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15.0 ACCIDENT ANALYSES

The ANS classification of plant conditions divides plant conditions into four categories in accordance with anticipated frequency of occurrence and potential radiological consequences to the public. The four categories are as follows:

Condition I: Normal Operation and Operational Transients

Condition II: Faults of Moderate Frequency

Condition III: Infrequent Faults

Condition IV: Limiting Faults

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

In the evaluation of the radiological consequences associated with initiation of a spectrum of accident conditions numerous assumptions must be postulated. In many instances these assumptions are a product of extremely conservative judgments. This is due to the fact that many physical phenomena, in particular fission product transport under accident conditions, are presently not understood to the extent that accurate predictions can be made. Therefore, the set of assumptions postulated would predominantly determine the accident classification.

This chapter addresses the accident conditions listed in Table 15-1 of the NRC Standard Format and Content Guide, Regulatory Guide 1.70, Revision 3, which apply to WBN.

15.1 CONDITION I - NORMAL OPERATION AND OPERATIONAL TRANSIENTS

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

Typical Condition I events are listed below:

(1) Steady-state and shutdown operations

(a) Power operation (>5% to 100% of full power)

(b) Startup (critical, 0% to $\leq$5% of full power)

(c) Hot shutdown (subcritical, residual heat removal system isolated)

(d) Cold shutdown (subcritical, residual heat removal system in operation)

(e) Refueling (reactor vessel head open)

(2) Operation with permissible deviations

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

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

(b) Leakage from fuel with cladding defects

(c) Radioactivity in the reactor coolant

(i) Fission products

(ii) Activation products

(iii) Tritium

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

(e) Testing as allowed by the Technical Specifications

(3) Operational transients

(a) Plant heatup and cooldown (up to 100°F/hour for the reactor coolant system; 200°F/hour for the pressurizer)

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

(c) Ramp load changes (up to 5%/minute)

(d) Load rejection up to and including design load rejection transient

15.1.1 Optimization of Control Systems

A setpoint study was performed to simulate performance of the reactor control and protection systems. In this study, emphasis was placed on the development of a control system to automatically maintain prescribed conditions in the plant even under the most conservative set of reactivity parameters with respect to both system stability and transient performance.

For each mode of plant operation, a group of optimum controller setpoints was determined. In areas where the resultant setpoints were different, compromises based on the optimum overall performance were made and verified. A consistent set of control system parameters was derived, satisfying plant operational requirements throughout the core life and for power levels between 15 and 100%.

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

15.1.2 Initial Power Conditions Assumed In Accident Analyses

15.1.2.1 Power Rating

Table 15.1-1 lists the principle power rating values which are used in analyses performed in this section. Two ratings are given:

(1) The guaranteed Nuclear Steam Supply System thermal power output rating. This power output includes the thermal power generated by the reactor coolant pumps.

(2) The Engineered Safety Features design rating. The Westinghouse supplied Engineered Safety Features are designed for thermal power higher than the guaranteed value in order not to preclude realization of future potential power capability. This higher thermal power value is designated as the Engineered Safety Features design rating. This power output includes the thermal power generated by the reactor coolant pumps.

Where initial power operating conditions are assumed in accident analyses, the "guaranteed Nuclear Steam Supply System thermal power output" plus allowance for errors in steady state power determination is assumed. Where demonstration of adequacy of the containment and Engineered Safety Features is concerned, the "Engineered Safety Features design rating" plus allowance for error is assumed. The thermal power values used for each transient analyzed are given in Table 15.1-2.

15.1.2.2 Initial Conditions

For accident evaluation, the initial conditions are obtained by adding the maximum steady state errors to rated values. The following steady state errors are considered:

1.	Core power	± 2% allowance for calorimetric error
2.	Average reactor coolant system temperature	± 6.0°F allowance for deadband and measurement error (bounds an instrument uncertainty of ± 5°F and instrument bias of -1°F)
3.	Pressurizer pressure	+70/-50 psi allowance for steady state fluctuations and measurement error (bounds an instrument uncertainty of ± 50 psi and instrument bias of -20 psi)

For most accidents which are departure from nucleate boiling (DNB) limited, nominal values of initial conditions are assumed. The allowance on power, temperature, and pressure are determined on a statistical basis and are included in the DNB limit ratio (DNBR) as described in reference [27]. This procedure is known as the Revised Thermal Design Procedure (RTDP). The minimum measured flow value is used in all RTDP transients.

Note that the signs of the errors used in the accident analyses are typically opposite of the signs describing the instrument uncertainties; e.g., an instrument error of +50, defined as indicated value of 50 greater than actual value, may be applied in the analysis as -50, i.e., the analysis assumes that the actual value may be 50 less than the nominal value.

For accidents which are not DNB limited or for which the RTDP is not employed, the initial conditions are obtained by adding the bounding steady-state errors to nominal values in such a manner to maximize the impact on the limiting parameter. The thermal design flow value, which is the minimum measured flow minus measurement uncertainty, is used for such analyses.

The thermal design ratings are given in Table 15.1-1.

15.1.2.3 Power Distribution

The transient response of the reactor system is dependent on the initial power distribution. The nuclear design of the reactor core minimizes adverse power distribution through the placement of control rods and operation instructions. The power distribution may be characterized by the radial factor $F_{\Delta H}$ and the total peaking factor F_q. The peaking factor limits are given in the Core Operating Limits Report.

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 increases in $F_{\Delta H}$ is included in the core limits illustrated in Figure 15.1-1. All transients that may be DNB limited are assumed to begin with a value of $F_{\Delta H}$ consistent with the initial power level defined in the Technical Specifications.

The axial power shape used in the DNB calculations is discussed in Section 4.4.3.2.2.

For transients which may be overpower-limited the total peaking factor F_q is of importance. The value of F_q may increase with decreasing power level such that full power hot spot heat flux is not exceeded (i.e., F_q x Power = design hot spot heat flux). All transients that may be overpower-limited are assumed to begin with a value of F_q consistent with the initial power level as defined in the Technical Specifications.

The value of peak kW/ft can be directly related to fuel temperature. For transients which are fast with respect to the fuel rod thermal time constant, for example, rod ejection, a detailed heat transfer calculation is made.

15.1.3 Trip Points And Time Delays To Trip Assumed In Accident Analyses

A reactor trip signal acts to open two trip breakers connected in series feeding power to the control rod drive mechanisms. The loss of power to the mechanism coils causes the mechanisms to release the rod cluster control assemblies which then fall by gravity into the core. There are various instrumentation delays associated with each trip function, including delays in signal actuation, in opening the trip breakers, and in the release of the rods by the mechanisms. The total delay to trip is defined as the time delay from the time that trip conditions are reached to the time the rods are free and begin to fall. Limiting trip setpoints assumed in accident analyses and the time delay assumed for each trip function are given in Table 15.1-3. Reference is made in that table to overtemperature and overpower ΔT trip shown in Figure 15.1-1.

Accident analyses which assume the steam generator low-low water level trip signal to initiate protection functions may be affected by the Trip Time Delay (TTD) (Reference 23) system, which was developed to reduce the incidence of unnecessary feedwater-related reactor trips.

The TTD imposes a system of pre-determined delays upon the steam generator low-low level reactor trip and auxiliary feedwater initiation. The values of these delays are based upon (1) the prevailing power level at the time the low-low level trip setpoint is reached, and by (2) the number of steam generators in which the low-low level trip setpoint is reached. The TTD delays the reactor trip and auxiliary feedwater actuation in order to provide time for corrective action by the operator or for natural stabilization of shrink/swell water level transients. The TTD is primarily designed for low power or startup operations.

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

15.1.4 Instrumentation Drift And Calorimetric Errors - Power Range Neutron Flux

The instrumentation drift and calorimetric errors used in establishing the power range high neutron flux setpoint are presented in References [22] & [28].

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

The secondary power is obtained from measurement of feedwater flow, feedwater inlet temperature to the steam generators and steam pressure. High accuracy instrumentation is provided for these measurements with accuracy tolerances much tighter than those which would be required to control feedwater flow.

15.1.5 Rod Cluster Control Assembly Insertion Characteristic

The rate of negative reactivity insertion following a reactor trip is a function of the acceleration 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% of the rod cluster travel. The most limiting insertion time to dashpot entry used for accident analyses is 2.7 seconds. The normalized rod cluster control assembly position versus time curve assumed in accident analyses is shown in Figure 15.1-2.

Figure 15.1-3 shows the fraction of total negative reactivity insertion 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. There is inherent conservatism in the use of this curve 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 curve corresponding to an insertion time to dashpot entry of 2.7 seconds is shown in Figure 15.1-4. The curve shown in this figure was obtained from Figures 15.1-2 and 15.1-3. A total negative reactivity insertion following a trip of 4% $\Delta k/k$ is assumed in the transient analyses except where specifically noted otherwise. This assumption is conservative with respect to the calculated trip reactivity worth available as shown in Table 4.3-3.

The normalized rod cluster control assembly negative reactivity insertion versus time curve for an axial power distribution skewed to the bottom (Figure 15.1-4) is the most limiting of those transient analyses for which a point kinetics core model is used. Where special analyses require use of three dimensional or axial one dimensional core models, the negative reactivity insertion resulting from the reactor trip is calculated directly by the reactor kinetics code and is not separable from the other reactivity feedback effects. In this case, the rod cluster control assembly position versus time of Figure 15.1-2 is used as code input.

15.1.6 Reactivity Coefficients

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 Chapter 4.

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 reactor coolant from cracks or ruptures in the reactor coolant system do not depend on reactivity feedback effects. The values used are given in Table 15.1-2; reference is made in that table to Figure 15.1-5 which shows the upper and lower bound Doppler power coefficients as a function of power, used in the transient analysis. The justification for use of conservatively large versus small reactivity coefficient values are treated on an event by event basis. To facilitate comparison, individual sections in which justification for the use of large or small reactivity coefficient values is to be found are referenced below:

	Condition II Events	Section
1.	Uncontrolled Rod Cluster Control Assembly Bank Withdrawal From a Subcritical Condition	15.2.1
2.	Uncontrolled Rod Cluster Control Assembly Bank Withdrawal at Power	15.2.2
3.	Rod Cluster Control Assembly Misalignment	15.2.3
4.	Uncontrolled Boron Dilution	15.2.4
5.	Partial Loss of Forced Reactor Coolant Flow	15.2.5
6.	Startup of an Inactive Reactor Coolant Loop	15.2.6
7.	Loss of External Electrical Load and/or Turbine Trip	15.2.7
8.	Loss of Normal Feedwater	15.2.8
9.	Coincident Loss of Onsite and External (Offsite) AC Power to the Station - Loss of Offsite Power to the Station Auxiliaries	15.2.9
10.	Excessive Heat Removal Due to Feedwater System Malfunctions	15.2.10
11.	Excessive Load Increase Incident	15.2.11
12.	Accidental Depressurization of the Reactor Coolant System	15.2.12
13.	Accidental Depressurization of the Main Steam System	15.2.13

14.	Inadvertent Operation of Emergency Core Cooling System During Power Operation	15.2.14
	Condition III Events	
1.	Complete Loss of Forced Reactor Coolant Flow	15.3.4
2.	Single Rod Cluster Control Assembly Withdrawal at Full Power	15.3.6
	Condition IV Events	
1.	Major Rupture of a Main Steam Line	15.4.2.1
2.	Major Rupture of a Main Feedwater Pipe	15.4.2.2
3.	Steam Generator Tube Rupture	15.4.3
4.	Single Reactor Coolant Pump Locked Rotor	15.4.4
5.	Rupture of a Control Rod Drive Mechanism Housing (Rod Cluster Control Assembly Ejection)	15.4.6

15.1.7 Fission Product Inventories

15.1.7.1 Radioactivity in the Core

The average core fission product-inventory is calculated by the ORIGEN-S Subcode within the SCALE-4.2 [2] computer code. The inventories of fission products important from a health hazard point of view are given in Table 15.1-5. The isotopes included in Table 15.1-5 are the isotopes controlling from considerations of inhalation dose (iodines) and from direct dose due to immersion (noble gases).

15.1.7.2 Radioactivity in the Fuel Pellet Clad Gap

The calculation of the maximum core fission product-inventories are also calculated by the ORIGEN-S computer code and are the basis for determining the gap activities used in single fuel assembly accident analyses. The gap activities are consistent with the guidance of Safety Guide 25 [3]: 10% of the total noble gases other than Kr-85 and 30% of Kr-85. For an accident analysis involving a fuel assembly, 10% of the total radioactive iodine in the rods at the time of the accident is also in the gap.

The radioactivity in the reactor coolant as well as in the volume control tank, pressurizer, and waste gas decay tanks are given in Chapter 11 along with the data on which these computations are based.

15.1.8 Residual Decay Heat

Residual heat in a subcritical core consists of:

(1) Fission product decay energy,

(2) Decay of neutron capture products, and

(3) Residual fissions due to the effect of delayed neutrons.

These constituents are discussed separately in the following paragraphs.

15.1.8.1 Fission Product Decay Energy

For short times (10^3 seconds) after shutdown, data on yields of short half life isotopes is sparse. Very little experimental data is available for the X-ray contributions and even less for the β-ray contribution. Several authors have compiled the available data into a conservative estimate of fission product decay energy for short times after shutdown, notably Shure[7] and Dudziak[8]. Of these two selections, Shure's curve is the highest, and it is based on the data of Stehn and Clancy[10] and Obenshain and Foderaro[11].

The fission product contribution to decay energy which has been assumed in the accident analyses is the curve of Shure increased by 20% for conservatism unless otherwise stated in the sections describing specific accidents. This curve with the 20% factor included is shown in Figure 15.1-6.

15.1.8.2 Decay of U-238 Capture Products

Betas and gammas from the decay of U-239 (23.5 minute half-life) and Np-239 (2.35 day half-life) contribute significantly to the heat generation after shutdown. The cross section for production of these isotopes and their decay schemes is relatively well known. For long irradiation times their contribution can be written as:

$$P_1/P_0 = \frac{E_{\gamma_1} + E_{\beta_1}}{200\,\mathrm{Mev}}\, c(1+\alpha)\, e^{-\lambda_1 t} \text{ watts/watt}$$

$$P_2/P_0 = \frac{E_{\gamma_2} + E_{\beta_2}}{200\,\mathrm{Mev}}\, c(1+\alpha) \left[\frac{\lambda_2}{\lambda_1 - \lambda_2}\left(e^{-\lambda_2 t} - e^{-\lambda_1 t}\right) + e^{-\lambda_2 t}\right] \text{ watts/watt}$$

where:

P_1/P_0 =the energy from U-239 decay

P_2/P_0 =the energy from Np-239 decay

t =the time after shutdown (seconds)

$c(1+\alpha)$ =the ratio of U-238 captures to total fissions = 0.6 (1 + 0.2)

λ_1 =the decay constant for U-239 = 4.91×10^{-4} second^{-1}

λ_2 =the decay constant for Np-239 = 3.41×10^{-6} second^{-1}

E_{γ_1} =total γ-ray energy from U-239 decay = 0.06 Mev

E_{γ_2} =total γ-ray energy from Np-239 decay = 0.30 Mev

E_{β_1} =total β-ray energy from U-239 decay = 1/3 x 1.18 Mev

E_{β_2} =total β-ray energy from Np-239 decay = 1/3 x 0.43 Mev

(Two-thirds of the potential β-energy is assumed to escape by the accompanying neutrinos.)

This expression with a margin of 10% has been assumed in the accident analysis unless otherwise stated in the sections describing specific accidents and is shown in Figure 15.1-6. The 10% margin, compared to 20% for fission product decay, is justified by the availability of the basic data required for this analysis. The decay of other isotopes, produced by neutron reactions other than fission, is neglected.

15.1.8.3 Residual Fissions

The time dependence of residual fission power after shutdown depends on core properties throughout a transient under consideration. Core average conditions are more conservative for the calculation of reactivity and power level than actual local conditions as they would exist in hot areas of the core. Thus, unless otherwise stated in the text, static power shapes were assumed in the analyses and these are factored by the time behavior of core average fission power calculated by a point model kinetics calculation with six delayed neutron groups.

For the purpose of illustration, only one delayed neutron group calculation, with a constant shutdown reactivity of negative 4% ΔK is shown in Figure 15.1-6.

15.1.8.4 Distribution of Decay Heat Following Loss of Coolant Accident

During a small break LOCA the core is rapidly shutdown by rod cluster control assembly insertion and a large fraction of the heat generation to be considered comes from fission product decay gamma rays. This heat is not distributed in the same manner as steady state fission power. Local peaking effects which are important for the neutron dependent part of the heat generation do not apply to the gamma-ray contribution. The steady state factor of 97.4% which represents the fraction of heat generated within the clad and pellet drops to 95% for the hot rod in a small break loss of coolant accident.

For example, for an Appendix K Small Break Loss Of Coolant Accident (SBLOCA) analysis, shortly after RCCA insertions about 30% of the heat generated in the fuel rods is from gamma-ray absorption. The gamma power shape is less peaked than the steady state fission power shape, reducing the energy deposited in the hot rod at the expense of adjacent colder rods. A conservative estimate of this effect is a reduction of 10% of the gamma-ray contribution or 3% of the total. Since the water density is considerably reduced at this time, an average of 98% of the available heat is deposited in the fuel rods, the remaining 2% being absorbed by water, thimbles, sleeves and grids. The net effect is a factor of 0.95 rather than 0.974, to be applied to the heat production in the hot rod.

For the BELOCA analysis, the energy deposition modeling is performed as described in Section 8 of Reference [47] in FSAR Chapter 15.4.

15.1.9 Computer Codes Utilized

Summaries of some of the principal computer codes used in transient analyses are given below. Other codes, in particular, are very specialized codes in which the modeling has been developed to simulate one given accident, such as those used in the analysis of the reactor coolant system pipe rupture (Section 15.4), and which consequently have a direct bearing on the accident itself, are summarized or referenced in their respective accident analyses sections. The codes used in the analyses of each transient have been listed in Table 15.1-2.

15.1.9.1 FACTRAN

FACTRAN calculates the transient temperature distribution in a cross section of a metal clad $U0_2$ fuel rod and the transient heat flux at the surface of the clad using as input the nuclear power and the 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-clad 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.

The gap heat transfer coefficient is calculated according to an elastic pellet model (refer to Figure 15.1-8). The thermal expansion of the pellet is calculated as the sum of the radial (one-dimensional) expansions of the rings. Each ring is assumed to expand freely. The cladding diameter is calculated based on thermal expansion and internal and external pressures.

If the outside radius of the expanded pellet is smaller than the inside radius of the expanded clad, there is no fuel-clad contact and the gap conductance is calculated on the basis of the thermal conductivity of the gas contained in the gap. If the pellet's outside radius so calculated is larger than the clad inside radius (negative gap), the pellet and the clad are pictured as exerting upon each other a pressure sufficiently important to reduce the gap to zero by elastic deformation of both. The contact pressure determines the gap heat transfer coefficient.

FACTRAN is further discussed in Reference [12].

15.1.9.2 LOFTRAN

LOFTRAN is used for studies of transient response of a pressurized water reactor system to specified perturbations in process parameters. LOFTRAN simulates a multi-loop system containing reactor vessel, hot and cold leg piping, steam generator (tube and shell sides) and the pressurizer. The pressurizer heaters, spray, 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 neutron flux, overpower and overtemperature reactor coolant ΔT, high and low pressure, low flow, and high pressurizer level. Control systems are also simulated including rod control, steam dump, feedwater control and pressurizer pressure control. The safety injection system including the accumulators is also modeled.

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

LOFTRAN also has the capability of calculating the transient value of DNB ratio based on the input from the core limits illustrated on Figure 15.1-1. The core limits represent the minimum value of DNBR as calculated for typical or thimble cell.

LOFTRAN is further discussed in Reference [15].

15.1.9.3 LEOPARD

LEOPARD determines fast and thermal neutron spectra, using only basic geometry and temperature data. The code optionally computes fuel depletion effects for a dimensionless reactor and recomputes the spectra before each discrete burnup step.

LEOPARD is further described in Reference [16].

15.1.9.4 TURTLE

TURTLE is a two-group, two-dimensional neutron diffusion code featuring a direct treatment of the nonlinear effects of xenon, enthalpy, and Doppler. Fuel depletion is allowed.

TURTLE was written for the study of azimuthal xenon oscillations, but the code is useful for general analysis. The input is simple, fuel management is handled directly, and a boron criticality search is allowed.

TURTLE is further described in Reference [17].

15.1.9.5 TWINKLE

TWINKLE is a multi-dimensional spatial neutron kinetics code patterned after steady-state codes 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-clad-coolant heat transfer model for calculating pointwise Doppler and moderator feedback effects. The code handles up to 2000 spatial points, and performs its own steady state initialization. Aside from basic cross-section data and thermal-hydraulic parameters, the code accepts as input basic driving functions such as inlet temperature, pressure, flow, boron concentration, control rod motion, and others. Various edits include channelwise power, axial offset, enthalpy, volumetric surge, pointwise power, and fuel temperatures.

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

TWINKLE is further described in Reference [18].

15.1.9.6 VIPRE-01

VIPRE-01 is described in Section 4.4.3.4.

15.1.9.7 LOFTTR

The steam generator tube rupture (SGTR) analyses were performed for Watts Bar using the analysis methodology developed in WCAP-10698[24] and Supplement 1 to WCAP-10698.[25] The methodology was developed by the SGTR Subgroup of the Westinghouse Owners Group (WOG) and was approved by the NRC in Safety Evaluation Reports (SERs) dated December 17, 1985 and March 30, 1987. The LOFTTR2 program, an updated version of the LOFTTR1 program, was used to perform the SGTR analysis for Watts Bar. The LOFTTR1 program was developed as part of the revised SGTR analysis methodology and was used for the SGTR evaluations.[24][25] However, the LOFTTR1 program was subsequently modified to accomodate steam generator overfill and the revised program, designated as LOFTTR2, and was used for the evaluation of the consequences of overfill in WCAP-11002.[26] The LOFTTR2 program is identical to the LOFTTR1 program, with the exception that the LOFTTR2 program has the additional capability to represent the transition from two regions (steam and water) on the secondary side to a single water region if overfill occurs, and the transition back to two regions again depending upon the calculated secondary conditions. Since the LOFTTR2 program has been validated against the LOFTTR1 program, the LOFTTR2 program is also appropriate for performing licensing basis SGTR analyses. The specific Watts Bar LOFTTR2 analysis utilizing this methodology is described in 15.4.3.

REFERENCES

(1) Deleted by Amendment 80.

(2) SCALE-4.2: A Modular Code System for Performing Standardized Computer Analyses for Licensing Evaluation, Volumes I-III, NUREG/CR-0200, Rev. 5 (ORNL/NUREG/CSD-2/R5), March 1997 (ORIGEN-S Subsection)

(3) Safety Guide 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.

(4) Toner, D. F. and Scott, J. S., "Fission-Product Release from UO", Nuc. Safety 3 No. 2, 15-20, December 1961.

(5) Belle, J., "Uranium Dioxide Properties and Nuclear Applications," Naval Reactors, Division of Reactor Development United States Atomic Energy Commission, 1961.

(6) Booth. A. H., "A Suggested Method for Calculating the Diffusion of Radioactive Rare Gas Fission Products From UO Fuel Elements," DCI-27, 1957.

(7) Shure, K., "Fission Product Decay Energy" in Bettis Technical Review, WAPD-BT-24, p. 1-17, December 1961.

(8) Shure, K. and Dudziak, D. J., "Calculating Energy Released by Fission Products," Trans. Am. Nucl. Soc. 4 (1) 30 (1961).

(9) Deleted by Amendment 65.

(10) Stehn, J.R. and Clancy, E. F., "Fission-Product Radioactivity and Heat Generation" and "Proceedings of the Second United Nations International Conference on the Peaceful Uses of Atomic Energy, Geneva, 1958," Volume 13, pp. 49-54, United Nations, Geneva, 1958.

(11) Obershain, F. E. and Foderaro, A. H., "Energy from Fission Product Decay," WAPD-P-652, 1955.

(12) Hargrove, H. G., "FACTRAN, a FORTRAN IV Code for Thermal Transients in a UO2 Fuel Rod," WCAP-7908, December 1989.

(13) Deleted by Amendment 72.

(14) Deleted by Unit 1 UFSAR Amendment 0.

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

(16) Barry, R. F., "LEOPARD, A Spectrum Dependent Non-Spatial Depletion Code for the IBM-7094," WCAP-3269-26, September 1963.

(17) Barry, R. F. and Altomare, S., "The TURTLE 24.0 Diffusion Depletion Code," WCAP-7213-P-A (Proprietary) and WCAP-7758-A (Non-Proprietary), January 1975.

(18) 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.

(19) Deleted by Amendment 80.

(20) Deleted by Amendment 80.

(21) Haessler, R. L. et al, “Methodology for The Analysis of the Dropped Rod Event,” WCAP-11394-P-A (Proprietary) and WCAP-11395-A (Non-Proprietary), January 1990.

(22) Reagan, J. R. and Tuley, C. R., "Westinghouse Setpoint Methodology for Protection Systems, Watts Bar Units 1 and 2, Eagle 21 Version," WCAP-12096, Rev. 5, May 1993 (Proprietary). UNIT 1 ONLY.

(23) Miranda, S., et.al., "Steam Generator Low Water Level Protection System Modifications to Reduce Feedwater Related Trips," WCAP-11325-P-A, Revision 1, February 1988.

(24) Lewis, Huang, Behnke, Fittante, Gelman, "SGTR Analysis Methodology to Determine the Margin to Steam Generator Overfill," WCAP-10698-P-A [PROPRIETARY]/WCAP-10750-A [NON-PROPRIETARY], August 1987.

(25) Lewis, Huang, Rubin, "Evaluation of Offsite Radiation Doses for a Steam Generator Tube Rupture Accident," Supplement 1 to WCAP-10698-P-A [PROPRIETARY]/Supplement 1 to WCAP-10750-A [NON-PROPRIETARY], March 1986.

(26) Lewis, Huang, Rubin, Murray, Roidt, Hopkins, "Evaluation of Steam Generator Overfill Due to a Steam Generator Tube Rupture Accident," WCAP-11002 [PROPRIETARY]/WCAP-11003 [NON-PROPRIETARY], February 1986.

(27) Friedland, A. J. and S. Ray, “Revised Thermal Design Procedure,” WCAP-11397-P-A (PROPRIETARY) and WCAP-11398-A (NON-PROPRIETARY), April 1989.

(28) Trozzo, R. W., “Westinghouse Setpoint Methodology for Protection Systems - Watts Bar Unit 2,” WCAP-17044-P, Revision 1, September 2012, (Unit 2 Only).

Table 15.1-1 Nuclear Steam Supply Power Ratings

Guaranteed Nuclear Steam Supply System thermal power output	3425 MWt
The Engineered Safety (Features) Design Rating (ESDR)(initial design maximum calculated turbine rating is 3579 MWt)	3650 MWt
Thermal power generated by the reactor coolant pumps	14 MWt
Guaranteed core thermal power	3411 MWt

Table 15.1-2 Summary Of Initial Conditions And Computer Codes Used (Page 1 of 4)

FAULTS	COMPUTER CODES UTILIZED	REACTIVITY COEFFICIENTS ASSUMED FOR: MODERATOR TEMPERATURE (Δk/°F)	MODERATOR DENSITY (Δk/gm/cc)	DOPPLER	INITIAL NSSS THERMAL POWER OUTPUT ASSUMED[1] (MWt)
CONDITION II					
Uncontrolled RCC Assembly Bank Withdrawal from Subcritical Condition	TWINKLE, FACTRAN, VIPRE-01	Refer to Section 15.2.1.2 (Part 2)	--	Least negative Doppler power coefficient- Doppler defect = 960 pcm	3411 (critical @ 0.0 fraction of Nominal [FON])
Uncontrolled RCC Assembly Bank Withdrawal at Power	LOFTRAN	---	0.00 and 0.43	lower and upper[2]	3425
RCC Assembly Misalignment	VIPRE-01, LOFTRAN	---	0.00	upper[2]	3425
Uncontrolled Boron Dilution	NA	NA	NA	NA	0 and 3425
Partial Loss of Forced Reactor Coolant Flow	LOFTRAN, VIPRE-01	---	0.00	upper[2]	3475
Startup of an Inactive Reactor Coolant Loop	N/A	---	N/A	N/A	N/A
Loss of External Electrical Load and/or Turbine Trip	LOFTRAN	---	0.00	upper[2]	3425
Loss of Normal Feedwater/ Loss of Off-Site Power to the Station Auxiliaries	LOFTRAN	---		upper[2]	3475
Excessive Heat Removal Due to Feedwater System Malfunctions[3]	LOFTRAN	---	0.43	lower[2]	3425
Excessive Load Increase Incident	N/A	---	N/A	N/A	N/A
Accidental Depressurization of the Reactor Coolant System	LOFTRAN		0.00	upper[2]	3425

Table 15.1-2 Summary Of Initial Conditions And Computer Codes Used (Continued)
(Page 2 of 4)

FAULTS	COMPUTER CODES UTILIZED	REACTIVITY COEFFICIENTS ASSUMED FOR: MODERATOR TEMPERATURE (Δk/°F)	MODERATOR DENSITY (Δk/gm/cc)	DOPPLER	INITIAL NSSS THERMAL POWER OUTPUT ASSUMED[1] (MWt)
CONDITION II (Cont'd)					
Accidental Depressurization of the Main Steam System	Accident evaluated; bounded by major rupture of a steam pipe				
Accidental Depressurization of the Reactor Coolant System	LOFTRAN		0.00	upper[2]	3425
Inadvertent Operation of ECCS During Power Operation	LOFTRAN	---	0.00 and 0.43	lower and upper[2]	3475[5]
CONDITION III					
Loss of Reactor Coolant from Small Ruptured Pipes or from Cracks in Large Pipes which Actuates Emergency Core Cooling	NOTRUMP, LOCTA-IV				3411[5]
Inadvertent Loading of a Fuel Assembly into an Improper Position	LEOPARD, TURTLE	---	Minimum	NA	3425
Complete Loss of Forced Reactor Coolant Flow	VIPRE-01, FACTRAN, LOFTRAN	---	0.00	upper[2]	3475
Waste Gas Decay Tank	NA	---	NA	NA	3579
Single RCC Assembly Withdrawal at Full Power	TURTLE, THINC, LEOPARD	---	NA	NA	3425

Table 15.1-2 Summary Of Initial Conditions And Computer Codes Used (Continued) (Page 3 of 4)

FAULTS	COMPUTER CODES UTILIZED	REACTIVITY COEFFICIENTS ASSUMED FOR: MODERATOR TEMPERATURE (Δk/°F)	MODERATOR DENSITY (Δk/gm/cc)	DOPPLER	INITIAL NSSS THERMAL POWER OUTPUT ASSUMED[1] (MWt)
CONDITION IV					
Major Rupture of Pipes Containing Reactor Coolant Up to and Including Double-ended Rupture of the Largest Pipe in the Reactor Coolant System (Loss of Coolant Accident)	WCOBRA/TRAC, HOTSPOT, LOTIC2	--	0.00	Function of fuel temperature.	3475
Major Rupture of a Steam Pipe	LOFTRAN, VIPRE-01,	Function of moderator density; see Section 15.2.13 (Figure 15.2-40)		Note 3	3425 (critical @ 0.0 fraction of nominal [FON]).
Major Rupture of a Main Feedwater Pipe	LOFTRAN	---	0.00	upper[2]	3425
Steam Generator Tube Rupture	LOFTTR2	0 pcm/°F @ 100 RTP	Figure 15.1-7	upper[2]	3427
Single Reactor Coolant Pump Locked Rotor	LOFTRAN, VIPRE-01, FACTRAN	---	0.00	upper[2]	3475
Fuel Handling Accident	NA	NA	NA		3579
Rupture of a Control Rod Drive Mechanism Housing (RCCA Ejection)	TWINKLE, FACTRAN	Refer to Section 15.4.6	---	Least negative Doppler defect, see Table 15.4-12	3411 (HZP 0)

Table 15.1-2 Summary Of Initial Conditions And Computer Codes Used (Continued)
(Page 4 of 4)

FAULTS	COMPUTER CODES UTILIZED	REACTIVITY COEFFICIENTS ASSUMED FOR: MODERATOR TEMPERATURE (Δk/°F)	MODERATOR DENSITY (Δk/gm/cc)	DOPPLER	INITIAL NSSS THERMAL POWER OUTPUT ASSUMED[1] (MWt)

[1] The values provided do not include the power uncertainty that is applied either directly (non-RTDP) or statistically (RTDP events).

[2] Refer to Figure 15.1-5.

[3] Refer to Figure 15.4-9.

.[4] LOCA M/E based on Engineering Safety Design Rating (ESDR) of 3650 MWt.

[5] Severalof these analyses are conservatively based upon a core power of 3459 MWt and NSSS power of 3475 MWt, based upon a redefinition of the 2% power uncertainty (2% to 0.6%), which bounds a core power of 3411 MWt and NSSS power of 3425 MWt.

Table 15.1-3 Trip Points And Time Delays To Trip Assumed In Accident Analyses

Trip Function	Limiting Trip Point Assumed in Analysis	Time Delay (Seconds)
Power Range High Neutron Flux, High Setting	118%	0.5
Power Range High Neutron Flux, Low Setting	35%	0.5
Overtemperature ΔT	Variable (see Figure 15.1-1)	8.0*
Overpower ΔT	Variable (see Figure 15.1-1)	8.0*
High Pressurizer Pressure	2445 psig	2.0
Low Pressurizer Pressure	1910 psig	2.0
*Total time delay (including 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.		
Low Reactor Coolant Flow (from loop flow detectors)	87% loop flow	1.2
Undervoltage Trip	68%	1.5
Turbine Trip	Not applicable	1.0
Low-Low Steam Generator Level	0% of narrow range span	2.0 + TTD*
High-High Steam Generator Level, Turbine Trip, and Feedwater Isolation	100% of narrow range level span	2.5
* Trip Time Delay (TTD) is applicable only below 50% RTP.		

Table 15.1-4 Determination Of Maximum Overpower Trip Point Power Range Neutron Flux Channel - Based On Nominal Setpoint Considering Inherent Instrumentation Errors
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Table 15.1-5 Core And Gap Activities Based On Full Power Operation For 1000 Days Full Power: 3565 MWt

Isotope	Curies/Assembly	Total Curies in Core
KR-83m	5.96E+04	1.15E+07
KR-85m	1.24E+05	2.39E+07
Kr-85	5.35E+03	1.03E+06
Kr-87	2.49E+05	4.81E+07
Kr-88	3.45E+05	6.66E+07
Kr-89	4.29E+05	8.28E+07
Xe-131m	5.43E+03	1.05E+06
Xe-133m	3.19E+04	6.16E+06
Xe-133	9.92E+05	1.91E+08
Xe-135m	2.10E+05	4.05E+07
XE-135	3.33E+05	6.43E+07
Xe-138	8.64E+05	1.67E+08
I-131	4.90E+05	9.46E+07
I-132	7.18E+05	1.39E+08
I-133	1.01E+06	1.95E+08
I-134	1.12E+06	2.16E+08
I-135	9.65E+05	1.86E+08

Table 15.1-6 Deleted by Amendment 97

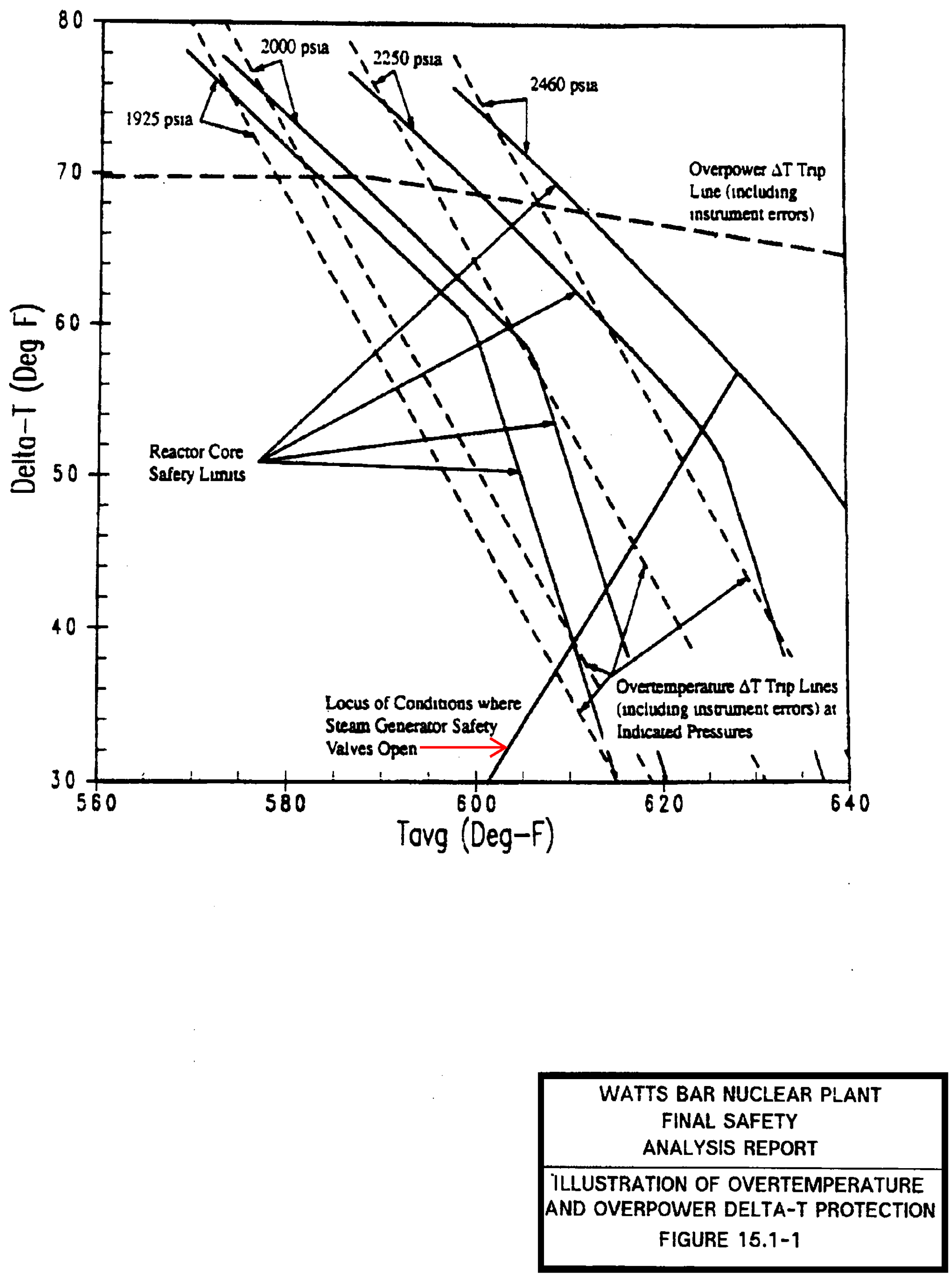
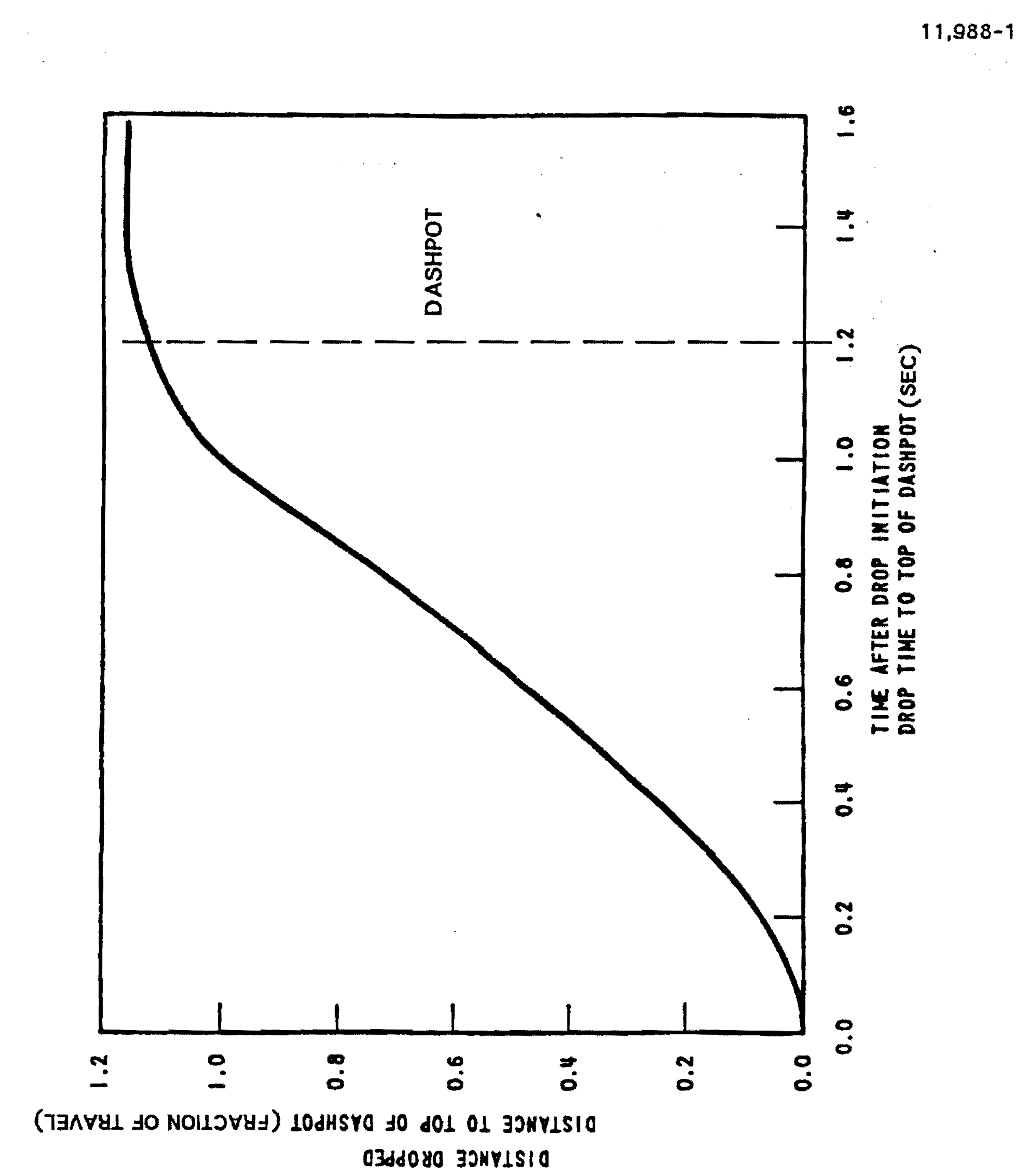


Figure 15.1-1 Ilustration of Overtemperature and Overpower ΔT Protection



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RCCA POSITION VERSUS TIME ON REACTOR TRIP

FIGURE 15.1-2

SCANNED DOCUMENT
THIS IS A SCANNED DOCUMENT MAINTAINED ON THE WBNP OPTIGRAPHICS SCANNER DATABASE

Figure 15.1-2 RCCA Position Versus Time On Reactor Trip

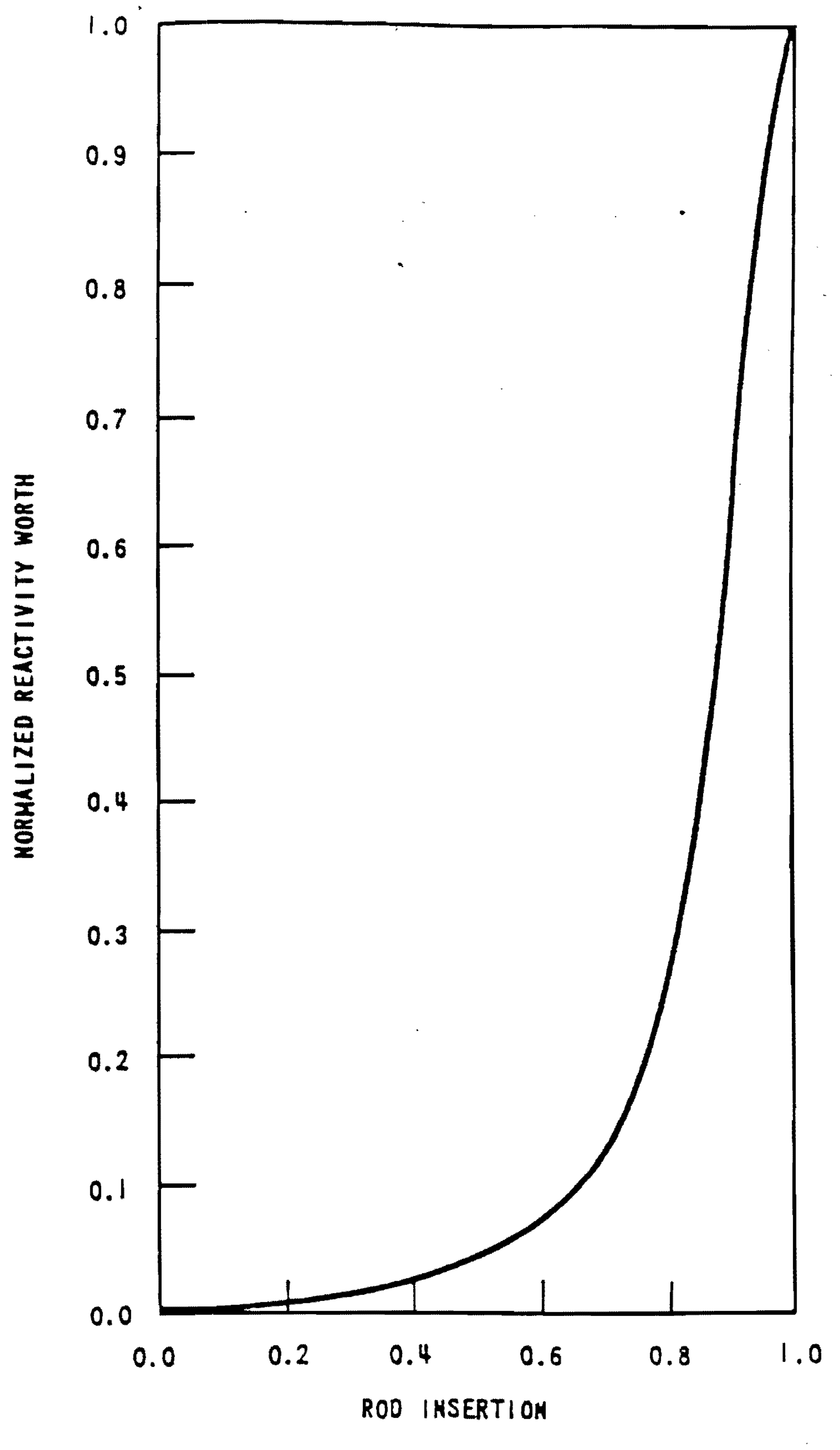

Figure 15.1-3 Normalized RCCA Reactivity Worth Versus Rod Insertion Fraction

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Figure 15.1-3 Normalized RCCA Reactivity Worth Versus Rod Insertion Fraction

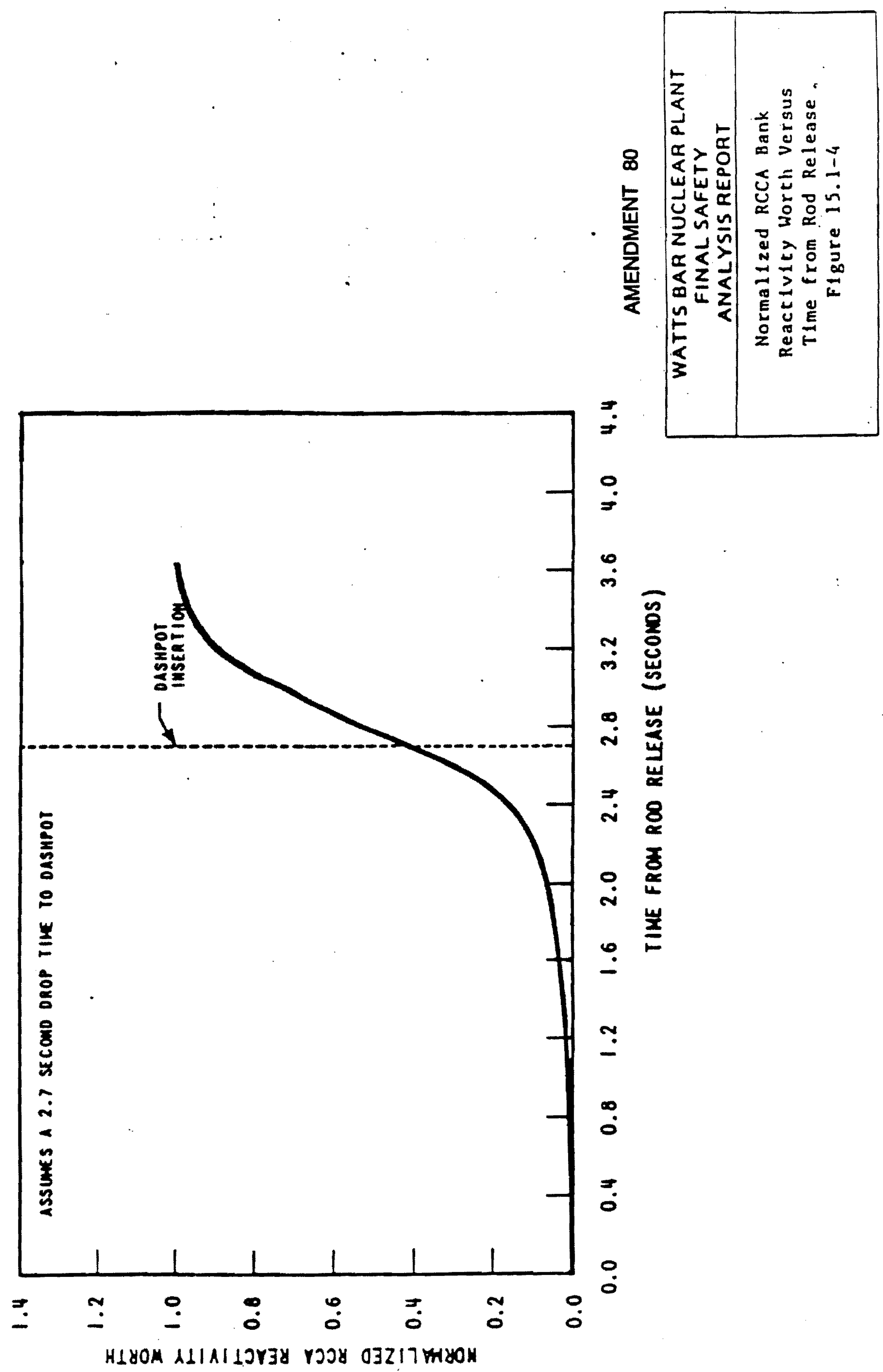
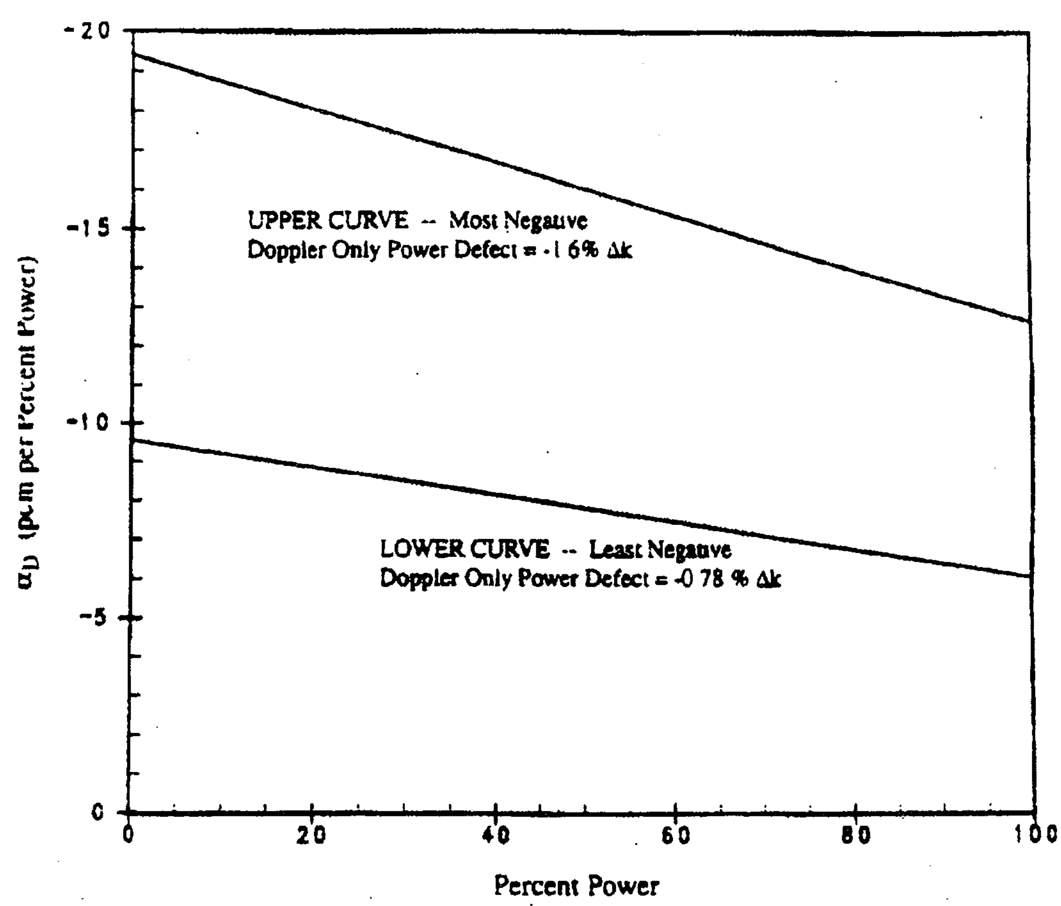


Figure 15.1-4 Normalized RCCA Bank Reactivity Worth Versus Time from Rod Re lease



WATTS BAR NUCLEAR PLANT FINAL SAFETY ANALYSIS REPORT
DOPPLER POWER COEFFICIENT USED IN ACCIDENT ANALYSIS FIGURE 15.1-5

Figure 15.1-5 Doppler Power Coefficient Used In Accident Analysis

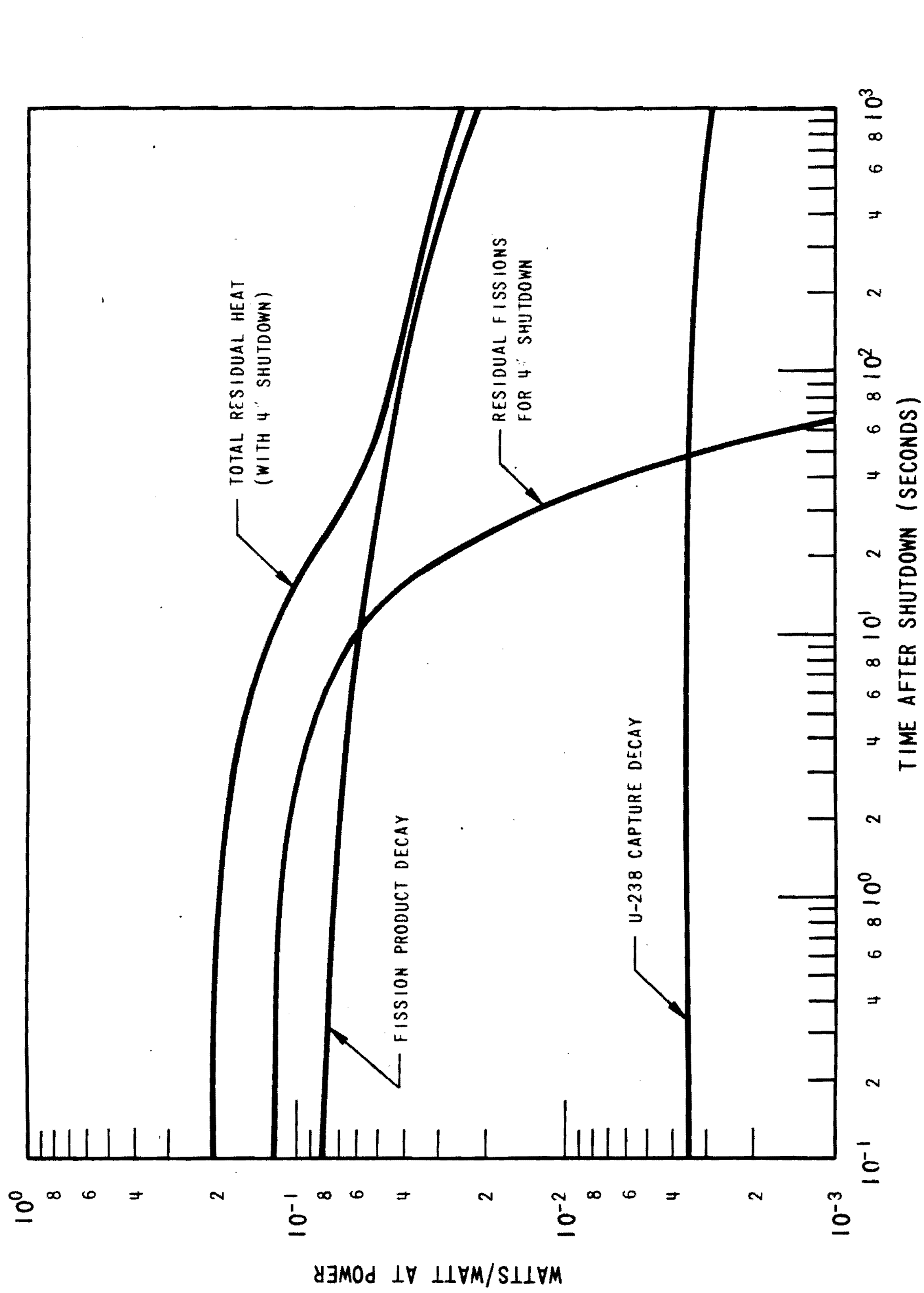
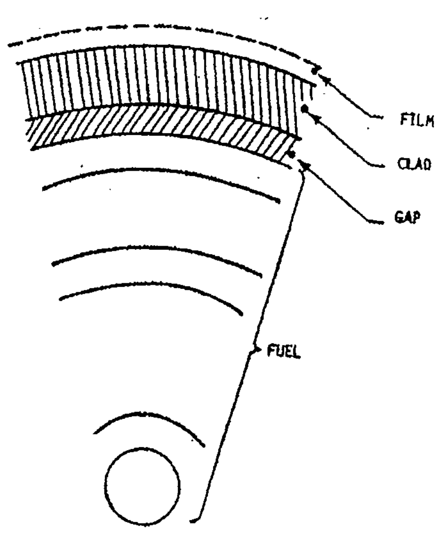


Figure 15.1-6 Residual Decay Heat

Figure 15.1-6 Residual Decay Heat

Figure 15.1-7 Deleted by Amendment 97



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FUEL ROD CROSS SECTION

FIGURE 15.1-8

Figure 15.1-8 Fuel Rod Cross Section

15.2 CONDITION II - FAULTS OF MODERATE FREQUENCY

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

(1) Uncontrolled rod cluster control assembly bank withdrawal from a subcritical condition.

(2) Uncontrolled rod cluster control assembly bank withdrawal at power.

(3) Rod cluster control assembly misalignment.

(4) Uncontrolled boron dilution.

(5) Partial loss of forced reactor coolant flow.

(6) Startup of an inactive reactor coolant loop.

(7) Loss of external electrical load and/or turbine trip.

(8) Loss of normal feedwater.

(9) Loss of offsite power to the station auxiliaries (station blackout).

(10) Excessive heat removal due to feedwater system malfunctions.

(11) Excessive load increase incident.

(12) Accidental depressurization of the reactor coolant system.

(13) Accidental depressurization of the main steam system.

(14) Inadvertent operation of emergency core cooling system during power operation.

(15) Chemical and Volume Control System Malfunction During Power Operation.

An evaluation of the reliability of the reactor protection system actuation following initiation of Condition II events is presented in Reference [1] for the relay protection logic. Standard reliability engineering techniques were used to assess likelihood of the trip failure due to random component failures. Common mode failures were also qualitatively investigated. It was concluded from the evaluation that the likelihood of no trip following initiation of Condition II events is extremely small (2×10^{-7} derived for random component failures).

The solid state protection system design has been evaluated by the same methods as used for the relay system and the same order of magnitude of reliability is provided.

The worst common mode failure which is postulated to occur is the failure to scram the reactor after an anticipated transient has occurred. A series of generic studies, References [2] and [11], on anticipated transients without scram (ATWS) showed acceptable consequences would result provided that the turbine trips and auxiliary feedwater flow is initiated in a timely manner. The effects of ATWS events are not considered as part of the design basis for transients analyzed in Chapter 15. The final NRC ATWS rule [12] requires that Westinghouse-designed plants install ATWS mitigation system circuitry (AMSAC) to initiate a turbine trip and actuate auxiliary feedwater flow independent of the reactor protection system. The Watts Bar AMSAC design is described in Section 7.7.1.12.

The time sequence of events during applicable Condition II events is shown in Table 15.2-1.

15.2.1 Uncontrolled Rod Cluster Control Assembly Bank Withdrawal from a Subcritical Condition

15.2.1.1 Identification of Causes and Accident Description

A rod cluster control assembly (RCCA) withdrawal accident is defined as an uncontrolled addition of reactivity to the reactor core caused by withdrawal of 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 either subcritical, hot zero power or at power. The "at power" case is discussed in Section 15.2.2.

Although the reactor is normally brought to power from a subcritical condition by means of RCCA withdrawal, initial startup procedures with a clean core call for boron dilution. The maximum rate of reactivity increase in the case of boron dilution is less than that assumed in this analysis (see Section 15.2.4).

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

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 a tolerable level during the delay time for protective action. Should a continuous RCCA withdrawal accident occur, the transient will be terminated by the following automatic features of the reactor protection system:

(1) Source Range High Neutron Flux Reactor Trip - actuated when either of two independent source range channels indicates a neutron flux level above a preselected manually adjustable setpoint. This trip function may be manually bypassed only after an intermediate range flux channel indicates a flux level above a specified level. It is automatically reinstated when both intermediate range channels indicate a flux level below a specified level.

(2) Intermediate Range High Neutron Flux Reactor Trip - actuated when either of two independent intermediate range channels indicates a neutron flux level above a preselected manually adjustable setpoint. This trip function may be manually bypassed only after two of the four power range channels are reading above approximately 10% 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% of full power. This trip function may be manually bypassed when two of the four power range channels indicate a power level above approximately 10% of full power and is automatically reinstated only after three of the four channels indicate a power level below this value.

(4) Power Range High Neutron Flux Reactor Trip (High Setting) - actuated when two out of the four power range channels indicate a power level above a preset setpoint. This trip function is always active.

(5) Power Range High Positive Neutron Flux Rate Trip - actuated when the positive rate of change of neutron flux on two out of four nuclear power range channels indicate a rate above the preset setpoint. This trip function is always active.

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

15.2.1.2 Analysis of Effects and Consequences

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 a DNBR calculation. The average core nuclear power calculation is performed using spatial neutron kinetics methods, TWINKLE[3], to determine the average power generation with time, including the various total core feedback effects, i.e., Doppler reactivity and moderator reactivity.

The average heat flux and temperature transients are determined by performing a fuel rod transient heat transfer calculation in FACTRAN[4]. The average heat flux is next used in VIPRE-01 (described in Section 4.4.3.4) for the transient DNBR calculation.

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

(1) Since the magnitude of the power peak reached during the initial part of the transient for any given rate of reactivity insertion is strongly dependent on the Doppler coefficient, conservative values (low absolute magnitude) as a function of power are used. See Section 15.1.6 and Table 15.1-2.

(2) Contribution of the moderator reactivity coefficient is negligible during the initial part of the transient because the heat transfer time between the fuel and the moderator is much longer than the neutron flux response time. However, after the initial neutron flux peak, the succeeding rate of power increase is affected by the moderator reactivity coefficient. A conservative value which is appropriate for beginning of core life at hot zero power, is used in the analysis to yield the maximum peak heat flux.

(3) The reactor is assumed to be at hot zero power. This assumption is more conservative than that of a lower initial system temperature. The higher initial system temperature yields a 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 the worst nuclear power transient.

(4) Reactor trip is assumed to be initiated by power range high neutron flux (low setting). The most adverse combination of instrument and setpoint errors, as well as delays for trip signal actuation and rod cluster control assembly release, is taken into account. A 10% increase is assumed for the power range flux trip setpoint raising it from the nominal value of 25% to 35%. Previous results, however, show that rise in the neutron flux is so rapid that the effect of errors in the trip setpoint on the actual time at which the rods are released is negligible. In addition, the reactor trip insertion characteristic is based on the assumption that the highest worth RCCA is stuck in its fully withdrawn position. See Section 15.1.5 for RCCA insertion characteristics.

(5) The maximum positive reactivity insertion rate assumed is greater than that for the simultaneous withdrawal of the combination of the two sequential control banks having the greatest combined worth at maximum speed (45 inches/minute). Control rod drive mechanism design is discussed in Section 4.2.3.

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

(7) The most limiting axial and radial power shapes, associated with having the two highest combined worth sequential control banks in their high worth position, are assumed in the DNB analysis.

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

Results

The calculated sequence of events for this accident is shown on Table 15.2-1.

Figures 15.2-1 through 15.2-3 show the transient behavior for the indicated reactivity insertion rate with the accident terminated by reactor trip at 35% nominal power. This insertion rate is greater than that for the two highest worth sequential control banks, both assumed to be in their highest incremental worth region.

Figure 15.2-1 shows the nuclear power transient. The nuclear power overshoots the full power nominal value but this occurs for only a very short time period. Hence, the energy release and the fuel temperature increases are relatively small. The heat flux response, of interest for DNB considerations, is shown on Figure 15.2-2. The beneficial effect of the inherent thermal lag in the fuel is evidenced by a peak heat flux less than the peak nuclear power value. Figures 15.2-3 and 15.2-3a show the response of the hot spot average fuel and cladding temperatures. The average fuel temperature increases to a value lower than the nominal full power value. The minimum DNBR remains above the limiting value at all times.

15.2.1.3 Conclusions

In the event of a RCCA withdrawal accident from the subcritical condition, the core and the reactor coolant system are not adversely affected, since the combination of thermal power and the coolant temperature result in a DNBR greater than the limiting value. Thus, no cladding damage and no release of fission products to the reactor coolant system is predicted as a result of DNB.

15.2.2 UNCONTROLLED ROD CLUSTER CONTROL ASSEMBLY BANK WITHDRAWAL AT POWER

15.2.2.1 Identification of Causes and Accident Description

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

The automatic features of the reactor protection system which prevent core damage following the postulated accident include the following:

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

(2) Reactor trip is actuated if any two out of four ΔT channels exceed an overtemperature ΔT setpoint. This setpoint is automatically varied with axial power imbalance, coolant 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.

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

(5) A high pressurizer water level reactor trip actuated from any two out of three level channels which is set at a fixed point.

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

(1) High neutron flux (one out of four)

(2) Overpower ΔT (two out of four)

(3) Overtemperature ΔT (two out of four)

The manner in which the combination of overpower and overtemperature ΔT trips provide protection over the full range of reactor coolant system conditions is described in Chapter 7. Figure 15.1-1 presents allowable reactor coolant loop average temperature and ΔT for the design power distribution and flow as a function of primary coolant pressure. The boundaries of operation defined by the overpower ΔT trip and the overtemperature ΔT trip are represented as "protection lines" on this diagram. The protection lines are drawn to include all adverse instrumentation and setpoint errors so that under nominal conditions 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 is above the limiting value. All points below and to the left of a DNB line for a given pressure have a DNBR greater than the limiting 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).

15.2.2.2 Analysis of Effects and Consequences

Method of Analysis

This transient is analyzed by the LOFTRAN[5] 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 15.1-1 are used as input to LOFTRAN to determine the minimum DNBR during the transient.

In order to obtain conservative values of DNBR the following assumptions are made:

(1) Nominal initial conditions of core power, reactor coolant average temperature, and reactor coolant pressure, are assumed in accordance with RTDP methodology[18].

(2) Reactivity Coefficients - Two cases are analyzed:

(a) Minimum Reactivity Feedback. A least negative 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 reactor trip on high neutron flux is assumed to be actuated at a conservative value of 118% of nominal full power. The ΔT trips include all adverse instrumentation and setpoint errors, while the delays for the trip signal actuation are assumed at their maximum values.

(4) The RCCA trip insertion characteristics are based on the assumption that the highest worth assembly is stuck in its fully withdrawn position.

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

Results

The calculated sequence of events for this accident is shown on Table 15.2-1.

Figures 15.2-4 and 15.2-5 show the response of neutron flux, pressurizer pressure, average coolant temperature, and DNBR to a rapid RCCA withdrawal incident starting from full power. Reactor trip on high neutron flux occurs shortly after the start of the accident. Since this is rapid with respect to the thermal time constants of the plant, small changes in T_{avg} and pressure result and a large margin to DNB is maintained.

The response of neutron flux, pressure, average coolant temperature, and DNBR for a slow control rod assembly withdrawal from full power is shown in Figures 15.2-6 and 15.2-7. Reactor trip on overtemperature ΔT occurs after a longer period of time than for the rapid RCCA withdrawal incident and the rise in temperature is consequently larger.

Following reactor trip, the plant approaches a stabilized condition at hot standby; normal plant operating procedures may then be followed. The operating procedures call for operator action to control RCS boron concentration and pressurizer level using the chemical and volume control system (CVCS), and to maintain steam generator level through control of the main or auxiliary feedwater system. Any action required of the operator to maintain the plant in a stabilized condition is in a time frame in excess of ten minutes following reactor trip.

Figure 15.2-8 shows the minimum DNBR as a function of reactivity insertion rate from initial full power operation for the minimum and maximum reactivity feedback. It can be seen that two reactor trip functions provide protection over the whole range of reactivity insertion rates. These are the high neutron flux and overtemperature ΔT trip functions. The minimum DNBR is never less than the limiting value.

Figures 15.2-9 and 15.2-10 show the minimum DNBR as function of reactivity insertion rate for RCCA withdrawal incidents starting at 60% and 10% power, respectively. The results are similar to the 100% power case except, as the initial power is decreased, the range over which the overtemperature ΔT trip is effective increases. In neither case does the DNBR fall below its minimum limit.

The shape of the curves of minimum DNB ratio versus reactivity insertion rate in the referenced figures is due both to reactor core and coolant system transient response and to protection system action in initiating a reactor trip.

Referring to Figure 15.2-8, for example, it is noted that

(1) For high reactivity insertion rates (i.e., between 4.0×10^{-4} Δk/k/sec and 8.0×10^{-4} Δk/k/sec) 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 DNB ratios remaining above the limiting value during the transient. As reactivity insertion rate decreases, core heat flux and coolant temperatures can remain more nearly in equilibrium with the neutron flux; minimum DNB ratio during the transient thus decreases with decreasing insertion rate.

(2) The overtemperature ΔT reactor trip circuit initiates a reactor trip when measured coolant loop ΔT exceeded a setpoint based on measured RCS average temperature and pressure. This trip circuit is described in detail in Chapter 7; however, it is important in this context to note that the average temperature contribution to the circuit is lead-lag compensated in order to decrease the effect of the thermal capacity of the RCS in response to power increases.

(3) With further decrease in reactivity insertion rate, the overtemperature ΔT and high neutron flux trips become equally effective in terminating the transient (e.g., at approximately 4.0×10^{-4} Δk/k/sec reactivity insertion rate).

For reactivity insertion rates between approximately 4.0×10^{-4} Δk/k/sec to 5.0×10^{-4} Δk/k/sec, the effectiveness of the overtemperature ΔT trip increases (in terms of increased minimum DNBR) 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.

(4) For reactivity insertion rates less than approximately 5.0×10^{-5} Δk/k/sec, the rise in the reactor coolant temperature is sufficiently high so that the steam generator safety valve setpoint is reached prior to trip. Opening of these valves, which act as an additional heat load of the RCS, sharply decreases the rate of rise of RCS average temperature. This decrease in rate of rise of the average coolant system temperature during the transient is accentuated by the lead-lag compensation causing the overtemperature ΔT trip setpoint to be reached later with resulting lower minimum DNBRs.

Figures 15.2-8, 15.2-9, and 15.2-10 illustrate minimum DNBR calculated for minimum and maximum reactivity feedback.

15.2.2.3 Conclusions

The high neutron flux and overtemperature ΔT trip channels provide adequate protection over the entire range of possible reactivity insertion rates, i.e., the minimum value of DNBR is always larger than the limiting value.

15.2.3 ROD CLUSTER CONTROL ASSEMBLY MISALIGNMENT

15.2.3.1 Identification of Causes and Accident Description

Rod cluster control assembly (RCCA) misalignment accidents include:

(1) One or more dropped RCCAs within the same group;

(2) A dropped RCCA bank;

(3) Statically misaligned RCCA

Each RCCA has a position indicator channel where the information is sent to monitors in the main control room. The monitors display the position of the assembly. The displays of assembly positions are grouped for the operator's convenience. Fully inserted assemblies are further indicated by a rod at bottom signal, which actuates a local alarm and a main control room annunciator. Group demand position is also indicated. The assemblies are always moved in preselected banks and the banks are always moved in the same preselected sequence.

Each bank of RCCAs is divided into two groups except Shutdown Banks C and D which have one group each. The rods comprising a group operate in paralled through multiplexing thyristors. The two groups in a bank move sequentially such that the first group is always within one step of the second group in the bank. A definite sequence of actuation of the stationary gripper, movable gripper, and lift coils of a mechanism is required to withdraw or insert the RCCA attached to the mechanism. Since the stationary gripper, movable gripper, and lift coils associated with the RCCAs of a rod group are driven in parallel, any single failure which would cause rod withdrawal would affect just that one group. Mechanical failures are in the direction of insertions, or immobility.

A dropped RCCA or RCCA bank is detected by:

(1) Sudden drop in the core power level is seen by the nuclear instrumentation system;

(2) Asymmetric power distribution as seen on:

- *(A)* the Out of Core Neutron Detectors,
- *(B)* the Core Exit Thermocouples,
- *(C)* the Power Distribution Monitoring System (PDMS);

(3) Rod at bottom signal;

(4) Rod deviation alarm (control banks only);

(5) Rod position indication;

Misaligned RCCAs are detected by:

(1) Asymmetric power distribution as seen on:

- *(A)* the Out of Core Neutron Detectors,
- *(B)* the Core Exit Thermocouples,
- *(C)* the Power Distribution Monitoring System (PDMS);

(2) Rod deviation alarm (control banks only);

(3) Rod position indicators;

The resolution of the rod position indicator channel is $\pm$ 12 steps. Deviation of any RCCA from its group by twice this distance (24 steps) will not cause power distributions worse than the design limits. The deviation alarm alerts the operator to rod deviation with respect to group demand position in excess of 12 steps. If the rod deviation alarm is not operable, the operator is required to take action as required by the Technical Specifications.

If one or more rod position indicator channels should be out of service, detailed plant instructions are followed to assure the alignment of the non-indicated RCCAs. The operator is also required to take action as required by the Technical Specifications. Plant instructions call for use of the Power Distribution Monitoring System to confirm indication of assembly misalignment.

15.2.3.2 Analysis of Effects and Consequences

Method of Analysis

(a) One or More Dropped RCCAs from the Same Group

For evaluation of the dropped RCCA event, the transient system response is calculated using the LOFTRAN[5] code. The 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.

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-01 code. The transient response, nuclear peaking factor analysis, and DNB design basis confirmation are performed in accordance with the methodology described in Section 4.4.3.4.

Results

(a) One or More Dropped RCCAs

Single or multiple dropped RCCAs within the same group result in a negative reactivity insertion. Power may be reestablished either by reactivity feedback or control bank withdrawal. Manual rod control (or with control stops) cases are bounded by automatic control because the reactivity insertions can only result from reactivity feedback and no power overshoot caused by control bank withdrawal can occur.

For a dropped RCCA event in the automatic rod control mode, the rod control system detects the drop in power and initiates control bank withdrawal. Power overshoot may occur due to this action by the automatic rod controller after which the control system will insert the control bank to restore nominal power. Figure 15.2-11 shows a typical transient response to a dropped RCCA (or RCCAs) in automatic control. Uncertainties in the initial condition are included in the DNB evaluation as described in Reference [13].

For evaluation of the dropped rod event, transient system conditions at the limiting point in the transient (i.e., statepoints) are calculated. No credit for any direct trip due to the dropped rod(s) is taken in the analysis.[13] The analysis also assumes no automatic power reduction features are actuated by the dropped rod(s). The statepoints are provided for conditions which cover the range of reactivity parameters expected to occur during core life. The minimum calculated pre rod drop hot channel factor is verified to be greater than the design value for each core cycle, demonstrating that in all cases, the minimum DNBR remains above the limiting value.

(b) Dropped RCCA Bank

A dropped RCCA bank typically results in a reactivity insertion greater than 500 pcm. The transient will proceed as described in part “a” above. The statepoint hot channel factor is used along with the transient statepoints and the dropped rod limit lines to confirm that the DNB design basis is met following a dropped rod event with no direct trip due to the dropped rods and no automatic power reduction features.

(c) Statically Misaligned RCCA

The most severe misalignment situations with respect to DNBR at significant power levels arise from cases in which one RCCA is fully inserted, or where bank D is fully inserted with one RCCA fully withdrawn. Multiple-independent alarms, including a bank insertion limit alarm, alert the operator well before the postulated conditions are approached. The bank can be inserted to its insertion limit with any one assembly fully withdrawn without the DNBR falling below the limit value.

The insertion limits in the Technical Specifications may vary from time to time depending on a number of limiting criteria. It is preferable, therefore, to analyze the misaligned RCCA case at full power for a position of the control bank as deeply inserted as the criteria on minimum DNBR and power peaking factor will allow. The full power insertion limits on control bank D must then be chosen to be above that position and will usually be dictated by other criteria. Detailed results will vary from cycle to cycle depending on fuel arrangements.

For this RCCA misalignment, with bank D inserted to its full power insertion limit and one RCCA fully withdrawn, DNBR does not fall below the limiting value. This case is analyzed assuming the initial reactor power, pressure, and RCS temperatures are at their nominal values including uncertainties but with the increased radial peaking factor associated with the misaligned RCCA.

DNB calculations have not been performed specifically for RCCAs missing from other banks; however, power shape calculations have been done as required for the RCCA ejection analysis. Inspection of the power shapes shows that the DNB and peak kW/ft situation is less severe than the bank D case discussed above assuming insertion limits on the other banks equivalent to a bank D full-in insertion limit.

For RCCA misalignments with one RCCA fully inserted, the DNBR does not fall below the limiting value. This case is analyzed assuming the initial reactor power, pressure, and RCS temperatures are at their nominal values, including uncertainties but with the increased radial peaking factor associated with the misaligned RCCA.

Violation of the DNB design basis does not occur for the RCCA misalignment incident and thus the ability of the primary coolant to remove heat from the fuel rod is not reduced. The peak fuel temperature corresponds to a linear heat generation rate based on the radial peaking factor penalty associated with the misaligned RCCA and the design axial power distribution. The resulting linear heat generation is well below that which would cause fuel melting.

15.2.3.3 Conclusions

For cases of dropped RCCAs or dropped banks, the DNBR remains greater than the limit value; therefore, the DNB design basis is met.

For all cases of any RCCA fully inserted, or bank D inserted to its rod insertion limits with a single RCCA in that bank fully withdrawn (static misalignment), the DNBR remains greater than the limiting value.

15.2.4 UNCONTROLLED BORON DILUTION

15.2.4.1 Identification of Causes and Accident Description

Reactivity can be added to the core by feeding primary grade water into the RCS via the reactor makeup portion of the CVCS. Boron dilution is a manual operation under strict administrative controls with procedures calling for a limit on the rate and duration of dilution. The primary causes of an inadvertent boron dilution event are the opening of the primary water control valve and failure of the blend system either by controller or mechanical failure. The CVCS, including the blend 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.

Inadvertent dilution from reactor water make-up can be readily terminated by closing the control valve. All expected sources of dilution may be terminated by closing isolation valves FCV-62-128 and FCV-62-144. In order for makeup water to be added to the RCS at pressure, at least one charging pump must be running in addition to a primary makeup water pump. The rate of addition of unborated makeup water to the

RCS when it is not at pressure is limited by the capacity of the primary water makeup pumps. Normally, only one primary water supply pump is operating while the other is on standby. With the RCS at pressure, the maximum delivery rate is limited by the control valve.

The boric acid from the boric acid tank is blended with primary grade water in the blender and the composition is determined by the preset flow rates of boric acid and primary grade water on the control board. In order to dilute, two separate operations are required:

(1) The operator must switch from the automatic makeup mode to the dilute or alternate dilute mode.

(2) The start handswitch must be actuated.

Failure to carry out either of these actions prevents the initiation of dilution. During normal operation the operator may add borated water to the RCS by blending boric acid from the boric acid storage tanks with primary grade water. This requires the operator to determine the concentration of the addition and setting the blended flow rate and the boric acid flow rate. The makeup controller will then limit the sum of the boric acid flow rate and primary water flow rate to the blended flow rate.

The status of the RCS makeup is continuously available to the operator by:

(a) Indication of the boric acid and blended flow rates,

(b) CVCS, boric acid, and primary water pump and valve status lights,

(c) Audible clicker on primary water addition

(d) Deviation alarms if the boric acid or blended flow rates deviate from the preset values

(e) Source range neutron flux – when the reactor is subcritical

  (1) High flux at shutdown alarm

  (2) Indicated source range neutron flux count rates,

  (3) Audible source range neutron flux count rate, and

  (4) Source range neutron flux – alarm on increase in base count rate above variable setpoint

(f) “Boron Dilution” alert alarms

  (1) VCT high level

  (2) Source range neutron flux increase

Primary water inadvertently added to the RCS via the charging system is a mass addition to the RCS. As primary water is added through the charging system, an equal amount of water is no longer being removed from the VCT. When this occurs, VCT level will increase. The system is designed to automatically divert water to the hold-up tank to prevent overfilling the VCT. A signal from redundant high VCT level switches result in a main control room alarm and lighting of an annunciator window. The alarm setpoint is the same level as when the divert valve starts to open. The divert valve will not fully open until VCT level reaches 93%. Thus letdown flow will not be diverted to the holdup tank prior to the alarm on high VCT level. The FSAR for Unit 1 and Unit 2 have described the high flux at shut down alarm and stated that the alarm set point is maintained within 1/2 decade of the source flux level. Following reactor shutdown, when in the hot standby, hot shutdown, or subsequently the cold shutdown condition, and once below the P-6 interlock setpoint, and 104 counts per second, the high flux at shutdown alarm setting is automatically adjusted downward as the count rate reduces. The actual setpoint is maintained at approximately three times background rather than at ½ decade above background as currently described in the FSAR. In addition to the high VCT level alarm set at 63% level, there is a high-high level alarm if the VCT level exceeds 93%.

15.2.4.2 Analysis of Effects and Consequences

15.2.4.2.1 Method of Analysis

Boron dilutions during refueling, cold shutdown, hot shutdown, hot standby, startup, and power operation are considered in this analysis. Table 15.2-1 contains the time sequence of events for this accident.

15.2.4.2.2 Dilution During Refueling

An uncontrolled boron dilution accident cannot occur during refueling. This accident is prevented by administrative controls which isolate the RCS from the potential source of unborated water.

Various combinations of valves will be closed during refueling operations. These valves will block the flow paths which could allow unborated makeup to reach the RCS. Any makeup which is required during refueling will be borated water supplied from the refueling water storage tank (RWST) by the RHR pumps. The operating procedures specify the various valve combinations.

15.2.4.2.3 Dilution During Cold Shutdown

In this mode, the plant is being taken from a long-term mode of operation (refueling) to a short term mode of operation (hot shutdown). Typically, the plant is maintained in the cold shutdown mode when reduced RCS inventory is necessary or ambient temperatures are required. The water level can be dropped to the mid-plane of the hot leg for maintenance work that requires the steam generators to be drained. Throughout the cycle, the plant may enter cold shutdown if reduced temperatures are required in containment or as the result of a Technical Specification action statement. The plant is maintained in cold shutdown at the beginning of the cycle for start-up testing of certain systems. Dilution with reduced inventory cannot occur due to administrative controls which isolate the RCS from the potential source of diluted water prior to terminating flow from the RCPs and initiating flow via the RHR system. Conditions used for the analysis are as follows:

(1) At operating temperature (between 68°F and 200°F) and pressure, dilution flow is limited by the maximum delivery capacity of one primary water pump, 150 gpm.

(2) A minimum RCS water volume of 8,451 ft^3. This corresponds to the active RCS volume excluding the pressurizer and the reactor vessel upper head.

(3) The initial boron concentration is 1,302 ppm, which corresponds to a concentration that maintains the reactor subcritical by the required shutdown margin (1.0 $\%\Delta\rho$), assuming all RCCAs inserted except for the most reactive RCCA.

(4) A conservative, maximum boron concentration at which the reactor will return to critical with all RCCAs inserted except for the most reactive RCCA, at the most reactive cycle burnup time without xenon, is 1,194 ppm. The 108 ppm change from the initial condition noted above is a conservative minimum value.

(5) Operator notification occurs via a high VCT level alarm with a setpoint of 68.1% span (including uncertainties). The alarm time is a function of the minimum letdown flow rate, which is 75 gpm.

15.2.4.2.4 Dilution During Hot Shutdown

In this mode, the plant is being taken from a short-term mode of operation (cold shutdown) to a long term mode of operation (hot standby). Typically, the plant is maintained in the hot shutdown mode to achieve plant heatup before entering hot standby. The plant is maintained in this mode at the beginning of cycle for startup testing of certain systems. Throughout the cycle, the plant will enter hot shutdown if reduced temperatures are required in containment or as a result of a Technical Specification action statement. In hot shutdown, primary coolant forced flow is provided by at least one Reactor Coolant Pump (RCP). Conditions used for the analysis are as follows:

(1) At operating temperature (200°F to 350°F) and pressure, dilution flow is limited by the maximum delivery capacity of one primary water pump, 150 gpm.

(2) A minimum RCS water volume of 8,451 ft^3. This corresponds to the active RCS volume excluding the pressurizer and the reactor vessel upper head.

(3) The initial boron concentration is 1,348 ppm, which corresponds to a concentration that maintains the reactor subcritical by the required shutdown margin (1.6 %$\Delta\rho$), assuming all RCCAs inserted except for the most reactive RCCA.

(4) A conservative, maximum boron concentration at which the reactor will return to critical with all RCCAs inserted except for the most reactive RCCA, at the most reactive cycle burnup time without xenon, is 1,165 ppm. The 183 ppm change from the initial condition noted above is a conservative minimum value.

(5) Operator notification occurs via a high VCT level alarm with a setpoint of 68.1% span (including uncertainties). The alarm time is a function of the minimum letdown flow rate, which is 75 gpm.

15.2.4.2.5 Dilution During Hot Standby

In this mode, the plant is being taken from one short-term mode of operation (hot shutdown) to another (startup). The plant is maintained in hot standby at the beginning of cycle for startup testing of certain systems and to achieve plant heatup before entering the startup mode and going critical. During operation of the cycle, the plant will enter this mode following a reactor trip or as the result of a Technical Specification action statement. During hot standby, all reactor pumps may not be in operation. In an effort to balance the heat loss through the RCS and the heat removal of the steam generators, one or more pumps may be shut off to decrease heat input into the system. The more limiting hot standby dilution scenario is with the control rods not withdrawn and the reactor shut down by boron to the Technical Specifications minimum requirement for this mode. Conditions used for the analysis are as follows:

(1) At operating temperature (350°F to 557°F) and pressure, dilution flow is limited to 160 gpm by the high charging flow alarm (including uncertainties). Any flow rate greater than this will result in an immediate alarm and ample operator action time.

(2) A minimum RCS water volume of 8,451 ft^3. This corresponds to the active RCS volume excluding the pressurizer and the reactor vessel upper head.

(3) The initial boron concentration is 1,300 ppm, which corresponds to a concentration that maintains the reactor subcritical by the required shutdown margin (1.6 %$\Delta\rho$), assuming all RCCAs inserted except for the most reactive RCCA.

(4) A conservative, maximum boron concentration at which the reactor will return to critical with all RCCAs inserted except for the most reactive RCCA, at the most reactive cycle burnup time without xenon, is 1,100 ppm. The 200 ppm change from the initial condition noted above is a conservative minimum value.

(5) Operator notification occurs via a high VCT level alarm with a setpoint of 68.1% span (including uncertainties). The alarm time is a function of the minimum letdown flow rate, which is 75 gpm.

15.2.4.2.6 Dilution During Startup

In this mode, the plant is being taken from one long-term mode of operation (hot standby) to another (power). Typically, the plant is maintained in the startup mode only for the purpose of startup testing at the beginning of each cycle. During this mode of operation, rod control is in manual. All normal actions required to change power level, either up or down, require operator initiation. Conditions used for the analysis are as follows:

(1) At operating temperature and pressure, dilution flow is limited to 235 gpm.

(2) A minimum RCS water volume of 8,451 ft^3. This corresponds to the active RCS volume excluding the pressurizer and the reactor vessel upper head.

(3) The initial boron concentration is 1,600 ppm, which corresponds to a critical, hot zero power condition with the control rods at the rod insertion limits providing a shutdown margin of 1.6%.

(4) The critical boron concentration following reactor trip is 1,400 ppm, which is a conservative maximum value corresponding to a hot zero power condition with all RCCAs inserted except for the most reactive RCCA at the most-reactive cycle burnup time without xenon. The 200 ppm change from the initial condition noted above is a conservative minimum value.

15.2.4.2.7 Dilution at Power

In this mode, the plant may be operated in either automatic or manual rod control. Conditions used for the analysis are as follows:

(1) At operating temperature and pressure, dilution flow is limited to 235 gpm.

(2) A minimum RCS water volume of 8,451 ft^3. This corresponds to the active RCS volume excluding the pressurizer and the reactor vessel upper head.

(3) The initial boron concentration is 1,500 ppm, which corresponds to a hot full power condition with the control rods at the rod insertion limits providing a shutdown margin of 1.6%.

(4) The critical boron concentration following reactor trip is 1,250 ppm, which is a conservative maximum value corresponding to a hot zero power condition with all RCCAs inserted except for the most reactive RCCA at the most-reactive cycle burnup time without xenon. The 250 ppm change from the initial conditions noted above is a conservative minimum value.

15.2.4.3 Conclusions

15.2.4.3.1 For Dilution During Refueling

Dilution during refueling cannot occur due to administrative controls (see Section 15.2.4.2). 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 control room. In addition, a source range high flux level is alarmed in the control room. The count rate increase is proportional to the subcritical multiplication factor.

15.2.4.3.2 For Dilution During Startup

This mode of operation is a transitory operational mode in which the operator intentionally dilutes and withdraws control rods to take the plant critical. During this mode, the plant is in manual control with the operator required to maintain a high awareness of the plant status. For a normal approach to criticality, the operator must manually initiate a limited dilution and subsequently manually withdraw the control rods, a process that takes several hours. The Technical Specifications require that the operator determine the estimated critical position of the control rods prior to approaching criticality, thus assuring that the reactor does not go critical with the control rods below the insertion limits. Once critical, the power escalation must be sufficiently slow to allow the operator to manually block the source range reactor trip after receiving P-6 from the intermediate range.

The accidental dilution increase causes a more rapid power escalation such that insufficient time would be available following P-6 to manually block the source range reactor trip. Failure to perform this manual action results in a reactor trip and immediate shutdown of the reactor. Continued dilution decreases the shutdown margin such that criticality could eventually be regained.

For dilution during startup, there are more than 15 minutes available for operator action from the time of alarm (reactor trip on source range high flux) to loss of shutdown margin.

15.2.4.3.3 For Dilution Following Reactor Shutdown

In cold shutdown, hot shutdown and hot standby, the reactor operators are relied upon to detect and recover from an inadvertent boron dilution event. Numerous alarms from the chemical and volume control system, the reactor makeup water system and the nuclear instrumentation system are available to provide assistance to the reactor operator in the detection of an inadvertent boron dilution event. In the analyses of the event initiated from these modes, the high Volume Control Tank (VCT) level alarm with an analysis setpoint of 68.1% span is modeled and provides the operator with timely indication that an event is occurring. The analyses have demonstrated that the reactor operators have at least 15 minutes in which to initiate actions to terminate the dilution and initiate boration of the RCS from the time of the alarm to loss of shutdown margin.

15.2.4.3.4 For Dilution During Full Power Operation

With the reactor in automatic rod control, the power and temperature increase from boron dilution results in insertion of the control rods and a decrease in the available shutdown margin. The rod insertion limit alarms (LOW and LOW-LOW settings) alert the operator that a dilution event is in progress. There are more than 15 minutes available for operator action from the time of alarm (LOW-LOW rod insertion limit) to loss of shutdown margin.

With the reactor in manual control and no operator action taken to terminate the transient, the power and temperature rise will cause the reactor to reach the overtemperature ΔT trip setpoint resulting in a reactor trip. The boron dilution transient in this case is essentially the equivalent to an uncontrolled RCCA bank withdrawal at power. The reactivity insertion rate for a boron dilution accident is conservatively estimated to be about 0.6 pcm/sec, which yields the longest time to reach reactor trip. There are more than 15 minutes available for operator action from the time of alarm (overtemperature ΔT) to loss of shutdown margin.

For all cases, the reactor will be in a stable condition following termination of the dilution flow. The operator will then initiate reboration to recover the shutdown margin, using the CVCS. If the reactor has tripped, operating procedures call for operator action to control pressurizer level using the CVCS and to maintain steam generator level through control of the main or auxiliary feedwater system. Any action required of the operator to maintain the plant in a stabilized condition are in a time frame in excess of ten minutes following reactor trip.

15.2.5 PARTIAL LOSS OF FORCED REACTOR COOLANT FLOW

15.2.5.1 Identification of Causes and Accident Description

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

Normal power for the reactor coolant pumps is supplied through individual electrical boards from a transformer connected to the generator. When a generator trip occurs, the boards are automatically transferred to a transformer supplied from external power lines, and the pumps will continue to provide forced coolant flow to the core. Following a turbine trip where there are no electrical faults or a thrust bearing failure which requires tripping the generator from the network, the generator remains connected to the network for approximately 30 seconds. The reactor coolant pumps remain connected to the generator thus ensuring full flow for approximately 30 seconds after the reactor trip before any transfer is made. Since each pump is on a separate board, a single board fault would not result in the loss of more than one pump.

The necessary protection against a partial loss of coolant flow accident is provided by the low primary coolant flow reactor trip which is actuated by two out of three low flow signals in any reactor coolant loop.

Above approximately 48% power (Permissive 8), low flow in any loop will actuate a reactor trip. Between approximately 10% power (Permissive 7) and the power level corresponding to Permissive 8, low flow in any two loops will actuate a reactor trip.

Following a RCP trip, if the cause of the shutdown is immediately resolved, a restart of the pump may be attempted if reactor power is reduced to less than 10% and there is ample time to meet the Technical Specifications Limiting Condition for Operation (LCO) action statement.

15.2.5.2 Analysis of Effects and Consequences

Method of Analysis

A partial loss of flow involving the loss of one pump with four loops in operation has been analyzed.

This transient is analyzed by three digital computer codes. First the LOFTRAN[5] Code is used to calculate the loop and core flow transients, the time of reactor trip based on the loop flow transient the nuclear power transient, and the primary system pressure and coolant temperature transients. The FACTRAN Code[4] is then used to calculate the heat flux transient based on the nuclear power and flow from LOFTRAN. Finally, the VIPRE-01 Code (see Section 4.4.3.4) is used to calculate the DNBR during the transient based on the heat flux from FACTRAN and flow from LOFTRAN. The DNBR transient presented represents the minimum of the typical or thimble cell.

Initial Conditions

Initial reactor power, pressure, and RCS temperature are assumed to be at their nominal values. Uncertainties in the initial conditions are included in the safety analysis DNBR limit as described in Reference [18]. The minimum measured flow value is also included.

Reactivity Coefficients

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

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.

Results

The calculated sequence of events for the limiting case analyzed is shown on Table 15.2-1. Figures 15.2-12, 15.2-13, and 15.2-15 through 15.2-17 show the transient response for the loss of power to one reactor coolant pump with four loop operation. The DNBR never goes below the design basis limit.

Following reactor trip, the plant will come to a stabilized condition at hot standby with one or more reactor coolant pumps in operation. Normal operating procedures may then be followed. The operating procedures would call for operator action to control RCS boron concentration and pressurizer level using the CVCS, and to maintain steam generator level through control of the main or auxiliary feedwater system. Any action required of the operator to maintain the plant in a stabilized condition will be in a time frame in excess of ten minutes following reactor trip.

15.2.5.3 Conclusions

The analysis has demonstrated for the partial loss of forced reactor coolant flow that the DNBR will not decrease below the design basis limit at any time during the transient.

15.2.6 Startup of an Inactive Reactor Coolant Loop

15.2.6.1 Identification of Causes and Accident Description

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

Starting an idle reactor coolant pump without first bringing the inactive loop hot leg temperature close to the core inlet temperature would result in the injection of cold water into the core. This injection would cause a reactivity insertion and subsequent power increase due to the moderator density reactivity feedback effect.

Based on the expected frequency of occurrence, the Startup of an Inactive Loop event is classified as a condition II event (an incident of moderate frequency) as defined by the American Nuclear Society Nuclear Safety Criteria for the Design of Stationary PWR Plants.

Sequence of Events and System Operation

Following the startup of the inactive reactor coolant pump, the flow in the inactive loop will accelerate to full flow in the forward direction over a period of several seconds. Since the Technical Specifications require all reactor coolant pumps to be operating while in modes 1 and 2, the maximum initial core power level for the Startup of an

Inactive Loop transient is approximately 0 MWt. Under these conditions, there can be no significant reactivity insertion because the RCS is initially at a nearly uniform temperature. Furthermore, the reactor will initially be subcritical by the Technical Specification requirement. Thus, there will be no increase in core power, and no automatic or manual protective action is required. [This analysis is normally run at high power levels for (N-1) loop operation plants. WBN design does not currently include this operating configuration.]

15.2.6.2 Conclusions

The Startup of an Inactive Loop event results in an increase in reactor vessel flow while the reactor remains in a subcritical condition. No analysis is required to show that the minimum DNBR limit is satisfied for this event.

Startup of an RCP at less than 10% power is allowed as a corrective measure taken during a recovery phase after a partial loss of forced reactor coolant event, and is not the same as the startup of an inactive loop. Refer to Section 15.2.5.1.

15.2.7 LOSS OF EXTERNAL ELECTRICAL LOAD AND/OR TURBINE TRIP

15.2.7.1 Identification of Causes and Accident Description

Major load loss on the plant can result from loss of external electrical load or from a turbine trip. For either case offsite power is available for the continued operation of plant components such as the reactor coolant pumps. This analysis, along with the Loss of Normal Feedwater (Section 15.2.8) and Complete Loss of Forced Reactor Coolant Flow (Section 15.3.4) addresses the case of loss of offsite power to the station auxiliaries (Section 15.2.9).

For a turbine trip, the reactor will be tripped directly (unless below approximately 50% power) from a signal derived from the turbine autostop oil pressure or turbine throttle valve position. The automatic steam dump system will accommodate the excess steam generation. Reactor coolant temperatures and pressure do not significantly increase if the steam dump system and pressurizer pressure control system are functioning properly. If the turbine condenser is not available, the excess steam generation will be dumped to the atmosphere. Additionally, main feedwater flow will be lost if the turbine condenser is not available. For this situation feedwater flow will be maintained by the auxiliary feedwater system.

For a loss of external electrical load without subsequent turbine trip, no direct reactor trip signal would be generated. A continued steam load of approximately 5% would exist after total loss of external electrical load because of the steam demand of plant auxiliaries.

Onsite power supplies plant auxiliaries during plant operation, e.g., the reactor coolant pumps. Safeguards loads are supplied from offsite power or, alternatively, from emergency diesels. Reactor protection system equipment is supplied from the 120V AC vital instrument power supply system, which in turn is supplied from the vital inverters; the inverters are supplied from a DC bus energized from vital batteries or rectified AC from safeguards buses. Thus, for postulated loss of load and subsequent turbine generator overspeed, any overfrequency condition is not seen by safety related pump motors, reactor protection system equipment, or other safeguards loads. Any increased frequency to the reactor coolant pump motors will result in slightly increased flowrate and subsequent additional margin to safety limits.

Should a safety limit be approached, protection would be provided by high pressurizer pressure and overtemperature ΔT trip. Power and frequency relays associated with the reactor coolant pump provide no additional safety function for this event. Following a complete loss of load the maximum turbine overspeed would be approximately 8% to 9%, resulting in an overfrequency of less than 6 Hz. This resulting overfrequency is not expected to damage the sensors (non-NSSS) in any way. However, it is noted that frequent testing of this equipment is required by the Technical Specifications. Any degradation in their performance could be ascertained at that time.

In the event the steam dump valves fail to open following a large loss of load, the steam generator safety valves may lift and the reactor may be tripped by the high pressurizer pressure signal, the high pressurizer water level signal, the overtemperature ΔT signal or the low-low steam generator water level signal. The sudden reduction in steam flow will result in an increase in pressure and temperature in the steam generator shell. As a result, the heat transfer rate in the steam generator is reduced causing the reactor coolant temperature to rise, which causes coolant expansion, pressurizer insurge, and RCS pressure rise. The pressurizer safety valves and steam generator safety valves are, however, sized to protect the RCS and steam generator against overpressure for all load losses without assuming the operation of the steam dump system, pressurizer spray, pressurizer power-operated relief valves, automatic rod cluster control assembly control nor direct reactor trip on turbine trip.

The steam generator safety valve capacity is sized to remove the steam flow at the Engineer Safety Features Rating (105% of steam flow at rated power) from the steam generator without exceeding 110% of the steam system design pressure. The pressurizer safety valve capacity is sized based on a complete loss of heat sink with the plant initially operating at the maximum calculated turbine load along with operation of the steam generator safety valves. The pressurizer safety valves are then able to maintain the RCS pressure within 110% of the RCS design pressure without direct or immediate reactor trip action.

A more complete discussion of overpressure protection can be found in Reference [9].

15.2.7.2 Analysis of Effects and Consequences

Method of Analysis

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.

The total loss of load transients are analyzed by employing the detailed digital computer program LOFTRAN[5], which is described in Section 15.1. The program simulates the neutron kinetics, RCS, presssurizer, pressurizer relief and safety valves, pressurizer spray, steam generator, and variables including temperatures, pressures, and power level.

Typical assumptions are:

(1) Initial Operating Conditions

(a) DNB case - The initial reactor power, pressurizer pressure, and RCS temperature are assumed at their nominal values, consistent with steady-state full-power operation, in accordance with the RTDP methodology [Reference 18]. Minimum measured RCS flow is also assumed for the DNB evaluation case in accordance with the RTDP methology.

(b) RCS Overpressure Case - The initial reactor power and RCS temperatures are assumed at their maximum values consistent with steady-state full-power operation including allowances for calibration and instrument errors. The initial RCS pressure is assumed at a minimum value (pressurizer pressure - 50 psi allowance for steady-state fluctuations and measurement error) consistent with steady-state full-power operation including allowances for calibration and instrument errors. Thermal design RCS flow is assumed, ensuring minimum primary-to-secondary heat transfer. This results in the maximum power difference for this load loss, and the minimum margin to core protection limits at the initiation of the accident.

(2) Moderator and Doppler Coefficients of Reactivity - the total loss of load is analyzed assuming beginning-of-life conditions. The least negative moderator temperature coefficients at beginning-of-life is used. A conservatively large (absolute value) Doppler power coefficient is used for all cases.

(3) Reactor Control - it is conservatively assumed that the reactor is in manual control.

(4) 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 setpoints where steam release through safety valves occurs to limit the secondary steam pressure.

(5) Pressurizer Spray and Power-Operated Relief Valves:

(a) DNB Case - Full credit is taken for the effect of pressurizer spray and power-operated relief valves in reducing or limiting the coolant pressure.

(b) 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.

(6) 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 each case, in Figures 15.2-19 through 15.2-26. The calculated sequence of events for the accident is shown in Table 15.2-1.

Figures 15.2-19 and 15.2-22 show the transient responses for the total loss of steam load at beginning-of-life with a zero moderator temperature coefficient 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 OTΔT signal trip channel. The minimum DNBR is well above the limiting value.

The total loss of load accident was also studied assuming the plant to be initially operating at full power, including uncertainty, with 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. Figures 15.2-23 through 15.2-26 show the RCS overpressurization transient with a zero moderator coefficient. In this case the pressurizer safety valves and main steam safety valves are actuated and maintain the system pressure below 110% of their respective design values.

Reference [9] presents additional results of analysis for a complete loss of heat sink including loss of main feedwater. This analysis shows the overpressure protection that is afforded by the pressurizer and steam generator safety valves.

15.2.7.3 Conclusions

Results of the analyses, including those in Reference [9], show that the plant design is such that a total loss of external electrical load without a direct or immediate reactor trip presents no hazard to the integrity of the RCS or the main steam system. Pressure relieving devices incorporated in the two systems are adequate to limit the maximum pressures to within the design limits.

The integrity of the core is maintained by operation of the reactor protection system, i.e., the DNBR will be maintained above the limiting value.

15.2.8 LOSS OF NORMAL FEEDWATER

15.2.8.1 Identification of Causes and Accident Description

A loss of normal feedwater (from pump failures, valve malfunctions, or loss of offsite AC power) results in a reduction in capability of the secondary system to remove the heat generated in the reactor core. If the reactor were not tripped during this accident, core damage would possibly occur from a sudden loss of heat sink. If an alternative 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. Significant loss of water from the RCS could conceivably lead to core damage. Since the plant is tripped well before the steam generator heat transfer capability is reduced, the primary system variables never approach a DNB condition.

The following provides the necessary protection against a loss of normal feedwater:

(1) Reactor trip on low-low water level in any steam generator.

(2) Two motor driven auxiliary feedwater pumps which are started on:

- *(a)* Low-low level in any steam generator
- *(b)* Trip of both turbine driven main feedwater pumps
- *(c)* Any safety injection signal
- *(d)* Loss of offsite power
- *(e)* Manual actuation

(3) One turbine driven auxiliary feedwater pump is started on:

- *(a)* Low-low level in any two steam generators
- *(b)* Trip of both turbine driven main feedwater pumps
- *(c)* Any safety injection signal
- *(d)* Loss of offsite power
- *(e)* Manual actuation

Refer to Section 10.4.9 for the design of the auxiliary feedwater system.

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

The analysis shows that, following a loss of normal feedwater, the auxiliary feedwater system is capable of removing the stored and residual heat, thus preventing water relief from the pressurizer and subsequently a loss of water from the reactor core.

15.2.8.2 Analysis of Effects and Consequences

Method of Analysis

A detailed analysis using the LOFTRAN[5] Code 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.

Two cases are examined for a loss of normal feedwater event. The first is the case where offsite ac power is maintained, and the second is the case where offsite ac power is lost, which results in reactor coolant pump coastdown as described in Section 15.2.5.2.

The case where offsite ac power is lost is limiting with respect to water relief from the pressurizer and loss of water from the reactor core.

Assumptions

(1) The initial steam generator water level (in all steam generators) at the time of reactor trip is at a conservatively low level. The low-low steam generator level trip setpoint is conservatively assumed to 0.0% of narrow range span.

(2) The plant is initially operating at 102% of the Nuclear Steam Supply System design rating. The heat added to the RCS by the reactor coolant pumps is assumed, as applicable.

(3) The core residual heat generation is based on the 1979 version of ANS 5.1[14] based upon long term operation at the initial power level. The decay of U-238 capture products is included as an integral part of this expression.

(4) A heat transfer coefficient in the steam generator associated with RCS natural circulation for the case where offsite power is lost.

(5) Two motor-driven auxiliary feedwater pumps are available one minute after the accident. (Failure of the turbine-driven auxiliary feedwater pump is assumed since this failure provides minimum auxiliary feedwater flow.)

(6) Constant auxiliary feedwater flow equal to 820 gpm from the two motor-driven auxiliary feedwater pumps is delivered to four steam generators.

(7) Auxiliary feedwater temperature is 120°F.

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

(9) LONF and LOOP cases run with both positive and negative initial average RCS temperature uncertainty (±6°F) and pressurizer pressure uncertainty (+70/-50 psi) have indicated that the case with negative temperature and pressurizer pressure uncertainties is conservative in terms of maximum pressurizer water volume.

(10) The pressurizer heaters and sprays are assumed operable during the transient. Heaters cause expansion of the pressurizer water while sprays reduce pressurizer pressure allowing a greater coolant in surge. Both scenarios conservatively maximize the pressurizer water inventory.

(11) The CVCS is not assumed to function for this event as operation of the system is a benefit with respect to long term core decay heat removal. Note, however, that charging pump operation will increase the reactor coolant inventory if the letdown isolation valve closes due to subsequent loss of instrument air. This scenario was examined to determine if the operators have sufficient time to terminate the net mass addition to the reactor coolant system to preclude water relief through the pressurizer safety valves. The heaters were assumed to operate as-designed on pressure effects.

The loss of normal feedwater analysis is performed to demonstrate the adequacy of the reactor protection and engineered safeguards systems (e.g., the auxiliary feedwater system) in removing long term decay heat and preventing excessive heatup of the RCS with possible loss of RCS water.

As such, the assumptions used in this analysis are designed to minimize the energy removal capability of the system and to maximize the possibility of water relief from the coolant system by maximizing the coolant system expansion, as noted in the assumptions listed above.

One such assumption is the loss of external (offsite) ac power. This assumption results in coolant flow decay down to natural circulation conditions reducing the steam generator heat transfer coefficient. Following a loss of offsite ac power, the first few seconds of a loss of normal feedwater transient will be virtually identical to the transient response (including DNBR and neutron flux versus time) presented in Section 15.3.4 for the complete loss of forced reactor coolant flow incident.

A separate case was run with charging flow initiated on a loss of offsite power signal. The addition of charging flow would provide a benefit regarding primary side temperature increase (post-trip heatup), and hence should not be credited to demonstrate the heat removal capacity of the auxiliary feedwater system. Therefore, the loss of normal feedwater event as presented herein is appropriate in terms of demonstrating auxiliary feedwater system heat removal capacity.

Further, this case did not result in the filling of the pressurizer prior to ten minutes following the initiation of the event. Thus, there is sufficient time available for the operator to terminate the net mass addition to the reactor coolant system to preclude water relief through the pressurizer safety valves. This case is analyzed similar to the inadvertent operation of the emergency core cooling system (ECCS) event, where operator action is required to terminate the ECCS flow, thereby, precluding water relief through the pressurizer safety valves. Also, as mentioned above, the loss of normal feedwater cases bound this case relative to demonstrating the long-term heat removal capacity of the auxiliary feedwater system.

Additional sensitivities were performed to determine if it was more conservative to model the pressurizer power operated relief valves as operable or inoperable.

Results

Figures 15.2-27a through 15.2-27i show the significant plant parameter transients following a loss of normal feedwater where offsite power is lost. The calculated sequence of events for this accident is listed in Table 15.2-1.

Following the reactor and turbine trip from full load, the water level in the steam generators will fall 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. One minute following the initiation of the low-low level trip, both of the motor-driven auxiliary feedwater pumps are automatically started and are at full speed, reducing the rate of water level decrease.

The capacity of the auxiliary feedwater pumps 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 15.2-27g, it can be seen that at no time is there water relief from the pressurizer. If the auxiliary feed delivered is greater than that of two motor-driven pumps, if the initial NSSS power is less than 100% of the NSSS design rating plus applicable uncertainty or if the steam generator water level in one or more steam generators is above the low-low level trip point at the time of trip, then the results of this transient will be bounded by the analysis presented.

The plant will slowly approach a stabilized condition at hot standby with auxiliary feedwater removing decay heat. The plant may be maintained at hot standby or further cooled through manual control of the auxiliary feed flow. The operating procedures also call for operator action to control RCS boron concentration and pressurizer level using the CVCS and to maintain steam generator level through control of the auxiliary feedwater system. Any action required of the operator to maintain the plant in a stabilized condition is in a time frame in excess of ten minutes following reactor trip.

15.2.8.3 Conclusions

Results of the analysis show that a loss of normal feedwater does not adversely affect the core, the RCS, or the steam system since the auxiliary feedwater capacity is such that the reactor coolant water is not relieved from the pressurizer relief or safety valves, and the water level in the steam generators receiving feedwater is maintained above the tubesheets.

15.2.9 COINCIDENT LOSS OF ONSITE AND EXTERNAL (OFFSITE) AC POWER TO THE STATION - LOSS OF OFFSITE POWER TO THE STATION AUXILIARIES

A complete loss of all offsite power (no-emergency AC power) may result in the loss of all power to the plant auxiliaries, i.e., the reactor coolant pumps, condensate pumps, etc. The loss of power may be caused by a complete loss of the offsite grid accompanied by a turbine generator trip at the station, or by a loss of the onsite AC distribution system. See analysis contained in Sections 15.2.7, 15.2.8 and 15.3.4.

For a LOOP event, the Emergency Diesel Generators (EDG) are available to supply AC power to support plant safe shutdown. A Station Blackout (SBO) event differs from a LOOP in that for the SBO Unit, both Emergency Diesel Generators are lost or are not available on the SBO unit for the SBO coping period (4 hours). The non-SBO unit is assumed to have a single failure such only that one EDG is available to support safe shutdown. A SBO event is beyond the design basis for Watts Bar. Section 4.41 of WB-DC-40-64 describes the plant's ability to mitigate this multiple failure scenario as required by 10 CFR 50.63.

15.2.10 EXCESSIVE HEAT REMOVAL DUE TO FEEDWATER SYSTEM MALFUNCTIONS

15.2.10.1 Identification of Causes and Accident Description

Additions of excessive feedwater cause increases in core power by decreasing reactor coolant temperature. Such transients are attenuated by the thermal capacity of the secondary plant and of the RCS. The overpower-overtemperature protection (overtemperature ΔT, and overpower ΔT trips) prevents any power increase which could lead to a DNBR less than the limiting value.

Excessive feedwater flow could be caused by a full opening of one or more feedwater control valves due to a feedwater control system malfunction or an operator error. At power, this excess flow causes a greater load demand on the RCS due to increased subcooling in the steam generator. With the plant at no-load conditions, the addition of cold feedwater may cause a decrease in RCS temperature and thus a reactivity insertion due to the effects of the negative moderator coefficient of reactivity. Continuous addition of excessive feedwater is prevented by the steam generator high-high level trip, which closes the feedwater control and isolation valves.

15.2.10.2 Analysis of Effects and Consequences

Method of Analysis

The excessive heat removal due to a feedwater system malfunction transient is analyzed by the detailed digital computer code LOFTRAN[5]. This code simulates a multi-loop system, neutron kinetics, the 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.

Excessive feedwater addition due to a control system malfunction or operator error which allows one or more feedwater control valves to open fully is considered. The most limiting cases are as follows:

1. *a* Accidental opening of one feedwater control valve with the reactor at zero load.

   b Accidental opening of all feedwater control valves with the reactor at zero load.

2. *a* Accidental opening of one feedwater control valve with the reactor at full power.

   b Accidental opening of all feedwater control valves with the reactor at full power.

The plant response following a feedwater system malfunction is calculated with the following assumptions:

(1) Reactor at zero load

(a) A generic study performed by Westinghouse demonstrated that the consequences of a hot zero power feedwater malfunction with an increased feedwater flow rate of less that 150% of the nominal full power flow rate are non-limiting and are bounded by the hot full power feedwater malfunction. The hot zero power discussion herein is maintained for historical purposes. The reactor is assumed to be just critical in the hot shutdown condition.

(b) Both automatic and manual rod control are considered for each of the zero-power cases.

(c) For case 1a, an increase in feedwater flow to one steam generator from zero flow to 100% of the nominal single steam generator full-load flow is assumed.

For case 1b, an increase in feedwater flow to each of the four steam generators from zero flow to 90%, 11%, 11%, and 12% of nominal flow is assumed.

(d) The feedwater temperature is assumed to be at a conservatively low value of 32 °F.

(e) For case 1a, no credit is taken for the heat capacity of the steam and water in the unaffected steam generators.

(2) Reactor at full power

(a) This accident is analyzed with the RTDP as described in Reference [18]; therefore initial reactor power, pressure, and RCS temperatures

are assumed to be at their nominal values. Uncertainties in initial conditions are included in the limit DNBR as described in Reference [18].

(b) Both automatic and manual rod control are considered for each of the full-power cases. The results from the most limiting scenario are presented.

(c) For case 2a, a step increase in feedwater flow to one steam generator from nominal flow to 200% of nominal flow (for one steam generator) is assumed.

For case 2b, a step increase in feedwater flow to each of the four steam generators from nominal flow to 173%, 155%, 154%, and 157% of nominal flow is assumed.

(d) For case 2a, no credit is taken for the heat capacity of the steam and water in the unaffected steam generators.

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

(3) For both cases 1 and 2 above:

(a) The initial water level in all steam generators is at a conservatively low level for the initial conditions.

(b) No credit is taken for the heat capacity of the reactor coolant system in attenuating the resulting plant cooldown.

(c) A conservatively large moderator coefficient of reactivity that is characteristic of end-of-life core conditions is used.

Results

The cases of an accidental full opening of one or more feedwater control valves with the reactor at hot zero power (HZP) are bounded by the hot full power cases as mentioned above. Therefore, the results of the analyses are not presented.

The full-power cases (end-of-life, with automatic rod control) give the largest reactivity feedback and result in the greatest power increase. Figures 15.2-28a through 15.2-28j show the transient response for the accidental full opening of one or all four feedwater control valves with the reactor at full power. The DNBR does not drop below the limit value.

Following reactor trip and feedwater isolation, the plant will approach a stabilized condition at hot standby. Normal plant operating procedures may then be followed. The operating procedures would call for operator action to control RCS boron concentration and pressurizer level using the CVCS, and to maintain steam generator level through control of the main or auxiliary feedwater system. Any action required of the operator to maintain the plant in a stabilized condition will be in a time frame in excess of ten minutes following reactor trip.

15.2.10.3 Conclusions

Results show that the DNBRs encountered for excessive feedwater addition at power are well above the limit value.

15.2.11 Excessive Load Increase Incident

15.2.11.1 Identification of Causes and Accident Description

An excessive load increase incident is defined as a rapid increase in the steam flow that causes a power mismatch between the reactor core power and the steam generator load demand. The RCS is designed to accommodate a 10% step load increase or a 5% per minute ramp load increase in the range of 15 to 100% of full power. Any loading rate in excess of these values may cause a reactor trip actuated by the reactor protection system .

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

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

Protection against an excessive load increase accident is provided by the following RPS signals:

(1) Overpower ΔT

(2) Overtemperature ΔT

(3) Power range high neutron flux

(4) Low Pressurizer Pressure

15.2.11.2 Analysis of Effects and Consequences

Method of Analysis

Four cases are considered to demonstrate that the fuel cladding integrity will not be adversely impacted following a 10 percent step-load increase from rated load. This is shown by demonstrating that the minimum DNBR will not go below the safety analysis limit value.

(1) Reactor control in manual at beginning of life.

(2) Reactor control in manual at end of life.

(3) Reactor control in automatic at beginning of life.

(4) Reactor control in automatic at end of life.

At beginning-of-life minimum moderator feedback conditions, the core has the least negative moderator temperature coefficient of reactivity and the least-negative Doppler only power coefficient curve, and therefore, the least inherent transient response capability. A zero moderator temperature coefficient is evaluated for the minimum feedback conditions. For the end of life maximum moderator feedback conditions, the moderator temperature coefficient of reactivity has its most-negative value and the most-negative Doppler only power coefficient curve. This results in the largest amount of reactivity feedback due to changes in coolant temperature.

The effect of this transient on the minimum DNBR is evaluated by applying conservatively large deviations to the initial conditions of core power, average coolant temperature, and pressurizer pressure at the normal full power operating conditions in order to generate a limiting set of statepoints. These deviations bound the variations which could occur as a result of an excessive load increase accident and are only applied in the direction that has the most adverse impact on the DNB ratio; namely increased power and coolant temperature and decreased pressure. No credit is taken for the decrease in coolant temperature expected for cases with manual rod control and no reactor trip is assumed.

The reactor condition statepoints (temperature, pressure, and power) are compared to the conditions corresponding to operation at the safety analysis DNB limit. These limits are illustrated in the figure showing the Overpower and Overtemperature ΔT Protection setpoints (Figure 15.1-1).

Normal reactor control systems and engineered safety systems are not required to function. A conservative limit on the turbine valve opening is assumed. The analysis does not take credit for pressurizer heaters. The cases which assume automatic rod control are evaluated to ensure that the worst case in bounded. The automatic function is not required.

The RPS is assumed to be operable. However, reactor trip is not encountered for most cases due to the error allowances assumed in the setpoints. No single active failure in any system or component required for mitigation will adversely affect the consequences of this accident.

This accident is evaluated with the RTDP as described in Reference [18]. Initial reactor power, RCS pressure, and temperature are assumed to be nominal values. Uncertainties in initial conditions are included in the limit DNBR as described in Reference [18].

Results

An excessive load increase accident typically does not result in a reactor trip, and the plant soon reaches a new equilibrium condition at a higher power level based on the increased steam load.

Transients assuming manual rod control yield decreased coolant temperatures and pressures resulting from the increased heat removal. If the automatic rod control system were available, coolant average temperature would be maintained at or near the programmed value while pressure would decrease. Figures 15.2-29 through 15.2-36 show a typical transient response for each case.

A comparison of the plant conditions assuming conservatively bounding deviations in core power, average coolant temperature, and pressure to the conditions corresponding to operation at the safety analysis DNB limit indicate that the minimum DNBR remains above the limit value for each of the cases.

15.2.11.3 Conclusions

It has been demonstrated that for an excessive load increase the minimum DNBR during the transient will not be below the limiting value.

15.2.12 ACCIDENTAL DEPRESSURIZATION OF THE REACTOR COOLANT SYSTEM

15.2.12.1 Identification of Causes and Accident Description

The most severe core conditions resulting from an accidental depressurization of the reactor coolant system are associated with an inadvertent opening of a pressurizer safety valve. Note that the event is limiting for core analysis only and is not a design basis load condition for pipe stress analysis. Initially the event results in a rapidly decreasing reactor coolant system pressure which could reach the hot leg saturation pressure. At that time, the pressure decrease is slowed considerably. The pressure continues to decrease throughout the transient. The effect of the pressure decrease would be to decrease the neutron flux via the moderator density feedback but the reactor control system (if in the automatic mode) functions to maintain the power essentially constant throughout the initial stage of the transient. The average coolant temperature decreases slowly, but the pressurizer level increases until reactor trip.

The reactor will be tripped by the following reactor protection system signals:

(1) Overtemperature ΔT

(2) Pressurizer low pressure

15.2.12.2 Analysis of Effects and Consequences

Method of Analysis

The accidental depressurization transient is analyzed by employing the detailed digital computer code LOFTRAN[5]. The 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.

In calculating the DNBR, the following conservative assumptions are made:

(1) Nominal initial conditions of core power, reactor coolant temperatures. and reactor coolant pressure are assumed in accordance with the RTDP methology [18].

(2) A least negative moderator coefficient of reactivity was assumed in this analysis. The spatial effect of void due to local or subcooled boiling is not considered in the analysis with respect to reactivity feedback or core power shape. The DNB evaluation is made assuming that core power peaking factors remain constant at their design values while, in fact, the effects of local or subcooled void would have the effect of flattening the power distribution (especially in hot channels) thus increasing the DNB margin.

(3) A high (absolute value) Doppler coefficient of reactivity such that the resultant amount of positive feedback is conservatively high in order to retard any power decrease due to moderator reactivity feedback.

Results

Figure 15.2-37 illustrates the nuclear power transient following the accident. Reactor trip on overtemperature ΔT occurs as shown in Figure 15.2-37. The pressure and core average temperature versus time following the accident is given in Figure 15.2-38. The resulting DNBR never goes below its limiting value as shown in Figure 15.2-39. The calculated sequence of events for this accident is listed in Table 15.2-1.

Following reactor trip, RCS pressure will continue to fall until flow through the inadvertently opened valve is terminated. Automatic actuation of the safety injection system may occur if the pressure falls to the low pressurizer pressure SI setpoint.

RCS pressure will stabilize following operator action to terminate flow to the inadvertently opened valve; normal operating procedures may then be followed. The operating procedures call for operator action to control RCS boron concentration and pressurizer level using the CVCS and to maintain steam generator level through control of the main or auxiliary feedwater system. Any action required of the operator to stabilize the plant is in a time frame in excess of ten minutes following reactor trip.

15.2.12.3 Conclusions

The pressurizer low pressure and the overtemperature ΔT reactor protection system signals provide adequate protection against this accident, and the minimum DNBR remains in excess of the limiting value.

15.2.13 ACCIDENTAL DEPRESSURIZATION OF THE MAIN STEAM SYSTEM

15.2.13.1 Identification of Causes and Accident Description

The most severe core conditions resulting from an accidental depressurization of the main steam system are associated with an inadvertent opening of a single steam dump, relief or safety valve. The analyses performed assuming a rupture of a main steam line are given in Section 15.4.2.1.

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

The evaluation performed demonstrates that the following criterion is satisfied: assuming a stuck RCCA and a single failure in the engineered safety features (ESF), there will be no consequent damage to the fuel or RCS after reactor trip for a steam release equivalent to the spurious opening, with failure to close, of the largest of any single steam dump, relief or safety valve, with or without offsite power.

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

(1) Safety injection system actuation from any of the following:

(a) Two out of three low pressurizer pressure signals.

(b) Two out of three high containment pressure signals.

(c) Two out of three low steamline pressure signals in any steamline.

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

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

(4) Trip of the fast-acting steamline stop valves (main steam isolation valves) (designed to close in less than 6 seconds) on:

(a) Two out of four high-high containment pressure signals.

(b) Two out of three low steamline pressure signals in any steamline.

(c) Two out of three high negative steamline pressure rate signals in any steamline (below Permissive P-11).

15.2.13.2 Analysis of Effects and Consequences

Method of Analysis

The following conditions are assumed to exist at the time of an accidental depressurization of the main steam system:

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

(2) A negative moderator coefficient corresponding to the end-of-life rodded core with the most reactive rod cluster control assembly in the fully withdrawn position. The variation of the coefficient with temperature and pressure is included. The k_{eff} versus temperature at 1100 psi corresponding to the negative moderator temperature coefficient used is shown in Figure 15.2-40.

(3) Minimum capability for injection of high concentration boric acid solution corresponding to the most restrictive single failure in the safety injection system. This corresponds to the flow delivered by one charging pump delivering its full contents to the cold leg header. Low concentration boric acid must be swept from the safety injection lines downstream of the RWST prior to the delivery of high concentration boric acid (2000 ppm) which is bounded for higher concentrations to the reactor coolant loops.

(4) The evaluation considers a maximum steam flow of 247 pounds per second at 1100 psia from one steam generator with offsite power available. This is the maximum capacity of any single steam dump, relief or safety valve.

Initial hot shutdown conditions at time zero are assumed since this represents the most conservative initial condition. Should the reactor be just critical or operating at power at the time of a steam release, the reactor will be tripped by the normal overpower protection when power level reaches a trip point. Following a trip at power, the reactor coolant system contains more stored energy than at no load, 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 release before the no load conditions of reactor coolant system temperature and shutdown margin assumed in the analyses are reached.

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

(5) In computing the steam flow, the Moody Curve for fl/D = 0 is used.

(6) Perfect moisture separation in the steam generator and a tube plugging level of 10% is assumed.

(7) A thermal design flowrate of 372,400 gpm is used based on the assumption of a 10% steam generator tube plugging level and instrumentation uncertainty.

Results

Since the conditions above for an accidental depressurization of the main steam system are significantly less limiting than those for the main steam line rupture (MSLB, 15.4.2) transient from HZP conditions and since these events are analyzed utilizing similar methodology, the analysis for the MSLB transient is used to bound the accidental depressurization of the main steam system event. This approach is supported by the fact that the maximum return to power for steam release transient is much lower than that for the HZP MSLB event. Hence, minimum DNBR is not a concern under these conditions.

15.2.13.3 Conclusions

The analysis shows that the criteria stated earlier in this section are satisfied since a DNBR less than the limiting value does not exist.

15.2.14 Inadvertent Operation of Emergency Core Cooling System

This analysis was performed after the boron injection tank (BIT) and associated 900 gallons of 20,000 ppm boron were deleted from the Watts Bar design basis. Therefore, the BIT is not referred to in this section.

15.2.14.1 Identification of Causes and Accident Description

Spurious Emergency Core Cooling System (ECCS) operation at power could be caused by operator error or a false electrical actuating signal. Spurious actuation may be assumed to be caused by any of the following:

(1) High containment pressure

(2) Low pressurizer pressure (above Permissive P11)

(3) Low steamline pressure (above Permissive P11)

(4) Manual actuation

Following the actuation signal, the suction of the centrifugal charging pumps is diverted from the volume control tank to the refueling water storage tank.

The charging pumps then force concentrated (3300 ppm) boric acid solution from the RWST, through the common injection header and injection lines and into the cold leg of each reactor coolant loop. The safety injection pumps also start automatically, but provide no flow when the reactor coolant system is at normal pressure. The passive injection system and the low head system provide no flow at normal reactor coolant system pressure.

A safety injection signal normally results in a reactor trip followed by a turbine trip. However, it cannot be assumed that any single fault that actuates a safety injection signal will also produce a reactor trip. Therefore, two different courses of events are considered.

(1) Case A - Trip occurs at the same time spurious injection starts.

The operator should determine if the spurious signal was transient or steady state in nature. The operator must also determine if the safety injection signal should be blocked. For a spurious occurrence, the operator would stop the safety injection and maintain the plant in the hot shutdown condition. If the ECCS actuation instrumentation must be repaired, future plant operation will be in accordance with the Technical Specifications.

(2) Case B -The reactor protection system produces a trip later in the transient.

The reactor protection system does not produce an immediate trip, and the reactor experiences a negative reactivity excursion due to the injected boron causing a decrease in reactor power. The power mismatch causes a drop in primary coolant temperature and coolant shrinkage. Pressurizer pressure and level drop. Load will decrease due to the effect of reduced steam pressure on load when the turbine throttle valve is fully open. If automatic rod control is used, these effects will be lessened until the rods have moved out of the core. The transient is eventually terminated by the reactor protection system low pressure trip or by manual trip.

The time to trip is affected by initial operating conditions including core burnup history which affects initial boron concentration, rate of change of boron concentration, Doppler and moderator coefficients.

Recovery from this incident for Case B is made in the same manner described for Case A. The only difference is the lower T_{avg} and pressure associated with the power mismatch during the transient. The time at which reactor trip occurs is of no concern for this occurrence. At lower loads coolant contraction will be slower resulting in a longer time to trip.

15.2.14.2 Analysis of Effects and Consequences

Method of Analysis

The spurious operation of the safety injection system is analyzed by employing the detailed digital computer program LOFTRAN[5]. The code simulates the neutron kinetics, reactor coolant system, pressurizer, pressurizer relief and safety valves, pressurizer spray, steam generator, steam generator safety valves, and the effect of the safety injection system. The program computes pertinent plant variables including temperatures, pressures, and power level.

Inadvertent operation of the ECCS at power is classified as a Condition II event, a fault of moderate frequency. The criteria established for Condition II events include the following:

(a) Pressure in the reactor coolant and main steam systems should be maintained below 110% of the design values,

(b) Fuel cladding integrity shall be maintained by ensuring that the minimum DNBR remains above the 95/95 DNBR limit for PWRs, and

(c) An incident of moderate frequency should not generate a more serious plant condition without other faults occurring independently.

To address criterion (c), the more restrictive criterion that pressurizer power-operated relief valves (PORVs) do not open while a water-solid condition exists in the pressurizer is used. This addresses any concerns regarding subcooled water relief through the pressurizer PORVs or pressurizer safety relief valves (PSRVs) and the downstream piping which are not qualified for this condition.

The inadvertent ECCS actuation at power event is analyzed to determine both the minimum DNBR value and maximum pressurizer water volume. The most limiting case with respect to DNB is a minimum reactivity feedback condition with the plant assumed to be in manual rod control. Because of the power and temperature reduction during the transient, operating conditions do not approach the core limits.

For maximizing the potential for pressurizer filling, the most limiting case is a maximum reactivity feedback condition with an immediate reactor trip, and subsequent turbine trip, on the initiating SI signal. The transient results are presented for each case.

Assumptions

(1) Initial Operating Conditions

The DNB case is analyzed with the Revised Thermal Design Procedure as described in WCAP-11397-P-A[18]. Initial reactor power, RCS pressure, and temperature are assumed to be at the nominal full power values. Uncertainties in initial conditions are included in the limit DNBR as described in Reference [18]. For the pressurizer filling case, initial conditions with uncertainties in their worst possible direction on power, vessel average temperature, pressurizer pressure, and pressurizer level are assumed in order to maximize the rate of coolant expansion and minimize the size of the steam bubble.

(2) Moderator and Doppler Coefficients of Reactivity

The minimum DNBR case is evaluated at beginning of life (BOL) conditions, so a low BOL moderator temperature coefficient and a low absolute value Doppler power coefficient are assumed. For the pressurizer filling case, conservative maximum feedback coefficients consistent with end of life operation are assumed.

(3) Reactor Control

For the minimum DNBR case (without direct reactor trip on SI), the reactor is assumed to be in manual rod control. For the pressurizer filling case, a reactor trip is assumed to occur coincident with initiation of the transient.

(4) Pressurizer Pressure Control

Pressurizer heaters are assumed to be inoperable for the minimum DNBR case, since this yields a higher rate of pressure decrease. The opposite is assumed for the pressurizer filling case, in which the operation of the pressurizer heaters has been found to result in an increase in the pressurizer filling rate.

PORVs are operable for both the minimum DNBR and pressurizer filling cases. For the minimum DNBR case, maintaining a low pressurizer pressure is conservative. For the pressurizer filling case, availability of the PORVs may provide earlier steam relief and therefore maximizes the pressurizer insurge. However, the PORV opening setpoint is not reached in this analysis.

Pressurizer spray is assumed available to minimize pressure for the minimum DNBR case and to increase the rate of the pressurizer level increase for the pressurizer filling case.

(5) Boron Injection

At the initiation of the event, two centrifugal charging pumps inject borated water into the cold leg of each loop. In addition, flow is included to account for the potential operation of the positive displacement charging pump (PDP) for the DNBR case. However, this analysis remains valid although the PDP has been abandoned and is no longer used for normal operation. No PDP flow is assumed for the overfill case since the pump is not used for normal operation.

(6) Turbine Load

For the minimum DNBR case (without direct reactor trip/turbine trip on SI), the turbine load remains constant until the governor drives the throttle valve wide open. After the throttle valve is fully open, turbine load decreases as steam pressure drops. In the case of pressurizer filling, the reactor and turbine both trip at the time of SI actuation with the turbine load dropping to zero simultaneously.

(7) Reactor Trip

Reactor trip is initiated by low pressure at 1925 psia for the minimum DNBR case. The pressurizer filling case assumes an immediate reactor trip on the initiating SI signal.

(8) Decay Heat

The decay heat has no impact on the DNB case (i.e., minimum DNBR occurs prior to reactor trip). For the pressurizer filling case, the availability of decay heat and its expansion effects on the RCS liquid volume is considered. Core residual heat generation is based on the 1979 version of ANSI 5.1[14] assuming long-term operation at the initial power level preceding the trip is assumed.

(9) Operator Actions

Operator action to terminate safety injection flow is assumed 10 minutes from event initiation, and thereby, mitigates the event.

(10) Auxiliary Feedwater System

For the pressurizer filling case only, the AFW System is assumed to actuate on the initiating SI signal. The AFW flow provides additional RCS cooling which slows the pressurizer insurge.

Results

The transient responses for the pressurizer filling cases are shown in Figures 15.2-42a through 15.2-42c. Table 15.2-1 shows the calculated sequence of events for both minimum DNBR case and the pressurizer filling case.

Minimum DNBR Case:

Nuclear power starts decreasing immediately due to boron injection, but steam flow does not decrease until the turbine throttle valve goes wide open. The mismatch between load and nuclear power causes T_{avg}, pressurizer water level, and pressurizer pressure to drop. The reactor trips on low pressurizer pressure. After trip, pressures and temperatures slowly rise since the turbine is tripped and the reactor is producing some power due to delayed neutron fissions and decay heat. The DNBR remains above its initial value throughout the transient.

Pressurizer Filling Case:

Reactor trip occurs at event initiation followed by a rapid initial cooldown of the RCS. Coolant contraction results in a short-term reduction in pressurizer pressure and water level. The combination of the RCS heatup, due to residual RCS heat generation, and ECCS injected flow causes the pressure and level transients to rapidly turn around. Pressurizer water level then increases throughout the transient. Spray flow helps to condense the pressurizer steam bubble, causing a pressurizer insurge and minimizing pressurizer pressure.The ECCS injection flow is terminated via operator action in accordance with plant emergency procedures and the increase in pressurizer level stops. SI flow termination at 10 minutes prevents the pressurizer from filling. As such, the integrity of the PORVs and PSRVs are not compromised.

Following the analyzed portion of the transient, the plant will approach a stabilized condition at hot standby; normal plant operating procedures may then be followed. The operating procedures call for operator action to control RCS boron concentration and pressurizer level using the CVCS, and to maintain generator level through control of the main or auxiliary feedwater system. Any action required of the operator to maintain the plant in a stabilized condition is in a time frame in excess of ten minutes following reactor trip.

15.2.14.3 Conclusions

Results of the analysis show that spurious ECCS operation without immediate reactor trip does not present any hazard to the integrity of the RCS with respect to DNBR. The minimum DNBR is never less than the initial value. Thus, there will be no cladding damage and no release of fission products to the RCS. If the reactor does not trip immediately, the low pressurizer pressure reactor trip will provide protection. This trips the turbine and prevents excess cooldown, which expedites recovery from the incident.

With respect to pressurizer filling, SI flow is terminated in sufficient time to prevent the pressurizer from going water solid. As such, the PORVs and PSRVs do not pass water and their integrity is not compromised. Termination of ECCS injection via operator action in accordance with plant emergency procedures, stops the further increase in pressure, thus preventing the safety valves from opening. This precludes possible damage to the valves which could potentially generate a more serious plant condition.

15.2.15 Chemical and Volume Control System Malfunction During Power Operation

15.2.15.1 Identification of Causes and Accident Description

Increases in reactor coolant inventory caused by a malfunction of the chemical and volume control system may be postulated to result from operator error or a control signal malfunction. Transients examined in this section are characterized by increasing pressurizer level, increasing pressurizer pressure, and a constant boron concentration. The transients analyzed in this section are done to demonstrate that there is adequate time for the operator to take corrective action to ensure that the integrity of the pressurizer Power Operated Relief Valves (PORVs) and the Pressurizer Safety Relief Valves (PSRVs) is maintained (i.e., the valves do not actuate with the pressurizer in a water-solid condition). An increase in reactor coolant inventory, which results from the addition of cold, unborated water to the RCS, is analyzed in Section 15.2.4, Uncontrolled Boron Dilution.

The most limiting CVCS Malfunction case would result if charging was in automatic control and the pressurizer level channel being used for charging control failed in a low direction. This would cause maximum charging flow to be delivered to the RCS and letdown flow to be isolated. The worst single failure for this event would be a second pressurizer level channel failing in an as-is condition or a low condition. This will defeat the reactor trip on two-out-of-three high pressurizer level channels. To ensure that the integrity of the PORVs and the PSRVs is maintained, the operator must be relied upon to terminate charging.

During a CVCS Malfunction event, several main control board alarms could be generated to alert the operator, including the following:

- High charging flow alarm
- High pressurizer water level alarm
- Pressurizer water level deviation alarm
- Low VCT level alarm

15.2.15.2 Analysis of Effects and Consequences

Method of Analysis

The CVCS malfunction is analyzed using the LOFTRAN computer code (WCAP-7907-P-A). The code simulates the neutron kinetics, RCS, pressurizer, pressurizer relief and safety valves, pressurizer spray, feedwater system, steam generator, and steam generator safety valves. The code computes pertinent plant variables including temperatures, pressures, and power level.

A Chemical and Volume Control System Malfunction at power event is classified as a Condition II event, a fault of moderate frequency. The criteria established for Condition II events include the following:

- Pressure in the reactor coolant and main steam systems should be maintained below 110% of the design values,
- Fuel cladding integrity shall be maintained by ensuring that the minimum DNBR remains above the 95/95 DNBR limit for PWRs, and
- An incident of moderate frequency should not generate a more serious plant condition without other faults occurring independently.

Of these, the limiting criterion is that the event will not propagate to a more serious event. To address this criterion, Westinghouse currently uses the more restrictive criterion that the pressurizer Power Operated Relief Valves (PORVs) do not open while a water-solid condition exists in the pressurizer. This addresses any concerns regarding subcooled water relief through the pressurizer PORVs or Pressurizer Safety Relief Valves (PSRVs) and the downstream piping which may not be qualified for this condition.

The analysis assumptions are the same as those discussed in Section 15.2.14.2 for the pressurizer filling case with a few exceptions:

- No reactor trip is assumed.
- The flow source is assumed to be at the RCS boron concentration.
- The same pumps are providing the flow as in the Section 15.2.14 event but the flow path has a higher resistance than the Safety Injection flow path. Thus, the CVCS Malfunction flow rates are lower than the Inadvertent ECCS flow rates
- Alarm actuation alerts the operator 60 seconds after event initiation
- Operator terminates charging flow 10 minutes after an alarm notifies the operator

Cases are examined with flow from both one and two centrifugal charging pumps to determine the time available for the operators to take the necessary corrective actions to maintain the integrity of the PORVs and PSRVs. The scenario analyzed with two charging pumps operating is slightly different than the one charging pump scenario. In the two pump scenario, it takes two failures to have two charging pumps operating at maximum capacity. Letdown isolation would require a third failure so letdown is not isolated in the two pump case. Minimum letdown flow is 75 gpm so the net inventory addition is decreased by 75 gpm for the two pump case.

Results

The transient responses for the limiting CVCS system malfunction cases are shown in Figures 15.2-44 through Figure 15.2-47. Table 15.2-6 shows the calculated sequence of events. In all the cases analyzed, core power and RCS temperatures remain relatively constant.

The pressurizer level increases as a result of the injected flow. In the case with one charging pump operating, the pressurizer reaches a peak water- volume of 1664 ft3 and the case with 2 charging pumps, the peak pressurizer water volume is 1635 ft3. Since the pressurizer does not fill in either case, there can be no water relief through either the PORVs or the PSRVs.

Figures 15.2-44 through Figure 15.2-47 provide transient information for both the one-pump and two-pump cases and Table 15.2-6 shows a sequence of events.

15.2.15.3 Conclusions

With respect to not creating a more serious plant condition, water relief out of the PORVs and PSRVs will not occur during a CVCS Malfunction event because operator action to terminate the charging flow occurs early enough to prevent a water-solid pressurizer. The sequence of events presented in Table 15.2-6 shows the operators have sufficient time to take corrective action.

References

(1) Gangloff, W. C., "An Evaluation of Anticipated Operational Transients in Westinghouse Pressurized Water Reactors," WCAP-7486-L (Proprietary), March 1971 and WCAP-7486 (Non-Proprietary), May 1971 .

(2) "Westinghouse Anticipated Transients Without Trip Analysis," WCAP-8330, August 1974.

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

(4) Hargrove, H. G., "FACTRAN, A FORTRAN IV Code for Thermal Transients in a $U0_2$ Fuel Rod," WCAP-7908-A, December 1989.

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

(6) Deleted in Amendment 80.

(7) Deleted in Amendment 80.

(8) Deleted by UFSAR Amendment 2

(9) Smith, M., et al, “Overpressure Protection Report for Watts Bar Nuclear Power Plant Unit 2,” March 2010.

(10) Deleted by Amendment 80.

(11) Letter from T.M. Anderson (Westinghouse) to S.H. Hanauer (NRC), "ATWS Submittal," Westinghouse Letter NS-TMA-2182, dated December 30, 1979.

(12) ATWS Final Rule - Code of Federal Regulations 10 CFR 50.62 and Supplementary Information Package, "Reduction of Risk from Anticipated Transients Without Scram (ATWS) Events for Light-Water-Cooled Nuclear Power Plants."

(13) Haessler, R. L. et al., “Methodology for the Analysis of the Dropped Rod Event,” WCAP-11394-P-A (Proprietary) and WCAP-11395-A (Non-Proprietary), January 1990.

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

(15) Deleted by Amendment 97.

(16) Deleted by Amendment 97.

(17) Deleted by Amendment 97.

(18) Friedland, A. J. and S. Ray, "Revised Thermal Design Procedure," WCAP-11397-P-A (Proprietary), WCAP-11398-A (Nonproprietary), April 1989.

Table 15.2-1 Time Sequence Of Events For Condition II Events (Page 1 of 5)

Accident	Event	Time (sec.)
Uncontrolled RCCA Withdrawal from a Subcritical Condition	Initiation of uncontrolled rod withdrawal 75 pcm/sec reactivity insertion rate from 10^{-9} of normal power	0
	Power range high neutron flux low setpoint reached	10.43
	Peak nuclear power occurs	10.57
	Rods begin to fall into core	10.93
	Peak heat flux occurs	12.40
	Minimum DNBR occurs	12.40
	Peak clad temperature occurs	12.931
	Peak average fuel temperature occurs	13.141
Uncontrolled RCCA Withdrawal at Power		
1. Case A	Initiation of uncontrolled RCCA withdrawal at maximum reactivity insertion rate (110 pcm/sec)	0
	Power range high neutron flux high trip point reached	1.1
	Rods begin to fall into core	1.6
	Minimum DNBR occurs	2.7
2. Case B	Initiation of uncontrolled RCCA withdrawal at a small reactivity insertion rate (1 pcm/sec)	0
	Overtemperature ΔT reactor trip signal initiated	61.1
	Rods begin to fall into core	62.6
	Minimum DNBR occurs occurs	63.6

Table 15.2-1 Time Sequence Of Events For Condition II Events (Page 2 of 5)

Accident	Event	Time (sec.)
Uncontrolled Boron Dilution		
1. Dilution During Cold Shutdown - RCS filled	Dilution Begins	0
	High VCT Level Alarm Sounds	820
	Shutdown Margin is Lost	≈2186
2. Dilution During Hot Shutdown	Dilution Begins	0
	High VCT Level Alarm Sounds	820
	Shutdown Margin is Lost	≈3552
3. Dilution During Hot Standby	Dilution Begins	0
	High VCT Level Alarm Souds	820
	Shutdown Margin is Lost	≈3563
4. Dilution During Startup	Dilution begins	(Unspecified)*
	Reactor trip on source range high flux	0
	Shutdown margin lost	≈1584
5. Dilution During Full Power Operation		
a. Automatic Reactor Control	Dilution begins	0
	Shutdown margin lost	≈2057
b. Manual Reactor Control	Dilution begins	0
	Reactor trip setpoint reached for overtemperature ΔT	77.5
	Rods begin to fall into core	79
	Shutdown margin lost (if dilution continues after trip)	≈2057
* The results of the analysis are not impacted by the time of dilution initiation		

Table 15.2-1 Time Sequence Of Events For Condition II Events (Page 3 of 5)

Accident	Event	Time (sec.)
Partial Loss of Forced Reactor Coolant Flow (four loops operating, one pump coasting down)	One pump begins coasting down	0
	Low flow trip setpoint reached	1.32
	Rods begin to drop	2.52
	Minimum DNBR occurs	3.7
Loss of External Electrical Load		
1. With pressurizer control (BOL)	Loss of electrical load	0
	High pressurizer pressure reactor trip point reached	9.6
	Rods begin to drop	11.1
	Minimum DNBR occurs	12.6
2. Without pressurizer control	Loss of electrical load	0
	High pressurizer pressure reactor trip point reached	4.3
	Rods begin to drop	6.3
	Peak Pressurizer pressure occurs	7.2
Loss of Normal Feedwater with Loss of Offsite Power (LOOP)	Main Feedwater Flow Stops	10.0
	Low-low steam generator water level reactor trip	62.1
	Rods begin to drop	64.1
	Reactor coolant pumps begin to coastdown	66.1
	Auxiliary Feedwater from Two Motor-Driven Auxiliary Feedwater Pumps Initiated	122.1

Table 15.2-1 Time Sequence Of Events For Condition II Events (Page 4 of 5)

Accident	Event	Time (sec.)
	Four steam generators begin to receive auxiliary feed from two motor-driven auxiliary feedwater pumps	175.0
	Longterm peak water level in pressurizer occurs	≈330
Single-Loop Feedwater Malfunction at Hot Full Power	One Main Feedwater Control Valve Fails Fully Open	0.0
	Minimum DNBR Occurs	26.5
	S/G High-High Water Level ESF Setpoint Reached	49.7
	Feedwater Isolation Occurs	57.7
	Overtemperature ΔT Reactor Trip Setpoint Reached	61.0
	Reactor Trip Occurs	62.5
Multi-Loop Feedwater Malfunction at Hot Full Power	All Four Main Feedwater Control Valves Fail Fully Open	0.0
	Overtemperature ΔT Reactor Trip Setpoint Reached	23.5
	Reactor Trip Occurs	25.0
	Minimum DNBR Occurs	25.5
	S/G High-High Water Level ESF Setpoint Reached	45.6
	Feedwater Isolation Occurs	53.6
Accidental Depressurization of the Reactor Coolant System	Inadvertent opening of one pressurizer safety valve	0.0
	OTΔT reactor trip setpoint reached	32.3
	Rods begin to drop	33.8
	Minimum DNBR occurs	34.4

Table 15.2-1 Time Sequence Of Events For Condition II Events (Page 5 of 5)

Accident	Event	Time (sec.)
Inadvertent Operation of ECCS During Power Operation		
DNBR Case:	Charging pumps begin injecting borated water; neutron flux starts decreasing	0.0
	Steam flow starts decreasing	44
	Low pressurizer pressure reactor trip setpoint reached	56
	Rods begin to drop	58
	Minimum DNBR occurs	(1)
Pressurizer Filling Case:	Charging pumps begin injecting borated water; reactor trip on 'S' signal; rod motion begins	0.0
	Operator terminates injection flow	600
	Maximum pressurizer water level occurs	667
(1)DNBR does not decrease below its initial value.		

Table 15.2-2 Deleted by Amendment 63.

Table 15.2-3 Deleted by Amendment 80

Table 15.2-4 Deleted by Amendment 80

Table 15.2-5 Deleted by Amendment 80

Table 15.2-6 Time Sequence Of Events For CVCS Malfunction

Accident	Event	Time (sec)
CVCS malfunction, One pump operating	Maximum charging flow initiated / letdown isolated	0.0
	An annunciator on the control board alerts the operator that an event is occuring	60.0
	Operator terminates charging flow	660.0
	Peak pressurizer water volume is reached	1479.1
CVCS malfunction, Two pumps operating	Maximum charging flow initiated from two charging pumps	0.0
	An annunciator on the control board alerts the operator thatn an event is occuring	60.0
	Operator terminates charging flow	660.0
	Peak pressurizer water volume is reached	688.2

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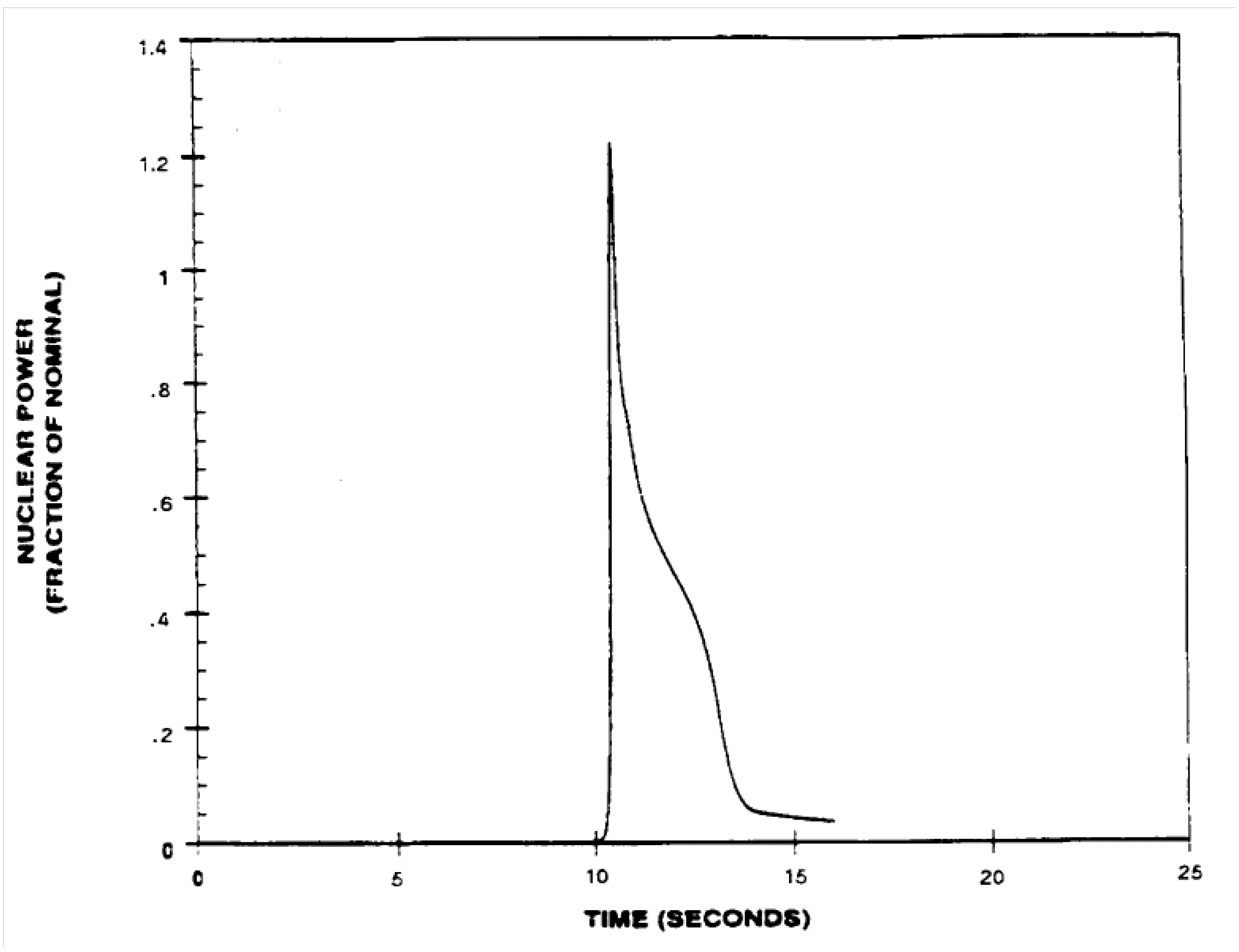

Figure 15.2-1 Uncontrolled RCCA Bank Withdrawal From Subcritical

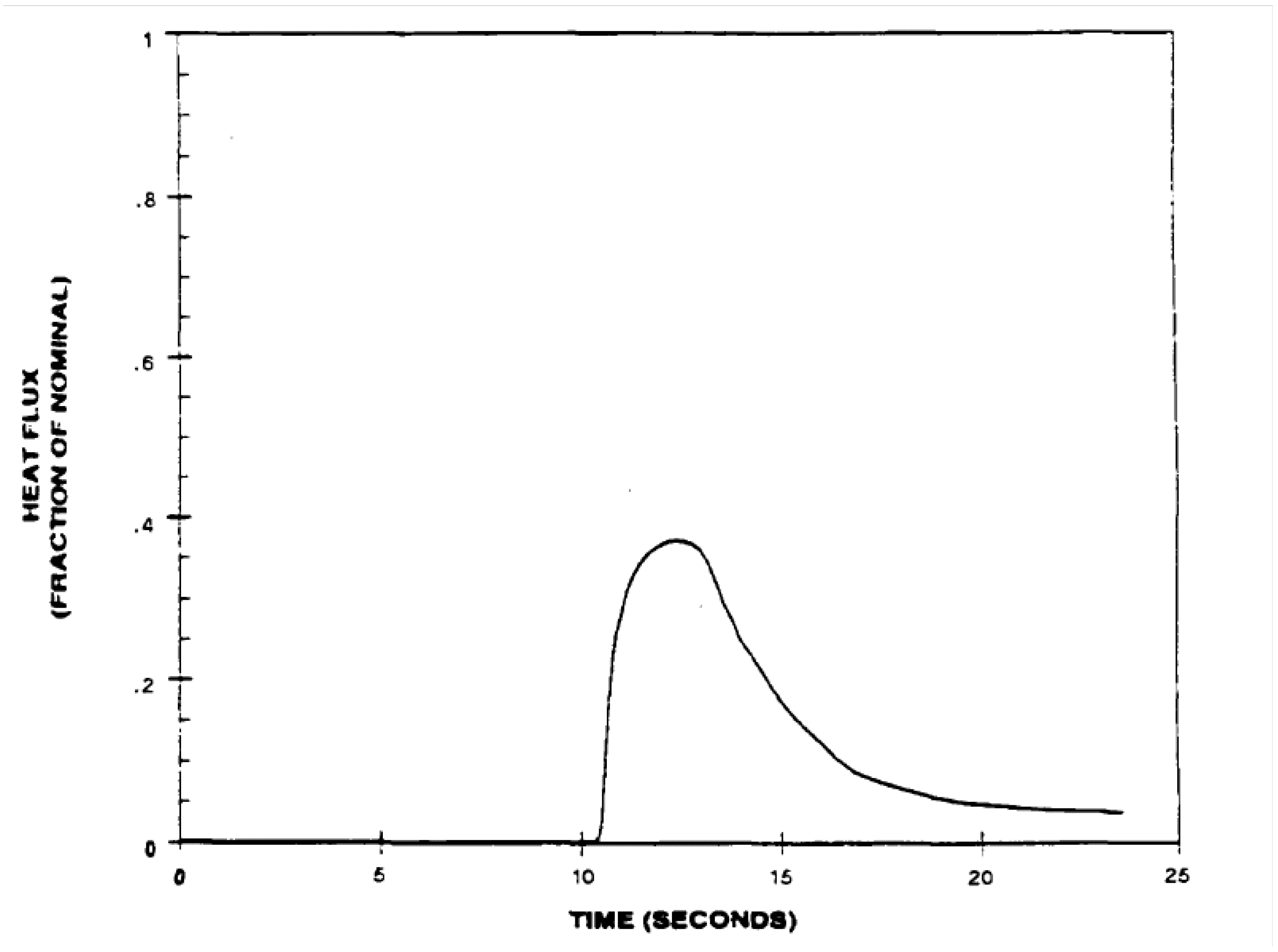

Figure 15.2-2 Uncontrolled RCCA Bank Withdrawal From Subcritical

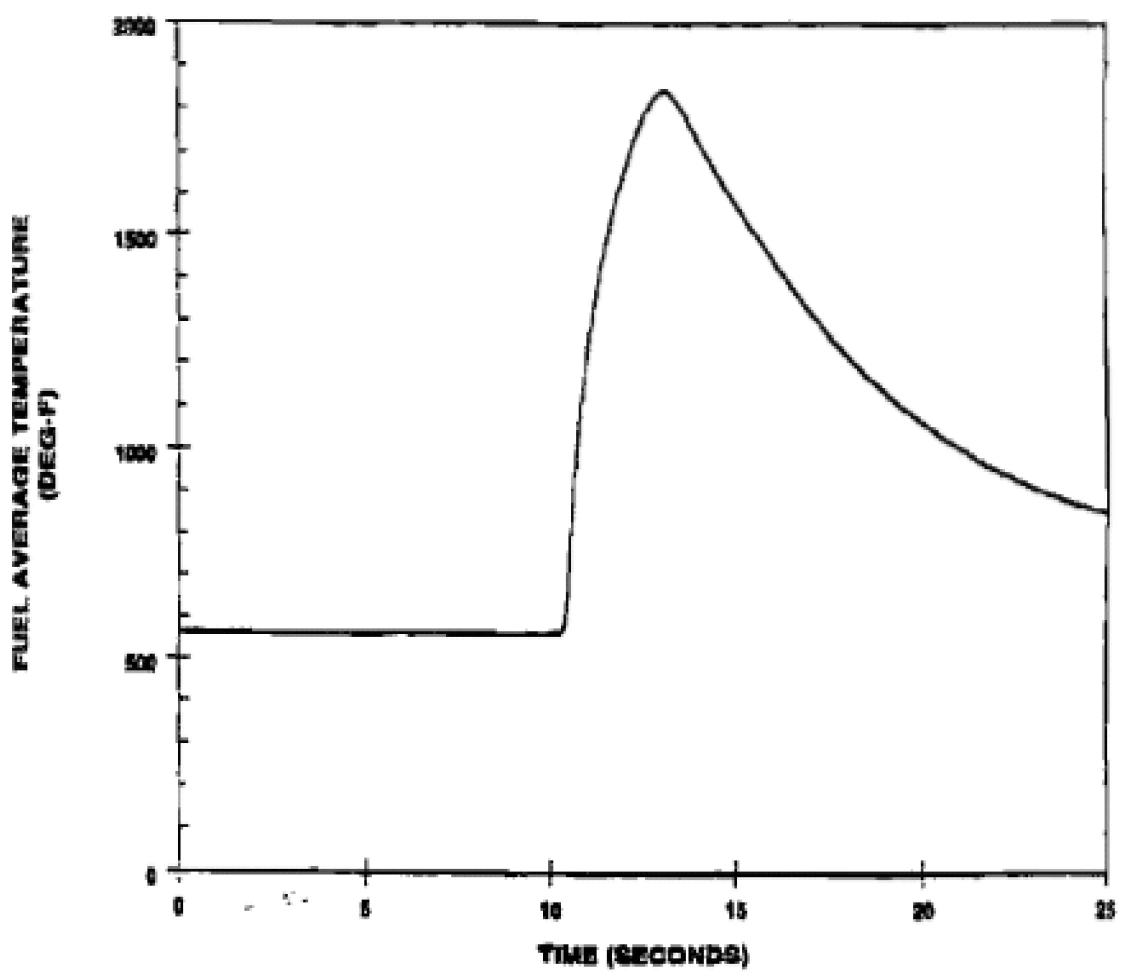

Figure 15.2-3 Uncontrolled RCCS Bank Withdrawal from a Subcritical

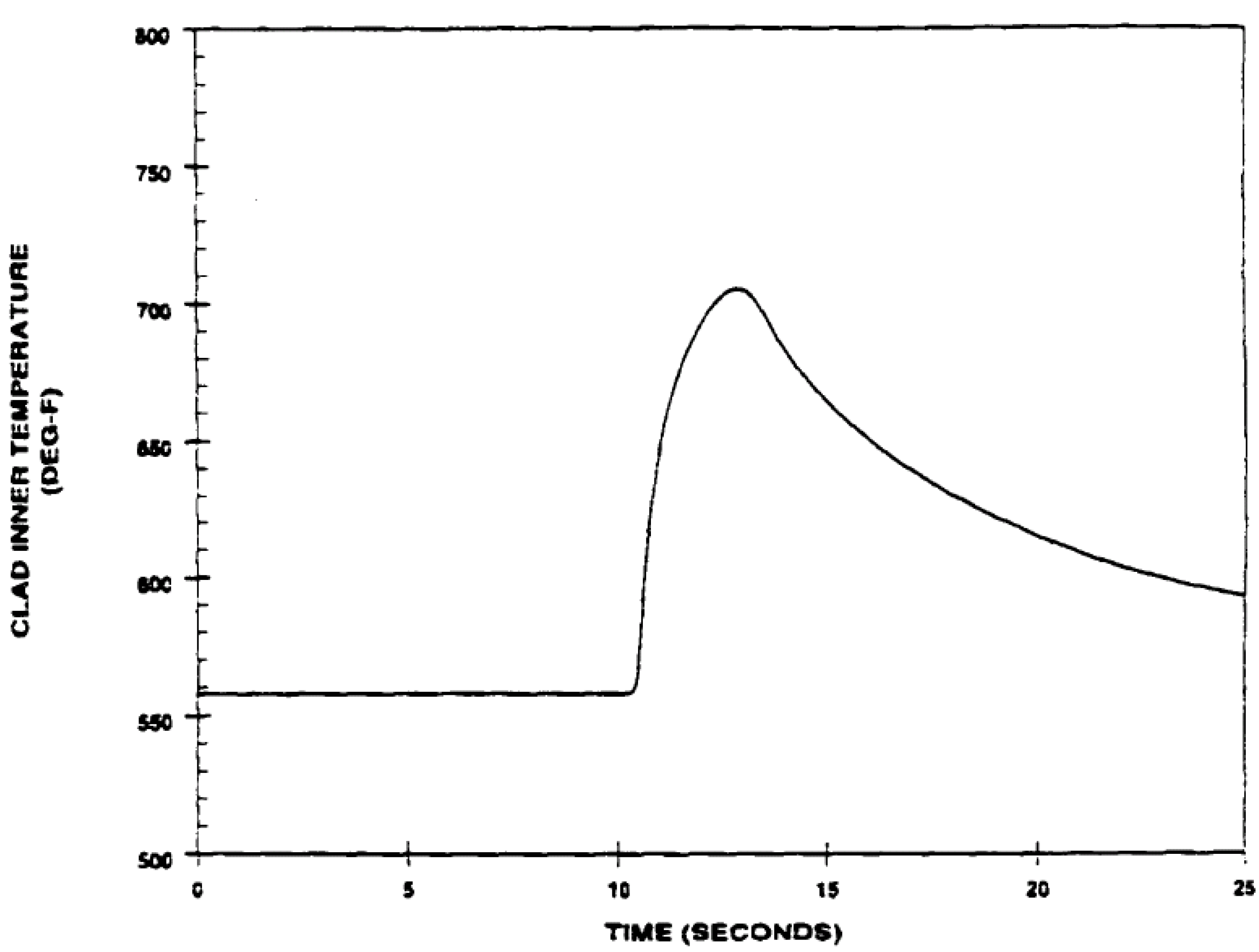

Figure 15.2-3a Uncontrolled RCCS Bank Withdrawal from a Subcritical

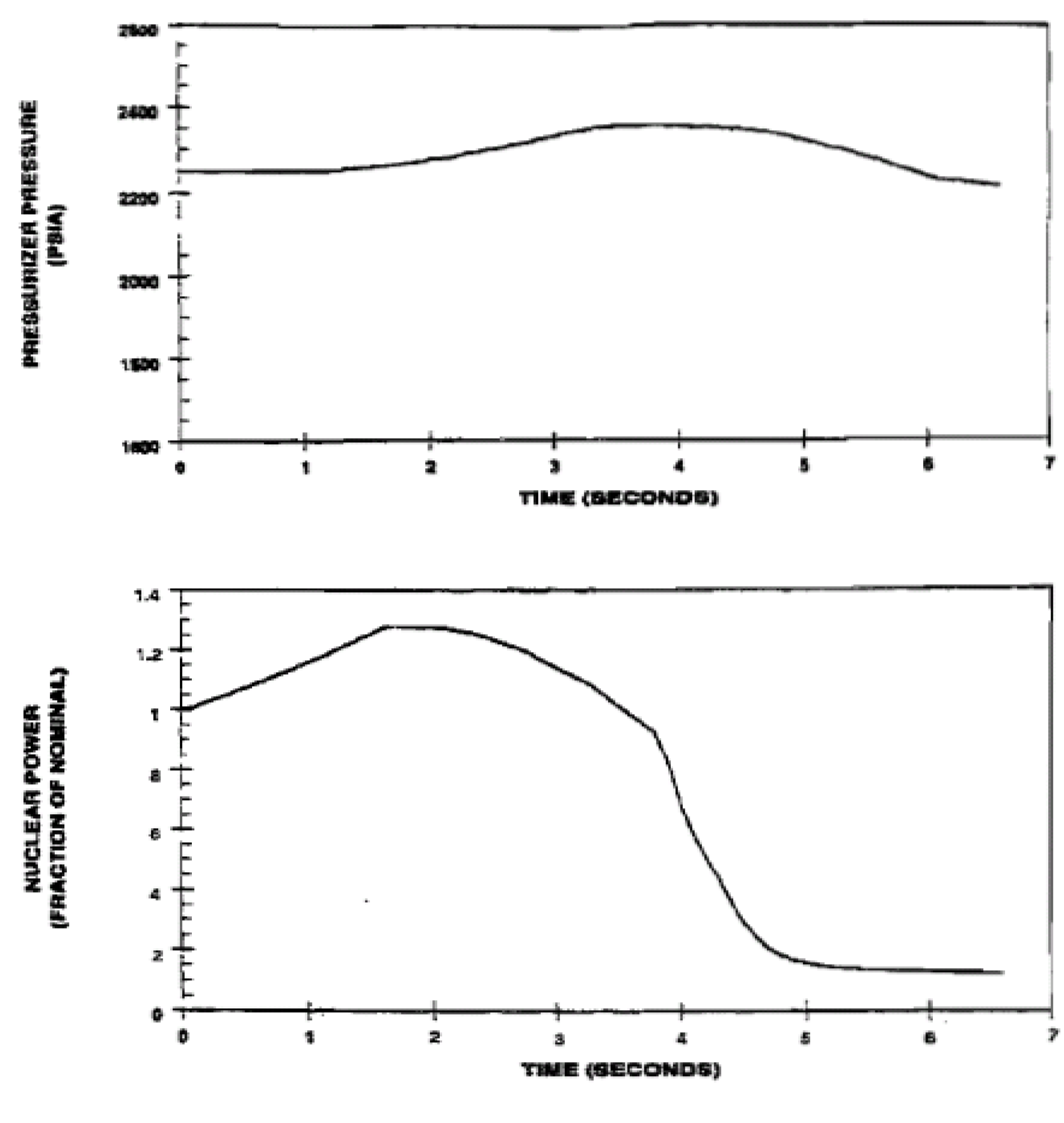

Figure 15.2-4 Uncontrolled Rod Withdrawal From Full Power, Minimum Feedback 110 PCM/Sec Withdrawal Rate

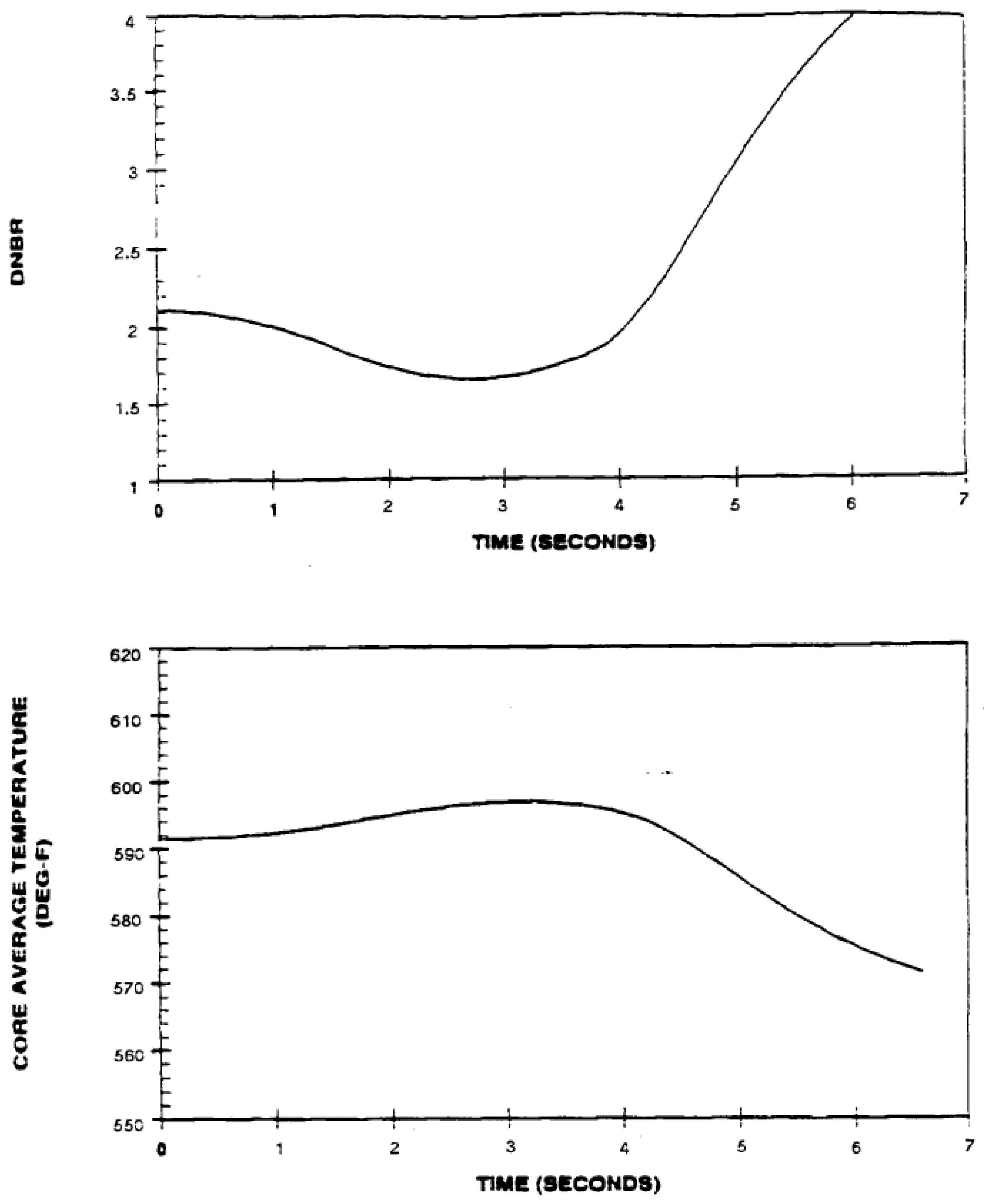

Figure 15.2-5 Uncontrolled Rod Withdrawal From Full Power, Minimum Feedback 110 PCM/Sec Withdrawal Rate

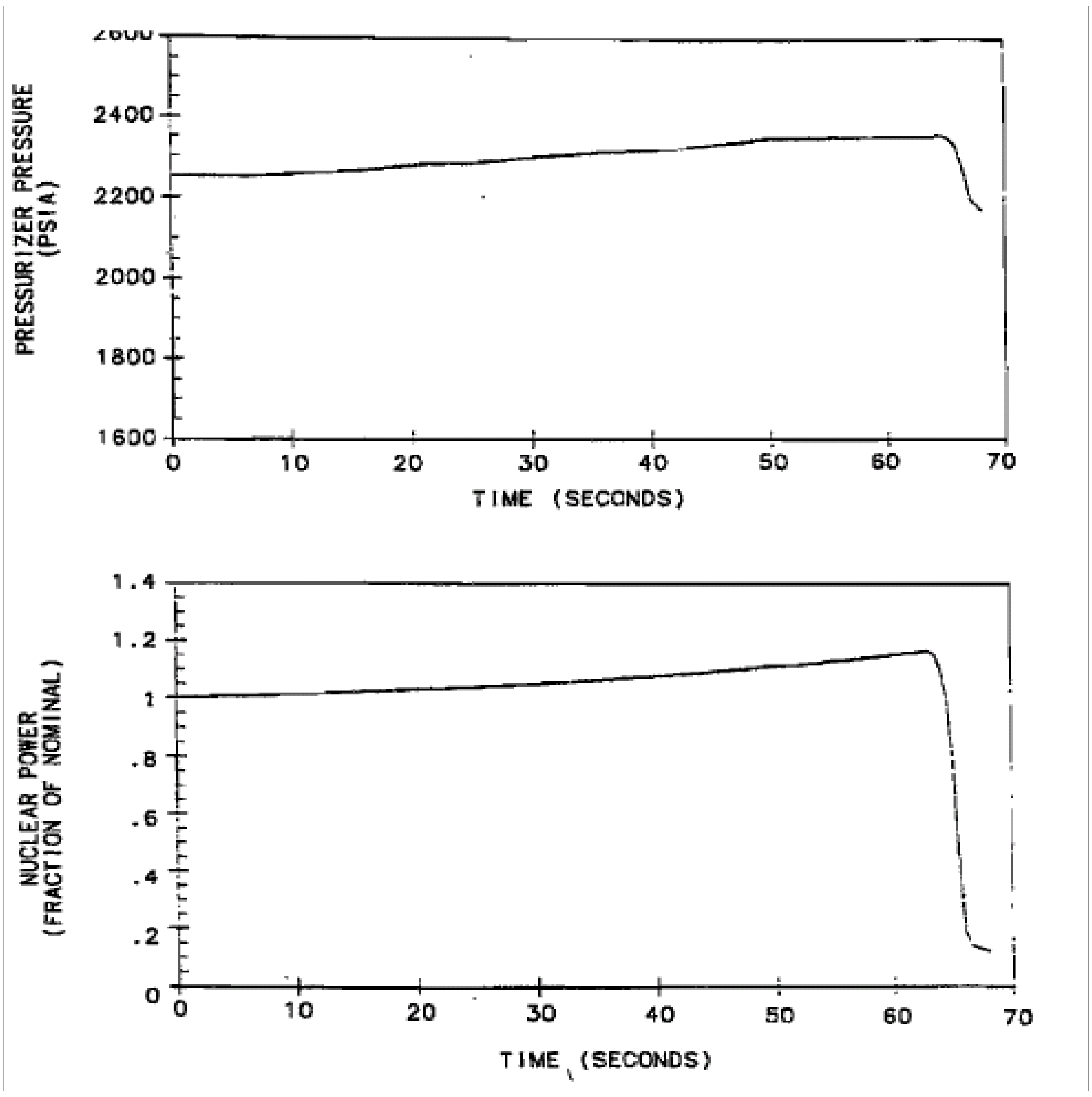

Figure 15.2-6 Uncontrolled Rod Withdrawal From Full Power, Minimum Feedback 1 PCM/Sec Withdrawal Rate

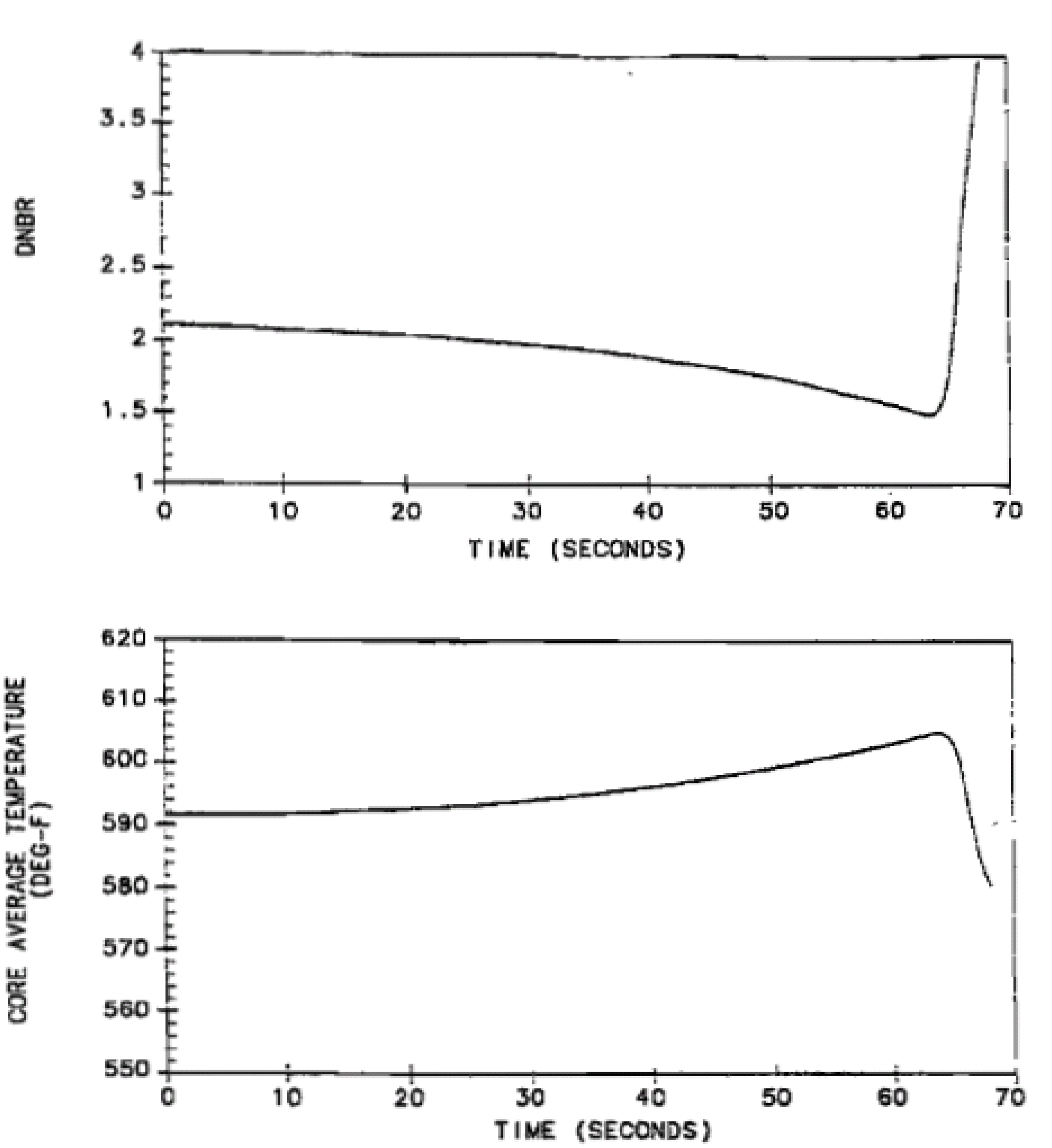

Figure 15.2-7 Uncontrolled Rod Withdrawal From Full Power, Minimum Feedback 1 PCM/Sec Withdrawal Rate

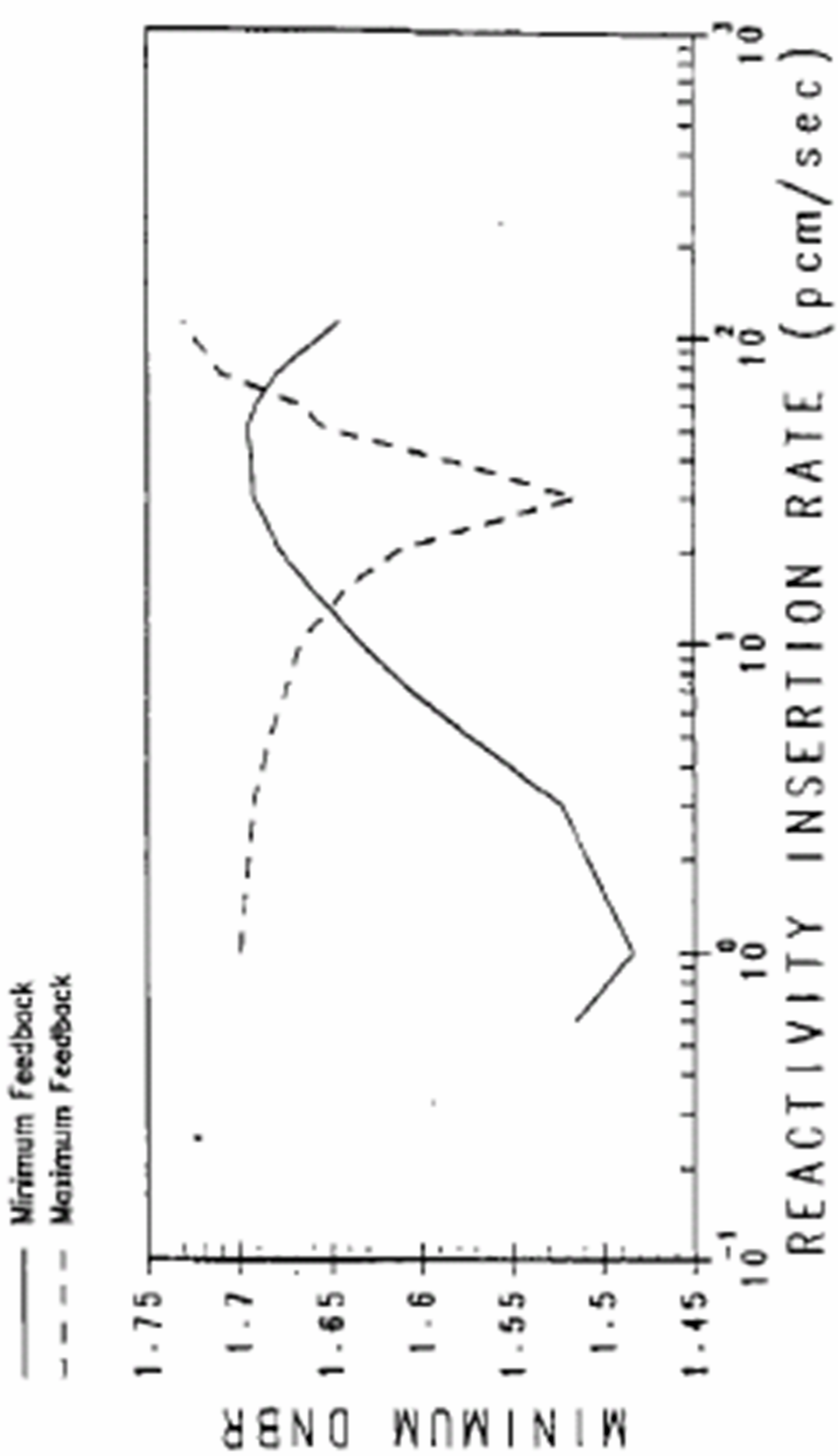

Figure 15.2-8 Uncontrolled Rod Withdrawal From 100% Power, Effect Of Reactivity Insertion Rate On Minimum DNBR

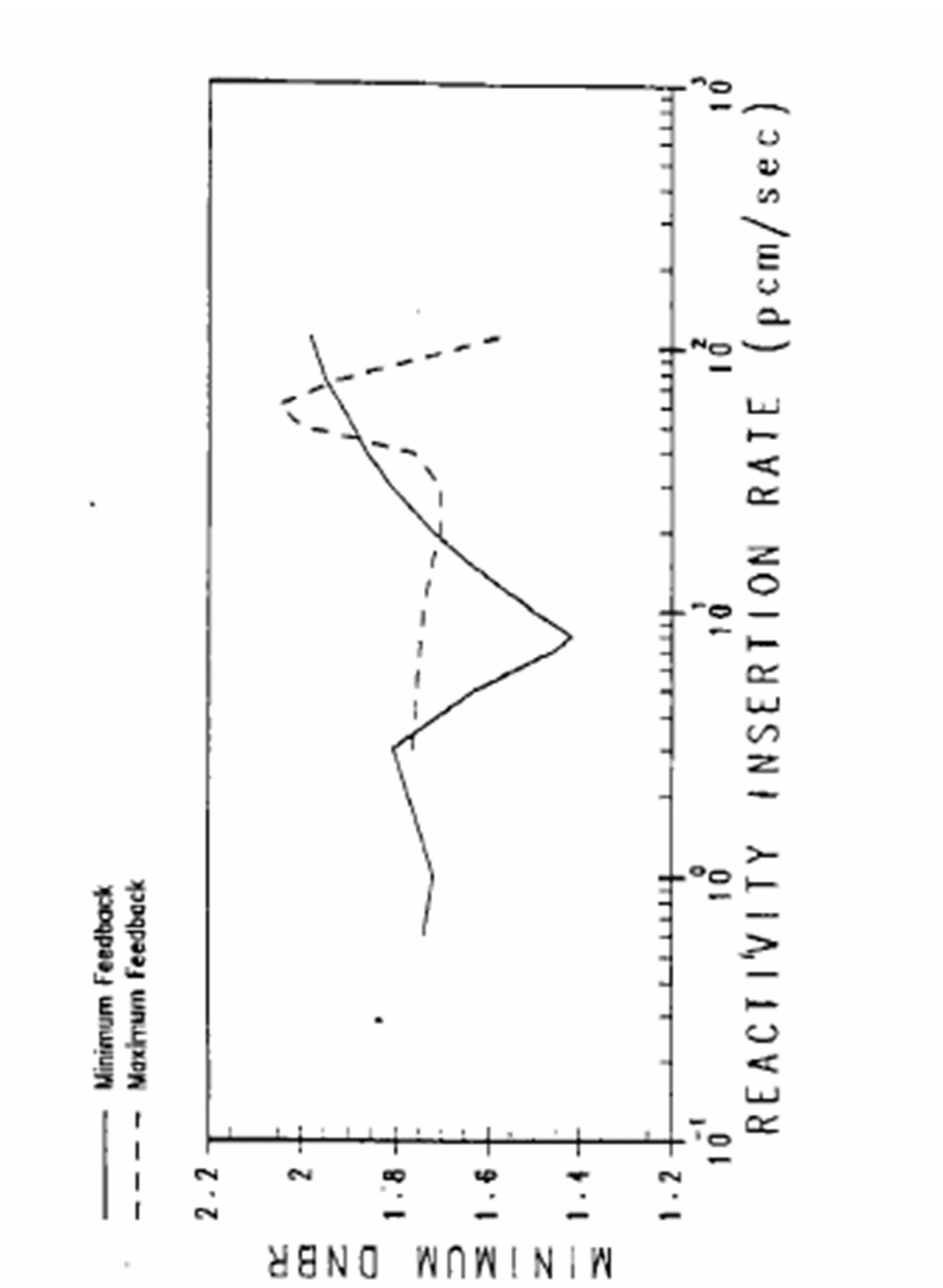

Figure 15.2-9 Uncontrolled Rod Withdrawal From 60% Power, Effect Of Reactivity Insertion Rate On Minimum DNBR

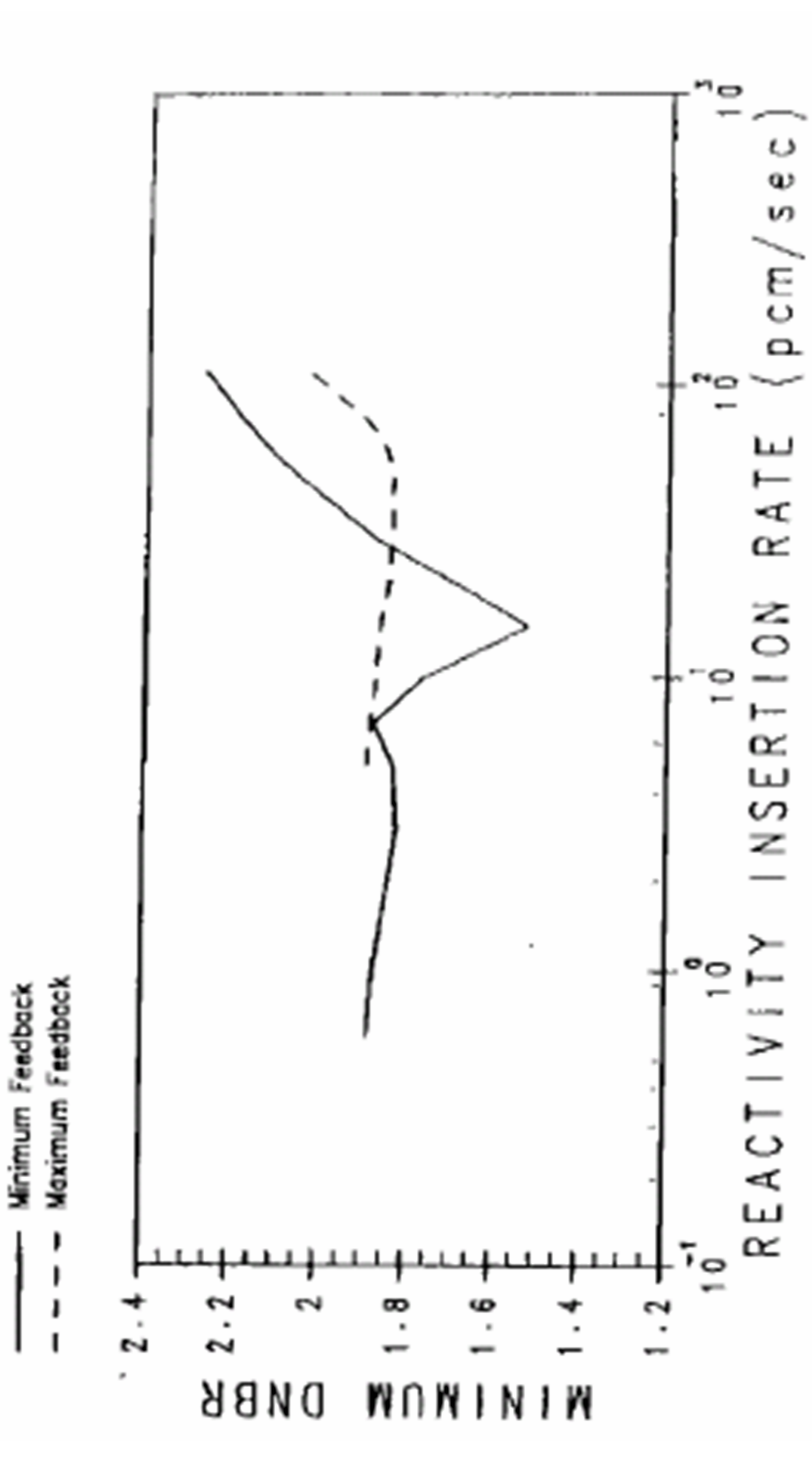

Figure 15.2-10 Uncontrolled Rod Withdrawal From 10% Power, Effect Of Reactivity Insertion Rate On Minimum DNBR

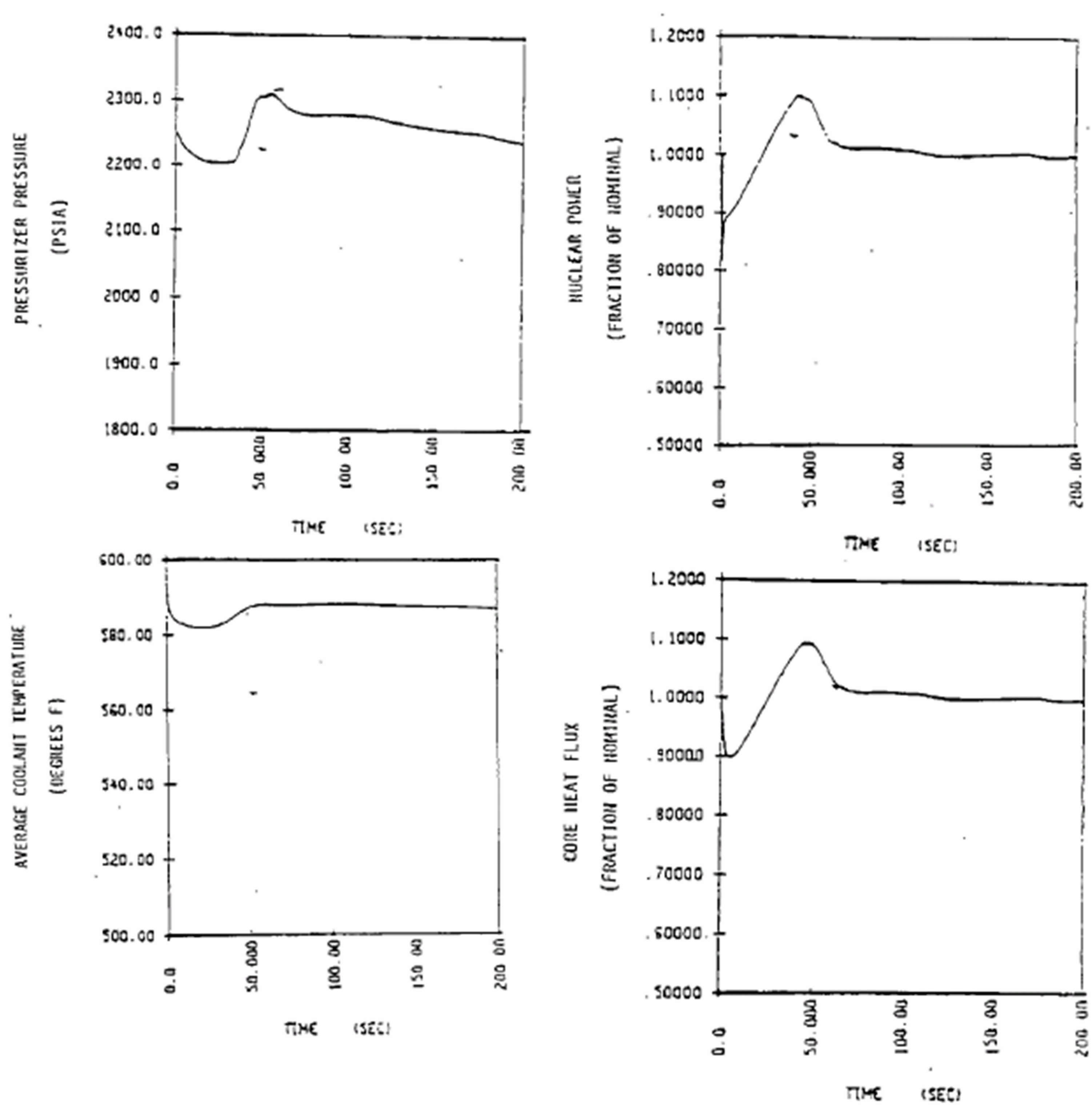

Figure 15.2-11 Pressurizer Pressure Transient, Nuclear Power, Core Average Temperature, and Core Heat Flux Transient for Dropped RCCA Assembly

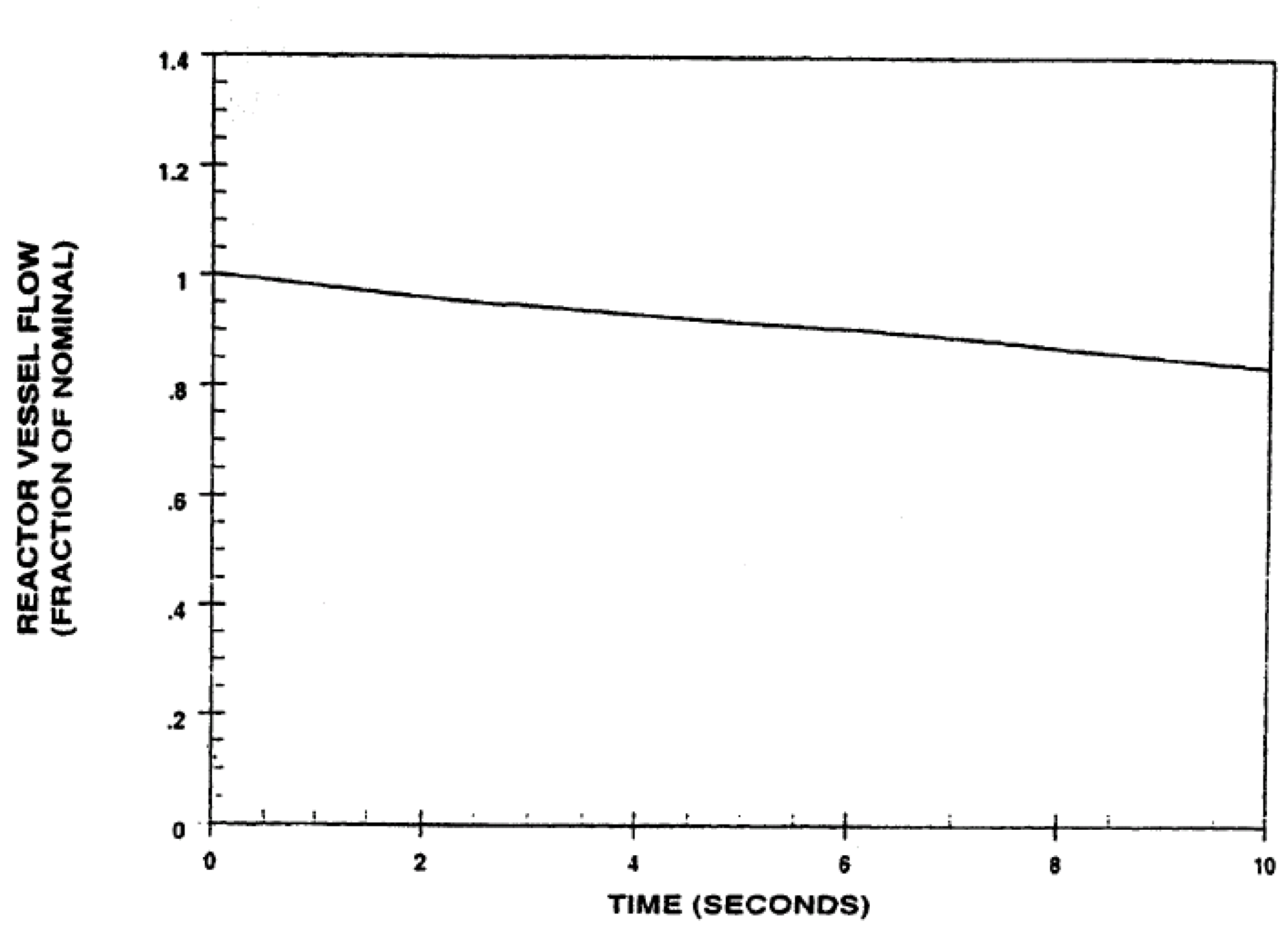

Figure 15.2-12 Reactor Vessel Flow Transient Four Pumps in Operation, One Pump Coasting Down

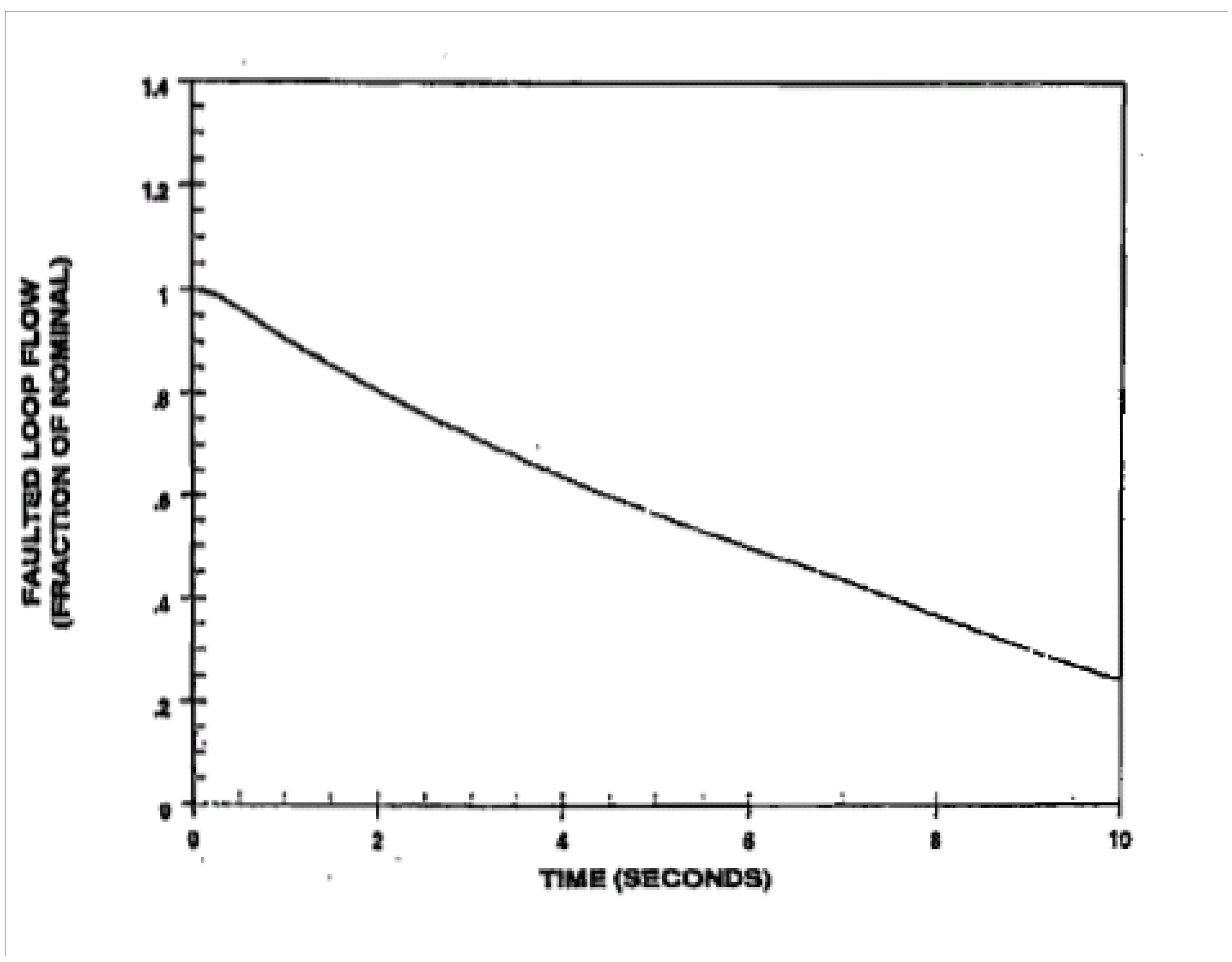

Figure 15.2-13 Loop Flow Transient Four Pumps In Operation One Pump Coasting Down

Figure 15.2-14 Deleted by Amendment 89

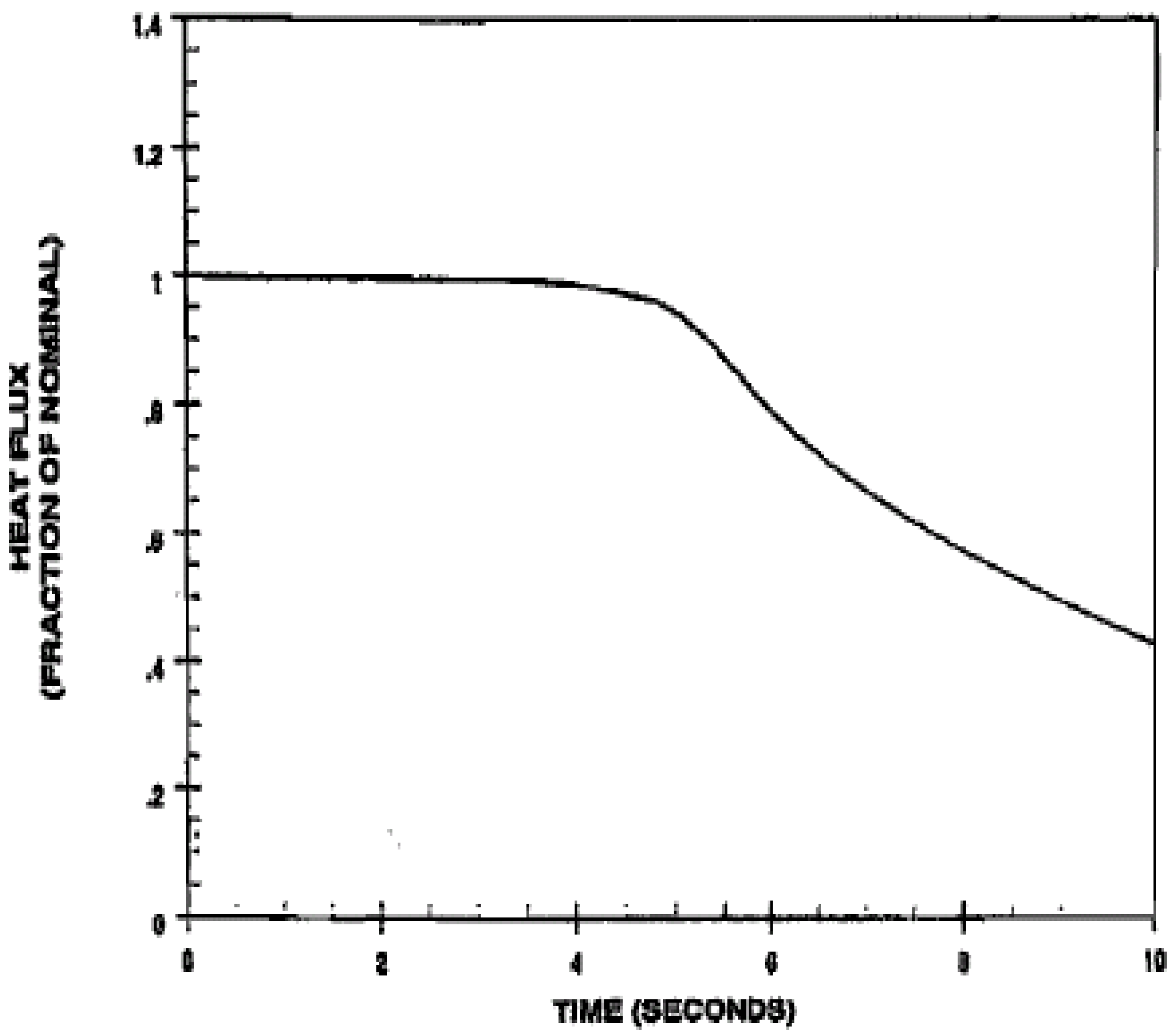

Figure 15.2-15 Hot Channel Heat Flux Transient Four Pumps in Operation, One Pump Coasting Down

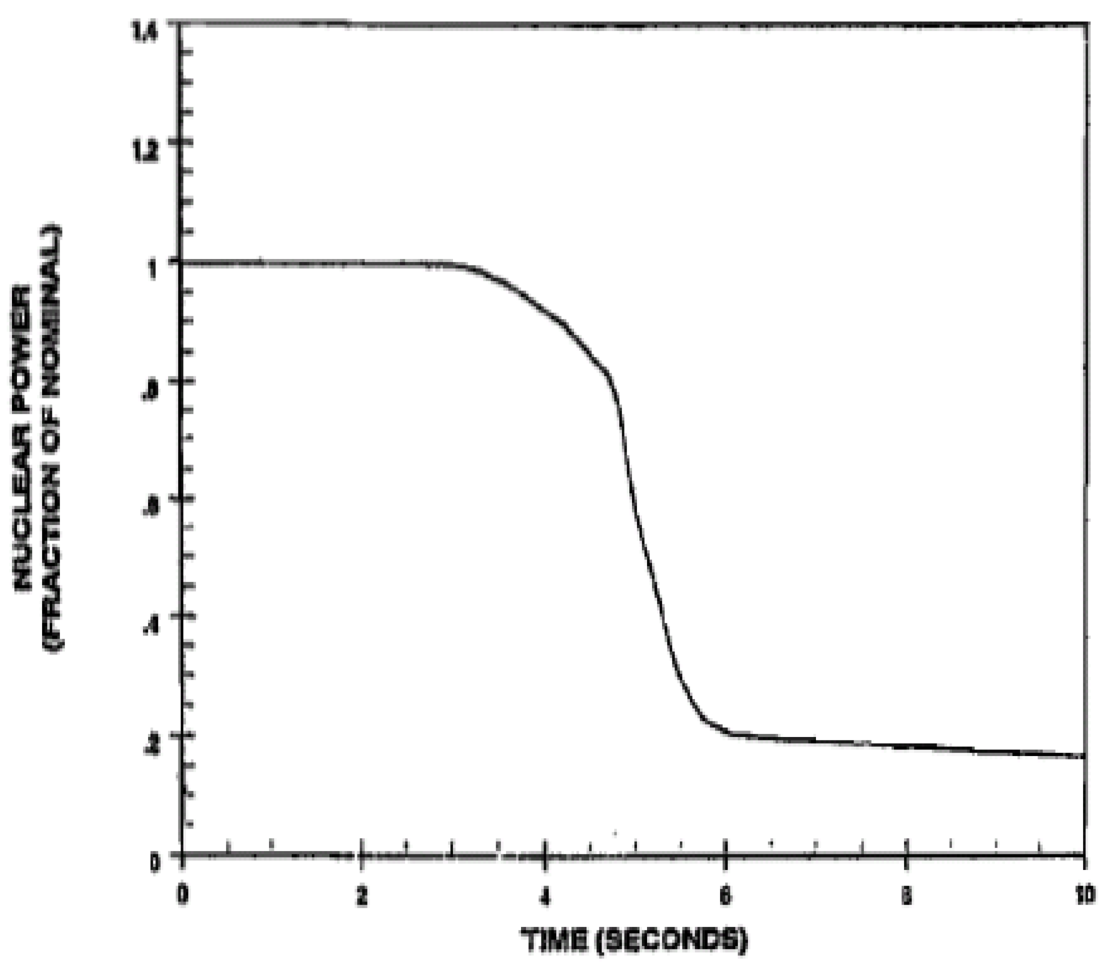

Figure 15.2-16 Nuclear Power Transient Four Pumps In Operation One Pump Coasting Down

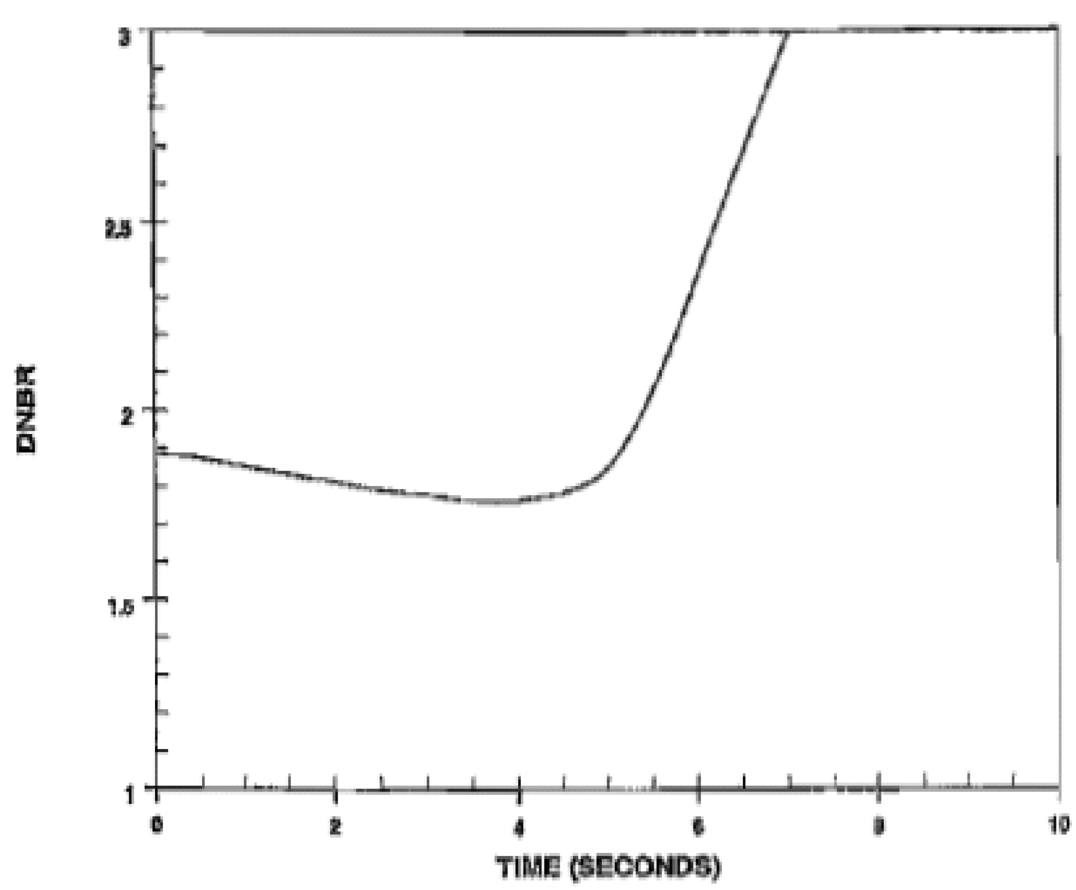

Figure 15.2-17 DNBR Versus Time Four Pumps In Operation One Pump Coasting Down

Figure 15.2-18a Deleted by Amendment 97

Figure 15.2-18b Deleted by Amendment 97

Figure 15.2-18c Deleted by Amendment 97

Figure 15.2-18d Deleted by Amendment 97

Figure 15.2-18e Deleted by Amendment 97

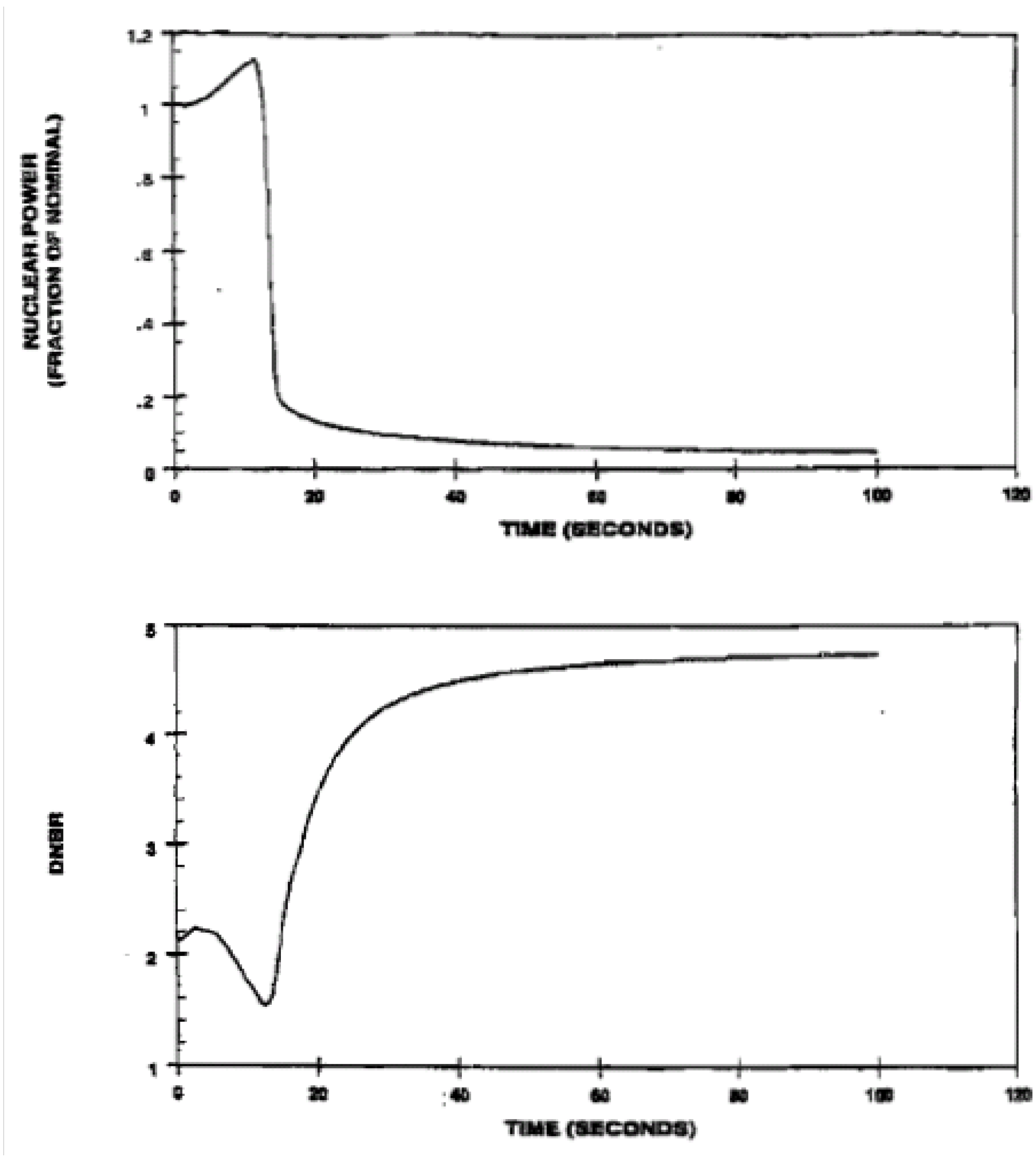

Figure 15.2-19 Loss Of Load Accident With Pressurizer Spray and Power-Operated Relief Valves

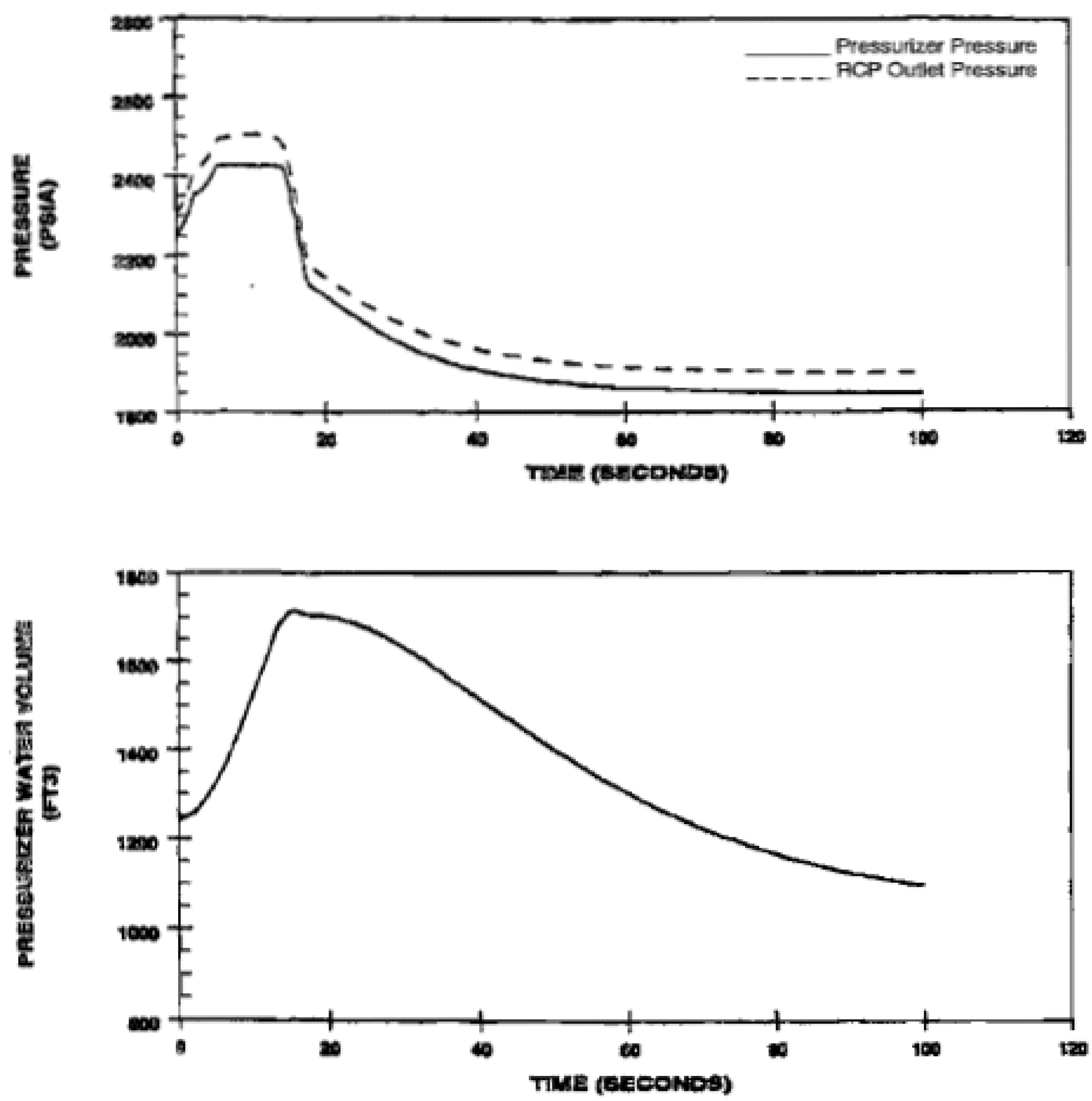

Figure 15.2-20 Loss Of Load Accident With Pressurizer Spray and Power-Operated Relief Valves

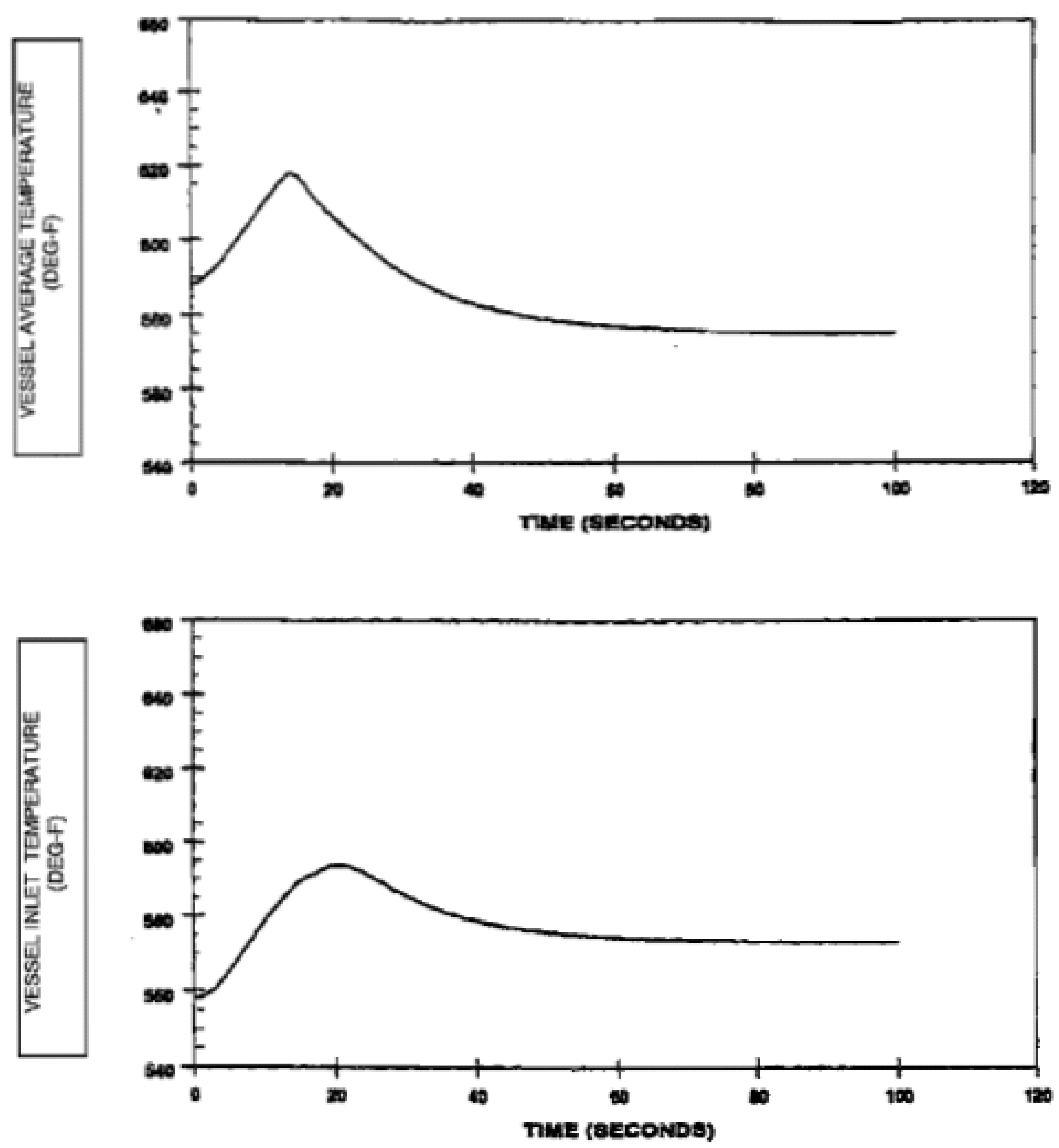

Figure 15.2-21 Loss of Load Accident with Pressurizer Spray and Power-Operated Relief Valves

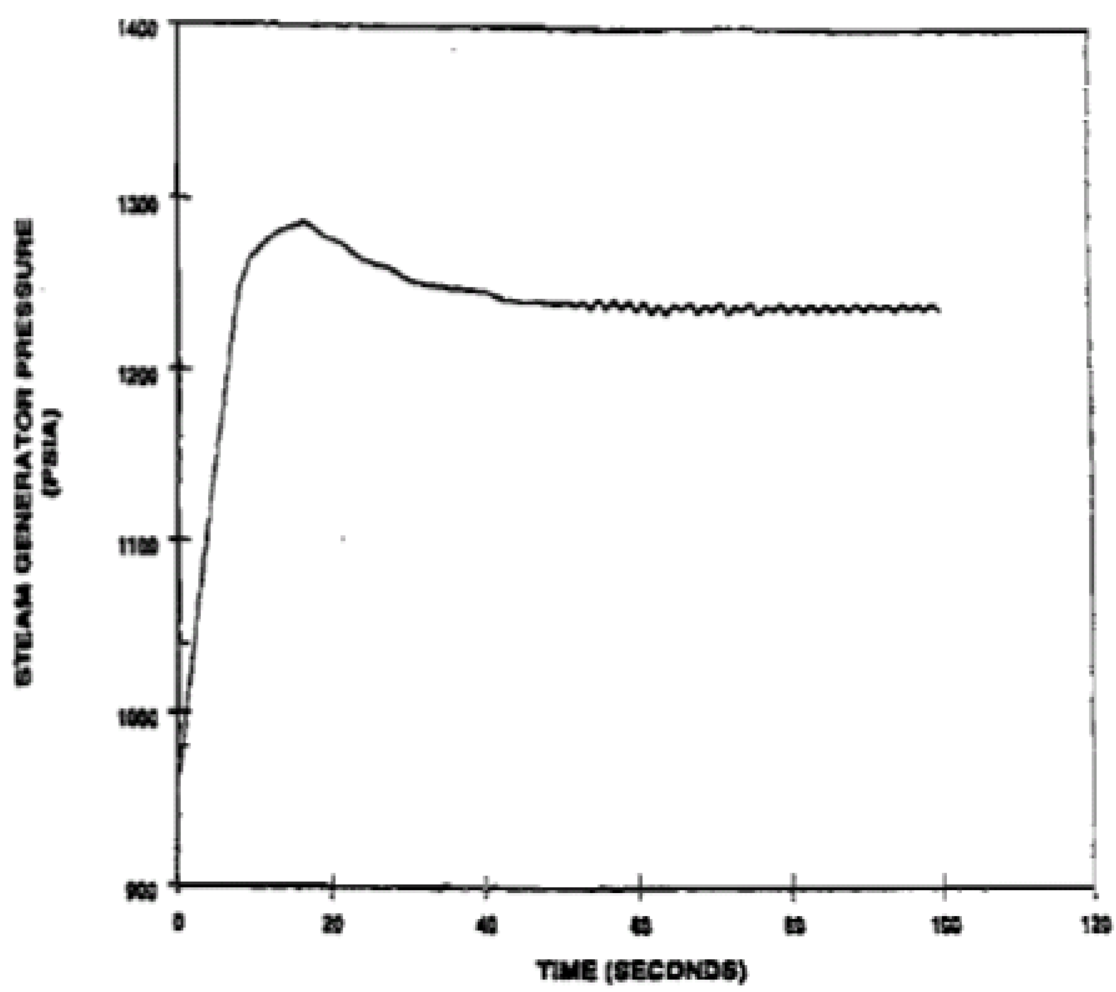

Figure 15.2-22 Loss of Load Accident with Pressurizer Spray and Power-Operated Relief Valves

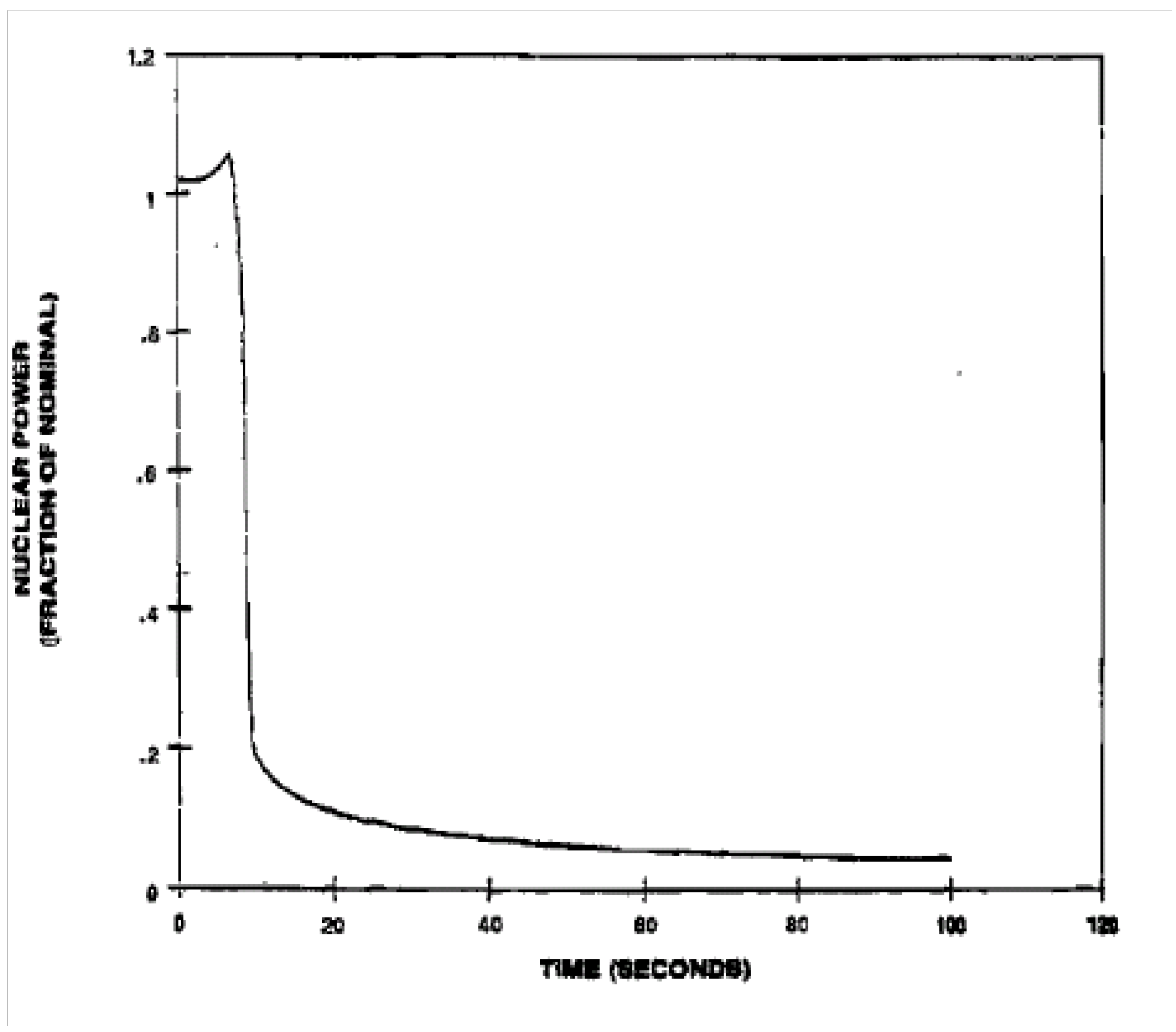

Figure 15.2-23 Loss of Load Accident Without Pressurizer Spray and Power-Operated Relief Valves

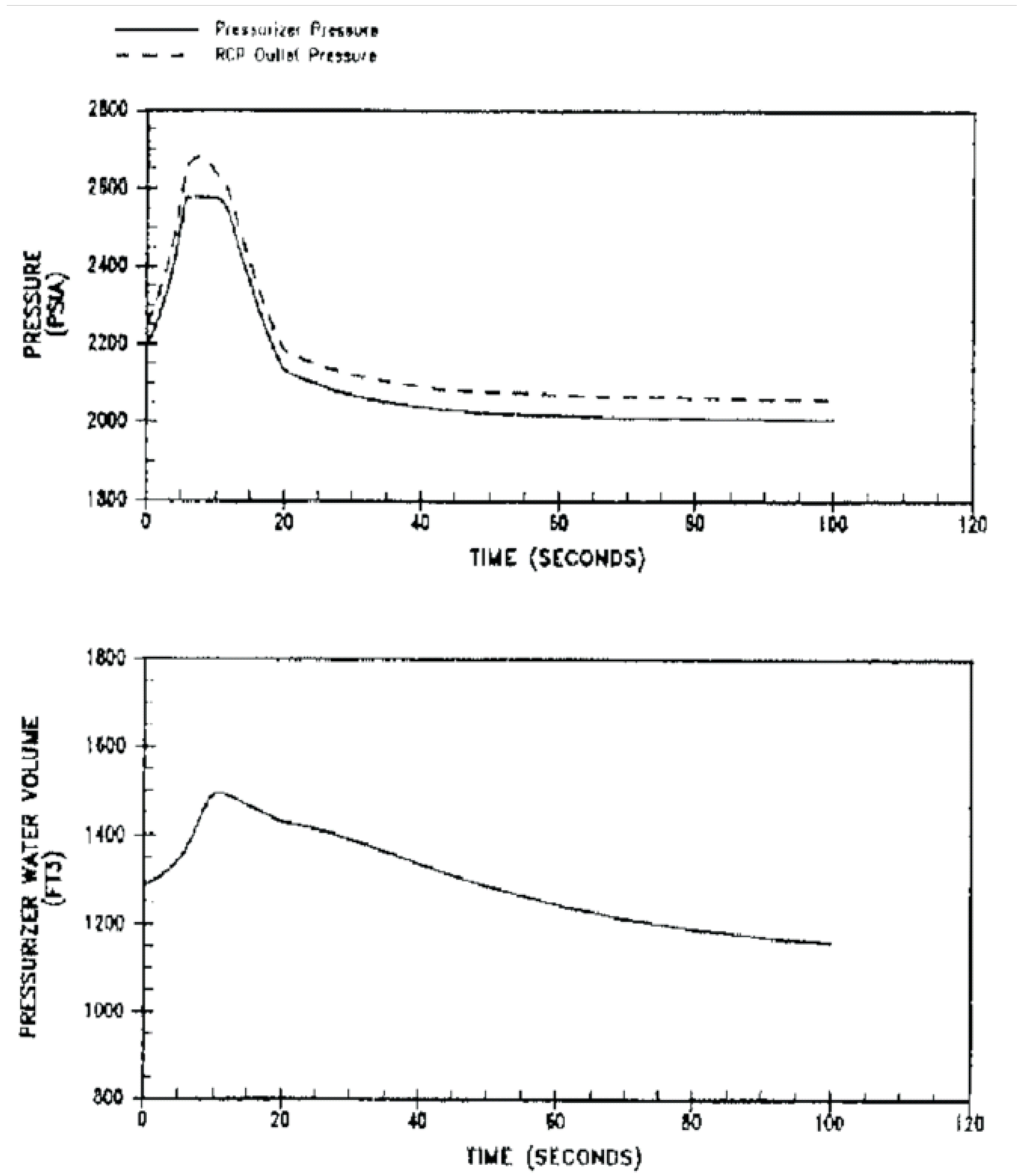

Figure 15.2-24 Loss of Load Accident Without Pressurizer Spray and Power-Operated Relief Valves

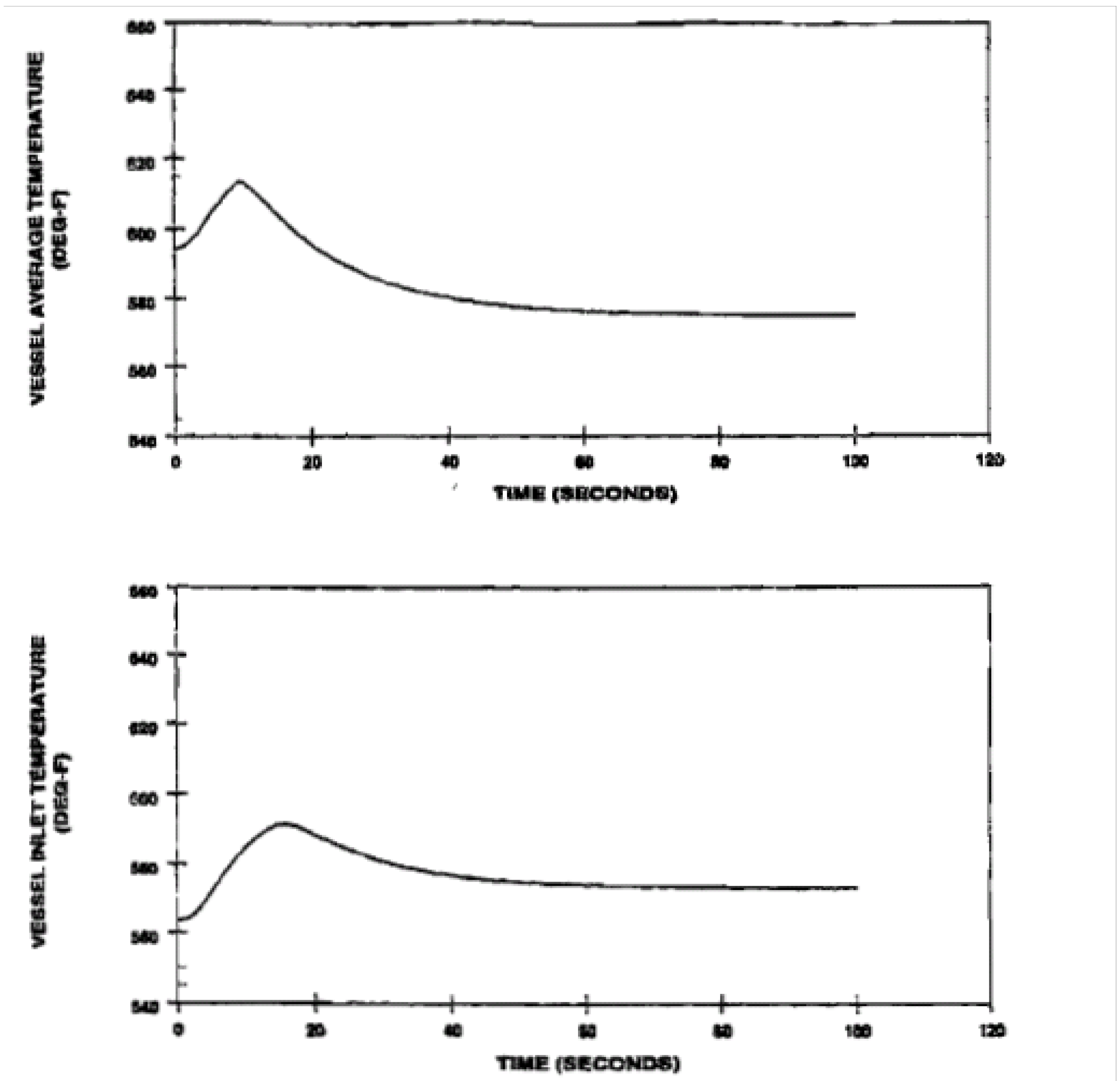

Figure 15.2-25 Loss of Load Accident Without Pressurizer Spray and Power-Operated Relief Valves

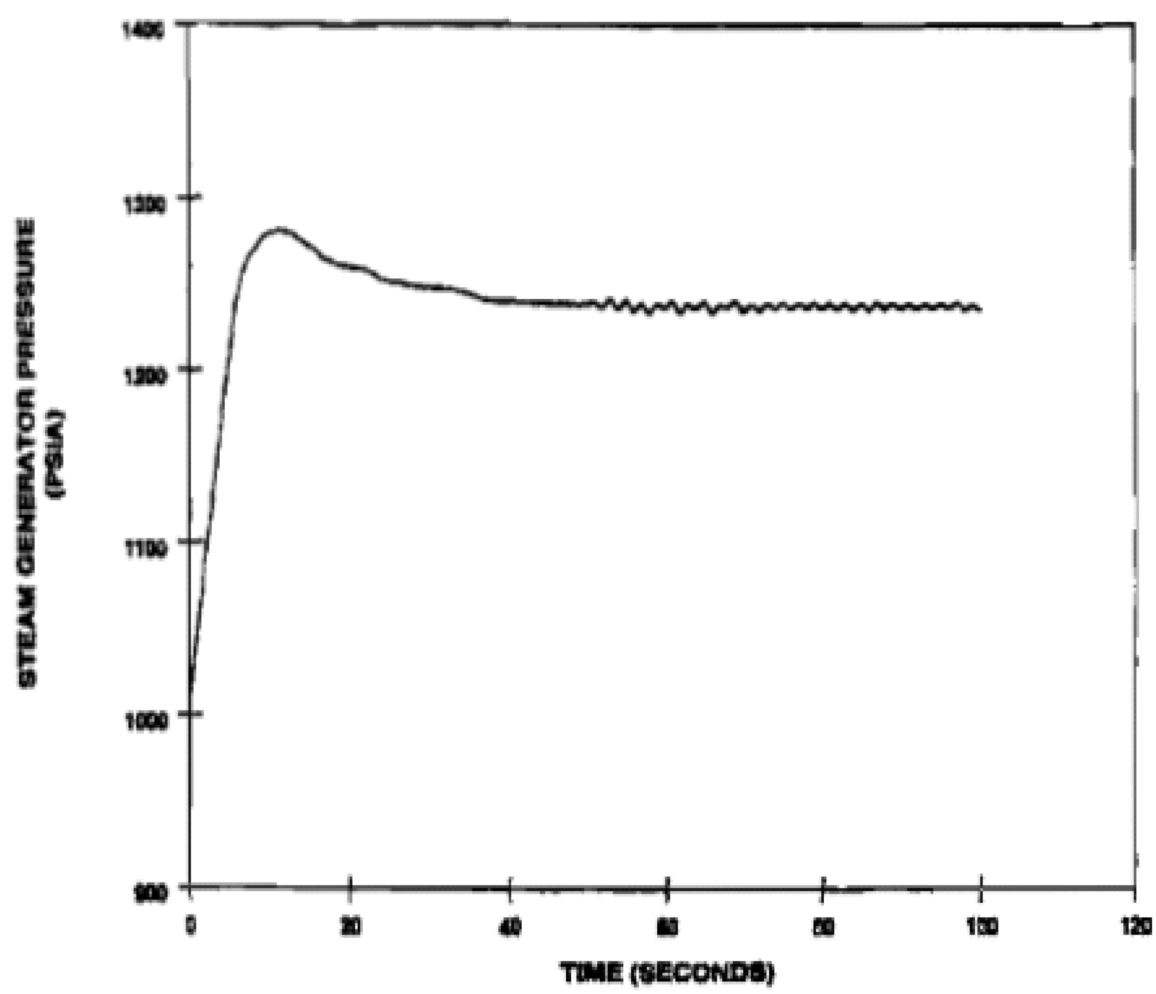

Figure 15.2-26 Loss of Load Accident Without Pressurizer Spray and Power-Operated Relief Valves

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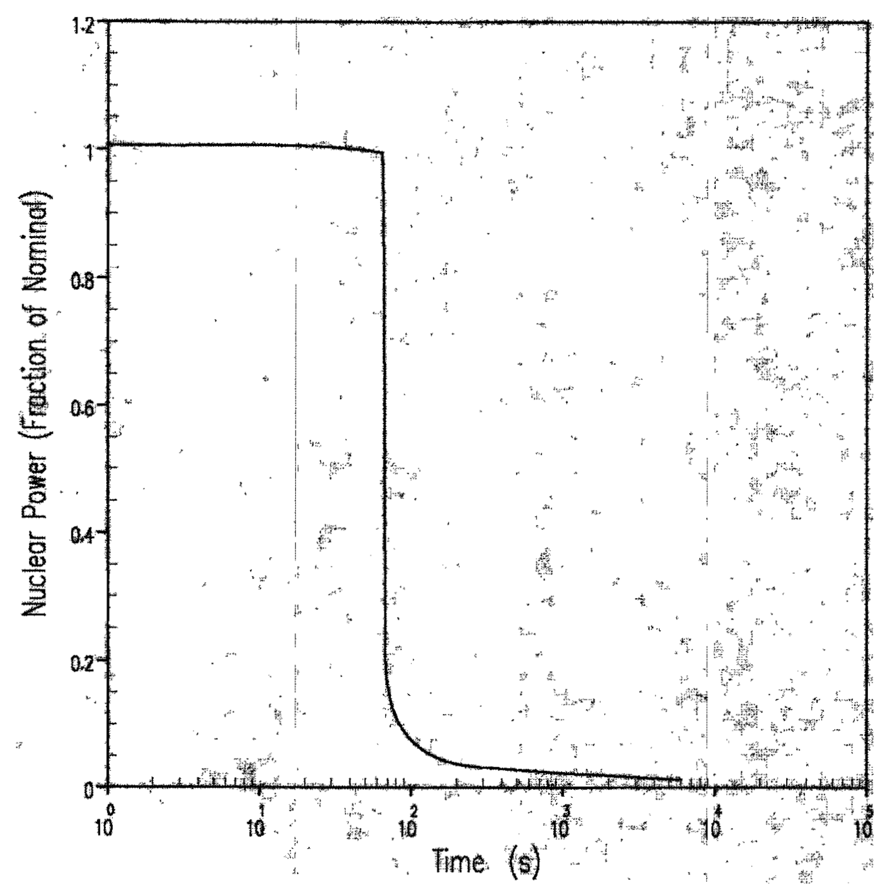

WATTS BAR NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

LOSS OF NORMAL FEEDWATER
NUCLEAR POWER VERSUS TIME
FIGURE 15.2-27a

Figure 15.2-27a Loss Of Normal Feedwater Nuclear Power Versus Time

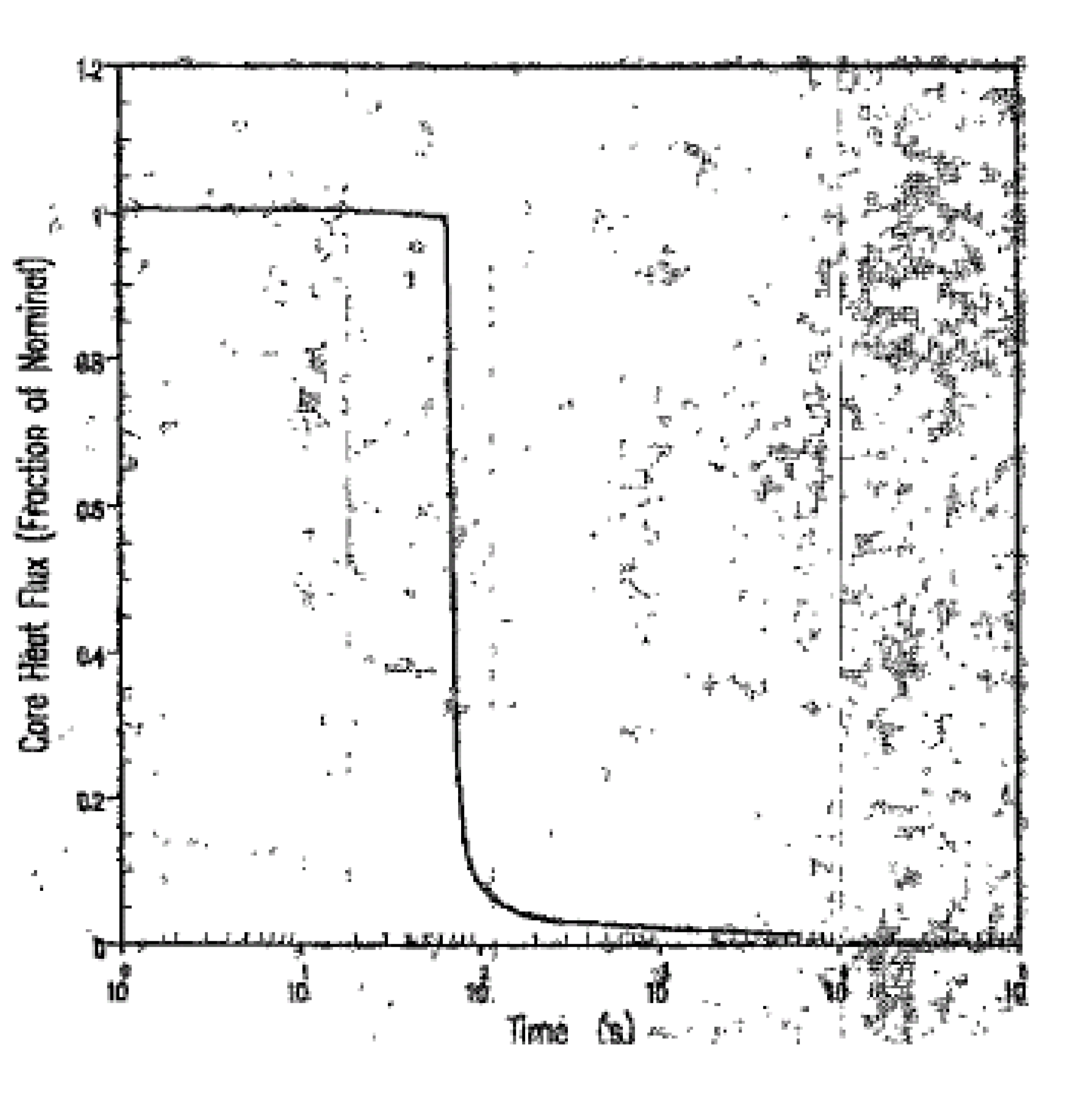

Figure 15.2-27b Loss of Normal Feedwater Core Heat Flux Versus Time

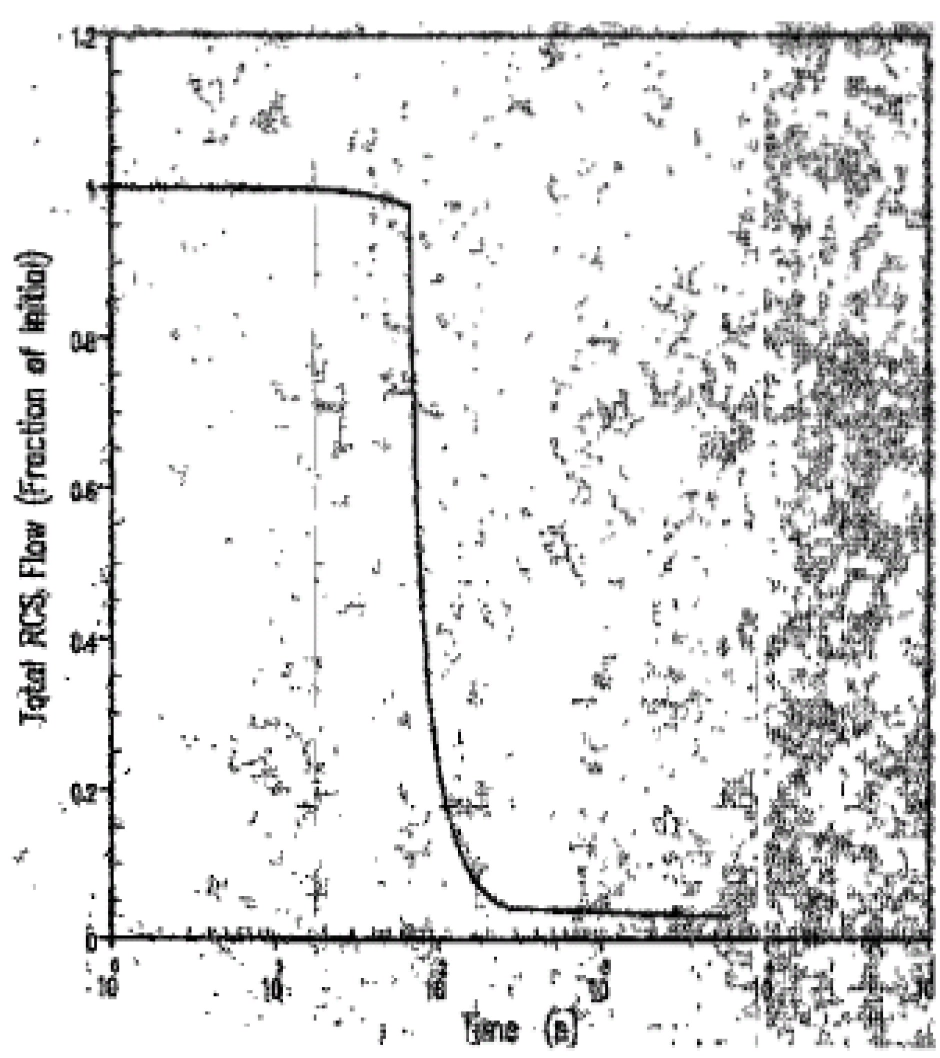

Figure 15.2-27c Loss of Normal FeedwaterTotal RCS Flow Versus Time

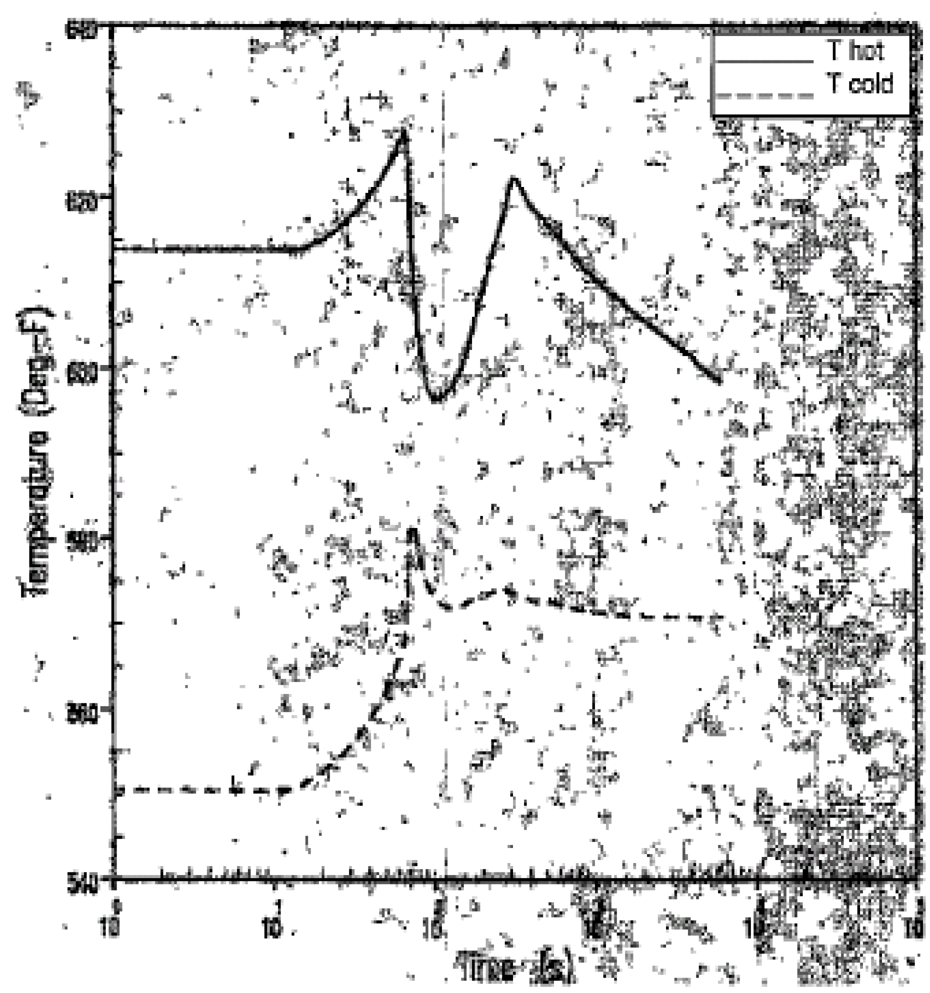

Figure 15.2-27d Loss of Normal Feedwater Reactor Coolant System Temperature Transient Versus Time

Figure 15.2-27e Deleted by Amendment 72

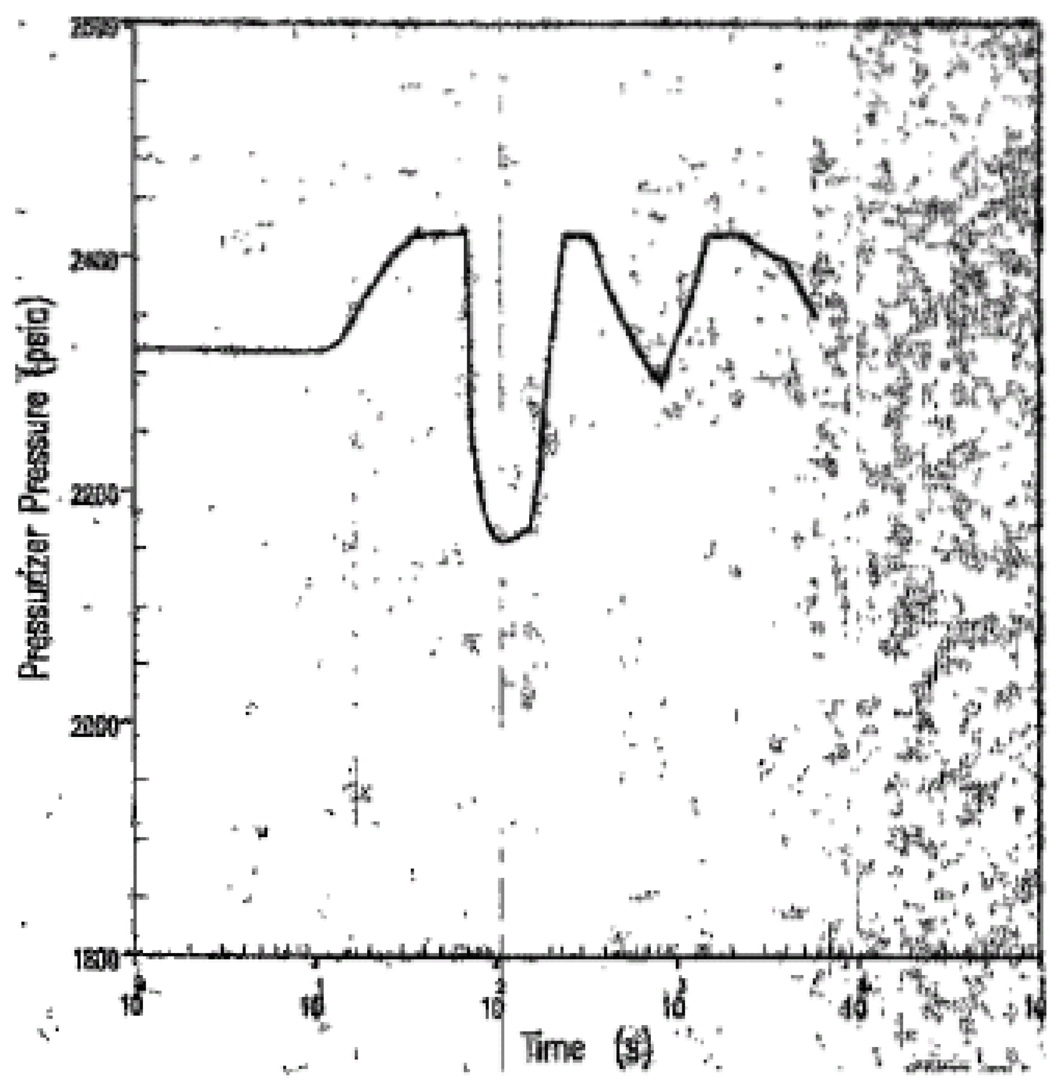

Figure 15.2-27f Loss of Normal Feedwater Pressurizer Pressure Versus time

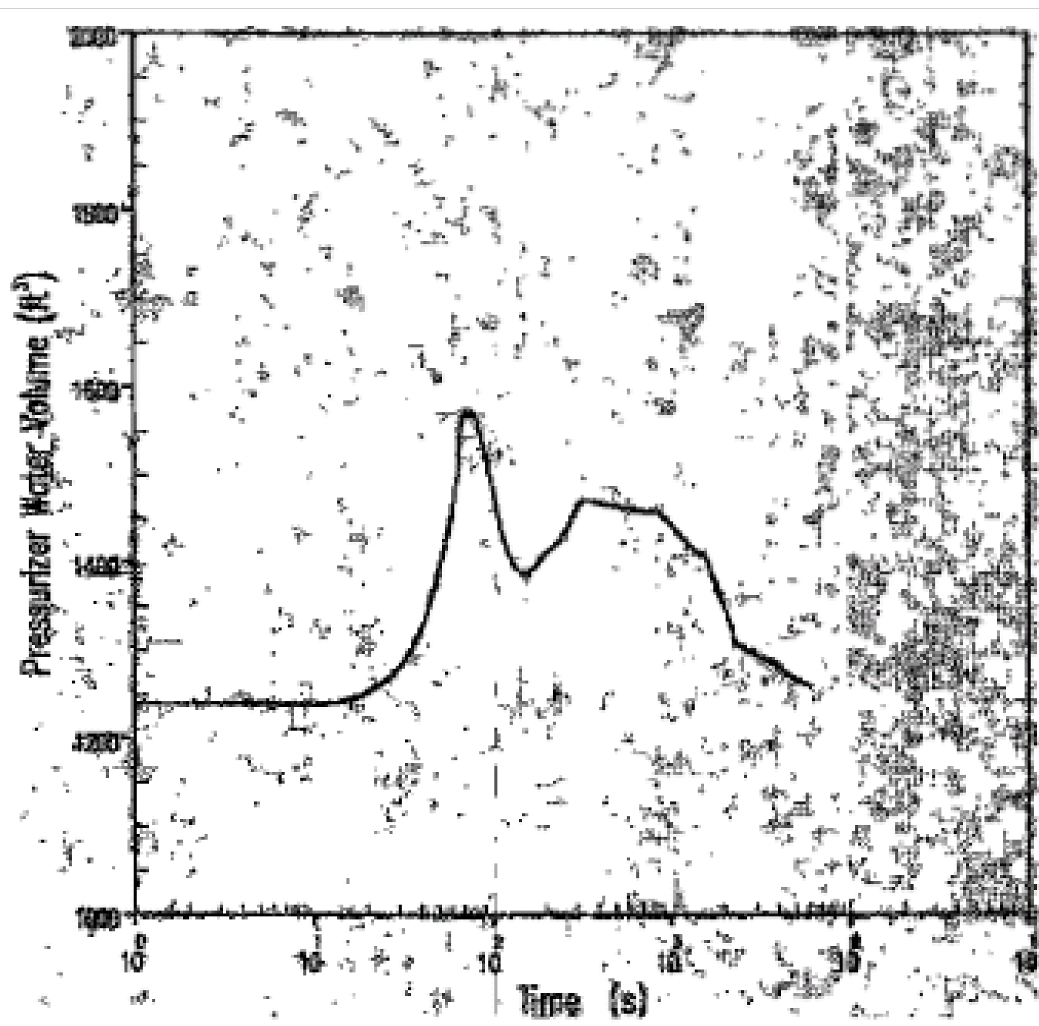

Figure 15.2-27g Loss of Normal Feedwater Pressurizer Water Volume Versus Time

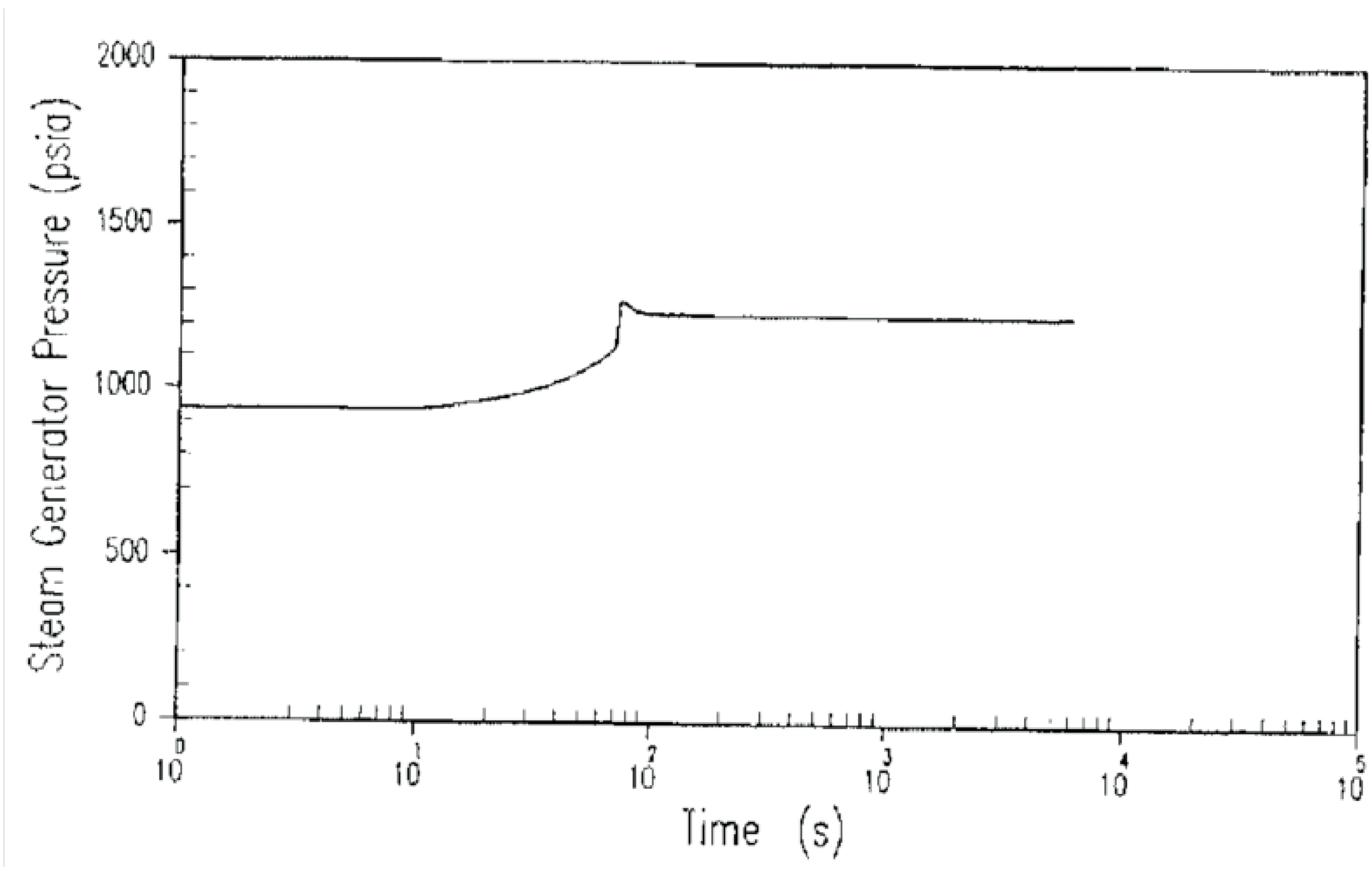

Figure 15.2-27h Loss of Normal Feedwater Steam Generator Pressure Versus Time

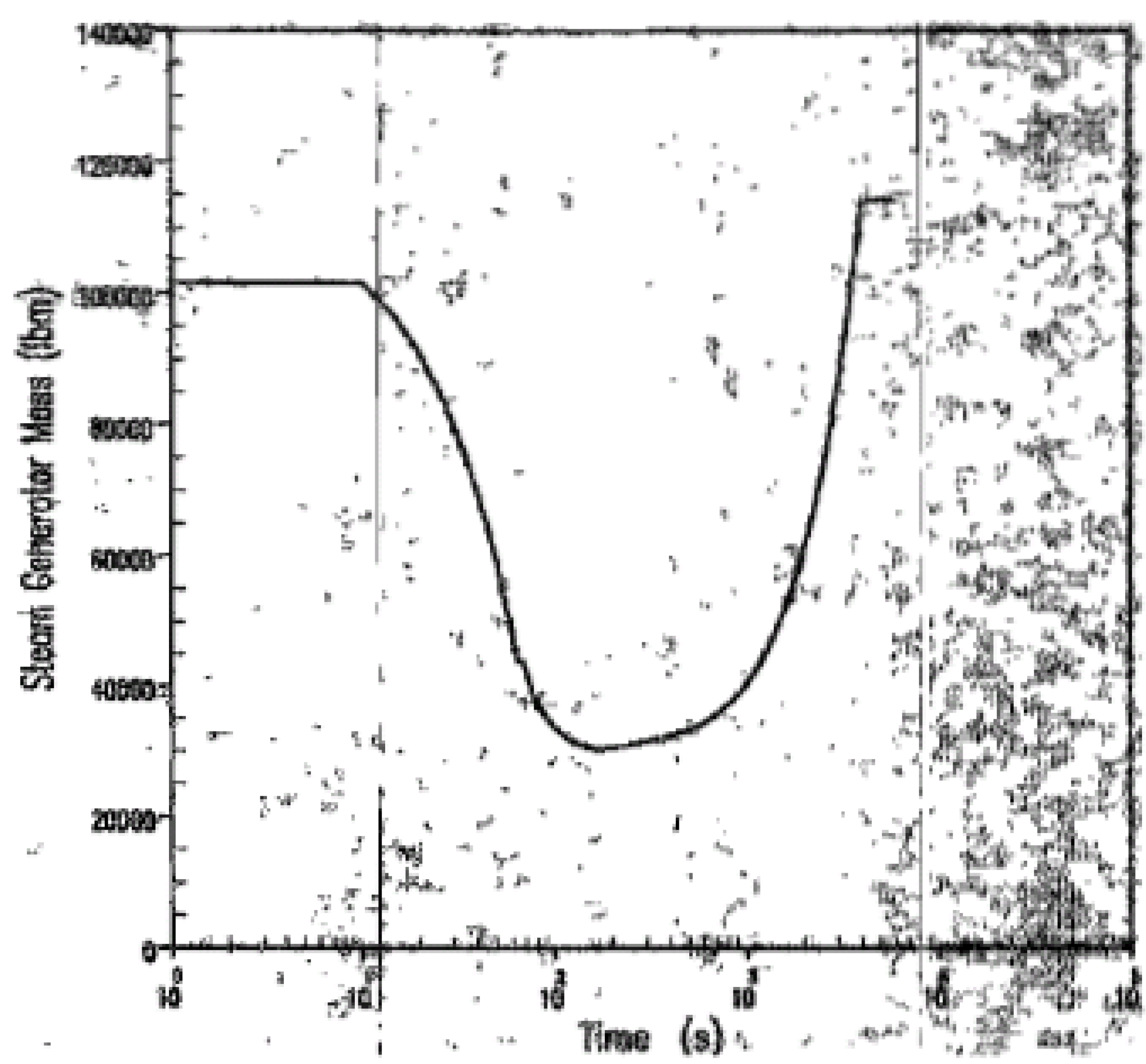

Figure 15.2-27i Loss of Normal Feedwater Steam Generator Mass Versus Time

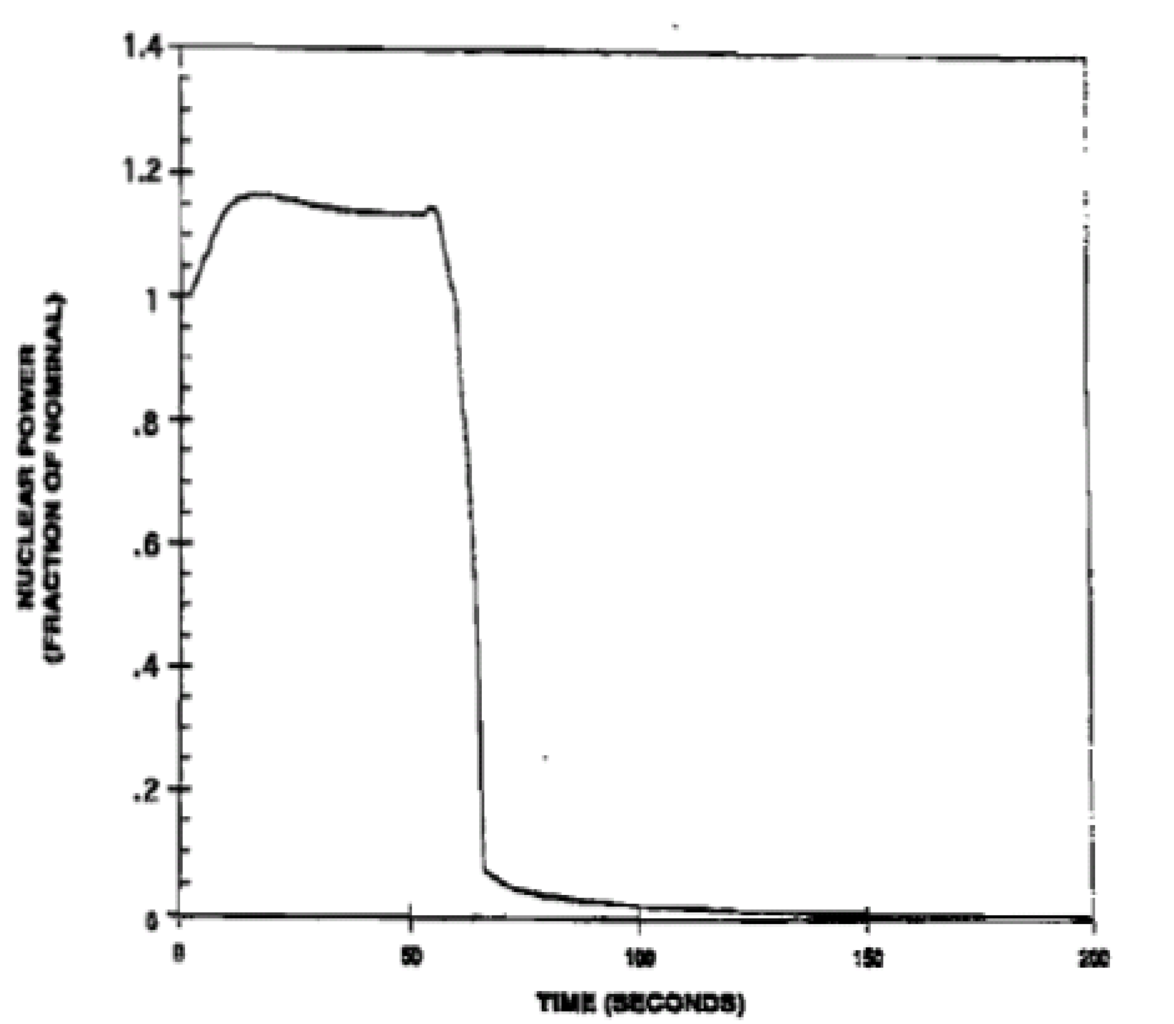

Figure 15.2-28a Single Feedwater Control Valve Malfunction, Excess Feedwater with Manual Rod Control - Nuclear Power Versus Time

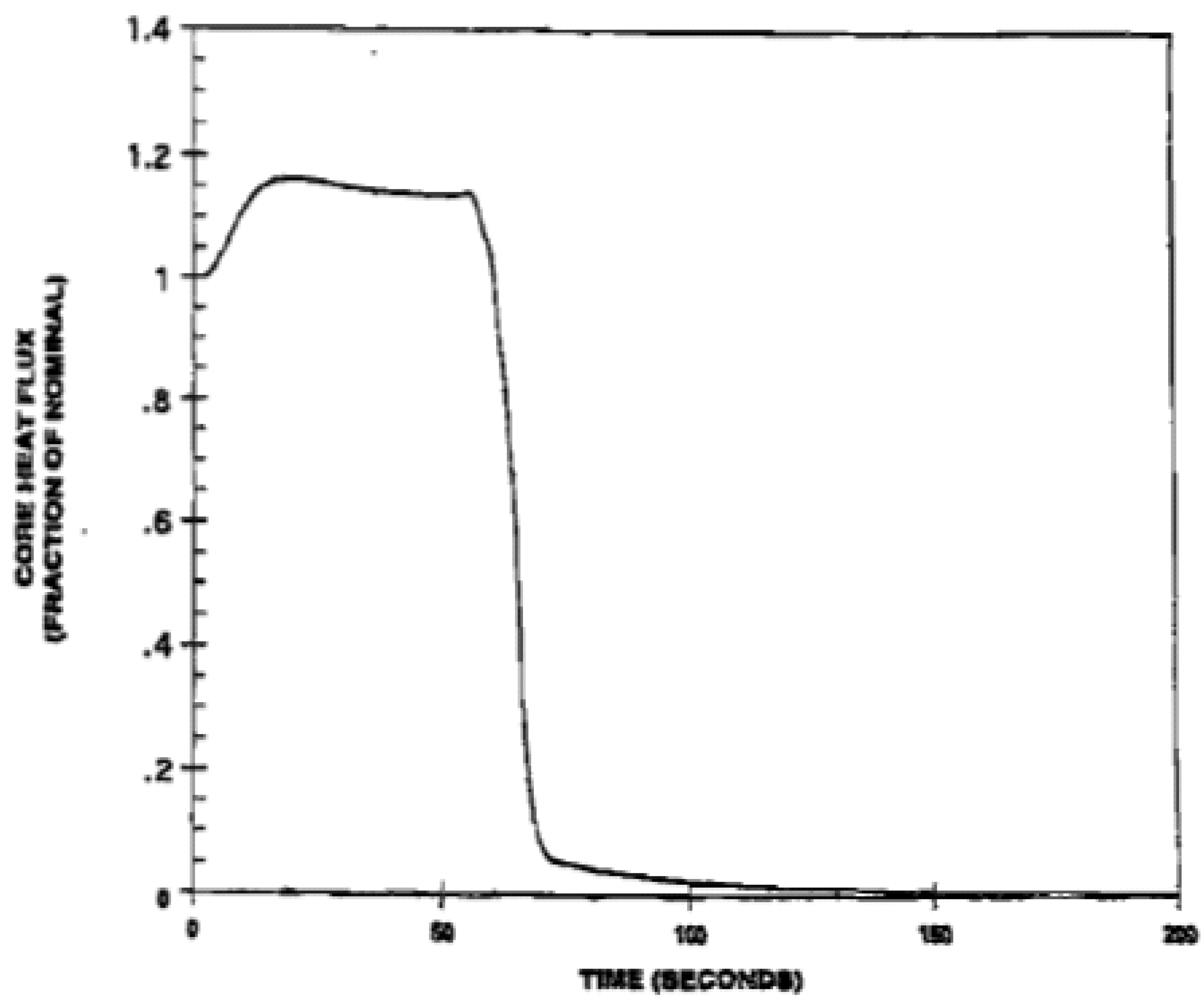

Figure 15.2-28b Single Feedwater Control Valve Malfunction, Excess Feedwater with Manual Rod Control - Core Heat Flux Versus Time

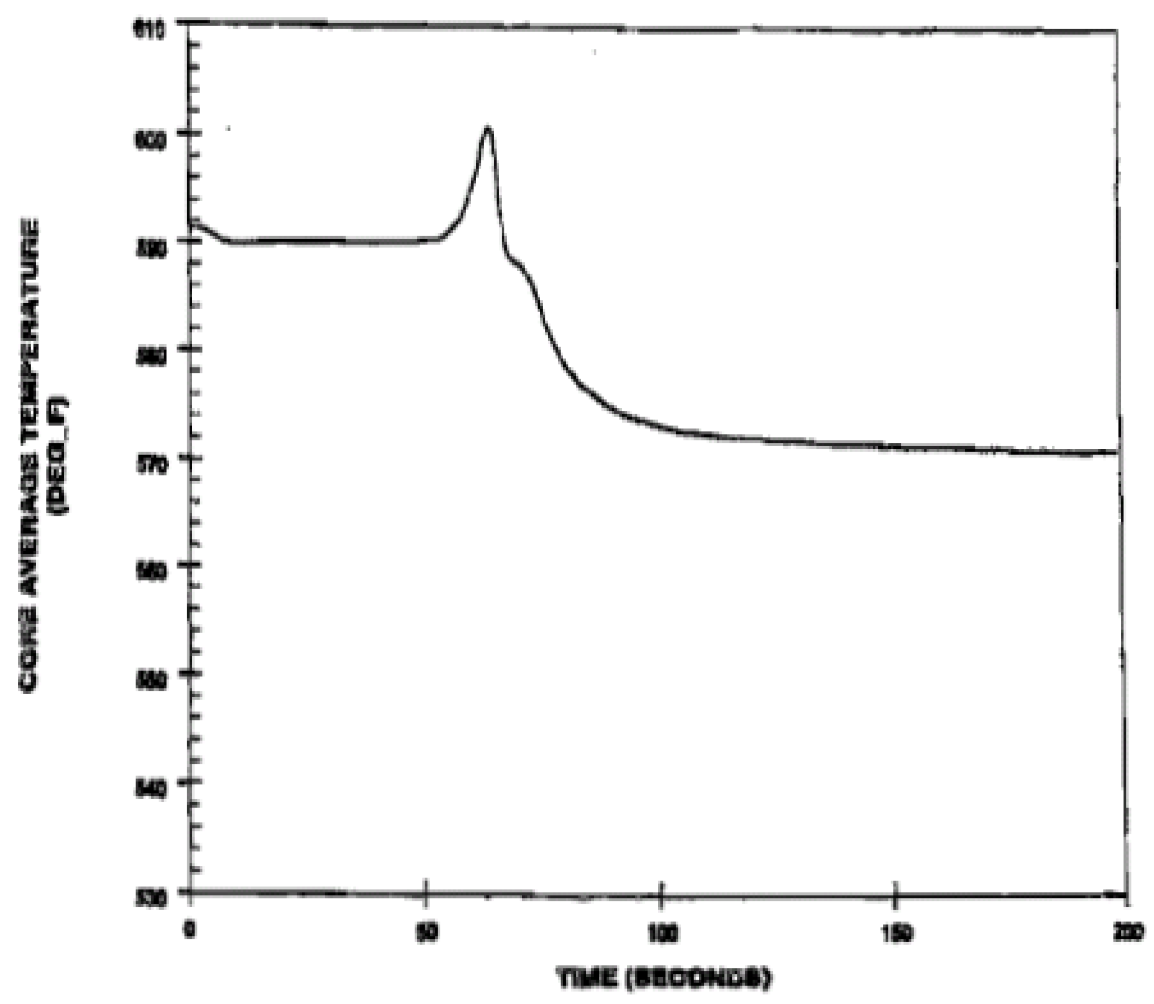

Figure 15.2-28c Single Feedwater Control Valve Malfunction, Excess Feedwater with Manual Rod Control - Core Average Temp Versus Time

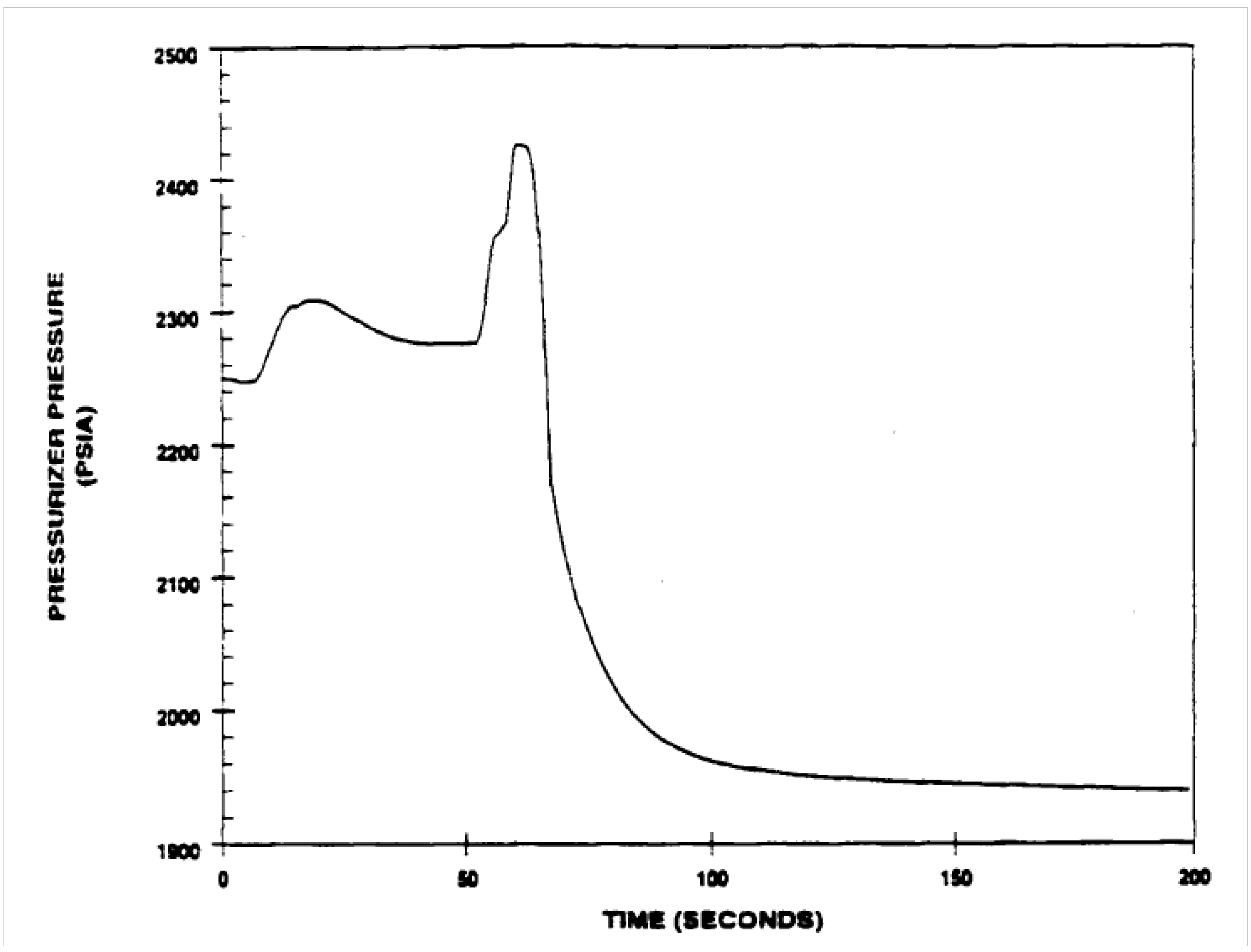

Figure 15.2-28d Single Feedwater Control Valve Malfunction, Excess Feedwater with Manual Rod Control - Pressurizer Pressure Versus Time

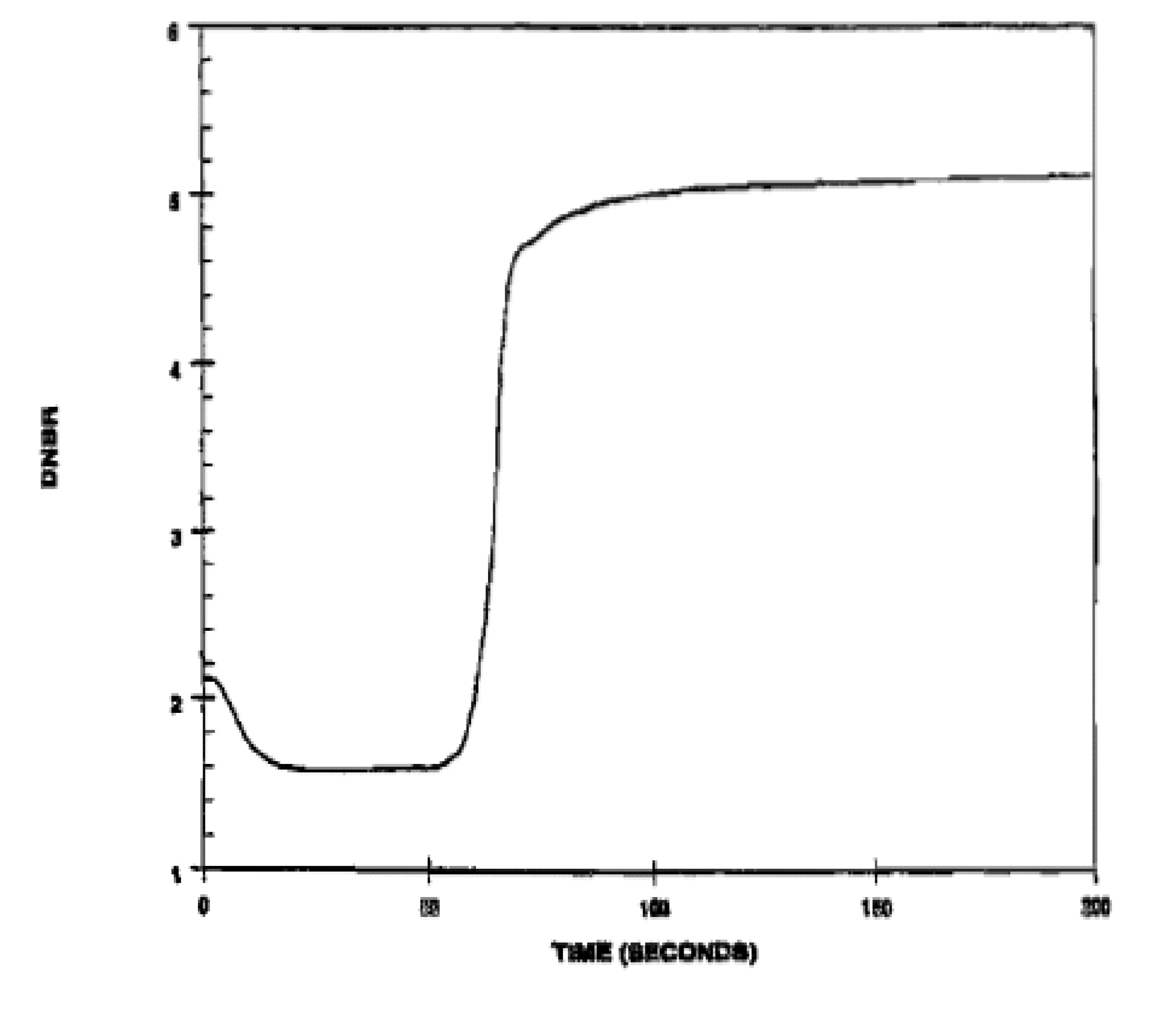

Figure 15.2-28e Single Feedwater Control Valve Malfunction, Excess Feedwater with Manual Rod Control -DNBR Versus Time

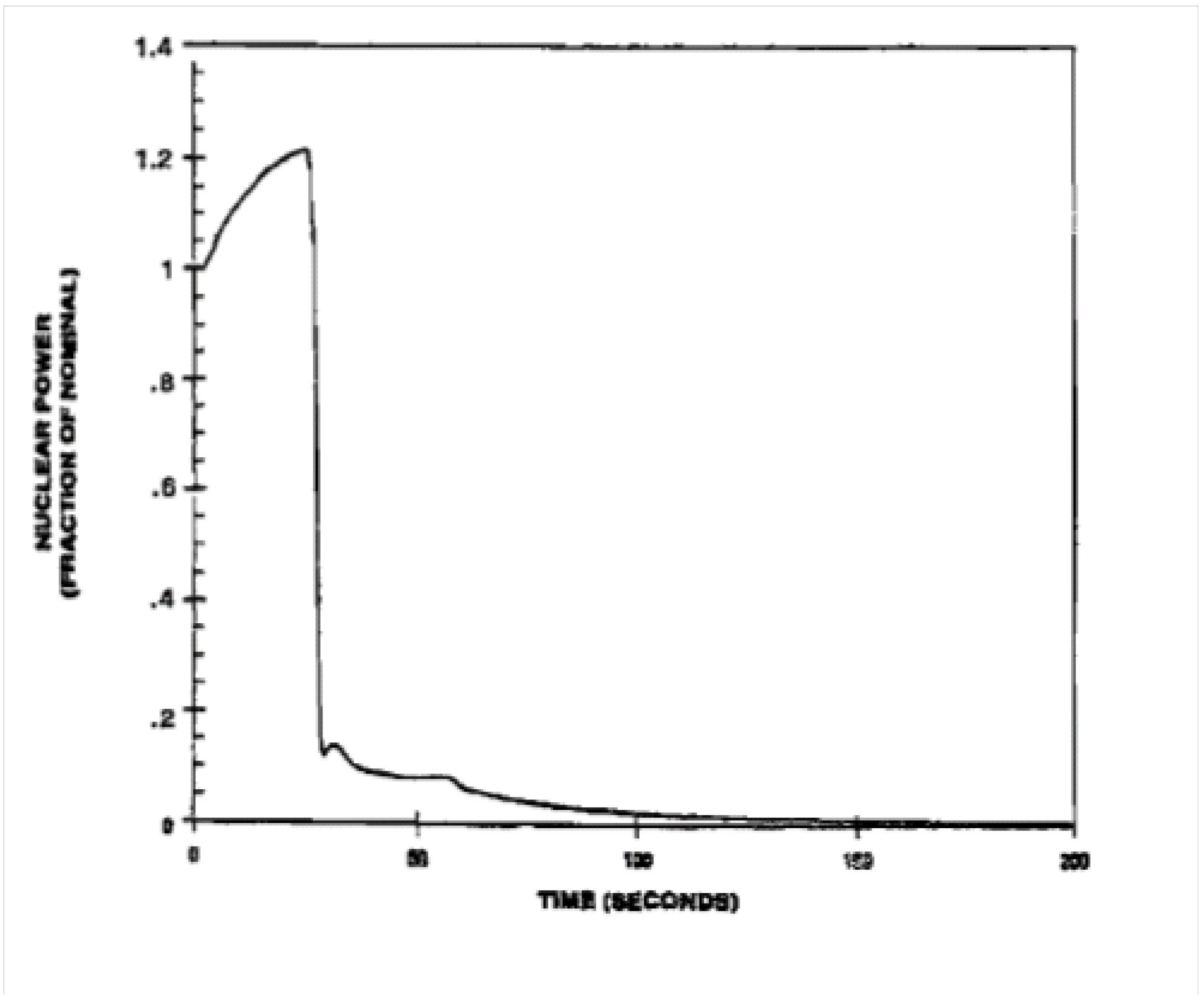

Figure 15.2-28f Multiple Feedwater Control Valve Malfunction, Excess Feedwater with Manual Rod Control -Nuclear Power Versus Time

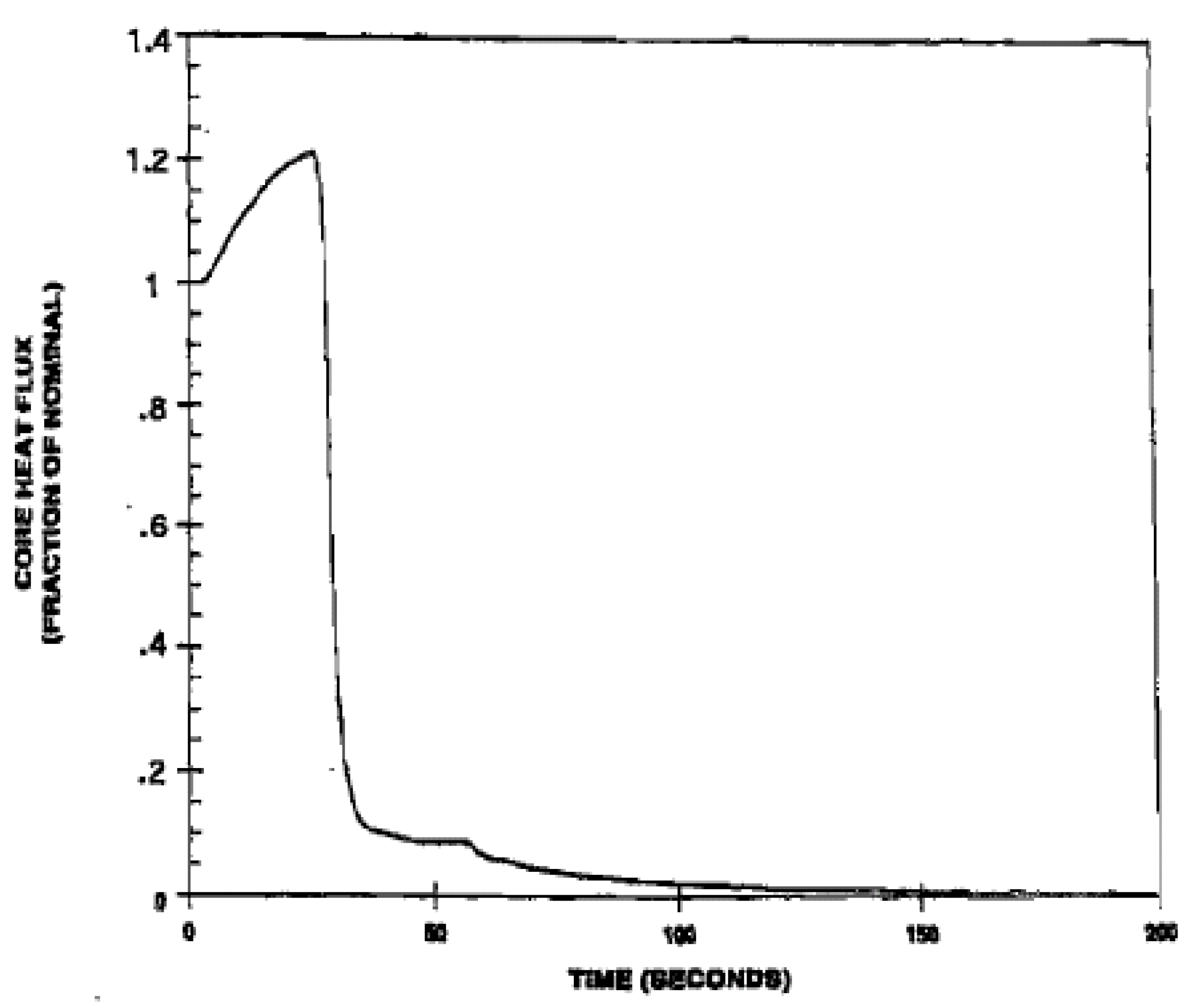

Figure 15.2-28g Multiple Feedwater Control Valve Malfunction, Excess Feedwater with Manual Rod Control -Core Heat Flux Versus Time

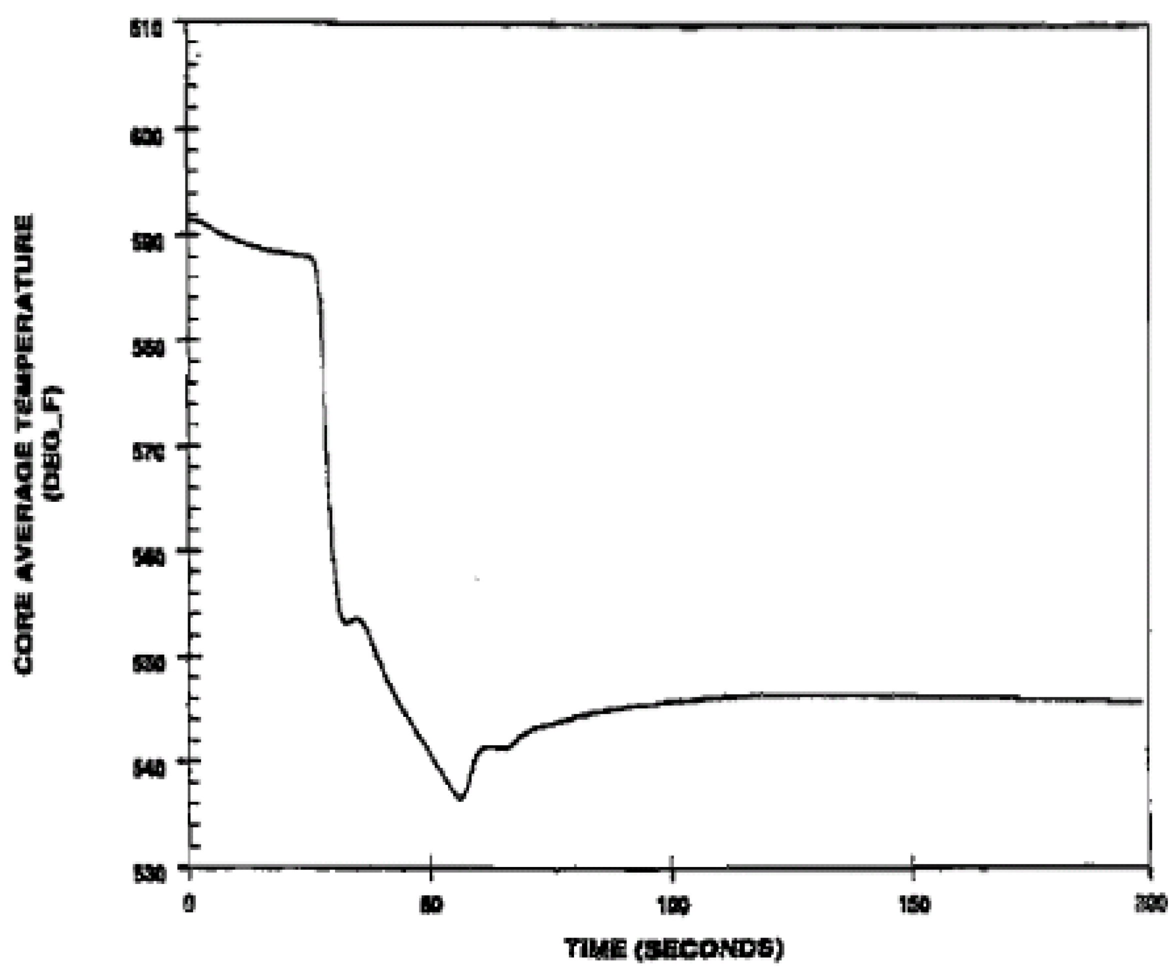

Figure 15.2-28h Multiple Feedwater Control Valve Malfunction, Excess Feedwater with Manual Rod Control - Core Average Temp Versus Time

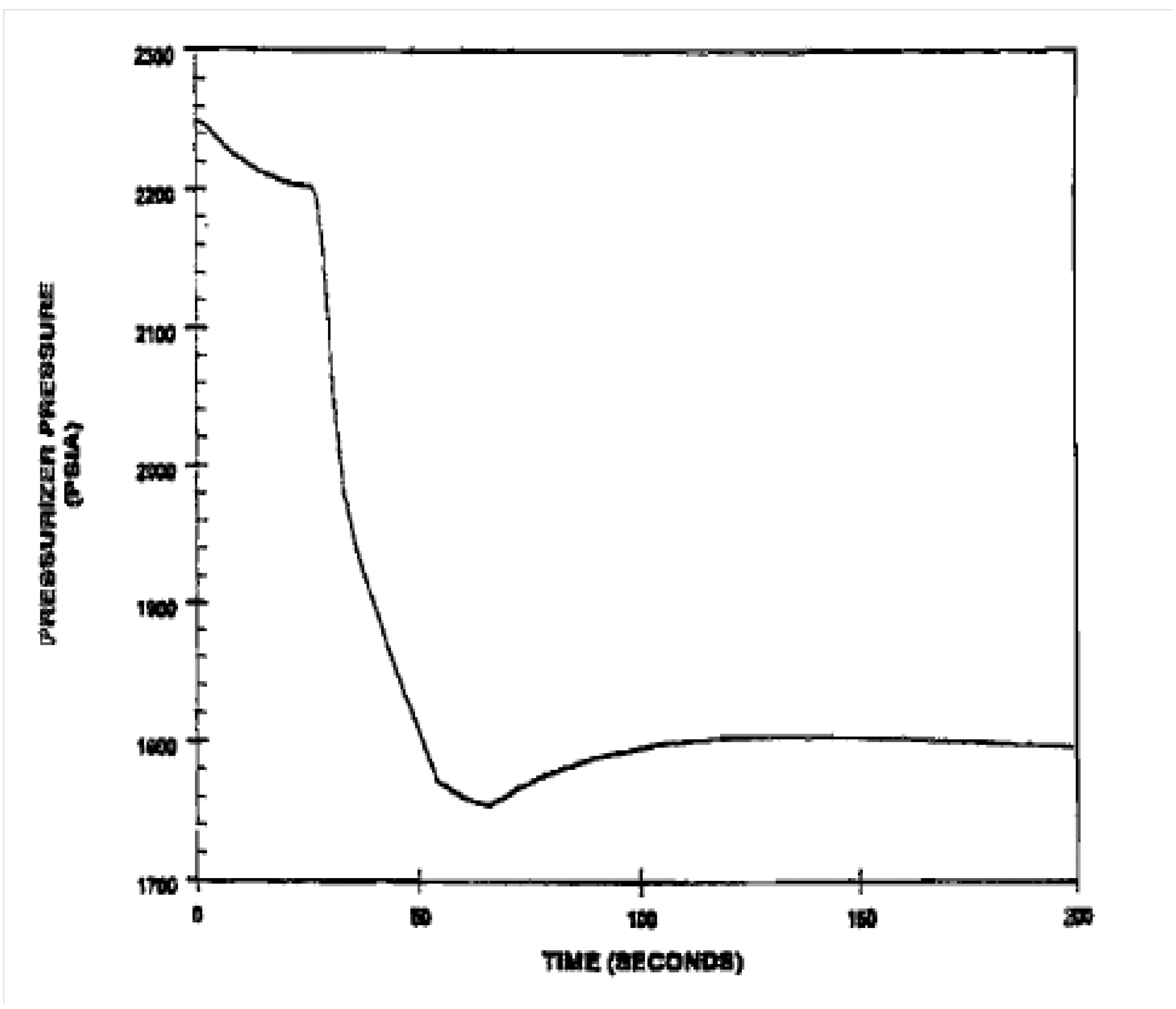

Figure 15.2-28i Multiple Feedwater Control Valve Malfunction, Excess Feedwater with Manual Rod Control - Pressurizer Pressure Versus Time

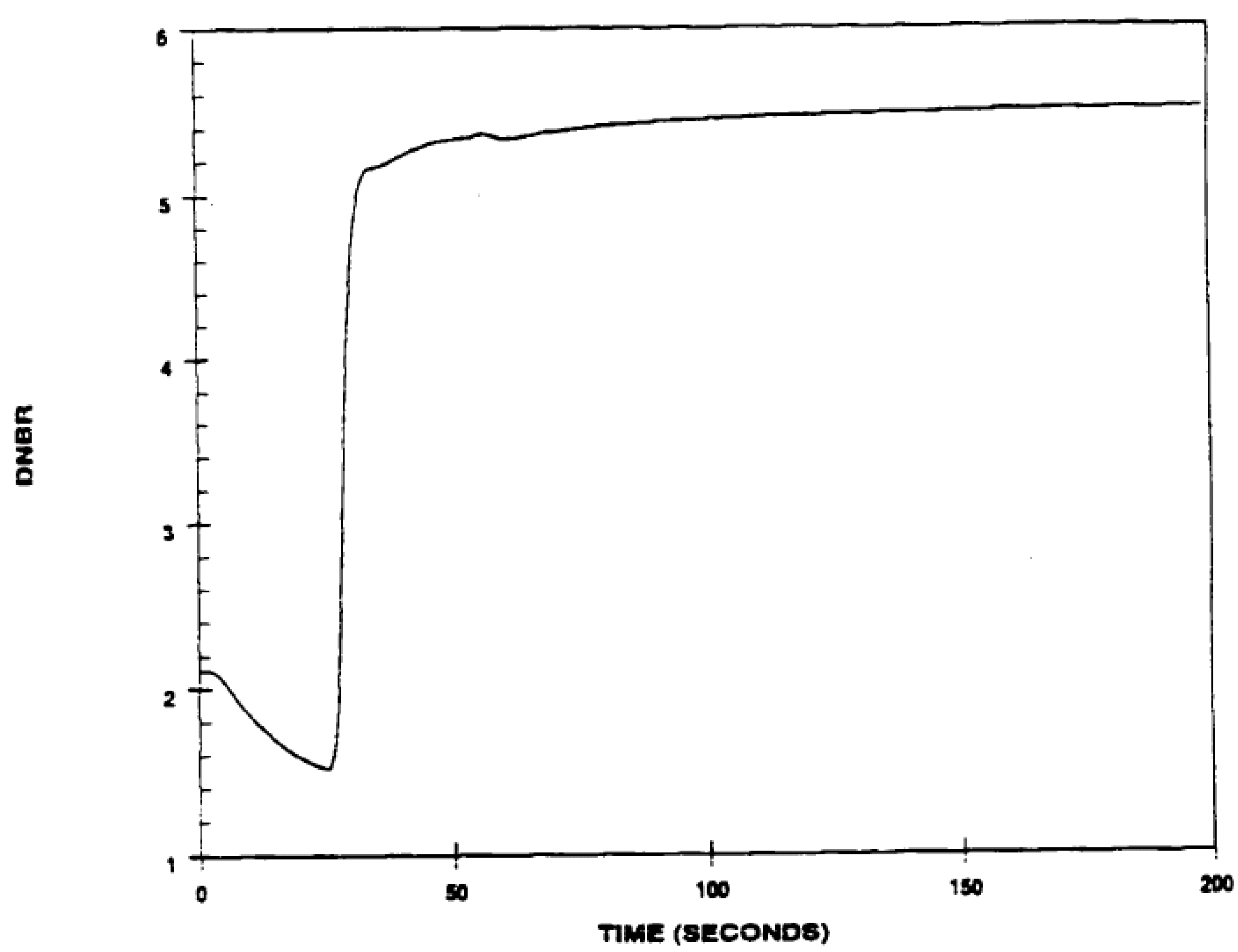

Figure 15.2-28j Multiple Feedwater Control Valve Malfunction, Excess Feedwater with Manual Rod Control -DNBR Versus Time

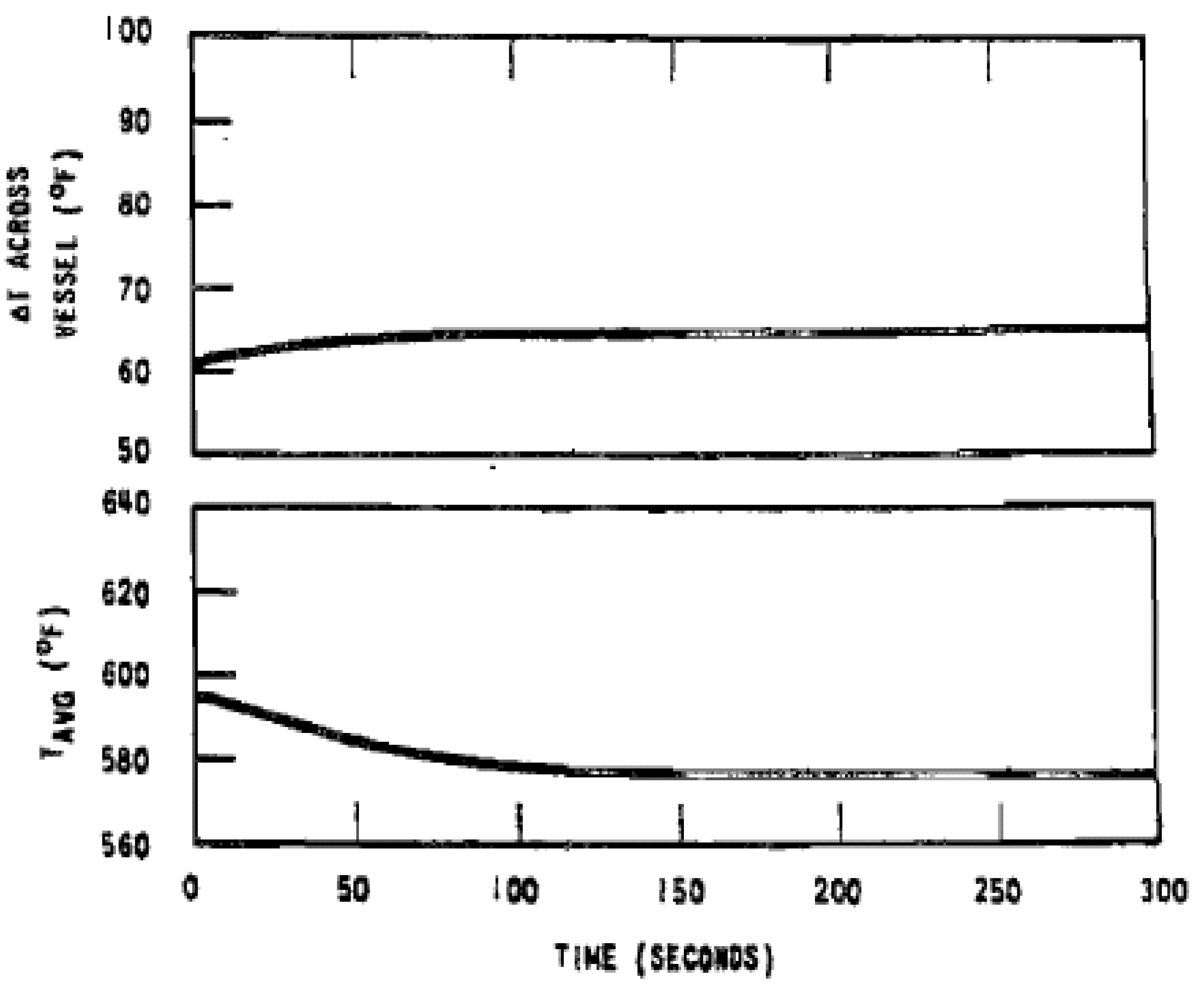

Figure 15.2-29 Typical Transient-10% Step Load Increase, Beginning of Life , Manual Reactor Control

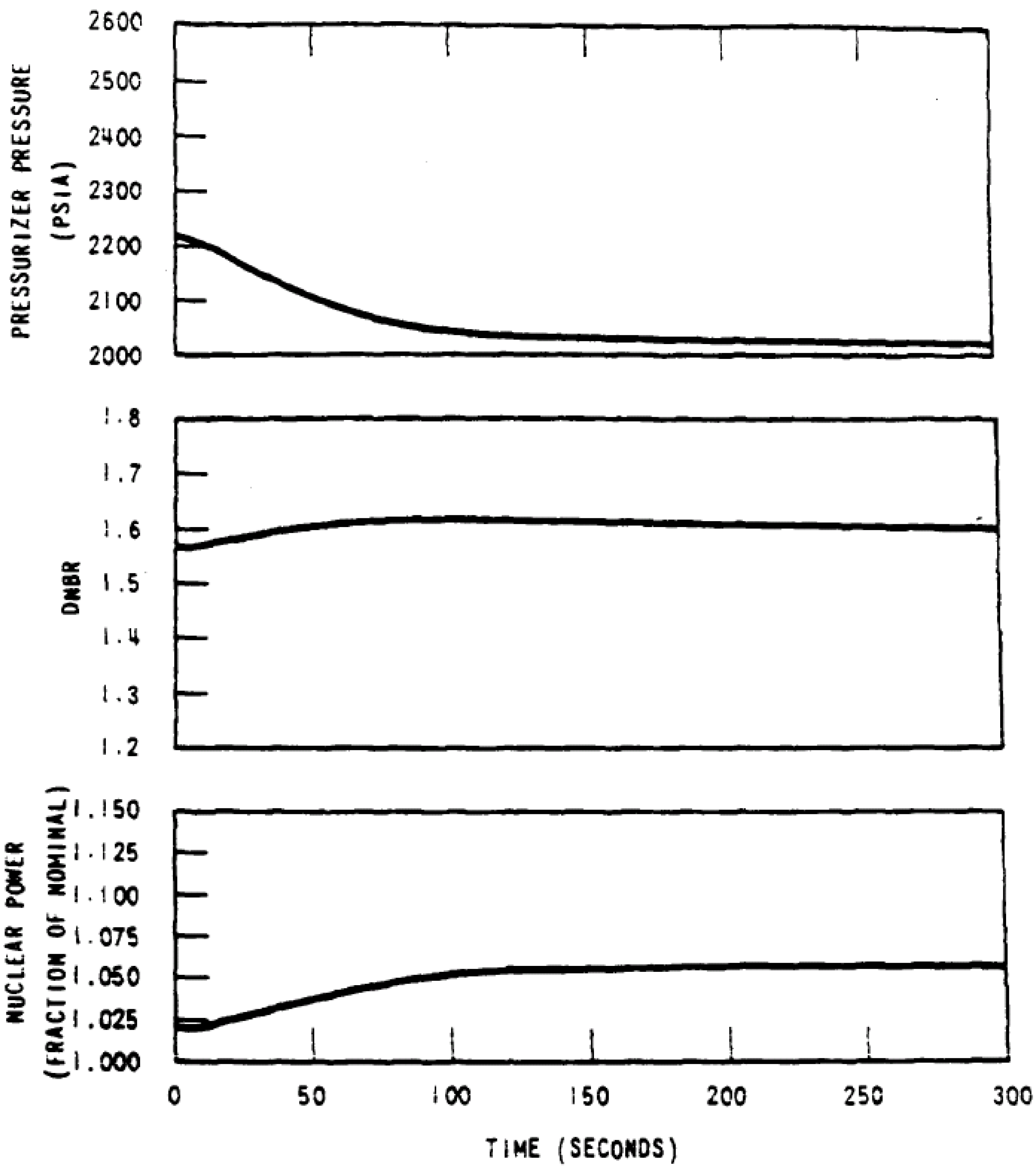

Figure 15.2-30 Typical Transient-10% Step Load Increase, Beginning of Life, Manual Reactor Control

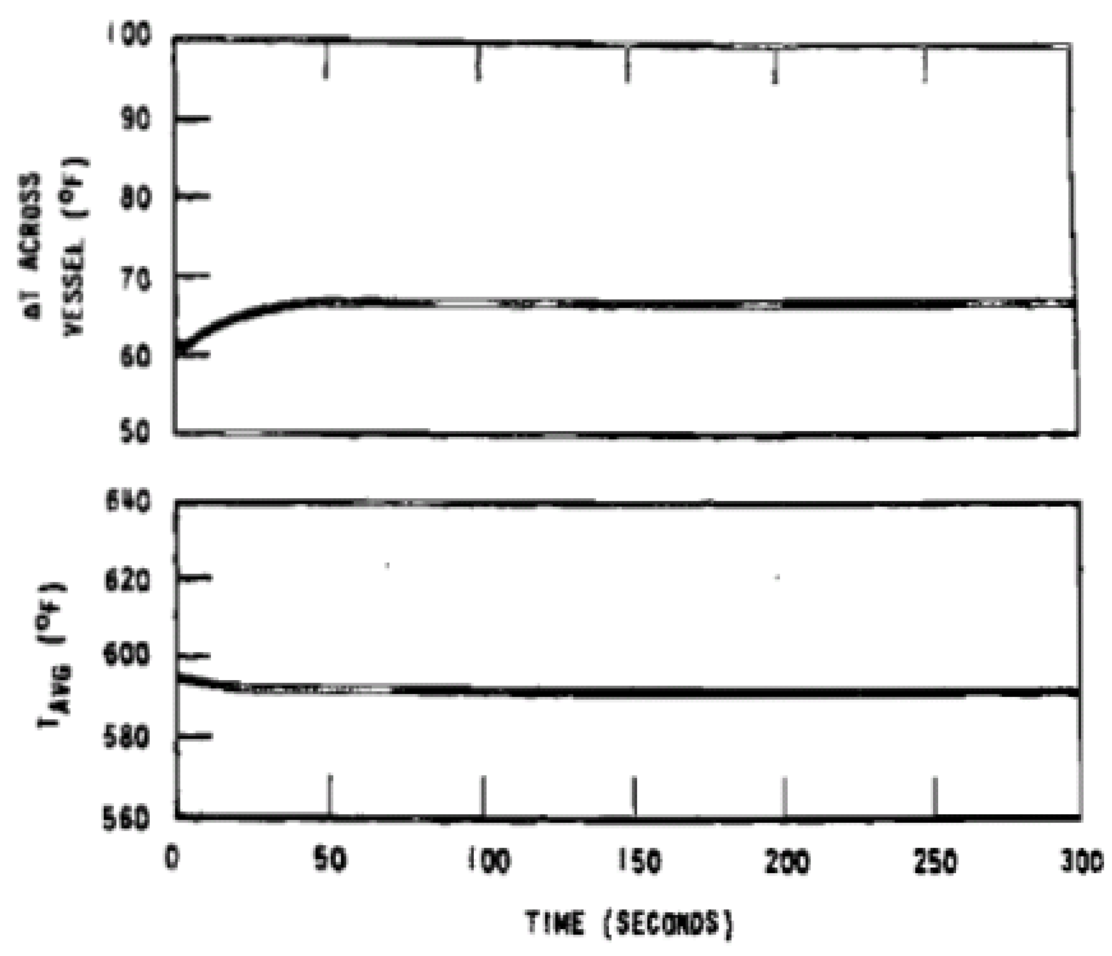

Figure 15.2-31 Typical Transient-10% Step Load Increase, End of Life, Manual Reactor Control

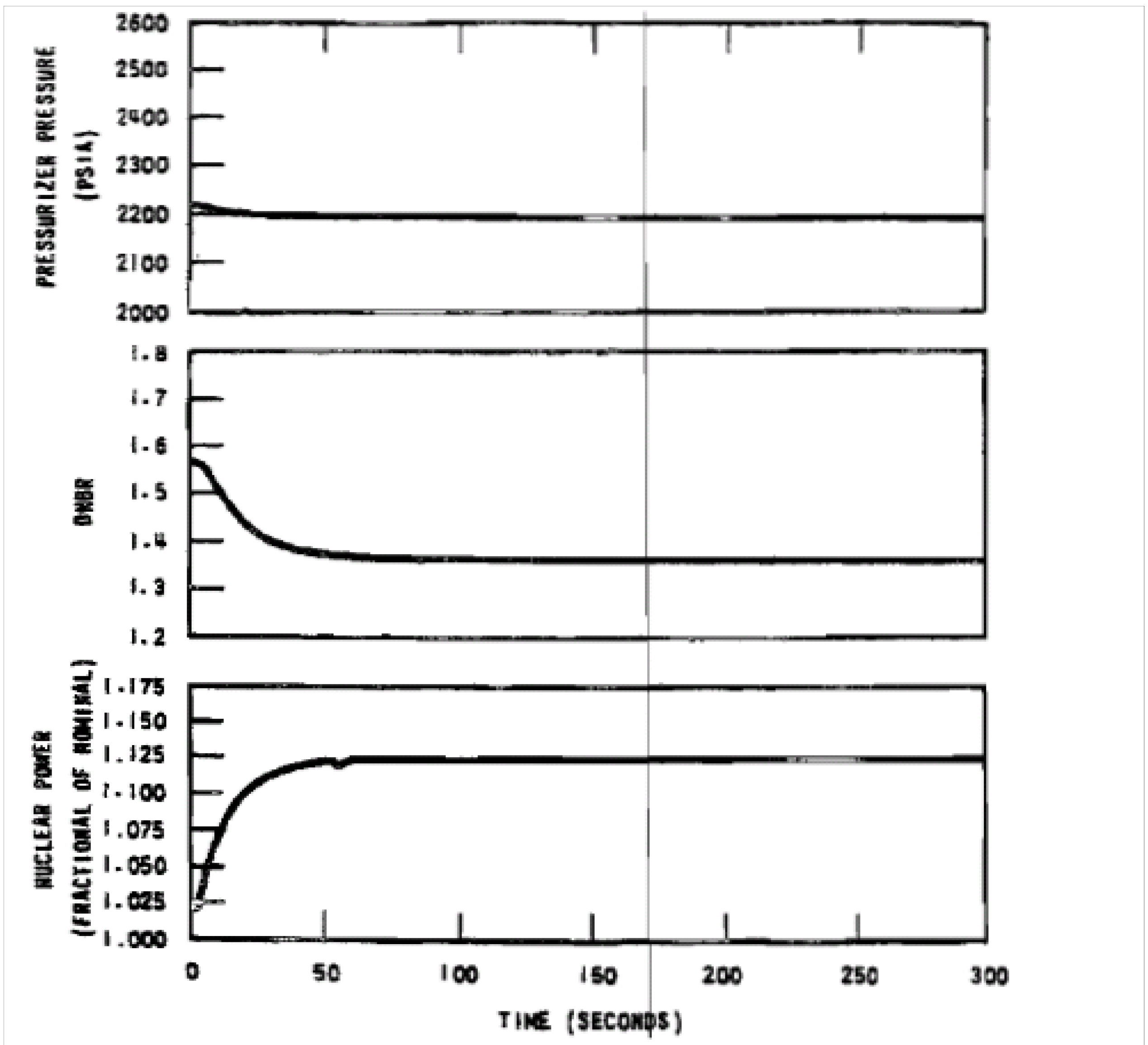

Figure 15.2-32 Typical Transient-10% Step Load Increase, End of Life, Manual Reactor Control

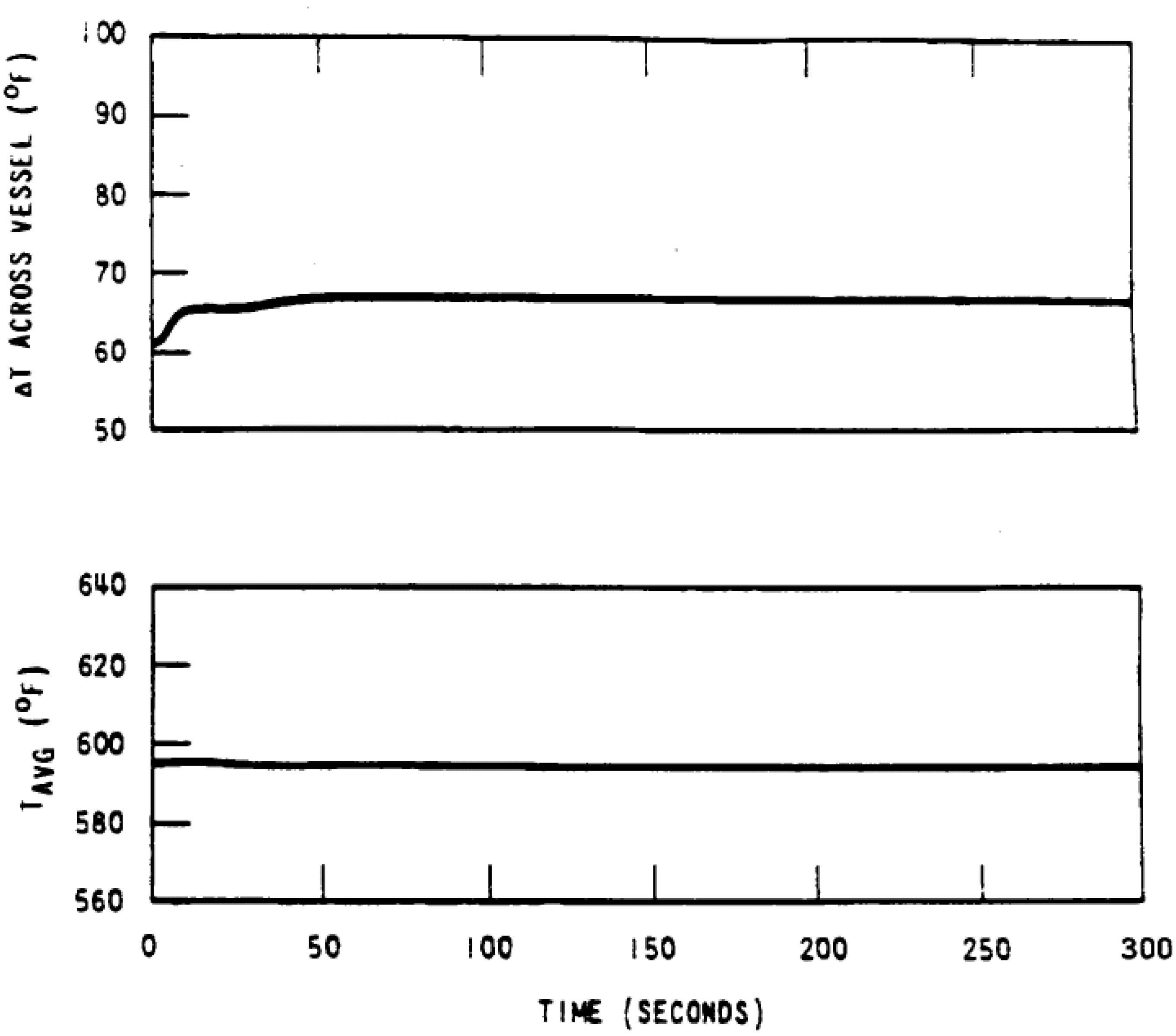

Figure 15.2-33 Typical Transient-10% Step Load Increase, Beginning of Life, Automatic Reactor Control

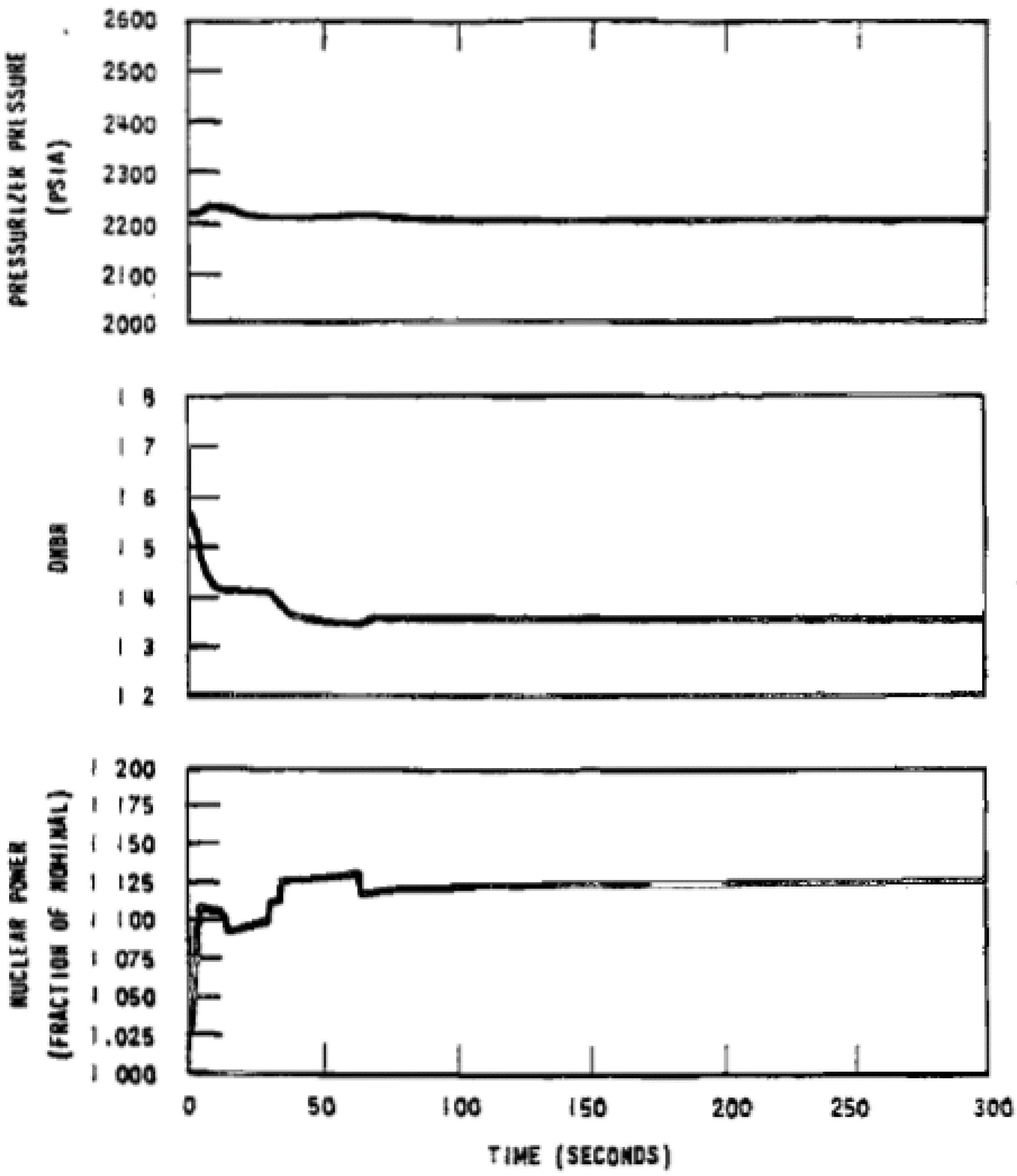

Figure 15.2-34 Typical Transient-10% Step Load Increase, Beginning of Life, Automatic Reactor Control

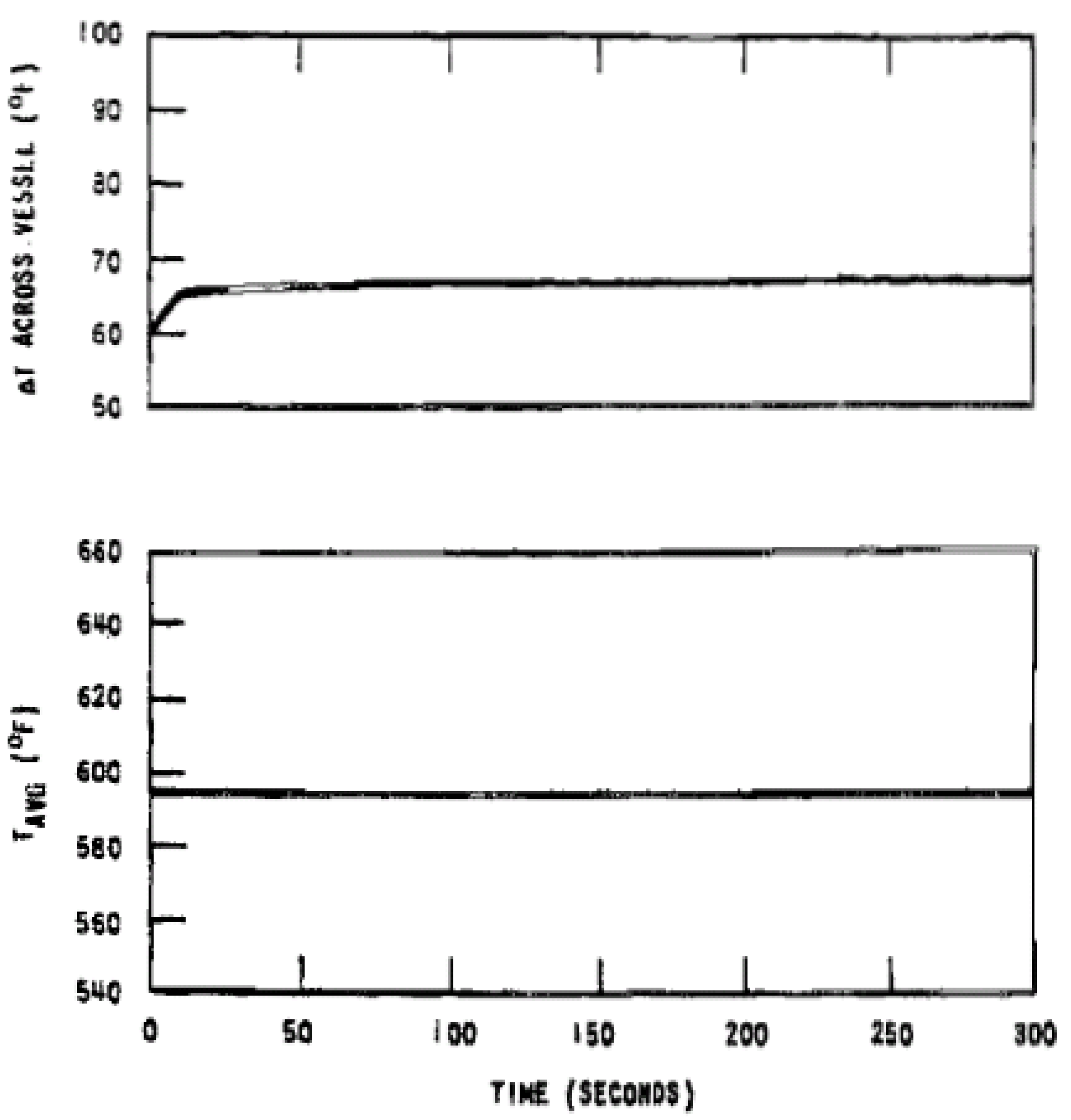

Figure 15.2-35 Typical Transient-10% Step Load Increase, End of Life, Automatic Reactor Control

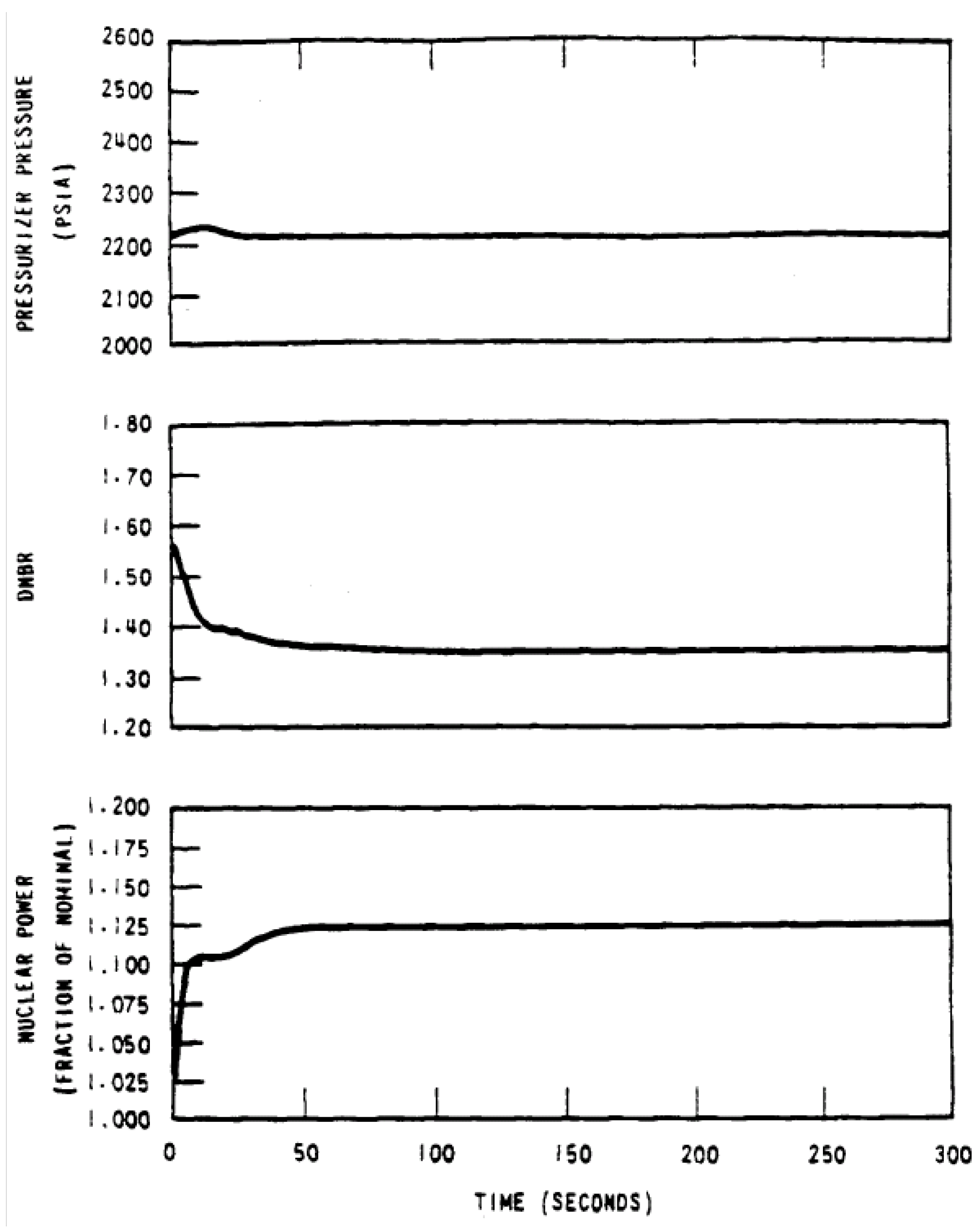

Figure 15.2-36 Typical Transient-10%t Step Load Increase, End of Life, Automatic Reactor Control

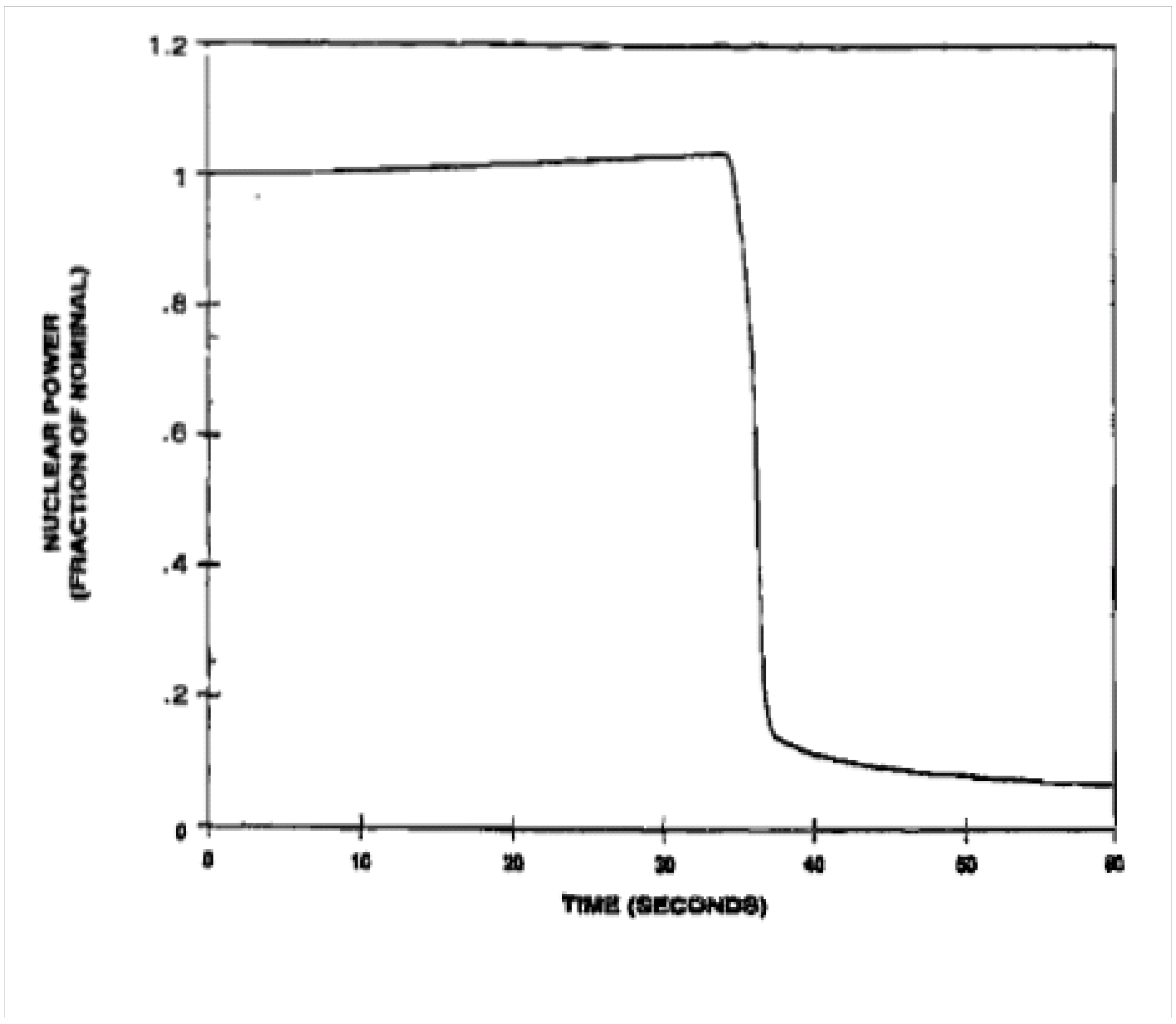

Figure 15.2-37 Accidental Depressurization of the Reactor Coolant System

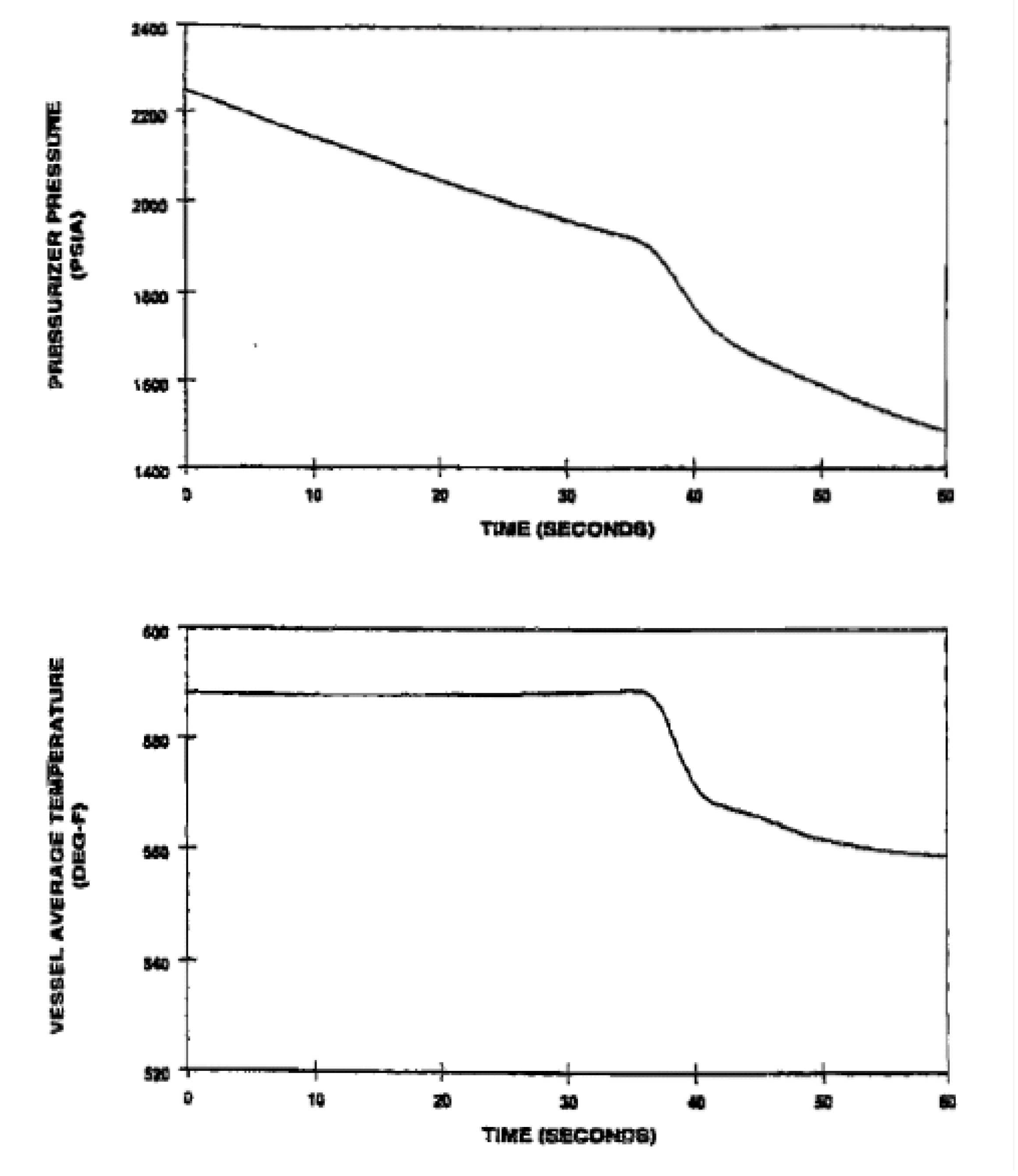

Figure 15.2-38 Accidental Depressurization of the Reactor Coolant System

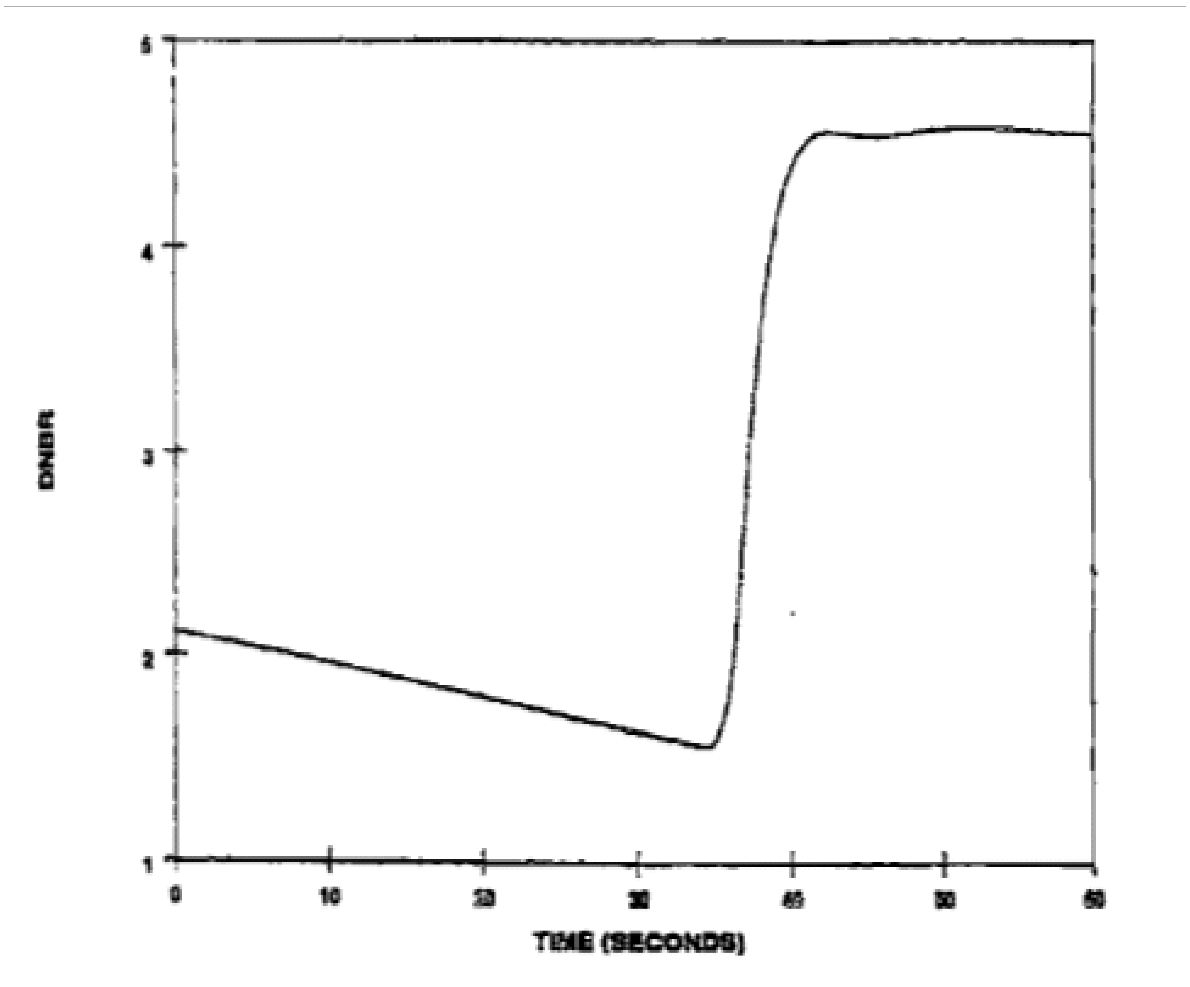

Figure 15.2-39 Accidental Depressurization of the Reactor Coolant System

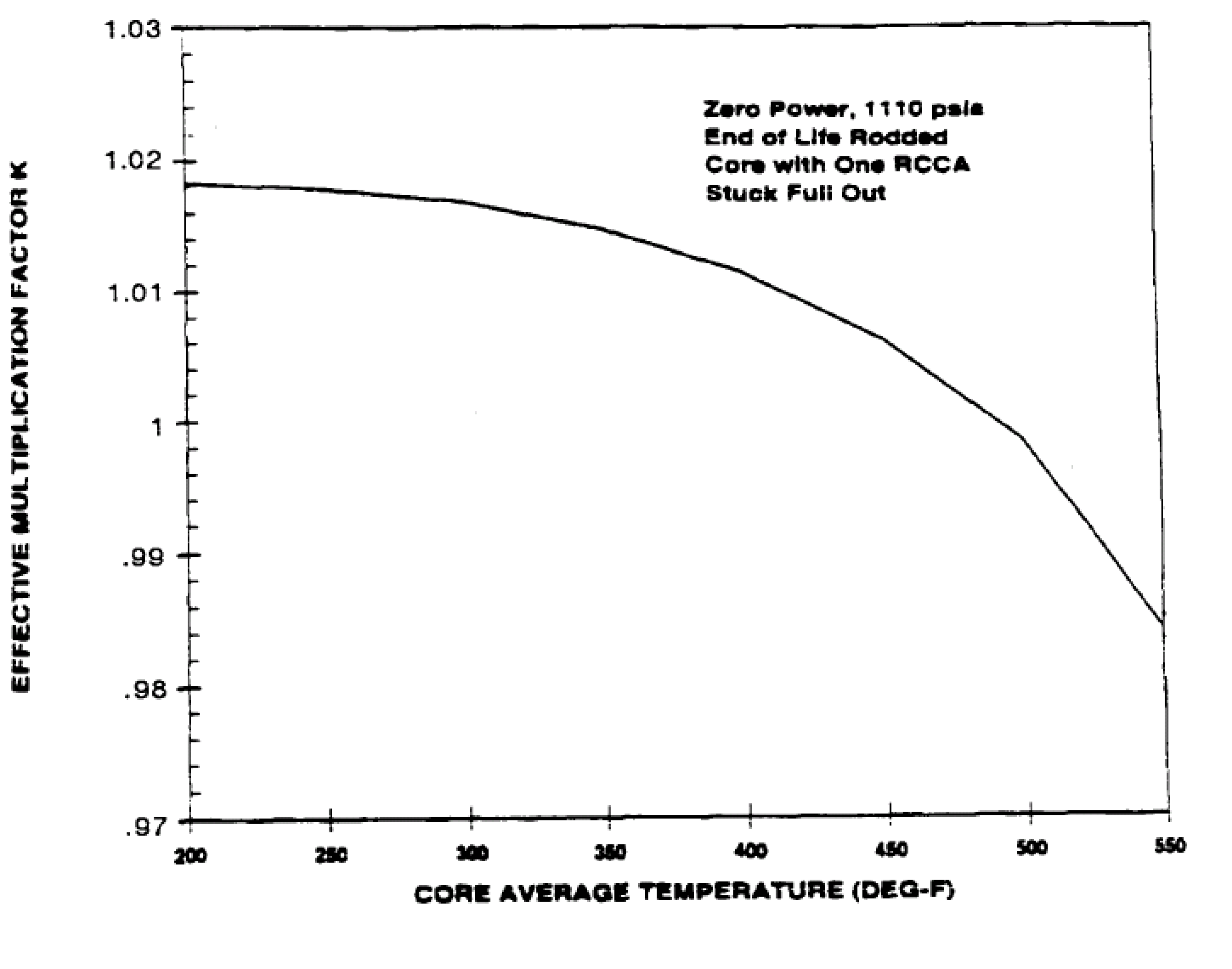

Figure 15.2-40 Variation of K_{eff} with Core Average Temperature

Figure 15.2-41 Deleted by Amendment 97

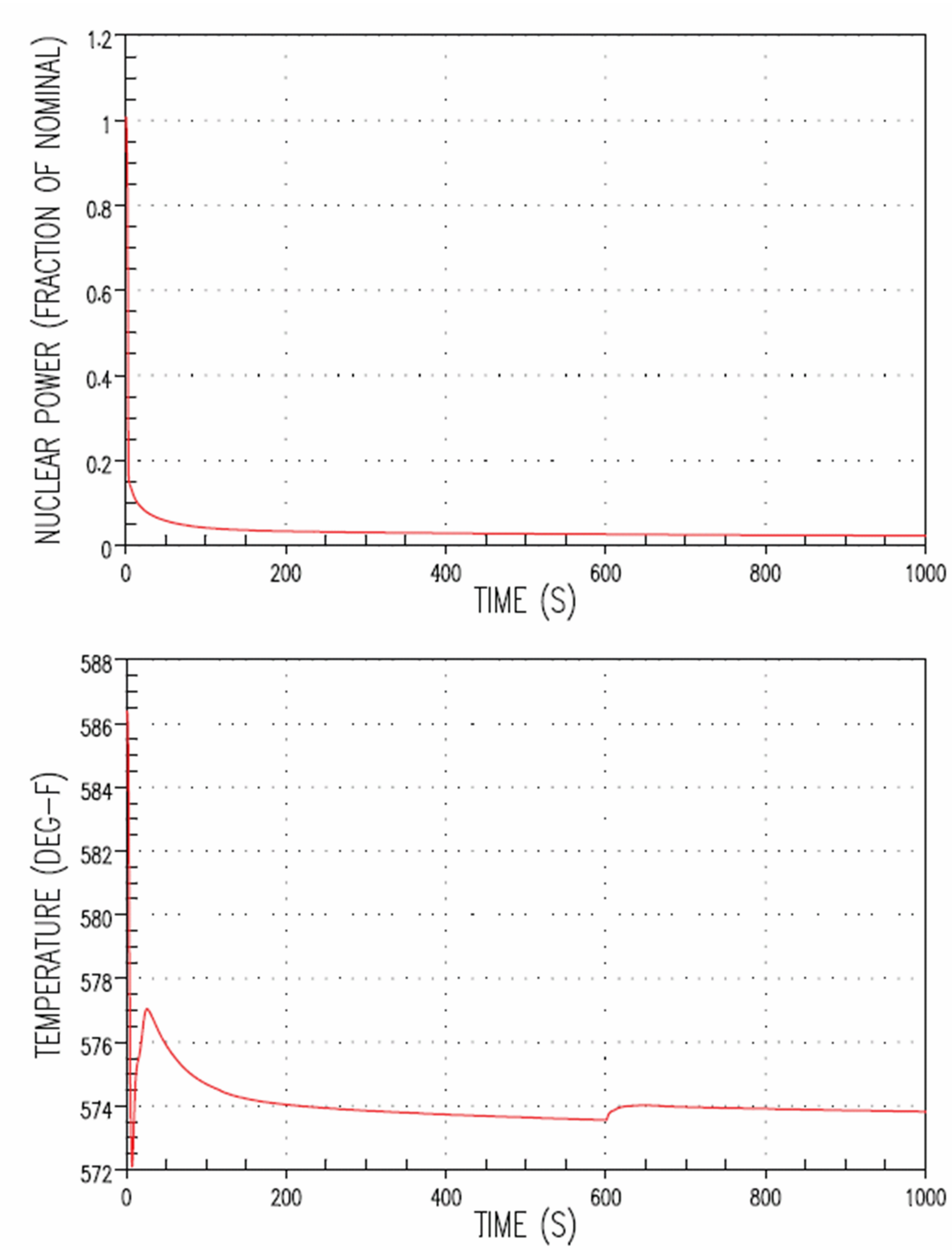

Figure 15.2-42a Inadvertent Operation of Emergency Core Cooling System - Nuclear Power and Core Average Temperature Response

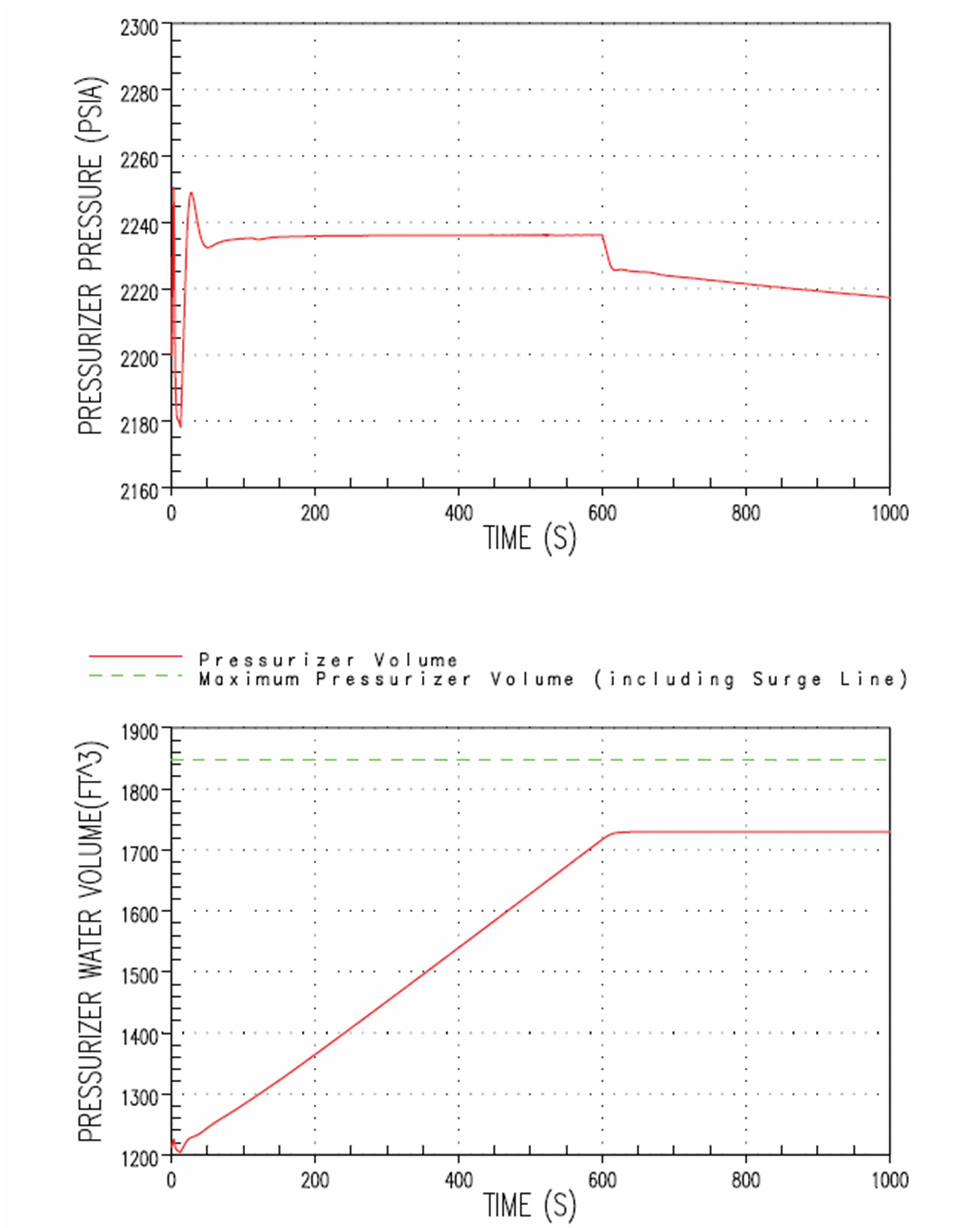

Figure 15.2-42b Inadvertent Operation of Emergency Core Cooling System - Pressurizer Pressure and Pressurizer Water Volume Response

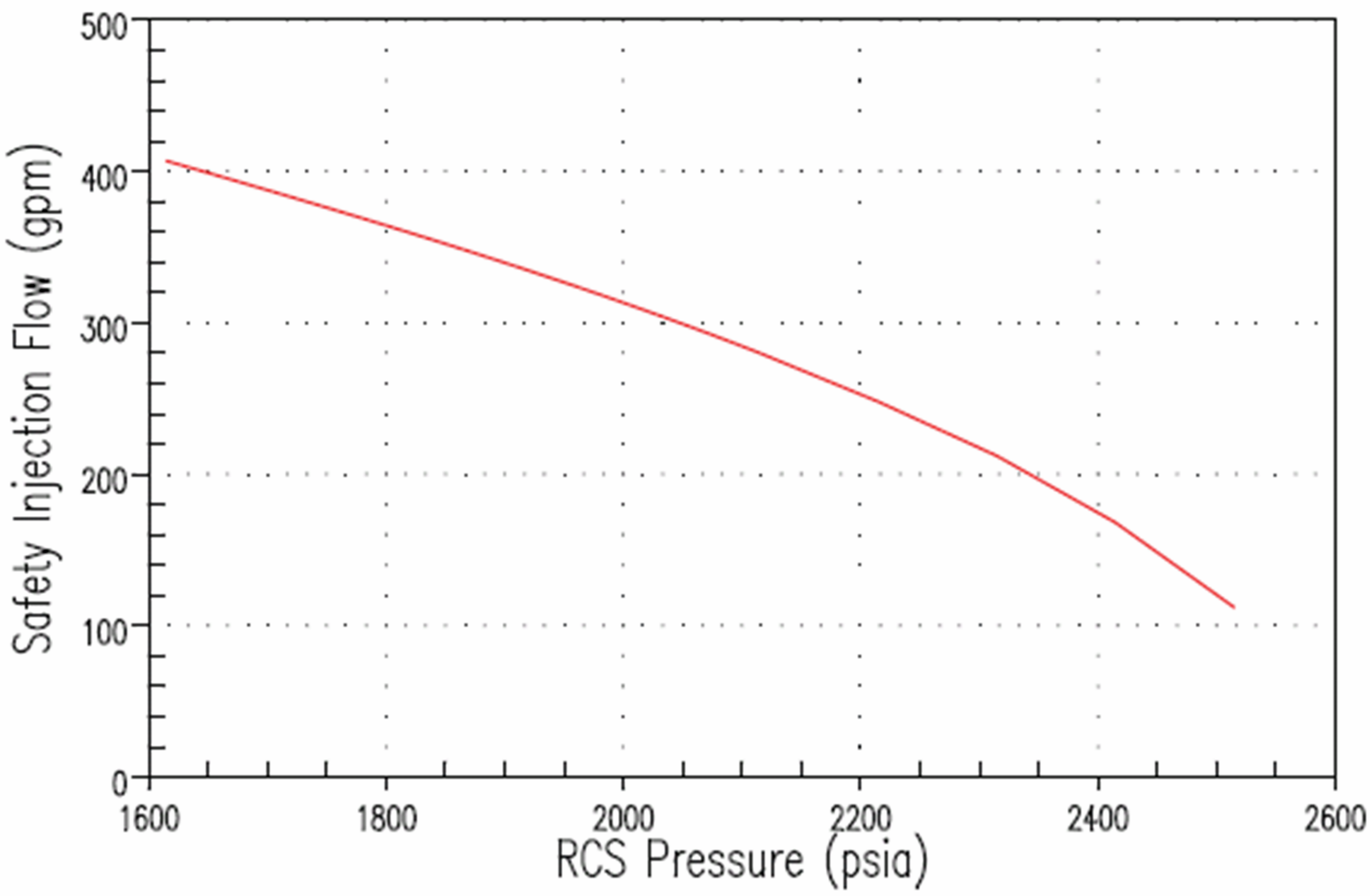

Figure 15.2-42c Inadvertent Operation of Emergency Core Cooling System - Maximum Emergency Core Cooling System Flow Rate

Figure 15.2-42d Deleted by Amendment 97

Figure 15.2-42e Deleted by Amendment 97

Figure 15.2-42f Deleted by Amendment 97

Figure 15.2-43a Deleted by Amendment 90

Figure 15.2-43b Deleted by Amendment 90

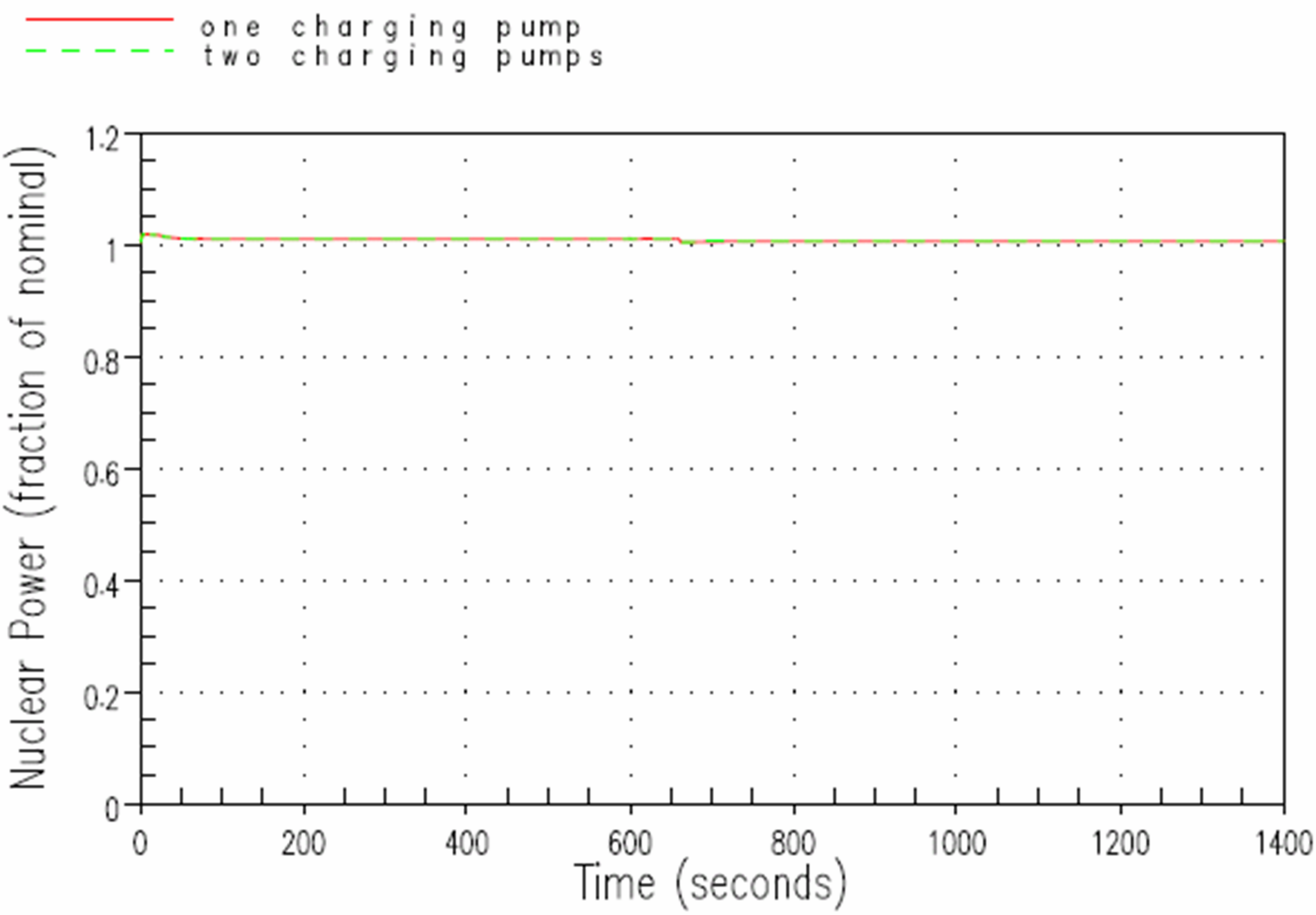

Figure 15.2-44 CVCS MALFUNCTIONNUCLEAR POWER VS. TIME

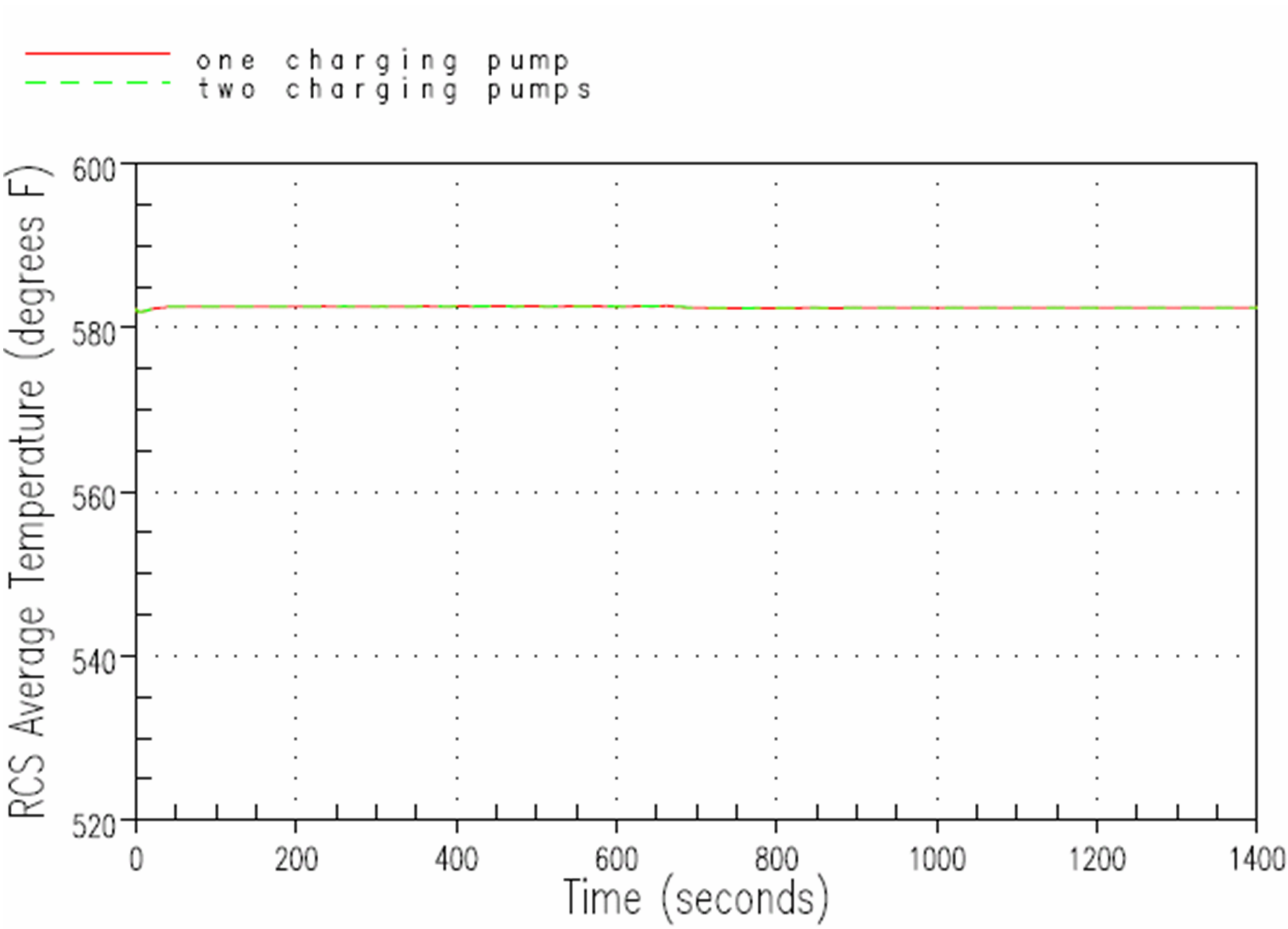

Figure 15.2-45 CVCS MALFUNCTION RCS AVERAGE TEMPERATURE VERSUS TIME

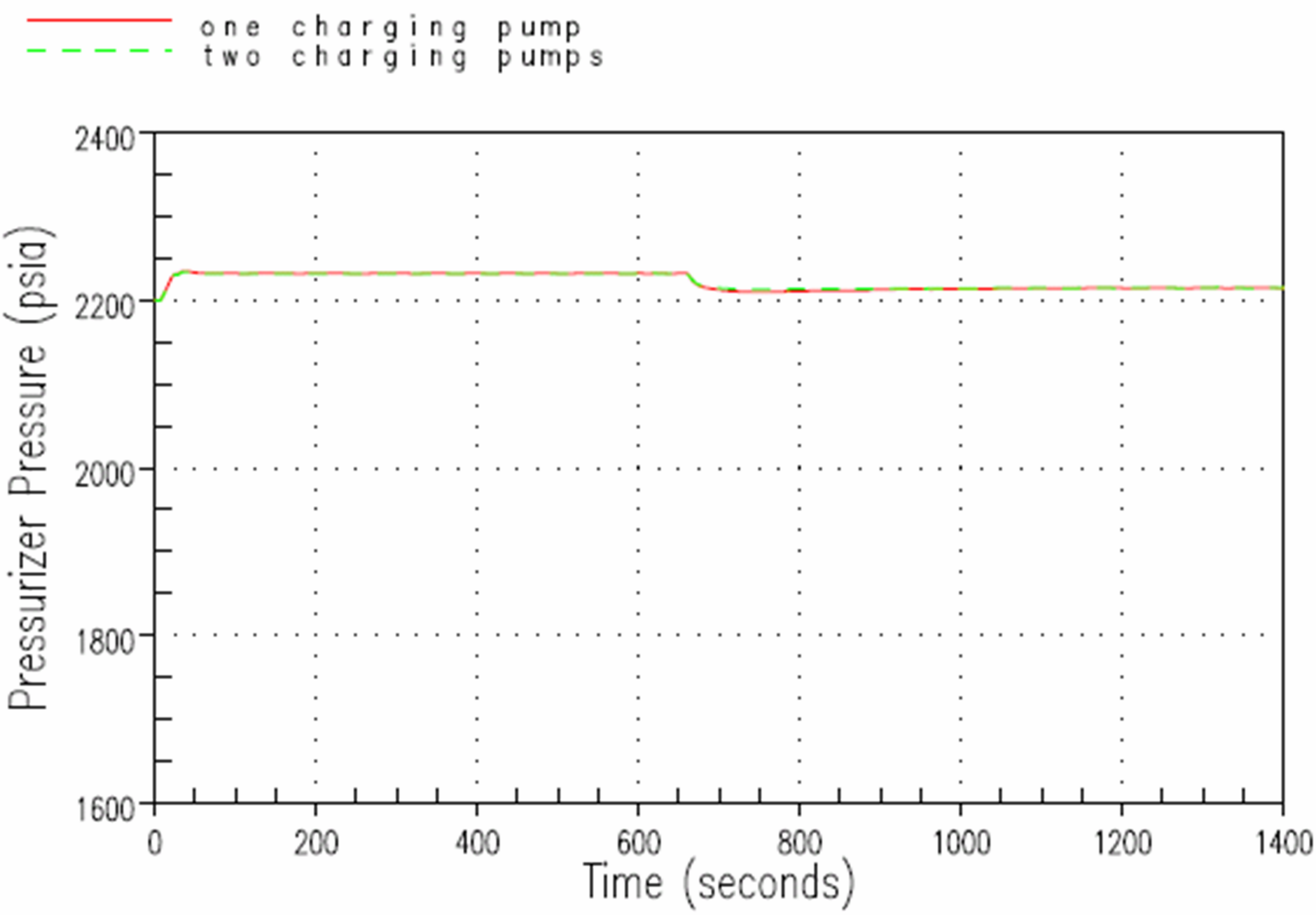

Figure 15.2-46 CVCS MALFUNCTION PRESSURIZER PRESSURE VERSUS TIME

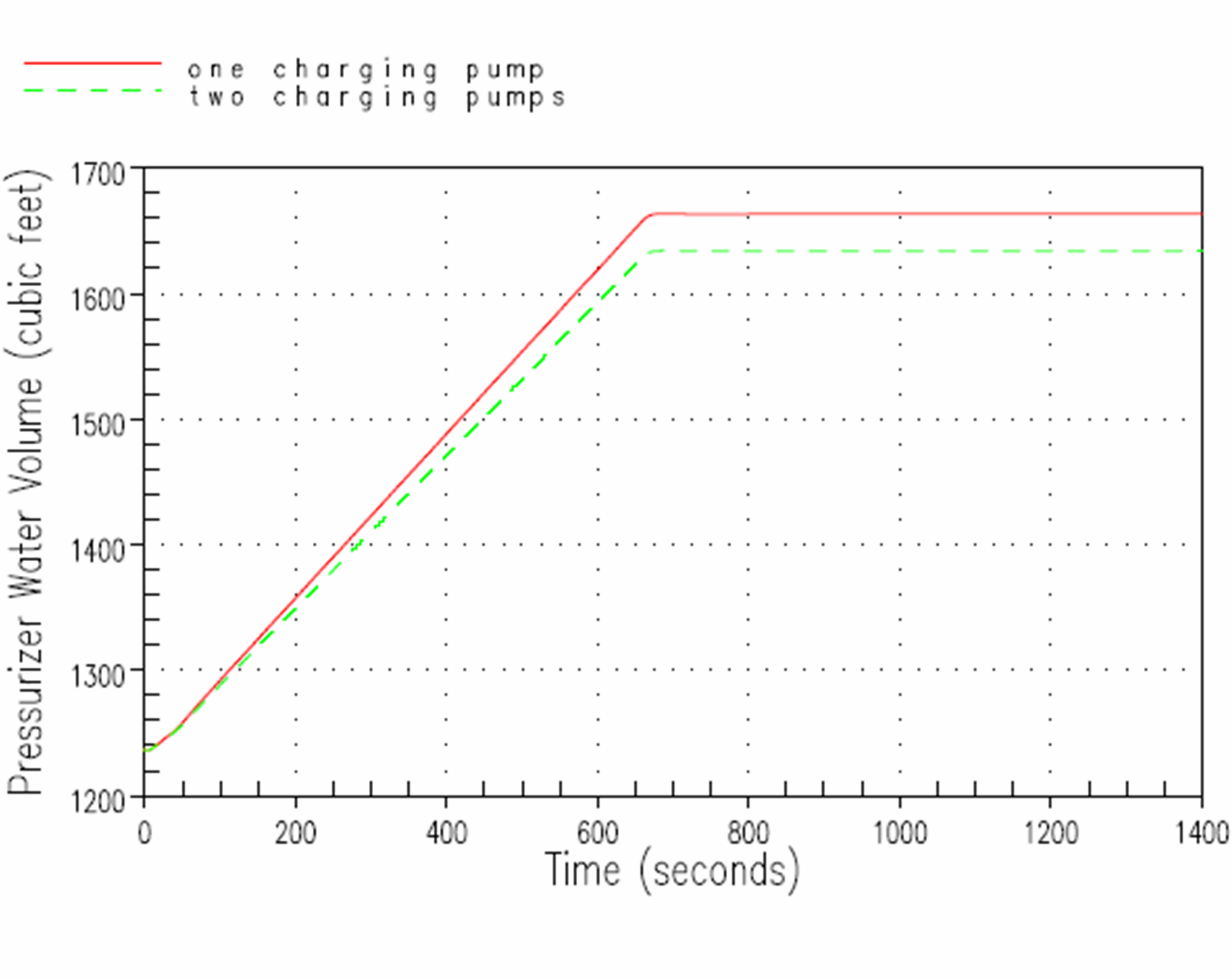

Figure 15.2-47 CVCS MALFUNCTION PRESSURIZER WATER VOLUME VERSUS TIME

15.3 CONDITION III - INFREQUENT FAULTS

By definition Condition III occurrences are faults which may occur very infrequently during the life of the plant. They will be accommodated with the failure of only a small fraction of the fuel rods although sufficient fuel damage might occur to preclude resumption of the operation for a considerable outage time. The release of radioactivity will not be sufficient to interrupt or restrict public use of those areas beyond the exclusion radius. A Condition III fault will not, by itself, generate a Condition IV fault or result in a consequential loss of function of the RCS or containment barriers. For the purposes of this report the following faults have been grouped into this category:

(1) Loss of reactor coolant, from small ruptured pipes or from cracks in large pipes, which actuates the ECCS.

(2) Minor secondary system pipe breaks.

(3) Inadvertent loading of a fuel assembly into an improper position.

(4) Complete loss of forced reactor coolant flow.

(5) Waste gas decay tank rupture.

(6) Single rod cluster control assembly withdrawal at full power.

15.3.1 Loss of Reactor Coolant From Small Ruptured Pipes or From Cracks in Large Pipes Which Actuate the Emergency Core Cooling System

15.3.1.1 Identification of Causes and Accident Description

A LOCA is defined as the loss of reactor coolant at a rate in excess of the reactor coolant normal makeup rate from breaks or openings in the RCPB inside primary containment up to, and including, a break equivalent in size to the largest justified pipe rupture (or in the absence of justification, a double-ended rupture of the largest pipe) in the reactor coolant pressure boundary (RCPB)(ANSI/ANS-51.1-1983). See Section 3.6 for a more detailed description of the loss of reactor coolant accident boundary limits. Ruptures of small cross section will cause expulsion 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. The coolant which would be released to the containment contains the existing fission products.

The maximum break size for which the normal makeup system can maintain the pressurizer level is obtained by comparing the calculated flow from the RCS through the postulated break against the charging pump makeup flow at normal RCS pressure, i.e., 2250 psia.

Should a larger break occur, depressurization of the RCS causes fluid to flow to the RCS from the pressurizer, resulting in a pressure and level decrease in the pressurizer.

A reactor trip occurs when the pressurizer low pressure trip setpoint is reached. The safety injection system is actuated when the appropriate pressure 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 clad 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. The heat transfer between the RCS 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 dump may occur. Makeup to the secondary side is automatically provided by the auxiliary feedwater pumps. The reactor trip signal coincident with low T_{avg} signal (with assumed coincident loss of offsite power), stops normal feedwater flow by closing the main feedwater isolation valves and flow control valves. The secondary flow aids in the reduction of RCS pressure.

When the RCS depressurizes to the cold leg accumulator tank pressure, the accumulators begin to inject water into the reactor coolant loops. The reactor coolant pumps are assumed to be tripped concurrent with the reactor trip, and effects of pump coastdown are included in the blowdown analyses.

15.3.1.2 Analysis of Effects and Consequences

Method of Analysis

For breaks less than 1.0 ft^2, the NOTRUMP[1,2,16] digital computer code is employed to calculate the transient depressurization of the RCS as well as to describe the mass and enthalpy of flow through the break.

Small Break LOCA Analysis Using NOTRUMP

The NOTRUMP computer code is 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 flooding limitations, mixture level tracking logic in multiple-stacked fluid nodes, and regime-dependent heat transfer correlations. 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." [Ref. 15]

In NOTRUMP, the RCS is nodalized into volumes interconnected by flowpaths. The broken loop is modeled explicitly with the intact loops lumped into a second loop. The transient behavior of the system is determined from the governing conservation equation of mass, energy, and momentum applied throughout the system. A detailed description of NOTRUMP is given in References [1], [2] and [16].

The use of NOTRUMP in the analysis involves, among other things, the representation of the reactor core as heated control volumes with an associated bubble rise model to permit a transient mixture height calculation. The multinode 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.

Cladding thermal analyses are performed with the LOCTA-IV[3] code which uses the RCS pressure, fuel rod power history, steam flow past the uncovered part of the core, and mixture height history from the NOTRUMP hydraulic calculations as input.

A schematic representation of the computer code interfaces is given in Figure 15.3-1.

Safety injection flow rate to the RCS as a function of system pressure is an input parameter. The SIS is assumed to begin delivering full flow to the RCS 27 seconds after the generation of a safety injection signal.

Also, minimum safeguards ECCS capability and operability has been assumed in these analyses including use of the COSI/safety injection in the broken loop model.

Hydraulic transient analyses are performed with the NOTRUMP code which determines the RCS pressure, fuel rod power history, steam flow past the uncovered part of the core and mixture height history. The core thermal transient is performed with the LOCTA-IV[3] code. Both calculations assume the core is operating at 102% of licensed power.

15.3.1.3 Reactor Coolant System Pipe Break Results

A spectrum of break sizes was analyzed to determine the limiting break size in terms of the highest peak cladding temperature. These break sizes were 2, 3, 4, 6, and 8.75 inches.

For all cases reported, during the earlier part of the small break transient, the effect of the break flow is not strong enough to overcome the flow maintained by the reactor coolant pumps through the core as they are coasting down following reactor trip. Therefore, upward flow through the core is maintained.

The resultant heat transfer cools the fuel rod cladding to very near the coolant temperatures as long as the core remains covered by a two-phase mixture. When the mixture level drops below the top of the core, the steam flow computed with NOTRUMP provides cooling to the upper portion of the core.

The typical core power (dimensionless) transient following the accident (relative) to reactor scram time is shown in Figure 15.3-9. Also shown is the typical hot rod axial power shape in Figure 15.3-10.

The reactor scram delay time is equal to the reactor trip signal time plus control rod insertion time, or a total of 4.7 seconds (conservatively modeled as 5.0 seconds). During this delay period, the reactor is conservatively assumed to continue to operate at the initial rated power level.

The safety injection flow vs. RCS pressure in Figure 15.3-2a is modeled for spill to RCS pressure cases (i.e., 2, 3, 4, and 6 inch break sizes). The safety injection flow vs. RCS pressure in Figure 15.3-2b is modeled for spill to containment pressure (0 psig) cases (i.e., 8.75 inch break size). Auxiliary feedwater flow is 660 gpm to four steam generators based on the operation of one motor-driven and one turbine driven auxiliary feedwater pump, each delivering to two steam generators. The flow rate is based on the conservative minimum flow of 165 gpm delivered by one motor-driven pump to one steam generator.

The 27 second delay time includes the time for diesel generator startup, loading on the 6.9 kV shutdown board, and sequential loading of the centrifugal charging and safety injection pumps onto the emergency buses, with acceleration to full speed and capability for injection.

The 4-inch break was determined to be the limiting break size, with a peak cladding temperature of 1183.9°F. The transient results for the limiting 4-inch break are presented in Figures 15.3-3 to 15.3-8. The depressurization transient for the 4-inch break is shown in Figure 15.3-3. The extent to which the core is uncovered is shown in Figure 15.3-4. The peak cladding temperature transient is shown in Figure 15.3-5. The steam flow rate for this break is shown in Figure 15.3-6. The heat transfer coefficients for the rod for this phase of the transient are given in Figure 15.3-7, and the hot spot fluid temperature is shown in Figure 15.3-8.

The comparable transient results for the 2-inch break are presented in Figures 15.3-11 to 15.3-11e, for the 3-inch break in Figures 15.3-12 to 15.3-12e, for the 6-inch break in Figures 15.3-13 to 15.3-13e, and for the 8.75-inch break in Figures 15.3-14 to 15.3–14b. Note that since there is no core uncovery for the 8.75-inch break, cladding heatup is not calculated.

An evaluation has been performed to determine the impact of change in the lower radial key stiffness value and concluded that the fuel assemblies on the core periphery are the only assemblies to experience grid deformation for Watts Bar Unit 2. An SBLOCA assessment has concluded that core coolable geometry is maintained if grid deformation remains in peripheral assembly locations. Therefore, it is further concluded that coolable core geometry is maintained for Watts Bar Unit 2 for cores of 17x17 RFA-2 fuel following a SBLOCA.

Calculated peak cladding temperatures for large breaks are presented in section 15.4.1.

15.3.1.4 Conclusions - Thermal Analysis

For cases considered, the emergency core cooling system meets the acceptance criteria as presented in 10 CFR 50.46. That is:

(1) The calculated peak fuel element cladding temperature provides margin to the limit of 2200°F, based on an F_q value of 2.50.

(2) The amount of fuel element cladding that reacts chemically with water or steam does not exceed 1% of the total amount of zircaloy in the reactor.

(3) The cladding temperature transient is terminated at a time when the core geometry is still amenable to cooling. The oxidation limit of 17% of the cladding thickness is not exceeded during or after quenching.

(4) The core temperature is reduced and decay heat is removed for an extended period of time, as required by the long-lived radioactivity remaining in the core.

The time sequence of events is shown in Table 15.3-1. Table 15.3-2 summarizes the results of these analyses.

15.3.2 Minor Secondary System Pipe Breaks

15.3.2.1 Identification of Causes and Accident Description

Included in this grouping are ruptures of secondary system lines which would result in steam release rates equivalent to a 6 inch diameter break or smaller.

15.3.2.2 Analysis of Effects and Consequences

Minor secondary system pipe breaks must be accommodated with the failure of only a small fraction of the fuel elements in the reactor. Since the results of analysis presented in Section 15.4.2 for a major secondary system pipe rupture also meet this criteria, separate analysis for minor secondary system pipe breaks is not required.

The evaluation of the more probable accidental opening of a secondary system steam dump, relief or safety valve is presented in Section 15.2.13. These analyses are illustrative of a pipe break equivalent in size to a single valve opening. These smaller equivalent pipe break sizes are also bounded by the analysis presented in Section 15.4.2 for the MSLB event.

15.3.2.3 Conclusions

The analyses presented in Section 15.4.2 demonstrate that the consequences of a minor secondary system pipe break are acceptable since a DNBR of less than the limiting value does not occur even for a more critical major secondary system pipe break.

15.3.3 Inadvertent Loading of a Fuel Assembly Into an Improper Position

15.3.3.1 Identification of Causes and Accident Description

Fuel and core loading errors such as can arise from the inadvertent loading of one or more fuel assemblies into improper positions, loading a fuel rod during manufacture with one or more pellets of the wrong enrichment or the loading of a full fuel assembly during manufacture with pellets of the wrong enrichment will lead to increased heat fluxes if the error results in placing fuel in core positions calling for fuel of lesser enrichment. Also included among possible core loading errors is the inadvertent loading of one or more fuel assemblies requiring burnable poison rods into a new core without burnable poison rods.

Any error in enrichment, beyond the normal manufacturing tolerances, can cause power shapes which are more peaked than those calculated with the correct enrichments. There is a 5% uncertainty margin included in the design value of power peaking factor assumed in the analysis of Condition I and Condition II transients. The Power Distribution Monitoring System [17] is capable of revealing any assembly enrichment error or loading error which causes power shapes to be peaked in excess of the design value.

To reduce the probability of core loading errors, each fuel assembly is marked with an identification number and loaded in accordance with a core loading diagram. During core loading the identification number is checked before each assembly is moved into the core. Serial numbers read during fuel movement are subsequently recorded on the loading diagram as a further check on proper placing after the loading is completed.

In addition to the Power Distribution Monitoring System, thermocouples are located at the outlet of about one third of the fuel assemblies in the core. There is a high probability that these thermocouples would also indicate any abnormally high coolant enthalpy rise.

15.3.3.2 Analysis of Effects and Consequences

Method Of Analysis

Steady-state power distributions in the x-y plane of the core are calculated by the TURTLE[6] Code based on macroscopic cross section calculated by the LEOPARD[7] Code. A discrete representation is used wherein each individual fuel rod is described by a mesh interval. The power distributions in the x-y plane for a correctly loaded core assembly are also given in Chapter 4 based on enrichments given in that section.

For each core loading error case analyzed, the percent deviations from detector readings for a normally loaded core are shown at all incore detector locations (see Figures 15.3-15 to 15.3-19, inclusive).

Results

The following core loading error cases have been analyzed.

Case A:

Case in which a Region 1 assembly is interchanged with a Region 3 assembly. The particular case considered was the interchange of two adjacent assemblies near the periphery of the core (see Figure 15.3-15).

Case B:

Case in which a Region 1 assembly is interchanged with a neighboring Region 2 fuel assembly. Two analyses have been performed for this case (see Figures 15.3-16 and 15.3-17).

In Case B-1, the interchange is assumed to take place with the burnable poison rods transferred with the Region 2 assembly mistakenly loaded into Region 1.

In Case B-2, the interchange is assumed to take place closer to core center and with burnable poison rods located in the correct Region 2 position but in a Region 1 assembly mistakenly loaded into the Region 2 position.

Case C:

Enrichment error: Case in which a Region 2 fuel assembly is loaded in the core central position (see Figure 15.3-18).

Case D:

Case in which a Region 2 fuel assembly instead of a Region 1 assembly is loaded near the core periphery (see Figure 15.3-19).

15.3.3.3 Conclusions

Fuel assembly enrichment errors would be prevented by administrative procedures implemented in fabrication.

In the event that a single pin or pellet has a higher enrichment than the nominal value, the consequences in terms of reduced DNBR and increased fuel and clad temperatures will be limited to the incorrectly loaded pin or pins.

Fuel assembly loading errors are prevented by administrative procedures implemented during core loading. In the unlikely event that a loading error occurs, analyses in this section confirm that resulting power distribution effects will either be readily detected by the Power Distribution Monitoring System or will cause a sufficiently small perturbation to be acceptable within the uncertainties allowed between nominal and design power shapes.

15.3.4 Complete Loss of Forced Reactor Coolant Flow

15.3.4.1 Identification of Causes and Accident Description

A complete loss of forced reactor coolant flow may result from a simultaneous loss of electrical supplies to all reactor coolant pumps (RCPs). If the reactor is at power at the time of the accident, the immediate effect of loss of forced reactor coolant flow is a rapid increase in the reactor coolant temperature and subsequent increase in reactor coolant pressure. The flow reduction and increase in coolant temperature could eventually result in DNB and subsequent fuel damage before the peak pressures exceed the values at which the integrity of the pressure boundaries would be jeopardized unless the reactor was tripped promptly.

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

The following reactor trips provide the necessary protection against a loss of coolant flow accident:

(1) Reactor coolant pump power supply undervoltage or underfrequency.

(2) Low reactor coolant loop flow.

The reactor trip on reactor coolant pump undervoltage is provided to protect against conditions which can cause a loss of voltage to all reactor coolant pumps, i.e., loss of power supply to all reactor coolant pumps. This function is blocked below the approximately 10% power (Permissive 7) interlock setpoint to permit startup.

The reactor trip on reactor coolant pump underfrequency is provided to trip the reactor for an underfrequency condition, resulting from frequency disturbances on the power grid. This function is also blocked below the approximately 10% power (Permissive 7) interlock setpoint to permit startup.

Reference [8] provides analyses of grid frequency disturbances and the resulting Nuclear Steam Supply System protection requirements which are applicable to current generation Westinghouse plants.

These analyses have shown that the reactor is adequately protected by the underfrequency reactor trip such that DNB will be above the limiting value for grid frequency decay rates less than 6.8 Hz/sec based on a trip setpoint of approximately 57 Hz. In addition, for a maximum frequency decay rate of 5 Hz/sec, the selected trip setpoint would have to be at least 54.3 Hz. The sensing relay connected to the load side of each RCP breaker for WBN is set at approximately 57 Hz. A grid analysis has been provided which determined that for the worst case the maximum system frequency decay rate is less than 5 Hz/sec.

The reactor trip on low primary coolant loop flow is provided to protect against loss of flow conditions which affect only one reactor coolant loop. This function is generated by two out of three low flow signals per reactor coolant loop. Above approximately 48% power (Permissive 8), low flow in any loop will actuate a reactor trip. Between approximately 10% power and 48% power (Permissive 7 and Permissive 8), low flow in any two loops will actuate a reactor trip.

The effect of low loop flow trip protection alone relative to frequency decay rate, although not the primary trip function taken credit for in WBN's design, is also addressed in Reference [8].

15.3.4.2 Analysis of Effects and Consequences

Method of Analysis

This transient is analyzed by three digital computer codes. The LOFTRAN[9] Code is used to calculate the loop flow, core flow, the time of reactor trip, the nuclear power transient, and the primary system pressure and coolant temperature transients. The FACTRAN[10] Code is then used to calculate the heat flux transient based on the nuclear power and flow from LOFTRAN. Finally, the VIPRE-01[13,14] Code (see Section 4.4.3.4) is used to calculate the DNBR during the transient based on the heat flux from FACTRAN and flow from LOFTRAN. The DNBR transients presented represent the minimum of the typical or thimble cell.

The method of analysis and the assumptions made regarding initial operating conditions and reactivity coefficients are identical to those discussed in Section 15.2, except that following the loss of supply to all pumps at power, a reactor trip is actuated by either reactor coolant pump power supply undervoltage or underfrequency.

Results

The calculated sequence of events for the case analyzed is shown on Table 15.3-3. The reactor is assumed to trip on an undervoltage signal. Figures 15.3-20 and 15.3-23 through 15.3-25 show the transient response for the loss of power to all reactor coolant pumps. The DNBR never goes below the design basis limit.

The most limiting statepoint occurred for the complete loss of flow under-frequency case for the DNB transient. The DNB evaluation showed that the minimum DNBR remained above the limiting value. An axial power shape that bounds the cycle specific conditions is used to perform the statepoint evaluation of the complete loss of flow analysis (also partial loss of flow analysis as presented in Section 15.2.5).

Following reactor trip, the pumps will continue to coast down until natural circulation flow is established and will approach a stabilized hot standby condition as shown in Section 15.2.8. The operating procedures call for operator action to control RCS boron concentration and pressurizer level using the CVCS, and to maintain steam generator level through control of the main or auxiliary feedwater system. Any action required of the operator to maintain the plant in a stabilized condition is in a time frame in excess of ten minutes following reactor trip.

15.3.4.3 Conclusions

The analysis performed has demonstrated that for the complete loss of forced reactor coolant flow, the DNBR will not decrease below the design basis limit at any time during the transient.

15.3.5 Waste Gas Decay Tank Rupture

15.3.5.1 Identification of Causes and Accident Description

The gaseous waste processing system, as discussed in Section 11.3, is designed to remove fission product gases from the reactor coolant. The system consists of a closed loop with waste gas compressors, waste gas decay tanks for service at power and other waste gas decay tanks for service at shutdown and startup.

The maximum amount of waste gases stored occurs after a refueling shutdown at which time the gas decay tanks store the radioactive gases stripped from the reactor coolant.

The accident is defined as an unexpected and uncontrolled release of radioactive xenon and krypton fission product gases stored in a waste decay tank as a consequence of a failure of a single gas decay tank or associated piping.

15.3.5.2 Analysis of Effects and Consequences

For the analyses and consequences of the postulated waste gas decay tank rupture, please refer to Section 15.5.2.

15.3.6 Single Rod Cluster Control Assembly Withdrawal at Full Power

15.3.6.1 Identification of Causes and Accident Description

The current WBN design basis for the single rod cluster control assembly (RCCA) withdrawal at full power event assumes no single electrical or mechanical failure in the rod control system could cause the accidental withdrawal of a single RCCA from the inserted bank at full power operation. The operator could deliberately withdraw a single RCCA in the control bank since this feature is necessary in order to retrieve an assembly should one be accidentally dropped. In the extremely unlikely event of simultaneous electrical failures which could result in single RCCA withdrawal, rod deviation and rod control urgent failure would both be displayed on the plant annunciator, and the rod position indicators would indicate the relative positions in the assemblies in the bank. The urgent failure alarm also inhibits automatic rod withdrawal. Withdrawal of a single RCCA by operator action would result in activation of the same alarm and the same visual indications.

Each bank of RCCAs in the system is divided into two groups of 4 mechanisms each (except group 2 of bank D which consists of 5 mechanisms). The rods comprising a group operate in parallel through multiplexing thyristors. The two groups in a bank move sequentially such that the first group is always within one step of the second group in the bank. A definite sequence of actuation of the stationary gripper, movable gripper, and lift coils of a mechanism is required to withdraw the RCCA attached to the mechanism. Since the stationary gripper, movable gripper, and lift coils associated with the RCCAs of a rod group are driven in parallel, any single failure which would cause rod withdrawal would affect a minimum of one group. Mechanical failures are in the direction of insertion, or immobility.

In the unlikely event of multiple failures which result in continuous withdrawal of a single RCCA, it is not possible, in all cases, to provide assurance of automatic reactor trip such that DNB safety limits are not violated. Withdrawal of a single RCCA results in both positive reactivity insertion tending to increase core power, and an increase in local power density in the core area associated with the RCCA.

15.3.6.2 Analysis of Effects and Consequences

Method of Analysis

Power distributions within the core are calculated by the TURTLE[6] Code based on macroscopic cross sections generated by LEOPARD[7]. The peaking factors calculated by TURTLE are then used by THINC[11] to calculate the minimum DNBR for the event. The case of the worst rod withdrawn from bank D inserted at the insertion limit, with the reactor initially at full power, was analyzed. This incident is assumed to occur at beginning-of-life since this results in the minimum value of moderator temperature coefficient. This maximizes the power rise and minimizes the tendency of increased moderator temperature to flatten the power distribution.

Results

Two cases have been considered as follows:

(1) If the reactor is in the manual control mode, continuous withdrawal of a single RCCA results in both an increase in core power and coolant temperature, and an increase in the local hot channel factor in the area of the failed RCCA. In terms of the overall system response, this case is similar to those presented in Section 15.2.2; however, the increased local power peaking in the area of the withdrawn RCCA results in lower minimum DNBRs than for the withdrawn bank cases. Depending on initial bank insertion and location of the withdrawn RCCA, automatic reactor trip may not occur sufficiently fast to prevent the minimum core DNB ratio from falling below the limiting value. Evaluation of this case at the power and coolant conditions at which the overtemperature ΔT trip would be expected to trip the plant shows that an upper limit for the number of rods with a DNBR less than the limiting value is 5%.

(2) If the reactor is in automatic control mode, the multiple failures that result in the withdrawal of a single RCCA will result in the immobility of the other RCCAs in the controlling bank. The transient will then proceed in the same manner as Case 1 described above. For such cases as above, a trip will ultimately ensue, although not sufficiently fast in all cases to prevent the minimum DNBR in the core from decreasing below the limiting value.

Following reactor trip, the plant will approach a stabilized condition at hot standby; normal plant operating procedures may then be followed. The operating procedures would call for operator action to control RCS boron concentration and pressurizer level using the CVCS, and to maintain steam generator level through control of the main or auxiliary feedwater system. Any action required of the operator to maintain the plant in a stabilized condition will be in a time frame in excess of ten minutes following reactor trip.

15.3.6.3 Conclusions

For the case of one RCCA fully withdrawn, with the reactor in the automatic or manual control mode and initially operating at full power with bank D at the insertion limit, an upper bound of the number of fuel rods experiencing DNBR at values less than the limiting value is 5% of the total fuel rods in the core.

For both cases discussed, the indicators and alarms mentioned would function to alert the operator to the malfunction. For case 1, the insertion limit alarms (low and low-low alarms) would also serve to alert the operator.

It is to be additionally noted that the current analysis methodology for the bank withdrawal at power uses point-kinetics and one-dimensional kinetics transient models, respectively. These models use conservative constant reactivity feedback assumptions which result in an overly conservative prediction of the core response for these events.

The accidental withdrawal of a bank or banks of RCCAs in the normal overlap mode is a transient which has been specifically considered in the safety analysis. The consequences of a bank withdrawal accident meet Condition II criteria (no DNB). If, however, it is assumed that less than a full group or bank of control rods is withdrawn, and these rods are not symmetrically located around the core, this then can cause a "tilt" in the core radial power distribution. The "tilt" could result in a radial power distribution peaking factor which is more severe than is normally considered in the safety analysis, and therefore cause a loss of DNB margin.

A more detailed DNBR analysis addressing the limiting transient setpoints has been conducted (References 11 and 12) and the Revised Thermal Design Procedure (RTDP) maximizes DNBR margins and determines setpoints that are conservatively low when compared to previous results.

Using these approaches, generic analyses and their plant-specific application demonstrate that for WBN DNB does not occur for the worst-case asymmetric rod withdrawal, and the licensing basis for the facility with regard to the requirements for system response to a single failure in the rod control system (GDC-25 or equivalent) is still satisfied.

REFERENCES

(1) Lee, N., et al., "Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code," WCAP-10054-P-A and WCAP-10081-A (Non-Proprietary), August 1985.

(2) Meyer, P. E., et al., "NOTRUMP: A Nodal Transient Small Break and General Network Code," WCAP-10079-P-A and WCAP-10080-A (Non-Proprietary), August 1985.

(3) F. M. Bordelon, et. al., "LOCTA-IV Program: Loss-of-Coolant Transient Analysis," WCAP-8305 (Non-Proprietary) and WCAP-8301 (Proprietary), June 1974.

(4) Deleted by Amendment 63.

(5) Deleted by Amendment 63.

(6) Barry, R. F. and Altomare, S. "The TURTLE 24.0 Diffussion Depletion Code," WCAP-7213-P-A (Proprietary) and WCAP-7758-A (Non-Proprietary), January 1975.

(7) Barry, F. R., "LEOPARD, A Spectrum Dependent Non-Spatial Depletion Code for the IBM-7094," WCAP-3269-26, September 1963.

(8) Balwin, M. S., Merrian, M. M., Schenkel, H. S., and Vandewalle, D. J., "An Evaluation of Loss of Flow Accidents Caused by Power System Frequency Transients in Westinghouse PWRs," WCAP-8424, Revision 1, June 1975.

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

(10) Hargrove, H. G., "FACTRAN, A FORTRAN IV Code for Thermal Transients in a UO2 Fuel Rod," WCAP-7908-A, December 1989.

(11) Friedland, A. J., and Ray, S., "Improved THINC IV Modeling for PWR Core Design," WCAP-12330-P, August 1989.

(12) Huegel, D., et al., "Generic Assessment of Asymmetric Rod Cluster Control Assembly Withdrawal," WCAP-13803, August 1993.

(13) C. W. Stewart, et al., “VIPRE-01; A Thermal-Hydraulic Code for Reactor Cores,” Volume 1-3 (Revision 3, August 1989), Volume 4 (April 1987), NP-2511-CCM-A, EPRI.

(14) WCAP-14565-P-A, “VIPRE-01 Modeling and Qualification for Pressurized Water Reactor Non-LOCA Thermal-Hydraulic Safety Analysis,” October 1999.

(15) "Generic Evaluation of Feedwater Transients and Small Break Loss-of-Coolant Accidents in Westinghouse-Designed Operating Plants," NUREG-0611, January 1980.

(16) 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.

(17) Beard, C.L. and Morita, T. "BEACON: Core Monitoring and Operations Support System", WCAP-12472-P-A (Proprietary), August 1994, Addendum 1-A, January 2000, Addendum 2-A, April 2002 and WCAP-12473-A (Non-proprietary), August 1994.

Table 15.3-1 Small Break Loca Analysis Time Sequence Of Events

Break Size:	2 inch	3 inch	4 inch	6 inch	8.75 inch
Break initiation [sec]	0.0	0.0	0.0	0.0	0.0
Reactor trip signal [sec]	143.3	52.1	26.8	13.4	7.7
Safety injection signal [sec]	143.3	52.1	26.8	13.4	7.7
Top of core uncovered	3688	901	629	401	N/A
Accumulator Injection Begins [sec]	N/A	2698	858	366	169
Peak cladding temperature occurs [sec]	4910.4	1409.2	976.6	468.4	N/A
Top of core recovered [sec]	5572	2540	1918	483	N/A

*Note: There is no core uncovery for the 8.75 inch break.

Table 15.3-2 Small Break LOCA Fuel Cladding Results

Break Size:	2-inch	3-inch	4-inch	6-inch	*8.75 inch
Peak cladding temperature (PCT) (°F)	1009.5	1043.2	1183.9	747.8	N/A
Location of PCT [ft.]	11.25	11.25	11.25	10.75	N/A
PCT Time [sec]	4910.4	1409.2	976.6	468.4	N/A
Maximum Local Zr-H_2O Reaction (%)	0.02	0.03	0.06	0.00	N/A
Maximum Local Zr-H_2O Reaction Location (ft)	11.25	11.25	11.25	11.00	N/A
Total Zr-H_2O Reaction (%)	<1	<1	<1	<1	N/A
Hot rod burst time (sec)	N/A	N/A	N/A	N/A	N/A
Hot rod burst location [ft.]	N/A	N/A	N/A	N/A	N/A

* Note: Three is no core uncovery for the 8.75-inch break.

Boundary Condition Assumptions

NSSS power	Equivalent to 102% of 3427 MWt
Core power (rod heatup analysis)	Equivalent to 102% of 3411 MWt
Peak linear power	13.89 kW/ft
Cold leg accumulators:	
Water volume (each)	1050 ft3
Pressure	600 psia

Table 15.3-3 Time Sequence Of Events For Condition III Events

Accident	Event	Time (seconds)
Complete Loss of Forced Reactor Coolant Flow		
Undervoltage		
1. All pumps in operation, all pumps coasting down	All operating pumps lose power (due to undervoltage event) and begin coasting down	0
	Rods begin to drop	1.5
	Minimum DNBR occurs	3.7
Underfrequency		
2. All pumps in operation, all pumps decelerating	All operating pumps lose power (due to underfrequency event) and begin coasting down	0
	Rods begin to drop	1.24
	Minimum DNBR occurs	3.6

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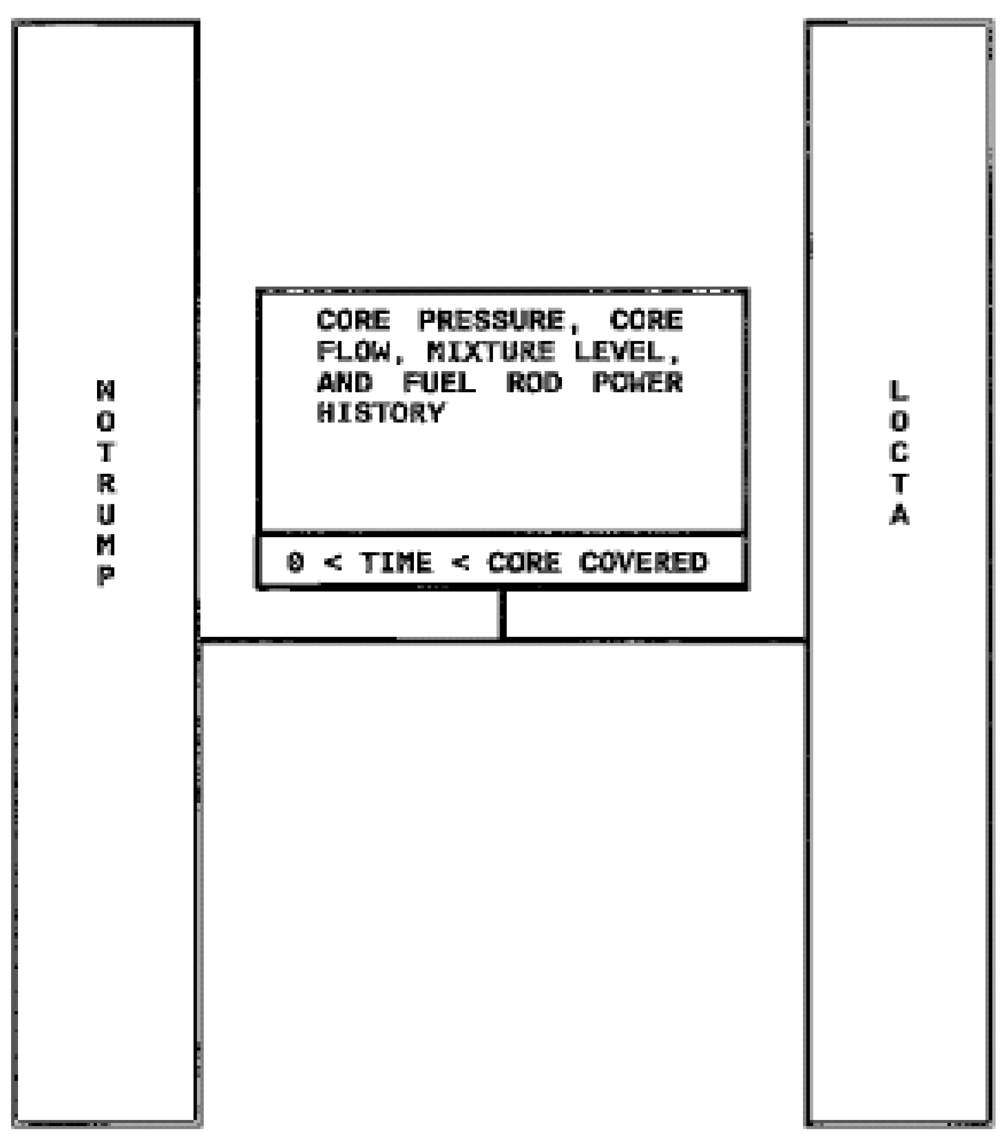


Figure 15.3-1 Code Interface Description for Small Break Model

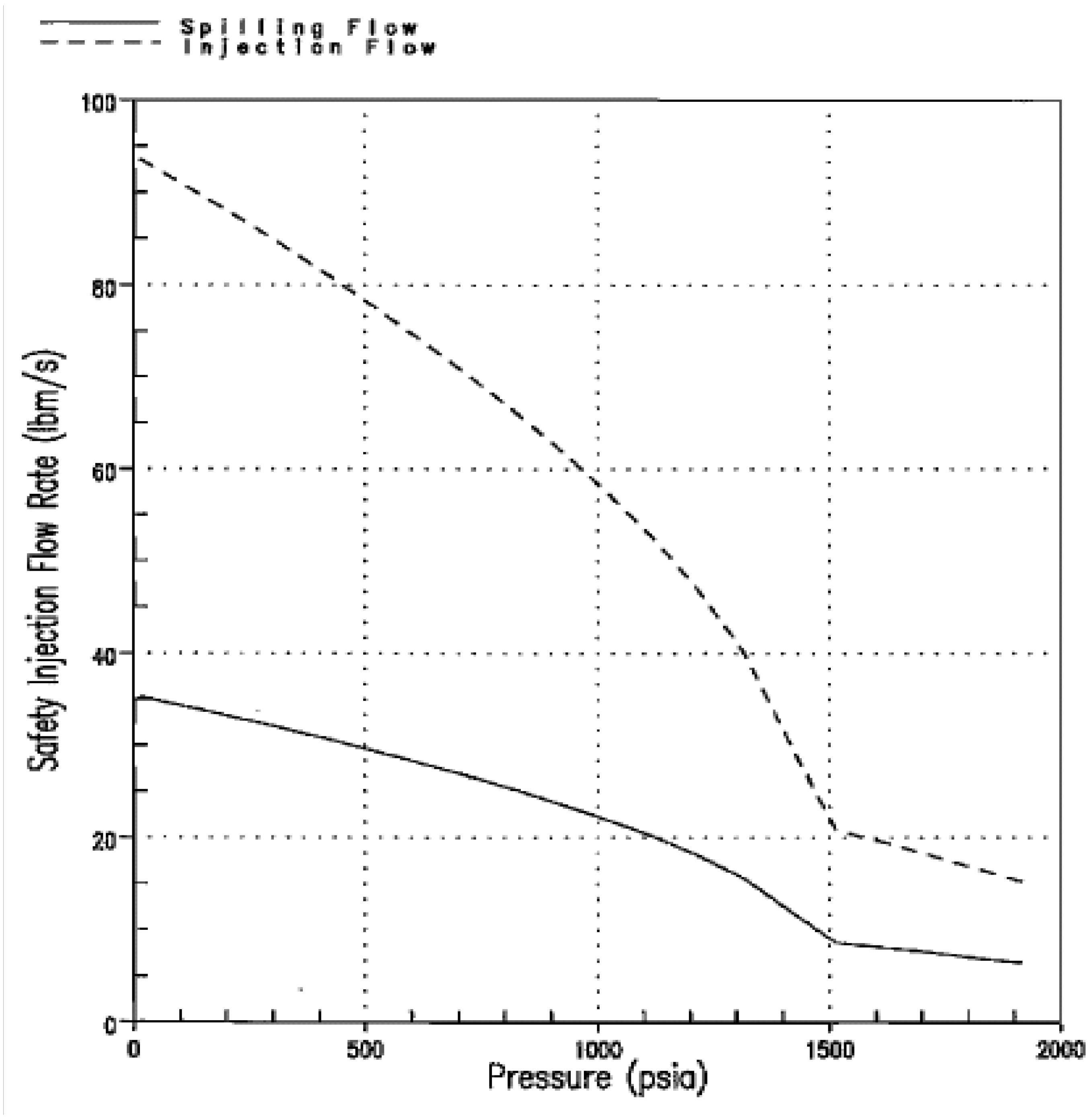


Figure 15.3-2a Pumped Safety Injection Flowrate VS. RCS Pressure (Spilling To RCS Pressure)

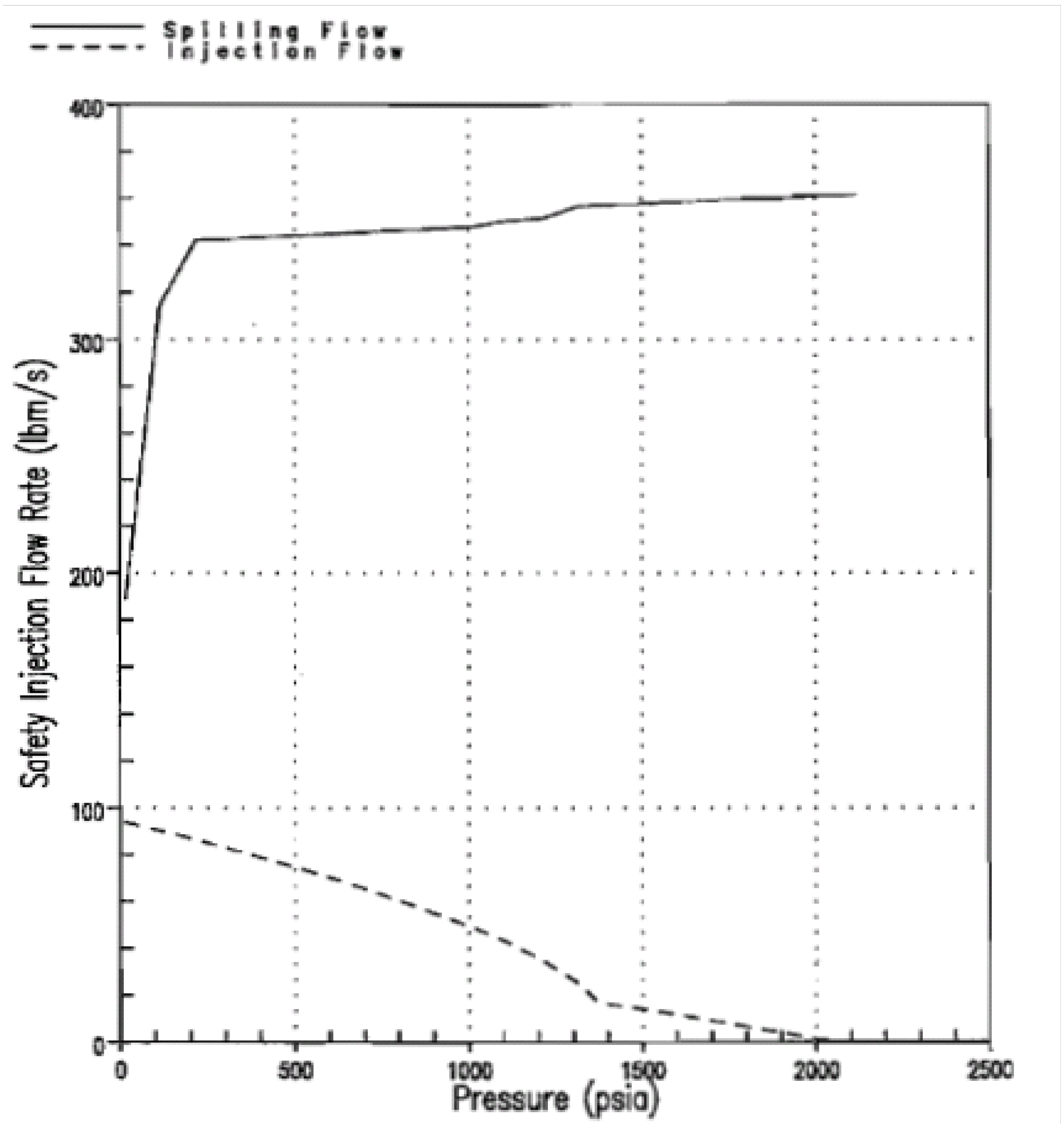


Figure 15.3-2b Pumped Safety Injection Flow Rate VS. RCS Pressure (Spilling To Containment Pressure)

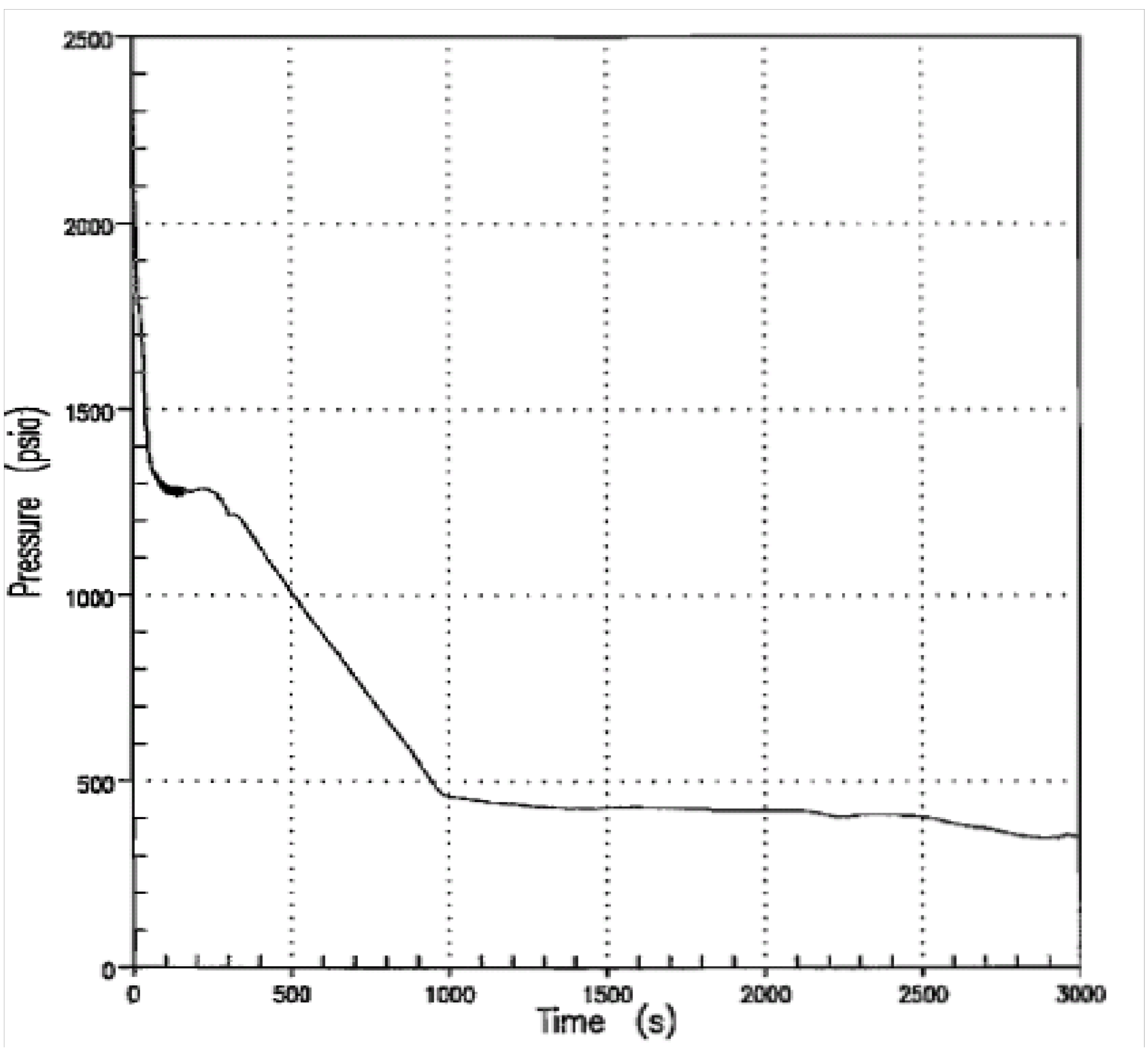


Figure 15.3-3 Reactor Coolant System Pressure for Limiting 4-Inch Break

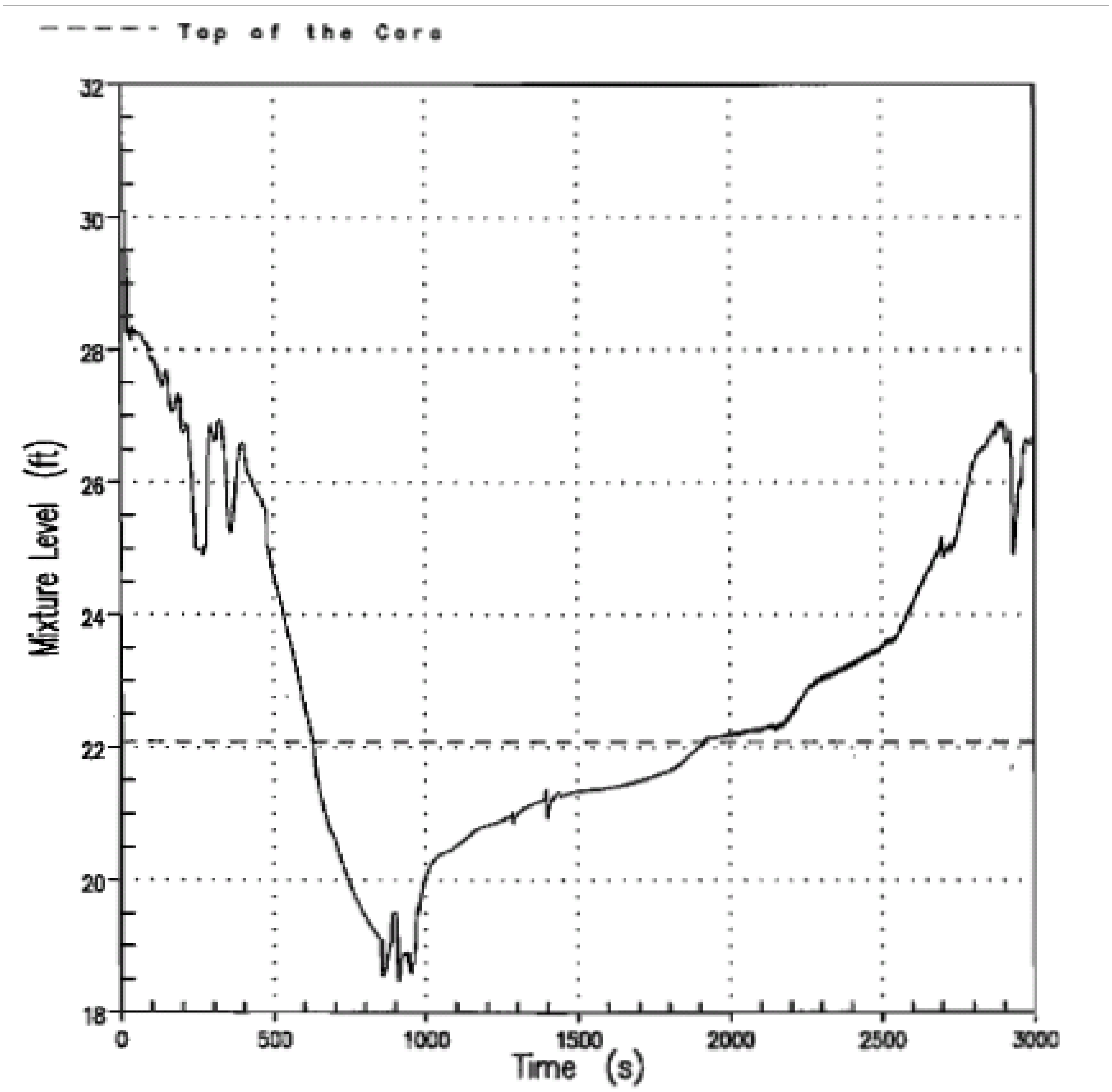


Figure 15.3-4 Core Mixture Level for Limiting 4-Inch Break

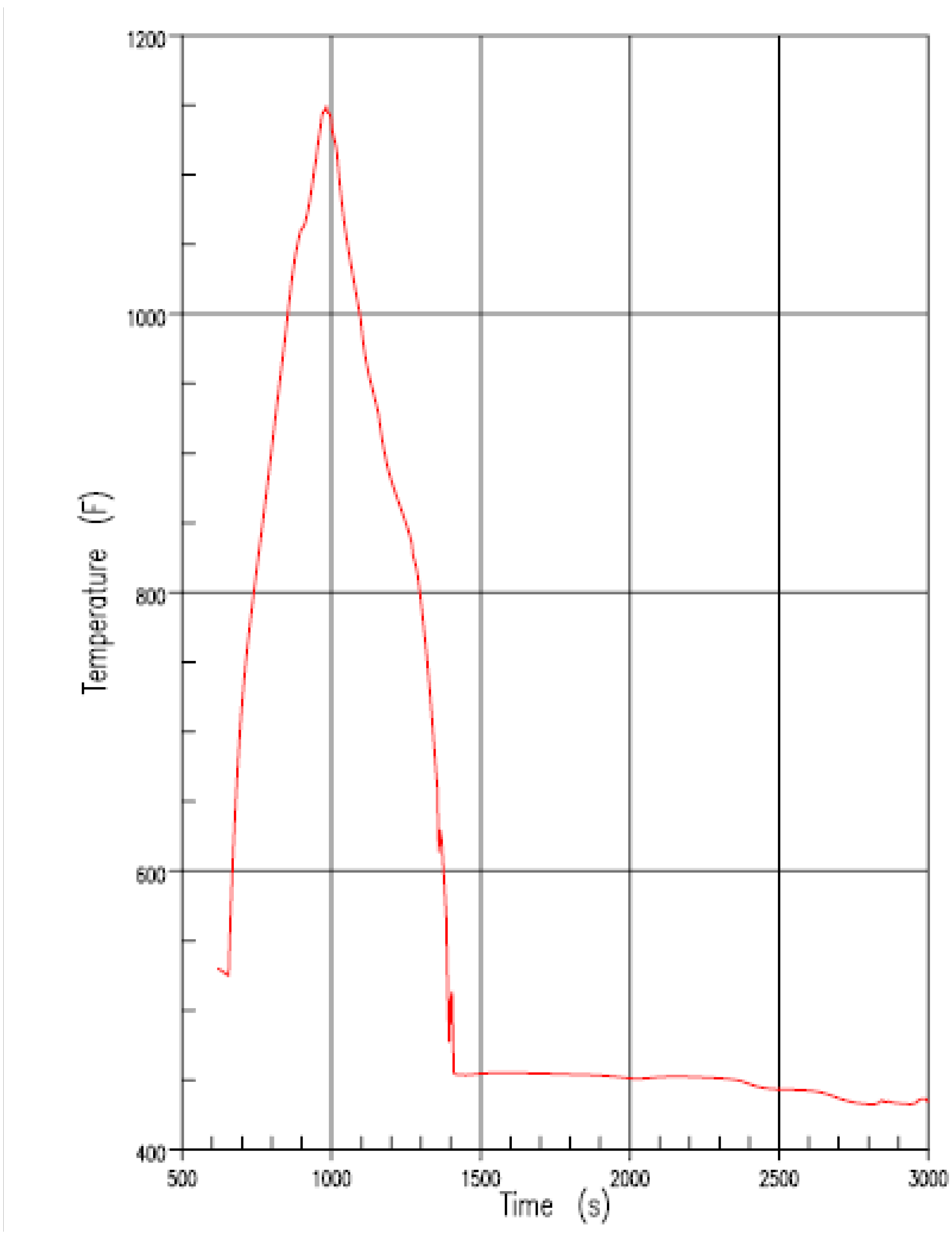


Figure 15.3-5 Cladding Temperature Transient at Peak CladdingTemperature Elevation for Limiting4-Inch Break

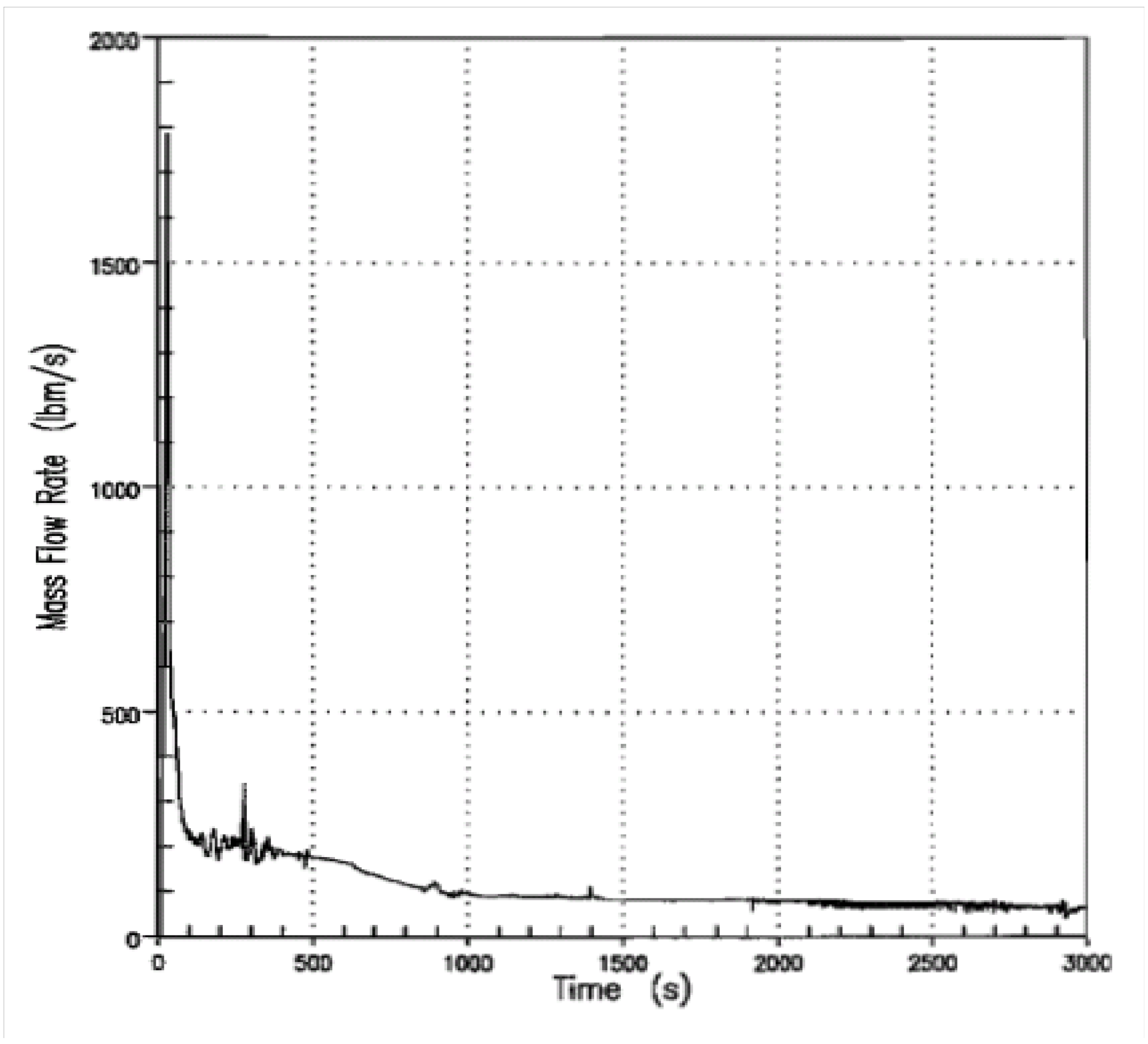


Figure 15.3-6 Core Outlet Steam Flow for Limiting 4-Inch Break

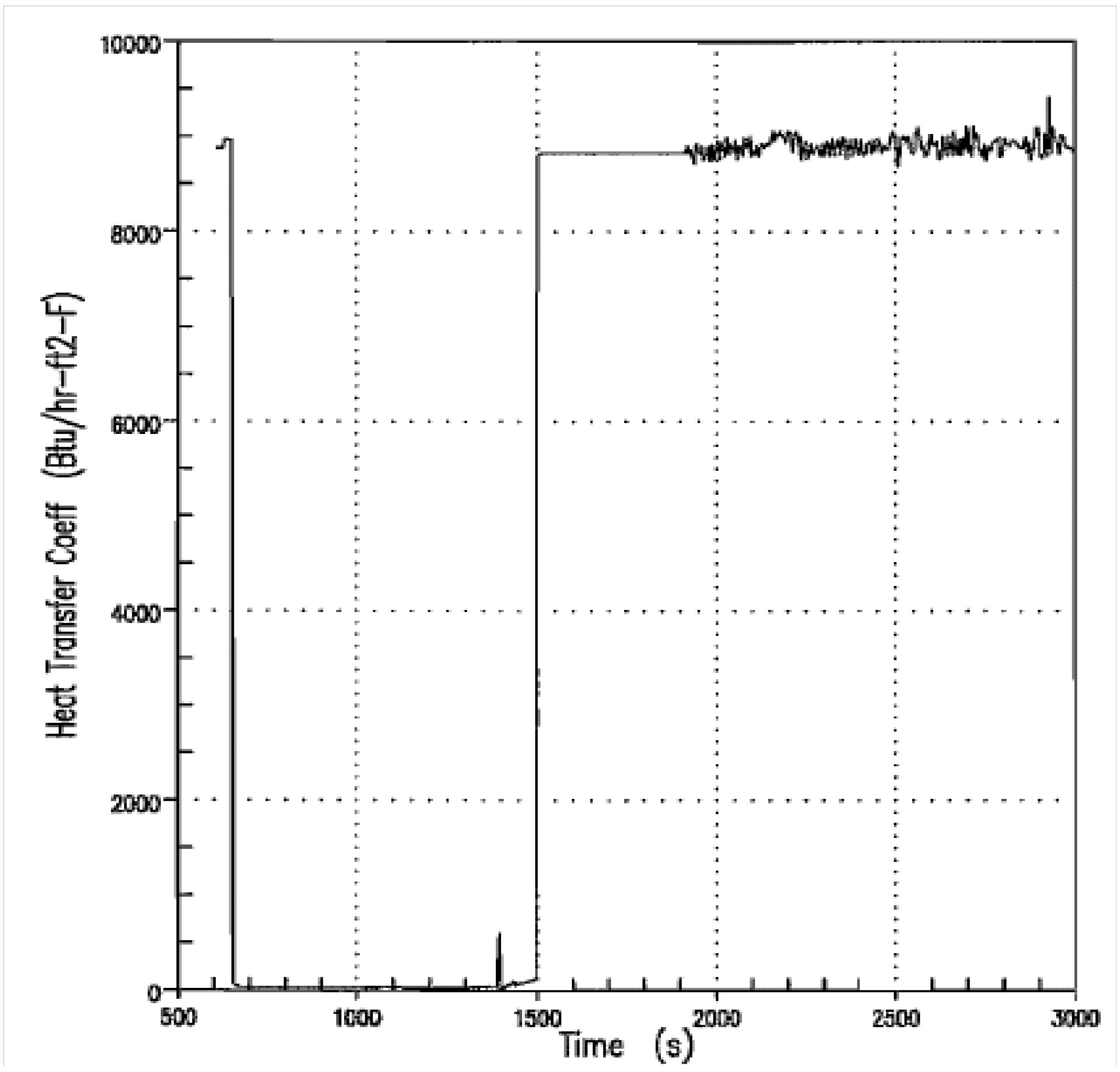


Figure 15.3-7 Cladding Surface Heat Transfer Coefficient at Peak CladdingTemperature Elevation for Limiting 4-Inch Break

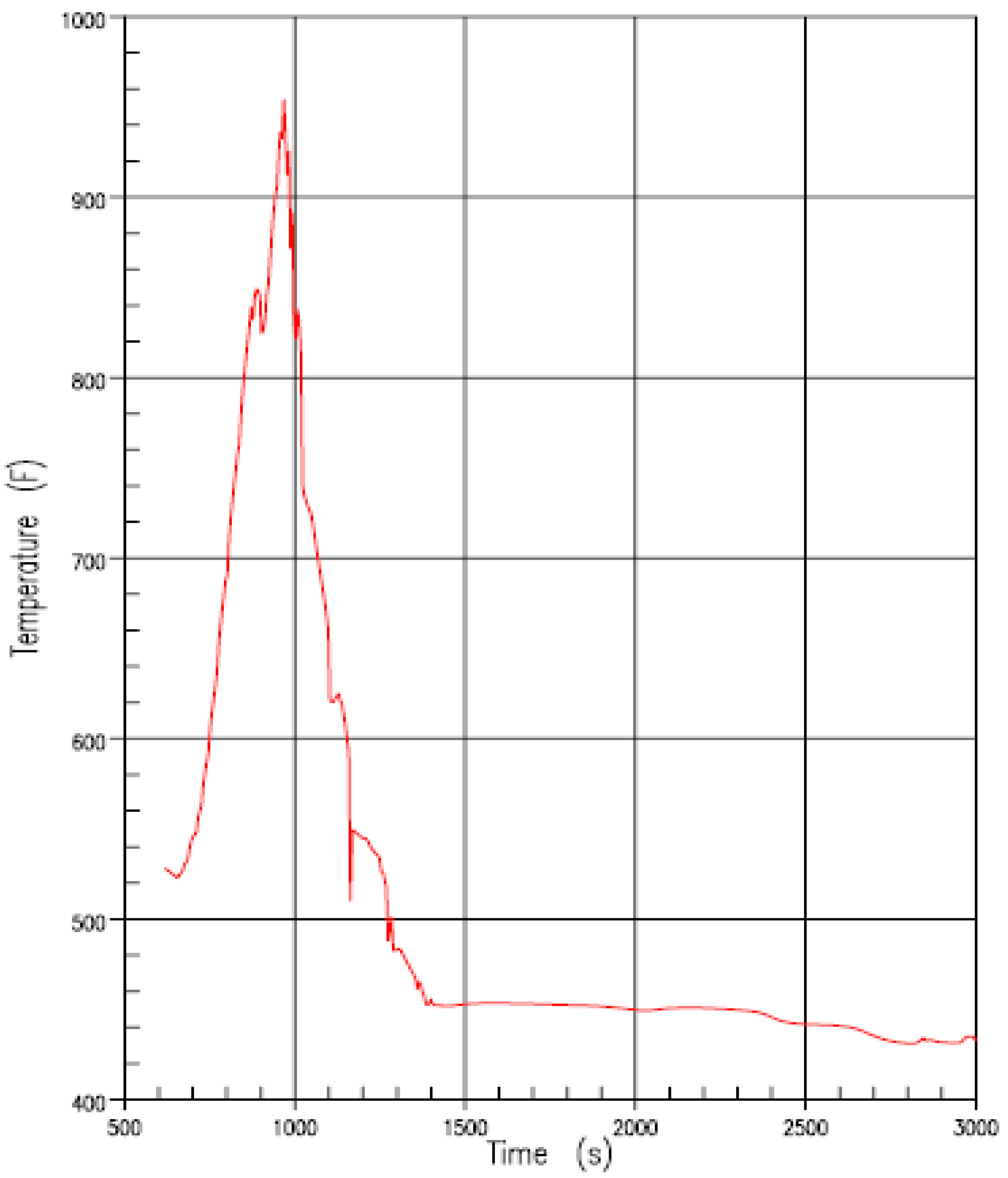


Figure 15.3-8 Fluid Temperature at Peak Cladding Temperature Elevation for Limiting 4-Inch Break

Figure 15.3-8b Deleted by Amendment 89

Figure 15.3-8c Deleted by Amendment 89

Figure 15.3-8d Deleted by Amendment 89

Figure 15.3-8e Deleted by Amendment 89

Figure 15.3-8f Deleted by Amendment 89

Figure 15.3-8g Deleted by Amendment 89

Figure 15.3-8h Deleted by Amendment 89

Figure 15.3-8i Deleted by Amendment 89

Figure 15.3-8j Deleted by Amendment 89

Figure 15.3-8k Deleted by Amendment 89

Figure 15.3-8I Deleted by Amendment 89

Figure 15.3-8m Deleted by Amendment 89

Figure 15.3-8n Deleted by Amendment 89

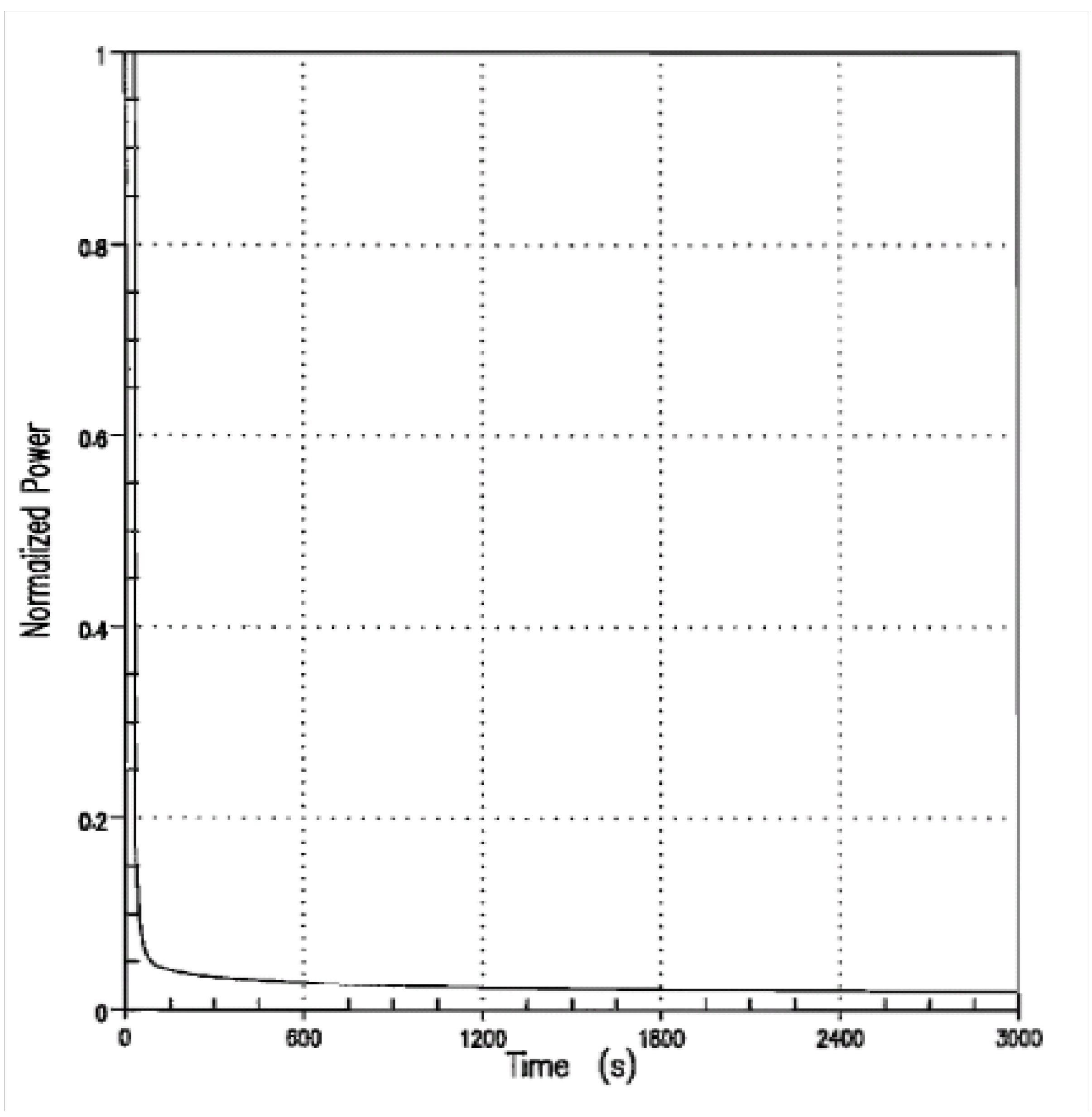


Figure 15.3-9 Core Power Transient

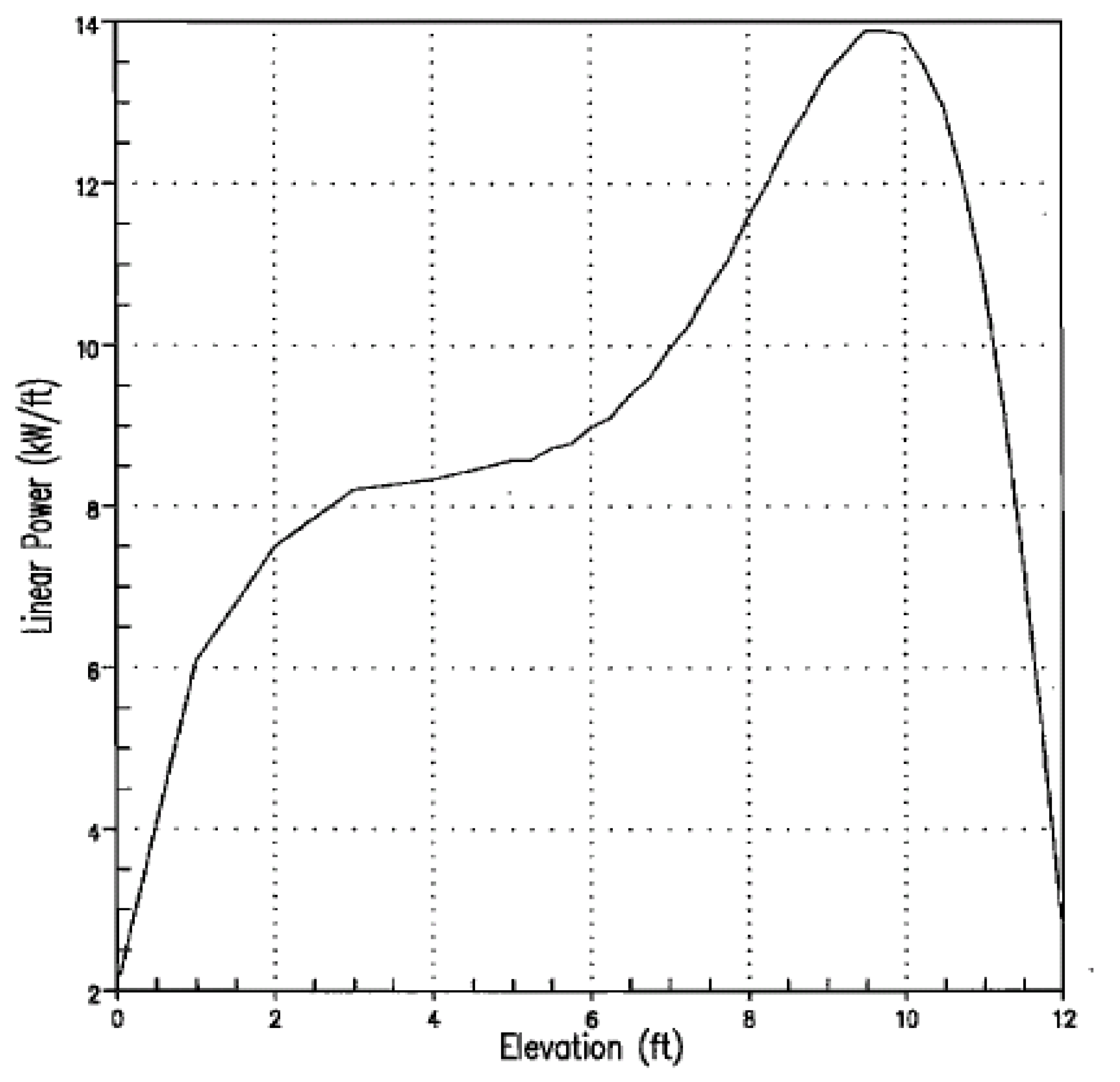


Figure 15.3-10 Hot Rod Axial Power Shape

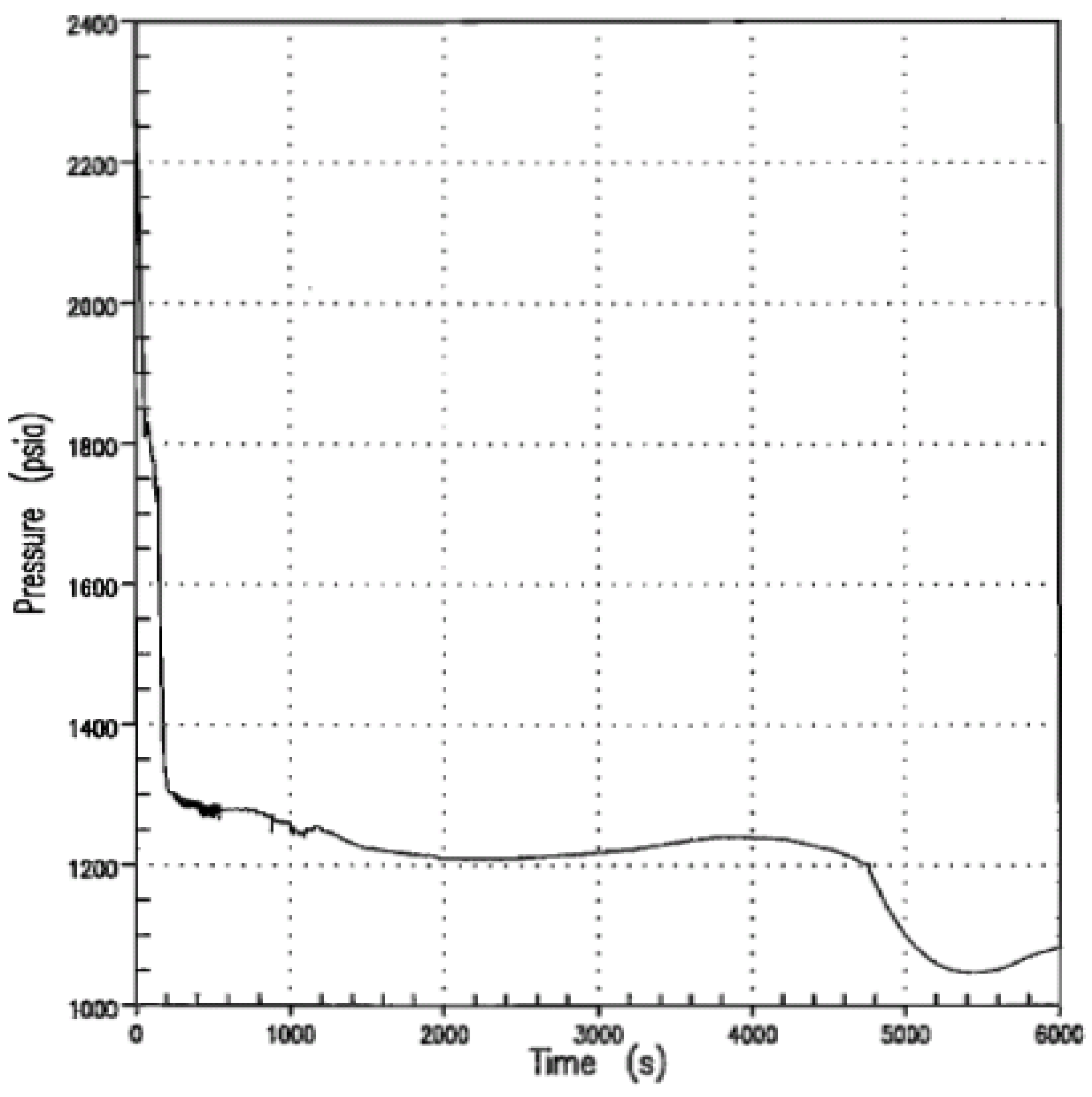


Figure 15.3-11 Reactor Coolant System Pressure for 2-Inch Break

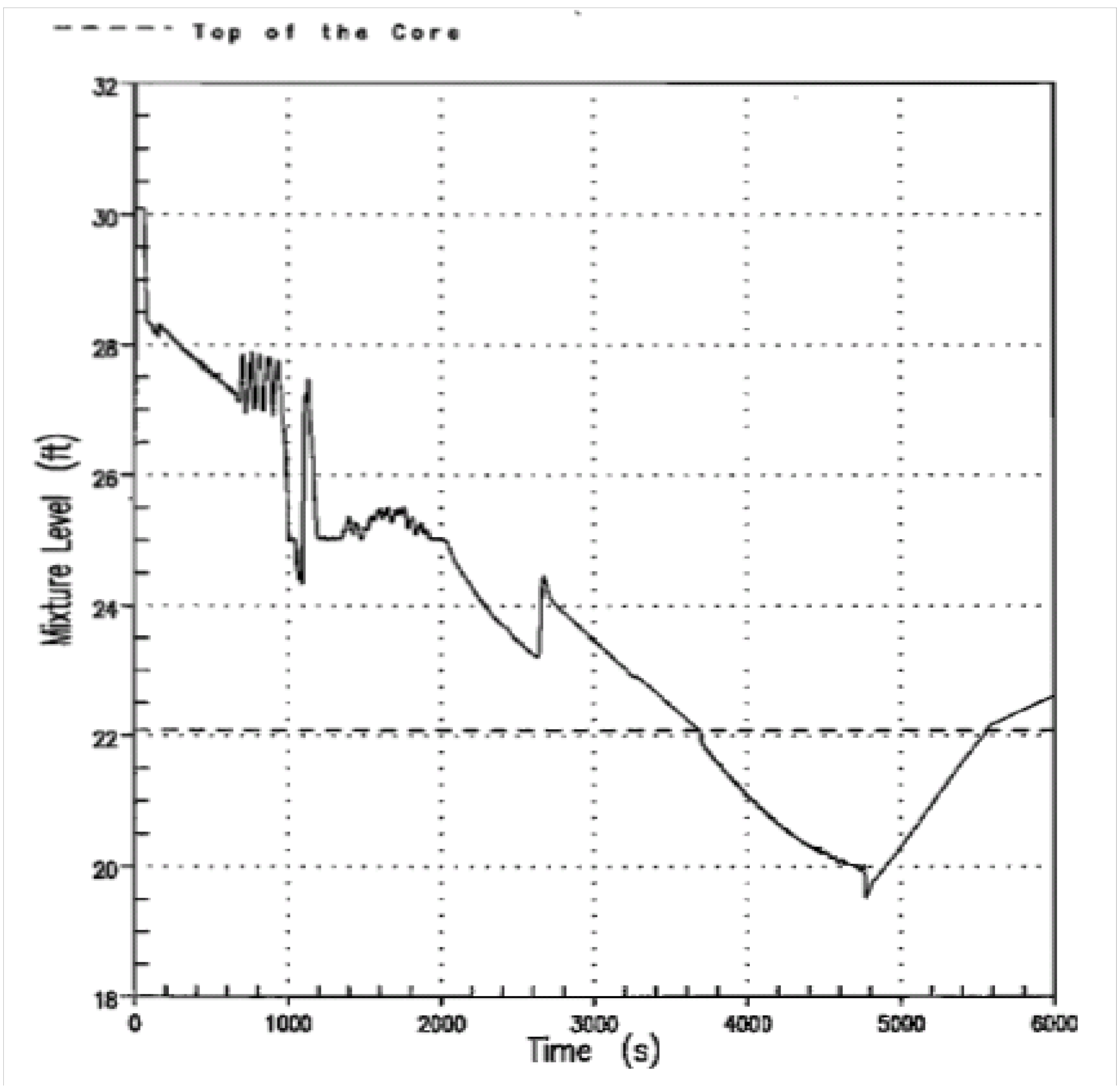


Figure 15.3-11a Core Mixture Level Transient for 2-inch Break

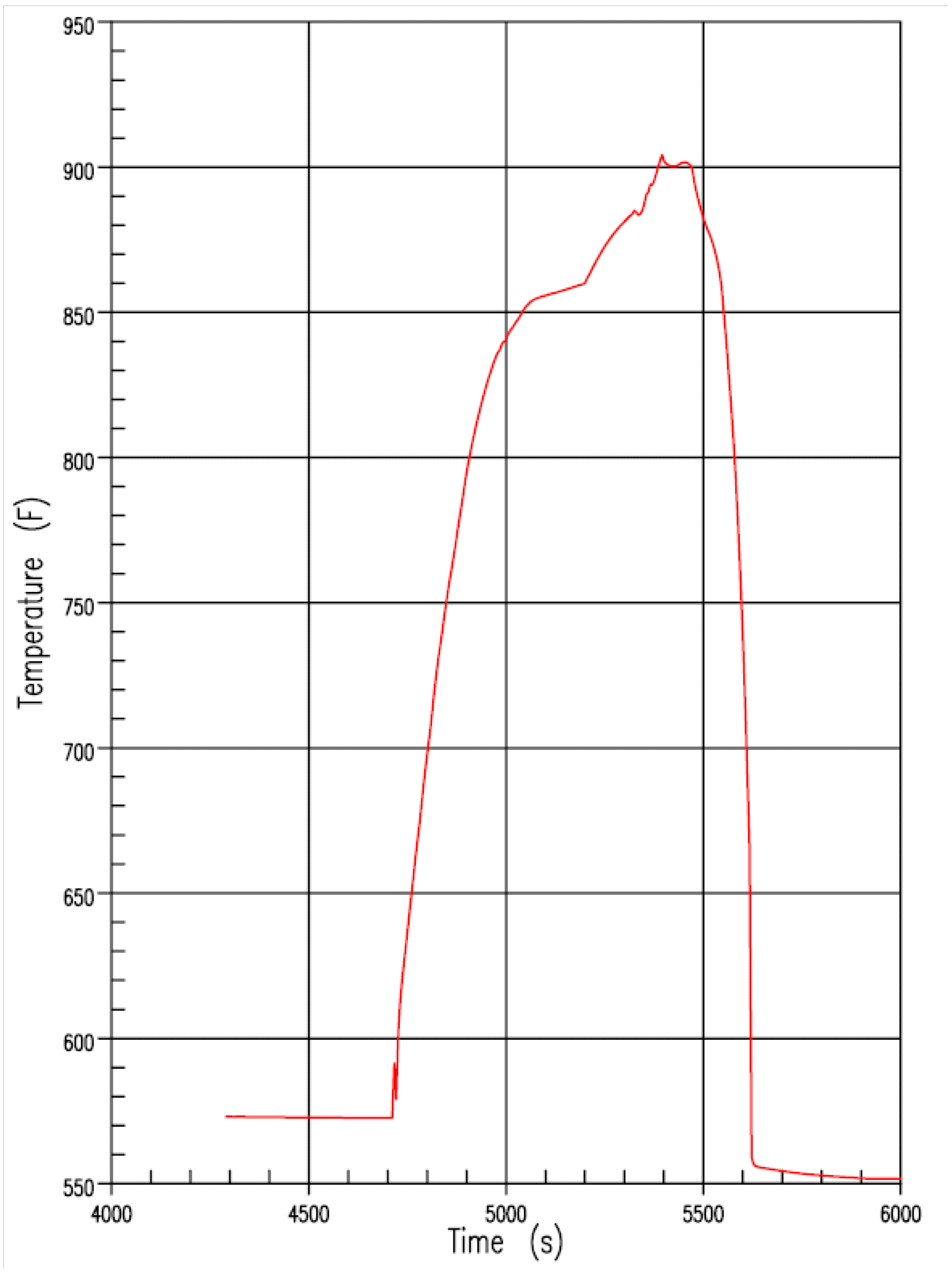


Figure 15.3-11b Cladding Temperature Transient at Peak Cladding Temperature Elevation for 2-Inch Break

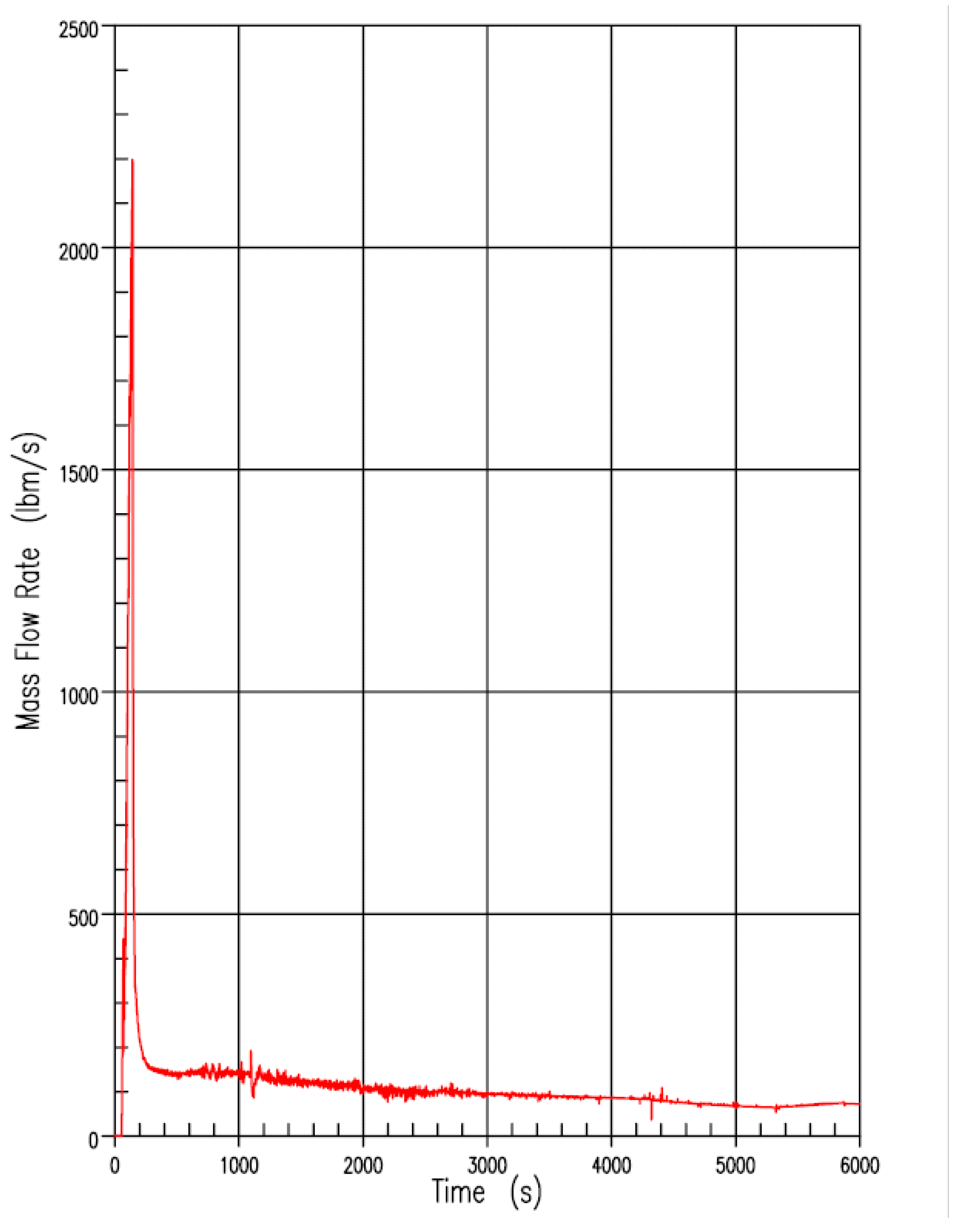


Figure 15.3-11c Core Outlet Steam Flow Rate for 2-Inch Break

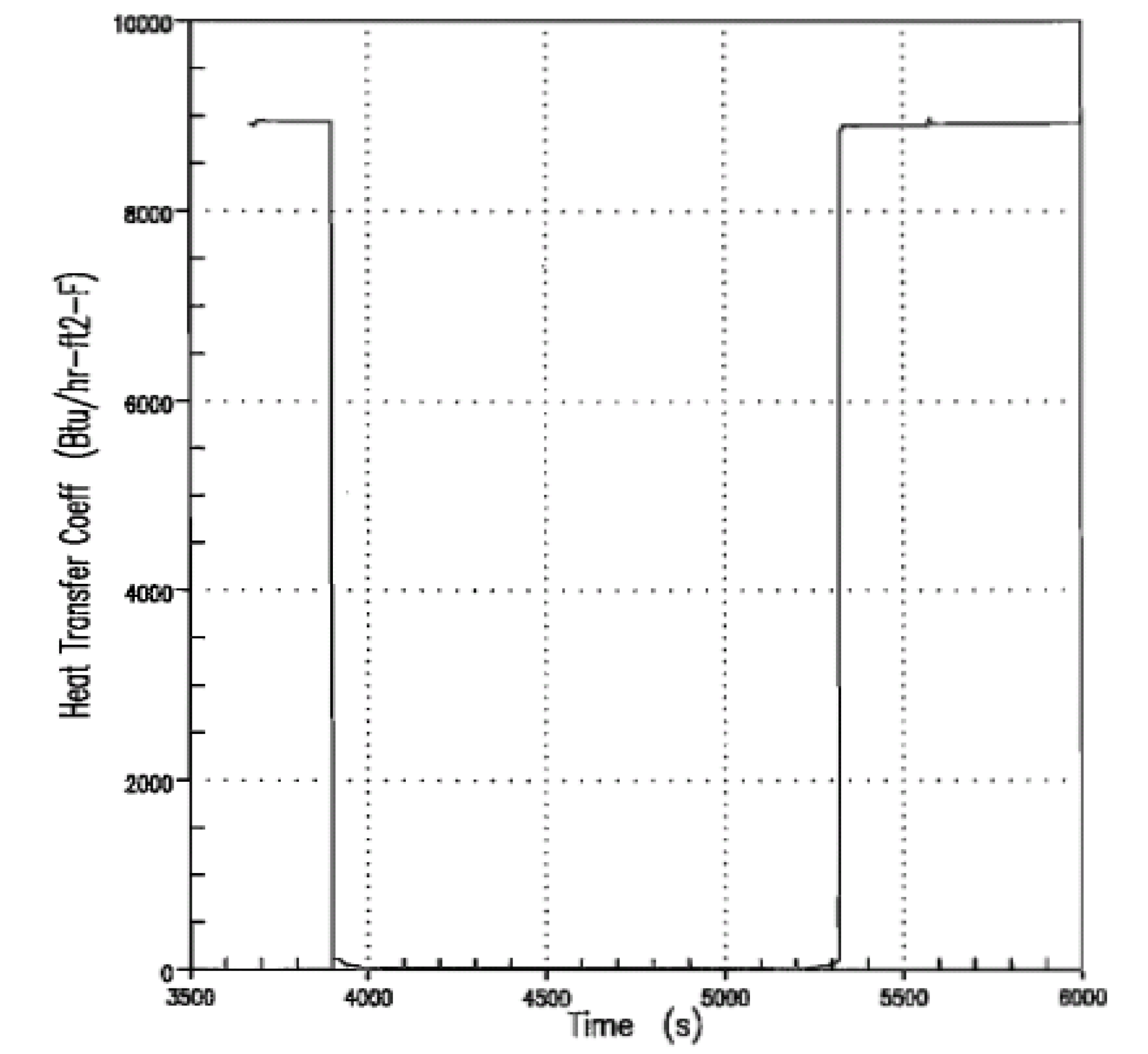


Figure 15.3-11d Cladding Surface Heat Transfer Coefficient at Peak Cladding Temperature Elevation for 2-Inch Break

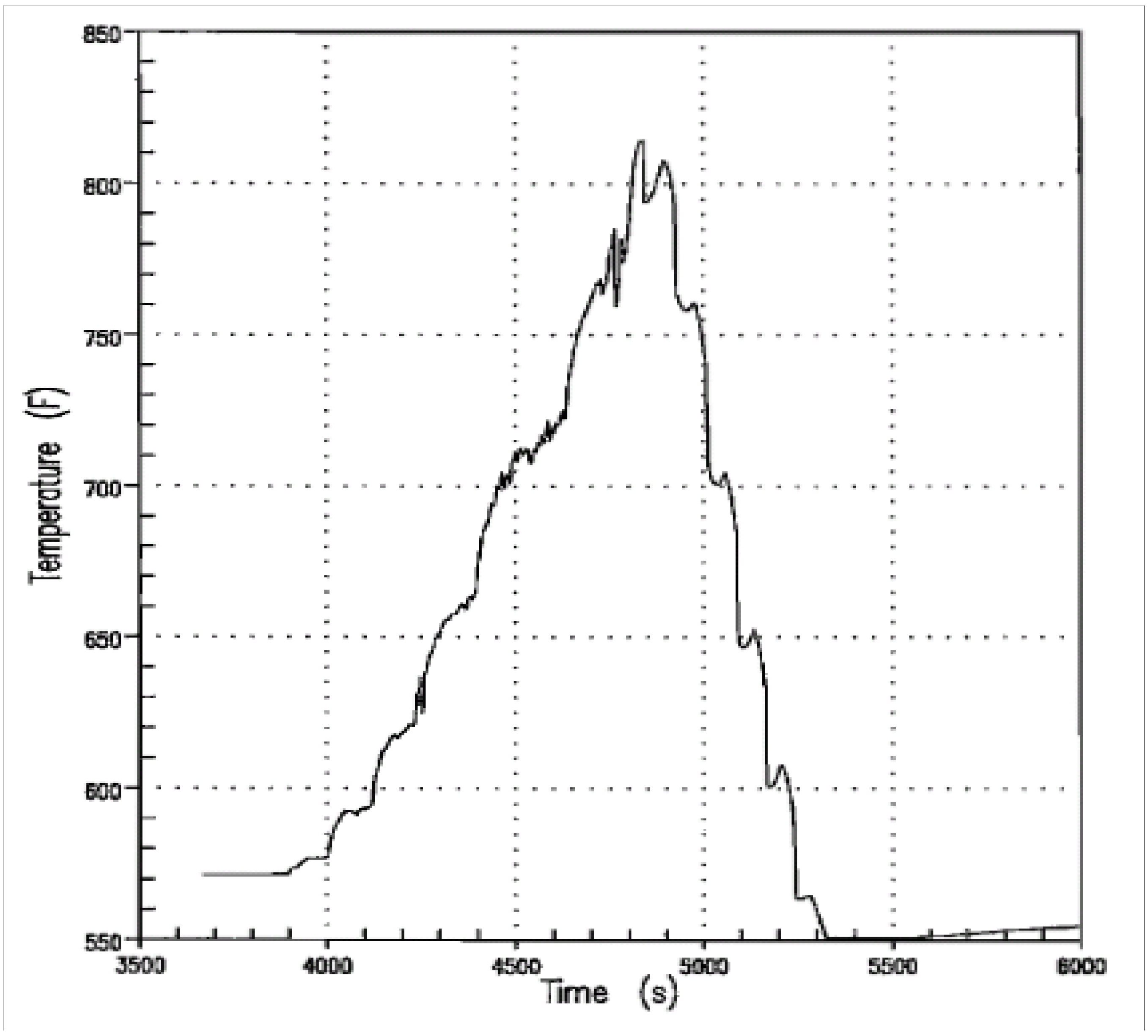


Figure 15.3-11e Fluid Temperature at Peak Cladding Temperature Elevation for 2-Inch Break

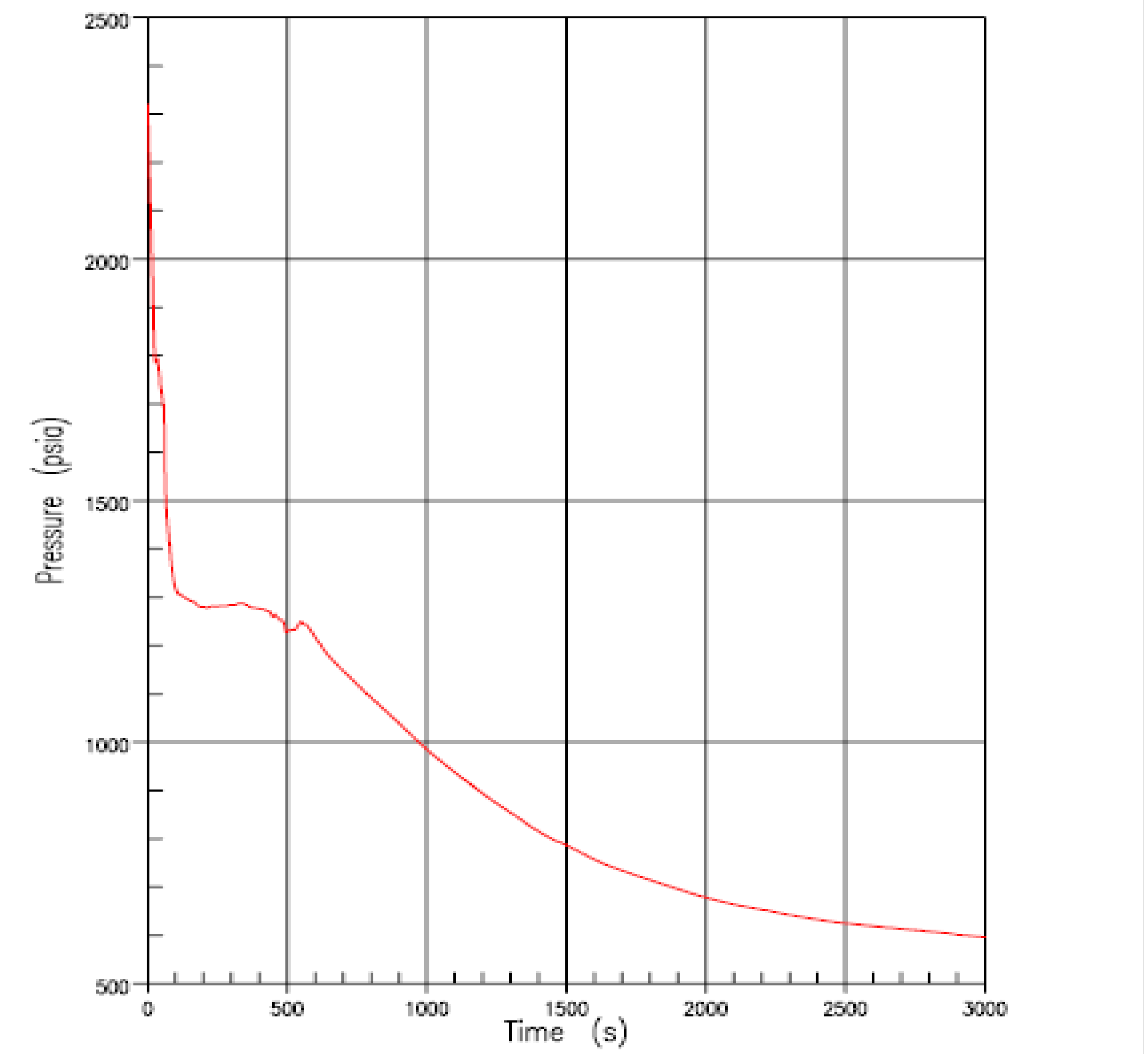
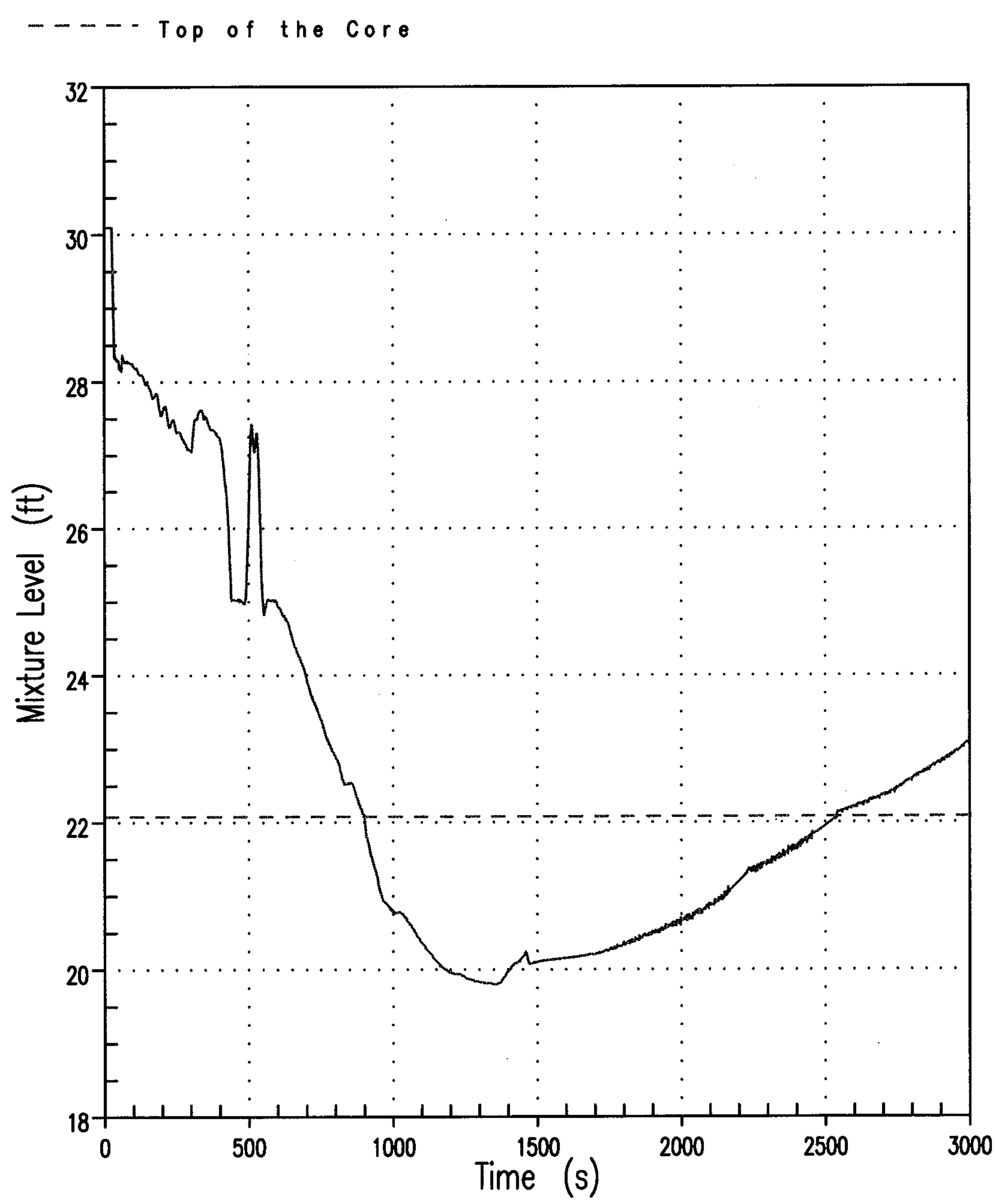


Figure 15.3-12 Reactor Coolant System Pressure for 3-Inch Break



WATTS BAR NUCLEAR PLANT FINAL SAFETY ANALYSIS REPORT
CORE MIXTURE LEVEL TRANSIENT FOR 3-INCH BREAK Figure 15.3-12a

Figure 15.3-12a Core Mixture Level Transient for 3-Inch Break

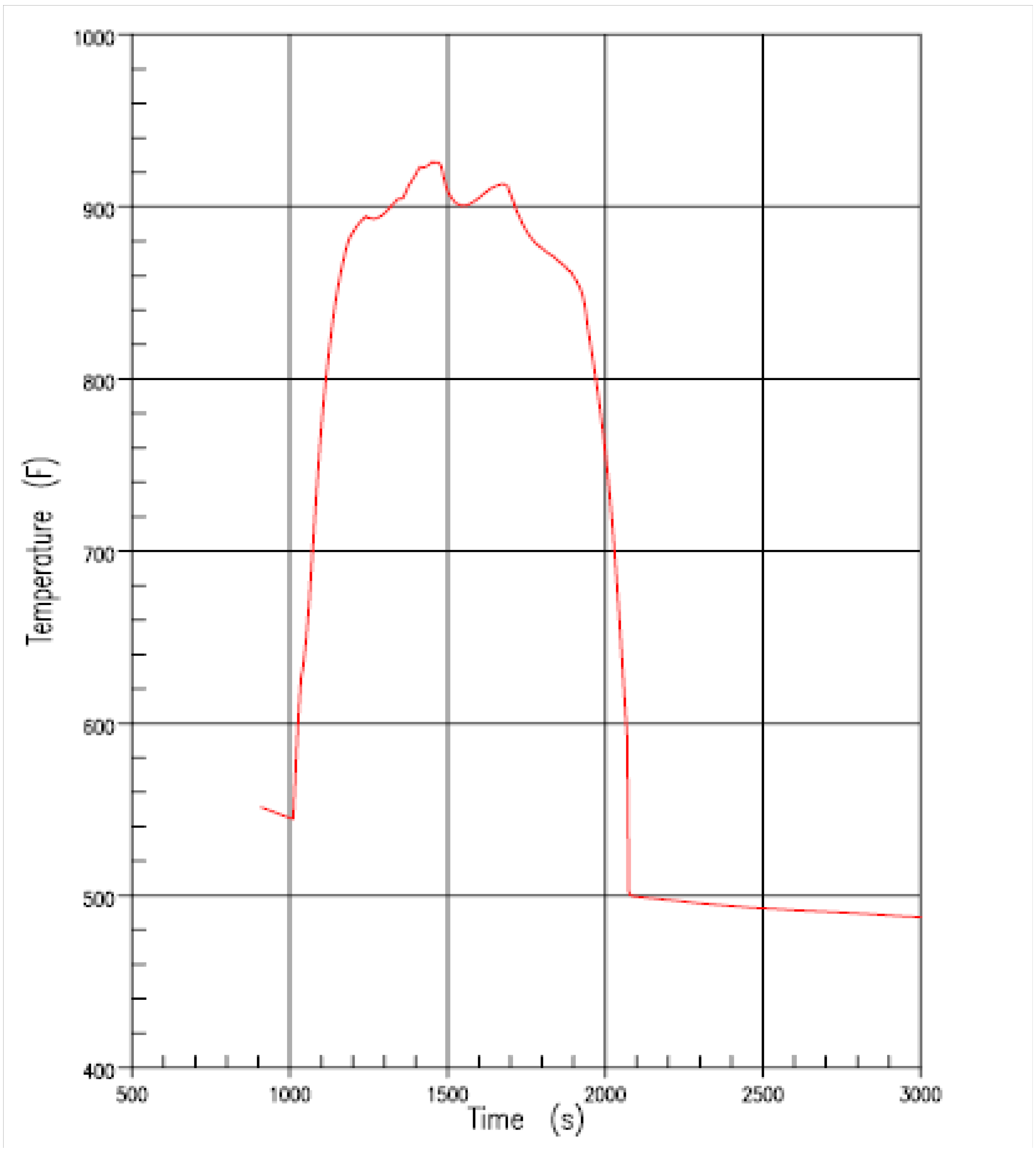
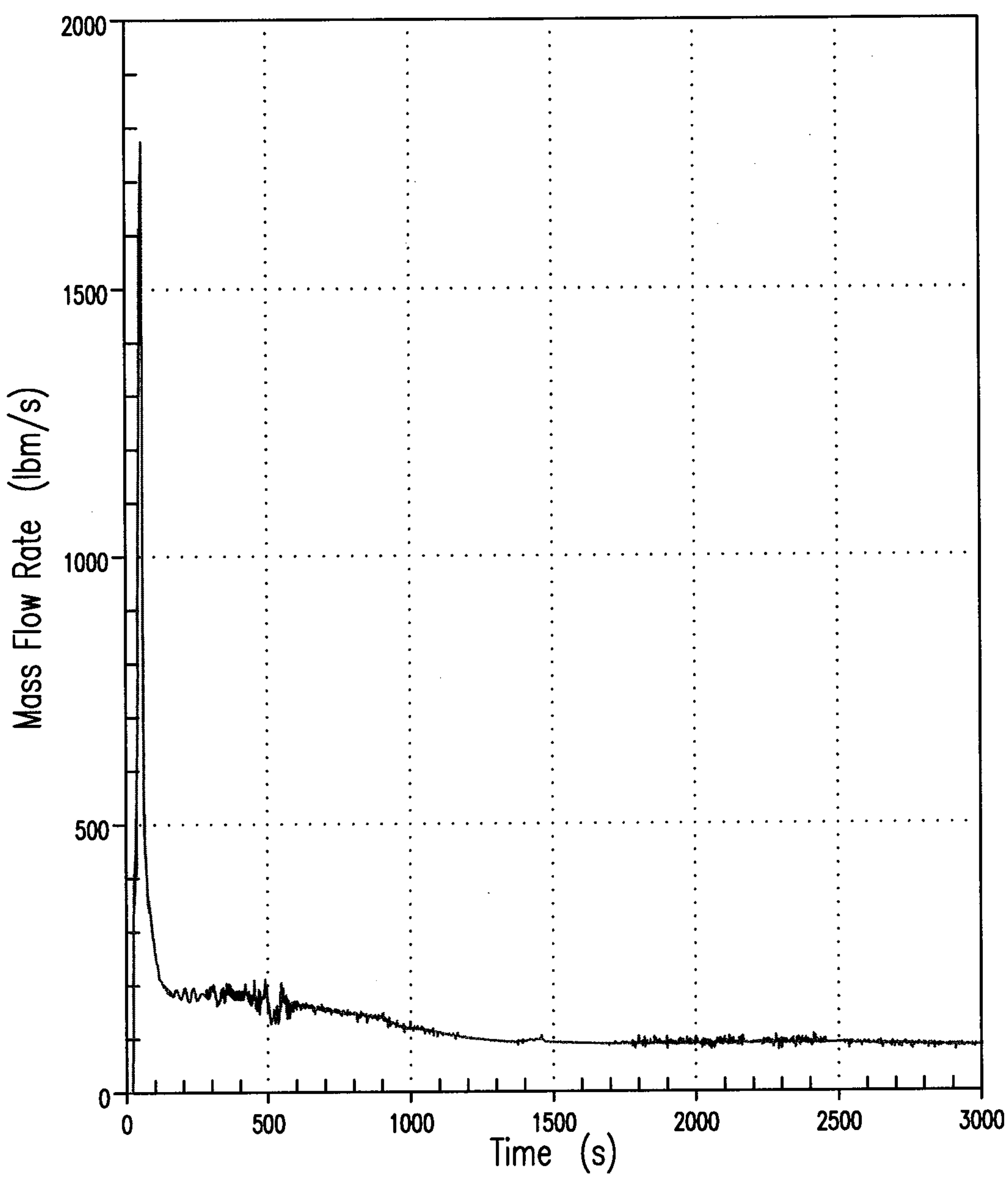


Figure 15.3-12b Clad Temperature Transient at Peak Temperature Elevation for 3-Inch Break



WATTS BAR NUCLEAR PLANT FINAL SAFETY ANALYSIS REPORT
CORE OUTLET STEAM FLOW RATE FOR 3-INCH BREAK Figure 15.3-12c

Figure 15.3-12c Core Outlet Steam Flow Rate for 3-Inch Break

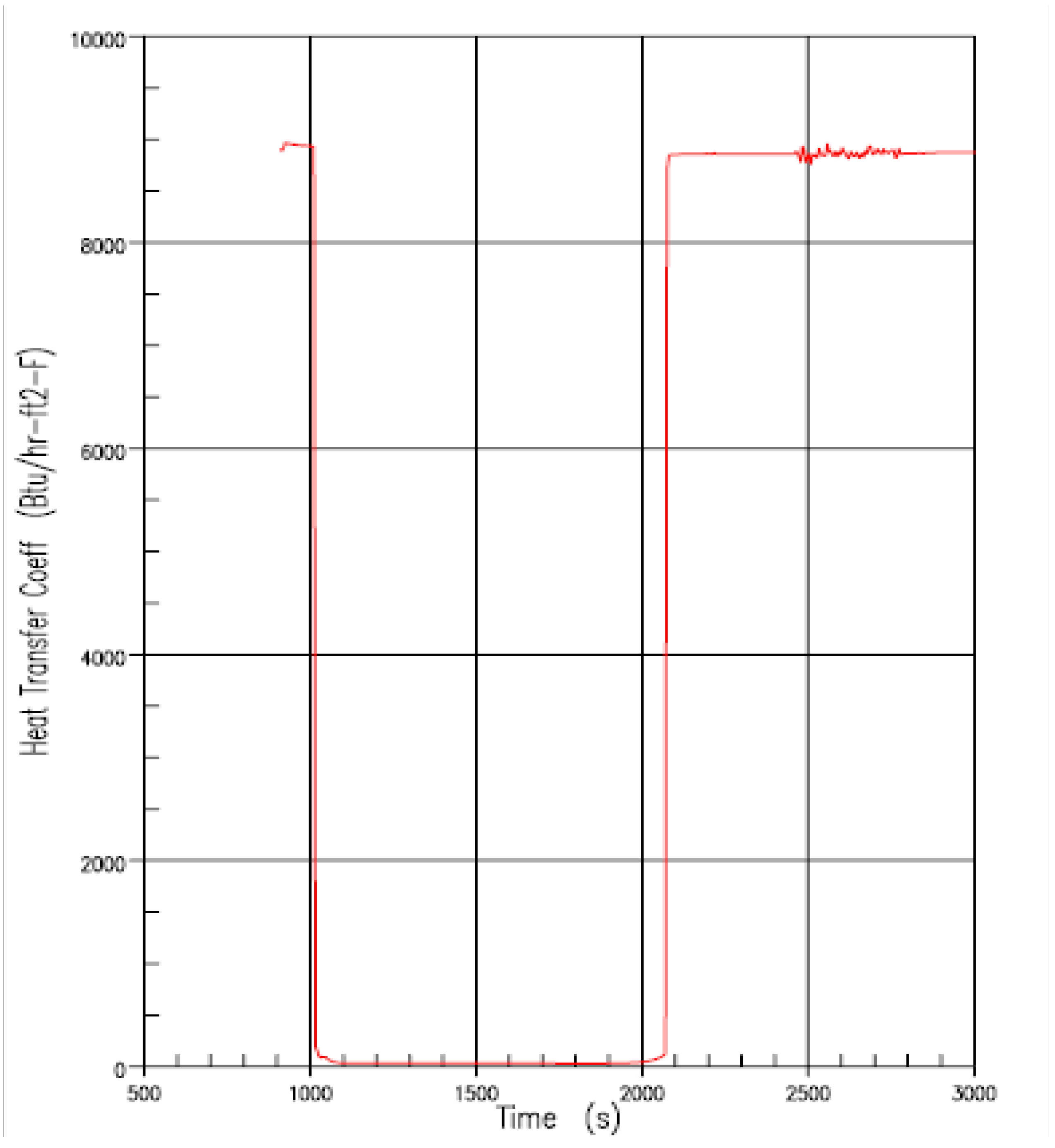


Figure 15.3-12d Clad Surface Heat Transfer Coefficient at Peak Clad Temperature Elevation for 3-Inch Break

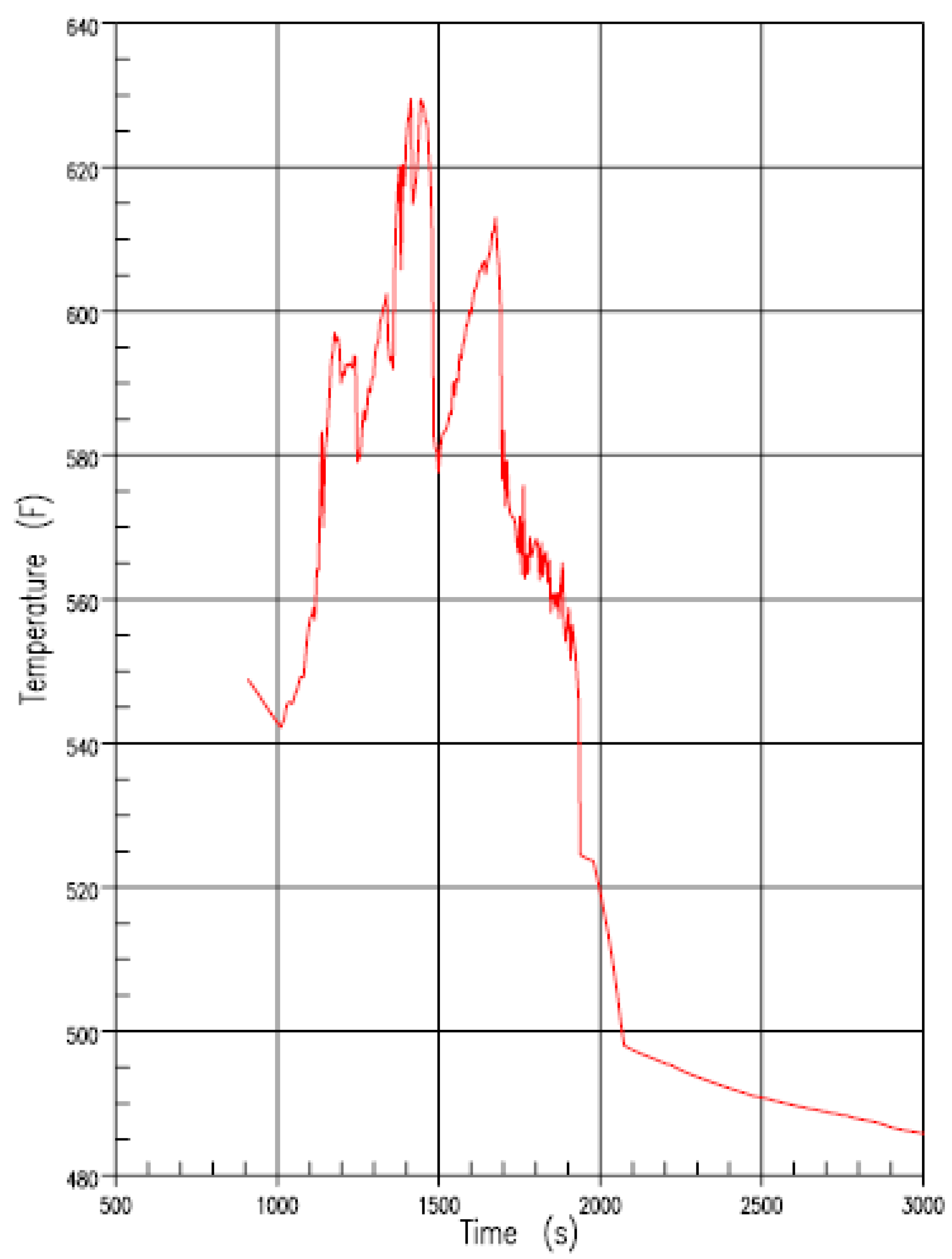


Figure 15.3-12e Fluid Temperature at Peak Clad Temperature Elevation for 3-Inch Break

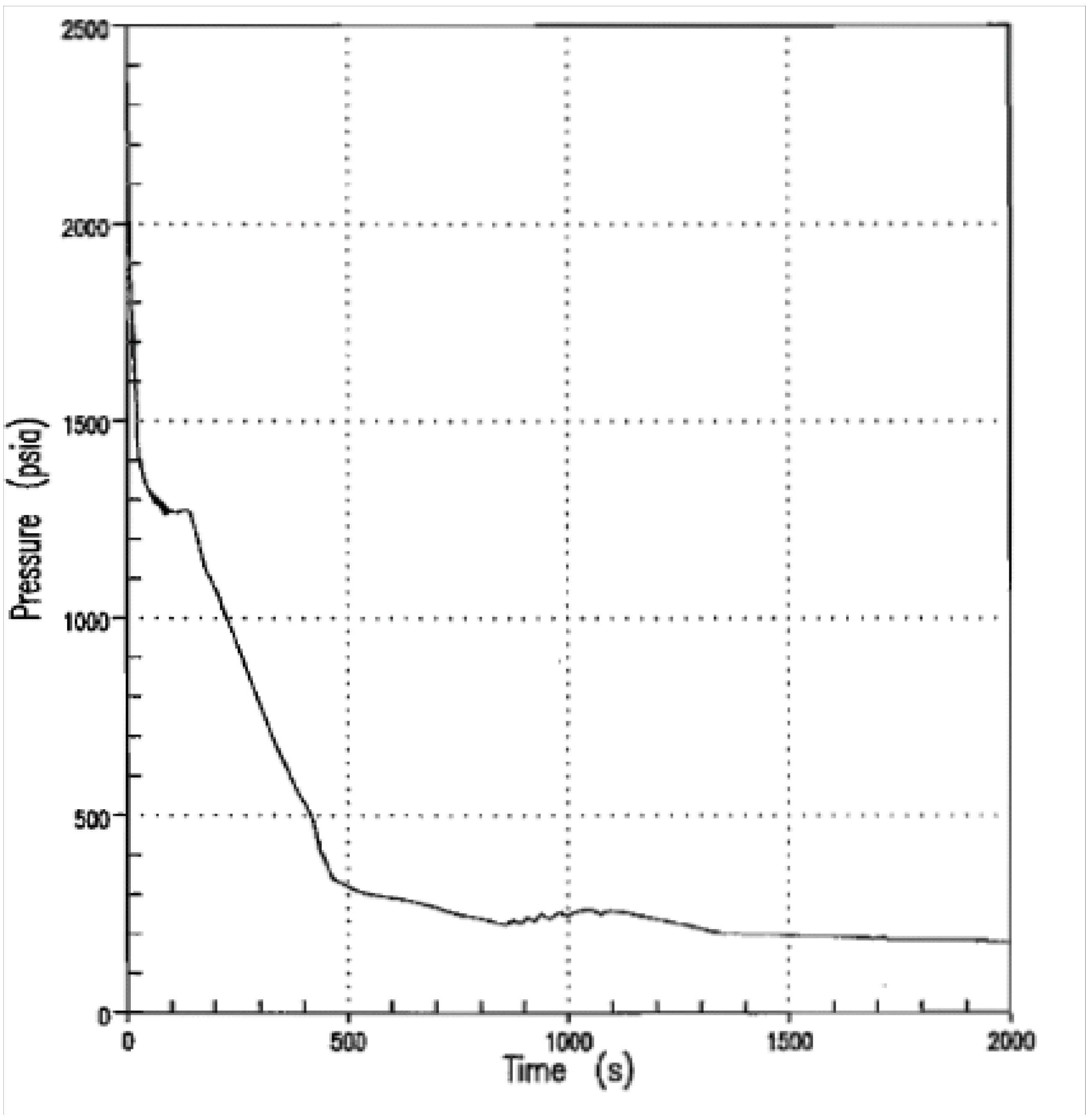


Figure 15.3-13 Reactor CoolantSystem Pressure For 6-Inch Break

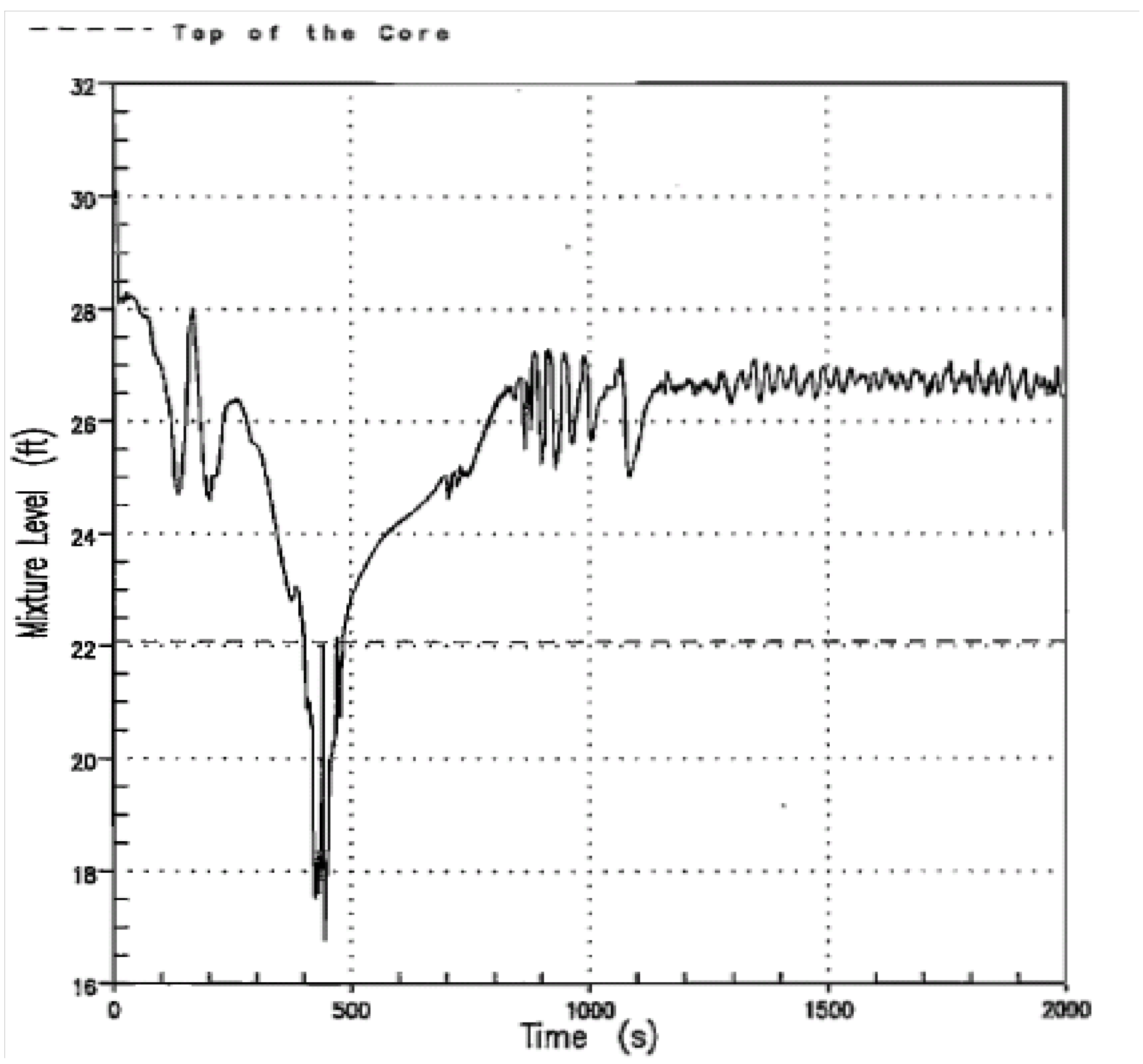
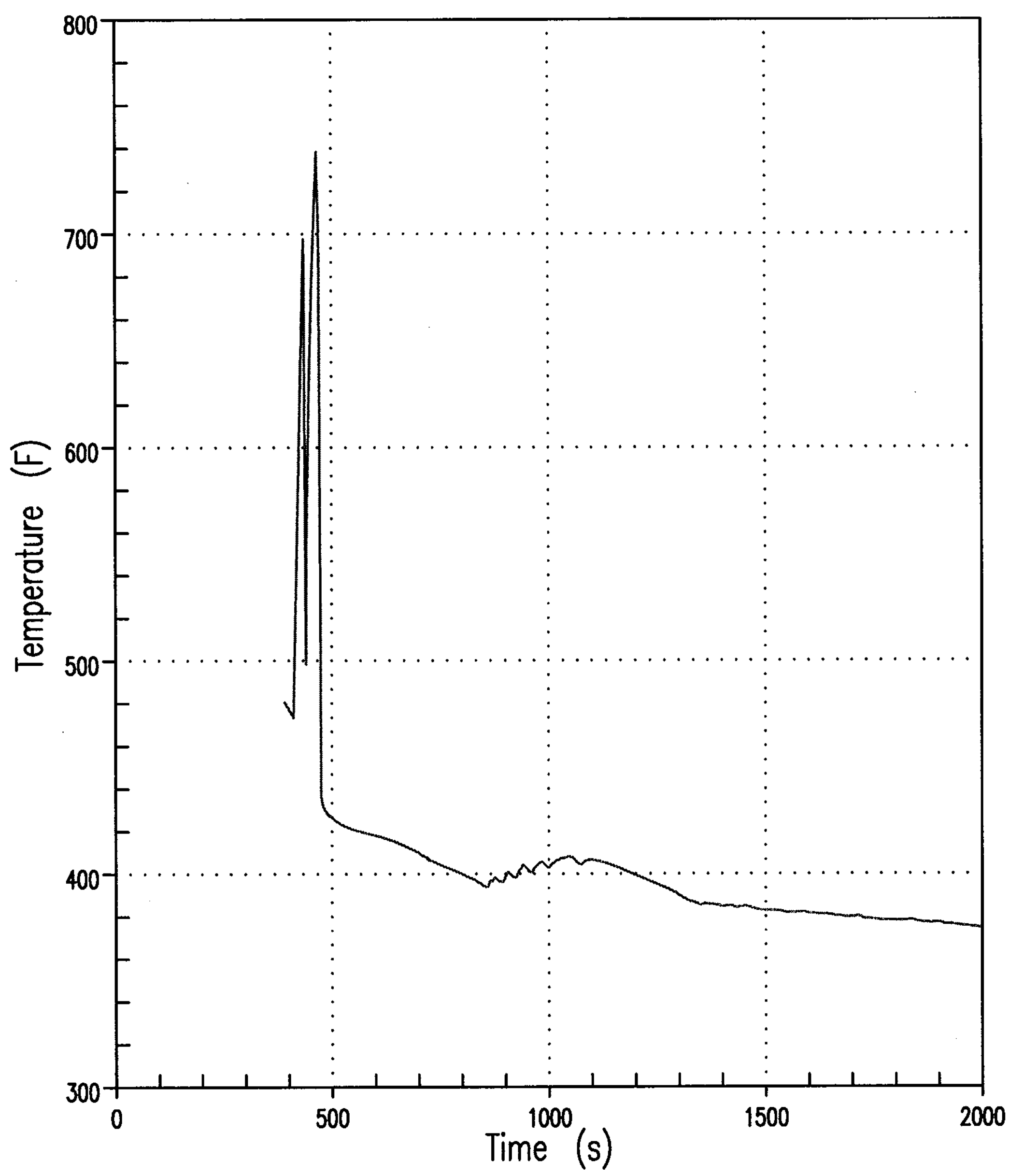


Figure 15.3-13a Core Mixture Level Transient For 6-Inch Break



WATTS BAR NUCLEAR PLANT FINAL SAFETY ANALYSIS REPORT
CLADDING TEMPERATURE TRANSIENT AT PEAK CLADDING TEMPERATURE ELEVATION FOR 6-INCH BREAK Figure 15.3-13b

Figure 15.3-13b Cladding Temperature Transient At Peak Cladding Temperature Elevation For 6-Inch Break

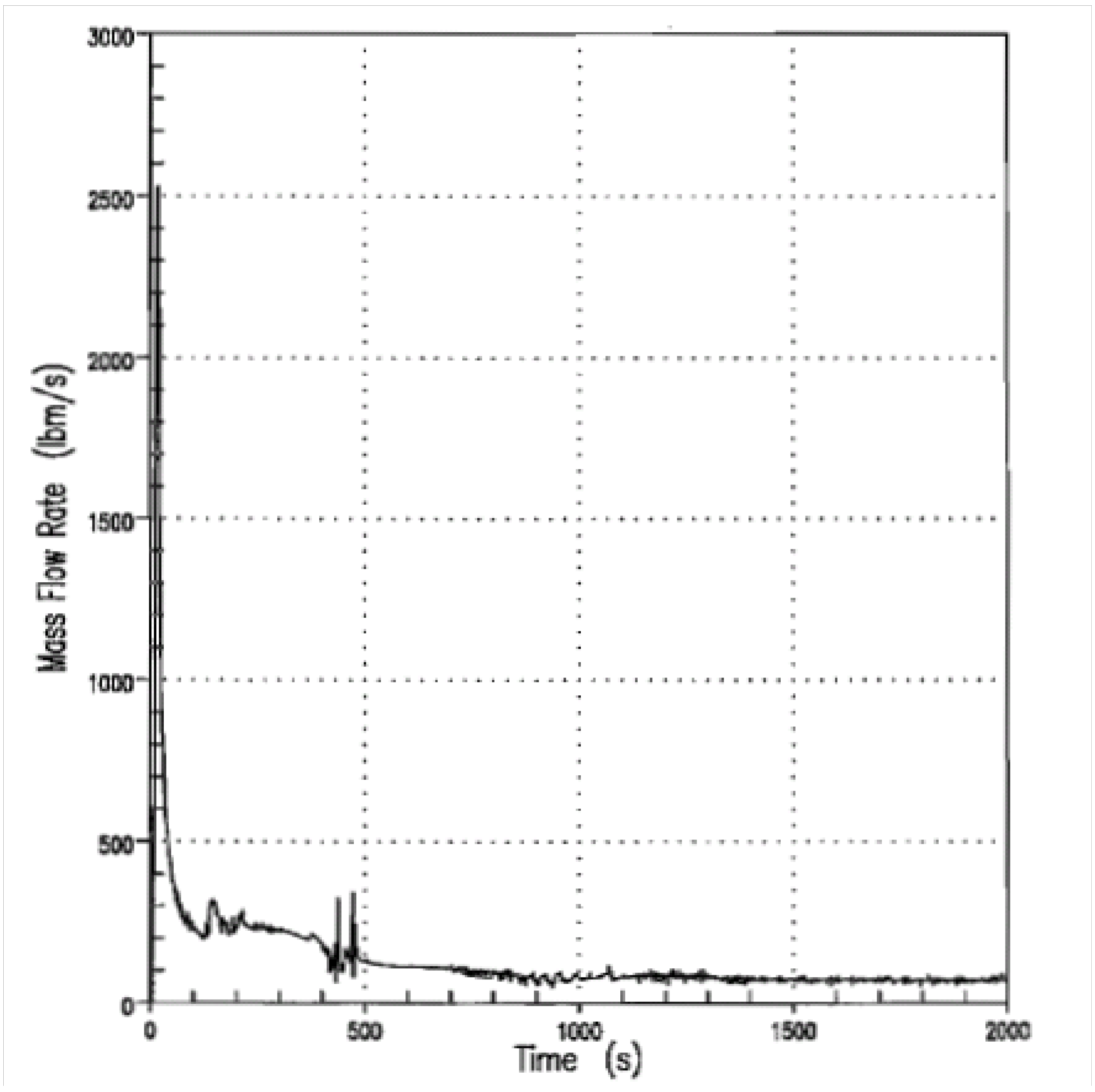


Figure 15.3-13c Core Outlet Steam Flow Rate For 6-Inch Break

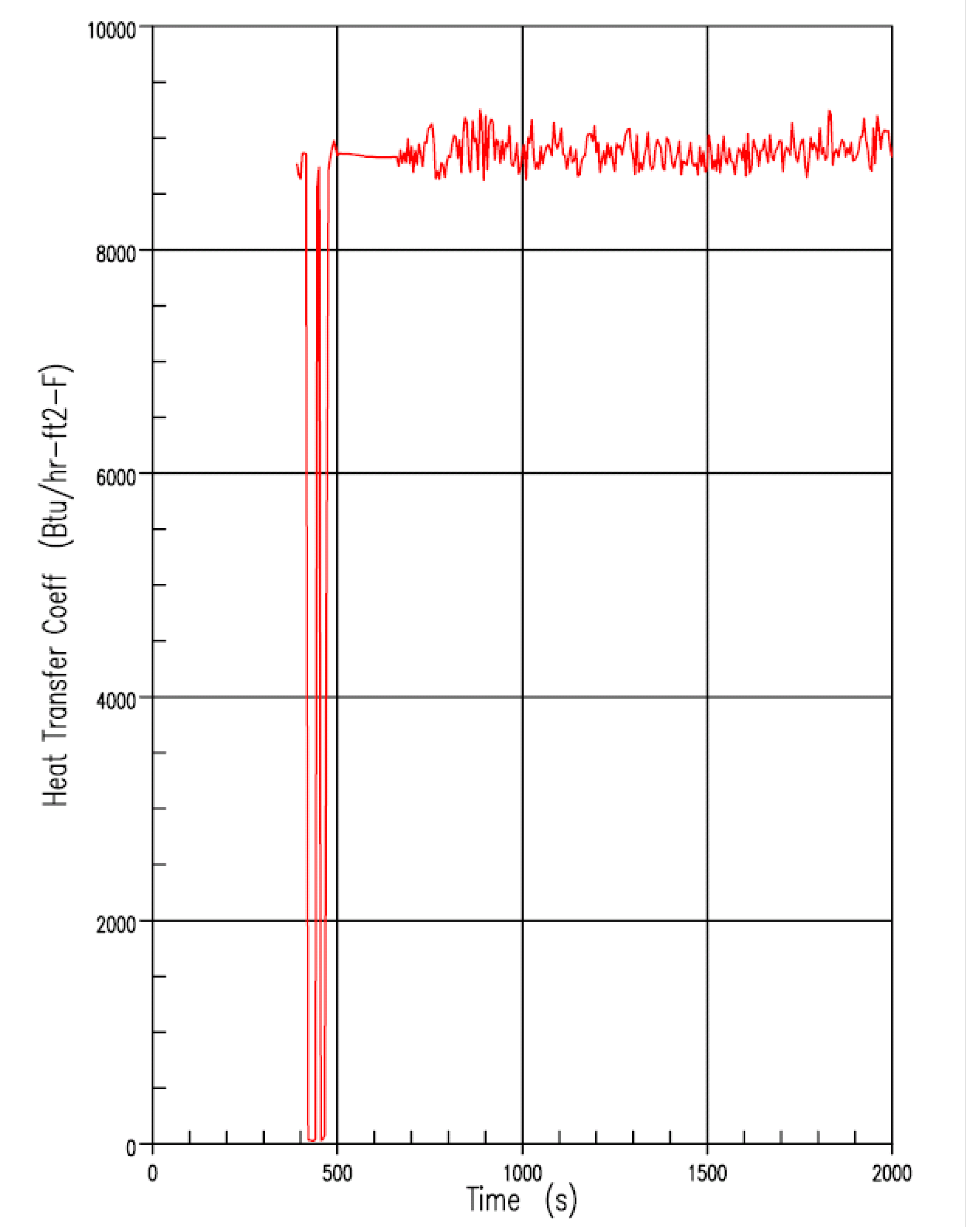


Figure 15.3-13d Cladding Surface Heat Transfer Coefficient At Peak Cladding Temperature Elevation For 6-Inch Break

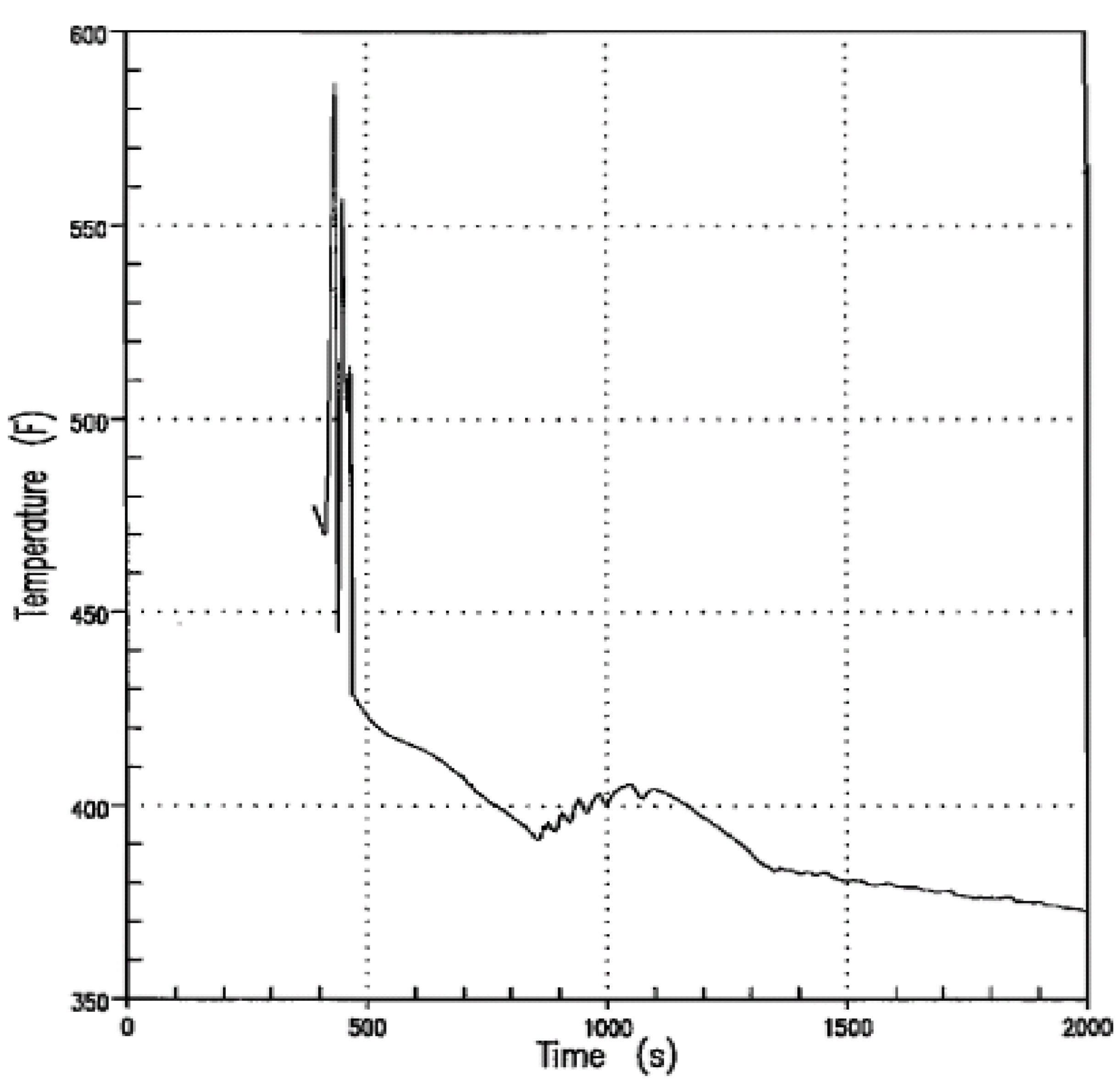


Figure 15.3-13e Fluid Temperature At Peak Cladding Temperature Elevation For 6-Inch Break

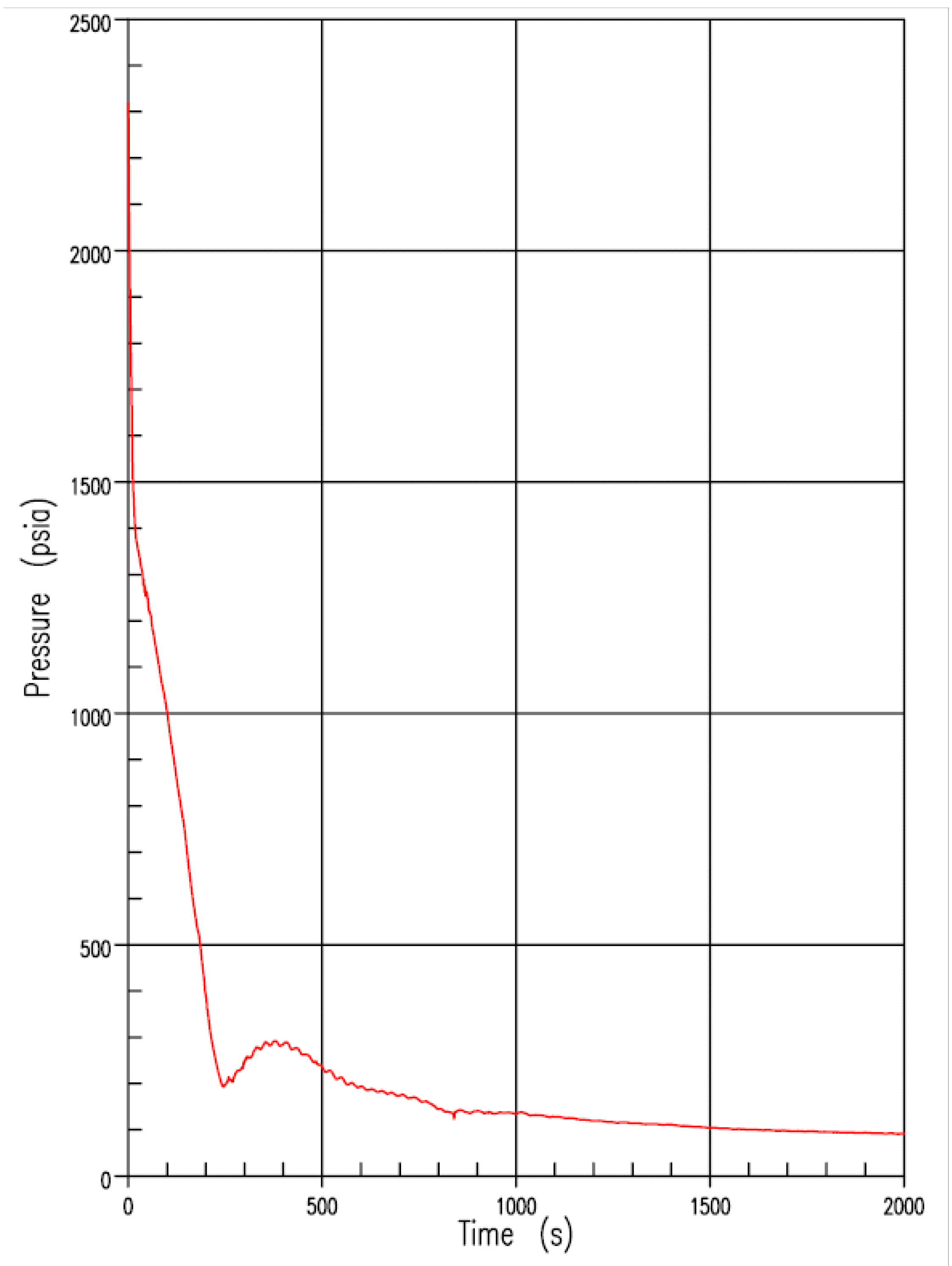


Figure 15.3-14 Reactor Coolant System Pressure For 8.75-Inch Break

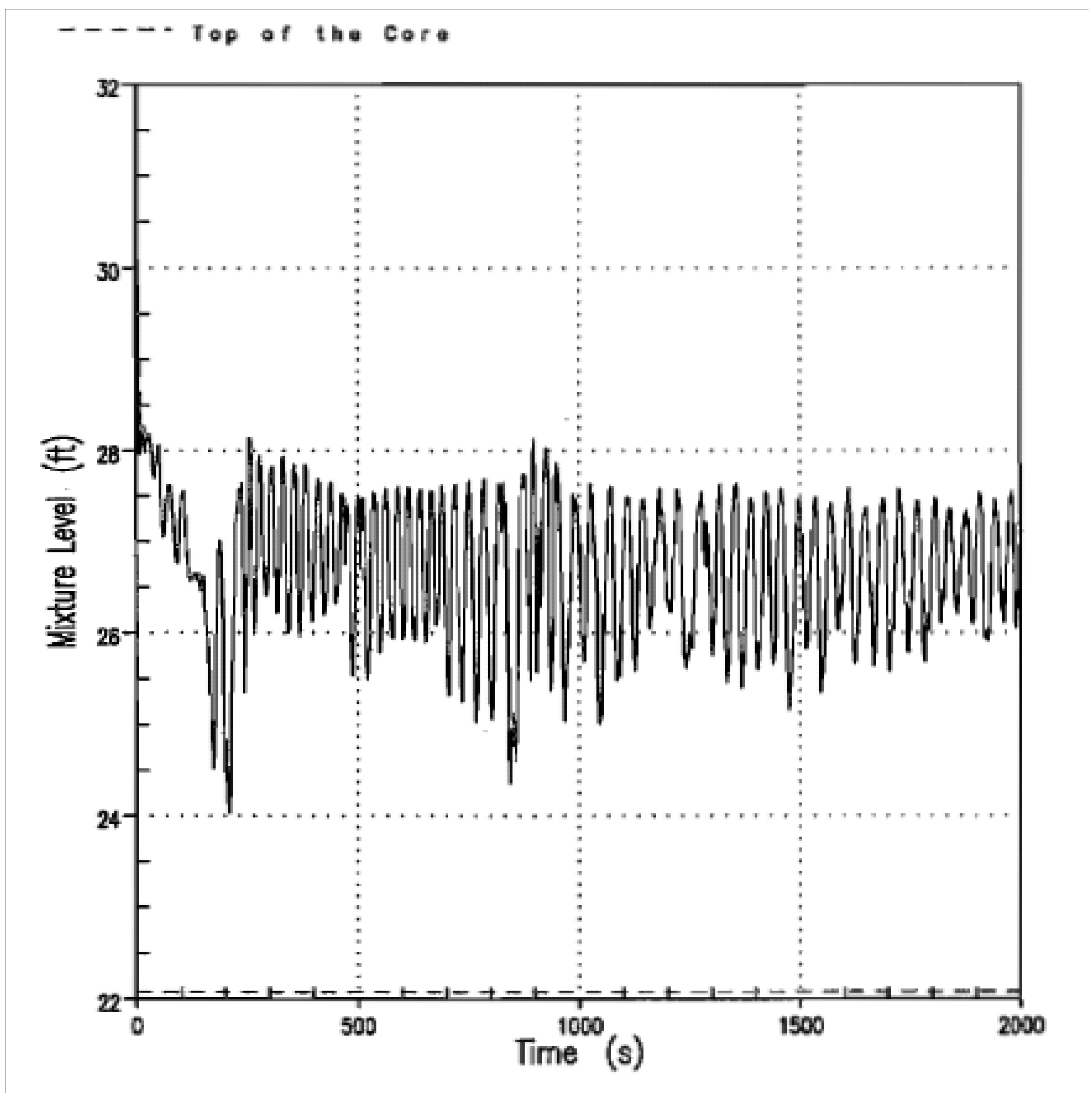


Figure 15.3-14a Core Mixture Level Transient For 8.75-Inch Break

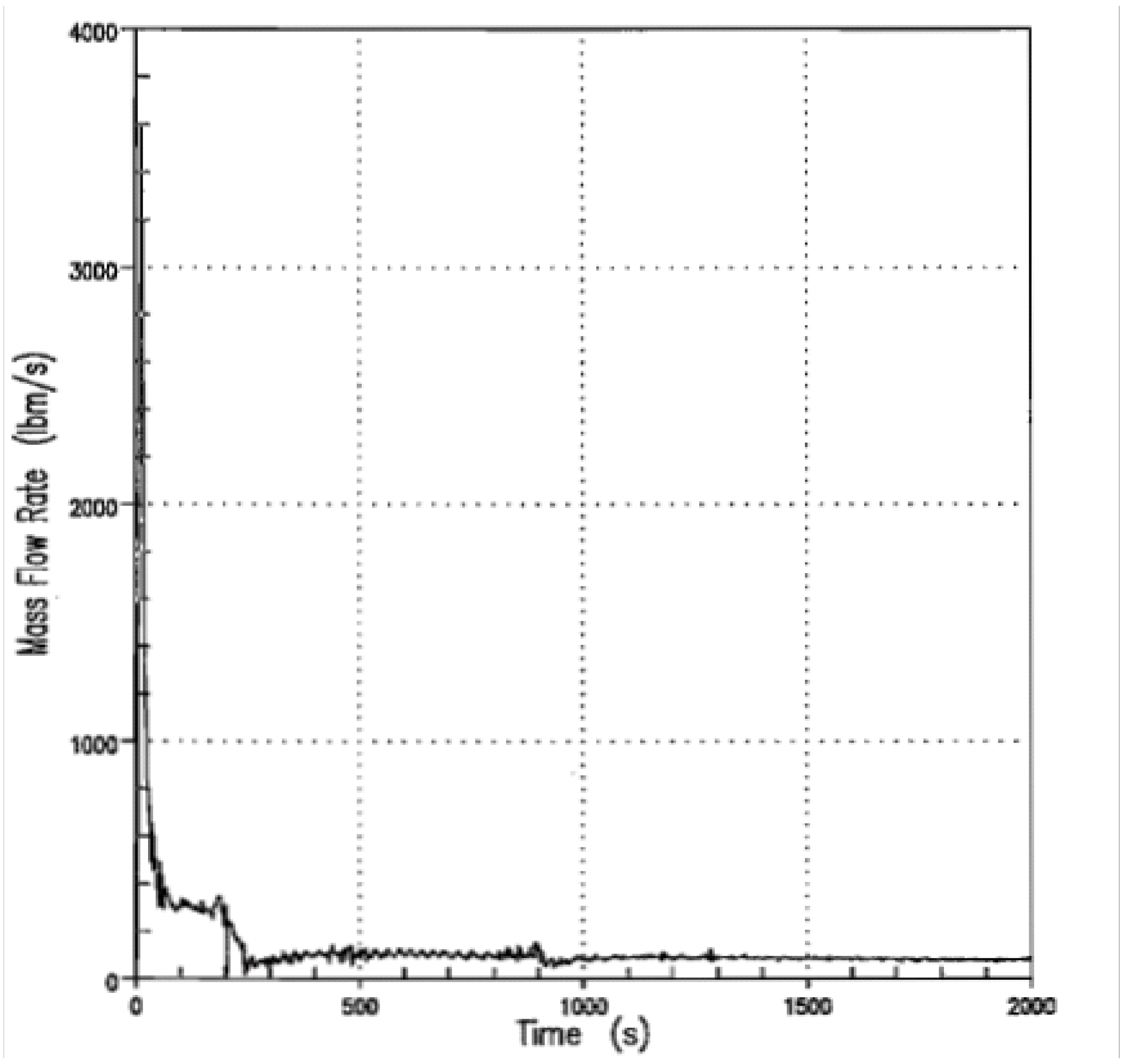


Figure 15.3-14b Core Outlet Steam Flow Rate For 8.75-Inch Break

4026-385

	R	P	N	M	L	K	J	H	G	F	E	D	C	B	A
1							-8.9			-7.4					
2			-5.6			-9.1		-8.5							
3								-8.2		-6.8		-4.1		0.2	
4		-7.9	-8.2					-7.7							
5					-8.4				-6.0		-3.8		-1.8		
6	-8.5		-8.4			-7.4		-5.5						-0.3	
7				-7.7			-5.0			-1.2			-1.0		
8	-7.7		-7.3		-5.9		-3.2			1.5		3.2	3.4	3.6	
9		-6.9							2.7		5.9				6.0
10					-3.4		0.7					10.6			
11	-5.3				-1.8			5.9			17.1				11.4
12						1.3			12.3			24.6			
13			0.1		0.7			7.7						23.6	
14			2.5				4.7			11.1		17.6			
15					2.1			6.5							

CASE A

Figure 15.3-15 Interchange Between Region 1 and Region 3 Assembly

4026-386

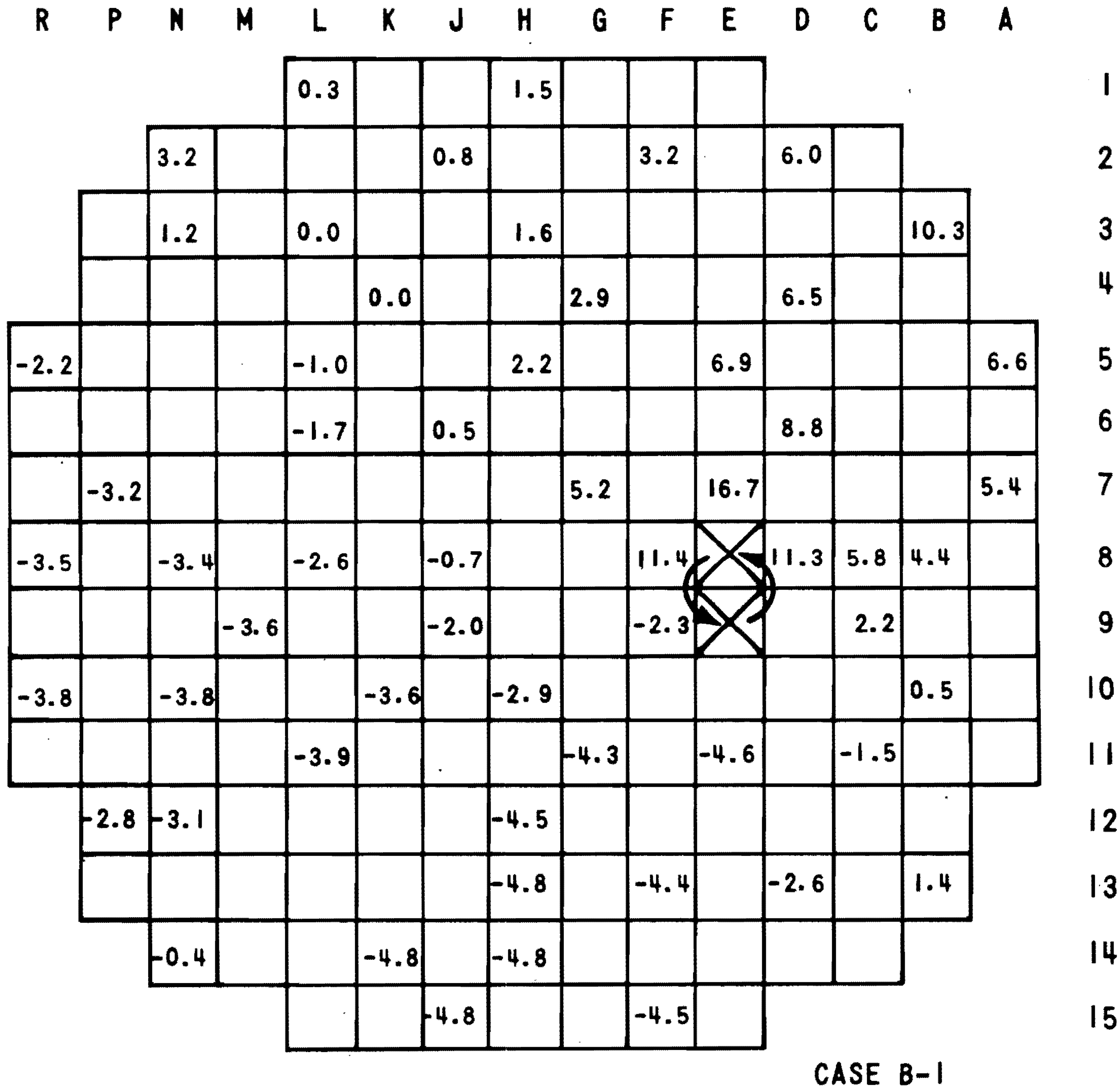


Figure 15.3-16 Interchange Between Region 1 and Region 2 Assembly, Burnable Poison Rods Being Retained by the Region 2 Assembly

Figure 15.3-16 Interchange Between Region 1 and Region 2 Assembly, Burnable Poison Rods Being Retained by the Region 2 Assembly

4026-387

	R	P	N	M	L	K	J	H	G	F	E	D	C	B	A
1							1.0			1.1					
2			5.1			1.0		1.0							
3								1.1		1.1		1.9		4.9	
4		1.7	1.7					1.4							
5					1.1				1.8		1.1		0.7		
6	0.0		0.2			1.8		3.9						4.0	
7				0.0			5.2			2.2			-0.3		
8	-0.7		-0.6		0.3		5.1			1.5		-0.3	-0.6	-0.7	
9		-1.0							-1.1		-0.8				-0.9
10					-1.4		-3.1					-1.3			
11	-0.9				-1.7						-1.7				-0.9
12						-2.5			-2.9			-1.1			
13			0.7		-1.9			-2.9						2.5	
14			2.3				-2.8			-2.4		-0.8			
15					-2.1			-2.8							

CASE B-2

Figure 15.3-17 Interchange Between Region 1 and Region 2 Assembly, Burnable Poison Rods Being Transferred to the Region 1 Assembly

4026-388

	R	P	N	M	L	K	J	H	G	F	E	D	C	B	A
1							-2.2			-2.1					
2			2.0			-2.0		-2.1							
3								-1.5		-1.6		-1.0		2.0	
4		-0.9	-1.0					-0.4							
5					-0.4				1.2		-0.5		-1.4		
6	-2.1		-1.6			2.3		5.7						-2.0	
7				-3.2			9.7			4.4			-1.7		
8	-2.3		-1.6		1.8		13.6	X		5.6		-0.4	-1.6	-2.1	
9		-2.2							9.7		1.1				-2.2
10					0.3		4.5					-0.9			
11	-1.9				-0.4			1.8			-0.5				-1.9
12						-0.9			-0.6			-1.1			
13			0.4		-1.4			-1.5						2.0	
14			2.0				-2.1			-2.0		-0.9			
15					-1.9			-2.2							

CASE C

Figure 15.3-18 Enrichment Error: A Region 2 Assembly Loaded into the Core Central Position

4026-389

	R	P	N	M	L	K	J	H	G	F	E	D	C	B	A
1							-11			-14					
2			0.4			-9.2		-12							
3								-12		-14		-15		-13	
4		3.2	1.2					-11							
5					-1.5				-12		-15		-16		
6	9.8		7.1			-1.6		-8.0						-16	
7				9.2			-2.3			-12			-14		
8	20.0		17.8		10.8		0.8			-10		-14	-15	-16	
9		27.2							-5.5		-11				-15
10					20.7		5.8					-12			
11	42.0		X		23.6			1.9			-8.6				-13
12						14.0			-1.7			-8.9			
13			38.6		20.4			2.8						-7.0	
14			35.9				7.0			-3.3		-6.3			
15					15.3			2.9							

CASE D

Figure 15.3-19 Loading a Region 2 Assembly into a Region 1 Position Near Core Periphery

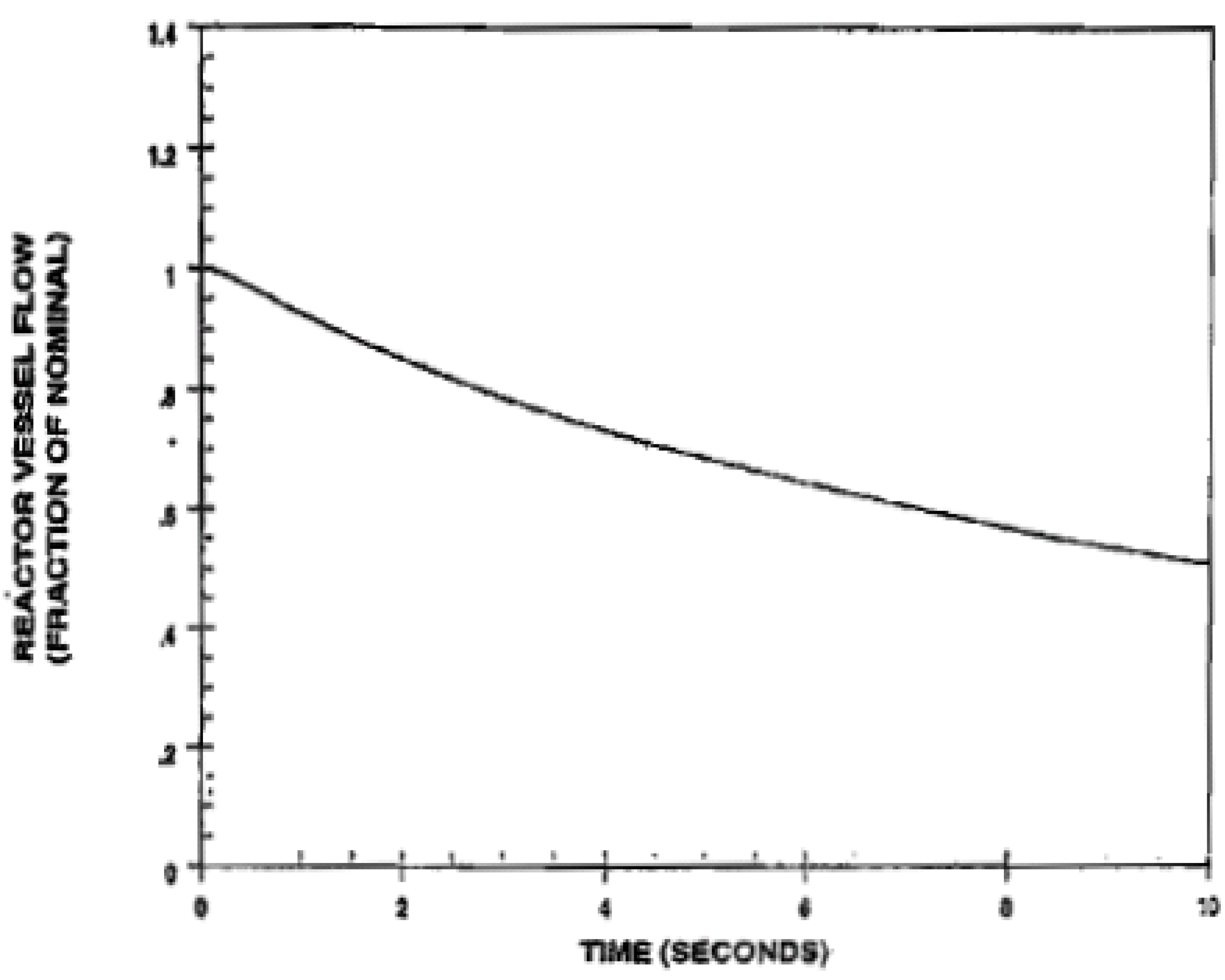


Figure 15.3-20 Reactor Vessel Flow Transient Complete Loss of Flow - Undervoltage Four Pumps in Operation, Four Pumps Coasting Down

Figure 15.3-21 Deleted by Amendment 80

Figure 15.3-22 Deleted by Amendment 97

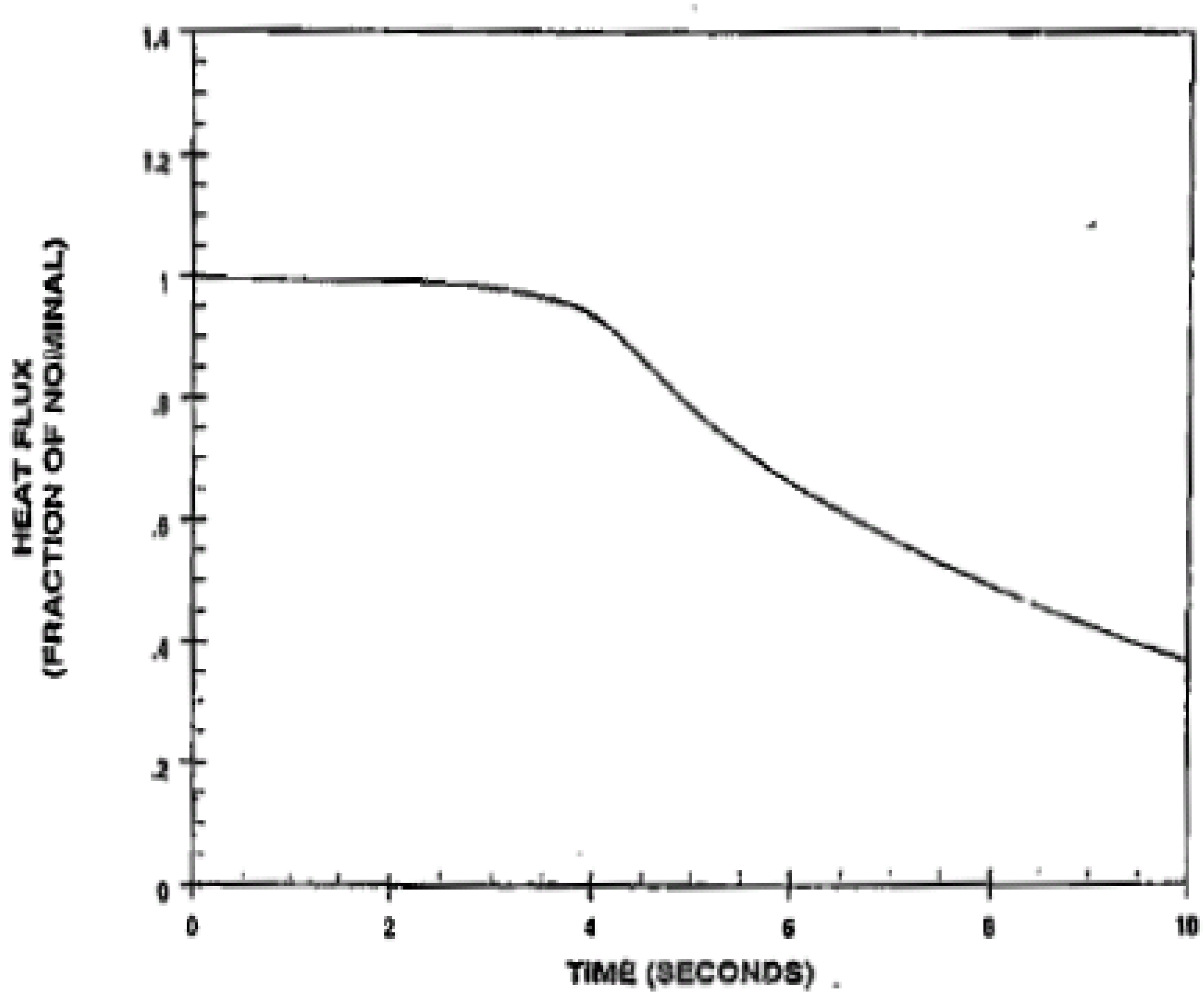


Figure 15.3-23 Hot Channel Heat Flux Transient Complete Loss Of Flow-Undervoltage; Four Pumps in Operation, Four Pumps Coasting Down

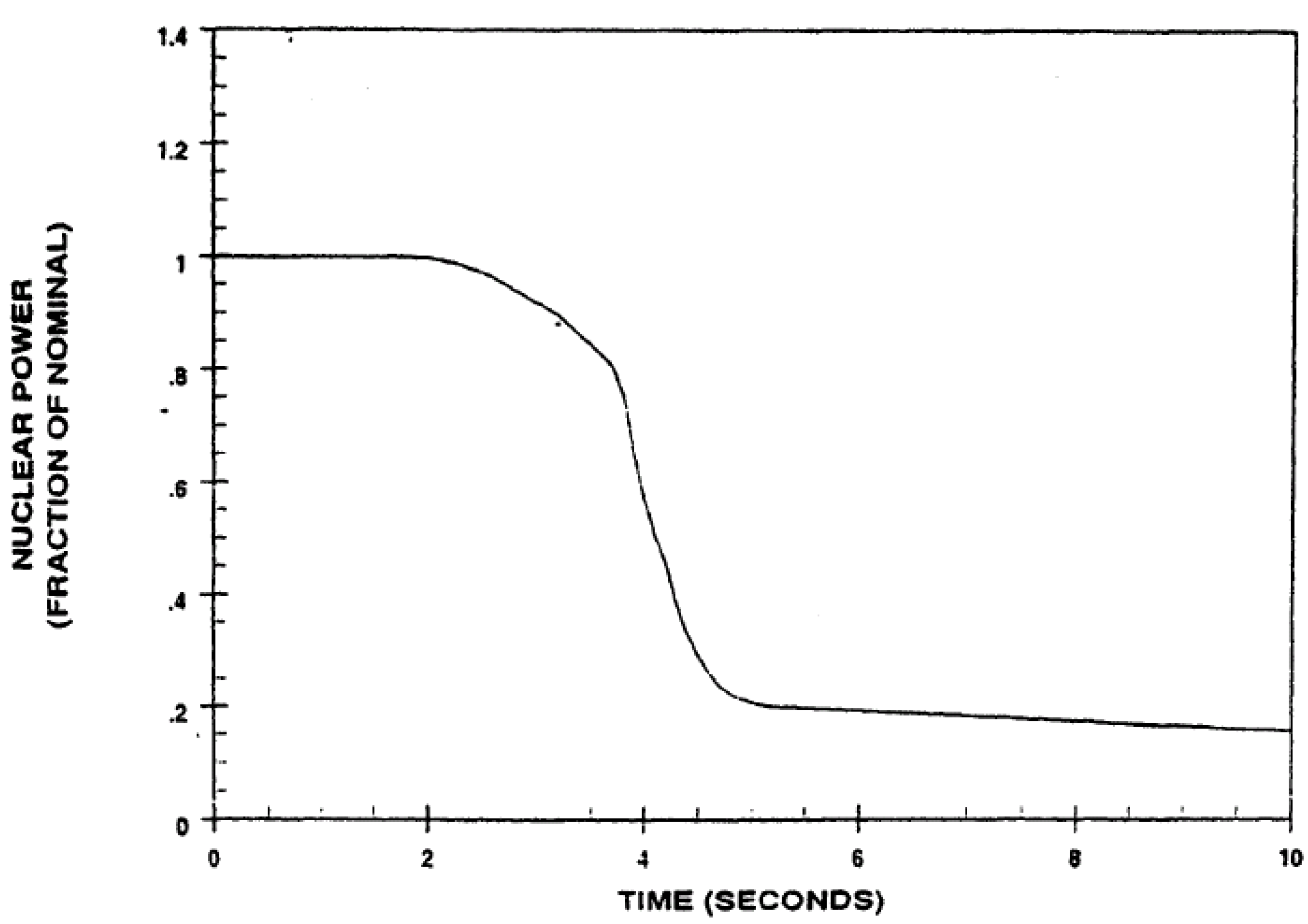


Figure 15.3-24 Nuclear Power Transient Complete Loss Of Flow-Undervoltage; Four Pumps in Operation, Four Pumps Coasting Down

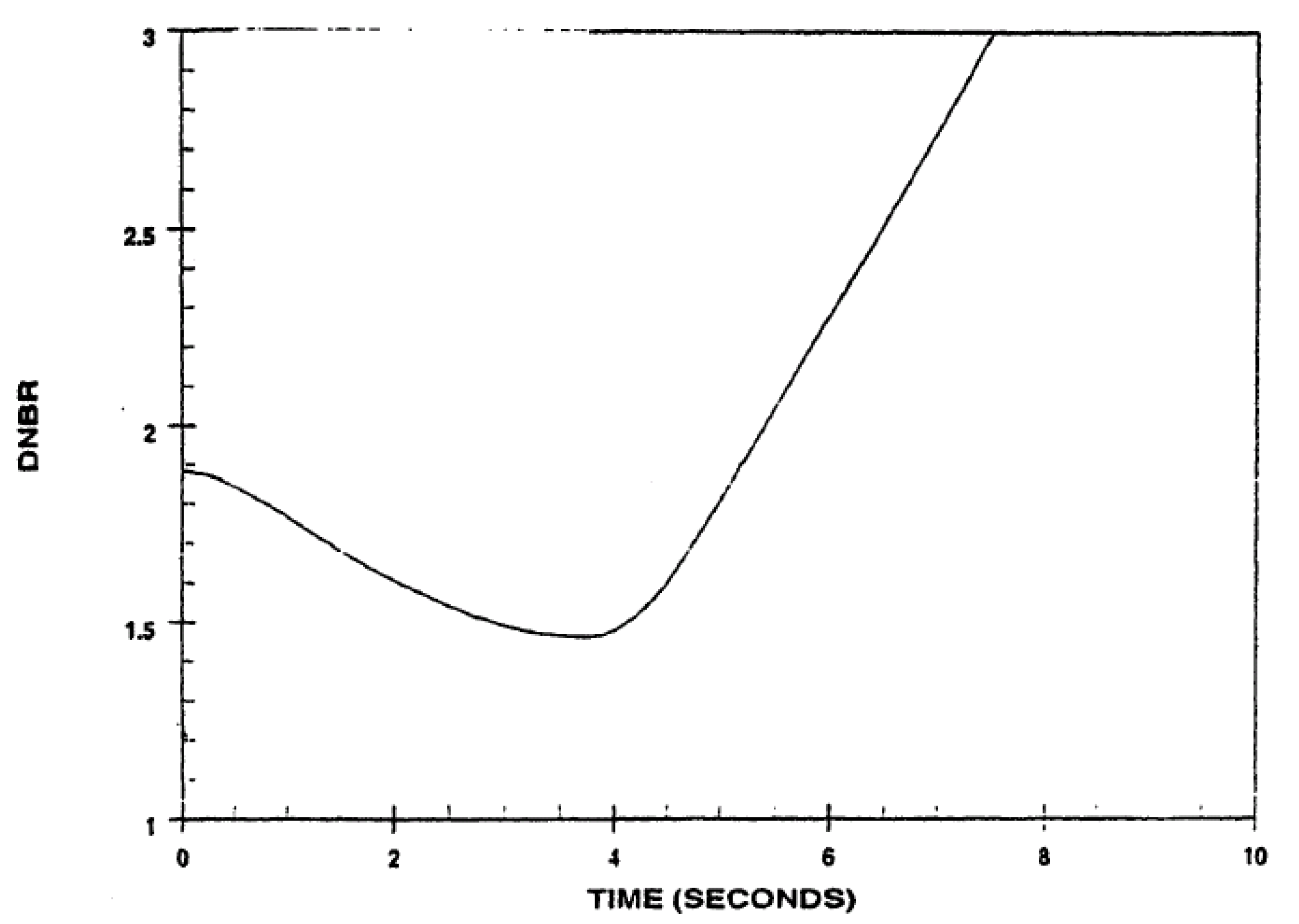


Figure 15.3-25 DNBR Versus TimeComplete Loss of Flow-Undervoltage Four Pumps in Operation, Four Pumps Coasting Down

Figure 15.3-26 Deleted by Amendment 97

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