CHAPTER 14 TABLE OF CONTENTS

14.0   SAFETY ANALYSIS

This section evaluates the safety aspects of either Unit 1 or Unit 2 of the plant, demonstrates that either or both units can be operated safely and that exposures from credible accidents do not exceed the guidelines of 10 CFR 50.67 or other applicable acceptance criteria.

This section is divided into three subsections, each dealing with a different behavior category:

Core and Coolant Boundary Protection Analysis, FSAR 14.1

With the exception of the Locked Rotor Accident, the abnormalities presented in FSAR 14.1 have no off-site radiation consequences. Radiological consequences, resulting from fuel cladding damage and a radioactivity release to the outside atmosphere, are assumed to occur as a result of the Locked Rotor Accident, presented in FSAR 14.1.8.

Standby Safety Features Analysis, FSAR 14.2

With the exception of the Locked Rotor Accident, the accidents presented in FSAR 14.2 are more severe than those discussed in FSAR 14.1 and may cause release of radioactive material to the environment.

Rupture of a Reactor Coolant Pipe, FSAR 14.3

The accident presented in FSAR 14.3, the rupture of a reactor coolant pipe, is the worst case accident and is the primary basis for the design of engineered safety features. It is shown that even the consequences of this accident are within the guidelines of 10 CFR 50.67.

Parameters and assumptions that are common to various accident analyses are described below to avoid repetition in subsequent sections.

Steady State Errors

For most accidents which are DNB limited, nominal values of initial conditions are assumed. The allowances on power, temperature, and pressure are determined on a statistical basis and are included in the limit DNBR, as described in WCAP-11397 (Reference 1). This procedure is known as the "Revised Thermal Design Procedure," and is discussed more fully in FSAR 3.2.

For accidents in which the Revised Thermal Design Procedure is not employed, the initial conditions are obtained by adding the maximum steady state errors to rated values. The following conservative steady state errors were assumed in the analyses:

1.   Core Power   ± 0.6%   allowance for calorimetric error
2.   Average Reactor Coolant Temp   ± 6.4°F   allowance for controller deadband
                                                     and measurement error
3.   Pressurizer Pressure   ± 50 psi   allowance for steady state
                                                     fluctuations and measurement error

Table 14.0-1 and Table 14.0-2 summarize initial conditions and computer codes used in the accident analyses, and show which accidents employed a DNB analysis using the Revised Thermal Design Procedure (RTDP).

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 and operating instructions. Power distribution may be characterized by the radial peaking factor ($F_{\Delta H}$) and the total peaking factor (F_Q). The peaking factor limits are given in the Technical Specifications.

For transients which may be DNB limited, the radial peaking factor is of importance. The radial peaking factor increases with decreasing power level due to rod insertion. This increase in $F_{\Delta H}$ is included in the core limits illustrated in Figure 14.0-1. All transients that may be DNB limited are assumed to begin with a $F_{\Delta H}$ consistent with the initial power level defined in the Technical Specifications. The axial power shape used in the DNB calculation is discussed in FSAR 3.2.

The radial and axial power distributions described above are input to the VIPRE code as described in FSAR 3.2.

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

For overpower transients which are slow with respect to the fuel rod thermal time constant (for example, the Chemical and Volume Control System malfunction which results in a decrease in the boron concentration in the reactor coolant, lasting many minutes, and the excessive increase in secondary steam flow incident which may reach equilibrium without causing a reactor trip), the fuel rod thermal evaluations are performed as discussed in FSAR 3.2. For overpower transients which are fast with respect to the fuel rod thermal time constant (for example, the uncontrolled rod cluster control assembly bank withdrawal from subcritical and rod cluster control assembly ejection incidents which result in a large power rise over a few seconds), a detailed fuel heat transfer calculation must be performed. Although the fuel rod thermal time constant is a function of system conditions, fuel burnup and rod power, a typical value at beginning-of-life for high power rods is approximately five seconds.

Reactivity Coefficients Assumed in the Accident Analyses

The transient response of the reactor system is dependent on reactivity feedback effects, in particular the moderator temperature coefficient and the Doppler power coefficient. These reactivity coefficients and their values are discussed in detail in FSAR 3.2.

In the analysis of certain events, conservatism requires the use of large reactivity coefficient values, whereas in the analysis of other events, conservatism requires the use of small reactivity coefficient values. Some analyses such as loss of coolant from cracks or ruptures in the Reactor Coolant System do not depend on reactivity feedback effects. The justification for use of conservatively large versus small reactivity coefficient values is treated on an event-by-event basis. In some cases conservative combinations of parameters are used to bound the effects of core life, although these combinations may represent unrealistic situations.

Rod Cluster Control Assembly Insertion Characteristics

The negative reactivity insertion following a reactor trip is a function of the position versus time of the rod cluster control assemblies 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. The rod cluster control assembly position versus time assumed in accident analyses is shown in Figure 14.0-2. The rod cluster control assembly insertion time to dashpot entry is taken as 2.2 seconds.

Figure 14.0-3 shows the fraction of total negative reactivity insertion versus normalized rod position for a core where the axial distribution is skewed to the lower region of the core. An axial distribution 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 which is input to all point kinetics core models used in transient analyses. The bottom-skewed power distribution itself is not input into the point kinetics core model. There is inherent conservatism in the use of Figure 14.0-3 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, significant negative reactivity would have been inserted due to the more favorable axial distribution existing prior to trip.

The normalized rod cluster control assembly negative reactivity insertion versus time is shown in Figure 14.0-4. The curve shown in this figure was obtained from Figure 14.0-2 and Figure 14.0-3. A total negative reactivity insertion following a trip of 5 percent $\Delta K/K$ is assumed in the transient analyses except where specifically noted otherwise. This assumption is conservative with respect to the calculated trip reactivity worth available. For Figure 14.0-2 and Figure 14.0-3, the rod cluster control assembly drop is normalized to 2.2 seconds, unless otherwise noted for a particular event.

Reactor Trip

A reactor trip signal acts to open the two series trip breakers feeding power to the control rod drive mechanisms. The loss of power to the mechanism coils causes the mechanisms to release the control rods, which then fall by gravity into the core. There are various instrumentation delays associated with each tripping function, including delays in signal actuation, in opening the trip breakers, and in the release of the rods by the mechanisms. The total delay to trip is defined as the time delay from the time that trip conditions are reached to the time the rods are free and begin to fall. The time delay assumed for each tripping function is given in Table 14.0-3.

Reference is made in Table 14.0-3 to overtemperature and overpower ΔT trip points shown in Figure 14.0-1. Figure 14.0-1 presents the allowable reactor coolant loop average temperature and ΔT for the design flow and power distribution, as described in FSAR 3.2, as a function of primary coolant pressure. The boundaries of operation defined by the overpower ΔT trip and the overtemperature ΔT trip are represented as "Protection Lines" on this diagram. The protection lines are drawn to include all adverse instrumentation and setpoint errors so that under nominal conditions a trip would occur well within the area bounded by these lines. The utility of this diagram is in the fact that the limit imposed by any given DNBR can be represented as a line. The DNB lines represent the locus of conditions for which the DNBR equals the safety analysis limit value. All points below and to the left of a DNB line for a given pressure have a DNBR greater than the limit value. The diagram shows that DNB is prevented for all cases if the area enclosed with the maximum protection lines is not traversed by the applicable DNBR line at any point.

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).

The limit value, which was used as the DNBR limit for all accidents analyzed with the Revised Thermal Design Procedure (see Table 14.0-1), is conservative compared to the actual design DNBR value required to meet the DNB design basis as discussed in FSAR 3.2.

The difference between the limiting trip point assumed for the analysis and the normal trip point represents an allowance for instrumentation channel error and setpoint error. Nominal trip setpoints are specified in the plant Technical Specifications.

Determining Reactor Power Level through Secondary Calorimetric

To assure that the initial reactor power level prior to an overpower transient is maintained within the accident analysis assumption of 100.6%, a secondary plant calorimetric is performed on a periodic basis to determine core thermal power and to set the power range flux instruments to this measured power. The calorimetric power level is calculated using measurement of secondary parameters such as feedwater flow, feedwater inlet temperature to the steam generators and steam pressure. High accuracy instrumentation is provided for these measurements, such that total instrument error is less than or equal to 0.6%. If the Leading Edge Flow Meter (LEFM) used to measure feedwater flow is out of service, the operating reactor power level is reduced to account for increased calorimetric measurement uncertainty of the feedwater flow venturis, so that reactor power continues to be maintained within the accident analysis assumption for initial reactor power level.

Plant-to-Plant Interaction

The safety evaluation of a two unit plant, where two reactors are situated in close physical proximity on the same site, sharing certain facilities and operated as combined power producing units, requires that the safety assessment treat the plant as a two unit facility rather than as two individual single unit facilities. However, for the reasons discussed below, the nature of the two unit plant design confines the location of a reactor fault condition to one of the two units at any time (with the exception of possible faults arising in the electrical grid system to which both units are connected, and these have no off-site radiation consequences). Thus, for the two unit plant, the potential consequences of each and every credible reactor fault condition are no different than those for a single unit plant.

Possible sources of interaction between the two units are discussed below:

Sharing of Systems

As noted in FSAR 1.0, FSAR 9.0, FSAR 10.0, and FSAR 11.0, all or part of certain systems (e.g., Chemical and Volume Control System, Waste Disposal System) are shared by the two units. A functional evaluation of the components of those systems which are shared by the two units is given in Appendix A.6.

The plant is provided with a control room which is common to both units. Physical separation of control panels in the control room essentially eliminates interaction of the control systems of the two units. The two units are connected to the same external electrical grid, and it is therefore possible that the following transients could affect both units simultaneously:

1.   Loss of external electrical load (FSAR 14.1.9)

2.   Loss of all AC power to the station auxiliaries (FSAR 14.1.11)

The design is such that the occurrence of either of these two transients, in both units simultaneously, can be accommodated without an unsafe condition arising in either unit.

Except for the electrical grid conditions noted above, all systems which are shared by both units are designed such that a shared system can neither cause a simultaneously unsafe condition in both units, nor propagate an accident condition, which may arise in one unit, to the other unit.

Physical Proximity

The positioning of the two units in close physical proximity introduces no possibility of external interaction. For each unit, the integrity of all systems whose functions are necessary to maintain the safety of the reactor is ensured by the nature of the design: e.g., through separation of redundant components such as wiring, and missile shielding both inside and outside the containment. Thus, with the exception of the electrical faults already noted, the two unit plant precludes by the nature of its design, any possibility of either (a) simultaneous occurrence in both units of fault conditions having a common origin, (b) the propagation from one unit to the other unit of a fault condition.

In addition, it is not considered credible that both units could develop unrelated accidents, either of the same or a different nature simultaneously. Thus, the criteria for plant design require the capability to deal with the affected unit while maintaining safe control of the other unit. Although these criteria do not directly imply that the other unit must be shut down following the occurrence of an accident condition in one unit, the two unit plant design includes the capability to meet all safety criteria in the affected unit, and simultaneously shut the second unit down and maintain it at hot shutdown, if required. In fact, continued on-line operation of the adjacent unit enhances the assurance of a continuous supply of electrical power for the engineered safety features of the affected unit.

In a two unit plant, the overall design of each unit represents no essential departure from the current design of the unit which comprises a single unit plant. Thus, the methods and techniques for the safety assessment of a single unit plant are directly applicable to a two unit plant. Further, since both units of a two unit plant are nearly identical, the safety assessment (presented in this section for a single unit) is equally applicable to either unit.

Computer Codes Utilized

Summaries of some of the principal computer codes used in transient analyses are given below. Other codes, in particular very specialized codes in which the modeling has been developed to simulate one given accident, such as those used in the analysis of the primary system pipe rupture (FSAR 14.3), are summarized in their respective accident analyses sections. The codes used in the analyses of each transient have been listed in Table 14.0-1.

Advanced Nodal Code (ANC) / SPNOVA (Reference 7, Reference 10, and Reference 11)

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

The SPNOVA code utilizes the same Westinghouse standard core design methodology with three-dimensional (3-D) nodal expansion methodology for static analysis of cores that is incorporated into the ANC computer program (Reference 11). SPNOVA includes a neutron kinetics capability and uses the Stiffness Confinement Method to solve time dependent equations.

The ANC licensing topical report, WCAP-10965 (Reference 7), was approved by the NRC via an SER from C. Berlinger (NRC) to E. P. Rahe (Westinghouse), dated June 23, 1986. The SPNOVA licensing topical report, WCAP-12983 (Reference 10), was approved by the NRC via an SER from A. C. Thadani (NRC) to W. J. Johnson (Westinghouse), dated November 26, 1990. A process improvement that has resulted in streamlining and consolidating the Westinghouse neutronics code system was discussed in a letter (Reference 11) from N. J. Liparulo (Westinghouse) to R. C. Jones (NRC), dated March 29, 1996. As concluded in that letter, the implementation of the ANC solution method in SPNOVA eliminated the solution differences between ANC and SPNOVA, and also eliminated the SPNOVA normalization step to the ANC conditions, addressing the SPNOVA SER conditions imposed due to the solution differences between ANC and SPNOVA.

FACTRAN (Reference 2)

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

1.   A sufficiently large number of radial space increments to handle fast transients such as rod ejection accidents.

2.   Material properties which are functions of temperature and a sophisticated fuel-to-cladding gap heat transfer calculation.

3.   The necessary calculations to handle post-DNB transients: film boiling heat transfer correlations, Zircaloy-water reaction and partial melting of the materials.

LOFTRAN (Reference 3)

The LOFTRAN program is used for studies of transient response of a PWR system to specified perturbations in process parameters. LOFTRAN simulates a multiloop system by a model containing reactor vessel, hot and cold leg piping, steam generator (tube and shell sides) and the pressurizer. The pressurizer heaters, spray, and relief and safety valves are also considered in the

program. Point model neutron kinetics, and reactivity effects of the moderator, fuel, boron, and rods are included. The secondary side of the steam generator utilizes a homogeneous, saturated mixture for the thermal transients and a water level correlation for indication and control. The Reactor Protection System is simulated to include reactor trips on high neutron flux, overtemperature ΔT, overpower ΔT, high and low pressurizer pressure, low flow, and high pressurizer level. Control systems are also simulated including rod control, steam dump, feedwater control, and pressurizer pressure control. The Emergency Core Cooling System, including the accumulators and upper head injection, is also modeled.

LOFTRAN is a versatile program which is suited to both accident evaluation and control studies as well as parameter sizing. It also has the capability of calculating the transient value of DNBR based on the input from the core limits illustrated in Figure 14.0-1. The core limits represent the minimum value of DNBR as calculated for typical or thimble cell.

RETRAN (Reference 8)

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

TWINKLE (Reference 4)

The TWINKLE program is a multi-dimensional spatial neutron kinetics code, which is patterned after steady state codes presently used for reactor core design. The code uses an implicit finite-difference method to solve the two-group transient neutron diffusion equations in one, two and three dimensions. The code uses six delayed neutron groups and contains a detailed multi-region fuel-cladding-coolant heat transfer model for calculating pointwise Doppler and moderator feedback effects. The code handles up to 8000 spatial points, and performs its own steady state initialization. Aside from basic cross-section data and thermal-hydraulic parameters, the code accepts as input basic driving functions such as inlet temperature, pressure, flow, boron concentration, control rod motion, and others. Various edits are provided, e.g., channelwise power, axial offset, enthalpy, volumetric surge, pointwise power, and fuel temperatures.

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

THINC

The THINC Code is described in Reference 7 and Reference 21, of FSAR 3.2.

VIPRE (Reference 9)

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

The VIPRE licensing topical report, WCAP-14565 (Reference 9), was approved by the NRC via an SER from T. H. Essig (NRC) to H. Sepp (Westinghouse), dated January 19, 1999.

14.0.1   REFERENCES

1.   Friedland, A. J., Ray, S., "Revised Thermal Design Procedure," WCAP-11397 (Proprietary), February 1987.

2.   Hargrove, H. G., "FACTRAN - A Fortran-IV Code for Thermal Transients in a UO2 Fuel Rod," WCAP-7908 (Proprietary), June 1972.

3.   Burnett, T. W. T., et al., "LOFTRAN Code Description," WCAP-7907-P-A, April 1984.

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

5.   WE letter to NRC, VPNPD-96-051, "Supplement to Technical Specifications Change Requests 188 and 189," dated August 5, 1996.

6.   NRC SE for License Amendment Nos. 207 and 212, "Measurement Uncertainty Recapture Power Uprate," dated November 29, 2002.

7.   Liu, Y. S., et al., "ANC: A Westinghouse Advanced Nodal Computer Code," WCAP-10965-P-A, December 1985.

8.   Huegel, D. S., et al., "RETRAN-02 Modeling and Qualification for Westinghouse Pressurized Water Reactor Non-LOCA Safety Analyses," WCAP-14882-P-A, April 1999.

9.   Sung, Y. X., et al., "VIPRE-01 Modeling and Qualification for Pressurized Water Reactor Non-LOCA Thermal-Hydraulic Safety Analyses," WCAP-14565-P-A, October 1999.

10.   Chao, Y. A., et al., "SPNOVA - A Multidimensional Static and Transient Computer Program for PWR Core Analysis," WCAP-12394-A (Proprietary) and WCAP-12983-A (Non-proprietary), June 1991.

11.   Letter from Liparulo, N. J. (Westinghouse) to Jones, R. C. (NRC), "Process Improvement to the Westinghouse Neutronics Code System," NTD-NRC-96-4679, March 29, 1996.

Table 14.0-1 SUMMARY OF INITIAL CONDITIONS AND COMPUTER CODES USED

Page 1 of 2

Event	Computer Codes Used	DNB Correlation	RTDP	Initial Power, %	Vessel Coolant Flow (gpm)	Vessel Average Coolant Temp.(°F)	RCS Pressure (psia)
Uncontrolled Rod Withdrawal from Subcritical	TWINKLE FACTRAN VIPRE	W-3[1] WRB-1[2]	No	0 (1,800 MWt - Core power)	79,922 [3]	547	2,200
Uncontrolled Rod Withdrawl at Power - Minimum DNBR Cases	RETRAN VIPRE	WRB-1	Yes	100, 60, 10 (1,806 MWt - NSSS power)	186,000	578.4 (100%) 566.4 (60%) 551.4 (10%)	2,250
Uncontrolled Rod Withdrawl at Power - Peak RCS Pressure Cases	RETRAN	N/A	No	100.6, 70, 55, 50, 45, 40, 35, 25, 8 (1,806 MWt - NSSS power)	178,000	583.4 (100.6%) 574.4 (70%) 569.9 (55%) 568.4 (50%) 566.9 (45%) 565.4 (40%) 563.9 (35%) 560.9 (25%) 555.8 (8%)	2,200
RCCA Drop	LOFTRAN[4] ANC VIPRE	WRB-1	Yes	100 (1,800 MWt - Core power)	186,000	577.0	2,250
Chemical and Volume Control System Malfunction	N/A	N/A	N/A	100 (MODE 1) 5 (MODE 2) 0 (MODES 5 and 6) (1,800 MWt - Core power)	N/A	583.4 (MODE 1) 554.9 (MODE 2) 200.0 (MODE 5) 140.0 (MODE 6)	2,250 (MODE 1) 2,250 (MODE 2) 14.7 (MODES 5 and 6)
Startup of an Inactive Reactor Coolant Loop				See FSAR 14.1.5			
Reduction in Feedwater Enthalpy Incident				Bounded by Excessive Load Increase Incident			
Excessive Load Increase Incident	RETRAN	WRB-1	Yes	100 (1,806 MWt - NSSS power)	186,000	578.4	2,250
Loss of Reactor Coolant Flow - All Cases	RETRAN SPNOVA VIPRE	WRB-1	Yes	100 (1,806 MWt - NSSS power)	186,000	578.4	2,250

Table 14.0-1 SUMMARY OF INITIAL CONDITIONS AND COMPUTER CODES USED

Page 2 of 2

Event	Computer Codes Used	DNB Correlation	RTDP	Initial Power, %	Vessel Coolant Flow (gpm)	Vessel Average Coolant Temp.(°F)	RCS Pressure (psia)
Locked Rotor - DNB Case	RETRAN SPNOVA VIPRE	WRB-1	Yes	100 (1,806 MWt - NSSS power)	186,000	578.4	2,250
Locked Rotor - Peak RCS Pressure Case	RETRAN SPNOVA VIPRE	N/A	No	100.6 (1,806 MWt - NSSS power)	178,000	583.4	2,300
Loss of External Elecrtical Load - Minimum DNBR Case	RETRAN	WRB-1	Yes	100 (1,806 MWt - NSSS power)	186,000	578.4	2,250
Loss of External Electrical Load - Peak RCS Pressure Case	RETRAN	N/A	No	100.6 (1,806 MWt - NSSS power)	178,000	577.0 (Unit 1)[5] 583.4 (Unit 2)[5]	2,200
Loss of External Electrical Load - Peak MSS Pressure Case	RETRAN	N/A	No	100.6 (1,806 MWt - NSSS power)	178,000	583.4	2,200
Loss of Normal Feedwater	RETRAN	N/A	No	100.6 (1,806 MWt - NSSS power)	178,000	570.6	2,300
Loss of All AC Power to Station Auxiliaries	RETRAN	N/A	No	100.6 (1,806 MWt - NSSS power)	178,000	570.6	2,300 (Unit 1) 2,200 (Unit 2)
Steam System Piping Failure - Zero Power (Core response only)	RETRAN ANC VIPRE	W-3	No	0 (1,806 MWt - NSSS power)	178,000	547.0	2,250
Steam System Piping Failure - Full Power (Core response only)	RETRAN ANC VIPRE	WRB-1	Yes	100 (1,806 MWt - NSSS power)	186,000	578.4	2,250
Rupture of a Control Rod Mechanism Housing (RCCA Ejection)	TWINKLE FACTRAN	N/A	No	102 (HFP) 0 (HZP) (1,800 MWt - Core power)	178,000 (HFP) 79,922 [3] (HZP)	583.4 (HFP) 547.0 (HZP)	2,200

Notes:
(1) Below the first mixing vane grid.
(2) Above the first mixing vane grid.
(3) Flow from one loop = 0.449* TDF.
(4) The LOFTRAN portion of the analysis was generic; the DNB evaluation performed with VIPRE utilized the plant-specific values presented.
(5) Unit specific values are based on sensitivity studies performed to address issues related to initial vessel average coolant temperature for this event.

Table 14.0-2 NOMINAL VALUES OF PERTINENT PLANT PARAMETERS FOR
NON-LOCA ACCIDENT ANALYSES

Parameter	Max T-avg With RTDP	Max T-avg non-RTDP	Min T-avg With RTDP	Min T-avg non-RTDP
Thermal Output of NSSS (MWt)	1806	1806	1806	1806
Maximum Core Power (MWt)	1800	1800	1800	1800
Vessel Coolant Average Temperature ($^{\circ}$F) [1]	577.0	577.0±6.4	558.0	558.0±6.4
Reactor Coolant System Pressure (psia)	2250	2250±50	2250	2250±50
Reactor Coolant Flow Per Loop (gpm)	93000	89000	93000	89000
Steam Generator Tube Plugging	0 to 10%	0 to 10%	0 to 10%	0 to 10%
Steam Generator Outlet Pressure (psia)	755 (0% SGTP) 727 (10% SGTP)	755 (0% SGTP) 727 (10% SGTP)	626 (0% SGTP) 601 (10% SGTP)	626 (0% SGTP) 601 (10% SGTP)
Assumed Feedwater Temperature at Steam Generator Inlet ($^{\circ}$F)	390.0/458.0	390.0/458.0	390.0/458.0	390.0/458.0
Average Core Heat Flux [2] (BTU/hr-ft^2)	209848	209848	209848	209848

_____

[1]   Accident analyses support a range of full-power T-avg from 558.0°F to 577.0°F.

[2]   Average Core Heat Flux = (1800 MWt * 0.974 * 156.401E6)/(121 * 179* 0.422 * 143.25/1.002), where, 1800 MWt is core power, 0.974 is the fraction of heat generated in the pellet, 156.401E6 is a conversion factor, 121 is the number of fuel assemblies, 179 is the number of rods per fuel assembly, 0.422 is the clad diameter in inches, 143.25 is the active fuel length in inches, and 1.002 is the fuel densification factor.

Table 14.0-3 TRIP POINTS AND TIME DELAYS TO TRIP ASSUMED IN ACCIDENT
ANALYSES

Trip Function	Limiting Trip Point Assumed in Analysis for 2250 psia Oper.	Time Delay (seconds)
Power range high neutron flux, high setting	116%	0.5
Power range high neutron flux, low setting	35%	0.5
Overtemperature DT	Variable see Figure 14.0-1	5.0[1]
Overpower DT	Variable see Figure 14.0-1	6.0[1]
High pressurizer pressure	2418 psia	1.0
Low pressurizer pressure	1855 psia	2.0
Low reactor coolant flow (from loop flow detectors)	87% loop flow	1.0
Turbine trip	N/A	2.0
Low-low steam generator level (% of level span)	20% of narrow range	2.0

[1]   Total time delay (including RTD bypass loop fluid transport delay effect, bypass loop piping thermal capacity, RTD time response, and trip circuit, channel electronics delay) from the time the temperature difference in the coolant loops exceeds the trip setpoint until the rods are free to fall.

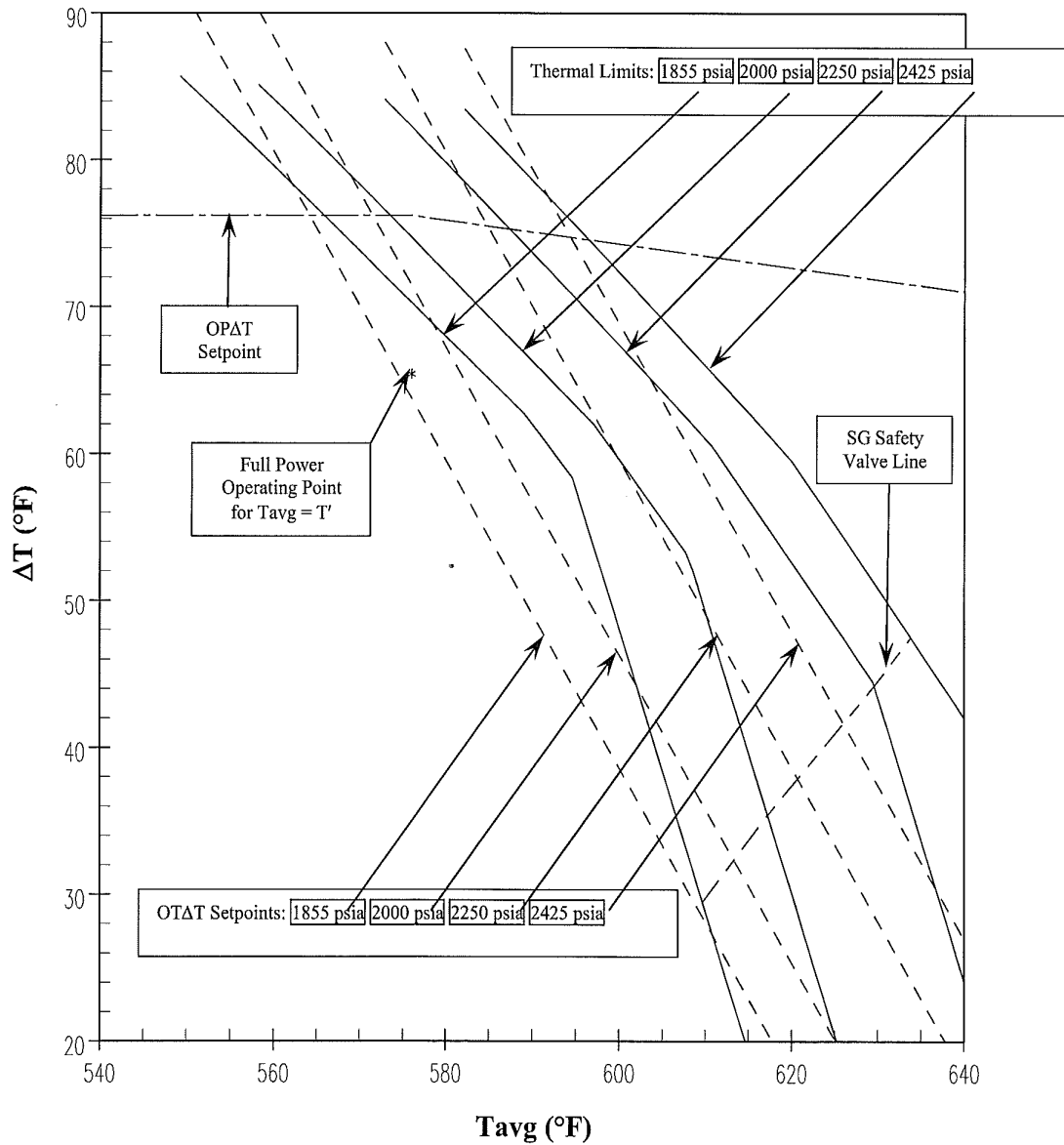Figure 14.0-1 ILLUSTRATION OF OVERTEMPERATURE AND OVERPOWER
DELTA-T PROTECTION

Figure 14.0-2 RCCA NORMALIZED ROD POSITION VS. TIME CURVE
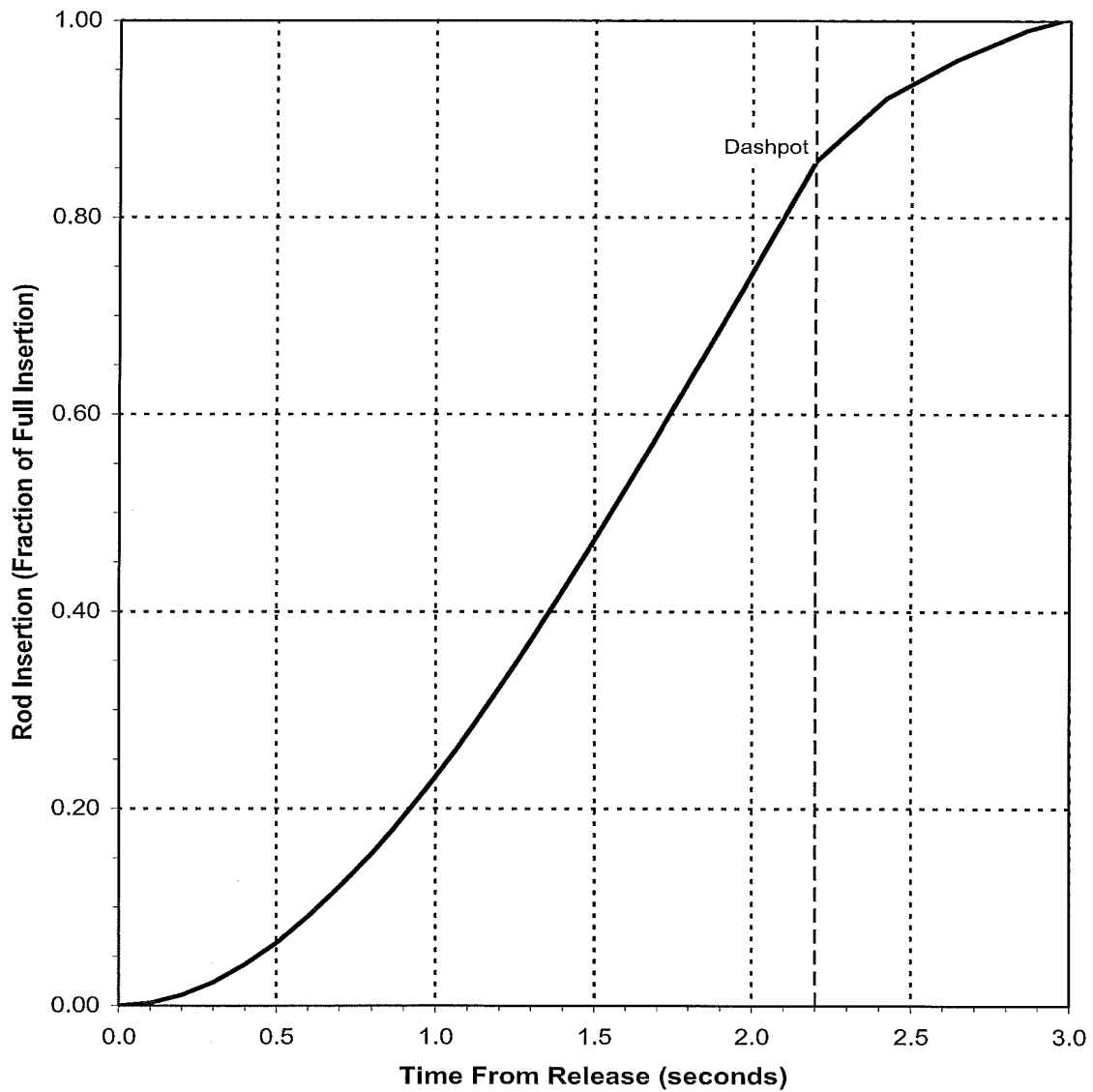
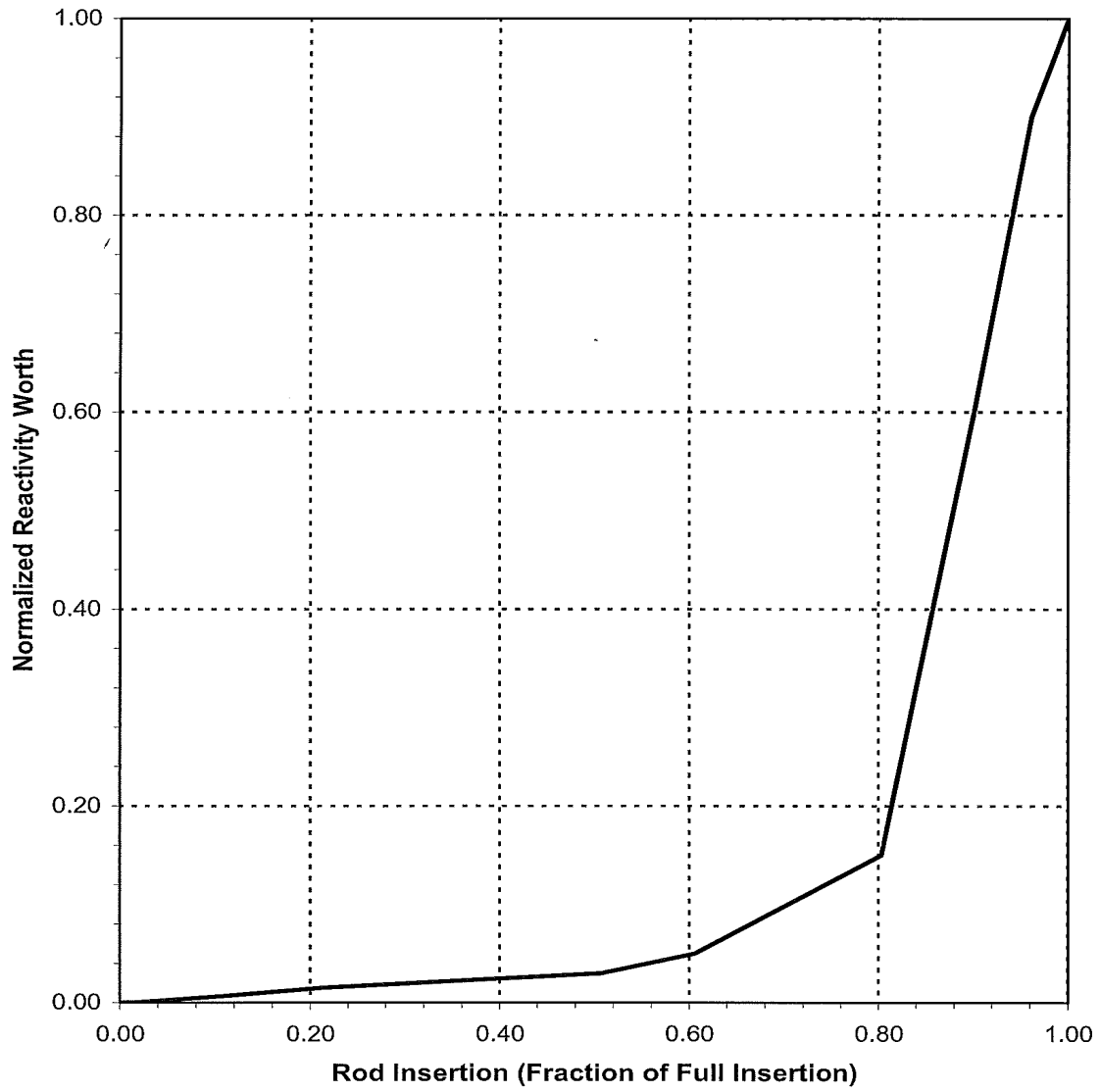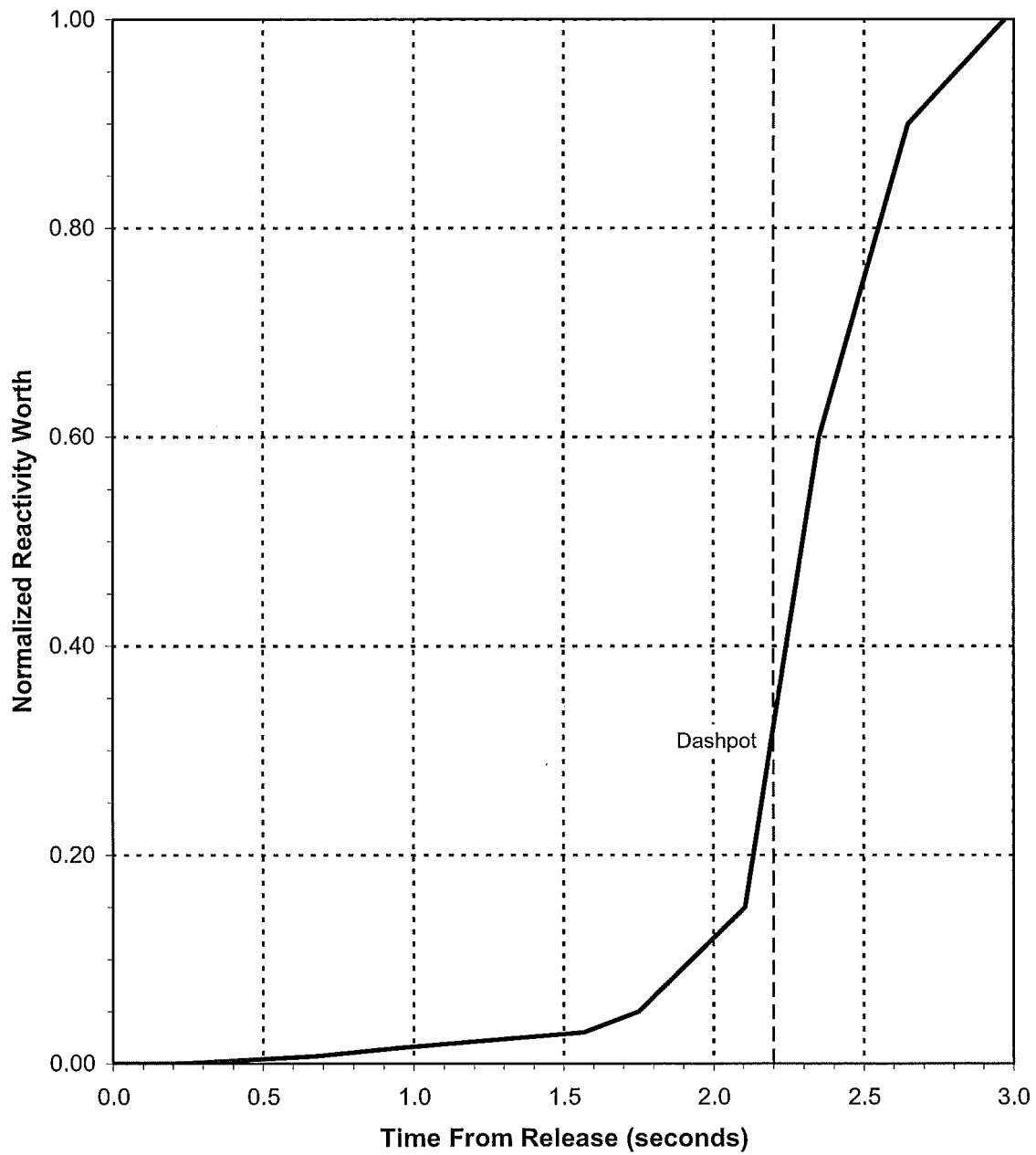Figure 14.0-3 NORMALIZED REACTIVITY VS ROD POSITION

Figure 14.0-4 NORMALIZED TRIP REACTIVITY VS TIME

14.1   CORE AND COOLANT BOUNDARY PROTECTION ANALYSIS

14.1.1   UNCONTROLLED ROD WITHDRAWAL FROM SUBCRITICAL

An RCCA withdrawal incident is defined as an uncontrolled addition of reactivity to the reactor core caused by withdrawal of RCCAs 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, at hot zero power or at power. The "at power" case is discussed in Section 14.1.2. Although the reactor is normally brought to power from a subcritical condition by means of RCCA withdrawal, procedures for the initial startup following refueling call for boron dilution. The maximum rate of reactivity increase in the case of boron dilution is less than that assumed in this analysis (Section 14.1.4).

The rod cluster drive mechanisms are wired into preselected banks, and these bank configurations are not altered during core life. The rods are therefore physically prevented from withdrawing in other than their respective banks. Power supplied to the rod banks is controlled such that no more than two banks can be withdrawn at any time. Additionally, with the Bank Selector Switch in either the Automatic (AUTO) or Manual (MAN) position, the banks can be withdrawn only in their proper withdrawal sequence. The rod drive mechanism is of the magnetic latch type and the coil actuation is sequenced to provide variable speed rod travel.   The maximum reactivity insertion rate is analyzed in the detailed plant analysis assuming the simultaneous withdrawal of the combination of the two control banks with the maximum combined worth at maximum speed.

The neutron flux response to a continuous reactivity insertion 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 an acceptable 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 flux level above a preselected manually adjustable setpoint. This trip function may be manually blocked 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 level. This trip function may be manually blocked only after two out of four power range channels are reading above approximately 10 percent of full power and is automatically reinstated when three 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 blocked 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.

      In addition, control rod stops on high intermediate range flux level (one of two) and high
      power range flux level (one 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.

Method of Analysis

The analysis of the uncontrolled RCCA bank withdrawal from subcritical accident is performed
in three stages: first an average core nuclear power transient calculation, then an average core heat
transfer calculation, and finally the DNBR calculation. The average nuclear power transient with
respect to time calculation is performed using a spatial neutron kinetics code, TWINKLE, which
includes the various total core feedback effects, i.e., Doppler and moderator reactivity. The
FACTRAN code is then used to calculate the thermal heat flux transient, based on the nuclear
power transient calculated by TWINKLE. FACTRAN also calculates the fuel and cladding
temperatures. The average heat flux is next used in VIPRE, Reference 43 and Reference 44,
(Section 3.2) for transient DNBR calculation.

Plant characteristics and initial conditions are discussed in Section 14.0. In order to give
conservative results for a startup accident, the following assumptions are made.

1.   Since the magnitude of the nuclear power peak reached during the initial part of the
      transient for any given rate of reactivity insertion is strongly dependent on the Doppler-only
      power defect, conservatively low (lowest absolute magnitude) values are used.

2.   Contribution of the moderator reactivity coefficient is negligible during the initial part of
      the transient because the heat transfer time between the fuel and the moderator is much
      longer than the nuclear flux response time. However, after the initial nuclear flux peak, the
      succeeding rate of power increase is affected by the moderator reactivity coefficient. The
      most positive value of the moderator temperature coefficient 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
      larger fuel-water heat transfer coefficient, larger specific heats, and a less negative (smaller
      absolute magnitude) Doppler coefficient, all of which tend to reduce the Doppler feedback
      effect thereby increasing the neutron flux peak. The initial effective multiplication factor is
      assumed to be 1.0 since this results in maximum neutron flux peaking and, thus, the most
      severe nuclear power transient.

4.   Reactor trip is assumed to be initiated by power range 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.

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 inches/minute).

6.   The most limiting axial and radial power shapes, associated with having the two highest combined worth sequential banks in their highest worth position, are assumed for DNB analysis.

7.   The initial power level was assumed to be below the power level expected for any shutdown condition (10^{-9} of nominal power). The combination of highest reactivity insertion rate and lowest initial power produces the highest peak heat flux.

8.   One reactor coolant pump is assumed to be in operation. This lowest initial flow minimizes the resulting DNBR.

9.   The RCS pressure is 50 psi below nominal pressure.

Results

Figure 14.1.1-1 through Figure 14.1.1-3 show the transient behavior for the uncontrolled RCCA bank withdrawal with the accident terminated by reactor trip at 35 percent nominal power. The reactivity insertion rate used is greater than that calculated for the two highest worth sequential control banks, both assumed to be in their highest incremental worth region. Figure 14.1.1-1 shows the neutron flux transient.

The energy release and the fuel temperature increases are relatively small. The thermal flux response, of interest for departure from nucleate boiling considerations, is shown on Figure 14.1.1-2. The beneficial effect of the inherent thermal lag in the fuel is evidenced by a peak heat flux less than the full-power nominal value. The minimum DNBR at all times remains above the safety analysis limit value and there is a high degree of subcooling at all times in the core. Figure 14.1.1-3 shows the response of the hot spot average fuel and cladding temperature. The average fuel temperature increases to a value lower than the nominal full-power value.

The calculated sequence of events for this accident is shown in Table 14.1.1-1. With the reactor tripped, the plant returns to a stable condition. The plant may subsequently be cooled down further by following normal plant shutdown procedures.

Conclusion

In the event of a RCCA withdrawal accident from the subcritical condition, the core and the reactor coolant system are not adversely affected. The minimum departure from nucleate boiling ratio remains above the safety analysis limit value and thus, no fuel or clad damage is predicted.

Reference

1.   NRC Safety Evaluation 2011-0004, "Issuance of License Amendments Regarding Extended Power Uprate," dated May 3, 2011.

Table 14.1.1-1 TIME SEQUENCE OF EVENTS FOR UNCONTROLLED RCCA
WITHDRAWAL FROM A SUBCRITICAL CONDITION

Event	Time of Each Event (Seconds)
Initiation of uncontrolled rod withdrawal from 10^{-9} of nominal power	0
Power range high neutron flux low setpoint reached	10.0
Peak nuclear power occurs	10.11
Rods begin to fall into core	10.48
Peak heat flux occurs	11.93
Minimum DNBR occurs	11.93
Peak cladding temperature occurs	12.23
Peak average fuel temperature occurs	12.43

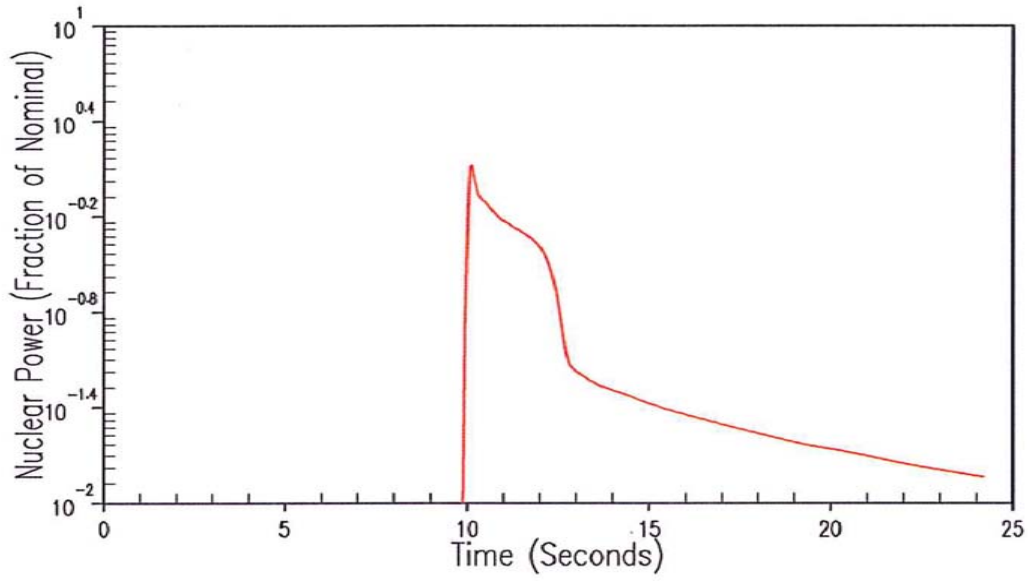Figure 14.1.1-1 UNCONTROLLED RCCA BANK WITHDRAWAL FROM SUBCRITICAL
NUCLEAR POWER TRANSIENT

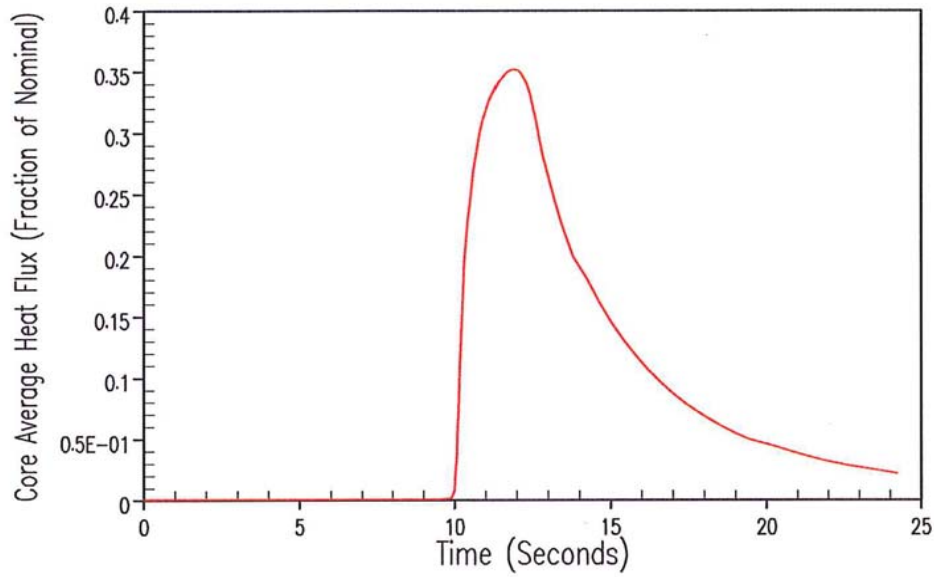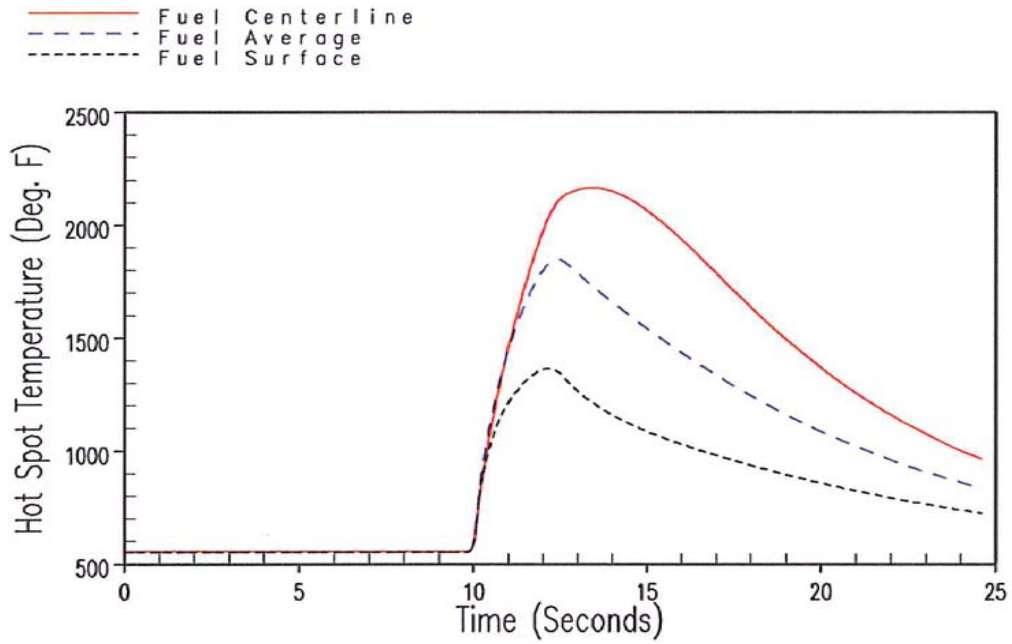Figure 14.1.1-2 UNCONTROLLED RCCA BANK WITHDRAWAL FROM SUBCRITICAL HEAT FLUX TRANSIENT

Figure 14.1.1-3 UNCONTROLLED RCCA BANK WITHDRAWAL FROM SUBCRITICAL
FUEL TEMPERATURE TRANSIENT

14.1.2 UNCONTROLLED ROD WITHDRAWAL AT POWER

An uncontrolled RCCA withdrawal at power results in an increase in core heat flux. Since the heat extraction from the steam generator remains constant, there is a net increase in reactor coolant temperature. Unless terminated by manual or automatic action, this power mismatch and resultant coolant temperature rise would eventually result in DNB. Therefore, to prevent the possibility of damage to the cladding, the Reactor Protection System is designed to terminate any such transient with an adequate margin to DNB.

The automatic features of the Reactor Protection System which prevent core damage in a rod withdrawal accident at power include the following:

1.   Nuclear power range instrumentation actuates a reactor trip if two out of the four channels exceed an overpower setpoint.

2.   Reactor trip is actuated if any two out of four ΔT channels exceed an overtemperature ΔT setpoint. This setpoint is automatically varied with power distribution, temperature and pressure to protect against DNB.

3.   Reactor trip is actuated if any two out of four ΔT channels exceed an overpower ΔT setpoint. This setpoint is automatically varied with temperature to ensure that the allowable full power rating is not exceeded.

4.   A high pressure reactor trip, actuated from any two out of three pressure channels, is set at a fixed point. This set pressure will be 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, is actuated at a fixed setpoint. This affords additional protection for RCCA withdrawal accidents.

The manner in which the combination of overpower and overtemperature ΔT trips provide protection over the full range of reactivity insertion rates is illustrated in Section 14.0. Figure 14.0-1 represents the possible conditions of reactor vessel average temperature and ΔT with the design power distribution in a two-dimensional plot. The boundaries of operation defined by the overpower ΔT trip and the overtemperature ΔT trip are represented as "protection lines" on this diagram. These protection lines are drawn to include all adverse instrumentation and setpoint errors, so that under nominal conditions trip would occur well within the area bounded by these lines. A maximum steady state operating condition for the reactor is also shown on the figure.

The utility of the diagram just described is in the fact that the operating limit imposed by any given DNB ratio can be represented as a line on this coordinate system. The DNB lines represent the locus of conditions for which the DNBR equals the safety analysis limit value. All points below and to the left of this line have a DNB ratio greater than this value. The diagram shows that DNB is prevented for all cases if the area enclosed within the maximum protection lines is not traversed by the applicable DNB ratio line at any point.

The region of permissible operation (power, pressure and temperature) is completely bounded by the combination of reactor trips: nuclear overpower (fixed setpoint); high pressure (fixed setpoint); low pressure (fixed setpoint); overpower and overtemperature ΔT (variable setpoints). These trips are designed to prevent overpower and a DNB ratio of less than the limit value.

Method of Analysis

Uncontrolled rod cluster control assembly bank withdrawal is analyzed by the RETRAN code. This code simulates the neutron kinetics, reactor coolant system, pressurizer, pressurizer relief and safety valves, pressurizer spray, steam generator, and steam generator safety valves. The code computes pertinent plant variables, including temperatures, pressures, and power level. The core limits, as illustrated in Figure 14.0-1, are used as input to RETRAN to determine the minimum departure from nucleate boiling ratio during the transient. Although RETRAN has the capability of conservatively approximating the transient value of the DNBR, a detailed DNB analysis was performed for the limiting cases with the VIPRE thermal-hydraulic computer code. This accident is analyzed with the Revised Thermal Design Procedure as described in Reference 22, Section 3.2. Plant characteristics and initial conditions are discussed in Section 14.0.

In order to obtain conservative values of departure from nucleate boiling ratio, the following assumptions are made:

1. Initial Conditions - Cases are analyzed at three initial power levels (100%, 60%, and 10%). Both minimum and maximum nominal RCS average temperature are analyzed at a power level of 100% with minimum reactivity feedback. Uncertainties in the initial conditions are included in the limit DNBR as described in Reference 22, of Section 3.2.

2. Reactivity Coefficients - Two cases are analyzed.

    a. Minimum Reactivity Feedback - A positive (5 pcm/°F) moderator coefficient of reactivity is assumed, corresponding to the beginning of core life. A variable Doppler power coefficient with core power is used in the analysis. A conservatively small (in absolute magnitude) value is assumed.

    b. Maximum Reactivity Feedback - A conservatively large positive moderator density coefficient and a large (in absolute magnitude) negative Doppler power coefficient are assumed.

3. The rod cluster control assembly trip insertion characteristic is based on the assumption that the highest worth assembly is stuck in its fully withdrawn position.

4. The reactor trip on high neutron flux is assumed to be actuated at a conservative value of 116% of nominal full power. The overtemperature ΔT trip includes all adverse instrumentation and setpoint errors; the delays for trip actuation are assumed to be the maximum values. The high pressurizer pressure reactor trip was credited to examine the effect of increasing the full-open areas of the pressurizer relief and safety valves; the valve areas were increased to correct a modeling error discovered with the licensing basis analysis. Crediting the high pressurizer pressure reactor trip resulted in showing that the licensing basis analysis remains bounding. No credit was taken for the other expected trip functions.

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

The uncontrolled RCCA bank withdrawl at-power accident was also analyzed to ensure that the RCS and MS peak pressures did not exceed 110% of the respective design pressures. Reactivity insertion rates at various power levels were analyzed. These cases were initiated from conditions that include uncertainties on power, RCS pressure, and RCS temperature.

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

Results

Figures shown are for Unit 1. Unit 2 is similar, but in the analysis Unit 1 is slightly more limiting.

Figure 14.1.2-1 shows the response of neutron flux, DNBR, pressurizer pressure, pressurizer water volume, and vessel T-avg to a rapid rod cluster control assembly withdrawal incident starting from full power. Reactor trip on high neutron flux occurs shortly after the start of the accident. Since this is rapid with respect to the thermal time constants of the plant, small changes in T-avg and pressure result, and a large margin to DNB is maintained.

The response of neutron flux, DNBR, pressurizer pressure, pressurizer water volume, and vessel T-avg for a slow control rod withdrawal from 100% power is shown in Figure 14.1.2-2. Reactor trip on overtemperature ΔT occurs after a longer period, and the rise in temperature and pressure is consequently larger than for rapid rod cluster control assembly withdrawal. Again, the minimum DNBR is greater than the limit value.

Figure 14.1.2-3 shows the minimum departure from nucleate boiling ratio as a function of the reactivity insertion rate for the three initial power levels (100%, 60%, and 10%), minimum and maximum reactivity feedback. It can be seen that the high neutron flux (HNF) and the overtemperature ΔT trip channels provide protection over the whole range of reactivity insertion rates. As previously indicated, the high pressurizer pressure reactor trip was required to be credited to show that these results remain bounding when considering the effect of increased full-open areas of the pressurizer relief and safety valves. For the cases that violated the safety analysis DNBR limit using the conservative RETRAN DNBR approximation model (Figure 14.1.2-3 Sh. 3), the DNBR response was recalculated using the detailed thermal-hydraulic computer code VIPRE in order to obtain acceptable results. Thus, in all cases, the DNBR remained above the safety analysis limit.

In the referenced figures, the shape of the curves of minimum departure from nucleate boiling ratio versus reactivity insertion rate is due both to reactor core and coolant system transient response and to protection system action in initiating a reactor trip.

Referring to Figure 14.1.2-3 (sheet 3) for example, it is noted that:

1.   For high reactivity insertion rates (i.e., between ~100 pcm/second and ~20 pcm/second), reactor trip is initiated by the high neutron flux trip. The neutron flux level in the core rises

rapidly for these insertion rates, while core heat flux and coolant system temperature lag behind due to the thermal capacity of the fuel and coolant system fluid. Thus, the reactor is tripped prior to significant increase in heat flux or water temperature with resultant high minimum departure from nucleate boiling ratios during the transient. Within this range, as the reactivity insertion rate decreases, core heat flux and coolant temperatures can remain more nearly in equilibrium with the neutron flux; minimum DNBR during the transient thus decreases with decreasing insertion rate.

2.   With further decrease in reactivity insertion rate, the overtemperature ΔT and high neutron flux trips become equally effective in terminating the transient. The overtemperature ΔT reactor trip circuit initiates a reactor trip when measured coolant trip ΔT exceeds a setpoint based on measured reactor coolant system average temperature and pressure. It is important in this context to note, however, that the average temperature contribution to the circuit is lead-lag compensated in order to decrease the effect of the thermal capacity of the reactor coolant system in response to power increases. It should also be noted that with the increased full-open areas of the pressurizer relief and safety valves, reactor trip can occur slightly earlier or slightly later, depending on the RCS pressurization rate and the pressurizer relief capacity. Delaying reactor trip typically has an adverse impact on the calculated minimum DNBR. Therefore, the high pressurizer pressure reactor trip was credited to show that the current analysis minimum DNBR results for the most limiting cases remain bounding.

For reactivity insertion rates between ~20 pcm/second and ~10 pcm/second, the effectiveness of the overtemperature ΔT trip increases (in terms of increased minimum departure from nucleate boiling ratio) due to the fact that, with lower insertion rates, the power increase rate is slower, the rate of rise of average coolant temperature is slower, and the system lags and delays become less significant.

3.   For reactivity insertion rates less than ~10 pcm/second, the rise in reactor coolant temperature is sufficiently high so the steam generator safety valves relieve a significant amount of steam prior to trip. Opening these valves, which act as an additional heat sink on the reactor coolant system, sharply decreases the rate of rise of reactor coolant system average temperature. This causes the overtemperature ΔT trip setpoint to be reached later with resulting lower minimum departure from nucleate boiling ratios.

The results obtained for the cases that were analyzed to address RCS and MS peak pressure concerns demonstrate that the limits were not exceeded when the maximum permissible insertion rate was conservatively limited to 50 pcm/second.

Conclusions

In the unlikely event of an at power (either from full power or lower power levels) control rod bank withdrawal incident, the core and reactor coolant system are not adversely affected since the minimum value of DNB ratio reached is in excess of the DNB limit value for all rod reactivity rates. Protection is provided by nuclear flux overpower, overtemperature ΔT, and high pressurizer pressure reactor trips. The peak RCS and MS pressures do not exceed 110% of the respective design pressures. Additional protection would be provided by the high pressurizer level and overpower ΔT reactor trips. The preceding sections have described the effectiveness of these protection channels.

References

1. NRC Safety Evaluation 2011-0004, "Issuance of License Amendments Regarding Extended Power Uprate," dated May 3, 2011.

2. Engineering Change 281807 Revision 0, "AR 01943430, Error In Vendor (Westinghouse) Non-LOCA Analysis," Approved June 12, 2014.

Table 14.1.2-1 TIME SEQUENCE OF EVENTS FOR UNCONTROLLED RCCA
WITHDRAWAL AT POWER (maximum nominal RCS Tavg; Minimum Feedback) (These are
Unit 1values; Unit 2 is similar but Unit 1 is slightly more limiting)

Event	Time of Each Event (Sec.)
Case A:	
Initiation of uncontrolled rod cluster control assembly withdrawal at full power and maximum reactivity insertion rate (100 pcm/sec)	0
Power range high neutron flux high trip point reached	1.3
Rods begin to fall into core	1.8
Minimum departure from nucleate boiling ratio occurs	2.0
Case B:	
Initiation of uncontrolled rod cluster control assembly withdrawal at 100% power and at a small reactivity insertion rate (1 pcm/sec)	0
Overtemperature ΔT reactor trip signal initiated	71.1
Rods begin to fall into core	73.1
Minimum departure from nucleate boiling ratio occurs	73.0

Figure 14.1.2-1 ROD WITHDRAWAL AT POWER 100%, MINIMUM FEEDBACK
100 PCM/SECOND

Sheet 1 of 3

Figure 14.1.2-1 ROD WITHDRAWAL AT POWER 100%, MINIMUM FEEDBACK
100 PCM/SECOND

Sheet 2 of 3

Figure 14.1.2-1 ROD WITHDRAWAL AT POWER 100%, MINIMUM FEEDBACK
100 PCM/SECOND

Sheet 3 of 3

Figure 14.1.2-2 ROD WITHDRAWAL AT POWER 100%, MINIMUM FEEDBACK
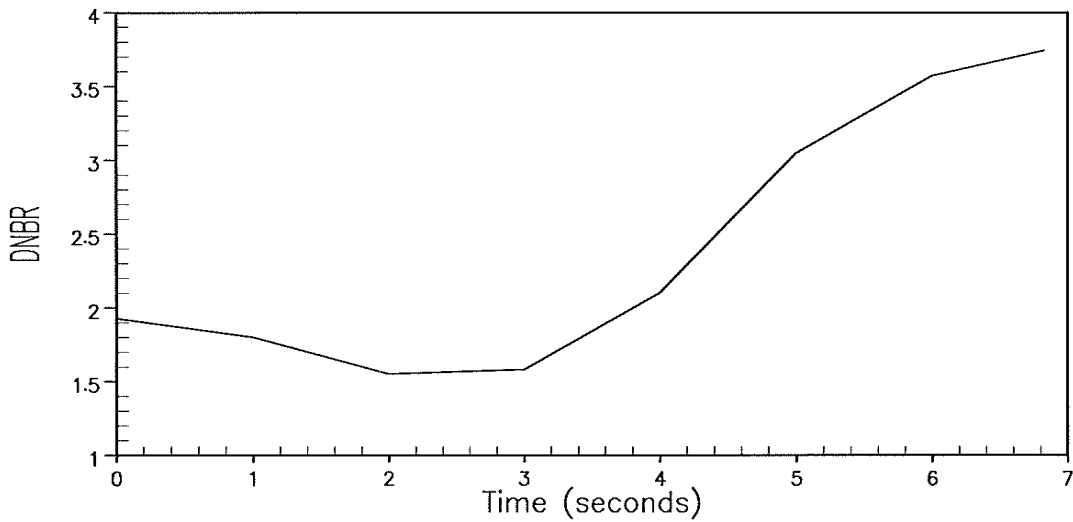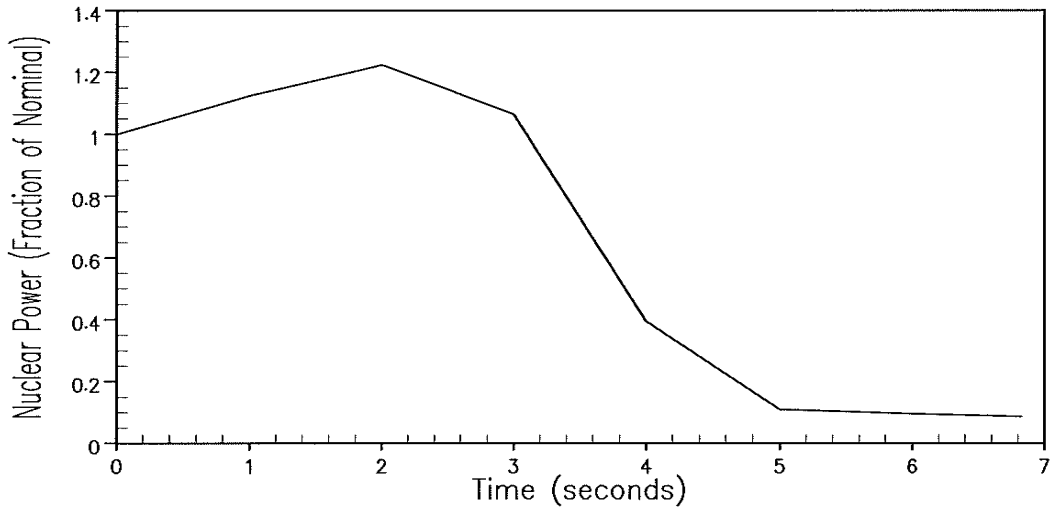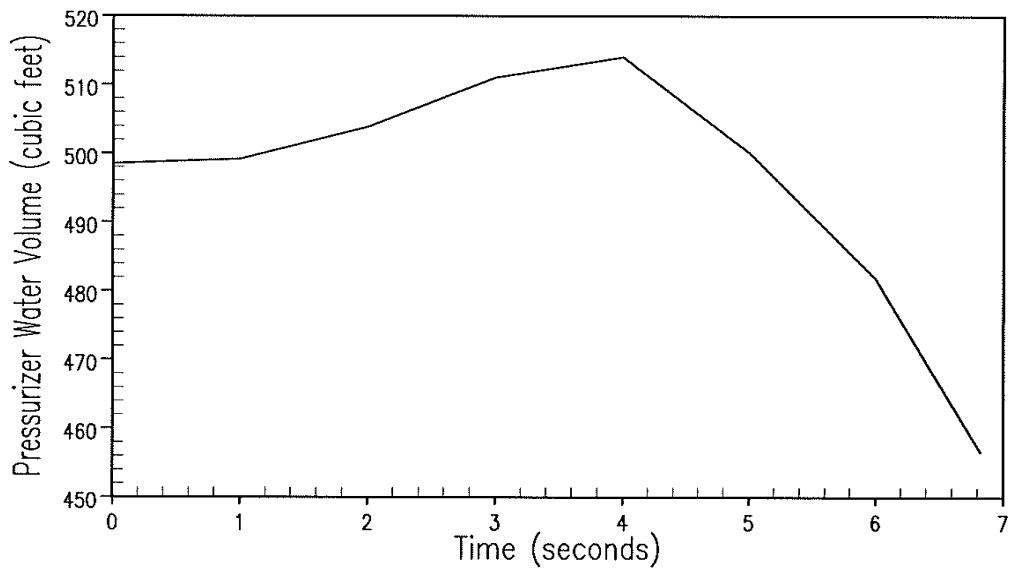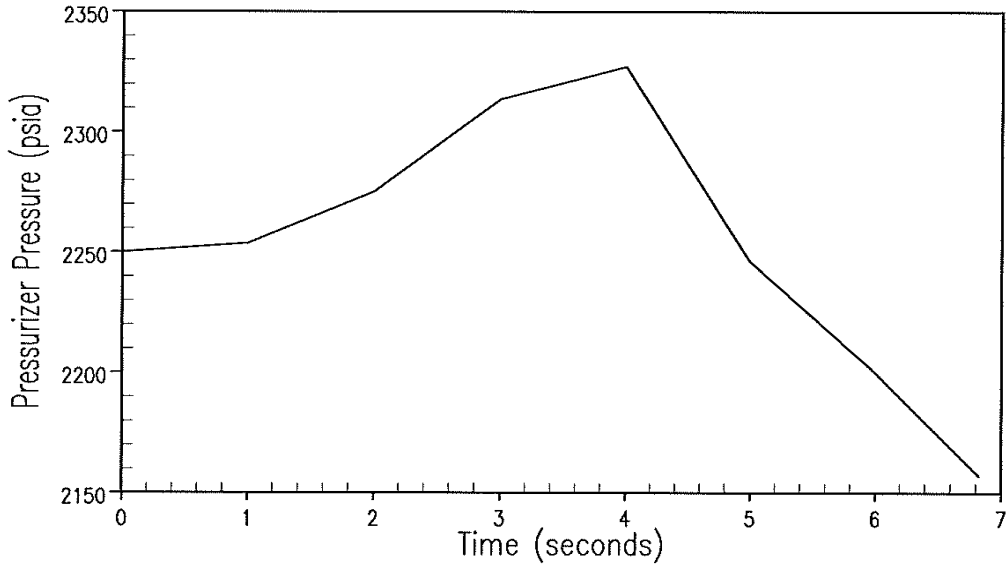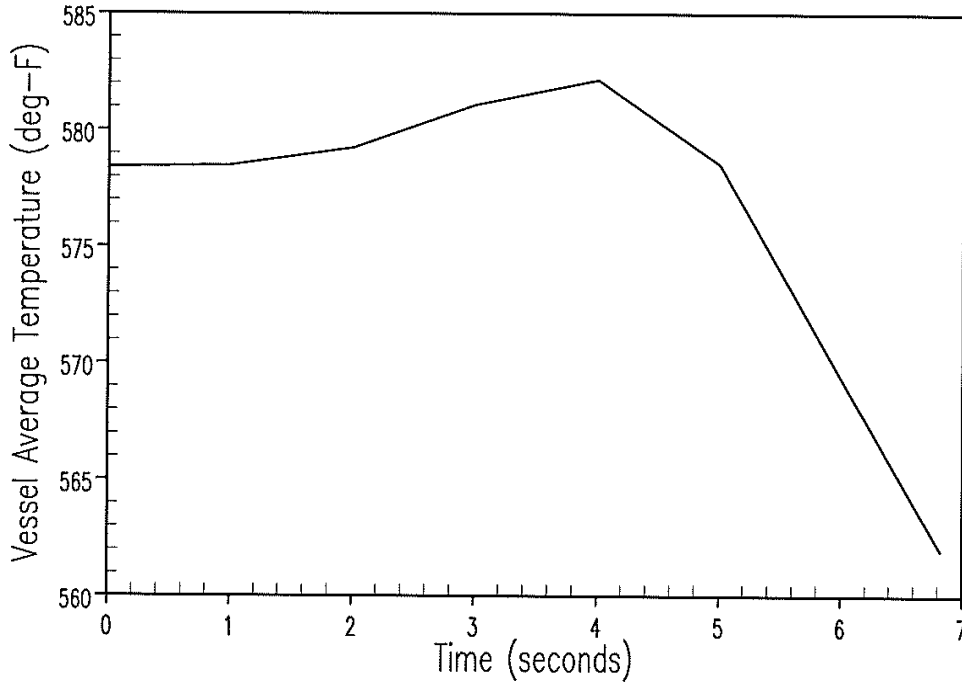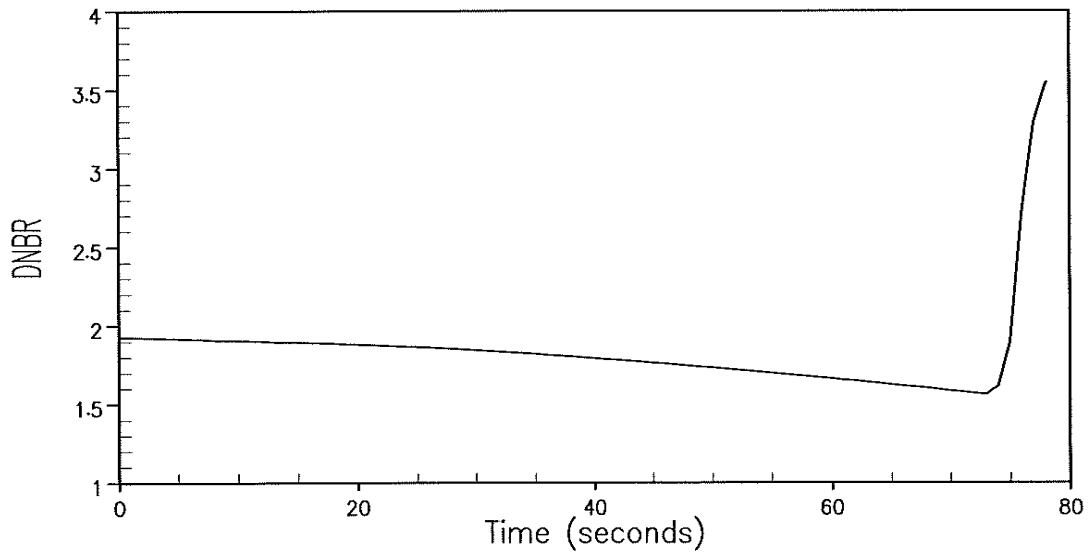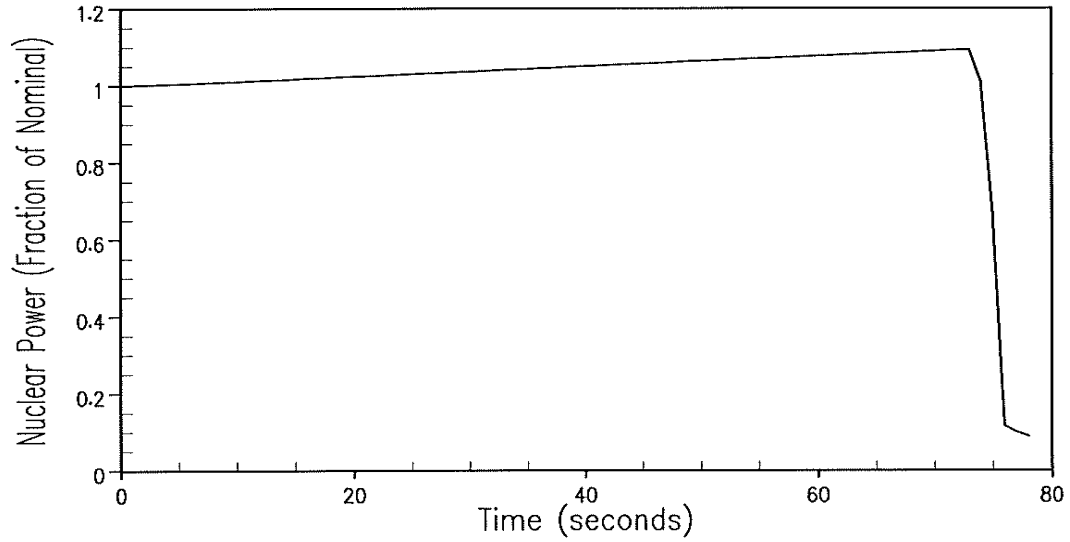1 PCM/SECOND

Sheet 1 of 3

Figure 14.1.2-2 ROD WITHDRAWAL AT POWER 100%, MINIMUM FEEDBACK
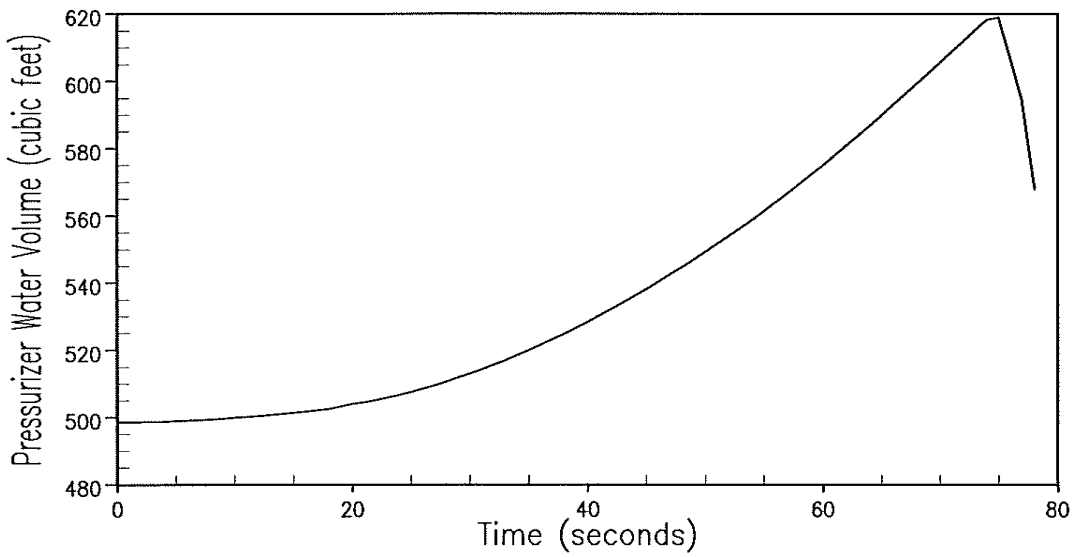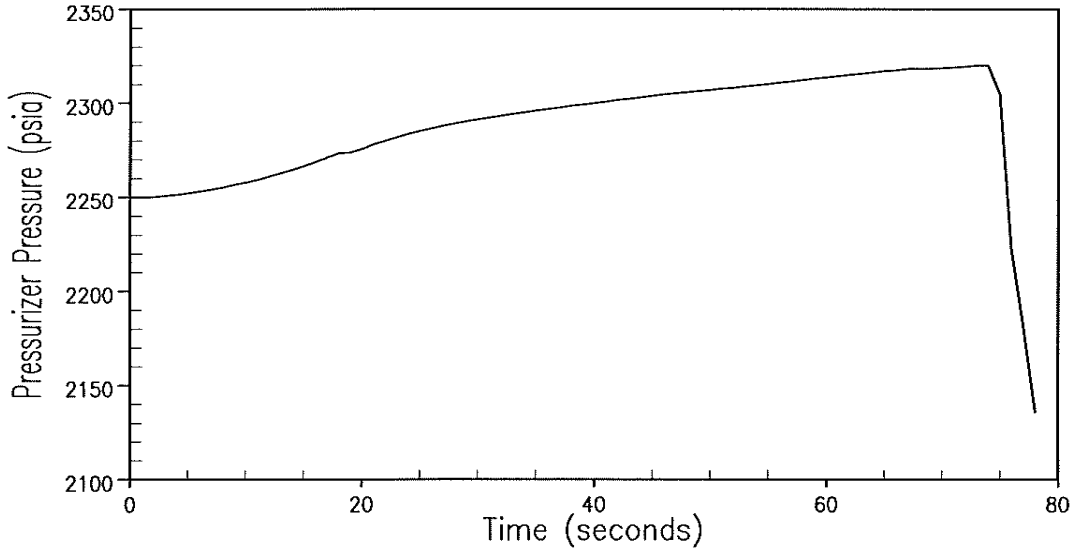1 PCM/SECOND

Sheet 2 of 3

Figure 14.1.2-2 ROD WITHDRAWAL AT POWER 100%, MINIMUM FEEDBACK
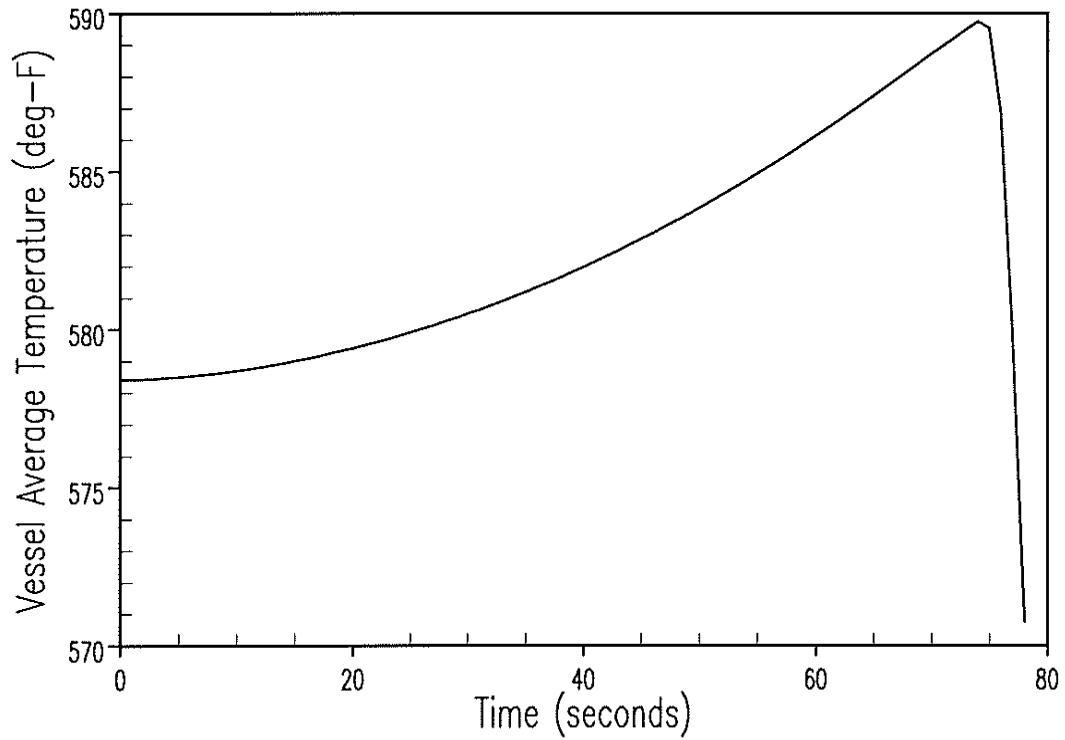   1 PCM/SECOND

Sheet 3 of 3

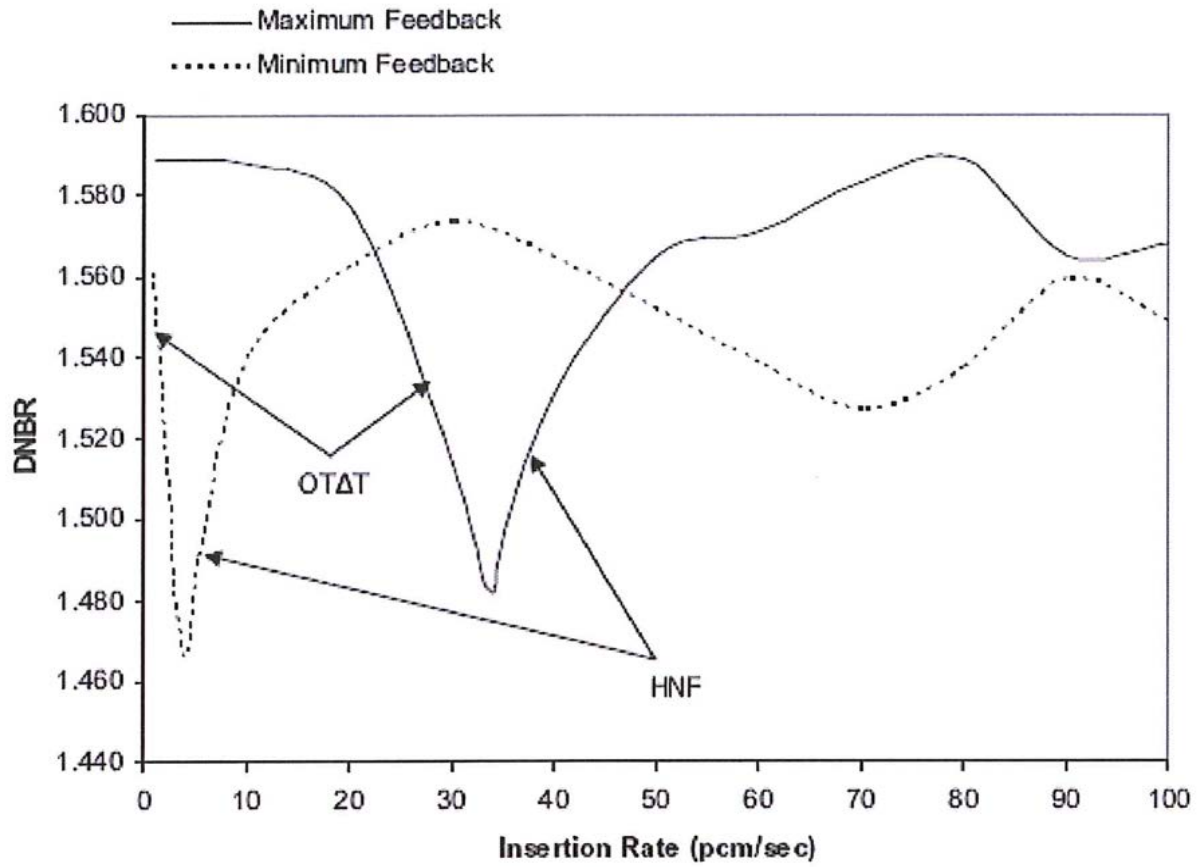Figure 14.1.2-3 ROD WITHDRAWAL AT POWER 100%
Sheet 1 of 3

Figure 14.1.2-3 ROD WITHDRAWAL AT POWER 60%
Sheet 2of 3
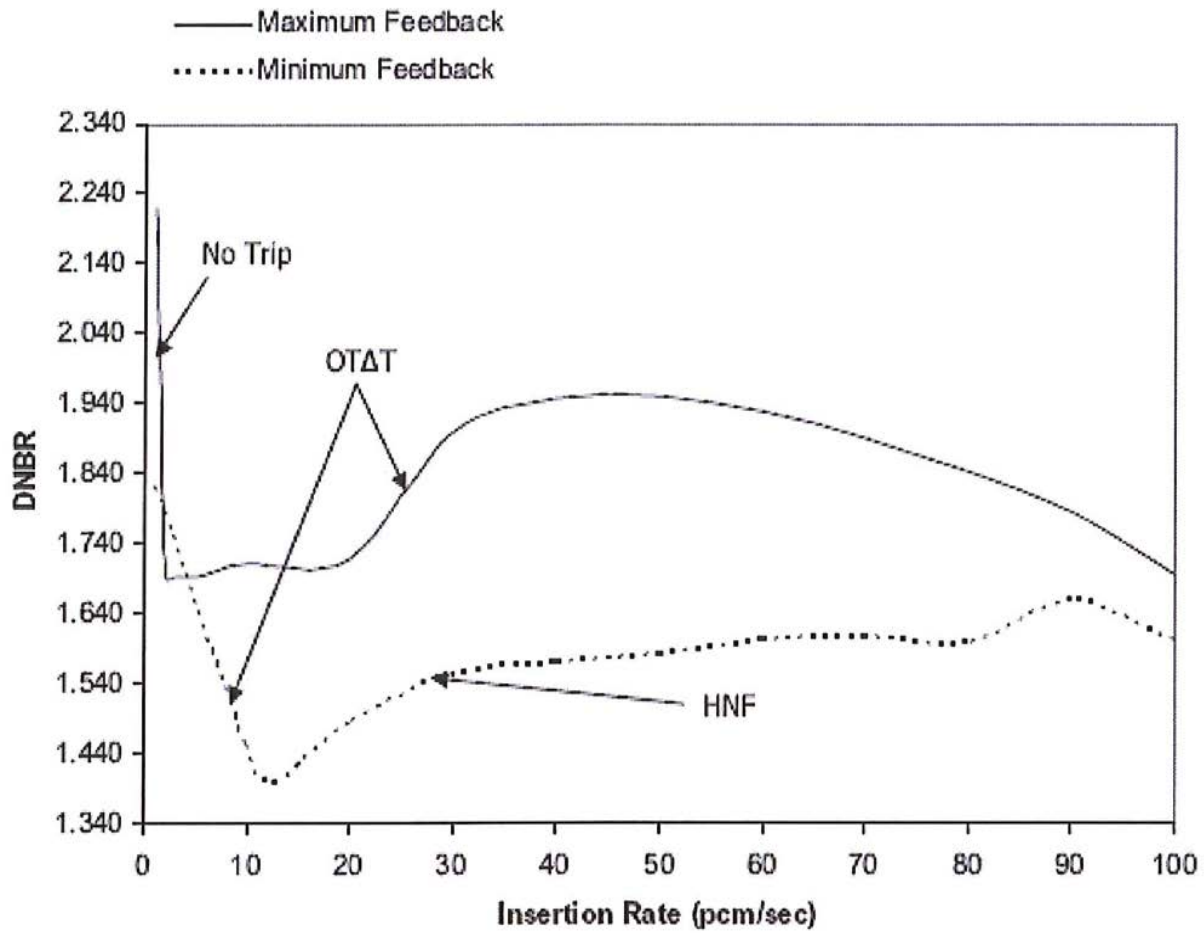
Figure 14.1.2-3 ROD WITHDRAWAL AT POWER 10%
Sheet 3 of 3
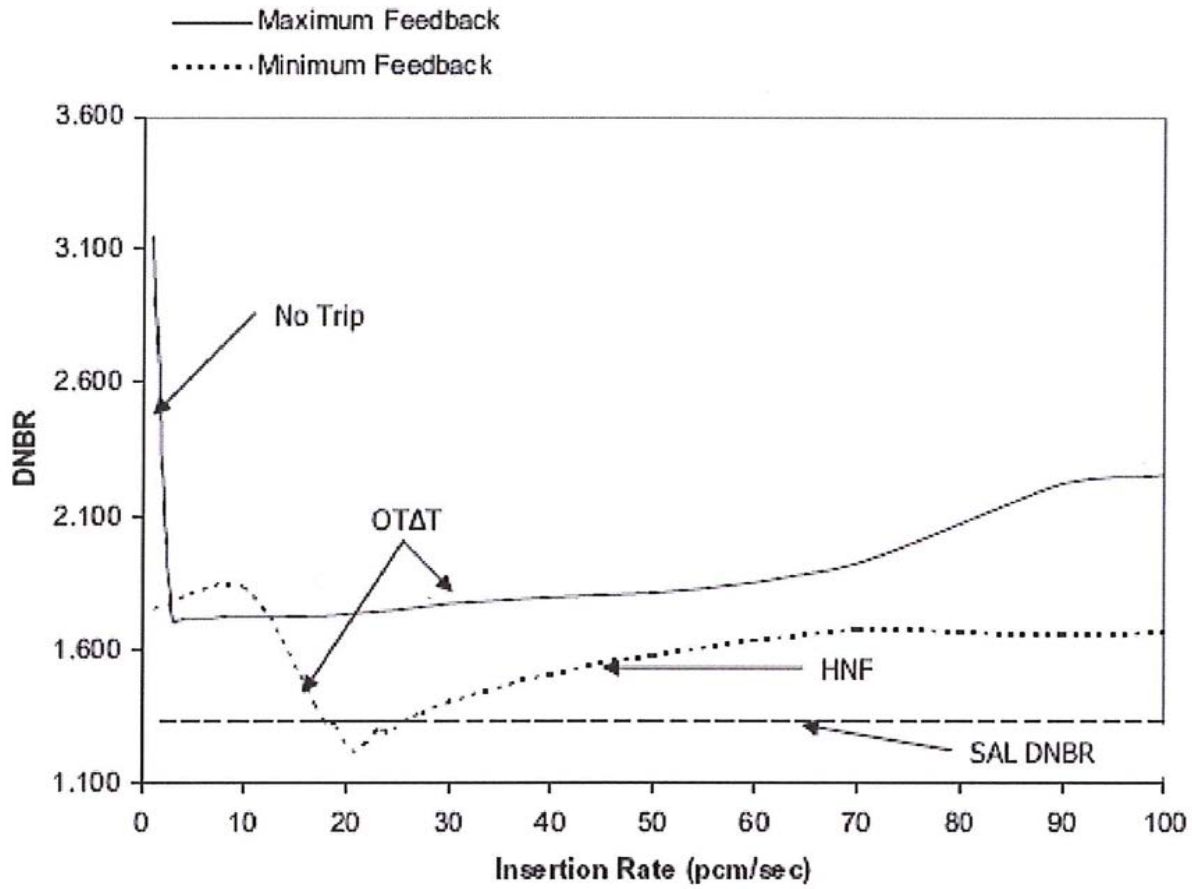
14.1.3   ROD CLUSTER CONTROL ASSEMBLY DROP

Dropping of a full length RCCA occurs when the drive mechanism is deenergized. The dropped RCCA causes a power reduction and an increase in the hot channel factor. The automatic rod control system tries to restore the power to the level which existed before the incident by withdrawing rods. An increased hot channel factor and automatic rod withdrawal may lead to a reduced safety margin depending upon the magnitude of the dropped RCCA worth.

Indication of an RCCA dropping into the core during power operation would be by either a rod bottom signal, by an out of core ion chamber, or both. The rod bottom signal device provides an indication signal for each RCCA. The other independent indication of a dropped RCCA is obtained by using the out of core power range channel signals. The rod drop detection circuit is actuated upon sensing a rapid decrease in local flux and is designed such that normal load variations do not cause it to be actuated.

Method of Analysis

For the evaluation of the dropped RCCA(s), the transient response is calculated using the LOFTRAN code. The code simulates the neutron kinetics, reactor coolant system, pressurizer, pressurizer relief and safety valves, pressurizer spray, steam generators, and steam generator safety valves. The code computes pertinent plant variables including temperatures, pressures and power level.

Statepoints are calculated and nuclear models are used to obtain a hot channel factor consistent with the primary system conditions and reactor power. By incorporating the primary conditions from the transient and the hot channel factor from the nuclear analysis, the DNB design basis is shown to be met using the VIPRE code (Reference 2). The transient response, nuclear peaking factor analysis, and DNB design basis confirmation are performed in accordance with the methodology described in WCAP-11394-A (Reference 1).

Results

For the dropped RCCA event, power may be reestablished either by reactivity feedback or control bank withdrawal.

Following a dropped RCCA(s) 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.

For a dropped RCCA(s) 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. Figure 14.1.3-1 and Figure 14.1.3-2 show a typical transient response to a dropped RCCA(s). Uncertainties in the initial conditions are included in the DNB evaluation as described in Reference 1 In all cases, the minimum DNBR remains greater than the limit value.

Conclusions

For all cases the DNB design is met by demonstrating that the DNBR is greater than the limit value.

References

1. Westinghouse Licensing Topical Report WCAP 11394-P-A (Proprietary), and WCAP 11395-A (Non-proprietary), "Methodology for the Analysis of the Dropped Rod Event," October 23, 1989.

2. NRC Safety Evaluation 2011-0004, "Issuance of License Amendments Regarding Extended Power Uprate," dated May 3, 2011.

Figure 14.1.3-1 NUCLEAR POWER TRANSIENT AND CORE HEAT FLUX TRANSIENT
FOR DROPPED RCCA
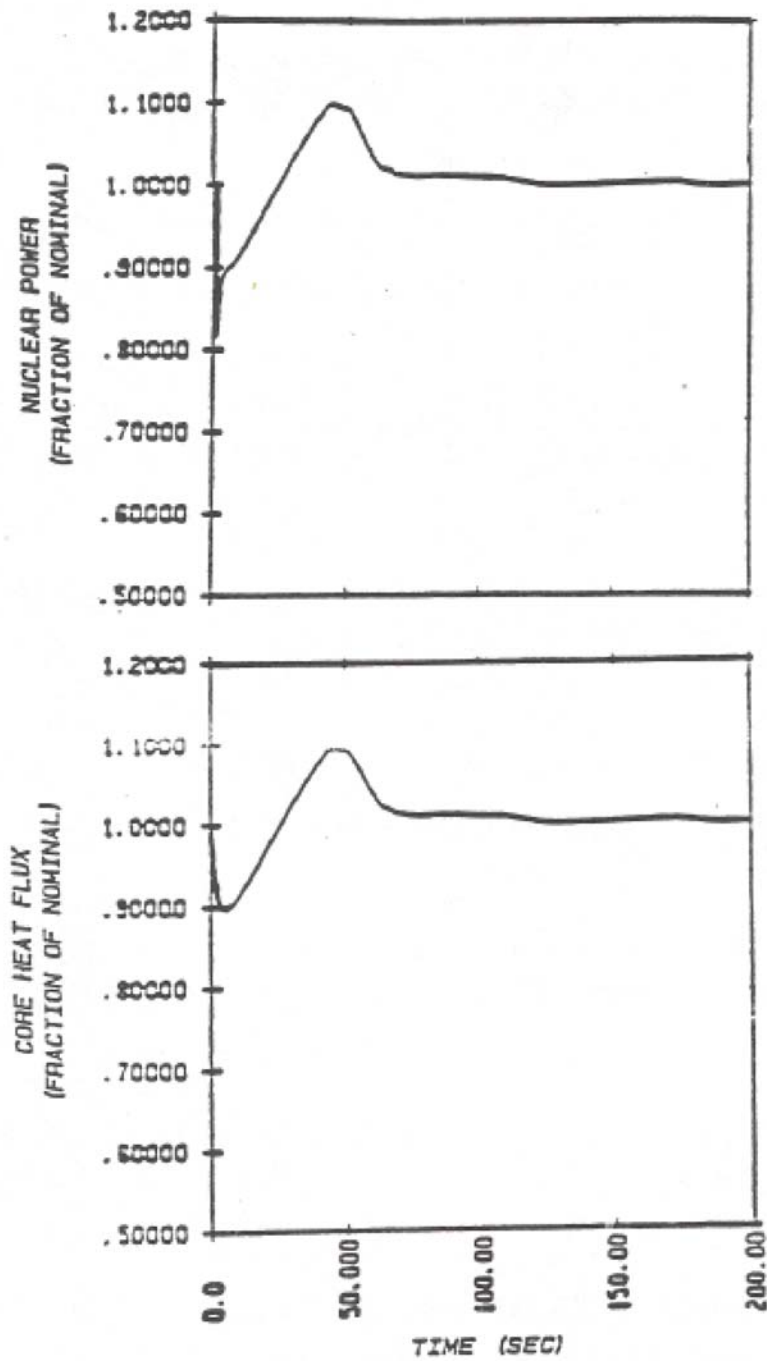
Figure 14.1.3-2 PRESSURIZER PRESSURE TRANSIENT AND VESSEL AVERAGE
TEMPERATURE TRANSIENT FOR DROPPED RCCA
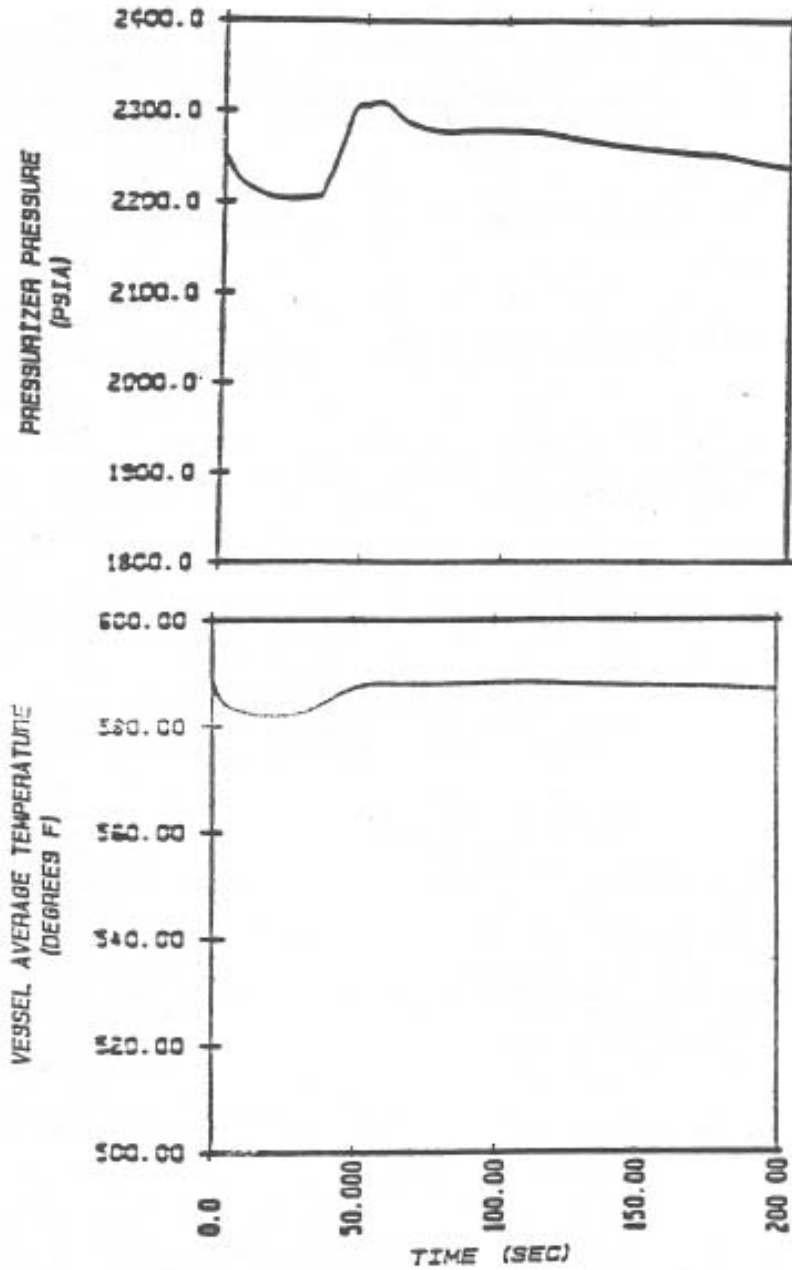
14.1.4   CHEMICAL AND VOLUME CONTROL SYSTEM MALFUNCTION

Positive reactivity can be added to the core with the Chemical and Volume Control System by feeding reactor makeup water into the Reactor Coolant System via the reactor makeup control system. The normal dilution procedures call for a limit on the rate and magnitude for any individual dilution, under strict administrative controls. Boron dilution is a manual operation. A boric acid blend system is provided to permit the operator to match the boron concentration of reactor coolant makeup water to that existing in the coolant at the time. The Chemical and Volume Control System is designed to limit, even under various postulated failure modes, the potential rate of dilution to a value which, after indication through alarms and instrumentation, provides the operator sufficient time to correct the situation in a safe and orderly manner.

The most limiting credible source of reactor makeup water to the reactor coolant system is from the reactor makeup water storage tank using the reactor makeup water pumps. Dilution via this pathway can be readily terminated by isolating this source.

The rate of addition of unborated makeup water to the reactor coolant system is limited to the capacity of the CVCS charging pumps and FCV-111. Normally one charging pump is operating in manual mode and one pump is operating in the automatic mode, responding to pressurizer level changes.

The boric acid from the boric acid tank is blended with the reactor makeup water in the blender and the composition is determined by the preset flow rates of boric acid and reactor makeup water on the reactor makeup control system. Two separate operations are required. First, the operator must switch from the automatic makeup mode to the dilute mode. Second, a manual start of the system is required. Omitting either step would prevent dilution. This makes the possibility of inadvertent dilution very small.

Information on the status of the reactor coolant makeup is continuously available to the operator. Lights are provided on the control board to indicate the operating condition of pumps in the chemical and volume control system. 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. An additional alarm is available to warn the operator of a potential dilution condition.

To cover all phases of plant operation, boron dilution during refueling, startup, and power operation are considered in this analysis.

Method of Analysis and Results

Dilution During Refueling

During refueling the following conditions exist:

1.   One residual heat removal pump is running to ensure continuous mixing in the reactor vessel,

2.   The valves on the suction side of the charging pumps are adjusted for addition of concentrated boric acid solution.

3.   The boron concentration of the refueling water corresponds to a shutdown margin of at least that required by COLR 2.12; periodic sampling ensures that this concentration is maintained, and

4.   Neutron sources can be installed in the core, if necessary, during startup to provide a minimum count rate. However, neutron source assemblies are not currently used in Unit 1 or Unit 2. BF_3 detectors connected to instrumentation giving audible count rates are installed to provide direct monitoring of the core.

A minimum active water volume in the reactor coolant system of 1884 ft^3 is considered. This corresponds to the volume necessary to fill the reactor vessel up to the midplane of the nozzles plus the volume of one RHR train. This ensures mixing via the residual heat removal loop.

The maximum dilution flow of 121 gpm and uniform mixing are also considered. Administrative procedures limit the charging flow available during this condition. The maximum dilution flow assumes a single failure, such that two pumps are delivering maximum flow. The actual amount of reactor makeup water delivered to the suction of the charging pumps would be determined by the position of FCV-111 which is normally set at no more than 40 gpm. At the full open position, FCV-111 would pass approximately 100 gpm.

The operator has prompt and definite indication of any boron dilution from the audible count rate instrumentation. High count rate is alarmed in the reactor containment and the main control room. The count rate increase is proportional to the inverse multiplication factor.

The Technical Specifications require that one source range audible count rate circuit be operable during MODE 6. If the required audible count rate circuit becomes inoperable, then actions are immediately taken to isolate all sources of unborated water. Isolating these flow paths ensures that an inadvertent dilution of the reactor coolant boron concentration is prevented. Therefore, the mitigative function of the audible count rate circuit is ensured to be available, or else conditions are established to prevent a boron dilution, through the control of the Technical Specifications (Reference 1).

A ratio of the initial refueling water boron concentration to the critical boron concentration that is greater than or equal to 1.3125 corresponds to more than 30 minutes before the loss of all shutdown margin. This is ample time for the operator to recognize the audible high count rate signal and isolate the reactor makeup water source by closing valves and stopping the reactor makeup water pumps.

Dilution During Cold Shutdown

This analysis was performed to determine the required boron concentration necessary to prevent criticality from an inadvertent boron dilution event with a reduced RCS volume for a duration of 15 minutes.

The analysis used a conservative RCS and RHR combined volume by assuming that the RCS is drained to the midplane of the nozzles (1884 ft^3). The RCS volume when drained to the midplane of the nozzles is the smallest volume that can result from any allowable scenario while in Cold Shutdown. Mixing of the diluting water (boron free) and the RCS water was assumed to take place at the vessel inlet nozzle which then proceeds in a "wave front fashion" through the rest of the RCS. A maximum RCS temperature of 200°F is assumed and a minimum temperature is

assumed for the dilutant. The dilution flow rate is conservatively increased to compensate for the density differences.

These calculations determine what boron concentration is required to ensure that the operator has 15 minutes to identify and terminate the boron dilution prior to a complete loss of shutdown margin. The calculations cover one, two or three charging pumps in operation and RHR flow rates up to approximately 6000 gpm. The results of the analysis are presented in Figure 14.1.4-1.

The assumptions and conclusions of this analysis are maintained by administratively limiting charging pump operation in accordance with Figure 14.1.4-1. A limit switch on the valve for the reactor makeup water pump is also provided to warn the operators of a potential dilution in progress. The limit switch will activate an alarm in the control room whenever the valve is not closed.

Dilution During Startup

Prior to refueling, the reactor coolant system is filled with borated water from the refueling water storage tank. Core monitoring is by external BF_3 detectors. Mixing of reactor coolant is accomplished by operation of the reactor coolant pumps. Again the maximum dilution flow (181.5 gpm) is considered. The volume of reactor coolant is approximately 5035 ft^3 which is the volume of the reactor coolant system excluding the pressurizer. The volume has been calculated taking into account steam generator tube plugging. High source level and all reactor trip alarms are effective.

The minimum time required to reduce the reactor coolant boron concentration from 1800 to 1600 ppm, where the reactor could go critical with all rods at the insertion limits, is greater than 15 minutes. Once again, this should be more than adequate time for operator action due to the high count rate signal, and for termination of dilution flow.

Dilution at Power

For dilution at power, it is necessary that the time to lose shutdown margin be sufficient to allow identification of the problem and termination of the dilution. As in the dilution during startup case, the RCS volume reduction due to steam generator tube plugging is considered. The effective reactivity addition rate is a function of the reactor coolant temperature and boron concentration. The reactivity insertion rate calculated is based on a conservatively high charging flow rate capacity (181.5 gpm). The reactor is assumed to have all rods at the insertion limits in either automatic or manual control. With the reactor in manual control and no operator action to terminate the transient, the power and temperature rise will cause the reactor to reach the reactor protection (i.e., OTΔT, high nuclear flux) trip setpoint, resulting in a reactor trip. After reactor trip there are greater than 15 minutes for operator action prior to return to criticality. The boron dilution transient in this case is essentially equivalent to an uncontrolled rod withdrawal at power. The maximum reactivity insertion rate for a boron dilution transient is conservatively estimated to be 0.90 pcm/sec and is within the range of insertion rates analyzed for an uncontrolled rod withdrawal at power. Prior to reaching the reactor protection trip, the operator will have received an alarm on overtemperature ΔT and turbine runback.

With the reactor in automatic control, a boron dilution will result in a power and temperature increase such that the rod controller will attempt to compensate by slow insertion of the control rods. This action by the controller will result in rod insertion limit and axial flux alarms. If the reactor is shutdown, the minimum time for operator action prior to return to criticality would be greater than 15 minutes.

Conclusions

Because of the procedures involved in the dilution process, an erroneous dilution is not considered credible. Nevertheless, if an unintentional dilution of boron in the reactor coolant does occur, numerous alarms and indications are available to alert the operator to the condition. The maximum reactivity addition due to the dilution is slow enough to allow the operator to determine the cause of the addition and take corrective action before the required shutdown margin is lost.

Reference:

1.	Technical Specification 3.9.2, Nuclear Instrumentation.

2.	NRC Safety Evaluation 2011-004, "Issuance of License Amendments Regarding Extended Power Uprate," dated May 3, 2011.

Figure 14.1.4-1 RATIO OF THE INITIAL BORON CONCENTRATION TO THE CRITICAL BORON CONCENTRATION (DILUTION FACTOR, DLF) AS A FUNCTION OF RHR FLOW RATE
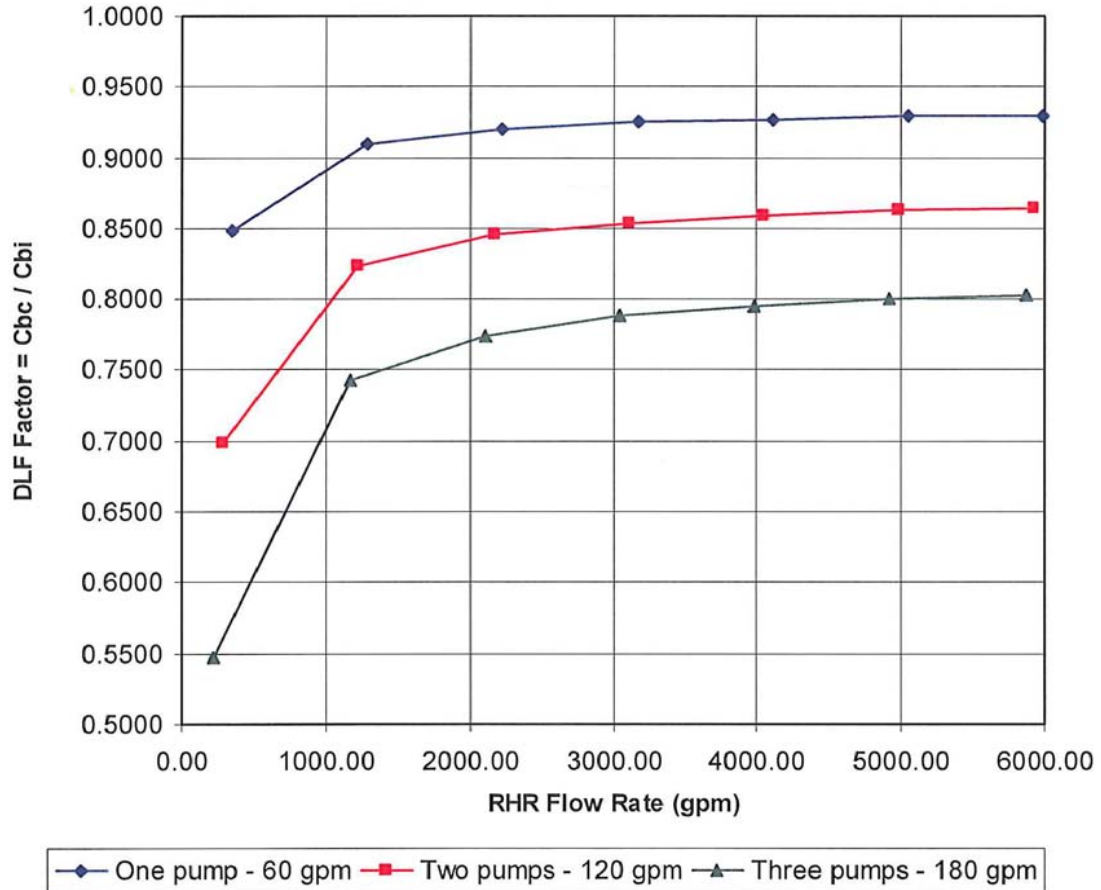
14.1.5   STARTUP OF AN INACTIVE REACTOR COOLANT LOOP

Operation of the plant with an inactive loop causes reversed flow through the inactive loop because there are no isolation valves or check valves in the reactor coolant loops.

If the reactor is operated at power in this condition, there is a decrease in the coolant temperature in that loop in comparison with the other loop. The subsequent startup of the idle reactor coolant pump, would result in the injection of colder water into the core. This colder water and increased flow rate causes an increase in reactivity and therefore a power increase.

The Point Beach Nuclear Plant Technical Specifications do not permit the reactor to be taken critical with only one reactor coolant pump (RCP) in operation. Because of this, an analysis of this event was determined not to be necessary. The discussion presented below corresponds to an analysis previously performed assuming a nominal initial power level of 10% and is retained for historical purposes.

Method of Analysis and Assumptions

This transient is analyzed by three digital computer codes. The LOFTRAN code (Reference 1) is used to calculate the loop and core flow, nuclear power and core pressure and temperature transients following the startup of an idle pump. FACTRAN (Reference 2) is used to calculate the core heat flux transient based on core flow and nuclear power from LOFTRAN. The THINC code is then used to calculate the DNBR during the transient based on system conditions (pressure, temperature, and flow) calculated by LOFTRAN and heat flux as calculated by FACTRAN.

The reverse flow in the inactive loop is calculated to be 15% of the nominal loop flow, which carries about 10% of the heat generated in the core to the secondary system and causes a temperature difference of -8.2°F between the temperature in the cold leg of the active loop and the temperature in the "hot" leg of the inactive loop. The cold water is introduced into the core upon the startup of the inactive loop.

The following assumptions are made:

1.   The idle pump, on starting, accelerates to full flow in 20 seconds.

2.   A conservative maximum moderator density coefficient of .43 Δk/gm/cc is assumed.

3.   A conservative large (absolute value) Doppler coefficient of -2.9 x 10^{-5} Δk/°F is taken.

4.   The water entering the core is assumed to exhibit the temperature of the water in the inactive loop. This assumption provides the analysis with a high degree of conservatism.

Results

Figure 14.1.5-1 through Figure 14.1.5-4 show the plant transients. The cold water slug reaches the reactor core with a delay of approximately 7 seconds and is sustained for 14 seconds. It decreases the core water temperature and causes the nuclear power excursion. The peak power is 30% of full power and does not cause a reactor trip.

The average temperature of the reactor coolant water increases due to the heating up of the cold water which existed in the inactive loop and this leads to the increase in the pressurizer pressure. The maximum pressure for this transient does not actuate the pressurizer relief valves.

Conclusion

The results show that for startup of an inactive loop at 10% power, the power and temperature excursions are not severe. These transients are given only to indicate the transient behavior of the reactor following an incident of this type. The conclusion is that the transient effects of this accident are not severe and place no undue restrictions on the plant, when operating at 10% power.

The Point Beach Nuclear Plant Technical Specifications do not permit the reactor to be taken critical with only one reactor coolant pump (RCP) in operation. Because of this, the startup of an inactive loop is non-limiting with respect to minimum DNBR. No analysis is required to show that the minimum DNBR is satisfied for this event.

REFERENCES

1.   Burnett, T. W. T., et.al., "LOFTRAN Code Description," WCAP-7907-P-A, April 1984.

2.   Hargrove, H. G., "FACTRAN - A Fortran - IV Code for Thermal Transients in a UO2 Fuel Rod," WCAP-7908 (Non-Proprietary), July 1972.

3.   NRC Safety Evaluation 2011-004, "Issuance of License Amendments Regarding Extended Power Uprate," dated May 3, 2011.

Figure 14.1.5-1 START-UP OF AN INACTIVE REACTOR COOLANT LOOP

START-UP OF AN INACTIVE REACTOR COOLANT LOOP
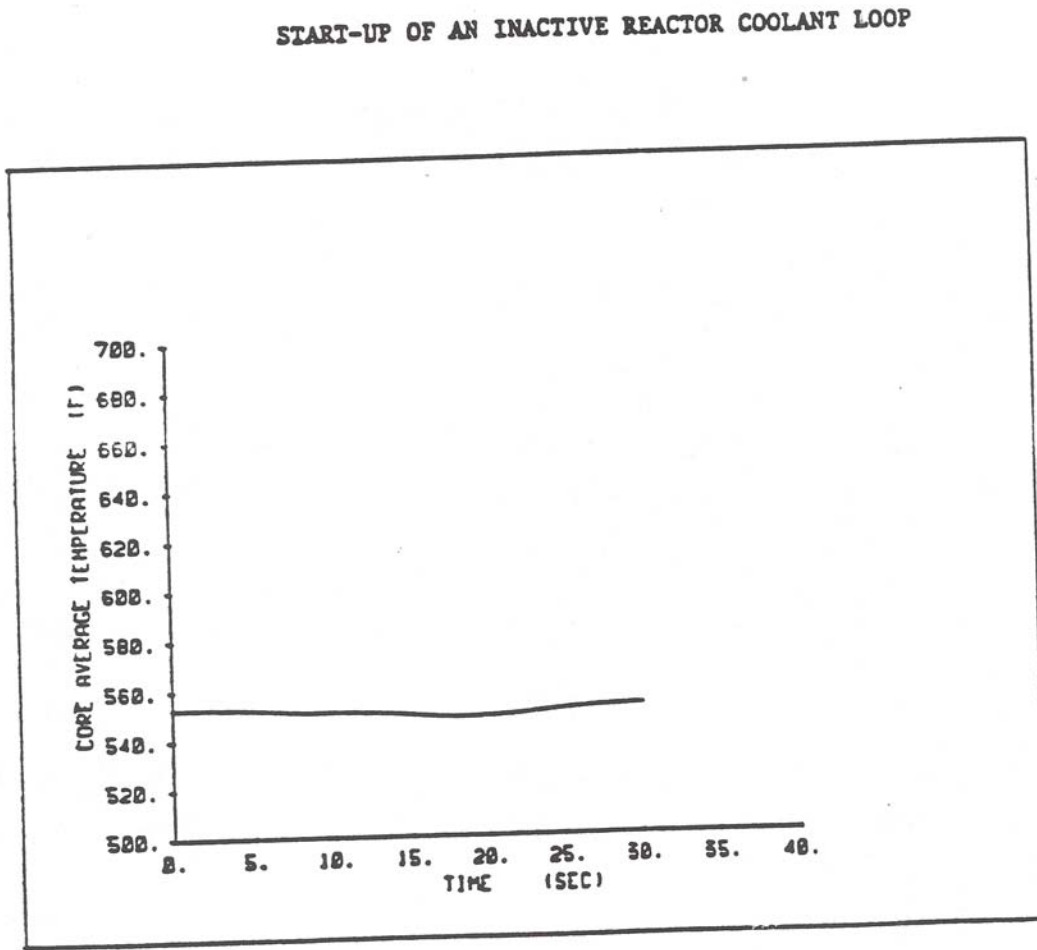
FIGURE 14.1.5-1

Figure 14.1.5-2 START-UP OF AN INACTIVE REACTOR COOLANT LOOP

START-UP OF AN INACTIVE REACTOR COOLANT LOOP
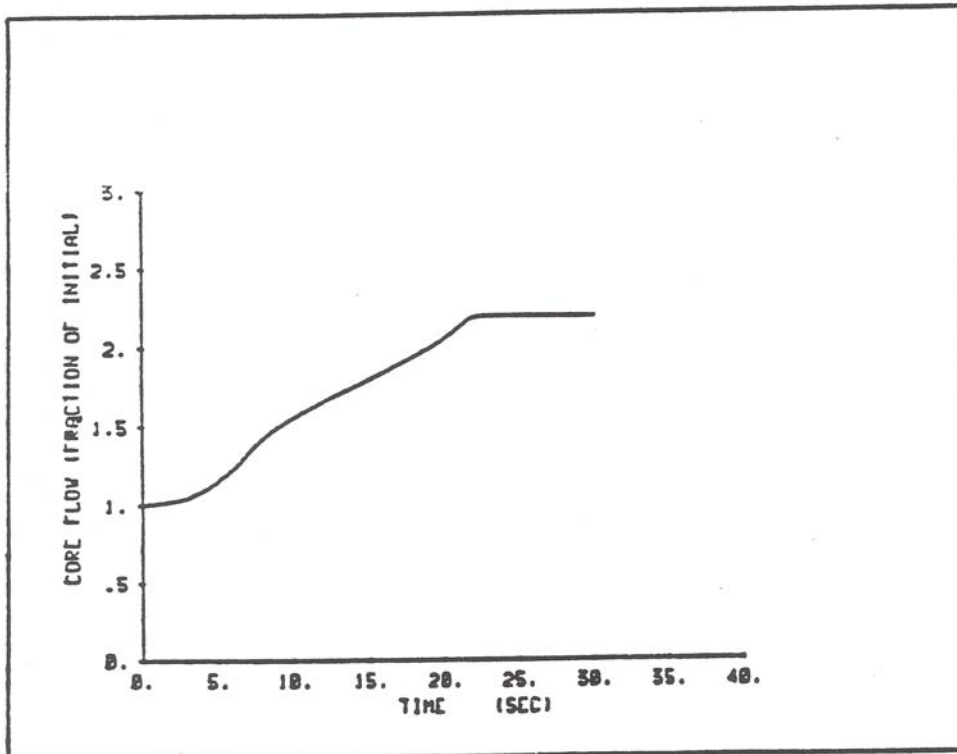
FIGURE 14.1.5-2

Figure 14.1.5-3 START-UP OF AN INACTIVE REACTOR COOLANT LOOP

START-UP OF AN INACTIVE REACTOR COOLANT LOOP
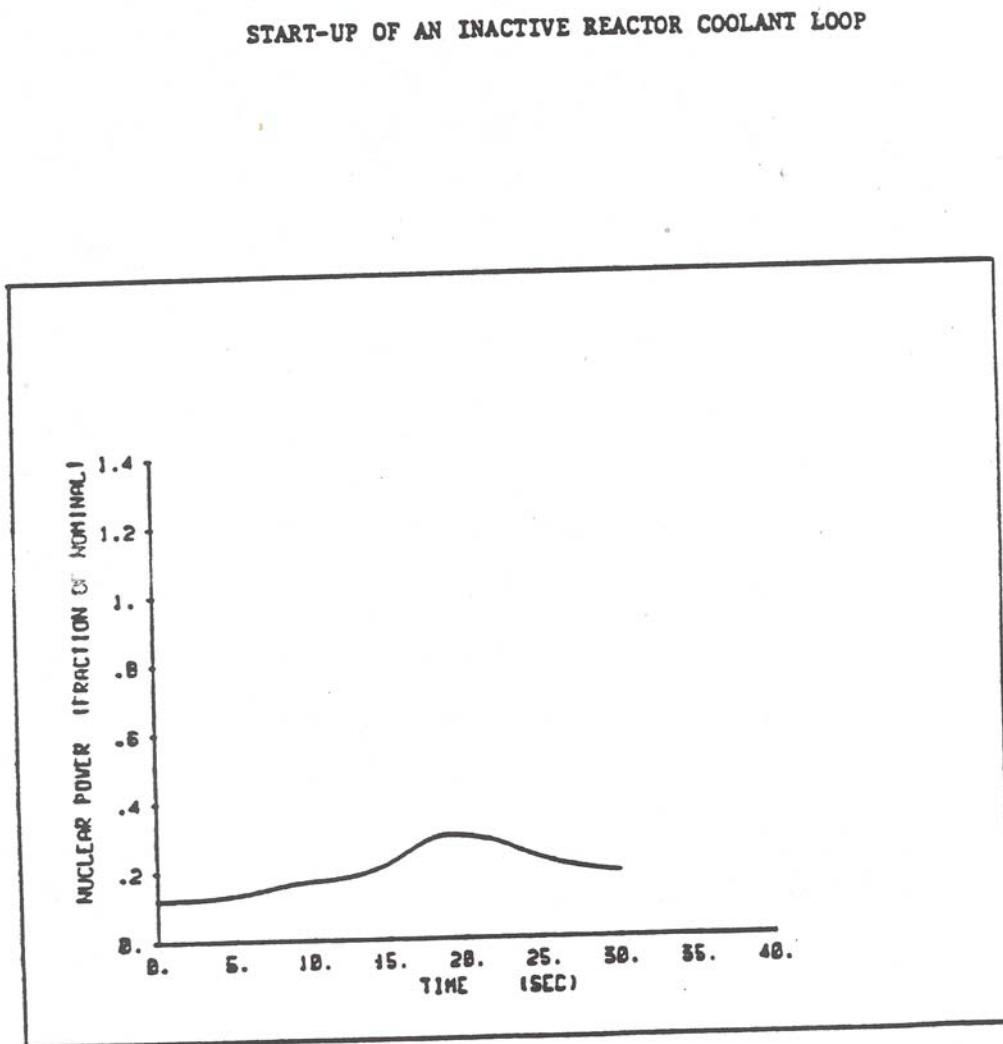
FIGURE 14.1.5-3

Figure 14.1.5-4 START-UP OF AN INACTIVE REACTOR COOLANT LOOP

START-UP OF AN INACTIVE REACTOR COOLANT LOOP
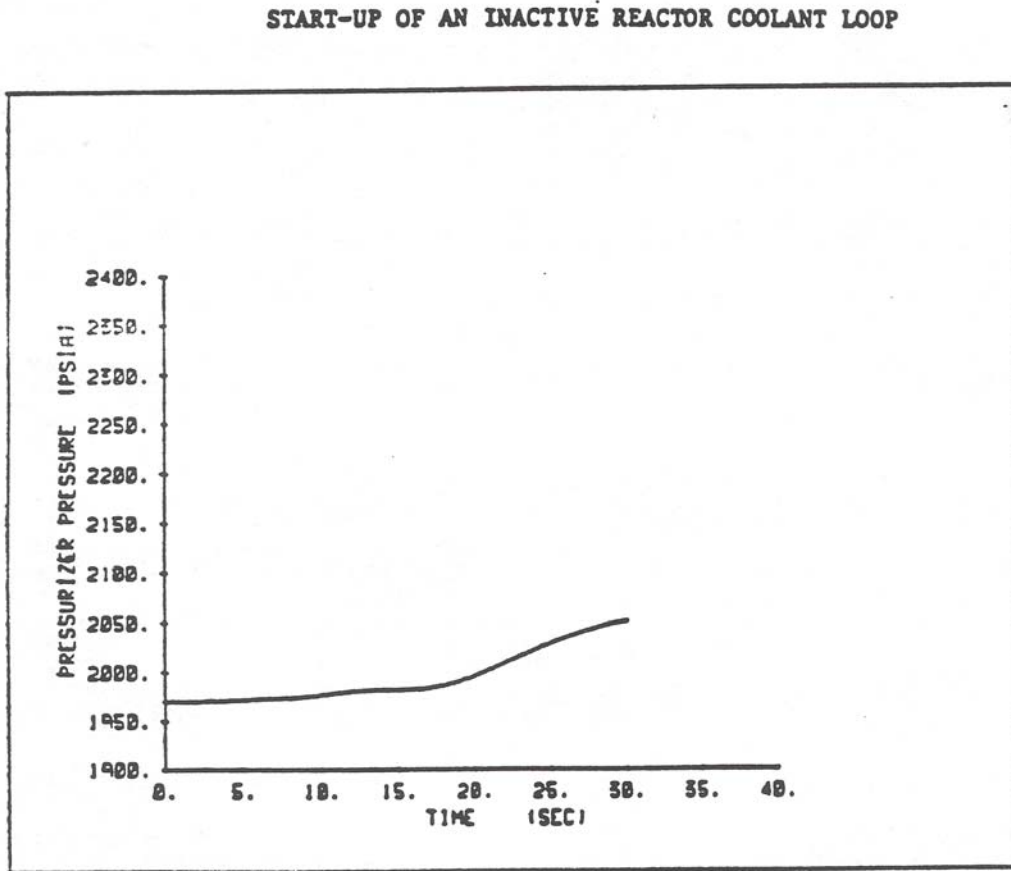
FIGURE 14.1.5-4

14.1.6   <u>REDUCTION IN FEEDWATER ENTHALPY INCIDENT</u>

The reduction in feedwater enthalpy is another means of increasing core power above full power. Such increases are attenuated by the thermal capacity in the secondary plant and in the reactor coolant system. The overpower-overtemperature protection (nuclear overpower and ΔT trips) prevents any power increase which could lead to a DNBR less than the safety analysis limit DNBR.

An extreme example of excess heat removal by the feedwater system is the transient associated with the accidental opening of the feedwater bypass valve which diverts flow around the low pressure feedwater heaters. The function of this valve is to maintain net positive suction head on the main feedwater pump in the event that the heater drain pump flow is lost, e.g., during a large load decrease.

In the event of accidental opening, there is a sudden reduction in inlet feedwater temperature to the steam generators. The increased subcooling will create a greater load demand on the primary system which can lead to a reactor trip.

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

The net effect on the RCS due to a reduction in feedwater enthalpy 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.

The protection available to mitigate the consequences of a decrease in feedwater enthalpy is the same as that for an excessive load increase, as discussed in Section 14.1.7.

<u>Method of Analysis</u>

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 of 1806 MWt.

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

<u>Results</u>

Opening of a low pressure heater bypass valve causes a reduction in feedwater temperature which increases the thermal load on the primary system. The reduction in feedwater temperature is less than 40°F (Reference 1) resulting in an increase in heat load on the primary system of less than

10% of full power. The reduction in feedwater temperature due to a 10% step load increase is 69°F. The increased thermal load, due to opening of the low pressure heater bypass valve, thus results in a transient very similar (but of reduced magnitude) to that present in Section 14.1.7 for an excessive load increase, which evaluates the consequences of a 10% step load increase. Therefore, the transient results of this analysis are not presented.

No explicit analysis was performed. However, an engineering evaluation performed at current and uprated power showed the 40°F evaluated was conservative even for the uprated power and the event remained bounded by the excessive load incident in Section 14.1.7.

Conclusions

The decrease in feedwater enthalpy incident is less severe than the excessive load increase incident (see Section 14.1.7). Based on results presented in Section 14.1.7, the applicable acceptance criteria for the reduction in feedwater enthalpy incident have been met.

References

1.   Shaw Calculation 129187-M-0001, Revision 0, "Condensate, Feedwater, and Heater Drain Systems Hydraulic Model for NSSS Power Level of 1806 MWt," November 24, 2008.

2.   NRC Safety Evaluation 2011-004, "Issuance of License Amendments Regarding Extended Power Uprate," dated May 3, 2011.

14.1.7   EXCESSIVE LOAD INCREASE INCIDENT

An excessive load increase incident is defined as a rapid increase in steam generator steam flow that causes a power mismatch between the reactor core power and the steam generator load demand. The reactor control system is designed to accommodate a 10% step load increase and/or a 5% per minute ramp load increase (without a reactor trip) in the range of 15 to 100% full power. Any loading rate in excess of these values may cause a reactor trip actuated by the reactor protection system. If the load increase exceeds the capability of the reactor control system, the transient is terminated in time to prevent DNBR less than the limiting value, by a combination of the nuclear overpower trip and the overpower-overtemperature ΔT trips, as discussed in Section 7.0. An excessive load increase incident could result from either an administrative violation such as excessive loading by the operator or an equipment malfunction such as steam bypass control or turbine speed control.

To avoid excessive load increases, either by manual operator action or by automatic system demand, the normal configuration at full load is for the turbine valve position limiter to be set slightly above the full load governor valve position.

During power operation, steam bypass to the condenser is controlled by reactor coolant condition signals, i.e., abnormally high reactor coolant temperature indicates a need for steam bypass. A single controller malfunction does not cause steam bypass; an interlock is provided which blocks the control signal to the valves unless a large turbine load decrease or a turbine trip has occurred.

Method of Analysis

This accident is analyzed using the RETRAN code. The code simulates the neutron kinetics, reactor coolant system, pressurizer, pressurizer relief and safety valves, pressurizer spray, steam generator, steam generator safety valves, and feedwater system. The code computes pertinent plant variables, including temperatures, pressures, and power level.

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

1.   Reactor control in manual with minimum reactivity feedback.
2.   Reactor control in automatic with minimum reactivity feedback.
3.   Reactor control in manual with maximum reactivity feedback.
4.   Reactor control in automatic with maximum reactivity feedback.

For the minimum reactivity feedback cases, the core has the least negative moderator temperature coefficient (0 pcm/°F) of reactivity and the least negative Doppler only power coefficient; therefore, the least inherent transient response capability. For the maximum reactivity feedback cases, the moderator temperature coefficient of reactivity has its most negative value and the most negative Doppler only power coefficient. This results in the largest amount of reactivity feedback due to changes in coolant temperature.

A conservative limit on the turbine valve opening is assumed, and all cases are studied without credit being taken for pressurizer heaters. This accident is analyzed with the Revised Thermal Design Procedure as described in Reference 1, Section 14.0. Plant characteristics and initial conditions are as discussed in Section 14.1. Initial reactor power, pressure, and RCS temperatures are assumed to be at their nominal values. Uncertainties in initial conditions are included in the limit DNBR, as described in Reference 1, Section 14.0.

Results

Figure 14.1.7-1 and Figure 14.1.7-3 illustrate the transient with the reactor in the manual control mode. For the beginning-of-life case, there is a slight power increase, and the average core temperature shows a large decrease. This results in a departure from nucleate boiling ratio that increases above its initial value. For the end-of-life, manually controlled case, there is a much larger increase in reactor power due to the moderator feedback. A reduction in departure from nucleate boiling ratio is experienced, but the departure from nucleate boiling ratio remains above the limit value. Figure 14.1.7-2 and Figure 14.1.7-4 illustrate the transient when the reactor is assumed to be in the automatic control mode. Both the beginning-of-life and the end-of-life cases show that core power increases, thereby reducing the rate of decrease in coolant average temperature and pressurizer pressure. For both the beginning-of-life and the end-of-life cases, the minimum departure from nucleate boiling ratio remains above the limit value. The calculated sequence of events is shown in Table 14.1.7-1.

The excessive load increase incident is an overpower transient for which the fuel temperatures rise. When a reactor trip does not occur, the plant reaches a new equilibrium condition at a higher power level corresponding to the increase in steam flow.

Conclusions

The result of the analysis presented above indicate that no applicable acceptance criterion is challenged during this event. The thermal core limit lines are not challenged, and that the minimum DNBR during this transient remains above the safety analysis limit value.

Table 14.1.7-1 TIME SEQUENCE OF EVENTS FOR EXCESSIVE LOAD INCREASE
              INCIDENT

Case	Event	Time of Event (Seconds)
1. Beginning of Core Life, Manual Reactor Control	10% step load increase	0
	Steady-state conditions reached (approximate)	250 (U1) 200 (U2)
2. Beginning of Core Life, Automatic Reactor Control	10% step load increase	0
	Steady-state conditions reached (approximate)	200
3. End of Core Life, Manual Reactor Control	10% step load increase	0
	Steady-state conditions reached (approximate)	200
4. End of Core Life, Automatic Reactor Control	10% step load increase	0
	Steady-state conditions reached (approximate)	200

Figure 14.1.7-1 EXCESSIVE LOAD INCREASE BOL, MANUAL CONTROL
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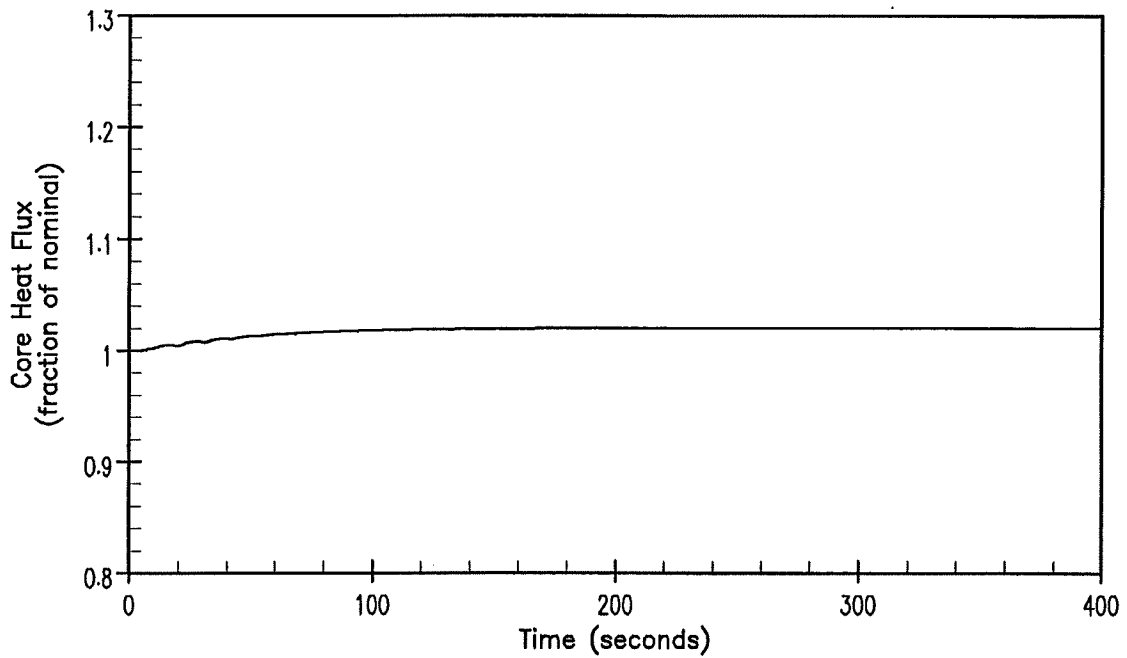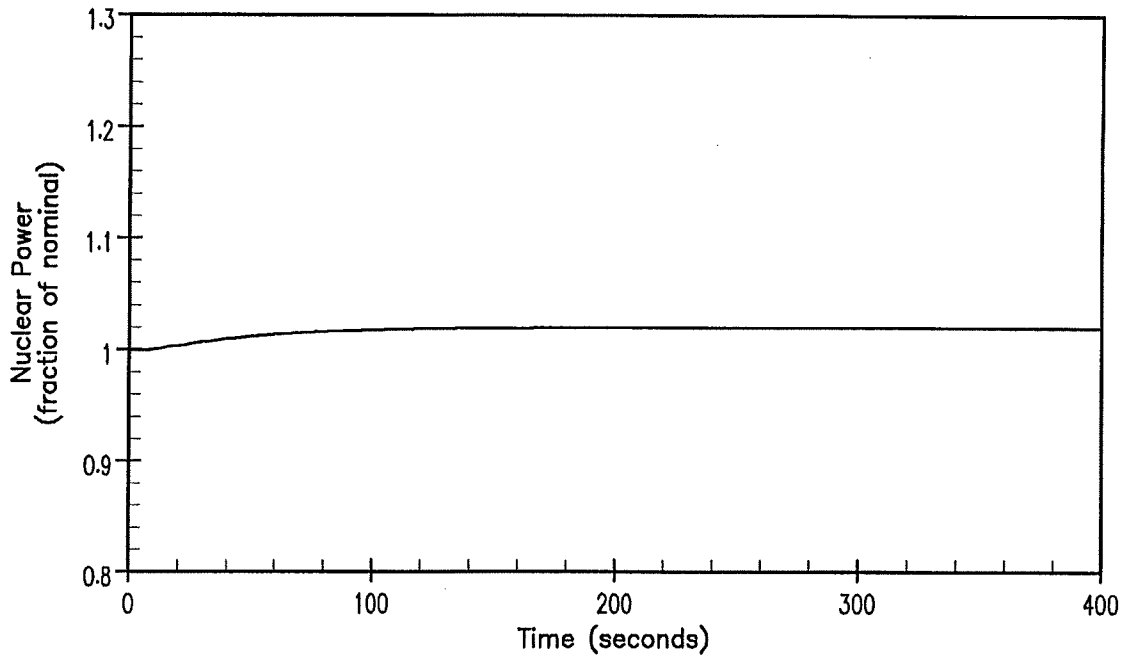
Figure 14.1.7-1 EXCESSIVE LOAD INCREASE BOL, MANUAL CONTROL
Sheet 2 of 4
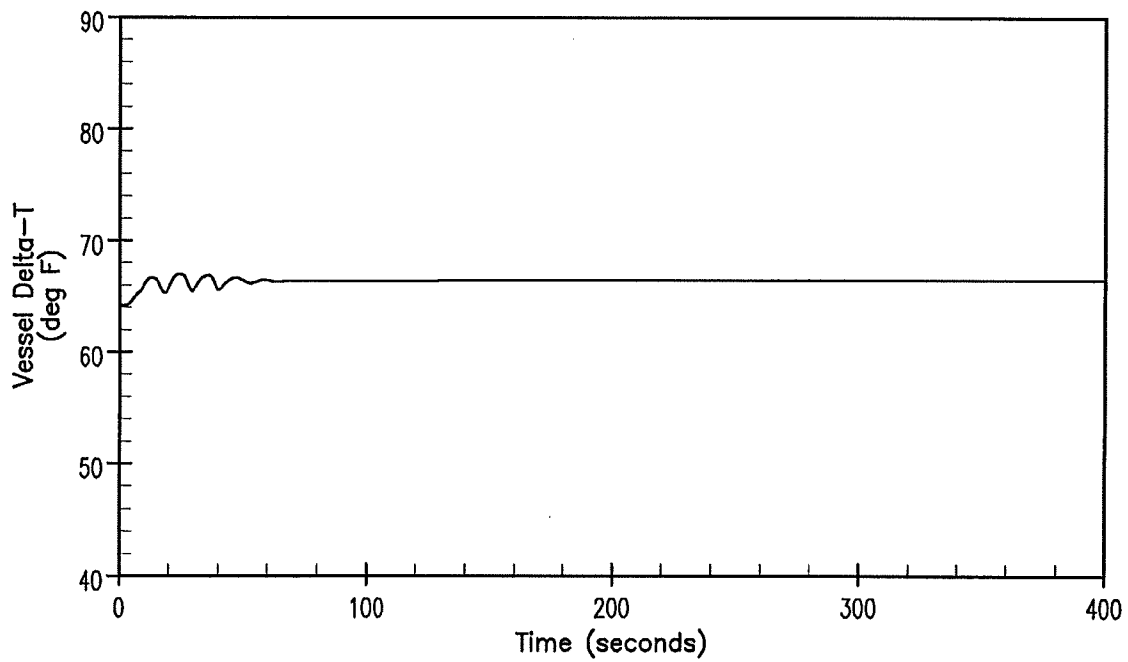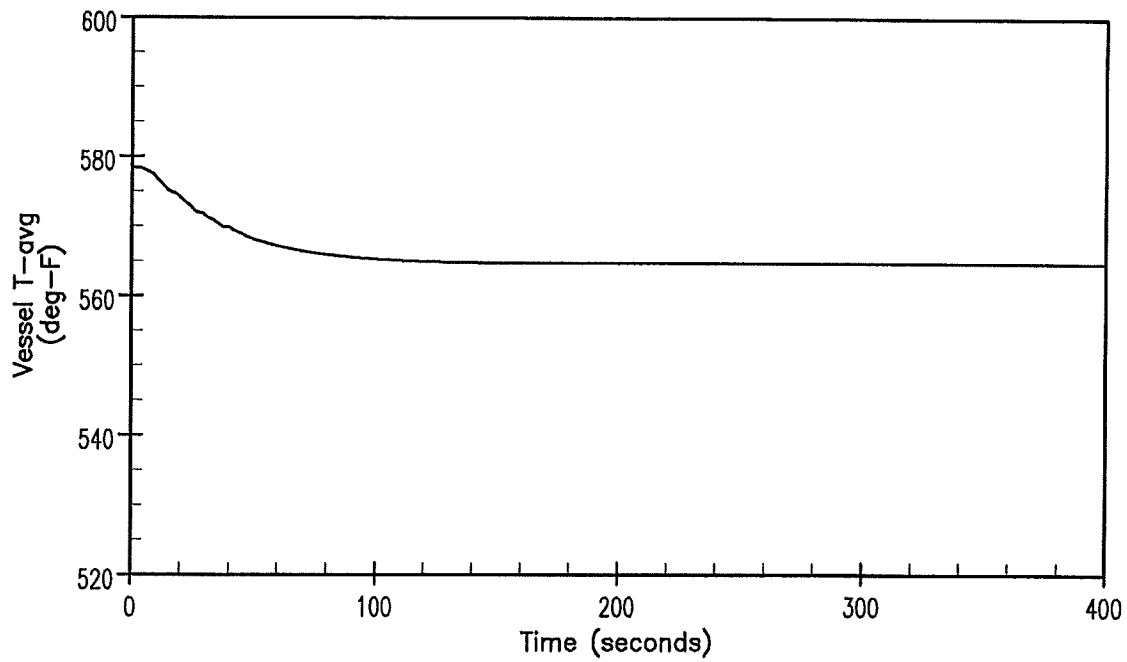
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Sheet 3 of 4
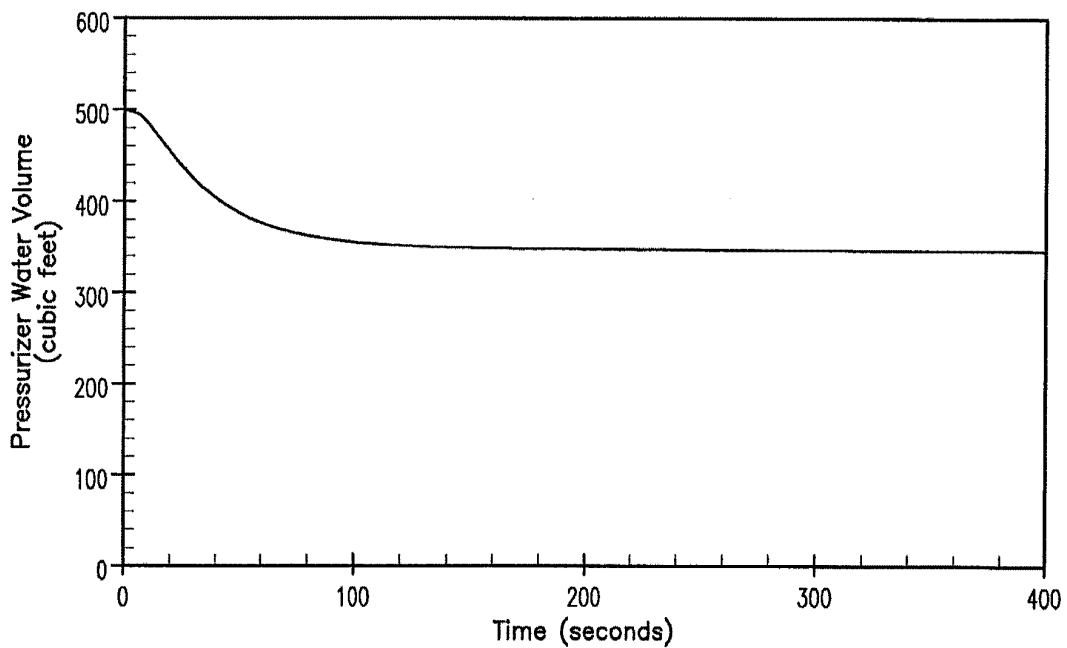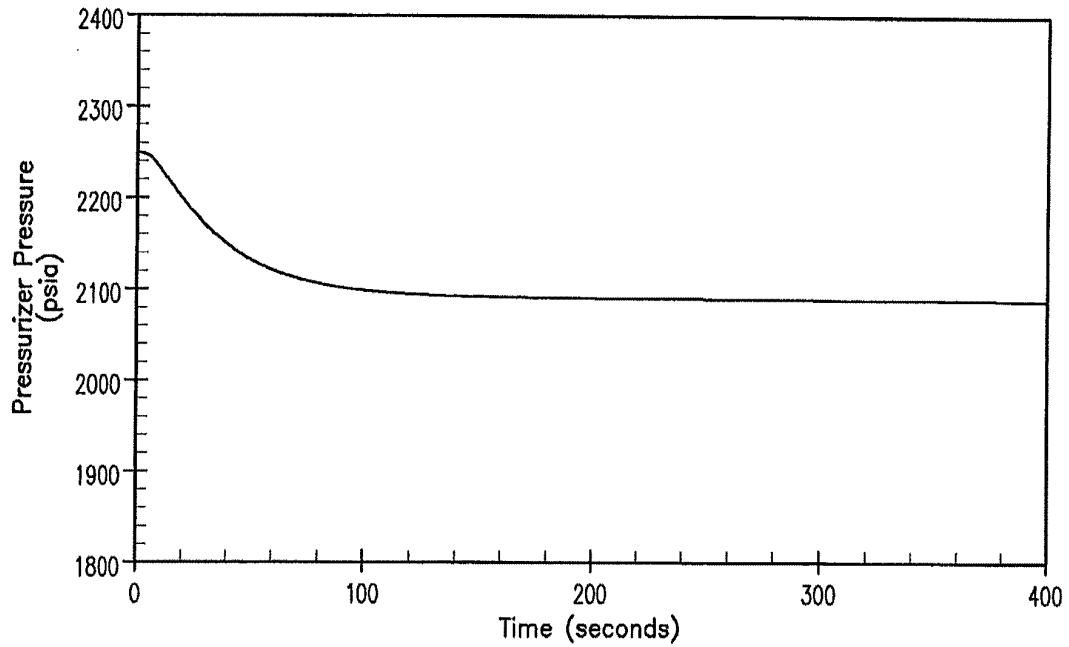
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Sheet 4 of 4
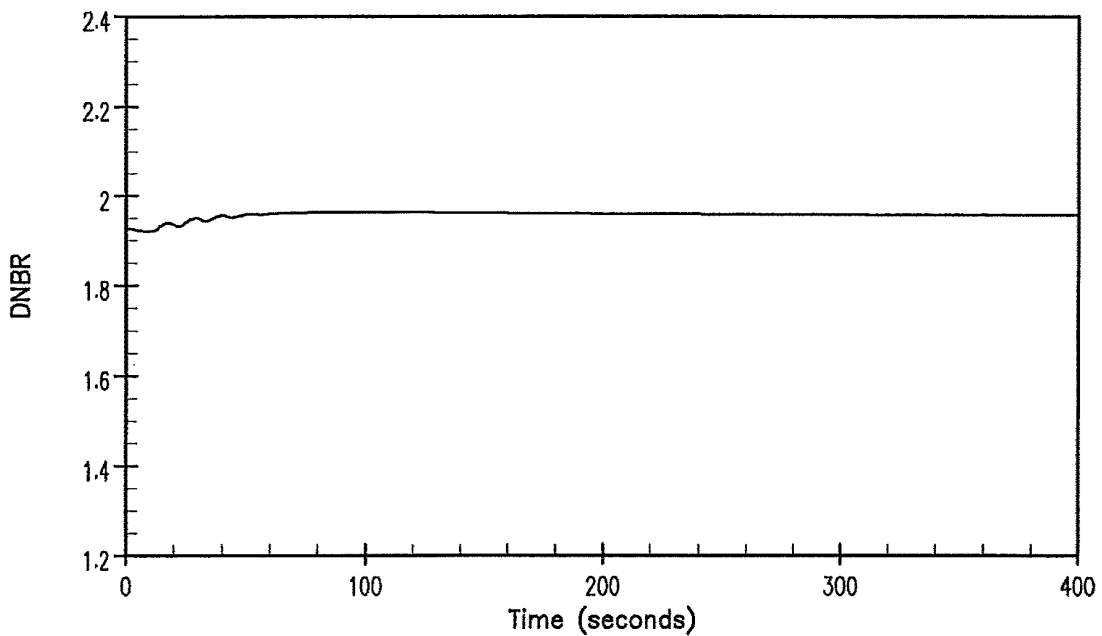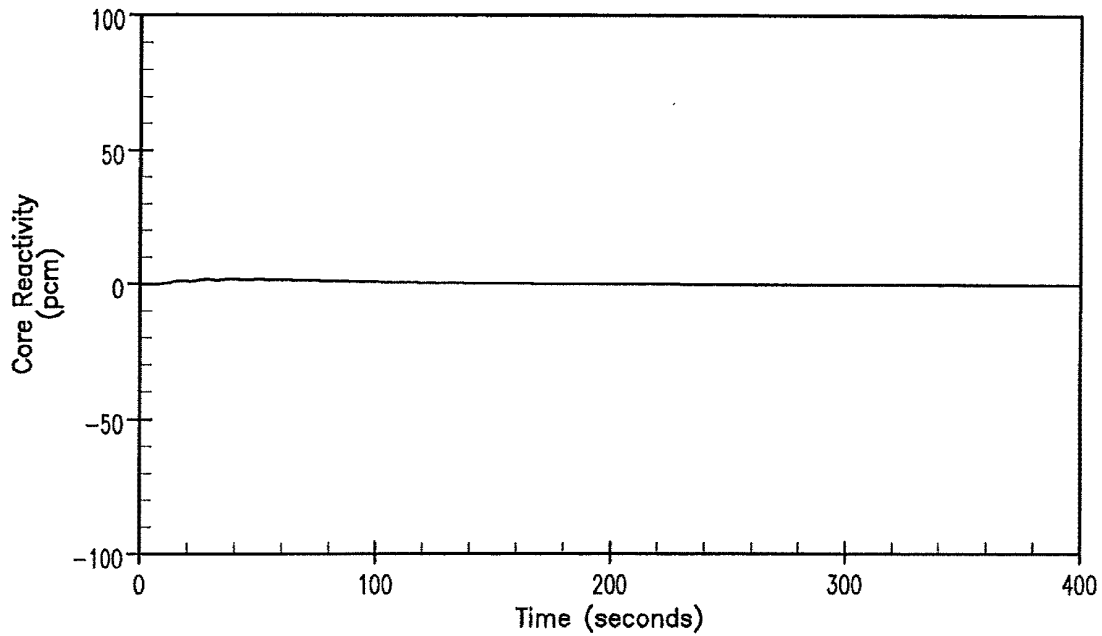
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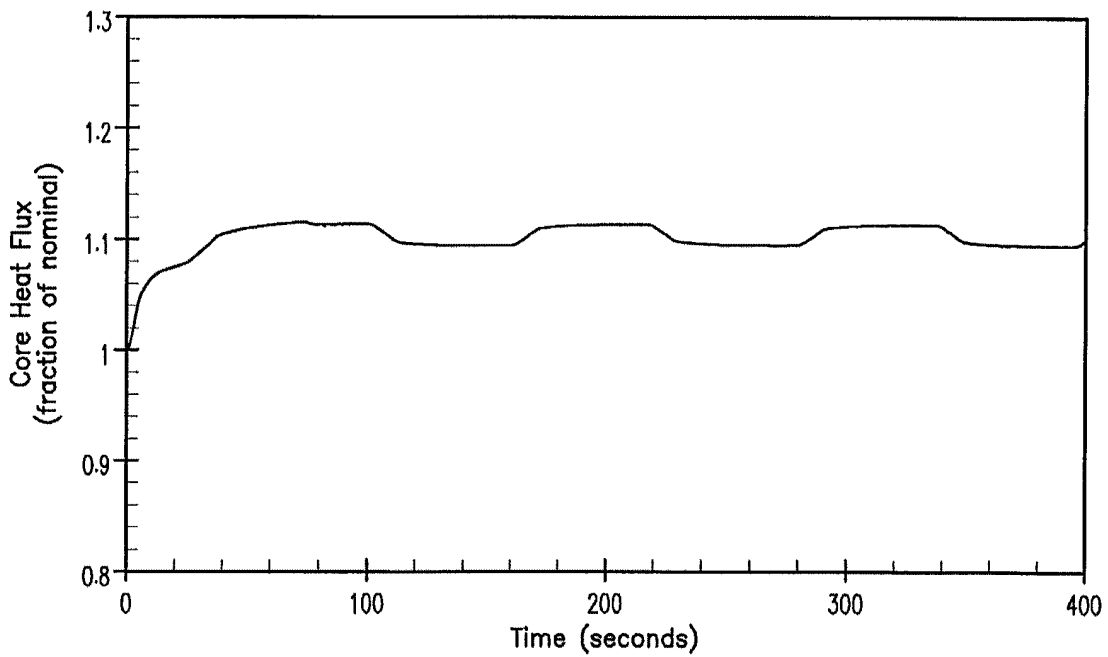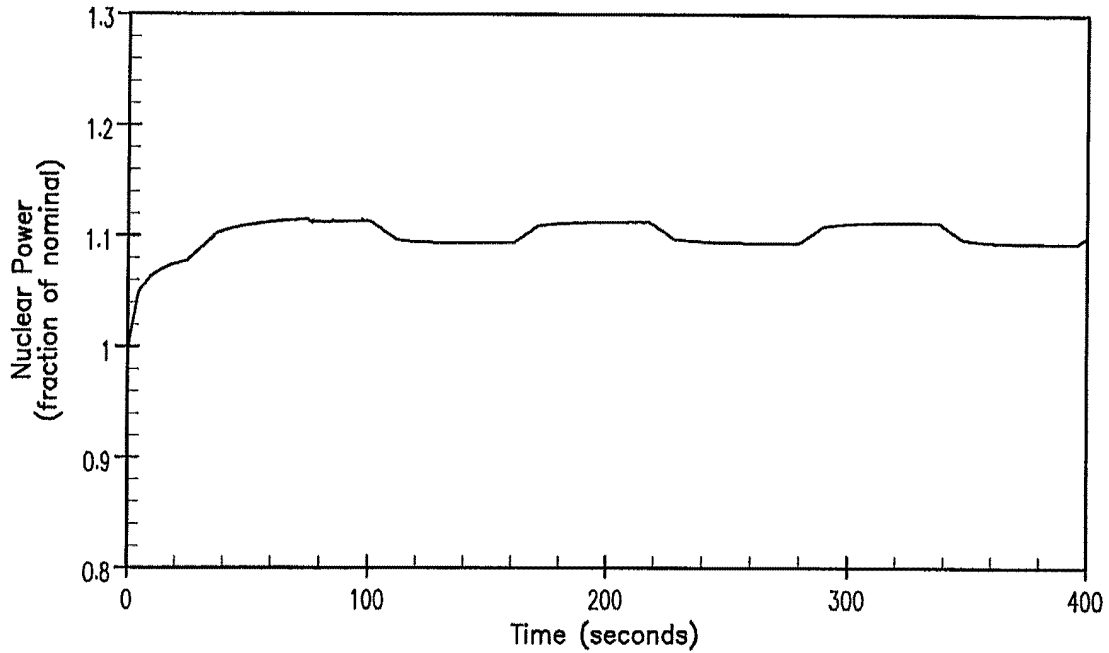
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Sheet 2 of 4
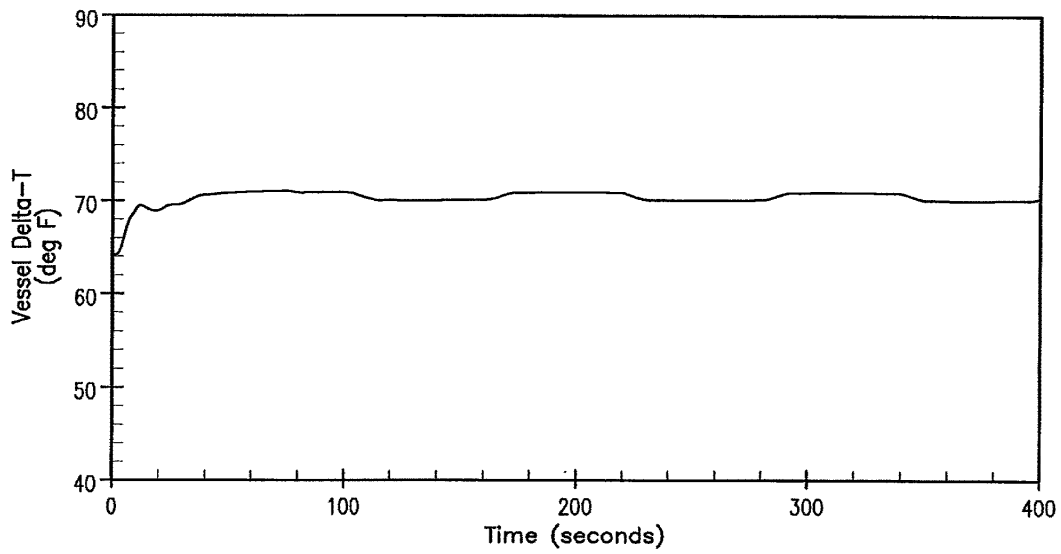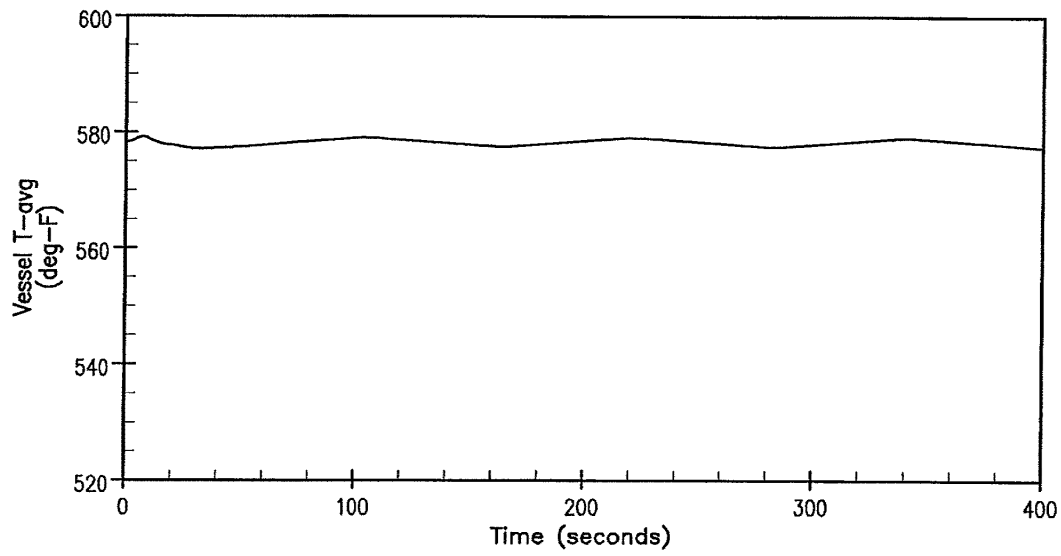
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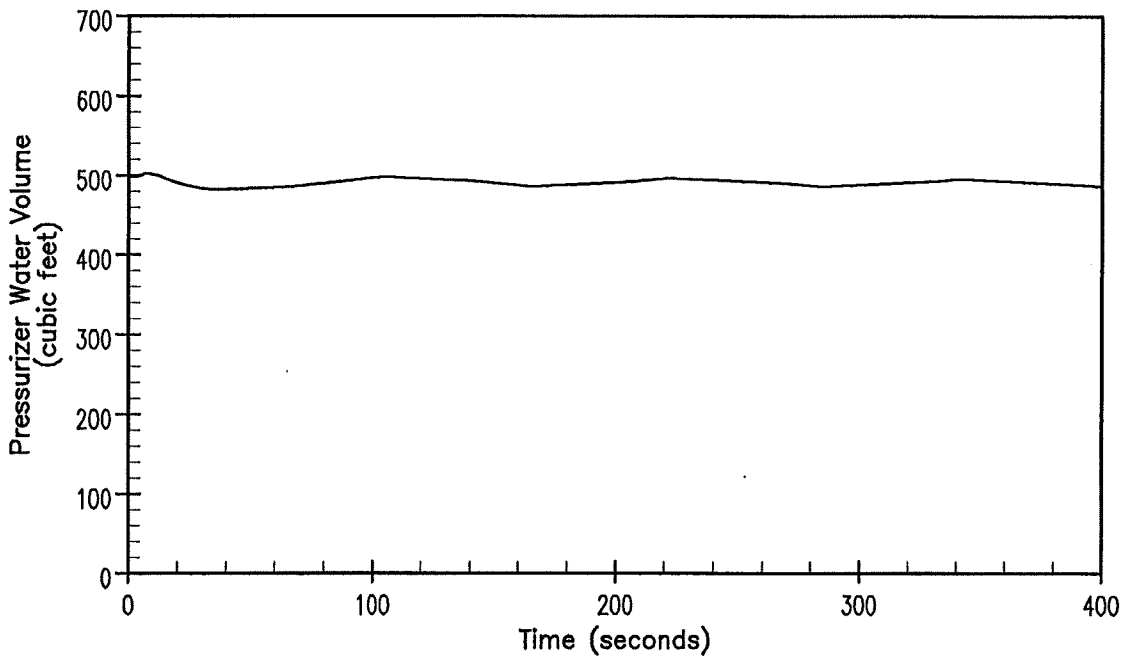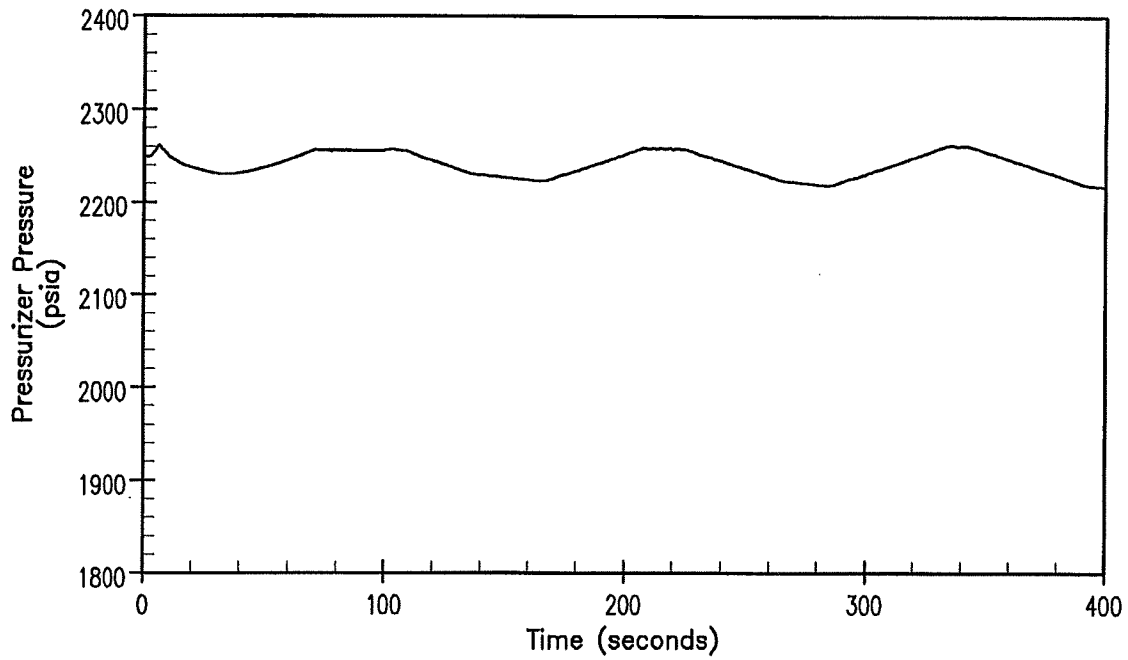
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Sheet 4 of 4
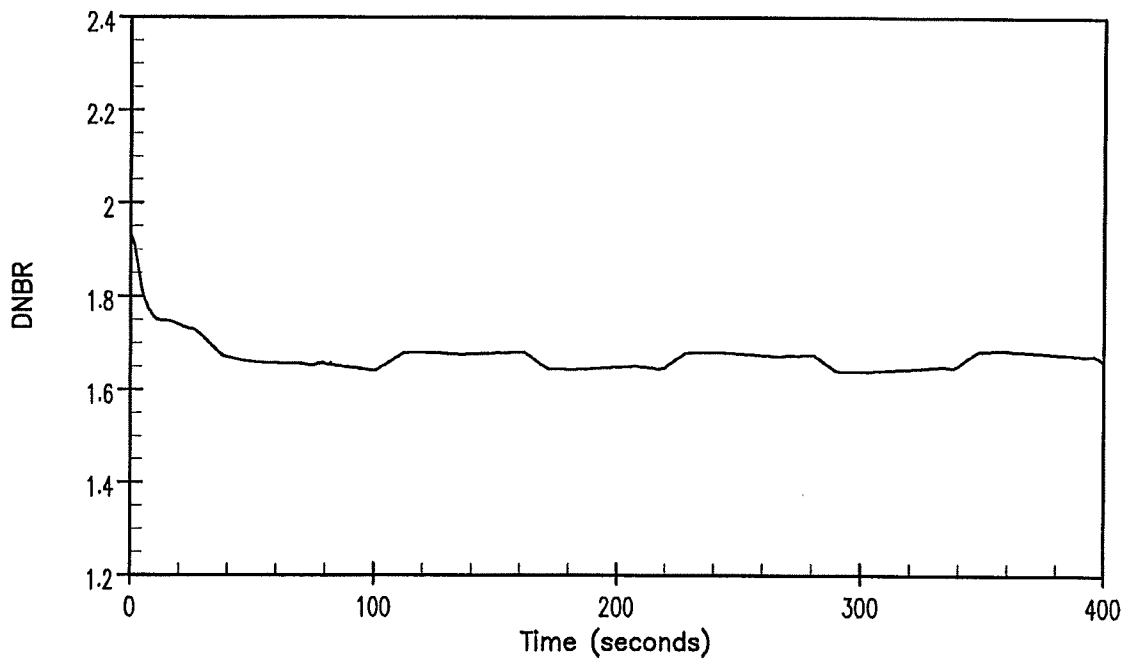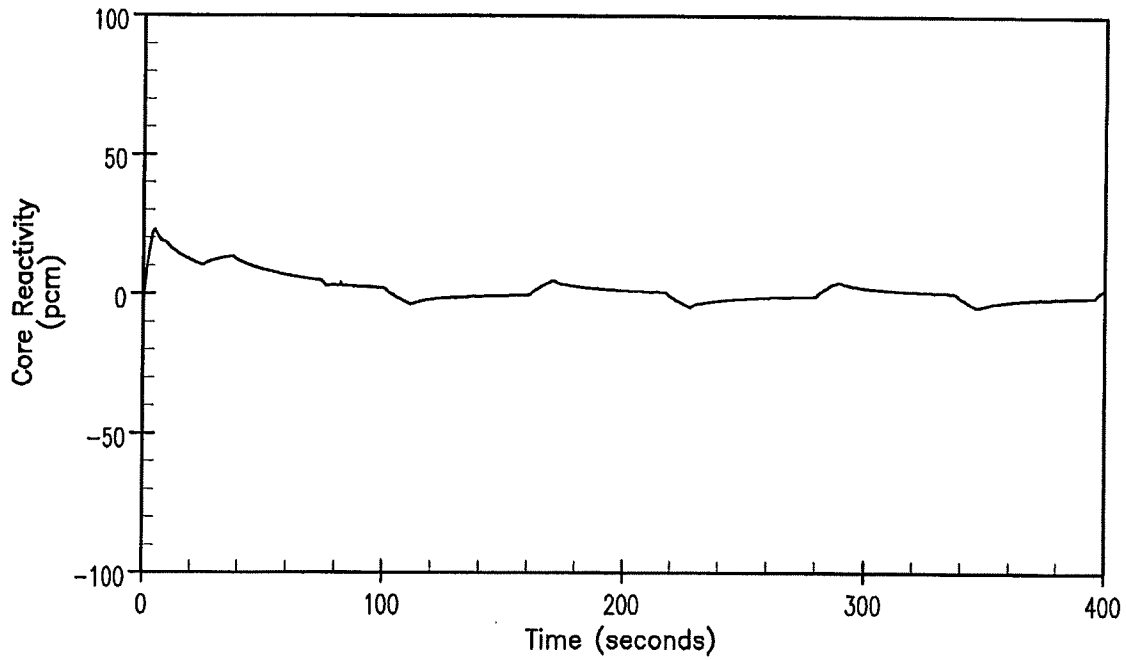
Figure 14.1.7-3 EXCESSIVE LOAD INCREASE EOL, MANUAL CONTROL
Sheet 1 of 4
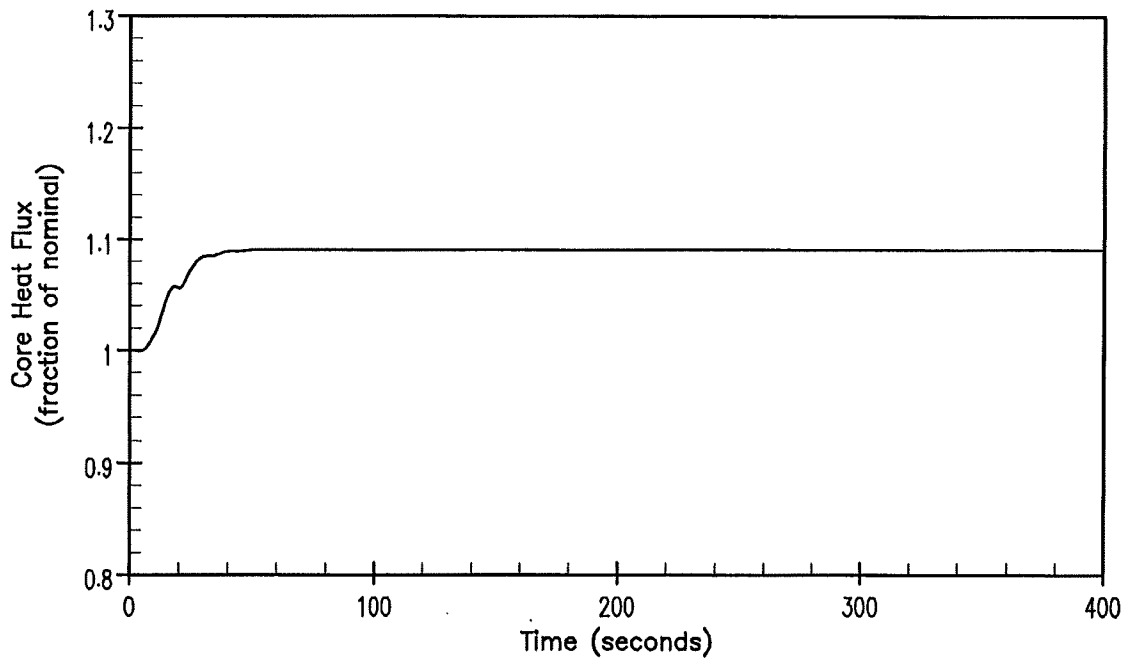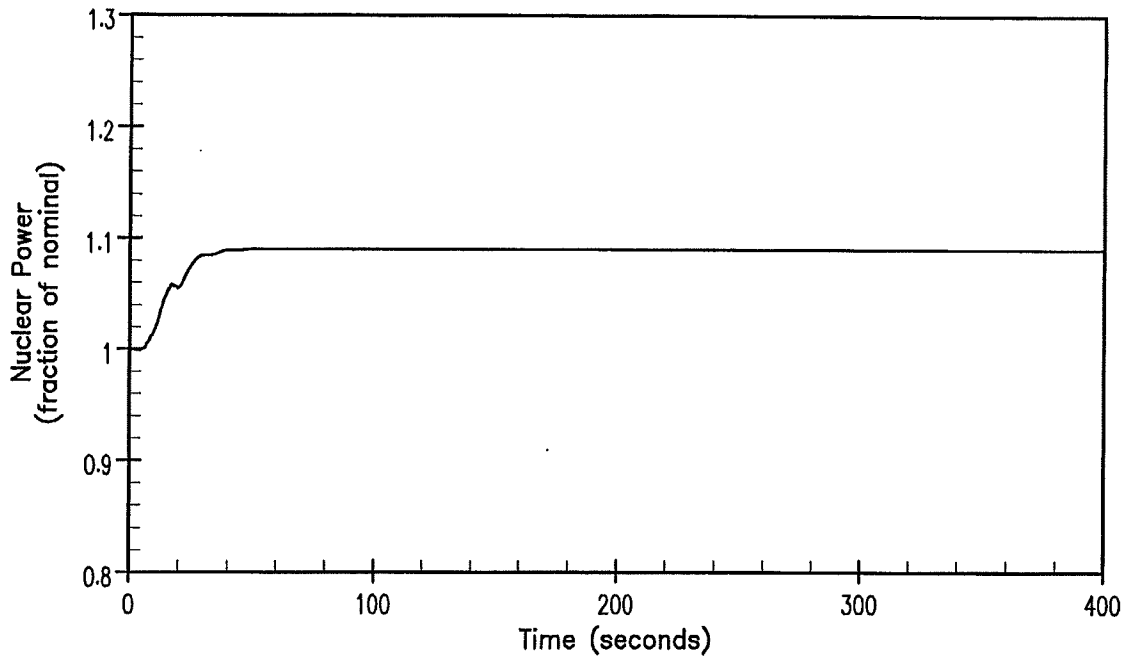
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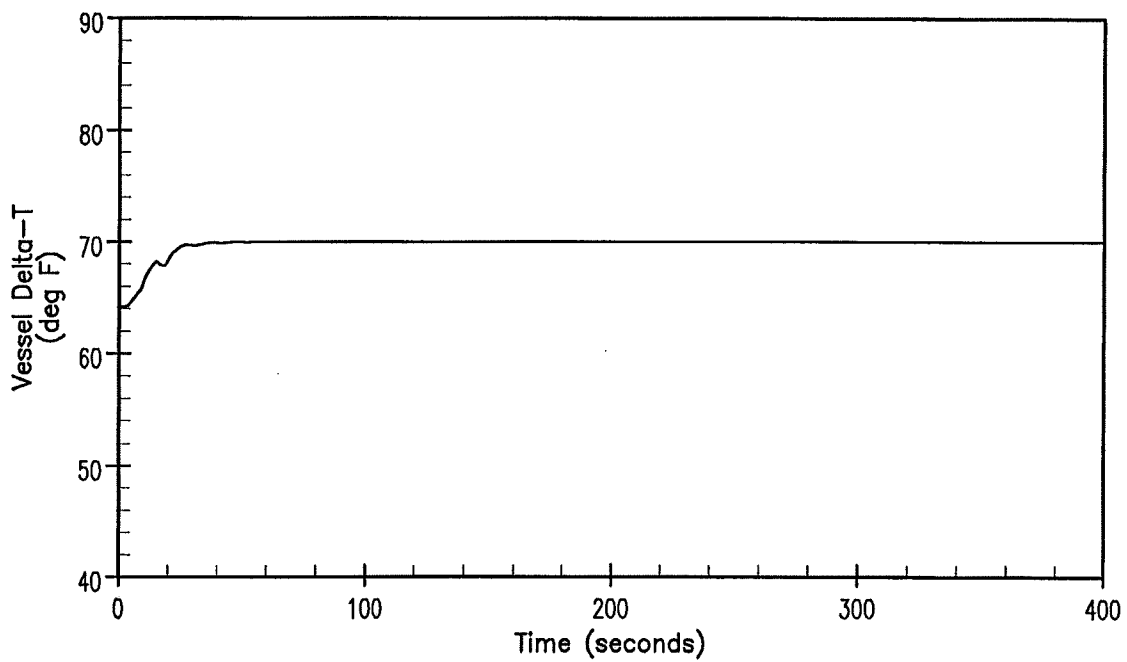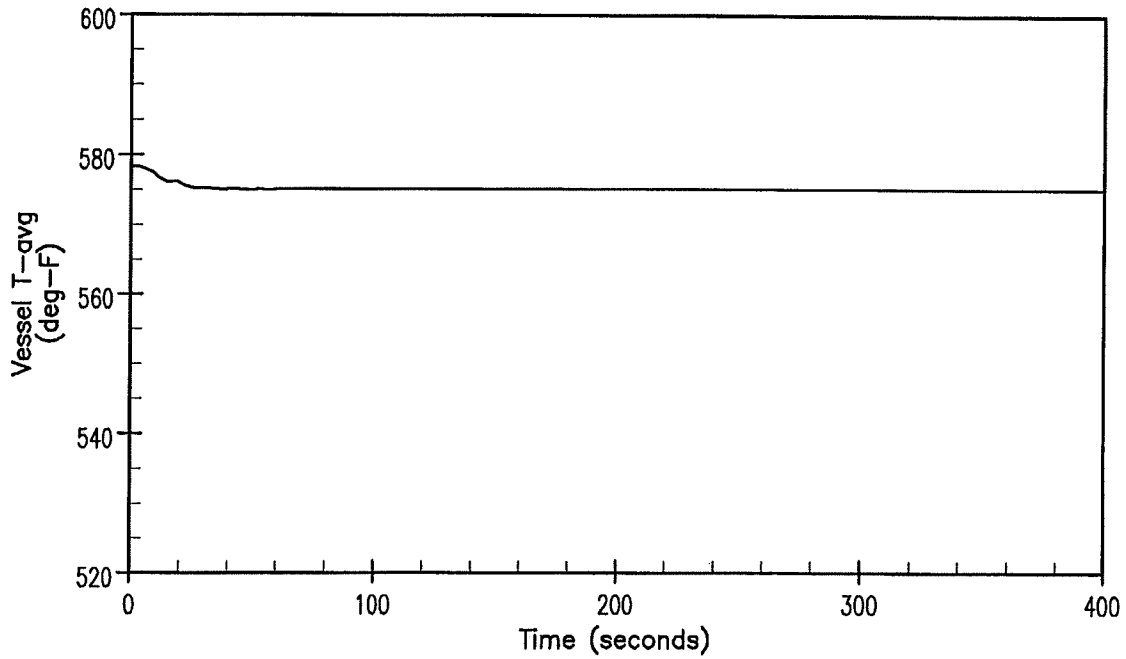
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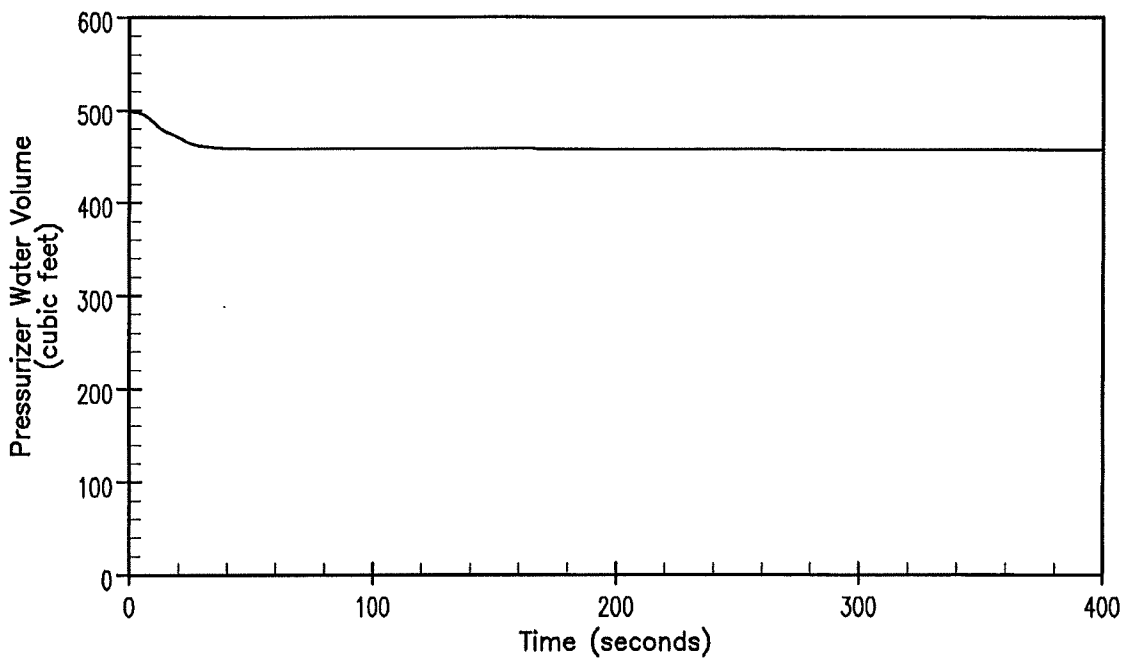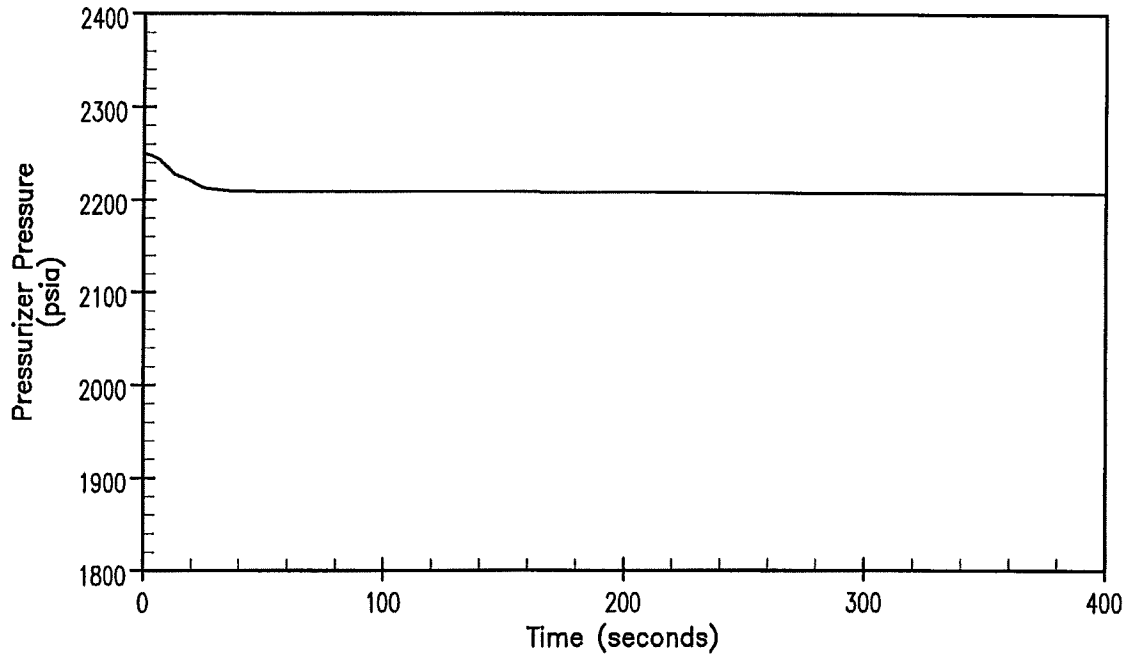
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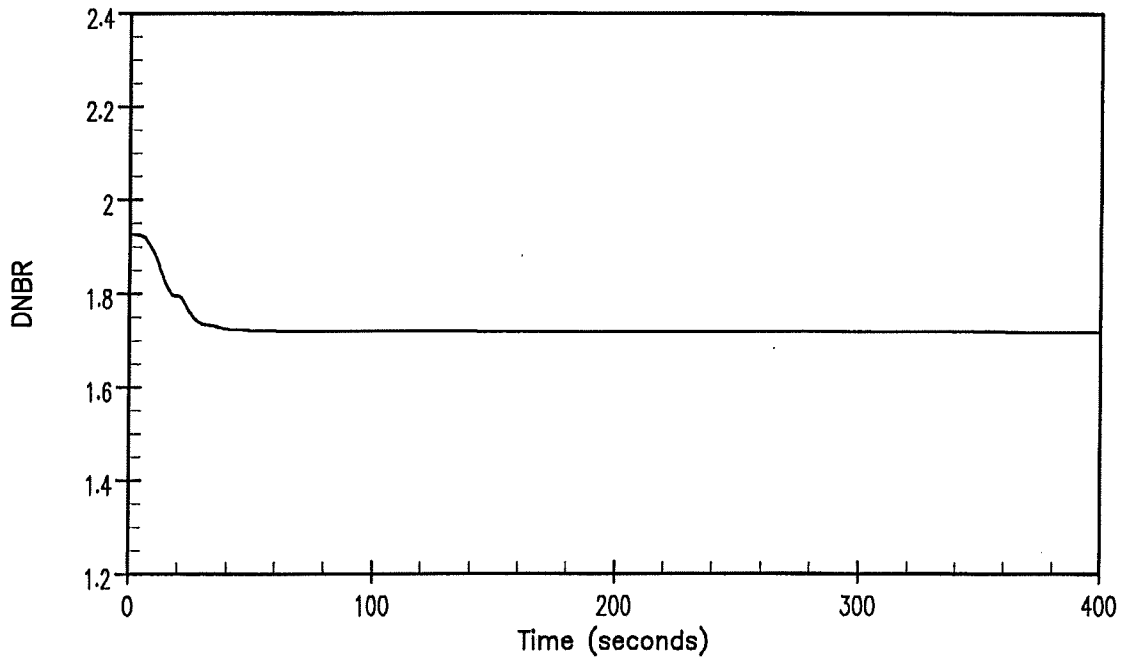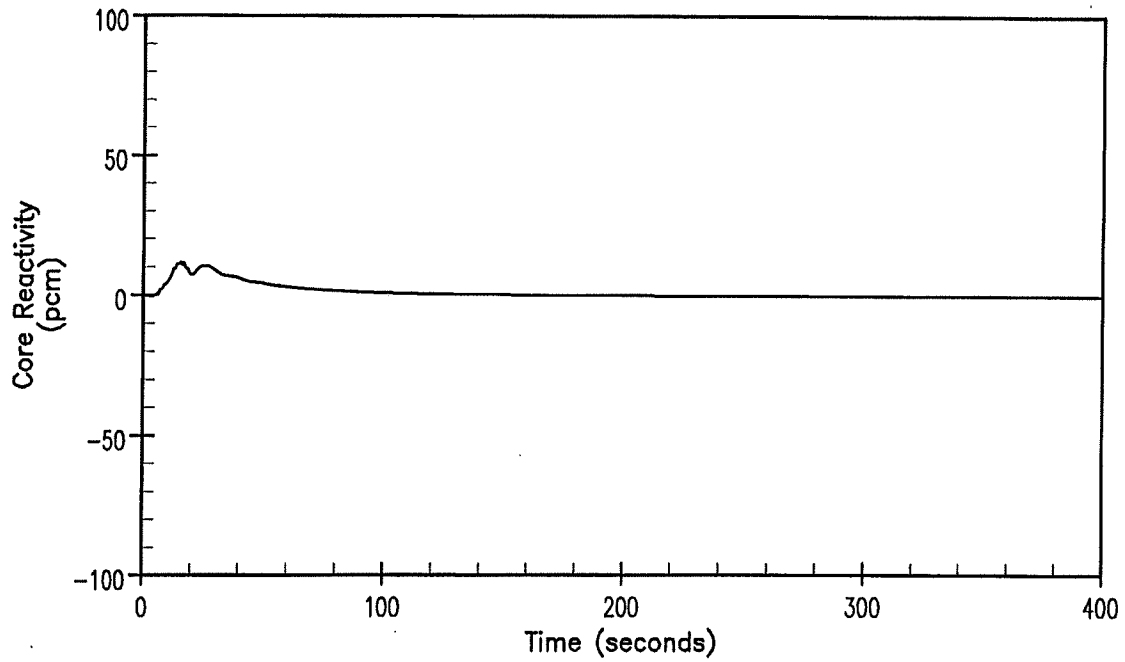
Figure 14.1.7-4 EXCESSIVE LOAD INCREASE EOL, AUTO CONTROL
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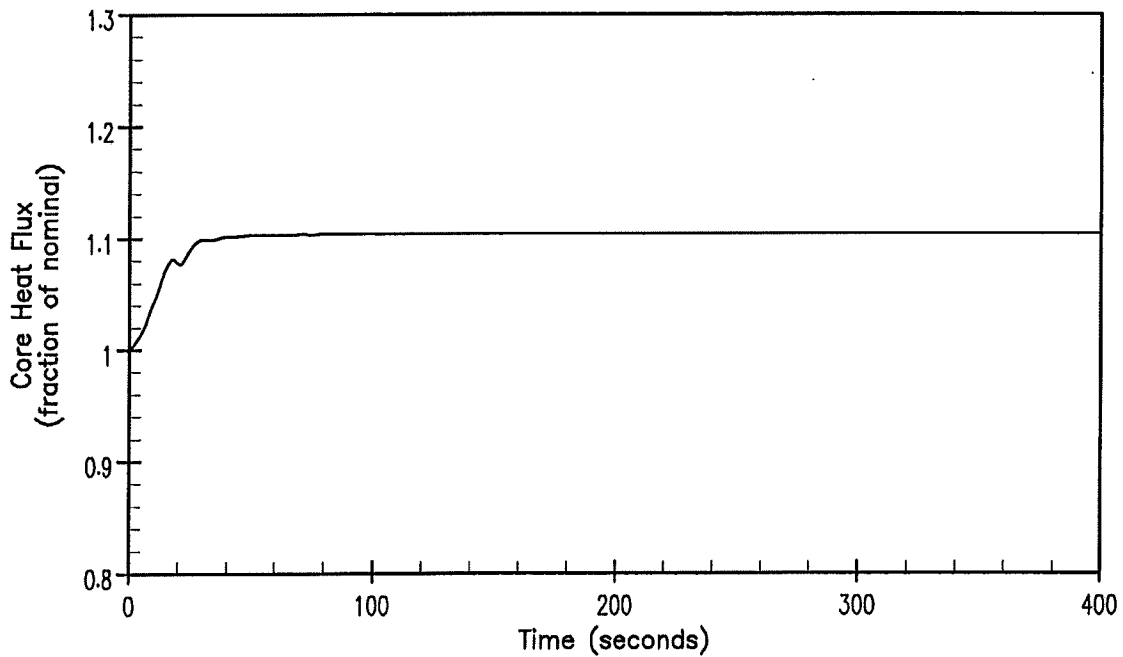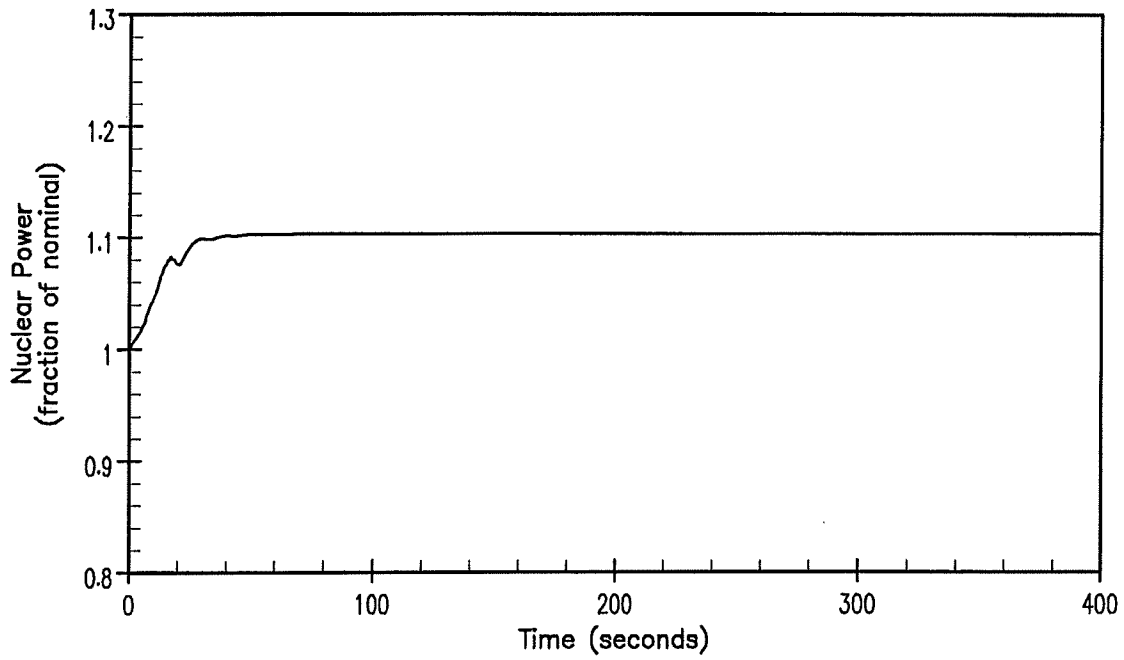
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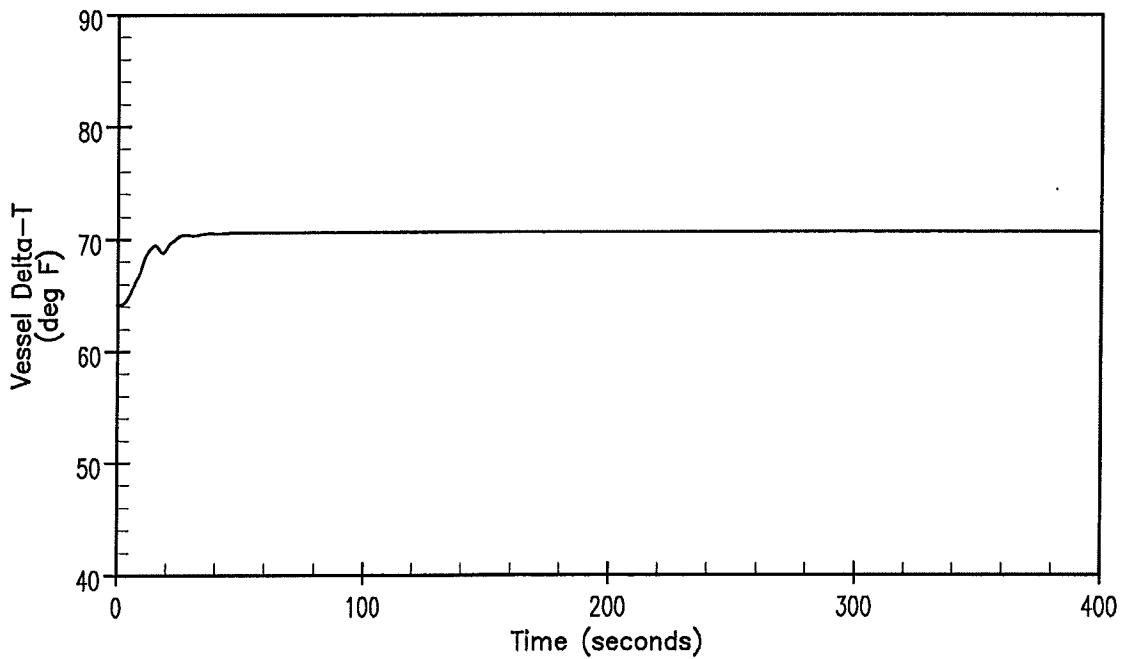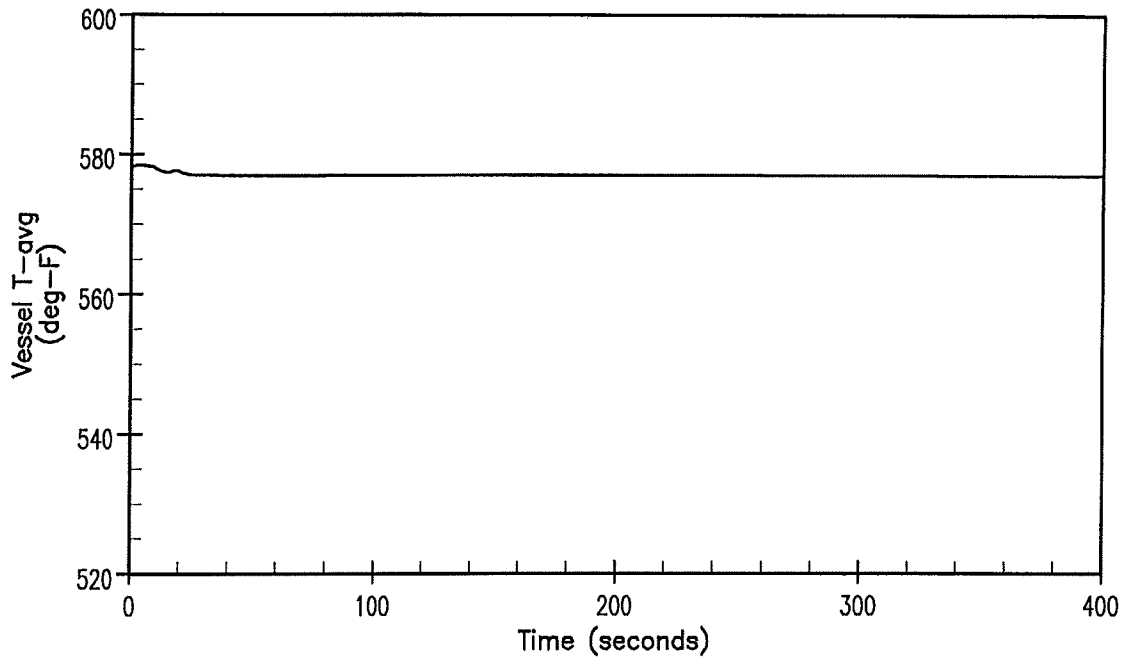
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Sheet 3 of 4
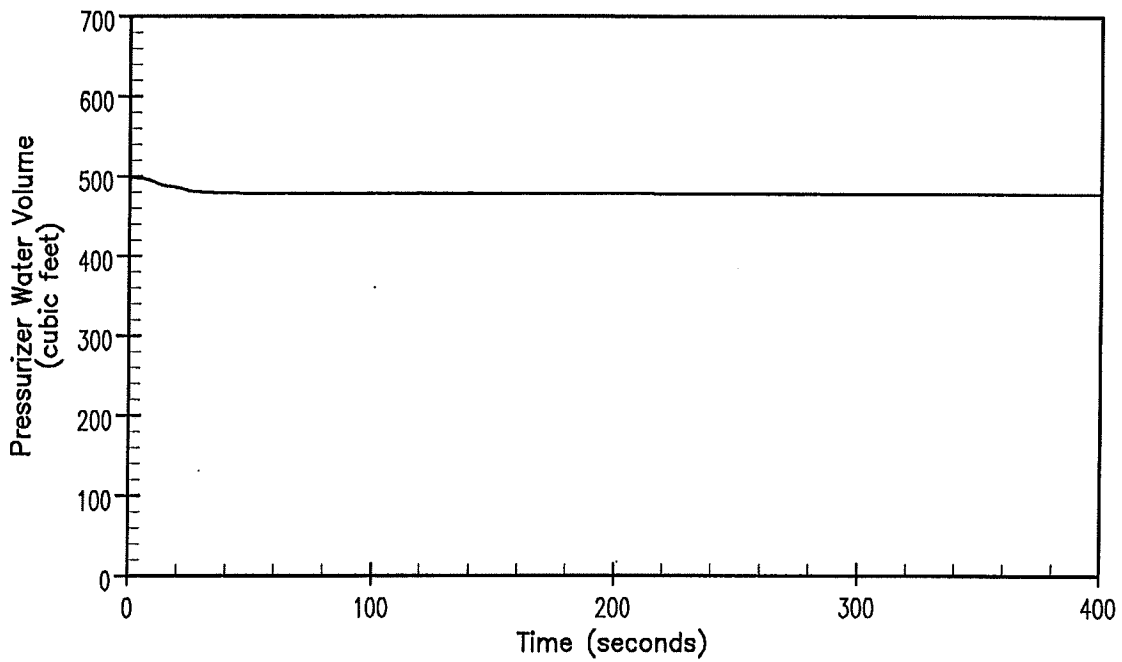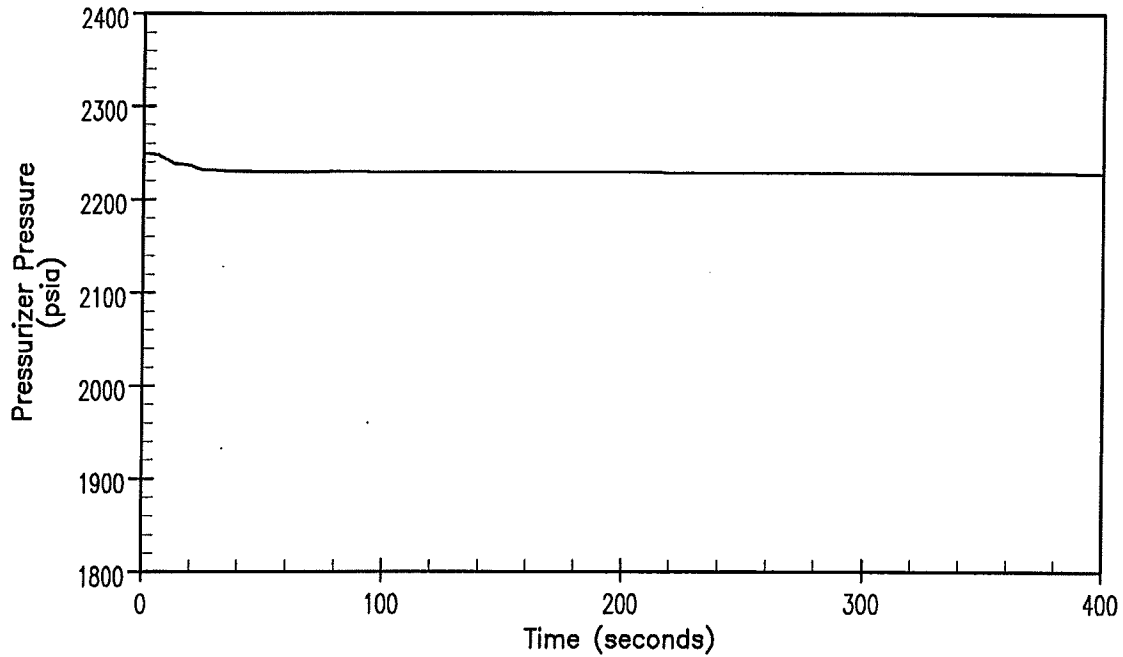
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Sheet 4 of 4
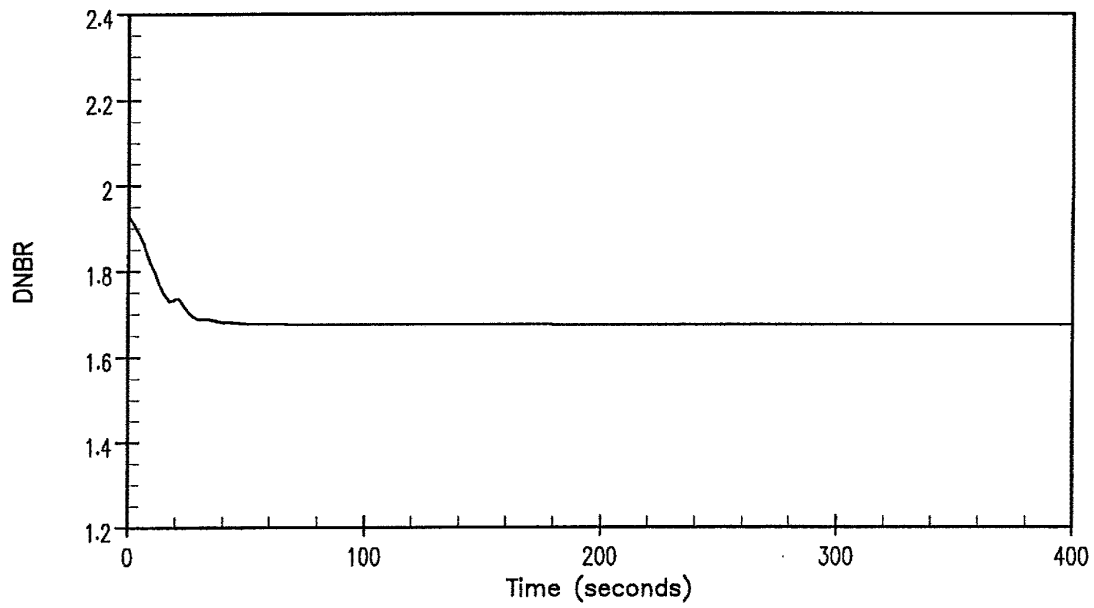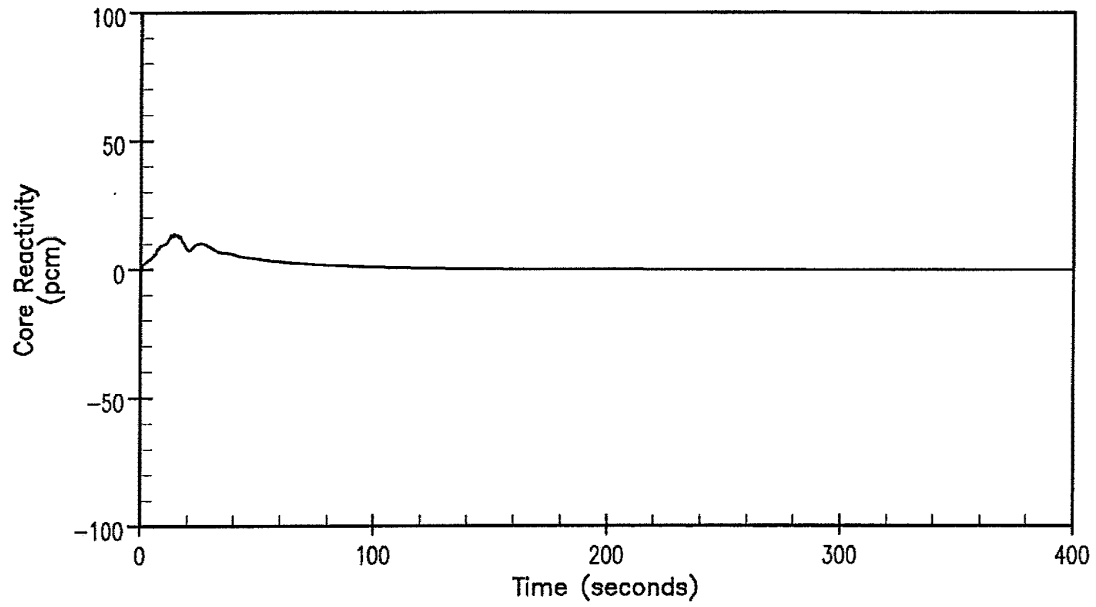
14.1.8   LOSS OF REACTOR COOLANT FLOW

Flow Coastdown Events

A loss of coolant flow incident can result from a mechanical or electrical failure in one or more reactor coolant pumps, or from a fault in the power supply to these pumps. If the reactor is at power at the time of the incident, the immediate effect of loss of coolant flow is a rapid increase in coolant temperature. This increase could result in departure from nucleate boiling (DNB) with subsequent fuel damage if the reactor is not tripped promptly. Trip circuits provide the necessary protection against a loss of coolant flow incident and are actuated by:

1.   Low voltage on pump power supply bus;

2.   Pump circuit breaker opening (low frequency on pump power supply bus opens pump circuit breaker); or

3.   Low reactor coolant flow.

These trip circuits and their redundancy are further described in Section 7.2, Reactor Protection System.

Frequency decay for both reactor coolant pumps during full power operation is the most severe credible loss-of-coolant flow condition. For this condition reactor trip together with flow sustained by the inertia of the coolant and rotating pump parts will be sufficient to prevent fuel failure, reactor coolant system overpressure and prevent the DNB ratio from going below the limit value.

Method of Analysis

The loss of flow analysis is performed for the following cases:

*   pump bus frequency decay (underfrequency) event,
*   loss of pump power supply voltage (undervoltage) event,
*   partial loss of flow (PLOF) event, and
*   complete loss of flow (CLOF) (reference case; bounds the undervoltage and partial loss of flow events).

The limiting loss of flow transient is the pump bus frequency decay (underfrequency) event with a 5 Hz/s frequency decay rate which conservatively does not credit the RCP underfrequency trip. The reactor trip occurs on a low flow signal. This underfrequency event is assumed to begin after a one-second null transient. The pumps slow down at a frequency decay rate of 5 Hz/s. The flow decreases as a response to the slower pump rotation throughout the transient and the reactor trips on a low flow signal at 87% of nominal flow.

A complete loss of flow (CLOF) reference case is analyzed for a loss of both RCPs with both reactor coolant loops in operation. This case bounds the undervoltage and the PLOF events. This CLOF event is assumed to initiate at 1.0 second and both RCPs start to coastdown. In order to bound the undervoltage event, the undervoltage trip is delayed until the low flow reactor trip signal is reached.

The normal power supplies for the pumps are the two buses connected to the generator, each of which supplies power to one of the two pumps. Following a turbine generator trip, the 19 kV main generator breaker opens. The auxiliaries on the 4.16 kV non-safeguards buses remain fed by the unit auxiliary transformer (X02) via the main transformer (X01). Therefore, the simultaneous loss of power to all reactor coolant pumps is a highly unlikely event.

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 one minute. Since both pumps are not on the same bus, a single bus fault would not result in the loss of both pumps.

The complete loss of flow transients were analyzed using the Westinghouse advanced 3-D methodology with three computer codes, linked by an external communication interface (Reference 3). The RETRAN code is used to calculate the RCS conditions versus time, including the reactor vessel, RCS loops, pressurizer, and steam generators. The RETRAN code also models the reactor trips, engineered safety feature (ESF) functions, and the RCS control functions. The SPNOVA code is used to perform steady-state and transient 3-D core neutronics calculations, using the VIPRE code to calculate the transient local coolant density and fuel effective temperature (Teff) for the core feedback calculations. The VIPRE code is used to calculate the local heat flux to the coolant in the RETRAN core model. The VIPRE code obtains its core inlet conditions (core inlet flow and temperature) and core exit pressure from the RETRAN calculation. Using boundary conditions from the core feedback calculations (SPNOVA/ RETRAN/VIPRE linked run), the VIPRE code is also used in a separate hot rod calculation to determine the minimum DNBR versus time.

Initial Operating Conditions

Initial reactor power, RCS temperature and pressure are assumed to be at the most limiting nominal conditions, i.e. 100% power, maximum RCS temperature, and reduced pressure operation. Uncertainties in initial conditions are included in the DNBR limits as described in Reference 2.

Initial Core Conditions

The loss of flow analysis was performed at Beginning of Cycle (BOC) Hot Full Power (HFP) conditions. The analysis used minimum moderator temperature feedback, maximum Doppler feedback and a maximum value of the delayed neutron fraction. The control rods were initially assumed to be at their fully withdrawn position to minimize the initial rate of reactivity insertion following a reactor trip. A conservative rod position vs. time curve was assumed for the reactor trip.

Flow Coastdown

The flow coastdown analysis is based on a momentum balance around each reactor coolant loop and across the reactor core. This momentum balance is combined with the continuity equation, a pump momentum balance and the pump characteristics and is based on high estimates of system pressure losses.

No single active failure in the plant systems and equipment which are necessary to mitigate the effects of the accident will adversely affect the consequences of the accident during the transient.

Fuel Type and SG Tube Plugging Level

The loss of coolant flow analysis is performed to bound operation with 422V+ fuel and an effective (i.e. sleeved and/or plugged) uniform steam generator tube plugging level of up to 10% for Units 1 and 2, with a maximum loop-to-loop steam generator tube plugging asymmetry of 10%.

Results

The limiting loss of flow event is the pump bus frequency decay (underfrequency) event. The transient results for this limiting case are presented in Figure 14.1.8-1. The transient results for the complete loss of flow case (bounding both the undervoltage and partial loss of flow cases) are presented in Figure 14.1.8-2. The sequence of events for the limiting frequency decay case and the reference complete loss of flow case are presented in Table 14.1.8-1.

Conclusions

Since the minimum DNBR remains above the design DNBR limit for all cases, there is no cladding damage and no release of fission products into the reactor coolant. Therefore, once the fault is corrected, the plant can be returned to service in the normal manner. The absence of fuel failures would, of course, be verified by analysis of reactor coolant samples.

Locked Rotor Accident

A hypothetical transient analysis is performed for the postulated instantaneous seizure of a reactor coolant pump rotor. Flow through the reactor coolant system is rapidly reduced, leading to a reactor trip on a low-flow signal. Following the trip, heat stored in the fuel rods continues to pass into the core coolant, causing the coolant to heat up and expand. At the same time, heat transfer to the shell side of the steam generator 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 coolant in the reactor core, combined with the reduced heat transfer in the steam generator causes an insurge into the pressurizer and a pressure increase throughout the reactor coolant system. The insurge into the pressurizer compresses the steam volume, actuates the automatic spray system, opens the power-operated relief valves, and opens the pressurizer safety valves, in that sequence. The two 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 is not included in the peak RCS pressure and peak cladding temperature (PCT) cases.

There are no credible sources of shaft seizure other than impeller rubs. Any seizure of the pump bearing is precluded by the graphite in the bearing. Any seizure in the seals results in a shearing of the anti-rotation pin in the seal ring. An inadvertent actuation of the shut down seal (SDS) on a rotating assembly will not have any measurable impact on RCP coastdown or on the pump's capability to provide sufficient cooling flow to the reactor core. The motor has adequate power to continue pump operation even after the above occurrences. Indications of pump malfunction in these conditions are first, by high-temperature signals from the bearing water temperature detector and second, by excessive No. 1 seal leakoff indications. Along with these signals, pump vibration levels are checked. When there are indications of a serious malfunction, the pump is shut down for investigation.

Method of Analysis

The locked rotor transients were analyzed using the Westinghouse advanced 3-D methodology with three computer codes, linked by an external communication interface (Reference 3). The RETRAN code is used to calculate the RCS conditions versus time, including the reactor vessel, RCS loops, pressurizer and steam generators. The RETRAN code also models the reactor trips, engineered safety feature (ESF) functions, and the RCS control functions. The SPNOVA code is used to perform steady-state and transient 3-D core neutronics calculations, using the VIPRE code to calculate the transient local coolant density and fuel effective temperature (Teff) for the core feedback calculations. The VIPRE code is used to calculate the local heat flux to the coolant in the RETRAN core model. The VIPRE code obtains its core inlet conditions (core inlet flow and temperature) and core exit pressure from the RETRAN calculation. Using boundary conditions from the core feedback calculations (SPNOVA/RETRAN/VIPRE linked run), the VIPRE code is also used in separate hot rod calculations to determine the minimum DNBR versus time and peak cladding temperature (PCT).

There were three locked rotor cases analyzed: one to determine the percentage of rods-in-DNB, a second to determine the peak RCS pressure and a third to determine the PCT.

The first case was run to establish the percentage of rods-in-DNB in support of the radiological analysis. One locked rotor and shaft break was assumed with both reactor coolant loops in operation. This case made assumptions designed to maximize the number of rods-in-DNB. Initial core power was assumed to be at its nominal value consistent with steady-state, full-power operation. The reactor coolant system pressure and vessel average temperature were assumed to be at their nominal values. Minimum Measured Flow (MMF) was also assumed. Uncertainties in initial conditions were accounted for in the DNBR limit value as described in the Revised Thermal Design Procedure (RTDP) (Reference 2). The pressure-reducing effects of the PORVs and the automatic pressurizer spray system were modeled in the rods-in-DNB analysis for conservatism.

The second and third cases were performed to evaluate the peak RCS pressure and PCT. As in the rods-in-DNB case, one locked rotor and shaft break was assumed with both reactor coolant loops in operation. These cases made assumptions designed to maximize the RCS pressure and cladding temperature, using the Standard Thermal Design Procedure (STDP). Initial core power, reactor coolant temperature, and pressure include allowances for calibration and instrument errors. Thermal Design Flow (TDF) was also assumed. The pressure-reducing effects of the pressurizer PORVS and automatic pressurizer spray system were not modeled.

The pressure response shown in Figure 14.1.8-3 is the response at the point in the reactor coolant system having the maximum pressure.

Evaluation of the Pressure Transient - The locked rotor peak RCS pressure and PCT calculations were performed at Beginning of Cycle (BOC) Hot Full Power (HFP) conditions. The analyses used minimum moderator temperature feedback and maximum Doppler feedback. The analyses assumed a maximum value of the delayed neutron fraction. The control rods were initially assumed to be at their fully withdrawn position to minimize the initial rate of reactivity insertion following a reactor trip. A conservative rod position vs. time curve was assumed for the reactor trip.

Since the Locked Rotor peak RCS pressure and PCT cases are analyzed using the Standard Thermal Design Procedure (STDP), the analysis was performed using a +0.6% uncertainty in the initial reactor power, a ±6.4°F combined uncertainty and bias in RCS temperature, and a +50 psi uncertainty in pressurizer pressure. The RCS flow rate was set to the Thermal Design Flow (TDF).

No credit is taken for the pressure-limiting effects of the pressurizer PORVs, pressurizer spray, steam dump or controlled feedwater flow after plant trip. Although these operations are expected to occur and would result in a lower peak pressure, an additional degree of conservatism is provided by ignoring their effects.

The lift pressure of the pressurizer safety valves is assumed to be 3.4% above the nominal set pressure of 2500 psia, including +0.9% set pressure shift due to the presence of pressurizer safety valve loop seals (Reference 3). The safety valve steam relief capacity is 288,000 lbm/hr per valve.

The accident was initiated by causing an immediate halt in the rotational speed of one RCP. A loss of offsite power was conservatively assumed to occur at the time of reactor trip (control rod release), causing the unaffected RCP to lose power and coast down freely. Reactor trip occurs on the low flow reactor trip function at 87% flow with a trip delay time of 1.0 second.

A separate VIPRE hot rod calculation to determine the peak cladding temperature is performed assuming that the hot rod is experiencing DNB throughout the flow transient. The initial hot rod power was increased such that the initial hot spot power was at the plant F_Q limit.

Evaluation of Departure from Nucleate Boiling in the Core During the Accident - Since the locked rotor rods-in-DNB evaluation is analyzed using the Revised Thermal Design Procedure (Reference 2), the core feedback calculations were performed using nominal HFP conditions for reactor power, RCS average temperature, and pressurizer pressure. The RCS flow rate was set to the Minimum Measured Flow (MMF). All other RCS initial conditions (pressurizer water volume, steam generator level, etc.) were also set to nominal conditions.

The VIPRE code was used in a separate time-dependent DNBR calculation to determine the number of rods-in-DNB. The DNBR calculation was based on the core average power, power distribution, inlet temperature, core inlet flow, and core exit pressure vs. time. The core average power and power distributions were obtained from the core feedback calculations, including the time-dependent changes in radial enthalpy rise hot channel factor (FDH) and the axial power distributions. The design pin-by-pin radial power distribution, with the peak rod power raised to a value consistent with the limit allowed by the plant Technical Specifications, was used as the initial condition for the DNBR calculations. The reactor coolant conditions (inlet temperature, core inlet flow and core exit pressure vs. time) were obtained from the core feedback calculations.

Film Boiling Coefficient - The film boiling coefficient is calculated in the VIPRE code using the Bishop-Sandberg-Tong film boiling correlation. The fluid properties are evaluated at film temperature, which is the average between the wall and bulk temperatures. The program calculates the film coefficient at every time step, based on the actual heat transfer conditions at the time. The nuclear power, system pressure, bulk density, and mass flow rate as a function of time were based on the core feedback calculations.

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

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

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

where:

w	=	amount reacted (mg/cm^2)
t	=	time (seconds)
T	=	temperature (°K).

The reaction heat is 1510 cal/gm.

Results

Figure 14.1.8-3 shows the core flow and loop flow transients, the nuclear power and maximum pressure transients, the average channel heat flux transient, and the cladding temperature transient. The results of these calculations are summarized in Table 14.1.8-2. The sequence of events is shown in Table 14.1.8-1.

Conclusions

Since the peak cladding surface temperature calculated for the hot spot during the more severe transient remains considerably less than the PCT non-LOCA limit of 2700 °F for ZIRLO® and 2375 °F for Optimized ZIRLO^{TM} and the amount of zirconium-water reaction is small, the core remains in place and intact with no consequential loss of core cooling capability.

Since the peak cladding surface temperature calculated for the hot spot during the more severe transient remains considerably less than the PCT non-LOCA limit of 2700 °F and the amount of zirconium-water reaction is small, the core remains in place and intact with no consequential loss of core cooling capability.

Radiological Consequence of the Locked Rotor Accident

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 is assumed to occur as a result of the reduced flow. Due to the pressure differential between primary and secondary systems, and assumed steam generator tube leaks, 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 dump valves or main steam safety valves. In addition, it is postulated that some of the activity contained in the secondary coolant prior to the accident is released to atmosphere as a result of steaming of the steam generators following the accident.

This section describes the assumptions and analyses performed to determine the amount of radioactivity released and the offsite and control room doses resulting from the release. The specific analyses conducted for the PBNP dose consequences were accepted by the NRC (Reference 6).

Input Parameters and Assumptions

The analysis of the locked rotor radiological consequences uses the analytical methods and assumptions outlined in the RG 1.183 (Reference 4).

It is conservatively assumed that 30% 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. The fuel clad gap activity fractions from Table 3 of RG 1.183 are applied in the analysis. The damaged rods are high power first or second cycle rods which meet the burnup criteria of RG 1.183, Table 3, Footnote 11. The activity released from the damaged fuel reflects a radial peaking factor of 1.7.

The concentrations of iodines and noble gasses in the RCS at time the accident occurs are based on the Technical Specification limits of 0.5 µCi/gm of dose equivalent (DE) 1-131 and 520 µCi/gm of DE Xe-133. The DE Xe-133 value is a pre-EPU limit, and is conservative. The alkali metal concentration in the RCS is based on the fuel defect level that corresponds to 0.5 µCi/gm of DE 1-131. The iodine activity concentration of the secondary coolant at the time the accident occurs is assumed to be equivalent to the Technical Specification limit of 0.1 µCi/gm of DE 1-131. The alkali metal activity concentration of the secondary coolant at the time the accident occurs is assumed to correspond to 0.1 µCi/gm of DE 1-131. The core activity and equilibrium nuclide concentrations are presented in Table 14.1.8-4.

An accident-induced primary-to-secondary leak rate of 1000 gm/min per SG is assumed for the duration of the accident.

An iodine partition factor in the SGs of 0.01 (curies iodine/gm steam) / (curies iodine/gm water) is used. Per RG 1.183, the retention of particulates in the SG is limited by moisture carry over which is modeled by a retention factor of 0.0025. This is the estimated full power moisture carryover fraction. All noble gas activity transferred to the secondary side of the SG through SG tube leakage is assumed to be directly released to the outside atmosphere. Plant cooldown to RHR operating conditions can be accomplished within 14 hours after initiation of the event. At 30 hours after the accident the RHR system is assumed to be placed into service, after which there is no further steam release to the atmosphere from the secondary system.

The specific assumptions applied to PBNP are summarized in Table 14.1.8-3, Table 14.1.8-4 and Table 14.1.8-5. The dose conversion factors, breathing rates, and atmospheric dispersion factors used in the dose calculations are given in Table 14.1.8-3. The core and coolant activities used in the radiological calculations are given in Table 14.1.8-4. The remaining major assumptions and parameters used specifically in the locked rotor analysis are itemized in Table 14.1.8-5.

Control Room Model

For the locked rotor accident it is assumed that the HVAC system begins in Mode 1 (normal operating Mode). The dose rates in the control room trip the control room monitors within 1 minute, switching the system to Mode 5 (emergency Mode) where it remains throughout the event. The parameters associated with the control room HVAC Modes assumed for the locked rotor accident are summarized in Table 14.1.8-6. FSAR 9.8 provides a complete description of the control room HVAC system.

Acceptance Criteria

The Standard Review Plan (SRP) 15.0.1 (Reference 5) offsite dose acceptance criterion for a locked rotor is 2.5 rem TEDE, which is 10% of the 10 CFR 50.67 limit of 25 rem TEDE. The control room personnel dose acceptance criterion is 5 rem TEDE per 10 CFR 50.67.

Results/Conclusions

The results of the offsite and control room dose analyses are provided in Table 14.1.8-2, and indicate that the acceptance criteria are met. The exclusion area boundary doses reported are for the worst 2 hour period, determined to be from 28 to 30 hours.

REFERENCES

1.   Goldberg, G., Westinghouse Electric Corporation, letter to E. J. Lipke, "Wisconsin Electric Power Company Point Beach Units 1 and 2 Final Reports for RCP Bus Frequency Decay Analysis," WEP-91-196, August 19, 1991.

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

3.   Beard, L.C., et al., "Westinghouse Methodology for Application of 3-D Transient Neutronics to Non-LOCA Accident Analysis," WCAP-16259-P-A (Proprietary) and WCAP-16259-NP-A (Non-Proprietary), August 2006.

4.   USNRC, Regulatory Guide 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," July 2000.

5.   US NRC, NUREG 0800, Standard Review Plan (SRP) Section 15.0.1, "Radiological Consequence Analyses Using Alternative Source Terms," July 2000.

6.   NRC Safety Evaluation 2011-0003, "Issuance of License Amendments Regarding Use of Alternate Source Term," April 14, 2011.

7.   NRC Safety Evaluation 2011-0004, "Issuance of License Amendments Regarding Extended Power Uprate," dated May 3, 2011.

Table 14.1.8-1 LOSS OF FORCED REACTOR COOLANT FLOW TIME SEQUENCE OF
EVENTS

Case	Event	Time (Seconds)
Complete Loss of Forced Reactor Coolant Flow	Transient begins	0.00
	Both operating RCPs lose power and begin coasting down	1.00
	RCP undervoltage trip setpoint reached	1.00
	Low flow reactor trip setpoint reached	2.93
	Rods begin to drop	3.93
	Minimum DNBR occurs	4.60
Underfrequency Event	Transient begins	0.00
	Frequency decay begins and RCPs begin to decelerate	1.00
	Low RCS flow reactor trip setpoint reached	2.76
	Rods begin to drop	3.76
	Minimum DNBR occurs	4.55
Locked RCP Rotor	Transient begins	0.00
	Rotor on one RCP locks	1.00
	Low RCS flow reactor trip setpoint reached	1.1
	Rods begin to drop	2.1
	Remaining pump loses power and begins to coast down	2.1
	Maximum cladding temperature occurs	4.5
	Maximum RCS pressure occurs	5.1

Table 14.1.8-2 SUMMARY OF LIMITING RESULTS FOR LOCKED ROTOR ACCIDENT

Reactor Plant Results

Maximum Reactor Coolant System Pressure	2653 psia
Maximum Cladding Temperature at Core Hot Spot	1810°F
Zr-H$_2$O Reaction at Core Hot Spot	0.4% by weight
Rods in DNB	25%

Radiological Results

Site Boundary (28 - 30 hr)	2.0 rem TEDE
Low Population Zone (0 - 30 hr)	0.5 rem TEDE
Control Room	4.6 rem TEDE

Table 14.1.8-3 ASSUMPTIONS USED FOR DOSE ANALYSES

RCP LOCKED ROTOR ACCIDENT (14.1.8)
STEAM GENERATOR TUBE RUPTURE ACCIDENT (14.2.4)
MAIN STEAM LINE BREAK ACCIDENT (14.2.5)
CONTROL ROD EJECTION ACCIDENT (14.2.6)

DOSE CONVERSION FACTORS, BREATHING RATES, ATMOSPHERIC DISPERSION FACTORS

Isotope	Committed Effective Dose Equivalent (Sv/Bq)	Effective Dose Equivalent (Sv-m3/Bq-sec)
I-130	7.14E-10	1.04E-13
I-131	8.89E-9	1.82E-14
I-132	1.03E-10	1.12E-13
I-133	1.58E-9	2.94E-14
I-134	3.55E-11	1.30E-13
I-135	3.32E-10	7.98E-14
Kr-85m	NA	7.48E-15
Kr-85	NA	1.19E-16
Kr-87	NA	4.12E-14
Kr-88	NA	1.02E-13
Xe-131m	NA	3.89E-16
Xe-133m	NA	1.37E-15
Xe-133	NA	1.56E-15
Xe-135m	NA	2.04E-14
Xe-135	NA	1.19E-14
Xe-138	NA	5.77E-14
Cs-134	1.25E-8	7.57E-14
Cs-136	1.98E-9	1.06E-13
Cs-137	8.63E-9	2.88E-14
Cs-138	2.74E-11	1.21E-13
Rb-86	1.79E-9	4.81E-15

Time Period	Breathing Rate (m^3 / second)
0 - 8 hr	3.5E-4*
8 - 24 hr	1.8E-4
24 - 720 hr	2.3E-4

Location	Atmospheric Dispersion Factors (second / m^3)
Site Boundary	5.0E-4*
Low Population Zone	
0 - 8 hr	3.0 E-5
8 - 24 hr	1.6 E-5
24 - 96 hr	4.2 E-6
96 -720 hr	8.6 E-7

* The breathing rate and site boundary atmospheric dispersion factor are held constant at the initial value for all time intervals in the determination of the limiting 2-hour period for the site boundary.

Table 14.1.8-4 ASSUMPTIONS USED FOR DOSE ANALYSES

RCP LOCKED ROTOR ACCIDENT (14.1.8)
STEAM GENERATOR TUBE RUPTURE ACCIDENT (14.2.4)
MAIN STEAM LINE BREAK ACCIDENT (14.2.5)
CONTROL ROD EJECTION ACCIDENT (14.2.6)

CORE AND COOLANT ACTIVITIES

Isotope	Total Core Activity at Shutdown (Ci)	Coolant Activity (µCi/gm)*
I-130	1.05E+06	2.89E-03
I-131	5.10E+07	3.77E-01
I-132	7.47E+07	4.24E-01
I-133	1.06E+08	6.55E-01
I-134	1.19E+08	9.97E-02
I-135	1.01E+08	3.76E-01
Kr-85m	1.36E+07	1.58E+00
Kr-85	6.15E+05	7.63E+00
Kr-87	2.68E+07	1.05E+00
Kr-88	3.60E+07	2.92E+00
Xe-131m	5.55E+05	2.35E+00
Xe-133m	3.21E+06	3.80E+00
Xe-133	1.02E+08	2.12E+02
Xe-135m	2.20E+07	4.33E-01
Xe-135	2.17E+07	6.72E+00
Xe-138	9.05E+07	5.77E-01
Cs-134	9.52E+06	3.29E-01
Cs-136	2.14E+06	3.44E-01
Cs-137	6.27E+06	2.79E-01
Cs-138	9.89E+07	1.62E-01
Rb-86	9.95E+04	3.64E-03

* Iodine activity corresponds to 0.5 µCi/gm dose equivalent (DE) I-131, noble gas activity is based on 520 µCi/gm DE Xe-133 and alkali metal activity is based on the fuel defect level that corresponds to 0.5 µCi/gm DE I-131. Values for other assumed DEs are multiples of these values.

Table 14.1.8-5 ASSUMPTIONS USED FOR DOSE ANALYSES
(Page 1 of 2)
RCP LOCKED ROTOR ACCIDENT

PARAMETER	VALUE
Initial Power	1811 MWt
Fraction of Fuel Rods in Core Assumed to Fail for Dose Considerations	30% of Core
Gap Fractions	
I-131	0.08
Kr-85	0.10
Other Iodines and Noble Gases	0.05
Alkali Metals	0.12
Radial Peaking Factor	1.7
RCS Activity Prior to Accident	
Iodine	0.5μCi/gm of DE I-131
Noble Gas	520 μCi/gm of DE Xe-133
Alkali Metals	Corresponds to 0.5 μCi/gm of DE I-131
Secondary Coolant Activity Prior to Accident	
Iodine	0.1 μCi/gm of DE I-131
Alkali Metals	Corresponds to 0.1 μCi/gm of DE I-131
Total SG Tube Leak Rate During Accident	2000 gm/min
Steam Release to Environment	See page 2
SG Iodine Partition Factor	0.01
SG Alkali Metal Retention Factor	0.0025
Iodine Species Released to the Atmosphere	
Elemental	97%
Organic	3%
RCS Mass	1.06E8 gm
Secondary Side Mass	
0-2 hours	5.98E7 gm
> 2 hours	7.37E7 gm

Table 14.1.8-5 ASSUMPTIONS USED FOR DOSE ANALYSES
(Page 2 of 2)

LOCKED ROTOR DOSE ANALYSIS STEAM RELEASE TO ENVIRONMENT

Hours	Mass (lbm)	Hours	Mass (lbm)
0-2	213,295	16-17	37,245
2-3	75,645	17-18	36,821
3-4	70,441	18-19	36,116
4-5	65,672	19-20	35,461
5-6	63,060	20-21	34,969
6-7	60,838	21-22	34,969
7-8	58,305	22-23	34,109
8-9	57,015	23-24	33,595
9-10	55,886	24-25	33,595
10-11	54,629	25-26	33,595
11-12	53,326	26-27	33,595
12-13	52,514	27-28	33,595
13-14	51,714	28-29	33,595
14-15	38,508	29-30	33,595
15-16	37,749		

Table 14.1.8-6 CONTROL ROOM PARAMETERS USED FOR DOSE ANALYSES

RCP LOCKED ROTOR ACCIDENT (14.1.8)
STEAM GENERATOR TUBE RUPTURE (14.2.4)
MAIN STEAM LINE BREAK ACCIDENT (14.2.5)
CONTROL ROD EJECTION ACCIDENT (14.2.6)

Volume	65,243 ft^3
Control Room Unfiltered In-Leakage	200 cfm$^{(1)}$
Normal Ventilation Flow Rates (Mode 1)	
Filtered Makeup Flow Rate	0 cfm
Filtered Recirculation Flow Rate	0 cfm
Unfiltered Makeup Flow Rate	2000 cfm
Emergency Mode Flow Rates (Mode 5)	
Filtered Makeup Flow Rate	2500 cfm
Filtered Recirculation FLow Rate	1955 cfm
Unfiltered Makeup Flow Rate	0 cfm
Filter Efficiencies	
Elemental Iodine	95%
Organic (Methyl) Iodine	95%
Particulate	99%
Delay to Switch CR HVAC from Normal Operation to Post Accident Operation after receiving an isolation signal (sec)	60 seconds
Breathing Rate - Duration of the Event	3.5E-04 m^3/second
Atmospheric Dispersion Factors	(second/m^3)
Steam Generator Safety Valves $^{(2)}$	
0-2 hr	4.66E-3
2-8 hr	3.40E-3
8-24 hr	1.17E-3
24-96 hr	1.07E-3
96-720 hr	9.05E-4
Facade $^{(3)}$	
0-2 hr	1.87E-2
2-8 hr	1.50E-2
8-24 hr	5.11E-3
24-96 hr	4.94E-3
96-720 hr	4.23E-3
Containment Surface $^{(4)}$	
0-2 hr	1.39E-3
2-8 hr	9.80E-4
8-24 hr	3.84E-4
24-96 hr	3.46E-4
96-720 hr	3.02E-4
Occupancy Factors	
0-24 hours	1.0
1-4 days	0.6
4-30 days	0.4

Notes:
(1) some analyses modeled a bounding flow rate of 300 cfm
(2) used secondary releases except for steam line break faulted steam generator releases
(3) used for steam line break faulted steam generator releases
(4) used for control rod ejection containment leakage releases

Figure 14.1.8-1 UNDERFREQUENCY EVENT (5 Hz/sec FREQUENCY DECAY RATE)
Sheet 1 of 3
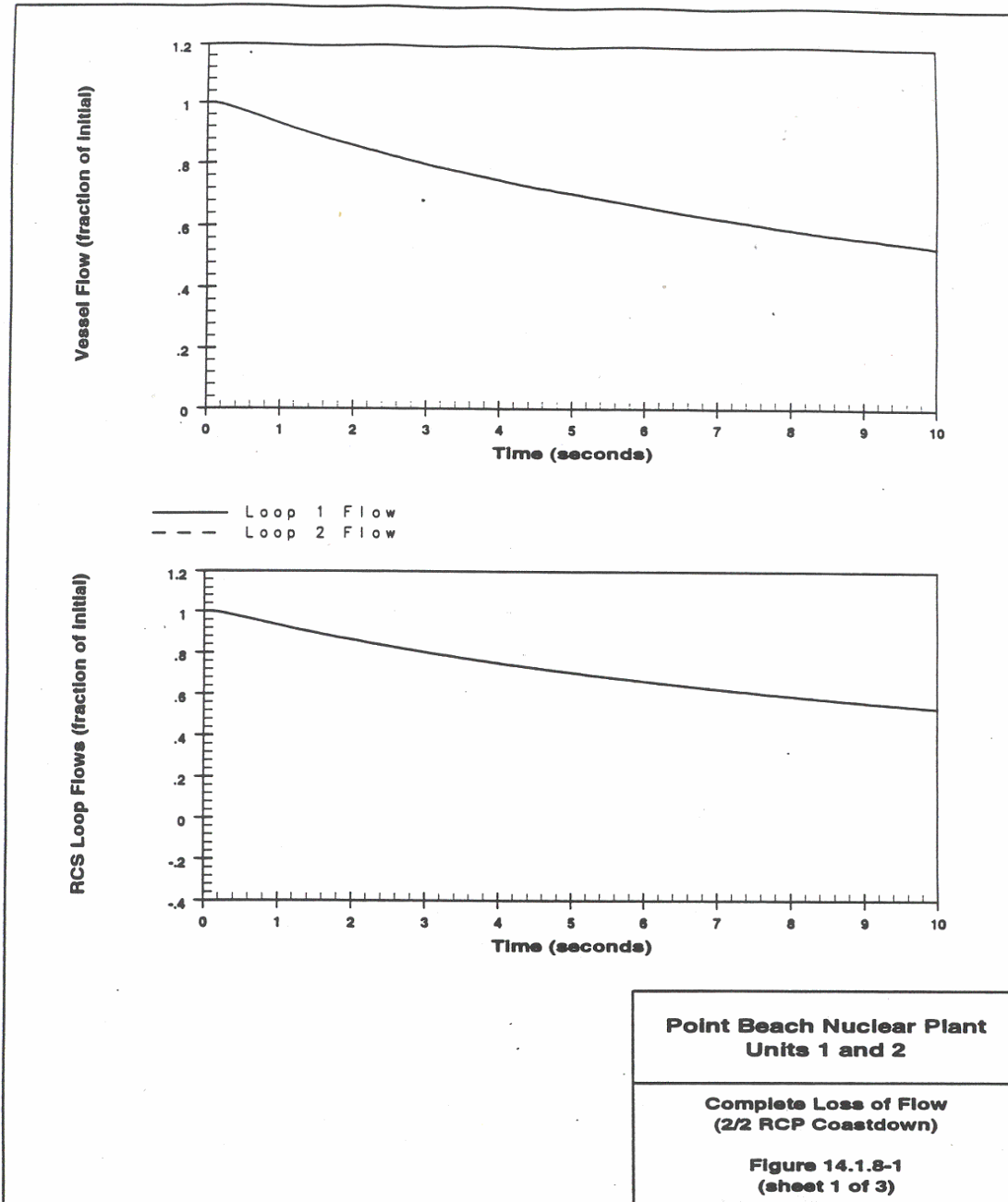
Point Beach Nuclear Plant
Units 1 and 2

Complete Loss of Flow
(2/2 RCP Coastdown)

Figure 14.1.8-1
(sheet 1 of 3)

Figure 14.1.8-1 UNDERFREQUENCY EVENT (5 Hz/sec FREQUENCY DECAY RATE)
Sheet 2 of 3
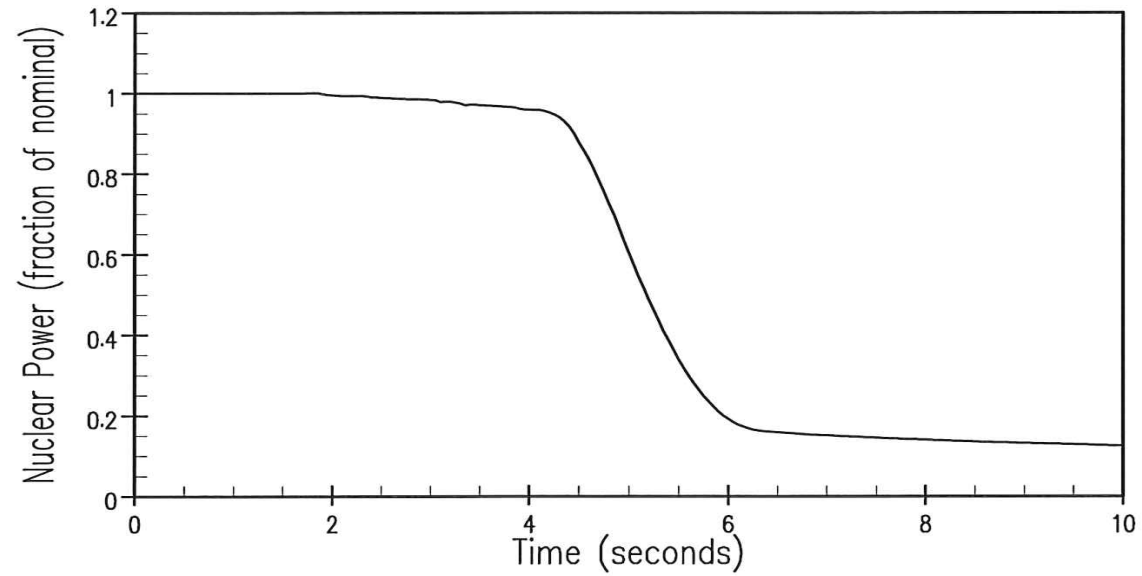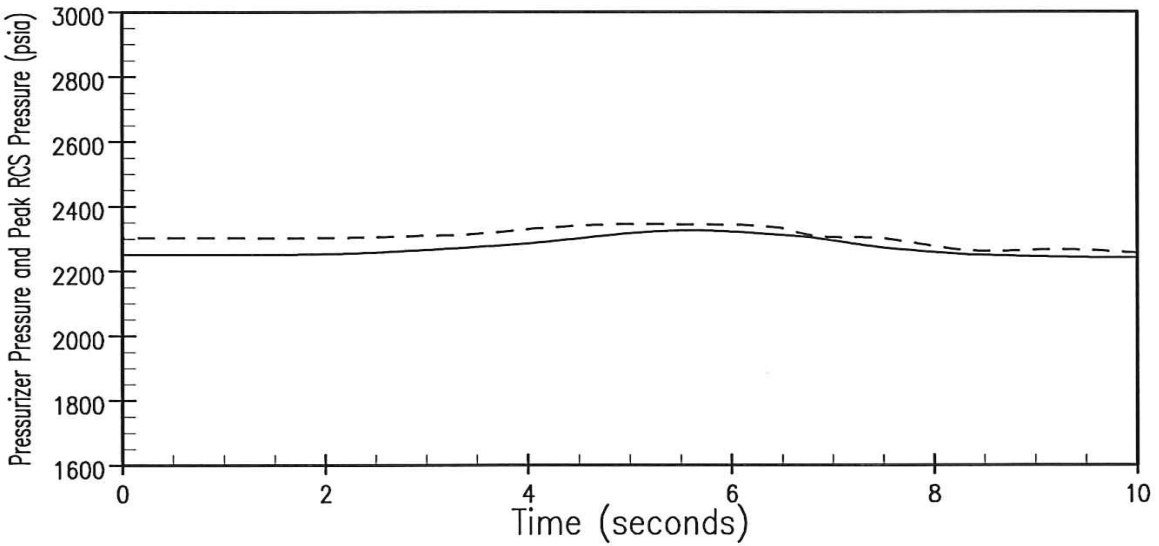
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Sheet 3 of 3
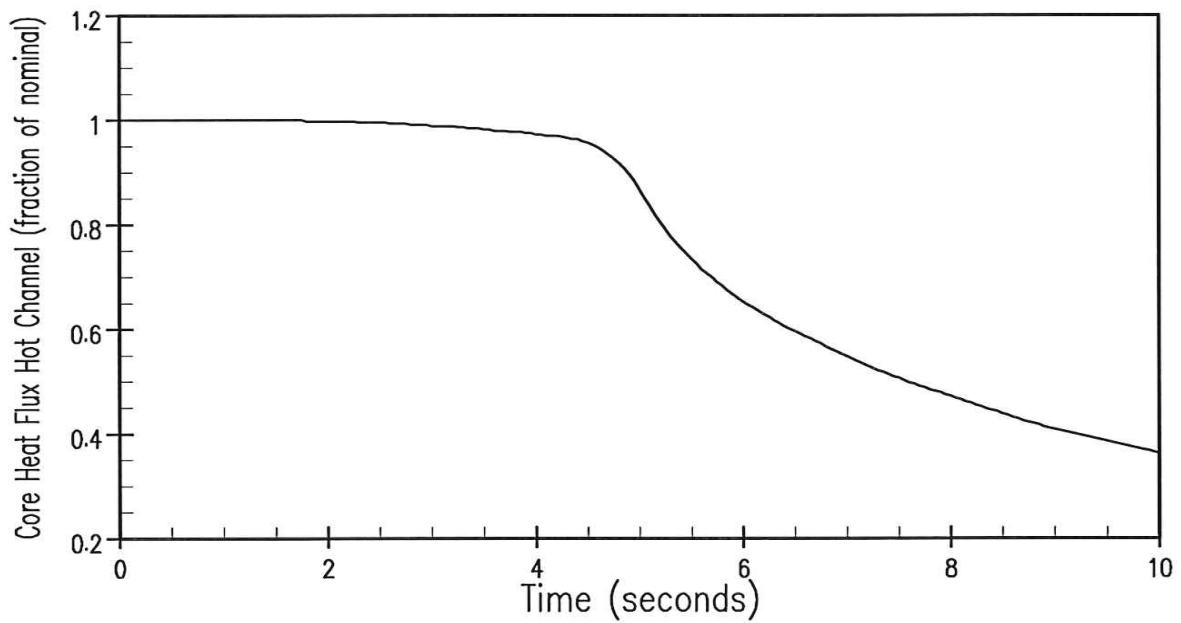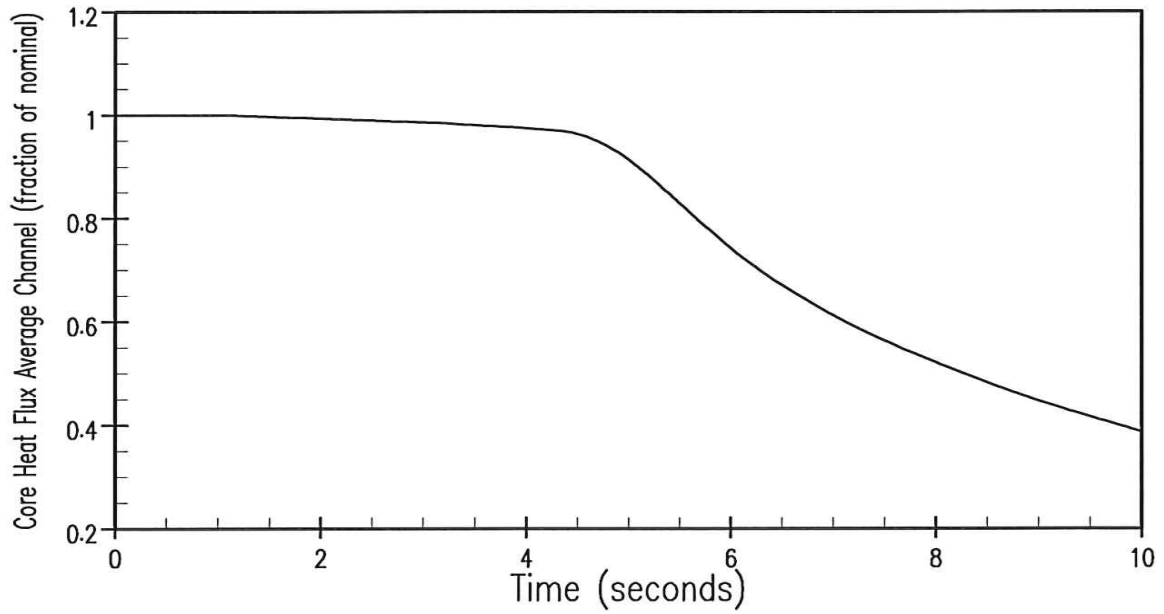
Figure 14.1.8-2 COMPLETE LOSS OF FLOW (2/2 RCP COASTDOWN)
Sheet 1 of 3
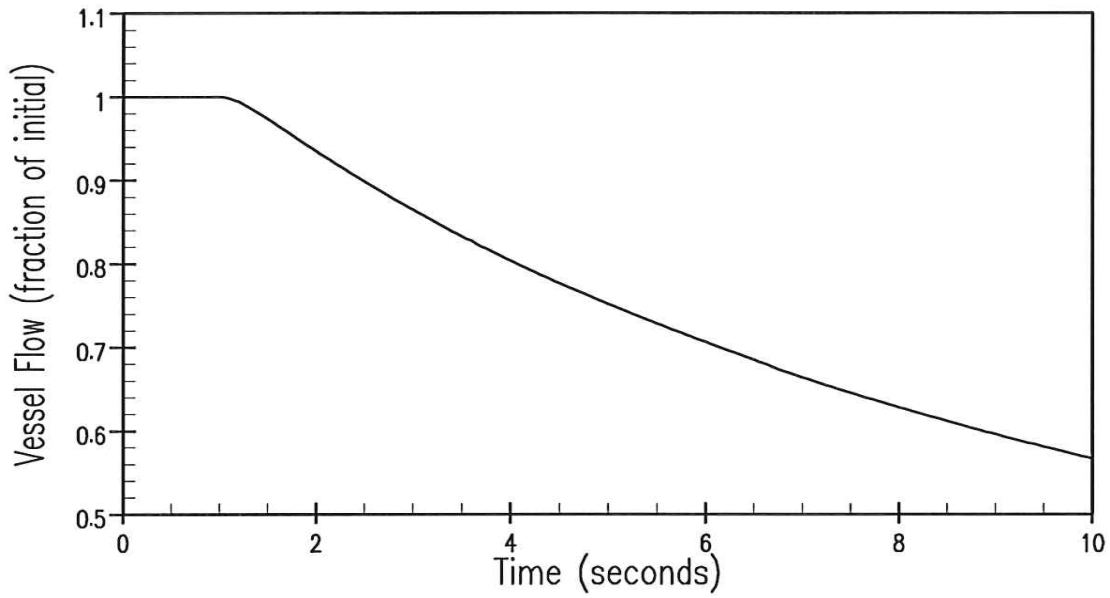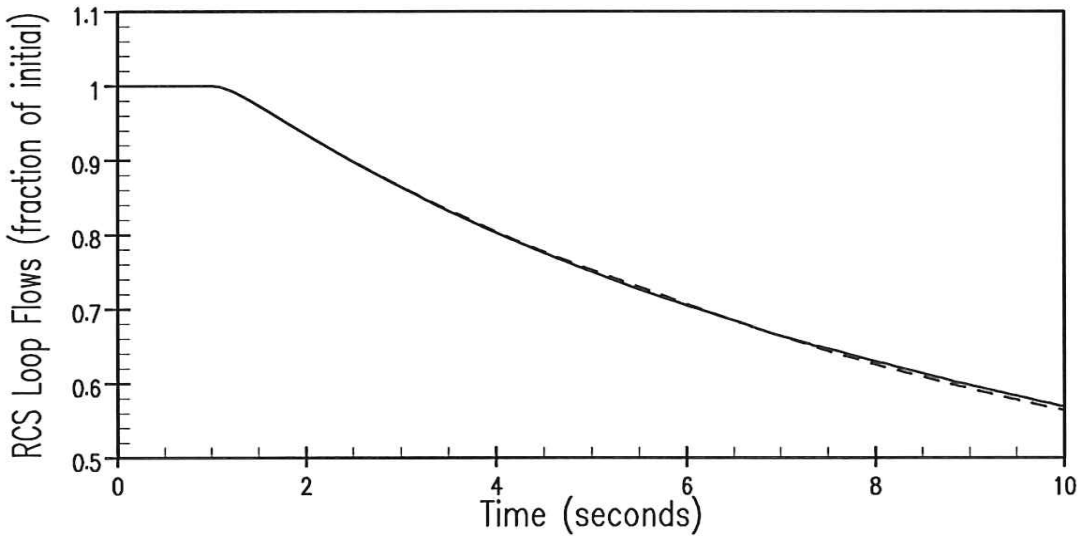
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Sheet 2 of 3
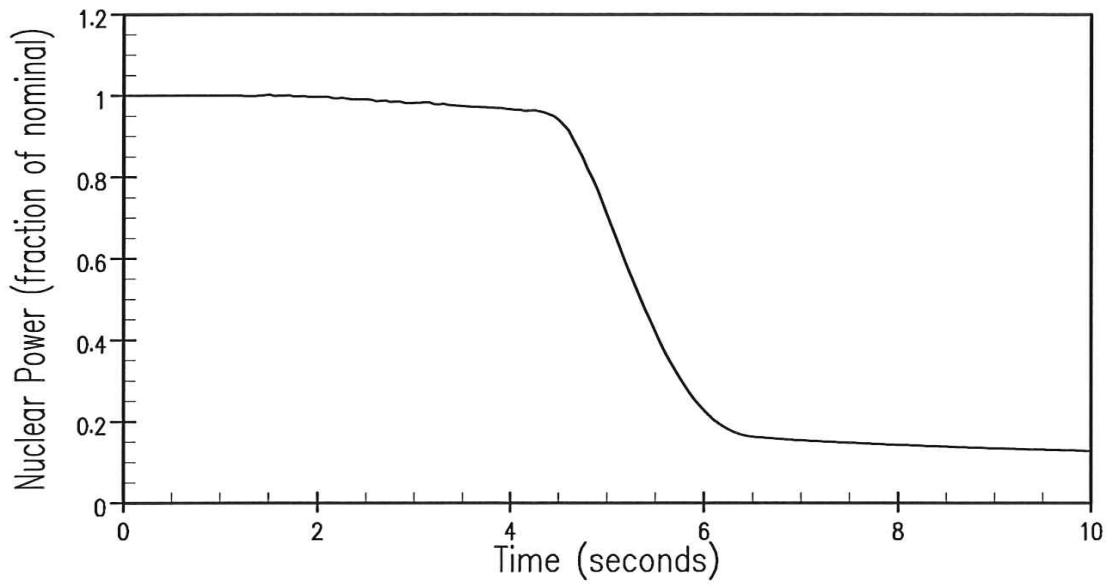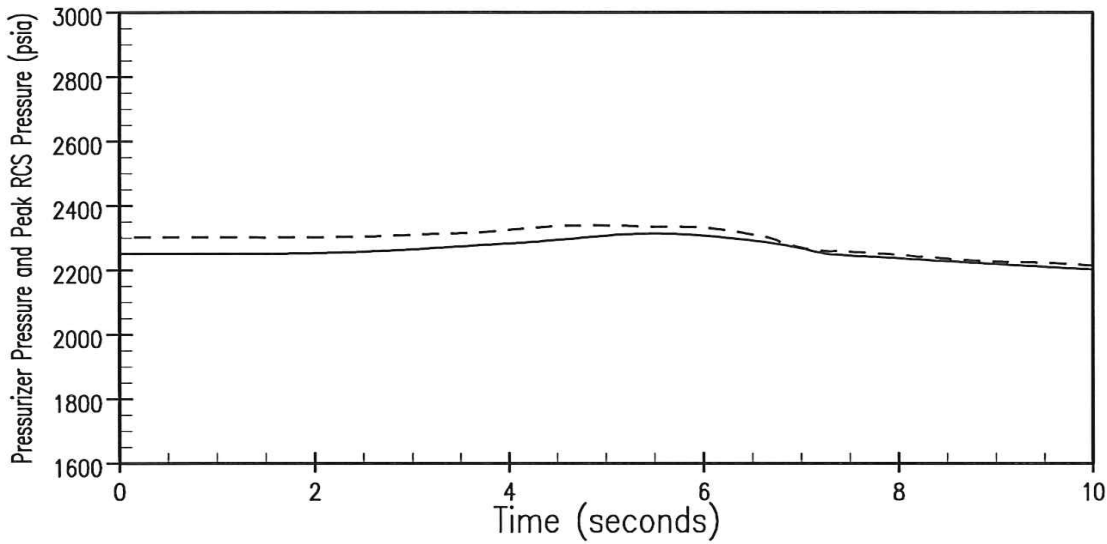
Figure 14.1.8-2 COMPLETE LOSS OF FLOW (2/2 RCP COASTDOWN)
Sheet 3 of 3
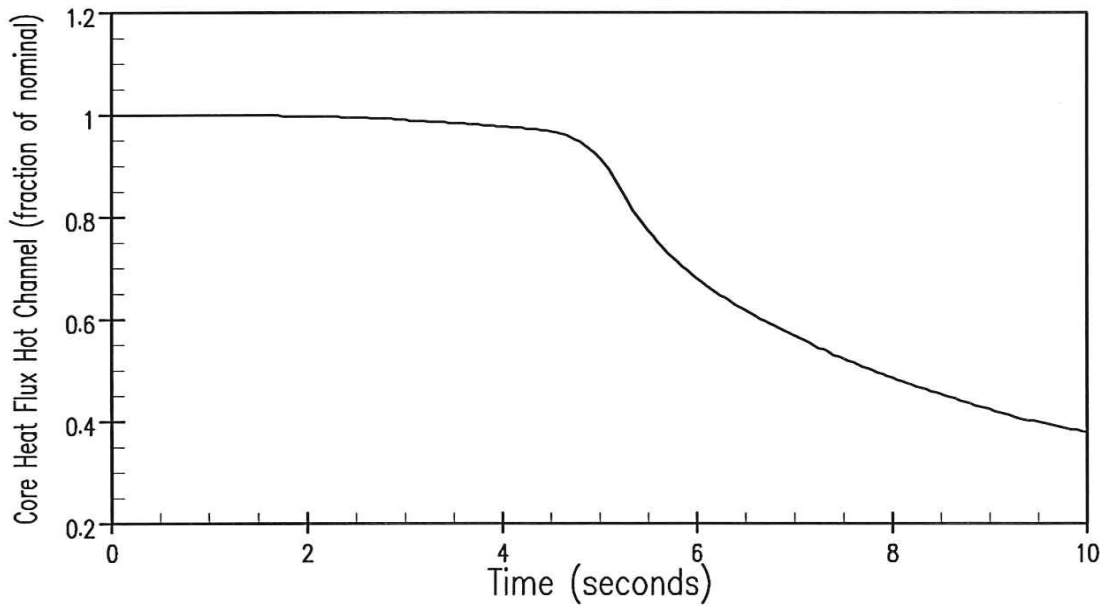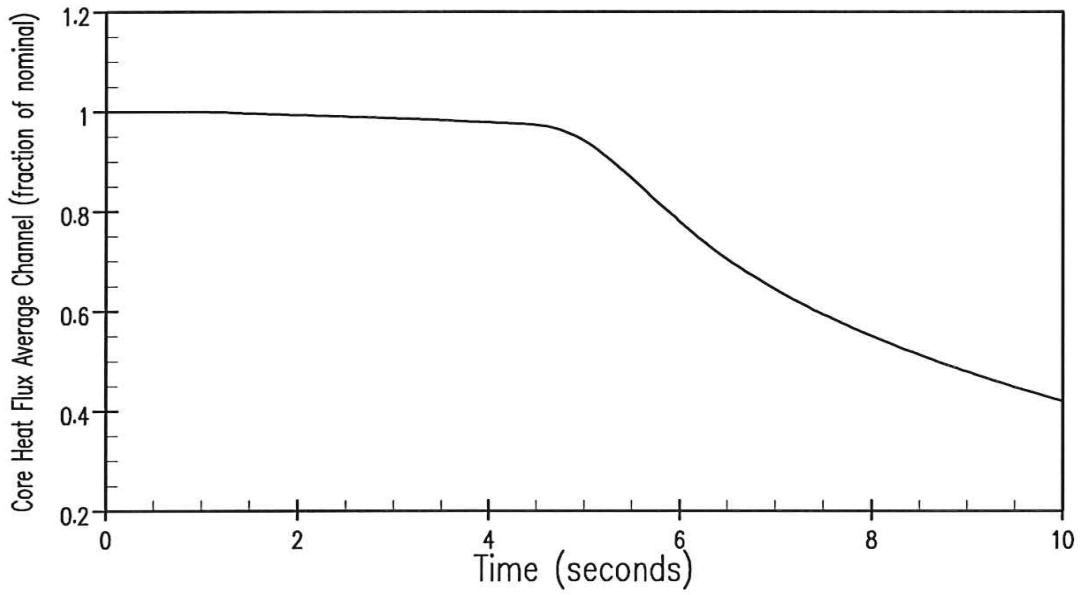
Figure 14.1.8-3 RCP LOCKED ROTOR

Sheet 1 of 3
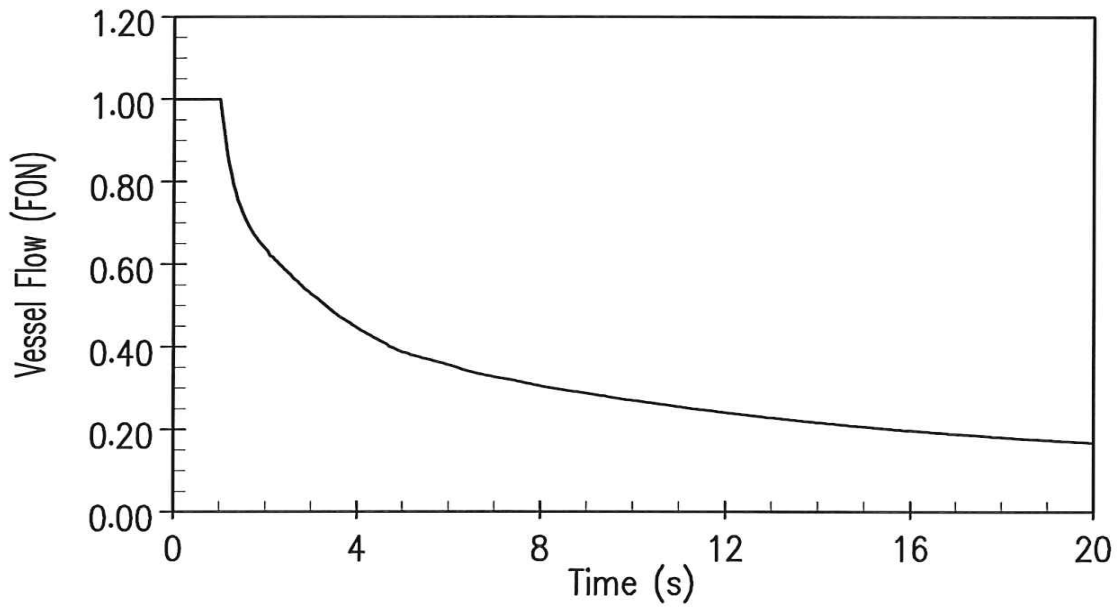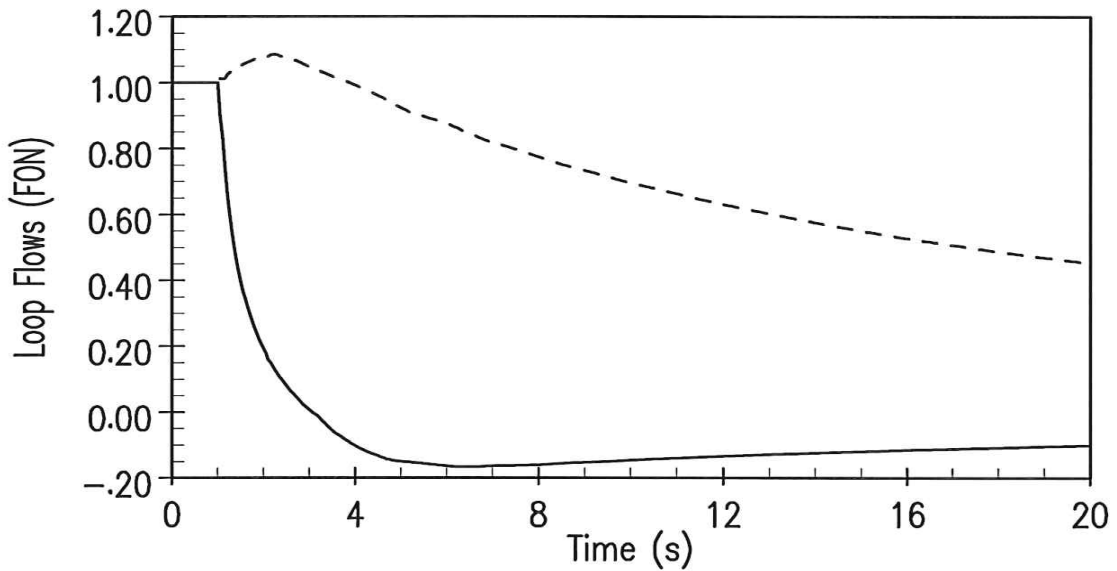
Figure 14.1.8-3 RCP LOCKED ROTOR

Sheet 2 of 3
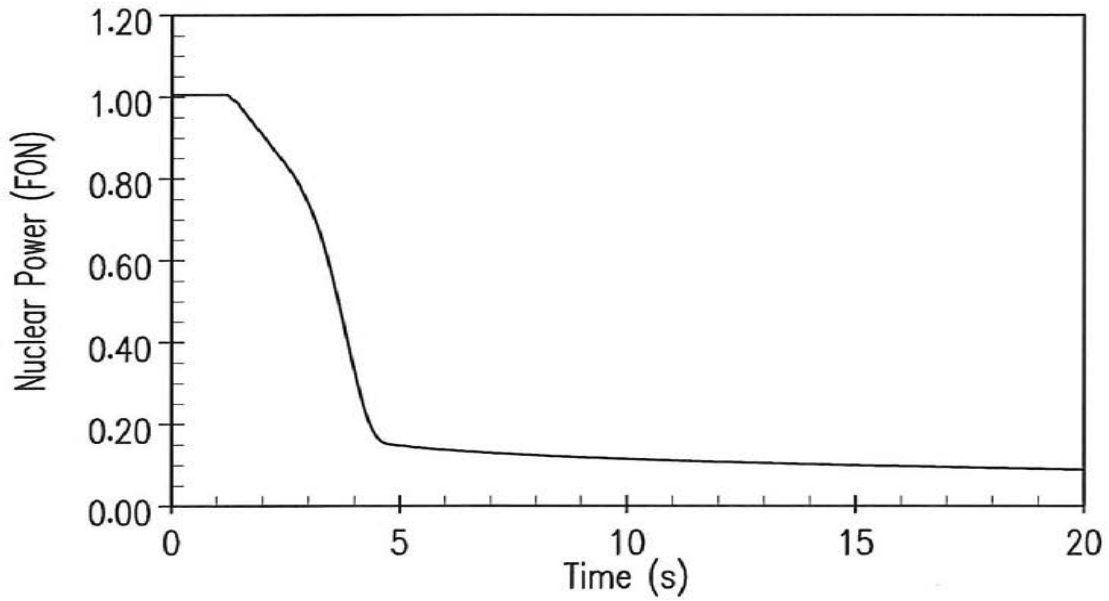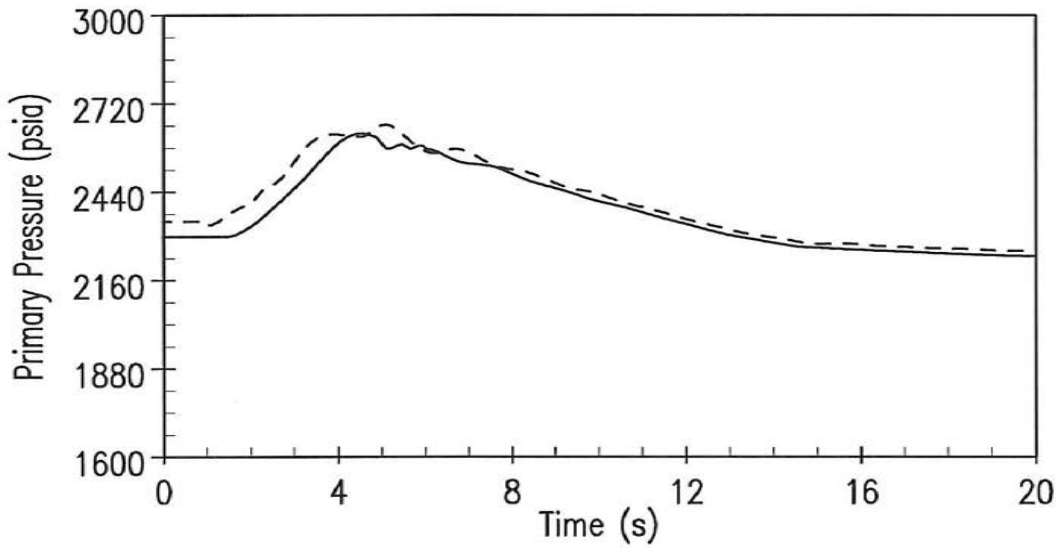
Figure 14.1.8-3 RCP LOCKED ROTOR

Sheet 3 of 3
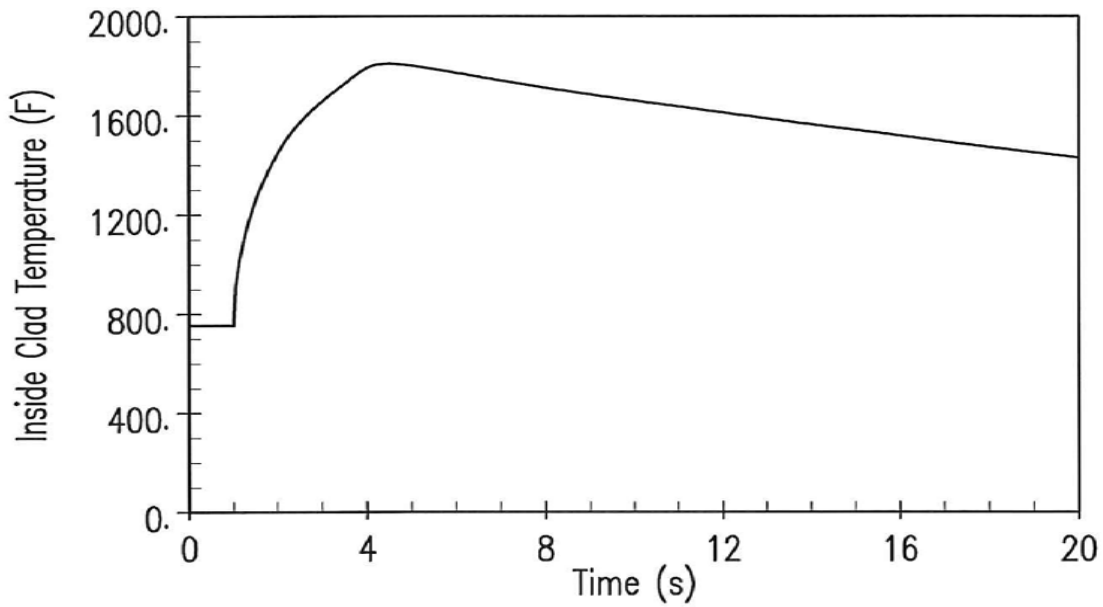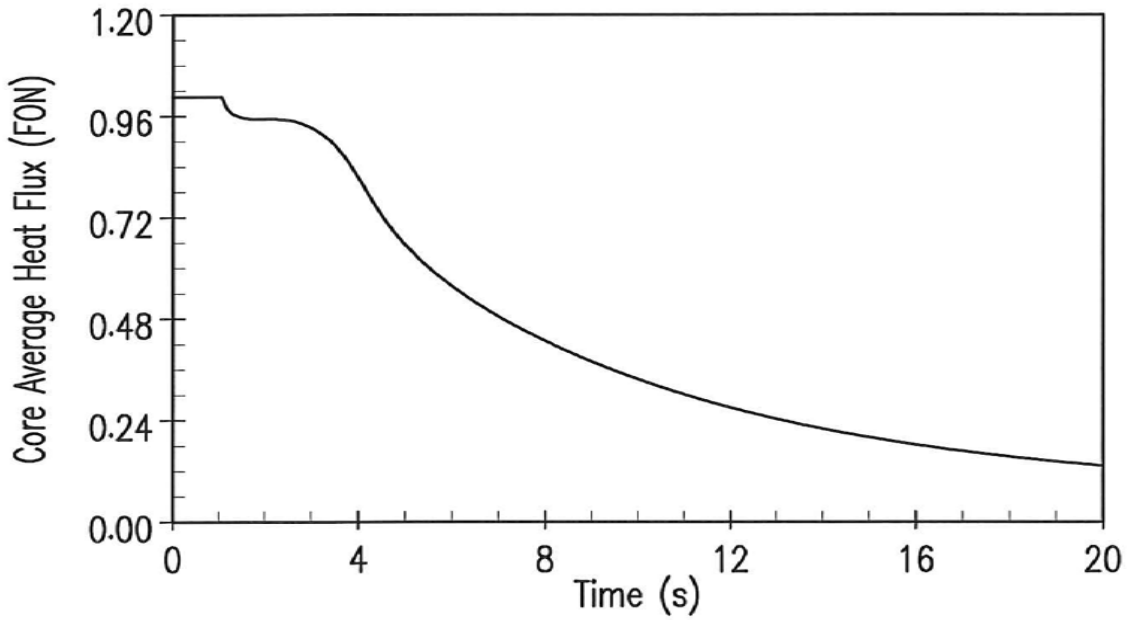
14.1.9   LOSS OF EXTERNAL ELECTRICAL LOAD (EPU Conditions)

The plant is able to accept a 50% loss of electrical load over 15 seconds while operating at full power or a complete loss of load while operating below the P-9 setpoint without actuating a reactor trip. The automatic steam bypass system (Section 10.1) is able to accommodate this load rejection by reducing the transient imposed upon the reactor coolant system. The reactor power is reduced to the new equilibrium power level at a rate consistent with the capability of the rod control system. Should the reactor suffer a complete loss of load from full power, the reactor protection system would automatically actuate a reactor trip.

The most likely source of a complete loss of load on the nuclear steam supply system is a trip of the turbine-generator. In this case, there is a direct reactor trip signal derived from either the turbine autostop oil pressure or a closure of the turbine stop valves, provided the reactor is operating above the P-9 interlock setpoint. Reactor temperature and pressure do not increase significantly if the steam bypass system and pressurizer pressure control system are functioning properly.   However, the plant behavior is evaluated for a complete loss of load from full power without a direct reactor trip, primarily to show the adequacy of the pressure relieving devices and also to show that no core damage occurs. The reactor coolant system and steam system pressure relieving capacities are designed to ensure the safety of the plant without requiring the automatic rod control, pressurizer pressure control, and/or steam bypass control systems.

Method of Analysis

The total loss of load transients are analyzed by employing the detailed digital computer program RETRAN (Reference 2 and Reference 3). The program simulates the neutron kinetics, reactor coolant system, pressurizer, pressurizer relief and safety valves, pressurizer spray, steam generator, and steam generator safety valves.

The program computes pertinent plant variables, including temperatures, pressures, and power level.

In this analysis, the behavior of the unit is evaluated for a complete loss of steam load from full power without direct reactor trip, primarily to show the adequacy of the pressure-relieving devices and also to demonstrate core protection margins.

Plant characteristics and initial conditions are discussed in Section 14.0.

Initial Operating Conditions - The initial core power, reactor coolant temperature, and reactor coolant pressure are assumed at the most limiting nominal values. The DNBR calculations are performed using the Revised Thermal Design Procedure (Reference 1), in which the uncertainties in the initial conditions are included in the DNBR limit value. For the peak RCS and SG pressure calculations, uncertainties of 0.6%, 50 psi, and 6.4°F are applied in the most limiting direction to the initial core power, reactor coolant pressure, and reactor coolant temperature.

Moderator and Doppler Coefficients of Reactivity - The loss of load accident is analyzed with minimum reactivity feedback. These cases assume a moderator temperature coefficient of 0 pcm/°F and the least negative Doppler coefficient.

The loss of load event results in a primary system heatup and therefore is conservatively analyzed with minimum reactivity feedback. Maximum feedback cases are no longer analyzed since they are non-limiting with respect to DNB and peak RCS and steam generator pressure.

Reactor Control - From the standpoint of the maximum pressures attained, it is conservative to assume that the reactor is in manual control.

Steam Release - No credit is taken for the operation of the steam dump system or steam generator power-operated relief valves. The steam generator pressure rises to the safety valve setpoint, where steam release through safety valves limits secondary steam pressure at the setpoint value. Main Steam Safety Valve performance is described in Table 14.1.9-2.

Pressurizer Spray and Power-Operated Relief Valves - Three cases are analyzed:

a.   For the DNB case, full credit is taken for the effect of pressurizer spray and power-operated relief valves in reducing or limiting the coolant pressure. Maximum steam generator tube plugging (10%) is assumed.

b.   For the RCS overpressure case, no credit is taken for the effect of pressurizer spray and power-operated relief valves in reducing or limiting the coolant pressure. Safety valves are operable. Maximum steam generator tube plugging (10%) is assumed.

c.   For SG overpressure case, full credit is taken for the effect of pressurizer spray and power-operated relief valves in reducing or limiting the primary coolant pressure, thereby delaying the time to reactor trip. Minimum steam generator tube plugging (0%) is assumed.

Feedwater Flow - Main feedwater flow to the steam generators is assumed to be lost at the time of loss of external electrical load. Reactor trip is actuated by the first reactor protection system trip setpoint reached, with no credit taken for the direct reactor trip on turbine trip.

Results

The transient responses for a total loss of load from full power operation are shown for three cases for minimum reactivity feedback illustrated in Figure 14.1.9-1 through Figure 14.1.9-3.

Figure 14.1.9-1 shows the transient response for the total loss of steam load (DNB case) with minimum reactivity feedback, maximum steam generator tube plugging (10%), and assuming full credit for the pressurizer spray and pressurizer power-operated relief valves. No credit is taken for the steam dump. The reactor is tripped by the high pressurizer pressure signal. The minimum departure from nucleate boiling ratio is well above the limit value. The pressurizer safety valves are actuated at a conservatively low setpoint.

Figure 14.1.9-2 shows the total loss of load accident (RCS overpressure case), assuming the plant to be initially operating at full power with maximum steam generator tube plugging (10%), and no credit taken for the pressurizer spray, pressurizer power-operated relief valves, or steam dump. The reactor is tripped on the high pressurizer pressure signal. In this case, the pressurizer safety valves are actuated. The peak RCS pressure of 2739.6 psia for Unit 1 and 2741.9 psia for Unit 2 occurs in the reactor vessel lower plenum.

Figure 14.1.9-3 shows the total loss of load accident (SG overpressure case), assuming the plant to be initially operating at full power with minimum steam generator tube plugging (0%), and assuming full credit for the pressurizer spray and pressurizer power-operated relief valves. No credit is taken for the steam dump. The reactor is tripped on the Overtemperature ΔT signal. The pressurizer safety valves are actuated at a conservatively low setpoint.

The calculated sequence of events for these three cases is shown in Table 14.1.9-1.

Conclusions

Results of the analyses show that the plant design is such that a total loss of external electrical load without a direct or immediate reactor trip presents no hazard to the integrity of the reactor coolant system or the main steam system. Pressure-relieving devices incorporated in the two systems are adequate to limit the maximum pressures within the design limits.

The integrity of the core is maintained by operation of the reactor protection system; i.e., the departure from nucleate boiling ratio is maintained above the limit value.

References

1. Friedland, A. J., Ray S., "Revised Thermal Design Procedure," WCAP- 11397-P-A (Proprietary), WCAP-11397-A (Non-Proprietary), April 1989.

2. Huegel, D. S., et. al., "RETRAN-02 Modeling and Qualification for Westinghouse Pressurized Water Reactor Non-LOCA Safety Analyses," WCAP-14882-P-A (Proprietary), WCAP-15234-A (Non-Proprietary), April 1999.

3. Calculation CN-TA-08-60, Revision 1, Point Beach Units 1 and 2 (WEP/WIS) Loss of Load / Turbine Trip (LOL/TT) Analysis for the Extended Power Uprate (EPU) Program, approved February 23, 2009, (see Reference 5).

4. NRC Safety Evaluation, PBNP Units 1 and 2 - Issuance of License Amendments Regarding Extended Power Uprate, May 3, 2011.

5. Engineering Change 281807, "AR 01943430, Error In Vendor (Westinghouse) Non-LOCA Analysis," Approved June 12, 2014

Table 14.1.9-1   TIME SEQUENCE OF EVENTS FOR LOSS OF EXTERNAL ELECTRICAL
LOAD

Case		Event	Time of Each Event (Seconds)	
			Unit 1	Unit 2
a.	With pressurizer control (DNB case)	Loss of electrical load	0	0
		Initiation of release from SG safety valves	10.2	9.6
		High pressurizer pressure reactor trip reached	11.7	11.4
		Rod begins to drop	12.7	12.4
		Minimum departure from nucleate boiling ratio occurs	14.1	13.7
b.	Without pressurizer control (RCS overpressure case)	Loss of electrical load	0	0
		High pressurizer pressure reactor trip point reached	6.0	5.9
		Rods begin to drop	7.0	6.9
		Peak RCS pressure occurs	9.2	9.0
		Initiation of release from SG safety valves	10.6	8.3
c.	With pressurizer control (SG overpressure case)	Loss of electrical load	0	0
		Initiation of release from SG safety valves	6.5	5.8
		Overtemperature ΔT reactor trip signal initiated	12.3	12.4
		Rods begin to drop	14.3	14.4
		Peak SG pressure occurs	18.3	18.0

Table 14.1.9-2 MSSV CHARACTERISTICS

Parameter	Bank 1	Bank 2	Bank 3	Bank 4
Nominal set pressure (psig)	1085	1100	1105	1105
Lift pressure (psia)	1166.4	1181.9	1187.0	1187.0
Full-open pressure (psia)	1171.4	1186.9	1192.0	1192.0

Table 14.1.9-2 Notes

1. The lift pressure is the nominal set pressure, plus 3% allowance for setpoint tolerance (3% of the nominal set pressure), plus the appropriate allowance for the frictional pressure drop between the main steamline and the valve at full MSSV relief conditions, plus atmospheric pressure (14.7 psi).

2. The full-open pressure is the lift pressure, plus 5 psia for valve accumulation.

3. The MSSV relief rate, which is based on a Moody choked flow model for saturated steam discharge versus steam pressure, is assumed to be a linear function of the pressure between the lift pressure and the full-open pressure.

4. The values listed above for the lift pressure and full-open pressure reflect the main steamline pressure. However, since the safety valves are actually located downstream of the SG, the pressure at the valve is only the same as that in the main steamline when the first safety valve opens. Once relief flow is established, a frictional pressure drop will exist between the main steamline and the valves (assumed to be 34.2 psi at full relief flow) and the steam pressure at the safety valve will actually be less than the values listed above. Thus, the appropriate allowance for the frictional pressure drop has been conservatively included in the values listed above for the lift pressure and full-open pressure.

Figure 14.1.9-1 LOSS OF ELECTRICAL LOAD WITH PRESSURE CONTROL (DNB Case)
Sheet 1 of 3
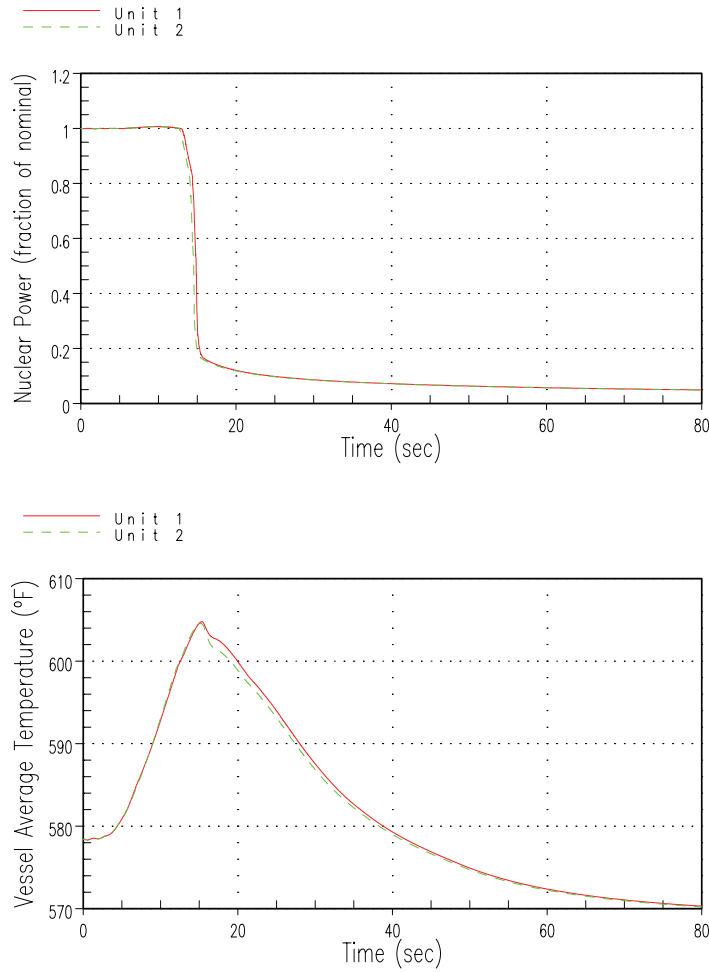
Figure 14.1.9-1 LOSS OF ELECTRICAL LOAD WITH PRESSURE CONTROL (DNB Case)
Sheet 2 of 3
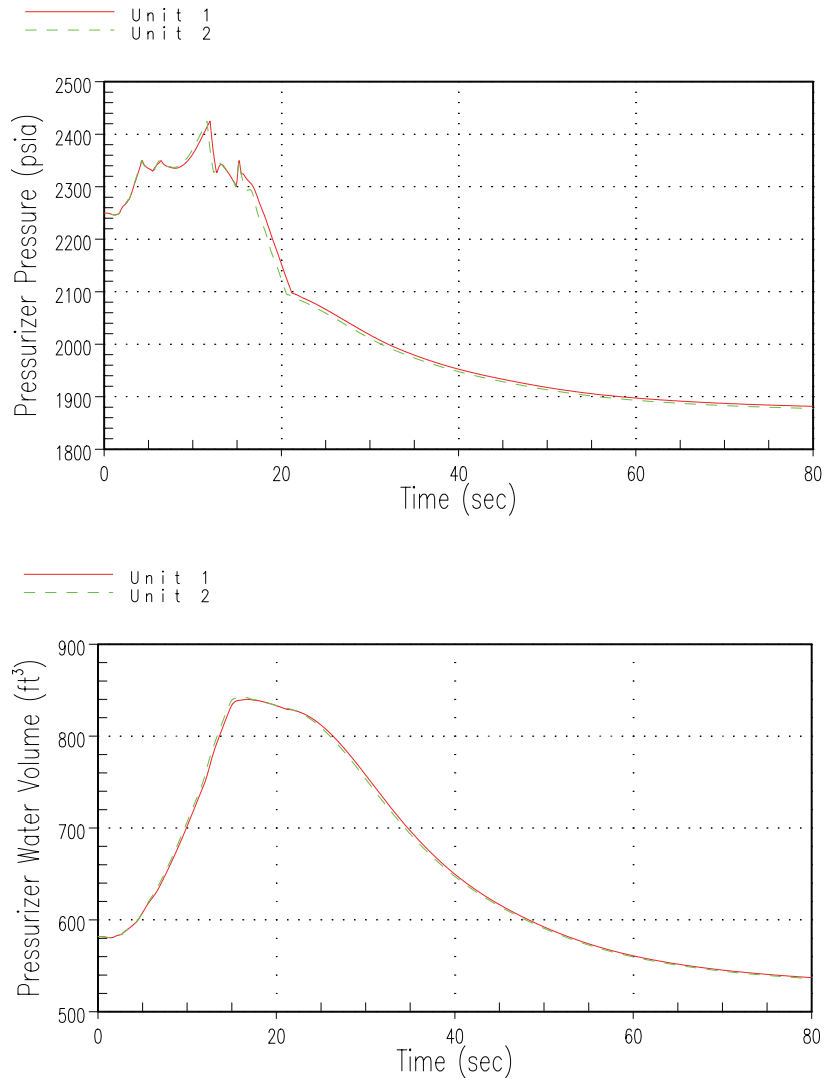
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Sheet 3 of 3
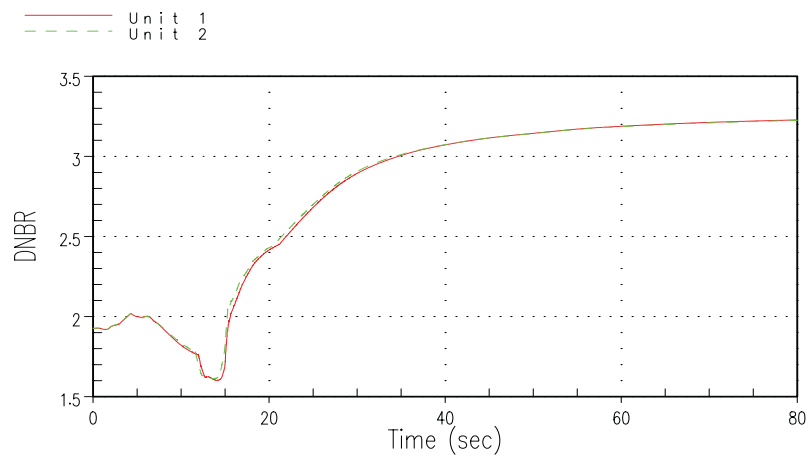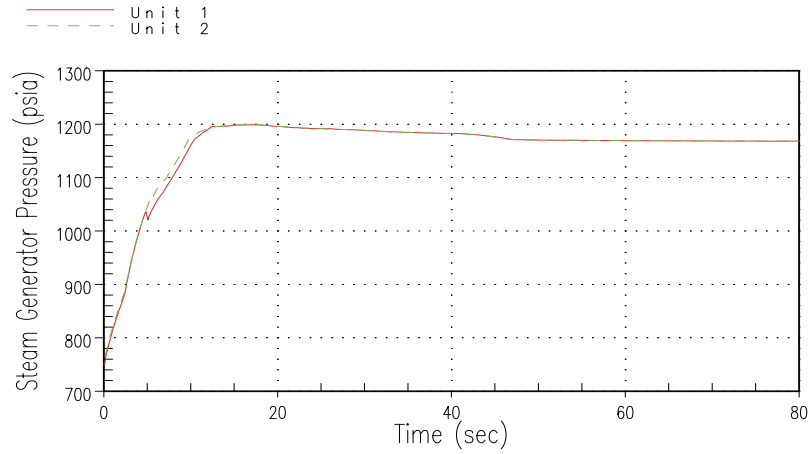
Figure 14.1.9-2 LOSS OF ELECTRICAL LOAD WITHOUT PRESSURE CONTROL
(RCS Overpressure Case)
Sheet 1 of 3

Point Beach Nuclear Plant
Units 1 and 2

Loss of Electrical Load
Without Pressure Control
(RCS Overpressure Case)
Figure 14.1.9-2
(Sheet 1 of 3)

Figure 14.1.9-2 LOSS OF ELECTRICAL LOAD WITHOUT PRESSURE CONTROL
(RCS Overpressure Case)
Sheet 2 of 3

Point Beach Nuclear Plant
Units 1 and 2

Loss of Electrical Load
Without Pressure Control
(RCS Overpressure Case)
Figure 14.1.9-2
(Sheet 2 of 3)

Figure 14.1.9-2 LOSS OF ELECTRICAL LOAD WITHOUT PRESSURE CONTROL
(RCS Overpressure Case)
Sheet 3 of 3

Point Beach Nuclear Plant
Units 1 and 2
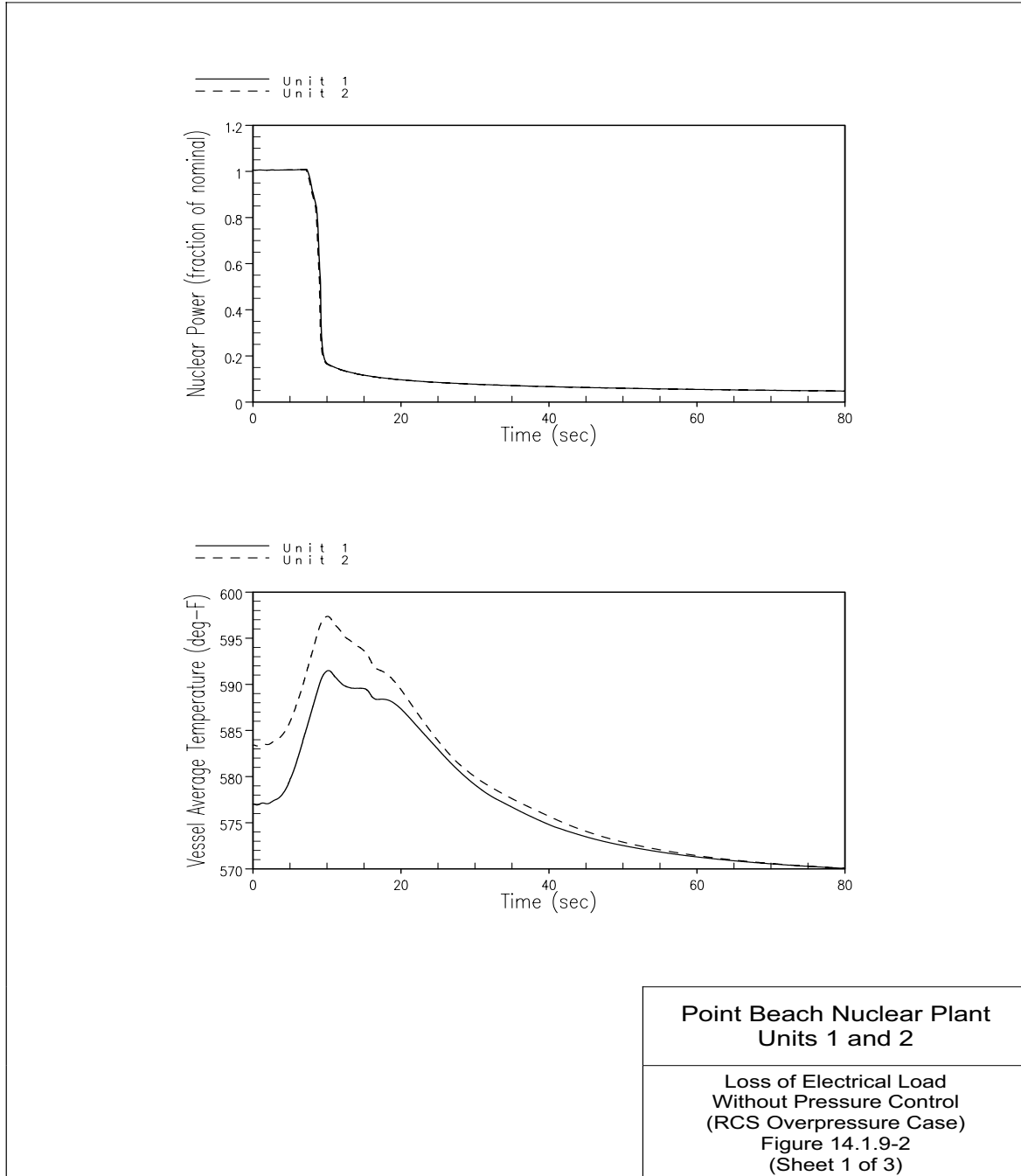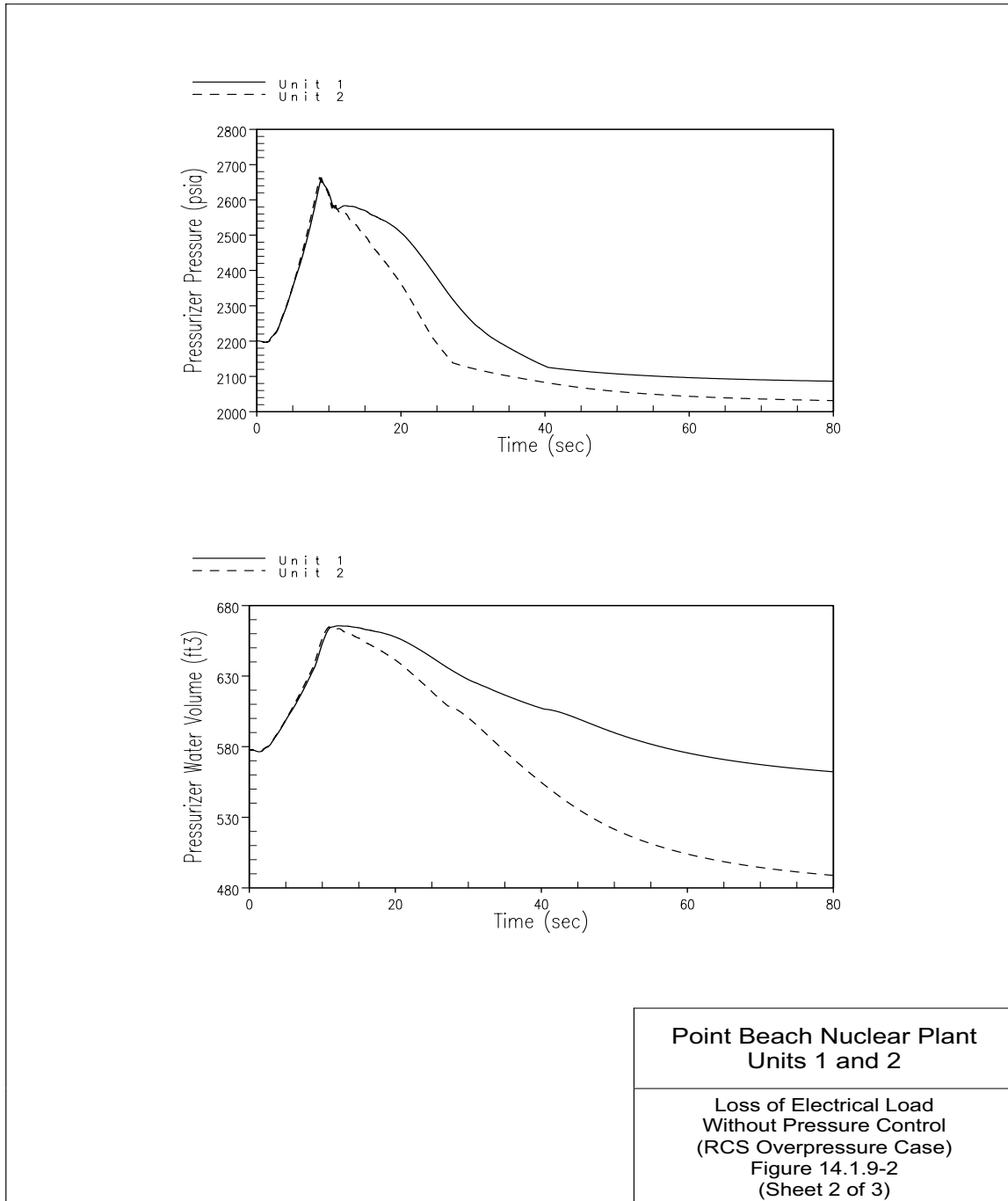
Loss of Electrical Load
Without Pressure Control
(RCS Overpressure Case)
Figure 14.1.9-2
(Sheet 3 of 3)
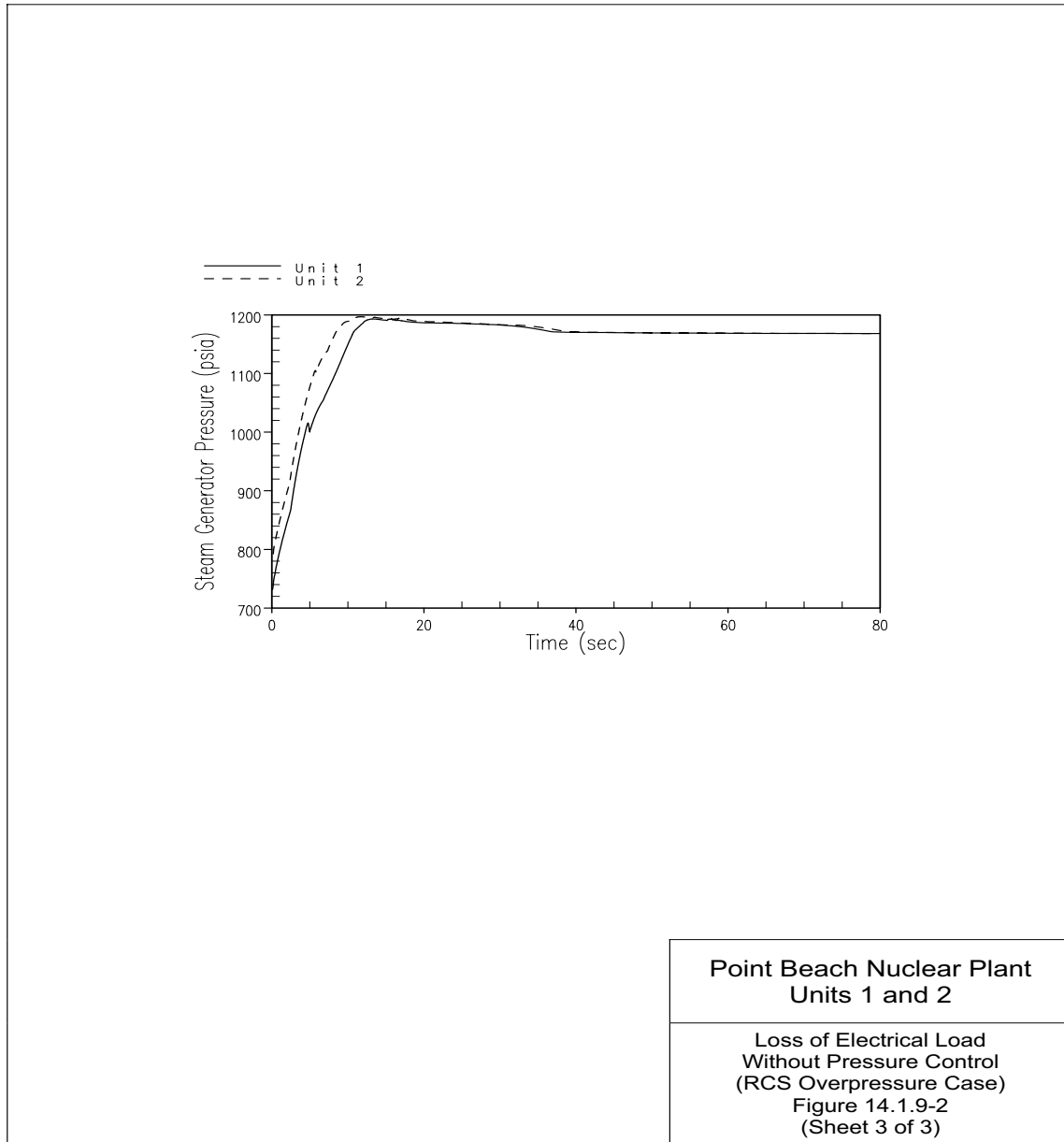
Figure 14.1.9-3 LOSS OF ELECTRICAL LOAD WITH PRESSURE CONTROL
(SG Overpressure Case)
Sheet 1 of 3

Figure 14.1.9-3 LOSS OF ELECTRICAL LOAD WITH PRESSURE CONTROL
(SG Overpressure Case)
Sheet 2 of 3
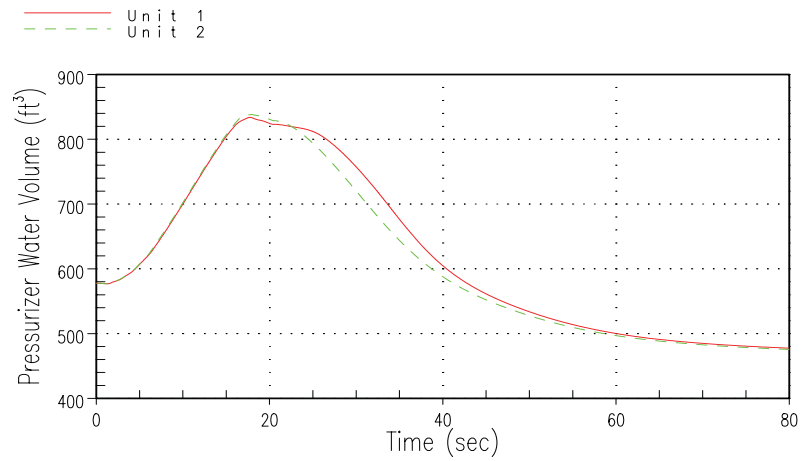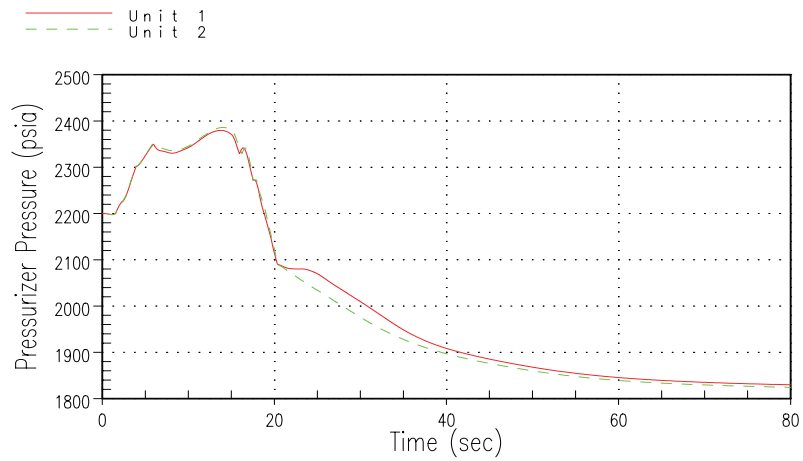
Figure 14.1.9-3 LOSS OF ELECTRICAL LOAD WITH PRESSURE CONTROL
(SG Overpressure Case)
Sheet 3 of 3
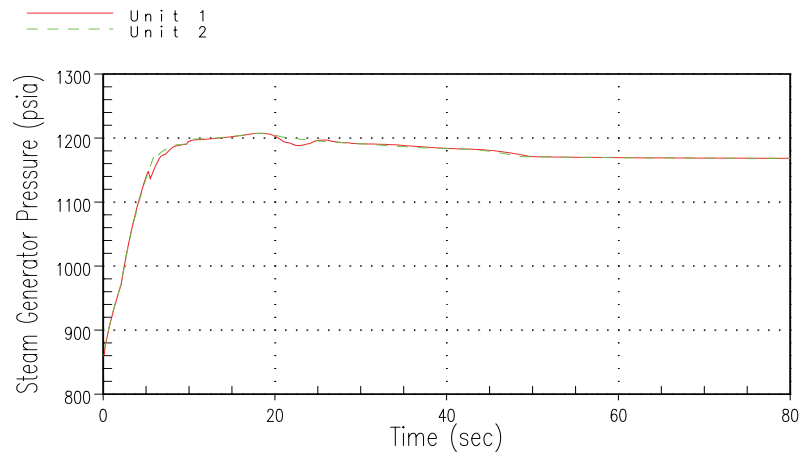
14.1.10 <u>LOSS OF NORMAL FEEDWATER</u>

A loss of normal feedwater (from a pipe break, pump failure, or valve malfunction) results in a reduction in capability of the secondary system to remove the heat generated in the reactor core. If the reactor is not tripped during this accident primary plant damage could possibly result from a sudden loss of heat sink. If an alternate supply of feedwater were not supplied to the plant, residual heat following reactor trip would heat the primary system water to the point where water relief from the pressurizer occurs, and significant loss of water from the reactor coolant system could conceivably lead to core damage. The following provides the protection in the event a loss of normal feedwater (LONF) occurs:

1. Reactor trip on low-low water level in either steam generator.

2. Reactor trip on steam flow-feedwater flow mismatch coincident with low water level in either steam generator.

3. A motor-driven and a steam driven auxiliary feedwater pump (275 gpm each) which are automatically started on any of the following:

    • Low-low water level in either steam generator.
    • Loss of voltage on both 4.16 kv buses supplying the main feedwater pump motors.
    • Trip or shutdown of both feedwater pumps or closure of either a feedwater isolation valve or a feedwater regulating valve in both main feedwater lines. These signals are processed through AMSAC at power levels above 40% (Reference Section 7.4).
    • Automatic or manual safety injection. In conjunction with a loss of AC the MDAFW pump start is sequenced a nominal 32.5 seconds after EDG breaker closure.

The motor driven auxiliary feedwater pumps is supplied by an emergency diesel generator if a loss of offsite power occurs. The turbine-driven pump utilizes steam from the secondary systems and exhausts the steam to the atmosphere. The auxiliary feedwater pumps take suction directly from the condensate storage tank (CST) for delivery to the steam generators, or from the Service Water System should the CST not be available. See Section 10.2.3 and Section 7.4.3 for a description of the automatic switchover of the AFW suction supply to Service Water.

The above protection provides considerable backup in equipment and control logic to ensure that reactor trip and automatic auxiliary feedwater flow will occur following any loss of normal feedwater including that caused by loss of AC power.

<u>Method of Analysis</u>

A detailed analysis using the RETRAN code (Reference 2) is performed in order to obtain the plant transient following a loss of normal feedwater. The simulation describes the plant thermal kinetics, RCS including the natural circulation, pressurizer, steam generators, and feedwater system. The digital program computes pertinent variables, including the steam generator level, pressurizer water level, and reactor coolant average temperature.

The following assumptions were made:

1. The plant is initially operating at 100.6% of 1806 MWt.

2.   Core residual heat generation is based on the 1979 version of ANS-5.1 (Reference 1) plus two standard deviations for uncertainty. ANSI/ ANS-5.1-1979 is a conservative representation of the decay heat release rates.

3.   The initiating signal for the reactor trip is a low-low steam generator level. No credit is taken for the reactor trip due to a steam flow/feed flow mismatch coincident with a low steam generator level.

4.   Both steam generators are affected equally, and both reach their low-low level trip setpoints simultaneously. This assumption conservatively minimizes the secondary heat sink available at the time of the reactor trip.

5.   The auxiliary feedwater system provides only 275 gpm of flow split to two steam generators. No credit is taken for AFW flow from the turbine driven pump.

6.   AFW flow of 275gpm is delivered to the steam generator(s) starting 30 seconds after the initiating signal (low-low steam generator level trip). From 30 to 60 seconds the AFW flow is ramped from 0% to 80% of total flow: from 60 to 120 seconds AFW flow is ramped from 80% to 100% of total flow: beyond 120 seconds 100 % of total AFW flow is maintained.

7.   The assumed steam generator models are 44F (Unit 1) and Delta-47 (Unit 2).

8.   The pressurizer sprays, function to produce the maximum peak pressurizer water volume. The backup heaters are assumed to be unavailable on high pressurizer level deviation signal and the PORVs are assumed to be inoperable. Cases with the PORVs operable were found to be less limiting.

9.   Secondary system steam relief is through the self-actuated safety valves.

Results

The calculated sequence of events for this event is listed in Table 14.1.10-1. Figure 14.1.10-1 and Figure 14.1.10-2 show the plant parameters following a loss of normal feedwater accident with the assumptions listed above for Units 1 and 2. Low-low level signal in either steam generator initiates the reactor trip. The reactor trip then initiates the turbine trip. Following the reactor and turbine trip from full load, the water level in the steam generators falls due to the reduction of steam generator void fraction and because steam flow through the safety valves continues to dissipate the stored and generated heat.

Upon the initiation of the low-low level signal, one auxiliary feedwater pumps is automatically started. The pumps will start to supply auxiliary feedwater to both steam generators within 30 seconds, reducing the rate of water level decrease.

The capacity of the auxiliary feedwater system is such that the water level in the steam generators does not recede below the lowest level at which sufficient heat transfer area is available to dissipate core residual heat without water relief from the RCS relief or safety valves. From Figure 14.1.10-1 and Figure 14.1.10-2 it can be seen that at no time is there water relief from the pressurizer.

Conclusion

The loss of normal feedwater does not result in any adverse condition in the core, because it does not result in water relief from the pressurizer relief or safety valves.

References

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

2. Huegel, D. S., et. al., "RETRAN-02 Modeling and Qualification for Westinghouse Pressurized Water Reactor Non-LOCA Safety Analyses," WCAP-14882-P-A (Proprietary), WCAP-15234-A (Non-Proprietary), April 1999.

3. Westinghouse CN-TA-08-79, Rev 1, Point Beach Unit 1 and 2 Loss of Normal Feedwater/Loss of AC Power (LONF/LOAC) Analysis for the EPU Program, Approved February 26, 2009.

4. NRC Safety Evaluation, PBNP Units 1 and 2 - Issuance of License Amendments Regarding Extended Power Uprate, May 3, 2011.

5. NRC Safety Evaluation, "Point Beach Nuclear Plant Units 1 and 2-Issuance of License Amendments Re: Auxiliary Feedwater System Modification," dated March 25, 2011.

6. Letter NRC 2011-0086, NextEra Energy to NRC, Clarification/Comments on the NRC Safety Evaluation Report, Amendment Nos. 238 (Unit 1) and 242 (Unit 2), Auxiliary Feedwater System Modification, September 16, 2011.

7. NRC Letter to NextEra Energy, Point Beach Nuclear Plant, Units 1 and 2-NRC Staff Response to Clarification/Comments Related to the Safety Evaluation Report Associated with the Auxiliary Feedwater System Modification License Amendment, December 6, 2011.

Table 14.1.10-1 TIME SEQUENCE OF EVENTS FOR LOSS OF NORMAL FEEDWATER
FLOW INCIDENTS

	Time of Each Event (Seconds)	
Event	Unit 1	Unit 2
Main feedwater flow stops	20	20
Low-Low steam generator water level trip actuated	56.0	54.4
Rods begin to drop	58.0	56.4
AFW flow to each loop begins	86.0	84.4
80% of full AFW flow reached	116.0	114.4
100% of full AFW flow reached	176.0	174.4
Peak water level in pressurizer occurs	1410	1378
Core decay heat decreases to auxiliary feedwater heat removal capacity	~1414	~1392

Figure 14.1.10-1 UNIT 1 (MODEL 44F SG) LOSS OF NORMAL FEEDWATER
Sheet 1 of 6

**Point Beach Nuclear Plant
Units 1 and 2**

**Unit 1 (Model 44F SG)
Loss of Normal Feedwater**

Figure 14.1.10-1 (sheet 1 of 6)

Figure 14.1.10-1 UNIT 1 (MODEL 44F SG) LOSS OF NORMAL FEEDWATER
Sheet 2 of 6

**Point Beach Nuclear Plant
Units 1 and 2**

Unit 1 (Model 44F SG)
Loss of Normal Feedwater

Figure 14.1.10-1 (sheet 2 of 6)

Figure 14.1.10-1 UNIT 1 (MODEL 44F SG) LOSS OF NORMAL FEEDWATER
Sheet 3 of 6

Point Beach Nuclear Plant
Units 1 and 2

Unit 1 (Model 44F SG)
Loss of Normal Feedwater

Figure 14.1.10-1 (sheet 3 of 6)

Figure 14.1.10-1 UNIT 1 (MODEL 44F SG) LOSS OF NORMAL FEEDWATER
Sheet 4 of 6

Point Beach Nuclear Plant
Units 1 and 2

Unit 1 (Model 44F SG)
Loss of Normal Feedwater

Figure 14.1.10-1 (sheet 4 of 6)

Figure 14.1.10-1 UNIT 1 (MODEL 44F SG) LOSS OF NORMAL FEEDWATER
Sheet 5 of 6

Point Beach Nuclear Plant
Units 1 and 2

Unit 1 (Model 44F SG)
Loss of Normal Feedwater

Figure 14.1.10-1 (sheet 5 of 6)

Figure 14.1.10-1 UNIT 1 (MODEL 44F SG) LOSS OF NORMAL FEEDWATER
Sheet 6 of 6

**Point Beach Nuclear Plant
Units 1 and 2**

**Unit 1 (Model 44F SG)
Loss of Normal Feedwater**

Figure 14.1.10-1 (sheet 6 of 6)

Figure 14.1.10-2 UNIT 2 (Delta - 47 SG) LOSS OF NORMAL FEEDWATER
Sheet 1 of 6

Point Beach Nuclear Plant
Units 1 and 2

Unit 2 (Delta-47 SG)
Loss of Normal Feedwater

Figure 14.1.10-2 (sheet 1 of 6)

Figure 14.1.10-2 UNIT 2 (Delta - 47 SG) LOSS OF NORMAL FEEDWATER
Sheet 2 of 6

**Point Beach Nuclear Plant
Units 1 and 2**
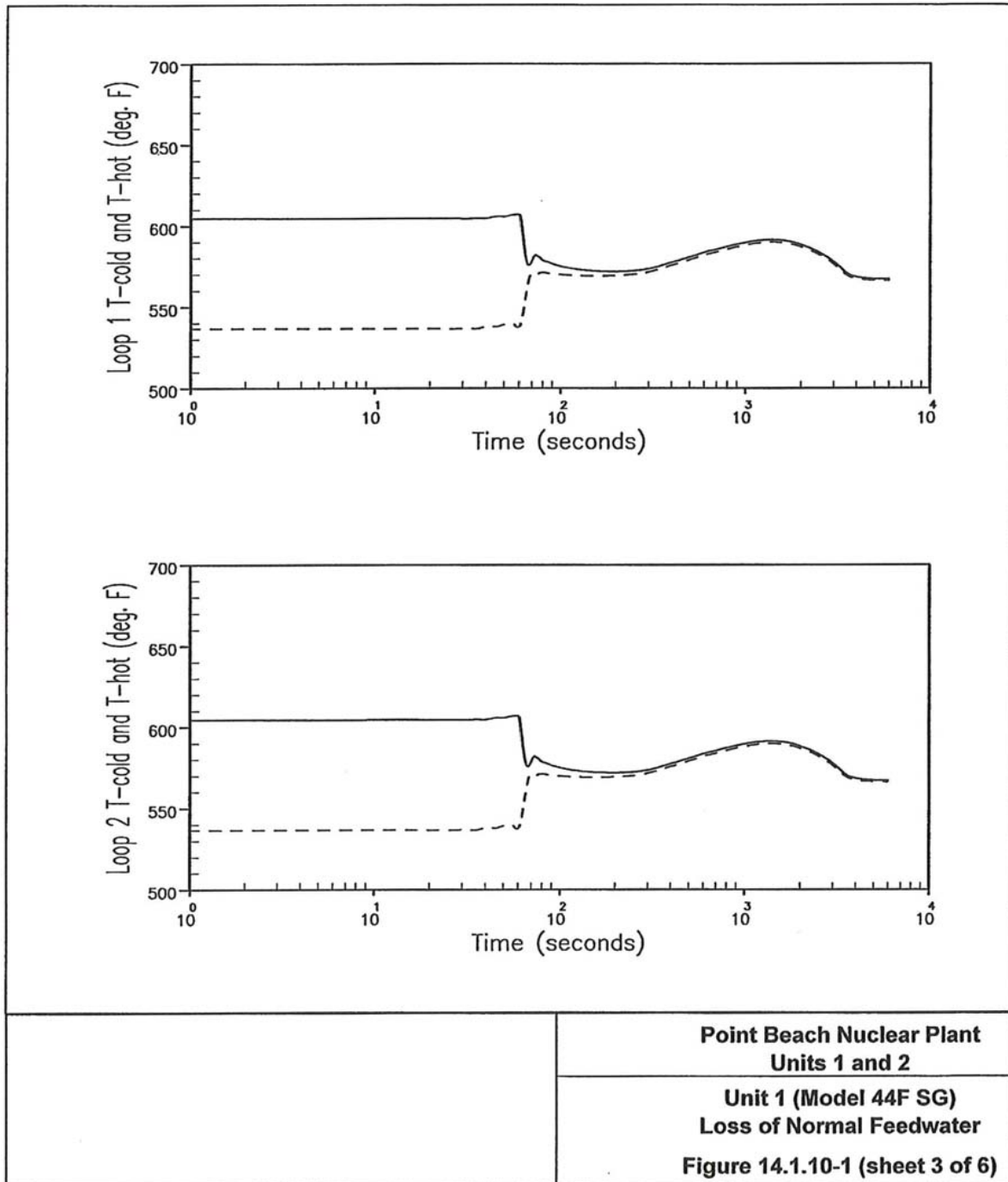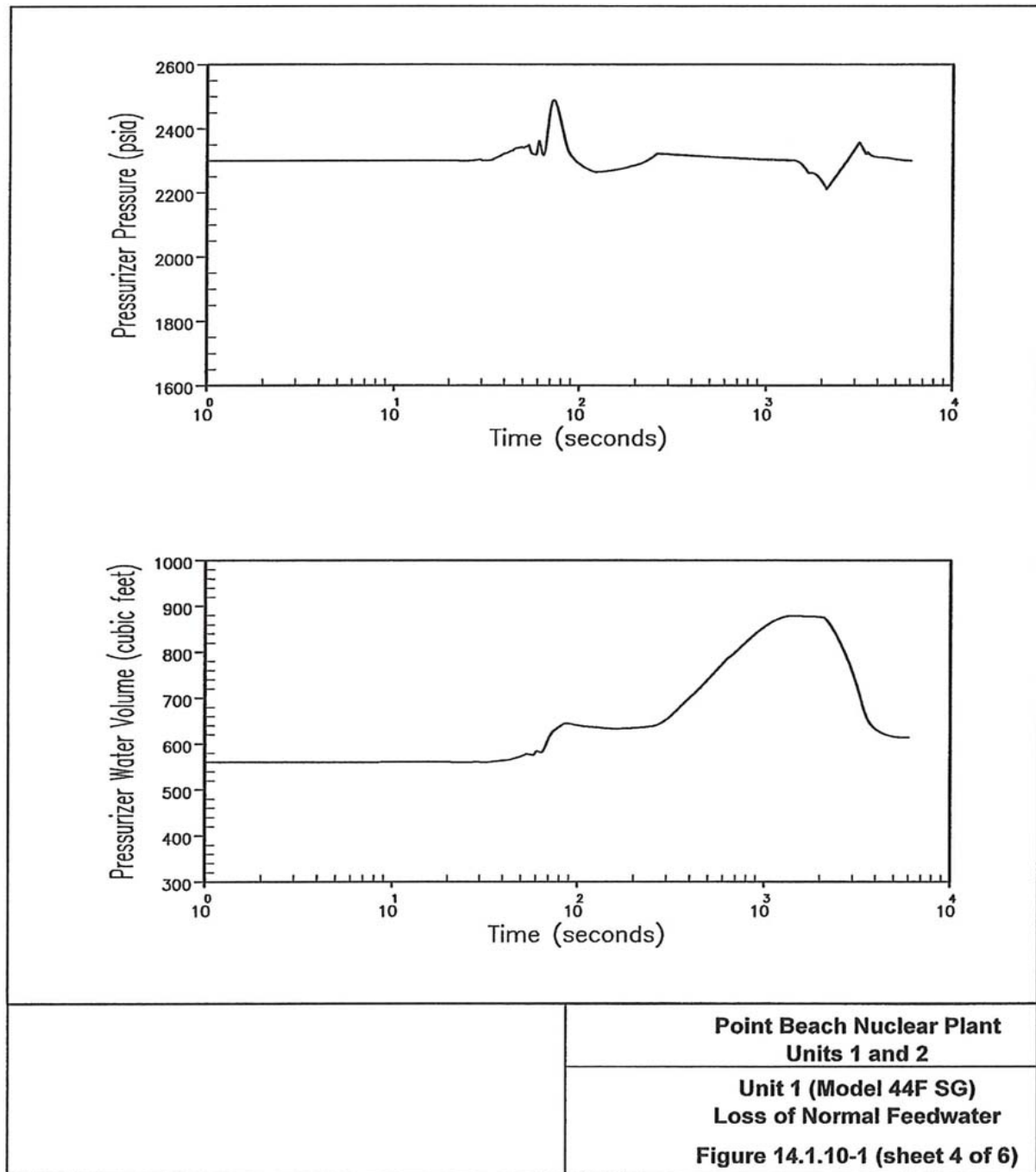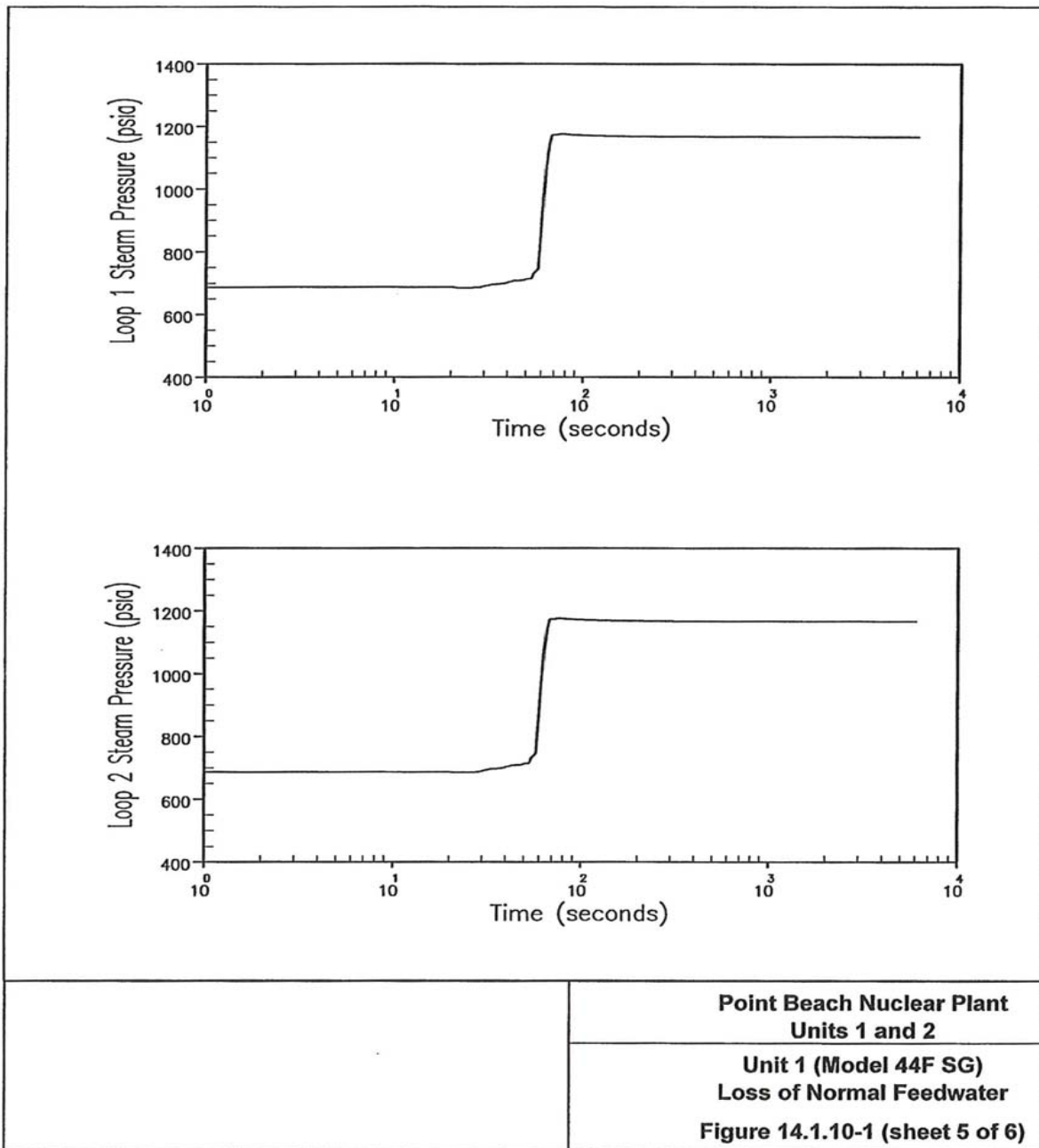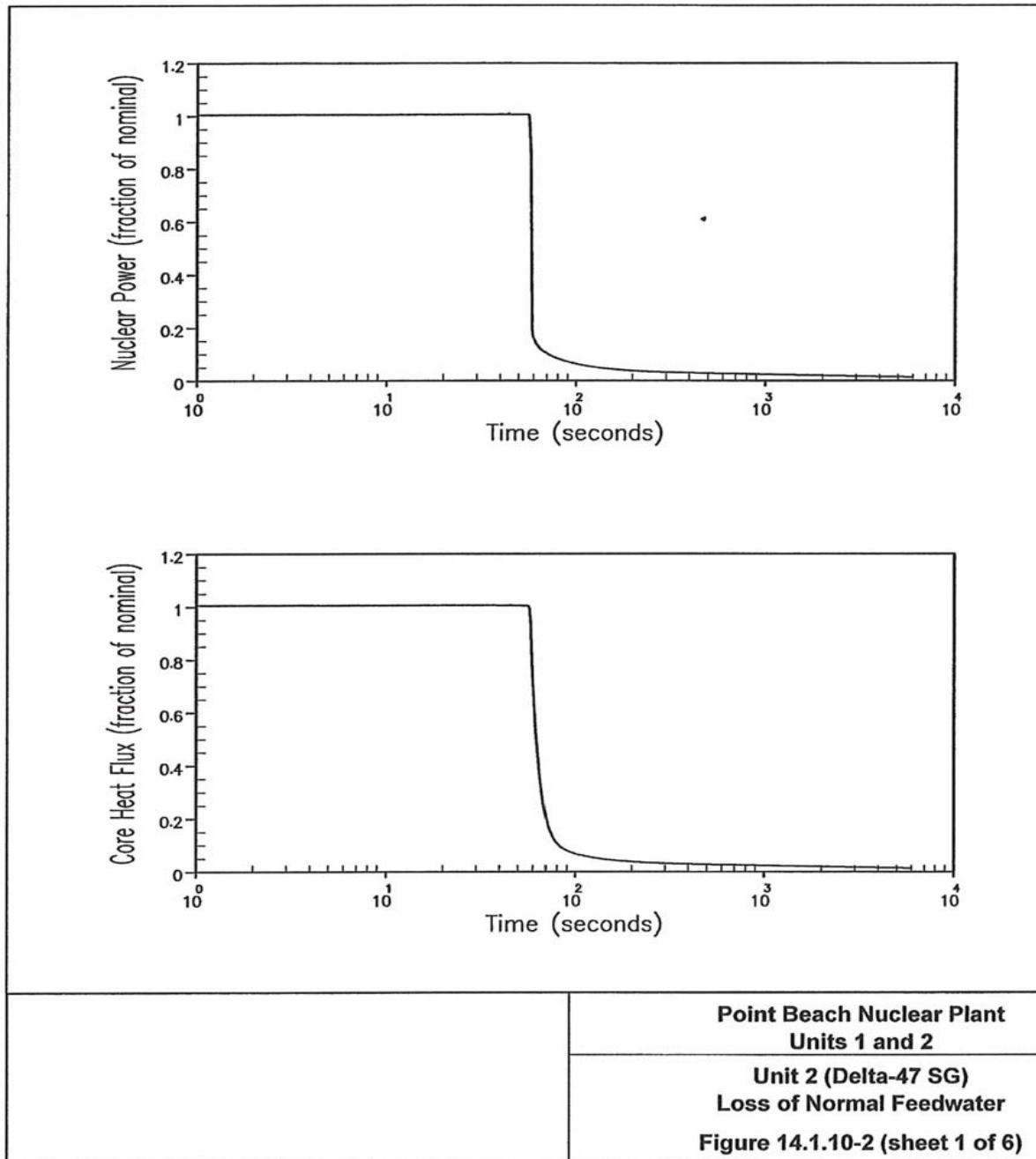
**Unit 2 (Delta-47 SG)
Loss of Normal Feedwater**

Figure 14.1.10-2 (sheet 2 of 6)

Figure 14.1.10-2 UNIT 2 (Delta - 47 SG) LOSS OF NORMAL FEEDWATER
Sheet 3 of 6

**Point Beach Nuclear Plant
Units 1 and 2**

**Unit 2 (Delta-47 SG)
Loss of Normal Feedwater**

Figure 14.1.10-2 (sheet 3 of 6)

Figure 14.1.10-2 UNIT 2 (Delta - 47 SG) LOSS OF NORMAL FEEDWATER
Sheet 4 of 6
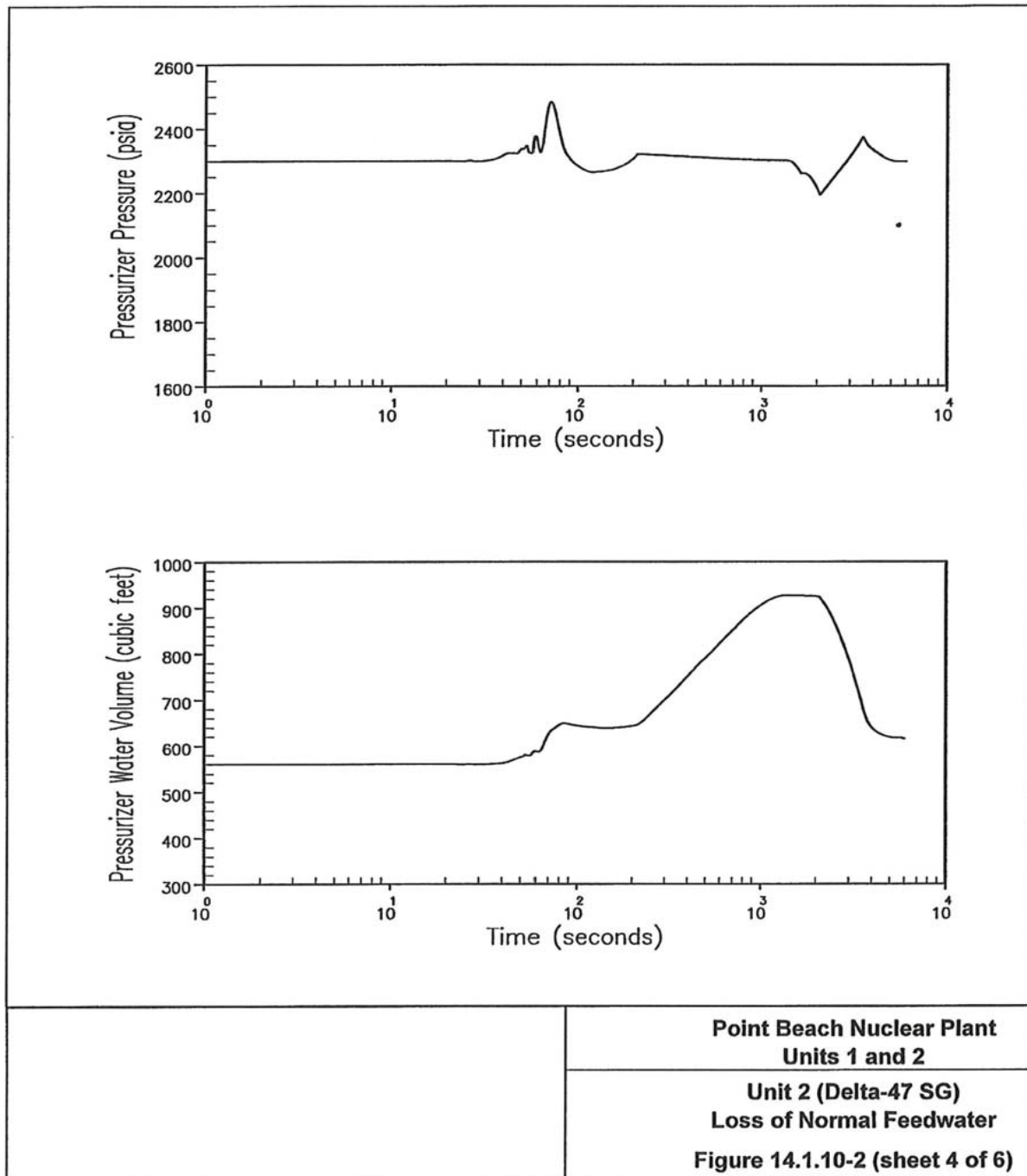
**Point Beach Nuclear Plant
Units 1 and 2**

**Unit 2 (Delta-47 SG)
Loss of Normal Feedwater**

Figure 14.1.10-2 (sheet 4 of 6)

Figure 14.1.10-2 UNIT 2 (Delta - 47 SG) LOSS OF NORMAL FEEDWATER
Sheet 5 of 6

**Point Beach Nuclear Plant
Units 1 and 2**

**Unit 2 (Delta-47 SG)
Loss of Normal Feedwater**

Figure 14.1.10-2 (sheet 5 of 6)

Figure 14.1.10-2 UNIT 2 (Delta - 47 SG) LOSS OF NORMAL FEEDWATER
Sheet 6 of 6

Point Beach Nuclear Plant
Units 1 and 2
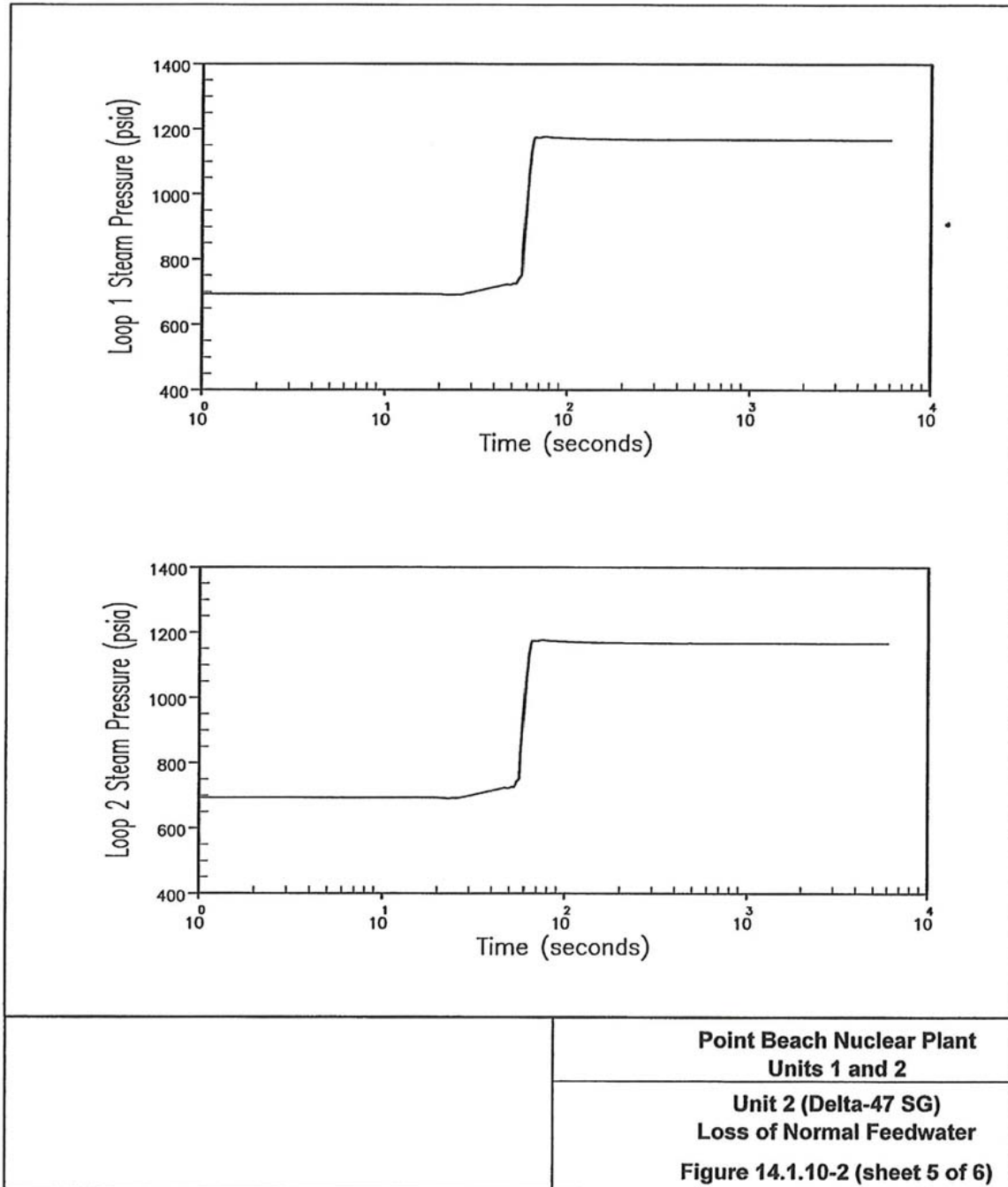
Unit 2 (Delta-47 SG)
Loss of Normal Feedwater

Figure 14.1.10-2 (sheet 6 of 6)
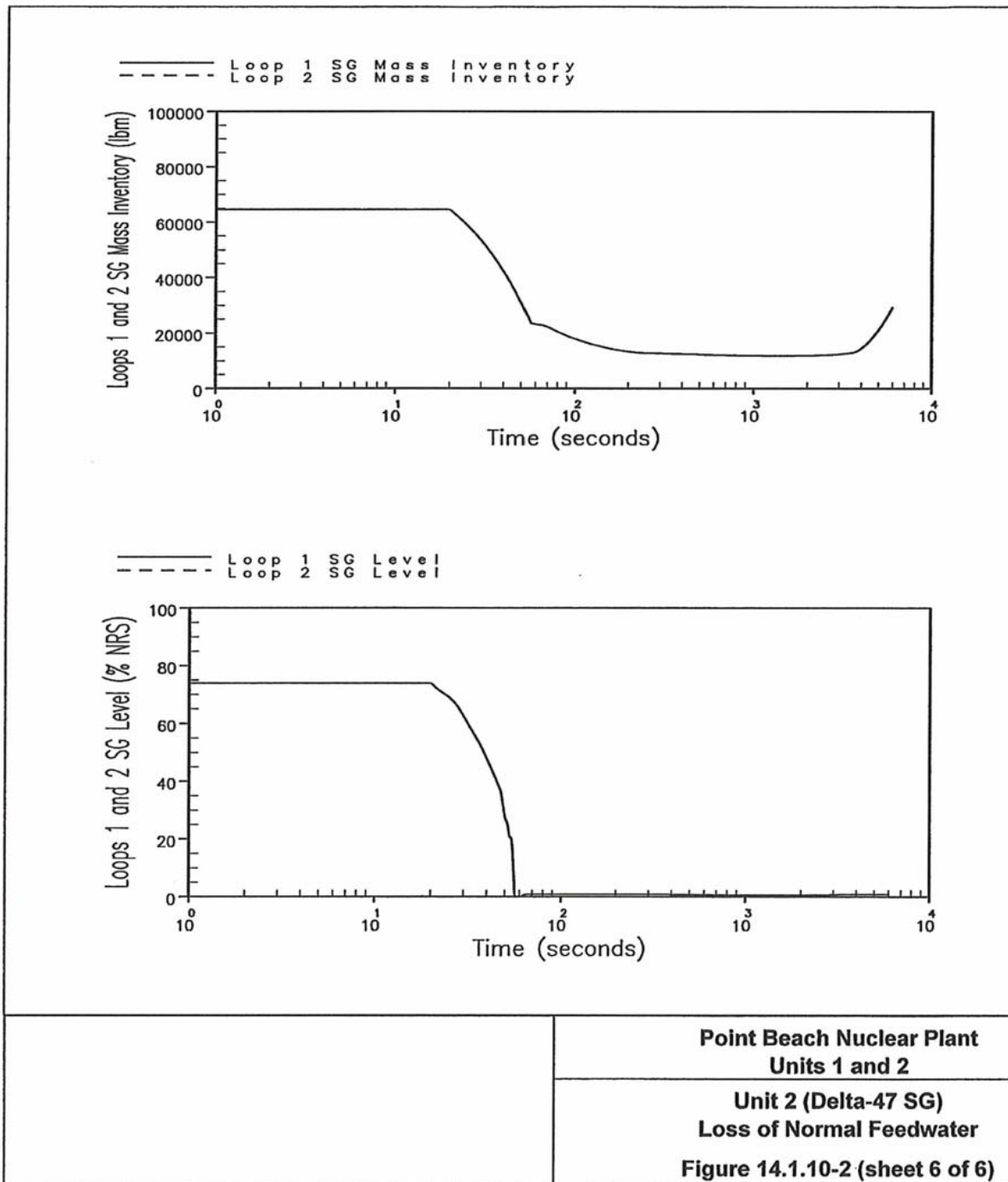
14.1.11 LOSS OF ALL AC POWER TO STATION AUXILIARIES

For the Point Beach EPU, the LONF (FSAR 14.1.10) refers to the loss of normal feedwater cases with offsite power available and the Loss of Offsite Power (LOOP) refers to the loss of normal feedwater cases with loss of offsite power. LOAC refers to the transient which is initiated by a loss of AC power. The LOAC cases are bounded by the LOOP cases, therefore the LOOP results are presented in this FSAR section as bounding results for the LOAC event.

In the unlikely event of a complete loss of all non-emergency AC power; the turbine will be tripped and there will be a loss of power to the station auxiliaries. The sequence below is described for the unit following a turbine trip:

1.   Plant vital instruments are supplied by the emergency power sources.

2.   As the steam system pressure subsequently increases, the steam generator power operated relief valves (also referred to as the atmospheric dump valves) are automatically opened to the atmosphere. Steam bypass to the condenser is not available because of loss of the circulating water pumps.

3.   As the steam flow rate through the power operated relief valves may not be sufficient, the steam generator self-actuated safety valves may temporarily lift to augment the steam flow until the rate of heat dissipation is sufficient to carry away the sensible heat of the fuel and coolant above no-load temperature plus the residual heat produced in the reactor.

4.   As the no-load temperature is reached, the steam generator power operated relief valves are used to dissipate the residual heat and to maintain the plant at the hot shutdown condition.

The steam turbine driven and motor driven auxiliary feedwater pumps are automatically started by the loss of AC power on the buses that supply power to the Main Feedwater Pumps. The turbine utilizes steam from the secondary system to drive the feedwater pump to deliver makeup water to the steam generators. The turbine driver exhausts the secondary steam to the atmosphere. The motor driven auxiliary feedwater pump is supplied by power from an emergency diesel generator. The pumps take suction directly from the condensate storage tanks (CSTs) for delivery to the steam generators, or from the Service Water System should the CSTs not be available. See Section 10.2.3 and Section 7.4.3 for a description of the automatic switchover of the AFW suction supply to Service Water. The auxiliary feedwater system insures feedwater supply of at least 275 gpm upon the loss of power to the station auxiliaries, since the steam turbine driven auxiliary feedwater pump and the motor driven auxiliary feedwater pumps have a minimum capacity of 275 gpm each.

Method of Analysis

A detailed analysis using the RETRAN code (Reference 2) is performed in order to obtain the plant transient following a loss of all AC power to the station auxiliaries. The simulation describes the plant thermal kinetics, RCS including the natural circulation, pressurizer, steam generators, and feedwater system. The digital program computes pertinent variables, including the steam generator level, pressurizer water level, and reactor coolant average temperature.

The following assumptions are made:

1. The plant is initially operating at 100.6% of 1806 MWt.

2. Core residual heat generation is based on the 1979 version of ANS-5.1 (Reference 1) plus two standard deviations for uncertainty. ANSI/ANS-5.1 - 1979 is a conservative representation of the decay heat release rates.

3. The initiating signal is a low-low steam generator level. This assumption conservatively disregards that a loss of AC power to the station auxiliaries would result in an immediate reactor trip due to loss of voltage to the 4.16kV busses.

4. Both steam generators are affected equally, and both reach their low-low level trip setpoints simultaneously. This assumption conservatively minimizes the secondary heat sink available at the time of the reactor trip.

5. The auxiliary feedwater system provides only 275 gpm of flow split equally to two steam generators. Note: 275 gpm to a single steam generator is sufficient to mitigate the transient but is not within the capability of the system as analyzed because the maximum AFW flow from a motor driven auxiliary feedwater pump to a single steam generator is limited to approximately 230 gpm by a cavitating venturi (Reference 6 and Reference 7).

6. AFW flow of 275 gpm is delivered to the steam generator(s) starting at 60 seconds after the initiating signal (low-low steam generator level trip). From 60 to 90 seconds the AFW flow is ramped from 0% to 80% of total flow: from 90 to 150 seconds AFW flow is ramped from 80% to 100% of total flow: beyond 150 seconds 100 % of total AFW flow is maintained.

7. The assumed steam generator models are 44F (Unit 1) and Delta-47 (Unit 2).

8. Secondary system steam relief is through the self-actuated safety valves.

9. The pressurizer sprays are assumed to function as designed which maximizes the peak pressurizer water volume. The backup heaters are assumed to be unavailable on high pressurizer level deviation signal and the PORVs are assumed to be inoperable for the Unit 1 limiting case and operable for the Unit 2 limiting case.

The remaining assumptions used in the analysis are similar to the loss of normal feedwater (14.1.10) except that power is assumed to be lost to the reactor coolant pumps at the time of reactor trip plus an appropriate delay time (2 sec. for reactor trip and 2 sec. for loss of power for a total of 4 sec.).

Results

The calculated sequence of events for this accident is listed in Table 14.1.11-1. The transient response of the RCS following a loss of AC power is shown in Figure 14.1.11-1 and Figure 14.1.11-2.

The first few seconds after the loss of power to the reactor coolant pumps will closely resemble the simulation of the loss of reactor coolant flow event (14.1.8), where core damage due to rapidly increasing core temperatures is prevented by promptly tripping the reactor. After the reactor trip, stored and residual decay heat must be removed to prevent damage to either the RCS or the core.

The results of the analysis show that the natural circulation flow available is sufficient to provide adequate core decay heat removal following reactor trip and RCP coastdown. An inadvertent actuation of the shut down seal (SDS) on a rotating pump shaft will not have any measurable impact on RCP coastdown or on the pump's capability to provide sufficient cooling flow to the reactor core (Reference 8).

Conclusion

The loss of AC power to the station auxiliaries does not cause any adverse condition in the core, since it does not result in water relief from the pressurizer relief or safety valves.

References

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

2.   Huegel, D. S., et. al., "RETRAN-02 Modeling and Qualification for Westinghouse Pressurized Water Reactor Non-LOCA Safety Analyses," WCAP-14882-P-A (Proprietary), WCAP-15234-A (Non-Proprietary), April 1999.

3.   Westinghouse Calculation Note CN-TA-08-79, Rev 1, Point Beach Unit 1 and 2 Loss of Normal Feedwater/Loss of AC Power (LONF/LOAC) Analysis for the EPU Program, Approved February 26, 2009.

4.   NRC Safety Evaluation, PBNP Units 1 and 2 - Issuance of License Amendments Regarding Extended Power Uprate, May 3, 2011.

5.   NRC Safety Evaluation, "Point Beach Nuclear Plant Units 1 and 2-Issuance of License Amendments Re: Auxiliary Feedwater System Modification," dated March 25, 2011.

6.   Letter NRC 2011-0086, NextEra Energy to NRC, Clarification/Comments on the NRC Safety Evaluation Report, Amendment Nos. 238 (Unit 1) and 242 (Unit 2), Auxiliary Feedwater System Modification, September 16, 2011.

7.   NRC Letter to NextEra Energy, Point Beach Nuclear Plant, Units 1 and 2-NRC Staff Response to Clarification/Comments Related to the Safety Evaluation Report Associated with the Auxiliary Feedwater System Modification License Amendment, December 6, 2011.

8.   Letter WEP-14-64, Westinghouse to NextEra Energy, Point Beach RCP SDS Final Documentation Deliverable, October 9, 2014.

Table 14.1.11-1 TIME SEQUENCE OF EVENTS FOR LOSS OF OFFSITE POWER
INCIDENTS*

	Time of Each Event (Seconds)	
Event	Unit 1	Unit 2
Main feedwater flow stops	20	20
Low-Low steam generator water level trip	63.5	55.0
Rods begin to drop	65.5	57.0
Reactor coolant pumps begin to coastdown	67.5	59.0
AFW flow to each loop begins	123.5	115.0
80% of full AFW flow reached	153.5	145.0
100% of full AFW flow reached	213.5	205.0
Peak water level in pressurizer occurs	285	776
Core decay heat decreases to auxiliary feedwater heat removal capacity	~844	~790

* Nonemergency AC power to station auxiliaries is lost at the times shown above for the start of RCP coastdown.

Figure 14.1.11-1 UNIT 1 (MODEL 44F SG) LOSS OF AC POWER
Sheet 1 of 6
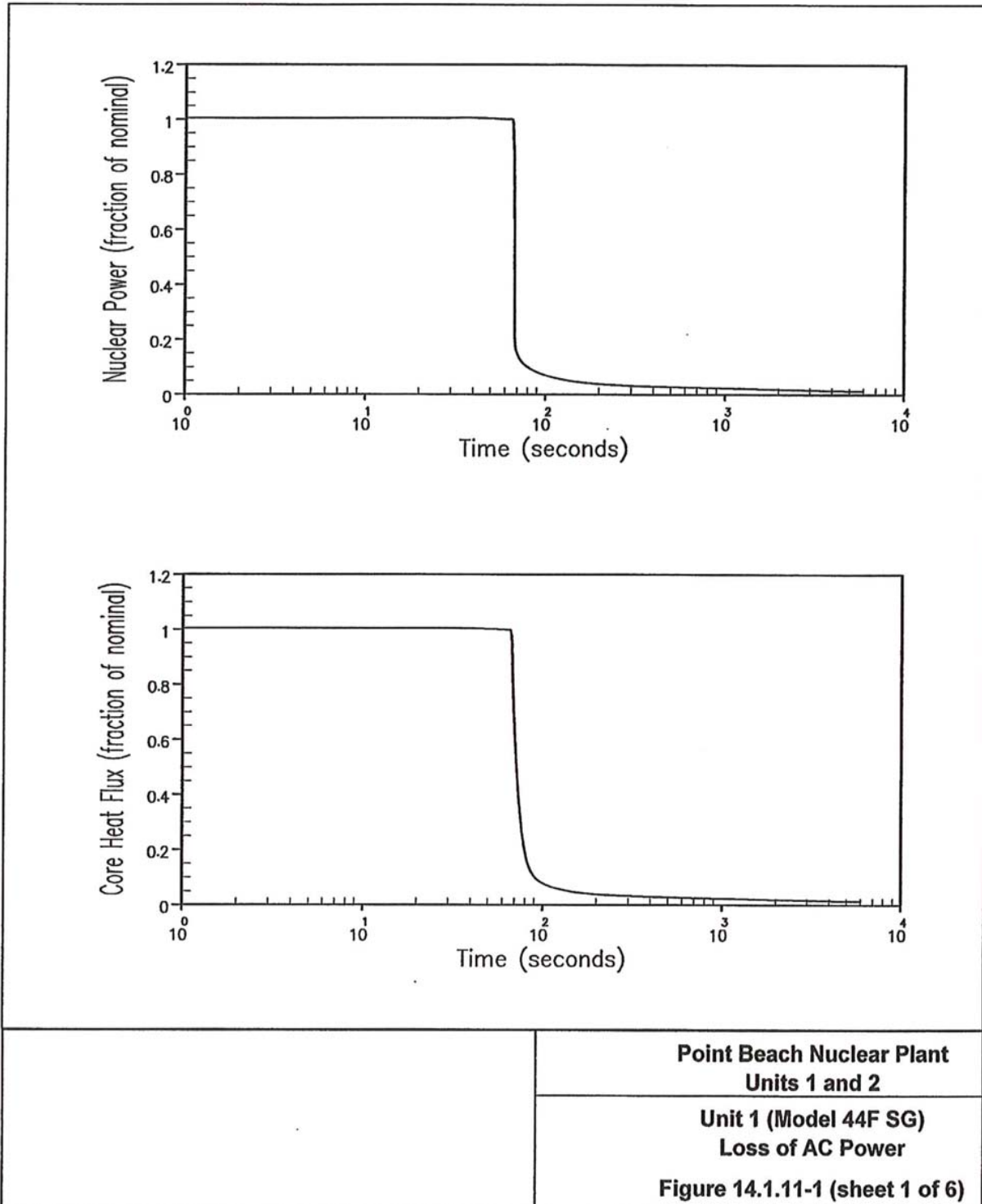
**Point Beach Nuclear Plant
Units 1 and 2**

**Unit 1 (Model 44F SG)
Loss of AC Power**

Figure 14.1.11-1 (sheet 1 of 6)

Figure 14.1.11-1 UNIT 1 (MODEL 44F SG) LOSS OF AC POWER
Sheet 2 of 6
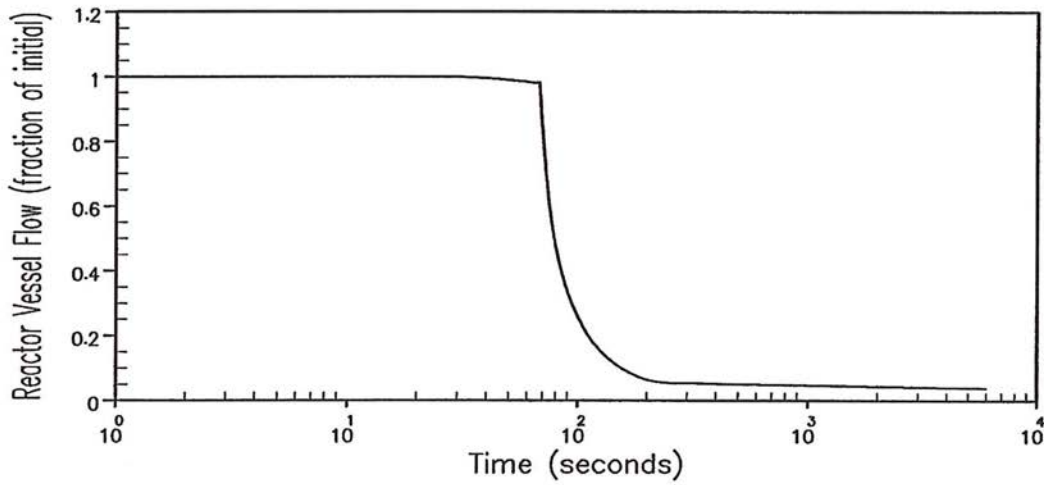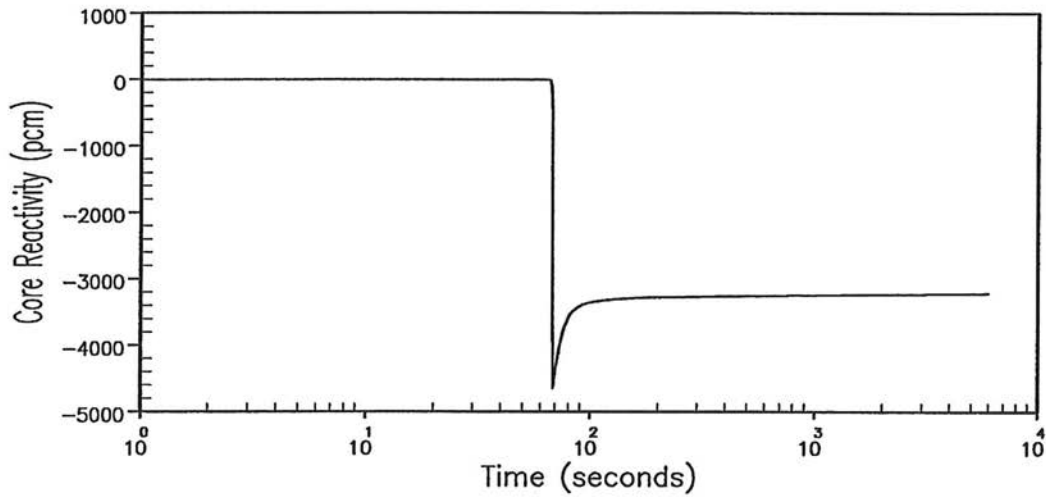
**Point Beach Nuclear Plant
Units 1 and 2**

**Unit 1 (Model 44F SG)
Loss of AC Power**

Figure 14.1.11-1 (sheet 2 of 6)

Figure 14.1.11-1 UNIT 1 (MODEL 44F SG) LOSS OF AC POWER
Sheet 3 of 6

Point Beach Nuclear Plant
Units 1 and 2

Unit 1 (Model 44F SG)
Loss of AC Power

Figure 14.1.11-1 (sheet 3 of 6)

Figure 14.1.11-1 UNIT 1 (MODEL 44F SG) LOSS OF AC POWER
Sheet 4 of 6

**Point Beach Nuclear Plant
Units 1 and 2**

**Unit 1 (Model 44F SG)
Loss of AC Power**

Figure 14.1.11-1 (sheet 4 of 6)

Figure 14.1.11-1 UNIT 1 (MODEL 44F SG) LOSS OF AC POWER
Sheet 5 of 6

Point Beach Nuclear Plant
Units 1 and 2

Unit 1 (Model 44F SG)
Loss of AC Power

Figure 14.1.11-1 (sheet 5 of 6)

Figure 14.1.11-1 UNIT 1 (MODEL 44F SG) LOSS OF AC POWER
Sheet 6 of 6

Point Beach Nuclear Plant
Units 1 and 2

Unit 1 (Model 44F SG)
Loss of AC Power

Figure 14.1.11-1 (sheet 6 of 6)

Figure 14.1.11-2 UNIT 2 (DELTA - 47 SG) LOSS OF AC POWER
Sheet 1 of 6

Point Beach Nuclear Plant
Units 1 and 2

Unit 2 (Delta-47 SG)
Loss of AC Power

Figure 14.1.11-2 (sheet 1 of 6)

Figure 14.1.11-2 UNIT 2 (DELTA - 47 SG) LOSS OF AC POWER
Sheet 2 of 6

**Point Beach Nuclear Plant
Units 1 and 2**

**Unit 2 (Delta-47 SG)
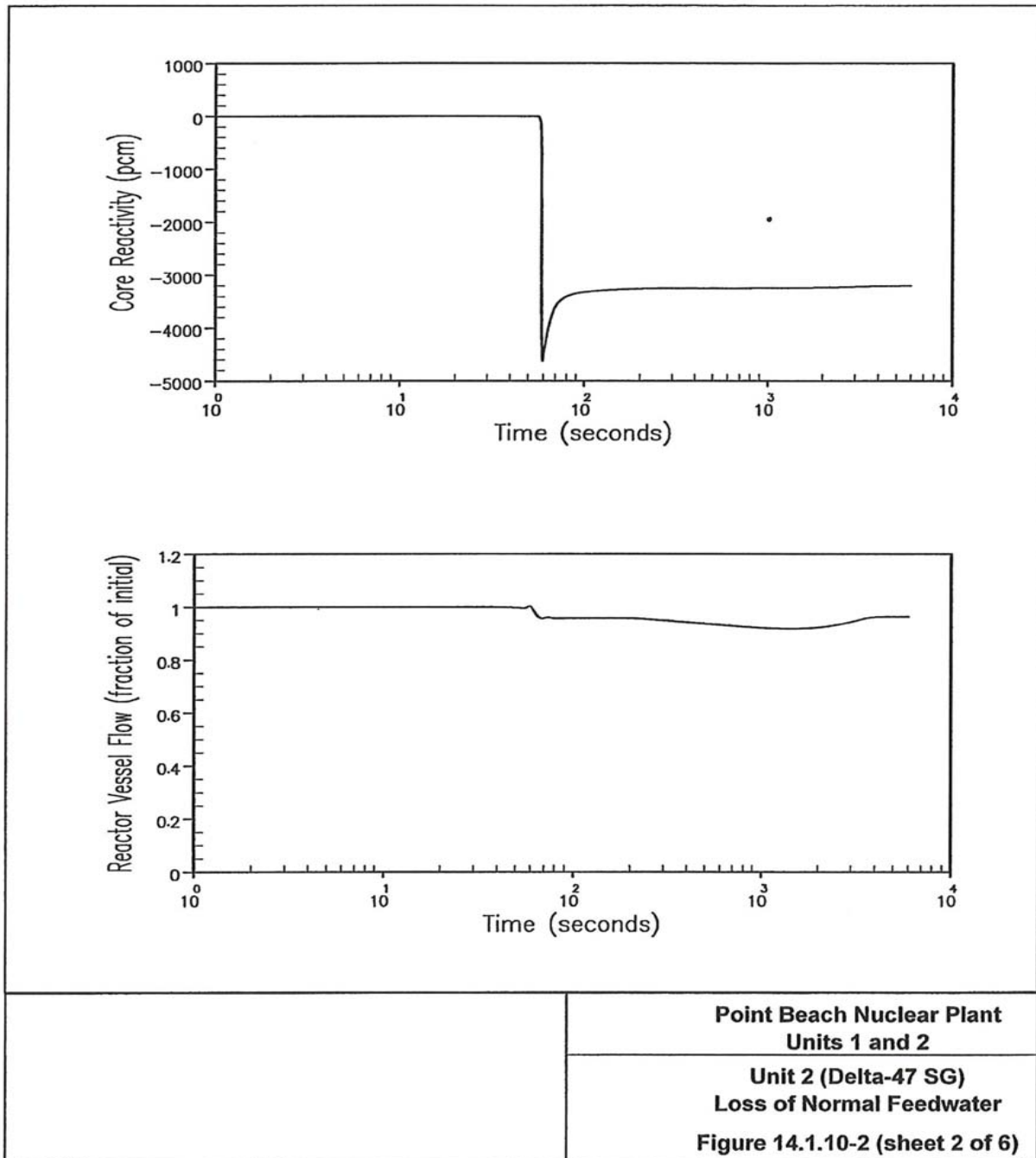Loss of AC Power**

Figure 14.1.11-2 (sheet 2 of 6)

Figure 14.1.11-2 UNIT 2 (DELTA - 47 SG) LOSS OF AC POWER
Sheet 3 of 6

**Point Beach Nuclear Plant
Units 1 and 2**

**Unit 2 (Delta-47 SG)
Loss of AC Power**

Figure 14.1.11-2 (sheet 3 of 6)
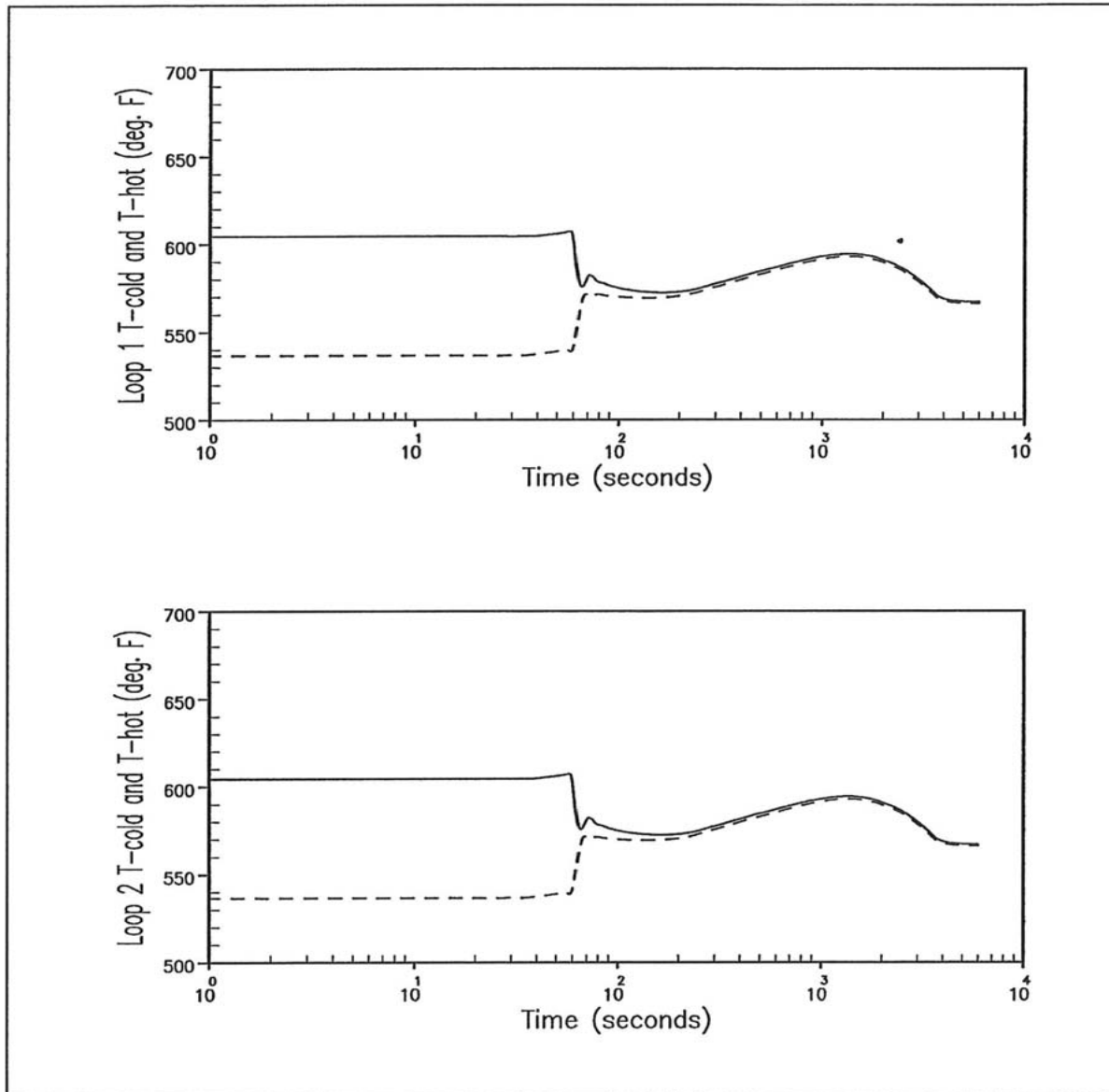
Figure 14.1.11-2 UNIT 2 (DELTA - 47 SG) LOSS OF AC POWER
Sheet 4 of 6

Point Beach Nuclear Plant
Units 1 and 2

Unit 2 (Delta-47 SG)
Loss of AC Power

Figure 14.1.11-2 (sheet 4 of 6)

Figure 14.1.11-2 UNIT 2 (DELTA - 47 SG) LOSS OF AC POWER
Sheet 5 of 6

Point Beach Nuclear Plant
Units 1 and 2

Unit 2 (Delta-47 SG)
Loss of AC Power

Figure 14.1.11-2 (sheet 5 of 6)

Figure 14.1.11-2 UNIT 2 (DELTA - 47 SG) LOSS OF AC POWER
Sheet 6 of 6

**Point Beach Nuclear Plant
Units 1 and 2**

Unit 2 (Delta-47 SG)
Loss of AC Power

Figure 14.1.11-2 (sheet 6 of 6)

14.1.12 LIKELIHOOD OF TURBINE-GENERATOR UNIT OVERSPEED

The present advanced state of the art of rotor forging and inspection techniques guarantees practically defect-free turbine rotors. Due to the redundancy and reliability of the turbine control protection system and of the steam system, the probability of occurrence of a unit over-speeding above the design value, i.e., 132%, is very remote.

A description and operation of the electro-hydraulic governing system is located in Section 10.1. WCAP-7525-L (Reference 2) contains a description of the additional overspeed protection system and a reliability analysis demonstrating that the likelihood of exceeding the maximum design speed (i.e., 132% of rated) is practically zero. Also included are the bases of the maximum design, the characteristics of the missiles that could be generated between rated and the maximum design speed, and an analysis of the plant's capability to withstand such a missile.

Besides the provisions in the design of the turbine control and protection system during plant operation, turbine stop and governor valves will be exercised on a periodic basis to further preclude the possibility of a valve stem sticking. In addition, the turbine is periodically tested to verify the tripping speed. The remaining tripping devices are periodically checked and oil samples are analyzed.

Consequences Of Turbine-Generator Unit Overspeeding

Prior to the original licensing of Point Beach Nuclear Plant (PBNP), the analysis on the consequences of turbine overspeed indicated there would be only a low energy missile generated external to the low pressure turbine casing. The basic assumptions used in the analysis that led to this conclusion were deemed reasonable, at the time, by all parties involved.   However, because of the potential serious consequences of external missiles, Westinghouse initiated a series of model tests to substantiate these assumptions.

The tests involved bursting of simulated low pressure turbine discs within various stationary steel cylinders modeled to approximate blade rings, inner cylinders and the outer casing. The most significant test findings were: (1) penetration occurs mostly by local punching with little bending or stretching of stationary steel; (2) a disc fragment can wedge a path between two blade rings if a blade ring is not directly opposite the rim of the disc. In either case, the stationary steel has less energy absorbing capability than originally expected, and as such, the energy required to penetrate is minimized.

As a result of these test series, new criteria were evolved for predicting the missile containing ability of the low pressure turbine structures. The previous calculations were redone using these new criteria and the results show the original position of only a low energy missile generated external to the turbine casing in the event of a turbine overspeed could no longer be maintained for original type rotor designs using shrunk-on blade discs.

Rupture of a low pressure turbine disk at or below design speed was postulated for design purposes, even though this failure is shown to have a very low probability because of design conservatism and original quality control. As a result of the updated missile generation studies, PBNP installed the independent overspeed protection system (IOPS) and the crossover steam dump system in both units. IOPS was designed and installed to be single failure proof to meet NRC commitments (Reference 12).

Since the original plant licensing, the bases for the consideration of the consequences of turbine-generator overspeeding have been modified. In a letter to Mr. James A. Martin, Westinghouse Electric Corporation Generation Technology Systems Division, dated February 2, 1987, the Nuclear Regulatory Commission (NRC) staff presented its views on precluding turbine missiles and consequential damage to safety-related structures, systems and components. The staff established that utilizing testing and inspection to maintain an initial small value of the probability of a turbine failure resulting in the ejection of fragments through the turbine casing simplifies and improves procedures for evaluation of turbine missile risks and ensures that the public health and safety are maintained. The staff provided Wisconsin Electric with the generic turbine failure guidelines for total turbine missile generation probabilities to be used for determining frequencies for turbine disc ultrasonic inspections and maintenance and testing schedules for turbine control and overspeed protection systems.

In response to the NRC guidance letter, Westinghouse prepared a report, Reference 3, for the Turbine Valve Test Frequency Evaluation Subgroup (which included Wisconsin Electric Power Company) of the Westinghouse Owners Group. That report provided a detailed probabilistic basis for extending the testing intervals of turbine valves. In performing the study, Westinghouse considered many variations of turbine stop valves and trip systems. As discussed on Page 5-2 of Reference 3, the availability of redundant overspeed protection systems, such as described in Reference 2, was not credited in the analyses because only eight of the nineteen units in the subgroup had one of these systems. The NRC approved the methodology developed in Reference 3 in the supplement to a safety evaluation dated February 7, 1989 for Northern States Power, which was the lead utility for the annual turbine valve testing. In another letter to the chairman of the turbine testing subgroup dated November 2, 1989, the NRC staff provided its generic conclusions regarding license amendment requests for changes in surveillance intervals for turbine valve tests and the applicability of Reference 3 to support these requests.

Subsequently, based on Reference 3, issuance of License Amendments 129 and 133 (Reference 4) dated October 16, 1991, for Units 1 and 2 respectively revised the turbine stop and governor valve testing from a monthly to an annual interval. They also required a commitment to work with the turbine vendor to maintain a turbine valve data base for the purpose of tracking changes in valve component failure rates, to accumulate and review valve failure rate information at least every three years to determine if the testing frequency requires modification, and to review the turbine valve test frequency anytime that major changes to the turbine system are made or a significant upward trend in turbine valve failure rate is identified.

In keeping with the Wisconsin Electric commitment to work with the turbine vendor, two reports (Reference 5 & Reference 6) were generated. These reports specified the appropriate turbine valve test intervals for PBNP. These reports recommended quarterly test, not annual. In September 2000, the WOG finalized the evaluation of BB-95/96 turbine valve failures. This report (Reference 10) concluded that the turbine valve testing frequency be quarterly and that the testing include checking the integrity of the stop valves in addition to freedom of movement and position verification. A more recent report (Reference 14 and Reference 15) recommends the maximum interval for turbine valve testing to be six months.

Unit 1 and Unit 2 each have had the original shrunk on blade disc design low pressure turbine rotors replaced with ruggedized monoblock rotors with integrally forged blade discs and complete low pressure turbine internal stream path. The replacements were done in 1997 (Unit 1) and 1998-1999 (Unit 2). Replacing the original design Low Pressure turbine rotors with ruggedized monoblock rotors with integrally forged blade discs significantly reduces concerns associated with missile generation associated with a turbine overspeed. Monoblock rotor construction with integrally forged blade discs are not susceptible to the same failure modes as the old rotors which use the shrunk-on discs.

The effect of the new rotors on turbine generator train overspeed is both positive and negative. Because of the increased efficiency of the new steam path, residual steam is allowed to perform more work towards increasing the overspeed of the unit during a trip. This effect is offset however, by the increased inertia of the new rotors with a net reduction in turbine overspeed for any given trip scenario. This effect is explained more fully in the PBNP overspeed analysis report (Reference 8) prepared by Westinghouse. Another overspeed evaluation (Reference 11) was performed by Siemens to support Extended Power Uprate (EPU). These reports provide the basis for the overspeed requirements contained in Technical Requirements Manual (TRM) 3.7.6. TRM 3.7.6 describes the operability requirements for the turbine mechanical overspeed trip system, the IOPS, and the crossover steam dump system for reactor power operation above 1518.5 MWt.

The missile generation report (Reference 9) developed by Westinghouse in 1984 and submitted to the NRC, examined the probability of various failure modes for their fully integral nuclear low pressure turbine rotors. Conclusions from this report state that the likelihood of missile generation from all mechanisms for the replacement monoblock rotors is significantly reduced. A condensed version of that report (Reference 13) was prepared and sent to PBNP in 1996 for the low pressure turbine replacements. Ductile bursting of the new rotors will not occur until the speed reaches greater than 177% rated speed. Based on the latest missile generation reports and the overspeed protection requirements and testing contained in TRM 3.7.6, the probability of missile generation for the PBNP turbine-generators is below the NRC safety criteria.

References

1.   "History of the Special ASTM Task Force on Large Turbine and Generator Rotors," R. M. Curran, ASTM Annual Meeting, 1965, Purdue University.

2.   WCAP-7525-L, (Proprietary), "Likelihood and Consequences of Turbine Overspeed at the Point Beach Nuclear Plant," J. N. Fox, June 1970.

3.   WCAP-11525, (Proprietary), "Probabilistic Evaluation of Reduction in Turbine Valve Test Frequency," June 1987.

4.   Robert B. Samworth, NRC, to James J. Zach, Amendment Nos. 129 and 133 to Facility Operating License Nos. DPR-24 and DPR-27, dated October 16, 1991.

5.   WOG-TVTF-93-17, "Westinghouse Owners Group, TVTF Subgroup, Final Report, Update of BB-95/96 Turbine Valve Failure Rates and Effect of Destructive Overspeed Probabilities," August 6, 1993.

6. WOG-TVTF-93-24, "Westinghouse Owners Group, TVTF Evaluation for Point Beach Units 1 & 2," December 17, 1993.

7. Technical Requirements Manual TLCO 3.7.6, Turbine Overspeed Protection.

8. "Overspeed Analysis for Wisconsin Electric Power Company Point Beach 1 & 2 with New LP Rotors and with Uprated Conditions", Westinghouse Electric Corporation Technical Report TR-98022, (Proprietary), dated March 6, 1998.

9. WSTG-4-NP, (Proprietary), Analysis of the Probability of the Generation of Missiles from Fully Integral Nuclear Low Pressure Rotors. Submitted to Nuclear Regulatory Commission in October, 1984.

10. "Final Update and Evaluation of BB95/96 Turbine Valve Failure Database", Westinghouse Owners Group report MUHP 7002/8002 dated September 2000.

11. Siemens EC 08109, "FPL Point Beach 1 & 2 Missile Analysis Report, Overspeed Setpoints, and Overspeed Analysis," (Confidential) dated July 25, 2008.

12. "Turbine Overspeed Protection & Crossover Steam Dump Operability", Design Basis Document Discussion Paper dated August 1994.

13. "Results of the Analysis of the Probability of the Generation of Missiles from Fully Integral Nuclear Low Pressure Rotors", (Proprietary), CT-27151, dated September 12, 1996.

14. WCAP-16054-P, (Proprietary), "Probabilistic Analysis of Reduction in Turbine Valve Test Frequency for Nuclear Plants with Siemens-Westinghouse BB-95/96 Turbines," dated April 2003

15. Westinghouse Electric Company WEP-11-81, "Evaluation of the Continued Applicability of WCAP- 16054-P to the Modified Point Beach Units 1 and 2 Turbine Control Valves - Final Report," (Proprietary) dated December 14, 2011.

14.2   STANDBY SAFETY FEATURES ANALYSIS

Adequate provisions have been included in the design of the plant and its standby engineered safety features to limit potential exposure of the public for situations which have a very low probability of occurrence, but which could conceivably involve uncontrolled releases of radioactive materials to the environment. The situations which have been considered are:

1.   Fuel Handling Accidents
2.   Accidental Release of Waste Liquid
3.   Accidental Release of Waste Gases
4.   Rupture of a Steam Generator Tube
5.   Rupture of a Steam Pipe
6.   Rupture of a Control Rod Drive Mechanism Housing - Rod Cluster Control Assembly (RCCA) Ejection

14.2.1   FUEL HANDLING ACCIDENT

The following handling accidents are evaluated to ensure that no hazards are created:

1.   A fuel assembly becomes stuck inside reactor vessel.
2.   A fuel assembly or control rod cluster is dropped onto the floor of the reactor cavity or spent fuel pool.
3.   A fuel assembly becomes stuck in the penetration valve.
4.   A fuel assembly becomes stuck in the transfer carriage or the carriage becomes stuck.

The possibility of a fuel handling incident is very remote because of the many administrative controls and physical limitations imposed on fuel handling operations. All refueling operations are conducted in accordance with prescribed procedures under direct surveillance of a supervisor technically trained in nuclear safety. Also, before any refueling operations begin, verification of complete rod cluster control assembly insertion is obtained by tripping the rods to obtain indication of rod drop and disengagement from the control rod drive mechanisms.   Boron concentration in the coolant is raised to the refueling concentration and verified by sampling. Refueling boron concentration is sufficient to maintain the clean, cold, fully loaded core subcritical with all rod cluster assemblies withdrawn. The refueling cavity is filled with water meeting the same boric acid specifications. As the vessel head is raised, a visual check is made to verify that the drive shafts are free in the mechanism housing.

After the vessel head is removed, the rod cluster control drive shafts are removed from their respective assemblies using the containment fuel handling crane and the shaft unlatching tool. A load cell is used to indicate that the drive shaft is free of the control cluster as the lifting force is applied.

The fuel handling manipulators and hoists are designed so that fuel cannot be raised above a position which provides adequate shield water depth for the safety of operating personnel. This safety feature applies to handling facilities in both the containment and in the spent fuel pool area. In the spent fuel pool, the design of storage racks and manipulation facilities is such that:

> Fuel at rest is positioned by positive restraints in an eversafe, always subcritical, geometrical array, with no credit for boric acid in the water.

Fuel can be manipulated only one assembly at a time.

Violation of procedures by placing one fuel assembly in juxtaposition with any group of assemblies in racks will not result in criticality.

Adequate cooling of fuel during underwater handling is provided by convective heat transfer to the surrounding water. The fuel assembly is immersed continuously while in the refueling cavity or spent fuel pool.

Even if a spent fuel assembly becomes stuck in the transfer tube, natural convection will maintain adequate cooling. The fuel handling equipment is described in detail in Section 9.0.

Two Nuclear Instrumentation System source range channels are continuously in operation and provide warning of any approach to criticality during refueling operations. This instrumentation provides a continuous audible signal in the containment, and would annunciate a local horn and a horn and light in the plant control room if the count rate increased above a preset low level.

Refueling boron concentration is sufficient to maintain the clean, cold, fully loaded core subcritical by at least 5% $\Delta \rho$ with all rod cluster control assemblies inserted. At this boron concentration, the core would also be more than 2% $\Delta \rho$ subcritical with all control rods withdrawn. The refueling cavity is filled with water meeting the same boric acid specifications.

All these safety features make the probability of a fuel handling incident very low.   Nevertheless, it is possible that a fuel assembly could be dropped during the handling operations. Therefore, this incident is analyzed both from the standpoint of radiation exposure and accidental criticality.

Special precautions are taken in all fuel handling operations to minimize the possibility of damage to fuel assemblies during transport to and from the spent fuel pool and during installation in the reactor. All handling operations on irradiated fuel are conducted under water. The handling tools used in the fuel handling operations are conservatively designed and the associated devices are of a fail-safe design.

In the fuel storage area, the fuel assemblies are spaced in a pattern which prevents any possibility of a criticality accident. In addition, the design is such that only one fuel assembly can be handled at a given time.

The motions of the cranes which move the fuel assemblies are limited to a relatively low maximum speed. Caution is exercised during fuel handling to prevent the fuel assembly from striking another fuel assembly or structures in the containment or fuel storage building. The fuel handling equipment suspends the fuel assembly in the vertical position during fuel movements, except when the fuel is moved through the transport tube.

The design of the fuel assembly is such that the fuel rods are restrained by grid clips which provide a total restraining force of approximately 60 lb. on each fuel rod. If the fuel rods are in contact with the bottom plate of the fuel assembly, any force transmitted to the fuel rods is limited due to the restraining force of the grid clips. The force transmitted to the fuel rods during fuel handling is not sufficient to breach the fuel rod cladding. If the fuel rods are not in contact with the bottom plate of the assembly, the rods would have to slide against the 60 lb. friction force. This would absorb the shock and thus limit the force on the individual fuel rods.   After the reactor

is shut down, the fuel rods contract during the subsequent cooldown and would not be in contact with the bottom plate of the assembly. Considerable deformation would have to occur before the rod would make contact with the top plate and apply any appreciable load on the fuel rod. Based on the above, it is felt that it is unlikely that any damage would occur to the individual fuel rods during handling. If one assembly is lowered on top of another, no damage to the fuel rods would occur that would breach the integrity of the cladding.

If during handling the fuel assembly strikes against a flat surface, the loads would be distributed across the fuel assemblies and grid clips and essentially no damage would be expected in any fuel rods. If the fuel assembly were to strike a sharp object, it is possible that the sharp object might damage the fuel rods with which it comes in contact, but breaching of the cladding is not expected.

The refueling operation experience that has been obtained with Westinghouse reactors has verified the expectation that no fuel cladding integrity failures occur during any fuel handling operations.

Rupture of all fuel elements in a withdrawn assembly is assumed as a conservative limit for evaluating the environmental consequences of a fuel handling incident. The remaining fuel assemblies are so protected by the storage rack structure that no lateral bending loads would be expected.

Radiological Consequences of a Fuel Handling Accident (FHA)

This section describes the assumptions and analyses performed to determine the potential offsite and control room radiological consequences for the postulated design basis fuel handling accident based on an Alternative Source Term (AST) in accordance with Regulatory Guide 1.183 (Reference 1). The analyses were performed such that the results are bounding for an accident occurring inside either containment or the spent fuel pool.

Input Parameters and Assumptions: The following assumptions were used in the analyses of the offsite and control room radiological consequences:

1. The reactor was assumed to have been operating at 1811 MWt prior to shutdown.

2. The reactor has been sub-critical for a minimum of 65 hours when the fuel handling accident occurs.

3. The fuel handling accident is assumed to result in damage to all of the fuel rods in the equivalent of one fuel assembly to the extent that all their gap activity is released.

4. The fission product gap inventories used are 12% for I-131, 30% for Kr-85, and 10% for all other noble gas and iodine nuclides. These values are higher than those in Table 3 to Reference 1. The gap fractions have been increased to reflect the fact that some nuclear fuel assemblies exceed the criteria of RG 1.183, Table 3, footnote 11. As a conservative approach, the gap fractions are those from RG 1.25 (Reference 6) with the value for I-131 increased by 20%, consistent with the recommendation in NUREG/CR-5009 (Reference 7).

5. The fission product inventory for the average fuel assembly at 65 hours after shutdown is provided in Table 14.2.1-1.

6. To account for differences in core power distribution across the core, the averaged fission product inventory in the dropped assembly is conservatively multiplied by a radial peaking factor of 1.7.

7. Consistent with the guidance of Reference 1, the iodine species in the pool is 99.85% elemental and 0.15% organic.

8. Consistent with the guidance of Reference 1, the effective decontamination factor (DF) used for iodine is 200 which accounts for scrubbing of the iodine as it evolves through the pool water. Applicability of this assumption is predicated on a minimum water level of 23 ft above the top of the reactor vessel flange and over the top of the assemblies in the spent fuel pool during movement of irradiated fuel assemblies. No DF is applied to the noble gas releases (i.e., no retention of the noble gases available for release) and an infinite DF is applied to the particulate radionuclides (i.e., the cesium and rubidium).

9. The activity released from the pool is assumed to be released from the containment refueling cavity or the spent fuel pool to the outside atmosphere over a two-hour period.

10. No credit is taken for removal of iodine by containment or spent fuel pool building ventilation systems' filters nor is credit taken for isolation of release paths. In addition, no credit is taken for the containment equipment hatch placement or closure nor is credit taken for having personnel air lock doors capable of closure.

11. The exclusion area boundary (EAB) and low population zone (LPZ) atmospheric dispersion factors values are found in Table 14.2.1-2.

12. The control room atmospheric dispersion factor is based on a release from the Unit 2 containment building purge stack. This release point results in a bounding analysis because the assumptions and parameters used to model the activity released due to a FHA inside either containment are identical to those for a FHA in the spent fuel pool. The control room atmospheric dispersion factor was developed using ARCON96 (Reference 5). The control room atmospheric dispersion factor value is found in Table 14.2.1-2. The meteorological data set used to develop the control room atmospheric dispersion factor was collected at the site from September 2000 to September 2005.

13. Breathing rates assumed are consistent with Reference 1 and are listed in Table 14.2.1-2.

14. The total effective dose equivalent (TEDE) doses are determined at each location. The TEDE is equivalent to the committed effective dose equivalent (CEDE) from inhalation and the deep dose equivalent (DDE) from external exposure. Effective dose equivalent (EDE) is used in lieu of DDE in determining the contribution of external dose to the TEDE consistent with Reference 1. The dose conversion factors (DCFs) used in determining the CEDE dose are from the EPA Federal Guidance Report No. 11 (Reference 2). The dose conversion factors used in determining the EDE dose are from the EPA Federal Guidance Report No. 12 (Reference 3).

15. The site-boundary (also called the exclusion area boundary (EAB)) dose is calculated for the worst two-hour period and the low population zone (LPZ) dose is calculated for the release duration, that is two-hours for FHA. The control room personnel dose is calculated for 30 days.

16. The control room HVAC system is assumed to be initially operating in normal MODE, whereby fresh air is being brought into the control room unfiltered at a rate of 2000 cfm. It is conservatively assumed that the emergency HVAC MODE is entered 10 minutes after event initiation based on the area monitor inside the control room reaching its alarm set-point. The emergency HVAC MODE is assumed to provide 2500 cfm of filtered outside air with 1955 cfm filtered recirculation.

17. Parameters used in the control room personnel dose calculations are provided in Table 14.2.1-2. These parameters include the normal operation flow rates, the post-accident operation flow rates, unfiltered inleakage rate, control room volume, filter efficiencies, and the control room operator breathing rates.

Acceptance Criteria

The EAB and LPZ dose Standard Review Plan (SRP) 15.0.1 (Reference 4) acceptance criteria for a fuel handling accident is 6.3 rem TEDE, which is approximately 25% of the 10 CFR 50.67 limit of 25 rem. The control room personnel dose acceptance criterion is 5 rem TEDE per 10 CFR 50.67.

Results and Conclusions

The offsite and control room personnel doses due to a design basis FHA are presented below. These doses are within the acceptance criteria of SRP 15.0.1 and the dose limits of 10 CFR 50.67.

Location	Acceptance Criteria (rem)	TEDE (rem)
Exclusion Area Boundary	6.3	2.7
Low Population Zone	6.3	0.2
Control Room	5	4.3

14.2.1.1   References:

1. USNRC, Regulatory Guide 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," July 2000.

2. US EPA, "Limiting values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation Submersion, and Ingestion," Federal Guidance Report No. 11, September 1988.

3. US EPA, "External Exposure to Radionuclides in Air, Water, and Soil," Federal Guidance Report No. 12, September 1993.

4. Standard Review Plan (SRP) Section 15.0.1, "Radiological Consequence Analyses Using Alternative Source Terms," July 2000.

5. J. V. Ramsdell, Jr. and C. A. Simonen, "Atmospheric Relative Concentrations in Building Wakes," NUREG/CR-6331, Revision 1, May 1997.

6. USNRC, Regulatory Guide 1.25, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Fuel Handling Accident in the Fuel Handling and Storage Facility for Boiling and Pressurized Water Reactors," March 1972.

7. NUREG/CR-5009, "Assessment of the Use of Extended Burnup Fuel in Light Water Power Reactors," February 1988.

8. Calculation CN-CRA-08-14, "EC 12732 Point Beach - Fuel Handling Accident Doses for the EPU," dated May 11, 2011.

Table 14.2.1-1   ACTIVITY IN AN AVERAGE FUEL ASSEMBLY AT 65 HOURS POST SHUTDOWN

Nuclide	Activity (Ci)
I-130	2.29E+02
I-131	3.43E+05
I-132	3.50E+05
I-133	1.03E+05
I-135	8.76E+02
Kr-85m	4.89E+00
Kr-85	5.07E+03
Kr-87	9.34E-11
Kr-88	3.84E-02
Xe-131m	4.52E+03
Xe-133m	1.65E+04
Xe-133	6.95E+05
Xe-135m	1.43E+02
Xe-135	1.45E+04

Note: Neither the gap fractions nor the radial peaking factor have been applied to these values.

Table 14.2.1-2   ASSUMPTIONS USED FOR THE FHA DOSE ANALYSIS

Parameter	Value
Core Power Level (1800 MWt x 1.006)	1811 MWt
Radial Peaking Factor	1.7
Number of Damaged Assemblies	1 assembly
Fission Product Decay Period	65 hr
Gap Fractions	
I-131	12% of activity
Kr-85	30% of activity
Other Iodine and Noble Gas	10% of activity
Water Level (minimum for reactor cavity or pool)	23 ft [a]
Overall Pool Iodine Decontamination Factor	200
Noble Gas Decontamination Factor	1
Particulate Decontamination Factor	Infinite
Filter Efficiency	No filtration
Isolation of Release	No isolation
Atmospheric Dispersion Factors (χ/Q)	
Exclusion Boundary Area	$5.0E-04 \ sec/m^3$
Low Population Zone	$3.0E-05 \ sec/m^3$
Control Room, Limiting Case – Unit 2 Purge Stack	$6.94E-03 \ sec/m^3$
Breathing Rate	$3.5E-04 \ m^3/sec$
Control Room HVAC Parameters	
Normal MODE Ventilation Flow Rates	
Filtered Makeup Flow Rate	0 cfm
Filtered Recirculation Flow Rate	0 cfm
Unfiltered Makeup Flow Rate	2000 cfm
Unfiltered Inleakage Flow Rate	300 cfm
Emergency MODE Ventilation Flow Rates	
Filtered Makeup Flow Rate	2500 cfm
Filtered Recirculation Flow Rate	1955 cfm
Unfiltered Makeup Flow Rate	0 cfm
Unfiltered Inleakage Flow Rate	300 cfm
Filter Efficiencies	
Elemental	95%
Organic	95%
Particulate	99%
Control Room Isolation Actuation Signal/Timing	
Area Monitor High Set-point	2 mrem/hr
Timing of High Radiation Signal	<10 min
Occupancy Factors	
0-24 hours	1.0
1 – 4 days	0.6
4 – 30 days	0.4

a. Measured from reactor vessel flange for reactor cavity or top of fuel assemblies for spent fuel pool.

14.2.2   <u>ACCIDENTAL RELEASE-RECYCLE OR WASTE LIQUID</u>

Section 14.2.2 relocated to Section 11.1.5

14.2.3 ACCIDENTAL RELEASE-WASTE GAS

Section 14.2.3 relocated to Section 11.2.5

14.2.4   STEAM GENERATOR TUBE RUPTURE

General

A complete single tube break adjacent to the tube sheet in a steam generator is examined. Since the reactor coolant pressure is greater than the steam generator shell side pressure, the contaminated reactor coolant discharges into the secondary system.

The activity release is limited by operator action to limit the primary to secondary fluid leakage and terminate the releases from the affected steam generator to the atmosphere.

Steam Release Analysis

A mass and energy balance is used to calculate the primary-to-secondary break flow and steam generator steam releases resulting from a steam generator tube rupture accident. The analysis provides conservatively high mass transfers for use in the radiological consequences analysis. The analysis incorporates and supports the following assumptions:

*   Core power of 1800 MWt.
*   Vessel average temperature range between 558°F to 577°F.
*   0-percent or 10-percent steam generator tube plugging levels.
*   Steam generator models 44F or Delta 47 in use at Unit 1 and Unit 2, respectively.
*   The ruptured steam generator pressure is maintained at the lowest steam generator safety valve re-seat pressure of 930 psia (which includes 12.6-percent blowdown and 3-percent uncertainty)
*   Maximum safety injection flow rates result in an equilibrium break flow rate of approximately 54 lbm/sec.
*   Consistent with the vintage of Point Beach Units 1 and 2, no single failures are modeled in the analysis.

For the purpose of this analysis, it is assumed that when reactor trip occurs station normal power is lost. The reactor coolant pumps will then coast down and the condenser circulating water pumps will stop. On-site emergency power is available from the diesel generators to supply the necessary engineered safeguards equipment.

Core decay heat is then removed by natural circulation of reactor coolant to the steam generators. The atmospheric steam relief valves will open automatically to relieve high pressure in the steam generators. Steam dump to the condenser is isolated when condenser vacuum is lost. During this time, secondary safety valves may also lift.

Main steam safety valves open to restore primary system temperature to the hot shutdown value. They are designed to blowdown to 12.6% below the setpoint pressure to remove decay heat while maintaining the hot shutdown (hot standby per Technical Specification definitions) system pressure. With no operator action, the main steam safety valves would maintain the primary system temperature between approximately 535 and 557°F.

The safety injection system borates the reactor coolant system within several minutes and will eventually refill the reactor coolant system and pressurize it to a pressure at which the injection flow is balanced by discharge through the broken tube. Initially, the water level in the unaffected

steam generator will decrease because the auxiliary feedwater supply will not match the steam relief needed to reduce the reactor coolant system to no-load temperature. When the steam dump is reduced to balance decay heat, the auxiliary feedwater supply exceeds decay heat requirements and the liquid level in the unaffected steam generator will increase. Because of the discharge from the reactor coolant system, the rate of increase in liquid level is greatest in the ruptured steam generator.

Up to this point, automatic actions will ensure safe shutdown of the reactor. Automatic actuation of safety injection will ensure that the core will not be damaged, and thus limit radioactivity releases to the level of the concentrations in the reactor coolant.

After the initial transient, the operator would isolate the affected steam generator, and perform a limited cooldown to assure subcooling margin. The safety injection system will maintain reactor coolant system pressure and pressurizer level, compensating for losses due to discharge in reaching pressure equilibrium between the reactor coolant system and the now isolated ruptured steam generator and for contraction losses during the remainder of cooldown. After cooldown, RCS depressurization would be performed to restore reactor coolant inventory, and subsequently the safety injection flow would be terminated to stop the primary-to-secondary break flow.

The analysis assumes that break flow to the ruptured steam generator is terminated 30 minutes following accident initiation. Steam releases from the ruptured steam generator are terminated when primary and secondary pressures are equalized at 30 minutes. Steam releases from the intact steam generator continue until the residual heat removal system takes over decay heat removal.

A fraction of the break flow flashes directly into steam upon entering the secondary side of the ruptured steam generator. The pre- and post-trip flashing fractions are conservatively calculated assuming the break flow is at the hot leg temperature. The flashing fractions are calculated to be 0.22 prior to reactor trip and 0.13 following reactor trip.

After the primary-to-secondary break flow has been terminated, the RCS would be cooled down to cold shutdown conditions. The cooldown is initiated by manually controlling the steam relief on the unaffected steam generator. During the cooldown, no further activity is discharged from the isolated steam generator.

The above assumptions lead to a conservative upper bound of 124,500 pounds for the total amount of reactor coolant transferred to the ruptured steam generator including 18,110 pounds of flashed primary coolant transferred to the ruptured steam generator. The assumptions also result in the release of 88,100 pounds of steam from the ruptured steam generator.

Because the licensing basis analysis described above is quite conservative it does not require operators to demonstrate the ability to terminate break flow within 30 minutes from the start of the event. It is recognized that the operators may not be able to terminate break flow within 30 minutes for all postulated steam generator tube rupture events. The operator actions applicable to the SGTR dose and margin-to-overfill analyses are specified below under the section for operator actions.

Margin to Overfill (MTO) Analysis)

Demonstration that the ruptured steam generator does not overfill during the accident has been performed by utilizing an NRC-approved thermal hydraulic analysis code. Reference 1 includes the NRC's approval of the LOFTTR2 computer code that has been used for the overfill analysis. This code simulates the plant response, and models specific operator actions. Thus, a more realistic representation of the break flow during the accident is obtained. The auxiliary feedwater flow is assumed to be maximum with a minimum delay time after reaching the low-low SG water level. Critical operator actions included in the LOFTTR2 simulation include: isolation of auxiliary feedwater flow to the ruptured steam generator (based on level indications), isolation of the ruptured steam generator, cooldown of the RCS by dumping steam from the intact steam generator, depressurizing the RCS, and terminating safety injection flow to terminate break flow.

Consistent with the vintage of Point Beach Units 1 and 2 and the licensing basis analysis, no single failures are modeled in the margin to overfill analysis. Instead, the following items from the unaffected unit are credited to mitigate the tube rupture event: (1) control-grade instrument air (IA), which is supplied from a shared system; and (2) in the event of a dual-unit LOOP, operator actions by the crew of the shared unit to restore power supply to the shared IA system. The ADV operator and the IA system are assumed to operate despite being non-safety related because both of the ADVs are required by Technical Specifications to be operable, and at least one IA compressor can be powered by an EDG, which would assure, with diversity, that IA is available to operate the ADV.

With the exception of the single failure assumption, the analysis is performed following the guidance in Reference 1. Conservative deviations from the Reference 1 method were taken to address the issues raised by NSAL-07-11 (Reference 2). The analysis demonstrates that following the complete severance of a steam generator tube break flow is terminated approximately 44 minutes after initiation of the tube rupture and that overfill of the ruptured steam generator does not occur.

Operator Actions for SGTR Dose and MTO Analyses

The operator actions used in the dose and MTO analyses are the following as documented in Reference 5:

- Isolate the ruptured SG within 6 minutes.
- Initiate RCS cooldown within 17 minutes after the ruptured SG is isolated.
- Initiate RCS depressurization within 3 minutes following the completion of cooldown.
- Secure ECCS within 2 minutes following the completion of depressurization.

Although not a direct operator action, the limiting SGTR analysis demonstrates that the SG tube break flow is terminated approximately 44 minutes after initiation of the tube rupture by crediting the above operator actions.

Radiological Consequences of a Steam Generator Tube Rupture Accident

The analysis of the SGTR radiological consequences uses the analytical methods and assumptions outlined in the RG 1.183 (Reference 3).

The quantity of radioactivity released to the environment due to a SGTR depends upon primary and secondary coolant activity, iodine spiking effects, primary-to-secondary break flow, break flow flashing, attenuation of activity carried by the flashed portion of the break flow, partitioning of iodine between the liquid and steam phases, moisture carryover, the mass of fluid released from the generators 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

The concentrations of iodines and noble gasses in the RCS at the time the accident occurs are based on 520 µCi/gm of DE Xe-133 and the Technical Specification limit of 0.5 µCi/gm of dose equivalent (DE) I-131. The alkali metal concentration in the RCS is based on the fuel defect level that corresponds to 0.5 µCi/gm DE I-131. The iodine activity concentration of the secondary coolant at the time the accident occurs is assumed to be equivalent to the Technical Specification limit of 0.1 µCi/gm of DE I-131. The alkali metal activity concentration of the secondary coolant at the time the accident occurs is assumed to correspond to 0.1 µCi/gm of DE I-131. The equilibrium nuclide concentrations are presented in Table 14.1.8-4. In addition, two iodine spikes are considered.

Pre-accident Spike - A reactor transient has occurred prior to the event and has raised the primary coolant iodine concentration to a conservative value of 60 µCi/gm DE I-131.

Accident Initiated Spike - The primary coolant iodine concentration is initially at the Technical Specification limit of 0.5 µCi/gm DE I-131. Following the primary system depressurization and reactor trip associated with the event, an iodine spike is initiated in the primary system. The spike increases the iodine appearance 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 duration of the spike is 8 hours

Offsite power is assumed to be lost at reactor trip. Prior to reactor trip, activity is released through the condenser air ejector exhaust and a partition factor of 0.01 for iodines and alkali metals is assumed for this release path. Although the air ejector exhausts through the auxiliary building vent stack to the environment, the atmospheric dispersion factors associated with the Unit 2 safety valves is used to determine the concentration of this release path at the control room intake. After reactor trip and loss of offsite power, flow to the condenser is isolated.

An iodine partition factor of 0.01 (curies iodine/gm steam) / (curies iodine/gm water) and a particulate retention factor of 0.0025 are applied to both SGs based on full power moisture carryover.

The iodine and alkali metal transport model used in this analysis accounts for break flow flashing, steaming and partitioning. The model assumes that a fraction of the activity carried by the break flow becomes airborne immediately due to flashing and atomization. All of the iodine and alkali metal in the flashed break flow is assumed to be transferred out of the steam generator. Droplet removal by the dryers is conservatively neglected. The time dependent iodine and alkali metal removal efficiency for scrubbing of steam bubbles as they rise from the rupture site to the water surface was not calculated and was conservatively neglected. The fraction of primary coolant iodine that 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.

Since there is no penalty taken for tube uncovery and scrubbing is not credited, the assumed location of the tube rupture is not significant for the radiological analysis. The thermal and hydraulic analysis has conservatively addressed the issue of the location of the tube rupture in the calculations of break flow rate and flashing fraction.

All noble gases in the break flow and primary-to-secondary leakage are assumed to be transferred instantly out of the steam generator to the atmosphere.

The integrated tube rupture break flow, flashed break flow, and integrated atmospheric steam releases are summarized in Table 14.2.4-2 for the different time intervals considered in the analysis. The time intervals considered are: from event initiation until reactor trip, reactor trip to 30 minutes, 30 minutes to 2 hours, 2 hours to 8 hours, 8 hours to 24 hours, and 24 hours to 30 hours. The plant cooldown to RHR operating conditions is assumed to be accomplished within 30 hours after initiation of the SGTR and steam releases are terminated at this time.

A total primary-to-secondary leak rate of 2000 gm/min is assumed to exist prior to the SGTR. The leak is assumed to be distributed with 1000 gm/min to the intact steam generator and 1000 gm/min to the ruptured steam generator. The leakage to the intact steam generator is assumed to persist for the duration of the accident.

Dose conversion factors, offsite atmospheric dispersion factors and breathing rates are provided in Table 14.1.8-3.

The control room HVAC begins in normal mode. Actuation of the emergency mode is conservatively assumed to occur when the SI/containment isolation actuation setpoint is reached at 220 seconds. Control room models are provided in Table 14.1.8-6.

Acceptance Criteria

The standard Review Plan (SRP) 15.0.1 (Reference 4) offsite dose acceptance criterion for a SGTR with pre-accident iodine spike is the 10 CFR 50.67 limit of 25 rem TEDE and the acceptance criterion for a SGTR with an accident initiated iodine spike is 2.5 rem TEDE, which is 10% of the 10 CFR 50.67 limit of 25 rem TEDE. The control room personnel dose acceptance criterion is 5 rem TEDE per 10 CFR 50.67.

Results and Conclusions

The results of the offsite and control room dose analyses are provided in Table 14.2.4-1, and indicate that the acceptance criteria are met. The exclusion area boundary doses reported are for the worst 2 hour period, determined to be from 0 to 2 hours.

Multiple Tube Ruptures

A much larger dose, e.g., TEDE dose of 25 rem at the exclusion radius, can only result from the rupture of sufficient steam generator tubes to cause fuel cladding failure.

Operating experience with steam generators of the type used in this plant has not shown significant numbers of single gross and immediate tube failures. Small leaks in a single tube which caused erosion type damage to adjacent tubes have been reported, but did not cause a

rupture of the adjacent tubes. Thus, if a single tube failure were postulated, it is probable that adjacent tubes would not be damaged but any adjacent failure would be an erosion-caused leak rather than a sudden gross failure.

To perform a rigorous analysis of the flow dynamics of blowdown through multiple tube ruptures, one must understand and define mathematically the physical configuration of the ruptures. Because no reasonable mechanism exists for the multiple ruptures, it is instead just as meaningful to analyze the consequences of a pipe rupture, equivalent in terms of discharge rate to various multiples of the single tube discharge rate.

Such an analysis reveals that the core cooling system will prevent clad damage for break discharge rates equal to or smaller than that resulting from a broken pipe between 4 inches and 6 inches in diameter. The discharge rates which bracket the onset of cladding damage correspond to 18 and 40 times the discharge from a single severed steam generator tube. Actually, the ratio would be much larger owing to the fact that the discharge from a tube failure will be limited by the back pressure in the steam generator. Ultimately, the tube discharge would terminate when the reactor coolant system and the steam generator reached pressure equilibrium. The operator can initiate cooldown through the unaffected steam generator.

These conclusions are based on single-failure mode performances of the core cooling system. The core does not become uncovered by the calculated quiet level in those cases where cladding damage is found to be prevented.

The incredibility of multiple simultaneous tube failures is supported by the following reasoning:

1. At the maximum operating internal pressure the tube wall sees only about 1530 psi compared with a calculated bursting pressure in excess of 11,100 psi based on ultimate strength at design temperature.

2. The above margin applies to the longitudinal failure modes, induced by hoop stress. This failure mode is the least likely to cause propagation of failure tube-to-tube. An additional factor of two applies to ultimate pressure strength in the axial direction tending to resist double-ended failure (total factor of 14.6).

3. Failures induced by fretting, corrosion, erosion, or fatigue are of such a nature as to produce tell-tale leakage in substantial quantity while ample metal remains to prevent severance of the tube (a small fraction of the original tube wall section) as indicated by the margin derived in 2 above. Thus, any incipient failures that would develop to the point of severe leakage requiring a shutdown for plugging or repair, in accordance with Technical Specifications, would happen long before the large safety margin in pressure strength is lost.

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10. CN-CRA-08-35, Rev 1, "Point Beach Units 1 & 2 Steam Generator Tube Rupture Doses for the Extended Power Uprate."

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Table 14.2.4-1   STEAM GENERATOR TUBE RUPTURE ACCIDENT DOSES

A. <u>With Pre-Accident Iodine Spike</u>

0 - 2 hr Dose at Site Boundary	0 - 30 hr Dose at LPZ	0 - 30 day Dose in CR
2.0 rem TEDE	0.2 rem TEDE	1.9 rem TEDE

B. <u>With Accident-Initiated Iodine Spike</u>

0 - 2 hr Dose at Site Boundary	0 - 30 hr Dose at LPZ	0-30 day Dose in CR
0.6 rem TEDE	0.1 rem TEDE	0.5 rem TEDE

Table 14.2.4-2   MASS TRANSFER USED FOR SGTR DOSE ANALYSES

Ruptured Steam Generator		
Pre-trip Break Flow	21,300 lbm	(0-220 sec)
Post-trip Break Flow	103,200 lbm	(0-220 sec)-30 min)
Pre-trip Flashed Break flow	4,690 lbm	(0-220 sec)
Post-trip Break Flow	13,420 lbm	(0-220 sec-30 min)
Steam Release	1,130 lbm/sec	(0-220 sec)
	88,100 lbm	(0-220 sec-30 min)
Intact Steam Generator		
Primary-to-Secondary Leakage	1000 gm/min	
Steam Release	1,130 lbm/sec	(0-220 sec)
	257,700 lbm	(0-220 sec-2hr)
	584,000lbm	(2-8 hr)
	866,000 lbm	(8-24 hr)
	54,100 lbm/hr	(>24 hr)

14.2.5 RUPTURE OF A STEAM PIPE

A.   Core Power and Reactor Coolant System Transient

A rupture of a steam pipe is assumed to include any accident which results in an uncontrolled
steam release from a steam generator. The release can occur due to a break in a pipe line or due to
a valve malfunction. The steam release results in an initial increase in steam flow which decreases
during the accident as the steam pressure falls. The energy removal from the Reactor Coolant
System causes a reduction of coolant temperature and pressure. With a negative moderator
temperature coefficient, the cool down results in a reduction of core shutdown margin. If the most
reactive control rod is assumed stuck in its fully withdrawn position, there is a possibility that the
core will become critical and return to power even with the remaining control rods inserted. A
return to power following a steam pipe rupture is a potential problem only because of the high hot
channel factors which may exist when the most reactive rod is assumed stuck in its fully
withdrawn position. Assuming the most pessimistic combination of circumstances which could
lead to power generation following a steam line break, the core is ultimately shut down by the
boric acid in the Safety Injection System.

The analysis of a steam pipe rupture is performed to demonstrate that with a stuck rod and
minimum engineered safety features, the core remains in place and essentially intact so as not to
impair effective cooling of the core.

Although DNB and possible cladding perforation (no cladding melting or zirconium-water
reaction) following a steam pipe rupture are not necessarily unacceptable, the following analysis,
in fact, shows that the DNB design basis is met for any rupture assuming the most reactive rod
stuck in its fully withdrawn position. The following functions provide the necessary protection
against a steam pipe rupture:

1.   Safety Injection System actuation on:

      a.   Two out of three pressurizer low pressure signals.

      b.   Two out of three low pressure signals in any steam line.

      c.   Two out of three high containment pressure signals.

2.   The overpower reactor trips (neutron flux and ΔT) and the reactor trip occurring upon
      actuation of the Safety Injection System.

3.   Redundant isolation of the main feedwater lines. Sustained high feedwater flow would
      cause additional cooldown, thus, in addition to the normal control action which will close
      the main feedwater valves, any safety injection signal will rapidly close all feedwater
      control valves and the feedwater isolation valves.

4.   Closure of the fast acting steam line isolation valves (designed to close in less than
      5 seconds upon receipt of a CLOSE signal) on:

      a.   One out of the two high steam flow signals in that steam line in coincidence with any
           safety injection signal. (Dual set points are provided, with the lower set point used in
           coincidence with two out of four indications of low reactor coolant average
           temperature.)

      b.   Two out of three high - high containment pressure signals.

Each steam line has a fast closing isolation valve and a check valve. These four valves prevent blowdown of more than one steam generator for any break location even if one valve fails to close. For example, for a break upstream of the isolation valve in one line, closure of either the check valve in that line or the isolation valve in the other line will prevent blowdown of the other steam generator.

Steam flow is measured by monitoring dynamic head in nozzles inside the steam pipes. The nozzles (16 in. I.D. vs. a pipe diameter of 28 in. I.D.) are located inside the containment near the steam generator. The Unit 1 and Unit 2 steam generators contain a steam nozzle flow limiting device which is designed to limit the steam generator depressurization rate by restricting the steam flow during any postulated steam line break accident.

Method of Analysis (Reference 11 and Reference 23)

The analysis of the steam pipe rupture has been performed to determine:

1. The core heat flux and reactor coolant system temperature and pressure resulting from the cooldown following the steam line break. The RETRAN code (Reference 12) has been used.

2. The conservatism of the core reactivity feedback model used in (1) above was confirmed with a detailed core analysis using the ANC code (Reference 13). ANC also calculates the core peaking factors used in the DNB analysis and the maximum fuel linear heat generation rate (kW/ft) to confirm that no fuel centerline melting is predicted for the steam line break transient.

3. The thermal and hydraulic behavior of the core following a steam line break. A detailed thermal and hydraulic digital computer code, VIPRE (Reference 14), has been used to determine if DNB occurs for the core conditions computed in (1) and (2) above.

4. The offsite consequences of the steam line break accident which include consideration of the additional secondary loop activity resulting from a steam generator tube leak prior to the accident.

5. The onsite consequences (e.g., control room habitability). These analyses are described in general terms in this section.

The following assumptions are made:

1. A 2.0% shutdown reactivity from the rods at no load conditions. This is the end of life design value including design margins with the most reactive rod stuck in its fully withdrawn position. Operation of the RCCA banks is restricted in accordance with the Technical Specifications such that the main steam line break analysis remains bounding.

2. The negative moderator temperature coefficient corresponding to the end of life core with all but the most reactive rod inserted. The variation of the coefficient with temperature and pressure has been included. In computing the power generation following a steam line break, the local reactivity feedback from the high neutron flux in the region of the core near the stuck control rod has been included in the overall reactivity balance. The local reactivity

feedback is composed of Doppler reactivity from the high fuel temperatures near the stuck control rod and moderator feedback from the high water enthalpy near the stuck rod.   For the cases analyzed where steam generation occurs in the high flux regions of the core the effect of void formation on the reactivity has been included. The effect of power generation in the core on overall reactivity is a function of the core temperature, pressure, and flow and thus is different for each case studied. The analysis assumes end of life core conditions with all rods in except the most reactive rod which is assumed stuck in its fully withdrawn position.

3.   Minimum capability for injection of boric acid solution corresponding to the most restrictive single failure in the safety injection system. The emergency core cooling system consists of three systems: 1) the passive accumulators, 2) the low head safety injection (residual heat removal) system, and 3) the high head safety injection system. Both the accumulators and the high head safety injection are modeled for the steam line break accident analysis. The boric acid solution of the high head safety injection is 2700 ppm and no credit is taken for boron in the accumulators.

The modeling of the safety injection system in RETRAN is described in Reference 12. The flow corresponds to that delivered by one safety injection pump delivering its full flow to both RCS cold legs. The accumulators are modeled to begin injection when the cold leg pressure drops to 694.7 psia.

For cases where offsite power is available, the sequence of events in the safety injection system is the following: After the generation of the safety injection signal (appropriate delays for instrumentation, logic, and signal transport included), the appropriate valves begin to operate and the high head safety injection pump starts. Ten seconds later, the valves are assumed to be in their final position and the pump is assumed to be at full speed. When the RCS pressure falls below the SI pump shutoff pressure net injection flow begins and the volume containing unborated water is swept into the core before the borated water reaches the core. This delay, described above, is included in the modeling.

In cases where offsite power is not available, maximum delay times are considered to account for SI signal processing (2 seconds), sequencer plus uncertainty (1 second), diesel generator start to full speed (15 seconds), and SI pump start to full speed (10 seconds), for a total delay of 28 seconds assumed in the analysis.

4.   In computing the steam flow during a steam line rupture, the Moody Curve (Reference 15) for f(L/D) = 0 is used.

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

6.   Since both the Unit 1 and Unit 2 steam generators are equipped with integral flow restrictors with a 1.388 ft^2 throat area, any rupture with a break greater than this size, regardless of the location, would have the same effect on the reactor as a 1.388 ft^2 break. The following two cases have been considered in determining the core power and RCS transients for each unit.

    a.   Complete severance of a pipe with the plant initially at no-load conditions, with offsite power available. Full reactor coolant flow is maintained.

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

    The cases above assume initial hot shutdown conditions with the rods inserted (except for one stuck rod) at time zero. Should the reactor be just critical or operating at power at the time of a steam line break, the reactor will be tripped by the normal overpower protection system when the power level reaches a trip point.

    Following a trip at power, the reactor coolant system contains more stored energy than at no load, the average coolant temperature is higher than at no load and there is appreciable energy stored in the fuel. Thus, the additional stored energy is removed via the cooldown caused by the steam line break before the no load conditions of reactor coolant system temperature and shutdown margin assumed in the analyses are reached. After the additional stored energy has been removed, the cooldown and reactivity insertions proceed in the same manner as in the analyses which assume no load conditions at time zero.

7.   Perfect moisture separation in the steam generator is assumed. This assumption leads to conservative results since considerable water would be expected to be discharged from the steam generator. Water entrainment in the steam reduces the steam generator inventory, thereby reducing the magnitude of the temperature decrease (cooldown) in the core.

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

9.   All main and auxiliary feedwater pumps are assumed to be operating at full capacity when the rupture occurs. This assumption maximizes the cooldown. The main feedwater temperature at no-load conditions is assumed to be 35°F. A conservatively high auxiliary feedwater flow rate of 1200 gpm at a minimum temperature of 35°F is assumed to be delivered to the affected steam generator. Main feedwater is isolated following the SI signal; however, auxiliary feedwater continues for the duration of the transient.

10.   The effect of heat transferred from thick metal in the RCS and the steam generators is not included in the cases analyzed. The heat transferred from these sources would be a net benefit since it would slow the cooldown of the RCS.

Results

The results presented are a conservative indication of the events which would occur assuming a steam line rupture. The worst case assumes that all of the following occur simultaneously.

1.   Minimum shutdown reactivity margin.

2.   The most negative moderator temperature coefficient for the rodded core at end of life.

3.   The rod having the most reactivity stuck in its fully withdrawn position.

4.   One safety injection pump fails to function as designed.

Rupture of a Steam Line at Zero Power Analysis

As described above, two cases were analyzed for each unit from zero power initial conditions. A time sequence of events for all cases analyzed is provided in Table 14.2.5-2. The peak heat flux and time of occurrence are also shown on the table for each case analyzed (Reference 11).

The limiting steam line rupture for each unit is the case in which offsite power is assumed to be available. The transient plots in Figure 14.2.5-1 show the limiting Unit 1 plant response following a main steam pipe rupture from zero power initial conditions with offsite power available. Figure 14.2.5-2 shows the plant response for the Unit 1 case with offsite power not available. Loss of offsite power results in a coastdown of the reactor coolant pumps and reduced core flow. This causes the core power to increase at a slower rate and reach a lower peak value. The Unit 2 transient plots are very similar to Unit 1, and thus are not presented.

The results of the major rupture of a main steam pipe event analysis confirm that the DNB and fuel centerline melt design bases are met for both units. The calculated minimum DNBR is above the applicable limit value of 1.45 (the W-3 DNB correlation limit with pressure less than 1000 psia). The calculated peak linear heat generation rate is less than the limit value of 22.54 kW/ft corresponding to fuel centerline melting. Primary and secondary pressure limits are not challenged because primary and secondary pressures decrease from their initial values during the transient. Therefore, this event does not adversely affect the core or the RCS, and all applicable acceptance criteria are met.

Rupture of a Steam Line at Full Power Analysis

To ensure safe shutdown during MODE 1 operation, the steam line rupture event was analyzed at hot full power conditions. For this analysis, initial conditions of core power and pressurizer pressure were assumed to be at their nominal values consistent with steady-state full power operation. RCS coolant temperature was assumed to be at its nominal, steady-state, full-power value plus a small temperature bias. Uncertainties in the initial conditions of these parameters are considered in the DNBR limit rather than explicitly modeled in the transient calculations, consistent with the application of the Revised Thermal Design Procedure (RTDP) methodology. Steam generator water level was assumed to be at its nominal value. Minimum measured reactor coolant flow was modeled according to the RTDP methodology. Zero steam generator tube plugging was assumed to maximize the primary-to-secondary heat transfer, which results in a more severe RCS cooldown transient.

For breaks outside containment, the overpower ΔT and Low Steam Line Pressure – Safety Injection protection functions are relied upon to provide the necessary protection to mitigate the event. For breaks inside containment, protection is provided by the Hi-1 Containment Pressure – Safety Injection function. The results of separate containment pressure response analyses showed that the Hi-1 Containment Pressure – Safety Injection signal would be reached before overpower ΔT on all inside containment break cases. A delayed reactor trip for this event

results in more limiting transient results. Based on this, the outside containment breaks, which rely on overpower ΔT and Low Steam Line Pressure – Safety Injection, are determined to be the most limiting scenario; therefore, it is this scenario that is explicitly modeled.

The most limiting full power case is typically the largest break that produces a reactor trip on overpower ΔT. Larger breaks result in a rapid reactor trip as a result of the Low Steam Line Pressure – Safety Injection signal, before core power increases significantly, and are therefore less limiting. Since PBNP has steam exit nozzle flow restrictors which limit the flow area to about 1.388 ft^2, the analysis modeled a spectrum of break sizes up to 1.4 ft^2. The analysis demonstrates that the most limiting break size is 0.59 ft^2 (Unit 1) and 0.63 ft^2 (Unit 2); reactor trip for both cases is on overpower ΔT.

The results of the full-power steam line rupture analysis demonstrate that the DNB design basis is met. In addition, the peak linear heat generation rate (expressed in kW/ft) does not exceed the fuel centerline melt limit. Since this event results in a decrease in both the primary and secondary side pressures, the maximum RCS and Main Steam System pressure criteria are not challenged.

B.   Radiological Consequences (Reference 24)

The complete severance of a main steamline outside containment is assumed to occur. The affected SG will rapidly depressurize and release to the outside atmosphere the activity initially contained in the secondary coolant and the activity transferred from the primary coolant through SG tube leakage. A portion of the activity initially contained in the intact SG and a portion of the activity due to tube leakage is released to the atmosphere through either the atmospheric dump valves or the main steam safety valves. This section describes the assumptions and analyses performed to determine the amount of radioactivity released and the doses resulting from the release.

The analysis of the main steamline break radiological consequences uses the analytical methods and assumptions outlined in the RG 1.183 (Reference 9).

The concentrations of iodines and noble gasses in the RCS at the time the accident occurs are based on 520 μCi/gm of DE Xe-133 and the Technical Specification limit of 0.5 μCi/gm of dose equivalent (DE) I-131. The alkali metal concentration in the RCS is based on the fuel defect level that corresponds to 0.5 μCi/gm DE I-131. The iodine activity concentration of the secondary coolant at the time the accident occurs is assumed to be equivalent to the Technical Specification limit of 0.1 μCi/gm of DE I-131. The alkali metal activity concentration of the secondary coolant at the time the accident occurs is assumed to correspond to 0.1 μCi/gm of DEI-131. The equilibrium nuclide concentrations are presented in Table 14.1.8-4. In addition, two iodine spikes are considered.

Pre-accident Spike - A reactor transient has occurred prior to the event and has raised the primary coolant iodine concentration to a conservative value of 60 μCi/gm DE I-131.

Accident-Initiated Spike - The primary coolant iodine concentration is initially at the Technical Specification limit of 0.5 μCi/gm DE I-131. Following the primary system depressurization and reactor trip associated with the event, an iodine spike is initiated in the primary system. The spike increases the iodine appearance rate from the fuel to the coolant to a value 500 times greater than the release rate corresponding to the initial primary system iodine concentration. The duration of the spike is 4 hours.

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

A total primary-to-secondary leak rate of 2000 gm/min is assumed to exist prior to the steam line rupture. The leak is assumed to be distributed with 1000 gm/min to the intact steam generator and 1000 gm/min to the ruptured steam generator. The leakage is assumed to persist for the duration of the accident.

An iodine partition factor of 0.01 (curies iodine/gm steam) / (curies iodine/gm water) and a particulate retention factor of 0.0025 based on full power moisture carryover are applied to the intact SG.

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

The plant cooldown to RHR operating conditions is assumed to be accomplished within 30 hours after initiation of the event and steam releases from the intact steam generator are terminated at this time. Within 60 hours after the accident the reactor coolant system has been cooled to below 212°F and there are no further steam releases to the atmosphere from the faulted steam generator.

Dose conversion factors, offsite atmospheric dispersion factors and breathing rates are provided in Table 14.1.8-3.

The control room HVAC begins in normal mode. In the event of a steamline break, the low steam line pressure SI setpoint will be reached shortly after event initiation. The SI/containment isolation signal or a radiation monitor signal cause the control room HVAC to switch from the normal operation mode to the post-accident mode of operation. The analysis conservatively did not credit the SI signal but relied on the ventilation system line radiation monitor signal for control room isolation. It was confirmed that the radiation monitor setpoint is reached within 15 seconds. The control room HVAC switches from normal operation to post-accident mode of operation at 75 seconds (15 seconds for radiation signal plus 60 second delay time). Control room models are provided in Table 14.1.8-6.

Acceptance Criteria

The Standard Review Plan (SRP) 15.0.1 (Reference 10) offsite dose acceptance criterion for a steamline break with a pre-accident iodine spike is the 10 CFR 50.67 limit of 25 rem TEDE and the acceptance criterion for a steamline break with an accident initiated iodine spike is 2.5 rem TEDE, which is 10% of the 10 CFR 50.67 limit of 25 rem TEDE. The control room personnel dose acceptance criterion is 5 rem TEDE per 10 CFR 50.67.

Results and Conclusions

The results of the offsite and control room dose analyses are provided in Table 14.2.5-1, and indicate that the acceptance criteria are met. The exclusion area boundary doses reported are for the worst 2 hour period, determined to be from 0 to 2 hours for the pre-accident iodine spike and from 3.9 to 5.9 hours for the accident initiated iodine spike.

C.   Containment Response Analysis (Reference 22)

An analysis is performed to predict the pressure and temperature response of the containment atmosphere to a main steamline break inside of containment. The steamline break is postulated as a full double-ended rupture (DER) of the steamline immediately downstream of the steam generator integral flow restrictor. The blowdown from the faulted steam generator is limited by the 1.4 ft^2 integral flow restrictor. The steamline non-return valve limits the reverse break flow to the steam in the steamline between the break and the non-return valve.

A spectrum of cases was considered in this analysis. All cases were analyzed at EPU conditions with a full DER. The cases included variations in initial power level and the single failure. The case resulting in the highest containment pressure was a full DER initiated from 30% power with the feedwater isolation valve (FIV) postulated to fail open. The open FIV allows additional main feedwater to be pumped into the faulted steam generator until the feedwater regulator valve (FRV) closes. Furthermore the FRV is located upstream of the FIV, creating a larger unisolable feedline volume. Additional hot water in the feedline will flash and enter the faulted steam generator when the feedwater becomes saturated due to the depressurization of the system.

Method of Analysis

The analysis consists of the calculation of the mass and energy releases from the steamline break and the calculation of the containment pressure and temperature response. The methods and assumptions of these calculations are summarized below.

Mass and Energy Release Calculation

WCAP-8822, "Mass and Energy Releases Following a Steam Line Rupture" (Reference 2) forms the basis for the assumptions and models used in the calculation of the mass and energy releases resulting from a steamline rupture. The steamline break mass and energy releases are generated using the NRC-approved LOFTRAN code (Reference 1). The Westinghouse steamline break mass and energy release methodology using LOFTRAN was approved by the NRC and is documented in Supplement 2 to WCAP-8822 (Reference 3).

The major inputs and assumptions affecting the mass and energy releases to containment are summarized below.

- The NSSS power level is 1806 MWt. Cases are analyzed at 100.6%, 70%, 30% and hot zero power.

- The full power RCS average temperature is 583.4°F, which includes a +6.4°F uncertainty.

- The core nuclear power transient due to the cooldown following the steamline rupture is based on end-of-core life conditions with the most reactive control rod stuck out of the core. The credited shutdown margin is 2.0%Δk.

- Two sources of latent energy to the reactor coolant system are modeled: the reactor vessel and primary system thick metal, and the fluid inventory in the intact steam generator.

- Offsite power is assumed to remain available. The largest effect of this assumption is the continued operation of the reactor coolant pumps, which maintains a high heat transfer rate to the steam generators.

- Minimum flowrates are modeled from ECCS injection, to conservatively minimize the amount of boron that provides negative reactivity feedback. The flowrates correspond to a single train of ECCS. The hydraulic performance of the ECCS systems assumed in the transient and accident analysis is based on certified pump curves lowered uniformly to provide head margin for periodic pump testing. The flowrates are assured by the plant in-service testing acceptance criteria.

- A high initial steam generator mass is assumed. The initial level corresponds to 64% NRS + 10% uncertainty.

- The calculation of secondary side break flow is based on the Moody critical flow correlation with fL/D=0.

- The main feedwater modeling accounts for an increase from the initial flowrate due to the depressurization of the faulted steam generator and the opening of the FRV in response to the increased steam flow. Main feedwater pumped flow is terminated by the closure of the FIV or FRV (when the FIV is assumed to fail open).

- Feedline flashing occurs when saturated conditions are reached in the 225 ft^3 unisolable volume between the faulted steam generator and the FIV or 355 ft^3 unisolable volume between the faulted steam generator and the FRV (when the FIV is assumed to fail open).

- Maximum flowrates of auxiliary feedwater (AFW) were assumed, with the AFW start conservatively modeled at the time of the SI signal, with no delay. AFW is assumed to be manually re-aligned at 600 seconds to prevent further water addition to the faulted steam generator (See Results Section for additional discussion). Cases have been analyzed with a control failure that increases AFW flowrates; however, these cases have been shown to be non-limiting and do not require isolation within 600 seconds (Reference 5).

- The steam in the unisolable volume of 1650 ft^3 between the faulted steam generator and the steamline non-return check valve comprises the reverse flow from the break.

- The break effluent is assumed to be dry, saturated steam throughout most of the transient. However, when a large double-ended break first occurs, it is expected that there will be a significant quantity of liquid in the break effluent. A conservative amount of liquid entrainment is assumed to occur in the beginning of the steam generator blowdown phase of the accident. The break effluent is assumed to return to all vapor within the first 25 seconds.

- The containment backpressure is modeled within LOFTRAN at a conservatively low value of 14.7 psia.

- The time to tube uncovery was modeled in the same manner as was used in Reference 3 for "predicted tube uncovery" cases. This affects the total amount of heat transfer to the secondary side, and the possible generation of superheated steam.

Containment Response Calculation

The GOTHIC computer code (Reference 4, Reference 16 and Reference 17) is used to calculate the containment pressure and temperature transient response following the postulated steamline break accident inside containment.

The initial conditions (Table 14.2.5-3) are selected to maximize the containment pressure response. The initial pressure has a direct relationship on the peak containment pressure, and thus is maximized. The initial temperature is maximized because the steady-state temperature of the containment heat sinks are assumed to be the same as the containment air temperature. The higher initial heat sink temperature causes them to be less effective in removing heat. The initial humidity is conservative when it is assumed to be low, since this maximizes the amount of air initially in the containment.

Two trains of containment fan coolers (four coolers) and two trains of containment spray are credited in the limiting case because the FIV failure has already been modeled in the mass and energy release calculation. Cases were analyzed with a containment safeguards train single failure, but were shown to be non-limiting. Conservative values for containment fan cooler heat removal performance were used. A conservatively high temperature has been assumed as the temperature of the spray water. The containment spray pump flow rates are conservatively low. Pump performance is based on certified pump curves lowered uniformly to provide head margin for periodic pump testing. The required system flowrate is assured by the plant in-service testing acceptance criteria.

Finally, the heat transfer through, and heat storage in, interior and exterior walls of the containment structure are considered. Structural heat sinks, consisting of steel and concrete, are modeled as slabs having specific areas and layers of varying thickness. The initial temperature of the structural heat sinks is assumed to be the initial containment air temperature of 120°F.

Results

The containment pressure and containment temperature transients are shown in Figure 14.2.5-3 and Figure 14.2.5-4. The peak containment pressure of 59.56 psig is reached at 264.2 seconds, which is below the 60 psig containment design pressure. The peak containment air temperature of 285.5°F is also reached at 254.2 seconds, which is below the containment design temperature of 286°F (see FSAR Section 5.1).

Both containment pressure and temperature trend consistently downward after peaking. This is due to heat removal by both active systems and passive heat sinks exceeding heat introduction from the break. At 600 seconds, the rate of pressure and temperature drop increases when AFW flow is isolated to the faulted steam generator. It is apparent from Figure 14.2.5-3 and Figure 14.2.5-4 however, that the isolation of AFW at 600 seconds does not affect the peak containment pressure and temperature experienced earlier in the transient because both parameters are already decreasing before AFW isolation occurs. Therefore, while the manual isolation of AFW is an input assumed by the analysis, the results of the analysis show that the manual action at 600 seconds is not necessary to ensure that containment integrity is maintained.

Conclusions

A DNB analysis has been performed. It was found that all cases have a minimum DNBR greater than the limit value. The calculated peak linear heat generation rate is less than a value corresponding to fuel centerline melting.

The analysis has shown that the criteria stated in Section 14.2.5 are satisfied. Although DNB and possible cladding perforation following a steam pipe rupture are not necessarily unacceptable and not precluded by the criteria, the above analysis, in fact, shows that the DNB design basis is met as stated in Section 3.2.

No significant exposure to the public would result from a rupture of a steam pipe.

The containment pressure and temperature responses to a MSLB inside of containment remain below the containment design pressure and temperature.

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6.   Deleted.

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Table 14.2.5-1   MAIN STEAMLINE BREAK ACCIDENT DOSES

Site Boundary	Dose (Rem)
Accident-Induced Spike TEDE Dose (3.9 - 5.9 hr)	0.2
Pre-Accident Spike TEDE Dose (0 - 2 hr)	0.14

Low Population Zone (0 - 60 hr)	Dose (Rem)
Accident-Induced Spike TEDE Dose	0.08
Pre-Accident Spike TEDE Dose	0.03

Control Room (0 - 30 days)	Dose (Rem)
Accident-Induced Spike TEDE Dose	4.0
Pre-Accident Spike TEDE Dose	1.9

Table 14.2.5-2   RUPTURE OF A STEAM PIPE ANALYSIS ASSUMPTIONS AND
                 SEQUENCE OF EVENTS

PBNP Unit Affected	Unit1	Unit 1	Unit 2	Unit 2
Steam Generator Model	44F	44F	Delta-47	Delta-47
Initial shutdown margin, %Δk	2.0	2.0	2.0	2.0
Offsite Power Available	Yes	No	Yes	No
Main steam line ruptures in loop 1, sec	0.0	0.0	0.0	0.0
High-High steam flow setpoint reached in loop 1, sec	0.01	0.01	0.01	0.01
High-high steam line flow setpoint reached in loop 2, sec	0.23	0.23	0.24	0.24
Low steam pressure SI setpoint reached in loop 1, steam line isolation logic satisfied in both loops, sec	1.5	1.5	1.4	1.4
RCPs begin to coastdown, sec	NA	3.0	NA	3.0
SI actuation occurs, sec	3.5	3.5	3.4	3.4
Steam line isolation completed in both loops, sec	8.5	8.5	8.4	8.4
SI pump starts, sec	3.5	19.5	3.4	19.4
Main feedwater isolation completed in both loops, sec	13.5	13.5	13.4	13.4
SI pump achieves full speed, sec	14.5	29.5	14.4	29.4
SI flow injection begins (cold leg pressure below SI pump shutoff pressure), sec	16.9	19.5	16.2	19.5
Criticality attained, sec	39.8	49.0	39.5	49.0
Accumulators begin to inject, sec	73.0	229.0	66.5	230.5
Boron reaches the core (> 1 ppm), sec	93.5	111.5	92.0	110.0
Time of maximum core heat flux, sec	112.5	115.3	108.8	264.8
Maximum core heat flux, fraction of nominal	0.2095	0.0332	0.2058	0.0360

Table 14.2.5-3   GOTHIC MODEL INPUTS MSLB CONTAINMENT RESPONSE
ANALYSIS

Input	Value
RWST water temperature for containment sprays (°F)	100
Initial containment temperature (°F)	120
Initial containment pressure (psia)	16.7
Initial relative humidity (%)	20
Net free volume (ft^3)	1.0×10^6

Figure 14.2.5-1 RUPTURE OF A STEAM PIPE UNIT 1 WITH OFFSITE POWER
Sheet 1 of 6
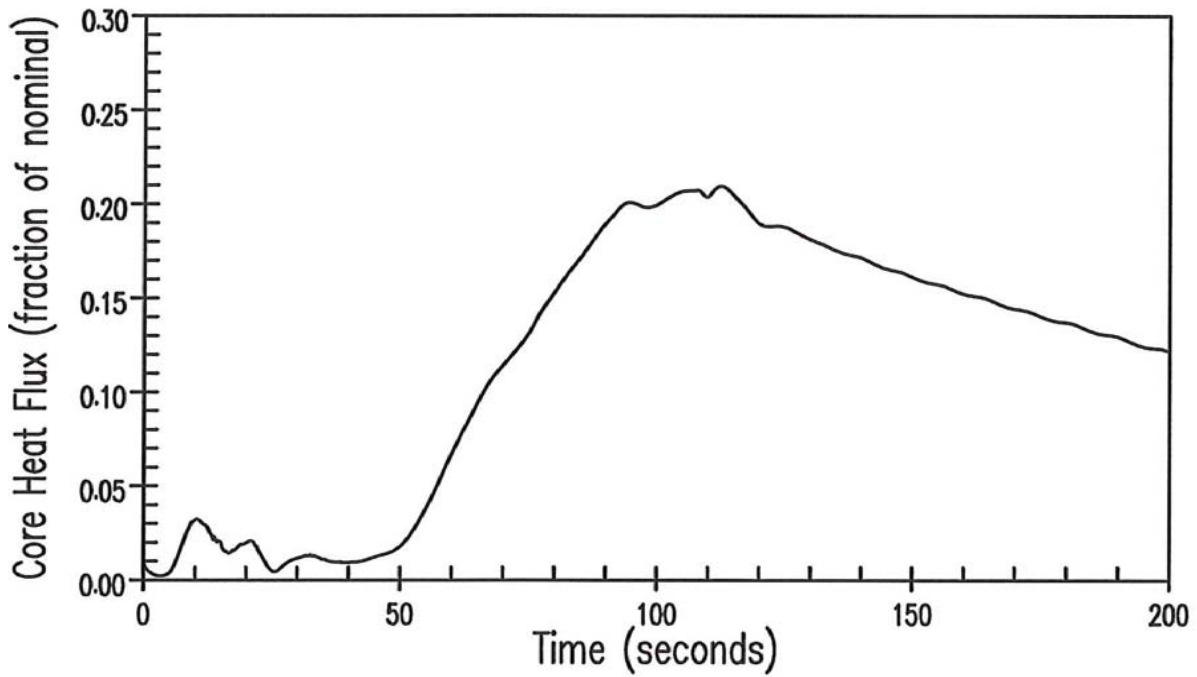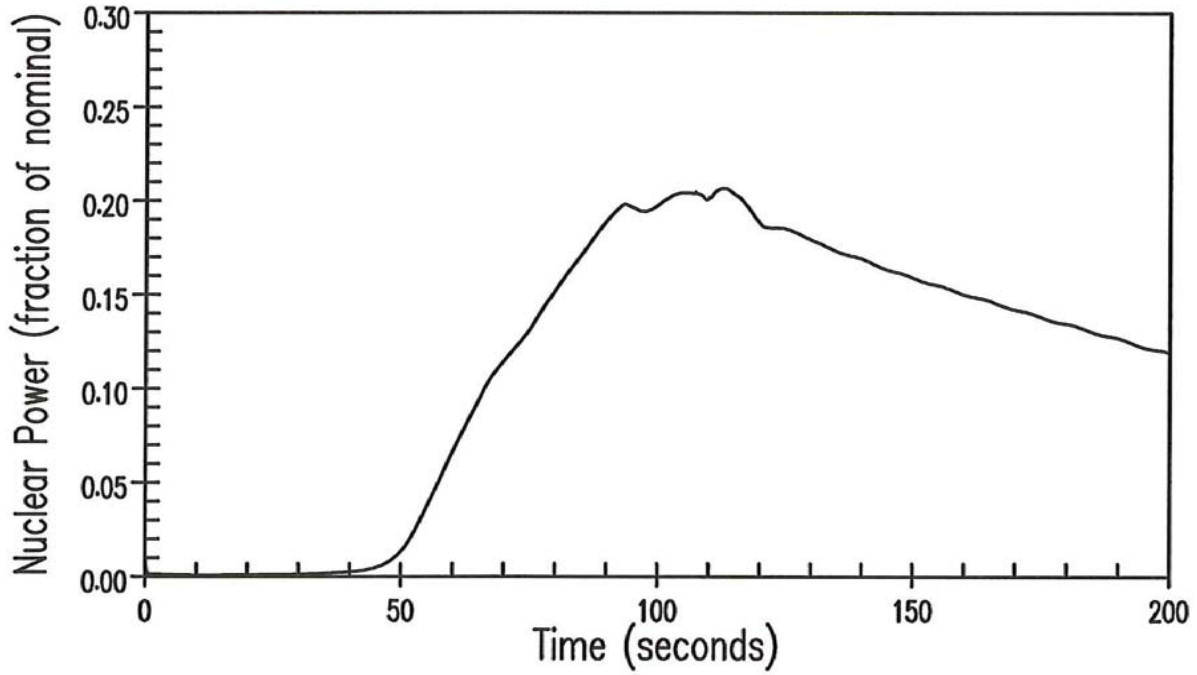
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Sheet 2 of 6
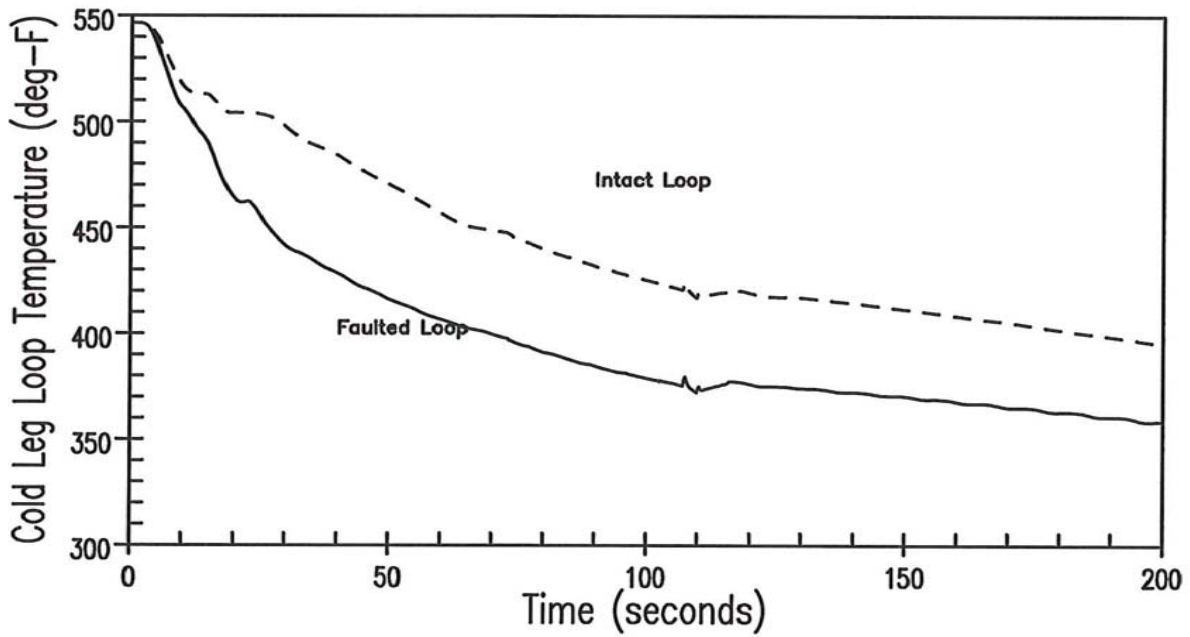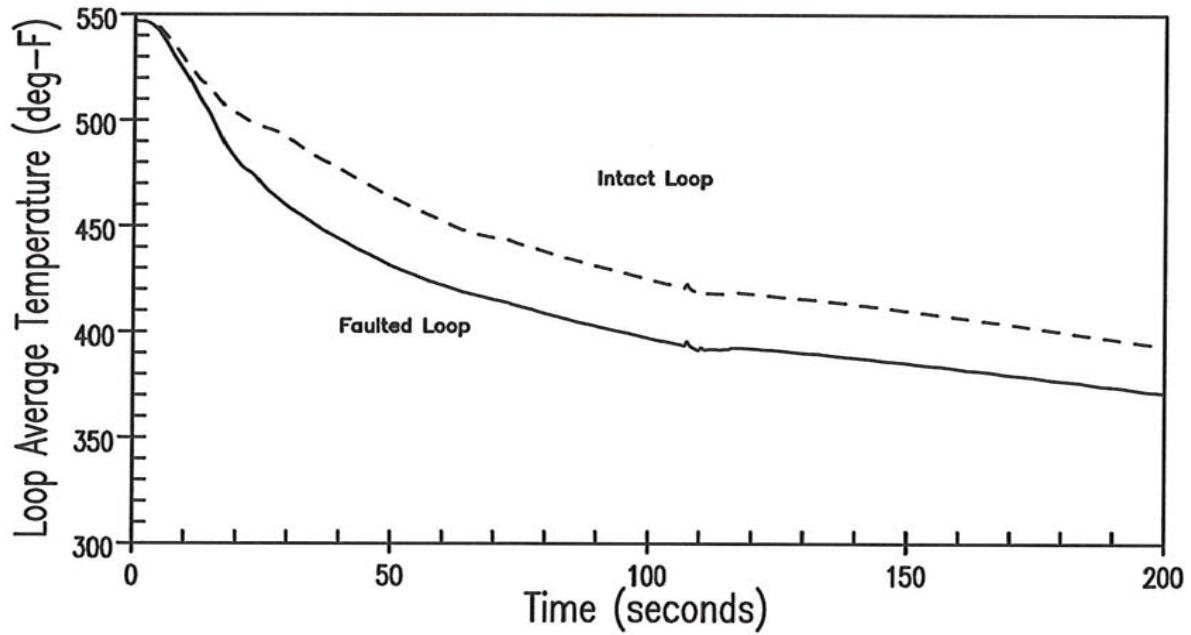
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Sheet 3 of 6
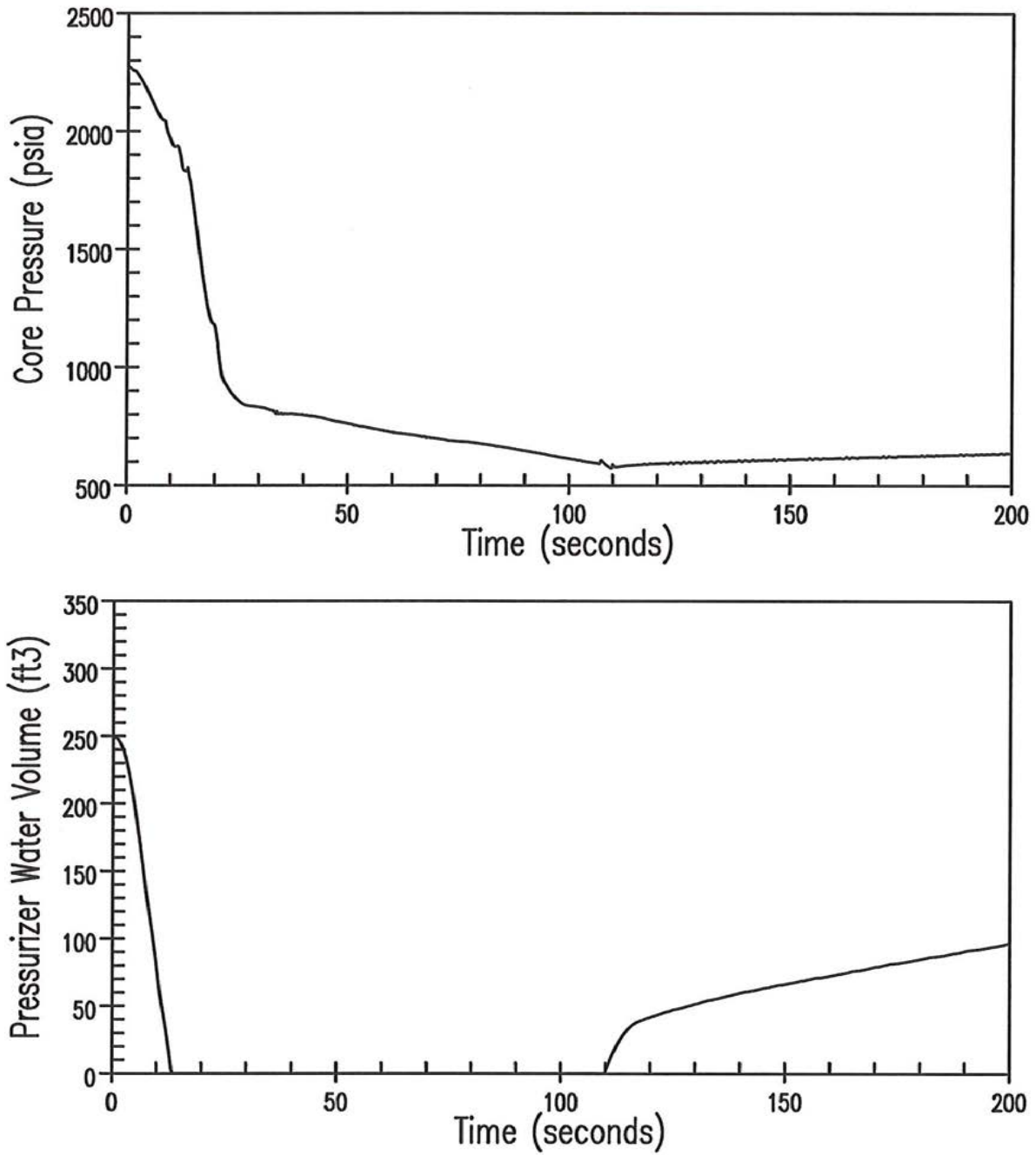
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Sheet 4 of 6
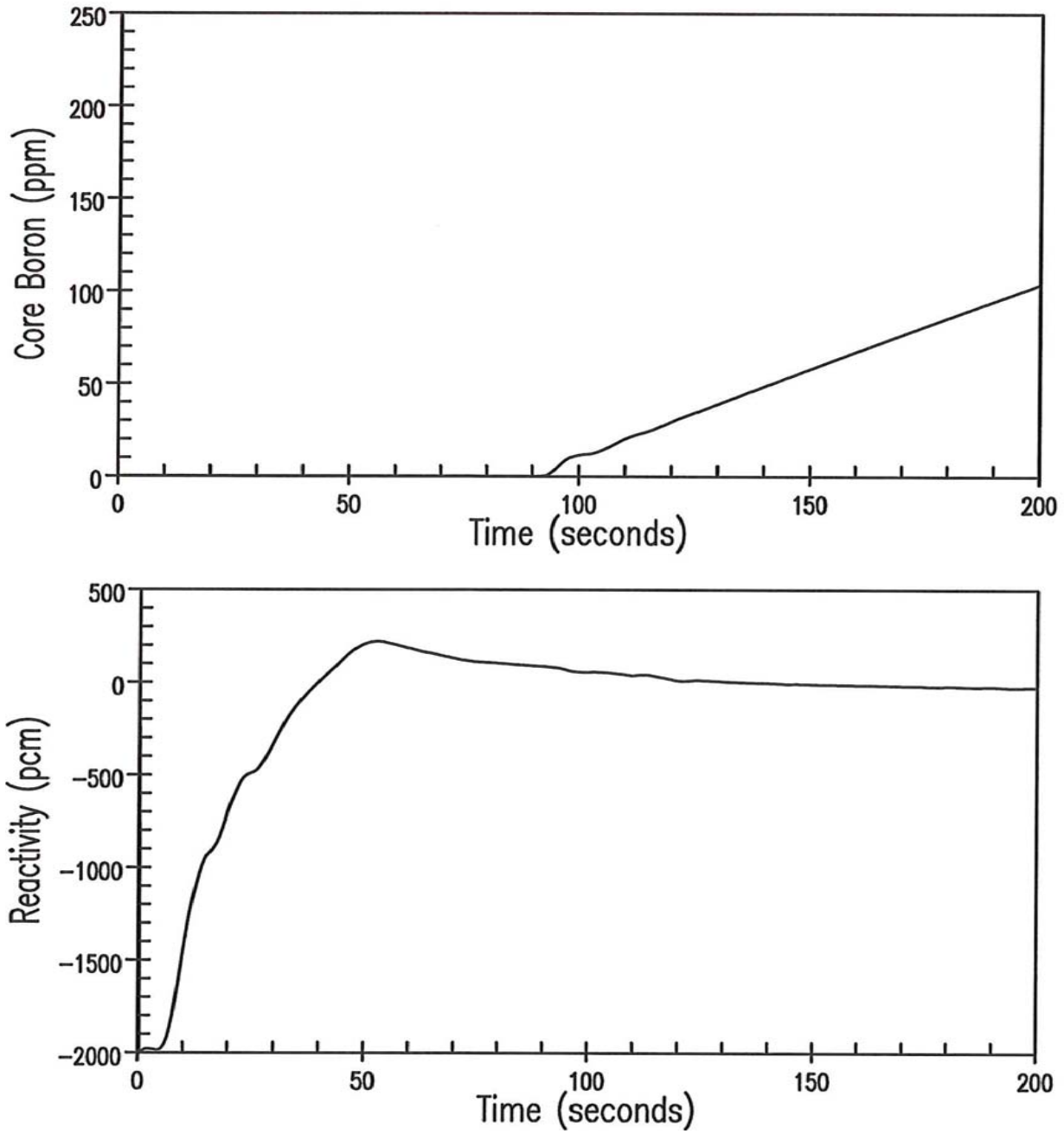
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Sheet 5 of 6
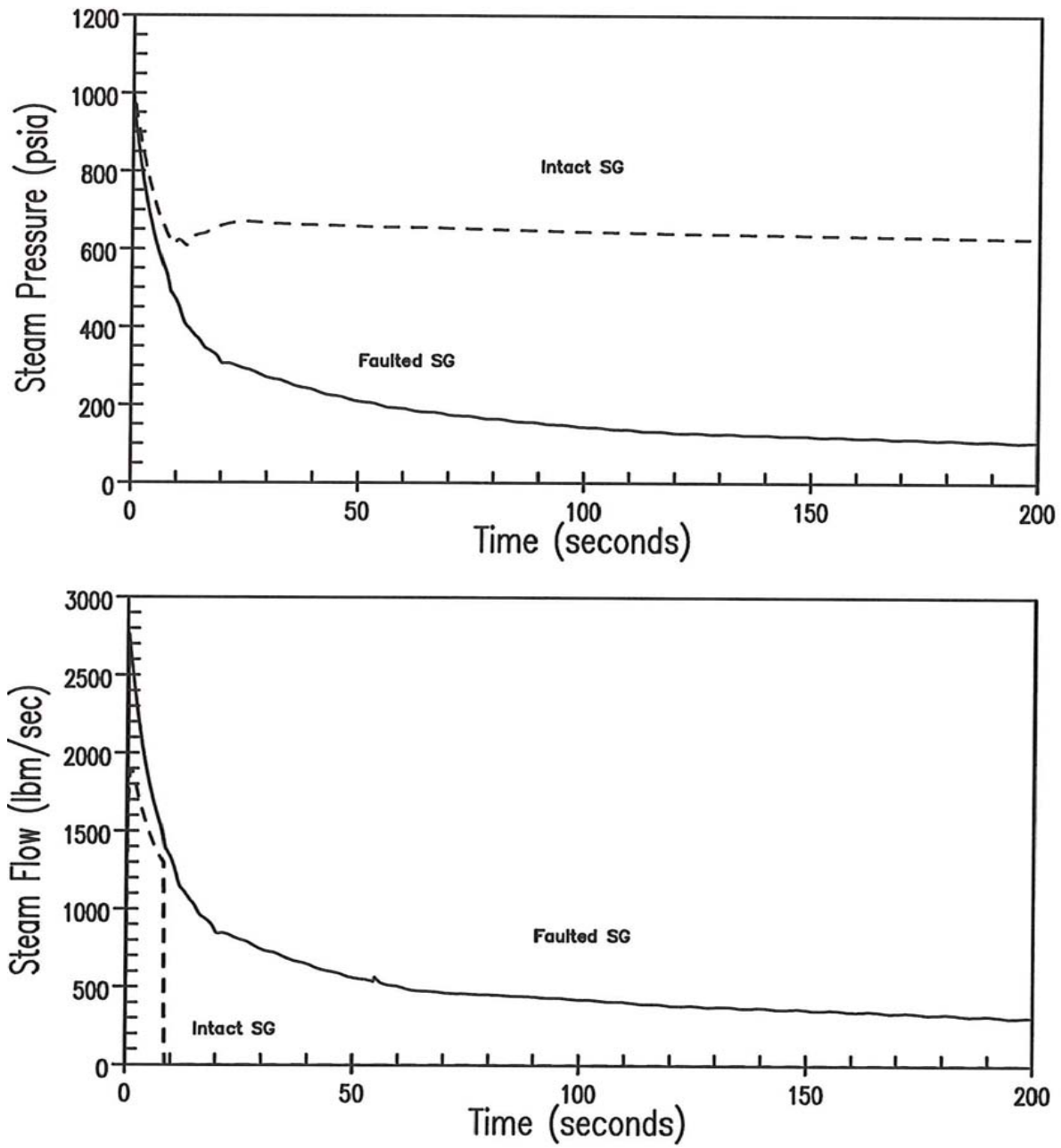
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Sheet 6 of 6
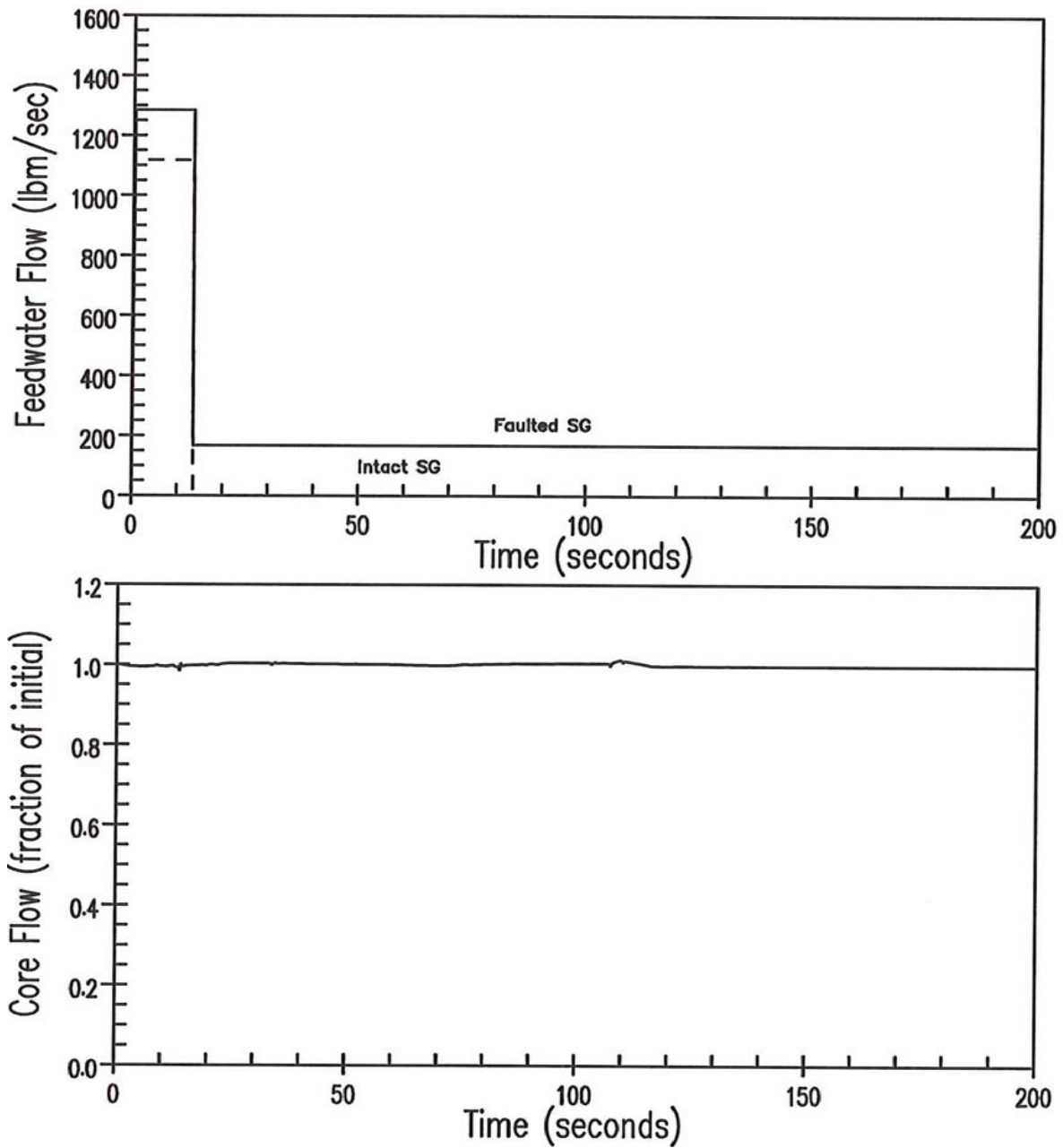
Figure 14.2.5-2 RUPTURE OF A STEAM PIPE UNIT 1 WITHOUT OFFSITE POWER
Sheet 1 of 6
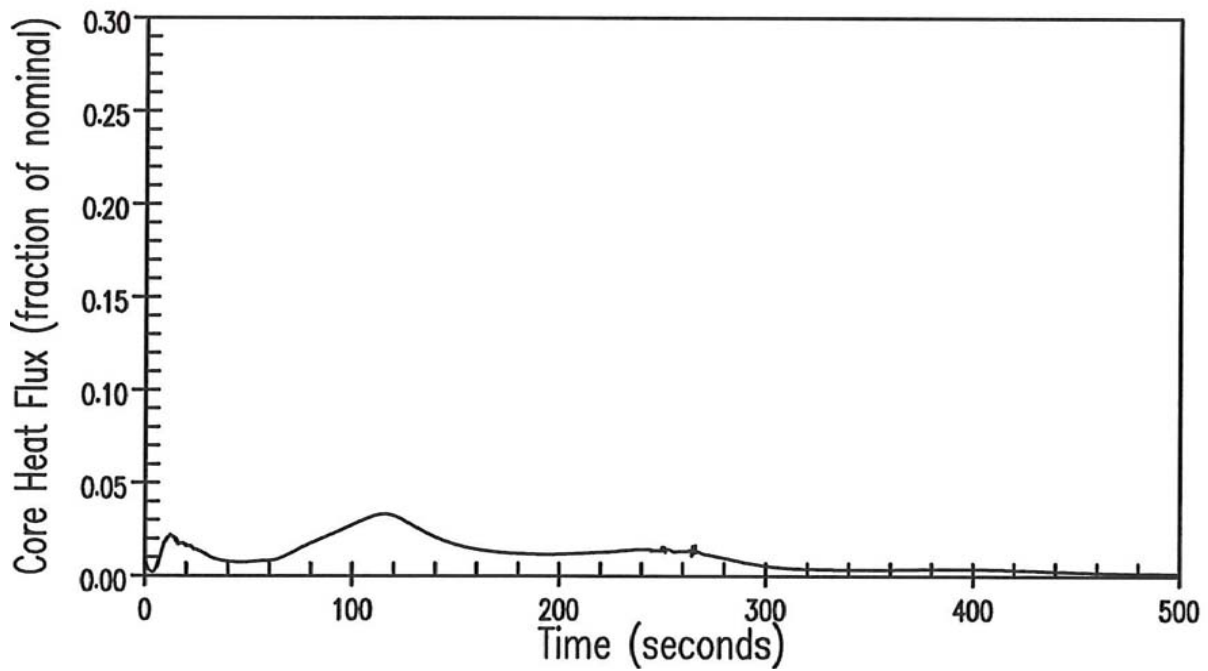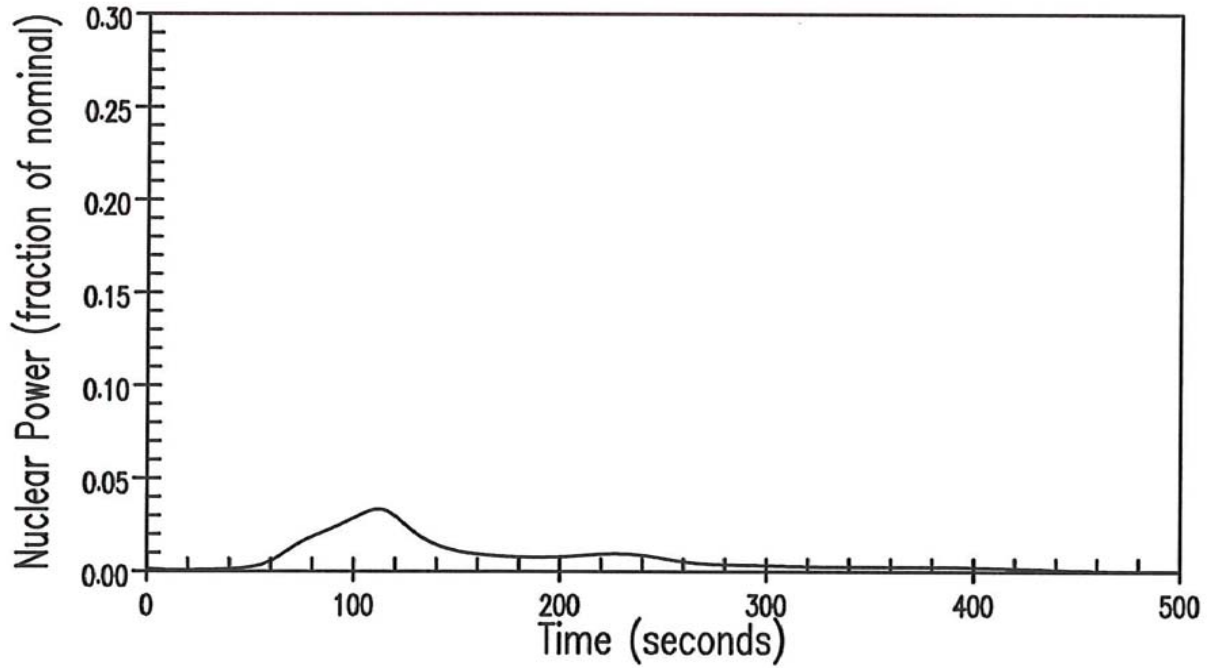
Figure 14.2.5-2 RUPTURE OF A STEAM PIPE UNIT 1 WITHOUT OFFSITE POWER
Sheet 2 of 6
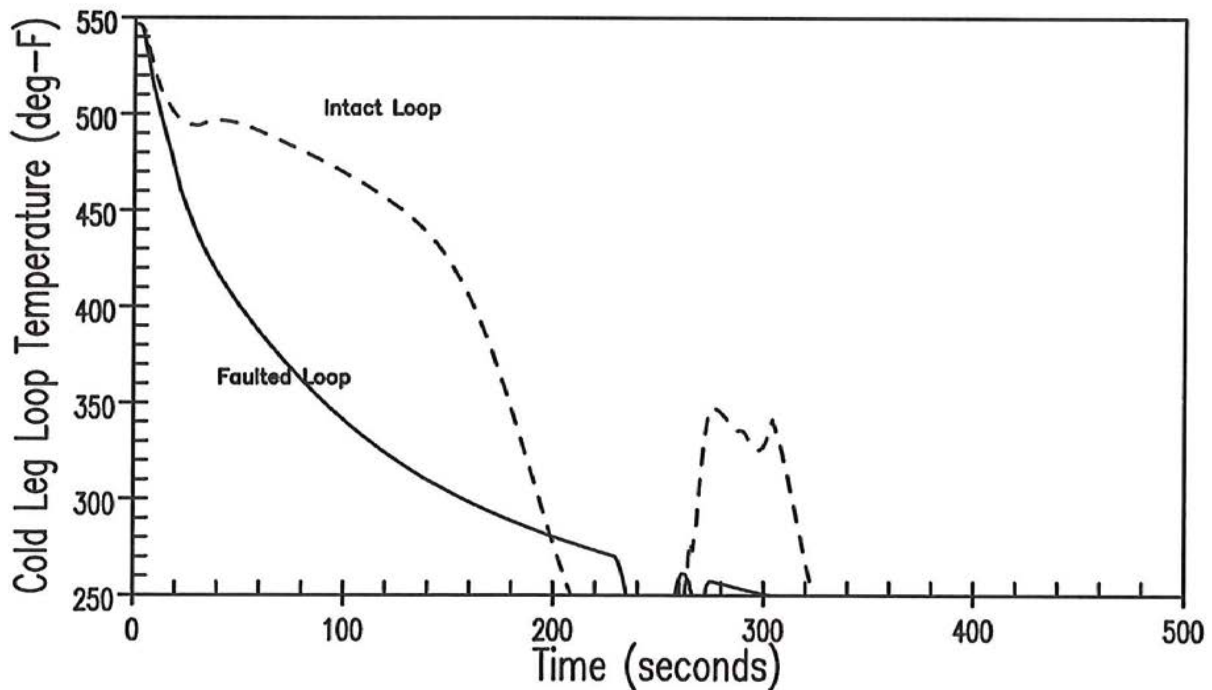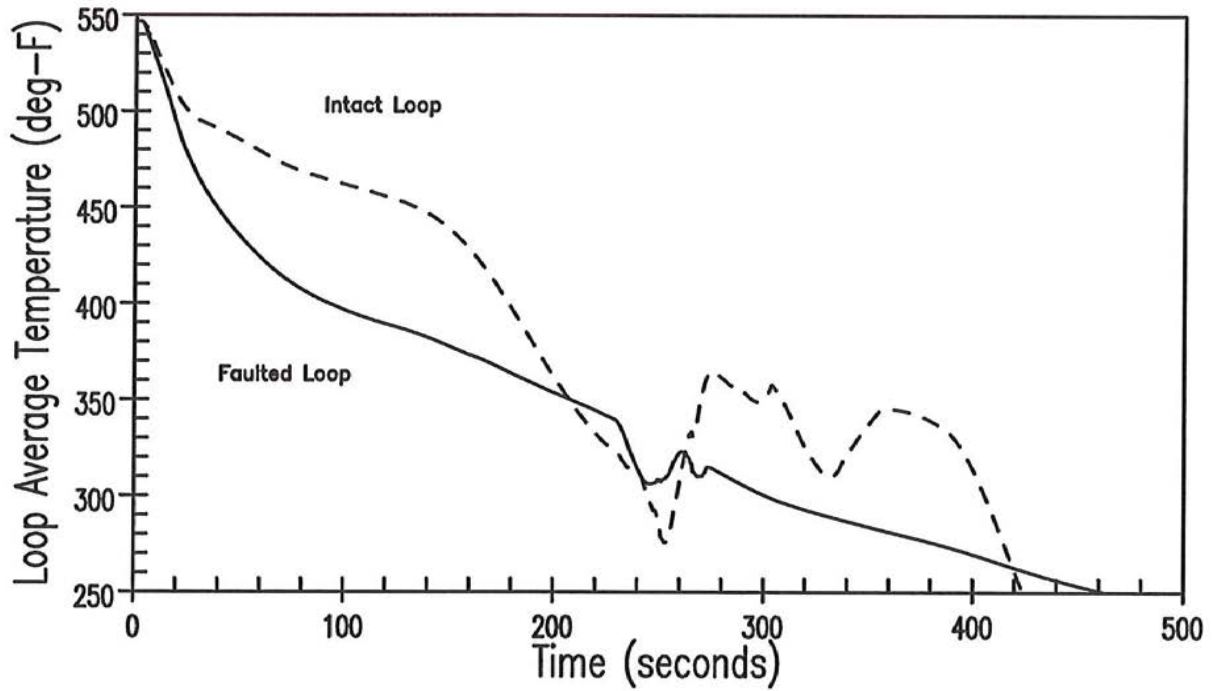
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Sheet 3 of 6
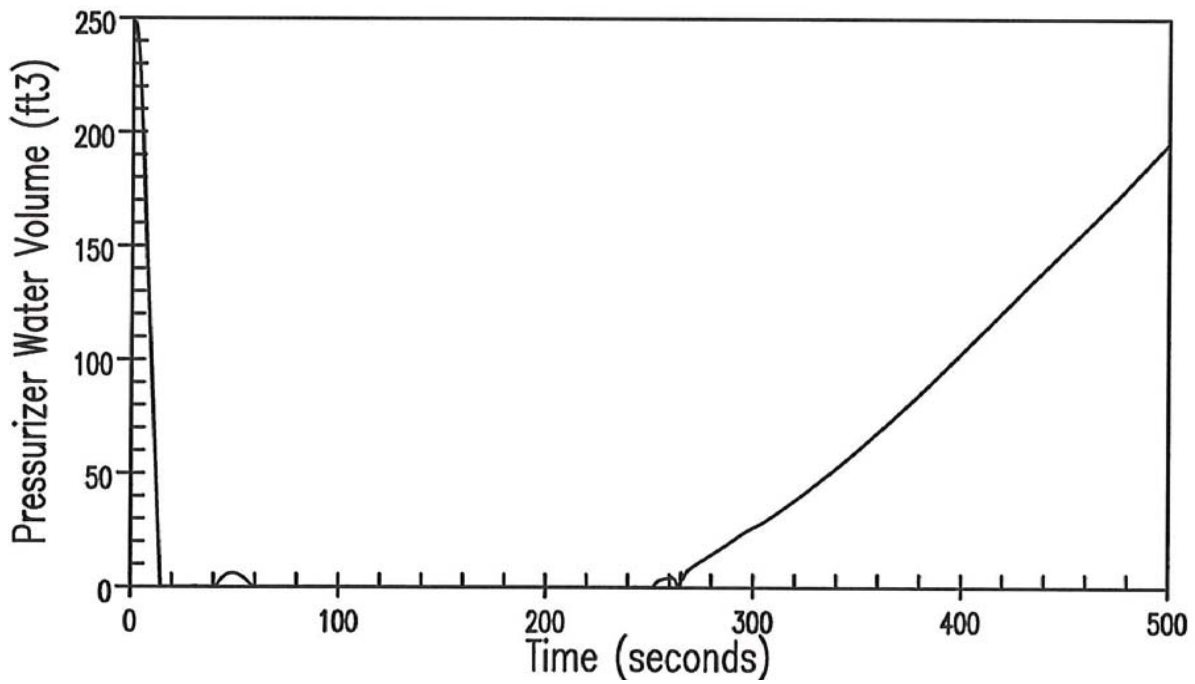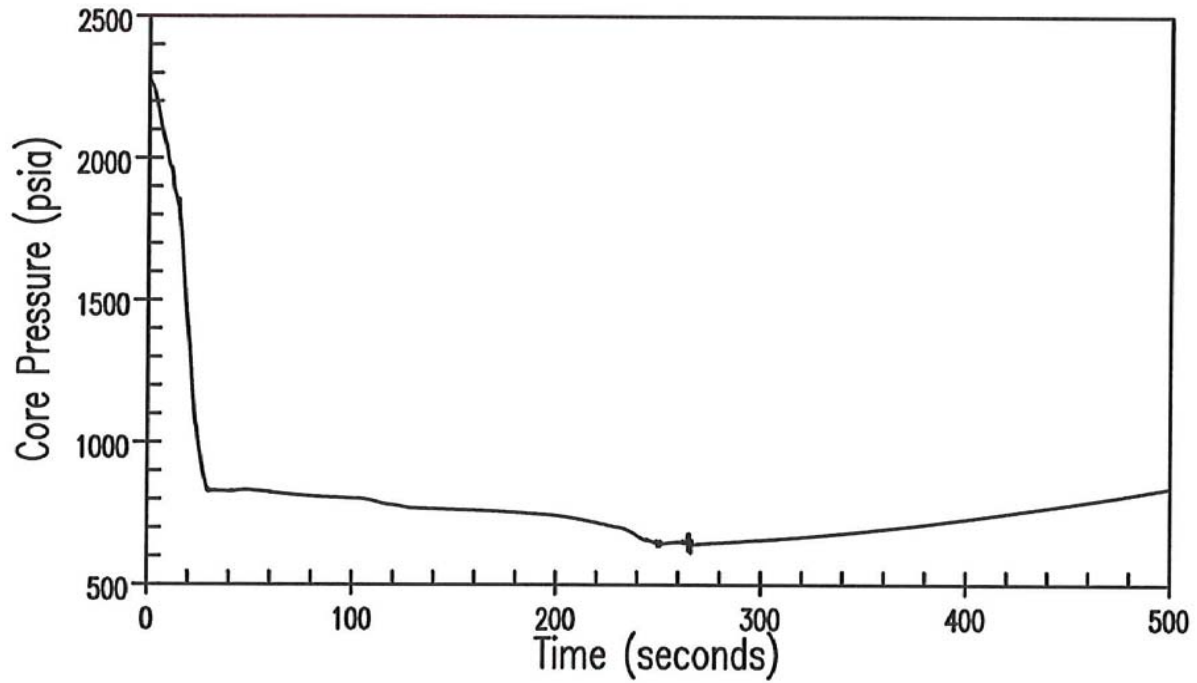
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Sheet 4 of 6
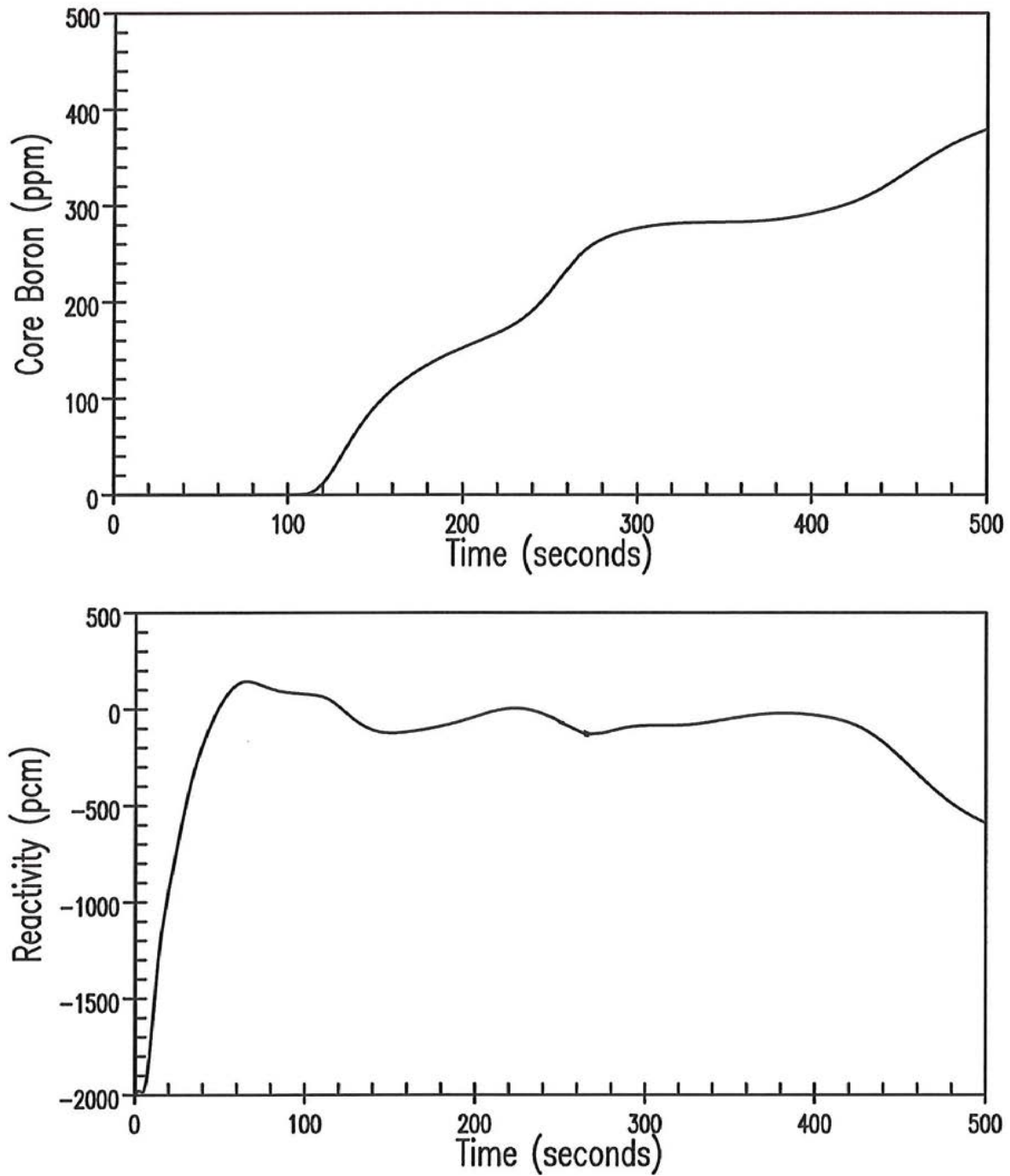
Figure 14.2.5-2 RUPTURE OF A STEAM PIPE UNIT 1 WITHOUT OFFSITE POWER
Sheet 5 of 6
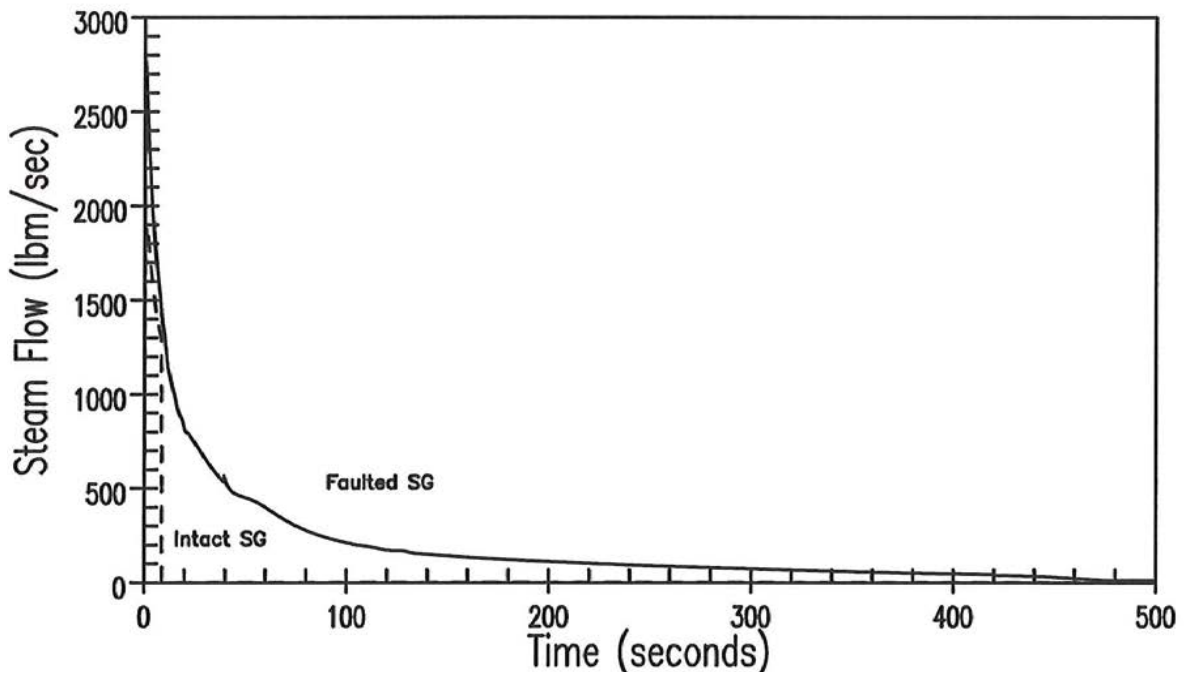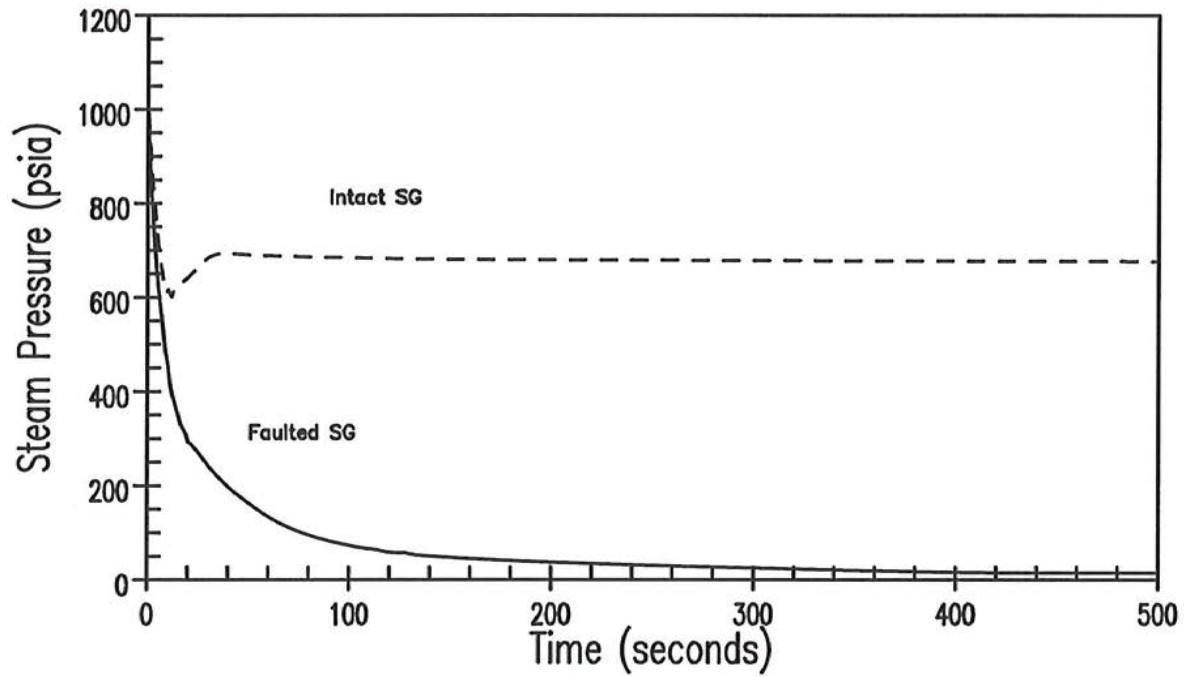
Figure 14.2.5-2 RUPTURE OF A STEAM PIPE UNIT 1 WITHOUT OFFSITE POWER
Sheet 6 of 6
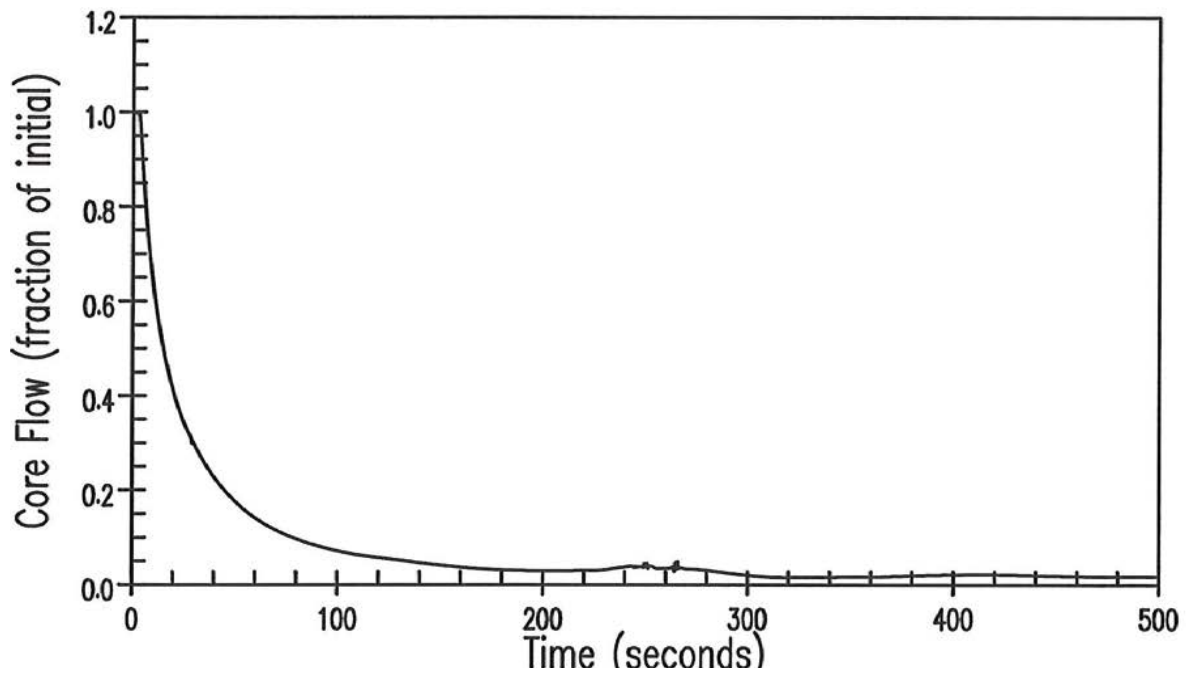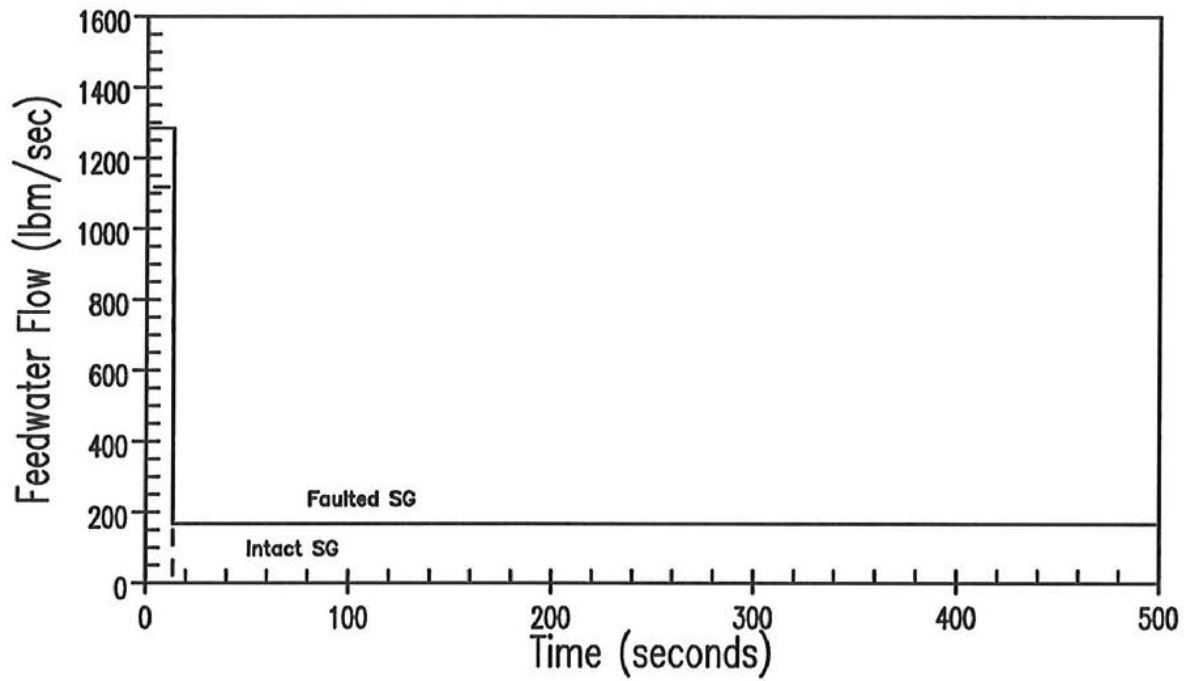
Figure 14.2.5-3 CONTAINMENT PRESSURE MSLB CONTAINMENT RESPONSE
ANALYSIS
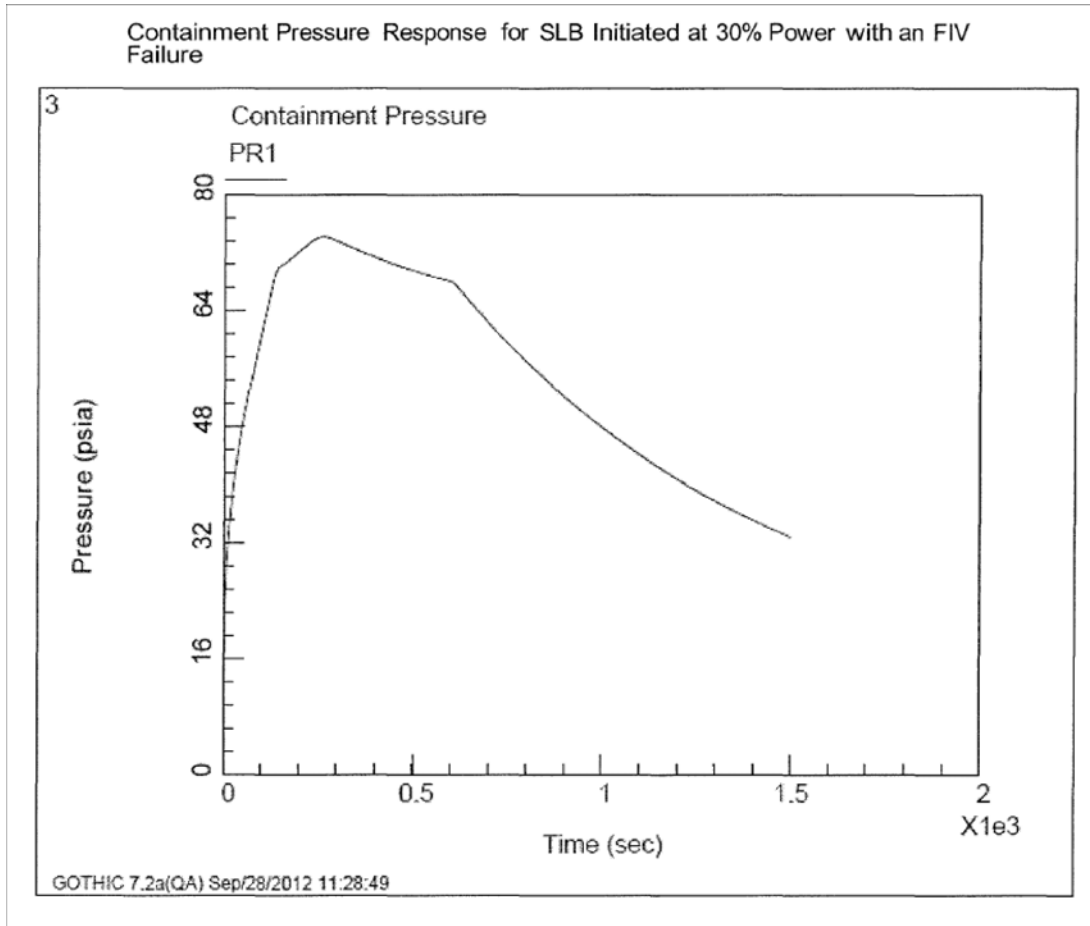
Figure 14.2.5-4 CONTAINMENT TEMPERATURE MSLB CONTAINMENT RESPONSE
ANALYSIS

Containment Temperature Response for SLB Initiated at 30% Power with an FIV Failure
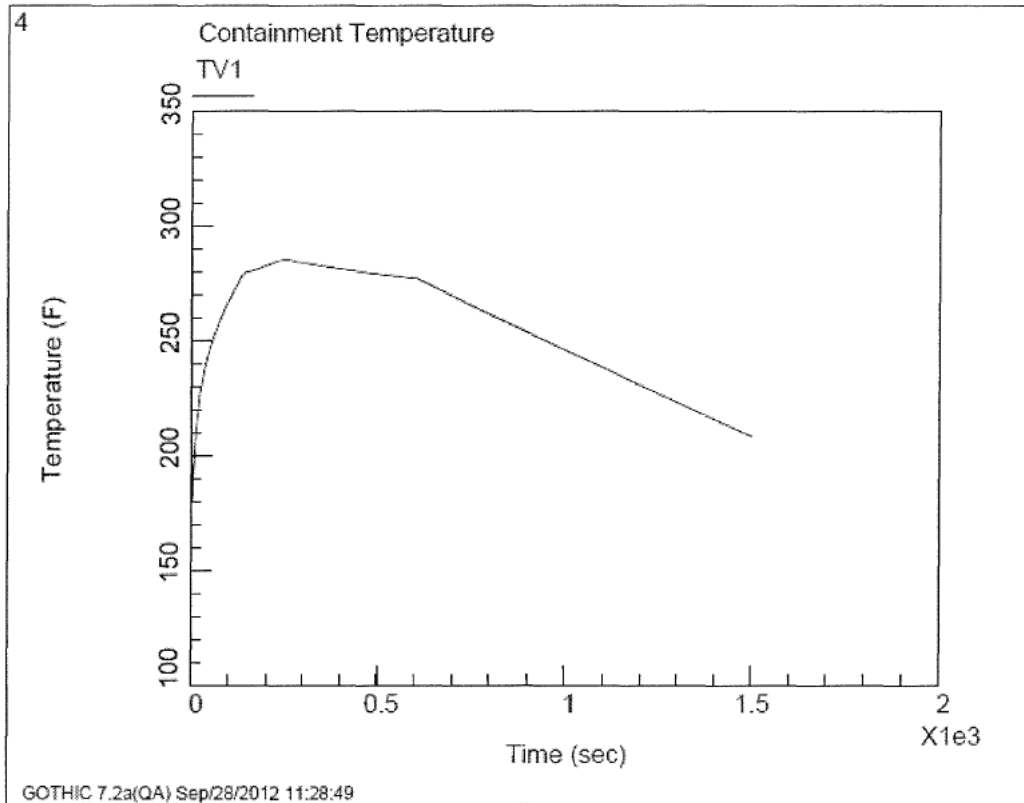
14.2.6   RUPTURE OF A CONTROL ROD MECHANISM HOUSING - RCCA EJECTION

In order for this accident to occur, a rupture of the control rod mechanism housing must be postulated, creating a full system pressure differential acting on the drive shaft. The resultant core thermal power excursion is limited by the Doppler reactivity effects of the increased fuel temperature and terminated by reactor trip actuated by high nuclear power signals.

A failure of a control rod mechanism housing sufficient to allow a control rod to be rapidly ejected from the core is not considered credible for the following reasons:

1.   Each control rod drive mechanism housing is completely assembled and shop-tested at 3105 psig (nominal).

2.   Stress levels in the mechanism are not affected by system transients at power, or by the thermal movement of the coolant loops. Moments induced by the design earthquake can be accepted within the allowable primary working stress range specified by the ASME code, Section III, for Class 1 components.

3.   The latch mechanism housing and rod travel housing are Grade F316 stainless steel. This material exhibits excellent notch toughness at all temperatures that will be encountered. The joints between the latch mechanism housing and head adapter, and between the latch mechanism housing and rod travel housing, are fabricated with full penetration welds.

Nuclear Design

Even if a rupture of a RCCA drive mechanism housing is postulated, the operation of a plant 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 RCCA's is continuously indicated in the control room. An alarm will occur if a bank of RCCA's approaches its insertion limit or if one RCCA deviates from its bank. Operating instructions require boration at the low-low alarm.

Reactor Protection

The reactor protection in the event of a rod ejection accident has been described in Reference 4. The protection for this accident is provided by high neutron flux trip (high and low setting). These protection functions are described in detail in Section 7.2 of the FSAR.

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. However, even if damage is postulated, it would not be expected to lead to a more severe transient, since RCCA's 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.

Limiting Criteria

This event is classified as an ANS Condition IV incident. Due to the extremely low probability of a RCCA ejection accident, some fuel damage could be considered an acceptable consequence.

Comprehensive studies, both of the threshold of fuel failure and of the threshold or significant conversion of the fuel thermal energy to mechanical energy, have been carried out as part of the SPERT project by the Idaho Nuclear Corporation. Extensive tests of UO_2 zirconium clad fuel rods representative of those in pressurized water reactor type cores have demonstrated failure thresholds in the range of 240 to 257 cal/gm. However, other rods of a slightly different design have exhibited failures as low as 225 cal/gm. These results differ significantly from the TREAT results, which indicated that this threshold decreases by about 10% with fuel burnup. The cladding failure mechanism appears to be melting for zero burnup rods and brittle fracture for irradiated rods. Also important is the conversion ratio of thermal to mechanical energy. This ratio becomes marginally detectable above 300 cal/gm for unirradiated rods and 200 cal/gm for irradiated rods; catastrophic failure (large fuel dispersal, large pressure rise) even for irradiated rods did not occur below 300 cal/gm.

In view of the above experimental results, criteria are applied to ensure that there is little or no possibility of fuel dispersal in the coolant, gross lattice distortion, or severe shock waves. These criteria are:

a.   Average fuel pellet enthalpy at the hot spot below 200 cal/gm (360 Btu/lbm) for irradiated fuel. This bounds non-irradiated fuel which has a slightly higher enthalpy limit.

b.   Peak reactor coolant pressure less than that which could cause stresses to exceed the faulted condition stress limits.

c.   Fuel melting limited to less than the innermost ten percent of the fuel pellet at the hot spot, even if the average fuel pellet enthalpy is below the limits of criterion (a) above.

Method of Analysis

The calculation of the transient is performed in two stages, first an average core calculation and then a hot region calculation. The average core calculation is performed using spatial neutron kinetics methods to determine the average power generation with time including the various total core feedback effects, i.e., Doppler reactivity and moderator density reactivity. Enthalpy and temperature transients in the hot spot are determined by adding a multiple of the average core energy generation to the hotter rods and performing a transient heat-transfer calculation. The asymptotic power distribution calculated without feedback is pessimistically assumed to persist throughout the transient.

Average Core Analysis

The spatial kinetics computer code, TWINKLE (Reference 4 in Section 14.0), is used for the average core transient analysis. This code solves the two group neutron diffusion theory kinetic equation in one, two or three spatial dimensions (rectangular coordinates) for six delayed neutron groups and up to 8000 spatial points. The computer code includes a detailed multiregion, transient fuel-cladding-coolant heat transfer model for calculation of pointwise Doppler and moderator feedback effects. In this analysis, the code is used as a one dimensional axial kinetics code, since it allows a more realistic representation of the spatial effects of axial moderator feedback and RCCA movement. However, since the radial dimension is missing, it is still necessary to employ very conservative methods (described in the following) of calculating the ejected rod worth and hot channel factor. Further description of TWINKLE appears in Section 14.0.

Hot Spot Analysis

In the hot spot analysis, the initial heat flux is equal to the nominal times the design hot channel factor. During the transient, the heat flux hot channel factor is linearly increased to the transient value in 0.1 second, the time for full ejection of the rod. Therefore, the assumption is made that the hot spots before and after ejection are coincident. This is very conservative, since the peak after ejection will occur in or adjacent to the assembly with the ejected rod, and prior to ejection the power in this region will necessarily be depressed.

The hot spot analysis is performed using the detailed fuel-and cladding transient heat transfer computer code, FACTRAN (Reference 2 in Section 14.0). This computer code calculates the transient temperature distribution in a cross section of a metal clad UO_2 fuel rod, and the heat flux at the surface of the rod, using as input the nuclear power versus time and the local coolant conditions. The zirconium-water reaction is explicitly represented, and all material properties are represented as functions of temperature. A conservative pellet radial power distribution is used within the fuel rod.

FACTRAN uses the Dittus-Boelter or Jens-Lottes correlation to determine the film heat transfer before DNB, and the Bishop-Sandburg-Tong (BST) correlation to determine the film boiling coefficient after DNB. The BST correlation is conservatively used assuming zero bulk fluid quality. The DNB ratio is not calculated, instead the code is forced into DNB by specifying a conservative DNB heat flux. The gap heat transfer coefficient can be calculated by the code; however, it is adjusted in order to force the full power steady-state temperature distribution to agree with the fuel heat transfer design codes. Further description of FACTRAN appears in Section 14.0.

System Overpressure Analysis

Because safety limits for fuel damage specified earlier are not exceeded, there is little likelihood of fuel dispersal into the coolant. The pressure surge may therefore be calculated on the basis of conventional heat transfer from the fuel and prompt heat generation in the coolant. The pressure surge is calculated by first performing the fuel heat transfer calculation to determine the average and hot spot heat flux versus time. Using this heat flux data, a THINC (Section 3.2) calculation is conducted to determine the volume surge. Finally, the volume surge is simulated in a plant transient computer code. This code calculates the pressure transient taking into account fluid transport in the reactor coolant system and heat transfer to the steam generators. No credit is taken for the pressure reduction caused by the assumed failure of the control rod pressure housing.

Calculation of Basic Parameters

Input parameters for the analysis are conservatively selected on the basis of values calculated for this type of core. The more important parameters are discussed below. Table 14.2.6-1 presents the parameters used in this analysis.

Ejected Rod Worths and Hot Channel Factors

The values for ejected rod worths and hot channel factors are calculated using either three dimensional static methods or by a synthesis method employing one dimensional and two dimensional calculations. Standard nuclear design codes are used in the analysis. No credit is taken for the flux flattening effects of reactivity feedback. The calculation is performed for the maximum allowed bank insertion at a given power level, as determined by the rod insertion limits. Adverse xenon distributions are considered in the calculation.

Appropriate margins are added to the ejected rod worth and hot channel factors to account for any calculational uncertainties, including an allowance for nuclear power peaking due to densification.

Power distributions before and after ejection for a "worst case" can be found in Reference 4. During plant startup physics testing, ejected rod worths and power distributions are measured in the zero and full power rodded configurations and compared to values used in the analysis. It has been found that the ejected rod worth and power peaking factors are consistently overpredicted in the analysis.

Reactivity Feedback Weighting Factors

The largest temperature rises, and hence the largest reactivity feedbacks occur in channels where the power is higher than average. Since the weight of a region is dependent on flux, these regions have high weights. This means that the reactivity feedback is larger than that indicated by a simple channel analysis. Physics calculations have been carried out for temperature changes with a flat temperature distribution, and with a large number of axial and radial temperature distributions. Reactivity changes have been compared and effective weighting factors determined. These weighting factors take the form of multipliers which when applied to single channel feedbacks correct them to effective whole core feedbacks for the appropriate flux shape. In this analysis, since a one dimensional (axial) spatial kinetics method is employed, axial weighting is not necessary if the initial condition is made to match the ejected rod configuration. In addition, no weighting is applied to the moderator feedback. A conservative radial weighting factor is applied to the transient fuel temperature to obtain an effective fuel temperature as a function of time accounting for the missing spatial dimension.   These weighting factors have also been shown to be conservative compared to three dimensional analysis (Reference 4).

Moderator and Doppler Coefficient

The critical boron concentrations at the beginning of life and end of life are adjusted in the nuclear core in order to obtain moderator density coefficient curves which are conservative compared to actual design conditions for the plant. As discussed above, no weighting factor is applied to these results.

The Doppler reactivity defect is determined as a function of power level using a one dimensional steady-state computer code with a Doppler weighting factor of 1.0. The Doppler defect used is given in Table 14.2.6-1. The Doppler weighting factor will increase under accident conditions, as discussed above.

Delayed Neutron Fraction, β_{eff}

Calculations of the effective delayed neutron fraction (β_{eff}) typically yield values no less than 0.70% at beginning of life and 0.50% at end of life for the first cycle. The accident is sensitive to β if the ejected rod worth is equal to or greater than β as in zero power transients. In order to allow for reload cycles, pessimistic estimates of β of 0.49% at beginning of cycle and 0.43% at end of cycle were used in the analysis.

Trip Reactivity Insertion

The trip reactivity insertion assumed is given in Table 14.2.6-1 and includes the effect of one stuck RCCA. These values are reduced by the ejected rod reactivity. The shutdown reactivity has been simulated by dropping a rod of the required worth into the core. The start of rod motion occurs 0.5 second after the high neutron flux trip point is reached. This delay is assumed to consist of 0.2 second for the instrument channel to produce a signal, 0.15 second for the trip breaker to open and 0.15 second for the coil to release the rods. A curve of trip rod insertion versus time is used which assumes that insertion to the dashpot does not occur until 2.2 seconds after the start of fall. The choice of such a conservative insertion rate means that there is over one second after the trip point is reached before significant shutdown reactivity is inserted into the core. This is a particularly important conservatism for hot full power accidents.

Reactor Protection

Reactor protection for a rod ejection is provided by high neutron flux trip (high and low setting). These protection functions are part of the reactor trip system. No single failure of the reactor trip system will negate the protection functions required for the rod ejection accident, or adversely affect the consequences of the accident.

Results

Cases are presented for both beginning and end of life at zero and full power.

1.   Beginning of Cycle, Full Power

      Control bank D is assumed to be inserted to its insertion limit. The worst ejected rod worth and hot channel factor are conservatively calculated to be 400 pcm and 4.2 respectively. The peak hot spot cladding average temperature is 2142°F. The peak hot spot fuel center temperature reaches melting, which is conservatively assumed to be 4900°F. However, melting is restricted to less than 10% of the pellet.

2.   Beginning of Cycle, Zero Power

      For this condition, control bank D is assumed to be fully inserted and banks B and C are at their insertion limits. The worst ejected rod is located in control bank D and has a worth of 790 pcm and a hot channel factor of 11.0. The peak hot spot cladding average temperature reaches 2676°F, the fuel center temperature is 3959°F.

3.   End of Cycle, Full Power

Control bank D is assumed to be inserted to its insertion limit. The ejected rod worth and hot channel factors are conservatively calculated to be 420 pcm and 5.69 respectively. This results in a peak cladding average temperature of 2168°F. The peak hot spot fuel temperature reaches melting, conservatively assumed to be 4800°F. However, melting is restricted to less than 10% of the pellet.

4.   End of Cycle, Zero Power

The ejected rod worth and hot channel factor for this case are obtained assuming control bank D to be fully inserted and banks C and B at their insertion limit. The results are 930 pcm and 18.0, respectively. The peak cladding average and fuel center temperatures are 2995°F and 4075°F, respectively.

A summary of the cases presented above is given in Table 14.2.6-1. The nuclear power and hot spot fuel and cladding temperature transients are presented in Figure 14.2.6-1 through Figure 14.2.6-4.

For all cases, reactor trip occurs very early in the transient, after which the nuclear power excursion is terminated. As discussed previously, the reactor will remain subcritical following reactor trip.

The ejection of an RCCA constitutes a break in the reactor coolant system, located in the reactor pressure vessel head. The effects and consequences of loss of coolant accidents are discussed in Section 14.3. Following the RCCA ejection, the operator would follow the same emergency instructions as for any other loss of coolant accident to recover from the event.

Rods-in-DNB

It is assumed that fission products are released from the gaps of all rods entering DNB. In all cases considered, less than 10% of the rods entered DNB based on a detailed three-dimensional THINC analysis.

Pressure Surge

A detailed calculation of the pressure surge for an ejection worth of one dollar at beginning of life, hot full power, indicates that the peak pressure does not exceed that which would cause stress to exceed the faulted condition stress limits. Since the severity of the present analysis does not exceed the "worst case" analysis, the accident for this plant will not result in an excessive pressure rise or further damage to the reactor coolant system.

Lattice Deformations

A large temperature gradient will exist in the region of the hot spot. Since the fuel rods are free to move in the vertical direction, differential expansion between separate rods cannot produce distortion. However, the temperature gradients across individual rods may produce a differential expansion tending to bow the midpoint of the rods toward the hotter side of the rod. Calculations have indicated that this bowing would result in a negative reactivity effect at the hot spot since

Westinghouse cores are undermoderated, and bowing will tend to increase the undermoderation at the hot spot. Since the 14 x 14 fuel design is also undermoderated, the same effect would be observed. In practice, no significant bowing is anticipated, since the structural rigidity of the core is more than sufficient to withstand the forces produced. Boiling in the hot spot region would produce a net flow away from that region. However, the heat from the fuel is released to the water relatively slowly, and it is considered inconceivable that crossflow will be sufficient to produce significant lattice forces. Even if massive and rapid boiling, sufficient to distort the lattice, is hypothetically postulated, the large void fraction in the hot spot region would produce a reduction in this ratio at the hot spot. The net effect would therefore be a negative feedback. It can be concluded that no conceivable mechanism exists for a net positive feedback resulting from lattice deformation. In fact, a small negative feedback may result. The effect is conservatively ignored in the analysis.

Conclusions

Conservative analyses indicate that the described fuel and cladding limits are not exceeded. It is concluded that there is no danger of sudden fuel dispersal into the coolant. Since the peak pressure does not exceed that which would cause stresses to exceed the faulted condition stress limits, it is concluded that there is no danger of further consequential damage to the reactor coolant system. The analyses have demonstrated that the number of fuel rods entering DNB is limited to less than 10% of the fuel rods in the core.

Radiological Consequences of a Rod Ejection Accident

This section presents an evaluation of the offsite consequences of a control rod ejection (CRE) accident. The analysis of the CRE radiological consequences uses the analytical methods and assumptions outlined in the RG 1.183 (Reference 7).

Following the accident, two release paths contribute to the total radiological consequences of the accident. The first is the leakage of radioactivity from the containment atmosphere to the environment and the second is the leakage of radioactivity from the secondary system through the steam generator relief valves. The radioactivity in the containment atmosphere is due to the radioactivity in the primary system coolant that has spilled out of the primary system into the containment through the hole in the reactor head created by the rod ejection. The radioactivity in the secondary system is due to the radioactivity in the primary system coolant that has leaked into the secondary system prior to the accident and also to the radioactivity that is transported to the secondary system by the primary system coolant that leaks through the steam generator tubes during the accident. Steam is released from the steam generator for heat removal purposes because condenser cooling is lost due to the assumed coincident loss of offsite power during the accident.

The major assumptions and parameters used in the CRE dose analysis are itemized in Table 14.2.6-2. Other assumptions for this dose analysis are presented in Table 14.1.8-3 and Table 14.1.8-6.

The concentrations of iodines and noble gasses in the RCS at the time the accident occurs are assumed to be 0.5 µCi/gm of dose equivalent (DE) I-131 and 520 µCi/gm of DE Xe-133. The alkali metal concentration in the RCS is based on the fuel defect level that corresponds to 0.5 µCi/gm DE I-131. The iodine activity concentration of the secondary coolant at the time the accident

occurs is assumed to be equivalent to the Technical Specification limit of 0.1 µCi/gm of DE I-131. The alkali metal activity concentration of the secondary coolant at the time the accident occurs is assumed to correspond to 0.1 µCi/gm of DE I-131. The equilibrium nuclide concentrations are presented in Table 14.1.8-4.

Core Release Model

The core activity is presented in Table 14.1.8-4. The quantity of radioactivity released from the reactor core either to the primary system or to the containment atmosphere during the accident was conservatively calculated using the following assumptions:

1. Ten percent of the fuel rods in the reactor core are assumed to suffer sufficient damage (DNB), as a result of the CRE, such that all of their gap activity is released to the RCS. Ten percent of the core iodine and noble gases and twelve percent of the core alkali metals are assumed to be in the fuel clad gap.

2. One quarter of one percent (0.25%) of the fuel in the reactor core suffers fuel melt. The fraction of melted fuel activity released to containment or the RCS is 100% for noble gases and 50% for iodines and alkali metals. The fuel melt fraction was determined using the following assumptions:

    a. Fifty percent of the fuel rods experiencing clad damage may also experience fuel melting at the centerline of the fuel rod

    b. Centerline fuel melting is limited to the inner 10% of the fuel

    c. Melting occurs over fifty percent of the axial length of the fuel rod

3. The activity releases from the damaged/melted fuel reflect the maximum radial peaking factor of 1.7.

Containment Release Pathway

The model for this release pathway assumes that all of the radioactivity initially present in the primary system and the radioactivity introduced by the fuel rod cladding failures and the melted fuel is instantaneously and homogeneously mixed throughout the net free volume of the containment atmosphere at the time of the accident. No credit is taken for plate out onto containment surfaces or for containment spray operation, which would remove airborne particulates and elemental iodine. The only removal processes considered are sedimentation of particulates, radioactive decay and leakage.

The containment is assumed to leak at the design leak rate of 0.2 weight percent per day for the first 24 hours of the accident and then to leak at half that rate (0.1 weight percent per day) for the remainder of the 30 day period following the accident considered in the analysis.

Primary-to-Secondary Leakage Release Pathway

When determining doses due to the primary-to-secondary SG tube leakage, all the iodine, alkali metals 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). An accident-induced primary-to-secondary leak rate of 1000 gm/min per SG is assumed. Although the

primary-to-secondary pressure differential drops throughout the event, the constant flow rate is conservatively maintained. The primary-to-secondary tube leakage continues until the RCS pressure drops below the secondary pressure. A conservative time of 0.556 hours was used for this analysis. Steam releases from the SGs are conservatively assumed to continue for 30 hours.

An iodine partition factor in the SGs of 0.01 (curies iodine/gm steam) / (curies iodine/gm water) and a particulate retention factor of 0.0025 are used. All noble gas activity, transferred to the secondary side of the SG through SG tube leakage, is assumed to be directly released to the outside atmosphere.

Acceptance Criteria

The Standard Review Plan (SRP) 15.0.1 (Reference 8) offsite dose acceptance criterion for a CRE accident is 6.3 rem TEDE, which is approximately 25% of the 10 CFR 50.67 limit of 25 rem TEDE. The control room personnel dose acceptance criterion is 5 rem TEDE per 10 CFR 50.67.

Results and Conclusions

The results of the offsite and control room dose analyses are provided in Table 14.2.6-3, and indicate that the acceptance criteria are met. The exclusion area boundary doses reported are for the worst 2 hour period, determined to be from 0 to 2 hours.

References

1.   Tong, "Post DNB Heat Transfer," WCAP 7247

2.   Redfield, J.A., "CHIC-KIN -- A Fortran Program for Intermediate and Fast Transients in a Water Moderated Reactor," WAPD-TM-479, January, 1965

3.   Barry, R. F., "The Revised LEOPARD Code - A Spectrum Dependent Non Spatial Depletion Program," WCAP-2759 (1965)

4.   "Power Distribution Control of Westinghouse PWR" WCAP 7208 (1968)

5.   Conway and Hein, Journal of Nuclear Materials (15.1), 1965

6.   Ogard & Leary, "High Temperature Heat Content and Heat Capacity of Uranium Dioxide - Plutonium Dioxide Solid Solutions," LA-DC-8620

7.   USNRC, Regulatory Guide 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," July 2000.

8.   Standard Review Plan (SRP) Section 15.0.1, "Radiological Consequence Analyses Using Alternative Source Terms," July 2000.

Table 14.2.6-1   PARAMETERS USED IN THE ANALYSIS OF THE ROD CLUSTER
CONTROL ASSEMBLY EJECTION ACCIDENT

Parameters	BOL-HZP	BOL-HFP	EOL-HZP	EOL-HFP
Initial core power level, percent of 1800 MWt	0%	102%	0%	102%
Ejected rod worth, pcm	790	400	930	420
Delayed neutron fraction	0.0049	0.0049	0.0043	0.0043
Doppler reactivity defect (absolute value), pcm	1000	1000	980	945
Doppler feedback reactivity weighting	2.008	1.139	2.704	1.316
Trip reactivity, percent ΔK	2.0	4.0	2.0	4.0
F_q before rod ejection	N/A	2.6	N/A	2.6
F_q after rod ejection	11.0	4.2	18.0	5.69
Number of operational pumps	1	2	1	2
Maximum fuel pellet average temperature, °F	3538	3995	3742	4041
Maximum fuel center temperature, °F	3959	>4900	4075	>4800
Maximum cladding average temperature, °F	2676	2142	2995	2168
Maximum fuel stored energy, cal/gm	150.5	174.1	161.0	176.4
Maximum fuel melt, %	nil	5.6	nil	9.8

Table 14.2.6-2   ASSUMPTIONS USED FOR CONTROL ROD EJECTION ACCIDENT
ANALYSIS

PARAMETER	VALUE
Initial Power	1811 MWt
Fraction of Fuel Rods in Core Assumed to Fail	10% of Core
Fraction of Fuel Rods in Core Assumed to Melt	0.25% of Core
Gap Fractions	
Iodines and Noble Gases	0.10
Alkali Metals	0.12
Fraction of Activity Released from Melted Fuel	
Iodines and Alkali Metals	0.5
Noble Gases	1.0
Radial Peaking Factor	1.7
RCS Activity Prior to Accident	
Iodine	0.5 µCi/gm of DE I-131
Noble Gas	520 µCi/gm of DE Xe-133
Alkali Metals	Corresponds to 0.5 µCi/gm of DE I-131
Secondary Coolant Activity Prior to Accident	
Iodine	0.1 µCi/gm of DE I-131
Alkali Metals	Corresponds to 0.1 µCi/gm of DE I-131
Containment Leak Rate	
0 - 24 hours	0.2 weight %/day
> 24 hours	0.1 weight %/day
Iodine Chemical Form in Containment	
Elemental	4.85%
Organic	0.15%
Particulate	95%
Spray Removal in Containment	Not Credited
Sedimentation Removal Credit	
Iodines	Not credited
Alkali Metals	0.1 hr^{-1}
Total SG Tube Leak Rate	
0 - 0.556 hours	2000 gm/min
> 0.556 hours	0.0 gm/min
Steam Release to Environment	
0 - 2 hours	8.063E5 gm/min
2 - 14 hours	4.530E5 gm/min
14 - 30 hours	2.651E5 gm/min
SG Iodine Partition Factor	0.01
SG Alkali Metal Retention Factor	0.0025
Iodine Species Released to the Atmosphere from SGs	
Elemental	97%
Organic	3%

Table 14.2.6-3   DOSES DUE TO THE RADIOACTIVITY RELEASED DURING THE
CONTROL ROD EJECTION ACCIDENT

Site Boundary (0 - 2 hr)	2.3 rem TEDE
Low Population Zone (0 - 30 days)	0.8 rem TEDE
Control Room (0 - 30 days)	2.9 rem TEDE

Figure 14.2.6-1 RCCA EJECTION TRANSIENT BEGINNING OF LIFE ZERO POWER
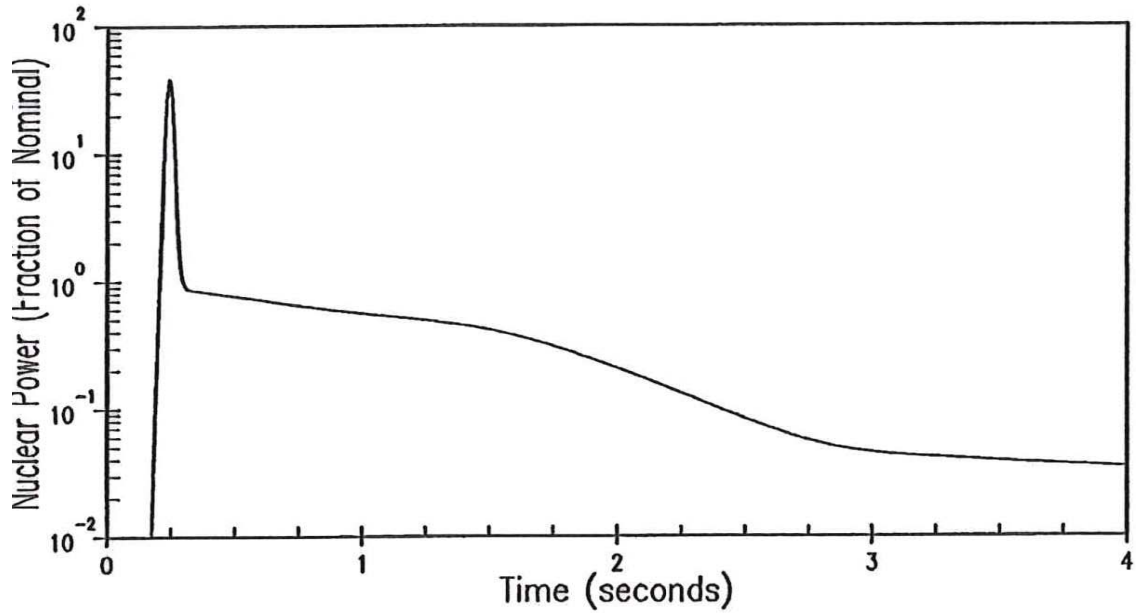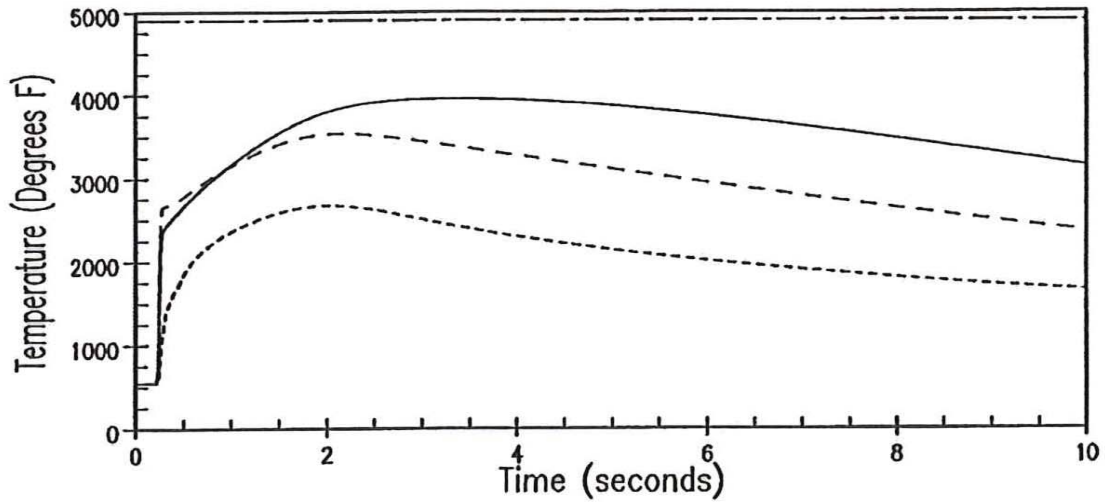
Figure 14.2.6-2 RCCA EJECTION TRANSIENT BEGINNING OF LIFE FULL POWER
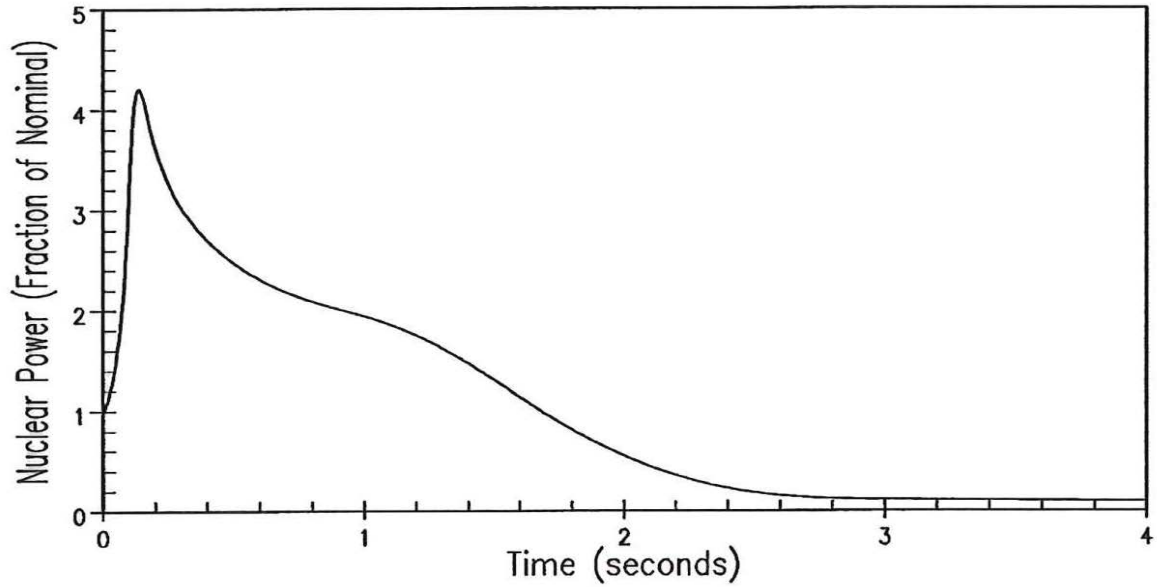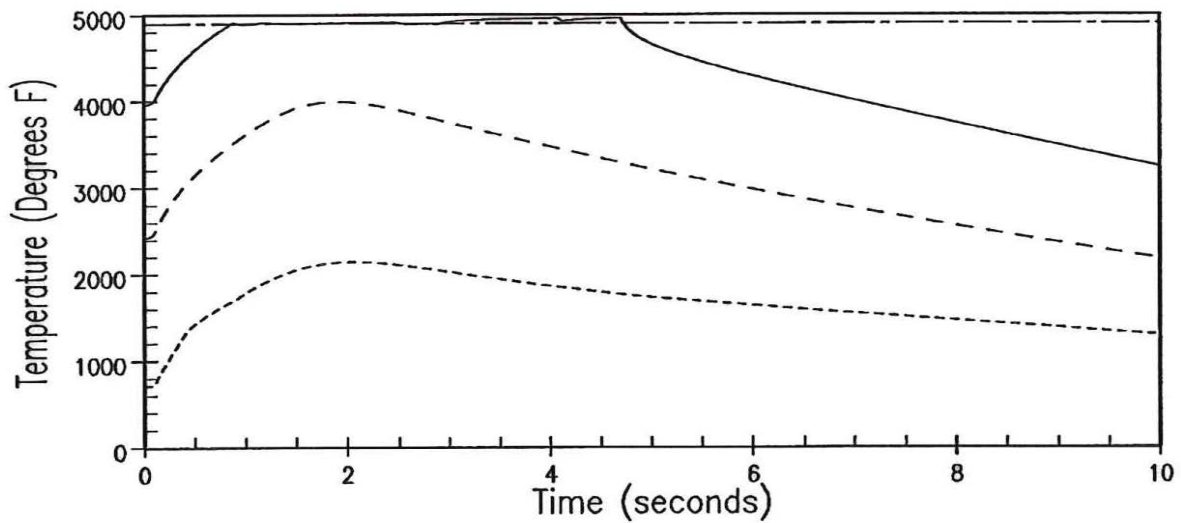
Figure 14.2.6-3 RCCA EJECTION TRANSIENT END OF LIFE ZERO POWER
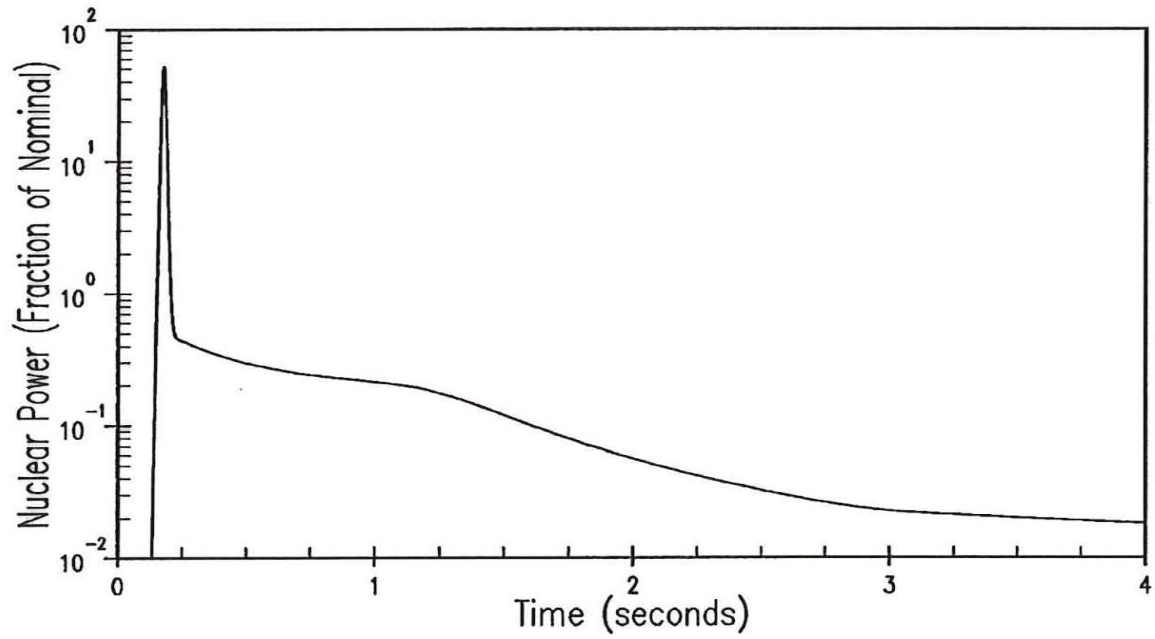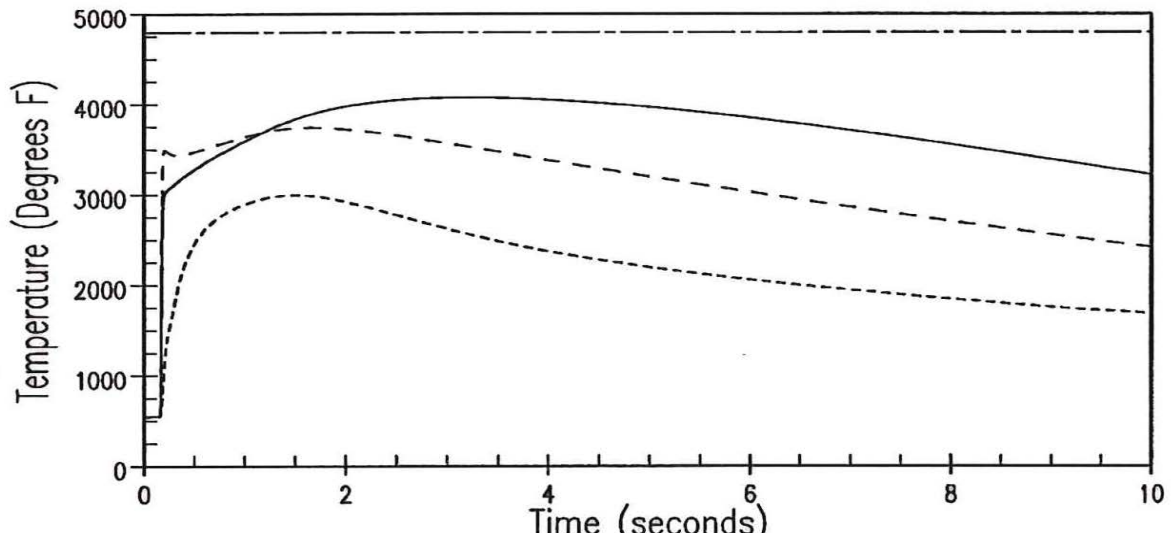
Figure 14.2.6-4 RCCA EJECTION TRANSIENT END OF LIFE FULL POWER
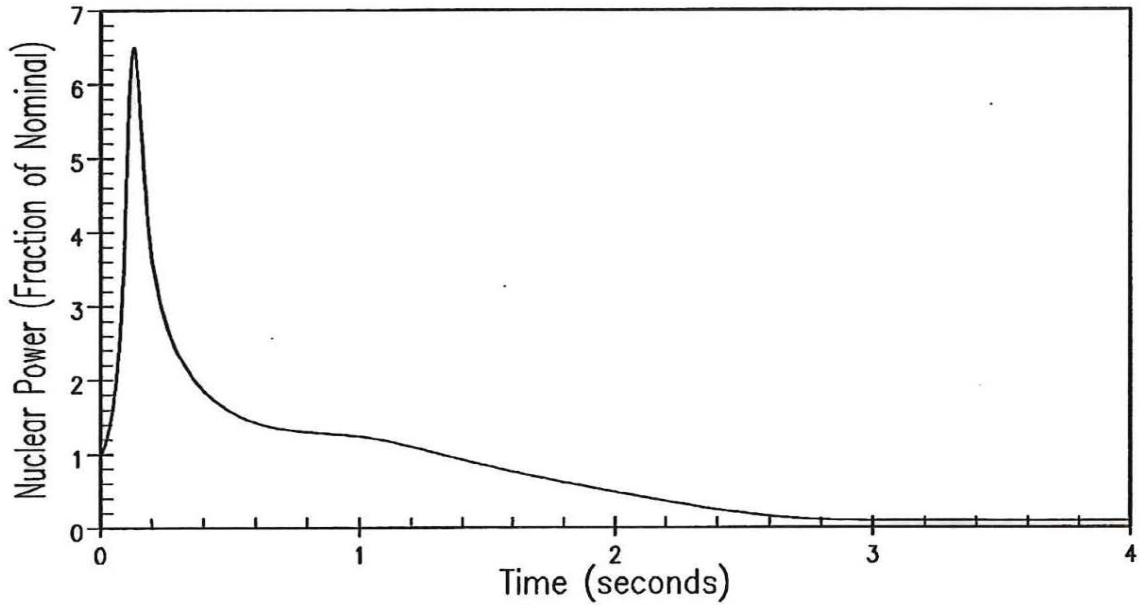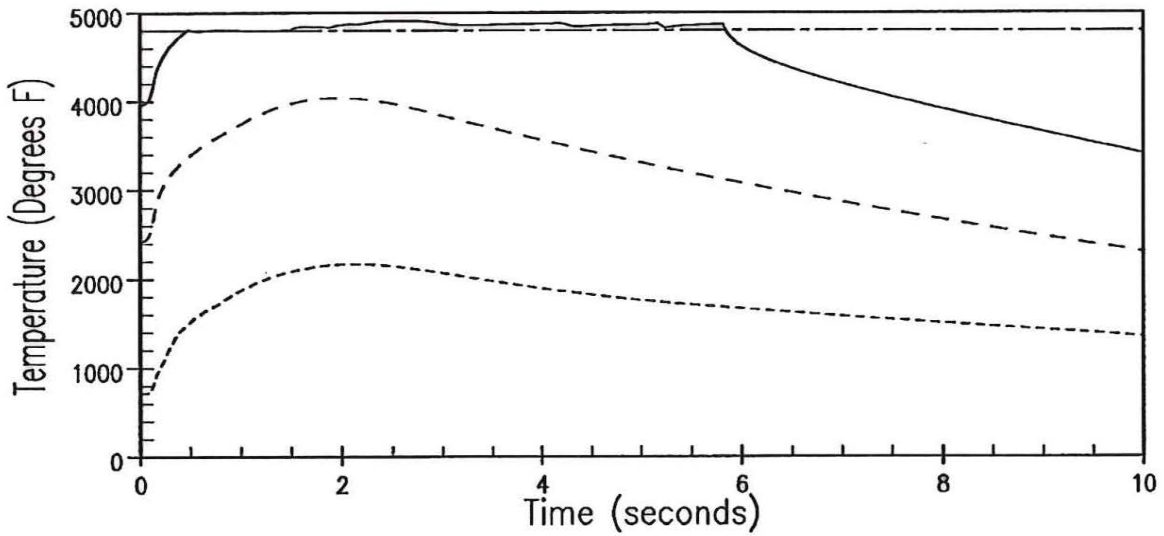
14.2.7 INADVERTENT OPENING OF A STEAM GENERATOR (SG) RELIEF OR SAFETY VALVE

The inadvertent opening of a steam generator (SG) relief or safety valve event is classified as an anticipated operational Occurrence (AOO) and considered to be an American Nuclear Society (ANS) Condition II event. The cooldown effects and transient results from an inadvertent opening of a SG relief or safety valve have been shown to be less severe than those for a hot zero power hypothetical steam line break (i.e., the double-ended rupture) (Reference 1 and Reference 2). The latter event, analyzed in the FSAR Section 14.2.5, is considered to be an ANS Condition IV event. The peak heat flux in the case of inadvertent opening of a SG relief or safety valve would be much less than that of the double-ended steam line break event due to the lower steam release rate.

The steam line break event is also analyzed in the FSAR Section 14.2.5 from hot full power conditions for a range of break sizes up to 1.4 ft^2, which would bound the inadvertent opening of a SG relief or safety valve.

The inadvertent opening of a SG relief or safety valve event is thus bounded by the limiting hot full power and hot zero power steam line break events described in the FSAR Section 14.2.5. Since the steam line break events in the FSAR Section 14.2.5 are analyzed to the same A00 acceptance criteria, the inadvertent opening of a SG relief or safety valve event is not explicitly analyzed for PBNP. The limiting steam line break accident described in FSAR Section 14.2.5, Rupture of A Steam Pipe, demonstrates that the DNBR and kW/ft limits are met.

Conclusions

The inadvertent opening of a steam generator relief or safety valve is less severe than that of a steam line break event (see Section 14.2.5). Based on results presented in [1], the applicable acceptance criteria for the inadvertent opening of a steam generator relief or safety valve have been met.

REFERENCES

1. NRC Safety Evaluation, PBNP Units 1 and 2 - Issuance of License Amendments Regarding Extended Power Uprate, May 3, 2011.

2. WCAP-12602, "Report for the Reduction of SI System Boron Concentration," September 1990.

1.

14.3   PRIMARY SYSTEM PIPE RUPTURES

14.3.1   SMALL BREAK LOSS-OF-COOLANT ACCIDENT ANALYSIS

Identification of Causes and Accident Description

A loss of coolant accident is defined as a rupture of the reactor coolant system piping or of any line connected to the system up to the first closed valve. Ruptures of small cross section will cause loss of the coolant at a rate which can be accommodated by the charging pumps which would maintain an operational water level in the pressurizer permitting the operator to execute an orderly shutdown. A moderate quantity of coolant containing such radioactive impurities as would normally be present in the coolant, would be released to the containment.

The maximum break size for which the normal makeup system can maintain the pressurizer level is obtained by comparing the calculated flow from the reactor coolant system through the postulated break against the charging pump makeup flow at normal reactor coolant system pressure, i.e., 2250 psia. A makeup flow rate from two charging pumps is typically adequate to maintain pressurizer level long enough for the operator to respond without activating the ECCS for a break through a 3/8 inch diameter hole.

Should a larger break occur, depressurization of the reactor coolant system causes fluid to flow to the reactor coolant system from the pressurizer resulting in a pressure and level decrease in the pressurizer. Reactor trip occurs when the pressurizer low pressure trip setpoint is reached. The consequences of the accident are limited in two ways:

1.   Reactor trip and borated water injection complement void formation in causing rapid reduction of nuclear power to a residual level corresponding to the delayed fission and fission product decay.

2.   Injection of borated water ensures sufficient flooding of the core to prevent excessive cladding temperatures.

Before the break occurs, the plant is in an equilibrium condition, i.e., the heat generated in the core is being removed via the secondary system. During blowdown, heat from decay, hot internals and the vessel continues to be transferred to the reactor coolant system. The heat transfer between the reactor coolant system and the secondary system may be in either direction depending on the relative temperatures. In the case of continued heat addition to the secondary, system pressure increases and steam dumping may occur. The safety injection signal stops normal feedwater flow by closing the main feedwater line isolation valves and initiates emergency feedwater flow by starting Auxiliary Feedwater (AFW) pumps. Although the AFW System may be initiated during the Small Break LOCA, the event has been analyzed with no credit for auxiliary feedwater. The designated motor-driven and turbine-driven AFW pumps would automatically start as a result of a Safety Injection signal and may start as a result of 4.16KV bus undervoltage or steam generator low-low levels in both steam generators of the accident unit. However, the event was analyzed without AFW due to asymmetries and limit the modeling required to address all possible combinations and time-delays of AFW System configurations. The secondary flow aids in the reduction of reactor coolant system pressure. When the RCS depressurizes to 695 psia, the accumulators begin to inject water into the reactor coolant loops. The reactor coolant pumps are assumed to be tripped at the initiation of the accident and effects of pump coastdown are included in the blowdown analyses.

Analysis of Effects and Consequences - Method of Analysis

For small breaks less than 1.0 ft^2 the NOTRUMP Evaluation Model (Reference 1, Reference 2 and Reference 4) is employed to calculate the transient depressurization of the reactor coolant system as well as to describe the mass and enthalpy of flow through the break and the subsequent rod heat-up.

Small Break LOCA Analysis Using NOTRUMP

The NOTRUMP and LOCTA-IV (Reference 1 and Reference 3) computer codes are used in the analysis of loss-of-coolant accidents due to small breaks in the Reactor Coolant System. The NOTRUMP computer code is a one-dimensional general network code consisting of a number of advanced features. Among these features are the calculation of thermal non-equilibrium in all fluid volumes, flow regime-dependent drift flux calculations with counter-current flow limitations, mixture level tracking logic in multiple-stacked fluid nodes, and regime-dependent heat transfer correlations. Safety injection into the broken loop is modeled along with the COSI condensation model (Reference 4). The NOTRUMP small break LOCA emergency core cooling system (ECCS) evaluation model was developed to determine the RCS response to design basis small break LOCAs and to address the NRC concerns expressed in NUREG-0611, "Generic Evaluation of Feedwater Transients and Small Break Loss-of-Coolant Accidents in Westinghouse Designed Operating Plants."

The reactor coolant system is nodalized into volumes interconnected by flowpaths. Both the broken and intact loops are modeled explicitly. The transient behavior of the system is determined from the governing conservation equations of mass, energy, and momentum applied throughout the system. A detailed description of NOTRUMP is given in Reference 1, Reference 2 and Reference 4.

The use of NOTRUMP in the analysis involves, among other things, the representation of the reactor core as heated control volumes with the associated bubble rise model to permit a transient mixture height calculation. The multi-node capability of the program enables an explicit and detailed spatial representation of various system components. In particular, it enables a proper calculation of the behavior of the loop seal during a loss-of-coolant transient.

Peak cladding temperature analyses are performed with the LOCTA IV code. Input for the code is obtained from the NOTRUMP calculations which determine the RCS pressure, fuel rod power history, steam flow past the uncovered part of the core, core inlet enthalpy, and mixture height history.

Table 14.3.1-1 lists important input parameters and initial conditions used in the analysis. Major assumptions included a total peaking factor of 2.60, F$_{\Delta H}$ of 1.68, 10% steam generator tube plugging, thermal design flow of 89,000 gpm/loop and 100.6% of a core thermal power of 1800 MWt. Note: The Small Break LOCA analysis was performed with ZIRLO$^{\circledR}$ cladding. However, Reference 7 concluded that the LOCA ZIRLO models are acceptable for application to Optimized ZIRLO^{TM} cladding in the Small Break analysis, and that no additional calculations are necessary for evaluating the use of Optimized ZIRLO^{TM} cladding provided that plant specific ZIRLO claculations were previously performed.

Safety injection flow rate to the reactor coolant system as a function of the system pressure is used as part of the input. The safety injection (SI) system is assumed to be delivering to the RCS 28 seconds after the generation of a safety injection signal. For this analysis, the ECCS delivery considers flow which is depicted in Figure 14.3.1-2 through Figure 14.3.1-3A as a function of RCS pressure; these figures represent injection flow from the HHSI and LHSI pumps based on Reference 5. The SI flows are assured by the plant in-service testing acceptance criteria. The 28 second delay includes time required for diesel startup and loading of the safety injection pumps onto the emergency buses. Also minimum Emergency Core Cooling System capability and operability has been assumed in these analyses.

The data used to generate Figure 14.3.1-1 through Figure 14.3.1-3A are provided in Table 14.3.1-2A through Table 14.3.1-2C, respectively.

Table 14.3.1-2A provides the broken and intact loop high head safety injection (HHSI) flows used for breaks less than the accumulator line inner diameter (8.75-inches). The faulted loop "Spills to RCS Pressure" when the assumed backpressure is the reactor coolant system, (RCS). Since the HHSI injects into the accumulator line, and the fault size is less than the inner diameter of the accumulator line, both the broken and intact loop HHSI flows inject ("spill") to RCS pressure.

Conversely, Table 14.3.1-2B provides the broken and intact loop HHSI flows for breaks greater than or equal to the accumulator line inner diameter. For these break cases, the faulted loop HHSI flow will inject ("spill") directly into containment. While the faulted loop does spill to containment pressure, the spilling rate for these breaks is a function of RCS pressure due to the communicating intact loop HHSI branch line.

Table 14.3.1-2C provides the low head safety injection (LHSI) flows. Since the LHSI injects directly into the upper plenum, and no fault is assumed on this line, these flows always inject to RCS pressure.

The reactor scram time is equal to the reactor trip signal time plus 4.2 seconds for signal transmission and rod insertion. During this period, the reactor is conservatively assumed to operate at 100.6% of 1800 MWt.

Figure 14.3.1-1 presents the axial power shape utilized to perform the small break analysis presented here. This power shape was chosen because it provides a conservative distribution of power versus core height by maximizing the local power in the upper regions of the reactor core, while minimizing the power in the lower regions of the core. This is limiting for small break analysis because of the uncovery process. As the core uncovers, the cladding in the upper elevation of the core heats up and is sensitive to the linear power at that elevation. The cladding temperatures in the lower elevations of the core, below the two phase mixture height, remains low reducing the amount of mixture level swell, thus providing a deeper core uncovery. The peak cladding temperatures occur above 10 ft.

Results of Small Break Analysis

This section presents results of the Point Beach Units 1 and 2 break spectrum. The analysis techniques for this evaluation allow for the Point Beach Units to operate at vessel average temperatures ranging from 558.0°F to 577.0°F + 6.1/-6.2°F, at a pressure of 2250 psia ± 50 psi. The Units 1 and 2 time sequence of events is summarized in Table 14.3.1-3A and

Table 14.3.1-3B. The rod heatup information is summarized in Table 14.3.1-4A and Table 14.3.1-4B. The depressurization transient for the limiting 3-inch breaks are shown in Figure 14.3.1-4 and Figure 14.3.1-14 for Units 1 and 2 , respectively. The extent to which the core is uncovered for the limiting breaks are shown in Figure 14.3.1-5 and Figure 14.3.1-15 for Units 1 and 2 , respectively.

During the early part of the small break LOCA transient positive core flow is maintained by the reactor coolant pump coastdown, overcoming any potential for the cold leg break to induce negative flow or flow stagnation in the core. The resultant heat transfer cools the fuel rod and cladding to very near the coolant temperatures as long as the core remains covered by a two phase mixture.

The maximum hot rod peak cladding temperatures calculated during the transient are 1049 and 1103°F for Units 1 and 2 , respectively. The limiting hot rod peak cladding temperature transients are shown in Figure 14.3.1-12 and Figure 14.3.1-22 for Units 1 and 2 , respectively. The calculated PCT may vary for each core reload analysis and is limited by federal regulations (10 CFR 50.46) to a maximum temperature of 2200°F for this event. The vapor mass flow rate for the limiting breaks is shown in Figure 14.3.1-8 and Figure 14.3.1-18. When the mixture level drops below the top of the core, the steam flow computed in NOTRUMP provides cooling to the upper portion of the core. The cladding surface heat transfer coefficients for this phase of the transient are given in Figure 14.3.1-10 and Figure 14.3.1-20. The fluid temperature at the PCT elevation are shown in Figure 14.3.1-11 and Figure 14.3.1-21.

Additional Break Sizes

Additional break sizes were analyzed to identify the limiting break size, including 1.5, 2, 4, 6, and 8.75 inch breaks. Figure 14.3.1-24 through Figure 14.3.1-65 show the RCS Pressure, Core Mixture Level, Core Exit Vapor Temperature, Peak Cladding Temperature, and Maximum Local Oxidation for each break size. The 6 and 8.75 inch breaks do not have rod heat up data since there was no core uncovery for those break sizes.

Conclusions

Analyses presented in this section show that the emergency core cooling system, together with accumulators, provide sufficient core flooding to keep the calculated peak cladding temperatures below required limits of 10 CFR 50.46. Hence, adequate protection is afforded by the emergency core cooling system in the event of a small break loss-of-coolant accident.

References

1.   Lee, N., et. al., "Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code," WCAP-10054-P-A (Proprietary) and WCAP-10081-A (Non-Proprietary), August 1985.

2.   Meyer, P. E., "NOTRUMP, A Nodal Transient Small Break and General Network Code," WCAP-10079-P-A (Proprietary) and WCAP-10080-A (Non-Proprietary), August 1985.

3.   Bordelon, F. M., et. al., "LOCTA-IV Program: Loss-of-Coolant Transient Analysis," WCAP-8301 (Proprietary), WCAP-8305 (Non-Proprietary), June 1974.

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

5.   Calculation 2006-0021-000-A, "ECCS System Accident Analysis Flow Inputs," dated
       April 5, 2008.

6.   Calculation CN-LIS-08-15 Revision 1, "Point Beach Units 1 and 2 (WEP/WIS) Extended
       Power Uprate (EPU) Small Break Loca (SBLOCA) Analysis," dated September 23, 2008.

7.   WCAP-12610-P-A and CENPD-404-P-A Addendum 1-A, "Optimized ZIRLO^{TM},
       July 2006.

Table 14.3.1-1   INPUT ASSUMPTIONS USED IN THE SMALL BREAK ANALYSIS

Input Parameters Used in the Small Break LOCA Analysis

100% Licensed Core Power	1800 MWt
Calorimetric Uncertainty	0.6%
Peak Hot Rod Linear Power	17.689 kW/ft
Fuel Type	14x14, 422V+ with ZIRLO$^{®}$ Cladding [2]
Total Core Peaking Factor, F_Q	2.6
Hot Channel Enthalpy Rise Factor, $F_{\Delta H}$	1.68
Hot Assembly Peaking factor, P_{HA}	1.62
Thermal Design Flow	89,000 gpm/loop
Nominal Vessel Average Temperature Range	558 - 557°F
Reactor Coolant Pressure (including uncertainties)	2300 psia
Accumulator Water Volume	1118 ft^3
Accumulator Gas Pressure (minimum, including uncertainties)	695 psia
Minimum AFW Flow Rate per Steam Generator	[1]
Steam Pressure	705 psia (Unit 1) / 726 psia (Unit 2)
Steam Generator Tube Plugging Level	10%

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[1] Since asymmetric AFW flow is not modeled in the standard NOTRUMP evaluation model, the AFW flow is assumed to be 0 gpm.

[2] Optimized ZIRLO^{TM} fuel cladding has been evaluated as an acceptable fuel cladding.

Table 14.3.1-2A HHSI FLOWS WITH THE FAULTED LOOP SPILLING TO RCS PRESSURE

High Head Safety Injection (HHSI) Flows vs. Pressure, Minimum Safeguards, Spill to RCS Pressure (Breaks < 8.75 in. diameter)

Pressure (psia)	Spilled Flow (lbm/s)	Injected Flow (lbm/s)
14.7	65.79	60.40
114.7	63.49	58.29
214.7	61.13	56.12
314.7	58.47	53.68
414.7	55.73	51.16
514.7	52.86	48.52
614.7	49.76	45.67
714.7	46.54	42.72
814.7	43.14	39.60
914.7	39.59	36.33
1014.7	35.38	32.46
1114.7	30.70	28.16
1214.7	25.22	23.10
1314.7	18.32	16.74
1364.7	13.53	12.29

Table 14.3.1-2B HHSI FLOWS WITH THE FAULTED LOOP SPILLING TO
CONTAINMENT PRESSURE

**High Head Safety Injection (HHSI) Flows vs. Pressure, Minimum Safeguards, Spill to 0
psig Containment Pressure (Breaks ≥ 8.75 in. diameter)**

Pressure (psia)	Spilled Flow (lbm/s)	Injected Flow (lbm/s)
14.7	71.09	65.46
114.7	72.75	61.48
214.7	74.45	57.31
314.7	76.10	52.79
414.7	77.81	48.00
514.7	79.61	42.87
614.7	81.53	37.32
714.7	83.56	31.16
814.7	85.78	24.17
914.7	88.33	15.96
934.7	88.90	14.10

Table 14.3.1-2C LHSI FLOWS INJECTING TO RCS PRESSURE

**Low Head Safety Injection (LHSI) Flows vs. Pressure, Minimum Safeguards,
Upper Plenum Injection**

Pressure (psia)	Injecting Flow (lbm/s)
14.7	235.2
24.7	224.8
34.7	214.2
44.7	202.8
54.7	190.8
64.7	178.3
74.7	164.9
84.7	150.7
94.7	133.3
104.7	113.9
114.7	90.9
134.7	0.0

Note:

RHR cut-in pressure is reached only for the 6- and 8.75-inch cases during the RWST injection
phase.

Table 14.3.1-3A TIME SEQUENCE OF EVENTS FOR UNIT 1

NOTRUMP Transient Results for Unit 1

Event (sec)	1.5-inch	2-inch	3-inch	4-inch	6-inch	8.75-inch
Break Initiated	0	0	0	0	0	0
Reactor Trip Signal	153.7	76.1	31.3	19.0	8.5	8.7
Safety Injection Signal	153.7	76.1	31.3	19.0	8.5	8.7
Safety Injection Begins [1]	176.7	99.1	54.3	42.0	31.5	31.7
Loop Seal Clearing Occurs [2]	1037	590	230	125	28	27
Core Uncovery	4115	1130	442	433	N/A[3]	N/A[3]
Accumulator Injection Begins	N/A	3697	690	385	156	154
Core Recovery	5649	2256	1142	450	N/A[3]	N/A[3]
RWST Low Level	2320	2269	2173	2133	1956	1880

(1) Safety Injection is assumed to begin 23.0 s after the Safety Injection Signal (a 5 second SI delay increase was evaluated qualitatively).

(2) Loop seal clearing is assumed to occur when the steam flow through the loop seal in the broken loop is sustained above 1 lbm/s and mixture level is at or below the loop seal elevation. Only the broken loop is allowed to clear for break sizes less than 6-inches in diameter. For the 6- and 8.75-inch breaks, the loop seal in the broken loop clears prior to the intact loop.

(3) There is no core uncovery for the 6- and 8.75-inch breaks.

Table 14.3.1-3B TIME SEQUENCE OF EVENTS FOR UNIT 2

NOTRUMP Transient Results for Unit 2

Event (sec)	1.5-inch	2-inch	3-inch	4-inch	6-inch	8.75-inch
Break Initiated	0	0	0	0	0	0
Reactor Trip Signal	150.6	75.5	31.0	11.8	8.4	8.5
Safety Injection Signal	150.6	75.5	31.0	11.8	8.4	8.5
Safety Injection Begins [1]	173.6	98.5	54.0	34.8	31.4	31.5
Loop Seal Clearing Occurs [2]	1083	553	237	129	28	28
Core Uncovery	4258	1175	335	355	N/A[3]	N/A[3]
Accumulator Injection Begins	N/A	3705	685	366	164	158
Core Recovery	5654	2288	1183	490	N/A[3]	N/A[3]
RWST Low Level	2032	2270	2173	2131	1957	1883

(1) Safety Injection is assumed to begin 23.0 s after the Safety Injection Signal (a 5 second SI delay increase was evaluated qualitatively).

(2) Loop seal clearing is assumed to occur when the steam flow through the loop seal in the broken loop is sustained above 1 lbm/s and mixture level is at or below the loop seal elevation. Only the broken loop is allowed to clear for break sizes less than 6-inches in diameter. For the 6- and 8.75-inch breaks, the loop seal in the broken loop clears prior to the intact loop.

(3) There is no core uncovery for the 6- and 8.75-inch breaks.

Table 14.3.1-4A SBLOCTA BOL RESULTS FOR UNIT 1

Beginning of Life (BOL) Rod Heatup Results for Unit 1

Results	1.5-inch	2-inch	3-inch	4-inch	6-inch	8.75-inch
PCT, °F	678	958	1049	532		
PCT Time, sec	4887	1516	769	445		
PCT Elevation, ft	11.75	10.75	10.75	11.75		
Burst Time[1], sec	N/A	N/A	N/A	N/A		
Burst Elevation[1], ft					N/A[2]	N/A[2]
Maximum Local Transient ZrO_2, %	0.00	0.01	0.01	0.00		
Maximum Local Transient ZrO_2 Elevation, %	11.75	10.75	10.75	11.75		
Average ZrO_2, %	0.00	0.00	0.00	0.00		

(1) Neither the hot rod nor the hot assembly average rod burst during the SBLOCTA calculations.

(2) The core either does not uncover or only uncovers for a very short time; therefore, SBLOCTA calculations are not warranted for 6- and 8.75-inch breaks.

Table 14.3.1-4B SBLOCTA BOL RESULTS FOR UNIT 2

Beginning of Life (BOL) Rod Heatup Results for Unit 2

Results	1.5-inch	2-inch	3-inch	4-inch	6-inch	8.75-inch
PCT, °F	669	955	1103	803		
PCT Time, sec	4943	1530	758	442		
PCT Elevation, ft	11.75	10.75	10.75	11.00		
Burst Time[1], sec	N/A	N/A	N/A	N/A		
Burst Elevation[1], ft					N/A[2]	N/A[2]
Maximum Local Transient ZrO$_2$, %	0.00	0.01	0.02	0.00		
Maximum Local Transient ZrO$_2$ Elevation, %	11.75	10.75	10.75	11.00		
Average ZrO$_2$, %	0.00	0.00	0.00	0.00		

(1) Neither the hot rod nor the hot assembly average rod burst during the SBLOCTA calculations.

(2) The core either does not uncover or only uncovers for a very short time; therefore, SBLOCTA calculations are not warranted for 6- and 8.75-inch breaks.

Figure 14.3.1-1 HOT ROD AXIAL POWER DISTRIBUTION
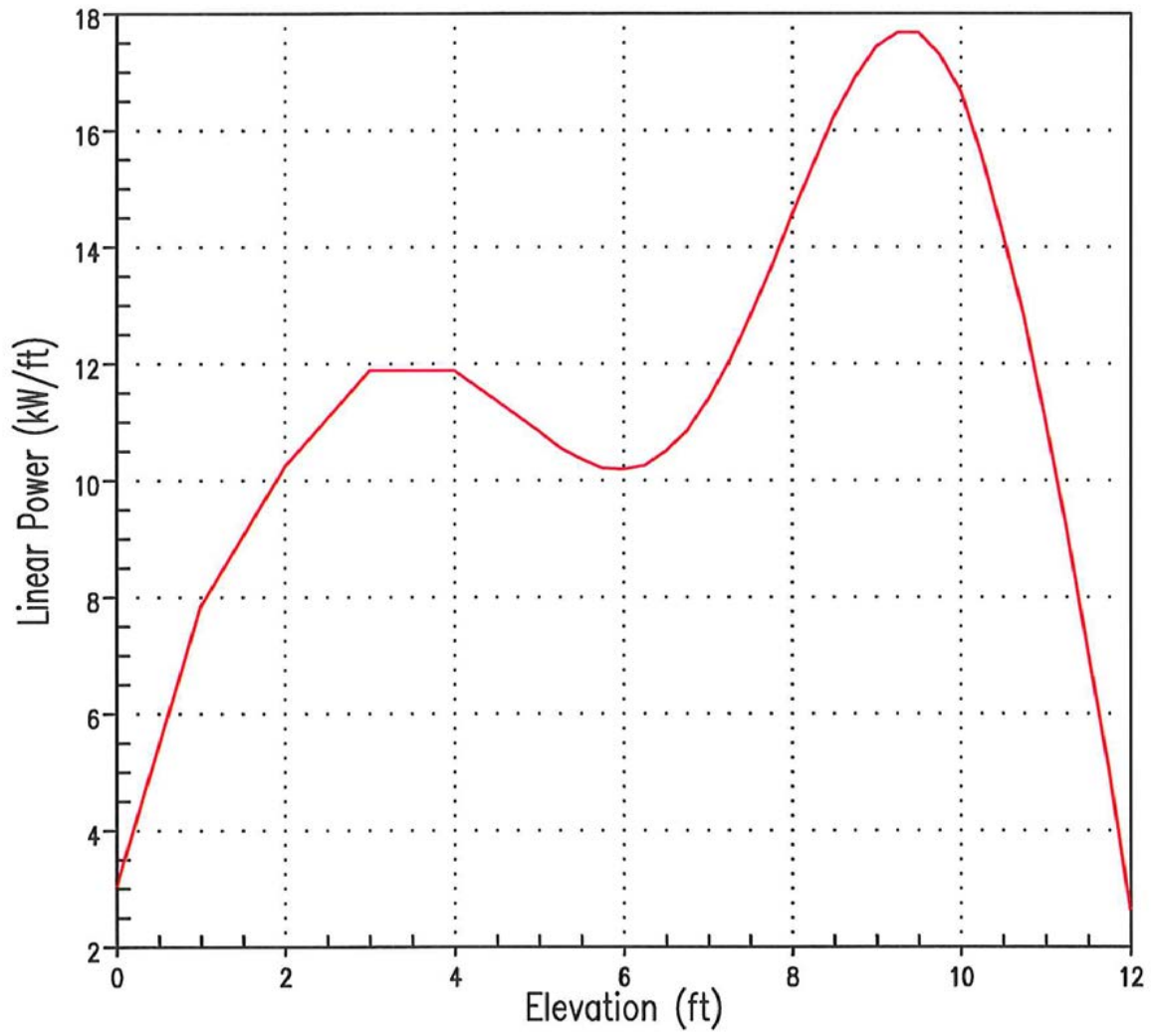
Figure 14.3.1-2 PUMPED HHSI SAFETY INJECTION FLOW RATE FAULTED LOOP
SPILLING TO RCS PRESSURE
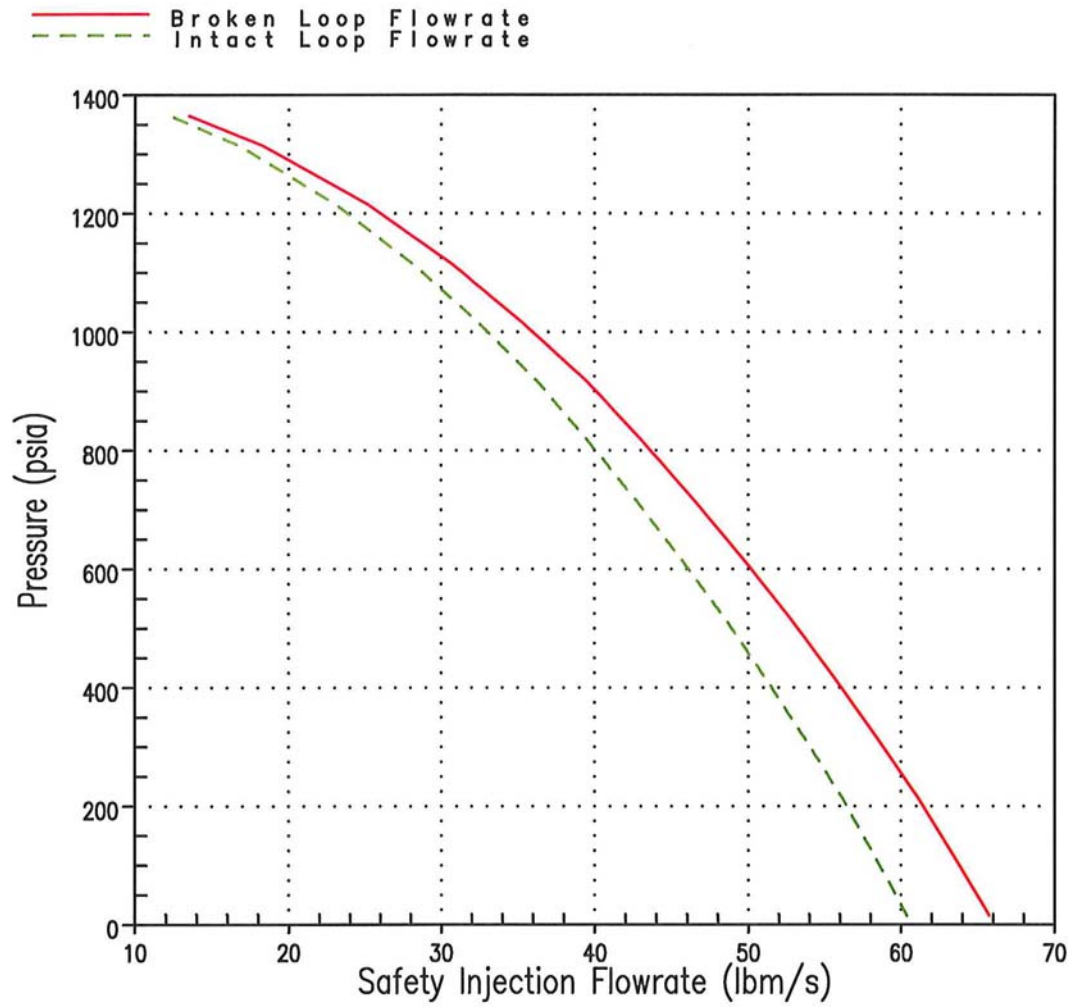
Figure 14.3.1-3 PUMPED HHSI SAFETY INJECTION FLOW RATE FAULTED LOOP
SPILLING TO CONTAINMENT PRESSURE
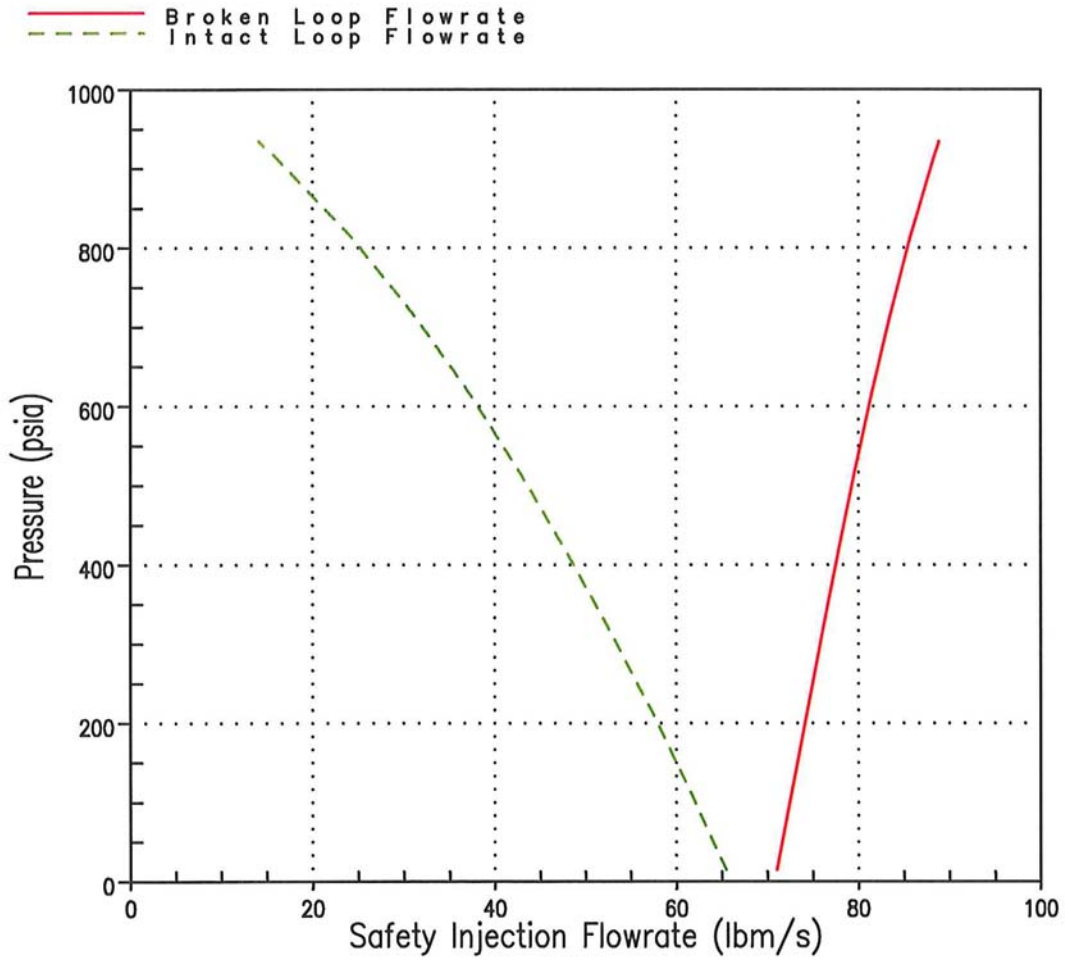
Figure 14.3.1-3A PUMPED LHSI SAFETY INJECTION FLOW RATE UPPER PLENUM
INJECTION
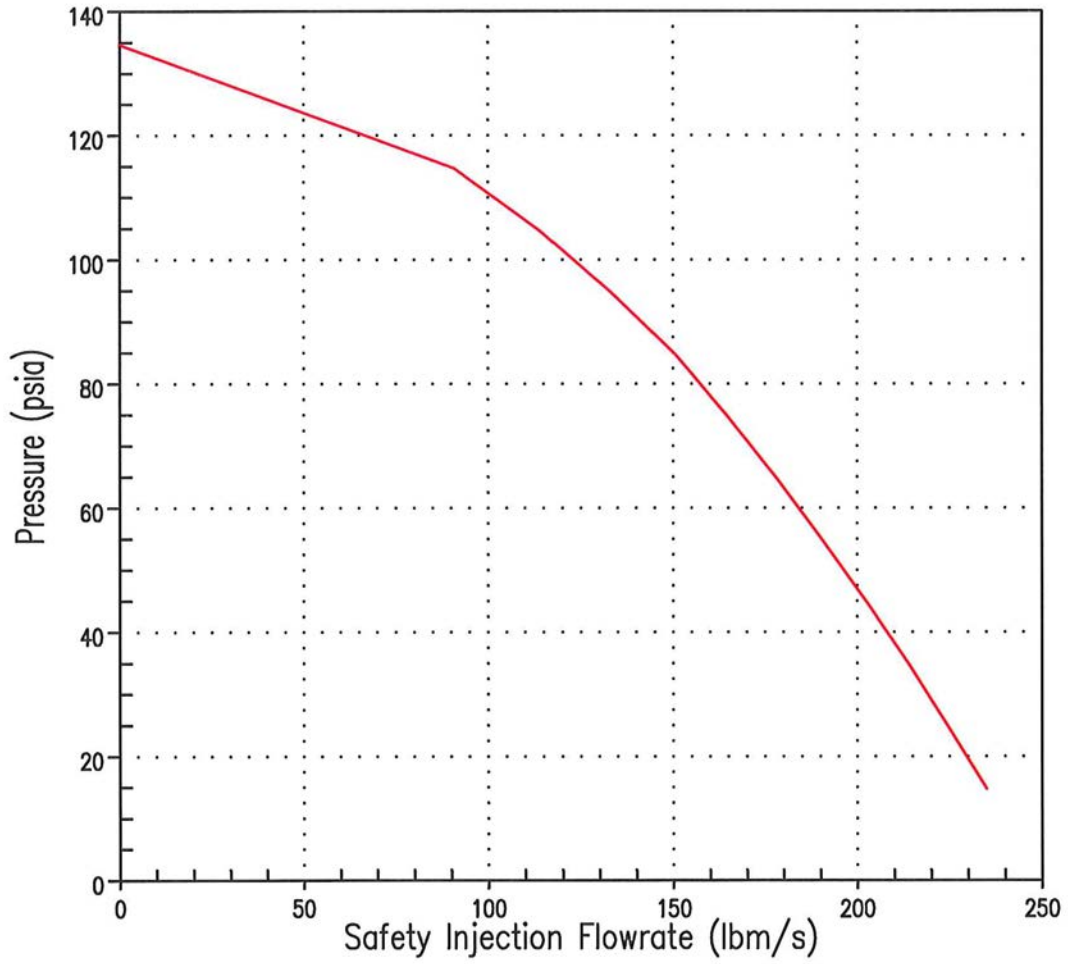
Figure 14.3.1-4   REACTOR COOLANT SYSTEM PRESSURE - 3 INCH BREAK
                  POINT BEACH UNIT 1
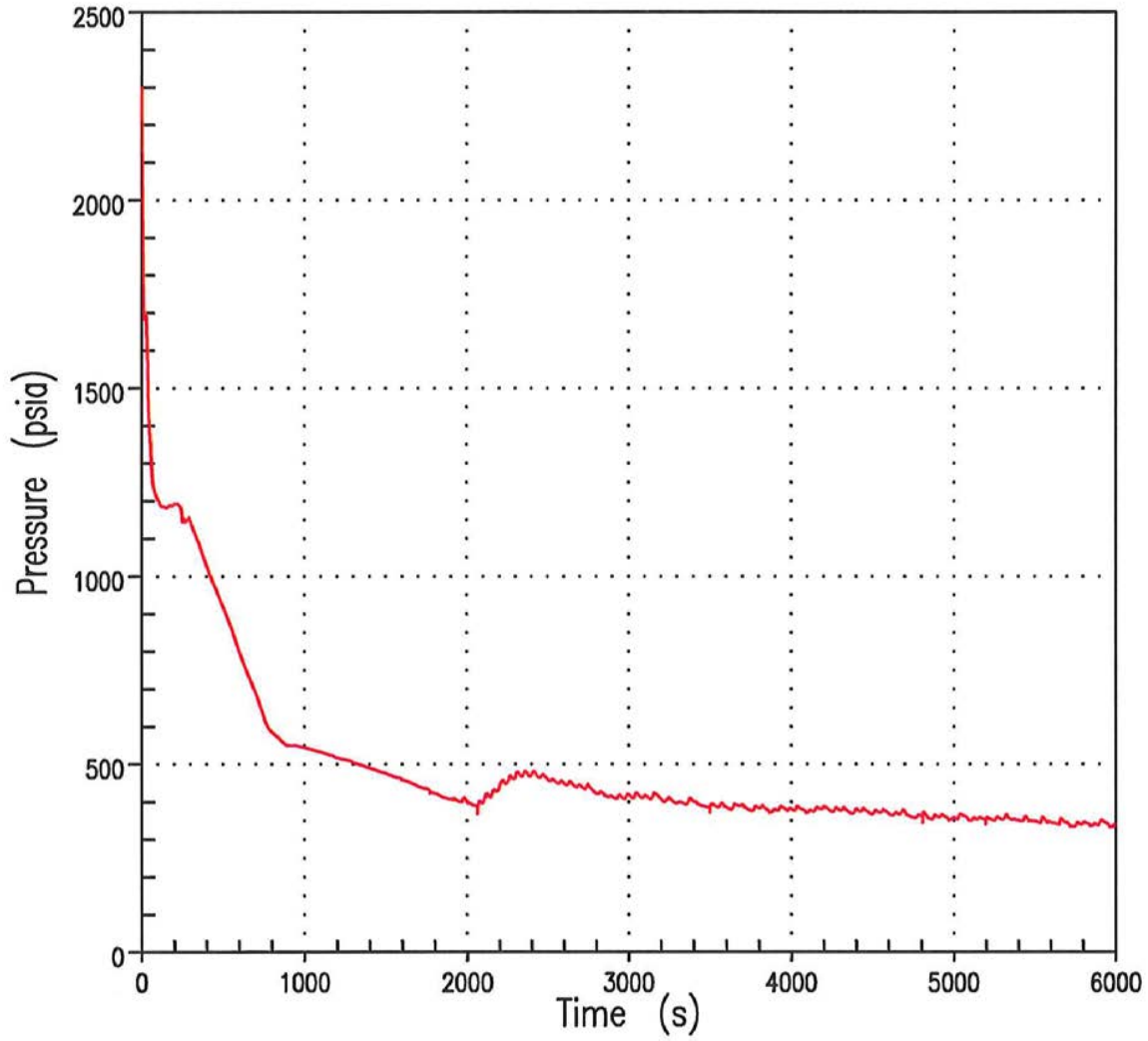
Figure 14.3.1-5 CORE MIXTURE LEVEL AND TOP OF CORE - 3 INCH BREAK
POINT BEACH UNIT 1
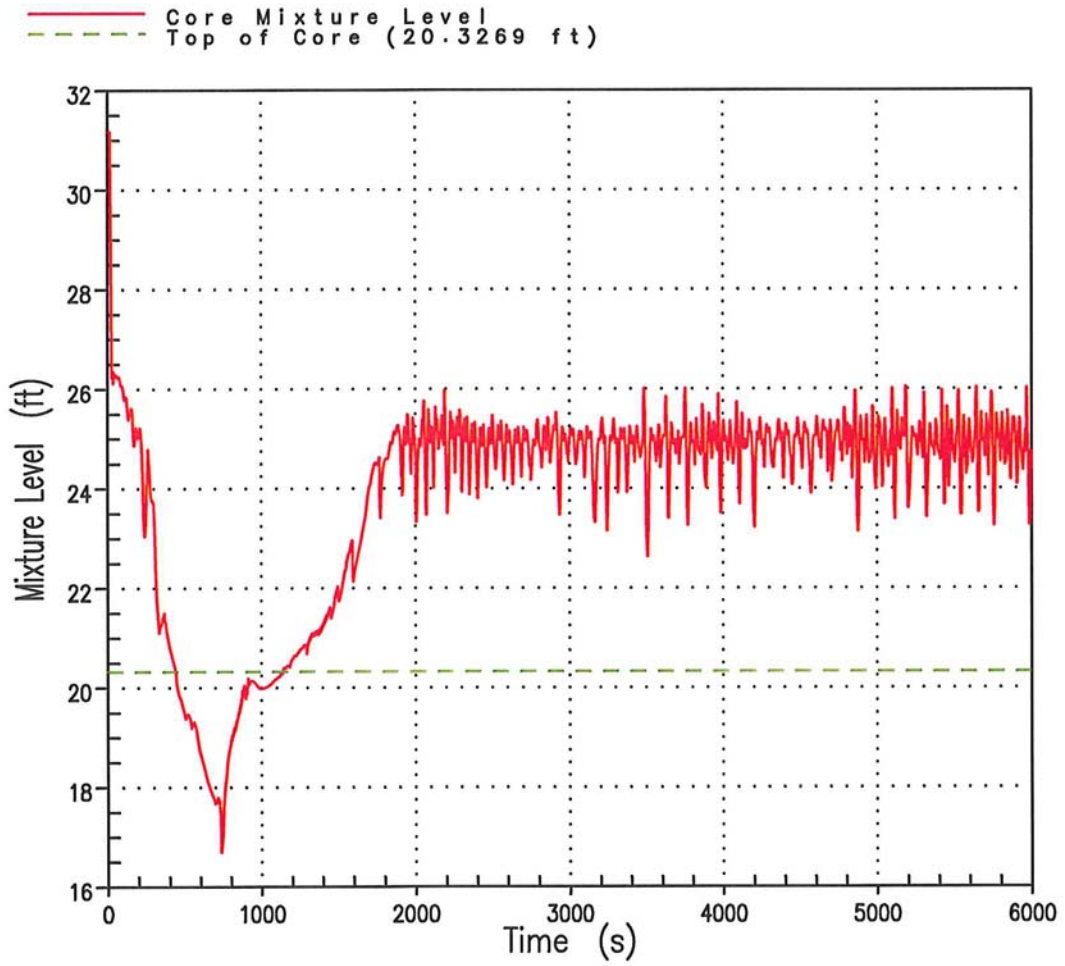
Figure 14.3.1-6 TOTAL REACTOR COOLANT SYSTEM MASS - 3 INCH BREAK
POINT BEACH UNIT 1
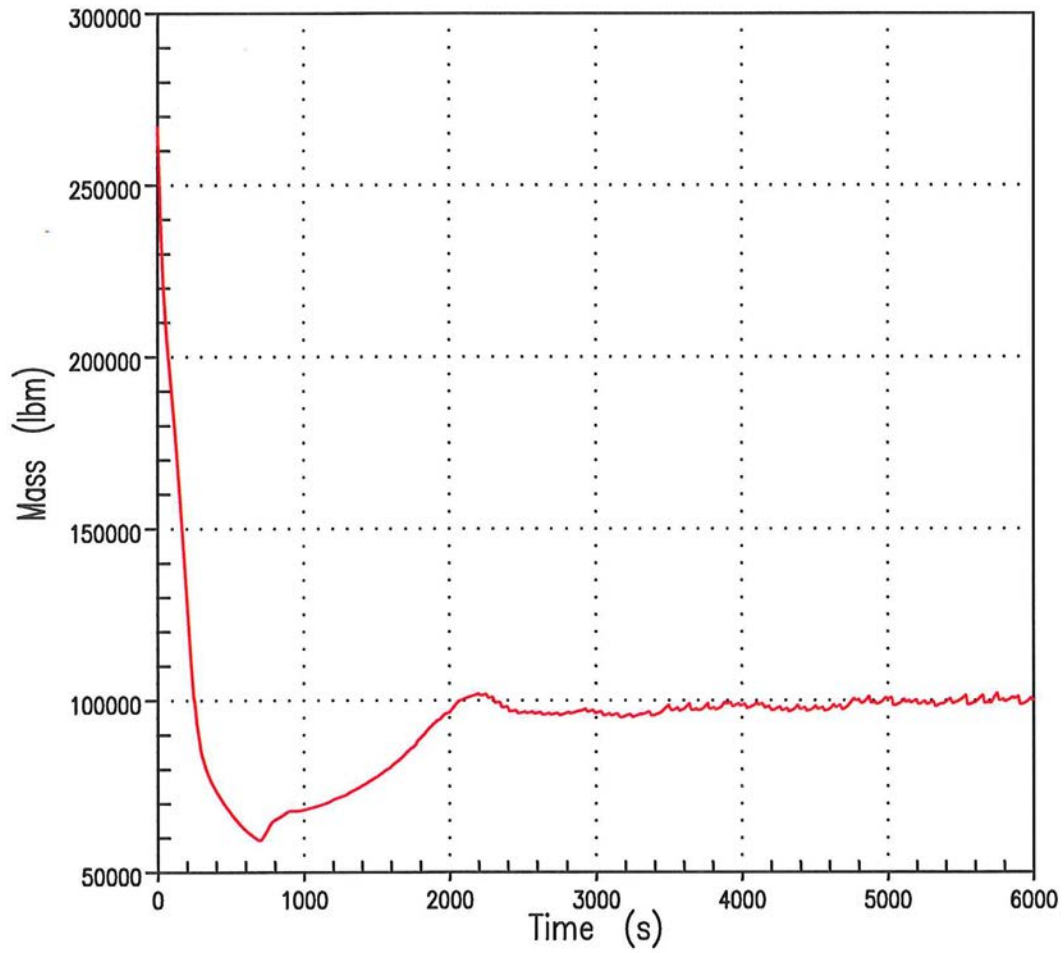
Figure 14.3.1-7 TOP CORE EXIT VAPOR TEMPERATURE - 3 INCH BREAK
POINT BEACH UNIT 1
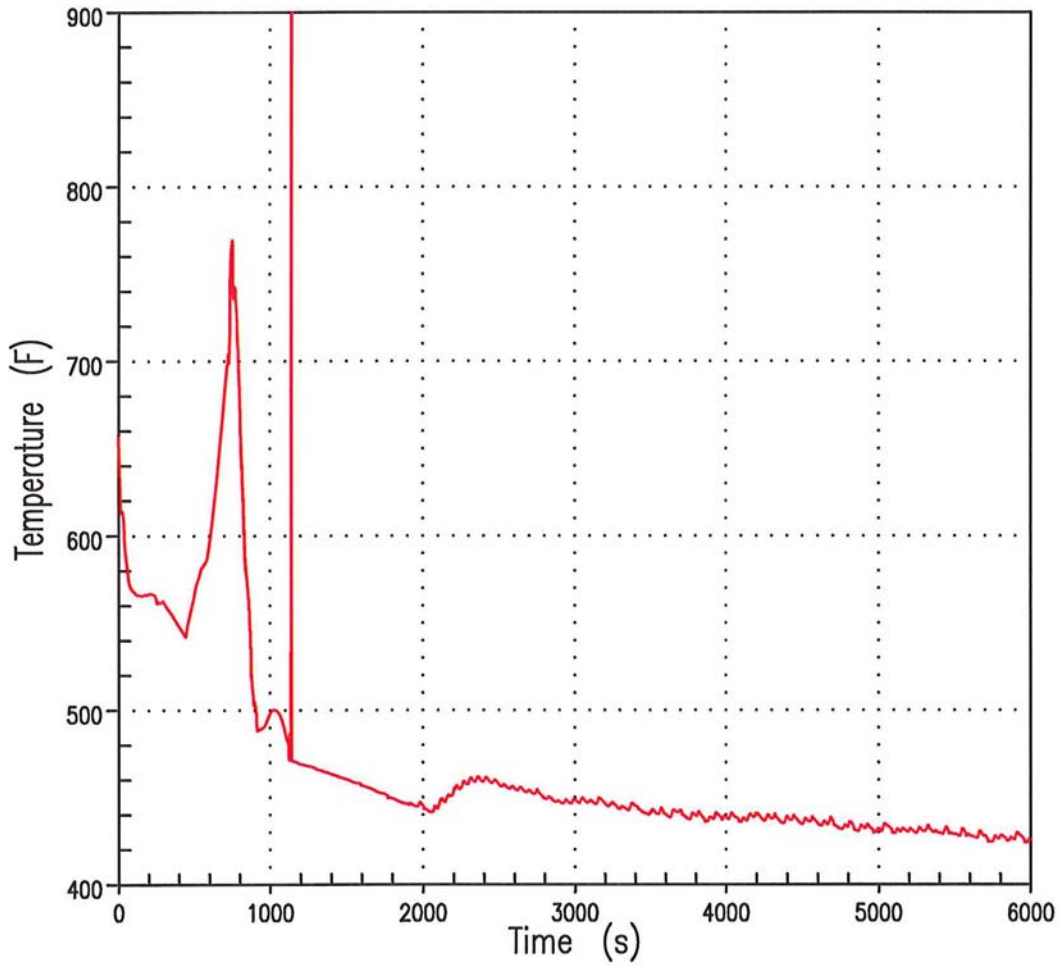
Figure 14.3.1-8 VAPOR MASS FLOW RATE OUT OF TOP OF CORE - 3 INCH BREAK
POINT BEACH UNIT 1
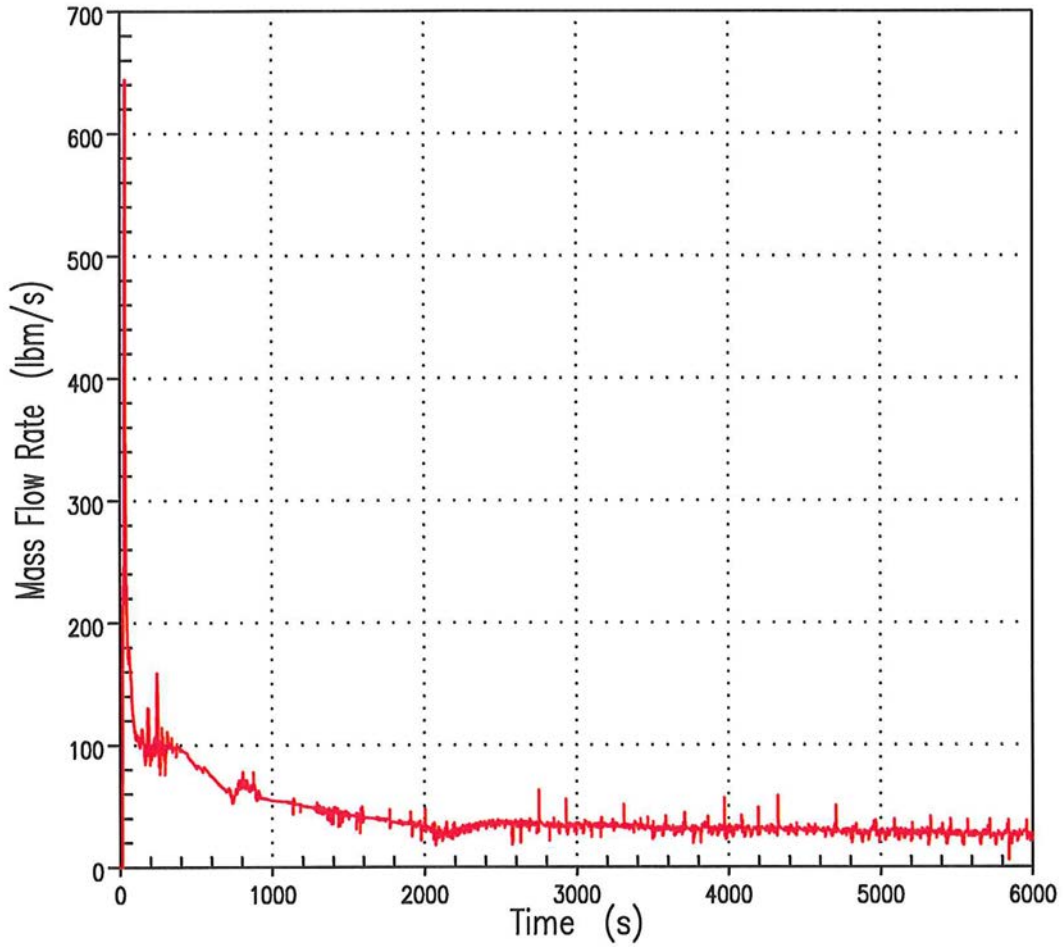
Figure 14.3.1-9 TOTAL BREAK FLOW AND SAFETY INJECTION FLOW - 3 INCH
BREAK POINT BEACH UNIT 1
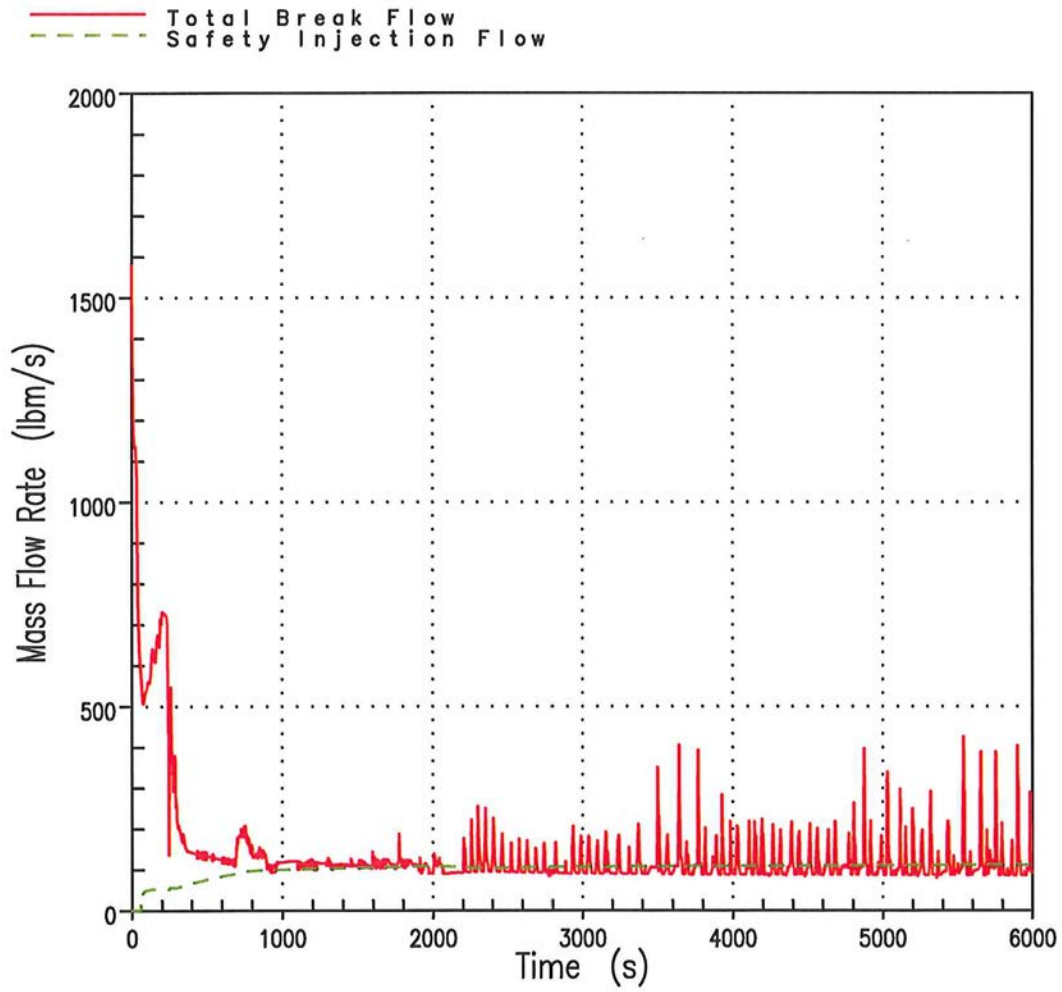
Figure 14.3.1-10 CLADDING SURFACE HEAT TRANSFER COEFFICIENT AT PCT
ELEVATION - 3 INCH BREAK POINT BEACH UNIT 1
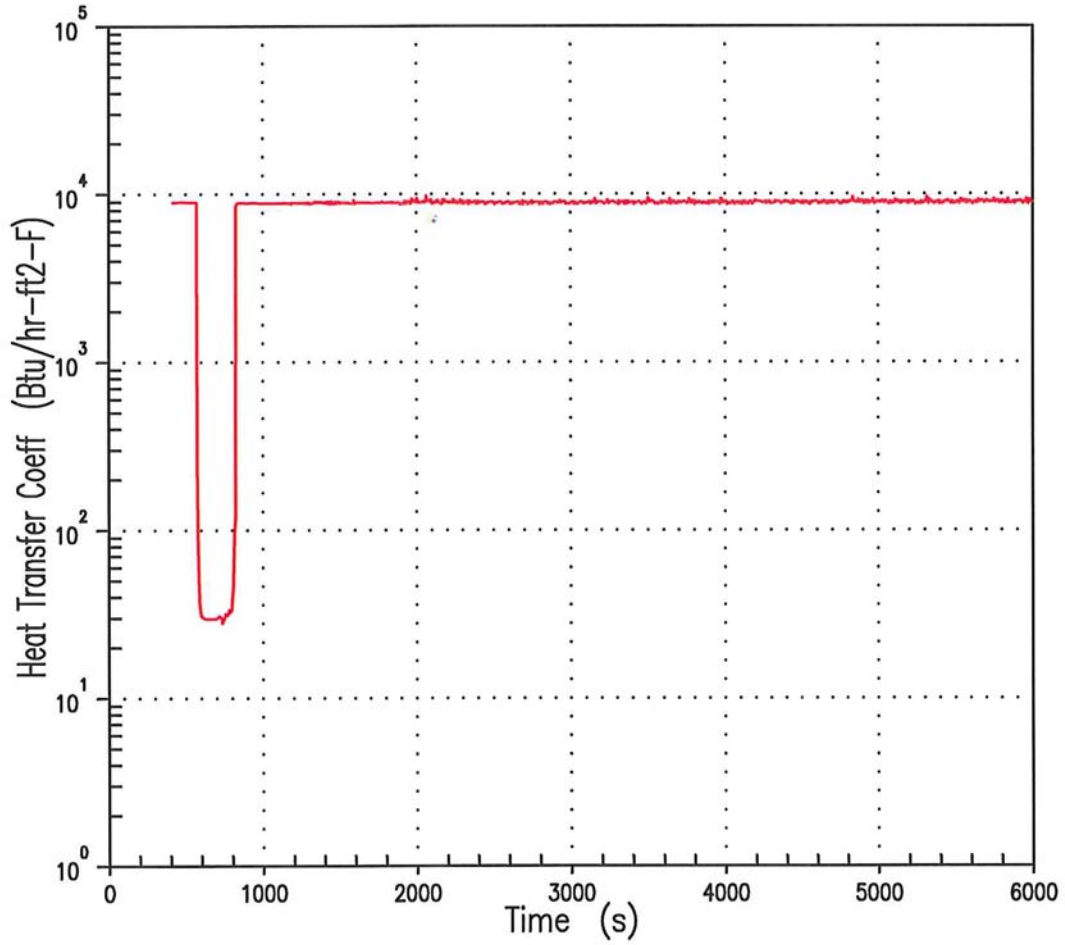
Figure 14.3.1-11 FLUID TEMPERATURE AT PCT ELEVATION - 3 INCH BREAK
POINT BEACH UNIT 1
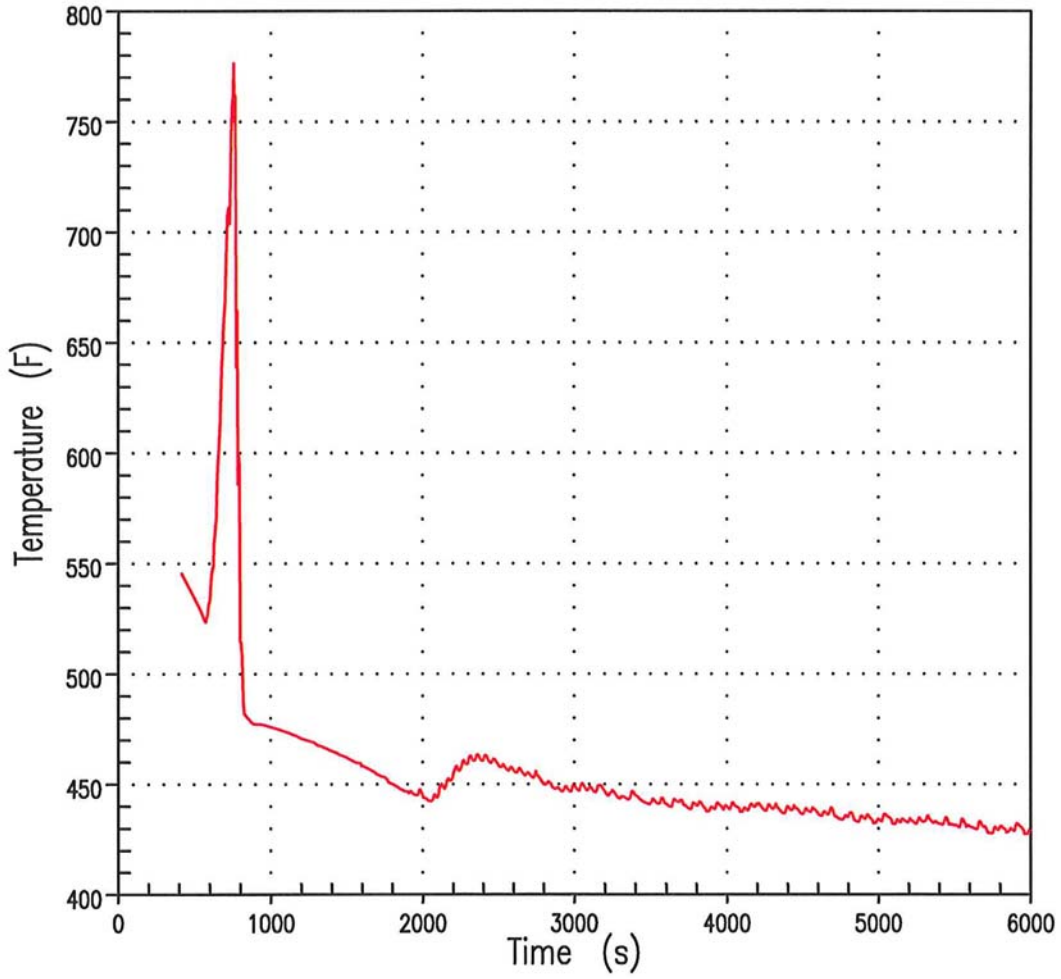
Figure 14.3.1-12 CLADDING TEMPERATURE TRANSIENT AT PCT ELEVATION - 3 INCH
BREAK POINT BEACH UNIT 1
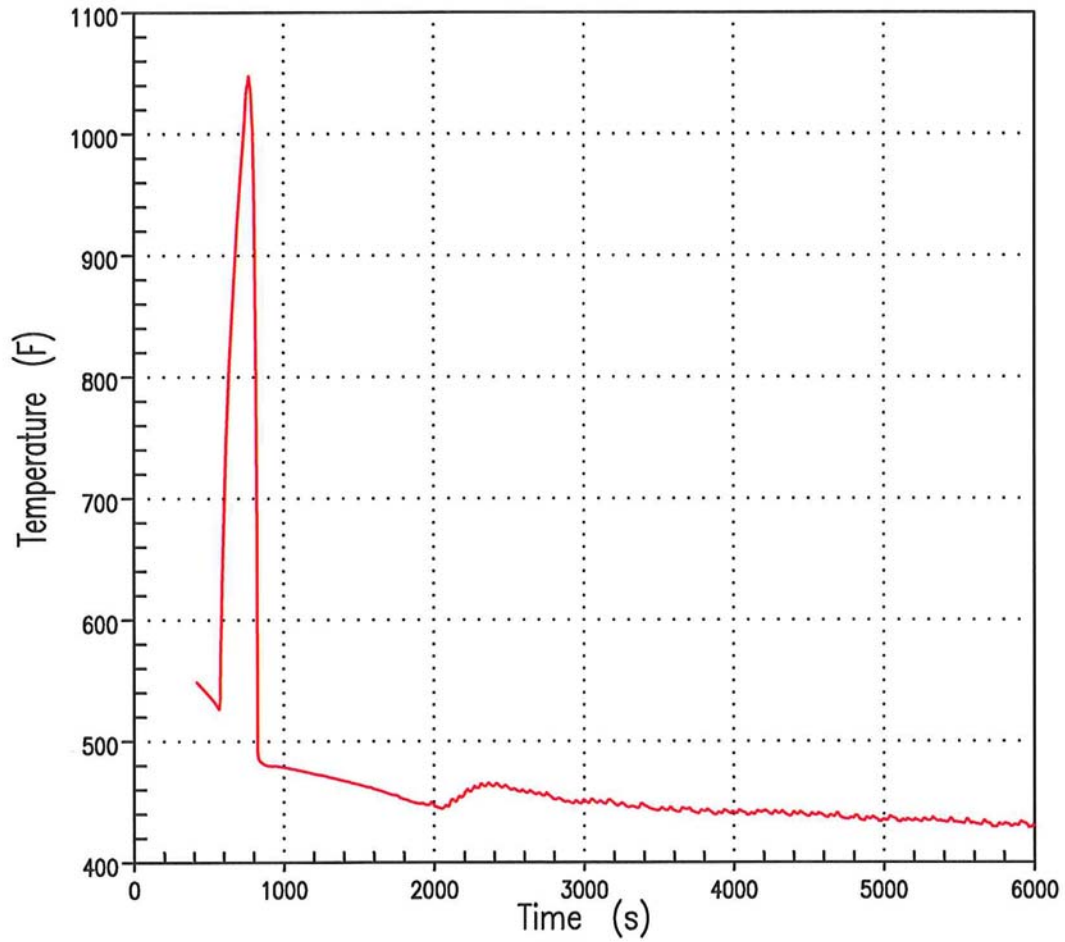
Figure 14.3.1-13 LOCAL ZRO2 THICKNESS AT MAXIMUM LOCAL ZRO2 ELEVATION -
3 INCH BREAK POINT BEACH UNIT 1
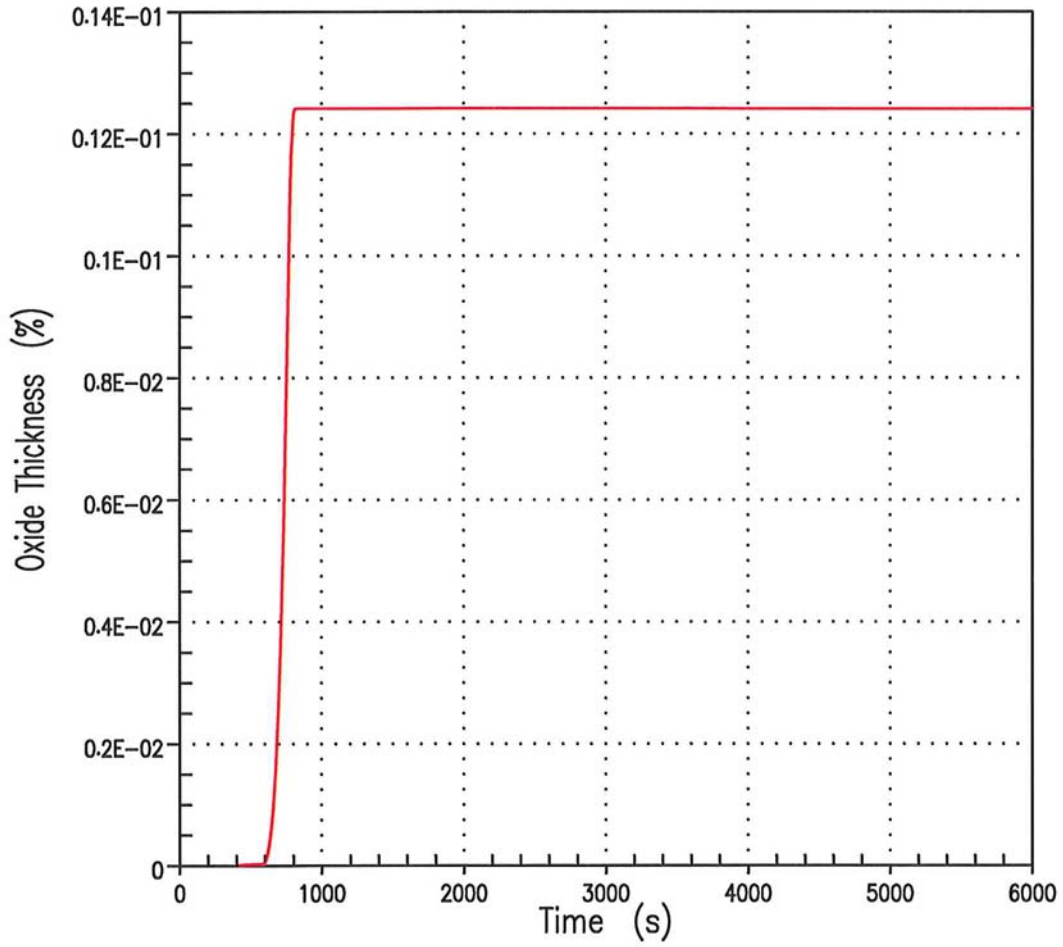
Figure 14.3.1-14 REACTOR COOLANT SYSTEM PRESSURE - 3 INCH BREAK
POINT BEACH UNIT 2
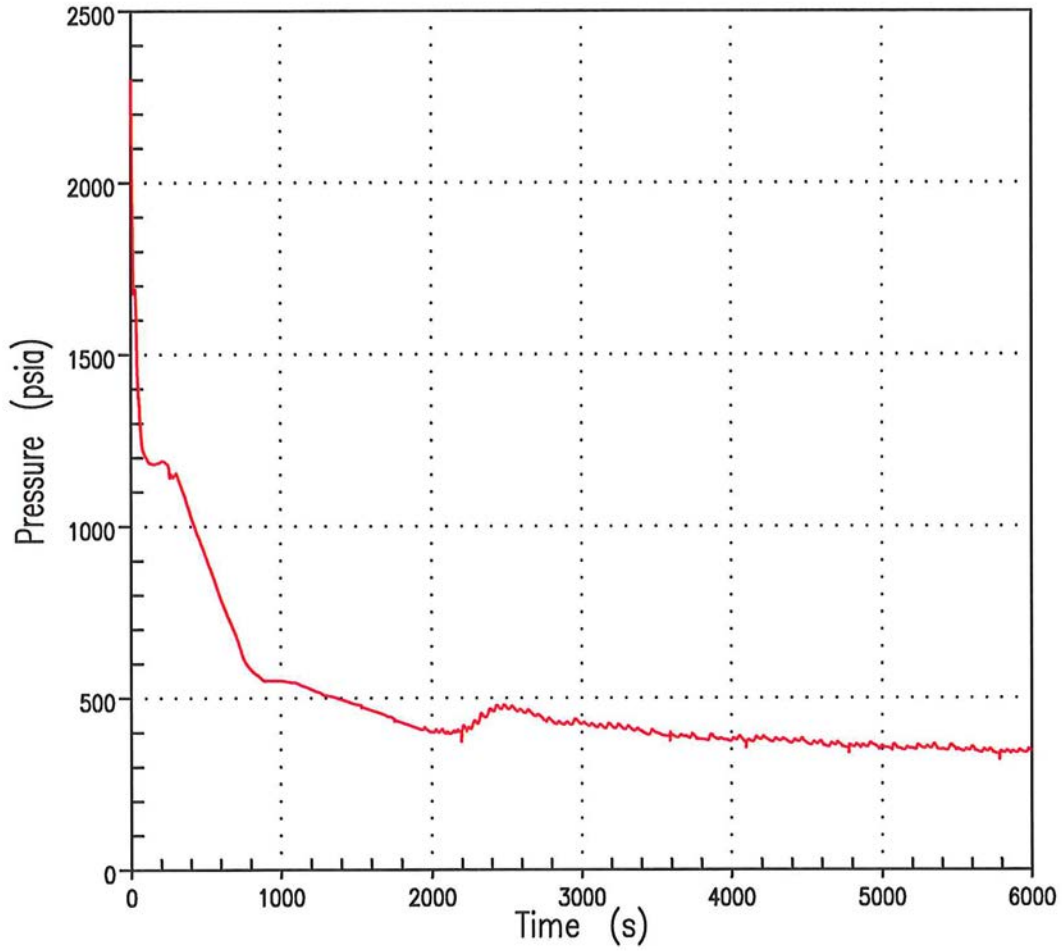
Figure 14.3.1-15 CORE MIXTURE LEVEL AND TOP OF CORE - 3 INCH BREAK
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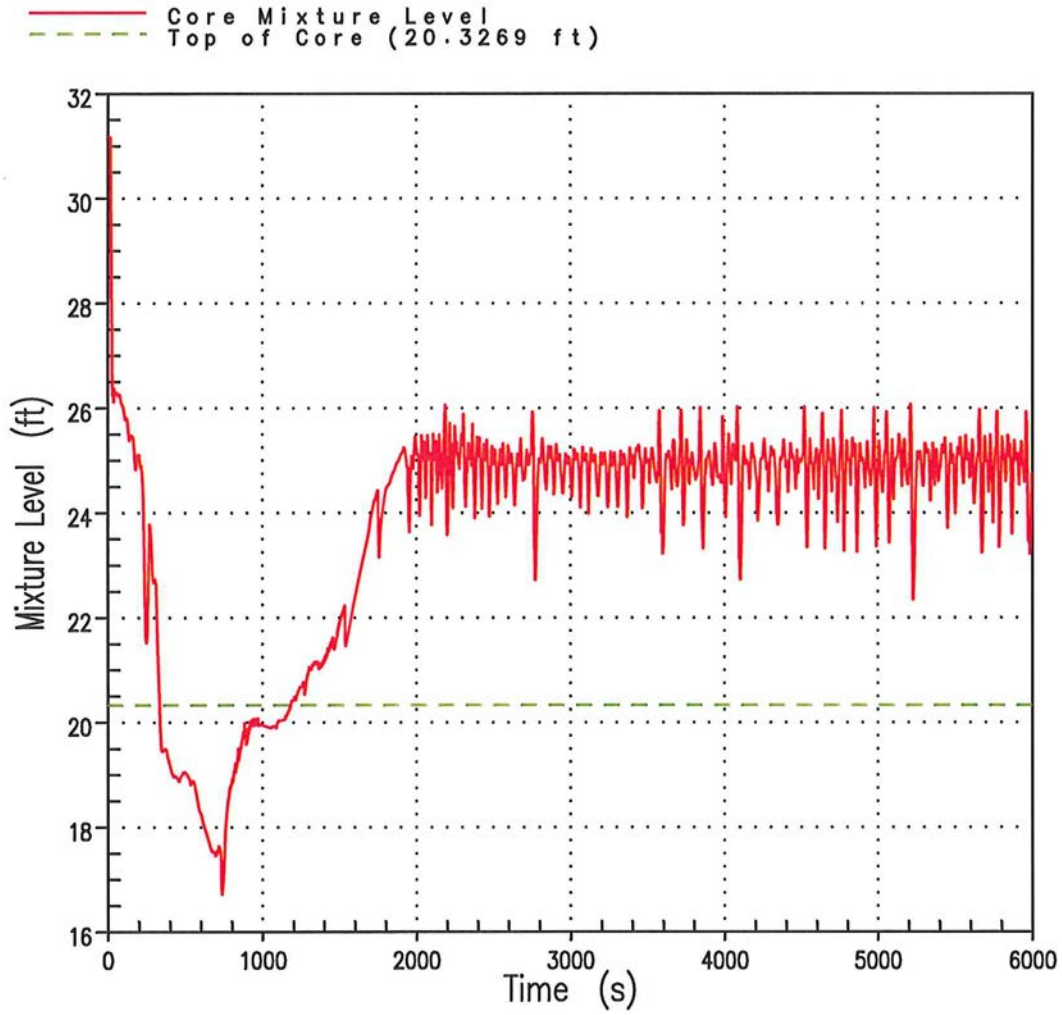
Figure 14.3.1-16 TOTAL REACTOR COOLANT SYSTEM MASS - 3 INCH BREAK
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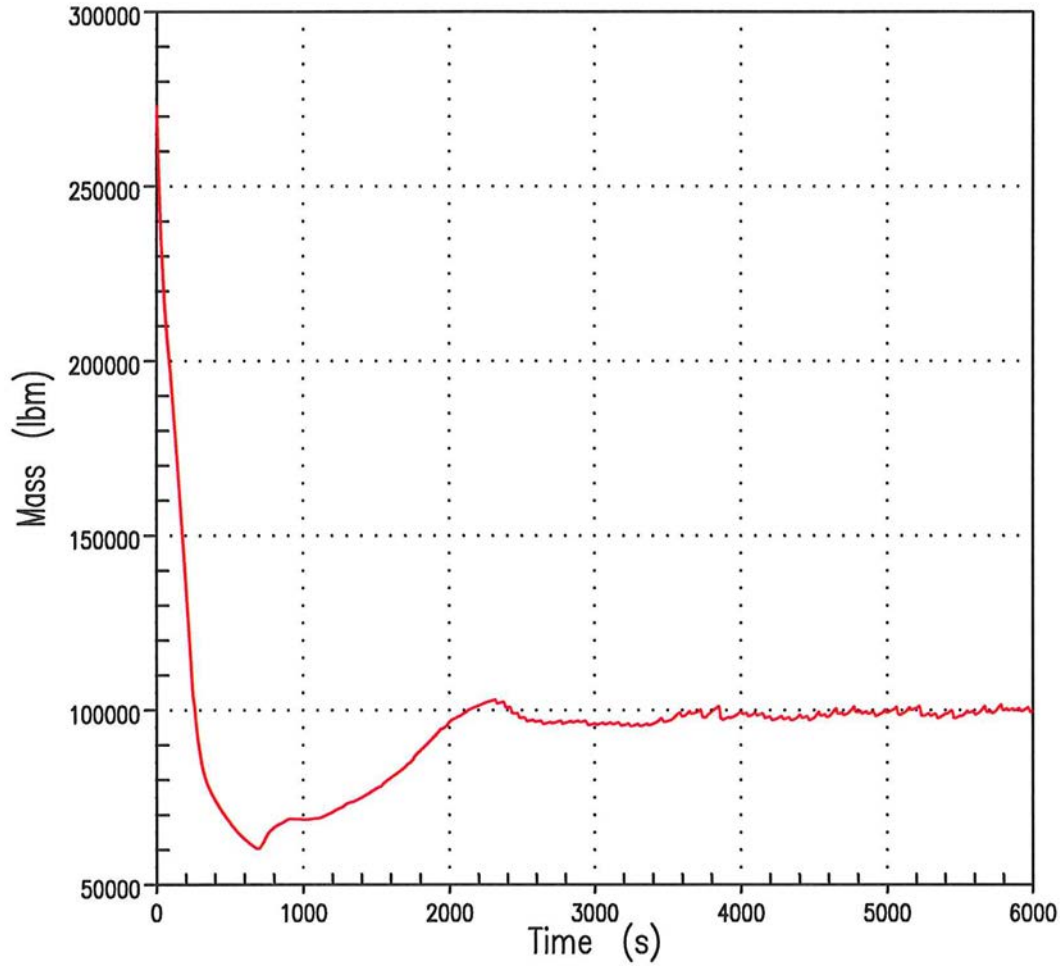
Figure 14.3.1-17 TOP CORE EXIT VAPOR TEMPERATURE - 3 INCH BREAK
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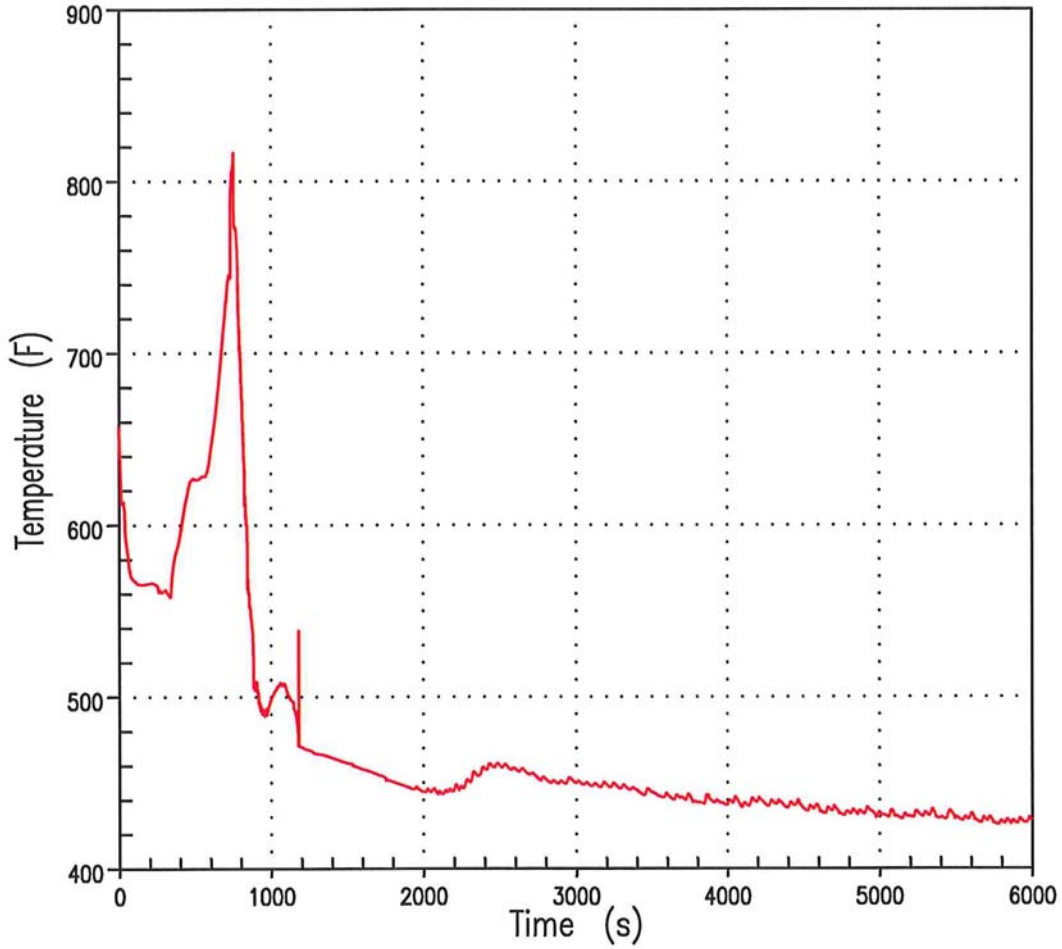
Figure 14.3.1-18 VAPOR MASS FLOW RATE OUT OF TOP OF CORE - 3 INCH BREAK
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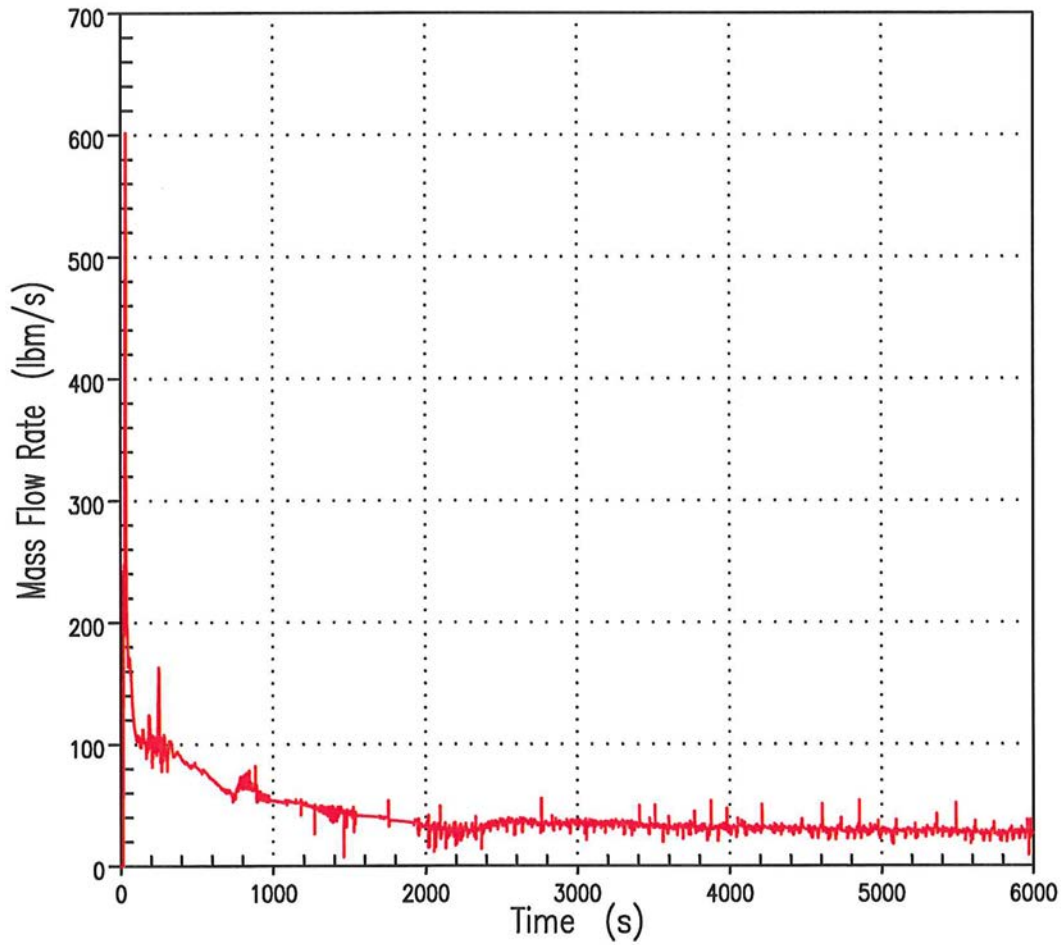
Figure 14.3.1-19 TOTAL BREAK FLOW AND SAFETY INJECTION FLOW - 3 INCH
BREAK POINT BEACH UNIT 2
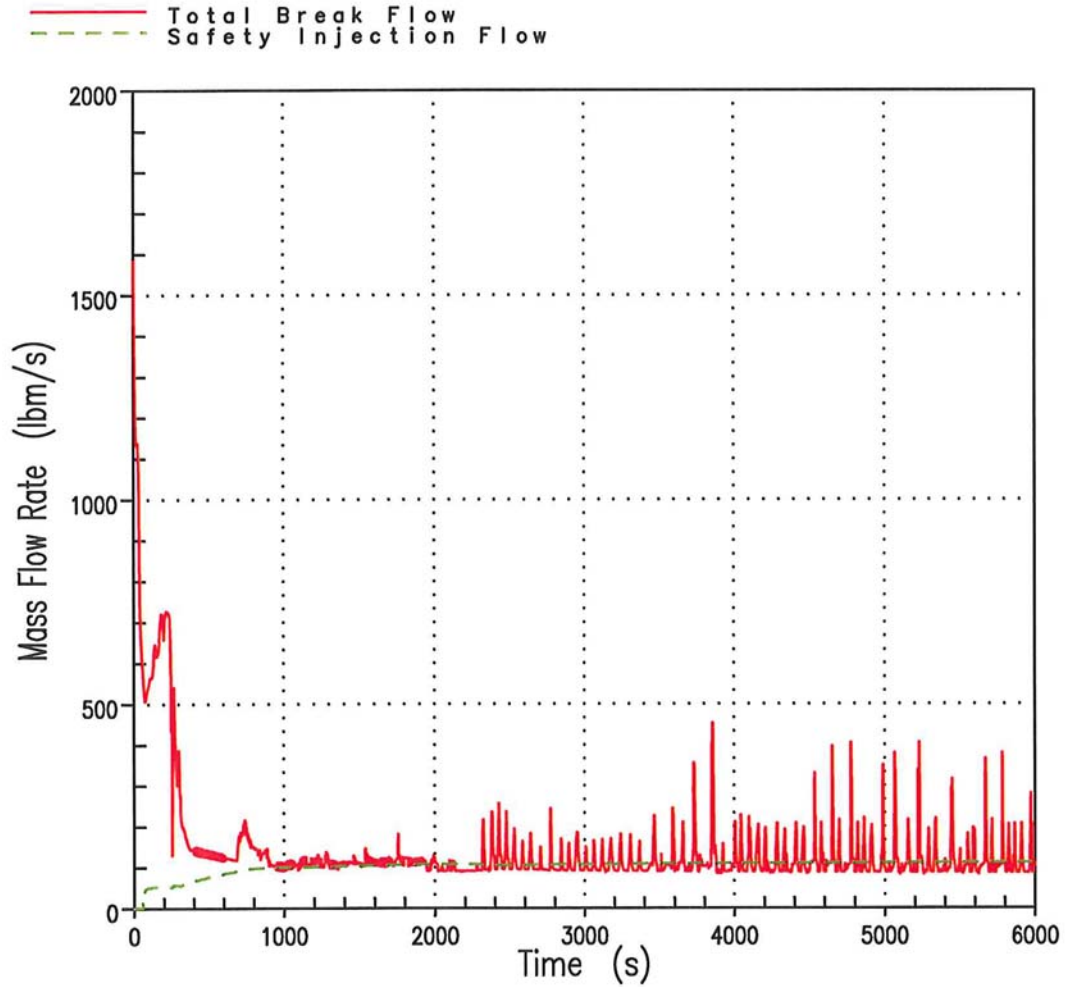
Figure 14.3.1-20 CLADDING SURFACE HEAT TRANSFER COEFFICIENT AT PCT
ELEVATION - 3 INCH BREAK POINT BEACH UNIT 2
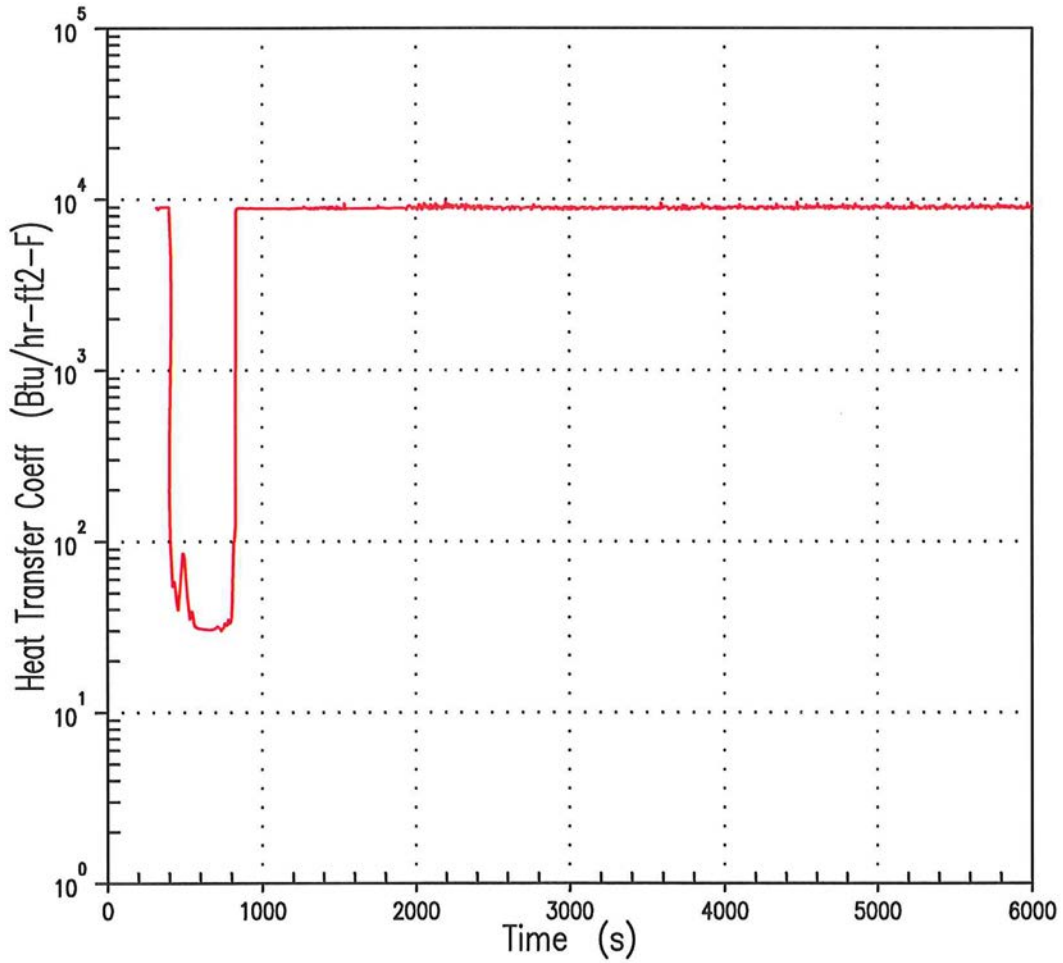
Figure 14.3.1-21 FLUID TEMPERATURE AT PCT ELEVATION - 3 INCH BREAK POINT
BEACH UNIT 2
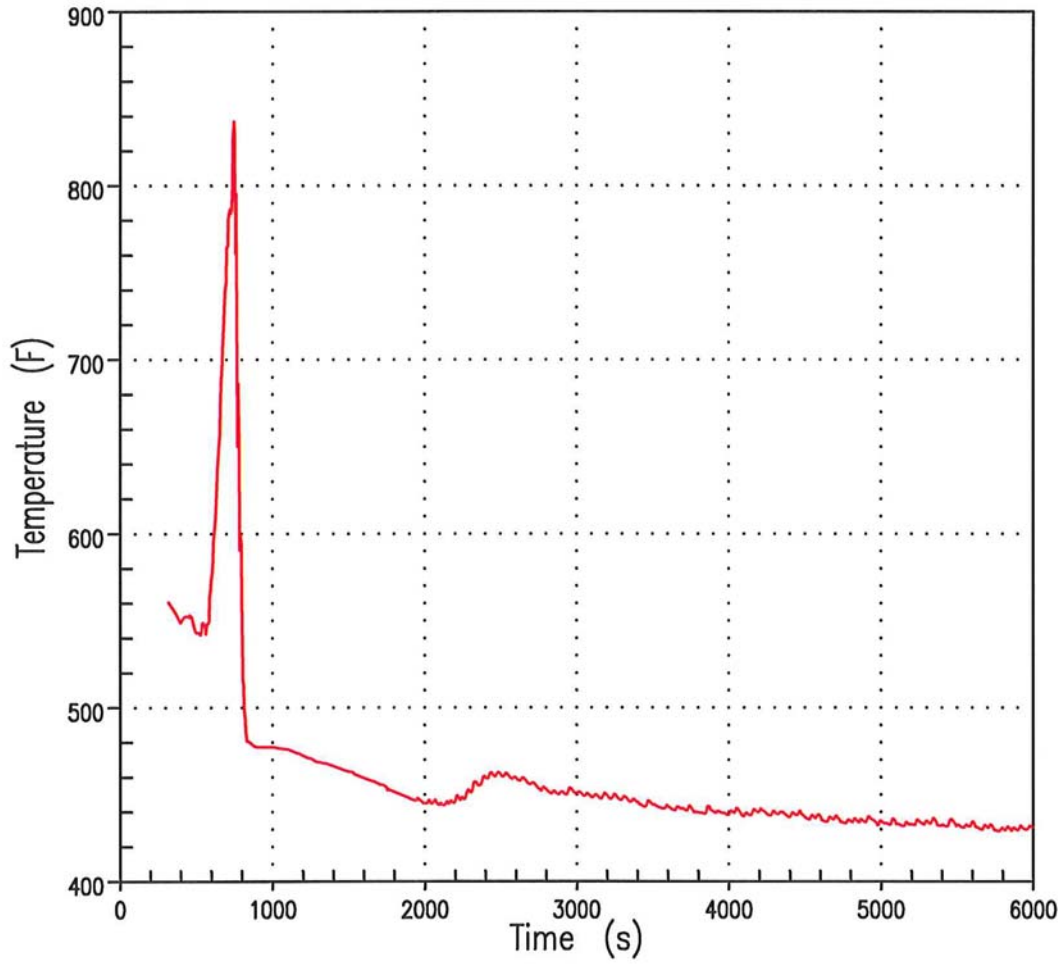
Figure 14.3.1-22 CLADDING TEMPERATURE TRANSIENT AT PCT ELEVATION - 3 INCH
BREAK POINT BEACH UNIT 2
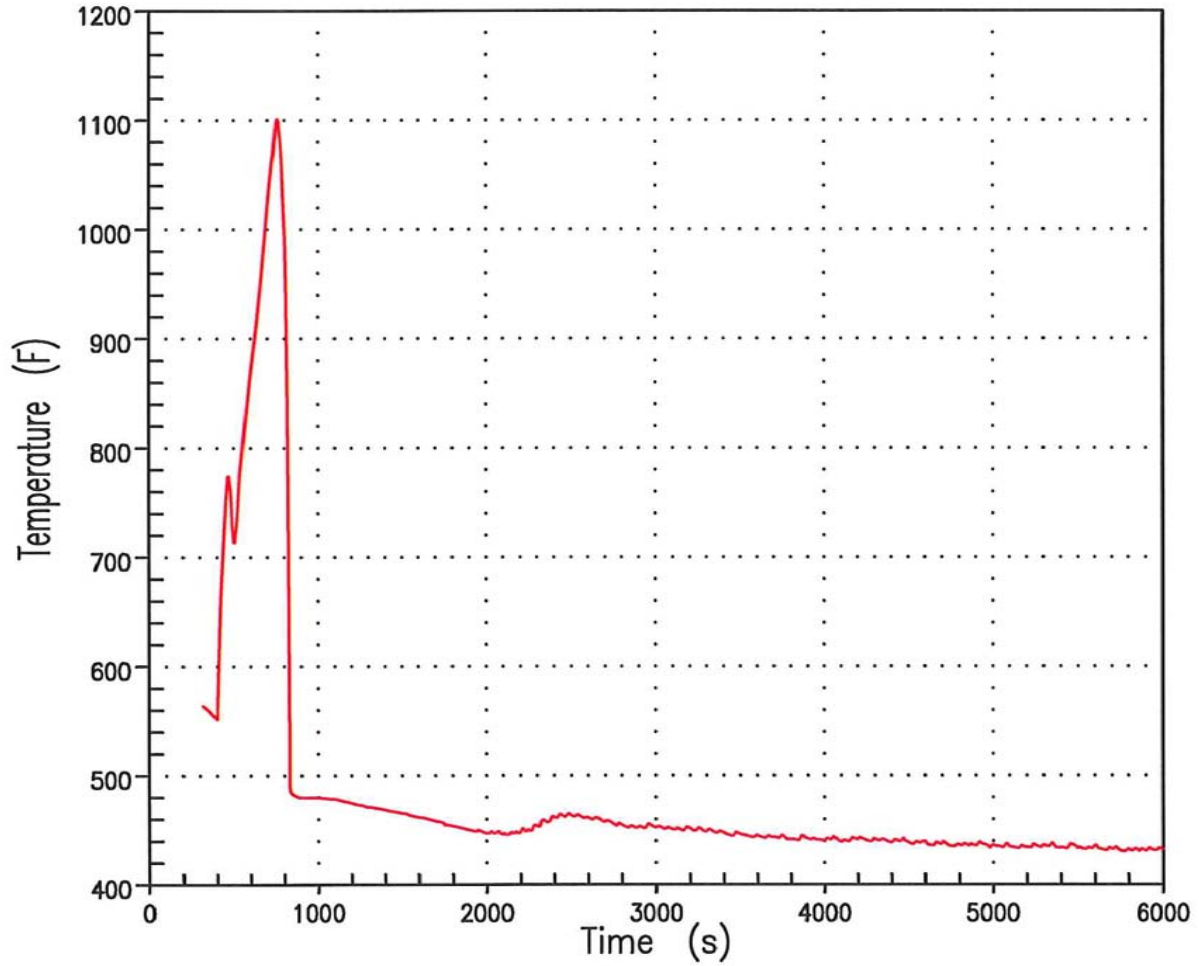
Figure 14.3.1-23 LOCAL ZR02 THICKNESS AT MAXIMUM LOCAL ZRO2 ELEVATION -
3 INCH BREAK POINT BEACH UNIT 2
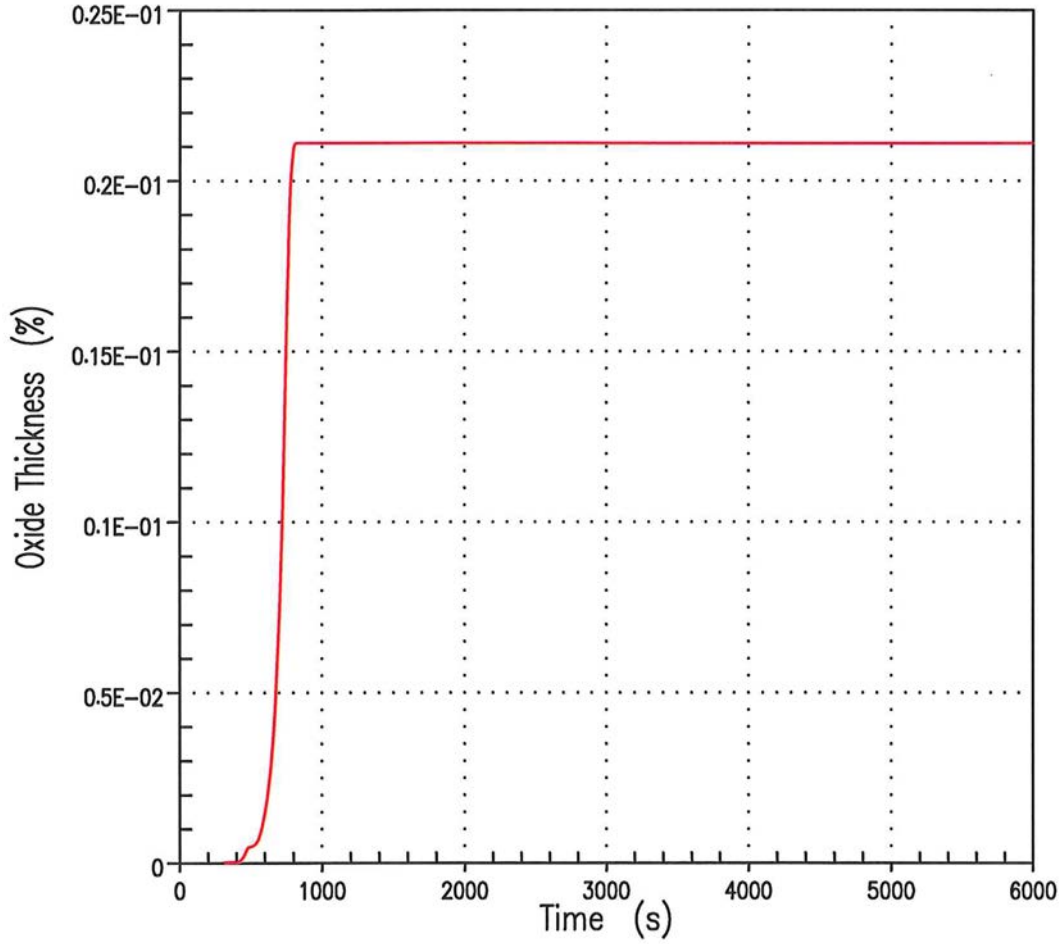
Figure 14.3.1-24 REACTOR COOLANT SYSTEM PRESSURE - 1.5 INCH BREAK
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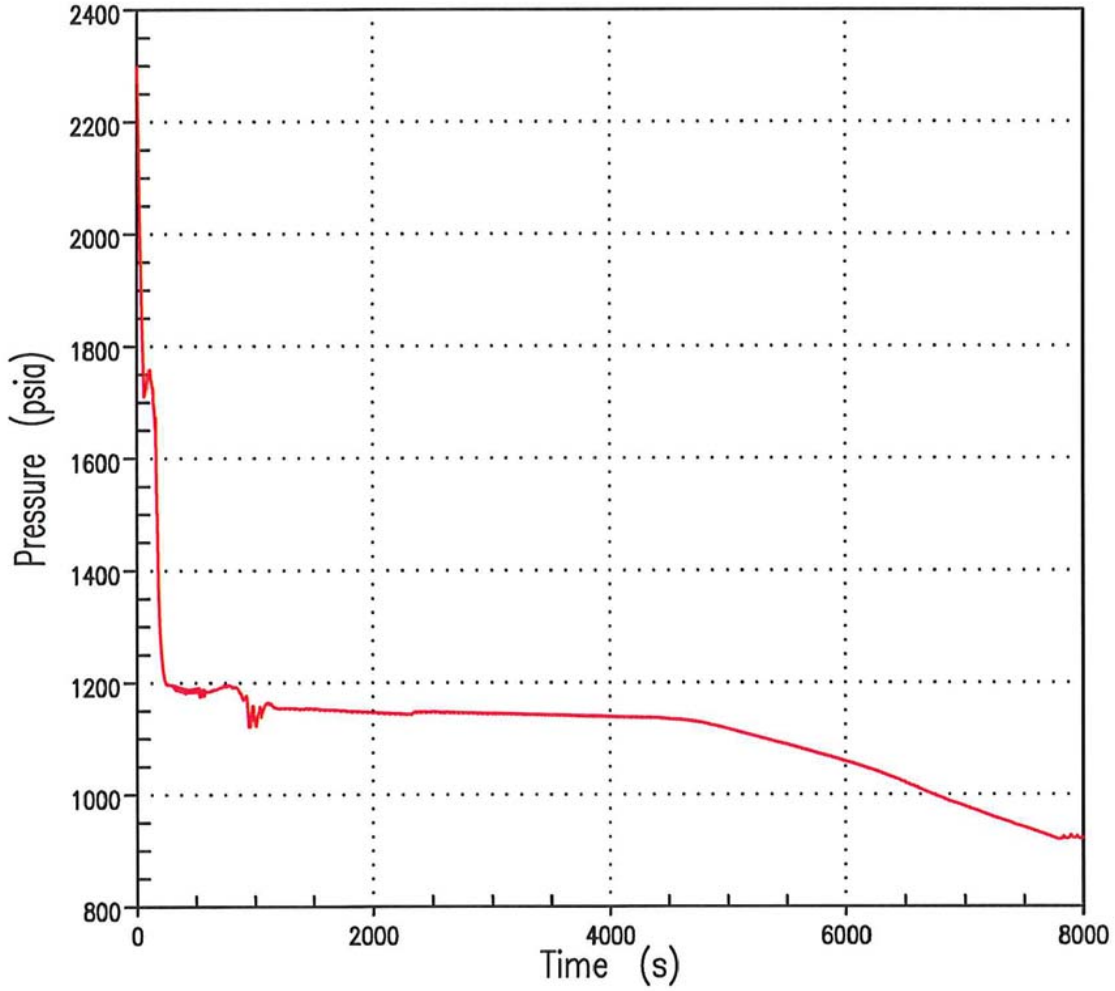
Figure 14.3.1-25 CORE MIXTURE LEVEL AND TOP OF CORE - 1.5 INCH BREAK
POINT BEACH UNIT 1
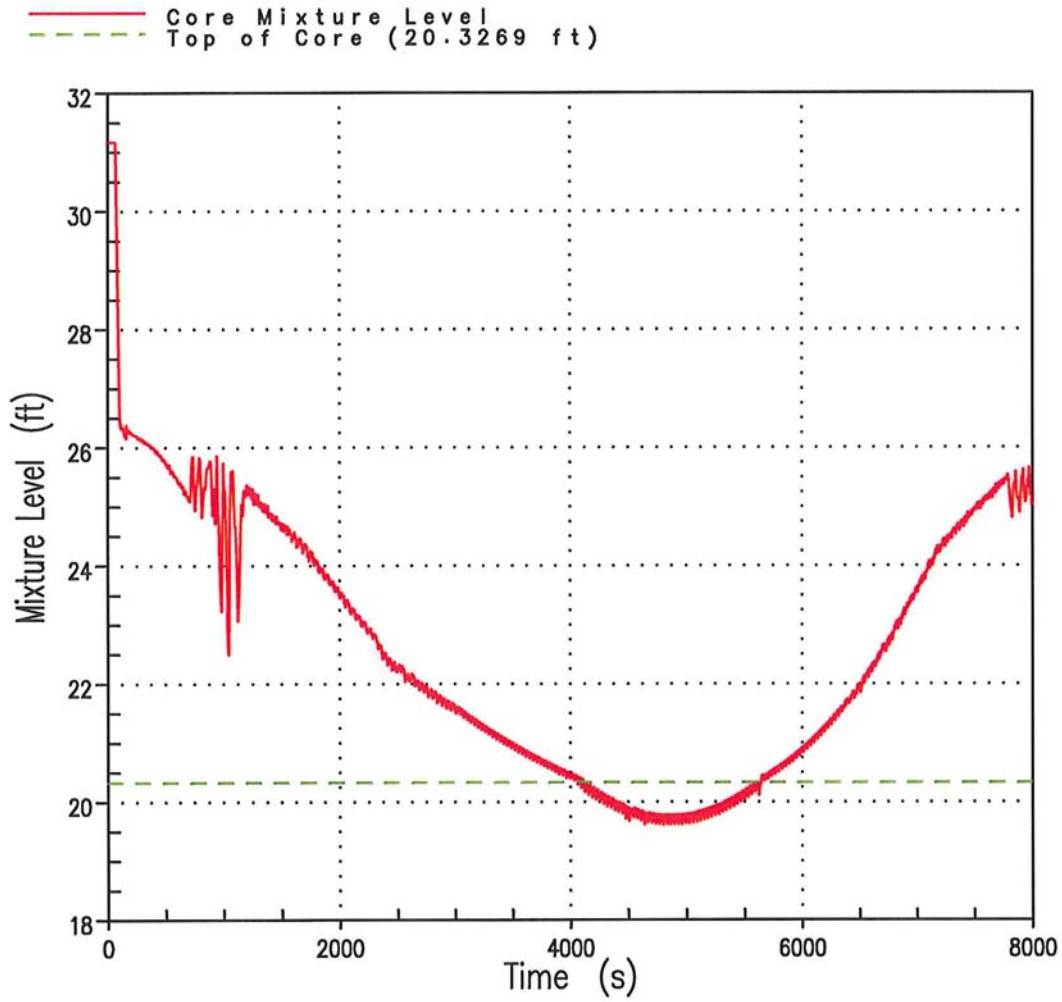
Figure 14.3.1-26 TOP CORE EXIT VAPOR TEMPERATURE - 1.5 INCH BREAK
POINT BEACH UNIT 1
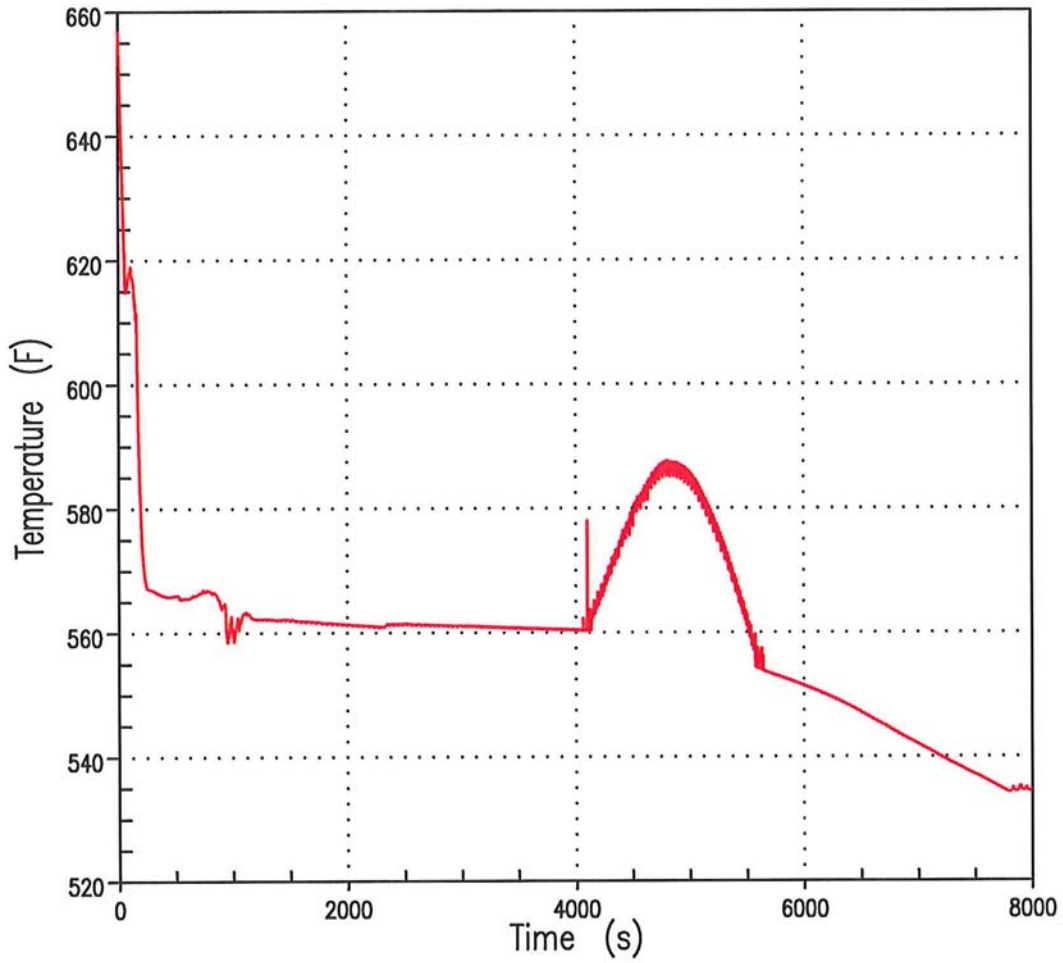
Figure 14.3.1-27 CLADDING TEMPERATURE TRANSIENT AT PCT ELEVATION -
1.5 INCH BREAK POINT BEACH UNIT 1
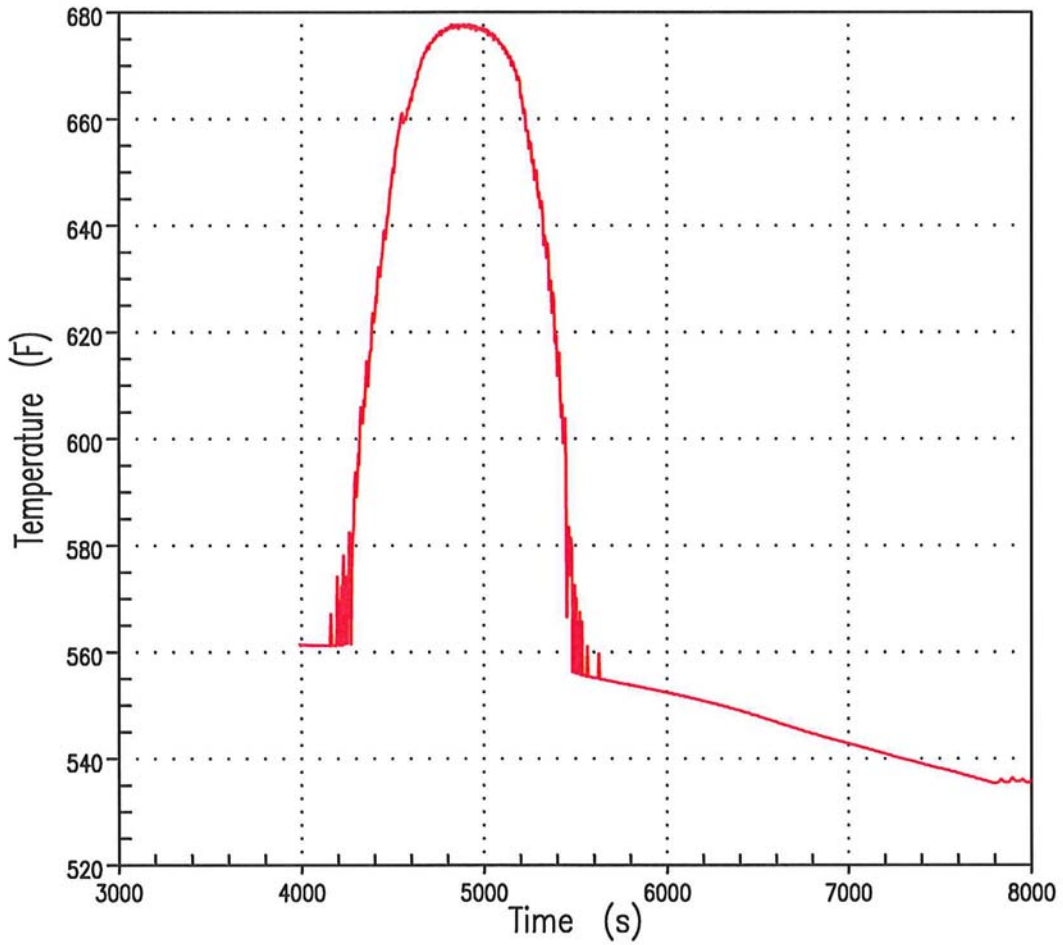
Figure 14.3.1-28 LOCAL ZRO2 THICKNESS AT MAXIMUM LOCAL ZRO2 ELEVATION -
1.5 INCH BREAK POINT BEACH UNIT 1
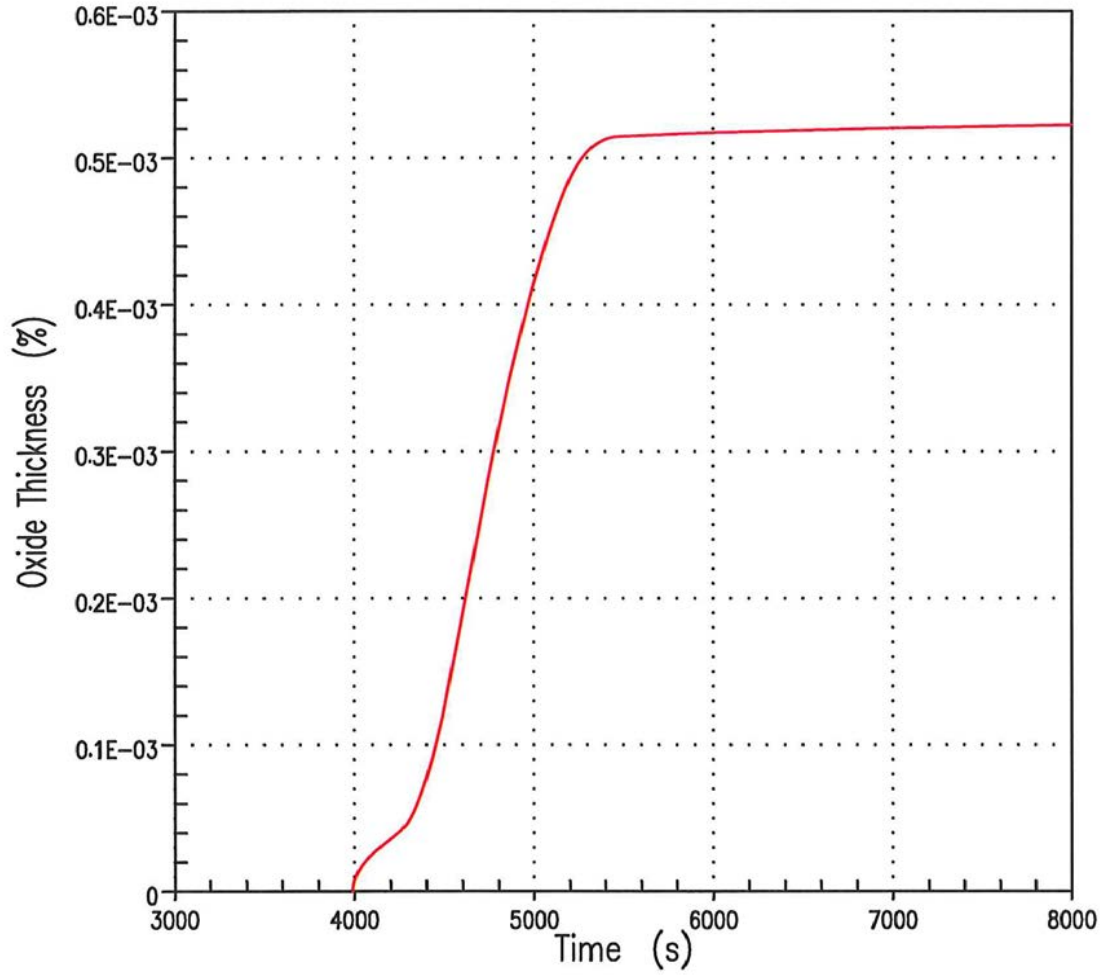
Figure 14.3.1-29 REACTOR COOLANT SYSTEM PRESSURE - 1.5 INCH BREAK
POINT BEACH UNIT 2
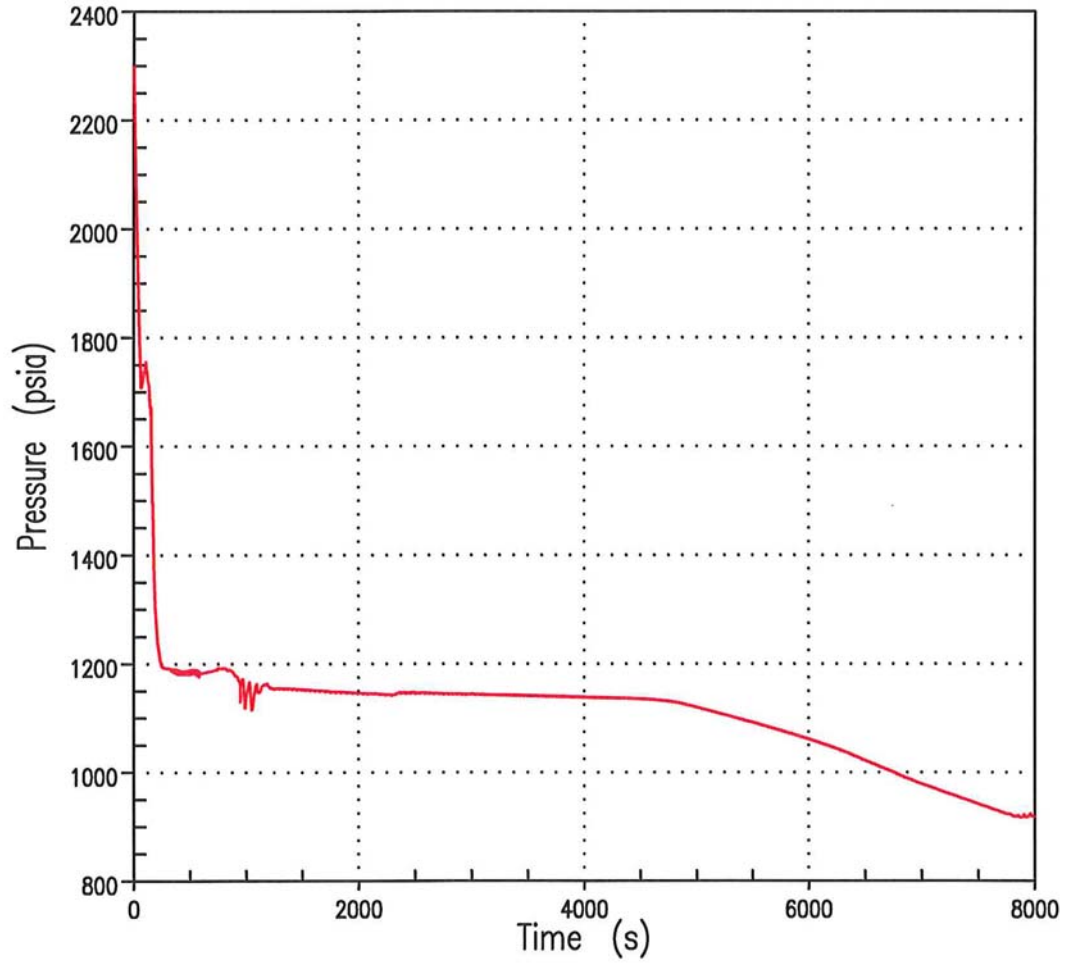
Figure 14.3.1-30 CORE MIXTURE LEVEL AND TOP OF CORE - 1.5 INCH BREAK
POINT BEACH UNIT 2
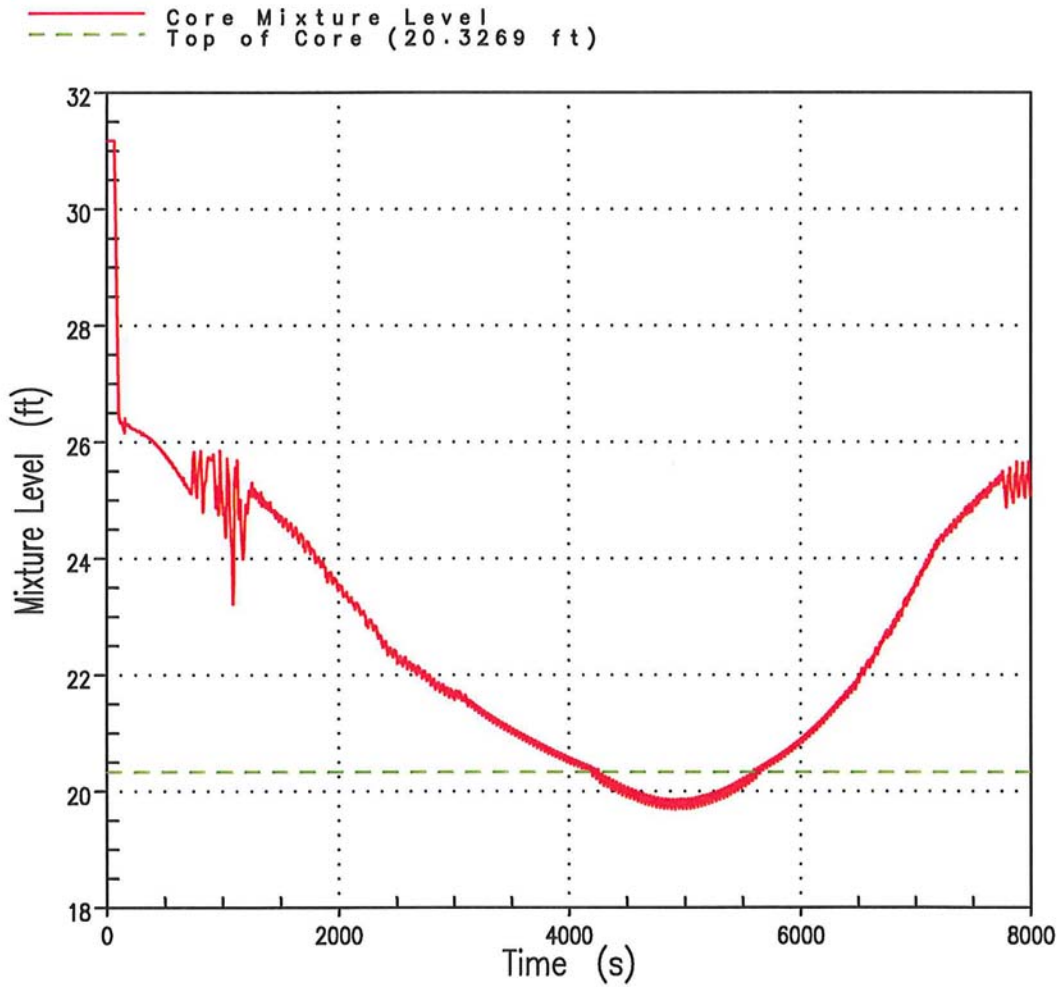
Figure 14.3.1-31 TOP CORE EXIT VAPOR TEMPERATURE - 1.5 INCH BREAK
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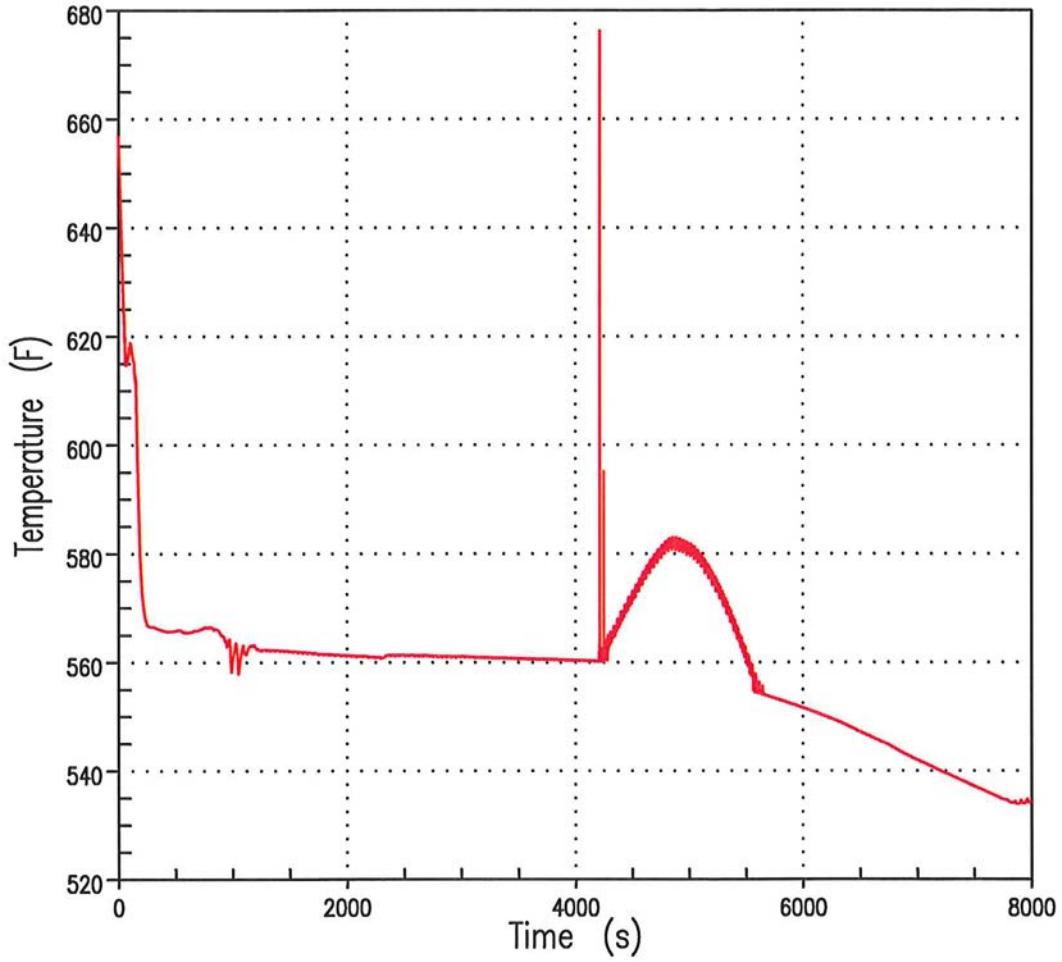
Figure 14.3.1-32 CLADDING TEMPERATURE TRANSIENT AT PCT ELEVATION -
1.5 INCH BREAK POINT BEACH UNIT 2
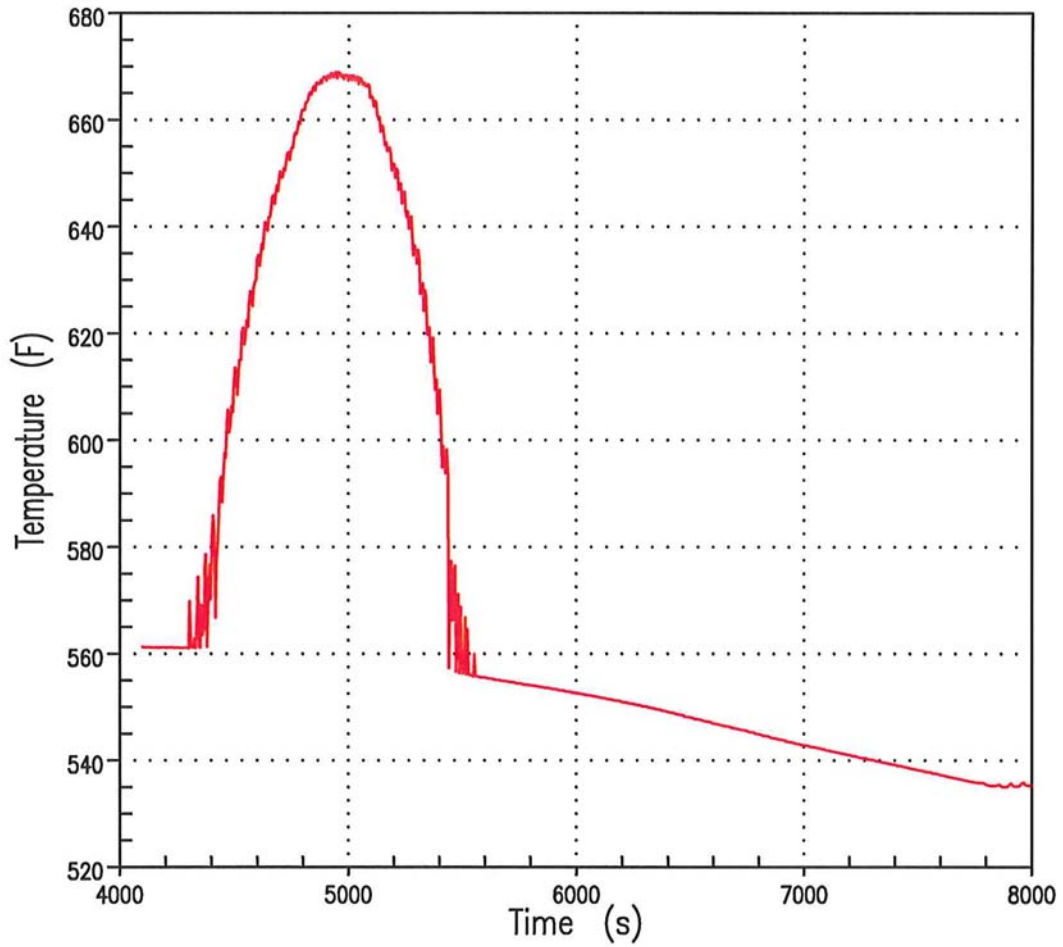
Figure 14.3.1-33 LOCAL ZRO2 THICKNESS AT MAXIMUM LOCAL ZRO2 ELEVATION -
1.5 INCH BREAK POINT BEACH UNIT 2
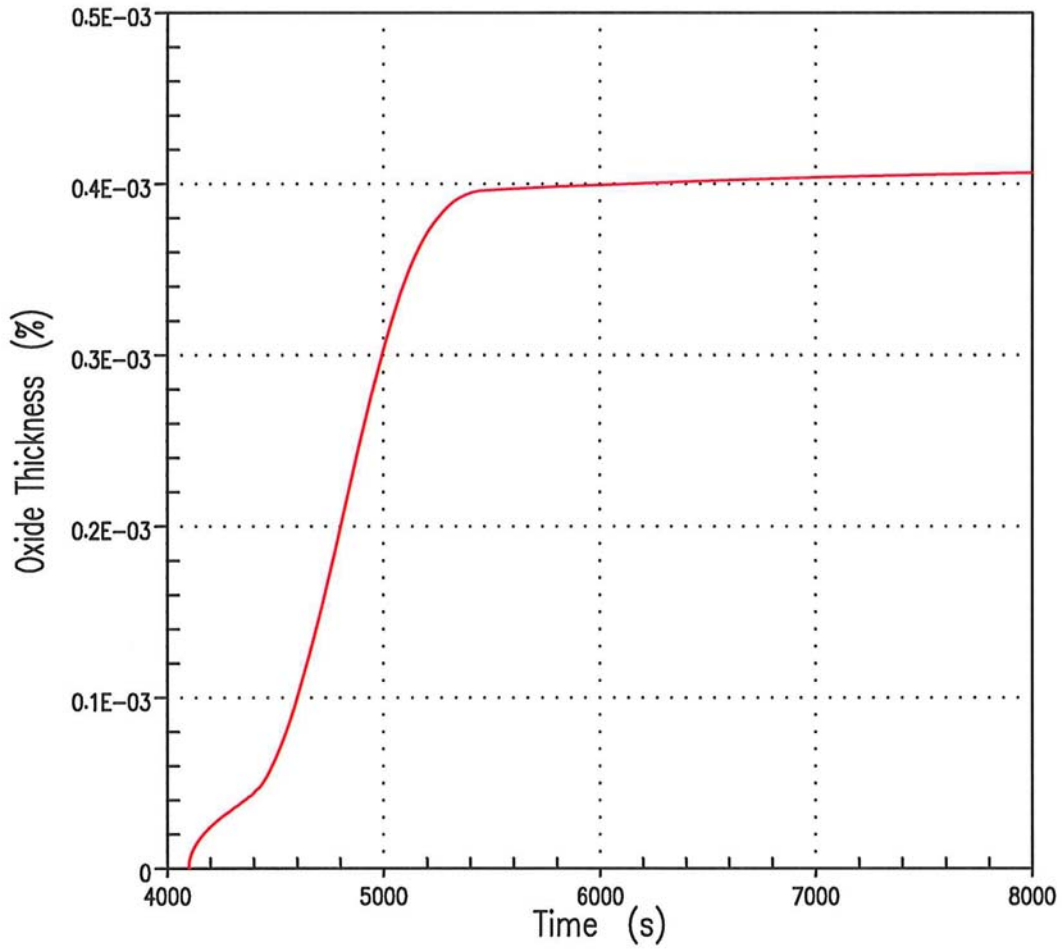
Figure 14.3.1-34 REACTOR COOLANT SYSTEM PRESSURE - 2 INCH BREAK
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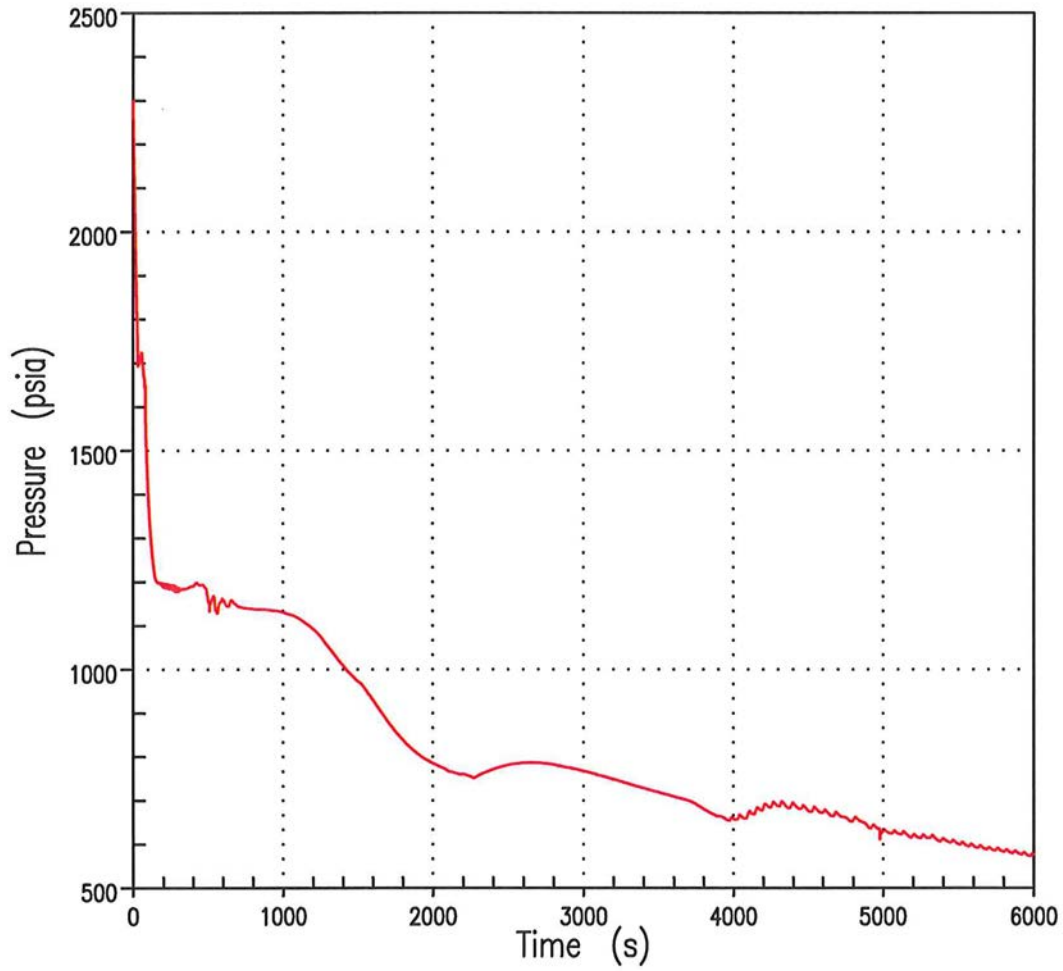
Figure 14.3.1-35 CORE MIXTURE LEVEL AND TOP OF CORE - 2 INCH BREAK
POINT BEACH UNIT 1
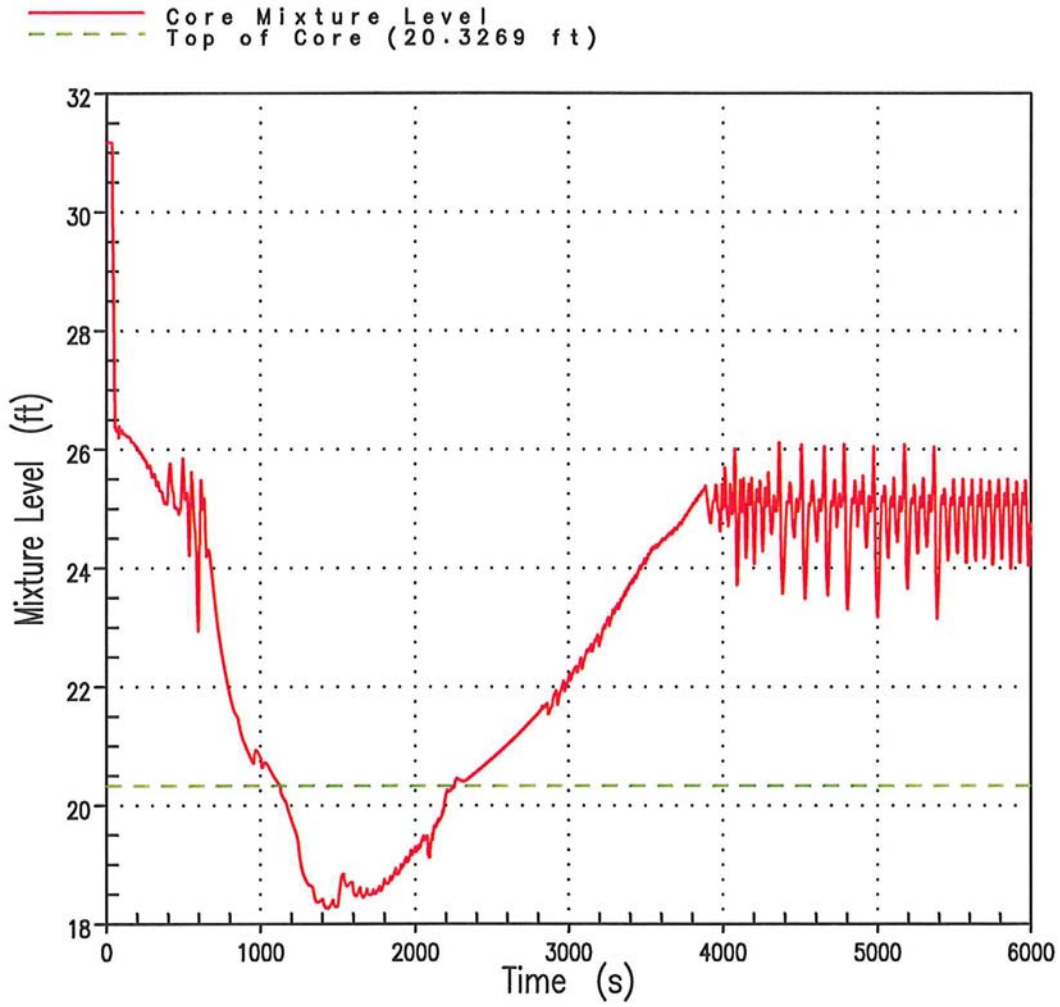
Figure 14.3.1-36 TOP CORE EXIT VAPOR TEMPERATURE - 2 INCH BREAK
POINT BEACH UNIT 1
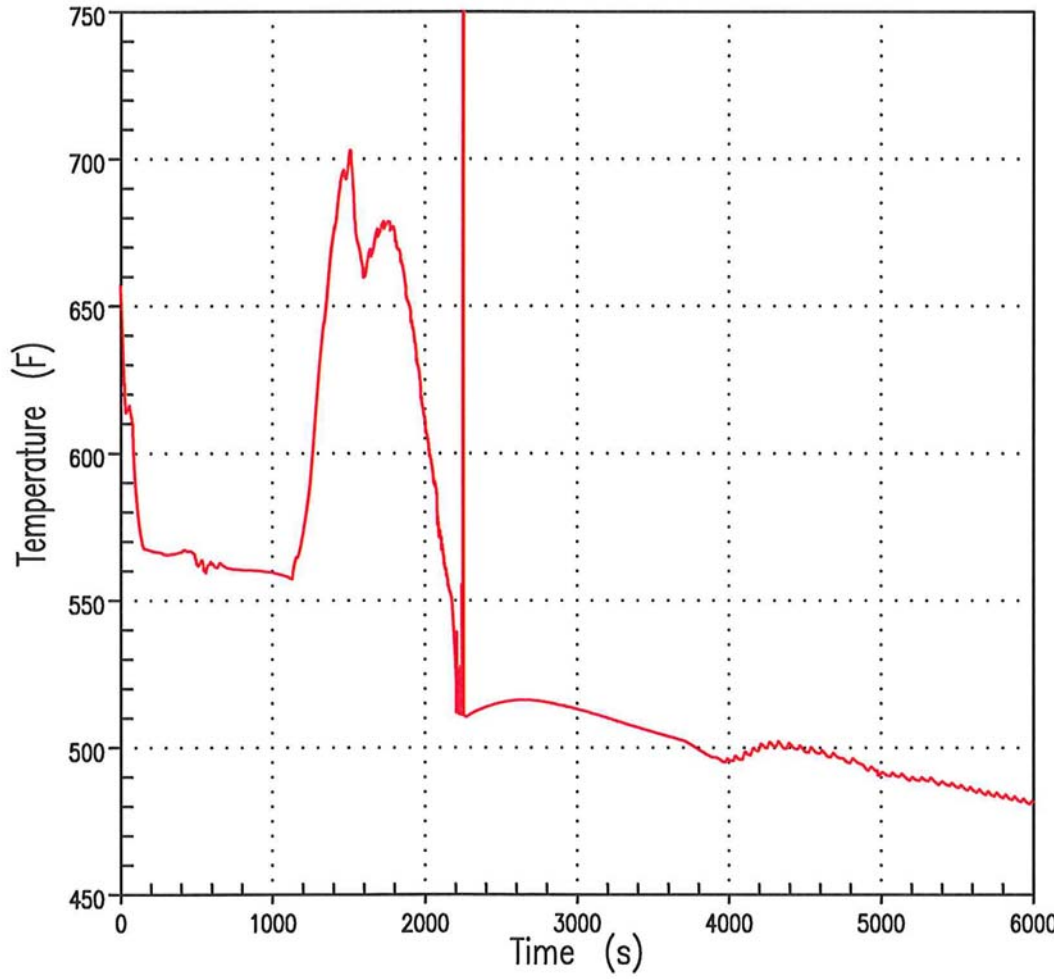
Figure 14.3.1-37 CLADDING TEMPERATURE TRANSIENT AT PCT ELEVATION - 2 INCH
BREAK POINT BEACH UNIT 1
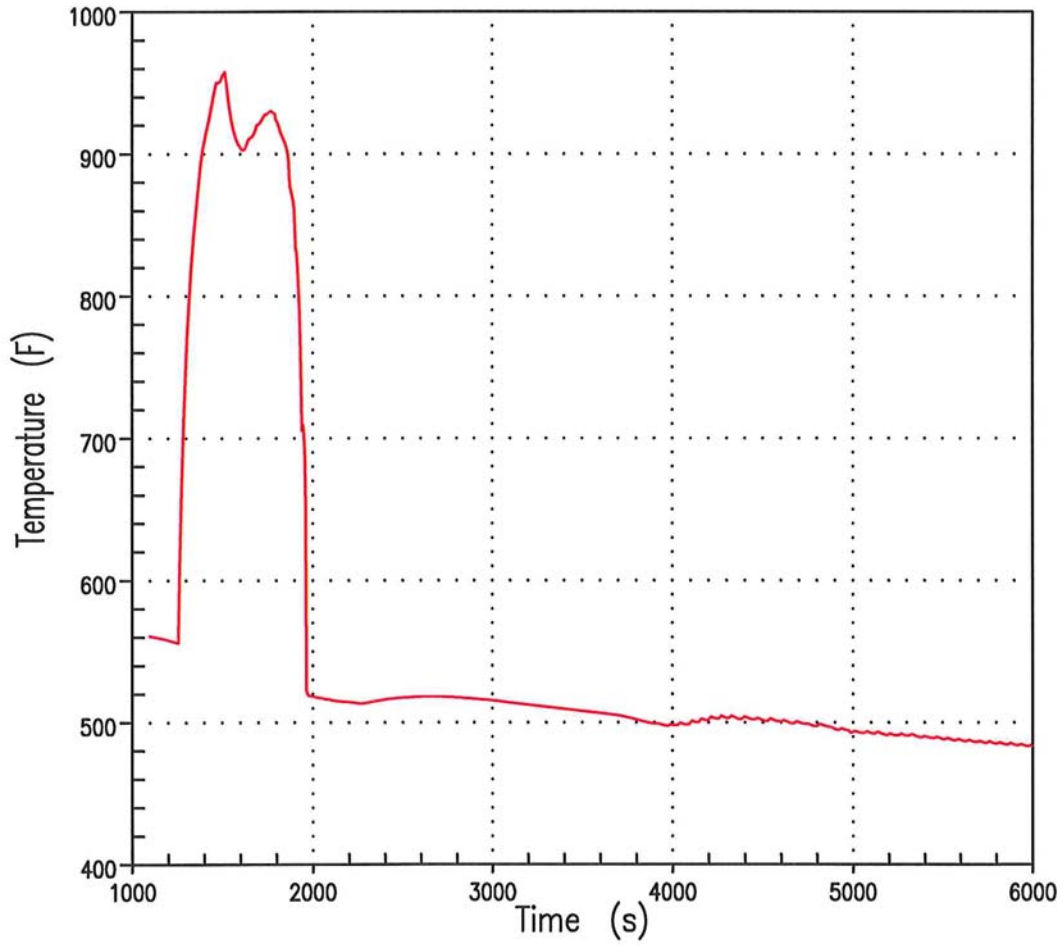
Figure 14.3.1-38 LOCAL ZRO2 THICKNESS AT MAXIMUM LOCAL ZRO2 ELEVATION -
2 INCH BREAK POINT BEACH UNIT 1
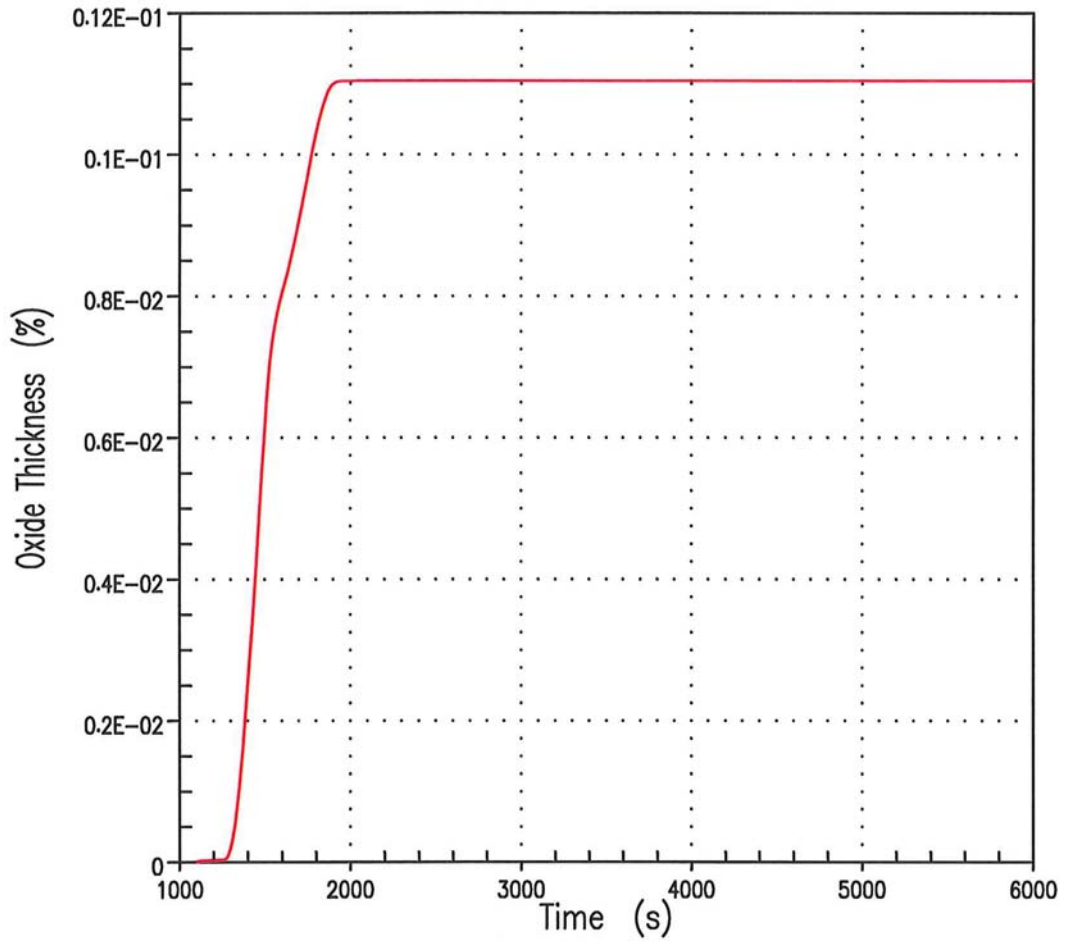
Figure 14.3.1-39 REACTOR COOLANT SYSTEM PRESSURE - 2 INCH BREAK
POINT BEACH UNIT 2
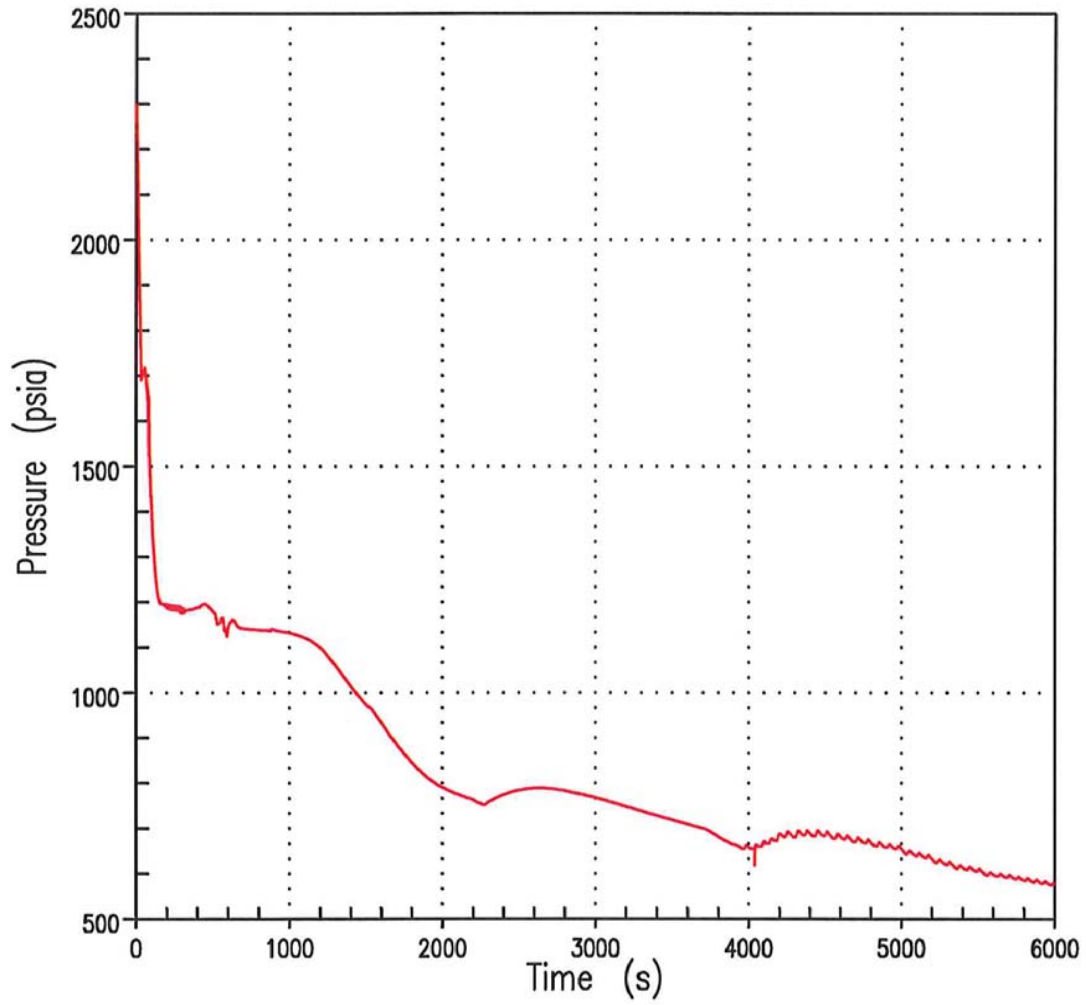
Figure 14.3.1-40 CORE MIXTURE LEVEL AND TOP OF CORE - 2 INCH BREAK POINT
BEACH UNIT 2
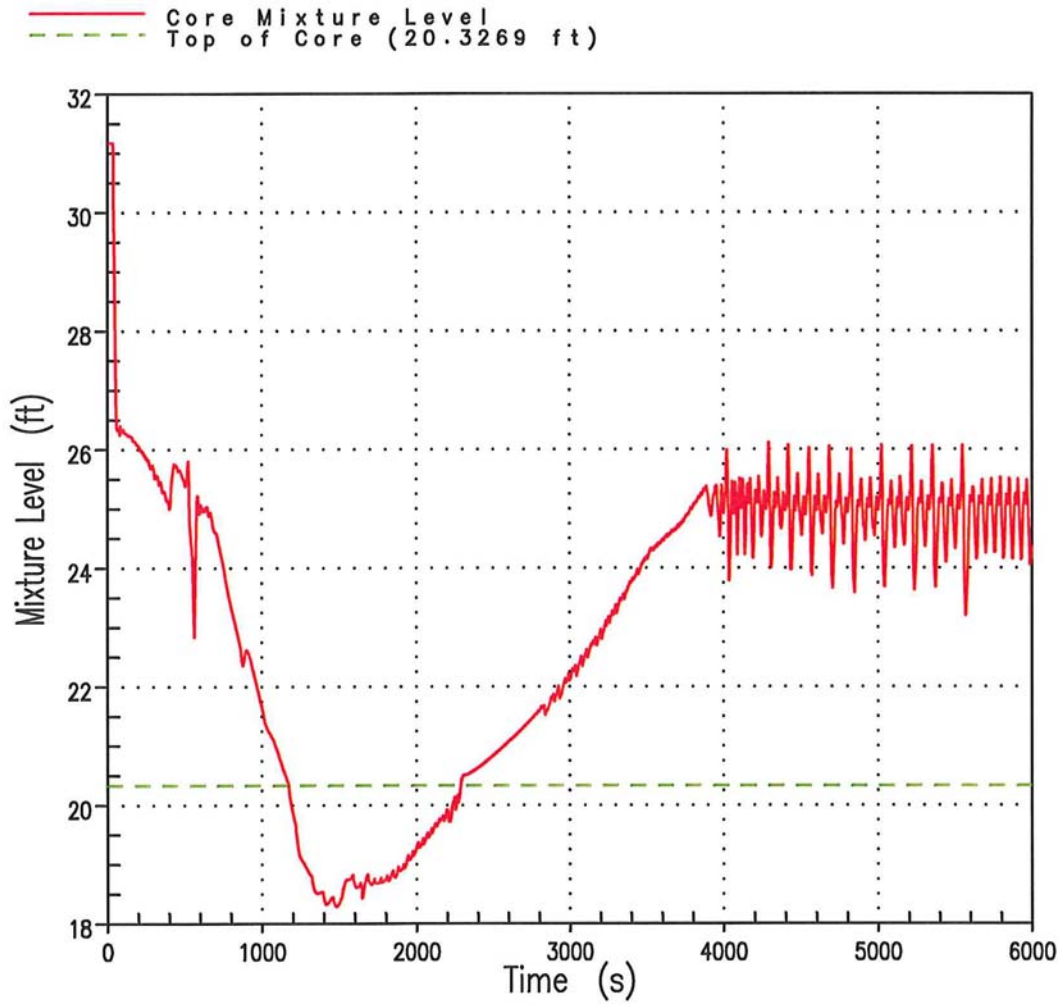
Figure 14.3.1-41 TOP CORE EXIT VAPOR TEMPERATURE - 2 INCH BREAK
POINT BEACH UNIT 2
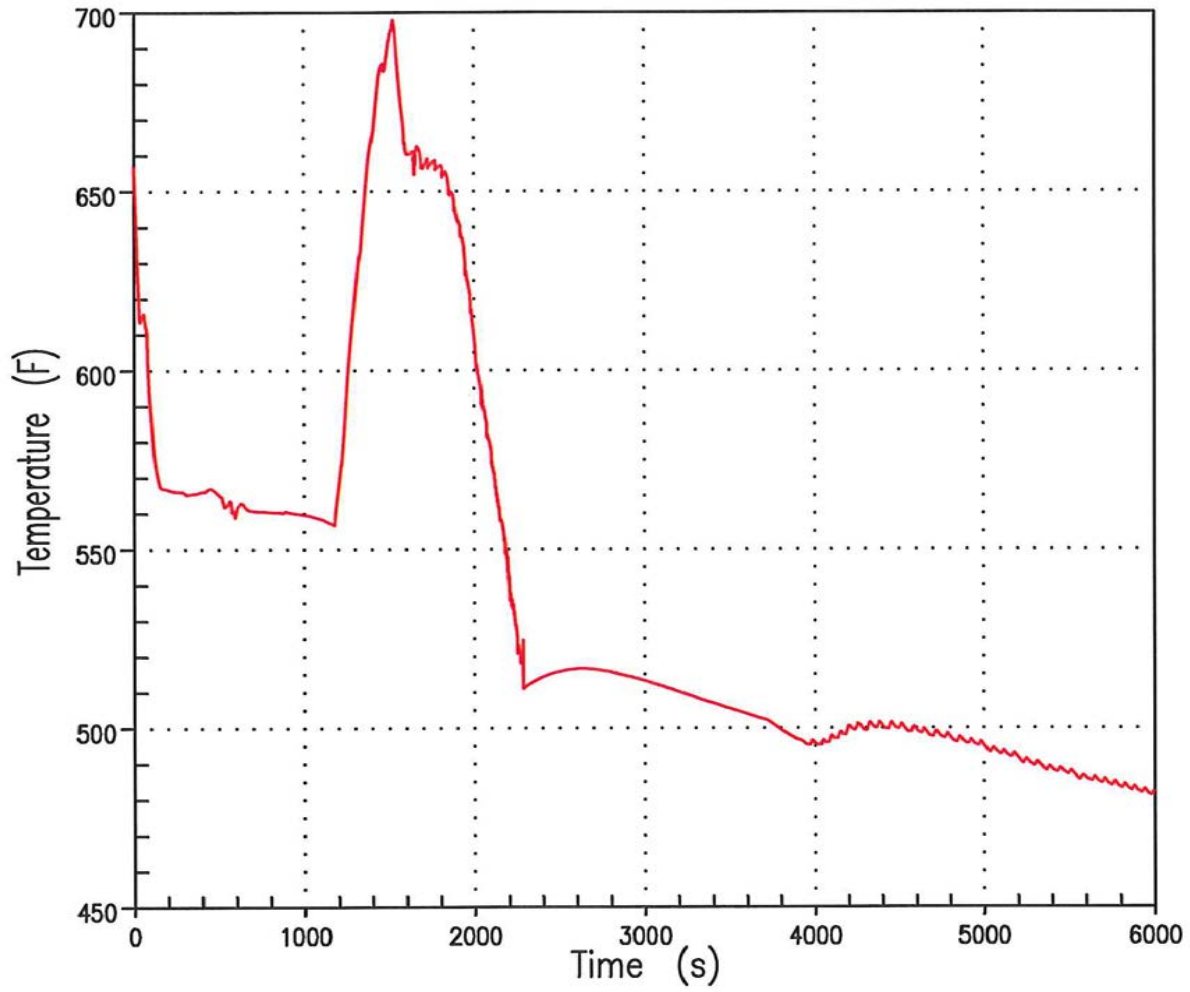
Figure 14.3.1-42 CLADDING TEMPERATURE TRANSIENT AT PCT ELEVATION - 2 INCH BREAK POINT BEACH UNIT 2
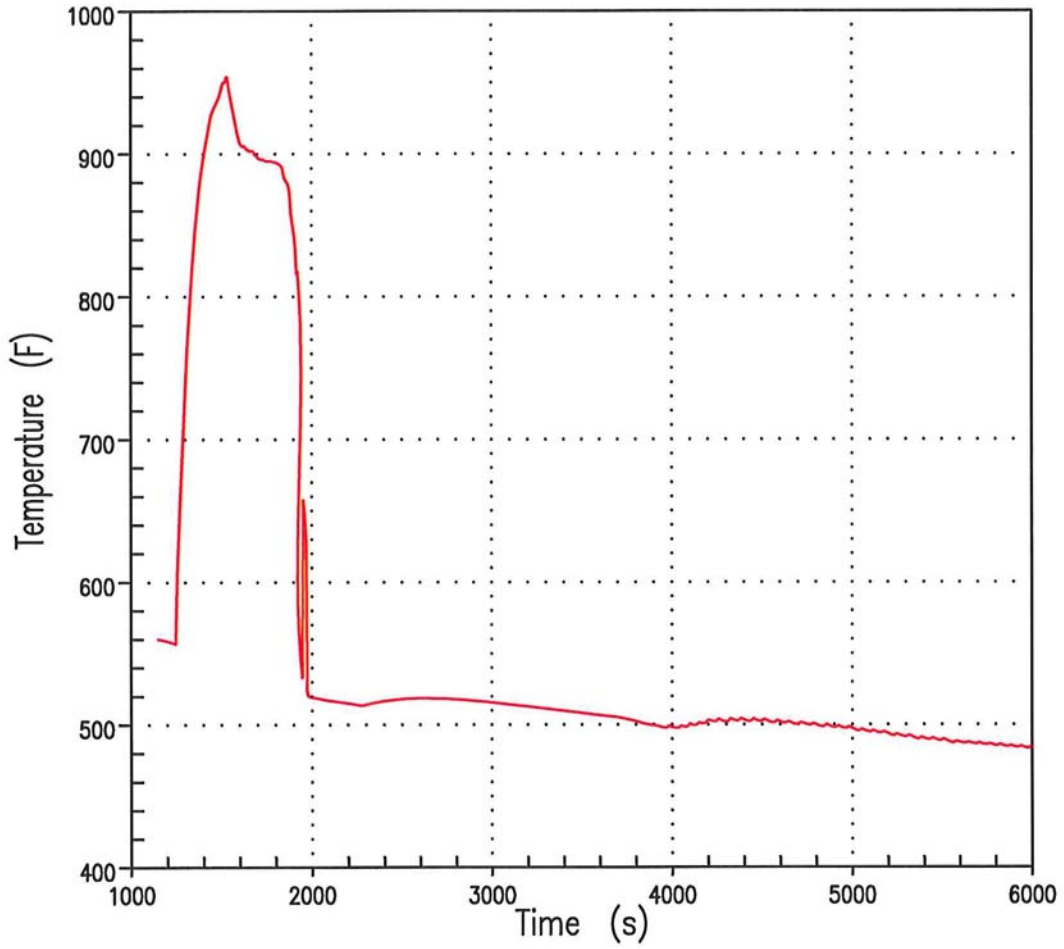
Figure 14.3.1-43 LOCAL ZRO2 THICKNESS AT MAXIMUM LOCAL ZRO2 ELEVATION -
2 INCH BREAK POINT BEACH UNIT 2
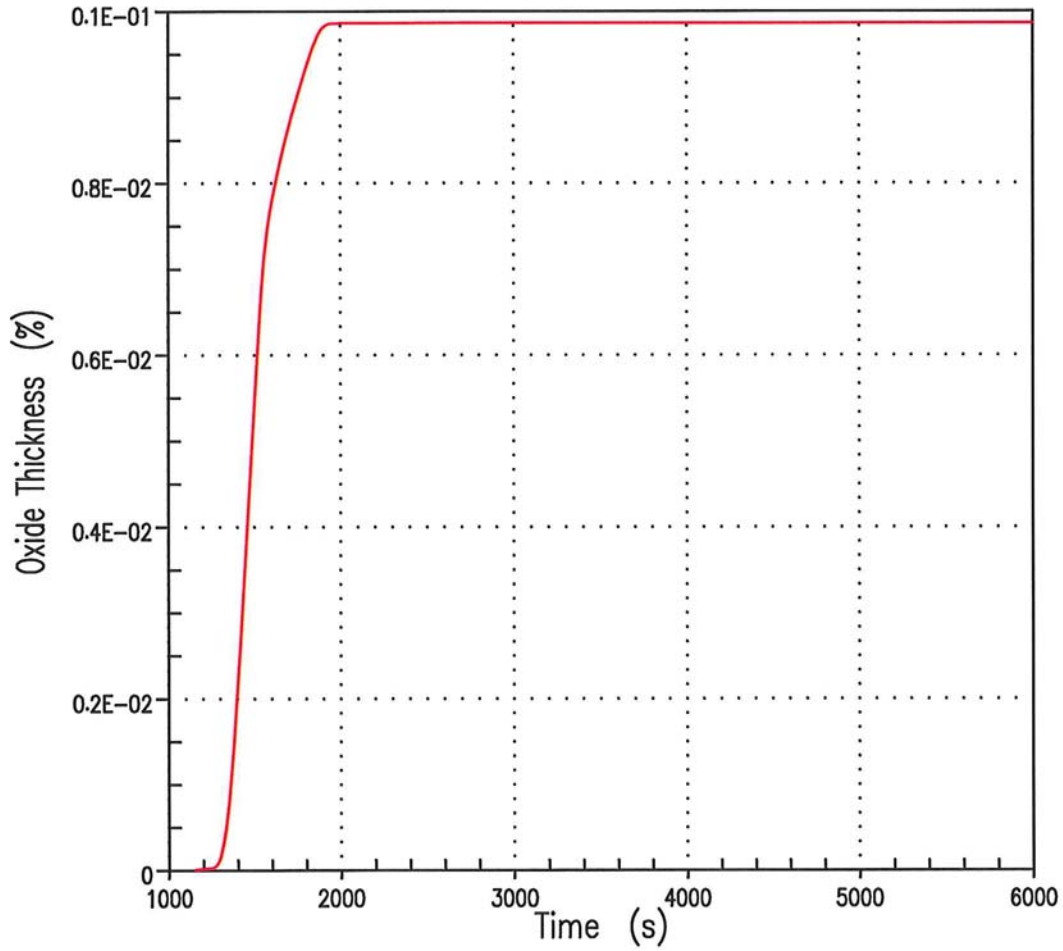
Figure 14.3.1-44 REACTOR COOLANT SYSTEM PRESSURE - 4 INCH BREAK
POINT BEACH UNIT 1
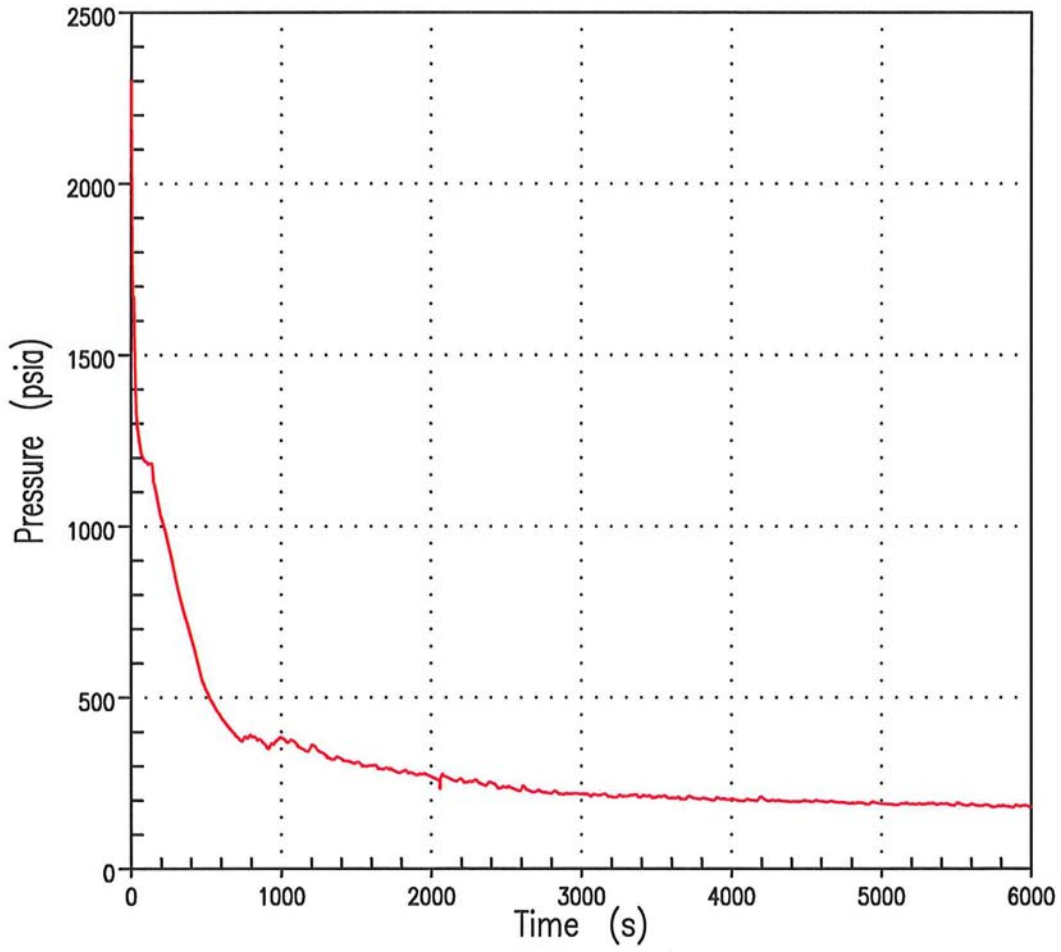
Figure 14.3.1-45 CORE MIXTURE LEVEL AND TOP OF CORE - 4 INCH BREAK
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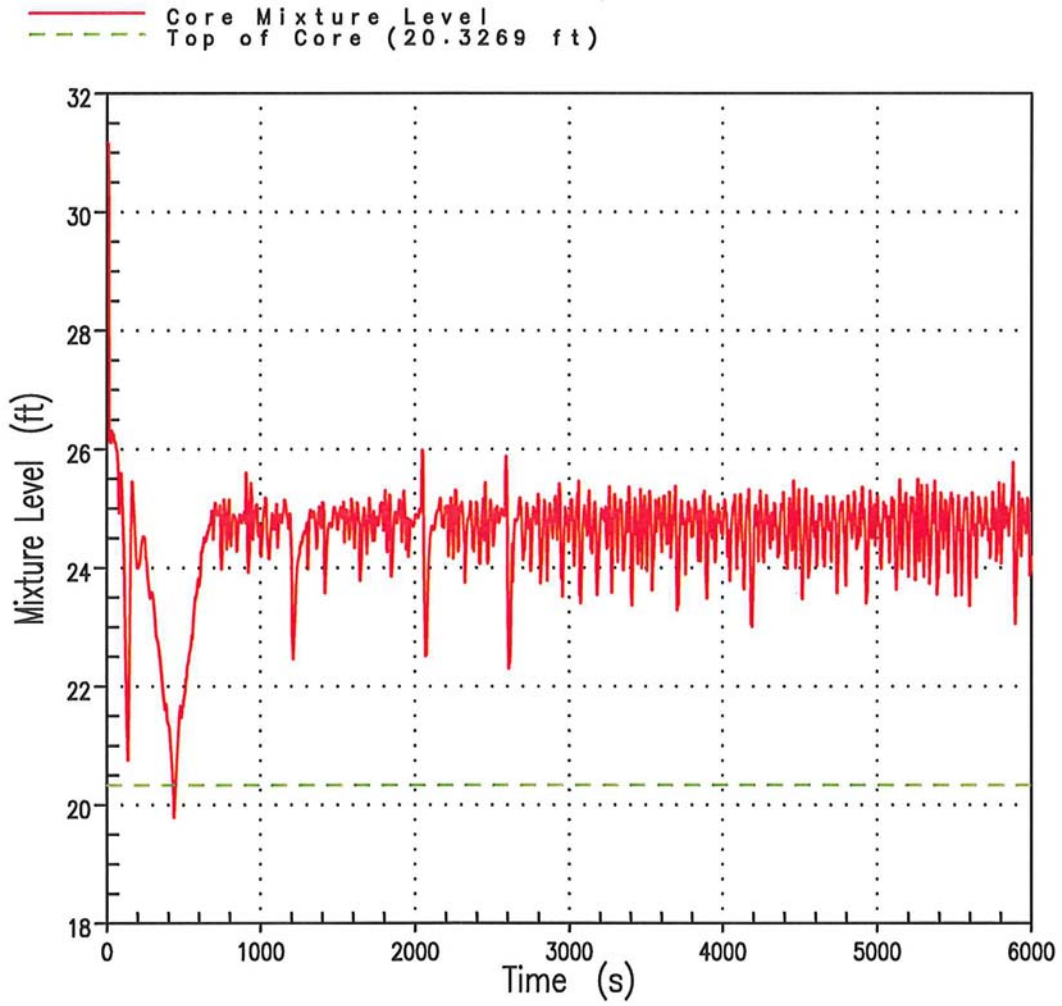
Figure 14.3.1-46 TOP CORE EXIT VAPOR TEMPERATURE - 4 INCH BREAK
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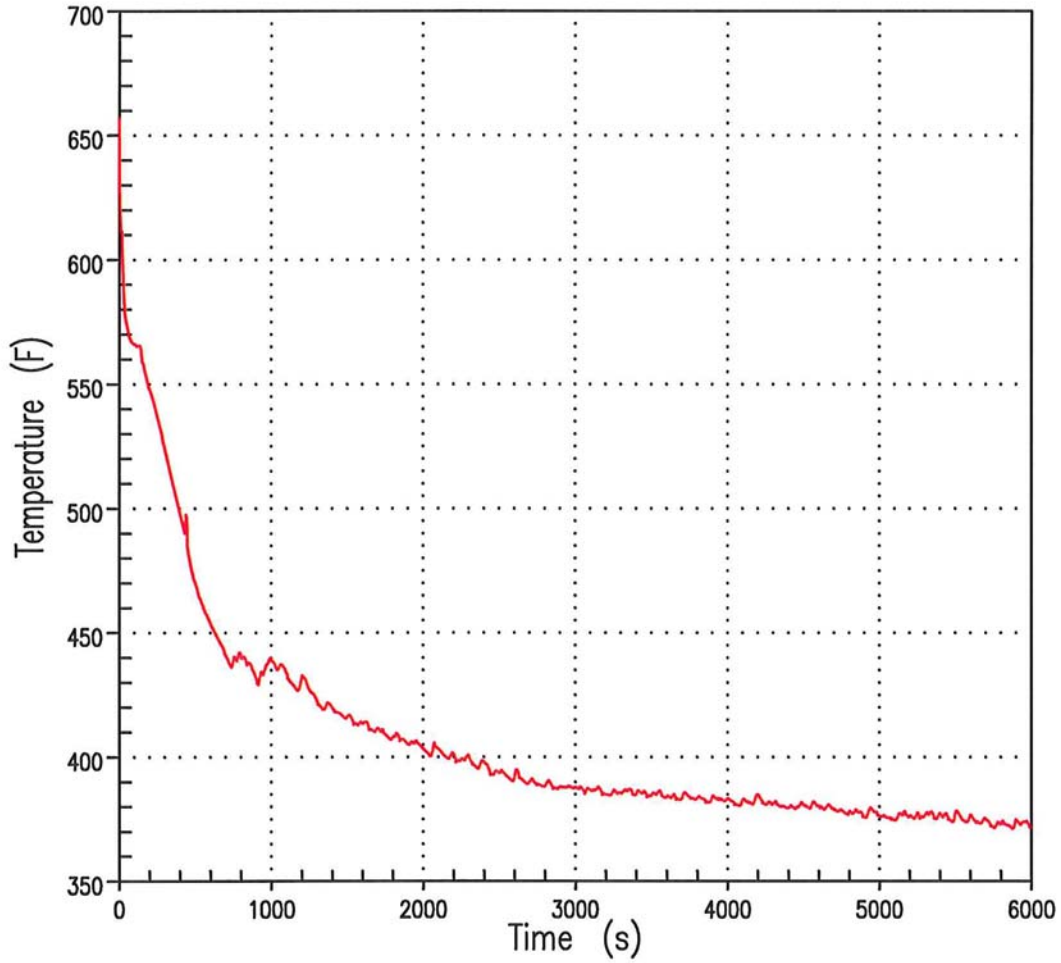
Figure 14.3.1-47 CLADDING TEMPERATURE TRANSIENT AT PCT ELEVATION - 4 INCH
BREAK POINT BEACH UNIT 1
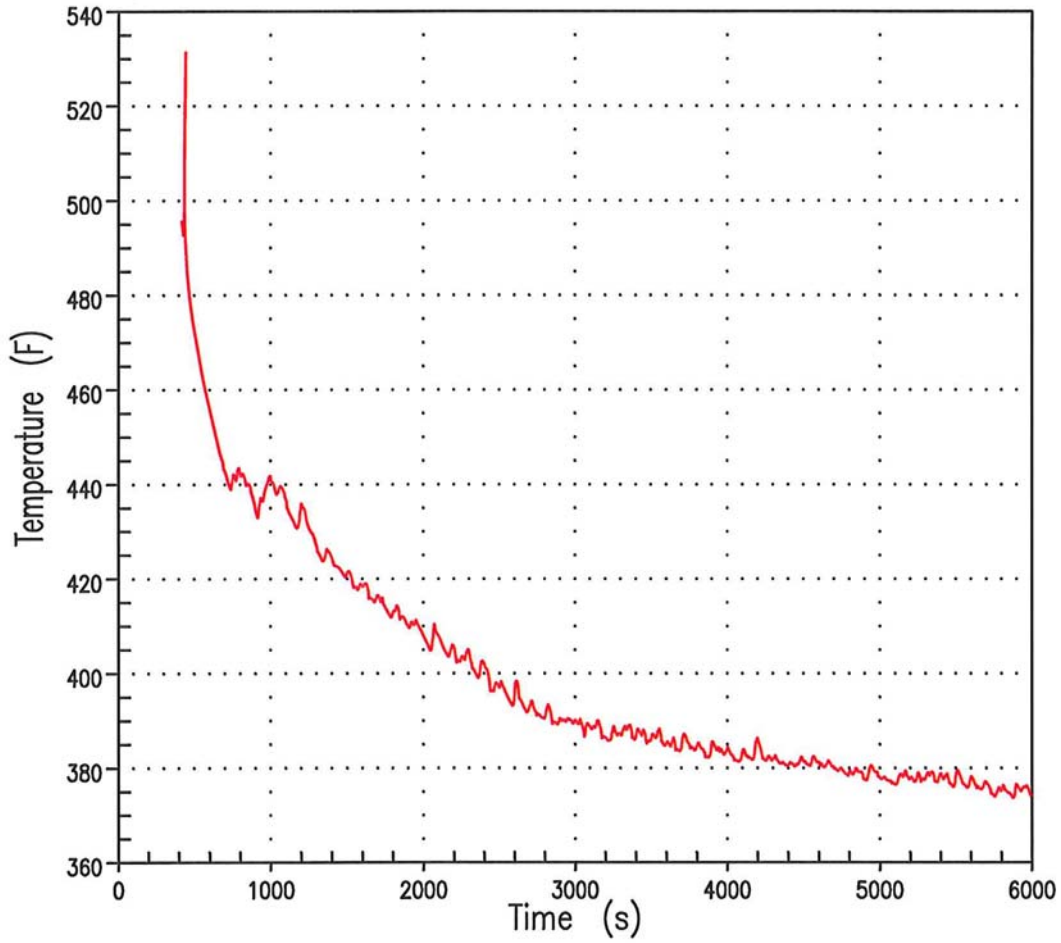
Figure 14.3.1-48 LOCAL ZRO2 THICKNESS AT MAXIMUM LOCAL ZRO2 ELEVATION -
4 INCH BREAK POINT BEACH UNIT 1
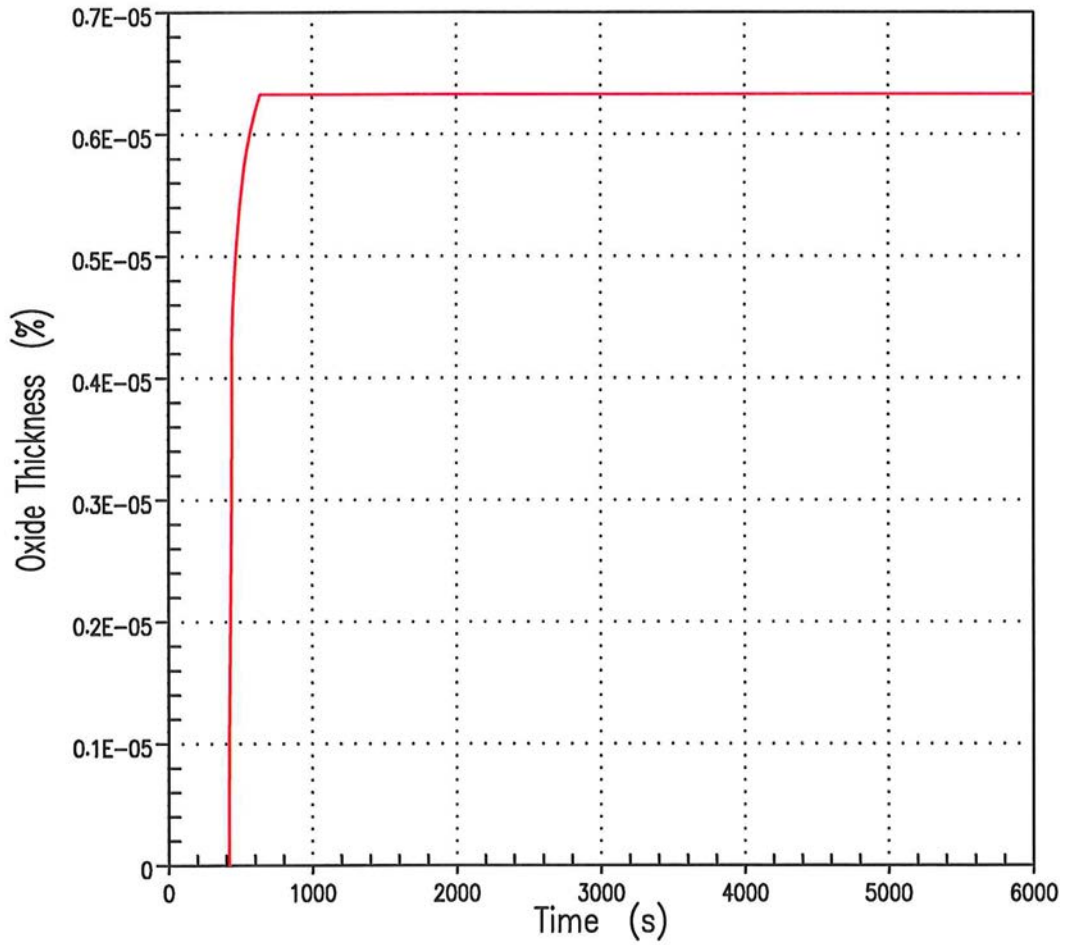
Figure 14.3.1-49 REACTOR COOLANT SYSTEM PRESSURE - 4 INCH BREAK
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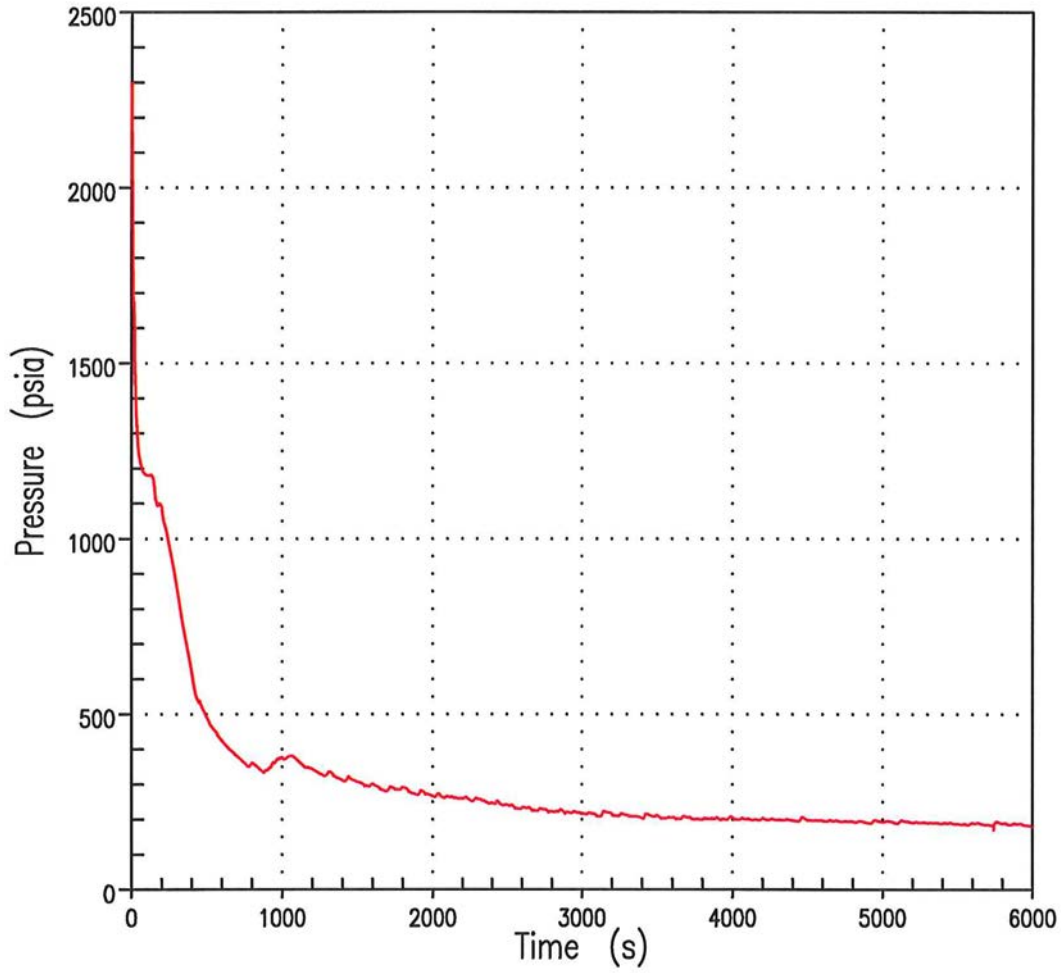
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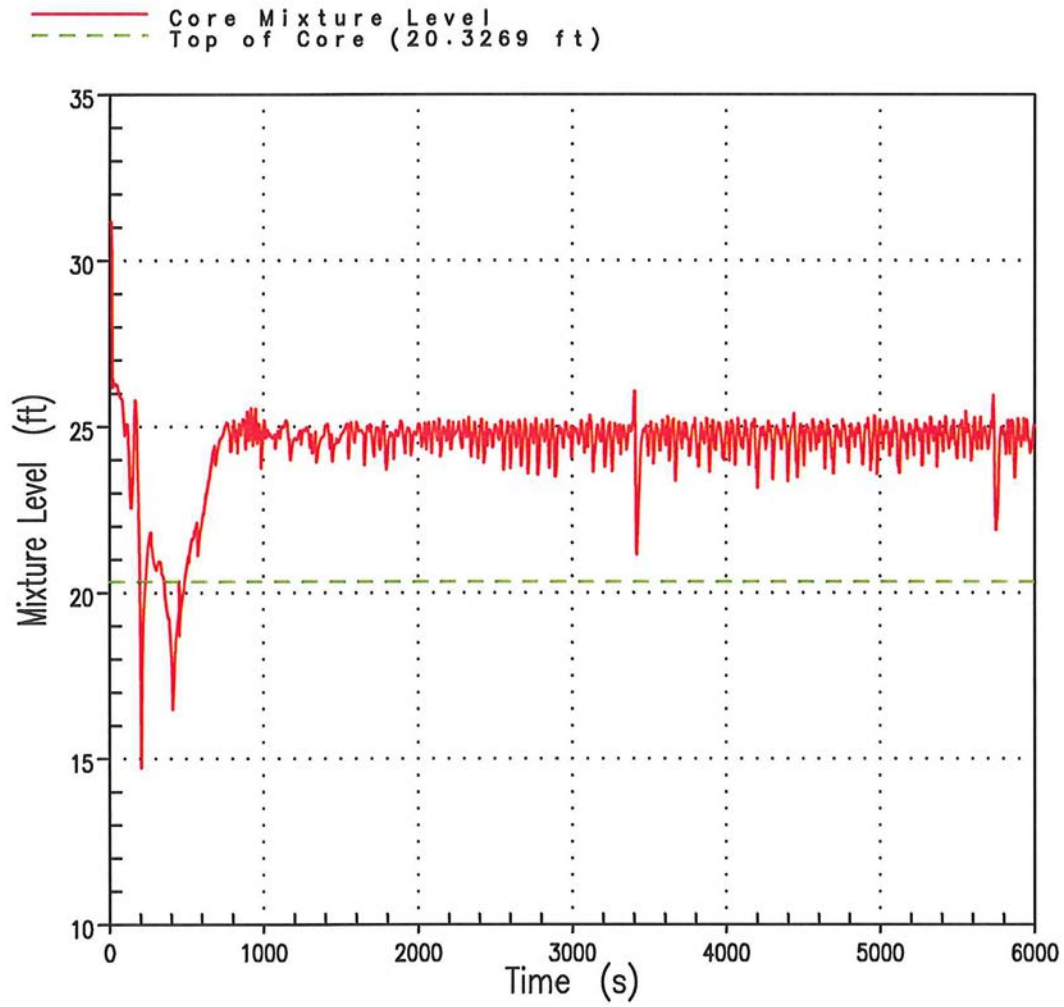
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UNIT 2
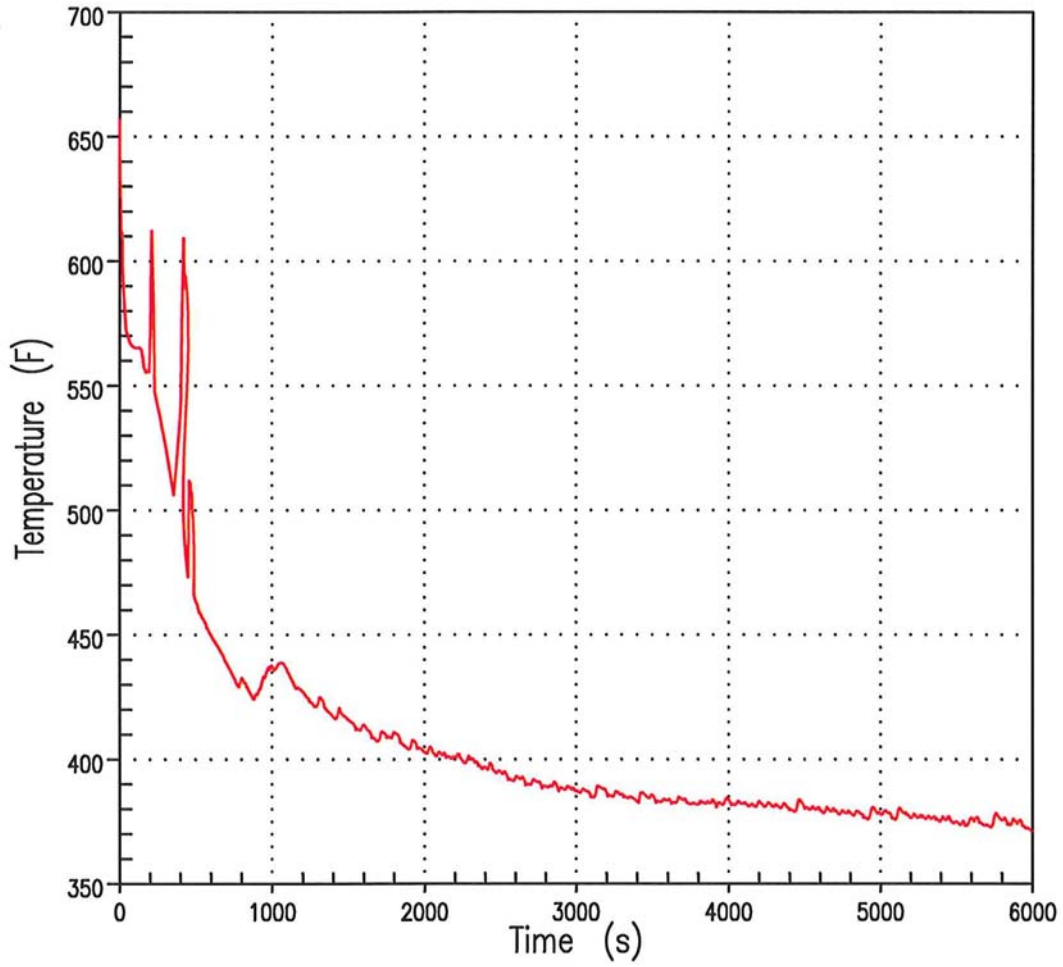
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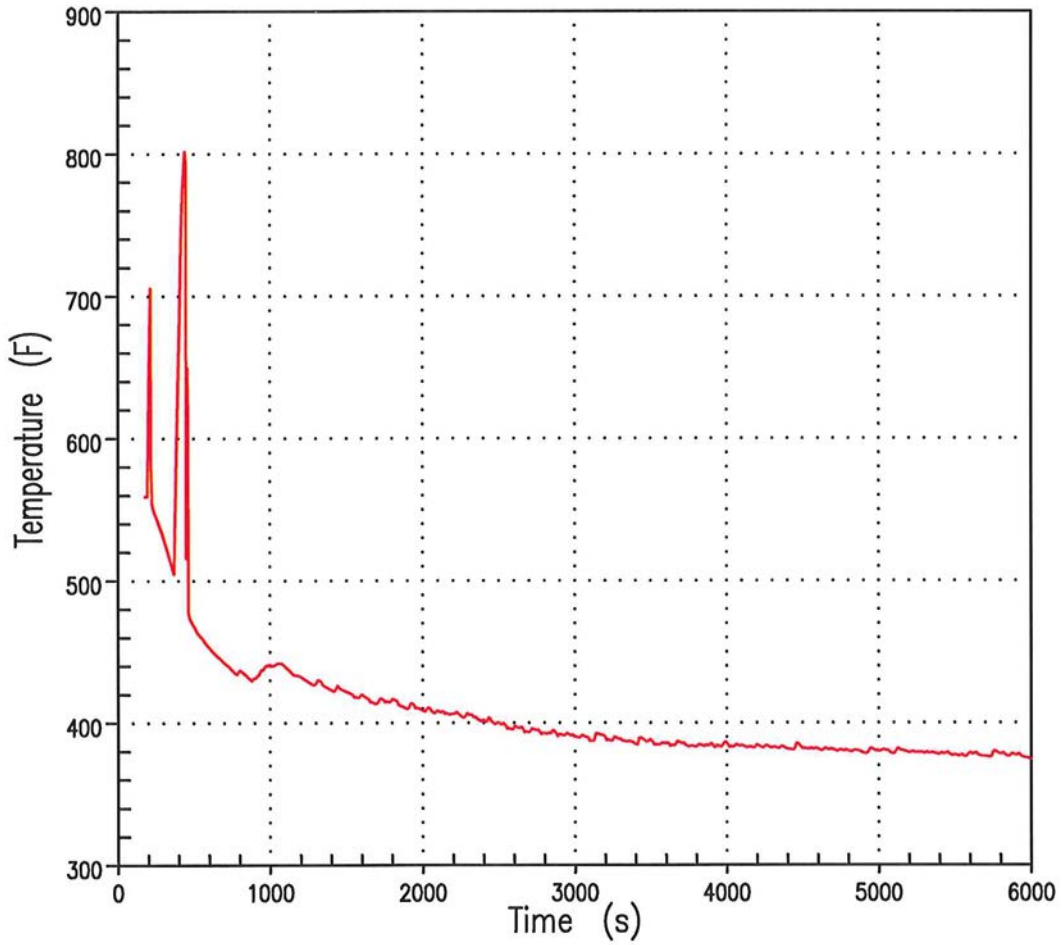
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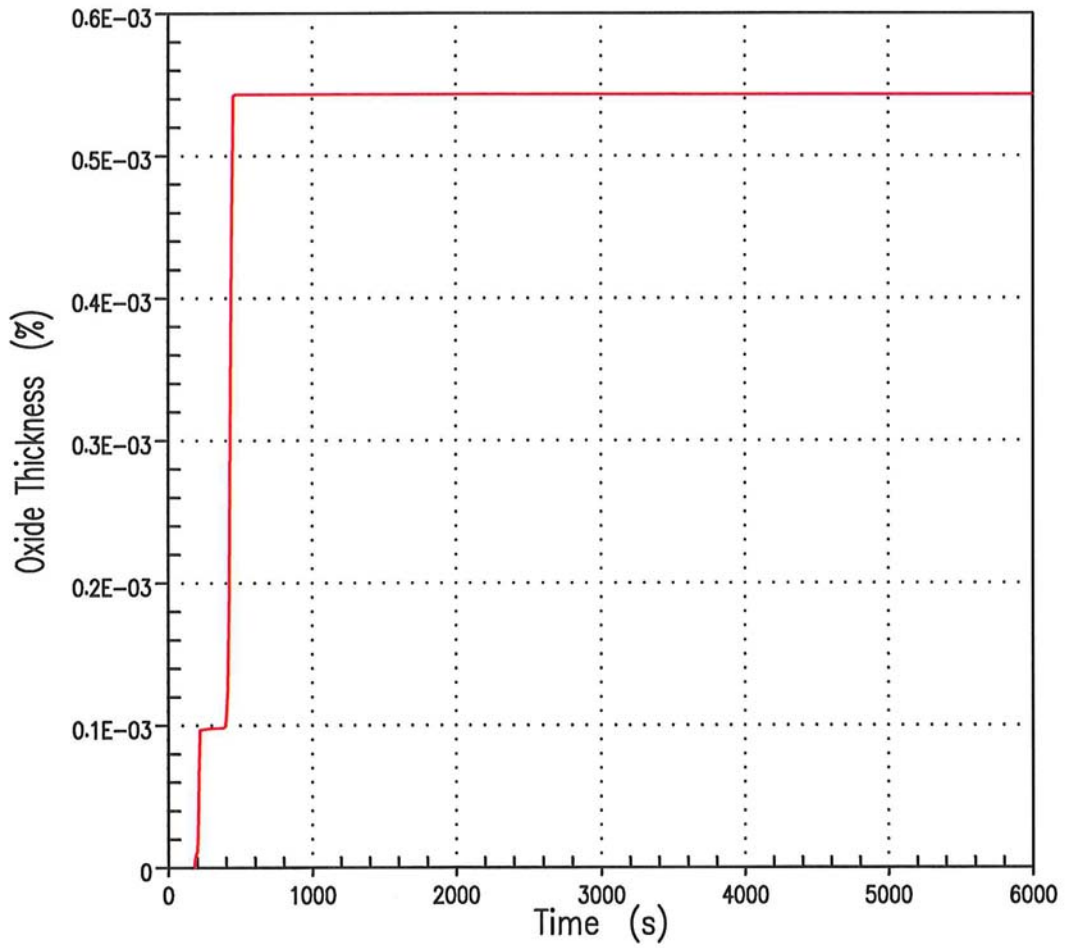
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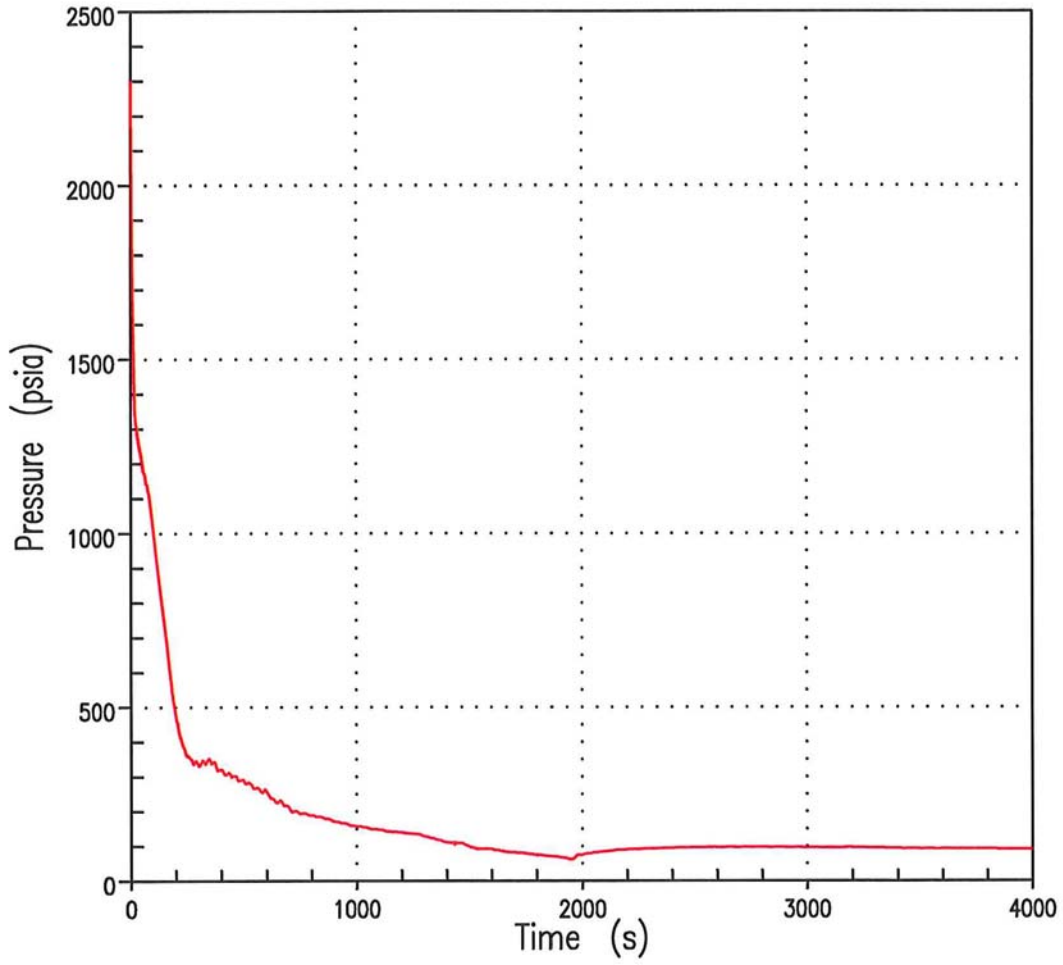
Figure 14.3.1-55 CORE MIXTURE LEVEL AND TOP OF CORE - 6 INCH BREAK
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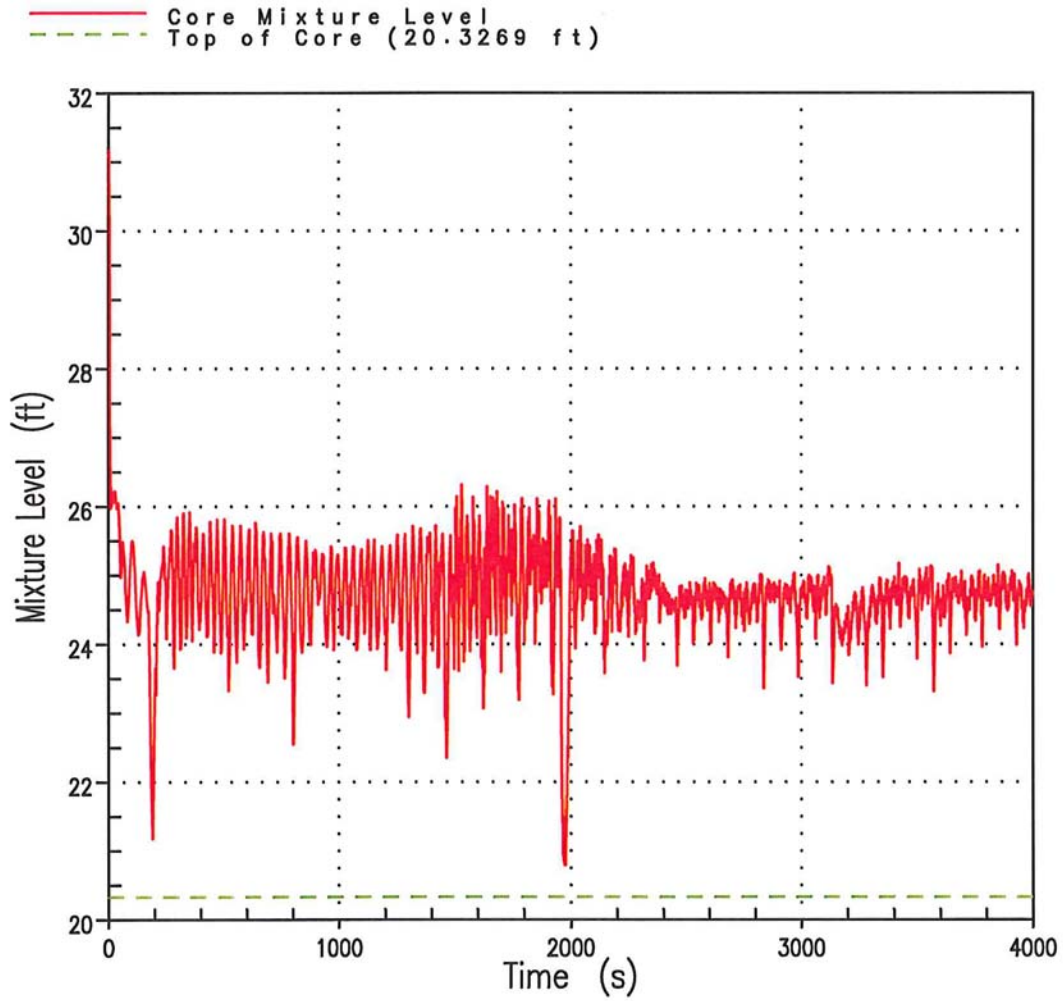
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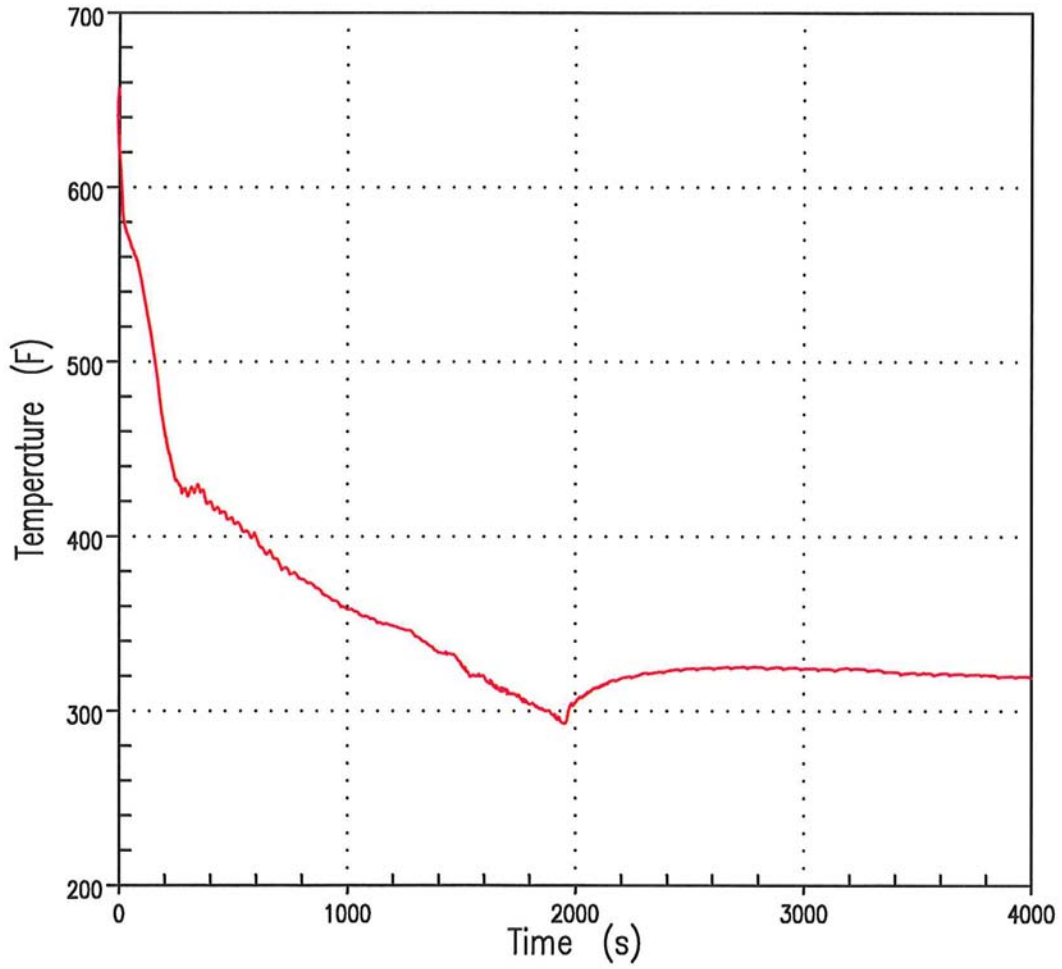
Figure 14.3.1-57 REACTOR COOLANT SYSTEM PRESSURE - 6 INCH BREAK
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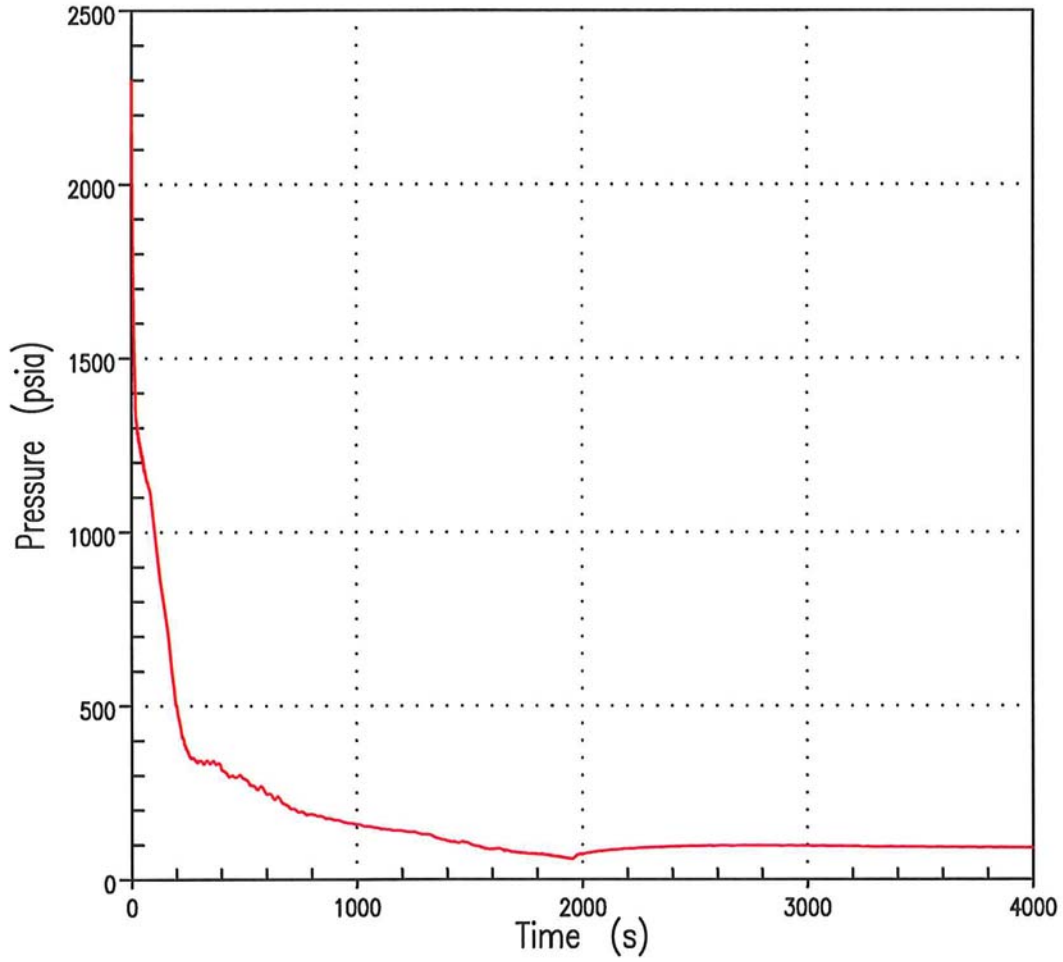
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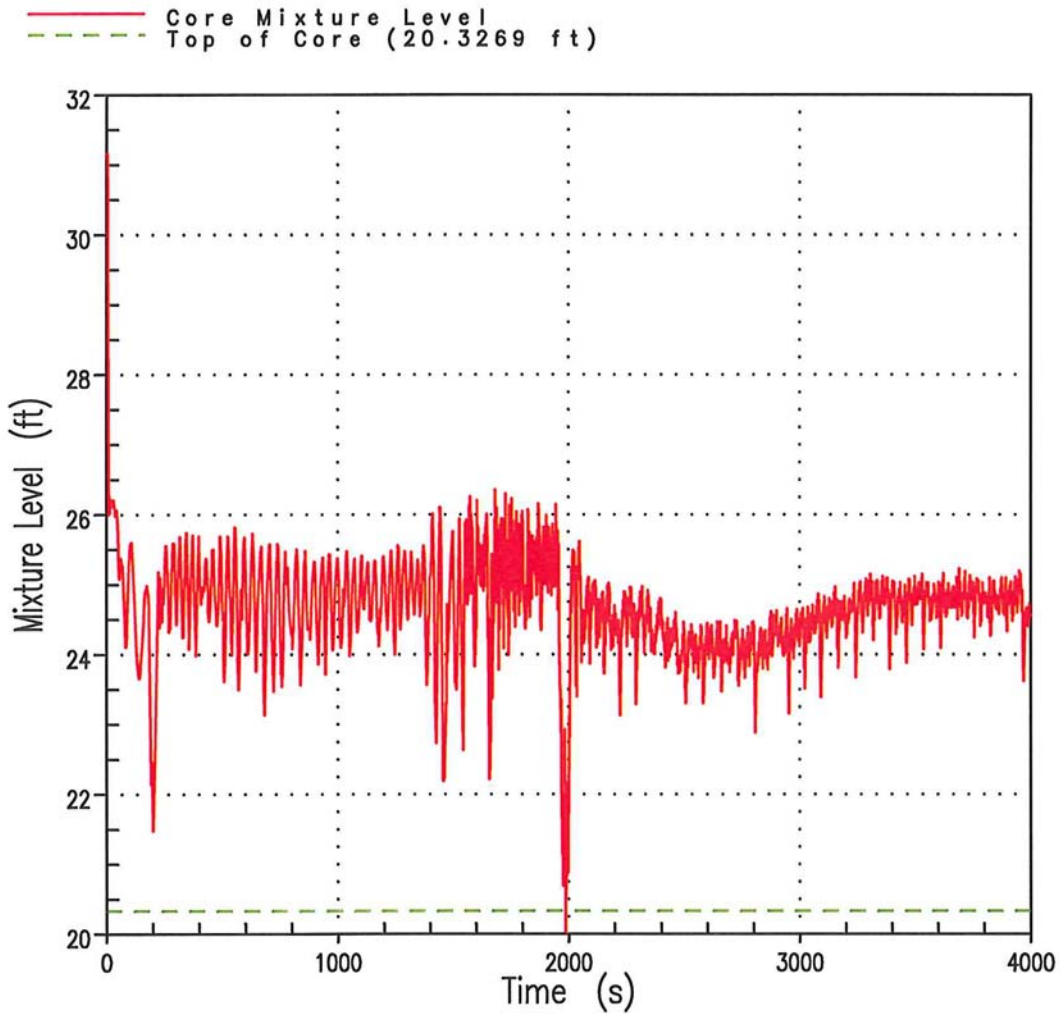
Figure 14.3.1-59 TOP CORE EXIT VAPOR TEMPERATURE - 6 INCH BREAK
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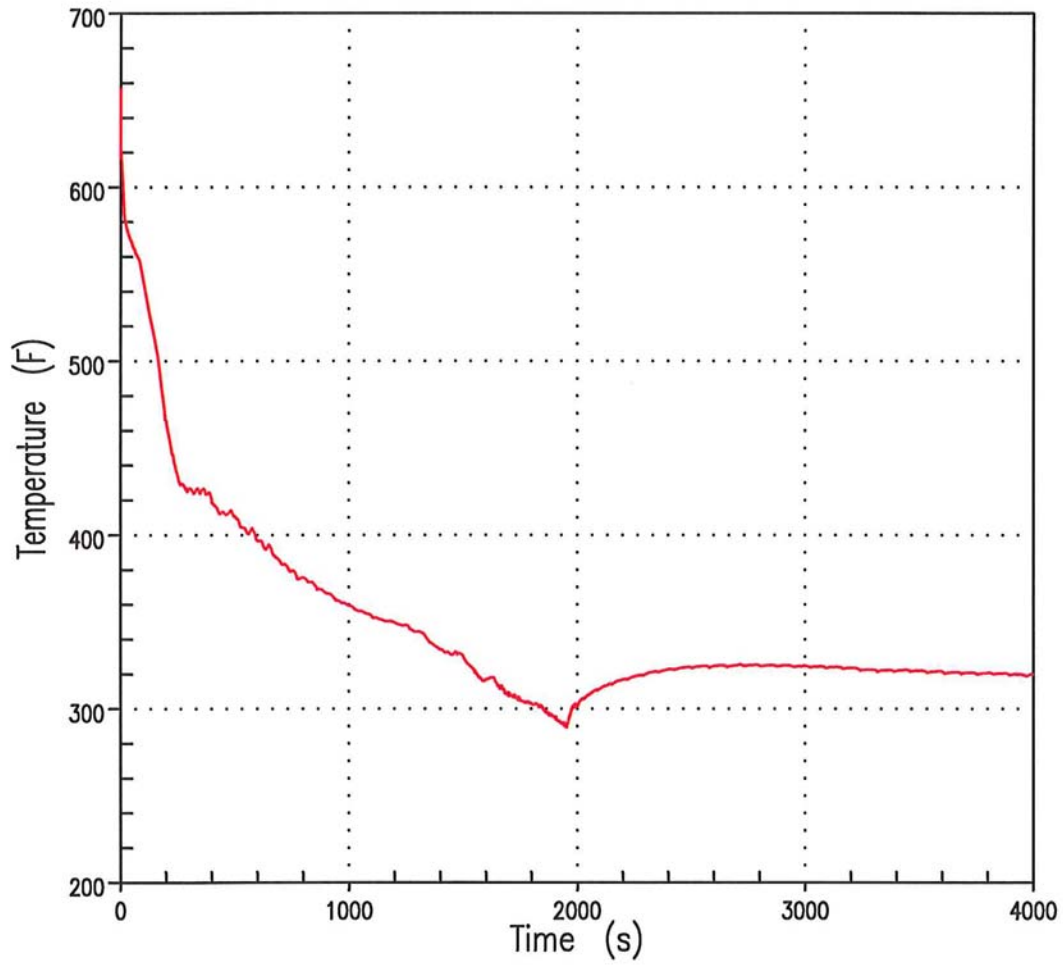
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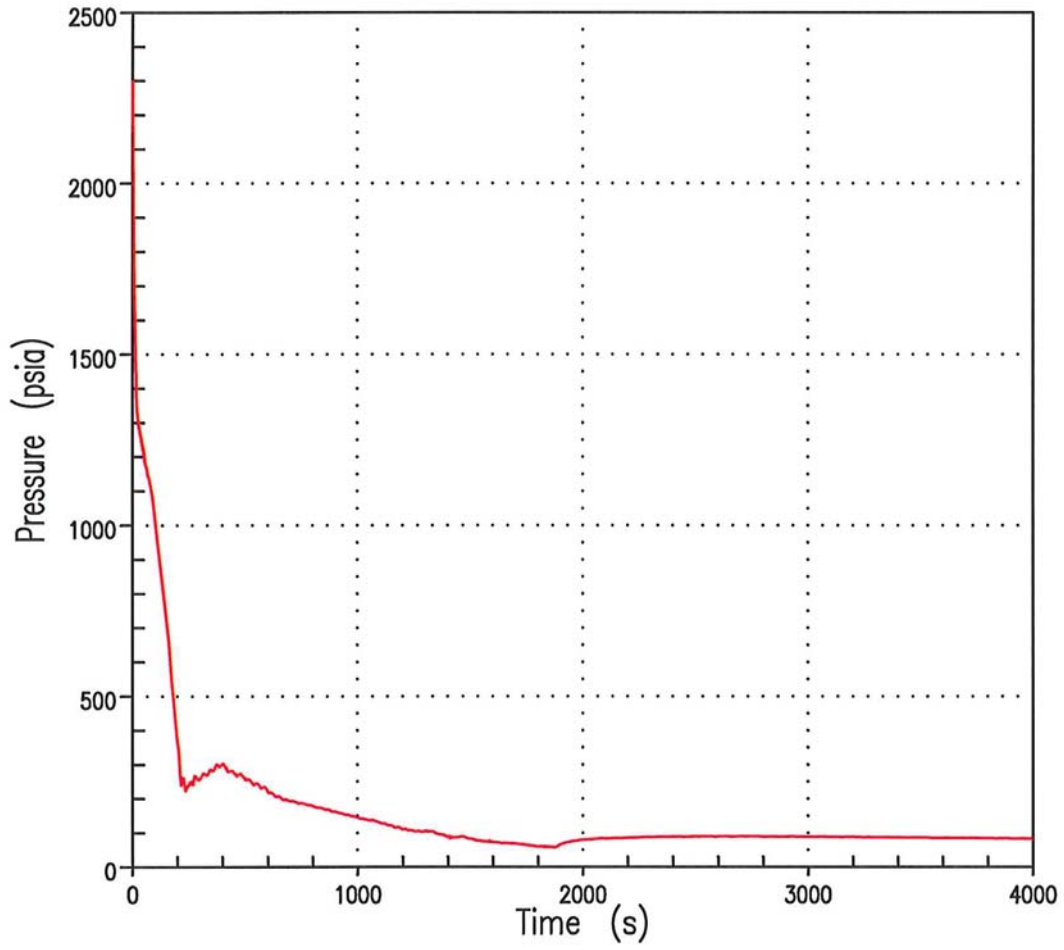
Figure 14.3.1-61 CORE MIXTURE LEVEL AND TOP OF CORE - 8.75 INCH BREAK
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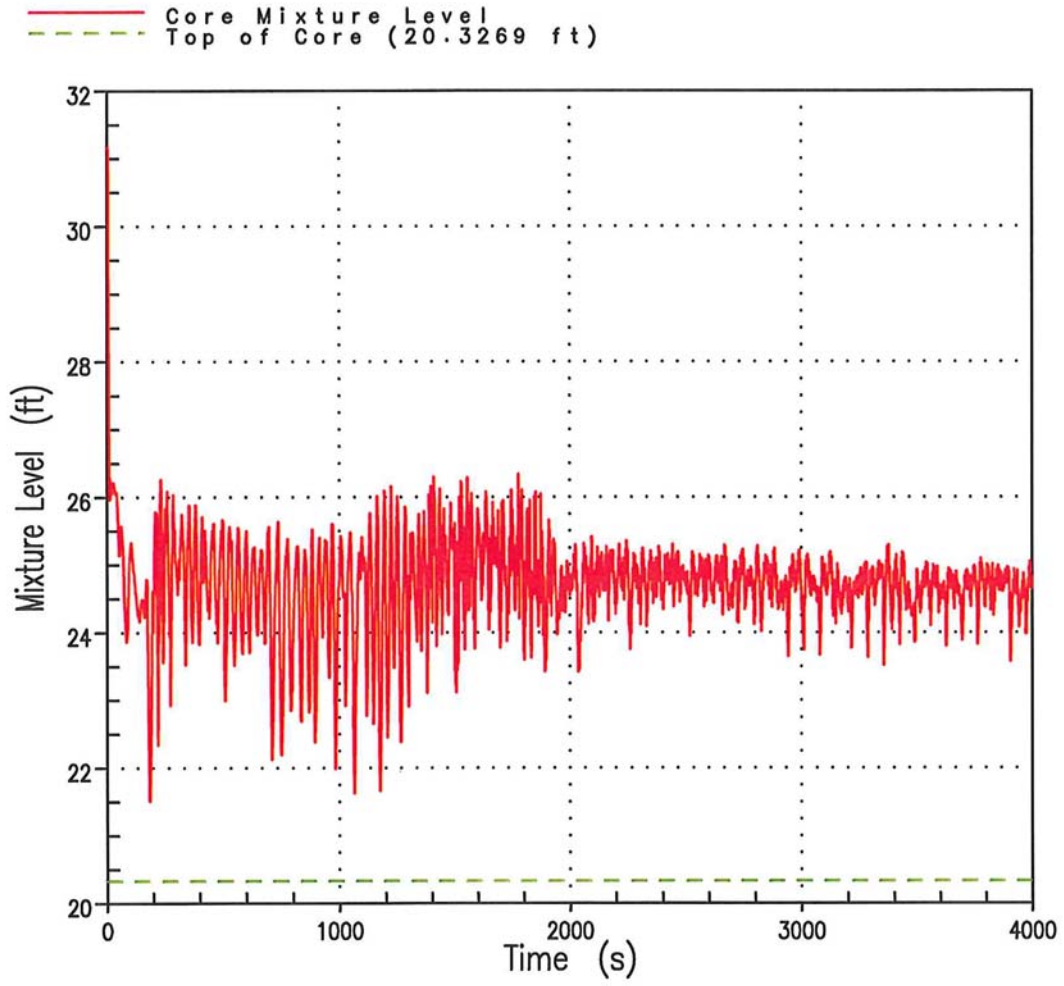
Figure 14.3.1-62 TOP CORE EXIT VAPOR TEMPERATURE - 8.75 INCH BREAK
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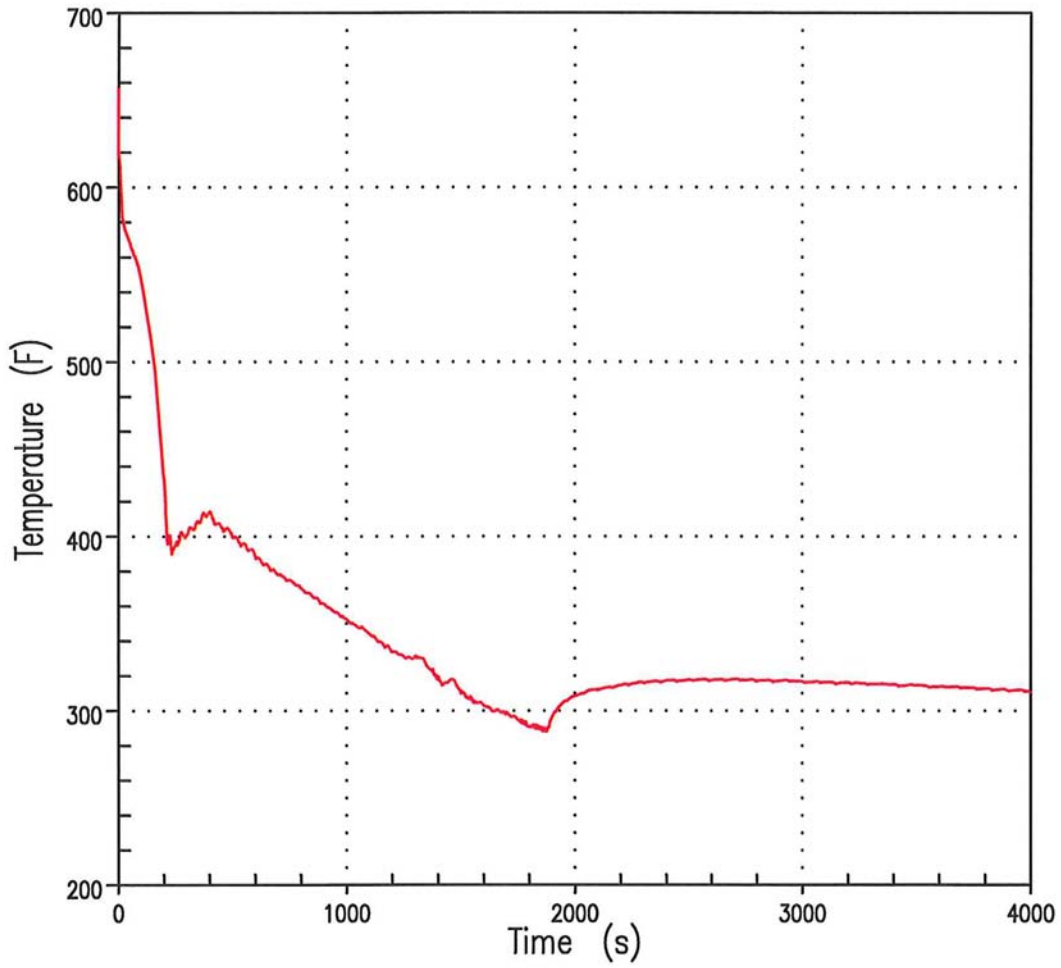
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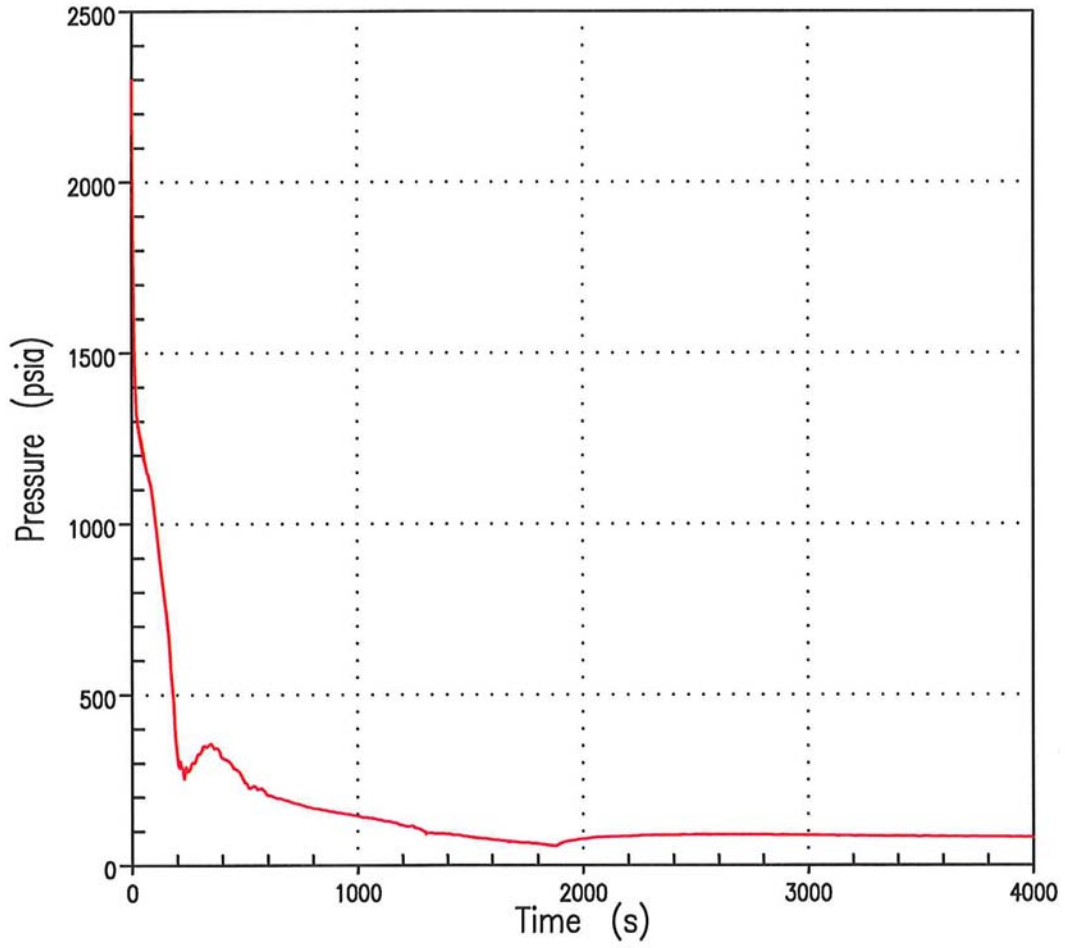
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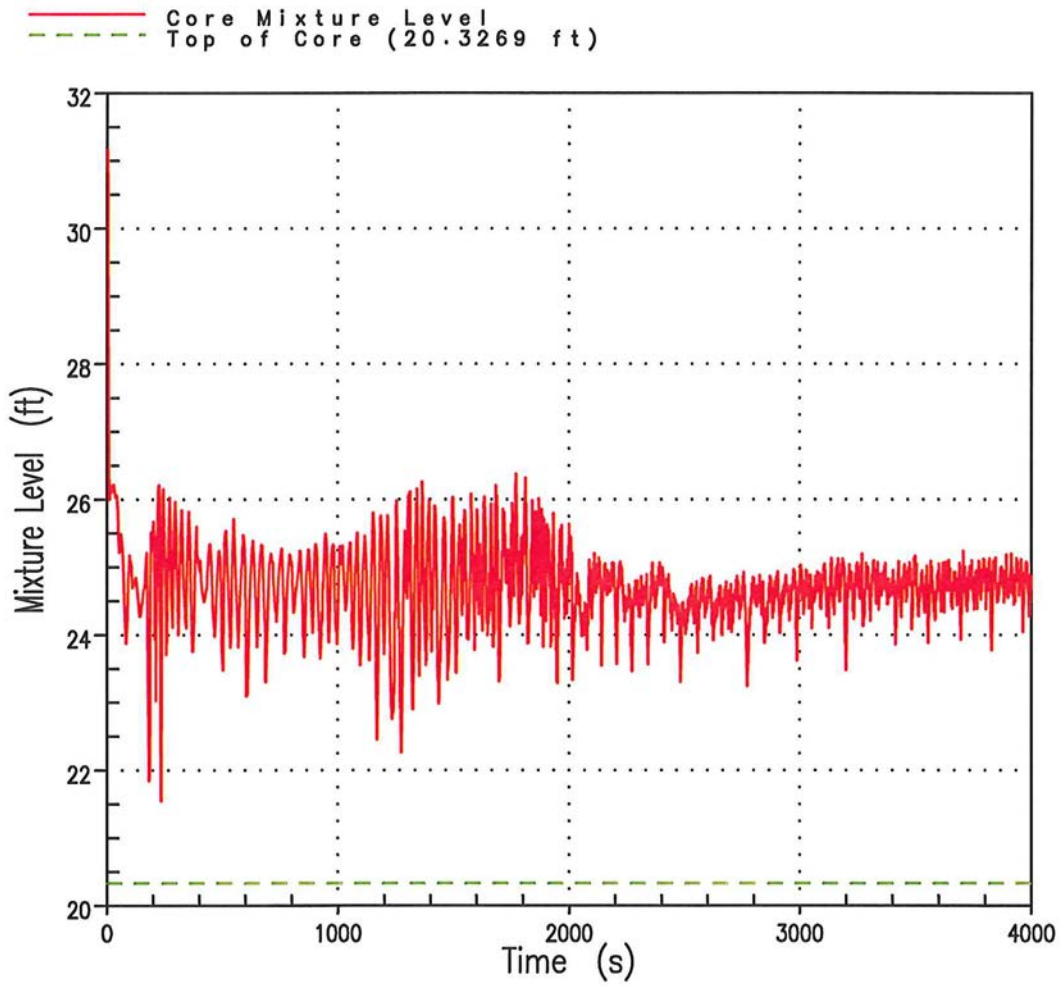
Figure 14.3.1-65 TOP CORE EXIT VAPOR TEMPERATURE - 8.75 INCH BREAK
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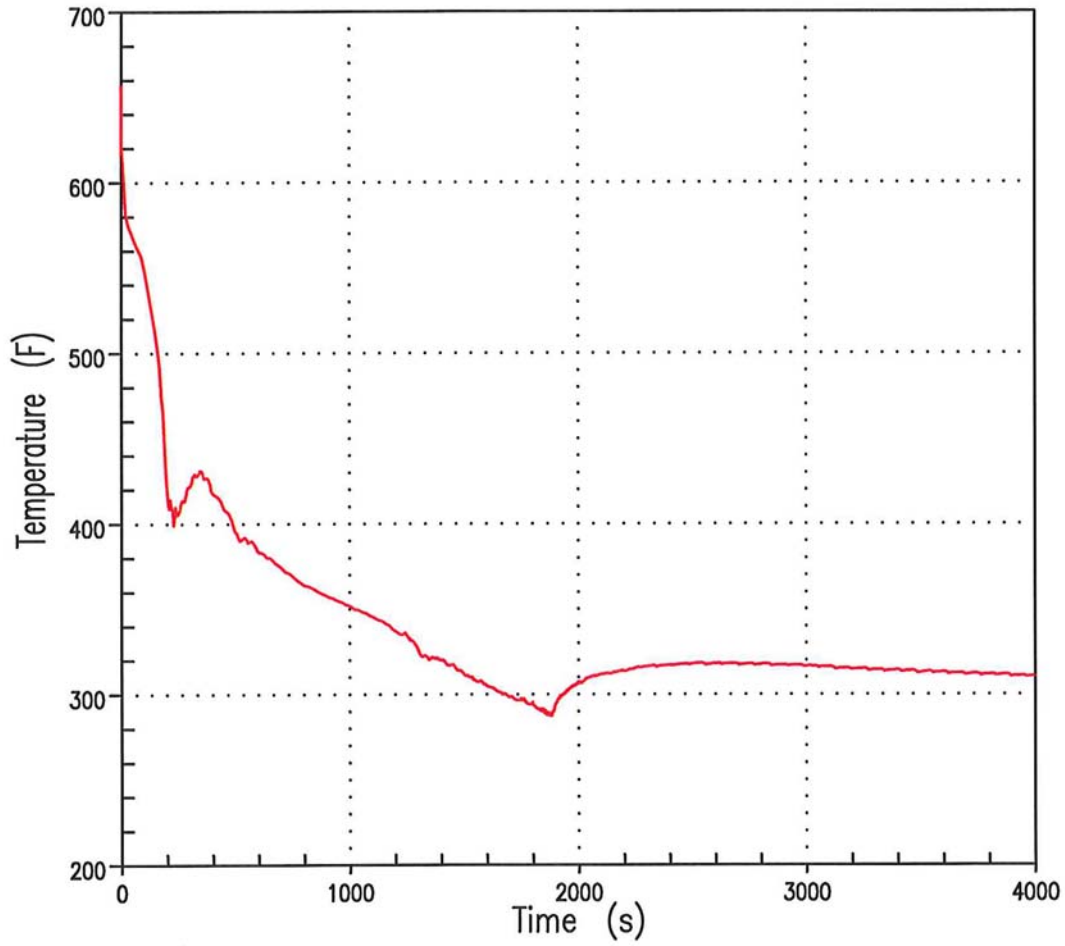
14.3.2   LARGE BREAK LOSS-OF-COOLANT ACCIDENT ANALYSIS

14.3.2.1   Summary

When the Final Acceptance Criteria (FAC) governing the loss-of-coolant accident (LOCA) for Light Water Reactors was issued in Appendix K of 10 CFR 50.46, (Reference 1), both the Nuclear Regulatory Commission (NRC) and the industry recognized that the stipulations of Appendix K were highly conservative. That is, using the then accepted analysis methods, the performance of the Emergency Core Cooling System (ECCS) would be conservatively underestimated, resulting in predicted Peak Clad Temperatures (PCTs) much higher than expected. At that time, however, the degree of conservatism in the analysis could not be quantified. As a result, the NRC began a large-scale confirmatory research program with the following objectives:

1.   Identify, through separate effects and integral effects experiments, the degree of conservatism in those models permitted in the Appendix K rule. In this fashion, those areas in which a purposely prescriptive approach was used in the Appendix K rule could be quantified with additional data so that a less prescriptive future approach might be allowed.

2.   Develop improved thermal-hydraulic computer codes and models so that more accurate and realistic accident analysis calculations could be performed. The purpose of this research was to develop an accurate predictive capability so that the uncertainties in the ECCS performance and the degree of conservatism with respect to the Appendix K limits could be quantified.

Since that time, the NRC and the nuclear industry have sponsored reactor safety research programs directed at meeting the above two objectives. The overall results have quantified the conservatism in the Appendix K rule for LOCA analyses and confirmed that some relaxation of the rule can be made without a loss in safety to the public. It was also found that some plants were being restricted in operating flexibility by the overly conservative Appendix K requirements. In recognition of the Appendix K conservatism that was being quantified by the research programs, the NRC adopted an interim approach for evaluation methods. This interim approach is described in SECY-83-472, (Reference 2). The SECY-83-472 approach retained those features of Appendix K that were legal requirements, but permitted applicants to use best-estimate thermal-hydraulic models in their ECCS evaluation model. Thus, SECY-83-472 represented an important step in basing licensing decisions on realistic calculations, as opposed to those calculations prescribed by Appendix K.

In 1998, the NRC Staff amended the requirements of 10 CFR 50.46 and Appendix K, "ECCS Evaluation Models," to permit the use of a realistic evaluation model to analyze the performance of the ECCS during a hypothetical LOCA. This decision was based on an improved understanding of LOCA thermal-hydraulic phenomena gained by extensive research programs. Under the amended rules, best-estimate thermal-hydraulic models may be used in place of models with Appendix K features. The rule change also requires, as part of the LOCA analysis, an assessment of the uncertainty of the best-estimate calculations. It further requires that this analysis uncertainty be included when comparing the results of the calculations to the prescribed acceptance criteria of 10 CFR 50.46. Further guidance for the use of best-estimate codes is provided in Regulatory Guide 1.157, (Reference 3).

To demonstrate use of the revised ECCS rule, the NRC and its consultants developed a method called the Code Scaling, Applicability, and Uncertainty (CSAU) evaluation methodology, NUREG/CR-5249, (Reference 4). This method outlined an approach for defining and qualifying a best-estimate thermal-hydraulic code and quantifying the uncertainties in a LOCA analysis.

A LOCA evaluation methodology for three- and four-loop Pressurized Water Reactor (PWR) plants based on the revised 10 CFR 50.46 rules was developed by Westinghouse with the support of EPRI and Consolidated Edison and has been approved by the NRC in WCAP-12945-P-A, (Reference 5). This methodology was later extended to Westinghouse two-loop plants equipped with upper plenum injection (UPI) as documented in WCAP-14449-P-A, (Reference 9).

More recently, Westinghouse developed an alternative uncertainty methodology called ASTRUM, which stands for Automated Statistical TReatment of Uncertainty Method as documented in WCAP-16009-P-A, (Reference 6). This method is still based on the Code Qualification Document (CQD) methodology and follows the steps in the CSAU methodology in NUREG/CR-5249. However, the uncertainty analysis (Element 3 in the CSAU) is replaced by a technique based on order statistics. The ASTRUM methodology replaces the response surface technique with a statistical sampling method where the uncertainty parameters are simultaneously sampled for each case. The ASTRUM methodology has received NRC approval for referencing in licensing calculations in WCAP-16009-P-A.

The three 10 CFR 50.46 criteria (peak clad temperature, maximum local oxidation, and core-wide oxidation) are satisfied by running a sufficient number of WCOBRA/TRAC calculations (sample size). In particular, the statistical theory predicts that 124 calculations are required to simultaneously bound the 95th percentile values of three parameters with a 95-percent confidence level.

This analysis is in accordance with the applicability limits and usage conditions defined in Section 13-3 of WCAP-16009-P-A, (Reference 6) as applicable to the ASTRUM methodology. Section 13-3 of WCAP-16009-P-A was found to acceptably disposition each of the identified conditions and limitations related to WCOBRA/TRAC and the CQD uncertainty approach per Section 4.0 of the ASTRUM Final Safety Evaluation Report appended to the topical report.

14.3.2.2   Method of Analysis

The methods used in the application of WCOBRA/TRAC to the large break LOCA with ASTRUM are described in WCAP-12945-P-A and WCAP-16009-P-A. A detailed assessment of the computer code WCOBRA/TRAC was made through comparisons to experimental data. These assessments were used to develop quantitative estimates of the code's ability to predict key physical phenomena in a PWR large break LOCA. Modeling of a PWR introduces additional uncertainties which are identified and quantified in the plant-specific analysis. WCOBRA/TRAC MOD7A was used for the execution of ASTRUM for Point Beach Units 1 and 2.

WCOBRA/TRAC combines two-fluid, three-field, multi-dimensional fluid equations used in the vessel with one-dimensional drift-flux equations used in the loops to allow a complete and detailed simulation of a PWR. This best-estimate computer code contains the following features:

1. Ability to model transient three-dimensional flows in different geometries inside the vessel

2. Ability to model thermal and mechanical non-equilibrium between phases

3. Ability to mechanistically represent interfacial heat, mass, and momentum transfer in different flow regimes

4. Ability to represent important reactor components such as fuel rods, steam generators, reactor coolant pumps, etc.

A typical calculation using <u>W</u>COBRA/TRAC begins with the establishment of a steady-state, initial condition with all loops intact. The input parameters and initial conditions for this steady-state calculation are discussed in the next section.

Following the establishment of an acceptable steady-state condition, the transient calculation is initiated by introducing a break into one of the loops. The evolution of the transient through blowdown, refill, and reflood proceeds continuously, using the same computer code (<u>W</u>COBRA/TRAC) and the same modeling assumptions. Containment pressure is modeled with the BREAK component using a time dependent pressure table. Containment pressure is calculated using the COCO code as described in WCAP-8327 (Proprietary) and WCAP-8326 (Non-Proprietary), ([Reference 7](#)) and mass and energy releases from the <u>W</u>COBRA/TRAC calculation.

The final step of the best-estimate methodology, in which all uncertainties of the LOCA parameters are accounted for to estimate a PCT, Local Maximum Oxidation (LMO), and Core-Wide Oxidation (CWO) at 95- percent probability, is described in the following sections.

1) Plant Model Development:

In this step, a <u>W</u>COBRA/TRAC model of the plant is developed. A high level of noding detail is used in order to provide an accurate simulation of the transient. However, specific guidelines are followed to ensure that the model is consistent with models used in the code validation. This results in a high level of consistency among plant models, except for specific areas dictated by hardware differences, such as in the upper plenum of the reactor vessel or the ECCS injection configuration.

2. Determination of Plant Operating Conditions:

In this step, the expected or desired operating range of the plant to which the analysis applies is established. The parameters considered are based on a "key LOCA parameters" list that was developed as part of the methodology. A set of these parameters, at mostly nominal values, is chosen for input as initial conditions to the plant model. A transient is run utilizing these parameters and is known as the "initial transient." Next, several confirmatory runs are made, which vary a subset of the key LOCA parameters over their expected operating range in one-at-a-time sensitivities. Because certain parameters are not included in the uncertainty analysis, these parameters are set at their bounding condition. This analysis is commonly referred to as the confirmatory analysis. The most limiting input conditions, based on these confirmatory runs, are then combined into the model that will represent the limiting state for the plant, which is the starting point for the assessment of uncertainties.

3.   Assessment of Uncertainty:

The ASTRUM methodology is based on order statistics. The technical basis of the order statistics is described in Section 11 of WCAP-16009-P-A, (Reference 6). The determination of the PCT uncertainty, LMO uncertainty, and CWO uncertainty relies on a statistical sampling technique. According to the statistical theory, 124 WCOBRA/TRAC calculations are necessary to assess against the three 10 CFR 50.46 criteria (PCT, LMO, CWO).

The uncertainty contributors are sampled randomly from their respective distributions for each of the WCOBRA/TRAC calculations. The list of uncertainty parameters, which are randomly sampled for each time in the cycle, break type (split or double-ended guillotine), and break size for the split break are also sampled as uncertainty contributors within the ASTRUM methodology.

Results from the 124 calculations are tallied by ranking the PCT from highest to lowest. A similar procedure is repeated for LMO and CWO. The highest rank of PCT, LMO, and CWO will bound 95 percent of their respective populations with 95 percent confidence level.

4.   Plant Operating Range:

The plant operating range over which the uncertainty evaluation applies is defined. Depending on the results obtained in the above uncertainty evaluation, this range may be the desired range or may be narrower for some parameters to gain additional margin.

The Large Break LOCA analysis was performed with ZIRLO® cladding. However, Reference 17 concluded that the LOCA ZIRLO models are acceptable for application to Optimized ZIRLO^{TM} cladding in the Large Break analysis, and that no additional calculations are necessary for evaluating the use of Optimized ZIRLO^{TM} cladding provided that plant specific ZIRLO claculations were previously performed.

14.3.2.3   Analysis Assumptions

Two ASTRUM analyses were executed: one for Point Beach Unit 1 and one for Point Beach Unit 2. The expected PCT and its uncertainty developed are valid for a range of plant operating conditions. The range of variation of the operating parameters has been accounted for in the uncertainty evaluation. Table 14.3.2-1 summarizes the operating ranges as defined for the proposed operating conditions which are supported by the Best-Estimate LBLOCA analyses for Point Beach Units 1 and 2. If operation is maintained within these ranges, the LBLOCA results developed in this report using WCOBRA/TRAC are considered to be valid. Note that some of these parameters vary over their range during normal operation (accumulator temperature) and other ranges are fixed for a given operational condition (T$_{avg}$). Table 14.3.2-2, Table 14.3.2-3, and Table 14.3.2-4 summarize the LBLOCA containment data used for calculating containment pressure (for both units). Nominal values are used for containment initial temperature and pressure (Reference 12).

14.3.2.4   Design Basis Accident

The Point Beach Units 1 and 2 PCT-limiting transients are split break transients which analyze conditions that fall within those listed in Table 14.3.2-1. Traditionally, cold leg breaks have been limiting for large break LOCA. This location is the one where flow stagnation in the core appears most likely to occur.

The large break LOCA transient can be divided into convenient time periods in which specific phenomena occur, such as various hot assembly heatup and cooldown transients. For a typical large break, the blowdown period can be divided into the Critical Heat Flux (CHF) phase, the upward core flow phase, and the downward core flow phase. These are followed by the refill, reflood, and long-term cooling periods. Specific important transient phenomena and heat transfer regimes are discussed below, with the transient results shown in Figure 14.3.2-1 to Figure 14.3.2-14 for Unit 1 and Figure 14.3.2-15 to Figure 14.3.2-28 for Unit 2. The PCT-limiting case for each unit was chosen to show a conservative representation of the response to a large break LOCA.

1.   Critical Heat Flux (CHF) Phase:

Immediately following the cold leg rupture, the break discharge rate is subcooled and high (Figure 14.3.2-2, Figure 14.3.2-3 for Unit 1 and Figure 14.3.2-16, Figure 14.3.2-17 for Unit 2). The regions of the RCS with the highest initial temperatures (core, upper plenum, upper head, and hot legs) begin to flash to steam, the core flow reverses and the fuel rods begin to go through departure from nucleate boiling (DNB). The fuel cladding rapidly heats up (Figure 14.3.2-1 for Unit 1 and Figure 14.3.2-15 for Unit 2) while the core power shuts down due to voiding in the core. This phase is terminated when the water in the lower plenum and downcomer begins to flash (Figure 14.3.2-7 and Figure 14.3.2-12 for Unit 1 and Figure 14.3.2-21 and Figure 14.3.2-26 for Unit 2, respectively). The mixture swells and intact loop pumps, still rotating in single-phase liquid, push this two-phase mixture into the core.

2.   Upward Core Flow Phase:

Heat transfer is improved as the two-phase mixture is pushed into the core. This phase may be enhanced if the pumps are not degraded, or if the break discharge rate is low due to saturated fluid conditions at the break. If pump degradation is high or the break flow is large, the cooling effect due to upward flow may not be significant. Figure 14.3.2-4 (Unit 1) and Figure 14.3.2-18 (Unit 2) show the void fraction for one intact loop pump and the broken loop pump. Each figure shows that the intact loop remains in single-phase liquid flow for several seconds, resulting in enhanced upward core flow cooling. This phase ends as the lower plenum mass is depleted, the loop flow becomes two-phase, and the pump head degrades.

3.   Downward Core Flow Phase:

The loop flow is pushed into the vessel by the intact loop pump and decreases as the pump flow becomes two-phase. The break flow begins to dominate and pulls flow down through the core, up the downcomer to the broken loop cold leg, and out the break. While liquid and entrained liquid flow provide core cooling, the top third of core vapor flow (Figure 14.3.2-5 for Unit 1 and Figure 14.3.2-19 for Unit 2) best illustrates this phase of core cooling. Once the system has depressurized to the accumulator pressure (Figure 14.3.2-6 for Unit 1 and Figure 14.3.2-20 for Unit 2), the accumulators begin to inject cold borated water into the intact cold legs (Figure 14.3.2-9 for Unit 1 and Figure 14.3.2-23 for Unit 2). During this period, due to steam upflow in the downcomer, a portion of the injected ECCS water is calculated to be bypassed around the downcomer and out the break. As the system pressure continues to fall, the break flow, and consequently the downward core flow, is reduced. The core begins to heat up as the system pressure approaches the containment pressure and the vessel begins to fill with ECCS water (Figure 14.3.2-8 for Unit 1 and Figure 14.3.2-22 for Unit 2).

4.   Refill Period:

As the refill period begins, the core begins a period of heatup and the vessel begins to fill with ECCS water (Figure 14.3.2-9, Figure 14.3.2-10A and Figure 14.3.2-10B for Unit 1 and Figure 14.3.2-23, Figure 14.3.2-24A and Figure 14.3.2-24B for Unit 2). This period is characterized by a rapid increase in cladding temperatures at all elevations due to the lack of liquid and steam flow in the core region. This period continues until the lower plenum is filled and the bottom of the core begins to reflood and entrainment begins.

5.   Reflood Period:

During the early reflood phase, the accumulators begin to empty and nitrogen enters the system. This forces water into the core, which then boils, causing system re-pressurization, and the lower core region begins to quench (Figure 14.3.2-11 for Unit 1 and Figure 14.3.2-25 for Unit 2). During this time, core cooling may increase due to vapor generation and liquid entrainment. During the reflood period, the core flow is oscillatory as cold water periodically rewets and quenches the hot fuel cladding, which generates steam and causes system re-pressurization. The steam and entrained water must pass through the vessel upper plenum, the hot legs, the steam generators, and the reactor coolant pumps before it is vented out of the break. This flow path resistance is overcome by the downcomer water elevation head, which provides the gravity driven reflood force. From the later stage of blowdown to the beginning of reflood, the accumulators rapidly discharge borated cooling water into the RCS, filling the lower plenum and contributing to the filling of the downcomer. The pumped ECCS water aids in the filling of the downcomer and subsequently supplies water to maintain a full downcomer and complete the reflood period. As the quench front progresses up the core, the PCT location moves higher into the top core region. As the vessel continues to fill, the PCT location is cooled and the early reflood period is terminated.

A second cladding heatup transient may occur due to boiling in the downcomer. The mixing of ECCS water with hot water and steam from the core, in addition to the continued heat transfer from the hot vessel and vessel metal, reduces the subcooling of ECCS water in the lower plenum and downcomer. The saturation temperature is dictated by the containment pressure. If the liquid temperature in the downcomer reaches saturation, subsequent heat transfer from the vessel and other structures will cause boiling and level swell in the downcomer. The downcomer liquid will spill out of the broken cold leg and reduce the driving head, which can reduce the reflood rate, causing a late reflood heatup at the upper core elevations. Figure 14.3.2-12 (Unit 1) and Figure 14.3.2-26 (Unit 2) show only a slight reduction in downcomer level and indicates that a late reflood heatup does not occur.

14.3.2.5   Post Analysis of Record Evaluations

In addition to the analyses presented in this section, evaluations and reanalyses may be performed as needed to address computer code errors and emergent issues, or to support plant changes. The issues or changes are evaluated, and the impact on the Peak Cladding Temperature (PCT) is determined. The resultant increase or decrease in PCT is applied to the analysis of record PCT. The PCT changes due to the evaluation model errors/changes are documented in the 10 CFR 50.46 reports. These PCT changes are not (or may not be) reflected in the PCT documented here. The impact on the analysis of record PCT due to the Thermal Conductivity Degradation (TCD) assessment is presented in Table 14.3.2-5 (Unit 1) and Table 14.3.2-7 (Unit 2) for the large break LOCA. The current PCT is demonstrated to be less than the 10 CFR 50.46(b) requirement of 2200°F.

In addition, 10 CFR 50.46 requires that licensees assess and report the effect of changes to or errors in the evaluation model used in the large break LOCA analysis. These reports constitute addenda to the analysis of record provided in the FSAR until the overall changes become significant as defined by 10 CFR 50.46. If the assessed changes or errors in the evaluation model result in significant changes in calculated PCT, a schedule for formal reanalysis or other action as needed to show compliance will be addressed in the report to the NRC.

Finally, the criteria of 10 CFR 50.46 requires that holders and users of the evaluation models establish a number of definitions and processes for assessing changes in the models or their use. Westinghouse, in consultation with the PWR Owner's Group (PWROG), has developed an approach for compliance with the reporting requirements. This approach is documented in WCAP-13451, (Reference 8). FPL Energy Point Beach provides the NRC with annual and 30-day reports, as applicable, for Point Beach Units 1 and 2. FPL Energy Point Beach intends to provide future reports required by 10 CFR 50.46 consistent with the approach described in WCAP-13451.

14.3.2.6   Conclusions

It must be demonstrated that there is a high level of probability that the limits set forth in 10 CFR 50.46 are met. The demonstration that these limits are met is as follows:

1.   The limiting PCT corresponds to a bounding estimate of the 95th percentile PCT at the 95-percent confidence level. Since the resulting PCT for the limiting case is 1975°F for Unit 1 and 1810°F for Unit 2, the analyses confirm that 10 CFR 50.46 acceptance criterion (b)(1), i.e., "Peak Clad Temperature less than 2200°F", is demonstrated. The results are shown in Table 14.3.2-6 for Unit 1 and Table 14.3.2-8 for Unit 2. Impact on the analysis of record PCT due to 10 CFR 50.46 assessments is addressed in Table 14.3.2-5 (Unit 1) and Table 14.3.2-7 (Unit 2) as discussed in Section 14.3.2.5.

2.   The maximum cladding oxidation corresponds to a bounding estimate of the 95th percentile Local Maximum Oxidation (LMO) at the 95-percent confidence level. Since the resulting LMO for the limiting case is 2.61 percent for Unit 1 and 2.57 percent for Unit 2, the analyses confirm that 10 CFR 50.46 acceptance criterion (b)(2), i.e., "Local Maximum Oxidation of the cladding less than 17 percent," is demonstrated. The results are shown in Table 14.3.2-6 for Unit 1 and Table 14.3.2-8 for Unit 2.

3.   The limiting Core-Wide Oxidation (CWO) corresponds to a bounding estimate of the 95th percentile CWO at the 95-percent confidence level. The limiting Hot Assembly Rod (HAR) total maximum oxidation is 0.386 percent for Unit 1 and 0.154 percent for Unit 2. A detailed CWO calculation takes advantage of the core power census that includes many lower power assemblies. Because there is significant margin to the regulatory limit, the CWO value can be conservatively chosen as that calculated for the limiting HAR. A detailed CWO calculation is therefore not needed because the outcome will always be less than the HAR value. Since the resulting HAR is less than 1.0 percent, the analyses confirm that 10 CFR 50.46 acceptance criterion (b)(3), i.e., "Core-Wide Oxidation less than 1 percent," is demonstrated. The results are shown in Table 14.3.2-6 for Unit 1 and Table 14.3.2-8 for Unit 2.

4.   10 CFR 50.46 acceptance criterion (b)(4) requires that the calculated changes in core geometry are such that the core remains amenable to cooling. This criterion has historically been satisfied by adherence to criteria (b)(1) and (b)(2), and by assuring that fuel deformation due to combined LOCA and seismic loads is specifically addressed. It has been demonstrated that the PCT and maximum cladding oxidation limits remain in effect for Best-Estimate LOCA applications. The approved methodology in WCAP-12945-P-A specifies that effects of LOCA and seismic loads on core geometry do not need to be considered unless grid crushing extends beyond the peripheral assemblies. The actions, automatic or manual, that are currently in place at these plants to maintain long-term cooling remain unchanged with the application of the ASTRUM methodology as documented in WCAP-16009-P-A.

5.   10 CFR 50.46 acceptance criterion (b)(5) requires that long-term core cooling be provided following the successful initial operation of the ECCS. Long-term cooling is dependent on the demonstration of continued delivery of cooling water to the core. The actions, automatic or manual, that are currently in place at these plants to maintain long-term cooling remain unchanged with the application of the ASTRUM methodology as documented in WCAP-16009-P-A, (Reference 6).

Based on the ASTRUM Analyses results (Table 14.3.2-6 and Table 14.3.2-8), it is concluded that Point Beach Units 1 and 2 continue to maintain a margin of safety to the limits prescribed by 10 CFR 50.46.

14.3.2.7   References

1.   10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Cooled Nuclear Power Reactors," and 10 CFR 50 Appendix K, "ECCS Evaluation Models," both dated January 4, 1974.

2.   NRC Staff Report, "Emergency Core Cooling System Analysis Methods," USNRC-SECY-83-472, November 1983.

3.   USNRC Regulatory Guide 1.157, "Best-Estimate Calculations of Emergency Core Cooling System Performances," May 1989.

4.   NUREG/CR-5249, "Qualifying Reactor Safety Margins: Application of Code Scaling Applicability and Uncertainty (CSAU) Evaluation Methodology to a Large Break Loss-of-Coolant-Accident," 1989.

5.   WCAP 12945-P-A (Proprietary), Volume I, Revision 2, and Volumes II-V, Revision 1, "Westinghouse Code Qualification Document for Best Estimate Loss of Coolant Accident Analysis," 1998.

6.   WCAP-16009-P-A, "Realistic Large-Break LOCA Evaluation Methodology Using the Automated Statistical Treatment of Uncertainty Method (ASTRUM)," (Proprietary), January 2005.

7.   WCAP-8327 (Proprietary) and WCAP-8326 (Non-Proprietary), "Containment Pressure Analysis Code (COCO)," July 1974.

8.   WCAP-13451, "Westinghouse Methodology for Implementation of 10 CFR 50.46 Reporting," October 1992.

9.   WCAP-14449-P-A, Revision 1, "Application of Best-Estimate Large-Break LOCA Methodology to Westinghouse PWRs with Upper Plenum Injection," 1999.

10.   EC 12604, "Install Debris Interceptors - Unit 1," dated January 13, 2009

11.   EC 14534, "GSI 191 Unit 1 Sump B Modifications."

12.   Letter WEP-10-105, Westinghouse to NextEra Energy Resources, "Report Request: Technical Evaluation of Containment Internal Temperature Assumptions for LBLOCA Analysis," dated October 19, 2010.

13.   NRC Safety Evaluation for License Amendments 235 and 239, "ASTRUM Implementation for Large Break LOCA Analysis," dated October 29, 2009.

14.   Westinghouse CALC Note CN-LIS-08-91, "Point Beach Extended Power Uprate BELOCA ASTRUM Analysis: Units 1 and 2 (WEP/WIS) Uncertainty Analysis," dated September 29, 2008.

15.   Letter NRC 2009-0027, FPL Energy to NRC, "Response to Request for Additional Information License Amendment Request 258, Incorporate Best Estimate Large Break Loss of Coolant Accident (LOCA) Analysis Using ASTRUM," dated March 4, 2009.

16.   Letter NRC 2012-0038, NextEra Energy to NRC, "ECCS 30-Day Report for the Thermal Conductivity Degradation Impact on Point Beach Nuclear Plants Units 1 and 2 Large Break Loss of Coolant Accident Analysis with ASTRUM," dated May 30, 2012.

17.   WCAP-12610-P-A and CENPD-404-P-A Addendum 1-A, "Optimized ZIRLO^{TM}, July 2006.

Table 14.3.2-1   PLANT OPERATING RANGE ANALYZED BY THE BEST-ESTIMATE
LARGE BREAK LOCA ANALYSIS (Sheet 1 of 2)

	Parameter	As-Analyzed Value or Range
1.0	Plant Physical Description	
	Dimensions	Nominal
	Pressurizer location	Assumed on Broken Loop
	Hot assembly location	Anywhere in core interior [1]
	Hot assembly type	14x14 422V+ fuel with ZIRLO [®] [6] cladding, IFM
	Steam generator tube plugging level	≤10%
	Fuel assembly type	14x14 422V+ fuel with ZIRLO [®] [6] cladding, IFM
	Steam generator type	U1: Model 44F U2: Model Delta-47
2.0	Plant Initial Operating Conditions	
	2.1 Reactor Power	
	Core power [5]	1811 MWt (100.6% of 1800 MWt)
	Peak heat flux hot channel factor (F_Q)	≤ 2.6
	Peak hot rod enthalpy rise hot channel factor ($F_{\Delta H}$)	≤ 1.68
	Hot assembly radial peaking factor ($\bar{P}_{HA}$)	≤ 1.68/1.04
	Hot assembly heat flux hot channel factor (F_{QHA})	≤ 2.6/1.04
	Axial power distribution (P_{BOT}, P_{MID})	U1: Figure 14.3.2-13 U2: Figure 14.3.2-27
	Low power region relative power (P_{LOW})	$0.2 \leq P_{LOW} \leq 0.6$
	Hot assembly burnup	≤ 75,000 MWD/MTU, lead rod [1][3]
	MTC	≤ 0 at hot full power (HFP)
	Typical cycle average burnup	20,000 MWD/MTU
	Minimum core average burnup	≥ 10,000 MWD/MTU
	Maximum steady state depletion, F_Q	2.1
	2.2 Fluid Conditions	
	T_{AVG}	$558.0 - 6.4°F \leq T_{AVG} \leq 577.0 + 6.4°F$
	Pressurizer pressure	$2250 - 50\ psia \leq P_{RCS} \leq 2250 + 50\ psia$
	Loop flow	TDF ≥ 89,000 gpm/loop
	Upper head temperature	Function of T_{AVG}, Between T_{AVG} and T_{HOT}
	Pressurizer level	31% of span at Low T_{AVG}
		50% of span at Hi T_{AVG}
	Accumulator temperature	60°F ≤ TACC ≤ 120°F
	Accumulator pressure	689.7 psia ≤ PACC ≤ 839.7 psia
	Accumulator liquid volume	$1068\ ft^3 \leq VACC \leq 1168\ ft^3$
	Accumulator fL/D [2]	7.056 +/- 20%
	Minimum accumulator boron	2600 ppm
3.0	Accident Boundary Conditions	
	Minimum safety injection flow	Table 14.3.2-9
	Safety injection temperature	32°F ≤ SI Temp ≤ 120°F

Table 14.3.2-1   PLANT OPERATING RANGE ANALYZED BY THE BEST-ESTIMATE
LARGE BREAK LOCA ANALYSIS (Sheet 2 of 2)

	Safety injection delay [4]	High Head: 13 seconds (with no-LOOP) 28 seconds (with LOOP) Low Head: 23.7 seconds (with no-LOOP) 37 seconds (with LOOP)
	Containment modeling	See Table 14.3.2-2, Table 14.3.2-3, and Table 14.3.2-4
	Minimum containment pressure	See Table 14.3.2-2
	Containment spray initiation delay	See Table 14.3.2-2
	Recirculation spray initiation delay	See Table 14.3.2-2
	Single failure	Loss of one ECCS train

Notes:

1. 24 peripheral locations will not physically be lead power assembly.

2. Based on average L/D of 504.0

3. Please note that the fuel temperature and rod internal pressure data is only provided up to 62,000 MWD/ MTU. In addition, the hot assembly/hot rod will not have a burnup this high in the ASTRUM analyses.

4. The BELOCA analysis originally modeled a High Head Safety Injection (HHSI) delay of 8 seconds with no-LOOP and 23 seconds with LOOP. However, the additional 5 second delay for HHSI was evaluated as negligible and thus the values of 13 and 28 seconds are reflected herein (See Page 82 of Reference 14).

5. It has been shown that LOCA analysis input values at 1811 MWt conservatively bound operation at lower power levels (Reference 13 and Reference 15).

6. Optimized ZIRLO^{TM} fuel cladding has been evaluated as an acceptable fuel cladding.

Table 14.3.2-2   LARGE BREAK LOCA CONTAINMENT DATA USED FOR
CALCULATION OF CONTAINMENT PRESSURE

Containment Net Free Volume	1,118,250 ft^3
Initial Conditions Initial containment pressure at full power operation Initial containment temperature at full power Minimum RWST temperature Minimum temperature outside containment Initial spray temperature	 14.7 psia 90.0°F 32.0°F -25.0°F 32.0°F
Spray System Number of containment spray pumps operating Minimum post-accident spray system initiation delay Maximum spray system flow from all containment spray pumps	 2 10 sec 3900 gal/min
Fan Coolers Maximum number of containment fan coolers in operation Minimum post-accident containment fan cooler initiation delay Fan Cooler Performance	 4 0 sec See Table 14.3.2-3
Recirculation Spray	Not Modeled

Table 14.3.2-3   CONTAINMENT FAN COOLER HEAT REMOVAL RATE FOR ECCS
CONTAINMENT BACKPRESSURE ANALYSIS

Containment Temperature (°F)	Heat Removal Rate for One Fan Cooler (Btu/sec)
120	3718
160	8893
190	14,953
210	19,425
220	21,558
240	25,539
260	29,047
270	30,725

Table 14.3.2-4   STRUCTURAL HEAT SINK DATA FOR ECCS CONTAINMENT
BACKPRESSURE ANALYSIS (Sheet 1 of 4)

Heat Sink	Description[2]	Area (ft²)	Material[1]	Thickness (inches)
1	Upper Dome	1,883.7	Paint Type 1	0.01404
			Carbon Steel	0.2496
			Gap	0.021
			Concrete	36
2	Middle Dome	6,917.0	Paint Type 1	0.01404
			Carbon Steel	0.2496
			Gap	0.021
			Concrete	36
3	Lower Dome	7,525.4	Paint Type 1	0.01404
			Carbon Steel	0.2496
			Gap	0.021
			Concrete	36
4	Upper Containment outer wall (above 66')	19,876.0	Paint Type 1	0.015
			Carbon Steel	0.2496
			Gap	0.021
			Concrete	42
5	Middle Containment outer wall (21' to 66')	17,367.5	Paint Type 1	0.015
			Carbon Steel	0.2496
			Gap	0.021
			Concrete	42
6	Lower Containment outer wall (8' to 21')	4,874.2	Paint Type 1	0.015
			Carbon Steel	0.2496
			Gap	0.021
			Concrete	42
7	Reactor Cavity: Shield wall/Reactor Pit	1,983.2	Paint Type 2	0.039
			Concrete	12
8	Reactor Cavity: tunnel walls	304.2	Paint Type 2	0.039
			Concrete	12
9	Reactor Cavity: Keyway tower/shaft	1,310.4	Paint Type 2	0.039
			Concrete	12
10	Reactor Cavity: Floor slab	413.0	Paint Type 2	0.015
			Concrete	12

Table 14.3.2-4   STRUCTURAL HEAT SINK DATA FOR ECCS CONTAINMENT
BACKPRESSURE ANALYSIS (Sheet 2 of 4)

Heat Sink	Description[2]	Area (ft²)	Material[1]	Thickness (inches)
11	Pressurizer walls (inside 46' to 86')	2,371.6	Paint Type 2	0.039
			Concrete	15
12	Pressurizer floor slab	182.5	Paint Type 2	0.015
			Concrete	24
			Paint Type 2	0.039
13	Pressurizer missile shields	205.9	Paint Type 2	0.039
			Carbon Steel	0.5
			Gap	0.021
			Concrete	15
			Paint Type 1	0.039
14	Upper Containment interior walls	6,341.4	Paint Type 2	0.039
			Concrete	15
15	Upper Containment Floor/ Annular Compartment ceiling	5,076.6	Paint Type 2	0.015
			Concrete	4
16	Annular Compartment: Interior wall (46' to 66')	6,285.2	Paint Type 2	0.039
			Concrete	15
17	Annular Compartment: Interior wall (21' to 46')	9,667.7	Paint Type 2	0.039
			Concrete	15
18	Annular Compartment: laydown area high wall (21' to 66')	684.5	Paint Type 2	0.039
			Concrete	18
19	Annular Compartment 46' floor slab	4,579.4	Paint Type 2	0.015
			Concrete	4
20	Annular Compartment floor/Annular Sump ceiling (21')	4,998.2	Paint Type 2	0.015
			Concrete	4
21	Annular Sump: interior walls (8' to 21')	5,249.8	Paint Type 2	0.039
			Concrete	15
22	Annular Sump floor slab (8')	5,091.8	Paint Type 2	0.015
			Concrete	12
23	Loop A: walls	7,828.5	Paint Type 2	0.039
			Concrete	15
24	Loop A: floor slab	954.7	Paint Type 2	0.015
			Concrete	12

Table 14.3.2-4   STRUCTURAL HEAT SINK DATA FOR ECCS CONTAINMENT
BACKPRESSURE ANALYSIS (Sheet 3 of 4)

Heat Sink	Description[2]	Area (ft²)	Material[1]	Thickness (inches)
25	Loop A: missile shields	293.8	Paint Type 2	0.015
			Concrete	15
			Paint Type 2	0.039
26	Loop B: walls	9,461.8	Paint Type 2	0.039
			Concrete	15
27	Loop B: floor slab	929.0	Paint Type 2	0.015
			Concrete	12
28	Loop B: missile shields	243.4	Paint Type 2	0.015
			Concrete	15
			Paint Type 2	0.039
29	Loop B: sub-pressurizer compartment walls	334.6	Paint Type 2	0.039
			Concrete	15
30	Loop B: sub-pressurizer compartment floor	205.9	Paint Type 2	0.015
			Concrete	24
			Paint Type 2	0.039
31	Refueling cavity wall	5,488.5	Stainless Steel	0.1875
			Gap	0.021
			Concrete	18
			Paint Type 2	0.039
32	Refueling cavity floor/ Annular sump ceiling	627.1	Stainless Steel	0.1875
			Gap	0.021
			Concrete	36
			Paint Type 2	0.039
33	Misc. steel in reactor cavity compartment	780.8	Paint Type 1	0.013
			Carbon Steel	1.263
34	Misc. steel in the pressurizer compartment	1.3	Paint Type 1	0.013
			Carbon Steel	0.005
35	Misc. steel in the upper containment	5,906.5	Paint Type 1	0.013
			Carbon Steel	0.377
36	Misc. steel in the annular compartment	26,333.6	Paint Type 1	0.013
			Carbon Steel	0.396
37	Misc. steel in the annular sump compartment	7,795.5	Paint Type 1	0.013
			Carbon Steel	0.23

Table 14.3.2-4   STRUCTURAL HEAT SINK DATA FOR ECCS CONTAINMENT
BACKPRESSURE ANALYSIS (Sheet 4 of 4)

Heat Sink	Description[2]	Area (ft^2)	Material[1]	Thickness (inches)
38	Misc. steel in the Loop A compartment	3,967.0	Paint Type 1	0.013
			Carbon Steel	0.372
39	Misc. steel in the Loop B compartment	3,967.0	Paint Type 1	0.013
			Carbon Steel	0.372
40	Misc. steel in the dome compartment	24,255.6	Paint Type 1	0.013
			Carbon Steel	0.148
41	Misc. steel in refueling cavity compartment	466.0	Paint Type 1	0.013
			Carbon Steel	1.475
42	1 CFC in upper containment compartment; unpainted copper	8,274.1	Copper	0.013
43	1 CFC in upper containment compartment	25.2	Stainless Steel	1.022
44	1 CFC in annular compartment	8,278.3	Copper	0.013
45	Unpainted stainless steel in Annular Compartment; 1 CFC	28.2	Stainless Steel	0.67
46	Polar crane and Rail grider in the upper containment	9,470.5	Paint Type 1	0.013
			Carbon Steel	0.906
47	A Reactor Coolant Pump in the Loop A compartment	667.5	Paint Type 1	0.0079
			Copper	2.583
48	B Reactor Coolant Pump in the Loop B compartment	667.5	Paint Type 1	0.0079
			Copper	2.583
49	Pressurizer Relief Tank Unpainted Stainless Steel	595.5	Stainless Steel	0.67

(1) Paint Type 1 is Amercote 66 top coating with a Dimecote 6 primer coating; Paint Type 2 is Phenoline 305 top coating with a Carboline 195 primer coating.

(2) Debris interceptors were installed in Unit 1 containment, and some were subsequently removed. This added a small amount of steel to the Unit 1 containment heat sink inventory not included in this table. See References 10 and 11.

Table 14.3.2-5   PEAK CLAD TEMPERATURE INCLUDING ALL PENALTIES AND
BENEFITS, BEST-ESTIMATE LARGE BREAK LOCA (BE LBLOCA)
UNIT 1

PCT for Analysis-of-Record (AOR)	1975°F
Impact due to Thermal Conductivity Degradation[1]	+151°F
BE LBLOCA PCT for Comparison to 10 CFR 50.46 Requirements	2126°F [2]

(1) Per Reference 16.

(2) The PCT changes due to the evaluation model errors/changes are documented in the
10 CFR 50.46 reports. These PCT changes are not (or may not be) reflected in the PCT
documented here.

Table 14.3.2-6   UNIT 1 BEST-ESTIMATE LARGE BREAK LOCA RESULTS

10 CFR 50.46 Requirement	Value	Criteria
95/95 PCT[1] (°F)	1975	< 2,200
95/95 LMO[2] (%)	2.61	< 17
95/95 CWO[3] (%)	0.386	< 1

1. Peak Cladding Temperature

2. Local Maximum Oxidation

3. Core-Wide Oxidation

Table 14.3.2-7   PEAK CLAD TEMPERATURE INCLUDING ALL PENALTIES AND
BENEFITS, BEST-ESTIMATE LARGE BREAK LOCA (BE LBLOCA)
UNIT 2

PCT for Analysis-of-Record (AOR)	1810°F
Impact due to Thermal Conductivity Degradation[1]	+285°F
BE LBLOCA PCT for Comparison to 10 CFR 50.46 Requirements	2095°F [2]

(1) Per Reference 16.

(2) The PCT changes due to the evaluation model errors/changes are documented in the
10 CFR 50.46 reports. These PCT changes are not (or may not be) reflected in the PCT
documented here.

Table 14.3.2-8   UNIT 2 BEST-ESTIMATE LARGE BREAK LOCA RESULTS

10 CFR 50.46 Requirement	Value	Criteria
95/95 PCT[1] (°F)	1810	< 2,200
95/95 LMO[2] (%)	2.57	< 17
95/95 CWO[3] (%)	0.154	< 1

1. Peak Clad Temperature

2. Local Maximum Oxidation

3. Core-Wide Oxidation

Table 14.3.2-9   INJECTED SAFETY INJECTION FLOW USED IN BEST-ESTIMATE
LARGE-BREAK LOCA ANALYSIS FOR UNITS 1 AND 2

RCS Pressure (psia)	High Head Injected Flow (gpm)	Low Head Injected Flow (gpm)
14.7	439.5	1,693.5
24.7	439.5	1,618.9
34.7	439.5	1,542.1
44.7	439.5	1,460.4
54.7	439.5	1,374.1
64.7	439.5	1,283.7
74.7	439.5	1,187.6
84.7	439.5	1,084.8
94.7	439.5	959.8
104.7	439.5	819.8
114.7	415.5	654.6
214.7	390.5	0
314.7	364.2	0
414.7	336.6	0
514.7	306.0	0
614.7	273.1	0
714.7	237.4	0

Figure 14.3.2-1 UNIT 1 LIMITING PEAK CLAD TEMPERATURE CASE PCT AND PEAK
CLAD TEMPERATURE LOCATION

PBNP FSAR FIG 14.3.2-1
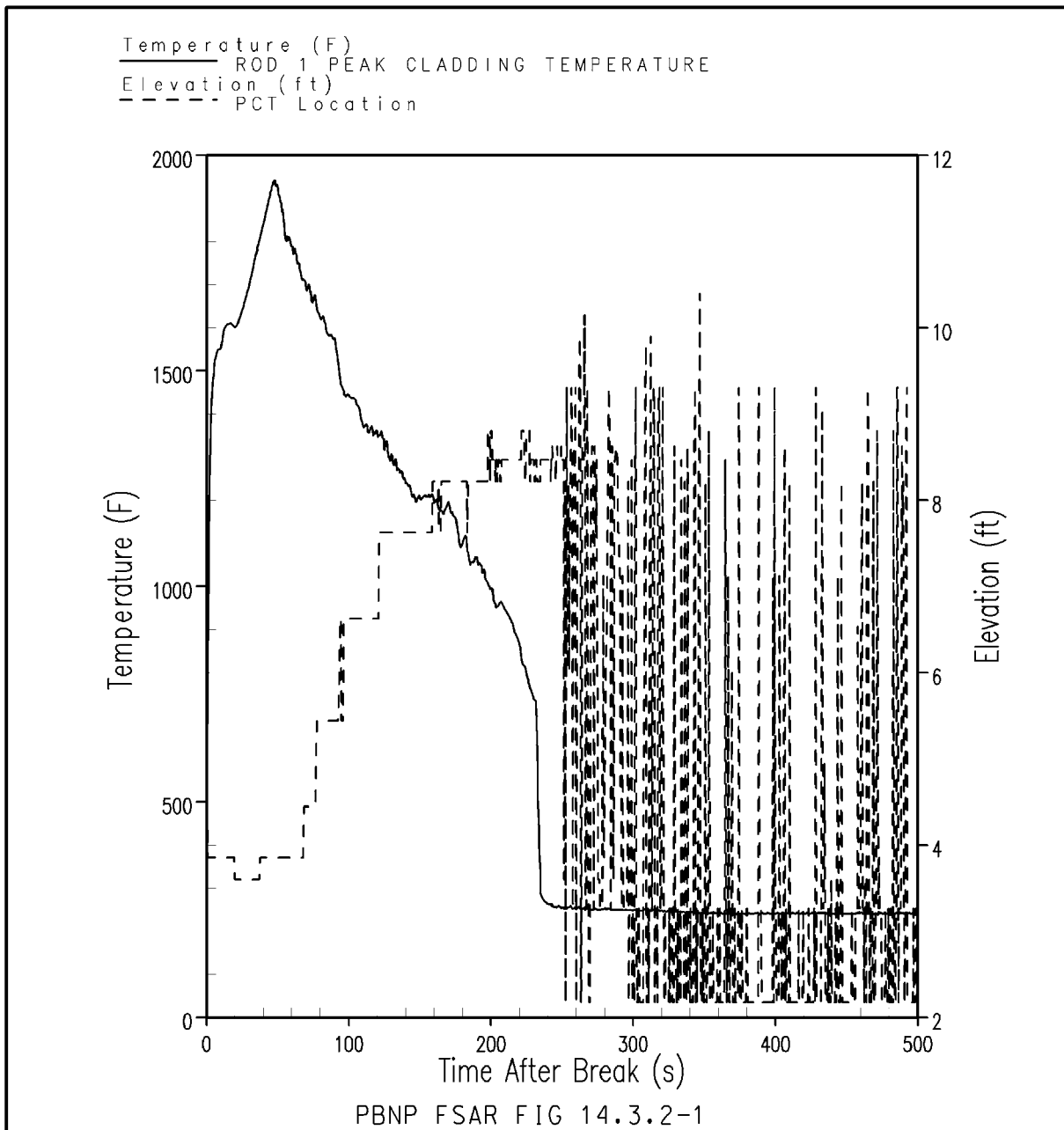
00	07-09-10	INCORP. INTO PDS PER EC #15920	KJB	JLF	KJB
REV NO.	DATE	REVISION DESCRIPTION	DRAWN	CHK'D	APPRV'D

UPDATE PRI: 2

Figure 14.3.2-2 UNIT 1 LIMITING PEAK CLAD TEMPERATURE CASE VESSEL SIDE
BREAK FLOW
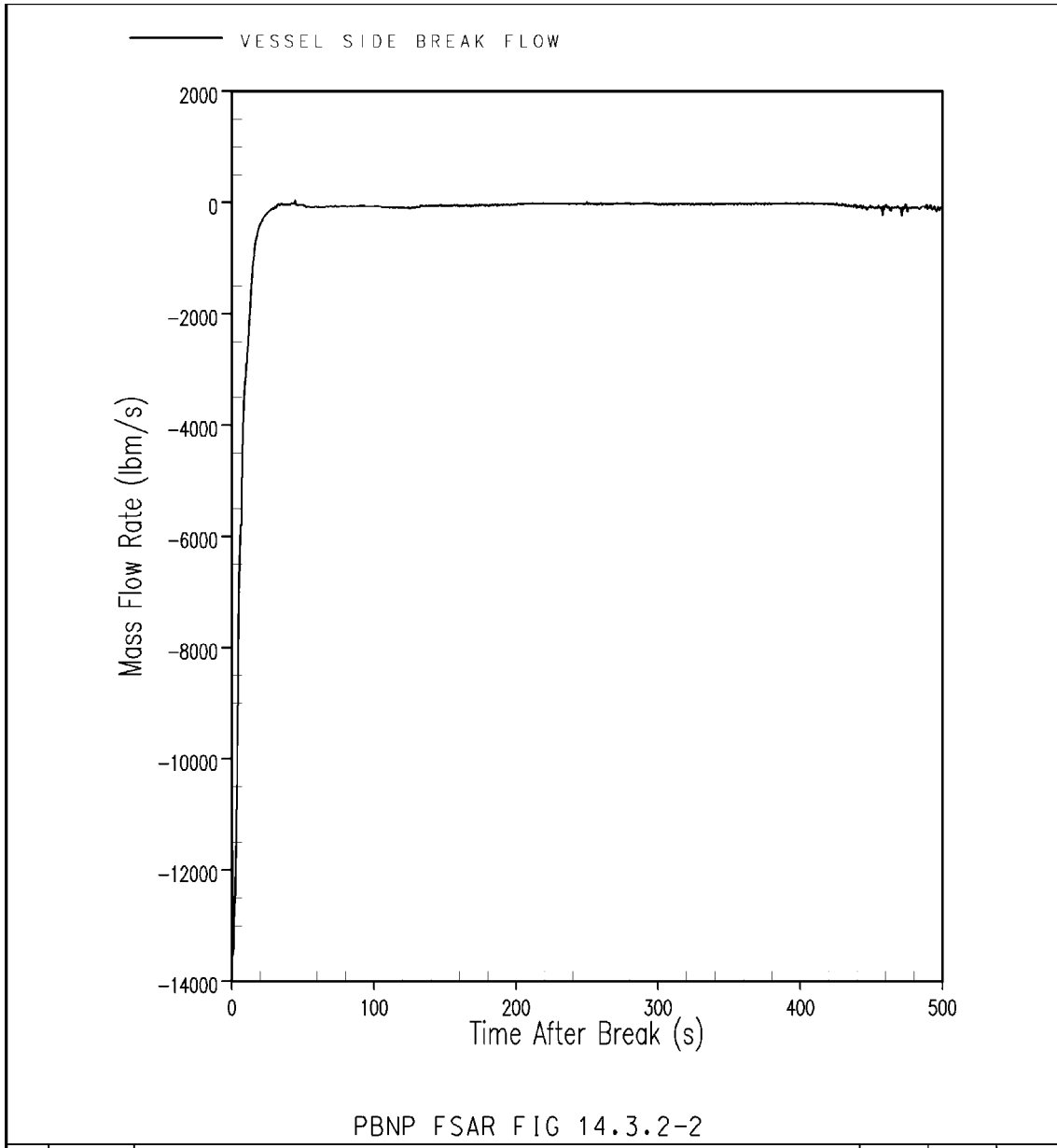
Figure 14.3.2-3 UNIT 1 LIMITING PEAK CLAD TEMPERATURE CASE PUMP SIDE
BREAK FLOW
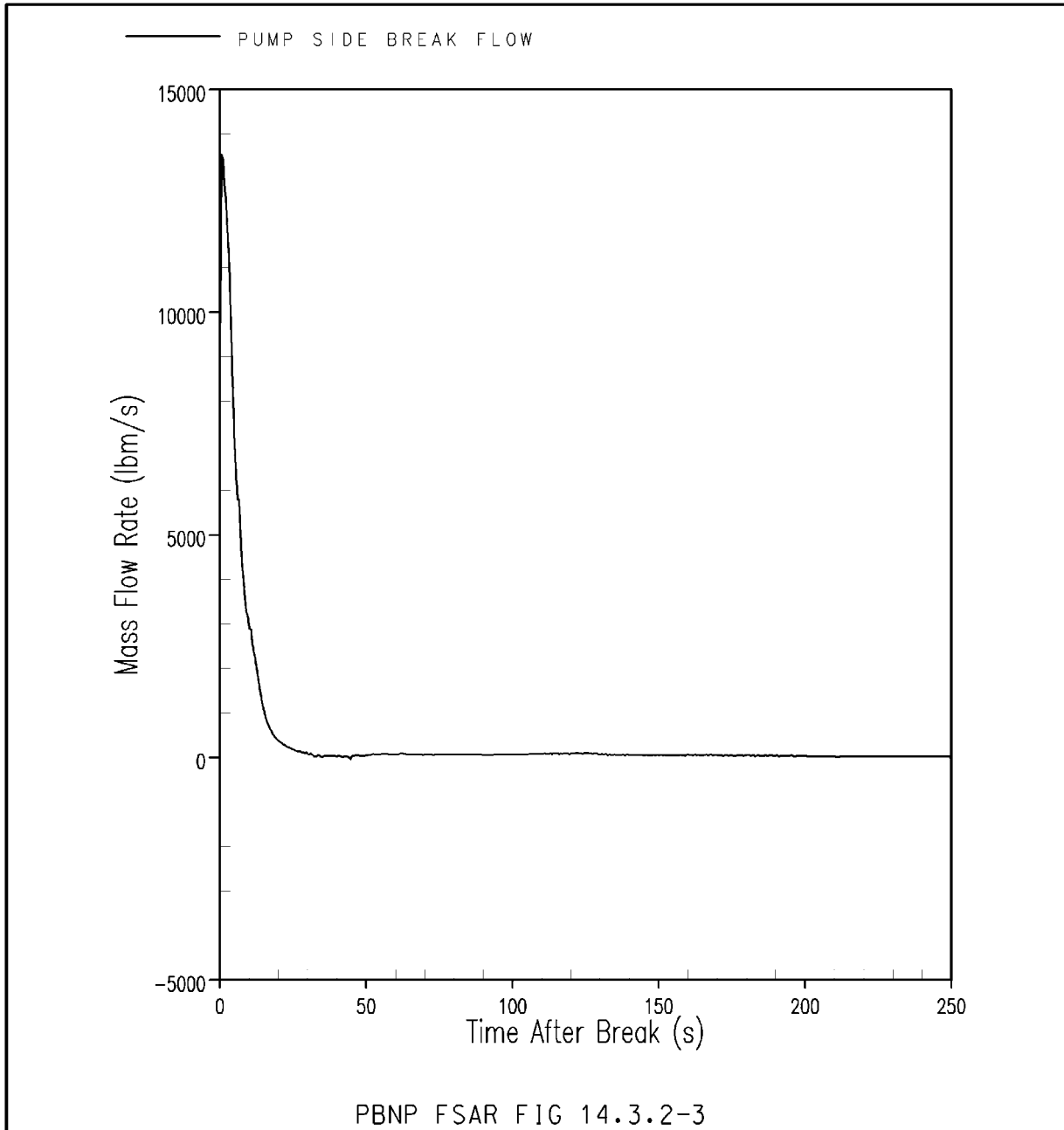
PBNP FSAR FIG 14.3.2-3

00	08-09-10	INCORP. INTO PDS PER EC # 15920		KJB	JLF	KJB
REV NO.	DATE	REVISION DESCRIPTION		DRAWN	CHK'D	APPRV'D

Figure 14.3.2-4   UNIT 1 LIMITING PEAK CLAD TEMPERATURE CASE BROKEN AND INTACT LOOP PUMP VOID FRACTION
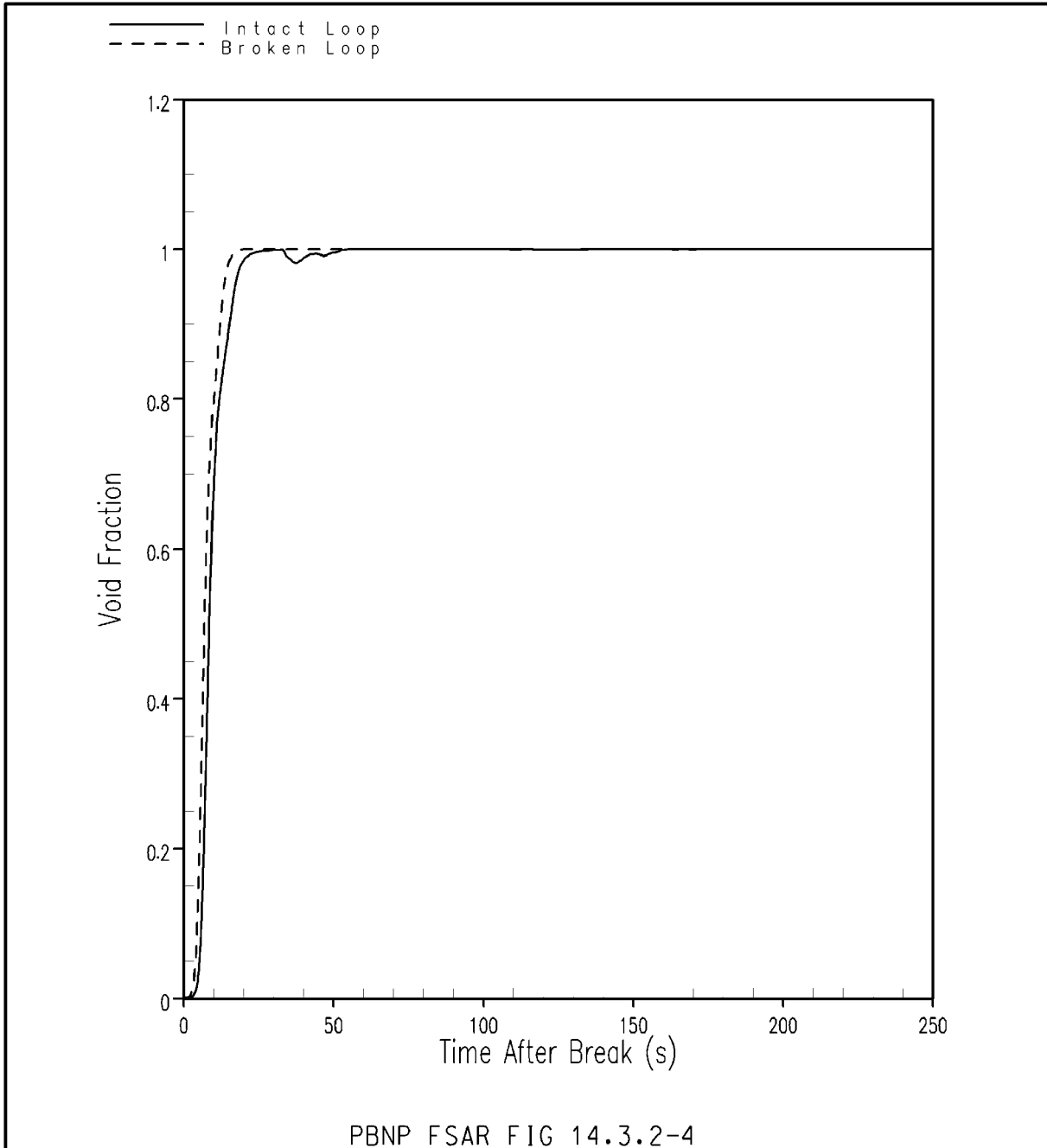
PBNP FSAR FIG 14.3.2-4

00	08-09-10	INCORP. INTO PDS PER EC #15920		KJB	JLF	KJB
REV NO.	DATE	REVISION DESCRIPTION		DRAWN	CHK'D	APPRV'D

UPDATE PPL. 2

Figure 14.3.2-5   UNIT 1 LIMITING PEAK CLAD TEMPERATURE CASE HOT ASSEMBLY
TOP THIRD OF CORE VAPOR FLOW
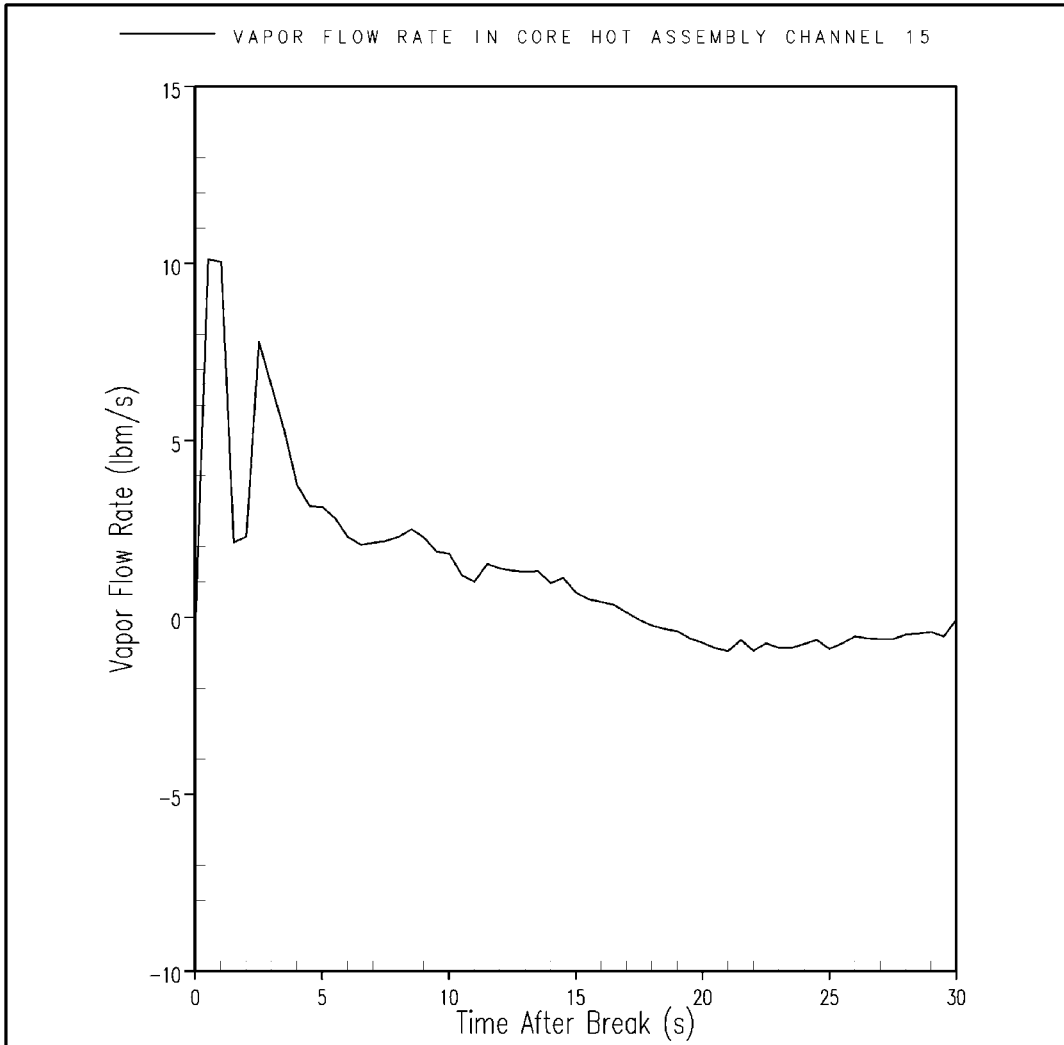
PBNP FSAR FIG 14.3.2-5

00	08-09-10	INCORP. INTO PDS PER EC #15920			KJB	JLF	KJB
REV NO.	DATE	REVISION DESCRIPTION			DRAWN	CHK'D	APPRV'D

UPDATE PRI: 2
SUB TYPE: FSAR
CBD: NO

DRAWN		DATE		NEXTera™	LIMITING PCT CASE HOT ASSEMBLY
KJB		09-18-09		ENERGY	TOP THIRD OF CORE VAPOR FLOW
CHECKED		DATE	SCALE	POINT BEACH	
WJH		06-30-10	NONE	MICROFILM	POINT BEACH N.P. UNIT 1
APPROVED		DATE	FILE NO. PB11440C	NO. PB11440	REV. NO. 00 DWG SIZE A

8/25/2010 2:02:44 PM

Figure 14.3.2-6 UNIT 1 LIMITING PEAK CLAD TEMPERATURE CASE PRESSURIZER
                 PRESSURE
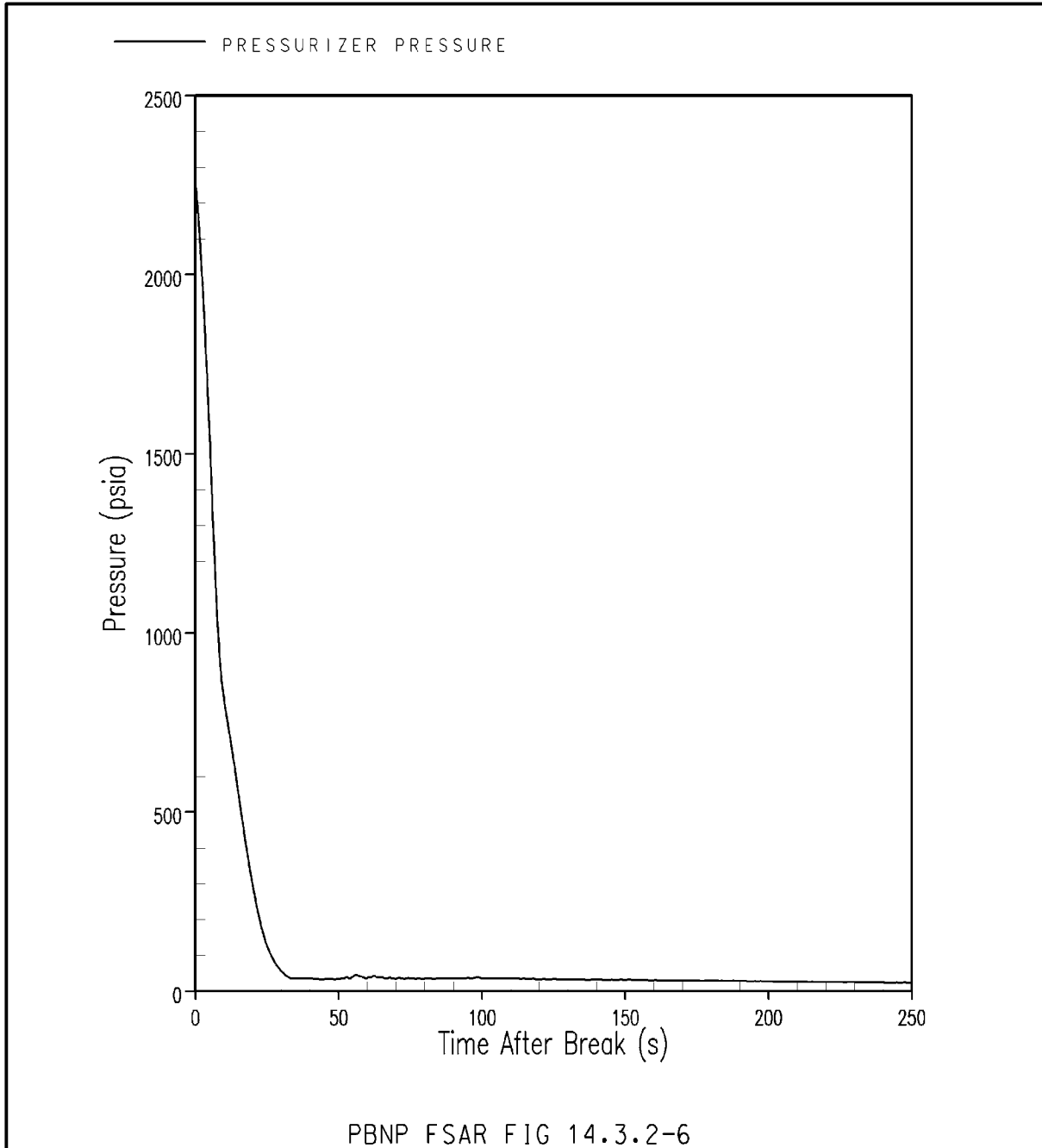
PBNP FSAR FIG 14.3.2-6

00	08-09-10	INCORP. INTO PDS PER EC #15920	KJB	JLF	KJB
REV NO.	DATE	REVISION DESCRIPTION	DRAWN	CHK'D	APPRV'D

UPDATE PBI: 2

Figure 14.3.2-7 UNIT 1 LIMITING PEAK CLAD TEMPERATURE CASE LOWER
                 PLENUM COLLAPSED LIQUID LEVEL

PBNP FSAR FIG 14.3.2-7

00	08-09-10	INCORP. INTO PDS PER EC #15920	KJB	JLF	KJB
REV NO.	DATE	REVISION DESCRIPTION	DRAWN	CHK'D	APPRV'D
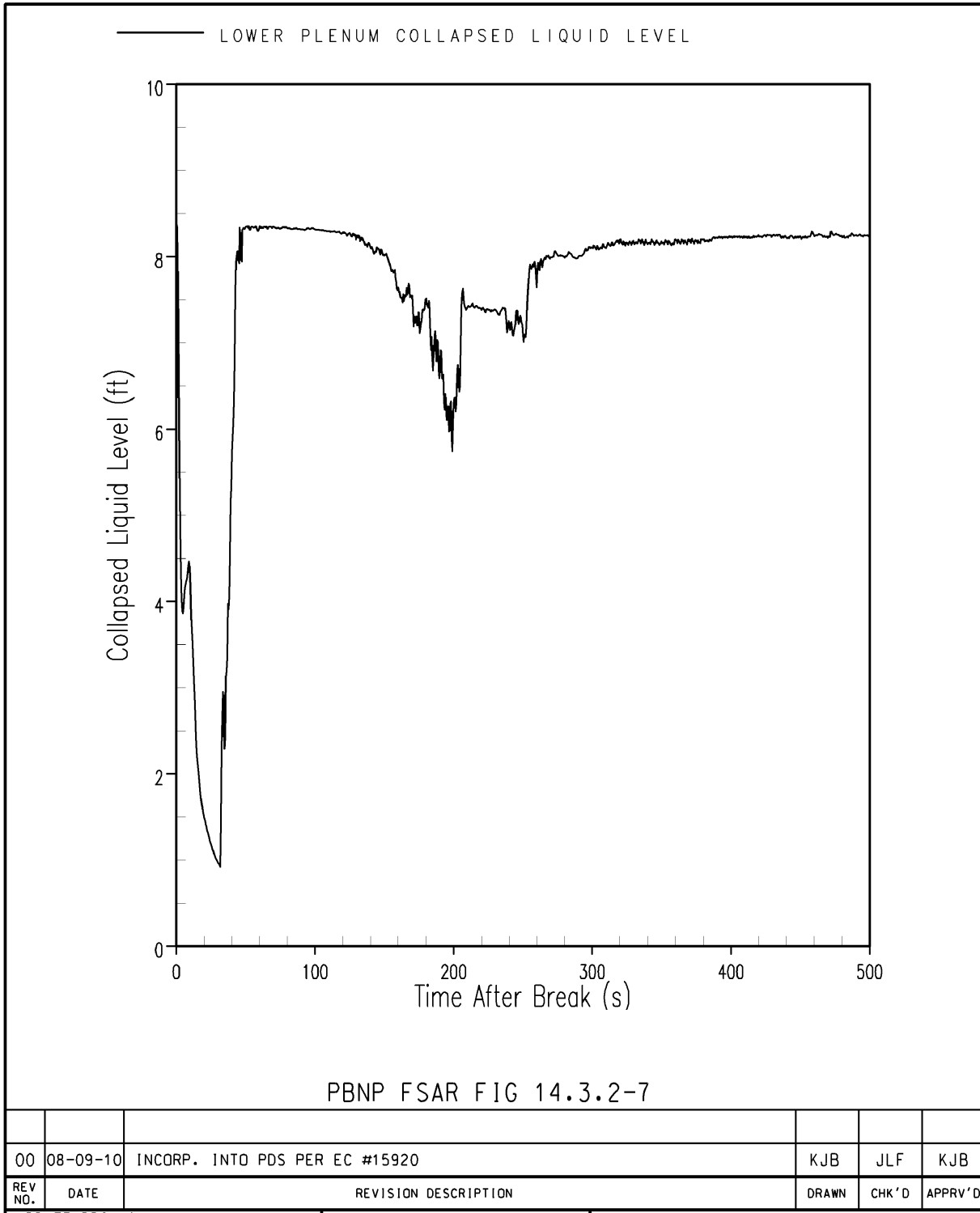
Figure 14.3.2-8   UNIT 1 LIMITING PEAK CLAD TEMPERATURE CASE VESSEL LIQUID
MASS
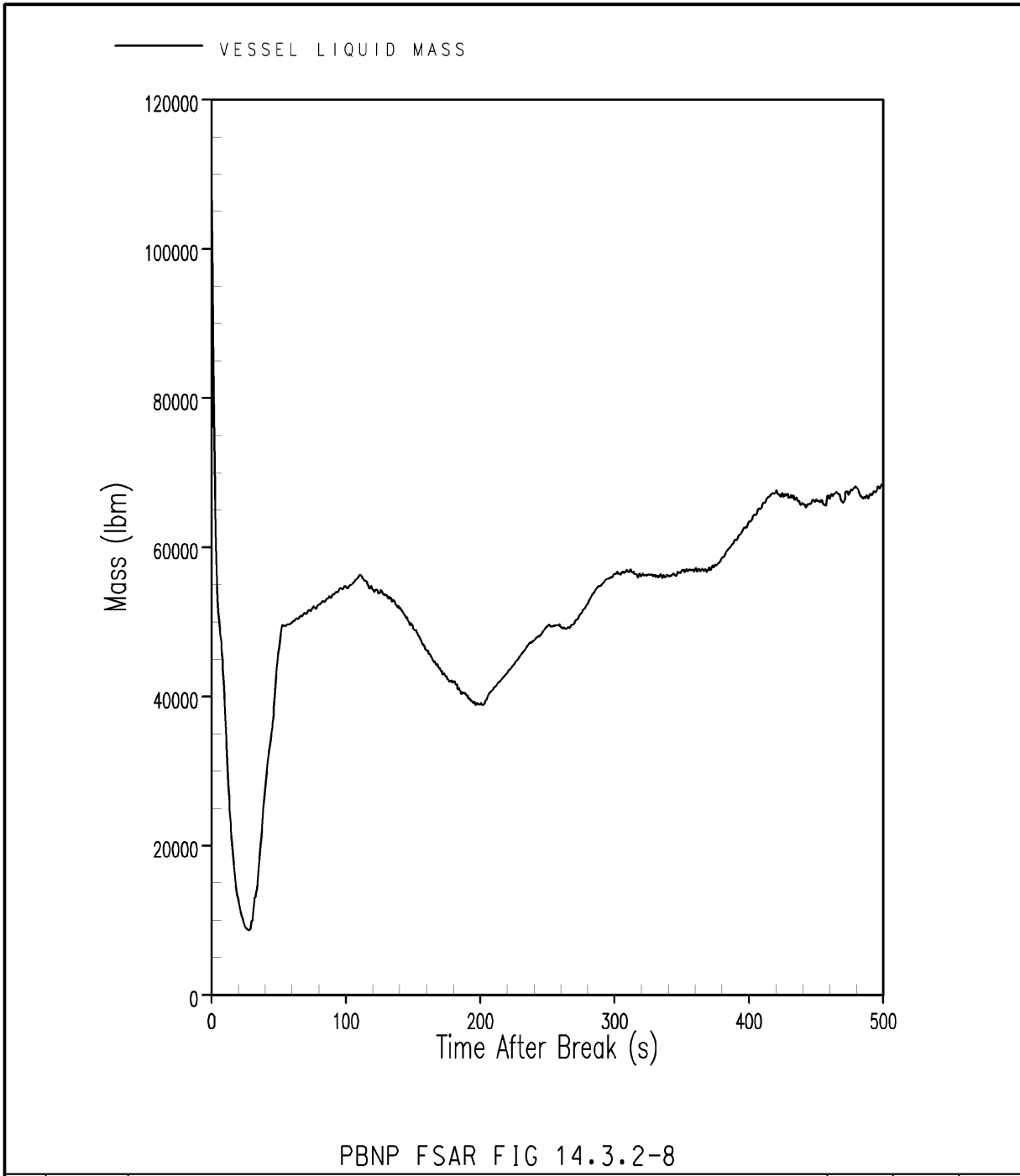
PBNP FSAR FIG 14.3.2-8

00	08-09-10	INCORP. INTO PDS PER EC #15920		KJB	JLF	KJB
REV NO.	DATE	REVISION DESCRIPTION		DRAWN	CHK'D	APPRV'D
UPDATE RRI: 2						

Figure 14.3.2-9 UNIT 1 LIMITING PEAK CLAD TEMPERATURE CASE LOOP 2
ACCUMULATOR FLOW

PBNP FSAR FIG 14.3.2-9

| 00 | 08-09-10 | INCORP. INTO PDS PER EC #15920 | KJB | JLF | KJB |
| REV NO. | DATE | REVISION DESCRIPTION | DRAWN | CHK'D | APPRV'D |

Figure 14.3.2-10A UNIT 1 LIMITING PEAK CLAD TEMPERATURE CASE LOOP 2 HIGH HEAD SAFETY INJECTION FLOW
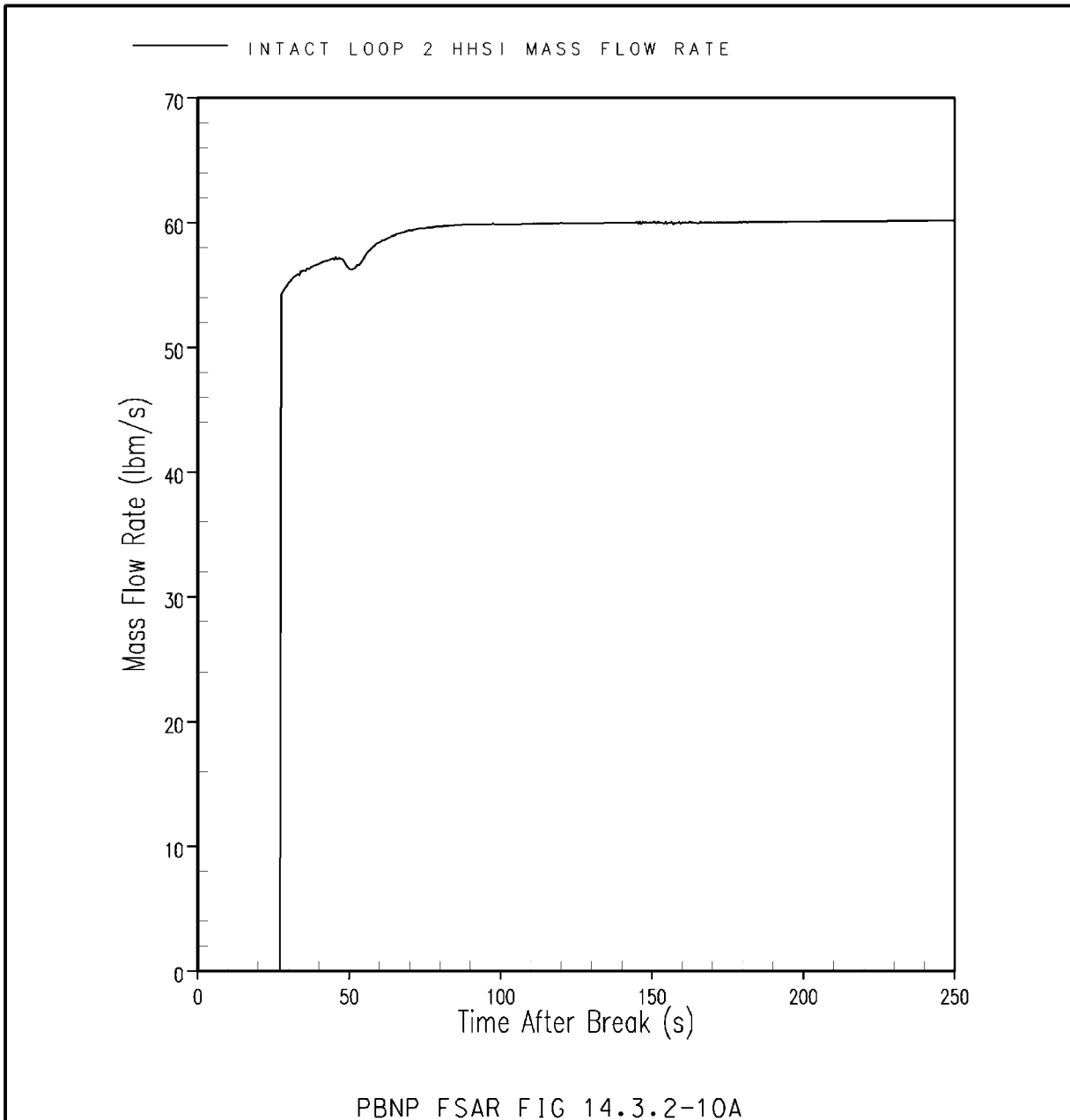
PBNP FSAR FIG 14.3.2-10A

00	08-09-10	INCORP. INTO PDS PER EC #15920	KJB	JLF	KJB
REV NO.	DATE	REVISION DESCRIPTION	DRAWN	CHK'D	APPRV'D

Figure 14.3.2-10B UNIT 1 LIMITING PEAK CLAD TEMPERATURE CASE LOOP 2 LOW HEAD SAFETY INJECTION FLOW
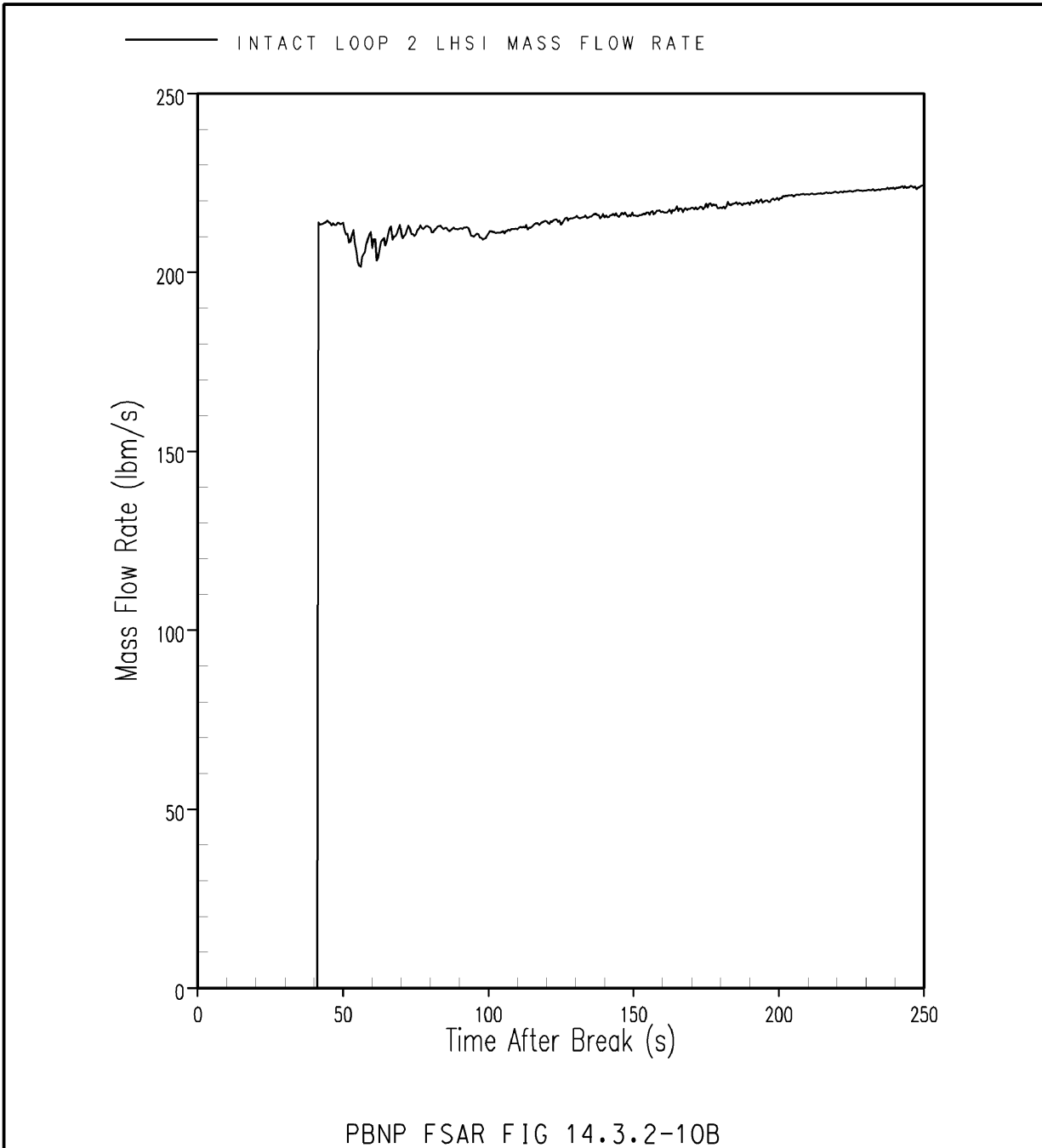
PBNP FSAR FIG 14.3.2-10B

00	08-09-10	INCORP. INTO PDS PER EC #15920		KJB	JLF	KJB
REV NO.	DATE	REVISION DESCRIPTION		DRAWN	CHK'D	APPRV'D

Figure 14.3.2-11 UNIT 1 LIMITING PEAK CLAD TEMPERATURE CASE CORE AVERAGE
CHANNEL COLLAPSED LIQUID LEVEL
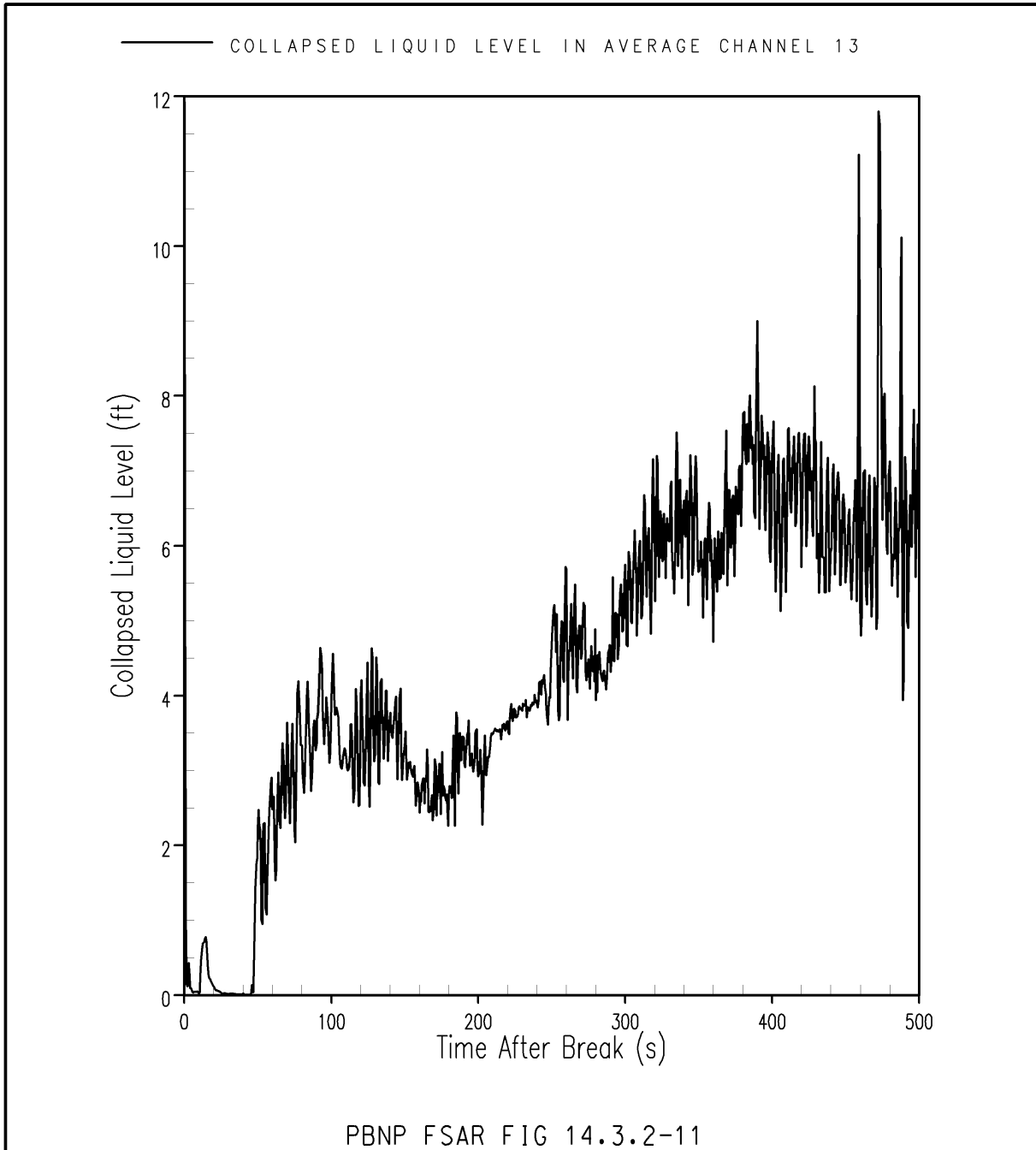
PBNP FSAR FIG 14.3.2-11

00	08-09-10	INCORP. INTO PDS PER EC #15920		KJB	JLF	KJB
REV NO.	DATE	REVISION DESCRIPTION		DRAWN	CHK'D	APPRV'D

Figure 14.3.2-12 UNIT 1 LIMITING PEAK CLAD TEMPERATURE CASE LOOP 2
DOWNCOMER COLLAPSED LIQUID LEVEL
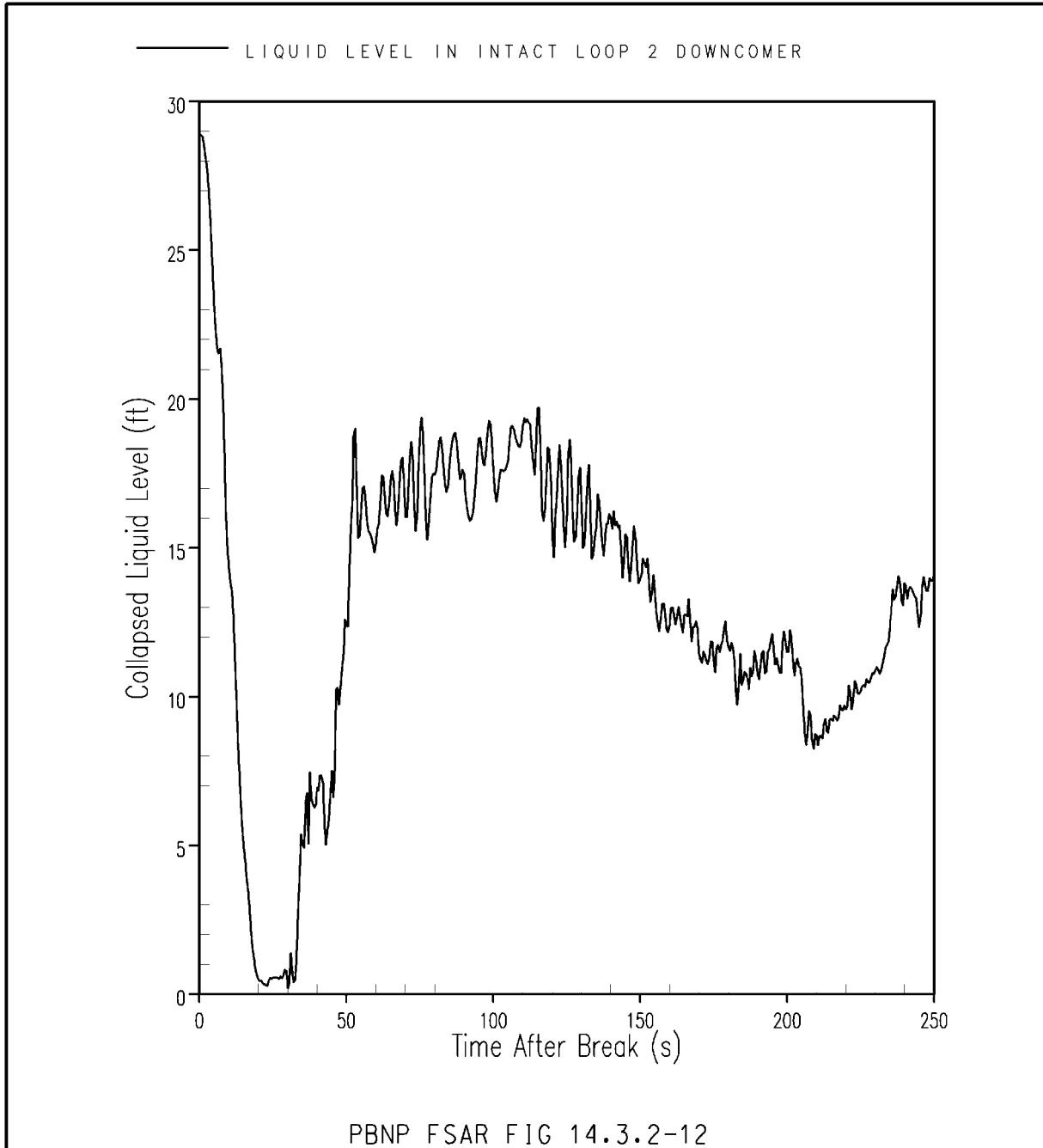
PBNP FSAR FIG 14.3.2-12

00	08-09-10	INCORP. INTO PDS PER EC #15920			KJB	JLF	KJB
REV NO.	DATE	REVISION DESCRIPTION			DRAWN	CHK'D	APPRV'D

Figure 14.3.2-13 UNIT 1 BELOCA ANALYSIS AXIAL POWER SHAPE OPERATING
SPACE ENVELOPE

PBOT/PMID Box for the Point Beach
BELOCA Project
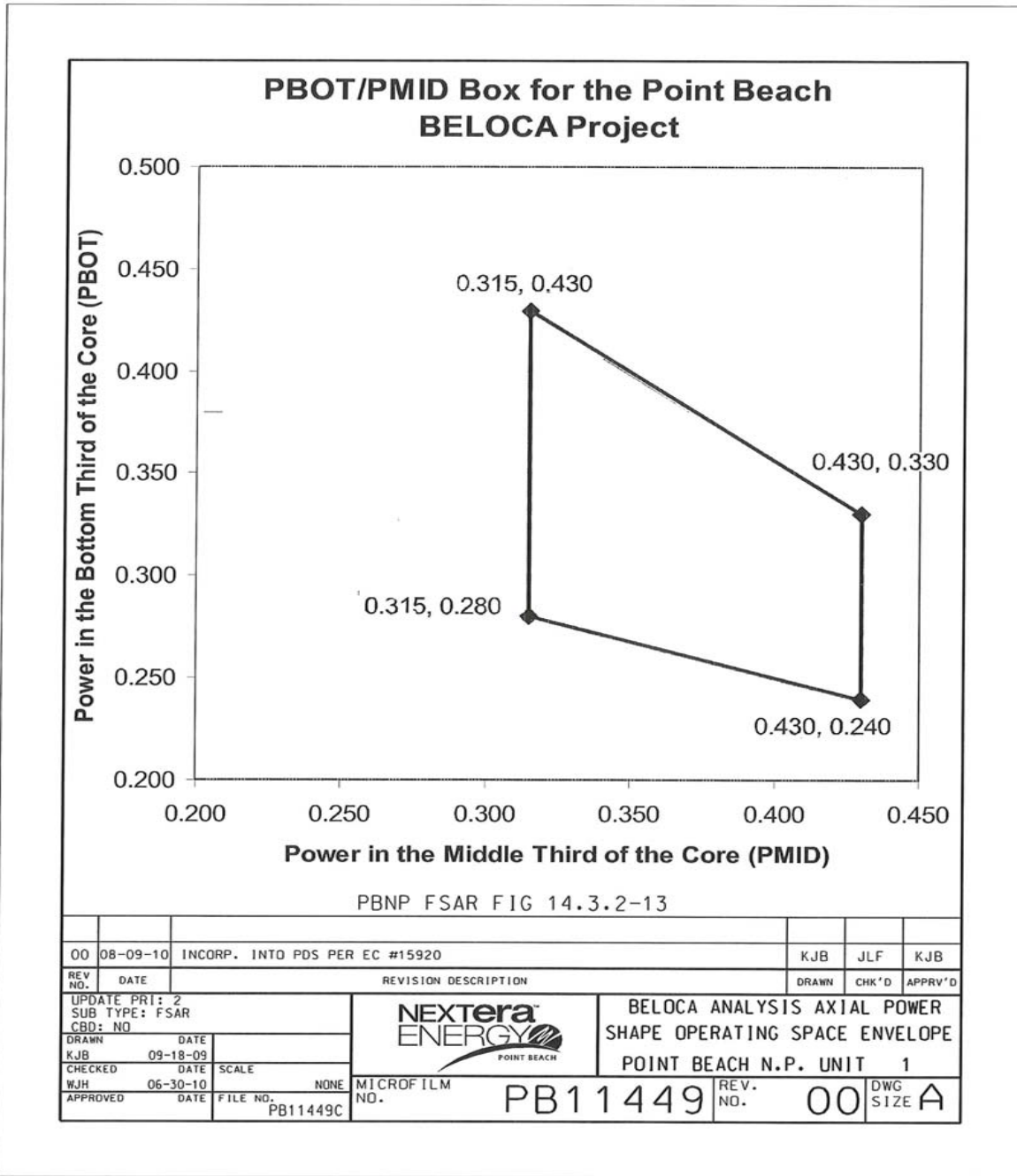
Figure 14.3.2-14 UNIT 1 LOWER BOUND CONTAINMENT PRESSURE

Point Beach Unit 1 Containment Pressure Comparison

PBNP FSAR FIG 14.3.2-14

00	08-09-10	INCORP. INTO PDS PER EC #15920	KJB	JLF	KJB
REV NO.	DATE	REVISION DESCRIPTION	DRAWN	CHK'D	APPRV'D
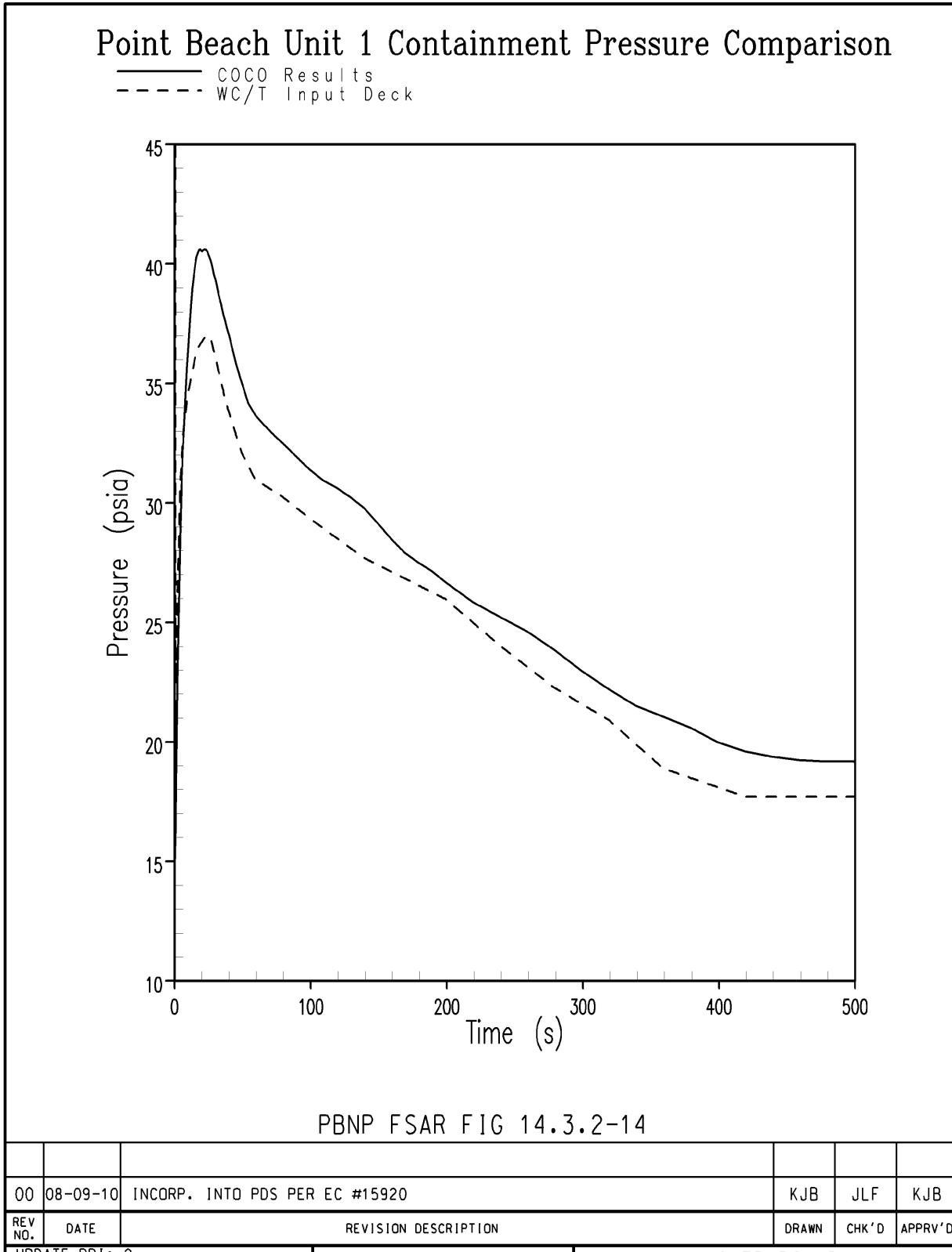
Figure 14.3.2-15 UNIT 2 LIMITING PEAK CLAD TEMPERATURE CASE PEAK CLAD
TEMPERATURE AND PEAK CLAD TEMPERATURE LOCATION

PBNP FSAR FIG 14.3.2-15

00	08-09-10	INCORP. INTO PDS PER EC #15920		KJB	JLF	KJB
REV NO.	DATE	REVISION DESCRIPTION		DRAWN	CHK'D	APPRV'D
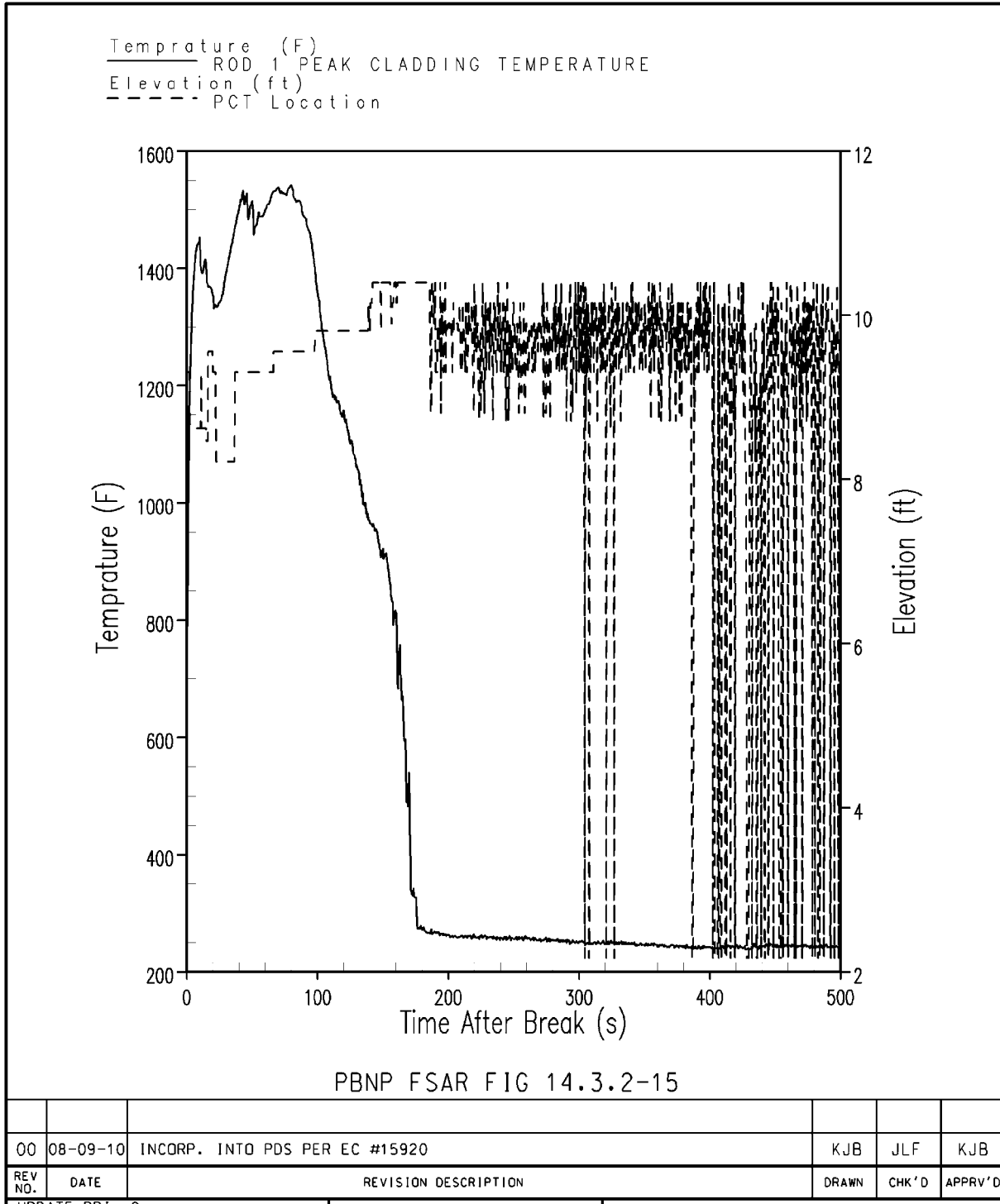
Figure 14.3.2-16 UNIT 2 LIMITING PEAK CLAD TEMPERATURE CASE VESSEL SIDE
BREAK FLOW
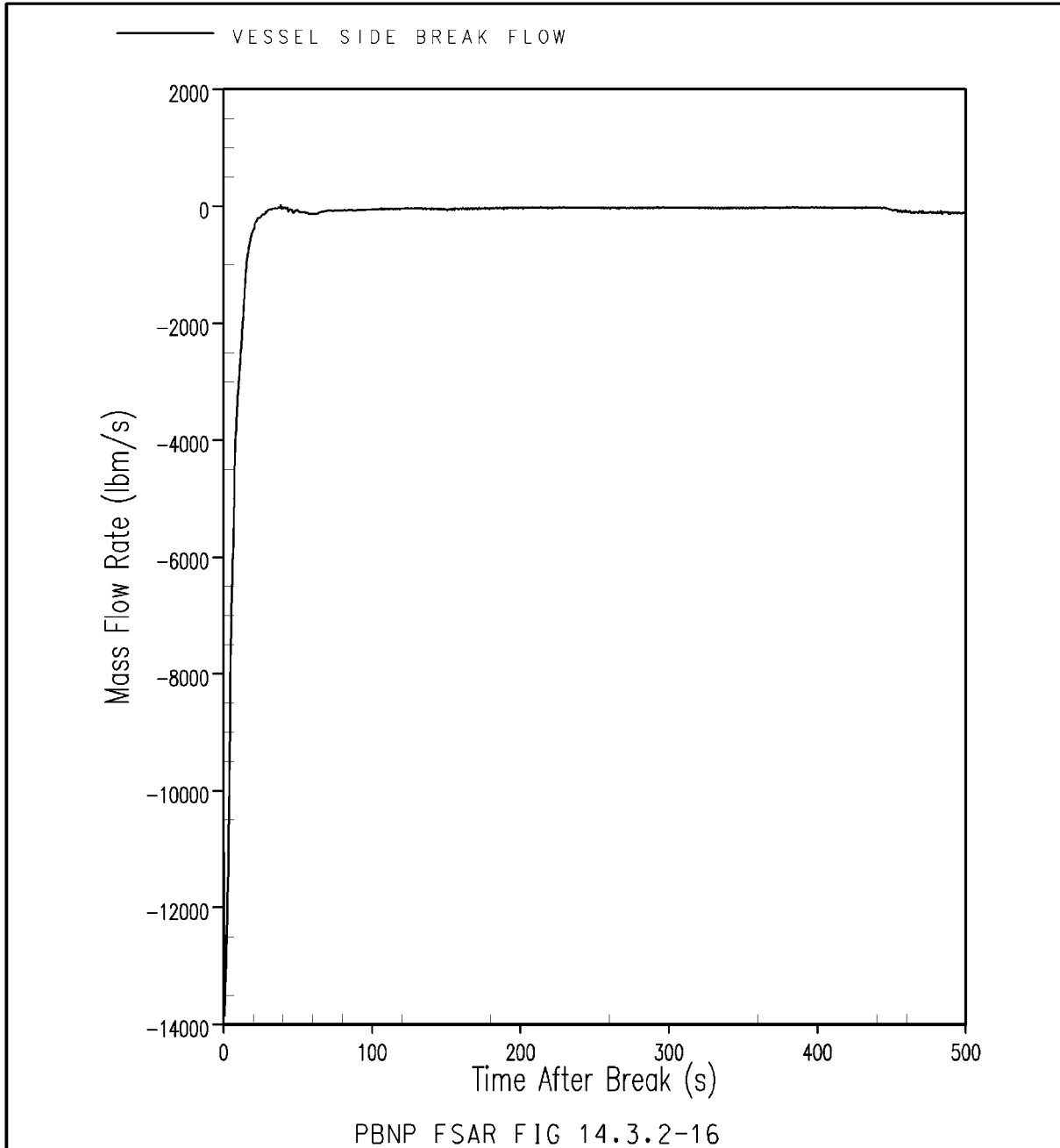
PBNP FSAR FIG 14.3.2-16

| 00 | 08-09-10 | INCORP. INTO PDS PER EC #15920 | | | KJB | JLF | KJB |
| REV NO. | DATE | REVISION DESCRIPTION | | | DRAWN | CHK'D | APPRV'D |

Figure 14.3.2-17 UNIT 2 LIMITING PEAK CLAD TEMPERATURE CASE PUMP SIDE
BREAK FLOW
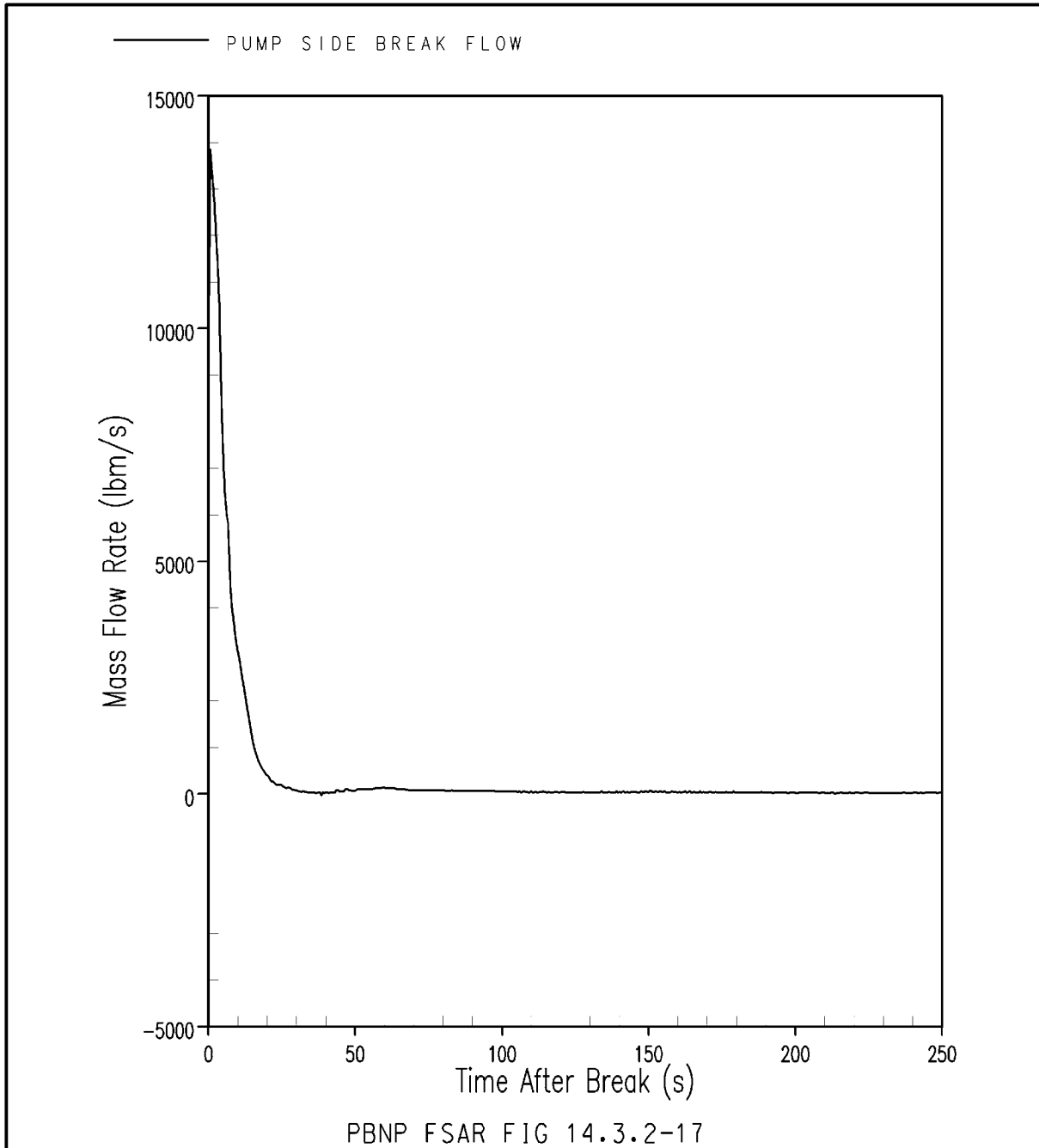
PBNP FSAR FIG 14.3.2-17

00	08-09-10	INCORP. INTO PDS PER EC #15920		KJB	JLF	KJB
REV NO.	DATE	REVISION DESCRIPTION		DRAWN	CHK'D	APPRV'D

Figure 14.3.2-18 UNIT 2 LIMITING PEAK CLAD TEMPERATURE CASE BROKEN AND INTACT LOOP PUMP VOID FRACTION
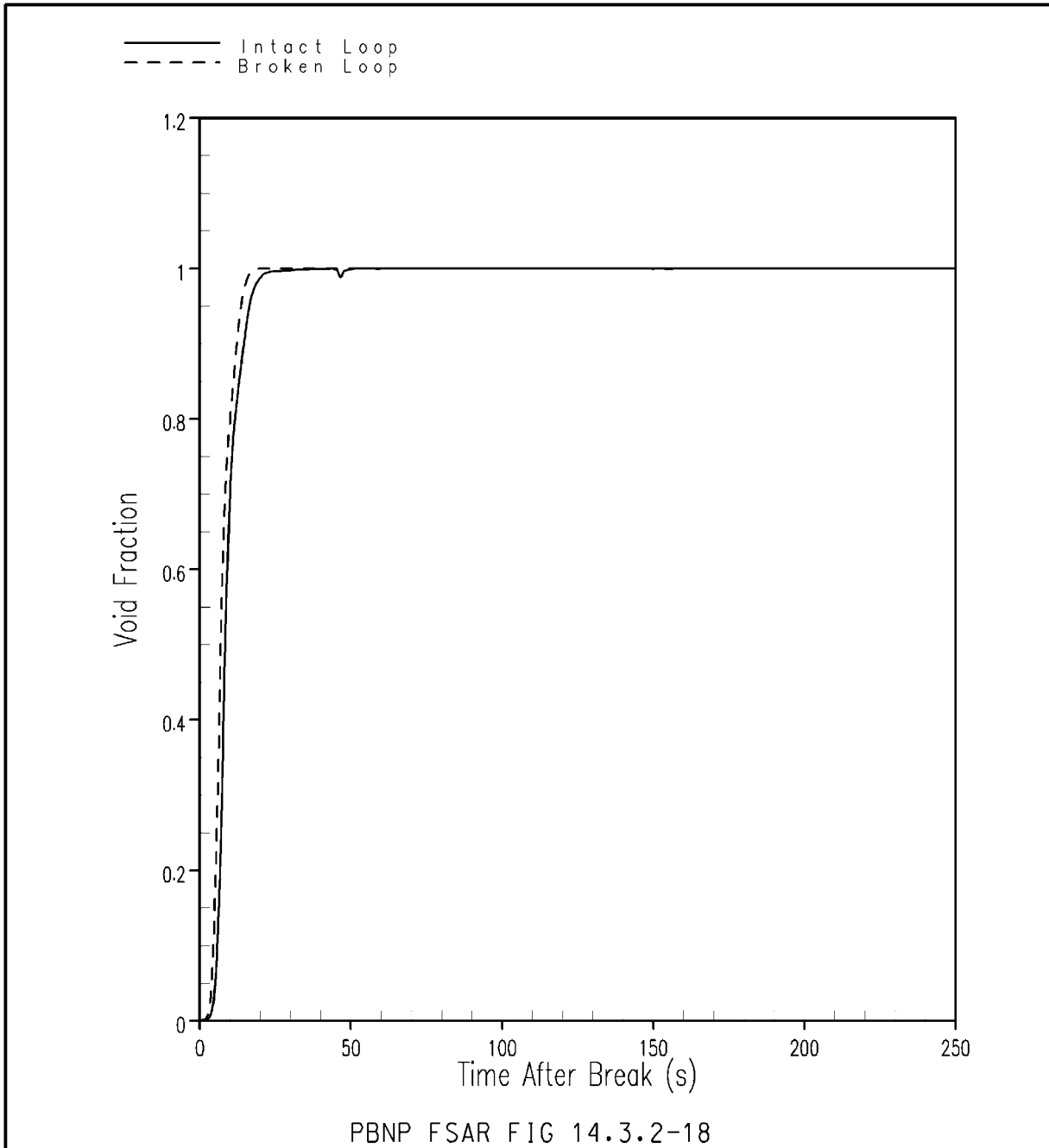
PBNP FSAR FIG 14.3.2-18

00	08-09-10	INCORP. INTO PDS PER EC #15920	KJB	JLF	KJB
REV NO.	DATE	REVISION DESCRIPTION	DRAWN	CHK'D	APPRV'D

Figure 14.3.2-19 UNIT 2 LIMITING PEAK CLAD TEMPERATURE CASE HOT ASSEMBLY
TOP THIRD OF CORE VAPOR FLOW
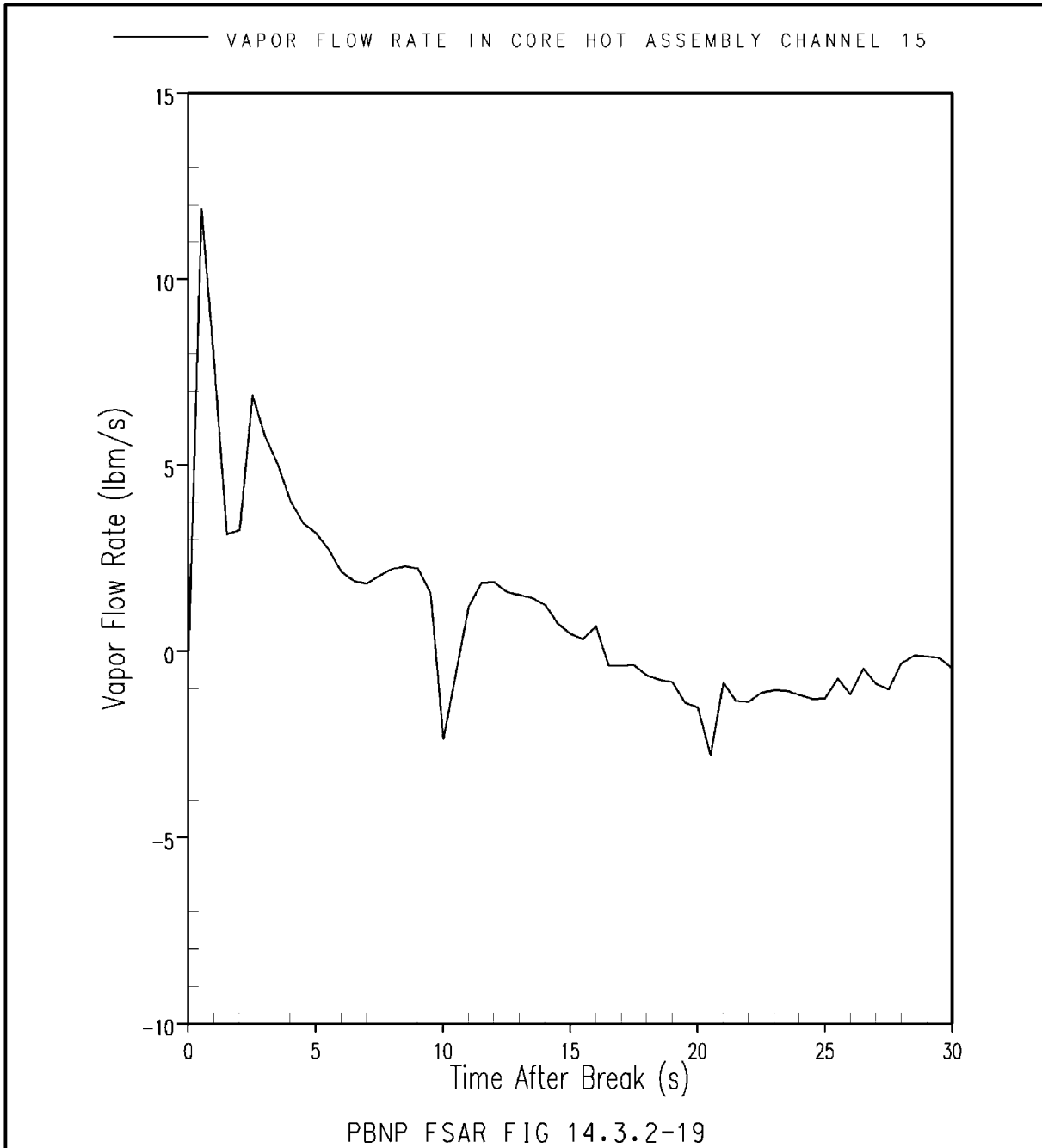
Figure 14.3.2-20 UNIT 2 LIMITING PEAK CLAD TEMPERATURE CASE PRESSURIZER
PRESSURE
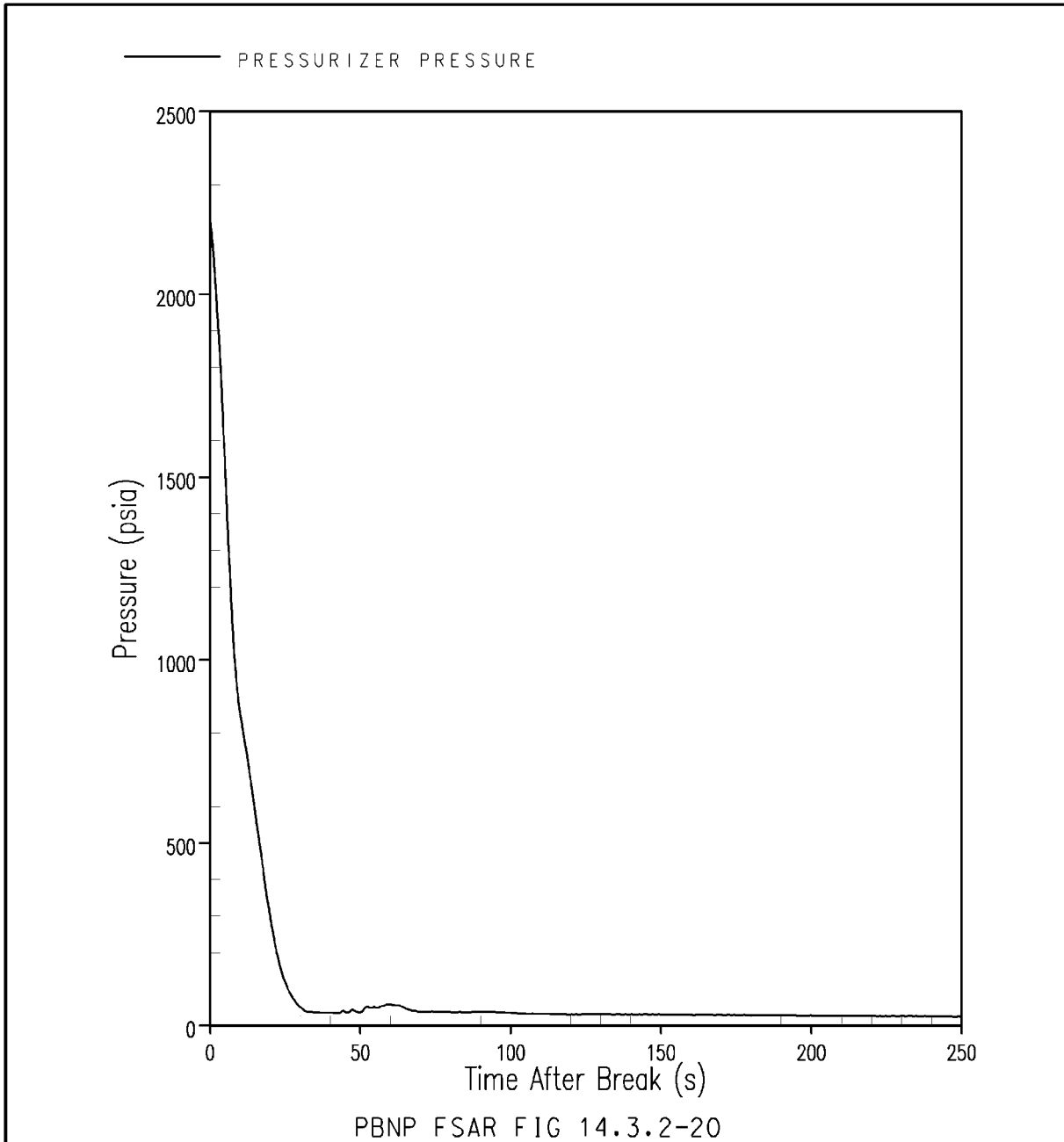
PBNP FSAR FIG 14.3.2-20

00	08-09-10	INCORP. INTO PDS PER EC #15920	KJB	JLF	KJB
REV NO.	DATE	REVISION DESCRIPTION	DRAWN	CHK'D	APPRV'D

Figure 14.3.2-21 UNIT 2 LIMITING PEAK CLAD TEMPERATURE CASE LOWER
PLENUM COLLAPSED LIQUID LEVEL
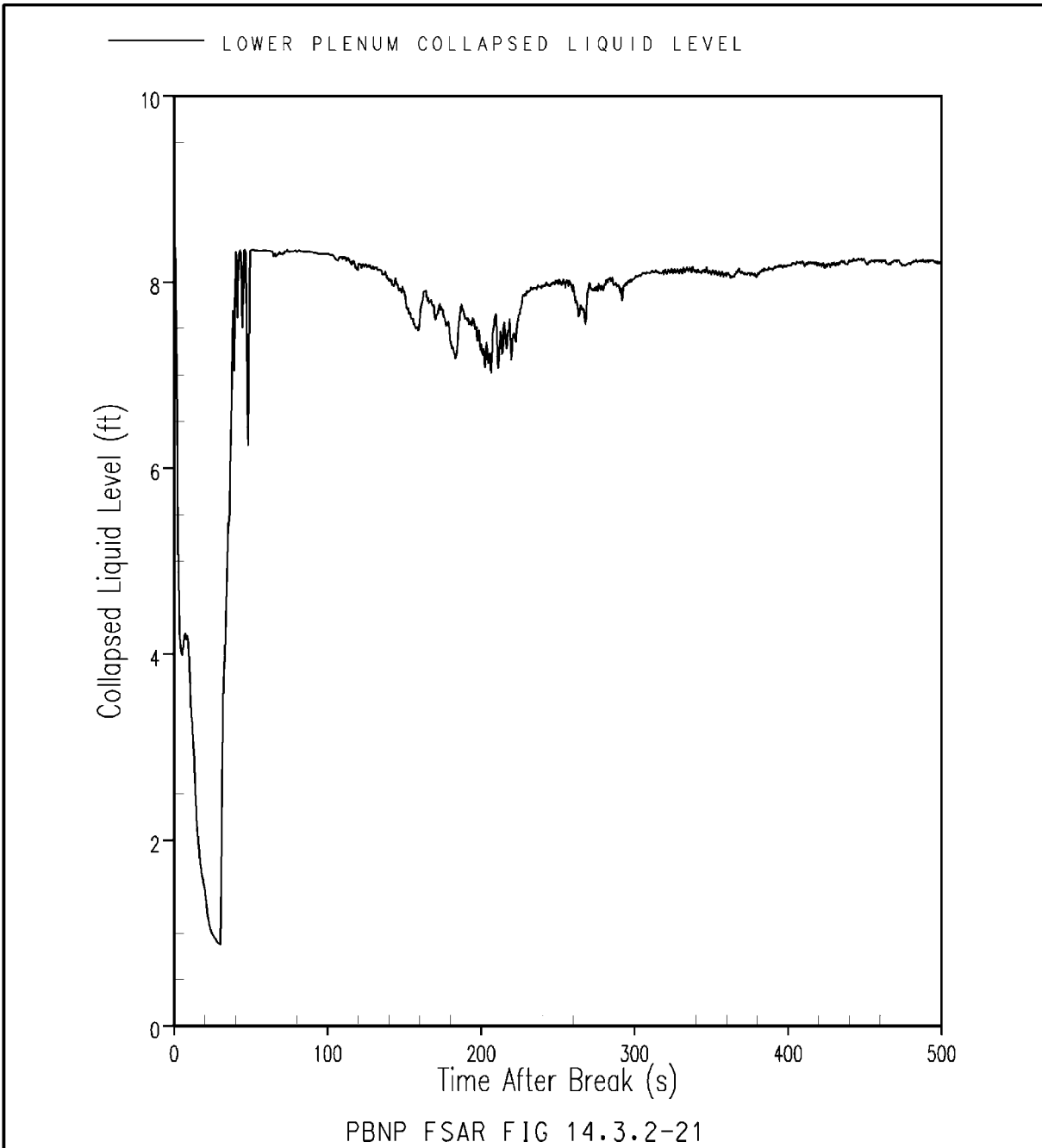
PBNP FSAR FIG 14.3.2-21

00	08-09-10	INCORP. INTO PDS PER EC #15920	KJB	JLF	KJB
REV NO.	DATE	REVISION DESCRIPTION	DRAWN	CHK'D	APPRV'D

Figure 14.3.2-22 UNIT 2 LIMITING PEAK CLAD TEMPERATURE CASE VESSEL LIQUID
         MASS
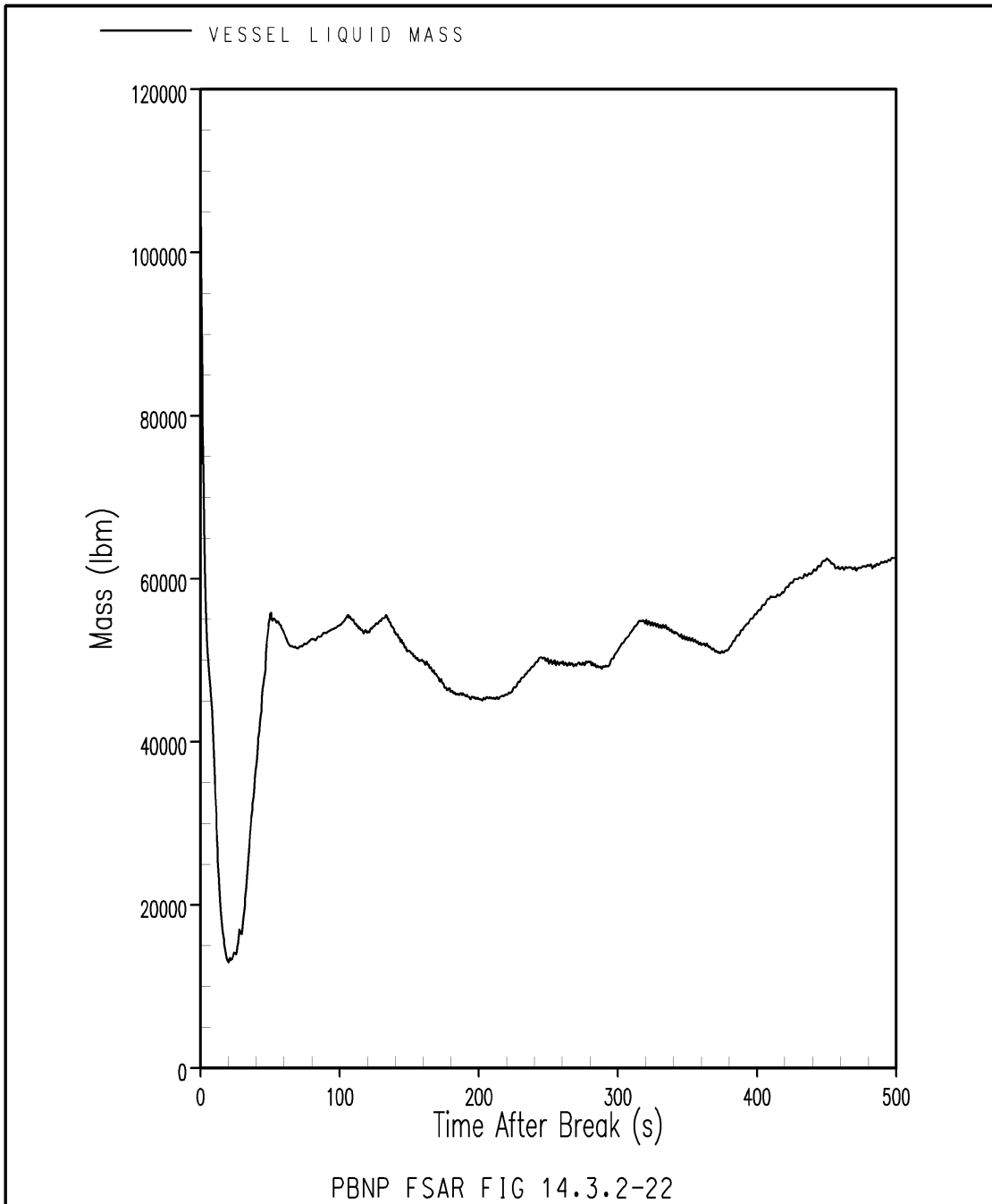
PBNP FSAR FIG 14.3.2-22

| 00 | 08-09-10 | INCORP. INTO PDS PER EC #15920 | | KJB | JLF | KJB |
| REV NO. | DATE | REVISION DESCRIPTION | | DRAWN | CHK'D | APPRV'D |

Figure 14.3.2-23 UNIT 2 LIMITING PEAK CLAD TEMPERATURE CASE LOOP 2
ACCUMULATOR FLOW
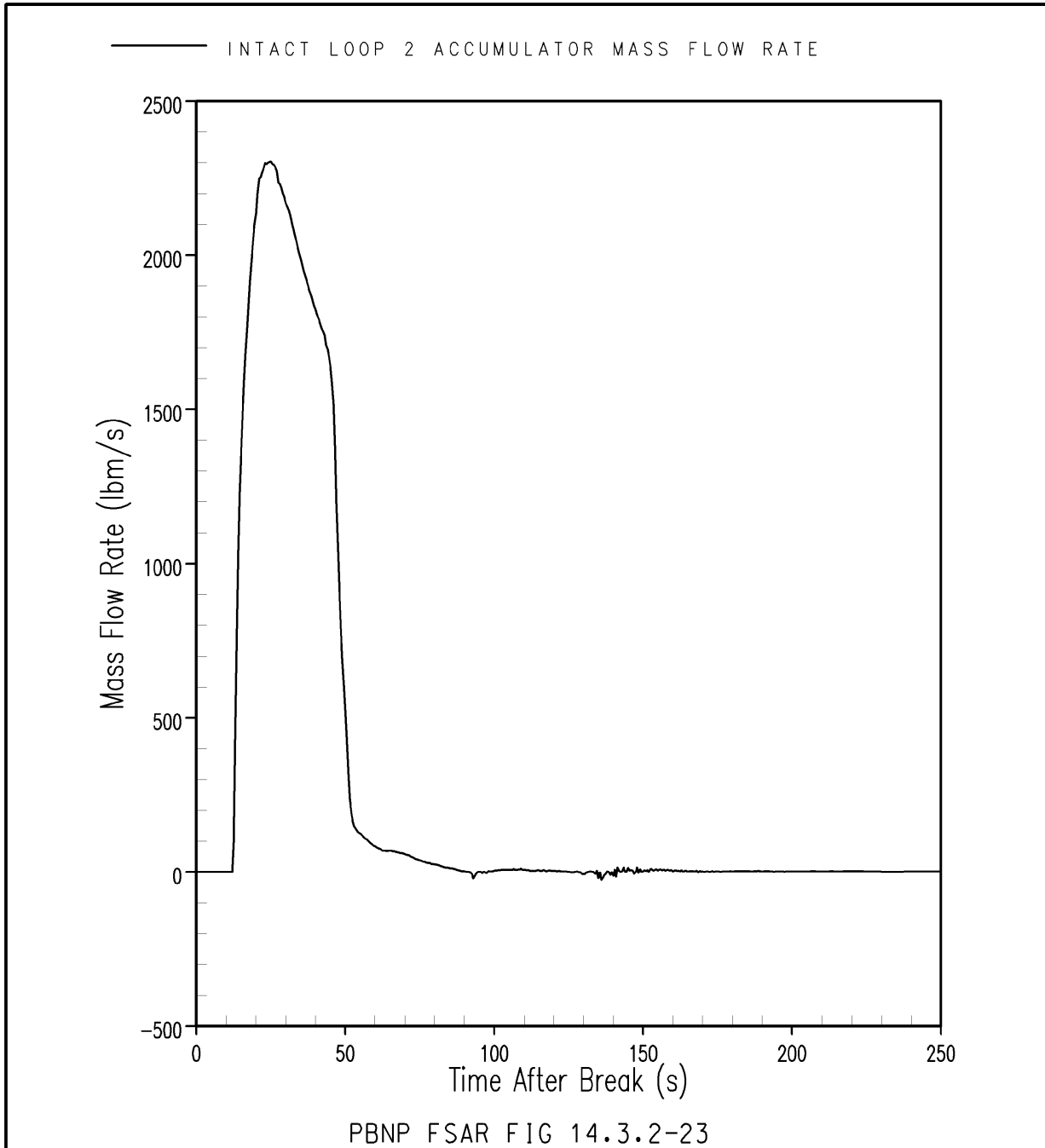
PBNP FSAR FIG 14.3.2-23

00	08-09-10	INCORP. INTO PDS PER EC #15920		KJB	JLF	KJB
REV NO.	DATE	REVISION DESCRIPTION		DRAWN	CHK'D	APPRV'D

Figure 14.3.2-24A UNIT 2 LIMITING PEAK CLAD TEMPERATURE CASE LOOP 2 HIGH
HEAD SAFETY INJECTION FLOW
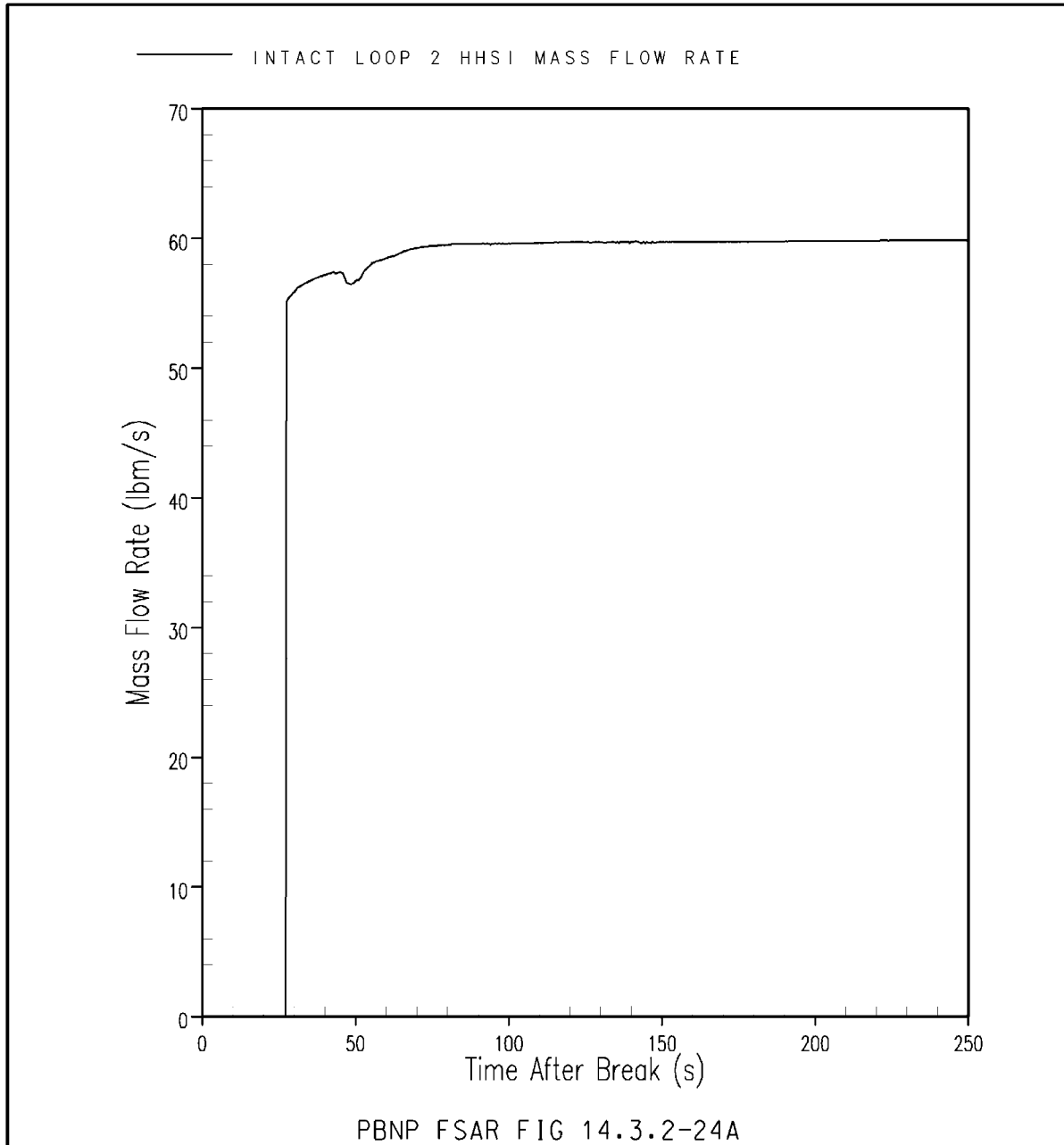
PBNP FSAR FIG 14.3.2-24A

REV NO.	DATE	REVISION DESCRIPTION	DRAWN	CHK'D	APPRV'D
00	08-09-10	INCORP. INTO PDS PER EC #15920	KJB	JLF	KJB

Figure 14.3.2-24B UNIT 2 LIMITING PEAK CLAD TEMPERATURE CASE LOOP 2 LOW
HEAD SAFETY INJECTION FLOW
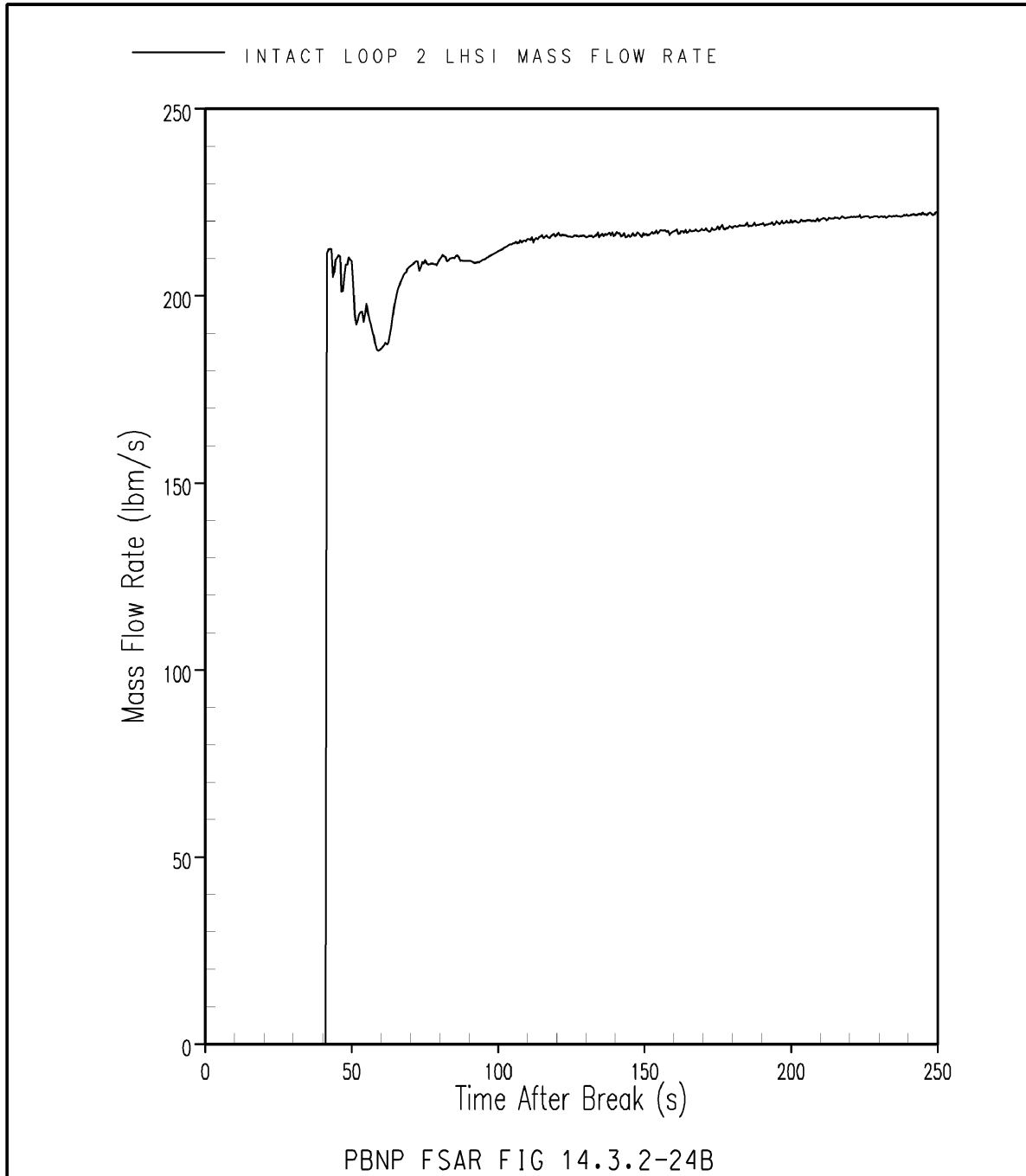
PBNP FSAR FIG 14.3.2-24B

00	08-09-10	INCORP. INTO PDS PER EC #15920	KJB	JLF	KJB
REV NO.	DATE	REVISION DESCRIPTION	DRAWN	CHK'D	APPRV'D

Figure 14.3.2-25 UNIT 2 LIMITING PEAK CLAD TEMPERATURE CASE CORE AVERAGE CHANNEL COLLAPSED LIQUID LEVEL
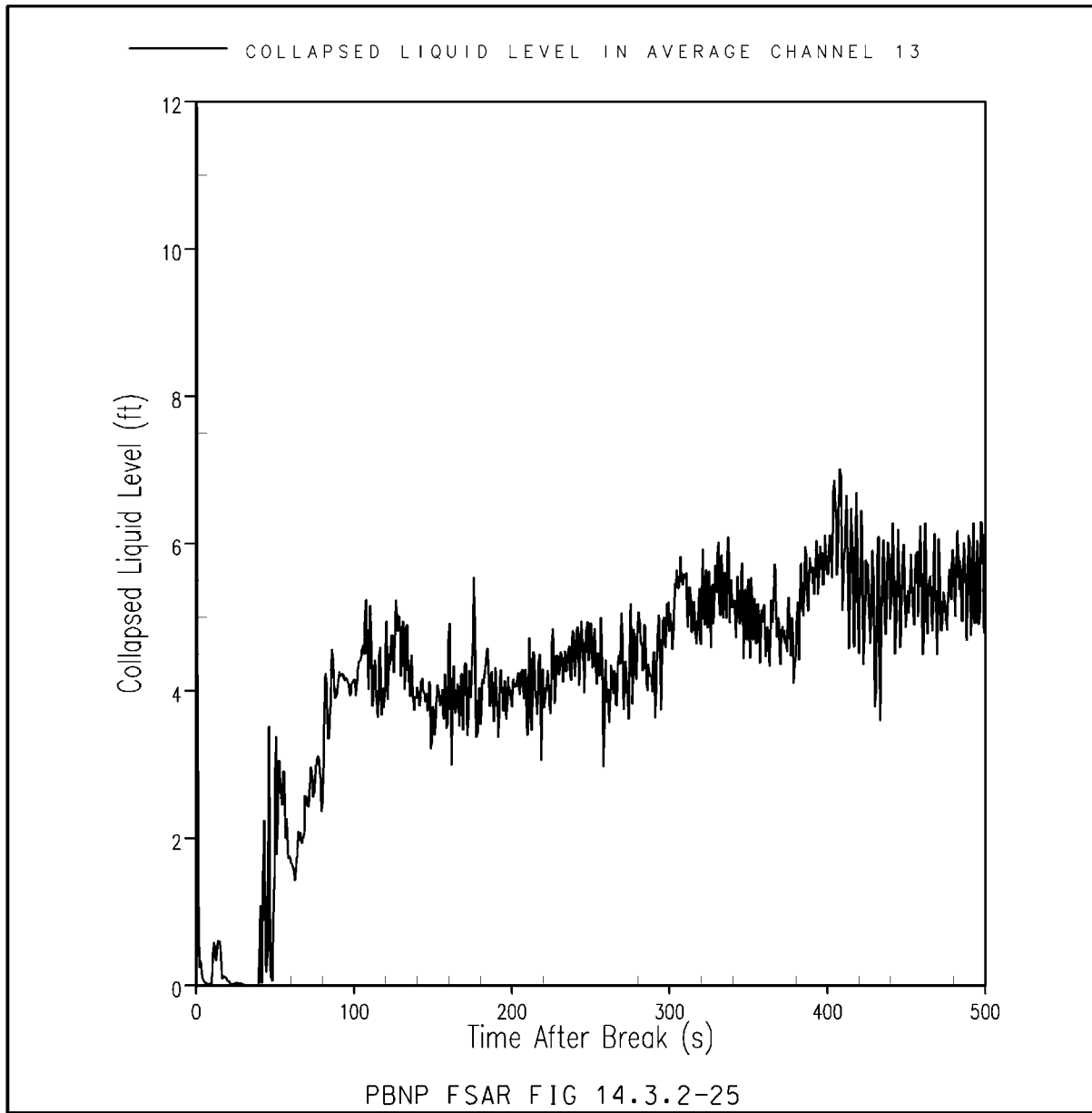
PBNP FSAR FIG 14.3.2-25

| 00 | 08-09-10 | INCORP. INTO PDS PER EC #15920 | | | KJB | JLF | KJB |
| REV NO. | DATE | REVISION DESCRIPTION | | | DRAWN | CHK'D | APPRV'D |

Figure 14.3.2-26 UNIT 2 LIMITING PEAK CLAD TEMPERATURE CASE LOOP 2 DOWNCOMER COLLAPSED LIQUID LEVEL
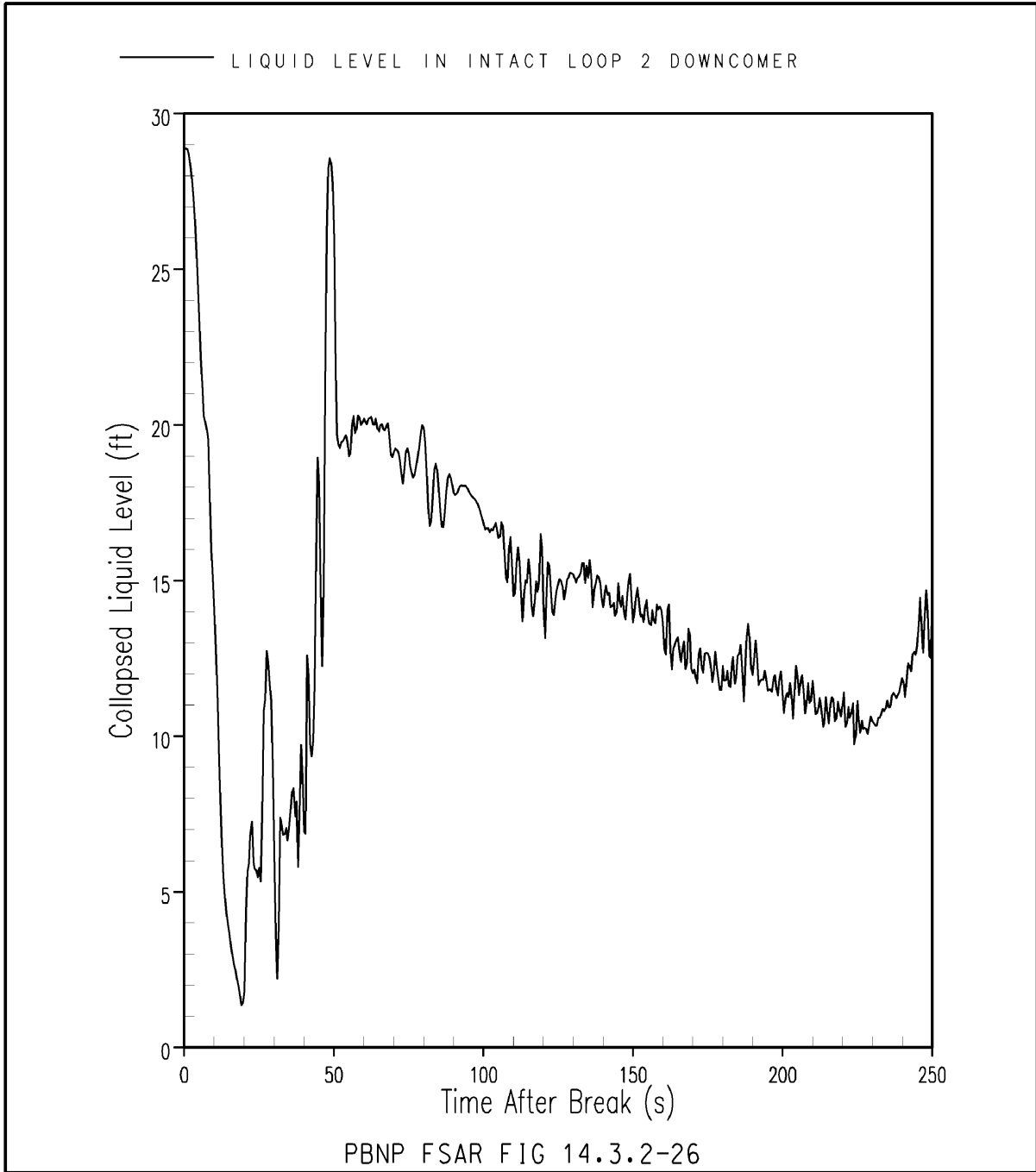
PBNP FSAR FIG 14.3.2-26

00	08-09-10	INCORP. INTO PDS PER EC #15920	KJB	JLF	KJB
REV NO.	DATE	REVISION DESCRIPTION	DRAWN	CHK'D	APPRV'D

Figure 14.3.2-27 UNIT 2 BELOCA ANALYSIS AXIAL POWER SHAPE OPERATING SPACE ENVELOPE

PBOT/PMID Box for the Point Beach BELOCA Project

PBNP FSAR FIG 14.3.2-27

00	08-09-10	INCORP. INTO PDS PER EC #15920		KJB	JLF	KJB
REV NO.	DATE	REVISION DESCRIPTION		DRAWN	CHK'D	APPRV'D
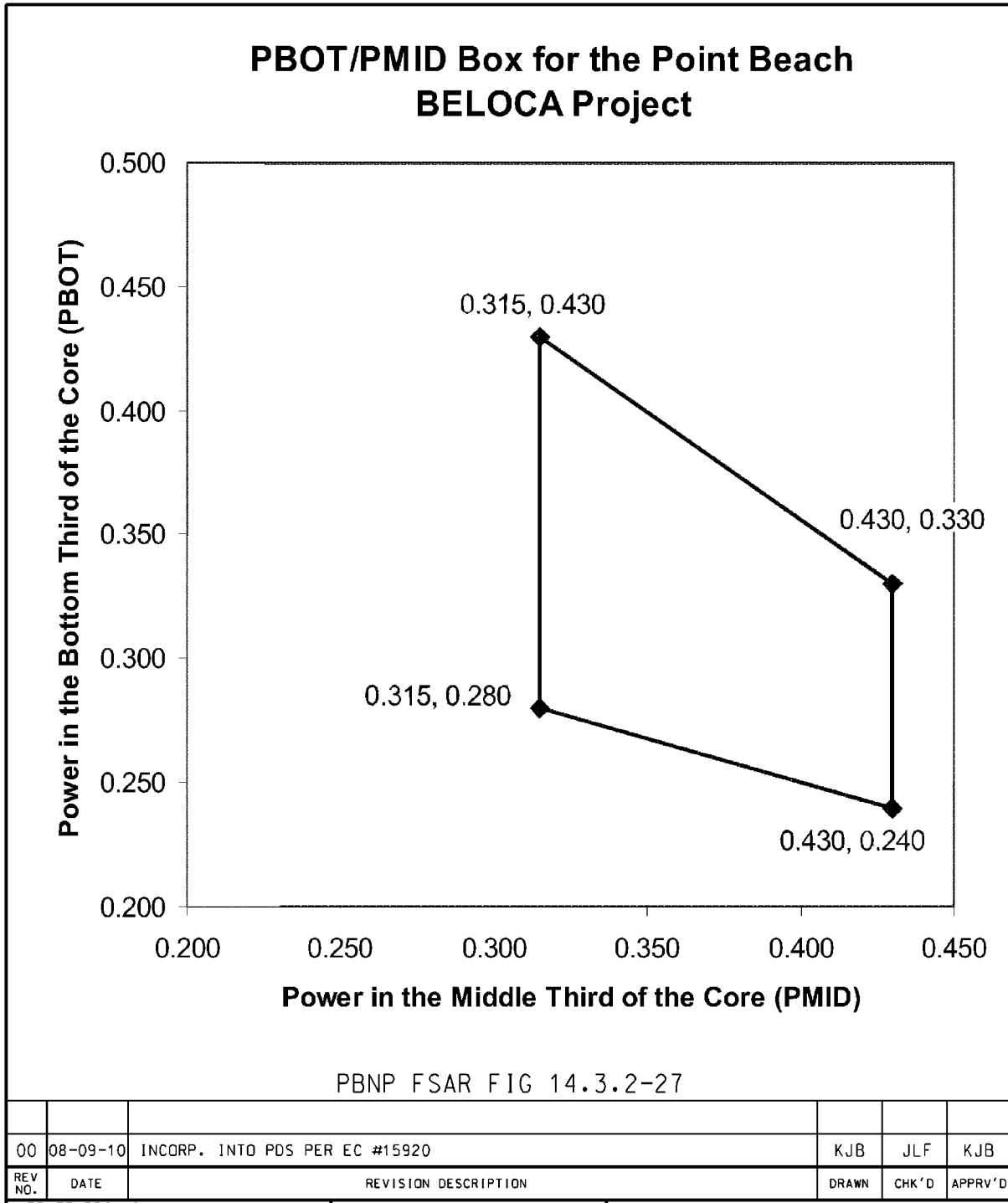
Figure 14.3.2-28 UNIT 2 LOWER BOUND CONTAINMENT PRESSURE

Point Beach 2 Containment Pressure Comparison

PBNP FSAR FIG 14.3.2-28

00	08-09-10	INCORP. INTO PDS PER EC #15920	KJB	JLF	KJB
REV NO.	DATE	REVISION DESCRIPTION	DRAWN	CHK'D	APPRV'D
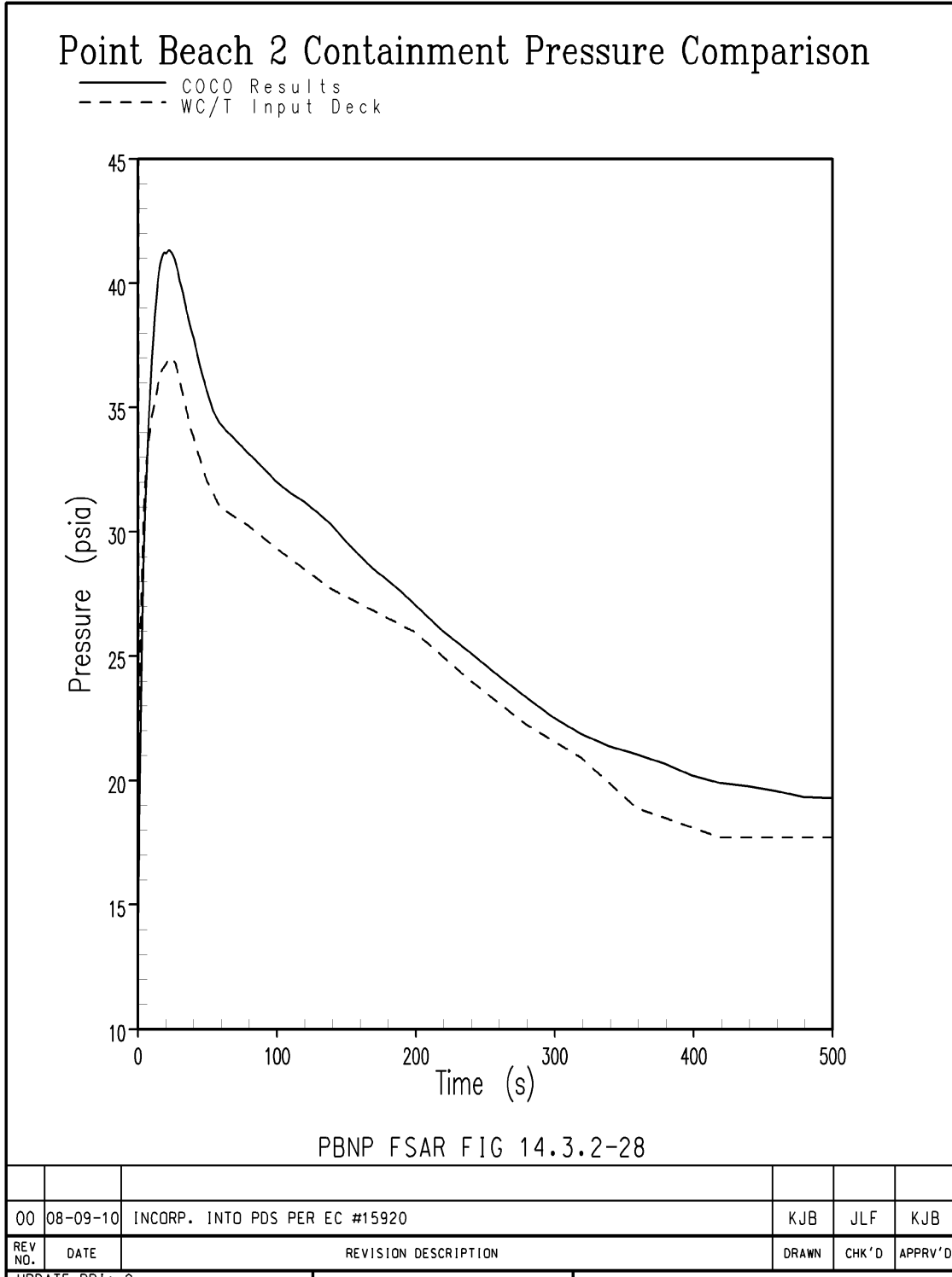
14.3.3 CORE AND INTERNALS INTEGRITY ANALYSIS

Internals Evaluation

The forces exerted on reactor internals and core, following a loss-of-coolant accident, are computed by employing the MULTIFLEX 3.0 (Reference 9) digital computer program developed for the space-time-dependent analysis of multi-loop PWR plants.

Design Criteria

The criteria for acceptability are that the core should be coolable and intact following a pipe rupture on the postulated 3" schedule 160 charging line (cold leg) or on the 6" schedule 120 capped line (hot leg) for extended power uprate (EPU) conditions. This implies that core cooling and adequate core shutdown must be assured. Consequently, the limitations established on the internals are concerned principally with the maximum allowable deflections and/or stability of the parts (Reference 7).

Critical Internals - Upper Barrel

The upper barrel deformation has the following limits:

> To assure reactor trip and to avoid disturbing the RCC guide structure, the barrel should not interfere with any guide tubes. This condition requires a stability check to assure that the barrel will not buckle under the accident loads.

Critical Internals - RCC Guide Tubes

The RCC guide tubes in the upper core support package have the following allowable limits. Tests were conducted on guide tubes to measure the insertion time versus guide tube deflection, with lateral forces simulating flow or inertia forces that the guide tubes are subjected to during a faulted event. The higher the lateral forces, the longer the insertion time. At a point, defined as the "allowable load," the guide tube would lose its function, and the control rods would no longer be able to insert. The allowable load for the guide tubes used at Point Beach Units 1 and 2 (i.e., 14x14 118-inch), derived from tests conducted on similar guide tubes, was established to be 15,500 pounds.

The maximum combined load during a safe-shutdown earthquake (SSE) and a loss-of-coolant accident (LOCA) event, as defined above, was calculated to be 14,580 pounds, which is less than the allowable load of 15,500 pounds. Therefore, control rod insertion for EPU conditions is not an issue (Reference 8).

Critical Internals - Fuel Assemblies

The limitations for this case are related to the stability of the thimbles at the upper end. During the accident, the fuel assembly will have a vertical displacement and could touch the upper package subjecting the components to dynamic stresses. The upper end of the thimbles shall not experience stresses above the buckling compressive stresses because any buckling of the upper end of the thimbles will distort the guide line and could affect the fall of the control rod.

Critical Internals - Upper Package

The maximum allowable local deformation of the upper core plate where a guide tube is located is 0.100 inch. This deformation will cause the plate to contact the guide tube since the clearance between plate and guide tube is 0.100 inch. This limit will prevent the guide tubes from being put in compression. In order to maintain the straightness of the guide tube, a maximum allowable total deflection of 1 inch for the upper support plate and deep beam has been established. The corresponding no loss of function deflection is above 2 inches.

Allowable Stress Criteria

The allowable stress criteria fall into two categories dependent upon the nature of the stress state: membrane or bending. A direct state of stress (Membrane) has a uniform stress distribution over the cross section. The allowable (Maximum) membrane or direct stress is taken to be equal to the stress corresponding to 0.2 of the uniform material strain or the yield strength, whichever is higher. For unirradiated 304 stainless steel at operating temperature, the stress corresponding to 20% of the uniform strain is:

$$(S_m)_{allowable} = 39500 \text{ psi}$$

For irradiated materials, the limit stress is higher. For a bending state of stress, the strain is linearly distributed over a cross section. The average strain value is, therefore, one half of the outer fiber strain where the stress is a maximum. Thus, by requiring the average strain to satisfy an allowable criterion similar to that for the direct state of stress, the outer fiber strain may be 0.4 times the uniform strain. The maximum allowable outer fiber bending stress is then taken to be equal to the stress corresponding to 40% of the uniform strain or the yield strength, whichever is higher. For unirradiated 304 stainless steel at operating temperature, we obtain from the stress strain curve:

$$(S_b)_{allowable} = 50,000 \text{ psi}$$

For combinations of membrane and bending stresses, the maximum allowable stress is taken to be equal to the stress corresponding to the maximum outer fiber strain not in excess of 40% uniform strain and average strain not in excess of 20% uniform strain.

Blowdown and Force Analysis - Blowdown Model

MULTIFLEX 3.0 (Reference 9) is a digital computer program for calculation of local fluid pressure, flow, and density transients that occur in the reactor primary coolant systems during a loss of coolant accident. This program applies to the subcooled, transition, and saturated two-phase blowdown regimes. This is in contrast to programs, such as WHAM (Reference 1), which are applicable only to the subcooled region and which, due to their method of solution, could not be extended into the region in which large changes in the sonic velocities and fluid densities take place.

MULTIFLEX 3.0 (Reference 9) is based on the method of characteristics wherein the resulting set of ordinary differential equations, obtained from the laws of conservation of mass, momentum, and energy, are solved numerically utilizing a fixed mesh in both space and time.

Although spatially one-dimensional conservation laws are employed, the code can be applied to describe 3-dimensional system geometries through the use of the equivalent piping networks. Such piping networks may contain any number of pipes or channels of various diameters, dead ends, branches (with up to six pipes connected to each branch), contractions, expansions, orifices, pumps, and free surfaces (such as in a pressurizer). All types of the system losses (such as friction, contraction, expansion, etc.) are considered.

Force Model

MULTIFLEX 3.0 (Reference 9) evaluates the pressure and velocity transients for a maximum of 2000 locations throughout the system. These pressure and velocity transients are made available to the programs LATFORC and FORCE 2 which utilize a detailed geometric description in evaluating the loadings on the reactor internals.

Each reactor component for which vertical force calculations are required is designated as an element and assigned an element number. Vertical forces acting upon each of the elements are calculated summing the effects of:

1. The pressure differential across the element
2. Flow stagnation on, and unrecovered orifice losses across the element
3. Friction losses along the element

Input to the code, in addition to the MULTIFLEX 3.0 (Reference 9) pressure and velocity transients, includes the effective area of each element on which acts the force due to the pressure differential across the element, a coefficient to account for flow stagnation and unrecovered orifice losses, and the total area of the element along which the shear forces act.

In addition to the vertical forces calculated by FORCE2, the horizontal forces on the vessel, core barrel, and thermal shield are calculated by LATFORC. The horizontal forces are calculated by summing the lateral force components around the vessel, core barrel and thermal shield, based on the pressure differential across each section, multiplied by the area of each section. This is done at ten different elevations. The total lateral force is calculated by summing the forces over the ten elevations.

Vertical Excitation - Structural Model and Method of Analysis

The response of reactor internals components due to an excitation produced by complete severance of a branch line pipe is analyzed. Assuming a pipe break occurs in a very short period of time, the rapid drop of pressure at the break produces a disturbance which propagates along the primary loop and excites the internal structure.

The internal structure is simulated by a multi-mass system connected with springs and dashpots representing the viscous damping due to structural and impact losses. The gaps between various components, as well as Coulomb type of friction, is also incorporated into the overall model. Since the fuel elements in the fuel assemblies are kept in position by friction forces originating from the preloaded fuel assembly grid fingers, any sliding that occurs between the fuel rods and assembly is considered as Coulomb type of friction. A series of mechanical models of local structures have been developed and analyzed so that certain basic nonlinear phenomena previously mentioned could be understood. Using the results of these models, a final eleven-mass

model is adopted to represent the internals structure under vertical excitation. Figure 14.3.3-1 is a schematic representation of the internals structures. The eleven-mass model is shown in Figure 14.3.3-2. A comparison between Figure 14.3.3-1 and Figure 14.3.3-2 shows the parallel between the plant and the model. The modeling is conducted in such a way that uniform masses are lumped into easily identifiable discrete masses while elastic elements are represented by springs. A legend for the different masses is given in Table 14.3.3-1. The masses are readily recognized as Items W1 through W11. The core barrel and the lower package are easily discernible. The fuel assemblies have been segregated into two groups. The majority of the fuel mass, W4, is indirectly connected to the deep beam structure represented by mass W8. There is also a portion of the fuel mass, W6, which connects through the long columns to the top plate. The stiffness of the top plate panels is represented by K8. The hold down spring, K1, is bolted-up between the flange of the deep beam structure and the core barrel flange with the preload, Pl. After preloading the hold down spring, a clearance, G1, exists between the core barrel flange and the solid height of the hold down spring. Within the fuel assemblies, the fuel elements W4 and W6 are held in place by frictional contact with the grid spring fingers. Coulomb damping is provided in the analysis to represent this frictional restraint.

The analytical model is also provided with viscous terms to represent the structural damping of the elastic elements. The viscous dampers are represented by C1 through C11.

Restrictions are placed on the displacement amplitudes by specifying the free travel available to the dynamic masses. Available displacements are designated by symbols G1 through G8.

The displacements are tested during the solution of the problem to see if the available travel has been achieved. When the limit of travel has been attained, stops are engaged to arrest further motion of the dynamic masses. The stops of snubbers are designated by the symbols S1 through S11. Contact with the snubbers results in some damping of the motion of the model. The impact damping of the snubbers is represented by the devices D1 through D11.

During the assembly of the reactor, bolt-up of the closure head presets the spring loading of the core barrel and the spring loading on the fuel assemblies. Since the fuel assemblies in the model have been segregated into two groups, two preload values are provided in the analysis. Preload values P1, P3, and P5 represent the hold down spring preload on the core barrel and the top nozzle springs preload values on the fuel assemblies.

The formulation of the transient motion response problem and digital computer programming have been performed. The effects of an earthquake vertical excitation are also incorporated into the program.

In order to program the multi-mass system, the appropriate spring rates, weights, and forcing function for the various masses were determined. The spring rates and weights of the reactor components are calculated separately for each plant. The forcing functions for the masses are obtained from the FORCE2 program described in the previous section. It calculates the transient forces on reactor internals during blowdown using transient pressures and fluid velocities.

For the blowdown analysis the forcing functions are applied directly to the various internal masses.

For the earthquake analysis of the reactor internals, the forcing function, which is simulated earthquake response, is applied to the multi-mass system at the ground connections (the reactor vessel). Therefore, the external excitation is transmitted to the internals through the springs at the ground connections.

Results

Analysis was performed for a 1 millisecond opening time, and for hot leg and cold leg branch line breaks. The response of the structure to these excitations indicates that the vertical motion is irregular with peaks of very short duration. The deflections and motion of some of the reactor components are limited by the solid height of springs as is also the hold down spring located above the barrel flange.

The internals behave as a nonlinear system during the vertical oscillations produced by the blowdown forces. The nonlinearities are due to the Coulomb frictional forces between grids and rods, and to gaps between components causing discontinuities in force transmission. The frequency response is consequently a function not only of the exciting frequencies in the system, but also of the amplitude. Different break conditions excite different frequencies in the system. This situation can be seen clearly when the response under blowdown forces is compared with the one due to vertical seismic acceleration. Under seismic excitation, the system behaves almost linearly because component motion is not sufficient to cause closing of the various gaps in the structure or slippage in the fuel rods.

Under certain blowdown excitation conditions, the core moves upward, touches the core plate, and falls down on the lower structure causing oscillations in all the components. During the time that the oscillations occur and, depending on its initial position, the fuel rods slide on the fuel assembly. The response shows that the case could be represented as two large vibrating masses (the core and the barrel), and the rest of the system oscillates with respect to the barrel and the core. Damping effects have also been considered; it appears that the higher frequencies disappear rapidly after each impact of slippage.

The results of the computer program give not only the frequency response of the components, but also the maximum impact force and deflections. From these results, the stresses are computed using the standard "Strength of Material" formulas. The impact stresses are obtained in an analogous manner using the maximum forces seen by the various structures during impact.

Baffle Former Bolt Replacement (Unit 2 Only)

Point Beach Unit 2 was selected as a lead plant to collect information regarding a baffle former bolt cracking phenomena observed in some foreign nuclear power plants. All baffle former bolts were inspected and a select pattern of baffle former bolts were replaced using methodology presented WCAP-15133 "Determination of Acceptable Baffle-Barrel-Bolting for Point Beach Units 1 and 2". This replacement also included one bolt in a "non-critical" location which was removed and not replaced. WCAP-15133 was based on an NRC accepted methodology presented in WCAP-15029 "Westinghouse Methodology for Evaluating the Acceptability of Baffle-Former-Barrel Bolting Distributions Under Faulted Load Conditions." This methodology and its use were strictly limited to the baffle former bolt project (Reference 3 through Reference 6).

Aging Management Program

The Aging Management Program, Reactor Vessel Internals Program (FSAR Section 15.2.17) provides additional information for monitoring during the period of extended operation (NRC SE dated 12/2005, NUREG-1839).

References

1. S. Fabic: "Computer Program WHAM for Calculation of Pressure, Velocity, and Force Transients in Liquid Filled Piping Networks," Kaiser Engineers Report No. 67-49-R (November 1967).

2. K. Takeuchi: "MULTIFLEX, A FORTRAN-IV Computer Program for Analyzing Thermal-Hydraulic-Structure System Dynamics," WCAP 8708-PA, WCAP 8709-A (Non-proprietary), September, 1977.

3. WCAP-15133 "Determination of Acceptable Baffle-Barrel-Bolting for Point Beach Units 1 and 2."

4. Westinghouse Engineering Letter NSD-E-MSI-99-036 "Evaluation of Case 132 and 132 no 722 Baffle Bolting Patterns," dated January 28, 1999.

5. WCAP-15029 "Westinghouse Methodology for Evaluating the acceptability of Baffle-Former-Barrel Bolting Distributions Under Faulted Load Conditions," by P.E. Schwirian, et. al., Westinghouse Electric Company, 1998.

6. NRC Safety Evaluation of Topical Report WCAP-15029 "Westinghouse Methodology for Evaluating the Acceptability of Baffle-Former-Barrel Bolting Distribution Under Faulted Load Conditions" (TAC NO. MA 1152)," dated November 10, 1998.

7. Westinghouse Calculation Note CN-RIDA-08-37, Rev. 2, "WEP/WIS (Point Beach Units 1 and 2) RPV System LOCA Analysis - EPU Program," November 20, 2008.

8. Westinghouse Calculation Note CN-RIDA-08-73, Rev. 0, "WEP/WIS (Point Beach Units 1 and 2) EPU - Guide Tube Control Rod Insertability," November 24, 2008.

9. K. Takeuchi: "MULTIFLEX 3.0, A FORTRAN-IV Computer Program for Analyzing Thermal-Hydraulic - Structural System Dynamics Advanced Beam Model," WCAP-9735 Rev. 2, WCAP-9736 (Non-proprietary) Rev. 1, February 1998.

Table 14.3.3-1   MULTI-MASS VIBRATIONAL MODEL-DEFINITION OF SYMBOLS

W1 - Core Barrel   K1 - Hold Down Spring
W2 - Lower Package   K2 - Lower Package Major
W3 - Fuel Assemblies Major   K3 - Top Nozzle Springs Major
W4 - Fuel Rods Major   K5 - Top Nozzle Springs Minor
W5 - Fuel Assemblies Minor   K7 - Short Columns
W6 - Fuel Rods Minor   K8 - Upper Core Plate
W7 - Core Plate & Short Column   K9 - Long Columns
W8 - Deep Beam   K10 - Top Plate
W9 - Core Plate & Long Columns   K11 - Core Barrel
W10 - Top Plate (Ctr.)
W11 - Core Barrel

    Snubbers   Impact Dampers

S1 - Core Barrel Flange   D1 - Barrel Flange
S2 - Hold Down Spring   D2 - Hold Down Spring
S3 - Top Nozzles Bars, Major   D3 - Top Nozzle Bars, Major
S4 - Pedestal Bars, Major   D4 - Pedestal Bars, Major
S5 - Top Nozzles Bars, Minor   D5 - Top Nozzle Bars, Minor
S6 - Pedestal Bars, Minor   D6 - Pedestal Bars, Minor
S7 - Top Nozzle Bumpers, Major   D7 - Top Nozzles, Major
S8 - Top Nozzle Bumpers, Minor   D8 - Top Nozzles, Minor
S9 - Pedestals, Major   D9 - Pedestal, Major
S10 - Pedestals, Minor   D10 - Pedestal, Minor
S11 - Deep Beam Flange   D11 - Deep Beam Flange

    Structural Dampers   Clearances

C1 - Hold Down Springs   G1 - Hold Down Spring
C2 - Lower Package   G3 - Fuel Rod Top, Major
C3 - Top Nozzle, Major   G4 - Fuel Rod Bottom, Major
C5 - Top Nozzle, Minor   G5 - Fuel Rod Top, Minor
C7 - Short Columns   G6 - Fuel Rod Bottom, Minor
C8 - Upper Core Plate   G7 - Fuel Assembly Major
C9 - Long Columns   G8 - Fuel Assembly Minor
C10 - Top Plate
C11 - Core Barrel

    Preloads

P1 - Hold Down Spring
P3 - Top Nozzle Springs Major
P5 - Top Nozzle Springs Minor

Figure 14.3.3-1 REACTOR VESSEL INTERNALS
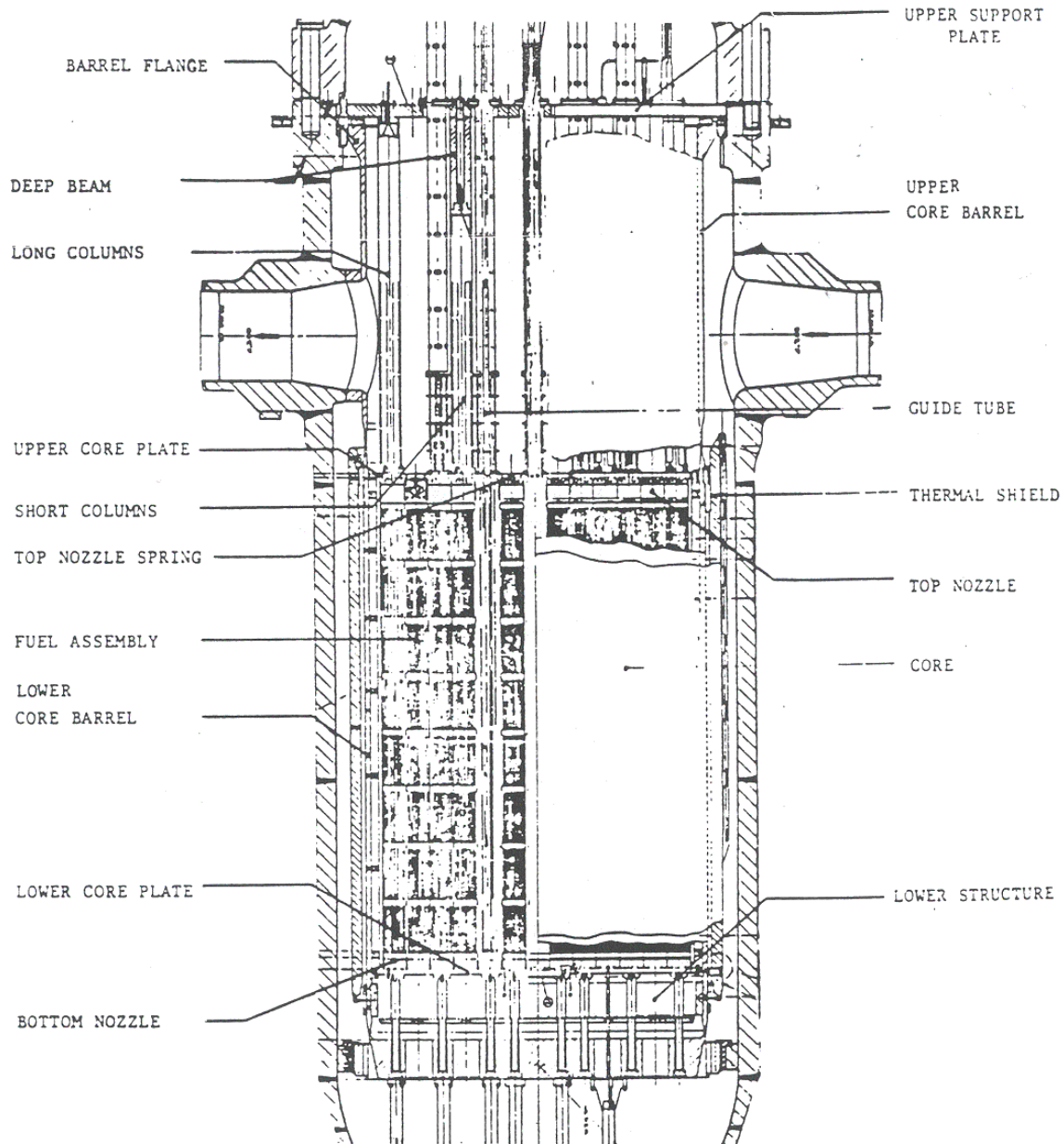
REACTOR VESSEL INTERNALS

Figure 14.3.3-1

Figure 14.3.3-2 MULTI-MASS VIBRATIONAL MODEL
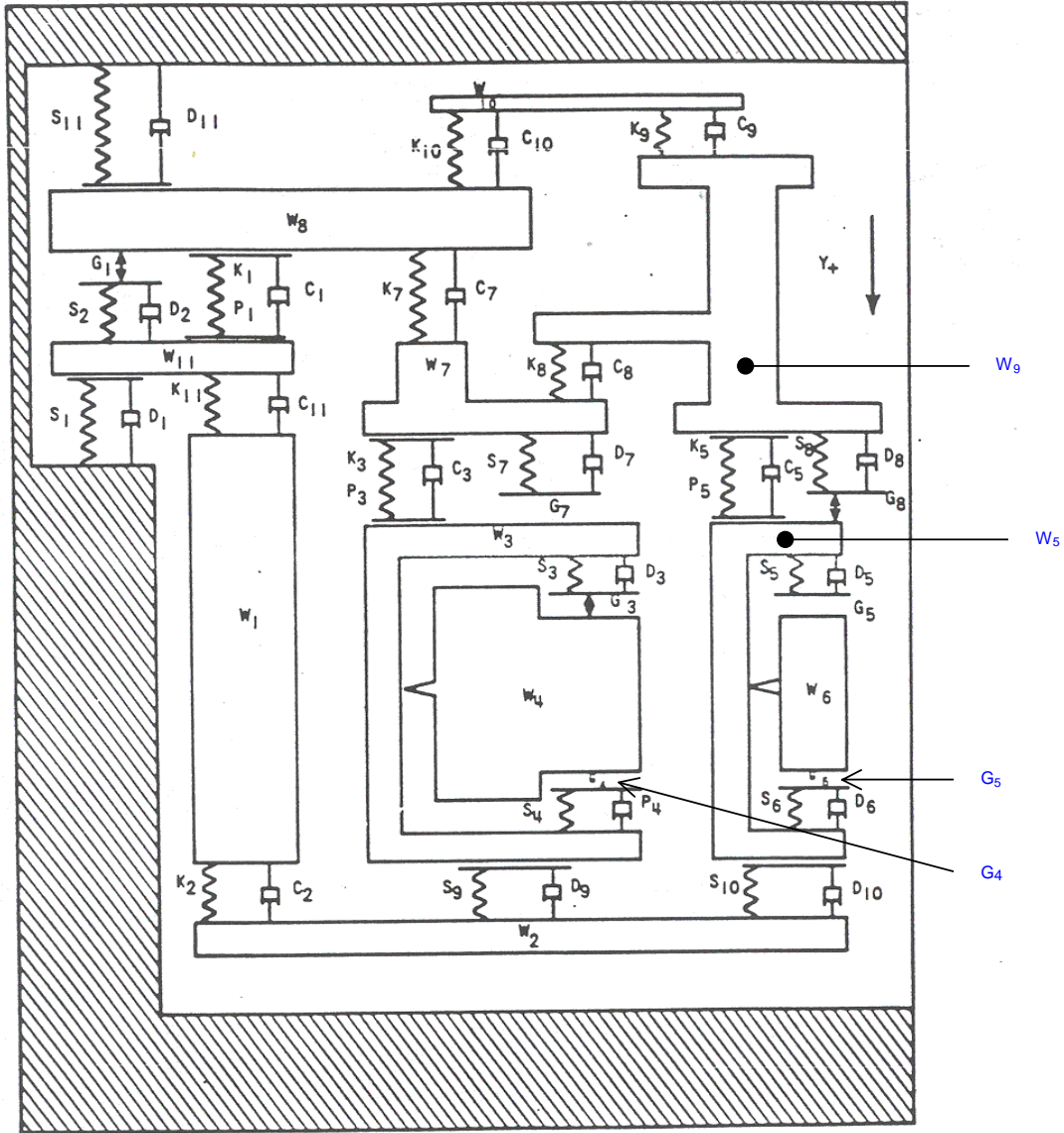
Multi-Mass Vibrational Model

Figure 14.3.3-2

14.3.4   LOCA M&E Release and Containment Response

14.3.4.1   Loss-of Coolant (LOCA) Mass and Energy Releases

The uncontrolled release of pressurized high-temperature reactor coolant, termed a loss-of-coolant accident (LOCA), will result in release of steam and water into the containment. This, in turn, will result in increases in the local subcompartment pressures, and an increase in the global containment pressure and temperature. Therefore, there are both long and short-term issues reviewed relative to a postulated LOCA that must be considered at the operating conditions for the Point Beach Units 1 and 2 EPU Program at a core power of 1,800 MWt (without uncertainty).

The long-term LOCA mass and energy (M&E) releases are analyzed and used as input to the containment integrity analysis [note that only the mass and energy releases up to 3,600 seconds will be utilized by GOTHIC for the containment integrity analysis (see Section 14.3.4.2)]. This demonstrates the acceptability of the containment safeguards systems to mitigate the consequences of a hypothetical large-break LOCA (LBLOCA). The containment safeguards systems must be capable of limiting the peak containment pressure to less than the design pressure and limiting the temperature excursion to less than the acceptance limits. For this program, Westinghouse generated the M&E releases using the March 1979 model, described in WCAP-10325-P-A (Reference 1) and associated support review documents (Reference 5 and Reference 7). The Nuclear Regulatory Commission (NRC) review and approval letter is included with WCAP-10325-P-A (Reference 1). Section 14.3.4.1.1 discusses the long-term LOCA M&E releases generated for this program. The results of this analysis were used in the containment integrity analysis.

The short-term LOCA-related M&E releases are used as input to the subcompartment analyses. These analyses are performed to ensure that the walls of a subcompartment can maintain their structural integrity during the short pressure pulse (generally less than 3 seconds) accompanying a high-energy line pipe rupture within that subcompartment. The subcompartments that are typically evaluated include the SG compartment, the reactor cavity region, and the pressurizer compartment. Point Beach Units 1 and 2 are approved for leak-before-break (LBB) (see Section 14.3.4.1.2) such that the only breaks that need to be evaluated are a 6 inch double-ended hot leg break and a 3 inch double-ended cold leg break. Any changes associated with the power uprate are typically offset by the LBB benefit of using the smaller Reactor Coolant System (RCS) nozzle breaks. The critical mass flux correlation utilized in the SATAN computer program (Reference 2) was used to conservatively estimate the impact of the changes in RCS temperatures on the short-term releases. The evaluation showed that the decrease in mass and energy releases associated with the smaller breaks more than offsets the potential penalties associated with increased releases associated with the EPU. Section 14.3.4.1.2 discusses the short-term evaluation conducted for this program.

14.3.4.1.1   Long-Term LOCA Mass and Energy Releases

The mass and energy release rates described in this section form the basis of further computations to evaluate the containment response (containment integrity peak pressure and the long-term containment temperature calculations) following the postulated accident to ensure that containment design margin is maintained. Discussed in this section are the long-term LOCA mass and energy releases for the hypothetical double-ended pump suction DEPS) rupture with minimum

and maximum safeguards and the double-ended hot leg (DEHL) rupture break case mass and energy release which is limiting for the blowdown portion of the LOCA transient. These LOCA cases are used for the long-term containment integrity analyses in subsection 14.3.4.2 (Long-Term LOCA Containment Response).

14.3.4.1.1.1   <u>Input Parameters and Assumptions</u>

The mass and energy release analysis is sensitive to the characteristics of various plant systems, in addition to other key modeling assumptions. Where appropriate, bounding inputs are utilized and instrumentation uncertainties are included. For example, the RCS operating temperatures are chosen to bound the highest average coolant temperature range of all operating cases and a temperature uncertainty allowance of +6.4°F is then added. Nominal parameters are used in certain instances. For example, the RCS pressure in this analysis is based on a nominal value of 2,250 psia plus an uncertainty allowance (a conservatively high uncertainty of +50.0 psi was used).

All input parameters are chosen consistent with accepted analysis methodology. Some of the most critical items are the RCS initial conditions, core decay heat, safety injection (SI) flow, and primary and secondary metal mass and steam generator heat release modeling. Specific assumptions concerning each of these items are discussed in the following paragraphs. Table 14.3.4-1 through Table 14.3.4-4 present key data used in the analysis.

The core rated power of 1,811 MWt adjusted for calorimetric error (that is, 100.6% of 1,800 MWt) was used in the analysis. As previously noted, RCS operating temperatures were used to bound the highest average coolant temperature range as bounding analysis conditions. The use of higher temperatures is conservative because the initial fluid energy is based on coolant temperatures that are at the maximum levels attained in steady-state operation. Additionally, an allowance to account for instrument error and deadband is reflected in the initial RCS temperatures. The selection of 2,300 psia (2,250 psia nominal value + 50 psi uncertainty allowance) as the limiting pressure is considered to impact the blowdown phase results only, since this represents the initial pressure of the RCS. The RCS rapidly depressurizes from this value until the point at which it equilibrates with containment pressure.

The rate at which the RCS blows down is initially more severe at the higher RCS pressure. Additionally the RCS has a higher fluid density at the higher pressure (assuming a constant temperature) and subsequently has a higher RCS mass available for releases. Thus, 2,250 psia plus uncertainty was selected for the initial pressure as the limiting case for the long-term mass and energy release calculations.

Core stored energy is the amount of energy in the fuel rods above the local coolant temperature. The selection of the fuel design features for the long-term mass and energy release calculation are based on the need to conservatively maximize the energy stored in the fuel at the beginning of the postulated accident. The following fuel features are considered; 1) rod geometry, 2) rod power, and 3) limiting time in life (e.g., burn-up). Uncertainty is included in the core stored energy value to conservatively address the thermal fuel model, considering uncertainties, margin, and fuel densification. Core stored energy is addressed in the analysis as full power seconds (FPS). The core-stored energy that was selected for the Point Beach analysis was 5.25 FPS.

A 3-percent margin in the RCS volume (of which is composed of 1.6-percent allowance for thermal expansion and 1.4-percent allowance for uncertainty) was modeled. This assumption maximizes the initial RCS mass and energy fluid inventory.

A uniform steam generator tube plugging level of zero-percent was modeled. This assumption maximizes the reactor coolant volume and fluid release by virtue of consideration of the RCS fluid in all steam generator tubes. During the post-blowdown period, the steam generators are active heat sources, since significant energy remains in the secondary metal and secondary mass that has the potential to be transferred to the primary side. The zero-percent tube plugging assumption maximizes the heat transfer area and therefore, the transfer of secondary heat across the steam generator tubes. Additionally, this assumption reduces the reactor coolant loop resistance, which reduces the pressure drop (i.e., ΔP) upstream of the break for the pump suction breaks and increases break flow. Thus, the analysis very conservatively accounts for the effects related to steam generator tube plugging.

Secondary-to-primary heat transfer is maximized by assuming conservative coefficients of heat transfer (i.e., steam generator primary-to-secondary heat transfer and reactor coolant system metal heat transfer). Maximum secondary-to-primary heat transfer is ensured by maximizing the initial steam generator inventory based upon 100% power conditions, nominal level plus level uncertainty, and then increasing this by 10% to maximize the available energy. The 10% uncertainty is part of the licensed methodology in Reference 1.

Following a large-break LOCA blowdown inside containment, the safety injection system (SIS) actuates to reflood the RCS. Regarding safety injection flow, the mass and energy release analysis considered configurations, component failures, and offsite power assumptions to conservatively bound respective alignments. The first phase of the SIS operation is the passive accumulator injection. Two accumulators are assumed available to inject. In the LOCA mass and energy re-lease analysis, when the RCS depressurizes below 834.7 psia [maximum accumulator gas cover pressure of 800 psig (814.7 psia) plus an allowance (adjustment) for pressure uncertainty of 20 psi], the accumulators begin to inject. The accumulator injection temperature was conservatively modeled high at 120°F. Relative to the active pumped emergency core cooling system (SI flow), the mass and energy release calculation considered configurations/failures, and offsite power assumptions to conservatively bound respective alignments. The cases include a minimum safeguards case [one high-head SI (HHSI) pump, and one low-head SI (LHSI) pump, see Table 14.3.4-2, and a maximum safeguards case (two HHSI and two LHSI pumps, see Table 14.3.4-3). In addition, the containment backpressure is assumed to be equal to the containment design pressure. This assumption was shown in WCAP-10325-P-A (Reference 1) to be conservative for the generation of mass and energy releases.

In summary, the following assumptions were employed to ensure that the mass and energy releases are conservatively calculated, thereby maximizing energy release to containment:

- Maximum expected operating temperature of the RCS (100-percent full-power operation)
- Allowance for RCS temperature uncertainty (+6.4°F)
- Analyzed Core power of 1,811 MWt
- Allowance for calorimetric error (0.6 percent of power)

- Conservative heat transfer coefficients (that is, steam generator primary/secondary heat transfer and RCS metal heat transfer)
- Allowance in core-stored energy for effect of fuel densification
- An allowance for RCS initial pressure uncertainty (+50psi)
- A total uncertainty for fuel temperature calculation based on a statistical combination of effects and dependent upon fuel type, power level, and burnup
- A maximum containment backpressure equal to the design pressure (74.7 psia)
- Steam Generator Tube Plugging (SGTP) level (0 percent uniform)
  - Maximizes reactor coolant volume and fluid release
  - Maximizes heat transfer area across the steam generator tubes
  - Reduces reactor coolant loop resistance, which reduces the ΔP upstream of the break for the pump suction breaks and increases break flow

Therefore, based on the previously discussed conditions and assumptions, an analysis for Point Beach Units 1 and 2 was performed for the release of mass and energy from the RCS in the event of a large break LOCA at 1,811 MWt core power.

Decay Heat Model

The American Nuclear Society (ANS) Standard 5.1 (Reference 4) was used in the LOCA mass and energy release model for the determination of decay heat energy. This standard was balloted by the Nuclear Power Plant Standards Committee (NUPPSCO) in October 1978 and subsequently approved. The official standard (Reference 4) was issued in August 1979. Table 14.3.4-4 lists the decay heat curve used in the Point Beach Units 1 and 2 mass and energy release analysis.

Significant assumptions in the generation of the decay heat curve for use in the LOCA mass and energy release analysis include the following:

- The decay heat sources considered are fission product decay and heavy element decay of U-239 and Np-239.
- The decay heat power from fissioning isotopes other than U-235 is assumed to be identical to that of U-235.
- The fission rate is constant over the operating history of maximum power level.
- The factor accounting for neutron capture in fission products has been taken from Table 10 of the ANS Standard 5.1 (Reference 4)
- The fuel has been assumed to be at full power for 10^8 seconds.
- The total recoverable energy associated with one fission has been assumed to be 200 MeV/fission.
- An uncertainty of two sigma (two times the standard deviation) has been applied to the fission product decay.

Based upon NRC staff review, (Safety Evaluation Report (SER) issued for the March 1979 evaluation model (Reference 1), use of the ANS Standard 5.1 (Reference 4) decay heat model was approved for the calculation of mass and energy releases to the containment following a LOCA.

Application of Single-Failure Criterion

An analysis of the effects of the single-failure criterion has been performed on the mass and energy release rates for each break analyzed. An inherent assumption in the generation of the mass and energy release is that offsite power is lost. This results in the actuation of the emergency diesel generators, which are required to power the emergency core cooling system (ECCS). Actuation of the emergency diesel generators results in a delay in the time to start both the ECCS and containment safeguards. A delay in the actuation of these accident mitigating components results in a higher containment pressure and temperature for the postulated LOCA. Since the M&E codes (Reference 1) are uncoupled from the containment pressure code (Reference 8, Reference 9, and Reference 12) an assumption on containment pressure is required in the Reference 2 M&E calculations. Maximum containment backpressure equal to the design pressure is modeled, which reduces the rate of safety injection, condensation of steam by the safety injection, and extends the reflood phase, which maximizes the steam release.

Two cases were analyzed to assess the effects of a single failure. The first case assumed minimum safeguards SI flow based on the postulated single failure of an EDG. This assumption results in the loss of one train of safeguards equipment. Therefore, the remaining ECCS was conservatively modeled as: one HHSI pump and one LHSI pump. The second case (maximum safeguards) was modeled as: two HHSI pumps and two LHSI pumps until the RWST water level is drained down to 60% and then only one HHSI pump and one LHSI pump are modeled for the remainder of the injection phase. The single failure assumption postulated is the failure of one containment spray train. Only a single train of recirculation flow was utilized for the maximum safeguards case. Typically a maximum safeguards case would include two trains of safety injection and recirculation flow. Point Beach plant procedures direct operators to shut down one of the two trains and place it in standby when the RWST water level is drained down to 60% through the remainder of the transient. The analysis of the cases described provides confidence that the effect of credible single failures is bounded.

Acceptance Criteria

A large break loss-of-coolant accident is classified as an ANS Condition IV event, an infrequent fault. To satisfy the Nuclear Regulatory Commission acceptance criteria, the relevant requirements are as follows:

1. Point Beach Nuclear Plant (PBNP) FSAR Chapter 1.3 General Design Criteria; as it relates to General Design Criteria 10, 49, and 52, with the respect to containment design integrity and containment heat removal.
2. 10 CFR 50, Appendix K, paragraph I.A: as it relates to sources of energy during the LOCA, provides requirements to assure that all energy sources have been considered.

In order to meet these requirements, the following must be addressed.

1. Source of Energy
2. Break Size and Location
3. Calculation of Each Phase of the Accident
4. Single Failure Criteria

The mass and energy release flowrate table and related analysis information for Point Beach Units 1 and 2 are presented in Table 14.3.4-5 through Table 14.3.4-21.

14.3.4.1.1.2   Description of Analysis

This report section presents the long-term LOCA mass and energy releases generated for Point Beach Units 1 and 2 at 1811 MWt core power. The evaluation model used for the long-term LOCA mass and energy release calculations is the March 1979 model described in Reference 1. This evaluation model has been reviewed and approved generically by the NRC. The approval letter is included with Reference 1. This LOCA mass and energy release methodology has been utilized and approved on the plant-specific dockets for other Westinghouse PWRs such as Catawba Units 1 and 2, Beaver Valley Unit 2, McGuire Units 1 and 2, Millstone Unit 3, Sequoyah Units 1 and 2, Surry Units 1 and 2, Indian Point Unit 2, and Indian Point Unit 3.

A description of the Reference 1 methodology is provided below.

14.3.4.1.1.3 LOCA Mass and Energy Release Phases

The containment system receives mass and energy releases following a postulated rupture in the RCS. These releases continue over a time period, which, for the LOCA mass and energy analysis, is typically divided into four phases:

- Blowdown~the period of time from accident initiation (when the reactor is at steady-state operation) to the time that the RCS and containment reach an equilibrium state.
- Refill - the period of time when the lower plenum is being filled by accumulator and emergency core cooling system (ECCS) water. At the end of blowdown, a large amount of water remains in the cold legs, downcorner, and lower plenum. To conservatively consider the refill period for the purpose of containment mass and energy releases, it is assumed that this water is instantaneously transferred to the lower plenum along with sufficient accumulator water to completely fill the lower plenum. This allows an uninterrupted release of mass and energy to containment. Thus, the refill period is conservatively neglected in the mass and energy release calculation.
- Reflood-begins when the water from the lower plenum enters the core and ends when the core is completely quenched.
- Post-reflood describes the period following the reflood phase. For the pump suction break, a two-phase mixture exits the core, passes through the hot legs, and superheated in the steam generators prior to exiting the break as steam. After the broken loop steam generator cools, the break flow becomes two phase.

14.3.4.1.1.4 Computer Codes

The Reference 1 mass and energy release evaluation model is comprised of mass and energy release versions of the following codes: SATAN-VI, WREFLOOD, FROTH, and EPITOME. These codes were used to calculate the long-term LOCA mass and energy releases for Point Beach.

SATAN-VI calculates blowdown, the first portion of the thermal-hydraulic transient for the RCS following break initiation, including pressure, enthalpy, density, mass and energy flowrates, and energy transfer between primary and secondary systems as a function of time.

The WREFLOOD code addresses the portion of the LOCA transient during the core reflood phase.

FROTH models the post-reflood portion of the transient. The FROTH code is used for the steam generator heat addition calculation from the broken and intact loop steam generators.

EPITOME continues the FROTH post-reflood portion of the transient from the time at which the secondary equilibrates to containment design pressure to the end of the transient.

14.3.4.1.1.5 Break Size and Location

Generic studies (Reference 2) have been performed to determine the effect of postulated break size on the LOCA mass and energy releases. The double-ended guillotine break has been found to be limiting due to larger mass flow rates during the blowdown phase of the transient. During the reflood and froth phases, the break size has little effect on the releases.

Three distinct locations in the reactor coolant system can be postulated for a pipe rupture for mass and energy release purposes:

- Hot leg (between vessel and steam generator)
- Cold leg (between pump and vessel)
- Pump suction (between steam generator and pump)

The break locations analyzed are the double-ended pump-suction (DEPS) rupture (10.48 ft^2) and the double-ended hot-leg (DEHL) rupture (9.17 ft^2). Break mass and energy releases have been calculated for blowdown, reflood, and post-reflood phases of the LOCA for the DEPS cases. For the DEHL case, the releases were calculated only for the blowdown. The following information provides a discussion on each break location.

The DEHL rupture has been shown in previous studies (Reference 1) to yield the highest blowdown mass and energy release rates. Although the core flooding rate would be the highest for this break location, the amount of energy released from the steam generator secondary is minimal because the bulk of the fluid that exits the core vents directly to containment bypassing the steam generators. As a result, the reflood mass and energy releases are reduced significantly as compared to either the pump-suction or cold-leg break locations where the core-exit mixture must pass through the steam generators before venting through the break. For the hot-leg break, generic studies (Reference 1, Section 3.3) have confirmed that there is no reflood peak, that is, from the end of the blowdown period, the containment pressure would continually decrease (Reference 1, Section 3.3). In addition, since none of the powered safety systems are assumed to be operational during the initial blowdown phase, the service water system has no impact on the DEHL break. Therefore, only the mass and energy releases for the hot-leg break (blowdown phase) are calculated and presented in this section of the report and no further evaluation is necessary.

The cold-leg break location has also been found in previous studies to be much less limiting in terms of the overall containment energy releases (Reference 1, Section 3.3). The cold-leg blowdown is faster than that of the pump-suction break, and more mass is released into the containment. However, the core heat transfer is greatly reduced, and this results in a considerably lower energy release into containment. The blowdown transient for the cold-leg is, in general less

limiting than that for the pump suction break. During the reflood phase, the flooding rate is greatly reduced and the energy release rate into the containment is reduced. Therefore, the cold-leg break is bounded by other breaks and no further evaluation is necessary.

The pump-suction break combines the effects of the relatively high core flooding rate, as in the hot-leg break, and the addition of the stored energy in the steam generators. As a result, the pump-suction break yields the highest energy flow rates during the post-blowdown period by including all of the available energy of the RCS in calculating the releases to containment.

Therefore, only DEHL and DEPS case are analyzed for long-term LOCA containment integrity. LOCA mass and energy releases have been calculated for the blowdown, reflood and post-reflood phases for the DEPS cases. For the DEHL case, the releases were calculated only for the blowdown phase with this methodology.

14.3.4.1.1.6   Blowdown Mass and Energy Release Data

The SATAN-VI code is used for computing the blowdown transient. The code utilizes the control volume (element or nodal) approach with the capability for modeling a large variety of thermal fluid system configurations. The fluid properties are considered uniform and thermodynamic equilibrium is assumed in each element. A point kinetics model is used with weighted feedback effects. The major feedback effects include moderator density, moderator temperature, and Doppler broadening. A critical flow calculation for subcooled (modified Zaloudek), two-phase (Moody), or superheated break flow is incorporated into the analysis. The methodology for the use of this model is described in WCAP-10325-P-A (Reference 1).

Table 14.3.4-5 presents the calculated mass and energy release for the blowdown phase of the DEHL break. For the hot-leg break mass and energy release tables, break path 1 refers to the mass and energy exiting from the reactor vessel side of the break. Break path 2 refers to the mass and energy exiting from the steam generator side of the break. Table 14.3.4-6 and Table 14.3.4-7 present the mass and energy balance data for the DEHL case.

Table 14.3.4-8 presents the calculated mass and energy releases for the blowdown phase of the DEPS break location (applicable for both minimum and maximum safeguards break cases). For the pump-suction breaks, break path 1 in the mass and energy release tables refers to the mass and energy exiting from the steam generator side of the break. Break path 2 refers to the mass and energy exiting from the pump side of the break.

14.3.4.1.1.7   Reflood Mass and Energy Release Data

The WREFLOOD code is used for computing the reflood transient. The WREFLOOD code consists of two basic hydraulic models: one for the contents of the reactor vessel and one for the coolant loops. The two models are coupled through the interchange of the boundary conditions applied at the vessel outlet nozzles and at the top of the downcomer. Additional transient phenomena, such as pumped safety injection and accumulators, reactor coolant pump performance, and steam generator releases are included as auxiliary equations that interact with the basic models as required. The WREFLOOD code permits the capability to calculate variations during the core reflooding transient of basic parameters such as core flooding rate, core and

downcomer water levels, fluid thermodynamic conditions (pressure, enthalpy, density) throughout the primary system, and mass flow rates through the primary system. The code permits hydraulic modeling of the two flow paths available for discharging steam and entrained water from the core to the break, that is, the path through the broken loop and the path through the unbroken loops.

A complete thermal equilibrium mixing condition for the steam and ECCS injection water during the reflood phase has been assumed for each loop receiving ECCS water. This is consistent with the usage and application of the WCAP-10325-P-A (Reference 1) mass and energy release evaluation model in recent analyses, for example, D. C. Cook Unit 1 Docket (Reference 5). Even though the WCAP-10325-P-A (Reference 1) model credits steam/water mixing only in the intact loop and not in the broken loop, the justification, applicability, and NRC approval for using the mixing model in the broken loop has been documented (Reference 5). Moreover, this assumption is supported by test data and is further discussed below.

The model assumes a complete mixing condition (that is, thermal equilibrium) for the steam/water interaction. The complete mixing process, however, is made up of two distinct physical processes. The first is a two-phase interaction with condensation of steam by cold ECCS water. The second is a single-phase mixing of condensate and ECCS water. Since the steam release is the most important influence to the containment pressure transient, the steam condensation part of the mixing process is the only part that must be considered. Any spillage directly heats only the sump.

The most applicable steam/water mixing test data have been reviewed for validation of the containment integrity reflood steam/water mixing model. This data was generated in 1/3-scale tests (Reference 6), which are the largest scale data available. Therefore, the test data most clearly simulates the flow regimes and gravitational effects that would occur in a PWR. These tests were designed specifically to study the steam/water interaction for PWR reflood conditions.

A group of 1/3-scale steam/water mixing tests discussed in Reference 6 corresponds directly to containment integrity reflood conditions. The injection flow rates for this group cover all phases and mixing conditions calculated during the reflood transient. The data from these tests were reviewed and discussed in detail in WCAP-10325-P-A (Reference 1). For all of these tests, the data clearly indicate the occurrence of very effective mixing with rapid steam condensation. The mixing model used in the containment integrity reflood calculation is, therefore, wholly supported by the 1/3-scale steam/water mixing data.

Additionally, the following justification is also noted. The post-blowdown limiting break for the containment integrity peak pressure analysis is the pump-suction double-ended rupture break. For this break, there are two flow paths available in the RCS by which mass and energy may be released to containment. One is through the outlet of the steam generator, the other via reverse flow through the RCP. Steam that is not condensed by ECCS injection in the intact RCS loops passes around the downcomer and through the broken-loop cold-leg and pump in venting to containment. This steam also encounters ECCS injection water as it passes through the broken-loop cold-leg, complete mixing occurs and a portion of it is condensed. It is this portion of steam that is condensed that is taken credit for in this analysis. This assumption is justified based upon the postulated break location, and the actual physical presence of the ECCS injection nozzle. A description of the test and test results are contained in WCAP-10325-P-A (Reference 1) and the operating license Amendment No. 126 for D. C. Cook (Reference 5).

Table 14.3.4-9 presents the calculated mass and energy releases for the reflood phase of the double-ended pump-suction rupture with minimum safeguards (i.e., the EDG failure) case. Table 14.3.4-10 presents the reflood phase mass and energy release for the maximum safeguards (i.e., failure of a containment spray pump) case.

The transient responses of the principal parameters during reflood are given in Table 14.3.4-11 for the DEPS minimum safeguards case and Table 14.3.4-12 for the DEPS maximum safeguards case.

14.3.4.1.1.8   Post-Reflood Mass and Energy Release Data

The FROTH code (Reference 1 and Reference 2) is used for computing the post-reflood transient. The FROTH code calculates the heat release rates resulting from a two-phase mixture present in the steam generator tubes. The mass and energy releases that occur during this phase are typically superheated (Reference 7) due to the depressurization and equilibration of the broken loop and intact loop steam generators. During this phase of the transient, the RCS has equilibrated with the containment pressure. However, the steam generators contain a secondary inventory at an enthalpy that is much higher than the primary side. Therefore, there is a significant amount of reverse heat transfer that occurs. Steam is produced in the core due to core decay heat. For a pump-suction break, a two-phase fluid exits the core, flows through the hot legs, and becomes superheated as it passes through the steam generator. Once the broken loop cools, the break flow becomes two-phase. During the FROTH calculation, ECCS injection is addressed for both the injection phase and the recirculation phase. The FROTH code calculation stops when the secondary side equilibrates to the saturation temperature (T_{sat}) at the containment design pressure. After this point, the EPITOME code completes the steam generator depressurization.

The methodology for the use of this model is described in WCAP-10325-P-A (Reference 1). The mass and energy release rates are calculated by the FROTH and EPITOME computer codes until the time of containment depressurization. After containment depressurization (14.7 psia), the mass and energy release available to containment is generated directly from core boil-off/decay heat.

Table 14.3.4-13 presents the two-phase post-reflood mass and energy release data for the minimum safeguards pump suction double-ended break case. Table 14.3.4-14 presents the two-phase post-reflood mass and energy release data for the double-ended pump suction maximum safeguards case.

14.3.4.1.1.9   Post-Reflood Mass and Energy Release Data-Steam Generator Equilibration and Depressurization

Steam generator equilibration and depressurization is the process by which secondary-side energy is removed from the steam generators in stages. The FROTH computer code calculates the heat removal from the secondary mass until the secondary temperature is the saturation temperature (T_{sat}) at the containment design pressure. After the FROTH calculations, the EPITOME code continues the FROTH calculation for steam generator cooldown removing steam generator secondary energy at different rates (that is, first- and second-stage rates). The first-stage rate is applied until the steam generator reaches the saturation temperature (T_{sat}) at the user specified intermediate equilibration pressure, when the secondary pressure is assumed to reach the actual

containment pressure. Then the second-stage rate is used until the final depressurization, when the secondary reaches the reference temperature of T_{sat} at 14.7 psia, or 212°F. The heat removal of the broken loop and intact loop steam generators are calculated separately.

During the FROTH calculations, steam generator heat removal rates are calculated using the secondary side temperature, primary side temperature and a secondary side heat transfer coefficient determined using a modified McAdam's correlation. Steam generator energy is removed during the FROTH transient until the secondary side temperature reaches saturation temperature (Tsat) at the containment design pressure. The constant heat removal rate used during the first heat removal stage is based on the final heat removal rate calculated by FROTH. The steam generator energy available to be released during the first stage interval is determined by calculating the difference in secondary energy available at the containment design pressure and that at the (lower) user-specified intermediate equilibration pressure, assuming saturated conditions. The intermediate equilibrium pressures are chosen as discussed in Reference 1, Section 2.3 and 3.3. This energy is then divided by the first stage energy removal rate, resulting in an intermediate equilibration time. At this time, the rate of energy release drops substantially to the second stage rate. The second stage rate is determined as the fraction of the difference in secondary energy available between the intermediate equilibration and final depressurization at 212°F, and the time difference from the time of the intermediate equilibration to the user-specified time of the final depressurization at 212°F. With this methodology, all of the secondary energy remaining after the intermediate equilibration is conservatively assumed to be released by imposing a mandatory cooldown and subsequent depressurization down to atmospheric pressure at 3600 seconds; that is, 14.7 psia and at 212°F (the mass and energy balance tables have this point labeled as "Available Energy").

14.3.4.1.1.10   Post One-Hour mass and Energy Releases

The long-term post-one hour mass and energy releases (boil-off from core at the decay heating rate) are performed through user defined input functions in the GOTHIC code (Reference 8). This method of determining the long-term mass and energy releases is consistent with past application of Westinghouse methodology. See subsection 14.3.4.2.3 Boundary Conditions LOCA Mass and Energy Release for discussion of long-term mass and energy calculations.

14.3.4.1.1.11   Sources of Mass and Energy

The sources of mass considered in the LOCA mass and energy releases are given in Table 14.3.4-6 for the DEHL break case. The sources of mass for the DEPS break cases, i.e., minimum and maximum ECCS, respectively, are given in Table 14.3.4-15 and Table 14.3.4-16. These sources are:

- The RCS water
- Accumulator water (two accumulators injecting)
- Pumped injection water (SI)

The energy inventories considered in the LOCA mass and energy release analysis are presented in Table 14.3.4-7, Table 14.3.4-17, and Table 14.3.4-18 (for the DEHL and DEPS minimum and maximum safeguards cases, respectively). The energy sources are as follows:

- RCS water
- Accumulator water (two accumulators injecting)
- Pumped SI water
- Decay heat
- Core-stored energy
- RCS metal (includes the reactor vessel and internals, hot and cold leg piping, steam generator inlet and outlet plenums, and steam generator tubes)
- Steam generator metal (includes transition cone, shell, wrapper, and other internals)
- Steam generator secondary energy (includes fluid mass and steam mass)
- Secondary transfer of energy (feedwater into and steam out of the steam generator secondary)

The analysis used the following energy reference points:

Available energy: 212°F; 14.7 psia [energy available that could be released]
Total energy content: 32°F; 14.7 psia [total internal energy of the RCS]

The mass and energy inventories are presented at the following times, as appropriate:

- Time zero (initial conditions)
- End of blowdown time
- End of refill time
- End of reflood time
- Time of broken loop steam generator equilibration to pressure setpoint
- Time of intact loop steam generator equilibration to pressure setpoint
- Time of full depressurization (3600 seconds)

The energy release from the zirc-water reaction is considered as part of the WCAP-10325-P-A (Reference 1) methodology. Based on the way that the energy in the fuel is conservatively released to the vessel fluid, the fuel cladding temperature does not increase to the point where the metal-water reaction is significant. For the LOCA mass and energy calculation, the energy created by the metal-water reaction value is small and is not explicitly provided in the energy balance tables. The energy that is determined is part of the mass and energy releases, and is therefore already included in the LOCA mass and energy release.

The sequence of events for the LOCA transients is shown in Table 14.3.4-19, Table 14.3.4-20, and Table 14.3.4-21 (for the DEHL and DEPS minimum and maximum safeguards cases, respectively).

14.3.4.1.1.12   Conclusions

The consideration of the various energy sources in the long-term mass and energy release analysis provides assurance that all available sources of energy have been included in this analysis. Thus, the review guidelines presented in SRP Section 6.2.1.3 (Reference 3) have been satisfied. The results of this analysis were available for use in the containment integrity analysis in a subsection of Section 14.3.4.2, Long-Term LOCA Containment Response.

14.3.4.1.2 Short-Term LOCA Mass and Energy Releases

An evaluation was performed to determine the effect of the Point Beach EPU program on the short-term LOCA-related M&E releases.

14.3.4.1.2.1 Accident Description

The short-term LOCA-related M&E releases are used as input to the subcompartment analyses. These analyses are performed to ensure that the walls of a subcompartment can maintain their structural integrity during the short pressure pulse (generally less than 3 seconds) accompanying a high-energy line pipe rupture within that subcompartment. The subcompartments that are typically evaluated include the SG compartment, the reactor cavity region, and the pressurizer compartment.

The magnitude of the pressure differential across the walls is a function of several parameters, which include the blowdown M&E release rates, the subcompartment volume, vent areas, and vent flow behavior. The blowdown M&E release rates are affected by the initial RCS temperature conditions.

Point Beach Units 1 and 2 were initially approved for Leak-Before-Break (LBB) via Reference 13. The pressurizer surge line was also eliminated from the structural design basis in Reference 14. Any changes associated with a major plant modification, such as a steam generator replacement or a power uprating, are typically offset by the LBB benefit of using the smaller RCS nozzle breaks. This demonstrates that the current licensing bases for the subcompartments would remain bounding for breaks postulated in the large, primary loop piping. All breaks larger than a 6 inch double-ended hot leg break and a 3 inch double-ended cold leg break have been eliminated by LBB. These specific breaks must be evaluated at the EPU conditions.

The critical mass flux correlation utilized in the SATAN-VI computer program (Reference 2) can be used to conservatively estimate the impact of the changes in RCS temperatures on the short-term releases. The following sections discuss the short-term evaluation conducted for this program.

14.3.4.1.2.2 Input Parameters and Assumptions

The short-term releases are linked directly to the critical mass flux, which increases with decreasing temperatures. The increase in mass flux is created by an increase in the differential pressure between the reservoir pressure and the saturation pressure at the RCS operating conditions. The critical mass flux is the maximum break flow per cross-sectional flow area based on a reservoir pressure and saturation temperature. The short-term LOCA releases would be expected to increase due to any reductions in RCS coolant temperature conditions.

It is noted that any changes in initial RCS inventory and SG liquid/steam mass and volume from the proposed parameters for the Point Beach Units I and 2 EPU Program have no effect on the releases because of the short duration of the postulated accident. The only change that needs to be addressed for this short-term LOCA M&E evaluation is the impact of the Point Beach Units 1 and 2 EPU Program on the RCS coolant temperatures.

Short-term releases are controlled by local pressures and temperatures, so the lower temperatures from the Point Beach Units 1 and 2 EPU Program operating conditions are more limiting. The hot leg temperature, cold leg temperature, RCS pressure, and system uncertainties for the EPU program can be found in Table 14.3.4-22.

14.3.4.1.2.3 Results

The short-term LOCA-related analyses for Point Beach Units 1 and 2 have been reviewed to assess the effects associated with the EPU Program. Based on the application of LBB methods,

the only breaks that need to be evaluated are a 6 inch double-ended hot leg break and a 3 inch double-ended cold leg break. The decrease in mass and energy releases associated with the smaller breaks more than offsets the potential penalties associated with increased releases associated with the EPU. Additionally, releases have been provided for the smaller breaks at the EPU conditions in Table 14.3.4-23.

14.3.4.2   Long-Term LOCA Containment Response

The purpose of the LOCA containment integrity analysis performed for Point Beach Units 1 and 2 at 1811 MWt core power (includes uncertainty) is to analyze the bounding peak pressure and temperature of a design basis LOCA event inside containment and to demonstrate the ability of the containment heat removal system to mitigate the accident. The impacts of LOCA mass and energy releases on the containment pressure and temperature are assessed to ensure that the containment pressure and temperature remain below their respective design limits.

The Point Beach LOCA containment response analysis considered a spectrum of cases as discussed in Section 14.3.4.1.1, Long-Term LOCA Mass and Energy Releases. The cases address break locations, and postulated single failure (minimum and maximum safeguards). The limiting cases that address the containment peak pressure cases are presented here.

Calculation of the containment response following a postulated LOCA was analyzed by use of the digital computer code GOTHIC version 7.2a. The GOTHIC technical manual (Reference 8) provides a description of the governing equations, constitutive models, and solution methods in the solver. The GOTHIC qualifications report (Reference 9) provides a comparison of the solver results with both analytical solutions and experimental data.

The GOTHIC containment modeling for Point Beach is consistent with the recent NRC approved Ginna evaluation model (Reference 11). The latest code version is used to take advantage of the diffusion layer model (DLM) heat transfer option. This heat transfer option was approved by the NRC (Reference 11) for use in Ginna containment analyses with the condition that mist be excluded from what was earlier termed as the mist diffusion layer model (MDLM). The GOTHIC containment modeling for Point Beach has followed the conditions of acceptance placed on Ginna. The Point Beach containment volume is similar because both Point Beach and Ginna are 2-loop plants. The differences in GOTHIC code versions are documented in Appendix A of the GOTHIC User Manual Release Notes (Reference 12). Version 7.2a is used consistent with the restrictions identified in Reference 11; none of the user-controlled enhancements added to Version 7.2a were implemented in the Point Beach containment model.

The Point Beach GOTHIC containment evaluation model for the LOCA events consisted of one volume. Additional boundary conditions, volumes, flow paths, and components are used to model accumulator nitrogen release and sump recirculation. Injection of accumulator nitrogen during a LOCA event is modeled by a boundary condition. The recirculation system model uses GOTHIC component models for the RHR and component cooling water (CCW) heat exchangers and the CCW pumps. Recirculation flow from the sump is modeled using a boundary condition.

14.3.4.2.1   Accident Description

A break in the primary RCS piping causes a loss-of-coolant, which results in a rapid release of mass and energy to the containment atmosphere. Typically, the blowdown phase for the large LOCA event is over in less than 30 seconds. This large and rapid release of high-energy, two-phase fluid causes a rapid increase in the containment pressure, which initiates safety injection and containment spray.

The RCS accumulators begin to refill the lower plenum and downcomer of the reactor vessel with water after the end of blowdown. The reflood phase begins after the vessel fluid level reaches the bottom of the fuel. During this phase, the core is quenched with water from both the accumulators and pumped SI. The quenching process creates a large amount of steam and entrained water that is released to containment through the break. This two-phase mixture would have to pass through the steam generators and also absorb energy from the secondary side coolant if the break were located in the cold-leg or pump-suction piping.

The LOCA mass and energy release decreases with time as the system cools. Core decay heat is removed by nucleate boiling after the reflood phase is complete. The core fluid level is maintained by pumping water back into the vessel by the SI system from either the RWST or the containment sump. The containment heat removal systems continue to condense steam and slowly reduce the containment pressure and temperature over time.

14.3.4.2.2   Input Parameters, Assumptions, and Acceptance Criteria

An analysis of containment response to the rupture of the RCS must start with knowledge of the initial conditions in the containment. The pressure, temperature, and humidity of the containment atmosphere prior to the postulated accident are specified in the analysis as shown in Table 14.3.4-24.

Also, values for the initial temperature of the service water (SW) and refueling water storage tank (RWST) are assumed, along with containment spray (CS) pump flowrate and containment fan cooler (CFC) heat removal performance. All of these values are chosen conservatively, as shown in Table 14.3.4-24. Long term sump recirculation is addressed via Residual Heat Removal System (RHR) heat exchanger performance. The primary function of the RHR system is to remove heat from the core by way of Emergency Core Cooling System (ECCS). Table 14.3.4-24 provides the RHR system parameters assumed in the analysis.

Design Basis Accident

A series of cases was performed for the LOCA containment response. Section 14.3.4.1.1 documented the M&E releases for the minimum and maximum safeguards cases for a DEPS break and the releases from the blowdown of a DEHL break.

For the maximum safeguards DEPS case a failure of a containment spray pump was assumed as the single failure, which leaves available as active heat removal systems: one containment spray pump and four CFCs. Table 14.3.4-25 provides the CFC performance per unit versus containment saturation temperature and Table 14.3.4-26 provides the performance data for one spray pump in operation. Note: For the maximum safeguards case a limiting assumption was made concerning the modeling of the recirculation system, i.e., heat exchangers. The minimum safeguards data was conservatively used to model the RHR heat exchangers, i.e., one RHR heat exchanger (Hx) was credited for residual heat removal. Emergency safeguards equipment data is given in Table 14.3.4-24.)

The minimum safeguards case was based upon a diesel train failure (which leaves available as active heat removal systems one containment spray pump and 2 CFCs). Due to the duration of the DEHL transient (i.e. blowdown only), no containment safeguards equipment is modeled.

The calculations for all of the DEPS cases were performed for 2.6 million seconds (approximately 30 days). The DEHL cases were terminated soon after the end of the blowdown. The sequence of events for each of these cases is shown in Table 14.3.4-19 through Table 14.3.4-21.

Modeling Assumptions

The following are the major assumptions made in the analysis.

- The mass and energy released to the containment are described in Section 14.3.4.1.1 for LOCA.
- Homogeneous mixing is assumed. The steam-air mixture and the water phases each have uniform properties. More specifically, thermal equilibrium between the air and the steam is assumed. However, this does not imply thermal equilibrium between the steam-air mixture and the water phase.
- Air is taken as an ideal gas, while compressed water and steam tables are employed for water and steam thermodynamic properties.
- For the blowdown portion of the LOCA analysis, the discharge flow separates into steam and water phases at the breakpoint. The saturated water phase is at the total containment pressure, while the steam phase is at the partial pressure of the steam in the containment. For the post blowdown portion of the LOCA analysis, steam and water releases are input separately.
- The saturation temperature at the partial pressure of the steam is used for heat transfer to the heat sinks and the fan coolers.
- The containment fan coolers are activated by a containment high pressure SI signal.

Acceptance Criteria

The containment response for design-basis containment integrity is an ANS Condition IV event, an infrequent fault. The relevant requirements to satisfy Nuclear Regulatory Commission acceptance criteria as follows.

- General Design Criteria (GDC) 10 and GDC 49 from the PBNP FSAR Table 1.3-1: In order to satisfy the requirements of GDC 10 and 49, the peak calculated containment pressure should be less than the containment design pressure of 60 psig (74.7 psia)

- PBNP FSAR Table 1.3-1, GDC 52: In order to satisfy the requirements of GDC 52, the calculated pressure at 24 hours should be less than 50% of the peak calculated value. (This is related to the criteria for doses at 24 hours.)

14.3.4.2.3   Description of the LOCA GOTHIC Containment Model

Noding Structure

The Point Beach GOTHIC containment evaluation model for the LOCA events consisted of one volume. Additional boundary conditions, volumes, flow paths, and components are used to model accumulator nitrogen release and sump recirculation. Injection of accumulator nitrogen during a LOCA event is modeled by a boundary condition. The recirculation system model uses GOTHIC component models for the RHR and component cooling water (CCW) heat exchangers and the CCW pumps. Recirculation flow from the sump is modeled using a boundary condition.

Volume Input

GOTHIC requires the volume, height, diameter, and elevation input values for each node. The containment is modeled as a single control volume in the containment model. The minimum free volume of 1,000,000 ft^3 was used.

Initial Conditions

The containment initial conditions for containment integrity cases are:

Pressure: 16.7 psia

Relative humidity: 20%

Temperature: 120°F

Flow Paths

Flow boundary conditions linked to functions that define the M&E releases model the LOCA break flow to the containment. The boundary conditions are connected to the containment control volume via flow paths. The injection spray is modeled as a boundary condition connected to the containment control volume via a flow path.

The flow rates through the flow paths are specified by boundary conditions, so the purpose of the flow path is to direct the flow to the proper control volume. The flow path input is mostly arbitrary. Standard values are used for the area, hydraulic diameter, friction length, and inertia length of the flow path. Since this is a single volume lumped-parameter model, the elevation of the break flow paths is arbitrarily set to 1 foot and the elevation of the spray flow paths is arbitrarily set to 50 feet above the containment floor.

Heat Sinks

The structural heat sinks in the containment are modeled as GOTHIC thermal conductors. The heat sink geometry data are based on conservatively low surface areas and are summarized in Table 14.3.4-27.

A thin air gap is assumed to exist between the steel and concrete for steel-jacketed heat sinks. A gap conductivity of 0.0174 Btu/hr-ft-°F is assumed between steel and concrete.

The thermophysical properties for the heat sink materials are summarized in Table 14.3.4-28.

Heat and Mass Transfer Correlations

GOTHIC has several heat transfer coefficient options that can be used for containment analyses. For the Point Beach GOTHIC model, the direct heat transfer coefficient set is used with the DLM mass transfer correlation for the heat sinks inside containment. This heat transfer methodology was reviewed and approved for use in the Ginna containment design basis accident analyses (Reference 11). The DLM correlation does not require the user to specify a revaporization input value.

The direct heat transfer coefficient set is used for the heat sinks representing floors, ceilings, and walls. The submerged conductors are essentially insulated from the vapor after the pool develops. Insulated surfaces are modeled with no heat loss (0.0 Btu/hr-ft^2-°F).

Containment Fan Coolers

The reactor containment fan coolers (CFCs) are another means of heat removal. Each CFC has a fan which draws in the containment atmosphere from the upper volume of the containment via a return air riser. The steam/air mixture is routed through the enclosed CFC unit, past service water cooling coils. The fan then discharges the air through ducting containing a check damper. The discharged air is directed at the lower containment volume. The CFCs are modeled in GOTHIC as a cooler/heater component in the containment volume. They are initiated by the containment high pressure safety injection signal at 6 psig (20.7 psia), with a time delay of 84 seconds. The heat removal rate for one CFC is defined by a function in GOTHIC. Mulitpliers are used to define the amount of operational CFCs (2 for minimum safeguards, 4 for maximum safeguards). See Table 14.3.4-25 for the CFC heat removal capability assumed for the containment response analyses.

Sump Recirculation

A sump recirculation model consisting of simplified RHR system and CCW system models was included in the Point Beach containment model to calculate the long-term LOCA containment pressure and temperature response. The RHR heat exchanger cools the water from the containment sump. The RHR system injects the cooled water into the RCS to cool the core. The RHR heat exchanger is cooled with CCW water and service water provides the ultimate heat sink cooling the CCW heat exchangers.

Boundary Conditions

LOCA Mass and Energy Release

The LOCA mass and energy release methodology generates the releases from both sides of the break (or two flow paths: M&E exiting from the vessel side of the break; and M&E exiting from the steam generator side of the break - as defined for a double-ended hot leg break). The LOCA transient M&E releases are calculated as separate flow paths (for the first 3,600 seconds) and input to the GOTHIC containment model via flow boundary conditions. The flow boundary conditions are linked to functions that define the mass break flow and the enthalpy of the break flow. The break mass and enthalpy are input to the containment model as external functions defined by control variables. The M&E releases from the boundary conditions are analyzed for Point Beach out to 3,600 seconds; that is, the time at which all energy in the primary heat structures and steam generator secondary system is released/depressurized to atmospheric pressure (14.7 psia and 212°F). The LOCA M&E release rates are generated using the Westinghouse M&E methodology (Reference 1).

During blowdown, the liquid portion of the break flow is released as drops with an assumed diameter of 100 microns (0.00394 inch). This is consistent with the methodology approved for Ginna (Reference 11) and is based on data presented in Reference 10. After blowdown, the liquid release is assumed to be a continuous pour into the sump.

The long-term, post 3,600 second, mass and release (boil-off from the core at the decay heating rate) calculations are performed through user defined functions by GOTHIC. These input functions are used to incorporate the sump water cooling in the long term and are consistent with the Westinghouse methodology previously approved by the NRC. After primary system and secondary system energy have been released (depressurized to atmospheric pressure, (14.7 psia and 212°F), the M&E release to the containment is assumed to be from long-term steaming of decay heat. A flow boundary condition is defined to provide the long-term boil-off M&E release to containment. The mass flow rate and enthalpy of the flow is calculated using GOTHIC control variables.

The American Nuclear Society (ANS) Standard 5.1 (Reference 4) decay heat model (2δ uncertainty) is used to calculate the long-term boil-off from the core. All the decay heat is assumed to produce steam from the recirculated ECCS water. The remainder of the ECCS water is returned to the sump region of the containment control volume. These assumptions are consistent with the long-term M&E methodology documented in Reference 1.

Containment Spray System

Containment spray is modeled with one boundary condition for the injection phase and two coupled boundary conditions for the recirculation phase. Point Beach has two trains of containment safeguards available, with one spray pump per train. An inherent assumption in the LOCA containment analysis is that offsite power is lost with the pipe rupture. Injection sprays are modeled with one operational train for both the minimum and maximum safeguards cases due to this assumption combined with a limiting single failure.

Injection spray actuation is modeled on the containment Hi-Hi pressure setpoint (44.7 psia). The sprays begin injecting 100°F water after a specified 70-second delay. The containment spray flow varies according to containment pressure and can be found in Table 14.3.4-26. The spray flow rate is modeled in GOTHIC as a control variable. The injection spray switches over to recirculation spray after a specified 1,200-second (i.e. 20 minutes) delay. Recirculation spray is modeled to terminate per Table 14.3.4-24.

Accumulator Nitrogen Gas Modeling

The accumulator nitrogen gas release is modeled with a flow boundary condition in the LOCA containment model. The nitrogen release rate was conservatively calculated by maximizing the mass available to be injected. The nitrogen gas release rate was used as input for the GOTHIC function, as a specified rate over a fixed time period. Nitrogen gas is released at a rate of 244.32 lbm/second; beginning at 40.73 seconds (average accumulator tank water volume empty time) and ending at 60.64 seconds.

14.3.4.2.4 LOCA Containment Integrity Results

The containment pressure and steam temperature profiles from each of the LOCA cases are shown in Figure 14.3.4-1 through Figure 14.3.4-6. The results of the DEHL break are shown in Figure 14.3.4-1 and Figure 14.3.4-2. The results of the DEPS break cases are shown in Figure 14.3.4-3 through Figure 14.3.4-6.

LOCA Containment Response Transient Description: Double Ended Hot Leg Break

This analysis assumes a loss-of-offsite power coincident with a double-ended rupture of the RCS piping between the reactor vessel outlet nozzle and the steam generator inlet (i.e., a break in the RCS hot leg).

The postulated RCS break results in a rapid release of mass and energy to the containment with a resulting rapid rise in both the containment pressure and temperature. As the containment pressure rises, the RCS rapidly depressurizes which results in the generation of a compensated pressurizer pressure reactor trip at 0.311 seconds and a low pressurizer pressure SI setpoint at 3.8 seconds. The containment pressure continues to rise rapidly in response to the release of mass and energy until the end of blowdown at 15.4 seconds, with the pressure reaching a value of 70.39 psia at 14.51 seconds. This is the highest peak containment pressure of the three cases analyzed. The highest peak containment temperature of 280.3°F also occurs coincident with the peak pressure. The end of blowdown marks a time when the initial inventory in the RCS has been exhausted and a process of filling the RCS downcomer in preparation for reflood has begun. Since the reflood for a hot leg break is very fast due to the low resistance to steam venting posed by the broken hot leg, the hot leg break mass and energy relies transients are terminated shortly after blowdown. Table 14.3.4-19 provides the transient sequence of events for the DEHL transient.

LOCA Containment Response Transient Description: Double Ended Pump Suction Break with Minimum Safeguards

This analysis assumes a loss-of-offsite power coincidence with a double-ended rupture of the RCS piping between the steam generator outlet and the RCS pump inlet (suction). The associated single failure assumption is the failure of a diesel to start, resulting in one train of ECCS and containment safeguards equipment being available. This combination results in a minimum set of safeguards being available. Further, loss of offsite power delays the actuation times of the safeguards equipment due to the required diesel startup time after receipt of the safety injection signal.

The postulated RCS break results in a rapid release of mass and energy to the containment with a resulting rapid rise in both the containment pressure and temperature. As the containment pressure rises, the RCS rapidly depressurizes which results in the generation of a compensated pressurizer pressure reactor trip at 0.418 seconds and a low pressurizerpressure SI setpoint at 4.1 seconds. The containment pressure continues to rise rapidly in response to the release of mass and energy until the end of the blowdown phase at 13.2 seconds.

The end of the blowdown phase marks a time when the initial inventory in the RCS has been exhausted and a slow process of filling the RCS downcomer in preparation for reflood has begun. Since the mass and energy release during this period is low, pressure decreases slightly and then increases in response to the reflood mass and energy release out to a second peak occurring at approximately 70 seconds. The turn around in containment pressure at 60 seconds is a result of the accumulator nitrogen cover gas flow ending at 60.73 seconds, initiation of the sprays at 72.73 seconds, and initiation of the containment fan coolers (CFCs) at 84.24 seconds. Reflood continues at a reduced flooding rate due to the buildup of mass in the RCS core which offsets the downcomer head. This reductions in flooding rate and the continued action of the CFCs and spray leads to a slowly decreasing pressure out to the end of reflood, which occurs at 207.4 seconds.

At this juncture, by design of the Reference 1 model, energy removal from the SG secondaries begins at a high rate, resulting in a rapid rise in containment pressure from the end of reflood out to approximately 797.5 seconds when energy has been removed from the SG in the faulted loop, bringing the SG in the faulted loop secondary pressure down to the containment design pressure of 74.7 psia. The result of the SG secondary energy release is a peak containment pressure of 69.98 psia and a peak containment temperature of 283.7°F at 1,007 seconds, the third major peak for this transient. After this event, the mass and energy released is reduced due to so much energy removal from the SGs having been accomplished and pressure slowly decreases out to the recirculation switchover time of 3,398.34 seconds.

At this time, the ECCS is realigned for recirculation resulting in an increase in the SI temperature due to delivery from the hot sump. At 6,000 seconds the injection sprays are terminated which results in a slight increase in pressure until the recirculation sprays are initiated at 7,200 seconds. The pressure once again decreases until the recirculation sprays are terminated at 14,400 seconds. After a slight increase in pressure, the containment pressure continues to decrease due to lower decay heat, SG energy release and continued CFC cooling. This trend continues to the end of the transient at 2.6E+06 seconds. Table 14.3.4-20 presents sequence of events for the DEPS with minimum safeguards transient.

LOCA Containment Response Transient Description: Double Ended Pump Suction Break with Maximum Safeguards

The DEPS break with maximum safeguards has a transient history very similar to the minimum safeguards case discussed above. Table 14.3.4-21 provides the key sequence of events and Table 14.3.4-29 shows that a peak pressure of 68.01 psia @ 12.51 seconds was calculated.

Steam Generator Tube Material Properties Evaluation

An evaluation was performed to determine the effect of correcting the steam generator tube material properties on the LOCA containment peak pressure (NSAL-14-2). The steam generator tube properties were corrected from stainless steel to Inconel 690® in the LOCA mass and energy release calculation which was updated to include the drift model and break flow with inertia model originally approved in Reference 1. The combination of these changes determined that the current results remain limiting and no further changes to the results are applied.

14.3.4.2.5   Conclusion

The LOCA containment response analyses have been performed as part of the transition Point Beach Units 1 and 2 EPU. The analyses included long-term pressure and temperature profiles for each case. As illustrated in the results found in Section 14.3.4.2.4, all cases resulted in a peak containment pressure that was less than 60 psig (74.7 psia). In addition, all long-term cases were well below 50% of the peak value within 24 hours. Based on the results, all applicable criteria for Point Beach have been met.

14.3.4.2.6   REFERENCES

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7. Letter from Herbert N. Berkow, Director (NRC) to James A. Gresham (Westinghouse), "Acceptance of Clarifications of Topical Report WCAP-10325-P-A, 'Westinghouse LOCA Mass and Energy Release Model for Containment Design - March 1979 Version' (TAC No. MC7980)," October 18,2005.

8. NAI 8907-06, Rev. 16, GOTHIC Containment Analysis Package Technical Manual, Version 7.2a, January 2006.

9. NAI 8907-09, Rev. 9, GOTHIC Containment Analysis Package Qualification Report, Version 7.2a, January 2006.

10. AICHE Journal Volume 8, #2, Sprays formed by Flashing Liquid Jets, Brown and York, May 1962.

11. Docket No. 50-244, Safety Evaluation by the Office of Nuclear Reactor Regulation Related to Amendment No. 97 to Renewed Facility Operating License No. DPR-I8 R E. Ginna Nuclear Power Plant Inc., R. E. Ginna Nuclear. Power Plant, Nuclear Regulatory Commission, July 11,2006.

12. NAI 8907-02, Rev. 17, *GOTHIC* Containment Analysis Package User Manual. Version 7.2a(QA), January 2006.

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

14. WCAP-15065-P-A, Revision 1, Technical Justification for Eliminating Pressurizer Surge Line Rupture as the Structural Design Basis for Point Beach Units 1 and 2 Nuclear Plants, June 2001.

15. NRC Safety Evaluation, PBNP Units 1 and 2-Issuance of License Amendments Regarding Extended Power Uprate, dated May 3, 2011.

16. NRC Safety Evaluation, PBNP Units 1 and 2-Issuance of License Amendments Regarding use of Alternate Source Term, dated April 14, 2011.

17. Westinghouse Calculation CN-CRA-08-6, LOCA Mass and Energy Release and Containment Response Analysis for the EPU Program, Revision 1, Approved December 29, 2008.

18. Westinghouse Calculation CN-CRA-08-6, LOCA Mass and Energy Release and Containment Response Analysis for the EPU Program, Revision 1-A, Approved June 29, 2011, Revision 1-B, Approved December 21, 2012, Revision 1-C, Approved March 15, 2013.

19. Westinghouse Memo to Harv Hanneman, WEP-08-4, Transmittal of Short-Term LOCA Mass and Energy Release Evaluation for the Point Beach Extended Power Uprate, dated January 9, 2008.

20.   NRC 2011-0025, NEXTera Energy to the NRC, License Amendment Request 261, Extended Power Uprate, Request for Additional Information, dated February 25, 2011.

21.   SCR 2013-0188-01, "Reduction of CFC Heat Removal Requirement," dated November 21, 2013.

Table 14.3.4-1   System Parameters Initial Conditions

Parameters	Value
Core Thermal Power (1) MWt)	1,811.0
Reactor coolant System Flow Rate, per Loop (lbm/sec)	18,777.8
Vessel Outlet Temperature[2] (°F)	617.5
Core Inlet Temperature[2] (°F)	549.3
Vessel Baffle-Barrel Configuration	Upflow
Initial SG Steam Pressure (psia)	833
SG Design (Unit 1/Unit 2)	Model 44F/Δ47
SGTP (%)	0
Initial SG Secondary-Side Mass[3] (lbm)	105,704.5
Assumed Maximum Containment Backpressure (psia)	74.7
Accumulator • Water volume (ft^3) (per accumulator)[4] • N_2cover Gas Pressure[5] (psia) • Temperature (°F)	1,100.0 834.7 120.0
SI Start Time (sec) [total time form beginning of event, which includes the maximum delay from reaching the setpoint]	40.8 (DEHL) 41.1 (DEPS)
Notes: (1) Includes allowance for calorimetric error (+0.6 percent of power). (2) Analysis value includes an additional +6.4°F allowance for instrument error and dead band. (3) SG secondary-side mass includes appropriate uncertainty and/or allowance. (4) Does not include accumulator line volume. (5) N_2 cover gas pressure includes uncertainty of +20 psi.	

Table 14.3.4-2   SAFETY INJECTION FLOW - MINIMUM SAFEGUARDS

RCS Pressure (psia)	Total Flow (lbm/sec)
Injection Mode (reflood phase)	
14.7	363.2
34.7	341.2
54.7	316.6
74.7	290.1
94.7	257.6
114.7	214.3
Injection Mode (post-reflood phase)	
74.7	290.1
Recirculation Mode	
14.7	270.9

Table 14.3.4-2   SAFETY INJECTION FLOW - MINIMUM SAFEGUARDS

Table 14.3.4-3   SAFETY INJECTION FLOW - MAXIMUM SAFEGUARDS

RCS Pressure (psia)	Total Flow (lbm/sec)
Injection Mode (reflood phase)	
14.7	800.0
34.7	766.0
54.7	728.0
74.7	686.0
94.7	642.0
114.7	592.0
Injection Mode (post-reflood phase)	
74.7	686.0
Recirculation Mode	
14.7	270.9

Table 14.3.4-4   LOCA MASS AND ENERGY RELEASE ANALYSIS - CORE DECAY HEAT
FRACTION

Time (sec)	Core Decay Heat Fraction of Full Power
10	0.053876
15	0.050401
20	0.048018
40	0.042401
60	0.039244
80	0.037065
100	0.035466
150	0.032724
200	0.030936
400	0.027078
600	0.024931
800	0.023389
1,000	0.022156
1,500	0.019921
2,000	0.018315
4,000	0.014781
6,000	0.013040
8,000	0.012000
10,000	0.011262
15,000	0.010097
20,000	0.009350
40,000	0.007778
60,000	0.006958
80,000	0.006424
100,000	0.006021
150,000	0.005323
200,000	0.004847
400,000	0.003770
600,000	0.003201
800,000	0.002834
1,000,000	0.002580
10,000,000	0.000808

Table 14.3.4-5   DEHL BREAK BLOWDOWN M&E RELEASE
Sheet 1 of 3

| Time Seconds | Break Path No. 1[1] | | Break Path No. 2[2] | |
| | Mass | Energy | Mass | Energy |
	lbm/sec	Thousand Btu/sec	lbm/sec	Thousand Btu/sec
0.0	0.0	0.0	0.0	0.0
0.0	43846.0	27887.8	43844.6	27886.0
0.0	45581.8	28990.5	45315.1	28815.3
0.1	36937.2	23746.7	26048.1	16529.1
0.2	34526.8	22145.0	22870.3	14419.6
0.3	33548.1	21468.3	20426.7	12701.9
0.4	32138.3	20561.1	19205.4	11746.6
0.5	31497.7	20152.1	18437.8	11098.0
0.6	31251.8	20008.9	17870.3	10605.8
0.7	30559.2	19615.8	17445.7	10227.6
0.8	30255.5	19507.5	17112.7	9927.4
0.9	29722.9	19281.0	16818.4	9668.5
1.0	28825.9	18816.3	16637.3	9488.7
1.1	27937.2	18363.2	16499.4	9344.2
1.2	27095.1	17943.8	16452.1	9259.3
1.3	26227.4	17508.9	16474.0	9218.7
1.4	25293.2	17020.8	16531.7	9203.2
1.5	24241.7	16443.5	16614.5	9205.2
1.6	23187.9	15850.7	16712.8	9220.1
1.7	22189.7	15293.5	16812.6	9240.2
1.8	21280.9	14810.1	16906.4	9261.4
1.9	20414.5	14373.2	16990.2	9281.4
2.0	19458.1	13889.5	17051.6	9293.6
2.1	18654.4	13424.7	17087.6	9296.0
2.2	18100.5	13036.7	17092.8	9285.3
2.3	17763.3	12712.1	17071.0	9263.0
2.4	17554.4	12445.7	17023.3	9229.4
2.5	17377.5	12209.2	16950.0	9184.5
2.6	17194.2	11991.8	16851.1	9127.9
2.7	17009.2	11794.5	16728.3	9060.1
2.8	16880.6	11659.4	16576.5	8978.2
2.9	16776.8	11553.0	16390.4	8879.2
3.0	16708.5	11462.7	16182.1	8769.3
3.1	16691.4	11387.9	15948.8	8647.2
3.2	16720.9	11331.1	15695.8	8515.4
3.3	16769.4	11281.7	15407.8	8365.6
3.4	16818.8	11235.7	15072.6	8190.8
3.5	16854.5	11187.2	14690.4	7991.2
3.6	16867.1	11132.8	14278.3	7776.0

Table 14.3.4-5   DEHL BREAK BLOWDOWN M&E RELEASE
Sheet 2 of 3

Time Seconds	Break Path No. 1[1]		Break Path No. 2[2]	
	Mass	Energy	Mass	Energy
	lbm/sec	Thousand Btu/sec	lbm/sec	Thousand Btu/sec
3.7	16859.7	11074.5	13861.8	7559.1
3.8	16829.5	11014.7	13443.0	7342.3
3.9	16737.2	10940.6	13044.9	7137.8
4.0	16613.0	10854.6	12659.7	6941.6
4.2	16358.5	10681.3	11903.2	6556.3
4.4	16106.2	10498.4	11121.9	6156.0
4.6	15872.8	10313.0	10357.0	5762.7
4.8	15653.1	10131.8	9667.2	5408.2
5.0	15430.3	9948.9	9061.8	5098.0
5.2	15197.8	9759.0	8532.1	4827.4
5.4	14962.9	9565.8	8074.3	4594.0
5.6	14729.0	9368.6	7663.1	4384.8
5.8	11491.4	7978.5	7283.5	4191.9
6.0	11092.1	7701.1	6911.9	4002.6
6.2	10674.3	7471.3	6542.8	3815.8
6.4	10230.5	7182.4	6179.3	3634.6
6.6	9830.2	6927.1	5823.5	3460.7
6.8	9417.5	6700.2	5478.9	3295.3
7.0	9035.7	6451.6	5150.1	3139.7
7.2	8662.6	6184.8	4838.3	2992.9
7.4	8123.5	5895.9	4548.2	2855.8
7.6	7769.5	5656.3	4282.9	2728.6
7.8	7292.9	5355.2	4057.0	2619.8
8.0	6806.4	5107.8	3867.0	2523.4
8.2	6269.9	4842.3	3672.8	2416.2
8.4	5753.8	4577.2	3481.8	2314.1
8.6	5267.8	4319.2	3289.8	2218.8
8.8	4802.4	4073.9	3092.6	2126.9
9.0	4343.4	3833.5	2893.3	2037.3
9.2	3908.2	3607.1	2698.7	1952.7
9.4	3471.4	3391.9	2511.2	1874.0
9.6	3035.9	3180.8	2330.3	1800.6
9.8	2619.6	2887.8	2157.0	1732.4
10.0	2390.1	2676.6	1991.2	1668.5
10.2	2270.9	2494.4	1832.8	1607.2
10.4	2144.4	2341.2	1683.6	1549.6
10.6	1980.7	2193.2	1545.1	1501.6
10.8	1811.0	2060.4	1423.7	1459.8
11.0	1662.9	1934.8	1314.6	1417.7

Table 14.3.4-5   DEHL BREAK BLOWDOWN M&E RELEASE
                  Sheet 3 of 3

Time Seconds	Break Path No. 1[1]		Break Path No. 2[2]	
	Mass	Energy	Mass	Energy
	lbm/sec	Thousand Btu/sec	lbm/sec	Thousand Btu/sec
11.2	1521.2	1804.9	1222.2	1378.7
11.4	1407.7	1692.4	1142.4	1324.7
11.4	1407.4	1692.0	1142.2	1324.5
11.4	1407.0	1691.6	1141.9	1324.3
11.4	1406.7	1691.3	1141.7	1324.2
11.4	1406.3	1690.9	1141.5	1324.0
11.4	1406.0	1690.5	1141.2	1323.8
11.4	1405.6	1690.1	1141.0	1323.7
11.4	1405.2	1689.7	1140.8	1323.5
11.6	1286.2	1558.7	1066.9	1265.9
11.8	1179.4	1443.7	961.0	1164.6
12.0	1060.0	1306.1	825.5	1014.6
12.2	916.1	1134.1	663.5	820.0
12.4	800.1	995.3	565.4	700.9
12.6	686.5	858.0	515.0	640.5
12.8	508.6	635.5	579.8	721.9
13.0	452.9	565.6	650.5	808.9
13.2	356.9	447.3	708.3	878.2
13.4	267.8	336.2	756.1	932.0
13.6	171.0	214.4	795.4	973.3
13.8	63.7	79.0	810.7	986.3
14.0	0.0	0.0	782.4	956.9
14.2	0.0	0.0	667.3	822.6
14.4	0.0	0.0	513.0	636.0
14.6	0.0	0.0	492.8	612.5
14.8	61.4	78.6	415.9	516.8
15.0	54.7	70.3	289.1	360.2
15.2	0.0	0.0	161.0	201.7
15.4	0.0	0.0	0.0	0.0

Notes:

1. Path 1: M&E exiting from the reactor vessel side of the break.

2. Path 2: M&E exiting from the steam generator side of the break.

Table 14.3.4-6   DEHL BREAK MASS BALANCE

Time (Seconds)		0.00	15.40	15.40+ε
		Mass (thousand lbm)		
Initial	In RCS and ACC	414.12	414.12	414.12
Added Mass	Pumped Injection	0	0	0
	Total Added	0	0	0
Total Available		414.12	414.12	414.12
Distribution	Reactor Coolant	272.88	51.57	68.81
	Accumulator	141.24	102.64	85.10
	Total Contents	414.12	153.91	153.91
Effluent	Break Flow	0	260.20	260.20
	ECCS Spill	0	0	0
	Total Effluent	0	260.20	260.20
Total Accountable		414.12	414.11	414.11

Note: +ε is used to indicate that the column represents the bottom of core recovery conditions that occurs instantaneously after blowdown.

Table 14.3.4-7   DEHL BREAK ENERGY BALANCE

Time (Seconds)		0.00	15.40	15.40+ε
			Energy (million Btu)	
InitialEnergy	In RCS, ACC, S GEN	432.19	432.19	432.19
Added Energy	Pumped Injection	0	0	0
	Decay Heat	0	2.91	2.91
	Heat from Secondary	0	13.04	13.04
	Total Added	0	15.96	15.96
Total Available		432.19	448.15	448.15
Distribution	Reactor Coolant	160.52	12.44	13.74
	Accumulator	12.73	9.25	7.95
	Core Stored	15.37	6.69	6.69
	Primary Metal	85.83	80.53	80.53
	Secondary Metal	43.54	42.14	42.14
	Steam Generator	114.21	125.80	125.80
	Total Contents	432.19	276.84	276.84
Effluent	Break Flow	0	170.82	170.82
	ECCS Spill	0	0	0
	Total Effluent	0	170.82	170.82
Total Accountable		432.19	447.66	447.66

Note: +ε is used to indicate that the column represents the bottom of core recovery conditions
       that occurs instantaneously after blowdown.

Table 14.3.4-8   DEPS BREAK BLOWDOWN M&E RELEASE (MINIMUM AND
MAXIMUM SAFEGUARDS)
Sheet 1 of 3

Time Seconds	Break Path No. 1[1]		Break Path No. 2[2]	
	Mass	Energy	Mass	Energy
	lbm/sec	Thousand Btu/sec	lbm/sec	Thousand Btu/sec
0.0	0.0	0.0	0.0	0.0
0.0	78894.0	42750.2	40705.8	22023.7
0.1	40724.8	22122.5	19870.4	10738.6
0.2	43133.1	25260.2	21792.8	11787.4
0.3	46398.1	25670.1	23269.5	12594.3
0.4	46379.4	25977.9	23648.7	12805.2
0.5	44127.4	25038.8	23196.0	12564.7
0.6	44661.9	25648.2	22624.4	12261.2
0.7	44352.4	25740.4	22179.8	12026.5
0.8	43212.9	25312.9	21896.2	11877.8
0.9	41893.4	24757.8	21658.3	11752.2
1.0	40609.1	24205.0	21395.7	11611.5
1.1	39333.7	23636.3	21091.7	11447.2
1.2	38065.0	23051.2	20756.6	11265.1
1.3	36798.2	22448.3	20388.1	11064.3
1.4	35479.7	21799.4	19991.7	10847.6
1.5	34051.2	21077.3	19612.3	10640.8
1.6	32698.9	20414.6	19310.0	10476.3
1.7	31635.6	19955.5	19028.6	10323.4
1.8	30622.2	19550.9	18711.2	10150.7
1.9	29432.6	19048.8	18365.8	9962.4
20.	27828.6	18297.9	18011.3	9769.4
2.1	23134.9	15447.2	17649.8	9572.6
2.2	19735.2	13443.4	17274.0	9368.1
2.3	17245.1	11959.5	16925.5	9179.3
2.4	15332.8	10777.2	16706.4	9062.3
2.5	14094.3	10004.0	16477.1	8939.4
2.6	13337.5	9529.6	15926.1	8640.6
2.7	12801.1	9186.6	15489.6	8405.5
2.8	12300.7	8865.8	15136.6	8216.5
2.9	11859.0	8601.9	14892.8	8087.8
3.0	11439.2	8368.8	14682.8	7977.5
3.1	11067.4	8177.5	14470.9	7865.9
3.2	10716.0	7999.1	14279.4	7765.6
3.3	10393.6	7836.4	14091.1	7667.1
3.4	10103.4	7694.3	14352.7	7817.7
3.5	9843.3	7569.4	14447.3	7873.3
3.6	9602.5	7451.6	14409.5	7856.4

Table 14.3.4-8   DEPS BREAK BLOWDOWN M&E RELEASE (MINIMUM AND
MAXIMUM SAFEGUARDS)
Sheet 2 of 3

Time Seconds	Break Path No. 1[(1)]		Break Path No. 2[(2)]	
	Mass	Energy	Mass	Energy
	lbm/sec	Thousand Btu/sec	lbm/sec	Thousand Btu/sec
3.7	9382.9	7341.4	14426.8	7870.2
3.8	9187.2	7241.2	14399.9	7859.5
3.9	9013.0	7147.4	14327.3	7823.4
4.0	8857.1	7056.5	14252.8	7786.2
4.2	8591.0	6880.9	14009.0	7658.6
4.4	8355.1	6691.6	13646.3	7466.5
4.6	8153.9	6491.1	13277.5	7273.7
4.8	7977.3	6276.3	12844.1	7047.2
5.0	7860.7	6083.4	14256.6	6847.3
5.2	7746.9	5889.1	11990.9	6603.8
5.4	7563.7	5677.8	11564.9	6383.3
5.6	7332.2	5438.9	11167.0	6164.8
5.8	7146.1	5218.4	10891.9	5982.2
6.0	7037.1	5041.3	10743.9	5847.2
6.2	7205.4	5047.0	10662.2	5738.7
6.4	7421.1	5125.4	10741.3	5719.0
6.6	7160.8	5132.1	10562.9	5575.6
6.8	6358.7	4820.2	10408.9	5444.6
7.0	5857.6	4517.7	10127.4	5255.7
7.2	5633.3	4325.4	9669.3	4976.4
7.4	5466.7	4176.4	9125.9	4655.0
7.6	5296.0	4042.2	8629.5	4365.1
7.8	5114.4	3916.7	8272.3	4157.3
8.0	4927.4	3795.8	7886.1	3941.6
8.2	4742.0	3684.4	7530.5	3741.3
8.4	4512.6	3569.6	7078.4	3485.7
8.6	4248.7	3436.5	6688.0	3247.7
8.8	4019.3	3319.2	6376.9	3042.9
9.0	3806.3	3210.2	6080.7	2845.4
9.2	3607.5	3113.6	5809.7	2663.9
9.4	3410.3	3029.9	5559.4	2498.5
9.6	3217.7	2956.3	5376.6	2370.1
9.8	3014.0	2883.3	5153.4	2230.2
10.0	2804.5	2820.6	4889.2	2078.2
10.2	2577.2	2758.3	4610.2	1924.7
10.4	2305.1	2654.5	4303.9	1765.5
10.6	1957.4	2392.5	3865.3	1557.8
10.8	1663.6	2057.8	3440.3	1356.6

Table 14.3.4-8   DEPS BREAK BLOWDOWN M&E RELEASE (MINIMUM AND
MAXIMUM SAFEGUARDS)
Sheet 3 of 3

Time Seconds	Break Path No. 1[1]		Break Path No. 2[2]	
	Mass	Energy	Mass	Energy
	lbm/sec	Thousand Btu/sec	lbm/sec	Thousand Btu/sec
11.0	1448.5	1798.8	3211.0	1222.3
11.2	1268.2	1578.9	3071.7	1116.1
11.4	1069.5	1335.1	2961.9	1028.9
11.6	894.3	1118.3	2797.3	936.0
11.8	740.1	926.5	2535.5	823.4
12.0	600.6	752.8	2196.3	697.3
12.2	480.7	603.0	1869.5	583.3
12.4	370.1	464.6	1489.0	458.1
12.6	253.7	318.8	1033.2	314.3
12.8	140.1	176.3	531.6	160.7
13.0	25.4	32.1	54.0	16.3
13.2	0.0	0.0	0.0	0.0

Notes: 1. Path 1: M&E exiting from the steam generator side of the break.
2. Path 2: M&E exiting from the broken loop reactor coolant pump side of the break.

Table 14.3.4-9   DEPS BREAK REFLOOD M&E RELEASE - MINIMUM SI
Sheet 1 of 5

Time Seconds	Break Path No. 1[1]		Break Path No. 2 Flow[2]	
	Mass	Energy	Mass	Energy
	lbm/sec	Thousand Btu/sec	lbm/sec	Thousand Btu/sec
13.2	0.0	0.0	0.0	0.0
13.7	0.0	0.0	0.0	0.0
13.8	0.0	0.0	0.0	0.0
14.0	0.0	0.0	0.0	0.0
14.1	0.0	0.0	0.0	0.0
14.11	0.0	0.0	0.0	0.0
14.2	82.9	97.9	0.0	0.0
14.3	27.4	32.4	0.0	0.0
14.4	20.8	24.5	0.0	0.0
14.6	25.8	30.5	0.0	0.0
14.7	34.6	40.9	0.0	0.0
14.8	41.8	49.4	0.0	0.0
14.9	48.6	57.5	0.0	0.0
15.0	55.2	65.2	0.0	0.0
15.1	61.4	72.6	0.0	0.0
15.2	67.4	79.7	0.0	0.0
15.21	68.8	81.4	0.0	0.0
15.3	72.4	85.5	0.0	0.0
15.4	76.9	90.8	0.0	0.0
15.5	81.2	95.9	0.0	0.0
15.6	85.3	100.8	0.0	0.0
15.7	89.3	105.5	0.0	0.0
15.8	93.1	110.0	0.0	0.0
15.9	96.8	114.4	0.0	0.0
16.0	100.4	118.7	0.0	0.0
16.1	103.9	122.8	0.0	0.0
16.2	107.3	126.9	0.0	0.0
16.3	110.6	130.8	0.0	0.0
17.3	139.9	165.4	0.0	0.0
18.3	164.3	194.3	0.0	0.0
18.9	265.0	313.7	2208.0	266.8
19.4	317.3	375.8	2920.0	353.2
20.4	326.4	386.7	2979.0	365.6
21.4	322.4	381.9	2935.5	361.6
22.4	318.1	376.8	2887.5	357.0
22.9	315.9	374.2	2863.1	354.6
23.4	313.8	371.7	2838.5	352.2
24.4	309.6	366.7	2789.7	347.4

Table 14.3.4-9   DEPS BREAK REFLOOD M&E RELEASE - MINIMUM SI
Sheet 2 of 5

Time Seconds	Break Path No. 1[1]		Break Path No. 2 Flow[2]	
	Mass	Energy	Mass	Energy
	lbm/sec	Thousand Btu/sec	lbm/sec	Thousand Btu/sec
25.4	305.5	361.8	2741.7	342.7
26.4	301.5	357.1	2694.8	338.1
27.4	297.7	352.6	2649.2	333.5
27.8	296.2	350.8	2631.3	331.8
28.4	294.1	348.2	2604.9	329.1
29.4	290.6	344.1	2562.0	324.9
30.4	287.2	340.1	2520.4	320.7
31.4	284.0	336.3	2480.1	316.7
32.4	280.9	332.6	2441.1	312.8
33.4	278.0	329.1	2403.2	309.1
34.4	275.2	325.8	2366.5	305.4
35.4	272.4	322.5	2330.9	301.8
36.4	269.8	319.4	2296.4	298.4
37.4	267.3	316.4	2262.8	295.0
38.4	264.8	313.5	2230.2	291.7
39.3	262.7	310.9	2201.6	288.8
39.4	223.5	264.5	921.5	170.0
40.4	188.7	223.2	970.9	164.3
41.4	198.8	235.2	1226.4	183.3
42.4	193.7	229.2	205.0	81.4
43.4	223.2	264.1	213.7	94.0
44.4	218.8	258.8	212.3	92.1
45.4	214.3	253.5	210.8	90.3
46.4	209.6	248.0	209.4	88.4
47.4	206.3	242.8	208.0	86.6
48.4	200.9	237.7	206.6	84.9
49.4	196.6	232.6	205.3	83.2
50.4	192.3	227.4	203.9	81.5
51.4	187.9	222.3	202.6	79.8
52.4	183.6	217.2	201.3	78.1
53.4	179.3	212.1	199.9	76.4
53.9	177.2	209.6	199.3	75.6
54.4	175.1	207.0	198.7	74.8
55.4	170.8	202.0	197.4	73.2
56.4	166.6	197.0	196.1	71.6
57.4	162.4	192.0	194.9	70.0
58.4	158.2	187.0	193.7	68.4
59.4	154.0	182.1	192.5	66.9
60.4	149.9	177.3	191.3	65.4

Table 14.3.4-9   DEPS BREAK REFLOOD M&E RELEASE - MINIMUM SI
Sheet 3 of 5

Time Seconds	Break Path No. 1[1]		Break Path No. 2 Flow[2]	
	Mass	Energy	Mass	Energy
	lbm/sec	Thousand Btu/sec	lbm/sec	Thousand Btu/sec
61.4	145.8	172.4	190.1	64.0
62.4	141.8	167.6	189.0	62.5
63.4	137.8	162.9	187.9	61.1
64.4	133.8	158.2	186.8	59.8
65.4	130.0	153.6	185.7	58.4
66.4	126.1	149.1	184.7	57.2
67.4	122.3	144.6	183.7	55.9
68.4	118.6	140.2	182.7	54.7
69.4	115.0	135.9	181.8	53.5
70.4	111.4	131.7	180.9	52.3
71.4	107.9	127.6	180.0	51.2
72.4	104.5	123.5	179.1	50.2
72.8	103.2	121.9	178.8	49.8
73.4	101.6	120.0	178.2	49.1
74.4	100.1	118.3	177.4	48.1
76.4	97.2	114.9	175.7	46.1
78.4	94.5	111.7	174.1	44.2
80.4	91.9	108.6	172.6	42.4
82.4	89.4	105.6	171.1	40.6
84.4	87.0	102.8	169.7	39.0
86.4	84.8	100.2	168.3	37.4
88.4	82.6	97.7	167.1	35.9
90.4	80.6	95.3	165.8	34.4
92.4	78.8	93.1	164.7	33.1
94.4	77.0	91.0	163.6	31.8
96.4	75.4	89.1	162.6	30.6
98.4	73.8	87.2	161.6	29.4
99.0	73.4	86.7	161.3	29.1
100.4	72.4	85.6	160.7	28.3
102.4	71.1	84.0	159.8	27.3
104.4	69.6	82.6	159.0	26.4
106.4	68.7	81.2	158.3	25.5
108.4	67.7	80.0	157.6	24.7
110.4	66.8	78.9	157.0	23.9
112.4	65.9	77.9	156.4	23.2
114.4	65.1	77.0	155.8	22.6
116.4	64.4	76.1	155.3	22.0
118.4	63.8	75.4	154.9	21.5
120.4	63.2	74.7	154.4	21.0

Table 14.3.4-9   DEPS BREAK REFLOOD M&E RELEASE - MINIMUM SI
Sheet 4 of 5

Time Seconds	Break Path No. 1[1]		Break Path No. 2 Flow[2]	
	Mass	Energy	Mass	Energy
	lbm/sec	Thousand Btu/sec	lbm/sec	Thousand Btu/sec
122.4	62.7	74.1	154.1	20.5
124.4	62.2	73.5	153.7	20.1
126.4	61.8	73.0	153.4	19.7
128.4	61.5	72.6	153.1	19.3
130.4	61.1	72.2	152.8	19.0
132.0	60.9	72.0	152.6	18.8
132.4	60.9	71.9	152.6	18.7
134.4	60.6	71.6	152.3	18.5
136.4	60.4	71.4	152.1	18.2
138.4	60.2	71.2	152.0	18.0
140.4	60.1	71.0	151.8	17.8
142.4	60.0	70.9	151.7	17.7
144.4	59.9	70.7	151.5	17.5
146.4	59.8	70.6	151.4	17.4
148.4	59.7	70.6	151.3	17.2
150.4	59.7	70.5	151.2	17.1
152.4	59.7	70.5	151.1	17.0
154.4	59.6	70.5	151.0	16.9
156.4	59.6	70.5	151.0	16.9
158.4	59.6	70.5	150.9	16.8
160.4	59.7	70.5	150.9	16.7
162.4	59.7	70.5	150.8	16.7
164.4	59.7	70.6	150.8	16.6
166.4	59.8	70.6	150.8	16.6
168.4	59.8	70.7	150.7	16.6
168.9	59.8	70.7	150.7	16.5
170.4	59.9	70.8	150.7	16.5
172.4	60.0	709.	150.7	16.5
174.4	60.0	70.9	150.7	16.5
176.4	60.1	71.0	150.6	16.5
178.4	60.2	71.1	150.6	16.5
180.4	60.3	71.2	150.6	16.4
182.4	60.4	71.3	150.6	16.4
184.4	60.5	71.5	150.6	16.4
186.4	60.6	71.6	150.6	16.4
188.4	60.7	71.7	150.6	16.4
190.4	60.8	71.8	150.6	16.5
192.4	60.9	71.9	150.6	16.5
194.4	61.0	72.0	150.6	16.5

Table 14.3.4-9   DEPS BREAK REFLOOD M&E RELEASE - MINIMUM SI
Sheet 5 of 5

| Time Seconds | Break Path No. 1[1] | | Break Path No. 2 Flow[2] | |
| | Mass | Energy | Mass | Energy |
	lbm/sec	Thousand Btu/sec	lbm/sec	Thousand Btu/sec
196.4	61.1	72.2	150.7	16.5
198.4	61.2	72.3	150.7	16.5
200.4	61.3	72.4	150.7	16.5
202.4	61.4	72.5	150.7	16.5
204.4	61.5	72.7	150.7	16.5
206.4	61.6	72.8	150.7	16.6
207.4	61.7	72.9	150.7	16.6

Notes: 1. Path 1: M&E exiting from the steam generator side of the break.
         2. Path 2: M&E exiting from the broken loop reactor coolant pump side of the break.

Table 14.3.4-10 DEPS BREAK REFLOOD M&E RELEASE - MAXIMUM SI
Sheet 1 of 4

Time Seconds	Break Path No. 1[1]		Break Path No. 2 Flow[2]	
	Mass	Energy	Mass	Energy
	lbm/sec	Thousand Btu/sec	lbm/sec	Thousand Btu/sec
13.2	0.0	0.0	0.0	0.0
13.7	0.0	0.0	0.0	0.0
13.8	0.0	0.0	0.0	0.0
14.0	0.0	0.0	0.0	0.0
14.1	0.0	0.0	0.0	0.0
14.11	0.0	0.0	0.0	0.0
14.2	82.9	97.9	0.0	0.0
14.3	27.4	32.4	0.0	0.0
14.4	20.8	24.5	0.0	0.0
14.6	25.8	30.5	0.0	0.0
14.7	34.6	40.9	0.0	0.0
14.8	41.8	49.4	0.0	0.0
14.9	48.6	57.5	0.0	0.0
15.0	55.2	65.2	0.0	0.0
15.1	61.4	72.6	0.0	0.0
15.2	67.4	79.7	0.0	0.0
15.21	68.8	81.4	0.0	0.0
15.3	72.4	85.5	0.0	0.0
15.4	76.9	90.8	0.0	0.0
15.5	81.2	95.9	0.0	0.0
15.6	85.3	100.8	0.0	0.0
15.7	89.3	105.5	0.0	0.0
15.8	93.1	110.0	0.0	0.0
15.9	96.8	114.4	0.0	0.0
16.0	100.4	118.7	0.0	0.0
16.1	103.9	122.8	0.0	0.0
16.2	107.3	126.9	0.0	0.0
16.3	110.6	130.8	0.0	0.0
17.3	139.9	165.4	0.0	0.0
18.3	164.3	194.3	0.0	0.0
18.9	265.0	313.7	2208.0	266.8
19.4	317.3	375.8	2920.0	353.2
20.4	326.4	386.7	2979.0	365.6
21.4	322.4	381.9	2935.5	361.6
22.4	318.1	376.8	2887.5	357.0
22.9	315.9	374.2	2863.1	354.6
23.4	313.8	371.7	2838.5	352.2
24.4	309.6	366.7	2789.7	347.4
25.4	305.5	361.8	2741.7	342.7

Table 14.3.4-10 DEPS BREAK REFLOOD M&E RELEASE - MAXIMUM SI
        Sheet 2 of 4

Time Seconds	Break Path No. 1[1]		Break Path No. 2 Flow[2]	
	Mass	Energy	Mass	Energy
	lbm/sec	Thousand Btu/sec	lbm/sec	Thousand Btu/sec
26.4	301.5	357.1	2694.8	338.1
27.4	297.7	352.6	2649.2	333.5
27.8	296.2	350.8	2631.3	331.8
28.4	294.1	348.2	2604.9	329.1
29.4	290.6	344.1	2562.0	324.9
30.4	287.2	340.1	2520.4	320.7
31.4	284.0	336.3	2480.1	316.7
32.4	280.9	332.6	2441.1	312.8
33.4	278.0	329.1	2403.2	309.1
34.4	275.2	325.8	2366.5	305.4
35.4	272.4	322.5	2330.9	301.8
36.4	269.8	319.4	2296.4	298.4
37.4	267.3	316.4	2262.8	295.0
38.4	264.8	313.5	2230.2	291.7
39.3	262.7	310.9	2201.6	288.8
39.4	223.5	264.5	921.5	170.0
40.4	188.7	223.2	970.9	164.3
41.4	217.0	256.8	1583.4	210.5
42.4	189.5	224.2	379.0	68.0
43.4	170.2	201.2	392.5	58.9
44.4	169.9	200.9	393.2	58.9
45.4	169.6	200.5	394.0	58.9
46.4	169.2	200.1	394.7	58.8
47.4	168.9	199.8	395.4	58.8
48.4	168.6	199.4	396.2	58.8
49.4	168.3	199.1	396.9	58.8
50.4	168.0	198.7	397.6	58.8
51.4	167.7	198.3	398.4	58.7
52.4	167.4	198.0	399.1	58.7
53.4	167.1	197.6	399.8	58.7
54.4	166.8	197.3	400.5	58.7
54.7	166.7	197.2	400.8	58.7
55.4	166.5	196.9	401.3	58.6
56.4	166.2	196.5	402.0	58.6
57.4	165.9	196.2	402.7	58.6
58.4	165.6	195.8	403.5	58.6
59.4	165.2	195.4	404.2	58.5
60.4	164.9	198.0	405.0	58.5
61.4	164.6	194.6	405.7	58.5

Table 14.3.4-10 DEPS BREAK REFLOOD M&E RELEASE - MAXIMUM SI
Sheet 3 of 4

Time Seconds	Break Path No. 1[1] Mass lbm/sec	Energy Thousand Btu/sec	Break Path No. 2 Flow[2] Mass lbm/sec	Energy Thousand Btu/sec
62.4	164.3	194.2	406.5	58.5
63.4	163.9	193.8	407.3	58.5
64.4	163.6	193.4	408.1	58.4
65.4	163.2	193.0	408.8	58.4
66.4	162.9	192.6	409.6	58.4
67.4	162.5	192.2	410.5	58.4
68.4	162.2	191.8	411.3	58.3
69.4	161.8	191.3	412.1	58.3
70.4	161.4	190.9	412.9	58.3
71.2	161.1	190.5	413.6	58.3
71.4	161.0	190.4	413.8	58.3
72.4	160.7	190.0	414.6	58.2
73.4	160.3	189.5	415.5	58.2
74.4	159.9	189.1	416.3	58.2
76.4	159.1	188.1	418.1	58.1
78.4	158.3	187.2	419.8	58.1
80.4	157.5	186.2	421.6	58.0
82.4	156.6	185.2	423.5	58.0
84.4	155.7	184.1	425.3	57.9
86.4	154.8	183.1	427.2	57.9
88.4	154.0	182.0	429.0	57.8
89.0	153.7	181.7	429.6	57.8
90.4	153.0	181.0	430.9	57.7
92.4	152.1	179.9	432.8	57.7
94.4	151.2	179.8	434.6	57.6
96.4	150.3	177.7	436.5	57.5
98.4	149.3	176.5	438.4	57.4
100.4	148.4	175.4	440.3	57.4
102.4	147.4	174.3	442.2	57.3
104.4	146.4	173.1	444.1	57.2
106.4	145.4	172.0	445.9	57.1
108.3	144.5	170.9	447.7	57.0
108.4	144.5	170.8	447.8	57.0
110.4	143.5	169.6	449.7	56.9
112.4	142.5	168.5	451.6	56.8
114.4	141.5	167.3	453.5	56.7
116.4	140.5	166.2	455.3	56.6
118.4	139.5	165.0	457.2	56.5
120.4	138.6	163.8	459.0	56.4

Table 14.3.4-10 DEPS BREAK REFLOOD M&E RELEASE - MAXIMUM SI
        Sheet 4 of 4

Time Seconds	Break Path No. 1[1]		Break Path No. 2 Flow[2]	
	Mass	Energy	Mass	Energy
	lbm/sec	Thousand Btu/sec	lbm/sec	Thousand Btu/sec
122.4	137.6	162.7	460.9	56.3
124.4	136.6	161.5	462.9	56.6
126.4	135.7	160.5	464.4	56.8
128.4	134.8	159.4	465.9	57.1
129.6	134.3	158.8	466.8	57.3
130.4	134.0	158.4	467.4	57.4
132.4	133.1	157.3	468.9	57.6
134.4	132.2	156.3	470.4	57.9
136.4	131.3	155.3	471.8	58.1
138.4	130.5	154.2	473.3	58.3
140.4	129.6	153.2	474.7	58.6
142.4	128.7	152.2	476.2	58.8
144.4	127.9	151.2	477.6	59.0
146.4	127.0	150.2	479.0	59.2
148.4	126.2	149.2	480.4	59.4
150.4	125.4	148.2	481.8	59.6
152.4	124.5	147.2	483.2	59.8
153.2	124.2	146.8	483.8	59.9

Notes: 1. Path 1: M&E exiting from the steam generator side of the break.
        2. Path 2: M&E exiting from the broken loop reactor coolant pump side of the break.

Table 14.3.4-11   DEPS - MINIMUM SAFETY INJECTION PRINCIPAL PARAMETERS DURING REFLOOD
Sheet 1 of 3

Time sec	Temp °F	Flooding Rate in/sec	Carry-over Fraction	Core Height ft	Downcomer Height ft	Flow Fraction	Total	Injector Accumulator	SI Spill	Enthalpy Btu/lbm
							(Pounds mass per second)			
13.2	152	0	0	0	0	0.5	0	0	0	0
14	150.8	24.8	0	0.77	1.49	0	4306.8	4306.8	0	90.1
14.1	150.5	25.817	0	0.98	1.45	0	4295	4295	0	90.1
14.11	150.4	25.712	0	1.09	1.42	0	4283.2	4283.2	0	90.1
14.6	150.2	3.115	0.156	1.37	2.46	0.39	4227.2	4227.2	0	90.1
15.2	150.4	3.17	0.289	1.5	4.42	0.544	4155.6	4155.6	0	90.1
15.21	150.4	3.164	0.294	1.5	4.5	0.546	4153	4153	0	90.1
15.9	150.7	3.087	0.4	1.61	6.69	0.579	4082.2	4082.2	0	90.1
18.9	152.2	4.473	0.622	2	15.67	0.709	3758	3758	0	90.1
19.4	152.4	4.936	0.641	2.07	15.82	0.747	3701.2	3701.2	0	90.1
20.4	153	4.805	0.669	2.21	15.83	0.746	3613.9	3613.9	0	90.1
22.9	154.6	4.461	0.702	2.51	15.83	0.743	3436.5	3436.5	0	90.1
27.8	158	4.108	0.724	3	15.83	0.735	3147.3	3147.3	0	90.1
33.4	162.3	3.863	0.732	3.51	15.83	0.726	2881.7	2881.7	0	90.1
39.3	167	3.678	0.735	4	15.83	0.717	2652.6	2652.6	0	90.1

Table 14.3.4-11 DEPS - MINIMUM SAFETY INJECTION PRINCIPAL PARAMETERS DURING REFLOOD
Sheet 2 of 3

Time sec	Temp °F	Flooding Rate in/sec	Carry-over Fraction	Core Height ft	Downcomer Height ft	Flow Fraction	Total	Injector Accumulator	SI Spill	Enthalpy Btu/lbm
							(Pounds mass per second)			
40.4	167.8	2.967	0.724	4.08	15.83	0.648	1311.4	1311.4	0	90.1
41.4	168.6	3.065	0.726	4.15	15.83	0.66	1578.6	1291.1	0	86.08
42.4	169.4	3.137	0.724	4.22	15.81	0.673	289.1	0	0	68
43.4	170.2	3.304	0.732	4.29	15.53	0.681	285.2	0	0	68
46.4	172.9	3.127	0.73	4.51	14.73	0.677	285.9	0	0	68
53.9	180.2	2.71	0.725	5.01	13.04	0.665	287.5	0	0	68
63.4	190.4	2.212	0.715	5.55	11.49	0.642	289.1	0	0	68
72.8	200.4	1.782	0.705	6	10.54	0.605	290.2	0	0	68
86.4	213.9	1.496	0.698	6.55	9.84	0.599	290.2	0	0	68
99	224	1.311	0.693	7	9.58	0.596	290.2	0	0	68
116.4	235.2	1.157	0.69	7.55	9.64	0.595	290.2	0	0	68
132	243.6	1.091	0.69	8	9.9	0.597	290.2	0	0	68
150.4	252.2	1.058	0.693	8.51	10.35	0.601	290.2	0	0	68
168.9	259.7	1.048	0.698	9	10.84	0.604	290.1	0	0	68
180.4	264	1.047	0.701	9.3	11.16	0.607	290.1	0	0	68

Table 14.3.4-11 DEPS - MINIMUM SAFETY INJECTION PRINCIPAL PARAMETERS DURING REFLOOD
    Sheet 3 of 3

Time sec	Temp °F	Flooding Rate in/sec	Carry-over Fraction	Core Height ft	Downcomer Height ft	Flow Fraction	Total	Injector Accumulator	SI Spill	Enthalpy Btu/lbm
							(Pounds mass per second)			
188.4	266.8	1.048	0.703	9.51	11.38	0.608	290.1	0	0	68
207.4	272.8	1.051	0.709	10	11.91	0.611	290.1	0	0	68

Table 14.3.4-12 DEPS - MAXIMUM SAFETY INJECTION PRINCIPAL PARAMETERS DURING REFLOOD
       Sheet 1 of 2

Time sec	Temp °F	Flooding Rate in/sec	Carry-over Fraction	Core Height ft	Downcomer Height ft	Flow Fraction	Total	Injector Accumulator	SI Spill	Enthalpy Btu/lbm
							(Pounds mass per second)			
13.2	152	0	0	0	0	0.5	0	0	0	0
14	150.8	24.8	0	0.77	1.49	0	4306.8	4306.8	0	90.1
14.1	150.5	25.817	0	0.98	1.45	0	4295	4295	0	90.1
14.11	150.4	25.712	0	1.09	1.42	0	4283.2	4283.2	0	90.1
14.6	150.2	3.115	0.156	1.37	2.46	0.39	4227.2	4227.2	0	90.1
15.2	150.4	3.17	0.289	1.5	4.42	0.544	4155.6	4155.6	0	90.1
15.21	150.4	3.164	0.294	1.5	4.5	0.546	4153	4153	0	90.1
15.9	150.7	3.087	0.4	1.61	6.69	0.579	4082.2	4082.2	0	90.1
18.9	152.2	4.473	0.622	2	15.67	0.709	3758	3758	0	90.1
19.4	152.4	4.936	0.641	2.07	15.82	0.747	3701.2	3701.2	0	90.1
20.4	153	4.805	0.669	2.21	15.83	0.746	3613.9	3613.9	0	90.1
22.9	154.6	4.461	0.702	2.51	15.83	0.743	3436.5	3436.5	0	90.1
27.8	158	4.108	0.724	3	15.83	0.735	3147.3	3147.3	0	90.1
33.4	162.3	3.863	0.732	3.51	15.83	0.726	2881.7	2881.7	0	90.1
39.3	167	3.678	0.735	4	15.83	0.717	2652.6	2652.6	0	90.1

Table 14.3.4-12 DEPS - MAXIMUM SAFETY INJECTION PRINCIPAL PARAMETERS DURING REFLOOD
Sheet 2 of 2

Time sec	Temp °F	Flooding Rate in/sec	Carry-over Fraction	Core Height ft	Downcomer Height ft	Flow Fraction	Total	Injector Accumulator	SI Spill	Enthalpy Btu/lbm
							(Pounds mass per second)			
40.4	167.8	2.967	0.724	4.08	15.83	0.648	1311.4	1311.4	0	90.1
41.4	168.6	3.247	0.73	4.15	15.83	0.679	1965.3	1285.6	0	82.46
42.4	169.4	2.914	0.727	4.22	15.83	0.619	682.3	0	0	68
47.4	173.6	2.768	0.723	4.54	15.83	0.623	685.7	0	0	68
54.7	181.3	2.72	0.725	5	15.83	0.624	685.6	0	0	68
63.4	192	2.659	0.728	5.54	15.83	0.625	685.6	0	0	68
68.4	198.6	2.623	0.729	5.84	15.83	0.626	685.6	0	0	68
71.2	202.3	2.601	0.73	6	15.83	0.626	685.6	0	0	68
80.4	214.7	2.527	0.733	6.53	15.83	0.628	685.5	0	0	68
89	225.3	2.453	0.735	7	15.83	0.629	685.4	0	0	68
98.4	235.4	2.37	0.738	7.5	15.83	0.63	685.4	0	0	68
108.3	244.5	2.282	0.74	8	15.83	0.632	685.3	0	0	68
120.4	254	2.174	0.743	8.58	15.83	0.634	685.2	0	0	68
129.6	260.2	2.099	0.745	9	15383	0.634	685.2	0	0	68
142.4	267.6	2.001	0.748	9.56	15.83	0.635	685.2	0	0	68
153.2	273	1.922	0.75	10	15.83	0.636	685.2	0	0	68

Table 14.3.4-13 DEPS BREAK POST-REFLOOD M&E RELEASE - MINIMUM SI
        Sheet 1 of 4

Time Seconds	Break Path No. 1[1]		Break Path No. 2 Flow[2]	
	Mass	Energy	Mass	Energy
	lbm/sec	Thousand Btu/sec	lbm/sec	Thousand Btu/sec
207.5	115.8	147.1	177.5	47.3
212.5	115.4	146.7	177.4	47.3
217.5	115.1	146.2	177.2	47.3
222.5	114.8	145.8	177.1	47.2
227.5	114.5	146.8	177.0	46.9
232.5	115.2	146.4	176.8	46.9
237.5	114.8	145.9	176.7	46.9
242.5	114.5	145.5	176.6	46.8
147.5	114.2	145.1	176.4	46.8
252.5	114.9	146.0	176.3	46.5
257.5	114.6	145.6	176.2	46.5
262.5	114.2	145.1	176.0	46.4
267.5	113.9	144.7	176.3	46.4
272.5	113.5	144.2	176.7	46.4
277.5	114.2	145.2	175.9	46.1
282.5	113.9	144.7	176.3	46.0
287.5	113.5	144.2	176.7	46.0
292.5	113.1	143.8	177.0	46.0
297.5	112.8	143.3	177.4	46.0
302.5	113.5	144.2	176.7	45.7
307.5	113.1	143.7	177.1	45.6
312.5	112.7	143.2	177.4	45.6
317.5	112.3	142.8	177.8	45.6
322.5	112.0	142.3	178.2	45.6
327.5	112.6	143.1	177.5	45.3
332.5	112.3	142.7	177.9	45.3
337.5	111.9	142.2	178.3	45.2
342.5	111.5	141.7	178.7	45.2
347.5	112.1	142.5	178.0	44.9
352.5	111.7	142.0	178.4	44.9
357.5	111.3	141.5	178.8	44.9
362.5	110.9	141.0	179.2	44.9
367.5	111.6	141.8	179.6	44.6
372.5	111.2	141.3	179.0	44.6
377.5	110.8	140.7	179.4	45.8
382.5	110.3	140.2	179.8	45.8
387.5	110.9	141.0	179.2	45.5
392.5	110.5	140.4	179.6	45.5
397.5	110.1	139.9	180.1	45.5

Table 14.3.4-13 DEPS BREAK POST-REFLOOD M&E RELEASE - MINIMUM SI
Sheet 2 of 4

Time Seconds	Break Path No. 1[1] Mass lbm/sec	Energy Thousand Btu/sec	Break Path No. 2 Flow[2] Mass lbm/sec	Energy Thousand Btu/sec
402.5	109.7	139.4	180.5	45.5
407.5	110.4	140.2	179.8	45.1
412.5	110.0	139.8	180.2	45.1
417.5	109.6	139.3	180.5	45.1
422.5	110.3	140.1	179.9	44.8
427.5	109.9	139.6	180.3	44.7
432.5	109.5	139.2	180.6	44.7
437.5	109.1	138.7	181.0	44.7
442.5	109.7	139.4	180.4	44.4
447.5	109.3	139.0	180.8	44.4
452.5	109.0	138.5	181.2	44.3
457.5	109.5	139.2	180.6	44.0
462.5	109.1	138.7	181.0	44.0
467.5	108.7	138.2	181.4	45.2
472.5	109.3	138.8	180.9	44.9
477.5	108.8	138.3	181.3	44.9
482.5	108.4	137.8	181.7	44.9
487.5	108.9	138.4	181.2	44.6
492.5	108.5	137.9	181.7	44.6
497.5	108.1	137.3	182.1	44.5
502.5	108.5	137.9	181.6	44.3
507.5	108.1	137.4	182.1	44.2
512.5	108.5	137.9	181.6	44.0
517.5	108.1	137.3	182.1	43.9
522.5	107.6	136.8	182.5	43.9
527.5	108.0	137.3	182.1	43.7
532.5	107.6	136.7	182.6	44.8
537.5	107.9	137.2	182.2	44.6
542.5	107.5	136.6	182.7	44.6
547.5	107.8	137.0	182.3	44.3
552.5	107.3	136.4	182.9	44.3
557.5	107.6	136.8	182.5	44.0
562.5	107.1	136.1	183.1	44.0
567.5	107.4	136.5	182.8	43.8
572.5	106.8	135.8	183.3	43.8
577.5	107.1	136.1	183.0	43.5
582.5	106.5	135.4	183.6	43.5
587.5	106.8	135.7	183.4	43.3
592.5	107.0	136.0	183.2	44.2

Table 14.3.4-13 DEPS BREAK POST-REFLOOD M&E RELEASE - MINIMUM SI
Sheet 3 of 4

| Time Seconds | Break Path No. 1[1] | | Break Path No. 2 Flow[2] | |
| | Mass | Energy | Mass | Energy |
	lbm/sec	Thousand Btu/sec	lbm/sec	Thousand Btu/sec
597.5	106.4	135.2	183.8	44.2
602.5	106.6	135.4	183.6	44.0
607.5	106.7	135.7	183.4	43.8
612.5	106.1	134.9	184.0	43.8
617.5	106.3	135.0	183.9	43.6
622.5	106.4	135.2	183.8	43.4
627.5	106.5	135.3	183.7	43.2
632.5	106.5	135.3	183.6	43.0
637.5	105.8	134.4	184.4	44.1
642.5	105.8	134.5	184.4	44.0
647.5	105.8	134.4	184.4	43.8
652.5	105.7	134.3	184.4	43.6
657.5	105.6	134.2	184.5	43.4
662.5	105.5	134.1	184.7	43.3
667.5	105.3	133.9	184.8	43.2
672.5	105.8	134.5	184.3	42.8
677.5	105.6	134.1	184.6	43.8
682.5	105.3	133.8	184.9	43.7
687.5	105.6	134.2	184.6	43.4
692.5	105.2	133.6	185.0	43.3
697.5	105.3	133.8	184.8	43.1
702.5	105.4	133.9	184.8	42.9
707.5	105.4	133.9	184.8	42.7
712.5	105.3	133.8	184.9	43.5
717.5	105.1	133.6	185.1	43.4
722.5	104.8	133.2	185.4	43.3
727.5	104.9	133.3	185.2	43.0
732.5	104.9	133.3	185.3	42.8
737.5	104.7	133.0	185.5	42.6
742.5	104.8	133.1	185.4	43.4
747.5	104.6	132.9	185.6	43.3
752.5	104.6	132.9	185.6	43.0
757.5	104.2	132.4	186.0	42.9
762.5	104.2	132.4	186.0	42.7
767.5	104.3	132.5	185.9	43.4
772.5	104.2	132.4	186.0	43.2
777.5	104.1	132.2	186.1	43.0
782.5	103.9	132.0	186.3	42.8
787.5	103.6	131.7	186.5	42.6

Table 14.3.4-13 DEPS BREAK POST-REFLOOD M&E RELEASE - MINIMUM SI
Sheet 4 of 4

| Time Seconds | Break Path No. 1[1] | | Break Path No. 2 Flow[2] | |
| | Mass | Energy | Mass | Energy |
	lbm/sec	Thousand Btu/sec	lbm/sec	Thousand Btu/sec
792.5	103.2	131.2	186.9	43.5
1006.1	103.2	131.2	186.9	43.5
1006.2	49.0	61.1	241.2	57.2
1007.5	49.0	61.1	241.2	57.2
1298.9	49.0	61.1	241.2	57.2
1299.0	45.9	52.8	244.3	18.7
3395.0	37.1	42.6	253.1	20.3
3395.1	37.1	42.6	233.8	40.6
3600.0	36.4	41.9	234.5	40.7

Notes: 1. Path 1: M&E exiting from the steam generator side of the break.
2. Path 2: M&E exiting from the broken loop reactor coolant pump side of the break.

Table 14.3.4-14 DEPS BREAK POST-REFLOOD M&E RELEASE - MAXIMUM SI
Sheet 1 of 3

| Time Seconds | Break Path No. 1[1] | | Break Path No. 2 Flow[2] | |
| | Mass | Energy | Mass | Energy |
	lbm/sec	Thousand Btu/sec	lbm/sec	Thousand Btu/sec
153.3	137.8	173.9	547.9	71.2
158.3	137.2	173.1	548.5	71.2
163.3	137.5	173.5	548.2	71.0
168.3	136.9	172.7	548.8	71.0
173.3	137.2	173.1	548.5	70.8
178.3	136.6	172.3	549.1	70.8
183.3	135.9	171.5	549.8	70.8
188.3	136.2	171.9	549.5	70.6
193.3	135.6	171.0	550.2	70.7
198.3	135.8	171.4	549.9	70.4
203.3	135.3	170.7	550.5	70.4
208.3	135.7	171.2	550.0	70.2
213.3	135.1	170.5	550.6	702.
218.3	134.6	169.8	551.1	70.2
223.3	135.0	170.3	550.7	71.2
228.3	134.4	169.6	551.3	71.2
233.3	134.8	170.1	550.9	71.0
238.3	134.2	169.3	551.5	71.0
243.3	134.6	169.8	551.2	70.8
248.3	134.0	169.0	551.7	70.8
253.3	134.3	169.4	551.4	70.5
258.3	133.7	168.7	552.0	70.6
263.3	134.0	169.1	551.7	70.3
268.3	133.4	168.3	552.3	70.3
273.3	133.6	168.6	552.1	70.1
278.3	133.0	167.8	552.7	70.1
283.3	133.3	168.1	552.5	69.9
288.3	132.6	167.3	553.1	70.0
293.3	132.8	167.6	552.9	69.7
298.3	133.0	167.8	552.7	69.5
303.3	132.3	167.0	553.4	69.6
308.3	132.5	167.2	553.2	69.4
313.3	132.7	167.4	553.1	69.2
318.3	131.9	166.5	553.8	69.2
323.3	132.1	166.6	553.7	70.2
328.3	132.2	166.7	553.6	70.0
333.3	131.4	165.8	554.3	70.1
338.3	131.5	165.9	554.3	69.9
343.3	131.5	165.9	554.2	69.7

Table 14.3.4-14 DEPS BREAK POST-REFLOOD M&E RELEASE - MAXIMUM SI
Sheet 2 of 3

Time Seconds	Break Path No. 1[1] Mass lbm/sec	Energy Thousand Btu/sec	Break Path No. 2 Flow[2] Mass lbm/sec	Energy Thousand Btu/sec
348.3	131.5	165.9	554.2	69.6
353.3	130.7	164.9	555.0	69.6
358.3	130.7	164.9	555.0	69.5
363.3	130.7	164.9	555.1	69.3
368.3	130.6	164.8	555.1	69.2
373.3	130.5	164.7	555.2	69.0
378.3	130.4	164.5	555.3	68.9
383.3	130.2	164.3	555.5	68.8
388.3	130.0	164.1	555.7	68.6
393.3	129.8	163.8	555.9	68.5
398.3	129.6	163.5	556.1	68.4
403.3	129.4	163.2	556.3	68.3
408.3	129.2	163.0	556.6	69.4
413.3	129.6	163.5	556.1	69.1
418.3	129.3	163.1	556.4	69.0
423.3	129.0	162.7	556.8	68.9
428.3	129.3	163.1	556.5	68.6
433.3	128.8	162.5	556.9	68.6
438.3	129.0	162.8	556.7	68.3
443.3	129.1	162.9	556.6	68.1
448.3	128.5	162.1	557.3	68.1
453.3	128.5	162.1	557.2	69.1
458.3	128.4	162.0	557.3	68.9
463.3	128.2	161.8	557.5	68.8
468.3	128.6	162.2	557.1	68.5
473.3	128.2	161.8	557.5	68.4
478.3	128.3	161.9	557.4	68.2
483.3	128.3	161.9	557.4	68.0
488.3	128.1	161.7	557.6	67.8
493.3	127.8	161.2	557.9	68.8
498.3	127.8	161.2	557.9	68.6
503.3	127.6	161.0	558.1	68.5
508.3	127.6	161.0	558.1	68.2
513.3	127.4	160.7	558.4	68.1
518.3	127.2	160.5	558.5	68.0
523.3	127.0	160.3	558.7	67.8
528.3	127.1	160.4	558.6	67.6
533.3	126.8	160.0	558.9	68.5
538.3	127.0	160.2	558.8	68.2

Table 14.3.4-14 DEPS BREAK POST-REFLOOD M&E RELEASE - MAXIMUM SI
Sheet 3 of 3

Time Seconds	Break Path No. 1[1]		Break Path No. 2 Flow[2]	
	Mass	Energy	Mass	Energy
	lbm/sec	Thousand Btu/sec	lbm/sec	Thousand Btu/sec
543.3	126.9	160.1	558.9	68.1
548.3	126.7	159.8	559.1	67.9
553.3	126.3	159.3	559.4	67.8
558.3	126.1	159.1	559.7	67.6
749.6	126.1	159.1	559.7	67.6
749.7	50.4	62.8	635.3	87.8
753.3	50.4	62.7	635.3	88.1
783.3	50.0	62.2	635.7	87.3
786.0	49.9	62.2	240.2	60.9
1244.4	49.9	62.2	240.2	60.9
1244.5	44.8	51.5	245.3	17.5
1646.0	41.9	48.3	248.2	18.0
1646.1	41.9	48.3	229.0	37.8
3600.0	34.9	40.2	236.0	39.1

Notes: 1. Path 1: M&E exiting from the steam generator side of the break.
2. Path 2: M&E exiting from the broken loop reactor coolant pump side of the break.

Table 14.3.4-15 DEPS BREAK MASS BALANCE MINIMUM SAFEGUARDS

	Time (Seconds)	.00	13.20	13.20+ε	207.40	1006.25	1298.87	3600.00
				Mass (Thousand lbm)				
Initial Mass	In RCS and Accumulator	414.12	414.12	414.12	414.12	414.12	414.12	414.12
Added Mass	Pumped Injection	0	0	0	48.19	279.95	364.86	1028.61
	Total Added	0	0	0	48.19	279.95	364.86	1028.61
*** Total Available ***		414.12	414.12	414.12	462.30	694.07	778.97	1442.73
Distribution	Reactor Coolant	272.88	18.13	42.74	71.08	71.08	71.08	71.08
	Accumulator	141.24	114.79	90.18	0	0	0	0
	Total Contents	414.12	132.92	132.92	71.08	71.08	71.08	71.08
Effluent	Break Flow	0	281.19	281.19	385.22	617.08	701.98	1365.73
	ECCS Spill	0	0	0	0	0	0	0
	Total Effluent	0	281.19	281.19	385.22	617.08	701.98	1365.73
*** Total Accountable ***		414.12	414.11	414.11	456.31	688.17	773.06	1436.82

Note: +ε is used to indicate that the column represents the bottom of core recovery conditions which occurs instantaneously after blowdown.

Table 14.3.4-16 DEPS BREAK MASS BALANCE MAXIMUM SAFEGUARDS

	Time (Seconds)	0.00	13.20	13.20+ε	153.20	749.72	1244.38	3600.00
				Mass (Thousand lbm)				
Initial Mass	In RCS and Accumulator	414.12	414.12	414.12	414.12	414.12	414.12	414.12
Added Mass	Pumped Injection	0	0	0	76.83	485.81	643.66	1289.51
	Total Added	0	0	0	76.83	485.81	643.66	1289.51
*** Total Available ***		414.12	414.12	414.12	490.94	899.93	1057.78	1703.63
Distribution	Reactor Coolant	272.88	18.13	42.74	75.05	75.05	75.05	75.05
	Accumulator	141.24	114.79	90.18	0	0	0	0
	Total Contents	414.12	132.92	132.92	75.05	75.05	75.05	75.05
Effluent	Break Flow	0	281.19	281.19	409.90	818.88	976.20	1622.05
	ECCS Spill	0	0	0	0	0	0	0
	Total Effluent	0	281.19	281.19	409.90	818.88	976.20	1622.05
*** Total Accountable ***		414.12	414.11	414.11	484.95	893.93	1051.25	1697.10

Note: +ε is used to indicate that the column represents the bottom of core recovery conditions which occurs instantaneously after blowdown.

Table 14.3.4-17 DEPS BREAK ENERGY BALANCE MINIMUM SAFEGUARDS

	Time (Seconds)	0.00	13.20	13.20+ε	207.40	1006.25	1298.87	3600.00
				Energy (Thousand Btu)				
Initial Energy	In RCS, Accumulators and Steam Generators	432.19	432.19	432.19	432.19	432.19	432.19	432.19
Added Energy	Pumped Injection	0	0	0	3.28	19.04	24.81	74.39
	Decay Heat	0	2.51	2.51	14.66	49.58	60.37	129.67
	Heat from Secondary	0	11.57	11.57	11.57	11.57	11.57	11.57
	Total Added	0	14.08	14.08	29.51	80.18	96.75	215.63
*** Total Available ***		432.19	446.27	446.27	461.70	512.38	528.94	647.82
Distribution	Reactor Coolant	160.52	5.03	7.25	19.19	19.19	19.19	19.19
	Accumulator	12.73	10.34	8.13	0	0	0	0
	Core Stored	15.37	9.47	9.47	2.77	2.63	2.54	1.81
	Primary Metal	85.83	82.17	82.17	68.99	41.71	38.46	27.37
	Secondary Metal	43.54	43.13	43.13	40.88	25.02	22.46	15.95
	Steam Generator	114.21	126.83	126.83	118.71	68.83	61.85	43.45
	Total Contents	432.19	276.97	276.97	250.55	157.38	144.50	107.78
Effluent	Break Flow	0	168.98	168.98	206.21	350.06	385.15	542.94
	ECCS Spill	0	0	0	0	0	0	0
	Total Effluent	0	168.98	168.98	206.21	350.06	385.15	542.94
*** Total Accountable ***		432.19	445.96	445.96	456.76	507.44	529.65	650.72

Note: +ε is used to indicate that the column represents the bottom of core recovery conditions which occurs instantaneously after blowdown.

Table 14.3.4-18 DEPS BREAK ENERGY BALANCE MAXIMUM SAFEGUARDS

	Time (Seconds)	0.00	13.20	13.20+ε	153.20	749.72	1244.38	3600.00
				Energy (Thousand Btu)				
Initial Energy	In RCS, Accumulators and Steam Generators	432.19	432.19	432.19	432.19	432.19	432.19	432.19
Added Energy	Pumped Injection	0	0	0	5.22	33.03	43.77	128.98
	Decay Heat	0	2.51	2.51	11.71	39.48	58.41	129.40
	Heat from Secondary	0	11.57	11.57	11.57	11.57	11.57	11.57
	Total Added	0	14.08	14.08	28.51	84.08	113.75	269.96
*** Total Available ***		432.19	446.27	446.27	460.70	516.27	545.94	702.15
Distribution	Reactor Coolant	160.52	5.03	7.25	20.32	20.32	20.32	20.32
	Accumulator	12.73	10.34	8.13	0	0	0	0
	Core Stored	15.37	9.47	9.47	2.77	2.56	2.43	1.81
	Primary Metal	85.83	82.17	82.17	67.85	42.32	36.79	27.41
	Secondary Metal	43.54	43.13	43.13	39.69	25.94	21.44	15.96
	Steam Generator	114.21	126.83	126.83	114.58	71.36	58.98	43.51
	Total Contents	432.19	276.97	276.97	245.21	162.50	139.96	109.01
Effluent	Break Flow	0	168.98	168.98	210.56	348.84	411.00	599.73
	ECCS Spill	0	0	0	0	0	0	0
	Total Effluent	0	168.98	168.98	210.56	348.84	411.00	599.76
*** Total Accountable ***		432.19	445.96	445.96	455.77	511.34	550.96	708.74

Note: +ε is used to indicate that the column represents the bottom of core recovery conditions which occurs instantaneously after blowdown.

Table 14.3.4-19 DOUBLE-ENDED HOT LEG BREAK SEQUENCE OF EVENTS

Time (sec)	Event Description
0.0	Break Occurs and Loss of Offsite Power is Assumed
0.311	Compensated Pressurizer Pressure for Reactor Trip (1968.7 psia) Reached and Turbine Trip Occurs
3.8	Low-Pressurizer Pressure Safety Injection (SI) Setpoint (1663 psia) Reached - Feedwater Isolation Signal
4.99	Broken Loop Accumulator Begins Injecting Water
5.03	Intact Loop Accumulator Begins Injecting Water
14.51	Peak Temperature Occurs (280.3°F)
14.51	Peak Pressure Occurs (70.39 psia)
15.4	End of Blowdown Phase
15.4	Feedwater Isolation Valves Closed
50.0	Transient Modeling Terminated

Table 14.3.4-20 DOUBLE-ENDED PUMP SUCTION BREAK SEQUENCE OF EVENTS
(MINIMUM SAFEGUARDS) Sheet 1 of 2

Time (sec)	Event Description
0	Break Occurs and Loss of Offsite Power is Assumed
0.24	Containment High Pressure Safety Injection Actuation Pressure Setpoint (20.7 psia; Analysis Value) Reached. (CFCs Actuated)
0.418	Compensated Pressurizer Pressure for Reactor Trip (1,968.7 psia) Reached and Turbine Trip Occurs
2.73	Containment Spray Actuation Pressure Setpoint (44.7 psia; Analysis Value) Reached
4.1	Low Pressurizer Pressure SI Setpoint (1,663 psia) Reached (Safety Injection Begins coincident with Low Pressurizer Pressure SI Setpoint)
5.29	Broken Loop Accumulator Begins Injecting Water
5.40	Intact Loop Accumulator Begins Injecting Water
13.2	End of Blowdown Phase
13.2	Accumulator Mass Adjustment for Refill Period
13.2	Feedwater Isolation Valves Closed
39.25	Broken Loop Acculumator Water Injection Ends
41.1	Pumped Safety Injection Begins (Included 37 Second Diesel Delay)
42.2	Intact Loop Accumulator Water Injection Ends
72.73	Containment Spray Pump (RWST) Begins
84.24	CFCs Begin Heat Removal (Includes 84 Second Delay)
207.4	End of Reflood for Minimum Safeguards Case
797.5	M&E Release Assumption: Broken Loop Steam Generator (SG) Equilibration When the Secondary Temperature is at Saturation (T_{SAT}) at Containment Design Pressure of 74.7 psia
1,006.25	M&E Release Assumption: Broken Loop SG Equilibration at Containment Pressure of 60.7 psia
1,007	Containment Peak Temperature Occurs (283.7°F)
1,007	Containment Peak Pressure Occurs (69.98 psia)
1,127.65	M&E Release Assumption: Intact Loop SG Equilibration When the Secondary Temperature is at Saturation (T_{SAT}) at Containment Design Pressure of 74.7 psia

Table 14.3.4-20 DOUBLE-ENDED PUMP SUCTION BREAK SEQUENCE OF EVENTS
(MINIMUM SAFEGUARDS) Sheet 2 of 2

1,298.87	M&E Release Assumption: Intact Loop SG Equilibration at Containment Pressure of 54.7 psia
3,398.34	Switchover to Recirculation Begins
6,000	Injection Sprays Terminated
7,200	Recirculation Sprays Initiated (Injection Spray Termination Plus 1,200 Second Delay)
14,400	Recirculation Spray Terminated
2.6E+6	Transient Modeling Terminated

Table 14.3.4-21 DOUBLE-ENDED PUMP SUCTION BREAK SEQUENCE OF EVENTS
(MAXIMUM SAFEGUARDS) Sheet 1 of 2

Time (sec)	Event Description
0.0	Break Occurs and Loss of Offsite Power is Assumed
0.24	Containment High Pressure SI Actuation Pressure Setpoint (20.7 psia; Analysis Value) Reached. (CFCs Actuated)
0.418	Compensated Pressurizer Pressure for Reactor Trip (1,968.7 psia) Reached and Turbine Trip Occurs
2.72	Containment Spray Actuation Pressure Setpoint (44.7 psia; Analysis Value) Reached
4.1	Low Pressurizer Pressure SI Setpoint (1,663 psia) Reached (Safety Injection Begins coincident with Low Pressurizer Pressure SI Setpoint)
5.29	Broken Loop Accumulator Begins Injecting Water
5.40	Intact Loop Accumulator Begins Injecting Water
12.51	Containment Peak Pressure Occurs (68.01 psia)
13.2	End of Blowdown Phase
13.2	Accumulator Mass Adjustment for Refill Period
13.2	Feedwater Isolation Valves Closed
39.25	Broken Loop Acculumator Water Injection Ends
41.1	Pumped Safety Injection Begins (Included 36 Second Diesel Delay)
42.2	Intact Loop Accumulator Water Injection Ends
72.72	Containment Spray Pump (RWST) Begins
84.24	CFCs Begin Heat Removal (Includes 84 Second Delay)
153.2	End of Reflood for Maximum Safeguards Case
563.3	M&E Release Assumption: Broken Loop Steam Generator (SG) Equilibration When the Secondary Temperature is at Saturation (T_{SAT}) at Containment Design Pressure of 74.7 psia
730.8	Containment Peak Temperature Occurs (280.7°F)
749.723	M&E Release Assumption: Broken Loop SG Equilibration at Containment Pressure of 54.7 psia

Table 14.3.4-21 DOUBLE-ENDED PUMP SUCTION BREAK SEQUENCE OF EVENTS
(MAXIMUM SAFEGUARDS) Sheet 2 of 2

975.023	M&E Release Assumption: Intact Loop SG Equilibration When the Secondary Temperature is at Saturation (T_{SAT}) at Containment Design Pressure of 74.7 psia
1,244.38	M&E Release Assumption: Intact Loop SG Equilibration at Containment Pressure of 44.7 psia
1,648.84	Switchover to Recirculation Begins
3,000	Injection Sprays Terminated
4,200	Recirculation Sprays Initiated (Injection Spray Termination Plus 1,200 Second Delay)
11,400	Recirculation Spray Terminated
2.6E+6	Transient Modeling Terminated

Table 14.3.4-22 RCS CONDITIONS FOR SHORT-TERM MASS AND ENERGY RELEASES

Minimum RCS Vessel Outlet Temperature	529.9°F
Minimum RCS Vessel / Core Inlet Temperature	525.0°F
Allowance for RCS Temperature Uncertainty	-6.4°F
Nominal RCS Pressure	2,250.0 psia
Allowance for RCS Pressure Uncertainty	+50.0 psia

Table 14.3.4-23 SHORT-TERM LOCA M&E RELEASES

Time (Sec)	Flow (lbm/sec)	Enthalpy (Btu/lbm)
Double-Ended Hot Leg 6" Break		
0.0	0.0	0.0
0.001	9615.02	598.04
3.0	9615.02	598.04
Double-Ended Cold Leg 3" Break		
0.0	0.0	0.0
0.001	2952.76	510.29
3.0	2952.76	510.29

Table 14.3.4-24 CONTAINMENT INTEGRITY LOCA ANALYSIS PARAMETERS
Sheet 1 of 2

Parameter	Value
Service Water Temperature (°F)	85
Refueling Water Storage Tank / Containment Injection Spray Water Temperature (°F)	100
Initial Containment Temperature (°F)	120
Initial Containment Pressure (psia)	16.7
Initial Relative Humidity (%)	20
Containment Net Free Volume (ft^3)	1,000,000
Reactor Containment Fan Coolers	
Total CFCs Available	4
Analysis Maximum Safeguards	4
Analysis Minimum Safeguards (with Diesel Failure)	2
Containment High Pressure Setpoint (psig)	6.0
Delay Time (sec) - Without Offsite Power	84.0
Air Flow Rate through Cooler (ft^3/min/CFC)	33,500
Containment Fan Cooler Heat Removal as a Function of Containment Saturation Temperature and Component Cooling Water Heat Exchanger Primary Side Outlet Temperature	See Table 14.3.4-25
Containment Spray Pumps	
Total CSPs Available	2
Analysis Maximum Safeguards	1
Analysis Minimum Safeguards (with Diesel Failure)	1

Containment Spray Pump Flow Rate (gpm/pump)
    Injection Phase
    Recirculation Phase
	See Table 14.3.4-26
	900

Containment Hi-Hi Pressure Setpoint (psig)	30.0
Spray Delay Time (Sec) Without Offsite Power (1 spray pump)	70

Table 14.3.4-24 CONTAINMENT INTEGRITY LOCA ANALYSIS PARAMETERS
Sheet 2 of 2

Parameter	Value
Containment Spray Termination time (sum of injection and recirculation phase including 1,200 second delay time) (sec) -Minimum Safeguards -Maximum Safeguards	 14,400.0 11,400.0
ECCS Recirculation	
ECCS Recirculation Switchover (sec) -Minimum Safeguards -Maximum Safeguards (after SI setpoint is reached)	 3,395.0 1,646.0
Residual Heat Removal System	
RHR Heat Exchangers[1]	
Modeled in analysis	1
Flows - Tube Side and Shell Side (gpm) -Tube Side -Shellside (component cooling water)	 1,951.0 2,780.0
Component Cooling Water Heat Exchangers	
Modeled in analysis	1
Flows - Shell Side and Tube Side (gpm) -Shellside -Tubeside (service water)	 2,895.0 2,700.0
CCW Misc. Heat Loads (MBtu/hr)	2.0
Notes: (1) Modeled with 10% tube plugging	

Table 14.3.4-25 CONTAINMENT FAN COOLER PERFORMANCE

Containment Temperature (°F)	Heat Removal Rate [Btu/sec] Per Reactor Containment Fan Cooler
120	839.25
160	2,070.5
190	3,799.25
210	5,390
220	5,830
230	6,167.75
240	6,508.75
260	7,184.25
270	7,516

Table 14.3.4-26 CONTAINMENT SPRAY PERFORMANCE

Containment Pressure (psig)	1 Pump (gpm)	2 Pumps (gpm)
0	1324.0	2665.8
10	1287.2	2589.8
20	1250.3	2515.6
30	1206.9	2431.0
40	1162.5	2342.7
50	1117.0	2252.4
60	1070.7	2160.2

Table 14.3.4-27 CONTAINMENT STRUCUTRAL HEAT SINK INPUT Sheet 1 of 7

GOTHIC Heat Sink Description	Area (ft²)	Material	Thickness (inches)
Upper Dome	1610	Paint type 1	0.01404
		Carbon Steel	0.2496
		Gap	0.021
		Concrete	36
Middle Dome	5912	Paint Type 1	0.01404
		Carbon Steel	0.2496
		Gap	0.021
		Concrete	36
Lower Dome	6432	Paint Type 1	0.01404
		Carbon Steel	0.2496
		Gap	0.021
		Concrete	36
Upper Containment outter wall (above 66')	16988	Paint Type 1	0.015
		Carbon Steel	0.2496
		Gap	0.021
		Concrete	42
Middle Containment outter wall (21' to 66')	14844	Paint Type 1	0.015
		Carbon Steel	0.2496
		Gap	0.021
		Concrete	42

Table 14.3.4-27 CONTAINMENT STRUCUTRAL HEAT SINK INPUT Sheet 2 of 7

GOTHIC Heat Sink Description	Area (ft²)	Material	Thickness (inches)
Lower Containment outer wall (8' top 21')	4166	Paint Type 1	0.015
		Carbon Steel	0.2496
		Gap	0.021
		Concrete	42
Rx Cavity: Shield wall / Rx Pit	1695	Paint Type 2	0.039
		Concrete	12
Rx Cavity: tunnel walls	260	Paint type 2	0.039
		Concrete	12
Rx Cavity: Keyway tower / shaft	1120	Paint Type 2	0.039
		Concrete	12
Rx Cavity: Floor slab	353	Paint Type 2	0.015
		Concrete	12
Pzr walls (inside 46'-86')	2027	Paint Type 2	0.039
		Concrete	15
Pzr floor slab	156	Paint Type 2	0.015
		Concrete	24
		Paint Type 2	0.039
Pzr missile shields	176	Paint Type 2	0.039
		Carbon Steel	0.5
		Gap	0.021
		Concrete	15

Table 14.3.4-27 CONTAINMENT STRUCUTRAL HEAT SINK INPUT Sheet 3 of 7

GOTHIC Heat Sink Description	Area (ft^2)	Material	Thickness (inches)
		Paint Type 1	0.039
Upper Ctmt interior walls	5420	Paint Type 2	0.039
		Concrete	15
Upper Ctmt floor / Annular Cmpt ceiling	4339	Paint Type 2	0.015
		Concrete	4
Annular Cmpt: Interior wall (46' to 66')	5372	Paint Type 2	0.039
		Concrete	15
Annular Cmpt: Interior wall (21' to 46')	8263	Paint Type 2	0.039
		Concrete	15
Annular Cmpt: lay-down area high wall (21' to 66')	585	Paint Type 2	0.039
		Concrete	18
Annular Cmpt 46' floor slab	3914	Paint Type 2	0.015
		Concrete	4
Annular Cmpt floor / Annular Sump ceiling (21')	4272	Paint Type 2	0.015
		Concrete	4
Annular Sump: interior walls (8' to 21')	4487	Paint type 2	0.039
		Concrete	15

Table 14.3.4-27 CONTAINMENT STRUCUTRAL HEAT SINK INPUT Sheet 4 of 7

GOTHIC Heat Sink Description	Area (ft²)	Material	Thickness (inches)
Annular Sump floor slab (8')	4352	Paint Type 2	0.015
		Concrete	12
Loop A: walls	6691	Paint Type 2	0.039
		Concrete	15
Loop A: floor slab	816	Paint Type 2	0.015
		Concrete	12
Loop A: missile shields	251.1	Paint Type 2	0.015
		Concrete	15
		Paint Type 2	0.039
Loop B: walls	8087	Paint Type 2	0.039
		Concrete	15
Loop B: floor slab	794	Paint Type 2	0.015
		Concrete	12
Loop B: missile shields	208	Paint Type 2	0.015
		Concrete	15
		Paint Type 2	0.039
Loop B: sub-pzr cmpt walls	286	Paint Type 2	0.039
		Concrete	15
Loop B: sub-pzr cmpt floor	176	Paint Type 2	0.015
		Concrete	24
		Paint Type 2	0.039

Table 14.3.4-27 CONTAINMENT STRUCUTRAL HEAT SINK INPUT Sheet 5 of 7

GOTHIC Heat Sink Description	Area (ft^2)	Material	Thickness (inches)
Refueling cavity wall	4691	Stainless Steel	0.1875
		Gap	0.021
		Concrete	18
		Paint Type 2	0.039
Refueling cavity floor / Annular sump ceiling	536	Stainless Steel	0.1875
		Gap	0.021
		Concrete	36
		Paint Type 2	0.039
Misc. steel in reactor cavity compartment	667.36	Paint Type 1	0.0130
		Carbon Steel	1.2630
Misc. steel in the pressurizer compartment	1.08	Paint Type 1	0.0130
		Carbon Steel	0.0050
Misc. steel in the upper containment	5048.27	Paint Type 1	0.0130
		Carbon Steel	0.3770
Misc. steel in the annular compartment	22507.34	Paint Type 1	0.0130
		Carbon Steel	0.3960
Misc. steel in the annular sump compartment	6662.86	Paint Type 1	0.0130

Table 14.3.4-27 CONTAINMENT STRUCUTRAL HEAT SINK INPUT Sheet 6 of 7

GOTHIC Heat Sink Description	Area (ft^2)	Material	Thickness (inches)
		Carbon Steel	0.2300
Misc. steel in the Loop A compartment	3390.63	Paint Type 1	0.0130
		Carbon Steel	0.3720
Misc. steel in the Loop B compartment	3390.63	Paint Type 1	0.0130
		Carbon Steel	0.3720
Misc. steel in the dome compartment	20731.29	Paint Type 1	0.0130
		Carbon Steel	0.1480
Misc. steel in refueling cavity compartment	398.26	Paint Type 1	0.0130
		Carbon Steel	1.4750
1 CFC in upper containment compartment; unpainted copper	7071.89	Copper	0.0130
1 CFC in upper containment compartment	21.53	Stainless Steel	1.0220
1 CFC in annular compartment	7075.48	Copper	0.0130
Unpainted stainless steel in Annular Compartment; 1 CFC	24.08	Stainless Steel	0.6700
Polar crane & rail girder in the upper containment	8094.46	Paint Type 1	0.0130

Table 14.3.4-27 CONTAINMENT STRUCUTRAL HEAT SINK INPUT Sheet 7 of 7

GOTHIC Heat Sink Description	Area (ft²)	Material	Thickness (inches)
		Carbon Steel	0.9060
A RCP in the Loop A compartment	570.49	Paint Type 1	0.0079
		Copper	2.583
A RCP in the Loop B compartment	570.49	Paint Type 1	0.0079
		Copper	2.583
PRT Unpainted SS	509	Stainless Steel	0.6700

Table 14.3.4-28 MATERIAL PROPERTIES FOR CONTAINMENT STRUCUTRAL HEAT SINKS

Material Type	Density	Thermal Conductivity	Specific Heat
	lbm/ft^3	Btu/hr-ft-°F	Btu/lbm-°F
Concrete	144	0.81	0.2
Stainless Steel	488	9.4	0.123
Carbon Steel	490	26	0.115
Copper (pure)	557.69	231.7	0.092
Gap (air)	0.06	0.0174	0.241
Amercote 66 top coating / Dimecote 6 primer coating (Paint Type 1)	1	0.25	21.7
Phenoline 305 top coating / Carboline 195 primer coating (Paint Type 2)	1	0.187	37.8

Table 14.3.4-29 SUMMARY OF PEAK CONTAINMENT PRESSURE AND
TEMPERATURES

Case	Peak Pressure (psia)	Time (sec)	Peak Temp (°F)	Time (sec)	Pressure @ 24 hours (psia)
DEHL	70.39	14.51	280.3	14.51	-
DEPS MINSI	69.98	1,007	283.7	1,007	23.7
DEPS MAXSI	68.01	12.51	280.7	730.8	20.7
Limit	74.7		286.0		

Figure 14.3.4-1 CONTAINMENT PRESSURE - DOUBLE-ENDED HOT-LEG BREAK
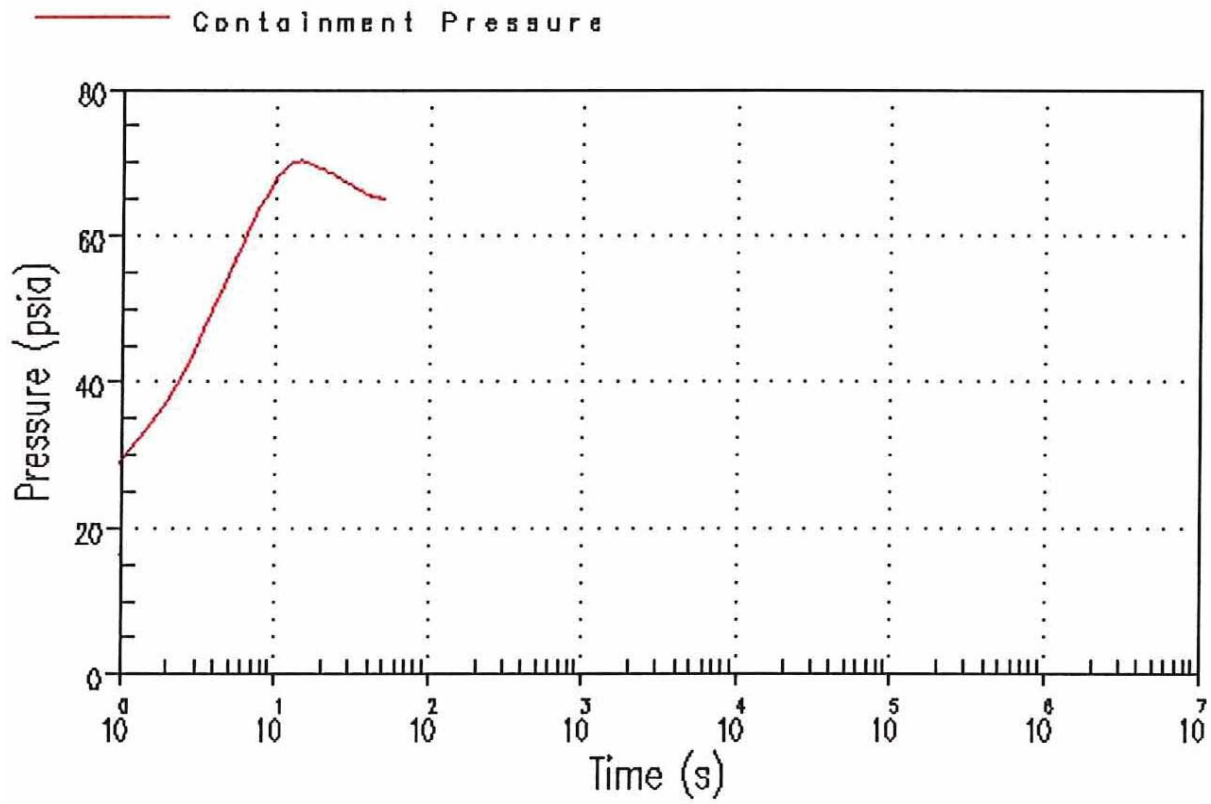
Figure 14.3.4-2 CONTAINMENT TEMPERATURE - DOUBLE-ENDED HOT-LEG BREAK

Figure 14.3.4-3 CONTAINMENT PRESSURE - DOUBLE-ENDED PUMP SUCTION
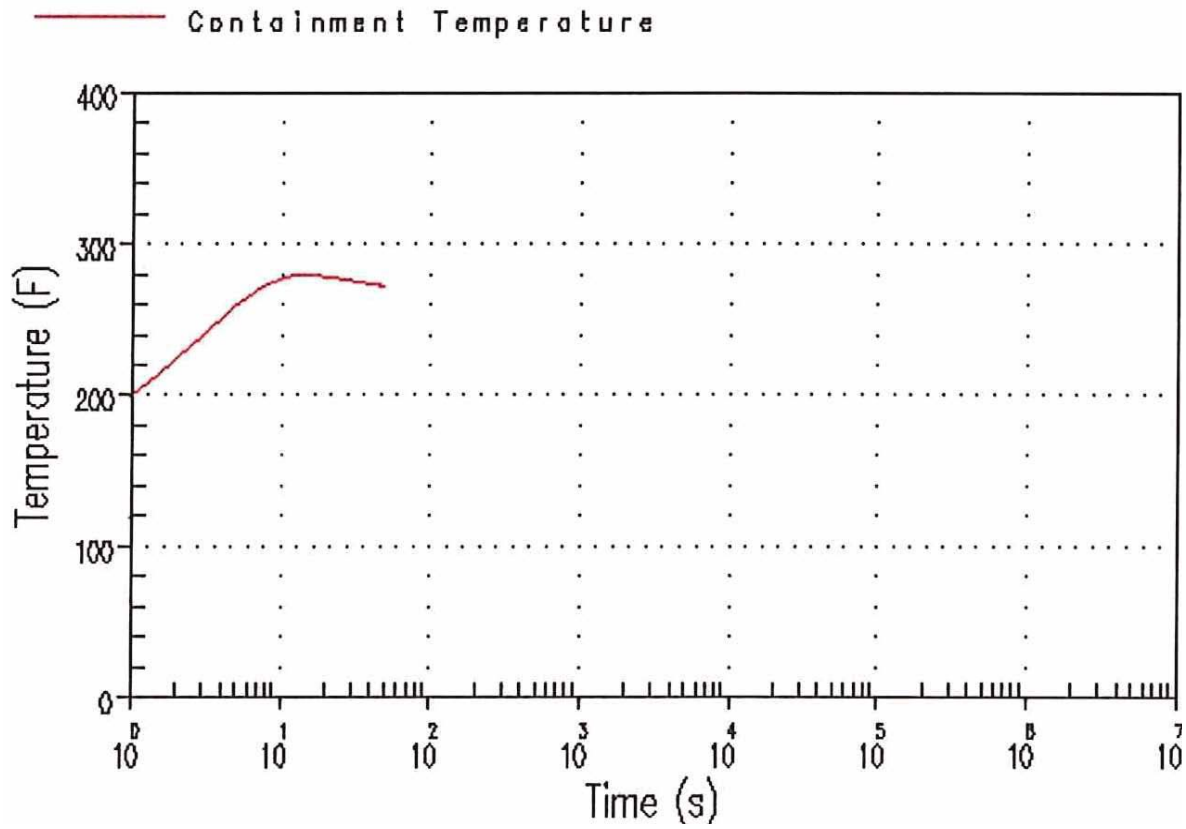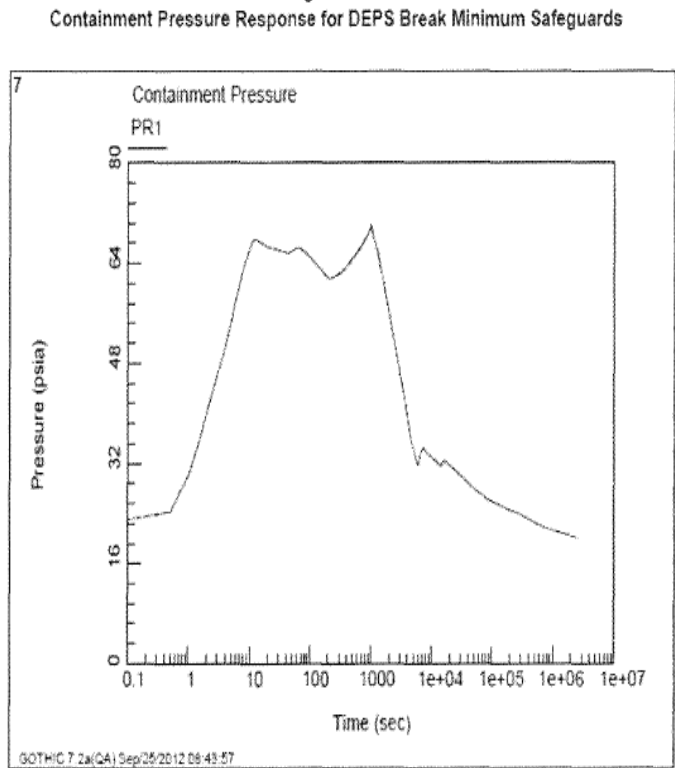BREAK (MINIMUM SAFEGUARDS)

Figure 14.3.4-4 CONTAINMENT TEMPERATURE - DOUBLE-ENDED PUMP SUCTION
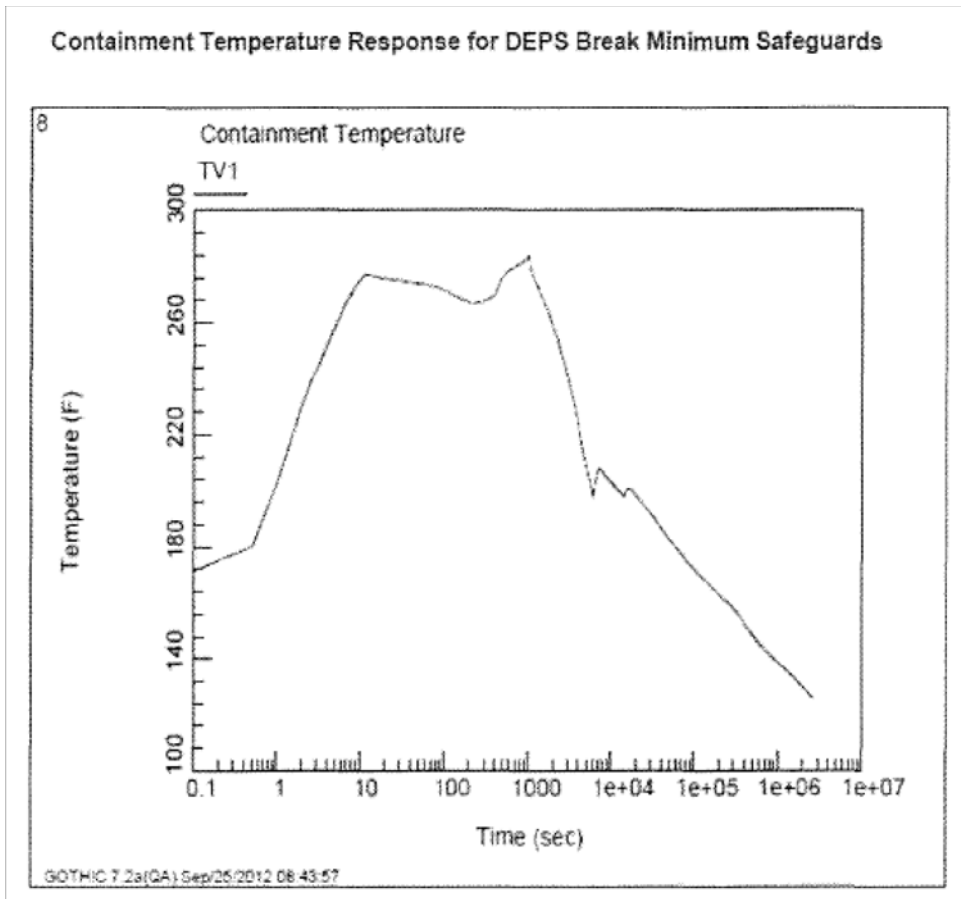BREAK (MINIMUM SAFEGUARDS)

Figure 14.3.4-5 CONTAINMENT PRESSURE - DOUBLE-ENDED PUMP SUCTION
BREAK (MAXIMUM SAFEGUARDS)
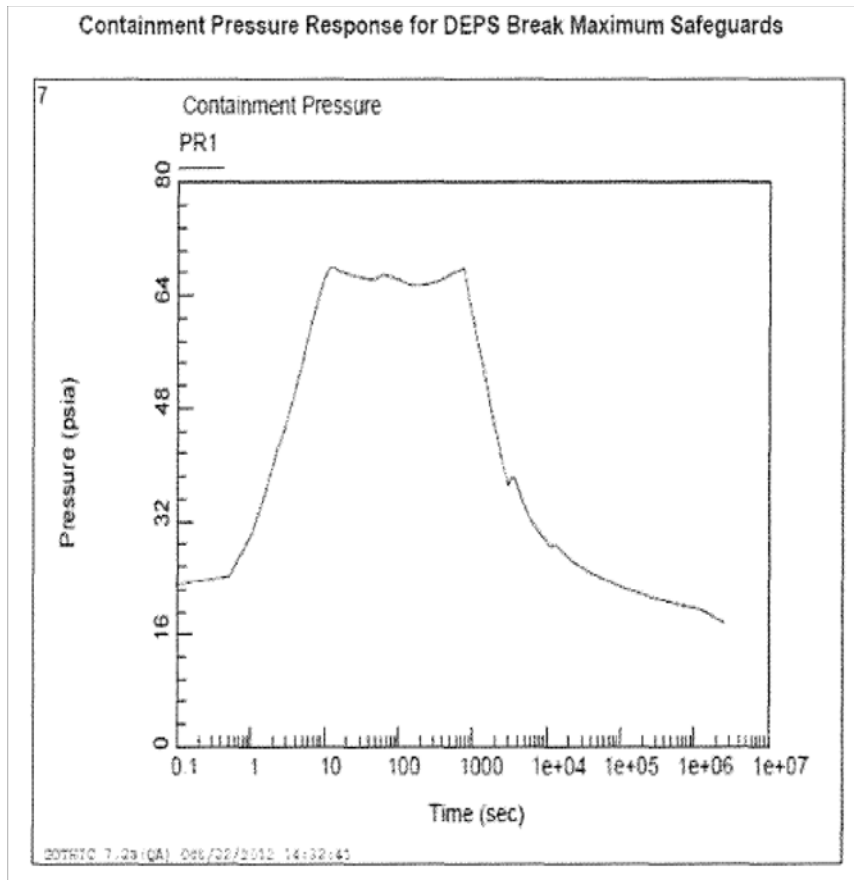
Figure 14.3.4-6   CONTAINMENT TEMPERATURE - DOUBLE-ENDED PUMP SUCTION
                  BREAK (MAXIMUM SAFEGUARDS)
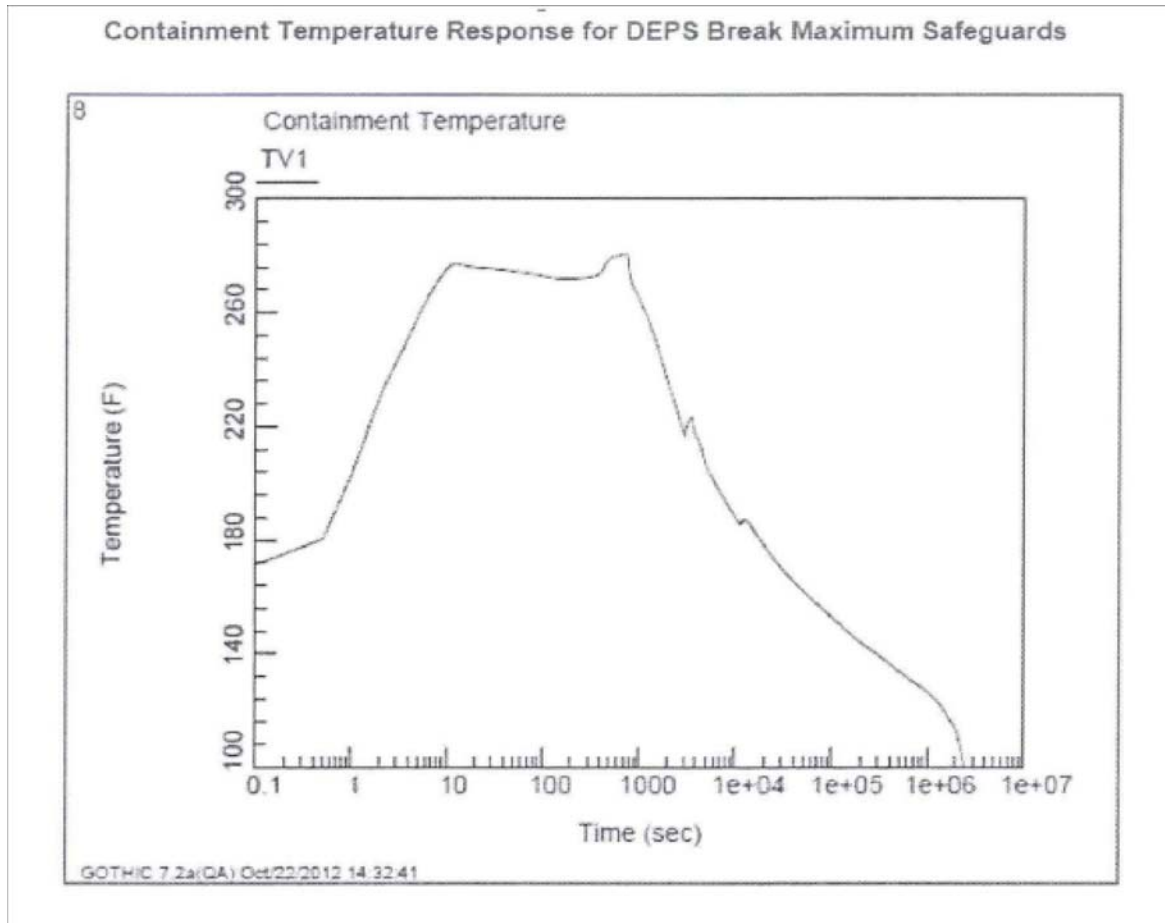
14.3.5   RADIOLOGICAL CONSEQUENCES OF LOSS-OF-COOLANT ACCIDENT

The results of analyses presented in this section demonstrate that the amounts of radioactivity released to the environment in the event of a loss-of-coolant accident result in calculated offsite radiological doses that do not exceed the limits specified in 10 CFR 50.67. The calculated doses are summarized in Table 14.3.5-6.

Basic Events and Release Fractions

There are two release pathways considered in this analysis: (1) radioactivity which enters containment from the reactor core and is released due to containment leakage, and (2) radioactivity which is released to the environment via ECCS equipment leakage.

The event causing the postulated releases is a double-ended rupture of a reactor coolant pipe, with subsequent blowdown, as described in Section . As demonstrated by the analysis in Section 14.3.2, the emergency core cooling system, using emergency power, keeps cladding temperatures well below melting and limits zirconium - water reactions to an insignificant level, assuring that the core remains intact and in place. As a result of the increase in cladding temperature and the rapid depressurization of the core, however, some cladding failure may occur in the hottest regions of the core. For analysis purposes, the entire core is assumed to fail. The release of activity from the core occurs over a 1.8 hour interval. The gap release phase occurs in the first half hour and the release from the melted fuel occurs over the next 1.3 hours. A wide spectrum of nuclides is taken into consideration. Table 14.3.5-1 lists the nuclides being considered for the LOCA with core melt.

Containment Vessel Inventory and Release Rate

Consistent with Regulatory Guide 1.183 (Reference 4), 95 percent of the radioiodine released to the containment is assumed to be cesium iodide (CsI), 4.85 percent is elemental iodine, and 0.15 percent is organic iodide. This includes releases from the gap and the fuel pellets. With the exception of elemental and organic iodine and noble gases, fission products are assumed to be in particulate form.

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.

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 and all activity is assumed to be released into the unsprayed volume. The containment volume is 1.0E6 ft3 with a sprayed fraction of 58.2 percent of the total (5.82E5 ft3).

The containment is assumed to leak at the design leak rate of 0.2 weight percent per day for the first 24 hours of the accident and then to leak at half that rate (0.1 weight percent per day) for the remainder of the 30 day period following the accident considered in the analysis.

Removal of Activity from the Containment Atmosphere

The reduction of activity available for release to the environment depends on the chemical form. The removal of elemental iodine from the containment atmosphere is accomplished only by containment sprays and radioactive decay. The removal of particulates from the containment atmosphere is accomplished by containment sprays, sedimentation and radioactive decay. The noble gases and the organic iodine are subject to removal only by radioactive decay.

One train of the containment spray system is assumed to operate in the injection mode following the LOCA. When the RWST drains to a predetermined level, the operators switch to recirculation of the sump liquid to provide a source to the sprays. The minimum injection spray duration until the level is reached is 40 minutes. The switchover is assumed to take 20 minutes. During these 20 minutes, the analysis does not credit any spray removal in the containment. The analysis assumed that the recirculation sprays operate for a 2-hour duration.

Containment Spray Removal of Elemental Iodine

The Standard Review Plan (SRP) Section 6.5.2 (Reference 5) identifies a methodology for the determination of spray removal of elemental iodine independent of the use of spray additive. The upper limit of the removal coefficient was specified as 20 hr-1 for this model. For PBNP the calculated elemental spray removal coefficients were higher than the upper limit of 20 hr-1, therefore the upper limit of 20 hr-1 was conservatively used. When sprays are operating in the recirculation phase the elemental removal coefficient is reduced to 9.20 hr-1 to address the loading of the recirculating solution with elemental iodine.

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 decontamination factor or DF of 200). With the RG 1.183 source term methodology, this is considered 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 at 2.71 hours.

Containment Spray Removal of Particulates

Particulate spray removal is determined using the model described in SRP Section 6.5.2 (Reference 5). For PBNP the calculated particulate spray removal coefficient is 4.42 hr-1 during injection and 3.72 hr-1 during recirculation. Following the model in SRP Section 6.5.2, these coefficients are applied until the time when the inventory in the containment is reduced to 2 percent of its original amount (DF of 50), at which time they are reduced by a factor of 10. With the RG 1.183 source term methodology, this is considered as being 2 percent of the total inventory of particulate iodine that is released to the containment atmosphere over the duration of gap and in-vessel release phases. In the analysis, the DF of 50 is not achieved prior to the termination of containment sprays.

Sedimentation Removal of Particulates

During spray operation, credit is taken for sedimentation removal of particulates in the unsprayed region. After sprays are terminated (and during the 20 minute switchover from injection to recirculation when sprays are not credited), 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. It is also conservatively assumed that sedimentation removal does not continue beyond a DF of 1000. A DF of 1000 is reached at 38.11 hours.

ECCS Equipment Iodine Inventory and Leakage Rate

When Emergency Core Cooling System (ECCS) recirculation is established following the LOCA, leakage is assumed to occur from ECCS equipment located outside containment. It is also assumed that all of the iodine released from the core is in the sump water being recirculated. Hence, the ECCS equipment leakage results in the release of a significant amount of iodine activity to the outside environment. For this activity release path, no credit is taken for plateout of elemental iodine on containment surfaces or for iodine removal by the atmosphere filtration system in the primary auxiliary building (PAB). The iodine release from this path is conservatively assumed to be 97% elemental and 3% organic.

Only iodine is released through this pathway since the noble gases are not assumed to dissolve in the sump and particulates would remain in the water of the ECCS leakage. It is assumed that the iodine is instantaneously and homogeneously mixed in the primary containment sump water at the time of release from the core. In the calculation of the dose resulting from ECCS leakage recirculation is conservatively initiated at 0 minutes. The leakage continues for the 30 day period following the accident considered in the analysis.

There are two pathways considered for the ECCS recirculation leakage. One is the leakage directly into the PAB and the other is back-leakage into the refueling water storage tank (RWST).

The total ECCS recirculation leakage modeled in the analysis is 800 cc/min. (Consistent with RG 1.183 guidance this includes a factor of two increase over the postulated leak rates based on the historical data for ECCS leakage collected from the PBNP Leakage Reduction and Preventive Maintenance Program). Of the 800 cc/min total ECCS recirculation leakage, 300 cc/min is assumed to leak into the PAB, and 500 cc/min is assumed to leak back to the RWST.

Leakage to the PAB

The analysis models a total ECCS recirculation leakage into the PAB of 300 cc/min. Instead of applying a 10% iodine airborne fraction as discussed in RG 1.183, the analysis applies iodine airborne fractions based on the calculated flashing fraction of ECCS recirculation leakage. The flashing fractions are developed using the calculated sump temperature and the constant enthalpy flashing fraction equation provided in RG 1.183. The maximum flashing fraction is calculated for several time intervals and a bounding airborne fraction is selected for use in the analysis. Once the sump temperature is less than 212°F, a constant airborne fraction of 2% is maintained for the duration of the event. The iodine airborne fractions are modeled as listed in Table 14.3.5-5.

Leakage to the RWST

ECCS back-leakage to the RWST is assumed at a rate of 500 cc/min. 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 conservatively assumed to be 3500 ppm. The initial pH of the RWST solution is determined to be approximately 4.5. The RWST water pH and iodine concentration are determined as a function of time. Figure 3.1 of NUREG-5950 (Reference 6) is used to determine the amount of iodine becoming elemental based on the pH and iodine concentration of the RWST solution. With an RWST pH of 4.5 and low iodine concentration, the fraction of conversion to elemental iodine is 2%. By 300 hours, the RWST liquid pH will exceed 5.0 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 45.4 using a relationship to solution temperature from NUREG-5950 (Reference 6). This is modeled by the transfer of a portion of the flow to the RWST liquid and a portion to the RWST gas space. The modeling of the air flow out of the RWST is based on a diurnal heating and cooling cycle. This model ignores the effect of the large heat sink provided by the mass of water in the tank that would tend to moderate the effects of the heating and cooling from atmospheric temperature variations. Temperature swings for the RWST are assumed to be the Technical Specification limits and do not result in pressurization of the RWST. No credit is assumed for evaporation. The transfer from the RWST gas space to the environment is calculated to be 2.71 cfm based on displacement by the in-leakage and air expansion from the heating/cooling cycle.

Meteorological Data

Five years of hourly onsite meteorological data collected between September 2000 and September 2005 were used to generate new CR air intake atmospheric dispersion factors (χ/Q values) for the Alternate Source term (AST) license amendment request (LAR). Wind speed and wind direction were measured at the 45 and 10 meter levels and the atmospheric stability categorization was based on temperature difference measurements between these two levels. The measurements were primarily from the primary tower located about 40 meters inland of the Lake Michigan shoreline.

The NRC reviewed available information relative to the onsite meteorological measurements program, the 2000 through 2005 meteorological data measured at the PBNP site, and the ARCON96 meteorological data input files provided to them. Based on this review, the NRC staff concluded that the data provides an acceptable basis for making estimates of atmospheric dispersion for the DBA control room dose assessments associated with the AST LAR.

Control Room Atmospheric Dispersion Factors

Releases from the following locations to the control room air intake were postulated:

Auxiliary Building Vent
Drumming Area Vent
Spent Fuel Pool
Unit 1 and Unit 2 Containment Wall
Unit 1 and Unit 2 Containment Facade
Unit 1 "A and Unit 1" B Main Steam Safety Valves (MSSVs)
Unit 2 "A" and Unit 2 "B" MSSVs
Units 1 and 2 Purge Stacks
Units 1 and 2 Refueling Water Storage Tanks (RWSTs)

χ/Q values were generated using the ARCON96 computer code and guidance provided in RG 1.194. All sources were modeled as ground level releases based upon guidance provided in RG 1.194. The shortest horizontal distance between each release location and the control room intake was input as the distance between the release and receptor locations. Releases from the Containment wall were modeled as diffuse sources. All other releases were modeled as point sources.

All potential release scenarios for the DBAs associated with the AST LAR were considered, including those due to single failures. The resultant χ/Q values were compared and the χ/Q values associated with releases from the Unit 2 containment wall, Auxiliary Building Vent and Unit 2 RWST were found to be limiting for the LOCA:

Subsequently a loss-of-coolant accident (LOCA) control room dose analysis was performed without credit for the Primary Auxiliary Building Ventilation System (VNPAB). This included a revision to the control room χ/Q values for the assessment considering Emergency Core Cooling System (ECCS) leakage to the Primary Auxiliary Building (PAB), by taking no credit for the VNPAB, i.e., χ/Q values were calculated for a postulated release from the PBNP Unit 2 facade roof instead of the PAB vent stack. χ/Q values are listed in Table 14.3.5-2.

Offsite Atmospheric Dispersion Factors

The EAB and EPZ χ/Q values listed in Table 14.3.5-2 did not require any changes due to the AST LAR. These χ/Q values were generated based upon guidance in RG 1.145, "Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants," using onsite data collected from 1991 through 1993. These χ/Q values were assessed against χ/Q values calculated using the PAVAN atmospheric dispersion computer code (NUREG/CR-2858, "PAVAN: An Atmospheric Dispersion Program for Evaluating Design Basis Accidental Releases of Radiological Materials from Nuclear Power Stations") and the onsite meteorological data collected between September 2000 and September 2005 and found to be conservative.

Control Room Doses

Control room modeling assumptions related to the calculation of control room dose from activity that enters the control room are described in Table 14.3.5-3.

The dose contribution in the CR due to direct shine from the external cloud and from contained sources is 0.32 rem. The external cloud contribution includes containment leakage, ECCS leakage, and RWST back-leakage. The contained sources include shine from the containment structure and the control room charcoal and HEPA ventilation filters. The analysis takes credit for control room walls and ceiling and shielding modifications to the control room envelope done per Engineering Change (EC) 11691 (258119) (Reference 3). The analysis assumed an operator located 5 feet from the control room east window (Reference 9).

Computer code SW-QADCGGP was used to calculate the direct shine dose to an operator in the control room from the airborne source inside containment, external plume source, and the control room charcoal and HEPA filter sources. SW-QADCGGP is a Shaw S&W version of the industry standard point-kernel radiation shielding computer code QAD-CGGP (Reference 2). Stone & Webster computer program PERC 2 was used to calculate the radiation source term in post-LOCA containment atmosphere, in the external plume passing the control room due to containment and ECCS leakage, and in the control room emergency filters due to containment and ECCS leakage.

Results and Conclusions

The major assumptions and parameters used in the LOCA dose analysis are itemized in Table 14.3.5-1 through Table 14.3.5-5.

The results of the offsite and control room dose analyses are provided in Table 14.3.5-6, and indicate that the acceptance criteria are met (Reference 8, Reference 9 and Reference 13). The exclusion area boundary doses reported are for the worst 2 hour period, determined to be from 0.5 to 2.5 hours.

Technical Support Center (TSC)

The TSC is required to meet the same post-accident radiological habitability criteria as the control room, i.e. less than 5 rem TEDE. Calculations confirm that the 30-day post-accident doses for the TSC following extended power uprate using alternate source term methodology is less than 5 rem TEDE (Reference 11 and Reference 12).

REFERENCES

1.   Eckerman, Keith F., Wolbarst, Anthony B., and Richardson, Allan C.B., Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion, Federal Guidance Report Number 11, EPA-520/1-88-020, September 1988.

2.   QAD-CGGP, "A Combinatorial Geometry Version of QAD-P5A, A Point Kernel Code System for Neutron and Gamma-Ray Shielding Calculations Using the GP Buildup Factor."

3. Engineering Change EC 11691 (258119), Revision 3, "Addition of Control Room Shielding," July 12, 2011.

4. USNRC, Regulatory Guide 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," July 2000.

5. NUREG-0800 Standard Review Plan, Section 6.5.2 entitled "Containment Spray as a Fission Product Cleanup System," Revision 4, March 2007.

6. NUREG/CR-5950, "Iodine Evolution and pH Control," E. C. Beahm, et al, December 1992.

7. K. F. Eckerman and J. C. Ryman, "External Exposure to Radionuclides in Air, Water, and Soil," Federal Guidance Report 12, EPA-402-R-93-081, 1993.

8. Calculation CN-CRA-08-21, Revision 1, Point Beach LOCA Doses for the Extended Power Uprate, Approved April 22, 2011.

9. Calculation 129187-M-0105, Revision 1, Control Room Direct Shine Dose Due to a Loss of Coolant Accident Following Extended Power Uprate and Using Alternative Source Term Methodology, Approved April 28, 2011.

10. NRC Safety Evaluation, PBNP Units 1 and 2-Issuance of License Amendments Regarding use of Alternate Source Term, dated April 14, 2011.

11. CN-CRA-10-53, Revision 1, Point Beach TSC Doses for LOCA and SLB, approved April 22, 2011.

12. Calculation 129187-M-0112, Revision 1, Technical Support Center Direct Shine Dose due to a Loss-of-Coolant Accident Following Extended Power Uprate and using Alternative Source Term Methodology, approved May 18, 2011.

13. SCR 2011-0275, Revise FSAR 14.3.5, Radiological Consequences of a LOCA for EPU per AR 1688483.

Table 14.3.5-1   CORE ACTIVITIES[1]

Core Total Nuclide Activities at Shutdown

Isotope	Activity (Ci)	Isotope	Activity (Ci)
GROUP 1 - Noble Gases		GROUP 6 - Barium	
Kr-85	6.15E+05	Ba-139	9.42E+07
Kr-85m	1.36E+07	Ba-140	9.05E+07
Kr-87	2.68E+07		
Kr-88	3.60E+07	GROUP 7 - Noble Metals	
Xe-131m	5.55E+05	Ru-103	7.79E+07
Xe-133	1.02E+08	Ru-105	5.42E+07
Xe-133m	3.21E+06	Ru-106	2.54E+07
Xe-135	2.17E+07	Rh-105	5.08E+07
Xe-135m	2.20E+07	Te-99m	8.47E+07
Xe-138	9.05E+07	Mo-99	9.62E+07
GROUP 2 - Halogens		GROUP 8 - Cerium	
I-130	1.05E+06	Ce-141	8.52E+07
I-131	5.10E+07	Ce-143	8.03E+07
I-132	7.47E+07	Ce-144	6.72E+07
I-133	1.06E+08	Pu-238	1.33E+05
I-134	1.19E+08	Pu-239	1.45E+04
I-135	1.01E+08	Pu-240	2.25E+04
		Pu-241	5.73E+06
GROUP 3 - Alkali Metals (Rb / Cs)		Np-239	9.65E+08
Rb-86	9.95E+04		
Cs-134	9.52E+06	GROUP 9 - Lanthanides	
Cs-136	2.14E+06	Y-90	5.01E+06
Cs-137	6.27E+06	Y-91	6.56E+07
Cs-138	9.89E+07	Y-92	6.82E+07
		Y-93	7.67E+07
GROUP 4 - Tellurium		Nb-95	8.87E+07
Te-127	4.54E+06	Zr-95	8.76E+07
Te-127m	7.48E+05	Zr-97	8.80E+07
Te-129	1.33E+07	La-140	9.69E+07
Te-129m	2.52E+06	La-142	8.25E+07
Te-131m	9.95E+06	Pr-143	7.75E+07
Te-132	7.30E+07	Nd-147	3.33E+07
Sb-127	4.63E+06	Am-241	6.16E+03
Sb-129	1.42E+07	Cm-242	1.70E+06
		Cm-244	1.58E+05
GROUP 5 - Strontium			
Sr-89	5.03E+07		
Sr-90	4.80E+06		
Sr-91	6.30E+07		
Sr-92	6.73E+07		

1.   These core activities are based on a core power level of 1811 MWt.

Table 14.3.5-2   DOSE CONVERSION FACTORS, BREATHING RATES, AND
ATMOSPHERIC DISPERSION FACTORS

Exclusion Area Boundary[1]		
Time (hr)	Breathing Rate (m^3/sec)	Atmospheric Dispersion Factor (X/Q sec/m^3)
0-720	3.5E-4	5.0E-4

Low Population Zone		
Time (hr)	Breathing Rate (m^3/sec)	Atmospheric Dispersion Factor (X/Q sec/m^3)
0-8	3.5E-4	3.0E-5
8-24	1.8E-4	1.6E-5
24-96	2.3E-4	4.2E-6
96-720	2.3E-4	8.6E-7

Control Room2			
Time (hr)	Containment Leakage Atmospheric Dispersion Factor (X/Q sec/m^3)	ECCS Leakage (PAB) Atmospheric Dispersion Factor (X/Q sec/m^3)	ECCS Leakage (RWST) Atmospheric Dispersion Factor (X/Q sec/m^3)
0 - 2 hr	1.39E-3	6.78E-3	9.89E-3
2 - 8 hr	9.80E-4	5.03E-3	7.98E-3
8 - 24 hr	3.84E-4	1.72E-3	2.88E-3
24 - 96 hr	3.46E-4	1.60E-3	2.75E-3
96 - 720 hr	3.02E-4	1.34E-3	2.35E-3

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1 The breathing rate and atmospheric dispersion factors for the exclusion area boundary are held constant at the initial value for all time intervals to determine the limiting 2-hour period.

2 The control room breathing rate is assumed constant at 3.5E-4 for 0-720 hours and VNPAB is assumed out of service.

Table 14.3.5-2A COMMITTED EFFECTIVE DOSE EQUIVALENT DOSE CONVERSION FACTORS (Page 1 of 2)

Committed Effective Dose Equivalent Dose Conversion Factors (Reference 1)

Isotope	DCF (Sv/Bq)	Isotope	DCF (Sv/Bq)
I-130	7.14E-10	Cs-134	1.25E-08
I-131	8.89E-09	Cs-136	1.98E-09
I-132	1.03E-10	Cs-137	8.63E-09
I-133	1.58E-09	Cs-138	2.74E-11
I-134	3.55E-11	Rb-86	1.79E-09
I-135	3.32E-10	Ru-103	2.42E-09
Kr-85m	N/A	Ru-105	1.23E-10
Kr-85	N/A	Ru-106	1.29E-07
Kr-87	N/A	Rh-105	2.58E-10
Kr-88	N/A	Mo-99	1.07E-09
Xe-131m	N/A	Tc-99m	8.80E-12
Xe-133m	N/A	Y-90	2.28E-09
Xe-133	N/A	Y-91	1.32E-08
Xe-135m	N/A	Y-92	2.11E-10
Xe-135	N/A	Y-93	5.82E-10
Xe-138	N/A	Nb-95	1.57E-09
Te-127	8.60E-11	Zr-95	6.39E-09
Te-127m	5.81E-09	Zr-97	1.17E-09
Te-129m	6.47E-09	La-140	1.31E-09
Te-129	2.42E-11	La-142	6.84E-11
Te-131m	1.73E-09	Nd-147	1.85E-09
Te-132	2.55E-09	Pr-143	2.19E-09
Sb-127	1.63E-09	Am-241	1.20E-04
Sb-129	1.74E-10	Cm-242	4.67E-06
Ce-141	2.42E-09	Cm-244	6.70E-05
Ce-143	9.16E-10	Sr-89	1.12E-08

Table 14.3.5-2A COMMITTED EFFECTIVE DOSE EQUIVALENT DOSE CONVERSION FACTORS (Page 2 of 2)

Committed Effective Dose Equivalent Dose Conversion Factors (Reference 1)

Isotope	DCF (Sv/Bq)	Isotope	DCF (Sv/Bq)
Ce-144	1.01E-07	Sr-90	3.51E-07
Pu-238	1.06E-04	Sr-91	4.49E-10
Pu-239	1.16E-04	Sr-92	2.18E-10
Pu-240	1.16E-04	Ba-139	4.64E-11
Pu-241	2.23E-06	Ba-140	1.01E-09
Np-239	6.78E-10		

Table 14.3.5-2B EFFECTIVE DOSE EQUIVALENT DOSE CONVERSION FACTORS
(Page 1 of 2)

Effective Dose Equivalent Dose Conversion Factors (Reference 7)

Isotope	DCF (Sv m3/Bq sec)	Isotope	DCF (Sv m3/Bq sec)
I-130	1.04E-13	Cs-134	7.57E-14
I-131	1.82E-14	Cs-136	1.06E-13
I-132	1.12E-13	Cs-137	2.88E-14
I-133	2.94E-14	Cs-138	1.21E-13
I-134	1.30E-13	Rb-86	4.81E-15
I-135	7.98E-14	Ru-103	2.25E-14
Kr-85m	7.48E-15	Ru-105	3.81E-14
Kr-85	1.19E-16	Ru-106	0.0
Kr-87	4.12E-14	Rh-105	3.72E-15
Kr-88	1.02E-13	Mo-99	7.28E-15
Xe-131m	3.89E-16	Tc-99m	5.89E-15
Xe-133m	1.37E-15	Y-90	1.90E-16
Xe-133	1.56E-15	Y-91	2.60E-16
Xe-135m	2.04E-14	Y-92	1.30E-14
Xe-135	1.19E-14	Y-93	4.80E-15
Xe-138	5.77E-14	Nb-95	3.74E-14
Te-127	2.42E-16	Zr-95	3.60E-14
Te-127m	1.47E-16	Zr-97	9.02E-15
Te-129m	1.55E-15	La-140	1.17E-13
Te-129	2.75E-15	La-142	1.44E-13
Te-131m	7.01E-14	Nd-147	6.19E-15
Te-132	1.03E-14	Pr-143	2.10E-17
Sb-127	3.33E-14	Am-241	8.18E-16
Sb-129	7.14E-14	Cm-242	5.69E-18
Ce-141	3.43E-15	Cm-244	4.91E-18
Ce-143	1.29E-14	Sr-89	7.73E-17

Table 14.3.5-2B COMMITTED EFFECTIVE DOSE EQUIVALENT DOSE CONVERSION
FACTORS (Page 2 of 2)

Committed Effective Dose Equivalent Dose Conversion Factors (Reference 7)

Isotope	DCF (Sv/Bq)	Isotope	DCF (Sv/Bq)
Ce-144	8.53E-16	Sr-90	7.53E-18
Pu-238	4.88E-18	Sr-91	3.45E-14
Pu-239	4.24E-18	Sr-92	6.79E-14
Pu-240	4.75E-18	Ba-139	2.17E-15
Pu-241	7.25E-20	Ba-140	8.58E-15
Np-239	7.69E-15		

Table 14.3.5-3   CONTROL ROOM PARAMETERS

Volume	65,243 ft^3
Unfiltered Inleakage	200 cfm
Normal Ventilation Flow Rates VNCR (Mode 1)	
Filtered Makeup Flow Rate	0 cfm
Filtered Recirculation Flow Rate	0 cfm
Unfiltered Makeup Flow Rate	2000 cfm
Emergency Mode Flow Rates VNCR (Mode 5)	
Filtered Makeup Flow Rate	2500 cfm
Filtered Recirculation Flow Rate	1955 cfm
Unfiltered Makeup Flow Rate	0 cfm
Filter Efficiency	
Elemental	95%
Organic (Methyl)	95%
Particulate	99%
Occupancy Factors	
0 - 24 hours	1.0
24 - 96 hours	0.6
4 - 30 days	0.4
Assumed time following SI signal to switch from normal to emergency filtration mode.	60 seconds

Table 14.3.5-4   ASSUMPTIONS USED FOR LARGE BREAK LOCA DOSE ANALYSIS
CONTAINMENT LEAKAGE

Core Power (including uncertainties)	1811 MWt	
Core Activity	See Table 14.3.5-1	
Activity release fractions	Gap	Core
Nobel gases	0.05	0.95
Iodines	0.05	0.35
Alkali Metals	0.05	0.25
Tellurium	0.	0.05
Strontium, Barium	0.	0.02
Nobel Metals	0.	0.0025
Cerium	0.	0.0005
Lanthanides	0.	0.0002
Iodine chemical form in containment		
Elemental	4.85%	
Organic (methyl)	0.15%	
Particulate (cesium iodide)	95%	
Containment net free volume	1.0E6 ft3	
Containment sprayed volume	5.82E5 ft3	
Fan Coolers		
Number in operation	2	
Flow rate (per unit)	33,500 cfm	
Containment leak rates		
0 - 24 hours	0.2 weight %/day	
> 24 hours	0.1 weight %/day	
Spray operation		
Time to initiate injection sprays	90 seconds	
Time that injection sprays are terminated	40 minutes	
Delay time for switchover to recirculation sprays	20 minutes	
Recirculation spray duration	2 hours	
Spray flow rates		
Injection	1,070 gpm	
Recirculation	900 gpm	
Spray fall height	65.58 ft	
Containment Spray Removal Coefficients		
Spray elemental iodine removal		
Injection	20 hr-1	
Recirculation	9.20 hr-1	
Spray particulate removal*		
Injection	4.42 hr-1	
Recirculation	3.72 hr-1	
Sedimentation particulate removal	0.1 hr-1	
(Unsprayed region: from start of event, Sprayed region: when		
sprays are not assumed to be operating)		
Containment Spray DF		
Elemental	200	
Particulate	1000	

* These coefficients are applicable until the time when the inventory in the containment is reduced to 2-percent of its original amount
(DF of 50) at which time they would be reduced by a factor of 10.

Table 14.3.5-5   ASSUMPTIONS USED FOR LARGE BREAK LOCA DOSE ANALYSIS
ECCS EQUIPMENT LEAKAGE

Core Power		1811 MWt
Core Activity		See Table 14.3.5-1

Activity release fractions	Gap	Core
Iodines	0.05	0.35

Containment Sump Volume	2.43E+05 gal
RWST Minimum Water Volume	25,500 gal
RWST Maximum Air Volume	270,000 gal
RWST Minimum Temperature	40°F
RWST Maximum Temperature	100°F
Time to Initiate ECCS Recirculation	0 min

ECCS Leak Rate

PAB Leak Rate	300 cc/min
RWST Leak Rate	500 cc/min

ECCS Leakage to PAB Iodine Airborne fraction	Time (sec)	Airborne Fraction
	0 - 2700	7%
	2700 - 3600	5%
	3600 - 4500	4%
	4500 - 6300	3%
	> 6300	2%

Transfer from the RWST gas space to the environment	2.71 cfm

Iodine chemical from released to atmosphere	
Elemental	97%
Organic (methyl)	3%
Particulate (cesium iodide)	0%

Table 14.3.5-6   LARGE BREAK OFFSITE AND CONTROL ROOM DOSES

	Dose (Rem TEDE)	Dose Limits (Rem TEDE)
Exclusion Area Boundary (0.5 - 2.5 hours)	14.0	25.0
Low Population Zone (0 - 30 days)	1.4	25.0
Control Room (0 - 30 days)		
All Pathways (excluded shine)	4.4	
Shine	0.32	
Total Dose	4.72	5.0

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14.3.6   REACTOR VESSEL HEAD DROP EVENT

PBNP committed to incorporate an analysis of the Reactor Vessel Head (RVH) drop into the PBNP FSAR by letter NRC 2005-0094, dated July 24, 2005 (Reference 2). The analyses presented in this section demonstrate that a limiting postulated RVH drop will not result in rupture of the RCS and associated pressure boundaries, that the core will remain covered, and core cooling remains available.

To resolve questions pertaining to a postulated RVH drop event initiated as a result of the Unit 2 reactor head replacement in 2005, analyses were performed and submitted for NRC review and approval. The analyses performed included two structural analyses that evaluated the effect of the impact on the impact load path through the reactor vessel, RCS piping, and the reactor vessel supporting structures. Included were radiological analyses predicated on an assumption that the impact would result in a clad gap release, and a presumptive failure of the bottom mounted instrumentation (BMI) conduits located beneath the reactor vessel. Reference 1 is the Safety Evaluation (SE) documenting NRC acceptance of those analyses and is applicable to both units.

Subsequent analyses performed in accordance with later approved NRC methods, and utilizing the previously performed structural analyses, found that the BMI conduits would remain intact, and that the previous presumption of a clad gap release was not necessary. The maintaining of core cooling capability with normal decay heat removal, and the removal of the assumed clad gap release permitted the elimination of most of the additional regulatory commitments associated with the Reactor Vessel Head Drop Event (Reference 1 and Reference 2). Reference 7 is the Safety Evaluation (SE) documenting NRC acceptance of the analyses demonstrating that the BMI conduits would remain intact, and is applicable to both units.

14.3.6.1   Initiating Event Occurrences

While the potential causes of an RVH drop event are not specified in the NRC safety evaluations or the supporting submittals, such an event can be postulated to occur from mechanical failure of the crane hoist mechanism, cable failure, or RVH lift rig failure. The main hoist of each polar crane is equipped with two independent upper travel limit switches to prevent the possibility of a "two-blocking" incident. The two independent upper travel limit devices are of different design and are activated by independent mechanical means. These devices independently de-energize either the hoist drive motor or the main power supply. Since the upper travel limit switches on the containment polar cranes are independent, are tested, and operational restrictions limit upward travel, it was established in Reference 1 that the potential for an RVH drop event due to "two-blocking" (i.e., exceeding the physical upper travel limits of the crane) is negligible. See FSAR Appendix A.3 for additional discussion on "two-blocking."

14.3.6.2   Event Frequency Classification

The initiating event in this assessment is the drop of the RVH while it is suspended over the reactor vessel. The RVH is assumed to fall onto the reactor vessel flange, resulting in damage to the reactor vessel support structure.

NUREG-1774, "A Survey of Crane Operating Experience at U.S. Nuclear Power Plants from 1968 through 2002," (Reference 3) was written to address NRC Candidate Generic Issue 186,

"Potential Risk and Consequences of Heavy Load Drops in Nuclear Power Plants." Crane operating history from 1968 through 2002 was reviewed as part of this report to provide a risk assessment associated with lifts of Very Heavy Loads (VHL). The risk analysis included in NUREG-1774 considers VHL lifts for any crane at any operating nuclear station. The analysis considers a postulated drop of load at any point during the movement of a load from the initial lift until set-down.

The probabilistic analysis contained within NUREG-1774 is primarily concerned with the probability of a VHL drop at an operating commercial nuclear power plant. A VHL is defined as any load over 30 tons. The generic probability for any VHL drop is given as 5.6E-5 per lift. This value is based upon three (3) drops per 54,000 VHL lifts.

Reference 1 established that a postulated RVH drop meets the frequency classification of an infrequent incident (i.e., an incident that may occur during the lifetime of the plant).

14.3.6.3   Sequence of Events

The analyzed event is a concentric drop of the RVH onto the reactor vessel flange from a height of 26.4 feet. This was determined to impart the maximum credible impact loads on the reactor vessel and supporting structures. The resultant impact displaces the reactor vessel downward. Downward movement of the vessel creates the potential for damage to piping and tubing directly or indirectly connected to the reactor vessel, thereby creating a potential for a decrease in reactor coolant inventory.

Upon impact with the vessel flange, the kinetic energy of the vessel head is partially dissipated and partially transferred to both the head (rebound) and the vessel through an elastic/plastic collision. The impact forces, if high enough, can lead to yielding of the vessel supporting structures and/or attached piping.

After the head and vessel have come to rest, decay heat removal can be maintained by one or both RHR trains. Damage to the point of rupture or shearing of other connected piping, including the main RCS loops, pressurizer surge line, core deluge lines, accumulator dump lines, normal charging, BMI conduits, and cold leg SI Lines, etc. are not expected.

14.3.6.4   Plant Characteristics Considered in the Safety Evaluation

To demonstrate the capability of the reactor vessel, RCS, and supporting systems and structures to sustain a postulated RVH drop event, two complementary inelastic structure and piping system analyses were performed (Reference 5 and Reference 6). A RVH drop is postulated to occur during refueling when the head is manipulated above the reactor vessel. The RVH is assumed to fall concentrically onto the reactor vessel. Established administrative controls limit the maximum RVH drop height to 26.4 feet. This drop height has been utilized in the analyses discussed below. Both analyses were performed prior to NRC issuance of (but consistent with) Reference 8 which established approved methods for analyses of postulated RVH drop events.

The Sargent & Lundy (S&L) analysis (Reference 5) evaluated the reactor vessel and vessel support behaviors using a finite element model. The Westinghouse analysis (Reference 6) evaluated the plastic deformation that may occur to connected RCS piping based on specified bounding reactor vessel displacements.

S&L Finite Element Analysis

This analysis considers a flat vertical impact of the new RVH, using weights of 200,000 lbs for Unit 1 and 194,000 lbs for Unit 2, dropping from a height of 26.4 feet onto the reactor vessel flange. This analysis also includes an evaluation of the structural integrity of supporting elements in the load path, and predicts the vertical downward displacement of the reactor vessel.

The load path consists of the reactor vessel, reactor vessel supports at the four RCS nozzles and two brackets under the RHR core deluge nozzles, the support girder box frame, and the six pipe columns and their supports, which rest on the concrete foundation. The reactor coolant system (RCS) piping provides additional stiffness to the reactor vessel nozzles under vertical impact loading, and also transfers a portion of the impact load to the steam generator (SG) and the reactor coolant pump (RCP) support structures under a postulated RVH scenario. The concrete and embedded reinforcing bar located between the support girder and the concrete foundation under the support columns is not considered to provide any vertical support, even if the predicted deflection of the vessel could result in contacting the concrete.

The analysis models used are static analysis models for stiffness calculations of various components and substructures, and a dynamic impact model. The finite element analyses are performed using the ANSYS computer code.

The static analysis models include:

(1)   A detailed model of reactor vessel flange and reactor vessel shell below the flange, including a nozzle resting on a supporting shoe.

(2)   A similar detailed model of reactor vessel flange and reactor vessel shell below the flange with a support bracket resting on a supporting shoe.

(3)   A detailed model of the hexagonal girder box frame supported by six pipe columns at the vertices.

(4)   Piping models for the RCS hot legs and cold legs.

These models are used to construct static load-displacement diagrams for all steel components that are within the impact load path. Static vertical displacement is applied to the components uniformly and a reaction force is calculated to construct the force-displacement diagram of the affected components. In the static analysis, non-linear material properties are modeled with a strength increase factor of 10 percent to account for the strain rate effects due to the dynamic impact. The large deformation analysis option was selected to account for potential buckling and yielding in the structural components along the impact load path.

The results of the static analysis are used as part of the input for dynamic analysis. In calculating the stiffness of RCS hot leg or cold leg, two bounding cases are analyzed:

1)   A fixed boundary condition is used at either the SG location or the RCP location.
2)   A pinned boundary condition is used at the SG location or the RCP location.

In both cases, the pipe axial movement is released to account for the potential horizontal movement of the SG or the RCP.

The dynamic impact model consists of a two-mass model with springs and dash-pot in a vertical configuration. The top mass represents the falling head, and the bottom mass represents the target reactor vessel model supported by various springs, which represent the stiffness of the nozzle/bracket support, the girder box frame/column supports, and the RCS piping.

In the dynamic impact analysis, an impact damping of 5% of the critical damping is used. This assumption is judged to be reasonable for this application in consideration of:

1)   energy loss due to plastic damage at the impact surface between the RVH and the reactor vessel flange;
2)   energy loss due to imparted damage to six lateral supports for the hexagonal girder box frame; and,
3)   energy loss due to local damage to the liner and concrete crushing at the top of the six support columns.

Results of the dynamic transient analysis for Unit 2 indicate that the maximum dynamic downward displacement of the reactor vessel is 2.72 and 3.20 inches for cases 1 and 2 respectively. These displacements are both less than the 3.375" necessary before the hexagonal girder box frame would come into contact with the concrete "shelf", and this is consistent with the assumption that the concrete shelf does not provide any resistance to downward motion.

Using the limiting downward displacement of 3.2", the maximum Von Mises stress in the nozzle due to membrane plus bending is less than the ASME Boiler and Pressure Vessel Code, Section III, Appendix F allowable stresses for membrane stress intensity of 0.7 S_u. Similarly, the Von Mises stress in the reactor vessel support brackets is also less than 0.7 S_u.

The S&L analysis also evaluated the maximum impact load on the column foundation, and the capability of the concrete shelf to provide lateral support for the stability of the support columns (i.e. to limit buckling) located within the shelf and found the results acceptable.

Westinghouse Plastic Analysis of RCS Loop Piping

The evaluation consisted of a plastic analysis of the PBNP reactor coolant loop piping for a downward vertical displacement of the reactor vessel nozzles. Two displacements were analyzed: (1) a 4-inch displacement, which bounds the displacement calculated by the S&L model, and (2) a 6.5-inch displacement, which represents the maximum possible displacement of the reactor vessel nozzles before the RCS piping comes in contact with the biological shield wall.

The results of the analysis were compared to the criteria specified in the 1998 Edition of ASME Code, Section III, Appendix F, Paragraph F-1340. The criteria allow for large RCS loop piping deformations, with the intent that violations of the RCS pressure boundary do not occur.

The analysis uses an ANSYS finite element model of the hot and cold legs. The hot and cold legs are fixed at both ends (the reactor vessel nozzles and the SG or RCP nozzles). Each leg was modeled as a straight run of piping with one elbow. The hot and cold leg material properties were represented by a piece-wise linear stress-strain curve. Two sets of material properties were used to represent the upper and lower bound properties of the piping and elbow materials.

The results of the analysis indicate that the maximum calculated stress intensity in the hot and cold leg piping is within the ASME Code, Section III, Appendix F limit of 0.7 S_u for general primary membrane stress for the 4-inch reactor vessel nozzle displacement. Since the 4 inch reactor vessel nozzle displacement bounds the maximum calculated vessel displacement predicted from the S&L model, there is reasonable assurance that the pressure boundary integrity of the RCS loop piping will be maintained in the event of a postulated RVH drop.

The results also indicate that the 0.7 S_u limit is exceeded for the cold leg for a 6.5-inch vessel nozzle displacement. The maximum stress intensity was calculated in the cold leg elbow. While the calculated stress intensity exceeds the ASME Code general primary membrane stress intensity limit, it is concluded that loss of the RCS piping pressure boundary integrity would not be expected even if the vessel nozzle displaced 6.5 inches. This is because the maximum calculated stress intensity is still well below the material ultimate strength.

Analysis of Reactor Vessel Deflection

Based on the Sargent & Lundy FEA provided in Reference 5 and the Westinghouse analysis provided in Reference 6, the following bounding conditions apply:

Following the postulated RVH drop, using a conservatively estimated RVH weight of 200,000 lbs (Unit 1), the reactor vessel deflection would not exceed 3.36 inches. This calculated deflection is slightly greater than the Unit 2 calculated vessel deflection due to the conservative weight assumed and slight dimensional differences between units. RCS piping remains intact following the postulated reactor vessel deflection.

The impact of the postulated reactor vessel deflection on the attached RCS piping was assessed. This assessment was performed by Westinghouse and is documented in Reference 6. Westinghouse performed an analysis for a 4-inch deflection, which bounds the projected reactor vessel deflection. The results of the analysis show that stress values are less than the more restrictive criteria of 0.7 S_u specified in ASME Section III Appendix F. In addition, a second case to analyze a deflection value of 6.5 inches, which is equivalent to the gap that exists between the RCS piping and the shield wall, was conducted. The results of this analysis yielded stress values of greater than 0.7 S_u but did not predict failure of the RCS piping.

The combined results of the Sargent & Lundy and the Westinghouse analyses show that the damage from a RVH drop would not result in a loss of decay heat removal. Based on these results, it was concluded that adequate reactor core cooling and makeup capability would be maintained following the expected deflection of the reactor vessel from a postulated RVH drop.

Piping attached to the reactor coolant system (RCS) was not modeled or specifically analyzed for deflection and stress values as a result of the vessel deflection from a RVH drop. Based on the ability to analyze and demonstrate RCS piping acceptability for a bounding deflection of 4 inches, it was determined that the attached piping would also be acceptable. This conclusion was based upon the fact that all connections to the RCS piping are outside of the biological shield wall; thus, the deflection would be much less than the total deflection of the RCS piping.   In addition, the attached piping is of smaller diameter and is more flexible. The main connections to the RCS, credited for maintaining core cooling and makeup following a RVH drop, are the residual heat removal (RHR) lines, cold leg safety injection (SI) injection lines and charging. The RHR suction and return lines are 10-inch lines; the cold leg SI flow path is through the 10-inch SI accumulator injection line connected to the RCS.

Charging and auxiliary charging are connected through a 3-inch and 2-inch line to the RCS. The 10-inch connections are the closest connections of concern to the reactor vessel, with one exception, and would therefore experience the greatest relative deflection. The only exception is that the Unit 1 Auxiliary Charging line is 10 inches closer to the reactor vessel than the corresponding Safety Injection line on the "B" cold leg. Since the Auxiliary Charging line is a 2-inch line with greater flexibility than the 10-inch SI line, the focus was on addressing the SI lines. For Unit 1, the ratio of the distance from the reactor vessel to the steam generators or reactor coolant pumps would yield a deflection of approximately 20 percent, or less, of the total vessel deflection. For a vessel deflection of 3.36 inches, the deflection at the connection would be approximately 0.67 inches.

In Unit 1, the shortest horizontal piping run from the 10-inch connections at the cold legs to the first vertical support (which is a spring hanger), is greater than 6 feet. The shortest vertical run is approximately 10 feet (on the opposite cold leg). Both connections have horizontal offsets that decrease their stiffness in the vertical direction. The shortest horizontal run to an anchor is greater than 14 feet with an intervening vertical loop.

The RHR return line connects to the SI accumulator injection line over 22 linear feet from the B loop cold leg connection. The condition is very similar for the RHR suction line connection to the A hot leg. The distance to the closest anchor is greater than 13 feet with an intervening vertical loop containing an additional 30 feet of piping.

In each case, the total linear distance between anchors for the attached piping is greater than the worst RCS piping case, and that case was shown to be acceptable for a deflection of 4 inches. Based on this, the added flexibility of smaller diameter piping and an equivalent deflection of approximately 0.67 inches, it was determined that a detailed analysis of the connected piping was not necessary.

Additionally, the integrity of the two 6-inch core deluge lines was evaluated based on comparing the section properties and applicable pipe spans to the RCS piping. This comparison, coupled with the fact that the core deluge lines are more flexible than the RCS piping, leads to the conclusion that the integrity of the core deluge lines are bounded by the assessment for the RCS piping.

Bottom-Mounted Instrument (BMI) Tubes

Reference 9 analyzed the stresses in the BMI conduits that result from the maximum downward displacement of the reactor vessel. The analysis was performed in accordance with the NRC approved guidance of Reference 10, and concluded that $0.7 \ S_u$ would not be exceeded in any of the BMI conduits. Therefore, no loss of integrity of the BMI conduits is expected, and the RCS inventory would be retained.

Conclusion

In the event of a worst case postulated RVH drop, the RCS pressure boundary remains intact, the core remains covered, and core cooling remains available. There would be no loss of RCS inventory, and no release of fission products from the reactor core. As such, no extraordinary measures are necessary to mitigate the consequences of a postulated RVH drop. Administrative controls limit the height of a reactor vessel head lift, ensuring that any real drop is bounded by the analyses of record.

14.3.6.5   References

1.   NRC Safety Evaluation, Point Beach Nuclear Plant, Units 1 and 2 - Issuance of Amendment Re: Incorporation of Reactor Vessel Head Drop Accident Analysis Into the Final Safety Analysis Report," September 23, 2005, as revised by NRC letter "Point Beach Nuclear Plant, Units 1 and 2 - Revision to Safety Evaluation for Amendment Nos. 220 and 226 (TAC Nos. MC7650 and MC7651," dated January 12, 2006.

2.   Point Beach Letter from D. L. Koehl to NRC, "Request for Review of Heavy Load Analysis," NRC 2005-0094, July 24, 2005.

3.   NUREG-1774, "A Survey of Crane Operating Experience at U.S. Nuclear Power Plants from 1968 through 2002," July 2003.

4.   NRC Regulatory Guide 1.70, "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants," Revision 3.

5.   Sargent & Lundy Calculation 2005-06760, Rev. 3, "Analysis of Postulated Reactor Head Load Drop Onto the Reactor Vessel Flange," July 22, 2005.

6.   Westinghouse Calculation Note CN-RCDA-05-68 Rev. 2, "Plastic Analysis of Point Beach Reactor Cooling Piping for Reactor Vessel Head Drop," July 21, 2005.

7.   NRC Safety Evaluation, Point Beach Nuclear Plant, Units 1 and 2 - "Issuance of License Amendment Related to the Revision to the Reactor Vessel Head Drop Methodology (TAC Nos. ME4006 and ME4007)" dated June 1, 2011.

8.   NRC Regulatory Issue Summary (RIS) 2008-28, "Endorsement of Nuclear Energy Institute Guidance for Reactor Vessel Head Heavy Load Lifts," dated December 1, 2008.

9.   Calculation CN-MRCDA-08-51 Revision 1, "Point Beach Units 1 and 2 Evaluation of Bottom Mounted Instrumentation (BMI) Conduits for a Postulated Closure Head Assembly Drop Event," dated January 4, 2010.

10.   Nuclear Energy Institute (NEI) 08-05, "Industry Initiative on Control of Heavy Loads," Revision 0, issued July 2008.