15.0 ACCIDENT ANALYSIS*

This chapter addresses the representative initiating events listed on pages 15-10, 15-11, and 15-12 of Regulatory Guide 1.70, Revision 3 as they apply to the Shearon Harris Nuclear Power Plant (SHNPP).

Certain items in the guide warrant comment, as follows:

Items 1.3 and 2.1 - There are no pressure regulators in the Nuclear Steam Supply System (NSSS) pressurized water reactor (PWR) design whose malfunction or failure could cause a steam flow transient.

Item 6.2 - No instrument lines from the reactor coolant system boundary in the NSSS PWR design penetrate the Containment. For the definition of the reactor coolant system boundary, refer to ANSI-N 18.2, "Nuclear Safety Criteria for the Design of Stationary PWR Plants," Section 5, 1973.

SHNPP has completed the transition from Westinghouse to AREVA fuel.

The only Westinghouse analyses in Chapter 15 are:

- Steam Generator Tube Rupture in Section 15.6.3 - Analysis of this event is not sensitive to specific fuel design, for example, the approach to minimum DNBR and peak linear heat generation rate limits are not close enough for values to be calculated for this event.

- Fission product inventory as source term for off-site and control room doses in Section 15.0.9, and the accident specific dose calculations.

- Anticipated Transient Without Scram in Section 15.8

Additionally, Duke Energy performs the turbine trip overpressure analysis in Section 15.2.3. AREVA continues to perform the Section 15.2.3 turbine trip DNB analysis. Other than these four exceptions, the accident analyses presented in Chapter 15 are supplied by AREVA.

Introduction of eight AREVA GAIA lead fuel assemblies is planned for Harris fuel reload cycle 20. This core change has been evaluated with respect to non-LOCA events, large break LOCA and small break LOCA. It has been determined that the existing analyses of record remain applicable and no new analyses are needed. As a result, the acceptance criteria continue to be met.

<u>Changes to Support ±3% MSSV Setpoint Tolerance</u>

In Reference 15.0-1, HNP requested a change to the Technical Specifications to increase the as-found lift setting tolerance for the mains steam line code safety valves from ±1% to ±3%. A consequence of the increased MSSV setpoint tolerance is a reduction of the credited AFW flow in the safety analyses at the lowest lifting MSSV setpoint plus tolerance. The change to AFW flow in certain accident and transient analyses is a reduction from 390 gpm to 374 gpm. To support the requested MSSV setpoint tolerance change, Reference 15.0-1 also requested a reduction to the pressurizer water level-high reactor trip setpoint from 92% to 87% of indicated

span, as well as a change to the maximum pressurizer water level Technical Specification limiting condition of operation (LCO) to 75% of indicated span. The 75% pressurizer water level LCO is tied to the initial level assumed in the Section 15.2.3 turbine trip overpressure evaluation performed by Duke Energy, which assess the impact of the requested changes to the transient and accident analyses in this chapter. The initial pressurizer levels assumed for all other events in this chapter are established in accordance with the applicable analysis methodology and are not associated with the pressurizer water level LCO.

Duke Energy has performed the overpressure evaluation of the Section 15.2.3 turbine trip event. The Section 15.2.3 turbine trip overpressure evaluation is performed using the RETRAN-3D computer code (Reference 15.0.11-23). The Section 15.2.3 turbine trip overpressure analyses consider MSSVs having at setpoint tolerance of +3%, a pressurizer water level - high reactor trip setpoint of 95% (87% requested Technical Specification value plus 8% allowance), and an initial pressurizer water level of 75% plus uncertainty. The initial conditions, inputs, assumptions, and boundary conditions applied in the overpressure analyses are described in detail in Section 15.2.3. The results from the Section 15.2.3 turbine trip overpressure analysis bound the overpressurization results for other ANS Condition II, III, and IV events.

Reference 15.0-1 examines the impact of the requested changes to the transient and accident analyses in this chapter and confirms that, with the exception of SBLOCA, all other events are either 1) bounded by current analysis of record, 2) bounded by the 15.2.3 turbine trip event regarding the overpressure, or 3) the fuel certerline melt and/or MDNBR limits are not affected by the requested changes. As a result, the remaining transient and accident analyses in this chapter are not revised to include the changes to the MSSV tolerance, pressurizer level trip setpoint, and AFW flow rate. Where applicable, these analyses continue to assume an MSSV setpoint tolerance of ±1% and an AFW flow rate of 390 gpm. For SBLOCA, the MSSV setpoint tolerance increase and AFW flow rate reduction result in an estimated peak cladding temperature penalty, which is discussed in Section 15.6.5.3.4.

15.0.1 CLASSIFICATION OF PLANT CONDITIONS**

Since 1970 the American Nuclear Society (ANS) classification of plant conditions has been used which divides plant conditions into four categories in accordance with anticipated frequency of occurrence and potential radiological consequences to the public. The four categories are as follows:

1) Condition I: Normal Operation and Operational Transients.

2) Condition II: Faults of Moderate Frequency.

3) Condition III: Infrequent Faults.

4) Condition IV: Limiting Faults.

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

15.0.1.1 ANS Condition I - Normal Operation and Operational Transients

ANS Condition I occurrences are those which are expected frequently or regularly in the course of power operation, refueling, maintenance, or maneuvering of the plant. As such, ANS Condition I occurrences are accommodated with margin between any plant parameter and the value of that parameter which would require either automatic or manual protective action. Since ANS Condition I occurrences occur frequently or regularly, they must be considered from the point of view of affecting the consequences of fault conditions (ANS Conditions II, III and IV). In this regard, analysis of each fault condition described is generally based on a conservative set of initial conditions corresponding to adverse conditions which can occur during ANS Condition I operation.

A typical list of ANS Condition I events is listed below:

1) Steady state and shutdown operations

    a. Power operation (>5 to 100 percent of rated thermal power).

    b. Startup ($K_{eff} \geq 0.99$, ≤ 5 percent of rated thermal power).

    c. Hot standby (subcritical, Residual Heat Removal System isolated).

    d. Hot shutdown (subcritical, Residual Heat Removal System in operation).

    e. Cold shutdown (subcritical, Residual Heat Removal System in operation).

    f. Refueling.

2) Operation with permissible deviations

    Various deviations, which may occur during continued operation as permitted by the plant Technical Specifications, must be considered in conjunction with other operational modes. These include:

    a) Operation with components or systems out of service (such as power operation with a reactor coolant pump out of service).

    b) Leakage from fuel with clad defects.

    c) Radioactivity in the reactor coolant

        1) Fission products.

        2) Corrosion products.

        3) Tritium.

    d) Operation with steam generator leaks up to the maximum allowed by the Technical Specifications.

e) Testing as allowed by the Technical Specifications.

4) Operational transients

a) Plant heatup and cooldown (up to 100F/hour for the Reactor Coolant System; 200F/hour for the pressurizer during cooldown and 100F/hour for the pressurizer during heatup).

b) Step load changes (up to $\pm$10 percent).

c) Ramp load changes (up to 5 percent/minute).

d) Load rejection up to and including design full load rejection transient.

15.0.1.2 ANS Condition II - Faults of Moderate Frequency

ANS Condition II occurrences are those which are expected to occur, in general, no more than once per year. These faults, at worst, result in a reactor trip with the plant being capable of returning to operation. By definition, these faults (or events) do not cause a more serious fault, i.e., ANS Condition III or IV events. In addition, ANS Condition II events are not expected to result in fuel rod failures or Reactor Coolant System or secondary system overpressure.

Table 15.0.1-1 lists the accident category used for each of the Chapter 15 events

15.0.1.3 ANS Condition III - Infrequent Faults

By definition, ANS Condition III occurrences are faults which may occur very infrequently during the life of the plant. They will be accommodated with the failure of only a small fraction of the fuel rods although sufficient fuel damage might occur to preclude resumption of the operation for a considerable outage time. An ANS Condition III fault will not, by itself, generate an ANS Condition IV fault or result in a consequential loss of function of the Reactor Coolant System or containment barriers. Table 15.0.1-1 lists the accident category used for each of the Chapter 15 events.

15.0.1.4 ANS Condition IV - Limiting Faults

ANS Condition IV occurrences are faults which are not expected to take place, but are postulated because their consequences would include the potential of the release of significant amounts of radioactive material. They are the most drastic events which must be designed against and represent limiting design cases.

ANS Condition IV faults are not to cause a fission product release to the environment resulting in an undue risk to public health and safety in excess of applicable limits. A single ANS Condition IV fault is not to cause a consequential loss of required functions of systems needed to cope with the fault including those of the Emergency Core Cooling System and the Containment. Table 15.0.1-1 lists the accident category used for each of the Chapter 15 events.

15.0.2 OPTIMIZATION OF CONTROL SYSTEMS

A control system setpoint study is performed in order to simulate performance of the Reactor Control and Protection Systems. In this study, emphasis is placed on the development of a control system which will automatically maintain prescribed conditions in the plant even under a conservative set of reactivity parameters with respect to both system stability and transient performance.

Nominal protection system setpoints on which the accident analysis is based are also used in the controls system setpoint study. Instrumentation errors are calculated consistent with the method used in the accident analysis. These errors are applied in an adverse direction with respect to maintaining system stability and transient performance. The accident analysis and setpoint study combine to show that the plant can be operated and meet both safety and operability requirements.

For each mode of plant operation, a group of optimum controller setpoints is determined. In areas where the resultant setpoints are different, compromises based on the optimum overall performance are made and verified. A consistent set of control system parameters is derived satisfying plant operational requirements throughout the core life and for various levels of power operation.

The study contains an analysis of the following control systems: rod cluster control assembly, steam dump, steam generator level, pressurizer pressure and pressurizer level.

15.0.3 PLANT CHARACTERISTICS AND INITIAL CONDITIONS ASSUMED IN THE ACCIDENT ANALYSES*

15.0.3.1 Design Plant Conditions.

AREVA Analyses - The accidents presented in Chapter 15 are analyzed at limiting conditions consistent with the Technical Specifications. The initial core power level for Chapter 15 analyses is assumed to be 2958 MWt. Pre MUR methodology presented the Rated Thermal Power as 2900 MWt with 2% added for uncertainty (2958 MWt). Under MUR conditions, this same value, 2958 MWt, corresponds to a Rated Thermal Power of 2948 MWt with 0.34% uncertainty. Going forward the FSAR 15.0 sections will identify the power condition as 2958 MW (rated + 0.34%). The values of pertinent plant parameters utilized in the accident analyses are given in Sections 15.1 through 15.7.

15.0.3.2 Initial Conditions

AREVA Analyses - The initial conditions assumed in the accident analyses are given in Sections 15.1 through 15.7. The following measurement uncertainties were considered in the AREVA analyses:

1) Core Power	$\pm$ 0.34% at full power
2) Pressurizer Pressure	-50 psi (for calculating DNBR only)
3) RCS Flow	Technical Specification minimum

4) RCS Average Temperature +6.0/-6.8°F

The component response times, setpoints, and capacities supported in the accident analyses are presented in Table 15.0.3-5. The vessel average temperature assumed in the accident analyses are presented in Sections 15.1 through 15.6.

15.0.3.3 Power Distribution

The transient response of the reactor system is dependent on the initial power distribution. The nuclear design of the reactor core minimizes adverse power distribution through the placement of control rods. Power distribution may be characterized by the radial peaking factor ($F_{\Delta H}$) and the total peaking factor (F_Q). The power distribution factor limits are given in the Technical Specifications.

AREVA Analyses - The $F_{\Delta H}$ and F_Q factors used in the AREVA analyses are presented in Table 15.0.3-6. Bounding axial power distributions generated from a three-dimensional core physics model were used in the AREVA analyses.

For centerline melt criteria, the bounding F_Q for a given event is used.

15.0.4 REACTIVITY COEFFICIENTS ASSUMED IN THE ACCIDENT ANALYSES*

AREVA Analyses - The reactivity coefficients assumed in the accident analyses are given in Sections 15.1 through 15.6.

15.0.5 ROD CLUSTER CONTROL ASSEMBLY INSERTION CHARACTERISTICS*

The negative reactivity insertion following a reactor trip is a function of the acceleration of the RCCA and the variation in rod worth as a function of rod position. With respect to accident analyses, the critical parameter is the time of insertion up to the dashpot entry, or approximately 85 percent of the rod cluster travel. For accident analyses, the insertion time to dashpot entry is conservatively taken as 2.7 seconds.

AREVA Analysis - The rod cluster control assembly position versus time assumed for AREVA fuel in accident analyses is shown in Figure 15.0.5-4.

Figure 15.0.5-5 shows the fraction of total negative reactivity insertion versus normalized rod position for a core where the axial power distribution is skewed to the lower region of the core. An axial distribution which is skewed to the lower region of the core can arise from an unbalanced xenon distribution. This curve is used to compute the negative reactivity insertion versus time following a reactor trip. This negative reactivity insertion curve is input to all point kinetics core models used in transient analyses. The bottom skewed power distribution itself is not an input into the point kinetics core model.

There is inherent conservatism in the use of Figure 15.0.5-5 in that it is based on a skewed flux distribution which would exist relatively infrequently. For cases other than those associated with unbalanced xenon distributions, significantly greater negative reactivity would have been inserted due to the more favorable axial distribution existing prior to the trip.

The normalized rod cluster control assembly negative reactivity insertion versus time is shown in Figure 15.0.5-6. The curve shown in this figure was obtained from Figures 15.0.5-4 and 15.0.5-5. The insertion worth used in the analysis of each event is indicated in the event description. The insertion worth has been decreased to account for the most reactive rod stuck out. For Figures 15.0.5-4 and 15.0.5-5, the rod cluster control assembly drop time is normalized to 2.7 seconds, in order to provide a bounding analysis for all rod cluster control assemblies to be used in the SHNPP cores, as previously stated.

15.0.6 TRIP POINTS AND TIME DELAYS TO TRIP ASSUMED IN ACCIDENT ANALYSES*

A reactor trip signal acts to open two trip breakers connected in series feeding power to the control rod drive mechanisms. The loss of power to the mechanism coils causes the mechanisms to release the rod cluster control assemblies, which then fall by gravity into the core. There are various instrumentation delays associated with each trip function, including delays in signal actuation, in opening the trip breakers, and in the release of the rods by the mechanisms. The total delay to trip is the difference between the time that trip conditions are reached and the time the rods are free and begin to fall.

Limiting trip setpoints assumed by AREVA in accident analyses and the time delay assumed for each trip function are given in Table 15.0.6-2. The Section 15.2.3 turbine trip overpressure analyses performed by Duke Energy also use the trip setpoints and time delays in Table 15.0.6-2

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

15.0.7 INSTRUMENTATION DRIFT AND CALORIMETRIC ERRORS - POWER* RANGE NEUTRON FLUX

The total allowance for instrument uncertainty for the power range high neutron flux reactor trip setpoint is presented in Technical Specifications:

108% of rated thermal power as nominal setpoint

+ (5.83% of span) (120% of rated thermal power per 100% channel span)

---------------- as total allowance for instrument uncertainty

115% of rated thermal power as value used in accident analysis

The total allowance for instrument channel uncertainty includes calorimetric error in the determination of core power and drift. Additional details are presented in Reference 7.2.1-6.

15.0.8 PLANT SYSTEMS AND COMPONENTS AVAILABLE FOR MITIGATION OF ACCIDENT EFFECTS*

The NSSS is protected by design from the possible effects of natural phenomena, postulated environmental conditions and dynamic effects of the postulated accidents. In addition, the design incorporates features which minimize the probability and effects of fires and explosions. The incorporation of these features in the NSSS, coupled with the reliability of the design, ensures that the normally operating systems and components listed in Table 15.0.8-1 will be available for mitigation of the events discussed in Chapter 15. In determining which systems are necessary to mitigate the effects of these postulated events, the classification system of ANSI-N18.2-1973 is utilized. The design of "systems important to safety" (including protection systems) is consistent with IEEE Standard 379-1972 and Regulatory Guide 1.53 in the application of the single failure criterion.

In the analysis of the Chapter 15 events, control system action is considered only if that action results in more severe accident results. No credit is taken for control system operation if that operation mitigates the results of an accident. For some accidents, the analysis is performed both with and without control system operation to determine the worst case.

For a dropped RCCA event in the automatic rod control mode, the Rod Control System detects the drop in power and initiates control bank withdrawal. Power overshoot may occur due to this action by the automatic rod controller after which the control system will insert the control bank to restore nominal power. Following a dropped rod event in manual rod control, the plant will establish a new equilibrium condition. The equilibrium process without control system interaction is monotonic, thus removing power overshoot as a concern and establishing the automatic rod control mode of operation as the limiting case.

Operation of the pressurizer heaters as a result of normal control action or a single failure will, depending on the transient, be less conservative or have negligible effects. See Section 5.4.10 for a discussion of the pressurizer heaters. For those events analyzed by AREVA, see Sections 15.1 through 15.6 for identification of whether the pressurizer heaters were assumed available.

The principal effect of the pressurizer heaters on transients would be to maintain higher RCS pressures.

During cooldown transients or DNB limited transients higher RCS pressure is less conservative. It could be conservative for these types of transients to not assume the pressurizer heaters are working.

The pressurizer heaters will not cause a more severe pressure transient for those accidents which have the potential to overpressurize the RCS. For overpower transients such as the loss of RCS flow, rod ejection or loss of load, the pressurizer heaters would have no effect on the pressure response since the pressure response during these types of transients is characterized by a rapid increase and then a decrease in pressurizer pressure. This increase and decrease of pressure occurs over a period less than 20 seconds, which is much faster than the response of the heaters.

For pressurizer heater effects during accident analyses of steam generator tube rupture, refer to Section 15.6.3.

15.0.9 FISSION PRODUCT INVENTORIES AND OTHER ISOTOPE SPECIFIC PARAMETERS

15.0.9.1 Fission Product Inventories

The calculation of the core fission product inventory is based on a safety analysis core power level of 2958 MWt. The fission product inventories are calculated using the ORIGEN Code (Reference 15.0.9-2) using the data library based on ENDF/B-IV and ENDF/B-V (Reference 15.0.9-3). These inventories are given in Table 15.0.9-1. The isotopes included in Table 15.0.9-1 are the isotopes controlling from considerations of inhalation dose and from direct dose due to immersion. Design basis primary and secondary coolant activities are shown in Tables 15.0.9-2 and 15.0.9-7. These activities are used to evaluate potential releases for events that do not directly result in fuel cladding failure or fuel centerline melt conditions.

The conservative iodine spiking model used for those events which must consider spiking calculates the equilibrium iodine appearance rates based on a nominal letdown flow of 120 gpm with perfect cleanup. The nominal RCS volume used in this equilibrium Iodine spiking model is provided in Tables 15.1.5-5, 15.2.6-5, 15.3.3-6, and 15.6.3-6 for the Steam Line break, Loss of AC Power, Locked Rotor and Steam Generator Tube Rupture events, respectively. The nominal 120 gpm letdown flow with perfect cleanup is increased by 10 percent to 132 gpm (to cover uncertainty), by 10 gpm for identified leakage from the RCS, by 1 gpm for unidentified leakage from the RCS, and by 31 gpm controlled leakage. The effective letdown flow is therefore 174 gpm. Inclusion of the controlled leakage in the effective removal flow is conservative, since this flow does not remove activity from the RCS.

The iodine spike appearance rates of iodines in the Reactor Coolant System assumed in the analysis are shown in Table 15.0.9-6.

15.0.9.2 Dose Conversion Factors

The total effective dose equivalent (TEDE) dose is equivalent to the committed effective dose equivalent (CEDE) or inhalation dose plus the acute dose (EDE) dose for the duration of exposure to the cloud. The dose conversion factors (DCFs) used in determining the CEDE dose are from Reference 15.0.9-9 and are given in Table 15.0.9-3. The DCFs used in determining the EDE dose are from Reference 15.0.9-10 and are given in Table 15.0.9-4. These are the DCFs suggested by Regulatory Guide 1.183 (Reference 15.0.9-8).

15.0.9.3 Nuclide Decay Constants

Decay constants for each nuclide are from Reference 15.0.9-11 and provided in Table 15.0.9-5.

15.0.9.4 Fuel Handling Accident Fission Product Source Term

The fission product inventory for the PWR and BWR fuel used in the Fuel Handling Accident Analysis is shown in Tables 15.7.4-1 and 15.7.4-3. For the FHA analysis in the Fuel Handling Building, the activity shown in Table 15.7.4-1 is the combined total activity of PWR fuel damaged plus the activity of the BWR fuel that is also assumed to be damaged. The gap release fractions used in the FHA analysis are listed in Tables 15.7.4-1 and 15.7.4-3.

15.0.10 RESIDUAL DECAY HEAT*

15.0.10.1 Small Break LOCA Decay Heat

The decay heat for the Small Break LOCA (SBLOCA) analysis is based upon 1.2 times the draft 1971 ANS standard fission product decay heat (Reference 15.0.10-6) as required by 10CFR Part 50.46 (Reference 15.0.10-7) and Appendix K (Reference 15.0.10-8). Refer to Reference 15.0.11-20 (SBLOCA Methodology)

15.0.10.2 Large Break LOCA Decay Heat

The Large Break LOCA (LBLOCA) decay heat is based on 1979 ANS Standard (Reference 15.0.10-3). The HNP LBLOCA Methodology (Reference 15.6.5-50) contains description on the application of the ANS standard.

15.0.10.3 Non-LOCA Decay Heat

Duke Energy Analyses - Decay heat for the turbine trip overpressure evaluation performed by Duke Energy in Section 15.2.3 is based on the 1979 ANS Standard (Reference 15.0.10-3). The turbine trip DNB analysis in Section 15.2.3 is performed by AREVA using the decay heat input described below.

AREVA Analyses - Decay heat for the non-LOCA analyses is based upon the 1973 ANS decay heat curve (Reference 15.0.10-4). Due to the rapidity of the majority of the non-LOCA transients analyzed, the decay heat has a negligible impact on the analysis results. For the longer non-LOCA transients, such as the Loss of Feedwater and the Feedwater Line Break events, a high value for the decay heat versus time will be conservative. The 1973 ANS decay heat curve provides a higher decay heat than the 1979 ANS decay heat curve, and is therefore conservative (Reference 15.0.10-5).

15.0.11 COMPUTER CODES UTILIZED*

15.0.11.1 Deleted by Amendment No. 48.

15.0.11.1.1 Deleted by Amendment No. 48.

15.0.11.1.2 Deleted by Amendment No. 48.

15.0.11.1.3 Deleted by Amendment No. 48.

15.0.11.1.4 Deleted by Amendment No. 48.

15.0.11.1.5 Deleted by Amendment No. 48.

15.0.11.1.6 Deleted by Amendment No. 48.

15.0.11.2 Computer Codes Utilized in AREVA Analyses

Summaries of some of the principal computer codes used in non-LOCA, Main Steam Line Break (MSLB), Large Break LOCA (LBLOCA), and Neutronics analyses performed by AREVA are given below.

15.0.11.2.1 Non-LOCA Transients and MSLB

The codes used for non-LOCA transients and for the Main Steam Line Break event are described below.

ANF-RELAP and S-RELAP5 - ANF-RELAP AND S-RELAP5 are the PWR system transient analysis codes used for simulation of the system response for the non-LOCA transients and for the Main Steam Line Break event. Control volumes and junctions are defined which describe all major components in the primary and secondary systems which are important for the event being analyzed. The ANF-RELAP and S-RELAP5 hydrodynamic model is a one-dimensional, transient, two-fluid model for flow of a two-phase steam-water mixture. ANF-RELAP and S-RELAP5 use a six equation model for the hydraulic solutions. These equations include two phasic continuity equations, two phasic momentum equations, and two phasic internal energy equations. The six equation model also allows both non-homogeneous and non-equilibrium situations encountered in reactor problems to be modeled. They have been generically approved by the NRC for use in non-LOCA transient analyses including the steamline break analysis.

The methodology for using the ANF-RELAP code is described in References 15.0.11-6, 15.0.11-7, and 15.0.11-13. S-RELAP5 was used to perform safety analysis for the uncontrolled rod cluster control assembly bank withdrawal at power (FSAR 15.4.2). The methodology for using the S-RELAP5 code is described in Reference 15.0.11-22.

XCOBRA-IIIC - XCOBRA-IIIC calculates the flow and enthalpy distribution within assemblies and sub-channels, and is used to calculate DNBRs based on these parameters. Core inlet boundary conditions for various plant transients and steamline break transients are input to XCOBRA-IIIC. This code was generically approved by the NRC in March 1985.

XCOBRA-IIIC is further discussed in References 15.0.11-8 and 15.0.11-9.

15.0.11.2.2 Large break LOCA analyses

AREVA uses a series of codes for LBLOCA analyses. All of these codes have been reviewed and generically accepted by the NRC as part of the AREVA LBLOCA evaluation models to be in compliance with the requirements of 10 CFR 50.46 and Appendix K. The AREVA PWR ECCS evaluation model is described in Reference 15.0.11-11.

The codes used for LBLOCA analyses are described below.

RODEX3A - RODEX3A calculates fuel, cladding and fuel-cladding gap properties as a function of exposure. The code models a fuel rod over the power history. The code is used to determine fuel rod temperature and gap conditions in the LBLOCA analysis.

S-RELAP5- S-RELAP5 is used for calculation of the system response. The field equations are basically the same form as RELAP5/MOD2 with the addition of full two-dimensional momentum equations. This two-dimensional capability is only applied within the reactor vessel in the Realistic Large Break LOCA methodology, but can be applied anywhere in the reactor coolant system through input. Initial fuel conditions are supplied by the realistic fuel performance code, RODEX3A. Capability for a concurrent calculation of containment backpressure based on the ICECON code was added.

15.0.11.2.3 Small break LOCA

The codes used to perform the SBLOCA analysis are described below:

RODEX2 - The RODEX2 code is utilized to determine the initial fuel stored energy and gap conditions for the initialization of the system blowdown and hot rod response calculations.

S-RELAP5 - The AREVA version of S-RELAP5 is used to model the primary system and secondary side of the steam generators during the entire transient. The governing conservation equations for mass, energy, and momentum transfer are used along with appropriate correlations consistent with Appendix K of 10 CFR 50.

The methodologies for applying the SBLOCA codes described above can be found in Reference 15.0.11-20.

15.0.11.2.4 Neutronics analyses

The codes used for neutronics analyses are described below:

MICBURN-3 - MICBURN-3 is a multi-group one-dimensional transmission probability code which calculates the microscopic burnup in an absorber rod containing initially homogeneously distributed gadolinia and generates effective cross sections as a function of the gadolinia number density to be used in a CASMO assembly depletion.

MICBURN-3 is further discussed in Reference 15.0.11-16.

CASMO-3G - CASMO-3G is a multi-group two-dimensional transmission probability code for burnup calculations on PWR or BWR assemblies or pin cells. Micro-group calculations (in 70 or 40 groups) are made to obtain detailed neutron energy spectra for energy consolidation and spatial homogenization for each pin type in an assembly. A two-dimensional seven energy group transport theory calculation is then made over the entire assembly with each pin cell explicitly modeled. The fluxes obtained from this calculation are used to generate both microscopic and macroscopic two-group cross sections.

CASMO-3G is further discussed in Reference 15.0.11-17.

XTG - XTG is a three-dimensional, coarse-mesh, simulated two-group diffusion theory reactor code. The large mesh spacing enables full-core calculations without prohibitive computer cost. Inner iterations are performed on the fast group flux. The thermal flux and nodal source term are updated after each outer iteration. After a specified number of outer iterations, the cross sections are updated to reflect power dependence on thermal-hydraulic, xenon, and Doppler feedback; this method of solution results in rapid convergence. For fuel management calculations, XTG has the following capabilities:

1) It calculates time- and power-dependent xenon concentrations, permitting load-follow and xenon oscillation calculations.

2) It determines margins to thermal operating limits.

3) It performs internal calculations of thermal-hydraulic effects, allowing moderator density changes as a function of axial core position.

4) It treats control rods and burnable poisons, including the capability of simulating reactor operation all the way from cold to hot, full-power conditions. Shutdown margin and excess reactivity calculations also can be routinely performed.

5) It performs calculations in 1/4, 1/2, and full core geometry, and it simulates fuel shuffling, insertion, and discharge.

XTG is further discussed in Reference 15.0.11-19.

PRISM - PRISM calculates core-wide power distributions in three dimensions. PRISM evaluates number densities and burnup of key isotopes on a nodal basis using microscopic cross sections. The nodal expansion method is utilized in solution of the two group diffusion theory representation of the reactor core. Pin power distributions are generated by PRISM using a pin power reconstruction technique.

PRISM is further discussed in Reference 15.0.11-12.

15.0.11.3 <u>Computer Codes Utilized in Duke Energy Analyses</u>

Duke Energy performs the overpressure evaluation of the Section 15.2.3 turbine trip event. AREVA continues to perform the Section 15.2.3 turbine trip DNB analysis. The Section 15.2.3 turbine trip overpressure evaluation is performed using the RETRAN-3D computer code (Reference 15.0.11-23).

RETRAN 3-D - RETRAN 3-D is a flexible, general-purpose, thermal-hydraulic computer code that can be used to represent light-water reactor systems. The code solves the governing conservation equations of mass, energy, and momentum, as applied to a network of fluid volumes and flow junctions. Conductive heat structures can be modeled, including the fuel elements in the reactor core. Changes in reactor power from nuetron kinetics and decay heat are calculated to occur with time. The name, RETRAN-3D, refers to ability of the code to perform three-dimensional neutronic calculations in the core, as opposed to three-dimensional fluid dynamic capability. RETRAN-3D was approved by the NRC staff in Reference 15.0.11-24 with 45 limitations and conditions of use.

15.0.12 LONG TERM EFFECTS AND EVENTS FOLLOWING CHAPTER 15 ACCIDENTS

For most of the events analyzed in Chapter 15, the plant will be in a safe and stable hot standby condition following the automatic actuation of reactor trip. This condition will in fact be similar to plant conditions following any normal, orderly shutdown of the reactor. At this point, the actions taken by the operator would be no different than normal operating procedures. The exact actions taken, and the time these actions would occur, will depend on what systems are available (e.g. steam dump system, main feedwater system, etc.) and the plans for further plant operation. As a minimum, to maintain the hot stabilized condition, decay heat must be removed via the steam generators. The main feedwater system and the steam dump or atmospheric relief system could be used for this purpose. Alternatively, the auxiliary feedwater system and the steam generator safety valves may be used, both of which are safety grade systems. Although the auxiliary feed system may be started manually, it will be automatically actuated if needed by one of the signals shown on Figure 7.2.1-1 sheet 14, such as low-low steam generator water level. If hot standby conditions are maintained for an extended period of time, operator action may be required to transfer to the auxiliary feedwater source. The time at which such action is required will be sufficiently long after initiation of the event to permit operator action. Also, if the hot standby condition is maintained for an extended period of time (greater than approximately 18 hours), operator action may be required to add boric acid via the CVCS to compensate for xenon decay and maintain shutdown margin. Again, the actions taken by the operator would be no different than during normal plant shutdown.

Many Chapter 15 events result in a stable condition being reached automatically following a reactor trip and only actions typical of normal operation are required from the operator. For several events involving breaks in the reactor coolant system or secondary system piping, additional requirements for operator action can be identified (see Sections 15.1.5.2 and 15.2.8.2). (Additional information about the impact of equipment failures or erroneous operator actions may be found in WCAP-9691 "NUREG-0578.2.1.9.C, Transient and Accident Analysis".)

15.0.13 SINGLE FAILURES ASSUMED IN THE ANALYSES OF CHAPTER 15 ACCIDENTS

All of the transients analyzed in Chapter 15 were analyzed assuming the most limiting single failure (e.g., loss of one protection signal of safety injection (SI) train failure). Table 15.0.13-1 lists the limiting single failures for each American Nuclear Society (ANS) Conditions II event (faults of moderate frequency). Table 15.0.13-2 lists the limiting single failures for Non-Condition II events.

The single failures listed in Tables 15.0.13-1 and 15.0.13-2 are the limiting failures for their respective event. Though operator errors do not appear to be explicitly addressed, they have been considered and found to be bounded by the most limiting single failures listed.

1) Single Failures Assumed for Accidents of Moderate Frequency - The incidents of moderate frequency were analyzed consistent with the acceptance criteria given in the Standard Review Plan (SRP) concerning peak pressure (less than 110 percent of design), fuel integrity (departure from nucleate boiling ratio (DNBR) limit), generation of more serious plant conditions, and single active failures.

Pressure transients for each event are provided in the FSAR and demonstrate that the pressure remains below 110 percent of design pressure. Fuel cladding integrity is demonstrated for each case by showing that the DNBR remains above the limit value. This is discussed in the results and conclusions sections for each event.

For each transient, its associated worst single failure within the protection system assumed in the FSAR analyses is given in Table 15.0.13-1. The protection system is defined as those safety functions required to mitigate the consequences of the event. This includes not only the Solid State Protection System (SSPS), but also the Engineered Safety Features (ESF) and pressurizer and steam generator safety valves.

These single failures were selected based on the requirements of 10 CFR 50 Appendix A, the SRP, and Reg. Guide 1.53 (which addresses IEEE-279 and IEEE-379). A single failure is ". . . an occurrence which results in the loss of capability of a component to perform its intended safety functions." (10 CFR 50 Appendix A). The single failure criterion states that a "single failure within the protection system shall not prevent proper protective action at the system level when required" (IEEE-279).

The single failures which are considered are active failures, consistent with the SRP acceptance criteria. Failures in the protection system which are not required to mitigate the consequences of an accident are not considered. These are failures of systems which are not challenged during the transient and are not active failures. Such failures are independent failures and are, therefore, not within the scope of the evaluation.

For each event listed in Table 15.0.13-1, a brief discussion of the assumed single failure is provided in this discussion. The purpose of the discussions is to justify that the single failure assumed is indeed the worst single failure. These failures are failures at the system level and consider the failure of a protective function. The cause or mechanical nature of the failure which causes the system failure is not discussed, since these are addressed in the failure modes and effects analyses (FMEA's) of the SSPS and ESF and in Chapters 6, 7, and 9 of the FSAR. Therefore, further detail beyond the systems level single failure of loss of one protection train is not provided.

The steam generator safety valves may be required to prevent a pressurization of the secondary system. Except where it is already stated in the FSAR, the steam generator valves are not challenged or required to mitigate the consequences of the event. Failures of these valves are not considered since they are not active failures. These independent failures are not applicable. Therefore, failure of these valves is not discussed below unless they are actuated as stated in the FSAR.

Finally, a loss of offsite power is not considered as a single failure for these events. The SRP does not require consideration of a loss of offsite power for the accidents listed in Table 15.0.13-1 (loss of AC power, FSAR Section 15.2.6, is by definition an exception). Furthermore, no

single active failure will cause a loss of offsite power to the emergency buses. Therefore, consideration of this failure is not applicable.

a) Feedwater Temperature Reduction - As stated in FSAR Section 15.1.1.1, this event is similar to the effect of increasing steam flow. This is bounded by the event in FSAR Sections 15.1.2 and 15.1.3, as stated in FSAR Section 15.1.1.3.

b) Excessive Feedwater Flow - As seen in FSAR Figure 15.1.2-4 for the HZP case, the pressurizer pressure initially decreases and then begins to increase just prior to the time of reactor scram. The pressure rise is caused once the shutdown reactivity is overcome by the positive moderator feedback with a resultant core power and temperature increase. The pressurizer power-operated relief valves (PORV's) are assumed to open to conservatively minimize MDNBR; however the safety valves do not open. Since the PORV's and safety valves are not required to mitigate the consequences of the event (i.e., this event is not a limiting overpressure event), a single failure in these valves to open is not applicable and has no impact. Failure of a feedwater isolation valve (FWIV) to close will have no impact since the minimum DNBR occurs before the time the FWIV closes (FSAR Table 15.1.2-4). The engineered safety features are not required for this event. Therefore, a single failure in the ESF is not applicable and has no impact. Therefore, the failure of one protection train as listed in Table 15.0.13-1 is the limiting single active failure.

c) Excessive Steam Flow - As stated in FSAR Section 15.1.3.1, depending on the magnitude of the increase in steam demand, a reactor trip may not be activated. Instead, the reactor system will reach a new steady-state condition at a power level greater than the initial power level which is consistent with the increased heat removal rate. No reactor trip is required, no pressurizer relief valves are required to reduce pressure (FSAR Figure 15.1.3-2), and no ESF actuation occurs. Since the protection system is not required to function for this event, a single failure does not apply and has no impact.

d) Inadvertent Secondary Depressurization - As stated in FSAR Section 15.1.4.1, it is the failure (opening) of a steam dump, relief, or safety valve which initiates the transient. As seen in relieving functions of the protection system are not challenged nor required to mitigate the consequences of the event. The only portion of the protection system required is the safety injection portion of the ESF. A single failure in a protection train of the signals which actuate SI (FSAR Section 15.1.4.1 Item a) will have no impact due to the redundancy, diversity, and independence of the SI actuations signals. The failure of one SI train (listed in Table 15.0.13-1) is the limiting single failure since it reduces SI flow, delays the injection of boron to the core, and, consequently allows a "closer" return to criticality. This is the single failure assumed in the FSAR as stated in FSAR Section 15.1.4.2. For this event, the DNB design basis is met by demonstrating no return to criticality (FSAR Section 15.1.4.3).

e) Loss of External Load - This is bounded by the event described in FSAR Section 15.2.3, as stated in FSAR Sections 15.2.2.1 and 15.2.3.1.

f) Turbine Trip - Unlike a depressurization transient, for this analysis, the ability to maintain Reactor Coolant System (RCS) pressure below 110 percent of design per the SRP criterion must be explicitly addressed. Since the DNBR increases with

pressure (assuming all other variables are held constant), the event is analyzed with and without pressure control to address both peak pressurizer and DNBR concerns. As stated in FSAR Section 15.2.3.2, both the pressurizer and steam generator safety valves may be required to operate. Assumptions relative to their operation are described under Items 2 and 3 in the FSAR.

If the pressurizer relief/safety valves fail to close once the pressure has been reduced, there will be no impact on the minimum DNBR. This is because the valves are not required to close until after the time of reactor trip, at which point the DNBR is rising and is very high (see FSAR Figures 15.2 3-1 through 15.2.3-7). As stated in FSAR Section 15.2.3.2, Item 2, steam relief is obtained by the steam generator safety valves. However, these or any other steam relief valves would not be required to close until after reactor trip, when both the RCS pressure and DNBR are past their maximum and minimum values respectively. Therefore, failure to close would have no impact. Although the ESF may be required to function to supply auxiliary feedwater, a failure in the ESF would have no impact since credit for auxiliary feedwater is not taken (FSAR Section 15.2.3.2, Item 4). Therefore, the limiting single failure is one protection train (Table 15.0.13-1).

g) Inadvertent Closure of Main Steam Isolation Valve (MSIV) - This is bounded by FSAR Section 15.2.3 as stated in the FSAR.

h) Loss of Condenser Vacuum - The results and conclusions of FSAR Section 15.2.3 apply to this event as stated in the FSAR.

i) Loss of AC Power - For this event, the ability of the protection system to provide long term cooling is verified. The worst single failure will minimize auxiliary feedwater delivery to the steam generators. As noted in Table 15.0.13-1, this results in a higher primary side heatup and pressure.

Although Table 10.4.9-2 presents the worst single failure affecting AFS flow as loss of one pump, SHNPP has taken credit for the ability to mitigate the Loss of Feedwater and Loss of AC Power Events using only one AFW pump in responding to TMI Action Item II.E.1.1. As documented in Chapter 10.4.9 App A of the SHNPP FSAR, taking credit for this capability enabled SHNPP to achieve the AFW System reliability criteria specified in the TMI Action Item. Therefore, this analysis was performed using the flow from only one MDAFW pump in order to demonstrate that capability. It should be noted that the corresponding loss of two of three AFS pumps is more conservative than any postulated single active failure (Reference 15.0.13-1).

Separate from pump or system flow, an additional consideration presented in Table 10.4.9-2 is the number of steam generators that receive auxiliary feedwater. An alternative potential failure is temporary isolation of one steam generator. Instead of dividing the minimum design (i.e., one pump) flow among three steam generators, inadvertent isolation of a steam generator would split the full (three pump) AFS flow between two steam generators.

For the cases in which forced RCS flow is maintained to sustain the normal operating heat transfer coefficient between the primary coolant and the inside of the steam generator tubes, the alternative of delivering three or four times as much auxiliary feedwater to two of three steam generators represents a much less restrictive limitation on cooling capability. However, when loss of offsite power is the initiating event, the relative cooling capability of AFW flow rate vs

number of steam generators receiving flow is not immediately obvious. Under RCP coastdown and natural circulation conditions, it is not clear whether the limiting factor for RCS cooling is steam generator secondary side inventory vs degraded heat transfer from the primary coolant to the SGs. As such, analysis of loss of AC power is based on conservatively combining these two alternative failure scenarios. With flow from only a single Auxiliary Feedwater pump, the analysis is performed with and without an additional failure of an auxiliary feedwater isolation valve in the closed position. Although postulation of multiple independent failures is not required, doing so is conservative. Therefore, the analysis basis listed in Table 15.0.13-1 is one motor driven AFS pump delivering flow to two or three steam generators (as separate cases).

In this case, a unique distinction exists where the AFW system configuration and performance assumed in the analysis (one AFW pump delivering to two steam generators) does not represent a legitimate performance requirement for the system. The analysis uses a "beyond licensing basis" system configuration that combines the independent failures in the AFW pump capability and the inadvertent isolation of one steam generator. The acceptable results obtained with the overly-conservative, beyond-licensing-basis model demonstrates that either of the two independent legitimate failure scenarios would also yield acceptable results if analyzed separately.

However, the analysis model itself (with its combination of independent failures) does not accurately represent a legitimate licensing basis configuration for the system and there is no commitment or requirement that the system must be able to perform as modeled in the analysis. Therefore, the AFW system does not have to be capable of delivering 390[a] gpm of flow from only one AFW pump with the coincident isolation of one steam generator. If only one AFW pump is assumed operational, then the required 390[a] gpm of flow can be assumed to be delivered to all three steam generators. If a failure has resulted in isolation of AFW flow to one steam generator, then it may be assumed that all three AFW pumps would be available to deliver the required 390[a] gpm of flow. The minimum capacity of a motor driven auxiliary feedwater (MDAFW) pump is 430 gpm. For conservatism, the flow capacity of any pump was assumed to be 390[a] gpm in the FSAR analysis.

For the case where the single active failure is the failure of the pressurizer PORV or safety valve to close, credit can be taken for complete auxiliary feedwater capability. This would reduce the peak pressure and cause the time at which decay heat equals heat removal capability to be sooner. As stated in FSAR Section 15.2.6.1, the steam generator safety and relief valves are used to dissipate decay heat during long term cooling. Since it is desirable to have these valves open, failure to close has no impact, especially since the emergency feedwater supplies sufficient heat removal capability. Single failures which result in loss of signals which actuate auxiliary feedwater, reactor trip, or valve openings have no impact due to their redundancy, diversity and independence.

j) Loss of Normal Feedwater - As for the loss of power event, the primary concern for the loss of normal feedwater is long term cooling capability which is provided by the emergency feedwater system. Because of the similarity of these two events, the discussion of the limiting failure for loss of AC power is generally applicable. However, loss of normal feedwater is analyzed slightly differently in that offsite power continues to be available to power the reactor coolant pumps. With the continuing

[a] Safety analyses support an AFW flow rate of 374 gpm at the lowest lifting MSSV setpoint plus 3% tolerance. See Section 15.0 for more details.

forced flow in the primary, the failure mode of minimum AFW flow is clearly more limiting than the inadvertent isolation of AFW flow to one steam generator. Therefore, the analysis basis listed in Table 15.0.13-1 is one motor-driven pump delivering flow to all three steam generators.

k) Loss of Flow - The protection for this event is discussed in FSAR Sections 15.3.1.1 and 15.3.2.1. A single failure in the ESF is not applicable since the ESF are not required to mitigate the consequences of the event. As can be seen in FSAR Figure 15.3.2 3 the pressurizer PORV's may open. However, failure to close will have no impact since the point of minimum DNBR is past and the DNBR is rising by the time the valves close. Therefore, the worst single failure is that of one protection train, as stated in Table 15.0.13-1.

l) Rod Cluster Control Assembly (RCCA) Bank Withdrawal from Subcritical - Two cases are analyzed to evaluate the challenge to system pressurization and the challenge to DNBR. An increase in RCS pressure is expected due to the increase in heat flux and temperature. However, if the PORV's opened and failed to close, there would be no impact on the minimum DNBR since DNBR increases with increases in pressure. The ESF are not required for this accident, therefore, a single failure in the ESF is not applicable. Therefore, a loss of one protection train is the limiting single failure.

m) RCCA Bank Withdrawal at Power - This event is primarily a DNB event and demonstrates the adequacy of the over temperature ΔT and high flux trips, as stated in FSAR Sections 15.4.2.1 and 15.4.2.3. Typical transients for the RCCA bank withdrawal at power event are provided in FSAR Figures 15.4.2-1 through 15.4.2-9. Operation of pressure relieving valves would serve to reduce pressure and thus minimize the DNBR.

(If no pressure control was available, the maximum pressure would be limited to that which results in a high pressurizer pressure trip. This is a less limiting pressure transient than those events discussed in FSAR Section 15.2.) Failure of valves to close would have no impact, since the point of minimum DNBR is passed by the time the pressure begins to fall (after trip) as seen in the transient figures.

As discussed in FSAR Section 15.4.2.1, for some cases, the steam generator safety valves are opened. However, failure to close has no impact since the point of minimum DNBR comes right after reactor trip. Failures in the ESF are not applicable since the ESF are not required. Therefore, the worst single failure is one protection train as stated in Table 15.0.13-1.

n) Dropped RCCA - The worst single failure for this event is the failure of one Nuclear Instrumentation System (NIS) channel. The failure of an NIS channel results in fewer dropped RCCA's being detected in order to initiate reactor trip via negative flux rate. When operating with the ARC system, failure of an NIS channel decreases the ability to reduce power on power mismatch. If no trip is generated, the plant reaches a new equilibrium condition, and no further protective action is required. Therefore, consideration of other single failures within the protection system is not applicable.

o) Statically Misaligned RCCA - As stated in Table 15.0.13-1, no transient analysis is required. Furthermore, no protective functions are required and single failures have no impact.

p) Inadvertent Boron Dilution - For inadvertent boron dilution at cold shutdown (as the case with the most severe consequences), the initiating event is taken as operator error, e.g., setting the boric acid concentration to zero while blending. The result is injection of pure water into the RCS at a flowrate corresponding to the setpoint for the Reactor Makeup Water System (RMWS) flow control valve. In addition to the initiating error, the worst single active failure is malfunction of the RMWS flow indication. If actual dilution flow is higher than indicated, the RMWS can deliver higher than programmed flow without actuating a flow deviation alarm. The result is increasing the maximum dilution flow up to the maximum charging/letdown rate. Even though higher flows may be possible from the RMWS, any flowrate above charging/letdown will automatically be diverted to the Volume Control Tank.

   In addition to the initiating error, failure of either flow controllers or valves would have less severe impact because early alarm due to flow deviation would result.

q) Inadvertent Actuation of the ECCS - As stated in FSAR Section 15.5.1.1, it is an operator error or an inadvertent safety injection system actuation signal from any such actuation channel which is the initiating event for this accident. The single failure considered is the failure of the ESF protection system to initiate a reactor trip upon spurious safety injection signal (for the DNBR). Other failures were considered, but these other failures had the effect of mitigating the challenge to DNBR by keeping system pressure high, or by reducing the fuel power levels. For PORV/Safety challenge (which would threaten the acceptance criteria that this event not lead to a more serious event, such as a SBLOCA), modeling the reactor trip on SI signal was conservative. There were no other single failures that were identified which would cause colder water to reach the PORV's and Safeties sooner, to increase the challenge to those components. Therefore, the failure of one protection train listed in Table 15.0.13-1 is the limiting single failure considered for this event.

r) Increase in RCS Inventory - As stated in the FSAR, this is bounded by FSAR Sections 15.5.1 and 15.4.6.

s) Inadvertent RCS Depressurization - As stated in FSAR Section 15.6.1.1, it is a single failure resulting in the opening of a pressurizer PORV or safety valve which initiates the transient. Although ESF features might be actuated, they are not required to mitigate the consequences of the event, since the DNBR rises after reactor trip. Therefore, ESF failures are not applicable. Therefore, the worst single failure is failure of one protection train.

t) Failure of Small Lines - No transient analysis is involved for this event. The protective system is not required to function, since operator action terminates this event as stated in FSAR Section 15.6.2.

2) Single Failures for Non-Condition II Events - Table 15.0.13-2 addresses events other than those of moderate frequency. Fuel failures (if any) which occur are discussed in the appropriate section of the FSAR.

The reliability criteria specified in TMI Action Item II.E.1.1 (FSAR Section 10.4.9 App A) that lead to modeling multiple AFW pump failures for the Loss of Feedwater and Loss of AC power Events is not applicable to the Feedline Break Event. Therefore, since no credible single-active-failure criteria has been identified that leads to loss of two AFW pumps, credit for two AFW pumps may be assumed in order to provide the AFW flow modeled to mitigate this event.

Although operator error is not explicitly considered in the accidents analyzed in Chapter 15, it should be noted that cognitive operator errors may induce the transient or cause the limiting single failures listed. With the exception of the steam generator tube rupture (SGTR), and the boron dilution, no credit for operator action or non-action is considered during the transient.

For the boron dilution event, operator action to terminate the dilution is taken within the time frames specified in the FSAR.

Operator action (subsequent to the action of the protection system) is also considered to mitigate the consequences for an SGTR event. The operator actions and the operator action times assumed for the SGTR analysis are discussed in Section 15.6.3

For the FLB with Loss of Offsite Power Case (natural circulation flow in the primary), a question remains as to whether the SHNPP-specific failure mode involving inadvertent isolation of AFW flow to a single steam generator is more limiting than the loss of AFW flow associated with the AFW pump failure. As discussed for the Loss of AC Power Event Failure Mode discussion (Section 15.0.13.1.i), sensitivity studies were performed to identify which of the independent failures was most limiting.

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APPENDIX 15.0A

15.0A.1 OFFSITE DOSE CALCULATION MODELS

15.0A.1.1 Computer Code and Model Description

The Westinghouse TITAN5 computer code was used to calculate offsite and control room personnel doses for the Alternative Source Term implementation under Reg Guide 1.183 (Reference 15.0.9-8). The code has been verified, documented, and controlled within Westinghouse, but has not been submitted to the NRC for generic approval under a topical report. Figure 15.0A.1-1 shows a simplified containment diagram containing elements of the model TITAN5 uses to evaluate the effects of containment sprays and containment leakage.

Using the appropriate event specific assumptions, the time dependent release of nuclides is calculated. No credit is taken for cloud depletion by ground deposition or by radioactive decay during transport to the exclusion area boundary or to the outer boundary of the low population zone. The time dependent release of nuclides is used to calculate TEDE doses to the individuals at specific offsite locations. The TEDE dose is equivalent to the committed effective

dose equivalent (CEDE) or inhalation dose plus the acute dose (EDE) dose for the duration of exposure to the cloud. The offsite TEDE doses are calculated using the following equations:

Offsite inhalation doses (CEDE) are calculated using the following equation:

$$D_{CEDE} = \sum_i \left[DCF_i \left(\sum_i (\mathrm{IAR})_{ij} (\mathrm{BR})_j (\chi/\mathrm{Q})_j \right) \right]$$

where:

D_{CEDE} = CEDE dose via inhalation (rem).

DCF_i = CEDE dose conversion factor via inhalation for isotope i (rem/Ci) (Table 15.0.9-3)

$(IAR)_{ij}$ = integrated activity of isotope i released during the time interval j (Ci)

$(BR)_j$ = breathing rate during time interval j (m^3/sec) (Table 15.6.3-10)

$(\chi/Q)_j$ = atmospheric dispersion factor during time interval j (sec/m^3) (Table 15.6.3-10)

Offsite external exposure (EDE) doses are calculated using the following equation:

$$D_{EDE} = \sum_i \left[DCF_i \left(\sum_j (\mathrm{IAR})_{ij} (\chi/\mathrm{Q})_j \right) \right]$$

where:

D_{EDE}= external exposure dose via cloud immersion (rem)

DCF_i = EDE dose conversion factor via external exposure for isotope i (rem x m^3/Ci x sec) (Table 15.0.9-4)

$(IAR)_{ij}$ = integrated activity of isotope i released during the time interval j (Ci)

$(\chi/Q)_j$ =atmospheric dispersion factor during time interval j (sec/m^3)(Table 15.6.3-10)

15.0A.1.2 Containment Leakage Pathway

After a major LOCA, containment leakage is assumed to be equal to 0.1 percent of the containment volume per day for the first 24 hours and to 50 percent of this value for the duration of this accident. For the purpose of dose calculations, this leakage has been assumed to reach the environs unfiltered, bypassing those portions of the Reactor Auxiliary Building which are maintained under negative pressure with engineered safety feature-grade filtered exhaust systems.

15.0A.1.3 Mixing Between Sprayed and Unsprayed Regions

As shown in Figure 15.0A.1-1, a portion of the containment does not receive direct coverage by the containment spray system (sprayed region is conservatively modeled as 85.9% of the

maximum net free volume of containment). To determine the effects of mixing between the sprayed and unsprayed regions, the two regions were modeled separately in TITAN5, with air and radionuclide transfer between the two regions determined by the minimum (one train, two units, half speed) containment fan cooler flow rate. No iodine removal or particulate filtration credit is applied to this transfer mechanism.

APPENDIX 15.0 - DELETED BY AMENDMENT NO. 41.

15.1 INCREASE IN HEAT REMOVAL BY THE SECONDARY SYSTEM

A number of events have been postulated which could result in an increase in heat removal from the Reactor Coolant System (RCS) by the Secondary System. Detailed analyses are presented for several such events which have been identified as limiting cases.

Discussions of the following RCS cooldown events are presented in this section:

1) Feedwater system malfunctions that result in a decrease in feedwater temperature (ANS Condition II event).

2) Feedwater system malfunctions that result in an increase in feedwater flow (ANS Condition II event).

3) Excessive increase in secondary steam flow (ANS Condition II event).

4) Inadvertent opening of a steam generator relief or safety valve (ANS Condition II event).

5) Steam system piping failure (ANS Condition III and IV events).

Section 15.0.1 contains a discussion of ANS classifications.

15.1.1 FEEDWATER SYSTEM MALFUNCTIONS THAT RESULT IN A DECREASE IN FEEDWATER TEMPERATURE

15.1.1.1 Identification of Causes and Accident Description

Reductions in feedwater temperature will cause an increase in core power by decreasing reactor coolant temperature. Such transients are attenuated by the thermal capacity of the secondary plant and of the RCS. The overpower/ overtemperature protection (overtemperature and overpower ΔT trips) prevents any power increase which could lead to a departure from nucleate boiling ratio (DNBR) less than the safety analysis limit.

A reduction in feedwater temperature may be caused by the accidental opening of a feedwater bypass valve, which diverts flow around a portion of the feedwater heaters. In the event of an accidental opening of the bypass valve, there is a sudden reduction in feedwater inlet temperature to the steam generators. At power, this increased subcooling will create a greater load demand on the RCS.

With the plant at no-load conditions the addition of cold feedwater may cause a decrease in RCS temperature and thus an effective reactivity insertion due to the effects of the negative

moderator coefficient of reactivity. However, the rate of energy change is reduced as load and feedwater flow decrease, so the no-load 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, i.e., the reactor will reach a new equilibrium condition at a power level corresponding to the new steam generator ΔT.

A decrease in normal feedwater temperature is classified as an ANS Condition II event, fault of moderate frequency. See Section 15.0.1 for a discussion of ANS Condition II events.

The protection available to mitigate the consequences of a decrease in feedwater temperature is the same as that for an excessive steam flow increase, as discussed in Section 15.0.8 and listed in Table 15.0.8-1.

15.1.1.2 Analysis of Effects and Consequences

1) Method of Analysis - This transient is analyzed by computing conditions at the feedwater pump inlet following opening of the heater bypass valve. These feedwater conditions are then used to recalculate a heat balance through the high pressure heaters. This heat balance gives the new feedwater conditions at the steam generator inlet.

   The following assumptions are made:

   a) Plant initial power level corresponding to the maximum design NSSS thermal output.

   b) Low pressure feedwater heater bypass valve opens, resulting in condensate flow splitting between the feedwater bypass line and the low pressure feedwater heaters; the flow through each path is proportional to the feedwater pressure drops.

   c) Heater drain pumps trip; this increases the effect of the cold bypass flow.

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

2) Results - Opening of the low pressure feedwater heater bypass valve and trip of the heater drain pumps causes a reduction in feedwater temperature which increases the thermal load on the RCS. The calculated reduction in feedwater temperature results in an increase in heat load on the RCS of less than 10 percent of full power. The increased thermal load, due to opening of the low pressure feedwater heater bypass valve, thus would result in a transient very similar (but of reduced magnitude) to that presented in Section 15.1.3 for an excessive increase in secondary steam flow, which evaluates the consequences of a 10 percent step load increase. Therefore, the transient results of this analysis are not presented.

15.1.1.3 Conclusions

The decrease in feedwater temperature transient is less severe than the increase in secondary steam flow event (Section 15.1.3). Based on results presented in Section 15.1.3, the applicable acceptance criteria for the decrease in feedwater temperature event have been met.

15.1.2 FEEDWATER SYSTEM MALFUNCTIONS THAT RESULT IN AN INCREASE IN FEEDWATER FLOW

15.1.2.1 Identification of Causes and Accident Description

Addition of excessive feedwater will cause an increase in core power by decreasing reactor coolant temperature. Such transients are attenuated by the thermal capacity of the secondary plant and of the RCS. The overpower/ overtemperature protection (overtemperature and overpower ΔT trips) prevents any power increase which could lead to a DNBR less than the safety analysis limit.

An example of excessive feedwater flow would be a full opening of a feedwater control valve due to a feedwater control system malfunction or an operator error. At power, this excess flow causes a greater load demand on the RCS due to increased subcooling in the steam generator. With the plant at no-load conditions, the addition of cold feedwater may cause a decrease in RCS temperature and thus an effective reactivity insertion due to the effects of the negative moderator coefficient of reactivity.

Continuous addition of excessive feedwater is prevented by the steam generator high-high level trip, which closes the feedwater valves.

An increase in normal feedwater flow is classified as an ANS Condition II event, fault of moderate frequency. See Section 15.0.1 for a discussion of ANS Condition II events. The acceptance criteria for this event are:

1) The pressures in the reactor coolant and main steam systems should be less than 110% of design values.

2) The fuel cladding integrity should be maintained by ensuring that fuel design limits are not exceeded. This is demonstrated by assuring that the minimum calculated departure from nucleate boiling ratio (DNBR) is not less than the applicable limits of the DNBR correlation being used and that fuel centerline melt does not occur.

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

4) For transients of moderate frequency in combination with a single failure, no loss of function of any fission product barrier, other than fuel element cladding, shall occur. Core geometry is maintained in such a way that there is no loss of core cooling capability and control rod insertability is maintained.

No single active failure will prevent operation of the RPS.

Plant systems and equipment which are available to mitigate the effects of the accident are discussed in Section 15.0.8 and listed in Table 15.0.8-1.

15.1.2.2 Analysis of Effects and Consequences

1) Method of Analysis - The system is analyzed to demonstrate plant behavior in the event that excessive feedwater addition, due to a control system malfunction or operator error which allows a feedwater control valve to open fully, occurs.

This event is analyzed at both HFP conditions and HZP conditions. The HFP and HZP cases analyzed are:

a) HFP conditions, minimum (BOC) reactivity feedback, automatic rod control, full open main feedwater valve used to initiate the transient

b) HFP conditions, maximum (EOC) reactivity feedback, manual rod control, full open main feedwater valve used to initiate the transient

c) HFP conditions, maximum reactivity feedback, automatic rod control, full open main feedwater valve used to initiate the transient

d) HZP conditions, maximum reactivity feedback, manual rod control, step change in main feedwater flow from 0 to 120% nominal used to initiate the transient

The feedwater flow resulting from a fully open control valve is terminated by a steam generator high-high level trip signal which closes all feedwater control and isolation valves, trips the main feedwater pumps, and trips the turbine.

This event is analyzed using the SGR/Uprating values for T_{avg} (T_{avg}=588.8°F at 100% power and T_{avg}=557°F at HZP).

The Increase in Feedwater event is asymmetric, affecting only one of the three steam generators. Therefore, the moderator reactivity feedback is conservatively computed using the moderator temperature in the cold leg of the affected loop rather than the core average moderator temperature. This method of computing moderator feedback is conservative for EOC cases because it maximizes the reactivity insertion rate. It is also conservative for the BOC case because it results in the most aggressive rod pull.

Conservative conditions established for the analysis of this event are presented in Table 15.1.2-1. Available reactor protection system trips are presented in Table 15.0.6-2. For conservatism, only the high power range and turbine trips were active in the analysis of this event. Key operating parameters used in the analysis of this event are presented in Table 15.1.2-2. The range of neutronics parameters supported by this analysis are presented in Table 15.1.2-3.

The transient response of the reactor system is calculated using the ANF RELAP (Reference 15.1.2-3) computer program. The core thermal hydraulic boundary conditions from the ANF-RELAP calculation are used as input to the XCOBRA-IIIC code (References 15.1.2-4 and 15.1.2-5) to predict the MDNBR for the event.

Results - The limiting case is the HZP case with maximum (EOC) reactivity feedback and manual rod control. This transient tripped on a high neutron flux trip. The sequence of events for the limiting case is given in Table 15.1.2-4. The responses to key system variables are given in Figures 15.1.2-1 to 15.1.2-5.

15.1.2.3 Conclusions

The results of the analysis demonstrate that the event acceptance criteria are met. Both the maximum reactor primary system pressure and the maximum secondary system pressure are less than the design limits of 2750 psia and 1320 psia, respectively. The predicted MDNBR is greater than the safety limit. The critical heat flux correlation limit ensures that, with 95% probability and 95% confidence, DNB is not expected to occur; therefore, no fuel is expected to fail. Fuel centerline melt threshold is not penetrated during this event.

The analysis for this event supports full power operation at nominal primary average temperature less than, or equal to, 588.8°F.

15.1.3 EXCESSIVE INCREASE IN SECONDARY STEAM FLOW

15.1.3.1 Identification of Causes and Accident Description

An excessive increase in secondary system steam flow (excessive load increase incident) is defined as a rapid increase in 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 in the range of 15 to 100 percent of full power. Any loading rate in excess of these values may cause a reactor trip actuated by the Reactor Protection System. Steam flow increases greater than 10 percent are analyzed in Sections 15.1.4 and 15.1.5. Steam flow increases due to a major steam line rupture are equivalent to a step increase in steam flow of over 200 percent.

The event initiator is a 10% step increase in steam flow. The feedwater regulating valves open to increase the feedwater flow in an attempt to match the increased steam demand and maintain steam generator water level. In response to the increased steam flow, the secondary system pressure decreases, resulting in an increase in the primary-to-secondary heat transfer rate. The primary side steam generator outlet temperature decreases due to the enhanced heat removal. As a consequence, the primary system core average temperature decreases and the primary system fluid contracts, resulting in an outsurge of fluid from the pressurizer. The pressurizer level and pressure decrease as fluid is expelled from the pressurizer. If the moderator temperature coefficient (MTC) is negative, the reactor core power increases as the moderator temperature decreases due to the mismatch between the power being removed by the steam generators and the power being generated in the core.

The reactor responds to the mismatch between the power being removed by the steam generators and the power being generated in the core. The rod control system is conservatively assumed to be in the automatic state.

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, i.e., high reactor coolant temperature indicates a need for steam dump. A single controller malfunction does not cause steam dump; an interlock is provided which blocks the opening of the valves unless a large turbine load decrease or a turbine trip has occurred.

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

1) Overpower ΔT.

2) Overtemperature ΔT.

3) Power range high neutron flux.

Depending on the magnitude of the increase in steam demand, a reactor trip may not be activated. Instead, the reactor system will reach a new steady-state condition at a power level greater than the initial power level which is consistent with the increased heat removal rate. The final steady-state conditions which are achieved will depend upon the capacity of the turbine control valves, the magnitude of the MTC, and whether or not the rod control system is in automatic. If the MTC is positive, the reactor power would decrease as the core average coolant temperature decreased, and this event would not produce a challenge to the acceptance criteria.

An excessive load increase incident is considered to be an ANS Condition II event, fault of moderate frequency. See Section 15.0.1 for a discussion of Condition II events. The acceptance criteria for this event are:

1) The pressures in the reactor coolant and main steam systems should be less than 110% of design values.

2) The fuel cladding integrity should be maintained by ensuring that fuel design limits are not exceeded. This is demonstrated by assuring that the minimum calculated departure from nucleate boiling ratio (DNBR) is not less than the applicable limits of the DNBR correlation being used and that fuel centerline melt does not occur.

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

4) For transients of moderate frequency in combination with a single failure, no loss of function of any fission product barrier, other than fuel element cladding, shall occur. Core geometry is maintained in such a way that there is no loss of core cooling capability and control rod insertability is maintained.

No single active failure will prevent operation of the RPS.

15.1.3.2 Analysis of Effects and Consequences

Method of Analysis - This event is predominantly a cooldown event and is evaluated at full power conditions. At full power, the margin to limits is the smallest and, therefore, bounds operation at lower power levels. The reactor control system is designed to accommodate a 10% increase in load (step increase) or a 5% per minute load ramp for power levels between 15% and 100% of full power. The 10% step increase in load is analyzed because it is the highest expected increase that would occur. Two cases are analyzed: one for minimum neutronics feedback (BOC conditions) and the other for maximum neutronics feedback (EOC conditions). Both cases are evaluated with automatic rod control.

This event is analyzed using the SGR/Uprating value for T_{avg} (588.8°F). By analyzing this event at 588.8°F, the DNB results bound all lower temperature operating conditions.

Conservative conditions established for the analysis of this event are presented in Table 15.1.3-1. Available reactor protection system trips are presented in Table 15.0.6-2. All trips listed in this table were modeled for the analysis of this event. Key operating parameters used in the analysis of this event are presented in Table 15.1.3-2. The range of neutronics parameters supported by this analysis are presented in Table 15.1.3-3.

The transient response of the reactor system is calculated using the ANF-RELAP (Reference 15.1.2-3) computer program. The core thermal hydraulic boundary conditions from the ANF-RELAP calculation are used as input to the XCOBRA-IIIC code (References 15.1.2-4 and 15.1.2-5) to predict the MDNBR for the event.

Results - The event is initiated by a rapid opening of the turbine control valves, the atmospheric dump valves and/or the turbine bypass valves resulting in a 10% step increase in steam flow. The maximum increased steam flow rate at full power is 110% of rated. The limiting case is the minimum (BOC) neutronics feedback case. There was no reactor trip for this transient.

The sequence of events is given in Table 15.1.3-4. The responses to key system variables are given in Figures 15.1.3-1 to 15.1.3-4.

15.1.3.3 Conclusions

The results of the analysis demonstrate that the event acceptance criteria are met. Both the maximum reactor primary system pressure and the maximum secondary system pressure are less than the design limits of 2750 psia and 1320 psia, respectively. The predicted MDNBR is greater than the safety limit. The critical heat flux correlation limit ensures that, with 95% probability and 95% confidence, DNB is not expected to occur; therefore, no fuel is expected to fail. The fuel centerline melt threshold is not penetrated during this event.

The analysis for this event supports nominal primary T_{avg} operation at full power at average primary temperatures of 588.8°F and less.

15.1.4 INADVERTENT OPENING OF A STEAM GENERATOR RELIEF OR SAFETY VALVE

15.1.4.1 Identification of Causes and Accident Description

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

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

The analysis is performed to demonstrate that the following criterion is satisfied:

Assuming a stuck rod cluster control assembly, and a single failure in the Engineered Safety Features System, the DNB will remain above the safety analysis limit following a steam release equivalent to the spurious opening, with failure to close, of the largest of any single steam dump, power operated relief, or safety valve. Accidental depressurization of the secondary system is classified as an ANS Condition II event. See Section 15.0.1 for a discussion of Condition II events.

The following systems provide the necessary protection against an accidental depressurization of the Main Steam Supply System.

1) Safety Injection System actuation from any of the following:

    a) Two out of three pressurizer pressure signals.

    b) Two out of three High-1 containment pressure signals.

    c) Two out of three low steamline pressure signals in any one main steam line.

2) The overpower reactor trips (neutron flux and ΔT), low pressurizer pressure reactor trip, and the reactor trip occurring in conjunction with receipt of the safety injection signal.

3) Redundant isolation of the main feedwater lines.

Sustained high feedwater flow would cause additional cooldown. Therefore, in addition to the normal control action which will close the main feedwater valves following reactor trip, a safety injection signal will rapidly close all feedwater control valves and back up feedwater isolation valves, trip the main feedwater pumps, and close the feedwater pump discharge valves.

4) Trip of the fast-acting main steam line isolation valves (designed to close in less than 7 seconds) on:

    a) High-2 containment pressure.

    b) Safety injection system actuation derived from two out of three low steam line pressure signal in any one main steam line (above Permissive P-11).

    c) High negative steam pressure rate indication from two out of three signals in any one main steam line (below Permissive P-11).

Plant systems and equipment which are available to mitigate the effects of the accident are also discussed in Section 15.0.8 and listed in Table 15.0.8-1.

15.1.4.2 Deleted by Amendment No. 48.

15.1.4.3 Deleted by Amendment No. 48.

15.1.4.4 Event Disposition for AREVA Analysis

Mode 1: The maximum steam flow through a single steam dump, power operated relief, or safety valve drives a thermal load increase less than that considered in Event 15.1.3. Ultimate reactor power level and the potential challenge to SAFDLs is greater for Event 15.1.3. The event is therefore bounded by Event 15.1.3 before trip and by Event 15.1.5 after trip, since the Condition II criteria are met by the more challenging Condition IV event.

Mode 2: The reactor will achieve a steady state power level equal to its initial power plus the additional load imposed by the steam flow through the failed valve. Because the initial power level is less than 5% of rated, and the additional load is less than 10% of rated power, the reactor power will not rise to a level at which a significant challenge to SAFDLs is posed. Event 15.1.4 is bounded in Mode 2 by the "Off-site Power Available" Case analyzed for Mode 2 of Event 15.1.5, "Steamline Break". The thermal load increase associated with the latter event is significantly greater than encountered here and the "Off-site Power Available" Steamline Break Cases continue to meet the Condition II acceptance criteria.

Mode 3: The event will proceed as described for Mode 2, except that the reactor is subcritical by at least 1000 pcm at event initiation for Mode 3. The initial margin to criticality for Mode 3 will slow the evolution of the event relative to Mode 2. Time available for operator response is greater than in Mode 2. Event 15.1.4 is bounded in Mode 3 by the "Off-site Power Available" Case analyzed for Mode 2 of Event 15.1.5, "Steamline Break". The thermal load increase associated with the latter event is significantly greater than encountered here and the "Off-site Power Available" Steamline Break Cases continue to meet the Condition II acceptance criteria.

Modes 4, 5, and 6: The reactor coolant temperature and secondary pressure are significantly reduced relative to Mode 3 for these modes. Thus, the potential cooldown is smaller than that in Mode 3 due to the decreased steam flow through the affected valve in these modes. The challenge to SAFDLs is reduced directly by the lower coolant temperatures. The Mode 3 event, therefore, bounds the event in Modes 4, 5, and 6. In Modes 5 and 6, the event cannot occur with consequences because the primary and secondary temperatures are below saturation at atmospheric pressure.

15.1.5 STEAM SYSTEM PIPING FAILURE

15.1.5.1 Identification of Causes and Accident Description

The steam release arising from a rupture of a main steam line would result in an initial increase in steam flow which decreases during the accident as the steam pressure falls. The energy removal from the Reactor Coolant System (RCS) causes a reduction of reactor coolant temperature and pressure. In the presence of a negative moderator temperature coefficient, the cooldown results in an insertion of positive reactivity. If the most reactive 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 line rupture is a potential problem mainly because of the high power peaking factors which exist assuming the most reactive RCCA to be stuck in its fully withdrawn position.

The core is ultimately shut down by the boric acid injection delivered by the Safety Injection System.

A major steam line rupture is classified as an ANS Condition IV event. See Section 15.0.1 for a discussion of ANS Condition IV events. The acceptance criteria for this event are:

1) Pressure in the reactor coolant and main steam systems should be maintained below acceptable design limits.

2) Any fuel failure calculated to occur must be of sufficiently limited extent that the core will remain in place and intact with no loss of core cooling capability.

3) The integrity of the reactor coolant pumps should be maintained, such that a loss of AC power and containment isolation will not result in pump seal damage.

4) The auxiliary feedwater system must be safety grade and, when required, automatically initiated.

5) Tripping of the reactor coolant pumps should be consistent with the resolution of TMI Action Plan item II.K.3.5.

6) The radiological analysis and acceptance criteria are described in the Alternative Source Term Regulatory Guide 1.183, Reference 15.0.9-8.

    a) For a Main Steam Line Break (MSLB) with an assumed pre-accident iodine spike or with accident induced fuel failure, the calculated doses should be less than the 10CFR50.67 limits of 25 rem TEDE offsite and 5 rem TEDE in the Control Room.

    b) For a Main Steam Line Break (MSLB) with an accident initiated iodine spike, the calculated offsite doses should be less than 10% of the 10 CFR 50.67 limits (i.e., less than 2.5 rem TEDE) and the control room doses should be less than the 10CFR50.67 limit of 5 rem TEDE.

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

The following functions provide the protection for a steam line rupture:

1) Safety Injection System actuation from any of the following:

    a) Two out of three low pressurizer pressure signals.

    b) Two out of three Hi-1 Containment pressure signals.

    c) Two out of three low steam line pressure signals in any one main steam line.

2) The overpower reactor trips (neutron flux and ΔT), low pressurizer pressure reactor trip, and the reactor trip occurring in conjunction with receipt of the safety injection signal.

3) Redundant isolation of the main feedwater lines - Sustained high feedwater flow would cause additional cooldown. Therefore, in addition to the normal control action which will close the main feedwater valves after a reactor trip, a safety injection signal will rapidly close all feedwater control valves and backup feedwater isolation valves, trip the main feedwater pumps, and close the feedwater pump discharge valves.

4) Trip of the fast acting main steam line isolation valves (designed to close in less than 7 seconds) on:

   a) Hi-2 Containment pressure.

   b) Safety Injection System actuation derived from two out of three low steam line pressure signals in any one main steam line (above P-11).

   c) High negative steam pressure rate indication from two signals in any main steam line (below Permissive P-11).

Fast-acting isolation valves are provided in each main steam line; these valves will fully close within five seconds of actuation following a large break in the steam line. For breaks downstream of the isolation valves, closure of all valves would completely terminate the blowdown. For any break, in any location, no more than one steam generator would experience an uncontrolled blowdown even if one of the isolation valves fails to close. A description of steam line isolation is included in Chapter 10.

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

The AFW system is designed to detect conditions indicative of a steam line or feedwater line break and automatically isolate AFW flow to the affected Steam Generator. For the MSLB Event, this helps to reduce the heat removal via the ruptured Steam generator loop and thus reduces the reactivity feedback to the "affected" sector of the core.

Table 15.1.5-1 lists the equipment required in the recovery from a high energy line rupture. Not all equipment is required for any one particular break, since the requirements will vary depending upon postulated break location and details of balance of plant design and pipe rupture criteria as discussed elsewhere in the FSAR. Design criteria and methods of protection of safety-related equipment from the dynamic effects of postulated piping ruptures are provided in Section 3.6.

15.1.5.2 Analysis of Effects and Consequences

Method of Analysis - Four base scenarios were considered. The reactor was assumed to be initially operating at either hot full power (HFP) or hot zero power (HZP) conditions. From both of these initial conditions, the transient was assumed to occur either with or without offsite power. With offsite power available, reactor coolant pumps remained operating throughout the entire transient. With loss of offsite power, the pumps were tripped at transient initiation. For these base scenarios, the most reactive control rod was assumed to be stuck out of the core.

The HFP cases were initiated at a nominal power level of 2900 MWt. The results of HFP cases are driven by the maximum rate of positive moderator reactivity insertion, which is predominantly a function of the largest break flow rate (i.e. largest break size) and the most negative moderator temperature coefficient (MTC). The maximum break size and most negative MTC remain unchanged for the Measurement Uncertainty Recapture (MUR) uprate to a rated condition of 2948 MW. The extent of cooldown of the RCS will be essentially the same for MUR as for the nominal starting condition of 2900 MW. The increase in rated thermal power for the MUR does not significantly change the system response and the analysis starting from a nominal condition of 2900 MW is acceptable.

The methodology for analyzing MSLB events is documented in Reference 15.1.5-3 and 15.1.5-9. Four computer codes are used in the analysis of this event: ANF-RELAP (Reference 15.1.2-3), XTGPWR (Reference 15.1.5-4), PRISM (Reference 15.1.5-10) and XCOBRA-IIIC (References 15.1.2-4 and 15.1.2-5). First, ANF-RELAP is used to calculate the general system thermal-hydraulic responses during an MSLB. ANF-RELAP produces a set of core boundary conditions for each case analyzed. The boundary conditions include core inlet flows and temperatures, core exit pressure, and core averaged fuel surface heat flux. ANF-RELAP monitors the MDNBR with the W-3 CHF correlation. The time of MDNBR as calculated by ANF-RELAP is used as a time for transferring boundary conditions to PRISM and XCOBRA-IIIC. In addition, the time of peak power as calculated by ANF-RELAP was also used as a time for transferring boundary conditions.

Second, PRISM is used to calculate the axial and radial power distributions and reactivity at the time of MDNBR and peak power as calculated by ANF-RELAP. The calculated power distributions are inputs to XCOBRA-IIIC. The reactivity calculated by PRISM is compared with the reactivity calculated by ANF-RELAP to ensure that conservatism exists in the ANF-RELAP representation of reactivity feedback.

Third, XCOBRA-IIIC is used to calculate detailed core flow and enthalpy distributions, which are coupled to the PRISM calculations. The iterative process between PRISM and XCOBRA-IIIC basically involves calculating nodal power distributions with PRISM for use in XCOBRA-IIIC and calculating nodal water density distributions with XCOBRA-IIIC for use in PRISM.

Fuel failure analysis is based on DNB and fuel centerline melt criteria. DNB criterion is evaluated using the XCOBRA-IIIC code and is based on the Biasi (Reference 15.1.5-2) correlation. DNB propagation is considered. Fuel centerline melt criterion is evaluated using the maximum post-scram core average heat flux from ANF-RELAP and the nuclear heat flux hot channel factor from PRISM.

The assumptions used in the analysis of this event are as follows:

1) A double ended guillotine break is assumed to occur in the loop 1 steam line, downstream of the flow restrictor and upstream of the main steam isolation valve (MSIV). A double ended guillotine break will lead to maximum steam flow out of the break which results in maximum cooldown and maximum return to power. The flow is choked at the flow restrictor, having an area of 1.4 ft^2. Only steam is allowed to flow out the break. The break flows are calculated based on the Moody critical flow model. On the steam generator side of the break, steam flows out the break throughout the entire transient. On the MSIV side of the break, break flow terminates after the MSIVs are fully closed. The MSIVs are fully closed 7 seconds after receiving the isolation signal.

2) For Conservatism, no steam generator tube plugging is assumed for the MSLB analysis.

3) The single failure assumed in this analysis is loss of one of two Charging/Safety Injection Pumps (CSIP's). Although three CSIP's are physically present, only two of the three are electrically connected and considered operational at any given time. Therefore, in the analysis, only one CSIP is assumed to be available to mitigate the event. The pump is assumed to take suction from the refueling water storage tank (RWST) at 40°F with a boron concentration of 2400 ppm and discharge through the boron injection tank (BIT). Initially, the BIT is assumed to be filled with unborated water. The water in the injection lines between the RWST and cold legs is also assumed to be unborated. The time required to flush this unborated water from the BIT and the injection lines is included in the ANF-RELAP model. The single failure assumed in this analysis plays an important role in ensuring long term subcriticality in the core.

4) At break initiation, normal main feed water (MFW) is assumed. MFW is isolated 8 seconds after receiving the isolation signal. Auxiliary feed water (AFW) is assumed to start at break initiation. The AFW pumps are assumed to take suction from the condensate storage tank (CST) at 40°F. All the AFW is assumed to be delivered to the broken steam generator. Although isolation of AFW flow to the faulted Steam Generator will occur within about 60 seconds, the analysis conservatively delayed AFW isolation until 600 seconds.

5) The reactor kinetics in ANF-RELAP is calculated using a point kinetics model. The total reactivity calculated by ANF-RELAP is compared with the reactivity calculated by PRISM at selected time points during the transient to ensure that conservatism exists in the ANF-RELAP representation of reactivity feedback. PRISM provides a three-dimensional, two-group diffusion calculation of the reactivity feedback.

6) The minimum technical specification value of moderator temperature coefficient (MTC), -50 pcm/°F is used in the MSLB analysis. Other neutronic parameters, are based on nominal end-of-cycle (EOC) values.

This analysis considers only the response of the primary and secondary systems and does not include containment response for structural or equipment qualification purposes.

<u>Results -</u>

The most limiting case with respect to fuel centerline melting was the HZP MSLB with offsite power available and with the stuck rod. The most limiting case with respect to the approach to DNBR was the HZP MSLB with offsite power available and with the stuck rod. The sequence of events for this case is summarized in Table 15.1.5-3. In addition, some key system parameters describing the transient are illustrated in Figures 15.1.5-1 through 15.1.5-6. With offsite power available, the primary reactor coolant pumps (RCP) operated throughout the event. In all cases, a 37-second delay in HHSI actuation time was assumed to account for diesel startup time and the time to switch suction from the volume control tank to the refueling water storage tank. The transient was terminated at 600 seconds.

As shown in Figure 15.1.5-1, the pressure in all three steam generators declined immediately after transient initiation due to steam flow out of the break (Figure 15.1.5-2). After MSIV closure, the flow from the unaffected steam generators was terminated and the pressure

recovered. The unaffected steam generators' pressure declined slowly thereafter due to heat transfer from the shell side fluid to the primary side fluid. Once the affected steam generator dried out, the secondary side pressure declined.

The break flow (Figure 15.1.5-2) from the unaffected steam generator was terminated when the MSIVs closed. After reaching an early maximum, the break flow from the affected steam generator gradually declined throughout the remainder of the transient. (The oscillatory behavior in the affected steam generator pressure and break flow beginning at approximately 360 seconds was due to the continual dryout and rewetting of the steam generator tubes.)

The mass inventory (Figure 15.1.5-3) in all three steam generators decreased until the MSIVs closed. Thereafter, the mass in the unaffected steam generators stabilized and remained essentially constant. The mass in the affected steam generator continued to decrease until it dried out at approximately 360 seconds.

The pressurizer pressure (Figure 15.1.5-4) decreased rapidly during the initial phase of the transient due to thermal contraction of the primary system coolant. The pressurizer pressure began to recover at approximately 40 seconds due to the addition of inventory via the SIS.

The reactor power was initially at 1 W. The cooldown resulted in power increasing significantly at about 20 seconds (Figure 15.1.5-5), reaching a peak of approximately 817 MW at 260 seconds, and then began to decline due to boron injection.

Initially, the total core reactivity (Figure 15.1.5-6) increased due to moderator and Doppler feedback associated with the primary cooldown. Once the reactor power began to increase, the Doppler feedback changed from positive to negative and the reactor was brought to a quasi steady-state. The boron component of reactivity began to show an effect at approximately 260 seconds as boron from the HHSI system began to reach the upper part of the stuck rod region. When the primary-to-secondary heat transfer deteriorated due to the affected steam generator dryout, the primary system began to heat up causing the Doppler reactivity to slightly increase.

Long Term Effects and Events - Following the hypothetical steamline break incident, a steamline isolation signal will be generated almost immediately, causing the steamline isolation valves to close within a few seconds. If the break is downstream of the isolation valves, all of which subsequently close, the break will be isolated. If the break is upstream of the isolation valves, or if one valve fails to close, the break will be isolated to two steam generators while the faulted one will continue to blow down. Only the case in which the steam generator continues to blow down is discussed here since the downstream break followed by isolation of all steam generators will terminate the transient.

An excessive cooldown protection signal will cause main feedwater isolation to occur. The only source of water available to the faulted steam generator is then the auxiliary feedwater system. The first required operator action is to identify the faulted steam generator and then isolate the auxiliary feedwater flow to that steam generator if this has not been accomplished by the Automatic AFW Isolation System. Table 15.1.5-7 provides a complete assessment of the operator's role. Following steamline isolation, steam pressure in the steamline with the faulted steam generator will continue to fall rapidly, while the pressure stabilizes in the remaining two steam lines. The indication of the different steam pressures will be available to the operator within a few seconds of the steamline isolation. This will provide the necessary information to identify the faulted steam generator so that auxiliary feedwater to it can be isolated. The

operator is instructed by Emergency Operating Procedures to isolate the affected steam generator by shutting the steam generator AFW isolation valve. Manual controls are provided in the control room for start and stop of the AFW pumps and for the control and isolation valves associated with the AFW system. The means for detecting the faulted steam generator and isolating auxiliary feedwater to it requires only the use of safety grade equipment available following the break.

Following the automatic safety injection actuation and after the faulted steam generator is completely isolated, the continued operation of the safety injection system will repressurize the reactor coolant system and continue to increase the RCS volume inventory. The second required operator action is to manually control the repressurization of the reactor coolant system and stop one Charging/Safety Injection Pump (CSIP) to control pressurizer level. The operator may then establish normal charging and letdown to restore pressure and level control. The operator has available, in the control room, an indication of pressurizer level from the instrumentation in the reactor protection system. After the indicated water level returns to the pressurizer, RCS pressure has returned to the normal range and the RCS is sufficiently subcooled, the operator is instructed to stop the CSIP's, reestablish normal charging and letdown flows, and reestablish operation of the pressurizer heaters to maintain a steam bubble in the pressurizer to limit system repressurization. The pressurizer level instrumentation and manual controls for operation of the CSIP's meet the required standards for safety systems.

The removal of decay heat in the long-term (following the initial cooldown) using the remaining intact steam generators requires only the auxiliary feedwater system as a water source and the secondary system safety valves to relieve steam.

The requirements to terminate auxiliary feedwater flow to the faulted steam generator, reestablishing normal charging and letdown flows, and reestablishing operation of the pressurizer heaters can be met by simple switch actions by the operator. Thus the required actions to limit the cooldown and repressurization can be easily recognized, planned and performed within the necessary time. For decay heat removal and plant cooldown the operator has a considerably longer time period in which to respond because of the large initial cooldown associated with a steamline break transient.

15.1.5.3 Radiological Consequences of a Postulated Steamline Break

The analysis of the steam line break (SLB) radiological consequences uses the analytical methods and assumptions outlined in the RG 1.183, Appendix E. A summary of input parameters and assumptions is provided in Table 15.1.5-5. Additional clarification is provided as follows:

a) The noble gas activity concentration in the RCS at the time the accident occurs is based on a one-percent fuel defect level. This is approximately equal to the Technical Specification value of 100/E bar μCi/gm for gross radioactivity. The noble gas concentrations in the RCS are given in Table 15.0.9-2. The iodine activity concentration of the secondary coolant at the time the SLB occurs is assumed to be equivalent to the Technical Specification limit of 0.1 μCi/gm of DE I-131. The iodine secondary coolant activity concentration is given in Table 15.0.9-7.

b) The amount of primary to secondary SG tube leakage is assumed to be 1 gpm total. The primary to secondary SG tube leakage is apportioned between the affected and

unaffected SGs to provide a conservative result. Leakage to the affected (faulted) SG is directly released to the atmosphere thus using 0.35 gpm to the affected SG and 0.65 gpm to the two unaffected SGs would maximize the dose.

c) The SG connected to the broken steam line is assumed to boil dry within the initial two minutes following the SLB. The entire liquid inventory of this SG is assumed to be steamed off and all of the iodine initially in this SG is released to the environment. In addition, iodine carried over to the faulted SG by tube leaks is assumed to be released directly to the environment with no credit taken for iodine retention in the SG.

d) In the intact SGs an iodine partition factor of 0.01 (curies iodine/gm steam)/ (curies iodine/gm water) is used.

e) All noble gas activity carried over to the secondary side through SG tube leakage is assumed to be immediately released to the outside atmosphere.

f) Eight hours after the accident, the RHR System is assumed to be placed into service for heat removal. After eight hours there are no further steam releases to the atmosphere from the intact steam generators.

g) Within 40 hours after the accident, the reactor coolant system has been cooled to below 212°F, and there are no further steam releases to atmosphere from the faulted steam generator.

15.1.5.3.1 Pre-accident Iodine Spike Case

It is assumed that a reactor transient has occurred prior to the SLB and has raised the RCS iodine concentration to the Technical Specification limit for a transient of 60 μCi/gm of dose equivalent (DE) I-131. The pre-accident spike iodine concentrations are given in Table 15.0.9-7.

15.1.5.3.2 Accident-Initiated Iodine Spike Case

The reactor trip associated with the SLB creates an iodine spike in the RCS which increases the iodine release rate from the fuel to the RCS to a value 500 times greater than the release rate corresponding to a maximum equilibrium RCS concentration of 1.0 μCi/gm of DE I-131. The iodine spike appearance rates are given in Table 15.0.9-6. The duration of the accident-initiated iodine spike is limited by the amount of activity available in the fuel-clad gap. Based on having 12 percent of the iodine in the fuel-clad gap, the gap inventory would be depleted within 5.0 hours and the spike is terminated at that time.

15.1.5.3.3 Fuel Failure Case

1% of the fuel is assumed to fail releasing its gap activity. Iodines, noble gases and alkali metals are considered in the fuel failure case. All of the gap activity is released from the fuel. Additionally, the radial peaking factor of 1.73 was applied to the failed fuel inventory as indicated in RG 1.183. The core fission product inventory is given in Table 15.0.9-1.

15.1.5.3.4 Offsite Doses

The offsite doses are calculated using the assumptions and equations in Section 15.0A.1

15.1.5.3.5 Control Room Doses

The control room assumptions are provided in Section 15.6.5.4.3 and Table 15.6.5-15. For the pre-accident and accident initiated iodine spike cases 500 cfm of control room unfiltered inleakage was modeled.

In the event of SLB, the low steamline pressure SI setpoint will be reached shortly after event initiation. The SI signal causes the control room HVAC to switch from the normal operation mode to the post-accident recirculation mode of operation. It is assumed that the Si signal is generated at 10 seconds. The control room HVAC switches from normal operation to post-accident recirculation mode of operation at 25 seconds (10 seconds for SI signal plus 15 second delay time). Two hours after the control room HVAC is in post-accident recirculation mode an operator action switches the control room to the pressurization mode.

15.1.5.3.6 Results

The potential radiological consequences resulting from the occurrence of a postulated main steam line break have been conservatively analyzed, using assumptions and modes in the previous sections. The TEDE doses have been analyzed for the worst two hours at the EAB and for the duration of the event at the LPZ and in the control room. The analytically predicted dose consequence to the Control Room (CR) operators due to LBLOCA increased by a small amount, based on the reduction in CR recirculation flow which reduces the iodine filtration provided by the charcoal filters relative to the analysis of record condition. The analytically predicted dose consequence to the control room operators for the MSLB event was not similarly revised or updated since the LBLOCA DBA is the limiting control room operator dose event. The radiological analysis results for this event are listed in Table 15.1.5-6. The resultant doses are within the applicable limits.

15.1.5.4 Conclusions

The methodology for analyzing the MSLB event provided a conservative method of calculating the system and core responses during an MSLB. The most limiting scenario from the fuel centerline melt standpoint was the case initiated from HZP with offsite power available and with the stuck rod. The most limiting scenario from the MDNBR standpoint was the case initiated from HZP with offsite power available and with the stuck rod. Although DNB and possible clad perforation following a steam pipe rupture are not necessarily unacceptable and not precluded by the acceptance criteria presented in Section 15.1.5.1, the analysis shows that no fuel failure is predicted to occur as a result of this accident. The radiological consequences presented in Section 15.1.5.3 are within the applicable limits.

This analysis supports nominal primary T_{avg} operation at full power between 588.8 and 580.8°F, inclusive.

REFERENCES: SECTION 15.1

15.1.2-1 Deleted by Amendment No. 48.

15.1.2-2 Deleted by Amendment No. 45.

15.1.2-3 ANF-RELAP Methodology for Pressurized Water Reactors: Analysis of Non-LOCA Chapter 15 Events, ANF-89-151(P)(A), Advanced Nuclear Fuels Corporation, Richland, WA 99352, May 1992.

15.1.2-4 Application of Exxon Nuclear Company PWR Thermal Margin Methodology to Mixed Core Configurations, XN-NF-82-21(P) (A), Revision 1, Exxon Nuclear Company, Richland, WA 99352, September 1983.

15.1.2-5 XCOBRA-IIIC: A Computer Code to Determine the Distribution of Coolant During Steady-State and Transient Core Operation, XN NF 75 21(P)(A), Revision 2, Exxon Nuclear Company, Richland, WA 99352, January 1986.

15.1.4-1 Deleted by Amendment No. 48.

15.1.5-1 Deleted by Amendment No. 48.

15.1.5-2 L. Biasi et al, "Studies on Burnout, Part 3 - A New Correlation For Round Ducts and Uniform Heating and Its Comparison with World Data", Energia Nucleare, vol 14, No. 9, September 1967.

15.1.5-3 Steamline Break Methodology for PWRs, EMF-84-093(P)(A) Revision 1 Siemens Power Corporation, Richland, WA 99352, February 1999.

15.1.5-4 XTG: A Two Group Three-Dimensional Reactor Simulator Utilizing Coarse Mesh Spacing, XN-CC-28(A), Revision 3, Exxon Nuclear Power Company, Richland, WA 99352, September 1975.

15.1.5-5 Deleted by Amendment No. 51

15.1.5-6 Deleted by Amendment No. 48.

15.1.5-7 Deleted by Amendment No. 51

15.1.5-8 EMF-92-153(P)(A) Revision 1, HTP: Departure from Nucleate Boiling Correlation for High Thermal Performance Fuel, January 2005.

15.1.5-9 EMF-84-093(P)(A) Revision 1, "Steam Line Break Methodology for PWRs", Siemens Power Corporation, February 1999.

15.1.5-10 EMF-96-029(P)(A) Volumes 1 and 2, "Reactor Analysis System for PWRs Volume 1 - Methodology Description, Volume 2 - Benchmarking Results", Siemens Power Corporation, January 1997.

15.1.5-11 EMF-2343(P), Revision 1, Harris Nuclear PLant Cycle 10 Safety Analysis Report, Siemens Power Corporation, April 2000.

15.2 DECREASE IN HEAT REMOVAL BY THE SECONDARY SYSTEM

A number of transients and accidents have been postulated which could result in a reduction of the capacity of the secondary system to remove heat generated in the Reactor Coolant System (RCS). Detailed analyses are presented in this section for several such events which have been identified as more limiting than the others.

Discussion of the following RCS coolant heatup events are presented in Section 15.2:

1) Steam pressure regulator malfunction or failure that results in decreasing steam flow (ANS Condition II event).

2) Loss of external electrical load (ANS Condition II event).

3) Turbine trip (ANS Condition II event).

4) Inadvertent closure of main steam isolation valves (ANS Condition II event.

5) Loss of condenser vacuum and other events resulting in turbine trip (ANS Condition II event).

6) Loss of nonemergency AC power to the station auxiliaries (ANS Condition II event).

7) Loss of normal feedwater flow (ANS Condition II event).

8) Feedwater system pipe break (ANS Condition IV event).

Section 15.0.1 contains a discussion of ANS classifications.

15.2.1 STEAM PRESSURE REGULATOR MALFUNCTION OR FAILURE THAT RESULTS IN DECREASING STEAM FLOW

There are no steam pressure regulators in the Shearon Harris Nuclear Power Plant (SHNPP) whose failure or malfunction could cause a steam flow transient.

15.2.2 LOSS OF EXTERNAL ELECTRICAL LOAD

15.2.2.1 Identification of Causes and Accident Description

A major load loss on the plant can result from loss of external electrical load due to some electrical system disturbance. Offsite alternating current (AC) power remains available to operate plant components such as the reactor coolant pumps; as a result, the onsite emergency diesel generators are not required to function for this event. Following the loss of generator load, an immediate fast closure of the turbine control valves will occur. This will cause a sudden reduction in steam flow, resulting in an increase in pressure and temperature in the steam generator shell. As a result, the heat transfer rate in the steam generator is reduced, causing the reactor coolant temperature to rise, which in turn causes reactor coolant expansion, pressurizer insurge, and RCS pressure rise.

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 if a safety limit were approached. A continued steam load of approximately 5 percent would exist after total loss of external electrical load because of the steam demand of plant auxiliaries.

In the event that a safety limit is approached, protection would be provided by the high pressurizer pressure and overtemperature ΔT trips. Voltage and frequency relays associated with the reactor coolant pump provide no additional safety function for this event. Following a complete loss of external load, the maximum turbine overspeed would be approximately 8 to 9 percent, resulting in an overfrequency of less than 6 hertz (hz). Any degradation in their performance could be ascertained at that time. Any increased frequency to the reactor coolant pump motors will result in slightly increased flow rate and subsequent additional margin to safety limits. For postulated loss of load and subsequent turbine generator overspeed, any overfrequency condition is not seen by other Reactor Protection System equipment. Reactor Protection System equipment is supplied from the 118 volt AC instrument power supply system, which in turn is supplied from the inverters; the inverters are supplied from a direct current (DC) bus energized from batteries or by a rectified AC voltage from safeguards buses. Safeguards loads are transferred to offsite power or, alternately, to standby diesel generators upon loss of the turbine.

In the event the steam dump valves fail to open following a large loss of load, the steam generator safety valves may lift and the reactor may be tripped by the high pressurizer pressure signal, the high pressurizer water level signal, or the overtemperature ΔT signal. The steam generator shell side pressure and reactor coolant temperatures will increase rapidly. The pressurizer safety valves and steam generator safety valves are, however, sized to protect the RCS and steam generator against overpressure for all load losses without assuming the operation of the Steam Dump System, pressurizer spray, pressurizer power operated relief valves, automatic rod cluster control assembly control, or direct reactor trip on turbine trip.

The steam generator safety valve capacity is sized to remove the steam flow from the steam generator without exceeding 110 percent of the steam system design pressure. The pressurizer safety valves capacity are sized based on a complete loss of heat sink with the plant initially operating at the maximum calculated turbine load, along with operation of the steam generator safety valves. The pressurizer safety valves are then able to relieve sufficient steam to maintain the RCS pressure within 110 percent of the RCS design pressure.

A more complete discussion of overpressure protection can be found in Section 5.2.2.

A loss of external load is classified as an ANS Condition II event, fault of moderate frequency. See Section 15.0.1 for a discussion of Condition II events.

The primary side transient is caused by a decrease in heat transfer capability from primary to secondary due to a rapid termination of steam flow to the turbine, accompanied by an automatic reduction of feedwater flow (should feed flow not be reduced, a larger heat sink would be available and the transient would be less severe). Termination of steam flow to the turbine following a loss of external load occurs due to automatic fast closure of the turbine control valves in approximately 0.3 seconds. Following a turbine trip event, termination of steam flow occurs via turbine stop valve closure, which occurs in approximately 0.1 seconds. Therefore, the transient in primary pressure, temperature, and water volume will be less severe for the loss

of external load than for the turbine trip due to a slightly slower loss of heat transfer capability. Therefore, a detailed transient analysis is not presented for the loss of external load.

The protection available to mitigate the consequences of a loss of external load is the same as that for a turbine trip, as listed in Table 15.0.8-1.

15.2.2.2 Analysis of Effects and Consequences

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

Normal reactor control systems are assumed to function only if their operation results in more severe accident conditions. The engineered safety systems are not required to function. The Auxiliary Feedwater System may, however, be automatically actuated following a loss of main feedwater; this will further mitigate the effects of the transient.

The Reactor Protection System may be required to function following a complete loss of external load to terminate core heat input and prevent departure from nucleate boiling (DNB). Depending on the magnitude of the load loss, pressurizer safety valves and/or steam generator safety valves may be required to open to maintain system pressure below allowable limits. No single active failure will prevent operation of any system required to function. Refer to Reference 15.8.0-1 for a discussion of anticipated transients without trip (ATWT) considerations.

15.2.2.3 Conclusions

Based on results obtained for the turbine trip event (Section 15.2.3) and considerations described in Section 15.2.2.1, the applicable acceptance criteria for a loss of external load event are met.

15.2.3 TURBINE TRIP

15.2.3.1 Identification of Causes and Accident Description

For a turbine trip event, the reactor would be tripped directly (unless below approximately 10 percent power) from a signal derived from the turbine stop emergency trip fluid pressure and turbine stop valves. The turbine stop valves close rapidly (typically 0.1 seconds) on loss of trip fluid pressure actuated by one of a number of possible turbine trip signals. Mechanical turbine trip initiation signals include:

1) Generator trip
2) Low condenser vacuum
3) Loss of lubricating oil
4) Turbine thrust bearing failure
5) Turbine overspeed

6) Manual trip

7) DEH DC Bus Trip

Upon initiation of stop valve closure, steam flow to the turbine stops abruptly. Sensors on the stop valves detect the turbine trip and initiate steam dump and, if above 10 percent power, a reactor trip. The loss of steam flow results in an almost immediate rise in secondary system temperature and pressure with a resultant primary system transient as described in Section 15.2.2.1 for the loss of external load event. A slightly more severe transient occurs for the turbine trip event due to the more rapid loss of steam flow caused by the more rapid valve closure.

The automatic Steam Dump System would normally accommodate the excess steam generation. Reactor coolant temperatures and pressure do not significantly increase if the Steam Dump System and Pressurizer Pressure Control System are functioning properly. If the turbine condenser was not available, the excess steam generation would be dumped to the atmosphere and main feedwater flow would be lost. For this situation feedwater flow would be maintained by the Auxiliary Feedwater System to ensure adequate residual and decay heat removal capability. Should the Steam Dump System fail to operate, the steam generator safety valves may lift to provide pressure control. See Section 15.2.2.1 for a further discussion of the transient.

A turbine trip is classified as an ANS Condition II event, fault of moderate frequency. See Section 15.0.1 for a discussion of ANS Condition II events. The acceptance criteria for this event are:

1) The pressures in the reactor coolant and main steam systems should be less than 110% of design values.

2) The fuel cladding integrity should be maintained by ensuring that fuel design limits are not exceeded. This is demonstrated by assuring that the minimum calculated departure from nucleate boiling ratio (DNBR) is not less than the applicable limits of the DNBR correlation being used and that fuel centerline melt does not occur.

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

4) For transients of moderate frequency in combination with a single failure, no loss of function of any fission product barrier, other than fuel element cladding, shall occur. Core geometry is maintained in such a way that there is no loss of core cooling capability and control rod insertability is maintained.

No single active failure will prevent operation of any system required to function.

Plant systems and equipment which are available to mitigate the effects of the accident are discussed in Section 15.0.8 and listed in Table 15.0.8-1.

Reactor Trip and OTΔT. Trip response times are presented in FSAR Section 15.0.6. The primary system is protected against overpressurization by the pressurizer safety and relief valves. Pressure relief on the secondary side is afforded by the steam line safety/relief valves.

A turbine trip event is more limiting than loss of external load, loss of condenser vacuum, and other decrease in heat removal events. As such, this event has been analyzed in detail. Results and discussion of the analysis are presented in Section 15.2.3.2.

15.2.3.2 Analysis of Effects and Consequences

Method of Analysis - The purpose of analyzing this event is to demonstrate that the primary and secondary pressure relief capability is sufficient to limit the pressures to less than 110% of their respective design values. This event is also analyzed to ensure that the reactor protection system is properly set to prevent penetration of the SAFDLs under the limiting assumptions of no credit for a direct reactor trip on turbine trip.

Three cases are analyzed for this event: one challenging the primary overpressurization criterion, one challenging the secondary overpressurization criterion, and one challenging the fuel design limits. In all cases, the input parameters are biased (BOC kinetics) to maximize the increase in reactor power during the transient. However, in the first case, the parameters and equipment operational states are selected to maximize the primary system overpressurization, in the second case the parameters and equipment states have been selected to maximize the secondary side overpressurization, and in the final case the parameters and equipment states have been selected to reduce the primary system pressurization which provides a conservative estimate of the minimum DNBR during the transient.

In this analysis, the behavior of the Unit is evaluated for a complete loss of steam load from the Measurement Uncertainty Recapture (MUR) power plus uncertainty without direct reactor trip primarily to show the adequacy of the pressure relieving devices and also to demonstrate core protection margins; that is, the turbine is assumed to trip without actuating any of the sensors for reactor trip on the turbine stop valves. This assumption delays reactor trip until conditions in the RCS result in a trip due to other signals (high pressurizer pressure, OTΔT, high neutron flux, high pressurizer water level, and low-low steam generator water level). Thus, the analysis assumes a worst transient. In addition, no credit is taken for steam dump. Main feedwater flow is terminated at the time of turbine trip, with no credit taken for auxiliary feedwater to mitigate the consequences of the transient.

The MDNBR case is analyzed using the SGR/Uprating value for T_{avg} (588.8°F). By analyzing this event at 588.8°F, the DNB results bound all lower temperature operating conditions.

The primary overpressurization case is conservatively analyzed at a T_{avg} of 580.8°F which bounds operation at 588.8°F. The secondary overpressurization case is conservatively analyzed at a Tavg of 595.6°F, which bounds operation at 588.8°F.

Conservative conditions established for the analysis of this event are presented in Table 15.2.3-1. Available reactor protection system trips are presented in Table 15.0.6-2. The trip setpoints and time delays assumed in the analysis of this event are also listed in Table 15.0.6-2. All trips listed in this table except reactor trip OPΔT were modeled for the analysis of this event. In the DNB analysis, the reactor trip on steam generator high-high level is disabled. For all cases analyzed, the reactor trip on turbine trip is also disabled. Key operating parameters used in the analysis of this event are presented in Table 15.2.3-2. The range of neutronics parameters supported by this analysis are presented in Table 15.2.3-3. Part power cases were analyzed for the DNB case to assure the Technical Specification limits on MTC are supported.

The primary and secondary system overpressure cases assume an MSSV setpoint tolerance of +3% and an initial pressurizer water level of 75% plus uncertainty. The primary overpressurization case assumes a PSV setpoint tolerance of +3%. The 75% pressurizer water level is tied to the Technical Specification pressurizer water level LCO. The initial pressurizer levels assumed for all other events in this chapter are established in accordance with the applicable analysis methodology and are not associated with the pressurizer water level LCO (Reference 15.2.3-4).

Major assumptions are summarized below:

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

2) Steam Release - No credit is taken for the operation of the Steam Dump System or steam generator power operated relief valves.

3) Pressurizer Spray and Power Operated Relief Valves:

   a) For the secondary side overpressurization and the MDNBR cases, the pressurizer spray and power operated relief valves are conservatively assumed to operate in reducing or limiting the reactor coolant pressure. Safety valves are also available.

   b) For the primary side overpressurization case, no credit is taken for the effect of pressurizer spray and power operated relief valves in reducing or limiting the reactor coolant pressure. Safety valves are operable.

4) Feedwater Flow - Main feedwater flow to the steam generators is assumed to be lost at the time of turbine trip. No credit is taken for auxiliary feedwater flow since a stabilized plant condition will be reached before auxiliary feedwater initiation is normally assumed to occur; however, the auxiliary feedwater pumps would be expected to start on a trip of the main feedwater pumps. The auxiliary feedwater flow would remove core decay heat following plant stabilization.

5) Reactor trip is actuated by the first reactor protection system trip setpoint reached with no credit taken for the direct reactor trip on the turbine trip. Trip signals are expected due to high pressurizer pressure, overtemperature ΔT, high neutron flux, high pressurizer water level, and low-low steam generator water level.

The transient response of the reactor system for the DNB case is calculated using the ANF-RELAP (Reference 15.2.3-1) computer program. The core thermal hydraulic boundary conditions from the ANF-RELAP calculation are used as input to the XCOBRA-IIIC code (References 15.2.3-2 and 15.2.3-3) to predict the MDNBR for the event. The transient resonse of the reactor system for the overpressurization cases is calculated using the RETRAN-3D (Reference 15.2.3-5) computer program.

Results - For the overpressurization cases, the peak reactor primary system pressure and the peak secondary system pressures are less than 110% of design limits. The primary and secondary overpressurization transients tripped on high pressurizer pressure and high

pressurizer water level, respectively. The DNB transient tripped on OTΔT. The sequences of events for the overpressurization cases are given in Table 15.2.3-4 and Table 15.2.3-5. The responses to key system variables for the primary overpressurization case are given in Figures 15.2.3-1 to 15.2.3-4 and for the secondary side overpressurization, these responses are presented in Figures 15.2.3-9 to 15.2.3-12.

The sequence of events for the MDNBR case is given in Table 15.2.3-6. The responses to key system variables are given in Figures 15.2.3-5 to 15.2.3-7. Part power cases indicated that the full power cases with 0.0 MTC was the limiting DNB case.

15.2.3.3 Conclusions

The results of the analysis demonstrate that the event acceptance criteria are met. For the overpressurization cases, both the maximum reactor primary system pressure and the maximum secondary system pressure are less than the design limits of 2750 psia and 1320 psia, respectively. For the MDNBR case, the predicted MDNBR is greater than the safety limit. The critical heat flux correlation limit ensures that, with 95% probability and 95% confidence, DNB is not expected to occur; therefore, no fuel is expected to fail. This event does not pose a credible challenge to FCM safety criteria.

The analysis for this event supports full power operation between nominal primary T_{avg} of 580.8°F and 588.8°F, inclusive.

15.2.4 INADVERTENT CLOSURE OF MAIN STEAM ISOLATION VALVES

Inadvertent closure of the main steam isolation valves (MSIVs) would result in a complete loss of steam flow similar to but less severe than the turbine trip event analyzed in Section 15.2.3. The main steam line isolation valves close more slowly than the turbine stop valves resulting in less severe transient. Therefore, this event is bounded by the results of the analysis for Event 15.2.3.

This event is classified as a Condition II event. See Section 15.0.1 for a discussion of ANS Condition II events.

15.2.5 LOSS OF CONDENSER VACUUM AND OTHER EVENTS RESULTING IN TURBINE TRIP

Loss of condenser vacuum is one of the events that can cause a turbine trip. Turbine trip initiating events are described in Section 15.2.3. A loss of condenser vacuum would preclude the use of steam dump to the condenser; however, since steam dump is assumed not to be available in the turbine trip analysis, no additional adverse effects would result if the turbine trip were caused by loss of condenser vacuum. Therefore, the analysis results and conclusions contained in Section 15.2.3 apply to loss of condenser vacuum.

This event is classified as a Condition II event. See Section 15.0.1 for a discussion of ANS Condition II events.

15.2.6 LOSS OF NONEMERGENCY AC POWER TO THE STATION AUXILIARIES

15.2.6.1 Identification of Causes and Accident Description

A complete loss of nonemergency AC power may result in the loss of all power to the station auxiliaries, i.e., the reactor coolant pumps, condensate pumps, etc. The loss of power may be caused by a complete loss of the offsite grid accompanied by a turbine generator trip at the station, or by a loss of the onsite AC distribution system. Loss of main feedwater occurs on loss of nonemergency AC power. The combination of the decrease in primary coolant flow rate, the cessation of main feedwater flow and trip of the turbine generator compounds the event consequences. The decrease of main feedwater to the steam generators decreases the primary-to-secondary system heat transfer rate resulting in heatup of the primary coolant system. The increase in primary coolant temperature results in overpressurization of the RCS.

The reactor will trip: 1) due to turbine trip; 2) upon reaching one of the trip setpoints in the primary and secondary systems, as a result of the flow coastdown and decrease in secondary heat removal; or 3) due to loss of power to the control rod drive mechanisms, as a result of the loss of power to the plant.

Following a loss of AC power with turbine and reactor trips, the sequence described below will occur:

1) Plant vital instruments are supplied from emergency DC power sources.

2) As the steam system pressure rises following the trip, the steam generator power operated relief valves may be automatically opened to the atmosphere. Steam dump to the condenser is assumed not to be available. If power is available, power operated relief valves will open and control the steam pressure eliminating the pressure rise from opening the self-actuated safety valves. The self-actuated safety valves will lift to dissipate the sensible heat of the fuel and reactor coolant plus the residual decay heat produced in the reactor.

3) As the no-load temperature is approached, the steam generator power operated relief valves (or the self-actuated safety valves, if the power operated relief valves are not available) are used to dissipate the residual decay heat and to maintain the plant at the hot standby condition.

4) The standby diesel generators, started on loss of voltage on the plant emergency buses, begin to supply plant vital loads.

The Auxiliary Feedwater System is started automatically as follows:

Two motor driven auxiliary feedwater pumps are started on any of the following:

1) Low-low level in any steam generator.

2) Trip of all main feedwater pumps.

3) A safety injection signal.

4) Loss of offsite power.

5) Manual actuation.

One turbine driven auxiliary feedwater pump is started on any of the following:

1) Low-low level in any two steam generators.

2) Loss of offsite power.

3) Manual actuation.

Refer to Section 10.4.9 for a description of the Auxiliary Feedwater System.

The motor driven auxiliary feedwater pumps are supplied power by the diesel generators and the turbine driven pump utilizes steam from the Main Steam System. Both type pumps are designed to start within one minute even if a loss of all AC power occurs simultaneously with loss of normal feedwater. The turbine exhausts the secondary steam to the atmosphere. The auxiliary feedwater pumps take suction from the condensate storage tank for delivery to the steam generators.

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

A loss of nonemergency AC power to the station auxiliaries is classified as an ANS Condition II event, a fault of moderate frequency. See Section 15.0.1 for a discussion of ANS Condition II events. The acceptance criteria for this event are:

1) The pressures in the reactor coolant and main steam systems should be less than 110% of design values.

2) The fuel cladding integrity should be maintained by ensuring that fuel design limits are not exceeded. This is demonstrated by assuring that the minimum calculated departure from nucleate boiling ratio (DNBR) is not less than the applicable limits of the DNBR correlation being used and that fuel centerline melt does not occur. In addition, the AFW system response maintains a sufficient steam generator secondary side water inventory to remove long term decay heat from the primary side (i.e., steam generator dryout is avoided).

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

4) For transients of moderate frequency in combination with a single failure, no loss of function of any fission product barrier, other than fuel element cladding, shall occur. Core geometry is maintained in such a way that there is no loss of core cooling capability and control rod insertability is maintained.

An actual loss of nonemergency AC power would cause a loss of all feedwater, loss of reactor coolant flow (flow coastdown) and a reactor trip due to either loss of power to the RCCA's or any

of the primary coolant flow trips within 1.5 seconds of the loss of AC power. The analysis presented in 15.2.6.2 is more conservative than an actual loss of AC in that it does not assume that these events occur simultaneously.

At 0 seconds, the loss of AC power and the resulting loss of feedwater occurs. However, the reactor trip and loss of RCS flow, which would normally occur, is not assumed to happen at this time. This causes the primary side coolant to heat up and the steam generator inventory to decrease. The reactor is finally tripped on a low-low steam generator level signal, and at this time, the loss of primary flow due to the loss of AC is assumed to occur.

The above assumptions are more conservative than an actual loss of nonemergency AC because the reactor power is maintained following the loss of AC/loss of feedwater at 0 seconds. This minimizes the steam generator heat transfer capability and increases the amount of RCS stored energy at the time of reactor trip and loss of primary coolant flow.

Following the reactor coolant pump coastdown caused by the loss of AC power, the natural circulation capability of the RCS will remove residual and decay heat from the core, aided by auxiliary feedwater in the secondary system. An analysis is presented here to show that the natural circulation flow in the RCS following a loss of AC power event is sufficient to remove residual heat from the core.

The plant systems and equipment available to mitigate the consequences of a loss of AC power event are discussed in Section 15.0.8 and listed in Table 15.0.8-1.

The worst single failure for this event is either the failure of an available auxiliary feedwater pump or the failure of an auxiliary feedwater isolation valve in the closed position. However, in compliance with SHNPP's TMI Action Plan Item II.E.1.1, this analysis conservatively accommodates failure of two of the three pumps. Furthermore, the analysis is performed with and without an additional failure of an auxiliary feedwater isolation valve in the closed position. These two scenarios are analyzed as separate cases. The case with a single AFW pump delivering to only two Steam Generators represents an especially conservative combination of multiple failures.

15.2.6.2 Analysis of Effects and Consequences

Method of Analysis - The transient response of the reactor system is calculated using the ANF-RELAP (Reference 15.2.3-1) computer program. Initial conditions, trip setpoints, plant systems availability, and boundary conditions are conservatively adjusted to maximize the calculated pressurizer water level.

The assumptions used in the analysis are as follows:

1) The plant is initially operating at 102 percent of the engineered safety features design rating.

2) Reactor trip occurs on steam generator low-low level. No credit is taken for immediate release of the control rod drive mechanisms caused by a loss of offsite power.

3) Auxiliary feedwater system delivers flow from one or two auxiliary feedwater pumps (see Section 15.0.13 for explanation of failure mode assumptions). For conservatism in this

analysis the flow capacity of one motor driven auxiliary feedwater pump is assumed to be 390[b] gpm and 760 gpm for two pumps.

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

5) The pressurizer power operated relief valves and pressurizer spray are conservatively assumed to function. This maximizes the peak transient pressurizer water volume.

6) RCS pump coastdown is conservatively assumed to occur at reactor scram to maximize primary system heatup and minimize steam generator inventory.

7) The worst single failure for this event is either the failure of the available turbine driven auxiliary feedwater pump or the failure of an auxiliary feedwater isolation valve in the closed position. In addition, to be in compliance with SHNPP's TMI Action Plan Item II.E.1.1, this analysis conservatively does not take credit for the operation of 1 available auxiliary feedwater pump.

The assumptions used in the analysis are similar to the loss of normal feedwater flow incident (Subsection 15.2.7), except that power is assumed to be lost to the reactor coolant pumps at the time of reactor trip.

This event is analyzed at the SGR/Uprating value for T_{avg} (588.8°), which bounds operation at 580.8°F.

Conservative conditions established for the analysis of this event are presented in Table 15.2.6-1. Available reactor protection system trips are presented in Table 15.0.6-2. The trip setpoints and time delays assumed in the analysis of this event are also listed in Table 15.0.6-2. For the analysis of this event, all trips listed in this table were disabled except the low-low steam generator level trip. Key operating parameters used in the analysis of this event are presented in Table 15.2.6-2. The range of neutronics parameters supported by the analysis are presented in Table 15.2.6-3.

Results - The case without failure of an AFW isolation valve (with one AFW pump delivering to all three Steam Generators) was determined to be more limiting than the corresponding case with one AFW pump delivering to (only) two Steam Generators. The sequence of events is given in Table 15.2.6-4. The responses to key system variables are given in Figures 15.2.6-1 to 15.2.6-5.

The transient is initiated from 2958 MW (rated + 0.34%) power initial conditions by a sudden and complete loss of main feedwater at time zero. The loss of feedwater results in degraded heat transfer capacity by the steam generators. Due to degraded heat transfer, the primary system begins to heat up. With a conservatively modeled positive moderator temperature coefficient, the increasing primary system temperature results in an increasing reactor core power. The pressurizer sprays and PORVs operate to mitigate the pressurizer pressure increase; however, the operation of these components also maximizes the pressurizer water level.

[b] The limiting case is obtained with only a single auxiliary feedwater pump providing 390 gpm to all 3 SGs. The safety analyses support an AFW flow rate of 374 gpm at the lowest lifting MSSV setpoint plus 3% tolerance. See Section 15.0 for more details.

Without feedwater, the steam generator water level decreases until a reactor trip and subsequent turbine trip occur at approximately 22 seconds due to a low-low steam generator water level trip. The reactor scram terminates the short term primary system power increase at 103% and short term pressurizer water level increase at 78% full.

The auxiliary feedwater system is also initiated by the low-low steam generator water level trip and delivers auxiliary feedwater flow to the steam generators by 80 seconds into the transient.

By approximately 300 seconds, the reactor coolant system is in a natural circulation flow mode.

After the turbine trip occurs, the main steam safety valves (MSSVs) begin cycling to remove the decay heat load. At 2100 seconds, the steam generator inventories reach a minimum value and then the steam generators begin to refill.

The transient simulation is terminated at 10,000 seconds with the primary system being cooled and the steam generators being refilled. No operator actions are credited in this event simulation.

15.2.6.3 Radiological Consequences of a Loss of Non-Emergency AC Power to Plant Auxiliaries

A loss of non-emergency AC power to plant auxiliaries would result in a turbine and reactor trip on loss of condenser vacuum. Heat removal from the secondary system would occur through the steam generator power-operated relief valves or safety valves. Due to the pressure differential between the primary and secondary systems and assumed steam generator tube leakage, fission products are discharged from the primary into the secondary system. A portion of this radioactivity is released to the outside atmosphere through either the atmospheric relief valves or safety valves.

In addition, iodine activity is contained in the secondary coolant before the accident and some of this activity is released to the atmosphere as a result of steaming from the steam generators following the accident.

The analysis of the loss of offsite power (LOOP) radiological consequences uses the analytical methods and assumptions outlined in RG 1.183, Appendix G (Locked Rotor) for secondary system leakage release path modeling and RG 1.183 Appendix E (Main Steam Line Break) for iodine spiking.

A summary of input parameters and assumptions is provided in Table 15.2.6-5. Additional clarification is provided as follows:

a) The noble gas and alkali metal activity concentrations in the RCS at the time the accident occurs are based on a one percent fuel defect level. The noble gas and alkali metal concentrations in the RCS are given in Table 15.0.9-2. The iodine activity concentration of the secondary coolant at the time the LOOP occurs is assumed to be equivalent to the Technical Specification limit of 0.1 μCi/gm of DE I-131. The secondary coolant iodine activity concentration is given in Table 15.0.9-7. The alkali metal activity concentration of the secondary coolant at the time the LOOP occurs is assumed to be 10% of the primary side concentration.

b) The amount of primary to secondary SG tube leakage is assumed to be equal to the Technical Specification limit of 1 gpm total.

c) An iodine partition factor in the SGs of 0.01 (curies iodine/gm steam)/(curies iodine/gm water) is used. This partition factor is also applied to alkali metals.

d) All noble gas activity carried over to the secondary side through SG tube leakage is assumed to be immediately released to the outside atmosphere.

e) At 8 hours after the accident, the RHR System is assumed to be placed into service for heat removal and there is no further steam release to the atmosphere from the secondary system.

15.2.6.3.1 Pre-Accident Iodine Spike Case

It is assumed that a reactor transient had occurred prior to the LOOP and had raised the RCS iodine concentration to the Technical Specification limit for a transient of 60 μCi/gm of dose equivalent (DE) I-131. The pre-accident spike iodine concentrations are given in Table 15.0.9-7.

15.2.6.3.2 Accident Initiated Iodine Spike Case

The reactor trip associated with the LOOP creates an iodine spike in the RCS which increases the iodine release rate from the fuel to the RCS to a value 500 times greater than the release rate corresponding to a maximum equilibrium RCS concentration of 1.0 μCi/gm of DE I-131. The iodine spike appearance rates are given in Table 15.0.9-6. The duration of the accident-initiated iodine spike is limited by the amount of activity available in the fuel-clad gap. Based on having 12 percent of the iodine in the fuel-clad gap, the gap inventory would be conservatively depleted within 5.0 hours and the spike is terminated at that time.

15.2.6.3.3 Offsite Doses

The offsite doses are calculated using the assumptions and equations in Section 15.0A.1.

15.2.6.3.4 Control Room Doses

The control room assumptions are provided in Section 15.6.5.4.3 and Table 15.6.5-15. For both the pre-accident and accident initiated iodine spike cases, 500 cfm of unfiltered inleakage was modeled. The control room HVAC is switched to the emergency post-accident recirculation mode after receiving a high radiation signal. The high radiation signal is reached at 3 seconds into the event. The control room HVAC is switched over to the emergency post-recirculation mode at 18 seconds (3 second signal initiation plus 15 second delay time for switching between modes). An operator action switches the control room from the post-accident recirculation mode to the pressurization mode at 2 hours after event initiation. The 15-second delay to switch between modes was also assumed with the operator action. Thus 2 hours and 33 seconds was actually modeled from the time of operator action switchover to the pressurization mode.

15.2.6.3.5 Results

The potential radiological consequences resulting from the occurrence of a loss of non-emergency AC power have been conservatively analyzed, using assumptions and models described in previous sections. The TEDE doses have been analyzed for the worst two hours at the EAB and for the duration of the event at the LPZ and in the control room. The analytically predicted dose consequence to the Control Room (CR) operators due to LBLOCA increased by a small amount, based on the reduction in CR recirculation flow which reduces the iodine filtration provided by the charcoal filters relative to the analysis of record condition. The analytically predicted dose consequence to the control room operators for the loss of non-emergency AC power event was not similarly revised or updated since the LBLOCA DBA is the limiting control room operator dose event. The radiological analysis results for this event are listed in Table 15.2.6-6. The resultant doses are within the guideline values. For the case with an assumed pre-accident iodine spike, the calculated doses are less than the 10CFR50.67 limits of 25 rem TEDE offsite and 5 rem TEDE in the control room. For the case with an accident initiated iodine spike, the offsite doses are less than 10% of the 10CFR50.67 limits (i.e., less than 2.5 rem TEDE) and the control room doses are less than the 10CFR50.67 limit of 5 rem TEDE.

15.2.6.4 Conclusions

The results of the analysis demonstrate that the event acceptance criteria are met since overpressurization of the primary and secondary system is avoided. Pressurizer filling is avoided and fluid discharge through the pressurizer relief or safety valves does not occur. Additionally, it is demonstrated that the assumed minimum motor driven auxiliary feedwater pump capacity of 390[c] gpm is sufficient to prevent steam generator dryout and to accomplish long term decay heat removal.

The analysis for this event supports full power operation at a nominal primary T_{avg} between 580.8°F and 588.8°F, inclusive.

15.2.7 LOSS OF NORMAL FEEDWATER FLOW

15.2.7.1 Identification of Causes and Accident Description

A Loss of Normal Feedwater Flow transient is initiated by main feedwater pump failure or a malfunction in the feedwater control valves. The loss of main feedwater flow decreases the amount of subcooling in the secondary-side downcomer which diminishes the primary-to-secondary system heat transfer and leads to an increase in the primary system coolant temperature. The increase in primary coolant temperature results in overpressurization of the RCS.

The opening of the secondary-side safety valves acts to remove decay heat load and to mitigate the primary system heatup. The long-term cooling of the primary system is governed by the heat removal capacity of the auxiliary feedwater flow. The auxiliary feedwater pumps are automatically started upon a steam generator low-low water level trip.

[c] Safety analyses support an AFW flow rate of 374 gpm at the lowest lifting MSSV setpoint plus 3% tolerance. See Section 15.0 for more details.

If an alternative supply of feedwater were not supplied to the plant, core residual heat following reactor trip would heat the Reactor Coolant System water to the point where water relief from the pressurizer would occur, resulting in a substantial loss of water from the RCS. Since the plant is tripped well before the steam generator heat transfer capability is reduced, the reactor coolant system variables never approach a DNB condition.

The following events occur upon loss of normal feedwater (assuming main feedwater pump failures or valve malfunctions).

1) As the steam system pressure rises following reactor and turbine trips, the steam generator power operated relief valves are automatically opened to the atmosphere. Steam dump to the condenser is assumed not to be available. If the steam flow rate through the power operated relief valves is not available, the steam generator self-actuated safety valves will lift to dissipate the sensible heat of the fuel and reactor coolant plus the residual decay heat produced in the reactor.

2) As the no-load temperature is approached, the steam generator power operated relief valves (or the self-actuated safety valves, if the power operated relief valves are not available) are used to dissipate the residual decay heat and to maintain the plant at the hot standby condition.

A loss of normal feedwater is classified as an ANS Condition II event, fault of moderate frequency. See Section 15.0.1 for a discussion of ANS Condition II events. The acceptance criteria for this event are:

1) The pressures in the reactor coolant and main steam systems should be less than 110% of design values.

2) The fuel cladding integrity should be maintained by ensuring that fuel design limits are not exceeded. This is demonstrated by assuring that the minimum calculated departure from nucleate boiling ratio (DNBR) is not less than the applicable limits of the DNBR correlation being used and that fuel centerline melt does not occur. In addition, fission product decay heat must be transferred from the reactor coolant system following a loss of normal feedwater flow (i.e., steam generator dryout is avoided).

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

4) For transients of moderate frequency in combination with a single failure, no loss of function of any fission product barrier, other than fuel element cladding, shall occur. Core geometry is maintained in such a way that there is no loss of core cooling capability and control rod insertability is maintained.

Reactor trip on low-low water level in any steam generator provides protection for a loss of normal feedwater.

The Auxiliary Feedwater System is started automatically as discussed in Section 15.2.6.1. The steam driven auxiliary feedwater pump utilizes steam from the Main Steam System and exhausts to the atmosphere. The motor driven auxiliary feedwater pumps are supplied by

power from the diesel generators. The pumps take suction directly from the condensate storage tank for delivery to the steam generators.

An analysis of the system transient is presented below to show that following a loss of normal feedwater, the Auxiliary Feedwater System is capable of removing the stored and residual heat, thus preventing either overpressurization of the RCS or loss of water from the reactor core, and returning the plant to a safe condition.

The worst single failure for this event is the failure of the available turbine driven auxiliary feedwater pump. In compliance with SHNPP's TMI Action Plan Item II.E.1.1, this analysis conservatively does not credit the operation of 1 of 2 available motor driven auxiliary feedwater pumps.

Plant systems and equipment which are available to mitigate the effects of a loss of normal feedwater accident are discussed in Section 15.0.8 and listed in Table 15.0.8-1.

15.2.7.2 Analysis of Effects and Consequences

Method of Analysis - The transient response of the reactor system is calculated using the ANF-RELAP (Reference 15.2.3-1) computer program. The primary concern in analyzing the loss of normal feedwater event is to avoid filling the pressurizer, resulting in liquid discharge through the pressurizer relief valves and loss of primary system water inventory. Therefore, the initial conditions, trip setpoints, plant systems availability, and boundary conditions are conservatively adjusted to maximize the calculated pressurizer water level.

Assumptions made in the analysis are:

1) The plant is initially operating at 102 percent of the engineered safety features design rating.

2) Reactor trip occurs on steam generator low-low level.

3) Auxiliary feedwater system delivers flow to the three steam generators from one auxiliary feedwater pump (see Section 15.0.13 for explanation of failure mode assumptions). For conservatism in this analysis, the flow capacity of the motor driven auxiliary feedwater pump is assumed to be 390[d] gpm.

4) Secondary system steam relief is achieved through the self-actuated safety valves. Note that steam relief will, in fact, be through the power operated relief valves or condenser dump valves for most cases of loss of normal feedwater. However, for the sake of analysis these have been assumed unavailable.

5) The pressurizer power operated relief valves and pressurizer spray are assumed to function, this maximizes the peak transient pressurizer water volume.

The assumptions used in the analysis are similar to the loss of AC power incident (Subsection 15.2.6), except that the reactor coolant pumps are assumed to continue to operate.

[d] Safety analyses support an AFW flow rate of 374 gpm at the lowest lifting MSSV setpoint plus 3% tolerance. See Section 15.0 for more details.

This event is analyzed at the SGR/Uprating value for T_{avg} (588.8°F), which bounds operation at 580.8°F.

Conservative conditions established for the analysis of this event are presented in Table 15.2.7-1. Available reactor protection system trips are presented in Table 15.0.6-2. For the analysis of this event, all trips listed in this table were disabled except the low-low steam generator level trip. Key operating parameters used in the analysis of this event are presented in Table 15.2.7-2. The range of neutronics parameters supported by this analysis are presented in Table 15.2.7-3.

Results - The sequence of events is given in Table 15.2.7-4. The responses to key system variables are given in Figures 15.2.7-1 to 15.2.7-5.

The transient is initiated from full power initial conditions by a sudden and complete loss of main feedwater at time zero. The loss of feedwater results in degraded heat transfer capacity by the steam generators. Due to degraded heat transfer, the primary system begins to heat up. With a conservatively modeled positive moderator temperature coefficient, the increasing primary system temperature results in an increasing reactor core power. The pressurizer sprays and PORVs operate to mitigate the pressurizer pressure increase. However, their operation also maximizes the pressurizer water inventory and level.

Without feedwater, the steam generator water level decreases until a reactor trip and subsequent turbine trip occur at approximately 22 seconds due to a low-low steam generator water level trip. The reactor scram terminates the short term primary system power increase at 103% and short term pressurizer water level increase at 76%

The auxiliary feedwater system is also initiated by the low steam generator water level signal and delivers auxiliary feedwater flow to the steam generators by 80 seconds into the transient.

After the turbine trip occurs, the MSSVs begin cycling to remove the decay heat load. At 2480 seconds, the steam generator water levels reach a minimum and then the steam generators begin to refill. The transient simulation is terminated at 10,000 seconds with the primary system being cooled and all three steam generators being refilled. No operator actions are credited in this event simulation.

15.2.7.3 Conclusions

Results of the analysis show that a loss of normal feedwater does not adversely affect the core, the RCS, or the Main Steam System since the auxiliary feedwater capacity is such that Reactor Coolant System does not overpressurize and water is not relieved from the pressurizer power operated relief or safety valves.

The results of the analysis demonstrate that the event acceptance criteria are met. Additionally, it is demonstrated that the assumed minimum motor driven auxiliary feedwater pump capacity of 390[e] gpm is sufficient to prevent steam generator dryout and to accomplish long term decay heat removal. The radiological consequences of this event would be less severe than the steamline break accident analyzed in Section 15.1.5.

[e] Safety analyses support an AFW flow rate of 374 gpm at the lowest lifting MSSV setpoint plus 3% tolerance. See Section 15.0 for more details.

The analysis for this event supports nominal primary T_{avg} operation at full power between 588.8 and 580.8°F, inclusive.

15.2.8 FEEDWATER SYSTEM PIPE BREAK

15.2.8.1 Identification of Causes and Accident Description

A major feedwater line rupture is defined as a break in a feedwater line large enough to prevent the addition of sufficient feedwater to the steam generators 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, liquid followed by steam from the steam generator will be discharged through the break. (A break upstream of the feedline check valve would affect the Nuclear Steam Supply System only as a loss of feedwater. This case is covered by the evaluation in Section 15.2.7). Depending upon the size of the break and the plant operating conditions at the time of the break, the break could cause either a RCS cooldown (by excessive energy discharge through the break) or a RCS heatup. Potential RCS cooldown resulting from a secondary pipe rupture is evaluated in Section 15.1.5. Therefore, only the RCS heatup effects are evaluated for a feedwater line rupture.

The event follows three phases. A mild primary system heatup occurs due to the loss of feedwater to the steam generators prior to a reactor scram. (This phase may not occur under certain kinetics assumptions.) This is followed by a cooldown phase of the primary-side coolant due to the energy removal during the steam generator blowdown stage. Finally, the eventual depletion of secondary-side inventory in the ruptured steam generator and lack of main feedwater to the intact steam generators results in a long term, primary system heatup much like a Loss of Normal Feedwater Flow event.

A feedwater line rupture reduces the ability to remove heat generated by the core from the RCS for the following reasons:

1) Feedwater flow to the steam generators is reduced. Since feedwater is subcooled, its loss may cause reactor coolant temperatures to increase prior to reactor trip.

2) Fluid in the steam generator may be discharged through the break, and would then not be available for decay heat removal after trip.

3) The break may be large enough to prevent the addition of any main feedwater after trip.

Since the auxiliary feedwater (AFW) flow is injected into the steam generators via a separate piping network than the main feedwater, the delivery of auxiliary feedwater will not be interrupted by the pipe rupture. An Auxiliary Feedwater System is provided to assure that adequate feedwater will be available such that:

1) No substantial overpressurization of the RCS shall occur.

2) Sufficient liquid in the RCS shall be maintained so that the core remains in place and geometrically intact with no loss of core cooling capability.

A major feedwater line rupture is classified as an ANS Condition IV event. See Section 15.0.1 for a discussion of ANS Condition IV events. The acceptance criteria for this event are:

1) Pressure in the reactor and main steam systems should be maintained below 110% of design pressures for low probability events and below 120% of design pressures for very low probability events such as double-ended guillotine breaks.

2) The potential for core damage is evaluated on the basis that it is acceptable if the minimum DNBR remains above the 95/95 DNBR limit. If DNBR falls below the 95/95 DNBR limit, fuel failure must be assumed for all rods that do not meet this criteria unless it can be shown, based on an acceptable fuel damage model, which includes the potential adverse effects of hydraulic instabilities, that fewer failures occur. Any fuel damage calculated to occur must be of sufficiently limited extent that the core will remain in place and intact with no loss of core cooling capacity.

3) Any radioactivity release must be such that the calculated doses are within applicable guidelines. For a case with an assumed pre-accident iodine spike or accident induced fuel failure, the calculated doses should be less than the 10CFR50.67 limits of 25 rem TEDE offsite and 5 rem TEDE in the control room. For a case with an accident initiated iodine spike, the offsite doses should be less than 10% of the 10CFR50.67 limits (i.e., less than 2.5 rem TEDE) and the control room doses should be less than the 10CFR50.67 limit of 5 rem TEDE.

4) The integrity of the reactor coolant pumps should be maintained, such that loss of AC power and containment isolation will not result in seal damage.

5) The auxiliary feedwater system must be safety grade and automatically initiated when required.

6) Tripping of the reactor coolant pumps should be consistent with the resolution to TMI Action Plan Item II.K.3.5.

The availability of offsite power affects the results of the Feedwater Line Break (FWLB) event. Therefore, two cases are considered to investigate the effects of offsite power availability, one with offsite power available and one without offsite power. For the case with offsite power, the worst single active failure is the failure of an auxiliary feedwater pump. For the case with loss of offsite power, the worst single active failure could be either the failure of an auxiliary feedwater pump or the failure of an auxiliary feedwater isolation valve in the closed position. For conservatism, and to minimize the number of cases to be analyzed, both worst single active failures are assumed to occur in the same transient.

Plant systems and equipment which are available to mitigate the effects of a feedwater line break accident are discussed in Section 15.0.8 and listed in Table 15.0.8-1.

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

1) A reactor trip on any of the following conditions.

   a) High pressurizer pressure.

   b) Overtemperature ΔT.

   c) Low-low steam generator water level in any steam generator.

d) Safety injection signals from any of the following:

   1) two out of three low steam line pressure in any one main steam line.

   2) two out of three high containment pressure (Hi-1).

   3) low pressurizer pressure.

2) An Auxiliary Feedwater System to provide an assured source of feedwater to the steam generators for decay heat removal. (Refer to Section 10.4.9 for a description of the Auxiliary Feedwater System). Automatic AFS isolation is actuated on high steam line differential pressure coincident with main steam line isolation to divert all available cooling flow to the intact steam generators. (Refer to Sections 7.3.1.3.3 and 10.4.9.3 and Figure 7.2.1-1, Sheet 7 of 15.)

3) Automatic main steam line isolation to terminate steam flow from the intact steam generators to the break (through the affected steam generator).

The analysis assumed operator control of safety injection (and auxiliary feedwater) 30 minutes after event initiation.

15.2.8.2 Analysis of Effects and Consequences

Method of Analysis - The availability of offsite power affects the results of the Feedwater Line Break event. Therefore, two cases are run to investigate the effects of offsite power availability, one with offsite power available and one with loss of offsite power.

In addition, sensitivity studies were run to identify the limiting single failure, safety injection assumption, timing of offsite power loss, and to verify that the possible case crediting operator manual action to trip off the reactor coolant pumps is bounded by the above FWLB cases. Plant operating procedures call for manual trip of the reactor coolant pumps on loss of subcooling margin. The reactor coolant pump trip criteria indicates that if both of the following occur, then stop all reactor coolant pumps:

- Safety Injection flow greater than 200 gpm
- Reactor coolant system pressure less than 1400 psig

The Feedwater Line Break event is postulated as a double ended rupture of the largest feedwater line that could result in steam generator blowdown. By assuming the largest possible break area, the steam generator blowdown rate is maximized, which maximizes the resulting reactor coolant system heatup.

The transient response of the reactor system is calculated using the ANF-RELAP (Reference 15.2.3-1) computer program. The primary concerns in analyzing the Feedwater Line Break event are to avoid primary and secondary side over pressurization and long term decay heat removal. The initial conditions, trip setpoints, plant systems availability, and boundary conditions are conservatively adjusted to maximize the ANF-RELAP calculated system pressures.

Major assumptions made in the analyses are as follows:

1) The plant is initially operating at 2958 MW (rated + 0.34%).

2) No credit is taken for the pressurizer power operated relief valves or pressurizer spray.

3) No credit is taken for the high pressurizer pressure reactor trip.

4) Main feedwater flow to all steam generators is assumed to stop at the time the break occurs (all main feedwater spills out through the break).

5) The worst possible break area is assumed. This maximizes the blowdown discharge rate, which maximizes the resultant heatup of the reactor coolant.

6) Reactor trip is assumed to be initiated when the low-low level steam generator trip setpoint.

7) The Auxiliary Feedwater System is actuated by the low-low steam generator water level signal. A 61.5 second delay was assumed following the low-low level signal to allow time for startup of the standby diesel generators and the auxiliary feedwater pumps.

8) No credit is taken for charging or letdown.

9) No credit is taken for the following potential protection logic signals to mitigate the consequences of the accident:

    1) High pressurizer pressure.

    2) Overtemperature ΔT.

    3) High pressurizer level.

    4) High containment pressure.

Receipt of a low-low steam generator water level signal in at least one steam generator starts both the motor driven auxiliary feedwater pumps, and in two out of three steam generators starts the turbine driven auxiliary feed pump, which in turn initiates auxiliary feedwater flow to the steam generators. Similarly receipt of a low steam line pressure signal in at least one steam line initiates a steam line isolation signal which closes the main steam line isolation valves in all steam lines. This signal also gives a safety injection signal which initiates flow of borated water into the RCS. The amount of safety injection flow is a function of RCS pressure.

10) For the feedwater line break case with offsite power, the worst single active failure is the failure of an auxiliary feedwater pump. For the feedwater line break case with loss of offsite power, the worst single active failure could be either the failure of an auxiliary feedwater pump or the failure of an auxiliary feedwater isolation valve in the closed position.

11) For the feedwater line break case with loss of offsite power, the reactor coolant pumps trip upon the loss of offsite power. The limiting time at which loss of offsite power occurs is determined by performing a series of sensitivity calculations.

Following the trip of the reactor coolant pumps, there will be a flow coastdown until reactor coolant loop flow reaches the natural circulation value. The natural circulation capability of the RCS has been shown in Section 15.2.6, for the loss of AC power transient, to be sufficient to remove core decay heat following reactor trip. Pump coastdown characteristics are demonstrated in Sections 15.3.1 and 15.3.2 for single and multiple reactor coolant pump trips, respectively.

Emergency operating procedures following a feedwater line break will call for the following actions to be taken by the reactor operator:

1) Isolation of feedwater flow spilling out the break of faulted steam generator and align system so that the level in the intact steam generators is recovered.

2) Stop Charging/Safety Injection Pumps (CSIP's) if: 1) wide range reactor coolant pressure is stable or increasing and subcooling exists, 2) the pressurizer level is on span plus errors and, 3) steam generator narrow range level indication exists in at least one steam generator or minimum auxiliary feedwater flow exists.

Subsequent to recovery of level in the intact steam generators, safety injection flow will be isolated and plant operating procedures will be followed in cooling the plant to hot shutdown conditions.

This event is analyzed at the SGR/Uprating value for T_{avg} (588.8°F), which bounds operation at 580.8°F.

Conservative conditions established for the analysis of this event are presented in Table 15.2.8-1. Available reactor protection system trips are presented in Table 15.0.6-2. For the analysis of this event, all trips listed in this table were disabled except low-low steam generator level. Key operating parameters used in the analysis of this event are presented in Table 15.2.8-2. The range of neutronics parameters supported by this analysis are presented in Table 15.2.8-3.

No reactor control systems are assumed to function. The Reactor Protection System is required to function following a feedwater line rupture as analyzed here. No single active failure will prevent operation of this system.

The engineered safety systems assumed to function are the Auxiliary Feedwater System, Main Steamline Isolation, and the Safety Injection System. For the Auxiliary Feedwater System, the minimum total flow of 390[f] gpm was used as a basis for this analysis. Although this is approximately the equivalent of flow from one AFW pump, there is no identified single failure that results in only one AFW pump being available. Unlike the Chapter 15.2.6 and 15.2.7 events, no commitments or requirements have been identified that would require this event to be analyzed using any failure scenario beyond the standard single-active failure condition specified in Section 15.0.13.a. Therefore, only the most limiting single AFW pump failure must

[f] Safety analyses support an AFW flow rate of 374 gpm at the lowest lifting MSSV setpoint plus 3% tolerance. See Section 15.0 for more details.

be assumed for this event. This allows credit for two AFW pumps to deliver the assumed 390[g] gpm of total flow.

Separate from pump or system flow, an additional consideration presented in Table 10.4.9-2 is the number of steam generators that receive auxiliary feedwater. An alternative potential failure is temporary isolation of one steam generator. Instead of dividing the minimum AFW flow among three steam generators, inadvertent isolation of a steam generator would split the full (three pump) AFS flow between two steam generators.

For the cases in which forced RCS flow is maintained to sustain the normal operating heat transfer coefficient between the primary coolant and the inside of the steam generator tubes, the alternative of delivering four times as much auxiliary feedwater to one less steam generator represents a much less restrictive limitation on cooling capability. Therefore, the analysis basis listed in Table 15.0.13-2 for the case in which offsite power continues to be available is minimum AFS flow delivering to two intact steam generators.

However, when loss of offsite power is assumed, the relative cooling capability of AFS flow rate vs number of steam generators receiving flow is not immediately obvious. Under RCP coastdown and natural circulation conditions, it is not clear whether the limiting factor for RCS cooling is steam generator secondary side inventory or degraded heat transfer from the primary coolant to the SGs. As such, analysis of feedwater line break with loss of offsite power is based on conservatively combining these two alternative failure scenarios:

1) the AFS delivering a minimum of 390[h] gpm total with a single failure of one AFW pump, in combination with

2) inadvertent isolation of AFW flow to one steam generator.

Although postulation of multiple independent failures is not required, doing so reduces the number of cases to be analyzed.

A detailed description and analysis of the Safety Injection System is provided in Section 6.3. The Auxiliary Feedwater System is described in Section 10.4.9.

Results

Various combinations of kinetics parameters, SI flow assumptions, and single-failures were analyzed to determine the limiting cases with offsite power available and under loss of offsite power conditions. Additional calculations were performed to determine the limiting time for loss of offsite power to occur. A case with manual pump trip due to loss of subcooling margin was also considered as a loss of offsite power case. The limiting cases are discussed below. The case with inadvertent isolation of one of two intact steam generators is not limiting and is not discussed further.

Feedwater Line Break with Offsite Power Available

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[g] Safety analyses support an AFW flow rate of 374 gpm at the lowest lifting MSSV setpoint plus 3% tolerance. See Section 15.0 for more details.
[h] Safety analyses support an AFW flow rate of 374 gpm at the lowest lifting MSSV setpoint plus 3% tolerance. See Section 15.0 for more details.

The limiting case with offsite power available includes BOC kinetics, maximum HHSI, and failure of the turbine-driven AFW pump.

The sequence of events for the Feedwater Line Break case with offsite power is given in Table 15.2.8-4. The responses to key system variables are given in Figures 15.2.8-1 through 15.2.8-10.

The Feedwater Line Break transient simulation is terminated at 7000 seconds. The MSSVs and AFW system are removing the decay heat load. The primary system is in a stable, slow cooldown mode. HHSI is off, but the primary system remains essentially water-solid. The operator is controlling the intact steam generators water level with AFW.

Operator actions are credited in this transient simulation with terminating HHSI and cycling AFW flow, both beginning at 30 minutes.

Both primary and secondary system maximum pressures occur during the "second peak" transient time period. The maximum reactor vessel pressure does not exceed 120% of the primary system design pressure, 3000 psia. The maximum SG pressure does not exceed 120% of the secondary side design pressure, 1440 psia.

Feedwater Line Break with Loss of Offsite Power

Based on sensitivity calculation results, this transient conservatively assumes that the reactor coolant pumps trip 15 minutes after the time of the subcooling margin trip. The limiting case with loss of offsite power includes EOC kinetics, maximum HHSI, and failure of the turbine-driven AFW pump.

The sequence of events for the Feedwater Line Break with Loss of Offsite Power case is given in Table 15.2.8-5. The responses to key system variables are given in Figures 15.2.8-11 through 15.2.8-20.

The Feedwater Line Break with Loss of Offsite Power transient simulation is terminated at 5000 seconds. The MSSV and AFW system are removing the decay heat load. The primary system is in a stable, slow cooldown mode. HHSI is off, but the primary system remains water-solid. The liquid levels in the intact steam generators are being controlled by the operator through AFW cycling.

Operator actions are credited to terminate HHSI and cycle AFW flow, both beginning at 30 minutes.

The maximum reactor vessel pressure does not exceed 120% of the primary system design pressure, 3000 psia. The maximum SG pressure does not exceed 120% of the secondary side design pressure, 1440 psia.

Long Term Effects and Events - For a feedwater line break, auxiliary feedwater is initiated automatically as is Safety Injection. For the feedline break downstream of the main feedwater isolation valves the required operator actions are similar in nature to the required actions for the steamline break. Table 15.2.8-6 provides a complete assessment of the operator's role.

Where possible, the operator should increase the auxiliary feedwater flow to the intact steam generators in order to shorten the time until primary temperatures begin to decrease. As a minimum, the operator must provide for decay heat removal through the intact steam generators by maintaining steam generator water level using auxiliary feedwater as a makeup supply. The operator can use the steam dump system or the steam generator PORV's to begin a controlled cooldown, or the unit may be maintained in hot standby using the steam side safety valves for decay heat removal.

Subsequent to recovery of level in the intact steam generators, the Charging/Safety Injection Pumps will be turned off and plant operating procedures will be followed in cooling the plant to shutdown conditions. The operator must observe the primary steam pressure-temperature relationship to ensure that voiding does not occur in the reactor coolant system. The operator uses safety grade instrumentation and controls to manually control the primary system pressure and pressurizer level.

Operator action in the postulated feedwater line rupture to terminate the safety injection within 30 minutes has been credited in the analysis. Refer to Section 5.2.2.2 for a discussion of pressurizer SRV qualification for this event.

15.2.8.3 Conclusions

The results of the analysis demonstrate that the event acceptance criteria are met since over pressurization of the primary and secondary systems is avoided. In addition, the analysis indicates that long term decay heat removal is adequate. The radiological consequences of this event would be less severe than the steamline break accident analyzed in Section 15.1.5

The analysis for this event supports full power operation at a nominal primary T_{avg} between 580.8°F and 588.8°F, inclusive.

REFERENCES: SECTION 15.2

15.2.3-1 ANF-RELAP Methodology for Pressurized Water Reactors: Analysis of Non-LOCA Chapter 15 Events, ANF-89-151(P)(A), Advanced Nuclear Fuels Corporation, Richland, WA 99352, May 1992.

15.2.3-2 Application of Exxon Nuclear Company PWR Thermal Margin Methodology to Mixed Core Configurations, XN-NF-82-21(P)(A), Revision 1, Exxon Nuclear Company, Richland, WA 99352, September 1983.

15.2.3-3 XCOBRA-IIIC: A Computer Code to Determine the Distribution of Coolant During Steady-State and Transient Core Operation, XN-NF-75-21(P)(A), Revision 2, Exxon Nuclear Company, Richland, WA 99352, January 1986.

15.2.3-4 Letter from B.C. Waldrep (Duke Energy) to NRC (Serial HNP-15-038) dated December 17,2015, "License Amendment Request for Main Steam Safety Valve Lift Setting Tolerance Change." (Safety Evaluation Report received by letter dated July 25, 2016).

15.2.3-5 RETRAN-3D - A Program for Transient Thermal-Hydraulic Analysis of Complex Fluid Flow Systems, EPRI NP-7450(A), Revision 10, September 2014.

15.3 DECREASE IN REACTOR COOLANT SYSTEM FLOW RATE

A number of faults are postulated which could result in a decrease in Reactor Coolant System flow rate. These events are discussed in this section. Detailed analyses are presented for the most limiting of these events.

Discussions of the following flow decrease events are presented in Section 15.3:

1) Partial loss of forced reactor coolant flow. (ANS Condition II event).

2) Complete loss of forced reactor coolant flow. (ANS Condition III event, however, is analyzed as a Condition II event).

3) Reactor coolant pump shaft seizure (locked rotor). (ANS Condition IV event).

4) Reactor coolant pump shaft break. (ANS Condition IV event).

Section 15.0.1 contains a discussion of ANS classifications.

15.3.1 PARTIAL LOSS OF FORCED REACTOR COOLANT FLOW

15.3.1.1 Identification of Causes and Accident Description

A partial loss of forced reactor coolant flow accident can result from a mechanical or electrical failure in a reactor coolant pump, or from a fault in a reactor coolant pump bus. If the reactor is at power at the time of the accident, the immediate effect of loss of forced reactor coolant flow is a rapid increase in the reactor coolant temperature. This increase could result in DNB with subsequent fuel damage if the reactor is not tripped promptly.

Normal power for the reactor coolant pumps is supplied through individual buses connected to the turbine generator. When a generator trip occurs, the buses are automatically transferred to a transformer supplied from external power lines, and the reactor coolant pumps will continue to supply coolant flow to the core. Following any turbine trip where there are no electrical faults which require tripping the generator from the network, the generator remains connected to the network for approximately 30 seconds. The reactor coolant pumps remain connected to the generator, thus ensuring full flow for approximately 30 seconds after the reactor trip before any transfer is made.

This event is classified as an ANS Condition II incident (a fault of moderate frequency) as defined in Section 15.0.1.

The necessary protection against a partial loss of forced reactor coolant flow accident is provided by the low reactor coolant flow reactor trip signal which is actuated in any reactor coolant loop by two out of three low flow signals. Above Permissive 8, low flow in any loop will actuate a reactor trip. Between approximately 10 percent power (Permissive 7) and the power level corresponding to Permissive 8, low flow in any two reactor coolant loops will actuate a reactor trip. Reactor Trip on Low flow is blocked below Permissive 7. Above Permissive 7, two or more reactor coolant pump circuit breakers opening will actuate the corresponding undervoltage relays. This results in a reactor trip which serves as a backup to the low flow trip.

Plant systems and equipment which are necessary to mitigate the effects of the accident are discussed in Section 15.0.8 and listed in Table 15.0.8-1. No single active failure in any of these systems or equipment will adversely affect the consequences of the accident.

15.3.1.2 Deleted by Amendment No. 48

15.3.1.3 Deleted by Amendment No. 48

15.3.1.4 Event Disposition for AREVA Analysis

Mode 1 (Above P-7) - The amount of flow reduction and the rate of flow decrease is less than that in Mode 1 of Event 15.3.2, because only 1 reactor coolant pump is affected in Event 15.3.1. The challenge to Specified Acceptable Fuel Design Limits (SAFDLs) is therefore bounded for this event by Event 15.3.2, as long as Event 15.3.2 satisfies the SAFDLs since Event 15.3.2 is classified as a Condition III event. Mode 1 (Above P-7) bounds all other modes of operation for this event.

In the AREVA analysis of Event 15.3.2 (Section 15.3.2), the Condition II acceptance criteria were met. Therefore, this event does not require reanalysis by AREVA.

15.3.2 COMPLETE LOSS OF FORCED REACTOR COOLANT FLOW

15.3.2.1 Identification of Causes and Accident Description

A complete loss of forced reactor coolant flow may result from a simultaneous loss of electrical supplies to all reactor coolant pumps. If the reactor is at power at the time of the accident, the immediate effect of loss of forced reactor coolant flow is a rapid increase in the reactor coolant temperature. This increase could result in DNB with subsequent fuel damage if the reactor were not tripped promptly.

Normal power for the reactor coolant pumps is supplied through buses from a transformer connected to the turbine generator. When a generator trip occurs, the buses are automatically transferred to a transformer supplied from external power lines, and the pumps will continue to supply reactor coolant flow to the core. Following any turbine trip where there are no electrical faults which require tripping the generator from the network, the generator remains connected to the network for approximately 30 seconds. The reactor coolant pumps remain connected to the generator, thus ensuring full flow for 30 seconds after the reactor trip before any transfer is made.

This event is classified as an ANS Condition III incident (an infrequent incident) as defined in Section 15.0.1. This transient has been analyzed against Condition II acceptance criteria in order to bound the Chapter 15.3.1 "Partial Loss of Forced Reactor Coolant Flow" Event. The Condition II acceptance criteria are:

1) The pressures in the reactor coolant and main steam systems should be less than 110% of design values.

2) The fuel cladding integrity should be maintained by ensuring that fuel design limits are not exceeded. This is demonstrated by assuring that the minimum calculated departure

from nucleate boiling ratio (DNBR) is not less than the applicable limits of the DNBR correlation being used and that fuel centerline melt does not occur.

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

4) For transients of moderate frequency in combination with a single failure, no loss of function of any fission product barrier, other than fuel element cladding, shall occur. Core geometry is maintained in such a way that there is no loss of core cooling capability and control rod insertability is maintained.

The following signals provide the necessary protection against a complete loss of flow accident:

1) Reactor coolant pump power supply undervoltage or underfrequency.

2) Low reactor coolant loop flow.

The reactor trip on reactor coolant pump undervoltage is provided to protect against conditions which can cause a loss of voltage to all reactor coolant pumps, i.e., station blackout. This function is blocked below approximately 10 percent power (Permissive 7).

The reactor trip on reactor coolant pump underfrequency is provided to trip the reactor for an underfrequency condition, resulting from frequency disturbances on the power grid. If the maximum grid frequency decay rate is less than approximately 5 Hz/sec., this trip function will protect the core from underfrequency events without requiring tripping of the RCP breakers. Refer to Chapter 7 for interface requirements concerning tripping of the RCP breakers for underfrequency events.

The reactor trip on low reactor coolant loop flow is provided to protect against loss of flow conditions which affect only one reactor coolant loop. This function is generated by two out of three low flow signals per reactor coolant loop. Above Permissive 8 power level, low flow in any reactor coolant loop will actuate a reactor trip. Between approximately 10 percent power (Permissive 7) and the power level corresponding to Permissive 8, low flow in any two reactor coolant loops will actuate a reactor trip. If the maximum grid frequency decay rate is less than approximately 2.5 Hz/sec., the low flow trip function will protect the core from underfrequency events.

Plant systems and equipment which are necessary to mitigate the effects of the accident are discussed in Section 15.0.8 and listed in Table 15.0.8-1. No single active failure will adversely affect the consequences of the event.

15.3.2.2 Analysis of Effects and Consequences

Method of Analysis - Two cases were analyzed: one for a reactor trip actuated by the pump power supply undervoltage trip and the other for a reactor trip actuated by pump power supply underfrequency trip with a maximum grid frequency decay rate of 5 Hz/sec.

The loss of flow cases are initiated from rated power plus uncertainty. This event is analyzed using the SGR/Uprating value for T_{avg} (588.8°F). The DNB results bound all lower temperature operating conditions.

Conservative conditions established for the analysis of this event are presented in Table 15.3.2-1. Available reactor protection system trips are presented in Table 15.0.6-2. For the analysis of this event, all trips listed in this table were disabled except low primary coolant flow, pump power supply underfrequency (for underfrequency case), and pump power supply undervoltage (for undervoltage case). Key operating parameters used in the analysis of this event are presented in Table 15.3.2-2. The range of neutronics parameters supported by this analysis are presented in Table 15.3.2-3.

The transient response of the reactor system is calculated using the ANF-RELAP (Reference 15.3.2-2) computer program. The core thermal hydraulic boundary conditions from the ANF-RELAP calculation are used as input to the XCOBRA-IIIC code (References 15.3.2-3 and 15.3.2-4) to predict the MDNBR for the event.

Results - The underfrequency event was more limiting than the undervoltage event. The sequence of events for the underfrequency event is given in Table 15.3.2-4. The responses to key system variables are given in Figures 15.3.2-1 to 15.3.2-7.

15.3.2.3 Conclusions

The results of the analysis indicate that the predicted MDNBR is greater than the safety limit. The critical heat flux correlation limit ensures that, with 95% probability and 95% confidence, DNB is not expected to occur; therefore, no fuel is expected to fail. This event does not pose a credible challenge to FCM safety criteria. Thus, Condition II acceptance criteria are met for this event. Since Condition II acceptance criteria are met for this event, Event 15.3.1 is bounded.

The analysis for this event supports nominal primary T_{avg} operation at full power between 588.8 and 580.8°F, inclusive.

15.3.3 REACTOR COOLANT PUMP SHAFT SEIZURE (LOCKED ROTOR)

15.3.3.1 Identification of Causes and Accident Description

The accident postulated is an instantaneous seizure of a reactor coolant pump rotor such as is discussed in Section 5.4.1. Flow through the affected reactor coolant loop is rapidly reduced, leading to an initiation of a reactor trip on a low flow signal.

Following initiation of the reactor trip, heat stored in the fuel rods continues to be transferred to the reactor coolant causing the coolant to expand. At the same time, heat transfer to the shell side of the steam generators is reduced, first because the reduced flow results in a decreased tube side film coefficient and then because the reactor coolant in the tubes cool down while the shell side temperature increases (turbine steam flow is reduced to zero upon plant trip). The rapid expansion of the reactor coolant in the reactor core, combined with reduced heat transfer in the steam generators, causes an insurge into the pressurizer and a pressure increase throughout the RCS. The insurge into the pressurizer compresses the steam volume, actuates the automatic pressurizer spray system, opens the pressurizer power operated relief valves, and opens the pressurizer safety valves, in that sequence. The three pressurizer power operated relief valves are designed for reliable operation and would be expected to function properly during the accident. However, for conservatism, their pressure reducing effect as well as the pressure reducing effect of the spray is not included in the overpressurization case of the analysis.

This event is classified as an ANS Condition IV incident (a limiting fault) as defined in Section 15.0.1. The acceptance criteria for this event are:

1) The pressure in the reactor coolant and main steam systems should be maintained below acceptable design limits (i.e., the pressure in the reactor coolant and main steam systems should be maintained below 120% of the design pressures).

2) The potential for core damage is evaluated on the basis that it is acceptable if the minimum DNBR remains above the 95/95 DNBR limit. If DNBR falls below the 95/95 limit, fuel failure must be assumed for all rods that do not meet this criteria unless it can be shown, based on an acceptable fuel damage model, which includes the potential adverse effects of hydraulic instabilities, that fewer failures occur. Any fuel damage calculated to occur must be of sufficiently limited extent that the core will remain in place and intact with no loss of core cooling capacity.

3) Any reactivity release must be such that the calculated offsite doses should be less than 10% of the 10CFR50.67 limits (i.e., 2.5 rem TEDE) and the control room doses should be less than the 10CFR50.67 limit of 5 rem TEDE.

4) A rotor seizure or shaft break in a reactor coolant pump should not, by itself, generate a more serious condition or result in a loss of function of the reactor coolant system or containment barriers.

No single active failure will adversely affect the consequences of the event.

Plant systems and equipment which are available to mitigate the effects of the accident are discussed in Section 15.0.8 and listed in Table 15.0.8-1.

15.3.3.2 Analysis of Effects and Consequences

Methods of Analysis -The analysis of the Locked Rotor event has considered the effect of a coincident loss of offsite power which causes the remaining two pumps to coastdown.

Both an overpressurization case and a MDNBR case are analyzed for this event.

The MDNBR case is analyzed at the nominal SGR/Uprating value for T_{avg} of 588.8°F. The DNB results bound all lower temperature operating conditions.

The overpressurization case is analyzed at a T_{avg} of 580.8°F, which bounds operation at 588.8°F.

The analysis is performed with BOC kinetics parameters.

Normal power for the reactor coolant pumps is supplied through individual buses connected to the turbine generator. When a generator trip occurs, the buses are automatically transferred to a transformer supplied from external power lines, and the reactor coolant pumps will continue to supply coolant flow to the core. Following any turbine trip where there are no electrical faults which require tripping the generator from the network, the generator remains connected to the network for approximately 30 seconds. The reactor coolant pumps remain connected to the generator ensuring full flow for approximately 30 seconds after the reactor trip before any supply

power transfer is made. This analysis conservatively assumes the remaining two pumps are connected to the generator for 3 seconds following the low flow trip.

Conservative conditions established for the analysis of this event are presented in Table 15.3.3-1. Available reactor protection system trips are presented in Table 15.0.6-2. For the analysis of this event, all trips listed in this table were disabled except low primary coolant flow. Key operating parameters used in the analysis of this event are presented in Table 15.3.3-2. The range of neutronics parameters supported by this analysis are presented in Table 15.3.3-3.

The transient response of the reactor system is calculated using the ANF-RELAP (Reference 15.3.2-2) computer program. The core thermal hydraulic boundary conditions from the ANF-RELAP calculation are used as input to the XCOBRA-IIIC code (References 15.3.2-3 and 15.3.2-4) to predict the MDNBR for the event.

Results - This transient tripped on low primary coolant flow. For the overpressurization case, both the maximum primary system pressure and the maximum secondary system pressures are less than the design limits. The sequence of events for the bounding overpressurization case is given in Table 15.3.3-4. The responses to key system variables are given in Figures 15.3.3-1 to 15.3.3-6.

The sequence of events for the MDNBR case is given in Table 15.3.3-5. This transient tripped on low primary coolant flow. The responses to key system variables are given in Figures 15.3.3-7 to 15.3.3-13.

15.3.3.3 Radiological Consequences of a Locked Rotor

An instantaneous seizure of a reactor coolant pump rotor is assumed to occur which rapidly reduces flow through the affected reactor coolant loop. Fuel clad damage may be predicted to occur as a result of this accident. Due to the pressure differential between the primary and secondary systems and assumed steam generator tube leakage, fission products are discharged from the primary into the secondary system. A portion of this radioactivity is released to the outside atmosphere through either the atmospheric relief valves or safety valves. In addition, iodine activity is contained in the secondary coolant before the accident and some of this activity is released to the atmosphere as a result of steaming from the steam generators following the accident.

The analysis of the locked rotor radiological consequences uses the analytical methods and assumptions outlined in RG 1.183, Appendix G (Locked Rotor) and RG 1.183, Appendix H (Rod Ejection) for the fuel melt model.

A summary of input parameters and assumptions is provided in Table 15.3.3-6. Additional clarification is provided as follows:

a) It is assumed that 8% of the fuel rods in the core suffer damage as a result of the locked rotor sufficient that all of their gap activity is released to the reactor coolant system. Additionally, 1% of the fuel rods are conservatively assumed to experience centerline melt. Eight percent of the total I-131 core activity is in the fuel-cladding gap. Ten percent of the total Kr-85 core activity is in the fuel-cladding gap. Five percent of other iodine isotopes and other noble gases and 12 percent of the total core activity for alkali metals are assumed to be in the fuel-cladding gap. In the calculation of activity releases

from the failed/melted fuel and maximum radial peaking factor of 1.73 was applied. All noble gas and alkali metal activity in the damaged fuel (both gap activity and activity contained in the melted fuel) is released to the primary coolant. All of the iodine contained in the gap of failed fuel and 50 percent of the iodine activity contained in the melted fuel are released to the reactor coolant system. The core fission product inventory is given in Table 15.0.9-1.

b) The iodine activity concentration of the secondary coolant at the time the locked rotor occurs is assumed to be equivalent to the Technical Specification limit of 0.1 μCi/gm of DE I-131. The iodine activity concentration of the secondary coolant is given in Table 15.0.9-7. The alkali metal activity concentration of the secondary coolant at the time the locked rotor occurs is assumed to be 10% of the primary side concentration.

c) The amount of primary to secondary SG tube leakage is assumed to be 1 gpm total.

d) An iodine partition factor in the SGs of 0.01 (curies iodine/gm steam) / (curies iodine/gm water) is used. This partition factor is also applied to alkali metals.

e) All noble gas activity carried over to the secondary side through SG tube leakage is assumed to be immediately released to the outside atmosphere.

f) At 8 hours after the accident, the RHR system is assumed to be placed into service for heat removal and there is no further steam release to the atmosphere from the secondary system.

15.3.3.3.1 Offsite Doses

The offsite doses are calculated using the assumptions and equations in Section 15.0A.1.

15.3.3.3.2 Control Room Doses

The control room assumptions are provided in Section 15.6.5.4.3 and Table 15.6.5-15. The locked rotor doses modeled 500 cfm of unfiltered inleakage.

The control room HVAC is switched to the emergency post-accident recirculation mode after receiving a high radiation signal. The high radiation signal is reached at 3 seconds into the event. The control room HVAC is switched over to the emergency post-recirculation mode at 18 seconds (3 seconds signal initiation plus 15 seconds delay time for switching between modes). An operator action switches the control room from the post-accident recirculation mode to the pressurization mode at 2 hours after event initiation. The 15-second delay to switch between modes was also assumed with the operator action. Thus 2 hours and 33 seconds actually modeled for the time of operator action switchover to the pressurization mode.

15.3.3.3.3 Results

The potential radiological consequences resulting from a locked rotor have been conservatively analyzed, using assumptions and models described in previous sections. The TEDE doses have been analyzed for the worst two hours at the EAB and for the duration of the event at the LPZ and in the control room. The analytically predicted dose consequence to the Control Room (CR) operators due to LBLOCA increased by a small amount, based on the reduction in CR

recirculation flow which reduces the iodine filtration provided by the charcoal filters relative to the analysis of record condition. The analytically predicted dose consequence to the control room operators for the locked rotor event was not similarly revised or updated since the LBLOCA DBA is the limiting control room operator dose event. The radiological analysis results for this event are listed in Table 15.3.3-7. The resultant doses are within the applicable limits.

15.3.3.4 Conclusions

For the overpressurization case, both the maximum reactor primary system pressure and the maximum secondary system pressures are less than the 120% design limits of 3000 psia and 1440 psia, respectively.

For the MDNBR case, the predicted MDNBR is less than the 95/95 safety limit. Less than 8% of the fuel is predicted to fail based on DNB criteria. This event does not pose a credible challenge to FCM safety criteria. The radiological doses have been calculated based on 8% assumed cladding failure, and 1% assumed fuel centerline melting. The offsite radiological doses are less that 10% of the 10CFR50.67 limits (i.e. 2.5 REM TEDE) and the control room doses are less than the 10CFR50.67 limit of 5 REM TEDE. Thus, the event acceptance criteria are met.

The analysis for this event supports nominal primary T_{avg} operation at full power between 588.8 and 580.8°F, inclusive.

15.3.4 REACTOR COOLANT PUMP SHAFT BREAK

15.3.4.1 Identification of Causes and Accident Description

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

Following initiation of the reactor trip, heat stored in the fuel rods continues to be transferred to the reactor coolant causing the coolant to expand. At the same time, heat transfer to the shell side of the steam generators is reduced, first because 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 reactor coolant in the reactor core, combined with reduced heat transfer in the steam generators causes an insurge into the pressurizer and a pressure increase throughout the RCS. The insurge into the pressurizer compresses the steam volume, actuates the automatic pressurizer spray system, opens the power operated relief valves, and opens the pressurizer safety valves, in that sequence. The three power operated relief valves are designed for reliable operation and would be expected to function properly during the accident. However, for conservatism, their pressure reducing effect as well as the pressure reducing effect of the spray is not included in the analysis. The peak RCS pressure is bounded by that for a locked rotor (see Sections 15.3.3.2. This event is classified as an ANS Condition IV incident (a limiting fault) as defined in Section 15.0.1.

No single active failure will adversely affect the consequences of the event.

Plant systems and equipment which are available to mitigate the effects of the accident are discussed in Section 15.0.8 and listed in Table 15.0.8-1.

15.3.4.2 Conclusions

The consequences of a reactor coolant pump shaft break are no worse than those calculated for the locked rotor incident (see Section 15.3.3). With a failed shaft, the impeller could conceivably be free to spin in a reverse direction as opposed to being fixed in position as assumed in the locked rotor analysis. However, the net effect on core flow is negligible, resulting in only a slight decrease in the end point (steady state) core flow. For both the shaft break and locked rotor incidents, reactor trip occurs very early in the transient. In addition, the locked rotor analysis conservatively assumes that DNB occurs at the beginning of the transient. The calculations performed for this event showed that the MDNBR, fuel centerline melt, and overpressurization consequences are bounded by those of the Pump Rotor Seizure/Locked Rotor event. The Condition IV acceptance criteria are met for the Pump Rotor Seizure/Locked Rotor event, as noted in Section 15.3.3.3.

REFERENCES: SECTION 15.3

15.3.1-1 Deleted by Amendment No. 48

15.3.1-2 Deleted by Amendment No. 48

15.3.2-1 Deleted by Amendment No. 45

15.3.2-2 ANF-RELAP Methodology for Pressurized Water Reactors: Analysis of Non-LOCA Chapter 15 Events, ANF-89-151(P)(A), Advanced Nuclear Fuels Corporation, Richland, WA 99352, May 1992.

15.3.2-3 Application of Exxon Nuclear Company PWR Thermal Margin Methodology to Mixed Core Configurations, XN-NF-82-21(P)(A), Revision 1, Exxon Nuclear Company, Richland, WA 99352, September 1983.

15.3.2-4 XCOBRA-IIIC: A Computer Code to Determine the Distribution of Coolant During Steady-State and Transient Core Operation, XN-NF-75-21(P)(A), Revision 2, Exxon Nuclear Company, Richland, WA 99352, January 1986.

15.4 REACTIVITY AND POWER DISTRIBUTION ANOMALIES

A number of faults have been postulated which could result in reactivity and power distribution anomalies. Reactivity changes could be caused by control rod motion or ejection, boron concentration changes, or addition of cold water to the Reactor Coolant System (RCS). Power distribution changes could be caused by control rod motion, misalignment, or ejection, or by static means such as fuel assembly mislocation. These events are discussed in this section. Detailed analyses are presented for the most limiting of these events.

Discussions of the following incidents are presented in Section 15.4:

a) Uncontrolled rod cluster control assembly bank withdrawal from a subcritical or low power startup condition. (ANS Condition II event).

b) Uncontrolled rod cluster control assembly bank withdrawal at power. (ANS Condition II event).

c) Rod cluster control assembly misalignment. (ANS Condition II and III events).

d) Startup of an inactive reactor coolant pump at an incorrect temperature. (ANS Condition II event).

e) Chemical and Volume Control System malfunction that results in a decrease in the boron concentration in the reactor coolant. (ANS Condition II event).

f) Inadvertent loading and operation of a fuel assembly in an improper position. (ANS Condition III event).

g) Spectrum of rod cluster control assembly ejection accidents. (ANS Condition IV event).

Section 15.0.1 contains a discussion of ANS classifications.

15.4.1 UNCONTROLLED ROD CLUSTER CONTROL ASSEMBLY BANK WITHDRAWAL FROM A SUBCRITICAL OR LOW POWER STARTUP CONDITION

15.4.1.1 Identification of Causes and Accident Description

A rod cluster control assembly (RCCA) withdrawal accident is defined as an uncontrolled addition of reactivity to the reactor core caused by withdrawal of RCCA's resulting in a power excursion. Such a transient could be caused by a malfunction of the reactor control or rod control systems. This could occur with the reactor subcritical or during startup. An "at power" case is discussed in Section 15.4.2.

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

The RCCA drive mechanisms are wired into preselected bank configurations which are not altered during reactor life. These circuits prevent the RCCAs from being automatically withdrawn in other than their respective banks. Power supplied to the banks is controlled such that no more than two banks can be withdrawn at the same time and in their proper withdrawal sequence. The RCCA drive mechanisms are of the magnetic latch type and coil actuation is sequenced to provide variable speed travel. The maximum reactivity insertion rate analyzed in the detailed plant analysis is that occurring with the simultaneous withdrawal of the combination of two sequential control banks having the maximum combined worth at maximum speed.

This event is classified as an ANS Condition II incident (a fault of moderate frequency) as defined in Section 15.0.1. The acceptance criteria for this event are:

1) The pressures in the reactor coolant and main steam systems should be less than 110% of design values.

2) The fuel cladding integrity should be maintained by ensuring that fuel design limits are not exceeded. This is demonstrated by assuring that the minimum calculated departure from nucleate boiling ratio (DNBR) is not less than the applicable limits of the DNBR correlation being used and that fuel centerline melt does not occur.

No single active failure will adversely affect the consequences of the event.

Plant systems and equipment which are available to mitigate the effects of the accident are discussed in Section 15.0.8 and listed in Table 15.0.8-1.

The neutron flux response to a continuous reactivity increase is characterized by a very fast rise terminated by the reactivity feedback effect of the negative Doppler coefficient. This self-limitation of the power excursion is of primary importance since it limits the power to a tolerable level during the delay time for protective action. Should a continuous RCCA withdrawal accident occur, the transient will be terminated by the following automatic features of the Reactor Protection System:

1) Source range high neutron flux reactor trip. Actuated when either of two independent source range channels indicates a neutron flux level above a preselected manually adjustable setpoint. This trip function may be manually bypassed only after an intermediate range flux channel indicates a flux level above a specified level. It is automatically reinstated when both intermediate range channels indicate a flux level below a specified level.

2) Intermediate range high neutron flux reactor trip. Actuated when either of two independent intermediate range channels indicates a flux level above a preselected manually adjustable setpoint. This trip function may be manually bypassed only after two out of the four power range channels are reading above approximately 10 percent of full power and is automatically reinstated when three out of the four channels indicate a power level below this value.

3) Power range high neutron flux reactor trip (low setting). Actuated when two out of the four power range channels indicate a power level above approximately 25 percent of full power. This trip function may be manually bypassed when two out of the four power range channels indicate a power level above approximately 10 percent of full power and is automatically reinstated only after three out of the four channels indicate a power level below this value.

4) Power range high neutron flux reactor trip (high setting). Actuated when two out of the four power range channels indicate a power level above a preset setpoint. This trip function is always active.

5) High nuclear flux rate reactor trip. Actuated when the positive rate of change of neutron flux on two out of four nuclear power range channels indicate a rate above the preset setpoint. This trip function is always active.

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

respectively. This event is precluded by plant procedures in Mode 3 when T_{avg} <551°F and in Modes 4-6.

15.4.1.2 Analysis of Effects and Consequences

Method of Analysis - The objective of this analysis is to bound plant operational modes below approximately 10% of rated power to Mode 3 when T_{avg} >551°F. The analysis examines the possible operational modes and state conditions between these two limits to develop a bounding case.

This event is driven by the magnitude and rate of reactivity insertion.

The reactivity insertion rate is rapid enough that very high neutron powers are calculated, but of short enough duration that excessive energy deposition does not occur. Because the event is very rapid, an increase in primary coolant temperature lags behind power. The low coolant flow rate in the core accompanied by a rapid surge of power makes this event a challenge to both the Specified Acceptable Fuel Design Limits (SAFDLs) and system pressurization. The challenge to the SAFDLs is controlled by the rate of energy dissipation from the fuel rod. The challenge to the system pressurization is due to the large and rapid thermal expansion of the coolant in the core.

A low initial power yields the maximum margin to trip and, hence, maximum time for withdrawal to trip. This will yield the largest prompt multiplication which maximizes overshoot past trip. The initial power selected conservatively bounds the shutdown condition. Therefore, this event is analyzed at HZP conditions. Two reactor coolant pumps are assumed operational to minimize the coolant flow. The event is analyzed using BOC neutronics conditions.

The overpressurization scenario is not analyzed because it is bounded by Event 15.2.3, Turbine Trip. An MDNBR case is analyzed to evaluate the challenge to the SAFDLs. The MDNBR case bounds operation in Mode 3 with T_{avg} >551°F the power range flux trip (low setting) reset setpoint, approximately 10% rated power.

Conservative conditions established for the analysis of this event are presented in Table 15.4.1-1. Available reactor protection system trips are presented in Table 15.0.6-2. For the analysis of this event, all trips listed in this table were disabled except power range neutron flux, low setting. Key operating parameters used in the analysis of this event are presented in Table 15.4.1-2. The range of neutronics parameters supported by this analysis are presented in Table 15.4.1-3.

In order to give conservative results for a startup accident, the following assumptions are made:

1) Since the magnitude of the power peak reached during the initial part of the transient for any given rate of reactivity insertion is strongly dependent on the Doppler coefficient, conservatively low values as a function of temperature are used.

2) Contribution of the moderator reactivity coefficient is secondary during the initial part of the transient because the heat transfer time between the fuel and the moderator is much longer than the neutron flux response time. However, after the initial neutron flux peak, the succeeding rate of power increase is affected by the moderator reactivity coefficient. A conservative value of +5 pcm/°F at hot zero power is used in the analysis to yield the maximum peak heat flux.

3) The reactor is assumed to be at hot zero power. This assumption is more conservative than that of a lower initial system temperature. The higher initial system temperature yields a less negative (smaller absolute magnitude) Doppler coefficient which tends to reduce the Doppler feedback effect thereby increasing the neutron flux peak.

4) Reactor trip is assumed to be initiated by power range high neutron flux (low setting). The most adverse combination of instrument and setpoint errors, as well as delays for trip signal actuation and RCCA release, is taken into account. A 10 percent increase is assumed for the power range flux trip setpoint raising it from the nominal value of 25 percent to 35 percent. Since the rise in the neutron flux is so rapid, the effect of errors in the trip setpoint on the actual time at which the rods are released is negligible. In addition, the reactor trip insertion characteristic is based on the assumption that the highest worth RCCA is stuck in its fully withdrawn position. See Section 15.0.5 for RCCA insertion characteristics.

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

6) The initial power level was assumed to be below the power level expected for any shutdown condition. The combination of highest reactivity insertion rate and lowest initial power produces the highest peak heat flux.

7) Two reactor coolant pumps are assumed to be in operation.

The transient response of the reactor system is calculated using the ANF-RELAP (Reference 15.4.1-3) computer program. The core thermal hydraulic boundary conditions from the ANF-RELAP calculation are used as input to the XCOBRA-IIIC code (References 15.4.1-4 and 15.4.1-5) to predict the MDNBR for the event.

Results

The event tripped on power range neutron flux, low setting. The sequence of events is given in Table 15.4.1-4. The transient response is given in Figures 15.4.1-1 through 15.4.1-4.

The results of this analysis also demonstrate that the hot spot centerline temperature is below the centerline melt temperature.

15.4.1.3 Conclusions

The results of the analysis indicate that the predicted MDNBR is greater than the safety limit. The critical heat flux correlation limit ensures that, with 95% probability and 95% confidence, DNB is not expected to occur; therefore, no fuel is expected to fail. The fuel centerline melt threshold temperature is not penetrated during this event. Thus, the acceptance criteria for this event are met.

The analysis for this event supports nominal no-load primary T_{avg} operation at 557°F.

15.4.2 UNCONTROLLED ROD CLUSTER CONTROL ASSEMBLY BANK WITHDRAWAL AT POWER

15.4.2.1 Identification of Causes and Accident Description

Uncontrolled RCCA bank withdrawal at power results in an increase in the core heat flux. Since the heat extraction from the steam generator lags behind the core power generation until the steam generator pressure reaches the power operated relief or safety valve setpoint, there is a net increase in the reactor coolant temperature. Unless terminated by manual or automatic action, the power mismatch and resultant reactor coolant temperature rise could eventually result in DNB. Therefore, in order to avert damage to the fuel clad, the Reactor Protection System is designed to terminate any such transient before the DNBR falls below the safety analysis value.

This event is classified as an ANS Condition II incident (a fault of moderate frequency) as defined in Section 15.0.1. The acceptance criteria for this event are:

1) The pressure in the reactor coolant and main steam systems should be less than 110% of design values.

2) The fuel cladding integrity should be maintained by ensuring that fuel design limits are not exceeded. This is demonstrated by assuring that the minimum calculated departure from nucleate boiling ratio (DNBR) is not less than the applicable limits of the DNBR correlation being used and that fuel centerline melt does not occur.

The automatic features of the Reactor Protection System which prevent core damage following the postulated accident include the following:

1) Power range neutron flux instrumentation actuates a reactor trip if two out of four channels exceed an overpower setpoint.

2) Reactor trip is actuated if any two out of three ΔT channels exceed an overtemperature ΔT setpoint. This setpoint is automatically varied with axial power imbalance, reactor coolant temperature and pressure to protect against DNB.

3) Reactor trip is actuated if any two out of three ΔT channels exceed an overpower ΔT setpoint. This setpoint is automatically varied with axial power imbalance to ensure that the allowable heat generation rate (kW/ft.) is not exceeded.

4) A high pressurizer pressure reactor trip actuated from any two out of three pressure channels, which is set at a fixed point. This set pressure is less than the set pressure for the pressurizer safety valves.

5) A high pressurizer water level reactor trip actuated from any two out of three level channels when the reactor power is above approximately 10 percent (Permissive-7).

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

1) High neutron flux (one out of four power range).

2) Overpower ΔT (two out of three).

3) Overtemperature ΔT (two out of three).

The manner in which the combination of overpower and overtemperature ΔT trips provide protection over the full range of RCS conditions is described in Chapter 7.

The area of permissible operation (power, pressure, and temperature) is bounded by the combination of reactor trips: high neutron flux (fixed setpoint); high pressure (fixed setpoint); low pressure (fixed setpoint); overpower and overtemperature ΔT (variable setpoints).

Overpressurization is not a concern since the secondary system is not isolated until after reactor trip. The overpressurization aspects of this event are bounded by the Turbine Trip event (Event 15.2.3).

Plant systems and equipment which are available to mitigate the effects of the accident are discussed in Section 15.0.8 and listed in Table 15.0.8-1. No single active failure in any of these systems or equipment will adversely offset the consequences of the accident.

15.4.2.2 Analysis of Effects and Consequences

Method of Analysis - The power range to be considered in this analysis is from the power range high flux (low setting) trip reset point (approximately 10% of rated power), up to full power. Uncontrolled RCCA bank withdrawal at power levels below 10% rated power are considered in Event 15.4.1. This analysis considers a spectrum of reactivity insertion rates at initial power levels of 10%, 60% and 100%. Since neutronic feedback as a function of cycle exposure and design also influences the results, these effects are also included in the analysis.

A broad range of reactivity insertion rates are possible. Therefore, a spectrum of reactivity insertion rates were evaluated in order to bound events ranging from a slow dilution of the primary system boron concentration to the maximum possible RCCA bank withdrawal rate at maximum bank worth. Specifically, the analysis encompasses reactivity insertion rates up to 50 pcm/sec.

Reactivity feedback effects are bounded by analyzing a series of BOC cases representing the minimum reactivity feedback (positive moderator coefficient and conservatively small Doppler coefficient) and a series of EOC cases representing maximum feedback (conservatively large negative moderator coefficient and large negative Doppler coefficient).

The reactivity insertion rate due to the bank withdrawal is varied until the OTΔT and power range high neutron flux trip actuate simultaneously. This represents the limiting point with respect to MDNBR.

This event is analyzed using the SGR/Uprating value for T_{avg} (588.8°F), which bounds operation at 580.8°F.

Conservative conditions established for the analysis of this event are presented in Table 15.4.2-1. Available reactor protection system trips are presented in Table 15.0.6-2. For the analysis of this event, all trips listed in this table were disabled except OTΔT and power range neutron flux, high setting. Key operating parameters used in the analysis of this event are presented in Table

15.4.2-2. The range of neutronics parameters supported by this analysis are presented in Table 15.4.2-3.

In order to obtain conservative results for an uncontrolled rod withdrawal at power accident, the following assumptions are made:

1) The reactor trip on high neutron flux is assumed to be actuated at a conservative value of 115 percent of nominal full power. The ΔT trips include all adverse instrumentation and setpoint errors; the delays for trip actuation are assumed to be the maximum values.

2) The RCCA trip insertion characteristic is based on the assumption that the highest worth assembly is stuck in its fully withdrawn position.

3) The maximum positive reactivity insertion rate is greater than that for the simultaneous withdrawal of the combinations of the two control banks having the maximum combined worth at maximum speed.

The effect of RCCA movement on the axial core power distribution is accounted for by causing a decrease in overtemperature ΔT trip setpoint proportional to a decrease in margin to DNB.

The transient response of the reactor system is calculated using the S-RELAP5 (Reference 15.4.2-3) computer program. The core thermal hydraulic boundary conditions from the S-RELAP5 calculation are used as input to the XCOBRA-IIIC code (References 15.4.1-4 and 15.4.1-5) to predict the MDNBR for the event.

Results - The uncontrolled RCCA bank withdrawal transients are analyzed for a spectrum of reactivity insertion rates at initial power levels of 10%, 60% and HFP. Figure 15.4.2-1 presents the MDNBR results for the range of reactivity addition rates analyzed for the 10%, 60%, and HFP cases. The limiting uncontrolled RCCA bank withdrawal transient occurred at 10% power with BOC kinetics and the most limiting Reactivity Insertion Rate among a spectrum of analyzed values.

The sequence of events for the limiting Uncontrolled RCCA Bank Withdrawal transient is given in Table 15.4.2-4. The responses to key system variables for the limiting transient are given in Figures 15.4.2-2 to 15.4.2-9.

15.4.2.3 Conclusions

Reactivity insertion transient calculations demonstrate that the DNB correlation limit will not be penetrated during any credible reactivity insertion transient at full power. Analysis results demonstrate that transients initiated at power levels below full power are less limiting. The critical heat flux correlation limit ensures that, with 95% probability and 95% confidence, DNB is not expected to occur; therefore, no fuel is expected to fail. The fuel centerline melt threshold is not penetrated during this event.

The analysis for this event supports full power operation at a nominal primary T_{avg} between 588.8°F and 580.8°F, inclusive.

15.4.3 ROD CLUSTER CONTROL ASSEMBLY MISOPERATION (SYSTEM MALFUNCTION OR OPERATOR ERROR)

RCCA misoperation accidents include:

1) One or more dropped assemblies within the same group.

2) A dropped full length assembly bank.

3) Statically misaligned full length assembly.

4) Withdrawal of a single full length assembly.

Each RCCA has a position indicator channel which displays position of the assembly. The displays of assembly positions are grouped for the operator's convenience. Fully inserted assemblies are further indicated by a rod at bottom signal, which actuates a local alarm and a Control Room annunciator. Group demand position is also indicated.

Full length RCCAs are always moved in preselected banks, and the banks are always moved in the same preselected sequence. Each bank of RCCAs is divided into two groups. The rods comprising a group operate in parallel through multiplexing thyristors. The two groups in a bank move sequentially such that the first group is always within one step of the second group in the bank. A definite schedule of actuation (or deactuation of the stationary gripper, movable gripper, and lift coils of a mechanism) is required to withdraw the RCCA attached to the mechanism. Since the stationary gripper, movable gripper, and lift coils associated with the RCCAs of a rod group are driven in parallel, any single failure which would cause rod withdrawal would affect a minimum of one group. Mechanical failures are in the direction of insertion, or immobility.

Plant systems and equipment which are available to mitigate the effects of the various control rod misoperations are discussed in Section 15.0.8 and listed in Table 15.0.8-1.

15.4.3.1 Dropped Full Length RCCA or RCCA Bank

15.4.3.1.1 Identification of causes and accident description

The dropped RCCA and dropped RCCA bank events are initiated by a de-energized control rod drive mechanism or by a malfunction associated with a RCCA bank during power operation. The result is that a single RCCA or RCCA bank falls into the core. The dropped RCCA promptly inserts negative reactivity which reduces reactor power and disturbs the power distribution, resulting in an increase (augmentation) of local power peaking. Two operational states are available: Manual and Automatic Rod Control (ARC).

In the automatic rod control mode, the ARC system receives signals from the excore detectors and the turbine to indicate a primary/secondary side power mismatch. In an attempt to eliminate a mismatch, the ARC system initiates the movement of a partially inserted control bank. The ARC system detects the drop in power and initiates control bank withdrawal. Power overshoot may occur, after which the control rod system will insert the control bank and return the plant to nominal power. The magnitude of the power overshoot is a function of:

- Core reactivity coefficients;
- Dropped rod worths;
- Differential bank worths; and
- Rod shadowing factors (RSF).

An automatic and redundantly actuated reduction in turbine load demand (turbine runback) is provided as protection for this event. When turbine runback occurs, the automatic rod controller setpoint reference temperature is also set back to an average primary coolant temperature corresponding to the new turbine load demand. With the turbine runback the reduction in load initially results in a load mismatch if the dropped rod (bank) reactivity does not match that required for the setback power level. If the dropped rod worth is not equal to the reactivity to match the power runback, a power mismatch between primary and secondary occurs and is detected by either coolant temperature or neutron power above or below setpoint. The controller output signal is sent to the control rod driver controller, which acts to minimize the mismatch.

Prior to Cycle 12 the ARC system was driven by excore detector N-44. For Cycle 12 and later, new circuitry will identify the highest NI channel reading of the four available signals for the ARC system. To incorporate a single failure, the analysis used the second highest NI channel reading to drive the ARC system.

In the manual mode, a decrease in moderator temperature results from the initial power reduction. At EOC conditions, automatic action taken by the turbine DEH system to open the Turbine Governor Valves in combination with a strongly negative moderator temperature coefficient can return the reactor to the full power condition with an elevated radial power peaking factor consequent to the dropped RCCA. Elevated clad heat flux in the hot assembly may result in an approach to the DNBR SAFDL.

An automatic and redundantly actuated reduction in turbine load demand (turbine runback) is also provided as protection for this event in the manual mode. When turbine runback occurs, the turbine load reduction reduces secondary steam flow, causing a tendency for the secondary side temperature and pressure to increase. Thus the primary coolant temperature decrease characteristic of the event is mitigated, reducing the reactivity insertion contingent on cooldown and reducing the ultimate power level at which the reactor stabilizes.

A dropped assembly or assembly bank is detected by:

1) Sudden drop in the core power level as seen by the Nuclear Instrumentation System.

2) Asymmetric power distribution as seen on out-of-core neutron detectors or core exit thermocouples.

3) Rod at bottom signal.

4) Rod deviation alarm.

5) Rod position indication.

The Dropped Full Length RCCA or RCCA Bank event is classified as a Condition II event. The acceptance criteria for this event are:

1) The pressures in the reactor coolant and main steam systems should be less than 110% of design values.

2) The fuel cladding integrity should be maintained by ensuring that fuel design limits are not exceeded. This is demonstrated by assuring that the minimum calculated departure from nucleate boiling ratio (DNBR) is not less than the applicable limits of the DNBR correlation being used and that fuel centerline melt does not occur.

The most adverse credible single failure assumption in the Dropped Full Length RCCA event is the loss of one NIS channel. This results in a reduced ability for the NIS and the ARC system to detect the core power redistribution characteristic of the event. In cases where a power range high neutron negative flux rate trip intercedes, the trip is delayed until two of the three remaining channels reach the trip setpoint. In cases where the plant stabilizes at a new equilibrium condition without a reactor trip, no further protective action is required and this assumption has no impact on the RPS response. Therefore, consideration of other single failures within the protection system is not applicable.

15.4.3.1.2 Analysis of effects and consequences

Method of Analysis - The characteristic system response for this event is strongly dependent on the neutron kinetics feedbacks, the worth of the dropped RCCA or RCCA bank, and on the availability of the ARC system. The Dropped RCCA transients are analyzed in two operational states: manual and automatic rod control. This event is evaluated at both BOC and EOC conditions for a spectrum of drop rod worths, differential worths (ARC case only), and rod shadowing factors (ARC case only).

The event considers the radial redistribution of power in the core, and can result in radial peaking factors in excess of Technical Specification limits. The analysis is performed by coupling a conservative power peak to transient response and DNB calculations. The power peak associated with each event is characterized through an augmentation factor which relates the maximum power peak to the steady-state power peak.

Conservative conditions established for the analysis of this event are presented in Table 15.4.3-1. Available reactor protection system trips are presented in Table 15.0.6-2. Key operating parameters used in the analysis of this event are presented in Table 15.4.3-2. The range of neutronics parameters supported by this analysis are presented in Table 15.4.3-3.

This event is analyzed using the SGR/Uprating value for T_{avg} (588.8°F).

The analysis of rod drop events is performed using PRISM (Reference 15.4.3-1), ANF-RELAP (Reference 15.4.1-3) and XCOBRA IIIC (References 15.4.1-4 and 15.4.1-5). The PRISM code is used to calculate neutronic parameters such as rod worth and power peaking augmentation factors. The transient response of the reactor system is calculated using the ANF-RELAP computer program. The core thermal hydraulic boundary conditions from the ANF-RELAP calculation are used as input to the XCOBRA-IIIC code to predict the MDNBR for the event. A radial power peaking factor on $F_{\Delta H}$ is included in the MDNBR calculation to account for radial power redistribution effects typical of the event. A statistical analysis of MDNBR was performed using the methodology in Reference 15.4.3-3.

The dropped RCCA bank is distinguished from the dropped RCCA in the ANF-RELAP/XCOBRA-IIIC analysis only by the greater magnitude of rod bank worth and radial peaking augmentation factors associated with the dropped RCCA bank. Cases involving dropped RCCA banks do not result in appreciable degradation of MDNBR due to the action of the high negative flux rate trip, which is used in this analysis, combined with the large initial power reduction.

Results - The dropped RCCA transients are analyzed in two operational states: manual and automatic rod control. The ARC cases are more limiting than the manual rod control cases. The limiting ARC dropped RCCA transient occurred with BOC kinetics, a 10 pcm/step differential worth, a 150 pcm dropped rod worth, and a 0.0% RSF.

The sequence of events for the limiting Dropped RCCA transient is given in Table 15.4.3-4a. The responses to key system variables for the limiting transient are given in Figures 15.4.3-1 to 15.4.3-11.

15.4.3.1.3 Conclusions

The results of the analysis of the Dropped RCCA/Bank event demonstrate that the acceptance criteria are met. Both the maximum reactor primary system pressure and the maximum secondary system pressure are less than the design limits of 2750 psia and 1320 psia, respectively. The predicted MDNBR is greater than the safety limit. The critical heat flux correlation limit ensures that, with 95% probability and 95% confidence, DNB is not expected to occur; therefore, no fuel is expected to fail. The fuel centerline melt threshold is not penetrated during this event.

The analysis for this event supports full power operation at or below a nominal primary T_{avg} of 588.8°F.

15.4.3.2 Withdrawal of a Single Full Length RCCA

15.4.3.2.1 Identification of causes and accident description

The rod withdrawal event is initiated by an electrical or mechanical failure in the Rod control System that causes the inadvertent withdrawal of a single RCCA. A rod is withdrawn from the reactor core causing an insertion of positive reactivity which results in a power excursion transient, increasing the core heat flux and creating a challenge to DNB margin. The DNB margin is further reduced by the mismatch between the constant energy removal rate of the steam generators and the increased energy generation rate in the core which increases the primary system temperature.

The system response is essentially the same as that occurring in the Uncontrolled Bank Withdrawal at Power event (Event 15.4.2). The single RCCA withdrawal is distinguished from the withdrawal of an RCCA bank by the severe radial power redistribution. High radial peaking is localized in the region of the single withdrawn RCCA and may, in severe cases, surpass the design limits.

Automatic protection for this event is afforded by the high neutron flux trip (high setting) and the OTΔT trip. Because of the localized power peaking, there is the possibility of violating the

SAFDLs. This is acceptable since this event is classified as a Condition III event, in which a small fraction of fuel failure is permitted. The acceptance criteria for this event are:

1) The pressures in the reactor coolant and main steam systems should be less than 110% of design values.

2) A small fraction of fuel failures may occur, but these fuel failures should not hinder core coolability.

3) Radiological consequences should be less than 10% 10 CFR 50.67 limits (i.e.,2.5 rem TEDE) offsite and less than the 10CFR50.67 limit of 5 rem TEDE in the control room.

4) The event should not generate a limiting fault or result in the consequential loss of the reactor coolant or containment barriers.

No single electrical or mechanical failure in the Rod Control System could cause the accidental withdrawal of a single RCCA from the inserted bank at full power operation. The operator could deliberately withdraw a single RCCA in the control bank since this feature is necessary in order to retrieve an assembly should one be accidentally dropped. The event analyzed must result from multiple wiring failures (probability for single random failure is on the order of 10-4/year (refer to Section 7.7.2.2) or multiple significant operator actions and subsequent and repeated operator disregard of event indication. The probability of such a combination of conditions is low; however, the limiting consequences may include slight fuel damage.

In the extremely unlikely event of simultaneous electrical failures which could result in single RCCA withdrawal, rod deviation and rod control urgent failure would both be displayed on the plant annunciator, and the rod position indicators would indicate the relative positions of the assemblies in the bank. The urgent failure alarm also inhibits automatic rod motion in the group in which it occurs. Withdrawal of a single RCCA by operator action, whether deliberate or by a combination of errors, would result in activation of the same alarm and the same visual indications. Withdrawal of a single RCCA results in both positive reactivity insertion tending to increase core power, and an increase in local power density in the core area associated with the RCCA. Automatic protection for this event is provided by the overtemperature ΔT reactor trip, although due to the increase in local power density it is not possible in all cases to provide assurance that the core safety limits will not be violated.

No single failure will prevent operation of any system required to function.

15.4.3.2.2 Analysis of effects and consequences

Method of Analysis - This event is analyzed at full power.

The overall system response for a single RCCA withdrawal is identical to that of the analysis of the Uncontrolled RCCA Bank Withdrawal event (Event 15.4.2). The difference is in the local peaking in the region of the single withdrawn RCCA that is not present if the entire bank is withdrawn. Therefore, the MDNBR calculation for the most limiting full power case from the RCCA bank withdrawal analysis will be reevaluated with a conservative radial peaking augmentation factor.

This event is analyzed using the SGR/Uprating value for T_{avg} (588.8°F).

Results - The event considers the radial redistribution of power in the core, and can result in radial peaking factors in excess of Technical Specification limits. The analysis is performed by coupling a conservative power peak to transient response and DNB calculations. The power peak associated with each event is characterized through an augmentation factor which relates the maximum power peak to the steady- state power peak. The steady-state power distributions and augmentation factors are calculated with the PRISM reactor simulator.

In the analysis of the single RCCA withdrawal event, the core boundary conditions of average heat flux, temperature, pressure and flow are selected from the limiting Uncontrolled RCCA Bank Withdrawal event (Event 15.4.2). These core boundary conditions are then combined in an XCOBRA-IIIC calculation with a radial augmentation peaking factor calculated to bound the possible single rod withdrawal radial power redistribution.

15.4.3.2.3 Conclusions

The results of the analysis of the Single Control Rod Withdrawal event demonstrate that Condition III acceptance criteria are met. The predicted MDNBR is greater than the 95/95 safety limit. No fuel is predicted to fail based on DNB criteria. No fuel is predicted to fail based on FCM criteria. Enveloping, conservative fuel damage assumptions were used to determine radiological consequences. The radiological doses have been calculated based on a 4% assumed cladding failure, and 1% assumed Fuel centerline melting. The remaining input and modeling parameters are identical to the Locked Rotor (Section 15.3.3.3 and specifically Table 15.3.3-6) except that conservative gap fractions of 10% for all iodines and noble gases were used for the Single RCCA Withdrawal analysis. The analytically predicted dose consequence for the Control Room (CR) operators due to LBLOCA increased by a small amount, based on the reduction in CR circulation flow which reduces the iodine filtration provided by the charcoal filters relative to the analysis of record condition. The analytically predicted dose consequence to the control room operators for the Single Control Rod Withdrawal event was not similarly revised or updated since the LBLOCA DBA is the limiting control room operator dose event. The radiological analysis results for this event are presented in Table 15.4.3-5.

The analysis for this event supports full power operation at a nominal primary T_{avg}, between 580.8°F and 588.8°F, inclusive.

15.4.3.3 Statically Misaligned RCCA or Bank

15.4.3.3.1 Identification of causes and accident description

The static misalignment events occur when a malfunction of the Control Rod Drive (CRD) mechanism causes a control rod to be out of alignment with its bank. Misalignment occurs when the rod is either higher or lower than any of the other control rods in the same bank. During this event, the reactor is at steady-state rated full power conditions, and no excursion of core temperature, pressure, flow, or power occurs. For extreme RCCA misalignments, the core radial power distribution may be characterized by peaking factors in excess of design limits. Highly localized increases in clad heat flux, coolant temperature, and flow diversions may occur. In severe cases, the SAFDL on DNB may be approached.

This event is classified as a Condition II event as defined in Section 15.0.1.

Misaligned assemblies are detected by:

1) Asymmetric power distribution as seen on out-of-core neutron detectors or core exit thermocouples.

2) Rod deviation alarm.

3) Rod position indicators.

The deviation alarm alerts the operator to rod-to-rod deviations within the same bank in excess of 12 steps. If the rod deviation alarm is not operable, the operator is required to take action as required by the Technical Specifications.

If one or more rod position indicator channels should be out of service, operating instructions shall be followed to assure the alignment of the non-indicated assemblies. The operator is also required to take action as required by the Technical Specifications. Indirect means of checking control rod position presented in plant operating instructions include review of temperature indications from core outlet thermocouples and review of data provided by movable in-core neutron detectors. Review of in-core detector data is specifically required following any significant movement of the non-indicated control rod assembly.

No single failure will prevent operation of any system required to function.

15.4.3.3.2 Analysis of effects and consequences

Cycle operation with a statically misaligned RCCA or Bank could result in core power distributions which are significantly more peaked than predicted. Steady state power distributions are calculated in three dimensions for several misaligned cases. Full power operation with the most severe peaking at any core location resulting from undetected misalignments will be analyzed.

This event is analyzed using the nominal SGR/Uprating value for T_{avg} (588.8°F). The results of this analysis are valid for a T_{avg} of 588.8°F or less.

Key operating parameters used in the analysis of this event are the steady state initial conditions and biasing for the Rod Drop Accident presented in Table 15.4.3-1.

The XCOBRA-IIIC code (References 15.4.1-4 and 15.4.1-5) is used to predict MDNBR for this event.

15.4.3.3.3 Conclusions

The results of the analysis demonstrate that Condition II event acceptance criteria are met. The predicted MDNBR is greater than the safety limit. The critical heat flux correlation limit ensures that, with 95% probability and 95% confidence, DNB is not expected to occur; therefore, no fuel is expected to fail. The fuel centerline melt threshold is not penetrated during this event.

15.4.3.3.4 Deleted by Amendment No. 50

15.4.4 STARTUP OF AN INACTIVE REACTOR COOLANT PUMP AT AN INCORRECT TEMPERATURE

15.4.4.1 Identification of Causes and Accident Description

If the plant is operating with one pump out of service, there is reverse flow through the inactive loop due to the pressure difference across the reactor vessel. The cold leg temperature in an inactive loop is identical to the cold leg temperature of the active loops (the reactor core inlet temperature). If the reactor is operated at power, and assuming the secondary side of the steam generator in the inactive loop is not isolated, there is a temperature drop across the steam generator in the inactive loop and, with the reverse flow, the hot leg temperature of the inactive loop is lower than the reactor core inlet temperature.

Administrative procedures require that the Unit be brought to a load of less than 25 percent of full power prior to starting the pump in an inactive loop in order to bring the inactive loop hot leg temperature closer to the core inlet temperature. Starting of an idle reactor coolant pump without bringing the inactive loop hot leg temperature close to the core inlet temperature would result in the injection of cold water into the core, which would cause a reactivity insertion and subsequent power increase.

This event is classified as an ANS Condition II incident (a fault of moderate frequency) as defined in Section 15.0.1.

Should the startup of an inactive reactor coolant pump accident occur, the transient will be terminated automatically by a reactor trip when the power range neutron flux (two out of four channels) exceeds the P-8 setpoint, which has been previously reset for two loop operation.

No single active failure will adversely affect the consequences of the event.

15.4.4.2 DELETED

15.4.4.3 DELETED

15.4.4.4 AREVA Event Disposition

<u>Modes 1 and 2</u>: The event is incredible in these modes because the Plant Technical Specifications require that three reactor coolant pumps operate in Modes 1 and 2.

<u>Modes 3 through 6</u>: In these modes, the reactor is subcritical and there is no significant load on the plant. The potential for a significant reactivity excursion is nil. Even low levels of backflow through the inactive loop will preclude a static condition in which significant cooling of the inactive loop water inventory might occur, and the primary system will remain essentially isothermal. The consequences of the event in these modes are bounded by those of Event 15.4.1.

15.4.5 A MALFUNCTION OR FAILURE OF THE FLOW CONTROLLER IN A BWR LOOP THAT RESULTS IN AN INCREASED REACTOR COOLANT FLOW RATE

This section is not applicable to the Shearon Harris Nuclear Power Plant.

15.4.6 CHEMICAL AND VOLUME CONTROL SYSTEM MALFUNCTION THAT RESULTS IN A DECREASE IN THE BORON CONCENTRATION IN THE REACTOR COOLANT

15.4.6.1 Identification of Causes and Accident Description

Reactivity can be added to the core by feeding primary grade water into the RCS via the reactor makeup portion of the Chemical and Volume Control System (CVCS), resulting in decreasing boron concentration in the reactor coolant system. The dilution of primary system boron adds positive reactivity to the core. This event can lead to an erosion of shutdown margin for subcritical initial conditions, or a slow power excursion for at-power conditions. A boron dilution for at-power conditions behaves in a similar manner to a slow Uncontrolled RCCA Bank Withdrawal event (Event 15.4.2).

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 CVCS 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 the operator sufficient time to correct the situation in a safe and orderly manner.

The opening of the reactor makeup water control valve provides makeup to the RCS which can dilute the reactor coolant. Inadvertent dilution from this source can be readily terminated by closing the control valve. In order for makeup water to be added to the RCS at pressure, at least one charging pump must be running in addition to a reactor makeup water pump.

The rate of addition of unborated makeup water to the RCS is limited by operator response to alarm setpoints for reactor makeup water flow, charging flow, and letdown flow.

The boric acid from the boric acid tank is blended with primary grade water in the blender and the composition is determined by the preset flow rates of boric acid and primary grade water on the main control board.

In order to dilute two separate operations are required:

1) The operator must switch from the automatic makeup mode to the dilute mode.

2) The Stop-Start switch must be turned to the start position.

Omitting either step would prevent dilution.

Information on the status of the reactor coolant makeup is continuously available to the operator. Lights are provided on the main control board to indicate the operating condition of the pumps in the CVCS. Alarms are actuated to warn the operator if boric acid or demineralized water flow rates deviate from preset values as a result of system malfunction.

This event is classified as an ANS Condition II incident (a fault of moderate frequency) as defined in Section 15.0.1. The acceptance criteria for this event are:

1) During cold shutdown (Mode 5):

If operator action is required to terminate the transient, a minimum time interval of 15 minutes must be available between the time when an alarm announces an unplanned moderator dilution and the time of loss of shutdown margin.

2) During hot shutdown, hot standby, startup, and power operation (Modes 4, 3, 2, and 1):

If operator action is required to terminate the transient, a minimum time interval of 15 minutes must be available from the time of initiation of the dilution to the time of loss of shutdown margin.

A boron dilution event can result from any of the following:

1) Resin sluice connections to the CVCS and BTRS demineralizers.

2) Reactor makeup water connection to the BTRS.

3) Pumping water of unknown boron concentration from the recycle holdup tanks to the charging pump suction.

4) Reactor makeup water connection to the boric acid batching tank(s).

5) Operation of the BTRS in the dilution mode.

6) Malfunction of the CVCS reactor makeup control system.

All of the above except items 5 and 6 require the opening of normally closed local manual valves, some of which are normally locked closed. Measures required to prevent a dilution from these sources during refueling are addressed in Section 15.4.6.2.

No dilution accident can originate from safeguard systems based on the premise that the technical specifications require periodic verification of the boron concentration in the RWST or the accumulators.

The containment spray system has been evaluated and determined not to be a boron dilution source during shutdown.

The effect of increasing boron worth with dilution was taken into account for all cases analyzed. A conservatively low (most negative pcm/ppm) value was used.

For all boron dilution events analyzed, all fuel assemblies are in the core.

15.4.6.2 Analysis of Effects and Consequences

1) Method of Analysis - To cover all phases of the plant operation, boron dilution during refueling, startup, cold shutdown, hot shutdown, hot standby and power operation are

considered in this analysis. Per the applicable methodology (EMF-2310, Reference 15.0.11-22) an instantaneous mixing model is used for modes where at least one reactor coolant pump is operating. A dilution front model is used for modes where no RCPs are operating.

2) Dilution During Refueling - An uncontrolled boron dilution accident cannot occur during refueling as a result of a reactor coolant makeup system malfunction. This accident is prevented by administrative controls which isolate the RCS from the potential source of unborated water.

   During refueling operations, potential boron dilution is prevented by administrative control of the valves listed on Table 15.4.6-2. Any makeup which is required during refueling will be from a borated water source.

3) Dilution During Cold Shutdown - The boron dilution analysis during cold shutdown must ensure that the operator has at least 15 minutes after the actuation of the high flux at shutdown alarm to terminate the dilution before losing shutdown margin. The shutdown margin requirement for cold shutdown is dependent on the reactor coolant system (RCS) temperature and cycle burnup, with a minimum requirement of 1000 pcm.

A minimum RCS water volume of 3186 ft^3 is assumed in the analysis. This volume corresponds to the minimum RCS volume for mid loop operation with residual heat removal (RHR) cooling. The water in the reactor vessel is assumed to be drained to -76.5" below the top of the lower vessel head flange, which corresponds to an alarm in the control room. The total volume consists of the volume of the reactor vessel filled to -76.5", the portions of the HL and CL piping filled to -76.5" that are in the recirculation flow path (e.g. from RHR injection on the 3 CL to the vessel and from the vessel to the RHR return on one HL), one train of RHR, and those portions of the CVCS system which are in operation during mid loop conditions.

The dilution flow rate is conservatively assumed to be 132 gpm. This flow rate corresponds to the maximum capacity of the reactor makeup water system with one makeup water pump locked out.

4) Dilution During Hot Shutdown - The boron dilution analysis during hot shutdown must ensure that the operator has at least 15 minutes to terminate the dilution before losing shutdown margin. The shutdown margin requirement is dependent on the RCS temperature and cycle burnup, and whether a RCP is in operation or cooling is being provided by RHR. The shutdown margin requirement is at least 1770 pcm for all hot shutdown conditions.

For RCP operation, a minimum RCS water volume of 9157 ft^3 is assumed in the analysis. This is the active RCS volume, excluding the pressurizer and assuming at least one RCP in operation and 3% SGTP. For RHR cooling, a minimum RCS water volume of 4202 ft^3 is assumed in the analysis. This volume corresponds to the minimum RCS volume that receives sufficient RHR system flow to be considered part of the RCS active volume. This includes the reactor pressure vessel (minus the upper head region), one reactor coolant loop, and one train of RHR. The calculated volume also assumes 3% SGTP.

The dilution flow rate is limited by the combined capacity of the two reactor makeup water pumps with the RCS at 135 psia and 200°F. This flow rate is conservatively assumed to be 321 gpm. Mixing of the reactor coolant is provided by the operation of one RHR pump.

5) Dilution During Hot Standby - The boron dilution analysis during hot standby must ensure that the operator has at least 15 minutes to terminate the dilution before losing shutdown margin. The shutdown margin requirement is temperature and exposure dependent, with a minimum requirement of 1770 pcm for required RCS boron concentrations below 1500 ppm and one RCP running. The shutdown margin requirement is at least 1770 pcm for all hot standby conditions.

   A minimum RCS water volume of 8744 ft^3 is assumed in the analysis. This is the active RCS volume, excluding the pressurizer and assuming at least one RCP in operation and 3% SGTP.

   The dilution flow rate is conservatively assumed to be 321 gpm, which is the combined capacity of the two reactor makeup water pumps with the RCS at 135 psia and 200°F. Mixing of the reactor coolant is provided by the operation of one RCP or one RHR pump.

6) Dilution During Startup - The boron dilution during startup is analyzed to ensure that the operator has at least 15 minutes to terminate the dilution before losing shutdown margin. The shutdown margin requirement is 1770 pcm.

   The analysis is performed with a minimum RCS water volume of 9001 ft^3, which conservatively excludes the pressurizer volume and assumes that at least one RCP is in operation and 3% SGTP.

   During startup, the capacity of the charging pumps limits the dilution flow rate. A conservatively high charging pump flow rate of 321 gpm is assumed in the analysis. Mixing of the reactor coolant is provided by one RCP.

7) Dilution During Full Power Operation - With the reactor at full power and in manual rod control, boron dilution will produce a power and temperature transient similar to that in a RCCA withdrawal accident. The Uncontrolled Bank Withdrawal at Power event (Event 15.4.2) bounds the boron dilution event with manual rod control since the reactivity insertion rates for the boron dilution are within the range analyzed for the RCCA withdrawal.

Due to how the boron concentrations for the Mode 2 (startup) dilution analysis are calculated, it will always bound operation in Mode 1 (full power operation) with respect to the time till loss of shutdown margin.

15.4.6.3 Results and Conclusions

The results of the boron dilution analysis show that the current Technical Specification shutdown margin requirements provide the operator with adequate time to manually terminate the source of dilution flow during operational Modes 1 to 5, including Mode 4 with no RCPs operable. Boron dilution during power operation is bounded by the analysis presented in Section 15.4.2. No analysis was performed for a boron dilution event in Mode 6, since

administrative controls are in place to prevent an uncontrolled boron dilution while the unit is in the refueling mode.

15.4.7 INADVERTENT LOADING AND OPERATION OF A FUEL ASSEMBLY IN AN IMPROPER POSITION

15.4.7.1 Identification of Causes and Accident Description

Fuel and core loading errors such as can arise from the inadvertent loading of one or more fuel assemblies into improper positions, loading a fuel rod during manufacture with one or more pellets of the wrong enrichment or the loading of a full fuel assembly during manufacture with pellets of the wrong enrichment will lead to increased heat fluxes if the error results in placing fuel in core positions calling for fuel of lesser enrichment. Also included among possible core loading errors is the inadvertent loading of one or more fuel assemblies requiring burnable poison rods into a new core without burnable poison rods. These loading errors can result in severe changes in the core power distribution which may be undetectable by the incore instrumentation.

Any error in enrichment, beyond the normal manufacturing tolerances, can cause power shapes which are more peaked than those calculated with the correct enrichments. There is a 5 percent uncertainty margin included in the design value of power peaking factor assumed in the analysis of ANS Condition I and ANS Condition II transients. The in-core system of moveable flux detectors which is used to verify power shapes at the start of life, is capable of revealing any assembly enrichment error or loading error which causes power shapes to be peaked in excess of the design value.

Fuel assembly loading errors are prevented by administrative procedures and controls implemented during fabrication. To reduce the probability of core loading errors, each fuel assembly is marked with an identification number and loaded in accordance with a core loading diagram. During core loading, the identification number will be checked before each assembly is moved into the core. After core loading is completed, a core map will be performed as a further check on proper placing of the fuel in the core.

The power distortion due to any combination of misplaced fuel assemblies would significantly raise peaking factors and would be readily observable with in-core flux monitors. In addition to the flux monitors, thermocouples are located at the outlet of about one third of the fuel assemblies in the core. There is a high probability that these thermocouples would also indicate any abnormally high coolant enthalpy rise. In-core flux measurements are taken during the startup subsequent to every refueling operation.

This event is classified as an ANS Condition III incident (an infrequent fault) as defined in Section 15.0.1. The acceptance criteria for this event are:

1) The pressures in the reactor coolant and main steam systems should be less than 110% of design values.

2) A small fraction of fuel failures may occur, but these fuel failures should not hinder core coolability.

3) Radiological consequences should be less than 10% of 10 CFR 50.67 limits (i.e., 2.5 rem TEDE) offsite and less than 10 CFR 50.67 limit of 5 rem TEDE in the control room.

15.4.7.2 Analysis of Effects and Consequences

Method of Analysis - Cycle operation with an improperly loaded core could result in core power distributions which are significantly more peaked than predicted. Steady state power distributions are calculated in three dimensions for several fuel misload cases. For each case analyzed, the assembly powers in instrumented core locations are compared to a normally loaded core to determine if the case would be detected at the time of the initial low-power flux map used in verifying that the core is properly loaded. Full power operation with the most severe peaking at any core location resulting from undetected misloadings will be analyzed.

Misloadings which exceed the criteria from the 30% flux map are detectable with the incore instrumentation system. Misloadings which are undetectable at the time of the 30% flux map are analyzed to ensure fuel failures in excess of allowed limits will not occur as a result of this event.

The misload assembly detection criteria are a function of the number of operable thimble locations during the initial 30% power flux map. The misload detection criteria are based on the maximum difference between measured and predicted reaction rates (detector signal) and the maximum ratio of reaction rates in symmetric thimbles. If the map is taken with fewer than 38 thimbles or has observed results exceeding the review criteria, further evaluation of the flux map for a potential misload is required.

This event is analyzed using the nominal SGR/Uprating value for T_{avg} (588.8°F). The results of this analysis are valid for a T_{avg} of 588.8°F or less.

Key operating parameters used in the analysis of this event are presented in Table 15.4.7-1.

The XCOBRA-IIIC code (References 15.4.1-4 and 15.4.1-5) is used to predict MDNBR for this event.

15.4.7.3 Conclusions

The results of the analysis demonstrate that Condition III event acceptance criteria are met. The predicted MDNBR is greater than the 95/95 safety limit. No fuel is predicted to fail based on DNB criteria. No fuel is predicted to fail based on FCM criteria. Enveloping, conservative fuel damage assumptions were used to determine the predicted radiological consequences. The radiological doses have been calculated based on 4% assumed cladding failure, and 1% assumed centerline fuel melting. These are the same assumptions as were made in Section 15.4.3.2.3, with the following clarification. The analytically predicted dose consequence to the Control Room (CR) operators due to the LBLOCA increased by a small amount, based on the reduction in CR recirculation flow which reduces the iodine filtration provided by the charcoal filters relative to the analysis of record condition. The analytically predicted dose consequence to the control room operators for the inadvertent loading and operation of a fuel assembly in an improper position event was not similarly revised or updated since the LBLOCA DBA is the limiting control room operator dose event.

The analysis for this event supports full power operation at nominal primary T_{avg} between 580.8°F and 588.8°F, inclusive.

15.4.8 SPECTRUM OF ROD CLUSTER CONTROL ASSEMBLY EJECTION ACCIDENTS

15.4.8.1 Identification of Causes and Accident Description

This accident is defined as the mechanical failure of a control rod mechanism pressure housing resulting in the ejection of a RCCA and drive shaft. The consequence of this mechanical failure is a rapid positive reactivity insertion together with an adverse core power distribution, possibly leading to localized fuel rod damage. The transient is terminated by the Doppler reactivity effects of increased fuel temperature, and by an automatic reactor trip on high neutron flux. This event challenges deposited enthalpy, radiological consequences and pressurization acceptance criteria.

15.4.8.1.1 Design precautions and protection

Certain features in the SHNPP pressurized water reactors are intended to preclude the possibility of a rod ejection accident, or to limit the consequences if the accident were to occur. These include a sound, conservative mechanical design of the rod housings, together with a thorough quality control (testing) program during assembly, and a nuclear design which lessens the potential ejection worth of RCCA's, and minimizes the number of assemblies inserted at high power levels.

Mechanical Design - The mechanical design is discussed in Section 4.6. Mechanical design and quality control procedures intended to preclude the possibility of a RCCA drive mechanism housing failure are listed below:

1) Each full length control rod drive mechanism housing is completely assembled and shop tested at 4100 psi.

2) The mechanism housings are individually hydrotested after they are attached to the head adapters in the reactor vessel head, and checked during the hydrotest of the completed Reactor Coolant System.

3) Stress levels in the mechanism are not affected by anticipated system transients at power, or by the thermal movement of the reactor coolant loops. Moments induced by the design earthquake can be accepted within the allowable primary working stress range specified by the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section III, for Class 1 components.

4) The latch mechanism housing and rod travel housing are each a single length of forged Type 304 stainless steel. This material exhibits excellent notch toughness at all temperatures which will be encountered.

A significant margin of strength in the elastic range together with the large energy absorption capability in the plastic range gives additional assurance that gross failure of the housing will not occur. The joints between the latch mechanism housing and head adapter, and between the latch mechanism housing and rod travel housing, are threaded joints reinforced by canopy type rod welds. Administrative regulations require periodic inspections of these (and other) welds.

Nuclear Design - Even if a rupture of a RCCA drive mechanism housing is postulated, the operation utilizing chemical shim is such that the severity of an ejected RCCA is inherently limited. In general, the reactor is operated with the RCCA's inserted only far enough to permit load follow. Reactivity changes caused by core depletion and xenon transients are compensated by boron changes. Further, the location and grouping of control RCCA banks are selected during the nuclear design to lessen the severity of a RCCA ejection accident. Therefore, should a RCCA be ejected from its normal position during full power operation, only a minor reactivity excursion, at worst, could be expected to occur.

However, it may be occasionally desirable to operate with larger than normal insertions. For this reason, a rod insertion limit is defined as a function of power level. Operation with the RCCA's above this limit guarantees adequate shutdown capability and acceptable power distribution. The position of all RCCAs is continuously indicated in the Control Room. An alarm will occur if a bank of RCCAs approaches its insertion limit or if one RCCA deviates from its bank. A low alarm requires boron addition by following normal procedures with the CVCS. A low-low alarm requires boron addition by following the emergency boration procedure.

Reactor Protection - The reactor protection in the event of a rod ejection accident is provided by high neutron flux trip (high and low setting) and high rate of neutron flux increase trip. These protection functions are described in detail in Section 7.2.

Effects on Adjacent Housings - Disregarding the remote possibility of the occurrence of a RCCA mechanism housing failure, investigations have shown that failure of a housing due to either longitudinal or circumferential cracking would not cause damage to adjacent housings. The full length control rod drive mechanism is described in Section 3.9.4.

Effects of Rod Travel Housing Longitudinal Failures - If a longitudinal failure of the rod travel housing should occur, the region of the position indicator assembly opposite the break would be stressed by the reactor coolant pressure of 2250 psia. The most probable leakage path would be provided by the radial deformation of the position indicator coil assembly, resulting in the growth of axial flow passages between the rod travel housing and the hollow tube along which the coil assemblies are mounted.

If failure of the position indicator coil assembly should occur, the resulting free radial jet from the failed housing could cause it to bend and contact adjacent rod housings. If the adjacent housings were on the periphery, they might bend outward from their bases. The housing material is quite ductile; plastic hinging without cracking would be expected. Housings adjacent to a failed housing, in locations other than the periphery, would not be bent because of the rigidity of multiple adjacent housings.

Effect of Rod Travel Housing Circumferential Failures - If circumferential failure of a rod travel housing should occur, the broken-off section of the housing would be ejected vertically because the driving force is vertical and the position indicator coil assembly and the drive shaft would tend to guide the broken-off piece upwards during its travel. Travel is limited by the missile shield, thereby limiting the projectile acceleration. When the projectile reached the missile shield it would partially penetrate the shield and dissipate its kinetic energy. The water jet from the break would continue to push the broken-off piece against the missile shield.

If the broken-off piece of the rod travel housing were short enough to clear the break when fully ejected, it would rebound after impact with the missile shield. The top end plates of the position

indicator coil assemblies would prevent the broken piece from directly hitting the rod travel housing of a second drive mechanism. Even if a direct hit by the rebounding piece were to occur, the low kinetic energy of the rebounding projectile would not cause significant damage.

Possible Consequences - From the above discussion, the probability of damage to an adjacent housing must be considered remote. However, even if damage is postulated, it would not be expected to lead to a more severe transient since RCCAs are inserted in the core in symmetric patterns, and control rods immediately adjacent to worst ejected rods are not in the core when the reactor is critical. Damage to an adjacent housing could, at worst, cause that RCCA not to fall on receiving a trip signal; however this is already taken into account in the analysis by assuming a stuck rod adjacent to the ejected rod.

Summary - The considerations given above lead to the conclusion that failure of a control rod housing, due either to longitudinal or circumferential cracking, would not cause damage to adjacent housings that would increase severity of the initial accident.

15.4.8.1.2 Limiting criteria

This event is classified as an ANS Condition IV incident. See Section 15.0.1 for a discussion of ANS classifications. Due to the extremely low probability of a RCCA ejection accident, some fuel damage could be considered an acceptable consequence. The acceptance criteria for this event are:

1) Reactivity excursions should not result in a radially averaged enthalpy greater than 280 cal/gm at any axial location in any fuel rod.

2) The maximum reactor pressure during any portion of the assumed excursion should be less than the value that will cause stresses to exceed the "Service Limit C" as defined in Section III of the ASME Boiler and Pressure Vessel Code (i.e., maximum reactor pressure should be less than 120% of design values).

3) Offsite Radiological consequences should be less than ~25% of the 10 CFR 50.67 limits (i.e., 6.3 rem TEDE) and less than the 10CFR50.67 limit of 5 rem TEDE in the control room.

No single failure of the reactor protection system will negate the protection functions required for this event.

15.4.8.2 Analysis of Effects and Consequences

Method of Analysis - Two sets of cases are analyzed for this event: one challenging the SAFDL, and one to determine the pellet energy deposition resulting from an ejected rod. In the MDNBR case the parameters and equipment states have been selected to reduce the primary system pressurization to provide a conservative calculation of the minimum DNBR during the transient.

The MDNBR cases are evaluated using both BOC and EOC conditions. The MDNBR cases are analyzed using the SGR/Uprating value for T_{avg} (588.8°F).

Conservative conditions established for the analysis of this event are presented in Table 15.4.8-1. Available reactor protection system trips are presented in Table 15.0.6-2. All trips listed in this table except OPΔT were modeled for the analysis of this event. Key operating parameters used in the cases evaluating MDNBR (and radiological consequences) for this event are presented in Table 15.4.8-2. The range of neutronics parameters supported by this analysis is presented in Table 15.4.8-3.

The transient response of the reactor system is calculated using the ANF-RELAP (Reference 15.4.1-3) computer program. The core thermal hydraulic boundary conditions from the ANF-RELAP calculation are used as input to the XCOBRA-IIIC code (References 15.4.1-4 and 15.4.1-5) to predict the MDNBR for the event.

The rod ejection accident is also evaluated with the procedures developed in Reference 15.4.8-8 to determine the pellet energy deposition resulting from an ejected rod.

Results

The amount of fuel failure is predicted by comparing the MDNBR to the 95/95 safety limit. Less than 4% of the fuel is predicted to fail based on DNB criteria. No fuel is predicted to fail based on FCM criteria. Enveloping conservative fuel damage assumptions were used to determine the predicted radiological consequences. The radiological doses have been calculated based on 4% assumed cladding failure, and 2% assumed centerline fuel melting.

The sequence of events for the EOC HZP MDNBR case is given in Table 15.4.8-4b. This transient tripped on power range neutron flux, high setting. The responses to key system variables are given in Figures 15.4.8-9 to 15.4.8-16.

The sequence of events for EOC HZP fuel centerline temperature case is given in Table 15.4.8-4b. The responses to key system variables are given in Figures 15.4.8-9 to 15.4.8-16.

The rod ejection accident is evaluated for deposited enthalpy with the methodology in Reference 15.4.8-8. The ejected rod worths and hot pellet peaking factors were calculated using the PRISM code (Reference 15.4.3-1). No credit was taken for the power flattening effects of Doppler or moderator feedback in the calculation of ejected rod worths or resultant post-transient peaking factors. The pellet energy deposition resulting from an ejected rod was conservatively evaluated explicitly for BOC and EOC conditions, at HFP and HZP. The rod ejection accident results in a peak pellet enthalpy less than the 280 cal/gm limit.

15.4.8.3 Radiological Consequences of a Postulated Rod Ejection Accident

It is assumed that a mechanical failure of a control rod mechanism pressure housing has occurred, resulting in the ejection of a rod cluster control assembly and drive shaft. As a result of the accident, some fuel clad damage and a small amount of fuel melt are assumed to occur. Due to the pressure differential between the primary and secondary systems, radioactive reactor coolant is discharged from the primary into the secondary system. A portion of this radioactivity is released to the outside atmosphere through either the atmospheric relief valves or the main steam safety valves. Iodine and alkali metals group activity is contained in the secondary coolant prior to the accident, and some of this activity is released to the atmosphere as a result of steaming the steam generators following the accident. Finally, radioactive reactor coolant is discharged to the containment via the spill from the opening in the reactor vessel

head. A portion of this radioactivity is released through containment leakage to the environment.

The analysis of the rod ejection radiological consequences uses the analytical methods and assumptions outlined in RG 1.183, Appendix H. Separate calculations are performed to calculate the dose resulting from the release of activity to containment and subsequent leakage to the environment and the dose resulting from the leakage to activity to the secondary system and subsequent release to the environment. The total offsite and control room doses are the sum of the doses resulting from each of the postulated release paths and nuclides considered. A summary of input parameters and assumptions is provided in Table 15.4.8-5. Additional clarification is provided as follows:

Source Term

a) In determining the offsite doses following a rod ejection accident, it is assumed that 4% of the fuel rods in the core suffer sufficient damage that all of their gap activity is released and that 2% of the fuel in the core melts. Ten percent of the total core activity of iodine and noble gases and 12 percent of the total core activity for alkali metals are assumed to be in the fuel-cladding gap. In the calculation of activity releases from the failed/melted fuel the maximum radial peaking factor of 1.73 was applied. The core fission product inventory is given in Table 15.0.9-1.

b) For both the containment leakage release path and the primary to secondary leakage release path all noble gas and alkali metal activity contained in the failed fuel gap and in the melted fuel is available for release.

c) For the containment leakage release path all of the iodine contained in the failed fuel gap and 25 percent of the activity contained in the melted fuel is available for release.

d) For the primary to secondary leakage release path all of the iodine contained in the failed fuel gap and 50 percent of the activity contained in the melted fuel is available for release from the reactor coolant system.

e) Prior to the accident the iodine activity concentration of the primary coolant is 1.0 μCi/gm of DE I-131. The iodine activity concentration in the primary coolant is given in Table 15.0.9-7. The noble gas and alkali metal activity concentration in the RCS at the time the accident occurs is based on a one percent fuel defect level. The noble gas and alkali metal activity concentration in the RCS is given in Table 15.0.9-2. The iodine activity of the secondary coolant at the time the rod ejection occurs is assumed to be equivalent to the Technical Specification limit of 0.1 μ Ci/gm of DE I-131. The iodine activity concentration of the secondary coolant is given in Table 15.0.9-7. The alkali metal activity concentration of the secondary coolant at the time the rod ejection occurs is assumed to be 10% of the primary side concentration.

f) Iodine in containment is assumed to be 4.85% elemental, 0.15% organic and 95% particulate.

g) Iodine released from the secondary system is assumed to be 97% elemental and 3% organic.

Containment Release Pathway

a) The containment is assumed to leak at the design leak rate of 0.1% per day for the first 24 hours of the accident and then to leak at half that rate (0.05% per day) for the remainder of the 30 day period following the accident considered in the analysis.

b) For the containment leakage pathway, no credit is taken for plateout onto containment surfaces or for containment spray operation which would remove airborne particulate and elemental iodine. Sedimentation of alkali metal particulate in containment is credited.

Primary to secondary Leakage Release Pathway

a) When determining doses due to the primary to secondary steam generator tube leakage, all the iodine, alkali metals group and noble gas activity (from prior to the accident and resulting from the accident) is assumed to be in the primary coolant (and not in the containment). The primary to secondary tube leakage and steaming from the steam generators continue until the reactor coolant system pressure drops below the secondary pressure. A conservative time of 2 hours was used for this analysis, although analyses of the small break LOCA pressure transient have shown that this would occur well before that time. The rod ejection pressure transient is similar to that of a small break LOCA.

b) The amount of primary to secondary SG tube leakage is assumed to be 1 gpm total. Although the primary to secondary pressure differential drops throughout the event, the constant flow rate is conservatively maintained.

c) An iodine partition factor in the SGs of 0.01 (curies iodine/gm steam) / (curies iodine/gm water) is used. This partition factor is also applied to alkali metals.

d) All noble gas activity carried over to the secondary side through SG tube leakage is assumed to be immediately released to the outside atmosphere.

15.4.8.3.1 Offsite Doses

The Offsite doses are calculated using the assumptions and equations in Section 15.0A.1.

15.4.8.3.2 Control Room Doses

The control room assumptions are provided in Section 15.6.5.4.3 and Table 15.6.5-15.

In the rod ejection accident, the SI setpoint will be reached within 30 seconds from event initiation. The SI signal causes the control room HVAC to switch from the normal operation mode to the post-accident recirculation mode of operation. A 15-second delay for the control room to switch between normal and post-accident recirculation modes is modeled. An operator action switches the control room from the post-accident recirculation mode to the pressurization mode at 2 hours after event initiation.

15.4.8.3.3 Results

The potential radiological consequences resulting from a rod ejection have been conservatively analyzed, using assumptions and models described in previous sections. The TEDE doses have been analyzed for the worst two hours at the EAB and for the duration of the event at the LPZ and in the control room. The analytically predicted dose consequence to the Control Room (CR) operators due to LBLOCA increased by a small amount, based on the reduction in CR recirculation flow which reduces the iodine filtration provided by the charcoal filters relative to the analysis of record condition. The analytically predicted dose consequence to the control room operators for the Control Rod Ejection event was not similarly revised or updated since the LBLOCA DBA is the limiting control room operator dose event. The radiological analysis results for this event are listed in Table 15.4.8-6. The resultant doses are within the guideline values. The doses reported in Table 15.4.8-6 are for the combined release pathways of containment leakage and steaming from the secondary system for 2 hours. These doses are bounding for the postulated situation in which the steaming from the secondary system is the only release pathway even though in that situation steaming would continue for 8 hours (as described in Reg Guide 1.183, Appendix H, Section 7).

15.4.8.4 Conclusions

The results of the analysis demonstrate that the event acceptance criteria are met. For both the overpressurization case and MDNBR case, the maximum reactor pressure is significantly less than the requirements of Section III of the ASME Boiler and Pressure Vessel Code per Regulatory Guide 1.77. For the limiting MDNBR case, the predicted fuel failures are less than the 4% supported by the radiological analysis. For the limiting FCM case, the predicted fuel failures are less than the 2% supported by the radiological analysis.

The rod ejection accident results in an energy deposition of less than the 280 cal/gm limit.

The analysis for this event supports nominal primary T_{avg} operation at full power between 588.8°F and 580.8°F, inclusive.

15.4.9 SPECTRUM OF ROD DROP ACCIDENTS IN A BWR

This section is not applicable to the Shearon Harris Nuclear Power Plant.

REFERENCES: SECTION 15.4

15.4.1-1 Deleted by Amendment No. 45.

15.4.1-2 Hargrove, H. G., "FACTRAN - A Fortran-IV Code for Thermal Transients in a UO_2 Fuel Rod," WCAP-7908-A, December 1989.

15.4.1-3 ANF-RELAP Methodology for Pressurized Water Reactors: Analysis of Non-LOCA Chapter 15 Events, ANF-89-151(P)(A), Advanced Nuclear Fuels Corporation, Richland, WA 99352, May 1992.

15.4.1-4 Application of Exxon Nuclear Company PWR Thermal Margin Methodology to Mixed Core Configurations, XN-NF-82-21(P) (A), Revision 1, Exxon Nuclear Company, Richland, WA 99352, September 1983.

15.4.1-5 XCOBRA-IIIC: A Computer Code to Determine the Distribution of Coolant During Steady-State and Transient Core Operation, XN-NF-75-21(P)(A), Revision 2, Exxon Nuclear Company, Richland, WA 99352, January 1986.

15.4.2-1 Deleted by Amendment No. 51

15.4.2-2 Deleted by Amendment No. 45.

15.4.2-3 SRP Chapter 15 Non-LOCA Methodology for Pressurized Water Reactors, EMF-2310(P)(A), Revision 1, Framatome ANP, Inc., May 2004

15.4.3-1 EMF-96-029(P)(A) Volumes 1 and 2, "Reactor Analysis System for PWRs Volume 1 - Methodology Description Volume 2 - Benchmarking Results", Siemens Power Corporation, January 1997.

15.4.3-2 Deleted by Amendment No. 45.

15.4.3-3 Statistical Setpoint/Transient Methodology for Westinghouse Type Reactors, EMF-92-081(P)(A), Revision 1, Framatome ANP Richland, Inc., February, 2000.

15.4.4-1 Deleted by Amendment No. 51

15.4.4-2 Deleted by Amendment No. 51

15.4.4-3 Deleted by Amendment No. 51

15.4.4-4 Deleted by Amendment No. 51

15.4.6-1 Memorandum from Rudy Oliver to Bill Slover dated June 24, 1994 "Harris Nuclear Plant Charging/Safety Injection Pump Flow Inadvertent Boron Dilution Analysis for Mode 2 Operation."

15.4.8-1 Deleted by Amendment No. 45.

15.4.8-2 Deleted by Amendment No. 45.

15.4.8-3 Deleted by Amendment No. 45.

15.4.8-4 Deleted by Amendment No. 45.

15.4.8-5 Deleted by Amendment No. 45.

15.4.8-6 Deleted by Amendment No. 45.

15.4.8-7 Deleted by Amendment No. 45.

15.4.8-8 A Generic Analysis of the Control Rod Ejection Transient for Pressurized Water Reactors, XN-NF-78-44(NP)(A), Exxon Nuclear Company, Richland, WA 99352, October 1983.

15.5 INCREASE IN REACTOR COOLANT INVENTORY

Discussion and analysis of the following events are presented in this section:

1) Inadvertent operation of the Emergency Core Cooling System (IOECCS) during power operation (ANS Condition II event).

2) Chemical and Volume Control system malfunction that increases reactor coolant inventory (ANS Condition II event).

15.5.1 INADVERTENT OPERATION OF THE EMERGENCY CORE COOLING SYSTEM DURING POWER OPERATION

15.5.1.1 Identification of Causes and Accident Description

Spurious Emergency Core Cooling System (ECCS) operation at power could be caused by operator error or a false electrical actuation signal. A spurious signal may originate from any of the safety injection system actuation channels as described in Section 7.3.

Following the actuation signal, the suction of the charging pumps is re-aligned to the Refueling Water Storage Tank (RWST) from the Volume Control Tank (VCT). The valves isolating the Boron Injection Tank (BIT) from the charging pumps and the valves isolating the BIT from the injection header automatically open. The charging pumps then force concentrated boric acid from the RWST into the Reactor Coolant System (RCS). If a reactor trip does not occur coincident with safety injection actuation, the turbine throttle valves will open to offset the addition of negative reactivity from the Safety Injection System (SIS). The transient is eventually terminated by the reactor protection system due to low pressurizer pressure or manual trip. The time to trip is affected by the initial operating conditions including core burnup history that affects boron concentration, rate of change of boron concentration, and Doppler and moderator coefficients.

The operator will determine if Safety Injection should be terminated. For spurious occurrence, the operator would stop the safety injection after ensuring satisfactory plant conditions per operating procedures and maintain the plant in hot standby conditions.

15.5.1.2 Description of Analysis

For the HNP Power Uprate project, three ANF-RELAP transient analyses were performed. Two of the analyses were performed to examine minimum departure from nucleate boiling ratio (MDNBR) with beginning of cycle (BOC) and end of cycle (EOC) kinetics. The third case examines pressurizer overfill and the thermal-hydraulic conditions at the pressurizer Power Operated Relief Valve (PORV) and Safety Relief Valve (SRV) inlets.

BOC MDNBR Case: The main emphasis for this case is to minimize RCS pressure during the transient, thereby posing a challenge to MDNBR. This event was initiated from hot full power and an average temperature of 588.8°F to minimize the initial DNBR margin.

EOC MDNBR Case: This case also examines MDNBR consequences but the biasing is established with the intent of avoiding a premature reactor trip on low pressurizer pressure. With minimum boron concentration and automatic rod control enabled there is a possibility of a

power increase due either to automatic rod control (ARC) initiated rod movement (T_{avg} error) and/or positive moderator reactivity feedback. This event was initiated from hot full power and an average temperature of 588.8°F to minimize the initial DNBR margin.

Pressurizer Overfill Case: The intent of this case is to examine the fluid thermal-hydraulic conditions at the inlet to the pressurizer PORVs and SRVs. The event was biased to conservatively ensure that the fluid temperatures seen by the pressurizer valves will be as low as realistically achievable.

The transient response of the reactor system is calculated using the ANF-RELAP computer program (Reference 15.5.1-4). The Reference 15.5.1-4 methodology contains the following safety evaluation report (SER) restriction that relates to this event:

> *The [ANF-RELAP] methodology cannot be used in situations where... the boron tracking model is needed without further justification. If... the steam line break boron tracking model is used with the [ANF-RELAP] methodology, then the applicability of the steam line break methods to the non-LOCA event under considerations should be justified.*

The boron reactivity feedback model from the NRC approved MSLB methodology was conservatively implemented in this analysis. This model is justified as conservative for use in this application because the negative reactivity associated with the boron injection is delayed until the borated water is near the top of the core. Any positive feedback effects due to colder water entering the core is seen before the boron effects. This delay increases potential for higher core power levels that could result in lower MDNBRs. Addition of boron in the pressurizer overfill cases does not impact the results.

15.5.1.3 Acceptance Criteria

1) The pressure in the reactor coolant and main steam systems should be maintained below 110% of design values.

2) Fuel cladding integrity shall be maintained by ensuring that the minimum DNBR remains above the 95/95 limit for PWRs.

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

15.5.1.4 Results

An analysis was performed to support plant operation following steam generator replacement and power uprate (SGR/PUR). The analysis bounds plant operation for up to 2948 MWt rated power plus uncertainty (0.34% for a total analyzed power of 2958 MWt), and up to 3% steam generator tube plugging, and with RWST boron concentrations between the Technical Specification limits of 2400 ppm and 2600 ppm. The analysis was performed at RCS T_{avg} of 588.8°F and bounds operation at a reduced RCS T_{avg} of 580.8°F.

The results of the analysis are discussed with respect to each of the acceptance criteria as follows:

Acceptance Criteria 1

The pressure in the reactor coolant and main steam systems should be maintained below 110% of design value. As seen in Figures 15.5.1-4 15.5.1-11, and 15.5.1-17, the pressurizer/RCS systems are maintained below, at, or near the PORV and/or Safety valve setpoints, which adequately protect the design pressure of the RCS system.

Acceptance Criteria 2

Fuel cladding integrity shall be maintained by ensuring that the minimum DNBR remains above the 95/95 limit for PWRs. The analyses of the BOC and EOC cases show that this criterion is met as described below.

> BOC MDNBR Case: The sequence of events and results are given in Table 15.5.1-3. The responses of key system variables are given in Figures 15.5.1-1 to 15.5.1-7. The ANF-RELAP calculated DNBR trend clearly shows that this event is not a challenge to minimum DNBR since the DNBR generally increases throughout the event. MDNBR is predicted to occur shortly after event initiation, but is insignificantly different from the initial value. Thus, this event poses no challenge to the DNB specified acceptable fuel design limit (SAFDL).
>
> EOC MDNBR Case: The sequence of events and results are given in Table 15.5.1-4. The responses of key system variables are given in Figures 15.5.1-8 to 15.5.1-14. The ANF-RELAP calculated DNBR trend clearly shows that this event is not a challenge to minimum DNBR since the DNBR generally increases throughout the event. MDNBR is predicted to occur after event initiation, but is insignificantly different from the initial value. Thus, this event poses no challenge to the DNB SAFDL.

Acceptance Criteria 3

An incident of moderate frequency should not generate a more serious plant condition without other faults occurring independently. If the Reactor Protection System (RPS) initiates a reactor trip on a SIS signal, the plant would be brought to hot standby or cold shutdown condition after ensuring satisfactory plant conditions per operating procedures and Technical Specifications. For this condition, the potential exists for the pressurizer to overfill from continued ECCS injection. The liquid flow capacity of the pressurizer PORVs and/or SRVs greatly exceeds the capacity of the high-head safety injection system; thus, RCS overpressurization is not a concern.

The pressurizer PORVs and SRVs were previously evaluated to determine if the valves remain operable during the discharge if subcooled water in accordance with NUREG 0737 II.D.1 as outlined in FSAR - TMI Appendix. The PORVs were shown to remain operable at inlet pressures of 2532 to 2545 psia and temperatures of 446 to 670°F. The SRVs were shown to remain operable at an inlet pressure of 2475 psia and an inlet temperature of 635°F.

An analysis of the IOECCS event was performed to evaluate the thermal hydraulic conditions at the inlet to the pressurizer SRVs and PORVs subsequent to SGR/PUR. The event analysis was biased to conservatively ensure that the fluid temperatures seen by the pressurizer PORVs and SRVs are as low as realistically achievable. The assumed state of plant systems and input biases for this case is given in the pressurizer overfill columns of Tables 15.5.1-1 and 15.5.1-2.

Two of the three pressurizer PORVs are safety-related. The associated pneumatic power and controls are designed to function by remote manual operation. The controls associated with manual operation of the valve are safety-related and the accumulator and piping leading from the accumulator to the valve operators are safety-related. The remaining portions of the pneumatic supply and the automatic actuation/control system are not safety-related but are very reliable for the following reasons:

1) Reference 15.5.1-2 concludes that the PORV circuitry meets the requirements of NUREG-0737, Item II.D.1 stating that "the PORVs were qualified under the pump and valve operability program *(PVORT), the actuation transmitters are environmentally qualified*, the cable is qualified (although not run as 1E), and the PIC cabinets are essentially the same hardware as the class 1E cabinets.

2) The PORVs have two diverse pneumatic supplies. One supply is the nitrogen system and the other is the instrument air system. The instrument air system 1A and 1B air compressors can be manually loaded onto the "A" and "B" train emergency diesel generators, respectively, in the event of a loss of offsite power.

The PORVs were assumed to initially open to increase the rate of pressurizer fill. The PORVs were then conservatively modeled as having exhausted their motive air supply. At this point, the PORVs are modeled in the closed position, in order to allow pressure increases that would potentially challenge the operation of the SRVs. The sequence of events and results of the analysis are given in Table 15.5.1-5. The responses of key system variables are given in Figures 15.5.1-15 through 15.5.1-21. Figure 15.5.1-15 shows that the pressurizer level is at 100% span at approximately 600 seconds (10 minutes) and the pressurizer is filled solid at 918.5 sec's. The liquid inlet pressures and temperatures for the pressurizer PORVs and SRVs remain above 2250 psia and 635°F for approximately 950 seconds or 15.8 minutes (see Figures 15.5.1-20 and 15.5.1-21). The liquid temperature from 950 seconds to event termination at 1200 seconds remains above 564°F. For additional discussion of valve operability, see Section 5.2.2.2.

15.5.1.5 Conclusions

Consistent with the current licensing basis, the results demonstrate that the acceptance criteria for this event are met for SGR/Uprating conditions. The analysis for this event supports operation with the Model Delta 75 replacement steam generators at core power of 2948 MWt (with 0.34% uncertainty or a total power of 2958 MWt) with nominal primary T_{avg} at full power from 580.8°F to 588.8°F for steam generator tube plugging from 0% to 3%. The analysis bounds plant operation with RWST boron concentrations between the Technical Specification limits of 2400 ppm and 2600 ppm.

Conclusions with respect to each of the identified acceptance criteria are provided as follows:

Acceptance Criteria 1

The pressure in the reactor coolant and main steam systems should be maintained below 110% of design value. This acceptance criterion is not challenged since the SRV and PORV relief capacity far exceeds the ECCS capacity to fill the pressurizer.

Acceptance Criteria 2

Fuel cladding integrity shall be maintained by ensuring that the minimum DNBR remains above the 95/95 limit for PWRs. This acceptance criterion is not challenged since the DNBR margin increases throughout the event, except for an insignificant decrease early in the event.

Acceptance Criteria 3

The results of the analysis performed for SGR/PUR indicate that conditions at the inlet to the pressurizer SRVs are well within the range of conditions previously evaluated as acceptable for compliance with NUREG-0737 II.D.1 for almost 16 minutes after event initiation. Conditions remain well within those previously evaluated for the PORVs up to event termination at 20 minutes after event initiation. The PORVs are expected to mitigate the event, thereby eliminating the challenge to SRVs.

15.5.2 CHEMICAL AND VOLUME CONTROL SYSTEM MALFUNCTION THAT INCREASES REACTOR COOLANT INVENTORY

An increase in reactor coolant inventory which results from the addition of cold, unborated water to the RCS is analyzed in Section 15.4.6. An increase in reactor coolant inventory which results from the injection of highly borated water into the RCS is analyzed in Section 15.5.1.

15.5.3 A NUMBER OF BWR TRANSIENTS

This section is not applicable to the SHNPP.

REFERENCES: SECTION 15.5

15.5.1-1 FANP Letter VNG:00:292, Revision 1, "Transmittal of the Letter Report and Calculation Notebook for the Evaluation of IOECCS Event for Harris at Uprated Conditions" dated October 27, 2000.

15.5.1-2 Letter from NRC's Richard A. Becker to CP&L's Lynn Eury dated May 31, 1989 "Evaluation of Carolina Power and Light Company's Shearon Harris Unit 1, Plant Specific Submittals in Response to NUREG-0737, TMI Action Plan Requirement, Item II.D.1 (TAC No. 63565)."

15.5.1-3 EPRI NP-2628-SR, "EPRI PWR Safety and Relief Valve Test Program" dated December 1982.

15.5.1-4 ANF-89-151 (P)(A), ANF-RELAP Methodology for Pressurized Water Reactors: Analysis of Non-LOCA Chapter 15 Events, May 1992.

15.6 DECREASE IN REACTOR COOLANT INVENTORY[i]

Events which result in a decrease in reactor coolant inventory as discussed in this section are as follows:

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[i] Further information is contained in the TMI Appendix.

1) Inadvertent opening of a pressurizer safety or relief valve (ANS Condition II event).

2) Break in instrument line or other lines from reactor coolant pressure boundary that penetrate containment (ANS Condition II event).

3) Steam generator tube failure (ANS Condition IV event).

4) Loss of coolant accident (LOCA) resulting from a spectrum of postulated piping breaks within the reactor coolant pressure boundary (ANS Condition IV event for large-break LOCA and condition III event for small-break LOCA).

15.6.1 INADVERTENT OPENING OF A PRESSURIZER SAFETY OR POWER OPERATED RELIEF VALVE

15.6.1.1 Identification of Causes and Accident Description

An accidental depressurization of the Reactor Coolant System (RCS) could occur as a result of an inadvertent opening of a pressurizer power operated relief or safety valve.

Since a safety valve is sized to relieve approximately twice the steam flow rate of a power operated 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 is to increase power as the coolant density decreases, due to a negative moderator density coefficient at full power beginning-of-cycle conditions. The pressurizer water 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 signals:

1. Overtemperature ΔT.

2. Pressurizer low pressure.

An inadvertent opening of a pressurizer safety valve is classified as an ANS Condition II event, a fault of moderate frequency. See Section 15.0.1 for a discussion of ANS Condition II events. The acceptance criteria for this event are:

1) The pressures in the reactor coolant and main steam systems should be less than 110% of design values.

2) The fuel cladding integrity should be maintained by ensuring that fuel design limits are not exceeded. This is demonstrated by assuring that the minimum calculated departure from nucleate boiling ratio (DNBR) is not less than the applicable limits of the DNBR correlation being used and that fuel centerline melt does not occur.

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

4) For transients of moderate frequency in combination with a single failure, no loss of function of any fission product barrier, other than fuel element cladding, shall occur. Core geometry is maintained in such a way that there is no loss of core cooling capability and control rod insertability is maintained.

Plant systems and equipment which are necessary to mitigate the effects of RCS depressurization caused by an inadvertent safety valve opening are discussed in Section 15.0.8 and listed in Table 15.0.8-1.

Normal reactor control systems are not required to function. The Reactor Protection System functions to trip the reactor on the appropriate signal. No single active failure will prevent the Reactor Protection System from functioning properly.

15.6.1.2 Analysis of Effects and Consequences

Method of Analysis - This event is a depressurization of the primary coolant system. No pressurization criteria need therefore be addressed. The challenge to the SAFDLs results from the depressurization which occurs prior to reactor scram. The analysis utilized a steam flow rate from the pressurizer of 475,000 lbs/hr at 2500 psia. This is in excess of the rated flow rate for one pressurizer safety valve.

This event is analyzed using the SGR/Uprating value for T_{avg} (588.8°F).

The DNB results bound all lower temperature operating conditions.

Conservative conditions established for the analysis of this event are presented in Table 15.6.1-1. Available reactor protection system trips are presented in Table 15.0.6-2. Key operating parameters used in the analysis of this event are presented in Table 15.6.1-2. The range of neutronics parameters supported by this analysis are presented in Table 15.6.1-3.

The transient response of the reactor system is calculated using the ANF-RELAP (Reference 15.6.1-1) computer program. The core thermal hydraulic boundary conditions from the ANF-RELAP calculation are used as input to the XCOBRA-IIIC code (References 15.6.1-2 and 15.6.1-3) to predict the MDNBR for the event.

Results - The sequence of events for this analysis is given in Table 15.6.1-4. This transient tripped on low pressurizer pressure. The responses to key system variables are given in Figures 15.6.1-1 to 15.6.1-7.

15.6.1.3 Conclusions

The results of the analysis demonstrate that the event acceptance criteria are met. The predicted MDNBR is greater than the safety limit. The critical heat flux correlation limit ensures that, with 95% probability and 95% confidence, DNB is not expected to occur; therefore, no fuel is expected to fail. This event does not pose a credible challenge to FCM safety criteria.

Both the maximum reactor primary system pressure and the maximum secondary system pressures are less than 110% of design pressure, 2750 psia and 1320 psia, respectively.

The analysis for this event supports nominal primary T_{avg} operation at full power between 588.8 and 580.8°F, inclusive.

15.6.2 BREAK IN INSTRUMENT LINE OR OTHER LINE FROM REACTOR COOLANT PRESSURE BOUNDARY THAT PENETRATE CONTAINMENT

15.6.2.1 Identification of Causes and Frequency Classification

The estimated frequency of a primary sample or instrument line rupture classifies it as a limiting fault incident. A primary sample or instrument line break provides a release path for reactor coolant outside Containment. The line break selected for analysis is the letdown line which penetrates the Containment. This is the largest penetration whose failure could result in an event in this category. This failure would result in larger releases than would be the case for the smaller instrument and sample lines.

This event is classified as an ANS Condition II event, a fault of moderate frequency. See Section 15.0.1 for a discussion of ANS Condition II events.

Plant systems and equipment which are necessary to mitigate the effects of the event are discussed in Section 15.0.8 and listed in Table 15.0.8-1. No single active failure in any of these systems or equipment will adversely affect the consequences of the event.

15.6.2.2 Sequence of Events and Systems Operation

The integrity of lines containing primary coolant external to the Containment is significant radiologically since a rupture of this barrier results in the release of reactor coolant outside Containment. Following such a break, the RCS pressure decreases due to the loss of reactor coolant. When the pressurizer pressure has reached the low pressure setpoint, a reactor trip is initiated. A turbine trip follows a reactor trip and results in an increase in secondary side pressure to the steam generator safety valve set pressure. The safety injection signal (SIS) on low pressurizer pressure terminates the break flow by isolating the letdown line inside Containment. The reactor coolant inventory is replenished by the charging pumps. Operation of these pumps ensures that the core will not be uncovered and prevents any significant increase in clad temperatures.

After 30 minutes, the operator is assumed to start a plant cooldown. At about 30 minutes into the transient, the ruptured line is isolated, terminating the leak flow. Prior to isolation of the line, reactor coolant has been released from the RCS to the Reactor Auxiliary Building at the rate of 200 gpm.

15.6.2.3 Analysis of Radiological Consequences

15.6.2.3.1 Design basis

15.6.2.3.1.1 Physical model

A break in fluid-bearing lines which penetrate the Containment could result in the release of radioactivity to the environment. There are no instrument lines connected to the RCS which penetrate the Containment. There are, however, other piping lines from the RCS to the Chemical and Volume Control System (CVCS) and the Process Sampling System which penetrate the Containment.

The most severe rupture with respect to radioactivity release during normal plant operation is the rupture of the letdown line outside Containment. For such a break, the reactor coolant letdown flow would have passed from the cold leg and through the regenerative heat exchanger.

15.6.2.3.1.2 Assumptions and Parameters

The analysis of the small line break outside of containment (SLBOC) radiological consequences uses the analytical methods and assumptions outlined in SRP 15.6.2 since this accident is not discussed in RG 1.183. The major assumptions and parameters used in the analysis are listed in Table 15.6.2-1 and discussed below:

a) The SRP indicates that accident-initiated iodine spiking be modeled. The accident-initiated iodine spike increases the iodine release rate from the fuel to the RCS to a value 500 times greater than the release rate corresponding to a maximum equilibrium RCS concentration of 1.0 μCi/gm of DE I-131. The iodine spike appearance rates are given in Table 15.0.9-6. The duration of the accident-initiated iodine spike is limited by the amount of activity available in the fuel-clad gap. Based on having 12 percent of the iodine in the fuel-clad gap, the gap inventory would be depleted within 5.0 hours and the spike is terminated at that time.

b) The noble gas activity concentration in the RCS at the time the accident occurs is based on a one-percent fuel defect level. This is approximately equal to the Technical Specification value of 100/E bar μCi/gm for gross radioactivity. The noble gas concentrations in the RCS are given in Table 15.0.9-2.

c) The transfer of the primary coolant to the environment through the letdown line break is 200 gpm until 30 minutes. The iodine flashing factor for the released letdown flow is 0.4. Therefore of the iodine contained in the water released in the letdown line break, only 40% of the iodine is released to the auxiliary building atmosphere and of that all is released to the environment.

15.6.2.3.1.3 Offsite Doses

The offsite doses are calculated using the assumptions and equations in Section 15.0A.1.

15.6.2.3.1.4 Control Room Doses

The control room assumptions are provided in Section 15.6.5.4.3 and Table 15.6.5-15. The SLBOC control room doses modeled 500 cfm unfiltered inleakage.

It is assumed that the control room HVAC system begins in normal mode. The activity level in the intake duct causes a high radiation signal almost immediately. It is conservatively assumed that the post-accident recirculation control room HVAC mode is entered 15 seconds after event initiation. The control room is assumed to be placed in pressurization mode at 2 hours after isolation signal.

15.6.2.3.1.5 Results

The potential radiological consequences resulting from a letdown line break outside of containment have been conservatively analyzed, using assumptions and models described in previous sections. The TEDE doses have been analyzed for the worst two hours at the EAB and for the duration of the event at the LPZ and in the control room. The analytically predicted dose consequence to the Control Room (CR) operators due to LBLOCA increased by a small amount, based on the reduction in CR recirculation flow which reduces the iodine filtration provided by the charcoal filters relative to the analysis of record condition. The analytically predicted dose consequence to the control room operators for the failure of a letdown line break outside containment event was not similarly revised or updated since the LBLOCA DBA is the limiting control room operator dose event. The radiological analysis results for this event are listed in Table 15.6.2-2. The resultant doses are within the guideline values. The calculated offsite doses are less than 10% of the 10CFR50.67 limits (i.e., 2.5 rem TEDE) and the calculated control room doses are less than the 10CFR50.67 limit of 5 rem TEDE.

15.6.3 STEAM GENERATOR TUBE RUPTURE

A steam generator tube rupture (SGTR) results in the leakage of contaminated reactor coolant into the secondary system and the subsequent release of a portion of that activity to the atmosphere. Therefore, the analysis must demonstrate that the offsite radiological consequences resulting from a SGTR are within allowable guidelines. One of the primary assumptions of the offsite dose assessment methodology is that the steam generator with the leaking tube can eventually be isolated from the release to the atmosphere. Therefore, it is essential to demonstrate that the secondary side of the ruptured Steam Generator does not overfill from water entering the secondary side of the steam generator from primary coolant flow through the broken tube and AFW flow delivered to the Steam Generator. Such an overfill condition in the Steam Generator could force water discharge through the Main Steam PORV's or Safeties potentially causing damage that could prevent their subsequent closure to isolate the release path.

The SGTR analysis then becomes a two-step process. The first part of the analysis is to ensure that overfill of the steam generator does not occur. Having demonstrated that Margin to Overfill exists, the Offsite Dose Assessment can be performed using the standard assumptions and methodology. Since the Margin to Overfill case and the Offsite Dose analysis have different sensitivities to plant parameters, two separate analyses must be performed to ensure that conservative scenarios are analyzed for each case.

Carolina Power & Light Company (CP&L) participated as a member of the Westinghouse Owners' Group (WOG) Steam Generator Tube Rupture (SGTR) Subgroup, which was responsible for the production of WCAP-10698 P-A, "SGTR Analysis Methodology to Determine the Margin to Steam Generator Overfill," (December 1984) and Supplement 1 to WCAP-10698-P-A, "Evaluation of Off Site Radiation Doses for a Steam Generator Tube Rupture Accident," (May 1985). Carolina Power & Light Company has used the methodology documents in these reports as a basis for the plant specific analysis of an SGTR. The NRC staff provided their assessment of this methodology and approval for referencing in plant specific submittals in a March 30, 1987 letter from Mr. C. E. Rossi to Mr. A. E. Ladeiu, Chairman, SGTR Subgroup.

The original design basis steam generator tube rupture (SGTR) analysis of record for the Shearon Harris Nuclear Power plant (SHNPP) is presented in WCAP-12403 (Reference 15.6.3-2). The SHNPP SGTR analysis in WCAP-12403 included an analysis of the margin to steam generator overfill, as well as an analysis of the offsite radiation doses for a design basis SGTR. The analysis results demonstrated that there was margin to steam generator overfill with the most limiting single failure with respect to overfill, and that the calculated offsite radiation doses would be acceptable assuming the most limiting single failure for the offsite dose evaluation.

Plant response to the event was modeled using the LOFTTR2 computer code with conservative assumptions of break size and location, condenser availability and initial secondary water mass in the ruptured steam generator. The analysis methodology includes the simulation of the operator actions for recovery from a steam generator tube rupture based on the SHNPP Emergency Operating Procedures (EOPs), which were developed from the Westinghouse Owners Group Emergency Response Guidelines (ERGs). The operator action times used in the simulation of the SGTR recovery actions for the SHNPP Analysis were based on the results from plant simulator studies.

The SGTR Event was subsequently reanalyzed to account for the recognition that the Main Steam PORV capacity was greater than assumed on the original analysis of record (WCAP-12403). The reanalysis was used as an opportunity to incorporate a few other existing conditions and to investigate the potential for possible future modifications. This reanalysis uses identical methodologies and operator action times as the original analysis of WCAP-12403 but it considers revised or expanded ranges for the AFW System performance and the SI system performance as well as the Main Steam PORV capacity. This revised analysis (WCAP-12403 Supplement 1/Ref. 15.6.3-7) was the analysis of record for SHNPP as amended by Reference 15.6.3-8, prior to installation of the model Δ75 replacement steam generators.

In support of the Shearon Harris Nuclear Power Plant (SHNPP) model Δ75 replacement steam generator 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 evaluation discussed herein considers operation with a full power average temperature (T_{avg}) of 588.8°F and assumes that up to 10% of the steam generator tubes are plugged. The analysis supports a main feedwater temperature window of 375°F to 440°F. Operation at the rated power of 2948 MWt with 0.34% uncertainty (analyzed core power of 2958 MWt) was also considered. As noted in Tables 15.0.3-5 and 15.0.6-2, values used in analysis of SGTR for Reactor Protection System time constants and Auxiliary Feedwater delivery are slightly different for this particular event. This analysis is documented in WCAP-14778 (Reference 15.6.3-9). Reference 15.6.3-9 presents the margin to overfill analysis.

The WCAP-14778 analysis was subsequently amended to address the decay heat issue presented in NSAL-07-11, (Reference 15.6.3-12). In addition to decay heat, safety injection and auxiliary feedwater temperatures were evaluated and single failure issues were addressed. The results of the amended SGTR Margin to Overfill analysis are presented in this FSAR section.

The dose analysis presented in the revised licensing submittal for Alternate Source Term implementation (Reference 15.6.3-13) contains the Regulatory Guide 1.183 results for the SGTR event. The thermal and hydraulic analysis in Reference 15.6.3-9 was used as input to the Regulatory Guide 1.183 dose calculations. Both of these analyses will be presented separately in this FSAR section based on credit for a measurement uncertainty recapture (MUR) power uprate being applied. The MUR uprate increased NSSS power to 2970.4 MWt.

15.6.3.1 Identification of Cause and Accident Description

The accident examined is the complete severance of a single steam generator tube. The accident is assumed to take place at power with the reactor coolant contaminated with fission products corresponding to continuous operation with a limited number of defective fuel rods. The accident leads to an increase in contamination of the secondary system due to leakage of radioactive coolant from the RCS. In the event of a coincident loss of offsite power, or failure of the Condenser Steam Dump System, discharge of radioactivity to the atmosphere takes place via the steam generator power operated relief valves (and safety valves if their setpoint is reached).

In view of the fact that the steam generator tube material is Inconel 690 and is a highly ductile material, it is considered that the assumption of a complete severance is somewhat conservative. The more probable mode of tube failure would be one or more minor leaks of undetermined origin. Activity in the steam and power conversion system is subject to continual surveillance and an accumulation of minor leaks which exceed the limits established in the Technical Specifications is not permitted during the Unit operation.

Due to a series of alarms as described below, the operator will readily determine that a steam generator tube rupture has occurred, identify and isolate the ruptured steam generator, and complete the required recovery actions to stabilize the plant and terminate the primary to secondary break flow. The recovery procedure can be completed on a time scale which ensures that break flow to the secondary system is terminated before water level in the affected steam generator rises into the main steam pipe. Sufficient indications and controls are provided to enable the operator to carry out these functions satisfactorily.

Assuming normal operation of the various plant control systems, the following sequence of events is initiated by a tube rupture:

1) Pressurizer low pressure and low level alarms are actuated and charging pump flow increases in an attempt to maintain pressurizer level. On the secondary side there is a steam flow/feedwater flow mismatch before trip as feedwater flow to the affected steam generator is reduced due to the break flow which is now being supplied to that steam generator from the primary side.

2) The condenser vacuum pump effluent radiation monitor, steam generator blowdown line radiation monitor, and main steamline radiation monitor will alarm, indicating a sharp increase in radioactivity in the secondary system.

3) Continued loss of reactor coolant inventory leads to a reactor trip signal generated by low pressurizer pressure or by overtemperature ΔT. Resultant plant cooldown following reactor trip leads to a rapid decrease in RCS pressure and pressurizer level. A safety injection signal, initiated by low pressurizer pressure, follows soon after the reactor trip. The safety injection signal automatically terminates steam generator blowdown, normal feedwater supply and initiates auxiliary feedwater addition via the motor-driven AFW pumps. If the steam generator level decreases below the low-low level setpoint in two of the three steam generators or a loss of offsite power occurs, the turbine-driven pump will also be started.

4) The reactor trip automatically trips the turbine and if offsite power is available the steam dump valves open permitting steam dump to the condenser. In the event of coincident loss of offsite power, the steam dump valves would automatically close to protect the condenser. The steam generator pressure would rapidly increase resulting in steam discharge to the atmosphere through the steam generator power-operated relief valves (and safety valves if their setpoint is reached).

5) Following reactor trip and safety injection actuation, the continued action of auxiliary feedwater supply and borated safety injection flow (supplied from the refueling water storage tank) provide a heat sink which absorbs some of the decay heat. This reduces the amount of steam bypass to the condenser, or in the case of loss of offsite power, steam relief to atmosphere.

6) Safety injection flow results in stabilization of RCS pressure and pressurizer water level, and the RCS pressure trends towards an equilibrium value where the safety injection flow rate equals the break flow rate.

In the event of an SGTR, the plant operators must diagnose the SGTR and perform the required recovery actions to stabilize the plant and terminate the primary to secondary leakage. The operator actions for SGTR recovery are provided in the plant Emergency Operating Procedures. The major operator actions 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 operators' actions are described below.

1) Identify the ruptured steam generator.

High secondary side activity, as indicated by the condenser vacuum pump effluent radiation monitor, steam generator blowdown line radiation monitor, or main steamline 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 steamline radiation monitor. For an SGTR that results in a reactor trip at high power, the steam generator water level will decrease to near the bottom of the narrow range scale for all of the steam generators. The AFW flow will begin to refill the steam generators, distributing flow to each of the steam generators. Since primary to secondary leakage adds additional liquid inventory to the ruptured steam generator, the water level will increase more rapidly in that 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 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 overfilling the ruptured steam generator with water 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.

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 only the intact steam generators. This ensures adequate subcooling will exist in the RCS after depressurization of the RCS 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 power operated relief valves (PORVs) on the intact steam generators.

4) Depressurize the RCS to terminate primary to secondary leakage.

When the cooldown is completed, SI flow will increase 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 RCS and ruptured steam generator pressures equalize, an "excess" amount of inventory is needed to ensure pressurizer level remains on span. The "excess" amount required depends on RCS pressure and reduces to zero when RCS pressure equals the pressure in the ruptured steam generator.

The RCS depressurization is performed using normal pressurizer spray if the reactor coolant pumps (RCPs) are running. However, if offsite power is lost or the RCPs are not running for some other reason, normal pressurizer spray is not available. In this event, RCS depressurization can be performed using the pressurizer PORVs or auxiliary pressurizer spray.

5) Terminate SI to stop primary to secondary leakage.

The previous actions will have established adequate RCS subcooling, secondary side heat sink and sufficient reactor coolant inventory to ensure that SI flow is no longer needed. When these actions have been completed, SI flow must be stopped to terminate primary to secondary leakage. Primary to secondary leakage will continue after SI flow is stopped until RCS and ruptured steam generator pressures equalize. Charging flow, letdown, and pressurizer heaters will then be controlled to prevent repressurization of the RCS and reinitiation of leakage into the ruptured steam generator.

Following SI termination, the plant conditions will be stabilized, the primary to secondary break flow will be terminated, and all immediate safety concerns will have been addressed. At this time, a series of operator actions are performed to prepare the plant for cooldown to cold shutdown conditions. Subsequently, actions are performed to cool down and depressurize the RCS to cold shutdown conditions and to depressurize the ruptured steam generator.

This event is classified as an ANS Condition IV event, a limiting fault (postulated accident). See Section 15.0.1 for a discussion of ANS Condition IV events.

Plant systems and equipment which are necessary to mitigate the effects of the event are discussed in Section 15.0.8 and listed in Table 15.0.8-1. No single active failure in any of these systems or equipment will adversely affect the consequences of the event. The analyses presented here consider the limiting single failure.

15.6.3.2 Westinghouse Thermal Hydraulic Analysis

Steam Generator Overfill. One of the primary assumptions of the offsite dose assessment methodology is that the steam generator with the leaking tube can eventually be isolated from relieving to the atmosphere. Therefore, it is essential to demonstrate that the secondary side of the ruptured steam generator does not overfill and potentially damage the Main Steam PORV's or Safeties by forcing water discharge through them. Therefore, the first part of the analysis is to ensure that overfill of the steam generator does not occur. Having demonstrated that margin to overfill exists, the Offsite Dose Assessment can be performed using the standard assumptions and methodology. Since the Margin to Overfill case and the Offsite Dose analysis have different sensitivities to plant parameters, two separate analyses must be performed to ensure that conservative scenarios are analyzed for each case. The Overfill Analysis is presented in this section.

15.6.3.2.1 Analysis assumptions

A detailed discussion of the Design Basis Accident, along with the SHNPP-specific inputs and Conservative Assumptions, is provided in the base analysis presented in WCAP-14778. The analysis is further revised to address the decay heat issue presented in NSAL-07-11 (Reference 15.6.3-12) and to address potential single failures. The following discussion summarizes those assumptions.

The accident modeled is a double ended break of one steam generator tube located at the top of the tube sheet on the outlet (cold leg) side of the steam generator. It was also assumed that loss of offsite power occurs at the time of reactor trip. The limiting single failure with respect to steam generator overfill used for the analysis is a failure within the turbine-driven AFW Pump speed controller, resulting in steady-state pump operation at the upper end of the speed control band (4100 rpm) and increased delivery to the ruptured steam generator.

Most of the conservative assumptions and initial conditions used in this analysis are discussed in detail in WCAP-14778 (Reference 15.6.3-9). These assumptions include simulation of turbine runback based on the calculated reactor trip time, a conservatively high initial steam generator secondary mass, and initiation of AFW flow from both motor-driven AFW pumps and the turbine driven AFW pump immediately after reactor trip.

Key assumptions include:

1) A maximum AFW flowrate of 750 gpm to the ruptured steam generator, until isolation, for the overfill analysis.

2) Steam generator PORV capacity of 795,000 lbm/hr per valve at 1200 psig (both MSPORVs are credited in the overfill analysis).

3) Safety Injection flow from two high head safety injection pumps.

To address NSAL-07-11 (Reference 15.6.3-12) and single failure effects, the following were modeled:

4) Low decay heat based on the 1979 ANS model -2σ uncertainty.

5) The minimum AFW enthalpy of 8.0 Btu/lbm (based on 40°F).

6) A failure within the turbine-driven AFW Pump speed controller, resulting in steady-state pump operation at the upper end of the speed control band (4100 rpm).

The major operator actions for SGTR recovery provided in the SHNPP Plant Operating Manual, Procedure Number EOP-E-3 (ERG E-3) were explicitly modeled in this analysis. The operator actions modeled include (1) identification and isolation of the ruptured steam generator, (2) cooldown of the RCS using the PORVs on the intact steam generators to ensure subcooling, (3) depressurization of the RCS using a pressurizer PORV to restore inventory, and (4) termination of SI to stop primary to secondary leakage. The operator action times used for the analysis are presented in Table 15.6.3-1.

15.6.3.2.2 Transient description

The analysis results for the SHNPP margin to overfill analysis are described below. The sequence of events for this transient is presented in Table 15.6.3-2.

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, pressurizer level decreases as shown in Figure 15.6.3-1. The RCS pressure also decreases as shown in Figure 15.6.3-2 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 at approximately 112 seconds on an overtemperature Δ-T trip signal.

After reactor trip, core power rapidly decreases to decay heat levels. The turbine stop valves close and steam flow to the turbine is terminated. 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 until the steam generator PORVs (and safety valves if their setpoints are reached) lift to dissipate the energy, as shown in Figure 15.6.3-3. After the plant sensible heat is dissipated, the steaming rate decreases to the level required to remove the core decay heat and the secondary pressure is maintained near the PORV setpoint of 1121 psia. The main feedwater flow will be terminated and AFW flow will be automatically initiated following the loss of offsite power assumed at the time of reactor trip.

The RCS pressure decreases more rapidly after reactor trip as energy transfer to the secondary shrinks the reactor coolant and the tube rupture break flow continues to deplete primary inventory. Pressurizer level also decreases more rapidly following reactor trip. The decrease in RCS inventory results in a low pressurizer pressure SI signal. After SI actuation, the SI flow

rate initially exceeds the tube rupture break flow rate and the pressurizer level begins to increase. This also results in an increase in the RCS pressure which trends toward the equilibrium value where the SI flow rate equals the break flow rate.

Since offsite power is assumed lost at reactor trip, the RCPs trip and a gradual transition to natural circulation flow occurs. Immediately following reactor trip the temperature differential across the core decreases as core power decays (see Figure 15.6.3-4); however, the temperature differential subsequently increases as the reactor coolant pumps coast down and natural circulation flow develops. The cold leg temperatures trend toward the steam generator temperature as the fluid residence time in the tube region increases. The hot leg temperature reaches a peak and then slowly decreases, as steady state conditions are reached, until operator actions are initiated to cool down the RCS.

15.6.3.2.3 Major operator actions

1) Identify and Isolate the Ruptured Steam Generator. Once a tube rupture has been identified, recovery actions begin by isolating steam flow from the ruptured steam generator and isolating the auxiliary feedwater flow to the ruptured steam generator. The ruptured steam generator is assumed to be identified and AFW flow to the ruptured steam generator isolated when the narrow range level reaches 30% on the ruptured steam generator or at 8.8 minutes after initiation of the SGTR, whichever is longer. For the Shearon Harris analysis, the time to reach 30% is less than 8.8 minutes; therefore, isolation of the AFW to the ruptured generator occurs at 8.8 minutes. The 8.8-minute time ensures a positive margin to overfill assuming a single failure within the turbine-driven AFW Pump speed controller that results in steady-state pump operation at the upper end of the speed control band (4100 rpm). This time limit ensures sufficient space for cooldown, depressurization, and SI termination following a speed-controller failure, which can also be accomplished if AFW is isolated prior to 80% narrow range level. Complete isolation of steam flow from the ruptured steam generator is verified when the narrow range level reaches 30% on the ruptured steam generator or at 12 minutes after initiation of the SGTR, whichever is longer. For the Shearon Harris analysis the time to reach 30% is less than 12 minutes, and thus the ruptured steam generator is assumed to be completely isolated at 12 minutes.

2) Cool down the RCS to Establish Subcooling Margin. After isolation of the ruptured steam generator, there is a 5 minute operator action time imposed prior to initiating the cooldown. After this time, actions are taken to cool the RCS as rapidly as possible by dumping steam from the intact steam generators. Since offsite power is lost, the RCS is cooled by dumping steam to the atmosphere using the PORVs on the intact steam generators. Both of the intact steam generator PORVs are assumed to be opened at 1020 seconds for RCS cooldown. The cooldown is continued until RCS subcooling at the ruptured steam generator pressure is 20°F plus an allowance for subcooling uncertainty. When these conditions are satisfied at 1356 seconds, it is assumed that the operator closes the intact steam generator PORVs to terminate the cooldown. This cooldown ensures that there will be adequate subcooling in the RCS after the subsequent depressurization of the RCS to the ruptured steam generator pressure. The reduction in the intact steam generator pressure required to accomplish the cooldown is shown in Figure 15.6.3-3, and the effect of the cooldown on the RCS temperatures is shown in Figure 15.6.3-4. The pressurizer level and RCS pressure also decrease during this cooldown process due to shrinkage of the reactor coolant as shown in Figures 15.6.3-1 and 15.6.3-2.

3) Depressurize RCS to Restore Inventory. After the RCS cooldown, a 4 minute 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 a pressurizer PORV. The RCS depressurization is initiated at 1598 seconds and continued until any of the following conditions are satisfied: RCS pressure is less than the ruptured steam generator pressure and pressurizer level is greater than 10%, or pressurizer level is greater than 75%, or RCS subcooling is less than 20°F. For this case, the RCS depressurization is terminated because the RCS pressure is reduced to less than the ruptured steam generator pressure and the pressurizer level is above 10%. The RCS depressurization reduces the break flow as shown in Figure 15.6.3-5, and increases SI flow to refill the pressurizer as shown in Figure 15.6.3-1.

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 beencompleted, the SI flow must be stopped to prevent repressurization of the RCS and to terminate primary to secondary leakage. The SI flow is terminated at this time if RCS subcooling is greater than 20°F, 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 greater than 10%. For the SHNPP analysis, SI was not terminated until the RCS pressure increased to 50 psi above the ruptured steam generator pressure to assure that RCS pressure is increasing.

   After depressurization is completed, an operator action time of 3 minutes was assumed prior to initiation of SI termination. Since the above requirements are satisfied, SI termination actions were performed at this time by closing off the SI flow path. After SI termination, the RCS pressure begins to decrease as shown in Figure 15.6.3-2. The intact steam generator PORVs are also opened to dump steam to maintain the prescribed RCS temperature to ensure that subcooling is maintained. When the PORVs are opened, the increased energy transfer from primary to secondary also aids in the depressurization of the RCS to the ruptured steam generator pressure. The primary to secondary leakage continues after the SI flow is terminated until the RCS and ruptured steam generator pressures equalize.

   The primary to secondary break flow rate throughout the recovery operations is presented in Figure 15.6.3-5, and the water volume in the ruptured steam generator is presented as a function of time in Figure 15.6.3-6. It is noted that the water volume in the ruptured steam generator when the break flow is terminated is approximately 66 cubic feet less than the total steam generator volume of 5545 ft^3. Therefore, it is concluded that overfill of the ruptured steam generator will not occur for a design basis SGTR for Shearon Harris.

15.6.3.3 Westinghouse Thermal Hydraulic Analysis - Offsite Dose Case

An analysis was also performed to determine the offsite radiological consequences, assuming the limiting single failure relative to offsite doses without steam generator overfill. Since steam generator overfill does not occur, the results of this analysis represent the limiting consequences for an SGTR for SHNPP, and the results of the offsite radiological consequences analysis are discussed below.

This analysis was performed to determine the plant response for a design basis SGTR and to determine the integrated primary to secondary break flow and the mass releases from the ruptured and intact steam generators to the condenser and to the atmosphere. This information was then used to calculate the quantity of radioactivity released to the environment and the resulting radiological consequences.

The plant response following an SGTR was analyzed with the LOFTTR2 program until the primary to secondary break flow is terminated. The reactor protection system and the automatic actuation of the engineered safeguards systems were modeled in the analysis. The major operator actions which are required to terminate the break flow for an SGTR were also simulated in the analysis.

15.6.3.3.1 Analysis assumptions

The accident modeled is a double-ended break of one steam generator tube located at the top of the tube sheet on the outlet (cold leg) side of the steam generator. It was assumed that the reactor is operating at full power at the time of the accident and the initial secondary mass was assumed to correspond to operation at the nominal steam generator mass minus an allowance for uncertainties. An initial minimum AFW flow of 390 gpm per steam generator was assumed for the offsite dose analysis of record. However, the analysis also finds that a reduction in AFW flow due to a 3% MSSV setpoint tolerance (i.e. an increase in SG pressure) will have no impact on calculated doses. 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.

The limiting single failure was assumed to be the failure of the PORV on the ruptured steam generator. Failure of this PORV in the open position will cause an uncontrolled depressurization of the ruptured steam generator which will increase primary to secondary leakage and the mass release to the atmosphere. It was assumed that the ruptured steam generator PORV fails open when the ruptured steam generator is isolated and that the PORV was subsequently isolated by locally closing the associated block valve.

The major operator actions required for the recovery from an SGTR are discussed in Section 15.6.3.1 and these operator actions were simulated in the analysis. The operator action times which were used for the analysis are presented in Table 15.6.3-3. It is noted that the PORV on the ruptured steam generator was assumed to fail open at the time the ruptured steam generator was isolated. Before proceeding with the recovery operations, the failed open PORV on the ruptured steam generator was assumed to be isolated by locally closing the associated block valve. It was assumed that the ruptured steam generator PORV is isolated at 20 minutes after the valve was assumed to fail open. After the ruptured steam generator PORV was isolated, the additional delay time of approximately five minutes (Table 15.6.3-3) was assumed for the operator action time to initiate the RCS cooldown.

15.6.3.3.2 Transient description

The LOFTTR2 analysis results are described below. The sequence of events for this transient is presented in Table 15.6.3-4.

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, pressurizer level decreases as shown in Figure 15.6.3-7. The RCS pressure also decreases as shown in Figure 15.6.3-8 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 an overtemperature ΔT trip signal.

After reactor trip, core power rapidly decreases to decay heat levels. The turbine stop valves close and steam flow to the turbine is terminated.

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 until the steam generator PORVs (and safety valves if their setpoints are reached) lift to dissipate the energy, as shown in Figure 15.6.3-9. The main feedwater flow will be terminated and AFW flow will be automatically initiated on loss of offsite power assumed at the time of reactor trip.

The RCS pressure decreases more rapidly after reactor trip as energy transfer to the secondary shrinks the reactor coolant and the leak flow continues to deplete primary inventory. Pressurizer level also decreases more rapidly following reactor trip. The decrease in RCS inventory results in a low pressurizer pressure SI signal. After SI actuation, the RCS pressure and pressurizer level tend to stabilize until the ruptured steam generator PORV is assumed to fail open.

Since offsite power is assumed lost at reactor trip, the RCPs trip and a gradual transition to natural circulation flow occurs. Immediately following reactor trip, the temperature differential across the core decreases as core power decays (see Figures 15.6.3-10 and 15.6.3-11); however, the temperature differential subsequently increases as natural circulation flow develops. The cold leg temperatures trend toward the steam generator temperature as the fluid residence time in the tube region increases. The intact steam generator loop temperatures continue to slowly decrease due to the continued AFW flow until operator actions are taken to control the AFW flow to maintain the specified level in the intact steam generators. The ruptured steam generator loop temperatures also continue to slowly decrease until the ruptured steam generator was isolated and the PORV was assumed to fail open.

15.6.3.3.3 Major operator actions

1) Identify and Isolate the Ruptured Steam Generator. The ruptured steam generator was assumed to be identified and AFW flow to the ruptured steam generator isolated at 10 minutes after the initiation of the SGTR or when the narrow range level reaches 30%, whichever time is greater. Since the time to reach 30% narrow range level is less than 10 minutes, it was assumed that the ruptured steam generator is isolated at 10 minutes. Complete isolation of steam flow from the ruptured steam generator is verified when the narrow range level reaches 30% on the ruptured steam generator or at 12 minutes after initiation of the SGTR, whichever is longer. For the Shearon Harris analysis the time to reach 30% is less than 12 minutes, and thus the ruptured steam generator is assumed to be completely isolated at 12 minutes. The ruptured steam generator PORV was also assumed to fail open at this time and the failure was simulated at 722 seconds. The failure causes the ruptured steam generator to rapidly depressurize as shown in Figure 15.6.3-9, which

results in an increase in primary to secondary leakage The depressurization of the ruptured steam generator increases the break flow and energy transfer from primary to secondary which results in a decrease in the ruptured loop temperatures as shown in Figure 15.6.3-11. As noted previously, the intact steam generator loop temperatures also decrease, as shown in Figure 15.6.3-10, until the AFW flow to the intact steam generators is throttled. After this time, the heat transfer to the intact steam generators decreases and the temperature differential across the intact steam generators decreases. As the intact steam generator hot leg temperatures decrease below the steam generator water temperature, reverse heat transfer takes place for a short time period as shown in Figure 15.6.3-10. It was assumed that the time required for the operator to identify that the ruptured steam generator PORV is open and to locally close the associated block valve is 20 minutes. Thus, at 1922 seconds the depressurization of ruptured steam generator was terminated.

2) Cool Down the RCS to Establish Subcooling Margin. After the ruptured steam generator PORV block valve was closed, there is an approximately five minute operator action time imposed prior to initiation of cooldown. The depressurization of the ruptured steam generator affects the RCS cooldown target temperature since the temperature is dependent upon the pressure in the ruptured steam generator. Since offsite power was lost, the RCS was cooled by dumping steam to the atmosphere using the intact steam generator PORVs. The cooldown was continued until RCS subcooling at the ruptured steam generator pressure is 20°F plus an allowance for instrument uncertainty. Because of the lower pressure in the ruptured steam generator, the associated temperature the RCS must be cooled to is also lower, which has the net effect of extending the time for cooldown. The cooldown was initiated at 2224 seconds and was completed at 2996 seconds.

   The reduction in the intact steam generator pressures required to accomplish the cooldown is shown in Figure 15.6.3-9 and the effect of the cooldown on the RCS temperature is shown in Figure 15.6.3-10. The pressurizer level and RCS pressure also decrease during this cooldown process due to shrinkage of the reactor coolant as shown in Figures 15.6.3-7 and 15.6.3-8.

3) Depressurizes RCS to Restore Inventory. After the RCS cooldown, a 4 minute operator action time is included prior to the RCS depressurization. The RCS is depressurized 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 a safety grade pressurizer PORV. The depressurization is initiated at 3236 seconds and continued until the criteria in the Emergency Operating Procedures are satisfied. The RCS depressurization reduces the break flow as shown in Figure 15.6.3-13, and increases SI flow to refill the pressurizer as shown in Figure 15.6.3-7.

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 repressurization of the RCS and to terminate primary to secondary leakage. The SI flow is terminated at this time if the SI termination criteria in the Emergency Operating Procedures are satisfied.

   After depressurization was complete, an operator action time of 3 minutes was assumed prior to initiation of SI termination. Since the SI termination requirements are satisfied, SI termination actions were performed at this time by closing off the SI flow path. After SI

termination, the RCS pressure begins to decrease as shown in Figure 15.6.3-8. The intact steam generator PORVs are also opened to dump steam to maintain the prescribed RCS temperature to ensure that subcooling is maintained. When the PORVs are opened, the increased energy transfer from primary to secondary also aids in the depressurization of the RCS to the ruptured steam generator pressure. The differential pressure between the RCS and the ruptured steam generator is shown in Figure 15.6.3-12. Figure 15.6.3-13 shows that the primary to secondary leakage continues after the SI flow is stopped until the RCS and ruptured steam generator pressures equalize.

The ruptured steam generator water volume is shown in Figure 15.6.3-14. The water volume in the ruptured steam generator when the break flow is terminated is less than the volume for the margin to overfill case in Reference 15.6.3-14 and is significantly less than the total steam generator volume of 5545 ft^3 for the radiological case. The mass of water in the ruptured steam generator is also shown as a function of time in Figure 15.6.3-15.

15.6.3.3.4 Mass releases

The mass releases were determined for use in evaluating the exclusion area boundary and low population zone radiation exposure. The steam releases from the ruptured and intact steam generators, the feedwater flows to the ruptured and intact steam generators, and primary to secondary break flow into the ruptured steam generator were determined for the period from accident initiation until two hours after the accident and from two to eight hours after the accident. The releases for 0-2 hours were used to calculate the radiation doses at the exclusion area boundary for a two-hour exposure, and the releases for 0-8 hours were used to calculate the radiation doses at the low population zone for the duration of the accident.

The operator actions for the SGTR recovery up to the termination of primary to secondary leakage were simulated in the LOFTTR2 analysis. Thus, the steam releases from the ruptured and intact steam generators, the feedwater flows to the ruptured and intact steam generators, and the primary to secondary leakage into the ruptured steam generator were determined from the LOFTTR2 results for the period from the initiation of the accident until the leakage is terminated. Following the termination of leakage, it was assumed that the actions are taken to cool down the plant to cold shutdown conditions. The PORVs for the intact steam generators were assumed to be used to cool down the RCS to the RHR system operating temperature of 325°F at the maximum allowable cooldown rate of 100°F/hr. The steam releases and the feedwater flows for the intact steam generators for the period from leakage termination until two hours were determined from a mass and energy balance using the calculated RCS and intact steam generator conditions at the time of leakage termination and at two hours. The RCS cooldown was assumed to be continued after two hours until the RHR system in-service temperature of 325°F is reached. Depressurization of the ruptured steam generator was then assumed to be performed to the RHR in-service pressure of 365 psia via steam release from the ruptured steam generator PORV. The RCS pressure was also assumed to be reduced concurrently as the ruptured steam generator is depressurized. It was assumed that the continuation of the RCS cooldown and depressurization to RHR operating conditions are completed within eight hours after the accident since there is ample time to complete the operations during this time period. The steam releases and feedwater flows from two to eight hours were determined for the intact and ruptured steam generators from a mass and energy balance using the conditions at two hours and at the RHR system in-service conditions.

After eight hours, it was assumed that further plant cooldown to cold shutdown as well as long-term cooling is provided by the RHR system. Therefore, the steam releases to the atmosphere were terminated after RHR in-service conditions were assumed to be reached at eight hours.

For the time period from initiation of the accident until leakage termination, the releases were determined from the LOFTTR2 results for the time prior to reactor trip and following reactor trip. Since the condenser is in service until reactor trip, any radioactivity released to the atmosphere prior to reactor trip would be through the condenser vacuum pump exhaust. After reactor trip, the releases to the atmosphere were assumed to be via the steam generator PORVs. The mass release rates to the atmosphere from the LOFTTR2 analysis are presented in Figures 15.6.3-16 and 15.6.3-17 for the ruptured and intact steam generators, respectively, for the time period until leakage termination. The mass releases calculated from the time of leakage termination until two hours and from 2-8 hours were also assumed to be released to the atmosphere via the steam generator PORVs. The mass releases for the SGTR event for the 0-2 hour and 2-8 hour time intervals considered are presented in Table 15.6.3-5.

15.6.3.4 Westinghouse Offsite Radiation Dose Analysis

The evaluation of the radiological consequences of a steam generator tube rupture (SGTR) assumes that the reactor has been operating at the Technical Specification limit for primary coolant activity and primary to secondary leakage for sufficient time to establish equilibrium concentrations of radionuclides in the reactor coolant and in the secondary coolant. Radionuclides from the primary coolant enter the steam generator via the ruptured tube and primary to secondary leakage and are released to the atmosphere through the steam generator safety or power operated relief valves (PORVs) and via the condenser air ejector exhaust.

The quantity of radioactivity released to the environment due to an SGTR depends upon primary and secondary coolant activity, iodine spiking effects, primary to secondary break flow, break flow flashing fractions, attenuation of iodine carried by the flashed portion of the break flow, partitioning of iodine between the liquid and steam phases, the mass of fluid released from the generator, and liquid-vapor partitioning in the turbine condenser hot well. All of these parameters were conservatively evaluated for a design basis double ended rupture of a single tube.

15.6.3.4.1 Design basis analytical assumptions

The major assumptions and parameters used in the analysis are itemized in Table 15.6.3-6.

15.6.3.4.2 Source term calculations

The radionuclide concentrations in the SHNPP primary and secondary system prior to and following the SGTR were determined as follows:

1) The iodine concentrations in the reactor coolant will be based upon pre-accident and accident-initiated iodine spikes.

    a. Accident-Initiated Spike - The initial primary coolant iodine concentration is 1.0 μCi/gm of Dose Equivalent (D.E.) I-131. Following the primary system depressurization associated with the SGTR, an iodine spike is initiated in the primary system which increases the iodine release rate from the fuel to the coolant to a value 335 times greater than the

release rate corresponding to the initial primary system iodine concentration. The iodine spike appearance rates are given in Table 15.0.9-6.

b. Pre-accident Spike - A reactor transient has occurred prior to the SGTR and has raised the primary coolant iodine concentration from 1 to 60 µCi/gram of D.E. I-131. The pre-accident iodine concentrations are given in Table 15.0.9-7.

2) The initial secondary coolant iodine concentration is 0.1 µCi/gram of D.E. I-131. The initial secondary coolant iodine concentrations are given in Table 15.0.9-7.

3) The chemical form of iodine in the primary and secondary coolant is assumed to be 97% elemental and 3% organic.

4) The initial noble gas concentration in the reactor coolant is based upon 1% fuel defects. The noble gas concentrations in the RCS are given in Table 15.0.9-2.

15.6.3.4.3 Dose calculations

The iodine transport model utilized in this analysis considers break flow flashing, steaming, and partitioning. The model assumes that a fraction of the iodine carried by the break flow becomes airborne immediately due to flashing and atomization. The fraction of primary coolant iodine which is not assumed to become airborne immediately mixes with the secondary water and is assumed to become airborne at a rate proportional to the steaming rate and the iodine partition coefficient. This analysis conservatively assumes an iodine partition coefficient of 100 between the steam generator liquid and steam phases. Droplet removal by the dryers is conservatively assumed to be negligible. The iodine transport model is illustrated in Figure 15.6.3-18. The radiological consequences analysis did not consider steam generator tube uncovery in the calculation, since the steam generator tube uncovery issue was investigated and closed by the Westinghouse Owners Group (WOG). Reference 15.6.3-10 documents the Westinghouse position on the issue, that the effects of tube uncovery on the limiting SGTR transient is essentially negligible and need not be considered in the analysis. Reference 15.6.3-11 documents the NRC agreement on the issue.

The following assumptions and parameters were used to calculate the activity released to the atmosphere and the offsite doses following an SGTR.

1) The mass of reactor coolant discharged into the secondary system through the rupture and the mass of steam released from the ruptured and intact steam generators to the atmosphere are presented in Table 15.6.3-5

2) The time dependent fraction of rupture flow that flashes to steam and is immediately released to the environment is presented in Figure 15.6.3-19. The break flow flashing fraction was conservatively calculated assuming that 100 percent of the break flow comes from the hot leg side of the steam generator, whereas the break flow actually comes from both the hot leg and cold leg sides of the steam generator.

3) In addition to the releases calculated in the thermal hydraulic analysis, steam released from the ruptured steam generator to the turbine driven auxiliary feedwater (TDAFW) pump is considered in the dose analysis. A flow of 41,310 lbm/hr is considered from the time of auxiliary feedwater initiation until the ruptured steam generator is isolated. All of the iodine

contained in this steam, determined from the steam generator activity and the water/steam partition coefficient of 100, is assumed to be released directly to the atmosphere.

4) The total primary to secondary leak rate is assumed to be 1.0 gpm. The leak rate is assumed to be 0.35 gpm for each of the intact steam generators and 0.30 gpm for the ruptured steam generator. The leakage to the intact steam generators is assumed to persist for the duration of the accident.

5) The iodine partition coefficient between the liquid and steam of the ruptured and intact steam generators is assumed to be 100.

6) No credit was taken for radioactivity decay during release and transport or for cloud depletion by ground deposition during transport to the control room, exclusion area boundary (EAB) or outer boundary of the low population zone (LPZ).

7) Short term atmospheric dispersion factors (x/Qs) for accident analysis and breathing rates are provided in Table 15.6.3-10. The breathing rates were obtained from NRC Regulatory Guide 1.183 (Reference 15.6.3-4).

15.6.3.4.4 Offsite Dose Calculation Model

No credit is taken for cloud depletion by ground deposition or by radioactive decay during transport to the exclusion area boundary or to the outer boundary of the low population zone.

Offsite inhalation doses (CEDE) are calculated using the following equation.

$$D_{CEDE} = \sum_i \left[DCF_i \left(\sum_j (\mathrm{IAR})_{ij} (\mathrm{BR})_j \, (\chi/\mathrm{Q})_j\right)\right]$$

where:

D_{CEDE} = CEDE dose via inhalation (rem).

DCF_i = CEDE dose conversion factor via inhalation for isotope i (rem/Ci)(Table 15.0.9-3)

$(IAR)_{ij}$ = integrated activity of isotope i released during the time interval j (Ci)

$(BR)_j$ = breathing rate during time interval j (m^3/sec)(Table 15.6.3-10)

$(\chi/Q)_j$ = atmospheric dispersion factor during time interval j (sec/m^3)(Table 15.6.3-10)

Offsite external exposure (EDE) doses are calculated using the following equation:

$$D_{EDE} = \sum_i \left[DCF_i \left(\sum_j (\mathrm{IAR})_{ij} \, (\chi/\mathrm{Q})_j\right)\right]$$

where:

D_{EDE} = external exposure dose via cloud immersion (rem)

DCF_i = EDE dose conversion factor via external exposure for isotope I (rem*m^3/Ci*sec)(Table 15.0.9.4)

$(IAR)_{ij}$ = integrated activity of isotope i released during the time interval j (Ci)

$(\chi/Q)_j$ = atmospheric dispersion factor during time interval j (sec/m^3)(Table 15.6.3-10)

15.6.3.4.5 Control Room Dose Calculation Models

Control room inhalation doses are calculated using the following equation:

$$D_{CEDE} = \sum_i \left[DCF_i \left(\sum_j \mathrm{Conc}_{ij} \, (\mathrm{BR})_j \right) \right]$$

where:

D_{CEDE} = CEDE dose via inhalation (rem)

DCF_i = CEDE dose conversion factor via inhalation for isotope i (rem/Ci)(Table 15.0.9-3)

Conc_{ij} = concentration in the control room of isotope i, during time interval j, calculated dependent upon inleakage, filtered recirculation and filtered inflow (Ci-sec/m^3)

$(BR)_j$ = breathing rate during time interval j (m^3/sec)(Table 15.6.5-15)

Control room external exposure doses are calculated using the following equation:

$$D_{EDE} = \frac{1}{GF} * \sum_i DCF_i \left(\sum_j \mathrm{Conc}_{ij} \right)$$

where:

D_{EDE} = external exposure dose via cloud immersion in rem.

GF = geometry factor, calculated based on Reference 15.6.5-23, using the equation

$GF = \frac{1173}{V^{0.338}}$ where V is the control room volume ft^3

DCF_I = EDE dose conversion factor via external exposure for isotope i (rem*m^3/Ci*sec)(Table 15.0.9-4)

Conc_{ij} = concentration in the control room of isotope i, during time interval j, calculated dependent upon inleakage, filtered recirculation and filtered inflow (Ci-sec/m^3)

15.6.3.4.6 Control Room Doses

The control room assumptions are provided in Section 15.6.5.4.3 and Table 15.6.5-15. Both the pre-accident and accident initiated iodine spike control room doses modeled 500 cfm unfiltered inleakage.

The control room HVAC begins in normal mode. Once the safety injection actuation setpoint is reached at ~178 seconds and after a delay of 15 seconds the control room HVAC is switched to the post-accident recirculation mode. After 2 hours of operation in post-accident recirculation mode the operator switches the control room HVAC system to the pressurized mode.

15.6.3.4.7 Results

Offsite and control room doses are calculated for the limiting thermal hydraulic analysis presented in Section 15.6.3.3. The doses at the EAB and the LPZ for an SGTR with an assumed pre-accident iodine spike must be within the 10 CFR 50.67 limit of 25 rem TEDE. The doses at the EAB and the LPZ for an SGTR with an assumed accident-initiated iodine spike must be within 10% of the 10CFR50.67 limit (i.e., less than 2.5 rem TEDE). The doses in the control room must be less than the 10CFR50.67 dose limit of 5 rem TEDE for all cases. The analytically predicted dose consequence to the Control Room (CR) operators due to LBLOCA increased by a small amount, based on the reduction in CR recirculation flow which reduces the iodine filtration provided by the charcoal filters relative to the analysis of record condition. The analytically predicted dose consequence to the control room operators for the SGTR event was not similarly revised or updated since the LBLOCA DBA is the limiting control room operator dose event. The radiological analysis results for this event are presented in Table 15.6.3-13 and are within the applicable limits.

15.6.3.5 AREVA Event Disposition

Following SGR/PUR, the SGTR event is supported by the bounding Westinghouse thermal hydraulic and AST radiological analyses. AREVA no longer performs an evaluation or disposition of this event for Harris plant.

15.6.4 SPECTRUM OF BWR STEAM SYSTEM PIPING FAILURES OUTSIDE CONTAINMENT

This section is not applicable to the Shearon Harris Nuclear Power Plant.

15.6.5 LOSS OF COOLANT ACCIDENTS

15.6.5.1 Identification of Causes and Frequency Classification

A loss-of-coolant accident (LOCA) is the result of a pipe rupture of the Reactor Coolant System (RCS) pressure boundary. A major pipe break (large break) is considered a limiting fault, an ANS Condition IV event, in that it is not expected to occur during the lifetime of the plant, but is postulated as a conservative design basis. See Section 15.0.1 for a discussion of Condition IV events.

A minor pipe break (small break) is defined as a rupture of the reactor coolant pressure boundary in which the normally operating charging system flow is not sufficient to sustain pressurizer level and pressure. This is considered a ANS Condition III event in that it is an infrequent fault that may occur during the life of the plant. See Section 15.0.1 for a discussion of Condition III events.

The acceptance criteria for the loss-of-coolant accident is described in 10 CFR 50 Paragraph 46 (Reference 15.6.5-1) as follows:

1) The calculated peak fuel element clad temperature is below the requirement of 2200°F.

2) 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 the metal in the cladding cylinders surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react.

3) The clad temperature transient is terminated at a time when the core geometry is still amenable to cooling. The localized cladding oxidation limits of 17 percent are not exceeded during or after quenching.

4) The core remains amenable to cooling during and after the break.

5) The core temperature is reduced and decay heat is removed for an extended period of time, as required by the longlived radioactivity remaining in the core.

These criteria were established to provide significant margin in ECCS performance following a LOCA.

15.6.5.2 Large Break LOCA Transient

15.6.5.2.1 Description of large break LOCA transient

The rupture of a RCS pipe is assumed to occur on the pump discharge side of a cold leg pipe. Loss-of-offsite power is assumed to occur co-incident with the LOCA. Primary coolant pump coastdown occurs co-incident with the loss-of-offsite power. Following the break, depressurization of the reactor coolant system, including the pressurizer, occurs. A reactor trip signal occurs when the pressurizer low pressure trip setpoint is reached. Reactor trip and scram are conservatively neglected in the LOCA analysis. Early in the blowdown, the reactor core experiences flow reversal and stagnation which causes the fuel rods to pass through critical heat flux (CHF). Following CHF, the fuel rods dissipate heat through the transition and film boiling modes of heat transfer. Rewet is precluded during blowdown by Appendix K of 10 CFR 50.

A Safety Injection System (SIS) signal is actuated when the appropriate setpoint (high containment pressure) is reached. Due to loss-of-offsite power, a time delay for startup of the diesel generators and SIS pumps is assumed. Once the time delay criteria is met and the system pressure falls below the shutoff head of the Charging/Safety Injection Pumps (CSIPs) and Low Head Safety Injection/Residual Heat Removal (LHSI/RHR) pumps, SIS flow is injected into the cold legs. The single failure criterion is met by assuming the loss of one ECCS pumped injection train. One HHSI pump, one LHSI pump and two containment spray pumps are assumed to be operating. When the system pressure falls below the Cold Leg Accumulator pressure, flow from the Cold Leg Accumulator is injected into the cold legs. Flow from the Emergency Core Cooling System (ECCS)is assumed to bypass the core and flow to the break until the end-of-bypass (EOBY) is predicted to occur (sustained downflow in the downcomer). Following EOBY, ECCS flow fills the downcomer and lower plenum until the liquid level reaches the bottom of the core (beginning-of-core-recovery or BOCREC time). During this downcomer and lower plenum refill period, heat is transferred from the fuel rods by radiation heat transfer.

Reflood begins at BOCREC time. ECCS fluid fills the downcomer and provides the driving head to move coolant through the core. As the mixture level moves up the core, steam is generated. Steam binding occurs as the steam flows through the intact and broken loop steam generators and pumps. The pumps are assumed to have a locked rotor (per Appendix K of 10 CFR 50) which tends to reduce the reflood rate. The fuel rods are eventually cooled and quenched by radiation and convective heat transfer as the quench front moves up the core. The reflood heat transfer rate is predicted through experimentally determined heat transfer and carry-over rate fraction correlations.

Continued operation of the ECCS pumps supplies water during long-term cooling. Core temperatures have been reduced to long-term steady state levels associated with dissipation of residual heat generation. After the water level of the refueling water storage tank (RWST) reaches a minimum allowable value, coolant for long-term cooling of the core is obtained by switching from the injection mode to the cold leg recirculation mode of operation in which spilled borated water is drawn from the containment sumps by the low head safety injection (RHR) pump and returned to the RCS cold legs. The Containment Spray System continues to operate to further reduce containment pressure. Hot leg recirculation is established to control boric acid concentration in the reactor vessel when the following criteria are met:

1) The safety injection system has previously been aligned for cold leg circulation (meaning that the Refueling Water Storage Tank level has been depleted), and

2) 6.5 hours have passed since the beginning of the event, and

3) Safety Injection has not been terminated such that a single Charging Safety Injection Pump has been realigned to the charging header (meaning that Reactor Coolant System subcooling and Pressurizer level have been established)(Reference 15.6.5-34). See Sections 6.3.2.5.2.3 and 6.3.2.8.

15.6.5.2.2 Large break LOCA evaluation model

The PWR ECCS evaluation model described in Reference 15.6.5-36 was used to perform the LBLOCA analysis. Additional restrictions are applied to the implementation of EMF-2103 as described in the plant specific implementation methodology ANP-3011 (Reference 15.6.5-50). The EMF-2103 model consists of the following computer codes:

1) RODEX2 3A for initial stored energy, fission gas release, and gap conductance;

2) S-RELAP5 for the system calculation and;

3) AUTORLBLOCA for generation of ranged parameter values, transient input, transient runs, and general output documentation.

The Shearon Harris nuclear reactor is a Westinghouse three-loop pressurized water reactor with a dry containment. The reactor coolant system (RCS) is divided into control volumes representing reasonably homogeneous regions, interconnected by flow paths or "junctions." The reactor coolant pump performance characteristics are the Westinghouse pump homologous curves built into the S-RELAP5 code. Three percent of the tubes in each steam generator are assumed to be plugged.

The transient behavior was determined from the governing equations for the conservation of mass, energy, and momentum. Energy transport, flow rates, and heat transfer are determined from appropriate correlations. The reactor core model uses heat generation rates determined from reactor kinetics equations which use reactivity feedback and decay heating from the 1979 ANSI/ANS standard.

15.6.5.2.3 Input parameters and initial conditions

Table 15.6.5-1 lists important input parameters, fuel design parameters and initial conditions used in the analysis. The LOCA analysis is based on a full core of AREVA fuel operating at a power of 2958 MWt (2948 MWt plus 10 MWt uncertainty), a peak rod average exposure less than 62,000 MWD/MTU, a total peaking factor (F_Q^T) of 2.52, a nuclear enthalpy rise factor ($F_{\Delta H}$) of 1.73 (including 4% uncertainty) and no axial or burnup dependent power peaking limit.

15.6.5.2.4 Large break LOCA results

The Peak Cladding Temperature (PCT) was calculated to be 1935°F for the limiting case which has a 3.6168 ft^2 Cold Leg Guillotine (CLG) (one sided break area shown) and models the loss of one ECCS train (i.e., one HHSI and one LHSI/RHR pump). The maximum local Zr-H_2O reaction was calculated to be 4.2%. The core wide Zr-H_2O reaction was less than 1.0%.

The NRC review of the HNP specific methodology (ANP-3011) identified a previously un-quantified issue with axial relocation of fuel pellet material in a rod that bursts during a LBLOCA. Progress Energy committed (Reference 15.6.5-51) to include a penalty to 138°F to the analysis PCT value (1935°F) to account for the uncertainties of the effect and the potential benefits of the rupture in the flow channel. The net result is a LOCA PCT of 2073°F for the purposes of NRC reporting against the regulatory limit of 2200°F.

This analysis supports full power operation T_{avg} ranging from 582°F to 594.8°F.

Figures 15.6.5-2 through 15.6.5-20 show transient results for the limiting case.

Break Spectrum versus Axial Shape Results - Calculations were performed with a range of DECLG break sizes from approximately 0.26 times to 1.0 times the full break area.

The limiting case has a 3.6168 ft^2 CLG break (total break area of 7.2336 ft^2). A constant value of K(z) = 1 was used for all core elevations.

Single Failure - The single failure evaluated was the loss of one diesel generator, namely a HHSI pump and a LHSI pump. However, all containment sprays and fans are assumed to function with minimum start time delays.

Exposure Study Results - The current AREVA methodology considers the effects of peak fuel rod exposures. The Linear Heat Generation Rate limit is therefore independent of exposure up to a peak rod average exposure of 62,000 MegaWatt Days/Metric Ton of Uranium.

Long Term Criticality Analysis Results - The long term post-LOCA analysis assures that the mixed mean boron concentration in the containment sump is higher than the corresponding critical boron concentration with no credit for control rod insertion.

15.6.5.3 Small Break LOCA Transient

15.6.5.3.1 Description of small break LOCA transient

The postulated SBLOCA covers a range of break areas that encompasses small lines which penetrate the primary pressure boundary. Small breaks could involve relief and safety valves, charging and letdown lines, drain lines, and instrumentation lines. The most limiting break location is in the cold leg pipe at the discharge side of the pumps. This break location results in the largest amount of inventory loss and the largest fraction of Emergency Core Cooling System (ECCS) fluid being ejected out through the break. This produces the greatest degree of core uncovery and the longest fuel rod heatup time.

The SBLOCA transient is characterized by a slow depressurization of the primary system with a reactor trip occurring at a low primary pressure (1934.7 psia). The Safety Injection Actuation Signal (SIAS) occurs when the system has further depressurized (1714.7 psia for Harris). The capacity and shutoff head of Charging/Safety Injection Pumps (CSIPs) are important parameters in the SBLOCA transient.

The SBLOCA transient can be categorized into three ranges of break sizes. The scenario is different for each range of break sizes. The "small" small breaks are characterized by inventory losses that are less than the makeup capacity of the CSIPs such that core uncovery is limited or precluded. The core level is eventually recovered and hot rod heatup is limited. "Large" small breaks are characterized by a larger primary system depressurization rate such that the accumulator pressure is reached in sufficient time to limit the core uncovery and hot rod heatup. The CSIPs have limited influence in the "large" small break transient. The break sizes between the "small" and "large" small breaks are generally the most limiting. For "medium" small breaks, the rate of inventory loss from the primary system is large enough that the CSIPs cannot preclude significant core uncovery. The primary system depressurization rate is very slow, extending the time required to reach the accumulator pressure. This tends to maximize the heatup time of the hot rod and produces the maximum Peak Cladding Temperature (PCT). It also results in the longest time with the core being at elevated temperatures, which maximizes the local cladding oxidation. Core recovery for the limiting break begins when intact loop SI flow and accumulator flow exceed primary system inventory lost out the break.

As contrasted with the large break, the blowdown phase of the small break occurs over a longer time period. Thus, for the small-break LOCA there are only three characteristic stages, i.e., a gradual blowdown in which the decrease in water level is checked, core recovery, and long-term recirculation.

15.6.5.3.2 Small break LOCA evaluation model

The AREVA NP SBLOCA evaluation model (References 15.6.5-40, 15.6.5-41 and 15.6.5-48) consists of two principal computer codes. The appropriate conservatisms, prescribed by Appendix K of 10 CFR 50, are incorporated. The sensitivity analyses, including time step analyses required by the NRC were run. The computer codes are:

1) The RODEX2 -2A code was utilized to determine the initial fuel stored energy and gap conditions for the initialization of the system blowdown and hot rod response calculations.

2) S-RELAP5 code was used to predict the thermal-hydraulic response of the primary and secondary sides of the reactor system and the hot rod response.

The Harris Nuclear Plant is a Westinghouse designed 3-loop PWR, having three hot leg pipes, three inverted U-tube generators, and three cold leg pipes with one reactor coolant pump in each cold leg.

The reactor coolant system of the plant is nodalized in the S-RELAP5 model into control volumes representing reasonably homogeneous regions, interconnected by flow paths or "junctions." The model includes three identical accumulators, a pressurizer, and three steam generators with both primary and secondary sides modeled. All three loops of the plant were simulated separately in order to provide an accurate representation of the plant. A steam generator tube plugging level of 3% was assumed. The HHSI pumps and lines were explicitly modeled to simulate minimum injection flow rates prescribed by the Technical Specifications. Since the system pressure does decay to the shutoff head of the Low Head Safety Injection (LHSI) pumps in the SBLOCA analysis, LHSI pumps and lines were included in the S-RELAP5 model. The primary coolant pump performance curves were characteristic of Westinghouse pumps.

The heat generation rate in the S-RELAP5 reactor core model was determined from reactor kinetics equations with actinide and decay heating as prescribed by Appendix K.

Single failure criteria were satisfied by the assumed loss of a diesel generator, which results in the disabling of one Charging/Safety Injection Pump (CSIP) and one of the two motor-driven auxiliary feedwater pumps. Although three CSIPs may be physically installed, only two of the three pumps are electrically connected and considered operational at any given time. Therefore, in the analysis, only one CSIP was available to mitigate the event and the analytically assumed pump performance bounds any combination of two of the possible three CSIPs that may have been in service at the initiation of the event.

For the Harris Nuclear Plant, the Reactor Coolant Pumps (RCPs) are manually tripped. The RCP trip criteria given in the Emergency Operating Procedure (Reference 15.6.5-43) are listed in Table 15.6.5-10. In the analysis, the RCPs were tripped at reactor scram. Tripping the RCPs at scram is conservative as this assumption impedes the loop seal clearing event and hence allows additional mass to escape from the system prior to the clearing of the loop seals. The reduced system water inventory, subsequently, contributes to the severity of the transient.

A conservative top skewed axial power shape was used in this analysis. The shape was adjusted to be consistent with the Technical Specification $F_{\Delta H}$ and $F\frac{T}{Q}$ limits.

Important system parameters used in this analysis are given in Table 15.6.5-10.

15.6.5.3.3 Small break LOCA results

SBLOCA break spectrum calculations were performed for break diameters from 0.75 inch to 9.0 inches in one of the cold legs of the reactor coolant system. The break spectrum calculations were based on the nominal auxiliary feedwater flow from one motor-driven pump.

The break spectrum calculations were performed at a nominal primary T_{avg} of 588.8°F.

Predicted event times from the break spectrum calculations are summarized in Table 15.6.5-11a. Results from S-RELAP5 hot rod response calculations for the limiting case are presented in Table 15.6.5-11b.

Review of the predicted event times in Table 15.6.5-11a shows that for the limiting break size (2.6-inch diameter) the RCP trip criterion is satisified and break uncovery occurs at approximately 15 minutes into the event. Thus, well over 5 minutes is available for operator action to manually trip the RCPs prior to break uncovery.

The 0.75-inch break represents the "small" SBLOCA for Harris. The coolant inventory loss rate for the 0.75-inch break case was slower than the makeup capacity of the SI system such that core uncovery was limited. The core level was eventually recovered and hot rod heatup was limited. Therefore, this transient was less severe.

The 9.0-inch break represents the "large" SBLOCA for Harris. The 9.0 inch break case experienced a rapid depressurization to the accumulator pressure which limited the length of time the core was uncovered and the depth of the core uncovery.

The 2.6-inch break represents the "medium" SBLOCA for Harris. The results show the 2.6-inch break to be the limiting break because it resulted in a slow rate of depressurization to the accumulator pressure, exposing the core for a long period of time, and causing the most severe fuel heatup.

System responses from the ANF-RELAP calculations for the limiting break are shown in Figures 15.6.5-30 through 15.6.5-35. The primary and secondary pressure responses are shown in Figure 15.6.5-30. The primary pressure decreased immediately after break initiation. Reactor scram occurred when the primary pressure reached 1934.7 psia. The secondary pressure increased rapidly after break initiation as the reactor scrammed and steam generator isolation took place. The secondary pressure continued to increase until the steam generator safety valves opened, causing the secondary pressure to stabilize. At approximately 878 seconds, liquid was expelled from the loop seal piping, allowing steam to flow directly to the break, which caused the primary pressure to decrease more rapidly.

The break flow rate is shown in Figure 15.6.5-31. The two-phase flow regime from 400 seconds to 878 seconds (loop seal clears) is characterized by an essentially fixed flow with a large oscillation band. The final single-phase flow regime is steam only. While a relatively large reduction in break mass flow occurs at the onset of this regime, the break volumetric flow actually increases, thereby causing an increase in the primary system depressurization rate. Oscillations in the break flow rate after 2020 seconds were caused by the initiation of accumulator flow, shown in Figure 15.6.5-34.

The downcomer and hot assembly collapsed liquid levels are shown in Figure 15.6.5-32. The mixture level remains above the core over the first 1300 seconds until the combination of decreasing pressure and inventory causes the level to begin to decrease. Sustained core uncovery begins at approximately 1300 seconds. The level continues to fall until accumulator flow is finally activated at 2020 seconds and the core begins to recover.

The total HHSI flow is shown in Figure 15.6.5-33. Flow begins at approximately 60 seconds and increases as primary system pressure decreases.

Flow from the accumulators, shown in Figure 15.6.5-34, begins at 2020 seconds and terminates the event.

The reactor vessel fluid mass, shown in Figure 15.6.5-35, declines rapidly after event initiation. After loop seal clearing at approximately 878 seconds, the amount of mass in the primary system continues to decline but at a reduced rate. The minimum primary system mass occurs at approximately 2038 seconds. By 2038 seconds, accumulator flow is active and the primary system mass begins to increase.

S-RELAP5 calculated cladding temperature for the limiting break is shown in Figure 15.6.5-36. The PCT for the limiting break SBLOCA analysis (2.6-inch break) was calculated to be 1664°F with a maximum local cladding oxidation of 2.2196%. Results from S-RELAP5 hot rod response calculations for the limiting break are presented in Table 15.6.5-11b.

15.6.5.3.4 Small break LOCA summary of results and conclusions

The SBLOCA analysis for the Harris Nuclear Plant identified the 2.6-inch diameter break to be the limiting break size. The PCT was calculated to be 1664°F with a maximum local cladding oxidation of 2.2196% for the 2.6-inch break size. The calculation was performed at T_{avg} of 588.8°F. An error of 17°F was reported by AREVA, with a revised PCT of 1681°F.

In Reference 15.6.5-53, HNP requested a change to Technical Specifications to increase the as-found lift setting tolerance for the mains steam line code safety valves from ± 1% to ± 3%. A consequence of the increased MSSV setpoint tolerance is a reduction of the credited AFW flow in the safety analyses at the lowest lifting MSSV setpoint plus tolerance. The change to AFW flow is a reduction from 390 gpm to 374 gpm. In the SBLOCA transients, secondary pressure rises to the MSSV setpoint upon reactor/turbine trip and remains there until primary phase change at the break occurs with a commensurate increase in energy release from the primary system. Early in a SBLOCA event, an increase in the MSSV setpoint tolerance can affect the energy balance during the transient because it results in a secondary heat sink temperature change. The higher setpoints of the MSSV's cause less heat transfer from the primary system and higher primary pressure. This results in less HPSI flow into the system, an earlier core uncovery, and more extensive cladding heatup. The estimated impact of this change on the SBLOCA analysis calculated peak cladding temperature is +32°F, with a revised PCT of 1713°F.

The analysis supports full power operation of the Harris Nuclear Plant at 2958 MWt (2948 MWt plus 10 MWt uncertainty) with a steam generator tube plugging level of up to 3.0% at full-power nominal T_{avg} of 588.8°F. The analysis supports an $F_{\Delta H}$ of 1.73, and an F_Q^T of 2.52 with an axially independent power peaking limit curve.

Operation of the Harris Nuclear Plant with AREVA fuel within the above stated criteria assures that the NRC acceptance criteria for SBLOCA (10 CFR 50.46(b)) will be met with the ECCS for the Harris Nuclear Plant.

15.6.5.4 Radiological Consequences Analysis of a Postulated Large Break Loss of Coolant Accident

An abrupt failure of the main reactor coolant pipe is assumed to occur and it is assumed that the emergency core cooling features fail to prevent the core from experiencing significant

degradation (i.e., melting). This sequence cannot occur unless there are multiple failures, and thus goes beyond the typical design basis accident that considers a single active failure. Activity from the core is released to the containment and from there released to the environment by means of containment leakage and leakage from the emergency core cooling system.

The input parameters and assumptions are listed in Table 15.6.5-12. Activity is released from the fuel into the containment using the timing and release fractions from RG 1.183. The analysis considers the release of activity from the containment via containment leakage. In addition, once the recirculation mode of the emergency core cooling system (ECCS) is established, activity in the sump solution may be released to the environment by means of leakage from ECCS equipment into the auxiliary building. Activity of the sump solution may also be released to the environment by means of leakage into the refueling water storage tank (RWST). The total offsite and control room doses are the sum of the doses resulting from each of the postulated release paths.

The release of activity from the core occurs over a 1.8 hour interval. Table 15.6.5-13 gives the fission product release timing assumed. A wide spectrum of nuclides is taken into consideration. Table 15.0.9-1 lists the nuclides being considered for the LOCA with core melt (eight groups of nuclides). Table 15.6.5-14 gives the core fission release fractions. The iodine is mainly in the form of cesium iodide, which exists as particulate. The iodine characterization from RG 1.183 is 4.85% elemental, 0.15% organic and 95% particulate.

For the containment leakage analysis, all activity released from the fuel is assumed to be in the containment atmosphere until removed by sprays, sedimentation, radioactive decay or leakage from the containment. For the ECCS leakage analyses, all iodine activity released from the fuel is assumed to be in the sump solution until removed by radioactive decay or leakage from the ECCS.

15.6.5.4.1 Containment Modeling

The containment building is modeled as two discrete volumes: sprayed and unsprayed. The volumes are conservatively assumed to be mixed only by the containment fan coolers. The containment volume is 2.344E6 ft^3 with a sprayed fraction of 85.9% of the total (2.014E6 ft^3).

The containment is assumed to leak at the design leak rate of 0.1% per day for the first 24 hours of the accident and then to leak at half that rate (0.05% per day) for the remainder of the 30 day period following the accident considered in the analysis.

One train of the containment spray system is assumed to operate following the LOCA. Injection spray is credited starting at 120 seconds in the event. This is conservative since it results in earlier spray termination and there is little activity in the containment at the time the sprays start. When the RWST drains to a predetermined setpoint level, the system automatically switches to recirculation of sump liquid to provide a source for the sprays. The analysis assumed that the sprays are terminated 4.0 hours from the start of the event. The elemental iodine spray removal coefficient is 20 hr^{-1}. Removal of elemental iodine from the containment atmosphere is assumed to be terminated when the airborne inventory drops to 0.5 percent of the total elemental iodine released to the containment (this is a DF of 200). With the RG 1.183 source term methodology this is interpreted as being 0.5 percent of the total inventory of elemental iodine that is released to the containment atmosphere over the duration of gap and in-vessel release phases. In the analysis, this occurs just before 2.0 hours. The particulate iodine spray

removal coefficient is 3.94 hr^{-1}. When the airborne inventory drops to 2 percent of the total particulate iodine released to the containment (this is a DF of 50) this removal coefficient is reduced by a factor of 10. In the analysis this occurs at 2.5 hours.

During spray operation, credit is taken for sedimentation removal of particulates in the unsprayed region. After sprays are terminated, credit for sedimentation is taken in both the sprayed and unsprayed regions. For the analysis the sedimentation removal coefficient is conservatively assumed to be 0.1 hr^{-1}.

15.6.5.4.2 ECCS Leakage

When ECCS recirculation is established following the LOCA, leakage is assumed to occur from ECCS equipment outside containment. There are two pathways considered for the ECCS recirculation leakage. One is the leakage directly into the Auxiliary Building and the other is back-leakage into the refueling water storage tank (RWST). Recirculation is initiated when the RWST has drained to the pre-determined setpoint level (at about 20 minutes).

It is assumed that the iodine is instantaneously mixed in the primary containment sump water at the time of release from the core.

The total ECCS recirculation leakage into the Auxiliary Building modeled in the analysis is 2 gpm (i.e., the assumed value of 1 gpm total ECCS leakage outside the containment, consistent with the amount of RCS unidentified leakage allowed by the HNP Technical Specifications, is doubled consistent with Regulatory Guide 1.183 guidance) and begins at 20 minutes. There is 2% partitioning of iodine in the leakage. Of this leakage, 1.934 gpm is inside the area served by the Reactor Auxiliary Building Emergency Exhaust System (RABEES) which filters out much of the iodine released to the atmosphere. The remaining 0.066 gpm is released outside of RABEES without filtration.

The 2 percent partitioning value for iodine releases from the fluid leaking into the ECCS area is based upon a conservative set of assumptions. In the analysis, the temperature of the sump water has been conservatively assumed to be constant at 230°F although the temperature is predicted to be reduced to 212°F after approximately 4 hours following the start of recirculation. It is also assumed that this flashing fraction is equivalent to the total partitioning fraction between iodine which stays in the ECCS leakage fluid, and that which evolves into airborne iodine for release. This 2% flashing fraction/partition fraction is retained following Steam Generator Replacement/Power Uprate, even though the maximum sump fluid temperature for SGR/PUR will exceed this temperature for a short period of time.

The fraction of recirculation sump water that would flash into steam after leaking into the ECCS area has been found to be 2 percent based on the Regulatory Guide 1.183 (Reference 15.0.9-8) Section 5.4 constant enthalpy, or "constant h" process. This reference specifies the following formula for determining the Flashing Fraction:

$$FF = \frac{h_{f_1} - h_{f_2}}{h_{fg}}$$

where:

h_{f_1} = enthalpy of liquid at sump fluid temperature and pressure

h_{f_2} = enthalpy of liquid at saturation conditions (14.7 psia, 212°F)

h_{fg} = heat of vaporization at 212°F

Treating this 2% flashing fraction as the total iodine Partition Factor, based on the 230°F sump fluid temperature, is acceptable based on the following evaluation. The fraction of iodine, (the Partition Factor, or PF) that would become airborne may be calculated using the following model (Reference 15.6.5-29):

PF = $\frac{S}{W} \chi\ 1700 \chi \frac{1}{PC}$

where,

PF = partition factor

S = mass fraction of steam

W = mass fraction of water

PC = partition coefficient, (μCi/cc liquid)/(μCi/cc gas); equivalent to the Flashing Fraction

1700 = the ratio of vapor to liquid specific volumes at 212°F

Standard Review Plan (SRP) 6.5.2, Rev. 1, indicates that long term iodine retention with no significant re-evolution may be assumed when the equilibrium sump pH, after mixing and dilution with the primary coolant and ECCS injection, is above 8.5. This view is supported by L. F. Parsly (Reference 15.6.5-30) by indicating high values of PC at pH of 9 and above, when iodate formation is significant. A value of 1.765E+09 has been indicated at 212 F, pH equal to 9 and concentration of aqueous iodine of 3E-03 moles/liter. The PC indicated in SRP Section 6.5.2, Rev. 1; Figure 5.2-1 is 5E+03. Conservatively, selecting 5E+03, PF is calculated as follows:

PF = $\frac{0.02}{0.98} \chi\ 1700 \chi \frac{1}{5E+03} = 6.9E-03$

This suggests that only 0.69 percent of the iodine leaking into the ECCS area would become airborne and be removed with the exhaust. Therefore, the 2 percent value, in effect, does not account for partition and is a conservative estimate in the dose evaluation.

Allowable back-leakage from ECCS or Containment Spray to the RWST has been evaluated up to a total maximum flow rate of 3.0 gpm. This leakage can enter the RWST below or above the remaining liquid level associated with post-LOCA sump recirculation, but must satisfy the following straight line tradeoff relationship: $L_{below} + 110\ L_{above} \leq 3.0$ gpm where L_{below} and L_{above} are the leakage rates into the RWST below and above the liquid level, respectively. The 3.0 gpm value is an analytical limit: measurement uncertainties associated with measurement of individual leakage path(s) and the impact of accident conditions on leakage measurements performed at normal operating conditions must be considered when establishing surveillance test acceptance criteria.

a) RWST to the RHR pumps - During recirculation isolation is provided by 1SI-320 and 1SI-322 (Train A RHR pump) and 1SI-321 and 1SI-323 (Train B RHR pump)

b) High Head Safety Injection Mini-flow to the RWST - During recirculation isolation is provided by 1CS-745 and 1CS-746 (Train A CSIP) and 1CS-752 and 1CS-753 (Train B CSIP)

c) RHR Test recirculation Line - During recirculation isolation is provided by 1-SI-331 (manual in-series isolation valves)

d) RWST to CSIP suction header - During recirculation isolation is provided by 1CS-291 (Train A) and 1CS-292 (Train B) which are in parallel and an in-series check valve (1CS-294)

e) Containment Spray Test line - During recirculation isolation is provided by 1CT 47 (Train A) and 1CT-95 (Train B)

f) RWST to the Containment Spray pumps - During recirculation isolation is provided by 1CT-26 and 1CT-27 (Train A Containment Spray Pump) and 1CT-71 and 1CT-72 (Train B Containment Spray Pump).

The iodine in the sump solution is assumed to all be in nonvolatile iodide or iodate form. However, when the solution leaks into the RWST, the iodine will be in an acidic solution such that there is the possibility of conversion of iodine compounds to form elemental iodine. The amount of iodine that will convert to the elemental form is dependent both on the concentration of iodine in the solution and the pH of the solution. The initial boron concentration in the RWST is ~2500 ppm. The initial pH of RWST solution is determined to be ~4.5. With an RWST pH of 4.5 and the low iodine concentration, the fraction of conversion to elemental iodine is 2%. After 24 hours, the RWST liquid pH will exceed 6.0 and the indicated conversion to elemental iodine is essentially zero; however, the fraction is conservatively assumed to be 1% for the remainder of the accident duration.

Elemental iodine is volatile and will partition between the liquid and the air in the RWST gas space. The partition coefficient for elemental iodine is determined to be 28.2 using a relationship to solution temperature. This is modeled by the transfer of a portion of the flow going to the RWST liquid and a portion going to the RWST gas space. The modeling of the air flow out of the RWST is based on diurnal heating and cooling cycle. This model ignores the effect of the large heat sink provided by the mass of water in the tank which would tend to moderate the effects of the heating and cooling from the sunlight and atmospheric temperature variations. The transfer from the RWST gas space to the environment is calculated to be 5.9 cfm based on displacement by the inleakage and air expansion from the heating/cooling cycle.

15.6.5.4.3 Offsite Doses

The offsite doses are calculated using the assumptions and equations in Section 15.0A.1.

15.6.5.4.4 Doses to Control Room Personnel

Control room inhalation doses are calculated using the following equation:

$$D_{CEDE} = \sum_{i}\left[DCF_i\left(\sum_{j} Conc_{ij} * (BR)_j\right)\right]$$

where:

D_{CEDE} = CEDE dose via inhalation (rem)

DCF_i = CEDE dose conversion factor via inhalation for isotope i (rem/Ci)(Table 15.0.9-3)

$Conc_{ij}$ = concentration in the control room of isotope i, during time interval j, calculated dependent upon inleakage, filtered recirculation and filtered inflow (Ci-sec/m3)

$(BR)_j$ = breathing rate during time interval j (m3/sec)(Table 15.6.5-15)

Control room external exposure doses are calculated using the following equation:

$$D_{EDE} = \left(\frac{1}{GF}\right) * \sum_{i} DCF_i \left(\sum_{j} Conc_{ij}\right)$$

where:

D_{EDE} = external exposure dose via cloud immersion in rem.

GF = geometry factor, calculated based on Reference 15.6.5-23, using the equation

GF= 1173V0.338 where V is the control room volume ft^3

DCF^i = EDE dose conversion factor via external exposure for isotope i ($rem*m^3/Ci*sec$)(Table 15.0.9-4)

$Conc_{ij}$ = concentration in the control room of isotope i, during time interval j, calculated dependent upon inleakage, filtered recirculation and filtered inflow ($Ci\text{-}sec/m^3$)

Parameters used in the control room personnel dose calculations are provided in Table 15.6.5-15. These parameters include the normal operation flowrates, the emergency operation flowrates, control room volume, filter efficiencies and control room operator breathing rates. The inflow (filtered and unfiltered) to the control room and the control room recirculation flow are used to calculate the activity introduced to the control room and cleanup of activity from that flow.

In the event of a large break LOCA, the SI setpoint will be reached shortly after event initiation. The SI signal causes the control room HVAC to switch from the normal operation mode to the post-accident recirculation mode of operation. It is assumed that the SI setpoint is reached immediately at the start of the event; only the 15 second delay time for switching from normal to emergency operating mode is modeled. An operator action switches the control room from the post-accident recirculation mode to the pressurization mode at 2 hours after event initiation.

CP&L Shine Dose Contribution:

As required by Reg Guide 1.183 (Reference 15.6.3-4), Section 4.2.1 on Control Room Dose Calculation Methodology, the control room dose contributions of the release plume, the containment building post-accident radionuclide inventory, and the control room HVAC filter shine doses were conservatively evaluated. The small incremental dose contributions from these sources are included in the total Control Room TEDE dose reported in Table 15.6.5-16.

15.6.5.4.5 Results

The potential radiological consequences resulting from a large break LOCA have been conservatively analyzed, using assumptions and models described in previous sections. The TEDE doses have been analyzed for the worst two hours at the EAB and for the duration of the event at the LPZ and in the control room. The results are listed in Table 15.6.5-16. The resultant doses are within the guideline values. The 10CFR50.67 guideline values are 25 rem TEDE at the EAB and LPZ and 5 rem TEDE in the control room.

15.6.5.5 Radiological Consequences Analysis of a Postulated Small Break Loss-of-Coolant Accident

An abrupt failure of the primary coolant system is assumed to occur and it is assumed that the break is small enough that the containment spray system is not immediately actuated by high containment pressure but that the core experiences substantial cladding damage such that the fission product gap activity of all fuel rods is released. Activity that is released to the containment is assumed to be released to the environment due to the containment leaking at its design rate. There is also a release path through the steam generators (primary to secondary) until the primary system becomes depressurized to below the secondary system pressure.

Separate calculations are performed to calculate the dose resulting from the release of activity to containment and subsequent leakage to the environment and the dose resulting from the leakage of activity to the secondary system and subsequent release to the environment. The total offsite and control room doses are the sum of the doses resulting from each of the postulated release paths and nuclides considered. A summary of input parameters and assumptions is provided in Table 15.6.5-17. As noted in Table 10.4.9B-1 and as an exception to the usual acceptance criteria for Condition III events, the same offsite dose acceptance criteria is applied to both the LBLOCA (discussed in Section 15.6.5.4) and Small Break LOCA.

In determining the offsite doses following a SBLOCA, it is assumed that all of the fuel rods in the core suffer sufficient damage that all of their gap activity is released and that 2% of the fuel in the core melts. Eight percent of the total I-131 core activity is in the fuel-cladding gap. Ten percent of the total Kr-85 core activity is in the fuel-cladding gap. Five percent of other iodine isotopes and other noble gases and 12 percent of the total core activity for alkali metals are assumed to be in the fuel-cladding gap. The core fission product inventory is given in Table 15.0.9-1.

Position 3.1 of RG 1.183 indicates that for accidents not involving the entire core the radial peaking factor should be applied. Since 100% of the rods are assumed to be damaged, this guidance does not apply to the gap release but does apply to the small fraction of the core involved in fuel melt. In the calculation of activity releases from the 2% melted fuel the maximum radial peaking factor of 1.73 was applied.

For both the containment leakage release path and the primary to secondary leakage release path all noble gas and alkali metal activity in the failed fuel gap and in the melted fuel is available for release.

For the containment leakage release path all of the iodine in the failed fuel gap and 25 percent of the activity in the melted fuel is available for release from containment.

For the primary to secondary leakage release path all of the iodine failed fuel gap and 50 percent of the activity in the melted fuel is available for release from the reactor coolant system.

Prior to the accident the iodine activity concentration of the primary coolant is 1.0 μCi/gm of DE I-131. The iodine activity concentration in the primary coolant is given in Table 15.0.9-7. The noble gas and alkali metal activity concentration in the RCS at the time the accident occurs is based on a one percent fuel defect level. The noble gas and alkali metal RCS concentrations are in Table 15.0.9-2. The iodine activity concentration of the secondary coolant at the time the SBLOCA occurs is assumed to be equivalent to the Technical Specification limit of 0.1 μCi/gm of DE I-131. The secondary coolant iodine activity is given in Table 15.0.9-7. The alkali metal activity concentration of the secondary coolant at the time the SBLOCA occurs is assumed to be 10% of the primary side concentration.

Iodine in containment is assumed to be 4.85% elemental, 0.15% organic and 95% particulate.

Iodine released from the secondary system is assumed to be 97% elemental and 3% organic.

15.6.5.5.1 Containment Modeling

The containment building is modeled as two discrete volumes: sprayed and unsprayed. The volumes are conservatively assumed to be mixed only by the containment fan coolers. The containment volume is 2.344E6 ft^3 with a sprayed fraction of 85.9% of the total (2.014E6 ft^3). The containment is assumed to leak at the design leak rate of 0.1% per day for the first 24 hours of the accident and then to leak at half that rate (0.05% per day) for the remainder of the 30 day period following the accident considered in the analysis.

One train of the containment spray system is assumed to operate following the SBLOCA. Injection spray is credited to automatically start at 30 minutes into the event. The analysis assumed that the sprays are terminated after 30 minutes of operation. The analysis used an elemental iodine spray removal coefficient of 20 hr^{-1} until the DF limit of 200 is reached after 15 minutes of spray operation. The particulate iodine spray removal coefficient of 3.94 hr^{-1} is credited for the entire 30 minutes of spray. These spray removal coefficients are the same as those used in the large break LOCA analysis discussed in Section 15.6.5.4.1.

After spray termination, credit is taken for sedimentation removal of particulates in both the sprayed and unsprayed regions of containment. The sedimentation removal coefficient of 0.1/hr is the same as that used in the large break LOCA analysis discussed in Section 15.6.5.4.1.

15.6.5.5.2 Primary to Secondary Leakage Release Pathway

When determining doses due to the primary to secondary steam generator tube leakage, all the iodine, alkali metals group and noble gas activity (from prior to the accident and resulting from the accident) is assumed to be in the primary coolant (and not in the containment). The primary

to secondary tube leakage and the steaming from the steam generators continue until the reactor coolant system pressure drops below the secondary pressure. A conservative time of 2 hours was used for this analysis, although analyses of the small break LOCA pressure transient have shown that this would occur well before that time.

The amount of primary to secondary SG tube leakage is assumed to be 1 gpm total. Although the primary to secondary pressure differential gradually drops, the constant flow rate is conservatively maintained until primary pressure is lower than secondary pressure.

An iodine partition factor in the SGs of 0.01 (curies iodine/gm steam) / (curies iodine/gm water) is used. This partition factor is also applied to alkali metals.

All noble gas activity carried over to the secondary side through SG tube leakage is assumed to be immediately released to the outside atmosphere.

15.6.5.5.3 Offsite Doses

The offsite doses are calculated using the assumptions and equations in Section 15.0A.1.

15.6.5.5.4 Control Room Doses

The control room assumptions are provided in Section 15.6.5.4.3 and Table 15.6.5-15. As noted in that table, a conservative factor of two is embedded in the X/Q factors used in the dose analysis results described below. The conservative factor yields numerical results that are artificially high, and explains why this SBLOCA event reported result is not the limiting control room dose consequence.

The SI setpoint will be reached within 60 seconds from event initiation. The SI signal causes the control room HVAC to switch from the normal operation mode to the post-accident recirculation mode of operation. A 15-second delay for the control room to switch between normal and post-accident recirculation modes is modeled. An operator action switches the control room from the post-accident recirculation mode to the pressurization mode at 2 hours after event initiation.

15.6.5.5.5 Results

The potential radiological consequences resulting from a small break LOCA have been conservatively analyzed, using assumptions and models described in previous sections. The TEDE doses have been analyzed for the worst two hours at the EAB and for the duration of the event at the LPZ and in the control room. The analytically predicted dose consequence to the Control Room (CR) operators due to LBLOCA increased by a small amount, based on the reduction in CR recirculation flow which reduces the iodine filtration provided by the charcoal filters relative to the analysis of record condition. The analytically predicted dose consequence to the control room operators for the SBLOCA event was not similarly revised or updated since the LBLOCA DBA is the limiting control room operator dose event. The radiological analysis results for this event are listed in Table 15.6.5-18. The resultant doses are within the guideline values. The 10CFR50.67 guideline values are 25 rem TEDE at the EAB and LPZ and 5 rem TEDE in the control room.

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15.6.5-5 Deleted by Amendment No. 45.

15.6.5-6 Deleted by Amendment No. 45.

15.6.5-7 Deleted by Amendment No. 45.

15.6.5-8 Deleted by Amendment No. 46.

15.6.5-9 Deleted by Amendment No. 46.

15.6.5-10 Deleted by Amendment No. 46.

15.6.5-11 Deleted by Amendment No. 46.

15.6.5-12 Deleted by Amendment No. 46.

15.6.5-13 Deleted by Amendment No. 46.

15.6.5-14 Deleted by Amendment No. 46.

15.6.5-15 Deleted by Amendment No. 46.

15.6.5-16 Deleted by Amendment No. 46.

15.6.5-17 Deleted by Amendment No. 46.

15.6.5-18 Deleted by Amendment No. 45.

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15.7 RADIOACTIVE RELEASE FROM A SUBSYSTEM OR COMPONENT

15.7.1 RADIOACTIVE WASTE GAS SYSTEM LEAK OR FAILURE

15.7.1.1 Identification of Causes

The most limiting waste gas accident is defined as an unexpected and uncontrolled release to the atmosphere of the radioactive xenon and krypton fission gases that are stored in one operating waste gas decay tank with maximum curie content as discussed below. Two waste gas decay tanks may be cross-connected for a short period of time to transfer tank contents only if both tanks are isolated from Waste Gas System influents and the total curie content is

less than the maximum curie content specified in this section. Rupture of these two tanks while cross-connected is limited by the results of this analysis. The Gaseous Waste Processing System (GWPS) is described in Section 11.3.

Since the components of the GWPS are subjected to pressures no greater than 100 psig, a failure is considered to be unlikely (design pressure 150 psig). A failure probability is further reduced since the GWPS is Safety Class 3, Seismic Category I, except for compressors which are classified as Rad-Q.

However, the probability of an accidental release resulting from such events as (1) failure of gas decay tank or associated piping, or (2) lifting and subsequent failure of a relief valve to close, is defined as a limiting fault and is analyzed to define the upper limit of a gaseous release that could result from any malfunction in the GWPS.

15.7.1.2 Analysis of Events and Consequences

Assumptions and methods used in this analysis are consistent with those of Regulatory Guide 1.24 (3/23/72) as discussed in Section 1.8. Table 15.7.1-1 lists the conservative assumptions used in the analysis.

It is assumed that the plant has been operating at the power level of 2958 MWt (rated power of 2948 MWt with 0.34% uncertainty) with one percent failed fuel for an extended period sufficient to achieve equilibrium radioactive concentrations in the Reactor Coolant System. As soon as possible after shutdown, all noble gases have been removed from the Reactor Coolant System and transferred to the gas decay tank which is assumed to release its contents in an uncontrolled manner. Radiological decay is assumed only for the minimum time period required to transfer the gases from the Reactor Coolant System to the waste gas decay tank. The release is assumed to occur immediately upon completion of the waste gas transfer, releasing the entire contents of the tank to the Waste Processing Building. All of the noble gases are assumed to leak out of the building at ground level over a two hour period. The activity released to the Waste Processing Building and subsequently to the environment is given in Table 15.7.1-2.

15.7.1.3 Radiological Consequences Analysis

15.7.1.3.1 Offsite Doses

The offsite doses are calculated using the assumptions and equations in Section 15.0A.1.

15.7.1.3.2 Control Room Doses

The control room assumptions are provided in Section 15.6.5.4.3 and Table 15.6.5-15. The gas decay tank rupture control room doses modeled 500 cfm unfiltered inleakage.

It is assumed that the control room HVAC system begins in the normal operational mode. The activity level causes a high radiation signal almost immediately. It is conservatively assumed that the post-accident recirculation control room HVAC mode is entered 15 seconds after event initiation. The control room is assumed to be placed in the pressurization mode by operator action at 2 hours after isolation signal.

15.7.1.3.3 Results

The analytically predicted dose consequence to the Control Room (CR) operators due to LBLOCA increased by a small amount, based on the reduction in CR recirculation flow which reduces the iodine filtration provided by the charcoal filters relative to the analysis of record condition. The analytically predicted dose consequence to the control room operators for the gas decay tank rupture event was not similarly revised or updated since the LBLOCA DBA is the limiting control room operator dose event. The radiological analysis results for the gas decay tank rupture are listed in Table 15.7.1-2. The TEDE doses have been analyzed for the worst two hours at the EAB and for the duration of the event at the LPZ and in the control room. The resultant doses are within the guideline values. The offsite dose limit for a gas decay tank rupture is given in HNP Technical Specifications 6.8.4j as 0.5 rem whole body. This translates to a dose limit of 0.5 rem TEDE. The 10CFR50.67 limit in the control room is 5 rem TEDE.

15.7.1.4 Conclusions

The dose analysis results in Table 15.7.1-2 are within the acceptance limits established for this event.

15.7.2 LIQUID WASTE SYSTEM LEAK OR FAILURE

HNP Liquid Waste System leaks or failures are bounded by the Analyzed Liquid Tank Failure in Section 15.7.3.

15.7.3 POSTULATED RADIOACTIVE RELEASES DUE TO LIQUID TANK FAILURE

The consequences of the postulated failure of a tank containing potentially contaminated liquid on the nearest potable water supply and the nearest surface water in an unrestricted area are discussed in Sections 2.4.12 and 2.4.13.

15.7.3.1 Results and Conclusions

The analysis of this event is based on operation at 2958 MWt (rated power of 2948 MWt with 0.34% uncertainty) with 1% failed fuel. The results shown in Sections 2.4.12 and 2.4.13 meet their acceptance criteria.

15.7.4 DESIGN BASIS FUEL HANDLING ACCIDENTS

15.7.4.1 Identification of Causes and Accident Description

The possibility of a fuel handling accident is remote because of the many interlocks, administrative controls, and physical limitations imposed on the fuel handling operations. All refueling operations are conducted in accordance with prescribed procedures under direct surveillance of a senior reactor operator (SRO). The analyzed Fuel Handling Accident inside containment involves dropping a spent fuel assembly resulting in the rupture of the cladding of all the fuel rods (264) in the assembly.

The projected worst case Fuel Handling Accident (FHA) in the Fuel Handling Building (FHB) involves dropping a recently discharged (100 hr decayed) PWR assembly (including the handling tool) on top of another recently discharged PWR assembly in a fuel storage rack. The

dropped assembly subsequently falls over landing on BWR fuel assemblies in an adjacent storage rack. Fifty fuel rods are projected to fail in the impacted PWR assembly in storage and all of the rods (264) in the dropped assembly fail when the assembly falls over (Reference 15.7.4.5). Due to the upper bail handle of the BWR fuel assemblies extending above the top of the BWR storage racks, up to 52 BWR assemblies could be impacted when the dropped PWR assembly falls over. All of the rods in the impacted BWR assemblies are assumed to fail.

15.7.4.2 Radiological Consequences Analysis

15.7.4.2.1 Input Assumptions Common to both FHA in the FHB and in Containment

Consistent with Regulatory Guide 1.183 (Position 1.2 of Appendix B), the radionuclides considered are xenons, kryptons, halogens, cesiums and rubidiums. The list of xenons, kryptons, and halogens considered is given in Tables 15.7.4-1 and 15.7.4-3. The cesium and rubidium are not included because they are not assumed to be released from the pool as discussed later.

The calculation of the radiological consequences following a FHA uses gap fractions of 8% for I-131, 10% for Kr-85, and 5% for all other nuclides.

Iodine species in the pool is 99.85% elemental and 0.15% organic iodine. This is based on the split leaving the fuel of 95% cesium iodide (CsI), 4.85% elemental iodine and 0.15% organic iodine. It is assumed that all CsI is dissociated in the water and re-evolves as elemental. This is assumed to occur instantaneously. Thus, 99.85% of the iodine released is elemental.

The water above the damaged fuel rods retains a large fraction of the gap activity of iodines. An overall effective decontamination factor (DF) of 200 is used. The split between elemental and organic iodine leaving the pool has no impact on the analyses since the filter efficiencies credited in the analyses for the two forms of iodine are the same.

The cesium and rubidium released from the damaged fuel rods is assumed to remain in a nonvolatile form and would not be released from the pool.

15.7.4.2.2 Postulated Fuel Handling Accident in the FHB

The major assumptions and parameters used in the analysis are itemized in Table 15.7.4-1. This analysis involves dropping a recently discharged (100 hour decay) PWR fuel assembly onto 52 Brunswick BWR fuel assemblies. This analysis also includes 50 PWR rods additionally damaged in the accident. the assembly inventory is based on the assumption that the PWR fuel assembly has been operated at 1.73 times the core average power and the BWR fuel assemblies have been operated at 1.5 times the core average power. All activity released from the fuel pool is assumed to be released to the atmosphere in two hours.

The BWR fuel inventory was conservatively evaluated at the IF-300 spent fuel shipping cask limits for GE- 7, 8, 9, 10, and 13 fuel assemblies with a maximum average lattice enrichment of 4.25 wt. % U-235 and a maximum assembly average burnup of 45 GWD/MTU. The decay time used in the analysis is 100 hours for the PWR fuel and 4 years for the BWR fuel. Thus, the analysis supports the design basis limit of 100 hours decay time prior to fuel movement.

It was determined that for the HNP specific water height above the failed fuel in the fuel handling building of 21 feet, the elemental DF would be at least 291, compared to the Reg. Guide 1.183 allowable elemental DF of 500. Using the elemental DF 291, it was determined that overall effective DF for 21 feet of coverage would be 203. Since this continues to exceed the Reg. Guide 1.183 cited overall effective DF of 200, it remains conservative to use the overall DF of 200 in the HNP dose calculations.

Credit is taken for removal of iodine by filters by the spent fuel pool ventilation system operation. Credit is not taken for isolation of release paths.

The activity released from the damaged assemblies is assumed to be released to the fuel building and subsequently to the atmosphere over a 2 hour period.

15.7.4.2.3 Postulated Fuel Handling Accident in Containment

A fuel assembly is assumed to be dropped in containment and damaged during refueling. Activity released from the damaged assembly is released to the outside atmosphere through the containment openings (such as the personnel air lock door or the equipment hatch).

The major assumptions are parameters used in the analysis are itemized in Table 15.7.4-3. This analysis involves dropping a recently discharged (100 hour decay) PWR fuel assembly. All activity released from the fuel pool is assumed to be released to the atmosphere in two hours. The pool referred to in RG 1.183 is interpreted as the flooded reactor cavity for the purposes of evaluating the fuel handling accident in containment. No credit is taken for isolation of containment for the FHA containment.

The calculation of the radiological consequences following a FHA uses gap fractions of 8% for I-131, 10% for Kr-85, and 5% for all other nuclides.

It is assumed that all of the fuel rods in the equivalent of one fuel assembly (264 rods) are damaged to the extent that all their gap activity is released. The assembly inventory is based on the assumptions that the subject fuel assembly has been operated at 1.73 times the core average power.

The decay time used in the analysis is 100 hours.

It was determined that for HNP specific water height above the failed fuel in the containment of 22 feet, the elemental DF would be at least 382, compared to the Reg. Guide 1.183 allowable elemental DF of 500. Using the elemental DF of 382, it was determined that the overall effective DF for 22 feet of coverage would be 243. Since this continues to exceed the Reg. Guide 1.183 cited overall effective DF of 200, it remains conservative to use the overall DF of 200 in the HNP dose calculations.

No credit is taken for removal of iodine by filters nor is credit taken for isolation of release paths.

Although the containment purge will be automatically isolated on a purge line high radiation alarm, isolation is not modeled in the analysis. The activity released from the damaged assembly is assumed to be released to the outside atmosphere over a 2 hour period. Since no filters or containment isolation is modeled, this analysis supports refueling operation with the equipment hatch or personnel air lock remaining open.

15.7.4.2.4 Offsite Doses

The offsite doses are calculated using the assumptions and equations in Section 15.0A.1.

15.7.2.2.5 Control Room Doses

The control room assumptions are provided in Section 15.6.5.4.3 and table 15.6.5-15. The FHA control room doses modeled 500 cfm unfiltered inleakage.

It is assumed that the control room HVAC system begins in normal mode. The activity level in the intake duct causes a high radiation signal almost immediately. It is conservatively assumed that the post-accident recirculation control room HVAC mode is entered 15 seconds after event initiation. The control room HVAC is placed into pressurization mode at 2 hours after isolation signal.

15.7.4.2.6 Results

The analytically predicted dose consequence to the Control Room (CR) operators due to LBLOCA increased by a small amount, based on the reduction in CR recirculation flow which reduces the iodine filtration provided by the charcoal filters relative to the analysis of record condition. The analytically predicted dose consequence to the control room operators for the FHA events were not similarly revised or updated since the LBLOCA DBA is the limiting control room operator dose event. The radiological analysis results for the FHA in FHB doses are listed in Table 15.7.4-2. The FHA in Containment doses are listed in Table 15.7.4-4. The TEDE doses have been analyzed for the worst two hours at the EAB and for the duration of the event at the LPZ and in the control room. The resultant doses are within the applicable limits. The offsite doses are less than ~25% of the 10CFR50.67 limits (i.e., 6.3 rem TEDE) and the control room dose is less than the 10CFR50.67 limit of 5 rem TEDE.

15.7.4.3 DELETED

15.7.4.3.1 DELETED

15.7.4.3.2 DELETED

15.7.4.4 Deleted

15.7.4.4.1 Deleted

15.7.4.4.2 Deleted

15.7.4.4.3 Deleted

15.7.4.4.4 Deleted

15.7.4.5 Other Fuel Handling Accidents

Fuel handling drop accidents involving the other fuel handling tools (BPRA, RCCA change tool, spent fuel handling tool), and items carried by the tools have also been evaluated (Reference 15.7.4-7) and are addressed in Section 9.1. The tool drop scenarios involve dropping the tools,

and items carried by the tools, onto PWR spent fuel racks, BWR spent fuel racks, and combinations of both. For all cases evaluated, the off-site dose consequences were determined to be bounded by the Fuel Handling Accident described in FSAR Section 15.7.4.5 which addresses a fuel handling drop accident which results in damage to 314 PWR spent fuel rods and 52 BWR spent fuel assemblies (Reference 15.7.4-7, pages 3.2.2-3.23).

15.7.5 SPENT FUEL CASK DROP ACCIDENTS

15.7.5.1 Cask Drop Into the New or Spent Fuel Pool

As discussed in Section 9.1, the cask handling crane is prohibited from traveling over the new and spent fuel pools or any unprotected safety related equipment. Thus, an accident resulting from dropping a cask or other major load into the new or spent fuel pools is not credible.

15.7.5.2 Cask Drop to Flat Surface

The cask has full integrity when the head is fully tensioned and the valve box covers are installed.

15.7.5.2.1 Cask with full integrity

As discussed in Section 9.1, the potential drop of a spent fuel cask is limited to less than an equivalent 30 ft. drop onto a flat, essentially unyielding, horizontal surface. Since the spent fuel cask, with the valve box covers installed and the head fully tensioned, is designed to withstand such loadings, the radiological consequences of dropping the cask in this condition are not evaluated.

15.7.5.2.2 Cask with less than full integrity

The loaded IF-300 series cask may be moved with the valve covers removed and, from the decon pit to the unloading pool, with only four cask head bolts installed. An evaluation of a 30-ft. drop during the movement from the decon pit to the unloading pool was performed and indicated that, while fuel components would be retained in the cask, the cask is not expected to be gas tight. Noble gas and iodine gap activity could be released to the Fuel Handling Building and subsequently to the environment. Damage to the valves caused by dropping the cask could cause the same type of release. The radiological consequences from this release were analyzed for an IF-300 series cask. This analysis utilized the worst case fuel types anticipated to be shipped in the cask, as shown in Table 15.7.5-1. The results of this analysis show that these consequences would be a small fraction of the 10CFR100 exposure guidelines. The analysis is documented in Harris Plant calculation HNP-M/FHB-1001.

REFERENCES: SECTION 15.7

15.7.4-1 Industrial Ventilation, 8th Edition, American Conference of Governmental Industrial Hygienists.

15.7.4-2 Deleted by Amendment No. 49

15.7.4-3 Deleted by Amendment No. 51

15.7.4-4 Regulatory Guide 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors", July 2000.

15.7.4-5 Westinghouse Letter, 97CP-G-0006; Christine M. Vertes to Leo Martin, dated April 9, 1997, "Limiting Fuel Handling Accident Assumptions."

15.7.4-6 Deleted by Amendment No. 49

15.7.4-7 ESR 98-00181 "Fuel Handling Tool Drop onto Spent Fuel Rack Evaluation"

15.7.4-8 Deleted by Amendment No. 51

15.7.4-9 Deleted by Amendment No. 51

15.7.4-10 CP&L Calculation HNP-M/FHB-1001 "Off-site Doses from FHB Cask Drop."

15.8 ANTICIPATED TRANSIENTS WITHOUT SCRAM

An anticipated transient without scram (ATWS) is an anticipated operational occurrence (such as loss of feedwater, loss of condenser vacuum, or loss of offsite power) that is accompanied by a failure of the reactor trip system to shut down the reactor. A series of generic studies (References 15.8.0-1, 15.8.0-2) on ATWS showed that acceptable consequences would result provided that the turbine trips and auxiliary feedwater flow is initiated in a timely manner.

The final USNRC ATWS rule (Reference 15.8.0-3) requires that all US Westinghouse-designed plants install ATWS mitigation system actuation circuitry (AMSAC) to initiate a turbine trip and actuate auxiliary feedwater independent of the reactor trip system. SHNPP is in compliance with the final ATWS rule by virtue of having installed a NRC-approved AMSAC. For the revised steam generator replacement/uprating conditions, the generic studies in References 15.8.0-1 through 15.8.0-3 have been shown to remain applicable to HNP (Reference 15.8.0-4).

For the Measurement Uncertainty Recapture (MUR) Power Uprate, the generic studies were updated to reflect the MUR plant conditions, and to include plant-specific AMSAC setpoints and delay times. The analyses demonstrated that the effects of the MUR would not result in unacceptable consequences (Reference 15.8.0-5).

REFERENCES: SECTION 15.8

15.8.0-1 Burnett, T.W.T., et al., "Westinghouse Anticipated Transients Without Trip Analysis," WCAP-8330, August 1974.

15.8.0-2 Letter from T. M. Anderson (Westinghouse) to S. H. Hanauer (USNRC), "Anticipated Transients Without Scram for Westinghouse Plants," NS-TMA-2182, December 1979.

15.8.0-3 ATWS Final Rule, Code of Federal Regulations 10 CFR 50.62, "Requirements for Reduction of Risk from Anticipated Transients Without Scram (ATWS) Events for Light-Water-Cooled Nuclear Power Plants."

15.8.0-4 "Steam Generator Replacement/Uprating Analysis and Licensing Project - NSSS Licensing Report," WCAP-15398 (Proprietary) and WCAP-15399 (Non-proprietary), June 2000.

15.8.0-5 "Harris Nuclear Plant Measurement Uncertainty Recapture Power Uprate Engineering Report," WCAP-17209-P (Priority), Revision 2, March 2012.

TABLE 15.0.1-1 ACCIDENT CATEGORY USED FOR EACH CHAPTER 15 EVENT

FSAR Event Designation	Event Name	Condition[a]
15.1	**INCREASE IN HEAT REMOVAL BY SECONDARY SYSTEM**	
15.1.1	Decrease in Feedwater Temperature	II (AOO)
15.1.2	Increase in Feedwater Flow	II (AOO)
15.1.3	Increase in Steam Flow	II (AOO)
15.1.4	Inadvertent Opening of a Steam Generator Relief or Safety Valve	II (AOO)
15.1.5	Steam Line Break	IV (PA)
15.2	**DECREASE IN HEAT REMOVAL BY THE SECONDARY SYSTEM**	
15.2.1	Steam Pressure Regulator Malfunction	N/A
15.2.2	Loss of External Load	II (AOO)
15.2.3	Turbine Trip	II (AOO)
15.2.4	Inadvertent Closure of MSIV's	II (AOO)
15.2.5	Loss of Condenser Vacuum	II (AOO)
15.2.6	Loss of Nonemergency AC Power	II (AOO)
15.2.7	Loss of Normal Feedwater	II (AOO)
15.2.8	Feedline Break	IV (PA)
15.3	**DECREASE IN REACTOR COOLANT SYSTEM FLOW**	
15.3.1	Partial Loss of Forced Reactor Coolant Flow	II (AOO)
15.3.2	Complete Loss of Forced Reactor Coolant Flow	III (PA)
15.3.3	RCP Shaft Seizure (Locked Rotor)	IV (PA)
15.3.4	RCP Shaft Break	IV (PA)
15.4	**REACTIVITY AND POWER DISTRIBUTION ANOMALIES**	
15.4.1	Uncontrolled RCCA Bank Withdrawal from a Subcritical or Low Power Startup Condition	II (AOO)
15.4.2	Uncontrolled RCCA Bank Withdrawal at Power	II (AOO)
15.4.3	RCCA Misoperation	
	1) Dropped Rod/Bank	II (AOO)
	2) Single Rod Withdrawal	III (PA)
	3) Statically Misaligned RCCA	II (AOO)
15.4.4	Startup of an Inactive RCP at an Incorrect Temperature	II (AOO)
15.4.5	A Malfunction or Failure of the Flow Controller in a BWR Loop that Results in an Increased Reactor Coolant Flow Rate	N/A
15.4.6	CVCS Malfunction that Results in a Decrease in the Boron Concentration in the Reactor Coolant	II (AOO)
15.4.7	Inadvertent Loading and Operation of a Fuel Assembly in an Improper Position	III (PA)
15.4.8	Spectrum of Rod Cluster Control Assembly Ejection Accidents	IV (PA)
15.5	**INCREASE IN REACTOR COOLANT INVENTORY**	
15.5.1	Inadvertent Operation of the ECCS During Power Operation	II (AOO)
15.5.2	CVCS Malfunction that Increases RCS Inventory	II (AOO)
15.6	**DECREASE IN REACTOR COOLANT INVENTORY**	
15.6.1	Inadvertent Opening of a Pressurizer Safety or PORV	II (AOO)

TABLE 15.0.1-1 ACCIDENT CATEGORY USED FOR EACH CHAPTER 15 EVENT

FSAR Event Designation	Event Name	Condition[a]
15.6.2	Radiological Consequences of the Failure of Small Lines Carrying Primary Coolant Outside Containment	II (AOO)
15.6.3	Radiological Consequences of Steam Generator Tube Rupture	IV (PA)
15.6.4	Radiological Consequences of a Main Steam Line Failure Outside Containment	N/A
15.6.5	Loss of Coolant Accidents	
	1) Large Break LOCA	IV (PA)
	2) Small Break LOCA	III (PA)
15.7	**RADIOACTIVE RELEASE FROM A SUBSYSTEM OR COMPONENT**	
15.7.1	Radioactive Waste Gas System Leak or Failure	III (PA)
15.7.2	Liquid Waste System Leak or Failure	III (PA)
15.7.3	Postulated Radioactive Releases Due to Liquid Tank Failure	III (PA)
15.7.4	Fuel Handling Accidents	IV (PA)
15.7.5	Spent Fuel Cask Drop Accidents	Not Credible/III

TABLE 15.0.3-5

COMPONENT RESPONSE TIME, SETPOINT, AND CAPACITY UTILIZED IN AREVA ANALYSES

Item	Response Time	Nominal Setpoint	Setpoint Uncertainty	Total Capacity
Pressurizer Safety Valves (3) loop seal clearing time	1.13 sec	2485 psig	1. +2%/-1%[a]	minimum required: 380,000 lbm/hr each valve of saturated steam at valve Setpoint
Pressurizer Power Operated Relief Valves				
Compensated PORV (1)	2 sec	100 psid		236,000 lbm/hr at 2155 psia and saturation temperature
Non-Compensated PORVs (2)	2 sec	2335 psig		
Steam Line Relief Valves (15)	---			
Group 1		1170.0 psig	±1%[b]	881,980 lbm/hr[c]
Group 2		1185.0 psig	±1%[b]	893,160 lbm/hr[c]
Group 3		1200.0 psig	±1%[b]	904,330 lbm/hr[c]
Group 4		1215.0 psig	±1%[b]	915,500 lbm/hr[c]
Group 5		1230.0 psig	±1%[b]	926,670 lbm/hr[c]
Pressurizer Sprays	---			
Spray Initiates		2260 psig		350 gpm/valve at 2485 psig and 650°F
Full On		2310 psig		
Main Steam Isolation Valves				
Signal Delay	2 sec			
Closing Time	5 sec			
Main Feedwater Isolation Valves				
Signal Delay	2 sec			
Close Time	8 sec			
Auxiliary Feedwater	61.5 sec[d]			390 gpm[e]
Auxiliary Feedwater Isolation Valves[f]	41 sec			

Table 15.0.3-5 (Continued)

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[a] The 2% uncertainty associated with the pressurizer safety valve setpoint includes a ±1% tolerance (Tech. Spec. 3.4.2.2), and a +1% set pressure shift.
[b] Safety analyses support operating with an MSSV setpoint tolerance of ±3%. See Section 15.0 for more details.
[c] Capacity at 3% accumulation
[d] Maximum response time.
[e] Safety analyese support an AFW flow rate of 374 gpm at the lowest lifting MSSV setpoint plus 3% tolerance. See Section 15.0 for more details.
[f] Values used in Westinghouse analysis of Steam Generator Tube Rupture are slightly different. (See Sections 15.0 and 15.6.3).

TABLE 15.0.3-6

CORE POWER DISTRIBUTION UTILIZED IN AREVA ANALYSES

$F_{\Delta H}$ Limit (including uncertainties)	1.66
Measurement Uncertainty	1.04
Fraction of Power Deposited in the Fuel	0.974
F_Q Limit (including uncertainties)	2.41
Measurement Uncertainty	1.05
Engineering Tolerance Uncertainty	1.03

TABLE 15.0.6-2 TRIP SETPOINTS AND TIME DELAYS TO TRIP ASSUMED IN AREVA ACCIDENT ANALYSES

Item	Tech. Spec. Trip Setpoint	Tech. Spec. Total Allowance	Analysis Value	Response Time[a]
Power Range, Neutron Flux				
High Setting	≤ 108% of RTP	+ (0.0583) (120% of RTP)	115% of RTP	≤ 0.5 sec
Low Setting	≤ 25% of RTP	+ (0.0783) (120% of RTP)	34.4% of RTP	
High Negative Flux Rate	≤ 5% of RTP with a time constant ≥ 2 sec	- (0.0233) (120% of RTP)	-7.8% of RTP with a time constant ≥ 2 sec	≤ 0.5 sec
High Pressurizer Water Level[b]	≤ 87% of instrument span	+ (0.08)(100%)	95% of instrument span	≤ 2.0 sec
High Pressurizer Pressure	≤ 2385 psig	+ (0.075) (800 psi)	2445 psig	≤ 2.0 sec
Low Pressurizer Pressure	≥ 1960 psig	- (0.05) (800 psi)	1920 psig	≤ 2.0 sec
Lead Time Constant, τ_4	2.0 sec		1.8 sec[c]	
Lag Time Constant, τ_5	1.0 Sec		1.1 sec[(c)]	
Low Primary Coolant Flow	≥ 90.5% of full flow	- (0.0458) (120%)	85% of full flow	≤ 1.0 sec
Low-Low Steam Generator Level	≥ 25.0% of Narrow Range Span	- (0.089) (100%) - (0.25%) (100%)	16.1% span[d] 0.0 span[e]	≤ 3.5 sec
Undervoltage – Reactor Coolant Pumps[f]	≥ 5148 volts			≤1.5 sec
Underfrequency – Reactor Coolant Pumps	≥ 57.5 Hz	- (0.05) (10Hz)	57.0 Hz	≤0.6 sec
Turbine Trip and Main Feedwater Isolation on SG Water Level – High-High	≤ 78.0% of Narrow Range Span	+ (0.22) (100%)	100% span	≤ 2.5 sec (Turbine Trip) ≤ 10 sec (Feedwater Isolation)
Over Temperature ΔT				
ΔT_0			$\Delta T/\Delta T_o = 1.0$	4.75 sec RTD lag time and 5.65 sec delay
K_1	1.185	+ (0.09) (150%)	1.32	
K_2	0.224/°F		0.224/°F	
K_3	0.0012/psig		0.0012/psig	
τ_1	0.0 sec		0.0 sec	
τ_2	0.0 sec		0.0 sec	
τ_3	4.0 sec		4.4 sec	
τ_4	22.0 sec	-10%	19.8 sec	
τ_5	4.0 sec	+10%	4.4 sec	

TABLE 15.0.6-2 TRIP SETPOINTS AND TIME DELAYS TO TRIP ASSUMED IN AREVA ACCIDENT ANALYSES

Item	Tech. Spec. Trip Setpoint	Tech. Spec. Total Allowance	Analysis Value
τ_6	0.0 sec		0.0 sec
f_1 (ΔI)	0 when +12% $\geq \Delta I \geq$ -21.6% 1.75% per neg. % ΔI 1.50% per pos. % ΔI		0 when +12% $\geq \Delta I \geq$ -21.6% 1.75% per neg. % ΔI 1.50% per pos. % ΔI
Over Power ΔT			$\Delta T/\Delta T_o$All Chapters = 1.0
ΔT_0			
K_4	1.12	+ (0.040) (150%)	1.18
K_5	0.02/°F for increasing average temperature 0.0 for decreasing average temperature		0.02/°F for increasing average temperature 0.0 for decreasing average temperature
K_6	0.002/°F for T > T" 0.0 for T $\leq$ T"		0.002/°F for T > T" 0.0 for T $\leq$ T"
τ_7	13.0 sec		11.7 sec
f_2 (ΔI)	0.0 for all ΔI		0.0 for all ΔI

[a] The Reactor Trip System Response Time is defined as the time interval from when the monitored parameter exceeds its trip setpoint at the channel sensor until loss of stationary gripper coil voltage.
[b] The high pressurizer water level trip is only used in the Section 15.2.3 turbine trip overpressure analyses performed by Duke Energy
[c] A specific undervoltage setpoint was not assumed in the analysis.
[d] For loss of normal feedwater.
[e] For feedwater or main steamline break.
[f] A specific undervoltage setpoint was not assumed in the analysis

TABLE 15.0.8-1 PLANT SYSTEMS AND EQUIPMENT AVAILABLE FOR TRANSIENT AND ACCIDENT CONDITIONS

Incident	Reactor Trip Functions	ESF Actuation Functions	Other Equipment	ESF Equipment
15.1 Increase in Heat Removed by the Secondary System				
Feedwater system malfunctions that result in an increase in feedwater flow	Power range high flux, high-high steam generator level, manual	High-high steam generator level-produced feed-water isolation and turbine trip	Feedwater isolation valves	-
Excessive increase in secondary steam flow	Power range high flux, over-temperature ΔT, overpower ΔT, manual	-	Pressurizer self-actuated safety valves, steam generator safety valves	-
Inadvertent opening of a steam generator relief or safety valve	Low pressurizer pressure, manual SIS, overpower ΔT, power range high flux	Low pressurizer pressure, low compensated steam line pressure, Hi-1 containment pressure, manual	Feedwater isolation valves, steam line isolation valves	Auxiliary Feedwater System, Safety Injection System
Steam system piping failure	SIS, low pressurizer pressure, manual, overpower ΔT, power range high flux	Low pressurizer pressure, low compensated steam line pressure, Hi-1 and Hi-3 containment pressure, manual	Feedwater isolation valves, steam line pressure, isolation valves	Auxiliary Feedwater System, Safety Injection System, Containment Heat Removal System
15.2 Decrease in Heat Removal by the Secondary System				
Loss of external electrical load/ turbine trip	High pressurizer pressure over-temperature ΔT, manual, steam generator low-low level, high pressurizer water level	-	Pressurizer safety valves, steam generator safety valves	-
Loss of non-emergency AC power to the station auxiliaries	Steam generator low-low level, manual	Steam generator low-low level	Steam generator safety valves	Auxiliary Feedwater System
Loss of normal feedwater flow	Steam generator low-low level, manual	Steam generator low-low level	Steam generator safety valves	Auxiliary Feedwater System

TABLE 15.0.8-1 PLANT SYSTEMS AND EQUIPMENT AVAILABLE FOR TRANSIENT AND ACCIDENT CONDITIONS

Incident	Reactor Trip Functions	ESF Actuation Functions	Other Equipment	ESF Equipment
Feedwater system pipe break	Steam generator low-low level, high pressurizer pressure, SIS, over-temperature ΔT, manual	High Containment pressure, steam generator low-low water level, low compensated steam line pressure High steamline differential pressure	Steam line isolation valves, feedline isolation, pressurizer self-actuated safety valves steam generator safety valves	Auxiliary Feedwater System Safety Injection System
15.3 Decrease in Reactor Coolant System Flow Rate				
Partial & complete loss of forced reactor coolant flow	Low flow, undervoltage underfrequency, manual	-	Steam generator safety valves	-
Reactor coolant pump shaft seizure (locked rotor)	Low flow, manual	-	Pressurizer safety valves, steam generator safety valves	-
15.4 Reactivity & Power Distribution Anomalies				
Uncontrolled rod cluster control assembly bank withdrawal from a subcritical or low power startup condition	Power range high flux, manual, source range high flux, high flux rate	-	-	-
Uncontrolled rod cluster control assembly bank withdrawal at power	Power range high Overtemperature ΔT, high pressurizer pressure, manual, high pressurizer level, overpower ΔT	-	Pressurizer safety valves, steam generator safety valves	-
Rod cluster control assembly misalignment	Power range negative flux rate, overtemperature ΔT, manual	-	-	-
Start up of an inactive reactor coolant loop at an incorrect temperature	Low flow interlocked with P-8, manual	-	-	-
Chemical & Volume Control System malfunction that results in a decrease in boron concentration in the reactor coolant	Source range high flux, power range high flux, overtemperature ΔT, manual	-	Low insertion limit annunciators for boration, Source Range count rate (while shut down)	-
Spectrum of rod cluster control	Power range high flux, high positive	-	-	-

TABLE 15.0.8-1 PLANT SYSTEMS AND EQUIPMENT AVAILABLE FOR TRANSIENT AND ACCIDENT CONDITIONS

Incident	Reactor Trip Functions	ESF Actuation Functions	Other Equipment	ESF Equipment
assembly ejection accidents	flux rate, manual			
15.5 Increase in Reactor Coolant Inventory				
Inadvertent operation of the ECCS during power operation	Low pressurizer pressure, manual safety injection trip	-	-	Safety Injection System
15.6 Decrease in Reactor Coolant Inventory				
Inadvertent opening of a pressurizer safety or relief valve	Pressurizer low pressure, overtemperature ΔT, manual	-	-	-
Steam generator tube failure	Low pressurizer pressure, overtemperature ΔT (See Section 15.6.3)	Engineered Safety Features Actuation System	Service Water System, Component Cooling Water System, steam generator shell side fluid operating system, steam generator safety and/or relief valves, steam line stop valves, pressurizer relief valves (PORV's).	Emergency Core Cooling System, Auxiliary Feedwater System, Emergency Power System
Loss of coolant accidents resulting from the spectrum of postulated piping breaks within the reactor coolant pressure boundary	Reactor Trip System	Engineered Safety Features Actuation System	Service Water System, Component Cooling Water System, steam generator safety and/or relief valves	Emergency Core Cooling System, Auxiliary Feedwater System, Containment Heat Removal System, Emergency Power System

TABLE 15.0.9-1

CORE TOTAL FISSION PRODUCT ACTIVITIES
Based on 2958 MWt

Isotope	Activity (Ci)	Isotope	Activity (Ci)
I-131	8.02E+07	Cs-134	1.53E+07
I-132	1.16E+08	Cs-136	4.27E+06
I-133	1.64E+08	Cs-137	9.17E+06
I-134	1.80E+08	Rb-86	1.78E+05
I-135	1.53E+08		
		Ru-103	1.22E+08
Kr-85	8.62E+05	Ru-105	8.39E+07
Kr-85m	2.19E+07	Ru-106	4.15E+07
Kr-87	4.22E+07	Rh-105	7.66E+07
Kr-88	5.95E+07	Mo-99	1.47E+08
Xe-131m	8.96E+05	Tc-99m	1.29E+08
Xe-133	1.60E+08		
Xe-133m	5.12E+06	Y-90	7.14E+06
Xe-135	3.83E+07	Y-91	1.04E+08
Xe-135m	3.21E+07	Y-92	1.08E+08
Xe-138	1.37E+08	Y-93	1.24E+08
		Nb-95	1.38E+08
Te-127	8.45E+06	Zr-95	1.37E+08
Te-127m	1.10E+06	Zr-97	1.36E+08
Te-129	2.53E+07	La-140	1.50E+08
Te-129m	3.76E+06	La-141	1.34E+08
Te-131m	1.16E+07	La-142	1.30E+08
Te-132	1.14E+08	Nd-147	5.38E+07
Sb-127	8.55E+06	Pr-143	1.22E+08
Sb-129	2.57E+07	Am-241	1.06E+04
		Cm-242	3.44E+06
Ce-141	1.35E+08	Cm-244	3.21E+05
Ce-143	1.25E+08		
Ce-144	1.01E+08	Sr-89	8.10E+07
Pu-238	2.58E+05	Sr-90	6.82E+06
Pu-239	2.38E+04	Sr-91	9.97E+07
Pu-240	3.26E+04	Sr-92	1.07E+08
Pu-241	1.02E+07	Ba-139	1.47E+08
Np-239	1.57E+09	Ba-140	1.42E+08

TABLE 15.0.9-2

RCS Coolant Fission Product Concentrations
Based on 1% Fuel Defects

Isotope	Activity (μCi/gm)
I-131	1.71
I-132	2.47
I-133	7.234
I-134	5.67E-01
I-135	1.84
Kr-85m	1.73
Kr-85	1.06E+01
Kr-87	1.10
Kr-88	3.21
Xe-131m	3.41
Xe-133m	4.86
Xe-133	2.76E+02
Xe-135m	4.36E-01
Xe-135	8.52
Xe-138	6.30E-01
Cs-134	1.55
Cs-136	3.21
Cs-137	1.61
Rb-86	1.97E-02

These fission product concentrations are based on projected RCS concentrations based on 1% fuel failure. They may be converted to Dose Equivalent I-131 using the dose conversion factors in ICRP-30 (Reference 15.0.9-12). For those events whose initial conditions are based on RCS Iodine concentrations at the Technical Specification Dose Equivalent I-131 limits (instead of the 1% fuel failure initial condition), see Table 15.0.9-7.

TABLE 15.0.9-3

Committed Effective Dose Equivalent Dose Conversion Factors

Isotope	DCF (rem/curie)	Isotope	DCF (rem/curie)
I-131	3.29E4	Cs-134	4.62E4
I-132	3.81E2	Cs-136	7.33E3
I-133	5.85E3	Cs-137	3.19E4
I-134	1.31E2	Rb-86	6.63E3
I-135	1.23E3		
		Ru-103	8.95E3
Kr-83m	N/A	Ru-105	4.55E2
Kr-85m	N/A	Ru-106	4.77E5
Kr-85	N/A	Rh-105	9.56E2
Kr-87	N/A	Mo-99	3.96E3
Kr-88	N/A	Tc-99m	3.3E1
Xe-131m	N/A		
Xe-133m	N/A	Y-90	8.44E3
Xe-133	N/A	Y-91	4.89E4
Xe-135m	N/A	Y-92	7.80E2
Xe-135	N/A	Y-93	2.15E3
Xe-138	N/A	Nb-95	5.81E3
		Zr-95	2.37E4
Te-127	3.18E2	Zr-97	4.33E3
Te-127m	2.15E4	La-140	4.85E3
Te-129m	2.39E4	La-141	5.81E2
Te-129	9.0E1	La-142	2.53E2
Te-131m	6.4E3	Nd-147	6.85E3
Te-132	9.44E3	Pr-143	1.09E4
Sb-127	6.04E3	Am-241	4.44E8
Sb-129	6.44E2	Cm-242	1.73E7
		Cm-244	2.48E8
Ce-141	8.96E3		
Ce-143	3.39E3	Sr-89	4.14E4
Ce-144	3.74E5	Sr-90	1.3E6
Pu-238	3.92E8	Sr-91	1.66E3
Pu-239	4.3E8	Sr-92	8.1E2
Pu-240	4.3E8	Ba-139	1.7E2
Pu-241	8.26E6	Ba-140	3.74E3
Np-239	2.51E3		

TABLE 15.0.9-4

Effective Dose Equivalent Dose Conversation Factors

Isotope	DCF (rem-m^3/Ci-sec)	Isotope	DCF (rem-m^3/Ci-sec)
I-131	6.734E-2	Cs-134	0.2801
I-132	0.4144	Cs-136	0.3922
I-133	0.1088	Cs-137	0.1066
I-134	0.4810	Rb-86	1.780E-2
I-135	0.2953		
Kr-83m	5.550E-6	Ru-103	8.325E-2
Kr-85m	2.768E-2	Ru-105	0.1410
Kr-85	4.403E-4	Ru-106	0.0
Kr-87	0.1524	Rh-105	1.376E-2
Kr-88	0.3774	Mo-99	2.694E-2
Xe-131m	1.439E-3	Tc-99m	2.179E-2
Xe-133m	5.069E-3		
Xe-133	5.772E-3	Y-90	7.030E-4
Xe-135m	7.548E-2	Y-91	9.620E-4
Xe-135	4.403E-2	Y-92	4.810E-2
Xe-138	0.2135	Y-93	1.776E-2
		Nb-95	0.1384
Te-127	8.954E-4	Zr-95	0.1332
Te-127m	5.439E-4	Zr-97	3.337E-2
Te-129m	5.735E-3	La-140	0.4329
Te-129	1.018E-2	La-141	8.843E-3
Te-131m	0.2594	La-142	0.5328
Te-132	3.811E-2	Nd-147	2.290E-2
Sb-127	0.1232	Pr-143	7.770E-5
Sb-129	0.2642	Am-241	3.027E-3
		Cm-242	2.105E-5
Ce-141	1.269E-2	Cm-244	1.817E-5
Ce-143	4.773E-2		
Ce-144	3.156E-3	Sr-89	2.860E-4
Pu-238	1.806E-5	Sr-90	2.786E-5
Pu-239	1.569E-5	Sr-91	0.1277
Pu-240	1.758E-5	Sr-92	0.2512
Pu-241	2.683E-7	Ba-139	8.029E-3
Np-239	2.845E-2	Ba-140	3.175E-2

TABLE 15.0.9-5

Nuclide Decay Constants

Isotope	Decay Constant (hr^{-1})	Isotope	Decay Constant (hr^{-1})
I-131	0.00359	Cs-134	3.84E-5
I-132	0.303	Cs-136	2.2E-3
I-133	0.0333	Cs-137	2.64E-6
I-134	0.791	Rb-86	1.55E-3
I-135	0.105		
Kr-83m	0.379	Ru-103	7.35E-4
Kr-85m	0.155	Ru-105	0.156
Kr-85	7.37E-6	Ru-106	7.84E-5
Kr-87	0.547	Rh-105	1.96E-2
Kr-88	0.248	Mo-99	1.05E-2
Xe-131m	0.00241	Tc-99m	0.115
Xe-133m	0.0130		
Xe-133	0.00546	Y-90	1.08E-2
Xe-135m	2.72	Y-91	4.94E-4
Xe-135	0.0756	Y-92	0.196
Xe-138	2.93	Y-93	0.0686
		Nb-95	8.22E-4
Te-127	7.41E-2	Zr-95	4.51E-4
Te-127m	2.65E-4	Zr-97	4.1E-2
Te-129m	8.6E-4	La-140	1.72E-2
Te-129	0.598	La-141	0.176
Te-131m	2.31E-2	La-142	0.45
Te-132	8.86E-3	Nd-147	2.63E-3
Sb-127	7.5E-3	Pr-143	2.13E-3
Sb-129	0.16	Am-241	1.83E-7
		Cm-242	1.77E-4
Ce-141	8.89E-4	Cm-244	4.37E-6
Ce-143	0.021		
Ce-144	1.02E-4	Sr-89	5.72E-4
Pu-238	9.02E-7	Sr-90	2.72E-6
Pu-239	3.29E-9	Sr-91	0.073
Pu-240	1.21E-8	Sr-92	0.256
Pu-241	5.5E-6	Ba-139	0.502
Np-239	0.0123	Ba-140	2.27E-3

Table 15.0.9-6

Iodine Spike Appearance Rates (Curies/Minute)
Based on 1 μCi/gm of D.E. I-131 Primary Coolant Activity

	I-131	I-132	I-133	I-134	I-135
335 times the equilibrium rate	127.6	422.4	608.4	186.3	197.3
500 times the equilibrium rate	190.5	657.0	915.0	294.0	301.0

Table 15.0.9-7

Iodine Specific Activities (μCi/gm)
Primary Coolant Based on 1.0 and 60.0 μCi/gm of D.E. I-131
Secondary Coolant Based on 0.1 μCi/gm of D.E. I-131

	Primary Coolant		Secondary Coolant
Nuclide	1 μCi/gm	60 μCi/gm	0.1 μCi/gm
I-131	0.570	34.20	0.0570
I-132	0.823	49.38	0.0823
I-133	2.408	144.48	0.2408
I-134	0.189	11.34	0.0189
I-135	0.613	36.78	0.0613

TABLE 15.0.13-1

SINGLE FAILURES ASSUMED FOR ACCIDENTS OF MODERATE FREQUENCY

Event Description	Section	Worst Failure Assumed	Effect
Feedwater temperature reduction	15.1.1	(1)	none
Excessive feedwater flow	15.1.2	One protection train	none
Excessive steam flow	15.1.3	(1)	none
Inadvertent secondary depressurization	15.1.4	One Safety injection train	delays boron to core
Loss of external load	15.2.2	One protection train	none
Turbine trip	15.2.3	One protection train	none
Inadvertent closure of MSIV	15.2.4	One protection train	none
Loss of condenser vacuum	15.2.5	One Protection train	none
Loss of ac power	15.2.6	Loss of two auxiliary feedwater pumps (3) with and without isolation of one SG (4)	increases primary heatup
Loss of normal feedwater	15.2.7	Loss of two auxiliary feedwater pumps (3)	increases primary heatup
Loss of forced reactor coolant flow	15.3.1 & 2	One protection train	none
RCCA bank withdrawal from subcritical	15.4.1	One protection train	none
RCCA bank withdrawal at power	15.4.2	One protection train	none
Dropped RCCA, dropped RCCA bank	15.4.3	One NIS channel	core power is not reduced on power mismatch
Statically misaligned RCCA	15.4.3	(2)	none
Single RCCA withdrawal	15.4.3	One protection train	none
Uncontrolled boron dilution	15.4.6	Malfunction of RMWS flow indication so that actual flow is higher than indicated	Reduces time to criticality
Inadvertent ECCS operation at power	15.5.1	One protection train	none
Increase in RCS inventory	15.5.2	One protection train	none
Inadvertent RCS depressurization	15.6.1	One protection train	none
Failure of small lines carrying primary coolant outside containment	15.6.2	(2)	none

(1) No protective action required.
(2) No transient analysis involved.
(3) The analysis assumes only one AFW pump is available to mitigate the Loss of Feedwater and Loss of AC Power Events even though no single failure has been identified that would result in loss of two AFW pumps. This was done to demonstrate a one-pump capability that was credited as a mechanism to obtain the system reliability criteria required by TMI Action Item II.E.1.1 (see UFSAR Section 10.4.9 Appendix A).
(4) No single failure has been identified that would lead to the combination of AFW pump failures and inadvertent steam generator isolation that is assumed in this analysis. The combination of failures in the analysis was simply a matter of analytical expediency and does not constitute a legitimate licensing basis system configuration (see discussion in FSAR Section 15.0.13.a.9).

TABLE 15.0.13-2

SINGLE FAILURES FOR NON CONDITION II EVENTS

Event Description	Section	Worst Failure Assumed	Effect
Single Rod Withdrawal	15.4.3	One Protection Train	None
Inadvertent Fuel Loading	15.4.7	(1) (2)	None
LOCA (Small Break)	15.6.5	Loss of One SI Train	Higher PCT
Gaseous Rad Waste Failure	15.7.1	(1)	None
Liquid Rad Waste Failure	15.7.2	(1)	None
Liquid Tank Failure	15.7.3	(1)	None
Fuel Cask Drop	15.7.5	(1)	None
Steamline Rupture	15.1.5	Loss of One SI Train	Delay Boron to Core
Feedline Rupture	15.2.8	Loss of One Aux Feed Pump with offsite power available (3)	Delay Cooling
Locked Rotor	15.3.3	Loss of One Protection Train	None
Shaft Break	15.3.4	Loss of One Protection Train	None
Rod Ejection	15.4.8	Loss of One Protection Train	None
Steam Generator Tube Rupture	15.6.3	Failed-open PORV on ruptured SG	Increased Break Flow and Increased Steam Release
Fuel Handling Accident	15.7.4	(1)	None

(1) No Protective Action Required.

(2) No Transient Analysis Involved.

(3) See discussions in Section 15.0.13.b and 15.2.8 for further discussion of event-specific failure scenarios.

TABLE 15.1.2-1

INPUT PARAMETERS AND BIASING FOR INCREASE IN FEEDWATER FLOW

Parameter	HFP Cases	HZP Case
Core Power	2958 MW(rated + 0.34%)	10-9 rated
Reactor Coolant System Pressure	Nominal	Nominal
Pressurizer Level	Nominal	Nominal
Core Average Temperature	Nominal at 2958 MW (rated + 0.34%)	Nominal at HZP Power
Reactor Coolant Flow	Tech. Spec. Minimum	Tech. Spec. Minimum
Steam Generator Pressure	Nominal at 2958 MW (rated + 0.34%)	Nominal at HZP Power
Initial Feedwater Flow Rate	Nominal at 2958 MW (rated + 0.34%)	~0
Feedwater Temperature	Nominal	Nominal
Cycle Exposure	BOC & EOC	EOC
Moderator Temperature Coefficient	Tech. Spec. Limits	Tech. Spec. Limit
Doppler Coefficient	0.8 * BOC 0.8 * EOC	0.8 * EOC
Delayed Neutron Fraction	Nominal for Exposure	Nominal
β/ℓ	Nominal for Exposure	Nominal
Reactor Trip Reactivity Insertion	Minimum (bounds the most reactive rod stuck out of the core	Minimum (bounds the most reactive rod stuck out of the core)
Pellet to Clad Heat Transfer Coefficient	Mean	Mean
Rod Position Controller	Auto for BOC Manual & Auto for EOC	Manual
Pressurizer Heaters	Disable	Disable
Pressurizer Spray	Available	Available
Pressurizer PORVs	Available	Available
Main Feedwater	Auto	Auto
Auxiliary Feedwater	Available	Disable

TABLE 15.1.2-2

KEY OPERATING PARAMETERS FOR INCREASE IN FEEDWATER FLOW

Parameter	HFP Value	HZP Value
Initial Reactor Power (MW)	2958.0	2.9E-6
Initial Pressurizer Pressure (psia)	2250	2250
Initial Pressurizer Level (% of level span)	60.0	25.0
Initial T_{avg} (°F)	588.8	557.7
Initial Total RCS Flow Rate (lbm/s)	30390	30351
Initial Steam Generator Pressure (psia)	977	1099
Initial Feedwater Flow Rate per SG (lbm/s)	1210	~0
Feedwater Temperature (°F)	440.0[2]	40

[2] Safety analyses support operating with a reduced main feedwater temperature of 375°F at full power.

TABLE 15.1.2-3

RANGE OF NEUTRONICS PARAMETERS SUPPORTED BY ANALYSIS FOR INCREASE IN FEEDWATER FLOW

Parameter	BOC	EOC
Moderator Temperature Coefficient (pcm/°F)	≤ +5	≥ -50(HFP)
Doppler Coefficient (pcm/°F)	≤ -1.072	≤ -1.256(HFP)
β	0.006146	0.005142
β/ℓ (sec^{-1})	398.5	298.0
Scram Worth (pcm)	≥ 1770	≥ 1770

TABLE 15.1.2-4

EVENT SUMMARY FOR INCREASE IN FEEDWATER FLOW - LIMITING CASE (HZP, EOC, MANUAL ROD CONTROL)

Event	Time (sec)
Initiate Transient (Step Increase in Feedwater Flow)	0.0
High Flux Reactor Trip	18.9
Turbine Trip (on reactor trip)	19.4
MDNBR	21.8
High-High Steam Generator Level Signal (MFW terminated)	47.7

TABLE 15.1.3-1

INPUT PARAMETERS AND BIASING FOR INCREASE IN STEAM FLOW

Parameter	Minimum Feedback Case	Maximum Feedback Case
Core Power	2958 MW (rated + 0.34%)	2958 MW (rated + 0.34%)
Reactor Coolant System Pressure	Nominal	Nominal
Pressurizer Level	Nominal	Nominal
Core Average Temperature	Nominal at 2958 MW (rated + 0.34%)	Nominal at 2958 MW (rated + 0.34%)
Reactor Coolant Flow	Tech. Spec. Minimum	Tech. Spec. Minimum
Steam Generator Pressure	Nominal at 2958 MW (rated + 0.34%)	Nominal at 2958 MW (rated + 0.34%)
Feedwater Flow Rate	Nominal at 2958 MW (rated + 0.34%)	Nominal at 2958 MW (rated + 0.34%)
Feedwater Temperature	Nominal	Nominal
Cycle Exposure	BOC	EOC
Moderator Temperature Coefficient	Tech. Spec. Limit	Tech. Spec. Limit
Doppler Coefficient	0.8 * BOC	0.8 * EOC
Delayed Neutron Fraction	Nominal	Nominal
β/ℓ	Nominal	Nominal
Reactor Trip Reactivity Insertion	Minimum (bounds the most reactive rod stuck out of the core	Minimum (bounds the most reactive rod stuck out of the core)
Pellet to Clad Heat Transfer Coefficient	Mean	Mean
Rod Position Controller	Auto	Auto
Pressurizer Heaters	Disable	Disable
Pressurizer Spray	Available	Available
Pressurizer PORVs	Available	Available
Main Feedwater	Auto	Auto
Auxiliary Feedwater	Available	Available

TABLE 15.1.3-2

KEY OPERATING PARAMETERS FOR INCREASE IN STEAM FLOW

Parameter	Value
Initial Reactor Power (MW)	2958.0
Initial Pressurizer Pressure (psia)	2250
Initial Pressurizer Level (% of level span)	60.0
Initial T_{avg} (°F)	588.8
Initial Total RCS Flow Rate (lbm/s)	30390
Initial Steam Generator Pressure (psia)	977
Initial Feedwater Flow Rate per SG	1212
Feedwater Temperature (°F)	440.0[2]

[2] Safety analyses support operating with a reduced main feedwater temperature of 375°F at full power.

TABLE 15.1.3-3

RANGE OF NEUTRONICS PARAMETERS SUPPORTED
BY ANALYSIS FOR INCREASE IN STEAM FLOW

Parameter	BOC Value	EOC Value
Moderator Temperature Coefficient (pcm/°F)	≤ +5	≥ -50
Doppler Coefficient (pcm/°F)	≤ -0.80	≤ -1.12
β	0.006146	0.005142
β/ℓ (sec^{-1})	398.5	298.0
Scram Worth (pcm)	≥ 1770	≥ 1770

TABLE 15.1.3-4

EVENT SUMMARY FOR INCREASE IN STEAM FLOW
(EXCESS LOAD) - MINIMUM (BOC) FEEDBACK CASE

Event	Time (sec)
10% Step Increase in Steam Flow	0.0
Peak Power	143.0
MDNBR	204.0

TABLE 15.1.5-1

EQUIPMENT REQUIRED FOLLOWING A RUPTURE OF A MAIN STEAM LINE

Short Term (Required for Mitigation of Accident)	Hot Standby	Required for Cooldown
Reactor trip and safeguards actuation channels including sensors, circuitry, and processing equipment (the protection circuits used to trip the reactor on undervoltage, underfrequency, and turbine trip may be excluded).	Auxiliary Feedwater System including pumps, water supply, and system valves and piping (this system must be placed in service to supply water to operable steam generators no later than 10 minutes after the incident).	Steam generator power operated relief valves (can be manually operated locally).
Safety Injection System including the pumps, the refueling water storage tank, the boron injection tank, and the systems valves and piping.	Containment ventilation cooling units.	Control for defeating automatic safety injection actuation during a cooldown and depressurization.
Diesels generators and emergency power distribution equipment.	Capability for obtaining a reactor coolant system sample.	Residual Heat Removal System including pumps, heat exchanger, and system valves and piping necessary to cool and maintain the Reactor Coolant System in a cold shutdown condition.
Emergency Service Water System.		
Containment safeguards cooling equipment.		
Auxiliary Feedwater System including pumps, water supplies, piping and valves.		
AFW Isolation System		

TABLE 15.1.5-3

SEQUENCE OF EVENTS FOR LIMITING MAIN STEAM LINE BREAK MDNBR CASE - HZP WITH OFFSITE POWER WITH THE STUCK ROD

Time(s)	Event
0.0	Reactor at EOC HZP conditions
0.0	Double-ended guillotine break in main steam line occurs
0.0	Full AFW flow to affected steam generator
7.1	MSIVs closed
18.0	Scram worth is overcome by moderator and Doppler feedback (total reactivity > 0 $)
37.1	HHSI pumps at rated speed (37 s delay)
258.6	Borated water has filled SI lines and begins to enter cold legs
260.0	Peak post-scram power reached (817.3 MW)
600.0	Calculation terminated

TABLE 15.1.5-5

Parameters Used in Steam Line Break Radiological Analysis

Reactor coolant noble gas activity prior to accident (% fuel defect level)	1.0
Reactor coolant iodine activity prior to accident (μCi/gm of DE I-131)	
Pre-accident iodine spike	60
Accident-initiated iodine spike (μCi/gm of DE I-131)	1.0
Reactor coolant iodine appearance rate increase due to the accident-initiated spike (times equilibrium rate)	500(Table 15.0.9-6)
Duration of accident-initiated iodine spike (hr)	5.0
Fraction of fuel rods in core assumed to fail for dose considerations (% of core)	1
Radial peaking factor	1.73
Gap Fractions (% of core activity)	
I-131	8
Kr-85	10
Other Iodine and Noble Gas nuclides	5
Alkali Metals	12
SG tube leak rate (gpm total)	1
SG tube leak rate to affected (faulted) SG (gpm)	0.35
SG tube leak rate to unaffected (intact) SGs (gpm)	0.65
Steam release from faulted SG to environment during first two minutes (lbm)	162,000
Time to release initial mass in faulted SG (min)	2
Steam releases from intact SGs (lbm)	
0 - 2 hours	401,000
2 - 8 hours	917,000
> 8 hours	0
Time to cool RCS below 212°F and stop releases from faulted SG (hr)	40
SG iodine water/steam partition coefficient	
Faulted SG	1
Intact SGs	0.01
Iodine chemical form after release to atmosphere (%)	
Elemental	97
Organic	3
Particulate (cesium iodide)	0
RCS mass (lbm)	4.11E5
Intact Secondary SG Side mass (lbm/per SG)	115,585
Faulted SG mass (lbm)	162,000

TABLE 15.1.5-6

Radiological Consequences of a Postulated Main Steam Line Break

For the pre-accident iodine spike:

Exclusion Area Boundary*	0.14 rem TEDE
Low Population Zone	0.15 rem TEDE
Control Room	0.38 rem TEDE

For the accident-initiated iodine spike:

Exclusion Area Boundary*	0.73 rem TEDE
Low Population Zone	1.09 rem TEDE
Control Room	2.58 rem TEDE

For the fuel failure:

Exclusion Area Boundary*	1.51 rem TEDE
Low Population Zone	2.62 rem TEDE
Control Room	4.11 rem TEDE

*The exclusion area boundary doses reported are for the worst two hour period, determined to be from 0.0 to 2.0 hours for the pre-accident iodine spike and for the fuel failure and from 5.0 to 7.0 hours for the accident-initiated iodine spike.

TABLE 15.1.5-7

STEAMLINE BREAK

Required Operator Action	Alarms to Alert the Operator to Initiate a Particular Action	Delay Time Assumed	Instructions Given to the Operator for Performing the Required Action	Components and Instrumentation Necessary to Complete Indicated Action	Impact of Single Active Component Failure	Impact of Operator's Failure to Take Action or the Operator Taking a Closely Related but Erroneous Action
A. Identify the faulted steam generator and isolate auxiliary feedwater to that steam generator.	A. No specific alarms provided for this function. Primary indication to the operator is steamline pressure indication. A possible alarm is the steam flow feed flow mismatch.	A. Within 10 minutes	A. Identify the faulted steam generator by comparing steamline pressures. Terminate auxiliary feedwater to that steam generator by shutting the AFW isolation valves.	A.1 Steamline pressure indicators A.2 Steam Generator AFW isolation valves A.3 Steam generator level indicators	None None None	A. The operator does not isolate AFW to any steam generator or isolates AFW to wrong steam generator; the faulted steam generator will continue to blowdown.
B. The Operator must reset the Safety Injection and manually control the repressurization of RCS and maintain normal pressure control.	B. No specific alarms for this purpose. Primary indications to the operator are: Pressurizer level, Pressurizer Pressure and RCS temperature. Possible alarms include: - High Pressurizer Level - High Pressurizer Pressure.	B. N/A	B. The conditions for resetting Safety Injection are given to the operator. The operator is instructed to manually control the high head SI pumps and re-establish normal pressurizer level control.	B.1 Pressurizer level indicators B.2 Pressurizer pressure indicators B.3 RCS Temperature Indicators B.4 High Head Safety Injection Pumps	None None None None	B.1 The operator fails to modulate SI pumps after the pressurizer level returns to the indicating range: Water relief thru pressurizer relief valves may occur. B.2 The operator stops SI before peak reactivity is reached: If criticality is attained the core power will increase until it reaches equilibrium with the steam demand.

TABLE 15.2.3-1 INPUT PARAMETERS AND BIASING FOR TURBINE TRIP			
Parameter	**Primary Overpressurization Case**	**Secondary Overpressurization Case**	**DNB Case**
Core power	2958 MW (MUR + 0.34%)	2958 MW (MUR + 0.34%)	2958 MW (MUR + 0.34%)
Pressurizer pressure	Nominal - Uncertainty	Nominal + Uncertainty	Nominal
Pressurizer level	75% + Uncertainty	75% + Uncertainty	Nominal
Reactor vessel T_{avg}	Minimum (580.8°F)	Nominal + Uncertainty	Nominal
RCS flow rate	Tech Spec minimum	Design maximum	Tech. Spec. minimum
Steam generator pressure	Low	High	Nominal
Initial feedwater flow rate	Nominal	Nominal	Nominal
Feedwater temperature	Nominal	Nominal	Nominal
Steam generator NR pressure	Nominal	High	Nominal
Cycle exposure	BOC	BOC	BOC
Moderator Temperature Coefficient	0.0 pcm/°F	0.0 pcm/°F	Tech. Spec. limit at 100% power
Doppler coefficient	-0.9 pcm/°F (BOC least negative)	-0.9 pcm/°F (BOC least negative)	0.8* BOC
Delayed neutron fraction, β	Maximum at BOC	Maximum at BOC	Minimum at BOC
β/ℓ	466.67 sec^{-1}	466.67 sec^{-1}	Nominal
Reactor trip reactivity insertion	Minimum (bounds the most reactive rod stuck out of the core)	Minimum (bounds the most reactive rod stuck out of the core)	Minimum (bounds the most reactive rod stuck out of the core)
Pellet-to-cladding heat transfer coefficient	N/A	N/A	Mean
Average core fuel temperature	BOC maximum	BOC maximum	N/A
Steam generator tube plugging	Maximum	Minimum	Nominal
Pressurizer SV Setpoint	PSVs with 3% drift	Nominal+tolerance	Nominal+tolerance
Pressurizer PORV setpoints	Disabled	Nominal	Nominal-tolerance
MSSV setpoints	Banks 1-5 with 3% drift	Banks 1-5 with 3% drift	Nominal+tolerance
Rod position controller	Manual	Manual	Manual
Pressurizer heaters	Available	Available	Available
Pressurizer spray	Disabled	Available	Available

TABLE 15.2.3-1 INPUT PARAMETERS AND BIASING FOR TURBINE TRIP			
Parameter	**Primary Overpressurization Case**	**Secondary Overpressurization Case**	**DNB Case**
Main feedwater	Auto	Auto	Auto
Auxiliary feedwater	Disabled	Disabled	Disabled

TABLE 15.2.3-2

KEY OPERATING PARAMETERS FOR TURBINE TRIP

Parameter	DNB Case Value	Secondary Side Overpressure Case Value	Primary Side Overpressure Case Value
Initial Reactor Power (MW)	2958	2958	2958
Initial Pressurizer Pressure (psia)	2250	2300	2212
Initial Pressurizer Level (% of level span)	60.0	81.75	81.75
Initial Reactor Vessel T_{avg} (°F)	588.8	595.6	580.8
Initial Total RCS Flow Rate (lbm/sec)	30390	32912 (321,000 gpm)	30854 (293,540 gpm)
Initial Steam Generator Pressure (psia)	977	1050	928
Initial Feedwater Flowrate per SG (lbm/sec)	1212	1217.5	1210.8
Feedwater Temperature (°F)	440.0[a]	440.0[a]	440.0[a]

[a] Safety analyses support operating with a reduced main feedwater temperature of 375°F at full power.

TABLE 15.2.3-3

RANGE OF NEUTRONICS PARAMETERS
SUPPORTED BY ANALYSIS FOR TURBINE TRIP

Parameter	Primary Side Overpressure Case and Secondary Side Overpressure Case Value	DNB Case Value
Moderator Temperature Coefficient (pcm/°F)	≤ 0.0	≤ 0.0
Doppler Coefficient (pcm/°F)	≤ -0.90	≤ -0.80
β	≤0.007	0.006
β/ℓ (sec^{-1})	466.67	398.5
Shutdown Margin (pcm)	≥ 1770	≥ 1770

TABLE 15.2.3-4

EVENT SUMMARY FOR TURBINE TRIP PRIMARY SIDE OVERPRESSURIZATION CASE

Event	**Time (sec)**
Turbine trips	0.0
Pressurizer high pressure trip signal reached	5.07
PSVs open	6.9
Reactor trips on pressurizer high pressure (rod motion starts)	7.07
Peak primary pressure at bottom of reactor vessel reached	7.84
PSVs close	9.4
Bank 1 MSSVs open	12.9
Bank 2 MSSVs open	14.0
End of simulation	60.0

TABLE 15.2.3-5

EVENT SUMMARY FOR TURBINE TRIP SECONDARY OVERPRESSURIZATION CASE

Event	**Time (sec)**
Turbine trips	0.0
Pressurizer spray initiates	0.0
Pressurizer compensated and non-compensated PORVs open and cycle	2.4
Bank 1 MSSVs open	5.4
Bank 2 MSSVs open	6.2
Bank 3 MSSVs open	7.3
High pressurizer level trip signal reached	9.34
Bank 4 MSSVs open	10.1
Reactor trips on high pressurizer level (rod motion starts)	11.34
Bank 5 MSSVs open	14.6
Pressurizer non-compensated and compensated PORVs close	15.2
Pressurizer spray terminates	15.4
Peak secondary pressure occurs at bottom of the SG downcorner	17.30
Bank 5 MSSVs close	33.4
Bank 4 MSSVs close	35.3
Bank 3 MSSVs close	38.3
Bank 2 MSSVs close	46.7
AFW on lo-lo SG level	57.9
End of simulation	60.0

TABLE 15.2.3-6

EVENT SUMMARY FOR TURBINE TRIP MDNBR CASE

Event	Time (s)
Turbine trip	0.0
Pressurizer spray on	1.0
Pressurizer compensated PORV open	1.24
Pressurizer uncompensated PORV open	4.8
SG 1st Stage MSSVs open	5.5
SG 2nd Stage MSSVs open	6.6
SG 3rd Stage MSSVs open	8.0
SG 4th Stage MSSVs open	10.4
OTΔT trip	11.55
Scram Initiation	12.80
Time of MDNBR	13.6
SG 5th Stage MSSVs open	13.9
Peak pressurizer level	16.1
Peak SG secondary pressure	18.6

TABLE 15.2.6-1

INPUT PARAMETERS AND BIASING LOSS OF
NON-EMERGENCY AC POWER TO THE STATION AUXILIARIES

Parameter	
Core Power	2958 MW (rated + 0.34%)
Reactor Coolant System Pressure	Nominal
Pressurizer Level	Nominal+Uncertainty
Core Average Temperature	Nominal at 2958 MW (rated + 0.34%)
Reactor Coolant Flow	Tech. Spec. Minimum
Steam Generator Pressure	Nominal at 2958 MW (rated + 0.34%)
Feedwater Flow Rate	Nominal at 2958 MW (rated + 0.34%)
Feedwater Temperature	Nominal
Steam Generator Level	Nominal
Cycle Exposure	BOC
Moderator Temperature Coefficient	Tech. Spec. Limit
Doppler Coefficient	0.8 * BOC
Delayed Neutron Fraction	Nominal BOC
β/ℓ	Nominal BOC
Reactor Trip Reactivity Insertion	Minimum (bounds the most reactive rod stuck out of the core)
Pellet to Clad Heat Transfer Coefficient	Mean
Rod Position Controller	Manual
Pressurizer Heaters	Available
Pressurizer Spray	Available
Pressurizer PORVs	Available
Main Feedwater	Auto until failure to deliver FW flow initiates transient
Auxiliary Feedwater	1 of 2 MDAFW pumps Available TDAFW pump Disable

TABLE 15.2.6-2

KEY OPERATING PARAMETERS FOR LOSS OF
NON-EMERGENCY AC POWER TO THE STATION AUXILIARIES

Parameter	Value
Initial Reactor Power (MW)	2958
Initial Pressurizer Pressure (psia)	2250
Initial Pressurizer Level (% of level span)	66.8
Initial Total T_{avg} (°F)	588.8
Initial Total RCS Flow Rate (lbm/s)	30390
Initial Steam Generator Pressure (psia)	981
Initial Feedwater Flow Rate per SG (lbm/s)	1216
Feedwater Temperature (°F)	440.0[b]

[b] Safety analyses support operating with a reduced main feedwater temperature of 375°F at full power.

TABLE 15.2.6-3

RANGE OF NEUTRONICS PARAMETERS SUPPORTED BY ANALYSIS
FOR LOSS OF NON-EMERGENCY AC POWER TO THE STATION AUXILIARIES

Parameter	BOC Value
Moderator Temperature Coefficient (pcm/°F)	≤ +5
Doppler Coefficient (pcm/°F)	≤ -0.80
β	0.006146
β/ℓ (sec^{-1})	398.5
Scram Worth (pcm)	≥ 1770

TABLE 15.2.6-4

EVENT SUMMARY FOR LOSS OF
NON-EMERGENCY AC POWER TO THE STATION AUXILIARIES

Event	Time (s)
Initiate transient (total loss of MFW)	0.0
Pressurizer PORV opens	12.0
AFW actuation signal on low-low SG level	18.18
Reactor trip on low-low SG level	21.70
Main turbine trip	21.73
RCP trip on loss of offsite power	21.75
MSSVs open	25.0
Maximum pressurizer level	26.5
Maximum post-trip RCS average temperature	28.5
Motor-driven AFW pump starts, blowdown isolated	79.68
Minimum SG liquid inventory	2100.0

TABLE 15.2.6-5

Parameters Used in Loss of AC Power Radiological Analysis

Reactor coolant noble gas and alkali metal activity prior to accident (% fuel defect level)	1.0
Reactor coolant iodine activity prior to accident (μCi/gm of DE I-131)	
Pre-accident iodine spike	60
Accident-initiated iodine spike (μCi/gm of DE I-131)	1.0
Reactor coolant iodine appearance rate increase due to the accident-initiated spike (times equilibrium rate)	500 (Table 15.0.9-6)
Duration of accident-initiated iodine spike (hr)	5.0
Secondary coolant iodine activity prior to accident (μCi/gm of DE I-131)	0.1
Secondary coolant alkali metal activity prior to accident (% of primary concentration)	10
Release Modeling	
SG tube leak rate (gpm total)	1
Steam release to environment (lbm)	
0 - 2 hours	378,000
2 - 8 hours	965,000
> 8 hours	0
SG iodine and alkali metal water/steam partition coefficient	0.01
Iodine chemical form after release to atmosphere (%)	
Elemental	97
Organic	3
Particulate (cesium iodide)	0
RCS mass (lbm)	4.11E5
Secondary Side mass (lbm/per SG)	115,585

TABLE 15.2.6-6

RADIOLOGICAL CONSEQUENCES OF A LOSS OF NON-EMERGENCY AC POWER TO THE PLANT AUXILIARIES

For the pre-accident iodine spike:	
Exclusion Area Boundary*	0.013 rem TEDE
Low Population Zone	0.0096 rem TEDE
Control Room	0.029 rem TEDE
For the accident-initiated iodine spike:	
Exclusion Area Boundary*	0.045 rem TEDE
Low Population Zone	0.023 rem TEDE
Control Room	0.069 rem TEDE

*The exclusion area boundary doses reported are for the worst two hour period, determined to be from 6.0 to 8.0 hours.

TABLE 15.2.7-1

INPUT PARAMETERS AND BIASING LOSS OF NORMAL FEEDWATER FLOW

Parameter	
Core Power	2958 MW (rated + 0.34%)
Reactor Coolant System Pressure	Nominal
Pressurizer Level	Nominal+Uncertainty
Core Average Temperature	Nominal at 2958 MW (rated + 0.34%)
Reactor Coolant Flow	Tech. Spec. Minimum
Steam Generator Pressure	Nominal at 2958 MW (rated + 0.34%)
Feedwater Flow Rate	Nominal at 2958 MW (rated + 0.34%)
Feedwater Temperature	Nominal
Steam Generator Level	Nominal
Cycle Exposure	BOC
Moderator Temperature Coefficient	Tech. Spec. Limit
Doppler Coefficient	0.8 * BOC
Delayed Neutron Fraction	Nominal BOC
β/ℓ	Nominal BOC
Reactor Trip Reactivity Insertion	Minimum (bounds the most reactive rod stuck out of the core)
Pellet to Clad Heat Transfer Coefficient	Mean
Rod Position Controller	Manual
Pressurizer Heaters	Available
Pressurizer Spray	Available
Pressurizer PORVs	Available
Main Feedwater	Auto until failure to deliver FW flow initiates transient
Auxiliary Feedwater	1 of 2 MDAFW pumps Available TDAFW pump Disable
Safety Injection	HHSI Available

TABLE 15.2.7-2

KEY OPERATING PARAMETERS FOR LOSS OF NORMAL FEEDWATER

Parameter	Value
Initial Reactor Power (MW)	2958
Initial Pressurizer Pressure (psia)	2250
Initial Pressurizer Level (% of level span)	66.8
Initial T_{avg} (°F)	588.8
Initial Total RCS Flow Rate (lbm/sec)	30390
Initial Steam Generator Pressure (psia)	982
Initial Feedwater Flow Rate per SG (lbm/s)	1216
Feedwater Temperature (°F)	440.0[b]

[b]Safety analyses support operating with a reduced main feedwater temperature of 375°F at full power.

TABLE 15.2.7-3

RANGE OF NEUTRONICS PARAMETERS
SUPPORTED BY ANALYSIS FOR LOSS OF NORMAL FEEDWATER

Parameter	BOC Value
Moderator Temperature Coefficient (pcm/°F)	$\leq$ +5
Doppler Coefficient (pcm/°F)	$\leq$ -0.80
β	0.006146
β/ℓ (sec^{-1})	398.5
Scram Worth (pcm)	$\geq$ 1770

TABLE 15.2.7-4

EVENT SUMMARY FOR LOSS OF NORMAL FEEDWATER

Event	Time (s)
Initiate transient (total loss of MFW)	0.0
Pressurizer PORV opens	12.0
AFW actuation signal on low-low SG level	18.18
Reactor trip on low-low SG level	21.70
Main turbine trip	21.73
MSSVs open	25.0
Maximum pressurizer level	26.0
Maximum post-trip RCS average temperature	26.5
Pressurizer PORV closes	27.5
Motor-driven AFW pump starts, blowdown isolated	79.68
Minimum SG liquid inventory	2480.0

TABLE 15.2.8-1

INPUT PARAMETERS AND BIASING FOR FEEDWATER LINE BREAK

Parameter	With Loss of Offsite Power	With Offsite Power
Core Power	2958 MW (rated + 0.34%)	2958 MW(rated + 0.34%)
Reactor Coolant System Pressure	Nominal	Nominal
Pressurizer Level	Nominal	Nominal
Core Average Temperature	Nominal at 2958 MW (rated + 0.34%)	Nominal at 2958 MW (rated + 0.34%)
Reactor Coolant Flow	Tech. Spec. Minimum	Tech. Spec. Minimum
Steam Generator Pressure	Nominal at 2958 MW (rated + 0.34%)	Nominal at 2958 MW (rated + 0.34%)
Feedwater Flow Rate	Nominal at 2958 MW (rated + 0.34%)	Nominal at 2958 MW (rated + 0.34%)
Feedwater Temperature	Nominal	Nominal
Steam Generator Level	Nominal	Nominal
Cycle Exposure	EOC	BOC
Moderator Temperature Coefficient	Tech. Spec. Limit	Tech. Spec. Limit
Doppler Coefficient	0.8 * EOC	0.8 * BOC
Delayed Neutron Fraction	Minimum Bounding EOC	Minimum Bounding BOC
β/ℓ	Nominal EOC	Nominal BOC
Reactor Trip Reactivity Insertion	Minimum allowed shutdown margin and the most reactive rod stuck out of the core	Minimum allowed shutdown margin and the most reactive rod stuck out of the core
Pellet to Clad Heat Transfer Coefficient	Mean	Mean
Rod Position Controller	Manual	Manual
Pressurizer Heaters	Disable	Disable
Pressurizer Spray	Disable	Disable
Pressurizer PORVs	Disable	Disable
Main Feedwater	Auto until FWLB initiates	Auto until FWLB initiates
Auxiliary Feedwater	1 pump Available	1 pump Available
Safety Injection	Maximum HHSI Available	Maximum HHSI Available

TABLE 15.2.8-2

KEY OPERATING PARAMETERS FOR FEEDWATER LINE BREAK EVENT

Parameter	Value
Initial Reactor Power (MW)	2958
Initial Pressurizer Pressure (psia)	2250
Initial Pressurizer Level (% of level span)	60
Initial T_{avg} (°F)	588.8
Initial Total RCS Flow Rate (lbm/s)	30390
Initial Steam Generator Pressure per SG (psia)	998
Initial Feedwater Flow Rate per SG (lbm/s)	1212
Feedwater Temperature (°F)	440.0[b]

[b] Safety analyses support operating with a reduced main feedwater temperature of 375°F at full power.

TABLE 15.2.8-3

RANGE OF NEUTRONICS PARAMETERS
SUPPORTED BY ANALYSIS FOR FEEDWATER LINE BREAK EVENT

Parameter	Offsite Power Available Value	Loss of Offsite Power Value
Moderator Temperature Coefficient (pcm/°F)	≤ +5	≤ -50
Doppler Coefficient	≤ -0.80	≤ -1.256
β	0.006146	0.005142
β/ℓ (sec^{-1})	398.5	298.0
Scram Worth (pcm)	≥ 5000	≥ 5000

TABLE 15.2.8-4

EVENT SUMMARY FOR FEEDLINE BREAK WITH OFFSITE POWER AVAILABLE

Time (sec)	Event
0	MFW line break initiated at SG-1
4.8	Low-Low SG liquid level signal
8.3	Reactor trip on Low-Low SG Level signal
8.8	Turbine trip on reactor trip
16	Intact SGs NR liquid level off scale low
54	SIS on low pressurizer pressure
66	AFW flow began to one intact SG
67	Pressurizer drained
83	HHSI flow initiated based on low pressurizer pressure and 29 sec delay
148	SIS actuation on low steam pressure
150	MSIS on low steam pressure; intact SGs isolated from blowdown through ruptured SG
159	Minimum pressurizer pressure (650 psia)
166	Minimum T_{AVG} (457°F), primary system began heatup
191	Pressurizer began to re-fill
198	AFW isolation on high steam pressure differential plus delay
472	Pressurizer SRVs first cycle
473	Maximum RV pressure (2628 psia) observed
662	Pressurizer liquid-full
1800	Operator control of HHSI and AFW assumed (30 minutes)
1800	HHSI terminated by operator due to high pressurizer liquid level
2570	First intact SG MSSV opened
6740	Maximum SG pressure observed (1209 psia)
7000	Transient simulation terminated;
	Gradual primary system cooldown maintained
	RCPs on
	Pressurizer water solid
	HHSI off (operator action)
	SG heat removal capacity exceeds decay heat load

TABLE 15.2.8-5

EVENT SUMMARY FOR FEEDLINE BREAK WITH LOSS OF OFFSITE POWER

Time (sec)	Event
0	MFW line break initiated at SG-1
4.8	Low-Low SG liquid level signal
8.3	Reactor trip on Low-Low SG Level signal
8.8	Turbine trip on reactor trip
16	Intact SGs NR liquid level off scale low
54	SIS on low pressurizer pressure
66	AFW flow began to one intact SG
67	Pressurizer drained
83	HHSI flow initiated based on low pressurizer pressure and 29 sec delay
148	SIS actuation on low steam pressure
150	MSIS on low steam pressure; intact SGs isolated from blowdown through ruptured SG
159	Minimum pressurizer pressure (650 psia)
166	Minimum T_{AVG} (457°F), primary system began heatup
191	Pressurizer began to re-fill
198	AFW isolation on high steam pressure differential plus delay
472	Pressurizer SRVs first cycle
473	Maximum RV pressure (2628 psia) observed
662	Pressurizer liquid-full
984	Loss of offsite power (RCP trip)
1800	Operator control of HHSI and AFW assumed (30 minutes)
1800	HHSI terminated by operator due to high pressurizer liquid level
3020	First intact SG MSSV opened
3294	Maximum SG pressure observed (1210 psia)
7000	Transient simulation terminated;
	Gradual primary system cooldown maintained
	RCPs coasting in natural circulation flow
	Pressurizer water solid
	HHSI off (operator action)
	SG heat removal capacity exceeds decay heat load

TABLE 15.2.8-6

FEEDLINE BREAK

Required Operator Action	Alarms to Alert the operator to initiate a Particular Action	Delay Time Assumed	Instructions Given to the Operator for Performing the Required Action	Components and Instrumentation Necessary to Complete Indicated Action	Impact of Single Active Component Failure	Impact of Operator's Failure to Take Action or the Operator Taking a Closely Related but Erroneous Action
A. The operator controls AFW to the intact steam generators and controls cooldown.	A. The operator will use individual S/G level indicators to control AFW flow to each of the steam generators. Hight level and low level alarms are provided.	Within 30 minutes	B. Maintain proper S/G level in intact S/Gs. If possible, maximize AFW flow to intact S/Gs to help lower primary temperature	B.1 Steam Generator AFW modulation valves and controls.	None	B.1 The operator fails to modulate AFW flows to intact steam generators: Overfilling of a steam generator may occur.
B. The operator controls safety injection	The operator will use pressurizer level indication to prevent/limit liquid discharge through the safety valves.	Within 30 minutes	Limit safety injection and use letdown flow if necessary to prevent liquid-solid pressurizer conditions	HHSI isolation valves	None	Liquid will be discharged from the pressurizer with loss of RCS pressure control.

TABLE 15.3.2-1

INPUT PARAMETERS AND BIASING FOR
LOSS OF FORCED REACTOR COOLANT FLOW

Parameter	
Core Power	2958 MW (rated + 0.34%)
Reactor Coolant System Pressure	Nominal
Core Average Temperature	Nominal at 2958 MW (rated + 0.34%)
Reactor Coolant Flow	Tech. Spec. Minimum
Steam Generator Pressure	Nominal at 2958 MW (rated + 0.34%)
Feedwater Flow Rate	Nominal at 2958 MW (rated + 0.34%)
Feedwater Temperature	Nominal
Cycle Exposure	BOC
Moderator Temperature Coefficient	Tech. Spec. Limit
Doppler Coefficient	0.8 * BOC
Delayed Neutron Fraction	Nominal BOC
β/ℓ	Nominal BOC
Reactor Trip Reactivity Insertion	Minimum allowed shutdown margin and the most reactive rod stuck out of the core
Pellet to Clad Heat Transfer Coefficient	Mean
Rod Position Controller	Manual
Pressurizer Heaters	Disable
Pressurizer Spray	Available
Pressurizer PORVs	Available
Main Feedwater	Auto
Auxiliary Feedwater	Available

TABLE 15.3.2-2

KEY OPERATING PARAMETERS FOR
LOSS OF FORCED REACTOR COOLANT FLOW

Parameter	Value
Initial Reactor Power (MW)	2958
Initial Pressurizer Pressure (psia)	2250
Initial Pressurizer Level (% of level span)	60
Initial Tavg (°F)	588.8
Initial Total RCS Flow Rate (lbm/s)	30390
Initial Steam Generator Pressure	977
Initial Feedwater Flow Rate per SG (lbm/s)	1212
Feedwater Temperature (°F)	440[a]

[a] Safety analyses support operating with reduced main feedwater temperature of 375°F at full power.

TABLE 15.3.2-3

RANGE OF NEUTRONICS PARAMETERS SUPPORTED BY ANALYSIS FOR LOSS OF FORCED REACTOR COOLANT FLOW

Parameter	Value
Moderator Temperature Coefficient (pcm/°F)	≤ +0
Doppler Coefficient (pcm/°F)	≤ -0.80
β	0.006146
β/ℓ (sec^{-1})	398.5
Scram Worth (pcm)	> 5000

TABLE 15.3.2-4

EVENT SUMMARY FOR LOSS OF FORCED
REACTOR COOLANT - UNDERFREQUENCY CASE

Event	Time (sec)
Initiate underfrequency event	0.0
Reactor Scram (Underfrequency)	1.2
Pressurizer PORVs begin to open	2.0
Peak Power Level	2.7
MDNBR	3.0
Peak core average temperature	3.6

TABLE 15.3.3-1

INPUT PARAMETERS AND BIASING FOR LOCKED ROTOR

Parameter	DNB Case	Overpressurization Case
Core Power	2958 MW (rated + 0.34%)	2958 MW(rated + 0.34%)
Reactor Coolant System Pressure	Nominal	Nominal
Pressurizer Level	Nominal	Nominal
Core Average Temperature	Nominal at 2958 MW (rated + 0.34%)	Minimum (580.8°F)
Reactor Coolant Flow	Tech. Spec. Minimum	Tech. Spec. Minimum
Steam Generator Pressure	Nominal at 2958 MW (rated + 0.34%)	Nominal at 2958 MW (rated + 0.34%)
Feedwater Flow Rate	Nominal at 2958 MW (rated + 0.34%)	Nominal at 2958 MW (rated + 0.34%)
Feedwater Temperature	Nominal	Nominal
Cycle Exposure	BOC	BOC
Moderator Temperature Coefficient	Tech. Spec. Limit	Tech. Spec. Limit
Doppler Coefficient	0.8 * BOC	0.8 * BOC
Delayed Neutron Fraction	Nominal BOC	Nominal BOC
β/ℓ	Nominal BOC	Nominal BOC
Reactor Trip Reactivity Insertion	Minimum (Bounds the most reactive rod stuck out of the core)	Minimum (Bounds the most reactive rod stuck out of the core)
Pellet to Clad Heat Transfer Coefficient	Mean	Mean
Rod Position Controller	Manual	Manual
Pressurizer Heaters	Disable	Available
Pressurizer Spray	Available	Disable
Pressurizer PORVs	Available	Disable
Main Feedwater	Auto	Auto
Auxiliary Feedwater	Available	Available

TABLE 15.3.3-2

KEY OPERATING PARAMETERS FOR LOCKED ROTOR

Parameter	DNB T_{avg} = 588.8°F Case Value	Overpressurization T_{avg} = 580.8°F Case Value
Initial Reactor Power (MW)	2958	2958
Initial Pressurizer Pressure (psia)	2250	2250
Initial Pressurizer Level (% of level span)	60	60
Initial T_{avg} (°F)	588.8	580.8
Initial total RCS Flow Rate (lbm/s)	30390	30390
Initial Steam Generator Pressure (psia)	977	905
Initial Feedwater Flow Rate per SG (lbm/s)	1212	1207
Feedwater Temperature (°F)	440[a]	440[a]

[a] Safety analysis support operating with a reduced main feedwater temperature of 375°F at full power.

TABLE 15.3.3-3

RANGE OF NEUTRONICS PARAMETERS
SUPPORTED BY ANALYSIS FOR LOCKED ROTOR

Parameter	Value
Moderator Temperature Coefficient (pcm/°F)	≤ +5
Doppler Coefficient (pcm/°F)	≤ -0.80
β	0.006146
β/ℓ (sec^{-1})	398.5
Scram Worth (pcm)	≥ 1700

TABLE 15.3.3-4

EVENT SUMMARY FOR LOCKED
ROTOR OVERPRESSURIZATION CASE

Event	Time (s)
Initiate transient (seizure of one RCP motor)	0.0
Main turbine trip	0.0
MFW isolated	0.0
Low RCS flow signal	0.125
Reactor scram	1.15
Reverse flow in affected loop	1.75
Peak power	2.50
Remaining RCS pumps tripped	3.00
Pressurizer safety valves open	3.50
Maximum primary side pressure	4.75
Maximum pressurizer level	6.25
Maximum secondary side pressure	12.0

TABLE 15.3.3-5

EVENT SUMMARY FOR LOCKED ROTOR MDNBR CASE

Event	Time (sec)
Initiate transient (seizure of one RCP motor)	0.0
Main turbine trip	0.0
MFW isolated	0.0
Low RCS flow signal	0.125
Reactor scam	1.15
Reverse flow in affected loop	1.75
Peak power	2.50
Remaining RCS pumps tripped	3.00
Peak core-average LHGR	3.25
Minimum DNBR	3.50
Pressurizer safety valves open	3.625
Maximum primary side pressure	3.75
Maximum pressurizer level	6.25
Maximum secondary side pressure	10.0

TABLE 15.3.3-6

PARAMETERS USED IN THE LOCKED ROTOR RADIOLOGICAL ANALYSIS

Source Term	
Core Activity	See Table 15.09-1
Fraction of fuel rods in core assumed to fail for dose considerations	8
Centerline melted fuel (%)	1
Radial peaking factor	1.73
Gap Fractions (% of core activity)	
I-131	8
Kr-85	10
Other Iodine and Noble Gas nuclides	5
Alkali Metals	12
Fraction of activity released from melted fuel (%)	
Primary to secondary leakage	
Iodine	50
Noble Gas	100
Alkali Metals	100
Iodine Chemical form after released to atmosphere (%)	
Elemental	97
Organic	3
Particulate (cesium iodide)	0
Reactor coolant noble gas activity prior to accident (%fuel defect level)	1.0
Secondary coolant iodine activity prior to accident (μCi/gm of DE I-131)	0.1
Secondary coolant alkali metal activity prior to accident (% of primary concentration)	10
Release Modeling	
SG tube leak rate (gpm total)	1
Steam release to environment (lbm)	
0 - 2 hours	378,000
2 - 8 hours	965,000
>8 hours	0
SG iodine and alkali metal water/steam partition coefficient	0.01
RCS mass (lbm)	4.11E5
Secondary Side mass (lbm/per SG)	115,585

TABLE 15.3.3-7

RADIOLOGICAL CONSEQUENCES OF A LOCKED ROTOR

Exclusion Area Boundary*	1.97 rem TEDE
Low Population Zone	1.46 rem TEDE
Control Room	3.31 rem TEDE

* The exclusion area boundary doses reported are for the worst two hour period, determined to be from 6.0 to 8.0 hours.

TABLE 15.4.1-1

INPUT PARAMETERS AND BIASING FOR UNCONTROLLED BANK WITHDRAWAL FROM A SUBCRITICAL OR LOW POWER CONDITION

Parameter	
Core Power	10^{-9} of rated
Reactor Coolant System Pressure	Nominal
Pressurizer Level	Nominal @ HZP
Core Average Temperature	Nominal at HZP
Reactor Coolant Flow	Tech. Spec. Minimum
Steam Generator Pressure	Nominal at HZP
Feedwater Flow Rate	Nominal at HZP
Feedwater Temperature	Nominal
Cycle Exposure	BOC
Moderator Temperature Coefficient	Tech. Spec. Limit
Doppler Coefficient	0.8 * BOC
Delayed Neutron Fraction	Maximum Bounding BOC
β/ℓ	Nominal BOC
Reactor Trip Reactivity Insertion	Minimum (bounds the most reactive rod stuck out of the core
Differential Worth	Bounding Maximum
Pellet to Clad Heat Transfer Coefficient	Bounding Maximum for BOC
Rod Position Controller	Manual
Pressurizer Heaters	Disable
Pressurizer Spray	Available
Pressurizer PORVs	Available
Reactor Coolant Pumps Operational	2

TABLE 15.4.1-2

KEY OPERATING PARAMETERS FOR UNCONTROLLED BANK WITHDRAWAL FROM A SUBCRITICAL OR LOW POWER CONDITION

Parameter	Value
Initial Reactor Power (MW)	2.9E-6
Initial Pressurizer Pressure (psia)	2250
Initial Pressurizer Level (%)	25%
Initial T_{avg} (°F)	557.0
Initial Total RCS Flow Rate (lbm/s)	
Two combined loops with pumps operating	22079
Loop without pump operating	-2597
Initial Steam Generator Pressure (psia)	1093
Initial Steam Flow Rate per SG (lbm/s)	
Loops with pumps operating	3.8
Loop without pump operating	2.6

TABLE 15.4.1-3

RANGE OF NEUTRONICS PARAMETERS SUPPORTED BY ANALYSIS FOR UNCONTROLLED BANK WITHDRAWAL FROM A SUBCRITICAL OR LOW POWER CONDITIONS

Parameter	BOC Value
Moderator Temperature Coefficient (PCM/°f)	≤ +5
Doppler Defect Table from 557°F	≤ -1008 pcm at 1400°F
β	0.007
β/ℓ (sec^{-1})	398.5
Scram Worth (pcm)	≥ 1770
Differential Bank Worth (pcm/step)	≤ 20

TABLE 15.4.1-4

EVENT SUMMARY FOR UNCONTROLLED RCCA BANK WITHDRAWAL FROM A SUBCRITICAL OR LOW POWER STARTUP CONDITIONS

Event	Time (sec)
RCCA Withdrawal Initiated	0.0
Power Range High Neutron Flux Trip (low setting) Setpoint Reached	27.9
Scram Initiated	28.4
Peak Nuclear Power	28.8
MDNBR	30.7
Peak Fuel Centerline Temperature	31.0

TABLE 15.4.2-1

INPUT PARAMETERS AND BIASING FOR UNCONTROLLED BANK WITHDRAWAL AT POWER

Parameter	100% Power Cases	60% Power Cases	10% Power Cases
Core Power	Rated + 0.34%	60% of Rated	10% of Rated
Reactor Coolant System Pressure	Nominal	Nominal	Nominal
Pressurizer Level	Nominal at 100.34% Power	Nominal at 60% Power	Nominal at 10% Power
Core Average Temperature	Nominal at 100.34% Power	Nominal at 60% Power	Nominal at 10% Power
Reactor Coolant Flow	Tech. Spec. Minimum	Tech. Spec. Minimum	Tech. Spec. Minimum
Steam Generator Pressure	Nominal at 100.34% Power	Nominal at 60% Power	Nominal at 10% Power
Feedwater Flow Rate	Nominal	Nominal	Nominal
Feedwater Temperature	Nominal at 100.34% Power	Nominal at 60% Power	Nominal at 10% Power
Cycle Exposure	BOC/EOC	BOC/EOC	BOC/EOC
Moderator Temperature Coefficient	Tech. Spec. Limits for Exposure	Tech. Spec. Limits for Exposure	Tech. Spec. Limits for Exposure
Doppler Coefficient	Bounding least negative (BOC) Bounding most negative (EOC)	Bounding least negative (BOC) Bounding most negative (EOC)	Bounding least negative (BOC) Bounding most negative (EOC)
Delayed Neutron Fraction	Nominal for Exposure	Nominal for Exposure	Nominal for Exposure
β/ℓ	Nominal for Exposure	Nominal for Exposure	Nominal for Exposure
Reactor Trip Reactivity Insertion	Minimum (bounds the most reactive rod stuck out of the core)	Minimum (bounds the most reactive rod stuck out of the core)	Minimum (bounds the most reactive rod stuck out of the core)
Differential Bank Worth	Spectrum	Spectrum	Spectrum
Pellet to Clad Heat Transfer Coefficient	Mean for Exposure	Mean for Exposure	Mean for Exposure
Rod Position Controller	Manual	Manual	Manual
Pressurizer Heaters	Disabled	Disabled	Disabled
Pressurizer Spray	Available	Available	Available
Pressurizer PORVs	Available	Available	Available
Main Feedwater	Auto	Auto	Manual
Auxiliary Feedwater	Available	Available	Available

TABLE 15.4.2-2

KEY OPERATING PARAMETERS FOR
UNCONTROLLED BANK WITHDRAWAL AT POWER

Parameter	100.34% Power Case Value	60% Power Case Value	10% Power Case Value
Initial Reactor Power (MW)	2958	1769	294.8
Initial Pressurizer Pressure (psia)	2250	2250	2250
Initial Pressurizer Level (%)	60	46	28
Initial T_{avg} (°F)	588.8	576.1	559.3
Initial Total RCS Flow Rate (lbm/s)	30419	30428	30410
Initial Steam Generator Pressure (psia)	979	1012	1070
Initial Feedwater Flow Rate per SG (lbm/s)	1214	673	91.9
Feedwater Temperature (°F)	440.0 [a]	380.0	180.0

[a] Safety analyses support operating with a reduced main feedwater temperature at 375°F at full power.

TABLE 15.4.2-3

RANGE OF NEUTRONICS PARAMETERS SUPPORTED BY
ANALYSIS FOR UNCONTROLLED BANK WITHDRAWAL AT POWER

Parameter	BOC Value	EOC Value
Moderator Temperature Coefficient (pcm/°F)	≤ +5	≥ -50
Doppler Coefficient (pcm/°F)	≤ -1.00	≥ -1.97
β	0.006490	0.005289
β/ℓ (sec^{-1})	442.7	279.2
Scram Worth (pcm)	≥ 5000	≥ 5000
Differential Bank Worth (pcm/step)	≤ 41.7	≤41.7

TABLE 15.4.2-4

EVENT SUMMARY FOR UNCONTROLLED RCCA BANK WITHDRAWAL AT POWER LIMITING MDNBR CASE WITH BOC KINETICS AND THE MOST LIMITING REACTIVITY INSERTION RATE AMONG A SPECTRUM OF ANALYZED VALUES.

Event	Time(s)
Bank withdrawal began	0.0
Peak RCS pressure	36.6
OTDT reactor trip setpoint reached	48.47
Indicated core power reached high flux trip setpoint	49.13
Scram occurred	49.64
MDNBR	50.6

TABLE 15.4.3-1

INPUT PARAMETERS AND BIASING FOR ROD DROP

Parameter	Value
Core power	2958 MW (rated + 0.34%)
RCS pressure	Nominal
Pressurizer level	Nominal
Core average temperature	Nominal
RCS flow rate	Tech. Spec. minimum
Steam generator pressure	Nominal
Initial feedwater flow rate	Nominal
Feedwater temperature	Nominal
Steam generator liquid level	Nominal
Cycle exposure	BOC & EOC
Moderator Temperature Coefficient	Tech. Spec. limit
Doppler coefficient	Bounding for exposure
Delayed neutron fraction, β	Nominal for exposure
β/ℓ	Nominal for exposure
U^{238} capture-to-fission ratio	Bounding maximum
Reactor trip reactivity insertion	Minimum allowed shutdown margin and the most reactive rod stuck out of the core
Differential bank worth	Spectrum
Dropped rod worth	Spectrum
Rod shadowing factor	Spectrum
Pellet-to-cladding heat transfer coefficient	Mean
PORV setpoint	Nominal
Pressurizer SRV setpoint	Nominal-tolerance
MSSV setpoint	Nominal+tolerance
Steam generator tube plugging	Maximum
Rod position controller	Manual & Auto
Pressurizer heaters	Disabled
Pressurizer spray	Available
Main feedwater	Auto
Auxiliary feedwater	Available
Turbine runback	Available

TABLE 15.4.3-2

KEY OPERATING PARAMETERS FOR ROD DROP

Parameter	Value
Initial Reactor Power (MW)	2958
Initial Pressurizer Pressure (psia)	2250
Initial Pressurizer Level (% of span)	60
Initial T_{avg} (°F)	588.8
Initial Total RCS Flow Rate (lbm/s)	30390
Initial Steam Generator Pressure (psia)	977
Initial Feedwater Flow Rate per SG (lbm/s)	1211
Feedwater Temperature (F°)	440[(a)]

[(a)] Safety analyses supports operating with a reduced main feedwater temperature of 375°F at full power.

TABLE 15.4.3-3

RANGE OF NEUTRONICS PARAMETERS SUPPORTED BY ANALYSIS FOR ROD DROP

Parameter	BOC Value
Moderator Temperature Coefficient (pcm/°F)	≤ +5
Doppler Coefficient (pcm/°F)	≤ -1.072
β	0.006393
β/ℓ(sec^{-1})	432.8
Scram Worth (pcm)	5842
Differential Bank Worth (ARC Cases Only) (pcm/step)	≤ 10
Dropped Rod Worth (pcm)	30 to 150
Rod Shadowing Factor (ARC Cases Only)	≤ 0.0%

TABLE 15.4.3-4a

EVENT SUMMARY FOR LIMITING DROPPED ROD CASE (BOC,10.0 PCM/STEP DIFFERENTIAL WORTH, 150 PCM DROPPED ROD WORTH, AND 0.0% RSF)

Event	Time (sec)
Initiate rod drop transient	0.0
Maximum pressurizer level	0.0
Maximum core average power	23.0
Maximum core average LHGR	26.0
MDNBR	26.0
Maximum pressurizer pressure	27.5

TABLE 15.4.3-5

RADIOLOGICAL CONSEQUENCES OF SINGLE RCCA WITHDRAWAL

Exclusion Area Boundary*	1.64 rem TEDE
Low Population Zone	1.28 rem TEDE
Control Room	2.74 rem TEDE

*The exclusion area boundary doses reported are for the worst two hour period, determined to be from 6.0 to 8.0 hours.

TABLE 15.4.6 2

ADMINISTRATIVE CONTROLS
TO PREVENT DILUTION DURING REFUELING

VALVE NO.	DESCRIPTION	REQUIRED POSITION DURING REFUELING
1-8455	Reactor Makeup Water to CVCS	Lock closed; may be opened to permit Makeup Control System makeup to Refueling Water Storage Tank provided valves 1-FCV-113B and 1-FCV-114A are maintained closed with their main control board control switches in "shut" position, and manual valves 1-8441, 1-8454 and 1-8439 are locked closed.
1-8308	Boric Acid Batch Tank Outlet	Locked closed; may be opened provided the boron concentration of the boric acid batch tank ≥ *ppmB and valve 1-8302 is closed.
1-8302	Reactor Makeup Water to Boric Acid Batch Tank	Lock closed, may be opened provided valve 1-8308 is closed.
1-8513	Resin Sluice to CVCS Demineralizers	Lock closed.
1-8629A	Boron Recycle Evaporator Feed Pump to Charging/SI Pumps	Lock closed.
1-7054	CVCS Letdown to Boron Thermal Regeneration System	Closed with main control board control switch in "shut" position, and BTRS function selector switch maintained in "off" position; no lock required.
1-7004	Reactor Makeup Water to Boron Thermal Regeneration System	Lock closed.
1-7052	Resin Sluice to BTRS Demineralizers	Lock closed.
1-8545	Boron Thermal Regeneration System Bypass	Opened with main control board control switch maintained in "open" position; no lock required.

* The boron concentration determined by the plant's Technical Specifications for Refueling.

TABLE 15.4.7-1

KEY OPERATING PARAMETERS FOR INADVERTENT LOADING AND OPERATION OF A FUEL ASSEMBLY IN AN IMPROPER POSITION

Parameter	MDNBR Case Value
Initial Reactor Power (MW)	2900
Initial Pressurizer Pressure (psia)	2250
Initial T_{avg} (°F)	588.8°F
Initial Total RCS Flow Rate (lbm/s)	30390

TABLE 15.4.8-1

INPUT PARAMETERS AND BIASING FOR ROD EJECTION ACCIDENTS

Parameter	HFP Cases	HZP Cases
Core Power	2958 MW (rated + 0.34%)	1 KW
Reactor Coolant System Pressure	Nominal	Nominal
Pressurizer Level	Nominal at HFP	Nominal at HZP
Core Average Temperature	Nominal at 2958 MW (rated + 0.34%)	Nominal at HZP
Reactor Coolant Flow	Tech. Spec. Minimum	Tech. Spec. Minimum
Steam Generator Pressure	Nominal at 2958 MW (rated + 0.34%)	Nominal at HZP
Feedwater Flow Rate	Nominal at 2958 MW (rated +0.34%)	Nominal at HZP
Feedwater Temperature	Nominal	Nominal
Cycle Exposure	BOC/EOC	BOC/EOC
Moderator Temperature Coefficient		
BOC	Most Positive T.S.Limit	Most positive T.S. Limit
EOC	Bounding Least Negative	Bounding Least Negative
Doppler Coefficient	0.8 x Nominal Temperature-Dependent Doppler Function	0.8 x Nominal Temperature-Dependent Doppler Function
Delayed Neutron Fraction	Minimum Bounding for Exposure	Minimum Bounding for Exposure
β/ℓ	Nominal for Exposure	Nominal for Exposure
Reactor Trip Reactivity Insertion	Minimum (bounds the most reactive rod stuck out of the core)	Minimum (bounds the most reactive rod stuck out of the core)
Pellet to Clad Heat Transfer Coefficient	Maximum	Maximum
Rod Position Controller	Manual	Manual
Pressurizer Heaters	Disable	Disable
Pressurizer Spray	Available	Available
Pressurizer PORVs	Available	Available
Main Feedwater	Auto	Auto
Auxiliary Feedwater	Available	Available

TABLE 15.4.8-2

KEY OPERATING PARAMETERS FOR ROD EJECTION ACCIDENTS (MDNBR CASES)

Parameter	HFP Value	HZP Value
Initial Reactor Power (MW)	2958	0.001
Initial Pressurizer Pressure (psia)	2250	2250
Initial Pressurizer Level (% of level span)	60	25
Initial T_{avg} (°F)	588.8	557
Initial Total RCSFlow Rate (lbm/s)	30390	30117
Initial Steam Generator Pressure (psia)	999	1094
Initial Feedwater Flow Rate per SG (lbm/s)	1212	~0.0
Feedwater Temperature (°F)	440.0[a]	440.0[a]

[a] Safety analyses support operating with a reduced main feedwater temperature of 375°F at full power.

TABLE 15.4.8-3

RANGE OF NEUTRONICS PARAMETERS SUPPORTED BY ANALYSIS FOR ROD EJECTION ACCIDENTS

Parameter	BOC Value	EOC Value
Moderator Temperature Coefficient (pcm/°F)	≤ +5	≤ 0
Doppler: Defect Table from 557° (BOC) or Coefficient (EOC)	≤ -1008 pcm at 1400°F	≤ -1.12pcm/°F[(a)]
β	0.006	0.005
β/ℓ (sec^{-1})	398.5	298.0
Scram Worth (pcm)	≥ 1770	≥ 1770
Ejected Rod Woth (pcm)	≤ 50 (HFP) ≤ 500 (HZP)	≤ 50 (HFP) ≤ 500 (HZP)

[(a)]The EOC analyses was performed with a constant Doppler temperature coefficient.

TABLE 15.4.8-4B

EVENT SUMMARY FOR CONTROL ROD EJECTION
LIMITING CASE (HZP, EOC KINETICS)

Event	Time(s)
Reactor at HZP, EOC conditions	0.0
RCCA ejection	0.50
RCCA fully ejected	0.60
Peak power	1.90
Power range high neutron flux (low setting) reactor trip	2.20
Turbine trip	2.20
Peak primary pressure	4.3
Maximum core-average LHGR	4.35
Maximum core-average temperature	4.55
Scram rods fully inserted	6.00

TABLE 15.4.8-5

Parameters Used for the Rod Cluster Assembly Ejection Radiological Analysis

Source Team	
Core Activity	See Table 15.0.9-1
Fraction of fuel rods in core that fail (% of core)	4
Gap Fractions (% of core activity)	
Iodine	10
Noble Gas	10
Alkali Metals	12
Fraction of fuel melting (% of core)	2
Radial peaking factor	1.73
Fraction of activity released from melted fuel (%)	
Containment leakage	
Iodine	25
Noble Gas	100
Alkali Metals	100
Primary to secondary leakage	
Iodine	50
Noble Gas	100
Alkali Metals	100
Reactor coolant noble gas and alkali metal activity prior to accident (% fuel defect level)	1.0
Reactor coolant iodine activity prior to accident (μCi/gm of DE I-131)	1.0
Secondary coolant iodine activity prior to accident (μCi/gm of DE I-131)	0.1
Secondary coolant alkali metal activity prior to accident (% of primary concentration)	10
Containment Leakage Release Path	
Containment net free volume (ft^3)	2.344E6
Containment leak rates (weight %/day)	
0 - 24 hours	0.1
> 24 hours	0.05
Iodine chemical form in containment (%)	
Element	4.85
Organic	0.15
Particulate (cesium iodide)	95
Spray removal in containment	Not Credited
Sedimentation removal in containment (hr^{-1})	
Iodines	Not Credited
Alcali metals	0.1
Primary to secondary Leakage Release Path	
SG tube leak rate (gpm total)	1.0
Steam release to environment (lbm)*	
0 - 2 hours	378,000
> 2 hours	0
SG iodine and alkali metal water/steam partition coefficient	0.01
Iodine Chemical form after release to atmosphere (%)	
Elemental	97l
Organic	3
Particulate (cesium iodide)	0

*A separate case was run with secondary steam release continuing for 8 hours (2 - 8 hour steam release = 965,000 lbm) per Reg. Guide 1.183, Appendix H, Section 7 with no containment release. The reported dose consequence results bound the 8 hour secondary release (only) dose consequence.

TABLE 15.4.8-6

Radiological Consequences of a Rod Cluster Control Assembly Ejection Accident

Exclusion Area Boundary*	4.04 rem TEDE
Low Population Zone	4.13 rem TEDE
Control Room	4.60 rem TEDE

*The exclusion area boundary doses reported are for the worst two hour period, determined to be from 0.0 to 2.0 hours.

TABLE 15.5.1-1

ASSUMED STATE OF PLANT SYSTEMS

Parameter	BOC MDNBR	EOC MDNBR	PZR Overfill
Automatic Rod Control	Manual	Auto	Manual
Pressurizer Heaters	Disabled	Disabled	Disabled
Pressurizer Spray	Available	Available	Available
Pressurizer PORVs	Available	Available	Available at first, then assumed to lose motive power
Steam Bypass Valves	Not Modeled	Not Modeled	Not Modeled
Steam Atmospheric Dump Valves	Not Modeled	Not Modeled	Not Modeled
Main Feedwater	Auto (no consequence)	Auto (no consequence)	Auto (no consequence)
Auxiliary Feedwater	Available	Available	Available
Charging/SIPumps	2 Pumps (Maximum Flow)	2 Pumps (Maximum Flow)	2 Pumps (Maximum Flow)
SI Boron Concentration	Maximum per Tech. Specs. (2600 ppm)	Minimum per Tech. Specs. (2400 ppm)	Maximum per Tech. Specs. (2600 ppm)
SI Unborated Purge Volume (total)	23.13 ft^3	23.13 ft^3	23.13 ft^3
SI Fluid Temperature	40°F	40°F	40°F
Letdown Flow	Not Modeled	Not Modeled	Not Modeled
Turbine Control	Auto / Load Demand	Manual	Auto / Load Demand
Cycle Exposure	BOC	EOC	EOC
Initial Core Power	2958 MWT (rated + 0.34%)	2958 MWt (rated + 0.34%)	2958 MWt (rated + 0.34%)
RCS Average Temperature	588.8 °F	588.8 °F	588.8 °F
Steam Generator Tube Plugging Level	3%	3%	3%
RCS Flow	Tech. Spec. minimum (293,540 gpm)	Tech. Spec. minimum (293,540 gpm)	Tech. Spec. minimum (293,540 gpm)

TABLE 15.5.1-2

INPUT BIASING

Parameter	BOC MDNBR Case	EOC MDNBR Case	PZR Overfill Case
Scram Reactivity Worth	Minimum allowed shutdown margin and the most reactive rod stuck out of the core	Minimum allowed shutdown margin and the most reactive rod stuck out of the core	Minimum allowed shutdown margin and the most reactive rod stuck out of the core
High Pressurizer Trip Setpoint	Not Modeled	Nominal + Uncertainty	Nominal + Uncertainty
Low Pressurizer Trip Setpoint	Nominal - Uncertainty	Nominal - Uncertainty	Nominal - Uncertainty
Moderator Temperature Reactivity Coefficient	1.2 X BOC limit	EOC Bounding Minimum	EOC Bounding Minimum
Doppler Reactivity Coefficient	0.8 X BOC	EOC Bounding Minimum	EOC Bounding Minimum
Delayed Neutron Data	Nominal BOC	Nominal EOC	Nominal EOC
Prompt Neutron Lifetime (β/ℓ)	Nominal BOC	Nominal EOC	Nominal EOC
Pressurizer Safety Valve Open Setpoint	Nominal - Tolerance	Nominal - Tolerance	Nominal + Tolerance
Pressurizer Safety Valve Stroke Time	Nominal (Initial lift includes delay for loop seal purge)	Nominal (Initial lift includes delay for loop seal purge)	Nominal (Initial lift includes delay for loop seal purge)
Non-Compensated Pressurizer PORV Open Setpoint	Nominal	Nominal	Nominal
MS Safety Valve Open Setpoint	Nominal + Tolerance	Nominal + Tolerance	Nominal + Tolerance
Initial Pressurizer Level	Nominal	Nominal	Nominal - Uncertainty
Initial Pressurizer Pressure	Nominal	Nominal	Nominal

TABLE 15.5.1-3

SEQUENCE OF EVENTS (BOC MDNBR CASE)

Event	Time (seconds)
Inadvertent actuation of HHSI system (maximum flow from 2 SI pumps)	0.0
Minimum DNBR (a negligible decrease from the initial value)	2.0
Non-borated water cleared from SI lines	25.0
Core power and pressurizer level begin to decrease	27.0
Main turbine valve fully open	50.5
Reactor trip signal on low pressurizer pressure	73.25
Main turbine trip	75.275
Minimum pressurizer pressure (1889.4 psia)	83.0
Minimum pressurizer level (23.4% of span)	83.5
Pressurizer spray actuates at 2260 psia	191.0
Maximum pressurizer level (~98% of span)	600.0
Transient terminated	600.0

TABLE 15.5.1-4

SEQUENCE OF EVENTS (EOC MDNBR CASE)

Event	Time (seconds)
Inadvertent actuation of HHSI system (minimum flow from 2 SI pumps)	0.0
Minimum DNBR (an insignificant decrease from the initial value)	52.5
Non-borated water cleared from SI lines	~71.0
ARC rods begin to step out of core	142.5
ARC banks borated fully out	599.5
Reactor power begins to decrease	600.0
RPS trip signal on high pressurizer level	812.65
Main turbine trip	813.20
Pressurizer outsurge begins	831.0
Pressurizer insurge resumes	856.0
Maximum pressurizer level (~98.8% of span)	1200.0
Transient terminated	1200.0

TABLE 15.5.1-5

SEQUENCE OF EVENTS (PRESSURIZER OVERFILL CASE)

Event	Time (seconds)
Inadvertent actuation of HHSI system: (maximum flow from 2 SI pumps)	0.0
RPS trip signal on SIS signal	2.025
Main turbine trip	2.05
Pressurizer spray actuates	3.2
Compensated pressurizer PORV opens	3.5
Pressurizer spray terminated	6.5
Compensated pressurizer PORV closes	7.5
Minimum pressurizer pressure (2200.1 psia)	8.5
Minimum pressurizer level (52.87% of span)	10.0
Non-borated water cleared from SI lines	~27.0
Compensated PORV opens	578.5
Pressurizer level = 100% span	591.5
Pressurizer liquid level = 463.15 inches	900.0
Pressurizer PORVs assumed disabled	900.0
Pressurizer pressure = 2267 psia	900.0
Pressurizer filled with liquid (level = 463.38 inches)	918.5
Pressurizer SRV opens	945.0
Pressurizer SRV closes	947.0
Pressurizer pressure ranging between 2450 and 2550 psia	950-1200
Minimum SRV inlet temperature = ~ 564°F (while valves are open)	950-1200
Transient terminated	1200.0

TABLE 15.6.1-1

INPUT PARAMETERS AND BIASING FOR INADVERTENT OPENING OF A PRESSURIZER PRESSURE SAFETY OR PORV

Parameter	
Core Power	2958 MW (rated + 0.34%)
Reactor Coolant System Pressure	Nominal
Pressurizer Level	Nominal
Core Average Temperature	Nominal at 2958 MW (rated + 0.34%)
Reactor Coolant Flow	Tech. Spec. Minimum
Steam Generator Pressure	Nominal at 2958 MW (rated + 0.34%)
Feedwater Flow Rate	Nominal at 2958 MW (rated + 0.34%)
Feedwater Temperature	Nominal
Cycle Exposure	BOC
Moderator Temperature Coefficient	Tech. Spec. Limit
Doppler Coefficient	0.8 * Maximum BOC
Delayed Neutron Fraction	Nominal BOC
β/ℓ	Nominal BOC
Reactor Trip Reactivity Insertion	Minimum (bounds the most reactive rod stuck out of the core)
Pellet to Clad Heat Transfer Coefficient	Mean
Rod Position Controller	Manual
Pressurizer Heaters	Disable
Pressurizer Spray	Available
Pressurizer PORVs	Available
Main Feedwater	Auto
Auxiliary Feedwater	Available

TABLE 15.6.1-2

KEY OPERATING PARAMETERS FOR INADVERTENT OPENING OF A PRESSURIZER PRESSURE SAFETY OR PORV

Parameter	Value
Initial Reactor Power (MW)	2958
Initial Pressurizer Pressure (psia)	2250
Initial Pressurizer Level (% of level span)	60
Initial T_{avg} (°F)	588.8
Initial Total RCS Flow Rate (lbm/s)	30390
Initial Steam Generator Pressure	977
Initial Feedwater Flow Rate per SG (lbm/s)	1212
Feedwater Temperature (°F)	440[b]

[b]Safety analyses support operating with a reduced main feedwater temperature of 375°F at full power.

TABLE 15.6.1-3

RANGE OF NEUTRONICS PARAMETERS SUPPORTED BY ANALYSIS FOR INADVERTENT OPENING OF A PRESSURIZER SAFETY OR PORV

Parameter	Value
Moderator Density Table Basis (pcm/°F)	≤ +5
Doppler Coefficient (pcm/°F)	≤ -1.088
β	0.006
β/ℓ (sec^{-1})	398.5
Scram Worth (pcm)	≥ 1770

TABLE 15.6.1-4

EVENT SUMMARY FOR INADVERTENT OPENING OF A PRESSURIZER SAFETY OR PORV

Event	Time (sec)
Pressurizer safety relief valve opened	0.0
Low pressurizer pressure trip reached	22.4
Reactor trip signal generated after 0.2-s delay	24.5
Turbine valve closed	24.5
Average core fuel rod heat flux maximum attained	24.7
Peak core power attained	24.8
Scram rod movement begins after holding-coil release delay period	25.2
MSSVs open	30.0
Peak steam generator pressure attained	32.0
Calculation terminated	50.0

TABLE 15.6.2-1

PARAMETERS USED FOR LETDOWN LINE BREAK OUTSIDE CONTAINMENT RADIOLOGICAL ANALYSIS

Reactor coolant noble gas activity prior to accident (% fuel defect level)	1.0
Reactor coolant iodine activity prior to accident (μCi/gm of DE I-131)	1.0
Reactor coolant iodine appearance rate increase due to the accident-initiated spike (times equilibrium rate)	500 (Table 15.0.9-6)
Letdown line break flow (gpm)	200
Duration of letdown line break (minutes)	30
Break flow flashing fraction	0.4

TABLE 15.6.2-2

RADIOLOGICAL CONSEQUENCES OF A LETDOWN LINE BREAK OUTSIDE CONTAINMENT

Exclusion Area Boundary*	2.34 rem TEDE
Low Population Zone	0.53 rem TEDE
Control Room	1.50 rem TEDE

* The exclusion area boundary doses reported are for the worst two hour period, determined to be from 0.0 to 2.0 hours.

TABLE 15.6.3-1

SHNPP SGTR ANALYSIS
OPERATOR ACTION TIMES FOR MARGIN TO OVERFILL ANALYSIS

Action	Time Intervals
Isolate auxiliary feedwater flow to ruptured SG	Maximum of 8.8 minutes or LOFTTR2 calculated time to reach 30% narrow range level in the ruptured SG
Isolate steam flow from ruptured SG	Maximum of 12 minutes or LOFTTR2 calculated time to reach 30% narrow range level in the ruptured SG
Operator action time to initiate cooldown	5 min
Cooldown	Calculated by LOFTTR2
Operator action time to initiate depressurization	4 min
Depressurization	Calculated by LOFTTR2
Operator action time to initiate SI termination	3 min
SI termination and pressure equalization	Calculated by LOFTTR2

TABLE 15.6.3-2

SHNPP SGTR ANALYSIS
SEQUENCE OF EVENTS
MARGIN TO OVERFILL ANALYSIS

Event	Time (sec)
SG Tube Rupture	0
Reactor Trip	112
Safety Injection	145
AFW Flow to Ruptured SG Isolated	528
Ruptured SG Isolated	720
RCS Cooldown Initiated	1020
RCS Cooldown Terminated	1356
RCS Depressurization Initiated	1598
RCS Depressurization Terminated	1680
SI Terminated	1860
Break Flow Terminated	2700

TABLE 15.6.3-3

OPERATOR ACTION TIMES FOR SGTR OFFSITE DOSE ANALYSIS

Action	Time (min)
Isolate auxiliary feedwater flow to ruptured SG	Maximum of 10 minutes or LOFTTR2 calculated time to reach 30% narrow range level in the ruptured SG
Isolate steam flow from ruptured SG	Maximum of 12 minutes or LOFTTR2 calculated time to reach 30% narrow range level in the ruptured SG
Close block valve to isolate failed open PORV on ruptured S/G	20 min after valve fails to close at the time of isolation of ruptured S/G
Operator action time to initiate cooldown	5 min
Cooldown	Calculated by LOFTTR2
Operator action time to initiate depressurization	4 min
Depressurization	Calculated by LOFTTR2
Operator action time to initiate SI termination	3 min
SI termination and pressure equalization	Calculated by LOFTTR2

TABLE 15.6.3-4

SEQUENCE OF EVENTS
OFFSITE DOSE ANALYSIS

Event	Time (sec)
SG Tube Rupture	0
Reactor Trip	114
Safety Injection	178
AFW Flow to Ruptured SG Isolated	600
Ruptured SG Isolated	720
Ruptured SG PORV Fails Open	722*
Ruptured SG PORV Block Valve Closed	1922
RCS Cooldown Initiated	2224*
RCS Cooldown Terminated	2996
RCS Depressurization Initiated	3236
RCS Depressurization Terminated	3312
SI Terminated	3492
Break Flow Terminated	4652

*The actual times listed above and simulated in the analysis were slightly longer than those obtained using the operator action times of Table 15.6.3 3, due to computer program limitations for modelling operator actions.

TABLE 15.6.3-5

SGTR MASS RELEASE RESULT
TOTAL MASS FLOW (POUNDS)

	Time Period			
	Time Zero to Time of Reactor Trip*	Time of Reactor Trip to Time at which Break Flow is Terminated*	Time at Which Break Flow is Terminated to 2 Hours	2 Hours to Time at Which RCS Reaches RHR In-Service Conditions*
Ruptured SG				
-Condenser	128,300	0	0	0
-Atmosphere	0	138,300	0	35,100,0
-Feedwater	123,400	33,000	0	
Intact SGs				
-Condenser	254,100	0	0	0
-Atmosphere	0	176,900	183,300	862,800
-Feedwater	254,100	292,400	201,800	894,900
Break Flow	4900	163,000	0	0
Flashed Break Flow	830	10,013	0	0

- For dose consequence analysis, reactor trip occurs at 114 seconds; break flow is terminated at 4652 seconds; RHR conditions are reached at 8 hours.

TABLE 15.6.3-6

PARAMETERS USED IN EVALUATING THE RADIOLOGICAL CONSEQUENCES OF A STEAM GENERATOR TUBE RUPTURE

I. Source Data	
A. Total Unit Thermal Output, MWt	2970.4*
B. Total steam generator tube leakage prior to accident, gpm	1.0
C. Reactor coolant iodine activity:	
1. Accident-Initiated Spike	The initial RC iodine activities are presented in Table 15.0.9-7. The iodine appearance rates assumed for the accident-initiated spike are presented in Table 15.0.9-6.
2. Pre-Accident Spike	Primary coolant iodine activities based on 60 µCi/gm of D.E. I-131 are presented on Table 15.0.9-7.
D. Noble Gas Activity	Primary coolant noble gas activities based on 1 percent fuel defects are presented in Table 15.0.9-2. No noble gases are contained in the secondary system.
E. Secondary system initial activity	Dose equivalent of 0.1 µCi/gm of I-131, presented in Table 15.0.9-7.
F. Reactor coolant initial mass, grams	1.73×10^8
G. Steam generator initial mass (each), grams	4.34×10^7
H. Offsite power	Lost at time of reactor trip
I. Primary-to-secondary leakage duration for intact SG, hours	8
J. Species of iodine	97 percent elemental, 3 percent organic
II. Activity Release Data	
A. Ruptured steam generator	
1. Initial Primary-to-secondary leakage, gpm	0.3
2. Ruptured flow flashing fraction	See Figure 15.6.3-19
3. Rupture flow	See Table 15.6.3-5 & Figure 15.6.3-13
4. Flashed rupture flow	See Table 15.6.3-5 & Figure 15.6.3-22
5. Steam releases	See Table 15.6.3-5 & Figure 15.6.3-23 An additional 41,310 lbm/hr to TDAFW pump is modeled until ruptured SG isolation.
6. Iodine partition factor for rupture flow	
Non-flashed	100
Flashed	1.0
B. Intact steam generators	
1. Primary-to-secondary leakage, gpm	0.7
2. Steam releases	See Table 15.6.3-5 & Figure 15.6.3-23
3. Iodine partition factor	100
C. Condenser	
1. Iodine partition factor	100
D. Atmospheric Dispersion Factors	See Table 15.6.3-10

Table 15.6.3-6 (Continued)

Notes:
* Includes 12.4 MWt of NSSS heat generation.
These analyses were done by Westinghouse for the steam generator replacement and power uprate program. Details of the inputs used in the calculations are discussed in the NSSS Licensing Report
(Reference 15.6.5-52). Both the analyses consider core powers up to 2900 MWt with an additional 2% increase applied. Therefore, the increase in the core power to 2958 MWt due to MUR has adequately been accounted for in these analyses. Note that the margin-to-overfill analysis as defined in the FSAR limits the full power T_{avg} value to a minimum of 588.8°F. This limitation continues to apply with the MUR power uprating.

TABLE 15.6.3-10

OFFSITE ATMOSPHERIC DISPERSION FACTORS AND BREATHING RATES

Time (hrs.)	Exclusion Area Boundary* x/Q (Sec/m^3)	Low Population Zone x/Q (Sec/m^3)	Breathing Rate (Sec/m^3) [Ref. 15.6.3-4]
0-2	6.17×10^{-4}	1.4×10^{-4}	3.5×10^{-4}
2-8	-	1.0×10^{-4}	3.5×10^{-4}
8-24	-	1.0×10^{-4}	1.8×10^{-4}
24-96	-	5.9×10^{-5}	2.3×10^{-4}
>96	-	2.4×10^{-5}	2.3×10^{-4}

* The exclusion area boundary atmospheric dispersion factor is conservatively applied during all time intervals in the determination of the limiting two hour period

TABLE 15.6.3-13

RADIOLOGICAL CONSEQUENCES OF A SGTR

Accident Initiated Iodine Spike TEDE Doses	
Exclusion Area Boundary*	1.30
Low Population Zone	0.40
Control Room	0.90
Pre-Accident Iodine Spike TEDE Doses	
Exclusion Area Boundary*	2.20
Low Population Zone	0.60
Control Room	1.60

* The exclusion area boundary doses reported are for the worst two hour period, determined to be from 0.0 to 2.0 hours for both the pre-accident iodine spike and accident-initiated iodine spike cases.

TABLE 15.6.5-1 KEY PARAMETERS FOR LBLOCA		
	Event	Operating Range
1.0	Plant Physical Description	
	1.1 Fuel	
	a) Cladding outside diameter	0.376 in.
	b) Cladding inside diameter	0.328 in.
	c) Cladding thickness	0.024 in.
	d) Pellet outside diameter	0.3215 in.
	e) Pellet density	96 percent of theoretical
	f) Active fuel length	144 in.
	g) $Gd_2 O_3$ concentration	2, 4, 6, 8 w/o
	1.2 RCS	
	a) Flow resistance	Analysis
	b) Pressurizer location	Analysis assumes location giving most limiting PCT (broken loop)
	c) Hot assembly location	Anywhere in core
	d) Hot assembly type	17 x 17
	e) SG tube plugging	≤ 3 percent
2.0	Plant Initial Operating Conditions	
	2.1 Reactor Power	
	a) Nominal reactor power	2958 MWt[1]
	b) F_Q	≤ 2.52[2]
	c) $F_{\Delta H}$	≤ 1.73[3]
	2.2 Fluid Conditions	
	a) Loop flow	109.2 Mlbm/hr ≤ M ≤ 115.3 Mlbm/hr
	b) RCS average temperature	582.0°F ≤ T ≤ 594.8°F
	c) Upper head temperature	~ Tcold Temperature[4]
	d) Pressurizer pressure	2200 psia ≤ P ≤2288 psia
	Event	Operating Range
	e) Pressurizer level	Gaussian distribution mean of 60% and standard deviation of 7.4%. Distribution on the high side up to 92%[a].
	f) Accumulator pressure	599.7 psia ≤ P ≤ 679.7 psia
	g) Accumulator liquid volume	994.6 ft^3 ≤ V ≤ 1029.4 ft^3

[a] The estimated impact to the LBLOCA PCT is 0°F for reducing the sampled pressurizer level high-side limit to 81% to bound the Technical Specitication pressurizer water level LCO of 75% indicated span.

TABLE 15.6.5-1 KEY PARAMETERS FOR LBLOCA

	Event	Operating Range
	h) Accumulator temperature	80°F ≤ T ≤ 130°F (It's coupled with containment temperature)
	i) Accumulator resistance fL/D	As-built piping configuration
	j) Minimum ECCS boron	≥ 2400 ppm
3.0	Accident Boundary Conditions	
	a) Break location	Any RCS piping location
	b) Break type	Double-ended guillotine or split
	c) Break size (each side, relative to cold leg pipe area)	0.26 ≤ A ≤ 1.0 full pipe area (split) 0.26 ≤ A ≤ 1.0 full pipe area (guillotine)
	d) Worst single-failure	Loss of Diesel (one train of ECCS)
	e) Offsite power	On or Off
	f) ECCS pumped injection temperature	125°F
	g) HHSI pump delay	29 s (w/ offsite power) 29 s (w/o offsite power)
	h) LHSI pump delay	29 s (w/ offsite power) 29 s (w/o offsite power)
	i) Containment pressure	14.7 psia, nominal value
	j) Containment temperature	80°F ≤ T ≤ 130°F
	k) Containment sprays delay	0 s
	l) Containment spray water temperature	40°F

[1] Consistent with rated core power of 2948 MWt and 0.34% uncertainty.

[2] The peaking factor includes measurement and engineering uncertainty.

[3] The $F_{\Delta H}$ value is listed in COLR; while 4% measurement uncertainty is a Technical Specifications limit.

[4] Upper head temperature will change based on sampling of RCS temperature.

TABLE 15.6.5-2

EVENT TIMES FOR LBLOCA

Event	Time(s)
Break Opened	0.0
RCP Trip	N/A
SIAS Issued	0.4
Start of Broken Loop Accumulator Injection	8.8
Start of Intact Loop Accumulator Injection (Loops 2 and 3 respectively)	12.5 and 12.5
Beginning of Core Recovery (Beginning of Reflood)	25.4
Broken Loop HHSI Delivery Began	29.4
Intact Loop HHSI Delivery Began (Loop 2 and 3 respectively)	29.4 and 29.4
LHSI Available	29.4
Broken Loop LHSI Delivery Began	29.4
Intact Loop LHSI Delivery Began (Loop 2 and 3 respectively)	29.4 and 29.4
Broken Loop Accumulator Emptied	35.0
Intact Loop Accumulators Emptied (Loops 2, and 3 respectively)	37.0 and 35.9
PCT Occurred (1935°F)	131.3
Transient Calculation Terminated	829.7

TABLE 15.6.5-3

SBLOCA SYSTEM ANALYSIS PARAMETERS

Parameter	Value
Reactor Power, MWt	2958[a]
Radial Peaking Factor ($F_{\Delta H}$) (includes uncertainty)	1.73
Total Power Peaking Factor (F_Q) (includes uncertainty)	2.52
RCS Flow Rate (minimum) (gpm)	293540
Pressurizer Pressure (nominal), psia	2250
RCS Operating Temperature (nominal), °F	588.8
Accumulator Pressure (minimum), psia	599.7
Accumulator Fluid Temperature (maximum), °F	130.0
Accumulator Water Volume (nominal), $ft.^3$	1012
SG Tube Plugging, %	3
SG Secondary Pressure (nominal), psia	985
MFW Temperature at 100% RTP (nominal), °F	440.0
AFW Temperature (maximum), °F	120
AFW Pump Delay Time on SIAS (LOOP), sec	61.5
HHSI, and LHSI/RHR Fluid Temperature, °F	125
Pressurizer Pressure - Low Reactor Trip (minimum), psia (includes uncertainty)	1934.7
Reactor Scram Delay on Low Pressurizer Pressure, (maximum) sec	2.0
SIAS Activation Setpoint Pressure (minimum), psia	1714.7
HHSI Pump Delay Time on SIAS (LOOP), sec	29
LHSI Pump Delay Time on SIAS (LOOP), sec	29
MSSV lift pressures (nominal; does not inclde 2%[b] tolerance), psia	1184.7 1199.7 1214.7 1229.7 1244.7

[a] Includes 0.34% (10 MWt) uncertainty.
[b] Safety analyses support operating with an MSSV setpoint tolerance of ±3%. See Section 15.0 for more details.

Table 15.6.5-3 (Continued)

HHSI Flow		LHSI Flow		
RCS Cold Leg Pressure (psia)	Flow per Loop (gpm)	RCS Cold Leg Pressure (psia)	Intact Loop Flow per Loop (gpm)	Broken Loop Flow (gpm)
0.00	165.82	0.00	924.49	1813.98
15.00	165.82	25.37	924.49	1813.98
398.83	151.94	35.43	888.15	1741.90
646.81	141.37	50.07	833.24	1632.95
829.46	134.06	70.06	753.65	1475.03
1012.11	126.52	90.03	655.05	1278.47
1142.57	120.60	100.02	600.22	1169.09
1390.45	108.60	105.21	569.63	1107.78
1638.33	95.47	110.07	539.84	1048.26
1755.74	88.49	115.72	503.46	973.90
1886.20	80.36	131.51	356.47	657.00
2003.62	70.44	142.79	108.86	137.47
2134.08	57.37	144.79	36.94	44.67
2251.50	42.19	144.89	0.00	0.00
2378.23	20.30			
2378.33	0.00			

TABLE 15.6.5-11a

SEQUENCE OF EVENTS DURING SBLOCA (LIMITING CASE)

Event Description for: Limiting Break Diameter = 2.60 inches (Limiting Break area = 0.03687 ft^{2})	Time (sec)
Break open	0.0
Low PZR Pressure Trip	19.6
RX, LOOP, RCPs, MFWP, and Turbine Trip	21.6
Low PZR Pressure SIAS Setpt	30.6
HHSI Flow Begins	59.6
Setpt to start Aux. FW Pmp	107.5
Loop seal 1 clears	878
Loop seal 2 clears	N/A
Loop seal 3 clears	N/A
Break uncovers	-900
Core uncovery begins	-1300
Hot Rod rupture occurs	2019
Accumulator Injection begins	2020
Minimum RV mass occurs	2038
PCT occurs	2060

TABLE 15.6.5-11b

SBLOCA ANALYSIS RESULTS (LIMITING CASE)

Parameter	Value
Break diameter (in)	2.60
Peak Clag Temperature (°F)	1664
Time of PCT (sec)	2060
PCT Elevation (ft)	11.13
Time of Rupture (sec)	2019
Core Wide Oxidation (%)	0.0556
Local Maximum Oxidation (%)	2.2196

TABLE 15.6.5-12

PARAMETERS USED FOR LARGE BREAK LOCA RADIOLOGICAL ANALYSIS

Source Term	
Core Activity	See Table 15.0.9-1
Activity release fractions and timing	See Tables 15.6.5-13 & Table 15.6.5-14
Iodine chemical form in containment (%)	
Elemental	4.85
Organic	0.15
Particulate (cesium iodide)	95
Containment	
Containment net free volume (ft^3)	2.344E6
Containment sprayed volume (ft^3)	2.014E6
Fan cooler units	
Number in operation	2
Flow rate (per unit)	31,250
Containment leak rates (weight %/day)	
0 - 24 hours	0.10
> 24 hours	0.05
Spray Operation	
Time to initiate sprays	120.0 seconds
Time to terminate spray operation	4.0 hours
Spray flow rates (gpm)	1730
Spray fall height (ft)	125
Removal Coefficients (hr^{-1})	
Spray elemental iodine removal	20.0
Spray particulate removal	3.94
Sedimentation particulate removal (after spray termination in sprayed region and from start of event in unsprayed region)	0.1
Containment sump volume (gal)	3.595E5
Time to initiate ECCS recirculation (min)	20
ECCS leak rate to Auxiliary Building (total, gpm)	2
Inside RABEES (gpm)	1.934
Outside RABEES (gpm)	0.066
ECCS leak rate to RWST (gpm)	1.5
Airborne fraction for ECCS leakage to RWST (%)	
Time < 24 hours and pH < 6.0	2.0
After 24 hours and pH ≥ 6.0	1.0
Partition Coefficient for Elemental Iodine for ECCS leakage to RWST	28.2
RABEES filter efficiencies (%)	
Elemental	95
Organic	95
Particulate	95

TABLE 15.6.5-13

LBLOCA CORE FISSION PRODUCT RELEASE TIMING

Release Phase	Duration
Coolant Activity	10 to 30 seconds
Gap Activity	0.5 hour
Early In-vessel	1.3 hour

TABLE 15.6.5-14

LBLOCA CORE FISSION PRODUCT RELEASE FRACTIONS

	Gap Release	Early In-Vessel
Noble gases	0.05	0.95
Halogens	0.05	0.35
Alkali Metals	0.05	0.25
Tellurium group	0	0.05
Barium, Strontium	0	0.02
Noble Materials (Ruthenium group)	0	0.0025
Cerium group	0	0.0005
Lanthanides	0	0.0002

The RG 1.183 source term assumes a release of gap activity (5% of core) followed by the in-vessel release as defined.

TABLE 15.6.5-15

CONTROL ROOM PARAMETERS USED FOR RADIOLOGICAL ANALYSIS

Volume (ft^3)	71,000
Normal Ventilation Flow Rates (cfm)	
Filtered Makeup Flow Rate	0.0
Filtered Recirculation Flow Rate	0.0
Unfiltered Makeup Flow Rate	1050.0
Unfiltered Recirculation Flow Rate (Not modeled-no impact on analyses)	
Post Accident Recirculation Flow Rates (cfm)	
Filtered Makeup Flow Rate	0.0
Filtered Recirculation Flow Rate	3600.0
Unfiltered Inleakage*	300
Unfiltered Recirculation Flow Rate (Not modeled-no impact on analyses)	
Pressurization Mode Flow Rates (cfm)	
Filtered Makeup Air Flow Rate	400.0
Filtered Recirculation Flow Rate	3600.0
Unfiltered Inleakage*	300
Unfiltered Recirculation Flow Rate (Not modeled-no impact on analyses)	
Filter Efficiencies (%)	
Elemental	99
Organic	99
Particulate	99
CR Radiation Monitor Sensitivity (μCi/ml for Xe-133)	3.0E-6
CR Radiation Monitor Location	Emergency & normal air intakes
Delay to Initiate Switchover of Post-Accident signal	15 seconds
Recirculation HVAC mode after radiation	
Operator Action Time to Switch to Pressurization Mode	2 hours
Breathing Rate – Duration of the Event (m^3/sec)	3.5E-4

Table 15.6.5-15 (Continued)

* Some analyses used more conservative unfiltered inleakage flow rates than 300 cfm.

Control Room Atmosphere Dispersion Factors for Large Break LOCA (sec/m^3)*	
0 - 8 hours	2.04E-3
8 - 24 hours	5.80E-4
1 - 4 days	1.63E-4
4 - 30 days	6.16E-6
Control Room Atmospheric Dispersion Factors for RWST vent release following a Large Break LOCA (sec/m^3)**	
0 - 8 hours	9.18E-3
8 - 24 hours	2.61E-3
1 - 4 days	7.31E-4
4 - 30 days	2.77E-5
Control Room Atmospheric Dispersion Factors for all accidents except Large Break LOCA (sec/m^3)***	
0 - 8 hours	4.08E-3
8 - 24 hours	1.16E-3
1 - 4 days	3.25E-4
4 - 30 days	1.23E-5
Occupancy Factors****	
0 - 24 hours	1.0
1 - 4 days	0.6
4 - 30 days	0.4

*These atmospheric dispersion factors incorporated a reduction factor of 4 based on credit for dual emergency air intakes being applied in accordance with SRP 6.4 Section III.3.d.(4).(ii).

**Although these factors only incorporate the reduction factor of 2, rather than the 4 as in the large break LOCA values, the doses for this release path have been reduced by a factor of 2, so further reduction is not appropriate.

***These atmospheric dispersion factors incorporate a reduction factor of 2 based on least credit for dual emergency air intakes being applied in accordance with SRP 6.4 Section III.3.d.(4).(ii). These could be further reduced by a factor of 2, similar to those used for the large break LOCA doses. The calculated analysis results using these factors therefore are conservative by a factor of 2.

****These occupancy factors have been conservatively incorporated in the atmospheric dispersion factors. This is conservative since it does not allow the benefit of reduced occupancy for activity already present in the control room from earlier periods.

TABLE 15.6.5-16

RADIOLOGICAL CONSEQUENCES OF A POSTULATED LARGE BREAK LOCA

Exclusion Area Boundary*	7.92 rem TEDE
Low Population Zone	6.05 rem TEDE
Control Room**	3.45 rem TEDE

*The exclusion area boundary dose reported is for the worst two hour period, determined to be from 0.4 hours to 2.4 hours.

**The control room dose includes the immersion dose, the ECCS leakage doses, and the indirect shine doses to operators.

TABLE 15.6.5-17

PARAMETERS USED FOR SMALL BREAK LOCA RADIOLOGICAL ANALYSIS

Source Term	
Core Activity	See Table 15.0.9-1
Fraction of fuel rods in core that fail (% of core)	100
Gap Fractions (% of core activity)	
I-131	8
Other Iodine	5
Kr-85	10
Other Noble Gas	5
Alkali Metals	12
Fraction of fuel melting (% of core)	2
Radial peaking factor applied to fuel melt	1.73
Fraction of activity released from melted fuel (%)	
Containment leakage	
Iodine	25
Noble Gas	100
Alkali Metals	100
Primary to secondary leakage	
Iodine	50
Noble Gas	100
Alkali Metals	100
Reactor coolant iodine activity (μCi/gm of DE I-131)	1.0
Reactor coolant noble gas and alkali metal activity prior to accident (% fuel defect level)	1.0
Secondary coolant iodine activity prior to accident (μCi/gm of DE I-131)	0.1
Secondary coolant alkali metal activity prior to accident (% of primary concentration)	10
Containment Leakage Release Path	
Containment net free volume (ft^3)	2.344E6
Containment sprayed volume (ft^3)	2.014E6
Fan cooler units	
Number in operation	2

Table 15.6.5-17 (Continued)

Flow rate (per unit)	31,250
Containment leak rates (weight %/day)	
0 - 24 hours	0.1
> 24 hours	0.05
Iodine chemical form in containment (%)	
Elemental	4.85
Organic	0.15
Particulate (cesium iodide)	95
Spray Operation	
Time to initiate sprays	30.0 minutes
Time to terminate spray operation	1.0 hour
Spray flow rates (gpm)	1730
Spray fall height (ft)	125
Removal Coefficients (hr^{-1})	
Spray elemental iodine removal	20.0
Spray particulate removal	3.94
Sedimentation particulate removal (after spray termination)	0.1
<u>Primary to Secondary Leakage Release Path</u>	
SG tube leak rate (gpm total)	1.0
Steam release to environment (lbm/sec)	
0 - 2 hours	378,000
> 2 hours	0
SG iodine and alkali metal water/steam partition coefficient	0.01
Iodine chemical form after release to atmosphere (%)	
Elemental	97
Organic	3
Particulate (cesium iodide)	0

TABLE 15.6.5-18

RADIOLOGICAL CONSEQUENCES OF A POSTULATED SMALL BREAK LOCA

Exclusion Area Boundary*	9.57 rem TEDE
Low Population Zone	3.70 rem TEDE
Control Room	4.92 rem TEDE

*The exclusion area boundary doses reported are for the worst two hour period, determined to be from 0.0 to 2.0 hours.

TABLE 15.7.1-1

ASSUMPTIONS FOR WASTE GAS DECAY TANK RELEASE ACCIDENT ANALYSIS

Design Basis Assumption

Source Data:

1. Power level prior to accident is 2958 MWt (rated power of 2948 MWt with 0.34% uncertainty)
2. RCS radioactive concentrations are maximum values based on 1 percent failed fuel
3. All gases stripped from processing one entire RCS volume are passed to the gas decay tank which fails. The flash tank is assumed to remove 100 percent of the noble gases and 0.1 percent of the iodines that enter it.
4. A decontamination factor of 10 is assumed for the CVCS purification ion exchanger for iodine
5. Accident occurs immediately following a cold shutdown

Activity Release:

1. All gases released from tank leak from the RAB at ground level within 2 hour period

TABLE 15.7.1-2

RADIOLOGICAL CONSEQUENCES OF A WASTE GAS DECAY TANK RELEASE

Gas Decay Tank Isotopes	Tank Inventory (Ci)
Kr-83m	19.1
Kr-85m	138.0
Kr-85	4100.0
Kr-87	46.0
Kr-88	172.0
Xe-131m	775.0
Xe-133m	903.0
Xe-133	58500.0
Xe-135m	56.6
Xe-135	900.0
Xe-138	5.16
Exclusion Area Boundary*	0.30 rem TEDE
Low Population Zone	0.069 rem TEDE
Control Room	0.049 rem TEDE

* The exclusion area boundary dose reported is for the worst two hour period, determined to be from 0.0 to 2.0 hours.

TABLE 15.7.4-1

PARAMETERS USED IN FUEL HANDLING ACCIDENT INSIDE THE FUEL HANDLING BUILDING RADIOLOGICAL ANALYSIS

Radial peaking factor (PWR fuel)	1.73
(BWR fuel)	1.5
Fuel damaged (number of assemblies)	1.2 PWR (314 rods) + 52 BWR
Time from shutdown before fuel movement (PWR)(hr)	100
(BWR fuel) (yr)	4
Activity in the damaged fuel assemblies (Ci)	
I-131	7.21E5
I-133	7.59E4
I-135	5.57E1
Kr-85	1.41E5
Xe-131m	9.06E3
Xe-133m	1.77E4
Xe-133	1.19E6
Xe-135	2.41E2
Gap Fractions (% of core activity)	
I-131	8
Kr-85	10
Other Iodine and Noble Gas nuclides	5
Water depth	21 feet
Overall pool iodine scrubbing factor	200
Iodine chemical form in release to atmosphere (%)	
Elemental	70
Organic	30
Particulate	0
Spent Fuel Pool Ventilation System Filter efficiency	
Elemental	95
Organic	95
Particulate	95
Isolation of release No isolation assumed	
Time to release all activity (hours) 2	
Activity Released, (Ci)	
I-131	1.439E1
I-133	9.47E-1
I-135	6.950E-4
Kr-85	1.410E4
Xe-131m	4.530E2
Xe-133m	8.850E2
Xe-133	5.950E4
Xe-135	1.205E1

TABLE 15.7.4-2

RADIOLOGICAL CONSEQUENCES OF A POSTULATED FUEL HANDLING ACCIDENT IN THE FUEL HANDLING BUILDING

Exclusion Area Boundary*	0.34 rem TEDE
Low Population Zone	0.077 rem TEDE
Control Room	0.12 rem TEDE

*The exclusion area boundary dose reported is for the worst two hour period, determined to be from 0 to 2 hours.

TABLE 15.7.4-3

Parameters Used in a Fuel Handling Accident Inside Containment Radiological Analysis

Radial peaking factor	1.73
Fuel damaged (number of assemblies)	1
Time from shutdown before fuel movement (hr)	100
Activity in the damaged fuel assembly (Ci)	
I-131	6.06E5
I-133	6.38E4
I-135	4.68E1
Kr-85	8.82E3
Xe-131m	7.61E3
Xe-133m	1.49E4
Xe-133	9.97E5
Xe-135	2.03E2
Gap Fractions (% of core activity)	
I-131	8
Kr-85	10
Other Iodine and Noble Gas nuclides	5
Water depth	22 feet
Overall pool iodine scrubbing factor	200
Iodine chemical form in release to atmosphere (%)	
Elemental	70
Organic	30
Particulate	0
Filter efficiency	No filtration assumed
Isolation of release	No isolation assumed
Time to release all activity (hours)	2
Activity Released (Ci)	
I-131	2.420E2
I-133	1.592E1
I-135	1.168E2
Kr-85	8.820E2
Xe-131m	3.805E2
Xe-133m	7.450E2
Xe-133	4.985E4
Xe-135	1.015E1

TABLE 15.7.4-4

RADIOLOGICAL CONSEQUENCES OF A POSTULATED FUEL HANDLING ACCIDENT INSIDE CONTAINMENT

Exclusion Area Boundary*	2.03 rem TEDE
Low Population Zone	0.46 rem TEDE
Control Room	1.39 rem TEDE

*The exclusion area boundary dose reported is for the worst two hour period, determined to be from 0 to 2 hours.

TABLE 15.7.5-1

FUEL CONDITIONS ANALYZED FOR IF-300 SERIES CASK DROP

Fuel Type	Assembly Average Burn-up (GWD/MTU)	Enrichment (w/o)	Cooling (Decay) (yrs)
PWR Fuel Types:			
PWR 15X15	35	2.33	2.5
PWR 15X15	35	5.00	2.5
PWR 15X15	45	2.33	5
PWR 15X15	45	5.00	5
BWR Fuel Types:			
BWR 7x7	35	2.33	3
BWR 7x7	35	5.00	3
BWR 8x8	35	2.33	3
BWR 8x8	35	5.00	3
BWR 8x8R	35	2.33	3
BWR 8x8R	35	5.00	3
GE- 7,8,9,10 & 13	45	3.19	4
GE- 7,8,9,10 & 13	45	4.25	4

FIGURE 15.0.5-4

ROD POSITION VERSUS TIME AFTER ROD DROP BEGINS
UTILIZED IN AREVA ANALYSES

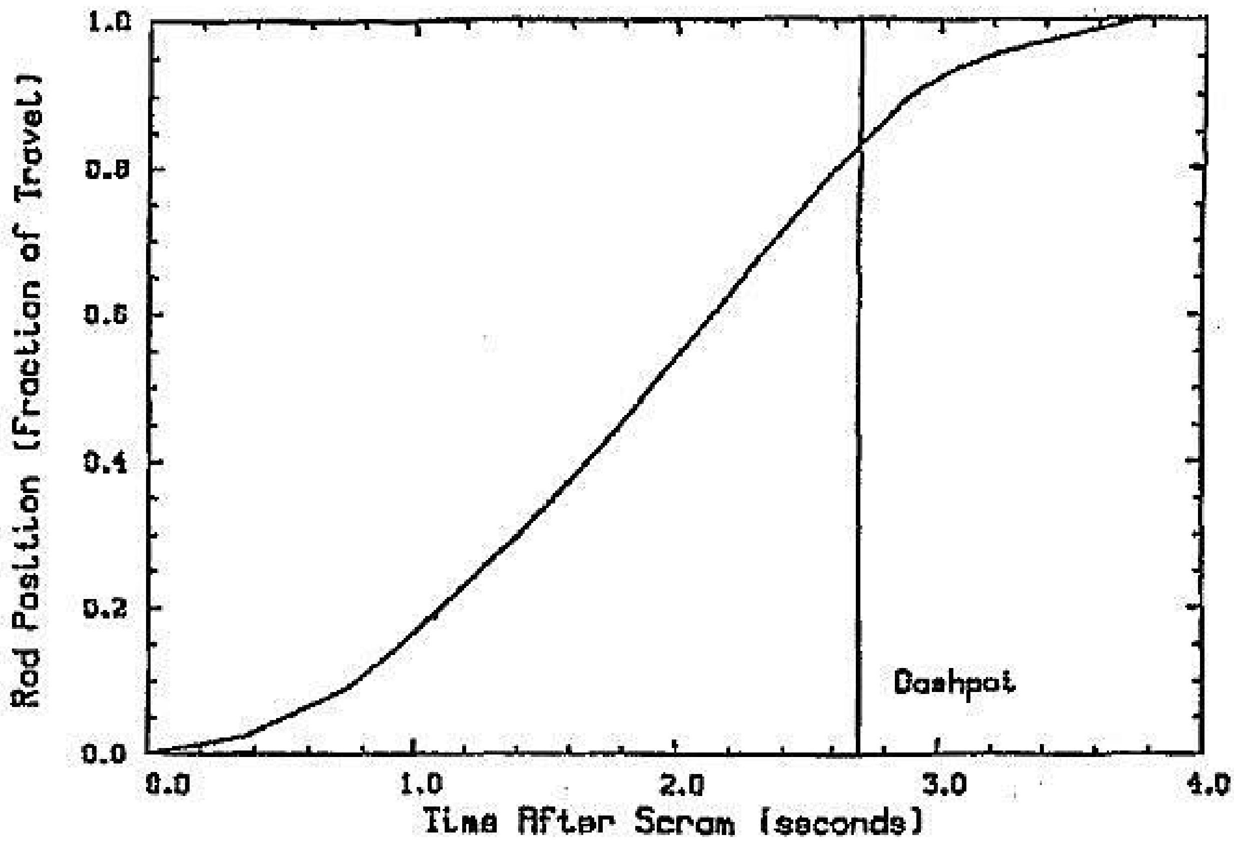


FIGURE 15.0.5-5

NORMALIZED RCCA REACTIVITY WORTH VERSUS ROD INSERTION
UTILIZED IN AREVA ANALYSES

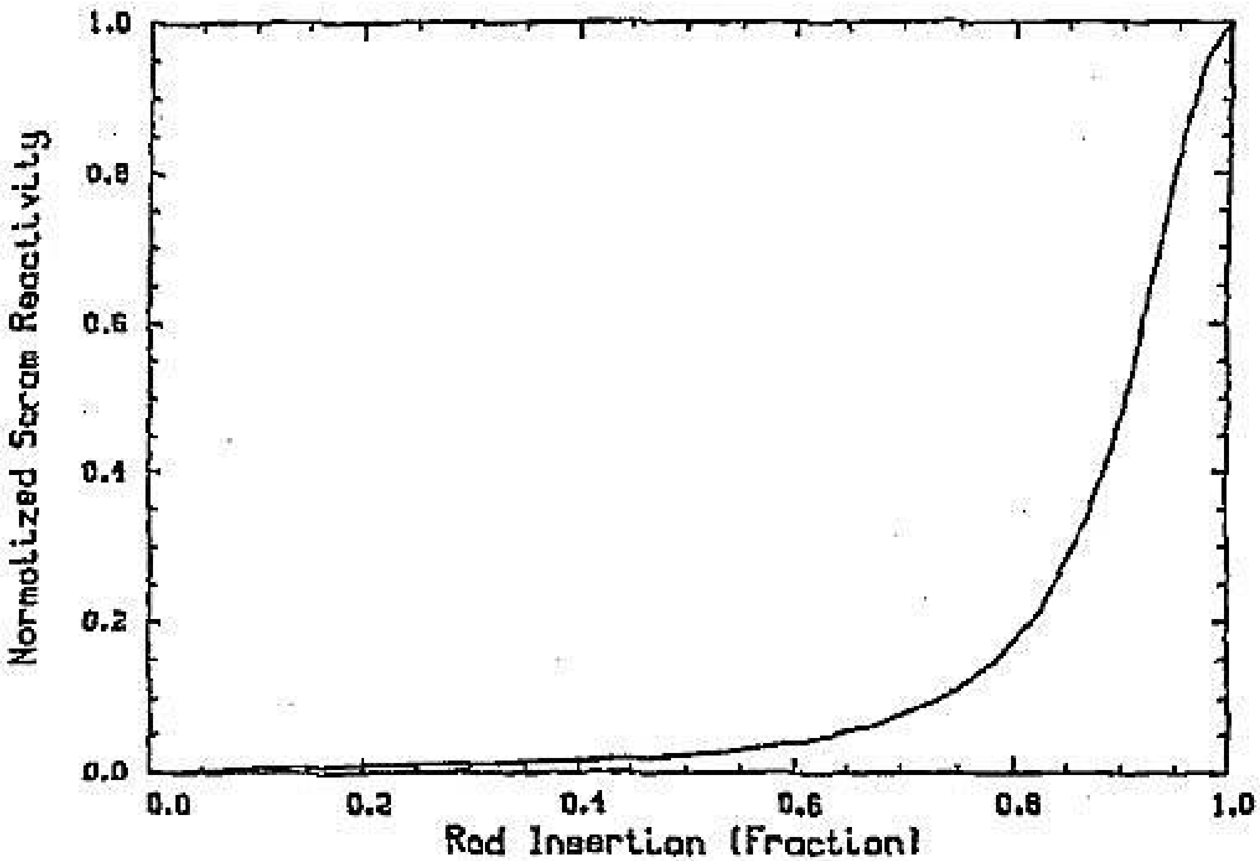


FIGURE 15.0.5-6

NORMALIZED RCCA REACTIVITY WORTH VERSUS TIME AFTER ROD DROP BEGINS UTILIZED IN AREVA ANALYSES

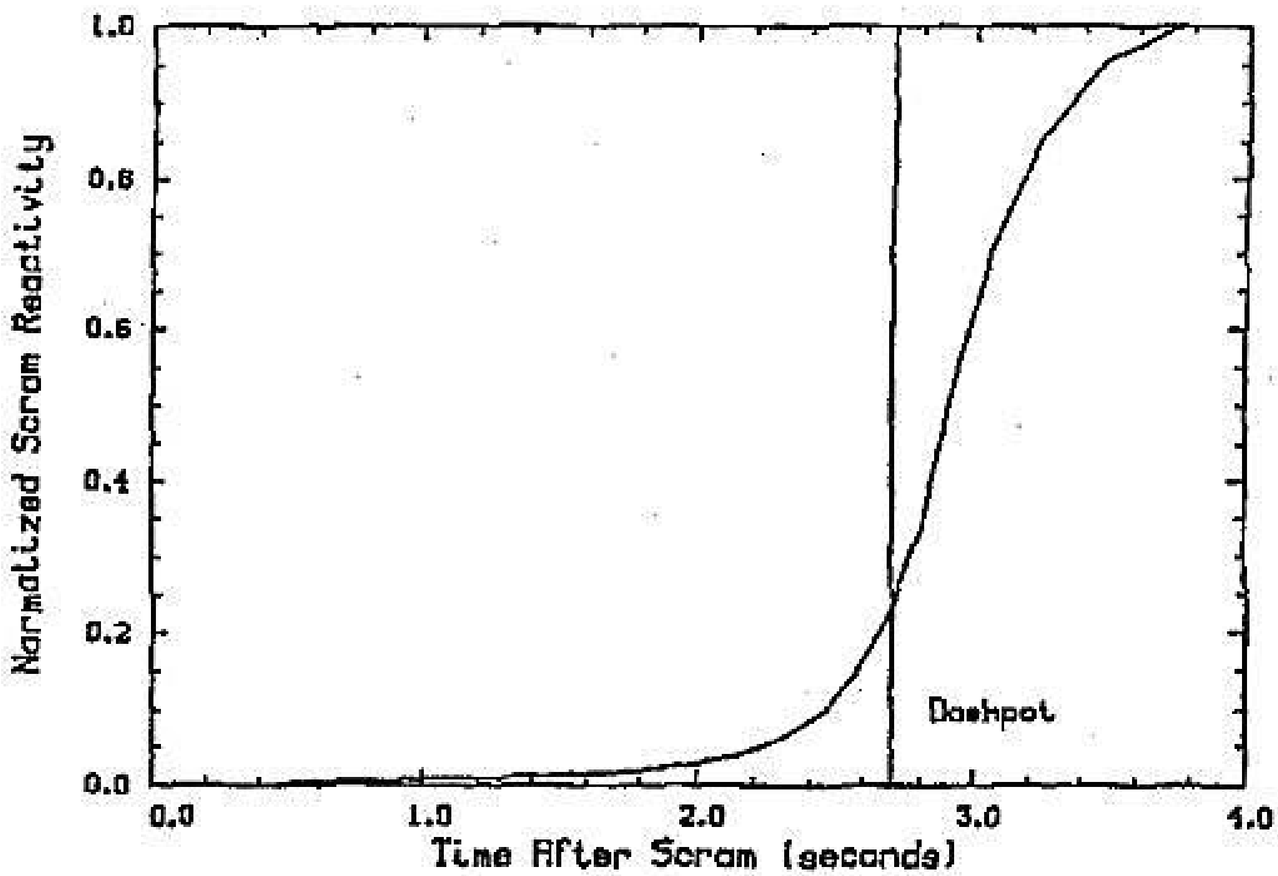
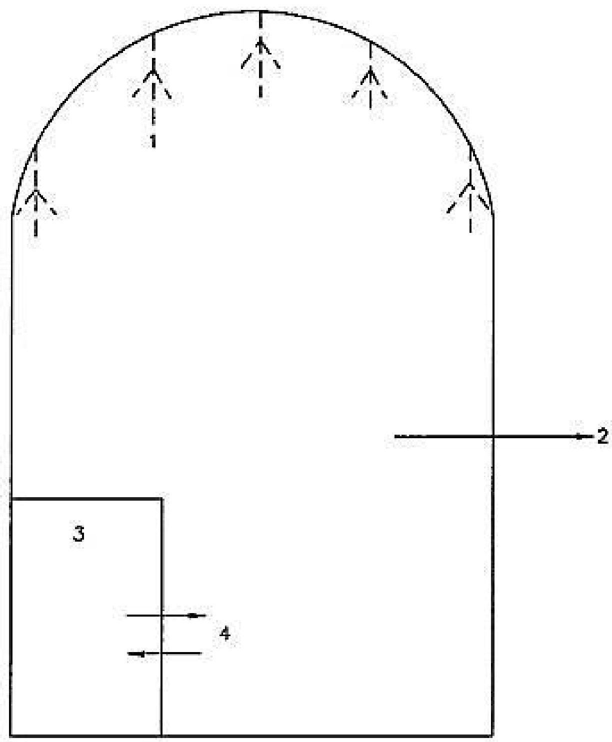


FIGURE 15.0A.1-1

CONTAINMENT LEAKAGE DOSE MODEL



NOTES:

1. CONTAINMENT SPRAY WITH SODIUM HYDROXIDE.
2. CONTAINMENT LEAKAGE TO ATMOSPHERE.
3. UNSPRAYED REGION OF CONTAINMENT.
4. AIR EXCHANGE VIA CONTAINMENT FAN COOLER SYSTEM.

FIGURE 15.1.2-1

REACTOR POWER FOR INCREASE IN FEEDWATER FLOW-LIMITING CASE

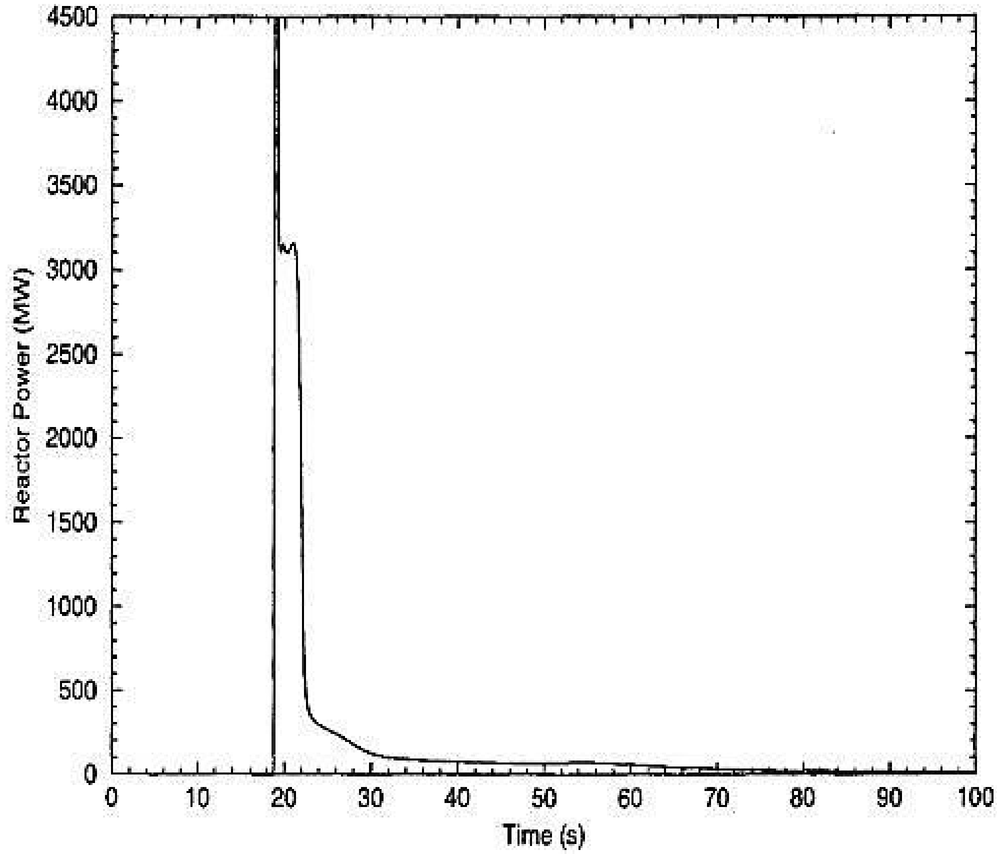


FIGURE 15.1.2-2

PRIMARY SYSTEM TEMPERATURES FOR INCREASE IN FEEDWATER FLOW - LIMITING CASE

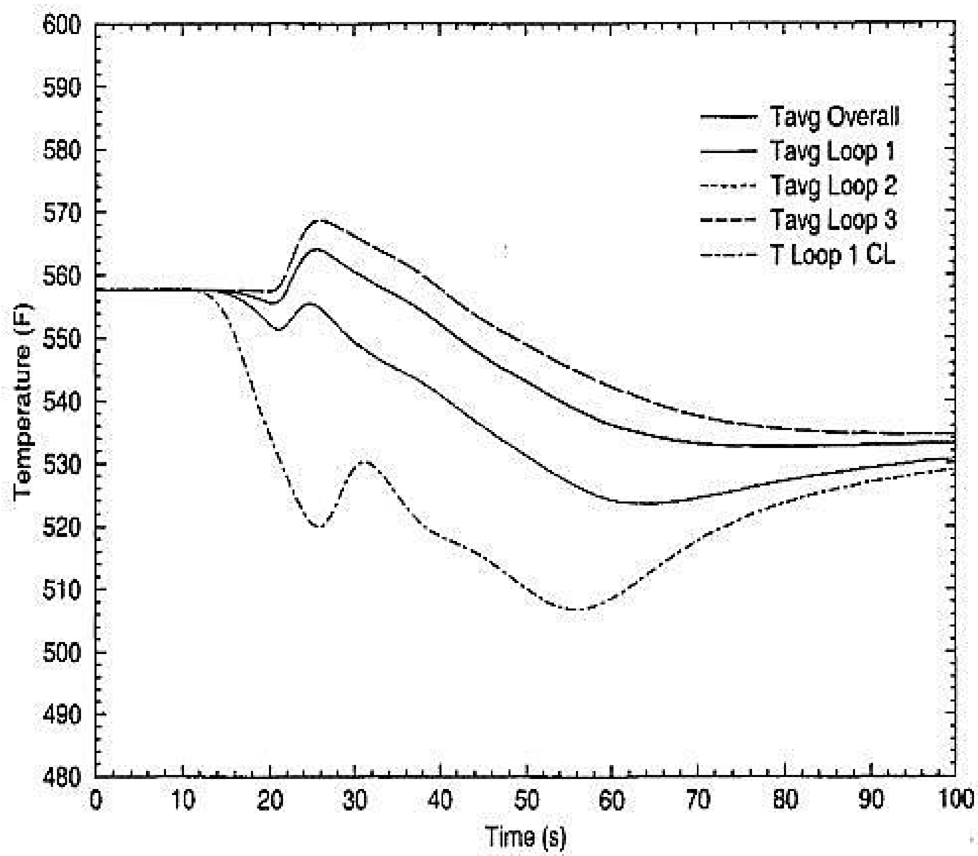


FIGURE 15.1.2-3

REACTIVITIES FOR INCREASE IN FEEDWATER FLOW – LIMITING CASE

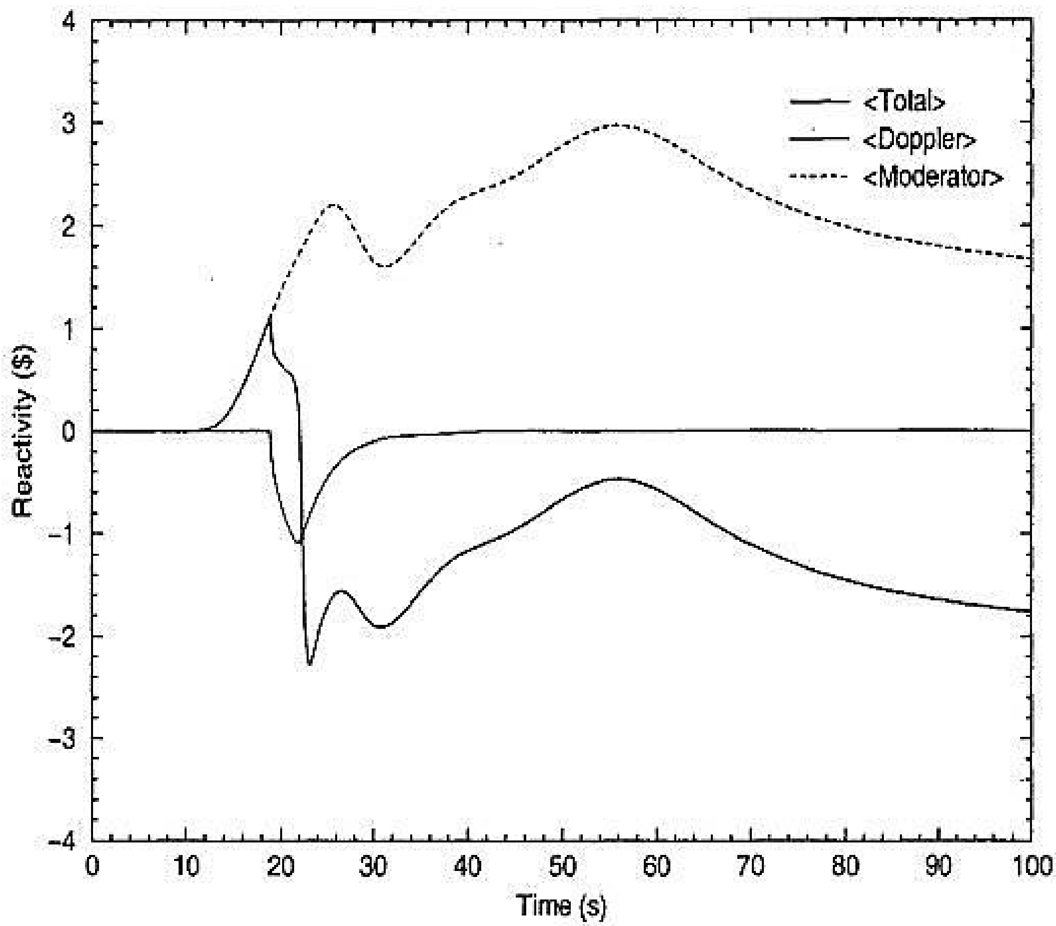


FIGURE 15.1.2-4

PRESSURIZER PRESSURE FOR INCREASE IN FEEDWATER FLOW – LIMITING CASE

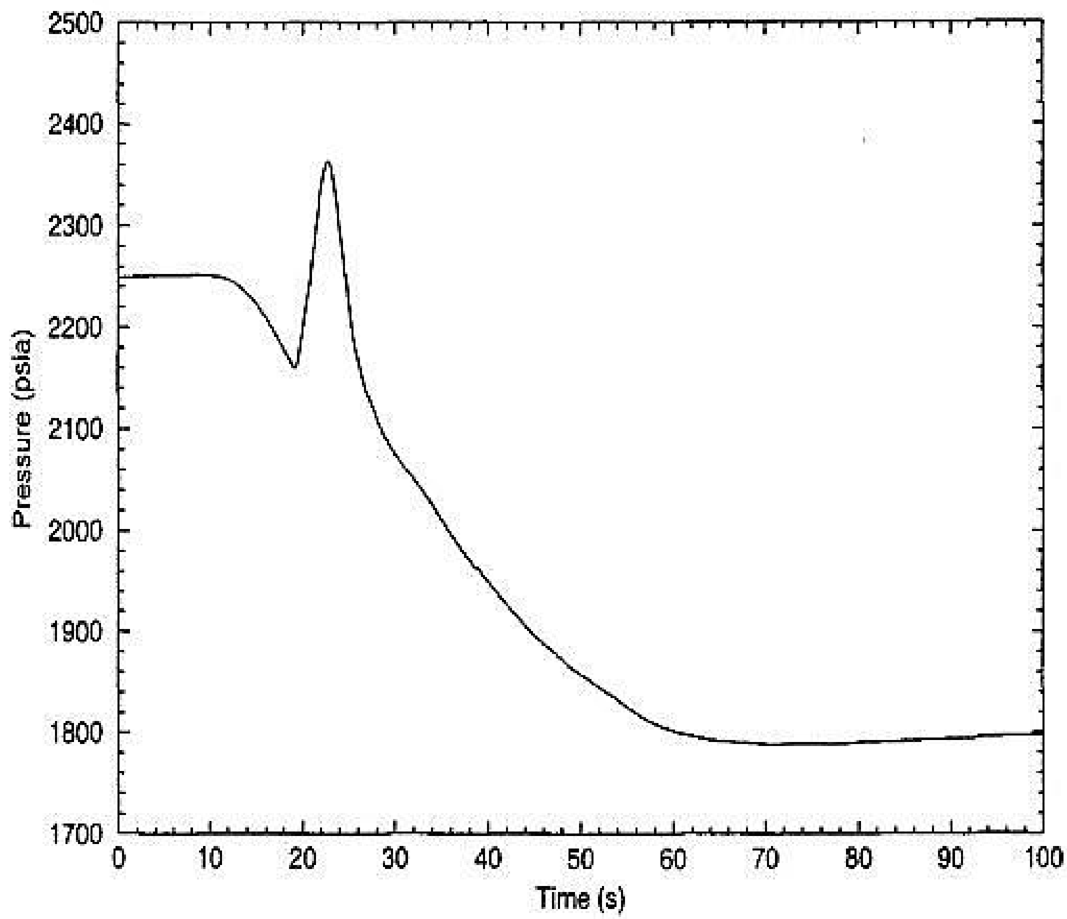


FIGURE 15.1.2-5

STEAM GENERATOR COLLAPSED LEVEL FOR INCREASE IN FEEDWATER FLOW - LIMITING CASE

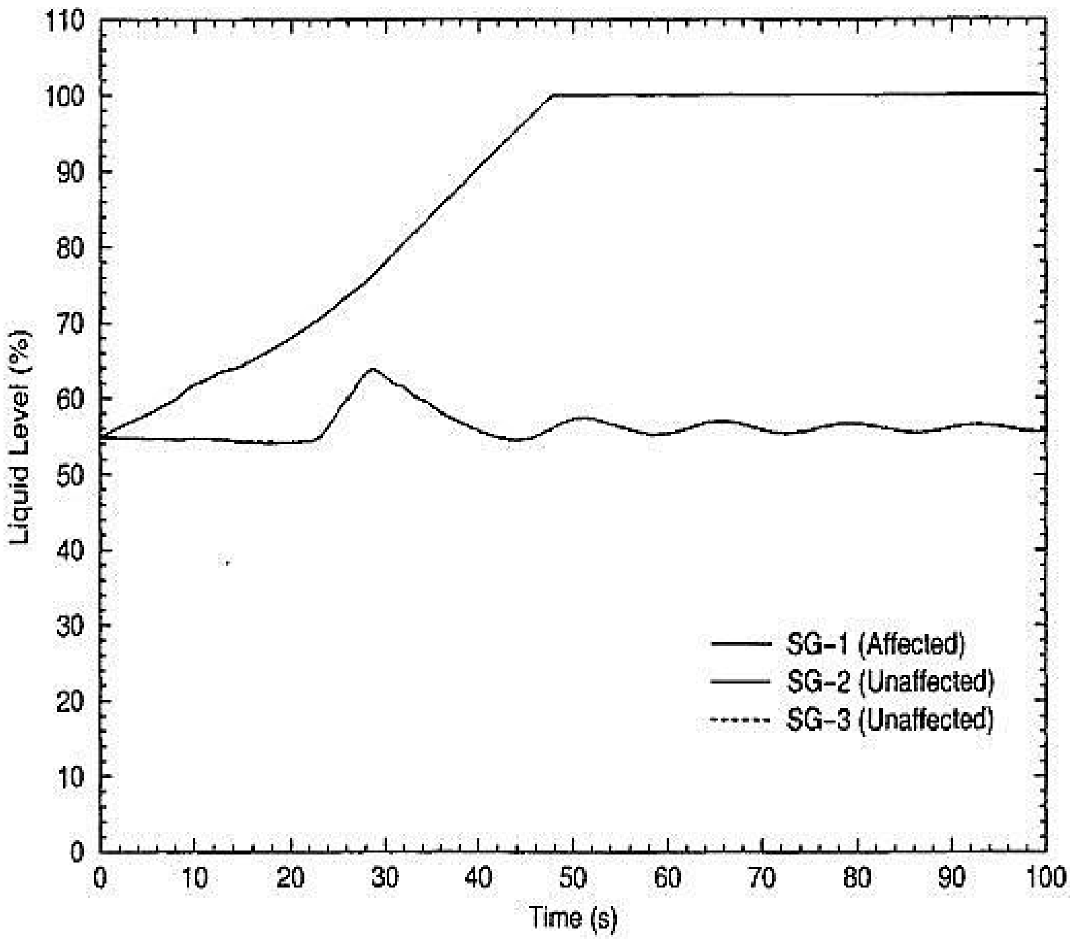


FIGURE 15.1.3-1

REACTOR POWER FOR INCREASE IN STEAM FLOW – BOC CASE

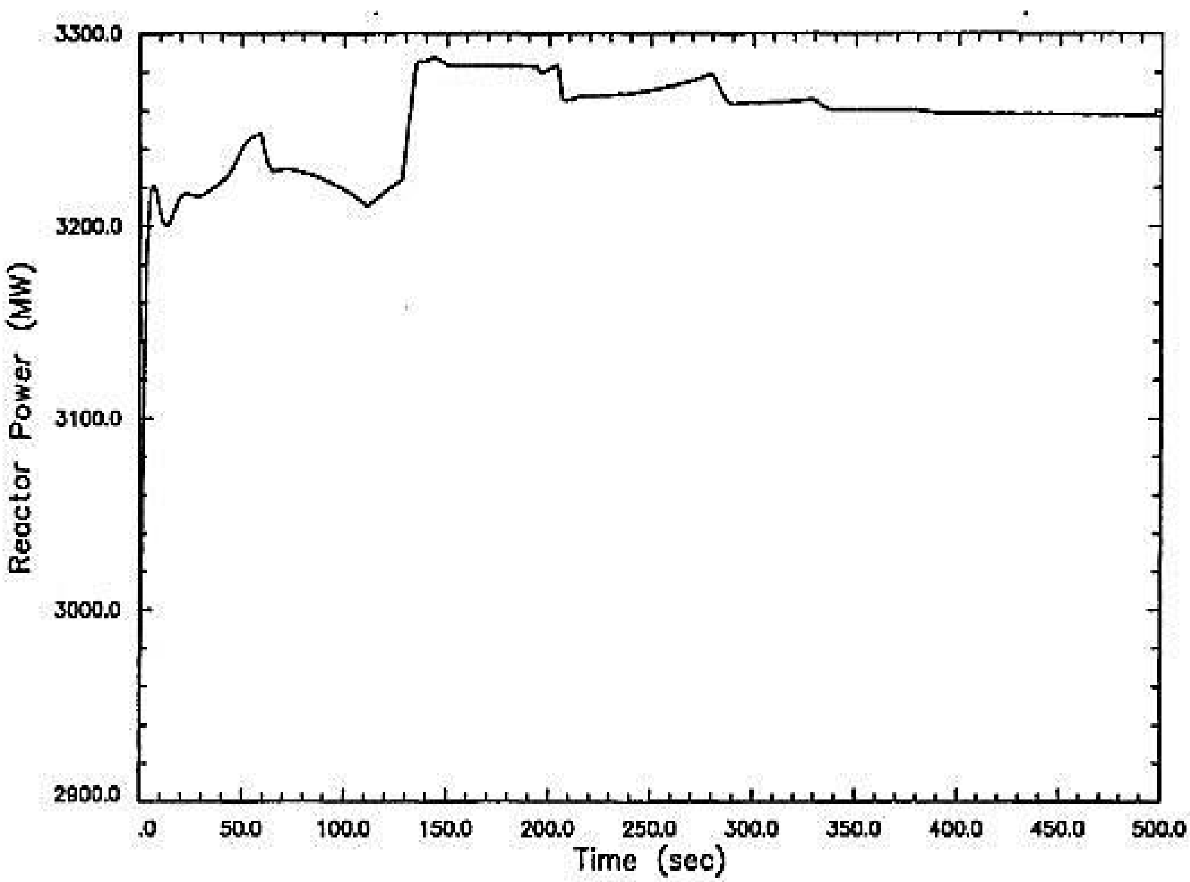


FIGURE 15.1.3-2

PRESSURIZER PRESSURE FOR INCREASE IN STEAM FLOW – BOC CASE

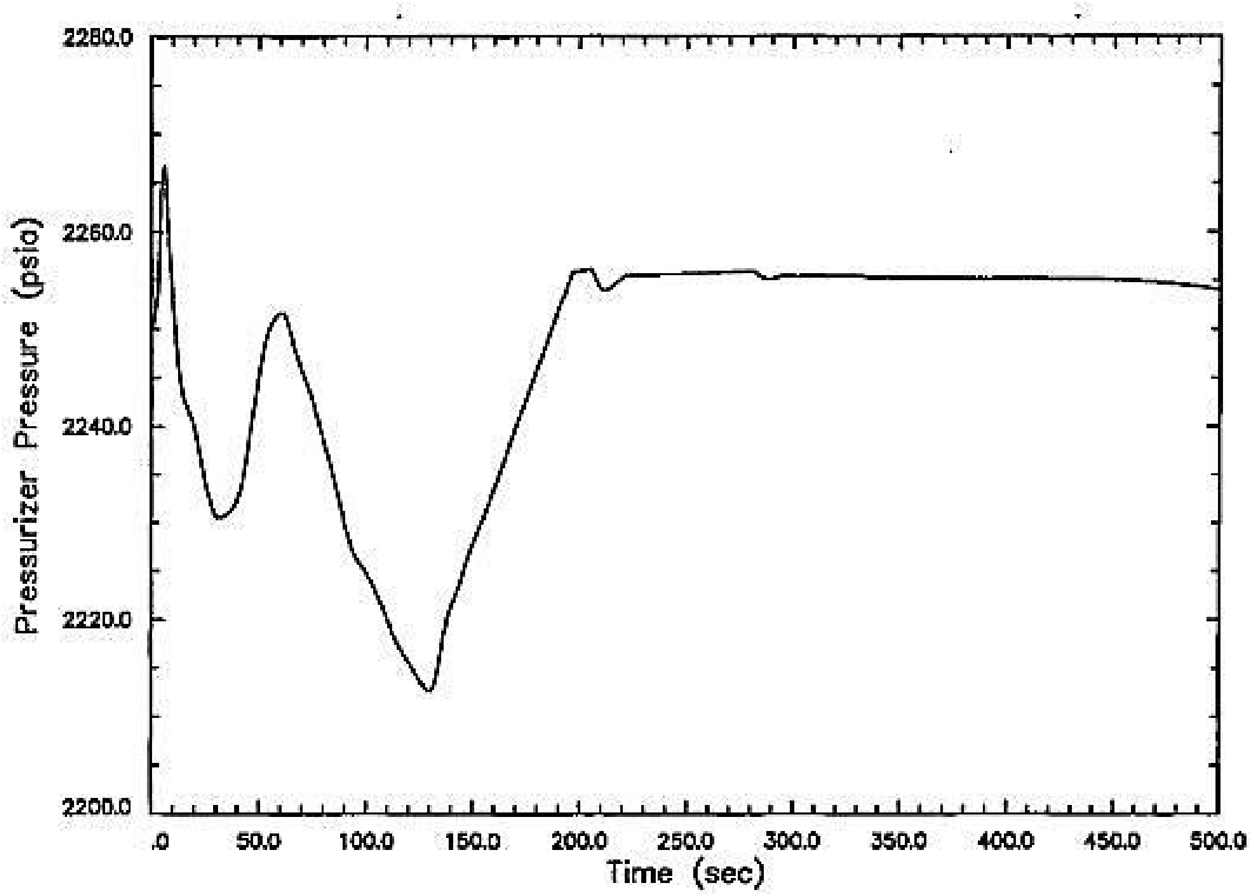


FIGURE 15.1.3-3

REACTIVITIES FOR INCREASE IN STEAM FLOW – BOC CASE

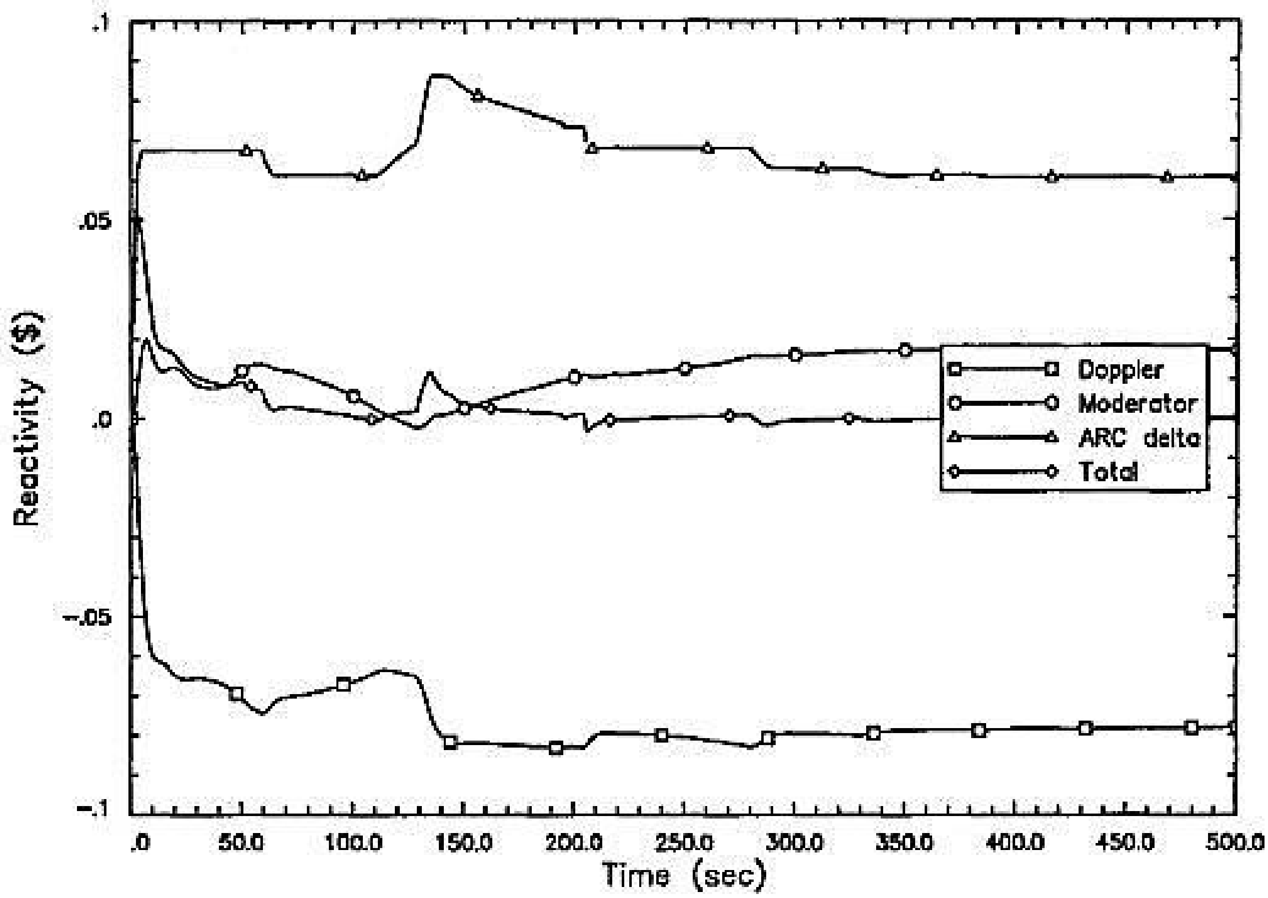


FIGURE 15.1.3-4

VESSEL AVERAGE TEMPERATURE FOR INCREASE IN STEAM FLOW – BOC CASE

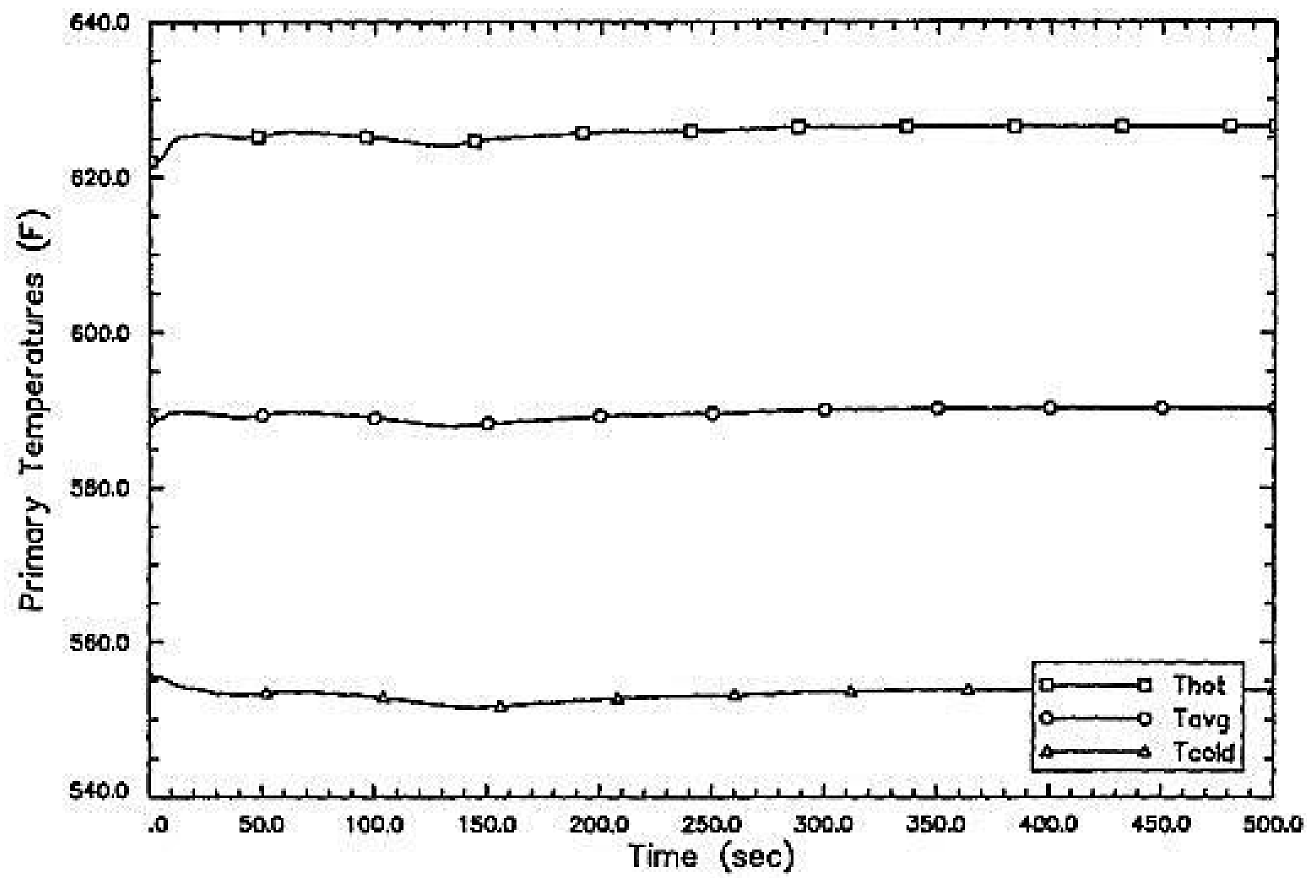


FIGURE 15.1.5-1

STEAM GENERATOR PRESSURES FOR THE HZP MSLB WITH OFFSITE POWER AVAILABLE AND WITH A STUCK ROD

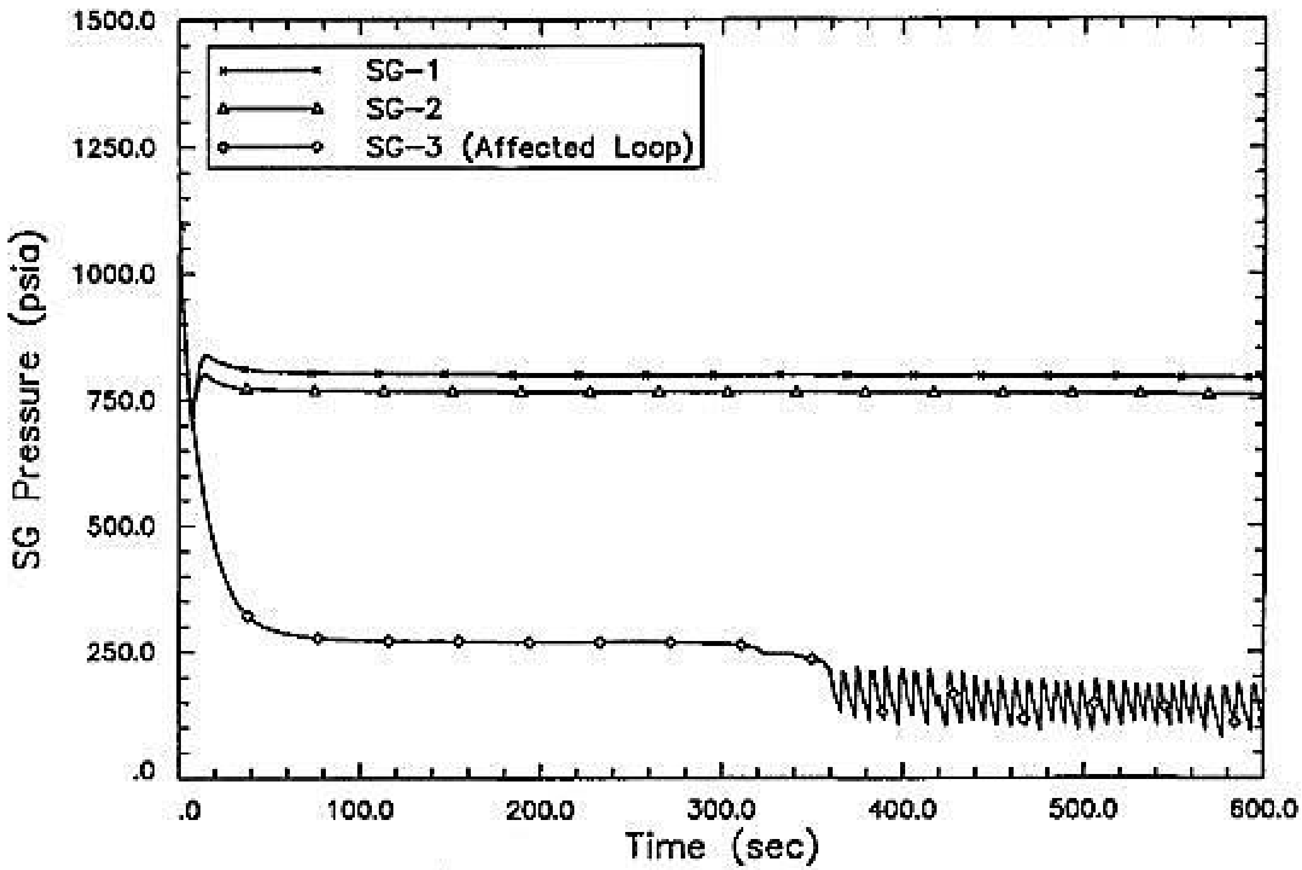


FIGURE 15.1.5-2

BREAK FLOW RATES FOR THE HZP MSLB WITH OFFSITE POWER AVAILABLE AND WITH A STUCK ROD

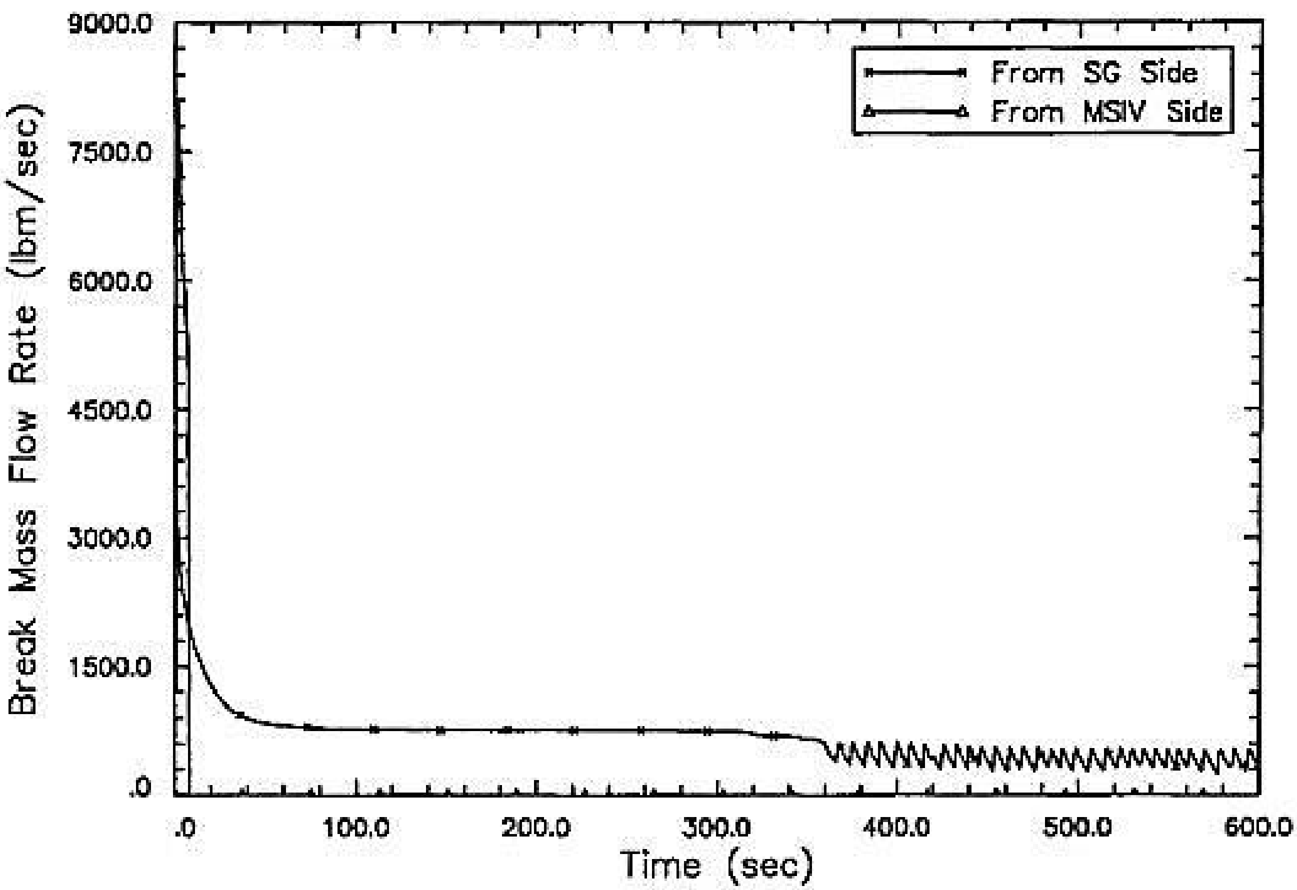


FIGURE 15.1.5-3

STEAM GENERATOR INVENTORIES FOR THE HZP MSLB WITH OFFSITE POWER AVAILABLE WITH A STUCK ROD

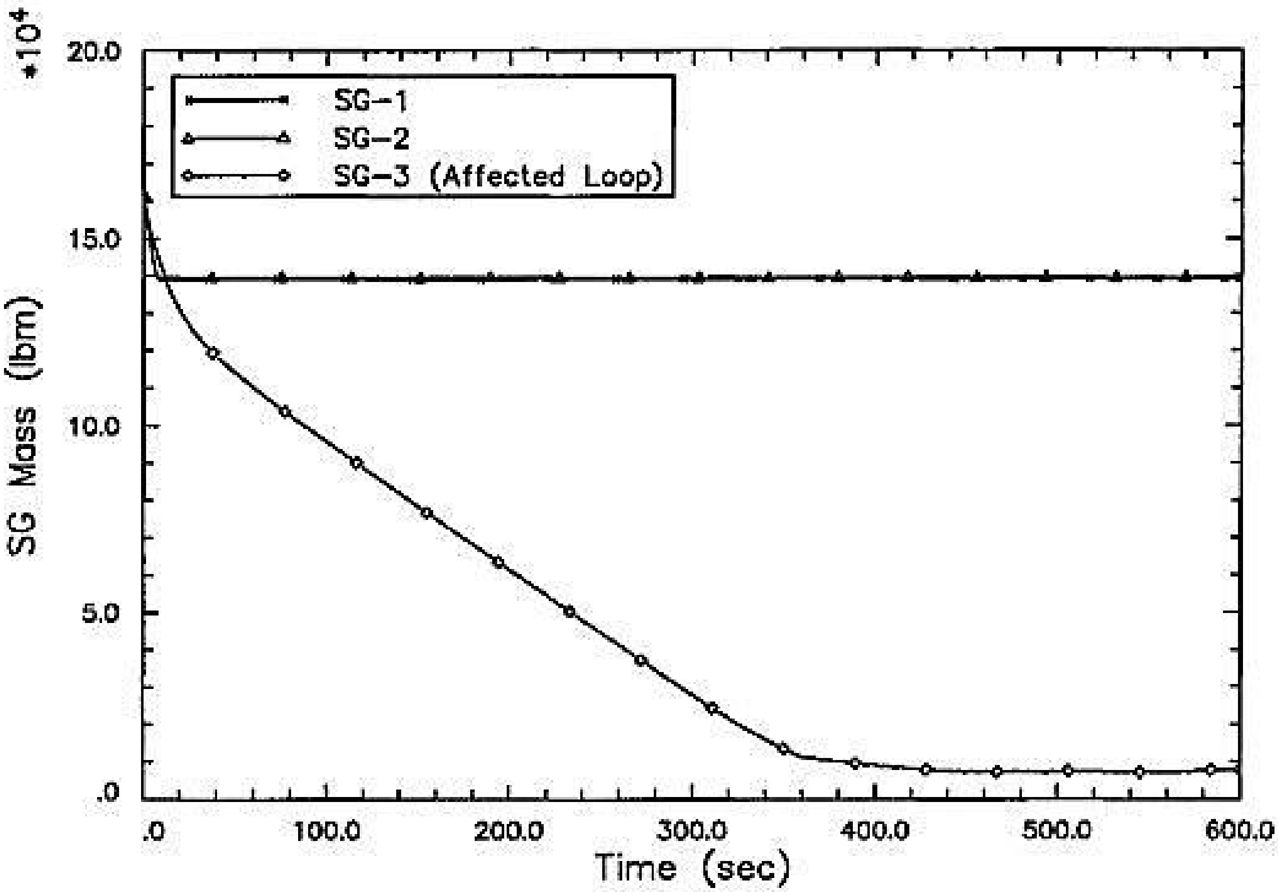


FIGURE 15.1.5-4

PRESSURIZER PRESSURE FOR THE HZP MSLB WITH OFFSITE POWER AVAILABLE AND WITH A STUCK ROD

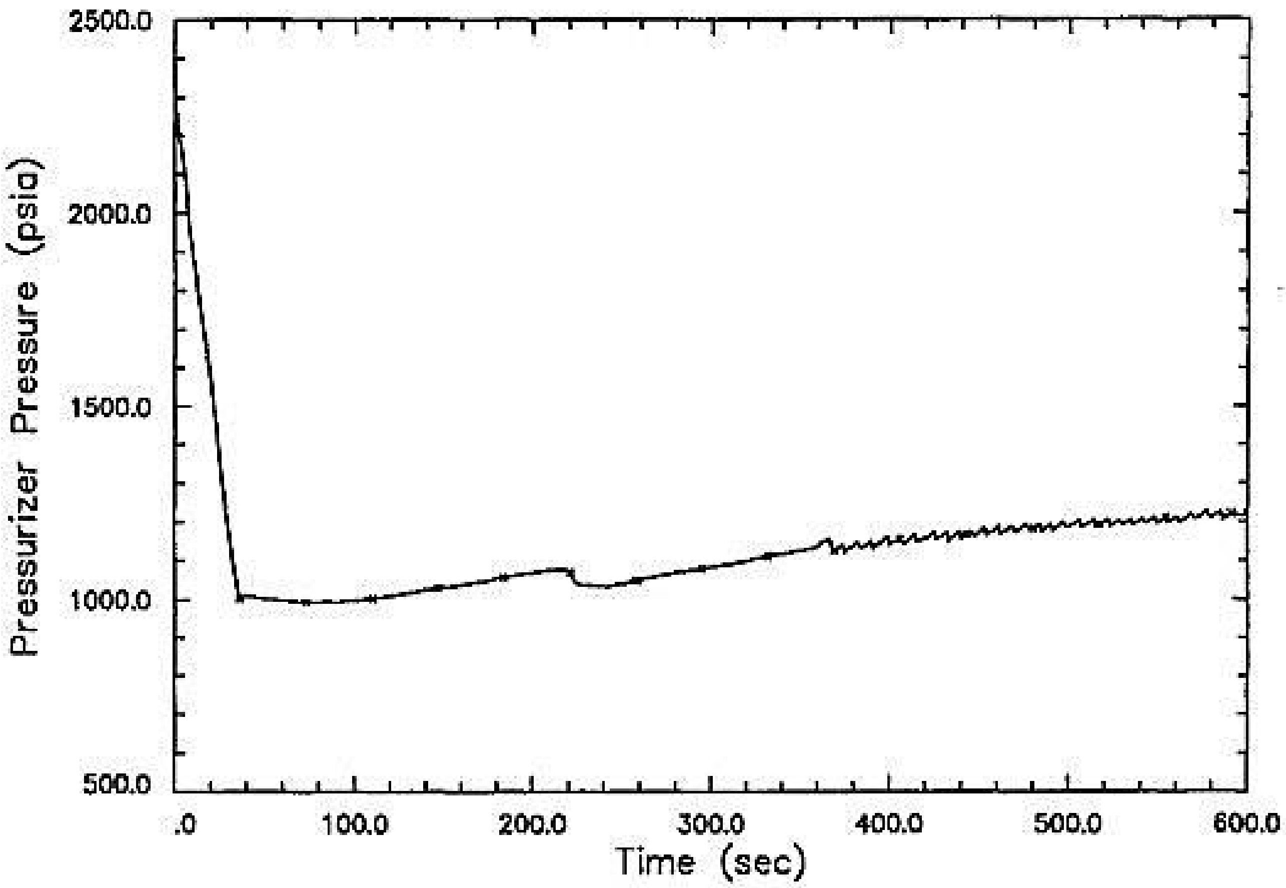


FIGURE 15.1.5-5

REACTOR POWER FOR THE HZP MSLB WITH OFFSITE POWER AVAILABLE AND WITH A STUCK ROD

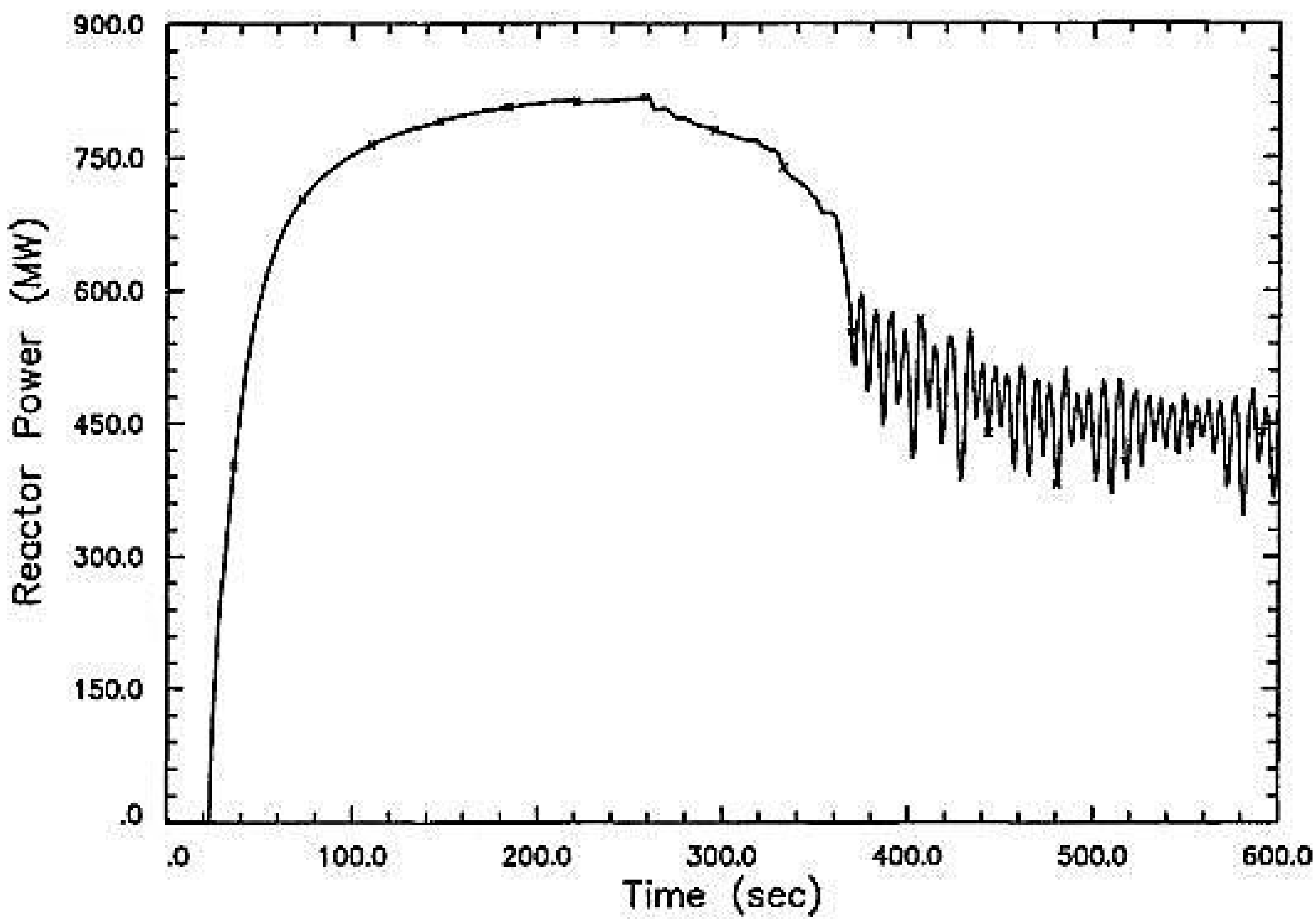


FIGURE 15.1.5-6

REACTIVITY COMPONENTS FOR THE HZP MSLB WITH OFFSITE POWER AVAILABLE AND WITH A STUCK ROD

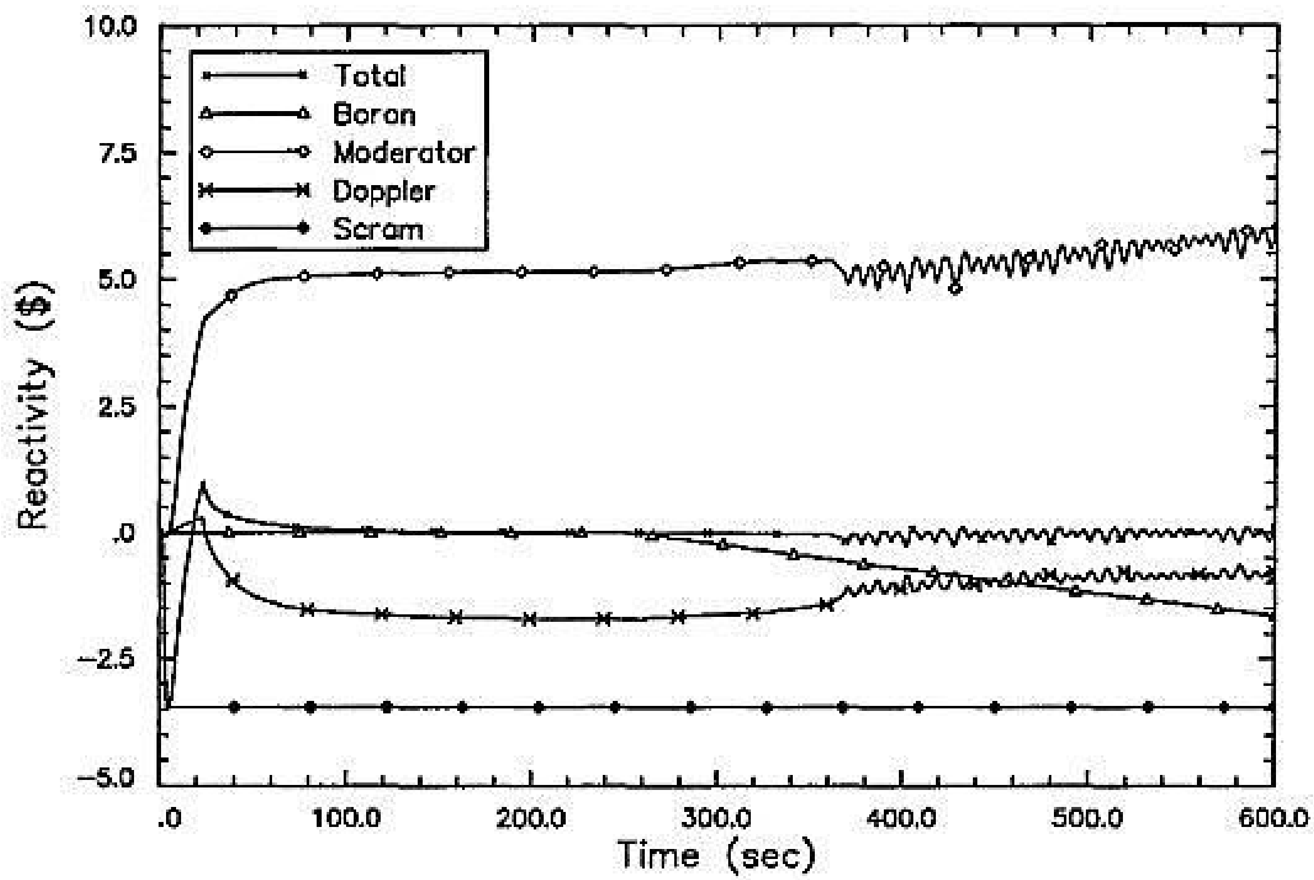


FIGURE 15.2.3-1

REACTOR POWER FOR TURBINE TRIP PRIMARY SIDE OVERPRESSURIZATION CASE

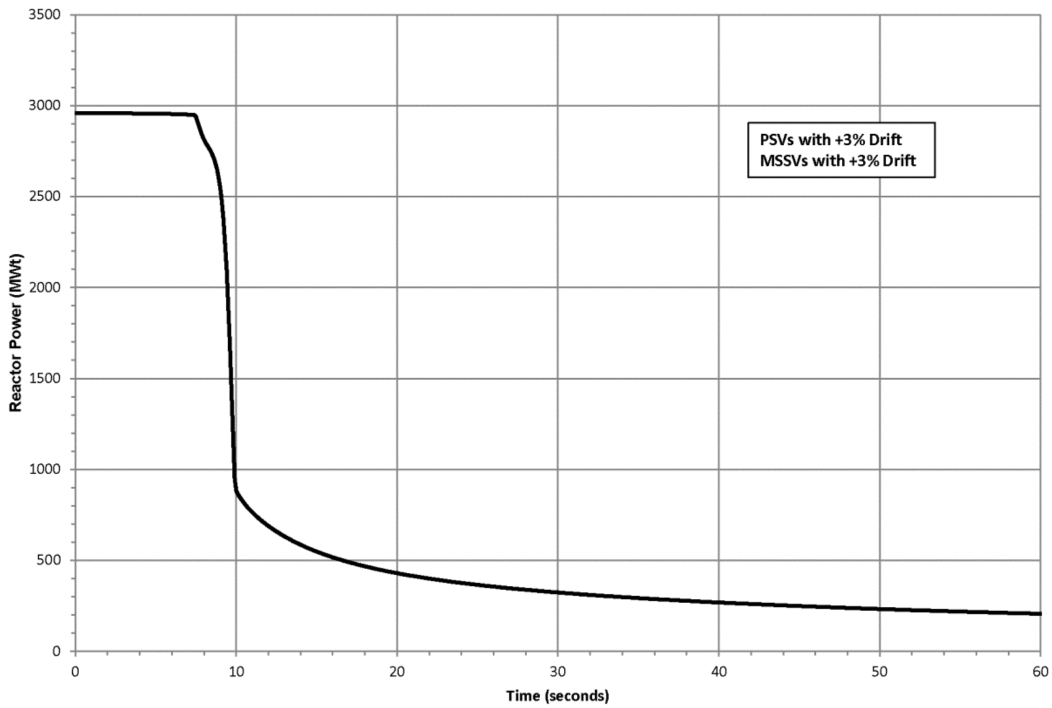


FIGURE 15.2.3-2

AVERAGE TEMPERATURES FOR TURBINE TRIP PRIMARY SIDE OVERPRESSURIZATION CASE

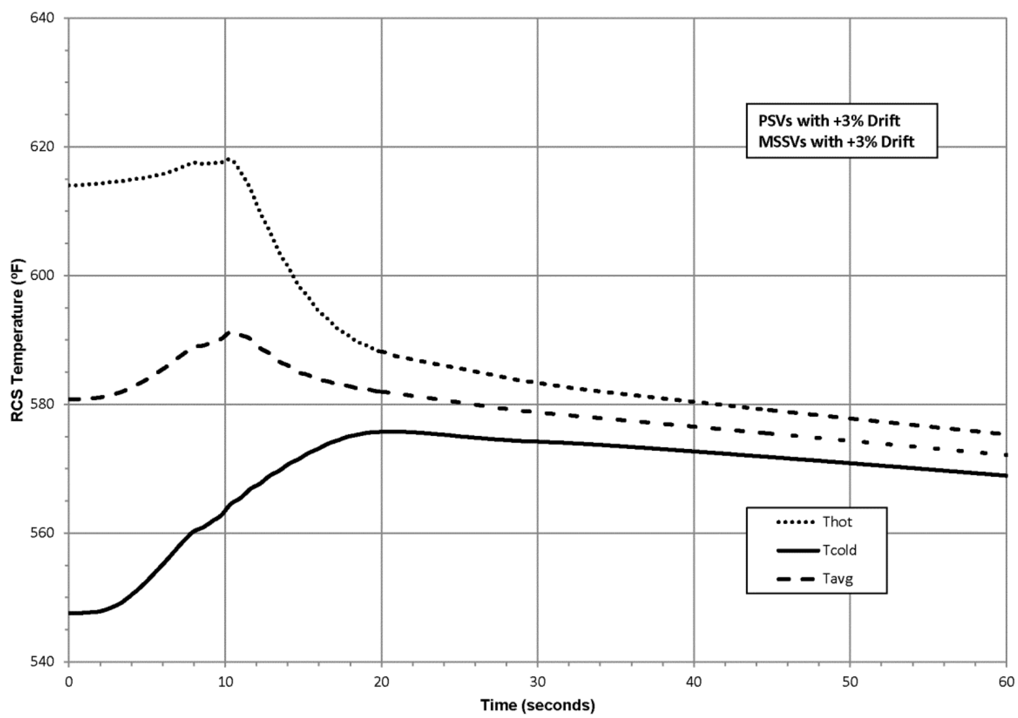
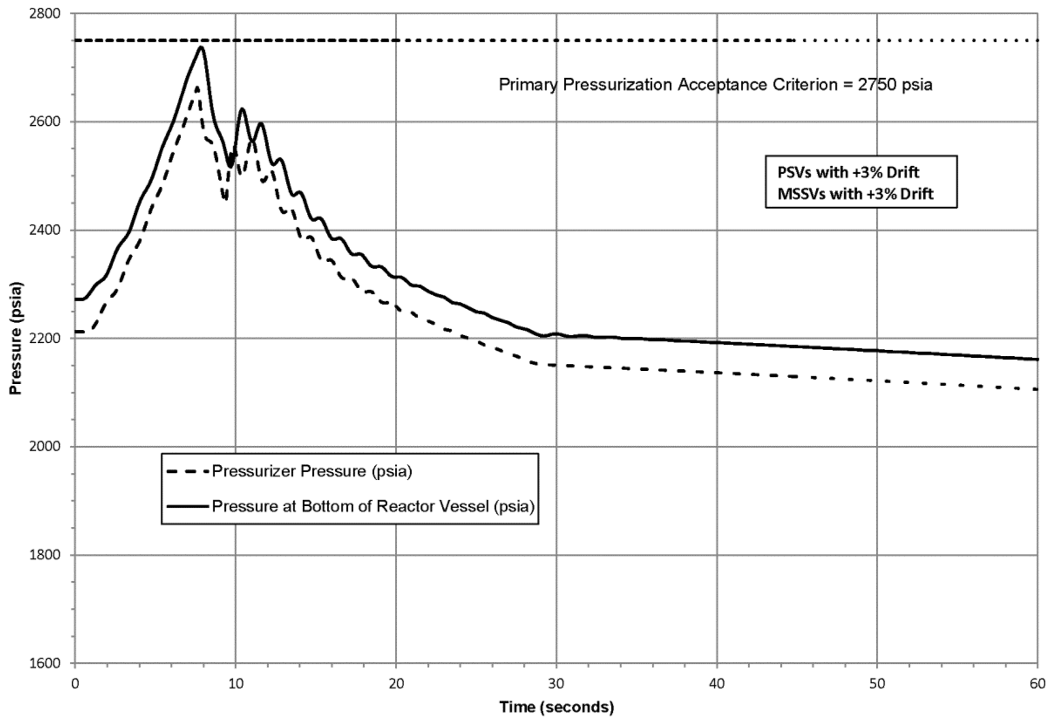


FIGURE 15.2.3-3

PRESSURIZER PRESSURE AND LOWER HEAD PRESSURE* FOR TURBINE TRIP PRIMARY SIDE OVERPRESSURIZATION CASE



* Maximum primary system pressure at the bottom of the RV downcomer is less than the pressure limit. Peak pressure at the bottom of the RV downcomer is approximately 1 psi higher than the pressure at the bottom of the RV lower head. The trace of pressure at the bottom of the RV lower head demonstrates the trends in RV pressure.

FIGURE 15.2.3-4

PRESSURIZER LEVEL FOR TURBINE TRIP PRIMARY SIDE OVERPRESSURIZATION CASE

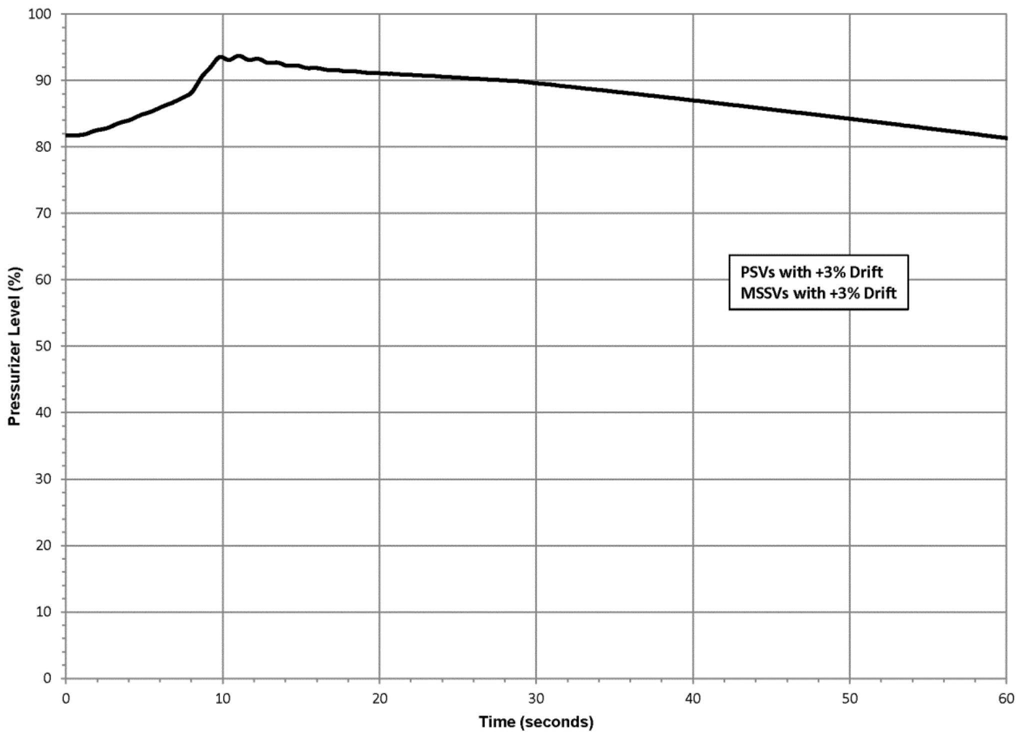


FIGURE 15.2.3-5

REACTOR POWER FOR TURBINE TRIP MDNBR CASE

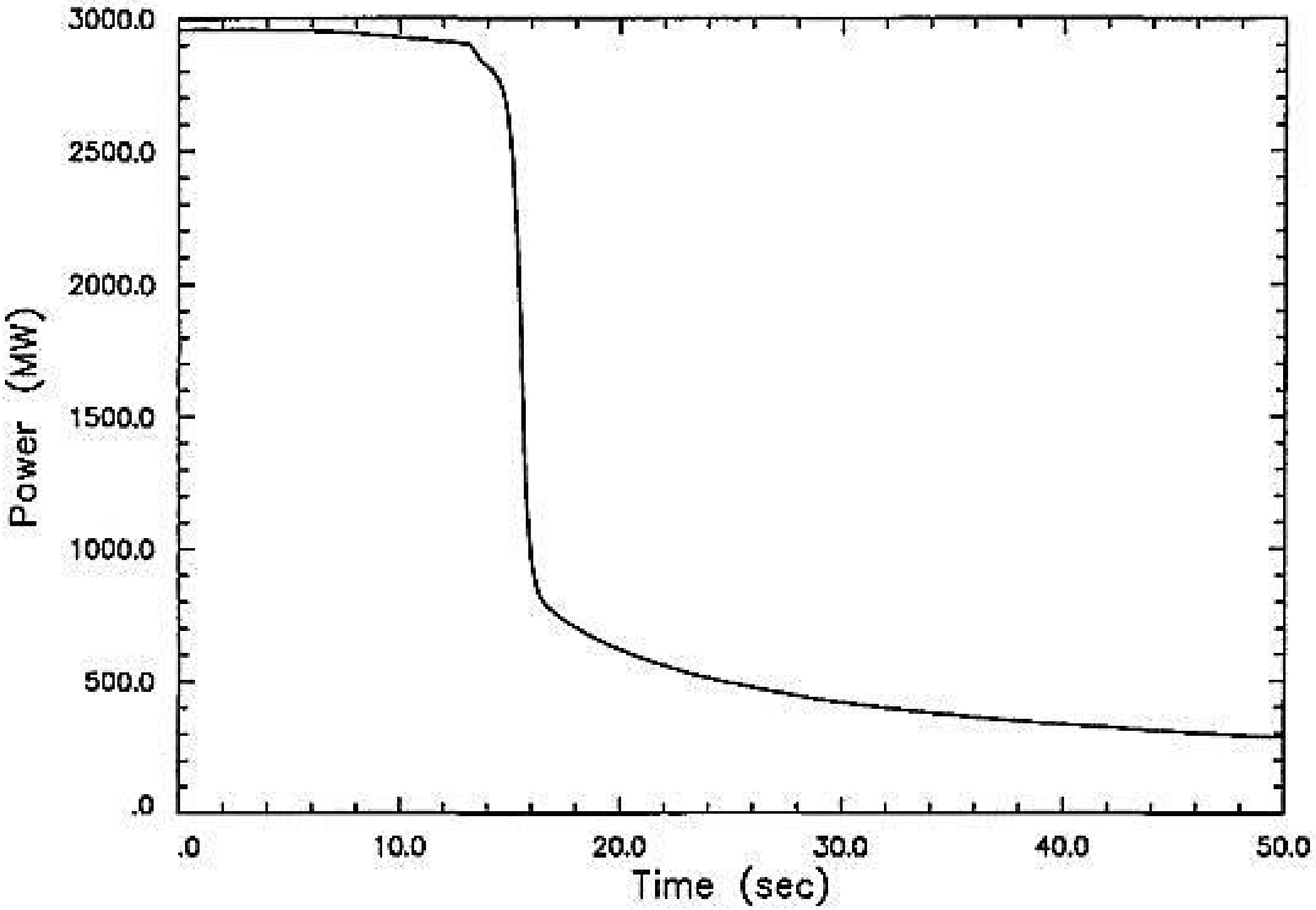


FIGURE 15.2.3-6

AVERAGE TEMPERATURES FOR TURBINE TRIP MDNBR CASE

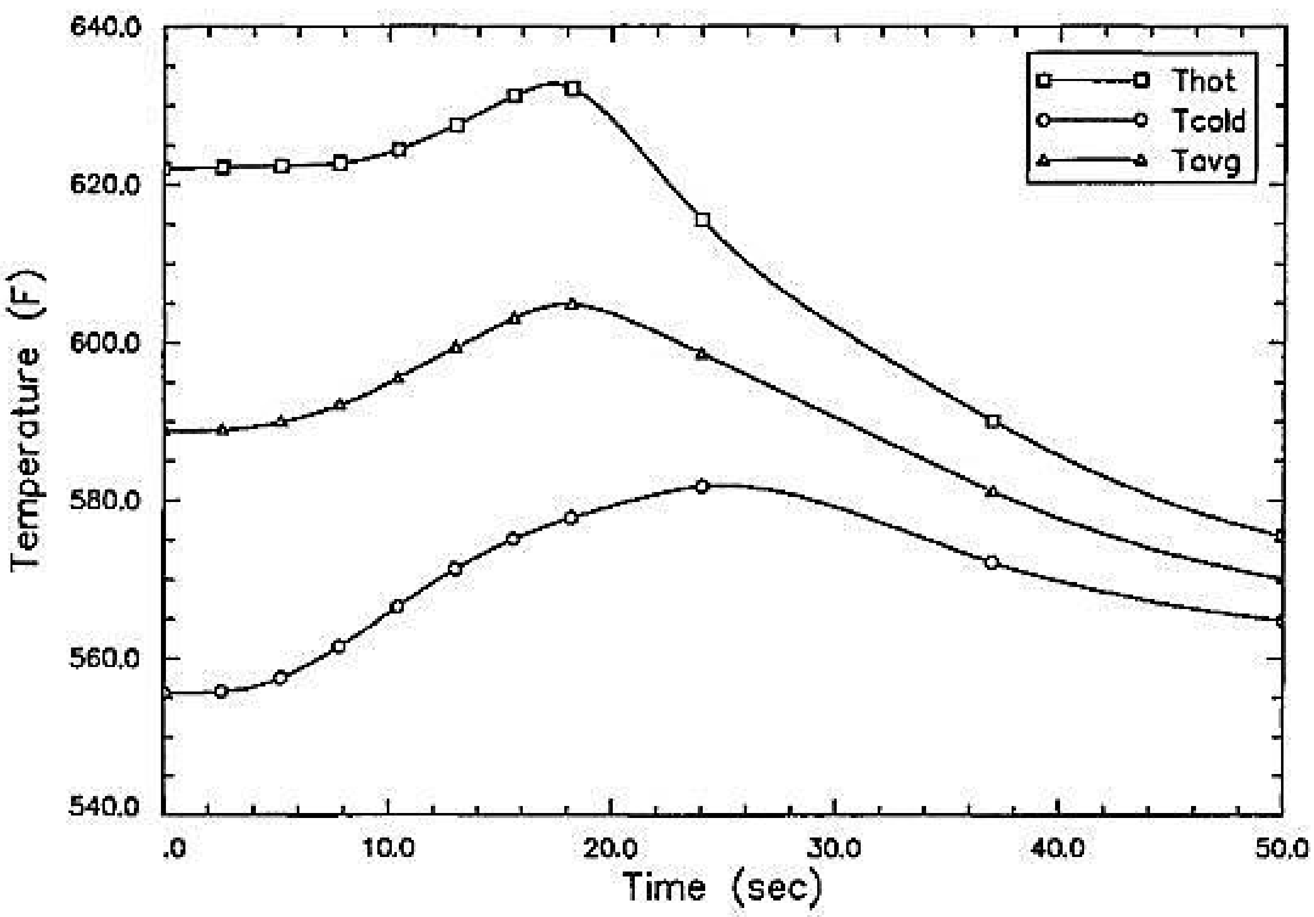


FIGURE 15.2.3-7

PRESSURIZATION PRESSURE AND LOWER HEAD PRESSURE FOR TURBINE TRIP MDNBR CASE

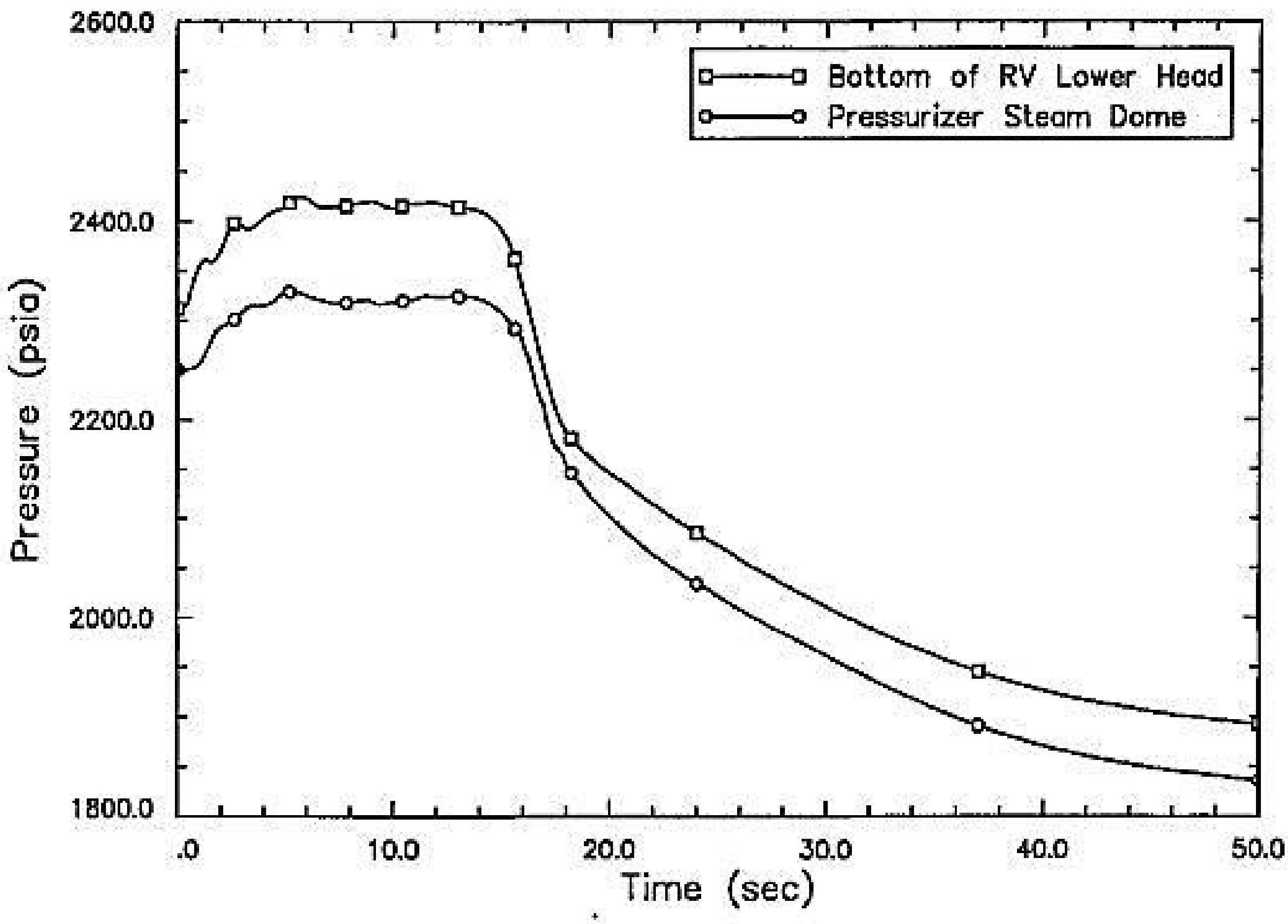


FIGURE 15.2.3-9

REACTOR POWER FOR TURBINE TRIP SECONDARY SIDE OVERPRESSURIZATION CASE

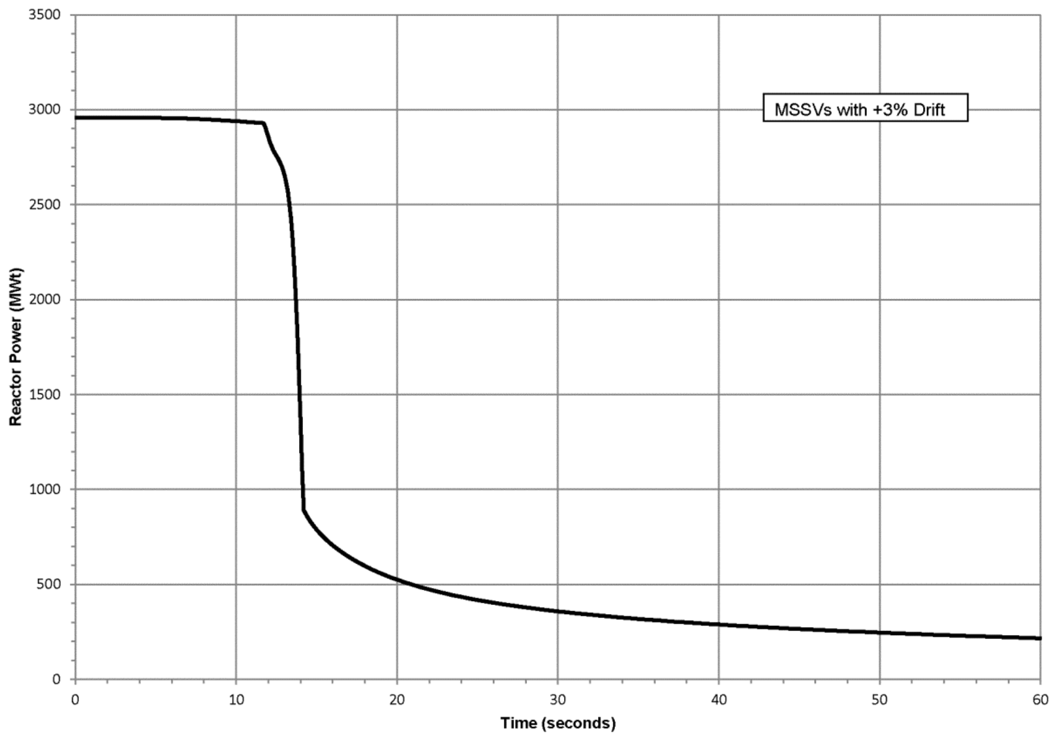


FIGURE 15.2.3-10

AVERAGE TEMPERATURES FOR TURBINE TRIP SECONDARY SIDE OVERPRESSURIZATION CASE

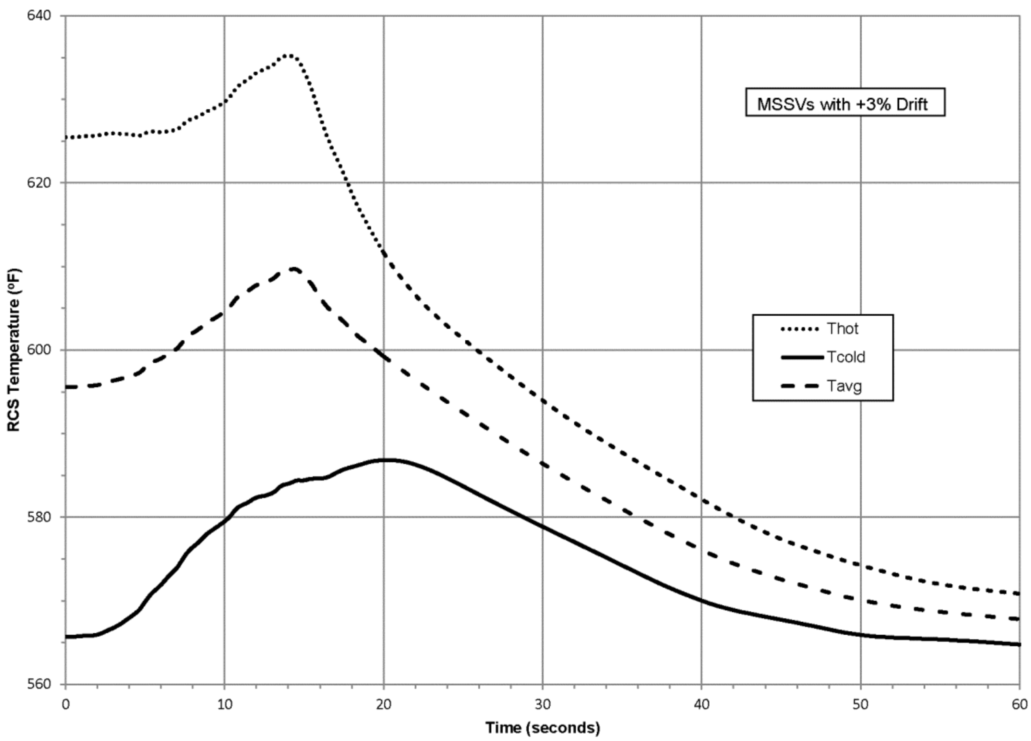


FIGURE 15.2.3-11

PRESSURIZER LEVEL FOR TURBINE TRIP SECONDARY SIDE OVERPRESSURIZATION CASE

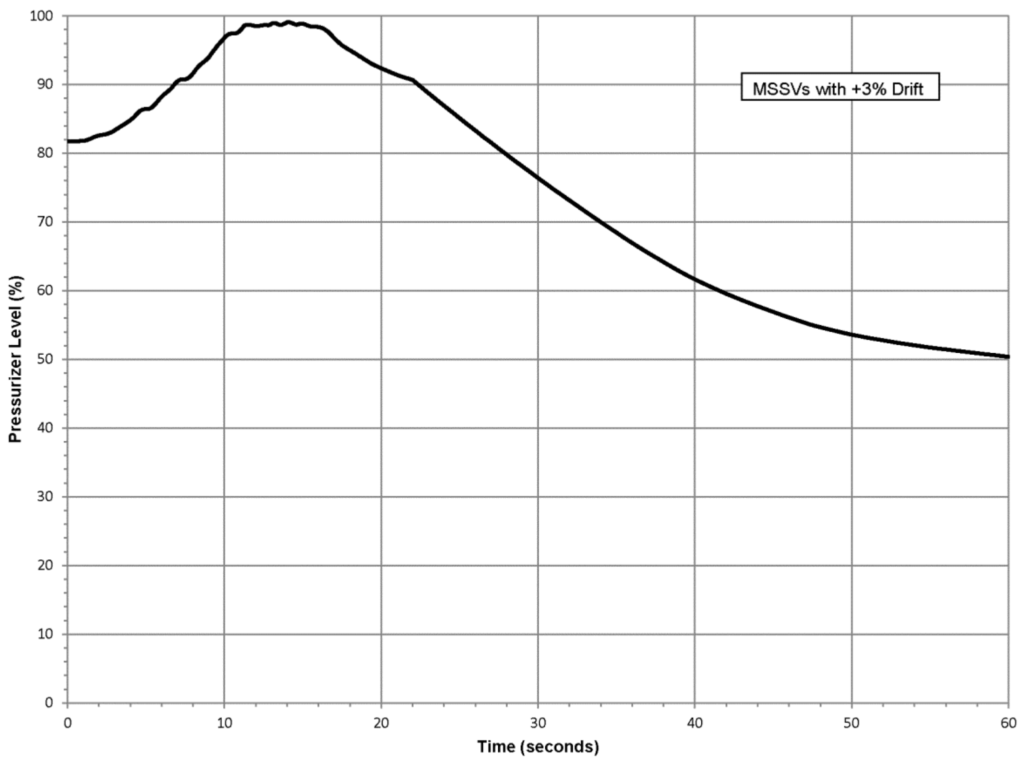


FIGURE 15.2.3-12

STEAM GENERATOR LOWER SHELL PRESSURE FOR TURBINE TRIP SECONDARY SIDE OVERPRESSURIZATION CASE

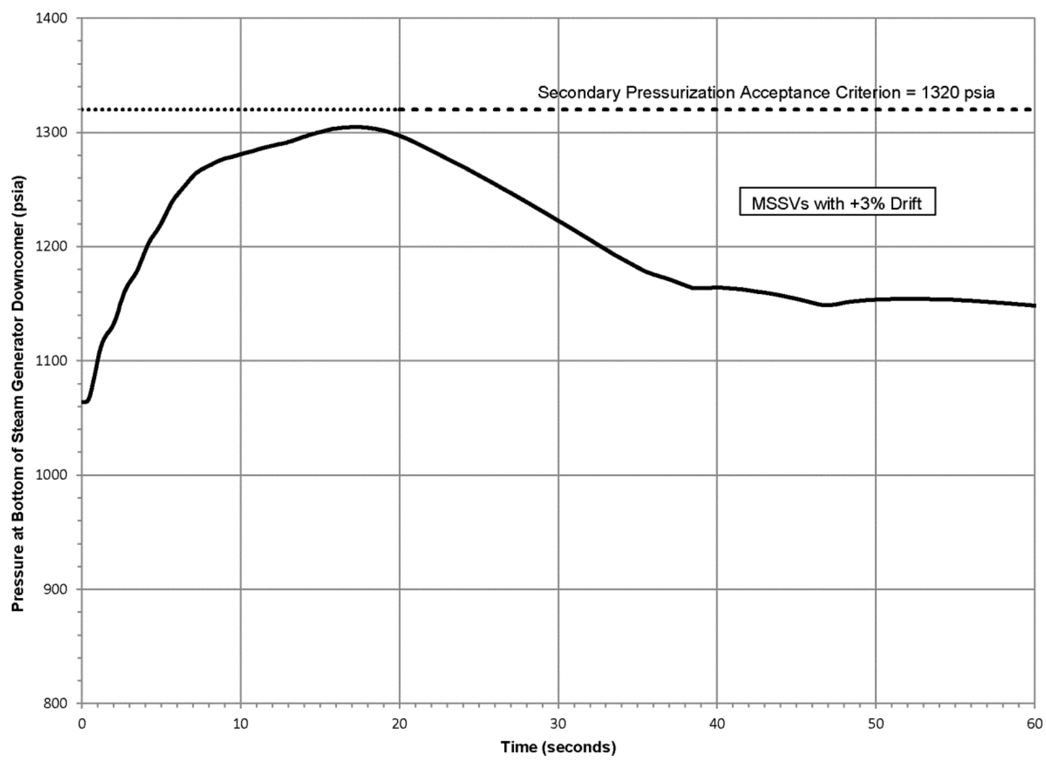


FIGURE 15.2.6-1

REACTOR POWER FOR LOSS OF NONEMERGENCY AC POWER TO THE STATION AUXILIARIES

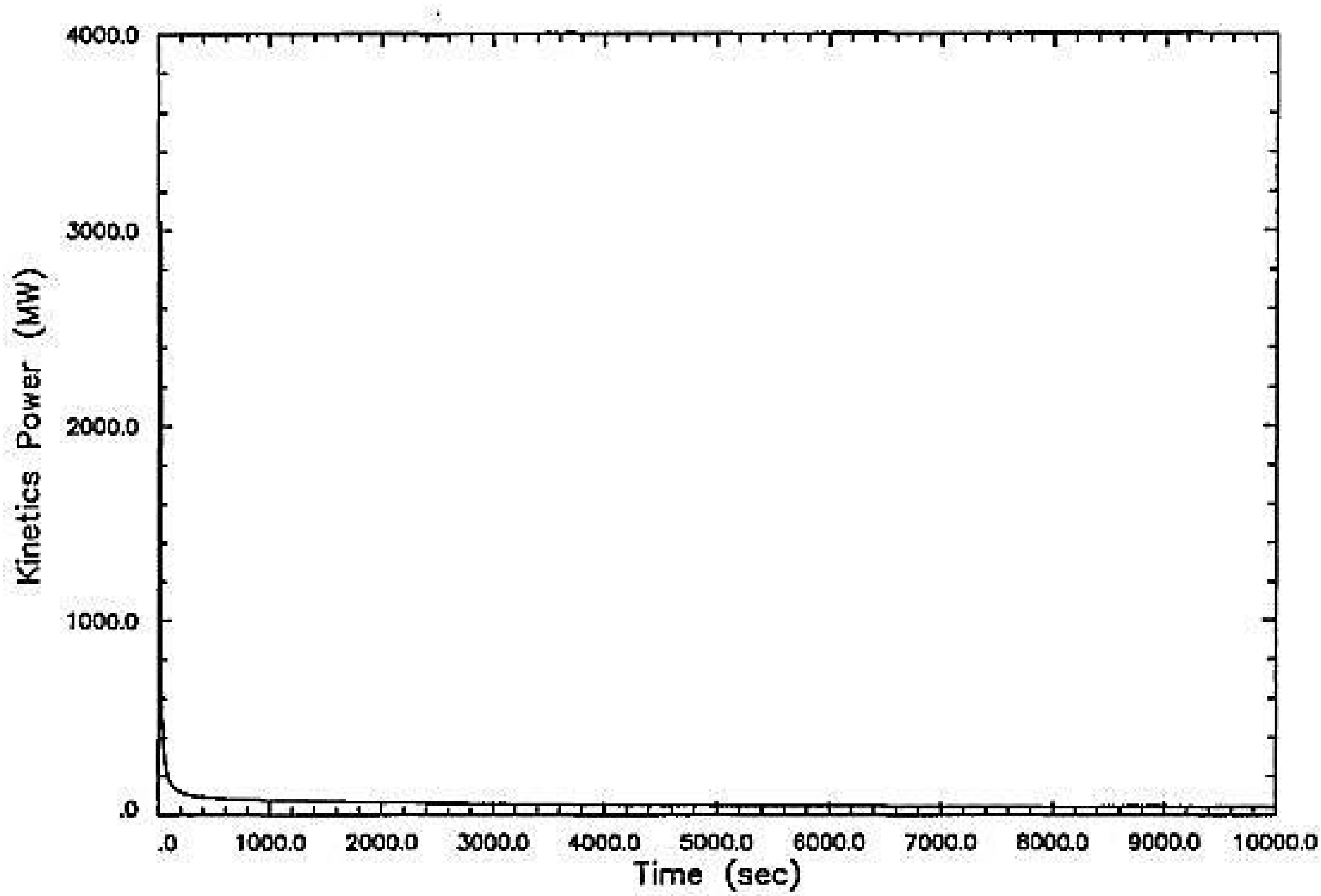


FIGURE 15.2.6-2

PRESSURIZER LEVEL FOR LOSS OF NONEMERGENCY AC POWER TO THE STATION AUXILIARIES

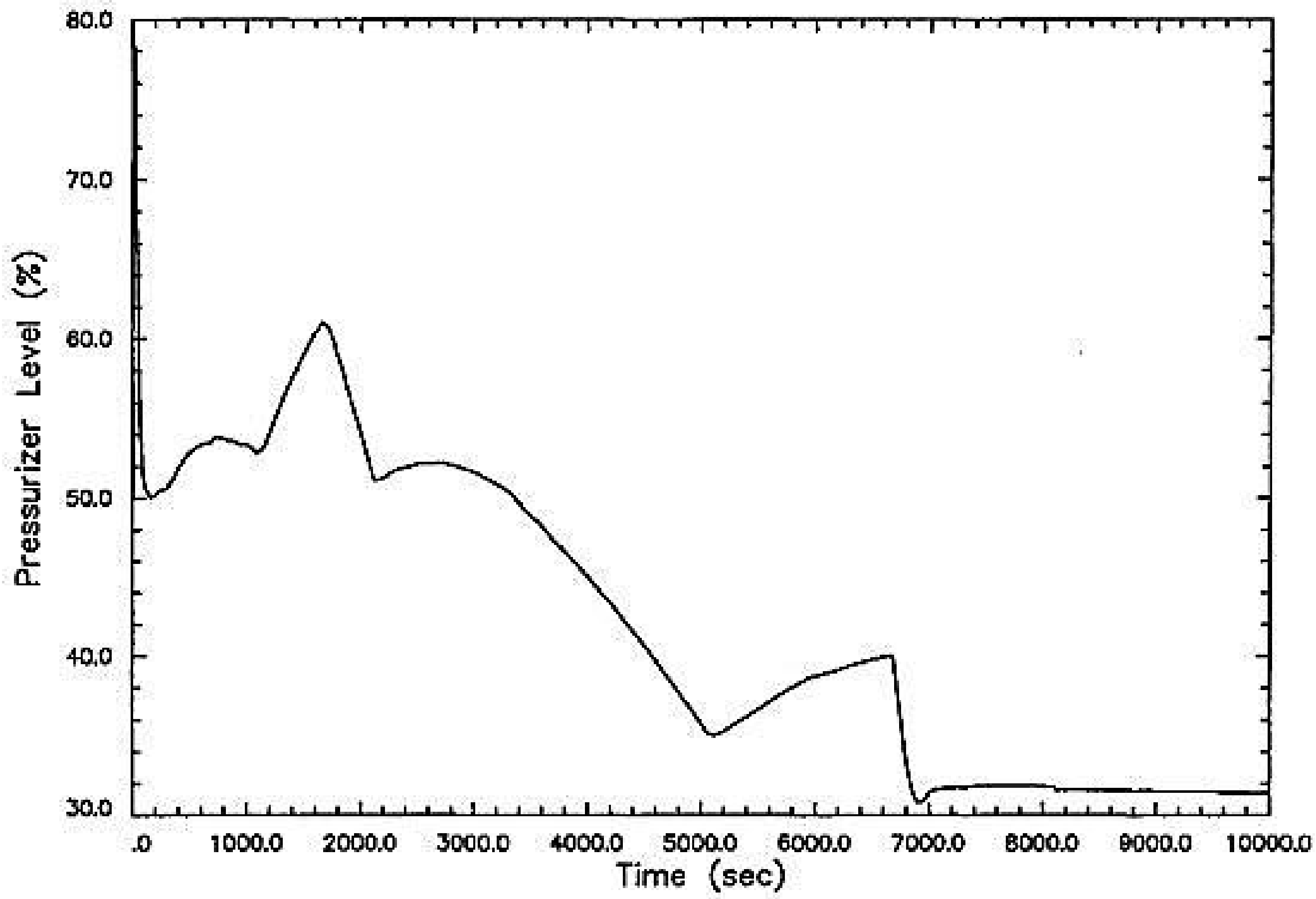


FIGURE 15.2.6-3

PRESSURIZER PRESSURE FOR LOSS OF NONEMERGENCY AC POWER TO THE STATION AUXILIARIES

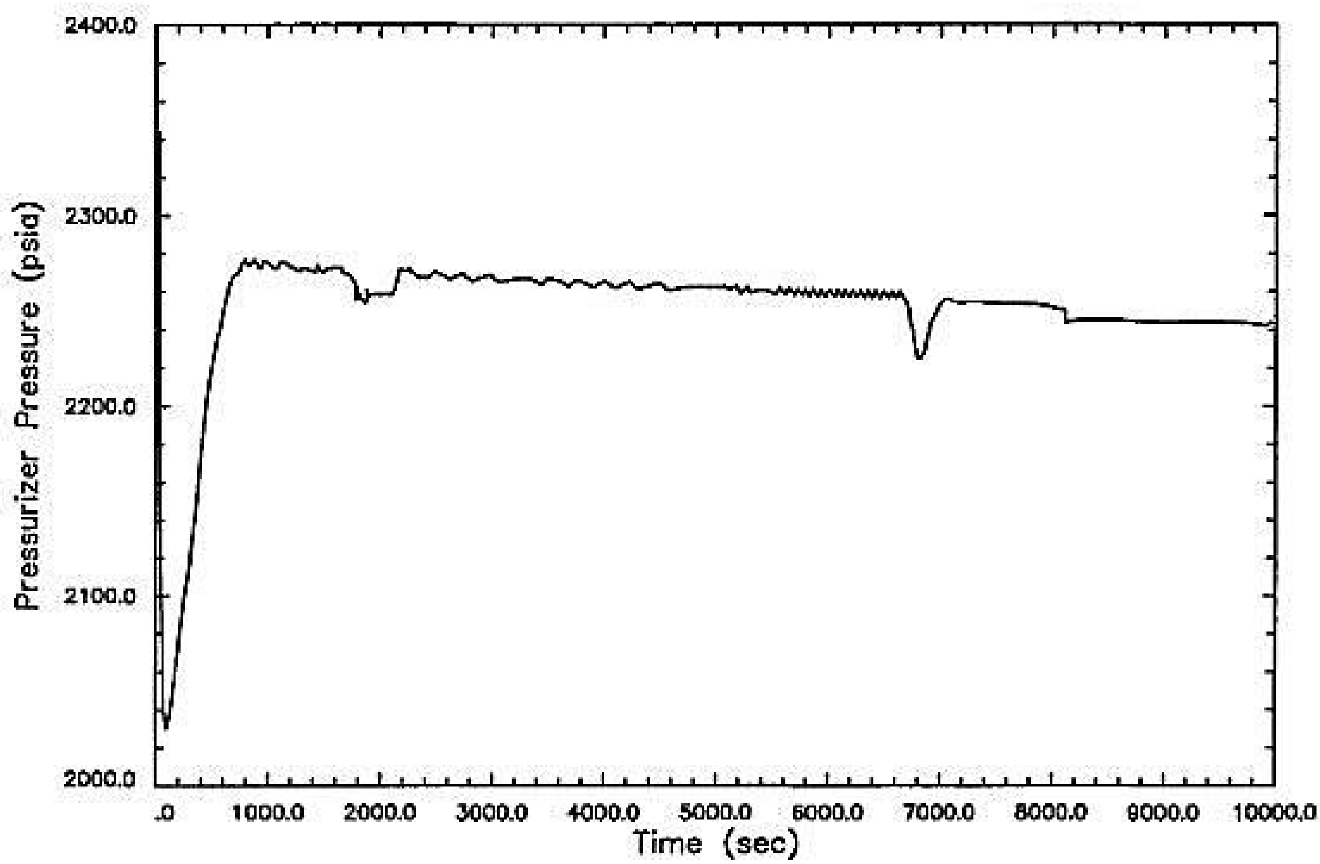


FIGURE 15.2.6-4

AUCTIONEERED T_{AVG}, LOOP 3 T_{COLD} AND T_{HOT} FOR LOSS OF NONEMERGENCY AC POWER TO THE STATION AUXILIARIES
(LOOP 3 IS LOOP WITHOUT AUXILIARY FEEDWATER FLOW)

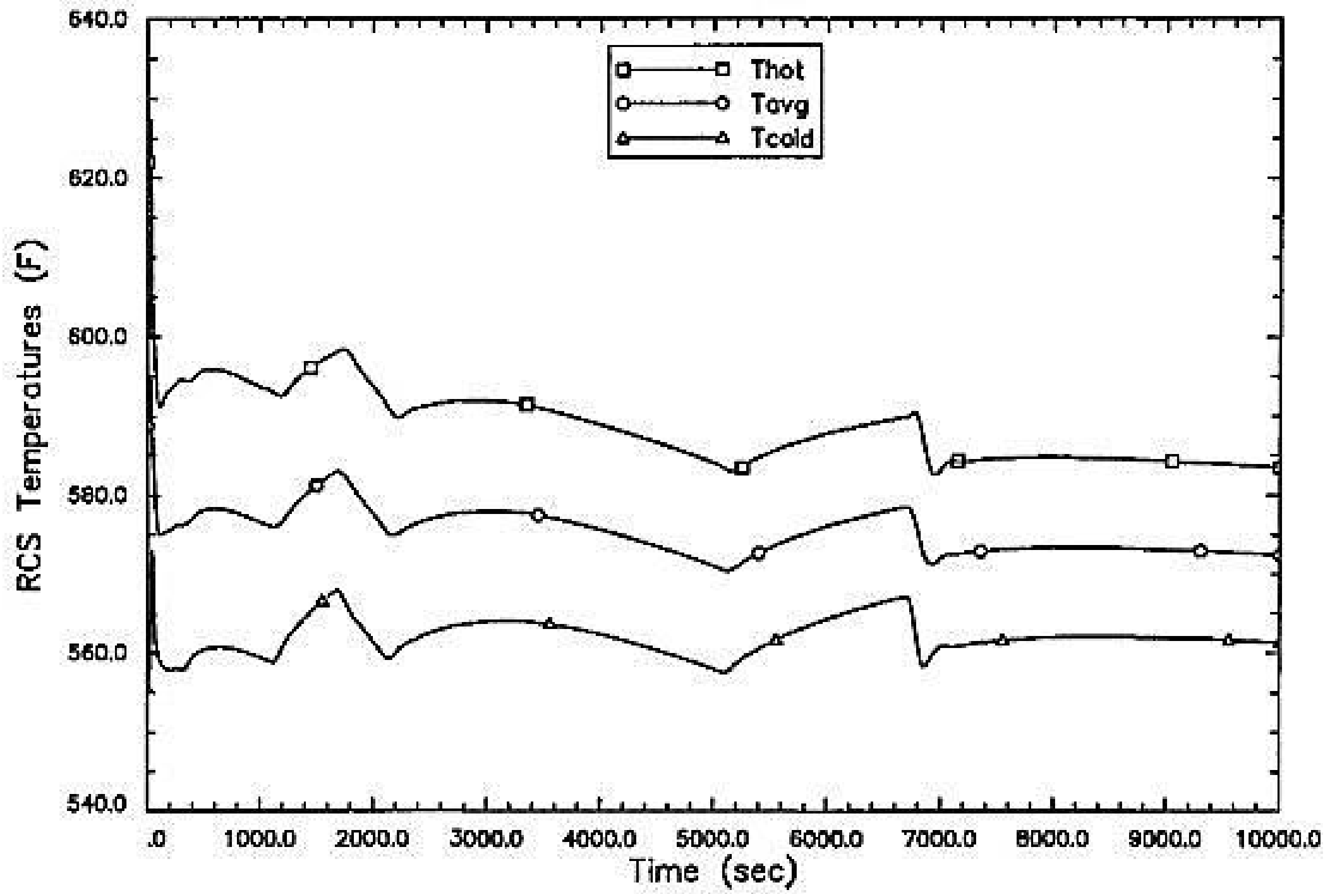


FIGURE 15.2.6-5

STEAM GENERATOR MASS INVENTORY FOR LOSS OF NONEMERGENCY AC POWER TO THE STATION AUXILIARIES

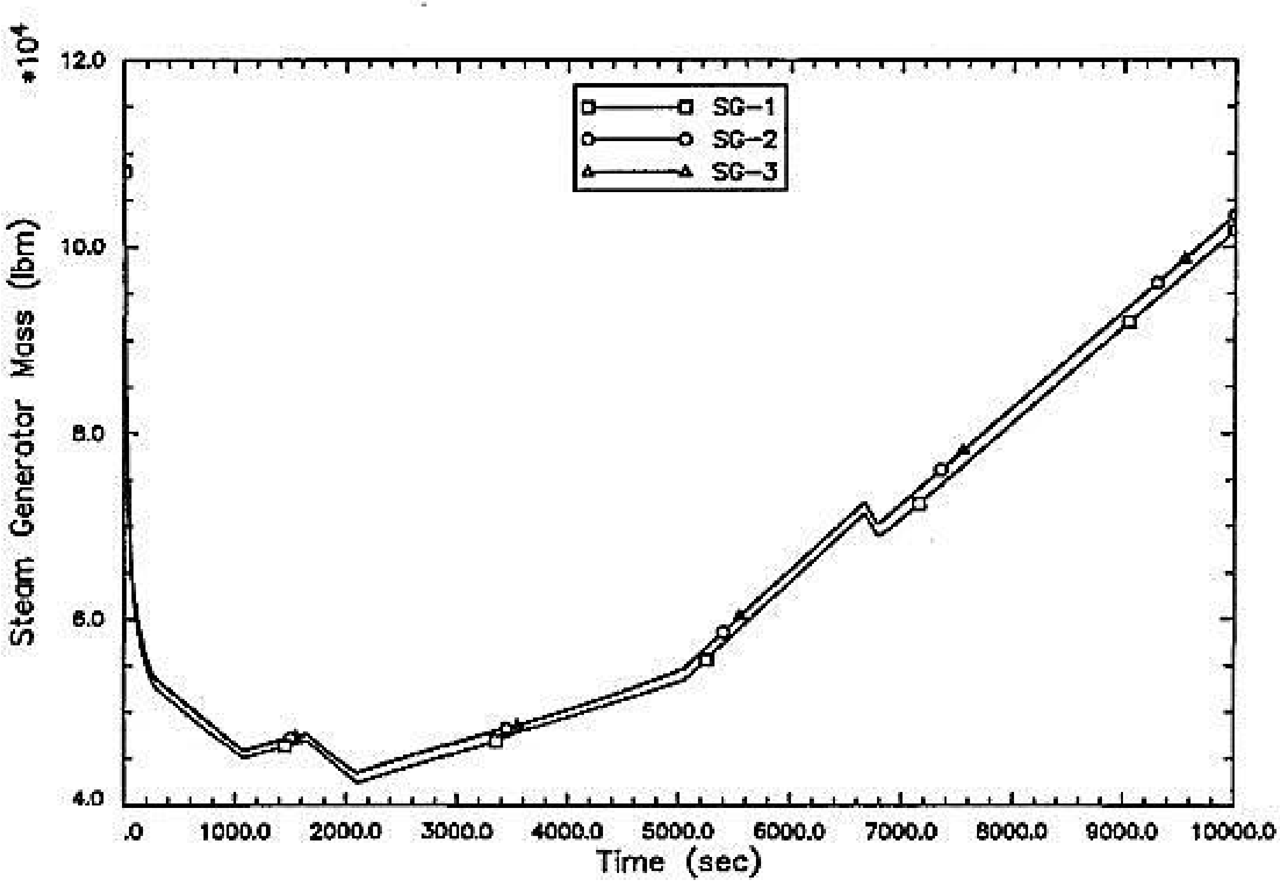


FIGURE 15.2.7-1

REACTOR POWER FOR LOSS OF NORMAL FEEDWATER FLOW

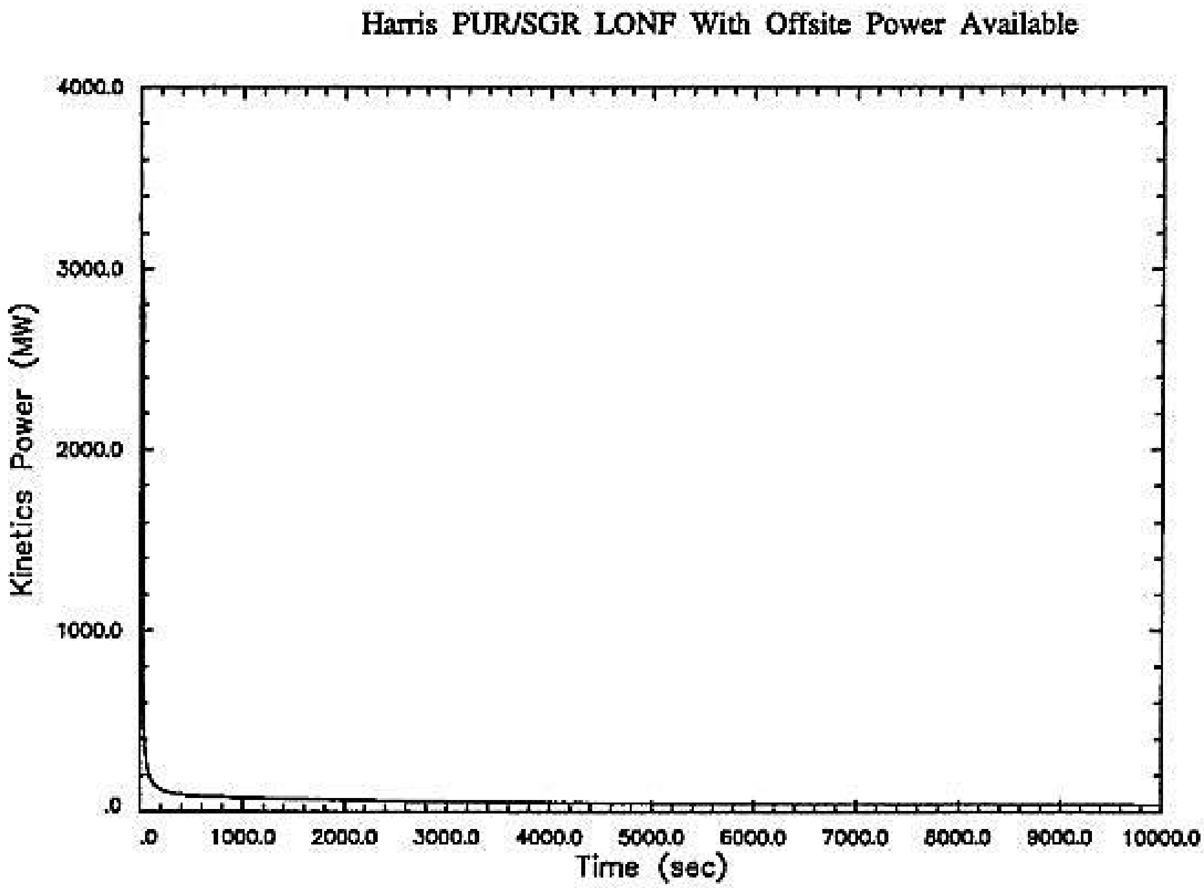


FIGURE 15.2.7-2

PRESSURIZER LEVEL FOR LOSS OF NORMAL FEEDWATER FLOW

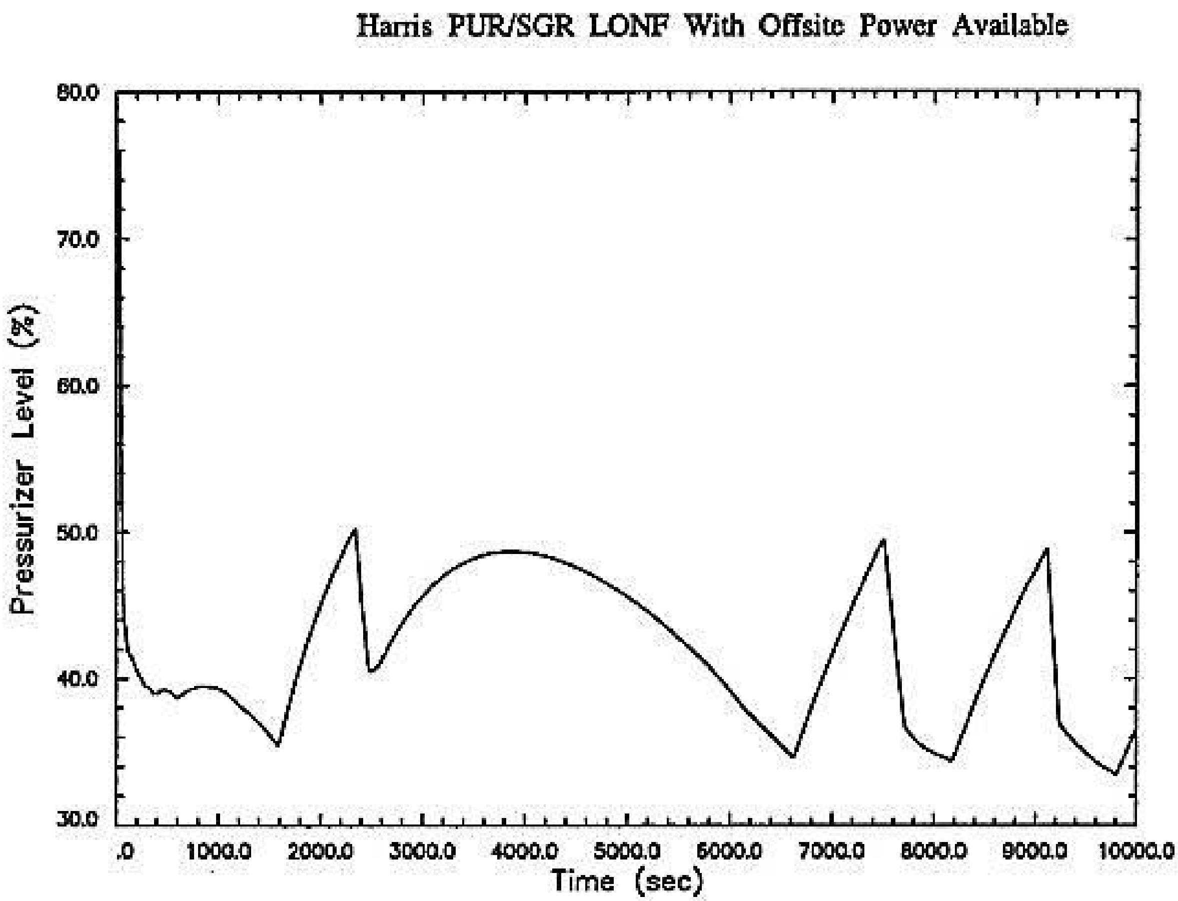


FIGURE 15.2.7-3

PRESSURIZER PRESSURE FOR LOSS OF NORMAL FEEDWATER FLOW

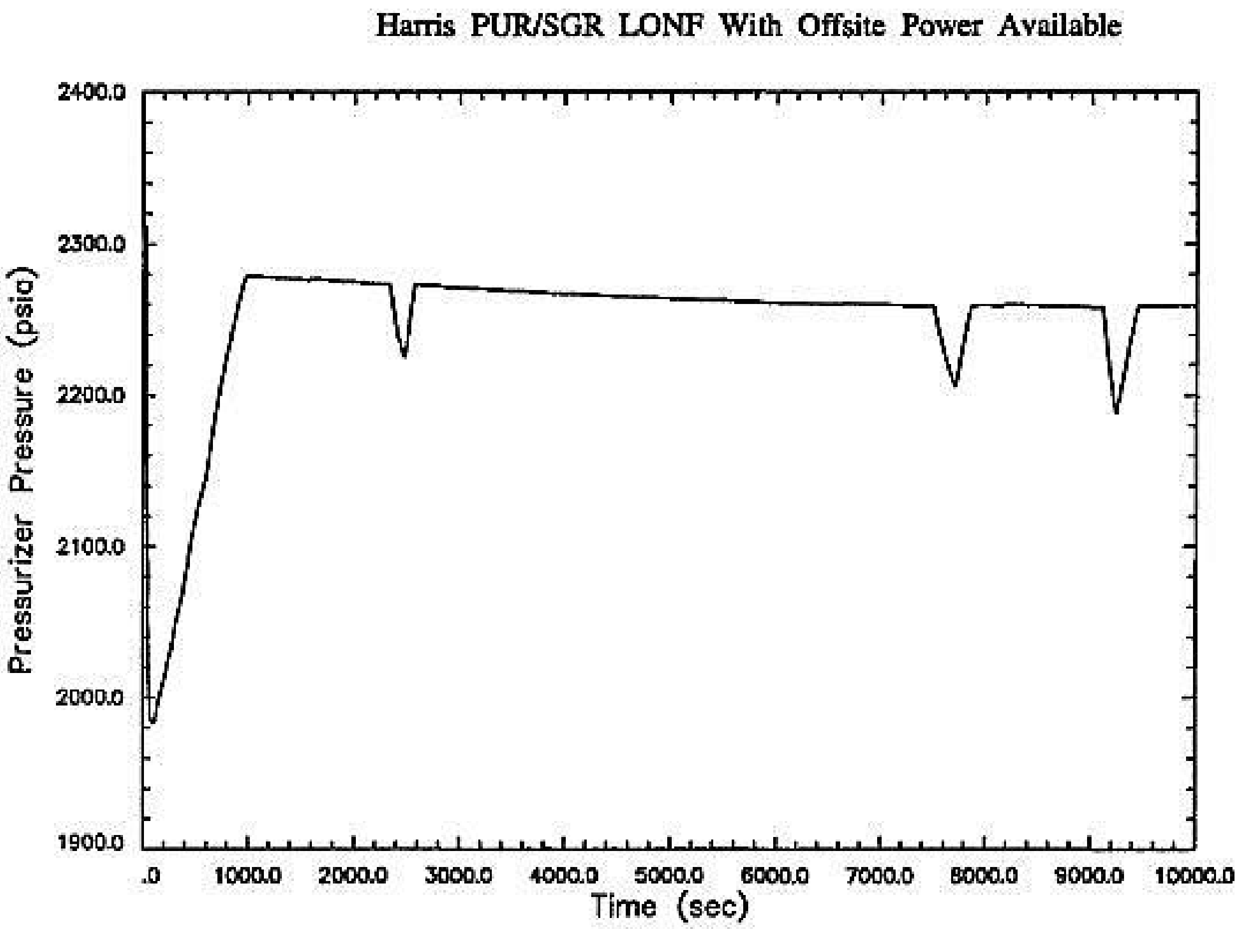


FIGURE 15.2.7-4

AUCTIONEERED T_{AVG} LOOP 3 T_{COLD} AND T_{HOT} FOR LOSS OF NORMAL FEEDWATER FLOW

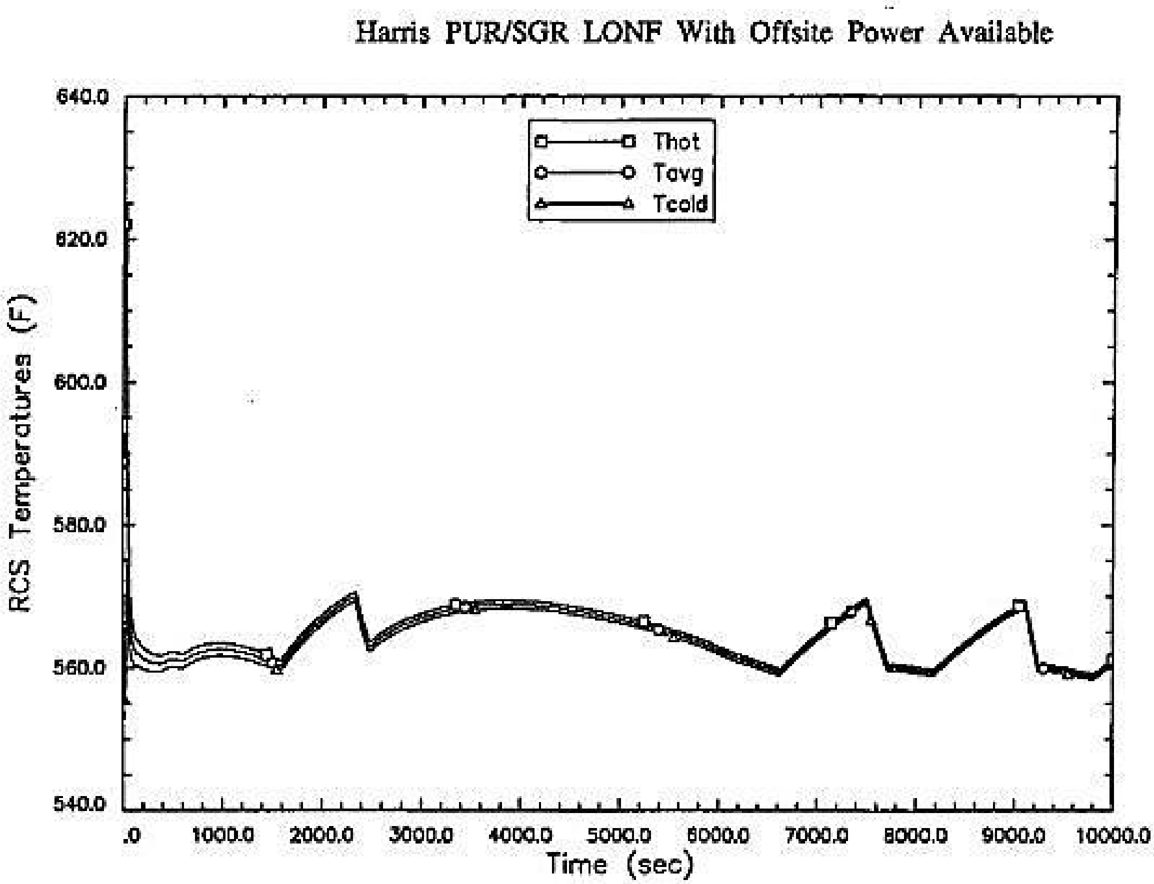


FIGURE 15.2.7-5

STEAM GENERATOR MASS INVENTORY FOR LOSS OF NORMAL FEEDWATER FLOW

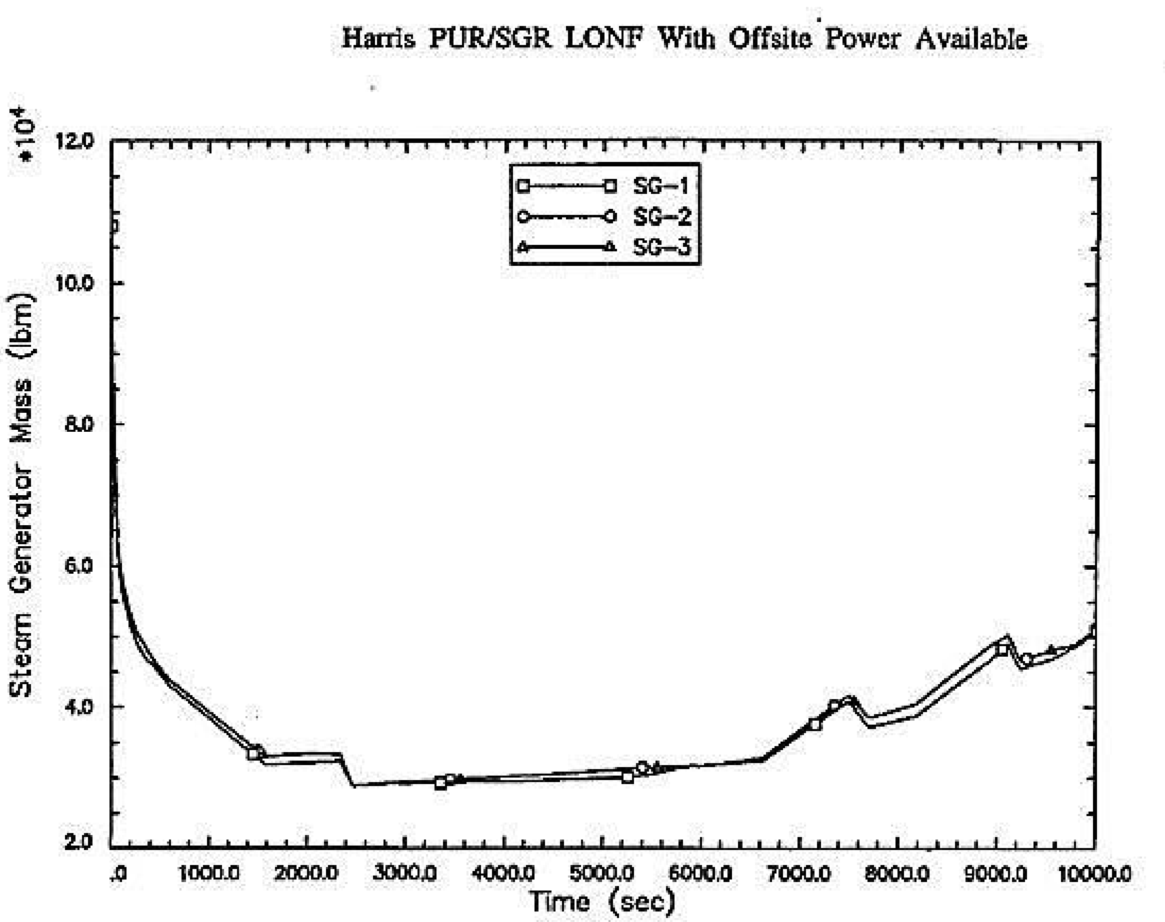


FIGURE 15.2.8-1

REACTOR POWER FOR FEEDWATER SYSTEM PIPE BREAK WITH OFFSITE POWER

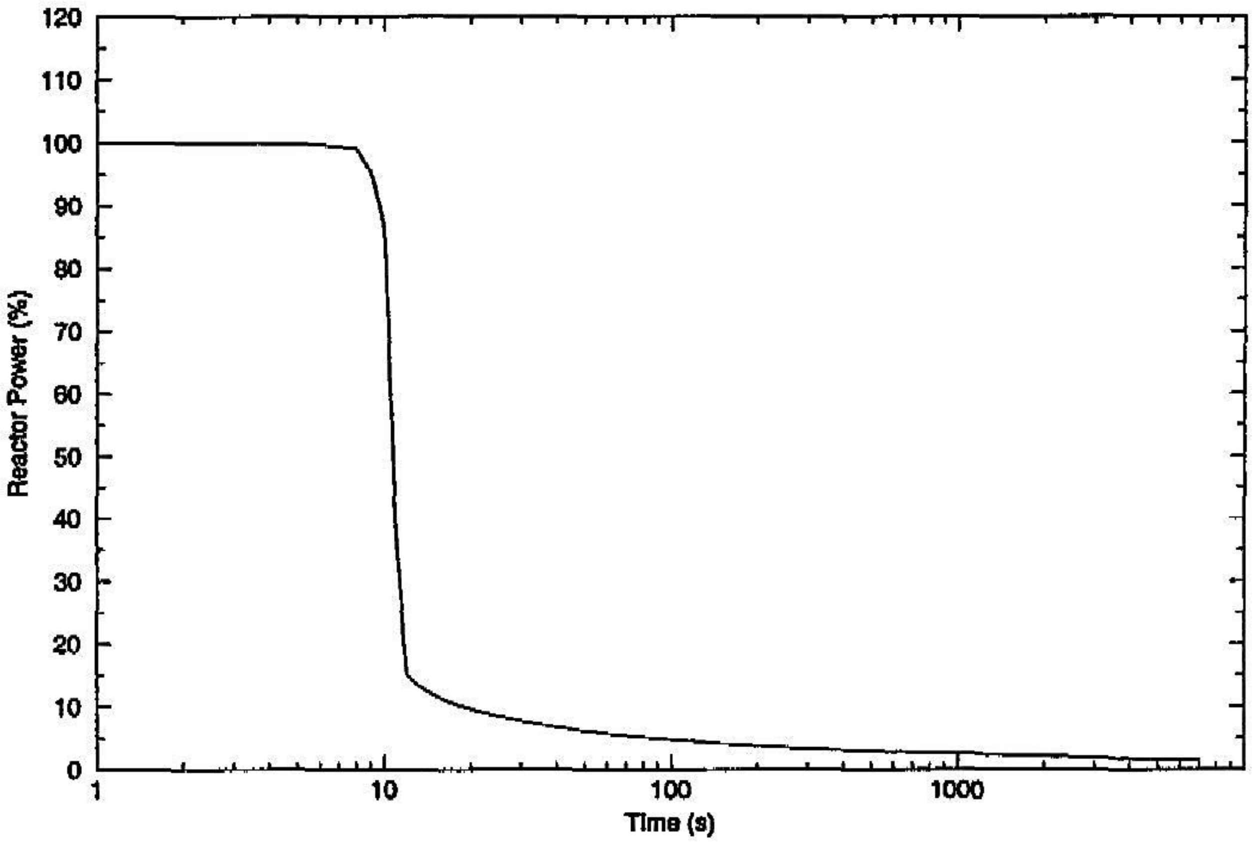


FIGURE 15.2.8-2

PRESSURIZER LEVEL FOR FEEDWATER SYSTEM PIPE BREAK WITH OFFSITE POWER

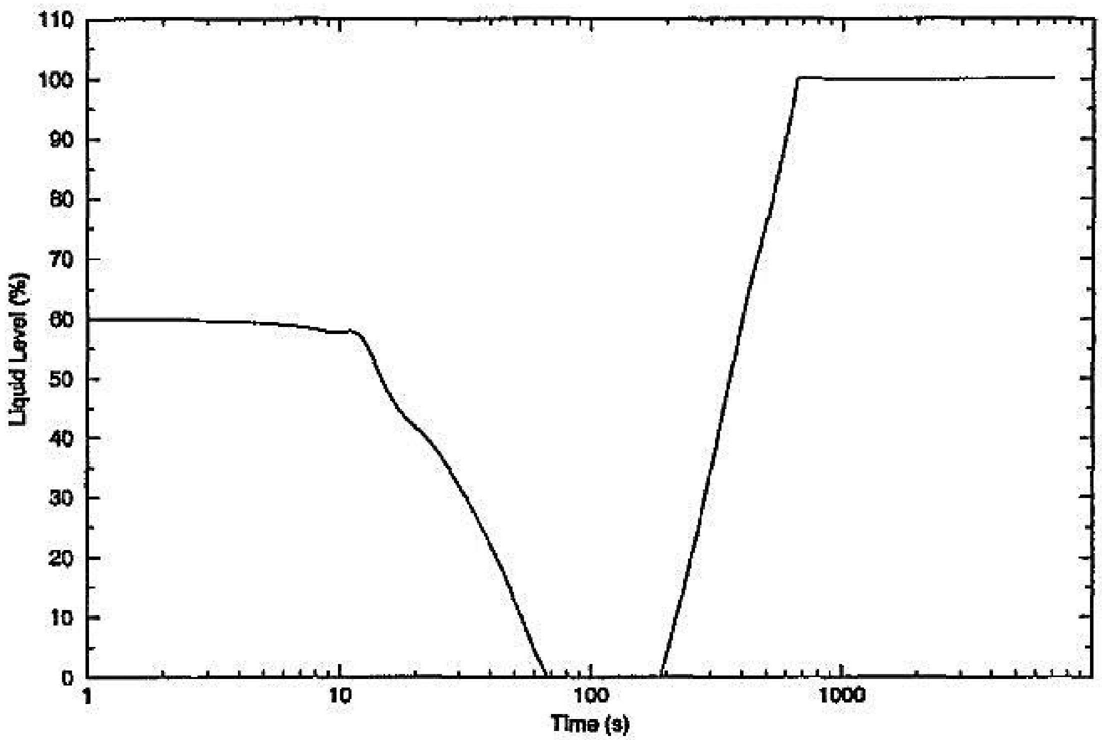


FIGURE 15.2.8-3

PRESSURIZER PRESSURE FOR FEEDWATER SYSTEM PIPE BREAK WITH OFFSITE POWER

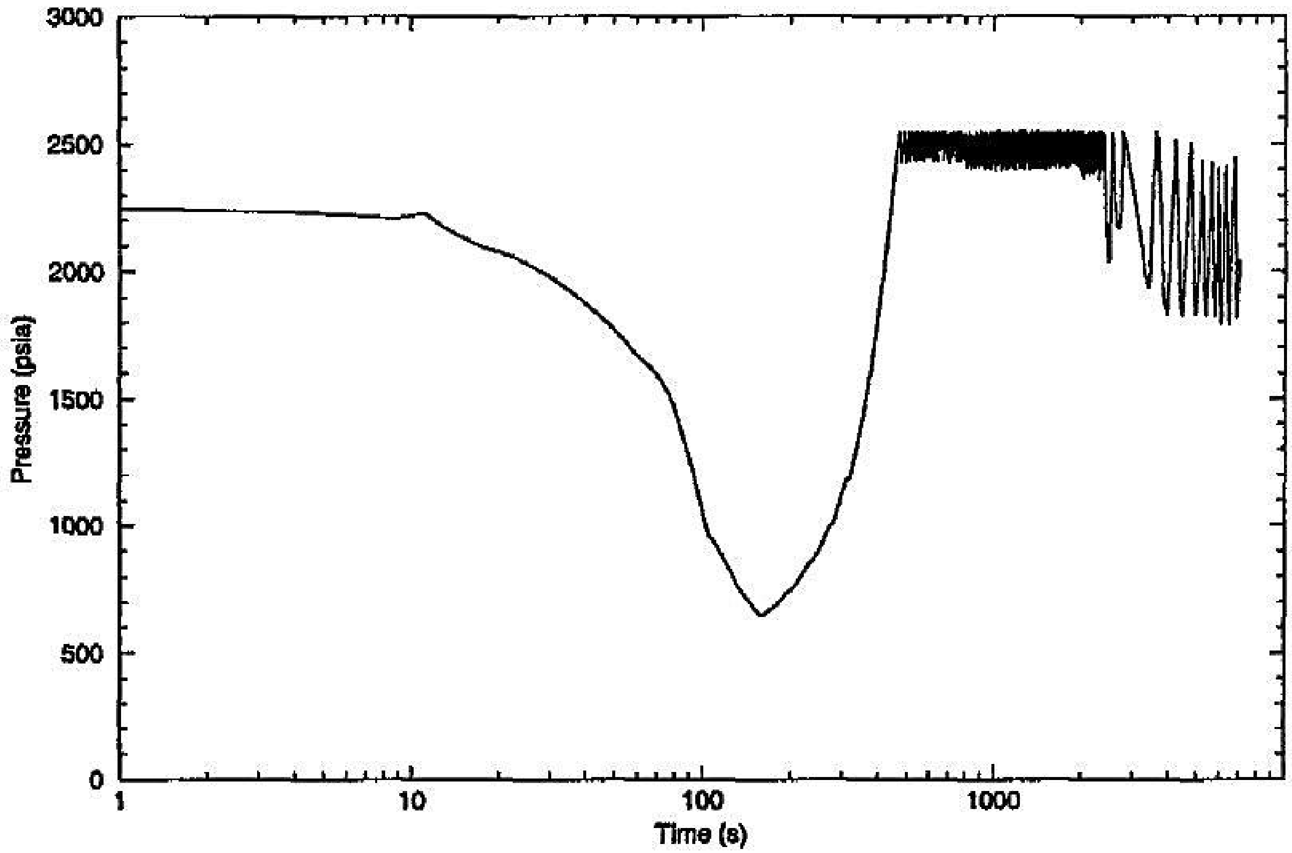


FIGURE 15.2.8-4

LOOP 1 PRIMARY SYSTEM TEMPERATURE FOR FEEDWATER SYSTEM PIPE BREAK WITH OFFSITE POWER

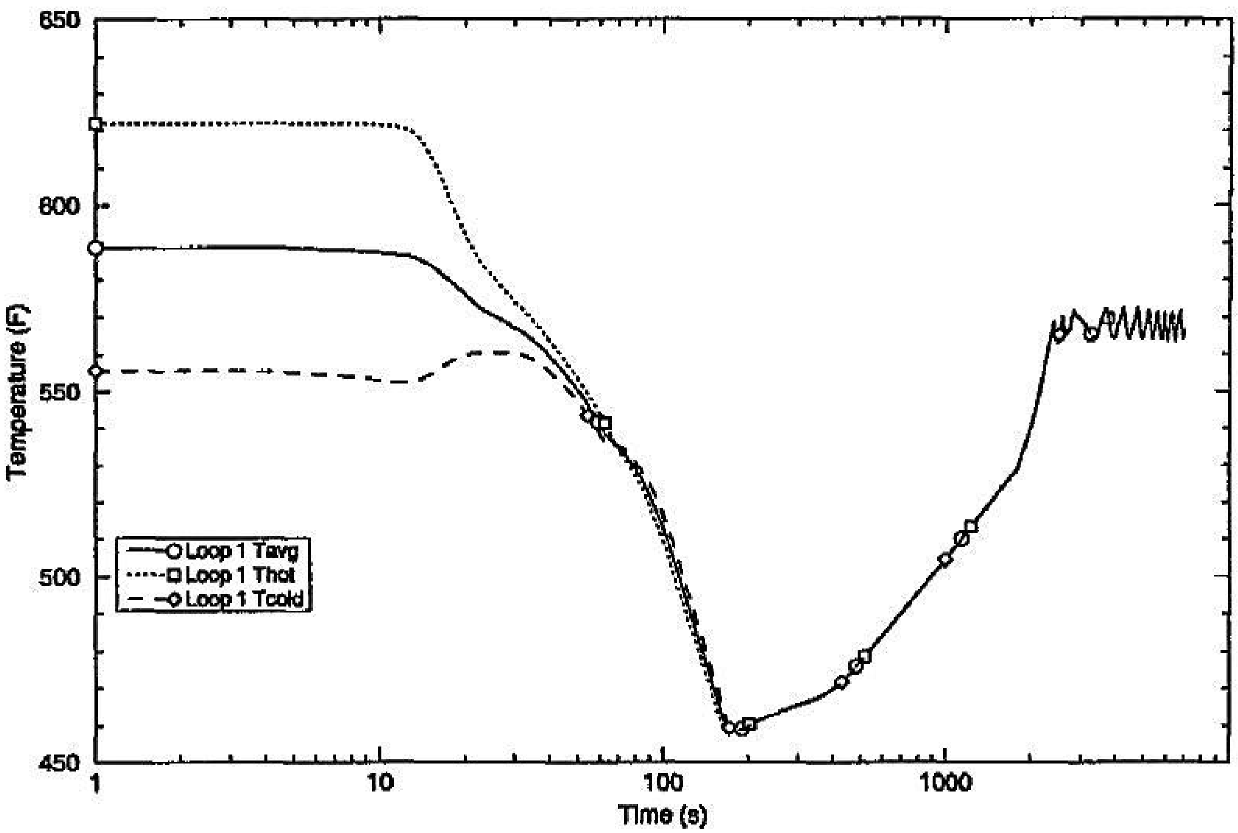


FIGURE 15.2.8-5

LOOP 2 PRIMARY SYSTEM TEMPERATURES FOR FEEDWATER SYSTEM PIPE BREAK WITH OFFSITE POWER

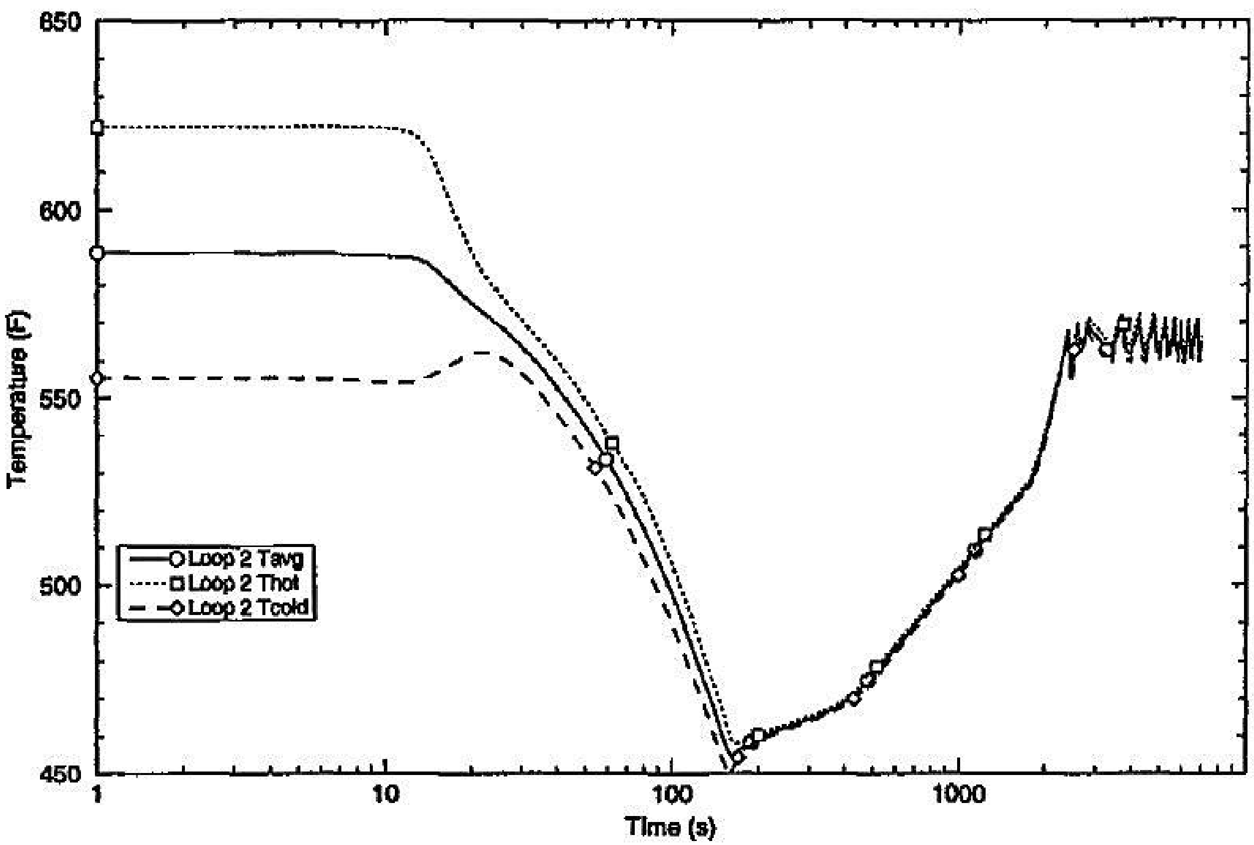


FIGURE 15.2.8-6

LOOP 3 PRIMARY SYSTEM TEMPERATURES FOR FEEDWATER SYSTEM PIPE BREAK WITH OFFSITE POWER

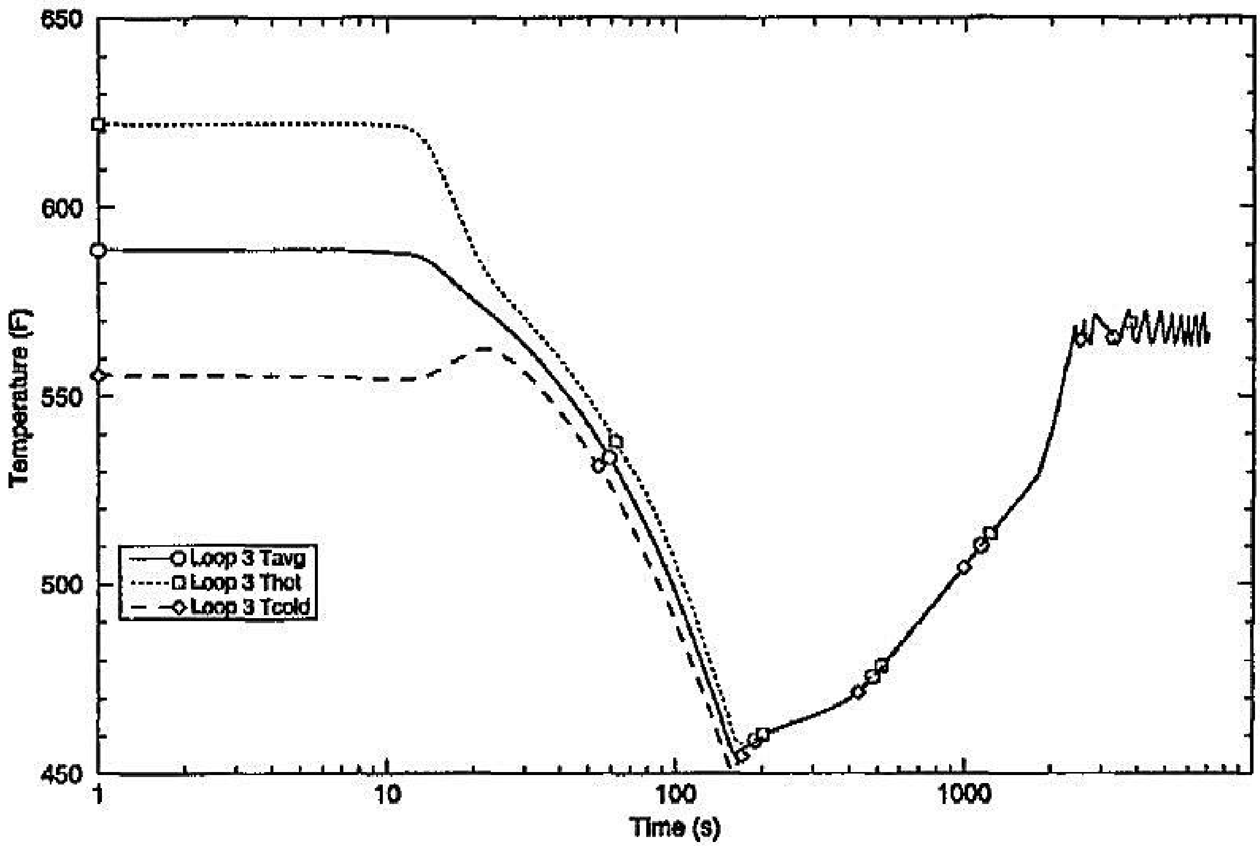


FIGURE 15.2.8-7

STEAM GENERATOR PRESSURE FOR FEEDWATER SYSTEM PIPE BREAK WITH OFFSITE POWER

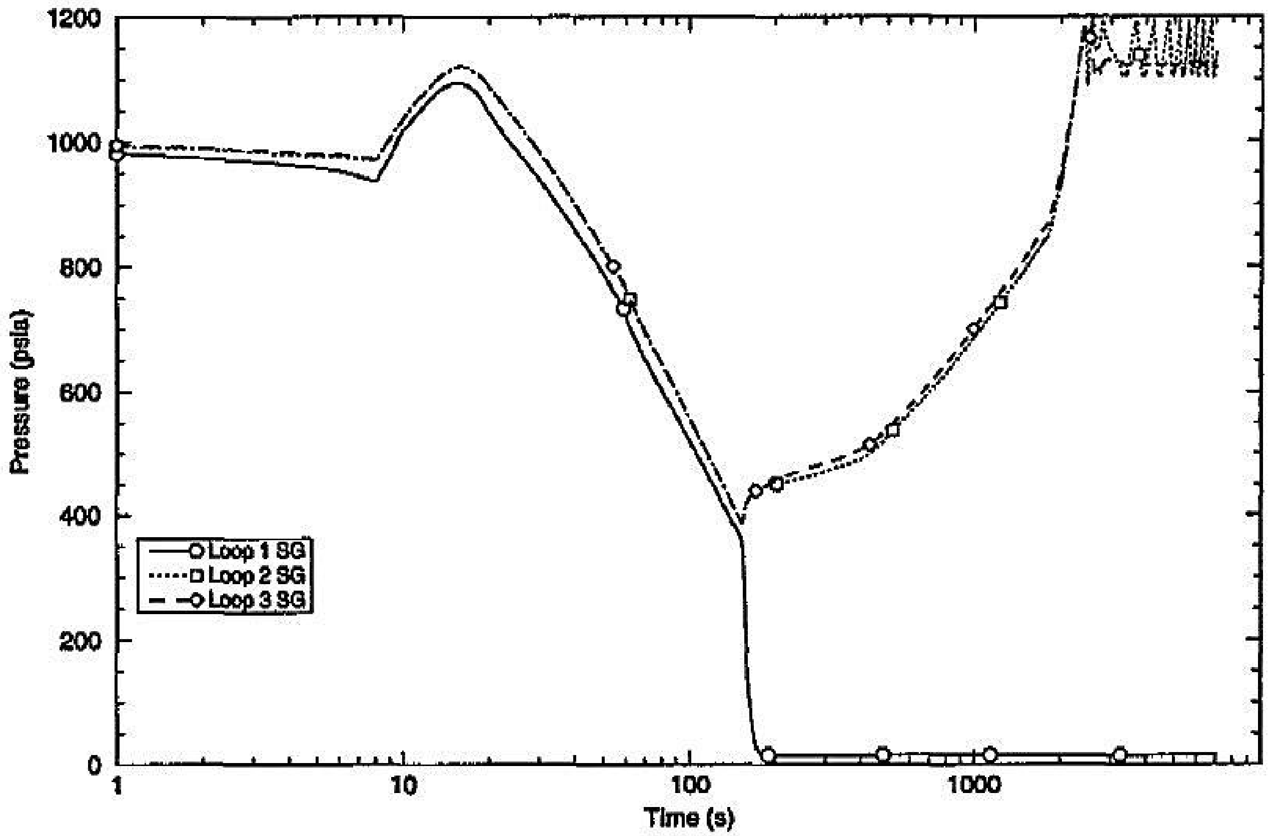


FIGURE 15.2.8-8

STEAM GENERATOR NARROW RANGE LEVEL FOR FEEDWATER SYSTEM PIPE BREAK WITH OFFSITE POWER

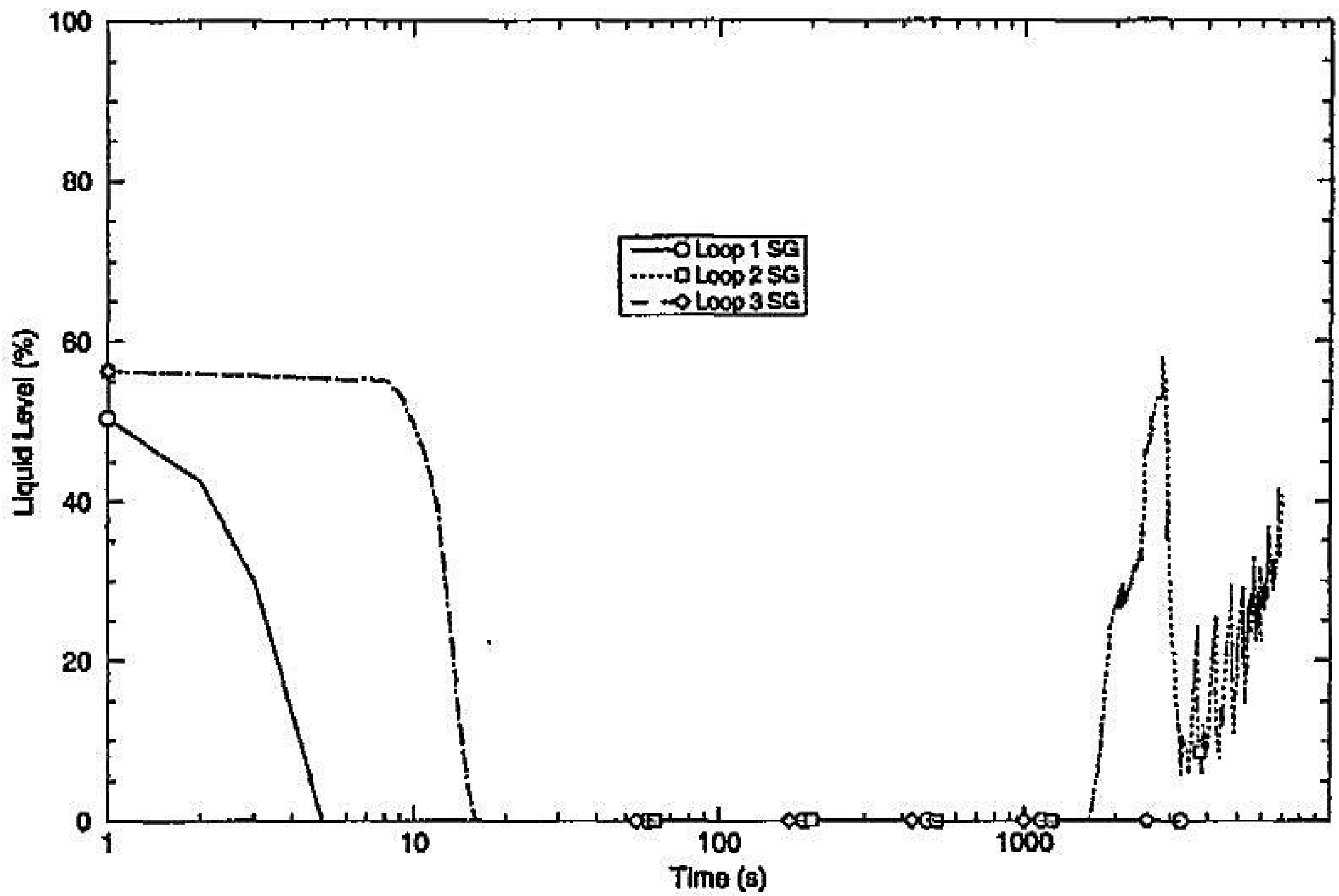


FIGURE 15.2.8-9

REACTOR COOLANT SYSTEM FLOW FOR FEEDWATER SYSTEM PIPE BREAK WITH OFFSITE POWER

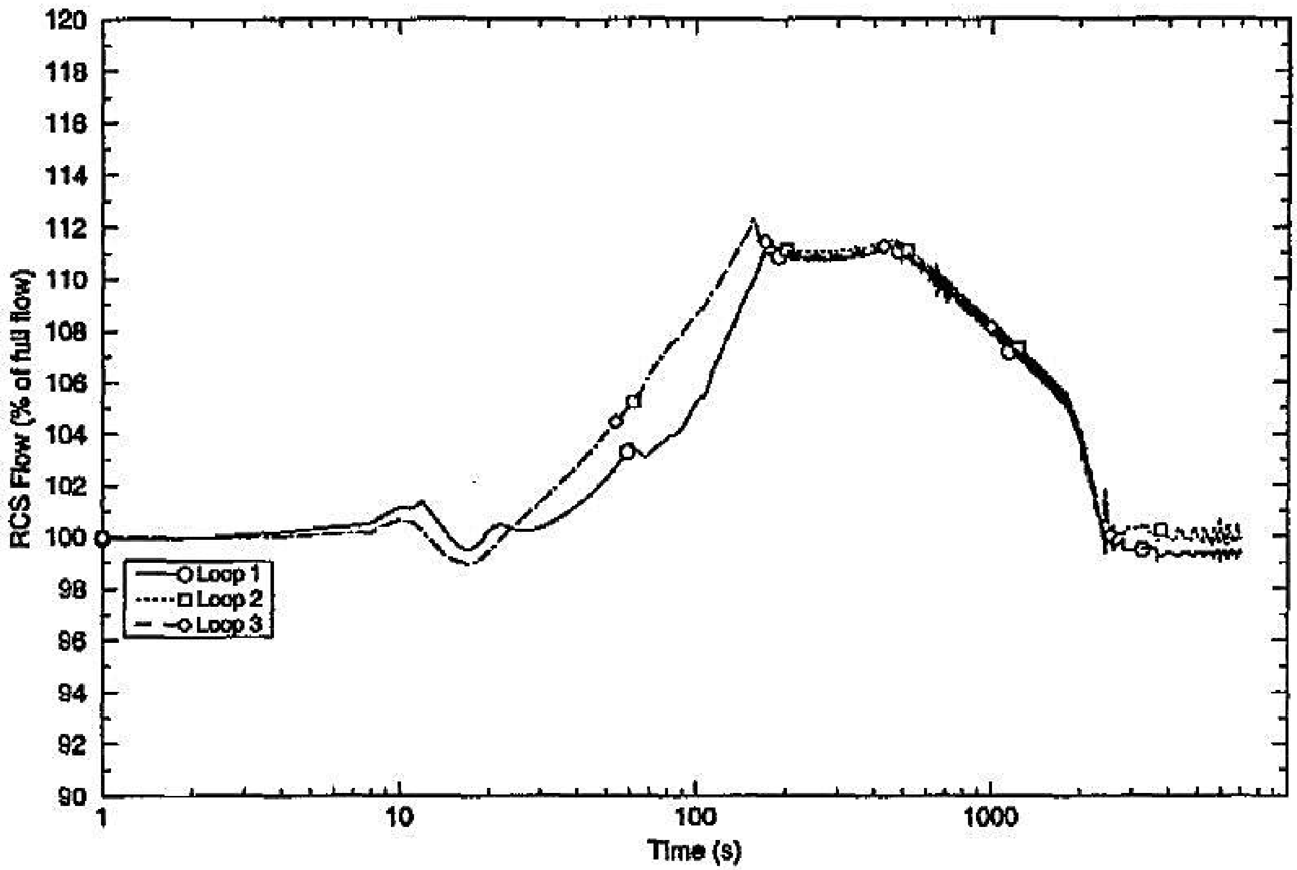


FIGURE 15.2.8-10

TOTAL PRESSURIZER RELIEF FLOW FOR FEEDWATER SYSTEM PIPE BREAK WITH OFFSITE POWER

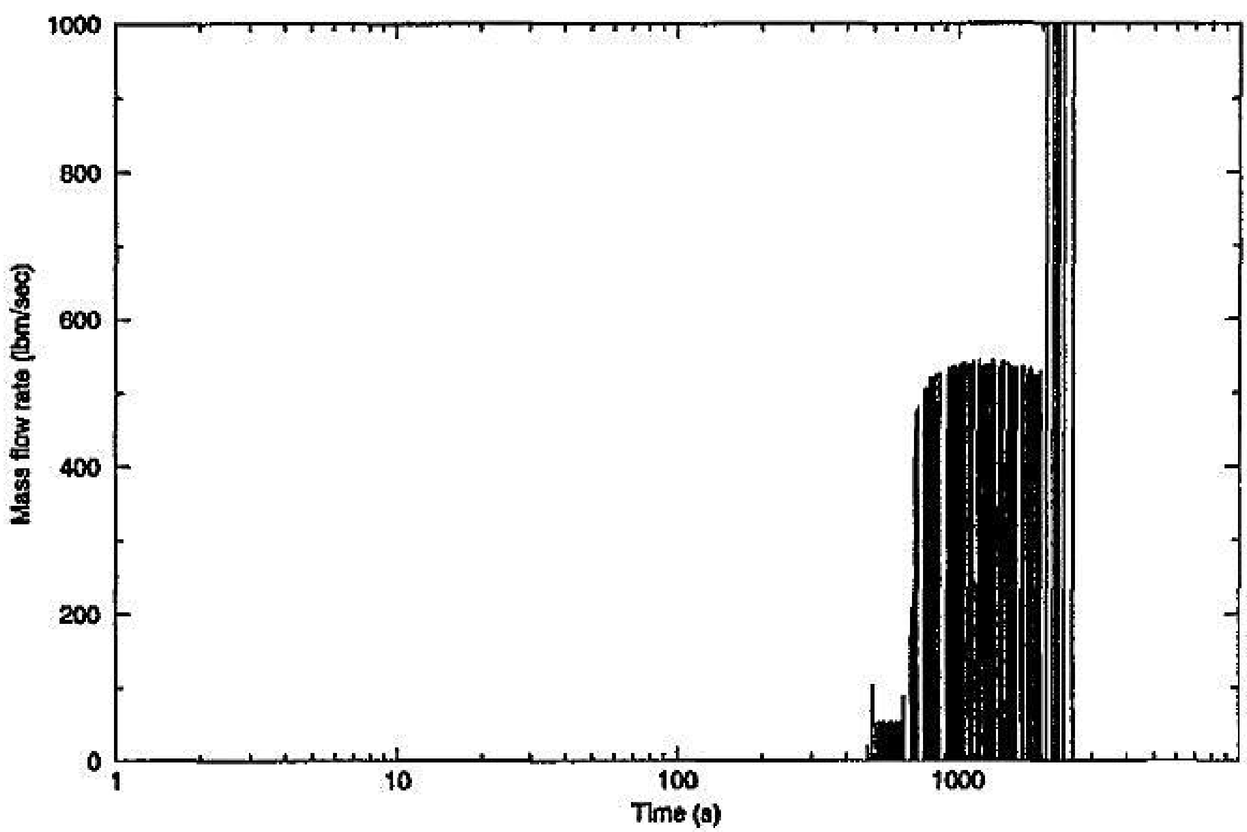


FIGURE 15.2.8-11

REACTOR POWER FOR FEEDWATER SYSTEM PIPE BREAK WITH LOSS OF OFFSITE POWER

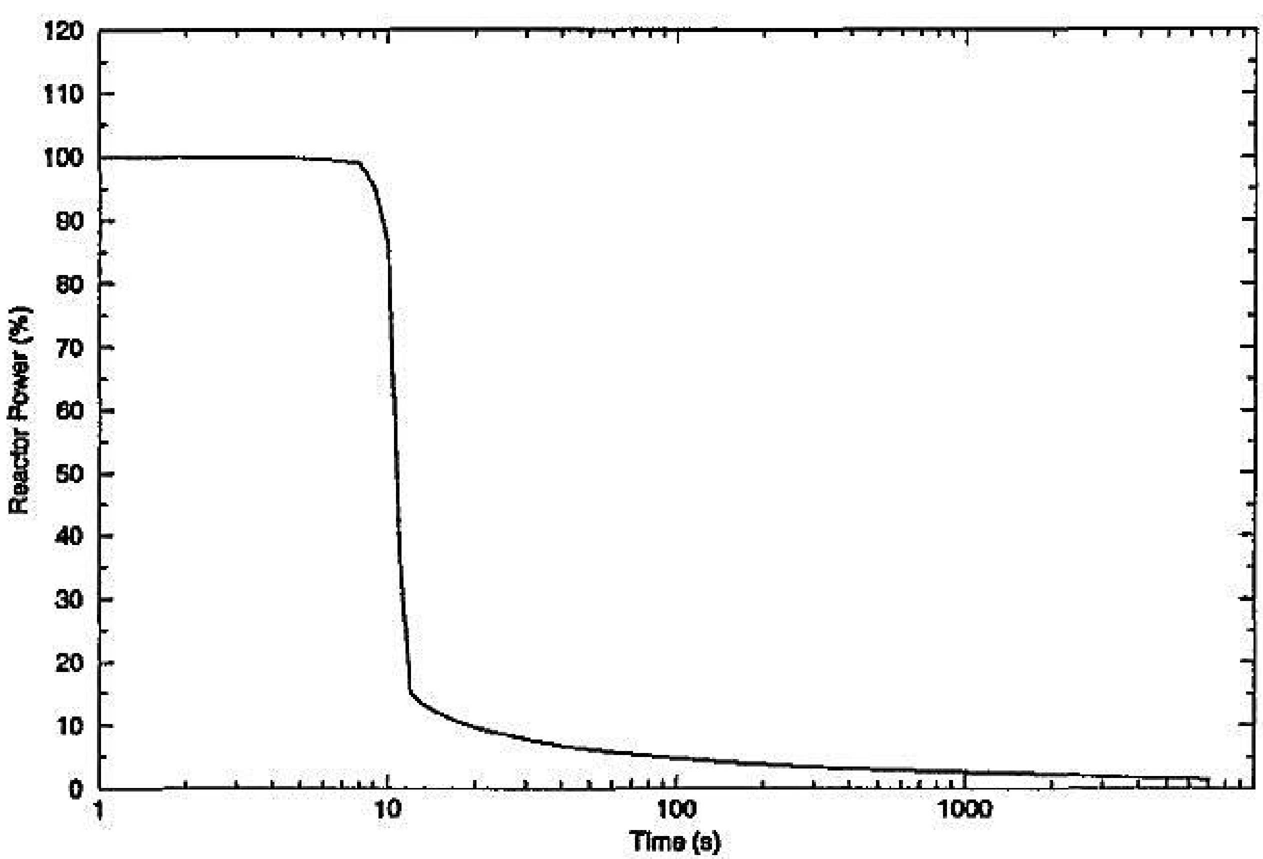


FIGURE 15.2.8-12

PRESSURIZER LEVEL FOR FEEDWATER SYSTEM PIPE BREAK WITH LOSS OF OFFSITE POWER

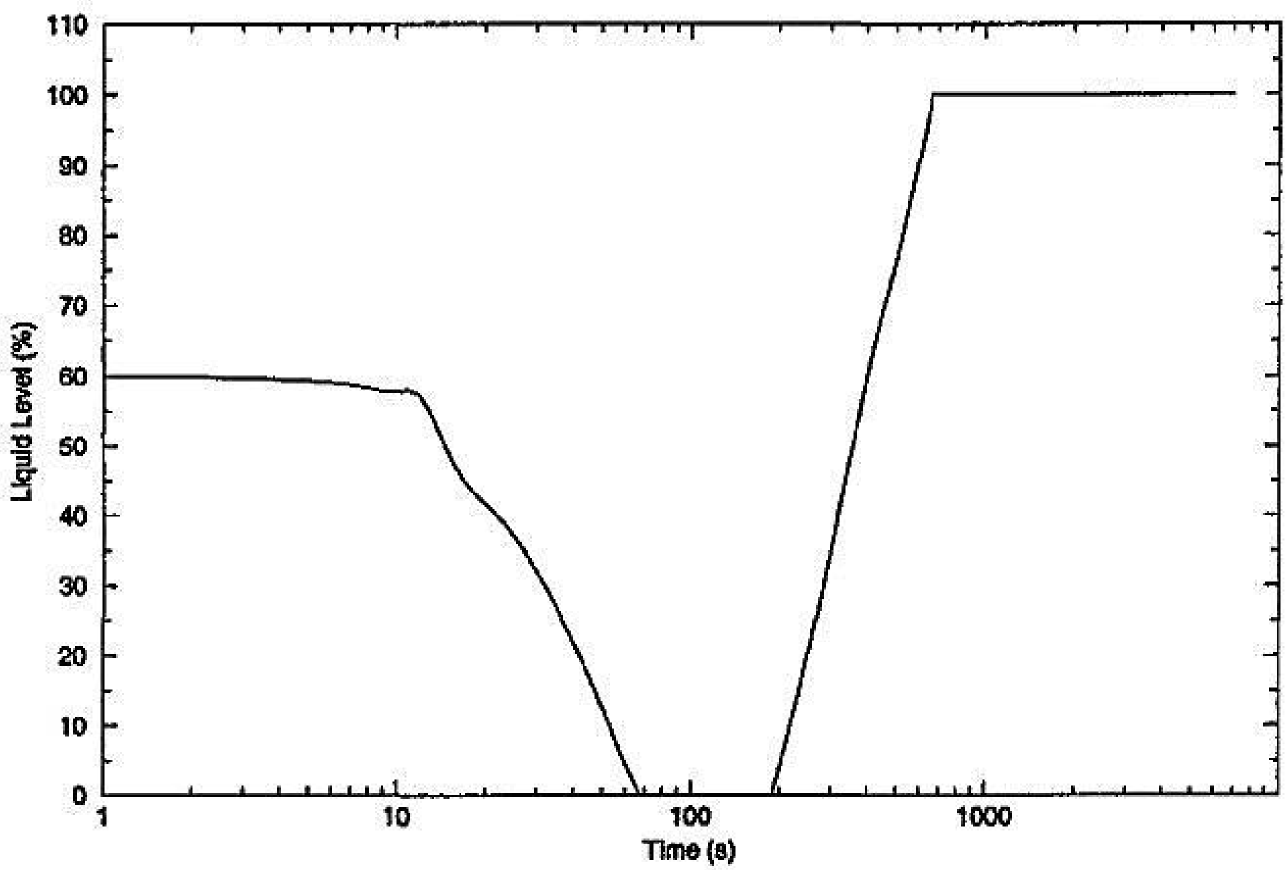


FIGURE 15.2.8-13

PRESSURIZER PRESSURE FOR FEEDWATER SYSTEM PIPE BREAK WITH LOSS OF OFFSITE POWER

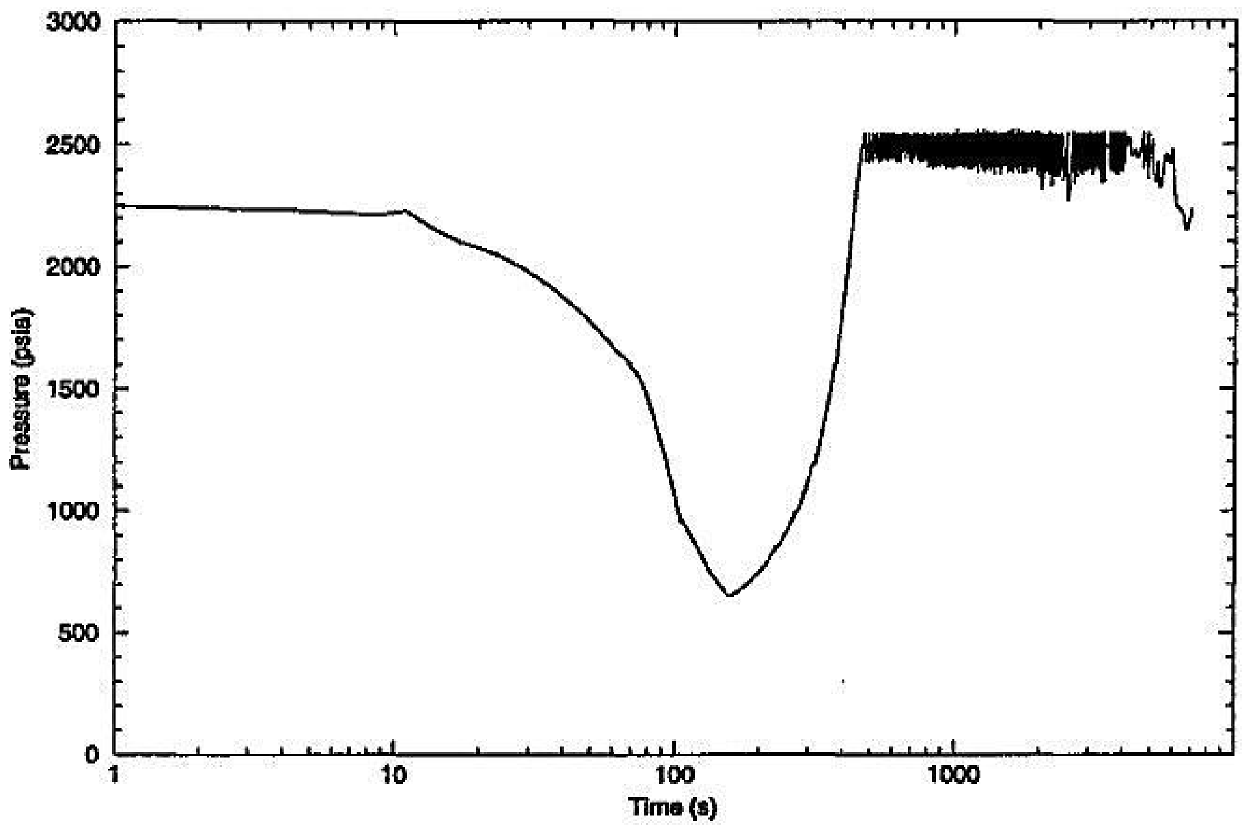


FIGURE 15.2.8-14

LOOP 1 PRIMARY SYSTEM TEMPERATURES FOR FEEDWATER SYSTEM PIPE BREAK WITH LOSS OF OFFSITE POWER

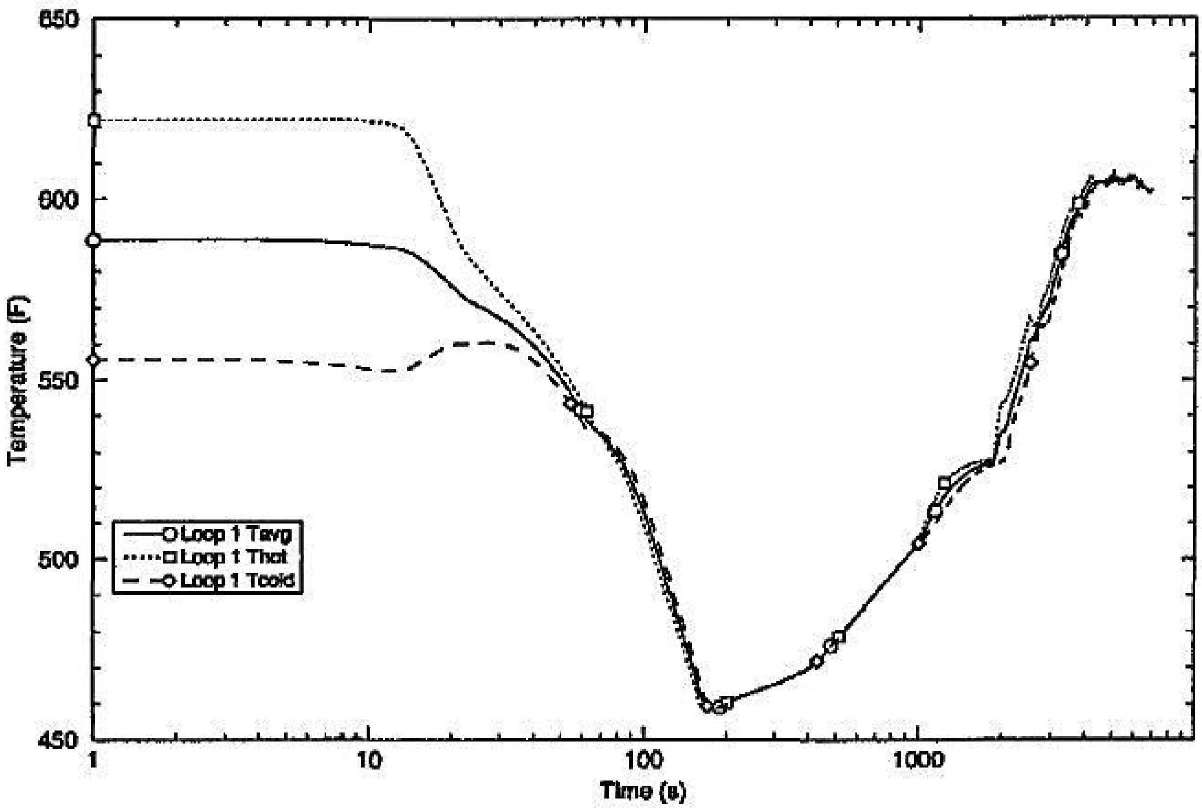


FIGURE 15.2.8-15

LOOP 2 PRIMARY SYSTEM TEMPERATURES FOR FEEDWATER SYSTEM PIPE BREAK WITH LOSS OF OFFSITE POWER

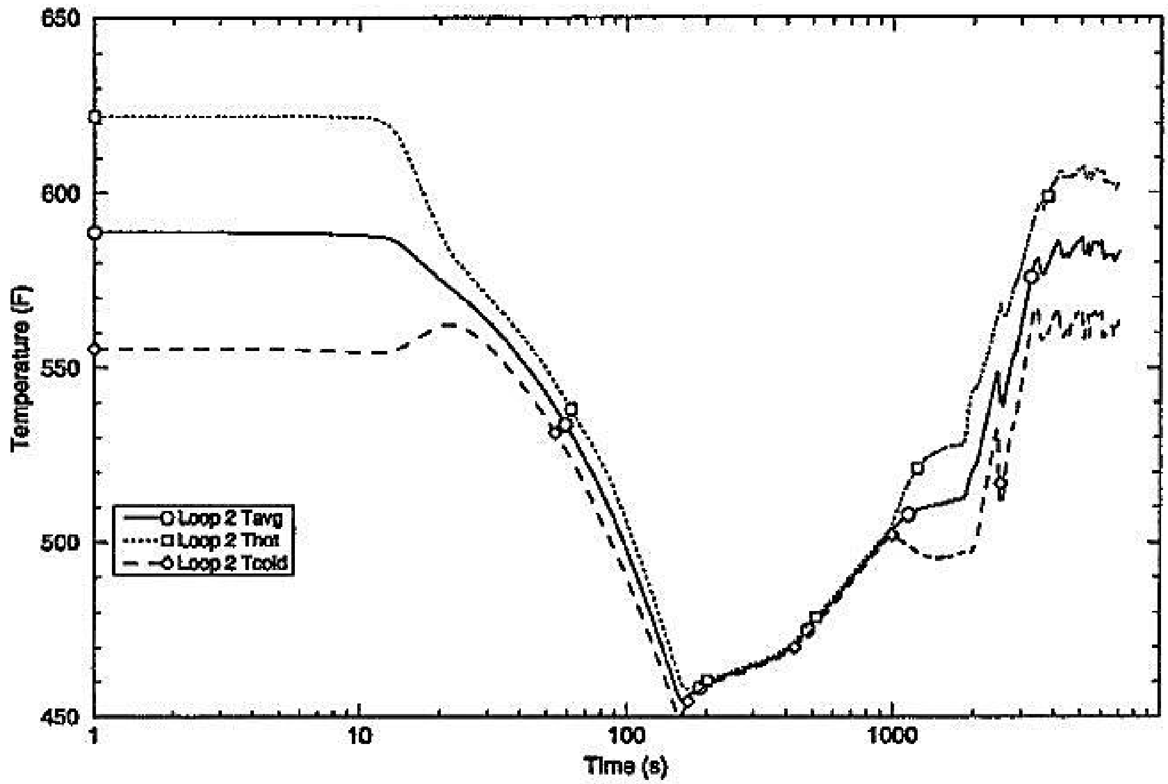


FIGURE 15.2.8-16

LOOP 3 PRIMARY SYSTEM TEMPERATURES FOR FEEDWATER SYSTEM PIPE BREAK WITH LOSS OF OFFSITE POWER

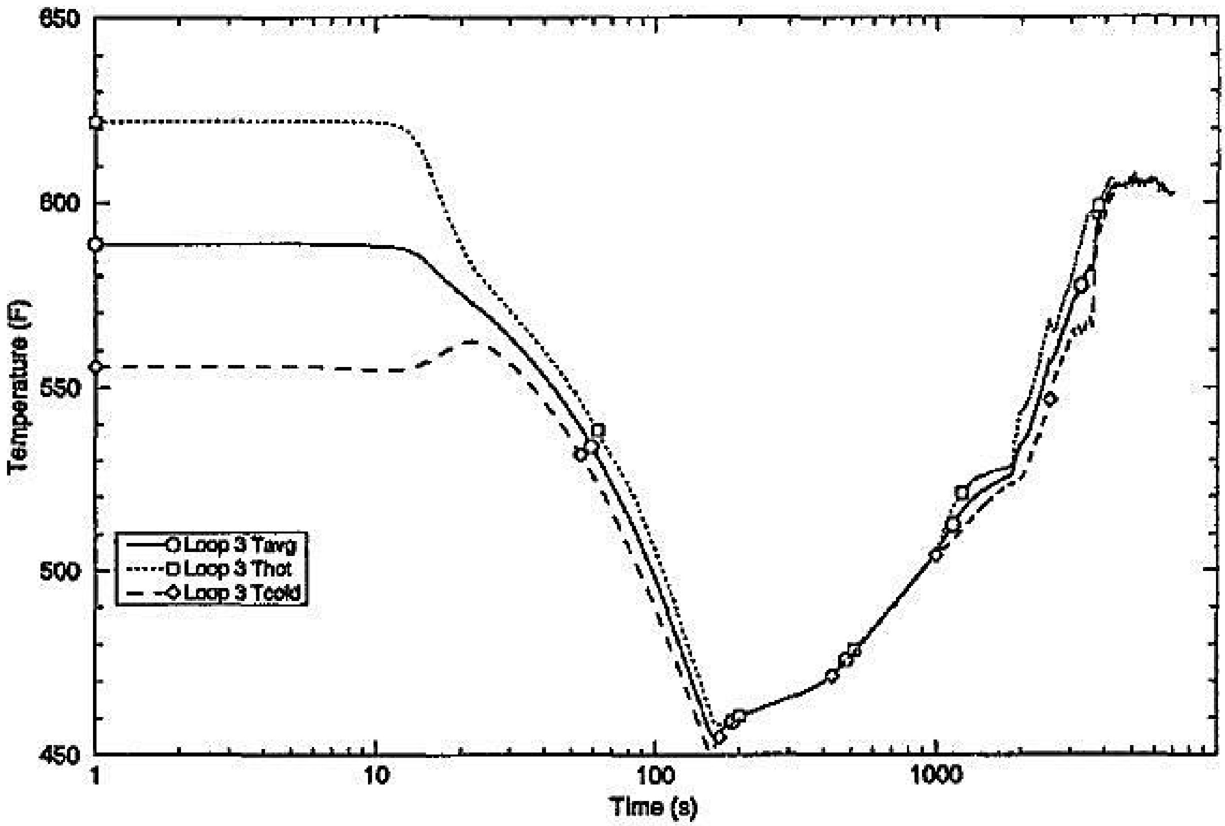


FIGURE 15.2.8-17

STEAM GENERATOR PRESSURE FOR FEEDWATER SYSTEM PIPE BREAK WITH LOSS OF OFFSITE POWER

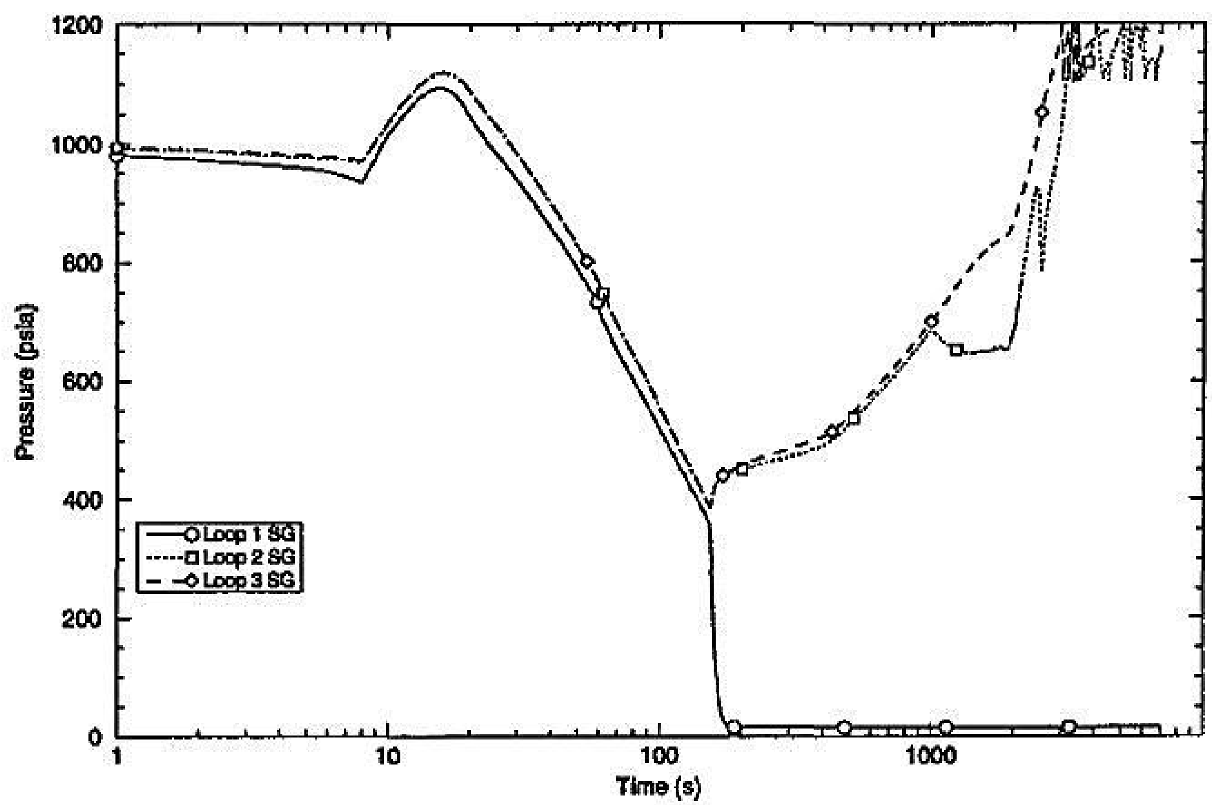


FIGURE 15.2.8-18

STEAM GENERATOR NARROW RANGE LEVEL FOR FEEDWATER SYSTEM PIPE BREAK WITH LOSS OF OFFSITE POWER

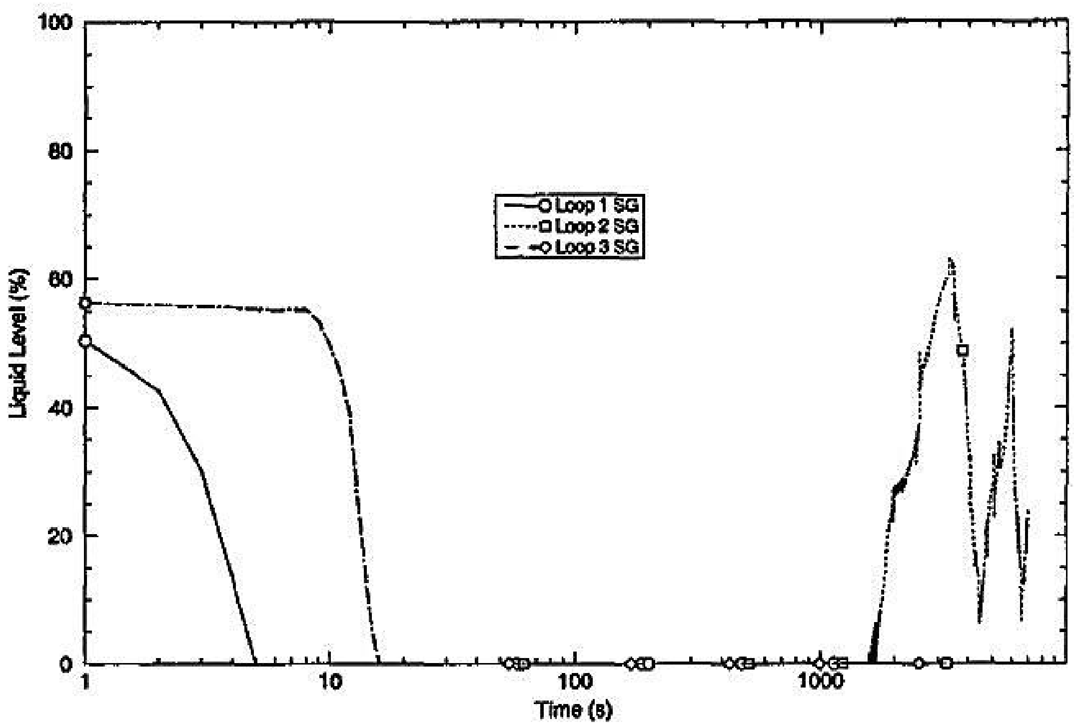


FIGURE 15.2.8-19

REACTOR COOLANT SYSTEM FLOW FOR FEEDWATER SYSTEM PIPE BREAK WITH LOSS OF OFFSITE POWER

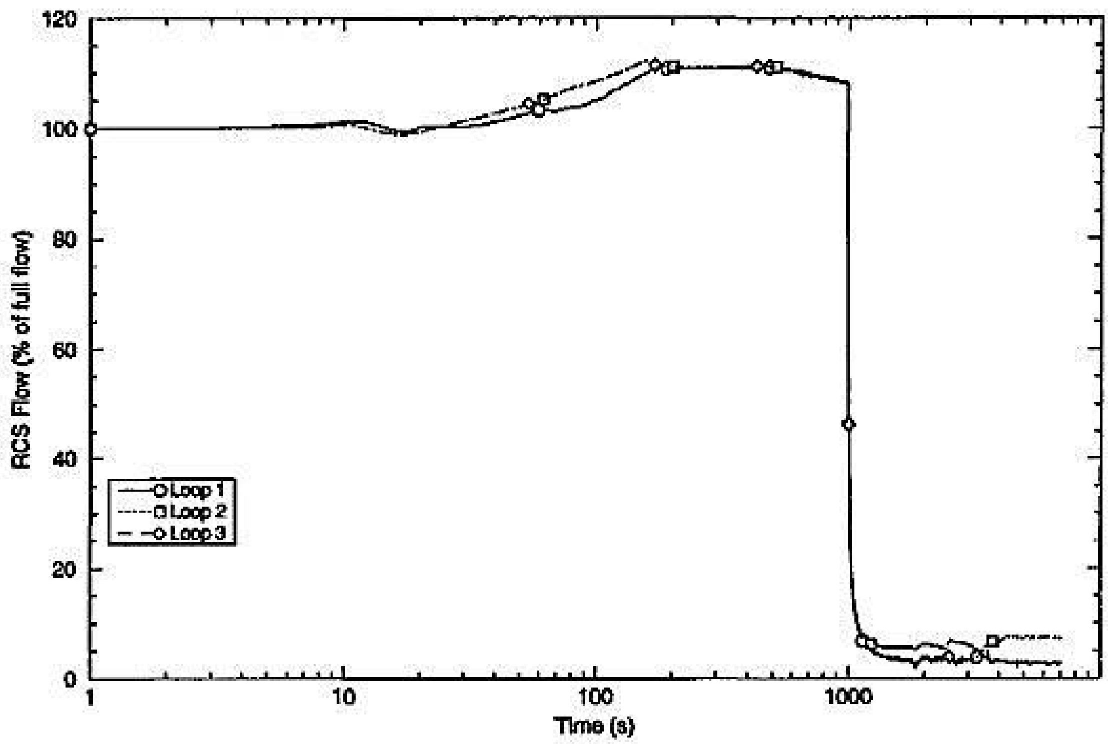


FIGURE 15.2.8-20

TOTAL PRESSURIZER RELIEF FLOW FOR FEEDWATER SYSTEM PIPE BREAK WITH LOSS OF OFFSITE POWER

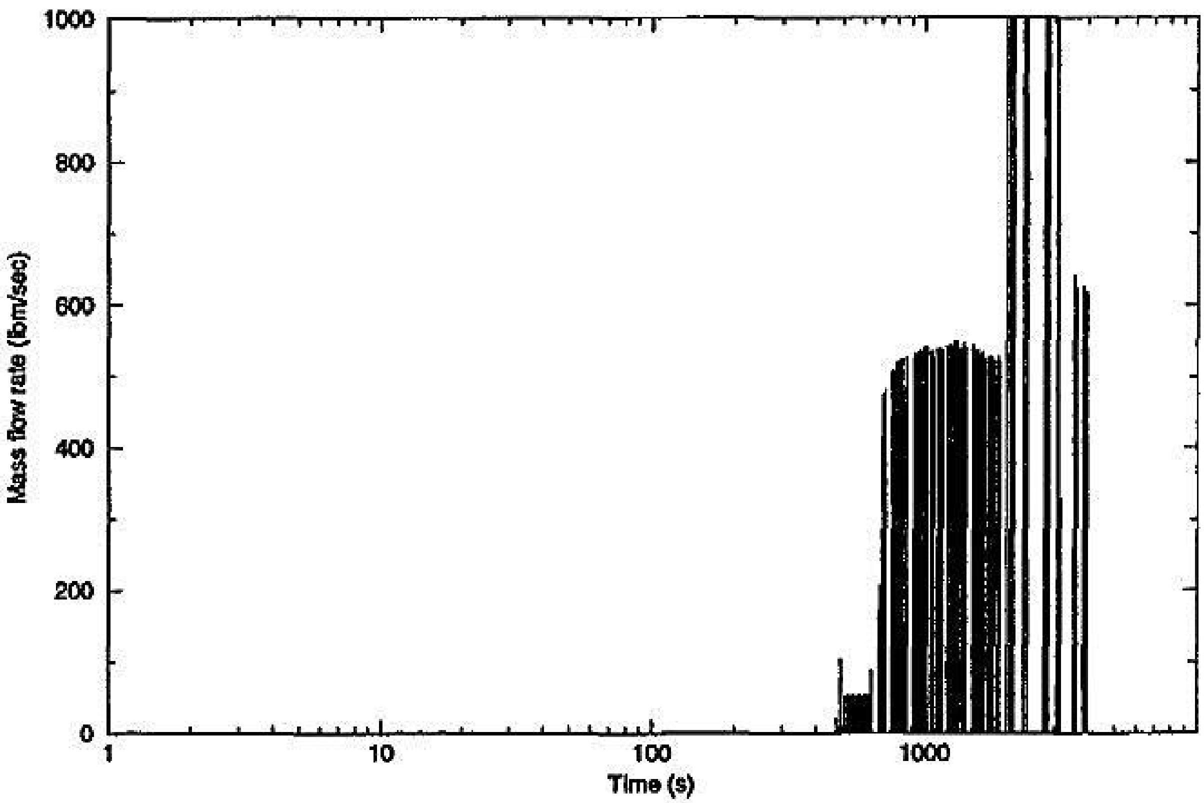


FIGURE 15.3.2-1

REACTOR POWER FOR LOSS OF FORCED REACTOR COOLANT FLOW – UNDERFREQUENCY CASE

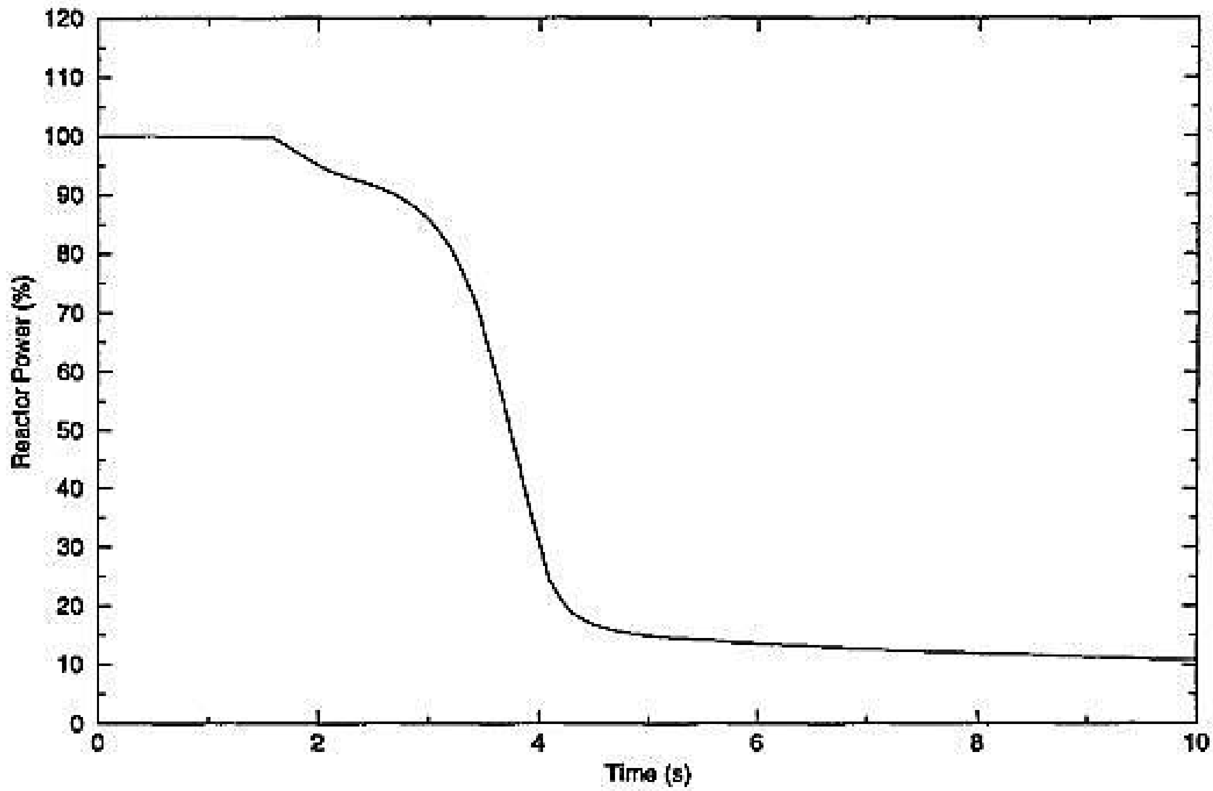


FIGURE 15.3.2-2

CORE AVERAGE HEAT FLUX FOR LOSS OF FORCED REACTOR COOLANT FLOW – UNDERFREQUENCY CASE

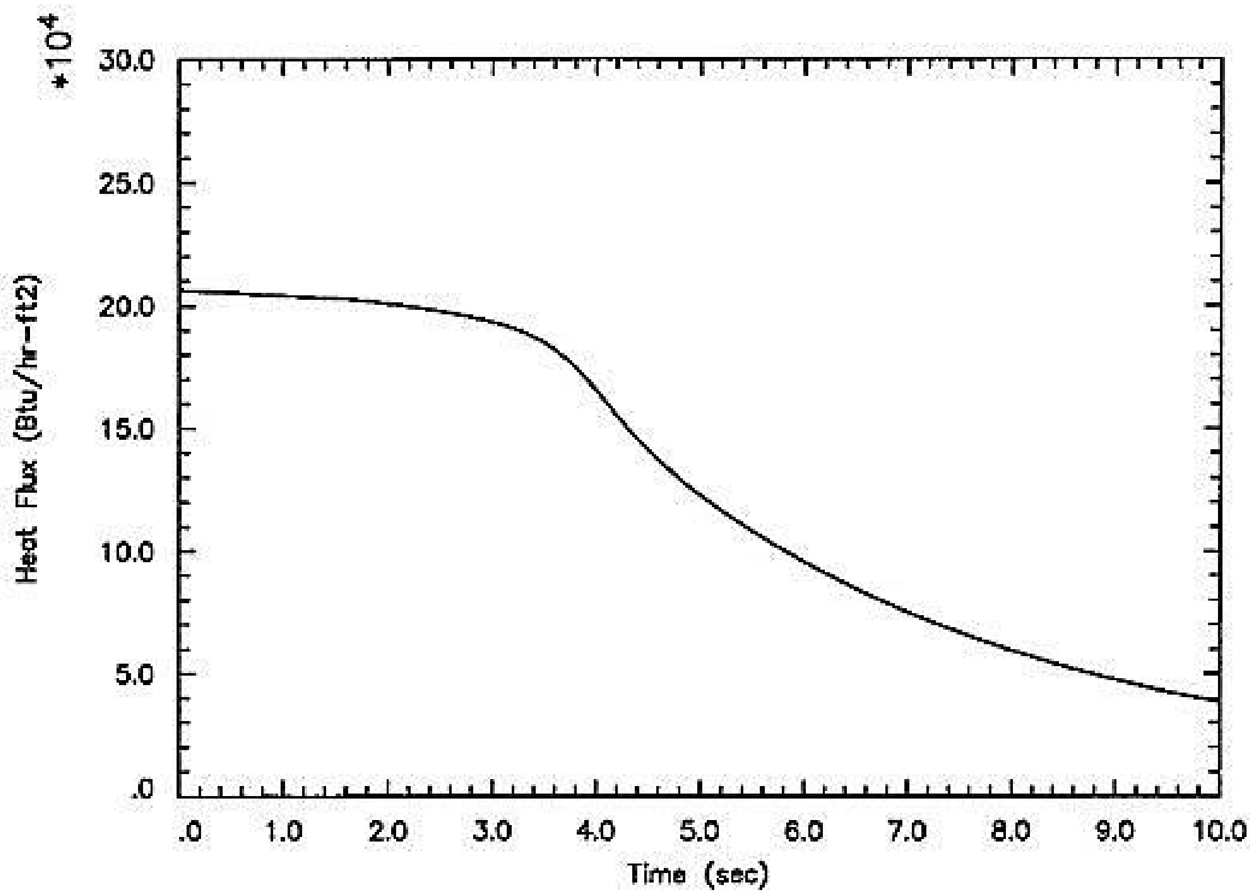


FIGURE 15.3.2-3

PRESSURIZER PRESSURE FOR LOSS OF FORCED REACTOR COOLANT FLOW – UNDERFREQUENCY CASE

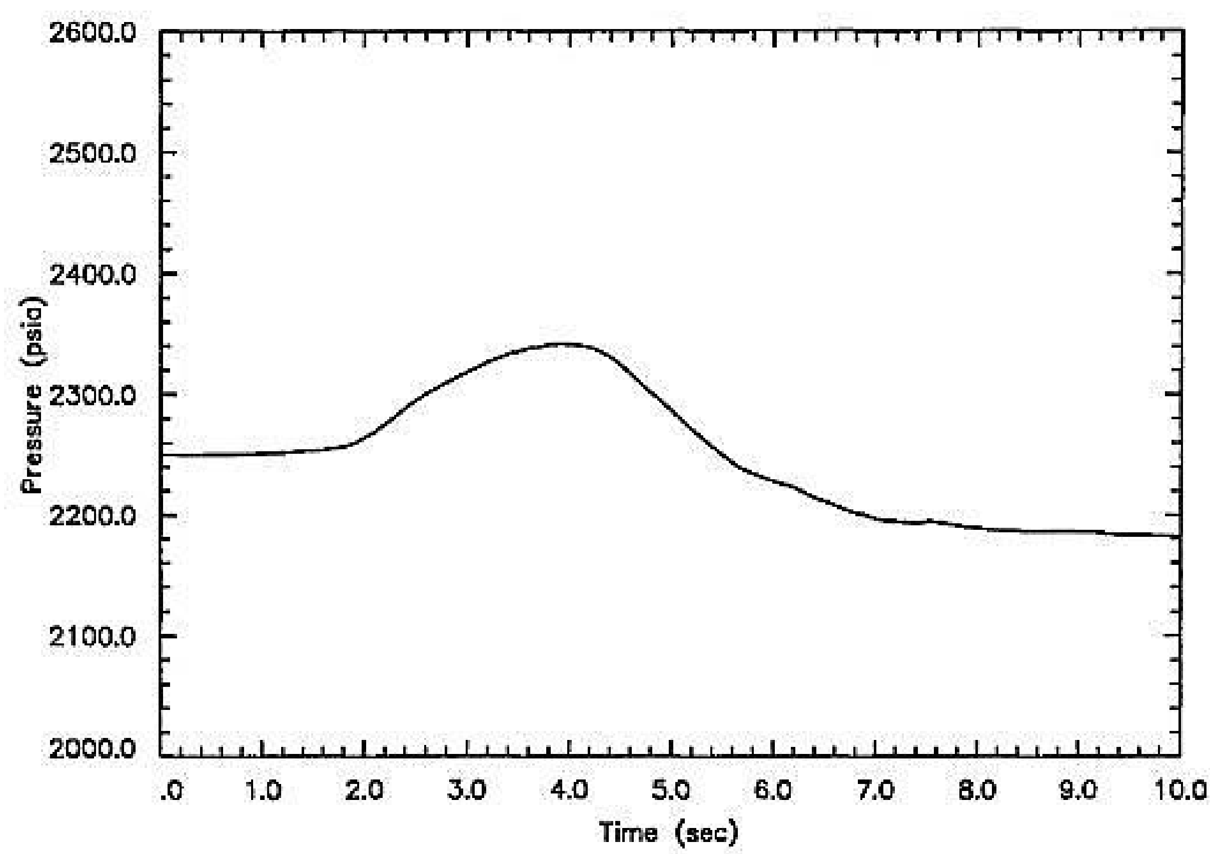


FIGURE 15.3.2-4

PRESSURIZER LEVEL FOR LOSS OF FORCED REACTOR COOLANT FLOW – UNDERFREQUENCY CASE

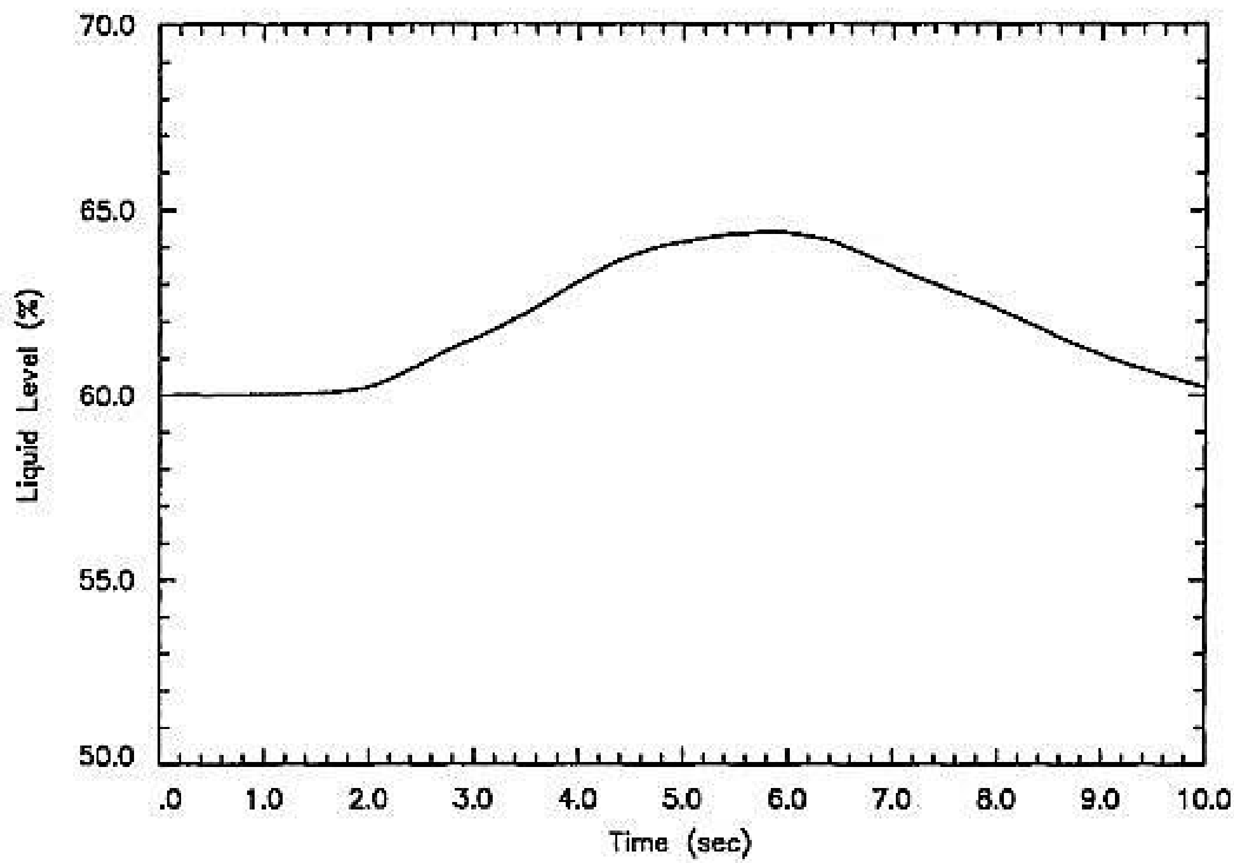


FIGURE 15.3.2-5

REACTOR COOLANT SYSTEM MASS FLOW RATE FOR LOSS OF FORCED REACTOR COOLANT FLOW – UNDERFREQUENCY CASE

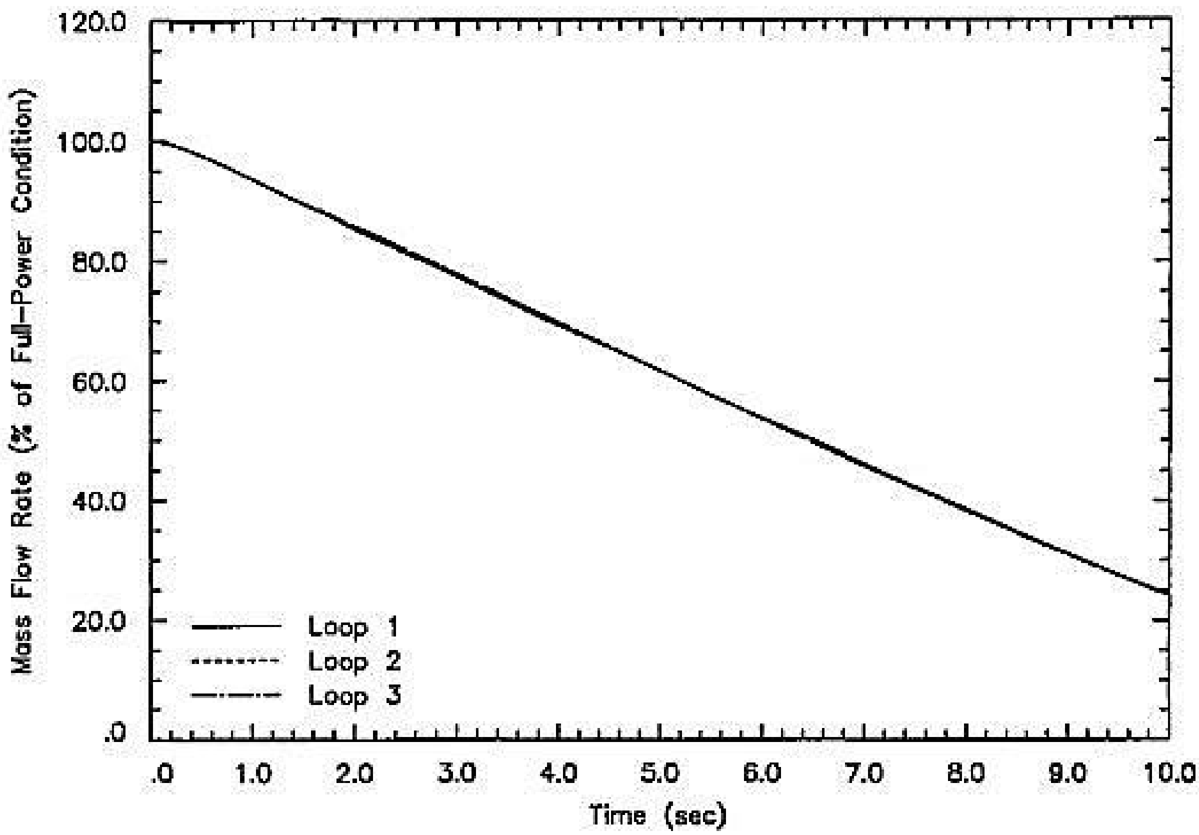


FIGURE 15.3.2-6

CORE INLET AND OUTLET TEMPERATURES FOR LOSS OF FORCED REACTOR COOLANT FLOW – UNDERFREQUENCY CASE

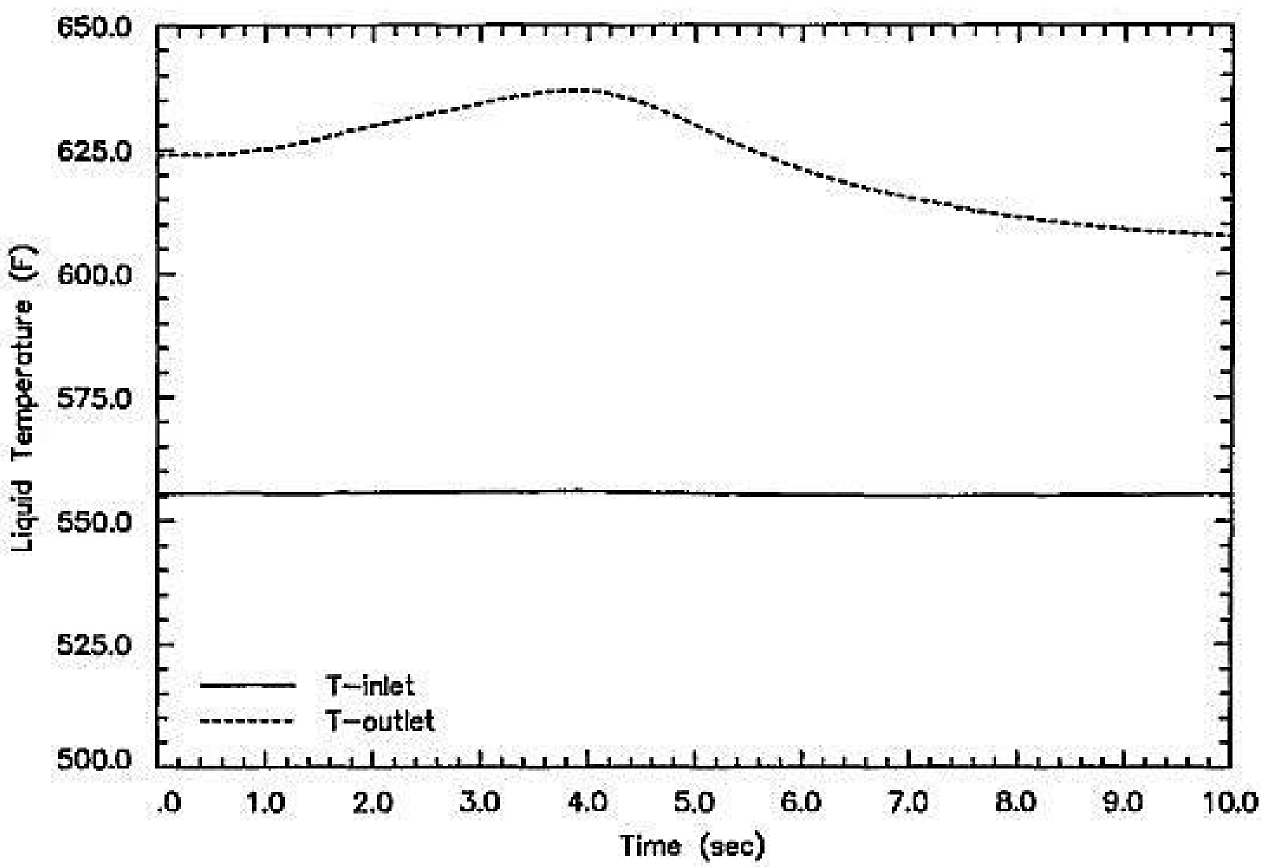


FIGURE 15.3.2-7

TOTAL CORE REACTIVITY FOR LOSS OF FORCED REACTOR COOLANT FLOW – UNDERFREQUENCY CASE

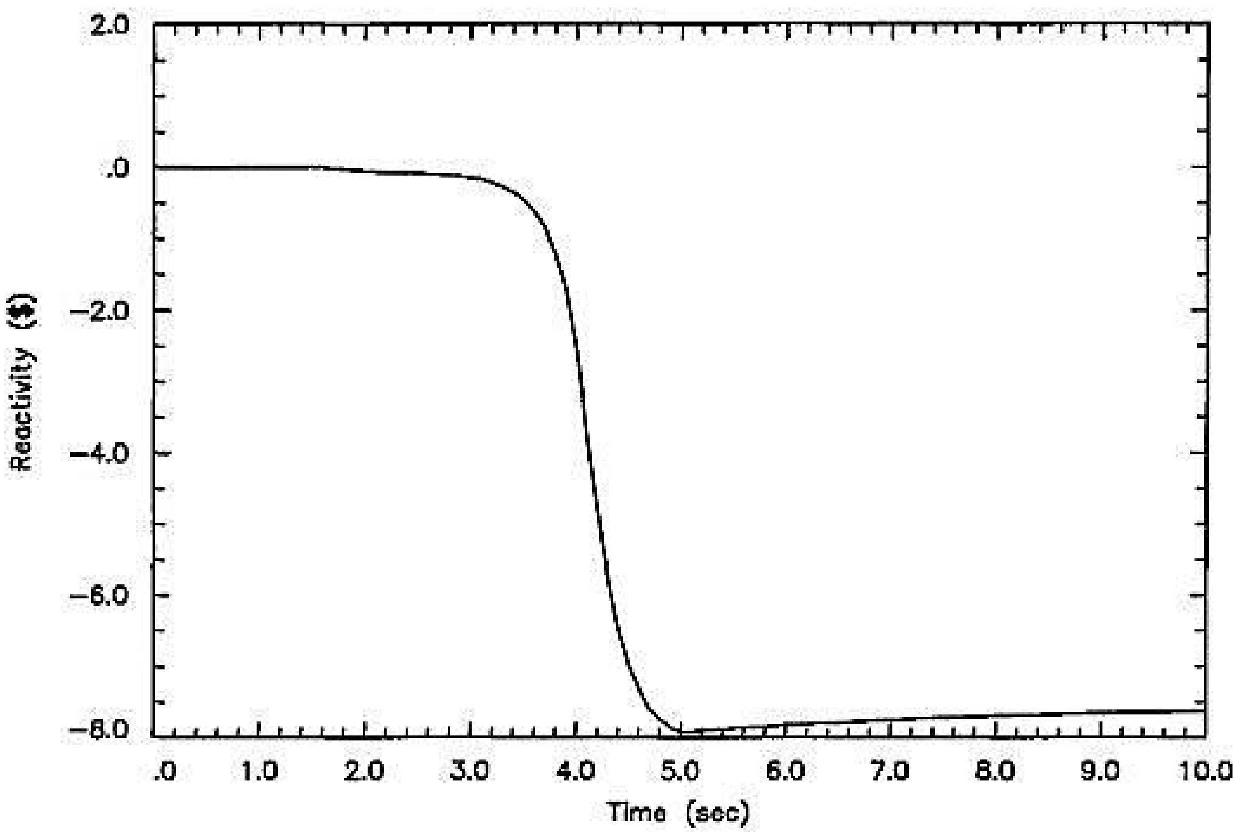


FIGURE 15.3.3-1

REACTOR POWER FOR LOCKED ROTOR OVERPRESSURIZATION CASE

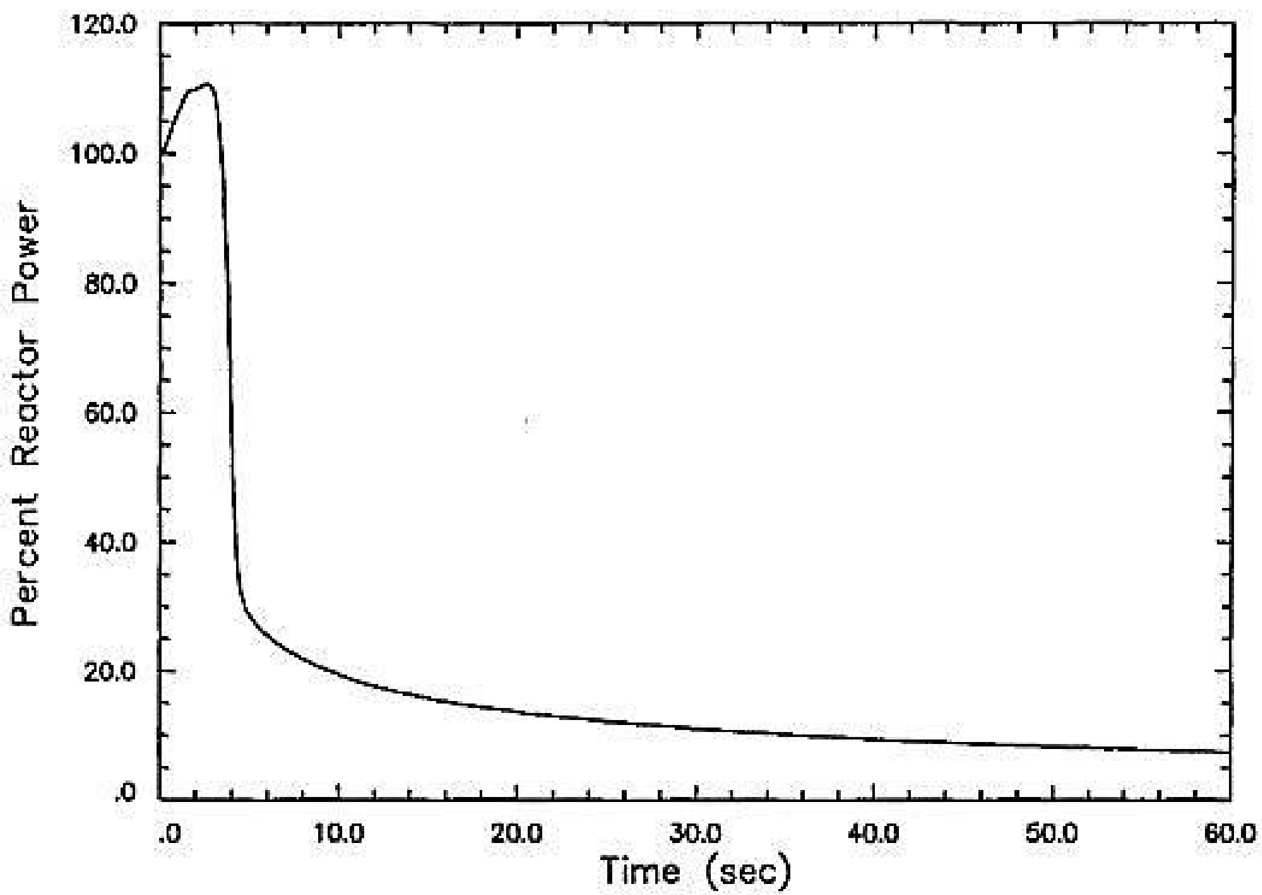


FIGURE 15.3.3-2

PRESSURIZER LEVEL FOR LOCKED ROTOR OVERPRESSURIZATION CASE

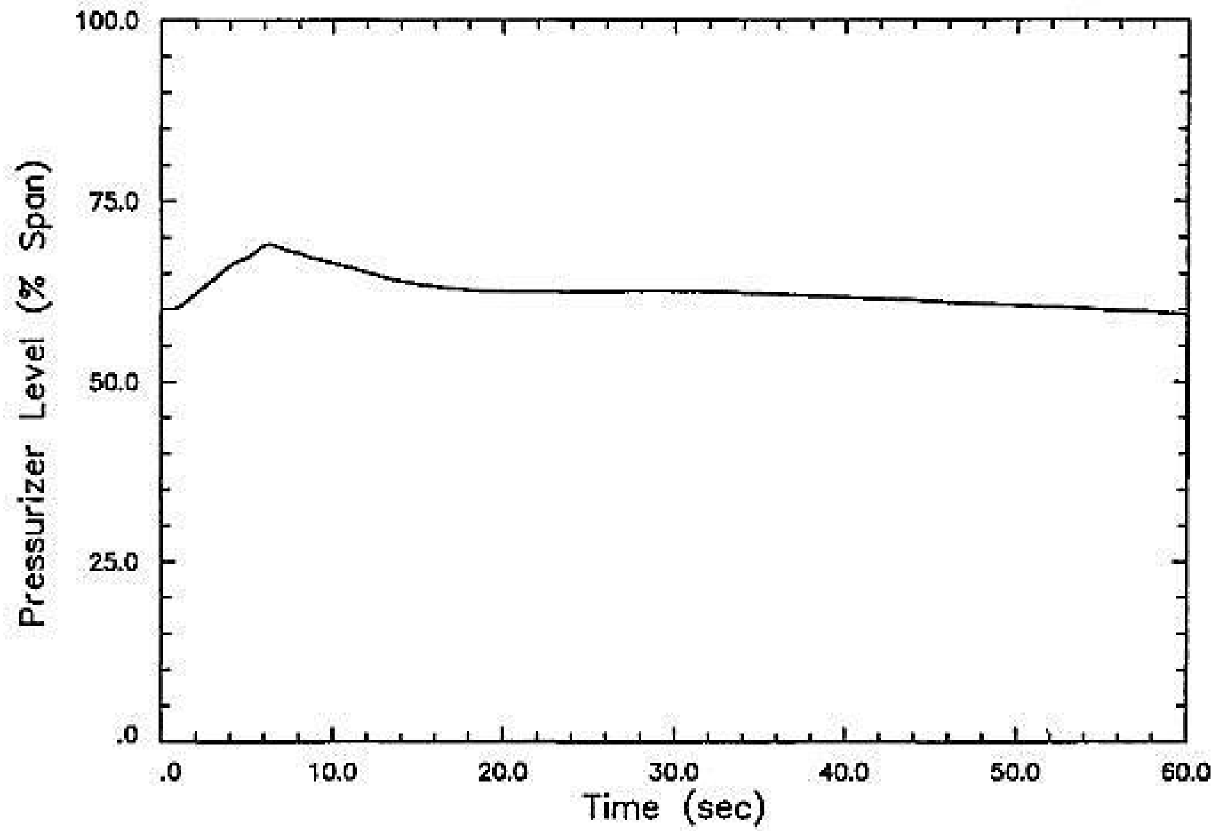


FIGURE 15.3.3-3

REACTOR COOLANT SYSTEM MASS FLOW RATE FOR LOCKED ROTOR OVERPRESSURIZATION CASE

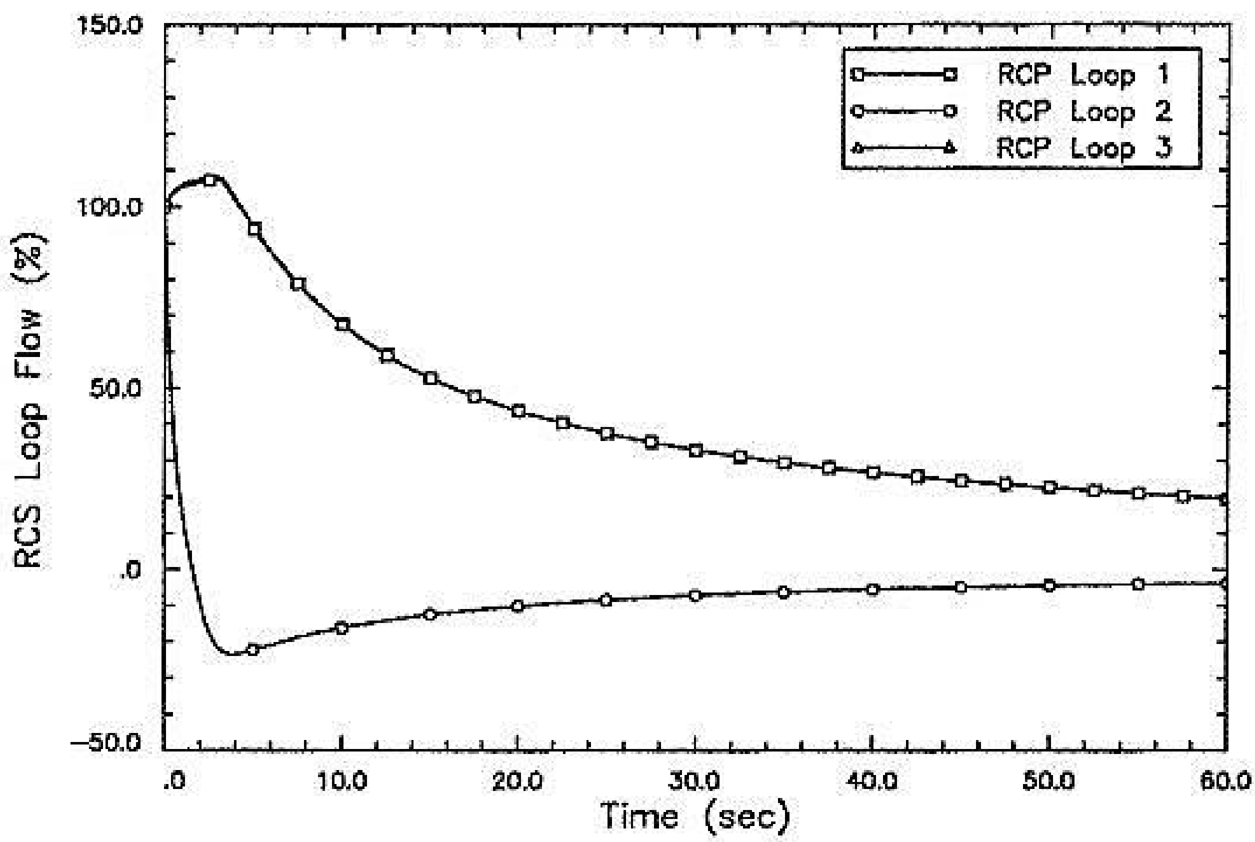


FIGURE 15.3.3-4

AVERAGE FLUID TEMPERATURES FOR LOCKED ROTOR OVERPRESSURIZATION CASE

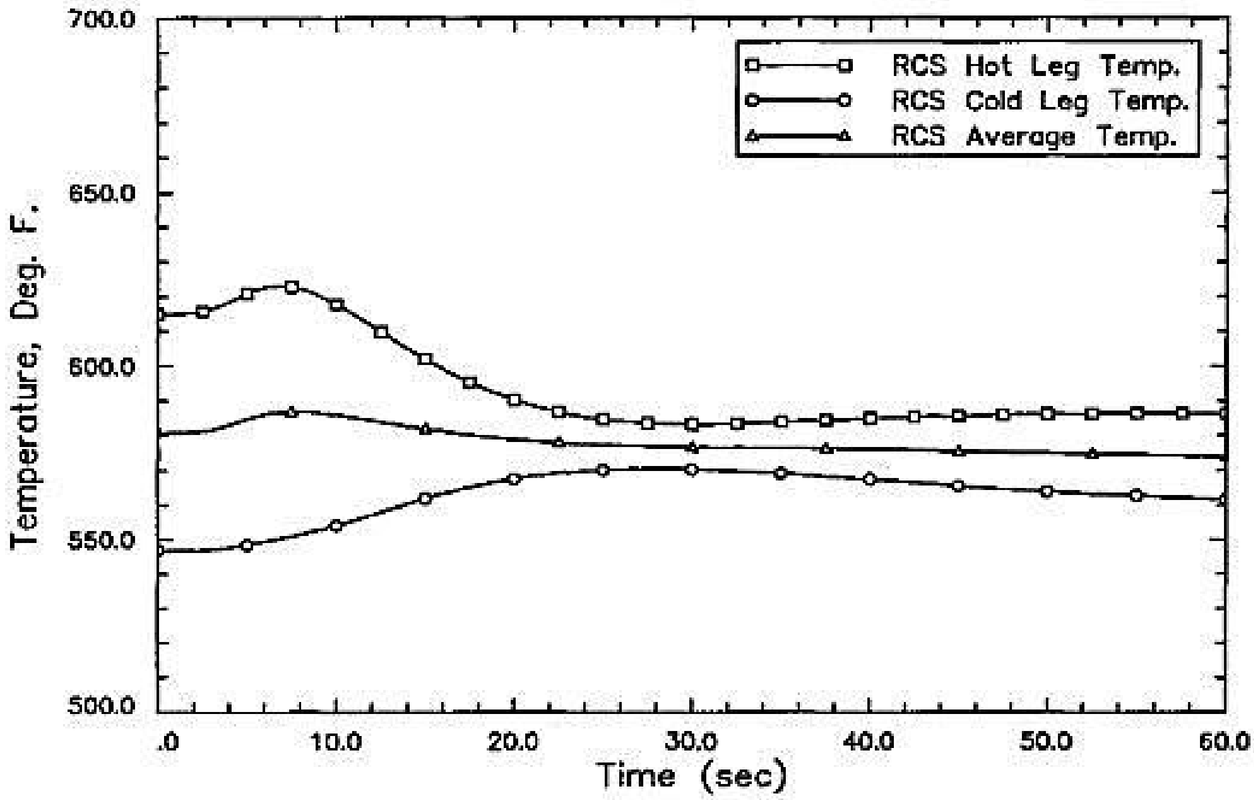


FIGURE 15.3.3-5

TOTAL CORE REACTIVITY FOR LOCKED ROTOR OVERPRESSURIZATION CASE

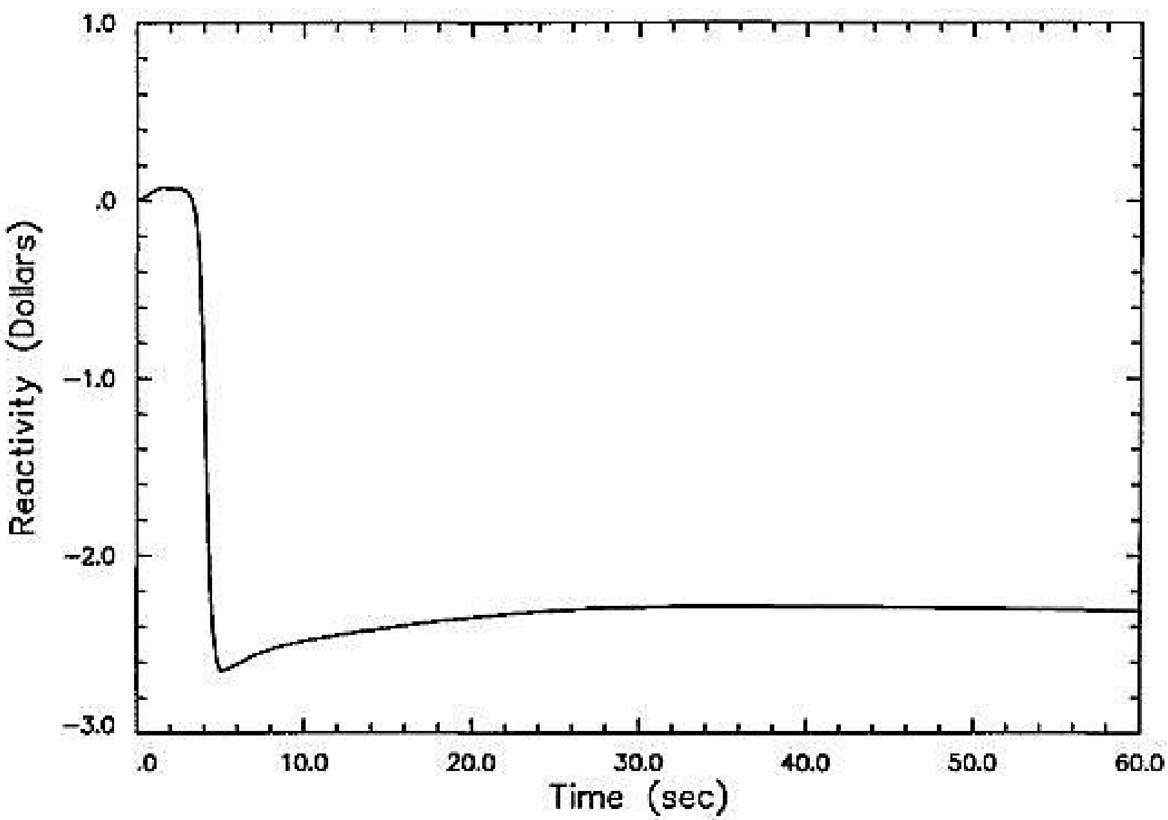


FIGURE 15.3.3-6

MAXIMUM PRIMARY SYSTEM PRESSURE FOR LOCKED ROTOR OVERPRESSURIZATION CASE

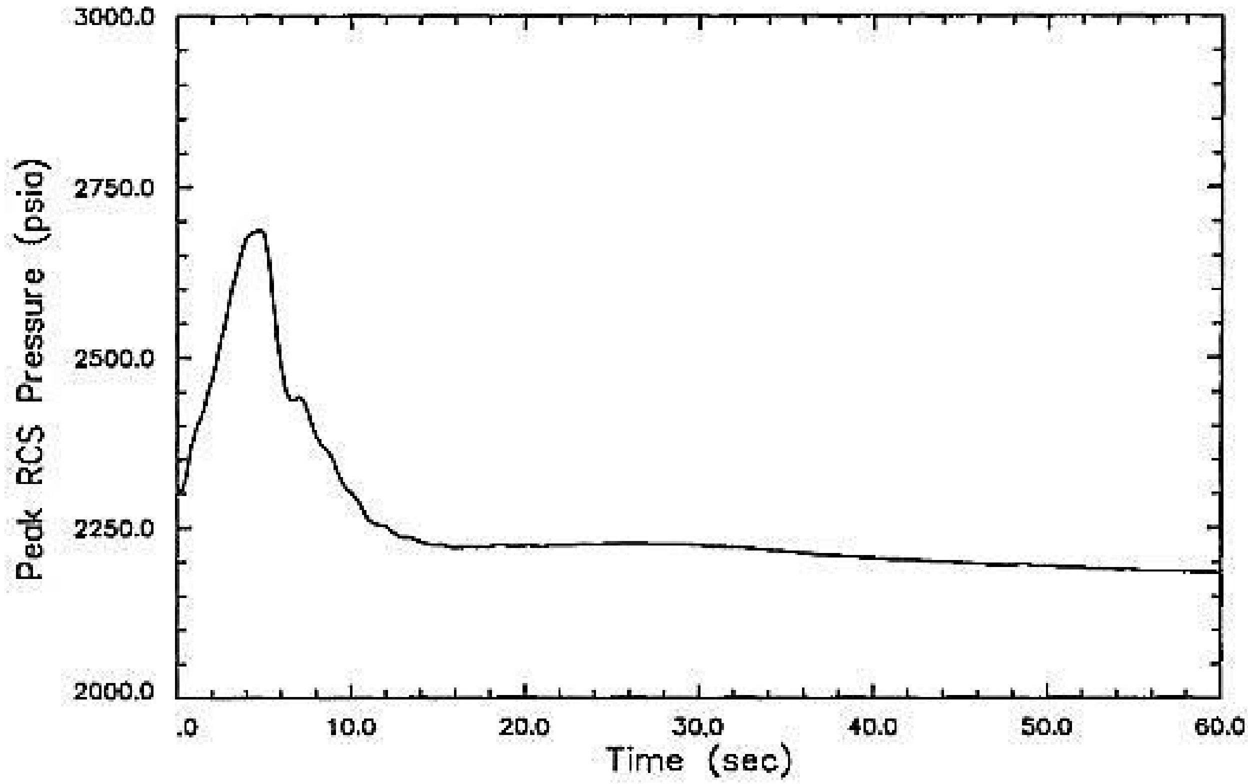


FIGURE 15.3.3-7

REACTOR POWER LEVEL FOR LOCKED ROTOR MDNBR CASE

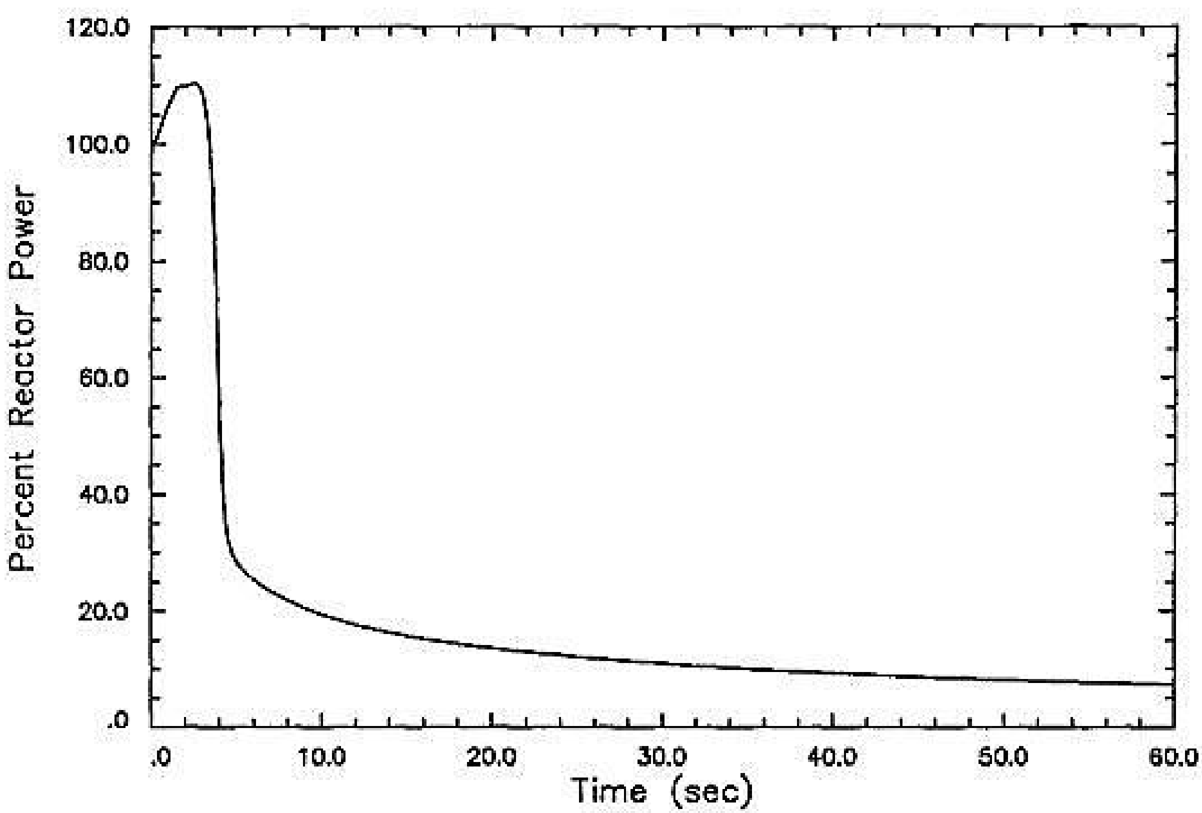


FIGURE 15.3.3-8

CORE AVERAGE HEAT FLUX FOR LOCKED ROTOR MDNBR CASE

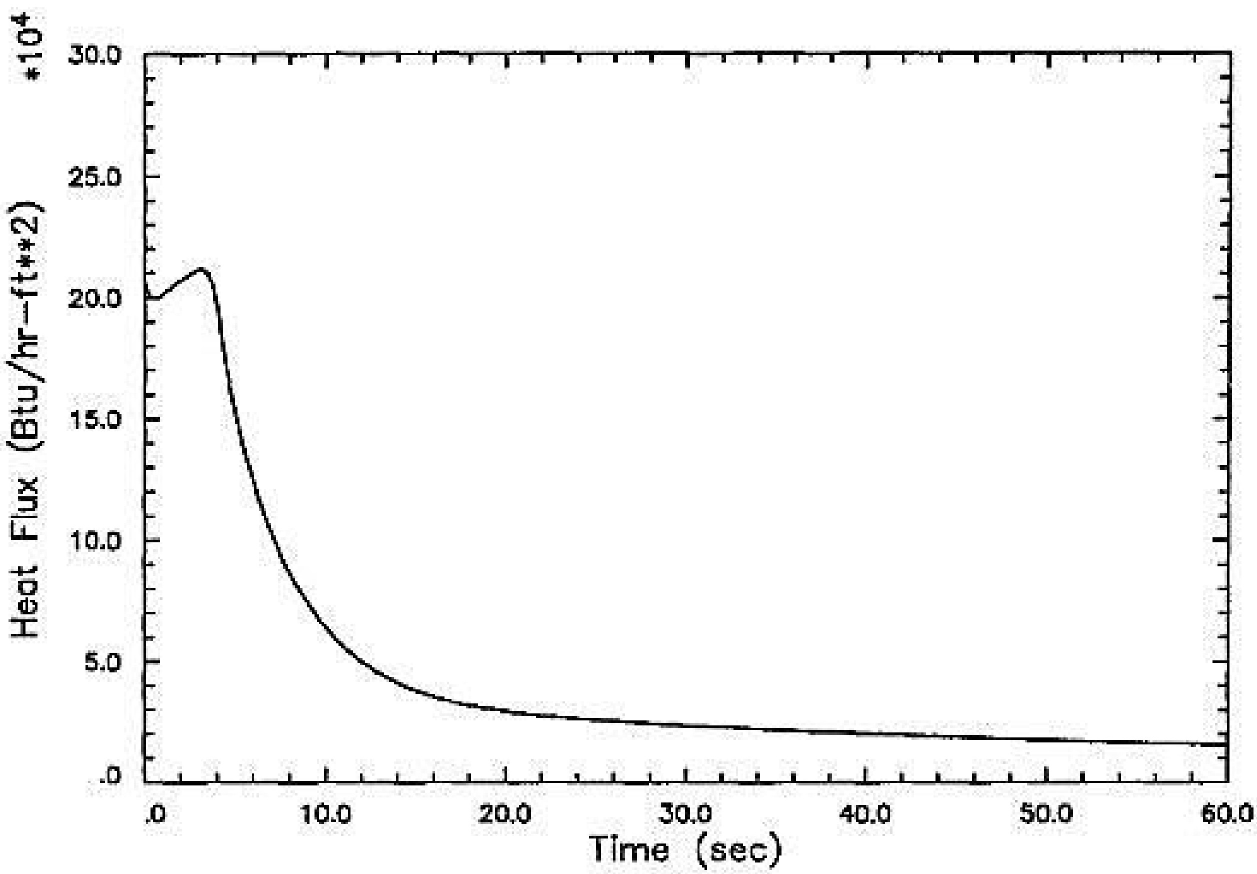


FIGURE 15.3.3-9

PRESSURIZER PRESSURE FOR LOCKED ROTOR MDNBR CASE

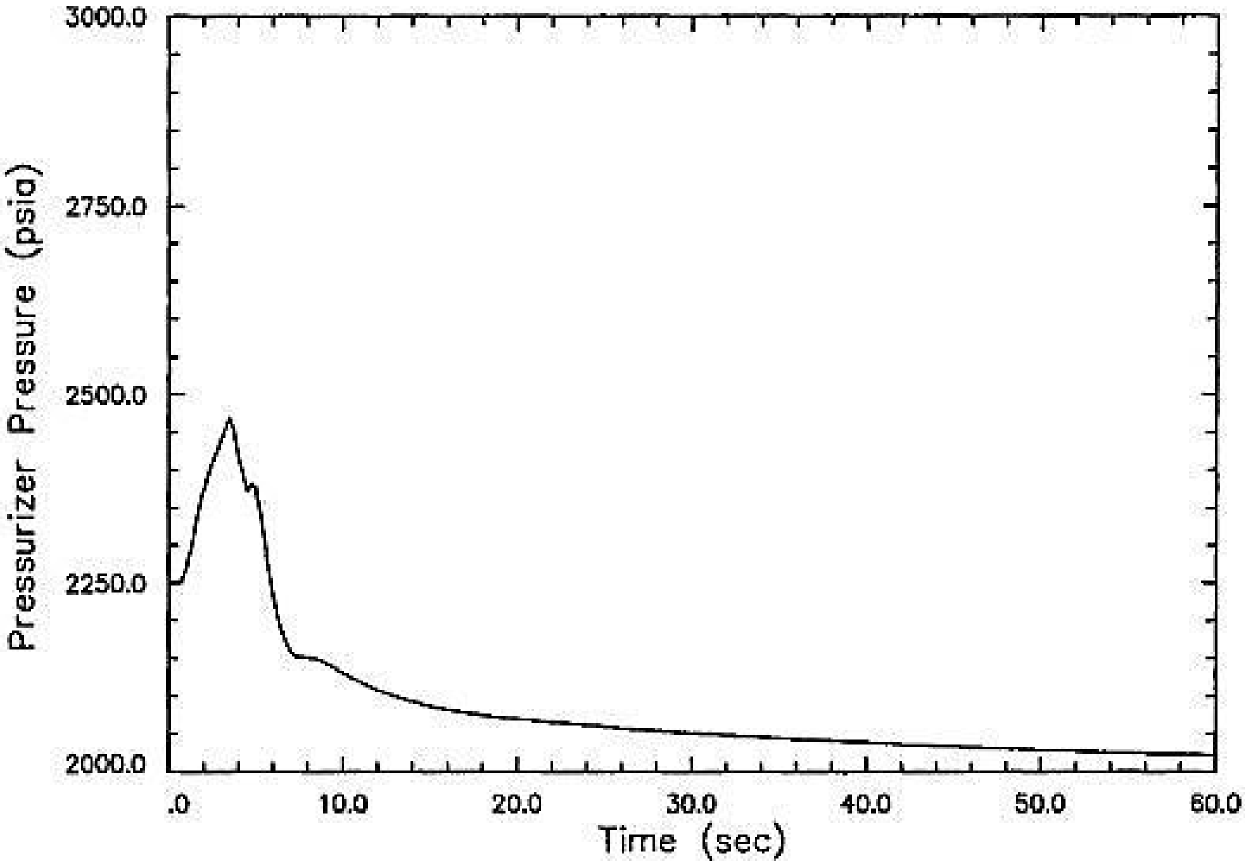


FIGURE 15.3.3-10

PRESSURIZER LEVEL FOR LOCKED ROTOR MDNBR CASE

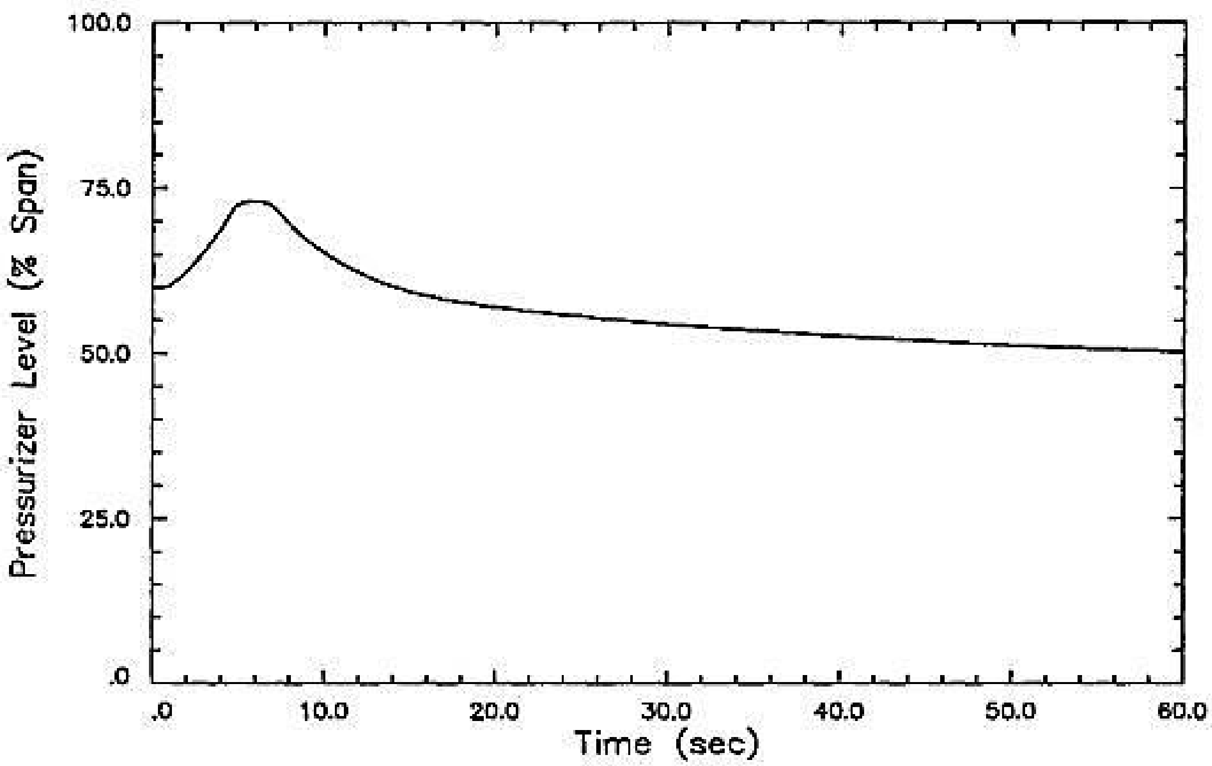


FIGURE 15.3.3-11

REACTOR COOLANT SYSTEM MASS FLOW RATE FOR LOCKED ROTOR MDNBR CASE

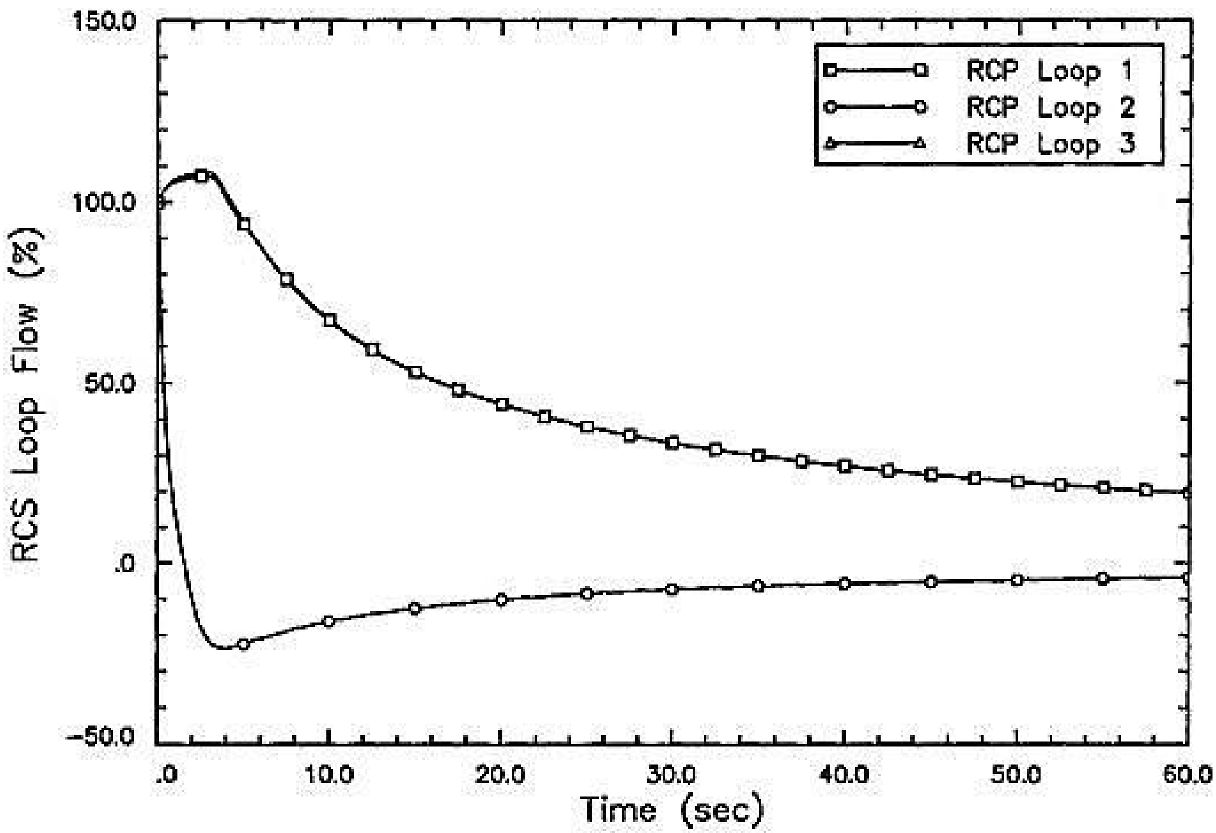


FIGURE 15.3.3-12

AVERAGE FLUID TEMPERATURES FOR LOCKED ROTOR MDNBR CASE

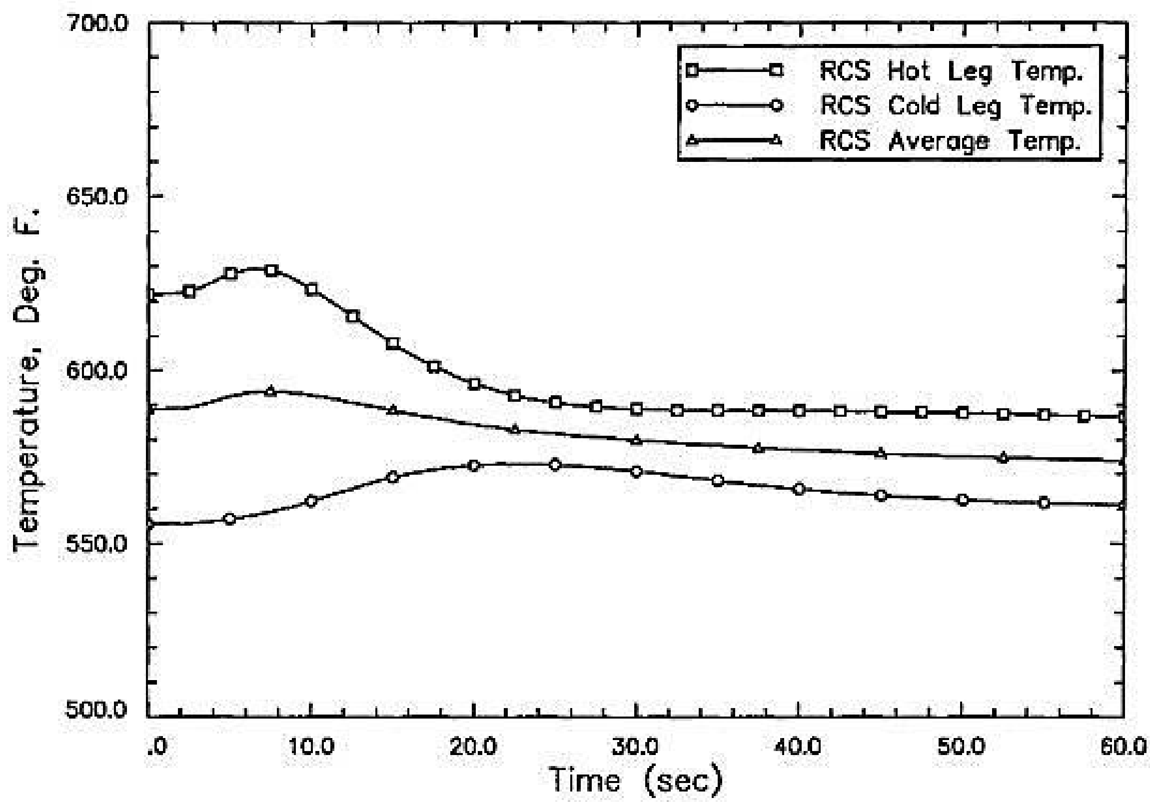


FIGURE 15.3.3-13

TOTAL CORE REACTIVITY FOR LOCKED ROTOR MDNBR CASE

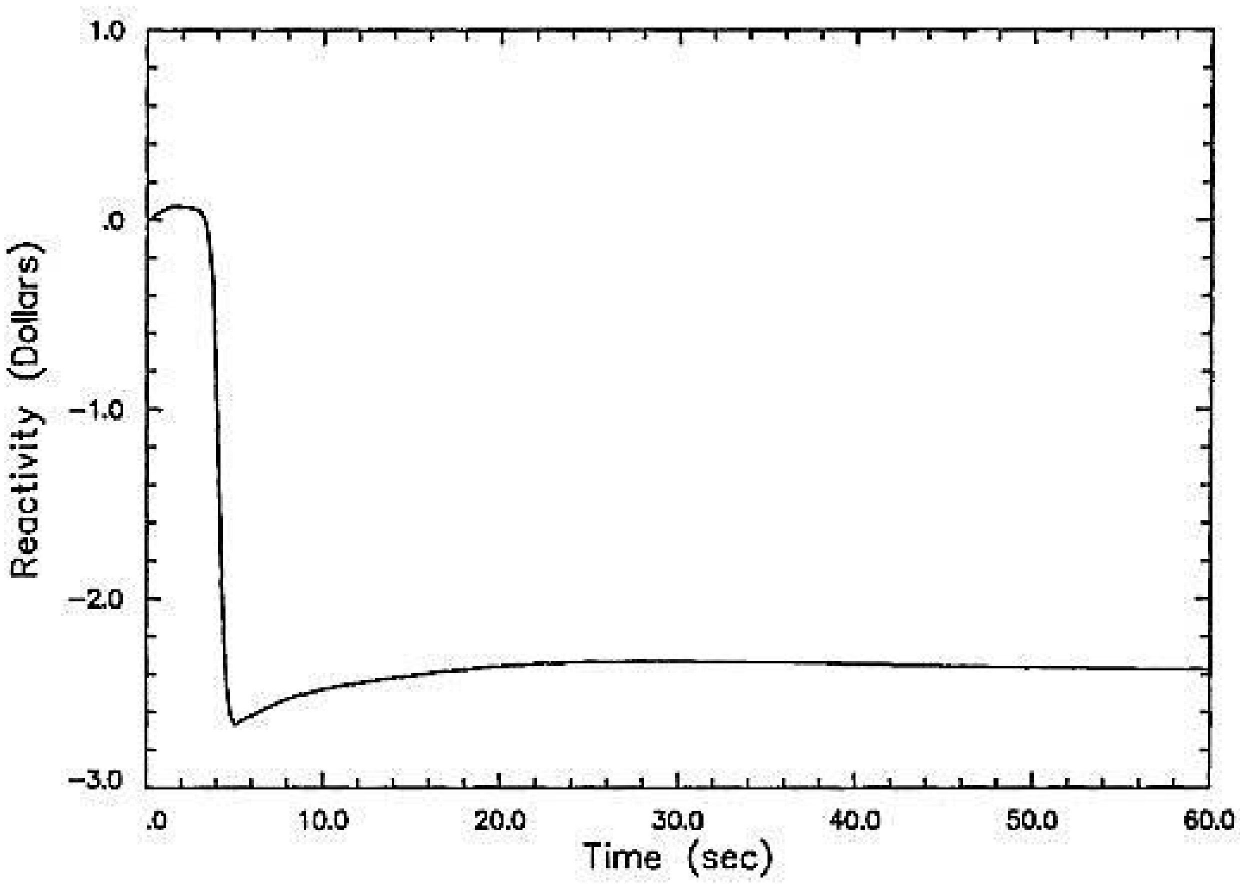


FIGURE 15.4.1-1

REACTOR POWER FOR UNCONTROLLED RCCA BANK WITHDRAWAL AT HZP

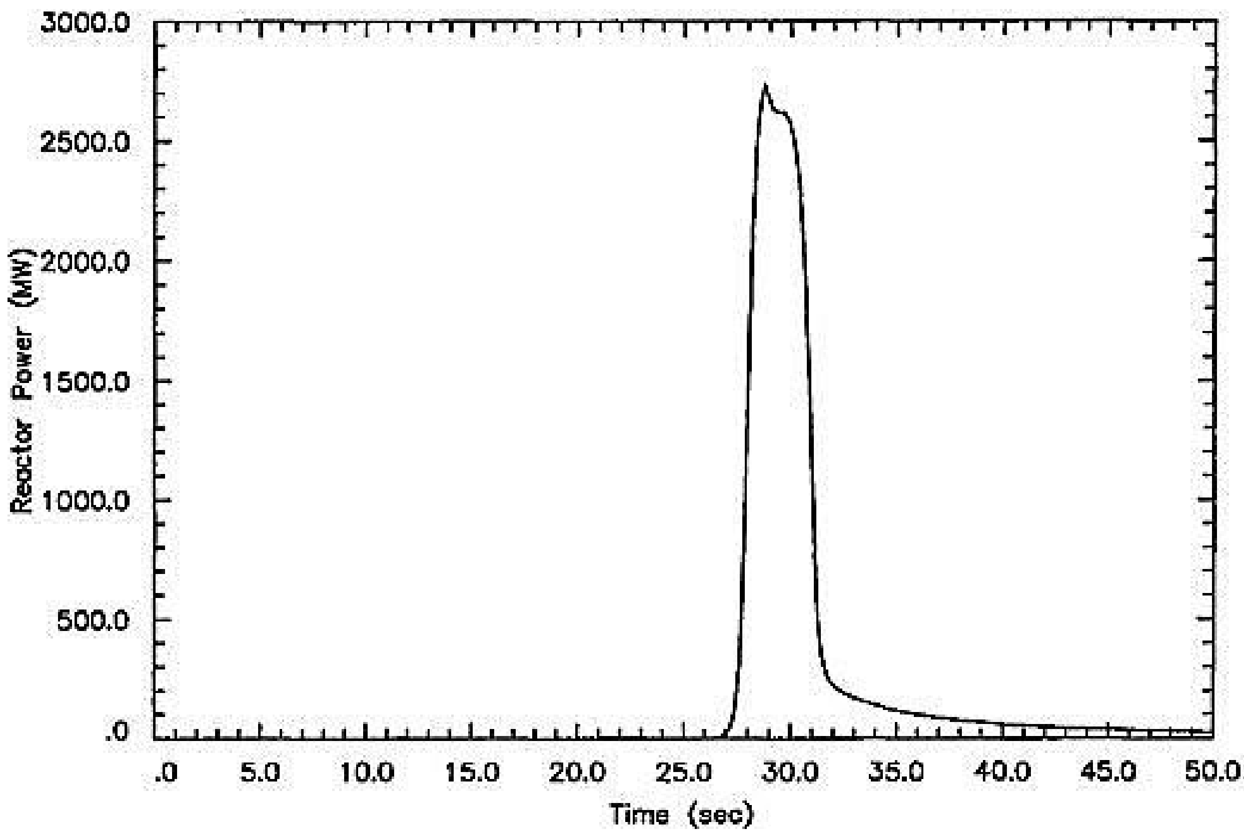


FIGURE 15.4.1-2

REACTIVITY COMPONENTS FOR UNCONTROLLED RCCA BANK WITHDRAWAL AT HZP

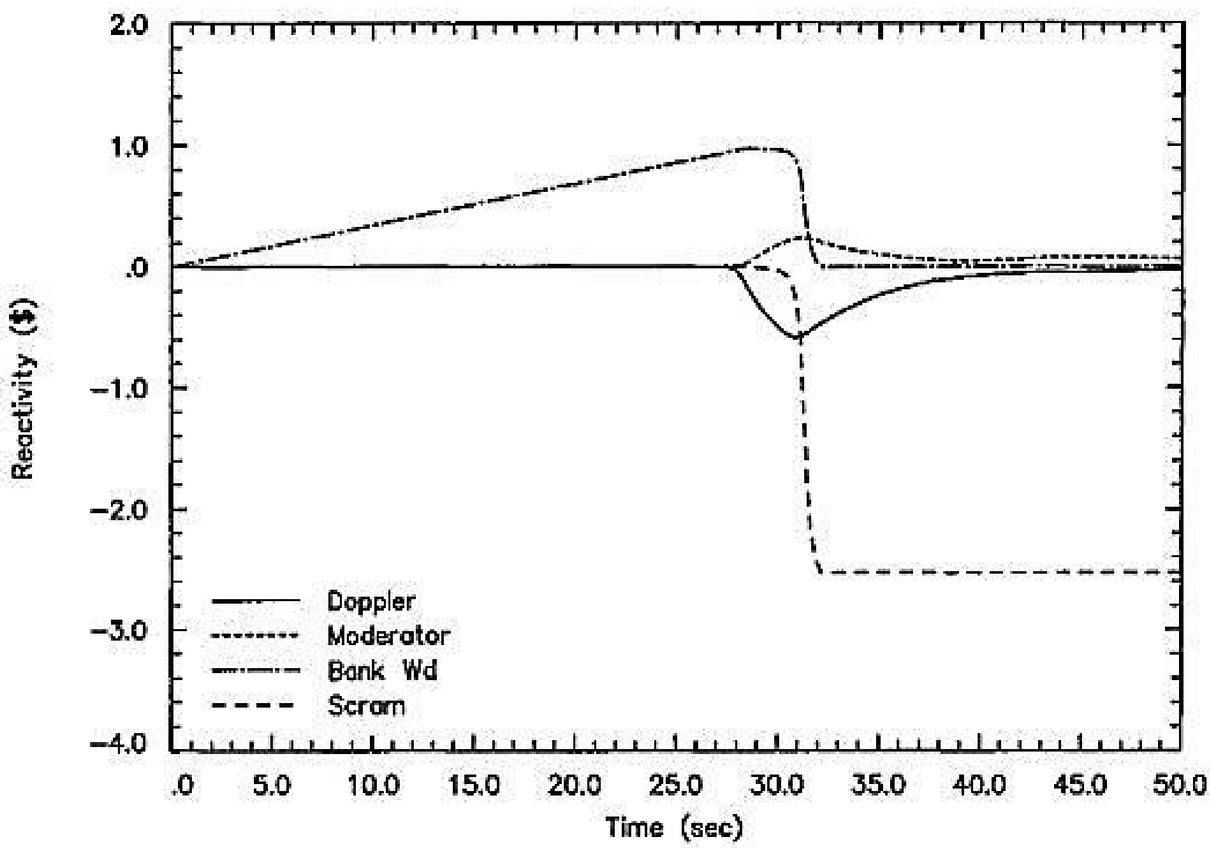


FIGURE 15.4.1-3

CORE AVERAGE HEAT FLUX FOR UNCONTROLLED RCCA BANK WITHDRAWAL AT HZP

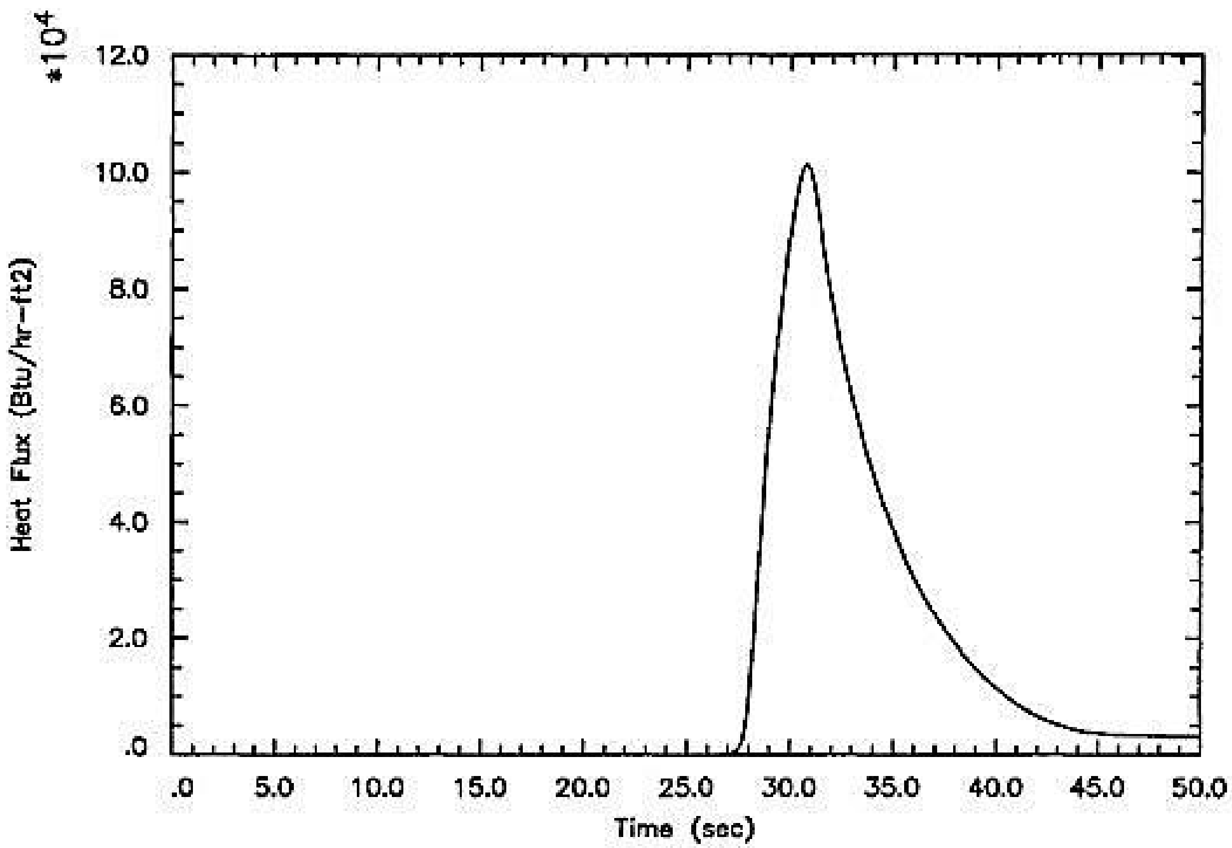


FIGURE 15.4.1-4

CORE AVERAGE FLUID AND FUEL TEMPERATURES FOR UNCONTROLLED RCCA BANK WITHDRAWAL AT HZP

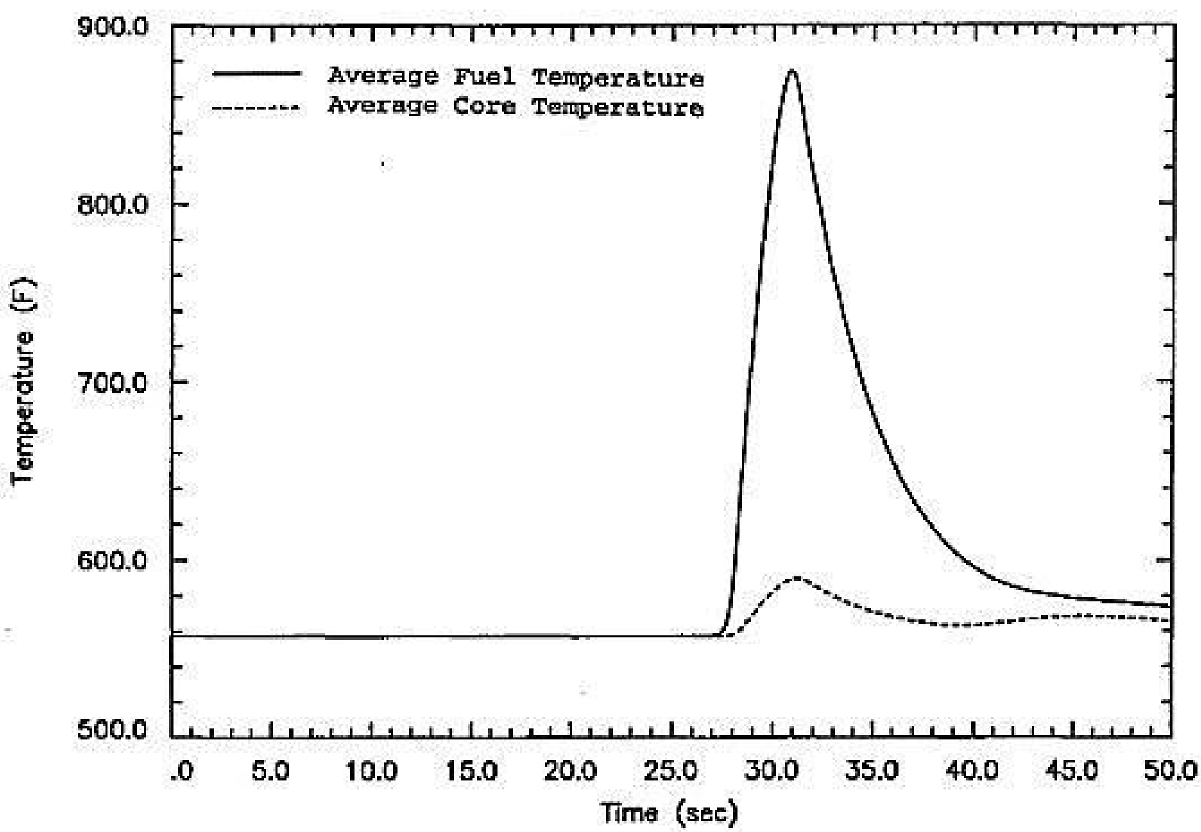


FIGURE 15.4.2-1

MINIMUM DNBR VS REACTIVITY INSERTION RATE FOR ALL RCCA BANK WITHDRAWAL CASES

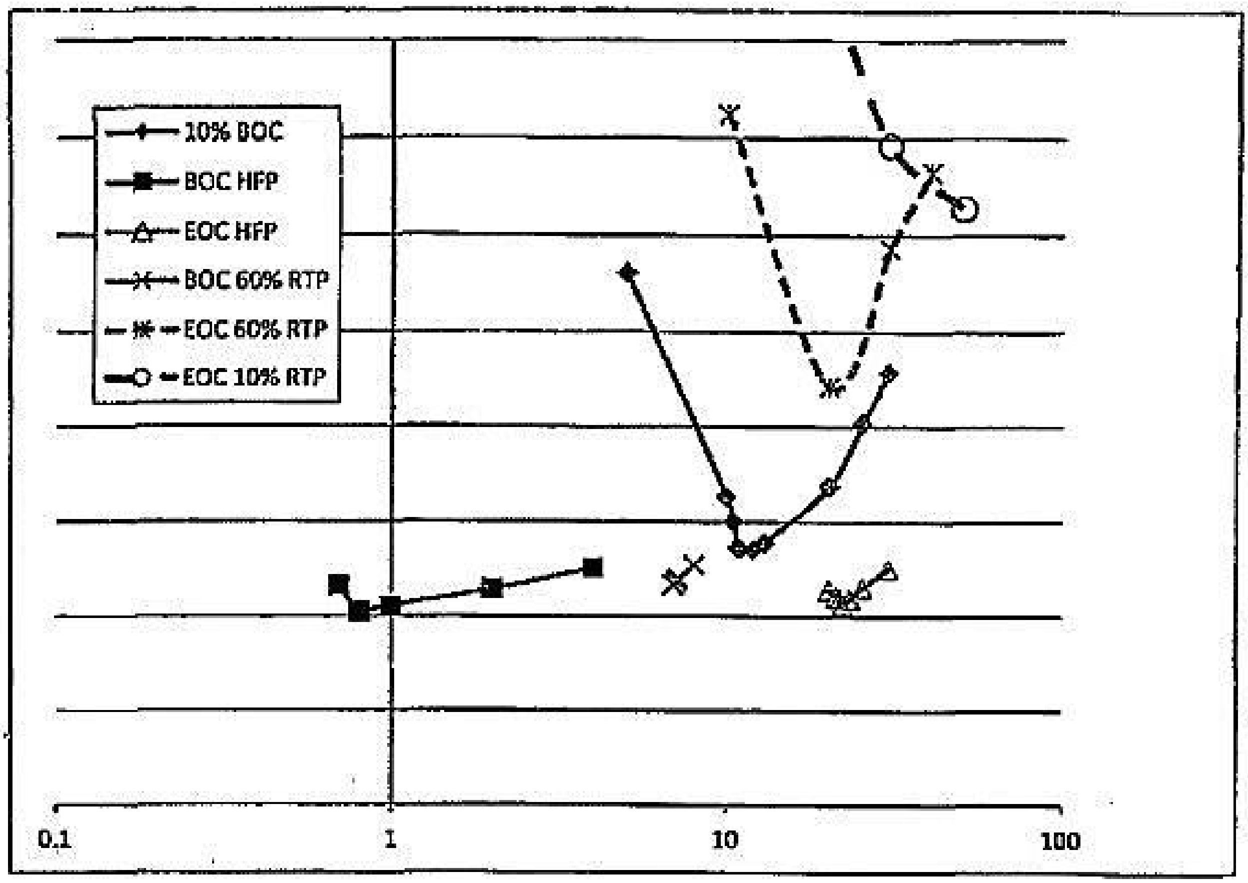


FIGURE 15.4.2-2

REACTOR POWER FOR LIMITING RCCA BANK WITHDRAWAL CASE

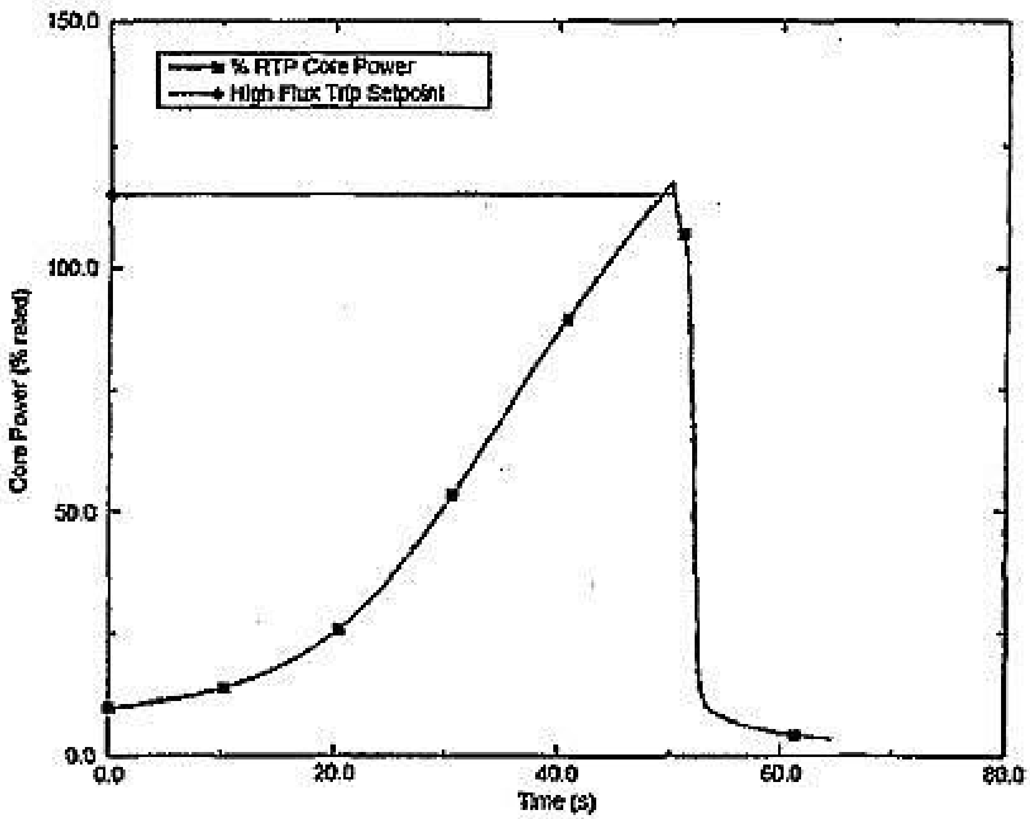


FIGURE 15.4.2-3

REACTIVITIES FOR LIMITING RCCA BANK WITHDRAWAL CASE

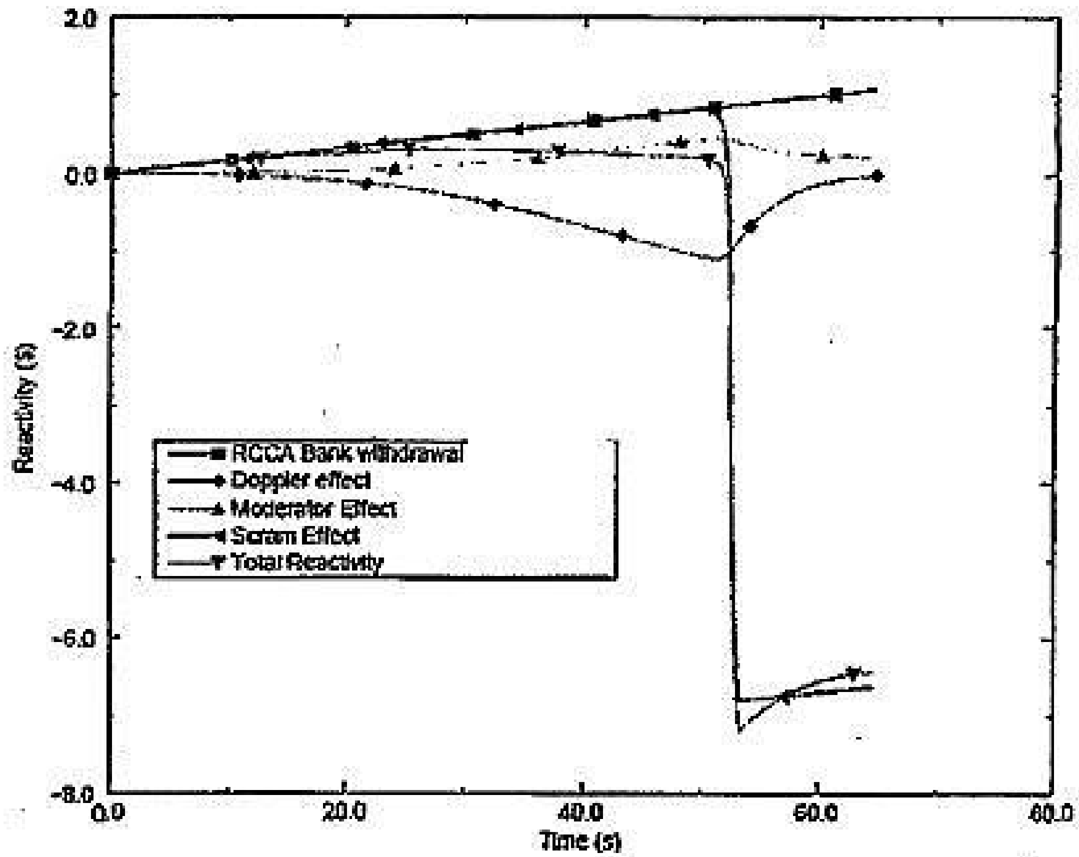


FIGURE 15.4.2-5

CORE POWER BASED ON ROD SURFACE HEAT FLUX FOR LIMITING RCCA BANK WITHDRAWAL CASE

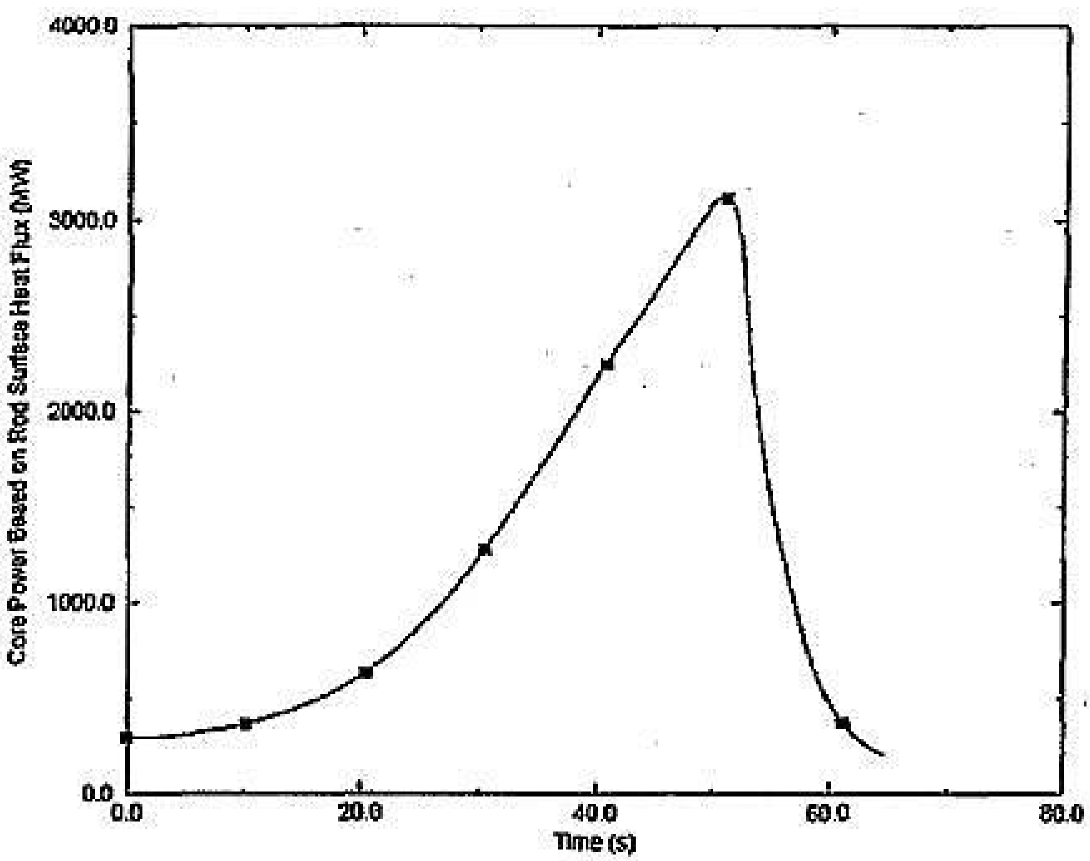


FIGURE 15.4.2-6

PRIMARY TEMPERATURES FOR LIMITING RCCA BANK WITHDRAWAL CASE

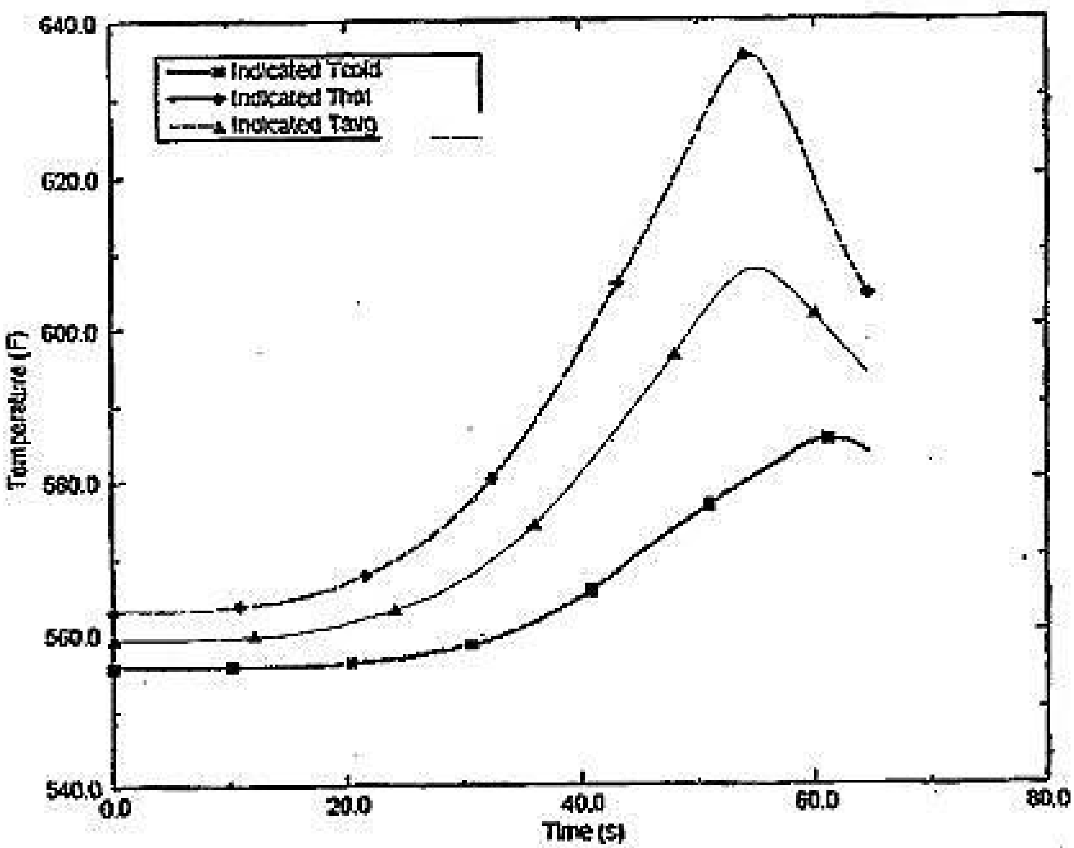


FIGURE 15.4.2-7

PRESSURIZER LEVEL FOR LIMITING RCCA BANK WITHDRAWAL CASE

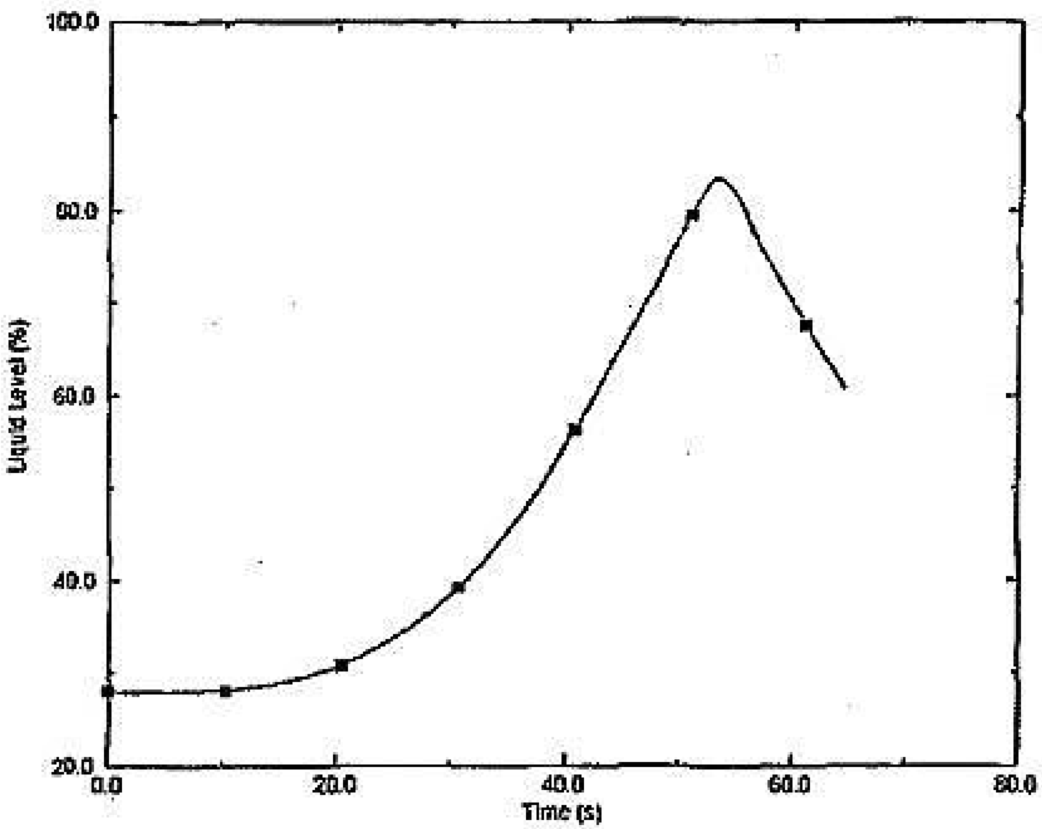


FIGURE 15.4.2-8

RCS MASS FLOW RATE FOR LIMITING RCCA BANK WITHDRAWAL CASE

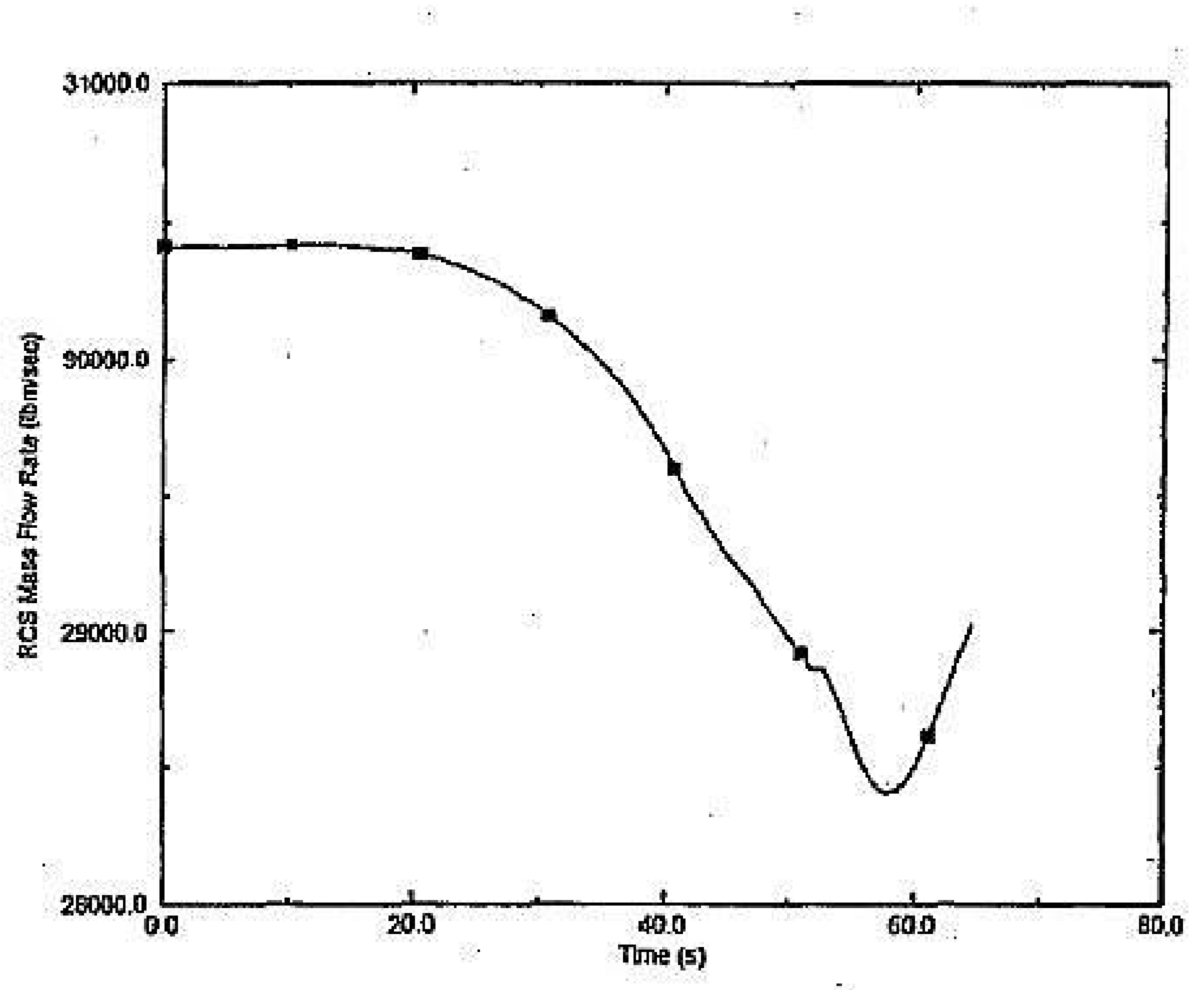


FIGURE 15.4.2-9

PRESSURIZER PRESSURE FOR LIMITING RCCA BANK WITHDRAWAL CASE

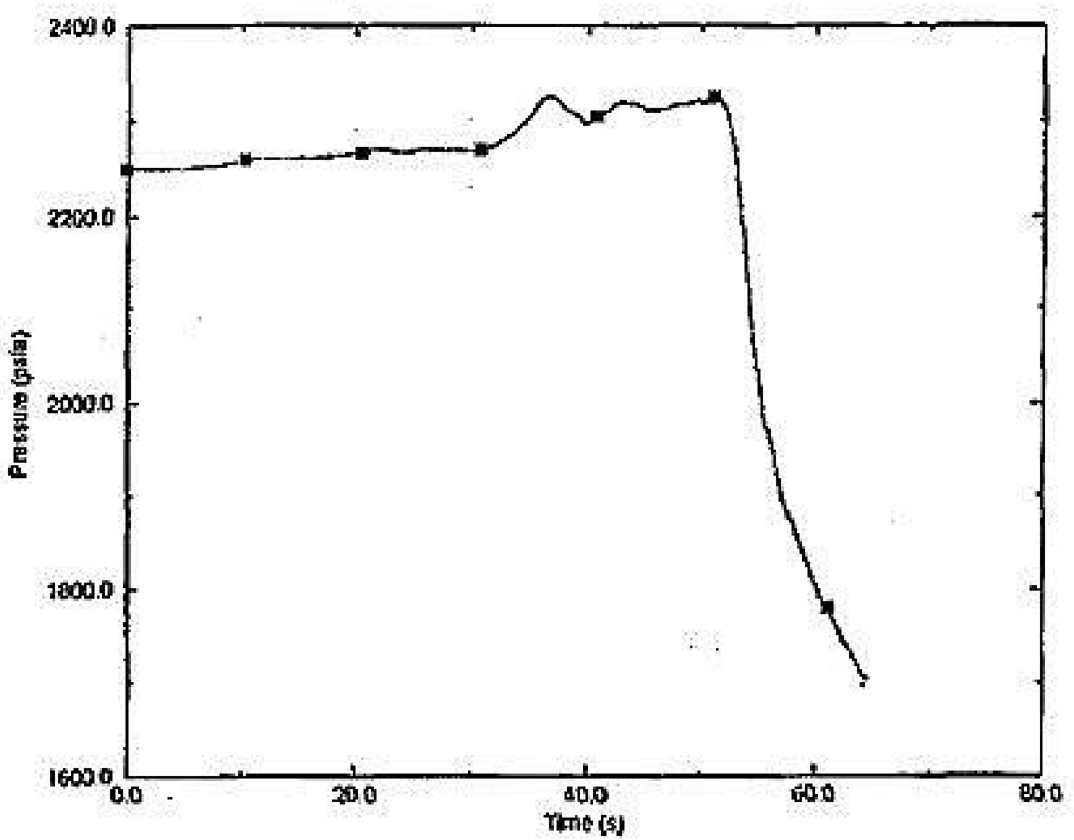


FIGURE 15.4.3-1

REACTOR POWER FOR LIMITING DROPPED ROD CASE

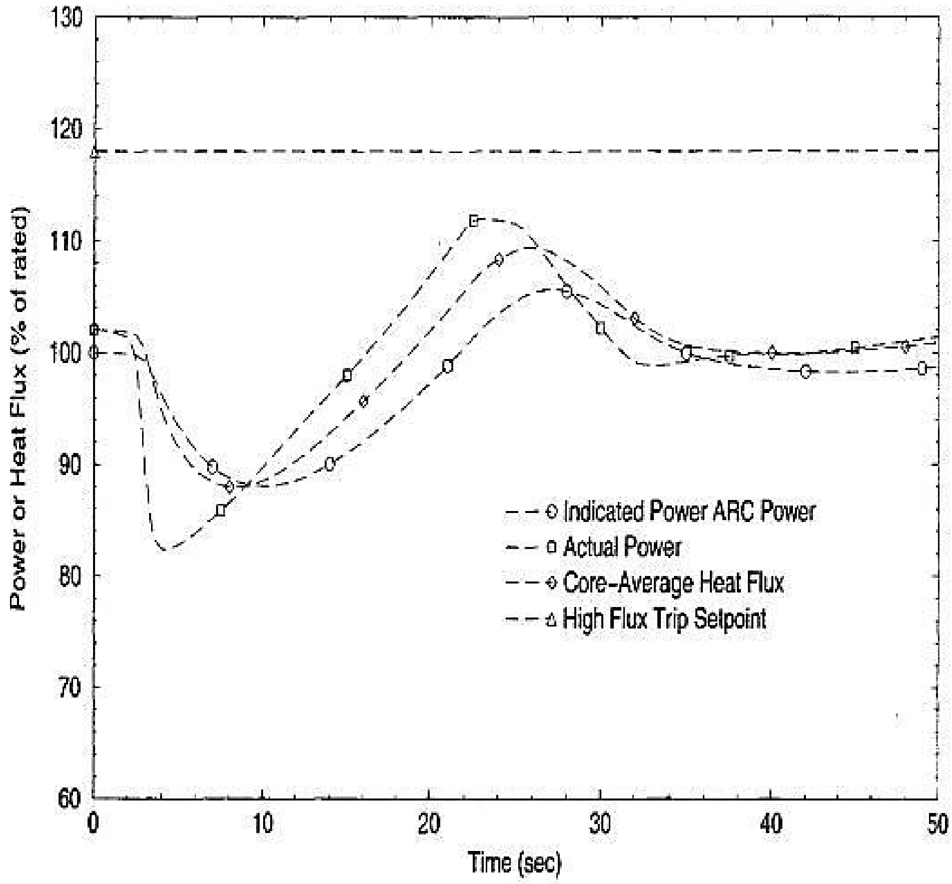


FIGURE 15.4.3-2

AVERAGE CORE LINEAR HEAT GENERATION RATE FOR LIMITING DROPPED ROD CASE

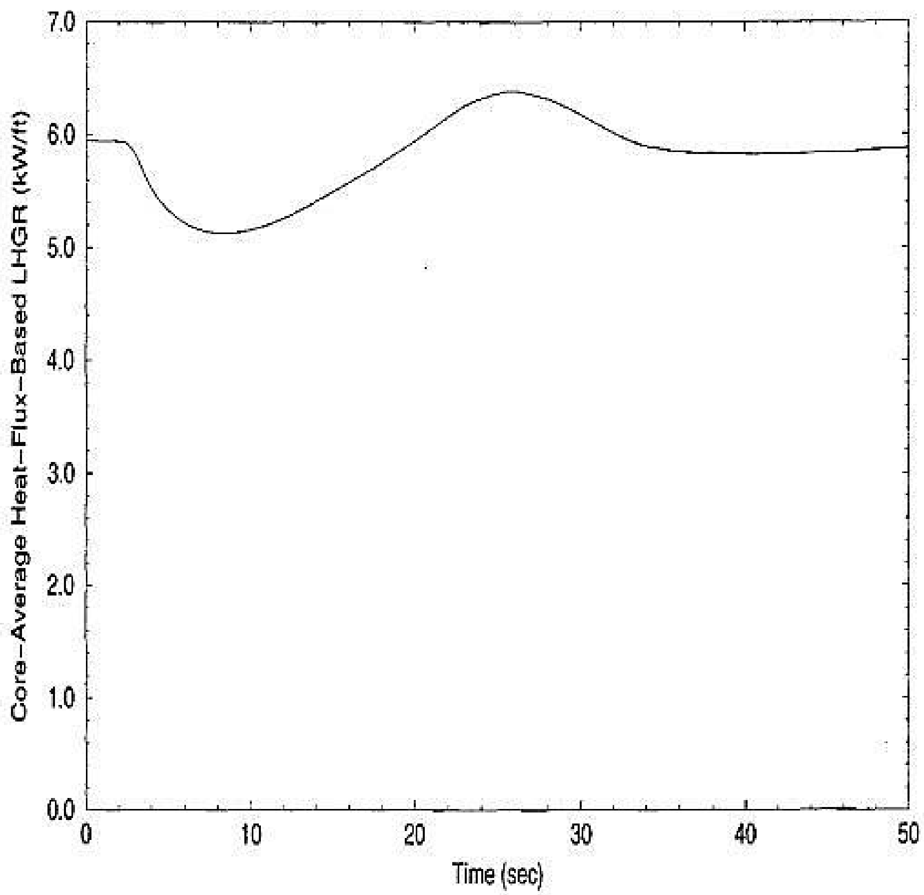


FIGURE 15.4.3-3

PRESSURIZER PRESSURE FOR LIMITING DROPPED ROD CASE

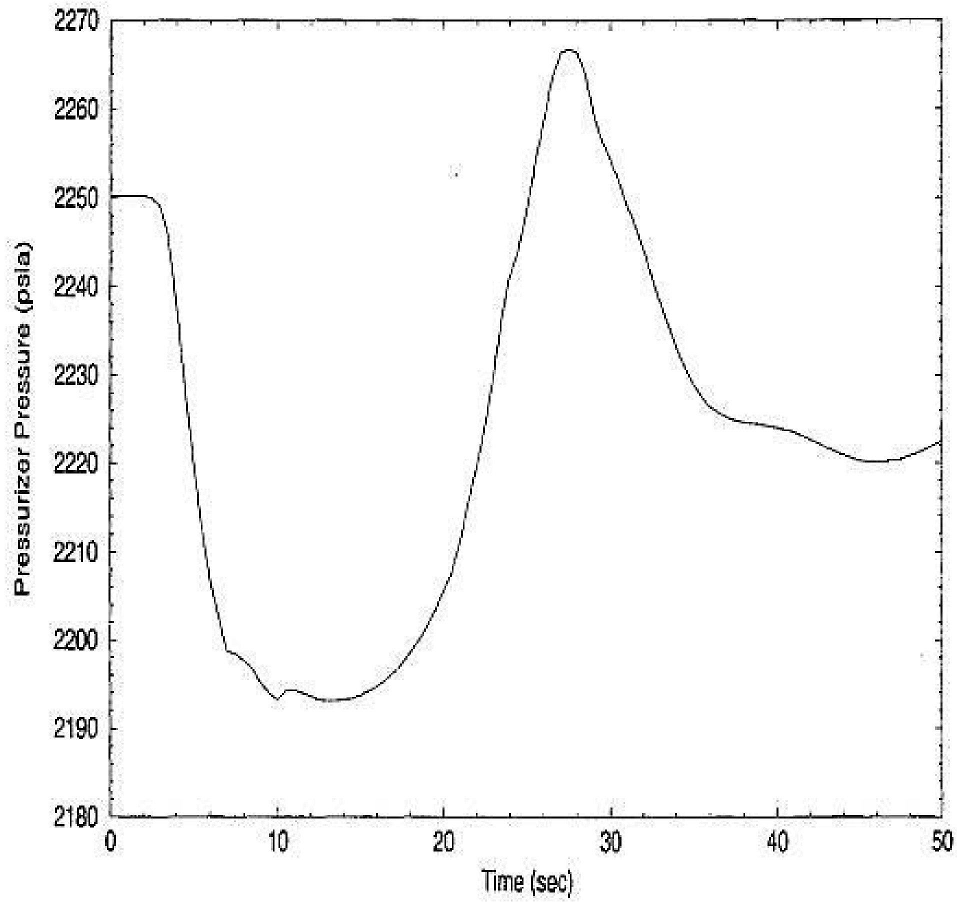


FIGURE 15.4.3-4

PRIMARY TEMPERATURES FOR LIMITING DROPPED ROD CASE

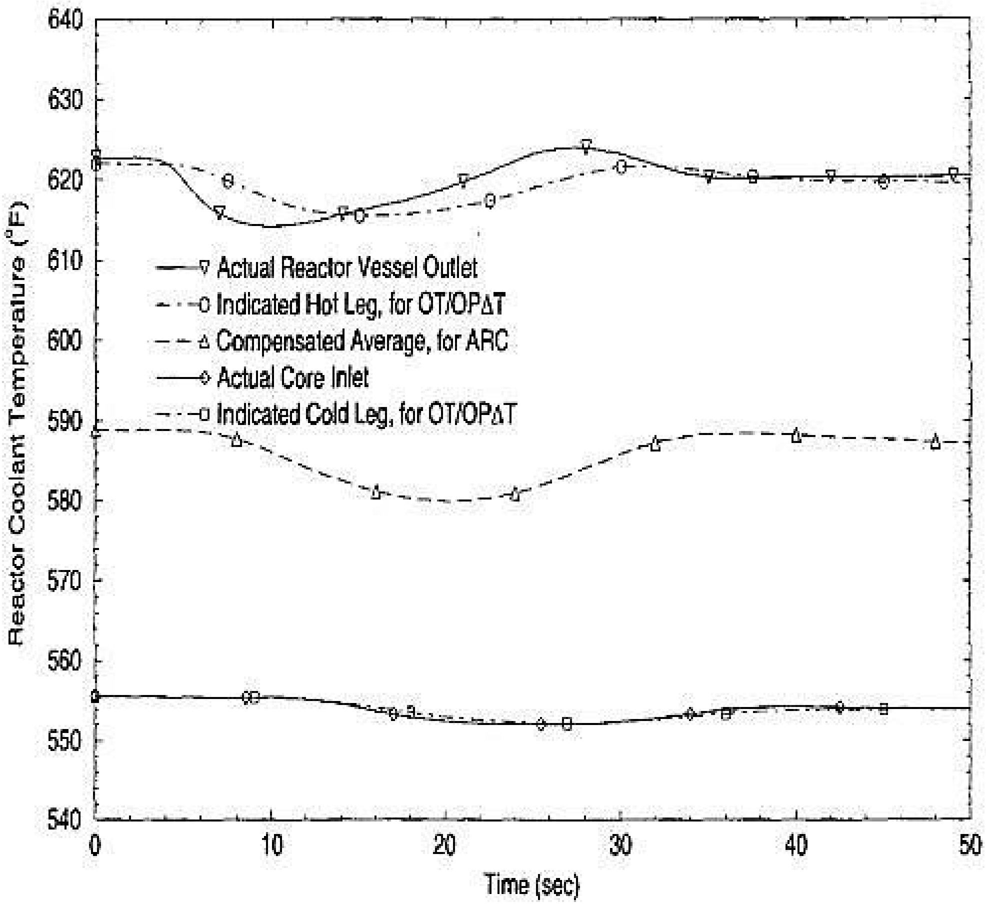


FIGURE 15.4.3-6

CORE INLET FLOW FOR LIMITING DROPPED ROD CASE

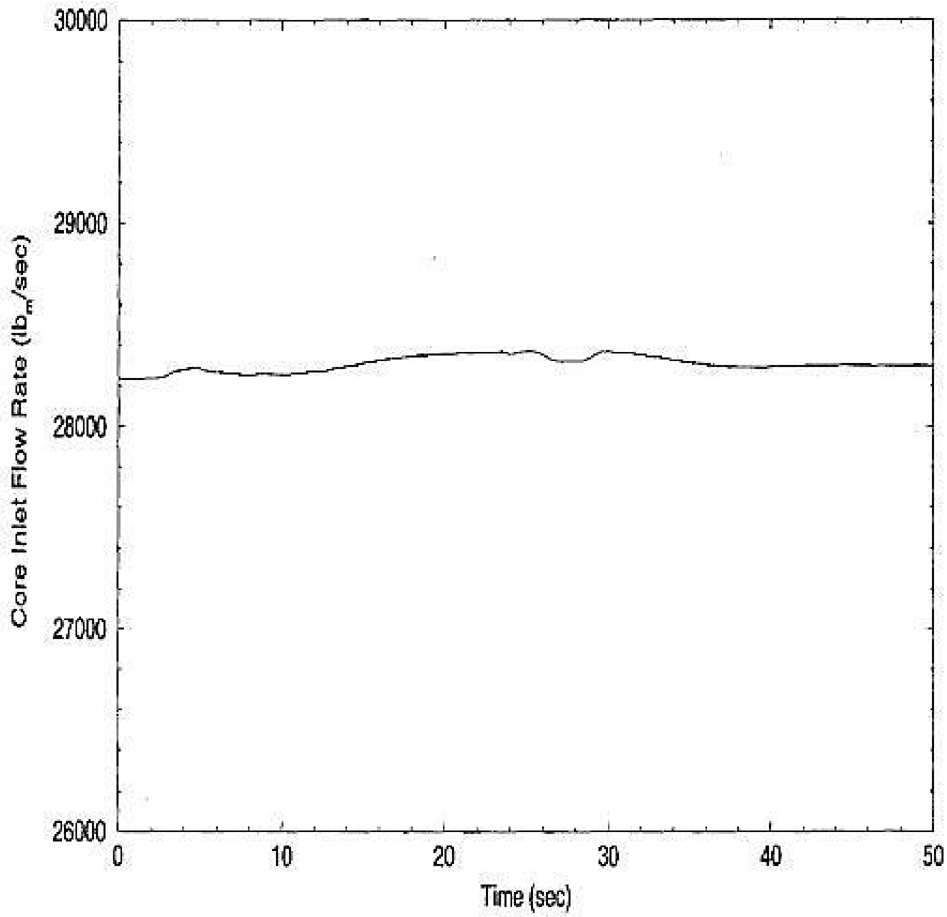


FIGURE 15.4.3-7

PRESSURIZER LEVEL FOR LIMITING DROPPED ROD CASE

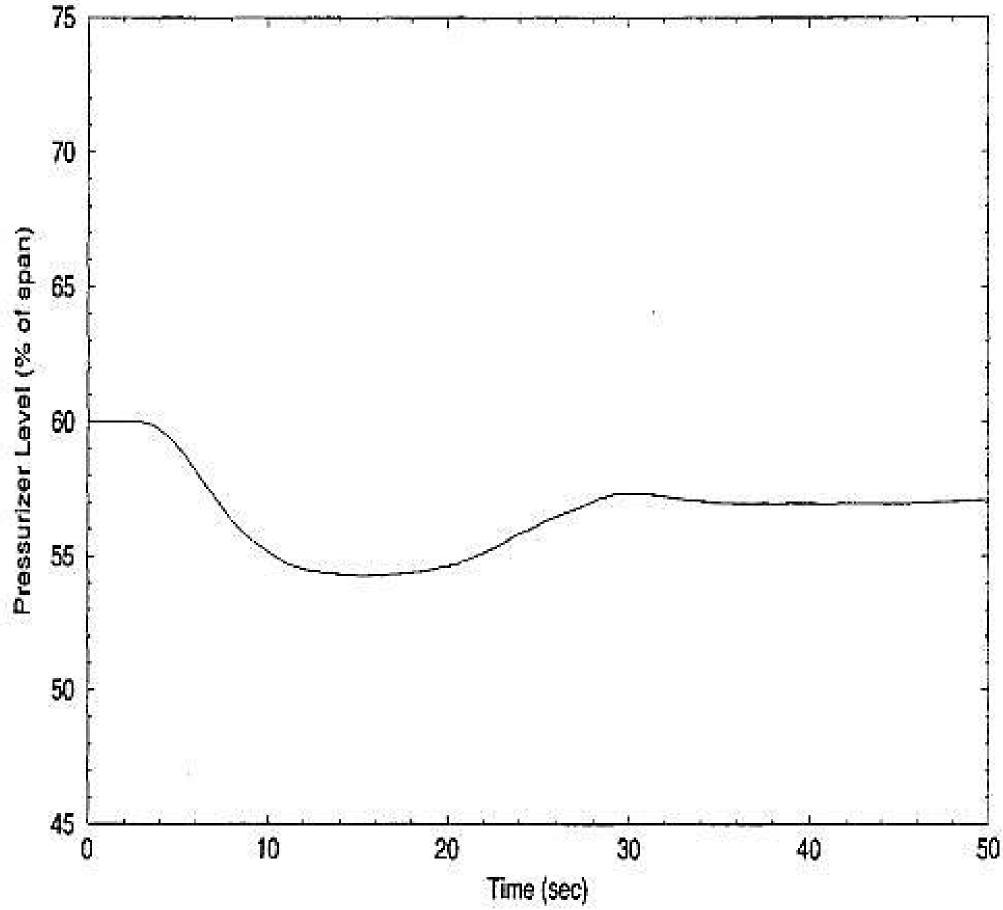


FIGURE 15.4.3-8

REACTIVITY FOR LIMITING DROPPED ROD CASE

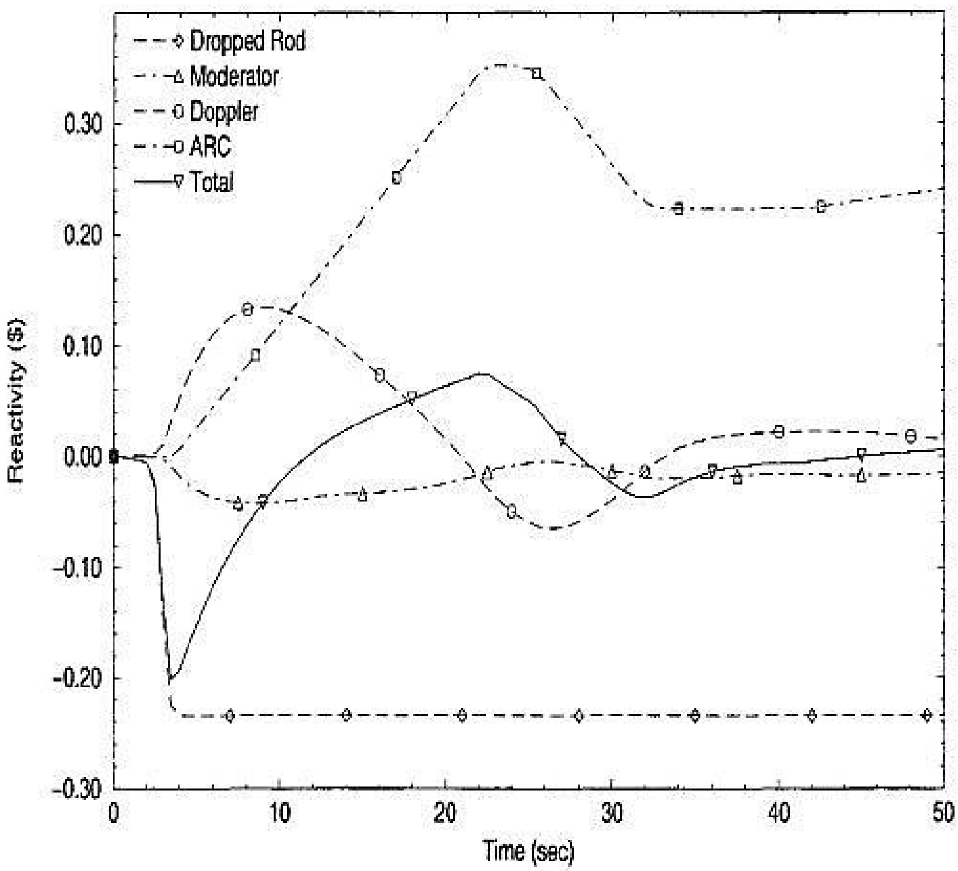


FIGURE 15.4.3-10

ROD POSITION FOR LIMITING DROPPED ROD CASE

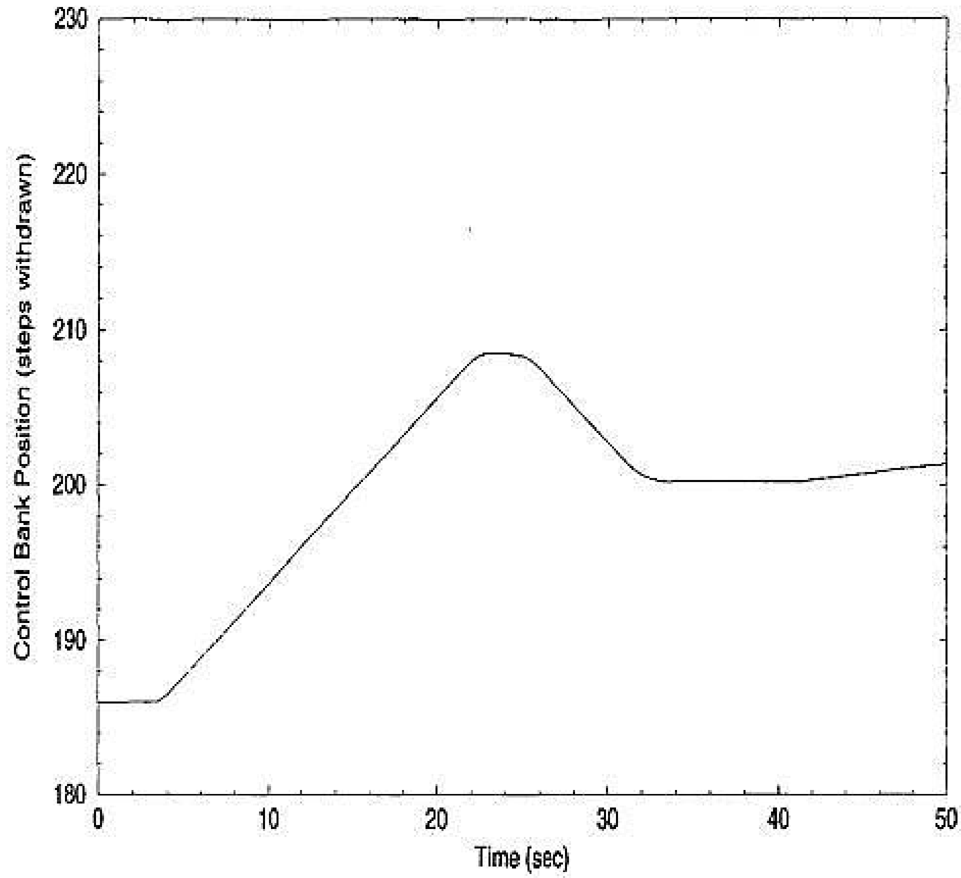


FIGURE 15.4.3-11

ROD SPEED FOR LIMITING DROPPED ROD CASE

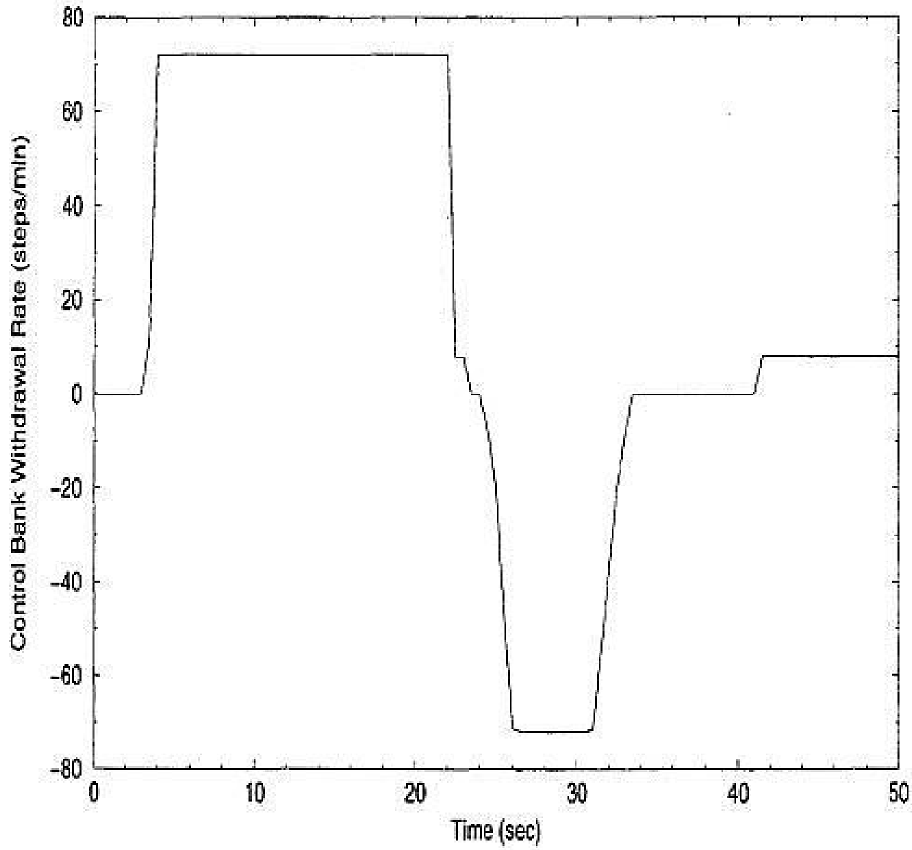


FIGURE 15.4.8-9

REACTOR POWER FOR RCCA EJECTION – LIMITING CASE

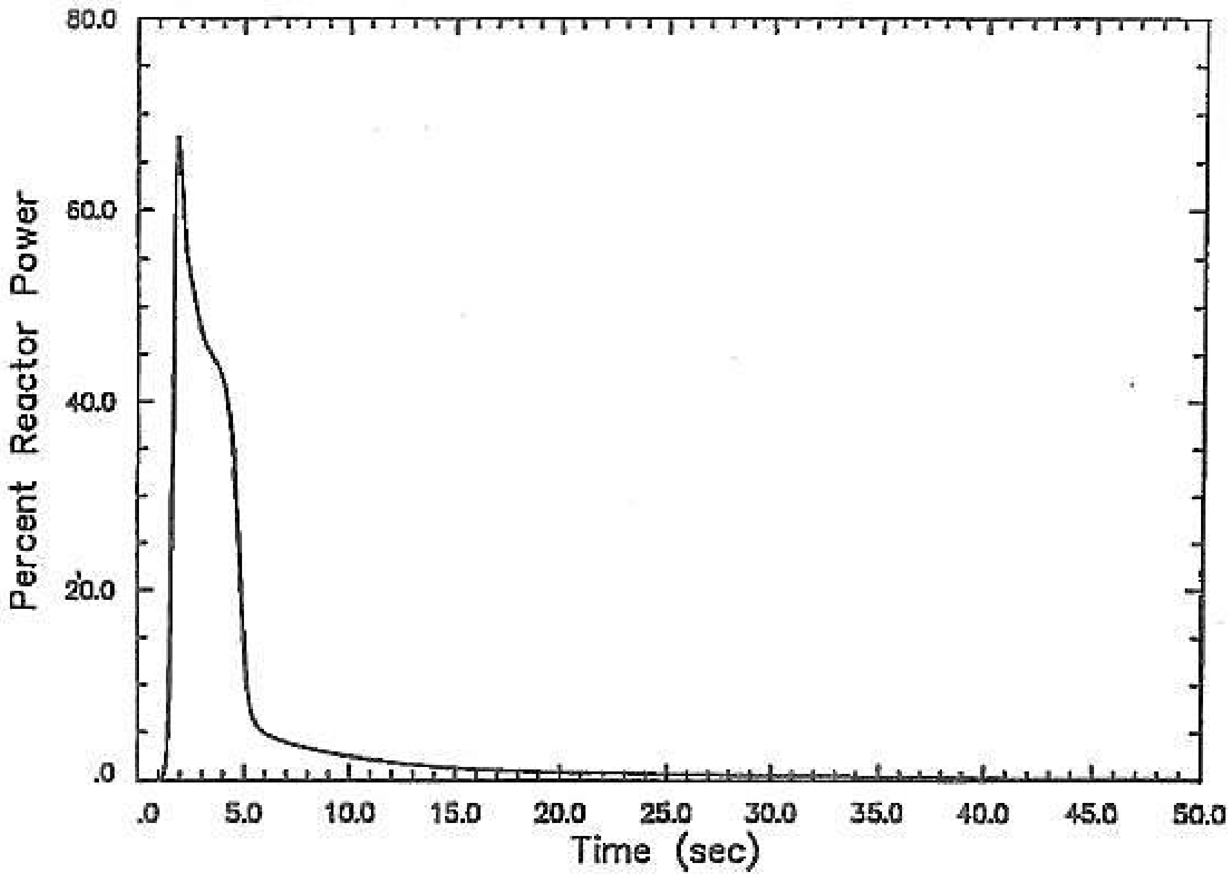


FIGURE 15.4.8-10

CORE AVERAGE HEAT FLUX FOR RCCA EJECTION – LIMITING CASE

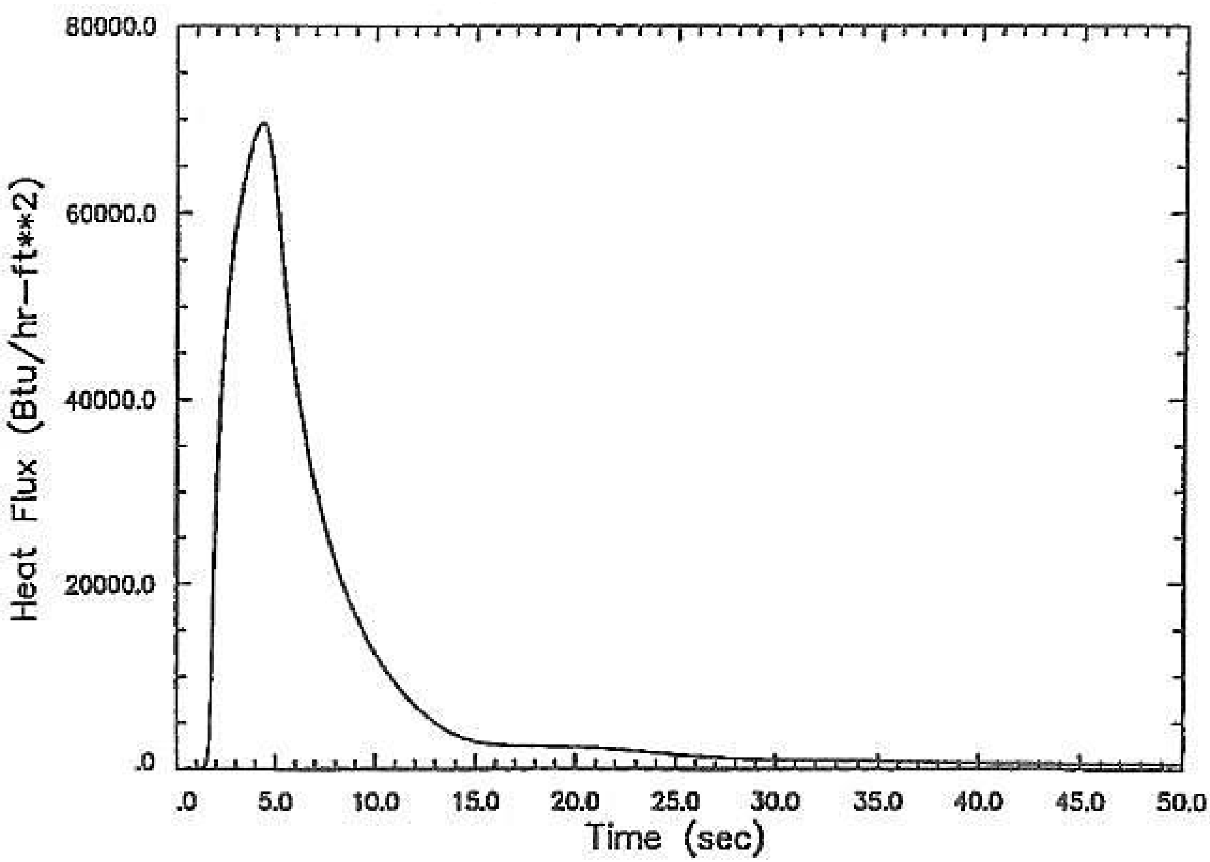


FIGURE 15.4.8-11

TOTAL CORE REACTIVITY FOR RCCA EJECTION – LIMITING CASE

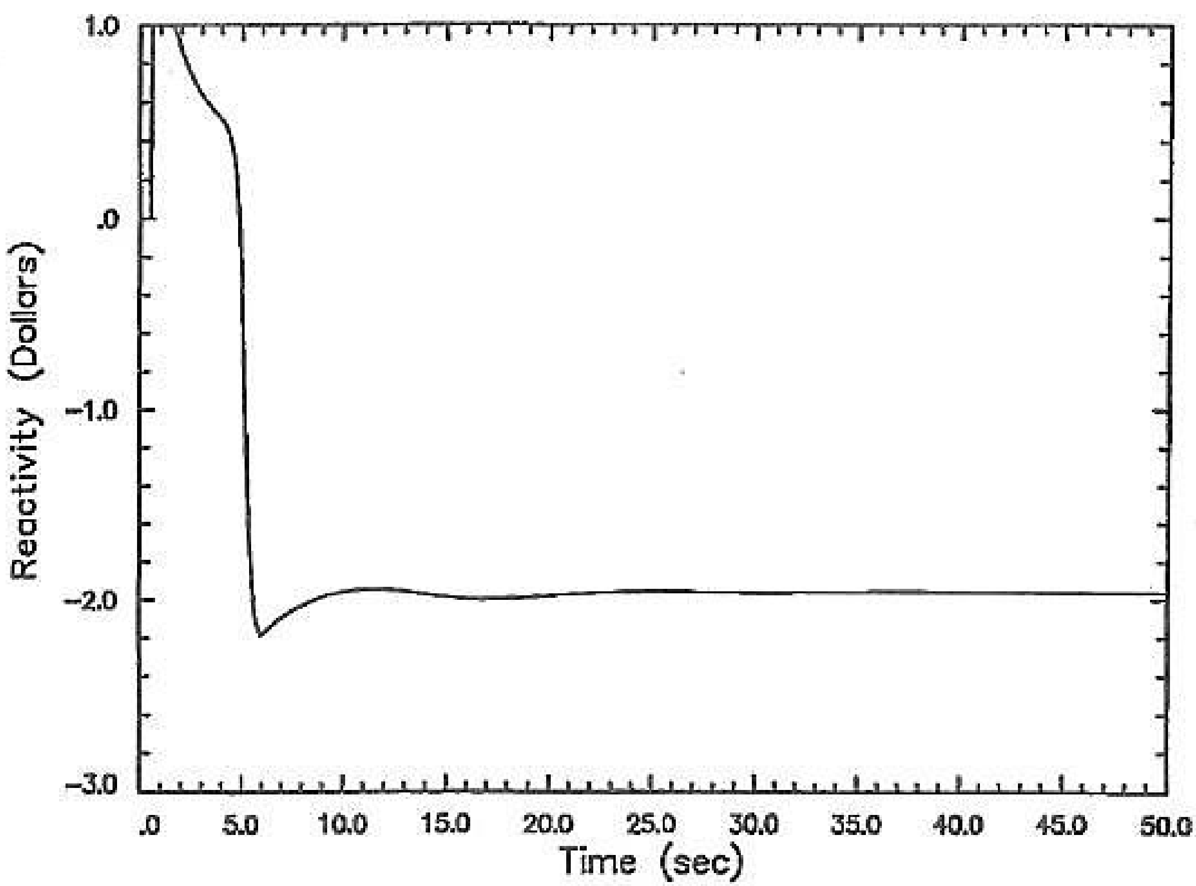


FIGURE 15.4.8-12

PRIMARY SYSTEM TEMPERATURES FOR RCCA EJECTION – LIMITING CASE

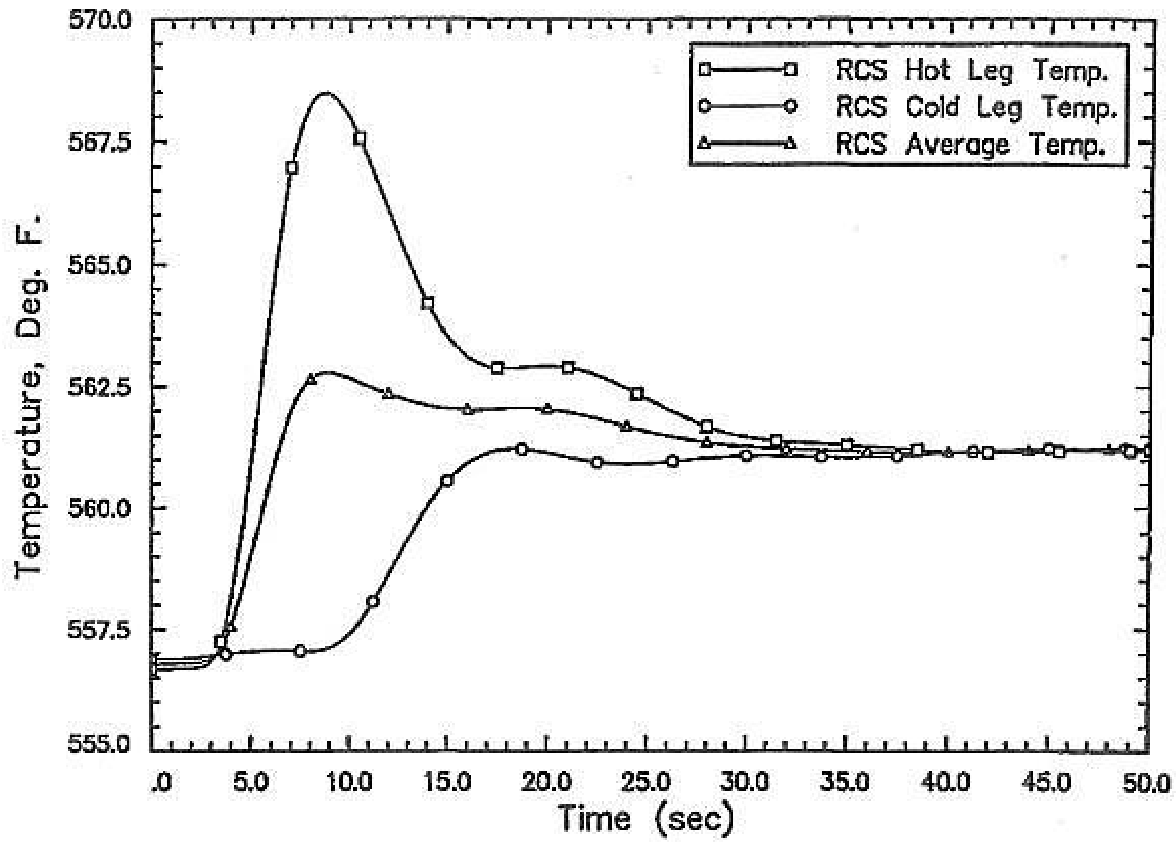


FIGURE 15.4.8-13

CORE INLET MASS FLUX FOR RCCA EJECTION – LIMITING CASE

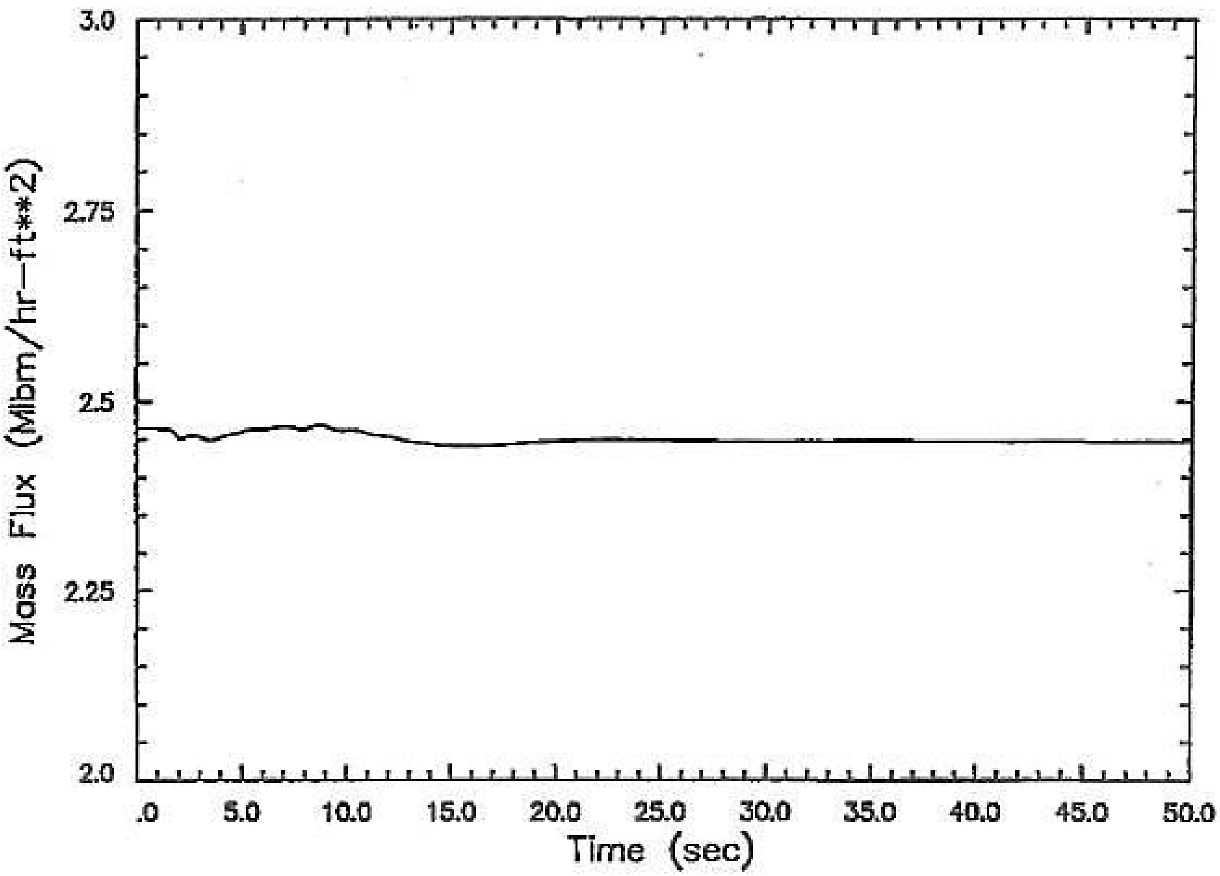


FIGURE 15.4.8-14

LOWER HEAD PRESSURE FOR RCCA EJECTION – LIMITING CASE

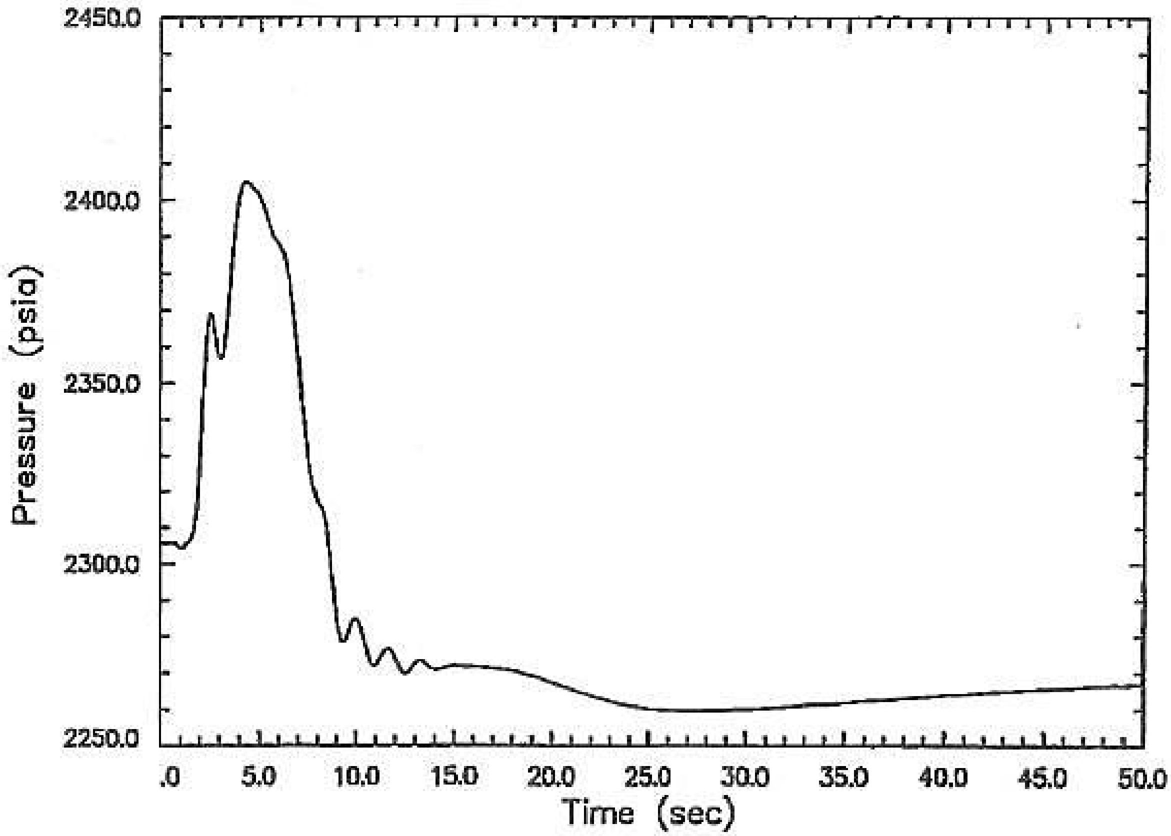


FIGURE 15.4.8-15

AVERAGE FUEL TEMPERATURE FOR RCCA EJECTION – LIMITING CASE

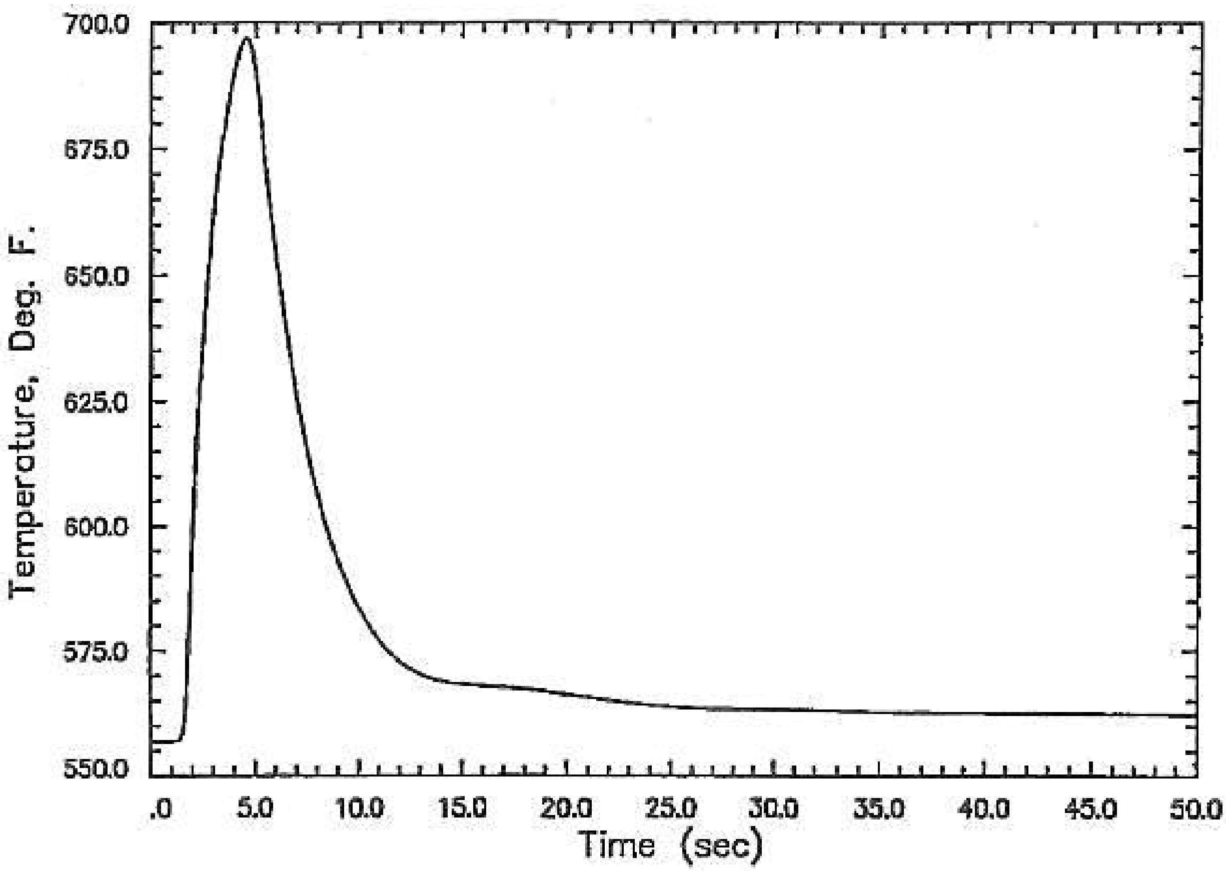


FIGURE 15.4.8-16

PRESSURIZER LEVEL FOR RCCA EJECTION – LIMITING CASE

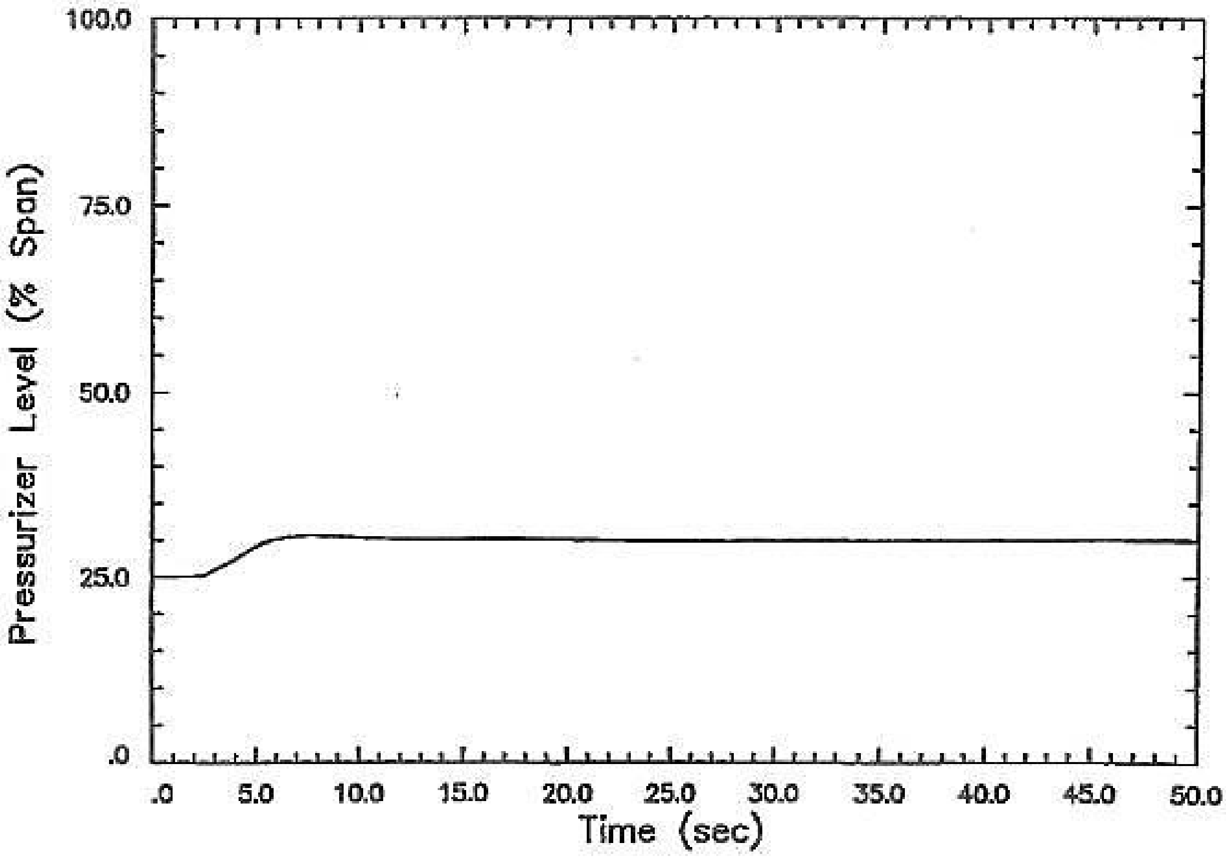


FIGURE 15.5.1-1

REACTOR POWER (BOC MDNBR CASE)

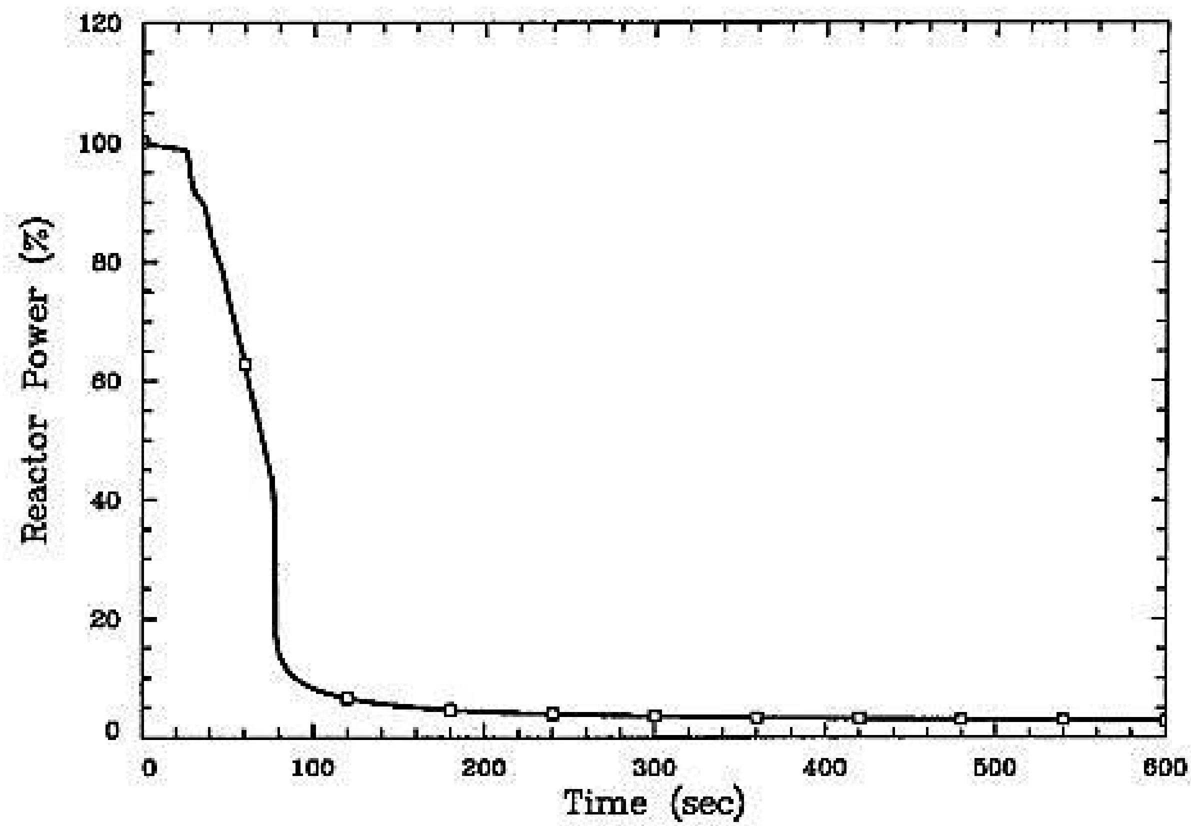


FIGURE 15.5.1-2

MAIN STEAM FLOW RATE (BOC MDNBR CASE)

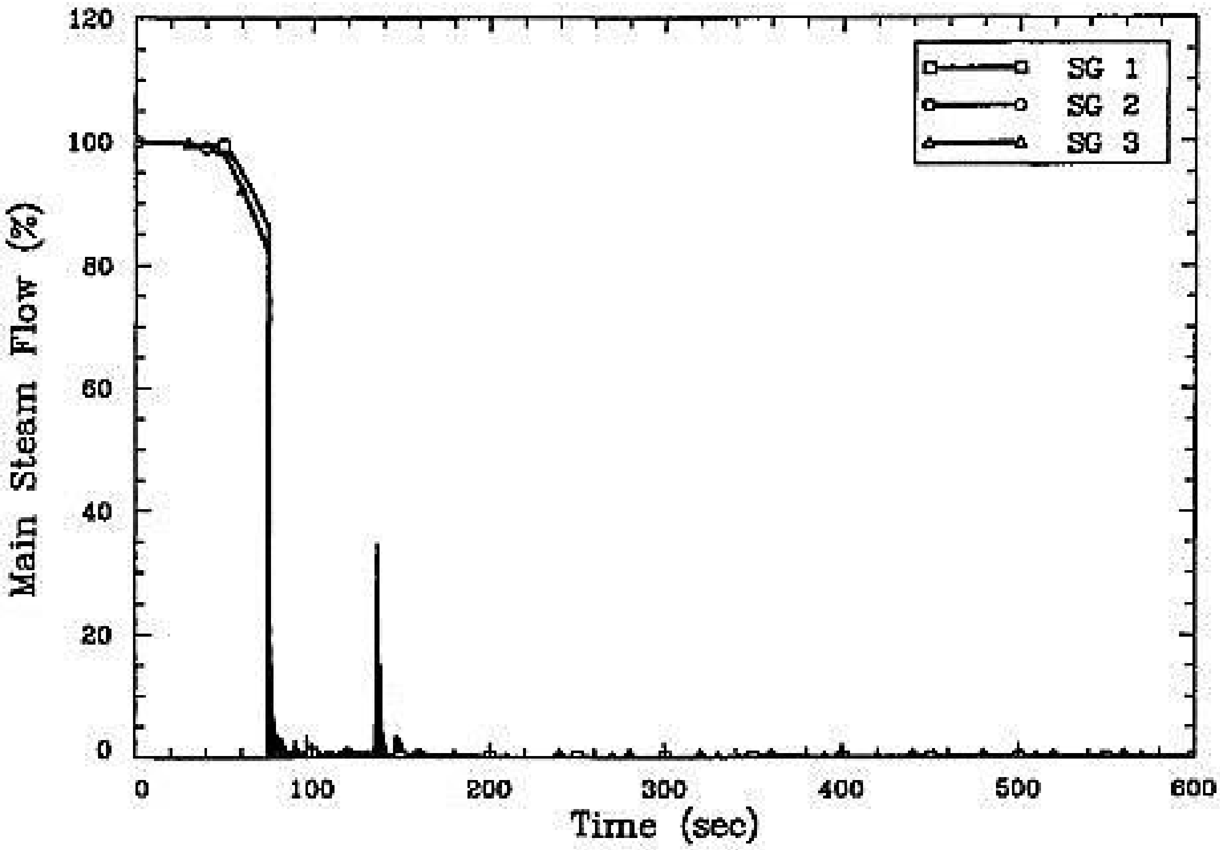


FIGURE 15.5.1-3

REACTOR COOLANT SYSTEM TEMPERATURES (BOC MDNBR CASE)

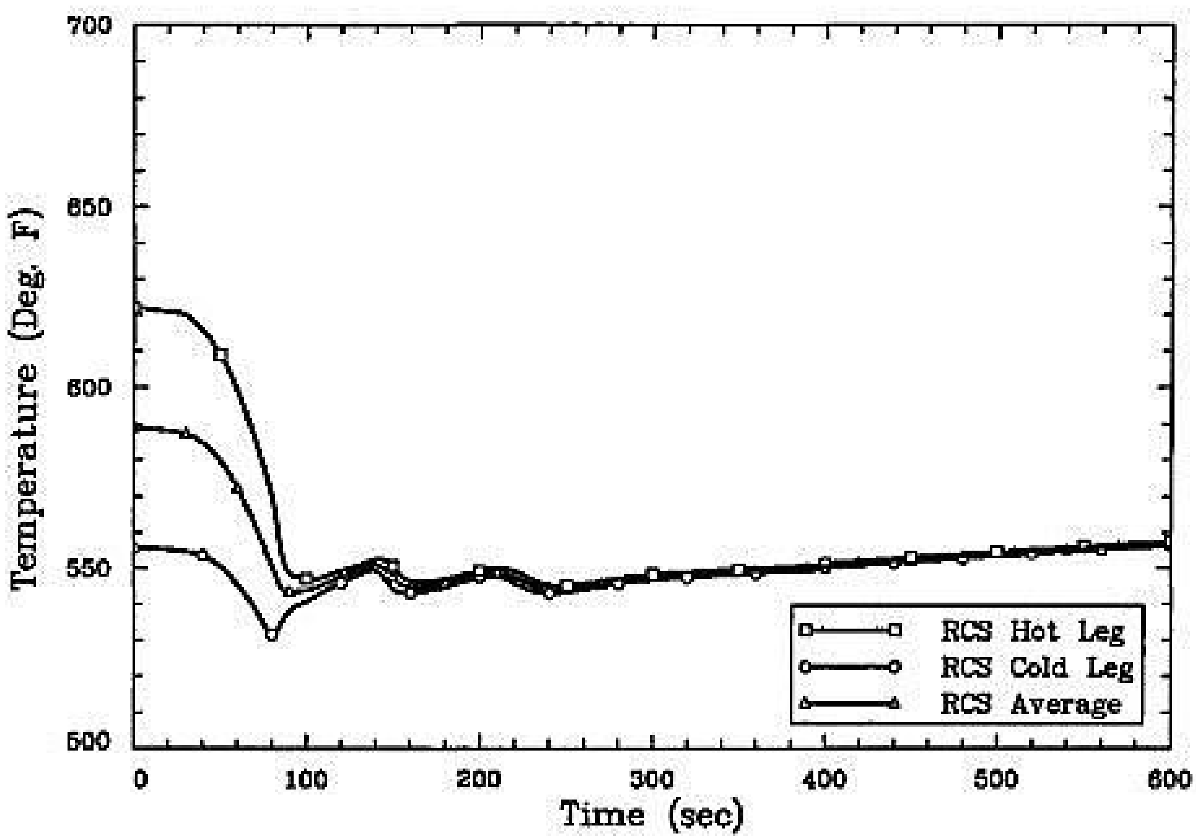


FIGURE 15.5.1-4

PRESSURIZER PRESSURE (BOC MDNBR CASE)

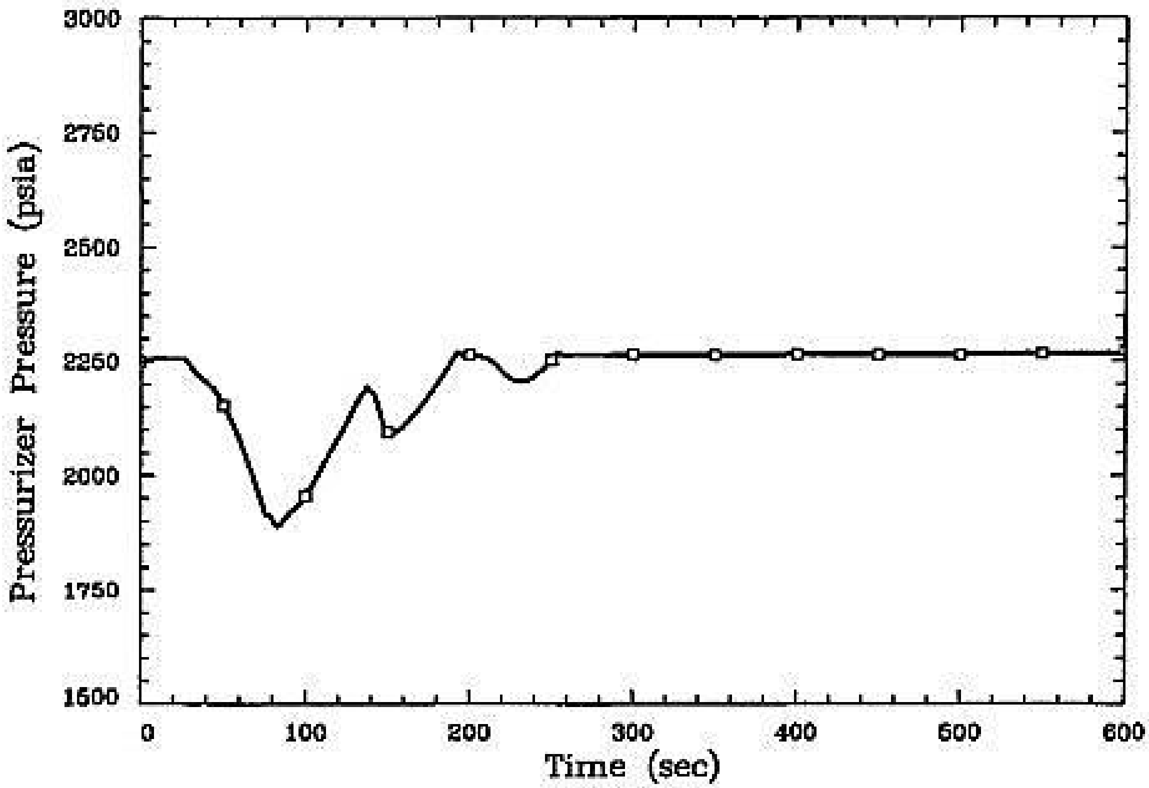


FIGURE 15.5.1-5

PRESSURIZER LEVEL (BOC MDNBR CASE)

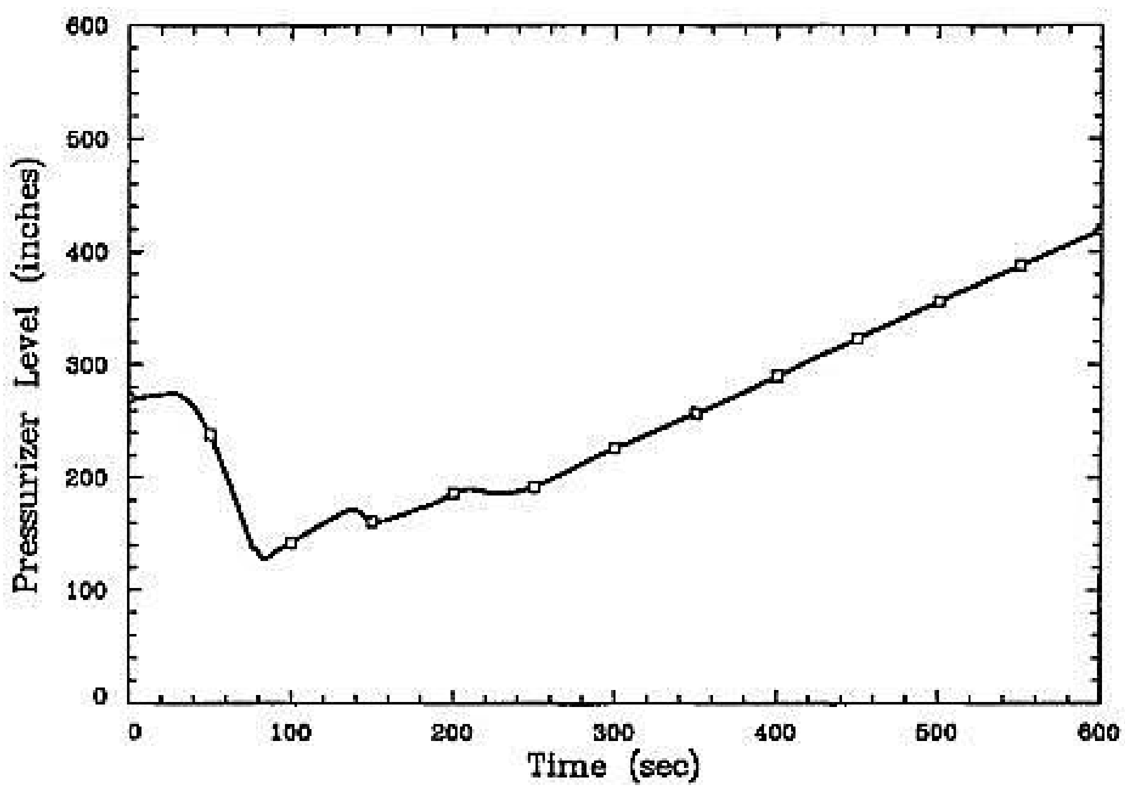


FIGURE 15.5.1-6

ANF-RELAP PREDICTED DNBR TREND (BOC MDNBR CASE)

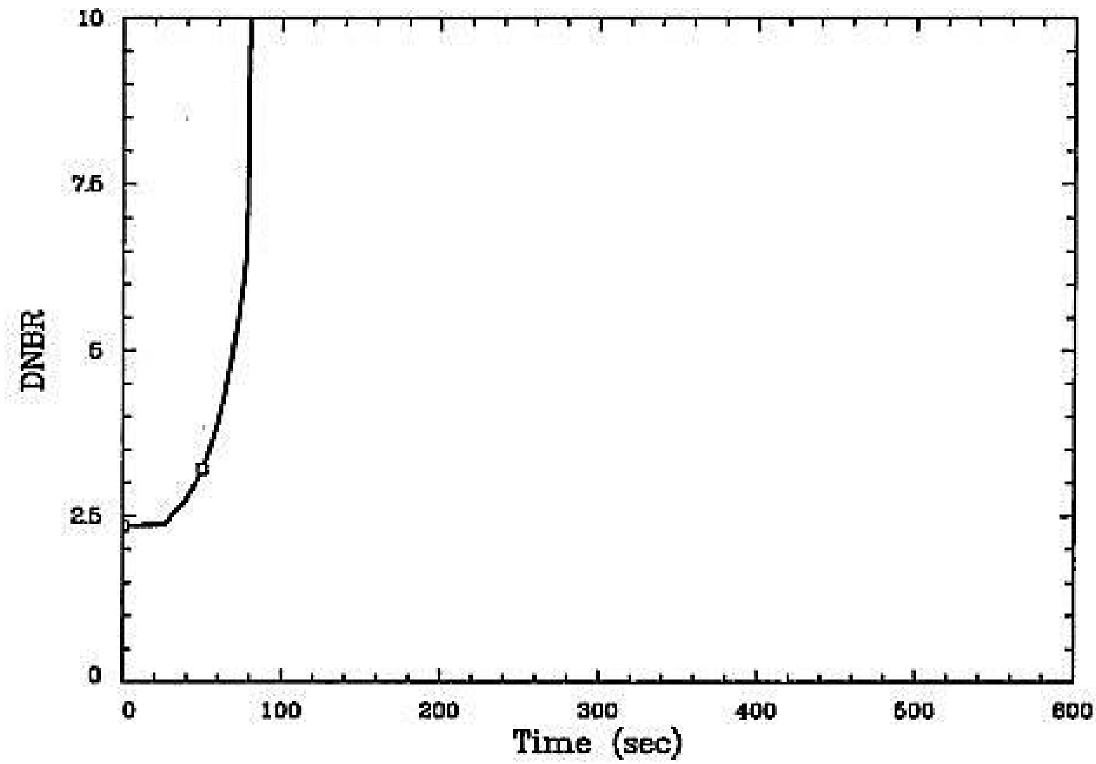


FIGURE 15.5.1-7

SAFETY INJECTION FLOW RATE (BOC MDNBR CASE)

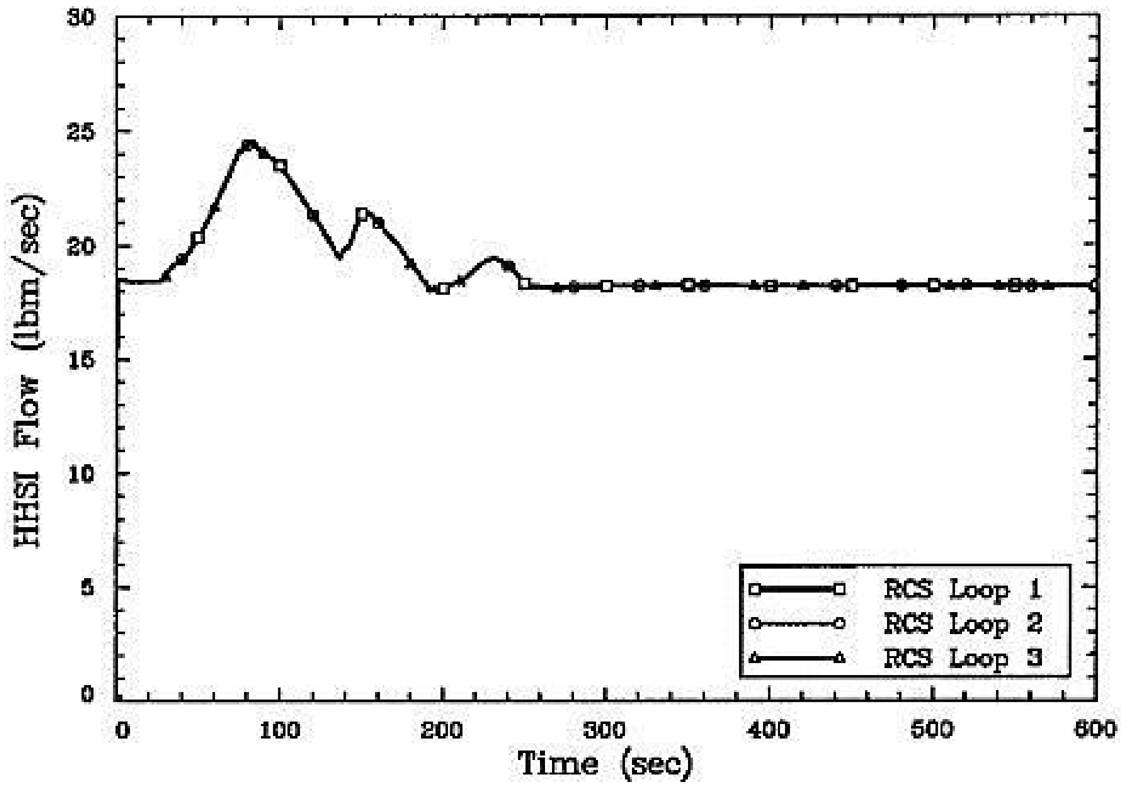


FIGURE 15.5.1-8

REACTOR POWER (EOC MDNBR CASE)

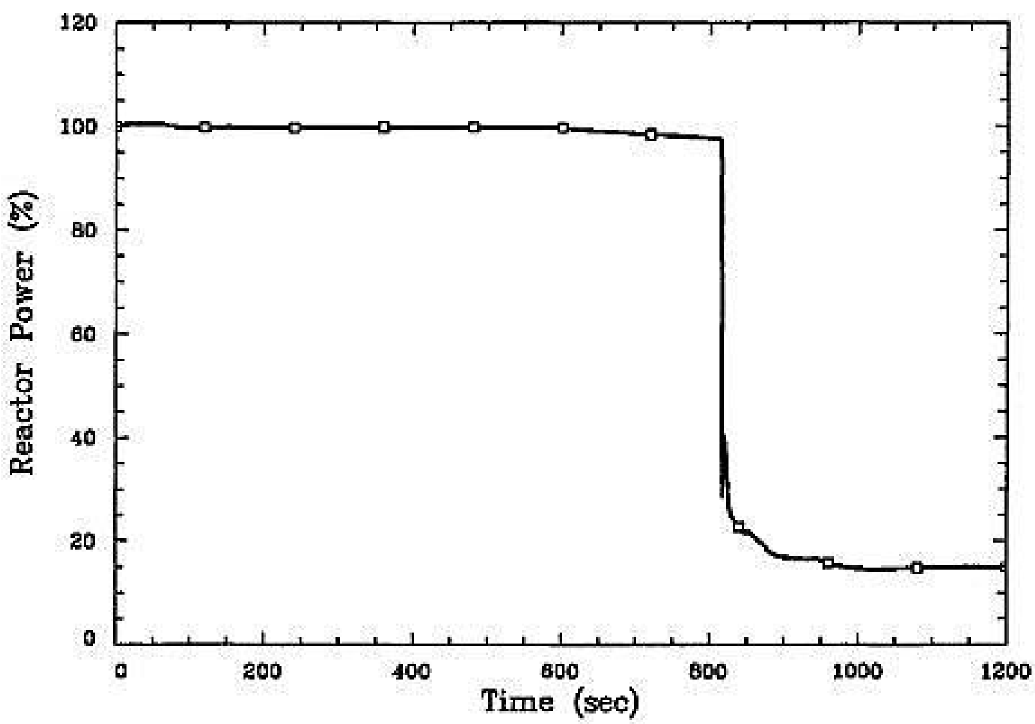


FIGURE 15.5.1-9

MAIN STEAM FLOW RATE (EOC MDNBR CASE)

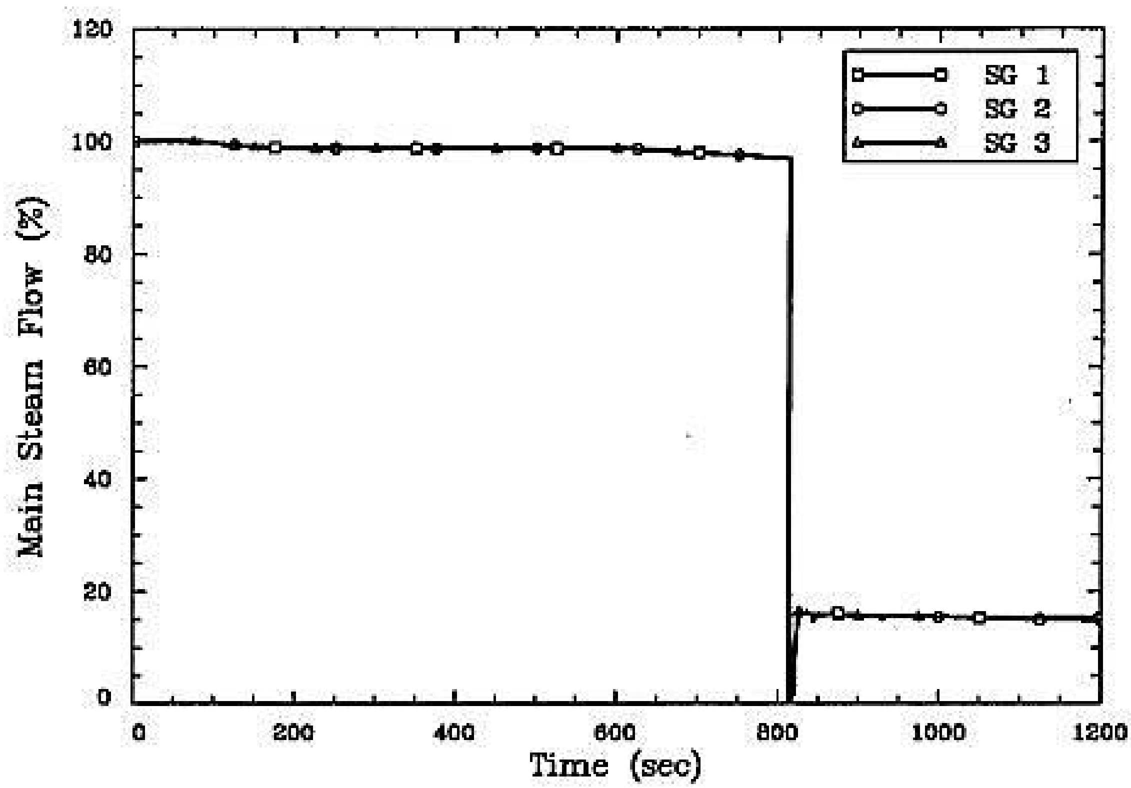


FIGURE 15.5.1-10

REACTOR COOLANT SYSTEM TEMPERATURES (EOC MDNBR CASE)

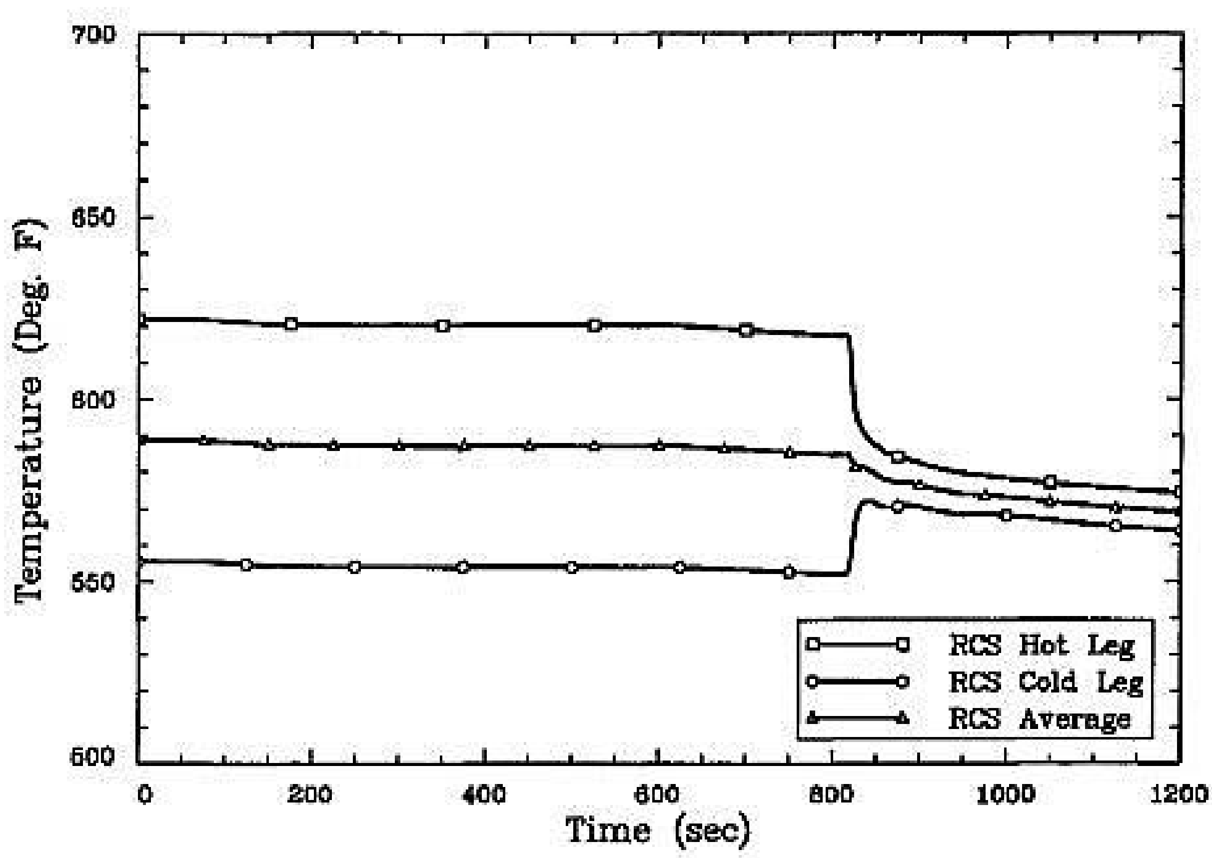


FIGURE 15.5.1-11

PRESSURIZER PRESSURE (EOC MDNBR CASE)

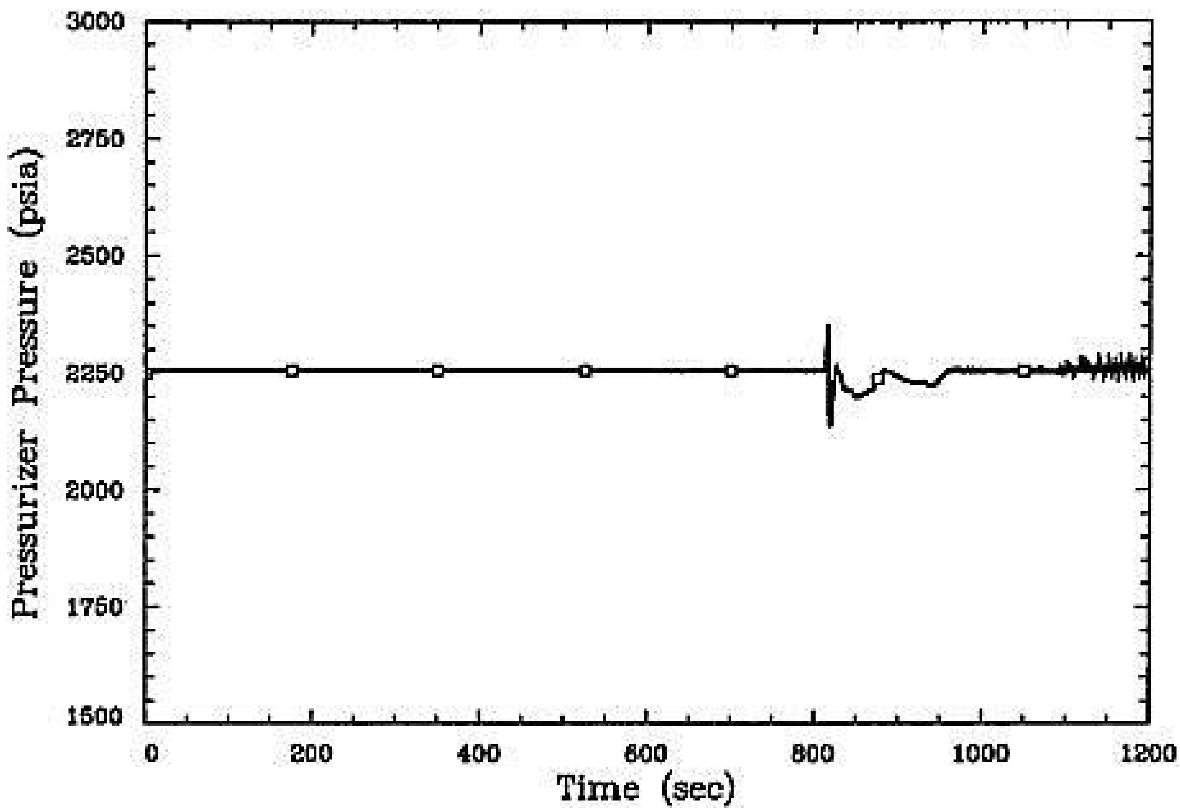


FIGURE 15.5.1-12

PRESSURIZER LIQUID LEVEL (EOC MDNBR CASE)

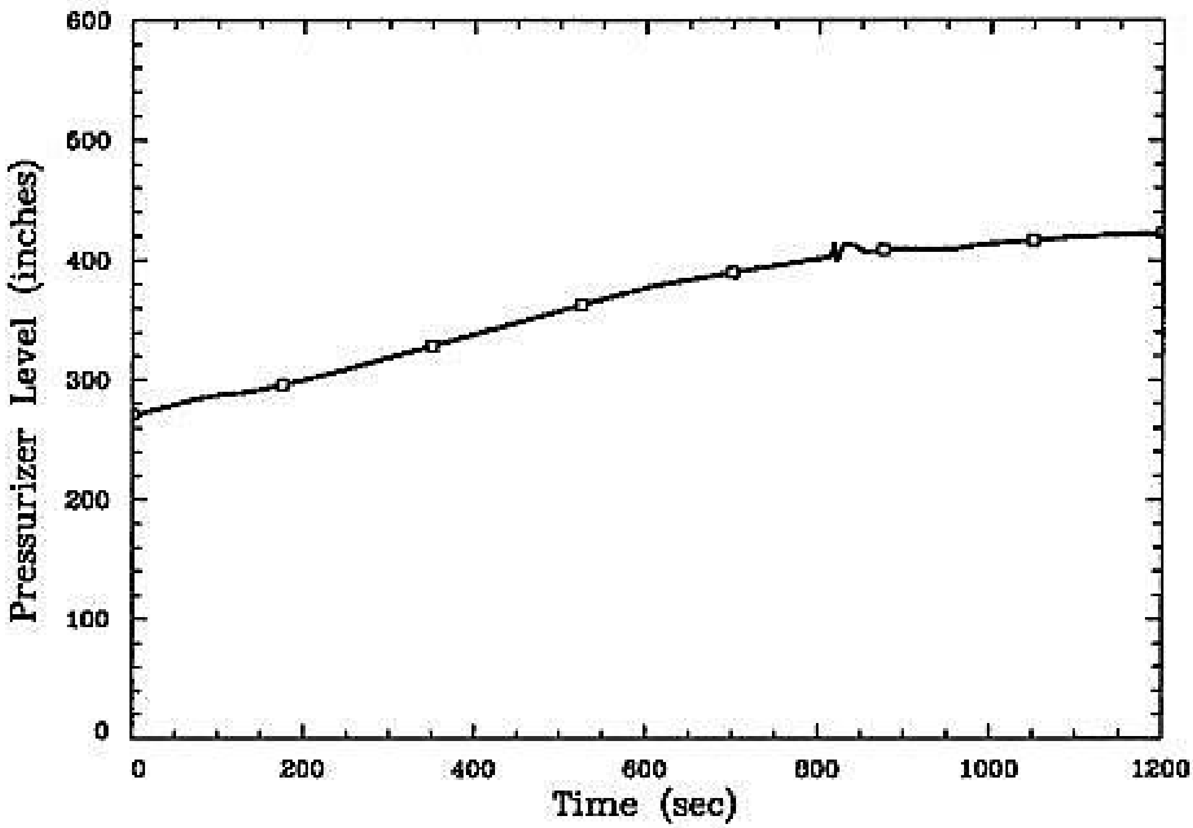


FIGURE 15.5.1-13

ANF-RELAP PREDICTED DNBR TREND (EOC MDNBR CASE)

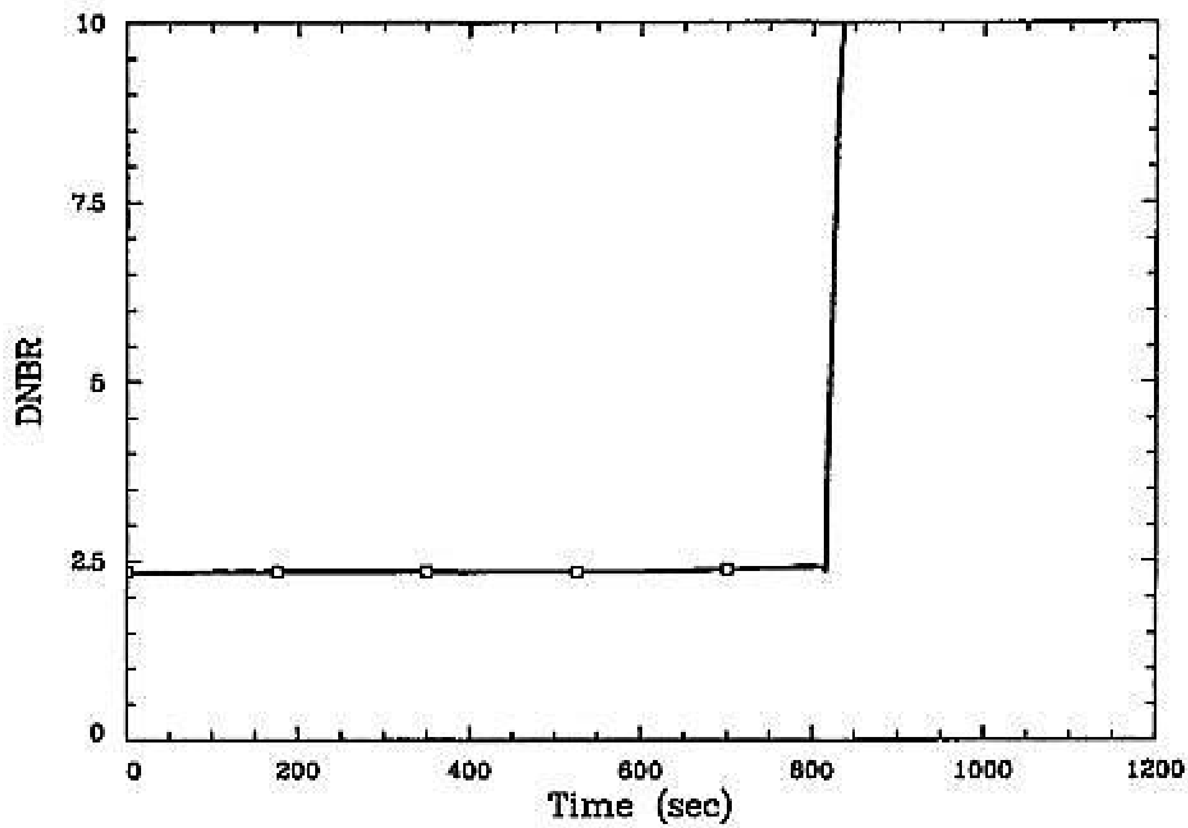


FIGURE 15.5.1-14

SAFETY INJECTION FLOW RATE (EOC MDNBR CASE)

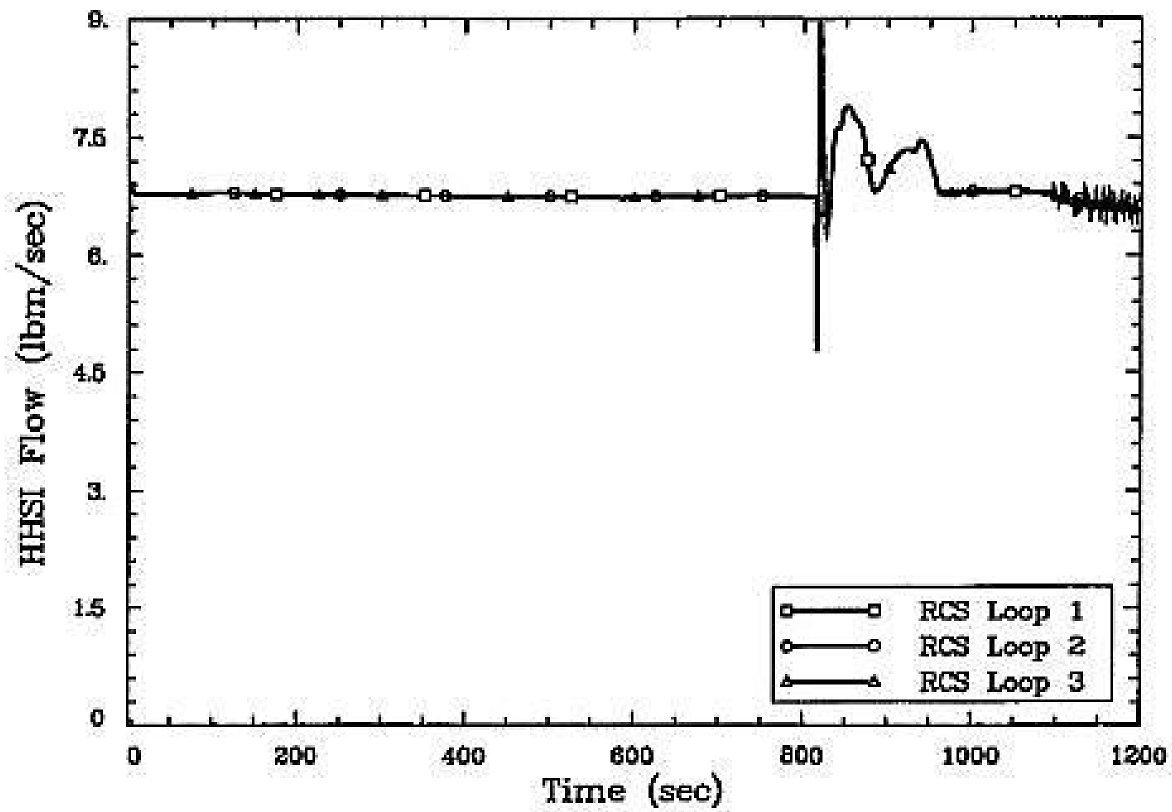


FIGURE 15.5.1-15

PRESSURIZER LIQUID LEVEL (PRESSURIZER OVERFILL CASE)

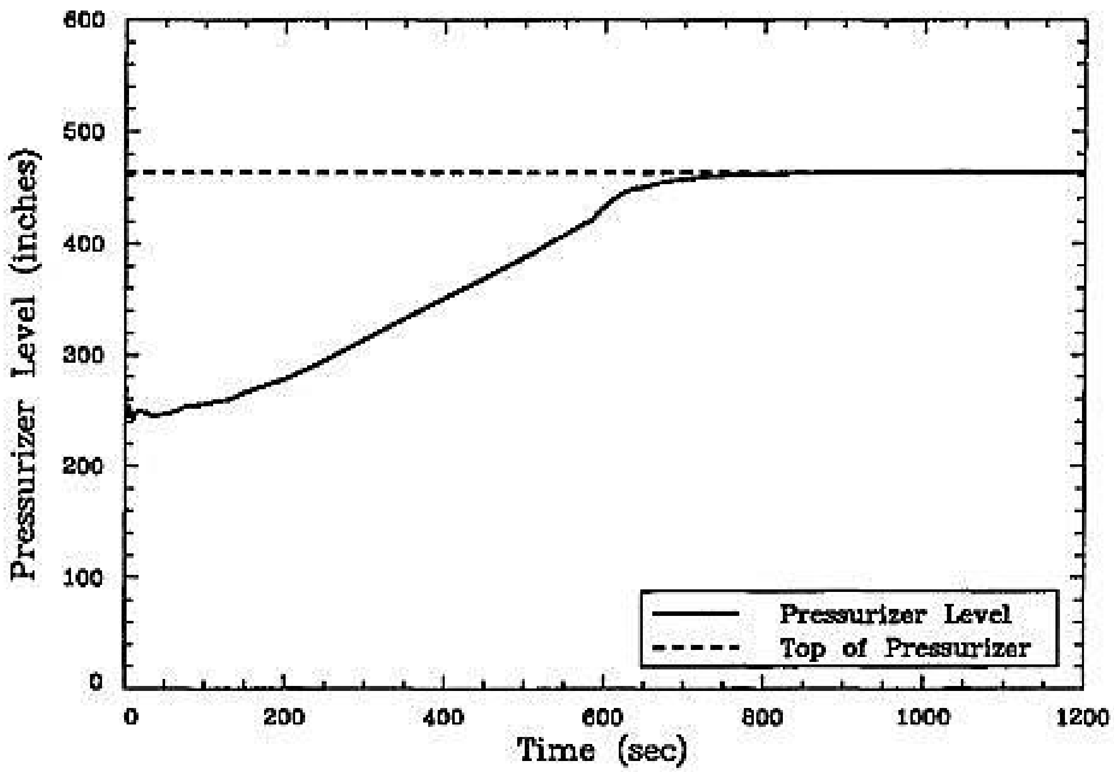


FIGURE 15.5.1-16

REACTOR COOLANT SYSTEM TEMPERATURES
(PRESSURIZER OVERFILL CASE)

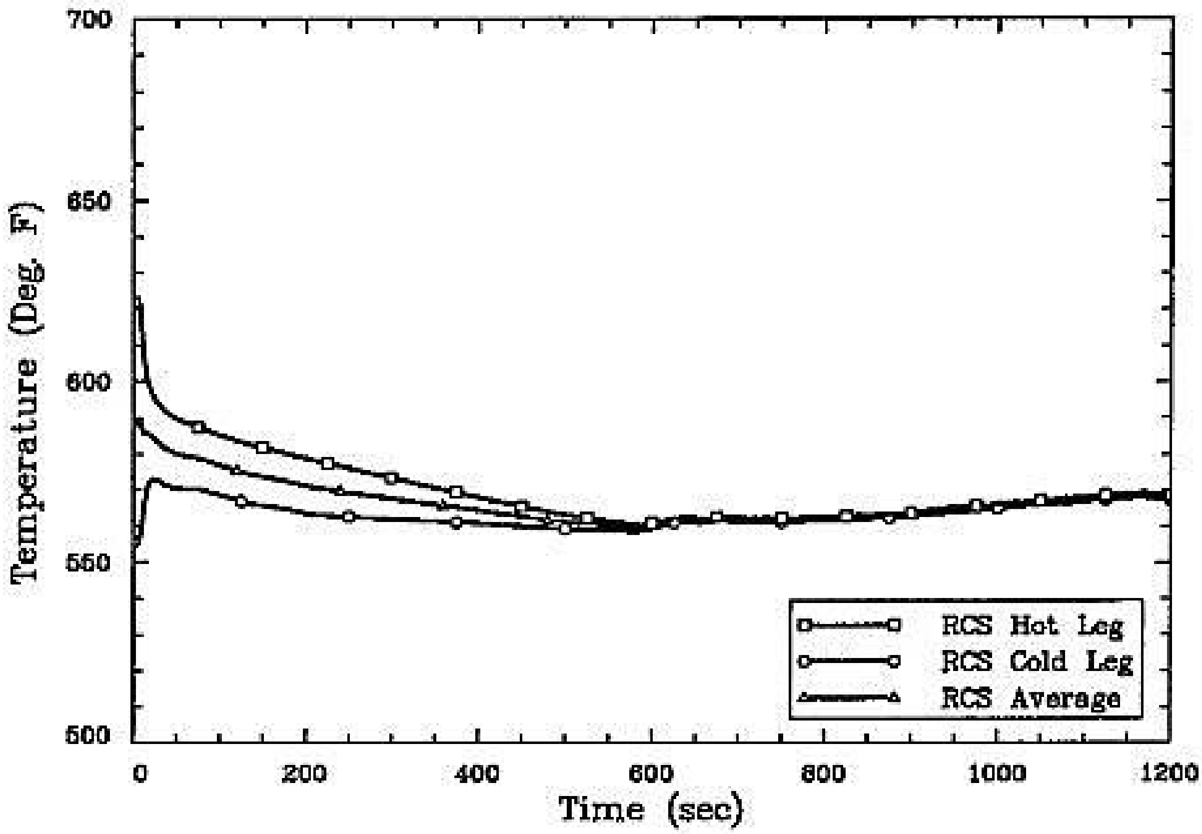


FIGURE 15.5.1-17

PRESSURIZER PRESSURE (PRESSURIZER OVERFILL CASE)

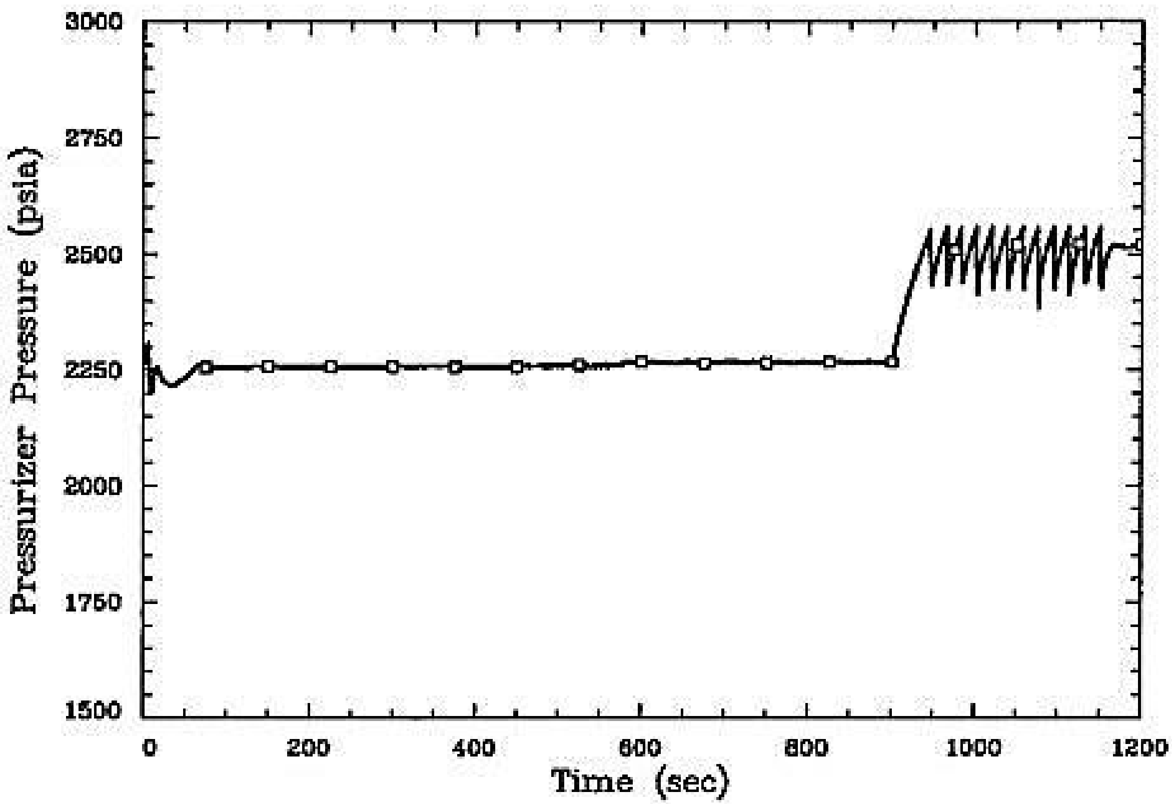


FIGURE 15.5.1-18

SAFETY INJECTION FLOW RATE (PRESSURIZER OVERFILL CASE)

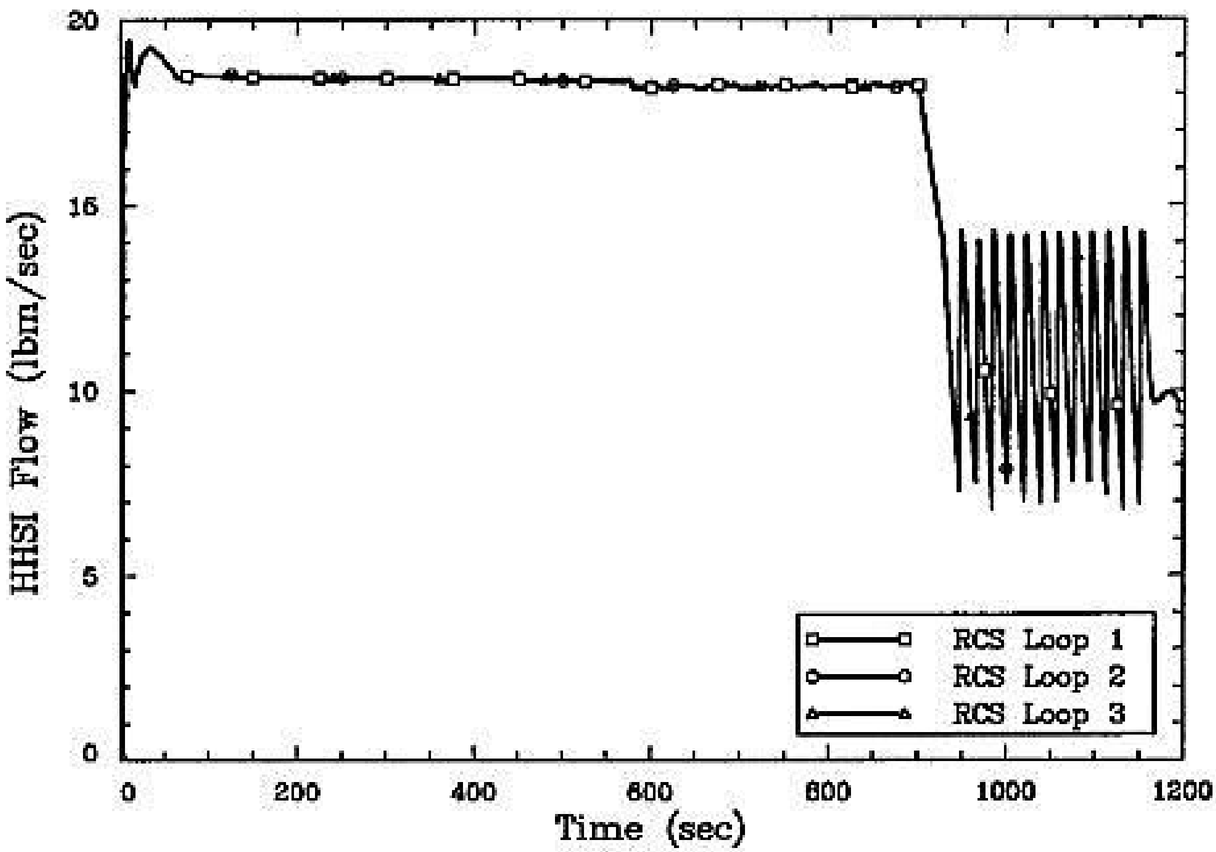


FIGURE 15.5.1-19

REACTOR POWER (PRESSURIZER OVERFILL CASE)

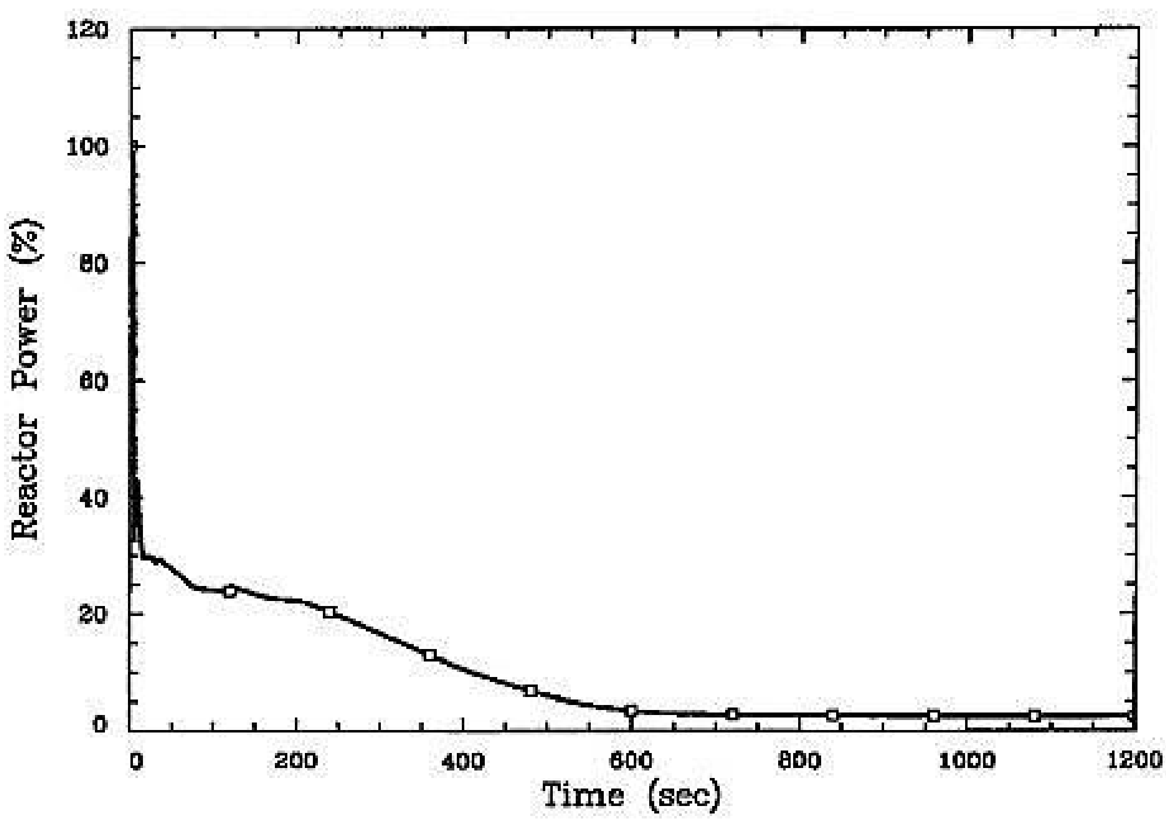


FIGURE 15.5.1-20

PRESSURIZER SAFETY RELIEF VALVE INLET TEMPERATURE
(PRESSURIZER OVERFILL CASE)

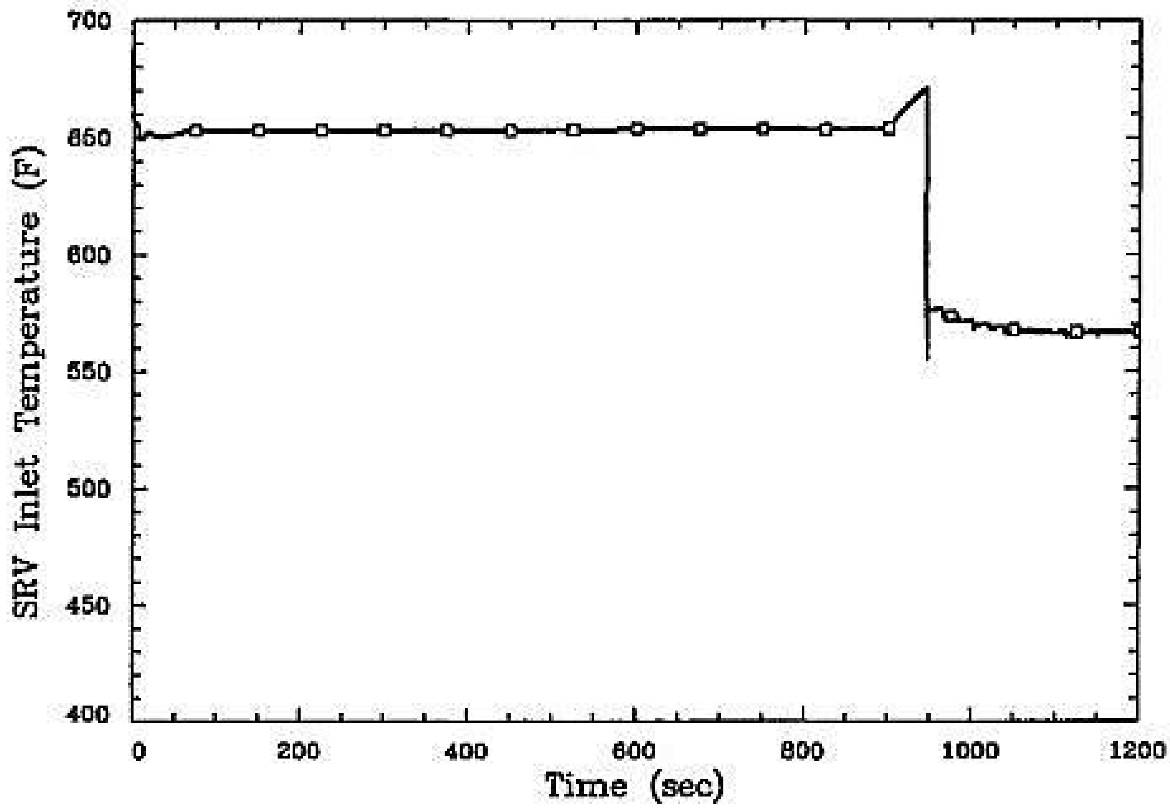


FIGURE 15.5.1-21

PRESSURIZER SAFETY RELIEF VALVE INLET PRESSURE
(PRESSURIZER OVERFILL CASE)

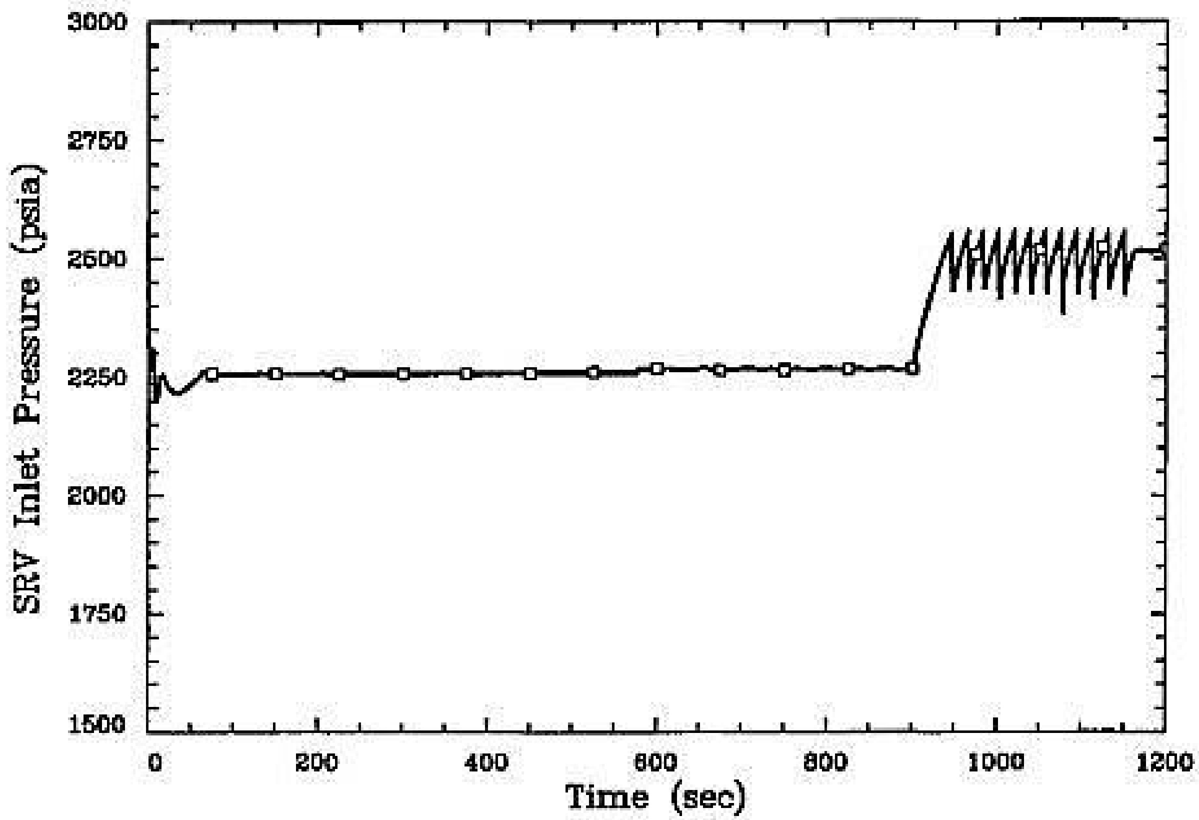


FIGURE 15.6.1-1

REACTOR POWER FOR INADVERTENT OPENING OF A PRESSURIZER PRESSURE SAFETY OR POWER OPERATED RELIEF VALVE

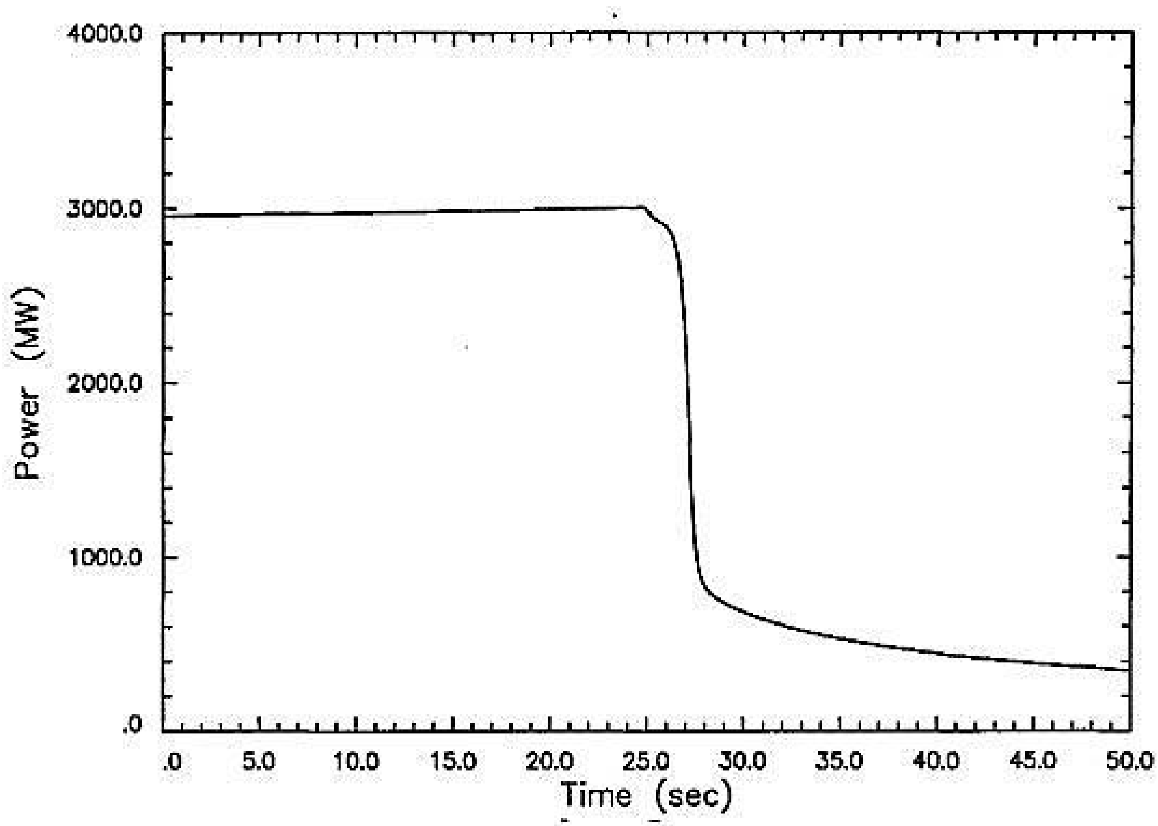


FIGURE 15.6.1-2

CORE AVERAGE HEAT FLUX FOR INADVERTENT OPENING OF A PRESSURIZER PRESSURE SAFETY OR POWER OPERATED RELIEF VALVE

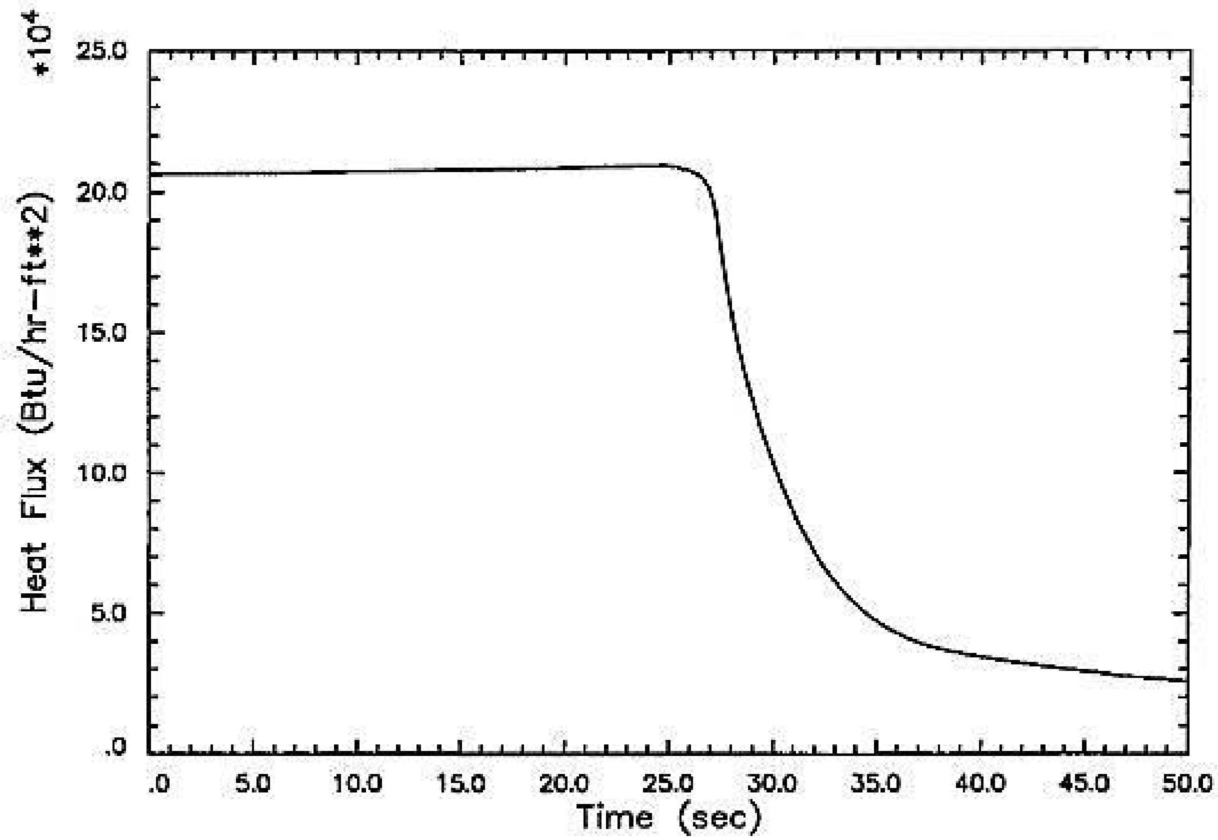


FIGURE 15.6.1-3

PRESSURIZER PRESSURE FOR INADVERTENT OPENING OF A PRESSURIZER PRESSURE SAFETY OR POWER OPERATED RELIEF VALVE

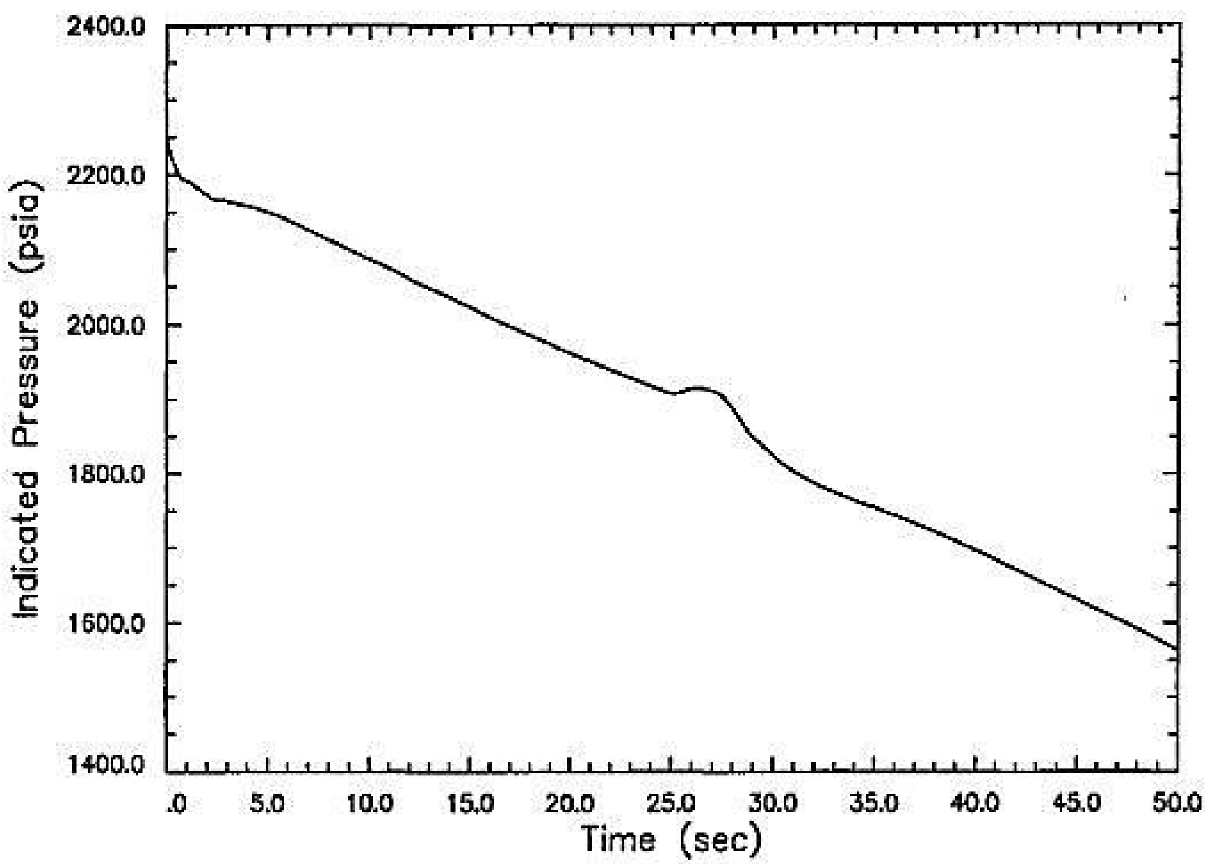


FIGURE 15.6.1-4

PRESSURIZER LEVEL FOR INADVERTENT OPENING OF A PRESSURIZER PRESSURE SAFETY OR POWER OPERATED RELIEF VALVE

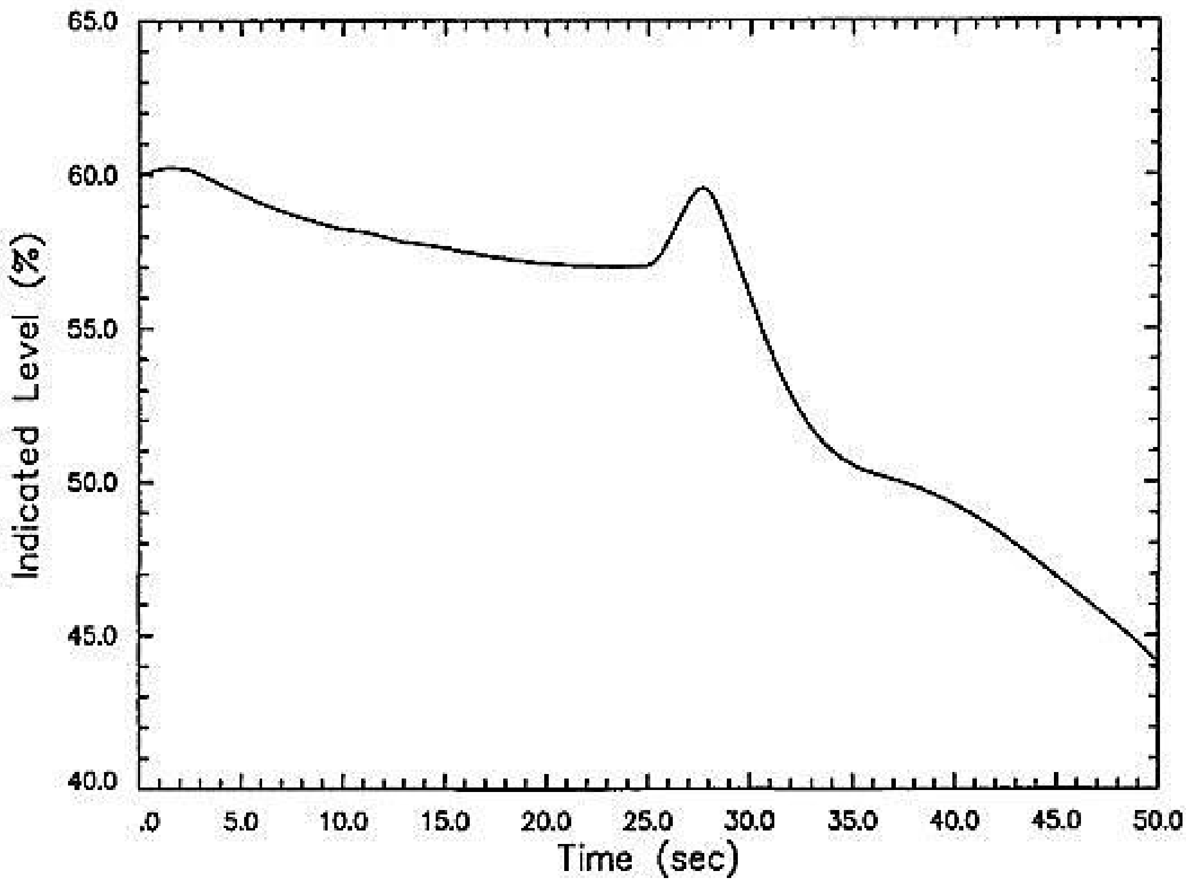


FIGURE 15.6.1-5

REACTOR COOLANT SYSTEM MASS FLOW RATE FOR INADVERTENT OPENING OF A PRESSURIZER PRESSURE SAFETY OR POWER OPERATED RELIEF VALVE

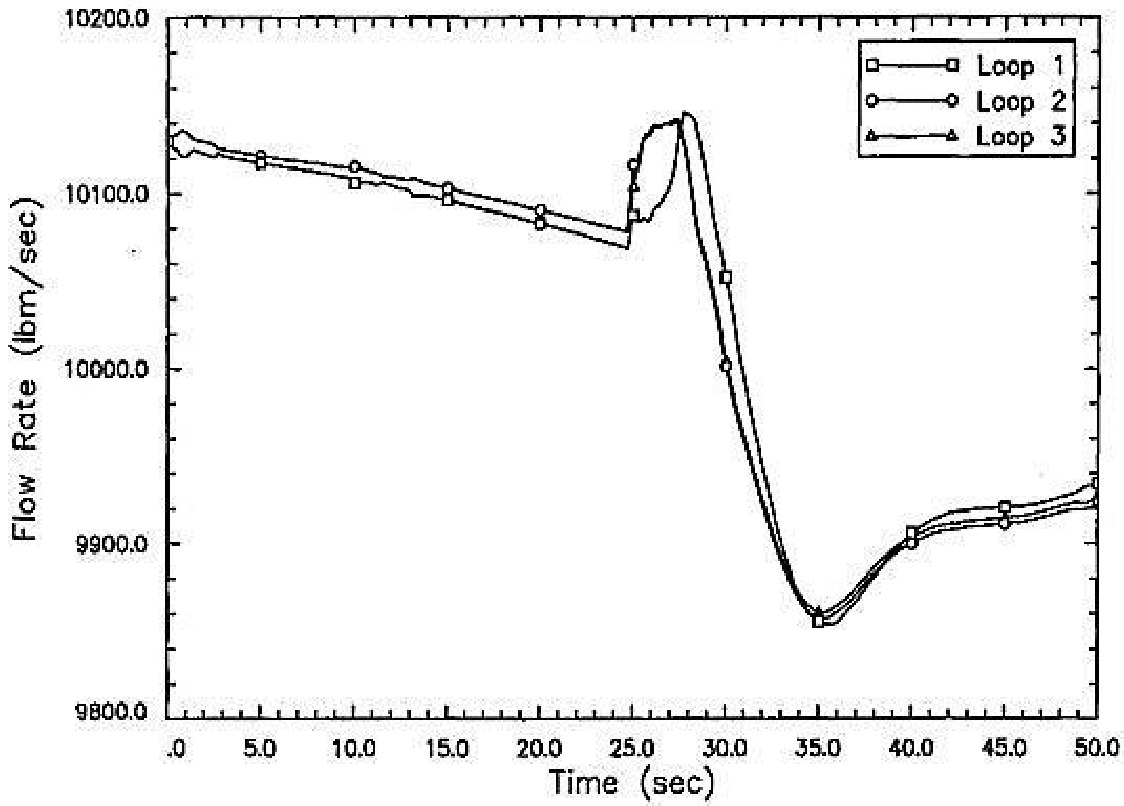


FIGURE 15.6.1-6

REACTOR COOLANT SYSTEM TEMPERATURE FOR INADVERTENT OPENING OF A PRESSURIZER PRESSURE SAFETY OR POWER OPERATED RELIEF VALVE

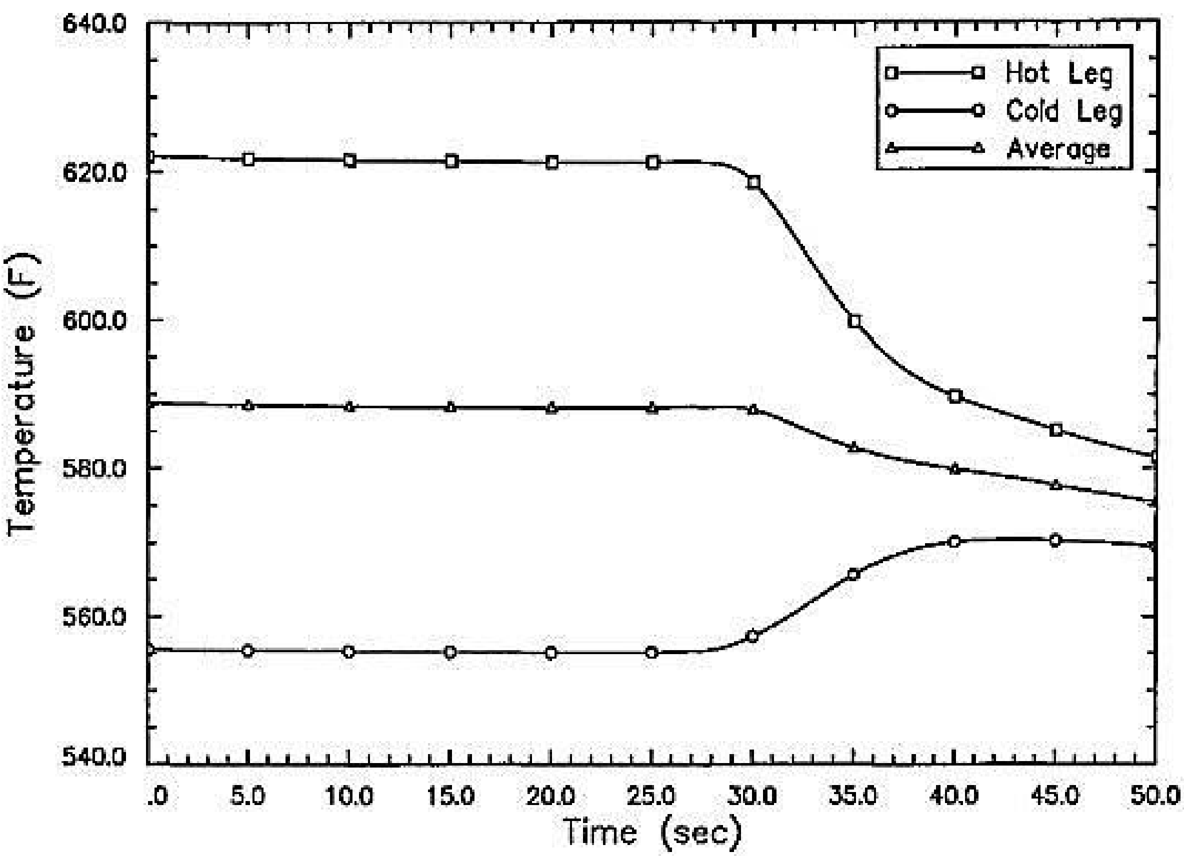


FIGURE 15.6.1-7

TOTAL CORE REACTIVITY FOR INADVERTENT OPENING OF A PRESSURIZER PRESSURE SAFETY OR POWER OPERATED RELIEF VALVE

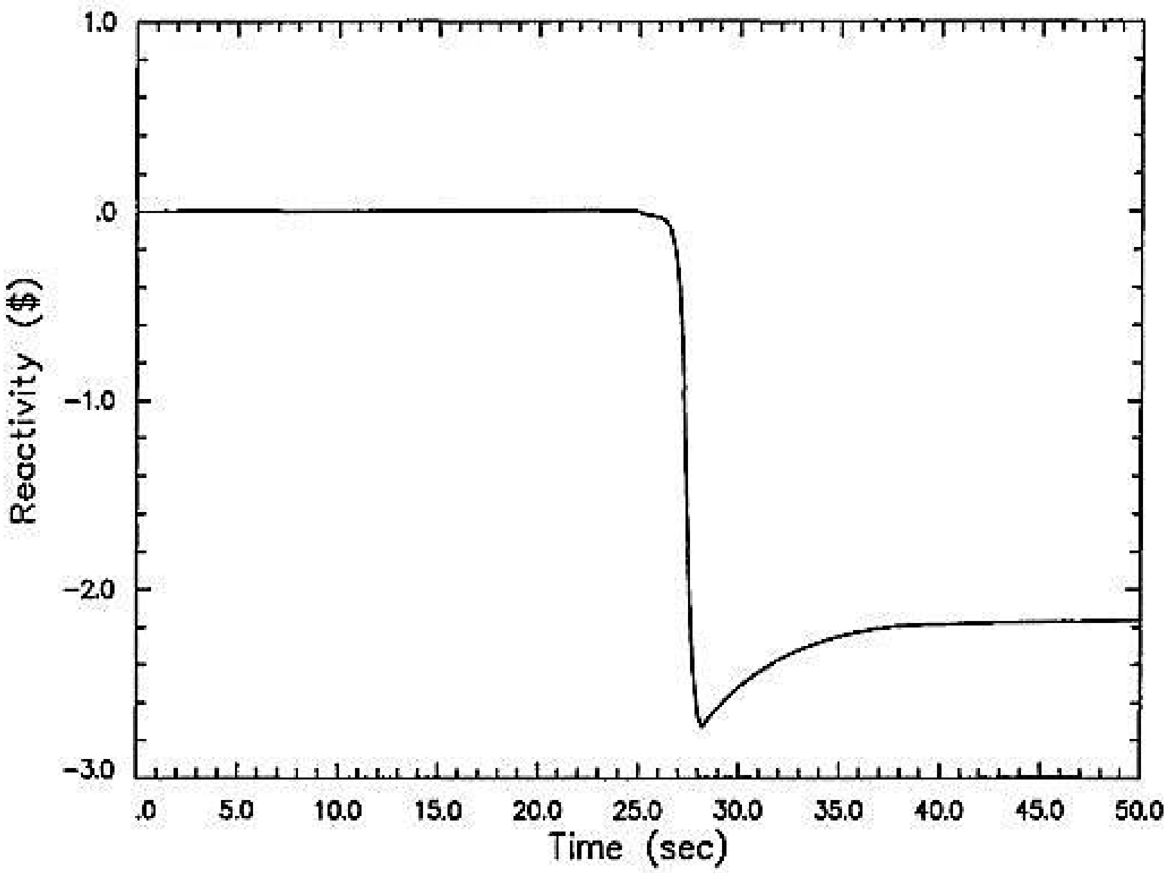
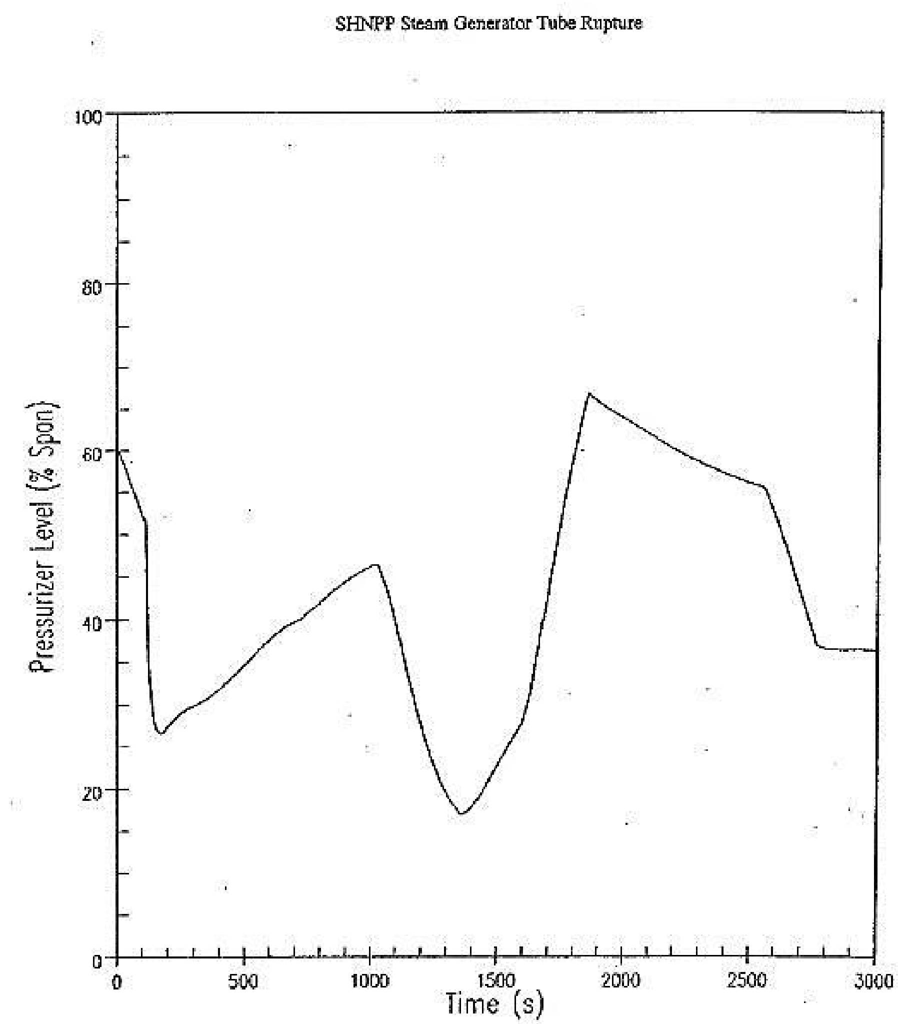


FIGURE 15.6.3-1

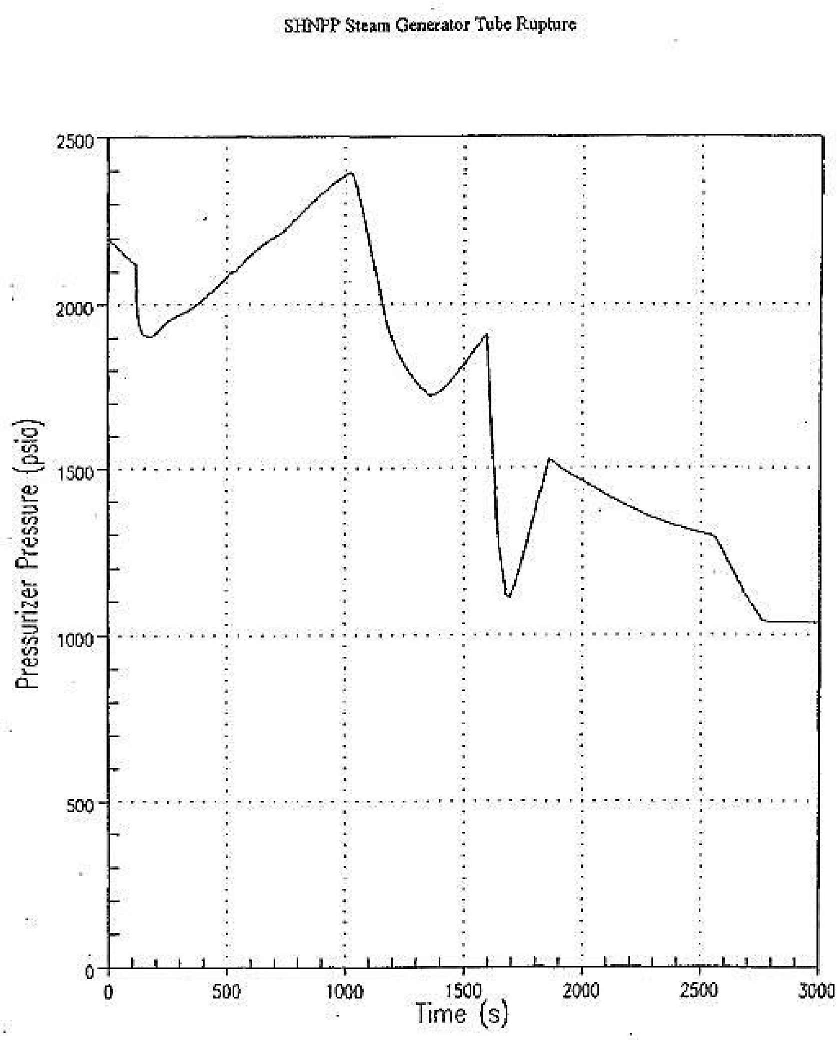
PRESSURIZER LEVEL – MARGIN TO OVERFILL ANALYSIS



The HNP MTO analysis has been amended to account for a TDAFW pump speed controller failure that results in increased AFW delivery over the first 528 seconds. The final MTO is unchanged (66 cu ft) but the level history in this figure applies to the original analysis.

FIGURE 15.6.3-2

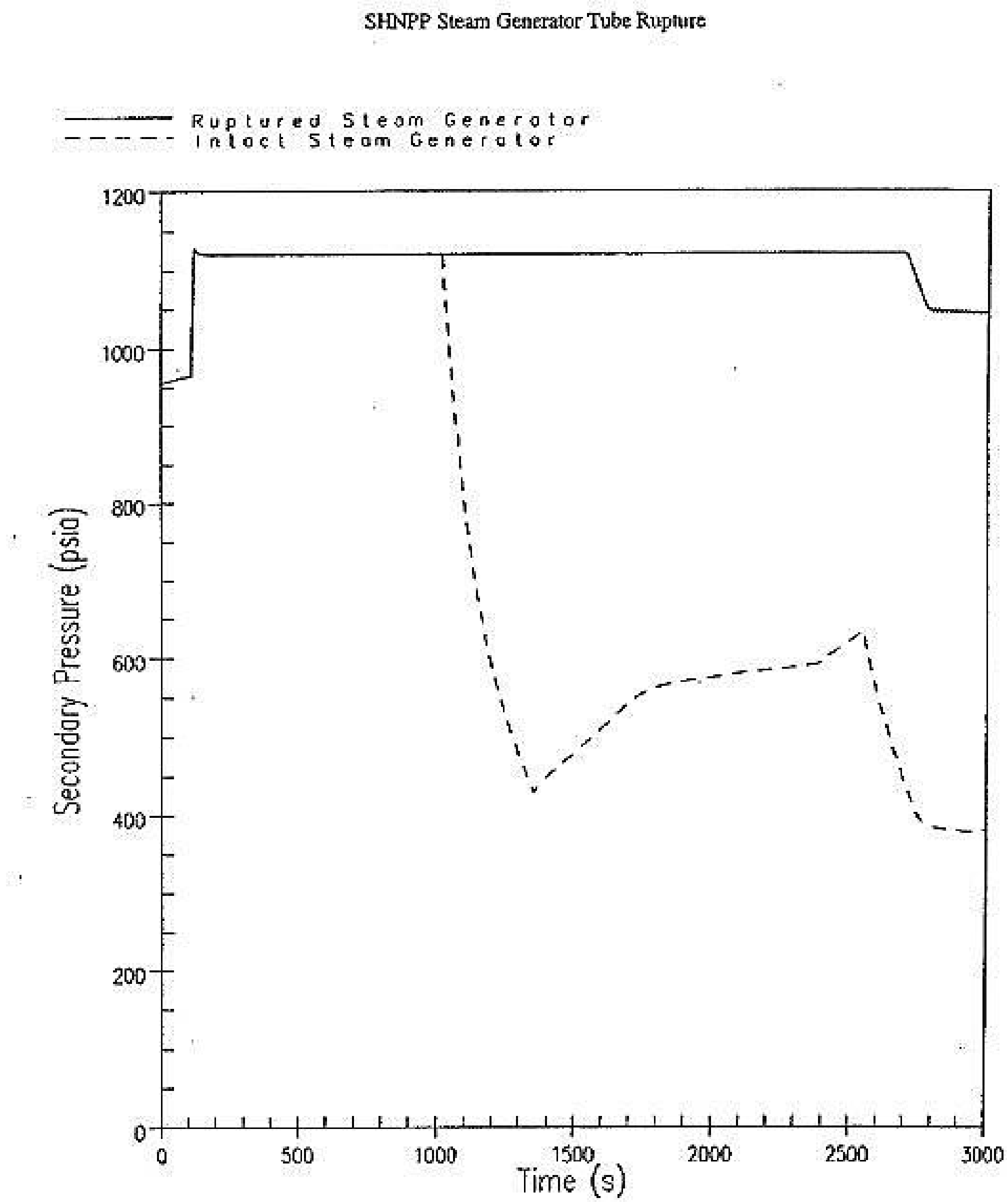
PRESSURIZER PRESSURE – MARGIN TO OVERFILL ANALYSIS



The HNP MTO analysis has been amended to account for a TDAFW pump speed controller failure that results in increased AFW delivery over the first 528 seconds. The final MTO is unchanged (66 cu ft) but the pressure history in this figure applies to the original analysis.

FIGURE 15.6.3-3

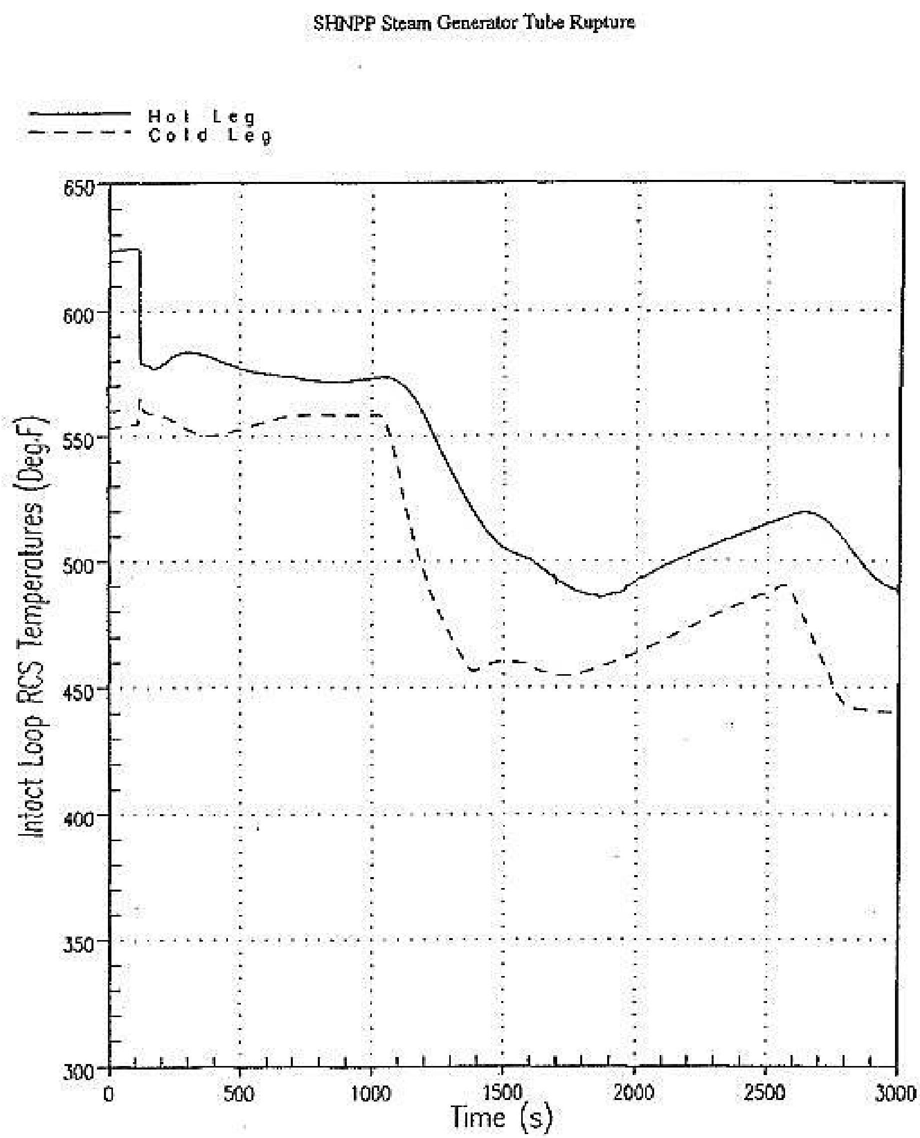
SECONDARY PRESSURE – MARGIN TO OVERFILL ANALYSIS



The HNP MTO analysis has been amended to account for a TDAFW pump speed controller failure that results in increased AFW delivery over the first 528 seconds. The final MTO is unchanged (66 cu ft) but the pressure history in this figure applies to the original analysis.

FIGURE 15.6.3-4

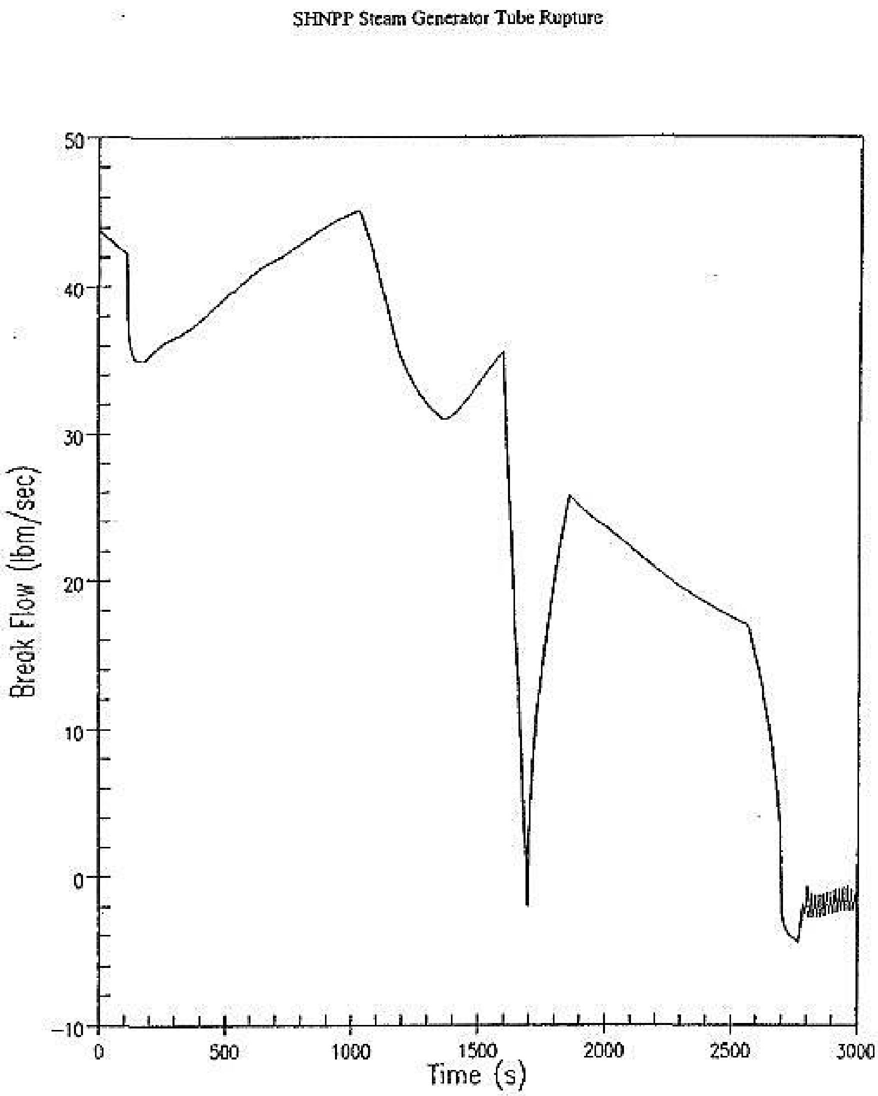
INTACT LOOP HOT AND COLD LEG TEMPERATURES – MARGIN TO OVERFILL ANALYSIS



The HNP MTO analysis has been amended to account for a TDAFW pump speed controller failure that results in increased AFW delivery over the first 528 seconds. The final MTO is unchanged (66 cu ft) but the temperature histories in this figure applies to the original analysis.

FIGURE 15.6.3-5

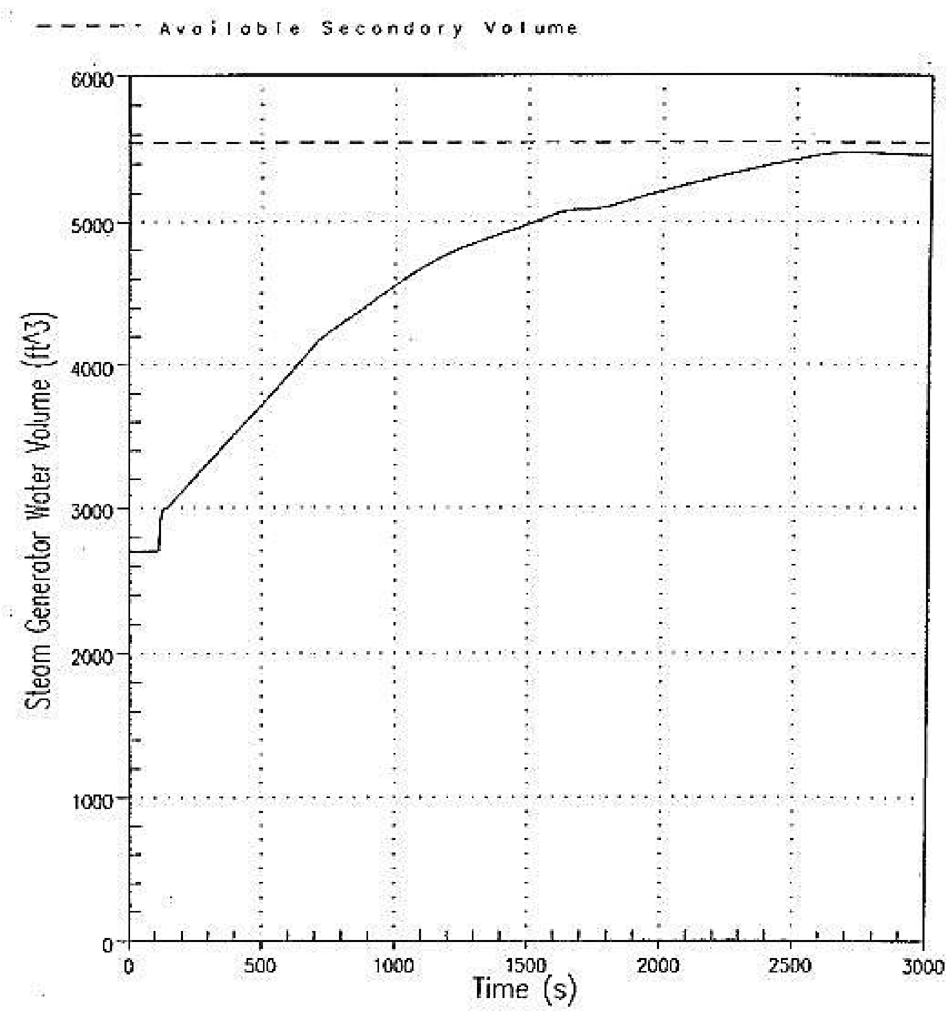
PRIMARY TO SECONDARY BREAK FLOW MARGIN TO OVERFILL ANALYSIS



The HNP MTO analysis has been amended to account for a TDAFW pump speed controller failure that results in increased AFW delivery over the first 528 seconds. The final MTO is unchanged (66 cu ft) but the break flow history in this figure applies to the original analysis.

FIGURE 15.6.3-6

RUPTURED SG WATER VOLUME MARGIN TO OVERFILL ANALYSIS



The HNP MTO analysis has been amended to account for a TDAFW pump speed controller failure that results in increased AFW delivery over the first 528 seconds. The final MTO remains as shown (66 cu ft) but the volume history in this figure applies to the original analysis.

FIGURE 15.6.3-7

PRESSURIZER LEVEL – OFFSITE RADIATION DOSE ANALYSIS

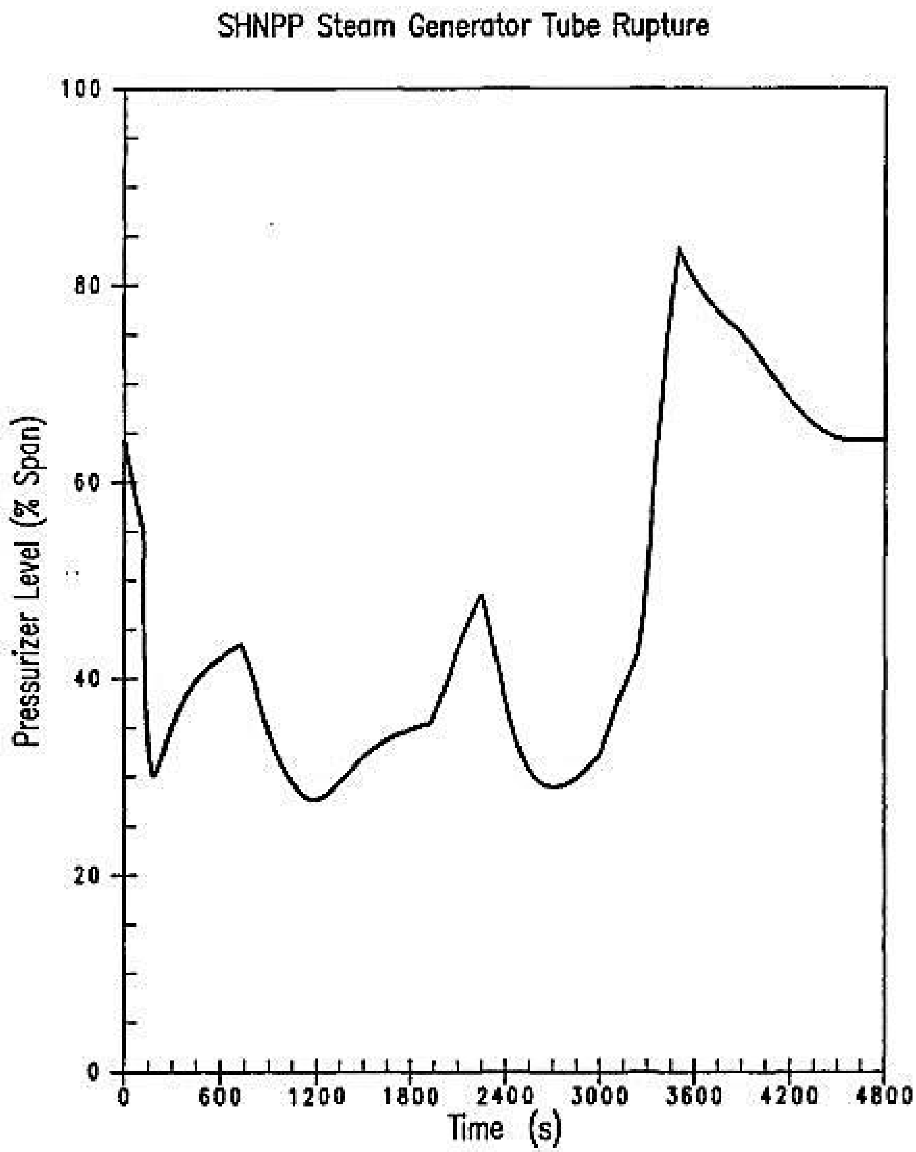


FIGURE 15.6.3-8

PRESSURIZER PRESSURE – OFFSITE RADIATION DOSE ANALYSIS

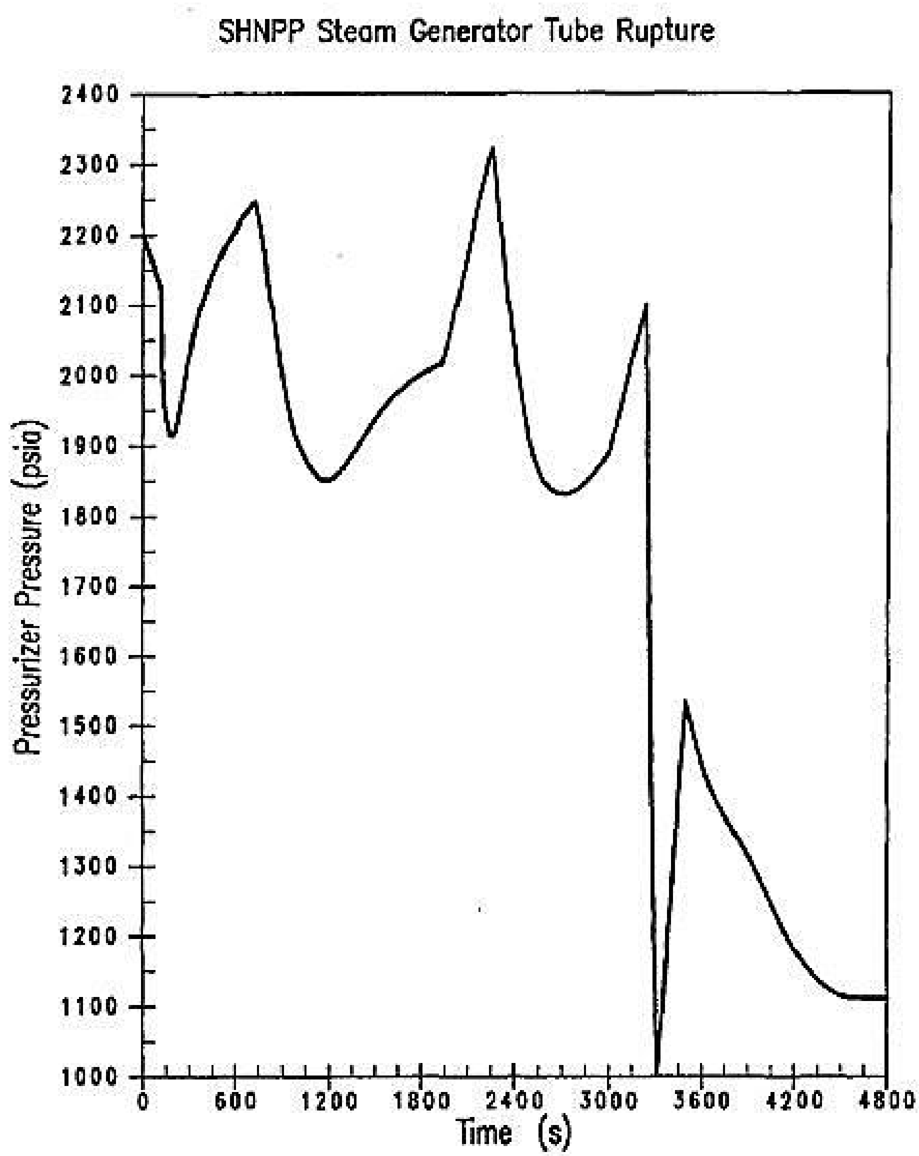


FIGURE 15.6.3-9

SECONDARY PRESSURE – OFFSITE RADIATION DOSE ANALYSIS

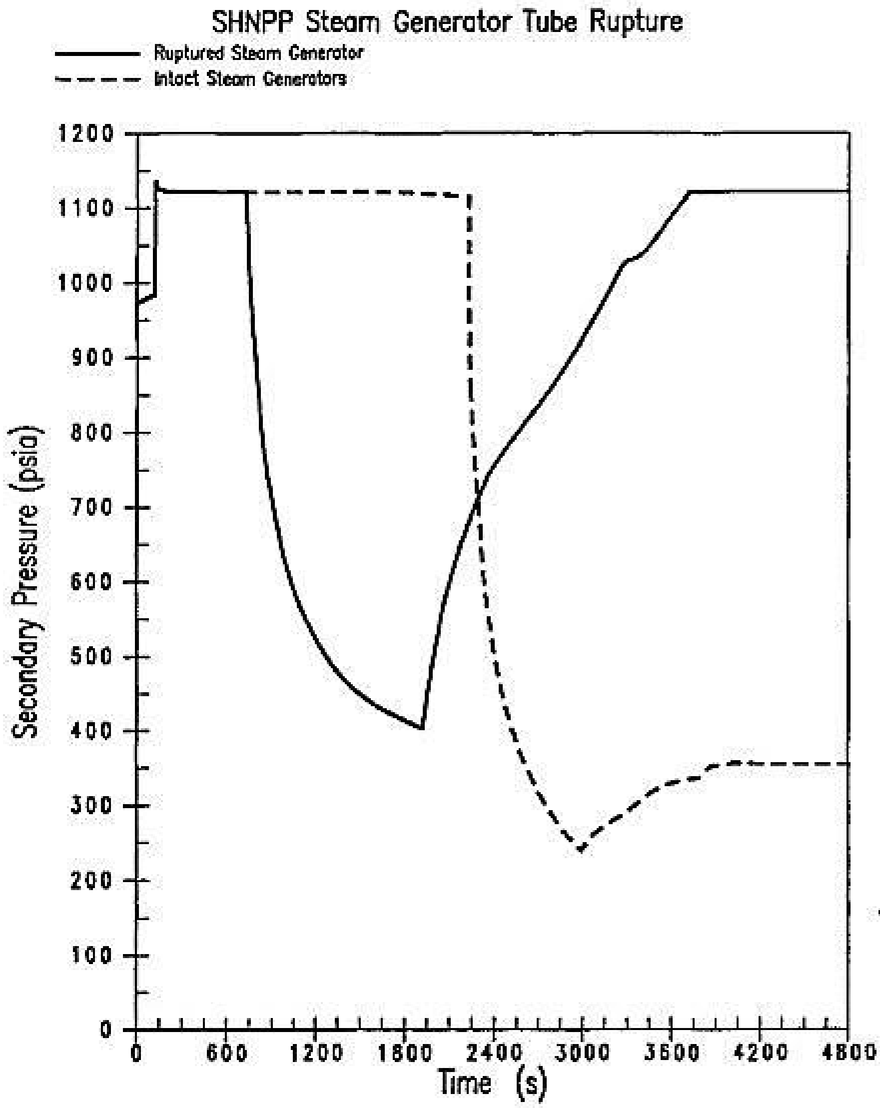


FIGURE 15.6.3-10

INTACT LOOP HOT & COLD LEG TEMPERATURES – OFFSITE RADIATION DOSE ANALYSIS

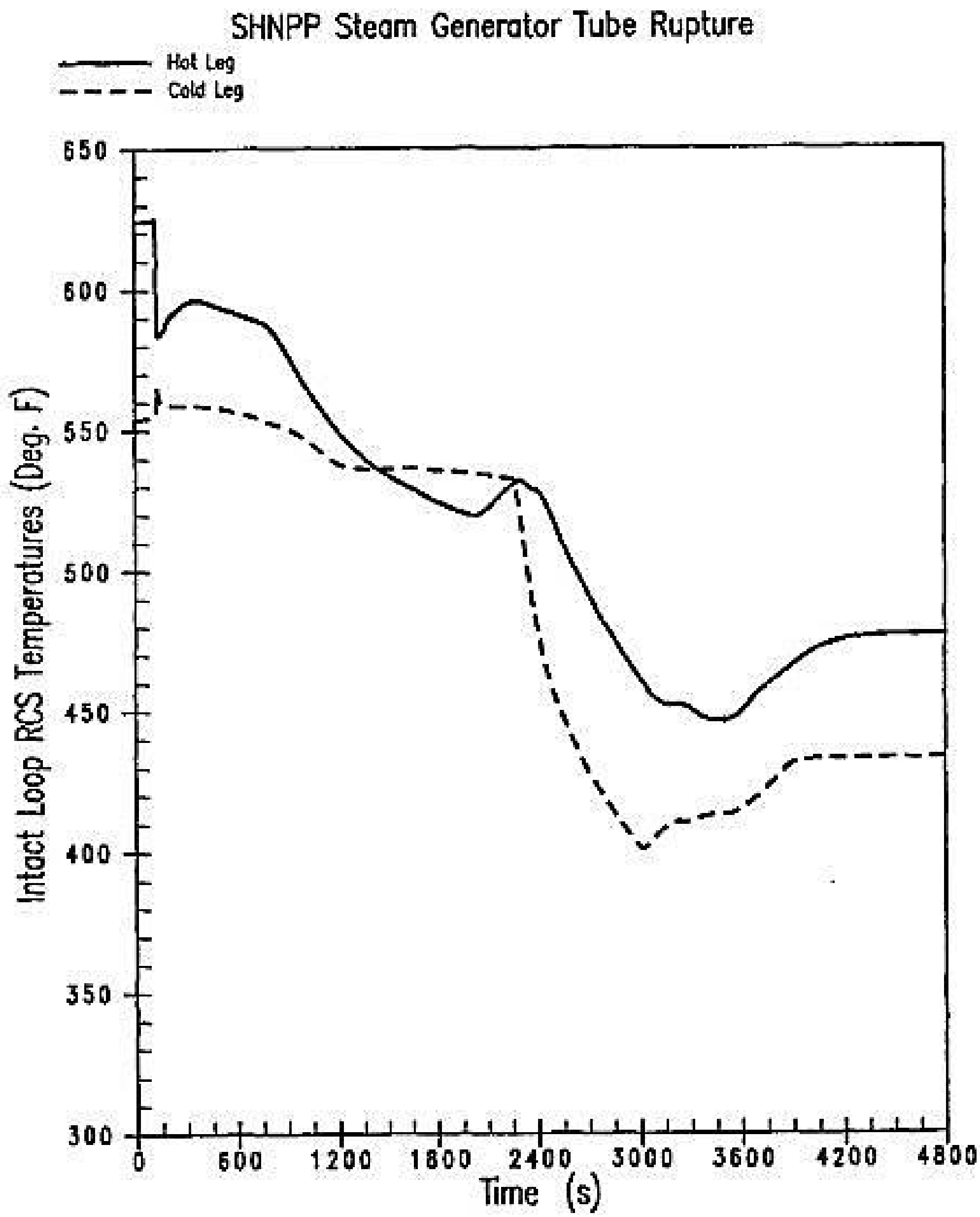


FIGURE 15.6.3-11

RUPTURED LOOP HOT AND COLD LEG TEMPERATURES – OFFSITE RADIATION DOSE ANALYSIS

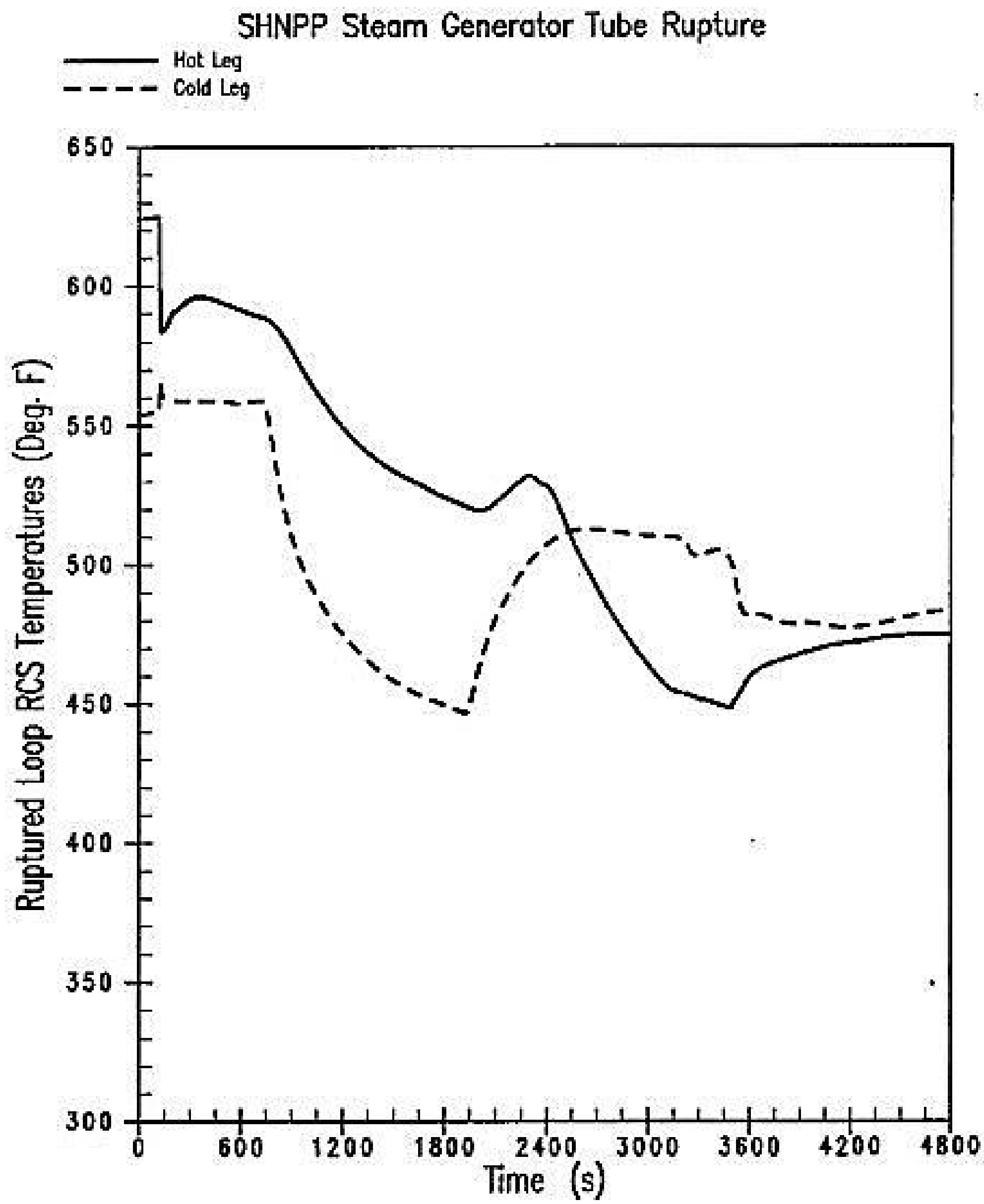


FIGURE 15.6.3-12

DIFFERENTIAL PRESSURE BETWEEN RCS AND RUPTURED SG – OFFSITE RADIATION DOSE ANALYSIS

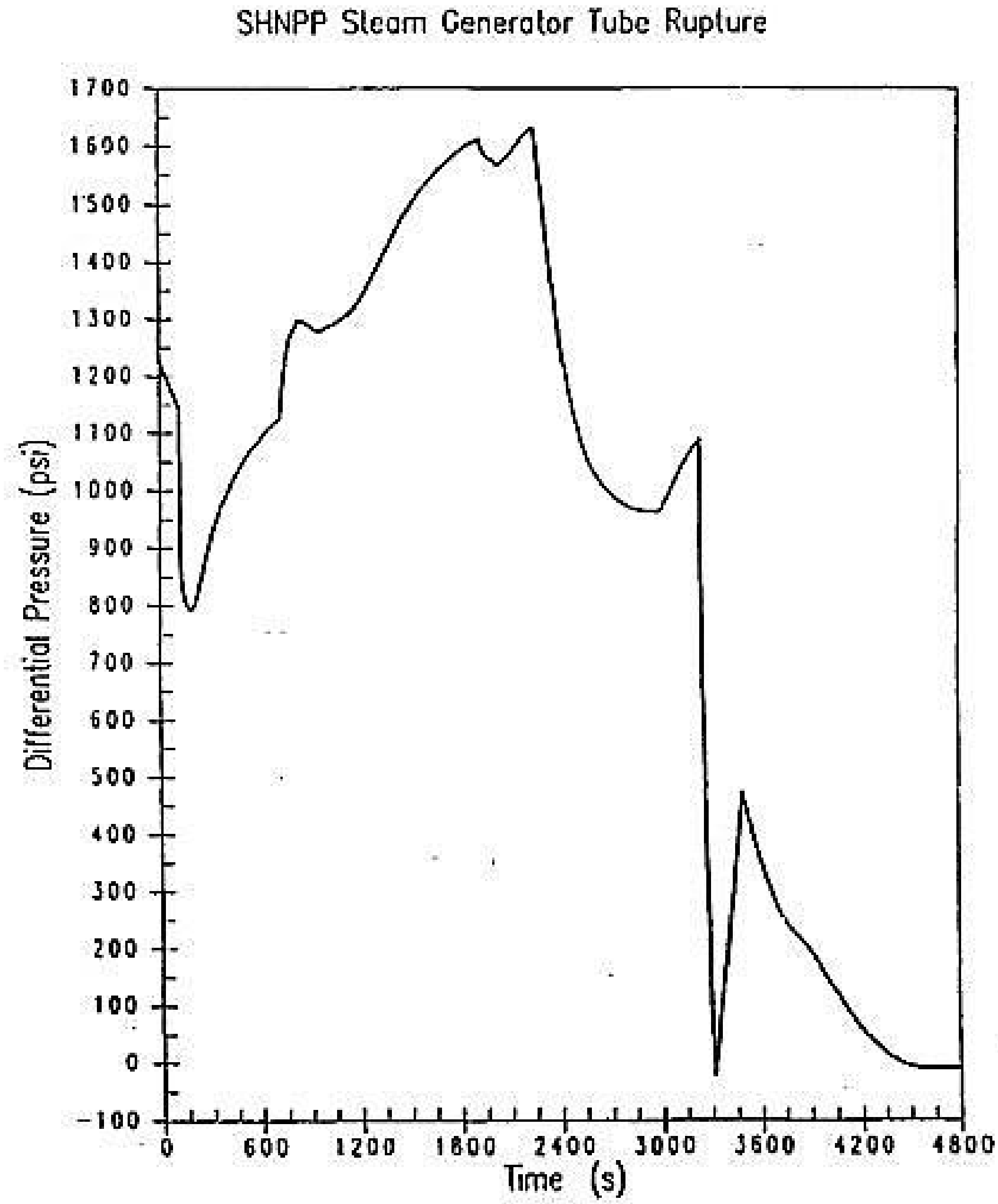


FIGURE 15.6.3-13

PRIMARY TO SECONDARY BREAK FLOW – OFFSITE RADIATION DOSE ANALYSIS

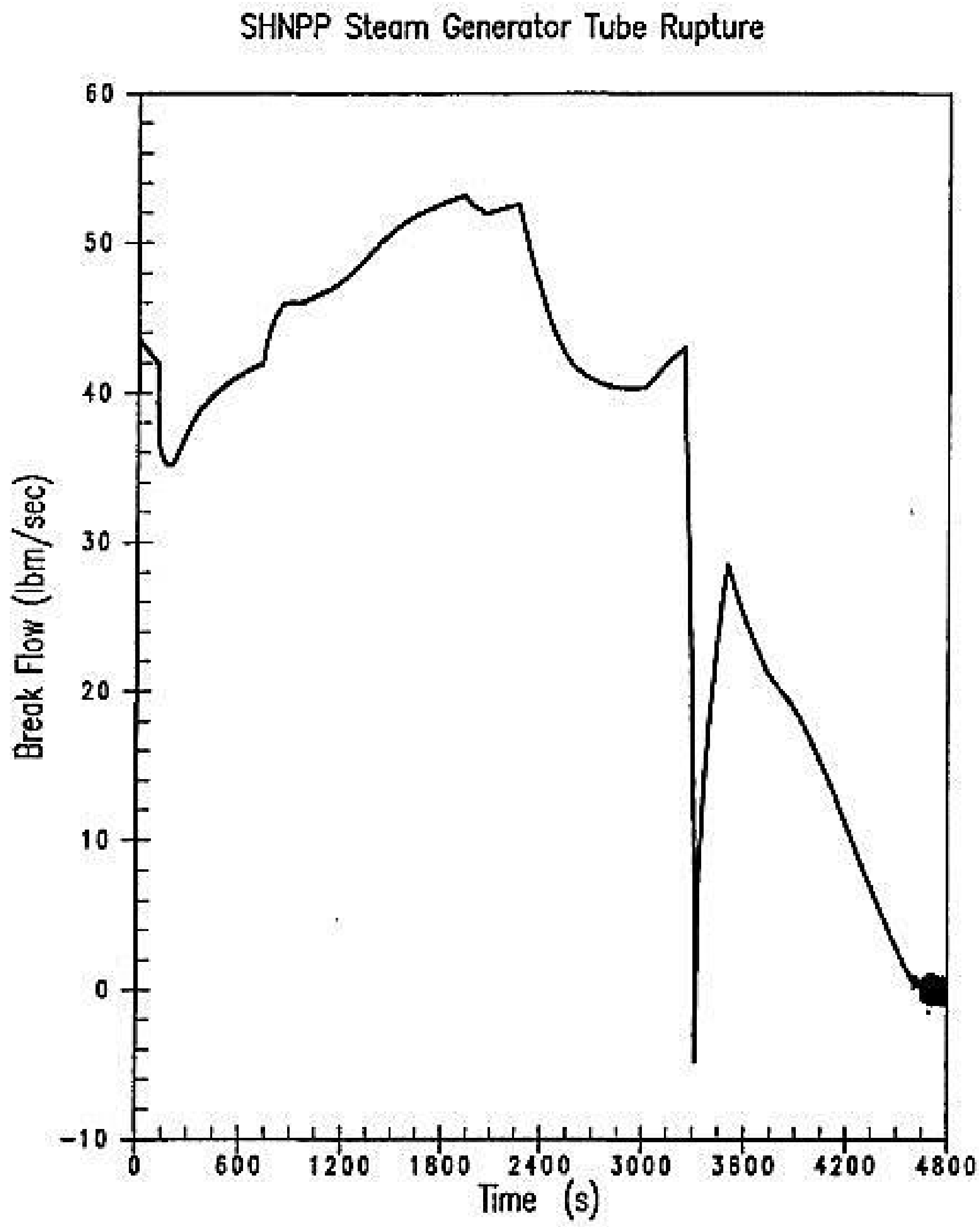


FIGURE 15.6.3-14

RUPTURED SG WATER VOLUME – OFFSITE RADIATION DOSE ANALYSIS

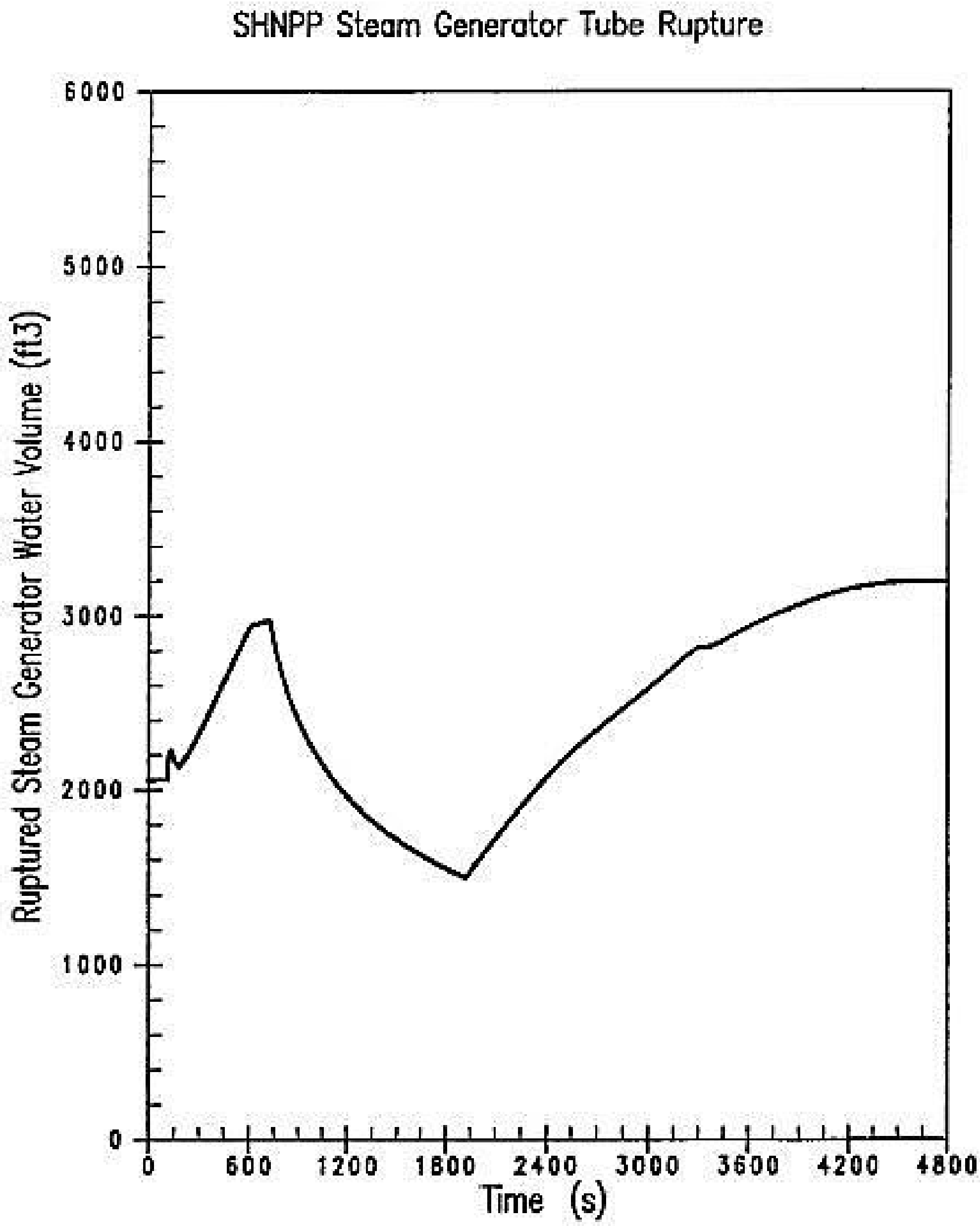


FIGURE 15.6.3-15

RUPTURED SG WATER MASS – OFFSITE RADIATION DOSE ANALYSIS

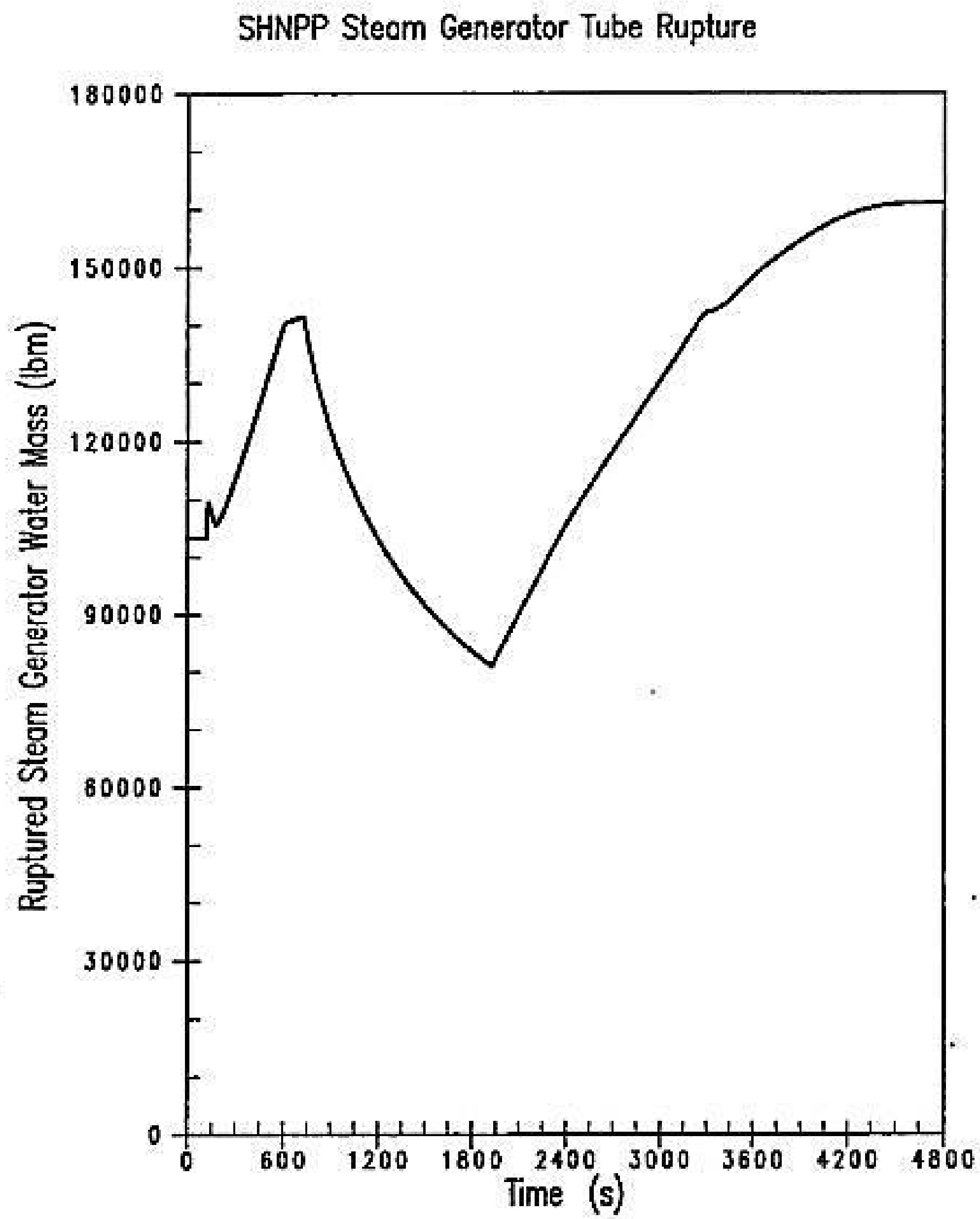


FIGURE 15.6.3-16

RUPTURED SG MASS RELEASE RATE TO THE ATMOSPHERE – OFFSITE RADIATION DOSE ANALYSIS

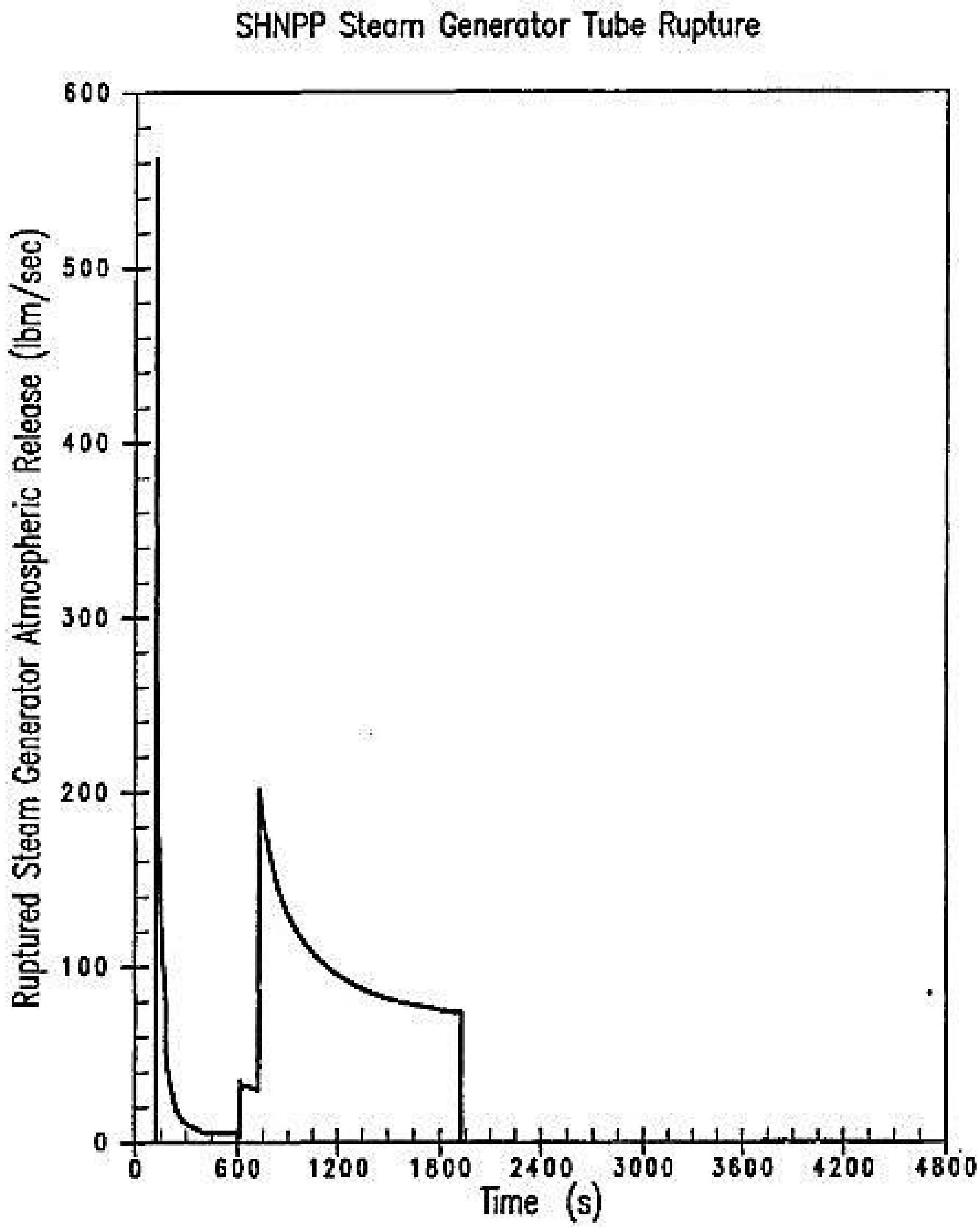


FIGURE 15.6.3-17

INTACT SGS MASS RELEASE RATE TO THE ATMOSPHERE – OFFSITE RADIATION DOSE ANALYSIS

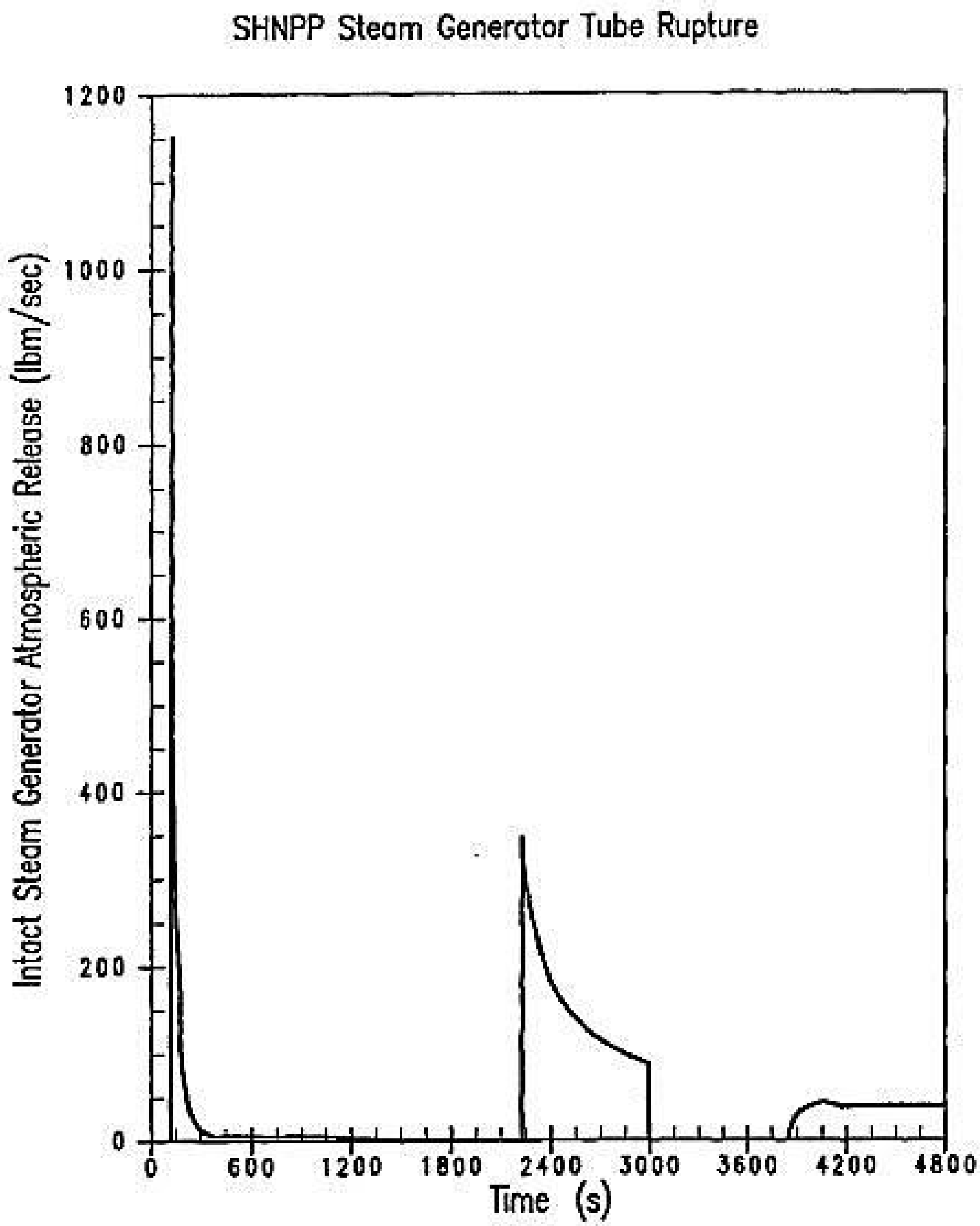


FIGURE 15.6.3-18

IODINE TRANSPORT MODEL – OFFSITE RADIATION DOSE ANALYSIS

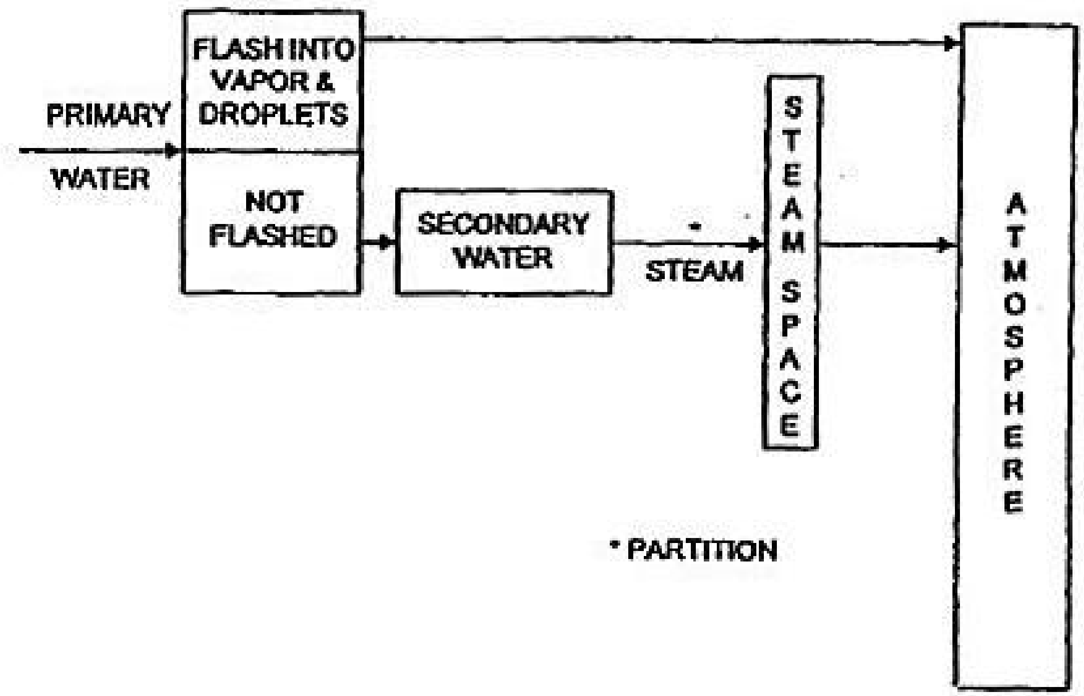


FIGURE 15.6.3-19

BREAK FLOW FLASHING FRACTION – OFFSITE RADIATION DOSE ANALYSIS

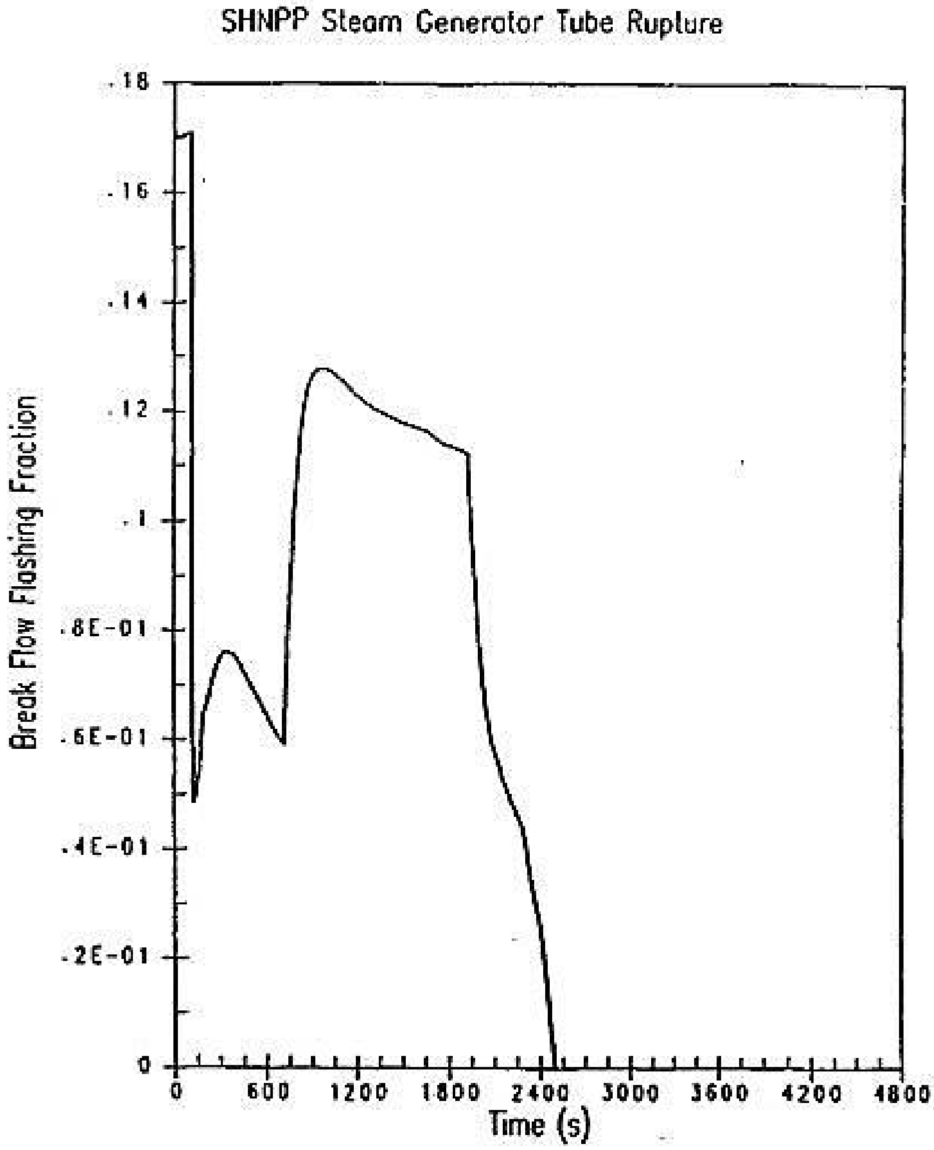


FIGURE 15.6.3-22

TOTAL FLASHED BREAK FLOW – SGTR OFFSITE RADIATION DOSE ANALYSIS

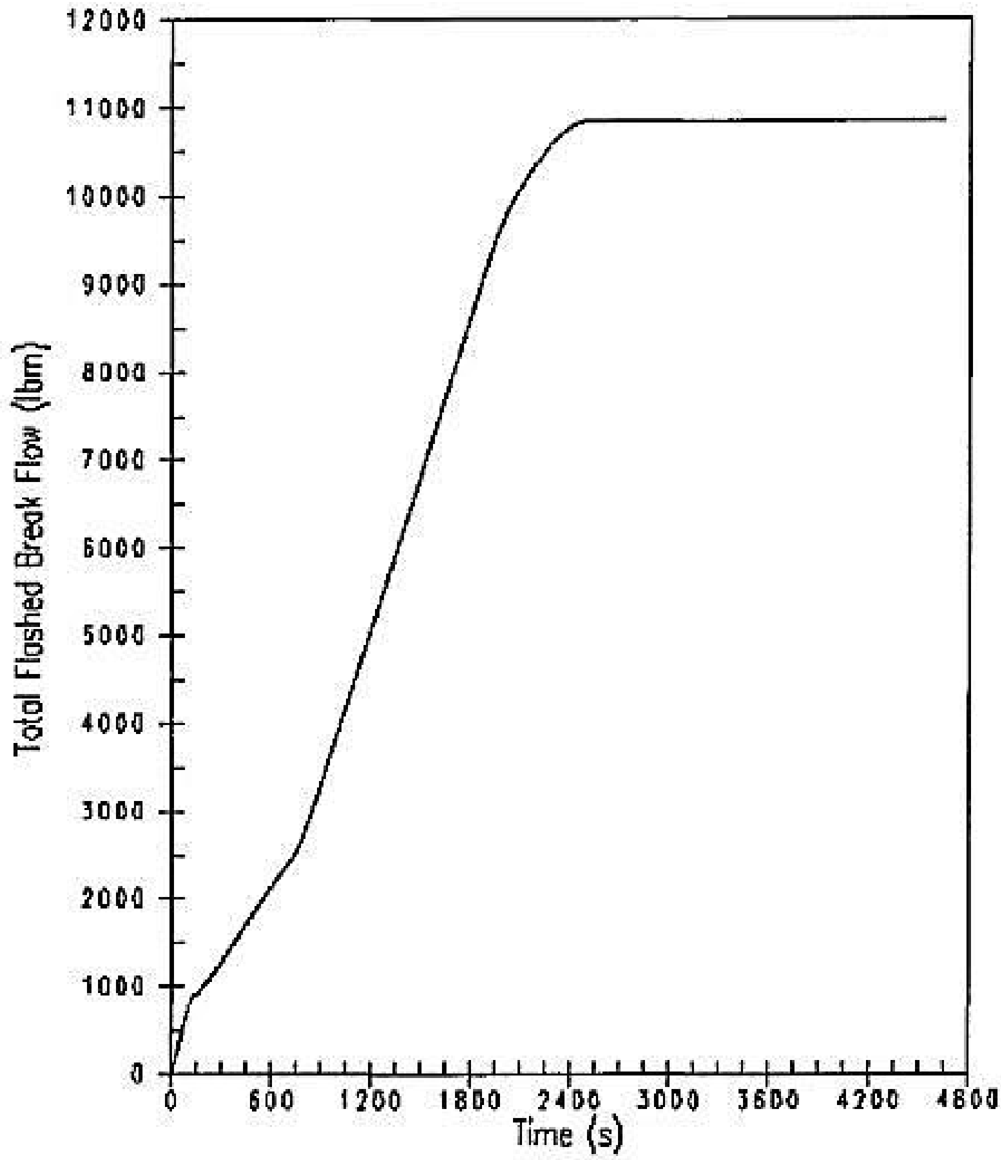


FIGURE 15.6.3-23

RUPTURED SG MASS RELEASE RATE TO THE ATMOSPHERE – SGTR OFFSITE RADIATION DOSE ANALYSIS

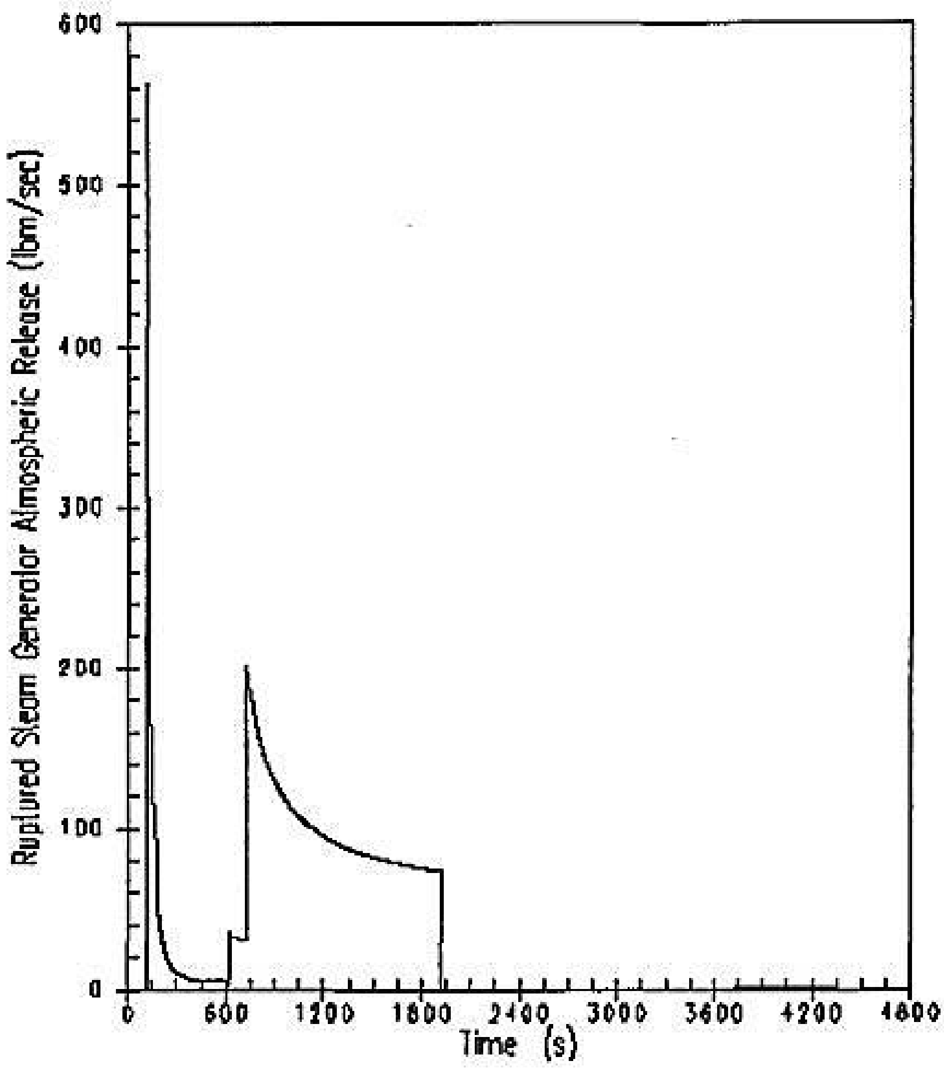


FIGURE 15.6.3-24

INTACT SG MASS RELEASE RATE TO THE ATMOSPHERE – SGTR OFFSITE RADIATION DOSE ANALYSIS

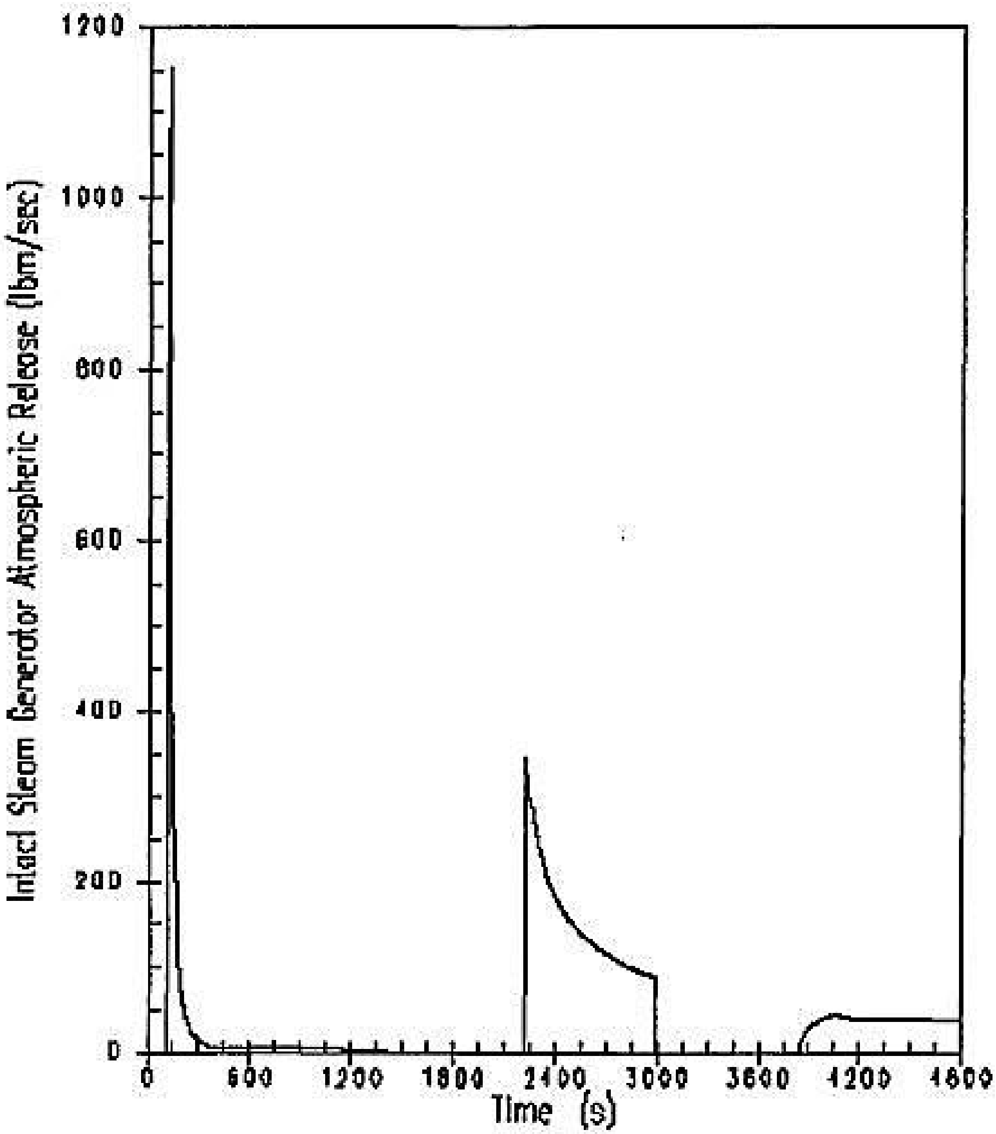


FIGURE 15.6.5-1

NORMALIZED AXIAL DEPENDENCE OF F_Q^T VERSUS ELEVATION

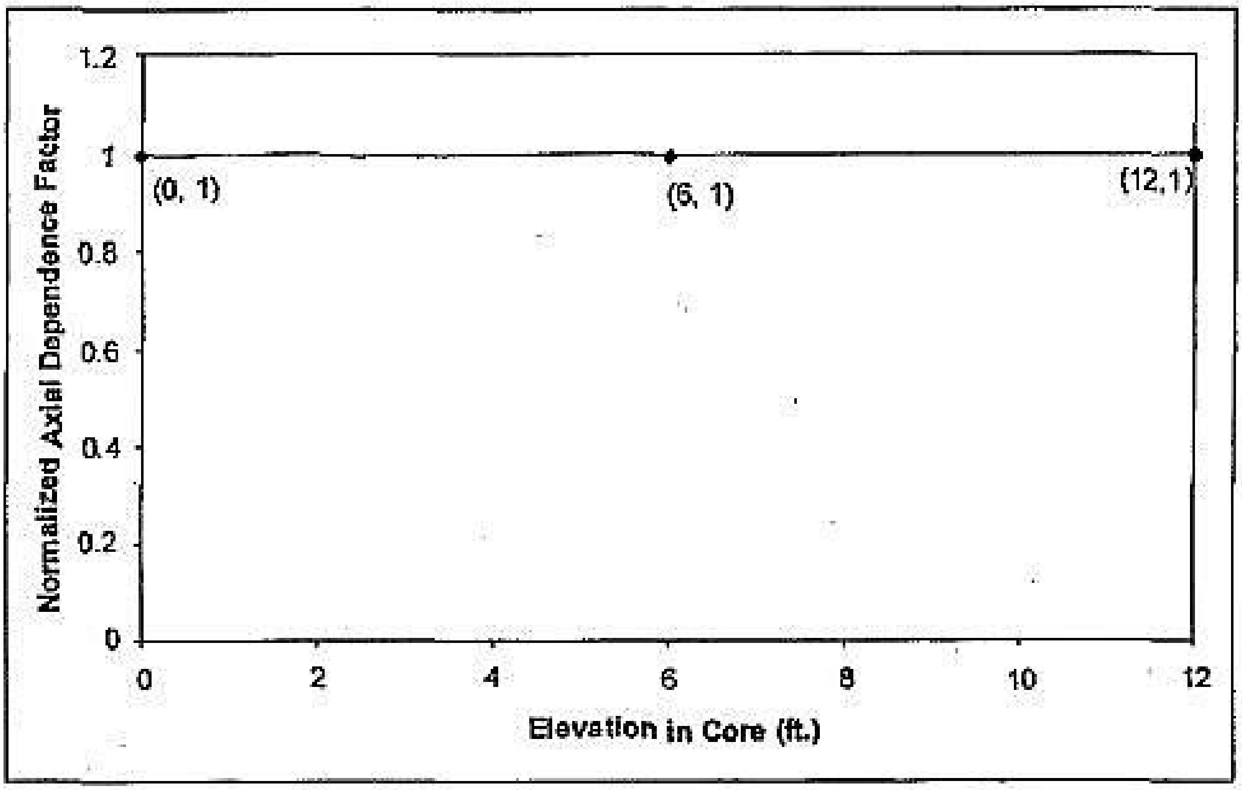


FIGURE 15.6.5-2

NORMALIZED THERMAL REACTOR POWER FOR LBLOCA (LIMITING CASE)

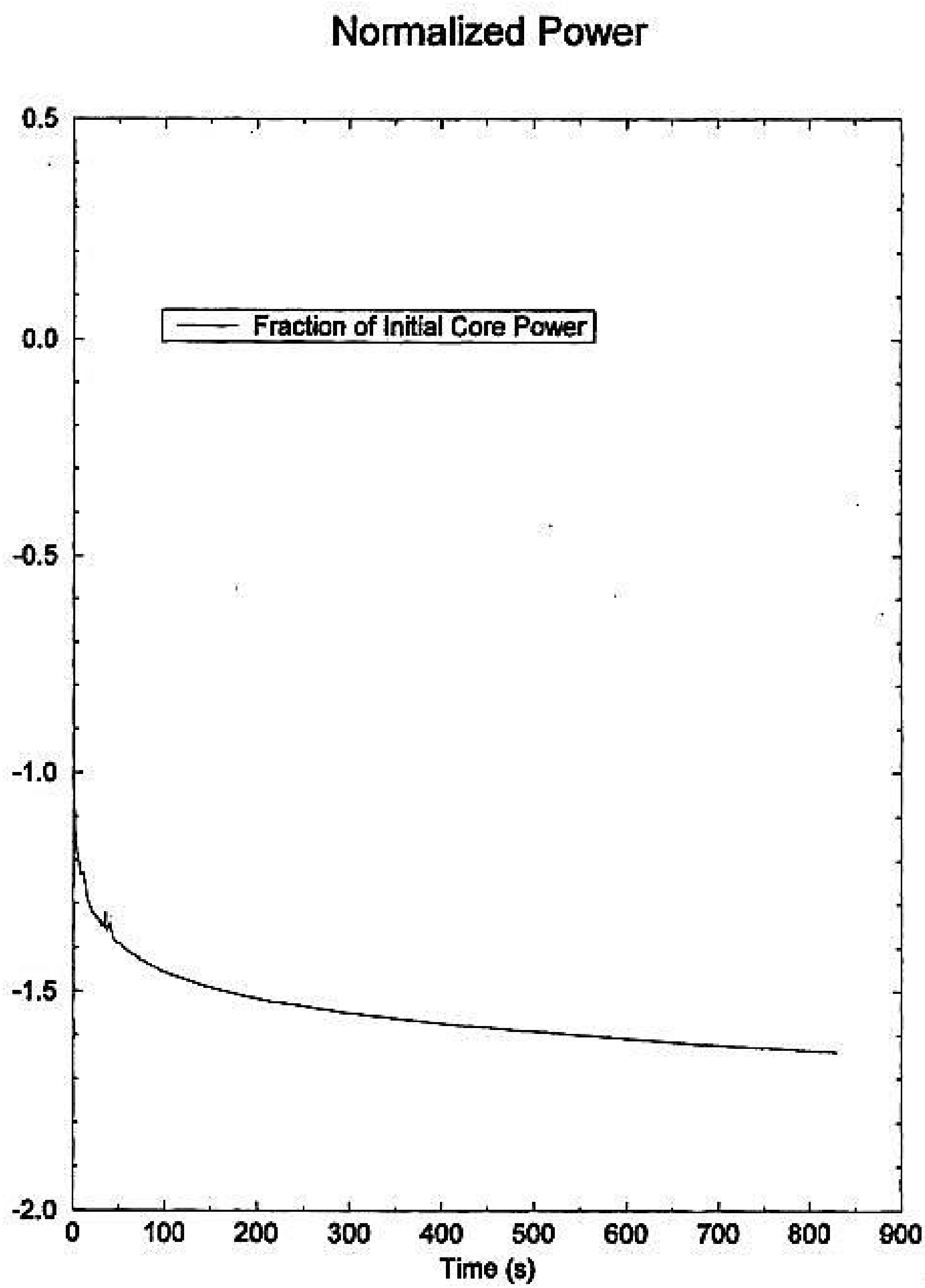


FIGURE 15.6.5-3

ECCS FLOWS (ACCUMULATOR, CHARGING, SI AND RHR) FOR LBLOCA (LIMITING CASE)

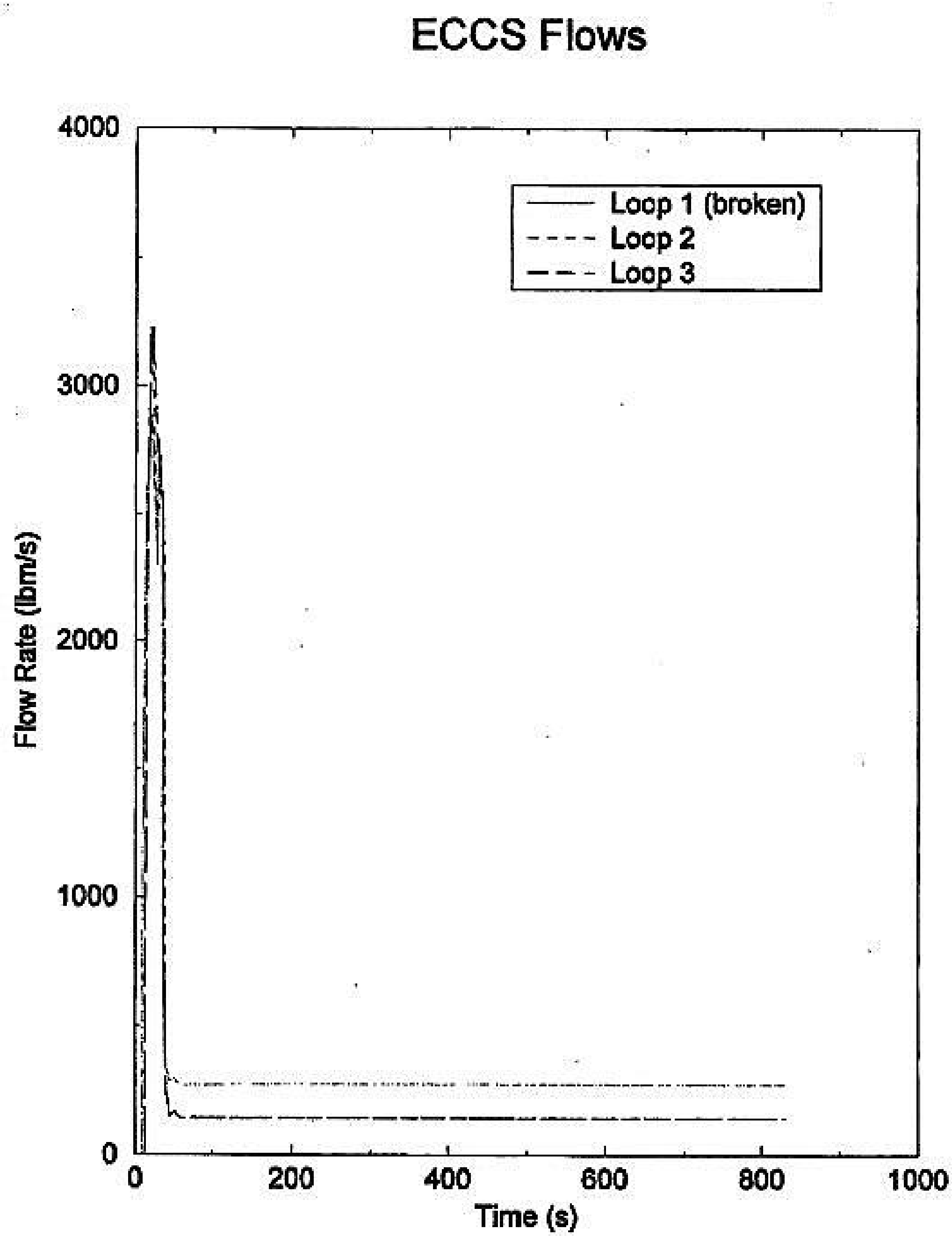


FIGURE 15.6.5-4

INTACT LOOP AND BROKEN LOOP HHSI FLOW FOR LBLOCA (LIMITING CASE)

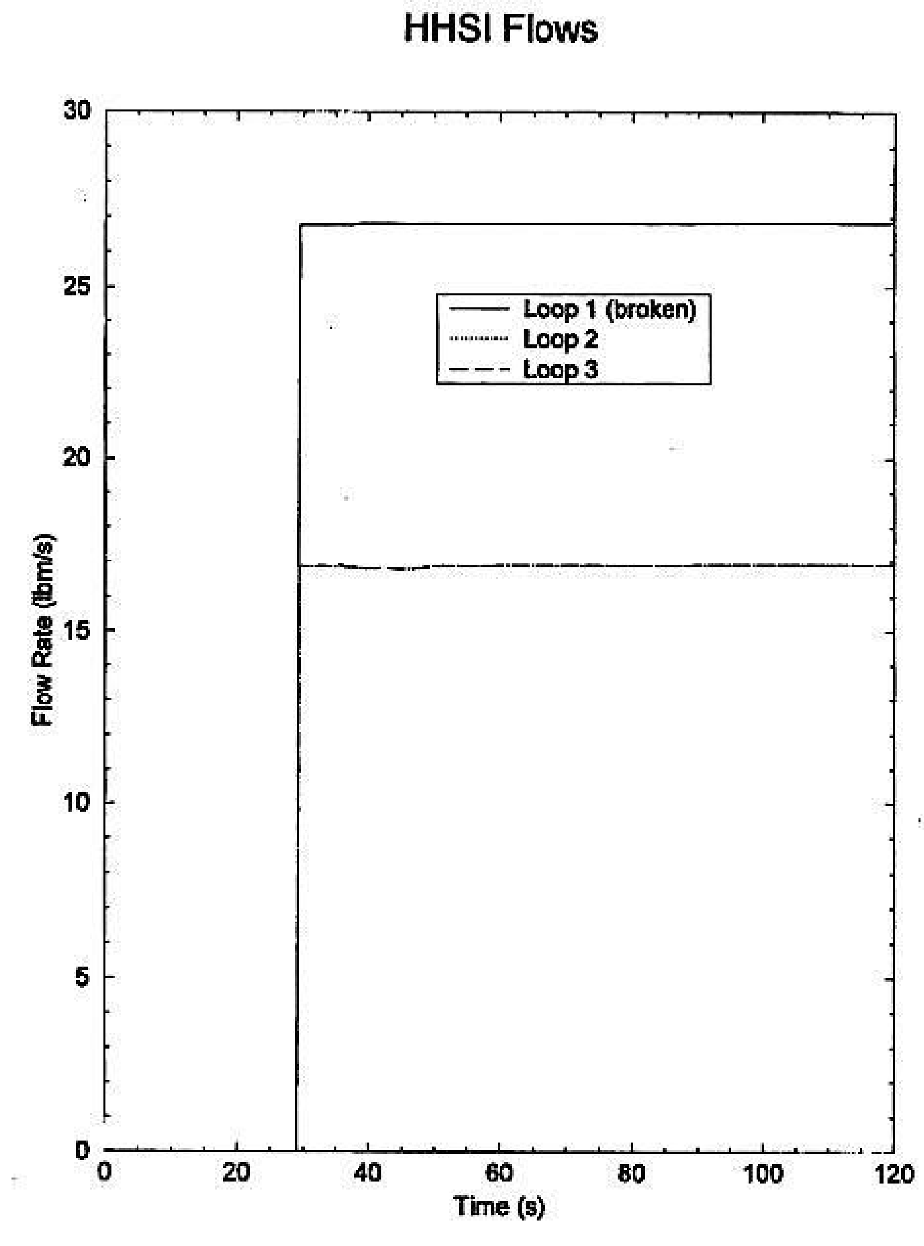


FIGURE 15.6.5-5

INTACT LOOP AND BROKEN LOOP LHSI FLOW FOR LBLOCA (LIMITING CASE)

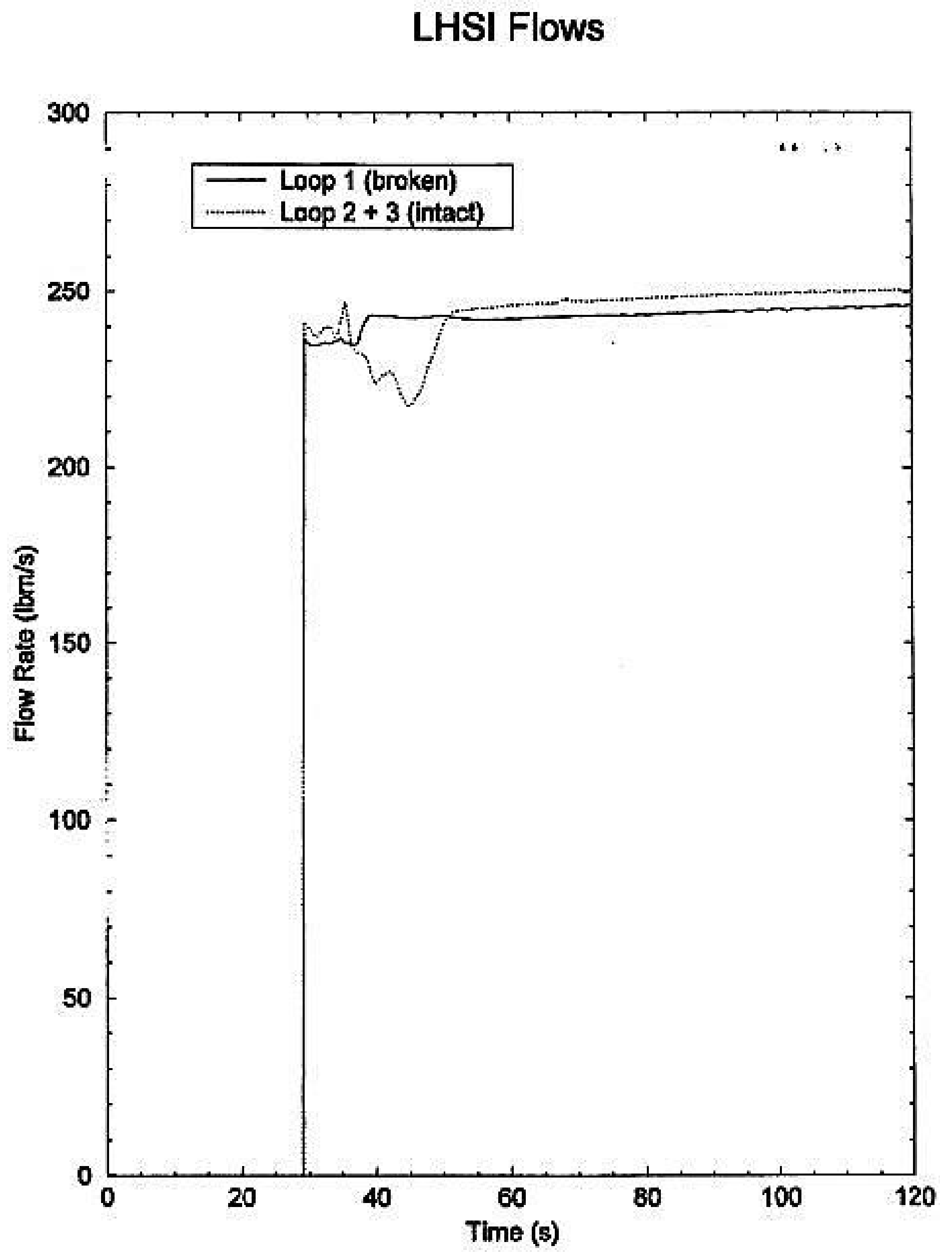


FIGURE 15.6.5-6

UPPER PLENUM PRESSURE FOR LBLOCA (LIMITING CASE)

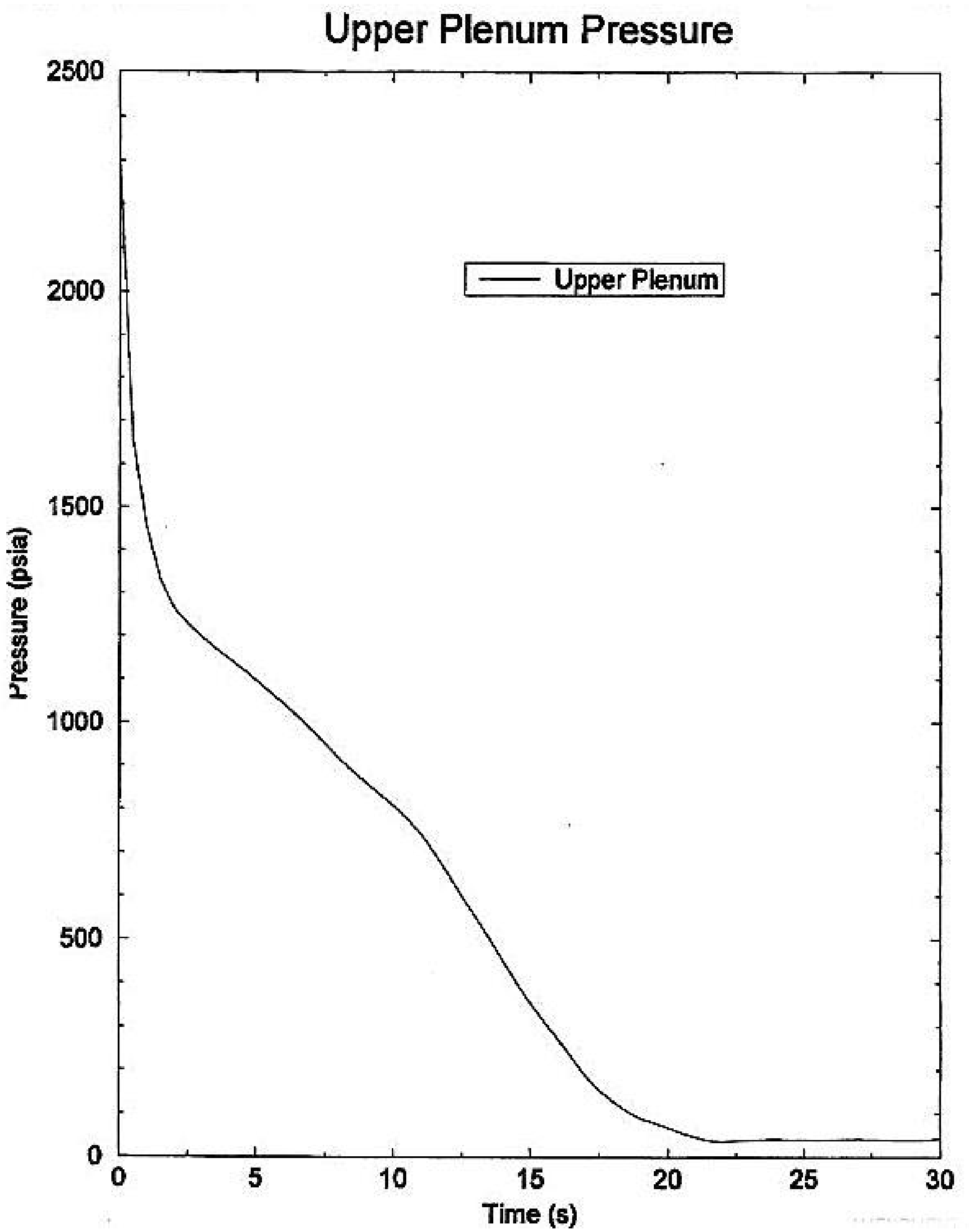


FIGURE 15.6.5-7

TOTAL BREAK FLOW RATE FOR LBLOCA (LIMITING CASE)

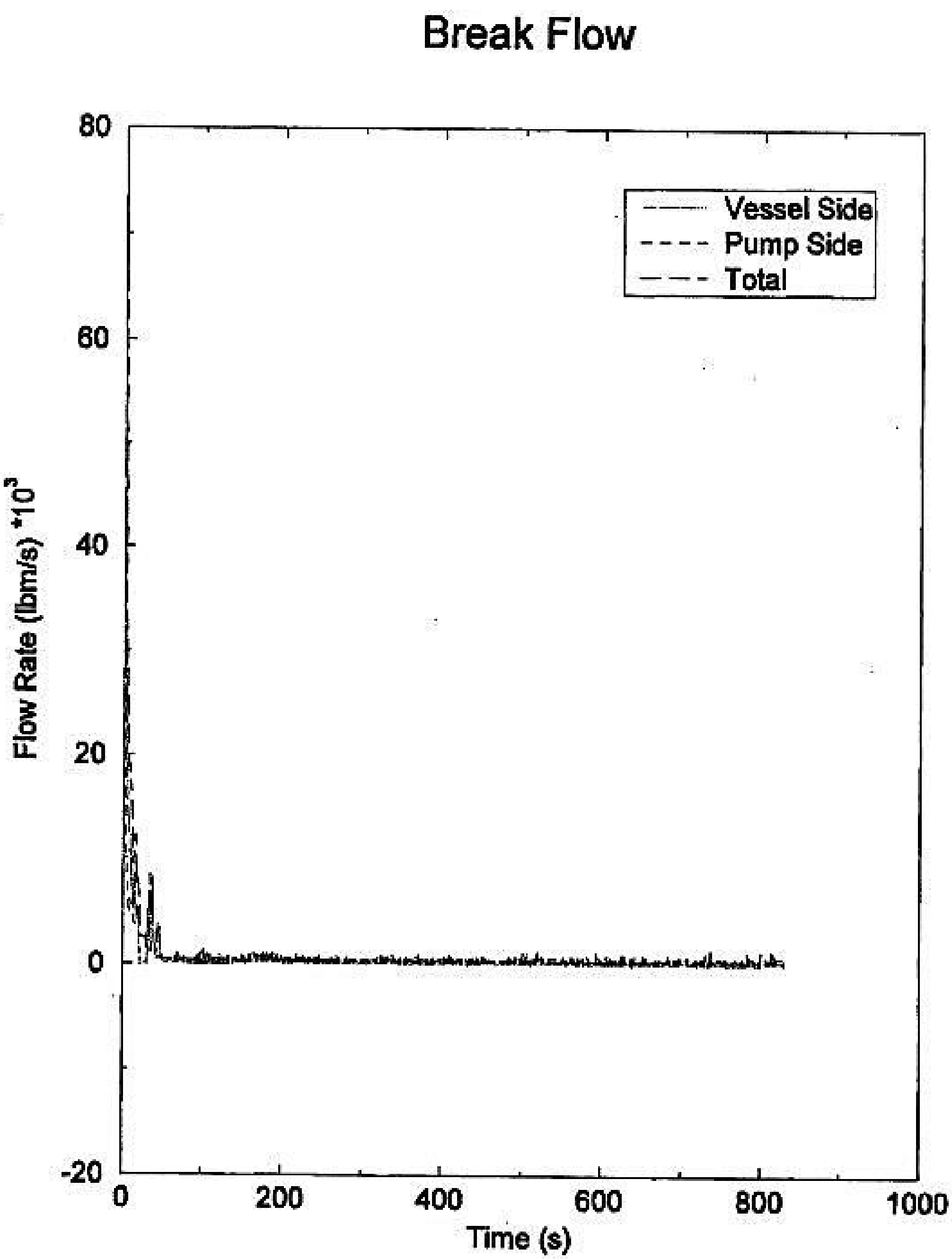


FIGURE 15.6.5-8

AVERAGE CORE INLET MAS FLUX FOR LBLOCA (LIMITING CASE)

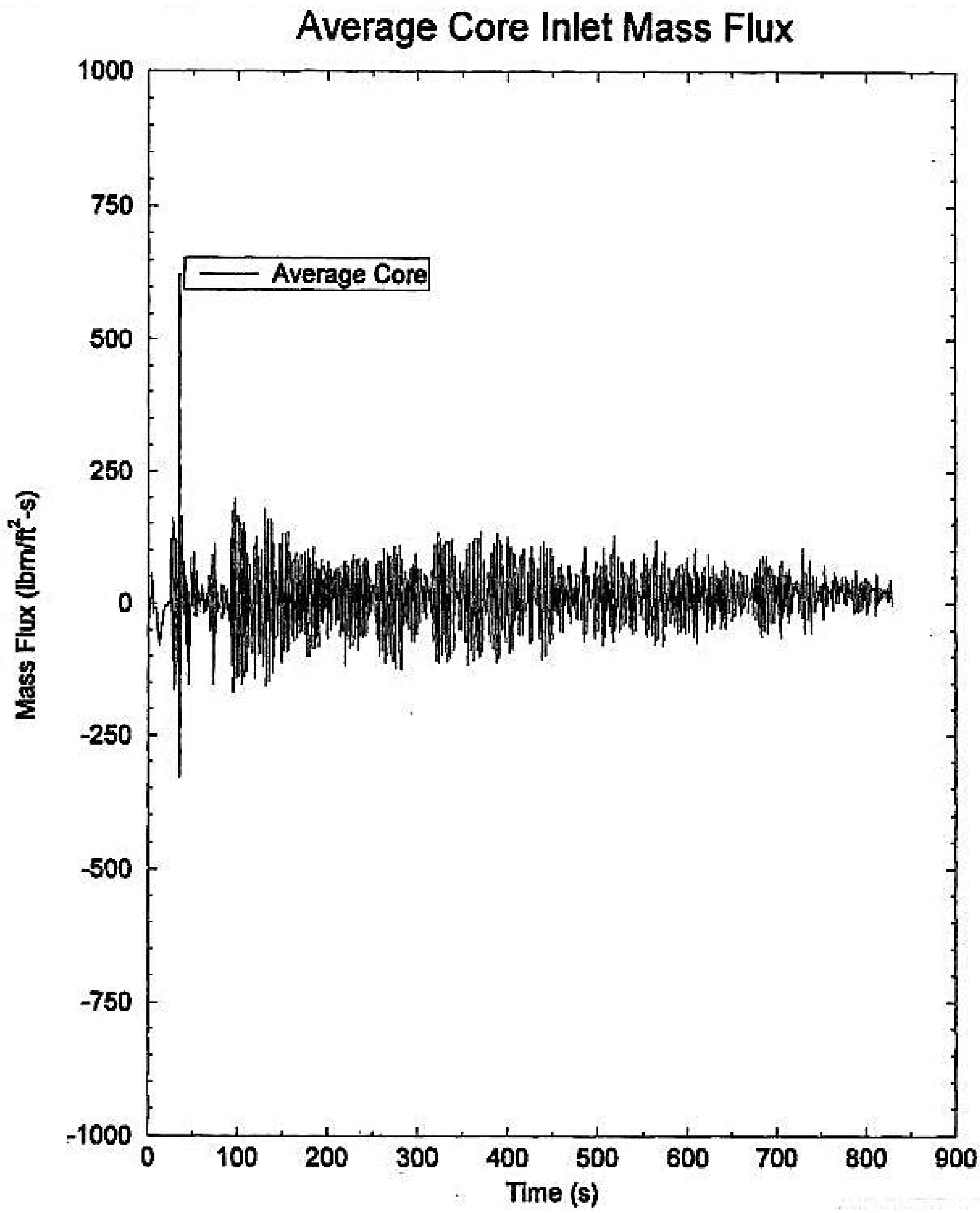


FIGURE 15.6.5-9

HOT ASSEMBLY INLET MASS FLUX FOR LBLOCA (LIMITING CASE)

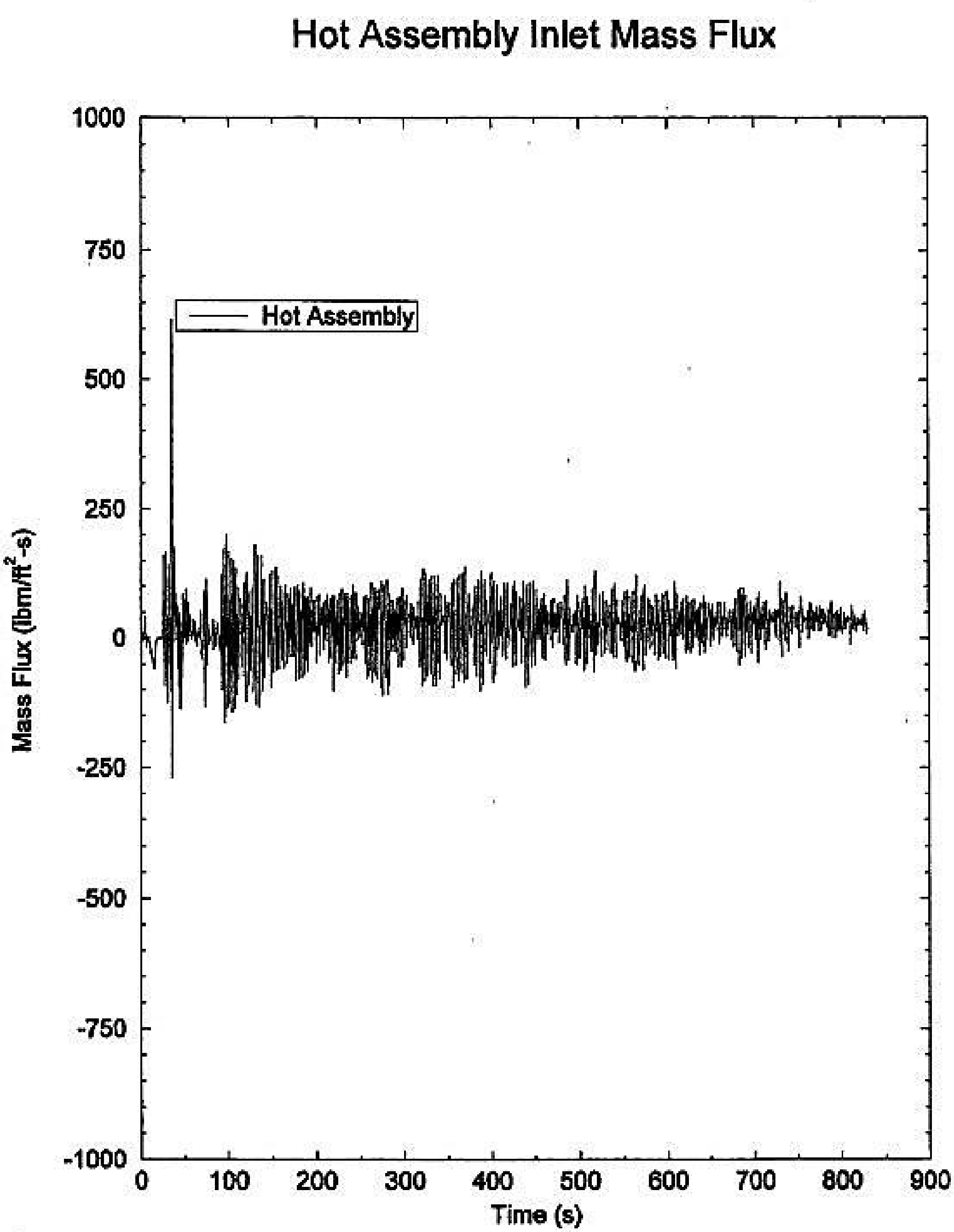


FIGURE 15.6.5-10

PCT NODE VOID FRACTION FOR LBLOCA (LIMITING CASE)

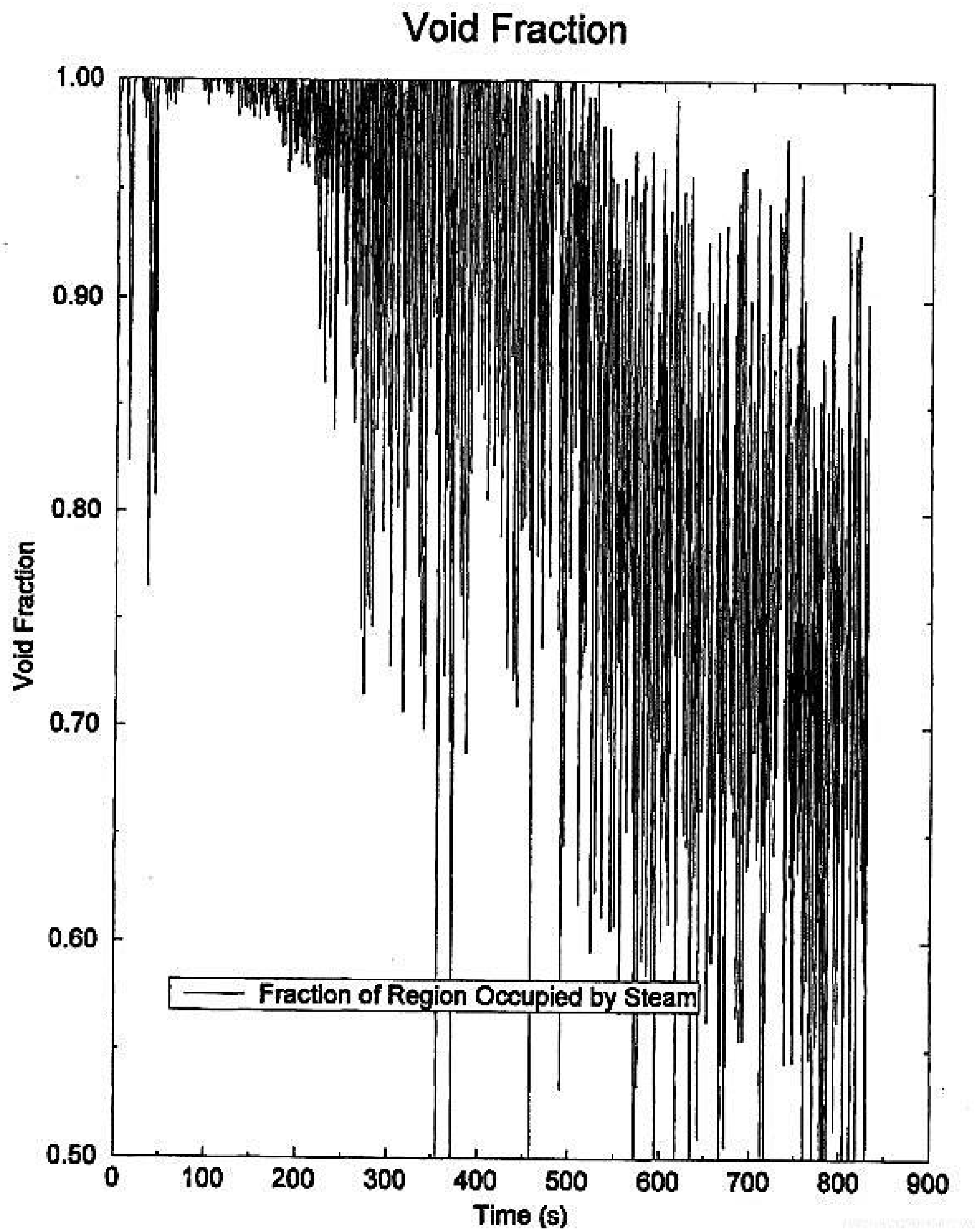


FIGURE 15.6.5-11

PCT NODE AVERAGE FUEL, CLADDING SURFACE AND FUEL CENTERLINE TEMPERATURES FOR LBLOCA (LIMITING CASE)

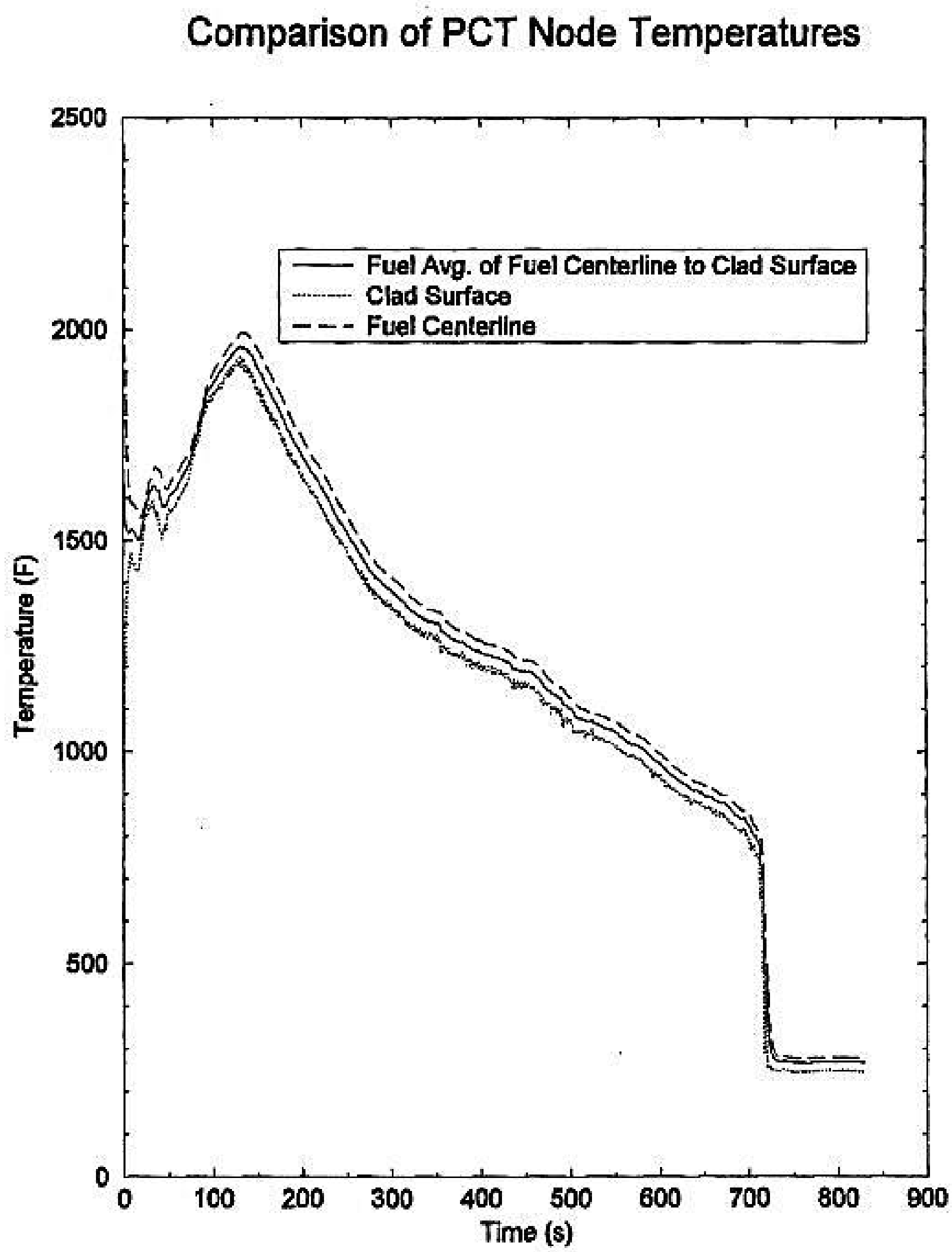


FIGURE 15.6.5-12

PCT NODE HEAT TRANSFER COEFFICIENT FOR LBLOCA (LIMITING CASE)

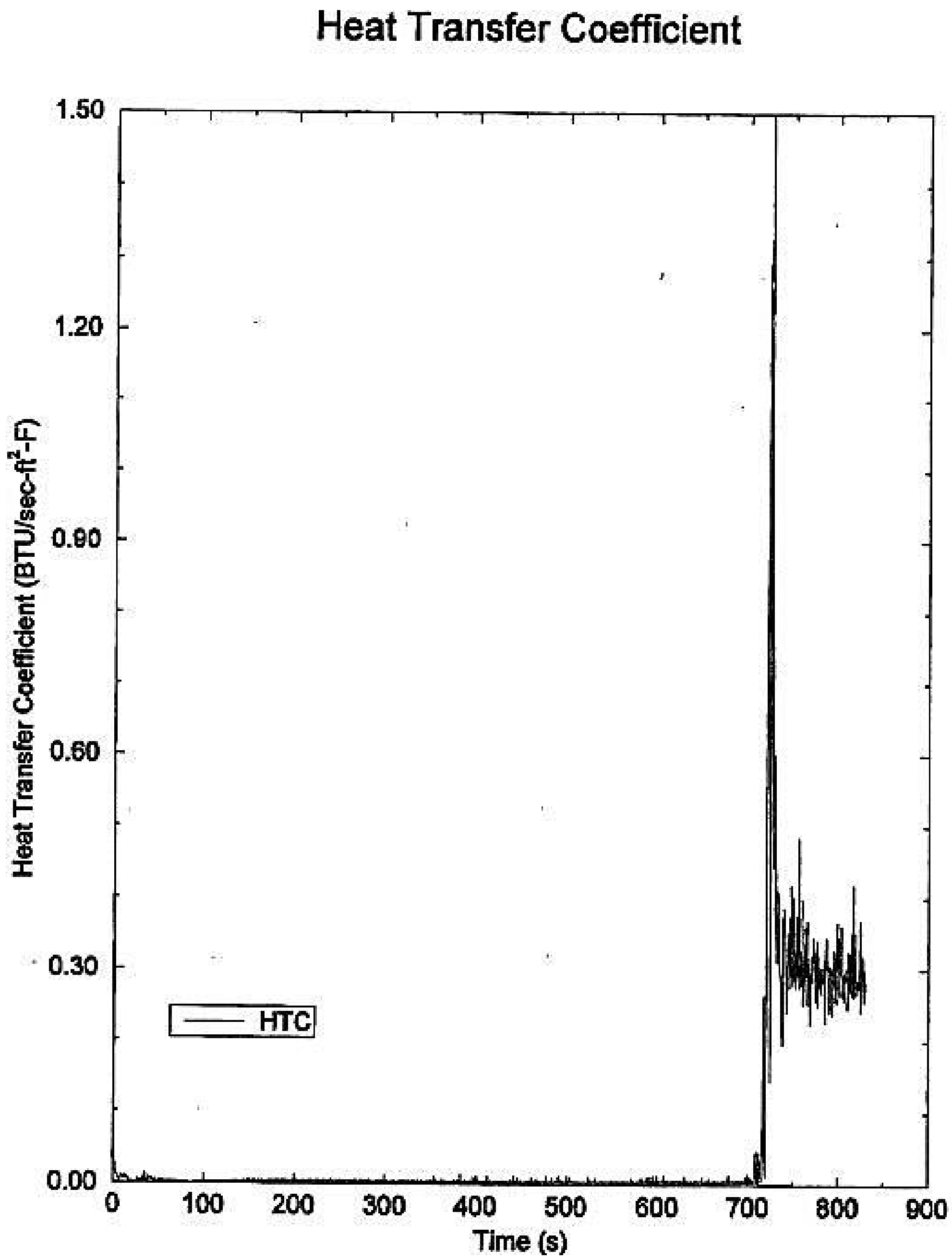


FIGURE 15.6.5-13

PCT NODE HEAT FLUX FOR LBLOCA (LIMITING CASE)

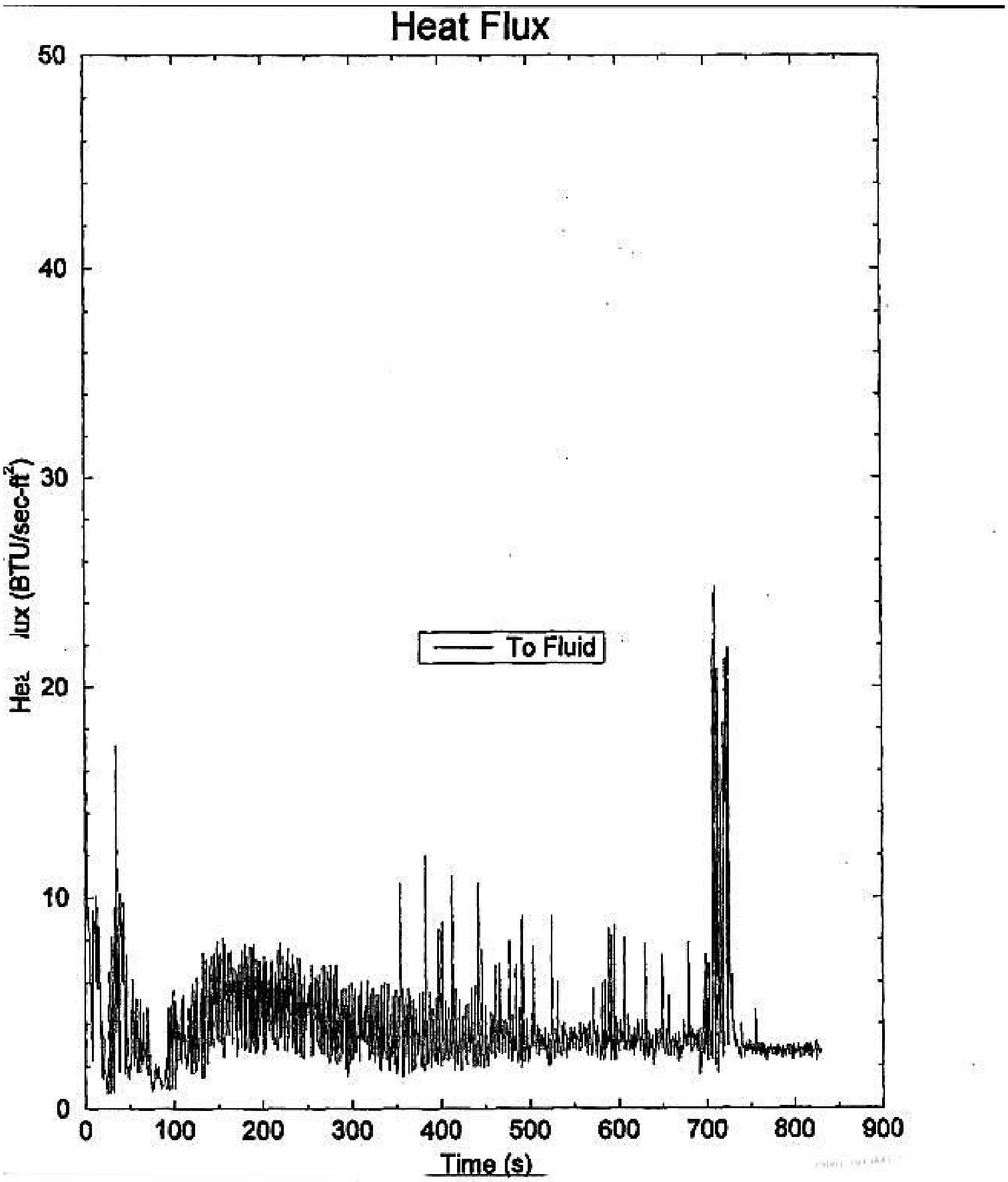


FIGURE 15.6.5-14

CONTAINMENT AND LOOP PRESSURES FOR LBLOCA (LIMITING CASE)

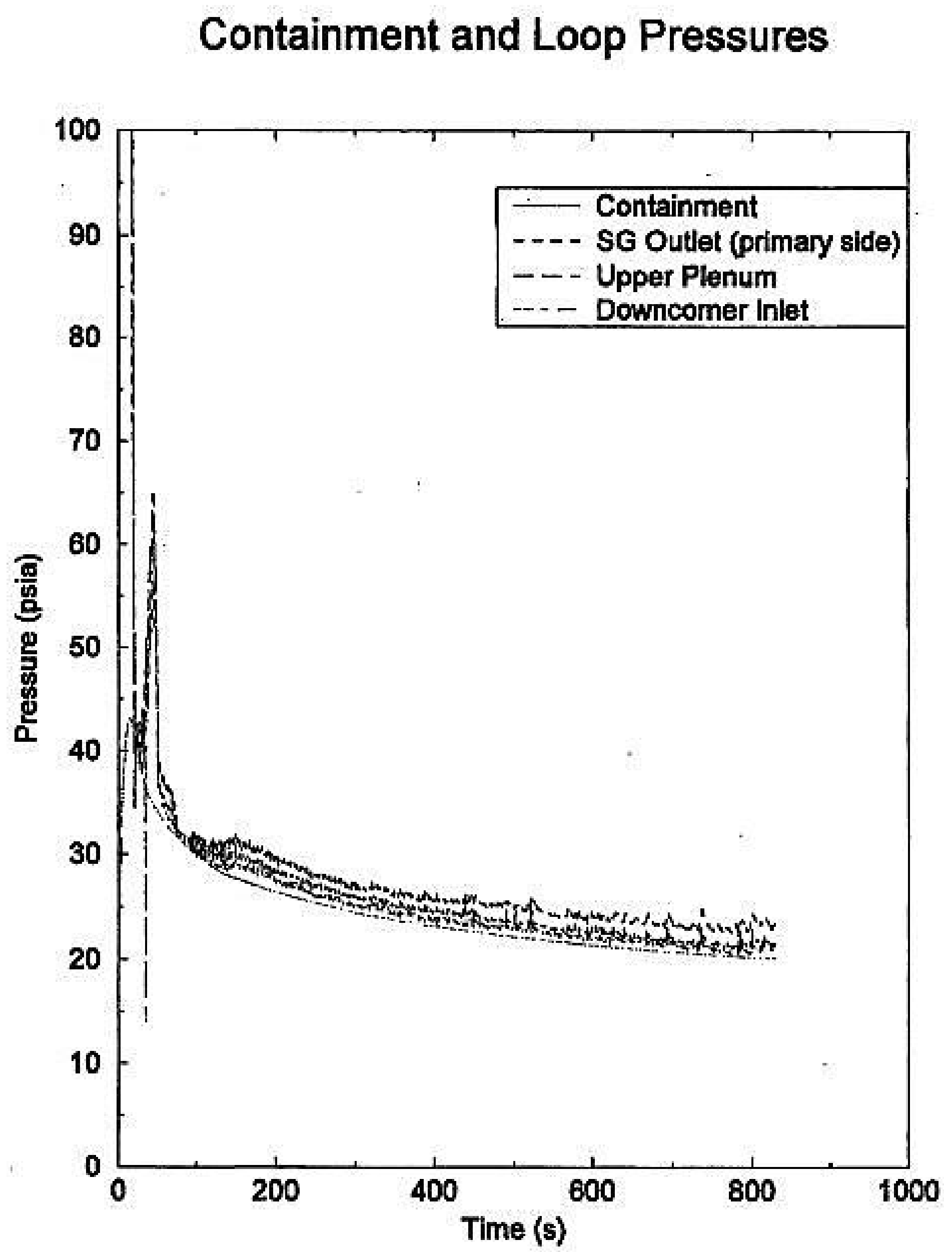


FIGURE 15.6.5-15

UPPER PLENUM PRESSURE FOR LBLOCA (LIMITING CASE)

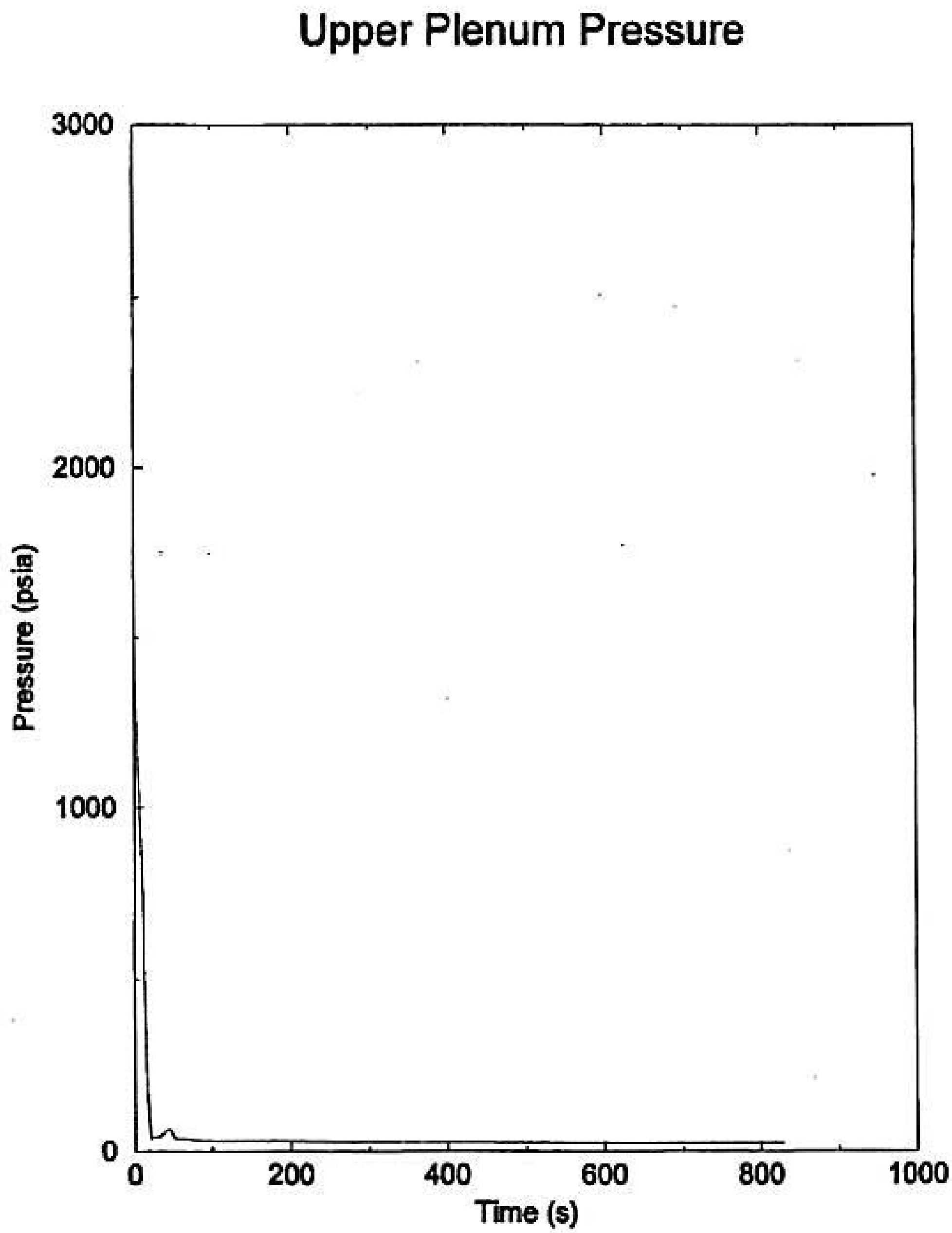


FIGURE 15.6.5-16

DOWNCOMER COLLAPSED LIQUID LEVEL FOR LBLOCA (LIMITING CASE)

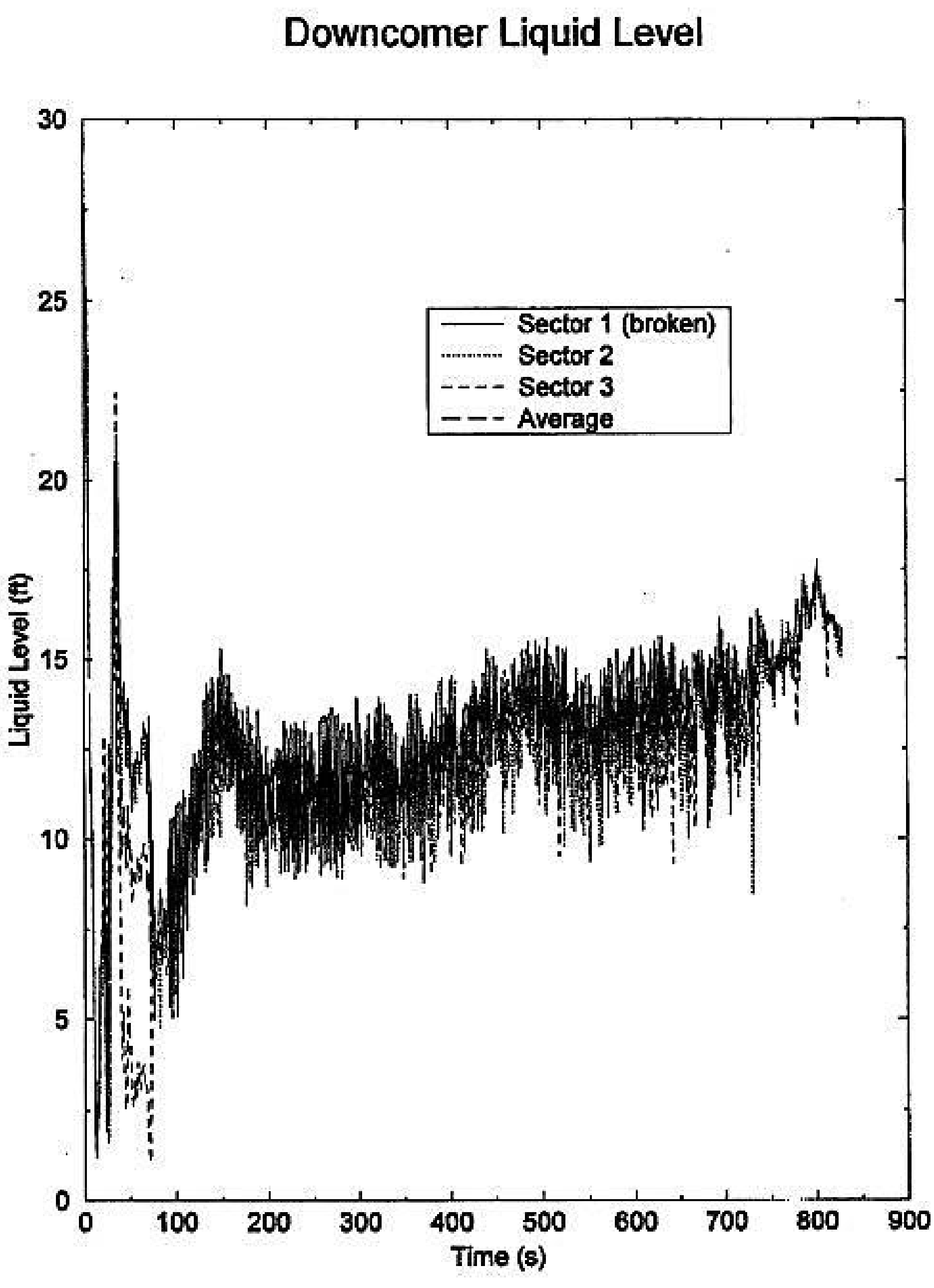


FIGURE 15.6.5-18

CORE COLLAPSED LIQUID LEVEL FOR LBLOCA (LIMITING CASE)

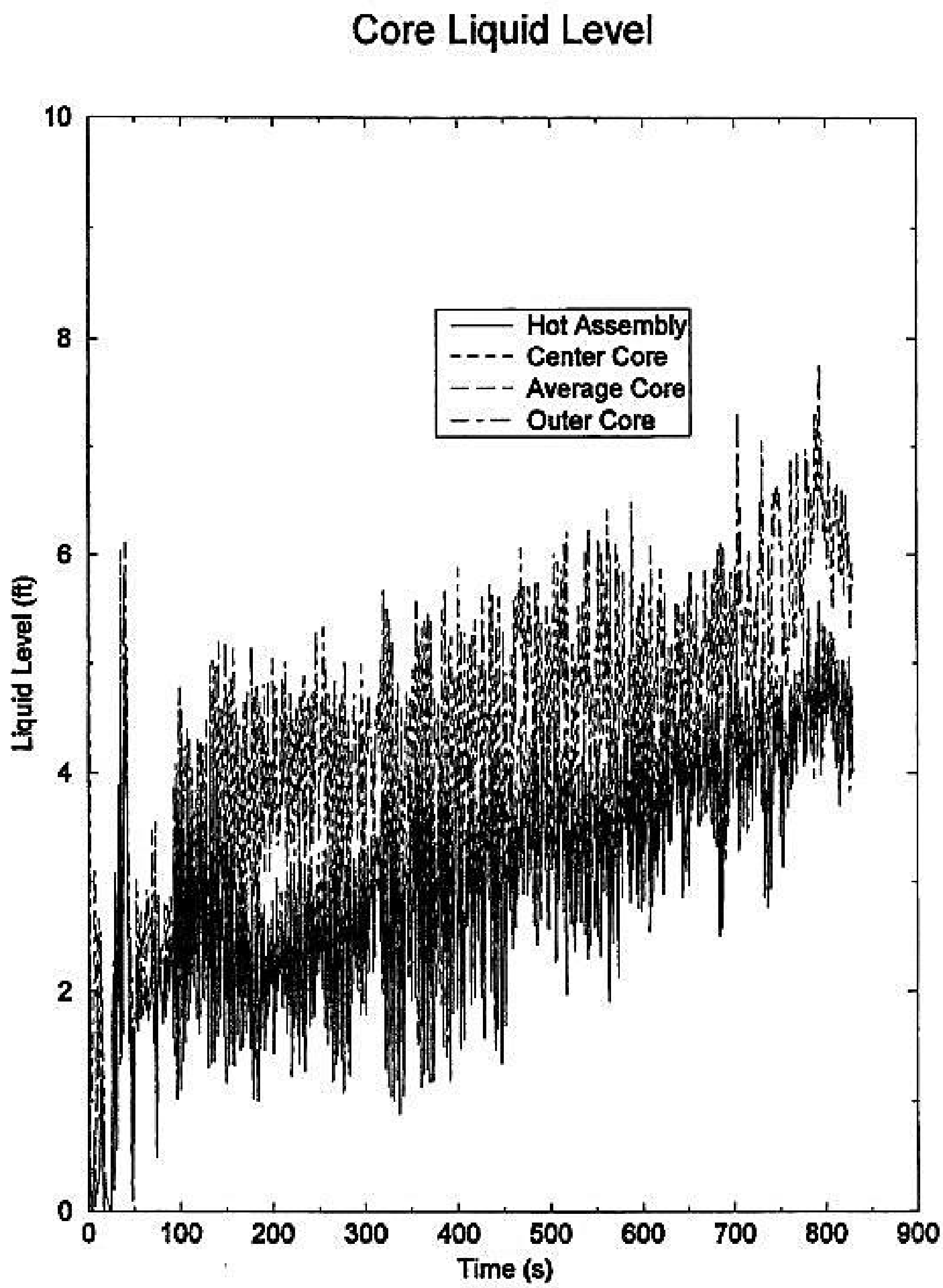


FIGURE 15.6.5-20

PEAK CLADDING TEMPERATURES FOR LBLOCA (LIMITING CASE)

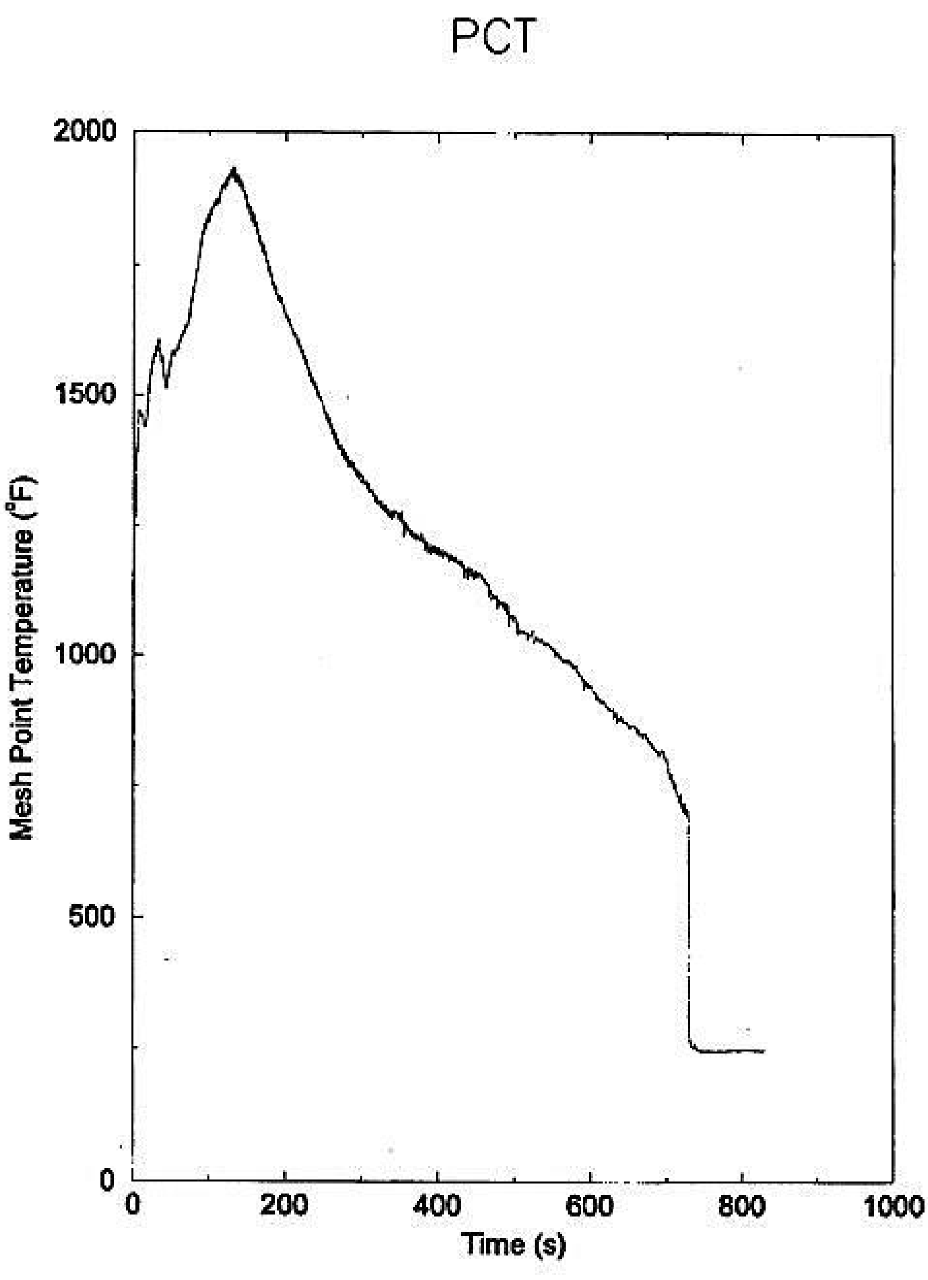


FIGURE 15.6.5-30

SYSTEM PRESSURES FOR LIMITING BREAK (2.6 INCH SBLOCA)

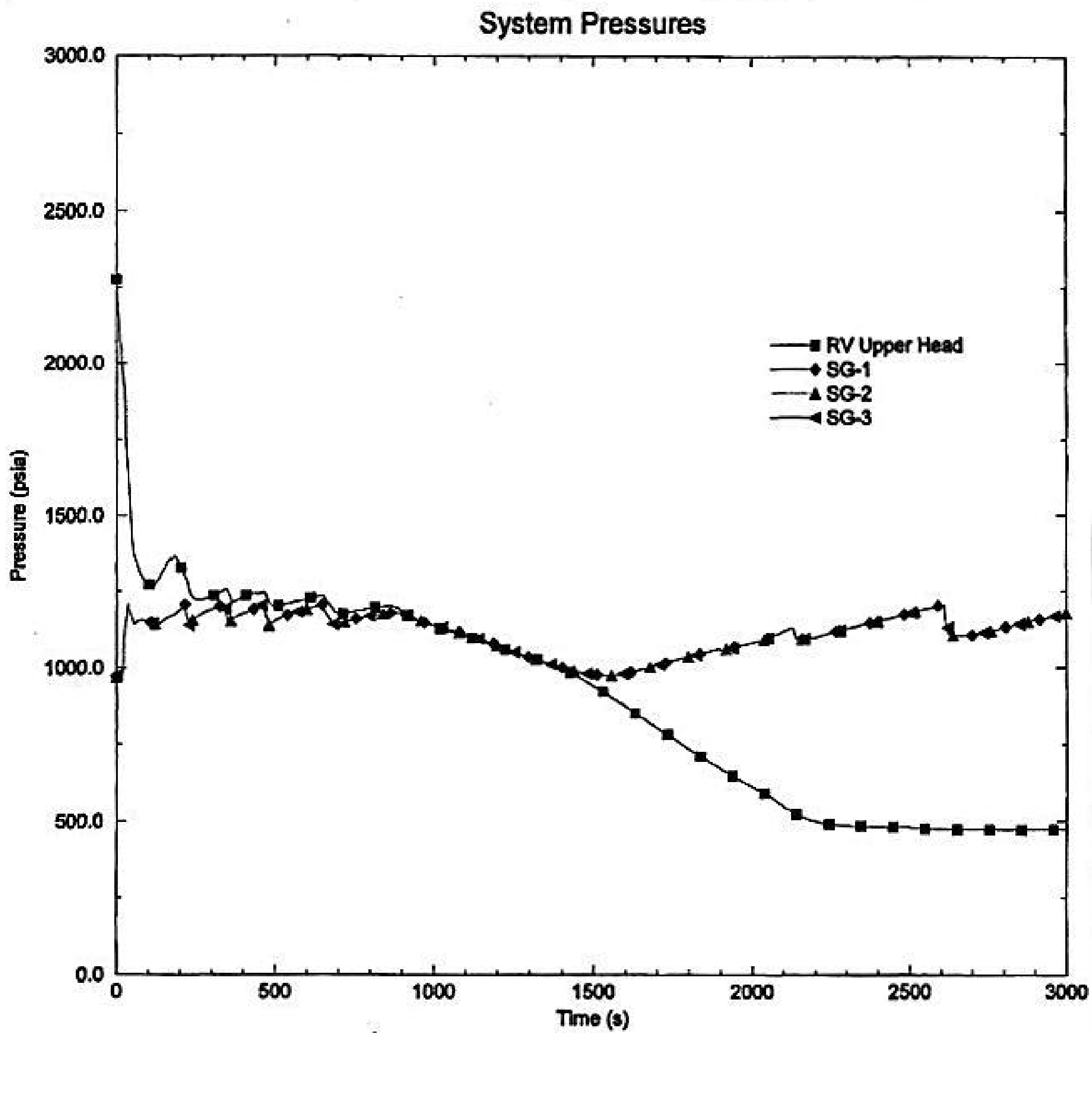


FIGURE 15.6.5-31

BREAK FLOW RATE FOR LIMITING BREAK (2.6 INCH SBLOCA)

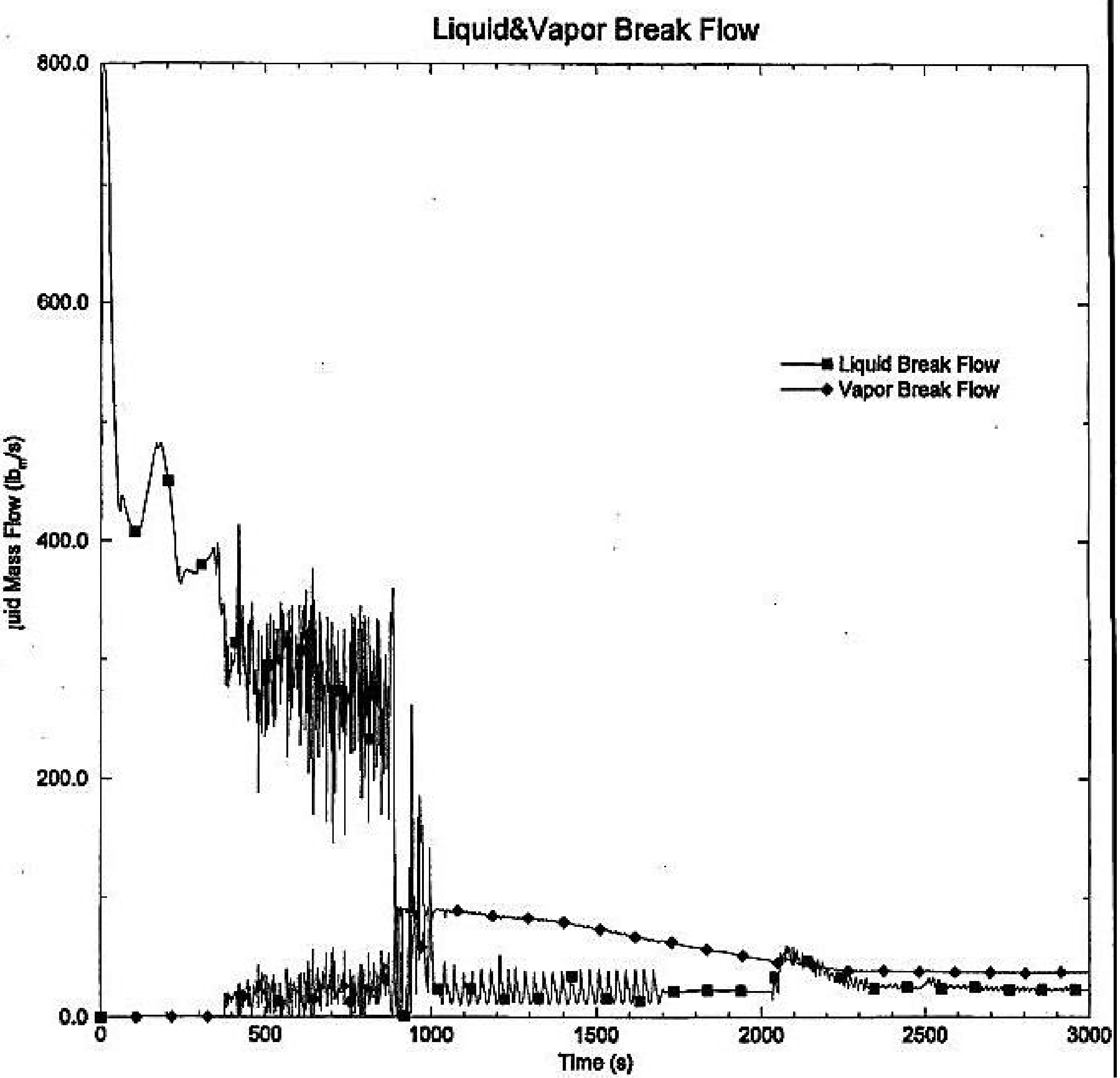


FIGURE 15.6.5-32

DOWNCOMER AND HOT ASSEMBLY COLLAPSED LIQUID LEVELS FOR LIMITING BREAK (2.6 INCH SBLOCA)

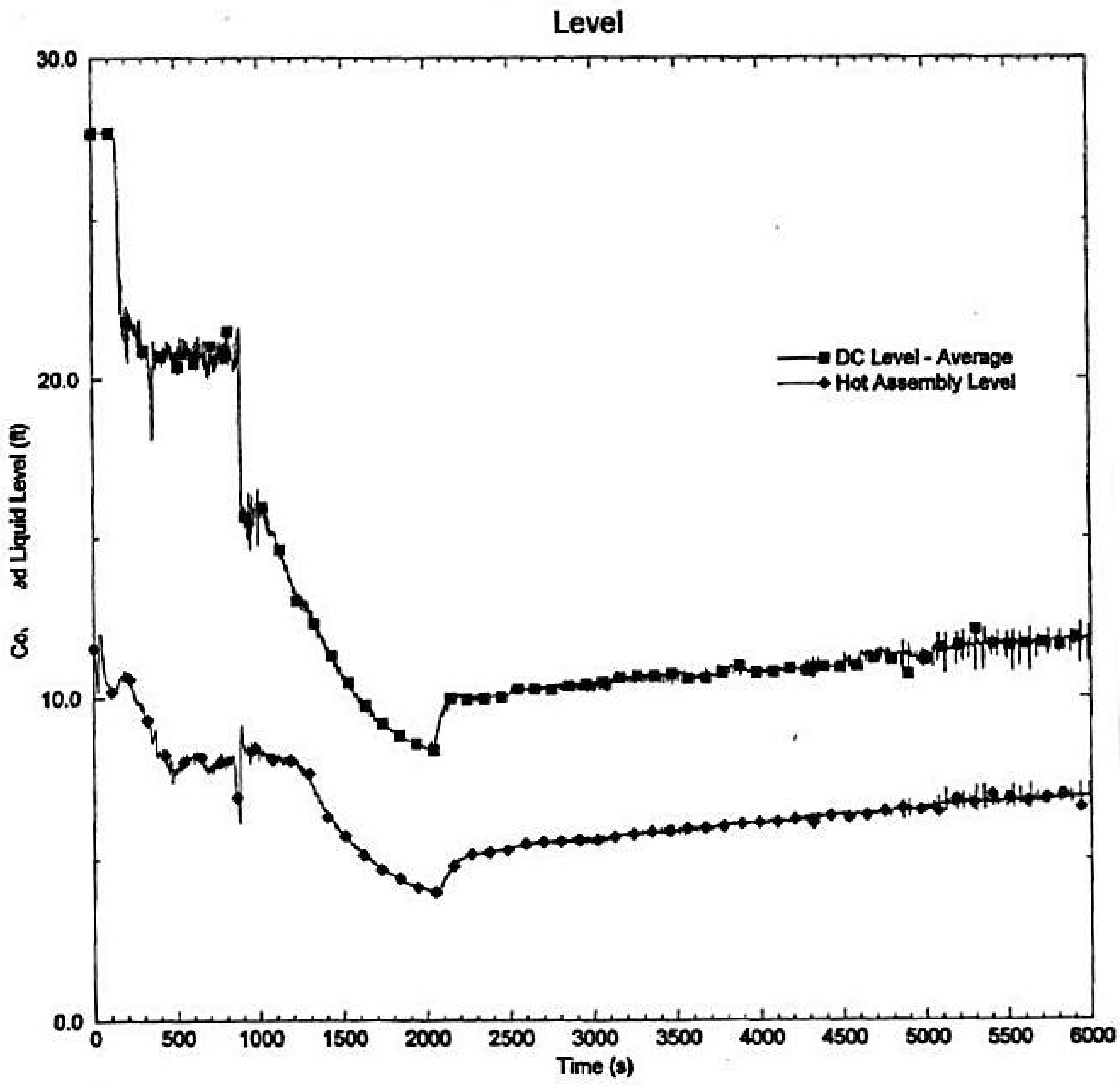


FIGURE 15.6.5-33

HHSI FLOWS FOR LIMITING BREAK (2.6 INCH SBLOCA)

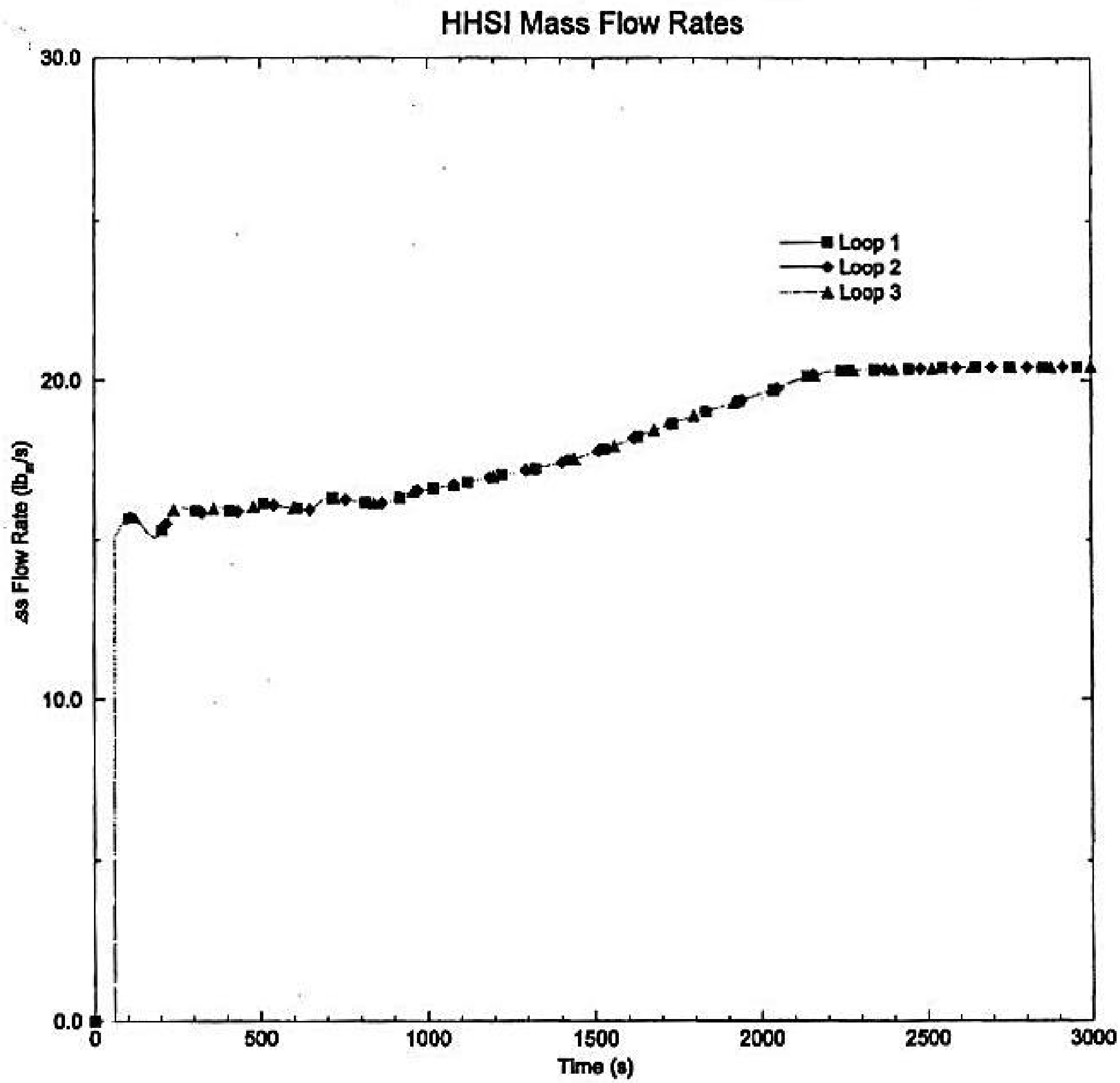


FIGURE 15.6.5-34

COMBINED ACCUMULATOR FLOW FOR LIMITING BREAK (2.6 INCH SBLOCA)

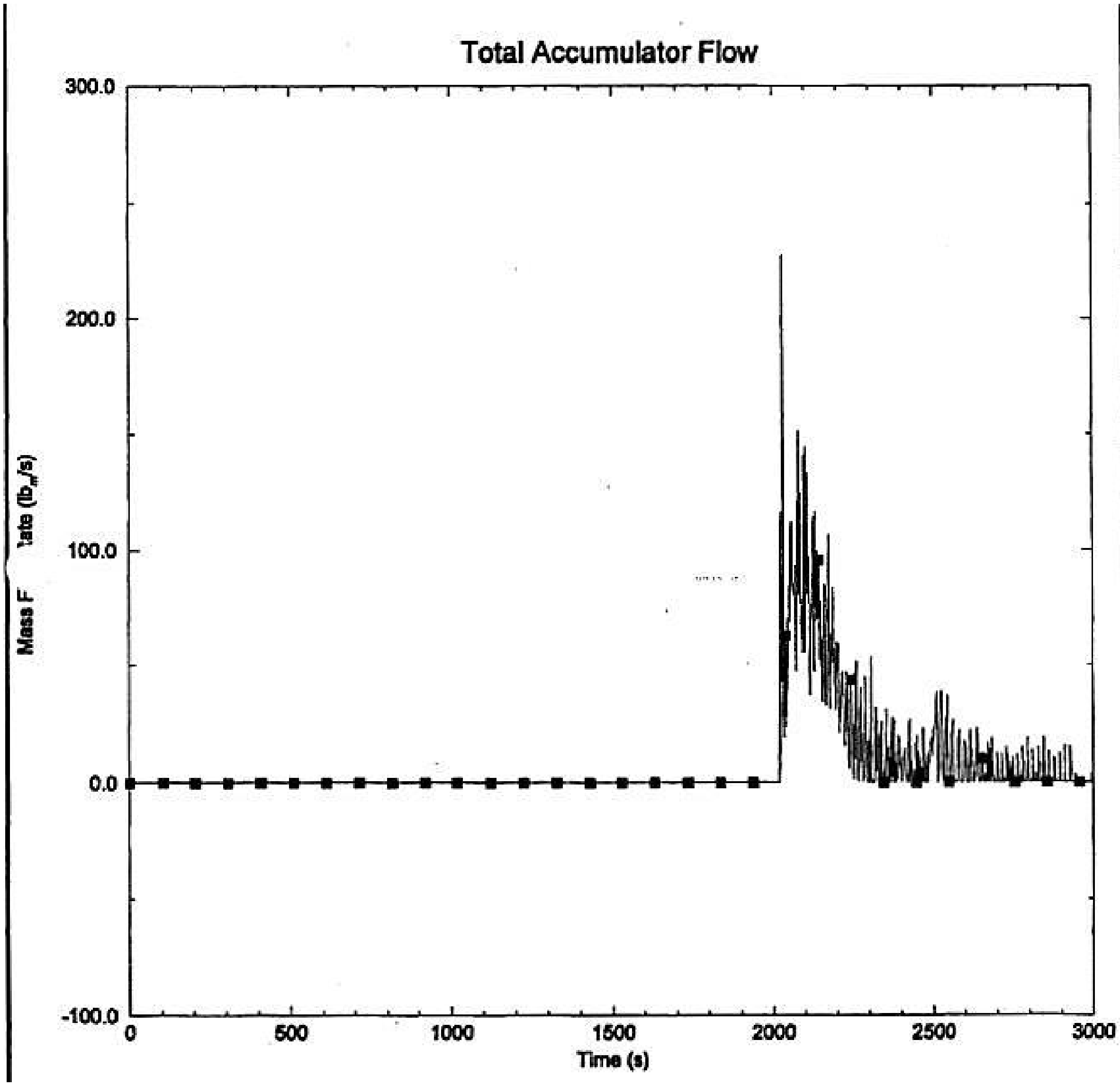


FIGURE 15.6.5-35

TOTAL RCS MASS AND REACTOR VESSEL MASS FOR LIMITING BREAK (2.6 INCH SBLOCA)

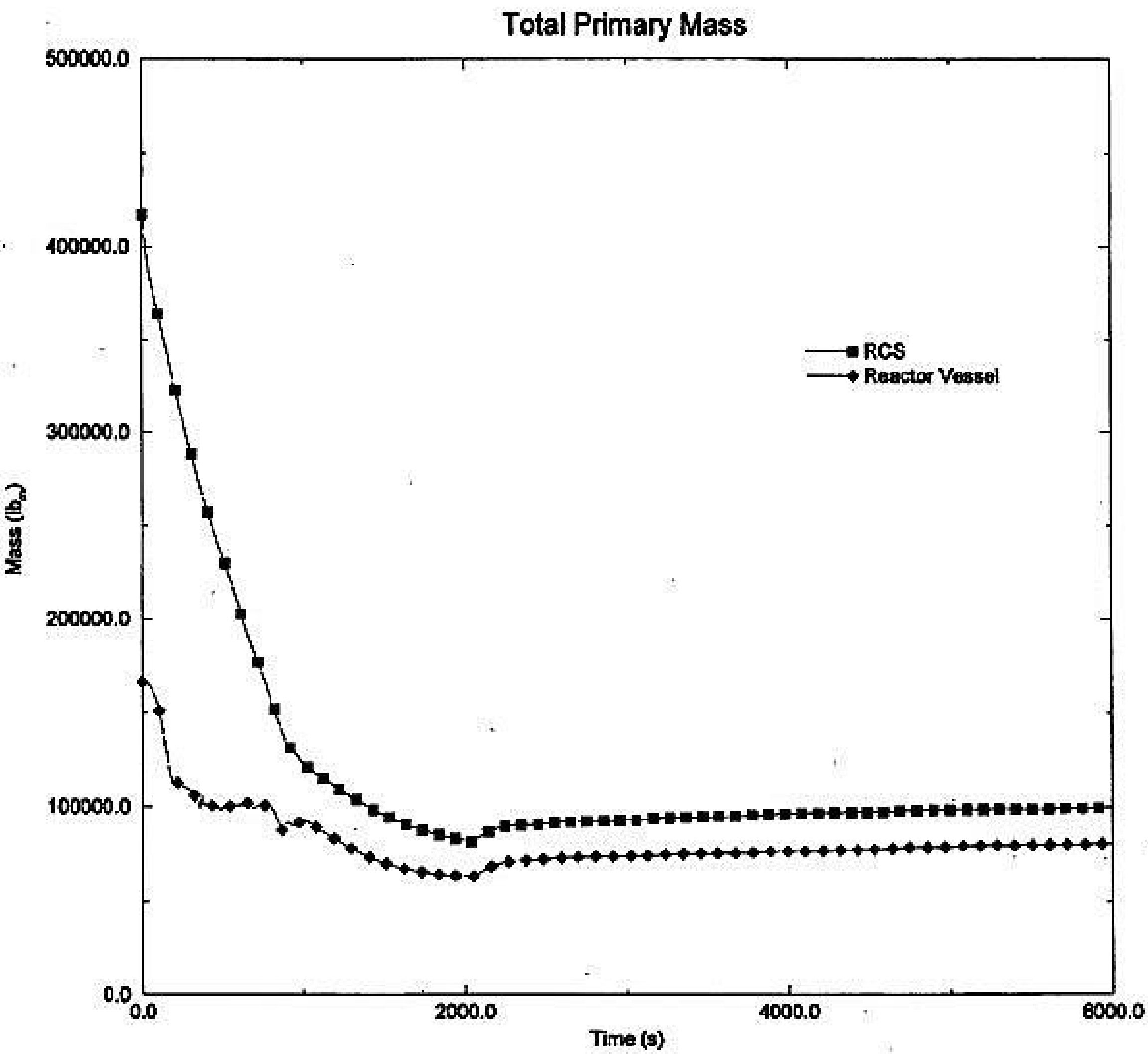


FIGURE 15.6.5-36

HOT ROD CLADDING TEMPERATURE FOR LIMITING BREAK (2.6 INCH SBLOCA)