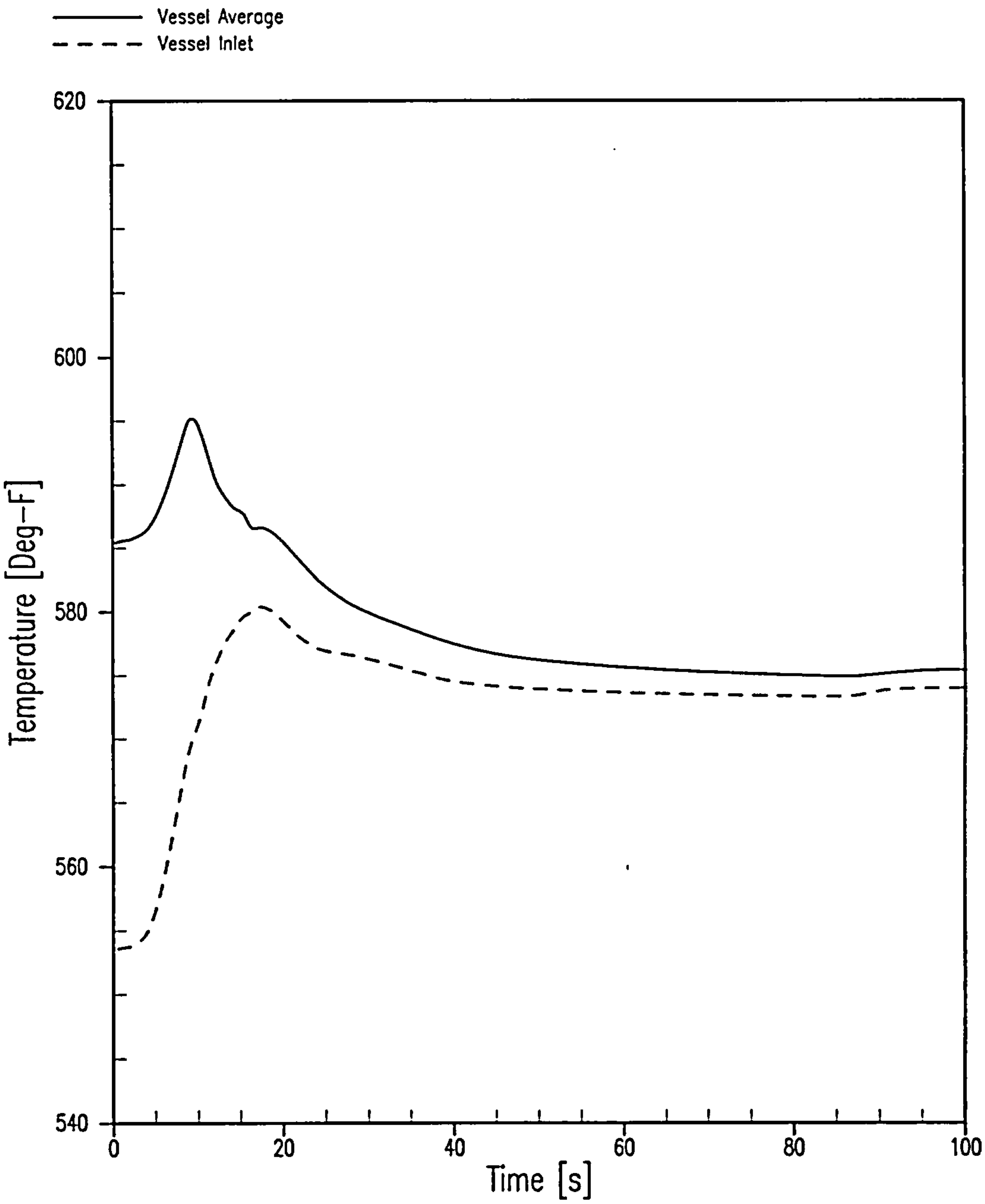


Figure 6.3.4-8 **Loss of External Electrical Load Without Automatic Pressure Control (RCS Overpressure Case) – Vessel Average and Vessel Inlet Temperatures versus Time**

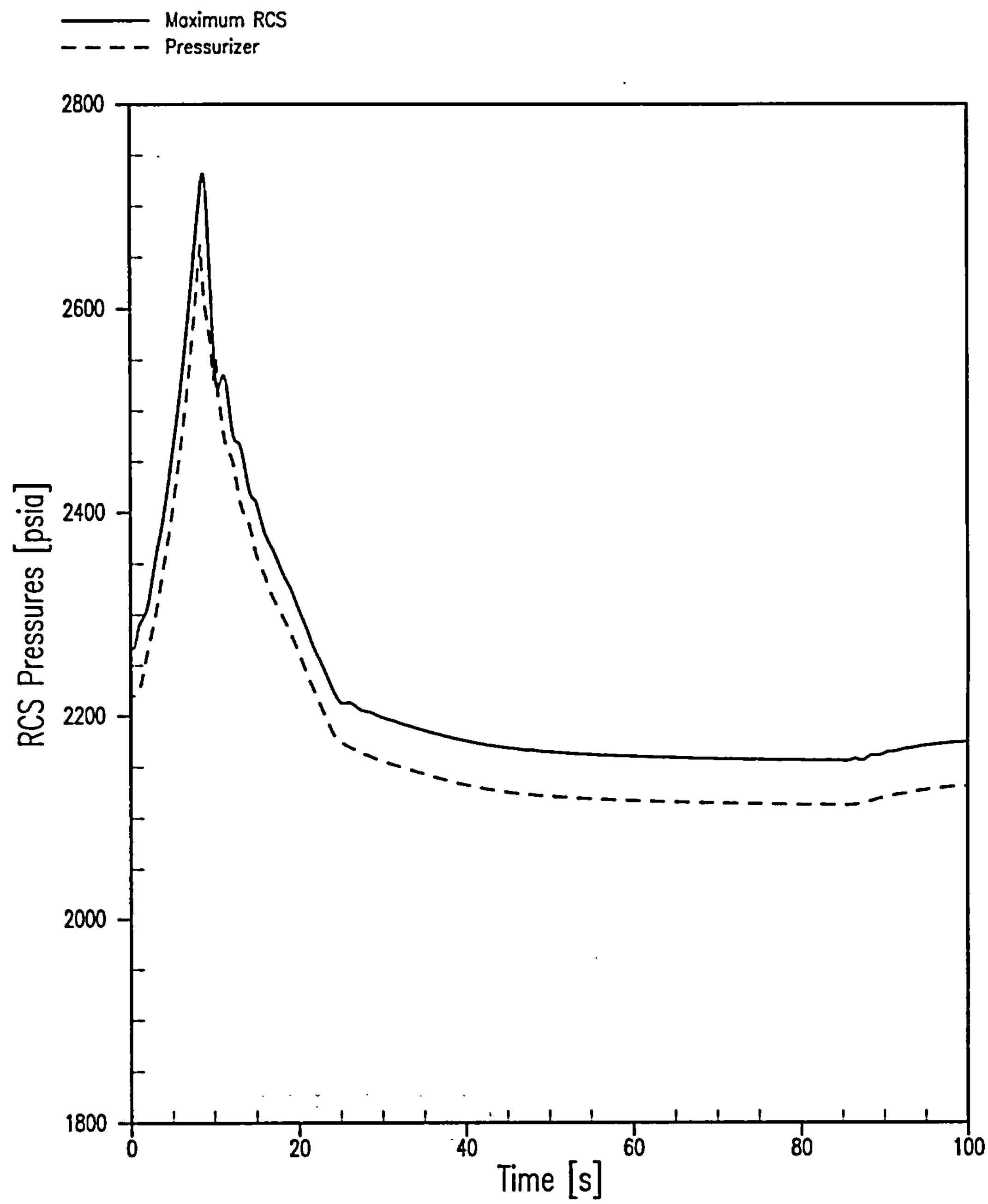


Figure 6.3.4-9 **Loss of External Electrical Load Without Automatic Pressure Control (RCS Overpressure Case) – Maximum RCS and Pressurizer Pressures versus Time**

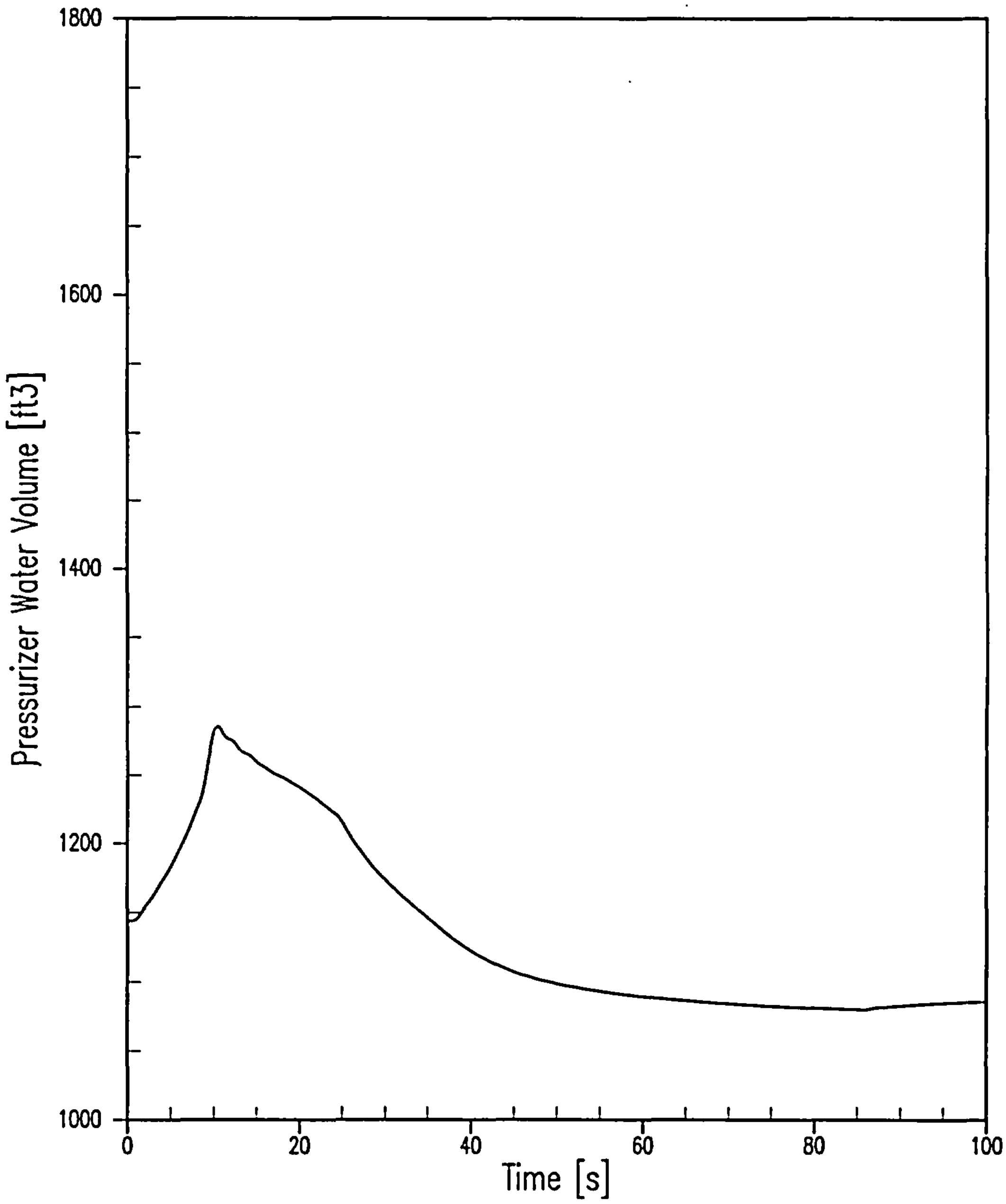


Figure 6.3.4-10 Loss of External Electrical Load Without Automatic Pressure Control (RCS Overpressure Case) – Pressurizer Water Volume versus Time

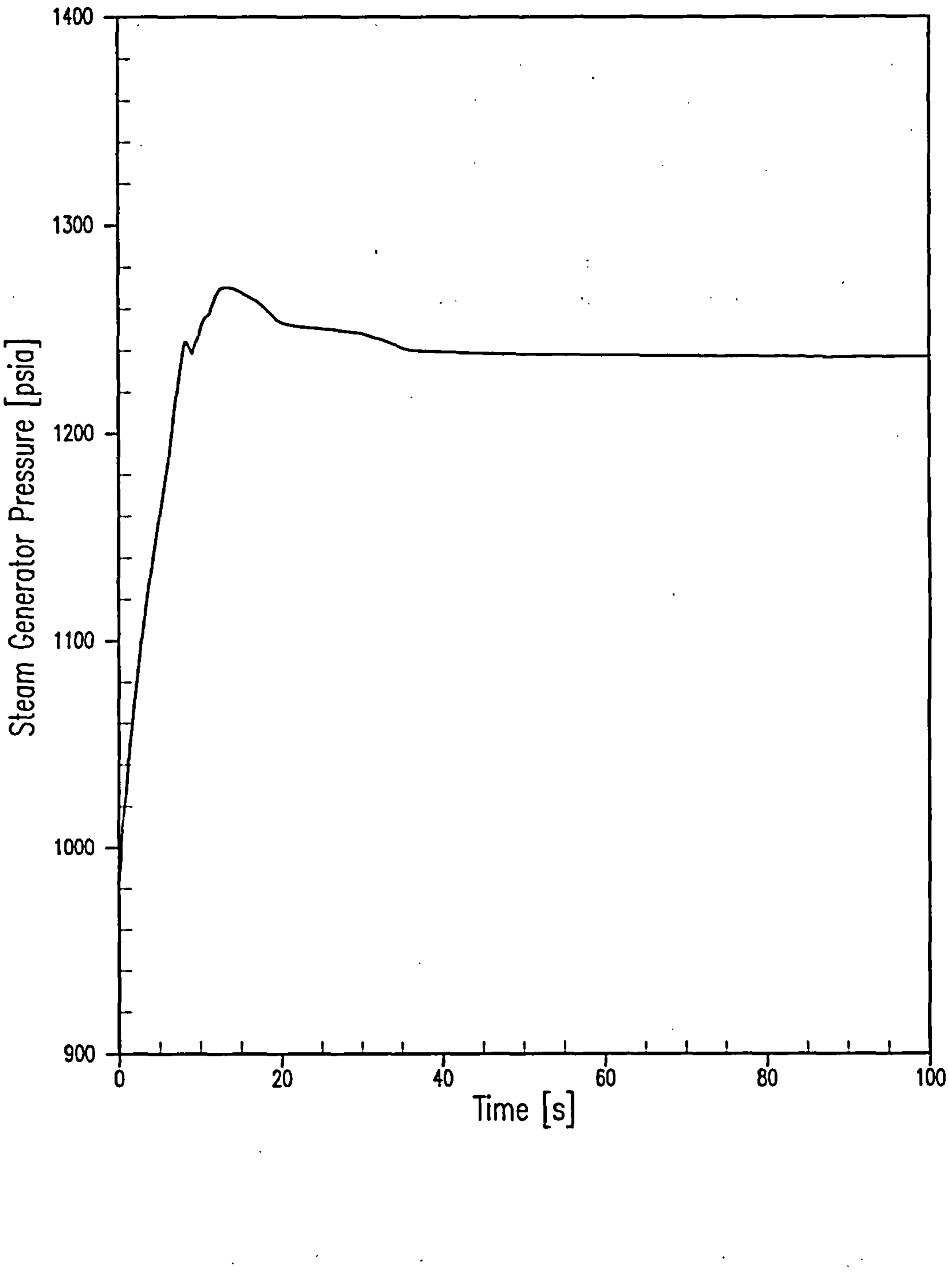


Figure 6.3.4-11 **Loss of External Electrical Load Without Automatic Pressure Control (RCS Overpressure Case) – Steam Generator Pressure versus Time**

6.3.5 Loss of Non-Emergency AC Power to the Station Auxiliaries/Loss of Normal Feedwater Flow (FSAR Sections 15.2.6 and 15.2.7)

6.3.5.1 Accident Description

A loss of normal feedwater (LONF) (from a pipe break, pump failure, or valve malfunction) results in a reduction of the ability of the secondary system to remove the heat generated in the reactor core. If the reactor were not tripped during this accident, core damage could possibly occur from a sudden loss of heat sink. If an alternate supply of feedwater were not supplied to the steam generators, residual heat following reactor trip and reactor coolant pump (RCP) heat would heat the primary system water to the point where water relief from the pressurizer would occur, resulting in a substantial loss of water from the reactor coolant system (RCS). A significant loss of water from the RCS could conceivably lead to core damage. Controlled operation of the reactor and RCS with a water-solid pressurizer is also very challenging. Since the reactor is tripped well before the steam generator heat transfer capability is reduced, the primary system never approaches a condition where the DNBR limit may be violated.

The RPS provides the protection against a LONF event via a reactor trip on low-low steam generator water level in one or more steam generator.

The auxiliary feedwater system (AFWS) is started automatically as described below. Two motor-driven AFW pumps are available for delivery of AFW; these are started on:

- Low-low steam generator water level in two-out-of-three level channels in one or more steam generators
- Any safety injection (SI) signal
- Loss of offsite power
- Trip of both main feedwater pumps
- Manual actuation

One turbine-driven AFW pump is also available and is started on:

- Low-low steam generator water level in two-out-of-three level channels in any two steam generators
- Loss of offsite power
- Manual actuation

Following a loss of offsite power, the emergency diesel generators supply electrical power to the two motor-driven AFW pumps. The turbine-driven AFW pump is powered via steam flow from the secondary system that exhausts to the atmosphere. All of the AFW pumps take suction from the non-safety grade condensate storage tank (CST) for delivery to the steam generators. For the purposes of safety analysis, it is assumed that suction is taken from the essential service water system.

An analysis is performed to demonstrate that following a loss of normal feedwater, the AFWS is capable of removing the stored energy, residual decay heat, and RCP heat. This is demonstrated by showing that the pressurizer does not become water-solid, which could lead to overpressurization of the RCS and a subsequent loss of water from the RCS via a pressurizer pressure relief or safety valve.

6.3.5.2 Method of Analysis

The LONF transient is analyzed using the RETRAN computer code (Reference 1). The RETRAN computer code is described in detail in Section 6.3.0.6 of this report.

The LONF analysis is performed to demonstrate the adequacy of the RPS to trip the reactor and of the AFWS to remove long-term decay heat, stored energy, and RCP heat. As such, the assumptions used in the analysis are designed to maximize the time to reactor trip and to minimize the energy removal capability of the AFWS. These assumptions maximize the possibility of water relief from the RCS by maximizing the expansion of the RCS inventory, as noted in the assumptions listed below.

Primary- and secondary-side overpressurization concerns for the LONF event (with and without offsite power) are bounded by the Loss of Load/Turbine Trip (LOL/TT) event discussed in Section 6.3.4 of this report. For the LONF event, turbine trip occurs after reactor trip, whereas for the LOL/TT the turbine trip is the initiating fault. Therefore, the primary-to-secondary power mismatch and resultant RCS and main steam system heatup and pressurization transients are always more severe for the LOL/TT event. The loss of load event also bounds the LONF event with offsite power with respect to departure from nucleate boiling ratio (DNBR) concerns. Both the LONF and LOL/TT events result in a reduction in the heat removal capability of the secondary system. For the LONF event, the RCS temperature increases gradually as the steam generators boil down to the low-low water level trip setpoint, at which time reactor trip occurs, followed by turbine trip. For the LOL/TT event, the turbine trip is the initiating event, and the loss of heat sink is much more severe. Therefore, the initial RCS heatup will be much more severe for the LOL/TT event. This results in a more limiting DNBR transient for the LOL/TT event. The DNBR results obtained for the Complete Loss of Flow event, discussed in Section 6.3.7, bound the DNBR transient for the LONF accident without offsite power. This is because the RCP coastdown for the Complete Loss of Flow event is the initiating fault and the reactor trip occurs when the core flow is already degraded, whereas for the LONF event, the flow coastdown is initiated following reactor trip. As such, no specific evaluation of DNBR or overpressurization concerns is performed in the analysis of the LONF event.

The major assumptions are summarized below:

The plant is initially operating at 102 percent of nominal NSSS power.

Reactor trip occurs on low-low steam generator water level at 0 percent of narrow range span (NRS). Turbine trip occurs coincident with reactor trip.

A conservative core residual heat generation is assumed, based on the American Nuclear Society (ANS) 5.1-1979 decay heat model plus 2 sigma (Reference 7).

In order to ensure delivery of auxiliary feedwater to the steam generators, given the unique location (inside containment) of the feedline check valves at Callaway, the feedwater isolation valves (FWIVs) must close on a low-low steam generator water level signal. The FWIV stroke time, although modeled as 15 seconds, is not critical, provided the valves are fully closed within the assumed auxiliary feedwater startup delay (60 seconds).

One minute after the low-low steam generator water level setpoint is reached, AFW flow (960 gpm) from both motor-driven AFW pumps is initiated with flow split equally among the four steam generators. The turbine-driven AFW pump (single failure) is assumed to be unavailable.

No credit is taken for the operation of the steam dump system or steam generator power-operated relief valves (PORVs). The steam generator pressure increases to the setpoint of the main steam safety valves (MSSVs) with the lowest setpoint, and the resultant steam release through the MSSVs limits the secondary-side steam pressure to the setpoint values. The MSSVs are explicitly modeled in the LONF licensing basis analysis assuming a +3.0-percent tolerance with a 5-psi accumulation to full open. The MSSV model also assumes a 10-psi pressure drop from the steam generator exit to the MSSV inlet in determining the opening setpoints and an additional 15-psi pressure drop at full-open and full-flow conditions.

Normal reactor control systems are assumed to not function. However, the pressurizer PORVs, pressurizer heaters, and pressurizer sprays are assumed to operate as designed. This assumption results in a conservative transient with respect to the peak pressurizer water level. If these control systems did not operate, the pressurizer safety valves would maintain peak RCS pressure around the actuation setpoint throughout the transient.

Loss of normal feedwater cases are analyzed with and without offsite power available, the difference being whether the RCPs coast down or not following reactor trip. A sensitivity study is also performed to determine the worst set of initial condition assumptions (such as, initial RCS average temperature or initial pressurizer water level).

The initial reactor coolant vessel average temperature modeled in the limiting cases (with and without offsite power) is assumed to be 3°F lower than the low nominal full-power value of 570.7°F. This results in a larger RCS water mass available for expansion during the transient, thus yielding a higher pressurizer water level.

The initial pressurizer water level modeled in the limiting cases is 5 percent of span above the nominal value of 43 percent of span, which corresponds to the low nominal full-power vessel average temperature of 570.7°F. This was shown to be slightly more limiting than a higher initial pressurizer water level of 65 percent of span (including uncertainty) and its corresponding initial reactor coolant vessel average temperature of 588.4°F.

The initial pressurizer pressure is assumed to be 30 psi below the nominal value of 2,250 psia. Given the large amount of margin calculated for all cases analyzed and the small sensitivity of the results to minor changes in the initial pressurizer pressure, additional sensitivities assuming a higher initial pressurizer pressure were deemed unnecessary. An initial feedwater temperature of 446°F, consistent with the high nominal value, was assumed. A sensitivity to a lower initial feedwater temperature of 390°F was not performed for the same reasons identified for the initial pressurizer pressure.

The initial steam generator water level is assumed to be 6.2 percent of NRS above the nominal value of 51.3 percent of NRS. A high initial steam generator water level is conservative because it maximizes the time to reach the low-low steam generator water level.

A separate analysis is performed to address the reliability of the AFWS. The analysis is performed in a manner similar to that described above for the Final Safety Analysis Report (FSAR) Chapter 15 analysis, but assuming that only a single motor-driven AFW pump is available to feed 2 of the 4 steam generators. The cases considered in this additional analysis assume better-estimate conditions for several key input parameters. Specifically, initial conditions (nuclear steam supply system (NSSS) power, RCS pressure and temperature, pressurizer level) and reactor trip and equipment setpoints are assumed to be at their nominal values. Most importantly, a better-estimate decay heat model, consistent with ANS 1971 full decay heat with no uncertainties, is used. This is the first implementation of the dual-analysis approach to separately address Chapter 15 and AFWS reliability concerns for the LONF event for Callaway. Previously, a single bounding analysis had been performed combining the conservative Chapter 15-type assumptions and the reduced AFW flow consistent with a single motor-driven AFW pump. This resulted in an analysis that was overly conservative. Utilizing the dual-analysis approach, with both analyses assuming the failure of the turbine-driven AFW pump as the limiting single failure, allows the plant to address both concerns separately while continuing to show that the conservative acceptance criterion used by Westinghouse for this event (preventing pressurizer filling) is met for both scenarios. By demonstrating that acceptable results are achieved in this separate analysis crediting a single motor-driven AFW pump, the Chapter 15 analysis can be performed assuming the operation of both available motor-driven AFW pumps. The dual-analysis approach has been previously used by Westinghouse in at least one other LONF analysis of a Westinghouse-designed plant.

6.3.5.3 Results

Figures 6.3.5-1 through 6.3.5-6 show the significant plant responses following a LONF with offsite power available. Similarly, Figures 6.3.5-7 through 6.3.5-12 show the plant responses for the case where offsite power is lost following reactor trip. The calculated sequences of events for both cases are listed in Tables 6.3.5-1 (with offsite power) and 6.3.5-2 (without offsite power).

Following the reactor and turbine trip from full load, the water level in each steam generator falls due to the reduction of the steam generator void fraction, and because steam flow through the MSSVs continues to dissipate the stored and generated heat. One minute after reaching the low-low steam generator water level trip setpoint, flow from the available motor-driven AFW pumps is credited, thus reducing the rate of water level decrease in the steam generators.

The capacity of both motor-driven AFW pumps is sufficient to dissipate core residual heat, stored energy, and RCP heat without water relief through the pressurizer PORVs or safety valves. Figures 6.3.5-4 and 6.3.5-10 show that at no time is there water relief from the pressurizer, as the peak pressurizer mixture volume is well below the pressurizer volume limit of 1,800 ft^3. Plant procedures may be followed to further cool down the plant.

The results of the LONF analysis (with and without offsite power), performed to address AFWS reliability concerns, are similar to those reported for the Chapter 15 cases described above and demonstrate that pressurizer filling does not occur. The limiting pressurizer filling case (with a loss of offsite power) reaches a maximum pressurizer mixture volume of 1,703 ft^3, which is below the pressurizer volume limit of 1,800 ft^3. Figures and sequences of events for these cases are not included in this report.

6.3.5.4 Conclusions

The results of the loss of normal feedwater analysis show that all applicable acceptance criteria are satisfied. The AFW capacity is sufficient to dissipate core residual heat, stored energy, and RCP heat such that reactor coolant water is not relieved through the pressurizer relief or safety valves.

Table 6.3.5-1 Time Sequence of Events for Loss of Normal Feedwater with Offsite Power Available

Event	Time (Seconds)
Main Feedwater Flow Stops	0.0
Low-Low Steam Generator Water Level Trip Setpoint (0% NRS) Reached	40.3
Rods Begin to Drop	42.3
Both Motor-Driven AFW Pumps Start	100.3
Peak Mixture Volume in the Pressurizer Occurs	4078.5
Results	
Peak Pressurizer Mixture Volume	1,231 ft^3
Pressurizer Mixture Volume Limit	1,800 ft^3

Table 6.3.5-2 Time Sequence of Events for Loss of Normal Feedwater Without Offsite Power Available

Event	Time (Seconds)
Main Feedwater Flow Stops	0.0
Low-Low Steam Generator Water Level Trip Setpoint (0% NRS) Reached	40.3
Rods Begin to Drop	42.3
Reactor Coolant Pumps Begin to Coast Down	44.3
Both Motor-Driven AFW Pumps Start	100.3
Peak Mixture Volume in the Pressurizer Occurs	2207.0
Results	
Peak Pressurizer Mixture Volume	1,425 ft^3
Pressurizer Mixture Volume Limit	1,800 ft^3

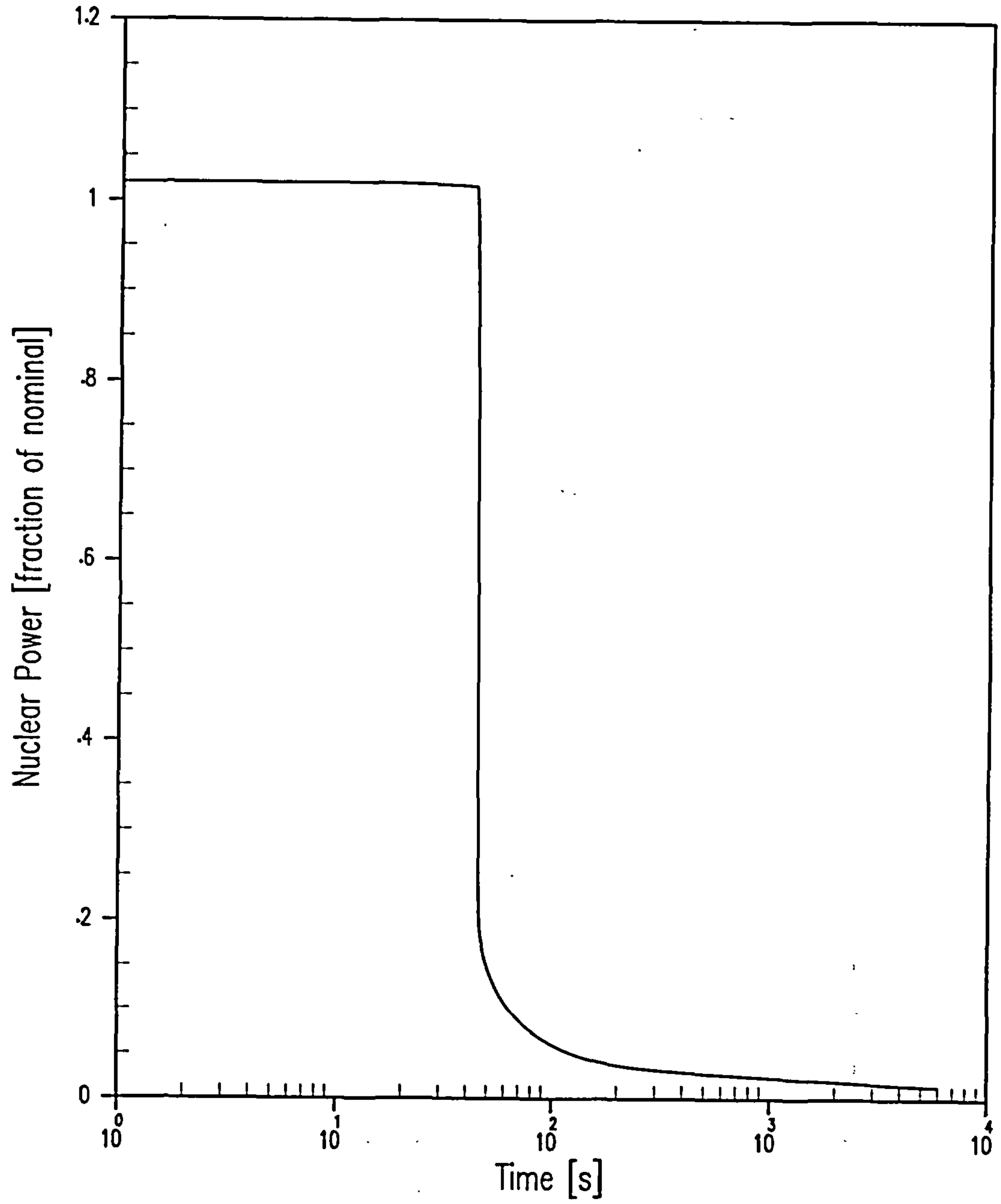


Figure 6.3.5-1 **Loss of Normal Feedwater with Offsite Power Available – Nuclear Power versus Time**

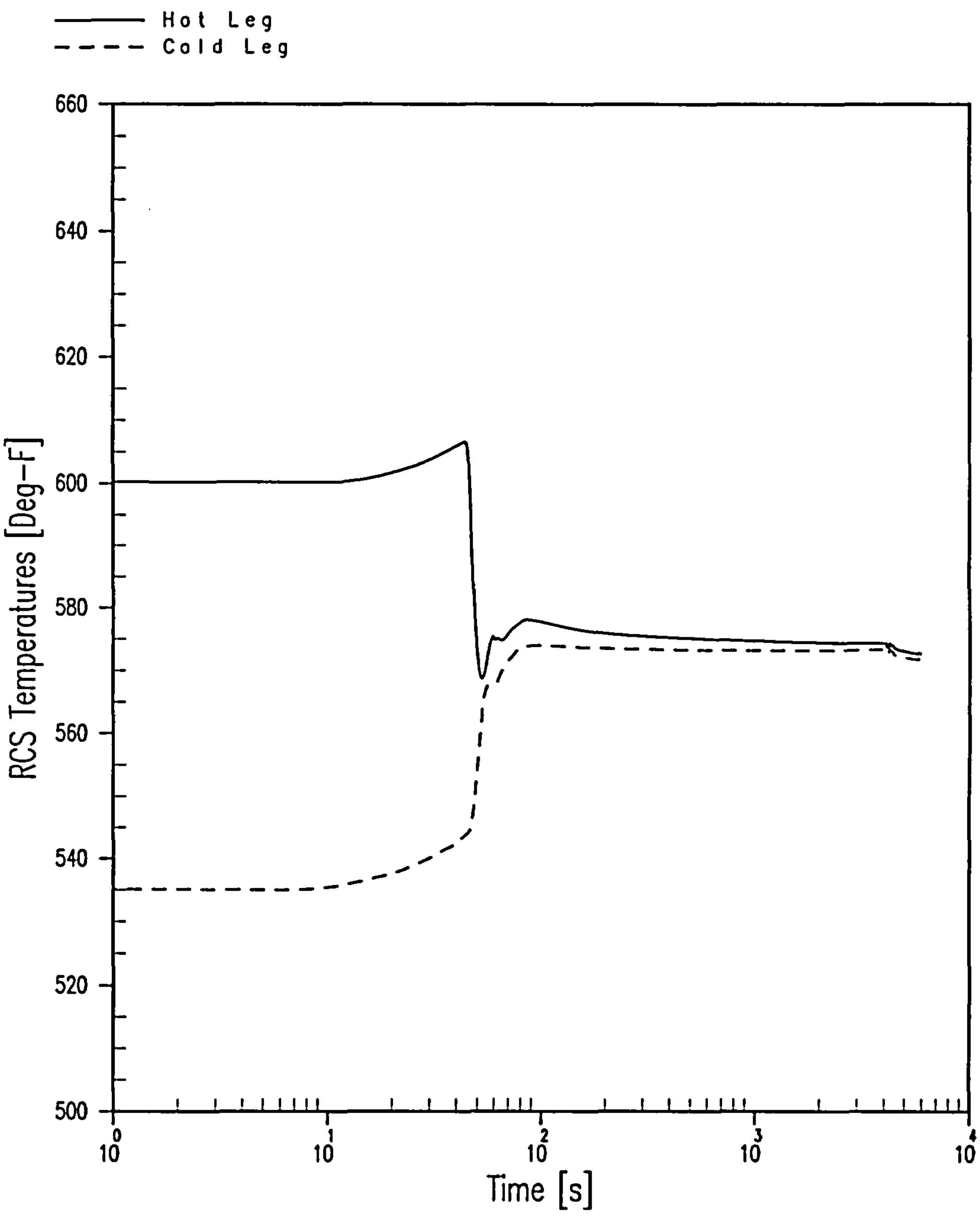


Figure 6.3.5-2 **Loss of Normal Feedwater with Offsite Power Available – RCS Temperatures versus Time**

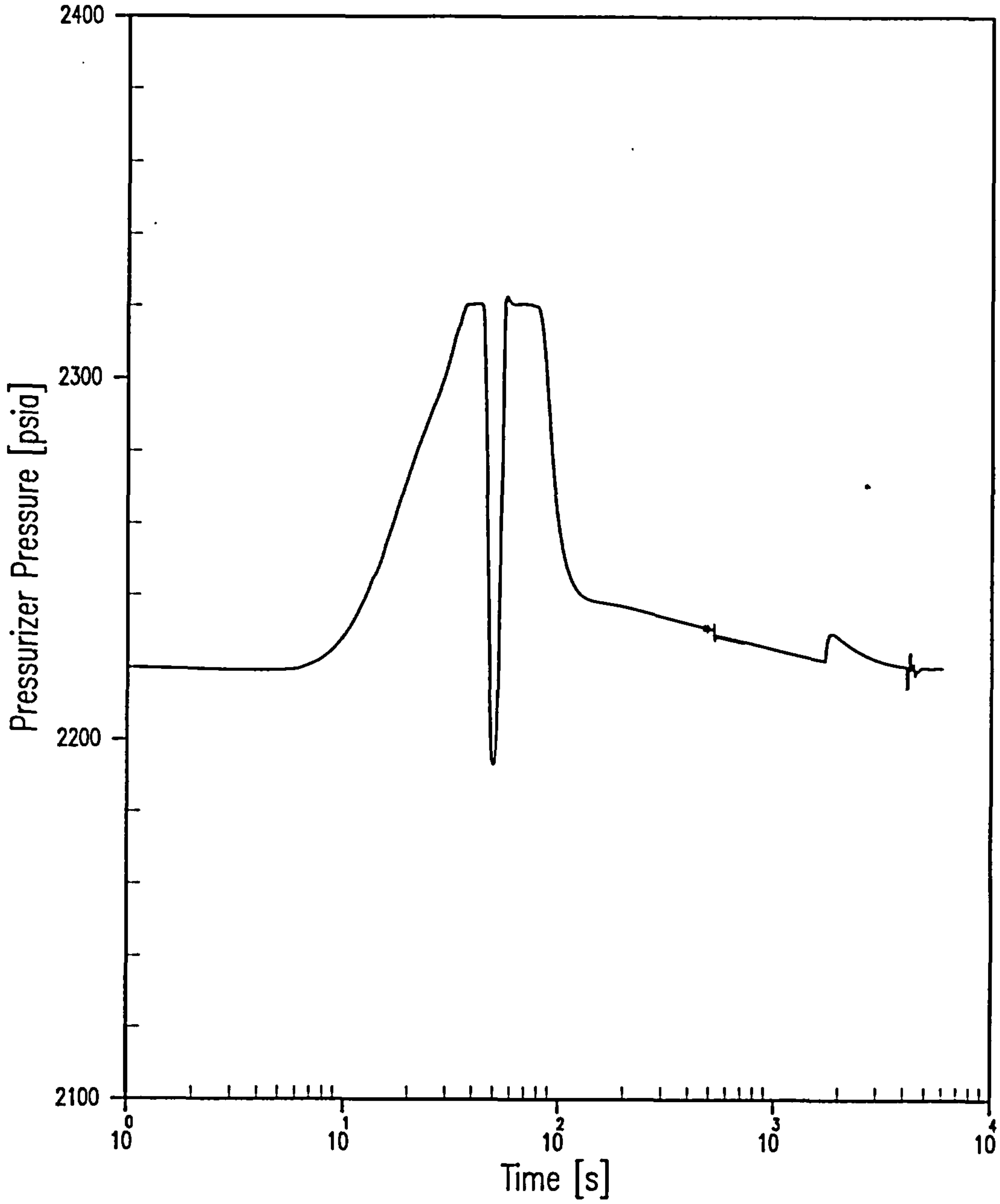


Figure 6.3.5-3 Loss of Normal Feedwater with Offsite Power Available – Pressurizer Pressure versus Time

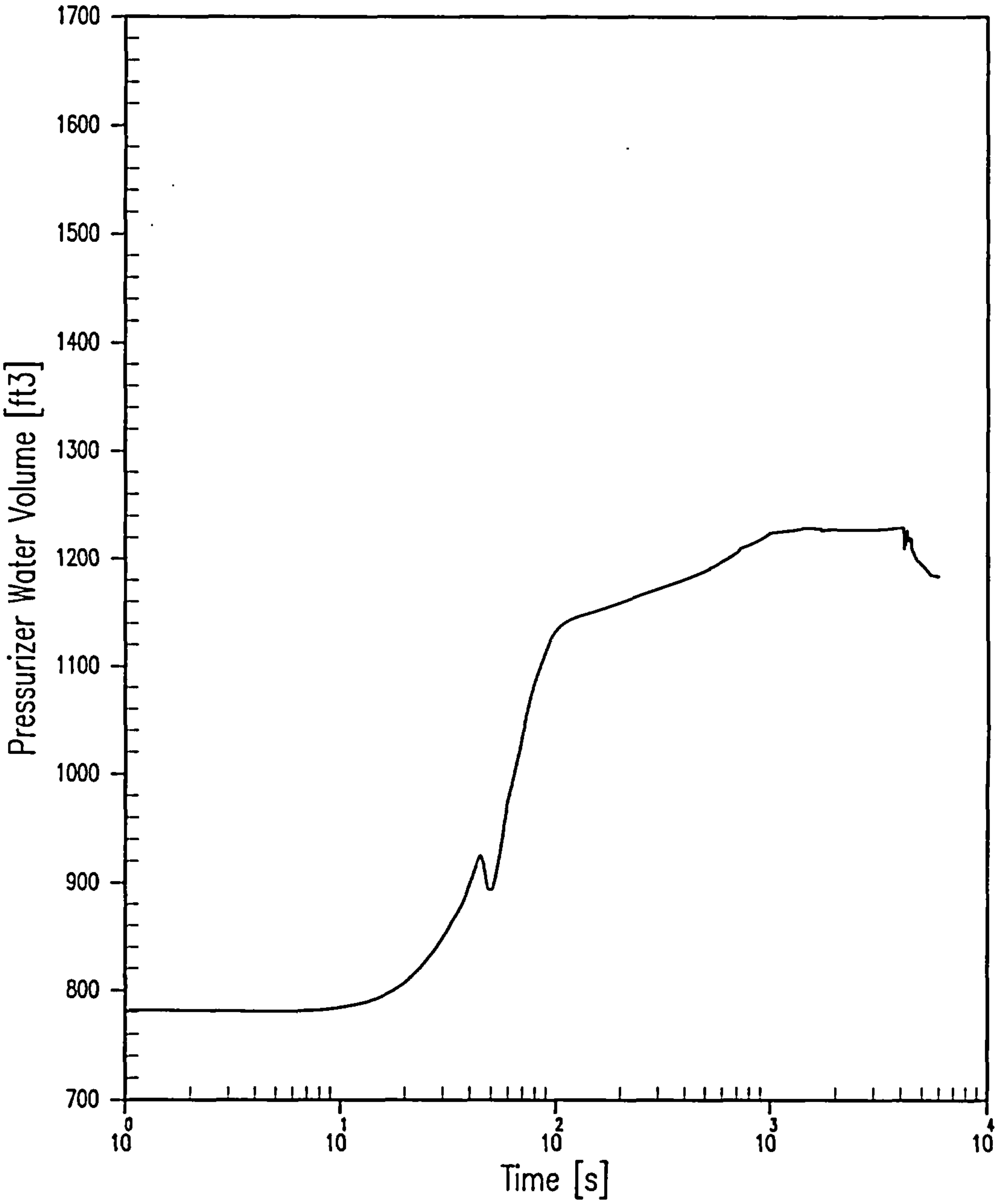


Figure 6.3.5-4 **Loss of Normal Feedwater with Offsite Power Available – Pressurizer Water Volume versus Time**

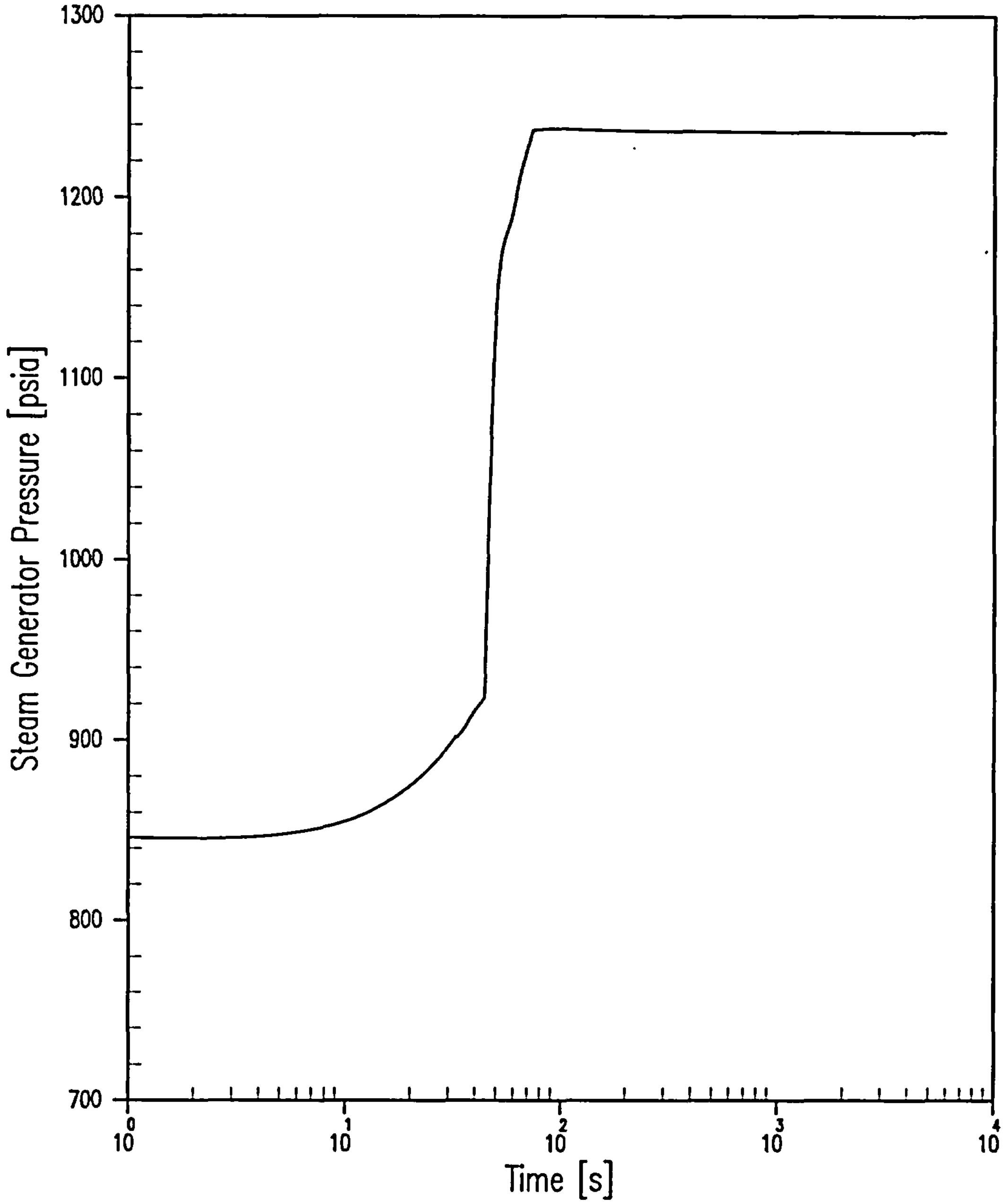


Figure 6.3.5-5 **Loss of Normal Feedwater with Offsite Power Available – Steam Generator Pressure versus Time**

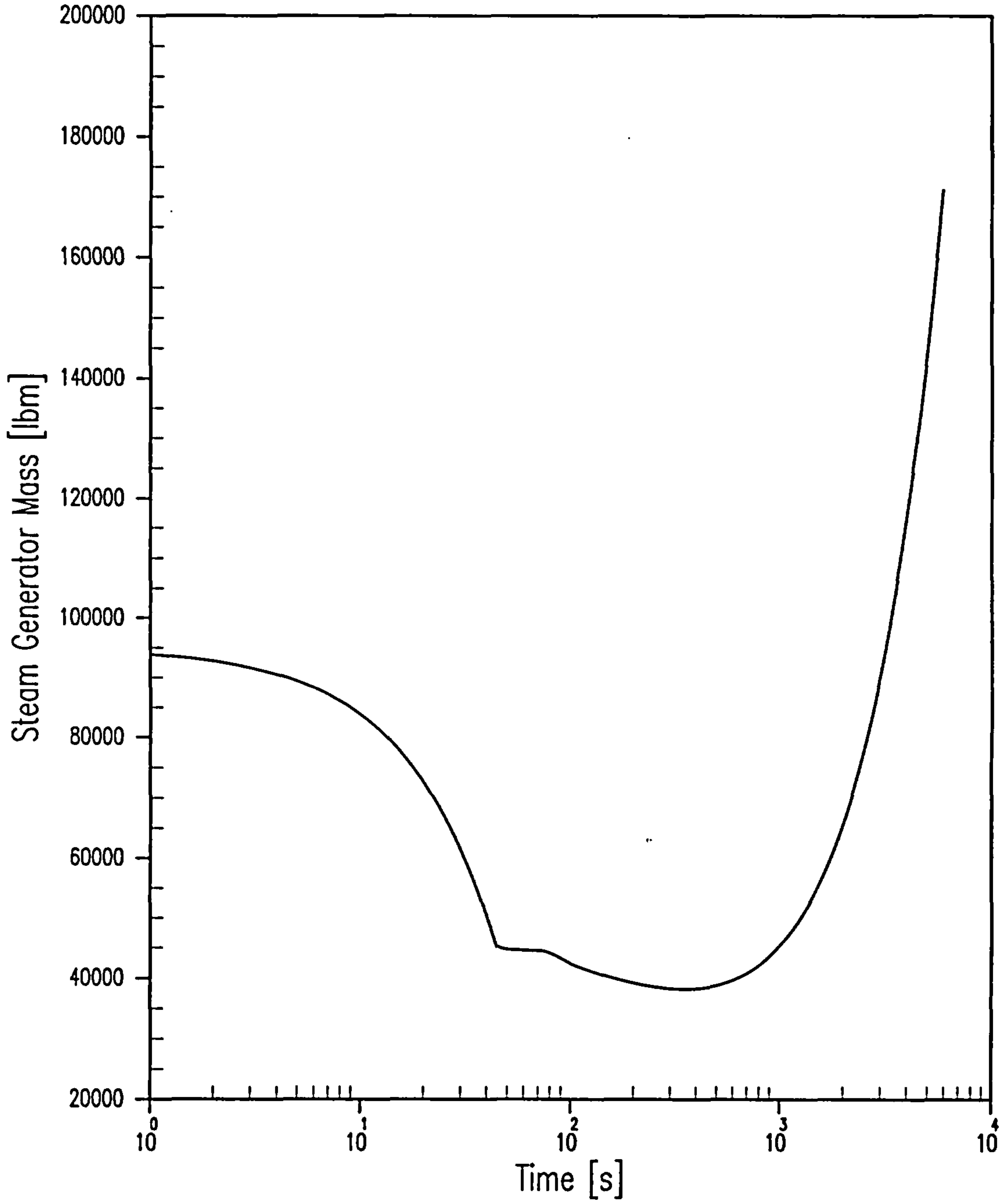


Figure 6.3.5-6 Loss of Normal Feedwater with Offsite Power Available – Steam Generator Mass versus Time

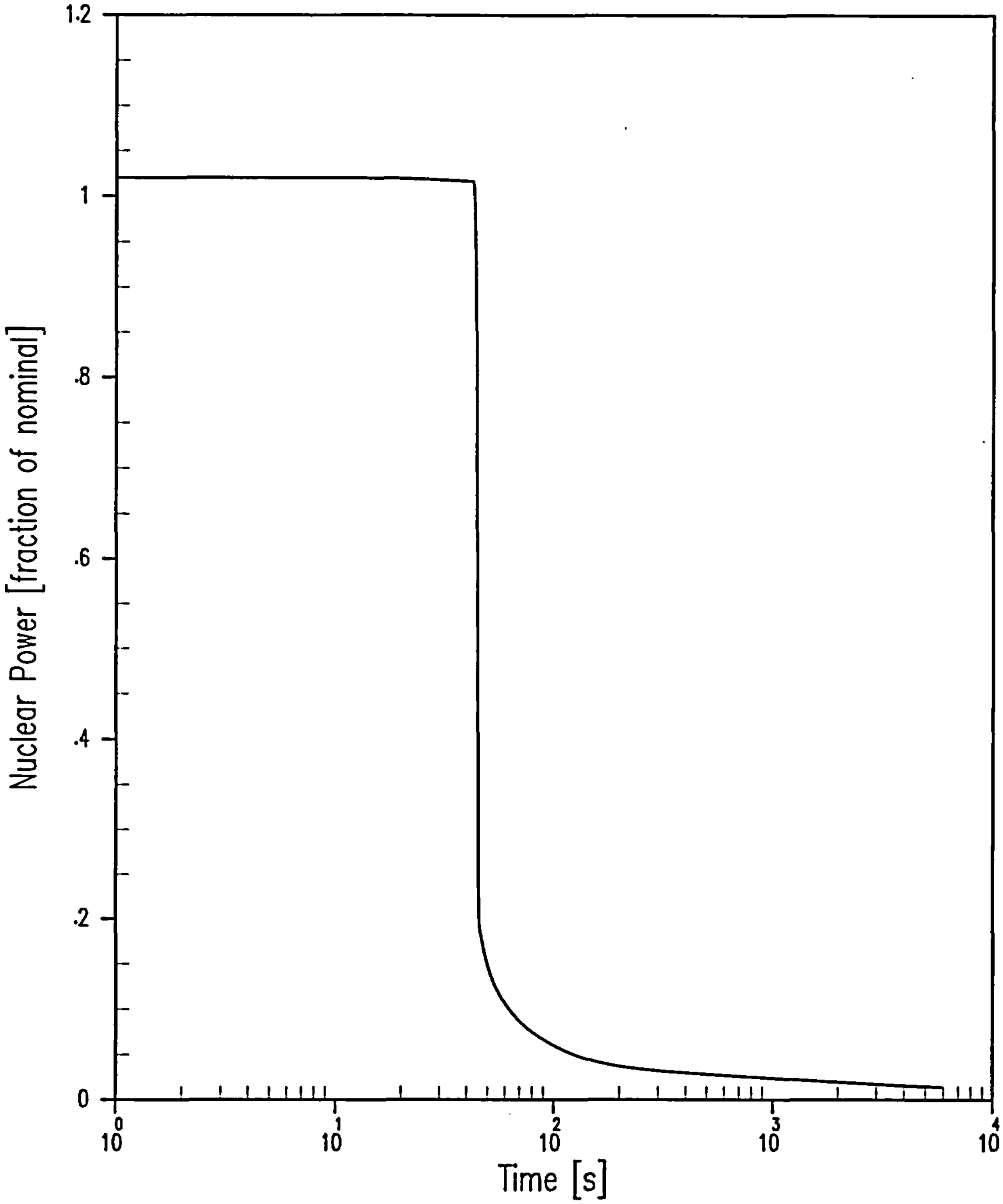


Figure 6.3.5-7 **Loss of Normal Feedwater Without Offsite Power Available – Nuclear Power versus Time**

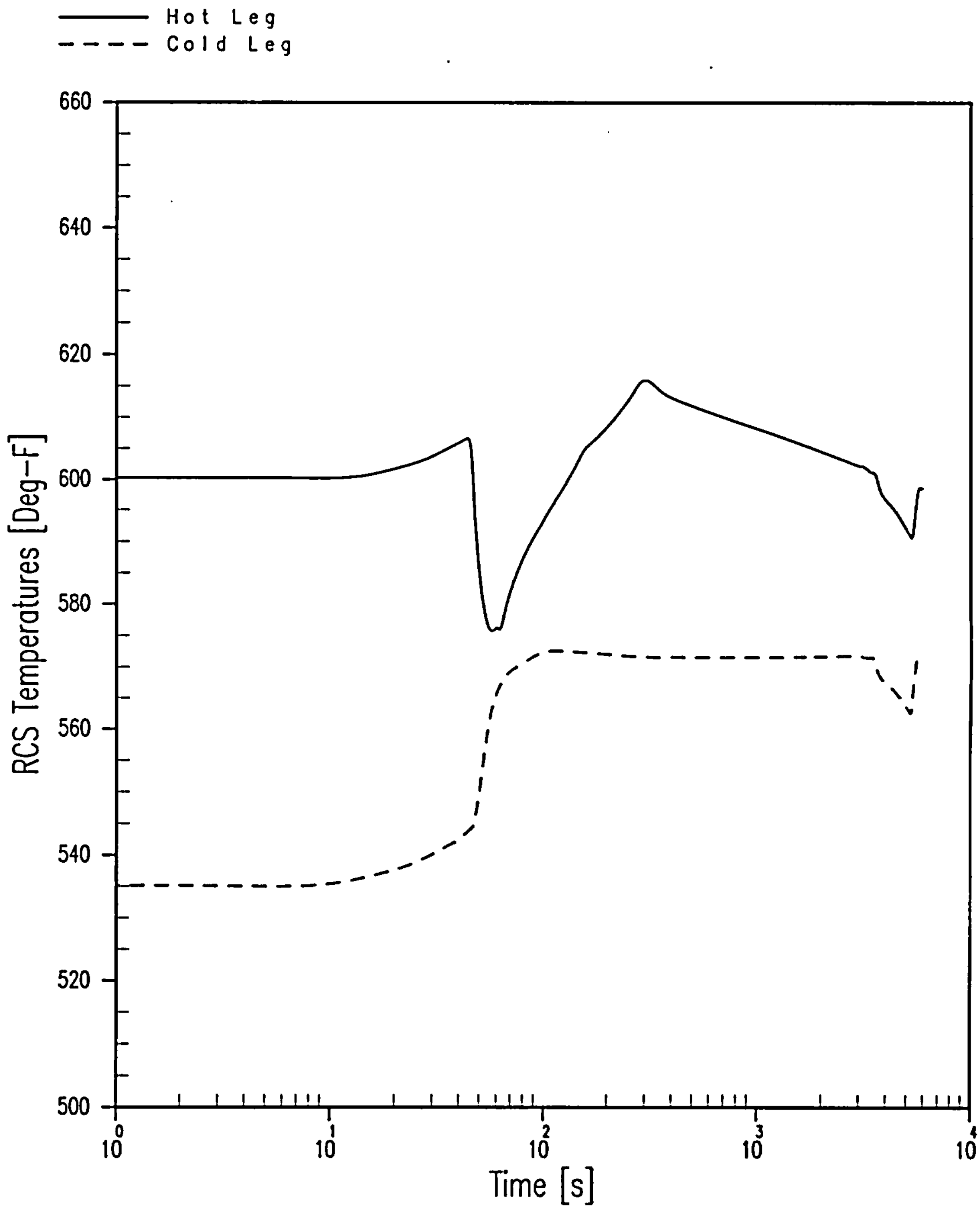


Figure 6.3.5-8 **Loss of Normal Feedwater Without Offsite Power Available – RCS Temperatures versus Time**

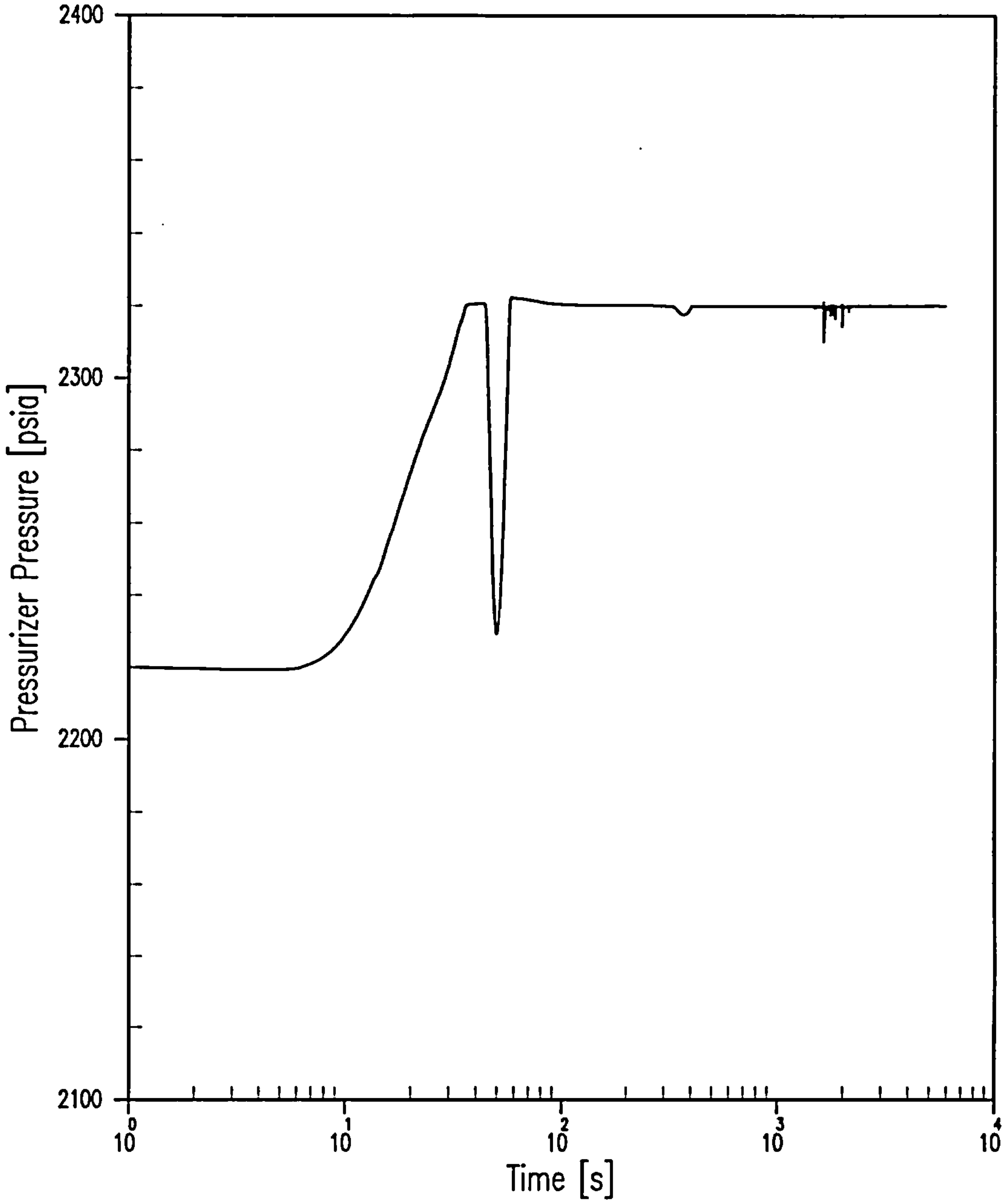


Figure 6.3.5-9 **Loss of Normal Feedwater Without Offsite Power Available – Pressurizer Pressure versus Time**

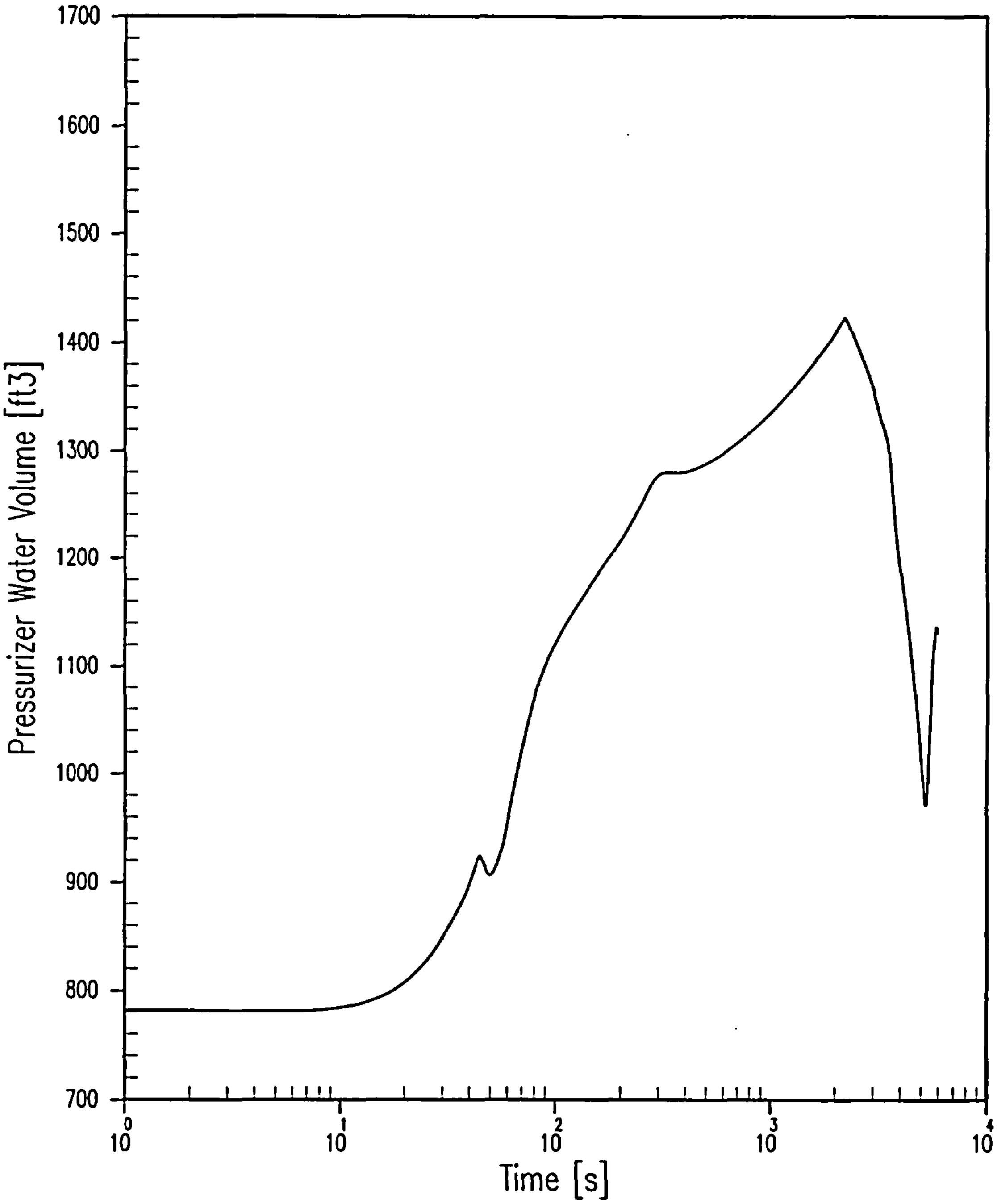


Figure 6.3.5-10 Loss of Normal Feedwater Without Offsite Power Available – Pressurizer Water Volume versus Time

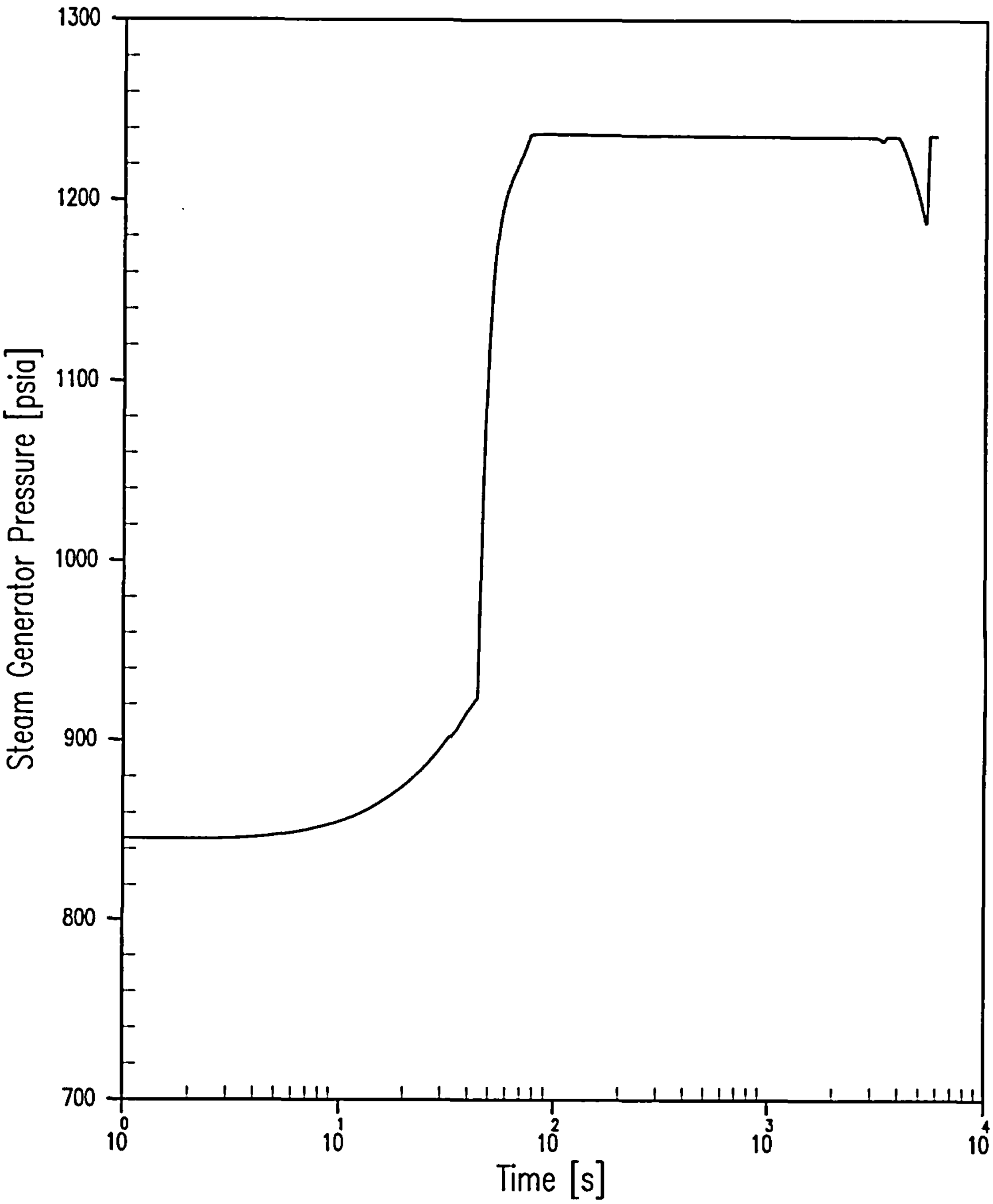


Figure 6.3.5-11 Loss of Normal Feedwater Without Offsite Power Available – Steam Generator Pressure versus Time

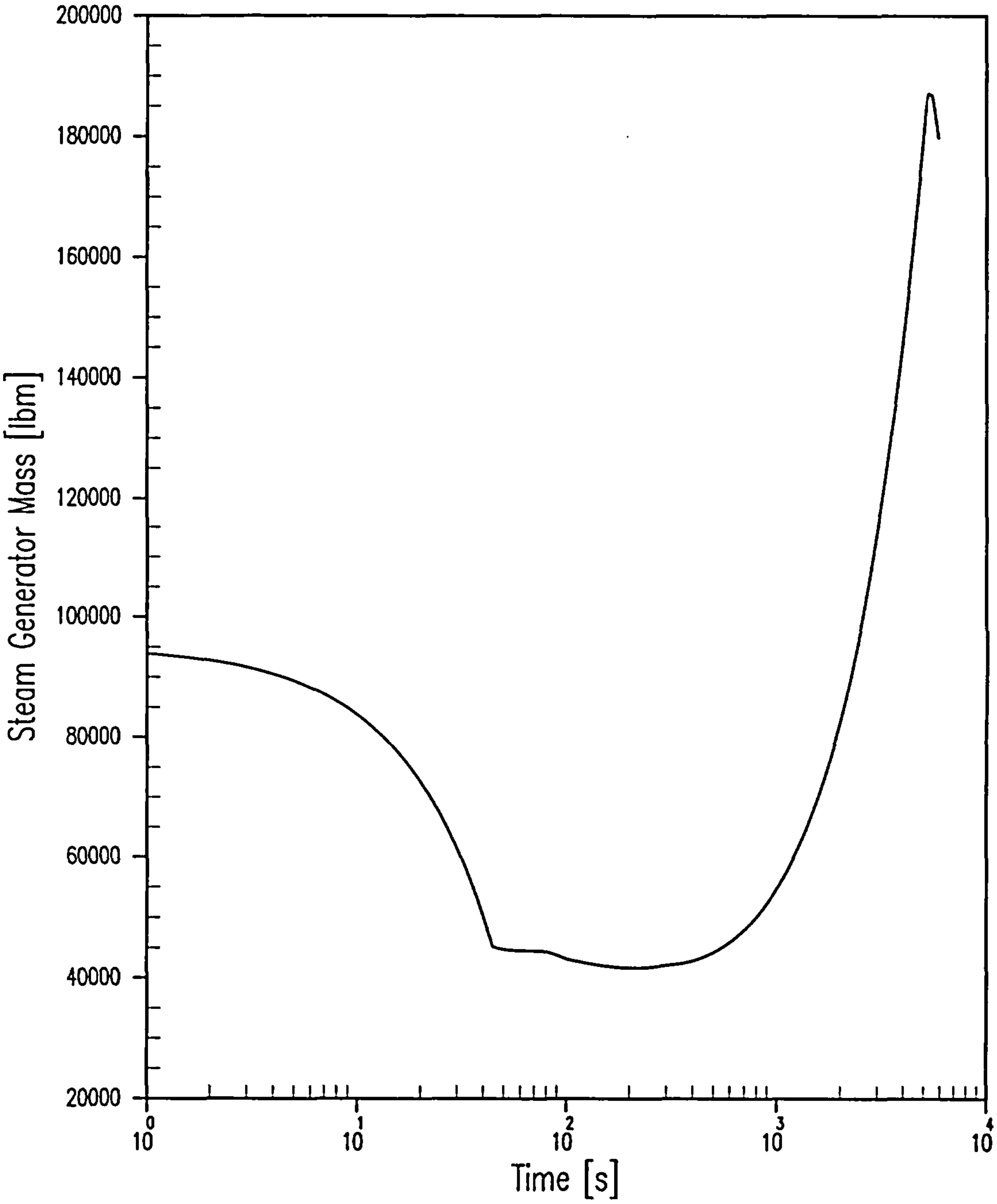


Figure 6.3.5-12 **Loss of Normal Feedwater Without Offsite Power Available – Steam Generator Mass versus Time**

6.3.6 Feedwater System Pipe Break (FSAR Section 15.2.8)

6.3.6.1 Accident Description

The Feedwater System Pipe Break (also referred to as the feedline break or FLB) incident is defined as a break in a feedwater pipe large enough to prevent the addition of sufficient feedwater to maintain shell-side fluid inventory in the steam generators. If the break is postulated in a feedline between the check valve and the steam generator, fluid from the steam generator will be discharged through the break. Furthermore, with feed-ring type steam generators, a break between the check valve and the steam generator could preclude the subsequent addition of auxiliary feedwater (AFW) to the affected steam generator because the AFW piping connects to the main feedline. In contrast, if a break occurs upstream of the feedline check valve, the transient progresses as a Loss of Normal Feedwater event, which is discussed in Section 6.3.5.

A break inside containment can produce adverse environmental conditions that can induce an error in the steam generator level indication, which is used for initiation of protective functions (reactor trip, AFW actuation, and feedwater line isolation). The environmental allowance modifier (EAM) automatically enables a higher steam generator low-low level trip setpoint, which accounts for this environmental error and guarantees that the low-low level trip signal is generated before the level assumed in this analysis (0-percent narrow range span (NRS)) is reached.

Depending upon the size of the break and the plant operating conditions at the time of the rupture, the break could cause either a cooldown (by excessive energy discharge through the break) or heatup of the reactor coolant system (RCS). Because the consequences of an RCS cooldown resulting from an FLB are bounded by the cooldown consequences of a Steam System Piping Failure (see Section 6.3.3), the FLB event is analyzed only with respect to RCS heatup effects.

Since the subcooled feedwater flow to the steam generators is reduced by an FLB, the long-term capacity of the secondary system to remove heat from the RCS is diminished. The feedwater flow reduction can cause the RCS temperatures to increase prior to reactor trip. Additionally, fluid inventory of the faulted steam generator may be discharged through the break, which will reduce the heat sink volume available for decay heat removal following reactor trip. The FLB event is analyzed to demonstrate the ability of Callaway safety systems to adequately remove long-term decay heat and prevent excessive heatup of the RCS.

The Reactor Protection System (RPS) provides the following protection against a feedwater line rupture event:

- High pressurizer pressure
- Overtemperature ΔT (OTΔT)
- Low-low steam generator water level in one or more steam generators
- High pressurizer water level
- High containment pressure
- Safety injection signal

The auxiliary feedwater system (AFWS) is started automatically as described below. Two motor-driven AFW pumps are available for delivery of AFW. These are started on:

- Low-low steam generator water level in two-out-of-four channels in any steam generator
- Loss-of-offsite power
- Trip of all main feedwater (MFW) pumps
- Manual actuation

One turbine-driven AFW pump is also available and is started on:

- Low-low steam generator water level in two-out-of-four channels in any two steam generators
- Loss-of-offsite power
- Manual actuation

An analysis is performed to demonstrate that following a feedwater line rupture, the AFWS is capable of removing the stored energy, residual decay heat, and reactor coolant pump (RCP) heat. This prevents overpressurizing the RCS and uncovering the reactor core.

6.3.6.2 Method of Analysis

The FLB event, as analyzed specifically for Callaway to address the control and protection interaction (CPI), is a loss of normal feedwater to all four loops up to the time of reactor trip followed by a full double-ended rupture in one loop between the check valve and the steam generator upon reactor trip. The FLB transient is analyzed using the RETRAN computer code (Reference 1). The RETRAN computer code is described in detail in Section 6.3.0.6 of this report.

The FLB analysis is performed to demonstrate the adequacy of the RPS to trip the reactor and of the AFWS to remove long-term decay heat and prevent excessive heatup of the RCS. For the analysis of the FLB event, Westinghouse has established an internal criterion that no bulk boiling occurs in the primary coolant system prior to event turnaround. Turnaround occurs when the heat removal capability of the steam generators being fed AFW exceeds nuclear steam supply system (NSSS) heat generation. This conservatively ensures that the core remains covered with water and thereby will remain in place and geometrically intact with no loss of core cooling capability. This single criterion is conservative and is chosen for convenience in interpreting the transient results. As such, the assumptions used in the analysis are designed to minimize the energy removal capability of the RCS and main steam system (MSS), and minimize the margin to saturated conditions in the RCS.

Primary- and secondary-side overpressurization concerns for the FLB event (with and without offsite power) are bounded by the Loss of Load/Turbine Trip (LOL/TT) event discussed in Section 6.3.4 of this report. For the FLB event, turbine trip occurs after reactor trip. For the LOL/TT, the turbine trip is the initiating fault. Therefore, the primary-to-secondary power mismatch and resultant RCS and MSS heatup and pressurization transients are always more severe for the LOL/TT event. With respect to fuel damage due to departure from nucleate boiling, the FLB event would be bounded by either the Steamline Break - Core Response event or the LOL/TT event.

The major assumptions for the FLB analysis are summarized below:

- The plant is initially operating at 102 percent of nominal NSSS power.
- Initial reactor coolant average temperature is 4.3°F above the high nominal full-power vessel average temperature of 588.4°F.
- The initial pressurizer pressure is 30 psi below the nominal value of 2,250 psia.
- The initial pressurizer level is set to the nominal full-power level plus 5 percent of span.
- The initial steam generator level in all steam generators is assumed to be at the nominal steam generator level plus 6.2 percent of NRS.
- Thermal design flow is assumed.
- Maximum tube plugging (5 percent) is assumed.
- A feedwater temperature of 446°F is assumed.

The MFW control system equipment is assumed to fail to conservatively account for CPI in an adverse environment. As such, MFW flow to all steam generators is assumed to be lost at the start of the transient.

Reactor trip is assumed to be initiated when the low-low steam generator water level trip setpoint (0-percent NRS) is reached. The analysis assumes the water level in all 4 steam generators decreases equally, at the same rate. Turbine trip occurs following reactor trip. Feedwater isolation valves will close 17 seconds following receipt of a low-low steam generator water level signal, thus ensuring proper delivery of auxiliary feedwater to the intact steam generator. This is a unique feature at Callaway to address the location of the feedline check valves inside containment.

The AFWS is actuated by the low-low steam generator water level signal and supplies AFW flow to the 3 intact steam generators. Failure of 1 protection train (worst single failure) is assumed; specifically, 1 of the 2 motor-driven AFW pumps has been assumed to fail. Therefore, 1 intact steam generator is delivered AFW flow from the turbine-driven and 1 motor-driven AFW pump, and the other 2 intact steam generators receive AFW from the turbine-driven AFW pump. AFW flow is modeled as a function of pressure. For example, at 1,200 psia, the total AFW flow modeled is 664.4 gpm.

A total AFW actuation delay of 301.4 seconds is assumed. This delay includes a 60-second delay following the low-low signal to allow time for startup of the emergency diesel generators and the AFW pumps, and an additional 241.4 seconds for the time it takes to fill the piping volume between the feedwater isolation valve and the feedline check valve in the intact feedwater lines.

A conservative core residual heat generation is assumed based on the American Nuclear Society (ANS) 5.1-1979 decay heat model plus 2 sigma (Reference 7). No credit is taken for heat energy deposited in RCS metal during the RCS heatup.

Normal reactor control systems are not assumed to function unless their operation results in more severe consequences. Therefore, the pressurizer power-operated relief valves (PORVs) are assumed to operate to minimize RCS pressure, which results in a lower saturation temperature. Pressurizer spray and heaters are assumed to be inoperable.

Although it is expected that the actuation of the safety injection system would occur during this event, the analysis conservatively does not model safety injection flow.

No credit is taken for the following RPS functions to mitigate the consequences of the FLB accident:

- High pressurizer pressure
- Low pressurizer pressure (reactor trip)
- Overtemperature ΔT
- High pressurizer water level
- High containment pressure

The FLB cases are analyzed with and without offsite power available. For the FLB without offsite power, RCPs coast down following reactor trip until the flow in the loops reaches natural circulation.

6.3.6.3 Results

Figures 6.3.6-1 through 6.3.6-11 show the significant plant responses following an FLB with offsite power available. Similarly, Figures 6.3.6-12 through 6.3.6-22 show the plant responses for the case where offsite power is lost following reactor trip. The calculated sequences of events for both cases are listed in Tables 6.3.6-1 (with offsite power) and 6.3.6-2 (without offsite power).

The system responses following the feedwater line break are similar for both cases analyzed. The results show that following reactor trip, the plant remains subcritical. The pressures in the RCS and MSS remain below 110 percent of the respective design pressures. Pressurizer pressure increases until reactor trip occurs on low-low steam generator water level. Pressure then decreases due to the relative loss of heat input. Coolant expansion then occurs due to reduced heat transfer capability in the steam generators. This in turn causes the pressurizer pressure to increase once again and remain at the PORV setpoint until the heatup portion of the transient is over. The pressurizer does not fill due to thermal expansion nor does the pressurizer empty. Therefore, the reactor remains covered with water throughout the transient. There is no bulk boiling in the RCS.

6.3.6.4 Conclusions

The results of the feedwater line rupture analysis show that all applicable acceptance criteria are satisfied. The AFWS is capable of removing the stored energy, residual decay heat, and RCP heat. This prevents overpressurizing the RCS and MSS and uncovering the reactor core.

Table 6.3.6-1 Time Sequence of Events for Feedwater Line Rupture with Offsite Power Available	
Event	**Time (Seconds)**
EAM Enables Harsh Environment Low-Low SG Water Level Trip Setpoint	0.0
Feedwater Control System Malfunction Occurs Due to Harsh Environment	0.0
Low-Low Steam Generator Water Level Reactor Trip Setpoint Reached in All Steam Generators	44.4
Low-Low SG Water Level Trip Signal is Generated	46.4
Rods Begin to Drop and Feedwater Line Rupture Occurs	46.4
Steam Generator Safety Valve Setpoint Reached (first occurrence)	47.8
MFW Isolation Valves Closed	61.4
Low Steam Line Pressure Setpoint Reached in Ruptured Steam Generator	118.4
Main Steam Line Isolation Valves Closed	135.4
AFW is Delivered to Intact Steam Generators	345.8
Steam Generator Safety Valve Setpoint Reached in Intact Steam Generators (second occurrence)	574.2
Core Decay Heat Plus Pump Heat Decreased to AFW Heat Removal Capacity	~1060.0
Results	
Minimum Margin to Hot Leg Saturation	67°F

Table 6.3.6-2 Time Sequence of Events for Feedwater Line Rupture Without Offsite Power Available	
Event	**Time (sec)**
EAM Enables Harsh Environment Low-Low SG Water Level Trip Setpoint	0.0
Feedwater Control System Malfunction Occurs Due to Harsh Environment	0.0
Low-Low Steam Generator Water Level Reactor Trip Setpoint Reached in All Steam Generators	44.4
Low-Low SG Water Level Trip Signal is Generated	46.4
Rods Begin to Drop and Feedwater Line Rupture Occurs	46.4
Steam Generator Safety Valve Setpoint Reached (first occurrence)	47.8
Power Lost to RCPs	48.4
MFW Isolation Valves Closed	61.4
Low Steam Line Pressure Setpoint Reached in Ruptured Steam Generator	98.9
Main Steam Line Isolation Valves Closed	113.9
AFW is Delivered to Intact Steam Generators	345.8
Steam Generator Safety Valve Setpoint Reached in Intact Steam Generators (second occurrence)	370.4
Core Decay Heat Plus Pump Heat Decreased to AFW Heat Removal Capacity	~790.0
Results	
Minimum Margin to Hot Leg Saturation	41.6°F

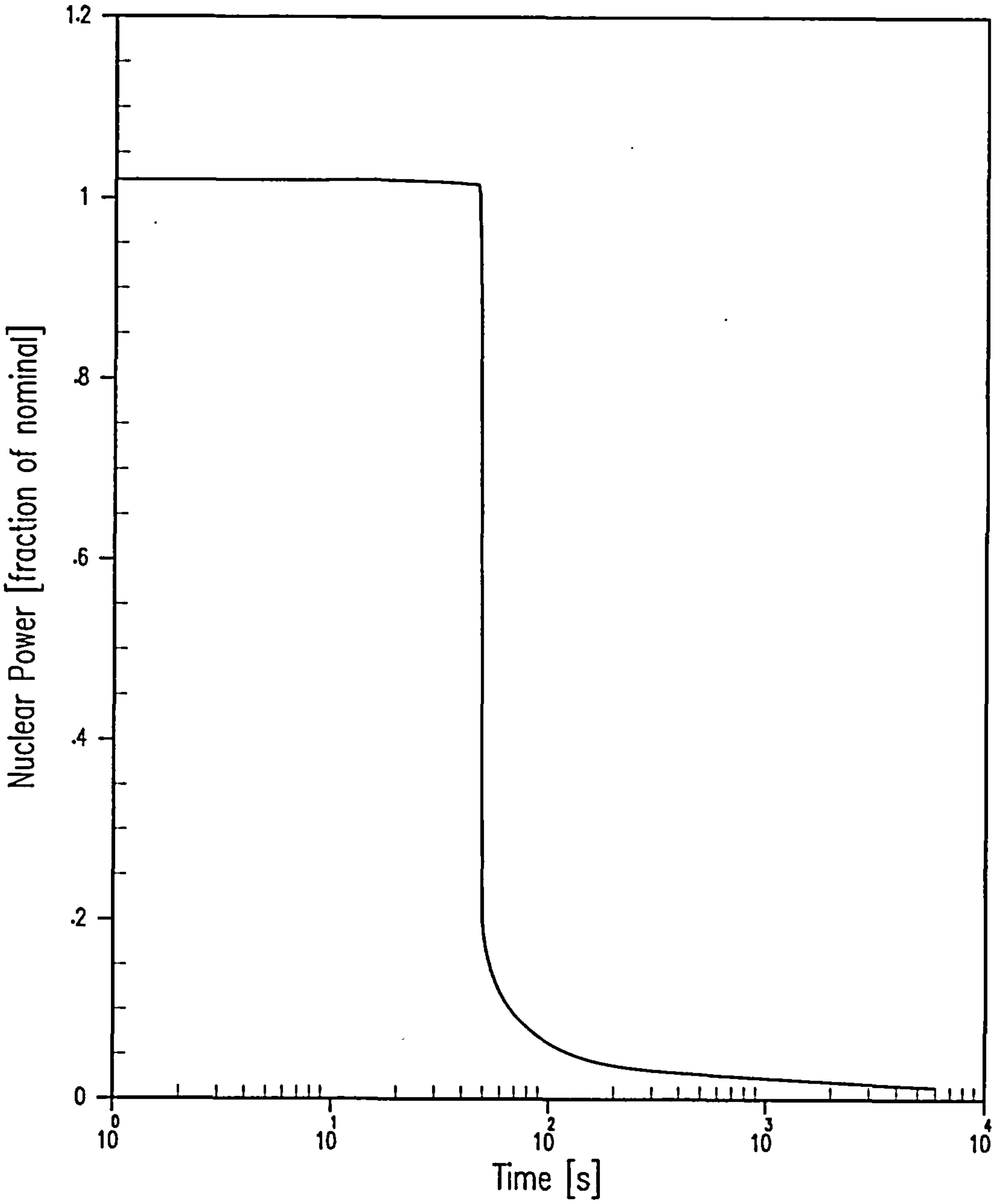


Figure 6.3.6-1 **Feedwater Line Rupture with Offsite Power Available – Nuclear Power Versus Time**

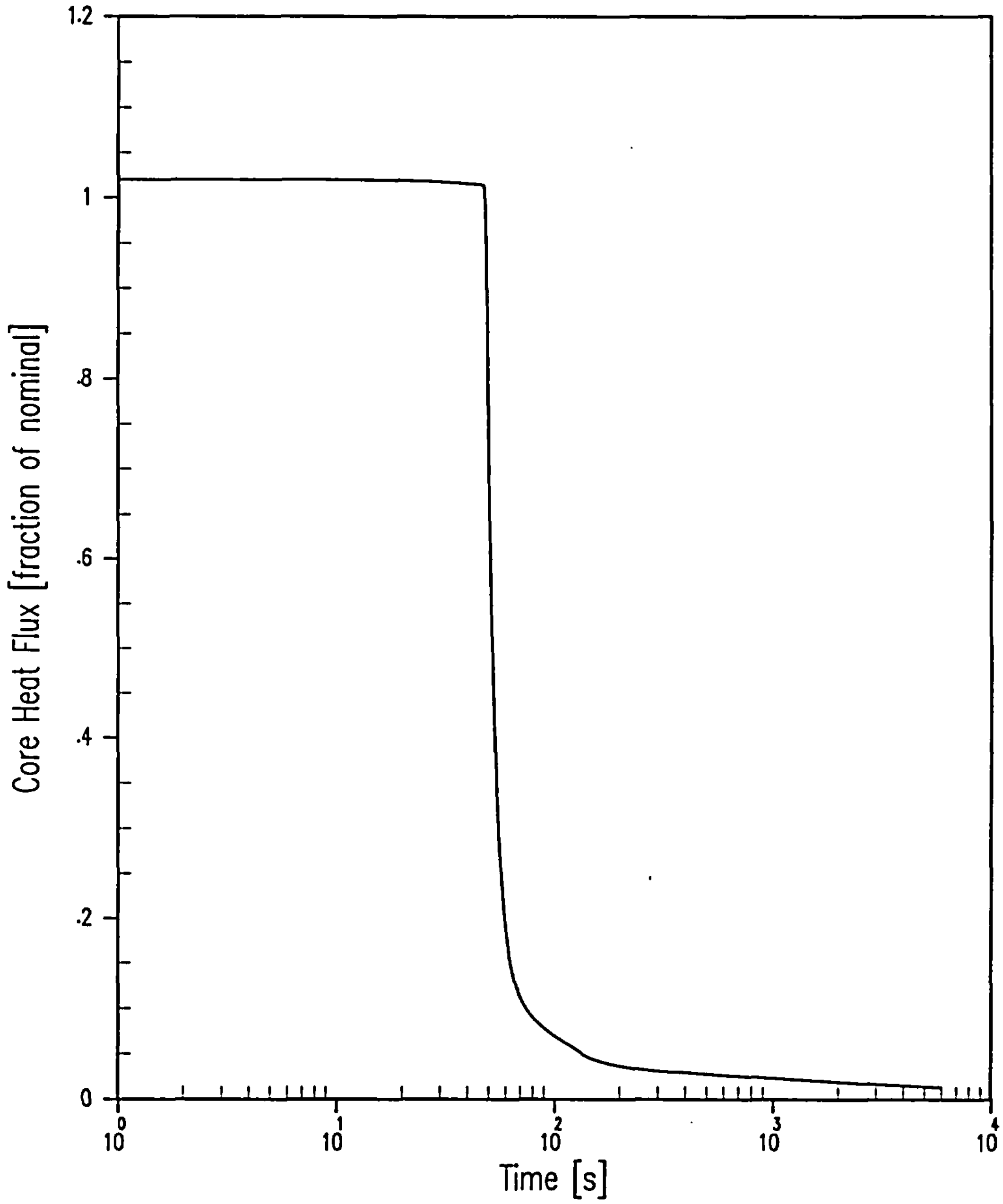


Figure 6.3.6-2 **Feedwater Line Rupture with Offsite Power Available – Core Heat Flux Versus Time**

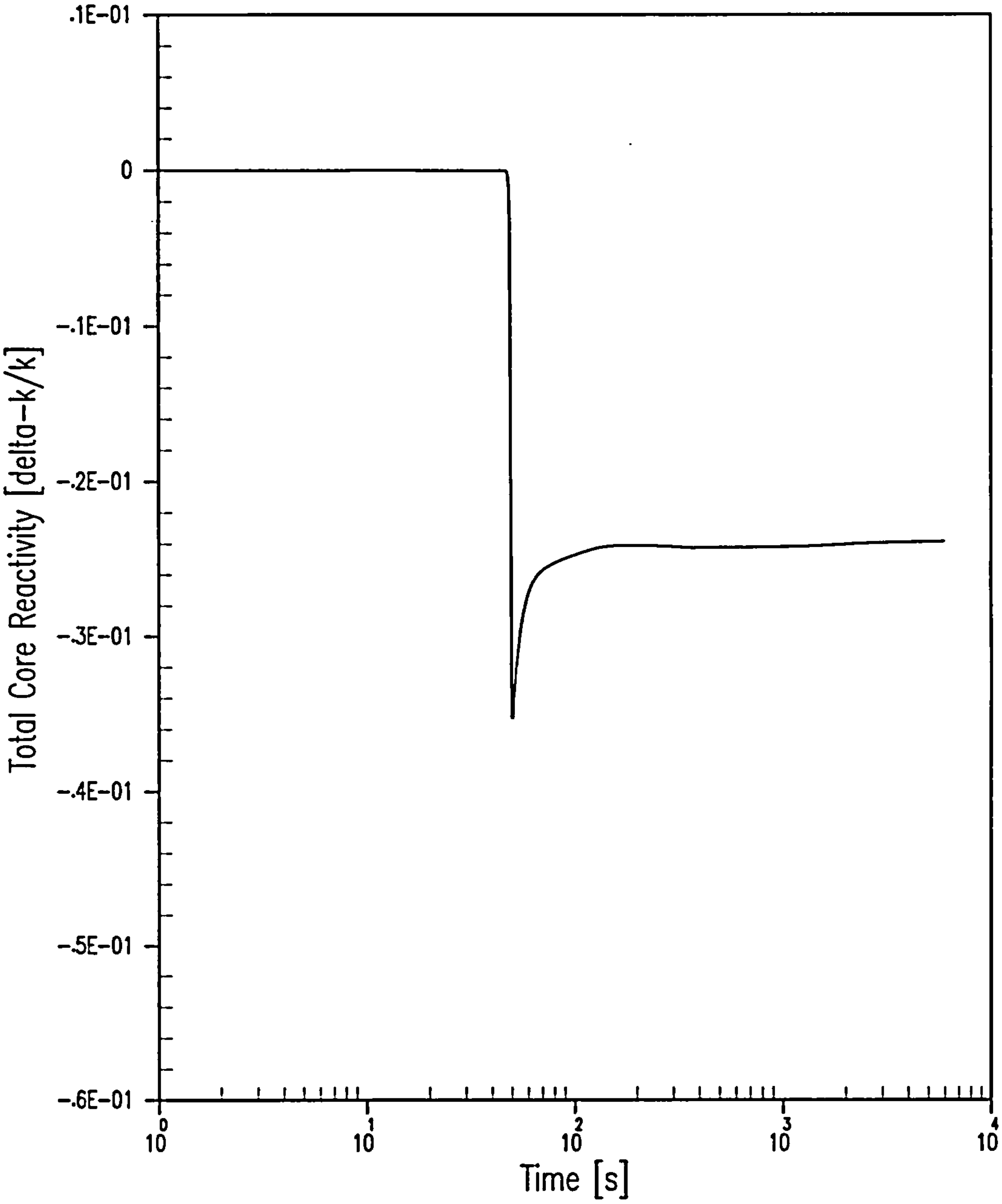


Figure 6.3.6-3 Feedwater Line Rupture with Offsite Power Available – Total Core Reactivity Versus Time

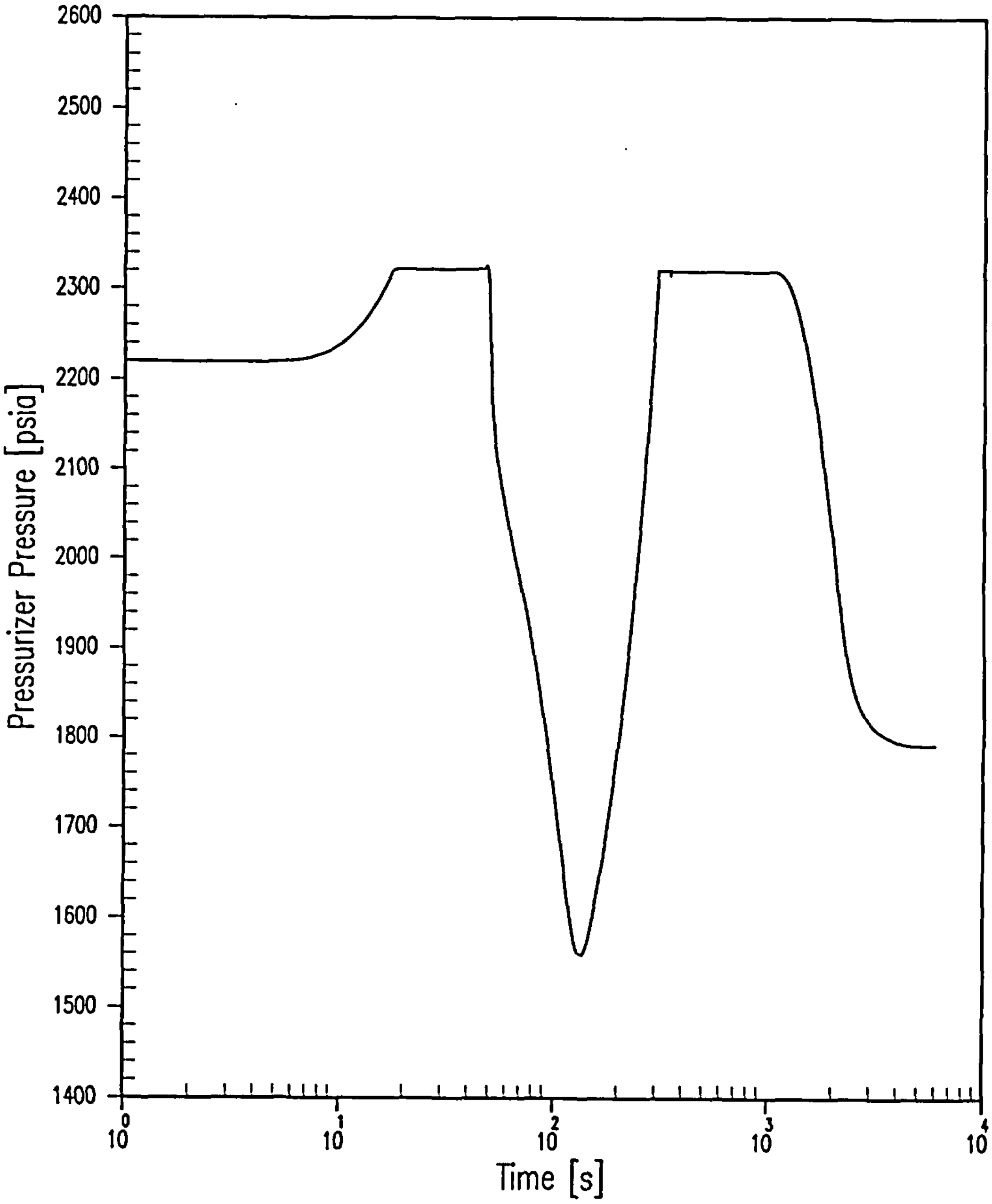


Figure 6.3.6-4 **Feedwater Line Rupture with Offsite Power Available – Pressurizer Pressure Versus Time**

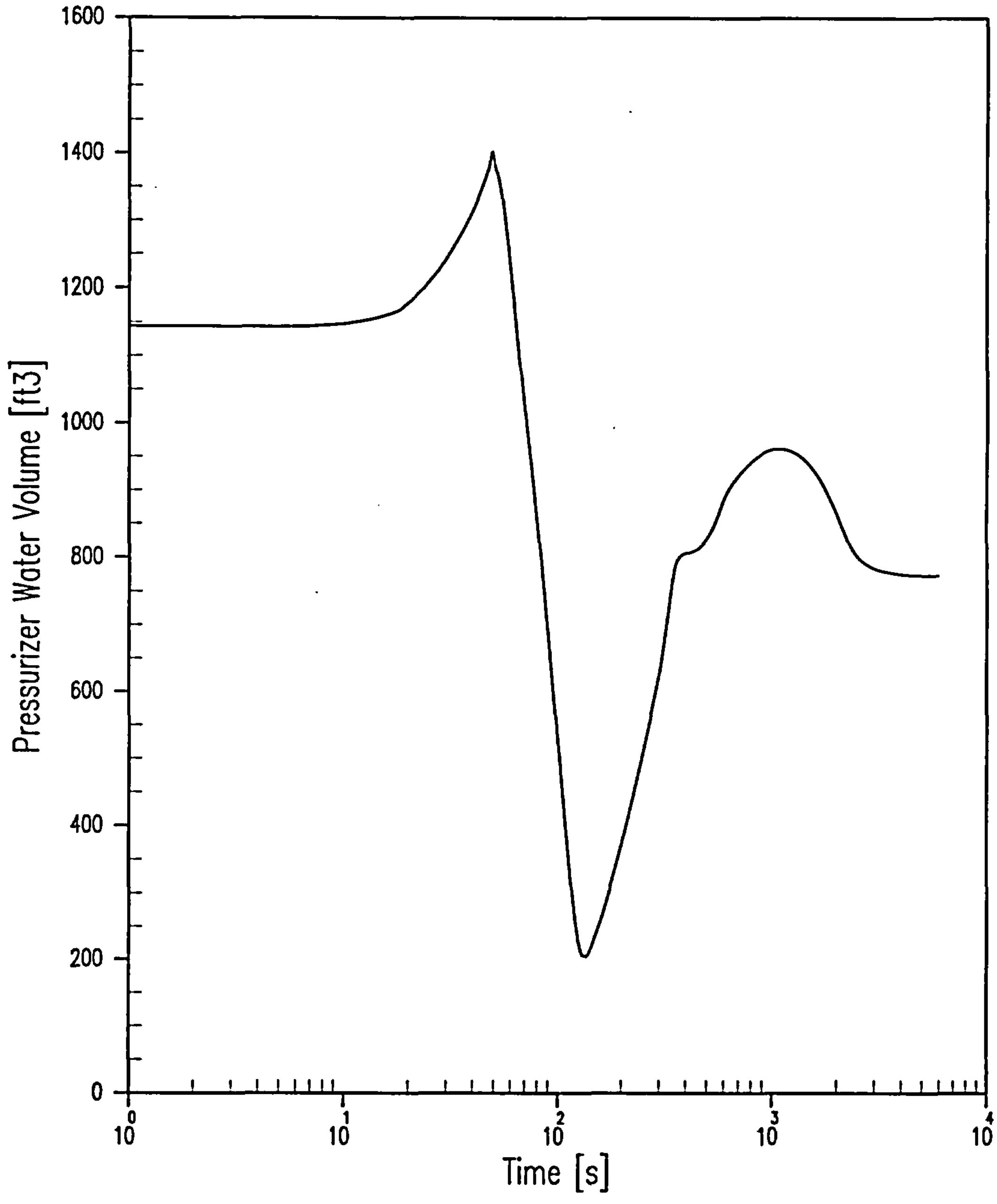


Figure 6.3.6-5 **Feedwater Line Rupture with Offsite Power Available – Pressurizer Water Volume Versus Time**

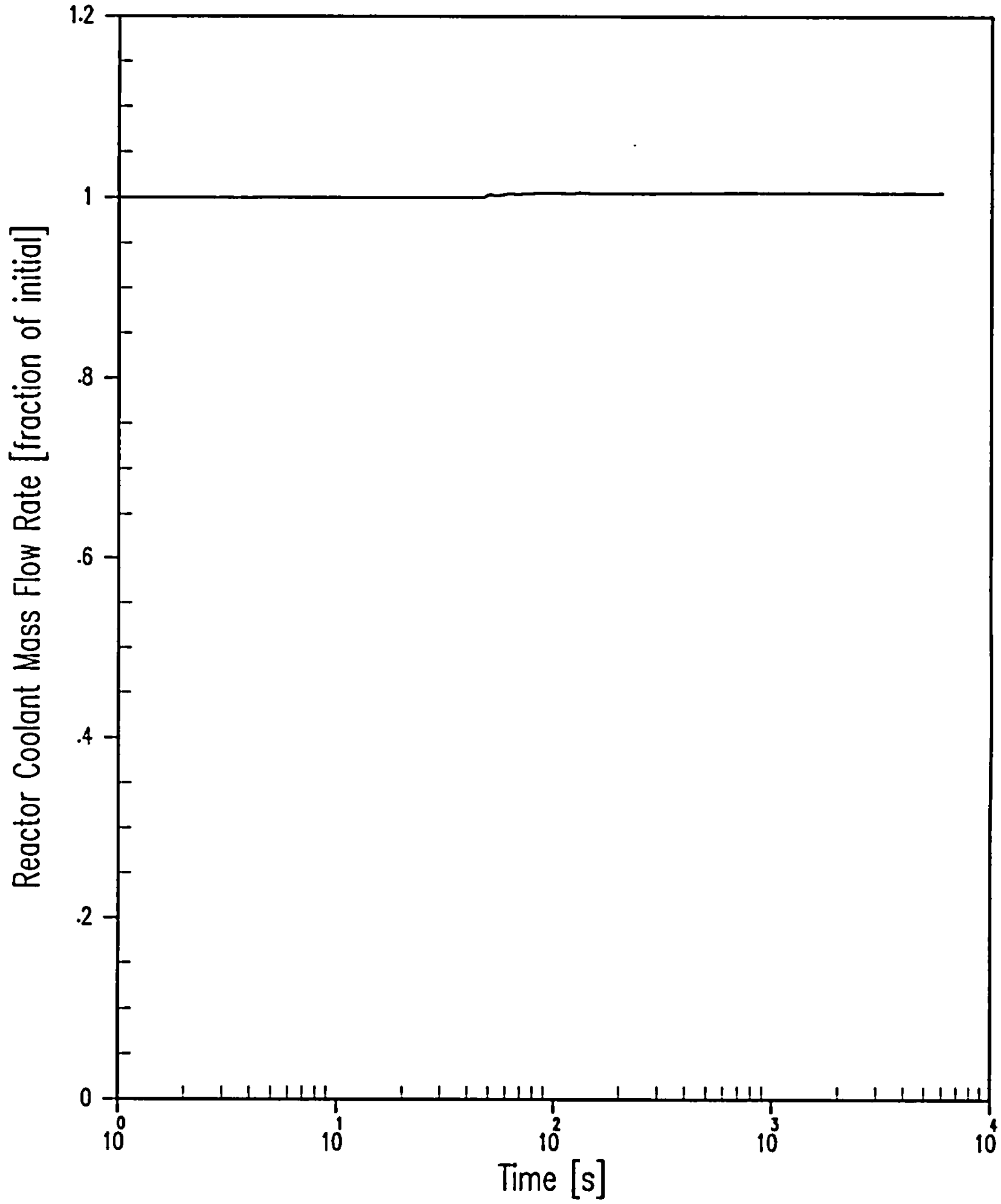


Figure 6.3.6-6 Feedwater Line Rupture with Offsite Power Available – Reactor Coolant Mass Flow Rate Versus Time

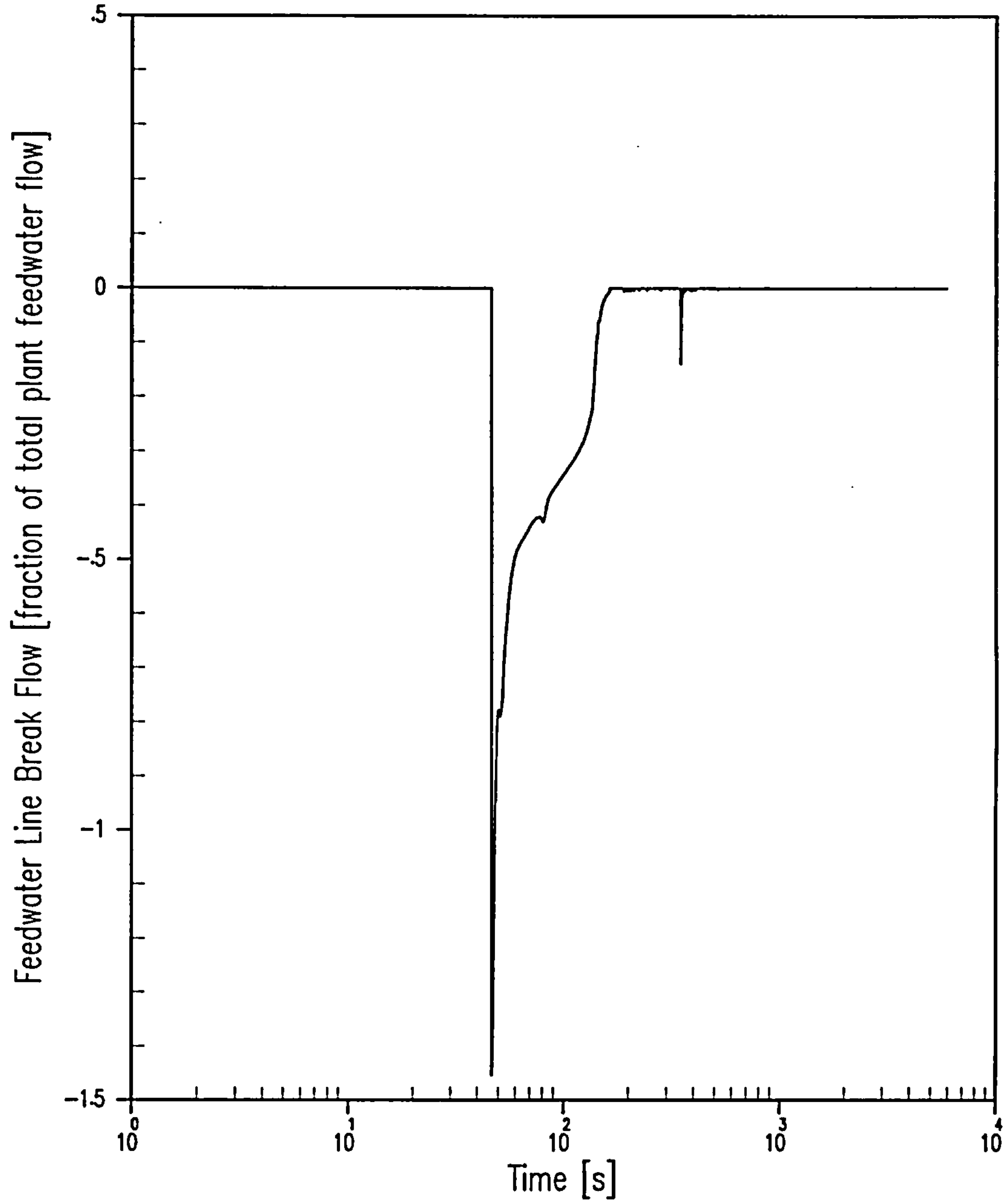


Figure 6.3.6-7 Feedwater Line Rupture with Offsite Power Available – Feedwater Line Break Flow Versus Time

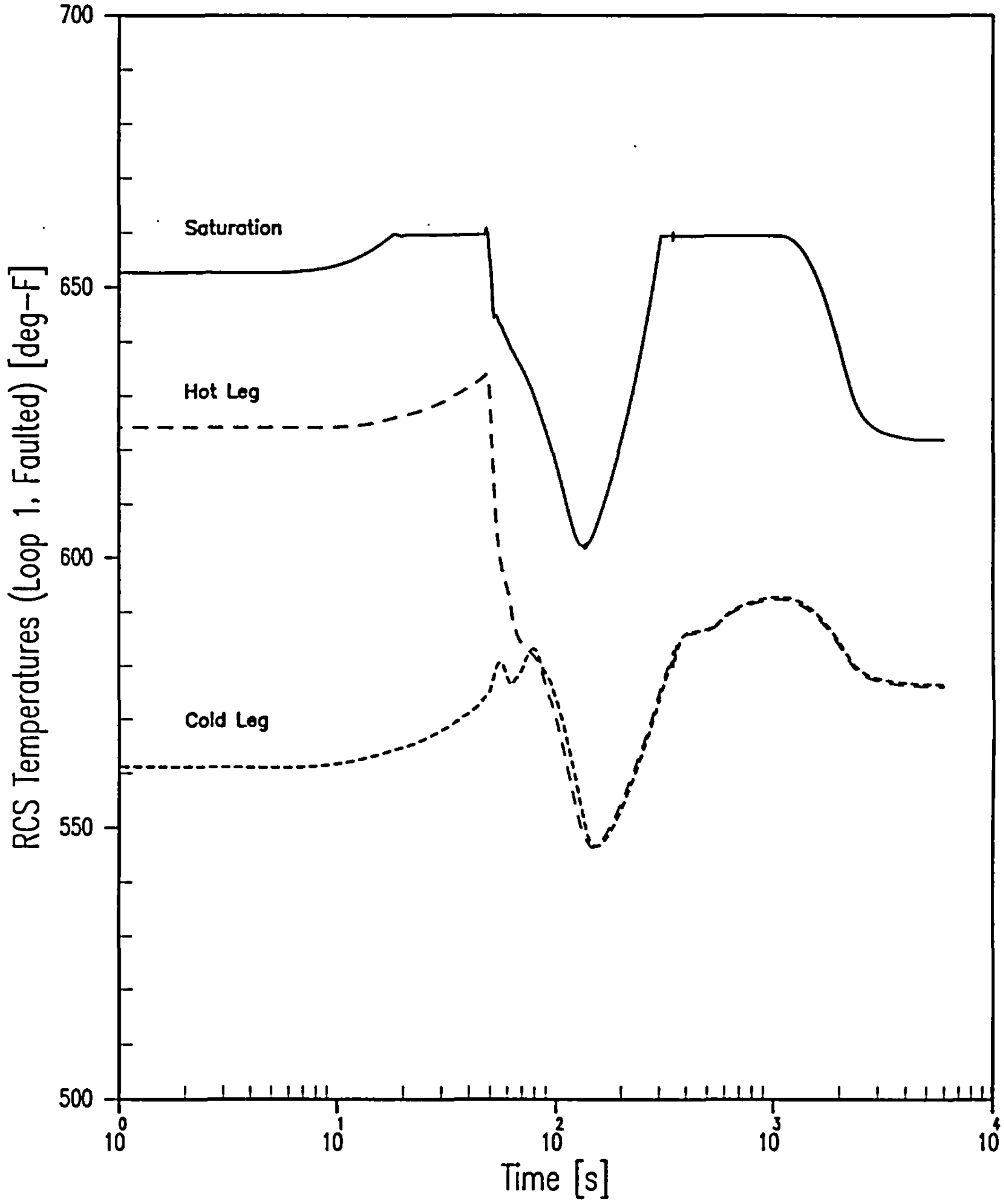


Figure 6.3.6-8 **Feedwater Line Rupture with Offsite Power Available – Reactor Coolant Temperatures – Loop 1 (Faulted Loop) Versus Time**

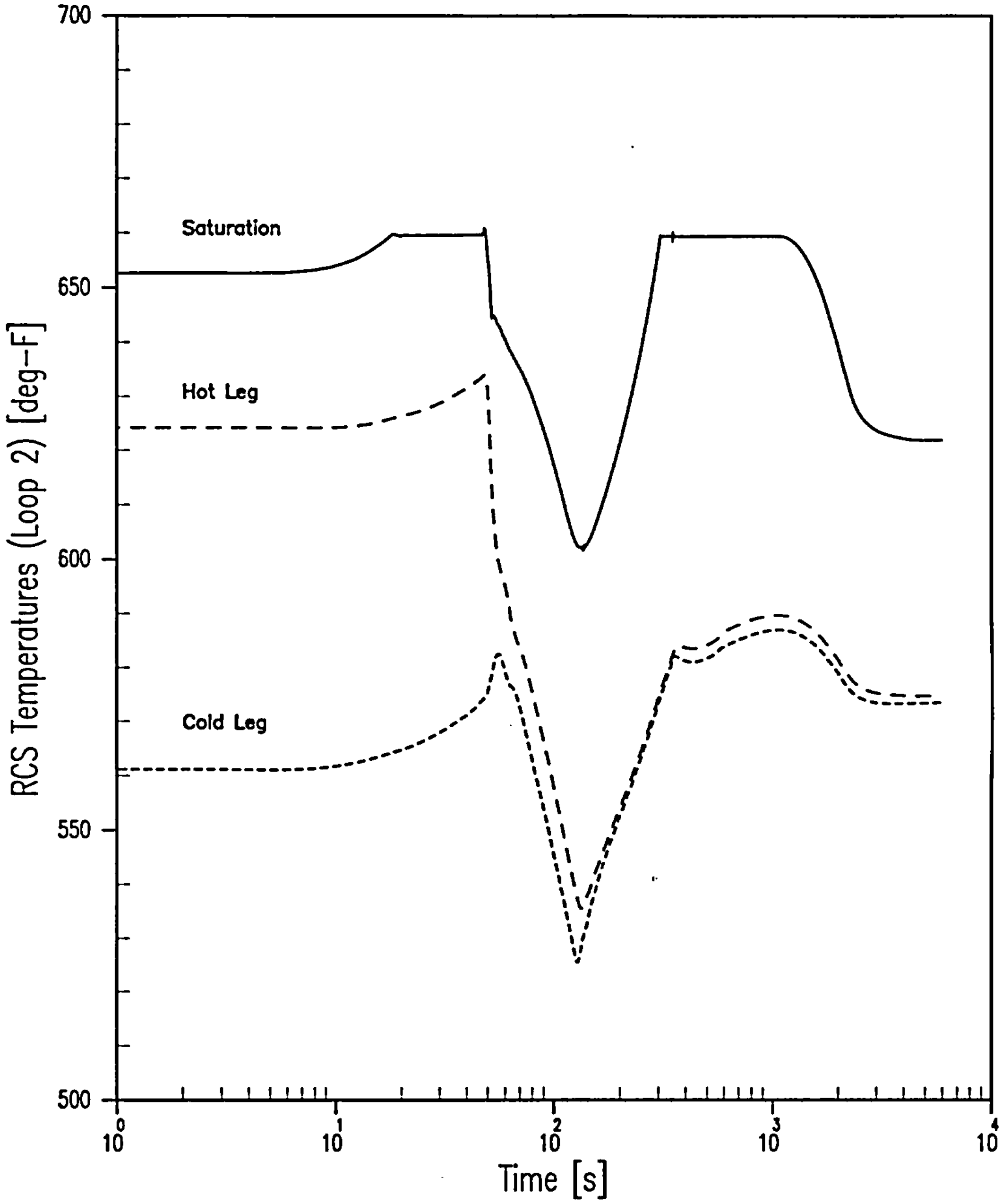


Figure 6.3.6-9 Feedwater Line Rupture with Offsite Power Available – Reactor Coolant Temperatures – Loop 2 Versus Time

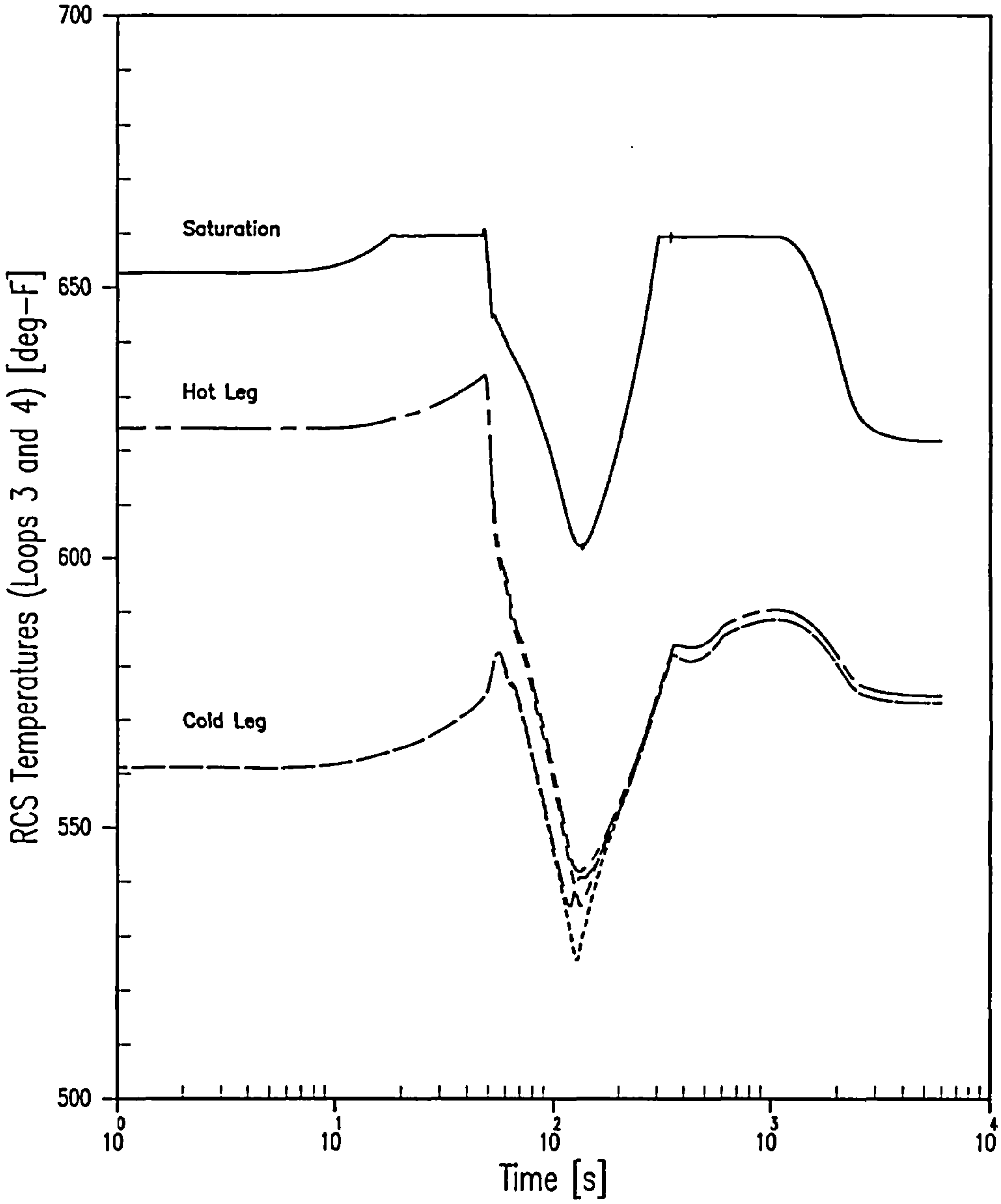


Figure 6.3.6-10 Feedwater Line Rupture with Offsite Power Available – Reactor Coolant Temperatures – Loops 3 and 4 Versus Time

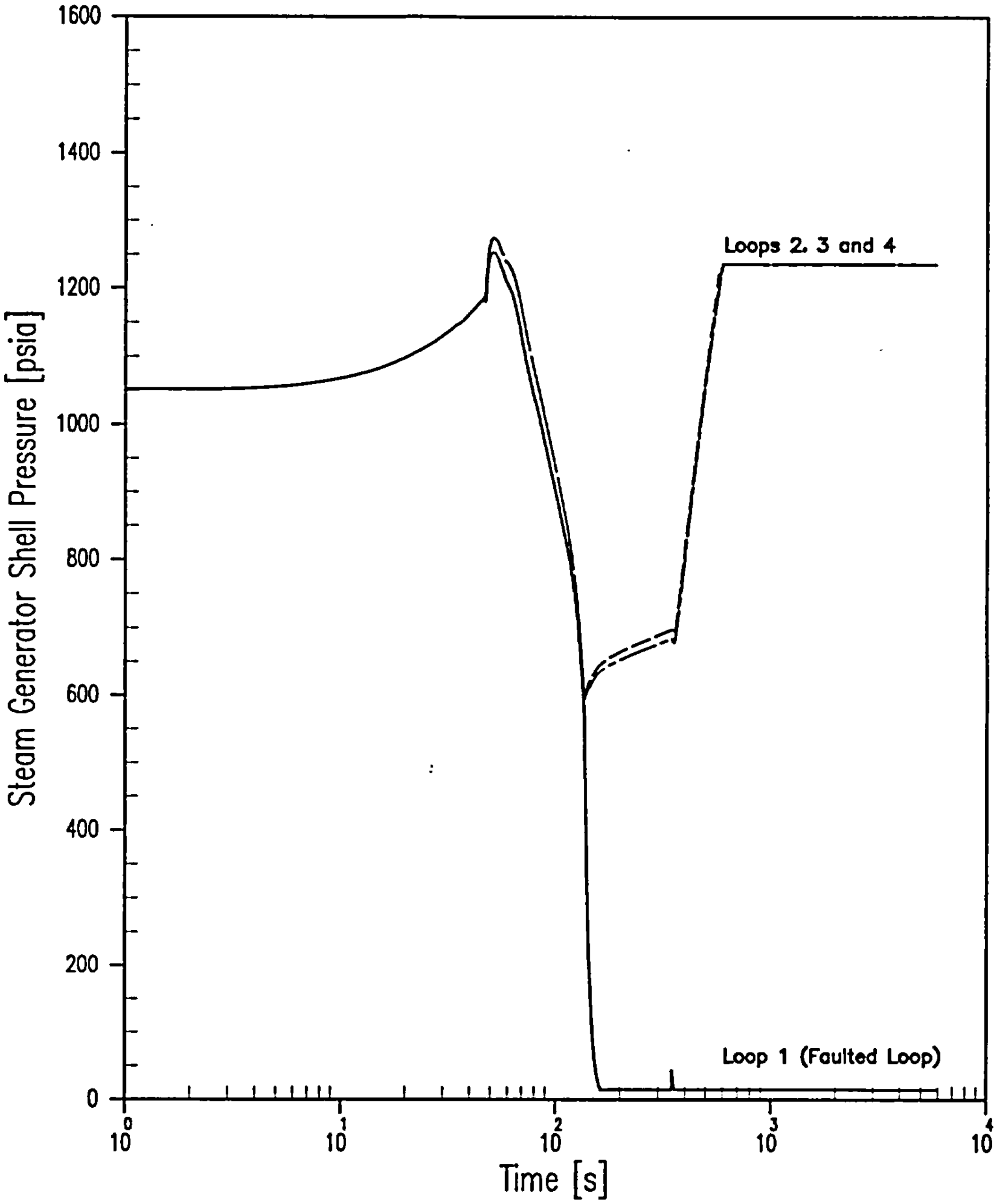


Figure 6.3.6-11 **Feedwater Line Rupture with Offsite Power Available – Steam Generator Shell Pressure Versus Time**

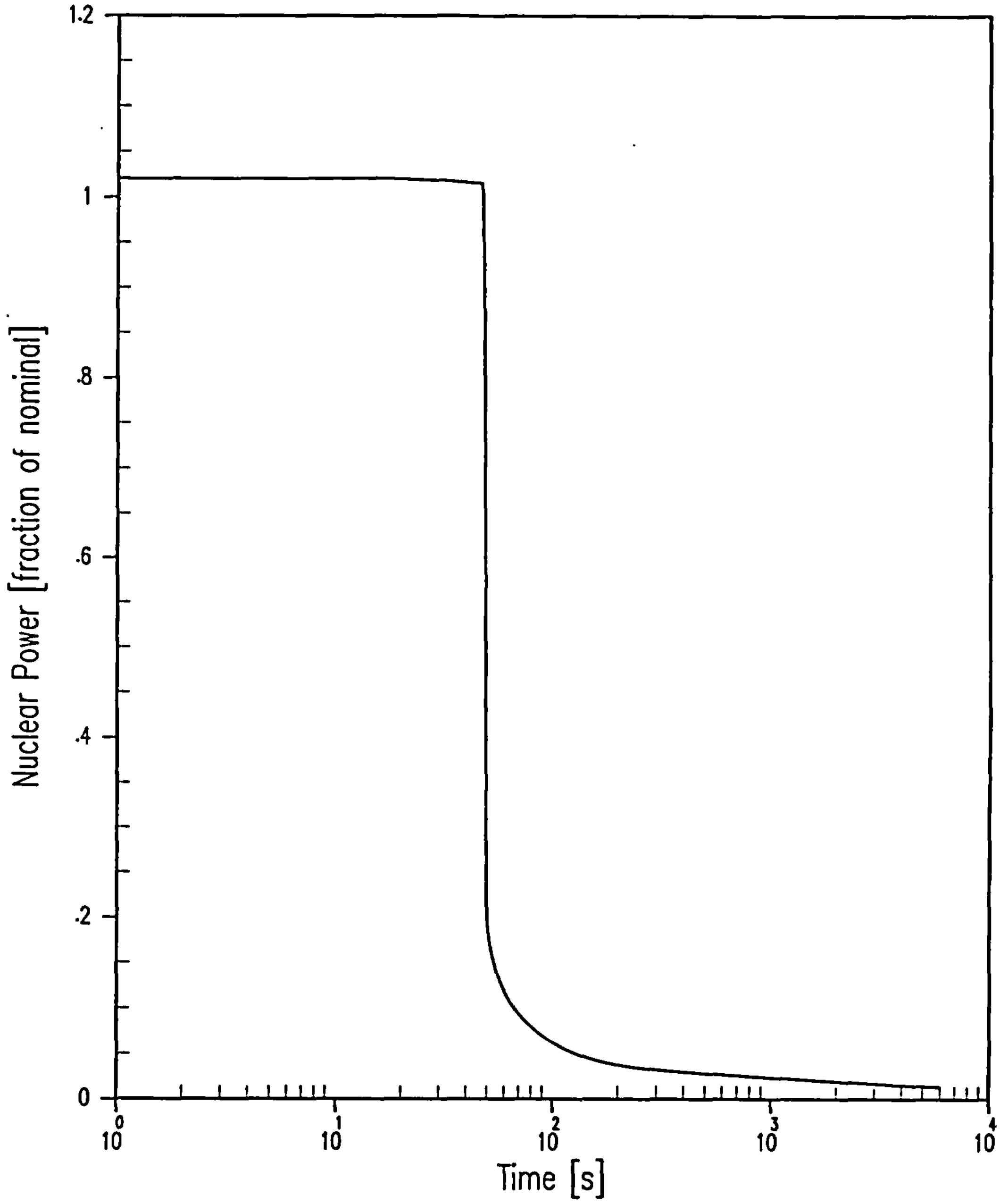


Figure 6.3.6-12 Feedwater Line Rupture Without Offsite Power Available – Nuclear Power Versus Time

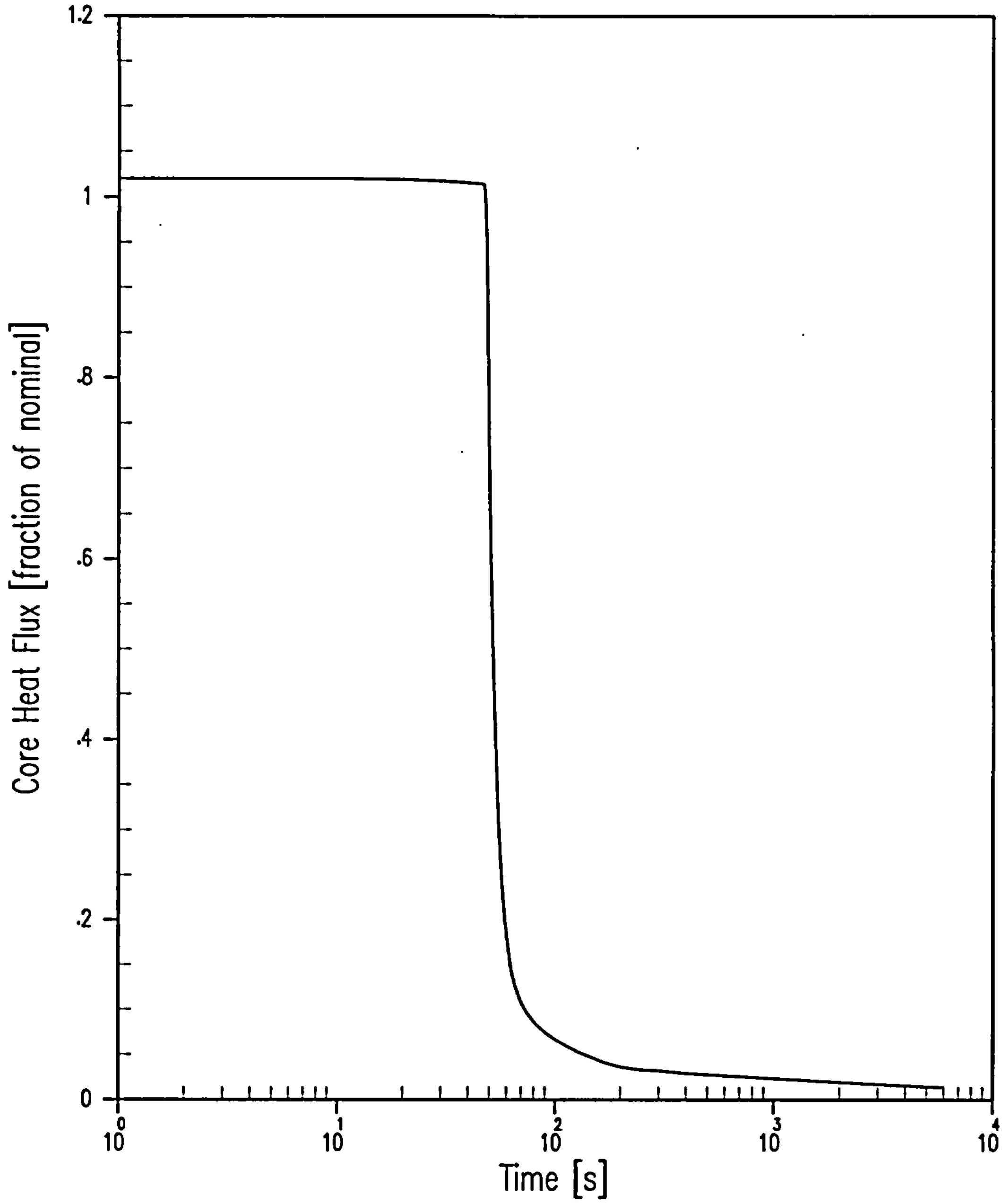


Figure 6.3.6-13 Feedwater Line Rupture Without Offsite Power Available – Core Heat Flux Versus Time

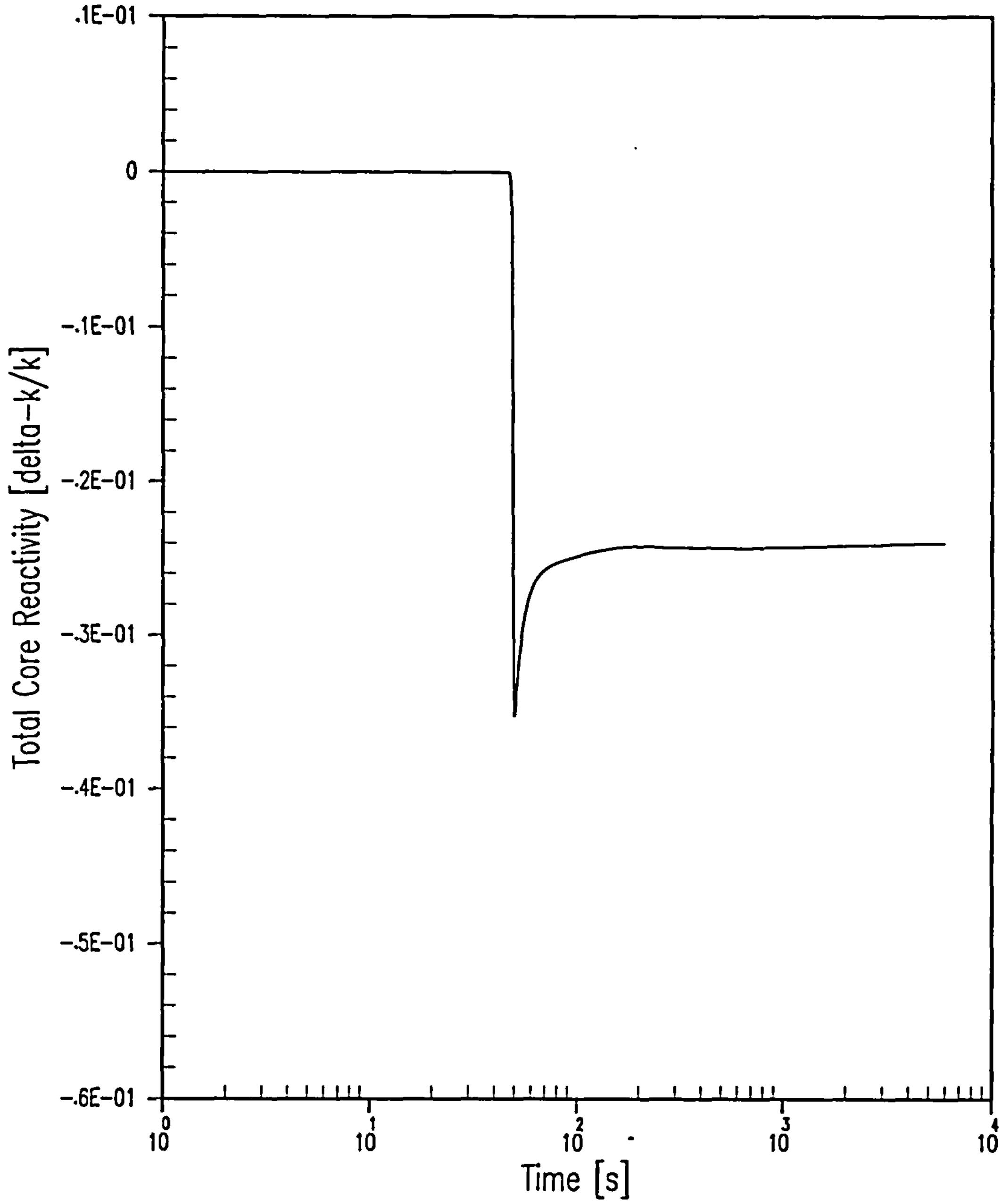


Figure 6.3.6-14 Feedwater Line Rupture Without Offsite Power Available – Total Core Reactivity Versus Time

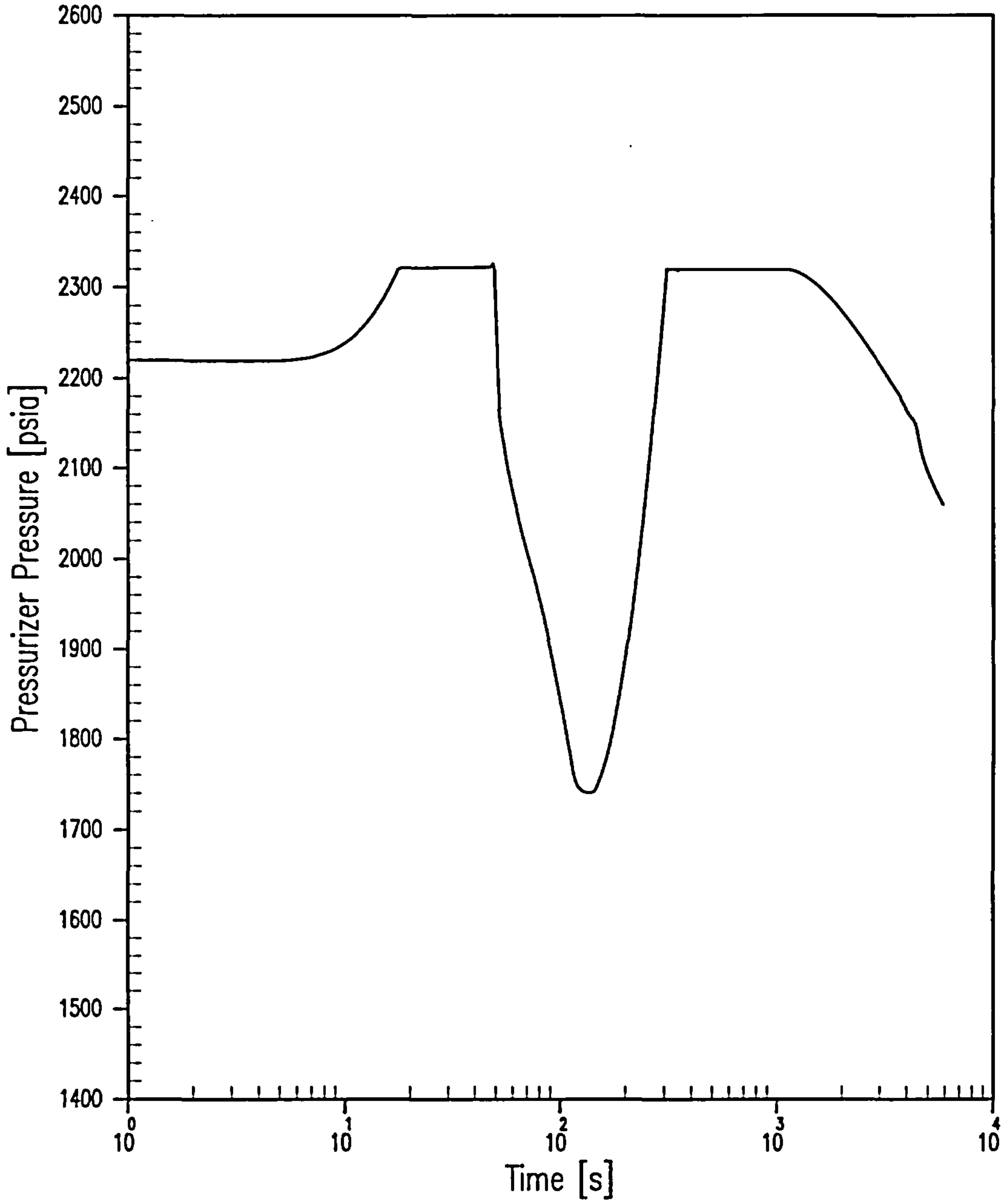


Figure 6.3.6-15 Feedwater Line Rupture Without Offsite Power Available – Pressurizer Pressure Versus Time

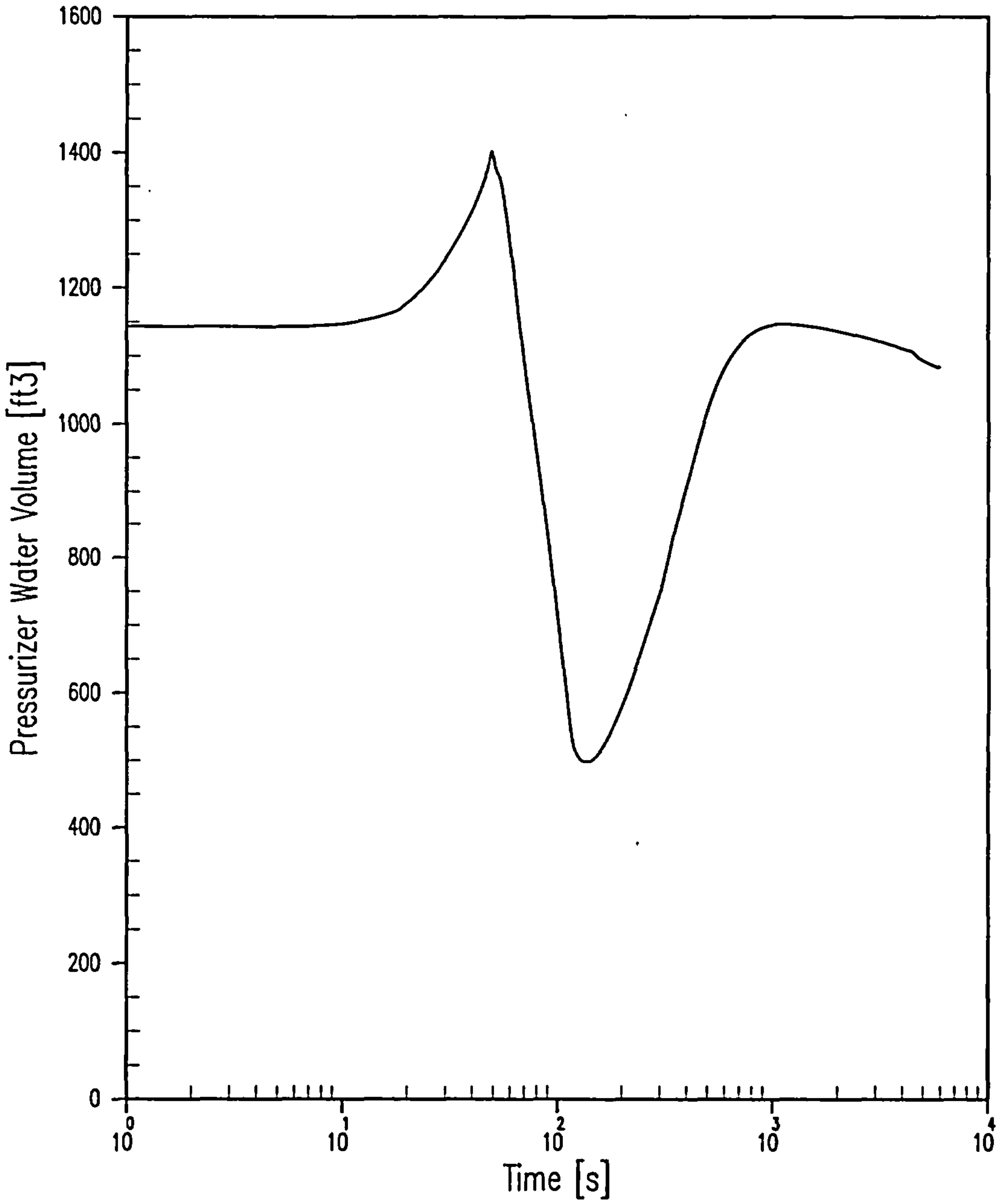


Figure 6.3.6-16 Feedwater Line Rupture Without Offsite Power Available – Pressurizer Water Volume Versus Time

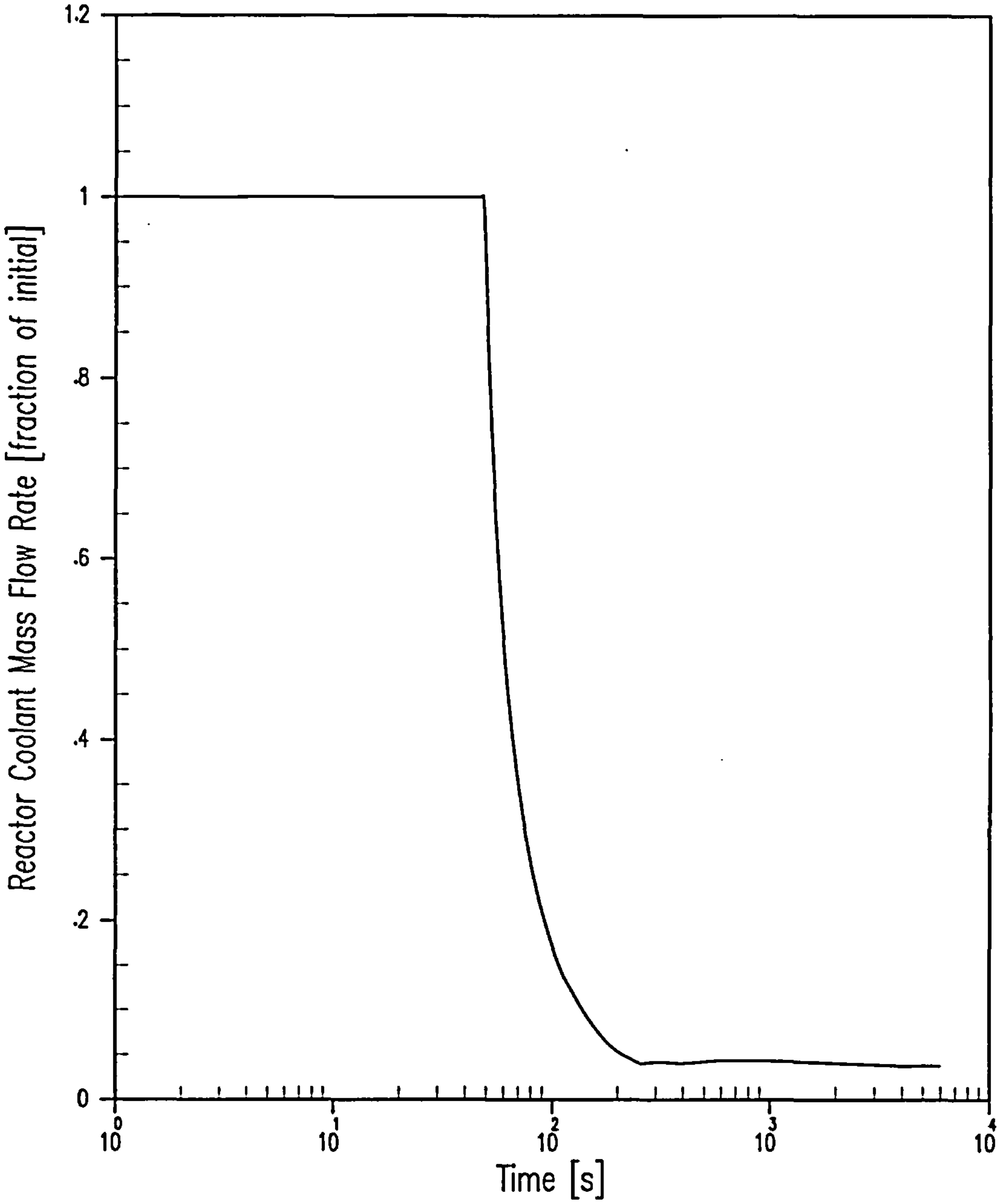


Figure 6.3.6-17 Feedwater Line Rupture Without Offsite Power Available – Reactor Coolant Mass Flow Rate Versus Time

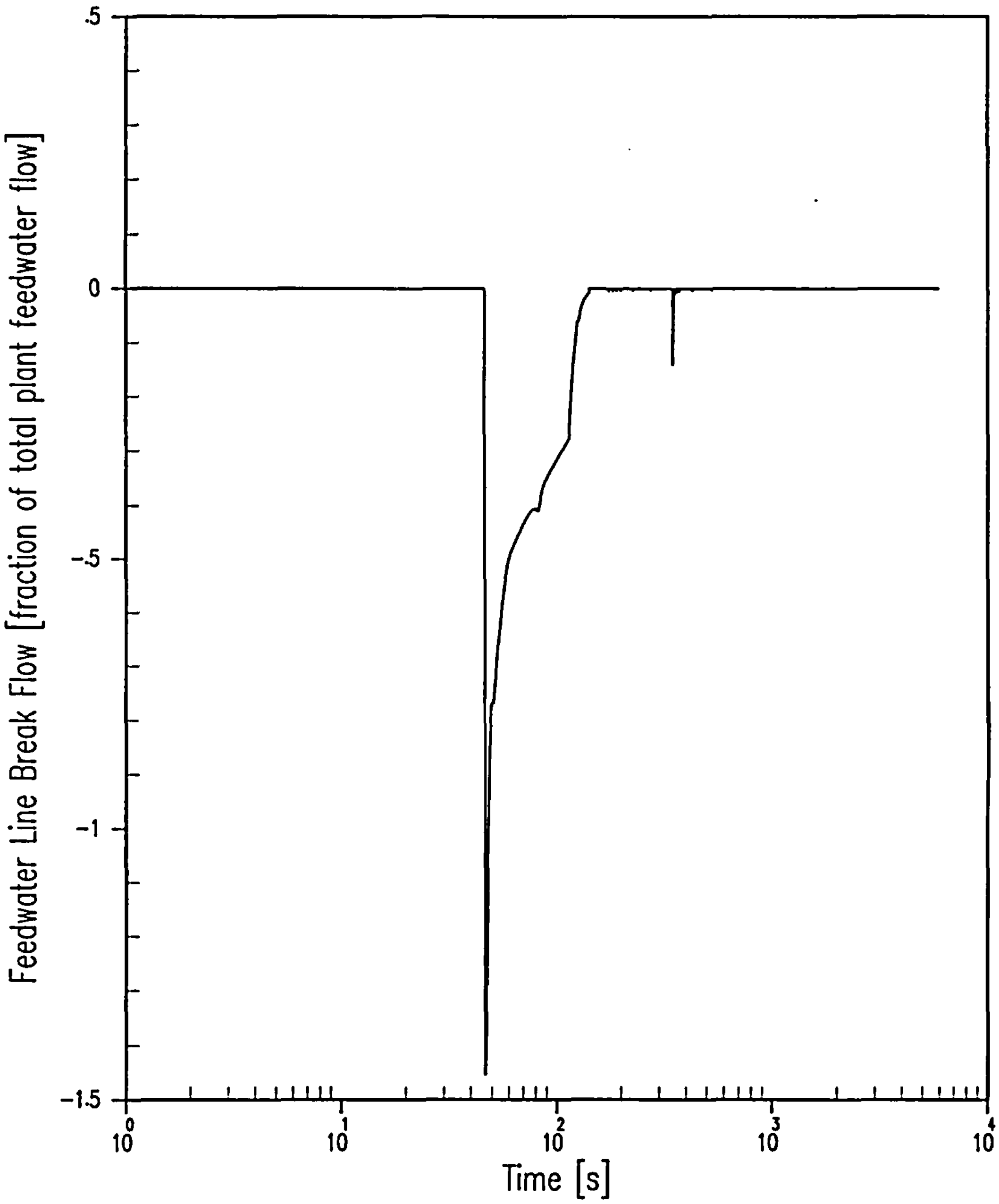


Figure 6.3.6-18 Feedwater Line Rupture Without Offsite Power Available – Feedwater Line Break Flow Versus Time

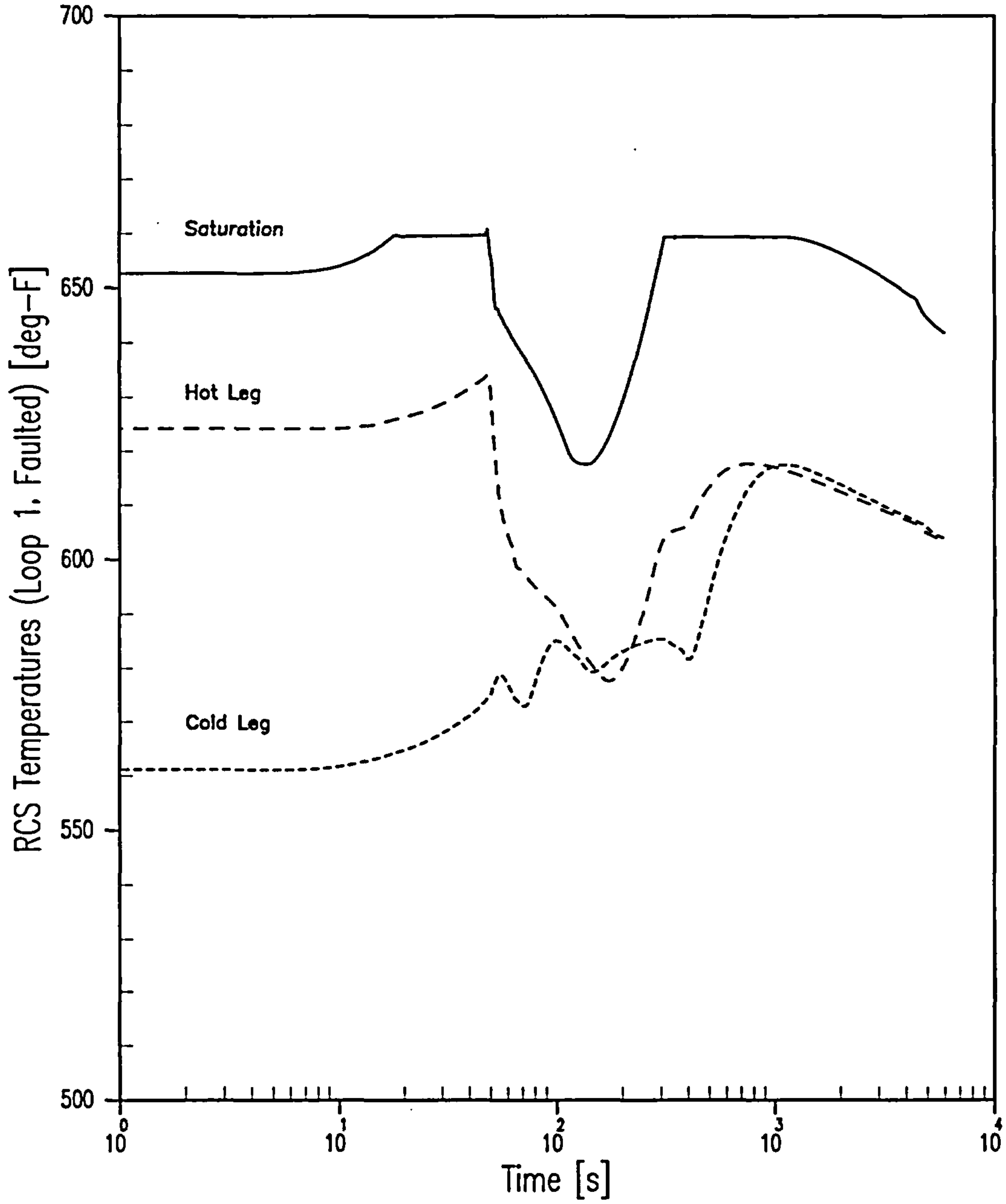


Figure 6.3.6-19 Feedwater Line Rupture Without Offsite Power Available – Reactor Coolant Temperatures – Loop 1 (Faulted Loop) Versus Time

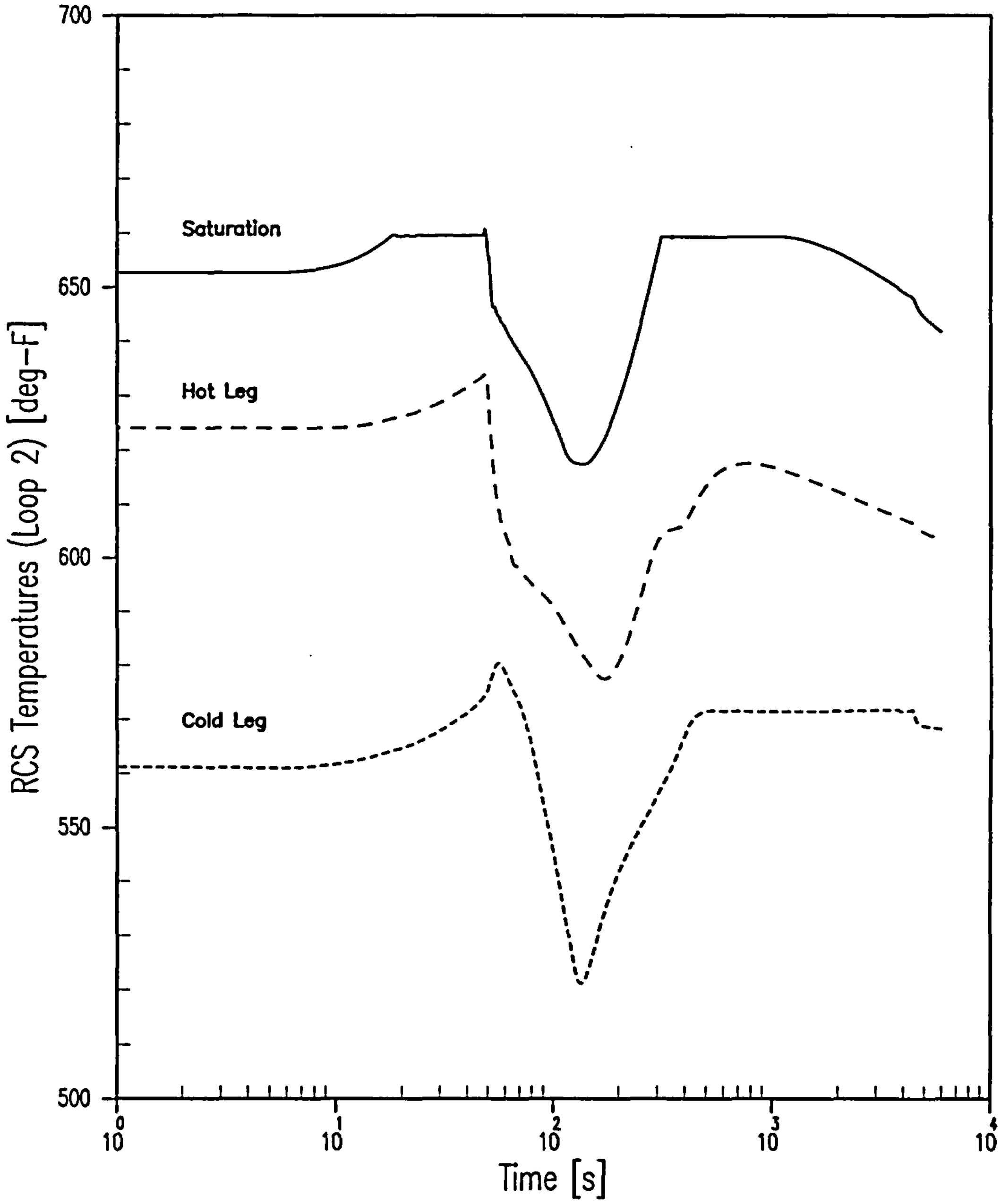


Figure 6.3.6-20 Feedwater Line Rupture Without Offsite Power Available – Reactor Coolant Temperatures – Loop 2 Versus Time

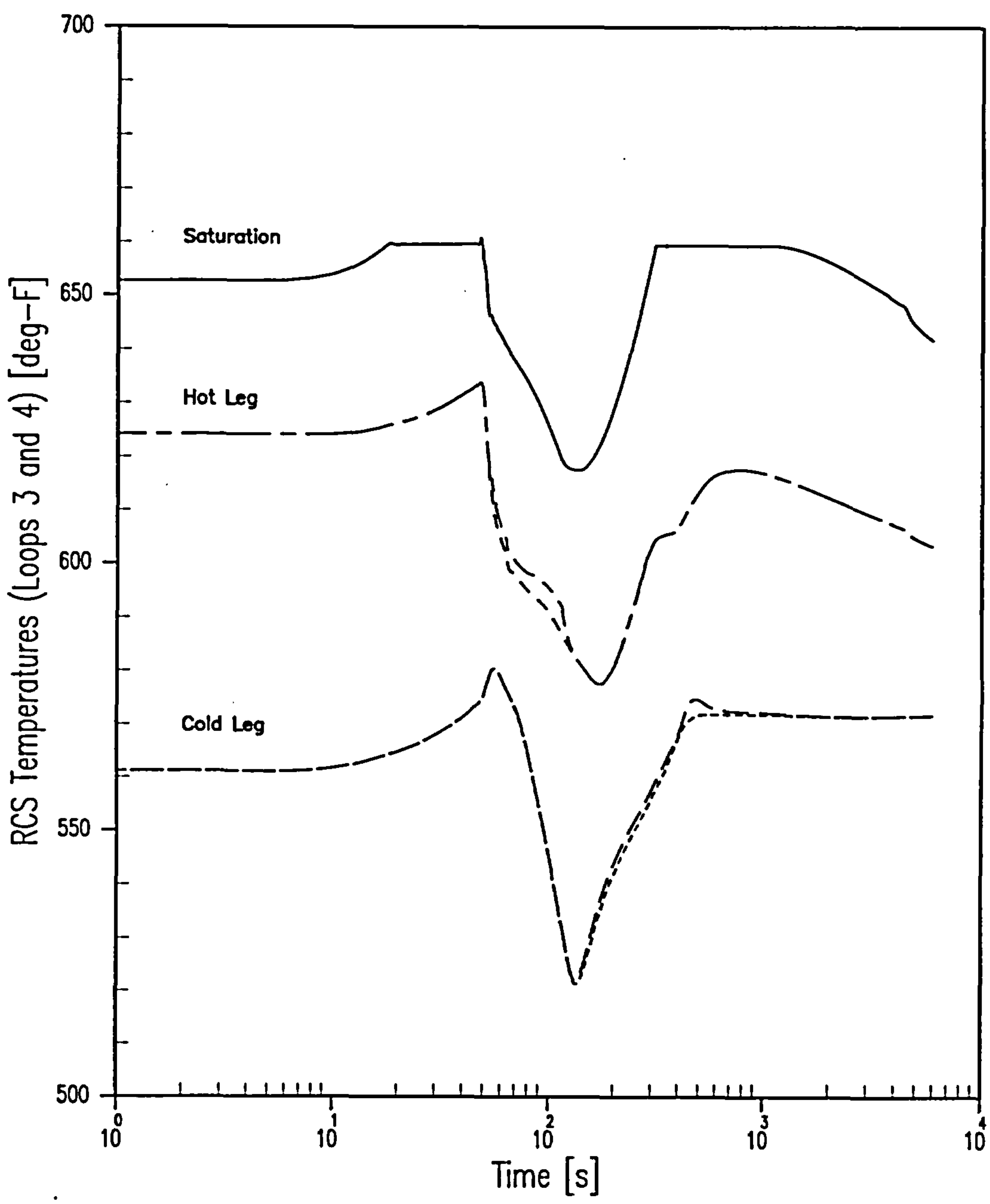


Figure 6.3.6-21 **Feedwater Line Rupture Without Offsite Power Available – Reactor Coolant Temperatures – Loops 3 and 4 Versus Time**

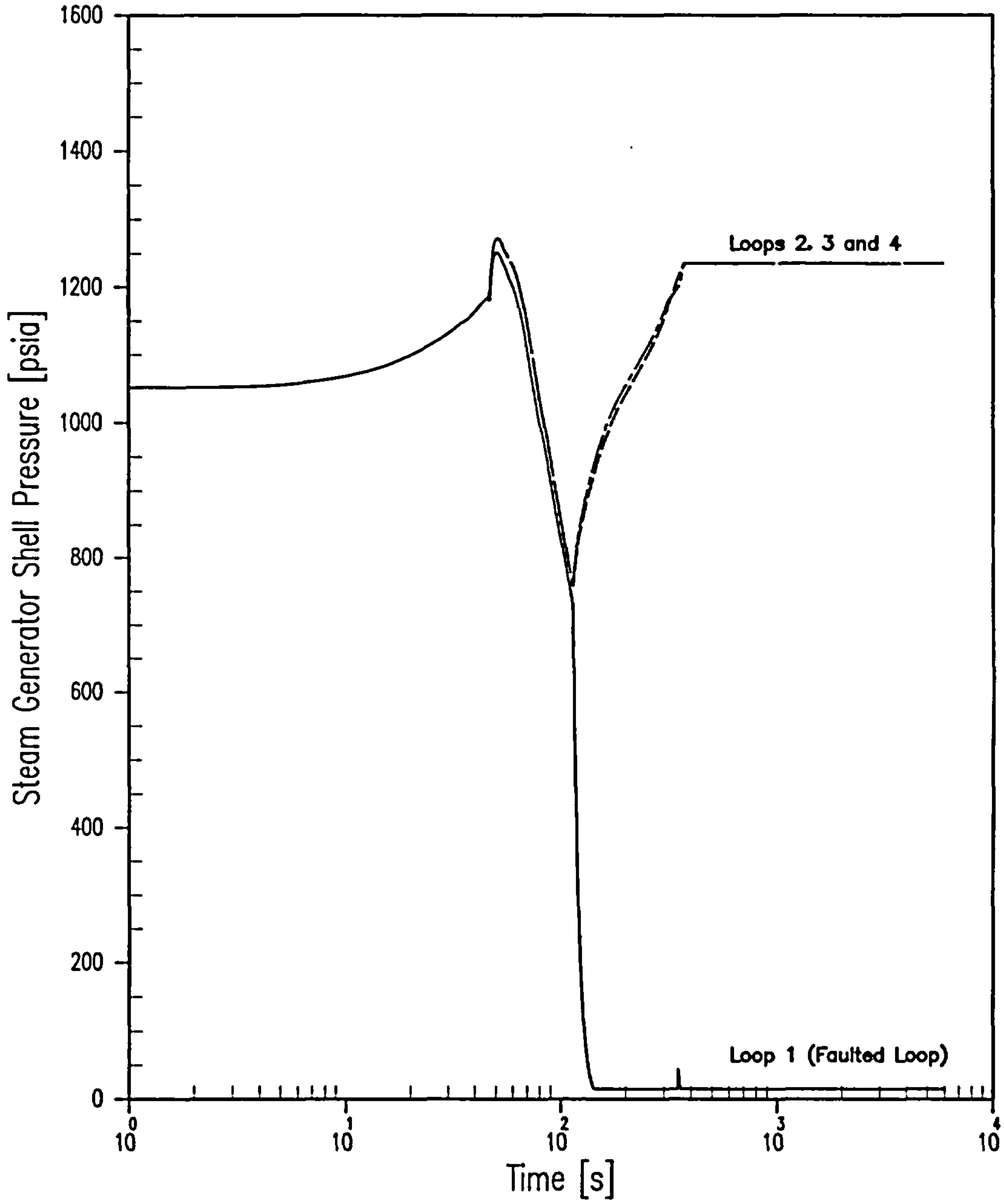


Figure 6.3.6-22 Feedwater Line Rupture Without Offsite Power Available – Steam Generator Shell Pressure Versus Time

6.3.7 Partial and Complete Loss of Forced Reactor Coolant Flow (FSAR Sections 15.3.1 and 15.3.2)

6.3.7.1 Partial Loss of Forced Reactor Coolant Flow

Accident Description

The partial loss-of-coolant-flow accident can result from a mechanical or electrical failure in a reactor coolant pump (RCP), or from a fault in the power supply to the RCP. If the reactor is at power at the time of the accident, the immediate effect of loss of coolant flow is a rapid increase in the coolant temperature. This increase could result in departure from nucleate boiling (DNB) with subsequent fuel damage if the reactor is not tripped promptly.

Normal power for the pumps is supplied through individual buses connected to the generator and the offsite power system. When a generator trip occurs, the buses continue to be supplied from external power lines, and the pumps continue to supply coolant to the core.

The necessary protection against a partial loss-of-coolant-flow accident is provided by the low primary coolant flow reactor trip signal, which is actuated in any reactor coolant loop by two-out-of-three low flow signals. Above 48-percent power (Permissive 8), low flow in any loop will actuate a reactor trip. Between 10-percent power (Permissive 7) and Permissive 8, low flow in any two loops will actuate a reactor trip.

Method of Analysis

The loss of two RCPs with all loops in operation event is analyzed to show that: (1) the integrity of the core is maintained as the DNB ratio (DNBR) remains above the safety analysis limit value, and (2) the peak reactor coolant system (RCS) and secondary system pressures remain below 110 percent of the design limits. Of these, the primary concern is assuring that the DNBR limit is met.

The partial loss-of-coolant-flow event is analyzed with two computer codes. First, the RETRAN computer code (Reference 1) is used to calculate the loop and core flow during the transient, the time of reactor trip based on the calculated flows, the nuclear power transient, and the primary-system pressure and temperature transients. The VIPRE computer code (Reference 5) is then used to calculate the hot channel heat flux transient and DNBR, based on the nuclear power and RCS temperature (enthalpy), pressure, and flow from RETRAN. The DNBR transients presented represent the minimum of the typical or thimble cell. The RETRAN and VIPRE computer codes are discussed in detail in Section 6.3.0.6 of this report.

This event is analyzed with the Revised Thermal Design Procedure (RTDP) (Reference 2). Initial reactor power, pressurizer pressure, and RCS temperature are assumed to be at their nominal values. Minimum measured flow is also assumed. A conservatively large absolute value of the Doppler-only power coefficient is used, along with the most-positive MTC limit for full-power operation (0 pcm/°F). These assumptions maximize the core power during the initial part of the transient when the minimum DNBR is reached.

A limiting end-of-cycle (EOC) DNB axial power shape is assumed in VIPRE for the calculation of DNBR. This shape provides the most limiting minimum DNBR for the loss-of-flow events.

A conservatively low trip reactivity value (4.0-percent $\Delta\rho$) is used to minimize the effect of rod insertion following reactor trip and maximize the heat flux statepoint used in the DNBR evaluation for this event. This value is based on the assumption that the highest worth rod cluster control assembly (RCCA) is stuck in its fully withdrawn position. A conservative trip reactivity worth versus rod position was modeled in addition to a conservative rod drop time (2.7 seconds to dashpot). The trip reactivity versus rod position curve is confirmed to be valid as part of the Reload Safety Analysis Checklist (RSAC) verification process.

The flow coastdown analysis is based on a momentum balance around each reactor coolant loop and across the reactor core. This momentum balance is combined with the continuity equation, a pump momentum balance, and the pump characteristics. Also, it is based on conservative estimates of system pressure losses.

A maximum, uniform, SGTP level of 5 percent was assumed in the RETRAN analysis.

Results

Figures 6.3.7-1 through 6.3.7-8 illustrate the transient response for the loss of two RCPs with all loops in operation. The minimum DNBR is 1.90/1.94 (thimble/typical), which occurred at 3.6 seconds. The applicable safety analysis DNBR limit is 1.55 for the thimble cell and 1.59 for the typical cell.

The calculated sequence of events table is shown in Table 6.3.7-1. This transient trips on a low primary reactor coolant flow trip setpoint, which is assumed to be 87.0 percent of loop flow. Following reactor trip, the affected RCPs continue to coast down, and the core flow reaches a new equilibrium value corresponding to the remaining pumps still in operation. With the reactor tripped, a stable plant condition is eventually attained. Normal plant shutdown may then proceed.

Conclusions

The analysis performed demonstrates that for the partial loss-of-coolant-flow event, the DNBR does not decrease below the safety analysis limit value at any time during the transient. Therefore, no fuel or cladding damage is predicted and all applicable acceptance criteria are met.

6.3.7.2 Complete Loss of Forced Reactor Coolant Flow

Accident Description

A complete loss of forced reactor coolant flow may result from a simultaneous loss of electrical supplies to all RCPs. If the reactor is at power at the time of the accident, the immediate effect of loss of coolant flow is a rapid increase in the coolant temperature. This increase could result in DNB with subsequent fuel damage if the reactor were not tripped promptly.

Normal power for the RCPs is supplied through buses from a transformer connected to the generator and the offsite power system. When a generator trip occurs, the buses continue to be supplied from external power lines and the pumps continue to supply coolant flow to the core.

The following signals provide the necessary protection against a complete loss-of-flow accident:

- Reactor coolant pump undervoltage reactor trip
- Low reactor coolant loop flow reactor trip
- Reactor coolant pump underfrequency reactor trip

The reactor trip on RCP undervoltage is provided to protect against conditions that can cause a loss of voltage to all RCPs; that is, station blackout. This function is blocked below the P-7 permissive.

The reactor trip on RCP underfrequency is available to trip the reactor for an underfrequency condition, resulting from frequency disturbances on the power grid.

The reactor trip on low primary coolant flow is provided to protect against partial loss-of-flow conditions. This function is generated by two-out-of-three low flow signals per reactor coolant loop. Above the P-8 permissive low flow in any loop will actuate a reactor trip. Between the 10-percent power (Permissive P-7) and the power level corresponding to Permissive P-8 (48-percent power), low flow in any two loops will actuate a reactor trip. This function serves as a backup to the RCP undervoltage and underfrequency functions for complete loss-of-flow cases.

This event is conservatively analyzed to the following acceptance criteria:

- Pressure in the RCS and main steam system (MSS) should be maintained below 110 percent of the design values.

- Fuel cladding integrity shall be maintained by ensuring that the minimum DNBR remains above the limit value.

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

Method of Analysis

The complete loss-of-flow transient is analyzed as a loss of all RCPs with all loops in operation. The event is analyzed to show that the integrity of the core is maintained as the DNBR remains above the safety analysis limit value. The loss-of-flow events do result in an increase in RCS and MSS pressures, but these pressure increases are generally not severe enough to challenge the integrity of the RCS and MSS. Since the maximum RCS and MSS pressures do not exceed 110 percent of their respective design pressures for the loss-of-load event, it is concluded that the maximum RCS and MSS pressures will also remain below 110 percent of their respective design pressures for the loss-of-flow events.

Two cases are analyzed:

- Complete loss-of-flow transient due to a loss of power to all pumps
- Complete loss-of-flow transient due to an underfrequency condition

The underfrequency case represents the worst credible coolant flow loss. For this case, flow decreases due to a constant frequency decay rate of 5 Hz/s. Reactor trip is then caused by an underfrequency signal.

The transients are analyzed with two computer codes. First, the RETRAN computer code (Reference 1) is used to calculate the loop and core flow during the transient, the time of reactor trip, the nuclear power transient, and the primary-system pressure and temperature transients. The VIPRE computer code (Reference 5) is then used to calculate the heat flux and DNBR transients based on the nuclear power and RCS temperature (enthalpy), pressure, and flow from RETRAN. The DNBR transients presented represent the minimum of the typical or thimble cell for the fuel. The RETRAN and VIPRE computer codes are discussed in detail in Section 6.3.0.6 of this report.

This event is analyzed with RTDP (Reference 2). Initial reactor power, pressurizer pressure, and RCS temperature are assumed to be at their nominal values. Minimum measured flow is also assumed. A conservatively large absolute value of the Doppler-only power coefficient is used, along with the most-positive MTC limit for full-power operation (0 pcm/°F). These assumptions maximize the core power during the initial part of the transient when the minimum DNBR is reached.

A limiting EOC DNB axial power shape is assumed in VIPRE for the calculation of DNBR. This shape provides the most limiting minimum DNBR for the loss-of-flow events.

A conservatively low trip reactivity value (4.0-percent $\Delta\rho$) is used to minimize the effect of rod insertion following reactor trip and maximize the heat flux statepoint used in the DNBR evaluation for this event. This value is based on the assumption that the highest worth RCCA is stuck in its fully withdrawn position. A conservative trip reactivity worth versus rod position was modeled in addition to a conservative rod drop time (2.7 seconds to dashpot). The trip reactivity versus rod position curve is confirmed to be valid as part of the RSAC verification process.

The flow coastdown analysis is based on a momentum balance around each reactor coolant loop and across the reactor core. This momentum balance is combined with the continuity equation, a pump momentum balance, and the pump characteristics. Also, it is based on conservative estimates of system pressure losses.

A maximum, uniform, SGTP level of 5 percent was assumed in the RETRAN analysis.

Results

Figures 6.3.7-9 through 6.3.7-16 illustrate the transient response for the complete loss of flow associated with a loss of power to all RCPs with all loops in operation. The minimum DNBR is 1.76/1.79 (thimble/typical) which occurred at 3.2 seconds. The applicable safety analysis DNBR limit is 1.55 for the thimble cell and 1.59 for the typical cell.

Figures 6.3.7-17 through 6.3.7-24 illustrate the transient response for the complete loss-of-flow (underfrequency) case. All RCPs decelerate at a constant rate until a reactor trip on underfrequency is initiated. The minimum DNBR is 1.78/1.80 (thimble/typical), which occurred at 3.0 seconds. The applicable safety analysis DNBR limit is 1.55 for the thimble cell and 1.59 for the typical cell.

The calculated sequence of events for both complete loss-of-flow cases are shown in Table 6.3.7-2. Following reactor trip, the RCPs will continue to coast down, and natural circulation flow will eventually be established. With the reactor tripped, a stable plant condition will eventually be attained. Normal plant shutdown may then proceed.

Conclusions

The analysis performed has demonstrated that for the complete loss-of-flow event, the DNBR does not decrease below the safety analysis limit value at any time during the transient. Therefore, no fuel or cladding damage is predicted and all applicable acceptance criteria are met.

Table 6.3.7-1 Time Sequence of Events for Partial Loss of Forced Reactor Coolant Flow	
Event	**Time (Seconds)**
Two Operating RCPs Lose Power and Begin Coasting Down	0.0
Low Flow Reactor Trip Setpoint is Reached	1.55
Rods Begin to Drop	2.55
Minimum DNBR Occurs	3.6
Results	
Minimum DNBR Value (thm/typ)	1.90/1.94
DNBR Limit (thm/typ)	1.55/1.59

Table 6.3.7-2 Time Sequence of Events for Complete Loss of Forced Reactor Coolant Flow	
Complete Loss of Flow	
Event	**Time (Seconds)**
All Operating RCPs Lose Power and Coastdown Begins	0.0
Reactor Coolant Pump Undervoltage Setpoint Reached	0.0
Rods Begin to Drop	1.5
Minimum DNBR Occurs	3.2
Results	
Minimum DNBR Value (thm/typ)	1.76/1.79
DNBR Limit (thm/typ)	1.55/1.59
Complete Loss of Flow - Underfrequency	
Event	**Time (Seconds)**
Frequency Decay Begins and All Operating RCPs Begin to Decelerate	0.0
Reactor Coolant Pump Underfrequency Setpoint Reached	0.6
Rods Begin to Drop	1.2
Minimum DNBR Occurs	3.0
Results	
Minimum DNBR Value (thm/typ)	1.78/1.80
DNBR Limit (thm/typ)	1.55/1.59

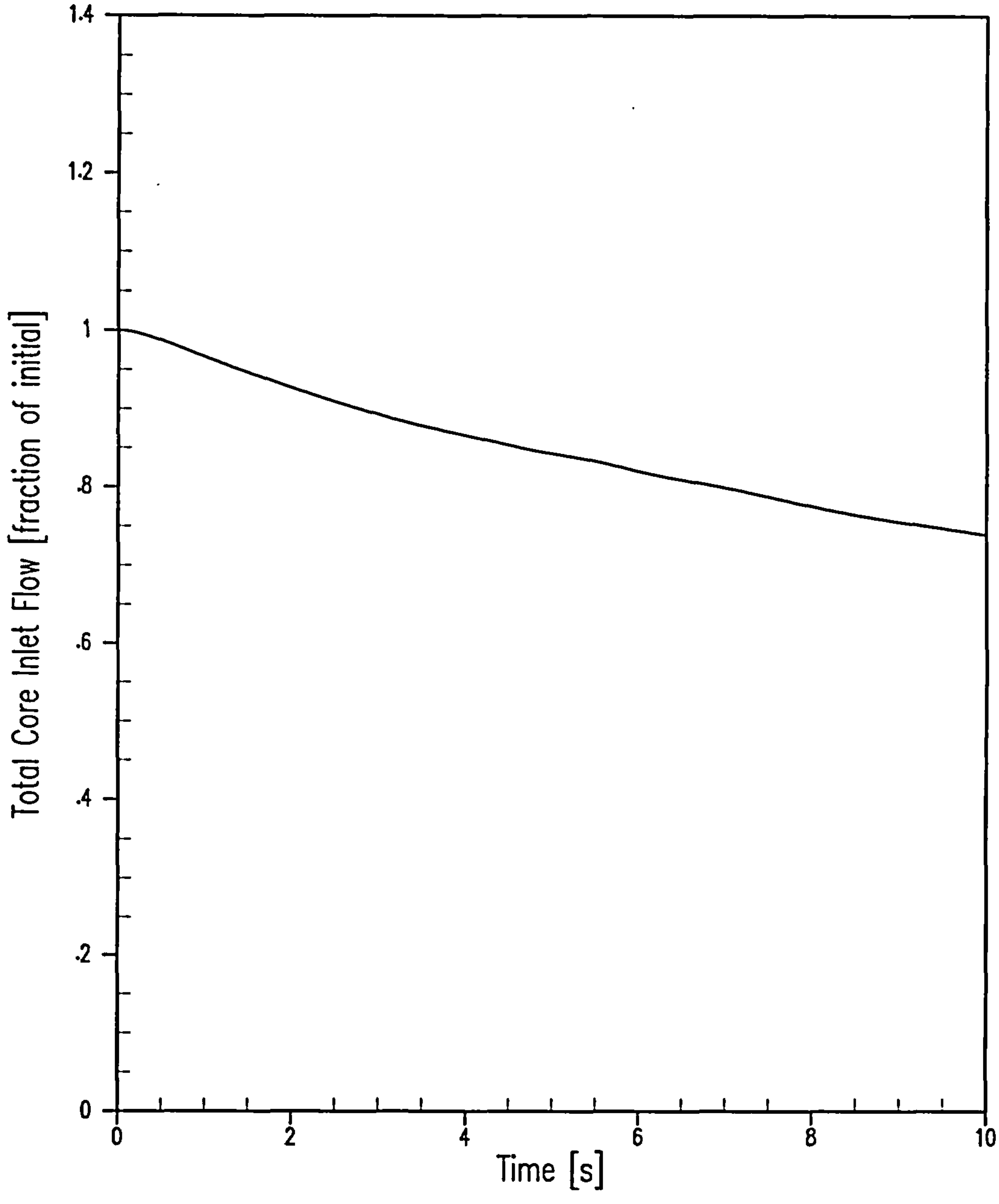


Figure 6.3.7-1 **Partial Loss of Flow, Two Pumps Coasting Down – Total Core Inlet Flow versus Time**

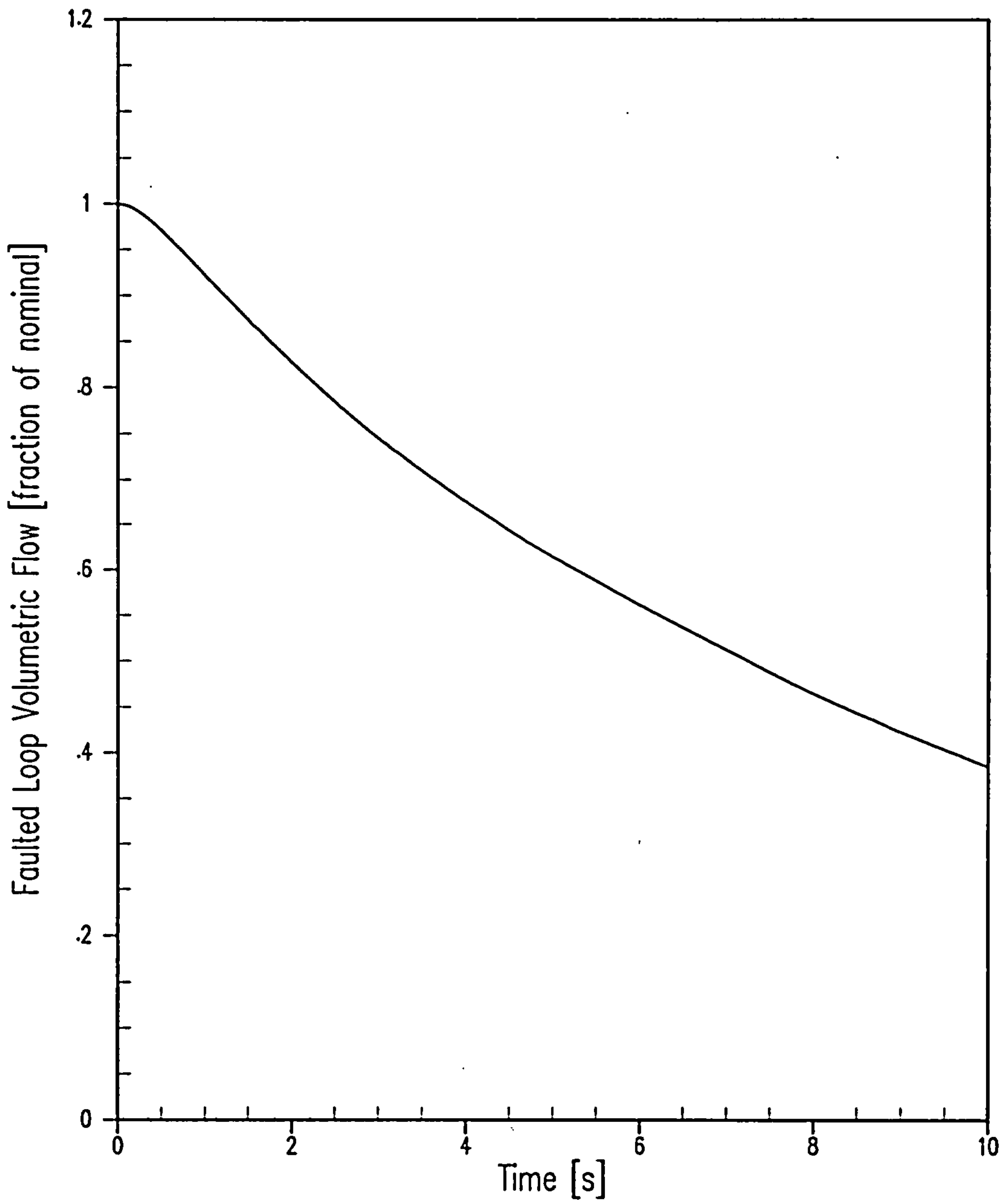


Figure 6.3.7-2 **Partial Loss of Flow, Two Pumps Coasting Down – RCS Faulted Loop Flow versus Time**

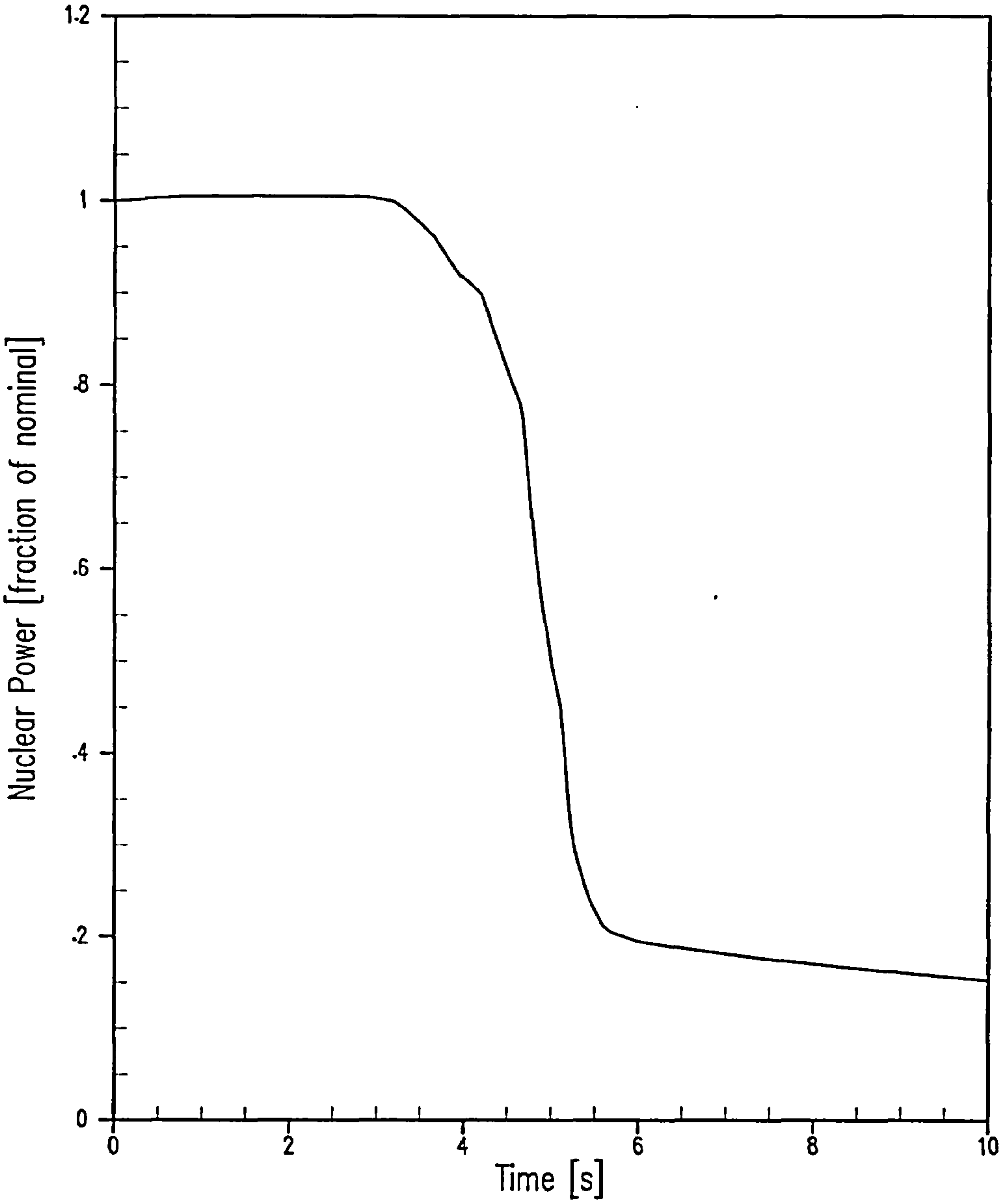


Figure 6.3.7-3 **Partial Loss of Flow, Two Pumps Coasting Down – Nuclear Power versus Time**

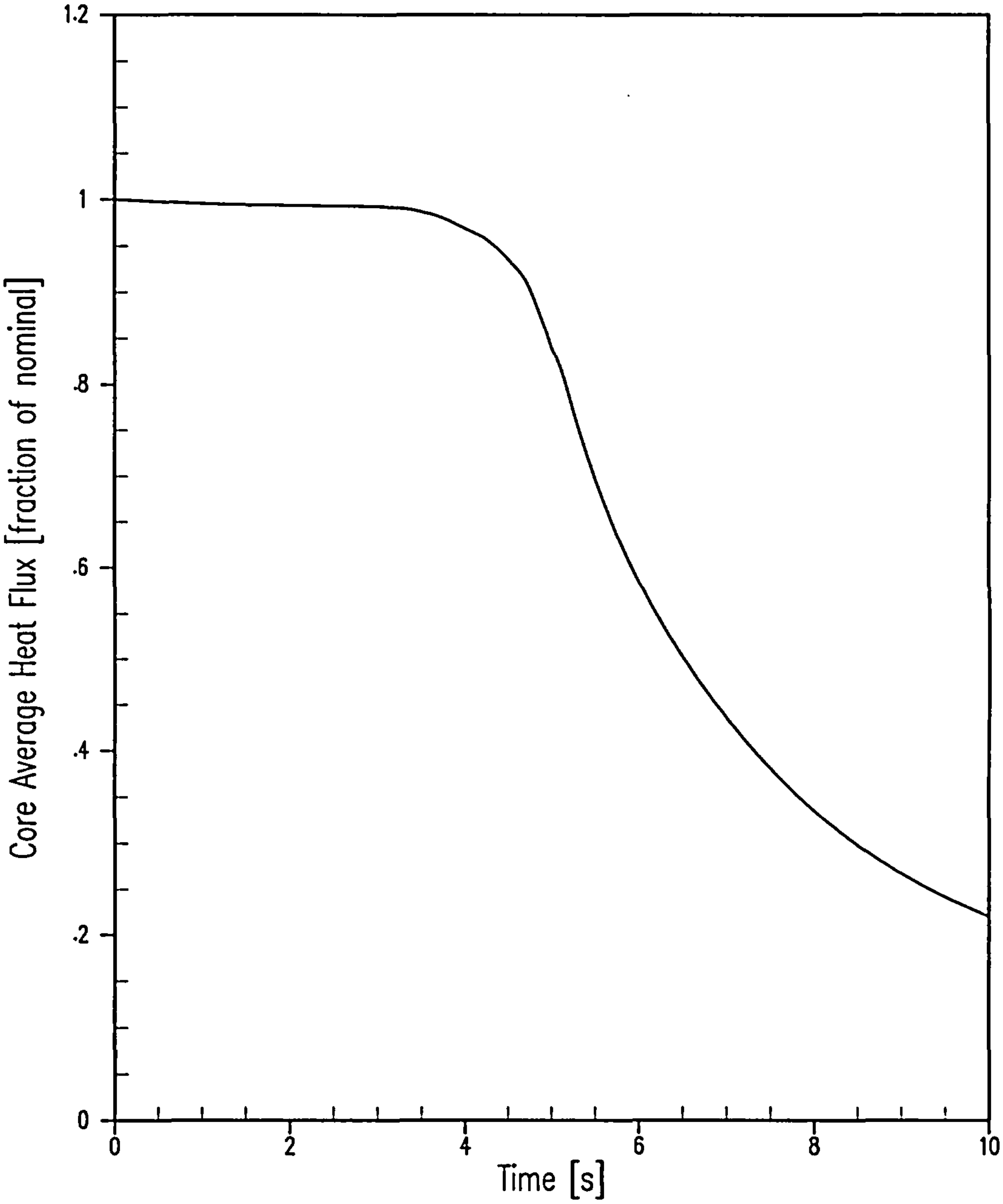


Figure 6.3.7-4 **Partial Loss of Flow, Two Pumps Coasting Down – Core Average Heat Flux versus Time**

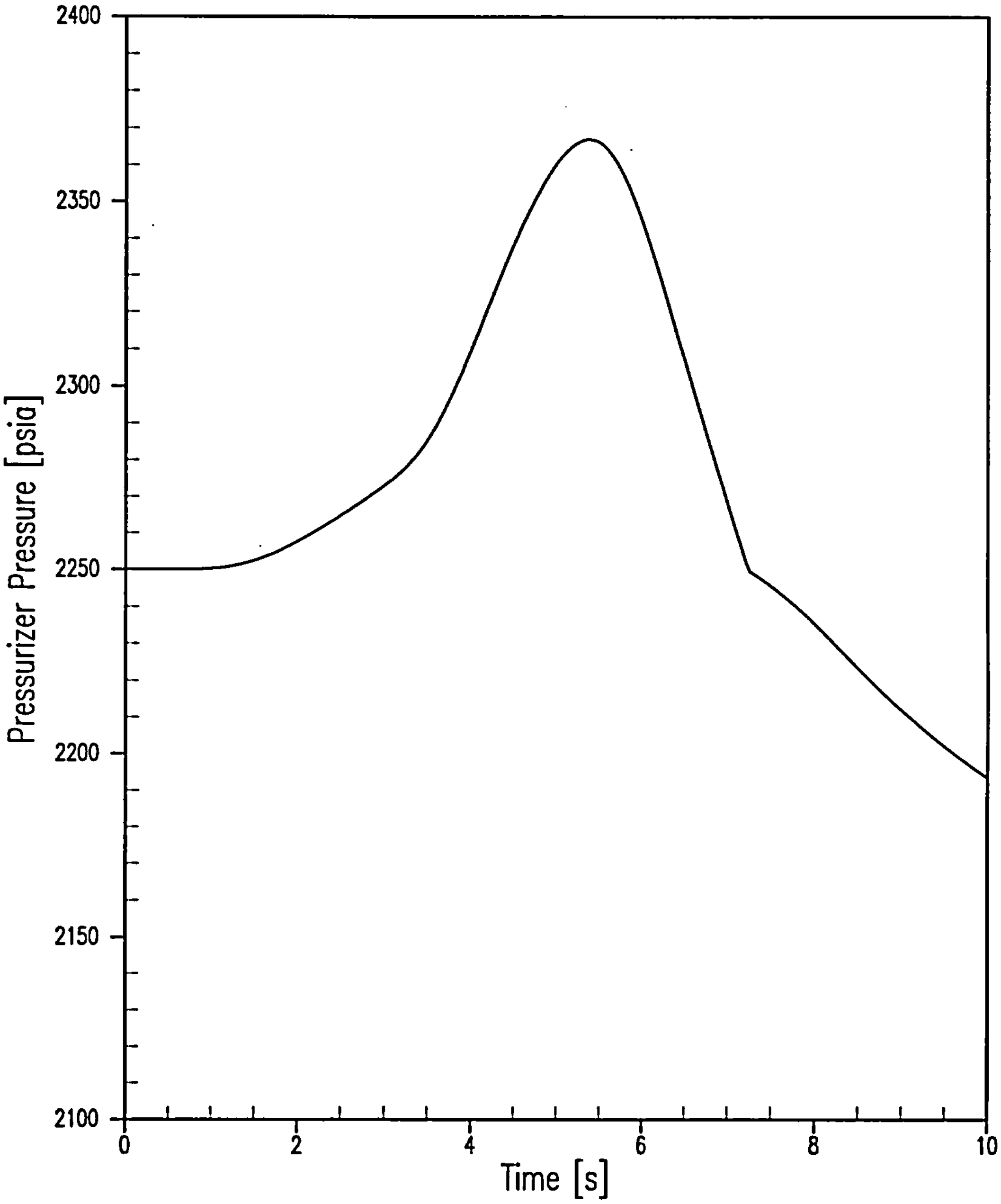


Figure 6.3.7-5 **Partial Loss of Flow, Two Pumps Coasting Down – Pressurizer Pressure versus Time**

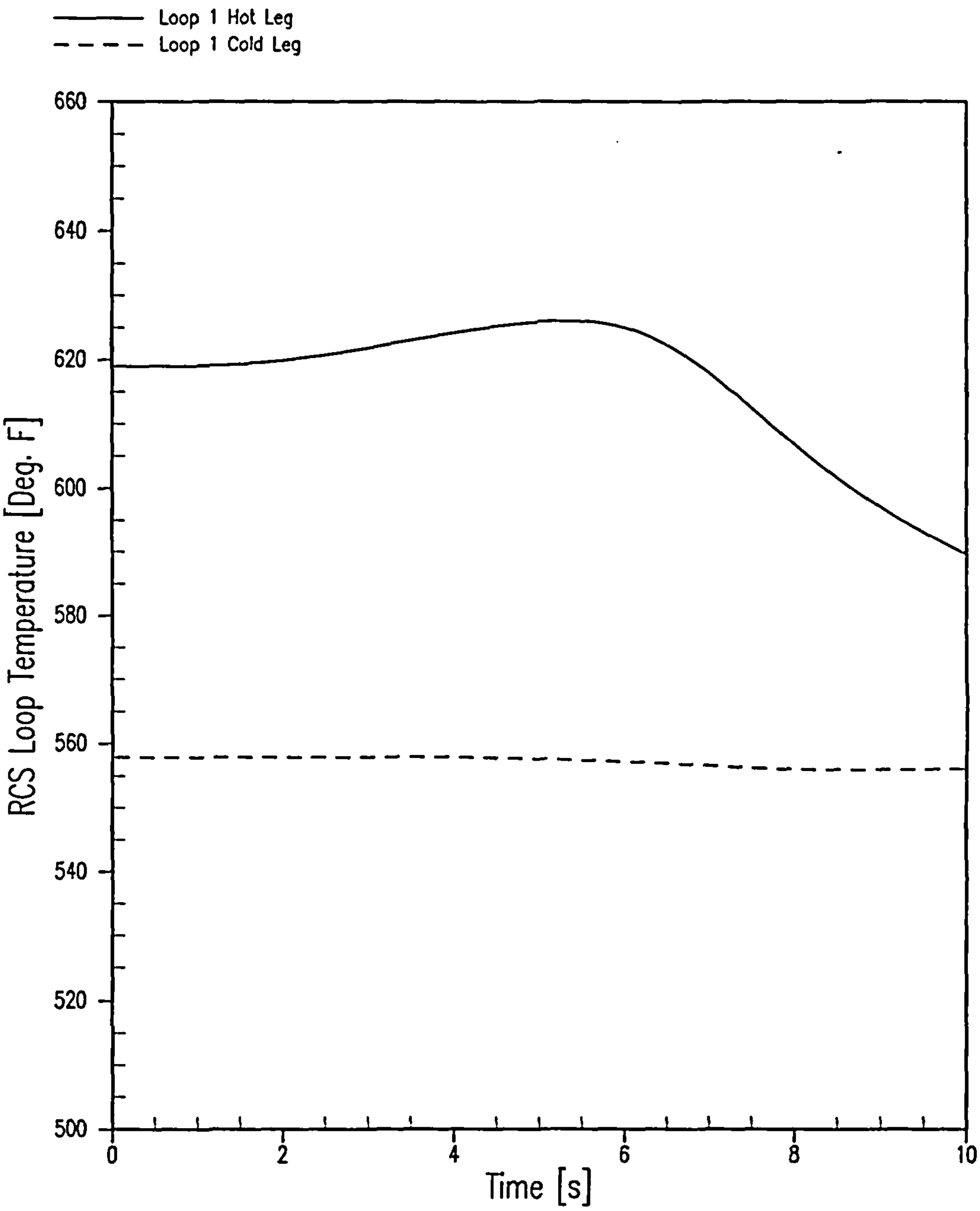


Figure 6.3.7-6 **Partial Loss of Flow, Two Pumps Coasting Down – RCS Faulted Loop Temperature versus Time**

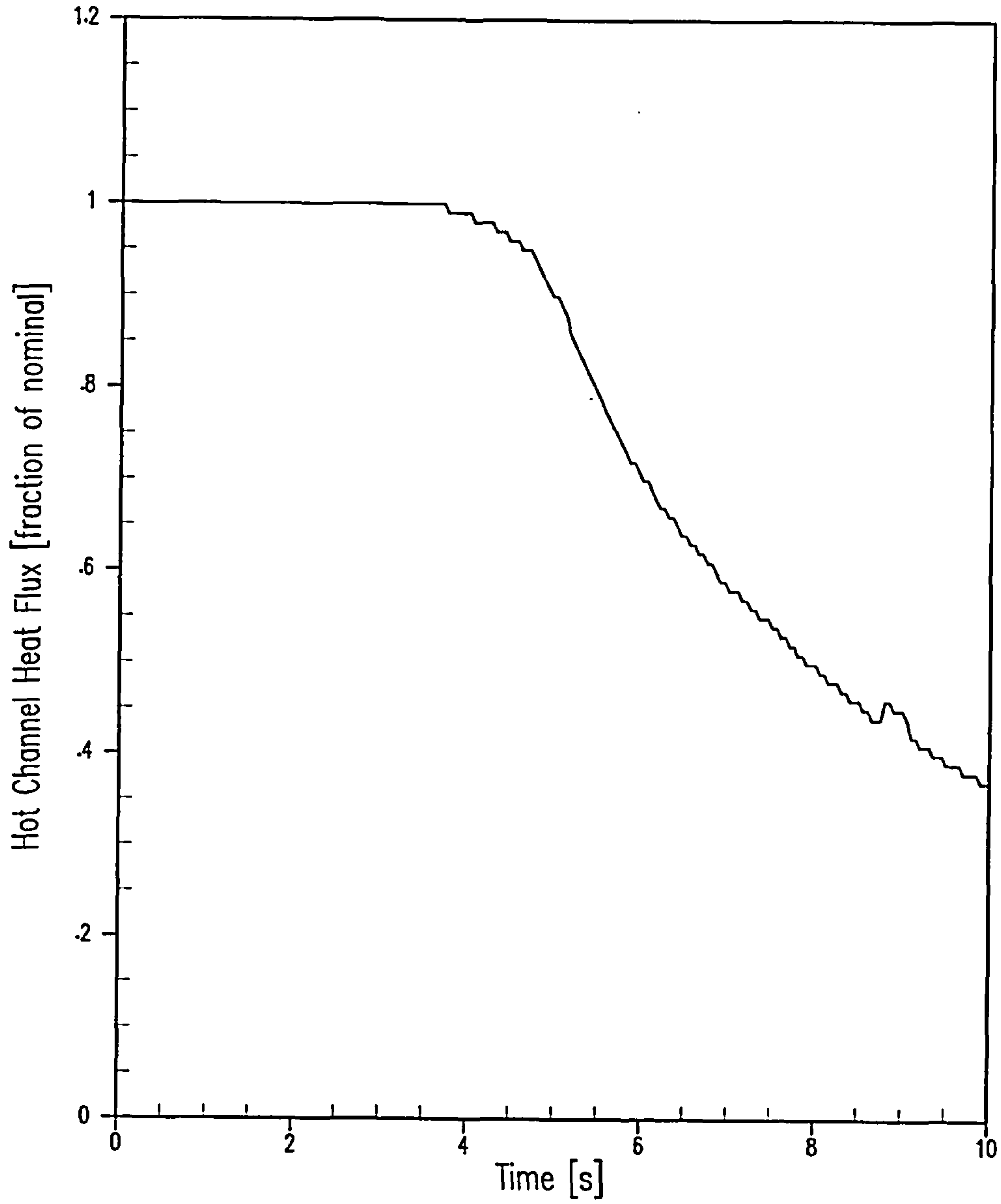


Figure 6.3.7-7 **Partial Loss of Flow, Two Pumps Coasting Down – Hot Channel Heat Flux versus Time**

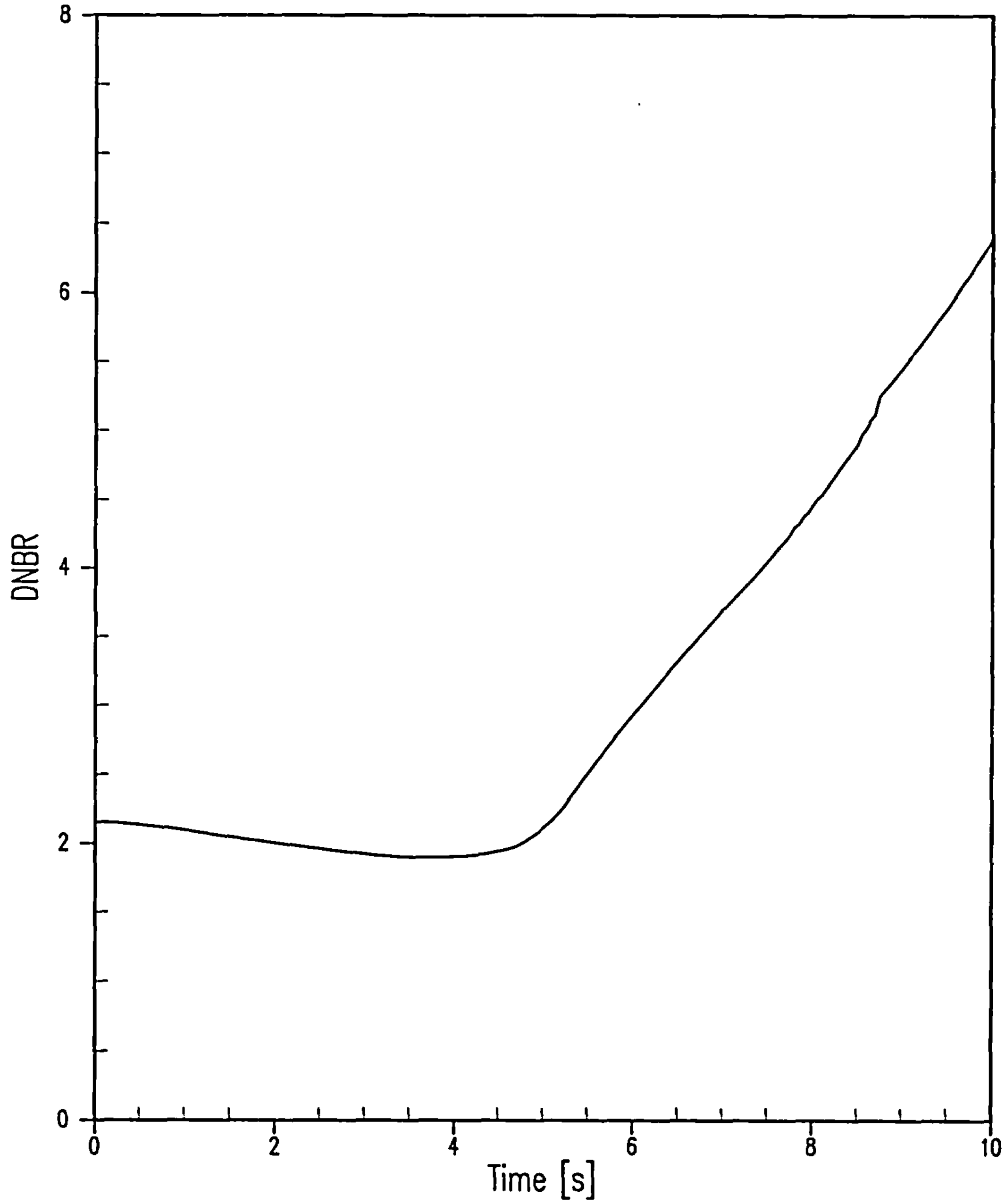


Figure 6.3.7-8 **Partial Loss of Flow, Two Pumps Coasting Down – DNBR versus Time**

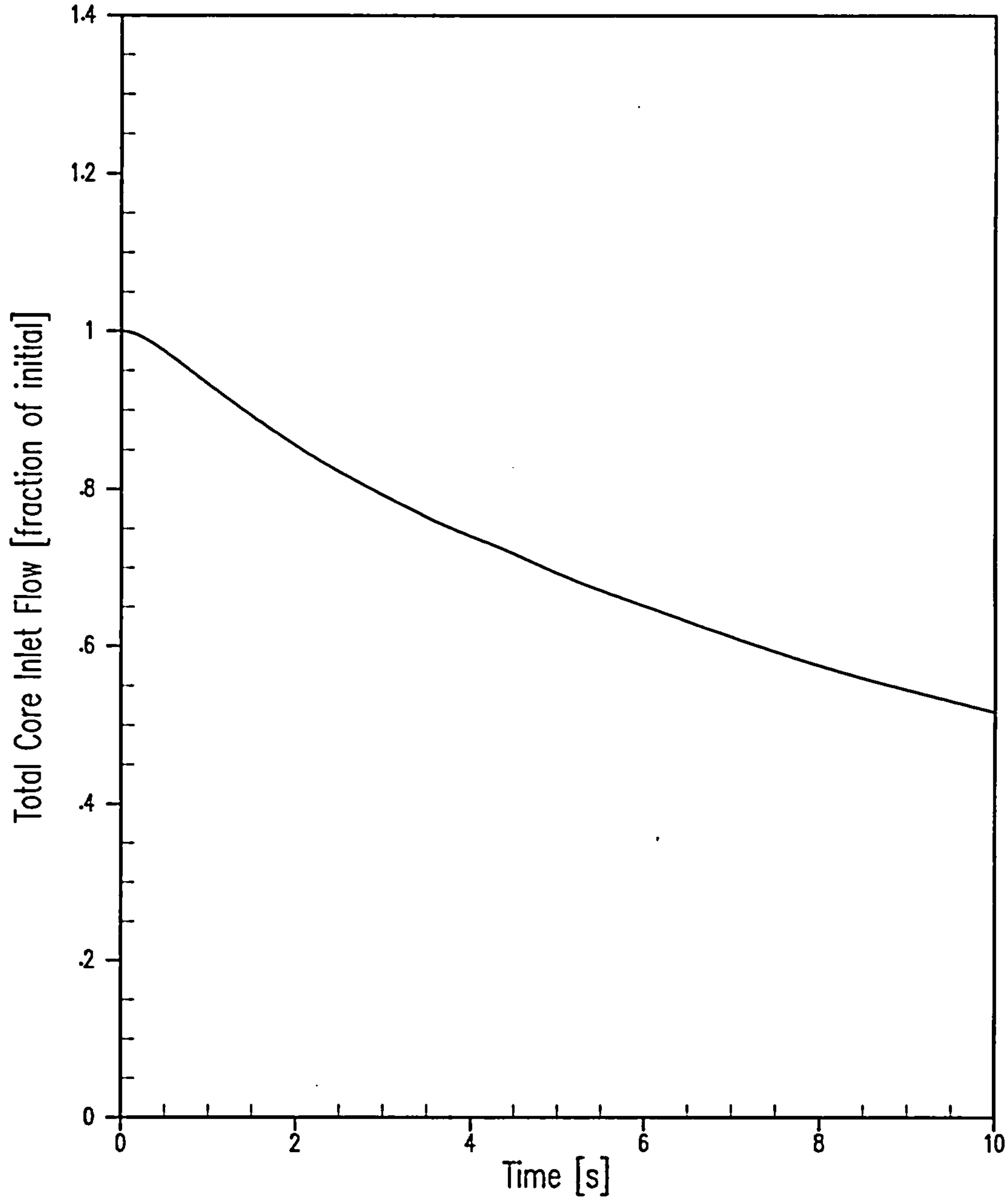


Figure 6.3.7-9 Complete Loss of Flow – Four Pumps Coasting Down – Total Core Inlet Flow versus Time

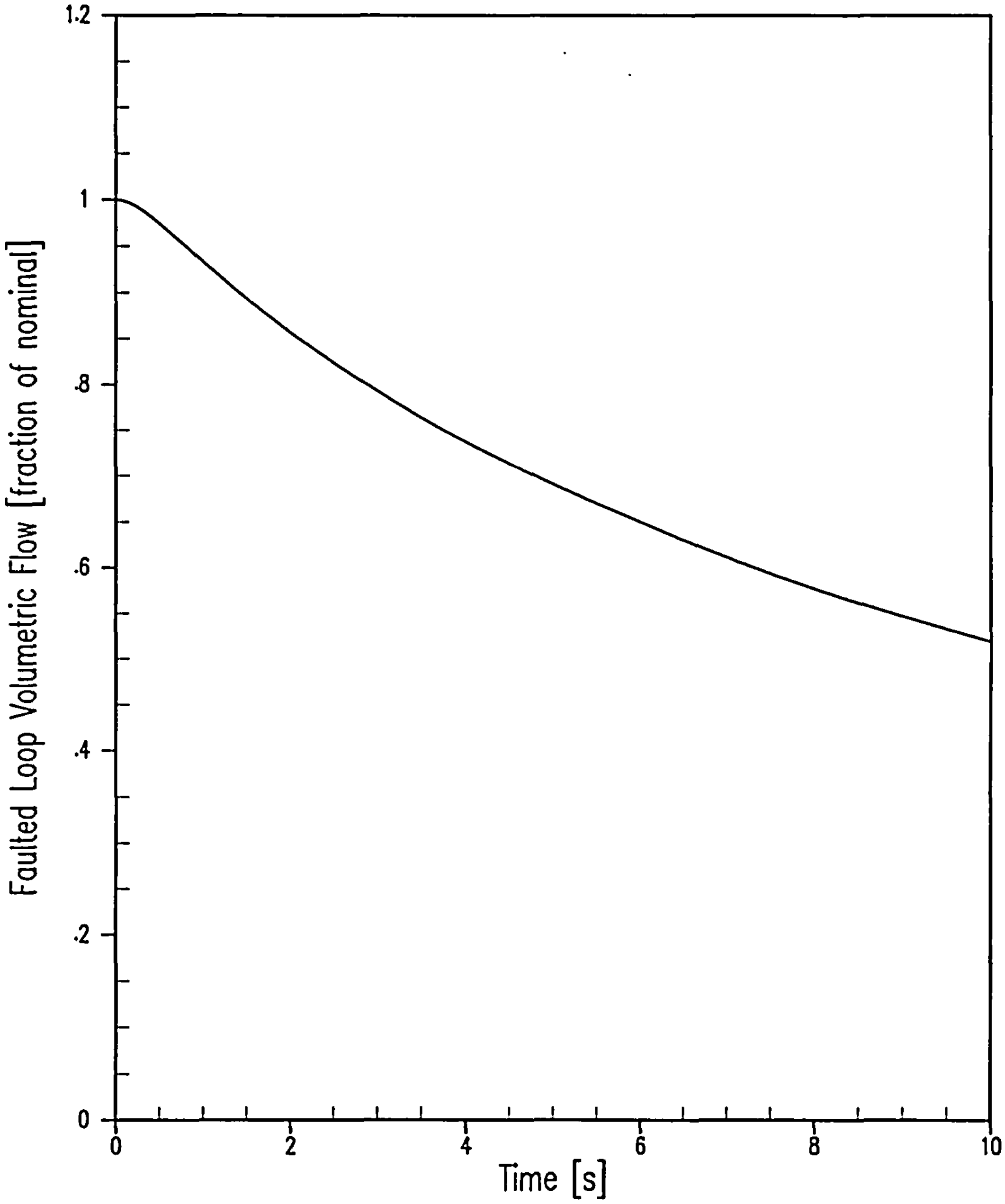


Figure 6.3.7-10 Complete Loss of Flow – Four Pumps Coasting Down – RCS Loop Flow versus Time

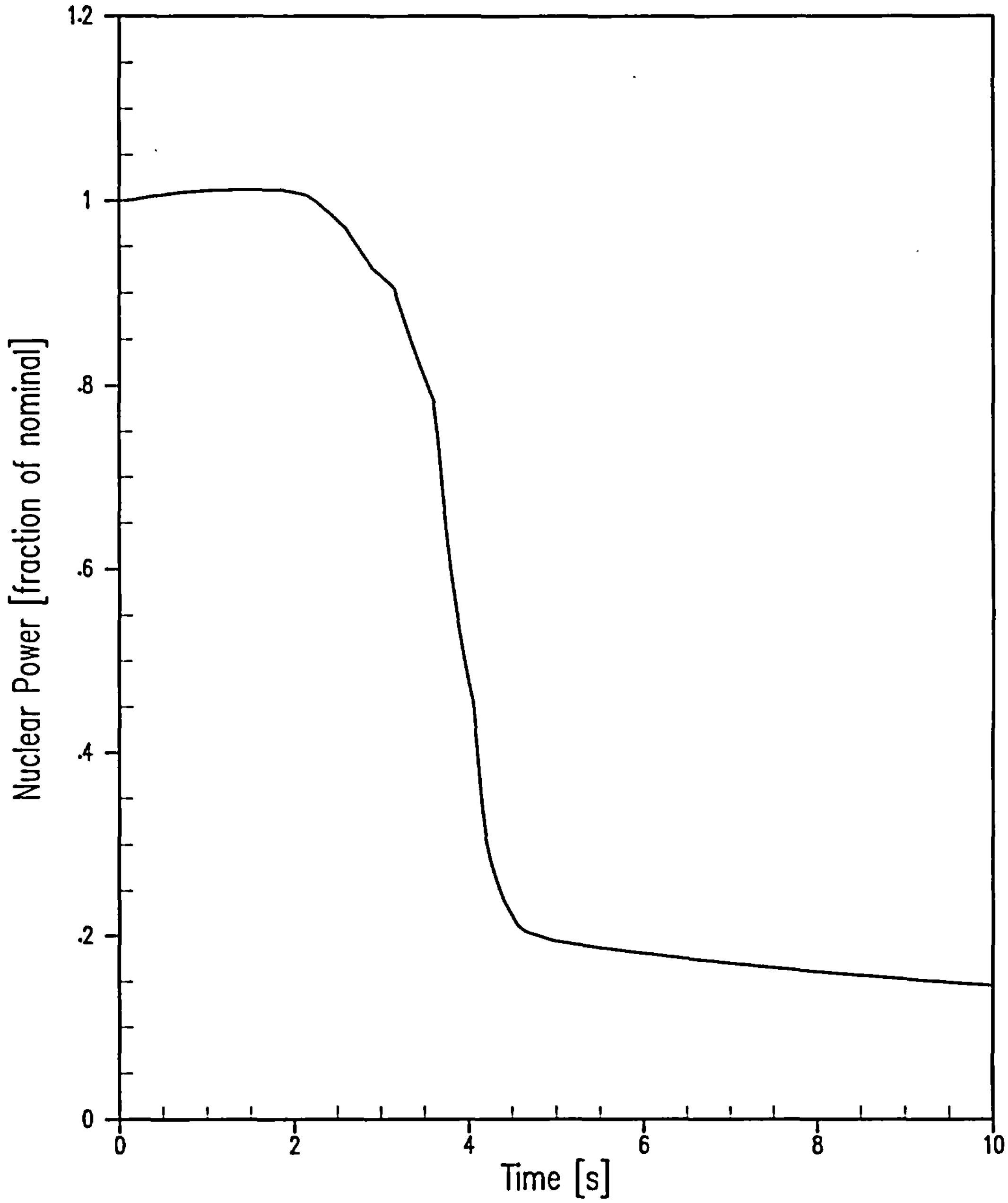


Figure 6.3.7-11 **Complete Loss of Flow – Four Pumps Coasting Down – Nuclear Power versus Time**

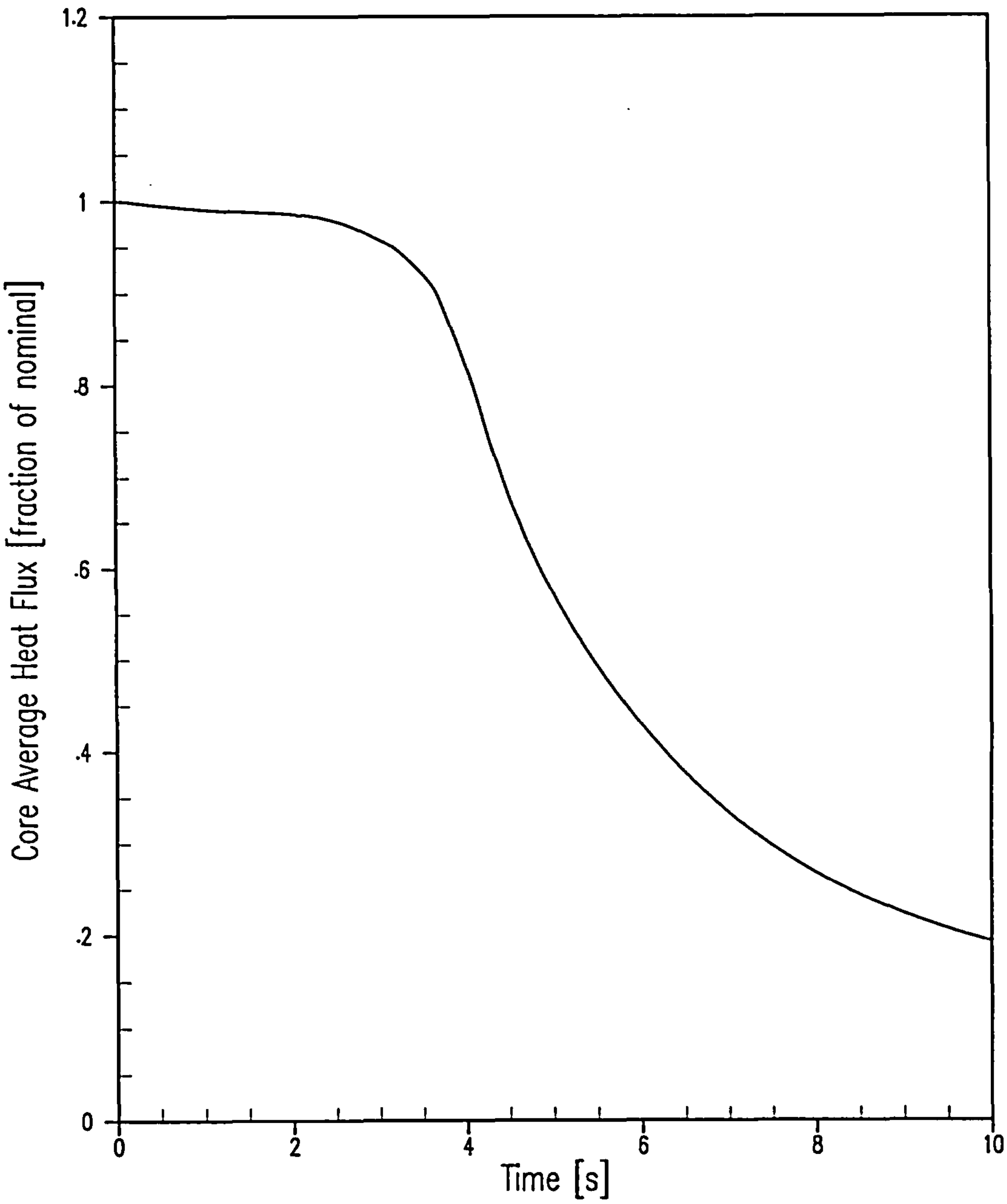


Figure 6.3.7-12 **Complete Loss of Flow – Four Pumps Coasting Down – Core Average Heat Flux versus Time**

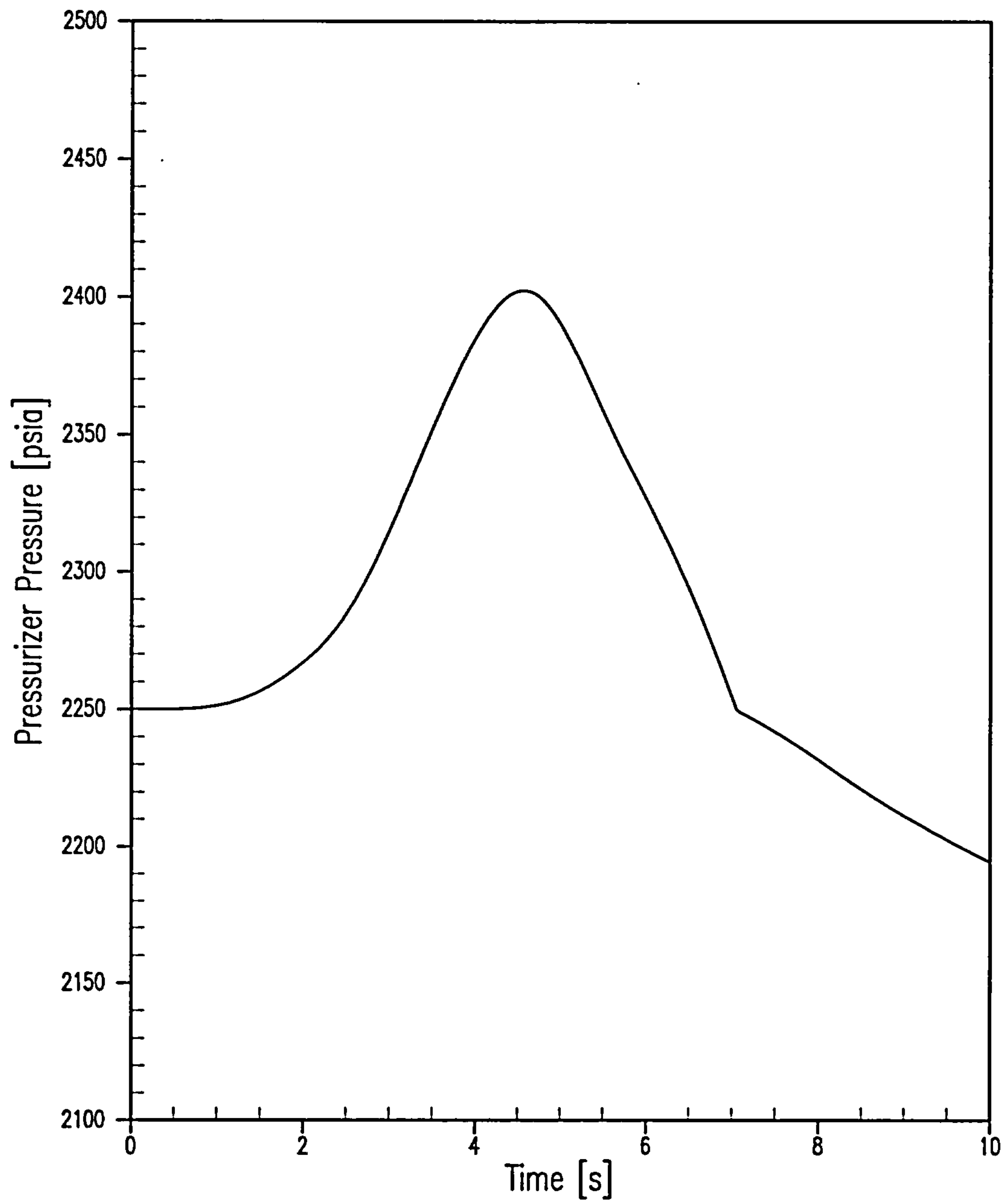


Figure 6.3.7-13 Complete Loss of Flow – Four Pumps Coasting Down – Pressurizer Pressure versus Time

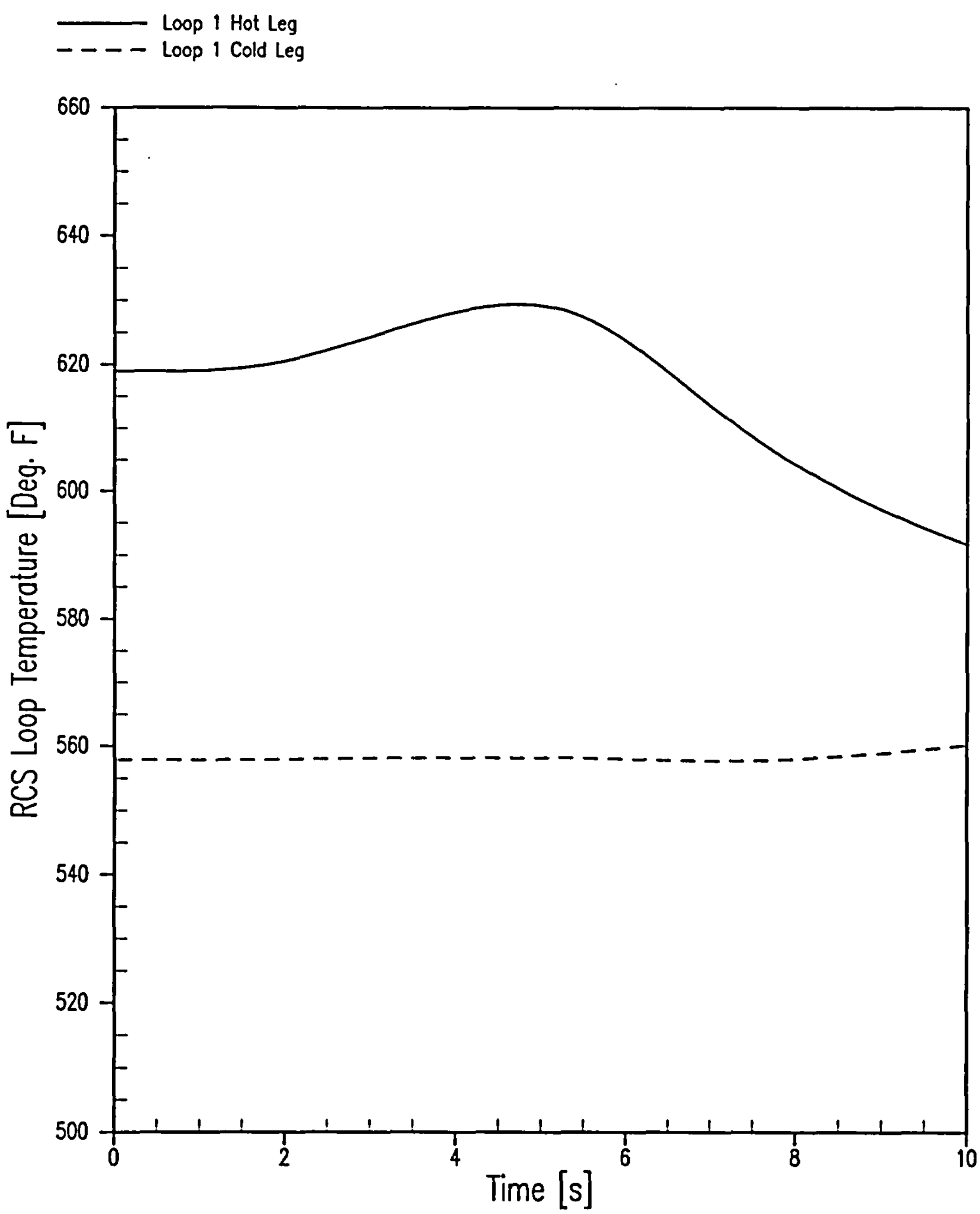


Figure 6.3.7-14 Complete Loss of Flow – Four Pumps Coasting Down – RCS Faulted Loop Temperature versus Time

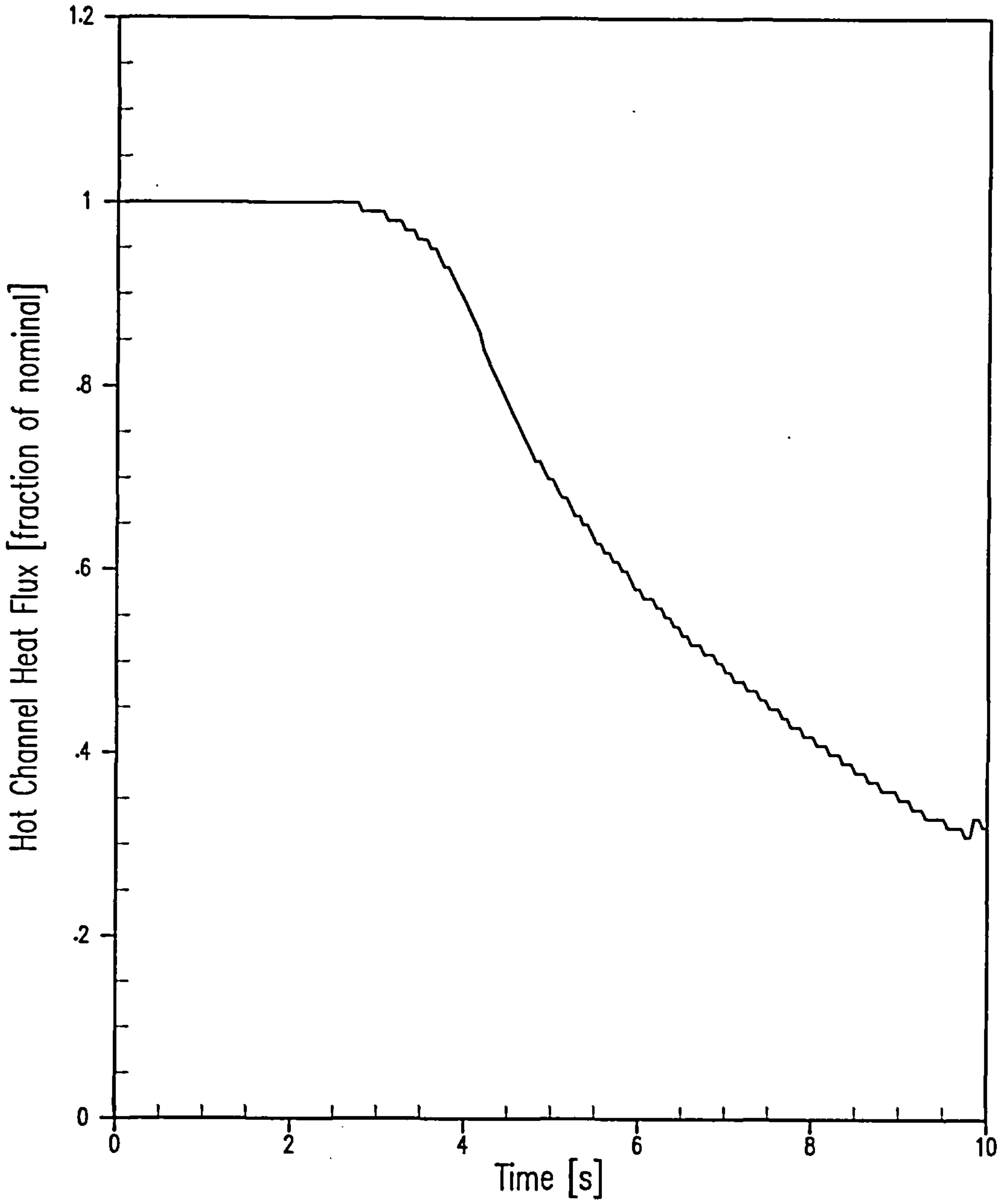


Figure 6.3.7-15 Complete Loss of Flow, Four Pumps Coasting Down – Hot Channel Heat Flux versus Time

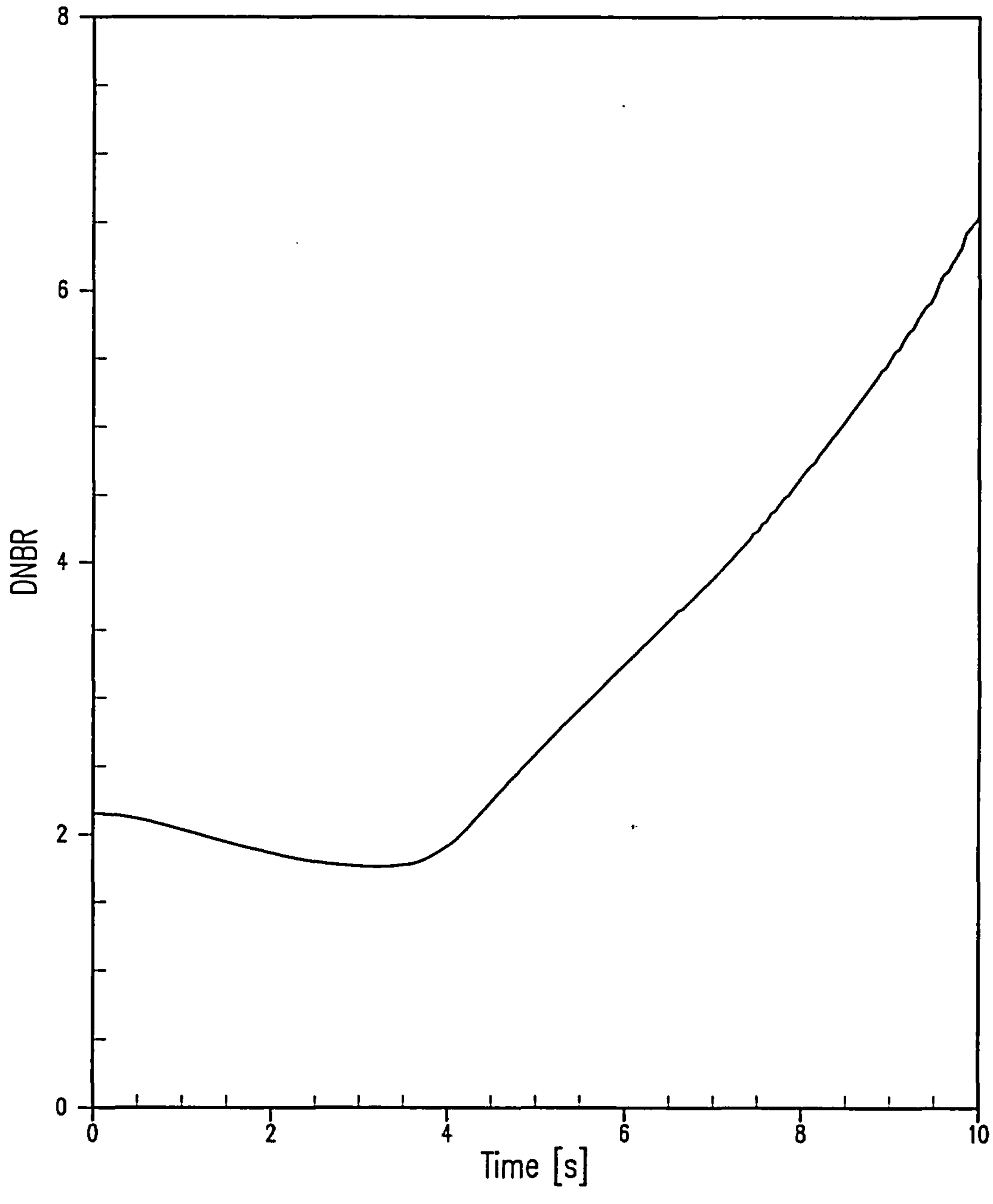


Figure 6.3.7-16 **Complete Loss of Flow, Four Pumps Coasting Down – DNBR versus Time**

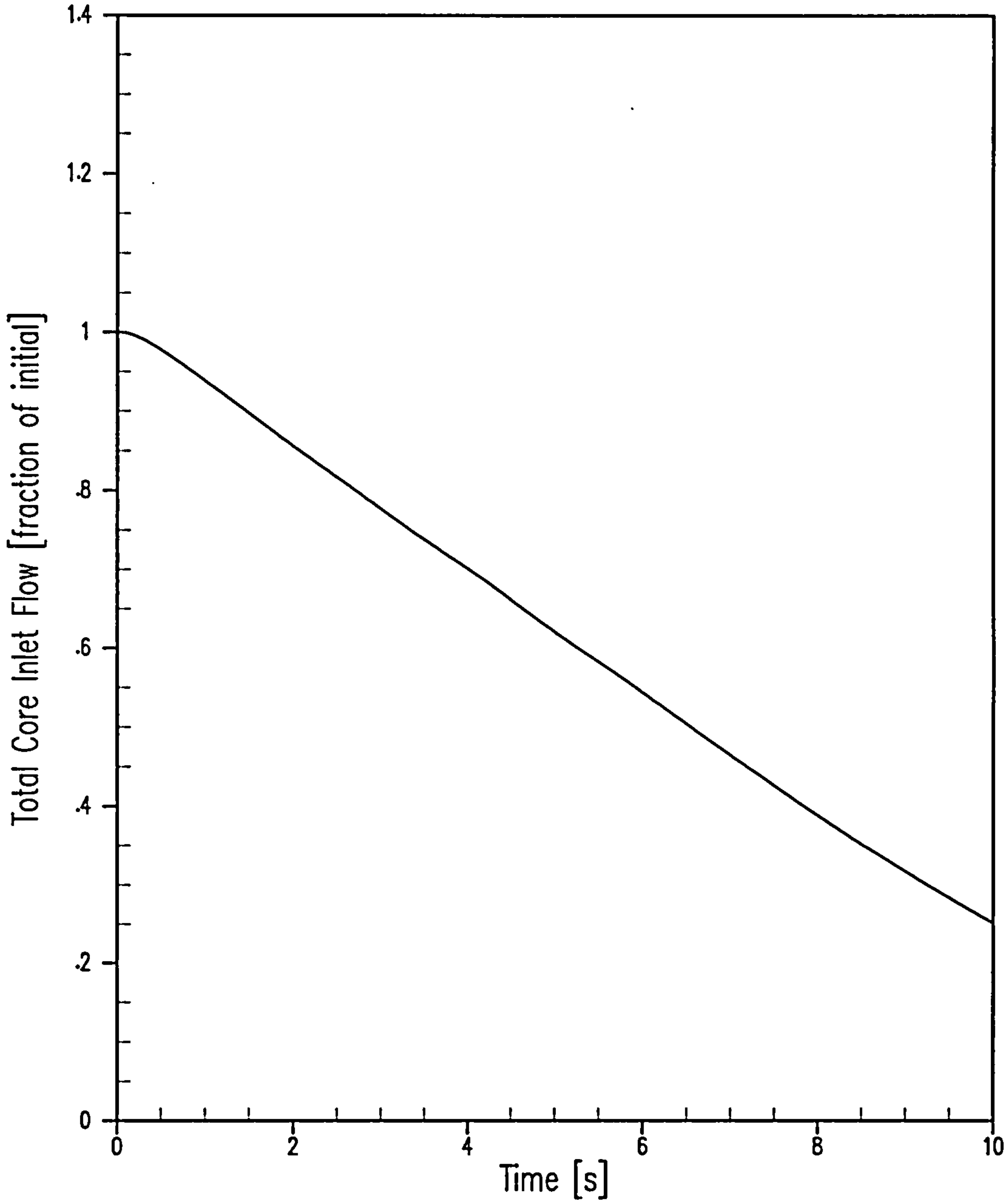


Figure 6.3.7-17 Complete Loss of Flow – Frequency Decay in Four Pumps – Total Core Inlet Flow versus Time

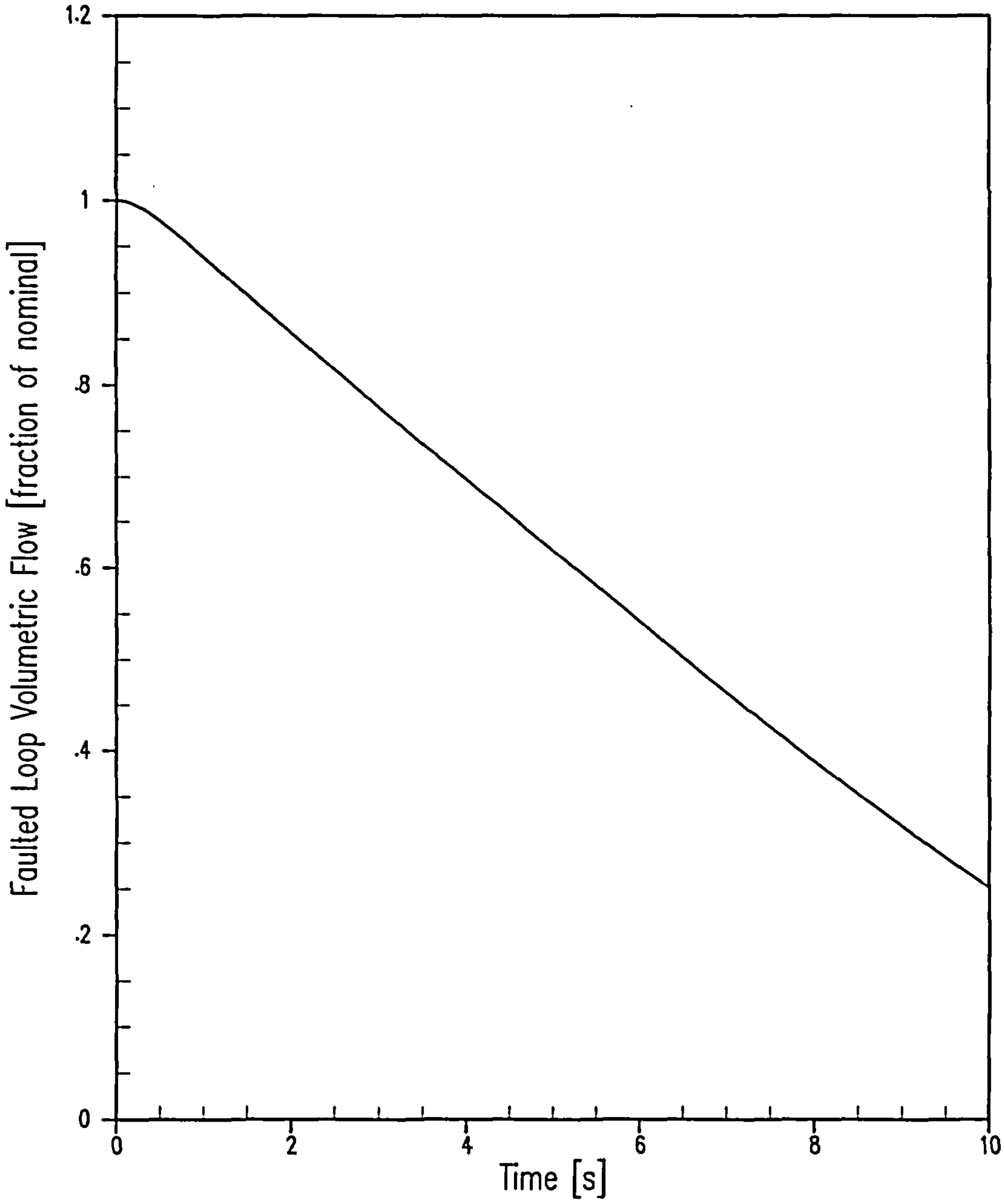


Figure 6.3.7-18 Complete Loss of Flow – Frequency Decay in Four Pumps – RCS Loop Flow versus Time

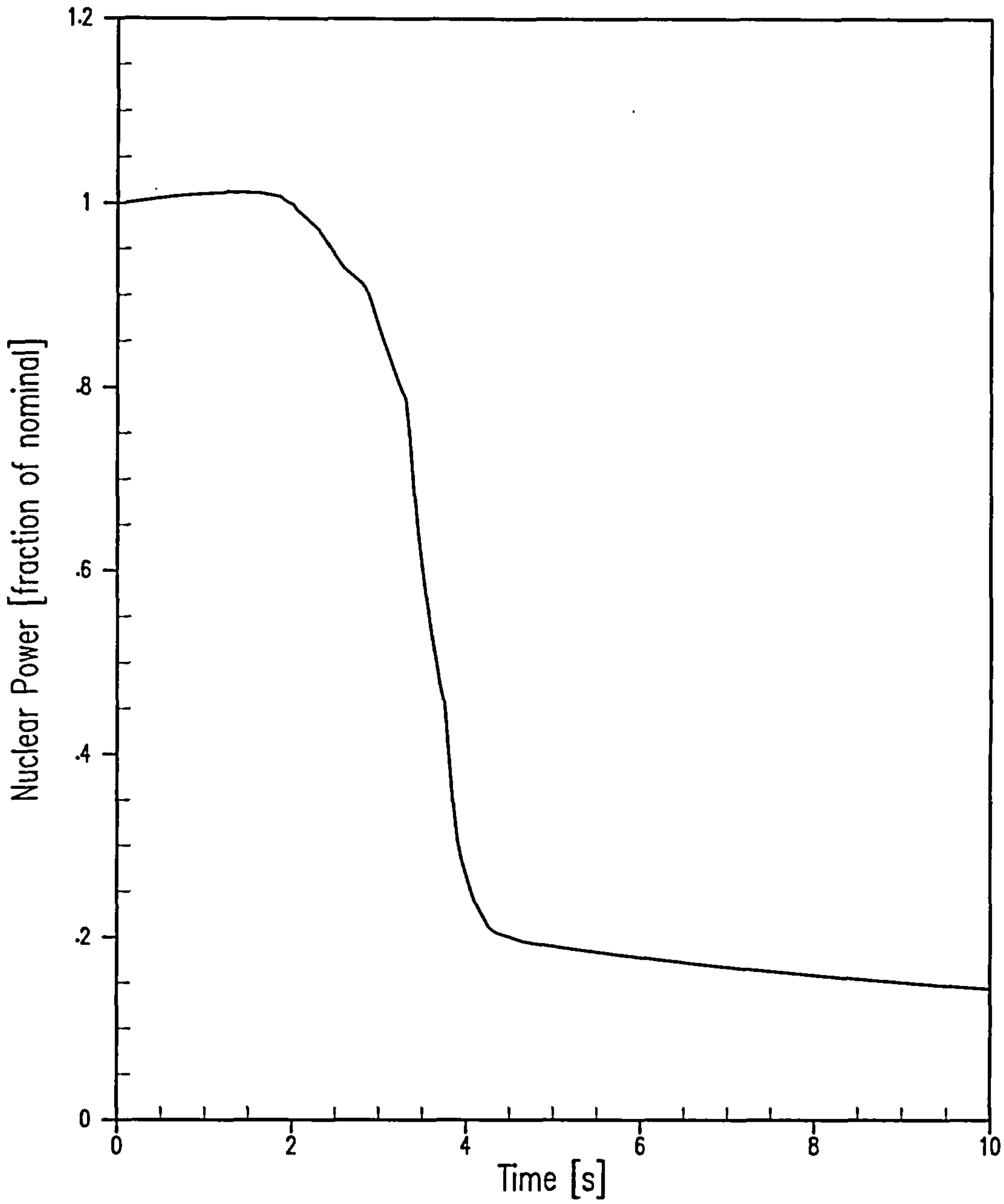


Figure 6.3.7-19 Complete Loss of Flow – Frequency Decay in Four Pumps – Nuclear Power versus Time

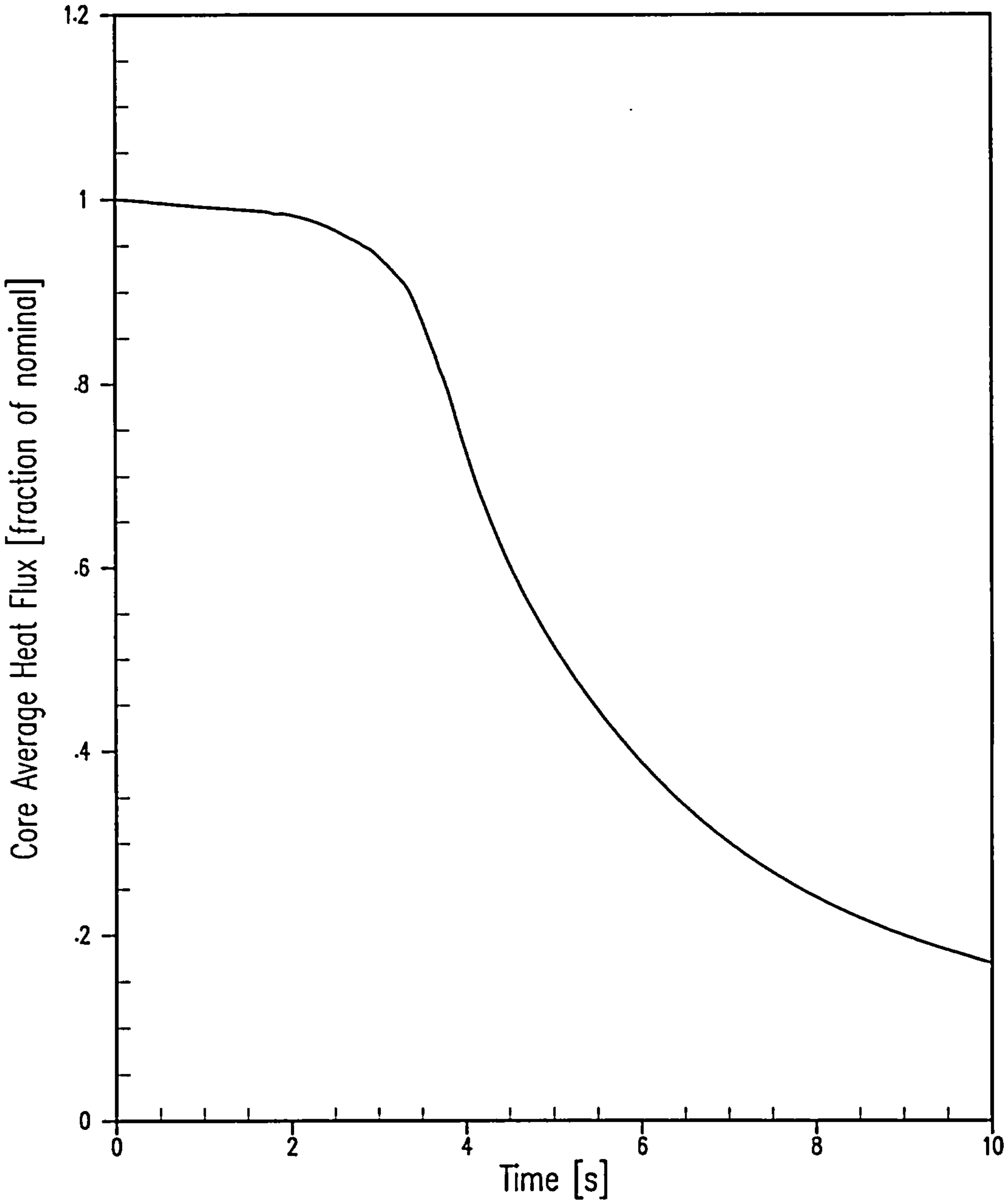


Figure 6.3.7-20 **Complete Loss of Flow – Frequency Decay in Four Pumps – Core Average Heat Flux versus Time**

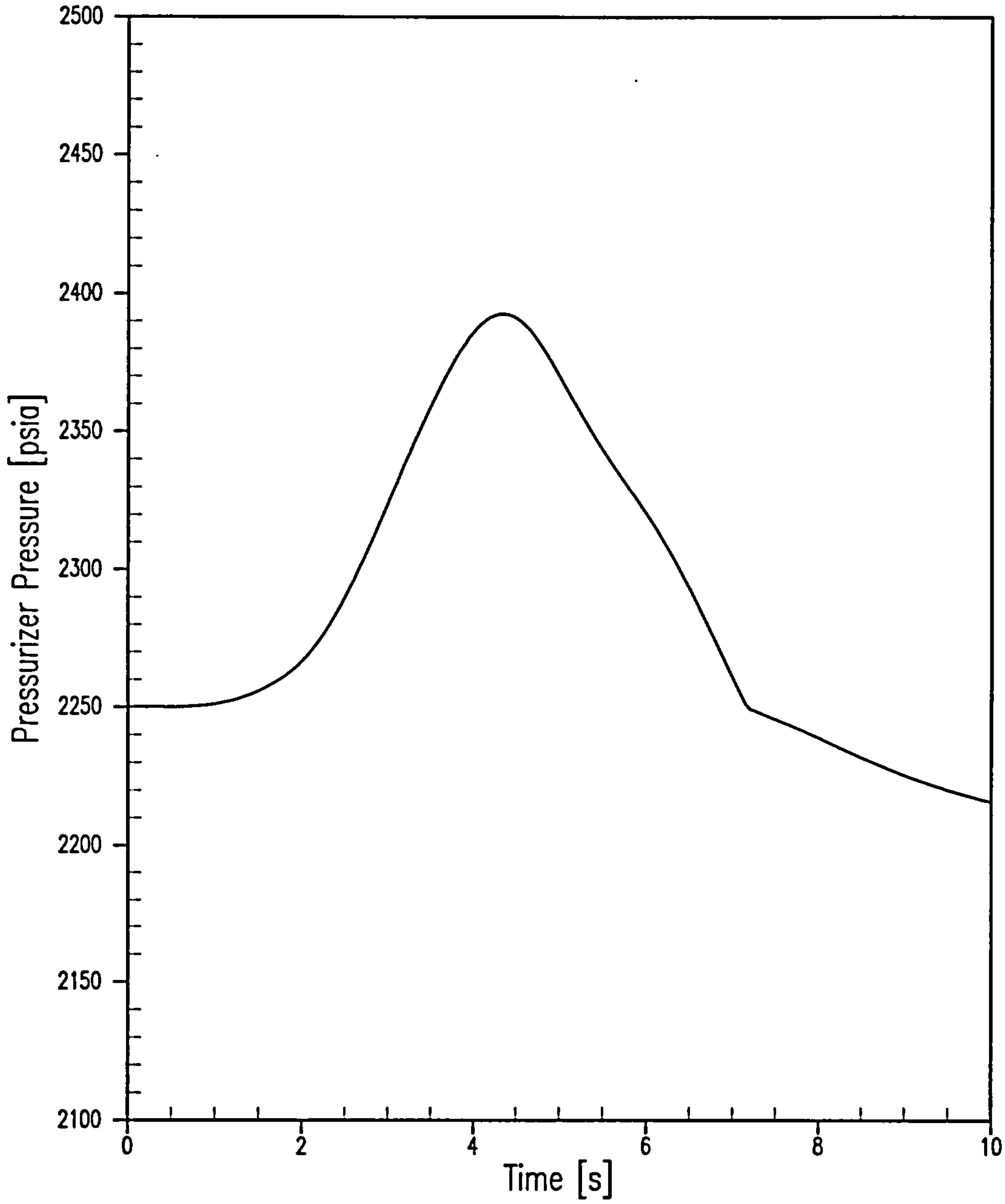


Figure 6.3.7-21 Complete Loss of Flow – Frequency Decay in Four Pumps – Pressurizer Pressure versus Time

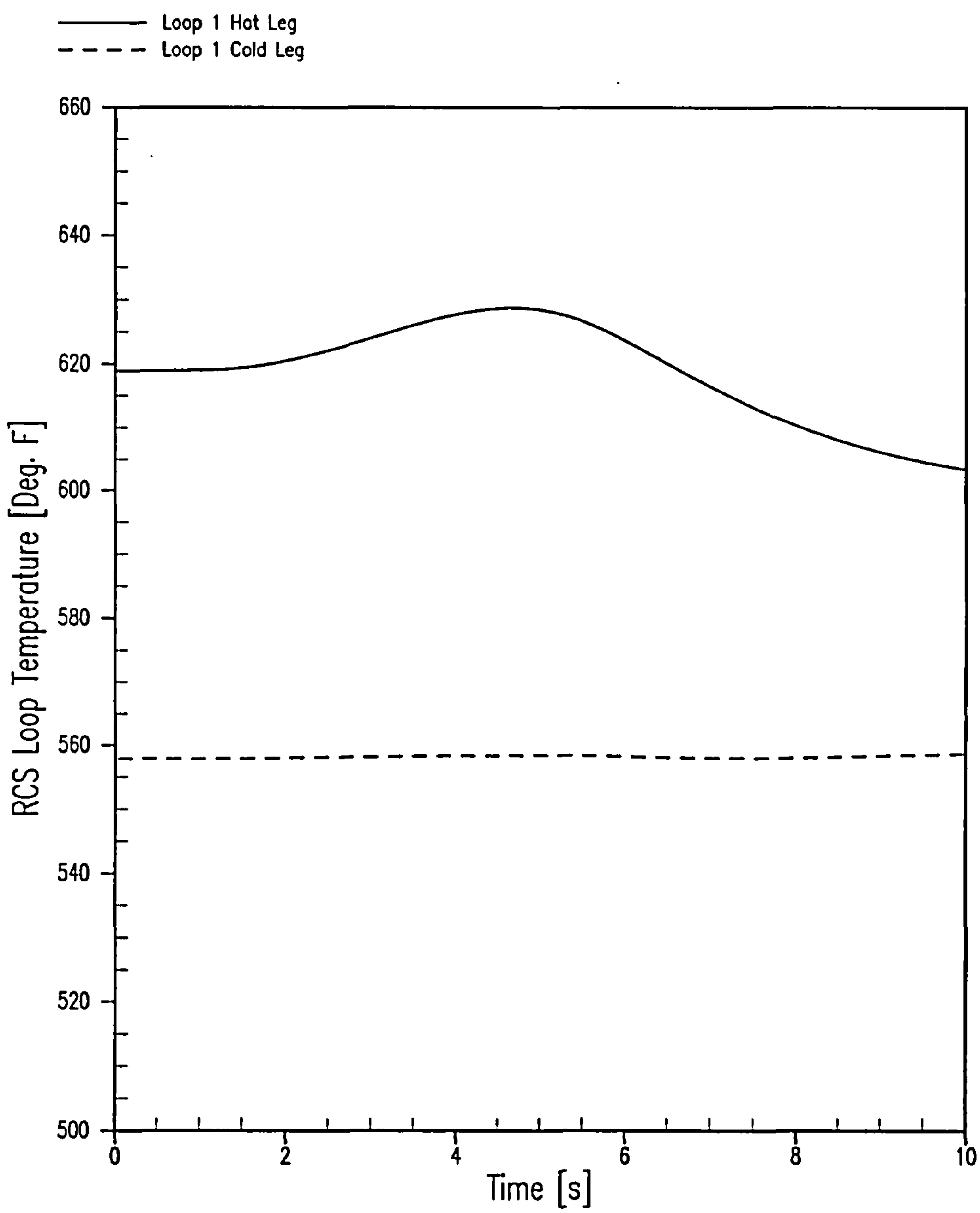


Figure 6.3.7-22 **Complete Loss of Flow – Frequency Decay in Four Pumps – RCS Loop Temperature versus Time**

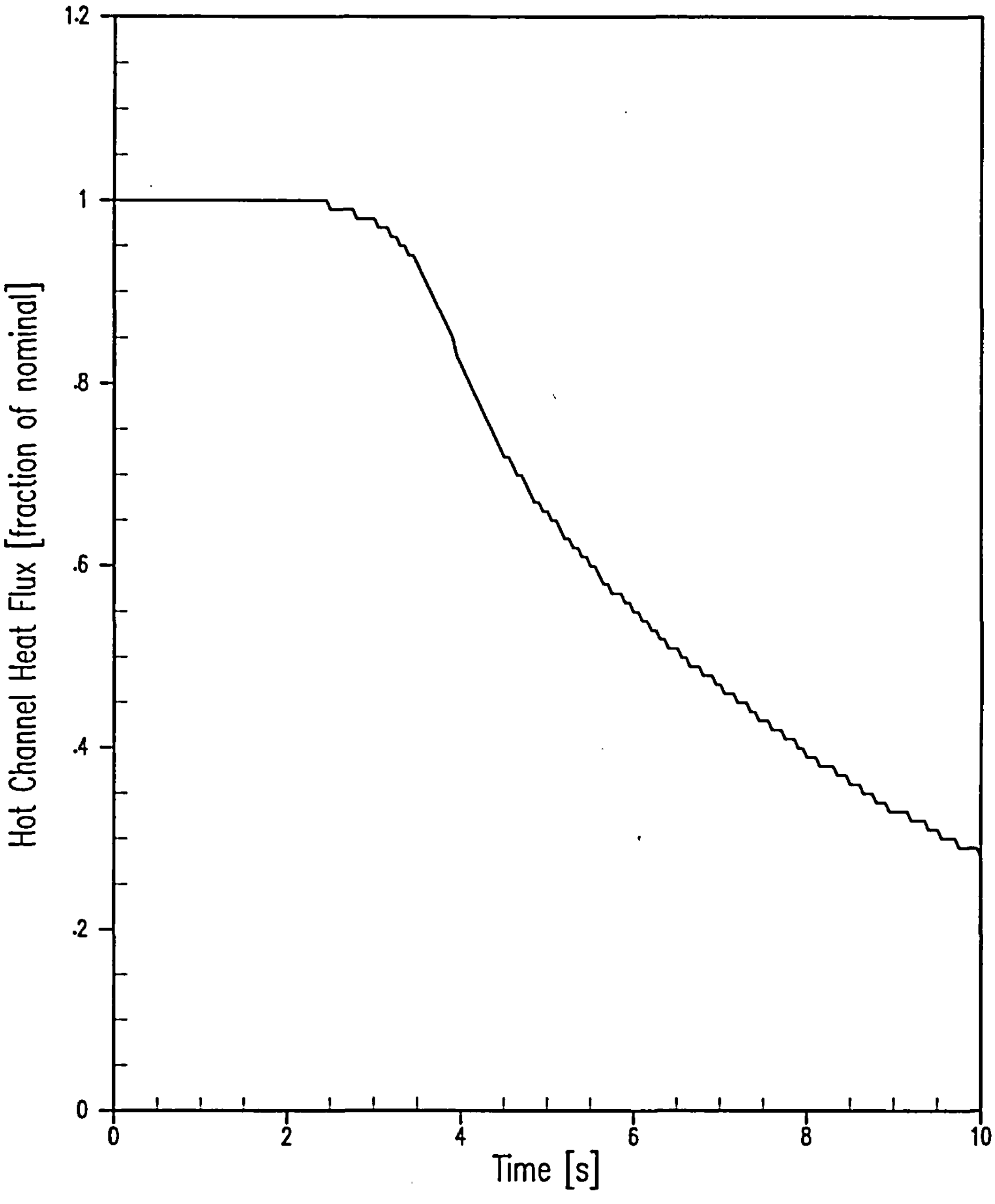


Figure 6.3.7-23 **Complete Loss of Flow – Frequency Decay in Four Pumps – Hot Channel Heat Flux versus Time**

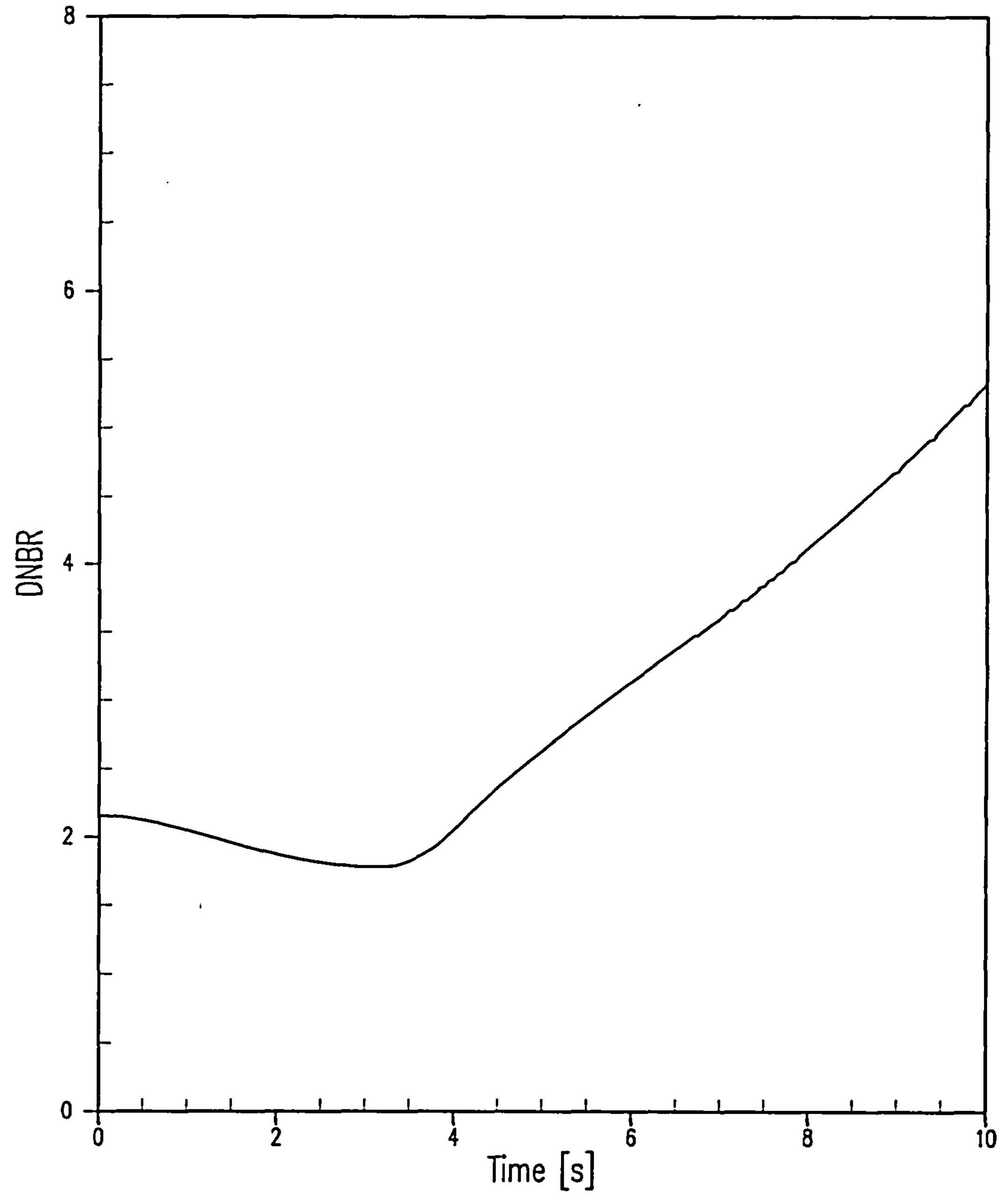


Figure 6.3.7-24 Complete Loss of Flow – Frequency Decay in Four Pumps – DNBR versus Time

6.3.8 Reactor Coolant Pump Shaft Seizure (Locked Rotor) / Reactor Coolant Pump Shaft Break (FSAR Sections 15.3.3 and 15.3.4)

6.3.8.1 Accident Description

The postulated locked rotor accident is an instantaneous seizure of a reactor coolant pump (RCP) rotor. Flow through the affected reactor coolant loop is rapidly reduced, leading to an initiation of a reactor trip on a low-flow signal. The consequences of a postulated pump shaft break accident are similar to the locked rotor event. With a broken shaft, the impeller is free to spin, as opposed to it being fixed in position during the locked rotor event. Therefore, the initial rate of reduction in core flow is greater during a locked rotor event than in a pump shaft break event because the fixed shaft causes greater resistance than a free-spinning impeller early in the transient, when flow through the affected loop is in the positive direction. As the transient continues, the flow direction through the affected loop is reversed. If the impeller is able to spin free, the flow to the core will be less than that available with a fixed shaft during periods of reverse flow in the affected loop. Because peak pressure, cladding temperature, and departure from nucleate boiling (DNB) occur very early in the transient, the reduction in core flow during the period of forward flow in the affected loop dominates the severity of the results. Consequently, the results are bounding for the locked rotor and RCP shaft break transients.

After the locked rotor, reactor trip is initiated on a reactor coolant loop low flow signal. At the time of reactor trip, the unaffected RCPs are assumed to lose power and coast down freely for cases where a loss of offsite power is assumed.

Following initiation of 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 generators is reduced. This is because, first, the reduced flow results in a decreased tube-side film coefficient; and then because the reactor coolant in the tubes cools down while the shell-side temperature increases (turbine steam flow is reduced to zero upon plant trip). The rapid expansion of the coolant in the reactor core, combined with reduced heat transfer in the steam generators, causes a coolant surge into the pressurizer and a pressure increase throughout the RCS. This coolant surge into the pressurizer compresses the steam volume, actuates the automatic spray system, opens the power-operated relief valves (PORVs), and opens the pressurizer safety valves, in that sequence. The PORVs are designed for reliable operation and would be expected to function properly during the accident. However, for conservatism in the peak-pressure evaluation, their pressure-reducing effect and the pressure-reducing effect of the pressurizer sprays are not included in the analysis.

The locked rotor event is analyzed to the following criteria:

- Pressure in the RCS should be maintained below the designated limit (see below).
- Coolable core geometry is ensured by showing that the peak cladding temperature and maximum oxidation level for the hot spot are below 2,700°F and 16 percent by weight, respectively.
- Activity release is such that the calculated doses meet 10 CFR Part 100 guidelines.

For Callaway, the locked rotor RCS pressure limit is equal to 110 percent of the design value, or 2,748.5 psia. For the secondary side, the locked rotor pressure limit is also assumed to be equal to 110 percent of design pressure, or 1,318.5 psia. Since the loss of load analysis bounds the locked rotor event, a specific MSS overpressurization analysis is not performed.

A hot-spot evaluation is performed to calculate the peak cladding temperature and maximum oxidation level. Finally, a calculation of the "rods-in-DNB" is performed for input to the radiological dose analysis.

6.3.8.2 Method of Analysis

The locked rotor transient is analyzed with two computer codes. First, the RETRAN computer code (Reference 1) is used to calculate the loop and core flow during the transient, the time of reactor trip based on the calculated flows, the nuclear power transient, and the primary-system pressure and temperature transients. The VIPRE computer code (Reference 5) is then used to calculate the thermal behavior of the fuel located at the core hot spot including the rods-in-DNB using the nuclear power and RCS temperature (enthalpy), pressure, and flow from RETRAN. The RETRAN and VIPRE computer codes are discussed in detail in Section 6.3.0.6 of this report.

For the case analyzed to determine the maximum RCS pressure and peak cladding temperature, the plant is assumed to be in operation under the most adverse steady-state operating conditions, that is, a maximum steady-state thermal power, maximum steady-state pressure, and maximum steady-state coolant average temperature thermal design flow is assumed. The case analyzed to determine the rods-in-DNB utilizes the Revised Thermal Design Procedure (RTDP) methodology (Reference 2). Initial reactor power, pressurizer pressure and RCS temperature are assumed to be at their nominal values. Minimum measured flow is also assumed. Both the maximum RCS pressure/peak cladding temperature and the rods-in-DNB cases are analyzed twice, once with continuous operation of the intact RCPs, and once with a loss of power to the intact RCPs. The RCPs are conservatively assumed to lose power simultaneously at the time of reactor trip in the cases where power is lost.

A maximum, uniform, steam generator tube plugging (SGTP) level of 5 percent was assumed in the RETRAN analysis.

A conservatively large absolute value of the Doppler-only power coefficient is used, along with the most positive moderator temperature coefficient (MTC) limit for full-power operation (0 pcm/°F). These assumptions maximize the core power during the initial part of the transient when the peak RCS pressures and hot spot results are reached.

A conservatively low trip reactivity value (4.0-percent $\Delta\rho$) is used to minimize the effect of rod insertion following reactor trip and maximize the heat flux statepoint used in the DNB ratio (DNBR) evaluation for this event. This value is based on the assumption that the highest worth rod cluster control assembly (RCCA) is stuck in its fully withdrawn position. A conservative trip reactivity worth versus rod position was modeled in addition to a conservative rod drop time (2.7 seconds from dashpot). The trip reactivity versus rod position curve is confirmed to be valid as part of the Reload Safety Analysis Checklist (RSAC) verification process.

For the peak RCS pressure evaluation, the initial pressure is conservatively estimated as 30 psi above the nominal pressure (2,250 psia) to allow 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. The peak RCS pressure occurs in the lower plenum of the vessel. The pressure transient in the lower plenum is shown in Figure 6.3.8-6.

For this accident, an evaluation of the consequences with respect to the fuel rod thermal transient is performed. The evaluation incorporates the assumption of rods going into DNB as a conservative initial condition to determine the cladding temperature and zirconium-water reaction resulting from the locked rotor. Results obtained from the 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 assumed to be 2.6 times the average rod power (that is, $F_Q = 2.6$) at the initial core power level.

Film Boiling Coefficient

The film boiling coefficient is calculated in the VIPRE code using the Bishop-Sandberg-Tong film boiling correlation. The fluid properties are evaluated at film temperature. 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 RCS flow rate as a function of time are based on the RETRAN results.

Fuel Cladding Gap Coefficient

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

Zirconium-Water Reaction

The zirconium-water reaction can become significant above 1,800°F (cladding temperature). The Baker-Just parabolic rate equation is used to define the rate of zirconium-water reaction. The effect of the zirconium-water reaction is included in the calculation of the hot spot cladding temperature transient.

6.3.8.3 Results

Figures 6.3.8-1 through 6.3.8-8 illustrate the transient response for the locked rotor event (peak RCS pressure/peak cladding temperature case). The peak RCS pressure is 2,559 psia in the lower plenum of the vessel and is less than the acceptance criterion of 2,748.5 psia. Also, the peak cladding temperature is 1790°F, which is considerably less than the limit of 2,700°F. The zirconium-water reaction at the hot spot is 0.30 percent by weight, which meets the criterion of less than 16-percent zirconium-water reaction. For the radiological dose evaluation, the total percentage of fuel rods calculated to experience DNB is less than 5 percent (rods-in-DNB case). The sequence of events for the peak RCS pressure/peak cladding temperature case is given in Table 6.3.8-1. This transient trips on a low primary reactor coolant loop flow trip setpoint, which is assumed to be 87.0 percent of initial.

6.3.8.4 Conclusions

The analysis performed demonstrates that for the locked rotor event, the RCS pressure remains below 110 percent of the design pressure and the hot spot cladding temperature and oxidation levels remain below the limit values. Therefore, all applicable acceptance criteria are met. In addition, the total percentage of rods calculated to experience DNB is less than 5 percent.

Table 6.3.8-1 Time Sequence of Events for Reactor Coolant Pump Locked Rotor

		Time (Seconds)	
Event		**With Power**	**Without Power**
Rotor on One Pump Locks		0.0	0.0
Low Flow Reactor Trip Setpoint Reached		0.05	0.05
Rods Begin to Drop		1.05	1.05
Loss-of-Offsite-Power (Remaining Active Pumps Begin to Coast Down)		----	1.05
Maximum RCS Pressure Occurs		3.40	5.10
Maximum Cladding Temperature Occurs		3.45	3.65
Results	**Limit**		
Peak RCS Pressure (psia)	2,748.5	2,498.3	2,559.5
Peak Cladding Temperature (°F)	2,700	1730	1790
Peak Zirconium-Water Reaction (%)	16	0.20	0.30
Rods-in-DNB (%)	5	0	<5

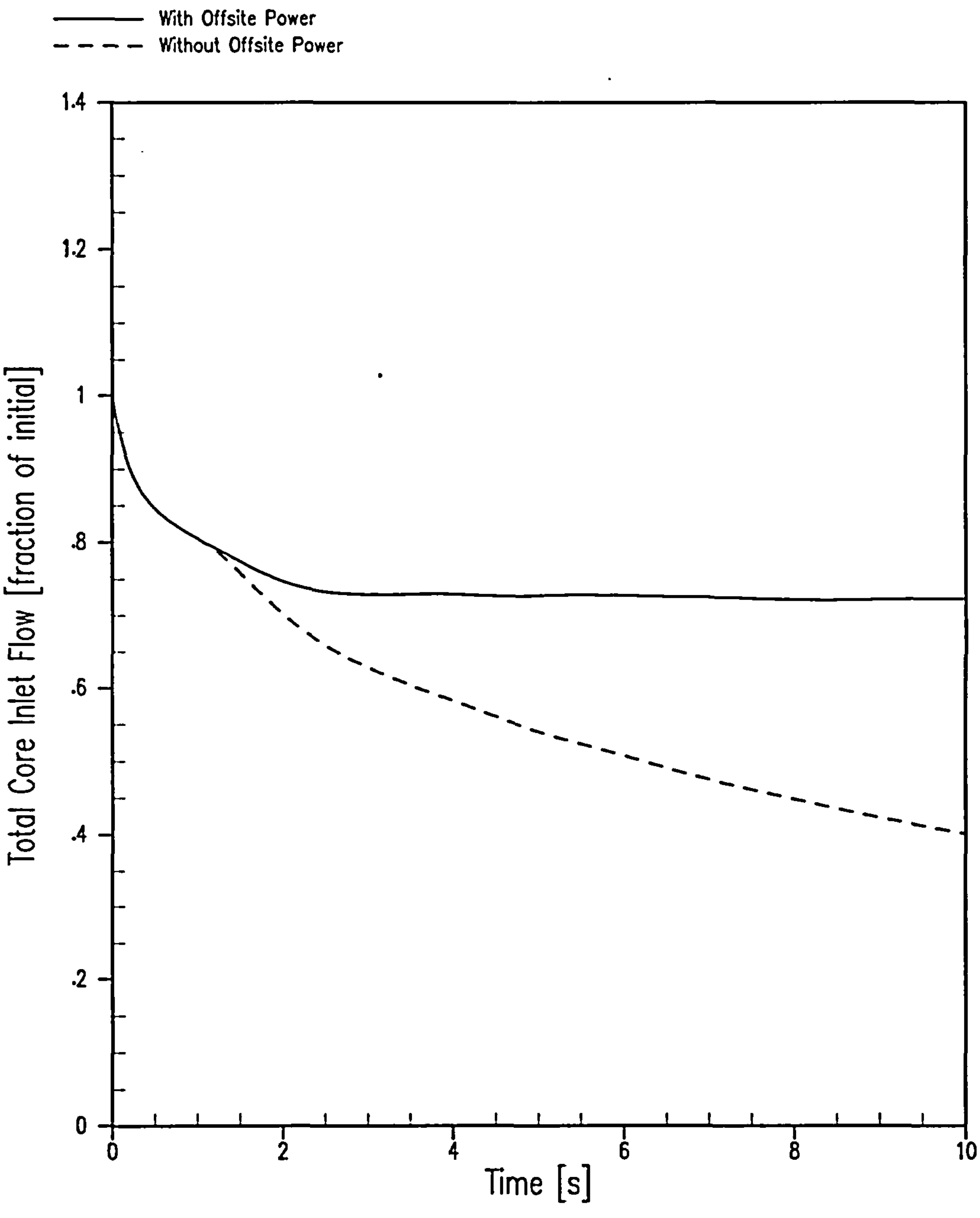


Figure 6.3.8-1 Locked Rotor/Shaft Break, RCS Pressure/Peak Cladding Temperature Case – Total Core Inlet Flow versus Time

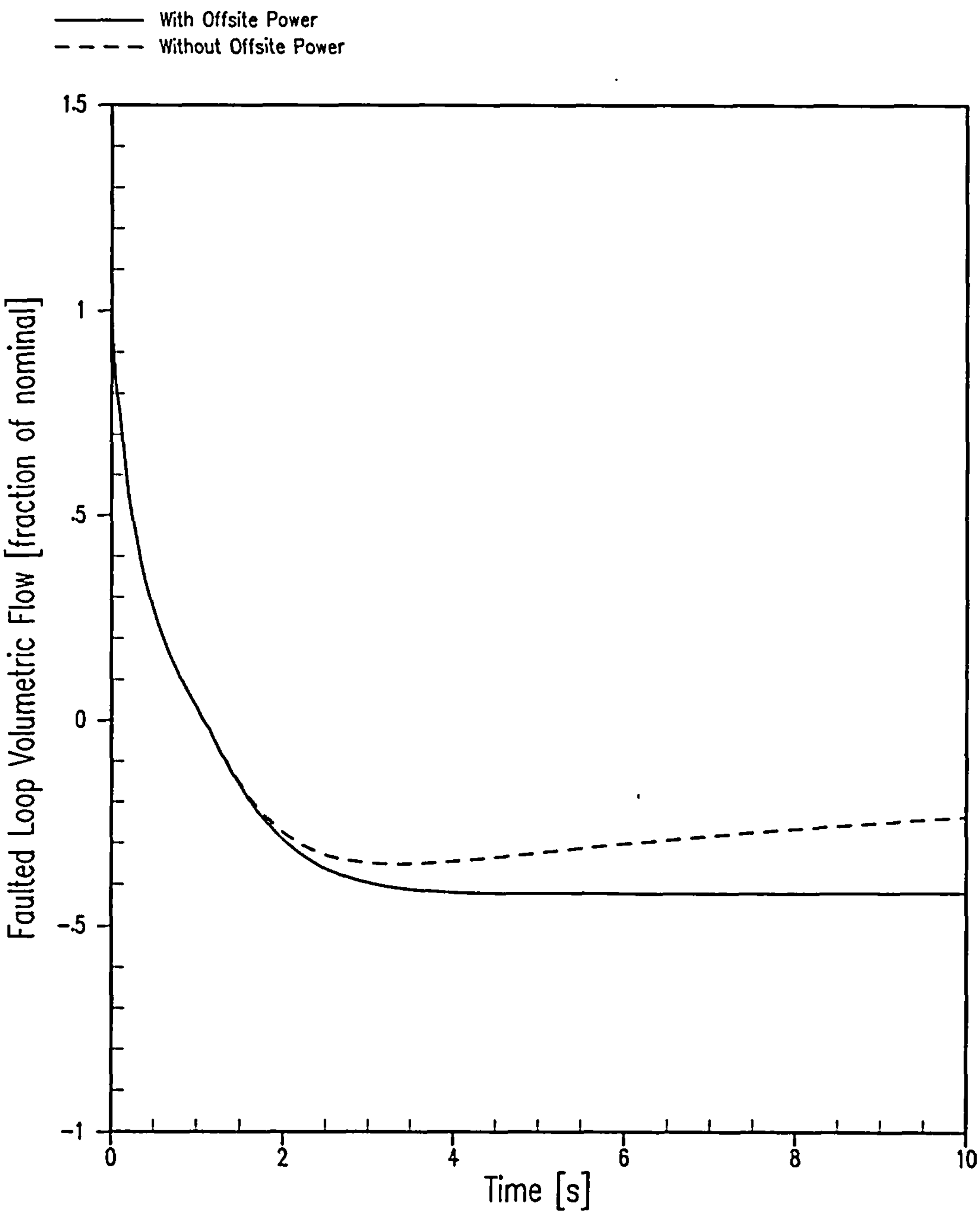


Figure 6.3.8-2 **Locked Rotor/Shaft Break, RCS Pressure/Peak Cladding Temperature Case – RCS Loop Flow versus Time**

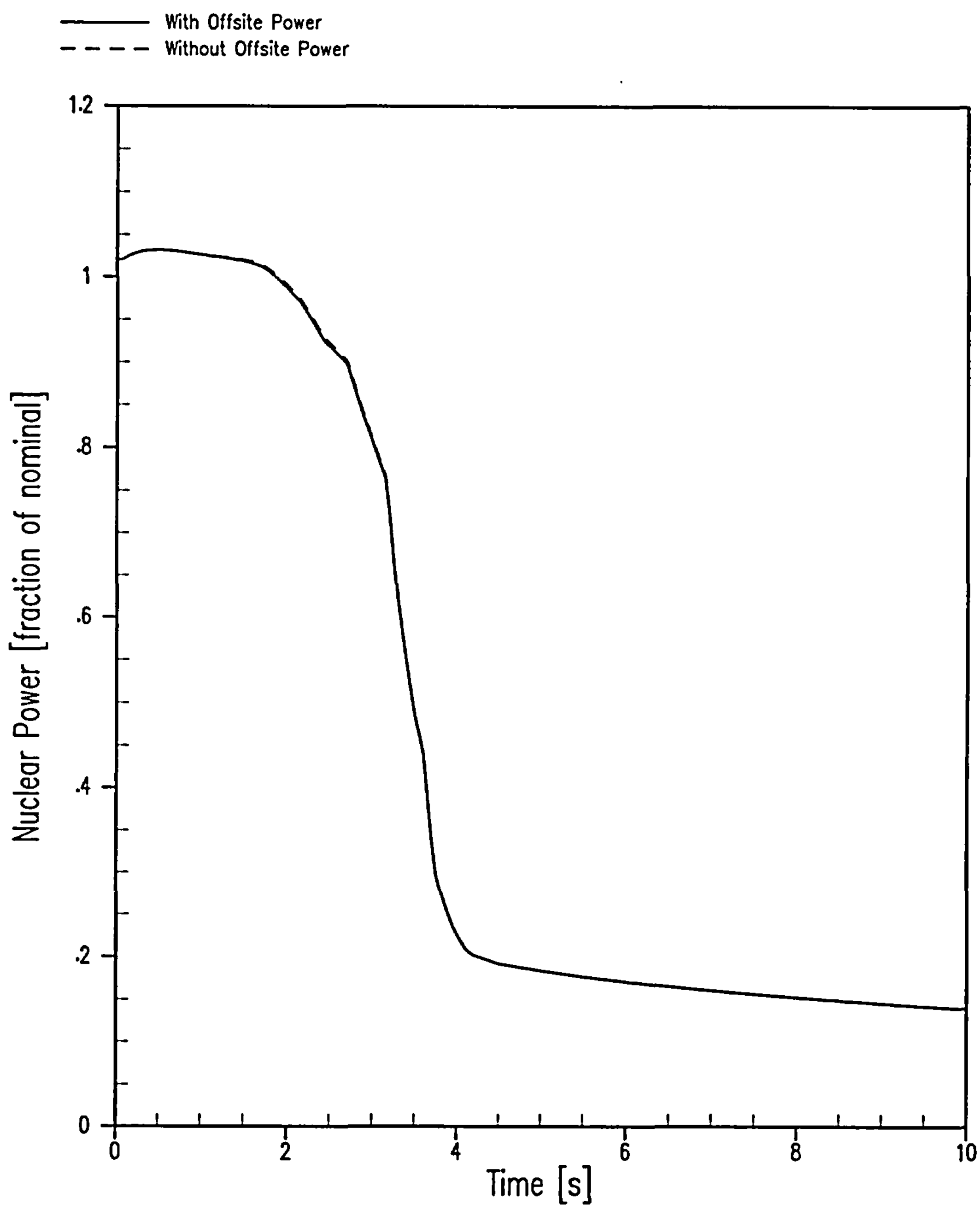


Figure 6.3.8-3 Locked Rotor/Shaft Break, RCS Pressure/Peak Cladding Temperature Case – Nuclear Power versus Time

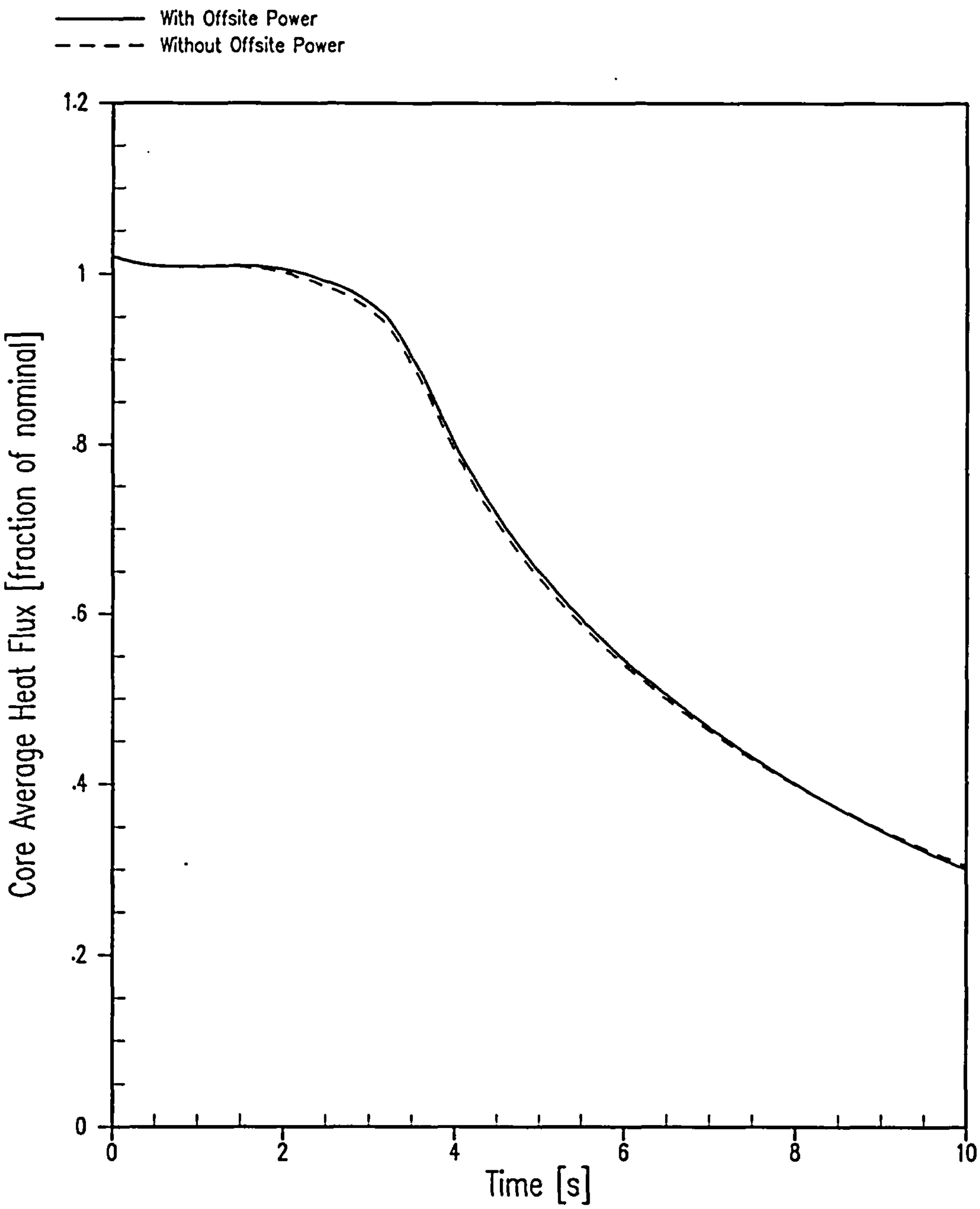


Figure 6.3.8-4 Locked Rotor/Shaft Break, RCS Pressure/Peak Cladding Temperature Case – Core Average Heat Flux versus Time

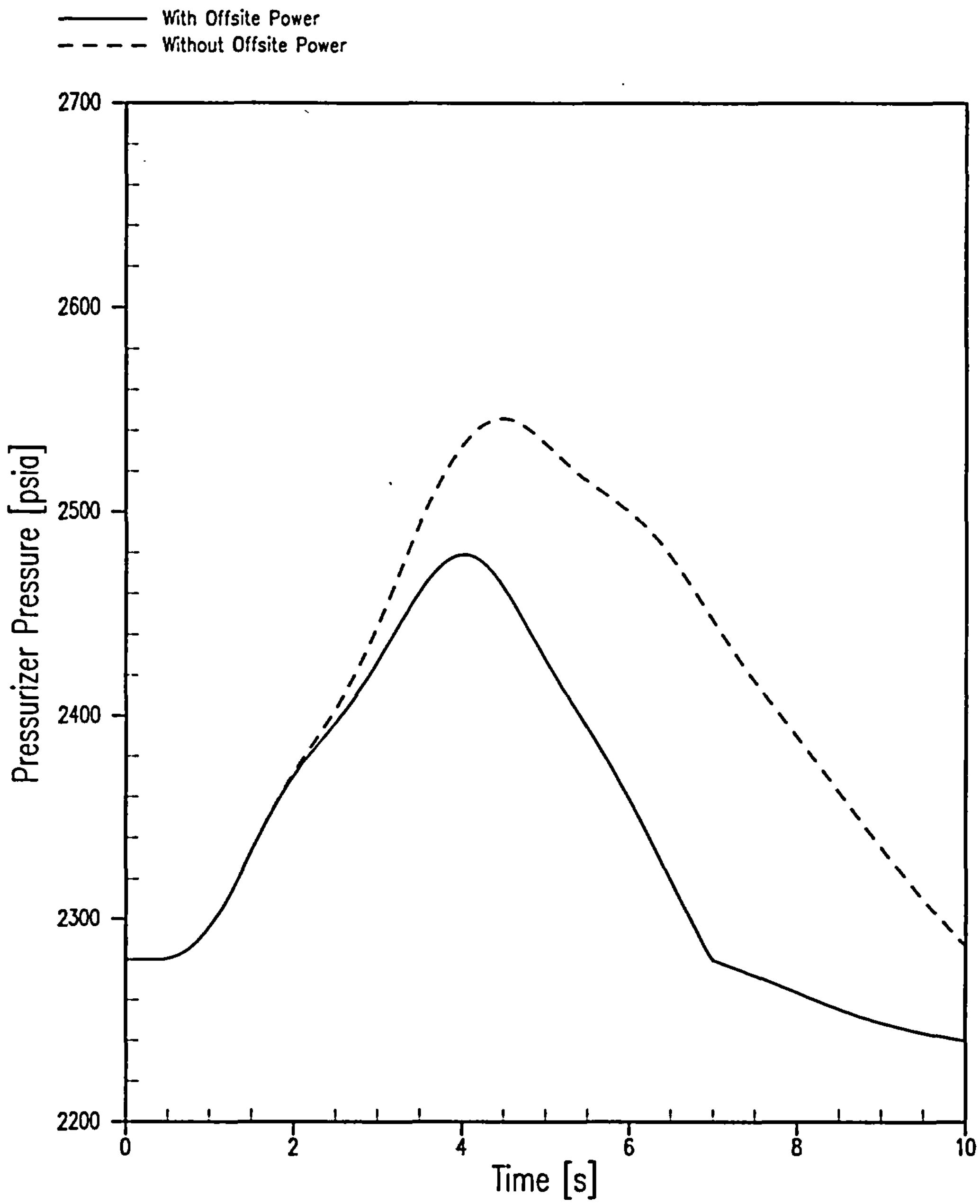


Figure 6.3.8-5 Locked Rotor/Shaft Break, RCS Pressure/Peak Cladding Temperature Case – Pressurizer Pressure versus Time

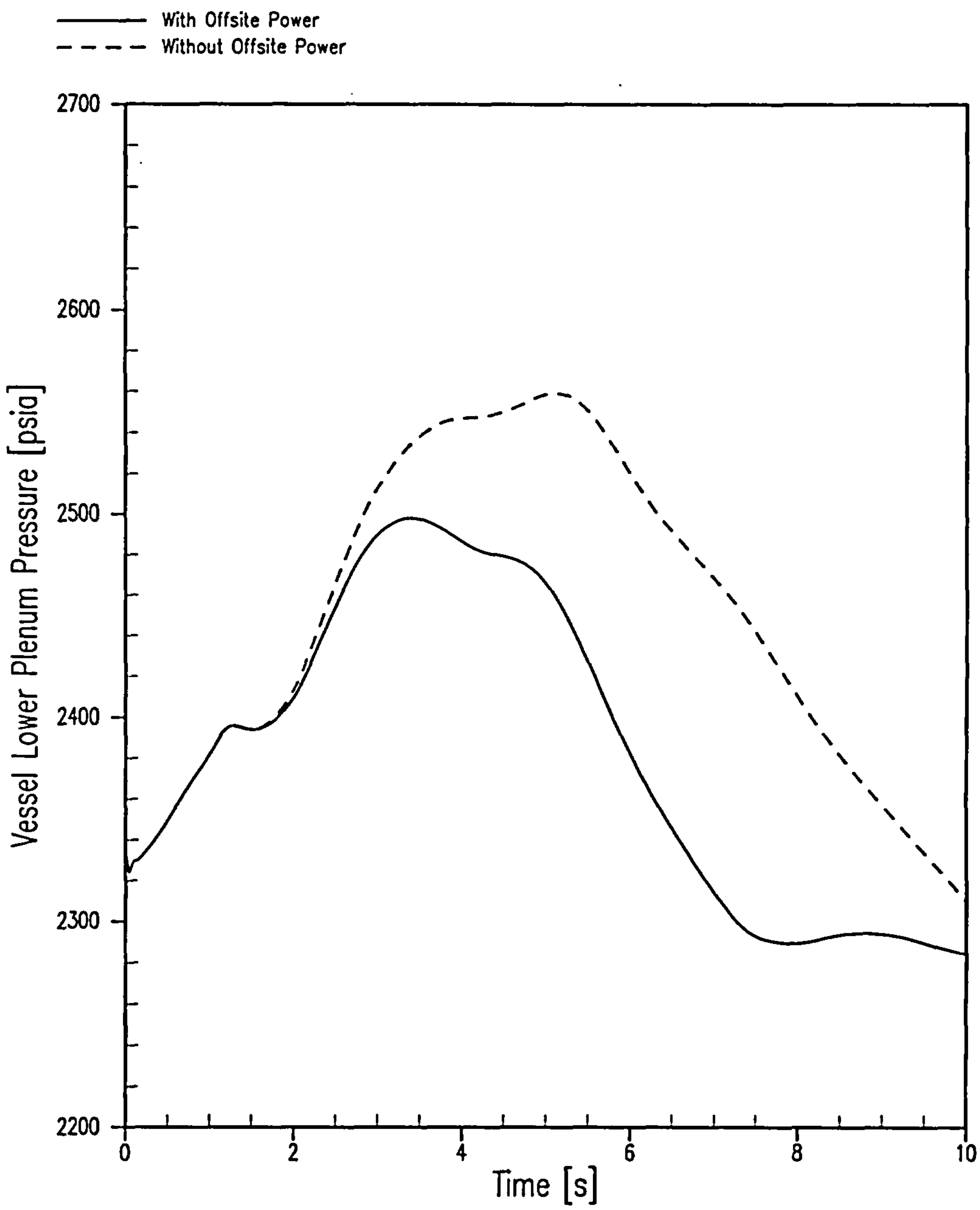


Figure 6.3.8-6 **Locked Rotor/Shaft Break, RCS Pressure/Peak Cladding Temperature Case – Vessel Lower Plenum Pressure versus Time**

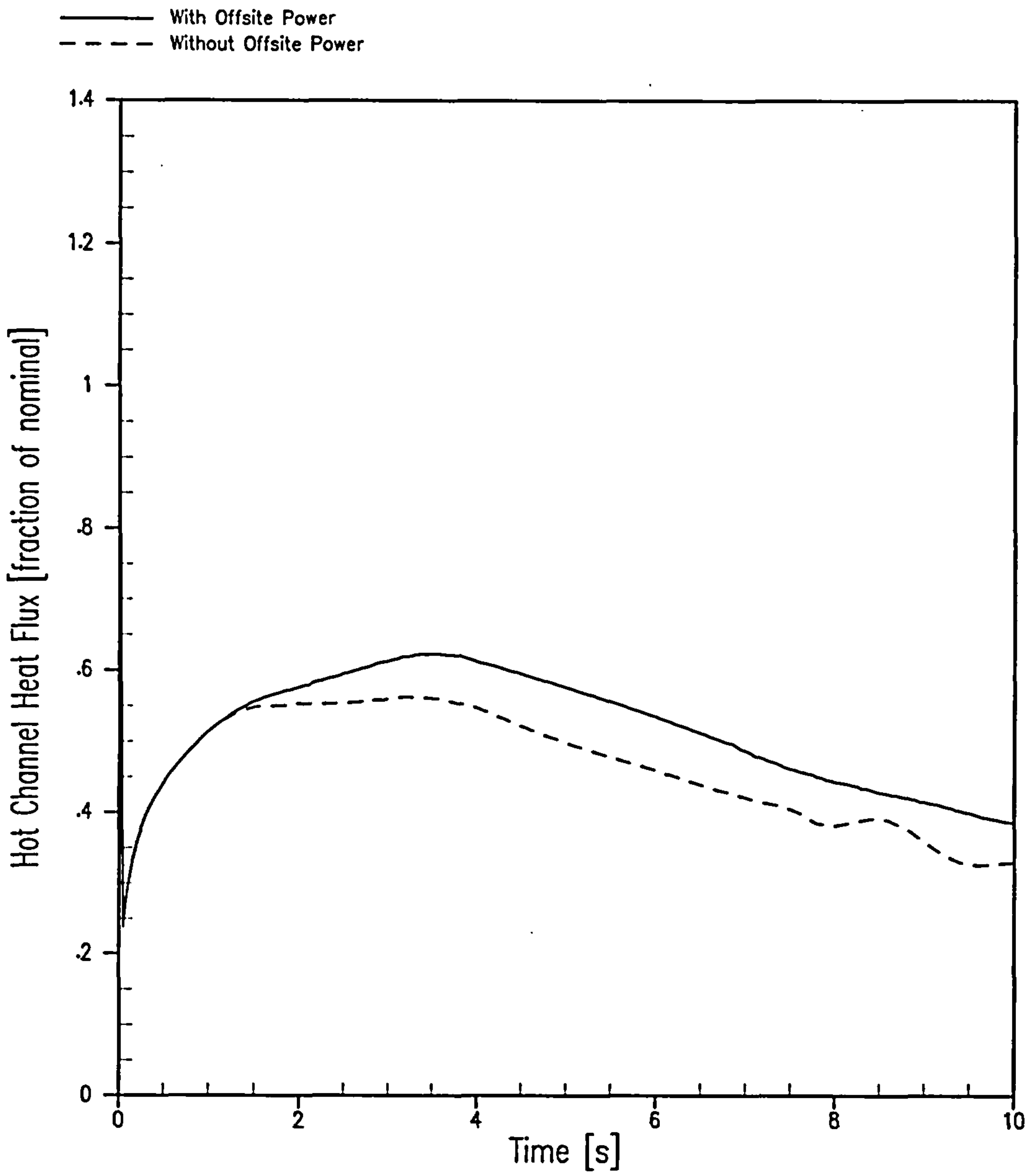


Figure 6.3.8-7 **Locked Rotor/Shaft Break, RCS Pressure/Peak Cladding Temperature Case – Hot Channel Heat Flux versus Time**

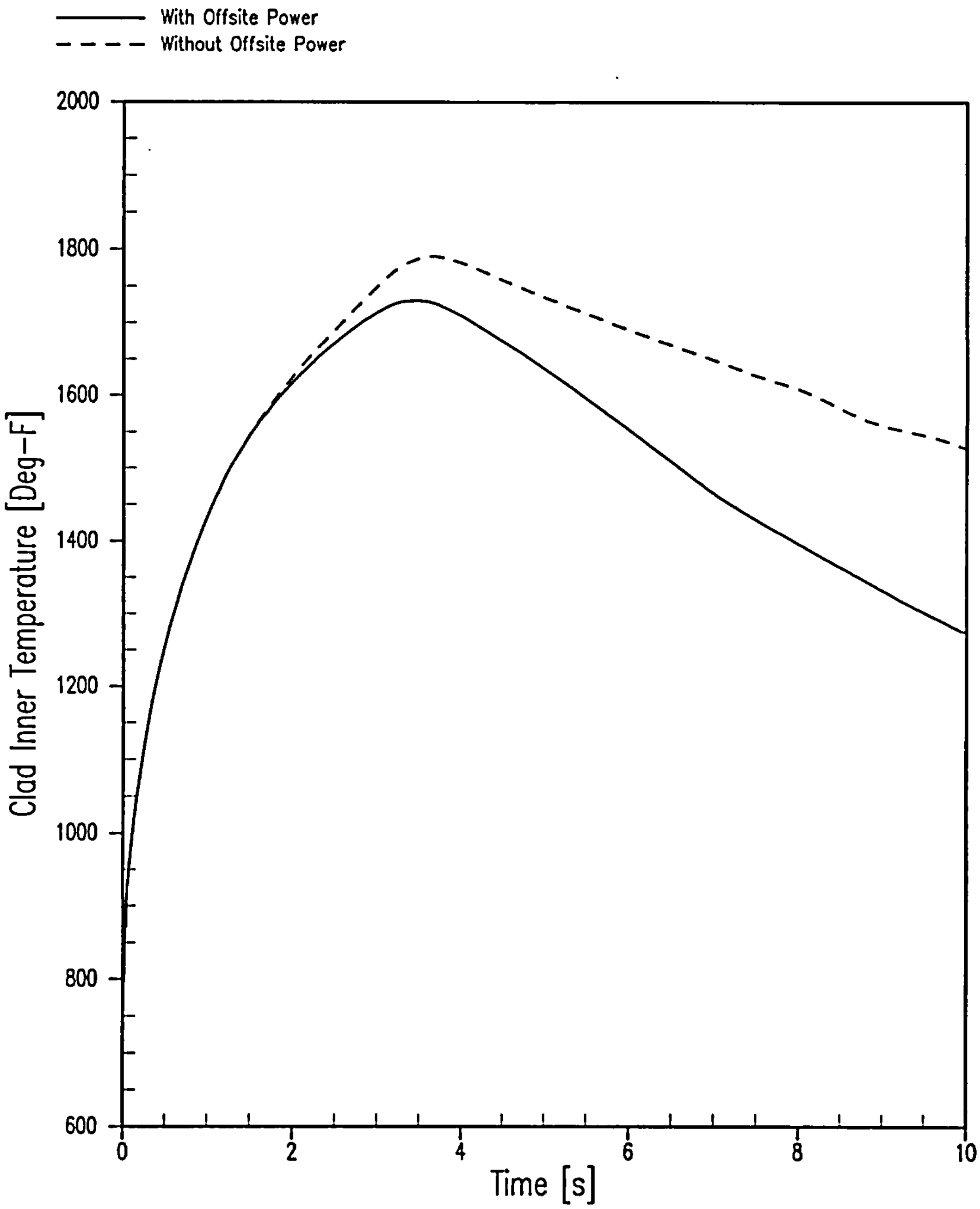


Figure 6.3.8-8 **Locked Rotor/Shaft Break, RCS Pressure/Peak Cladding Temperature Case – Hot Spot Cladding Inner Temperature versus Time**

6.3.9 Uncontrolled Rod Cluster Control Assembly Bank Withdrawal from a Subcritical or Low Power Startup Condition (FSAR Section 15.4.1)

The rod cluster control assembly (RCCA) withdrawal accident is defined as an uncontrolled addition of reactivity to the reactor core caused by withdrawal of RCCA banks resulting in a power excursion. While the occurrence of a transient of this type is unlikely, such a transient could be caused by a malfunction of the reactor control or the control rod drive system. This could occur with the reactor either subcritical, at hot zero power, or at power. The "at-power" case is discussed in Section 6.3.10.

The accident initiated from a subcritical or low power condition is not analyzed using the RETRAN code. Furthermore, the transient itself is not sensitive to secondary-side conditions (the steam generators are not explicitly modeled). Based on this, this event has not been re-analyzed in support of the Callaway Replacement Steam Generator (RSG) Program. The existing analysis of record for this event remains valid.

6.3.10 Uncontrolled Rod Cluster Control Assembly Bank Withdrawal at Power (FSAR Section 15.4.2)

6.3.10.1 Accident Description

The uncontrolled rod cluster control assembly (RCCA) bank withdrawal at power event is defined as the inadvertent addition of reactivity to the core caused by the withdrawal of RCCA banks when the core is above the no-load condition. The reactivity insertion resulting from the bank (or banks) withdrawal will cause an increase in core nuclear power and subsequent increase in core heat flux. An RCCA bank withdrawal can occur with the reactor subcritical, at hot zero power, or at power. The uncontrolled RCCA bank at power event is analyzed for Mode 1 (power operation). The uncontrolled RCCA bank withdrawal from a subcritical or low-power condition is considered as an independent event in Final Safety Analysis Report (FSAR) Section 15.4.1 and discussed in Section 6.3.9 of this report.

The event is simulated by modeling a constant rate of reactivity insertion starting at time zero and continuing until a reactor trip occurs. The analysis assumes a spectrum of possible reactivity insertion rates up to a maximum positive reactivity insertion rate greater than that occurring with the simultaneous withdrawal, at maximum speed, of two sequential RCCA banks having the maximum differential rod worth.

Unless the transient reactor coolant system (RCS) response to the RCCA bank withdrawal event is terminated by manual or automatic action, the power mismatch and resultant temperature rise could eventually result in departure from nucleate boiling (DNB) and/or fuel centerline melt. Additionally, the increase in RCS temperature caused by this event will increase the RCS pressure, and if left unchecked, could challenge the integrity of the RCS pressure boundary or the main steam system (MSS) pressure boundary.

To avert the core damage that might otherwise result from this event, the reactor protection system (RPS) is designed to automatically terminate any such event before the DNB ratio (DNBR) falls below the limit value, the fuel rod kW/ft limit is reached, the peak pressures exceed their respective limits, or the pressurizer fills. Depending on the initial power level and the rate of reactivity insertion, the reactor may be tripped and the RCCA withdrawal terminated by any of the following trip signals:

- Power-range high neutron flux
- Positive flux rate
- Overtemperature ΔT (OTΔT)
- Overpower ΔT (OPΔT)
- High pressurizer pressure
- High pressurizer water level

In addition to the previously listed reactor trips, there are the following withdrawal blocks for the control rod assemblies:

- High nuclear power (one-out-of-four channels)
- OPΔT (two-out-of-four channels)
- OTΔT (two-out-of-four channels)

6.3.10.2 Method of Analysis

The uncontrolled RCCA bank withdrawal at power event is analyzed to show that the integrity of the core is maintained by the RPS because the DNBR and peak kW/ft remain within the safety analysis limit values. The analysis also demonstrates that the peak MSSS pressure reached does not exceed the applicable safety analysis limit. A separate generic evaluation, performed to bound several Westinghouse-designed plants, demonstrates that the applicable RCS pressure limit is not exceeded during this event.

The RCCA bank withdrawal at power transient is analyzed with the RETRAN computer program (Reference 1). The RETRAN computer code is described in detail in Section 6.3.0.6.

To obtain a conservative value for the minimum DNBR, the following analysis assumptions are made:

1. This accident is analyzed with the Revised Thermal Design Procedure (RTDP) (Reference 2). Therefore, initial reactor power, pressure, and RCS temperatures are assumed to be at their nominal values. Uncertainties in initial conditions are included in the limit DNBR.

2. Reactivity coefficients – Two conditions are analyzed:

    a. Minimum reactivity feedback – A zero moderator temperature coefficient (MTC) of reactivity (0 pcm/°F) is assumed at full power. For transients initiated at 60-percent and 10-percent power, a positive MTC of reactivity (+5 pcm/°F) is conservatively assumed, corresponding to the beginning of core life. A conservatively small (in absolute magnitude) Doppler power coefficient (DPC) is used in the analysis.

    b. Maximum reactivity feedback – A conservatively large positive moderator density coefficient and a large (in absolute magnitude) negative DPC are assumed.

3. The reactor trip on high neutron flux is actuated at a conservative value of 118 percent of nominal full power. The OTΔT trip includes all adverse instrumentation and setpoint errors. The delays for trip actuation are assumed to be the maximum values. No credit was taken for the other expected trip functions.

4. The RCCA trip insertion characteristic is based on the assumption that the highest worth assembly is stuck in its fully withdrawn position.

5. A range of reactivity insertion rates is examined. The maximum positive reactivity insertion rate is greater than that which would be obtained from the simultaneous withdrawal of the two control rod banks having the maximum combined differential rod worth at a conservative speed (45 inches/minute, which corresponds to 72 steps/minute).

6. Initial power levels of 10, 60, and 100 percent of full power are considered.

7. The impact of a full-power RCS vessel Tavg window was considered for the uncontrolled RCCA bank withdrawal at power analysis. A conservative calculation modeling the high end of the RCS vessel Tavg window was explicitly analyzed.

6.3.10.3 Results

The calculated time sequences of events for two representative cases are listed in Table 6.3.10-1. The limiting results were calculated for the RCCA bank withdrawal at power transient and are provided in Table 6.3.10-2. Reactivity insertions rates of up to 110 pcm/s are considered in the analysis. This reactivity insertion rate bounds that calculated for the simultaneous withdrawal, at maximum speed, of two sequential RCCA banks having the maximum differential rod worth.

Figures 6.3.10-1 to 6.3.10-4 show the responses of nuclear power, pressurizer pressure, RCS vessel Tavg, and DNBR to a rapid RCCA withdrawal incident starting from full power. Reactor trip on high neutron flux occurs shortly after the start of the accident. Since this is rapid with respect to the thermal time constants of the plant, small changes in reactor core Tavg and pressurizer pressure result, and a large margin to DNB is maintained.

The responses of nuclear power, pressurizer pressure, RCS vessel Tavg, and DNBR for a slow control rod assembly withdrawal from 100-percent power are shown in Figures 6.3.10-5 to 6.3.10-8. Reactor trip on OTΔT occurs after a longer period of time and the rise in temperature is consequently larger than for a rapid RCCA withdrawal. Again, the minimum DNBR is greater than the limit value.

Figures 6.3.10-9 to 6.3.10-11 show the minimum DNBR as a function of the reactivity insertion rate for the three initial power levels (100, 60, and 10 percent, respectively) and for minimum and maximum reactivity feedback. It can be seen that the high neutron flux and OTΔT trip channels provide protection over the whole range of reactivity insertion rates because the minimum DNBR is never less than the limit value. In the referenced figures, the shapes of the curves of minimum DNBR versus reactivity insertion rate are due both to the reactor core and coolant system transient response and to protection system action in initiating a reactor trip.

Referring to Figure 6.3.10-11, for example, it is noted that for the minimum reactivity feedback case:

1. For high reactivity insertion rates (that is, between ~110 pcm/second and ~30 pcm/second) when modeling minimum reactivity feedback, reactor trip is initiated by the high neutron flux trip. The neutron flux level in the core rises rapidly for these insertion rates, while core heat flux and coolant temperature lag behind due to the thermal capacity of the fuel and coolant system fluid. Therefore, the reactor is tripped prior to a significant increase in the heat flux or core water temperature with resultant high minimum DNBRs during the transient. Within this range, as the reactivity insertion rate decreases, core heat flux and coolant temperatures can remain more nearly in equilibrium with the neutron flux. Therefore, minimum DNBR during the transient decreases with decreasing reactivity insertion rate.

2. With a further decrease in the reactivity insertion rate, the OTΔT and high neutron flux trips become equally effective in terminating the transient (such as, at a reactivity insertion rate of approximately 30 pcm/second).

   The OTΔT reactor trip function initiates a reactor trip when the measured ΔT exceeds an OTΔT setpoint that is based on the measured vessel Tavg and pressurizer pressure. It is important to note, however, that the contribution of RCS vessel Tavg to the OTΔT trip function is lead-lag compensated to compensate for the effect of the thermal capacity of the RCS response to power increases.

For reactivity insertion rates between ~30 pcm/second and ~10 pcm/second, the effectiveness of the OTΔT trip increases (in terms of increased minimum DNBR). This is due to the fact that, with lower insertion rates, the power increase rate is slower, the rate of rise of RCS vessel Tavg is slower, and the system lags and delays become less significant.

3. For reactivity insertion rates of ~10 pcm/second and lower, the rise in RCS temperature is sufficiently high so that there is an increased steam relief through the steam generator safety valves prior to trip. This steam relief acts as an additional heat sink on the RCS and sharply slows the increase of the RCS vessel Tavg. This causes the OTΔT trip setpoint to be reached later with resulting lower minimum DNBRs.

6.3.10.4 Conclusions

The results for the uncontrolled RCCA bank withdrawal at power transient analyzed show that the high neutron flux and OTΔT trip channels provide adequate protection over the entire range of possible reactivity insertion rates; that is, the minimum calculated DNBR is always greater than the safety analysis limit value. In addition, analysis results show that the peak kW/ft is less than the limit value. Peak MSS pressure remains below the safety analysis limit of 1,310.5 psia. The RCS pressure safety analysis limit of 2,748.5 psia is confirmed to be met via a generic evaluation.

Thus, all pertinent criteria are met for the uncontrolled RCCA bank withdrawal at power transient.

Table 6.3.10-1 Time Sequence of Events for Uncontrolled RCCA Withdrawal at Power (Maximum Nominal RCS Tavg; Minimum Feedback)	
Event	**Time (Seconds)**
Case A:	
Initiation of Uncontrolled RCCA Withdrawal at Full Power with Minimum Reactivity Feedback (110 pcm/sec)	0.0
Power-Range High Neutron Flux High Trip Point Reached	1.15
Rods Begin to Fall into Core	1.65
Minimum DNBR Occurs	2.58
Case B:	
Initiation of Uncontrolled RCCA Withdrawal at Full Power with Minimum Reactivity Feedback (1 pcm/sec)	0
OTΔT Reactor Trip Signal Initiated	77.06
Rods Begin to Fall into Core	79.06
Minimum DNBR occurs	79.50

Table 6.3.10-2 Limiting Results for RCCA Bank Withdrawal at Power Transient			
Criterion	**Limiting Value**	**Analysis Limit**	**Case**
DNBR	1.572	1.55	60% power, minimum reactivity feedback, 12.5 pcm/second reactivity insertion rate
Core Heat Flux (FON)	1.1712	1.1852	Full power, maximum reactivity feedback 34 pcm/second reactivity insertion rate
MSS Pressure (psia)	1,283.8	1,318.5	10% of full power, maximum reactivity feedback, 14 pcm/second reactivity insertion rate

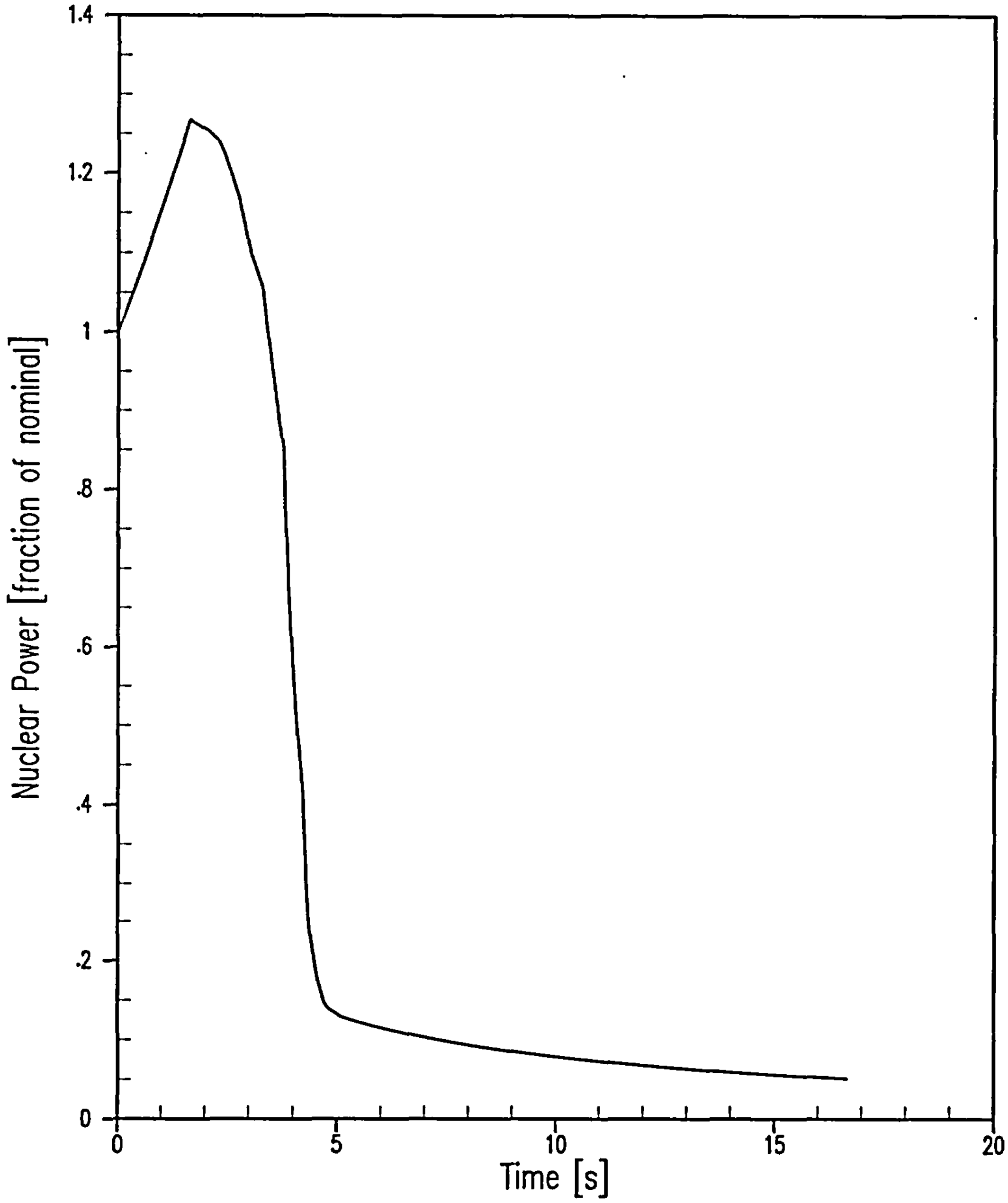


Figure 6.3.10-1 **Rod Withdrawal at Power (110 pcm/second Withdrawal Rate) – Nuclear Power versus Time**

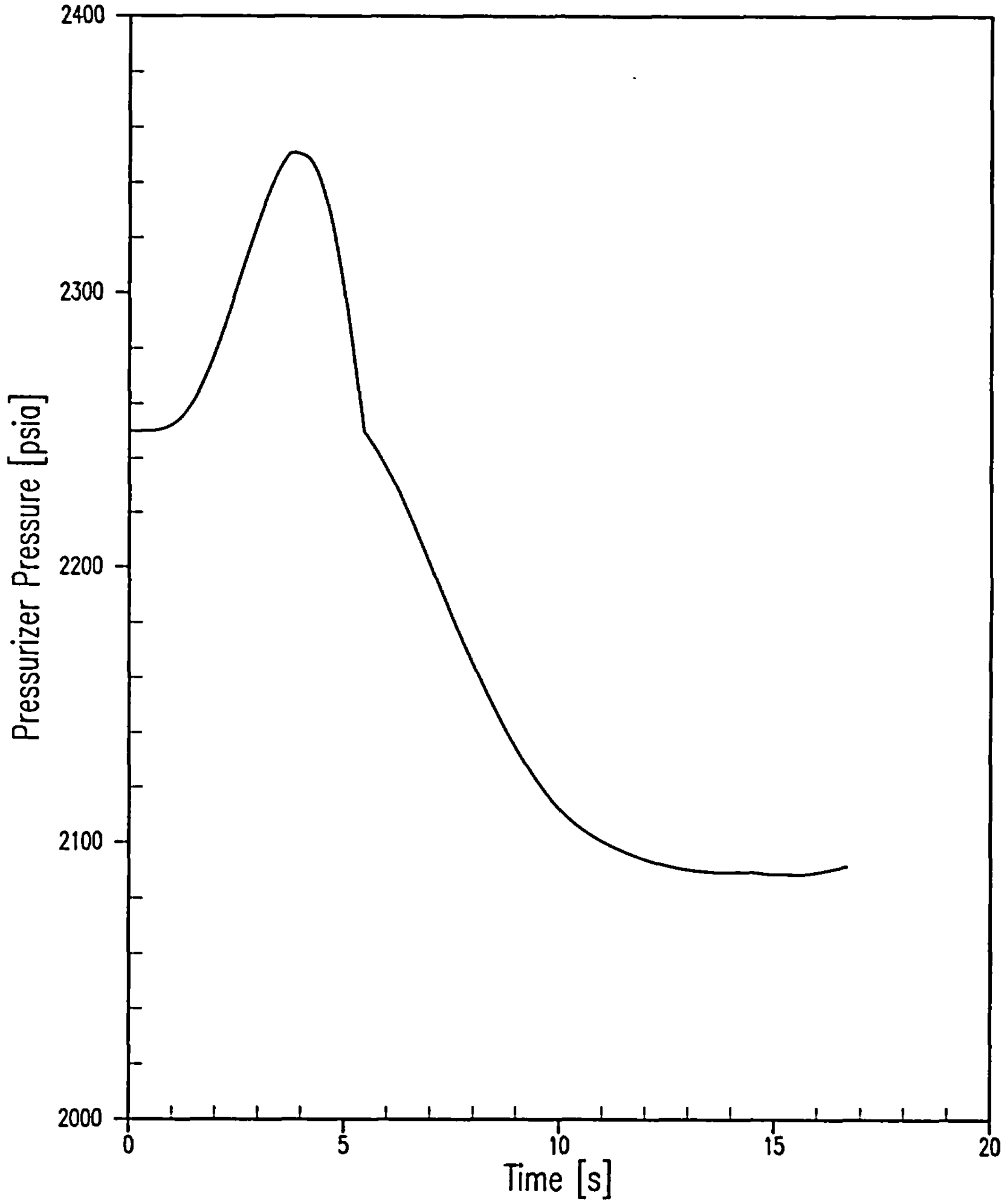


Figure 6.3.10-2 Rod Withdrawal at Power (110 pcm/second Withdrawal Rate) – Pressurizer Pressure versus Time

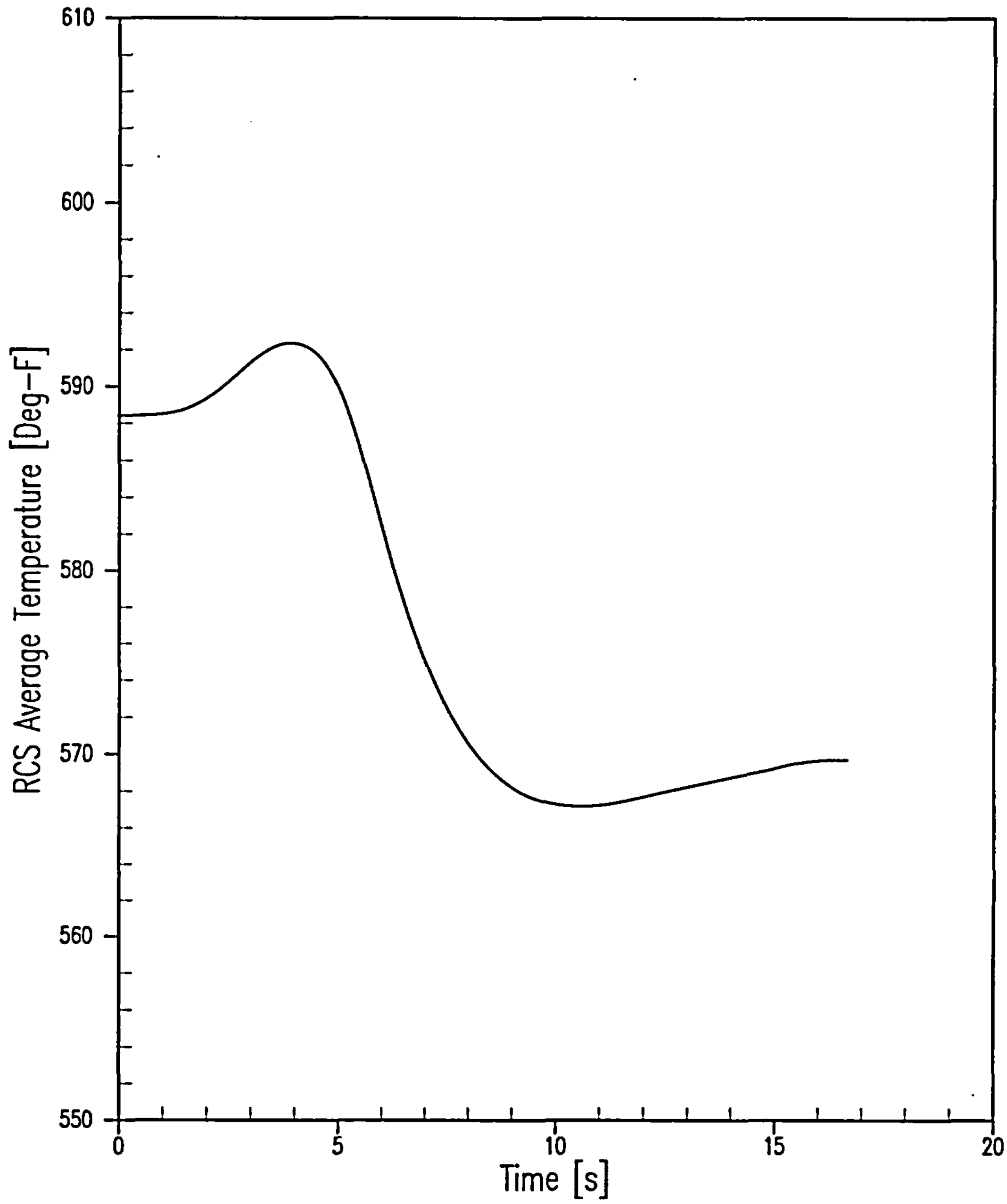


Figure 6.3.10-3 **Rod Withdrawal at Power (110 pcm/second Withdrawal Rate) – RCS Average Temperature versus Time**

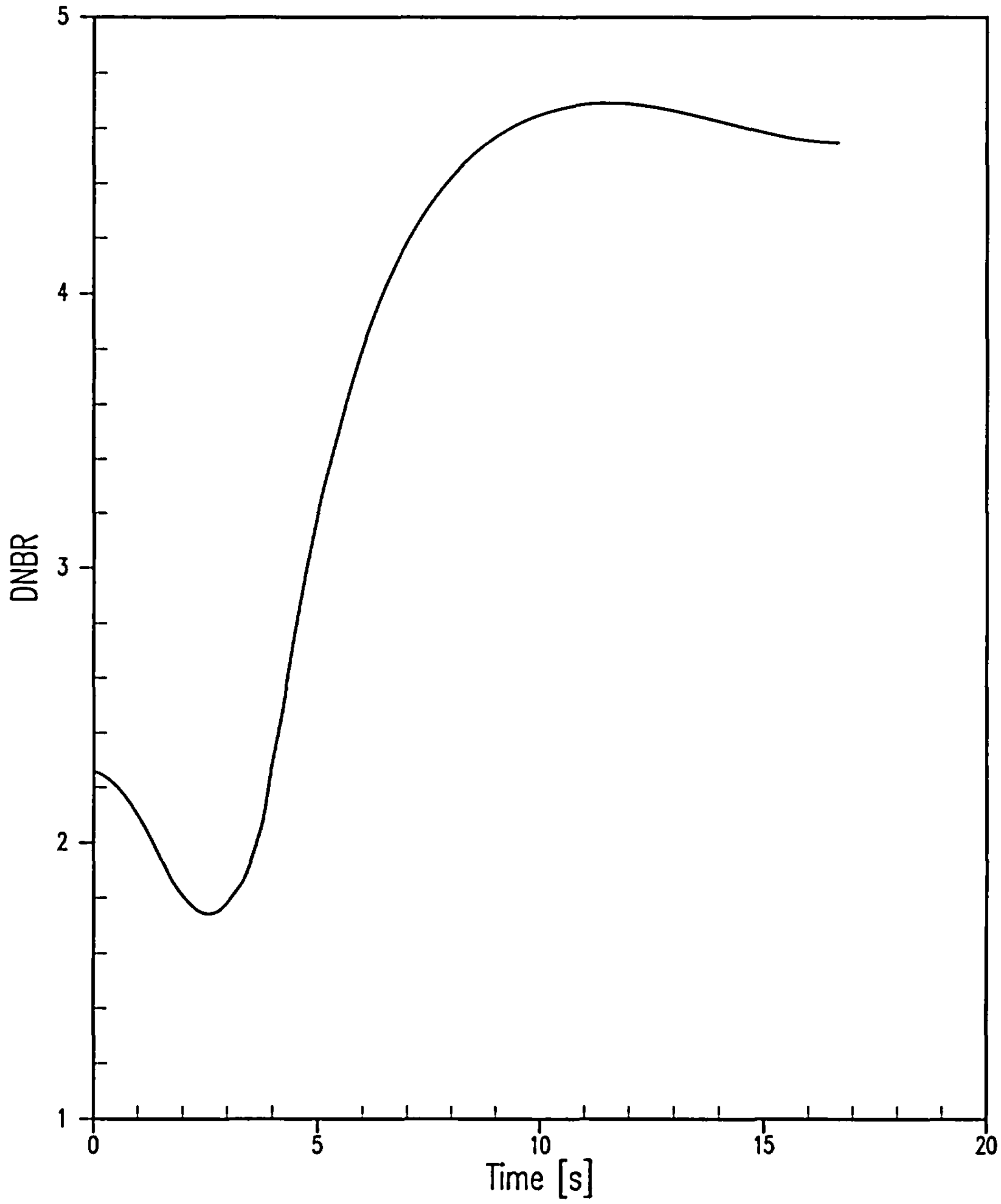


Figure 6.3.10-4 Rod Withdrawal at Power (110 pcm/second Withdrawal Rate) – DNBR versus Time

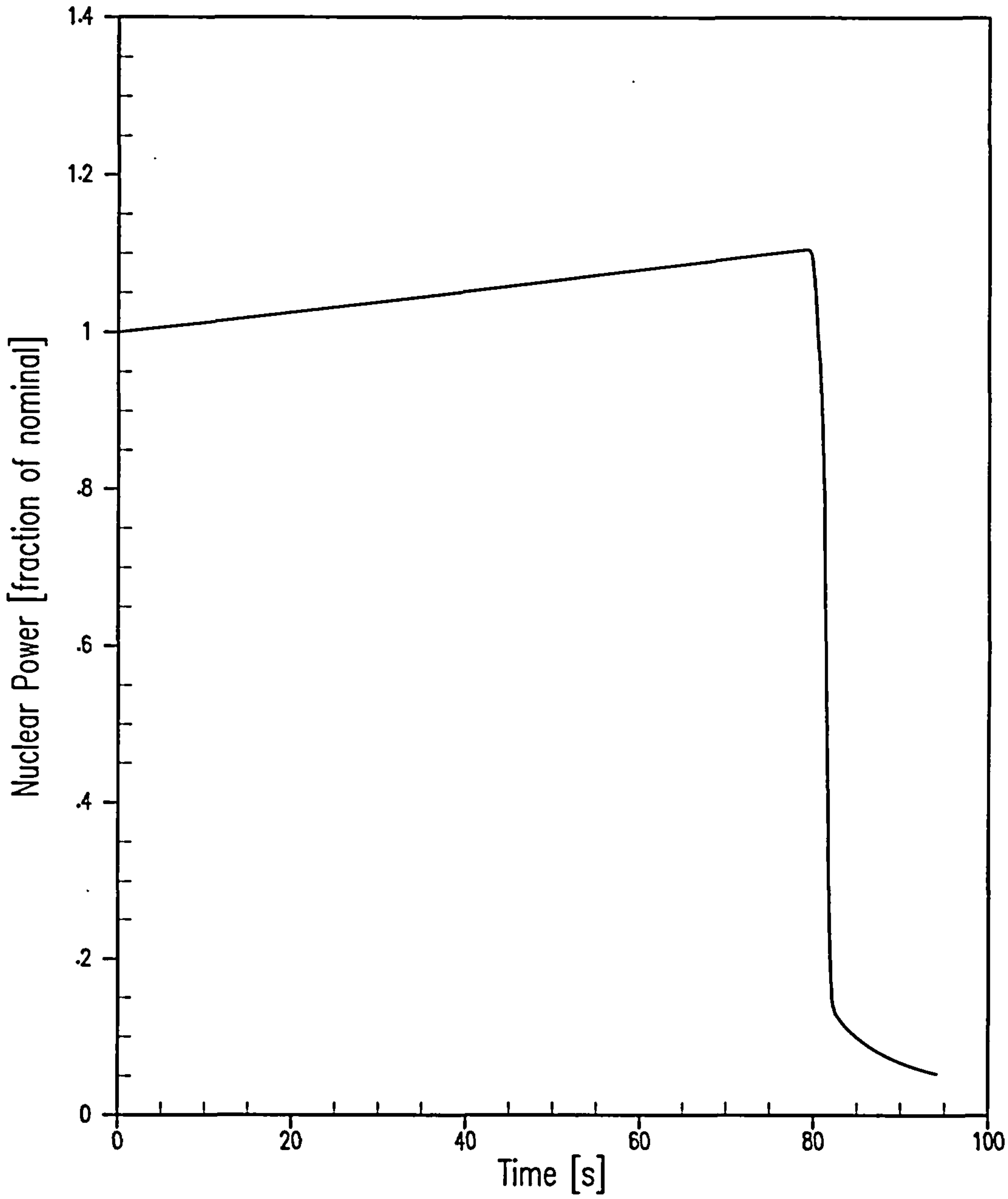


Figure 6.3.10-5 Rod Withdrawal at Power (1 pcm/second Withdrawal Rate) – Nuclear Power versus Time

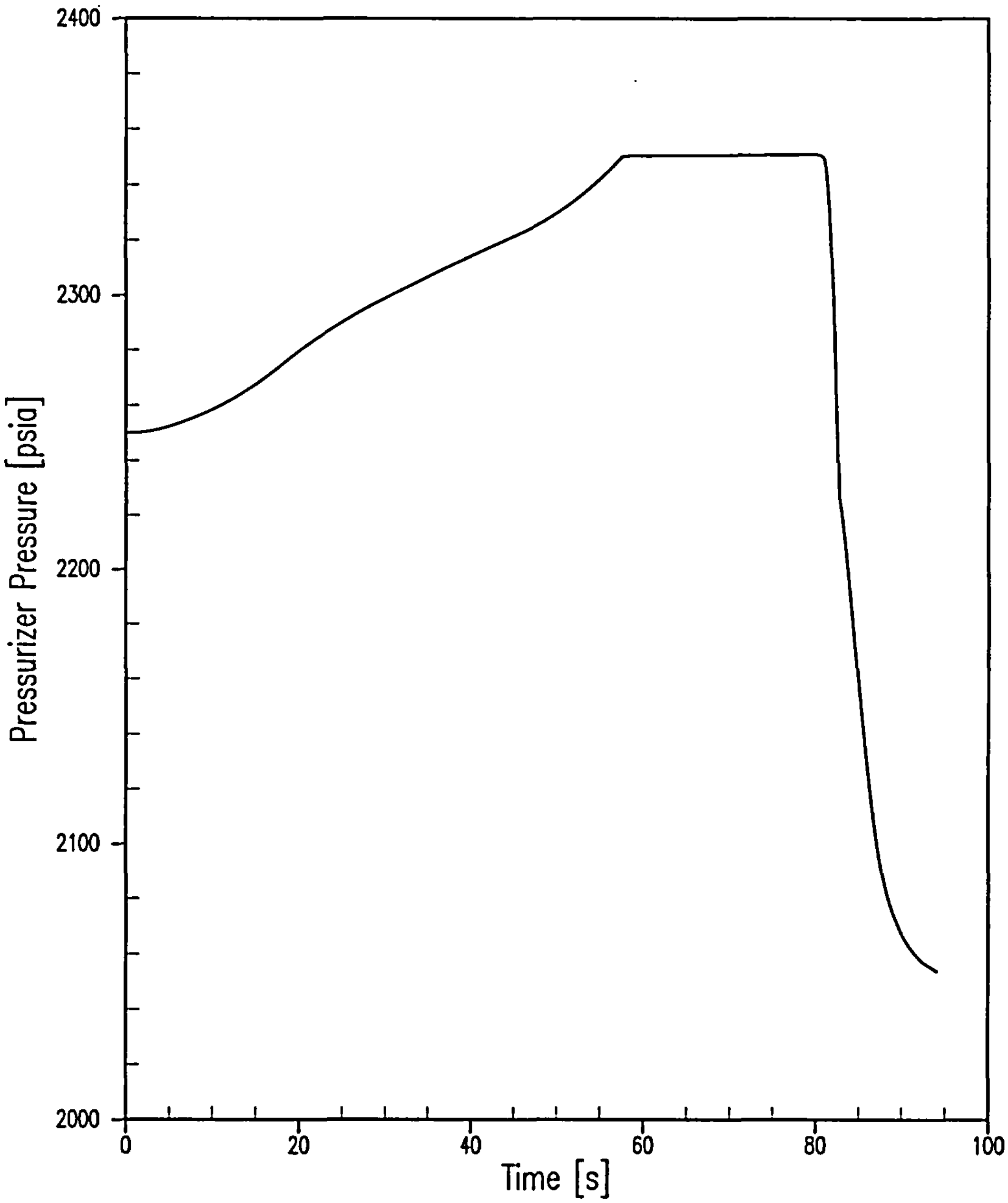


Figure 6.3.10-6 Rod Withdrawal at Power (1 pcm/second Withdrawal Rate) – Pressurizer Pressure versus Time

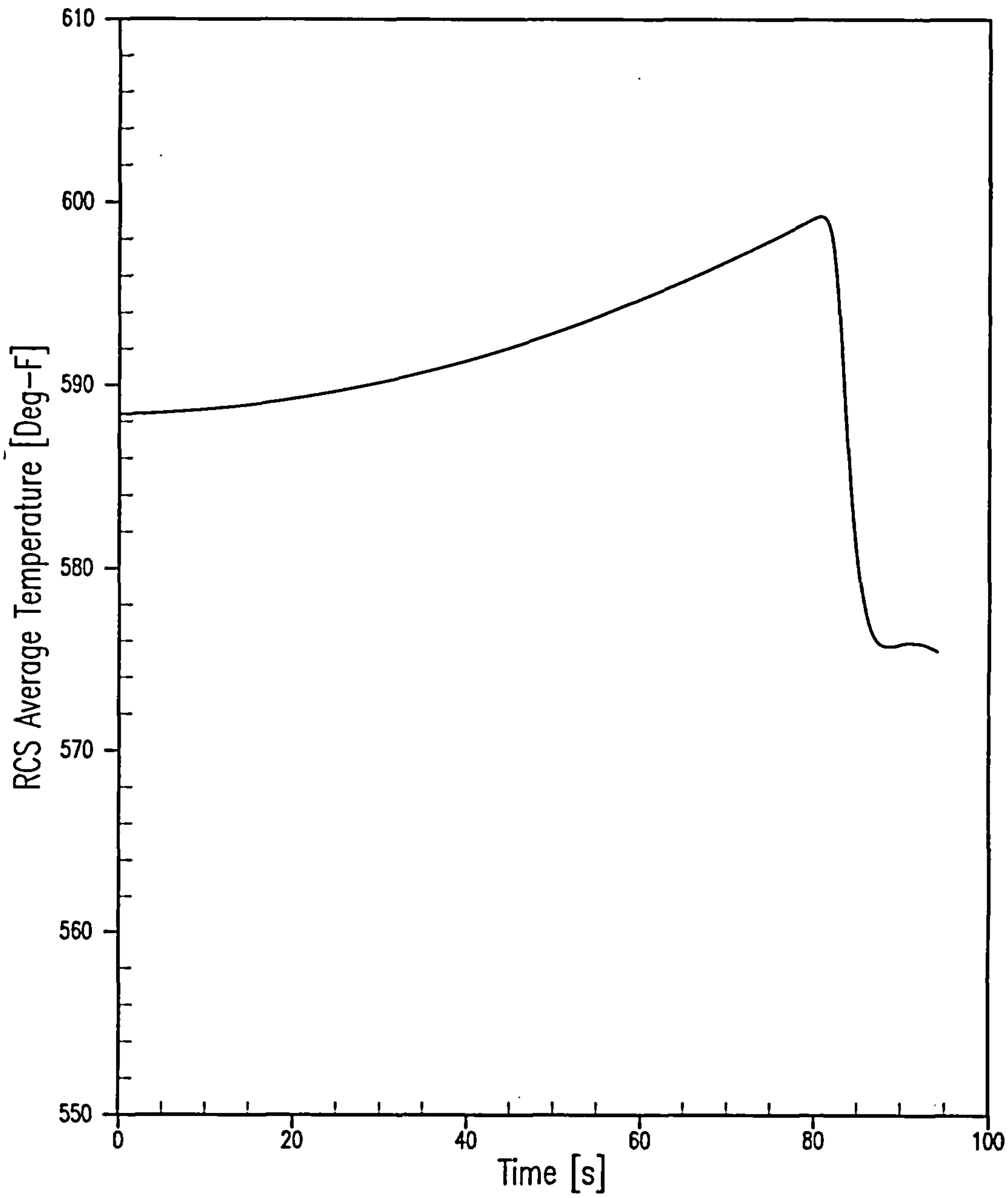


Figure 6.3.10-7 Rod Withdrawal at Power (1 pcm/second Withdrawal Rate) – RCS Average Temperature versus Time

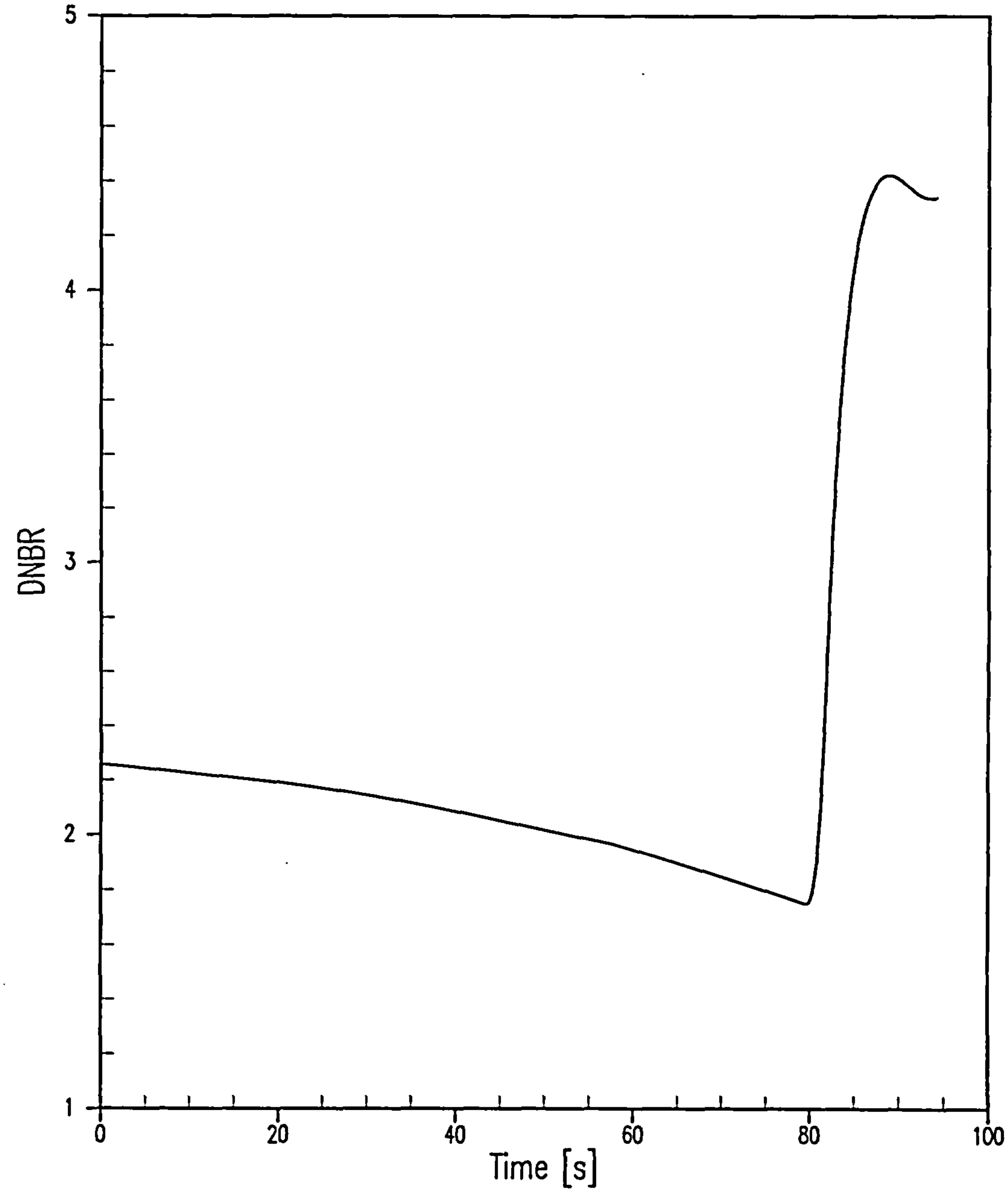


Figure 6.3.10-8 Rod Withdrawal at Power (1 pcm/second Withdrawal Rate) – DNBR versus Time

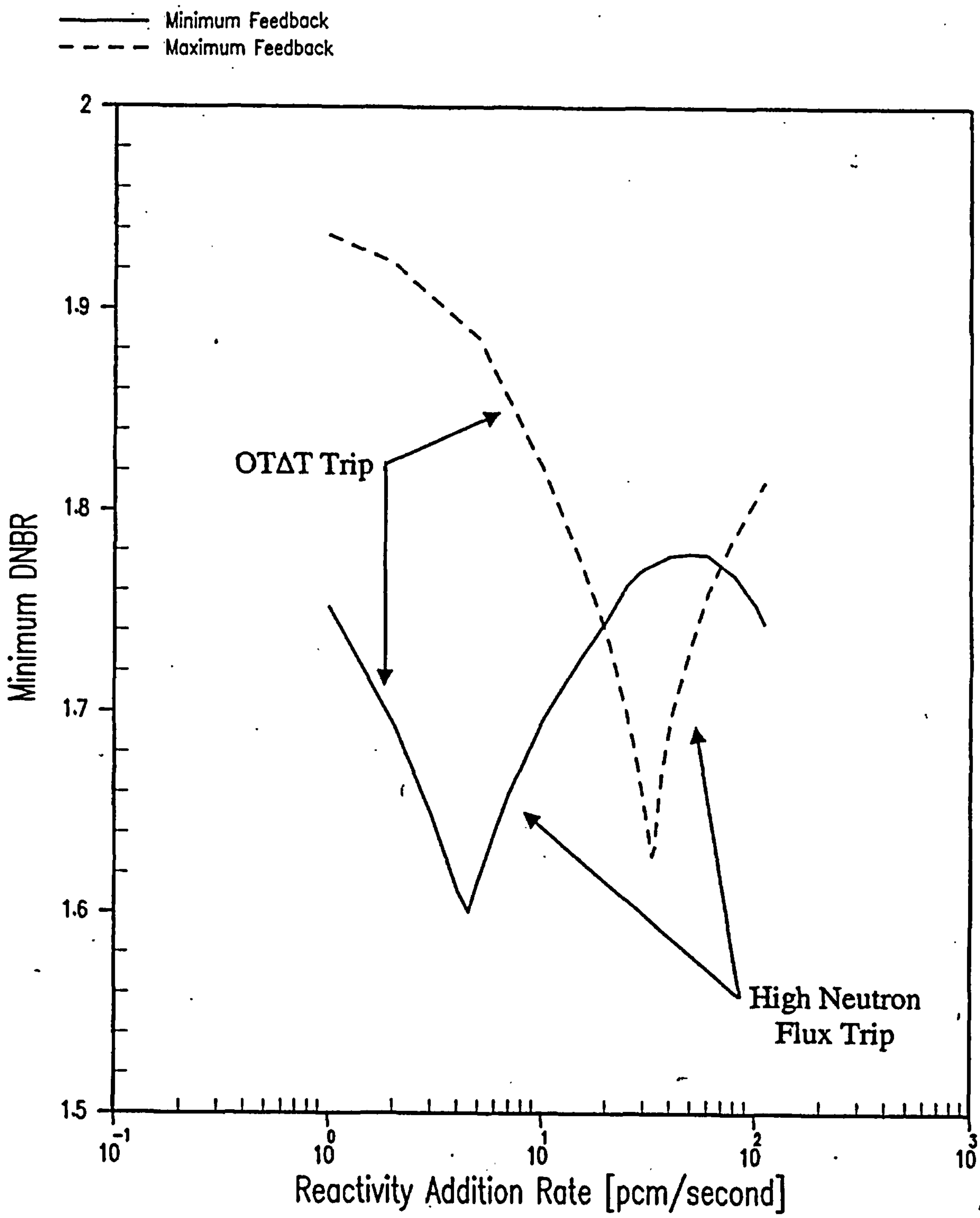


Figure 6.3.10-9 Rod Withdrawal at Power from Full Power – Minimum DNBR versus Reactivity Addition Rate (pcm/second)

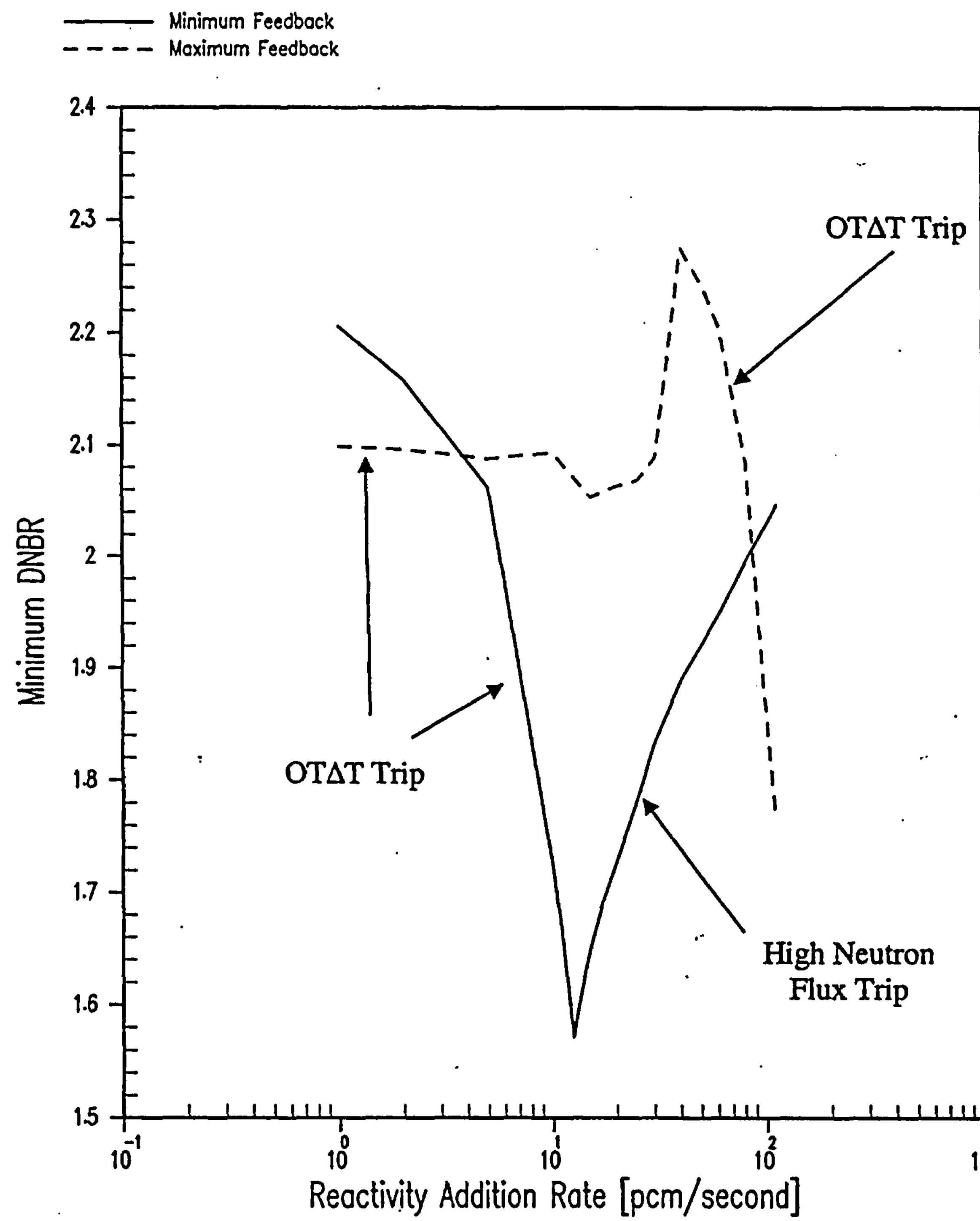


Figure 6.3.10-10 **Rod Withdrawal at Power from 60% Power – Minimum DNBR versus Reactivity Addition Rate (pcm/second)**

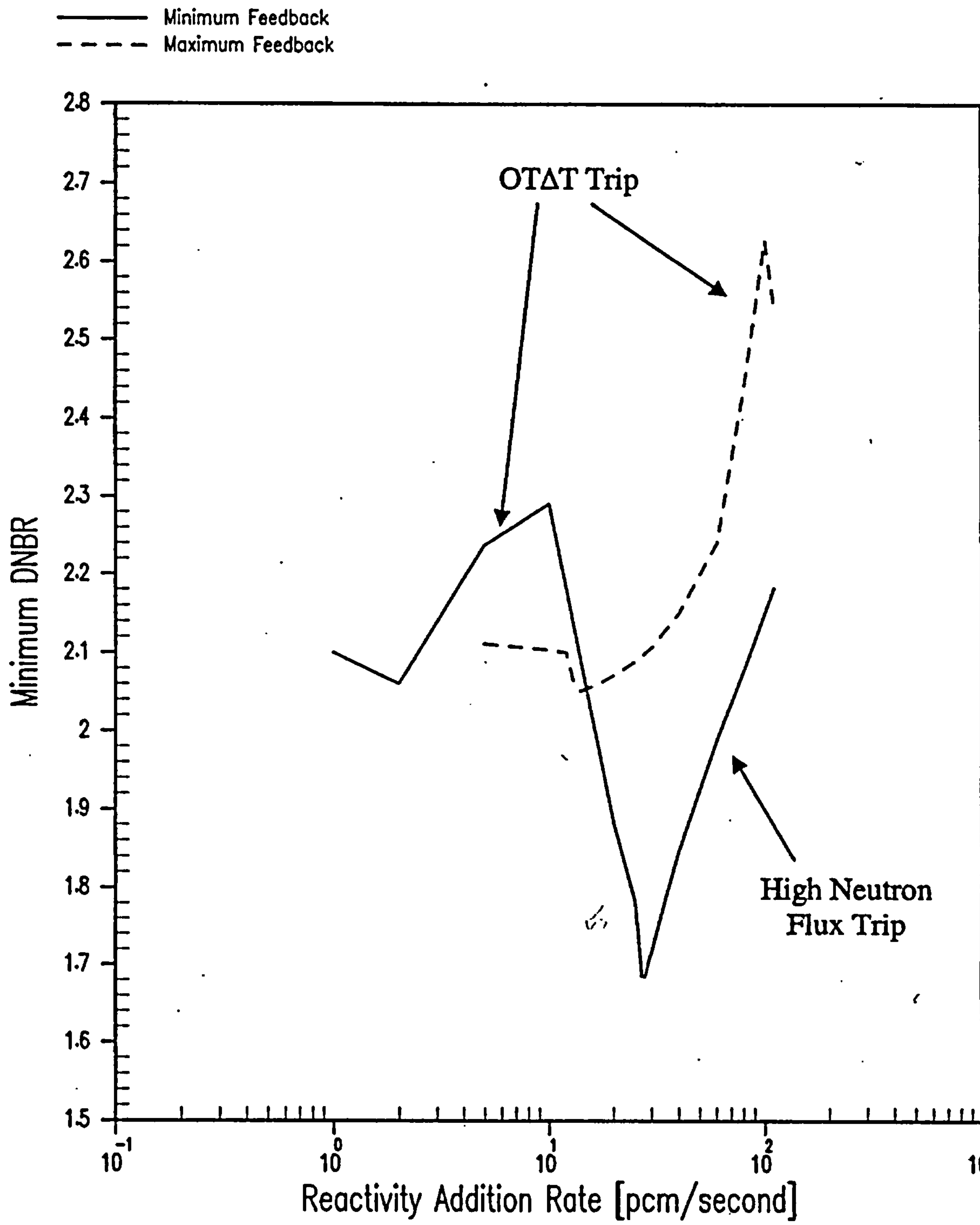


Figure 6.3.10-11 Rod Withdrawal at Power from 10% Power – Minimum DNBR versus Reactivity Addition Rate (pcm/second)

6.3.11 Rod Cluster Control Assembly Misoperation (FSAR Section 15.4.3)

The rod cluster control assembly (RCCA) misoperation events include incidents such as:

- One or more dropped full-length RCCAs
- A dropped full-length RCCA bank
- Statically misaligned full-length RCCA
- Withdrawal of a single RCCA

Each RCCA has a rod position indicator channel that displays the position of the assembly. The displays of assembly positions are grouped for operator convenience. Fully inserted assemblies are further indicated by rod bottom lights. The full-length assemblies are always moved in pre-selected banks and the banks are always moved in the same pre-selected sequence.

These accident scenarios have not been re-analyzed using the RETRAN code. Furthermore, these transients are not very sensitive to secondary-side conditions. In the case of the statically misaligned RCCA or the withdrawal of a single RCCA, the steam generators are not explicitly modeled. The cases covering dropped RCCAs or a dropped RCCA bank have been addressed in a generic statepoint analysis, performed in 1986 using the LOFTRAN code to bound a number of four-loop pressurized water reactors. These generic statepoints have been confirmed to remain valid for the Callaway Replacement Steam Generator (RSG) Program. Based on this, this event has not been re-analyzed in support of the Callaway RSG Program.

The generic dropped RCCA statepoints are evaluated every cycle as part of the reload safety evaluation (RSE) process to demonstrate that the applicable departure from nucleate boiling (DNB) design basis is met. They will once again be evaluated, based on cycle-specific data, prior to the installation of the RSGs at Callaway, planned to be complete for Cycle 15 operation.

6.3.12 Chemical and Volume Control System Malfunction that Results in a Decrease in the Boron Concentration in the Reactor Coolant (FSAR Section 15.4.6)

Reactivity can be added to the core by feeding primary-grade water into the reactor coolant system (RCS) via the reactor makeup portion of the chemical and volume control system. Boron dilution is a manual operation under strict administrative controls, with procedures calling for a limit on the rate and duration of dilution. A boric acid blend system is provided to permit the operator to match the boron concentration of reactor coolant makeup water during normal charging to that in the RCS. The chemical and volume control system is designed to limit, even under various postulated failure modes, the potential rate of dilution to a value which, after indication through alarms and instrumentation, provides for sufficient operator or automatic response time to correct the situation in a safe and orderly manner.

This event is explicitly analyzed for Callaway to cover operation in Modes 1 (manual and automatic rod control) through 5. The RETRAN code is not used in the analysis of this event. Although not directly modeled in the analysis, the steam generators are assumed to form part of the available mixing volume for this event. Thus, the replacement of the existing steam generators with the Framatome units could potentially impact the results currently presented in the plant's Final Safety Analysis Report (FSAR). No other change being considered under the Callaway Replacement Steam Generator (RSG) Program can adversely impact the current analysis of record for this event.

The active mixing volume assumed in the current analysis has been determined to remain conservative with respect to (that is, lower than) the total mixing volume with the RSGs installed. As such, the analysis of record remains valid following RSG implementation at Callaway.

6.3.13 Spectrum of Rod Cluster Control Ejection Accidents (FSAR Section 15.4.8)

This accident is the result of the extremely unlikely mechanical failure of a control rod drive mechanism pressure housing such that the reactor coolant system (RCS) pressure would eject the RCCA and drive shaft. The consequences of this mechanical failure, in addition to being a minor LOCA, may also be a rapid reactivity insertion together with an adverse core power distribution, possibly leading to localized fuel rod damage.

The accident, initiated from zero- or full-power conditions, is not analyzed using the RETRAN code. Furthermore, the transient itself is not sensitive to secondary-side conditions (the steam generators are not explicitly modeled). Based on this, this event has not been re-analyzed in support of the Callaway RSG Program. The existing analysis of record for this event remains valid.

6.3.14 Inadvertent ECCS Actuation at Power (FSAR Section 15.5.1)

6.3.14.1 Accident Description

An inadvertent or spurious actuation of the emergency core cooling system (ECCS) at-power event results in an increase in the reactor coolant system (RCS) inventory leading to the potential filling of the pressurizer. The inadvertent ECCS actuation at-power event could be caused by operator error or a false electrical actuating signal. Spurious actuation in plants designed with the "new" steam line break protection system may be caused by any of the following signals:

- High containment pressure
- Low pressurizer pressure
- Low steam line pressure
- Manual actuation

Following actuation of one or more of the above signals, the safety injection system (SIS) is actuated which results in borated water being pumped from the refueling water storage tank (RWST). The valves isolating the boron injection header (BIH) from the centrifugal charging pumps and the valves isolating the BIH from the injection header then automatically open. The centrifugal charging pumps then inject borated water into the cold leg of each RCS loop. The SI pumps also start automatically but provide no flow when the RCS is at normal pressure.

Normally, an SI signal results in an immediate automatic reactor trip, which in turn generates a turbine trip. However, it cannot be assumed that any single fault that actuated the ECCS will also produce a reactor trip. Even without an immediate reactor trip, the reactor will experience a negative reactivity excursion as a result of the injected borated water. This negative reactivity excursion results in a decrease in reactor power.

In manual rod control, the power mismatch causes a drop in Tavg and a shrinkage of the reactor coolant. Prior to a reactor trip, this results in a decrease in pressurizer pressure and water level and the turbine load will decrease due to the effect of reduced steam pressure on load when the turbine throttle valves are fully open. The decrease in RCS pressure results in an increase in SI flow associated with the ECCS pump performance characteristics.

In automatic rod control, the above effects may be compensated for by RCCA withdrawal as the control system responds to maintain programmed Tavg. Once the rods have been fully withdrawn, the event continues as described for operation in manual rod control. The transient is eventually terminated by the RPS low pressurizer pressure trip or by manual trip.

This event is trivial with respect to departure from nucleate boiling (DNB) since the conditions resulting from injecting borated water into the RCS are beneficial with respect to DNB. Depending on the control systems in operation, core power and RCS temperatures either remain near the initial nominal conditions or decrease during this event. The RCS flow remains constant throughout the event. A decrease in RCS pressure is the only condition that may occur which would adversely affect DNB. However, for the decrease in RCS pressure that may occur, the effects are more than offset by beneficial changes in power and temperature. The net effect is a DNBR that remains near the initial DNBR or increases throughout the event. Nevertheless, DNB is still considered for this event to satisfy NRC requirements as specified in Regulatory Guide 1.70 (Reference 8) and the Standard Review Plan (NUREG-0800, Reference 9).

The major concern that results from an Inadvertent ECCS Actuation at Power is associated with pressurizer overfill and subsequent water relief through the pressurizer safety valves (PSVs). The pressurizer water volume increases for this event as a result of the SI flow and operator action is ultimately required to terminate SI. The event is analyzed to demonstrate that sufficient time is available for appropriate operator action to preclude the opening of PSVs with the pressurizer in a water-solid condition.

6.3.14.2 Method of Analysis

The Inadvertent ECCS at-power event is analyzed using the RETRAN computer code (Reference 1). The RETRAN computer code is described in detail in Section 6.3.0.6 of this report.

Although DNB is not generally a concern for this event, the most limiting case for DNB is typically the case with minimum reactivity feedback with the plant assumed to be in manual rod control.

For maximizing the potential for pressurizer filling (and the possibility for subsequent water relief through the PSVs), the most limiting case is typically the maximum reactivity feedback condition with an immediate reactor trip on SI followed by a turbine trip on reactor trip.

To ensure that water relief through the PSVs is precluded for this event, operator actions are modeled. These operator actions include (1) reduction of SI flow by turning off the normal charging pump (NCP) (6 minutes after event initiation), and (2) assuring availability of both pressurizer power-operated relief valves (PORVs) (9 minutes after event initiation). Note that, although a single PORV is capable of controlling RCS pressure below the PSVs setpoint, both are required to be unblocked to address the potential failure of one PORV to open on demand.

Some of the key assumptions made in the two cases analyzed are listed below.

1. The DNBR case is analyzed using the RTDP (Reference 2). For this case, initial core power, reactor coolant temperature, and reactor coolant pressure are assumed to be at the nominal values consistent with steady-state full-power operation. Uncertainties in initial conditions are included in determining the DNBR limit value. For the case analyzed to address pressurizer filling concerns, the transient is analyzed using the STDP. For this case, initial core power is assumed at the maximum value consistent with steady-state full-power operation, including allowances for calibration and instrument errors. Initial pressurizer pressure is assumed at the nominal value minus uncertainties. Initial reactor coolant temperature is assumed to be at the nominal low Tavg value consistent with steady-state full-power operation minus uncertainties.

2. The DNBR case is analyzed assuming minimum reactivity feedback consistent with beginning-of-cycle (BOC) conditions. This includes assuming an moderator temperature coefficient (MTC) value consistent with BOC hot-full-power (HFP) conditions (that is, zero MTC) and a least negative Doppler power coefficient. The pressurizer filling case models maximum feedback (end

of cycle – EOC) including a large negative moderator temperature coefficient and a most negative Doppler power coefficient.

3. No credit is taken for the operation of the steam dump system or steam generator PORVs. The steam generator pressure rises to the lowest set safety valve setpoint, where steam release through the main steam safety valves (MSSVs) limits the secondary-side steam pressure to the setpoint values. The MSSVs are explicitly modeled in the analysis assuming a +3.0-percent tolerance with a 5 psi accumulation to full open. The MSSV model also accounts for a 10 psi pressure drop from the steam generator exit to the MSSV inlet in determining the opening setpoints and an additional 15 psi pressure drop at full-open and full-flow conditions.

4. The modeling of the pressurizer pressure control is as follows:

   a. For the case analyzed for DNB, full credit is taken for the effect of the pressurizer spray and PORVs in maintaining the primary coolant pressure. Pressurizer heaters are assumed to be inoperable.

   b. For the case analyzed for pressurizer filling concerns, only the proportional heaters and spray are assumed to be available from the start of the transient. For the purposes of maximizing the potential for water relief through the pressurizer safety valves, it is conservatively assumed that the pressurizer PORVs are not available at the start of the transient. Operator action to unblock the PORVs is assumed to occur no later than 9 minutes from the start of the event.

5. The PSVs are assumed operable with an opening setpoint of 2,425.8 psia, which accounts for a -2-percent setpoint tolerance on their nominal setpoint of 2,475 psia. The analysis, specifically the pressurizer filling case, is performed to demonstrate that appropriate and timely operator actions can be taken to ensure that the safety valves will not actuate with the pressurizer in a water-solid condition.

6. Maximum decay heat, consistent with the 1979 version of ANS 5.1, is assumed.

6.3.14.3 Results

Figures 6.3.14-1 through 6.3.14-6 and Figures 6.3.14-7 through 6.3.14-11 show the evolution of the key parameters during the transient for the DNBR and pressurizer filling cases, respectively. Tables 6.3.14-1 and 6.3.14-2 present the sequence of events for the two analyzed cases.

Pressurizer Filling Concern

Reactor trip occurs at event initiation followed by a rapid initial cooldown of the RCS. Coolant contraction results in a short-term reduction in pressurizer pressure and water level. The combination of the RCS heatup, due to residual RCS heat generation, and ECCS injected flow causes the pressure and level transient to rapidly turn around. Pressurizer water level then increases throughout the transient.

At six minutes into the transient, the analysis assumes that the normal charging pump flow is terminated via operator action. The pressurizer becomes water solid shortly before nine minutes into the transient. At nine minutes into the transient, it is assumed that appropriate operator actions have been taken to assure that at least one pressurizer PORV is available for pressure relief. The PORVs then actuate and the RCS depressurizes. At no time is the pressurizer safety valves setpoint challenged.

DNBR Concern

Nuclear power starts decreasing immediately due to boron injection, but steam flow does not decrease until later in the transient, when the turbine throttle valve is wide open. The mismatch between load and nuclear power causes Tavg, pressurizer water level and pressurizer pressure to drop. The reactor trips and control rods start moving into the core when the pressurizer pressure reaches the low pressurizer pressure trip setpoint. The DNBR remains at or above its initial value throughout the transient.

6.3.14.4 Conclusions

With respect to the pressurizer filling concern, although the pressurizer may reach a water-solid condition, operator actions to isolate the NCP and assure that at least one pressurizer PORV is available for pressure relief prevent the pressurizer safety valves from actuating with the pressurizer in a water-solid condition. Water relief through the pressurizer safety valves is therefore precluded.

With respect to the DNBR concern, the results of the analysis show that inadvertent ECCS actuation at power doesn't present any hazard to the integrity of the RCS. The DNBR remains at or above its initial value throughout the transient.

Table 6.3.14-1 Time Sequence of Events for Inadvertent ECCS Actuation at Power Pressurizer Filling Case	
Event	**Time (Seconds)**
Inadvertent SI Actuation (manual)	0.0
Reactor Trip	0.0
Turbine Trip	0.01
MSSV Actuation	172
Operator switches the NCP Off	360
Pressurizer Fills	525
Operator Makes PORVs Available	540
Pressurizer PORV Begins Relief	540
Pressurizer Safety Valve Opening	---

Table 6.3.14-2 Time Sequence of Events for Inadvertent ECCS Actuation at Power DNBR Case	
Event	**Time (Seconds)**
Inadvertent SI Actuation (Manual)	0.0
Minimum DNBR Reached	0.2
Low Pressurizer Pressure Reactor Trip Setpoint Reached	67.6
Rod Motion Begins	69.6
Results	
Minimum DNBR Value	2.335
DNBR Limit	1.55

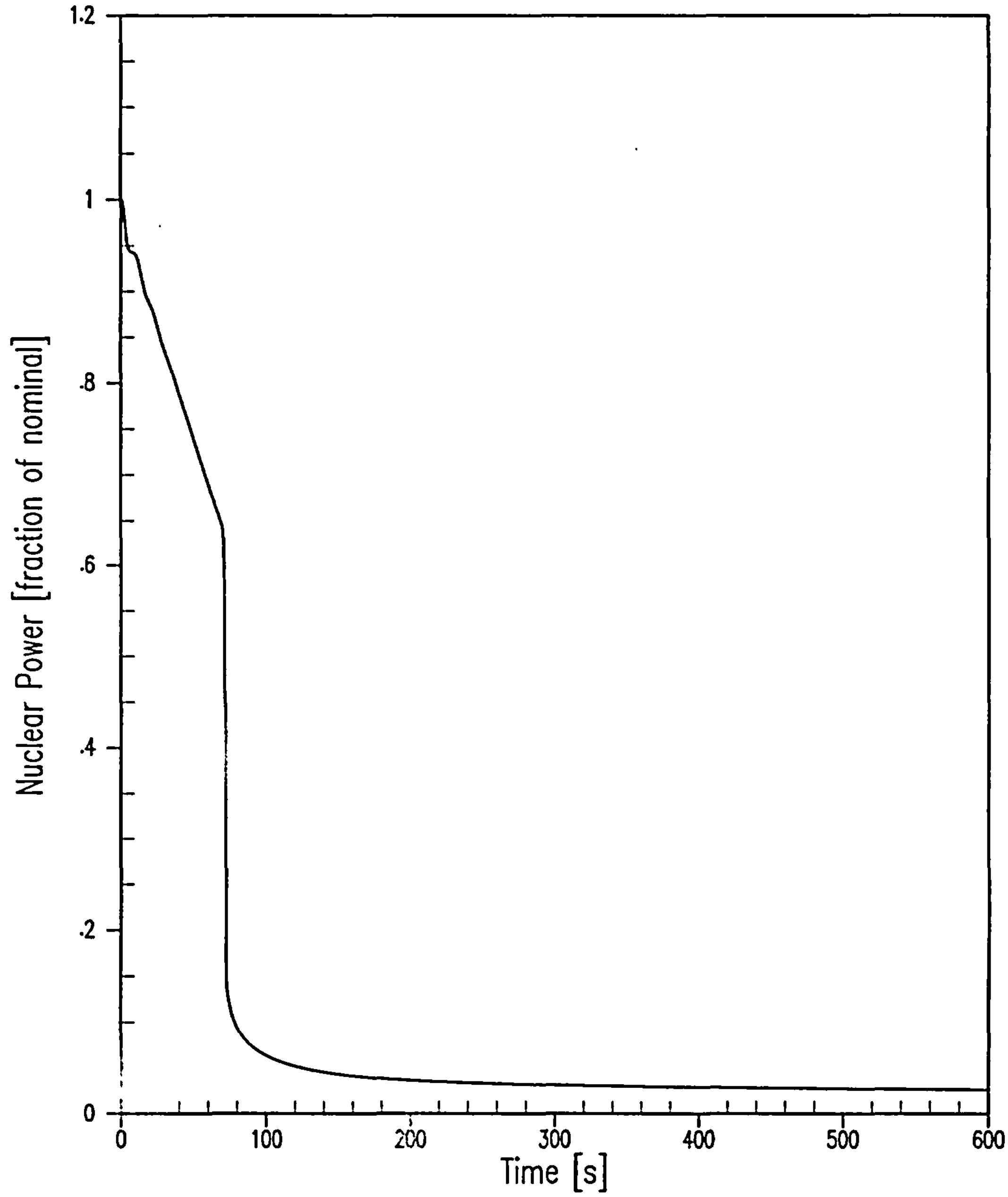


Figure 6.3.14-1 Inadvertent ECCS Actuation at Power (DNBR Case) – Nuclear Power versus Time

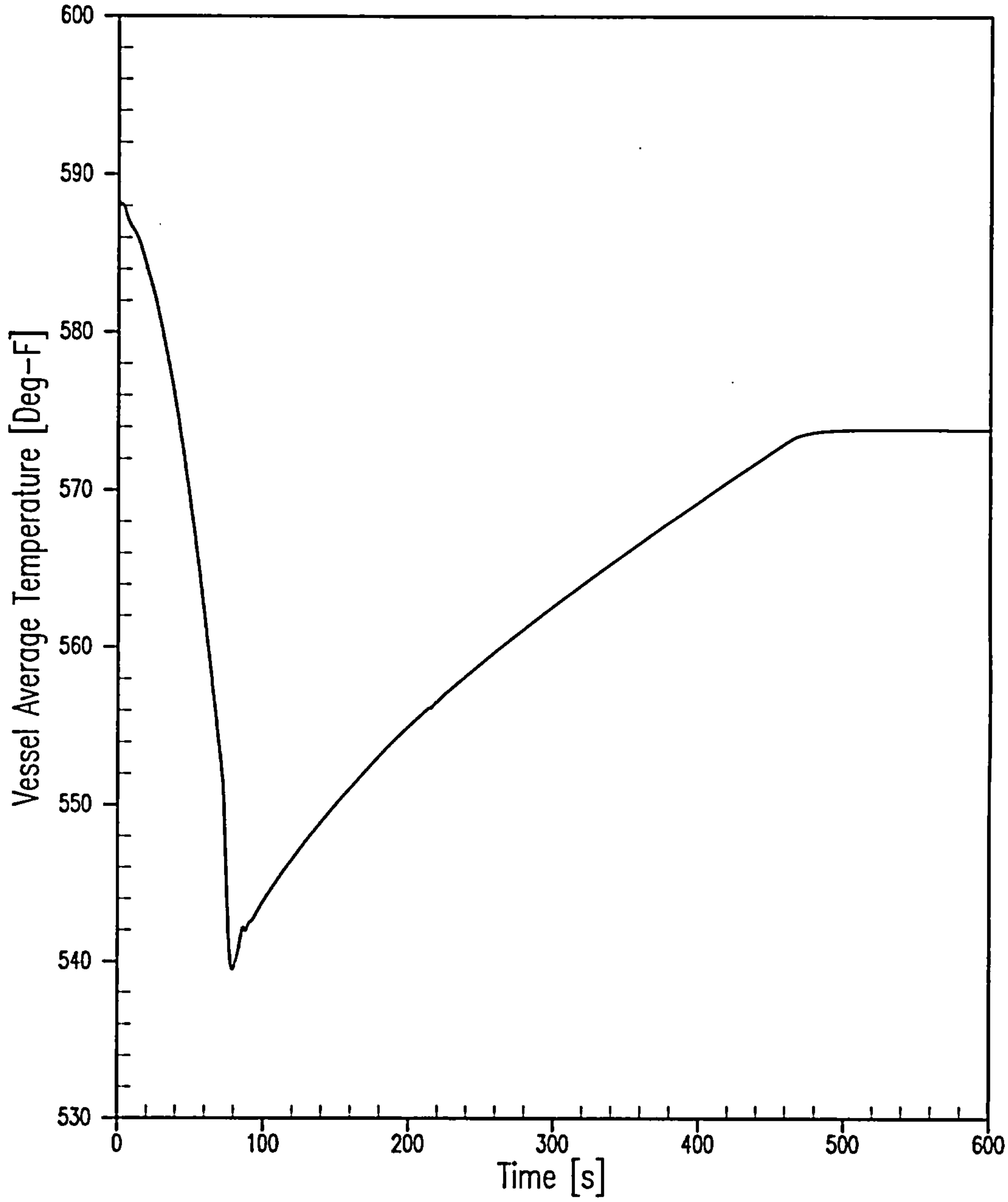


Figure 6.3.14-2 Inadvertent ECCS Actuation at Power (DNBR Case) – Vessel Average Temperature versus Time

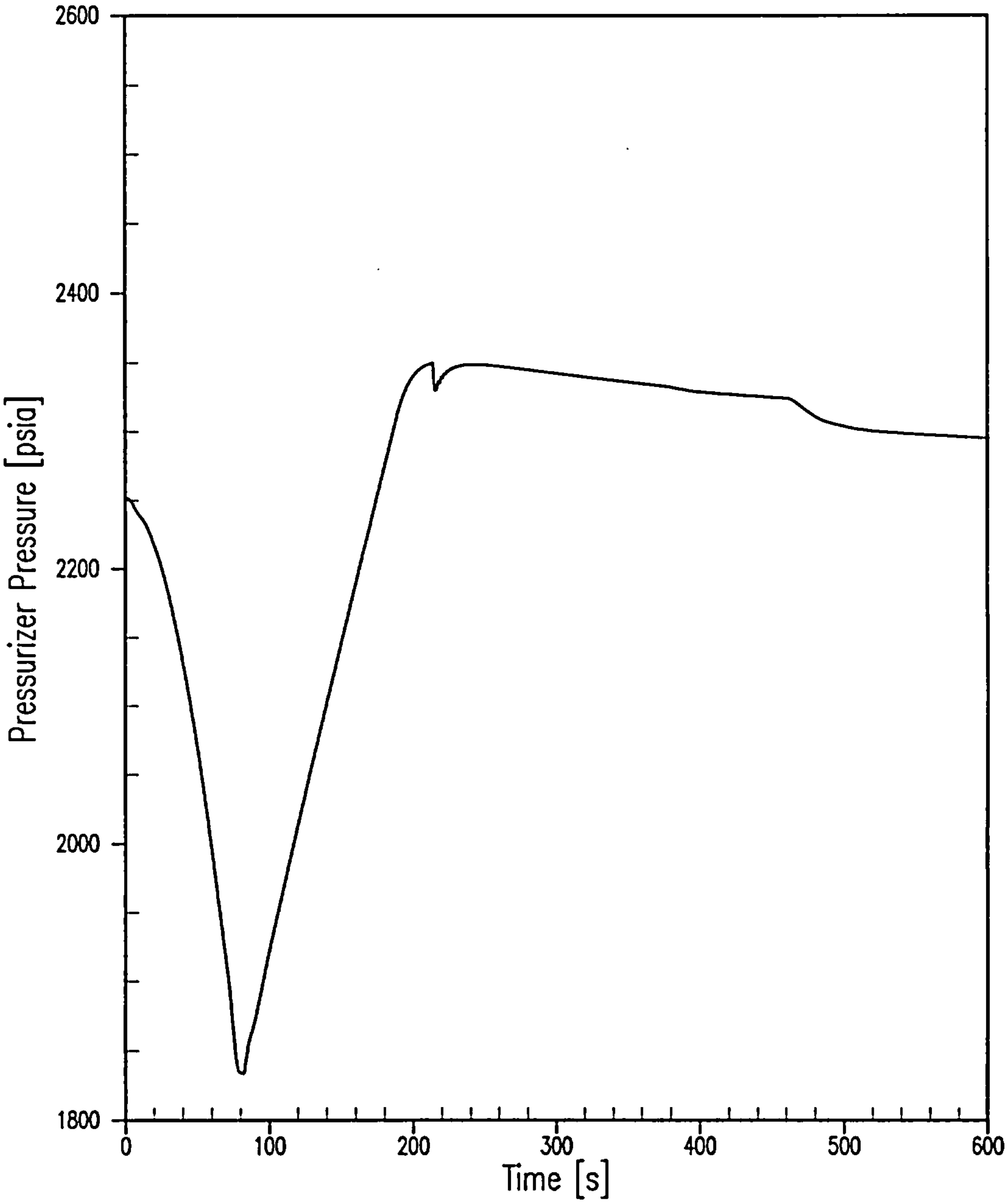


Figure 6.3.14-3 Inadvertent ECCS Actuation at Power (DNBR Case) – Pressurizer Pressure versus Time

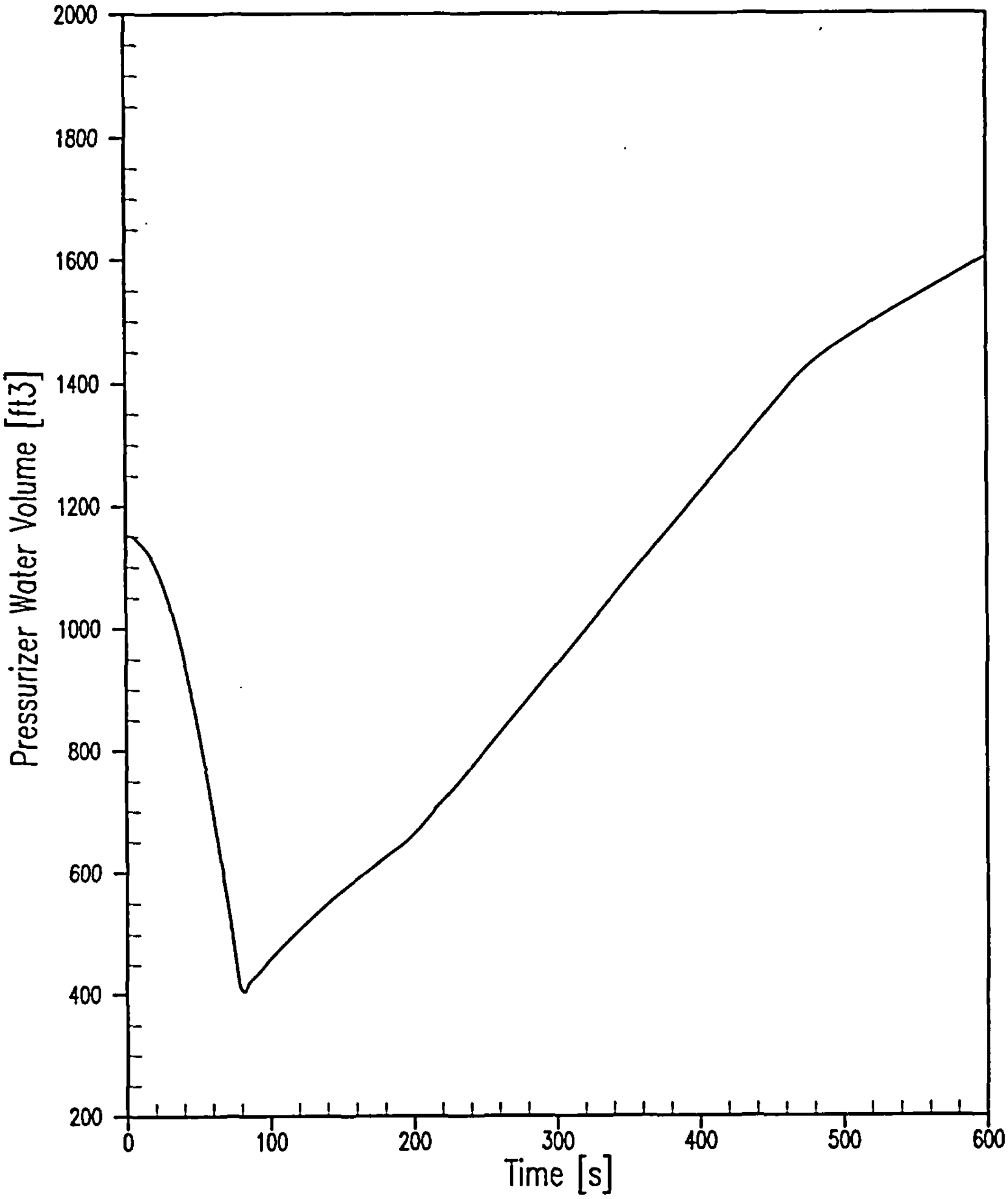


Figure 6.3.14-4 Inadvertent ECCS Actuation at Power (DNBR Case) – Pressurizer Water Volume versus Time

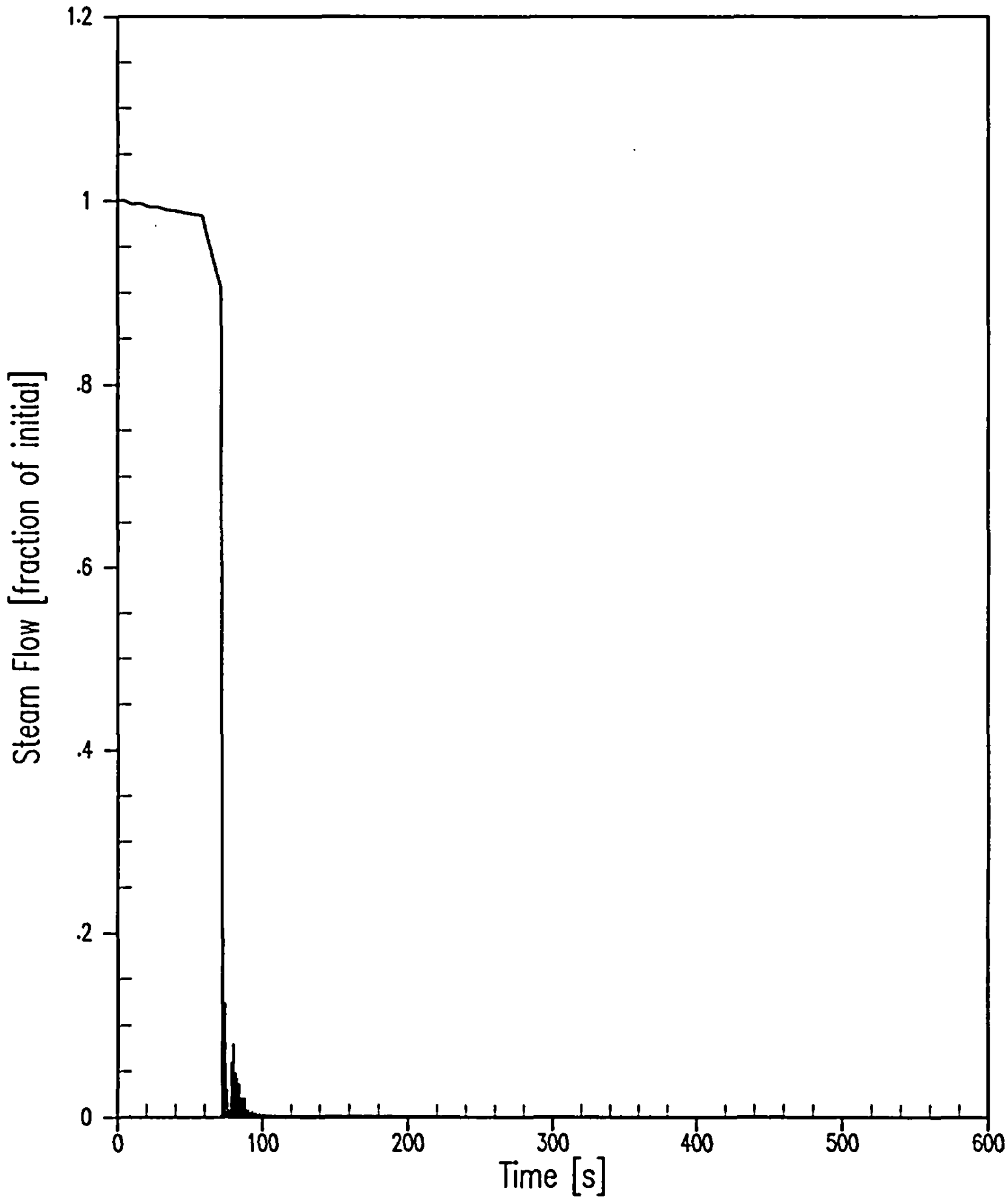


Figure 6.3.14-5 Inadvertent ECCS Actuation at Power (DNBR Case) – Steam Flow versus Time

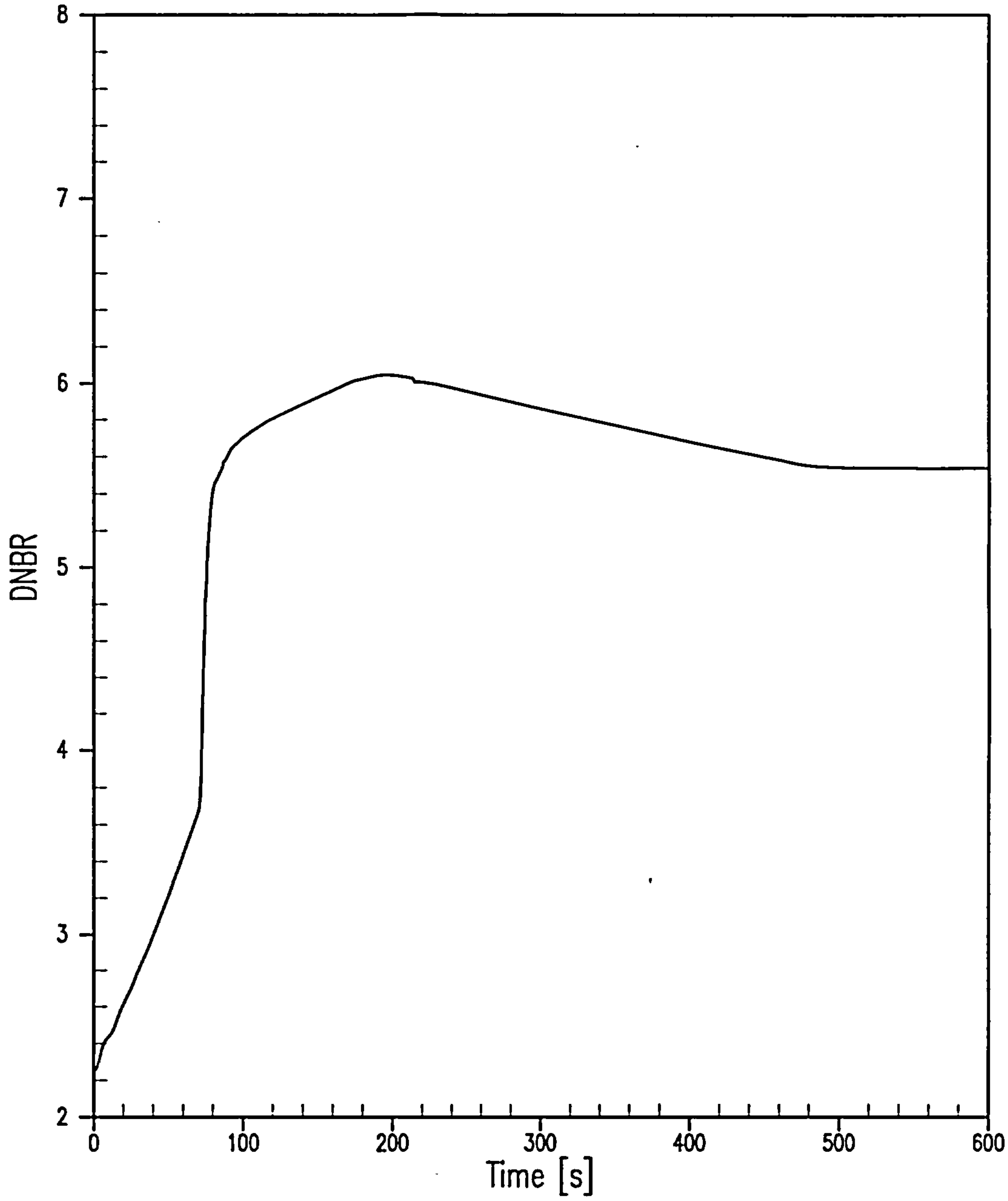


Figure 6.3.14-6 Inadvertent ECCS Actuation at Power (DNBR Case) – DNBR versus Time

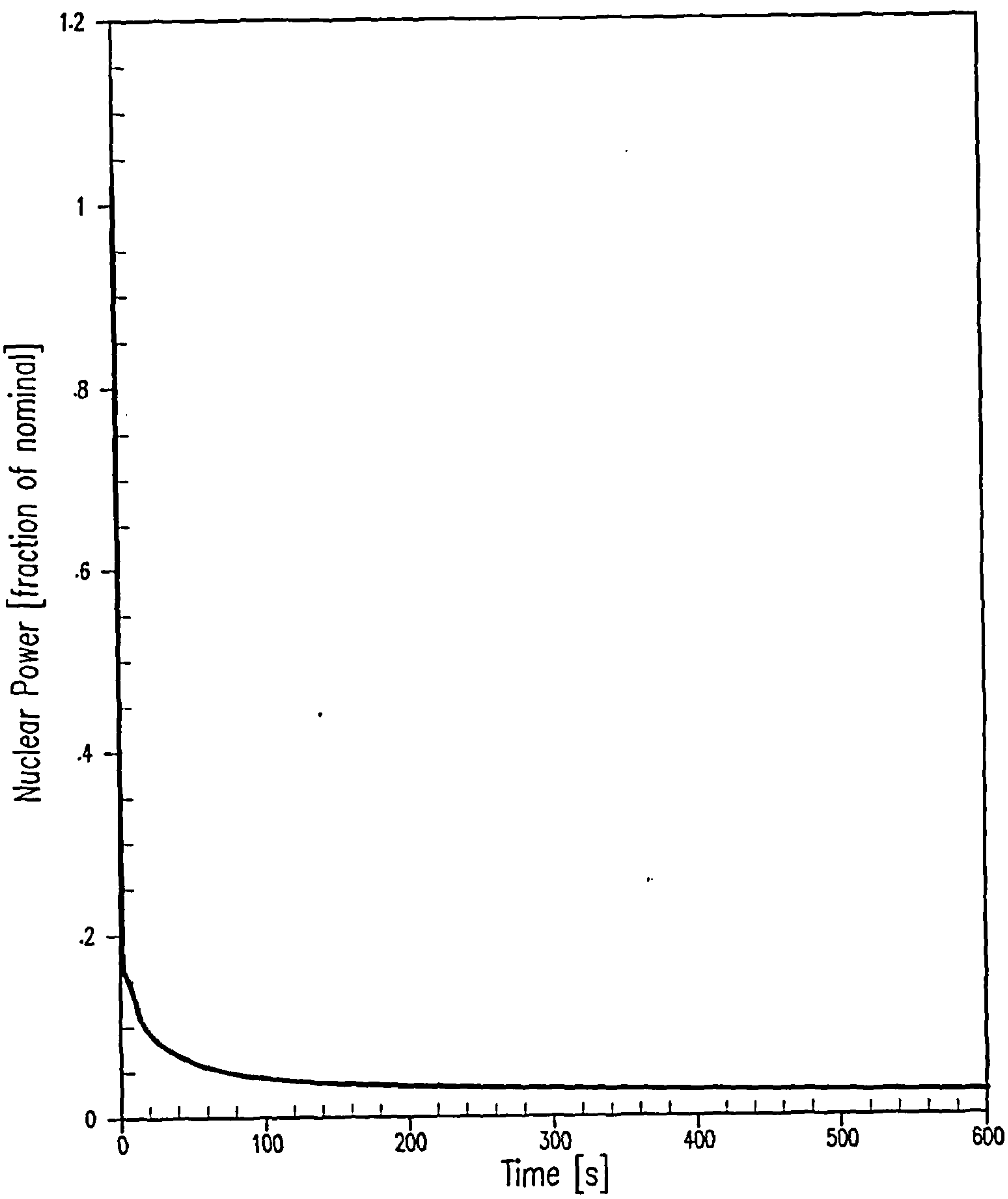


Figure 6.3.14-7 Inadvertent ECCS Actuation at Power (Pressurizer Filling Case) – Nuclear Power versus Time

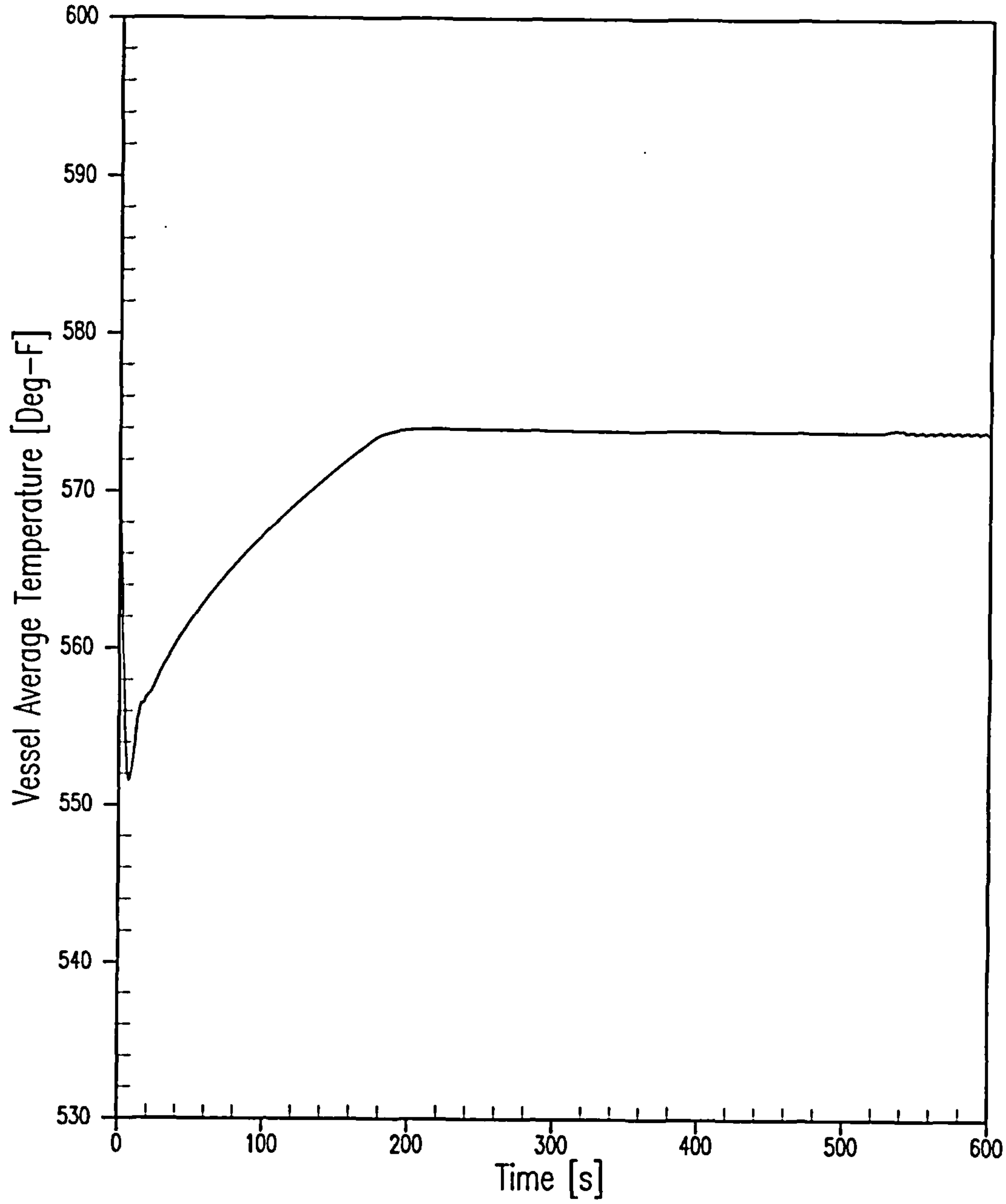


Figure 6.3.14-8 Inadvertent ECCS Actuation at Power (Pressurizer Filling Case) – Vessel Average Temperature versus Time

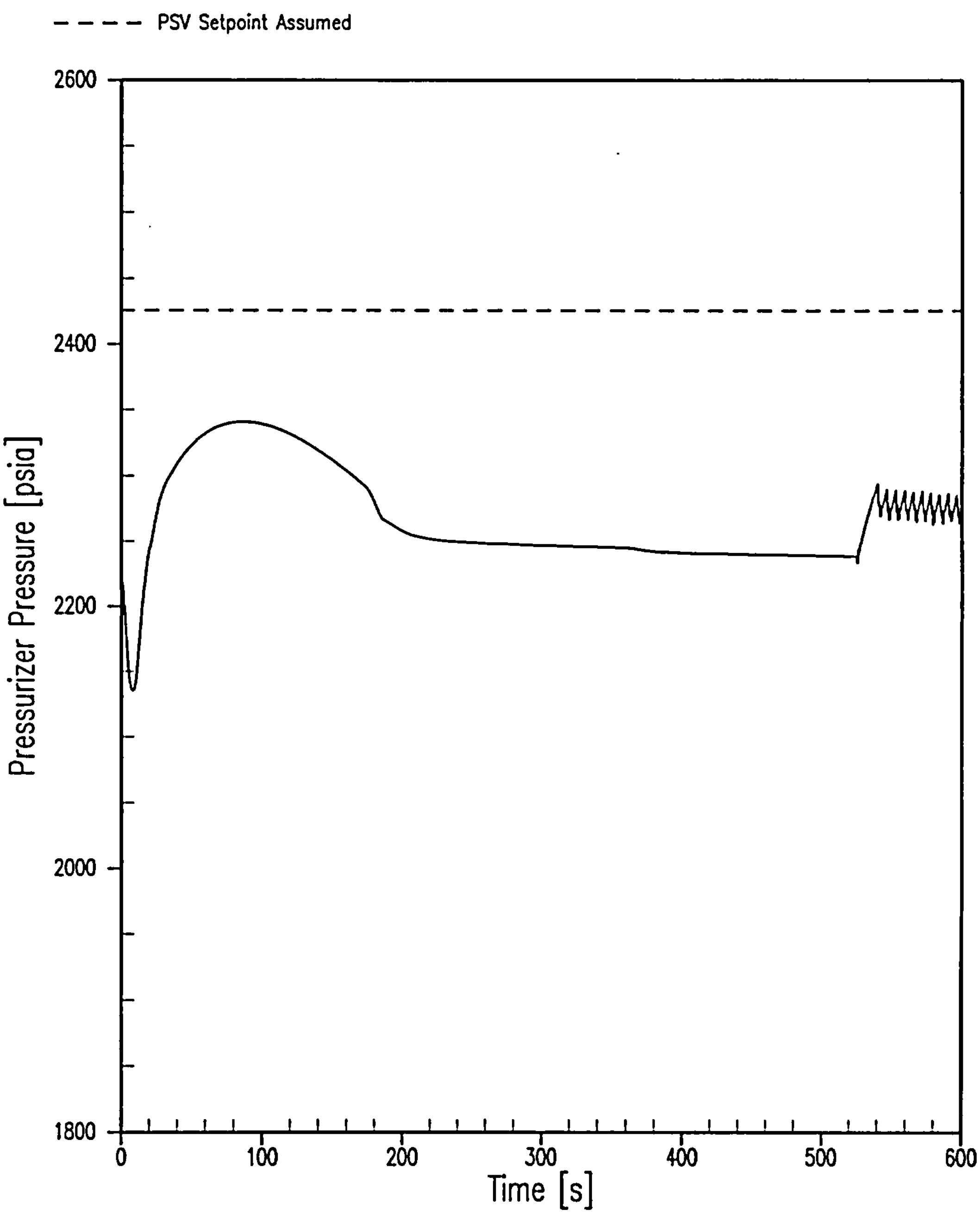


Figure 6.3.14-9 Inadvertent ECCS Actuation at Power (Pressurizer Filling Case) – Pressurizer Pressure versus Time

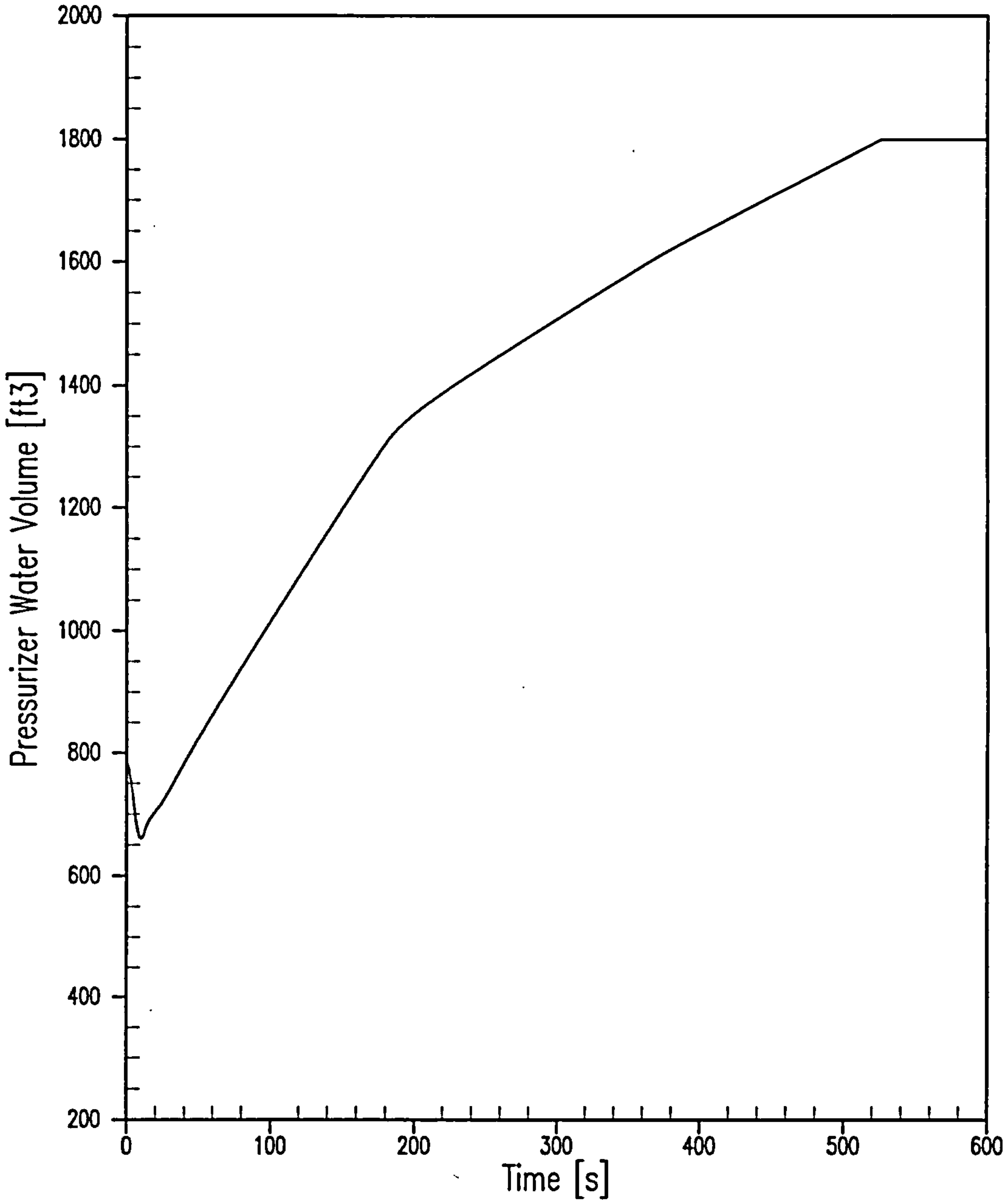


Figure 6.3.14-10 Inadvertent ECCS Actuation at Power (Pressurizer Filling Case) – Pressurizer Water Volume versus Time

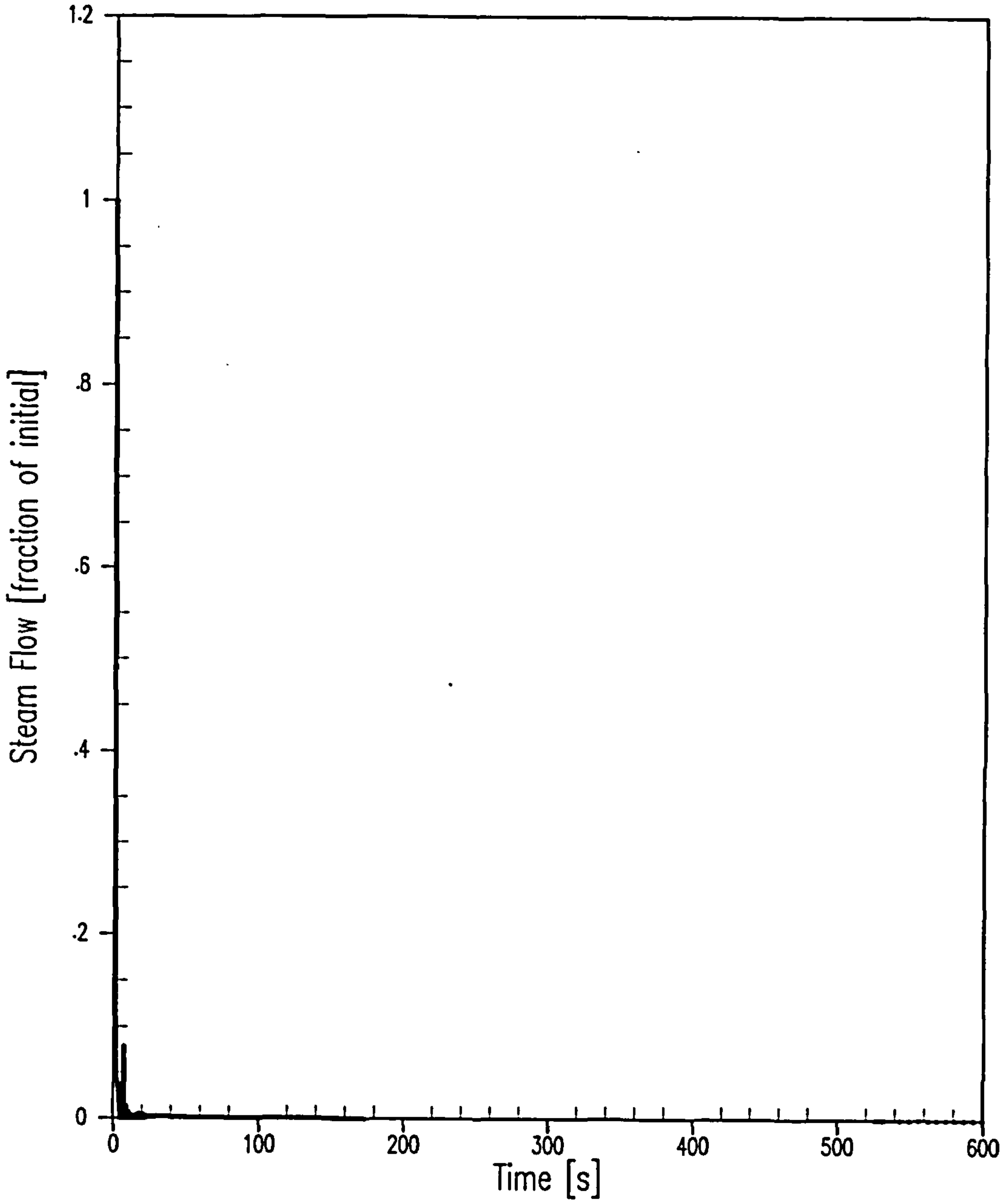


Figure 6.3.14-11 Inadvertent ECCS Actuation at Power (Pressurizer Filling Case) – Steam Flow versus Time

6.3.15 Chemical and Volume Control System Malfunction that Increases Reactor Coolant Inventory (FSAR Section 15.5.2)

At the request of AmerenUE, this event has not been re-analyzed in support of the Callaway Replacement Steam Generator (RSG) Program. The method of analysis and potential consequences are similar to those of the Inadvertent Operation of the Emergency Core Cooling System During Power Operation, discussed in Final Safety Analysis Report (FSAR) Section 15.5.1 and Section 6.3.14 of this report.

6.3.16 Inadvertent Opening of a Pressurizer Safety or Relief Valve (FSAR Section 15.6.1)

6.3.16.1 Accident Description

An accidental depressurization of the reactor coolant system (RCS) could occur as a result of an inadvertent opening of a pressurizer relief or safety valve. Since a safety valve is sized to relieve approximately twice the steam flow rate of a relief valve, and will therefore allow a much more rapid depressurization upon opening, the most severe core conditions resulting from an accidental depressurization of the RCS are associated with an inadvertent opening of a pressurizer safety valve. Initially, the event results in a rapidly decreasing RCS pressure until this pressure reaches a value corresponding to the hot leg saturation pressure. At this time, the pressure decrease is slowed considerably. The pressure continues to decrease throughout the transient. The effect of the pressure decrease would be to decrease power via the moderator density feedback, but the reactor control system (if in the automatic mode) functions to maintain the power essentially constant throughout the initial stage of the transient. The average coolant temperature decreases slightly. Pressurizer level increases initially due to expansion caused by depressurization and then decreases following reactor trip.

The reactor may be tripped by the following reactor protection system (RPS) signals:

- Overtemperature ΔT (OTΔT)
- Low pressurizer pressure

An inadvertent opening of a pressurizer relief valve is classified as an American Nuclear Society (ANS) Condition II event, a fault of moderate frequency. The failure of a pressurizer safety valve is classified as an ANS Condition III event. The analysis performed conservatively bounds the more limiting failure, while still applying the more restrictive Condition II acceptance criterion of ensuring the departure from nucleate boiling (DNB) design basis is met.

6.3.16.2 Method of Analysis

The accidental depressurization transient is analyzed by employing the detailed digital computer code RETRAN (Reference 1). The RETRAN computer code is described in detail in Section 6.3.0.6 of this report.

This accident is analyzed with the RTDP, as described in Reference 2. In order to give conservative results in calculating the DNB ratio (DNBR) during the transient, the following assumptions are made:

1. Consistent with Revised Thermal Design Procedure (RTDP) methodology, the initial reactor power, pressure, and RCS temperatures are assumed to be at their nominal values. Uncertainties in initial conditions are included in the limit DNBR.

2. Minimum reactivity feedback is assumed to minimize the resulting DNBR, including a moderator temperature coefficient of 0 pcm/°F and the least-negative Doppler coefficient. The spatial effect of the voids due to local or subcooled boiling is not considered in the analysis with respect to reactivity feedback or core power shape. The voids would tend to flatten the core power distribution.

Normal reactor control systems are not required to function. However, the rod control system is assumed to be in the automatic mode in order to hold the core at full power longer and thus delay the trip. This is a worst-case assumption. If the reactor were in manual control, an earlier trip could occur. Although automatic rod control is not normally used, and automatic rod withdrawal is no longer available, it was still assumed for this analysis. The RPS functions to trip the reactor on the appropriate signal. No single active failure will prevent the RPS from functioning properly.

6.3.16.3 Results

The system response to an inadvertent opening of a pressurizer safety valve is shown in Figures 6.3.16-1 and 6.3.16-2. Figure 6.3.16-1 illustrates the nuclear power and DNBR transients following the depressurization. Nuclear power is maintained at the initial value until reactor trip occurs on OTΔT. The pressure decay transient and average temperature transient following the accident are given in Figure 6.3.16-2.

Pressure drops more rapidly while core heat generation is reduced via the trip, and would then slow once saturation temperature is reached in the hot leg. The DNBR decreases initially, but increases rapidly following the trip, as shown in Figure 6.3.16-1.

The DNBR remains above the safety analysis limit values throughout the transient. The calculated sequence of events for the inadvertent opening of a pressurizer safety valve incident is shown in Table 6.3.16-1.

6.3.16.4 Conclusions

The results of the analysis show that the low pressurizer pressure and the OTΔT RPS signals provide adequate protection against the RCS depressurization event. No fuel or cladding damage is predicted for this accident.

Table 6.3.16-1 Time Sequence of Events for Inadvertent Opening of a Pressurizer Safety or Relief Valve	
Event	**Time (Seconds)**
Inadvertent Opening of Pressurizer Safety or Relief Valve	0.0
OTΔT Reactor Trip Setpoint Reached	27.8
Rod Motion Begins	29.8
Minimum DNBR Occurs	30.5
Results	
Minimum DNBR Value	1.859
DNBR Limit	1.55

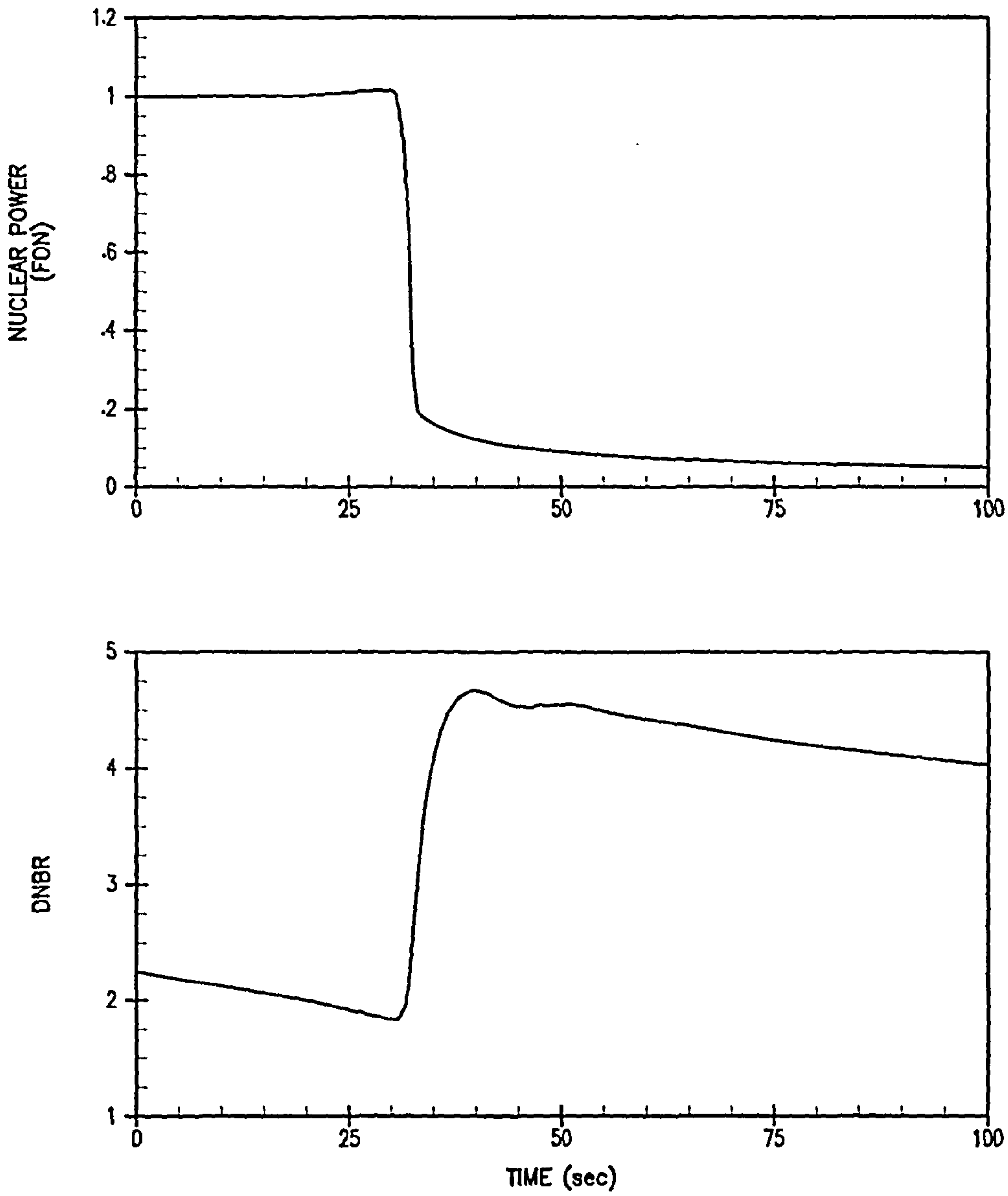


Figure 6.3.16-1 Inadvertent Opening of a Pressurizer Safety or Relief Valve – Nuclear Power and DNBR versus Time

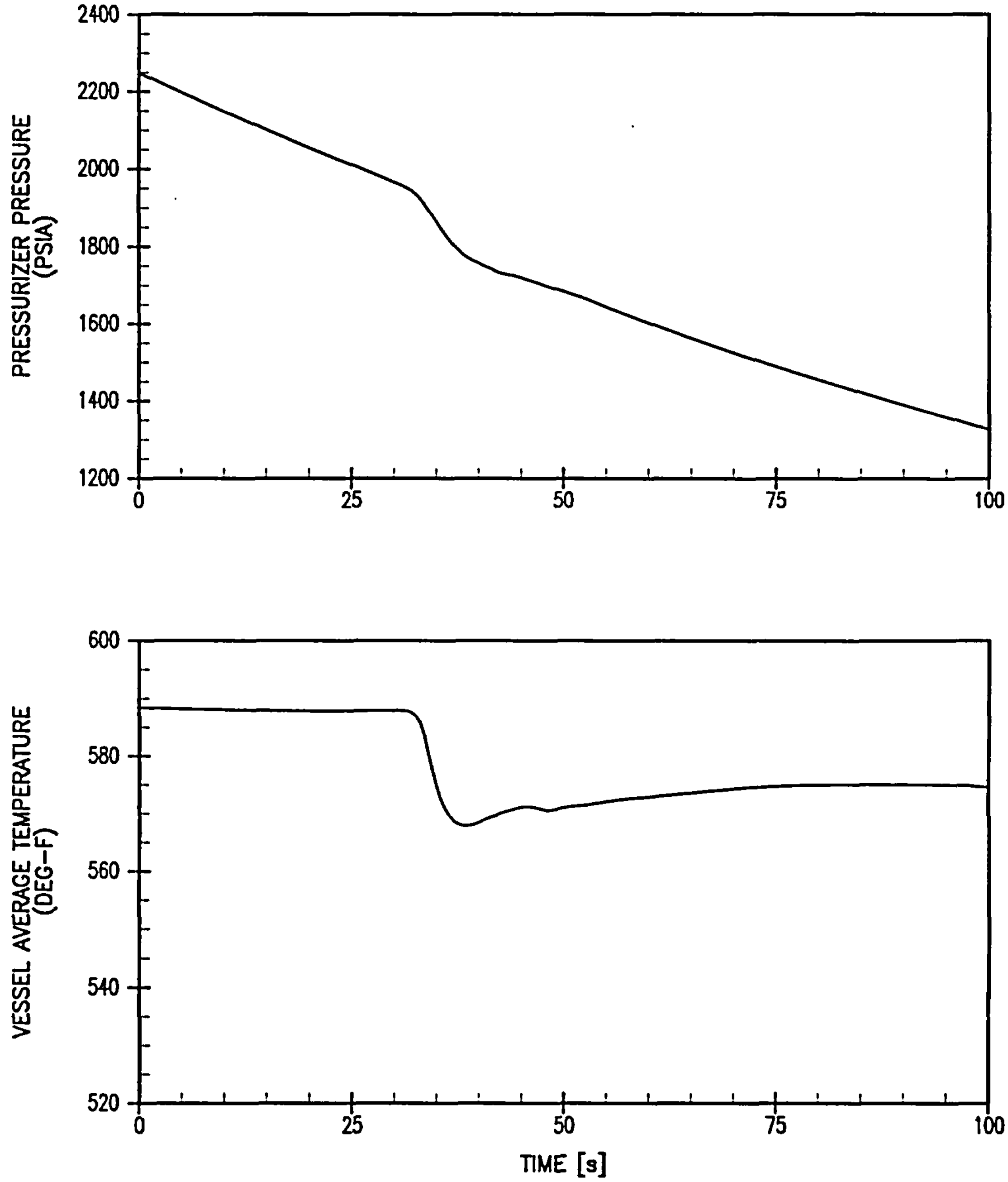


Figure 6.3.16-2 **Inadvertent Opening of a Pressurizer Safety or Relief Valve – Pressurizer Pressure and Vessel Average Temperature versus Time**

6.3.17 Anticipated Transients Without Scram (FSAR Section 15.8)

6.3.17.1 Accident Description

An anticipated transient without scram (ATWS) is defined as an anticipated operational occurrence (such as a loss of normal feedwater, loss of condenser vacuum, or loss of offsite power) combined with an assumed failure of the reactor trip system to shut down the reactor.

The final ATWS rule, the Code of Federal Regulations (CFR) 10 CFR 50.62(c) (Reference 14), requires Westinghouse designed pressurized water reactors (PWRs) (such as the Callaway plant) to incorporate an actuation device that is diverse from the reactor trip system, to automatically initiate the auxiliary feedwater system, and initiate a turbine trip for conditions indicative of an ATWS. The installation of the U.S. NRC approved ATWS Mitigating System Actuation Circuitry (AMSAC) satisfies this final ATWS rule. AMSAC has been installed at Callaway. Therefore, the requirements of 10 CFR 50.62 have been satisfied.

6.3.17.2 Method of Analysis

The basis for the final ATWS rule and the AMSAC design are supported by Westinghouse reference analyses reported in NS-TMA-2182 (Reference 15). These analyses were performed based on guidelines published in NUREG-0460 (1978) (Reference 16). Appendix A of WASH-1270 (Reference 17) states that in evaluating the reactor coolant system (RCS) boundary for ATWS events, "the calculated reactor coolant system transient pressure should be limited such that the maximum primary stress anywhere in the system boundary is less than that of the "emergency conditions" as defined in the ASME Nuclear Power Plant Components Code, Section III" (currently termed Service Limit C). Based on a review of reactor vessels for 2- 3- and 4-loop plants, the maximum allowable pressure for the reactor vessel is 3,200 psig. This value corresponds to the maximum allowable pressure for the weakest component in the reactor pressure vessel (the nozzle safe end). Therefore, the Reference 15 analyses were performed to demonstrate that the RCS pressure did not exceed 3200 psig (3215 psia). Reference 15 describes the methods used in the analyses and provides reference analyses for 2-loop, 3-loop, and 4-loop plant designs with several different steam generator models available in plants at the time.

The loss of load/turbine trip (LOL/TT) and loss of normal feedwater (LONF) ATWS events are the two most limiting RCS overpressure transients reported in Reference 15. To address the installation of RSGs at Callaway, these two events were re-analyzed to ensure that the analytical basis for the final ATWS rule continues to be met.

Although it is assumed that the AmerenUE will maintain and operate the AMSAC consistent with its design and as approved by the U.S. NRC, a conservative analysis approach was taken. The primary input to the LOL/TT and LONF ATWS analysis is the 4-loop reference LOL/TT and LONF ATWS models from the analysis supporting NS-TMA-2182. The nominal and initial conditions were made consistent with the current NSSS power of 3579 MWt, and the steam generator data was revised to reflect the Framatome Model 73/19T steam generators.

6.3.17.3 Results and Conclusions

To remain consistent with the basis of the final ATWS rule (10 CFR 50.62), the peak RCS pressure calculated in both the LOL/TT and the LONF ATWS analyses shall be less than 3,215 psia. The peak RCS pressure obtained for the LOL/TT and LONF ATWS analyses with Model 73/19T steam generators is 3,177 psia and 2,973 psia, respectively. Therefore, it has been demonstrated that the analytical basis for the final ATWS rule continues to be met for operation of Callaway with Framatome Model 73/19T steam generators.

6.3.18 Non-LOCA Transients Conclusions

All non-loss-of-coolant (non-LOCA) Final Safety Analysis Report (FSAR) Chapter 15 events have been analyzed or evaluated in support of the Callaway Replacement Steam Generator (RSG) Program and the transition to the Westinghouse RETRAN methodology. Detailed description of each event-specific analysis or evaluation performed is presented in Sections 6.3.1 through 6.3.17 of this report. A summary of the limiting results obtained for all non-LOCA events was presented in Table 6.3-2. These results demonstrate that all applicable acceptance criteria are met. Based on this, it can therefore be concluded that the implementation of the RSG Program at Callaway is acceptable with respect to the non-LOCA FSAR Chapter 15 safety analyses.

6.3.19 References

1. WCAP-14882-P-A (Proprietary), "RETRAN-02 Modeling and Qualification for Westinghouse Pressurized Water Reactor Non-LOCA Safety Analyses," D. S. Huegel, et al., April 1999.

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

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

4. WCAP-12910, Rev. 1-A (Proprietary), "Pressurizer Safety Valve Set Pressure Shift," G. O. Barrett, et al., May 1993.

5. WCAP-14565-P-A (Proprietary), "VIPRE-01 Modeling and Qualification for Pressurized Water Reactor Non-LOCA Thermal-Hydraulic Safety Analysis," Y. X. Sung, et al., October 1999.

6. WCAP-9226-P-A, Rev. 1 (Proprietary) and WCAP-9227, Rev. 1 (Non-Proprietary), "Reactor Core Response to Excessive Secondary Steam Releases," S. D. Hollingsworth and D. C. Wood, January 1978 and February 1998, respectively.

7. ANSI/ANS-5.1-1979, "Decay Heat Power in Light Water Reactors," August 29, 1979.

8. Nuclear Regulatory Guide 1.70, Rev. 3, "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants," November 1978.

9. NUREG-0800, Rev. 1, "Standard Review Plan" Sections 15.5.1 - 15.5.2, "Inadvertent Operation of ECCS and Chemical Volume Control System Malfunction that Increases Reactor Coolant Inventory," July 1981.

10. WCAP-8745-P-A (Proprietary), "Design Bases for the Thermal Overpower ΔT and Thermal Overtemperature ΔT Trip Functions," S. L. Ellenberger, et al., September 1986.

11. WCAP-9272-P-A (Proprietary), "Westinghouse Reload Safety Evaluation Methodology," F. M. Bordelon, et al., July 1985.

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

13. WCAP-10965-P-A, "ANC: A Westinghouse Advanced Nodal Computer Code," Y. S. Liu, et al., September 1986.

14. U.S. Federal Government 10 CFR 50.62, ATWS Final Rule.

15. Westinghouse Letter NS-TMA-2182, "Anticipated Transients Without Scram for Westinghouse Plants," December 1979.

16. NRC Staff Report NUREG-0460, Volume 3, "Anticipated Transients Without Scram for Light Water Reactors," December 1978.

17. USNRC, WASH-1270, "Technical Report on Anticipated Transients Without Scram for Water-Cooled Power Reactors," September 1973.

6.4 STEAM GENERATOR TUBE RUPTURE TRANSIENT

In support of the Callaway Replacement Steam Generator (RSG) Program, an evaluation for a design-basis steam generator tube rupture (SGTR) event has been performed to demonstrate that the potential consequences are acceptable. The analysis supports a full-power average temperature (Tavg) operating window of 570.7° to 588.4°F as well as a main feedwater temperature window of 390° to 446°F. The evaluation discussed herein assumes that up to 5 percent of the steam generator tubes are plugged.

The major hazard associated with an SGTR event is the radiological consequences resulting from the transfer of radioactive reactor coolant to the secondary side of the ruptured steam generator and subsequent release of radioactivity to the atmosphere. Therefore, an analysis must be performed to ensure that the offsite radiation doses resulting from an SGTR are within the allowable guidelines. Two SGTR scenarios have been identified that result in the most limiting radioactive releases to the environment. Detailed analyses are presented for the following two scenarios:

1. SGTR with postulated stuck-open atmospheric relief valve (ARV) on the ruptured steam generator.

   In the stuck-open ARV scenario, the radioactive releases are maximized by assuming the ruptured steam generator ARV is stuck open for 20 minutes. This analysis is presented in Section 6.4.1

2. SGTR with postulated failure of the ruptured steam generator auxiliary feedwater (AFW) flow control valve.

   In the failed control valve scenario, AFW flow is maximized in order to increase the probability for the ruptured steam generator to overfill and to maximize subsequent water relief from its safety valve. The radioactive releases are maximized by assuming that the safety valve is stuck open following water relief with an effective flow area equal to 5 percent of the total safety valve flow area. This analysis is presented in Section 6.4.2.

Plant response to the event was modeled using the RETRAN computer code. The analysis methodology includes the simulation of the operator actions for recovery from an SGTR based on the Callaway Emergency Operating Procedures (EOPs). In the analyses, the primary-to-secondary break flow and the releases to the atmosphere from both the ruptured and intact steam generators were calculated for use in determining the radioactivity released to the atmosphere. The mass release information is used to calculate the radiological consequences of the accident.

6.4.1 SGTR with Stuck-Open Atmospheric Relief Valve

6.4.1.1 Introduction and Background

The analysis of the SGTR with a failed-open ARV on the ruptured steam generator was performed for the RSG Program in a manner consistent with that documented in Reference 1 for operation with the original steam generators and currently reported in the Callaway Final Safety Analysis Report (FSAR). The current FSAR analysis used the RETRAN02 and the plant model (volumes, junctions, etc.) described in Reference 1. For the RSG analysis, the Westinghouse version of RETRAN02 (Reference 2) is used with the plant model described in Reference 2. The analysis method remains essentially consistent with that in Reference 1, that is, only the code version and plant model follow Reference 2, not the analysis methodology. This section includes the methods and assumptions used to analyze the SGTR event, as well as the sequence of events for the recovery and the calculated results.

6.4.1.2 Input Parameters and Assumptions

The thermal-hydraulic analysis for the case modeling the failure of the ARV on the ruptured steam generator includes assumptions selected (as discussed in Reference 1) to maximize the release of activity with the steam from the ruptured steam generator. This analysis models the plant operating at the higher end of the Tavg window, since a higher operating temperature results in increased steaming from the ruptured steam generator and a higher fraction of the break flow flashing to steam inside the ruptured steam generator. The analysis assumes that no tubes are plugged and that the plant is operating with the feedwater temperature at the higher end of the temperature window, although it was determined that these assumptions have a minimal impact on the results. In addition to the base analysis with these assumptions, sensitivity runs are included to investigate the impact of these operating parameters specifically for the Callaway RSG Program.

Design-Basis Accident

The design-basis accident modeled is a double-ended break of one steam generator tube located at the top of the tubesheet on the inlet (hot leg) side of the steam generator. The location of the break on the hot side of the steam generator results in a higher break flow flashing fraction than a break on the cold side of the steam generator, as determined by Reference 1. However, for the RSG Program, the break flow flashing fraction calculation was conservatively based on assuming that all of the break flow comes from the hot leg side of the steam generator. It was also assumed that a loss of offsite power occurs at the time of reactor trip, and the highest worth control assembly was assumed to be stuck in its fully withdrawn position at reactor trip. After reactor trip and loss of offsite power, the reactor coolant pumps (RCPs) begin to coast down. Due to the assumed loss of offsite power, the condenser is not available for steam releases once the reactor is tripped. Consequently, after reactor trip, steam is released to the atmosphere through the steam generator ARVs.

The single failure considered in this scenario is a failed-open ARV on the ruptured steam generator. Failure of this ARV will cause an uncontrolled depressurization of the steam generator, which will increase primary-to-secondary leakage, the break flow flashing fraction, and the mass release to the atmosphere. Pressure in the ruptured steam generator will remain below that in the primary system until the failed ARV can be isolated and recovery actions completed.

Analysis Assumptions

The primary-to-secondary break flow and the mass releases from the ruptured and intact steam generators to the condenser and to the atmosphere until break flow termination were calculated with the RETRAN program. This section includes a discussion of the methods and assumptions used to analyze the SGTR event and to calculate the mass releases, the sequence of events during the recovery operations, and the calculated results.

The major assumptions used for the RETRAN analysis are the following:

1. The following initial conditions are chosen:

   a. Core power = 3,565 MWt plus 2-percent power uncertainty

   b. Additional net power input to RCS = 14 MWt

   c. Pressurizer pressure = 2,280 psia (nominal pressure of 2,250 psia +30 psia uncertainty)

   d. Initial pressurizer level = 60 percent

   e. Vessel average temperature = 592.7°F. This reflects the high end of the Tavg band of 570.7° – 588.4°F and the uncertainty of -4.3°F. (A sensitivity case modeling a vessel average temperature of 567.7°F, to reflect the low end of the Tavg band and the uncertainty of +3°F, is included since this assumption would maximize the tube rupture mass flow rate, even though it would not otherwise be limiting.)

   f. RCS flow = thermal design flow = 374,400 gpm (93,600 gpm/loop)

   g. Feedwater temperature = 446°F. This is the high end of the feedwater temperature band. (Additional cases are included modeling the lowest feedwater temperature of 390°F, to evaluate the sensitivity to this parameter.)

   h. No steam generator tube plugging is assumed. (Additional cases are included modeling the maximum plugging level of 5 percent, to evaluate the sensitivity to this parameter.)

   i. Steam generator pressure: corresponding to conditions being analyzed

   j. Steam generator level = 43.4-percent narrow-range span (NRS). This is the nominal level of 51.3-percent NRS minus 7.9-percent NRS for uncertainty.

2. Reactor trip and safety injection (SI) occur coincidentally as a result of low pressurizer pressure. Overtemperature ΔT trip is not considered. This allows more break flow. Loss of offsite power occurs at reactor trip.

3. The tube rupture is a double-ended guillotine break of a single steam generator tube at the tubesheet on the inlet (hot leg) side of the steam generator. This break location maximizes the flashed fraction of the reactor coolant system (RCS) break flow. In the analysis, the break flow flashing fraction calculation was conservatively based on assuming that all of the break flow comes from the hot leg side of the steam generator.

4. The low pressurizer pressure safety analysis limit (SAL) for reactor trip is 1,845 psig. This reactor trip SAL is lower than the actual setpoint, which thereby delays the reactor trip and results in increased break flow. Safety injection is assumed concurrent with reactor trip, which decreases the time for initiation of safety injection. This also results in increased break flow. Safety

injection occurs 15 seconds after the SI signal. This minimum expected delay results in an early rise in RCS pressure due to SI and results in increased break flow.

5. Break flow is characterized by resistance-limited flow. An additional 5-percent uncertainty is added to the flow.

6. The assumption of a loss of offsite power at reactor trip prevents steam dump to the condenser resulting in steam discharge to the atmosphere via the ARVs. With the condenser unavailable for retention of any leaked radioactivity, offsite doses are maximized.

7. Pressurizer heaters and spray are not modeled.

8. Prior to reactor trip, feedwater is controlled to maintain the steam generator secondary-side level. In the intact steam generators, the normal feedwater matches the steam flow. For the ruptured steam generator, the feedwater flow is reduced such that the total flow into the steam generator secondary side (feed flow and including the break flow) matches the steam flow. Feedwater isolation is completed 4.3 seconds after reactor trip/SI.

9. Auxiliary feedwater flow is maintained to achieve a narrow-range level of at least 50 percent in the intact steam generators, but flow to the ruptured steam generator is isolated when the level rises above 4 percent. Auxiliary feedwater is initiated 60 seconds after reactor trip and attains a flow rate of 250 gpm to all steam generators. This maximum expected delay for AFW initiation maximizes break flow and maintains high RCS temperatures. This minimum expected AFW flow to the ruptured steam generator results in decreased RCS heat removal, maintains high RCS temperatures, and thereby maximizes the flashed fraction of leaked reactor coolant.

10. The ruptured steam generator's ARV is set at 1,184.7 psia. This is 4-percent higher than the nominal setpoint, which delays the release of pressure from the ruptured steam generator. This results in increased valve discharge flow and integrated break flow. The ARV on the ruptured steam generator fails open for 20 minutes, beginning on initial demand, shortly after reactor trip. This is the single failure considered in this scenario to maximize offsite doses.

11. The decay heat is modeled using the 1979 American Nuclear Society (ANS) 2σ model. This maximizes the heat to be transferred, which increases the break flow flashing fraction.

12. The ruptured steam generator ARV is isolated at 20 minutes after the valve is assumed to fail open.

13. The RCS cooldown is initiated 10 minutes after the failed ARV is isolated. The narrow-range level in all steam generators must be greater than 4 percent and the ruptured steam generator pressure must be greater than 430 psig prior to initiating RCS cooldown.

14. The RCS depressurization is assumed to begin 3 minutes after completion of cooldown. When the ruptured steam generator pressure is higher than the RCS pressure, the pressurizer power-operated relief valves (PORVs) are closed.

15. Safety injection is terminated 5 minutes after completion of RCS depressurization.

16. A second depressurization to terminate break flow is completed approximately 15 minutes after SI termination.

17. When calculating the fraction of break flow that flashes to steam, 100 percent of the break flow is assumed to come from the hot leg side of the break. Since the tube rupture flow actually consists of flow from the hot leg and cold leg sides of the steam generator, the temperature of the combined flow will be less than the hot leg temperature and the flashing fraction will be correspondingly lower. Thus the assumption is conservative for an SGTR analysis.

Operator Actions

In the event of an SGTR, the operator is required to take actions to stabilize the plant and terminate the primary-to-secondary leakage. The operator actions for SGTR recovery are provided in the Callaway Plant EOPs, and major actions were explicitly modeled in this analysis. The operator actions modeled include identification and isolation of the ruptured steam generator, cooldown, and depressurization of the RCS to restore inventory, and termination of SI to stop primary-to-secondary leakage. These operator actions are described below.

1. Identify the ruptured steam generator.

   High secondary-side activity, as indicated by the condenser air discharge radiation monitor, steam generator blowdown liquid monitor, or main steam line radiation monitor, typically will provide the first indication of an SGTR event. The ruptured steam generator can be identified by a mismatch between steam and feedwater flow, high activity in a steam generator water sample, or a high radiation indication on the corresponding main steam line radiation monitor. For an SGTR that results in a reactor trip at high power as assumed in this analysis, the steam generator water level as indicated on the narrow range will decrease significantly for all of the steam generators. The AFW flow will begin to refill the steam generators, distributing approximately equal flow to each of the steam generators. Since primary-to-secondary leakage adds additional inventory to the ruptured steam generator, the water level will increase more rapidly in the ruptured steam generator. This response, as displayed by the steam generator water level instrumentation, provides confirmation of an SGTR event and also identifies the ruptured steam generator.

2. Isolate the ruptured steam generator from the intact steam generators and isolate feedwater to the ruptured steam generator.

   Once the steam generator with a tube rupture has been identified, recovery actions begin by isolating steam flow from and stopping feedwater flow to the ruptured steam generator. In addition to minimizing radiological releases, this also reduces the possibility of filling the ruptured steam generator by (1) minimizing the accumulation of feedwater flow and (2) enabling the operator to establish a pressure differential between the ruptured and intact steam generators as a necessary step toward terminating primary-to-secondary leakage. In the Callaway EOP for SGTR, the operator is directed to maintain the level in the ruptured steam generator between 4 and 50 percent on the narrow-range instrument. For the analysis with the failed-open ARV on the ruptured steam generator, it was assumed that AFW flow to the ruptured steam generator would be isolated when level in the steam generator reached 4-percent narrow-range level. Isolation of the steam lines connecting the ruptured and intact steam generators to the header is modeled just after reactor trip in the analysis. This is conservative since it allows the ruptured steam generator to depressurize further and more rapidly than if it was not isolated from the other steam generators. This depressurization maximizes the break flow rate and the break flow flashing fraction.

3. Cool down the RCS using the intact steam generators.

After isolation of the ruptured steam generator, the RCS is cooled as rapidly as possible to less than the saturation temperature corresponding to the ruptured steam generator pressure by dumping steam from the intact steam generators. This ensures adequate subcooling in the RCS after depressurization to the ruptured steam generator pressure in subsequent actions. If offsite power is available, the normal steam dump system to the condenser can be used to perform this cooldown. However, if offsite power is lost, the RCS is cooled using the ARVs on the intact steam generators. Since offsite power is assumed to be lost at reactor trip for this analysis, the cooldown was performed by dumping steam via the ARVs on the intact steam generators.

4. Depressurize the RCS to restore reactor coolant inventory.

When the cooldown is completed, SI flow will tend to increase the RCS pressure until break flow matches SI flow. Consequently, SI flow must be terminated to stop primary-to-secondary leakage. However, adequate reactor coolant inventory must first be assured. This includes both sufficient reactor coolant subcooling and pressurizer inventory to maintain a reliable pressurizer level indication after SI flow is stopped. Since leakage from the primary side will continue after SI flow is stopped until the RCS and ruptured steam generator pressures equalize, an "excess" amount of inventory is needed to ensure the pressurizer level remains on span. The "excess" amount required depends on the RCS pressure and reduces to zero when the RCS pressure equals the pressure in the ruptured steam generator.

The RCS depressurization is performed using normal pressurizer spray if the RCPs are running. Since offsite power is assumed to be lost at the time of reactor trip, the RCPs are not running and thus normal pressurizer spray is not available. Therefore, the depressurization is modeled using a pressurizer PORV.

5. Terminate SI to stop primary-to-secondary leakage.

The previous actions will have established adequate RCS subcooling, a secondary-side heat sink, and sufficient reactor coolant inventory to ensure that the SI flow is no longer needed. When these actions have been completed, the SI flow must be stopped to terminate primary-to-secondary leakage. Primary-to-secondary leakage will continue after the SI flow is stopped until the RCS and ruptured steam generator pressures equalize. Charging flow, letdown, and pressurizer heaters will then be controlled to prevent re-pressurization of the RCS and re-initiation of leakage into the ruptured steam generator. The analysis assumes that a second depressurization is performed using the pressurizer PORVs to assure that break flow is terminated.

Since these major recovery actions will be modeled in the SGTR analysis, it is necessary to establish the times required to perform these actions. Although the intermediate steps between the major actions will not be explicitly modeled, it is also necessary to account for the time required to perform the steps. It is noted that the total time required to complete the recovery operations consists of both operator action time and system, or plant, response time. For instance, the time for each of the major recovery operations (such as, RCS cooldown) is primarily due to the time required for the system response, whereas the operator action time is reflected by the time required for the operator to perform the intermediate action steps.

The operator action times were determined by AmerenUE. The operator actions and the corresponding operator action times used in this analysis are listed in Table 6.4-1.

For this analysis scenario, the ARV on the ruptured steam generator was assumed to fail wide open, when it first opens just after reactor trip. Before proceeding with the recovery operations, the failed-open PORV on the ruptured steam generator is assumed to be isolated by locally closing the associated block valve. AmerenUE has determined that an operator can locally close the block valve for the ARV on the ruptured steam generator within 20 minutes after the failure. Thus, it was assumed that the ruptured steam generator ARV is isolated at 20 minutes after the valve is assumed to fail open. After the ruptured steam generator ARV is isolated, the additional delay time of 10 minutes (Table 6.4-1) was assumed for the operator action time to initiate the RCS cooldown.

Mass Releases

The mass releases were determined for use in evaluating the offsite and control room radiological consequences of the SGTR. The steam releases from the ruptured and intact steam generators, primary-to-secondary break flow into the ruptured steam generator, and the break flow flashing fraction were calculated by the RETRAN program for the period from accident initiation until break flow termination. Consistent with the analysis of record, simplifying assumptions are used as input to the dose analysis to bound the RETRAN results.

6.4.1.3 Description of Analyses and Evaluations

The RETRAN analysis results for the Callaway analysis of the SGTR with a failed-open ARV on the ruptured steam generator are described below. The sequence of events for the analysis is presented in Table 6.4-2. The results are first presented for the base case modeling operation at the high end of the vessel average temperature band plus uncertainty (592.7°F), the high end of the feedwater temperature band (446°F), and no steam generator tube plugging. The results of cases run to evaluate the sensitivity to the assumed feedwater temperature, tube plugging, and vessel average temperature are then compared to the base case. All cases are considered in the development of data for input to the radiological consequences analysis to assure that it is conservative.

Following the tube rupture, reactor coolant flows from the primary into the secondary side of the ruptured steam generator since the primary pressure is greater than the steam generator pressure. In response to this loss of reactor coolant, the pressurizer water volume decreases as shown in Figure 6.4-1. The RCS pressure also decreases as shown in Figure 6.4-2 as the steam bubble in the pressurizer expands. As the RCS pressure decreases due to the continued primary-to-secondary leakage, automatic reactor trip occurs on a low pressurizer pressure signal at 597 seconds. Safety injection actuation is assumed to occur at the same setpoint as reactor trip, resulting in main feedwater isolation and auxiliary feedwater initiation. As shown in Figure 6.4-3, the RCS mass drops until SI actuation and then increases as SI flow replaces the mass transferred through the ruptured tube.

After reactor trip, core power rapidly decreases to decay heat levels. Since offsite power is assumed lost at reactor trip, the RCPs trip and a gradual transition to natural circulation flow occurs. The turbine stop valves close and steam flow to the turbine is terminated. The analysis assumes that the steam lines are isolated as well. The steam dump system is designed to actuate following reactor trip to limit the increase in secondary pressure, but the steam dump valves remain closed due to the loss of condenser vacuum resulting from the assumed loss of offsite power at the time of reactor trip. Thus, the energy transfer from the primary system causes the secondary-side pressure to increase rapidly after reactor trip, as shown in Figure 6.4-2, until the steam generators' ARVs (and safety valves if their setpoints are reached) lift to dissipate the energy. Steam flow to the atmosphere from the ruptured and intact steam generators is shown in Figures 6.4-4 and 6.4-5.

The ruptured steam generator ARV is assumed to fail open at 604 seconds, when it first lifts. The failure causes the steam generator to rapidly depressurize, which results in an increase in primary-to-secondary leakage. The assumption of steam line isolation at reactor trip prevents the intact steam generators from depressurizing, and maximizes the depressurization of the ruptured steam generator. The depressurization of the ruptured steam generator increases the break flow and energy transfer from primary to secondary, which results in the RCS pressure and temperature decreasing more rapidly. The ruptured steam generator depressurization causes a cooldown in the intact steam generators loops. It is assumed that the time required for the operator to identify that the ruptured steam generator ARV is open and to locally close the associated block valve is 20 minutes. At 1,804 seconds, the depressurization of the ruptured steam generator is terminated and the ruptured steam generator pressure begins to increase as shown in Figure 6.4-2.

As shown in Figures 6.4-6 and 6.4-7, the ruptured steam generator secondary mass and volume remain constant until reactor trip. The flow out the failed-open ARV reduces the mass initially, but as the pressure in the ruptured steam generator drops, the ARV flow rate is reduced, while the break flow rate is increased. This causes the secondary mass to increase even before the ARV is isolated. Once the ARV is isolated, the break flow fills the ruptured steam generator more rapidly.

Major Operator Actions

1. Identify and isolate the ruptured steam generator.

   Recovery actions begin by throttling the AFW flow to the ruptured steam generator and isolating steam flow from the ruptured steam generator. As indicated previously, the steam flow from the ruptured steam generator to the header is isolated just after reactor trip, steam flow out the failed-open ARV is isolated 20 minutes after the valve fails open, and isolation of the AFW flow to the ruptured steam generator was assumed to be completed when the narrow-range level reached 4 percent. In this analysis, the ruptured steam generator level reaches 4-percent span within 10 minutes of isolating the failed-open ARV.

2. Cool down the RCS to establish subcooling margin.

After the block valve for the ruptured steam generator ARV is closed, there is a 10-minute operator action time imposed prior to initiation of cooldown. The depressurization of the ruptured steam generator due to the failed-open ARV affects the RCS cooldown target temperature since the temperature is determined based upon the pressure in the ruptured steam generator. Since offsite power is lost, the RCS is cooled by dumping steam to the atmosphere using the intact steam generators' ARVs. The cooldown is continued until the RCS subcooling at the ruptured steam generator pressure is 20°F plus an allowance for instrument uncertainty. The cooldown would not be initiated until all steam generators indicate greater than 4-percent level and the ruptured steam generator pressure is greater than 430 psig. Since both of these criteria are met within 10 minutes of isolating the failed ARV, the cooldown is initiated after 10 minutes. The cooldown begins at 2,404 seconds, using all 3 intact steam generator's ARVs, and is completed at 2,980 seconds.

The reduction in the intact steam generators' pressure required to accomplish the cooldown is shown in Figure 6.4-2 and the effect of the cooldown on the RCS temperatures is shown in Figures 6.4-8 and 6.4-9. The pressurizer water volume and RCS pressure also decrease during this cooldown process due to shrinkage of the reactor coolant as shown in Figures 6.4-1 and 6.4-2. The break flow flashing fraction is calculated throughout the transient based on the difference between the enthalpy of the break flow (conservatively calculated assuming all break flow is at the hot leg temperature) and the saturation enthalpy at the ruptured steam generator pressure as shown in Figure 6.4-10. Break flow is calculated to stop flashing at approximately 2,730 seconds as a result of the reduction in primary coolant temperature associated with the cooldown (Figure 6.4-8) and the increase in ruptured steam generator pressure following isolation of the failed-open ARV (Figure 6.4-2).

Although the operators would maintain cooling with the intact steam generators' ARVs after the cooldown was completed, this is not modeled in the analysis.

3. Depressurize to restore inventory.

After the RCS cooldown is completed, a 180-second operator action time is included prior to the RCS depressurization. The RCS depressurization is performed to ensure adequate coolant inventory prior to terminating SI flow. With the RCPs stopped, normal pressurizer spray is not available and thus the RCS is depressurized by opening the pressurizer PORVs. The RCS depressurization is initiated at 3,160 seconds and continues until the RCS pressure is less than the ruptured steam generator pressure and pressurizer level is on span. (The depressurization would also be terminated if pressurizer level rises too high or subcooling margin is lost, but these conditions do not occur in this analysis.) For this case, the RCS depressurization is terminated at 3,219 seconds when the RCS pressure is reduced to less than the ruptured steam generator pressure and the pressurizer level is on span. The RCS depressurization reduces the break flow as shown in Figure 6.4-11 and increases SI flow to refill the pressurizer, as shown in Figure 6.4-1.

4. Terminate SI to stop primary-to-secondary leakage.

   The previous actions establish adequate RCS subcooling, a secondary-side heat sink, and sufficient reactor coolant inventory to ensure that SI flow is no longer needed. When these actions have been completed, the SI flow must be stopped to prevent re-pressurization of the RCS and to terminate primary-to-secondary leakage. The SI flow is terminated at this time if RCS subcooling is greater than the allowance for subcooling uncertainty, minimum AFW flow is available, or at least one intact steam generator level is in the narrow range, the RCS pressure is stable or increasing, and the pressurizer level is on span.

   After depressurization is completed, an operator action time of 5 minutes was assumed prior to SI termination. Since the above requirements are satisfied, SI termination actions were performed at 3,519 seconds by closing off the SI flow path. After SI termination, the RCS pressure begins to decrease as shown in Figure 6.4-2.

   An operator action to depressurize using the pressurizer PORVs is modeled 15 minutes after the first depressurization. This results in termination of break flow at 4,413 seconds, as shown in Figure 6.4-11. (Figure 6.4-11 shows break flow continuing after 4,413 seconds since the RETRAN analysis does not model operator actions to maintain the pressure equilibrium and prevent re-initiation of break flow.)

Other Cases Considered

The case analyzed to provide thermal and hydraulic input to the radiological consequences analysis considers operation at the high end of the vessel average temperature band plus uncertainty (592.7°F), the high end of the feedwater temperature band (446°F), and no steam generator tube plugging. Four other cases were analyzed and included in the development of the input to the radiological consequences analysis to assure a bounding dose result.

	Tavg (°F)	Feedwater (°F)	Tube Plugging (%)
Case 1 (Base Case)	592.7	446	0
Case 2	592.7	446	5
Case 3	592.7	390	0
Case 4	592.7	390	5
Case 5	567.7	446	0

The sequences of events for these cases are compared in Table 6.4-3.

The first four cases showed little impact on the transient for variations in feedwater temperature and steam generator tube plugging. The analysis results used to determine input to the radiological consequences analysis are compared in Figures 6.4-12 through 6.4-18. These include the break flow rate, break flow flashing fraction, ruptured steam generator atmospheric steam release, intact steam generators atmospheric steam release, and primary and secondary water masses. Although the variations are taken into account in the development of the input for the radiological consequences analysis, the base-case transient response is reported as the design-basis event.

The fifth case showed a significant increase in the initial break flow for a lower initial RCS temperature, as shown in Figure 6.4-19. Other parameters are compared in Figures 6.4-20 through 6.4-24. These include the break flow flashing fraction, ruptured steam generator atmospheric steam release, intact steam generators atmospheric steam release, and primary and secondary water masses. While the variations are taken into account in the development of the input for the radiological consequences analysis, the base case transient response is reported as the design-basis event consistent with the methodology presented in Reference 1.

6.4.1.4 Acceptance Criteria and Results

The analysis is performed to calculate the mass transfer data for input to the radiological consequences analysis. As such, no acceptance criteria are defined. The results of the analysis are used as input to the radiological consequences analysis.

RETRAN Analysis Results

The radiological consequences analysis uses the RETRAN transient results as input. Bounding values are developed based on the results of the 5 cases presented in this report.

Reactor Coolant System Mass (lbm)

The radiological consequences analysis models the maximum initial RCS mass at the start of the event and the minimum RCS mass for all other times. The initial value provides a conservatively high initial RCS activity (in μCi), while the minimum mass from the RETRAN transient provides a conservatively high RCS activity concentration (in μCi/lbm).

The maximum initial RCS mass is obtained from the case modeling operation at the low end of the vessel average temperature band minus uncertainty (567.7°F) (because of the higher density) with no tube plugging (to maximize the primary-side volume) as shown in Figure 6.4-23. This mass is conservatively rounded up to 5.8E5 lbm.

The minimum transient RCS mass from the cases considered is obtained from the cases modeling operation at the high end of the vessel average temperature band plus uncertainty (592.7°F) (because of the lower density) with 5-percent tube plugging (to minimize the primary-side volume) as shown in Figure 6.4-16. This mass is conservatively rounded down to 5.2E5 lbm.

Secondary Mass (lbm)

The radiological consequences analysis models the intact steam generators at the initial secondary mass and the ruptured steam generator at the transient minimum secondary mass. These values are modeled for all times during the calculation. Minimizing the secondary mass provides a conservatively high secondary activity concentration (in μCi/lbm).

The minimum steam generator initial mass is conservatively is obtained from the case modeling operation at the low end of the vessel average temperature band minus uncertainty (567.7°F) and feedwater temperature at the high end of the band (446°F), as shown in Figure 6.4-24. This mass is conservatively rounded down to 2.8E5 lbm (total for three steam generators).

The minimum transient value for the ruptured steam generator mass is obtained from the cases modeling operation at the high end of the vessel average temperature band plus uncertainty (592.7°F) and feedwater temperature at the high end of the band (446°F), as shown in Figure 6.4-17. This mass is conservatively rounded down to 7.5E4 lbm.

Break Flow (lbm/sec)

The radiological consequences analysis models a constant tube rupture break flow rate rather than the RETRAN calculated flow, up until the time of SI isolation. The maximum break flow rate is obtained from the case modeling operation at the low end of the vessel average temperature band minus uncertainty (567.7°F), as shown in Figure 6.4-19. A conservative value of 65 lbm/sec for the first hour is selected to bound the results of the cases considered.

(Break flow after the failed-open ARV is isolated and flashing stops has a minor impact on the dose analysis since the activity that enters with the flow is not released until later in the transient progression when the operators cool/depressurize the ruptured steam generator to residual heat removal (RHR) conditions. The radiological consequences analysis makes conservative assumptions regarding those later releases, which are separate from the RETRAN analysis.)

Flashing Fraction

The radiological consequences analysis models a constant break flow flashing until the failed-open ARV is isolated. After isolation, the RETRAN calculated reduction in flashing fraction over time is used. For the RSGs, the flashing fraction is set to cover all of the cases considered. A constant flashing fraction of 0.16 is selected from the start of the event until the failed-open ARV is isolated. Rather than use exact RETRAN calculated flashing fractions for the remainder of the transient (until flashing stops) a linear reduction in flashing will be assumed. In all cases, the ARV is isolated before 1,860 seconds and in all cases flashing stops before 2,820 seconds as shown in Figures 6.4-13 and 6.4-20. These 2 times will be used as the end points of the line modeling the decrease in the flashing fraction. At 1,860 seconds (31 minutes) the flashing fraction is assumed to be 0.16, and at 2,820 seconds (47 minutes) the flashing fraction is assumed to be 0.0. Assuming a linear reduction over the 16-minute interval, the flashing fraction drops by 0.01 per minute. For times greater than 1,860 seconds (31 minutes), the flashing fraction is expressed by an equation based on the transient time: $0.16 - 0.01*(t-1860)/60$, where t is the transient time in seconds. Figure 6.4-25 shows the flashing fraction selected for use in the radiological consequences analysis compared to the RETRAN calculated values for the cases considered.

Ruptured Steam Generator Steam Release

The radiological consequences analysis models the RETRAN calculated flow from the ruptured steam generator to the atmosphere. Prior to reactor trip and the assumed loss of offsite power, steam releases are passed through the condenser with a partition factor of 0.01. Since the water/steam partition in the steam generators is also 0.01, only a small fraction of the iodine in the steam will be released. The contribution to the offsite dose from pre-trip steam releases is, therefore, neglected. Figure 6.4-4 shows the atmospheric steam releases selected for use in the radiological consequences analysis. Margin is added to the steam flows as part of the radiological consequences analysis to assure a conservative analysis.

Intact Steam Generator Steam Release

The radiological consequences analysis models the RETRAN calculated flow from the intact steam generators to the atmosphere. Prior to reactor trip and the assumed loss of offsite power, steam releases are passed through the condenser with a partition factor of 0.01. Since the water/steam partition in the steam generators is also 0.01, only a small fraction of the iodine in the steam will be released. The contribution to the offsite dose from pre-trip steam releases is, therefore, neglected. Figure 6.4-5 shows the atmospheric steam releases selected for use in the radiological consequences analysis. Margin is added to the steam flows as part of the radiological consequences analysis to assure a conservative analysis.

6.4.1.5 Conclusions

The analysis performed to calculate the mass transfer data for input to the radiological consequences analysis for the case of a SGTR with a failed-open ARV on the ruptured steam generator has been completed and data prepared for use in the radiological consequences analysis. The inputs are summarized below.

- RCS mass at the start of event = 5.8E5 lbm
- RCS mass for all other times = 5.2E5 lbm
- Ruptured steam generator mass = 7.5E4 lbm
- Intact steam generators mass = 2.8E5 lbm
- Break flow rate for first hour = 65 lbm/sec
- Flashing fraction from start of event until ARV is isolated (0 to 1,860 sec) = 0.16
- Flashing fraction after ARV isolation until flashing stops (1,860 to 2,820 sec) = 0.16 - 0.01*(t-1,860)/60, where t = transient time in seconds
- Ruptured steam generator releases: Figure 6.4-4
- Intact steam generators releases: Figure 6.4-5

6.4.2 SGTR with Failure of the Ruptured Steam Generator AFW Flow Control Valve

6.4.2.1 Introduction and Background

The failure of the ruptured steam generator AFW control valve potentially results in water relief from the ruptured steam generator. An SGTR analysis that resulted in water relief from the ruptured steam generator considering this single failure was performed by the Callaway plant in response to Nuclear Regulatory Commission (NRC) questions, in 1987 (Reference 3). The analysis included assumptions selected to force water relief from the ruptured steam generator so that the offsite doses could be calculated with water release and the associated higher activity releases. The analysis demonstrated that even in the event of steam generator overfill and water relief, the offsite doses remained within the applicable limits of Standard Review Plan (SRP) 15.6.3. The analysis was updated, as documented in Reference 4, in support of proposed modifications to the Callaway plant's main feedwater isolation valves (MFIVs) and AFW pump discharge check valves. The re-analysis also incorporated revised input and modeling assumptions to assure that the analysis is consistent with the plant's current configuration and operation.

The analysis of the SGTR with failure of the ruptured steam generator AFW flow control valve was performed for the RSG Program in a manner consistent with that for operation with the original steam generators. The Reference 4 analysis used the Westinghouse version of RETRAN02 (Reference 2) while retaining the plant model (volumes, junctions etc.) described in Reference 3. For the RSG analysis, the Westinghouse version of RETRAN02 is used with the plant model described in Reference 2. This section includes the methods and assumptions used to analyze the SGTR event, as well as the sequence of events for the recovery and the calculated results.

6.4.2.2 Input Parameters and Assumptions

The thermal-hydraulic analysis for the case modeling a failure of the ruptured steam generator AFW flow control valve includes assumptions selected (as discussed in Reference 4) to maximize the release of activity with water from the ruptured steam generator. This analysis models the plant operating at the lower end of the Tavg window, since a lower operating temperature results in a higher mass flow rate through the broken tube and less steam released from the ruptured steam generator. The analysis assumes that the plant is operating with the feedwater temperature at the low end of the temperature window, since this results in a higher mass of water in the steam generator at the start of the event, which limits the amount of break flow and auxiliary feedwater that can accumulate in the ruptured steam generator without forcing water into the steam lines. Maximum (5-percent) tube plugging is assumed since this reduces the heat transfer to the ruptured steam generator, minimizing the amount of mass released from the steam generator due to steaming, which in turn reduces the margin to overfill.

Design-Basis Accident

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 side of the steam generator results in higher primary-to-secondary leakage than a break on the hot side of the steam generator. The analysis assumes that, concurrent with the SGTR, a reactor trip occurs and a safety injection signal is generated. It was also assumed that a loss of offsite power (LOOP) occurs at the time of reactor trip, and the highest worth control assembly was assumed to be stuck in its fully withdrawn position at reactor trip. Following reactor trip, SI actuation, and the LOOP, the feedwater flow stops, and the MSIVs close. The secondary pressure rises and approaches the secondary ARVs' and safety valves' (SVs') setpoints. In response to the reactor trip and LOOP, AFW is delivered to the secondary. It is assumed that the AFW control valve fails full open on the ruptured steam generator and delivers excessive

AFW to the ruptured steam generator. The excessive AFW flow quickly rebounds the ruptured steam generator water level and drives the steam generator toward overfill.

The ruptured steam generator overfills and water fills the steam line up to the main steam isolation valve (MSIV). When the steam generator and steam line go water solid, a pressure spike (on the secondary) occurs as the primary side (driven by SI) drives the secondary pressure toward equilibrium. Therefore, an SV opens and contaminated water is dumped to the atmosphere. Water continues to be relieved from the ruptured steam generator SV until equilibrium is reached between the primary and secondary pressures as a result of operator actions, effectively terminating flow into the ruptured steam generator. To assure continued relief as requested by the NRC for this scenario, an active failure of the SV is assumed to occur, that is, after water relief the valve remains partially open (5 percent) Eventually, water relief depletes the secondary mass and creates a steam void. This steam void grows until water is no longer able to pass out of the SV.

It is assumed that steam relief continues until RHR cut-in, since steam relief continues to shrink the ruptured steam generator mass via cooling and mass depletion. Following break flow termination, it is assumed that the operators transition to the cooldown procedures and initiate cooldown via the intact steam generators' ARVs.

The major assumptions used for the RETRAN analysis are the following.

1. The following initial conditions are chosen:

   a. Core power = 3,565 MWt (power uncertainty is not added)

   b. Net RCS heat input = 14 MWt

   c. Pressurizer pressure = 2,280 psia (nominal pressure of 2,250 psia +30 psia uncertainty)

   d. Initial pressurizer level = 38 percent. This reflects the pressurizer level control used for operation at the low end of the Tavg window.

   e. Vessel average temperature = 567.7°F. This reflects the low end of the Tavg band of 570.7°F - 588.4°F and the uncertainty of +3°F.

   f. RCS flow = thermal design flow = 374,400 gpm (93,600 gpm/loop)

   g. Feedwater temperature = 390°F. This is the low end of the feedwater temperature band.

   h. 5-percent steam generator tube plugging is assumed.

   i. Steam generator pressure: corresponding to conditions being analyzed.

   j. Steam generator level = 57.5-percent NRS. This is the nominal level of 51.3-percent NRS plus 6.2-percent NRS for uncertainty.

2. Single failure: The ruptured steam generator's auxiliary feedwater control valve fails in the full-open position.

3. Additional active failure: The ruptured steam generator's safety valve fails partially open (5-percent effective area) after water relief.

4. The ARV on the ruptured steam generator is assumed inoperable.

5. The tube rupture is modeled as a double-ended-guillotine break of a single tube at the cold leg tube sheet. An additional 5-percent uncertainty is added to the flow predicted for resistance limited flow.

6. Reactor trip occurs at time zero.

7. LOOP occurs at reactor trip that is, at time zero)

8. The MSIV isolation is modeled at reactor trip and the assumed LOOP, although it could be significantly delayed based on the expected operator response. Early isolation of the MSIV allows the ruptured steam generator to depressurize due to the addition of the (maximum) AFW flow, while the intact steam generator pressure stays relatively high. This results in increased break flow to the ruptured steam generator, which is conservative. It also leads to higher AFW flow to the ruptured steam generator. If the MSIV would be left open, the steam generators would tend to be at the same pressure, which would be closer to that of the intact steam generators. Also, with the MSIV open overfilling the ruptured steam generator would not necessarily lead to water relief, since the water could go to the intact steam generators. The secondary pressure would not spike and the safety valve would not lift.

9. The MFIV closure is modeled as a step function after a 17-second delay. The SI signal generated at reactor trip initiates the isolation.

10. Decay heat = 0.8 x 1979 ANS 2σ model.

11. The following maximum AFW flow rates are modeled prior to partial/full isolation of AFW flow to the ruptured steam generator:

    a. The AFW flow to the ruptured steam generator before isolation of the turbine-driven AFW pump flow to the ruptured steam generator, at the intact steam generator pressure of 1,235.7 psia is used as a base. As the intact steam generator pressure drops, the flow to the ruptured steam generator is reduced. This model is reflected below:

Ruptured SG Pressure (psia)	AFW to Ruptured SG (gpm)	Intact SG Pressure (psia)	Reduction in AFW to Ruptured SG (gpm)
414.7	1,317.0	414.7	72.6
614.7	1,214.0	614.7	55.4
814.7	1,104.0	814.7	37.8
1,014.7	982.0	1,014.7	20.0
1,139.7	895.0	1,139.7	8.6
1,235.7	823.0	1,235.7	0.0

b. The AFW flow to the intact steam generators (total for the 3) before isolation of the turbine-driven AFW pump flow to the ruptured steam generator is provided below.

Intact SG Pressure (psia)	AFW to Intact SGs (gpm)
214.7	1,691.0
414.7	1,576.0
614.7	1,455.0
814.7	1,326.0
1,014.7	1,186.0
1,139.7	1,091.0
1,235.7	1,013.0

12. The following maximum AFW flow rates are modeled after partial/full isolation of AFW flow to the ruptured steam generator:

a. The AFW flow to the ruptured steam generator after isolation of the turbine-driven AFW pump flow to the ruptured steam generator is provided below:

Ruptured SG Pressure (psia)	AFW to Ruptured SG (gpm)
414.7	770.
614.7	712.
814.7	651.
1,014.7	586.
1,139.7	537.
1,235.7	498.

b. The AFW flow to the intact steam generators (total for the 3) after isolation of the turbine-driven AFW pump flow to the ruptured steam generator, and after complete isolation of AFW to the ruptured steam generator, is provided below:

Intact SG Pressure (psia)	AFW to Intact SGs (gpm)
214.7	1,760.
414.7	1,656.
614.7	1,546.
814.7	1,425.
1,014.7	1,295.
1,139.7	1,205.
1,235.7	1,129.

13. AFW flow is initiated 5 seconds after reactor trip, with a 30-second ramp up to full flow.

14. Safety injection modeling: High and intermediate injection pumps assumed with maximum expected flow. Injection starts 15 seconds after the SI signal (which is generated at the start of the event).

15. The break flow flashing fraction is conservatively determined assuming all break flow is at the ruptured loop hot leg temperature.

16. Pressurizer heaters and sprays are not modeled.

17. Ruptured steam generator secondary-side volume modeling: The secondary-side volume of a single steam generator is ~5,489 ft^3. The steam line volume up to the MSIV is ~733 ft^3. The estimated volume in the horizontal section of the steam pipe up to the MSIV is 201 ft^3. Only 100 ft^3 of this volume (about half) is credited. Water relief is not started until the pressure spike, which occurs when the defined RETRAN volume becomes water solid, lifts the SV. Until that time water is filling the steam line up to the MSIV but does not force the SV open so no water is released. Once water release starts it continues until the water in the ruptured steam generator steam line drops below the 100 ft^3 of horizontal steam line.

18. Turbine-driven AFW flow to the ruptured steam generator is isolated 10 minutes from the start of the event.

19. All AFW flow to the ruptured steam generator is isolated 20 minutes from the start of the event.

20. The RCS cooldown is initiated 10 minutes after AFW flow to the ruptured steam generator is isolated. The narrow-range level in all steam generators must be greater than 4 percent and the ruptured steam generator pressure must be greater than 430 psig prior to initiating RCS cooldown.

21. All 3 intact steam generator ARVs are credited in the cooldown.

22. RCS depressurization is assumed to begin 3 minutes after completion of cooldown. When the ruptured steam generator pressure is higher than the RCS pressure, the pressurizer PORVs are closed.

23. Safety injection is terminated 5 minutes after completion of RCS depressurization.

24. A second depressurization to terminate break flow is completed approximately 15 minutes after SI termination.

25. Cooldown to RHR cut in is initiated after break flow is terminated, and is completed within 5 hours after break flow termination. The RETRAN analysis does not include the complete cooldown to RHR conditions. The initial part of the cooldown is shown to demonstrate that once the cooldown is initiated the pressure differential (and break flow) is minimal.

Operator Actions

The major operator actions required for the recovery from a SGTR are discussed in subsection 6.4.1.2, and the operator action times used for the analysis of the case with the failed open ARV on the ruptured steam generator are presented in Table 6.4-1. These operator action times were also used for the offsite dose failure of the ruptured steam generator AFW flow control valve. However, for this scenario, instead of the 20 minutes required to isolate the failed open ARV, operator action times are included to model isolation of AFW flow to the ruptured steam generator. The operator actions and the corresponding operator action times used for this analysis are listed in Table 6.4-4.

Mass Releases

The mass releases were determined for use in evaluating the offsite and control room radiological consequences of the SGTR. The steam and water releases from the ruptured steam generator, steam releases from the intact steam generators, primary-to-secondary break flow into the ruptured steam generator, and the break flow flashing fraction were calculated by the RETRAN program for the period from accident initiation until after water release from the ruptured steam generator stops. Input to the dose analysis is developed consistent with the analysis submitted in Reference 4 for operation with the original steam generators. For most input parameters, the RETRAN transient results are used directly.

6.4.2.3 Description of Analysis and Evaluation

The RETRAN analysis results for the Callaway plant analysis of the SGTR with failure of the ruptured steam generator AFW flow control valve are described below. The sequence of events for the analysis is presented in Table 6.4-5.

Reactor trip and SI actuation are assumed to occur coincident with the tube rupture and LOOP is assumed to result from the trip. The SI signal initiates AFW flow and isolates main feedwater. The AFW is initiated to provide cooling for decay heat removal. The secondary pressure rises slightly after trip, but no steam is released. The cold AFW flow absorbs the secondary side energy and depressurizes all of the steam generators as shown in Figure 6.4-26.

Major Operator Actions

1. Identify and isolate the ruptured steam generator.

   Recovery actions begin by throttling the AFW flow to the ruptured steam generator and isolating steam flow from the ruptured steam generator. As indicated previously, the steam flow from the ruptured steam generator to the header is isolated just after reactor trip. Within 10 minutes, the operators are assumed to isolate flow from the turbine-driven AFW pump to the ruptured steam generator in response to the increase in level in that steam generator. Due to the assumed failure of the valve from the motor-driven AFW pump, AFW flow to the ruptured steam generator continues until 20 minutes when the operators are assumed to have isolated all AFW flow to the ruptured steam generator. AFW flow is shown together with the secondary level in Figures 6.4-27 and 6.4-28.

   Due to the assumed high initial secondary-side water inventory, maximum AFW and conservatively high break flow modeling, the ruptured steam generator overfills before it is completely isolated (Figure 6.4-29). When the steam line volume up to the MSIV fills with water a pressure spike occurs in that steam line (Figure 6.4-26), and the SV lifts. Initial flow out of the SV (Figure 6.4-30) is high, matching the flow into the steam generator (AFW plus break flow). After AFW isolation and as break flow is reduced, the flow out the SV drops. It is assumed that the valve sticks open with 5-percent effective area, leading to continued water relief at rates that exceed the flow into the steam generator.

2. Cool down the RCS to establish subcooling margin.

   After AFW flow to the ruptured steam generator is terminated, there is a 10-minute operator action time imposed prior to initiation of cooldown. Since offsite power is lost, the RCS is cooled by dumping steam to the atmosphere using the intact steam generators' ARVs. The cooldown is continued until RCS subcooling at the ruptured steam generator pressure is 20°F plus an allowance for instrument uncertainty. The cooldown would not be initiated until all steam generators indicate greater than 4-percent level and the ruptured steam generator pressure is greater than 430 psig. Since both of these criteria are met, the cooldown is initiated after 10 minutes. The cooldown begins at 1,800 seconds, using all 3 intact steam generator's ARVs, and is completed at 2,379 seconds.

   The reduction in the intact steam generators' pressure required to accomplish the cooldown is shown in Figure 6.4-26 and the effect of the cooldown on the RCS temperatures is shown in Figures 6.4-31 and 6.4-32. The RCS pressure and pressurizer water volume also decrease during this cooldown process due to shrinkage of the reactor coolant as shown in Figures 6.4-26 and 6.4-33. The break flow flashing fraction is calculated throughout the transient based on the difference between the enthalpy of the break flow (conservatively calculated assuming all break flow is at the hot leg temperature) and the saturation enthalpy at the ruptured steam generator pressure as shown in Figure 6.4-34. Break flow is calculated to stop flashing initially at approximately 960 seconds as a result of the increase in ruptured steam generator pressure following overfill (Figure 6.4-26). Flashing starts again after the pressure drops and finally stops at approximately 2,150 seconds as a result of the reduction in primary coolant temperature associated with the cooldown (Figure 6.4-34).

Although the operators would maintain cooling with the intact steam generators' ARVs after the cooldown was completed, this is not modeled in the analysis. Cooldown to RHR conditions using the intact steam generators ARVs is assumed to be initiated after break flow termination.

3. Depressurize to restore inventory.

After the RCS cooldown is completed, a 180-second operator action time is included prior to the RCS depressurization. The RCS depressurization is performed to assure adequate coolant inventory prior to terminating SI flow. With the RCPs stopped, normal pressurizer spray is not available and thus the RCS is depressurized by opening the pressurizer PORVs. The RCS depressurization is initiated at 2,559 seconds and continued until RCS pressure is less than the ruptured steam generator pressure and the pressurizer level is on span. (The depressurization would also be terminated if pressurizer level rises too high or subcooling margin is lost, but these conditions do not occur in this analysis.) For this case, the RCS depressurization is terminated at 2,626 seconds when the RCS pressure is reduced to less than the ruptured steam generator pressure and the pressurizer level is on span. The RCS depressurization reduces the break flow as shown in Figure 6.4-35 and increases SI flow to refill the pressurizer, as shown in Figure 6.4-33.

4. Terminate SI to stop primary-to-secondary leakage.

The previous actions establish adequate RCS subcooling, a secondary-side heat sink, and sufficient reactor coolant inventory to ensure that SI flow is no longer needed. When these actions have been completed, the SI flow must be stopped to prevent re-pressurization of the RCS and to terminate primary-to-secondary leakage. The SI flow is terminated at this time if RCS subcooling is greater than the allowance for subcooling uncertainty, minimum AFW flow is available or at least one intact steam generator level is in the narrow range, the RCS pressure is stable or increasing, and the pressurizer level is on span.

After depressurization is completed, an operator action time of 5 minutes was assumed prior to SI termination. Since the above requirements are satisfied, SI termination actions were performed at 2,926 seconds by closing off the SI flow path. After SI termination, the RCS pressure begins to decrease as shown in Figure 6.4-26.

An operator action to depressurize using the pressurizer PORVs is modeled 15 minutes after the first depressurization. This results in complete termination of break flow at 3,841 seconds, as shown in Figure 6.4-35. Cooldown to RHR conditions is initiated at approximately this time.

Eventually, the steam void resulting from continued water relief from the assumed stuck-open SV on the ruptured steam generator grows (Figure 6.4-29) to the extent that the valve no longer passes water (Figure 6.4-30). This occurs at approximately 5,000 seconds from the start of the event.

Cooldown to RHR conditions is not modeled in the RETRAN analysis. It is assumed that the cooldown is completed within 5 hours after break flow termination.

6.4.2.4 Acceptance Criteria and Results

The analysis is performed to calculate the mass transfer data for input to the radiological consequences analysis. As such no acceptance criteria are defined. The results of the analysis are used as input to the radiological consequences analysis.

RETRAN Analysis Results

The radiological consequences analysis uses the RETRAN transient results as input.

RCS Mass (lbm)

The radiological consequences analysis models the maximum initial RCS mass at the start of the event and the RETRAN calculated transient mass for other times. The initial value provides a conservatively high initial RCS activity (in μCi). The increase in RCS mass during the transient which results from the safety injection flow and cooldown of the system provides a benefit for the dose analysis by diluting the activity.

From subsection 6.4.1.4, the conservative maximum initial RCS mass for use in the dose analysis is 5.8E5 lbm.

The transient RCS mass is shown in Figure 6.4-36.

Secondary Mass (lbm)

The radiological consequences analysis models the RETRAN calculated transient mass. For the ruptured steam generator, this includes the mass of water in the steam line following overfill, since the activity mixes in this volume prior to release. The increase in secondary mass during the transient resulting from AFW flow and break flow provides a benefit for the dose analysis by diluting the activity.

The transient ruptured and intact secondary masses are shown in Figure 6.4-36.

Break Flow (lbm/sec)

The radiological consequences analysis models the RETRAN calculated transient break flow up until break flow termination. The transient break flow is shown in Figure 6.4-35. For the dose analysis a constant break flow rate of 4 lbm/sec is imposed following break flow termination, although it is not included in the RETRAN analysis.

Flashing Fraction

The radiological consequences analysis models the RETRAN calculated transient break flashing fraction. The transient flashing fraction is shown in Figure 6.4-34.

Ruptured Steam Generator Steam/Water Release

The radiological consequences analysis models the RETRAN calculated flow from the ruptured steam generator to the atmosphere. Figure 6.4-30 shows the atmospheric steam/water releases for use in the radiological consequences analysis. Margin is added to the steam flows as part of the radiological consequences analysis to assure a conservative analysis. For the dose analysis a constant steam flow rate of 8 lbm/sec is imposed following termination of water relief, although it is not included in the RETRAN analysis.

Intact Steam Generator Steam Release

The radiological consequences analysis models the RETRAN calculated flow from the intact steam generators to the atmosphere. Figure 6.4-37 shows the atmospheric steam releases for use in the radiological consequences analysis. Margin is added to the steam flows as part of the radiological consequences analysis to assure a conservative analysis.

6.4.2.5 Conclusions

The analysis performed to calculate the mass transfer data for input to the radiological consequences analysis for the case of an SGTR with failure of the ruptured steam generator AFW flow control valve has been completed and data prepared for use in the radiological consequences analysis. The inputs are summarized below.

- The RCS mass at the start of event = 5.8E5 lbm
- The RCS mass for all other times = Figure 6.4-36
- Ruptured steam generator mass = Figure 6.4-36
- Intact steam generators mass = Figure 6.4-36
- Break flow rate until 64 minutes = Figure 6.4-35
- Flashing fraction = Figure 6.4-34
- Ruptured steam generator releases: Figure 6.4-4
- Intact steam generators releases: Figure 6.4-5

6.4.3 References

1. SLNRC 86-3, "Steam Generator Tube Rupture Analysis – SNUPPS," February 11, 1986.

2. WCAP-14882-P-A, "RETRAN-02 Modeling and Qualification for Westinghouse Pressurized Water Reactor Non-LOCA Safety Analyses," Huegel et al., April 1999.

3. ULNRC-1518, "Steam Generator Tube Rupture Analysis," May 27, 1987.

4. ULNRC-04592, "Proposed Revision to Technical Specification 1.1, "Definitions"; Technical Specification 3.7.3 "Main Feedwater Isolation Valves (MFIVS)"; and Steam Generator Tube Rupture With Overfill; Re-analysis," June 27, 2003.

Table 6.4-1 Operator Action Times for Analysis of SGTR with Failed-Open ARV	
Action	**Time**
Isolate Steam Flow from Ruptured SG	Conservatively Modeled at Reactor Trip
Isolate Failed-Open ARV on Ruptured SG	20 Minutes from Failure
Isolate AFW Flow to Ruptured SG	RETRAN Calculated Time To Reach 4% Narrow-Range Level in the Ruptured SG
Operator Action Time to Initiate Cooldown	10 Minutes from Time of Complete Isolation of Ruptured Steam Generator (Including Failed-Open ARV)
Cooldown	Calculated by RETRAN
Operator Action Time to Initiate Depressurization	3 Minutes from End of Cooldown
Depressurization	Calculated by RETRAN
Operator Action Time to Terminate SI Following Depressurization	5 Minutes from End of Depressurization
Operator Action for Second Depressurization	15 Minutes from SI Termination

Table 6.4-2 Sequence of Events SGTR with Failed-Open ARV on Ruptured Steam Generator	
Event	**Time (seconds)**
Tube Rupture Occurs	0
Reactor Trip Signal	597
Safety Injection Signal	597
Rod Motion	599
Feedwater Terminated	603
Ruptured Steam Generator Atmospheric/Steam Dump Valve Opens	604
Safety Injection Begins	612
Auxiliary Feedwater Injection	659
Operator Isolates Faulted Steam Generator by Closing Manual Block Valve	1,804
Operator Initiates RCS Cooldown Via Intact Steam Generators Atmospheric Steam Dump Valves	2,404
Operator Completes RCS Cooldown	2,980
Operator Initiates RCS Depressurization via Pressurizer PORVs	3,160
Operator Completes RCS Depressurization	3,219
Operator Terminates SI	3,519
Operator Equalizes Primary-Secondary Pressure	4,413

Table 6.4-3 Sequence of Events for Feedwater Temperature/SGTP Sensitivity Cases SGTR with Failed-Open ARV on Ruptured Steam Generator

Event	Base Case 0% SGTP High Tavg High FW Temperature Time (sec)	Case 2 5% SGTP High Tavg High FW Temperature Time (sec)	Case 3 0% SGTP High Tavg Low FW Temperature Time (sec)	Case 4 5% SGTP High Tavg Low FW Temperature Time (sec)	Case 5 0% SGTP Low Tavg High FW Temperature Time (sec)
Tube Rupture Occurs	0	0	0	0	0.0
Reactor Trip Signal	597	593	607	603	439
Safety Injection Signal	597	593	607	603	439
Rod Motion	599	595	609	605	441
Feedwater Terminated	603	600	613	609	445
Ruptured Steam Generator ARV Opens	604	600	614	610	588
Safety Injection Begins	612	608	622	618	454
Auxiliary Feedwater Injection	659	655	669	665	501
Operator Isolates Faulted Steam Generator	1,804	1,800	1814	1,810	1,788
Operator Initiates RCS Cooldown	2,404	2,400	2,414	2,410	2,389
Operator Completes RCS Cooldown	2,980	2,972	2,999	2,990	2,984
Operator Initiates RCS Depressurization	3,160	3,152	3,179	3,170	3,164
Operator Completes RCS Depressurization	3,219	3,211	3,237	3,228	3,224
Operator Terminates SI	3,519	3,511	3,538	3,529	3,524
Operator Equalizes Primary-Secondary Pressure	4,413	4,407	4,432	4,422	4,418

Table 6.4-4 Operator Action Times for Analysis of SGTR with Failed AFW Control Valve	
Action	**Time**
Isolate Steam Flow from Ruptured SG	Conservatively Modeled at Reactor Trip
Isolate Auxiliary Feedwater Flow to Ruptured SG	Flow from Turbine Driven AFW Pump Isolated at 10 Minutes from Start of Event. Flow from Motor-Driven AFW Pump Isolated at 20 Minutes from the Start of the Event.
Operator Action Time to Initiate Cooldown	10 Minutes from Time of Complete Isolation of Ruptured Steam Generator (Including Failed-Open ARV)
Cooldown	Calculated by RETRAN
Operator Action Time to Initiate Depressurization	3 Minutes from End of Cooldown
Depressurization	Calculated by RETRAN
Operator Action Time to Terminate SI Following Depressurization	5 Minutes from End of Depressurization
Operator Action for Second Depressurization	15 Minutes from SI Termination

Table 6.4-5 Sequence of Events SGTR with Failed Ruptured Steam Generator AFW Control Valve	
Event	**Time (seconds)**
Tube Rupture Occurs	0
Reactor Trip and Loop	0
Safety Injection Signal	0
Auxiliary Feedwater Injection Starts	5
Safety Injection Flow Begins	15
Feedwater Terminated	17
Operator Terminates AFW Flow from Turbine-Driven Pump to Ruptured Steam Generator	600
Ruptured Steam Generator Water Relief Begins	970
Operator Terminates AFW Flow from Motor-Driven Pump To Ruptured Steam Generator	1,200
Operator Initiates RCS Cooldown via Intact Steam Generators Atmospheric Steam Dump Valves	1,800
Operator Completes RCS Cooldown	2,379
Operator Initiates RCS Depressurization via Pressurizer PORVs	2,559
Operator Completes RCS Depressurization	2,626
Operator Terminates SI	2,926
Operator Initiates Cooldown to RHR Conditions	3,840
Operator Equalizes Primary-Secondary Pressure	3,841
Stuck-Open Ruptured SG Safety Valve Begins to Relieve Steam	5,005
RHR Cut-in Conditions Reached	21,800

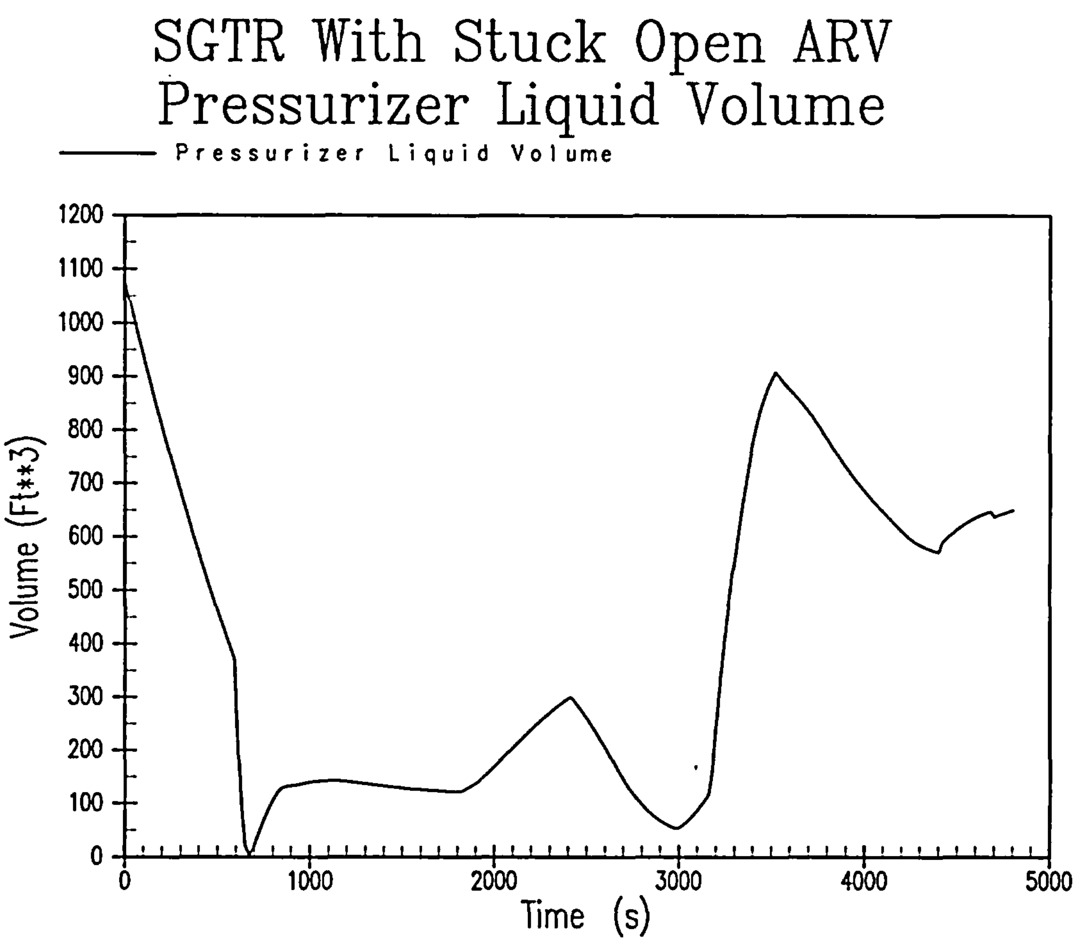


Figure 6.4-1 Pressurizer Liquid Volume

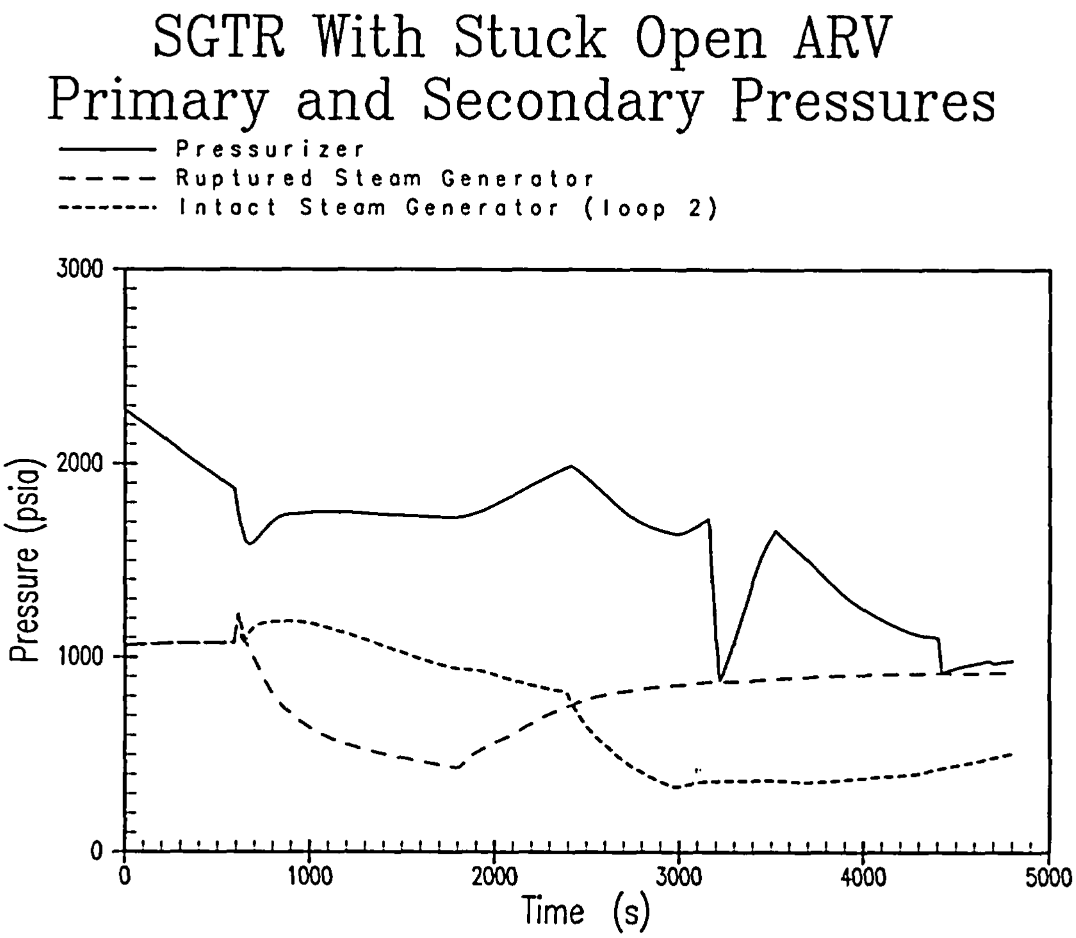


Figure 6.4-2 Primary and Secondary Pressure

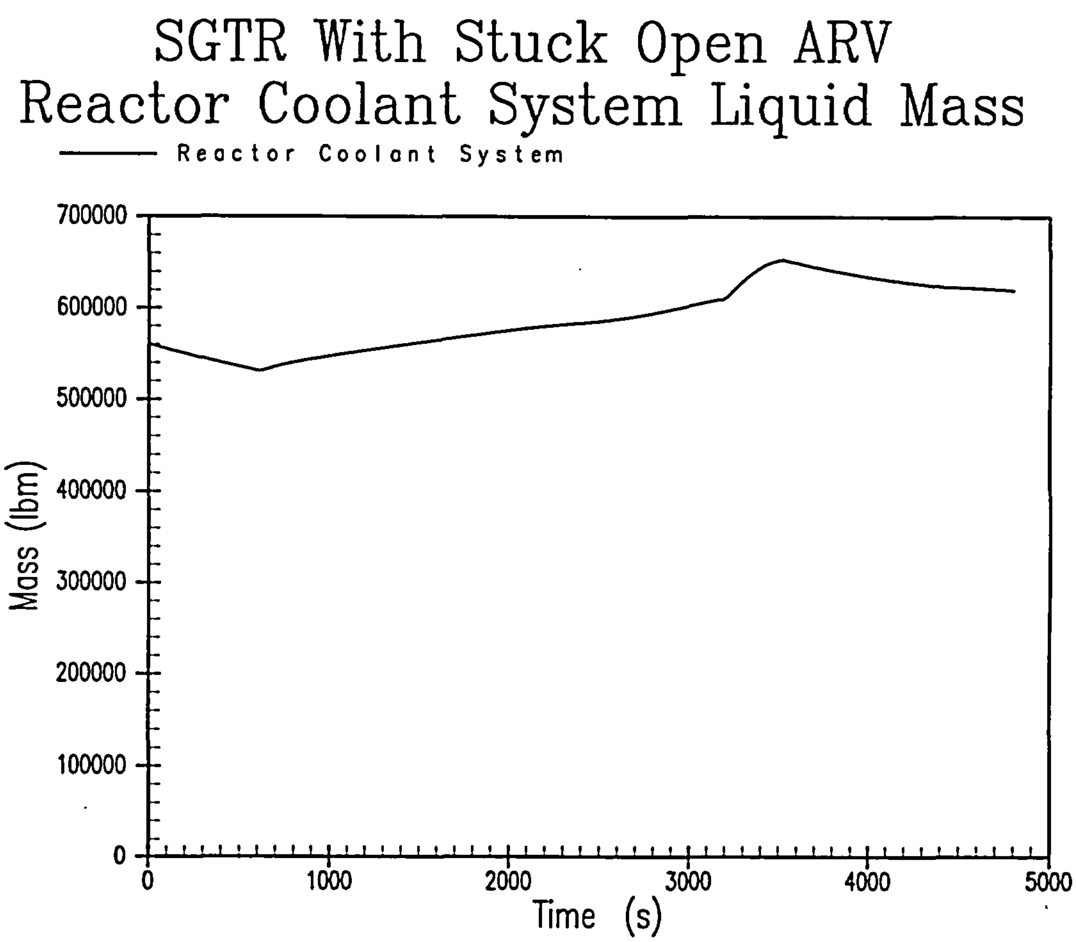


Figure 6.4-3 Reactor Coolant System Liquid Mass

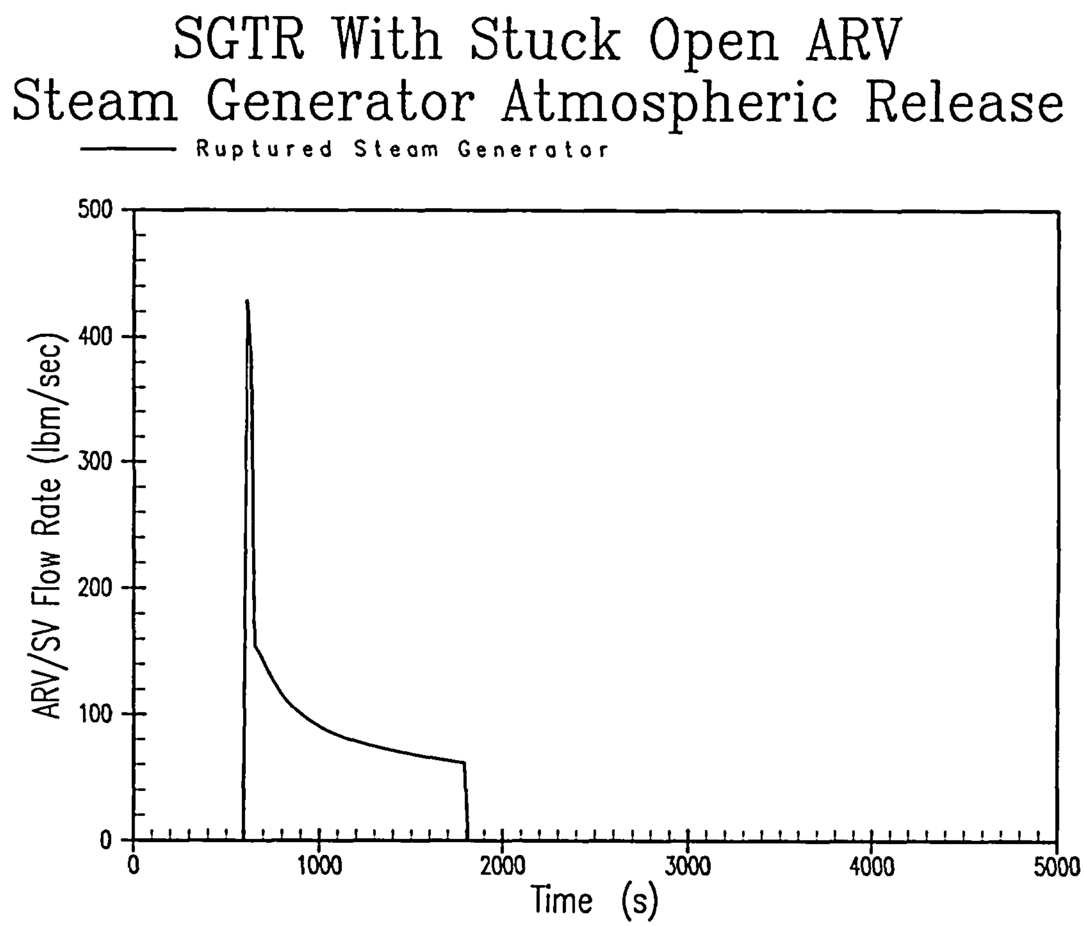


Figure 6.4-4 Ruptured SG Atmospheric Releases

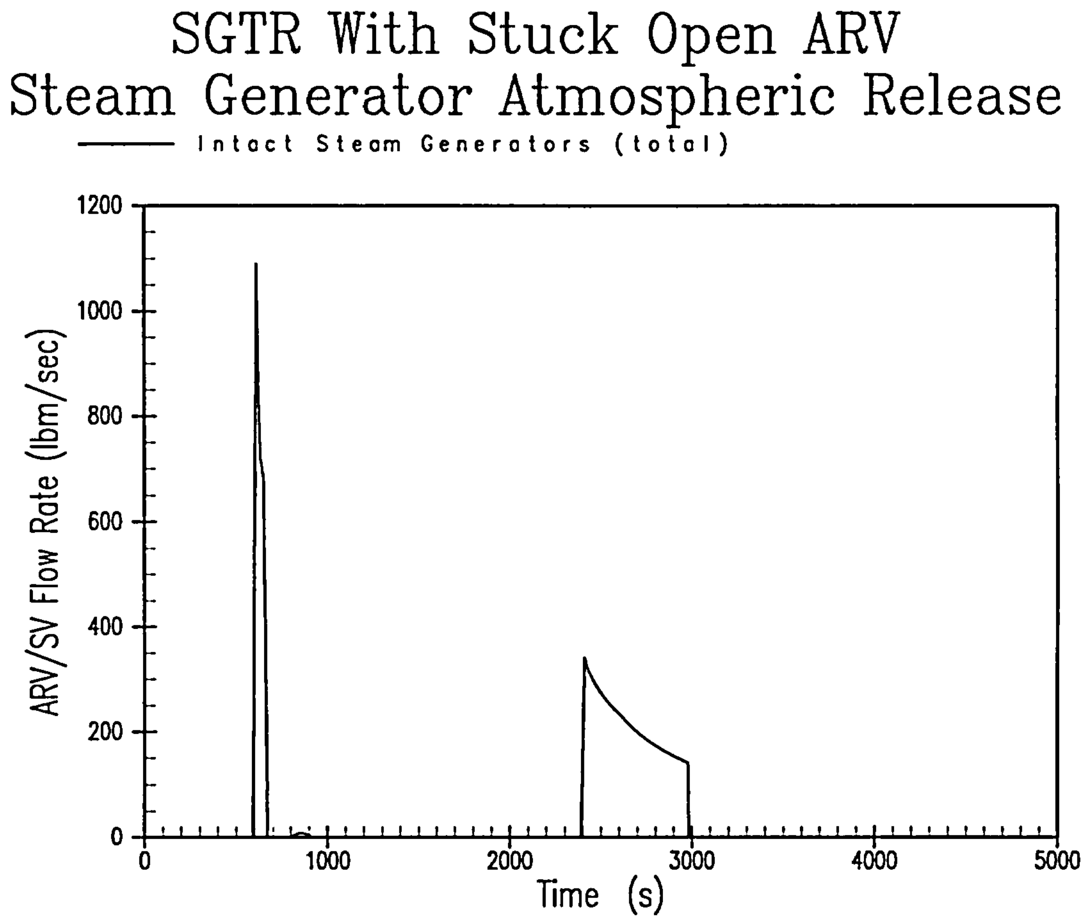


Figure 6.4-5 Intact SGs Atmospheric Releases

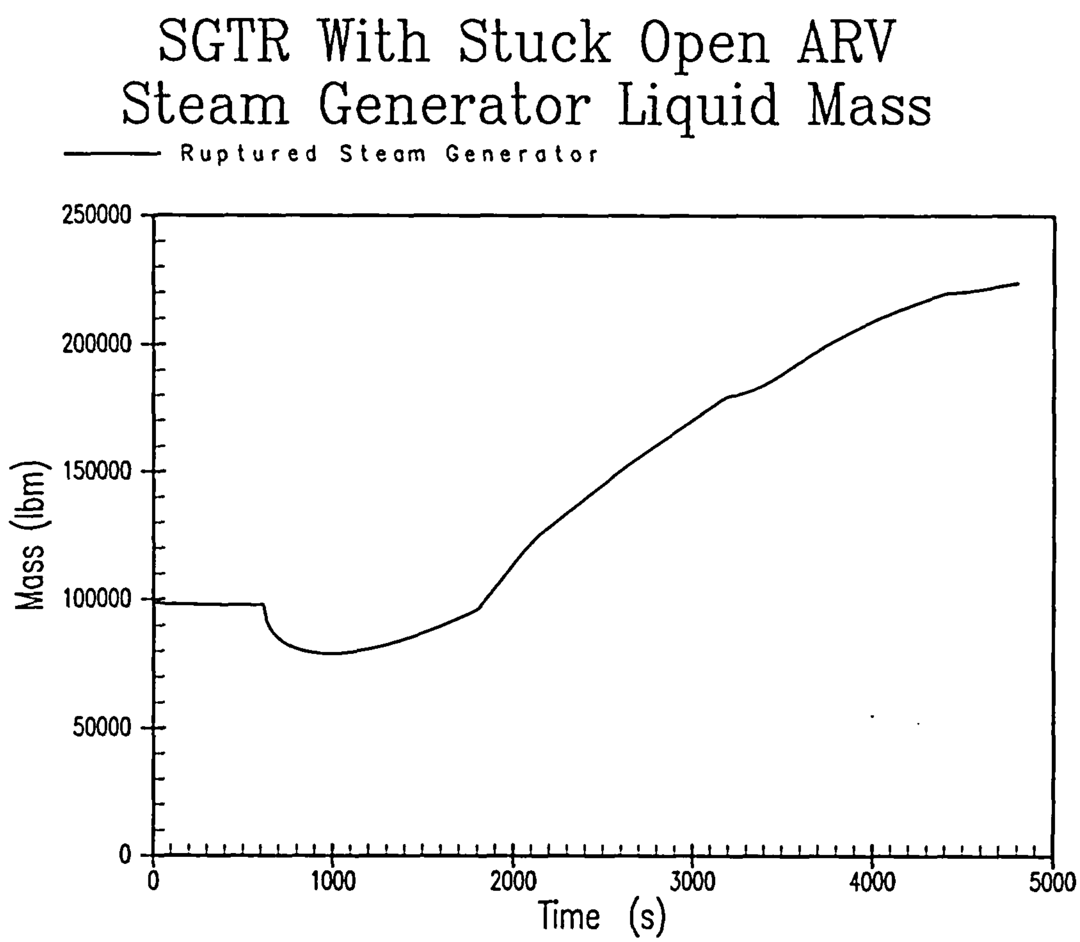


Figure 6.4-6 Ruptured Steam Generator Liquid Mass

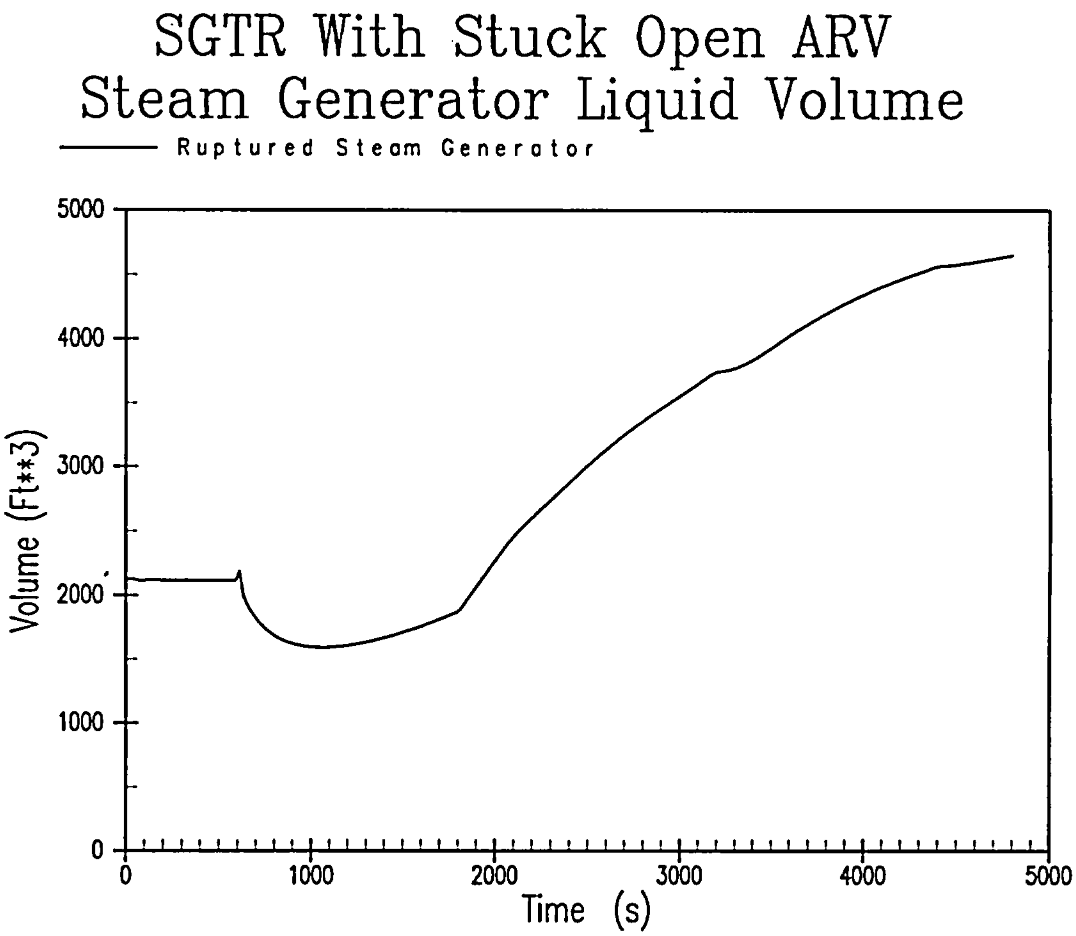


Figure 6.4-7 Ruptured Steam Generator Liquid Volume

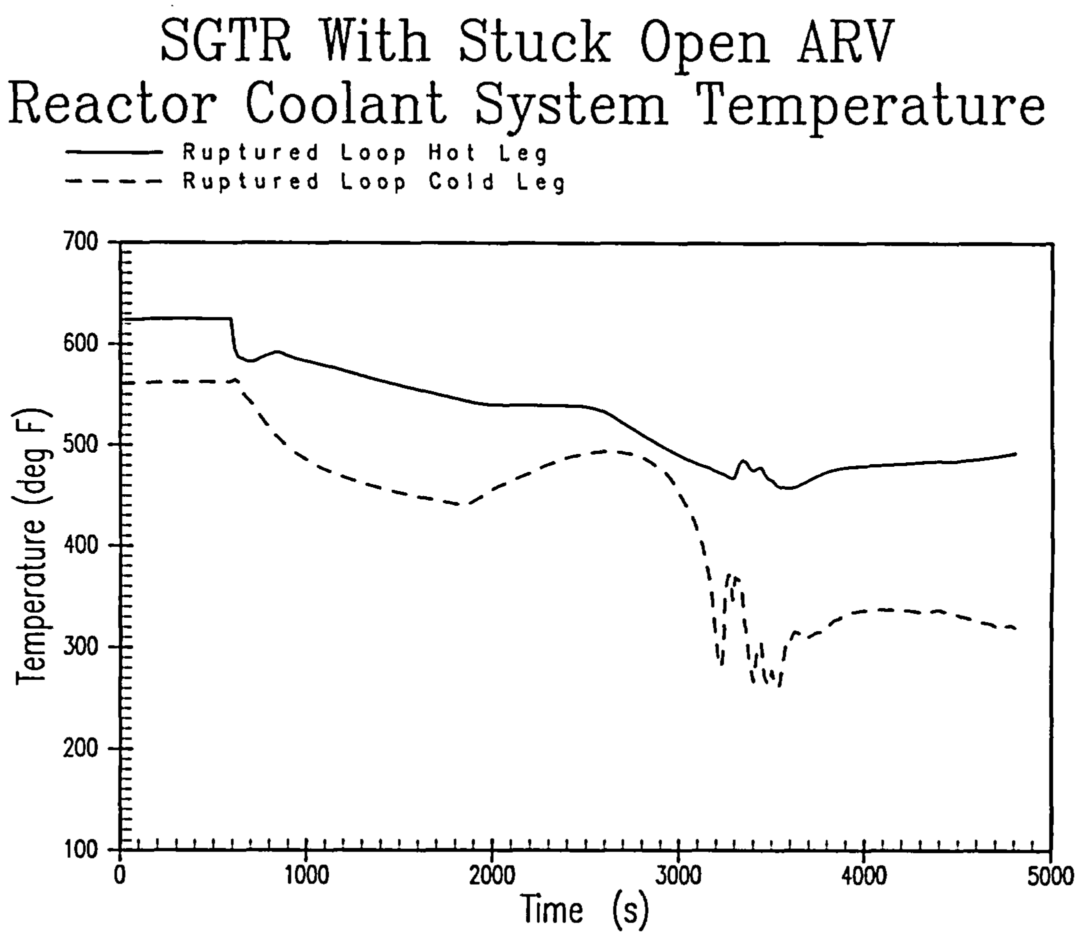


Figure 6.4-8 Ruptured Loop Temperatures

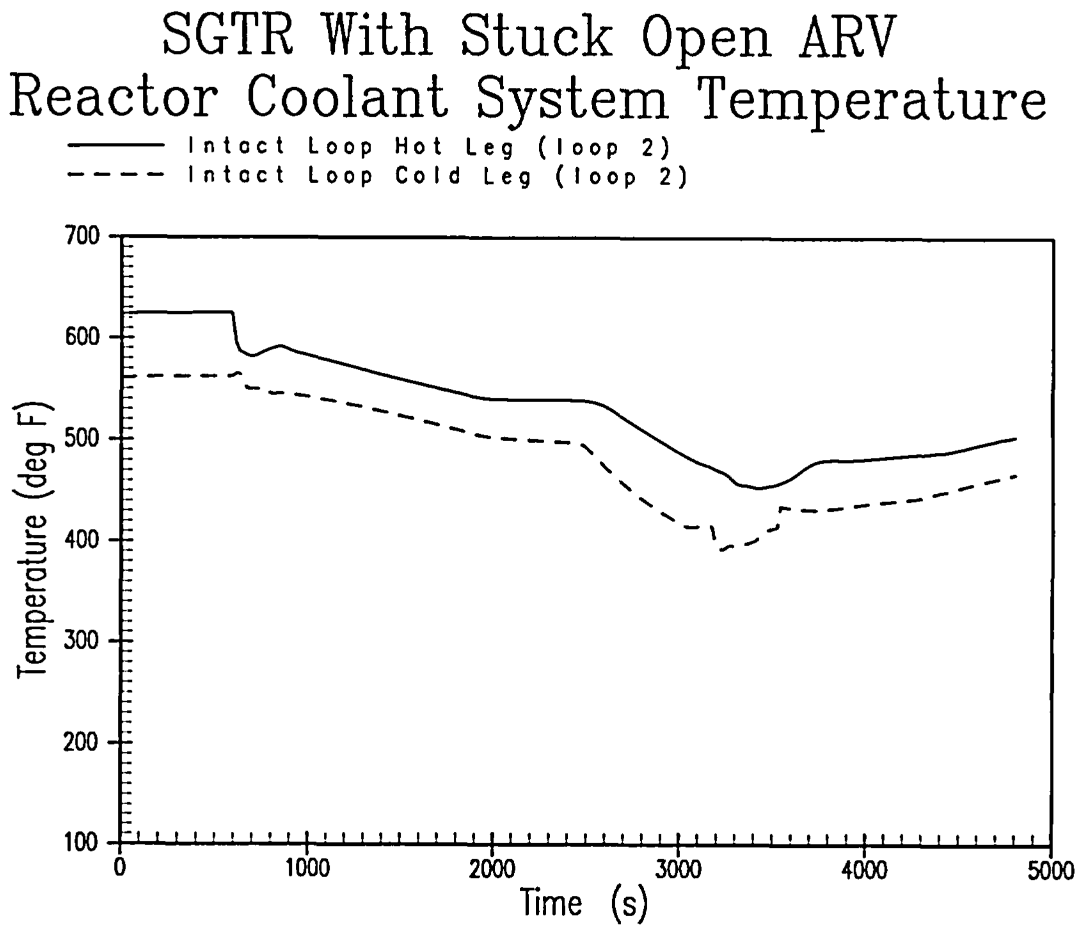


Figure 6.4-9 Intact Loop Temperatures

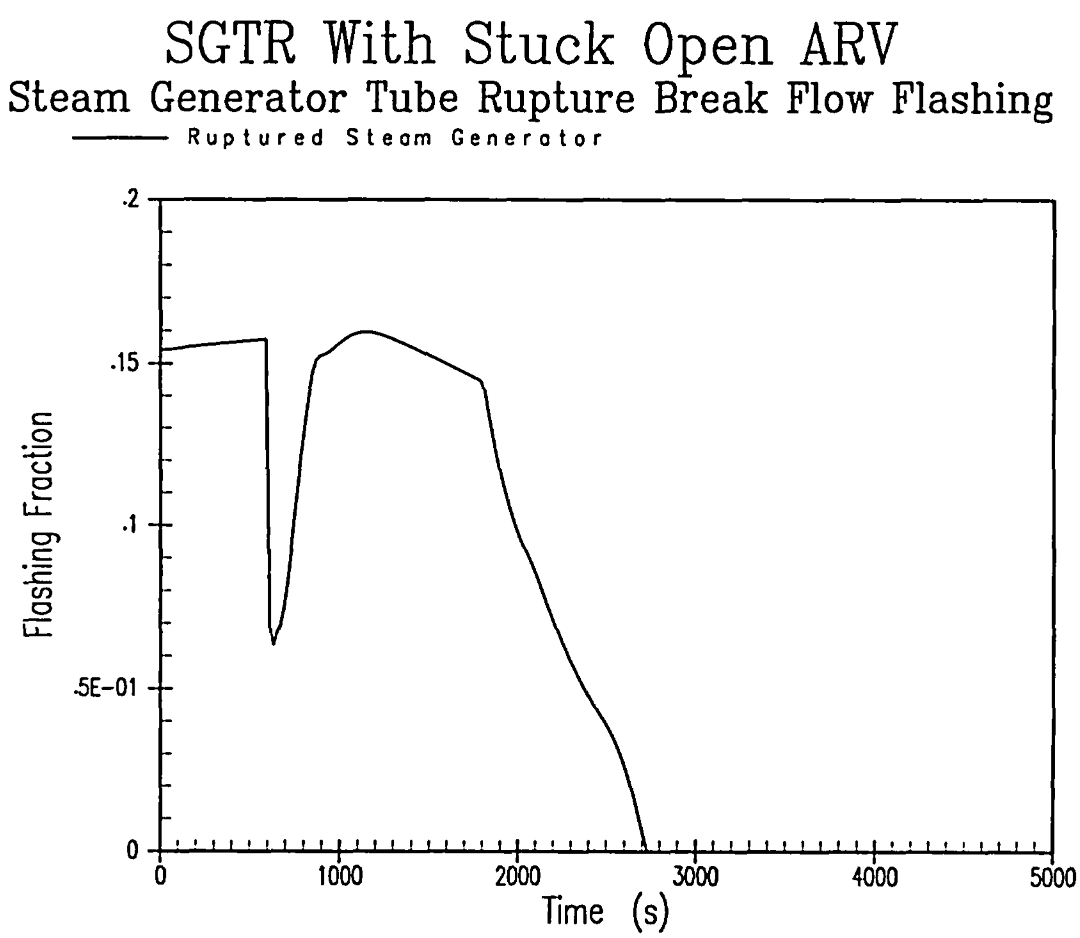


Figure 6.4-10 Break Flow Flashing Fraction

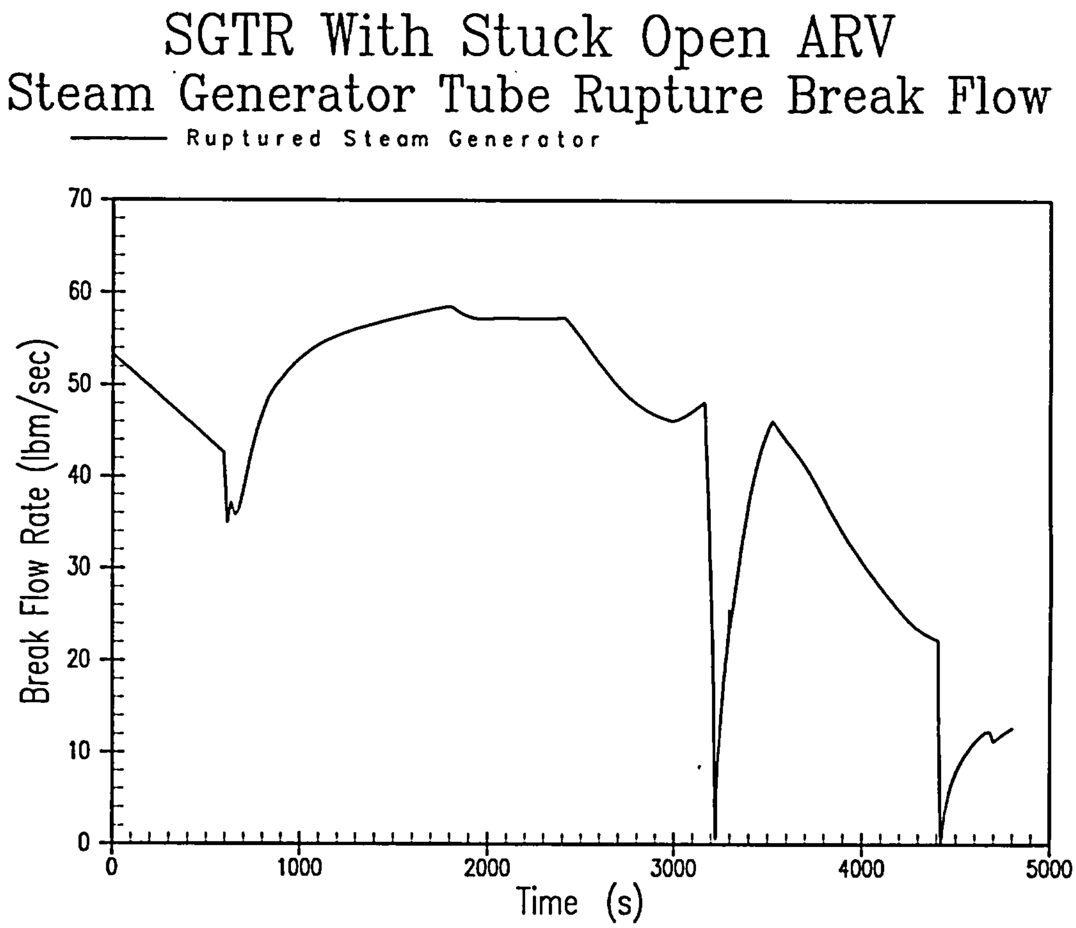


Figure 6.4-11 Primary-to-Secondary Break Flow

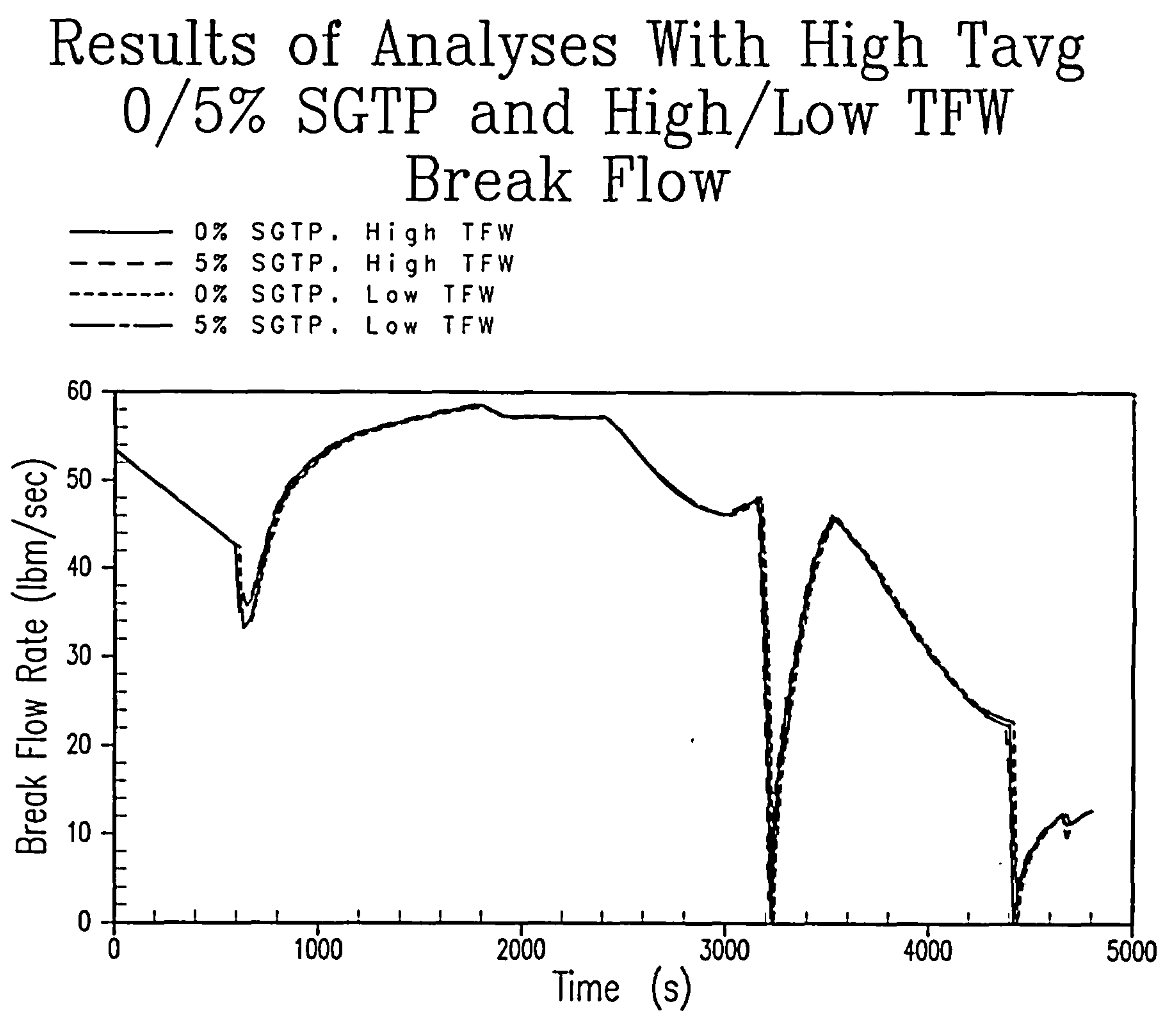


Figure 6.4-12 **Primary-to-Secondary Break Flow – SGTP/Feedwater Temperature Sensitivity**

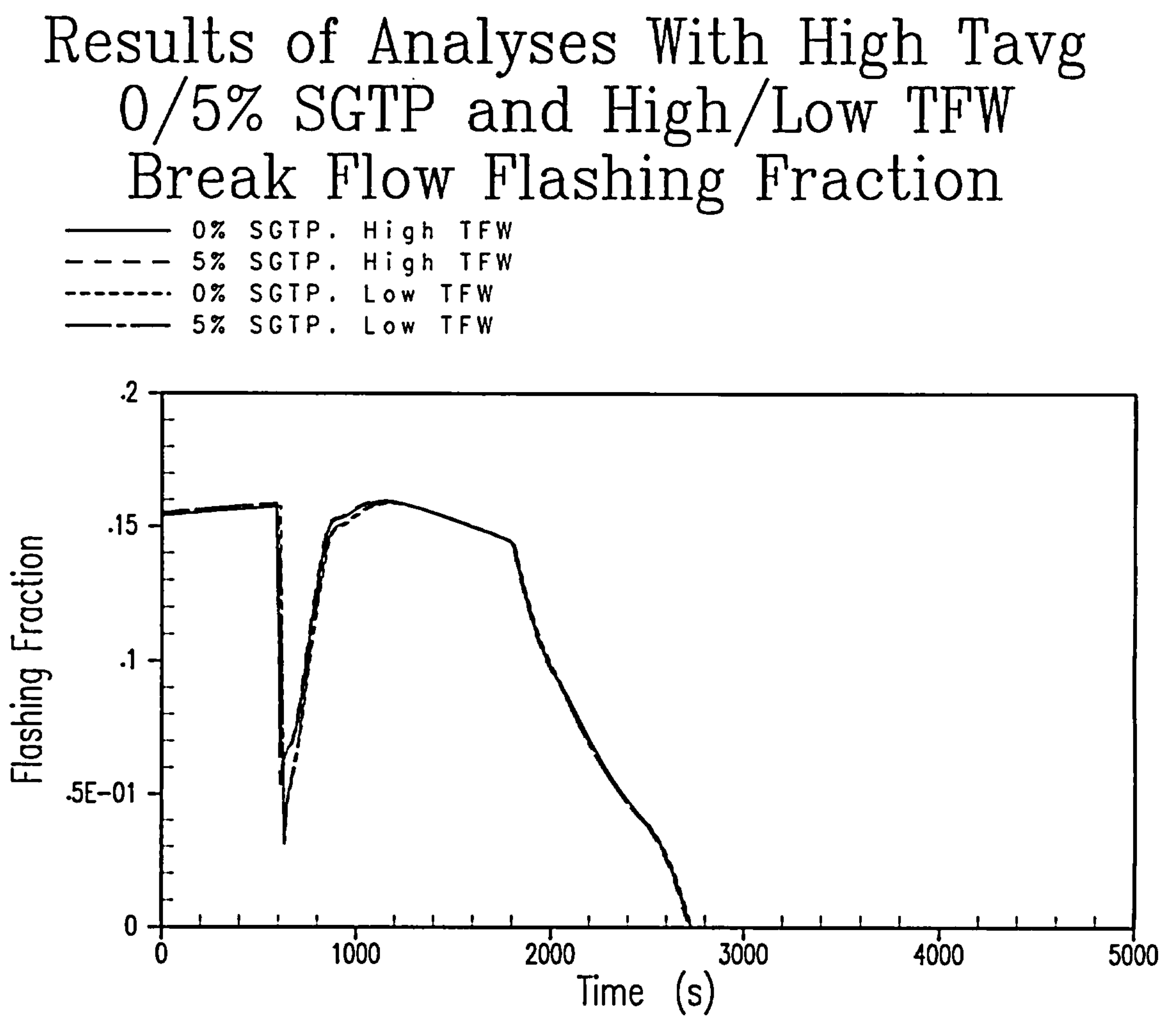


Figure 6.4-13 **Break Flow Flashing Fraction – SGTP/ Feedwater Temperature Sensitivity**

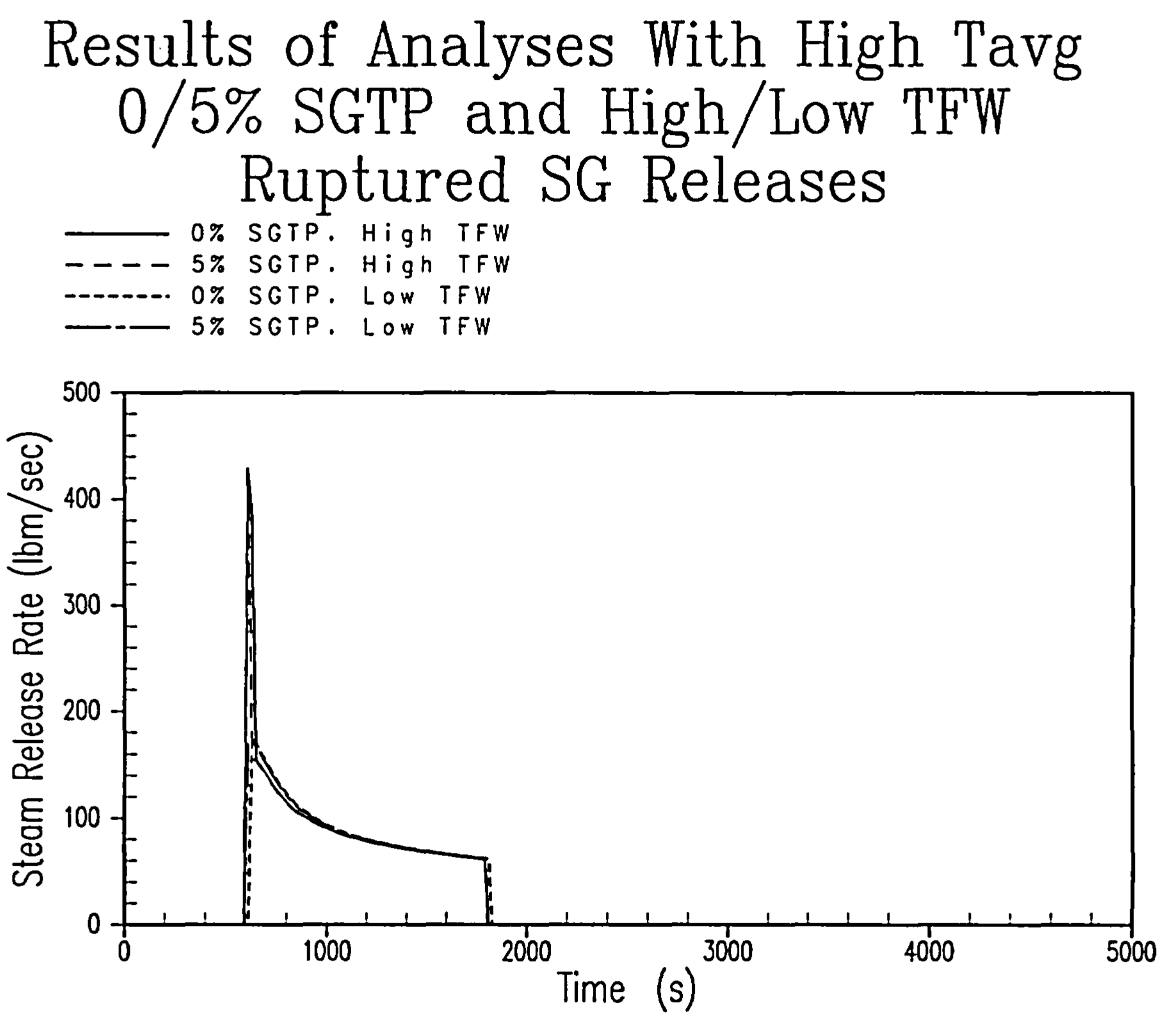


Figure 6.4-14 **Ruptured Steam Generator Atmospheric Release – SGTP/ Feedwater Temperature Sensitivity**

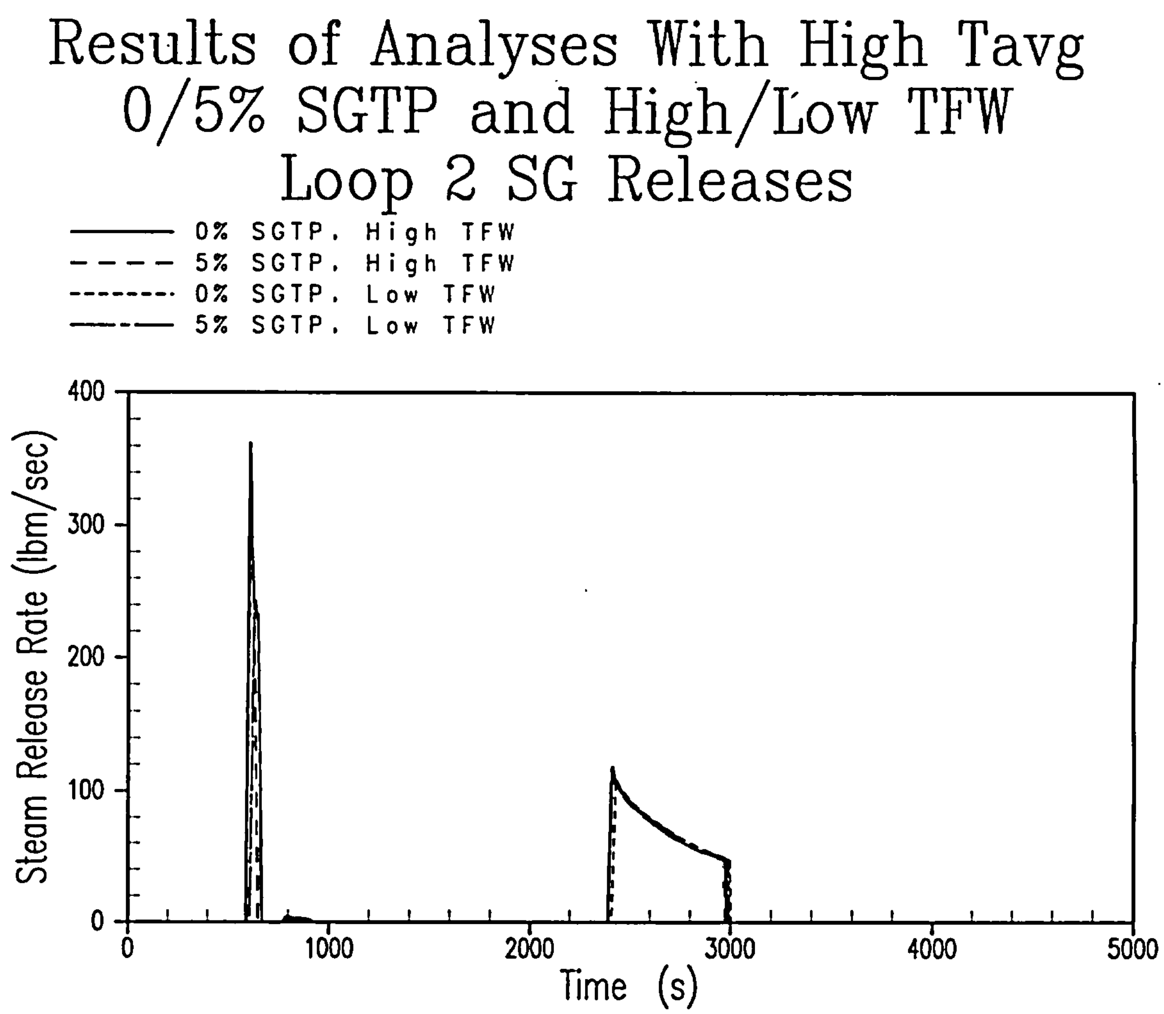


Figure 6.4-15 **Intact Steam Generators Atmospheric Release – SGTP/ Feedwater Temperature Sensitivity**

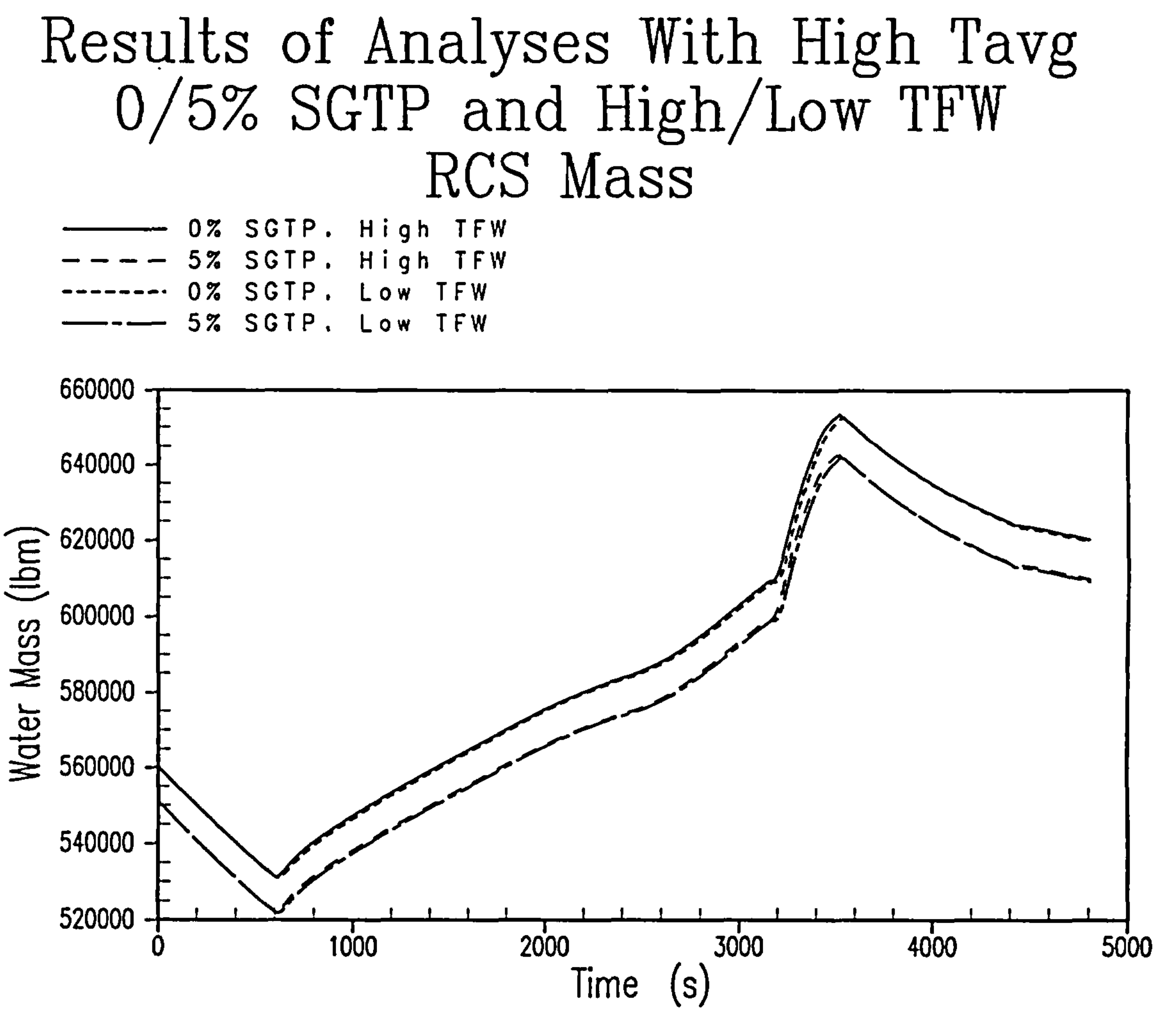


Figure 6.4-16 **Reactor Coolant System Liquid Mass – SGTP/ Feedwater Temperature Sensitivity**

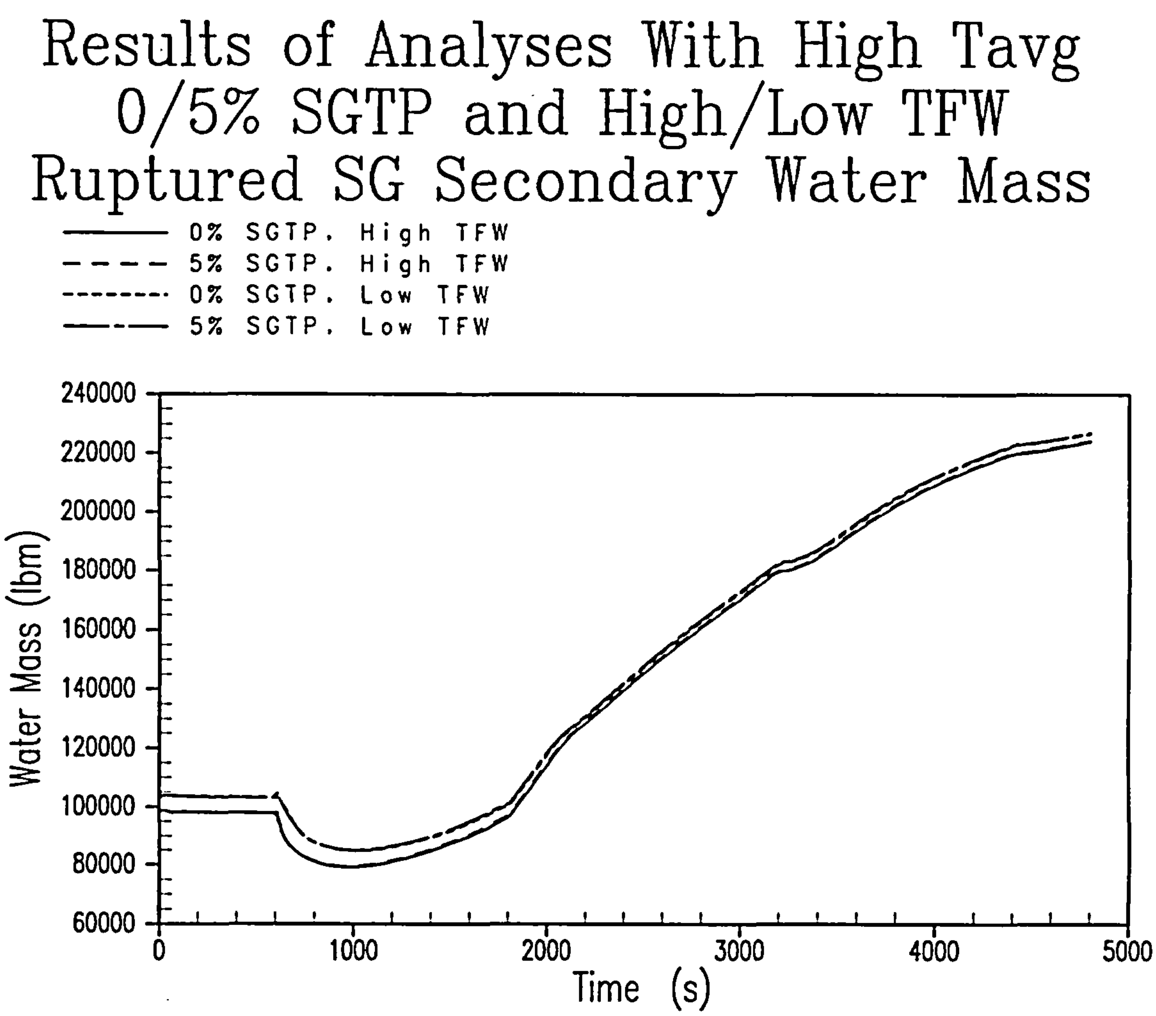


Figure 6.4-17 **Ruptured Steam Generator Liquid Mass – SGTP/ Feedwater Temperature Sensitivity**

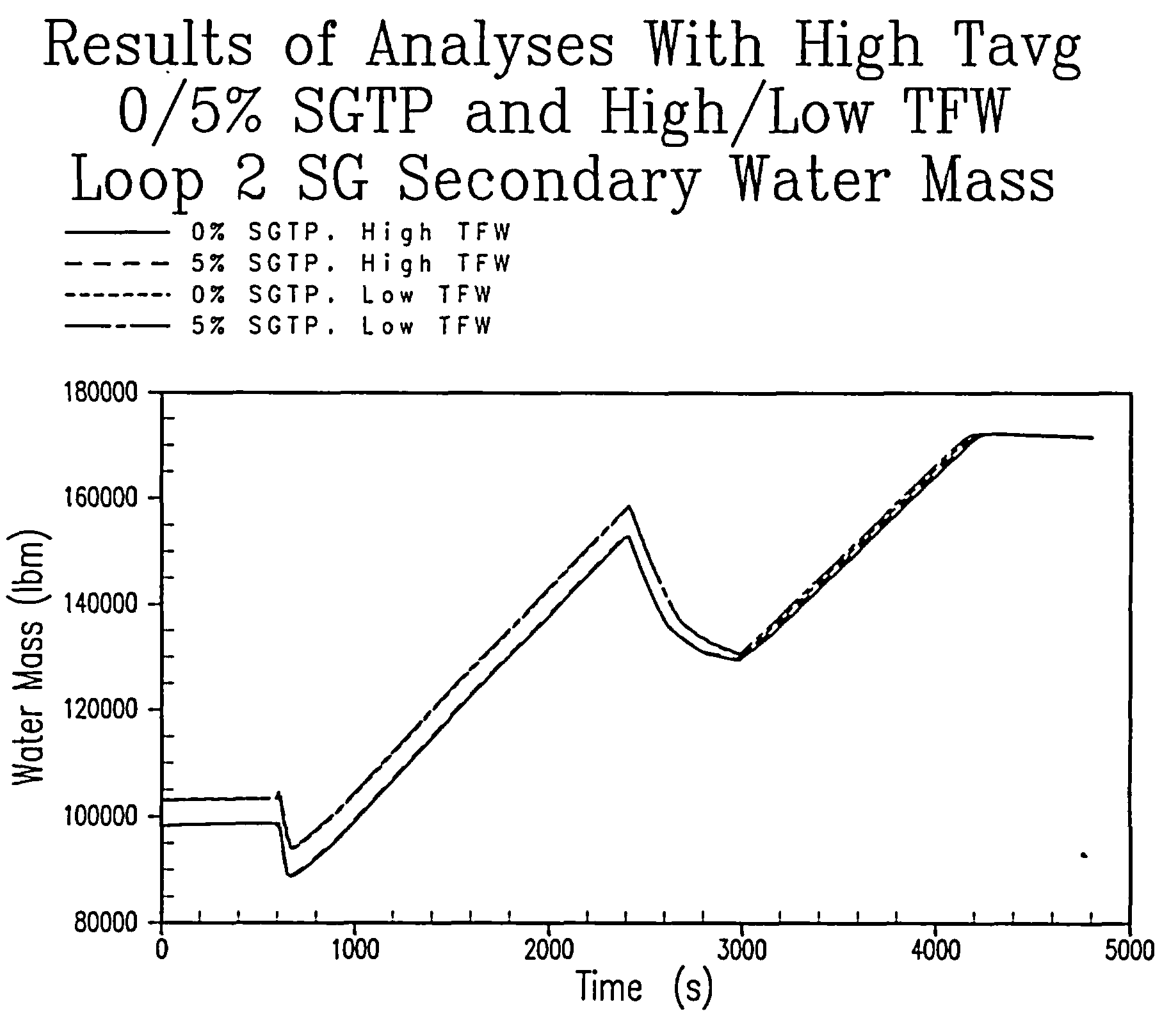


Figure 6.4-18 **Intact Steam Generator Liquid Mass – SGTP/ Feedwater Temperature Sensitivity**

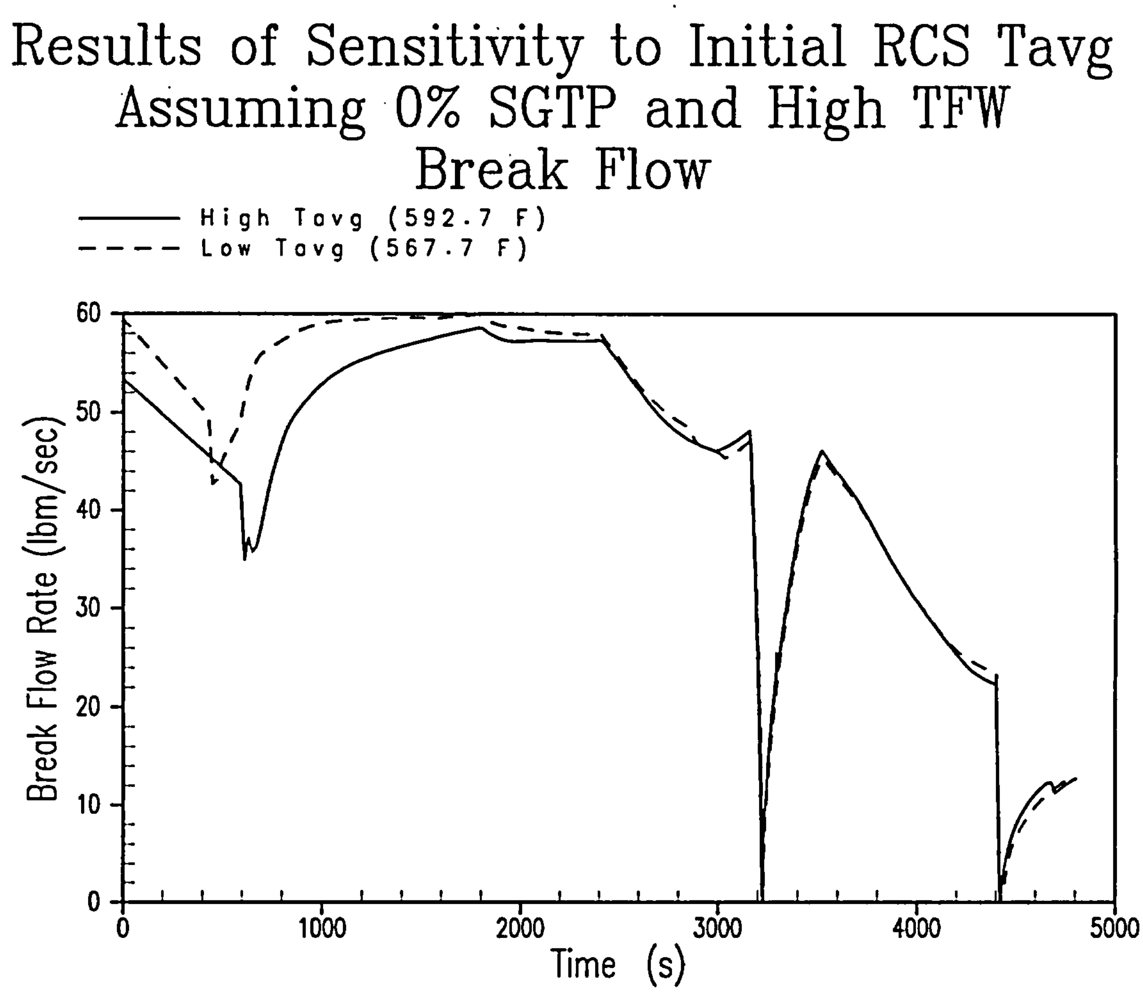


Figure 6.4-19 **Primary-to-Secondary Break Flow Rate – Tavg Sensitivity**

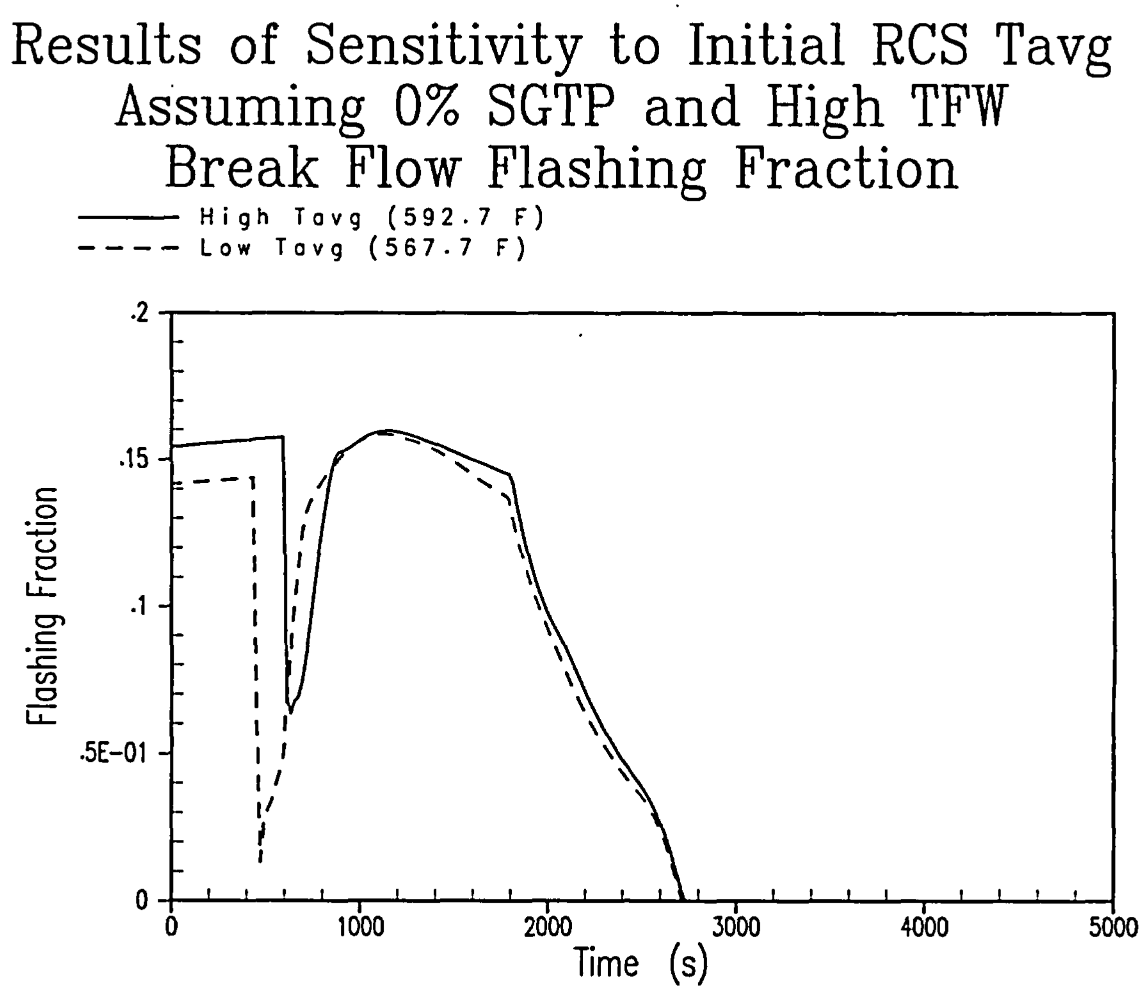


Figure 6.4-20 **Break Flow Flashing Fraction – Tavg Sensitivity**

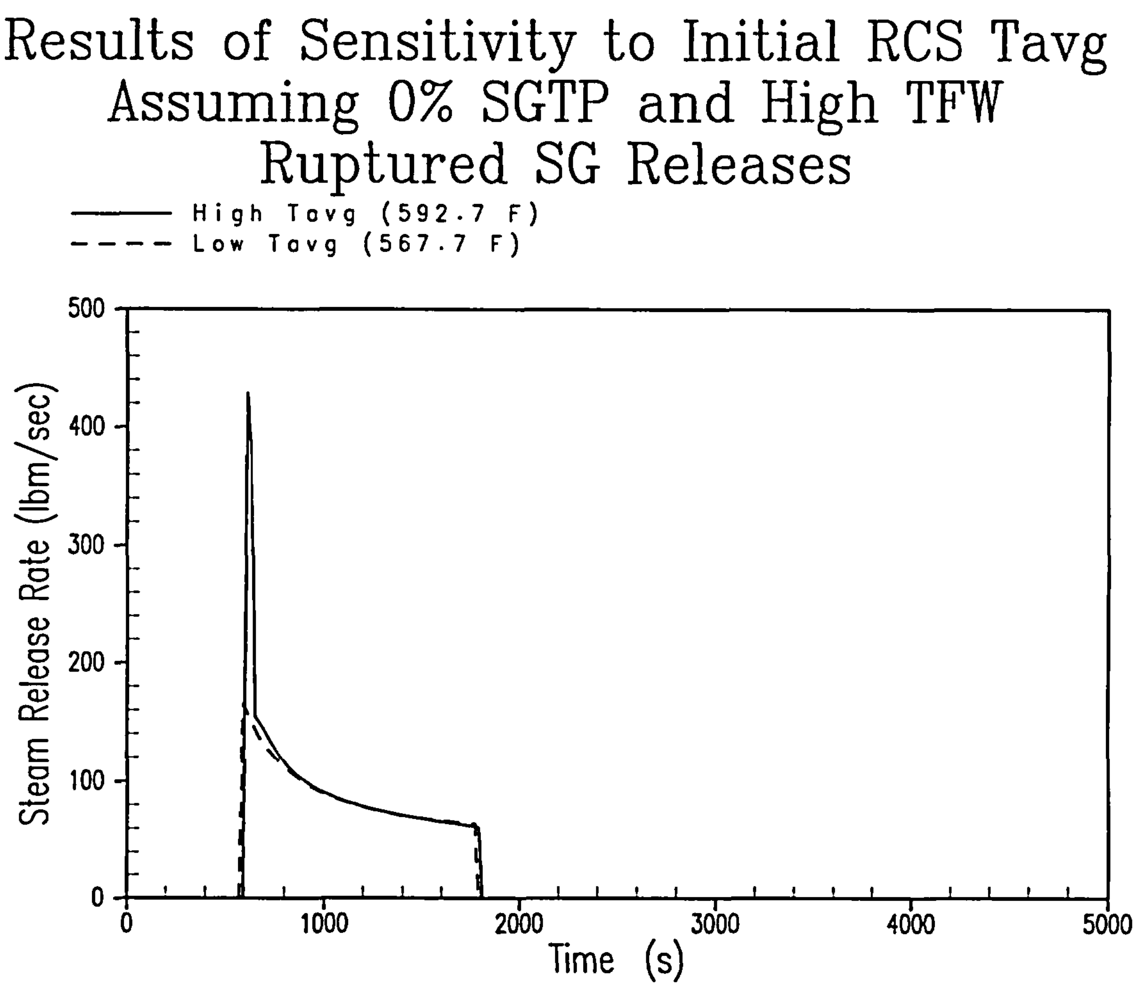


Figure 6.4-21 **Ruptured Steam Generator Atmospheric Release – Tavg Sensitivity**

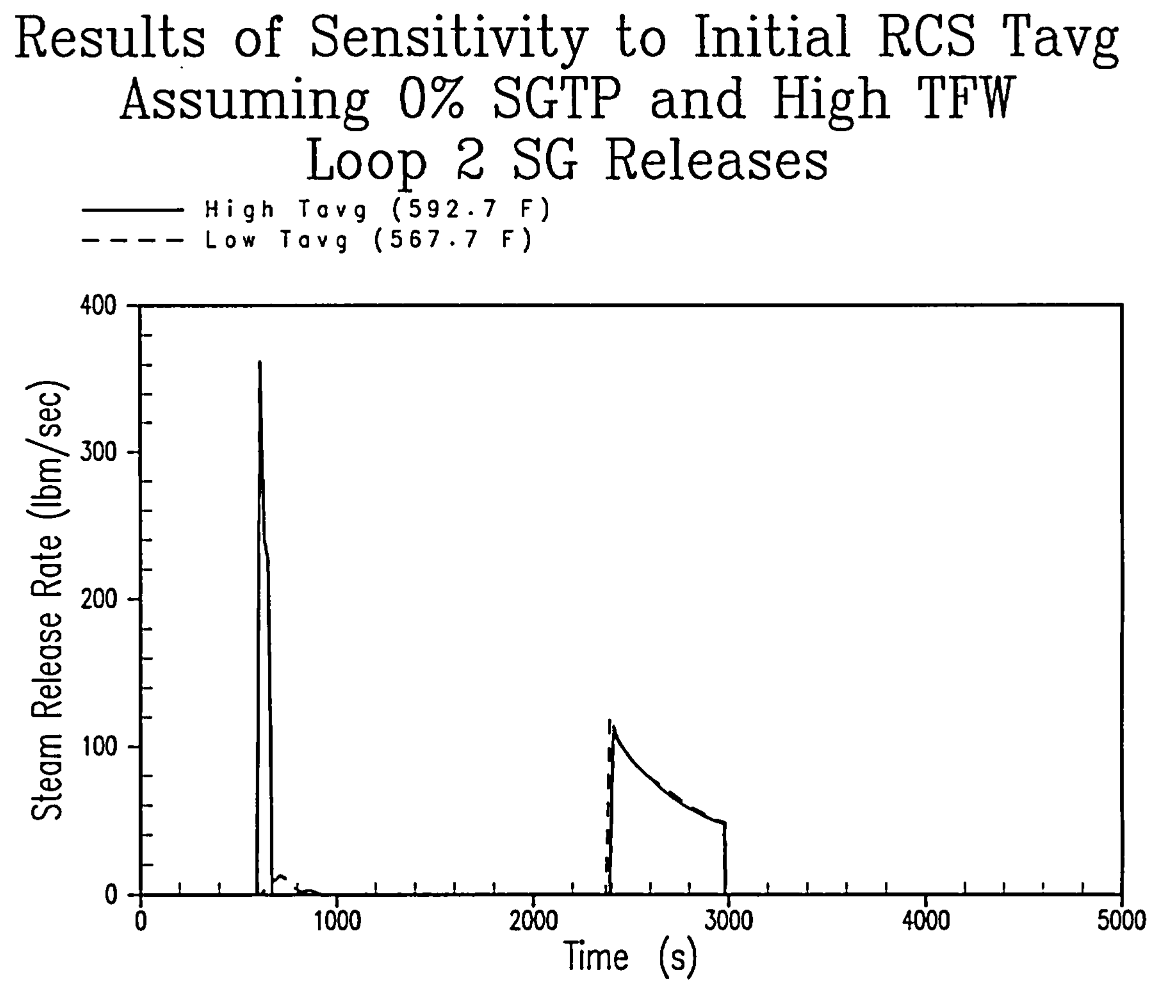


Figure 6.4-22 **Intact Steam Generators Atmospheric Release – Tavg Sensitivity**

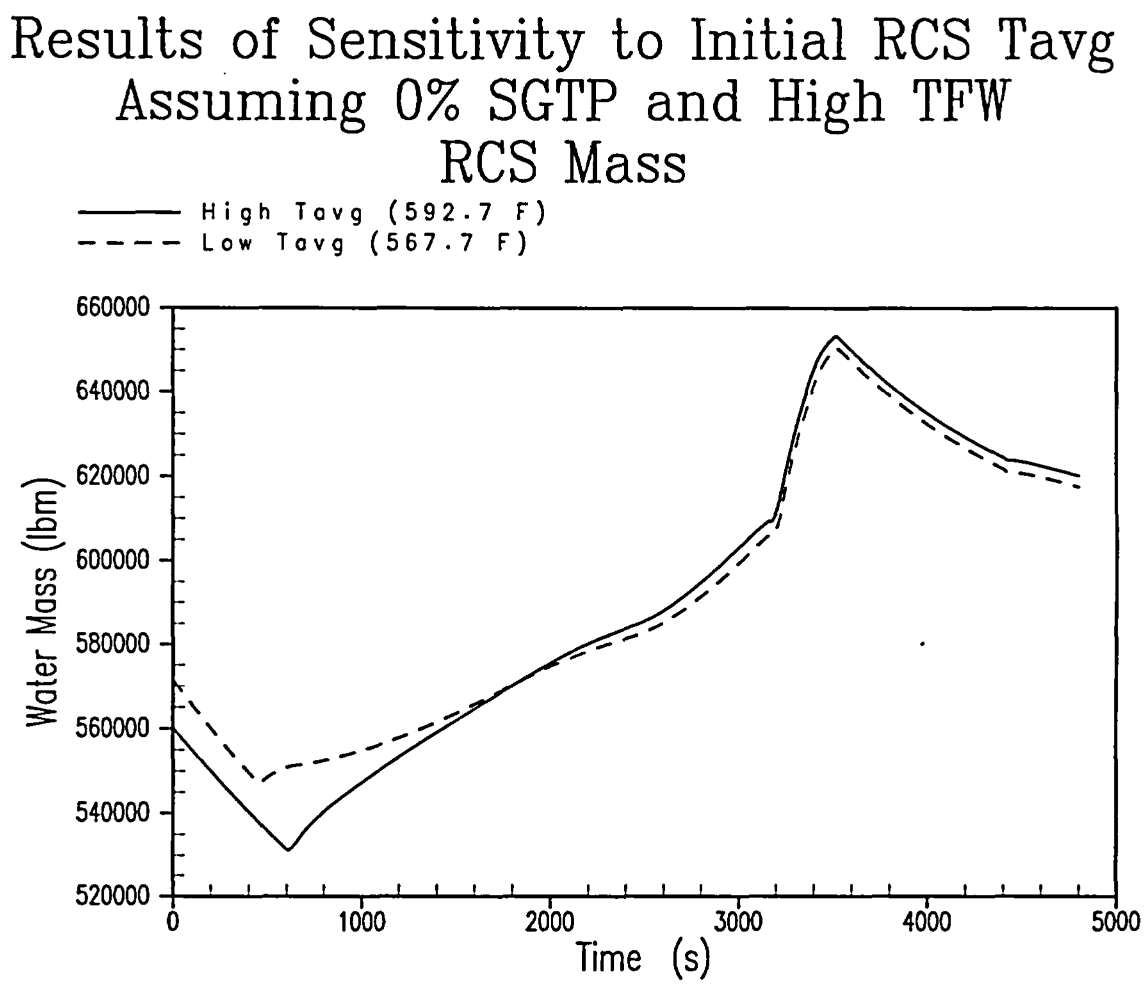


Figure 6.4-23 **Reactor Coolant System Liquid Mass – Tavg Sensitivity**

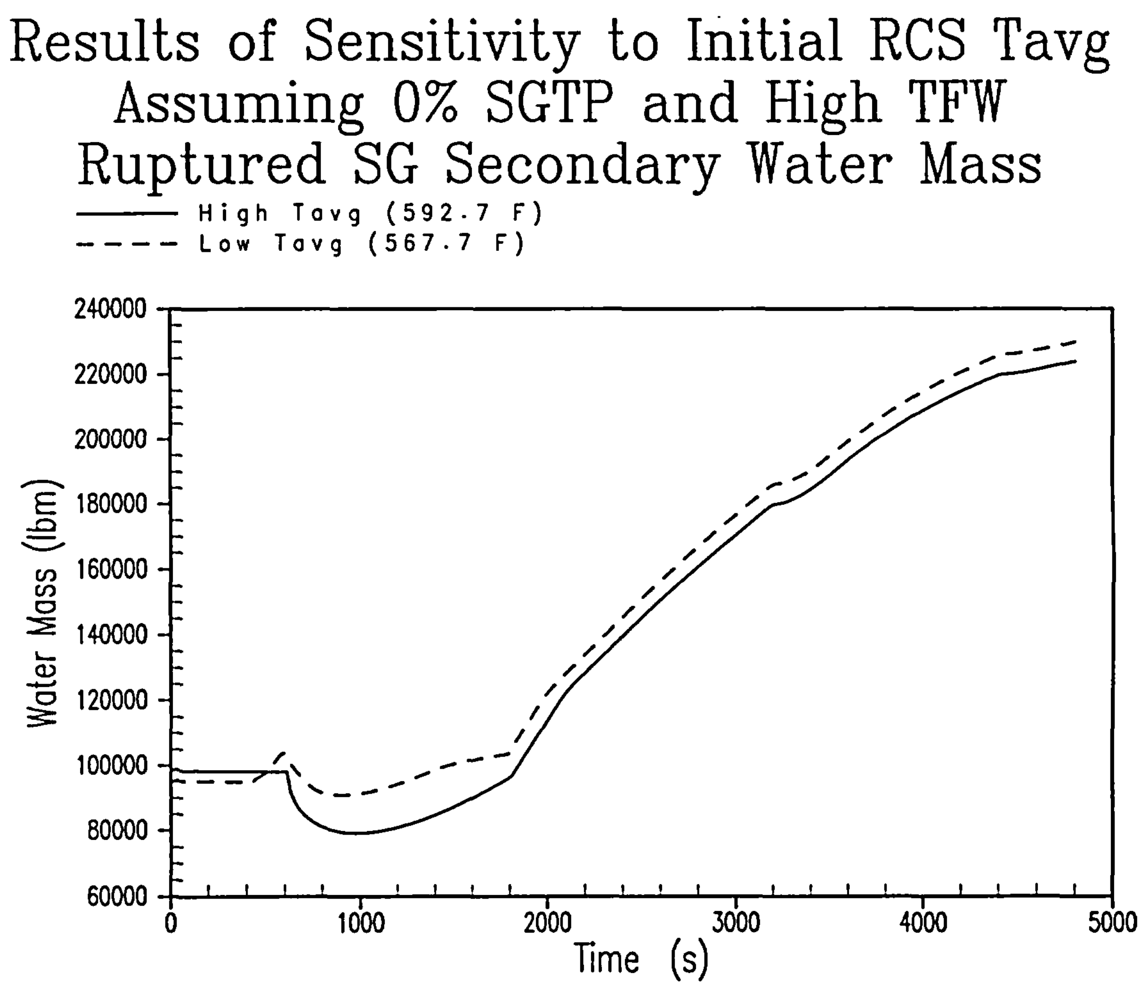


Figure 6.4-24 **Ruptured Steam Generator Liquid Mass – Tavg Sensitivity**

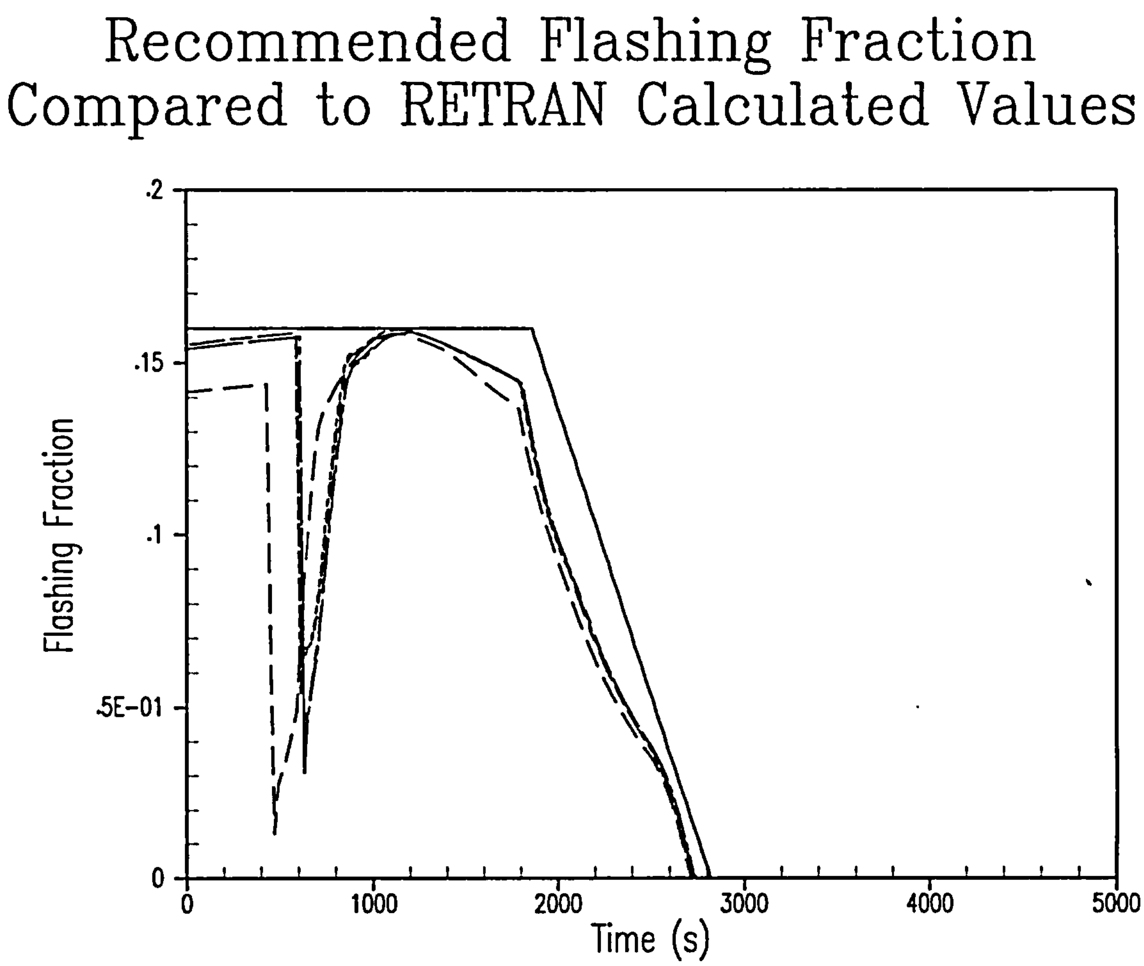


Figure 6.4-25 Comparison of Bounding Flashing Fraction to RETRAN Calculated Values

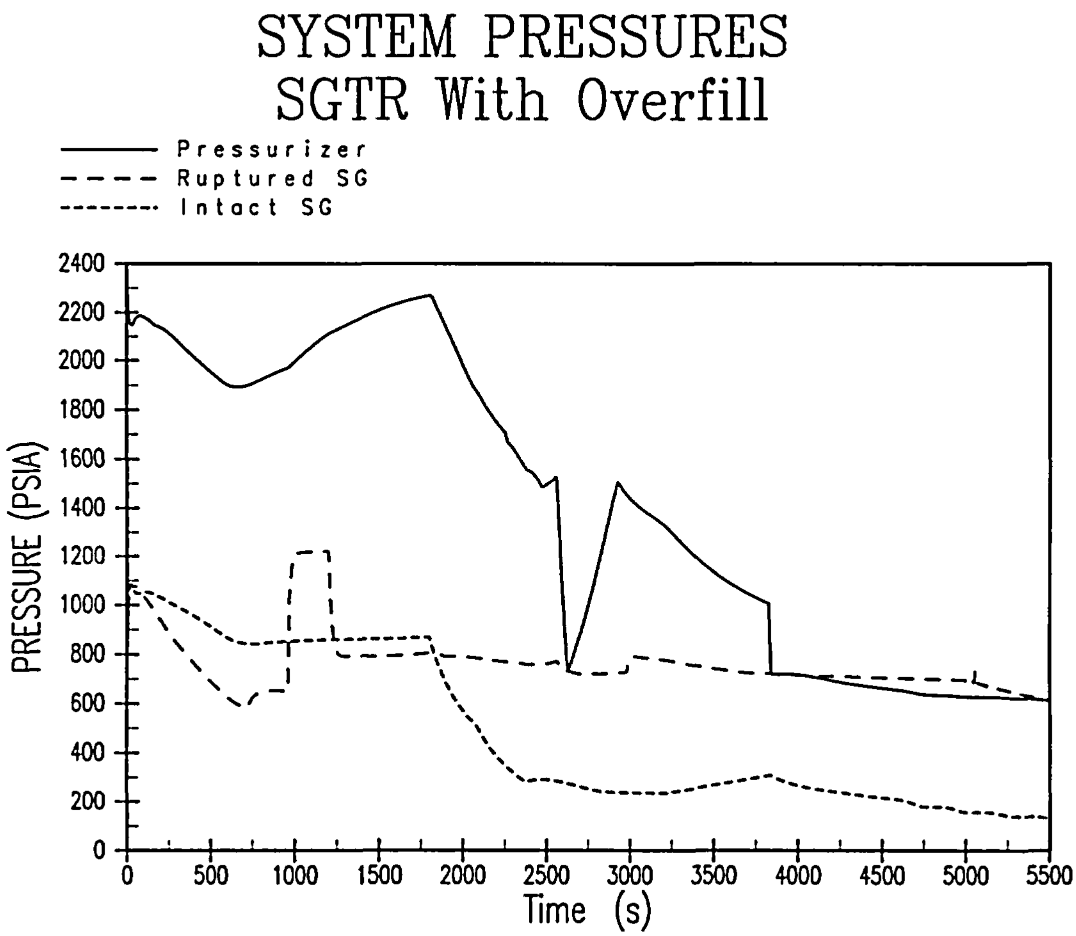


Figure 6.4-26 Primary and Secondary Pressure

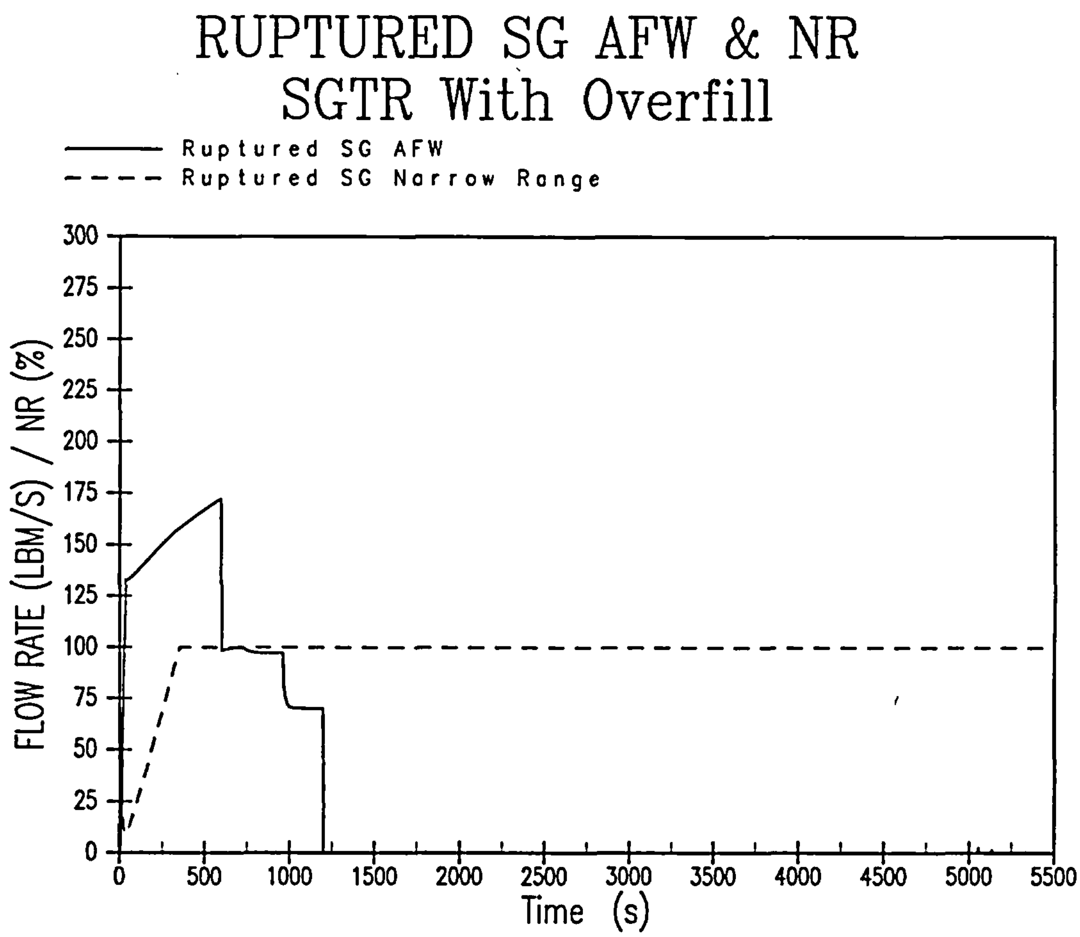


Figure 6.4-27 Ruptured Steam Generator AFW Flow and Steam Generator Level

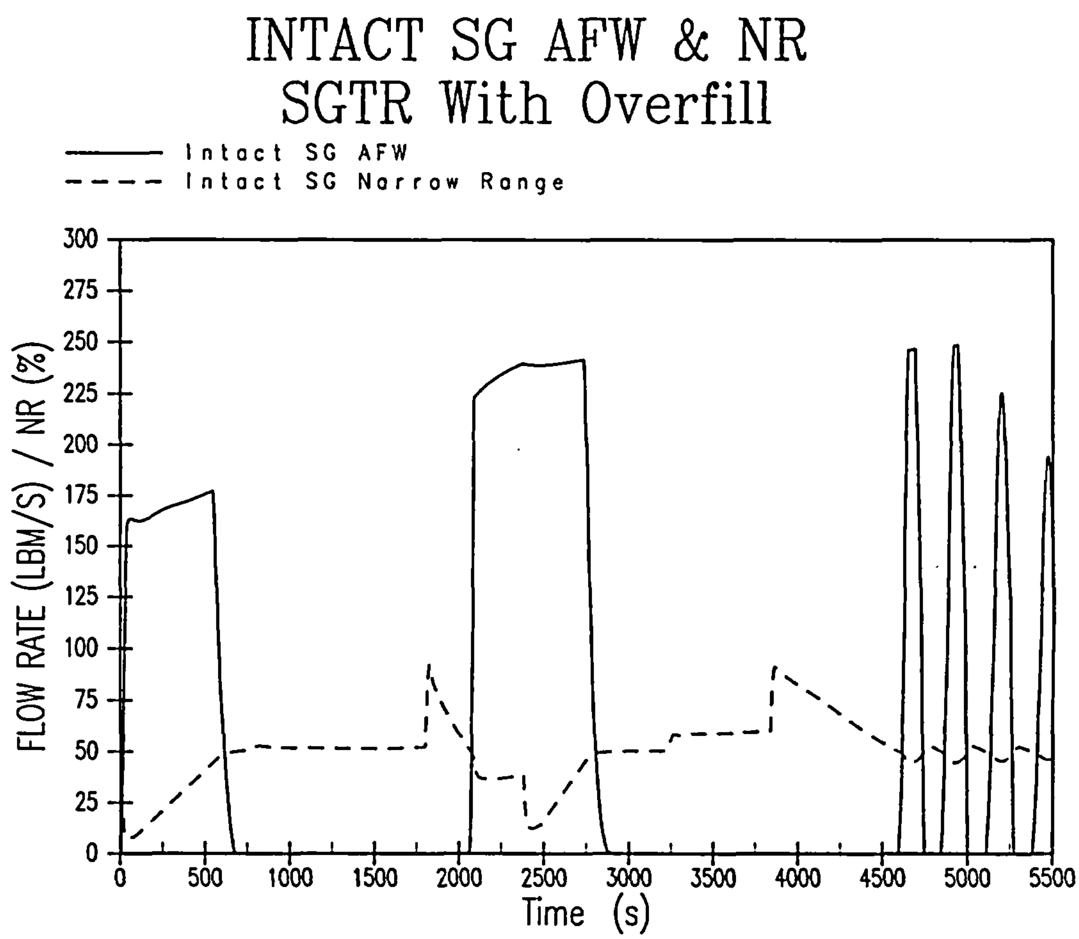


Figure 6.4-28 Intact Steam Generator AFW Flow and Steam Generator Level

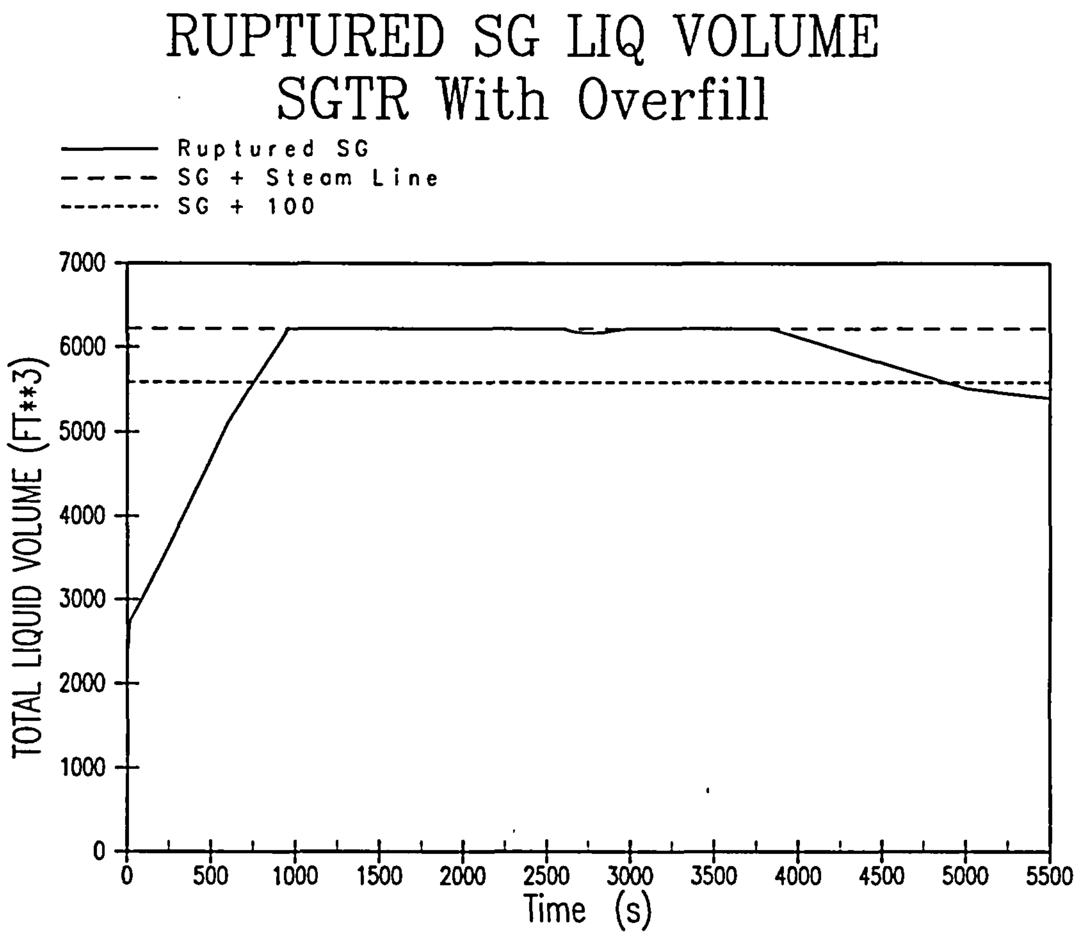


Figure 6.4-29 Ruptured Steam Generator and Steam Line Liquid Volume

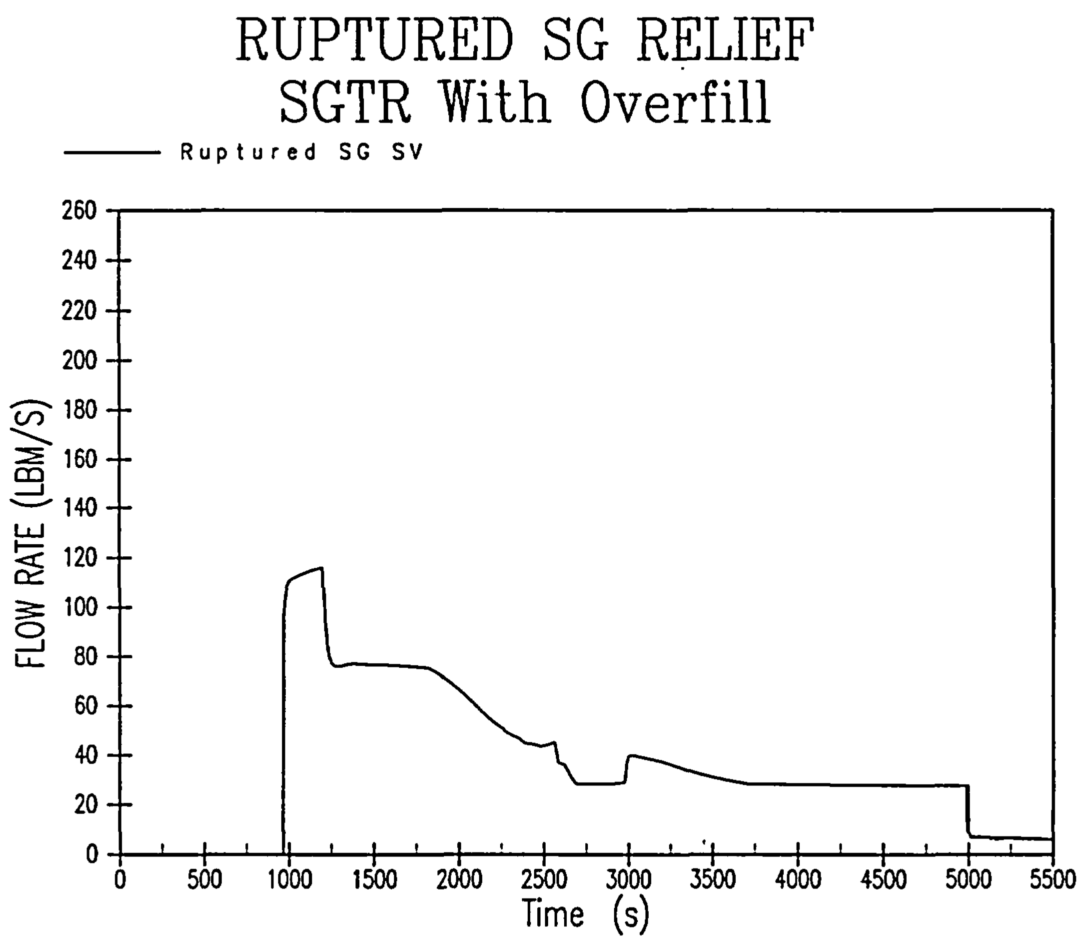


Figure 6.4-30 Ruptured Steam Generator Atmospheric Releases

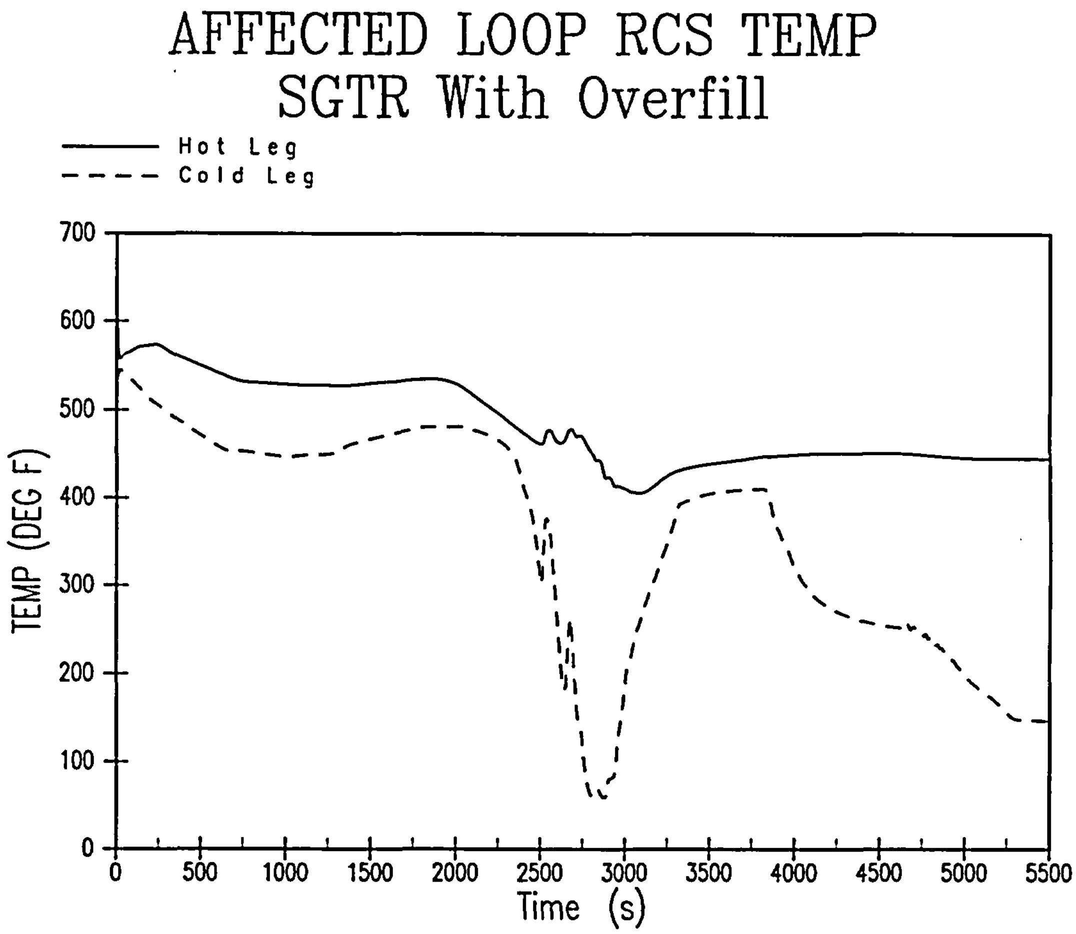


Figure 6.4-31 Ruptured Loop Temperatures

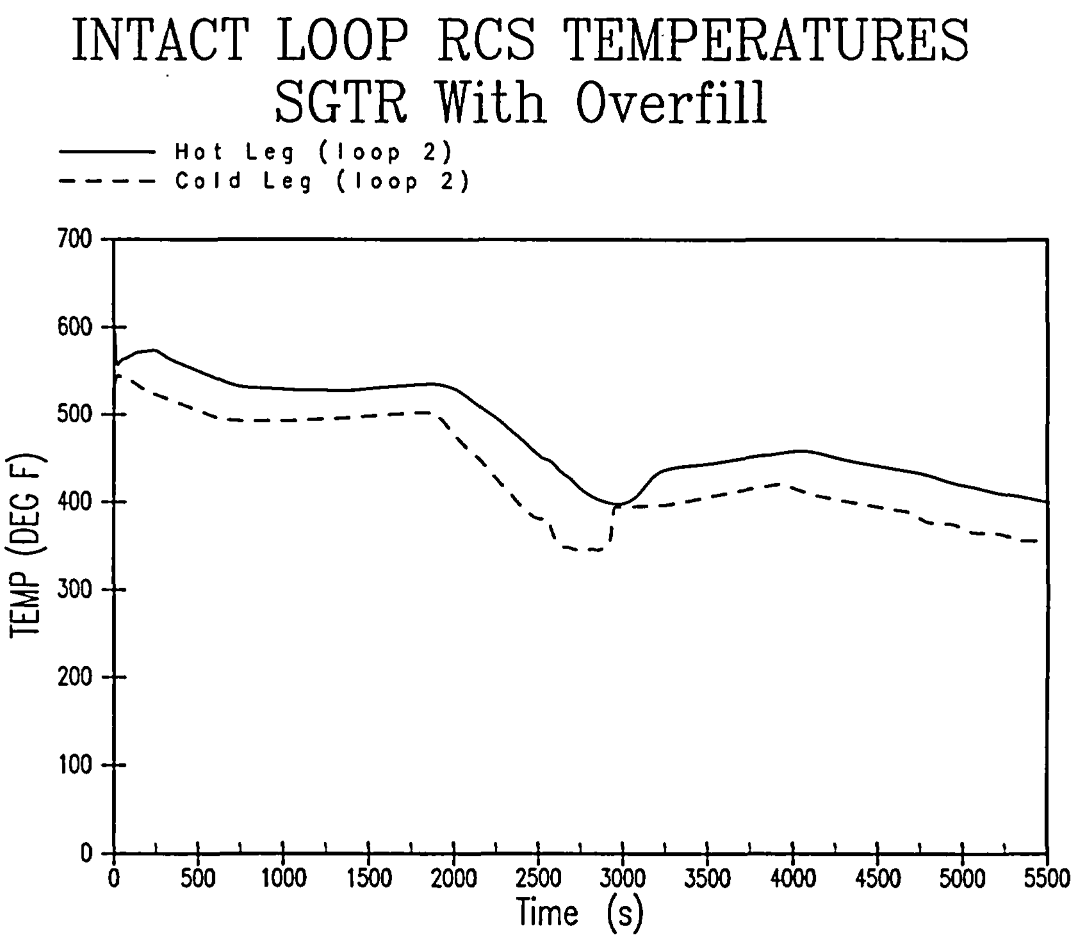


Figure 6.4-32 Intact Loop Temperatures

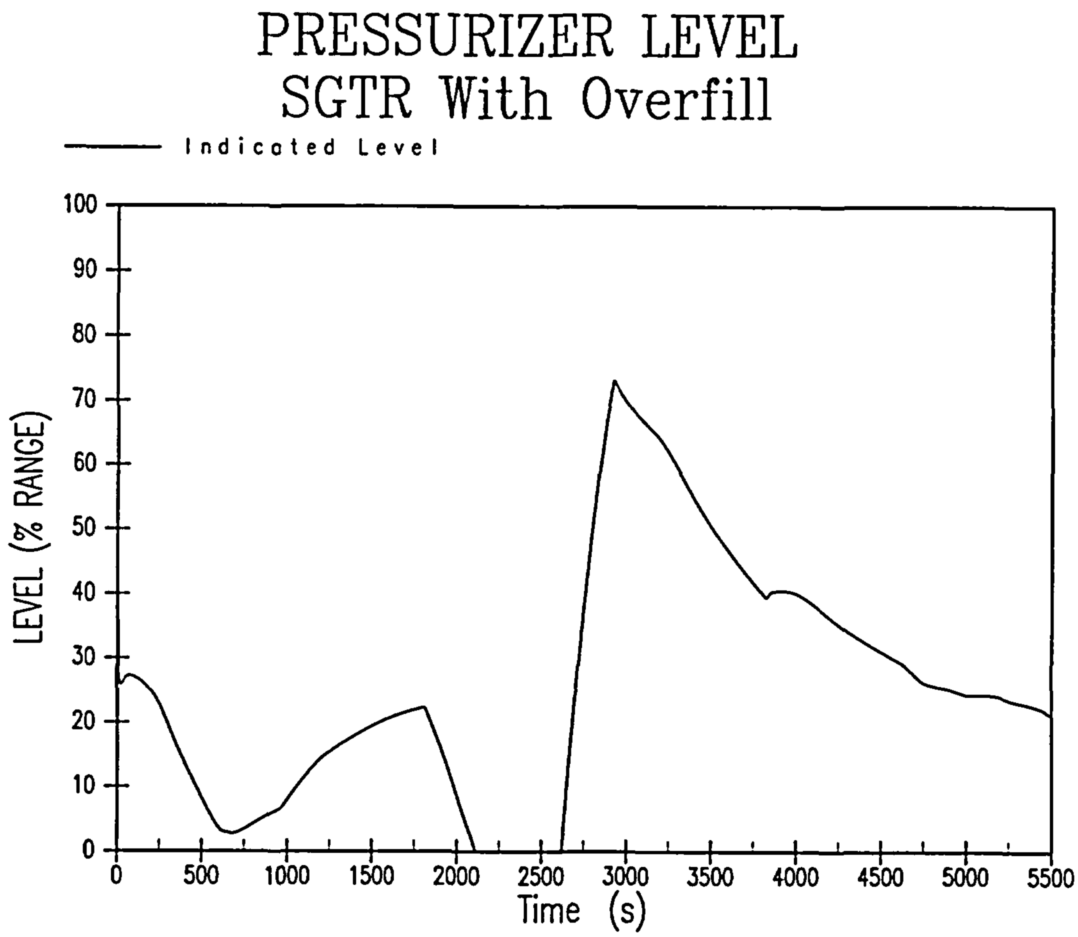


Figure 6.4-33 Pressurizer Level

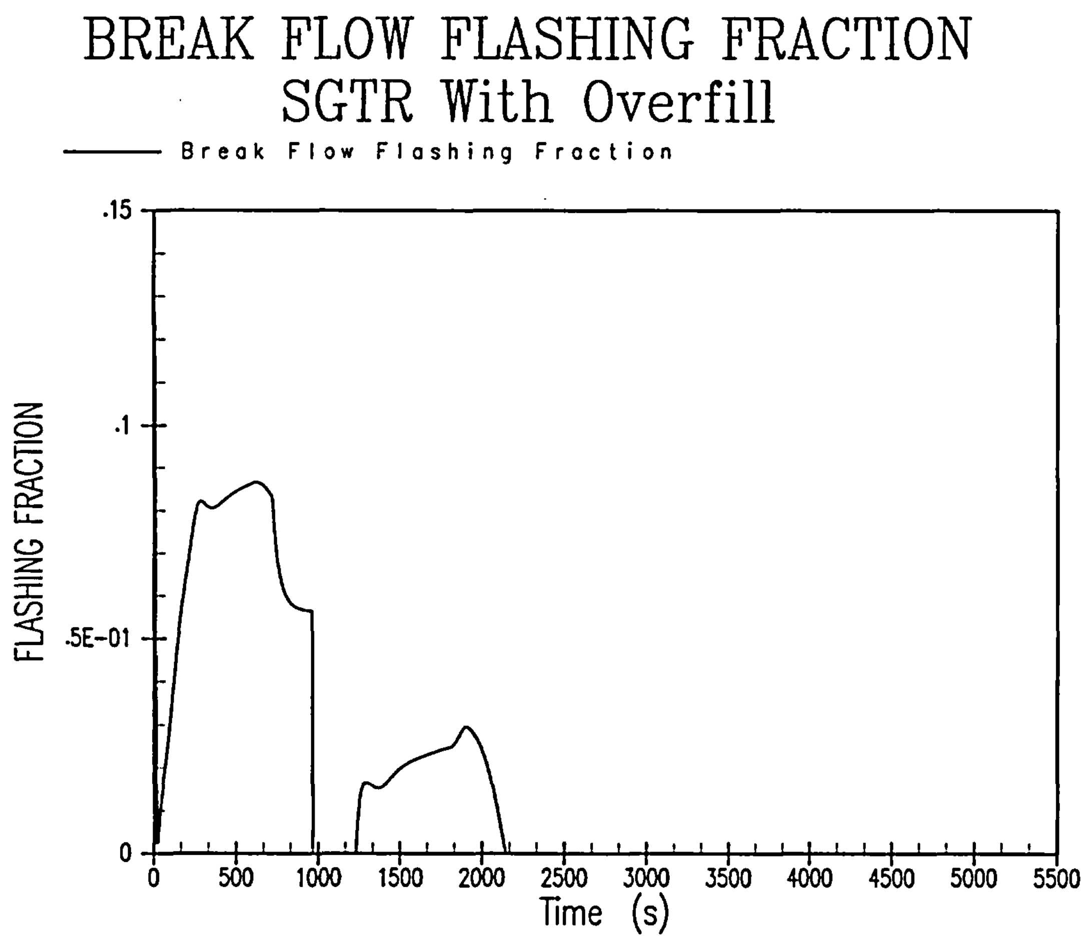


Figure 6.4-34 Break Flow Flashing Fraction

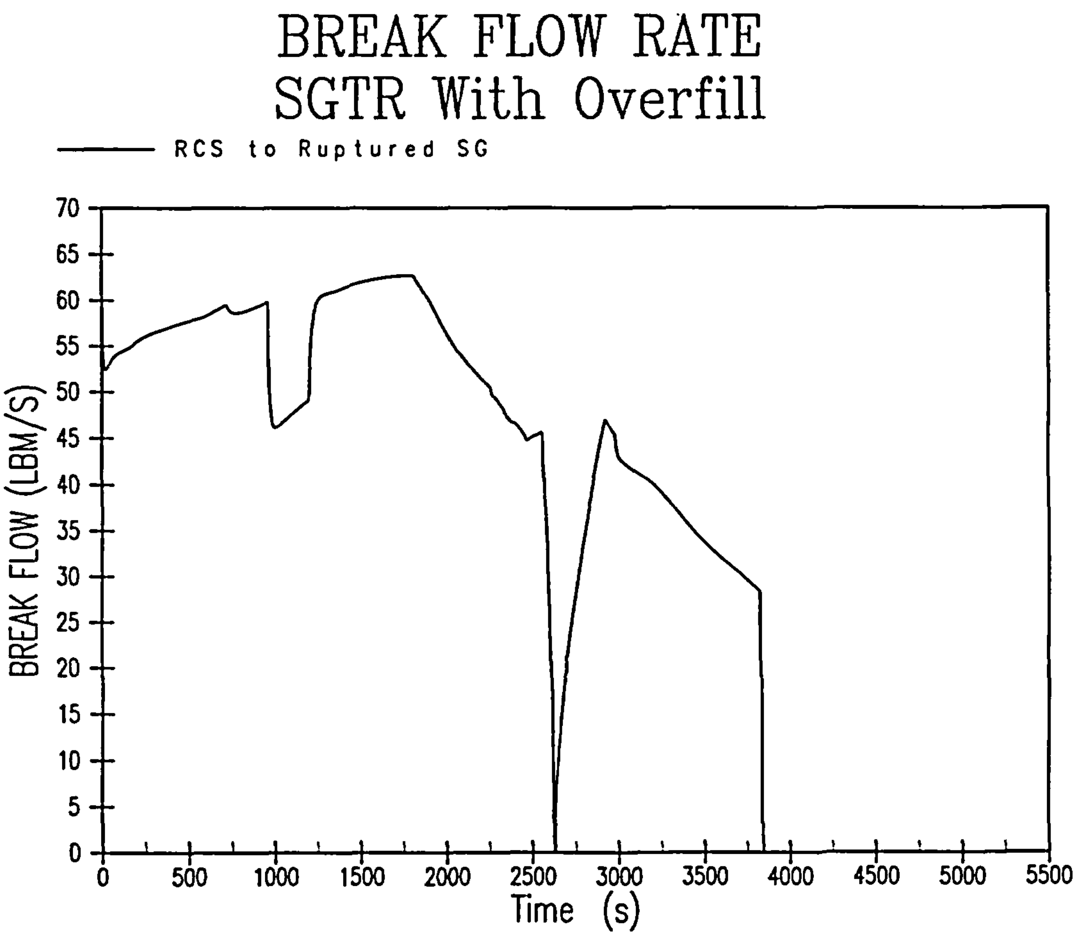


Figure 6.4-35 Primary-to-Secondary Break Flow

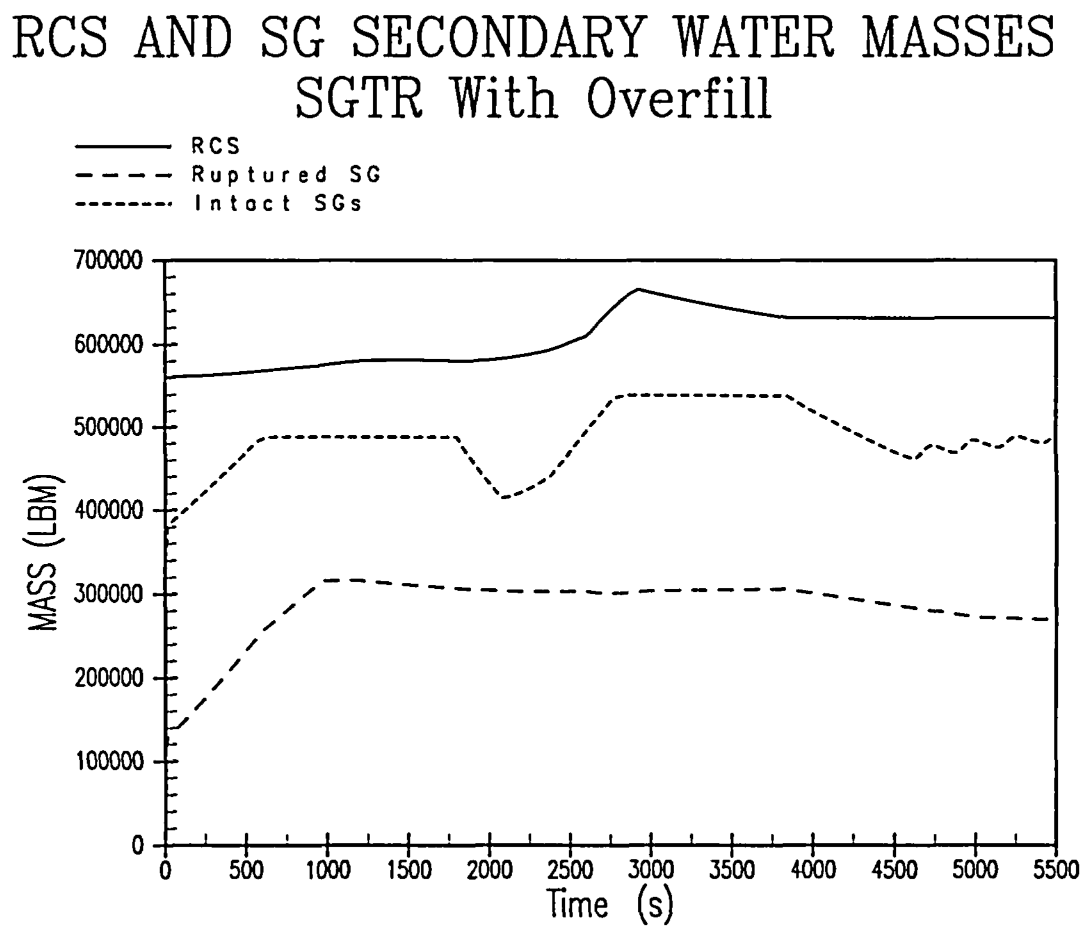


Figure 6.4-36 RCS, Ruptured Steam Generator and Intact Steam Generators Liquid Masses

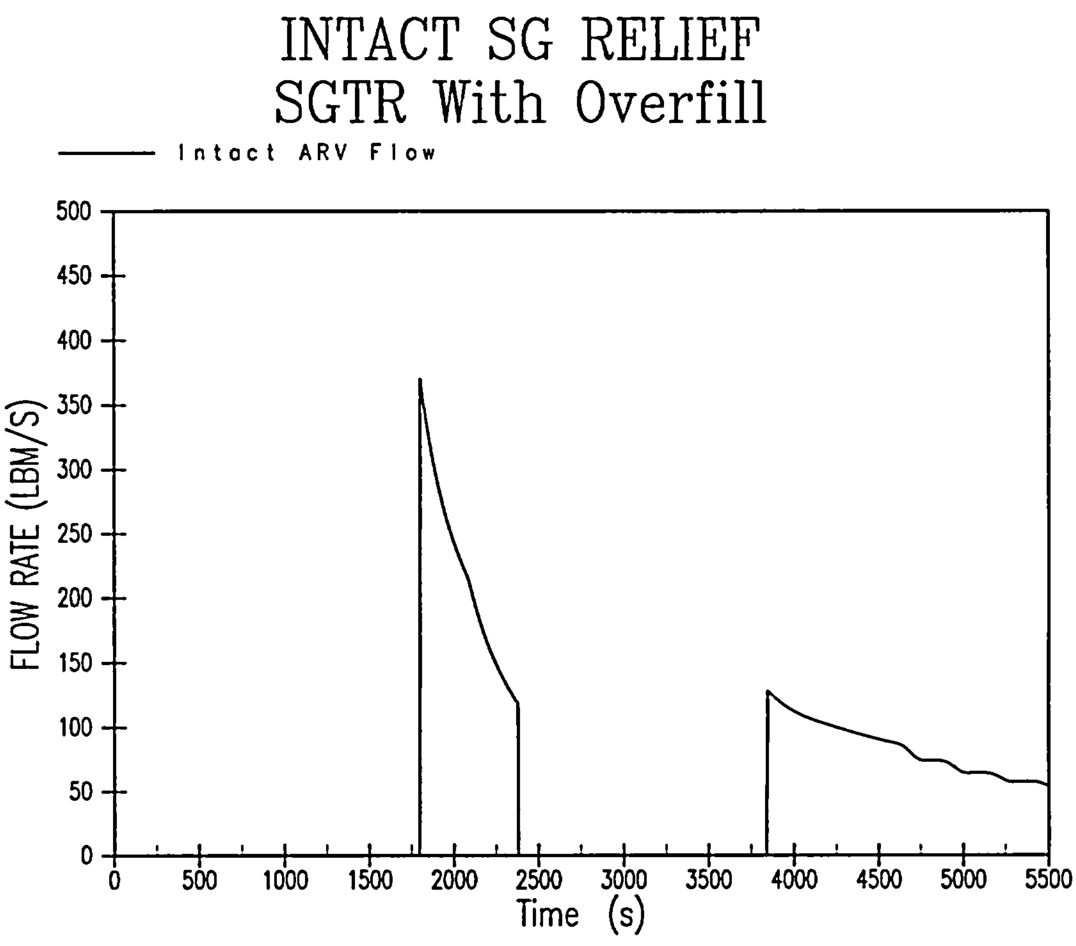


Figure 6.4-37 Intact Steam Generators Atmospheric Releases

6.5 LOCA MASS AND ENERGY RELEASE

The uncontrolled release of pressurized high temperature reactor coolant, termed a loss-of-coolant accident (LOCA), will result in release of steam and water into the containment. This, in turn, will result in increases in the local subcompartment pressures, and an increase in the global containment pressure and temperature. Therefore, there are both long- and short-term issues reviewed relative to a postulated LOCA that must be considered for the Replacement Steam Generator (RSG) Program for the Callaway Plant.

The long-term LOCA mass and energy releases are analyzed to approximately 10^7 seconds and are utilized as input to the containment integrity analysis, which demonstrates the acceptability of the containment safeguards systems to mitigate the consequences of a hypothetical large-break LOCA. The containment safeguards systems must be capable of limiting the peak containment pressure to less than the design pressure and to limit the temperature excursion to less than the environmental qualification (EQ) acceptance limits. For this program, Westinghouse generated the mass and energy releases using the March 1979 model, described in Reference 1. The Nuclear Regulatory Commission (NRC) review and approval letter is included with Reference 1. Even though this is a first time application of this methodology for the Callaway Plant, it has also been utilized and approved on many plant-specific dockets. Section 6.5.1 discusses the long-term LOCA mass and energy releases generated for this program. The results of this analysis were provided for use in the containment integrity analysis.

The short-term LOCA-related mass and energy releases are used as input to the subcompartment analyses, which are performed to ensure that the walls of a subcompartment can maintain their structural integrity during the short pressure pulse (generally less than 3 seconds) accompanying a high energy line pipe rupture within that subcompartment. The subcompartments evaluated include the steam generator compartment and the pressurizer compartment. For the steam generator compartment, the fact that Callaway Plant is approved for leak before break (LBB) on the main reactor coolant system (RCS) loop was used to qualitatively demonstrate that any changes associated with the replacement steam generator and Tavg reduction program are offset by the LBB benefit of using the smaller RCS nozzle breaks, thus demonstrating that the current licensing basis for this subcompartment remains bounding. AmerenUE plans to license LBB for the pressurizer surge line as part of the licensing effort for the replacement steam generators. Assuming surge line LBB is licensed, any changes associated with the replacement steam generator and the Tavg reduction program will be offset by the LBB benefit and the current Final Safety Analysis Report (FSAR) releases documented in the Callaway Plant FSAR will remain bounding for the pressurizer subcompartment. Section 6.5.2 discusses the short-term evaluation conducted for this program.

6.5.1 Long-Term LOCA Mass and Energy Releases

The mass and energy release rates described in this section form the basis for the containment pressure calculations to be performed by AmerenUE/Callaway. Discussed in this section are the long-term LOCA mass and energy releases for the hypothetical double-ended pump suction (DEPS) rupture and double-ended hot leg (DEHL) rupture break cases for the Callaway Plant with the replacement steam generators.

6.5.1.1 Input Parameters and Assumptions

The mass and energy release analysis is sensitive to the assumed characteristics of various plant systems, in addition to other key modeling assumptions. Where appropriate, bounding inputs are utilized and instrumentation uncertainties are included. For example, the RCS operating temperatures are chosen to bound the highest average coolant temperature range of all operating cases, and a temperature uncertainty allowance of (+4.3°F) is then added. Nominal parameters are used in certain instances. For example, the RCS pressure in this analysis is based on a nominal value of 2,250 psia plus an uncertainty allowance (+30 psi). All input parameters are chosen consistent with accepted analysis methodology.

Some of the most critical items are the RCS initial conditions, core decay heat, safety injection (SI) flow, and primary and secondary metal mass and steam generator (SG) heat release modeling. Specific assumptions concerning each of these items are next discussed. Tables 6.5.1-1 through 6.5.1-3 present key data assumed in the analysis.

The core rated power of 3,565 MWt adjusted for calorimetric error (+2 percent of power) was used in the analysis. As previously noted, the analysis used RCS operating temperatures that bound the highest average coolant temperature range, with uncertainty added. This is a bounding analysis condition for the analysis. The use of higher temperatures is conservative because the initial fluid energy is based on coolant temperatures, which are at the maximum levels attained in steady-state operation. Additionally, an allowance to account for instrument error and deadband is reflected in the initial RCS temperatures. As previously discussed, the initial system (RCS) pressure in this analysis is based on a nominal value of 2,250 psia plus an allowance which accounts for the measurement uncertainty on pressurizer pressure. The selection of 2,280 psia as the limiting pressure is considered to affect the blowdown phase results only, since this represents the initial pressure of the RCS. The RCS rapidly depressurizes from this value until the point at which it equilibrates with containment pressure.

The rate at which the RCS blows down is initially more severe at the higher RCS pressure. Additionally the RCS has a higher fluid density at the higher pressure (assuming a constant temperature) and subsequently has a higher RCS mass available for releases. Therefore, 2,250 psia plus uncertainty was selected for the initial pressure as the limiting case for the long-term mass and energy release calculations.

The selection of the fuel design features for the long-term mass and energy release calculation is based on the need to conservatively maximize the energy stored in the fuel at the beginning of the postulated accident (that is, to maximize the core stored energy). The margin in core-stored energy is a statistical value that is dependent upon fuel type, power level, and burnup. Therefore, the analysis very conservatively accounts for the stored energy in the core.

A margin in RCS volume of 3 percent (which is consists of 1.6-percent allowance for thermal expansion and 1.4-percent for uncertainty) is modeled.

A uniform steam generator tube plugging level of 0 percent is modeled. This assumption maximizes the reactor coolant volume and fluid release by virtue of consideration of the RCS fluid in all steam generator tubes. During the post-blowdown period, the steam generators are active heat sources since significant energy remains in the secondary metal and secondary mass that has the potential to be transferred to the primary side. The 0-percent tube plugging assumption maximizes the heat transfer area and, therefore, the transfer of secondary heat across the steam generator tubes. Additionally, this assumption reduces the reactor coolant loop resistance, which reduces the Δp upstream of the break for the pump suction breaks and increases break flow. Therefore, the analysis very conservatively accounts for the level of steam generator tube plugging.

Regarding safety injection flow, the mass and energy release calculation considered configurations/failures to conservatively bound respective alignments. The cases include (a) a Minimum Safeguards Case (1 charging (CH)/SI, 1 high-head safety injection (HHSI), and 1 low-head safety injection (LHSI) Pumps); and (b) Maximum Safeguards, (2 CH/SI, 2 HHSI, and 2 LHSI Pumps).

The following assumptions were employed to ensure that the mass and energy releases are conservatively calculated, thereby maximizing energy release to containment.

1. Maximum expected operating temperature of the RCS (100-percent full-power conditions)

2. Allowance for RCS temperature uncertainty (+4.3°F)

3. Margin in RCS volume of 3 percent (which is composed of 1.6-percent allowance for thermal expansion, and 1.4 percent for uncertainty)

4. Core rated power of 3,565 MWt

5. Allowance for calorimetric error (+2 percent of power)

6. Conservative heat transfer coefficient (i.e., steam generator primary/secondary heat transfer and RCS metal heat transfer)

7. Allowance in core stored energy for effect of fuel densification

8. A margin in core-stored energy that is a statistical value that is dependent upon fuel type, power level, and burnup

9. An allowance for RCS initial pressure uncertainty (+30 psi)

10. A maximum containment backpressure equal to design pressure (60 psig)

11. Minimum RCS loop flow (93,600 gpm/loop)

12. Steam generator tube plugging leveling (0-percent uniform)

    a. Maximizes reactor coolant volume and fluid release

    b. Maximizes heat transfer area across the steam generator tubes

    c. Reduces coolant loop resistance, which reduces the Δp upstream of the break for the pump suction breaks and increases break flow

Therefore, based on the previously discussed conditions and assumptions, a bounding analysis of the Callaway Plant was made for the release of mass and energy from the RCS in the event of a LOCA at 3,565 MWt.

6.5.1.2 Description of Analyses

The evaluation model used for the long-term LOCA mass and energy release calculations is the March 1979 model described in Reference 1. This evaluation model has been reviewed and approved generically by the NRC. The approval letter is included with Reference 1. Even though this is a first-time application for the Callaway Plant, it has also been utilized and approved on the plant-specific dockets for other Westinghouse pressurized water reactors (PWRs).

This report section presents the long-term LOCA mass and energy releases generated in support of the Callaway Plant RSG Program. These mass and energy releases are then subsequently used in the containment integrity analysis.

6.5.1.3 LOCA Mass and Energy Release Phases

The containment system receives mass and energy releases following a postulated rupture in the RCS. These releases continue over a time period, which, for the LOCA mass and energy analysis, is typically divided into the following four phases:

1. Blowdown – the period of time from accident initiation (when the reactor is at steady-state operation) to the time that the RCS and containment reach an equilibrium state.

2. Refill – the period of time when the lower plenum is being filled by accumulator and emergency core cooling system (ECCS) water. At the end of blowdown, a large amount of water remains in the cold legs, downcomer, and lower plenum. To conservatively consider the refill period for the purpose of containment mass and energy releases, it is assumed that this water is instantaneously transferred to the lower plenum along with sufficient accumulator water to completely fill the lower plenum. This allows an uninterrupted release of mass and energy to containment. Therefore, the refill period is conservatively neglected in the mass and energy release calculation.

3. Reflood – begins when the water from the lower plenum enters the core and ends when the core is completely quenched.

4. Post-reflood (FROTH) – describes the period following the reflood phase. For the pump suction break, a two-phase mixture exits the core, passes through the hot legs, and is superheated in the steam generators prior to exiting the break as steam. After the broken loop steam generator cools, the break flow becomes two phase.

6.5.1.4 Computer Codes

The Reference 1 mass and energy release evaluation model is comprised of mass and energy release versions of the following codes: SATAN VI, WREFLOOD, FROTH, and EPITOME. These codes were used to calculate the long-term LOCA mass and energy releases for the Callaway Plant.

SATAN VI calculates blowdown, the first portion of the thermal-hydraulic transient following break initiation, including pressure, enthalpy, density, mass and energy flow rates, and energy transfer between primary and secondary systems as a function of time.

The WREFLOOD code addresses the portion of the LOCA transient where the core reflooding phase occurs after the primary coolant system has depressurized (blowdown) due to the loss of water through the break and when water supplied by the ECCS refills the reactor vessel and provides cooling to the core. The most important feature of WREFLOOD is the steam/water mixing model (see subsection 6.5.1.8.2).

FROTH models the post-reflood portion of the transient. The FROTH code is used for the steam generator heat addition calculation from the broken and intact loop steam generators.

EPITOME continues the FROTH post-reflood portion of the transient from the time at which the secondary equilibrates to containment design pressure to the end of the transient. It also compiles a summary of data on the entire transient, including formal instantaneous mass and energy release tables and mass and energy balance tables with data at critical times.

6.5.1.5 Break Size and Location

Generic studies have been performed with respect to the effect of postulated break size on the LOCA mass and energy releases. The double-ended guillotine break has been found to be limiting due to larger mass flow rates during the blowdown phase of the transient. During the reflood and froth phases, the break size has little effect on the releases.

Three distinct locations in the RCS loop can be postulated for pipe rupture for any release purposes:

1. Hot leg (between vessel and steam generator)
2. Cold leg (between pump and vessel)
3. Pump suction (between steam generator and pump)

The break locations analyzed for this program are the DEPS rupture (10.46 ft^2), and the DEHL rupture (9.20 ft^2). Break mass and energy releases have been calculated for the blowdown, reflood, and post-reflood phases of the LOCA for the DEPS cases. For the DEHL case, the releases were calculated only for the blowdown. The following paragraphs provide a discussion on each break location.

The DEHL rupture has been shown in previous studies to result in the highest blowdown mass and energy release rates. Although the core flooding rate would be the highest for this break location, the amount of energy released from the steam generator secondary is minimal because the majority of the fluid, which exits the core, vents directly to containment bypassing the steam generators. As a result, the reflood mass and energy releases are reduced significantly as compared to either the pump suction or cold leg break locations where the core exit mixture must pass through the steam generators before venting through the break. For the hot leg break, generic studies have confirmed that there is no reflood peak (that is, from the end of the blowdown period the containment pressure would continually decrease). Therefore, only the mass and energy releases for the hot leg break blowdown phase are calculated and presented in this section of the report.

The cold leg break location has also been found in previous studies to be much less limiting in terms of the overall containment energy releases. The cold leg blowdown is faster than that of the pump suction break, and more mass is released into the containment. However, the core heat transfer is greatly reduced, and this results in a considerably lower energy release into containment. Studies have determined that the blowdown transient for the cold leg is, in general, less limiting than that for the pump suction break. During reflood, the flooding rate is greatly reduced and the energy release rate into the containment is reduced. Therefore, the cold leg break is bounded by other breaks and no further evaluation is necessary.

The pump suction break combines the effects of the relatively high core flooding rate, as in the hot leg break, and the addition of the stored energy in the steam generators. As a result, the pump suction break yields the highest energy flow rates during the post-blowdown period by including all of the available energy of the RCS in calculating the releases to containment.

6.5.1.6 Application of Single-Failure Criterion

An analysis of the effects of the single-failure criterion has been performed on the mass and energy release rates for each break analyzed. An inherent assumption in the generation of the mass and energy release is that offsite power is lost. This results in the actuation of the emergency diesel generators, required to power the safety injection system. This is not an issue for the blowdown period, which is limited by the DEHL break.

Two cases have been analyzed to assess the effects of a single failure. The first case assumes minimum safeguards SI flow based on the postulated single failure of an emergency diesel generator. This results in the loss of one train of safeguards equipment. The other case assumes maximum safeguards SI flow based on no postulated failures that would impact the amount of ECCS flow. The analysis of the cases described provides confidence that the effect of credible single failures is bounded.

6.5.1.7 Acceptance Criteria for Analyses

A large break loss-of-coolant accident is classified as an American Nuclear Society (ANS) Condition IV event, an infrequent fault. To satisfy the NRC acceptance criteria presented in the Standard Review Plan Section 6.2.1.3, the relevant requirements are as follows:

1. 10 CFR 50, Appendix A
2. 10 CFR 50, Appendix K, paragraph I.A

In order to meet these requirements, the following must be addressed:

1. Sources of energy
2. Break size and location
3. Calculation of each phase of the accident

6.5.1.8 Mass and Energy Release Data

6.5.1.8.1 Blowdown Mass and Energy Release Data

The SATAN-VI code is used for computing the blowdown transient. The code utilizes the control volume (element) approach with the capability for modeling a large variety of thermal fluid system configurations. The fluid properties are considered uniform and thermodynamic equilibrium is assumed in each element. A point kinetics model is used with weighted feedback effects. The major feedback effects include moderator density, moderator temperature, and Doppler broadening. A critical flow calculation for subcooled (modified Zaloudek), two-phase (Moody), or superheated break flow is incorporated into the analysis. The methodology for the use of this model is described in Reference 1.

Table 6.5.1-4 presents the calculated mass and energy release for the blowdown phase of the DEHL break. For the hot leg break mass and energy release tables, break path 1 refers to the mass and energy exiting from the reactor vessel side of the break; break path 2 refers to the mass and energy exiting from the steam generator side of the break.

Table 6.5.1-7 presents the calculated mass and energy releases for the blowdown phase of the DEPS break with minimum ECCS flows. Table 6.5.1-13 presents the calculated mass and energy releases for the blowdown phase of the DEPS break with maximum ECCS flows. For the pump suction breaks, break path 1 in the mass and energy release tables refers to the mass and energy exiting from the steam generator side of the break; break path 2 refers to the mass and energy exiting from the pump side of the break.

6.5.1.8.2 Reflood Mass and Energy Release Data

The WREFLOOD code is used for computing the reflood transient. The WREFLOOD code consists of two basic hydraulic models - one for the contents of the reactor vessel, and one for the coolant loops. The two models are coupled through the interchange of the boundary conditions applied at the vessel outlet nozzles and at the top of the downcomer. Additional transient phenomena such as pumped safety injection and accumulators, reactor coolant pump performance, and steam generator release are included as auxiliary equations that interact with the basic models as required. The WREFLOOD code permits the capability to calculate variations during the core reflooding transient of basic parameters such as core flooding rate, core and downcomer water levels, fluid thermodynamic conditions (pressure, enthalpy, density) throughout the primary system, and mass flow rates through the primary system. The code permits hydraulic modeling of the two flow paths available for discharging steam and entrained water from the core to the break; that is, the path through the broken loop and the path through the unbroken loops.

A complete thermal equilibrium mixing condition for the steam and ECCS injection water during the reflood phase has been assumed for each loop receiving ECCS water. This is consistent with the usage and application of the Reference 1 mass and energy release evaluation model in the previous analyses, such as, D. C. Cook Docket (Reference 2). Even though the Reference 1 model credits steam/water mixing only in the intact loop and not in the broken loop; the justification, applicability, and NRC approval for using the mixing model in the broken loop has been documented (Reference 2). Moreover, this assumption is supported by test data and is further discussed in the following paragraphs.

The model assumes a complete mixing condition (that is, thermal equilibrium) for the steam/water interaction. The complete mixing process, however, is made up of two distinct physical processes. The first is a two-phase interaction with condensation of steam by cold ECCS water. The second is a single-phase mixing of condensate and ECCS water. Since the steam release is the most important influence to the containment pressure transient, the steam condensation part of the mixing process is the only part that need be considered. (Any spillage directly heats only the sump.)

The most applicable steam/water mixing test data has been reviewed for validation of the containment integrity reflood steam/water mixing model. This data was generated in 1/3-scale tests (Reference 3), which are the largest scale data available and thus most clearly simulates the flow regimes and gravitational effects that would occur in a PWR. These tests were designed specifically to study the steam/water interaction for PWR reflood conditions.

A group of 1/3-scale tests corresponds directly to containment integrity reflood conditions. The injection flow rates for this group cover all phases and mixing conditions calculated during the reflood transient. The data from these tests were reviewed and discussed in detail in Reference 1. For all of these tests, the data clearly indicates the occurrence of very effective mixing with rapid steam condensation. The mixing model used in the containment integrity reflood calculation is, therefore, wholly supported by the 1/3-scale steam/water mixing data.

Additionally, the following justification is also noted. The post-blowdown limiting break for the containment integrity peak pressure analysis is the pump suction double-ended rupture break. For this break, there are two flow paths available in the RCS by which mass and energy may be released to containment. One is through the outlet of the steam generator, the other via reverse flow through the reactor coolant pump. Steam that is not condensed by ECCS injection in the intact RCS loops passes around the downcomer and through the broken loop cold leg and pump in venting to containment. This steam also encounters ECCS injection water as it passes through the broken loop cold leg. Complete mixing occurs and a portion of it is condensed. It is this portion of steam which is condensed that is taken credit for in this analysis. This assumption is justified based upon the postulated break location, and the actual physical presence of the ECCS injection nozzle. A description of the test and test results are contained in References 1 and 3.

Tables 6.5.1-8 and 6.5.1-14 present the calculated mass and energy releases for the reflood phase of the pump suction double-ended rupture, minimum safeguards, and maximum safeguards cases, respectively.

The transient response of the principal parameters during reflood are given in Tables 6.5.1-9 and 6.5.1-15 for the DEPS cases.

6.5.1.8.3 Post-Reflood Mass and Energy Release Data

The FROTH code (Reference 4) is used for computing the post-reflood transient. The FROTH code calculates the heat release rates resulting from a two-phase mixture present in the steam generator tubes. The mass and energy releases that occur during this phase are typically superheated due to the depressurization and equilibration of the broken loop and intact loop steam generators. During this phase of the transient, the RCS has equilibrated with the containment pressure, but the steam generators contain a secondary inventory at an enthalpy that is much higher than the primary side. Therefore, there is a significant amount of reverse heat transfer that occurs. Steam is produced in the core due to core decay heat. For a pump suction break, a two-phase fluid exits the core, flows through the hot legs and becomes superheated as it passes through the steam generator. Once the broken loop cools, the break flow becomes two phase. During the FROTH calculation, ECCS injection is addressed for both the injection phase and the recirculation phase. The FROTH code calculation stops when the secondary side equilibrates to the saturation temperature (Tsat) at the containment design pressure, after this point the EPITOME code completes the steam generator depressurization (see subsection 6.5.1.8.5 for additional information).

The methodology for the use of this model is described in Reference 1. The mass and energy release rates are calculated by FROTH and EPITOME until the time of containment depressurization. After containment depressurization (14.7 psia), the mass and energy release available to containment is generated directly from core boiloff/decay heat.

Tables 6.5.1-10 and 6.5.1-16 present the two-phase post-reflood mass and energy release data for the pump suction double-ended cases, minimum and maximum ECCS assumptions.

6.5.1.8.4 Decay Heat Model

On November 2, 1978, the Nuclear Power Plant Standards Committee (NUPPSCO) of the ANS approved ANS Standard 5.1 (Reference 5) for the determination of decay heat. This standard was used in the mass and energy release. Table 6.5.1-22 lists the decay heat curve used in the mass and energy release analysis, post blowdown, for the Callaway Plant RSG Program.

Significant assumptions in the generation of the decay heat curve for use in the LOCA mass and energy releases analysis include the following:

1. Decay heat sources considered are fission product decay and heavy element decay of U-239 and Np-239.

2. Decay heat power from fissioning isotopes other than U-235 is assumed to be identical to that of U-235.

3. Fission rate is constant over the operating history of maximum power level.

4. The factor accounting for neutron capture in fission products has been taken from Equation 11 of Reference 5, up to 10,000 seconds and from Table 10 of Reference 5, beyond 10,000 seconds.

5. The fuel has been assumed to be at full power for 10^8 seconds.

6. The number of atoms of U-239 produced per second has been assumed to be equal to 70 percent of the fission rate.

7. The total recoverable energy associated with one fission has been assumed to be 200 MeV/fission.

8. Two-sigma uncertainty (two times the standard deviation) has been applied to the fission product decay.

Based upon the NRC staff Safety Evaluation Report (SER) of the March 1979 evaluation model (Reference 5), use of the ANS Standard-5.1, November 1979 decay heat model was approved for the calculation of mass and energy releases to the containment following a LOCA.

6.5.1.8.5 Steam Generator Equilibration and Depressurization

Steam generator equilibration and depressurization is the process by which secondary-side energy is removed from the steam generators in stages. The FROTH computer code calculates the heat removal from the secondary mass until the secondary temperature is the saturation temperature (Tsat) at the containment design pressure. After the FROTH calculations, the EPITOME code continues the FROTH calculation for steam generator cooldown removing steam generator secondary energy at different rates (that is, first- and second-stage rates). The first-stage rate is applied until the steam generator reaches Tsat at the user-specified intermediate equilibration pressure, when the secondary pressure is assumed to reach the actual containment pressure. Then the second-stage rate is used until the final depressurization, when the secondary reaches the reference temperature of Tsat at 14.7 psia, or 212°F. The heat removal of the broken-loop and intact-loop steam generators are calculated separately.

During the FROTH calculations, steam generator heat removal rates are calculated using the secondary-side temperature, primary-side temperature, and a secondary-side heat transfer coefficient determined using a modified McAdam's correlation. Steam generator energy is removed during the FROTH transient until the secondary-side temperature reaches saturation temperature at the containment design pressure. The constant heat removal rate used during the first heat removal stage is based on the final heat removal rate calculated by FROTH. The steam generator energy available to be released during the first-stage interval is determined by calculating the difference in secondary energy available at the containment design pressure and that at the (lower) user-specified intermediate equilibration pressure, assuming saturated conditions. This energy is then divided by the first-stage energy removal rate, resulting in an intermediate equilibration time. At this time, the rate of energy release drops substantially to the second-stage rate. The second-stage rate is determined as the fraction of the difference in secondary energy available between the intermediate equilibration and final depressurization at 212°F, and the time difference from the time of the intermediate equilibration to the user-specified time of the final depressurization at 212°F. With current methodology, all of the secondary energy remaining after the intermediate equilibration is conservatively assumed to be released by imposing a mandatory cooldown and subsequent depressurization down to atmospheric pressure at 3,600 seconds (that is, 14.7 psia and 212°F).

6.5.1.8.6 Sources of Mass and Energy

The sources of mass considered in the LOCA mass and energy release analysis are given in Tables 6.5.1-5, 6.5.1-11, and 6.5.1-17. These sources are the RCS, accumulators, and pumped safety injection.

The energy inventories considered in the LOCA mass and energy release analysis are given in Tables 6.5.1-6, 6.5.1-12, and 6.5.1-18. The energy sources include:

1. RCS water
2. Accumulator water (all four inject)
3. Pumped safety injection water
4. Decay heat
5. Core stored energy
6. RCS metal (includes steam generator tubes)
7. Steam generator metal (includes transition cone, shell, wrapper, and other internals)
8. Steam generator secondary energy (includes fluid mass and steam mass)
9. Secondary transfer of energy (feedwater into and steam out of the steam generator secondary)

The energy reference points include:

1. Available energy: 212°F; 14.7 psia
2. Total energy content: 32°F; 14.7 psia

The mass and energy inventories are presented at the following times, as appropriate:

1. Time zero (initial conditions)
2. End of blowdown time
3. End of refill time (that is, end of blowdown with accumulator mass adjustment)
4. End of reflood time
5. Time of broken loop steam generator equilibration to pressure setpoint
6. Time of intact loop steam generator equilibration to pressure setpoint
7. Time of full depressurization (3600 seconds)

In the mass and energy release data presented, no Zirc-water reaction heat was considered because the cladding temperature is assumed not to rise high enough for the rate of the Zirc-water reaction heat to be of any significance.

The sequence of events for the LOCA transients are shown in Tables 6.5.1-19 through 6.5.1-21.

6.5.1.8.7 Conclusions

The consideration of the various energy sources in the long-term mass and energy release analysis provides assurance that all available sources of energy have been included in this analysis. Therefore, the review guidelines presented in Standard Review Plan Section 6.2.1.3 have been satisfied. The results of this analysis were provided for use in the containment integrity analysis.

6.5.1.9 References

1. WCAP-10325-P-A, (Proprietary) and WCAP-10326-A (Non-Proprietary), "Westinghouse LOCA Mass and Energy Release Model for Containment Design - March 1979 Version," May 1983.

2. NRC Docket No. 50-315, "Amendment No. 126, Facility Operating License No. DPR-58 (TAC No. 71062), for D. C. Cook Nuclear Plant Unit 1," June 9, 1989.

3. EPRI 294-2, "Mixing of Emergency Core Cooling Water with Steam; 1/3-Scale Test and Summary," (WCAP-8423), Final Report, June 1975.

4. WCAP-8264-P-A, Rev. 1, (Proprietary) and WCAP-8312-A (Non-Proprietary), "Westinghouse Mass and Energy Release Data For Containment Design," August 1975.

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

Table 6.5.1-1 System Parameters Initial Conditions for Replacement Steam Generator	
	Value
Parameters	**Replacement Steam Generator**
Core Thermal Power (MWt)	3636.3
Reactor Coolant System Total Flow rate (lbm/sec)	38,511.8
Vessel Outlet Temperature (°F)	624.3
Core Inlet Temperature (°F)	561.1
Vessel Average Temperature (°F)	592.7
Initial Steam Generator Steam Pressure (psia)	1033
Steam Generator Tube Plugging (%)	0
Initial Steam Generator Secondary Side Mass (lbm)	132,944
Assumed Maximum Containment Backpressure (psia)	74.7
Accumulator	
Water Volume (ft^3) per accumulator	916.2
N_2 Cover Gas Pressure (psia)	663
Temperature (°F)	120
Safety Injection Delay, total (sec) (from beginning of event)(Minimum ECCS case)	48.3
(Maximum ECCS case)	34.3

Note:
Core thermal power, RCS total flow rate, RCS coolant temperatures, and steam generator secondary-side mass include appropriate uncertainty and/or allowance.

Table 6.5.1-2 Safety Injection Flow Minimum Safeguards	
RCS Pressure (psia)	**Total Flow (gpm)**
Injection Mode (Reflood Phase)	
14.7	4973.0
114.7	3527.4
214.7	933.60
1014.7	633.0
Injection Mode (Post-Reflood Phase)	
74.7	4105.2
Cold Leg Recirculation Mode	
74.7	4800

Table 6.5.1-3 Safety Injection Flow Maximum Safeguards	
RCS Pressure (psia)	**Total Flow (gpm)**
Injection Mode (Reflood Phase)	
14.7	11727.9
114.7	8904.8
214.7	3474.39
1014.7	1225.3
Injection Mode (Post-Reflood Phase)	
74.7	10032.4
Cold Leg Recirculation Mode	
74.7	9600

Table 6.5.1-4 Double-Ended Hot Leg Break Blowdown Mass and Energy Releases Callaway Plant Utilizing the Replacement Steam Generator

Time Seconds	Break Path No.1*		Break Path No.2**	
	Flow lbm/sec	Energy Thousands Btu/sec	Flow lbm/sec	Energy Thousands Btu/sec
0.00	0.00	0.00	0.00	0.00
0.001	47326.99	30634.09	47325.13	30631.53
0.002	46756.23	30262.62	46414.55	30034.69
0.10	43575.79	28468.62	28067.30	18128.89
0.20	36659.40	23935.99	24836.47	15944.55
0.30	35422.77	23098.17	22486.00	14277.83
0.40	34434.82	22451.86	21307.95	13350.75
0.50	33856.43	22079.21	20599.45	12741.68
0.60	33792.11	22046.64	20087.18	12281.30
0.70	33485.46	21887.33	19687.89	11915.97
0.80	32850.10	21538.57	19416.41	11651.12
0.90	32210.16	21203.96	19171.57	11419.75
1.00	31707.34	20973.39	18988.57	11239.15
1.10	31464.30	20924.86	18805.30	11070.34
1.20	31145.24	20823.85	18676.27	10940.44
1.30	30685.67	20621.23	18572.48	10832.37
1.40	30106.85	20328.27	18498.95	10747.41
1.50	29539.75	20030.48	18457.47	10684.96
1.60	29116.11	19821.87	18445.14	10642.86
1.70	28796.58	19682.08	18457.96	10618.02
1.80	28415.65	19495.65	18485.52	10604.28
1.90	27880.77	19191.82	18515.54	10594.43
2.00	27284.68	18833.18	18544.81	10586.76
2.10	26772.79	18529.65	18576.12	10582.33
2.20	26383.84	18314.45	18609.60	10581.24
2.30	26015.03	18110.15	18640.02	10580.80
2.40	25578.06	17846.22	18660.63	10576.87

Table 6.5.1-4 (cont.) Double-Ended Hot Leg Break Blowdown Mass and Energy Releases Callaway Plant Utilizing the Replacement Steam Generator				
	Break Path No.1*		Break Path No.2**	
Time Seconds	Flow lbm/sec	Energy Thousands Btu/sec	Flow lbm/sec	Energy Thousands Btu/sec
2.50	25131.13	17564.54	18670.96	10569.28
2.60	24715.73	17299.52	18674.22	10559.61
2.70	24340.31	17058.54	18669.38	10547.23
2.80	24009.06	16845.10	18654.41	10530.86
2.90	23709.03	16649.98	18630.15	10510.77
3.00	23406.72	16444.78	18595.11	10486.02
3.10	23118.72	16241.07	18548.51	10456.12
3.20	22869.34	16061.23	18492.72	10422.17
3.30	22641.75	15892.33	18429.06	10384.79
3.40	22427.58	15727.14	18355.48	10342.73
3.50	22247.46	15583.00	18273.58	10296.78
3.60	22086.04	15448.56	18183.59	10246.94
3.70	21935.69	15317.41	18084.30	10192.45
3.80	21808.15	15199.59	17977.51	10134.20
3.90	21697.17	15090.78	17856.75	10068.52
4.00	21599.81	14989.68	17731.73	10000.85
4.20	21448.92	14814.73	17478.52	9864.63
4.40	21348.77	14672.65	17217.27	9724.81
4.60	21313.02	14575.68	16953.21	9584.04
4.80	21281.45	14485.74	16687.16	9442.68
5.00	21251.22	14409.31	16420.00	9301.00
5.20	21254.44	14361.76	16154.95	9161.26
5.40	21257.27	14310.10	15895.87	9025.36
5.60	21289.18	14272.71	15584.46	8859.90
5.80	21357.78	14249.69	15281.95	8700.91
6.00	21524.73	14264.80	14993.40	8550.72
6.20	21762.79	14313.33	14679.47	8386.62
6.40	22063.53	14390.79	14365.05	8222.94

Table 6.5.1-4 (cont.) Double-Ended Hot Leg Break Blowdown Mass and Energy Releases Callaway Plant Utilizing the Replacement Steam Generator				
	Break Path No.1*		Break Path No.2**	
Time Seconds	Flow lbm/sec	Energy Thousands Btu/sec	Flow lbm/sec	Energy Thousands Btu/sec
6.60	22498.23	14549.26	14045.34	8057.03
6.80	16987.96	11964.27	13725.73	7891.51
7.00	17169.50	12008.24	13389.52	7716.10
7.20	17328.23	11970.72	13060.44	7544.72
7.40	17491.25	11996.53	12754.33	7386.36
7.60	17668.47	12004.04	12436.13	7219.84
7.80	17814.20	12042.73	12132.49	7061.13
8.00	17979.96	12071.85	11835.31	6905.37
8.20	18130.16	12055.85	11552.54	6756.95
8.40	18238.50	12055.68	11273.81	6609.88
8.60	18365.78	12100.84	11004.28	6467.14
8.80	18495.06	12100.57	10746.94	6330.68
9.00	18313.32	11958.51	10499.54	6199.15
9.20	18506.06	12011.00	10259.85	6071.51
9.40	18780.59	12113.30	10030.22	5949.21
9.60	19159.82	12276.43	9807.56	5830.72
9.80	19894.70	12640.30	9592.54	5716.39
10.00	20765.47	13152.74	9380.04	5603.43
10.20	22657.67	14326.97	9174.20	5494.16
10.40	27098.16	17104.56	8960.27	5380.34
10.60	25991.03	16322.61	8744.87	5266.16
10.80	25498.89	15927.97	8512.27	5142.27
11.00	25168.42	15669.62	8255.38	5005.85
11.20	23764.21	14757.98	7991.34	4868.20
11.40	23363.91	14497.82	7718.30	4728.17
11.60	23189.66	14402.11	7446.75	4591.87
11.80	22969.51	14293.97	7179.73	4460.79
12.00	22666.90	14108.55	6922.05	4336.30

Table 6.5.1-4 (cont.) Double-Ended Hot Leg Break Blowdown Mass and Energy Releases Callaway Plant Utilizing the Replacement Steam Generator				
	Break Path No.1*		Break Path No.2**	
Time Seconds	Flow lbm/sec	Energy Thousands Btu/sec	Flow lbm/sec	Energy Thousands Btu/sec
12.20	22366.19	13923.32	6674.10	4218.29
12.40	22078.48	13746.17	6436.48	4106.13
12.60	21738.59	13553.17	6213.06	4001.21
12.80	21346.01	13339.06	6001.38	3902.20
13.00	19381.10	12110.72	5801.61	3808.55
13.20	15155.48	9240.06	5613.01	3719.96
13.40	10736.85	7365.64	5438.05	3638.31
13.60	10604.79	7345.08	5290.50	3572.68
13.80	10719.21	7399.43	5173.59	3520.40
14.00	10764.13	7439.62	5079.63	3474.51
14.20	10718.22	7446.38	5007.49	3434.08
14.40	10622.53	7438.77	4951.99	3397.70
14.60	10476.78	7409.39	4907.18	3364.53
14.80	10307.56	7368.71	4865.88	3332.93
15.00	10115.68	7312.81	4821.51	3300.51
15.20	9906.56	7241.43	4770.04	3266.15
15.40	9663.37	7145.95	4703.57	3225.90
15.60	9381.75	7025.12	4617.82	3178.66
15.80	9048.18	6872.84	4507.50	3122.64
16.00	8656.24	6670.97	4373.44	3059.43
16.20	8254.91	6300.27	4213.34	2986.80
16.40	7850.10	5899.22	4031.09	2906.25
16.60	7546.05	5651.17	3831.77	2817.38
16.80	6360.90	5379.48	3625.84	2727.21
17.00	5590.33	5174.71	3418.62	2638.68
17.20	5114.34	4936.56	3220.07	2554.87
17.40	4801.43	4696.53	3028.04	2473.48
17.60	4521.35	4461.25	2850.76	2398.08

Table 6.5.1-4 (cont.) Double-Ended Hot Leg Break Blowdown Mass and Energy Releases Callaway Plant Utilizing the Replacement Steam Generator

Time Seconds	Break Path No.1* Flow lbm/sec	Break Path No.1* Energy Thousands Btu/sec	Break Path No.2** Flow lbm/sec	Break Path No.2** Energy Thousands Btu/sec
17.80	4240.90	4235.41	2686.02	2326.77
18.00	3942.19	3993.06	2533.80	2260.25
18.20	3625.13	3745.38	2394.08	2199.95
18.40	3291.53	3499.38	2262.81	2147.71
18.60	2965.21	3262.28	2136.16	2089.91
18.80	2650.88	3023.91	2012.74	2035.44
19.00	2411.43	2821.86	1885.70	1992.13
19.20	2258.40	2690.12	1754.69	1944.41
19.40	2138.05	2555.69	1624.03	1880.56
19.60	2041.97	2449.73	1517.32	1808.84
19.80	1917.70	2307.57	1415.83	1717.06
20.00	1798.86	2169.55	1341.84	1639.38
20.20	1691.93	2039.24	1289.96	1580.58
20.40	1572.71	1905.75	1250.14	1533.95
20.60	1457.96	1777.10	1207.92	1485.04
20.80	1366.44	1672.31	1149.25	1417.76
21.00	1265.13	1554.64	1044.19	1292.08
21.20	1184.55	1461.87	932.61	1157.49
21.40	1103.93	1363.69	842.10	1047.12
21.60	1013.80	1251.60	714.92	890.18
21.80	932.33	1146.69	593.21	740.47
22.00	892.65	1096.21	476.06	595.72
22.20	857.02	1050.25	404.16	507.19
22.40	817.43	999.64	359.03	451.22
22.60	795.97	970.62	195.20	246.38
22.80	765.67	932.01	181.19	229.08
23.00	763.07	932.23	0.00	0.00
23.20	737.91	899.68	0.00	0.00

Table 6.5.1-4 (cont.) Double-Ended Hot Leg Break Blowdown Mass and Energy Releases Callaway Plant Utilizing the Replacement Steam Generator				
	Break Path No.1*		Break Path No.2**	
Time Seconds	Flow lbm/sec	Energy Thousands Btu/sec	Flow lbm/sec	Energy Thousands Btu/sec
23.40	728.01	885.78	0.00	0.00
23.60	716.99	871.07	0.00	0.00
23.80	686.55	829.60	0.00	0.00
24.00	629.81	760.29	0.00	0.00
24.20	666.23	812.72	0.00	0.00
24.40	644.92	779.92	0.00	0.00
24.60	649.24	773.84	0.00	0.00
24.80	643.14	737.44	0.00	0.00
25.00	648.50	689.47	0.00	0.00
25.20	656.65	663.93	0.00	0.00
25.40	655.27	642.08	0.00	0.00
25.60	657.19	631.19	0.00	0.00
25.80	656.11	624.02	0.00	0.00
26.00	354.11	376.04	0.00	0.00
26.20	0.00	0.00	0.00	0.00

* Mass and energy exiting from the reactor vessel side of the break
**Mass and energy exiting from the SG side of the break

Table 6.5.1-5 Double-Ended Hot Leg Break Mass Balance Callaway Plant Utilizing the Replacement Steam Generator				
Time (Seconds)		.00	26.20	26.20+δ
		Mass (Thousand lbm)		
Initial	In RCS and Accumulator (ACC)	807.4	807.4	807.4
Added Mass	Pumped Injection	.00	.00	.00
	Total Added	.00	.00	.00
*** Total Available[1] ***		807.4	807.4	807.4
Distribution	Reactor Coolant	580.70	93.98	123.92
	Accumulator	226.70	149.37	119.43
	Total Contents	807.40	243.36	243.36
Effluent	Break Flow	.00	564.00	564.00
	ECCS Spill	.00	.00	.00
	Total Effluent	.00	564.00	564.00
*** Total Accountable[2] ***		807.40	807.36	807.36

1. Total available is the sum of all sources of mass that could be released to the containment post-LOCA.
2. Total accountable represents the mass that was calculated to be released to the containment. The difference between total available and total accountable is basically the mass calculated to be stored in the RCS.

Table 6.5.1-6 Double-Ended Hot Leg Break Energy Balance Callaway Plant Utilizing the Replacement Steam Generator

Time (Seconds)		.00	26.20	26.20+δ
		Energy (Million Btu)		
Initial Energy	In RCS, ACC, SG	957.34	957.34	957.34
Added Energy	Pumped Injection	.00	.00	.00
	Decay Heat	.00	8.83	8.83
	Heat From Secondary	.00	-2.20	2.20
	Total Added	.00	6.62	6.62
*** Total Available[1] ***		957.34	963.97	963.97
Distribution	Reactor Coolant	343.94	21.31	23.99
	Accumulator	20.29	13.37	10.69
	Core Stored	24.01	7.89	7.89
	Primary Metal	158.77	148.19	148.19
	Secondary Metal	106.08	103.87	103.87
	Steam Generator	304.25	296.38	296.38
	Total Contents	957.34	591.00	591.00
Effluent	Break Flow	.00	372.35	372.35
	ECCS Spill	.00	.00	.00
	Total Effluent	.00	372.35	372.35
*** Total Accountable[2] ***		957.34	963.36	963.36

1. Total available is the sum of all sources of mass that could be released to the containment post-LOCA.
2. Total accountable represents the mass that was calculated to be released to the containment. The difference between total available and total accountable is basically the mass calculated to be stored in the RCS.

Table 6.5.1-7 Double-Ended Pump Suction Break Minimum ECCS Flows Blowdown Mass and Energy Releases Callaway Plant Utilizing the Replacement Steam Generator				
Time	Break Path No.1*		Break Path No.2**	
Seconds	Flow lbm/sec	Energy Thousand Btu/sec	Flow lbm/sec	Energy Thousand Btu/sec
.00000	.0	.0	.0	.0
.00108	92731.6	51719.5	42543.9	23666.8
.101	42376.2	23625.9	21837.7	12136.4
.201	47506.4	26607.6	24082.9	13394.7
.302	46705.9	26301.8	24213.4	13478.3
.401	46291.0	26236.5	23439.1	13058.7
.502	46253.8	26410.9	22302.0	12433.3
.602	45801.4	26365.3	21359.8	11912.5
.702	45595.0	26463.2	20529.0	11451.4
.802	45727.1	26745.3	19892.4	11098.3
.901	45538.9	26829.7	19439.1	10849.3
1.00	44956.6	26666.5	19186.9	10711.1
1.10	44097.2	26328.4	19030.0	10625.4
1.20	43236.3	25980.7	18944.4	10578.7
1.30	42430.9	25660.9	18892.2	10550.2
1.40	41668.1	25363.5	18858.9	10531.7
1.50	40947.3	25083.8	18846.3	10524.7
1.60	40223.7	24800.6	18853.0	10528.5
1.70	39491.1	24509.7	18871.5	10538.9
1.80	38728.1	24204.9	18883.0	10545.5
1.90	37912.3	23872.7	18875.3	10541.1
2.00	37001.9	23491.2	18857.1	10531.0
2.10	36016.2	23068.5	18832.6	10517.4
2.20	34954.2	22603.9	18781.8	10489.2
2.30	33868.8	22125.4	18703.0	10445.5
2.40	32661.7	21567.3	18579.1	10376.4
2.50	31487.7	21016.5	18420.3	10287.9
2.60	30288.4	20436.2	17986.7	10044.6
2.70	29026.8	19793.8	17753.6	9915.9

Table 6.5.1-7 (cont.) Double-Ended Pump Suction Break Minimum ECCS Flows Blowdown Mass and Energy Releases Callaway Plant Utilizing the Replacement Steam Generator				
Time	Break Path No.1*		Break Path No.2**	
Seconds	Flow lbm/sec	Energy Thousand Btu/sec	Flow lbm/sec	Energy Thousand Btu/sec
2.80	26299.7	18094.2	17579.4	9819.7
2.90	24111.5	16751.8	17374.7	9706.1
3.00	22715.0	15934.5	17158.0	9585.8
3.10	21383.7	15115.4	16967.0	9480.5
3.20	20227.1	14389.0	16793.7	9385.2
3.30	19300.3	13804.8	16615.7	9287.2
3.40	18473.9	13275.2	16453.4	9198.2
3.50	17763.3	12817.1	16305.8	9117.5
3.60	17157.2	12425.7	16164.7	9040.5
3.70	16614.6	12072.7	16021.7	8962.3
3.80	16146.2	11766.9	15894.8	8893.3
3.90	15747.4	11506.2	15776.9	8829.4
4.00	15401.5	11278.0	15660.1	8766.0
4.20	14809.2	10877.6	15433.6	8643.3
4.40	14348.3	10557.6	15233.9	8535.9
4.60	13984.3	10291.2	15028.6	8425.3
4.80	13702.1	10072.4	14832.2	8319.9
5.00	13486.1	9890.5	14632.0	8212.3
5.20	13336.1	9748.0	15770.8	8864.1
5.40	13231.2	9630.9	16146.1	9074.8
5.60	13170.0	9541.6	15830.9	8901.9
5.80	13185.7	9503.5	15544.3	8744.8
6.00	13232.7	9484.4	15442.6	8692.9
6.20	13289.2	9475.1	15274.9	8603.5
6.40	13358.9	9477.5	15117.8	8520.5
6.60	13436.0	9485.9	14968.6	8441.3
6.80	13511.3	9494.4	14793.6	8346.5
7.00	13570.3	9494.5	14606.6	8244.3
7.20	13606.6	9484.5	14449.6	8158.5

Table 6.5.1-7 (cont.) Double-Ended Pump Suction Break Minimum ECCS Flows Blowdown Mass and Energy Releases Callaway Plant Utilizing the Replacement Steam Generator				
Time	Break Path No.1*		Break Path No.2**	
Seconds	Flow lbm/sec	Energy Thousand Btu/sec	Flow lbm/sec	Energy Thousand Btu/sec
7.40	13812.8	9593.2	14382.8	8122.5
7.60	13582.7	9430.5	14427.1	8147.9
7.80	13311.9	9474.3	14208.7	8020.6
8.00	12155.3	9067.4	13993.6	7897.1
8.20	11417.5	8735.5	13897.0	7843.5
8.40	11312.8	8660.5	13727.6	7748.3
8.60	11372.0	8636.3	13551.9	7648.2
8.80	11477.9	8630.4	13378.9	7549.6
9.00	11636.5	8656.8	13201.9	7449.1
9.20	11830.4	8698.0	13037.8	7355.5
9.40	12013.0	8725.1	12858.7	7253.1
9.60	12170.4	8739.4	12692.1	7158.2
9.80	12292.1	8736.3	12534.6	7068.1
10.0	12333.4	8687.1	12373.9	6976.1
10.2	12273.7	8582.9	12228.8	6893.1
10.4	12146.0	8448.7	12093.8	6815.6
10.6	11975.4	8298.0	11958.8	6738.0
10.8	11725.1	8104.3	11837.3	6668.1
11.0	11404.8	7878.8	11733.5	6608.3
11.2	11125.6	7695.6	11626.3	6546.2
11.4	10921.6	7566.8	11507.3	6477.5
11.6	10708.0	7430.4	11404.6	6418.7
11.8	10469.0	7281.8	11308.5	6363.6
12.0	10279.7	7170.4	11193.7	6297.5
12.2	10105.9	7063.9	11087.5	6236.8
12.4	9897.6	6931.9	10998.0	6185.7
12.6	9706.0	6815.6	10893.4	6125.5
12.8	9544.8	6719.4	10789.9	6066.1
13.0	9362.1	6606.2	10702.6	6016.3

Table 6.5.1-7 (cont.) Double-Ended Pump Suction Break Minimum ECCS Flows Blowdown Mass and Energy Releases Callaway Plant Utilizing the Replacement Steam Generator				
Time	Break Path No.1*		Break Path No.2**	
Seconds	Flow lbm/sec	Energy Thousand Btu/sec	Flow lbm/sec	Energy Thousand Btu/sec
13.2	9182.9	6498.5	10607.5	5961.6
13.4	9031.2	6410.2	10507.2	5904.0
13.6	8867.7	6311.1	10424.5	5856.9
13.8	8709.5	6216.9	10329.9	5802.7
14.0	8563.1	6130.6	10240.6	5751.8
14.2	8419.6	6045.6	10153.2	5702.2
14.4	8287.2	5967.8	10060.8	5649.9
14.6	8153.0	5886.8	9963.0	5594.9
14.8	8010.5	5799.6	9855.5	5534.8
15.0	7859.9	5707.1	9735.5	5468.5
15.2	7699.7	5606.6	9622.0	5406.6
15.4	7542.8	5504.6	9508.7	5344.9
15.6	7402.8	5407.6	9408.0	5290.2
15.8	7286.1	5320.6	9311.3	5237.5
16.0	7189.8	5244.8	9219.7	5188.0
16.2	7105.6	5178.1	9130.7	5140.5
16.4	7024.7	5116.1	9044.6	5095.3
16.6	6943.2	5057.1	8959.5	5051.6
16.8	6859.3	5000.3	8875.6	5009.4
17.0	6773.5	4946.3	8793.0	4969.0
17.2	6685.0	4894.8	8710.2	4929.6
17.4	6593.3	4845.8	8627.1	4891.4
17.6	6499.0	4799.3	8540.9	4852.9
17.8	6402.0	4754.7	8420.5	4790.8
18.0	6300.2	4710.0	8322.0	4724.1
18.2	6196.4	4667.5	8245.7	4651.6
18.4	6088.9	4626.8	8208.5	4585.4
18.6	5974.7	4586.2	8204.8	4527.9
18.8	5852.6	4544.5	8129.0	4428.1

Table 6.5.1-7 (cont.) Double-Ended Pump Suction Break Minimum ECCS Flows Blowdown Mass and Energy Releases Callaway Plant Utilizing the Replacement Steam Generator				
Time	Break Path No.1*		Break Path No.2**	
Seconds	Flow lbm/sec	Energy Thousand Btu/sec	Flow lbm/sec	Energy Thousand Btu/sec
19.0	5725.2	4503.8	8113.9	4364.9
19.2	5590.6	4461.8	7974.6	4241.2
19.4	5453.8	4422.1	7838.9	4132.9
19.6	5358.3	4413.8	7532.0	3962.1
19.8	5269.0	4435.7	7175.2	3781.0
20.0	5000.3	4439.9	6684.9	3512.9
20.2	4489.2	4360.2	6289.3	3306.5
20.4	3930.9	4216.2	5863.0	3109.6
20.6	3442.6	4010.0	5432.2	2914.5
20.8	3090.9	3766.2	4740.2	2476.0
21.0	2780.6	3425.2	4512.0	2190.4
21.2	2551.8	3158.2	4420.3	2019.6
21.4	2354.8	2923.7	4183.0	1826.0
21.6	2168.0	2698.6	4354.5	1827.1
21.8	1965.3	2452.5	5340.0	2187.7
22.0	1784.9	2233.1	5118.8	2077.3
22.2	1649.2	2067.9	3707.1	1493.7
22.4	1537.4	1931.1	3291.8	1323.6
22.6	1433.1	1802.6	2496.3	995.7
22.8	1318.2	1660.2	1970.0	742.5
23.0	1197.5	1510.1	3871.0	1307.6
23.2	1049.1	1325.7	6257.8	2032.4
23.4	925.6	1171.0	5130.9	1651.2
23.6	812.6	1029.0	4329.9	1386.2
23.8	709.9	899.8	3528.0	1122.5
24.0	621.0	787.7	3005.0	947.9
24.2	547.3	694.9	2563.9	799.3
24.4	496.9	631.1	2141.0	658.4
24.6	465.5	591.7	1683.7	510.6

Table 6.5.1-7 (cont.) Double-Ended Pump Suction Break Minimum ECCS Flows Blowdown Mass and Energy Releases Callaway Plant Utilizing the Replacement Steam Generator				
Time	Break Path No.1*		Break Path No.2**	
Seconds	Flow lbm/sec	Energy Thousand Btu/sec	Flow lbm/sec	Energy Thousand Btu/sec
24.8	432.5	550.0	1178.5	353.2
25.0	381.5	485.3	613.6	182.5
25.2	327.7	417.1	77.0	22.9
25.4	271.1	345.2	.0	.0
25.6	210.0	267.7	.0	.0
25.8	146.4	186.8	.0	.0
26.0	92.6	118.4	.0	.0
26.2	10.1	12.9	.0	.0
26.4	.0	.0	.0	.0

* Mass and energy exiting the SG side of the break

** Mass and energy exiting the pump side of the break

Table 6.5.1-8 Double-Ended Pump Suction Break Minimum Safeguards Reflood Mass and Energy Releases Callaway Plant Utilizing the Replacement Steam Generator

Time	Break Path No.1*		Break Path No.2**	
Seconds	Flow lbm/sec	Energy Thousand Btu/Sec	Flow lbm/sec	Energy Thousand Btu/Sec
26.4	.0	.0	.0	.0
27.0	.0	.0	.0	.0
27.1	.0	.0	.0	.0
27.2	.0	.0	.0	.0
27.3	.0	.0	.0	.0
27.4	.0	.0	.0	.0
27.4	.0	.0	.0	.0
27.5	91.4	108.1	.0	.0
27.6	39.9	47.2	.0	.0
27.8	28.3	33.4	.0	.0
27.9	32.5	38.4	.0	.0
28.0	41.1	48.6	.0	.0
28.1	47.3	55.9	.0	.0
28.2	54.4	64.3	.0	.0
28.3	59.7	70.5	.0	.0
28.4	64.7	76.4	.0	.0
28.5	69.4	82.0	.0	.0
28.6	73.9	87.3	.0	.0
28.6	75.0	88.6	.0	.0
28.7	78.2	92.4	.0	.0
28.8	82.3	97.3	.0	.0
28.9	86.3	102.0	.0	.0
29.0	90.1	106.5	.0	.0
29.1	93.8	110.9	.0	.0
29.2	97.4	115.2	.0	.0
29.3	100.9	119.3	.0	.0
29.4	104.3	123.3	.0	.0
29.5	107.6	127.2	.0	.0
30.5	136.9	161.9	.0	.0

Table 6.5.1-8 (cont.) Double-Ended Pump Suction Break Minimum Safeguards Reflood Mass and Energy Releases Callaway Plant Utilizing the Replacement Steam Generator				
Time	Break Path No.1*		Break Path No.2**	
Seconds	Flow lbm/sec	Energy Thousand Btu/Sec	Flow lbm/sec	Energy Thousand Btu/Sec
31.5	161.3	190.8	.0	.0
32.5	384.0	455.4	3360.2	483.4
32.7	460.4	546.6	4137.1	603.1
33.5	508.8	604.6	4543.4	695.5
34.5	502.0	596.4	4484.2	691.8
35.5	492.7	585.3	4403.9	683.5
36.5	483.2	573.9	4320.0	674.4
36.8	480.3	570.5	4294.7	671.6
37.5	473.7	562.6	4235.8	665.0
38.5	464.5	551.5	4152.8	655.7
39.5	455.6	540.9	4071.5	646.4
40.5	447.0	530.6	3992.4	637.3
41.5	438.7	520.7	3915.5	628.5
42.0	434.7	515.9	3877.9	624.1
42.5	430.7	511.1	3840.9	619.8
43.5	423.0	502.0	3768.7	611.4
44.5	415.6	493.2	3698.6	603.3
45.5	408.5	484.7	3630.6	595.3
46.5	401.7	476.5	3564.7	587.6
47.5	395.1	468.6	3500.8	580.1
48.0	391.9	464.8	3469.5	576.4
48.5	418.6	496.8	3761.6	598.5
49.5	417.8	495.7	3743.3	591.7
50.5	411.9	488.7	3687.3	585.0
51.5	406.2	481.9	3632.7	578.4
52.5	400.7	475.3	3579.6	572.0
53.6	330.9	392.2	2837.9	493.5
54.5	326.4	386.7	2809.7	488.2
54.6	326.0	386.2	2805.6	487.7

Table 6.5.1-8 (cont.) Double-Ended Pump Suction Break Minimum Safeguards Reflood Mass and Energy Releases Callaway Plant Utilizing the Replacement Steam Generator				
Time	Break Path No.1*		Break Path No.2**	
Seconds	Flow lbm/sec	Energy Thousand Btu/Sec	Flow lbm/sec	Energy Thousand Btu/Sec
55.6	483.0	573.5	341.1	259.8
56.6	502.0	596.5	349.2	271.2
57.6	487.1	578.5	342.5	262.4
58.6	470.9	559.2	335.2	252.9
59.6	456.4	541.8	328.8	244.4
60.6	442.5	525.2	322.6	236.3
61.6	429.1	509.3	316.7	228.6
62.6	416.3	493.9	311.0	221.1
63.6	403.9	479.1	305.5	214.0
64.6	392.0	464.9	300.3	207.2
65.6	380.5	451.2	295.3	200.7
66.6	369.5	438.1	290.4	194.5
67.5	359.9	426.7	286.3	189.1
67.6	358.9	425.4	285.8	188.5
68.6	348.7	413.3	281.4	182.8
69.6	338.9	401.6	277.2	177.4
70.6	329.5	390.4	273.1	172.2
71.6	320.4	379.7	269.2	167.2
72.6	311.8	369.4	265.5	162.5
73.6	303.5	359.5	262.0	157.9
74.6	295.5	350.0	258.6	153.6
75.6	287.9	341.0	255.4	149.5
76.6	280.6	332.3	252.3	145.6
77.6	273.6	324.0	249.3	141.8
78.6	266.9	316.0	246.5	138.3
79.6	260.5	308.4	243.9	134.9
80.6	254.4	301.2	241.3	131.7
81.6	248.6	294.2	238.9	128.6
82.6	243.0	287.6	236.6	125.7

Table 6.5.1-8 (cont.)	Double-Ended Pump Suction Break Minimum Safeguards Reflood Mass and Energy Releases Callaway Plant Utilizing the Replacement Steam Generator			
Time	Break Path No.1*		Break Path No.2**	
Seconds	Flow lbm/sec	Energy Thousand Btu/Sec	Flow lbm/sec	Energy Thousand Btu/Sec
83.6	237.7	281.3	234.4	122.9
84.6	232.6	275.3	232.3	120.3
85.3	229.2	271.2	231.0	118.6
85.6	227.8	269.5	230.4	117.8
86.6	223.2	264.1	228.5	115.5
87.6	218.8	258.9	226.7	113.2
89.6	210.7	249.3	223.5	109.1
91.6	203.4	240.6	220.5	105.5
93.6	196.8	232.8	217.9	102.2
95.6	190.9	225.9	215.6	99.3
97.6	185.7	219.7	213.6	96.7
99.6	181.1	214.2	211.8	94.5
101.6	176.9	209.3	210.2	92.5
103.6	173.3	205.0	208.7	90.7
105.6	170.1	201.2	207.5	89.2
107.6	167.3	197.9	206.4	87.8
108.9	165.7	195.9	205.8	87.0
109.6	164.9	195.0	205.5	86.7
111.6	162.8	192.5	204.7	85.6
113.6	160.9	190.3	204.0	84.8
115.6	159.4	188.5	203.4	84.0
117.6	158.1	186.9	202.9	83.4
119.6	157.0	185.6	202.4	82.8
121.6	156.0	184.5	202.1	82.4
123.6	155.3	183.6	201.8	82.0
125.6	154.7	182.9	201.5	81.7
127.6	154.2	182.3	201.3	81.4
129.6	153.8	181.9	201.1	81.2
131.6	153.5	181.6	201.0	81.1

Table 6.5.1-8 (cont.) Double-Ended Pump Suction Break Minimum Safeguards Reflood Mass and Energy Releases Callaway Plant Utilizing the Replacement Steam Generator

Time	Break Path No.1*		Break Path No.2**	
Seconds	Flow lbm/sec	Energy Thousand Btu/Sec	Flow lbm/sec	Energy Thousand Btu/Sec
133.6	153.3	181.3	200.9	80.9
135.6	153.2	181.2	200.8	80.9
136.1	153.2	181.2	200.8	80.8
137.6	153.2	181.2	200.8	80.8
139.6	153.2	181.2	200.8	80.8
141.6	153.3	181.3	200.8	80.8
143.6	153.4	181.4	200.8	80.8
145.6	153.6	181.6	200.8	80.8
147.6	153.8	181.8	200.8	80.9
149.6	154.0	182.1	200.9	80.9
151.6	154.2	182.4	200.9	81.0
153.6	154.5	182.6	201.0	81.1
155.6	154.7	183.0	201.1	81.1
157.6	155.0	183.3	201.1	81.2
159.6	155.3	183.6	201.2	81.3
161.6	155.6	184.0	201.3	81.4
163.6	155.9	184.4	201.4	81.6
164.8	156.1	184.6	201.5	81.6
165.6	156.3	184.8	201.5	81.7
167.6	156.6	185.2	201.6	81.8
169.6	156.9	185.6	201.7	81.9
171.6	157.3	186.0	201.8	82.0
173.6	157.6	186.4	201.9	82.2
175.6	158.0	186.8	202.0	82.3
177.6	158.4	187.3	202.1	82.4
179.6	158.7	187.7	202.2	82.5
181.6	159.1	188.1	202.3	82.7
183.6	159.5	188.6	202.4	82.8
185.6	159.8	189.0	202.5	83.0

Table 6.5.1-8 (cont.) Double-Ended Pump Suction Break Minimum Safeguards Reflood Mass and Energy Releases Callaway Plant Utilizing the Replacement Steam Generator				
Time	Break Path No.1*		Break Path No.2**	
Seconds	Flow lbm/sec	Energy Thousand Btu/Sec	Flow lbm/sec	Energy Thousand Btu/Sec
187.6	160.2	189.5	202.6	83.1
189.6	160.6	189.9	202.8	83.2
191.6	161.0	190.4	202.9	83.4
193.6	161.4	190.8	203.0	83.5
194.5	161.6	191.1	203.0	83.6

* Mass and energy exiting the SG side of the break

** Mass and energy exiting the pump side of the break

Table 6.5.1-9 Double-Ended Pump Suction Break – Minimum Safeguards Principle Parameters During Reflood Callaway Plant Utilizing the Replacement Steam Generator										
Time	Flooding		Carryover Fraction	Core Height	Downcomer Height	Flow Fraction	Injection			
	Temp	Rate					Total	Accum	Spill	Enthalpy
Seconds	(°F)	(in/sec)	(---)	(feet)	(feet)	(---)	(Pounds Mass Per Second)			Btu/Lbm
26.4	181.8	.000	.000	.00	.00	.250	.0	.0	.0	.00
27.2	179.1	21.943	.000	.63	1.56	.000	7669.8	7669.8	.0	89.50
27.4	177.7	24.307	.000	1.02	1.46	.000	7615.3	7615.3	.0	89.50
27.8	176.7	2.964	.132	1.35	2.08	.261	7490.1	7490.1	.0	89.50
27.9	176.7	3.027	.153	1.37	2.39	.264	7464.4	7464.4	.0	89.50
28.6	176.6	2.767	.299	1.50	4.66	.334	7272.5	7272.5	.0	89.50
29.4	176.6	2.696	.405	1.61	7.07	.353	7092.6	7092.6	.0	89.50
32.7	177.0	4.701	.626	2.00	16.07	.588	5968.4	5968.4	.0	89.50
33.5	177.0	4.818	.655	2.12	16.12	.593	5693.8	5693.8	.0	89.50
34.5	177.2	4.610	.678	2.25	16.12	.591	5542.6	5542.6	.0	89.50
36.8	177.8	4.289	.705	2.51	16.12	.585	5247.5	5247.5	.0	89.50
42.0	179.8	3.867	.727	3.00	16.12	.570	4705.0	4705.0	.0	89.50
48.0	183.0	3.555	.735	3.50	16.12	.553	4209.8	4209.8	.0	89.50
48.5	183.3	3.719	.736	3.54	16.12	.565	4539.8	4014.9	.0	87.02
54.5	186.9	3.154	.737	4.00	16.12	.519	3438.3	2892.5	.0	86.09
54.6	186.9	3.151	.737	4.01	16.12	.518	3433.5	2887.6	.0	86.09
55.6	187.6	4.016	.743	4.08	15.96	.607	516.9	.0	.0	68.03
56.6	188.4	4.082	.743	4.17	15.62	.610	510.4	.0	.0	68.03
60.6	191.8	3.646	.742	4.50	14.39	.603	522.1	.0	.0	68.03

Table 6.5.1-9 (cont.) Double-Ended Pump Suction Break – Minimum Safeguards Principle Parameters During Reflood Callaway Plant Utilizing the Replacement Steam Generator

Time	Flooding		Carryover Fraction	Core Height	Downcomer Height	Flow Fraction	Injection			
	Temp	Rate					Total	Accum	Spill	Enthalpy
Seconds	(°F)	(in/sec)	(---)	(feet)	(feet)	(---)	(Pounds Mass Per Second)			Btu/Lbm
67.5	198.5	3.052	.739	5.00	12.82	.588	536.5	.0	.0	68.03
76.6	207.9	2.489	.733	5.55	11.56	.567	548.2	.0	.0	68.03
85.3	217.1	2.128	.729	6.00	10.96	.545	554.5	.0	.0	68.03
97.6	228.6	1.823	.725	6.55	10.73	.518	559.0	.0	.0	68.03
108.9	237.1	1.681	.725	7.00	10.89	.502	560.9	.0	.0	68.03
123.6	246.5	1.602	.727	7.55	11.34	.493	561.8	.0	.0	68.03
136.1	253.4	1.580	.730	8.00	11.81	.491	562.0	.0	.0	68.03
149.6	260.0	1.575	.734	8.48	12.35	.492	562.0	.0	.0	68.03
151.6	260.9	1.575	.735	8.55	12.43	.492	562.0	.0	.0	68.03
164.8	266.6	1.578	.739	9.00	12.96	.494	561.8	.0	.0	68.03
179.6	272.3	1.584	.745	9.50	13.55	.497	561.7	.0	.0	68.03
194.5	277.4	1.591	.750	10.00	14.13	.500	561.5	.0	.0	68.03

Table 6.5.1-10 Double-Ended Pump Suction Break Minimum Safeguards Post-Reflood Mass and Energy Releases Callaway Plant Utilizing the Replacement Steam Generator				
Time	Break Path No.1*		Break Path No.2**	
Seconds	Flow lbm/sec	Energy Thousand Btu/sec	Flow lbm/sec	Energy Thousand Btu/sec
194.6	232.3	290.5	334.9	134.4
199.6	232.1	290.3	335.1	134.2
204.6	231.0	289.0	336.1	134.3
209.6	231.1	289.1	336.0	134.0
214.6	231.2	289.2	336.0	133.8
219.6	230.1	287.9	337.0	133.8
224.6	230.2	287.9	337.0	133.6
229.6	229.0	286.5	338.1	133.6
234.6	229.0	286.5	338.1	133.4
239.6	229.0	286.5	338.2	133.2
244.6	227.9	285.0	339.3	133.2
249.6	227.8	284.9	339.4	133.0
254.6	227.7	284.8	339.5	132.8
259.6	227.5	284.6	339.6	132.6
264.6	226.3	283.1	340.9	132.7
269.6	226.1	282.8	341.1	132.5
274.6	225.9	282.6	341.3	132.3
279.6	225.6	282.3	341.5	132.1
284.6	225.4	281.9	341.8	131.9
289.6	224.0	280.2	343.2	132.1
294.6	223.7	279.8	343.5	131.9
299.6	223.3	279.3	343.9	131.8
304.6	222.9	278.8	344.3	131.6
309.6	222.4	278.2	344.8	131.5
314.6	221.9	277.6	345.2	131.4
319.6	221.4	276.9	345.8	131.3
324.6	220.8	276.2	346.4	131.2
329.6	221.2	276.6	346.0	130.9
334.6	220.5	275.8	346.7	130.8

Table 6.5.1-10 (cont.) Double-Ended Pump Suction Break Minimum Safeguards Post-Reflood Mass and Energy Releases Callaway Plant Utilizing the Replacement Steam Generator				
Time	Break Path No.1*		Break Path No.2**	
Seconds	Flow lbm/sec	Energy Thousand Btu/sec	Flow lbm/sec	Energy Thousand Btu/sec
339.6	219.7	274.8	347.5	130.7
344.6	218.9	273.9	348.2	130.7
349.6	219.0	273.9	348.2	130.4
354.6	218.1	272.8	349.1	130.4
359.6	218.0	272.7	349.2	130.2
364.6	217.8	272.5	349.4	130.0
369.6	216.7	271.0	350.5	130.1
374.6	216.3	270.6	350.8	129.9
379.6	215.9	270.0	351.3	129.8
384.6	215.3	269.3	351.9	129.7
389.6	214.6	268.4	352.6	129.6
394.6	214.6	268.4	352.6	129.3
399.6	213.6	267.2	353.6	129.3
404.6	213.4	266.9	353.8	129.2
409.6	213.0	266.5	354.2	129.0
414.6	212.4	265.7	354.8	128.9
419.6	212.3	265.5	354.9	128.7
424.6	211.9	265.0	355.3	128.6
429.6	211.1	264.1	356.1	128.5
434.6	210.7	263.5	356.5	128.4
439.6	209.8	262.4	357.4	128.4
444.6	209.6	262.1	357.6	128.2
449.6	209.2	261.7	358.0	128.0
454.6	208.5	260.8	358.7	128.0
459.6	208.1	260.3	359.1	127.8
464.6	207.4	259.5	359.7	127.7
469.6	206.8	258.7	360.3	127.6
474.6	95.2	119.1	471.9	158.3
635.2	95.2	119.1	471.9	158.3

Table 6.5.1-10 (cont.) Double-Ended Pump Suction Break Minimum Safeguards Post-Reflood Mass and Energy Releases Callaway Plant Utilizing the Replacement Steam Generator				
Time	Break Path No.1*		Break Path No.2**	
Seconds	Flow lbm/sec	Energy Thousand Btu/sec	Flow lbm/sec	Energy Thousand Btu/sec
635.3	100.1	124.2	467.1	154.8
639.6	100.0	124.0	467.2	154.5
1442.5	100.0	124.0	467.2	154.5
1442.6	82.4	94.9	484.7	49.6
3063.0	69.2	79.6	498.0	52.0
3063.1	69.2	79.6	577.2	107.1
3600.0	65.8	75.7	580.5	107.7
3600.1	53.1	61.1	593.2	86.0
8200.0	40.9	47.0	605.4	87.8
8200.1	40.4	46.4	606.0	80.0
10000.0	38.1	43.9	608.2	80.3
21000.0	31.4	36.1	615.0	81.2
21000.1	30.9	35.6	615.4	72.0
48000.0	24.8	28.6	621.5	72.7
48000.1	24.6	28.3	621.7	66.5
100000.0	19.9	22.9	626.4	67.0
100000.1	19.8	22.8	626.5	64.5
1000000.0	8.5	9.8	637.8	65.7
1000000.1	8.5	9.8	637.8	65.7
10000000.0	2.7	3.1	643.7	66.3

* Mass and energy exiting the SG side of the break

** Mass and energy exiting the pump side of the break

Table 6.5.1-11 Double-Ended Pump Suction Mass Balance Minimum Safeguards Callaway Plant Utilizing the Replacement Steam Generator								
		Mass Balance						
Time (seconds)		.00	26.40	26.40+δ	194.54	635.25	1442.53	3600.00
		Mass (Thousands lbm)						
Initial	In RCS and ACC	807.47	807.47	807.47	807.47	807.47	807.47	807.47
Added Mass	Pumped Inj	.00	.00	.00	81.14	331.07	788.93	2055.11
	Total Added	.00	.00	.00	81.14	331.07	788.93	2055.11
*** Total Available[1] ***		807.47	807.47	807.47	888.61	1138.54	1596.41	2862.58
Distribution	Reactor Coolant	580.78	50.07	77.83	135.75	135.75	135.75	135.75
	Accumulator	226.70	172.61	144.85	.00	.00	.00	.00
	Total Contents	807.47	222.67	222.67	135.75	135.75	135.75	135.75
Effluent	Break Flow	.00	584.78	584.78	741.36	991.29	1449.15	2715.32
	ECCS Spill	.00	.00	.00	.00	.00	.00	.00
	Total Effluent	.00	584.78	584.78	741.36	991.29	1449.15	2715.32
*** Total Accountable[2] ***		807.47	307.46	807.46	877.11	1127.03	1584.90	2851.07

1. Total available is the sum of all sources of mass that could be released to the containment post-LOCA.
2. Total accountable represents the mass that was calculated to be released to the containment. The difference between total available and total accountable is basically the mass calculated to be stored in the RCS.

Table 6.5.1-12 Double-Ended Pump Suction Break Energy Balance Minimum Safeguards Callaway Plant Utilizing the Replacement Steam Generator								
		Energy Balance						
Time (Seconds)		.00	26.40	26.40+δ	194.54	635.25	1442.53	3600.00
		Energy (Million Btu)						
Initial Energy	In RCS ACCS SG	957.60	957.60	957.60	957.60	957.60	957.60	957.60
Added Energy	Pumped Injection	.00	.00	.00	5.52	22.52	53.67	166.52
	Decay Heat	.00	8.57	8.57	29.38	70.99	132.72	262.04
	Heat From Secondary	.00	15.70	15.70	15.70	23.25	34.96	34.96
	Total Added	.00	24.27	24.27	50.60	116.76	221.34	463.52
*** Total Available[1] ***		957.60	981.87	981.87	1008.20	1074.37	1178.95	1421.12
Distribution	Reactor Coolant	343.98	11.83	14.32	37.14	37.14	37.14	37.14
	Accumulator	20.29	15.45	12.96	.00	.00	.00	.00
	Core Stored	24.00	12.13	12.13	5.09	4.91	4.48	3.33
	Primary Metal	161.73	152.98	152.98	127.74	94.22	72.32	53.29
	Secondary Metal	103.35	100.59	100.59	92.81	74.45	53.66	39.39
	Steam Generator	304.25	322.90	322.90	294.38	237.85	180.72	136.82
	Total Contents	957.60	615.88	615.88	557.16	448.57	348.33	269.97
Effluent	Break Flow	.00	365.41	365.41	440.02	614.79	822.88	1145.92
	ECCS Spill	.00	.00	.00	.00	.00	.00	.00
	Total Effluent	.00	365.41	365.41	440.02	614.79	822.88	1145.92
*** Total Accountable[2] ***		957.60	981.29	981.29	997.18	1063.35	1171.21	1415.89

1. Total available is the sum of all sources of energy that could be released to the containment post-LOCA and is referenced to 32°F.
2. Total accountable represents the energy that was calculated to be released to the containment above 212°F (or 14.7 psia). The difference between total available and total accountable is the energy between 32°F and 212°Fand the energy calculated to be stored in the RCS.

Table 6.5.1-13 Double-Ended Pump Suction Break Maximum ECCS Flows Blowdown Mass and Energy Releases Callaway Plant Utilizing the Replacement Steam Generator				
Time	Break Path No. 1*		Break Path No. 2**	
Seconds	Flow lbm/sec	Energy Thousands Btu/sec	Flow lbm/sec	Energy Thousands Btu/sec
0.00	0.00	0.00	0.00	0.00
0.001	92992.03	51865.73	42544.52	23667.09
0.10	42308.19	23586.60	21959.08	12204.75
0.20	43041.57	24100.35	24538.77	13650.45
0.30	45477.97	25600.84	24957.80	13896.01
0.40	45160.22	25584.11	24391.62	13594.45
0.50	45348.49	25875.03	23461.42	13084.63
0.60	45012.85	25890.25	22706.59	12669.23
0.70	45444.68	26353.83	22106.90	12337.38
0.80	45533.50	26615.19	21613.72	12063.47
0.90	45328.93	26696.49	21264.02	11870.77
1.00	44714.77	26522.24	21056.14	11756.51
1.10	43882.52	26207.92	20940.46	11693.20
1.20	43017.60	25869.53	20872.05	11655.64
1.30	42197.04	25551.94	20835.41	11635.49
1.40	41403.36	25246.96	20819.68	11626.79
1.50	40636.77	24951.72	20819.12	11626.50
1.60	39866.63	24651.31	20830.84	11633.14
1.70	39091.49	24344.23	20846.66	11642.14
1.80	38286.05	24020.43	20855.37	11647.19
1.90	37441.65	23675.57	20852.15	11645.64
2.00	36547.40	23309.04	20818.64	11627.14
2.10	35577.54	22904.24	20743.34	11585.36
2.20	34538.94	22462.27	20646.52	11531.79
2.30	33392.80	21954.48	20517.20	11460.21
2.40	32264.56	21456.53	20187.67	11275.75
2.50	31077.70	20911.25	19819.06	11070.79

Table 6.5.1-13 (cont.) Double-Ended Pump Suction Break Maximum ECCS Flows Blowdown Mass and Energy Releases Callaway Plant Utilizing the Replacement Steam Generator				
Time	Break Path No. 1*		Break Path No. 2**	
Seconds	Flow lbm/sec	Energy Thousands Btu/sec	Flow lbm/sec	Energy Thousands Btu/sec
2.60	29875.17	20339.75	19599.37	10949.70
2.70	27877.63	19182.82	19392.44	10835.68
2.80	25133.36	17469.86	19157.60	10705.96
2.90	23205.81	16304.96	18905.47	10566.76
3.00	21804.86	15472.20	18675.66	10440.57
3.10	20356.15	14557.54	18472.99	10329.97
3.20	19211.93	13835.69	18268.38	10218.33
3.30	18246.48	13220.33	18081.30	10116.69
3.40	17434.55	12700.82	17910.20	10024.15
3.50	16756.74	12267.33	17739.49	9931.88
3.60	16184.51	11900.31	17576.40	9843.94
3.70	15707.03	11594.26	17431.37	9766.30
3.80	15307.61	11337.79	17292.80	9692.31
3.90	14956.45	11109.35	17150.51	9616.17
4.00	14637.99	10898.61	17013.02	9542.79
4.20	14121.24	10551.16	16777.77	9418.49
4.40	13717.38	10267.66	16542.38	9294.09
4.60	13411.75	10039.91	16339.44	9188.17
4.80	13183.65	9855.29	16137.78	9082.87
5.00	13021.50	9708.33	15957.92	8989.87
5.20	12913.28	9592.44	15791.86	8904.61
5.40	12852.67	9505.04	15634.46	8824.17
5.60	12852.22	9457.20	16572.89	9365.23
5.80	12895.13	9435.07	16403.50	9277.38
6.00	12951.03	9420.28	16396.46	9282.67
6.20	13038.50	9427.66	16241.77	9202.11
6.40	13141.29	9445.62	16093.79	9125.88
6.60	13220.91	9451.06	15961.29	9057.53

Table 6.5.1-13 (cont.) Double-Ended Pump Suction Break Maximum ECCS Flows Blowdown Mass and Energy Releases Callaway Plant Utilizing the Replacement Steam Generator				
Time	Break Path No. 1*		Break Path No. 2**	
Seconds	Flow lbm/sec	Energy Thousands Btu/sec	Flow lbm/sec	Energy Thousands Btu/sec
6.80	13253.63	9431.49	15790.98	8966.69
7.00	13234.83	9385.85	15597.35	8861.56
7.20	13164.37	9314.13	15417.60	8762.94
7.40	13055.60	9225.78	15236.32	8661.72
7.60	12918.17	9118.79	15186.43	8634.80
7.80	12924.28	9226.04	15222.99	8652.79
8.00	12178.38	9090.28	14908.26	8466.69
8.20	10989.51	8649.01	14759.68	8381.24
8.40	10444.53	8389.44	14624.99	8306.26
8.60	10323.84	8276.62	14451.11	8206.57
8.80	10352.29	8217.25	14263.54	8098.00
9.00	10456.61	8190.04	14075.50	7990.21
9.20	10610.14	8185.22	13923.40	7902.49
9.40	10772.42	8180.53	13744.96	7798.75
9.60	10924.41	8172.39	13573.95	7699.86
9.80	11062.06	8163.84	13426.85	7614.64
10.00	11152.08	8133.72	13266.80	7521.58
10.20	11166.81	8065.59	13121.04	7437.06
10.20	11166.68	8065.14	13120.33	7436.65
10.40	11113.61	7968.04	12990.30	7361.08
10.60	11011.22	7852.54	12857.13	7283.35
10.80	10856.18	7714.69	12736.48	7213.04
11.00	10667.05	7567.86	12626.48	7148.85
11.20	10484.82	7437.84	12513.66	7082.90
11.40	10309.90	7319.35	12399.21	7016.33
11.60	10129.58	7201.54	12296.32	6956.84
11.80	9953.44	7090.40	12187.09	6893.65
12.00	9791.18	6989.74	12075.56	6829.40

Table 6.5.1-13 (cont.) Double-Ended Pump Suction Break Maximum ECCS Flows Blowdown Mass and Energy Releases Callaway Plant Utilizing the Replacement Steam Generator				
Time	Break Path No. 1*		Break Path No. 2**	
Seconds	Flow lbm/sec	Energy Thousands Btu/sec	Flow lbm/sec	Energy Thousands Btu/sec
12.20	9622.08	6883.25	11973.84	6771.12
12.40	9454.86	6779.18	11869.76	6711.36
12.60	9299.09	6683.93	11762.02	6649.58
12.80	9138.67	6585.18	11664.84	6594.21
13.00	8981.20	6489.97	11564.18	6536.76
13.20	8834.29	6402.89	11460.38	6477.72
13.40	8684.67	6313.34	11366.81	6425.00
13.60	8541.35	6228.93	11263.68	6366.74
13.80	8404.76	6148.68	11166.17	6312.38
14.00	8268.37	6068.26	11069.44	6258.80
14.20	8132.75	5988.18	10954.50	6195.16
14.40	7985.72	5900.67	10842.43	6134.36
14.60	7833.85	5811.40	10706.39	6060.94
14.80	7676.80	5717.39	10576.79	5992.66
15.00	7517.87	5616.92	10447.29	5924.64
15.20	7379.78	5524.68	10334.90	5865.93
15.40	7261.78	5440.17	10222.18	5806.32
15.60	7163.21	5367.72	10118.01	5752.10
15.80	7074.00	5303.47	10014.21	5699.19
16.00	6985.35	5242.80	9911.82	5648.26
16.20	6895.53	5184.95	9810.87	5599.48
16.40	6804.55	5130.15	9709.84	5552.05
16.60	6710.54	5077.52	9606.76	5505.14
16.80	6613.59	5027.43	9503.74	5460.19
17.00	6514.00	4980.06	9398.15	5415.90
17.20	6410.99	4934.77	9290.22	5372.82
17.40	6304.50	4891.28	9160.93	5320.00
17.60	6195.83	4850.11	8903.78	5195.11

Table 6.5.1-13 (cont.) Double-Ended Pump Suction Break Maximum ECCS Flows Blowdown Mass and Energy Releases Callaway Plant Utilizing the Replacement Steam Generator				
Time	Break Path No. 1*		Break Path No. 2**	
Seconds	Flow lbm/sec	Energy Thousands Btu/sec	Flow lbm/sec	Energy Thousands Btu/sec
17.80	6086.51	4811.70	8696.73	5099.22
18.00	5973.01	4773.63	8529.56	4999.21
18.20	5857.25	4737.09	8374.70	4880.72
18.40	5736.73	4700.36	8266.18	4771.67
18.60	5563.85	4629.66	7960.74	4526.17
18.80	5282.31	4487.77	7794.10	4330.42
19.00	4955.97	4317.95	7547.26	4087.33
19.20	4642.13	4143.97	7230.17	3833.57
19.40	4367.28	3977.31	6930.19	3626.99
19.60	4162.78	3845.51	6685.50	3513.97
19.80	4049.23	3774.04	6268.74	3391.87
20.00	3927.68	3766.24	5752.81	3222.62
20.20	3638.93	3742.91	5273.17	3060.85
20.40	3232.03	3666.69	4611.64	2569.53
20.60	2859.83	3477.50	4506.30	2275.50
20.80	2534.59	3128.35	4478.05	2101.68
21.00	2311.13	2866.58	4297.90	1906.71
21.20	2115.86	2631.79	4410.71	1866.30
21.40	1919.55	2393.87	5082.65	2089.22
21.60	1737.43	2172.43	4977.33	2017.56
21.80	1596.95	2001.50	3777.60	1514.04
22.00	1480.97	1859.58	3327.19	1326.99
22.20	1372.81	1726.43	2612.35	1023.46
22.40	1271.41	1600.88	2467.48	915.93
22.60	1141.96	1440.44	3485.47	1209.62
22.80	997.94	1260.59	5007.11	1669.07
23.00	877.36	1109.63	5053.37	1646.64
23.20	769.14	973.76	4342.61	1395.25

Table 6.5.1-13 (cont.) Double-Ended Pump Suction Break Maximum ECCS Flows Blowdown Mass and Energy Releases Callaway Plant Utilizing the Replacement Steam Generator				
Time	Break Path No. 1*		Break Path No. 2**	
Seconds	Flow lbm/sec	Energy Thousands Btu/sec	Flow lbm/sec	Energy Thousands Btu/sec
23.40	669.04	847.78	3603.93	1144.24
23.60	586.02	743.19	3122.20	979.34
23.80	505.62	641.68	2628.56	813.64
24.00	442.94	562.55	2161.39	659.73
24.20	391.96	498.06	1681.16	506.23
24.40	369.73	470.13	1163.37	346.36
24.60	344.57	438.34	631.81	186.74
24.80	303.37	386.07	149.55	44.13
25.00	247.62	315.30	0.00	0.00
25.20	184.84	235.57	0.00	0.00
25.40	118.71	151.53	0.00	0.00
25.60	43.47	55.64	0.00	0.00
25.80	0.00	0.00	0.00	0.00

* Mass and energy exiting the SG side of the break
** Mass and energy exiting the pump side of the break

Table 6.5.1-14 Double-Ended Pump Suction Break Maximum Safeguards Reflood Mass and Energy Releases Callaway Plant Utilizing the Replacement Steam Generator

Time (seconds)	Break Path No. 1 Flow*		Break Path No. 2 Flow**	
	(lbm/sec)	Thousand (Btu/sec)	(lbm/sec)	Thousand (Btu/sec)
25.81	0.00	0.00	0.00	0.00
26.38	0.00	0.00	0.00	0.00
26.48	0.00	0.00	0.00	0.00
26.58	0.00	0.00	0.00	0.00
26.68	0.00	0.00	0.00	0.00
26.78	0.00	0.00	0.00	0.00
26.83	0.00	0.00	0.00	0.00
26.93	94.30	111.46	0.00	0.00
27.03	38.83	45.88	0.00	0.00
27.15	26.54	31.36	0.00	0.00
27.25	32.17	38.01	0.00	0.00
27.35	39.82	47.05	0.00	0.00
27.45	48.46	57.27	0.00	0.00
27.55	53.93	63.74	0.00	0.00
27.65	59.44	70.24	0.00	0.00
27.75	64.60	76.35	0.00	0.00
27.85	69.48	82.11	0.00	0.00
27.95	74.12	87.60	0.00	0.00
27.98	75.25	88.93	0.00	0.00
28.05	78.56	92.85	0.00	0.00
28.15	82.82	97.88	0.00	0.00
28.25	86.92	102.73	0.00	0.00
28.35	90.88	107.41	0.00	0.00
28.45	94.70	111.93	0.00	0.00
28.55	98.41	116.31	0.00	0.00
28.65	102.00	120.57	0.00	0.00
28.75	105.50	124.70	0.00	0.00
28.85	108.90	128.73	0.00	0.00

Table 6.5.1-14 (cont.) Double-Ended Pump Suction Break Maximum Safeguards Reflood Mass and Energy Releases Callaway Plant Utilizing the Replacement Steam Generator

Time (seconds)	Break Path No. 1 Flow*		Break Path No. 2 Flow**	
	(lbm/sec)	Thousand (Btu/sec)	(lbm/sec)	Thousand (Btu/sec)
29.85	138.94	164.28	0.00	0.00
30.85	163.94	193.86	0.00	0.00
31.85	234.62	277.63	1441.90	200.70
32.16	437.62	519.22	3724.59	549.53
32.86	517.77	615.06	4455.63	682.08
33.86	516.26	613.29	4436.43	687.86
34.86	599.92	713.53	5197.40	745.24
35.86	590.49	702.24	5118.60	737.99
36.06	588.71	700.11	5104.01	736.32
36.86	581.57	691.54	5045.16	729.47
37.86	572.72	680.93	4971.51	720.79
38.86	564.05	670.54	4898.76	712.13
39.86	555.63	660.44	4827.48	703.59
40.56	549.89	653.56	4778.61	697.72
40.86	547.47	650.67	4757.94	695.23
41.86	539.58	641.22	4690.25	687.06
42.86	531.97	632.09	4624.47	679.10
43.86	524.61	623.29	4560.56	671.35
44.86	517.51	614.78	4498.49	663.80
45.76	511.32	607.37	4444.16	657.19
45.86	510.64	606.56	4438.21	656.46
46.86	504.00	598.62	4379.65	649.32
47.86	497.58	590.93	4322.74	642.36
48.86	491.35	583.49	4267.42	635.57
49.86	485.32	576.27	4213.60	628.96
50.86	479.47	569.28	4161.23	622.50
51.46	476.04	565.18	4130.47	618.70
51.86	473.79	562.48	4110.24	616.20

Table 6.5.1-14 (cont.) Double-Ended Pump Suction Break Maximum Safeguards Reflood Mass and Energy Releases Callaway Plant Utilizing the Replacement Steam Generator

Time (seconds)	Break Path No. 1 Flow*		Break Path No. 2 Flow**	
	(lbm/sec)	Thousand (Btu/sec)	(lbm/sec)	Thousand (Btu/sec)
52.86	468.27	555.88	4060.56	610.04
53.86	462.90	549.46	4012.15	604.02
54.91	395.45	468.99	3368.78	532.64
55.96	219.88	260.16	943.81	232.14
56.96	219.23	259.39	945.01	231.79
57.96	218.60	258.64	946.36	231.50
58.96	217.97	257.90	947.78	231.22
59.96	217.35	257.16	949.21	230.94
60.96	216.72	256.42	950.65	230.67
61.96	216.09	255.67	952.09	230.40
62.96	215.47	254.93	953.53	230.13
63.96	214.84	254.18	954.97	229.86
64.96	214.21	253.44	956.41	229.59
65.96	213.58	252.69	957.84	229.32
66.96	212.95	251.94	959.28	229.06
67.46	212.63	251.57	960.00	228.92
67.96	212.32	251.20	960.71	228.79
68.96	211.69	250.45	962.15	228.52
69.96	211.05	249.70	963.59	228.26
70.96	210.42	248.95	965.03	227.99
71.96	209.79	248.19	966.47	227.73
72.96	209.15	247.44	967.91	227.47
73.96	208.51	246.68	969.36	227.21
74.96	207.87	245.93	970.81	226.95
75.96	207.24	245.17	972.26	226.69
76.96	206.59	244.41	973.71	226.43
77.96	205.95	243.65	975.17	226.17
78.96	205.31	242.88	976.63	225.92

Table 6.5.1-14 Double-Ended Pump Suction Break Maximum Safeguards Reflood Mass and Energy Releases (cont.) Callaway Plant Utilizing the Replacement Steam Generator

Time (seconds)	Break Path No. 1 Flow*		Break Path No. 2 Flow**	
	(lbm/sec)	Thousand (Btu/sec)	(lbm/sec)	Thousand (Btu/sec)
79.96	204.66	242.12	978.10	225.66
80.96	204.01	241.34	979.57	225.41
81.96	203.35	240.57	981.05	225.16
82.96	202.70	239.79	982.53	224.90
83.96	202.04	239.01	984.02	224.65
84.96	201.38	238.23	985.51	224.40
85.96	200.72	237.44	987.01	224.15
86.26	200.52	237.20	987.46	224.08
86.96	200.05	236.65	988.51	223.91
88.96	198.71	235.07	991.53	223.42
90.96	197.37	233.47	994.56	222.93
92.96	196.01	231.86	997.61	222.45
94.96	194.64	230.24	1000.68	221.97
96.96	193.26	228.61	1003.77	221.50
98.96	191.88	226.96	1006.87	221.03
100.96	190.47	225.30	1009.97	220.56
102.96	189.05	223.62	1013.08	220.08
104.96	187.62	221.93	1016.19	219.61
106.66	186.40	220.47	1018.83	219.21
106.96	186.18	220.22	1019.29	219.14
108.96	184.73	218.50	1022.40	218.67
110.96	183.27	216.76	1025.51	218.20
112.96	181.80	215.02	1028.61	217.73
114.96	180.32	213.27	1031.72	217.26
116.96	178.83	211.50	1034.82	216.79
118.96	177.35	209.75	1038.73	216.55
120.96	176.65	208.92	1040.37	216.35
122.96	175.98	208.13	1041.94	216.13

Table 6.5.1-14 (cont.) Double-Ended Pump Suction Break Maximum Safeguards Reflood Mass and Energy Releases Callaway Plant Utilizing the Replacement Steam Generator				
	Break Path No. 1 Flow*		Break Path No. 2 Flow**	
Time (seconds)	(lbm/sec)	Thousand (Btu/sec)	(lbm/sec)	Thousand (Btu/sec)
124.96	175.32	207.35	1043.51	215.91
126.96	174.66	206.57	1045.07	215.67
128.96	174.01	205.79	1046.61	215.44
130.96	173.36	205.02	1048.15	215.20
132.96	172.71	204.26	1049.67	214.95
134.96	172.07	203.49	1051.18	214.69
136.96	171.43	202.74	1052.68	214.43
138.96	170.79	201.99	1054.18	214.17
140.96	170.16	201.24	1055.66	213.90
142.96	169.54	200.50	1057.13	213.62
144.96	168.92	199.76	1058.59	213.34
146.96	168.30	199.03	1060.05	213.05
148.96	167.68	198.30	1061.49	212.76
150.96	167.07	197.58	1062.93	212.46
152.96	166.46	196.85	1064.36	212.16
153.36	166.34	196.71	1064.64	212.10
154.96	165.85	196.13	1065.78	211.85
156.96	165.24	195.41	1067.20	211.54
158.96	164.64	194.70	1068.60	211.22
160.96	164.04	193.99	1070.00	210.89
162.96	163.45	193.29	1071.39	210.56
164.96	162.86	192.59	1072.77	210.23
166.96	162.27	191.89	1074.14	209.89
168.96	161.69	191.21	1075.50	209.55
170.96	161.11	190.52	1076.86	209.20
172.96	160.54	189.84	1078.20	208.85
174.96	159.97	189.17	1079.54	208.49
176.96	159.41	188.50	1080.87	208.13

Table 6.5.1-14 (cont.) Double-Ended Pump Suction Break Maximum Safeguards Reflood Mass and Energy Releases Callaway Plant Utilizing the Replacement Steam Generator

Time (seconds)	Break Path No. 1 Flow*		Break Path No. 2 Flow**	
	(lbm/sec)	Thousand (Btu/sec)	(lbm/sec)	Thousand (Btu/sec)
178.96	158.85	187.84	1082.18	207.77
180.06	158.55	187.48	1082.90	207.57
180.10	174.91	219.14	1211.16	208.44

* Mass and energy exiting the SG side of the break
** Mass and energy exiting the pump side of the break

Table 6.5.1-15 Double-Ended Pump Suction Break – Maximum Safeguards Principle Parameters During Reflood Callaway Plant Utilizing the Replacement Steam Generator

Time Seconds	Flooding		Carryover Fraction	Core Height (ft)	Downcomer Height (ft)	Flow Frac	Total	Injection Accum	Spill	Enthalpy Btu/lbm
	Temp °F	Rate in/sec					(Pounds Mass per Second)			
25.8	177.3	.000	.000	.00	.00	.250	.0	.0	.0	.00
26.6	175.1	21.764	.000	.62	1.52	.000	7607.0	7607.0	.0	89.50
26.8	174.0	24.126	.000	1.01	1.43	.000	7553.7	7553.7	.0	89.50
28.0	173.1	2.780	.301	1.50	4.48	.331	7217.7	7217.7	.0	89.50
28.9	173.2	2.704	.418	1.63	7.11	.356	7019.0	7019.0	.0	89.50
32.2	173.8	4.556	.629	2.00	16.00	.568	5969.6	5969.6	.0	89.50
32.9	173.9	4.904	.653	2.10	16.12	.597	5690.4	5690.4	.0	89.50
33.9	174.1	4.711	.677	2.23	16.12	.596	5523.8	5523.8	.0	89.50
34.9	174.4	5.106	.694	2.36	16.12	.624	6385.4	5140.2	.0	85.31
36.1	174.8	4.935	.708	2.51	16.12	.621	6234.9	4986.3	.0	85.20
40.6	176.8	4.527	.731	3.00	16.12	.613	5786.2	4520.6	.0	84.80
45.8	179.6	4.225	.739	3.50	16.12	.604	5367.9	4086.0	.0	84.37
51.5	183.0	3.977	.743	4.00	16.12	.595	4988.0	3691.9	.0	83.92
56.0	185.8	2.562	.731	4.35	16.12	.434	1377.9	.0	.0	68.03
59.0	187.6	2.539	.732	4.53	16.12	.433	1378.0	.0	.0	68.03
67.5	194.0	2.476	.733	5.00	16.12	.433	1378.3	.0	.0	68.03
77.0	202.7	2.404	.735	5.51	16.12	.432	1378.6	.0	.0	68.03

Table 6.5.1-15 (cont.) Double-Ended Pump Suction Break – Maximum Safeguards Principle Parameters During Reflood Callaway Plant Utilizing the Replacement Steam Generator

Time Seconds	Flooding		Carryover Fraction	Core Height (ft)	Downcomer Height (ft)	Flow Frac	Total	Injection Accum	Spill	Enthalpy Btu/lbm
	Temp °F	Rate in/sec					(Pounds Mass per Second)			
86.3	212.1	2.334	.737	6.00	16.12	.431	1379.0	.0	.0	68.03
97.0	223.1	2.251	.739	6.54	16.12	.429	1379.4	.0	.0	68.03
106.7	232.5	2.174	.741	7.00	16.12	.427	1379.9	.0	.0	68.03
119.0	242.6	2.078	.743	7.56	16.12	.424	1380.4	.0	.0	68.03
129.0	249.6	2.022	.746	8.00	16.12	.426	1380.3	.0	.0	68.03
141.0	257.0	1.957	.748	8.50	16.12	.429	1380.3	.0	.0	68.03
153.4	263.6	1.892	.751	9.00	16.12	.432	1380.2	.0	.0	68.03
167.0	269.9	1.823	.754	9.52	16.12	.436	1380.1	.0	.0	68.03
180.1	275.0	1.758	.756	10.00	16.12	.441	1380.0	.0	.0	68.03

Table 6.5.1-16 Double-Ended Pump Suction Break Maximum Safeguards Post-Reflood Mass and Energy Releases Callaway Plant Utilizing the Replacement Steam Generator				
Time (seconds)	Break Path No. 1 Flow*		Break Path No. 2 Flow**	
	(lbm/sec)	Thousand (Btu/sec)	(lbm/sec)	Thousand (Btu/sec)
185.10	174.07	218.10	1212.00	208.40
190.10	174.46	218.58	1211.62	208.03
195.10	173.61	217.52	1212.46	207.99
200.10	173.98	217.98	1212.09	207.61
205.10	173.43	217.29	1212.65	207.51
210.10	174.08	218.10	1212.00	207.07
215.10	173.51	217.39	1212.56	206.97
220.10	174.14	218.18	1211.93	206.53
225.10	173.56	217.46	1212.51	206.43
230.10	172.99	216.74	1213.09	206.33
235.10	173.59	217.50	1212.48	205.91
240.10	173.01	216.76	1213.07	205.81
245.10	173.59	217.50	1212.48	205.38
250.10	173.00	216.75	1213.08	205.28
255.10	172.40	216.00	1213.68	205.19
260.10	172.96	216.70	1213.12	204.77
265.10	172.35	215.94	1213.73	204.67
270.10	172.89	216.61	1213.19	204.26
275.10	172.27	215.83	1213.81	204.16
280.10	172.79	216.48	1213.29	203.75
285.10	172.15	215.69	1213.92	203.66
290.10	172.65	216.31	1213.42	203.26
295.10	172.00	215.50	1214.07	203.17
300.10	172.48	216.10	1213.60	202.77
305.10	171.82	215.27	1214.25	202.68
310.10	172.27	215.84	1213.80	202.29
315.10	171.60	214.99	1214.48	202.20
320.10	172.02	215.53	1214.05	201.81

Table 6.5.1-16 Double-Ended Pump Suction Break Maximum Safeguards Post-Reflood Mass and Energy (cont.) Releases Callaway Plant Utilizing the Replacement Steam Generator

Time (seconds)	Break Path No. 1 Flow*		Break Path No. 2 Flow**	
	(lbm/sec)	Thousand (Btu/sec)	(lbm/sec)	Thousand (Btu/sec)
325.10	171.34	214.67	1214.74	201.73
330.10	171.74	215.17	1214.33	201,35
335.10	172.12	215.65	1213.95	200.97
340.10	171.41	214.76	1214.67	200.89
345.10	171.76	215.20	1214.31	200.52
350.10	171.03	214.29	1215.04	200.45
355.10	171.36	214.70	1214.71	200.08
360.10	171.67	215.09	1214.40	199.72
365.10	170.91	214.13	1215.16	199.65
370.10	171.19	214.48	1214.88	199.30
375.10	170.41	213.51	1215.66	199.23
380.10	170.66	213.82	1215.42	198.89
385.10	170.89	214.10	1215.19	198.54
390.10	171.09	214.36	1214.98	198.21
395.10	170.26	213.32	1215.81	198.15
400.10	170.44	213.54	1215.64	197.82
405.10	170.74	213.92	1215.34	197.47
410.10	170.02	213.02	1216.05	197.39
415.10	170.29	213.35	1215.79	197.04
420.10	170.53	213.65	1215.55	196.70
425.10	169.77	212.71	1216.30	196.63
430.10	169.97	212.95	1216.10	196.30
435.10	170.14	213.17	1215.93	195.97
440.10	170.28	213.35	1215.79	195.66
445.10	170.40	213.49	1215.68	195.34
450.10	169.55	212.43	1216.52	195.30
455.10	169.62	212.51	1216.46	195.00
460.10	169.65	212.56	1216.42	194.71
465.10	169.66	212.56	1216.42	194.42

Table 6.5.1-16 (cont.) Double-Ended Pump Suction Break Maximum Safeguards Post-Reflood Mass and Energy Releases Callaway Plant Utilizing the Replacement Steam Generator				
	Break Path No. 1 Flow*		Break Path No. 2 Flow**	
Time (seconds)	(lbm/sec)	Thousand (Btu/sec)	(lbm/sec)	Thousand (Btu/sec)
470.10	169.63	212.52	1216.45	200.06
475.10	169.56	212.44	1216.51	199.77
480.10	169.46	212.31	1216.62	199.49
485.10	169.32	212.14	1216.76	199.22
490.10	169.14	211.91	1216.94	198.96
495.10	169.76	212.70	1216.31	198.48
500.10	169.48	212.35	1216.59	198.24
505.10	169.16	211.94	1216.92	198.02
510.10	169.60	212.49	1216.47	197.59
515.10	169.16	211.94	1216.91	197.39
520.10	169.46	212.32	1216.61	197.00
525.10	168.89	211.60	1217.18	196.84
530.10	169.03	211.78	1217.04	196.48
535.10	169.08	211.84	1217.00	196.15
540.10	169.02	211.76	1217.05	195.85
545.10	168.85	211.56	1217.22	195.58
550.10	168.57	211.20	1217.50	195.34
555.10	168.86	211.57	1217.21	194.94
560.10	168.98	211.71	1217.10	194.58
565.10	168.90	211.62	1217.17	194.28
570.10	168.62	211.26	1217.46	194.04
575.10	168.73	211.40	1217.34	193.68
580.10	168.54	211.17	1217.53	193.41
585.10	168.60	211.24	1217.47	193.07
590.10	168.22	210.76	1217.86	192.85
595.10	168.39	210.98	1217.68	192.47
600.10	90.36	113.21	1295.72	213.78
811.43	90.36	113.21	1295.72	213.78
811.53	95.49	118.51	1290.59	208.03

Table 6.5.1-16 (cont.) Double-Ended Pump Suction Break Maximum Safeguards Post-Reflood Mass and Energy Releases Callaway Plant Utilizing the Replacement Steam Generator

Time (seconds)	Break Path No. 1 Flow* (lbm/sec)	Break Path No. 1 Flow* Thousand (Btu/sec)	Break Path No. 2 Flow** (lbm/sec)	Break Path No. 2 Flow** Thousand (Btu/sec)
815.10	95.41	118.41	1290.67	207.74
1307.15	95.41	118.41	1290.67	207.74
1307.25	84.82	97.59	1301.25	101.95
2221.00	74.66	85.91	1311.41	103.78
2221.10	74.66	85.91	1218.00	208.60
3600.00	66.00	75.95	1226.66	210.16
3600.10	53.58	61.64	1239.09	190.86
10000.00	38.96	44.83	1253.70	193.11
100000.00	20.83	23.97	1271.83	195.91
1000000.00	8.92	10.27	1283.74	197.74
10000000.00	2.80	3.22	1289.87	198.68

* Mass and energy exiting the SG side of the break
** Mass and energy exiting the pump side of the break

Table 6.5.1-17 Double-Ended Pump Suction Break Mass Balance Maximum Safeguards Callaway Nuclear Plant Utilizing the Replacement Steam Generator

		Mass Balance						
Time (Seconds)		0.00	25.80	25.80+δ	180.06	811.53	1307.15	3600.00
		Mass (Thousand lbm)						
Initial	In RCS and ACC	807.40	807.40	807.40	807.40	807.40	807.40	807.40
Added Mass	Pumped Injection	0.00	0.00	0.00	198.89	1074.11	1761.06	4810.31
	Total Added	0.00	0.00	0.00	198.89	1074.11	1761.06	4810.31
*** Total Available[1] ***		807.40	807.40	807.40	1006.29	1881.51	2568.47	5617.71
Distribution	Reactor Coolant	580.70	47.82	78.62	141.77	141.77	141.77	141.77
	Accumulator	226.70	173.93	143.12	0.00	0.00	0.00	0.00
	Total Contents	807.40	221.75	221.75	141.77	141.77	141.77	141.77
Effluent	Break Flow	0.00	585.63	585.63	853.00	1728.21	2415.17	5464.42
	ECCS Spill	0.00	0.00	0.00	0.00	0.00	0.00	0.00
	Total Effluent	0.00	585.63	585.63	853.00	1728.21	2415.17	5464.42
*** Total Accountable[2] ***		807.44	807.38	807.38	994.77	1869.98	2556.94	5606.19

1. Total available is the sum of all sources of mass that could be released to the containment post-LOCA.
2. Total accountable represents the mass that was calculated to be released to the containment. The difference between total available and total accountable is basically the mass calculated to be stored in the RCS.

Table 6.5.1-18 Double-Ended Pump Suction Break Energy Balance Maximum Safeguards Callaway Plant Utilizing the Replacement Steam Generator

		Energy Balance						
Time (Seconds)		.00	25.80	25.80+δ	180.06	811.53	1307.15	3600.00
		Energy (Million Btu)						
Initial Energy	In RCS, ACC, SG	957.34	957.34	957.34	957.34	957.34	957.34	957.34
Added Energy	Pumped Injection	.00	.00	.00	13.53	73.07	119.81	480.55
	Decay Heat	.00	8.36	8.36	27.70	85.49	123.04	261.78
	Heat From Secondary	.00	-.79	-.79	-.79	6.40	10.75	10.75
	Total Added	.00	7.57	7.57	40.44	164.95	253.60	753.08
*** Total Available[1] ***		957.34	964.92	964.92	997.78	1122.30	1210.94	1710.43
Distribution	Reactor Coolant	343.94	11.51	14.26	38.68	38.68	38.68	38.68
	Accumulator	20.29	15.57	12.81	.00	.00	.00	.00
	Core Stored	24.01	11.92	11.92	5.09	4.91	4.63	3.33
	Primary Metal	158.77	150.51	150.51	124.88	87.25	72.97	53.21
	Secondary Metal	106.08	105.29	105.29	95.81	70.87	56.56	41.48
	Steam Generator	304.25	301.38	301.38	269.45	198.47	161.59	120.68
	Total Contents	957.34	596.18	596.18	533.91	400.17	334.42	257.37
Effluent	Break Flow	.00	368.16	368.16	452.94	711.19	857.44	1435.63
	ECCS Spill	.00	.00	.00	.00	.00	.00	.00
	Total Effluent	.00	368.16	368.16	452.94	711.19	857.44	1435.63
*** Total Accountable[2] ***		957.34	964.33	964.33	986.85	1111.36	1191.86	1693.01

1. Total available is the sum of all sources of mass that could be released to the containment post-LOCA.
2. Total accountable represents the mass that was calculated to be released to the containment. The difference between total available and total accountable is basically the mass calculated to be stored in the RCS.

Table 6.5.1-19 Double-Ended Hot Leg Break Sequence of Events Callaway Plant Utilizing the Replacement Steam Generator	
Time (sec)	**Event Description**
0.0	Break Occurs, Reactor Trip and Loss of Offsite Power are Assumed
3.9	Low Pressurizer Pressure SI Setpoint – 1,715 psia Reached in Blowdown
14.6	Broken Loop Accumulator Begins Injecting Water
14.9	Intact Loop Accumulator Begins Injecting Water
26.2	End of Blowdown Phase

Table 6.5.1-20 Double-Ended Pump Suction Break Minimum Safeguards Sequence of Events Callaway Plant Utilizing the Replacement Steam Generator	
Time (sec)	**Event Description**
0.0	Break Occurs, Reactor Trip and Loss of Offsite Power are Assumed
4.3	Low Pressurizer Pressure SI Setpoint – 1,715 psia Reached in Blowdown
17.5	Broken Loop Accumulator Begins Injecting Water
17.8	Intact Loop Accumulator Begins Injecting Water
26.4	End of Blowdown Phase
48.3	Safety Injection Begins
53.3	Broken Loop Accumulator Water Injection Ends
55.0	Intact Loop Accumulator Water Injection Ends
194.5	End of Reflood Phase
3063.0	Cold Leg Recirculation Begins
1.0E+07	Transient Modeling Terminated

Table 6.5.1-21 Double-Ended Pump Suction Break Maximum Safeguards Sequence of Events Callaway Plant Utilizing the Replacement Steam Generator	
Time (sec)	**Event Description**
0.0	Break Occurs, Reactor Trip and Loss of Offsite Power are Assumed
4.3	Low Pressurizer Pressure SI Setpoint – 1,715 psia Reached in Blowdown
17.6	Broken Loop Accumulator Begins Injecting Water
18.0	Intact Loop Accumulator Begins Injecting Water
25.8	End of Blowdown Phase
34.3	Safety Injection Begins
54.56	Broken Loop Accumulator Water Injection Ends
54.91	Intact Loop Accumulator Water Injection Ends
180.6	End of Reflood Phase
2221.	Cold Leg Recirculation Begins
1.0E+07	Transient Modeling Terminated

Table 6.5.1-22 LOCA Mass and Energy Release Analysis Core Decay Heat Fraction	
Time (sec)	Decay Heat Generation Rate (Btu/Btu)
1.00E+01	0.053876
1.50E+01	0.050401
2.00E+01	0.048018
4.00E+01	0.042401
6.00E+01	0.039244
8.00E+01	0.037065
1.00E+02	0.035466
1.50E+02	0.032724
2.00E+02	0.030936
4.00E+02	0.027078
6.00E+02	0.024931
8.00E+02	0.023389
1.00E+03	0.022156
1.50E+03	0.019921
2.00E+03	0.018315
4.00E+03	0.014781
6.00E+03	0.013040
8.00E+03	0.012000
1.00E+04	0.011262
1.50E+04	0.010097
2.00E+04	0.009350
4.00E+04	0.007778
6.00E+04	0.006958
8.00E+04	0.006424
1.00E+05	0.006021
1.50E+05	0.005323
4.00E+05	0.003770
6.00E+05	0.003201
8.00E+05	0.002834
1.00E+06	0.002580

6.5.2 Short-Term LOCA Mass and Energy Releases

6.5.2.1 Introduction

An evaluation was conducted to determine the effect of the replacement steam generators on the short-term LOCA-related mass and energy releases that support subcompartment analyses discussed in Chapter 6.2.1.2.3.b.1 of the Callaway Plant FSAR. From the FSAR (Reference 1) the following licensing position is stated. "On May 31, and October 26, 1984, Union Electric submitted Westinghouse topical reports (WCAPs-10500, -10501, -10690, and -10691) to the NRC in order to demonstrate compliance with the revised GDC GDC-4, which provides for the application of "leak-before-break" technology to eliminate protective devices against dynamic loads resulting form postulated ruptures of primary coolant loops. By letter dated October 28, 1986, the NRC confirmed its finding that, based on the UE submittals, Callaway is in compliance with the revised GDC-4."

However, the evaluation (that is, the original work) of loads on the containment subcompartment are retained in the FSAR. The following LOCA breaks are analyzed in the FSAR for the following regions of the containment:

1. For the steam generator loop compartments, the design-basis break is a steam generator inlet elbow longitudinal split with a break flow area of 863 square inches, a double-ended steam generator outlet nozzle break restrained to a break flow area of 426 square inches, and a double-ended reactor coolant pump outlet nozzle break restrained to a break flow area of 236 square inches.

2. The pressurizer compartment is divided into 2 compartments: a) the pressurizer compartment and b) the pressurizer surge line compartment.

The design basis-break is the double-ended pressurizer surge line break. This evaluation addresses the impact of the RSG Program and other relevant issues on the current licensing basis for postulated breaks in the containment subcompartments.

6.5.2.2 Input Parameters and Assumptions

Any changes in RCS volume or steam generator liquid/steam mass and volume have no effect on the releases because of the short duration of the postulated accident. Any volumetric changes are small and have no impact on the subcompartment model. Therefore, the only change that needs to be addressed for this program is the decreased RCS coolant temperatures.

For this evaluation, an RCS pressure of 2,300 psia, a vessel/core inlet temperature of 535.2°F, and a hot leg temperature of 600.2°F, were considered for the RSG.

6.5.2.3 Description of Analysis

The subcompartment analysis is performed to ensure that the walls of a subcompartment can maintain their structural integrity during the short pressure pulse (generally less than 3 seconds) that accompanies a high energy line pipe rupture within the subcompartment. The magnitude of the pressure differential across the walls is a function of several parameters, which include the blowdown mass and energy release rates, the subcompartment volume, vent areas, and vent flow behavior. The blowdown mass and energy release rates are affected by the initial RCS temperature conditions. Since short-term releases are linked directly to the critical mass flux, which increases with decreasing temperatures, the short-term LOCA releases would be expected to increase due to any reductions in RCS coolant temperature conditions. Short-term blowdown transients are characterized by a peak mass and energy release rate that occurs during a subcooled condition. Therefore, the Zaloudek correlation, which models this condition, is currently used in the short-term LOCA mass and energy release analyses with the SATAN computer program. Details of the Zaloudek correlation are presented in Reference 4.

This calculation was used to conservatively evaluate the impact of the changes in RCS temperature conditions due to the RSG Program on the short-term releases. This was accomplished by maximizing the reservoir pressure and maximizing the RCS inlet and outlet temperatures for the current analysis of record, and, by minimizing the RCS inlet and outlet temperatures for the RSG data. Since this maximizes the change in short-term LOCA mass and energy releases, data representative of the lowest inlet and outlet temperatures with uncertainty subtracted was used for the RSG evaluation.

Steam Generator Compartments

Callaway is approved for LBB (Reference 1) for the primary loop and LBB eliminates the dynamic effects of these pipe ruptures from the design basis. This means that the current reactor coolant loop breaks no longer have to be considered for subcompartment short-term effects. Since these breaks have been eliminated, the next largest branch nozzles must be considered for design verification. This typically includes the accumulator line (120 in^2), and residual heat removal (RHR) line (173 in^2). The current RCS breaks analyzed for the Callaway Plant (Callaway FSAR Section 6.2.1.2.1.a (Reference 1)) are a 763 in^2 steam generator inlet elbow break (hot leg), a 436 in^2 steam generator outlet nozzle break (pump suction), and a 236 in^2 reactor coolant pump outlet nozzle break (cold leg). The release from these breaks, even after consideration for the reduced temperatures, would bound the smaller breaks permitted under LBB. The differential loadings due to these smaller breaks are significantly reduced when compared to the RCS breaks. For example, the mass and energy release from an accumulator line (cold leg break) would be (120 in^2/236 in^2) = 0.5085 or 51 percent of the FSAR reactor coolant pump outlet nozzle release. The release from the RHR recirculation piping hot leg piping break would be (173 in^2/436 in^2) = 0.3968 or 40 percent of the FSAR steam generator inlet nozzle break. LBB has eliminated any break for the pump suction piping including the FSAR 436 in^2 steam generator outlet nozzle break. Therefore, based on the large RCS breaks in the Callaway Plant FSAR, these current releases remain bounding.

Pressurizer Compartment and Surge Line Compartment

The Callaway Plant FSAR discusses the design of the pressurizer compartment as having two subcompartments, a pressurizer compartment and a surge line compartment. These two regions can be affected by breaks in the surge line, pressurizer spray line, and the steam generator loop breaks listed above. However, the Callaway Plant FSAR states that design basis for the pressurizer compartment is the double-ended pressurizer surge line break. For the pressurizer region, the spray line results could increase by $\simeq$ 7 percent. However, the current surge line releases would remain bounding for this region since the surge line break area is much larger than the spray line (that is, > a factor of 4). Using the current surge line break releases would bound the effect of a 7-percent increase in releases for the spray line.

The pressurizer surge line break was modeled as a double ended break of 0.683 ft^2 (total area 2 * 0.683). The decrease in RCS hot leg temperature coupled with the increase in pressure to 2,300 psia will result in an increase in the mass and energy releases for the pressurizer surge line break. On the hot leg side of the break, the increase would be 21.167 percent in the mass flow rate and 15.12 percent in the energy release rate. These increases are significant and would require re-evaluation of the pressurizer compartment pressure loads from a LOCA. However, AmerenUE is currently in the process of licensing LBB for the pressurizer surge line (Reference 2). Based on the assumption of successful licensing of LBB for the pressurizer surge line, the surge line break in the FSAR, without the increase indicated above, would continue to be bounding for pressurizer and surge line compartment pressure loading following a postulated LOCA.

6.5.2.4 Acceptance Criteria

The NSSS design parameters discussed in Section 2.0 were used in this evaluation.

The NRC's NUREG-800, Section 6.2.1.3, "Mass and Energy Release Analysis for Postulated Loss-of-Coolant Accidents," subsection II, provides guidance on the NRC's expectations for what must be included in a LOCA mass and energy release calculation. The NRC has determined that the Westinghouse mass and energy models described in WCAP-8264-P-A, Rev. 1 (Reference 3) satisfy those expectations.

6.5.2.5 Results and Conclusion

The short-term LOCA-related mass and energy releases discussed in Chapter 6.2.1.2 of the FSAR have been reviewed to assess the effects associated with the replacement steam generator program conditions for the Callaway Plant. Since the Callaway Plant is approved for LBB, the decrease in mass and energy releases associated with the smaller RCS branch line breaks, as compared to the larger RCS pipe breaks, more than offsets the increased releases associated with the Callaway Plant RSG Program conditions. Therefore, the current licensing basis subcompartment analyses that consider breaks in the primary loop RCS piping (that is, steam generator subcompartment region) remain bounding. Likewise, successful licensing of LBB for the pressurizer surge line will allow the current FSAR releases for the pressurizer and surge line compartments to remain bounding.

6.5.2.6 References

1. "Callaway Nuclear Plant Final Safety Analysis Report," Rev. OL-13, May 2003.

2. WCAP-15983-P, "Technical Justification for Eliminating Pressurizer Surge Line Rupture as the Structural Design Basis for Callaway Nuclear Plant," February 2003.

3. WCAP-8264-P-A, Rev. 1, (Proprietary) and WCAP-8312-A (Non-Proprietary), "Westinghouse Mass and Energy Release Data For Containment Design," August 1975.

4. WCAP-8302-P, "SATAN VI Program: Comprehensive Space-Time Dependent Analysis of Loss-of-Coolant," June 1974.

6.6 MAIN STEAM LINE BREAK MASS AND ENERGY RELEASES

6.6.1 MSLB Mass and Energy Releases Inside Containment

6.6.1.1 Introduction and Background

Steam line ruptures occurring inside the reactor containment structure may result in significant releases of high-energy fluid to the containment environment, producing elevated containment temperatures and pressures. The magnitude of the releases following a steam line rupture is dependent upon the plant initial operating conditions and the size of the rupture as well as the configuration of the plant steam system and the containment design. These variations make it difficult to determine the absolute worst cases for either containment pressure or temperature evaluation following a steam line break. The analysis considers a variety of postulated pipe breaks encompassing wide variations in plant operation, safety system performance, and break size in determining the main steam line break (MSLB) mass and energy (M&E) releases for use in containment analysis.

The only event other than the MSLB that results in a breach of the secondary-side piping is the feedwater line break (FLB). The feedwater enthalpy at any power is less than the enthalpy of saturated steam at the secondary-side operating pressures. Therefore, the long-term integrated energy released following an FLB is bounded by the long-term integrated energy released following an MSLB. It is expected that the containment pressure and temperature responses to the M&E releases from an FLB would be bounded by the containment responses following the MSLB event.

6.6.1.2 Input Parameters and Assumptions

The analysis inputs, assumptions, and methods pertaining to the MSLB M&E releases inside containment are presented in this section.

To determine the effects of plant power level and break area on the M&E releases from a ruptured steam line, spectra of both variables have been evaluated. At plant power levels of 102 percent, 70 percent, 30 percent, and near 0 percent of nominal full-load power, 2 break sizes have been defined. These break areas are defined as the following.

1. A full double-ended rupture (DER) downstream of the flow restrictor in one steam line. Note that a DER is defined as a rupture in which the steam pipe is completely severed and the ends of the break fully displace from each other. The full DER represents the largest break of the main steam line producing the highest mass flow rate from the faulted-loop steam generator.

2. A small split rupture that will neither generate a steam line isolation signal from the Westinghouse solid-state protection system (SSPS) nor result in water entrainment in the break effluent. Reactor protection and safety injection actuation functions are obtained from containment pressure signals.

The 24 cases included in the Callaway analysis for the Replacement Steam Generator (RSG) Program have been chosen based on the selection of similar steam line ruptures included in the analyses presented in the Callaway Final Safety Analysis Report (FSAR), subsection 6.2.1.4. The cases, listed in subsection 6.6.1.3 of this report, have been analyzed at the conditions associated with the Framatome-design Model 73/19T RSGs. Other assumptions regarding important plant conditions and features are discussed in the following paragraphs.

6.6.1.2.1 Initial Power Level

Steam line breaks can be postulated to occur with the plant in any operating condition ranging from hot shutdown to full power. Since steam generator water mass decreases with increasing power level, breaks occurring at lower power levels will generally result in a greater total mass release to the containment. However, because of increased stored energy in the primary side of the plant, increased heat transfer in the steam generators, and additional energy generation in the fuel, the energy release to the containment from breaks postulated to occur during full-power, or near full-power, operation may be greater than for breaks occurring with the plant in a low-power, or hot-shutdown, condition. Additionally, pressure in the steam generators changes with increasing power and has a significant influence on the rate of blowdown.

Because of the opposing effects on mass versus energy release for the MSLB due to a change in initial power level, a single power level cannot be specified as the worst case for either the containment pressure response or the containment temperature response. Therefore, representative power levels including 102 percent, 70 percent, 30 percent, and near 0 percent of nominal full nuclear steam supply system (NSSS) power conditions have been investigated for Callaway based on the information in Reference 1. Reference 1 has been reviewed and approved by the Nuclear Regulatory Commission (NRC) for use in MSLB analysis inside containment. Additional discussion is provided in subsection 6.6.1.3 of this report.

In general, the plant initial conditions are assumed to be at the nominal value corresponding to the initial power for that case, with appropriate uncertainties included. Tables 6.6.1-1 and 6.6.1-2 identify the values assumed for NSSS power, reactor coolant system (RCS) pressure, RCS vessel average temperature, RCS flow, pressurizer water volume, steam generator water level, steam generator temperature and pressure, and feedwater enthalpy corresponding to each power level analyzed. Steam line break M&E releases assuming an RCS average temperature at the high end of the Tavg window are conservative with respect to similar releases at the low end of the Tavg window. At the high end, there is more M&E available for release into containment. The thermal design flow rate has been used for the RCS flow input consistent with the assumptions documented in Reference 1. The thermal design flow rate is also consistent with other MSLB analysis assumptions related to nonstatistical treatment of uncertainties, as well as RCS thermal-hydraulic inputs related to pressure drops and rod drop time.

Uncertainties on the initial conditions assumed in the analysis for the RSG Program have been applied only to the RCS average temperature (4.3°F), the steam generator mass (6.2-percent narrow-range span), and the power fraction at full power (2 percent). Nominal values are adequate for the initial conditions associated with pressurizer pressure and pressurizer water level. Uncertainty conditions are only applied to those parameters that could increase the amount of mass or energy discharged into containment.

6.6.1.2.2 Single-Failure Assumption

In a manner consistent with the standard approach for licensing-basis analyses, various single failures have been identified and used in the spectrum of MSLB cases analyzed. Most cases analyzed consider only one single failure. One of these failures is considered as part of the containment response analysis.

The postulated single failures (discussed also in Reference 1) that increase the MSLB M&E releases to containment are discussed below:

1. Failure of the Main Steam Isolation Valve (MSIV) in the Faulted Loop

   The main steam line isolation function is accomplished via the MSIV in each of the four steam lines. Each valve closes on an isolation signal to terminate steam flow from the associated steam generator. The main steam line rupture upstream of this valve, as postulated for the inside-containment analysis, creates a situation in which the steam generator on the faulted loop cannot be isolated, even when the MSIV successfully closes. The break location allows a continued blowdown from the faulted-loop steam generator until it is empty and all sources of feedwater and auxiliary feedwater addition are terminated. If the faulted-loop MSIV fails to close, blowdown from more than one steam generator is terminated by the closure of the corresponding MSIV for each intact-loop steam generator. Therefore, there is no failure of a single MSIV that could cause continued blowdown from multiple steam generators.

   In addition to the continued blowdown from the faulted-loop steam generator after MSIV closure, the steam in the unisolable sections of the main steam system need to be considered. The MSIV failure impacts the M&E releases since a failed MSIV will result in a larger unisolable steam line volume.

2. Failure of the Main Feedwater Isolation Valve (MFIV) in the Faulted Loop

   If the MFIV in the feedwater line to the faulted steam generator is assumed to fail in the open position, backup isolation is provided via the main feedwater flow regulator valve (FRV) closure. The additional inventory between the MFIV and the FRV in the faulted loop would be available to be released to containment.

6.6.1.2.3 Main Feedwater System

The rapid depressurization that occurs following a steam line rupture typically results in large amounts of water being added to the steam generators through the main feedwater system. Rapid-closing MFIVs or FRVs in the main feedwater lines limit this effect. The feedwater addition that occurs prior to closing of the MFIV or FRV influences the steam generator blowdown in several ways. First, because the water entering the steam generator is subcooled, it lowers the steam pressure thereby reducing the flow rate out of the break. As the steam generator pressure decreases, the fluid in the feedwater lines downstream of the isolation valves will flash into the steam generators providing additional secondary fluid which may exit out of the rupture. Secondly, the increased flow causes an increase in the total heat transfer from the primary to secondary systems resulting in greater integrated energy being released out of the break.

Following the initiation of the MSLB, main feedwater flow is conservatively modeled by assuming that sufficient feedwater flow is provided to match or exceed the steam flow prior to reactor trip, as shown in Table 6.6.1-3. The initial increase in feedwater flow (until fully isolated) is in response to the feedwater control valve opening up in response to the steam flow/feedwater flow mismatch, or the decreasing steam generator water level as well as due to a lower backpressure on the feedwater pump as a result of the depressurizing steam generator. This maximizes the total mass addition prior to feedwater isolation. The feedwater isolation response time, following the safety injection signal, is assumed to be a total of 17 seconds, accounting for delays associated with signal processing plus MFIV stroke time. For the circumstance in which the MFIV in the faulted loop fails to close, there is no effect on the feedwater isolation response time since the total delay for the FRV closure is also 17 seconds.

Following feedwater isolation, as the steam generator pressure decreases, some of the fluid in the feedwater lines downstream of the isolation or regulator valve may flash to steam if the feedwater becomes saturated. This unisolable feedwater line volume is an additional source of fluid that can increase the mass discharged out of the break. The unisolable volume in the feedwater lines is maximized for the faulted loop. Feedwater line piping volumes available for steam flashing in this analysis are shown in Table 6.6.1-3.

Steam line break M&E releases assuming a main feedwater temperature at the high end of the feedwater temperature window are conservative with respect to similar releases at the low end of the feedwater temperature window. At the high end, there is more energy available for release into containment.

6.6.1.2.4 Auxiliary Feedwater System

Generally, within the first minute following a steam line break, the auxiliary feedwater (AFW) system is initiated on any one of several protection system signals. Addition of AFW to the steam generators will increase the secondary mass available for release to containment. The AFW flow to the faulted and intact steam generators has been assumed to be a function of the backpressure on the AFW pumps as a result of the depressurizing steam generator in the steam line break analysis inside containment. Auxiliary feedwater flow to the faulted-loop steam generator has been assumed up until the time of operator action at 10 minutes after event initiation to isolate the flow to the steam generator near the break location. Auxiliary feedwater system assumptions that have been used in the analysis are presented in Table 6.6.1-3.

6.6.1.2.5 Steam Generator Fluid Mass

A maximum initial steam generator mass in the faulted-loop steam generator has been used in all of the analyzed cases. The use of a high faulted-loop initial steam generator mass maximizes the steam generator inventory available for release to containment. The initial mass has been calculated as the value corresponding to the programmed level +6.2-percent narrow-range span and assuming 0-percent tube plugging. This assumption is conservative with respect to the RCS cooldown through the faulted-loop steam generator resulting from the steam line break.

6.6.1.2.6 Steam Generator Reverse Heat Transfer

Once the steam line isolation is complete, the steam generators in the intact loops may become sources of energy that can be transferred to the steam generator with the broken steam line. This energy transfer occurs via the primary coolant. As the primary plant cools, the temperature of the coolant flowing in the steam generator tubes could drop below the temperature of the secondary fluid in the intact steam generators, resulting in energy being returned to the primary coolant. This energy is then available to be transferred to the steam generator with the broken steam line. When applicable, the effects of reverse steam generator heat transfer are included in the results.

6.6.1.2.7 Break Flow Model

Piping discharge resistances are not included in the calculation of the releases resulting from the steam line ruptures (Moody Curve for an $f(\ell / D) = 0$ is used). This is consistent with the expectations of the NRC as presented in subsection 6.2.1.4 of the Standard Review Plan (SRP). No entrainment is assumed in the break effluent. The assumption of saturated steam being released for all break types is a conservative assumption that maximizes the energy release into containment.

6.6.1.2.8 Steam Line Volume Blowdown

The contribution from the secondary plant steam piping is included in the M&E release calculations. The flow rate is determined using the Moody correlation, the pipe cross-sectional area, and the initial steam pressure. For all steam line break cases analyzed for the RSG Program, the unisolable steam line mass is included in the mass exiting the break from the time of steam line isolation until the unisolable mass is completely released to containment. The steam piping volume assumed in the analysis is 787 ft^3 with no MSIV failure and 5,130 ft^3 assuming an MSIV failure in the faulted-loop steam line.

6.6.1.2.9 Main Steam Line Isolation

Steam line isolation is assumed in all 4 loops to terminate the blowdown from the 3 intact steam generators. A conservative delay time of 17 seconds, accounting for delays associated with signal processing plus MSIV stroke time, with unrestricted steam flow through the valve during the valve stroke, has been assumed.

6.6.1.2.10 Protection System Actuations

The protection systems available to mitigate the effects of an MSLB inside containment include reactor trip, safety injection, steam line isolation, and feedwater isolation. The protection system actuation signals and associated setpoints that have been modeled in the analysis are identified in Table 6.6.1-4. The setpoints used are conservative values with respect to the plant-specific values delineated in the Technical Specifications for the RSG Program. The specific functions credited in the Callaway plant-specific analysis are documented in subsection 6.6.1.5.

6.6.1.2.11 Safety Injection System

Minimum safety injection system (SIS) flow rates corresponding to the failure of one SIS train have been assumed in this analysis. A minimum SI flow is conservative since the reduced boron addition maximizes a return to power resulting from the RCS cooldown. The higher power generation increases heat transfer to the secondary side, maximizing steam flow out of the break. The delay time to achieve full SI flow is assumed to be 27 seconds for this analysis with offsite power available. A coincident loss of offsite power is not assumed for the analysis of the steam line break inside containment since the M&E releases would be reduced due to the loss of forced reactor coolant flow, resulting in less primary-to-secondary heat transfer.

6.6.1.2.12 Reactor Coolant System Metal Heat Capacity

As the primary side of the plant cools, the temperature of the reactor coolant drops below the temperature of the reactor coolant piping, the reactor vessel, the reactor coolant pumps, and the steam generator thick-metal mass and tubing. As this occurs, the heat stored in the metal is available to be transferred to the steam generator with the broken line. The effects of this RCS metal heat are included in the results using conservative thick-metal masses and heat transfer coefficients.

6.6.1.2.13 Core Decay Heat

Core decay heat generation assumed in calculating the steam line break M&E releases is based on the 1979 American Nuclear Society (ANS) Decay Heat + 2σ model (Reference 2).

6.6.1.2.14 Rod Control

The rod control system is conservatively assumed to be in manual operation for all steam line break analyses. Assuming that the reactor is in manual rod control allows for a greater RCS cooldown prior to the reactor trip signal, which maximizes the reactivity feedback at end-of-cycle conditions and produces a greater post-trip power increase.

6.6.1.2.15 Core Reactivity Coefficients

Conservative core reactivity coefficients corresponding to end-of-cycle conditions are used to maximize the reactivity feedback effects resulting from the steam line break. Use of maximum reactivity feedback results in higher power generation if the reactor returns to criticality, thus maximizing heat transfer to the secondary side of the steam generators.

6.6.1.3 Description of Analyses and Evaluations

The system transient that provides the break flows and the energy release rates from the steam line break inside containment has been analyzed with the RETRAN (Reference 3) computer code. Blowdown M&E releases determined using RETRAN include the effects of core power generation, main and auxiliary feedwater additions, engineered safeguards systems, RCS thick-metal heat storage including steam generator thick-metal mass and tubing, and reverse steam generator heat transfer. The use of the RETRAN code for the analysis of the MSLB M&E releases inside containment is documented in WCAP-14882-P-A (Reference 4), which has been reviewed and approved by the NRC for use in Westinghouse non-loss-of-coolant-accident (non-LOCA) safety analyses. As noted in Reference 4, no entrainment is assumed in the break effluent. The assumption of saturated steam being released from the break location is a conservative assumption that maximizes the energy release into containment.

The Callaway NSSS has been analyzed to determine the transient mass releases and the energy release rates inside containment following a steam line break event. The tables of M&E releases are used as input conditions to the analysis of the containment response.

The following licensing-basis cases of the MSLB inside containment have been analyzed at the noted conditions for the RSG Program:

- Full double-ended (1.390 ft^2) rupture at 102-percent power – no single failure in MSLB transient
- Full double-ended (1.390 ft^2) rupture at 102-percent power – MSIV failure
- Full double-ended (1.390 ft^2) rupture at 102-percent power – MFIV failure
- 0.750 ft^2 split rupture at 102-percent power – no single failure in MSLB transient
- 0.750 ft^2 split rupture at 102-percent power – MSIV failure
- 0.750 ft^2 split rupture at 102-percent power – MFIV failure
- Full double-ended (1.390 ft^2) rupture at 70-percent power – no single failure in MSLB transient
- Full double-ended (1.390 ft^2) rupture at 70-percent power – MSIV failure
- Full double-ended (1.390 ft^2) rupture at 70-percent power – MFIV failure
- 0.852 ft^2 split rupture at 70-percent power – no single failure in MSLB transient
- 0.852 ft^2 split rupture at 70-percent power – MSIV failure
- 0.852 ft^2 split rupture at 70-percent power – MFIV failure
- Full double-ended (1.390 ft^2) rupture at 30-percent power – no single failure in MSLB transient
- Full double-ended (1.390 ft^2) rupture at 30-percent power – MSIV failure
- Full double-ended (1.390 ft^2) rupture at 30-percent power – MFIV failure
- 0.905 ft^2 split rupture at 30-percent power – no single failure in MSLB transient
- 0.905 ft^2 split rupture at 30-percent power – MSIV failure
- 0.905 ft^2 split rupture at 30-percent power – MFIV failure
- Full double-ended (1.390 ft^2) rupture at 2-percent power – no single failure in MSLB transient
- Full double-ended (1.390 ft^2) rupture at 2-percent power – MSIV failure
- Full double-ended (1.390 ft^2) rupture at 2-percent power – MFIV failure
- 0.803 ft^2 split rupture at 2-percent power – no single failure in MSLB transient
- 0.803 ft^2 split rupture at 2-percent power – MSIV failure
- 0.803 ft^2 split rupture at 2-percent power – MFIV failure

For the double-ended rupture cases, the forward-flow cross-sectional area from the faulted-loop steam generator is limited by the integral flow restrictor area of 1.390 ft^2, which is less than the actual area of 3.565 ft^2 for the reverse-direction cross-sectional flow area of the piping inside containment. The full DER represents the break producing the highest mass flow rate from the faulted-loop steam generator. Smaller DER break sizes are represented by a reduction in the initial steam blowdown rate at the time of the break. Therefore, no other DER break sizes have been considered other than the full DER.

For the split-break MSLB cases, the break area is smaller than the area of a single integral flow restrictor. The flow rate from all steam generators prior to MSIV closure and the flow rate from a single steam generator after MSIV closure supply the steam flow to the break. All the cross-sectional split-rupture areas have been redefined based on the assumption of operation with the Framatome-design Model 73/19T RSGs with plant-specific values for the secondary-side protection system setpoints incorporated into the RETRAN NSSS model. Each break size as a function of power is the largest area that does not produce a steam line isolation signal from the Westinghouse SSPS, a primary-side reactor trip signal such as overpower ΔT, nor results in water entrainment in the break effluent as discussed in Reference 1.

6.6.1.4 Acceptance Criteria

The MSLB is classified as an ANS Condition IV event, an infrequent fault. The acceptance criteria associated with the steam line break event resulting in an M&E release inside containment is based on an analysis that provides sufficient conservatism to show that the containment design margin is maintained. The specific criteria applicable to this analysis are related to the assumptions regarding power level, stored energy, the break flow model, main and AFW flow, steam line and feedwater isolation, and single failure such that the containment peak pressure and temperature are maximized. These analysis assumptions have been included in this steam line break M&E release analysis as discussed in Reference 1 and subsection 6.6.1.2 of this report.

6.6.1.5 Results

Using the MSLB analysis methodology documented in Reference 1 as a basis, including parameter changes associated with the RSG Program, the M&E release rates for each of the steam line break cases noted in subsection 6.6.1.3 have been developed for use in containment pressure and temperature response analyses. Table 6.6.1-5 provides the sequence of events for each of the 24 steam line break sizes analyzed for Callaway with the Model 73/19T Framatome RSGs.

For the double-ended rupture MSLB at all power levels, the first protection system signal actuated is low steam line pressure (2-of-3 channels per loop, lead/lag compensated in each channel) in any loop that initiates steam line isolation and safety injection; the SI signal produces a reactor trip signal. Feedwater system isolation and AFW actuation occur as a result of the SI signal.

For the split-rupture steam line breaks at all power levels, no mitigation signals are received from either the reactor protection system or any secondary-side signals produced by the engineered safety features actuation system. The first protection system signal actuated is assumed to be the high-1 containment pressure (2-of-3 channels), which initiates SI; the SI signal produces a reactor trip signal. Feedwater system isolation and AFW actuation occur as a result of the SI signal. Steam line isolation is initiated following receipt of the high-2 containment pressure (2-of-3 channels) signal.

The turbine stop valve is assumed to close instantly following the reactor trip signal; the delay time used in the steam line break mass and energy releases inside containment is 0.0 seconds.

6.6.1.6 Conclusions

The M&E releases from the 24 steam line break cases inside containment have been analyzed at the conditions defined by the RSG Program. The assumptions delineated in subsection 6.6.1.2 have been included in the steam line break analysis such that conservative M&E releases are calculated. The steam mass and energy releases discussed in this section have been provided for use in the containment response analysis for Callaway.

6.6.1.7 References

1. WCAP-8822 (Proprietary) and WCAP-8860 (Non-Proprietary), "Mass and Energy Releases Following a Steam Line Rupture," September 1976; WCAP-8822-S1-P-A (Proprietary) and WCAP-8860-S1-A (Non-Proprietary), "Supplement 1 – Calculations of Steam Superheat in Mass/Energy Releases Following a Steam Line Rupture," September 1986; WCAP-8822-S2-P-A (Proprietary) and WCAP-8860-S2-A (Non-Proprietary), "Supplement 2 – Impact of Steam Superheat in Mass/Energy Releases Following a Steam Line Rupture for Dry and Subatmospheric Containment Designs," September 1986.

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

3. EPRI NP-1850-CCMA, "RETRAN-02 - A Program for Transient Thermal-Hydraulic Analysis of Complex Fluid Flow Systems," J. H. McFadden, et al., April 1984.

4. WCAP-14882-P-A (Proprietary) and WCAP-15234-A (Non-Proprietary), "RETRAN-02 Modeling and Qualification for Westinghouse Pressurized Water Reactor Non-LOCA Safety Analysis," April 1999 (Proprietary) and May 1999 (Non-Proprietary).

Table 6.6.1-1 Nominal and Initial Plant Parameters for Steam Generator Replacement[1] MSLB M&E Releases Inside Containment

Plant Parameter	Nominal	Full Power Initial
NSSS Power, MWt	3,579	3,650.6
Core Power, MWt	3,565	3,559
Net RCS Heat Input, MWt	14	20
Reactor Coolant Flow (total), gpm (Thermal Design Flow)	374,400	374,400
Pressurizer Pressure, psia	2250	2250
Pressurizer Water Volume (% span)	60.0	60.0
Reactor Coolant Vessel Average Temperature, °F	588.4	592.7
Steam Generator[2]		
Steam Temperature, °F	547.2	553.1
Steam Pressure, psia	1,022	1,072
Feedwater Temperature, °F	446	446
Water Level, % narrow-range span	51.35	>57.5 (58.4)
Zero-Load Temperature, °F	557	557

Notes:

1. Noted values correspond to plant conditions defined by 0% steam generator tube plugging and the high end of the RCS Tavg window, initial temperature includes applicable calorimetric uncertainties and bias.
2. Steam generator performance data used in the analysis is conservatively high for steam temperature and pressure.

Table 6.6.1-2 Part-Power Initial-Condition Plant Parameters for Steam Generator Replacement[1] MSLB M&E Releases Inside Containment

Initial Conditions	Power Level (%)		
Parameter	**70**	**30**	**2**
RCS Average Temperature (°F)	583.2	570.7	558.2
RCS Flow Rate (gpm) (Thermal Design Flow)	374,400	374,400	374,400
RCS Pressure (psia)	2,250	2,250	2,250
Pressurizer Water Volume (% span)	46.3	31.6	26.0
Feedwater Enthalpy (Btu/lbm)	381.37	307.55	70.72
SG Pressure (psia)[2]	1,087	1,111	1,097
SG Water Level (% NRS)	>57.5 (58.4)	>57.5 (58.4)	>57.5 (58.4)

Notes:

1. Noted values correspond to plant conditions defined by 0% steam generator tube plugging and the high end of the RCS Tavg window, temperatures includes applicable calorimetric uncertainties and/or bias.
2. The noted SG pressures are determined at the steady-state conditions defined by the initial power level, and the RCS average temperatures, including applicable uncertainties and/or bias.

Table 6.6.1-3 Main and Auxiliary Feedwater System Assumptions for Steam Generator Replacement MSLB M&E Releases Inside Containment

Parameter	Analysis Assumption
Main Feedwater System	
Flow Rate (Until Main Feedwater Isolation)	Feedwater flow matches or exceeds steam flow.
Unisolable Volume from SG Nozzle to MFIV (Faulted Loop), ft^3	130
Unisolable Volume from SG Nozzle to FRV Assuming a Single Failure of the MFIV (Faulted Loop), ft^3	152.32
Auxiliary Feedwater System	
Flow Rate to all Steam Generators	Maximum flow to the SG in the faulted loop; minimum flow to each of the other 3 SGs. The actual data used is a function of SG pressure.
Temperature (Maximum Value), °F	120
Piping Purge Volume, ft^3	31.5
Actuation Delay Time, seconds	0

Table 6.6.1-4 Protection System Actuation Signals and Safety System Setpoints for Steam Generator Replacement – MSLB M&E Releases Inside Containment

Actuation Signal	Safety Analysis Setpoint
Reactor Trip	
2/4 Low Pressurizer Pressure, psia	1,860
Safety Injection	Yes
Safety Injection	
2/4 Low Pressurizer Pressure, psia	1,715
2/3 Low Steam Line Pressure in any loop, psia	559[1,2]
Dynamic Compensation Lead, seconds	50
Dynamic Compensation Lag, seconds	5
2/3 High-1 Containment Pressure	implicit
Steam Line Isolation	
2/3 Low Steam Line Pressure in any Loop, psia	559[1,2]
Dynamic Compensation Lead, seconds	50
Dynamic Compensation Lag, seconds	5
2/3 High-2 Containment Pressure	implicit
Feedwater Isolation and Auxiliary Feedwater Initiation	
Safety Injection	Yes

1. This value is different with respect to prior analysis.
2. For the MSLB inside containment, adverse environmental allowances need not be considered since the transmitters are outside containment. Per the information associated with the steam pressure uncertainty calculation, the safety analysis limit is 544 psig for the MSLB inside containment.

Table 6.6.1-5 Transient Summary for the Spectrum of Steam Line Breaks Inside Containment

Initial Power, Single Failure	Break Type	Reactor Trip Signal	Rod Motion (sec)	AFW Initiation/ Termination (sec)	Main Feedwater Isolation, Faulted SG (sec)	Steam Line Isolation (sec)	Faulted SG Dryout (sec)
102%, none	DER	SI-LSP	2.003	0.003 / 600.0	17.003	17.003	647
102%, MSIV	DER	SI-LSP	2.003	0.003 / 600.0	17.003	17.003	639
102%, MFIV	DER	SI-LSP	2.003	0.003 / 600.0	17.003	17.003	663
102%, none	split	High-1	19.0	17.0 / 600.0	34.0	80.0	700
102%, MSIV	split	High-1	19.0	17.0 / 600.0	34.0	80.0	700
102%, MFIV	split	High-1	19.0	17.0 / 600.0	34.0	80.0	700
70%, none	DER	SI-LSP	2.017	0.017 / 600.0	17.017	17.017	694
70%, MSIV	DER	SI-LSP	2.017	0.017 / 600.0	17.017	17.017	647
70%, MFIV	DER	SI-LSP	2.017	0.017 / 600.0	17.017	17.017	655
70%, none	split	High-1	17.0	15.0 / 600.0	32.0	75.0	700
70%, MSIV	split	High-1	17.0	15.0 / 600.0	32.0	75.0	700
70%, MFIV	split	High-1	17.0	15.0 / 600.0	32.0	75.0	700
30%, none	DER	SI-LSP	2.021	0.021 / 600.0	17.021	17.021	663
30%, MSIV	DER	SI-LSP	2.021	0.021 / 600.0	17.021	17.021	655
30%, MFIV	DER	SI-LSP	2.021	0.021 / 600.0	17.021	17.021	647
30%, none	split	High-1	16.0	14.0 / 600.0	31.0	74.0	800
30%, MSIV	split	High-1	16.0	14.0 / 600.0	31.0	74.0	800
30%, MFIV	split	High-1	16.0	14.0 / 600.0	31.0	74.0	795
2%, none	DER	SI-LSP	2.010	0.010 / 600.0	17.010	17.010	647
2%, MSIV	DER	SI-LSP	2.010	0.010 / 600.0	17.010	17.010	655
2%, MFIV	DER	SI-LSP	2.010	0.010 / 600.0	17.010	17.010	678
2%, none	split	High-1	18.0	0.0 / 600.0	33.0	88.0	692
2%, MSIV	split	High-1	18.0	0.0 / 600.0	33.0	88.0	700
2%, MFIV	split	High-1	18.0	0.0 / 600.0	33.0	88.0	692

Key SI – safety injection
LSP – low steam pressure
High-1 – containment high pressure

6.6.2 MSLB Mass and Energy Releases Outside Containment

6.6.2.1 Introduction

Steam line ruptures occurring outside the reactor containment structure may result in significant releases of high-energy fluid to the structures surrounding the steam systems. Superheated steam blowdowns following the steam line break have the potential to raise compartment temperatures outside containment. Early uncovery of the steam generator tube bundle maximizes the enthalpy of the superheated steam releases out of the break. The impact of the steam releases depends on the plant configuration at the time of the break, the plant response to the break, as well as the size and location of the break. Because of the interrelationship among many of the factors that influence steam line break M&E releases, an appropriate determination of a single limiting case with respect to M&E releases cannot be made. Therefore, it is necessary to analyze the steam line break event outside containment for a range of conditions.

6.6.2.2 Input Parameters and Assumptions

The analysis inputs, assumptions, and methods pertaining to the MSLB M&E releases outside containment are presented in this section.

To determine the effects of plant power level and break area on the M&E releases from a ruptured steam line, spectra of both variables have been evaluated as part of the methodology development program documented in Reference 1. The Callaway Plant had been included as part of the Category 1 plants in the analysis presented in Reference 1. At plant power levels of 102 percent and 70 percent, various break sizes were defined ranging from 0.1 ft^2 to the equivalent of a full single-ended rupture (4.6 ft^2) of a main steam line.

Consistent with the current licensing-basis analysis, assumptions are made that minimize the time to achieve steam generator tube uncovery, which maximizes the superheated release duration. A full break spectrum initiated from 102 percent of full power has been analyzed at the conditions associated with the Framatome-design Model 73/19T RSGs. The emphasis on maximizing the value of the superheated steam enthalpy due to early steam generator tube uncovery is consistent with the methodology development program documented in Reference 1. Other assumptions regarding important plant conditions and features are discussed in the following paragraphs.

6.6.2.2.1 Initial Power Level

The initial power that is assumed for steam line break analyses outside containment affects the M&E releases and steam generator tube bundle uncovery in two ways. First, the steam generator water mass inventory increases with decreasing power levels. This will tend to delay uncovery of the steam generator tube bundle, although the increased steam pressure associated with lower power levels will cause a faster blowdown at the beginning of the transient. Second, the amount of stored energy and decay heat, as well as feedwater temperature, are less for lower power levels. This will result in lower primary temperatures and less primary-to-secondary heat transfer during the steam line break event.

Overall, steam line breaks initiated from lower power levels result in lower levels of steam superheating than breaks analyzed at full-power conditions. For this reason, the licensing basis of the steam line break outside containment M&E release calculations for the Callaway Plant reflects only a full-power initial condition. The initial power is the maximum allowable NSSS power plus uncertainty, i.e., 102 percent of rated power. For this steam generator replacement analysis, the power level and steam line break sizes are noted in subsection 6.6.2.3 of this report.

In general, the plant initial conditions are assumed to be at the nominal value, with appropriate uncertainties included. Table 6.6.2-1 identifies the values assumed for NSSS power, RCS pressure, RCS vessel average temperature, RCS flow, pressurizer water volume, steam generator water level, steam generator temperature and pressure, and feedwater enthalpy. Steam line break mass releases and superheated steam enthalpies assuming an RCS average temperature at the high end of the Tavg window are conservative with respect to similar releases at the low end of the Tavg window. At the high end, there is a larger value for the superheated steam enthalpy available for release outside containment. The thermal design flow rate has been used for the RCS flow input consistent with the assumptions documented in Reference 2. The thermal design flow rate is also consistent with other MSLB analysis assumptions related to nonstatistical treatment of uncertainties, as well as RCS thermal-hydraulic inputs related to pressure drops and rod drop time.

Uncertainties on the initial conditions assumed in the analysis for the RSG Program have been applied only to the RCS average temperature (4.3°F), the steam generator mass (8-percent narrow-range span), and the power fraction (2 percent). For the control function water level uncertainty, the value is +7.9 percent, indicating that the channel indicates higher than actual. For the MSLB outside containment, the analysis uncertainty is applied in the negative direction for conservatism. The 7.9 percent has been rounded to 8 percent for this analysis. Nominal values are adequate for the initial conditions associated with pressurizer pressure and pressurizer water level. Uncertainty conditions are only applied to those parameters that could increase the enthalpy of superheated steam discharged out of the break.

6.6.2.2.2 Single-Failure Assumption

The main steam lines outside containment up to the isolation valves conform to Branch Technical Position MEB 3-1. Therefore, a rupture in these pipes is not postulated. The main steam lines comprise part of the "no break zone" and a break postulated here is assumed to be the single failure. Therefore, no additional single-failure assumption has been made in the plant-specific analysis of the steam line break outside containment for the Callaway Plant.

6.6.2.2.3 Main Feedwater System

The rapid depressurization that typically occurs following a steam line rupture results in large amounts of water being added to the steam generators through the main feedwater system. To maximize the superheated steam releases, main feedwater flow is conservatively modeled by assuming no increase in feedwater flow in response to the increases in steam flow following the steam line break event. This minimizes the total mass addition and associated cooling effects in the steam generators and causes the earliest onset of superheated steam released out of the break.

Isolation of the main feedwater flow is assumed to be initiated following either a safety injection signal or a low RCS Tavg signal following (coincident with) a reactor trip. The feedwater isolation response time is assumed to be 2 seconds, accounting only for the delay associated with signal processing. Closing of the feedwater valves in the main feedwater lines is conservatively assumed to be instantaneous with respect to valve stroke time.

Steam line break mass and energy releases, assuming a main feedwater temperature at the high end of the feedwater temperature window, are conservative with respect to similar releases at the low end of the feedwater temperature window. At the high end, there is more energy available for release outside containment.

6.6.2.2.4 Auxiliary Feedwater System

Generally, within the first few minutes following a steam line break, the AFW system is initiated on any one of several protection system signals. Addition of AFW to the steam generators will increase the secondary mass available to cover the tube bundle and reduces the amount of superheated steam produced. For this reason, AFW flow is minimized while actuation delays are maximized to accentuate the depletion of the initial secondary-side inventory. The volume of the AFW piping is maximized. A purging of the AFW piping is assumed, since maximum volume delays the injection of colder AFW into the steam generator, following any hotter feedwater resident in the piping up to the isolation valve closest to the steam generator. The less dense resident AFW exhibits a decreased mass addition to the faulted-loop steam generator than if the AFW is introduced directly into the steam generator. The large volume also delays the introduction of colder AFW into any steam generator, which reduces the amount of the cooldown effect on the primary side of the RCS. Auxiliary feedwater system assumptions that have been used in the analysis are presented in Table 6.6.2-2.

6.6.2.2.5 Steam Generator Fluid Mass

A minimum initial steam generator mass in all the steam generators has been used in all the steam line break cases. The use of a reduced initial steam generator mass minimizes the availability of the heat sink afforded by the steam generators and leads to earlier tube bundle uncovery. The initial mass has been calculated as the value corresponding to the programmed level minus 8-percent narrow-range span. For the control function water level uncertainty, the value is +7.9 percent, indicating that the channel indicates higher than actual. For the MSLB outside containment, the analysis uncertainty is applied in the negative direction for conservatism. The 7.9 percent has been rounded to 8 percent for this analysis. All steam generator fluid masses are calculated assuming 0-percent tube plugging. This assumption is conservative with respect to the RCS cooldown through the steam generators resulting from the steam line break.

6.6.2.2.6 Steam Generator Reverse Heat Transfer

Once the steam line isolation is complete, the steam generators in the intact loops become sources of energy that can be transferred to the steam generator with the broken steam line. This energy transfer occurs via the primary coolant. As the primary plant cools, the temperature of the coolant flowing in the steam generator tubes could drop below the temperature of the secondary fluid in the intact steam generators, resulting in energy being returned to the primary coolant. This energy is then available to be transferred to the steam generator with the broken steam line. When applicable, the effects of reverse steam generator heat transfer are included in the results.

6.6.2.2.7 Break Flow Model

Piping discharge resistances are not included in the calculation of the releases resulting from the steam line ruptures (Moody Curve for an $f(\ell / D) = 0$ is used). This maximizes the break flow rate and increases the energy release into the compartment, resulting in a maximum temperature for the assumed break area.

6.6.2.2.8 Steam Line Volume Blowdown

There is no contribution to the M&E releases from the steam in the secondary plant main steam loop piping and header because the initial volume is saturated steam. With the focus of the MSLB analysis outside containment on maximizing the superheated steam enthalpy, it is presumed that the saturated steam in the loop piping and the header has no adverse effects on the results. The blowdown of the steam in this volume serves to delay the time of tube uncovery in the steam generators and is conservatively ignored.

6.6.2.2.9 Main Steam Line Isolation

Steam line isolation is assumed for all steam line break cases since the break location is upstream of the main steam line isolation valves. A conservative delay time of 17 seconds, accounting for delays associated with signal processing plus MSIV stroke time (with unrestricted steam flow through the valve during the valve stroke), has been assumed.

6.6.2.2.10 Protection System Actuations

The protection systems available to mitigate the effects of a MSLB outside containment include reactor trip, SI, and AFW. The protection system actuation signals and associated setpoints that have been modeled in the analysis are identified in Table 6.6.2-3. The setpoints used are conservative values with respect to the plant-specific values delineated in the Technical Specifications for the steam generator replacement. The specific functions credited in the Callaway Plant-specific analysis are documented in subsection 6.6.2.5.

6.6.2.2.11 Safety Injection System

Minimum SIS flow rates corresponding to the failure of one SIS train have been assumed in this analysis. A minimum SI flow is conservative since the reduced boron addition maximizes a return to power resulting from the RCS cooldown. The higher power generation increases heat transfer to the secondary side, maximizing steam flow out of the break. The delay time to achieve full SI flow is assumed to be 27 seconds for this analysis with offsite power available. A coincident loss of offsite power is not assumed for the analysis of the steam line break outside containment since the M&E releases would be reduced due to the loss of forced reactor coolant flow, resulting in less primary-to-secondary heat transfer.

6.6.2.2.12 Reactor Coolant System Metal Heat Capacity

As the primary side of the plant cools, the temperature of the reactor coolant drops below the temperature of the reactor coolant piping, the reactor vessel, the reactor coolant pumps, and the steam generator thick-metal mass and tubing. As this occurs, the heat stored in the metal is available to be transferred to the steam generator with the broken line. The effects of this RCS metal heat are included in the results using conservative thick-metal masses and heat transfer coefficients.

6.6.2.2.13 Core Decay Heat

Core decay heat generation assumed in calculating the steam line break mass and energy releases is based on the 1979 ANS Decay Heat + 2σ model (Reference 3).

6.6.2.2.14 Rod Control

The rod control system is conservatively assumed to be in manual operation for all steam line break analyses. Assuming that the reactor is in manual rod control allows for a greater RCS cooldown prior to the reactor trip signal, which maximizes the reactivity feedback at end-of-cycle conditions and produces a greater post-trip power increase.

6.6.2.2.15 Core Reactivity Coefficients

Conservative core reactivity coefficients corresponding to end-of-cycle conditions are used to maximize the reactivity feedback effects resulting from the steam line break. Use of maximum reactivity feedback results in higher power generation if the reactor returns to criticality, thus maximizing heat transfer to the secondary side of the steam generators.

6.6.2.3 Description of Analyses and Evaluations

The system transient that provides the break flows and enthalpies of the steam release through the steam line break outside containment has been analyzed with the LOFTRAN (Reference 4) computer code. Blowdown M&E releases determined using LOFTRAN include the effects of core power generation, main and auxiliary feedwater additions, engineered safeguards systems, RCS thick-metal heat storage including steam generator thick-metal mass and tubing, and reverse steam generator heat transfer. The use of the LOFTRAN code for the analysis of the MSLB with superheated steam mass and energy releases is documented in Supplement 1 of WCAP-8822 (Reference 2), which has been reviewed and approved by the NRC for use in analyzing main steam line breaks, and in Reference 1 for MSLBs outside containment.

The Callaway Plant NSSS has been analyzed to determine the transient mass releases and associated superheated steam enthalpy values outside containment following a steam line break event. The tables of mass flow rates and steam enthalpies are used as input conditions to the environmental evaluation of safety-related electrical equipment in the main steam / main feedwater isolation valve compartment located in the northeast part of the Auxiliary Building.

The following licensing-basis cases of the MSLB outside containment have been analyzed at the noted conditions for the RSG Program:

- At 102-percent power, break sizes of 4.6, 2.0, 1.4, 1.2, 1.0, 0.9, 0.8, 0.7, 0.6, 0.5, 0.4, 0.3, 0.2, 0.1, and 0.05 ft^2.

Each MSLB outside containment is represented as a nonmechanistic split rupture (crack area). The largest break is postulated as a crack area equivalent to a single-ended pipe rupture, defined by the actual cross-sectional flow area of the steam line outside containment, 4.6 ft^2. Prior to steam line isolation, the steam flow is supplied by all 4 steam generators through the break area represented by the spectrum noted above. After steam line isolation, the steam release through the break is supplied by a single steam generator and is limited by the smaller of the integral steam generator flow restrictor (1.4 ft^2) or the defined break area.

6.6.2.4 Acceptance Criteria

The main steam line break is classified as an ANS Condition IV event, an infrequent fault. The acceptance criteria associated with the steam line break event resulting in an M&E release outside containment are based on an analysis that provides sufficient conservatism that the equipment qualification temperature envelope is maintained. The specific criteria applicable to this analysis are related to the assumptions regarding power level, stored energy, the break flow model, steam line and feedwater isolation, and main and AFW flow such that superheated steam resulting from tube bundle uncovery in the steam generators is accounted for and maximized. These analysis assumptions have been included in this steam line break M&E release analysis as discussed in subsection 6.6.2.2 of this report. The tables of mass flow rates and steam enthalpy values for each of the steam line break cases noted in the previous section are used as input to the environmental evaluation of safety-related electrical equipment in the main steam / main feedwater isolation valve compartment in the Auxiliary Building.

6.6.2.5 Results

Using the MSLB analysis methodology documented in Reference 1 as a basis, including parameter changes associated with the RSG Program, the mass and energy release rates for each of the steam line break cases noted in subsection 6.6.2.3 have been developed for use in the environmental evaluation of safety-related electrical equipment in the main steam / main feedwater isolation valve compartment for the Callaway Plant. Table 6.6.2-4 provides the sequence of events for each of the 15 steam line break sizes initiated from a full-power condition, for the Model 73/19T Framatome RSGs.

For the largest of the assumed break sizes from 4.6 ft^2 to 1.4 ft^2, reactor trip is actuated following the SI signal produced by the low steam pressure setpoint (2-of-3 channels per loop). Main feedwater flow is isolated and steam line isolation both occur due to the SI signal caused by low steam pressure. The AFW initiation also occurs as a result of the low steam pressure SI signal. The AFW flow continues until 1,800 seconds, at which time it is isolated to the faulted-loop steam generator. Dryout of the steam generator occurs shortly after the AFW flow is discontinued.

For the intermediate-sized breaks from 1.2 ft^2 to 0.4 ft^2, reactor trip is actuated following the overpower ΔT (OPΔT) (2-of-4 channels) signal. Safety injection is started as a result of a low pressurizer pressure (2-of-4 channels) signal. Steam line isolation occurs later due to receipt of the low steam pressure signal. Main feedwater flow is assumed to be isolated following reactor trip upon receipt of a low RCS Tavg signal. The AFW initiation occurs as a result of the SI signal. For these break sizes, the AFW flow also continues until 1,800 seconds, at which time it is isolated to the faulted-loop steam generator. Dryout of the steam generator occurs shortly after the AFW flow is discontinued.

For the smallest break sizes from 0.3 ft^2 to 0.05 ft^2, the first protection signal received is the low-low steam generator water level (2-of-4 channels per loop) signal, which actuates reactor trip and the AFW system. Main feedwater flow is assumed to be isolated following reactor trip upon receipt of a low RCS Tavg signal. Safety injection occurs following a low pressurizer pressure signal. The AFW flow continues until 1,800 seconds, at which time it is isolated to the faulted-loop steam generator. Dryout of the steam generator occurs later in time after the AFW flow is discontinued. An automatic steam line isolation signal is not received since the steam generator depressurization does not actuate the low steam line pressure setpoint within 1,800 seconds. Operator action at 30 minutes following event initiation is assumed to isolate the main steam lines for each of these smallest break sizes. For the smallest break included in the spectrum (0.05 ft^2), steam line isolation is not assumed since the break location is postulated upstream of the MSIVs in a cross-connect pipe supplying the AFW system. The steam generators blow down slowly after AFW isolation.

Steam line break mass flow rate and steam enthalpy data, accounting for early steam generator tube uncovery and superheated steam releases from a Model 73/19T steam generator, are presented in Tables 6.6.2-5 through 6.6.2-19.

6.6.2.6 Conclusions

The mass releases and associated steam enthalpy values from the spectrum of steam line break cases outside containment have been analyzed at the conditions defined by the RSG Program. The assumptions delineated in subsection 6.6.2.2 have been included in the steam line break analysis such that conservative M&E releases are calculated. The mass releases and associated steam enthalpy values discussed in this section have been provided for use in the environmental evaluation of safety-related electrical equipment in the main steam / main feedwater isolation valve compartment of the Auxiliary Building for the Callaway Plant.

6.6.2.7 References

1. WCAP-10961, Rev. 1, (Proprietary), "Steamline Break Mass/Energy Releases for Equipment Environmental Qualification Outside Containment, Report to the Westinghouse Owners Group High Energy Line Break/Superheated Blowdowns Outside Containment Subgroup," October 1985.

2. WCAP-8822 (Proprietary) and WCAP-8860 (Non-Proprietary), "Mass and Energy Releases Following a Steam Line Rupture," September 1976; WCAP-8822-S1-P-A (Proprietary) and WCAP-8860-S1-A (Non-Proprietary), "Supplement 1 – Calculations of Steam Superheat in Mass/Energy Releases Following a Steam Line Rupture," September 1986; WCAP-8822-S2-P-A (Proprietary) and WCAP-8860-S2-A (Non-Proprietary), "Supplement 2 – Impact of Steam Superheat in Mass/Energy Releases Following a Steam Line Rupture for Dry and Subatmospheric Containment Designs," September 1986.

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

4. WCAP-7907-P-A (Proprietary) and WCAP-7907-A (Non-Proprietary), "LOFTRAN Code Description," April 1984.

Table 6.6.2-1 Nominal and Initial Plant Parameters for Steam Generator Replacement[1] MSLB M&E Releases Outside Containment

Plant Parameter	Nominal	Initial
NSSS Power, MWt	3579	3650.6
Core Power, MWt	3565	not an input
Net RCS Heat Input, MWt	14	20
Reactor Coolant Flow (total), gpm (Thermal Design Flow)	374,400	374,400
Pressurizer Pressure, psia	2250	2250
Reactor Coolant Vessel Average Temperature, °F	588.4	592.7
Steam Generator[2]		
Steam Temperature, °F	547.2	548.6
Steam Pressure, psia	1022	1033
Feedwater Temperature, °F	446	446
Water Level, % Narrow-Range Span	51.35	43.35
Zero-Load Temperature, °F	557	557

Notes:

1. Noted values correspond to plant conditions defined by 0% steam generator tube plugging and the high end of the RCS Tavg window.
2. Steam generator performance data used in the analysis is conservatively high for steam temperature and pressure.

Table 6.6.2-2 Main and Auxiliary Feedwater System Assumptions for Steam Generator Replacement MSLB M&E Releases Outside Containment

Parameter	Analysis Assumption
Main Feedwater System	
Flow Rate	Nominal flow to all loops
Unisolable volume from SG Nozzle to Feedwater Isolation Valve (FWIV)	None assumed
Auxiliary Feedwater System	
Flow Rate to all Steam Generators	Minimum flow to each SG. The actual data used is a function of SG pressure.
Temperature (maximum value), °F	120
Piping Purge Volume (faulted loop), ft^3	125
Actuation Delay Time, seconds	60

Table 6.6.2-3 Protection System Actuation Signals and Safety System Setpoints for Steam Generator Replacement MSLB M&E Releases Outside Containment	
Actuation Signal	**Safety Analysis Setpoint**
Reactor Trip	
2/4 Low-Low Steam Generator Water Level in Any Loop, % Narrow-Range Span	0
2/4 Low Pressurizer Pressure, psia	1,860
2/4 Power-Range High Neutron Flux, % Rated Thermal Power	118
2/4 Overtemperature ΔT	
K1	1.290
K2	0.0251
K3	0.00116
Dynamic compensation lead, seconds	28
Dynamic compensation lag, seconds	4
2/4 Overpower ΔT	
K4	1.165
K5	0.0
K6	0.0015
Dynamic compensation rate lag, seconds	10
Safety Injection	Yes
Safety Injection	
2/4 Low Pressurizer Pressure, psia	1,715
2/3 Low Steam Line Pressure in Any Loop, psia	472.7[1,2]
Dynamic compensation lead, seconds	50
Dynamic compensation lag, seconds	5
Steam Line Isolation	
2/3 Low Steam Line Pressure in Any Loop, psia	472.7[1,2]
Dynamic compensation lead, seconds	50
Dynamic compensation lag, seconds	5
Feedwater Isolation	
Low RCS Average Temperature Following a Reactor Trip, °F	568.3
Safety Injection	Yes
Auxiliary Feedwater Initiation	
2/4 Low-Low Steam Generator Water Level in Multiple Loops % Narrow-Range Span	0
Safety Injection	Yes

Notes:

1. This value is different with respect to the prior analysis.
2. For the MSLB outside containment, adverse environmental allowances are considered since the setpoint transmitters are assumed to be in the location of the break. Per the information associated with the steam pressure uncertainty calculation, the safety analysis limit is 458 psig for the MSLB outside containment.

Table 6.6.2-4 Transient Summary for the Spectrum of Full-Power Steam Line Breaks Outside Containment

Case	Break Size (ft^2)	Reactor Trip Signal	Rod Motion (sec)	Safety Injection Signal	Safety Injection (sec)	Feedwater Isolation (sec)	Steam Line Isolation (sec)	Auxiliary Feedwater (sec)	SG Tube Uncovery (sec)
1	4.6	SI-LSP	3.1	LSP	28.2	3.1	18.1	61.1	54.0
2	2.0	SI-LSP	4.5	LSP	29.6	4.5	19.5	62.5	61.2
3	1.4	SI-LSP	6.6	LSP	31.7	6.6	21.6	64.6	64.2
4	1.2	OPΔT	13.7	LPP	66.6	28.2(1)	249.0	99.5	192.6
5	1.0	OPΔT	14.9	LPP	74.7	30.9(1)	315.5	107.6	225.6
6	0.9	OPΔT	15.7	LPP	80.3	32.8(1)	353.7	113.2	237.8
7	0.8	OPΔT	16.8	LPP	87.9	35.3(1)	408.0	120.8	254.2
8	0.7	OPΔT	18.3	LPP	97.8	38.1(1)	478.4	130.7	275.0
9	0.6	OPΔT	20.6	LPP	112.2	42.3(1)	583.4	145.1	305.2
10	0.5	OPΔT	24.7	LPP	134.0	48.7(1)	746.8	166.9	349.2
11	0.4	OPΔT	42.9	LPP	181.4	70.6(1)	1034.	214.3	472.8
12	0.3	LSGWL	107.6	LPP	294.6	141.5(1)	1800. (M)	165.6	819.4
13	0.2	LSGWL	156.7	LPP	458.1	199.2(1)	1800. (M)	214.7	1970.
14	0.1	LSGWL	299.8	LPP	956.7	357.4(1)	1800. (M)	357.8	2576.
15	0.05	LSGWL	570.6	LPP	6917.	640.0(1)	none	628.6	6514.

Note:
1. Main feedwater isolation function generated by an RCS low Tavg signal following reactor trip

Key: SI – safety injection
LSP – low steam pressure
LPP – low pressurizer pressure
OPΔT – overpower ΔT
LSGWL – low-low steam generator water level
M – manual

Table 6.6.2-5 Steam Line Break M&E Releases Outside Containment 102% Power, 4.6 ft^2 Break		
Time (sec)	**Mass Flow (lbm/sec)**	**Enthalpy (Btu/lbm)**
.00	.0000E+00	.0000E+00
.20	.8687E+04	.1190E+04
2.00	.7808E+04	.1193E+04
3.00	.7467E+04	.1195E+04
3.20	.7951E+04	.1195E+04
7.80	.7545E+04	.1197E+04
11.60	.6985E+04	.1199E+04
18.20	.5696E+04	.1202E+04
18.40	.1914E+04	.1202E+04
22.80	.1613E+04	.1204E+04
26.60	.1422E+04	.1204E+04
30.40	.1287E+04	.1204E+04
36.20	.1168E+04	.1204E+04
42.00	.1104E+04	.1204E+04
53.80	.1051E+04	.1204E+04
59.20	.9787E+03	.1227E+04
62.00	.9128E+03	.1237E+04
64.60	.8305E+03	.1247E+04
67.40	.7131E+03	.1258E+04
70.00	.5760E+03	.1268E+04
72.60	.4273E+03	.1278E+04
74.00	.3230E+03	.1283E+04
75.40	.2472E+03	.1287E+04
76.20	.2132E+03	.1288E+04
76.80	.1922E+03	.1290E+04
78.20	.1545E+03	.1292E+04
79.00	.1366E+03	.1293E+04
80.80	.1090E+03	.1296E+04
82.20	.9384E+02	.1297E+04

Table 6.6.2-5 Steam Line Break M&E Releases Outside Containment 102% Power, 4.6 ft^2 Break (cont.)		
Time (sec)	Mass Flow (lbm/sec)	Enthalpy (Btu/lbm)
83.40	.8517E+02	.1299E+04
84.80	.7761E+02	.1300E+04
86.20	.7168E+02	.1301E+04
91.40	.5736E+02	.1304E+04
94.20	.5257E+02	.1306E+04
96.80	.5125E+02	.1306E+04
108.80	.5803E+02	.1308E+04
112.80	.5852E+02	.1309E+04
129.60	.5734E+02	.1311E+04
161.00	.5769E+02	.1312E+04
210.60	.6878E+02	.1312E+04
400.00	.6853E+02	.1313E+04
600.00	.6853E+02	.1311E+04
800.00	.6853E+02	.1309E+04
1000.00	.6853E+02	.1307E+04
1200.00	.6852E+02	.1304E+04
1400.00	.6852E+02	.1301E+04
1600.00	.6852E+02	.1298E+04
1800.40	.6865E+02	.1295E+04
1801.60	.6693E+02	.1296E+04
1802.20	.6496E+02	.1296E+04
1802.40	.5994E+02	.1297E+04
1802.80	.5232E+02	.1297E+04
1803.40	.3889E+02	.1297E+04
1803.60	.3362E+02	.1297E+04
1803.80	.2742E+02	.1298E+04
1804.20	.1693E+02	.1298E+04
1804.40	.0000E+00	.0000E+00
1900.00	.0000E+00	.0000E+00

Table 6.6.2-6 Steam Line Break M&E Releases Outside Containment 102% Power, 2.0 ft^2 Break		
Time (sec)	Mass Flow (lbm/sec)	Enthalpy (Btu/lbm)
.00	.0000E+00	.0000E+00
.20	.4109E+04	.1189E+04
2.20	.3835E+04	.1192E+04
4.40	.3617E+04	.1194E+04
4.60	.3846E+04	.1194E+04
7.60	.4057E+04	.1192E+04
10.80	.4157E+04	.1191E+04
14.20	.4032E+04	.1192E+04
19.60	.3674E+04	.1195E+04
19.80	.2617E+04	.1196E+04
21.20	.2347E+04	.1198E+04
23.00	.2112E+04	.1200E+04
24.80	.1934E+04	.1202E+04
28.20	.1694E+04	.1203E+04
31.80	.1515E+04	.1204E+04
35.20	.1388E+04	.1204E+04
38.80	.1287E+04	.1204E+04
44.40	.1177E+04	.1204E+04
50.00	.1107E+04	.1204E+04
61.00	.1038E+04	.1204E+04
66.40	.9674E+03	.1226E+04
69.20	.9081E+03	.1235E+04
71.80	.8326E+03	.1244E+04
74.60	.7237E+03	.1255E+04
77.40	.5833E+03	.1266E+04
80.60	.4036E+03	.1279E+04
82.20	.2899E+03	.1284E+04
83.40	.2308E+03	.1287E+04
84.20	.2000E+03	.1289E+04
85.40	.1652E+03	.1291E+04
86.80	.1333E+03	.1293E+04
88.20	.1122E+03	.1295E+04

Table 6.6.2-6 Steam Line Break M&E Releases Outside Containment 102% Power, 2.0 ft^2 Break (cont.)		
Time (sec)	Mass Flow (lbm/sec)	Enthalpy (Btu/lbm)
89.40	.9789E+02	.1297E+04
90.80	.8669E+02	.1298E+04
92.20	.7860E+02	.1300E+04
93.60	.7253E+02	.1301E+04
99.00	.5751E+02	.1304E+04
101.60	.5286E+02	.1306E+04
104.40	.5125E+02	.1307E+04
116.40	.5804E+02	.1309E+04
120.40	.5852E+02	.1310E+04
137.20	.5735E+02	.1312E+04
164.20	.5789E+02	.1313E+04
208.60	.6850E+02	.1313E+04
400.00	.6853E+02	.1313E+04
600.00	.6853E+02	.1312E+04
800.00	.6853E+02	.1310E+04
1000.00	.6853E+02	.1308E+04
1200.00	.6852E+02	.1306E+04
1400.00	.6852E+02	.1303E+04
1600.00	.6852E+02	.1300E+04
1801.00	.6817E+02	.1297E+04
1802.00	.6567E+02	.1298E+04
1802.20	.6426E+02	.1298E+04
1802.40	.5945E+02	.1298E+04
1803.00	.4755E+02	.1299E+04
1803.40	.3820E+02	.1299E+04
1803.60	.3285E+02	.1299E+04
1803.80	.2641E+02	.1299E+04
1804.20	.1625E+02	.1300E+04
1804.40	.0000E+00	.0000E+00
1900.00	.0000E+00	.0000E+00

Table 6.6.2-7 Steam Line Break M&E Releases Outside Containment 102% Power, 1.4 ft^2 Break		
Time (sec)	Mass Flow (lbm/sec)	Enthalpy (Btu/lbm)
.00	.0000E+00	.0000E+00
.20	.2933E+04	.1189E+04
3.20	.2737E+04	.1192E+04
6.40	.2592E+04	.1194E+04
6.60	.2730E+04	.1193E+04
10.20	.2986E+04	.1190E+04
13.80	.3103E+04	.1189E+04
16.40	.3052E+04	.1189E+04
21.60	.2852E+04	.1192E+04
25.00	.2260E+04	.1199E+04
26.80	.2055E+04	.1201E+04
28.60	.1898E+04	.1202E+04
32.20	.1673E+04	.1204E+04
36.00	.1502E+04	.1204E+04
39.60	.1380E+04	.1204E+04
43.20	.1286E+04	.1204E+04
48.40	.1189E+04	.1204E+04
53.60	.1122E+04	.1204E+04
64.00	.1047E+04	.1204E+04
69.40	.9704E+03	.1226E+04
72.20	.9095E+03	.1235E+04
74.80	.8336E+03	.1244E+04
77.60	.7253E+03	.1255E+04
80.40	.5861E+03	.1266E+04
83.80	.3921E+03	.1279E+04
85.20	.2944E+03	.1284E+04
86.40	.2344E+03	.1287E+04
87.20	.2029E+03	.1288E+04
88.40	.1672E+03	.1290E+04
89.80	.1350E+03	.1293E+04

Table 6.6.2-7 Steam Line Break M&E Releases Outside Containment 102% Power, 1.4 ft^2 Break (cont.)		
Time (sec)	**Mass Flow (lbm/sec)**	**Enthalpy (Btu/lbm)**
91.20	.1134E+03	.1295E+04
92.40	.9872E+02	.1296E+04
93.80	.8724E+02	.1298E+04
95.20	.7901E+02	.1300E+04
96.60	.7284E+02	.1301E+04
102.00	.5767E+02	.1304E+04
104.60	.5295E+02	.1306E+04
107.40	.5126E+02	.1307E+04
119.40	.5802E+02	.1309E+04
123.40	.5852E+02	.1310E+04
140.20	.5735E+02	.1312E+04
166.60	.5791E+02	.1313E+04
215.20	.6879E+02	.1313E+04
400.00	.6853E+02	.1314E+04
600.00	.6853E+02	.1313E+04
800.00	.6853E+02	.1311E+04
1000.00	.6853E+02	.1309E+04
1200.00	.6852E+02	.1306E+04
1400.00	.6852E+02	.1304E+04
1600.00	.6852E+02	.1301E+04
1800.80	.6843E+02	.1298E+04
1802.00	.6566E+02	.1299E+04
1802.20	.6417E+02	.1299E+04
1802.40	.5938E+02	.1299E+04
1802.80	.5168E+02	.1299E+04
1803.40	.3810E+02	.1300E+04
1803.60	.3274E+02	.1300E+04
1803.80	.2626E+02	.1300E+04
1804.20	.1616E+02	.1300E+04
1804.40	.0000E+00	.0000E+00
1900.00	.0000E+00	.0000E+00

Table 6.6.2-8 Steam Line Break M&E Releases Outside Containment 102% Power, 1.2 ft^2 Break		
Time (sec)	Mass Flow (lbm/sec)	Enthalpy (Btu/lbm)
.00	.0000E+00	.0000E+00
.20	.2530E+04	.1189E+04
3.60	.2367E+04	.1192E+04
7.00	.2256E+04	.1193E+04
13.60	.2115E+04	.1195E+04
13.80	.2231E+04	.1195E+04
17.20	.2461E+04	.1192E+04
19.00	.2544E+04	.1191E+04
21.00	.2582E+04	.1190E+04
23.60	.2540E+04	.1191E+04
28.20	.2394E+04	.1193E+04
64.80	.1854E+04	.1200E+04
79.40	.1705E+04	.1201E+04
94.00	.1594E+04	.1202E+04
108.60	.1541E+04	.1203E+04
139.60	.1523E+04	.1203E+04
214.00	.1420E+04	.1206E+04
241.80	.1297E+04	.1219E+04
249.40	.1247E+04	.1227E+04
250.00	.1147E+04	.1232E+04
251.20	.1006E+04	.1239E+04
253.40	.8149E+03	.1248E+04
255.80	.6605E+03	.1256E+04
258.00	.5447E+03	.1263E+04
260.40	.4339E+03	.1269E+04
262.80	.3381E+03	.1275E+04
266.20	.2183E+03	.1282E+04
267.40	.1847E+03	.1283E+04
268.60	.1592E+03	.1285E+04
270.80	.1264E+03	.1287E+04
272.00	.1133E+03	.1288E+04

Table 6.6.2-8 Steam Line Break M&E Releases Outside Containment 102% Power, 1.2 ft^2 Break (cont.)		
Time (sec)	Mass Flow (lbm/sec)	Enthalpy (Btu/lbm)
274.40	.9487E+02	.1289E+04
276.80	.8030E+02	.1290E+04
279.00	.7146E+02	.1291E+04
281.20	.6582E+02	.1291E+04
283.60	.6159E+02	.1292E+04
286.00	.5959E+02	.1292E+04
288.20	.5907E+02	.1292E+04
294.80	.6191E+02	.1292E+04
301.20	.6638E+02	.1292E+04
307.80	.6899E+02	.1293E+04
314.40	.6973E+02	.1293E+04
430.40	.6852E+02	.1294E+04
600.00	.6852E+02	.1292E+04
800.00	.6851E+02	.1290E+04
1000.00	.6851E+02	.1287E+04
1200.00	.6851E+02	.1284E+04
1400.00	.6851E+02	.1281E+04
1600.00	.6851E+02	.1278E+04
1801.60	.6796E+02	.1278E+04
1803.80	.6298E+02	.1280E+04
1805.20	.4463E+02	.1280E+04
1805.60	.3832E+02	.1281E+04
1806.00	.3119E+02	.1281E+04
1806.20	.2712E+02	.1281E+04
1806.40	.2210E+02	.1281E+04
1806.80	.1427E+02	.1281E+04
1807.00	.0000E+00	.0000E+00
1900.00	.0000E+00	.0000E+00

Table 6.6.2-9 Steam Line Break M&E Releases Outside Containment 102% Power, 1.0 ft^2 Break		
Time (sec)	Mass Flow (lbm/sec)	Enthalpy (Btu/lbm)
.00	.0000E+00	.0000E+00
.20	.2121E+04	.1189E+04
3.20	.2017E+04	.1191E+04
7.60	.1914E+04	.1193E+04
14.80	.1810E+04	.1195E+04
15.00	.1899E+04	.1194E+04
18.60	.2114E+04	.1191E+04
20.40	.2186E+04	.1189E+04
22.40	.2219E+04	.1189E+04
30.80	.2050E+04	.1192E+04
72.80	.1611E+04	.1199E+04
89.60	.1481E+04	.1201E+04
106.60	.1384E+04	.1202E+04
123.40	.1326E+04	.1202E+04
254.60	.1256E+04	.1211E+04
278.40	.1216E+04	.1219E+04
302.20	.1130E+04	.1228E+04
315.80	.1041E+04	.1240E+04
317.00	.9035E+03	.1248E+04
318.40	.7815E+03	.1255E+04
320.00	.6653E+03	.1262E+04
321.40	.5787E+03	.1268E+04
323.00	.4934E+03	.1273E+04
327.40	.2919E+03	.1284E+04
328.80	.2423E+03	.1287E+04
330.20	.2054E+03	.1289E+04
331.80	.1720E+03	.1290E+04
333.20	.1497E+03	.1292E+04
334.80	.1298E+03	.1293E+04
336.20	.1158E+03	.1294E+04

Table 6.6.2-9 Steam Line Break M&E Releases Outside Containment 102% Power, 1.0 ft^2 Break (cont.)		
Time (sec)	Mass Flow (lbm/sec)	Enthalpy (Btu/lbm)
337.80	.1023E+03	.1295E+04
339.20	.9331E+02	.1295E+04
340.80	.8535E+02	.1296E+04
343.80	.7349E+02	.1296E+04
346.80	.6631E+02	.1297E+04
349.80	.6257E+02	.1297E+04
354.40	.6089E+02	.1297E+04
374.80	.6854E+02	.1295E+04
382.40	.6925E+02	.1295E+04
400.00	.6859E+02	.1295E+04
600.00	.6851E+02	.1293E+04
800.00	.6851E+02	.1290E+04
1000.00	.6851E+02	.1287E+04
1200.00	.6850E+02	.1284E+04
1400.00	.6850E+02	.1281E+04
1600.00	.6856E+02	.1280E+04
1801.60	.6816E+02	.1280E+04
1804.00	.6339E+02	.1281E+04
1804.60	.6137E+02	.1281E+04
1806.40	.4426E+02	.1282E+04
1807.60	.3211E+02	.1282E+04
1808.00	.2724E+02	.1283E+04
1808.40	.2157E+02	.1283E+04
1808.60	.1797E+02	.1283E+04
1809.00	.1233E+02	.1283E+04
1809.20	.0000E+00	.0000E+00
1900.00	.0000E+00	.0000E+00

Table 6.6.2-10 Steam Line Break M&E Releases Outside Containment 102% Power, 0.9 ft^2 Break		
Time (sec)	Mass Flow (lbm/sec)	Enthalpy (Btu/lbm)
.00	.0000E+00	.0000E+00
.20	.1914E+04	.1189E+04
4.20	.1807E+04	.1191E+04
8.00	.1738E+04	.1193E+04
15.60	.1652E+04	.1194E+04
15.80	.1728E+04	.1194E+04
17.40	.1830E+04	.1192E+04
19.40	.1928E+04	.1190E+04
21.40	.2002E+04	.1189E+04
24.20	.2027E+04	.1188E+04
32.00	.1881E+04	.1191E+04
75.60	.1502E+04	.1198E+04
93.20	.1381E+04	.1200E+04
110.60	.1291E+04	.1201E+04
128.20	.1226E+04	.1202E+04
145.60	.1201E+04	.1202E+04
252.60	.1192E+04	.1205E+04
272.40	.1172E+04	.1210E+04
312.00	.1108E+04	.1220E+04
331.80	.1051E+04	.1226E+04
341.60	.1011E+04	.1230E+04
353.80	.9471E+03	.1236E+04
355.20	.8341E+03	.1247E+04
356.20	.7608E+03	.1251E+04
357.40	.6842E+03	.1257E+04
358.80	.6064E+03	.1262E+04
360.00	.5476E+03	.1266E+04
361.20	.4954E+03	.1269E+04
363.80	.3967E+03	.1276E+04
366.20	.3197E+03	.1281E+04

Table 6.6.2-10 Steam Line Break M&E Releases Outside Containment 102% Power, 0.9 ft^2 Break (cont.)		
Time (sec)	Mass Flow (lbm/sec)	Enthalpy (Btu/lbm)
370.00	.2170E+03	.1287E+04
371.20	.1919E+03	.1288E+04
372.40	.1711E+03	.1290E+04
373.60	.1533E+03	.1291E+04
374.80	.1384E+03	.1292E+04
376.00	.1252E+03	.1293E+04
377.20	.1142E+03	.1293E+04
378.40	.1048E+03	.1294E+04
379.80	.9537E+02	.1295E+04
381.40	.8662E+02	.1295E+04
383.40	.7929E+02	.1296E+04
386.00	.7158E+02	.1296E+04
388.40	.6656E+02	.1296E+04
391.80	.6247E+02	.1296E+04
394.20	.6114E+02	.1296E+04
396.80	.6072E+02	.1296E+04
400.60	.6138E+02	.1296E+04
412.20	.6636E+02	.1295E+04
420.00	.6849E+02	.1294E+04
428.00	.6912E+02	.1294E+04
494.00	.6852E+02	.1294E+04
600.00	.6851E+02	.1292E+04
800.00	.6850E+02	.1290E+04
1000.00	.6850E+02	.1287E+04
1200.00	.6849E+02	.1284E+04
1400.00	.6849E+02	.1281E+04
1600.00	.6856E+02	.1280E+04
1800.60	.6871E+02	.1279E+04
1802.00	.6800E+02	.1280E+04
1803.40	.6581E+02	.1280E+04

Table 6.6.2-10 Steam Line Break M&E Releases Outside Containment 102% Power, 0.9 ft^2 Break (cont.)

Time (sec)	Mass Flow (lbm/sec)	Enthalpy (Btu/lbm)
1805.60	.6039E+02	.1281E+04
1805.80	.5799E+02	.1281E+04
1806.40	.5316E+02	.1281E+04
1808.60	.3691E+02	.1282E+04
1809.40	.3022E+02	.1283E+04
1809.80	.2644E+02	.1283E+04
1810.20	.2221E+02	.1283E+04
1810.40	.1984E+02	.1283E+04
1810.60	.1710E+02	.1283E+04
1811.00	.1224E+02	.1283E+04
1811.20	.0000E+00	.0000E+00
1900.00	.0000E+00	.0000E+00

Table 6.6.2-11 Steam Line Break Ṁ&E Releases Outside Containment 102% Power, 0.8 ft² Break		
Time (sec)	Mass Flow (lbm/sec)	Enthalpy (Btu/lbm)
.00	.0000E+00	.0000E+00
.20	.1706E+04	.1189E+04
4.20	.1620E+04	.1191E+04
8.40	.1560E+04	.1192E+04
16.60	.1489E+04	.1194E+04
16.80	.1554E+04	.1194E+04
18.60	.1656E+04	.1192E+04
20.60	.1746E+04	.1190E+04
22.60	.1812E+04	.1188E+04
24.60	.1836E+04	.1187E+04
27.60	.1803E+04	.1188E+04
34.20	.1700E+04	.1191E+04
82.00	.1368E+04	.1198E+04
101.20	.1259E+04	.1200E+04
120.20	.1176E+04	.1201E+04
139.40	.1115E+04	.1202E+04
158.60	.1086E+04	.1202E+04
267.80	.1080E+04	.1204E+04
346.20	.1013E+04	.1218E+04
372.20	.9663E+03	.1224E+04
385.20	.9316E+03	.1228E+04
408.00	.8439E+03	.1236E+04
409.80	.7300E+03	.1248E+04
411.40	.6437E+03	.1254E+04
413.00	.5708E+03	.1259E+04
414.80	.5005E+03	.1265E+04
416.40	.4467E+03	.1269E+04
418.00	.3988E+03	.1272E+04
419.60	.3551E+03	.1275E+04
421.20	.3153E+03	.1278E+04
423.00	.2753E+03	.1281E+04

Table 6.6.2-11 Steam Line Break M&E Releases Outside Containment 102% Power, 0.8 ft^2 Break (cont.)		
Time (sec)	Mass Flow (lbm/sec)	Enthalpy (Btu/lbm)
424.60	.2435E+03	.1283E+04
426.20	.2148E+03	.1285E+04
428.00	.1867E+03	.1287E+04
429.60	.1650E+03	.1289E+04
431.20	.1460E+03	.1290E+04
432.80	.1296E+03	.1291E+04
434.40	.1155E+03	.1292E+04
436.00	.1040E+03	.1293E+04
437.80	.9369E+02	.1294E+04
439.40	.8617E+02	.1294E+04
441.00	.8003E+02	.1295E+04
442.60	.7538E+02	.1295E+04
444.20	.7201E+02	.1295E+04
447.40	.6676E+02	.1295E+04
450.80	.6336E+02	.1295E+04
453.80	.6189E+02	.1295E+04
456.80	.6149E+02	.1295E+04
461.80	.6230E+02	.1295E+04
476.40	.6703E+02	.1294E+04
486.20	.6858E+02	.1293E+04
600.00	.6850E+02	.1292E+04
800.00	.6850E+02	.1289E+04
1000.00	.6849E+02	.1287E+04
1200.00	.6848E+02	.1284E+04
1400.00	.6848E+02	.1281E+04
1600.00	.6854E+02	.1281E+04
1801.40	.6864E+02	.1279E+04
1803.00	.6730E+02	.1280E+04
1804.60	.6460E+02	.1280E+04
1806.80	.5932E+02	.1281E+04

Table 6.6.2-11 Steam Line Break M&E Releases Outside Containment 102% Power, 0.8 ft^2 Break (cont.)

Time (sec)	Mass Flow (lbm/sec)	Enthalpy (Btu/lbm)
1807.20	.5598E+02	.1281E+04
1808.80	.4529E+02	.1282E+04
1810.60	.3494E+02	.1283E+04
1811.60	.2866E+02	.1283E+04
1812.40	.2276E+02	.1283E+04
1812.80	.1937E+02	.1283E+04
1813.00	.1748E+02	.1284E+04
1813.20	.1531E+02	.1284E+04
1813.40	.1219E+02	.1284E+04
1813.60	.0000E+00	.0000E+00
1900.00	.0000E+00	.0000E+00

Table 6.6.2-12 Steam Line Break M&E Releases Outside Containment 102% Power, 0.7 ft^2 Break		
Time (sec)	Mass Flow (lbm/sec)	Enthalpy (Btu/lbm)
.00	.0000E+00	.0000E+00
.20	.1496E+04	.1189E+04
4.60	.1425E+04	.1191E+04
9.20	.1376E+04	.1192E+04
18.20	.1321E+04	.1194E+04
18.40	.1375E+04	.1193E+04
20.20	.1467E+04	.1191E+04
22.20	.1549E+04	.1189E+04
24.20	.1608E+04	.1187E+04
26.20	.1631E+04	.1187E+04
28.60	.1612E+04	.1187E+04
36.80	.1511E+04	.1190E+04
89.00	.1231E+04	.1197E+04
110.00	.1136E+04	.1199E+04
130.80	.1061E+04	.1200E+04
151.80	.1003E+04	.1201E+04
172.60	.9711E+03	.1202E+04
195.40	.9626E+03	.1202E+04
253.40	.9688E+03	.1202E+04
321.40	.9492E+03	.1208E+04
389.40	.9142E+03	.1216E+04
423.40	.8761E+03	.1222E+04
440.40	.8458E+03	.1225E+04
457.40	.8059E+03	.1230E+04
478.60	.7402E+03	.1239E+04
479.00	.7136E+03	.1241E+04
480.80	.6268E+03	.1248E+04
482.80	.5494E+03	.1255E+04
484.80	.4849E+03	.1260E+04

Table 6.6.2-12 Steam Line Break M&E Releases Outside Containment 102% Power, 0.7 ft^2 Break (cont.)		
Time (sec)	Mass Flow (lbm/sec)	Enthalpy (Btu/lbm)
487.00	.4247E+03	.1265E+04
489.20	.3736E+03	.1269E+04
491.40	.3283E+03	.1273E+04
493.40	.2913E+03	.1276E+04
495.60	.2549E+03	.1279E+04
497.80	.2228E+03	.1281E+04
499.80	.1968E+03	.1284E+04
501.80	.1738E+03	.1285E+04
504.00	.1520E+03	.1287E+04
506.20	.1334E+03	.1289E+04
508.40	.1177E+03	.1290E+04
510.60	.1046E+03	.1291E+04
512.80	.9410E+02	.1292E+04
515.00	.8607E+02	.1292E+04
517.00	.8030E+02	.1293E+04
519.00	.7562E+02	.1293E+04
521.20	.7161E+02	.1293E+04
523.40	.6863E+02	.1294E+04
525.40	.6662E+02	.1294E+04
528.40	.6508E+02	.1294E+04
532.20	.6399E+02	.1294E+04
544.20	.6463E+02	.1293E+04
569.40	.6838E+02	.1292E+04
600.00	.6860E+02	.1292E+04
800.00	.6848E+02	.1289E+04
1000.00	.6848E+02	.1286E+04
1200.00	.6847E+02	.1283E+04
1400.00	.6849E+02	.1280E+04
1600.00	.6851E+02	.1281E+04
1801.40	.6876E+02	.1279E+04

Table 6.6.2-12 Steam Line Break M&E Releases Outside Containment 102% Power, 0.7 ft^2 Break (cont.)		
Time (sec)	Mass Flow (lbm/sec)	Enthalpy (Btu/lbm)
1803.60	.6708E+02	.1280E+04
1805.80	.6408E+02	.1280E+04
1808.20	.5905E+02	.1281E+04
1808.60	.5693E+02	.1281E+04
1809.40	.5133E+02	.1282E+04
1810.40	.4546E+02	.1282E+04
1812.60	.3513E+02	.1283E+04
1813.60	.3084E+02	.1284E+04
1814.80	.2510E+02	.1284E+04
1815.40	.2181E+02	.1284E+04
1816.00	.1814E+02	.1284E+04
1816.40	.1533E+02	.1285E+04
1816.80	.1163E+02	.1285E+04
1817.20	.9222E+01	.1285E+04
1817.40	.0000E+00	.0000E+00
1900.00	.0000E+00	.0000E+00

Table 6.6.2-13 Steam Line Break M&E Releases Outside Containment 102% Power, 0.6 ft^2 Break		
Time (sec)	Mass Flow (lbm/sec)	Enthalpy (Btu/lbm)
.00	.0000E+00	.0000E+00
.20	.1286E+04	.1189E+04
5.20	.1228E+04	.1191E+04
10.40	.1189E+04	.1192E+04
20.40	.1149E+04	.1193E+04
20.60	.1193E+04	.1193E+04
22.40	.1275E+04	.1191E+04
24.40	.1347E+04	.1188E+04
26.40	.1401E+04	.1187E+04
28.60	.1424E+04	.1186E+04
42.40	.1308E+04	.1190E+04
51.80	.1280E+04	.1191E+04
75.00	.1182E+04	.1194E+04
121.40	.1010E+04	.1198E+04
144.60	.9419E+03	.1200E+04
167.80	.8883E+03	.1201E+04
191.20	.8543E+03	.1201E+04
216.20	.8445E+03	.1202E+04
271.20	.8509E+03	.1201E+04
409.80	.8265E+03	.1211E+04
456.00	.8078E+03	.1215E+04
502.20	.7693E+03	.1220E+04
525.40	.7415E+03	.1224E+04
548.40	.7062E+03	.1228E+04
583.60	.6316E+03	.1238E+04
583.80	.6203E+03	.1239E+04
586.00	.5422E+03	.1247E+04
588.80	.4652E+03	.1255E+04
591.60	.4033E+03	.1261E+04
594.60	.3486E+03	.1266E+04
597.40	.3055E+03	.1270E+04

Table 6.6.2-13 Steam Line Break M&E Releases Outside Containment 102% Power, 0.6 ft^2 Break (cont.)		
Time (sec)	Mass Flow (lbm/sec)	Enthalpy (Btu/lbm)
600.20	.2675E+03	.1274E+04
603.00	.2338E+03	.1277E+04
606.00	.2023E+03	.1280E+04
609.00	.1751E+03	.1282E+04
611.80	.1532E+03	.1284E+04
614.60	.1346E+03	.1286E+04
617.60	.1179E+03	.1288E+04
620.40	.1051E+03	.1289E+04
623.40	.9395E+02	.1290E+04
626.40	.8527E+02	.1291E+04
629.20	.7924E+02	.1291E+04
632.00	.7476E+02	.1292E+04
635.00	.7125E+02	.1292E+04
638.00	.6873E+02	.1292E+04
642.80	.6628E+02	.1292E+04
647.00	.6531E+02	.1292E+04
651.20	.6506E+02	.1292E+04
675.40	.6751E+02	.1291E+04
709.60	.6858E+02	.1290E+04
800.00	.6848E+02	.1289E+04
1000.00	.6847E+02	.1286E+04
1200.00	.6846E+02	.1284E+04
1400.00	.6853E+02	.1282E+04
1600.00	.6850E+02	.1281E+04
1801.40	.6883E+02	.1280E+04
1802.60	.6846E+02	.1280E+04
1805.40	.6548E+02	.1281E+04
1810.40	.5608E+02	.1282E+04
1812.00	.4726E+02	.1283E+04
1813.40	.4066E+02	.1284E+04

Table 6.6.2-13 Steam Line Break M&E Releases Outside Containment 102% Power, 0.6 ft^2 Break (cont.)		
Time (sec)	Mass Flow (lbm/sec)	Enthalpy (Btu/lbm)
1814.60	.3615E+02	.1284E+04
1816.00	.3137E+02	.1285E+04
1817.40	.2693E+02	.1285E+04
1818.60	.2284E+02	.1286E+04
1819.40	.1975E+02	.1286E+04
1820.00	.1719E+02	.1286E+04
1820.60	.1433E+02	.1286E+04
1821.00	.1209E+02	.1287E+04
1821.20	.1072E+02	.1287E+04
1821.40	.8452E+01	.1287E+04
1821.60	.0000E+00	.0000E+00
1900.00	.0000E+00	.0000E+00

Table 6.6.2-14 Steam Line Break M&E Releases Outside Containment 102% Power, 0.5 ft^2 Break		
Time (sec)	Mass Flow (lbm/sec)	Enthalpy (Btu/lbm)
.00	.0000E+00	.0000E+00
.20	.1074E+04	.1189E+04
6.20	.1027E+04	.1191E+04
12.40	.9977E+03	.1192E+04
24.60	.9715E+03	.1193E+04
24.80	.1006E+04	.1193E+04
26.80	.1083E+04	.1190E+04
28.80	.1144E+04	.1188E+04
30.80	.1189E+04	.1186E+04
33.00	.1207E+04	.1185E+04
41.00	.1148E+04	.1188E+04
48.80	.1108E+04	.1189E+04
66.40	.1068E+04	.1191E+04
126.80	.9027E+03	.1196E+04
172.40	.8016E+03	.1199E+04
197.20	.7591E+03	.1200E+04
221.80	.7300E+03	.1201E+04
248.00	.7204E+03	.1201E+04
299.40	.7266E+03	.1201E+04
487.80	.7129E+03	.1209E+04
574.20	.6835E+03	.1214E+04
617.40	.6614E+03	.1218E+04
660.60	.6313E+03	.1222E+04
703.80	.5876E+03	.1229E+04
747.00	.5245E+03	.1238E+04
747.80	.5009E+03	.1241E+04
749.00	.4719E+03	.1244E+04
751.00	.4304E+03	.1249E+04
753.00	.3950E+03	.1253E+04
755.20	.3611E+03	.1257E+04
759.40	.3071E+03	.1263E+04

Table 6.6.2-14 Steam Line Break M&E Releases Outside Containment 102% Power, 0.5 ft^2 Break (cont.)		
Time (sec)	Mass Flow (lbm/sec)	Enthalpy (Btu/lbm)
763.60	.2636E+03	.1268E+04
767.60	.2281E+03	.1272E+04
771.80	.1957E+03	.1276E+04
776.00	.1682E+03	.1279E+04
780.20	.1449E+03	.1282E+04
784.40	.1256E+03	.1284E+04
788.40	.1106E+03	.1286E+04
792.60	.9804E+02	.1287E+04
796.80	.8825E+02	.1288E+04
801.00	.8087E+02	.1289E+04
805.20	.7551E+02	.1290E+04
809.40	.7179E+02	.1290E+04
817.80	.6787E+02	.1290E+04
824.20	.6673E+02	.1290E+04
830.80	.6644E+02	.1290E+04
859.40	.6785E+02	.1289E+04
1000.00	.6846E+02	.1287E+04
1200.00	.6845E+02	.1284E+04
1400.00	.6850E+02	.1282E+04
1600.00	.6851E+02	.1282E+04
1802.80	.6865E+02	.1281E+04
1804.40	.6761E+02	.1281E+04
1807.80	.6336E+02	.1282E+04
1812.80	.5440E+02	.1283E+04
1814.20	.4759E+02	.1284E+04
1816.40	.3961E+02	.1285E+04
1818.00	.3430E+02	.1286E+04
1819.60	.3008E+02	.1286E+04
1821.40	.2583E+02	.1287E+04
1823.00	.2232E+02	.1288E+04

Table 6.6.2-14 Steam Line Break M&E Releases Outside Containment 102% Power, 0.5 ft^2 Break (cont.)

Time (sec)	Mass Flow (lbm/sec)	Enthalpy (Btu/lbm)
1824.60	.1847E+02	.1288E+04
1825.40	.1630E+02	.1288E+04
1826.40	.1325E+02	.1289E+04
1826.80	.1188E+02	.1289E+04
1827.20	.1035E+02	.1289E+04
1827.40	.9466E+01	.1289E+04
1827.60	.8389E+01	.1289E+04
1828.00	.7736E+01	.1289E+04
1828.20	.0000E+00	.0000E+00
1900.00	.0000E+00	.0000E+00

Table 6.6.2-15 Steam Line Break M&E Releases Outside Containment 102% Power, 0.4 ft^2 Break		
Time (sec)	Mass Flow (lbm/sec)	Enthalpy (Btu/lbm)
.00	.0000E+00	.0000E+00
.20	.8613E+03	.1189E+04
5.20	.8348E+03	.1190E+04
10.00	.8180E+03	.1191E+04
19.80	.7986E+03	.1192E+04
42.80	.7895E+03	.1192E+04
43.00	.8155E+03	.1192E+04
44.80	.8749E+03	.1189E+04
47.00	.9319E+03	.1187E+04
49.00	.9701E+03	.1185E+04
51.20	.9857E+03	.1184E+04
61.00	.9366E+03	.1187E+04
70.80	.9008E+03	.1188E+04
96.20	.8662E+03	.1190E+04
191.80	.7066E+03	.1197E+04
251.60	.6288E+03	.1200E+04
281.40	.6011E+03	.1200E+04
312.20	.5902E+03	.1201E+04
360.40	.5952E+03	.1201E+04
411.20	.5915E+03	.1201E+04
542.80	.5953E+03	.1202E+04
613.00	.5887E+03	.1206E+04
753.40	.5611E+03	.1212E+04
823.80	.5423E+03	.1215E+04
894.00	.5160E+03	.1220E+04
964.20	.4757E+03	.1227E+04
999.40	.4477E+03	.1232E+04
1034.60	.4142E+03	.1238E+04
1037.20	.3723E+03	.1244E+04
1039.80	.3392E+03	.1249E+04
1042.40	.3111E+03	.1253E+04

Table 6.6.2-15 Steam Line Break M&E Releases Outside Containment 102% Power, 0.4 ft^2 Break (cont.)		
Time (sec)	Mass Flow (lbm/sec)	Enthalpy (Btu/lbm)
1045.00	.2866E+03	.1257E+04
1050.20	.2452E+03	.1262E+04
1055.40	.2116E+03	.1267E+04
1060.60	.1829E+03	.1272E+04
1065.80	.1581E+03	.1275E+04
1071.00	.1373E+03	.1278E+04
1076.20	.1201E+03	.1281E+04
1081.40	.1060E+03	.1283E+04
1086.60	.9502E+02	.1284E+04
1091.80	.8654E+02	.1285E+04
1097.00	.8014E+02	.1286E+04
1102.20	.7547E+02	.1286E+04
1107.40	.7218E+02	.1287E+04
1112.60	.6996E+02	.1287E+04
1123.40	.6765E+02	.1287E+04
1136.60	.6704E+02	.1287E+04
1279.20	.6806E+02	.1285E+04
1400.00	.6806E+02	.1283E+04
1600.00	.6812E+02	.1281E+04
1800.00	.6812E+02	.1281E+04
1801.80	.6855E+02	.1281E+04
1804.00	.6791E+02	.1281E+04
1806.20	.6612E+02	.1282E+04
1810.40	.6044E+02	.1283E+04
1814.20	.5387E+02	.1284E+04
1814.80	.5160E+02	.1285E+04
1817.00	.4457E+02	.1286E+04
1819.20	.3798E+02	.1287E+04
1823.40	.2846E+02	.1288E+04
1824.40	.2655E+02	.1289E+04

Table 6.6.2-15 Steam Line Break M&E Releases Outside Containment 102% Power, 0.4 ft^2 Break (cont.)		
Time (sec)	**Mass Flow (lbm/sec)**	**Enthalpy (Btu/lbm)**
1827.80	.2121E+02	.1290E+04
1832.20	.1473E+02	.1291E+04
1833.40	.1261E+02	.1291E+04
1834.40	.1064E+02	.1291E+04
1834.80	.9777E+01	.1291E+04
1835.40	.8333E+01	.1291E+04
1835.80	.7155E+01	.1291E+04
1836.00	.6359E+01	.1291E+04
1836.40	.5804E+01	.1291E+04
1836.60	.0000E+00	.0000E+00
1900.00	.0000E+00	.0000E+00

Table 6.6.2-16 Steam Line Break M&E Releases Outside Containment 102% Power, 0.3 ft^2 Break		
Time (sec)	Mass Flow (lbm/sec)	Enthalpy (Btu/lbm)
.00	.0000E+00	.0000E+00
.20	.6475E+03	.1189E+04
8.20	.6260E+03	.1190E+04
16.00	.6147E+03	.1191E+04
46.40	.6059E+03	.1192E+04
107.60	.6061E+03	.1192E+04
107.80	.6243E+03	.1191E+04
109.60	.6718E+03	.1189E+04
111.80	.7172E+03	.1186E+04
113.80	.7468E+03	.1184E+04
116.00	.7588E+03	.1183E+04
122.40	.7339E+03	.1185E+04
141.60	.6860E+03	.1188E+04
156.60	.6809E+03	.1188E+04
293.80	.5526E+03	.1196E+04
407.80	.4589E+03	.1200E+04
426.80	.4483E+03	.1201E+04
445.80	.4446E+03	.1201E+04
505.40	.4530E+03	.1200E+04
683.80	.4582E+03	.1200E+04
800.00	.4523E+03	.1200E+04
985.00	.4380E+03	.1204E+04
1100.00	.4282E+03	.1207E+04
1200.00	.4193E+03	.1209E+04
1300.00	.4097E+03	.1211E+04
1436.80	.3945E+03	.1214E+04
1587.40	.3722E+03	.1218E+04
1662.60	.3571E+03	.1222E+04
1800.00	.3203E+03	.1230E+04
1803.80	.2918E+03	.1236E+04
1808.40	.2634E+03	.1242E+04

Table 6.6.2-16 Steam Line Break M&E Releases Outside Containment 102% Power, 0.3 ft^2 Break (cont.)

Time (sec)	Mass Flow (lbm/sec)	Enthalpy (Btu/lbm)
1813.20	.2381E+03	.1247E+04
1822.60	.1935E+03	.1257E+04
1836.20	.1328E+03	.1271E+04
1839.00	.1144E+03	.1275E+04
1841.40	.1010E+03	.1277E+04
1843.80	.8883E+02	.1279E+04
1846.20	.7807E+02	.1282E+04
1848.40	.6957E+02	.1283E+04
1850.80	.6162E+02	.1285E+04
1853.20	.5441E+02	.1286E+04
1855.40	.4844E+02	.1287E+04
1857.80	.4272E+02	.1288E+04
1860.20	.3783E+02	.1289E+04
1865.40	.2882E+02	.1291E+04
1869.60	.2330E+02	.1292E+04
1872.00	.2048E+02	.1292E+04
1874.40	.1820E+02	.1293E+04
1881.40	.1243E+02	.1294E+04
1883.60	.1041E+02	.1294E+04
1884.80	.9187E+01	.1294E+04
1886.00	.7835E+01	.1295E+04
1886.60	.7090E+01	.1295E+04
1887.40	.5960E+01	.1295E+04
1887.80	.5258E+01	.1295E+04
1888.00	.4802E+01	.1295E+04
1888.40	.4518E+01	.1295E+04
1888.60	.0000E+00	.0000E+00
1900.00	.0000E+00	.0000E+00

Table 6.6.2-17 Steam Line Break M&E Releases Outside Containment 102% Power, 0.2 ft^2 Break		
Time (sec)	Mass Flow (lbm/sec)	Enthalpy (Btu/lbm)
.00	.0000E+00	.0000E+00
.20	.4326E+03	.1189E+04
15.20	.4181E+03	.1190E+04
156.60	.4136E+03	.1191E+04
157.00	.4304E+03	.1190E+04
160.80	.4878E+03	.1185E+04
163.00	.5097E+03	.1182E+04
165.20	.5174E+03	.1182E+04
199.20	.4652E+03	.1187E+04
248.40	.4555E+03	.1188E+04
633.00	.3066E+03	.1200E+04
661.20	.3047E+03	.1200E+04
779.40	.3149E+03	.1200E+04
897.80	.3163E+03	.1199E+04
1000.00	.3133E+03	.1200E+04
1100.00	.3089E+03	.1200E+04
1200.00	.3051E+03	.1200E+04
1300.00	.3015E+03	.1200E+04
1400.00	.2981E+03	.1201E+04
1500.00	.2949E+03	.1201E+04
1600.00	.2918E+03	.1201E+04
1700.00	.2890E+03	.1201E+04
1799.80	.2861E+03	.1201E+04
1808.60	.2709E+03	.1202E+04
1817.60	.2626E+03	.1203E+04
1835.40	.2546E+03	.1203E+04
1972.40	.2489E+03	.1205E+04
2005.40	.2360E+03	.1218E+04
2021.80	.2174E+03	.1228E+04
2030.20	.1991E+03	.1236E+04
2038.40	.1729E+03	.1248E+04

Table 6.6.2-17 Steam Line Break M&E Releases Outside Containment 102% Power, 0.2 ft^2 Break (cont.)		
Time (sec)	Mass Flow (lbm/sec)	Enthalpy (Btu/lbm)
2042.40	.1570E+03	.1254E+04
2046.60	.1356E+03	.1261E+04
2050.60	.1180E+03	.1267E+04
2058.80	.8881E+02	.1277E+04
2063.00	.7650E+02	.1280E+04
2071.20	.5732E+02	.1286E+04
2075.20	.4970E+02	.1288E+04
2079.40	.4304E+02	.1290E+04
2087.60	.3240E+02	.1293E+04
2091.80	.2799E+02	.1294E+04
2100.00	.2119E+02	.1296E+04
2104.20	.1836E+02	.1296E+04
2112.40	.1374E+02	.1298E+04
2120.60	.1046E+02	.1299E+04
2128.40	.7615E+01	.1300E+04
2128.80	.7551E+01	.1300E+04
2132.60	.5680E+01	.1300E+04
2132.80	.5746E+01	.1300E+04
2133.00	.5476E+01	.1300E+04
2133.20	.5544E+01	.1300E+04
2133.40	.5266E+01	.1300E+04
2133.60	.5337E+01	.1300E+04
2133.80	.5049E+01	.1300E+04
2134.00	.5123E+01	.1300E+04
2134.20	.4825E+01	.1300E+04
2134.40	.4902E+01	.1300E+04
2134.60	.4592E+01	.1300E+04
2134.80	.4672E+01	.1300E+04
2135.00	.4345E+01	.1300E+04
2135.20	.4432E+01	.1300E+04

Table 6.6.2-17 Steam Line Break M&E Releases Outside Containment 102% Power, 0.2 ft^2 Break (cont.)		
Time (sec)	**Mass Flow (lbm/sec)**	**Enthalpy (Btu/lbm)**
2135.40	.4082E+01	.1300E+04
2135.60	.4176E+01	.1300E+04
2135.80	.3792E+01	.1300E+04
2136.00	.3898E+01	.1300E+04
2136.20	.3453E+01	.1300E+04
2136.40	.3583E+01	.1300E+04
2136.60	.2956E+01	.1300E+04
2136.80	.3183E+01	.1300E+04
2137.00	.0000E+00	.0000E+00
2200.00	.0000E+00	.0000E+00

Table 6.6.2-18 Steam Line Break M&E Releases Outside Containment 102% Power, 0.1 ft^2 Break		
Time (sec)	Mass Flow (lbm/sec)	Enthalpy (Btu/lbm)
.00	.0000E+00	.0000E+00
.20	.2168E+03	.1189E+04
114.40	.2119E+03	.1190E+04
299.60	.2119E+03	.1190E+04
300.20	.2216E+03	.1189E+04
304.40	.2527E+03	.1183E+04
306.80	.2630E+03	.1181E+04
309.20	.2651E+03	.1181E+04
317.00	.2578E+03	.1182E+04
355.80	.2373E+03	.1186E+04
402.40	.2397E+03	.1186E+04
580.80	.2320E+03	.1187E+04
700.00	.2184E+03	.1190E+04
800.00	.2076E+03	.1192E+04
900.00	.1973E+03	.1194E+04
1054.60	.1824E+03	.1196E+04
1380.60	.1566E+03	.1200E+04
1497.20	.1640E+03	.1199E+04
1799.80	.1573E+03	.1200E+04
1931.60	.1469E+03	.1201E+04
2000.00	.1466E+03	.1201E+04
2100.00	.1451E+03	.1201E+04
2200.00	.1433E+03	.1201E+04
2300.00	.1416E+03	.1201E+04
2400.00	.1400E+03	.1202E+04
2500.00	.1385E+03	.1202E+04
2603.20	.1362E+03	.1208E+04
2633.20	.1335E+03	.1212E+04
2663.40	.1272E+03	.1218E+04
2678.40	.1202E+03	.1225E+04
2693.40	.1078E+03	.1236E+04

Table 6.6.2-18 Steam Line Break M&E Releases Outside Containment 102% Power, 0.1 ft² Break (cont.)		
Time (sec)	Mass Flow (lbm/sec)	Enthalpy (Btu/lbm)
2703.40	.9533E+02	.1246E+04
2723.40	.6752E+02	.1265E+04
2738.40	.5219E+02	.1274E+04
2753.40	.4010E+02	.1281E+04
2768.40	.3078E+02	.1286E+04
2776.00	.2685E+02	.1288E+04
2783.60	.2348E+02	.1290E+04
2798.80	.1812E+02	.1293E+04
2813.80	.1393E+02	.1295E+04
2828.80	.1082E+02	.1297E+04
2836.40	.9448E+01	.1298E+04
2851.40	.7282E+01	.1299E+04
2858.80	.6418E+01	.1300E+04
2884.40	.4062E+01	.1301E+04
2888.60	.3595E+01	.1301E+04
2888.80	.3617E+01	.1301E+04
2892.20	.3195E+01	.1302E+04
2893.20	.3145E+01	.1302E+04
2893.40	.3053E+01	.1302E+04
2893.80	.3074E+01	.1302E+04
2894.40	.3002E+01	.1302E+04
2895.20	.2832E+01	.1302E+04
2895.60	.2854E+01	.1302E+04
2896.40	.2677E+01	.1302E+04
2897.20	.2626E+01	.1302E+04
2898.40	.2463E+01	.1302E+04
2898.80	.2346E+01	.1302E+04
2899.20	.2372E+01	.1302E+04
2900.20	.2197E+01	.1302E+04
2900.40	.2192E+01	.1302E+04

Table 6.6.2-18 Steam Line Break M&E Releases Outside Containment 102% Power, 0.1 ft^2 Break (cont.)		
Time (sec)	Mass Flow (lbm/sec)	Enthalpy (Btu/lbm)
2900.60	.2065E+01	.1302E+04
2900.80	.2101E+01	.1302E+04
2901.00	.2096E+01	.1302E+04
2901.20	.1960E+01	.1302E+04
2901.40	.1999E+01	.1302E+04
2901.60	.1993E+01	.1302E+04
2901.80	.1845E+01	.1302E+04
2902.00	.1889E+01	.1302E+04
2902.20	.1882E+01	.1302E+04
2902.40	.1712E+01	.1302E+04
2902.60	.1765E+01	.1302E+04
2902.80	.1757E+01	.1302E+04
2903.00	.1539E+01	.1302E+04
2903.20	.1615E+01	.1302E+04
2903.40	.1604E+01	.1302E+04
2903.80	.1348E+01	.1302E+04
2904.00	.0000E+00	.0000E+00
3000.20	.0000E+00	.0000E+00

Table 6.6.2-19 Steam Line Break M&E Releases Outside Containment 102% Power, 0.05 ft^2 Break		
Time (sec)	Mass Flow (lbm/sec)	Enthalpy (Btu/lbm)
.00	.0000E+00	.0000E+00
.20	.1085E+03	.1189E+04
103.20	.1073E+03	.1190E+04
570.40	.1073E+03	.1190E+04
570.80	.1111E+03	.1189E+04
573.00	.1202E+03	.1185E+04
575.40	.1283E+03	.1182E+04
577.80	.1332E+03	.1180E+04
580.20	.1341E+03	.1180E+04
588.00	.1309E+03	.1181E+04
640.80	.1192E+03	.1186E+04
692.00	.1230E+03	.1184E+04
820.00	.1269E+03	.1183E+04
898.80	.1266E+03	.1183E+04
1000.00	.1248E+03	.1184E+04
1100.00	.1228E+03	.1184E+04
1200.00	.1207E+03	.1185E+04
1350.20	.1175E+03	.1187E+04
1500.00	.1140E+03	.1188E+04
1600.00	.1117E+03	.1189E+04
1700.00	.1093E+03	.1190E+04
1800.00	.1069E+03	.1191E+04
1927.80	.1091E+03	.1190E+04
2505.40	.1140E+03	.1188E+04
2890.40	.1156E+03	.1187E+04
3660.40	.1163E+03	.1187E+04
4430.40	.1149E+03	.1188E+04
7047.80	.1031E+03	.1197E+04
7181.00	.1009E+03	.1202E+04
7217.00	.9944E+02	.1205E+04
7257.40	.9650E+02	.1211E+04

Table 6.6.2-19 Steam Line Break M&E Releases Outside Containment 102% Power, 0.05 ft² Break (cont.)		
Time (sec)	Mass Flow (lbm/sec)	Enthalpy (Btu/lbm)
7303.40	.9048E+02	.1223E+04
7434.00	.6956E+02	.1257E+04
7502.40	.6037E+02	.1270E+04
7570.80	.5207E+02	.1280E+04
7648.60	.4397E+02	.1288E+04
7705.20	.3898E+02	.1294E+04
7799.80	.3170E+02	.1301E+04
7899.00	.2566E+02	.1308E+04
8002.40	.2041E+02	.1315E+04
8055.00	.1819E+02	.1318E+04
8118.80	.1584E+02	.1321E+04
8210.60	.1289E+02	.1326E+04
8288.80	.1090E+02	.1329E+04
8361.00	.9347E+01	.1332E+04
8451.20	.7669E+01	.1336E+04
8486.20	.7117E+01	.1337E+04
8545.80	.6272E+01	.1339E+04
8576.00	.4814E+01	.1341E+04
8607.00	.3683E+01	.1343E+04
8620.20	.3294E+01	.1344E+04

6.6.3 Steam Releases for Radiological Dose Analysis

6.6.3.1 Introduction

In support of radiological dose analyses, steam and radioactivity releases to the environment are postulated to occur by way of the following scenarios:

- An activity level exists in the RCS: The activity level in the RCS may be low, resulting from activated corrosion products or from the potential minute release of fission material from defective fuel assemblies. The activity level may also be moderate to high, resulting from potential fuel cladding failures and the subsequent fission product release.

- A primary-to-secondary leak occurs: The most common primary-to-secondary leak would be a leak through the wall of one or more steam generator tubes. A maximum allowable leak rate is specified in the Technical Specifications based on tube integrity requirements. The Technical Specifications leakage limit is used to determine radioactivity releases to the environment.

- Secondary-side activity is released into the atmosphere: Given that a primary-to-secondary leak exists and the condenser is not available for steam dump following an accident that produces a reactor trip, steam and radioactivity will be released through the atmospheric dump valves while the plant is being brought to a cold shutdown condition. The Loss of Non-Emergency AC Power event, and other events that result in a loss of offsite power, are situations that result in the unavailability of the condenser.

Vented steam releases have been calculated for the Loss of External Electrical Load, Loss of Non-Emergency AC Power, Locked Rotor, and Steam Line Break events to support the Callaway RSG Program.

6.6.3.2 Input Parameters and Assumptions

The following general assumptions associated with the RSG have been used in the calculation of the steam releases and feedwater flows:

- NSSS power (3,579 MWt) plus 2-percent uncertainty

- RCS average temperature (588.4°F)

- Nominal RCS pressure (2,250 psia)

- Nominal steam temperature (548.6°F) for the Model 73/19T Framatome RSGs

- Steam generator tube plugging chosen to maximize secondary-side mass inventory (The operating conditions used in this analysis reflect the high end of the Tavg RCS temperature range, high secondary-side (steam) temperature, the low end of the main feedwater temperature range, and no steam generator tube plugging.)

- A 3-percent blowdown at the lowest setpoint of the main steam safety valves
- RCS volumes and thick-metal masses associated with the Model 73/19T Framatome RSGs
- Residual heat removal (RHR) cut-in conditions of 350°F and 198 psig
- Steam releases determined for the intervals 0 to 2 hours and 2 to 8 hours

Steam dump will be required until the reactor can be placed on the RHR system. It has been assumed that 8 hours of steam release could occur prior to placing the plant in the RHR mode of operation. After the first 2 hours, it is assumed the plant will have cooled down and stabilized at no-load conditions. The additional 6 hours are required to cool down and depressurize the plant from no-load conditions to the RHR operating conditions.

An assumption in this analysis is that the entire inventory of the steam generators is released to the environment and no loss of inventory through the blowdown line is accounted for. This provides a conservative calculation of the quantity of steam vented during the noted time periods.

6.6.3.3 Description of Analyses and Evaluations

The amount of steam released to the atmosphere depends on the sensible heat and decay heat generated while reducing the temperature from the full-power value to the shutdown conditions. No computer program is used for this calculation. A hand calculation is performed to determine the amount of steam that is dissipated through the atmospheric steam release.

The total RCS energy at the end of the first 2-hour interval is subtracted from the sum of the initial RCS energy and the decay heat generated during this interval. For the Steam Line Break event, it is conservative to assume that the contents of the faulted-loop steam generator blow down within the first 2 hours with no energy extraction from the RCS (that is, no temperature decrease) due to the blowdown. Likewise, the total RCS energy at the end of the 2-to-8-hour interval is subtracted from the sum of the RCS energy and the decay heat generated during this 6-hour interval.

An energy balance during both of these intervals is used to calculate the mass of auxiliary feedwater injected during the cooldown interval. The mass of feedwater injected is used to calculate the steam mass vented to the environment through the intact-loop steam generators. For the Loss of Load, Loss of AC Power, and Locked Rotor events, 4 intact loops have been used in the steam release calculations; for the Steam Line Break event, 3 intact loops have been used for this calculation. An additional calculation is performed for the Steam Line Break event in which the contents of the faulted-loop steam generator blow down during the first 2-hour interval.

6.6.3.4 Acceptance Criteria

There are no specific acceptance criteria associated with the calculation of the steam releases and feedwater flows used as input to the radiological dose analysis. Tables of steam releases and feedwater flows for each of the cooldown intervals, for each of these 4 transients, are used as input to the radiological dose analysis in support of the RSG Program.

6.6.3.5 Results

Table 6.6.3-1 summarizes the vented steam releases from the intact-loop steam generators as well as feedwater flows for the 0-to-2-hour time period and the 2-to-8-hour time period for the Loss of Load, Loss of AC Power, Locked Rotor, and Steam Line Break events. These 2 time periods are documented to support the RSG Program.

For the Steam Line Break event, additional steam is released through the faulted-loop steam generator from the initiation of the transient up through the time at which the faulted loop is isolated from main and auxiliary feedwater flows. The steam mass from the faulted-loop steam generator is 155,500 lbm for the Callaway Model 73/19T Framatome RSGs. This steam mass is not included in the data presented in Table 6.6.3-1.

The steam releases discussed in this section are used as input to the radiological dose analysis in support of the RSG Program.

6.6.3.6 Conclusions

The steam releases and feedwater flows have been calculated at the conditions defined with Model 73/19T Framatome RSGs for the Loss of External Electrical Load, Loss of Non-Emergency AC Power, Locked Rotor, and Steam Line Break events. The assumptions delineated in subsection 6.6.3.2 have been included in the steam release calculations for each transient such that the results are consistent with and continue to comply with the current licensing-basis and acceptance requirements associated with the radiological analysis. The steam releases and feedwater flows discussed in this section have been provided for use in the radiological dose analysis to support the RSG Program.

Table 6.6.3-1 Callaway Steam Releases and Auxiliary Feedwater Flows with Model 73/19T Framatome Replacement Steam Generators

Event	Steam Release (lbm)		Feedwater Flow (lbm)	
	0 to 2 hours	2 to 8 hours	0 to 2 hours	2 to 8 hours
Loss of External Electrical Load	714,000	1,076,000	874,000	1,208,000
Loss of Non-Emergency AC Power and Locked Rotor	444,000	991,000	604,000	1,123,000
Steam Line Break	435,000	951,000	555,000	1,050,000

6.7 STEAM TUNNEL (AREA 5) ANALYSIS

6.7.1 Introduction

The steam line break scenarios documented in this report consider Callaway plant conditions with Framatome Model 73/19T steam generators operating at 102-percent core power. The specific assumptions associated with the development of these mass and energy (M&E) releases outside containment are discussed in Section 6.6. The M&E release transients are postulated to occur in the west main steam tunnel of the Auxiliary Building.

This analysis determines the compartment temperature transients and peak pressures resulting from the postulated steam line breaks.

6.7.2 GOTHIC Computer Code Model

The compartment analysis is performed with the GOTHIC version 7.1p1 code (Reference 1). GOTHIC is a multi-node containment code developed by Numerical Applications, Inc. (NAI). GOTHIC is becoming the industry standard for performing containment and outside containment compartment transients for design-basis events.

The compartment model for the main steam tunnel is comprised of two nodes representing the west and east compartments. A third node represents the environment. A boundary node and "environmental heat sink" are employed to maintain the environment node closely to atmospheric conditions throughout the transient. The east and west steam tunnel compartments are connected by a flow path that models the clear areas through column AC between the compartments. Each node has a vent area to the environment and a heat loading from the intact steam lines. The break is assumed to occur in the west compartment.

The GOTHIC model schematic is presented in Figure 6.7-1. Table 6.7-1 provides a summary of the compartment volume, flow path and heat sink data for the main steam tunnel GOTHIC model.

6.7.3 Transient Results

Fifteen break cases were generated for the GOTHIC analysis of the main steam tunnel. Each break is assumed to occur in the west compartment of the steam tunnel. These included 102-percent power level and break sizes ranging from 0.05 ft^2 to 4.6 ft^2. The details of the input assumptions that determine the M&E releases for these postulated transients are provided in Section 6.6.

For breaks less than 0.2 ft^2, the momentum of the break is not sufficient to mix the 2 compartments of the steam tunnel during the blowdown. The break compartment heats up as the steam plume fills the compartment and vents to the environment through the vents at the top of the building. The compartment temperatures remains elevated on the order of 200°F until the break flow ends and the compartments begin to cool (Figures 6.7-2 and 6.7-3). The east (non-break) compartment remains relatively cool throughout the transient.

For each case with breaks of 0.2 ft^2 and higher, the compartments mix and heat up together during the initial blowdown from the steam line. The compartment gas temperatures increases even more rapidly as the steam generator tubes uncover and the break flow becomes superheated. Compartment gas temperatures peak at approximately 450°F (Figures 6.7-4 through 6.7-16). As steam generator inventory depletes, the break flow rate deceases, and the compartments become less pressurized. The large density difference between the hot compartment gases and the ambient air begins to overcome the momentum of the break and natural circulation of ambient air into the steam tunnel begins. The break flow exits the steam tunnel through the vent in the west compartment, and the natural circulation draws air into the steam tunnel through the vent in the east compartment. Cool air enters the break compartment from the east compartment through the clear areas in column AC.

The addition of ambient air into the steam tunnel rapidly cools down the system, and the gas temperatures decrease and approach the ambient temperature. Based on a review of the data, the natural circulation begins when the superheated break flow drops below approximately 100 lbm/sec for the break greater than 0.2 ft^2.

The peak temperatures and pressures that were predicted in each compartment for the fifteen transients are provided in Table 6.7-2.

Figures 6.7-2 through 6.7-16 show plots of the temperature response of each break case in each region.

6.7.4 Conclusions

A GOTHIC model was created to simulate the steam tunnel in the Callaway plant's Auxiliary Building and compartment response to postulated main steam line break transients at conditions with the replacement steam generators (RSGs). The compartment steam temperatures within each region were calculated. The determination of the acceptability of the steam tunnel equipment qualification was performed by AmerenUE.

6.7.5 References

1. NAI-1105-04 Rev. 2, "Support for GOTHIC Containment Analysis for the Kewaunee Nuclear Power Plant," July 2002.

Table 6.7-1 Callaway Main Steam Tunnel GOTHIC Model Parameters

NODE	Description	Volume (ft³)	Bottom Elevation	Top Elevation	Initial Relative Humidity
1	MST West	59,098.92	2,026'	2,102'-6"	0.70
2	MST East	59,239,92	2,026'	2,102'-6"	0.70
3	Atmosphere	1,000,000	2,026'	3026'	0.50

Initial Pressure = 14.696 psia for all nodes
Initial Node Temperature = 120°F for all nodes

Flow Path	Description	Upstream Node	Downstream Node	Flow Area (ft²)	Loss Coefficient
1	MST West Vent	1	3	203.14	0.82
2	MST East Vent	2	3	203.14	0.82
3	Clear Areas through Column AC	1	2	633.84	0.82
4	Break Flow	Boundary Node	1	–	–
5	Atmospheric Pressure	3	Boundary Node	–	–

Heat Sink	Material	Node	Surface Area (ft²)	Thickness (ft)	Boundary Condition	
					Left	Right
1	Structural Steel	1	4,287.13	0.042	Uchida	Adiabatic
2	Structural Steel	2	4,300.96	0.042	Uchida	Adiabatic
3	Concrete Floor	1	945	2	Uchida	Adiabatic
4	Concrete Floor	2	945	2	Uchida	Adiabatic
5	Concrete Column	1	3,200	1	Uchida	Adiabatic
6	Concrete Column	2	3,200	1	Uchida	Adiabatic
7	Concrete Column	1	3,352.5	2	Uchida	Adiabatic
8	Concrete Column	2	3,352.5	2	Uchida	Adiabatic
9	Interior Concrete	1	1,665	1	Uchida	Adiabatic
10	Interior Concrete	2	1,665	1	Uchida	Adiabatic
11	Concrete Column	1	1,639	2	Uchida	Adiabatic
12	Concrete Column	2	1,639	2	Uchida	Adiabatic
13	Concrete Wall	1	1,865	4	Uchida	Adiabatic
14	Concrete Wall	2	1,865	4	Uchida	Adiabatic
15	Concrete Roof	1	365	2	Uchida	Adiabatic
16	Concrete Roof	2	365	2	Uchida	Adiabatic
17	Environmental Heat Sink	3	–	–	Infinite HTC	95°F

Table 6.7-2 Callaway Main Steam Tunnel Steam Line Break Analysis Peak Temperature and Pressure Results

Break Size (ft^2)	Peak Temperature (°F)		Peak Pressure (psia)	
	West	East	West	East
0.05	219.2	205.0	14.79	14.79
0.1	238.5	129.2	14.78	14.78
0.2	409.9	329.2	14.77	14.77
0.3	421.2	355.2	14.79	14.79
0.4	435.7	373.8	14.81	14.81
0.5	443.5	383.7	14.83	14.82
0.6	448.7	391.4	14.85	14.84
0.7	452.6	396.4	14.87	14.86
0.8	454.8	401.4	14.89	14.87
0.9	459.3	400.9	14.91	14.89
1.0	460.2	403.1	14.94	14.91
1.2	446.9	397.3	15.01	14.97
1.4	453.3	392.6	15.09	15.03
2.0	453.5	392.4	15.32	15.24
4.6	453.8	394.1	16.56	16.45

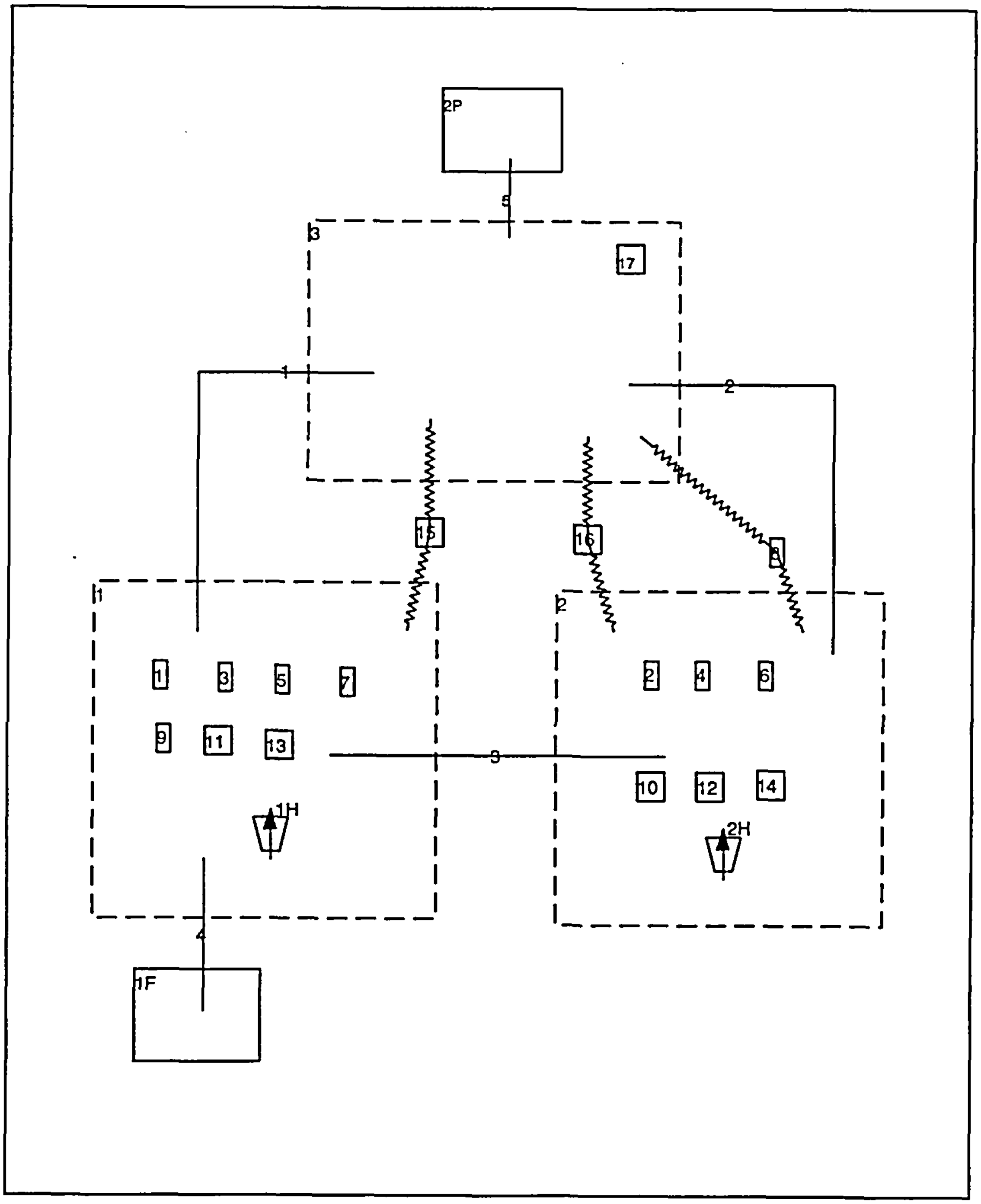


Figure 6.7-1 Schematic Drawing of the Callaway Steam Tunnel GOTHIC Model

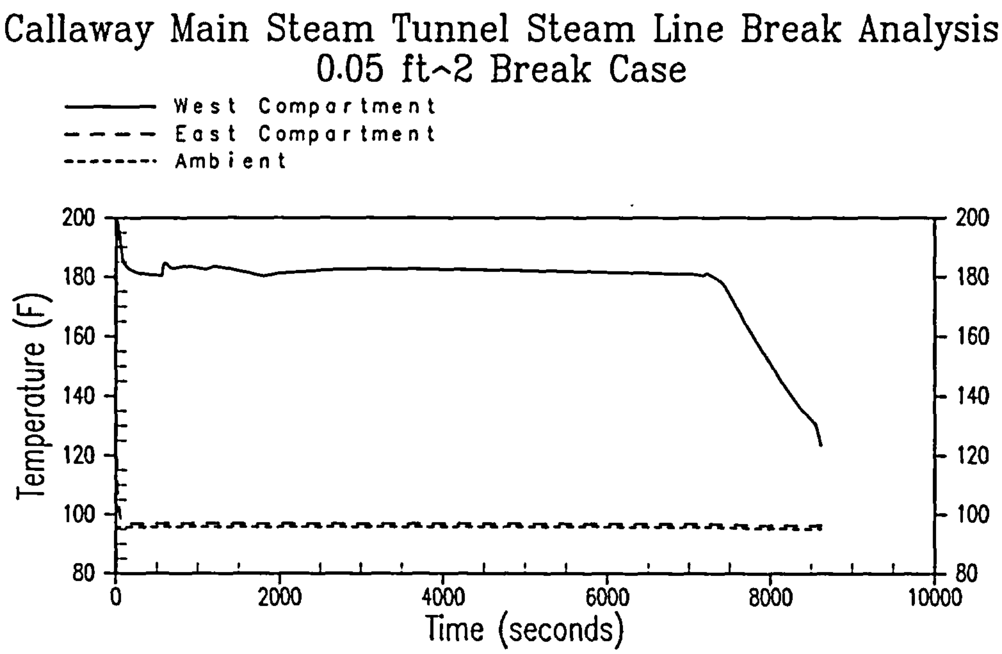


Figure 6.7-2 Steam Tunnel Temperature Results, 0.05 ft^2 Steam Line Break

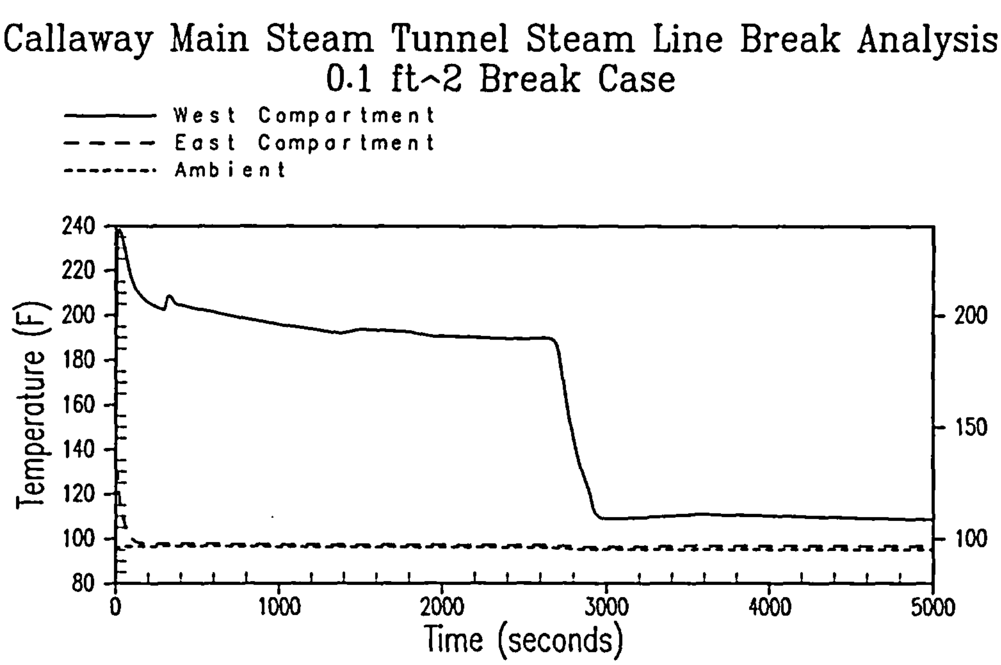


Figure 6.7-3 Steam Tunnel Temperature Results, 0.1 ft^2 Steam Line Break

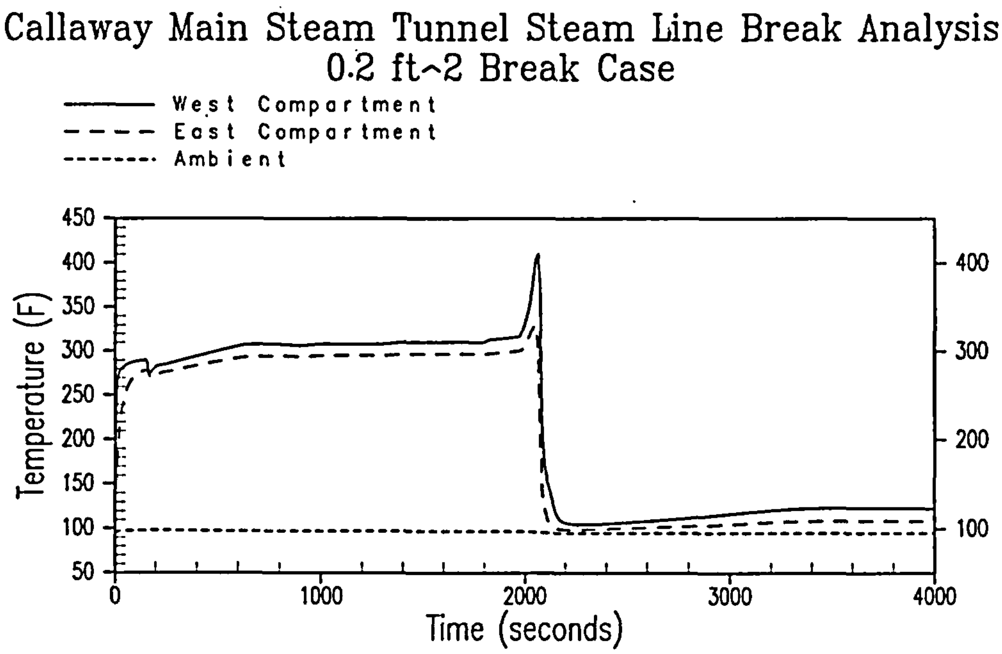


Figure 6.7-4 Steam Tunnel Temperature Results, 0.2 ft² Steam Line Break

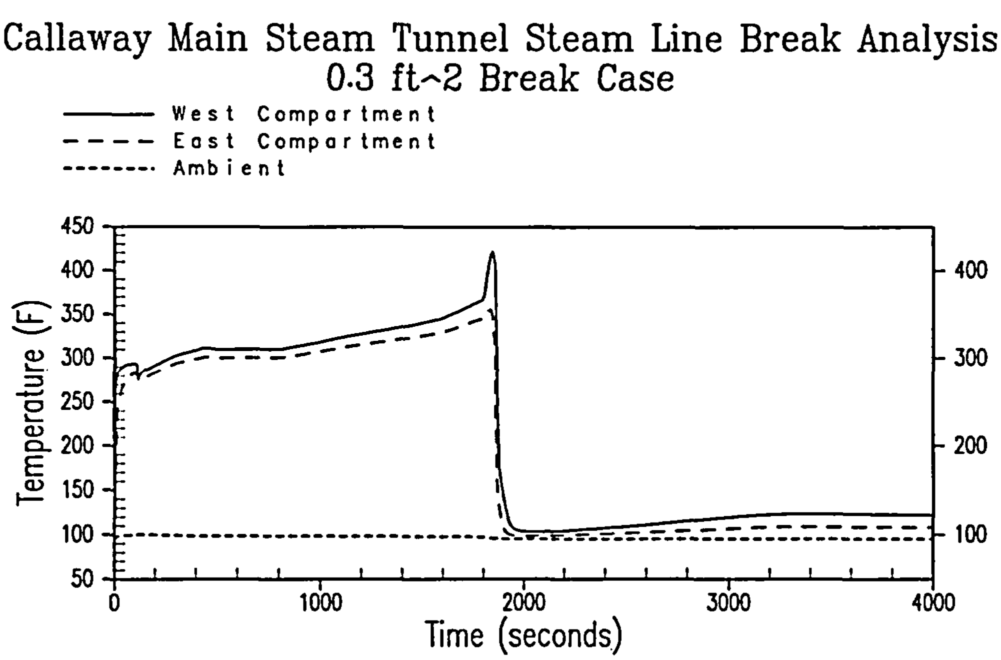


Figure 6.7-5 Steam Tunnel Temperature Results, 0.3 ft² Steam Line Break

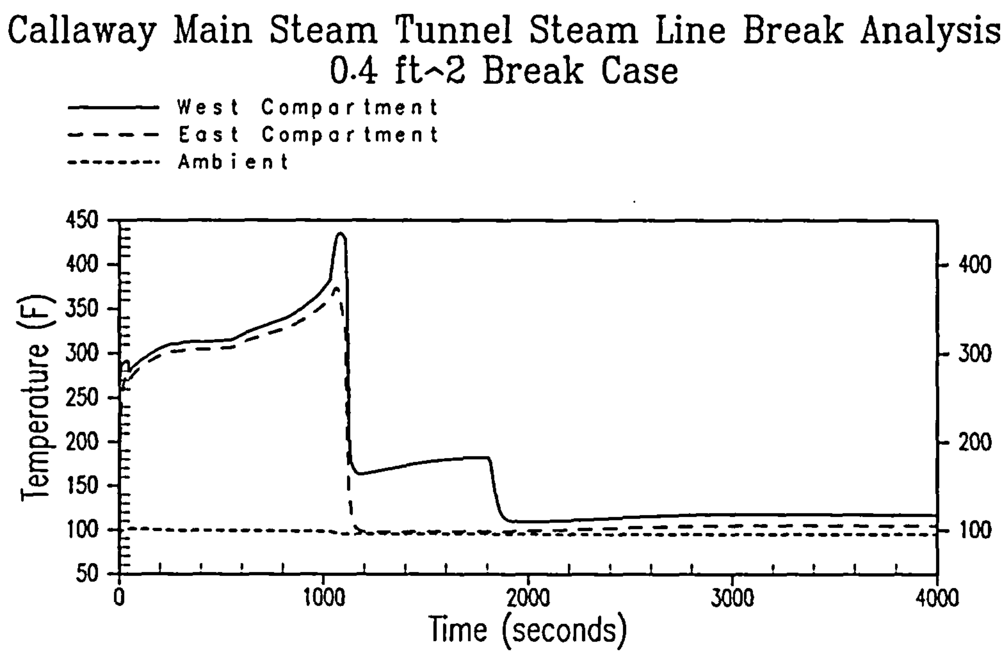


Figure 6.7-6 Steam Tunnel Temperature Results, 0.4 ft^2 Steam Line Break

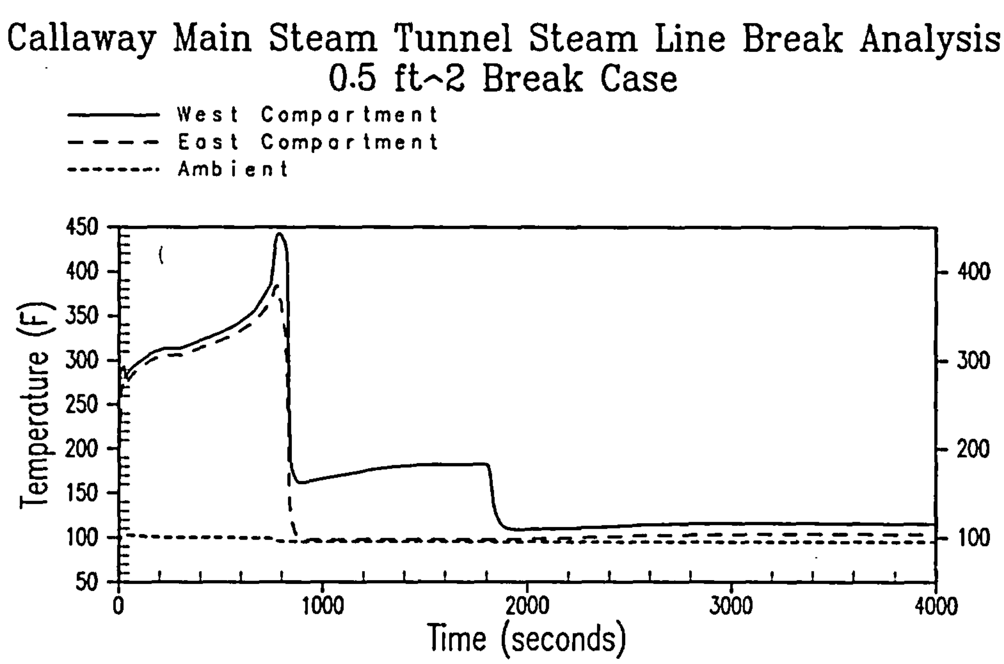


Figure 6.7-7 Steam Tunnel Temperature Results, 0.5 ft^2 Steam Line Break

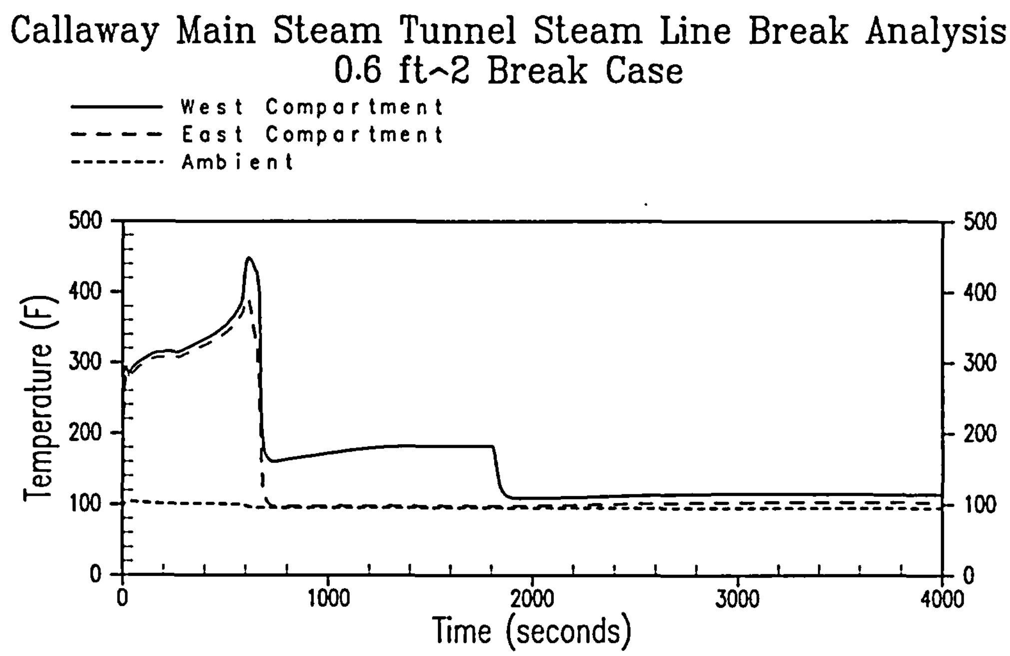


Figure 6.7-8 Steam Tunnel Temperature Results, 0.6 ft^2 Steam Line Break

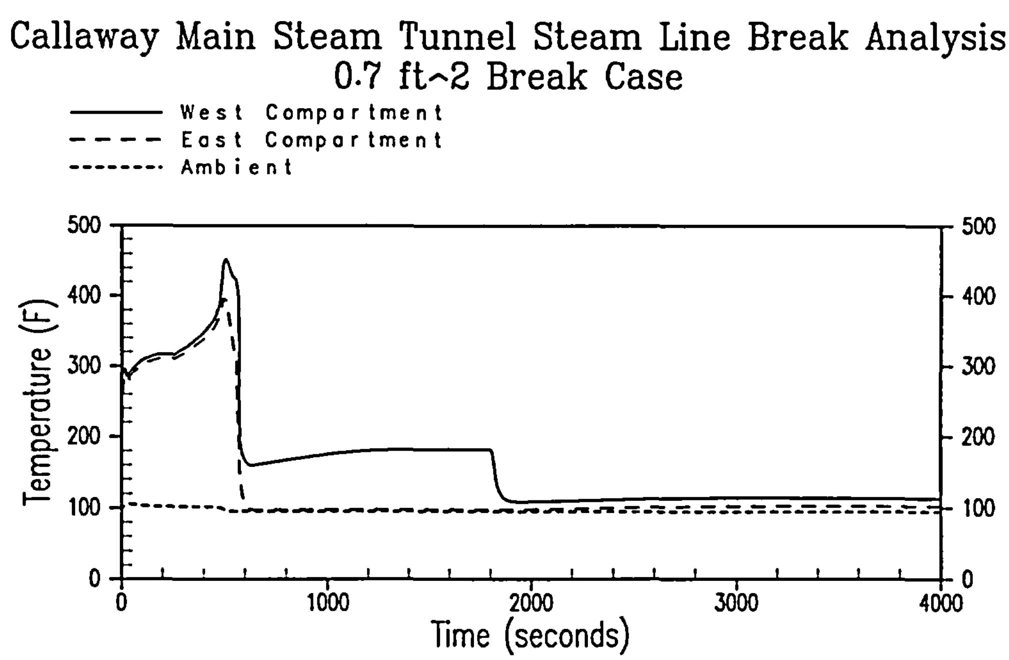


Figure 6.7-9 Steam Tunnel Temperature Results, 0.7 ft^2 Steam Line Break

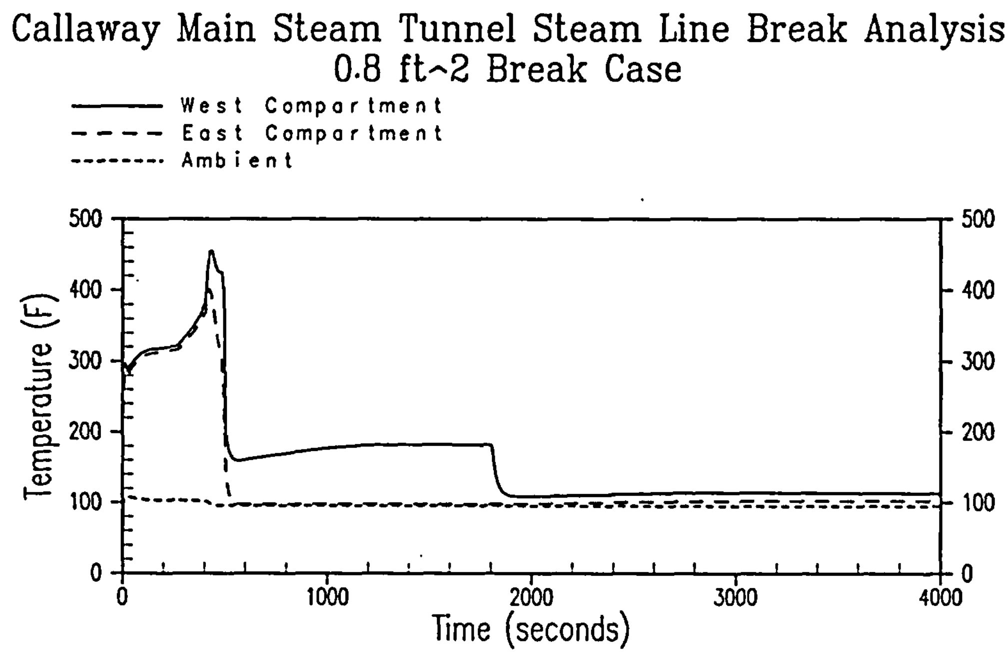


Figure 6.7-10 Steam Tunnel Temperature Results, 0.8 ft^2 Steam Line Break

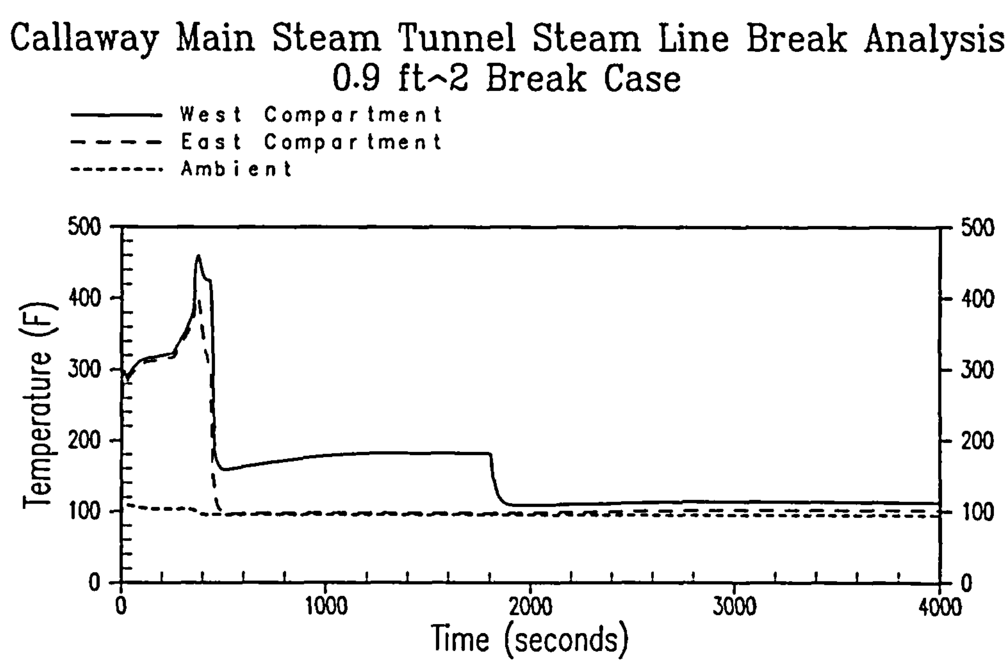


Figure 6.7-11 Steam Tunnel Temperature Results, 0.9 ft^2 Steam Line Break

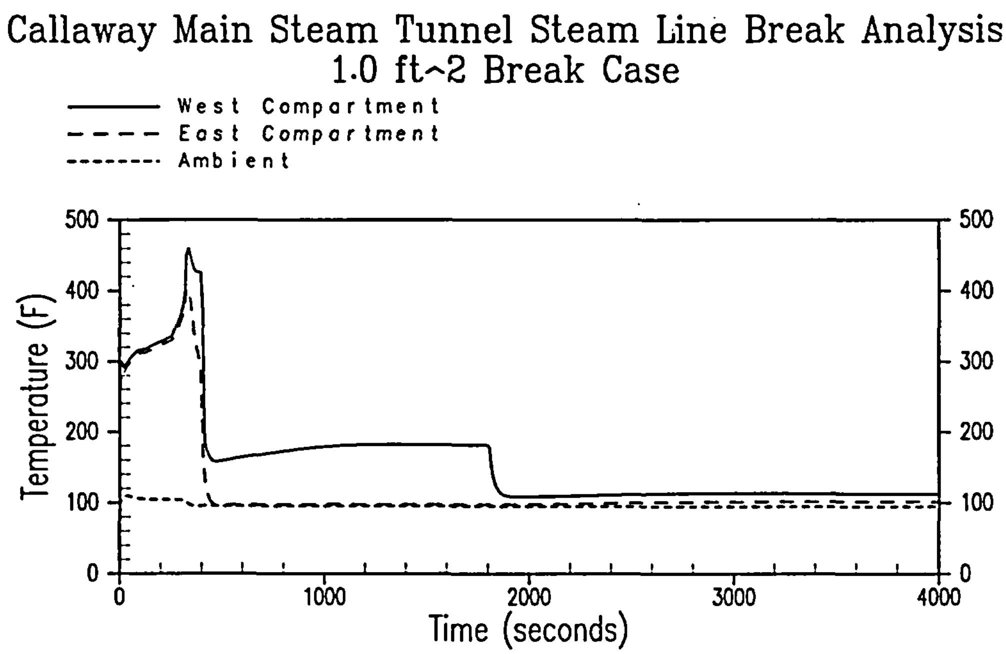


Figure 6.7-12 Steam Tunnel Temperature Results, 1.0 ft² Steam Line Break

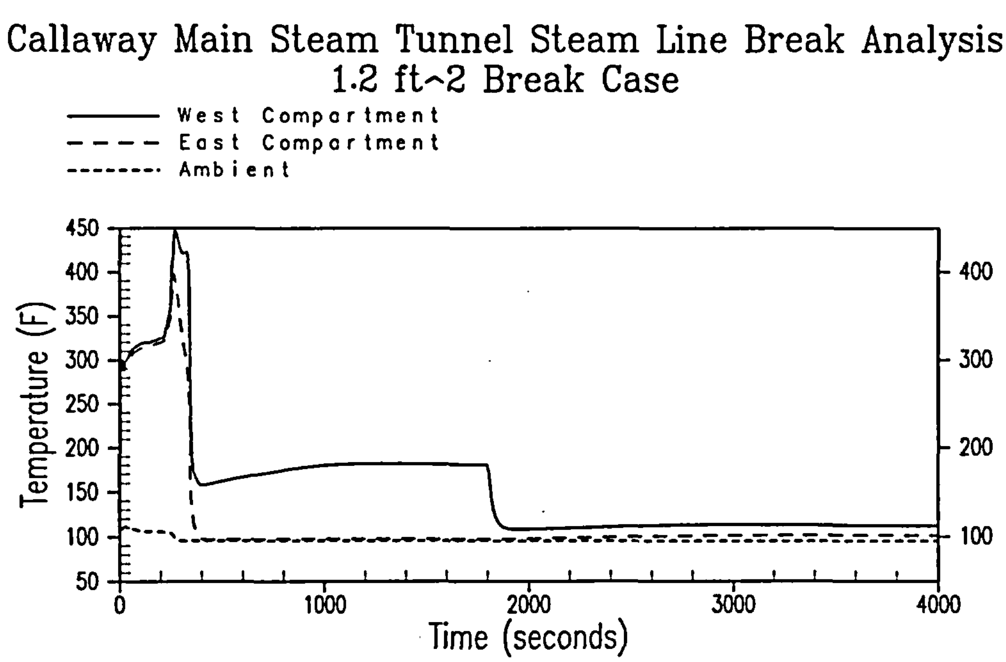


Figure 6.7-13 Steam Tunnel Temperature Results, 1.2 ft² Steam Line Break

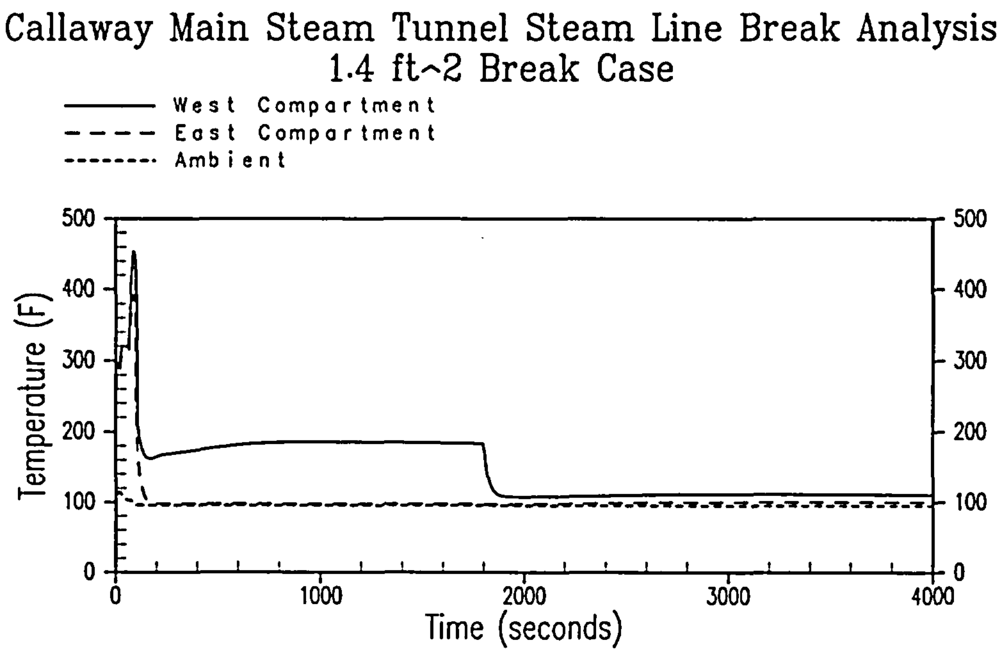


Figure 6.7-14 Steam Tunnel Temperature Results, 1.4 ft^2 Steam Line Break

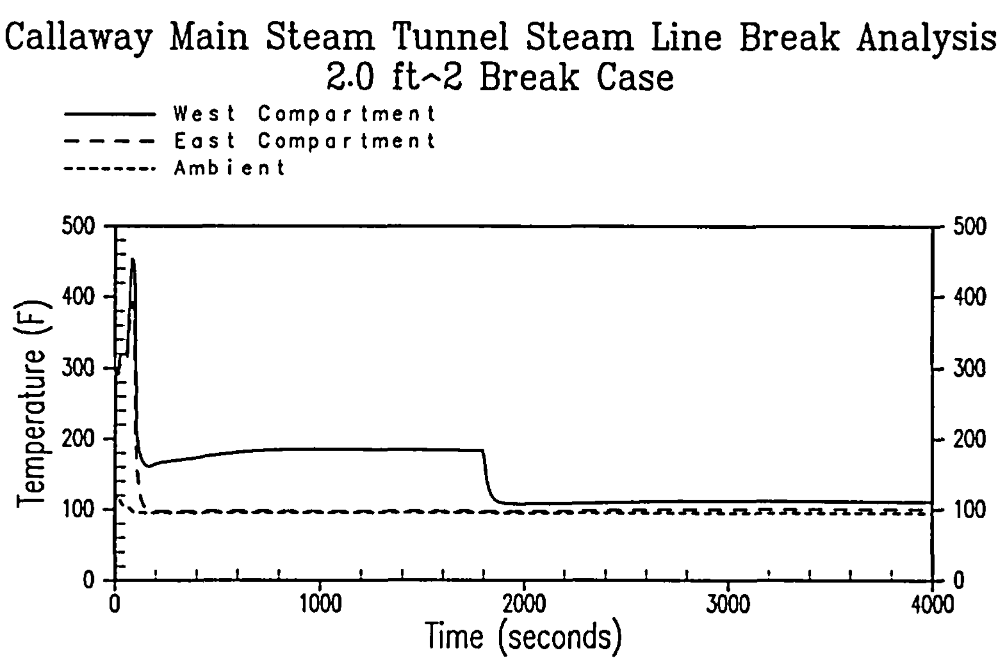


Figure 6.7-15 Steam Tunnel Temperature Results, 2.0 ft^2 Steam Line Break

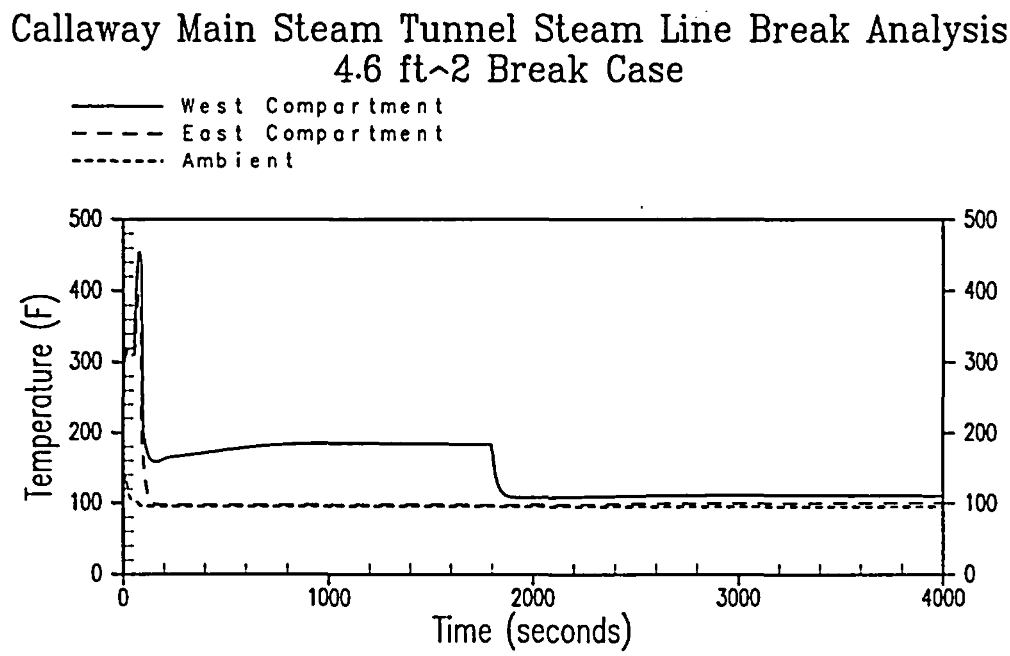


Figure 6.7-16 Steam Tunnel Temperature Results, 4.6 ft^2 Steam Line Break

6.8 LOCA HYDRAULIC FORCES

6.8.1 Introduction

The purpose of an analysis of the loss-of-coolant-accident (LOCA) hydraulic forces is to generate the hydraulic forcing functions that occur on reactor coolant system (RCS) components as a result of a postulated LOCA.

The forces created by a postulated break in the RCS piping are principally caused by the motion of the decompression wave through the RCS. The strength of the decompression wave is primarily a function of the assumed break opening time, break area, and RCS fluid density, which is a function of the operating conditions for temperature and pressure. The LOCA forces are relatively insensitive to initial RCS conditions of flow and power except as they relate to coolant density, where higher coolant density leads to increased LOCA forces. Lower flow rates and higher power lead to lower cold leg temperatures (and higher coolant density) for the same average coolant temperature. Therefore, limiting LOCA hydraulic forces are generated for conditions consistent with minimum thermal design flow and maximum RCS power.

6.8.2 Input Parameters and Assumptions

To conservatively assess LOCA forces analyses for the Callaway Plant, the following operating conditions were considered in establishing the limiting temperatures and pressures:

- Initial RCS conditions associated with a minimum thermal design flow of 93,600 gpm per loop
- Core power of 3,565 MWt (nuclear steam supply system (NSSS) power of 3,579 MWt)
- Framatome Model 73/19T replacement steam generators (RSGs)
- An RCS hot-full-power (HFP) Tavg range of 570.7°F to 588.4°F
- A feedwater temperature range of 390.0°F to 446.0°F
- An RCS pressure of 2,250 psia
- An RCS temperature uncertainty of ± 6.0°F
- A pressurizer pressure uncertainty of ± 50.0 psi

Based on these conditions, the LOCA forces were generated at a minimum Tcold of 532.2°F, including uncertainty, and a pressurizer pressure of 2,300 psia, including uncertainty.

The hydraulic forcing functions that occur as a result of a postulated LOCA are calculated assuming a limiting break location and break area. The limiting break location and area vary with the RCS component under consideration, but historically the limiting postulated breaks are a limited displacement reactor pressure vessel (RPV) inlet/outlet nozzle break or a double-ended guillotine (DEG) reactor coolant pump (RCP)/steam generator (SG) inlet/outlet nozzle break. General Design Criterion (GDC)-4 allows main coolant piping breaks to be "excluded from the design basis when analyses reviewed and approved by the Commission demonstrate that the probability of fluid system piping rupture is extremely low under conditions consistent with the design basis for the piping." This exemption is generally referred to as leak before break (LBB). The technical justification for application of LBB to the Callaway Plant is documented in Reference 1.

Leak-before-break licensing allows RCS components to be evaluated for LOCA integrity considering the next most limiting auxiliary line breaks. The next most limiting auxiliary line breaks considered were the accumulator line, the pressurizer surge line, and the residual heat removal (RHR) line, which have smaller areas than postulated breaks in the main RCS loop piping.

Other key inputs for the analysis include the RCS geometry and hydraulic losses, as well as the structural beam model of the core barrel (significant in vessel analyses).

6.8.3 Description of Analysis and Results

The LOCA forces are generated with a focus on the component of interest (loop and vessel). Minor differences in one component generally have no significant effect on forces generated for the other components. For example, differences in fuel type, while significant to vessel LOCA forces, have been found to have no significant effect on loop forces.

The LOCA system thermal-hydraulics transient data used for the original qualification of the loops and the reactor vessel and internals were performed using the Nuclear Regulatory Commission (NRC)-approved MULTIFLEX 1.0 computer code, Reference 2. The most recent qualification of the vessel internals and fuel was performed using an advanced beam model version of MULTIFLEX 3.0, Reference 3, in accordance with methodology approved by the NRC in Reference 4. Both versions of the MULTIFLEX code share a common hydraulic modeling scheme, with the differences being confined to a more realistic downcomer hydraulic network and a more realistic core barrel structural model that accounts for non-linear boundary conditions and vessel motion. Generally, this improved modeling results in lower, more realistic, but still conservative hydraulic forces on the core barrel.

The MULTIFLEX computer code calculates the thermal-hydraulic transient within the RCS and considers subcooled, transition, and early two-phase (saturated) blowdown regimes. The code employs the method of characteristics to solve the conservation laws, assuming one-dimensional flow and a homogeneous liquid-vapor mixture. The RCS is divided into subregions in which each subregion is regarded as an equivalent pipe. A complex network of these equivalent pipes is used to represent the entire primary RCS.

A coupled fluid-structure interaction is incorporated into the MULTIFLEX code by accounting for the deflection of the constraining boundaries, which are represented by separate spring-mass oscillator systems. For the reactor vessel/internals analysis, the reactor core barrel is modeled as an equivalent beam with the structural properties of the core barrel in a plane parallel to the broken inlet nozzle. Mass and stiffness matrices that are obtained from an independent modal analysis of the reactor core barrel are applied in the equations of structural vibration at each of the mass point locations. Horizontal forces are then calculated by applying the spatial pressure variation to the wall area at each of the elevations representative of the mass points of the beam model. The resultant core barrel motion is then translated into an equivalent change in flow area in each downcomer annulus flow channel. At every time increment, the code iterates between the hydraulic and structural subroutines of the program at each location confined by a flexible wall. For the reactor pressure vessel and specific vessel internal components, the MULTIFLEX code generates the LOCA pressure transient that is input to the LATFORC and FORCE2 post-processing codes (Reference 2). These codes, in turn, are used to calculate the actual forces on the various components.

The LATFORC computer code employs the field pressures generated by MULTIFLEX code, together with geometric vessel information (component radial and axial lengths), to determine the horizontal forces on the vessel wall, core barrel, and thermal shield. The LATFORC code represents the downcomer region with a model that is consistent with the model used in the MULTIFLEX blowdown calculations. The downcomer annulus is subdivided into cylindrical segments, formed by dividing this region into circumferential and axial zones. The results of the MULTIFLEX/LATFORC analysis of the horizontal forces are calculated for the initial 500 msec of the blowdown transient and are stored in a computer file. These forcing functions, combined with vertical LOCA hydraulic forces, seismic, thermal, and system shaking loads, are used by the cognizant structural groups to determine the resultant mechanical loads on the reactor pressure vessel and vessel internals.

The FORCE2 computer code calculates the hydraulic forces that the RCS coolant exerts on the vessel internals in the vertical direction. The FORCE2 code uses a detailed geometric description of the vessel components and the transient pressures, mass velocities, and densities computed by the MULTIFLEX code. The analytical basis for the derivation of the mathematical equations employed in the FORCE2 code is the one-dimensional conservation of linear momentum. Note that the computed vertical forces do not include body forces on the vessel internals, such as deadweight or buoyancy. When the vertical forces on the reactor pressure vessel internals are calculated, pressure differential forces, flow stagnation forces, unrecoverable orifice losses, and friction losses on the individual components are considered. These force components are then summed together, depending upon the significance of each, to yield the total vertical force acting on a given component. The results of the MULTIFLEX/FORCE2 analysis of the vertical forces are calculated for the initial 500 msec of the blowdown transient and are stored in a computer file. These forcing functions, combined with horizontal LOCA hydraulic forces, seismic, thermal, and system shaking loads, were used in the structural evaluations to determine the resultant mechanical loads on the vessel and vessel internals.

The loop forces analysis was completed using the THRUST post-processing code. The THRUST code is used to generate the X, Y, and Z directional component forces during a LOCA blowdown from the RCS pressure, density, and mass flux. The THRUST code is described and documented in Reference 5.

For the Callaway Plant RSG Program, the loop and reactor vessel and internals LOCA hydraulic forces analyses directly modeled the Framatome Model 73/19T RSGs and reduced Tavg operating conditions. These analyses included reactor vessel and internals (including fuel) and loop piping forces. The vessel and loop piping forces analyses considered 3 possible break locations, 1 cold leg break, and 2 hot leg breaks. These were the accumulator branch line (10" Schedule 140, 0.4176 ft^2), pressurizer surge line (14" Schedule 160, 0.6827 ft^2), and the RHR branch line (12" Schedule 140, 0.6013 ft^2). The results of these analyses were then used as input to the structural analyses for component qualification.

Note that Westinghouse recently completed LBB qualification analyses for the surge line (Reference 6), accumulator line (Reference 7), and RHR line (Reference 8). At this time, the NRC has granted approval for LBB on the accumulator and RHR lines (Reference 9). The LOCA forces for the RSG were conservatively generated based on a break in the surge line, accumulator, and RHR line so that margin would be available for an easy evaluation if future changes affecting the LOCA forces calculation were to occur. This margin will exist for the LOCA forces that are based on breaks in the RHR and accumulator lines. If the NRC grants approval of the surge line LBB at some point, margin will also be available then for the forces based on that break.

6.8.4 Acceptance Criteria

The LOCA hydraulic forces are provided as input to structural qualification analyses, and as such have no independent regulatory acceptance criteria. The structural analyses performed using these forcing functions are done to demonstrate compliance with the Code of Federal Regulations (CFR) 10 CFR 50, Appendix A, GDC- 4.

6.8.5 Conclusions and Results

LOCA hydraulic forces were generated for the Callaway Plant for the Framatome Model 73/19T RSG and reduced Tavg specified in subsection 6.8.2. Since the reactor vessel and internals (subsection 5.2.2), fuel (Section 5.3), and loop analyses (Section 5.5) show acceptable results, the steam generator replacement and reduced Tavg are acceptable from a LOCA hydraulic forces standpoint.

6.8.6 References

1. WCAP-14059 (Proprietary) and WCAP-14060 (Non-Proprietary), "Technical Justification for Eliminating Large Primary Loop Rupture as the Structural Design Basis for the Callaway and Wolf Creek Plants After Elimination of SG Snubbers," D. C. Bhowmick et al., August 1994.

2. WCAP-8708-P-A (Proprietary) and WCAP-8709-A (Non-Proprietary), "MULTIFLEX A FORTRAN-IV Computer Program for Analyzing Thermal-Hydraulic-Structure System Dynamics," K. Takeuchi et al., September 1977.

3. WCAP-9735, Rev. 2 (Proprietary) and WCAP-9736, Revision 1 (Non-Proprietary), "MULTIFLEX 3.0 A FORTRAN-IV Computer Program for Analyzing Thermal-Hydraulic-Structural System Dynamics Advanced Beam Model," K. Takeuchi et al., February 1998.

4. WCAP-15029-P-A (Proprietary) and WCAP-15030-NP-A (Non-Proprietary), "Westinghouse Methodology for Evaluating the Acceptability of Baffle-Former-Barrel Bolting Distributions Under Faulted Load Conditions," R. E. Schwirian et al., January 1999.

5. WCAP-8252, Rev. 1 (Non-Proprietary), "Documentation of Selected Westinghouse Structural Analysis Computer Codes," K. M. Vashi, May 1977.

6. WCAP-15983-P, "Technical Justification for Eliminating Pressurizer Surge Line Rupture as the Structural Design Basis for the Callaway Nuclear Power Plant," February 2003.

7. WCAP-16019-P, "Technical Justification for Eliminating 10" Accumulator Lines Rupture as the Structural Design Basis for the Callaway Nuclear Power Plant," February 2003.

8. WCAP-16020-P, "Technical Justification for Eliminating 12" Residual Heat Removal (RHR) Lines Rupture as the Structural Design Basis for Callaway Nuclear Power Plant," February 2003.

9. NRC Docket No. 50-483, "Safety Evaluation by Office of Nuclear Reactor Regulation Related to Amendment No. 161 to Facility Operating License Number NPF-30, Union Electric Company, Callaway Plant Unit 1," April 12, 2004.

6.9 REACTOR TRIP SYSTEM/ENGINEERED SAFETY FEATURE ACTUATION SYSTEM SETPOINTS

6.9.1 Introduction and Background

The Technical Specification reactor trip system (RTS)/engineered safety feature actuation system (ESFAS) setpoints that were judged to be affected by the replacement steam generator (RSG) have been reviewed for operation at RSG conditions, and Technical Specification changes have been identified consistent with the Westinghouse setpoint methodology.

6.9.2 Input Parameters and Assumptions

The uncertainty calculations for the Callaway Plant were performed based on the plant-specific instrumentation, plant calibration procedures, and customer-issued letters.

6.9.3 Description of Analyses and Evaluations

The setpoint analysis uses the square-root-sum-of-the-squares (SRSS) technique to combine the uncertainty components of an instrument channel in an appropriate combination of those components, or groups of components, which are statistically independent. Those uncertainties that are not independent are arithmetically summed to produce groups that are independent of each other, which can then be statistically combined. The determination of the RTS/ESFAS setpoints included the effects identified in Reference 1.

6.9.4 Acceptance Criteria and Results

Margin is defined as the difference between the Total Allowance (TA) and the Channel Statistical Allowance (CSA). Total Allowance is the difference between the limiting safety analysis limit and the Technical Specification trip setpoint (in percent of instrument span). Channel Statistical Allowance is the statistical combination of the instrument channel uncertainty components (in percent of instrument span). The acceptance criterion for the RTS/ESFAS setpoints is that margin is greater than or equal to zero.

The allowable values for the Callaway Technical Specifications are determined by adding (or subtracting) the calibration accuracy of the device tested during the channel operational test to the nominal trip setpoint (NTS) in the non-conservative direction (i.e., toward or closer to the safety analysis limit (SAL)) for the application. For those channels that provide trip actuation via a bistable in the process racks, the calibration accuracy is defined by the rack calibration accuracy term. The magnitude of the calibration accuracy term is specified in the plant procedures (Refer to Table 6.9-1).

6.9.5 Conclusions

The RTS/ESFAS functions analyzed are acceptable for the RSG Program.

6.9.6 References

1. WCAP-16115-P, "Steam Generator Level Uncertainties," September 2003.

Table 6.9-1 Summary of the Technical Specification Allowable Value Changes					
Parameter	**NTS**	**CSA**	**Margin**	**Current Allowable Value**	**Proposed Allowable Value**
Steam Line Pressure-Low Steam Line Isolation	615 psig	[ ]^{a,c} psi$^{(1)}$ [ ]^{a,c} psi$^{(2)}$	[ ]^{a,c} psi$^{(1)}$ [ ]^{a,c} psi$^{(2)}$	≥ 571 psig	≥ 609 psig
Steam Line Pressure-Low SI	615 psig	[ ]^{a,c} psi$^{(1)}$ [ ]^{a,c} psi$^{(2)}$	[ ]^{a,c} psi$^{(1)}$ [ ]^{a,c} psi$^{(2)}$	≥ 571 psig	≥ 609 psig
Steam Generator Level Low-Low (Normal)	17% Span	[ ]^{a,c} % Span	[ ]^{a,c} % Span	≥ 19.8% Span	≥ 16.6% Span
Steam Generator Level Low-Low (Adverse)	21.0% Span	[ ]^{a,c} % Span	[ ]^{a,c} % Span	≥ 25.2% Span	≥ 20.6% Span
Steam Generator Level High-High	91.0% Span	[ ]^{a,c} % Span	[ ]^{a,c} % Span	≤ 79.8% Span	≤91.4% Span

Notes:

1. Results based on an inside containment steam line break.
2. Results based on an outside containment steam line break.

Bracketed [ ]^{a,c} information designates data that is Westinghouse proprietary, as discussed in Section 1.6 of this report.

6.10 ROD EJECTION RELEASES FOR DOSE

6.10.1 Introduction

The analysis of the radiological consequences of a rod ejection accident models two release pathways. The first pathway is a release of activity into the containment followed by leakage from the containment to the outside atmosphere. The second release pathway is the transfer of activity from the primary system to the secondary system via steam generator tube leakage, followed by the release of activity from the steam generators as steam.

6.10.2 Input Parameters and Assumptions

The plant response to a rod ejection can be approximated by the predicted response to a small break loss-of-coolant accident (SBLOCA) since the rod ejection results in a hole in the upper head (and is, in effect, an SBLOCA).

The SBLOCA transients analyzed for the Callaway replacement Steam Generator (RSG) Program (Section 6.2.2) are all cold leg breaks. A break in the upper head would result in a more rapid primary-side depressurization than a cold leg break since steam would be vented through the break early in the event, compared to the break in the cold leg where steam is not vented until the loop seal clears. The size of the break corresponding to a rod ejection is approximately 2.75 inches. The closest break size considered in the SBLOCA analyses is the 3-inch break. Although a 3-inch cold leg break would depressurize more rapidly than a 2.75-inch cold leg break, it would still be slower than the depressurization for a 2.75-inch upper head break. Therefore, the 3-inch cold leg break is used to develop rod ejection data for use in the dose analysis.

6.10.3 Description of Analyses and Evaluations

When the primary system pressure drops below the secondary system pressure, the leakage stops. Heat transfer to the steam generator's secondary also stops, terminating the release of steam from the steam generators. The SBLOCA analysis also provides information on the time for reactor trip and safety injection (SI) system actuation (resulting from the initial depressurization), which can be used (depending on plant systems) in determining the time for control room isolation. Three pieces of information are obtained from the SBLOCA analysis for use in the rod ejection dose analysis:

- Time for SI system actuation on low pressurizer pressure
- Time when primary-system pressure falls below secondary-system pressure
- Steam releases from steam generators

Significant conservatism is applied in determining the data to be used in the rod ejection dose analysis (discussed in the following paragraphs) in order to assure that the data obtained from the SBLOCA analysis is bounding to support use of the 3-inch SBLOCA analysis results to represent the rod ejection and to limit the chances that an SBLOCA re-analysis would impact the rod ejection dose analysis.

6.10.4 Acceptance Criteria

There are no specific acceptance criteria associated with the calculation of the rod ejection releases used as input to the radiological dose analysis. The results are used as input to the radiological dose analysis in support of the RSG Program.

6.10.5 Results

The SI actuation time for the 3-inch cold leg break is 32 seconds (See subsection 6.2.2, Table 6.2.2-4). This is conservatively increased to 60 seconds for use in the control room dose analysis.

The primary and secondary pressures from the 3-inch SBLOCA transient are presented in Figure 6.10-1. From the figure, the primary pressure falls below the secondary pressure before 800 seconds. This is conservatively increased to 1,200 seconds for use in the dose analysis.

The post-trip atmospheric steam releases from the 3-inch SBLOCA transient are presented in Figure 6.10-2. Pre-trip flow is not included since a rod ejection results in a reactor trip signal generated in the first seconds of the transient. Also the flow would be passed through the condenser with a partition coefficient of 100 before trip and the assumed loss of offsite power. The conservative assumption of loss of offsite power at the start of the event and immediate initiation of steam releases at a high rate assure that the analysis is bounding. From Figure 6.10-2, the steam releases stop before the primary pressure falls below the secondary pressure. The integrated flow is approximately 90,000 lbm. For use in the dose analysis, this is conservatively increased to 200,000 lbm over the assumed steam release period of 1,200 seconds.

6.10.6 Conclusions

The rod ejection releases to be used in the dose analysis have been calculated and provided for use in support of the RSG Program.

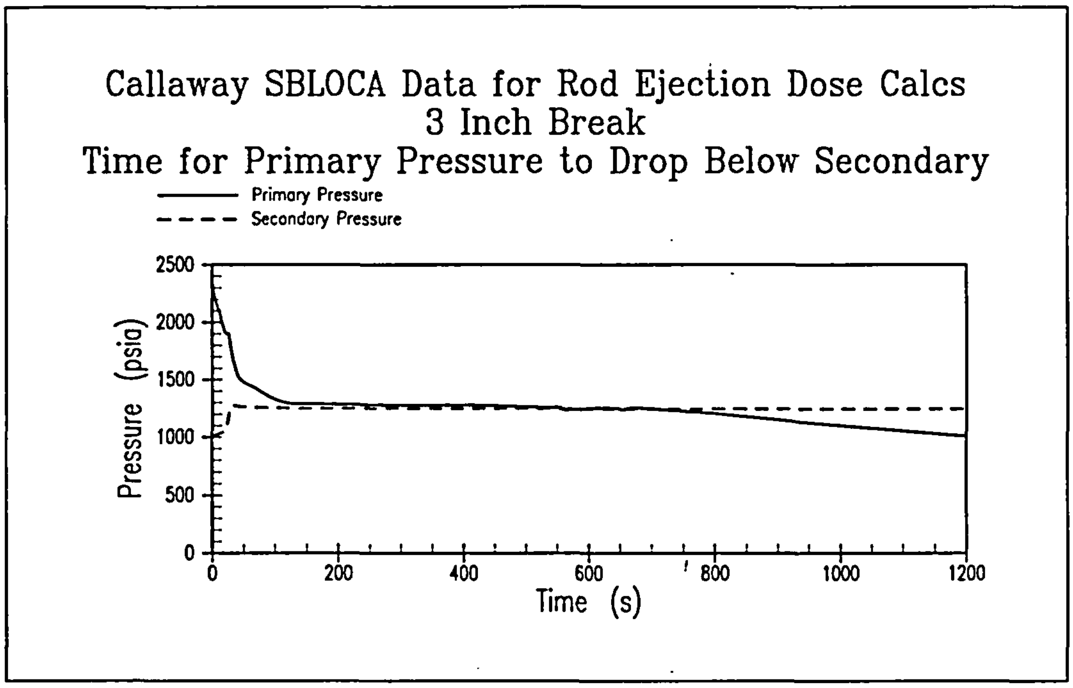


Figure 6.10-1 Pressure Transient

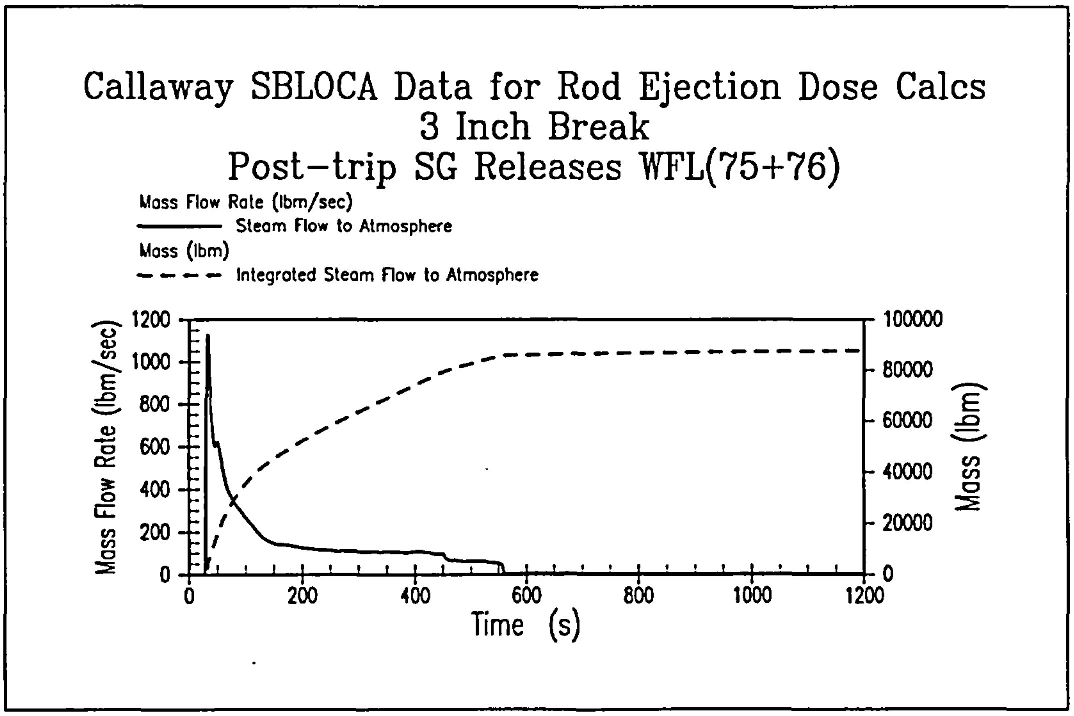


Figure 6.10-2 Steam Releases

7 NUCLEAR FUEL

7.1 CORE THERMAL-HYDRAULIC DESIGN

7.1.1 Introduction and Background

This section describes the core thermal-hydraulic analyses and evaluations performed in support of the Callaway Replacement Steam Generator (RSG) Program.

7.1.2 Input Parameters and Assumptions

The nuclear steam supply system (NSSS) design parameters for the Callaway RSG are specified in Table 2-1. Also, Table 7.1-1 summarizes the design parameters used in the core thermal-hydraulic analyses. As shown in Table 7.1-1, the RSG Program primary system parameters are mostly identical to the parameters used in the current cycle reload design for Callaway as reflected in the Final Safety Analysis Report (FSAR). Some Callaway thermal-hydraulic parameters were adjusted for the RSG Program due to the use of the Revised Thermal Design Procedure (RTDP) methodology (Reference 1), the VIPRE-W code (Reference 2), and the 8.6-percent bypass flow limit.

7.1.3 Description of Analyses and Evaluations

7.1.3.1 Calculational Methods

The core thermal-hydraulic design criteria for the RSG Program remain the same as those presented in the Callaway FSAR. The departure from nucleate boiling (DNB) design criterion is that there will be at least a 95-percent probability (at 95-percent confidence level) that DNB will not occur on the limiting fuel rods for any Conditions I and II event.

The RTDP, Reference 1, is employed as the primary design method to meet the DNB design basis for the fuel, replacing the previously used Improved Thermal Design Procedure (ITDP). With the RTDP methodology, uncertainties in plant operating parameters, nuclear and thermal parameters, fuel fabrication parameters, computer codes, and DNB correlation predictions are combined statistically to obtain the overall DNB uncertainty factor that is used to define the design limit DNB ratio (DNBR) that satisfies the DNB design criterion. Since the parameter uncertainties are considered in determining the RTDP design-limit DNBR, the plant safety analyses are performed using input parameters at their nominal values. The RTDP methodology is similar to ITDP, except that the DNB correlation statistics are combined with other uncertainties. Due to this combination of DNB correlation statistics, the RTDP methodology realizes more DNB margin than ITDP in which the DNB correlation limit was used when determining the design-limit DNBR.

The VIPRE-W code, Reference 2, is used as the primary DNB analysis code for the Callaway RSG Program replacing the previously used THINC-IV code. The VIPRE-W transient model is used for DNB analysis of such transients as Loss of Flow (LOF) and Locked Rotor (LR), corresponding to the use of the RETRAN code by the Westinghouse Transient Analysis group for their statepoints calculations (see Section 6.3 of this report).

As shown in Table 7.1-2, the uncertainties in the plant operating parameters (pressurizer pressure, primary coolant temperature, reactor power, and reactor coolant system (RCS) flow) were revised for Callaway for the RSG Program compared to the thermal-hydraulic design Analysis of Record. Based on those uncertainties, the RTDP design-limit DNBR values are calculated for the RSG Program (see Table 7.1-1). In addition, the safety analysis limit (SAL) DNBR values and the DNBR margin summary for the Callaway RTDP analyses are revised for the RSG Program in Table 7.1-3.

The standard thermal design procedure (STDP) is used for those analyses where RTDP is not applicable. The DNBR limit for STDP is the appropriate DNB correlation limit increased by sufficient margin to offset the applicable DNBR penalties. The use of STDP and the DNBR limits for the STDP analyses are unaffected by the RSG Program.

7.1.3.2 DNB Performance

To support the RSG Program in Callaway, DNBR re-analyses were performed using the RTDP methodology and the VIPRE-W code. Those analyses are addressed below.

7.1.3.2.1 Core and AO Limits

The existing core and axial offset (AO) limits were updated and confirmed to be applicable for the RSG Program using the RTDP methodology and the VIPRE-W code. The steady-state VIPRE-W model developed for the Callaway RSG Program consists of one-pass modeling, geometric input, rod input and power distribution, operating conditions, mixing and crossflow, and grid input. DNBR penalties were added to the DNBR margin rack-up in Table 7.1-3 in order to support the high-power positive runback AO limits used for the setpoint confirmation analysis and to cover the penalty associated with the new relaxed axial offset control (RAOC) analysis.

7.1.3.2.2 Dropped Rod

Dropped rod limit lines were calculated using the RTDP methodology and VIPRE-W code to address the acceptability of the plant's response to this accident. These limit lines are the loci of points that would result in the SAL DNBR being reached, based on a wide span of core conditions (inlet temperature, power, and pressure).

There is no explicit DNBR calculation for the Dropped Rod event. Calculation of the effects of the accident on the cycle-specific core conditions is checked each cycle by the nuclear designer using these limit lines. This demonstrates that the DNB design basis is met.

7.1.3.2.3 Static Rod Misalignment

The allowable $F_{\Delta H}$ limit for rod cluster control assembly (RCCA) misalignment was calculated using the RTDP methodology and VIPRE-W code. This is the value of $F_{\Delta H}$ at normal operating conditions that gives a minimum DNBR equal to the RTDP safety analysis DNBR limits. The acceptability of this limit is determined each cycle by the nuclear designer.

7.1.3.2.4 Hypothetical Steam Line Break at Hot Zero Power

The core thermal-hydraulic design methods and criteria for the analysis of the Hypothetical Steam Line Break event at hot zero power (HZP-SLB) for the RSG Program remain unchanged. The STDP is used, because this event is initiated from HZP condition for which the WRB-2 correlation and plant measurement uncertainties used for RTDP are not applicable. The analysis of the HZP-SLB is conservatively required to meet the Condition II DNB design criterion.

The time during the transient of the peak return to power was chosen for the DNBR analysis. The system statepoint conditions for RCS pressure and temperature, along with the power and peaking factors from the ANC model of a representative core, were analyzed to confirm that the Condition II DNB design criterion was met. Since the system pressure for the limiting statepoint was in the range of 500 to 1,000 psia, the W-3 correlation was used with the appropriate low pressure W-3 correlation limit of 1.45. The SAL DNBR is [ ]^{a,c} to account for the [ ]^{a,c} DNBR penalty for RCS flow anomaly. The results of the representative DNB analysis showed that there exists significant margin to the SAL DNBR for the limiting HZP-SLB statepoint selected for the RSG Program. It should be noted that a cycle-specific DNBR calculation of this event is normally performed for each fuel reload.

7.1.3.2.5 Full Power Steam Line Break

The core thermal-hydraulic design criterion for the analysis of the Steam Line Break Event At full Power (SLBAP) remains unchanged for the RSG Program. This full-power event is required to meet the Condition II DNB design criterion. The DNB analysis of SLBAP is based on the RTDP methodology, replacing the previously used ITDP.

The time of peak power during the transient was chosen for the DNBR analysis. The system statepoint conditions for RCS pressure and temperature, along with the power and peaking factors from the ANC model of a representative core, were analyzed to confirm that the Condition II DNB design criterion was met. The primary DNB correlation listed in Table 7.1-1 is applicable at the statepoint conditions. Based on the analysis of a representative core, the results for the limiting SLBAP statepoint for the RSG Program showed significant margin to the RTDP SAL DNBR listed in Table 7.1-3. Therefore, the DNB design criterion is met for this event. It should be noted that a cycle-specific DNBR calculation of this event is normally performed for each fuel reload.

7.1.3.2.6 Loss of Flow

The transient DNB analysis of LOF was performed for the Callaway RSG Program using the VIPRE-W code, the RTDP, and the WRB-2 DNBR correlation.

Based on the steady-state VIPRE-W model, the transient VIPRE-W model was developed for LOF analyses. The conduction rod model for VIPRE-W transient calculation was developed based on fuel rod and pellet geometrical parameters and the PAD4 fuel temperature data. The VIPRE-W fuel gap conductance at each axial level was determined to match the maximum fuel surface temperature from PAD4 calculations.

Transient DNB calculations were performed for each time step of the transient statepoints provided by the Transient Analysis (TA) group. The results of the limiting statepoint conditions show that there is significant margin to the SAL DNBRs (1.59/1.55 typical/thimble). Therefore, it is concluded that the Callaway RSG DNB design basis is met for the LOF events.

7.1.3.2.7 Locked Rotor

The LR VIPRE-W analysis of the Callaway RSG Program consists of 2 parts: 1) calculate the percentage of rods in DNB using the RTDP methodology and the WRB-2 correlation, and 2) calculate the peak cladding temperature (PCT) and local zirconium-water reaction.

An LR accident occurs when either the reactor coolant pump (RCP) rotor seizes or breaks. The reactor trips due to the rapidly-reduced flow in the coolant loop. The LR accident is classified as a Condition IV event due to its low probability of occurrence. Therefore, the percentage of fuel rods that fail during this transient is calculated for use in the dose calculations, and a hot rod analysis is performed to ensure that the PCT is below the limit of 2,700°F.

The transient VIPRE-W model for LR analysis is similar to that for LOF analysis. The conduction rod model was developed using PAD4 fuel temperature data as a comparison basis. The rods-in-DNB analysis of the limiting statepoints yields the minimum DNBR values violating the SAL DNBR. Using the generic rod power census normalized to a peak $F^N_{\Delta H}$ of 1.59, the percent of rods in DNB was determined as less than 5 percent.

For the LR PCT analysis, a time-dependent fuel rod gap forcing function was input to simulate the gap closure at the start of the transient such that the maximum gap coefficient value was approximately 10,000 Btu/(hr-ft^2°F). The Bishop-Sandberg-Tong film boiling heat transfer correlation was selected for post-critical-heat-flux (CHF) heat transfer. VIPRE-W calculations for the limiting statepoints showed that the peak cladding inner temperature was [ ]^{a,c}, much less than the 2,700°F limit. The amount of zirconium reacted is [ ]^{a,c}, much less than the 16-percent cladding oxidation limit during the LR transient.

7.1.3.3 Hydraulic Evaluation

The best-estimate RCS flows associated with different steam generator tube plugging levels are increased for the RSG Program. The fuel assembly lift forces that were used to determine the fuel assembly hold-down spring design (Section 5.3) were calculated using the maximum best-estimate RCS flow. The increased best-estimate RCS flows have a negligible impact on the fraction of the flow that bypasses the core through the thimble guide tubes. The core bypass flow limit with thimble plug removal remains unchanged for this program.

7.1.3.4 Fuel Temperatures and Rod Internal Pressures

The fuel temperatures and rod internal pressures have been updated using the U.S. Nuclear Regulatory Commission (NRC) approved Westinghouse PAD 4.0 fuel performance models (Reference 3). The integrated fuel burnable absorber (IFBA) and non-IFBA fuel temperatures and/or rod internal pressures were used as initial conditions for Loss-of-Coolant-Accident (LOCA) and Non-LOCA transients.

7.1.4 Conclusions

Core thermal-hydraulic analyses and evaluations were performed in support of the RSG Program for Callaway. The results showed that the core thermal-hydraulic design criteria are satisfied.

7.1.5 References

1. WCAP-11397-P-A, "Revised Thermal Design Procedure," A. J. Friedland and S. Ray, April 1989.

2. WCAP-14565-P-A, "VIPRE-01 Modeling and Qualification for Pressurized Water Reactor Non-LOCA Thermal-Hydraulic Safety Analysis," Y. X. Sung, P. Schueren, and A. Meliksetian, October 1999.

3. WCAP-15063-P-A, "Westinghouse Improved Performance Analysis and Design Model (PAD 4.0)," Revision 1 with Errata, J. P. Foster, and S. Sidener, July 2000.

Table 7.1-1 Thermal-Hydraulic Design Parameters for Callaway (Sheet 1 of 3)		
Thermal and Hydraulic Design Parameters	**Cycle 14 Reload Design Parameters**	**RSG Program Parameters**
Reactor Core Heat Output, MWt	3,565	3,565
Reactor Core Heat Output, 10^6 Btu/hr	12,164	12,164
Heat Generated in Fuel, %	97.4	97.4
Pressurizer Pressure, Nominal, psia	2,250	2,250
Core Pressure, Nominal, psia	2,280	2,274
$F_{\Delta H}$, Nuclear Enthalpy Rise Hot Channel Factor	1.59	1.59
Part Power Multiplier for $F_{\Delta H}$	[1+0.3(1-P)]	[1+0.3(1-P)]
Minimum DNBR at Nominal Conditions		
Typical Flow Channel	2.43(1)	2.45(2)
Thimble (Cold Wall) Flow Channel	2.29(1)	2.35(2)
Design Limit DNBR		
Typical Flow Channel	1.34(1, 3)	1.22(2, 4)
Thimble (Cold Wall) Flow Channel	1.33(1, 3)	1.21(2, 4)
Safety Analysis Limit DNBR		
Typical Flow Channel	1.69	1.59
Thimble (Cold Wall) Flow Channel	1.61	1.55
DNB Correlation	WRB-2	WRB-2

(1) Using THINC-IV

(2) Using VIPRE-W

(3) Using ITDP

(4) Using RTDP

Table 7.1-1 Thermal-Hydraulic Design Parameters for Callaway (Sheet 2 of 3) (cont.)		
HFP Nominal Coolant Conditions	Cycle 14 Reload Design Parameters	RSG Program Parameters
Vessel Minimum Measured Flowrate, MMF, (Including Bypass)	Based on Tin 560.1°F	Based on Tin 557.4°F
10^6 lbm/hr	141.9	142.4
gpm	382,640	382,630
Vessel Thermal Design Flowrate, TDF, (Including Bypass)	Based on Tin 560.1°F	Based on Tin 556.8°F
10^6 lbm/hr	138.8	139.4
gpm	374,400	374,400
Core Flowrate (Excluding Bypass, Based on Thermal Design Flow (TDF))	Based on 6.3% Bypass	Based on 8.6% Bypass
10^6 lbm/hr	130.0	127.4
gpm	350,810	342,200
Fuel Assembly Flow Area for Heat Transfer, ft^2	54.13	54.13
Core Inlet Mass Velocity, (Based on TDF)	Based on 6.3% Bypass	Based on 8.6% Bypass
10^6 lbm/hr-ft^2	2.40	2.35

Table 7.1-1 Thermal-Hydraulic Design Parameters for Callaway (Sheet 3 of 3) (cont.)		
Thermal & Hydraulic Design Parameters (Based on TDF)	**Cycle 14 Reload Design Parameters**	**RSG Program Parameters**
Nominal Vessel/Core Inlet Temperature, °F	556.8	556.8
Vessel Average Temperature, °F	588.4	588.4
Core Average Temperature, °F	592.2	593.1
Vessel Outlet Temperature, °F	620.0	620.0
Average Temperature Rise in Vessel, °F	63.2	63.2
Average Temperature Rise in Core, °F	66.9	68.4
Heat Transfer		
Active Heat Transfer Surface Area, ft^2	57,505	57,505
Average Heat Flux, Btu/hr-ft^2	211,529	206,085
Average Linear Power, kW/ft$^{(1)}$	5.69	5.69
Peak Linear Power for Normal Operation, kW/ft$^{(2)}$	14.22	14.23

(1) Based on densified active fuel length

(2) Based on F_Q of 2.50

Table 7.1-2 Initial Condition Uncertainties Used in DNBR Analyses		
Parameters	**Current Thermal-Hydraulic Analysis of Record**	**RSG Analyses**
Pressurizer Pressure	+ 27.0 psi	+ 30.0 psi
Temperature	+ 4.4°F	+ 3.0°F
Power	+2.0% Rated Thermal Power (RTP)	+2.0% RTP
RCS Flow	+ 2.2% flow	+ 2.1% Flow

Table 7.1-3 DNBR Margin Summary for Callaway RSG Program RTDP Analyses		
	Typical Cell	**Thimble Cell**
Design Limit DNBR	1.22	1.21
Safety Analysis Limit DNBR	1.59	1.55
DNBR Margin (Between Design and Safety Analysis Limit DNBR)	23.2%	21.9%
Total DNBR Penalties	14.8%	14.7%
Net DNBR Margin	8.4%	7.2%

7.2 FUEL CORE DESIGN

7.2.1 Introduction and Background

The credible and hypothetical steam line break transients were analyzed to incorporate the impact of the Callaway replacement steam generators (RSGs). The pressure, temperature, and reactivity results of that analysis (provided in Section 6.3) were used to verify that fuel limits are met for this transient. For the credible break for Callaway, no analysis is required because the accident is bounded by the analysis of the hypothetical steam line break. For the hypothetical break, this means verifying that the percentage of fuel failure meets the limit. In practice, this means confirming that the departure from nucleate boiling (DNB) design-basis limits are met. Additionally, the reactivity conditions of the transient are reviewed to ensure that the point kinetics model of the transient code is providing consistent results with the core design modeling codes.

The purpose of the full-power steam line break analysis is to demonstrate that the DNB design basis is met and that the peak linear heat generation rate (kW/ft) does not exceed a value which would cause fuel center line melting.

7.2.2 Input Parameters and Assumptions

The results of the transient analysis (presented in Section 6.3) were used as input to the core design evaluation. The 3-dimensional ANC core model (Reference 1) used was the Callaway Cycle 13 model. This core is representative in energy, burnable absorber usage, core leakage, and crud-induced power shift (CIPS) risk (such as, peaking factors or power distributions) of typical Callaway reload cores. Specifically, this Callaway design uses the VANTAGE + fuel assembly (VANTAGE 5 fuel with ZIRLO™ cladding). The region of fresh fuel, Region 15, consists of 96 assemblies with enriched annular axial blankets. Region 15A consists of 60 assemblies enriched to 4.00 w/o U^{235}. Region 15B consists of 36 assemblies enriched to 4.40 w/o U^{235}. The axial blanket design uses 2.6 w/o U^{235} enriched uranium dioxide pellets to replace the top six inches of the fuel stack. Both integral fuel burnable absorbers (IFBA) and wet annular burnable absorbers (WABA) are used for power distribution control. There are a total of 8,704 fresh IFBA rods in this core, with a linear B^{10} loading of 2.25 mg/in. There are a total of 288 fresh WABA rods in Cycle 13. The WABA rods have 132 inches of absorbing material centered axially about the core midplane at hot conditions. The Cycle 13 core was designed for a desired cycle length of 478 effective full-power days (EFPDs), or 21,205 MWD/MTU, at the end of full-power reactivity. The Cycle 13 low-leakage loading pattern was optimized for CIPS risk by flattening the radial power distribution across the core, and by keeping the nuclear enthalpy rise hot channel factor, $F_{\Delta H}$, at a relatively low value. $F_{\Delta H}$ was designed to 1.40 at the beginning of the cycle at hot zero power (HZP), with all rods out and no xenon conditions.

7.2.3 Description of Analyses and Evaluations

Hypothetical HZP Steam Line Break

The time in the transient of peak power was chosen for the DNB ratio (DNBR) analysis. The statepoint conditions of pressure, temperature, and reactivity along with limiting worst stuck rod assumptions were used with the 3-dimensional ANC core model to determine if the power generated by a detailed core model was consistent with that generated by the transient code point kinetics model.

The power from the 2 models was found to be sufficiently similar. The 3-dimensional ANC power was found to be 10.2 percent, resulting in a difference of 4.2 percent from the point kinetics model, which is within the 5.0-percent criteria required by Westinghouse methodology. This provided indirect confirmation of the adequacy of the reactivity coefficients used in the transient code point kinetics model. All methods and codes used for the steam line break core analysis are consistent with the current licensing basis for Callaway. It should be noted that this power consistency check is done with each fuel reload.

The power and peaking factors from this analysis were then used in the confirmation of the DNB design basis.

Full-Power Steam Line Break

The maximum power statepoint condition generated by the transient code point kinetics model was used in the core design 3-dimensional ANC core model.

The 3-dimensional ANC reactivity insertion was computed to be less than zero for the transient, thus validating assumptions used in the transient analysis. The peaking factors from this analysis were then used in the confirmation of the DNB design and peak linear heat generation rate (kW/ft) bases.

Design Evaluation - Physics Characteristics, Key Safety Parameters, Power Distributions, Peaking Factors

The physics characteristics, power distributions, and peaking factors of the Callaway core designs will not change substantially due to the RSGs, as changes in vessel flow rates have little influence on fuel management strategy and nuclear design calculations. The key safety parameters that are reviewed each cycle have not been significantly altered by the analyses described in Section 6.3. The key safety parameters assumed in these analyses will be reviewed each cycle after the RSGs have been installed.

7.2.4 Conclusions

The installation of the RSGs will have minimal impact on the fuel design for Callaway. The impact of the RSGs on peaking factors, rod worths, reactivity coefficients, shutdown margin, and kinetics parameters is well within normal cycle-to-cycle variation of these values and will be addressed on a cycle-specific basis consistent with the Reload Safety Evaluation Methodology.

No new Technical Specifications or modifications to existing Technical Specifications are anticipated to result from the RSGs due to nuclear-design-related aspects.

7.2.5 References

1. WCAP-11596-P-A (Proprietary) and WCAP-11597-A (Non-Proprietary), "Qualification of the PHOENIX-P/ANC Nuclear Design System for Pressurized Water Reactor Cores," T. Q. Nguyen, et al., June 1988.

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7.3 FUEL ROD DESIGN AND PERFORMANCE

7.3.1 Introduction and Background

The purpose of this evaluation was to review fuel rod design analyses to determine the acceptability of the Callaway Replacement Steam Generator (RSG) Program.

7.3.2 Input Parameters and Assumptions

The primary system parameters and fuel rod design assumptions used in the fuel rod design evaluation for the RSG Program are summarized in Table 7.3-1.

7.3.3 Description of Evaluations and Results

As shown in Table 7.3-1, the changed parameters for fuel rod design by the RSG Program are reactor coolant system (RCS) vessel average temperature and RCS core hot full-power (HFP) inlet temperature. These changed parameters are shown in bold in Table 7.3-1. All other parameters are the same as those assumed in the current fuel rod design (Cycle 13). Only the lower bound value of vessel average temperature and core inlet temperature are changed in a non-conservative direction for fuel rod performance. In the current fuel rod design analysis, upper bound values of the vessel average temperature and core inlet temperature ranges are conservatively assumed. Therefore, these parameters are also unaffected by the RSG Program. In view of fuel rod performance, all changed parameters for the RSG Program are bounded by or identical to the conditions assumed in the current design analysis (Cycle 13), performed with the Nuclear Regulatory Commission (NRC) approved PAD 3.4 fuel performance model and methodology (References 1 through 5).

7.3.4 Conclusions

All fuel rod design criteria have been evaluated for the RSG Program. All fuel rod design criteria are met. Furthermore, cycle-specific fuel rod design analyses are performed for each fuel region during each reload cycle to confirm that all fuel rod design criteria are satisfied for the operating conditions specified for each cycle of operation. These analyses support the Reload Safety Evaluation (RSE), which is provided for each cycle of operation.

7.3.5 References

1. WCAP-10851-P-A (Proprietary) and WCAP-11873-A (Non-Proprietary), "Improved Fuel Performance Models for Westinghouse Fuel Rod Design and Safety Evaluations," R. A. Weiner et al., August 1988.

2. WCAP-12610-P-A (Proprietary), "VANTAGE+ Fuel Assembly Reference Core Report," S. L. Davidson and D. L. Nuhfer, September 1994.

3. WCAP-12488-A, "Westinghouse Fuel Criteria Evaluation Process," S. L. Davidson, October 1994.

4. WCAP-10125-P-A (Proprietary), "Extended Burnup Evaluation of Westinghouse Fuel," S. L. Davidson et al., December 1985.

5. WCAP-13589-A, "Assessment of Clad Flattening and Densification Power Spike Factor Elimination in Westinghouse Nuclear Fuel," P. J. Kersting et al., March 1995.

Table 7.3-1 Summary of Callaway Replacement Steam Generator Parameters Analyzed in Fuel Rod Design Evaluation

Parameter	Current Condition	RSG Condition
Reactor Power (MWt)	3565	3565
Core Average Linear Power (kw/ft)	5.69	5.69
RCS Vessel Average Temperature (°F)	583.4 –588.4	570.7 – 588.4
RCS Core HFP Inlet Temperature (°F)	551.5 – 556.8 (557.5*)	538.2 – 556.8
Thermal Design Flow Rate (gpm / loop)	93,600	93,600
RCS Minimum Measured Flow Rate (gpm/loop)	95,660	95,660
Core Bypass Flow – Best Estimate (%)	4.8	4.8
System Pressure (psia)	2250	2250
FΔH Limit	1.65	1.65
Fuel Designs Considered	ZIRLO Cladding 17x17 V+ Annular Axial Blanket 1.5X IFBA	ZIRLO Cladding 17x17 V+ Annular Axial Blanket 1.5X IFBA

Note:
*557.5°F is used in the current fuel rod design, based upon the upper bound value of the vessel average temperature, 588.4°F