Seabrook Station

North Atlantic

Updated Final Safety Analysis Report

Revision 7

CHAPTER 15

ACCIDENT ANALYSES

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15.0.3.3 Power Distribution

The power distribution in the core, and in particular, the radial peaking factor ($F_{\Delta}\Delta H$) and the total peaking factor (F_q), are of major importance in determining the transient margin. Initial power distributions for the transients are selected from a range of possible conditions within the allowable axial flux difference LCO band. Such a band, corresponding to Wide-band operation for Seabrook Station, is illustrated in Figure 15.0-32. Power distributions used to generate the axial flux difference LCO band consider both steady-state operation and xenon transients. Details of the method for treating power distributions are provided in YAEC-1854P, Reference 13.

The radial peaking factor ($F_{\Delta H}$), the total peaking factor (F_q), and the axial flux difference LCO band are controlled through the COLR. Transient power peaking involving rod motion or rod misalignment is explicitly treated on an event-by-event basis.

15.0.3.4 Component Response Times and Capacities

A tabulation of the component response-time and design capacities, as assumed for the various accidents, is presented in Tables 15.0-7 and 15.0-8.

15.0.3.5 Non-LOCA Accidents

This section summarizes the non-LOCA analyses and evaluations performed to support the V5H (w/IFMs) implementation at Seabrook Unit 1. The analyses were performed to bound all applicable V5H fuel features (w/IFMs and w/o IFMs).

15.0.3.5.1 Fuel Features

The fuel features which were evaluated are:

a. Zirc-4 Intermediate Flow Mixing (IFMs) grids;
b. Zirc-4 mid grids;
c. ZIRLO$_{TM}$ fuel clad;
d. ZIRLO$_{TM}$ instrument and thimble tubes;
e. Removable top nozzles;
f. Protective bottom grids;
g. Debris filter bottom nozzles

15.0.3.5.2 Other Major Assumptions

a. An NSSS power level of 3429.4 MWt
b. A reactor thermal power of 3411 MWt
c. A reactor coolant system Thermal Design Flow (TDF) of 95,700 gpm/loop
d. A reactor coolant system Minimum Measured Flow (MMF) of 98,200 gpm/loop

e. An average vessel average coolant temperature of 588.5°F
f. An average reactor coolant pressure of 2250 psia
g. An average steam generator tube plugging (SGTP) of 8%

For most accidents which are DNB-limited, nominal values of the initial conditions are assumed. The uncertainty allowances on power, temperature, pressure, and RCS flow are included on a statistical basis and are included in the limit DNBR value by using the Revised Thermal Design Procedure (RTDP)[19].

For accidents analyses which are not DNB limited, or for which RTDP is not employed, the initial conditions are obtained by applying the maximum steady-state errors to rated values. The following steady-state errors are considered in the analyses:

a. For reactor power, a ± 2%
b. For average RCS temperature, a ± 5.8°F
c. For pressurizer pressure, a ± 50 psi

Accidents employing RTDP assume a Minimum Measured Flow (MMF), while others assume the Thermal Design Flow (TDF). In addition to being the flow used in the DNB analysis for RTDP methodology, the MMF is bounded by the Tech Specs minimum flow measurement requirement. The MMF includes allowance for plant flow measurement uncertainty.

15.0.3.5.3 Overtemperature-ΔT and Overpower-ΔT

The overtemperature-ΔT and overpower-ΔT setpoints were recalculated for the V5H (w/ IFMs) fuel upgrade program based on the most conservative core limits. These core limits were used to bound mixed cores and full cores with the V5H (w/ IFMs) features analyzed. The core limits used to calculate the OT/OPΔT setpoints are provided in Tech Spec Figure 2.1-1. All of the FSAR events which rely on OTΔT and OPΔT for protection were analyzed to reflect the setpoint changes, as provided in the revised COLR. It has been confirmed that these OTΔT and OPΔT setpoints protect the core safety limits as shown in Figure 5.0-1.

15.0.3.5.4 RCCA Reactivity Characteristics

The negative reactivity insertion following a reactor trip is a function of the acceleration of the RCCAs and the variation in rod worth as a function of rod position. With respect to the accident analyses, the critical parameter is the time from beginning of RCCA insertion to dashpot entry, or approximately 85% of the RCCA travel. For the accident analyses, the insertion time from fully withdrawn to dashpot entry remains at the Tech Spec limit of 2.4 seconds from the beginning of stationary gripper coil voltage decay.

While the rod drop time remains at 2.4 seconds from fully withdrawn to the dashpot, the reactivity worth has changed to reflect the fuel upgrade. The normalized RCCA position (fraction insertion) versus the

normalized time from release is presented in Figure 15.0-4. The reactivity worth versus rod insertion (fraction) assumed in the safety analyses is shown in Figure 15.0-5.

For analyses requiring the use of a dimensional diffusion theory code, the negative reactivity insertion resulting from the reactor trip is calculated directly by the reactor kinetic code and is not separable from other reactivity feedback effects. In this case, the RCCA position versus time of Figure 15.0-4 is used.

15.0.4 Reactivity Coefficients Assumed in the Accident Analyses

The transient response of the RCS is dependent on reactivity feedback effects, in particular the moderator density coefficient and the Doppler Power Coefficient (DPC). Depending upon event specific characteristics, conservatism dictates use of either large or small reactivity coefficient values. Justification for the use of the reactivity coefficient values is treated on an event-specific basis.

Maximum and minimum integrated DPCs assumed in the safety analyses are provided in Figure 15.0-2. The formulas for calculating the DPCs used are $[(.034Q^2) -19.4Q]x10^{-5}$ for the maximum and $[(.0175Q^2)-9.55Q]x10^{-5}$ for the minimum, where Q is the power level. Note that Steamline Break Core Response uses a different DPC based on a stuck RCCA.

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. the values used for each accident are given in Table 15.0-3. Conservative combinations of parameters are used for each event selected on a case-by-case basis.

15.0.5 Rod Cluster Control Assembly Insertion Characteristics

The negative reactivity insertion following a reactor trip is a function of the acceleration of the Rod Cluster Control Assemblies and the variation in rod worth as a function of rod position. Another critical parameter is the time of insertion up to the dashpot entry, or approximately 85 percent of the rod cluster travel. For accident analyses, the insertion time to dashpot entry is conservatively taken as 2.4 seconds. The rod Cluster Control Assembly position versus time assumed in accident analyses is shown in Figure 15.0-4.

Figure 15.0-5 illustrates the fraction of total negative reactivity insertion versus normalized rod position for a core where the axial power distribution is skewed to the bottom. This curve is used to compute the negative reactivity insertion versus time following a reactor trip, for the majority of cases presented in Chapter 15.

There is inherent conservatism in the use of Figure 15.0-5, particularly for DNB related events which are typically limiting for top skewed power distributions. For DNB related events a curve based on a slightly bottom skewed shape was used.

The normalized Rod Cluster Control Assembly negative reactivity insertion versus time is shown in Figure 15.0-6. The curve shown in this figure was obtained from Figures 15.0-4 and 15.0-5. Transient analyses performed with less conservative yet still bounding scram curves are specifically identified in subsequent sections. A total negative reactivity insertion following a trip of 4 percent ΔK is assumed in the transient analyses except where specifically noted otherwise. This assumption is conservative with respect to the calculated trip reactivity worth available. For Figures 15.0-4 and 15.0-5, the rod cluster control assembly drop time is normalized to 2.4 seconds.

15.0.6 Trip Points and Time Delays to Trip Assumed in Accident Analyses

A reactor trip signal acts to open two trip breakers connected in series feeding power to the control rod drive mechanisms. Opening either trip breaker initiates a turbine trip. 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 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 at the sensor to the time the rods are free and begin to fall. Limiting trip setpoints assumed in accident analyses and the total time delay assumed for each trip function are given in Table 15.0-4. The Overtemperature ΔT trip functions are illustrated in Figure 15.0-1.

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

15.0.9.2 Activities in the Fuel Pellet Cladding Gap

The fuel-clad gap activities were determined using the model given in Regulatory Guide 1.77. Thus, the amount of activity accumulated in the fuel-clad gap is assumed to be 10 percent of the iodines and 10 percent of the noble gases accumulated at the end of core life. The gap activities are given in Table 15.0-6.

15.0.9.3 Activities in the Secondary Side Coolant

See Table 11.1-4 for secondary side coolant activities.

15.0.10 Residual Decay Heat

15.0.10.1 Total Residual Heat

Residual heat in a subcritical core is calculated for the loss-of-coolant accident per the requirements of Appendix K of 10 CFR 50.46 (Reference 4), as described in References 5 and 6. These requirements include assuming infinite irradiation time before the core goes subcritical to determine fission product decay energy. For all other accidents, the same models are used except that fission product decay energy is based on core average exposure at the end of the equilibrium cycle.

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

During a loss-of-coolant accident the core is rapidly shut down by void formation or Rod Cluster Control Assembly insertion, or both, 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 percent which represents the fraction of heat generated within the clad and pellet drops to 95 percent for the hot rod in a loss-of-coolant accident.

For example, consider the transient resulting from the postulated double-ended break of the largest reactor coolant system pipe; 1/2 second after the rupture about 30 percent of the heat generated in the fuel rods is from gamma ray absorption. The gamma power shape is less peaked than the steady-state fission power shape, reducing the energy deposited in the hot rod at the expense of adjacent colder rods. A conservative estimate of this effect is a reduction of 10 percent of the gamma ray contribution or 3 percent of the total. Since the water density is considerably reduced at this time, an average of 98 percent of the available heat is deposited in the fuel rods, the remaining 2 percent 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.

15.0.11 Computer Codes Utilized

Summaries of some of the principal computer codes used in transient analyses are given below. Other codes, in particular very specialized codes in which the modeling has been developed to simulate one given accident, such as those used in the analysis of the reactor coolant system pipe rupture (Section 15.6), are summarized in their respective accident analyses sections. The codes used in the analyses of each transient have been listed in Table 15.0-3.

15.0.11.1 CHICK-KIN

CHICK-KIN calculates the transient temperature distribution in a cross section of a metal clad UO_2 fuel rod and the transient heat flux at the surface of the 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:

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

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

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

CHIC-KIN is further discussed in Reference 7.

15.0.11.2 LOFTRAN

Transient response studies of a Pressurized Water Reactor (PWR) to specified perturbations in process parameters use the LOFTRAN[8] program. The LOFTRAN program models all four reactor coolant loops. This code simulates a multi-loop system by a model containing the reactor vessel, hot and cold leg piping, steam generators (tube and shell sides), the pressurizer and the pressurizer heaters, spray, relief valves, and safety valves. LOFTRAN also includes a point neutron kinetics model and reactivity effects of the moderator, fuel, boron, and rods. The secondary side of the steam generator uses a homogeneous, saturated mixture for the thermal transients and a water level correlation for indication and control.

The code simulates the Reactor Protection System (RPS) which includes reactor trips on high neutron flux, $OT_{\Delta}T$, $OP_{\Delta}T$, high and low pressurizer pressure, low flow, and high pressurizer level. Control systems are also simulated including rod control, steam dump, feedwater control, and pressurizer pressure control. The safety injection system (SIS), including the accumulators, is also modeled. LOFTRAN also has the capability of calculating transient values of DNBR based on the input from the core limits.

15.0.11.3 RETRAN

The RETRAN program is used for studies of transient response of a PWR system to specified perturbations in process parameters. RETRAN simulates a multi-loop system by a model containing reactor vessel, hot and cold leg piping, steam generator (tube and shell sides) and the pressurizer. The pressurizer heaters, spray, relief and safety valves are also considered in the program. Point model neutron kinetics and reactivity effects of the moderator, fuel, boron and rods are included. The secondary side of the steam generator utilizes a homogeneous, saturated mixture for the thermal transients and a water level correlation for indication and control. The Reactor Protection System is simulated to include reactor trips on high neutron flux, Overtemperature ΔT. Overpower ΔT, high and low pressure, low flow, and high pressurizer level. Control systems are also simulated including rod control, steam dump, feedwater control, and pressurizer pressure and level control. The Emergency Feedwater and Emergency Core Cooling System (except accumulators) are also modeled.

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

RETRAN is further discussed in Reference 10.

15.0.11.4 STAR

The STAR program is a three-dimensional nodal reactor static and kinetic code. The code uses the analytic method to solve for the two-group neutron fluxes in static and kinetic cases. The code uses six delayed neutron groups and uses the VIPRE-01 thermal-hydraulic model to obtain coolant conditions and fuel temperatures. The code accepts basic geometry data, cross section data from the SIMULATE-3 code, and perturbations in core pressure, inlet temperature, in let flow, control position, and boron concentration. Edits can be obtained of core, assembly, and node powers, nodal temperatures, and coolant densities. The STAR Code is used to predict the kinetic behavior of the reactor for transients which cause a major perturbation in the spatial neutron flux distribution (Rod Cluster control assembly Ejection, and Steam System Piping Failure with stuck RCCA). See Reference 9.

15.0.11.5 SIMULATE-3 and CASMO-3

SIMULATE-3 is a two group, advanced nodal code, capable of determining detailed pin by pin power distributions for steady state and xenon transient conditions. All cross section data for SIMULATE-3 is given by CASMO-3 infinite lattice calculation. CASMO-3 uses neutron transport methods in forty neutron groups and collapses the results into two neutron group cross sections and discontinuity factors. Both codes have been extensively benchmarked and proven accurate in current safety analysis calculations performed by Yankee Atomic Electric Company and other utilities. Generic approval of both codes for this type of work was granted in YAEC-1363-A for CASMO-3 and YAEC-1659-A for SIMULATE-3.

Power distributions and local peaking factors are obtained from SIMULATE-3 calculations. Core conditions such as: control rod position, power level, and other parameters, are explicitly modelled within SIMULATE-3. The code uses the plant operating history, cross sections from CASMO-3, core conditions and control rod position to start the neutronic calculations. An industry standard advanced nodal technique is used to determine the incore flux and power distribution for each of nearly 20,000 nodes. Each node is defined as a quarter of an assembly in the radial direction and six inches in the axial direction. SIMULATE-3 has pin power reconstruction capabilities which will determine the power of each pin within each node.

SIMULATE-3 is further described in Reference 11.

15.0.11.6 VIPRE

The VIPRE computer program performs thermal-hydraulic calculations. This code calculates coolant density, mass velocity, enthalpy, void fractions, static pressure and DNBR distributions along flow channels within a reactor core. The VIPRE code is described in References 12 and 18, and Section 4.4.

15.0.11.7 FACTRAN

FACTRAN[16] calculates the transient temperature distribution in a cross-section of a metal clad UO_2 fuel rod and the transient heat flux at the surface of the clad, using as input the nuclear power and the time-dependent coolant parameters of pressure, flow, temperature and density. The code uses a fuel model which simultaneously contains the following features:

a. A sufficiently large number of radial space increments to handle fast transients such as a rod ejection accident;
b. Material properties which are functions of temperature and a sophisticated fuel-to-clad gap heat transfer calculation; and
c. The necessary calculations to handle post-DNB transients: film boiling heat transfer correlations, Zircaloy-water reaction, and partial melting of the fuel.

15.0.11.8 TWINKLE

The TWINKLE[17] program is a multi-dimensional spatial neutron kinetics code. 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 2,000 spatial points and performs 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. The code provides various output, e.g., channelwise power, axial offset, enthalpy, volumetric surge, pointwise power and fuel temperatures. It also predicts the kinetic behavior of a reactor for transients which cause a major perturbation in the spatial neutron flux distribution.

15.0.12 Radiological Consequences

Radiological consequences have been calculated for each hypothetical accident which can potentially result in radioactivity releases in excess of those expected to be experienced during normal plant operating conditions. In general, two hour thyroid doses, whole body gamma and beta-gamma skin doses are presented at the 914 meter site boundary and duration of accident doses for the outer boundary of the low-population zone (2012 meters). Parameters and assumptions used to evaluate the radiological consequences for both conservative and realistic cases are presented in the following discussions of each hypothetical accident, and are summarized in Appendices 15A and 15B.

The physical and mathematical models used in calculating radioactivity source terms are discussed in Section 11.1. Core fission products (halogens and noble gases) used to calculate accident doses are given in Table 15.0-6.

The radioactive fission product source terms are determined for the fuel, fuel rod gap and reactor coolant for full power operation at 3654 MWt core thermal power as discussed in Appendix 15B.

The effect of V5H (w/ IFMs) fuel upgrade implementation on each of the Seabrook Non-LOCA FSAR transients were evaluated or analyzed. These transient evaluations and analyses demonstrate that all applicable safety analysis acceptance criteria continue to be met for the intended V5H (w/ IFMs) fuel upgrade implementation at Seabrook Unit 1.

The hypothetical accident analyses show that the radiological consequences result in no offsite consequences, are bounded by radiological consequences calculated for other related accidents, or are below the guideline values of 10 CFR 100. Therefore, it is concluded that the Seabrook plant, Units 1 and 2, have been adequately designed to mitigate the potential radiological consequences of postulated accidents, and that they do not represent an undue hazard to public health and safety.

15.0.13 References

1. DiNunno, J. J., et al., "Calculation of Distance Factors for Power and Test Reactor Sites," TID-14844, March 1962.

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3. "ORIGEN Yields and Cross Sections - Nuclear Transmutation and Decay Data from ENDF/B-IV," RSIC-DLC-38, Radiation Shielding Information Center, Oak Ridge National Library, September 1975.

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6. Not used

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15. Not used

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TABLE 15.0-3
(Sheet 1 of 3)

SUMMARY OF INITIAL CONDITIONS AND COMPUTER CODES USED

FAULTS	COMPUTER CODES UTILIZED	REACTIVITY COEFFICIENTS ASSUMED		THERMAL POWER OUTPUT ASSUMED(b)
		MODERATOR TEMPERATURE	DOPPLER	%
15.1 Increase in Heat Removal by the Secondary System				
- Feedwater System Malfunction Causing an Increase in Feedwater Flow	LOFTRAN, FACTRAN, VIPRE	Most Negative	Least Negative	0 and 100
- Excessive Increase in Secondary Steam Flow	LOFTRAN, VIPRE	Most and Least Negative	Least Negative	100
- Accidental Depressurization of the Main Steam System	Bounded by Steam System Piping Failure Analysis	---	---	---
- Steam System Piping Failure	LOFTRAN, VIPRE,	Most Negative	Most Negative	0
15.2 Decrease in Heat Removal by the Secondary System				
- Loss of External Load and/or Turbine Trip	LOFTRAN, VIPRE	Most and Least Negative	Most and Least Negative	100
- Loss of Nonemergency AC Power to the Station Auxiliaries	LOFTRAN	Most Positive	Least Negative	100(a)
- Loss of Normal Feedwater Flow	LOFTRAN	Most Positive	Least Negative	100(a)
- Feedwater System Pipe Break	LOFTRAN	Most Positive	Least Negative	100(a)

TABLE 15.0-3
(Sheet 2 of 3)

SUMMARY OF INITIAL CONDITIONS AND COMPUTER CODES USED

FAULTS	COMPUTER CODES UTILIZED	REACTIVITY COEFFICIENTS ASSUMED: MODERATOR TEMPERATURE	REACTIVITY COEFFICIENTS ASSUMED: DOPPLER	THERMAL POWER OUTPUT ASSUMED(b) %
15.3 Decrease in Reactor Coolant System Flow Rate				
- Partial and Complete Loss of Forced Reactor Coolant Flow	LOFTRAN, FACTRAN, VIPRE	Most Positive	Most and least negative	100
- Reactor Coolant Pump Shaft Seizure (Locked Rotor)	LOFTRAN, FACTRAN, VIPRE	Most Positive	Least Negative	100
15.4 Reactivity and Power Distribution Anomalies				
- Uncontrolled Rod Cluster Control Assembly Bank Withdrawal from a Subcritical or Low power Startup Condition	TWINKLE, FACTRAN, VIPRE	Most Positive	Least Negative	0
- Uncontrolled Rod Cluster Control Assembly Bank Withdrawal at Power	LOFTRAN, VIPRE,	Most and Least Negative	Most and Least Negative	10,60,100
- Control Rod Misalignment	VIPRE, LOFTRAN	---	Least Negative	50 to 100
- Chemical and Volume Control System Malfunction that Results in a Decrease in Boron Concentration in the Reactor Coolant	NA	NA	NA	0 and 100

TABLE 15.0-3
(Sheet 3 of 3)

SUMMARY OF INITIAL CONDITIONS AND COMPUTER CODES USED

FAULTS	COMPUTER CODES UTILIZED	REACTIVITY COEFFICIENTS ASSUMED		THERMAL POWER OUTPUT ASSUMED(b)
		MODERATOR TEMPERATURE	DOPPLER	%
- Inadvertent Loading and Operation of a Fuel Assembly in an Improper Position	NA	NA	NA	NA
- Spectrum of Rod Cluster Control Assembly Ejection Accidents	TWINKLE, FACTRAN, LOFTRAN, VIPRE	Predicted values plus uncertainty	Predicted values plus uncertainty	0 and 100
15.5 Increase in Reactor Coolant Inventory				
- Inadvertent Operation of ECCS During Power Operation	LOFTRAN, RETRAN, VIPRE	Figure 15.0-3	Lower (see Figure 15.0-2)	100
15.6 Decrease in Reactor Coolant Inventory				
- Inadvertent Opening of a Pressurizer Safety or Relief Valve	LOFTRAN, VIPRE	Most Positive	Least Negative	100
- Steam Generator Tube Rupture	RETRAN	NA	NA	100
- Loss-of-Coolant Accident Resulting from Spectrum of Postulated Piping Breaks within the Reactor Coolant System	SATAN-VI, WREFLOOD, BASH, BART, COCO, NOTRUMP, LOCTA-IV LOCBART, THRIVE	See Subsection 15.6.5 References	See Subsection 15.6.5 References	100

NOTE:

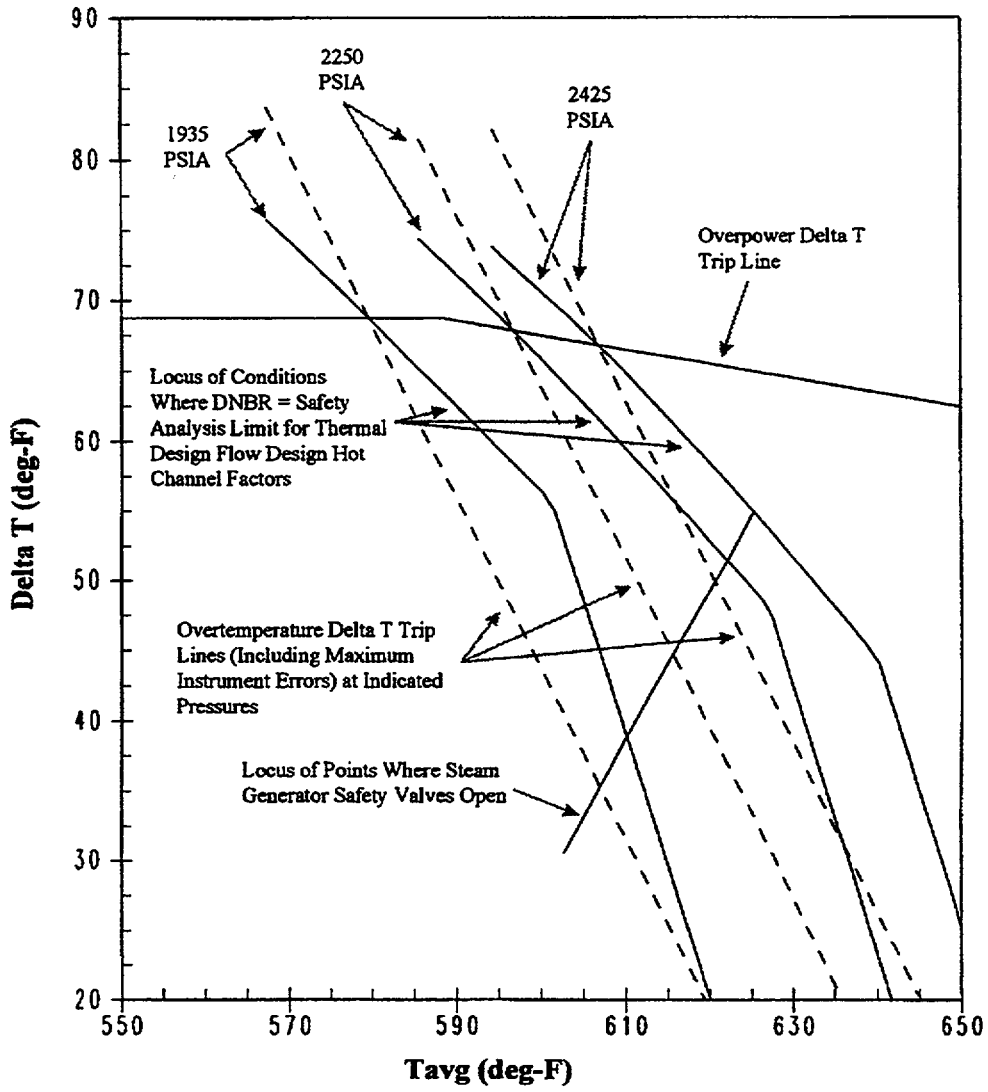
(a) Event is analyzed at the Engineered Safety Features design rating.
(b) The analysis includes a 2% allowance for calorimetric error.

TABLE 15.0-4
(Sheet 1 of 2)

TRIP POINTS AND TIME DELAYS TO TRIP
ASSUMED IN ACCIDENT ANALYSES

Trip Function	Limiting Trip Point Assumed In Analysis	Time Delays (Seconds)
Power Range High Neutron Flux, High Setting	118%	0.5
Power Range High Neutron Flux, Low Setting	35%	0.5
Power Range, Neutron Flux High Negative Rate	6.9%	0.5
Power Range Neutron Flux High Positive Rate	6.9%	0.65
Power Range Neutron Flux, P-8	50%	0.5
Power Range Neutron Flux, P-10	10%	N/A
Overtemperature ΔT	Variable	6.0*
Overpower ΔT	Variable	6.0*
High pressurizer pressure	2425 psia	2.0
Low pressurizer pressure	1935 psia	2.0
Low reactor coolant flow (from loop flow detectors)	87% loop flow	1.0
Undervoltage Trip	70% nominal (9660 volts)	1.5
Underfrequency Trip	55 HERTZ	0.6
Turbine Trip	Not applicable	1.0

* 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.



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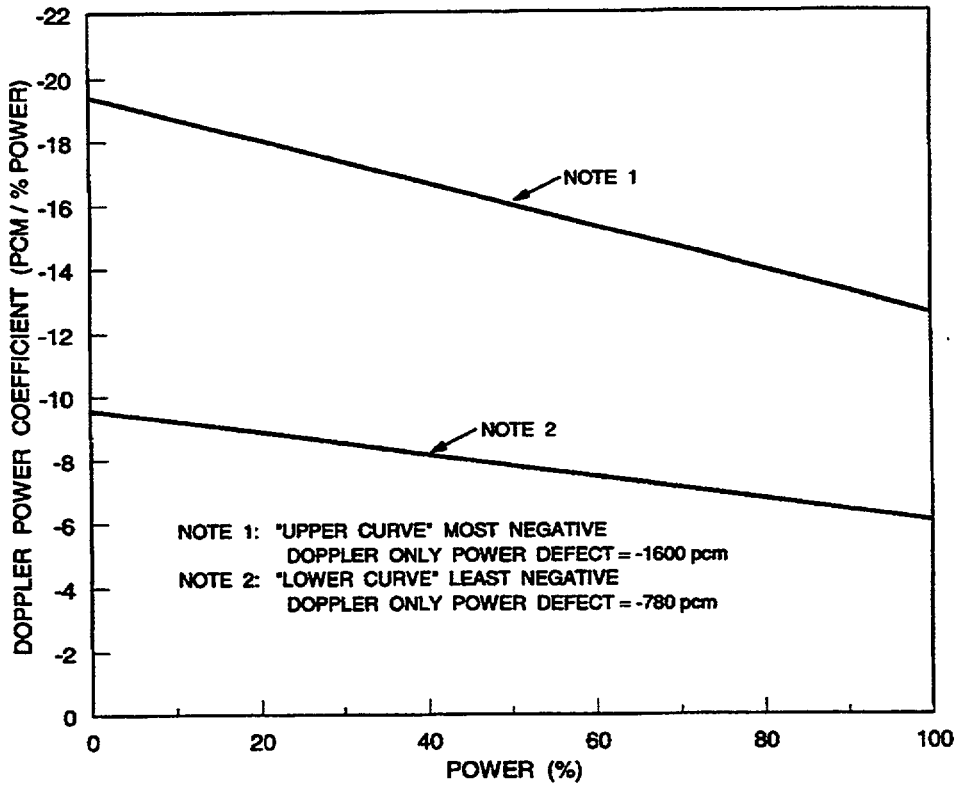
SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Illustration of Overtemperature ΔT and Overpower ΔT Protection	
	REV. 07	FIGURE 15.0-1



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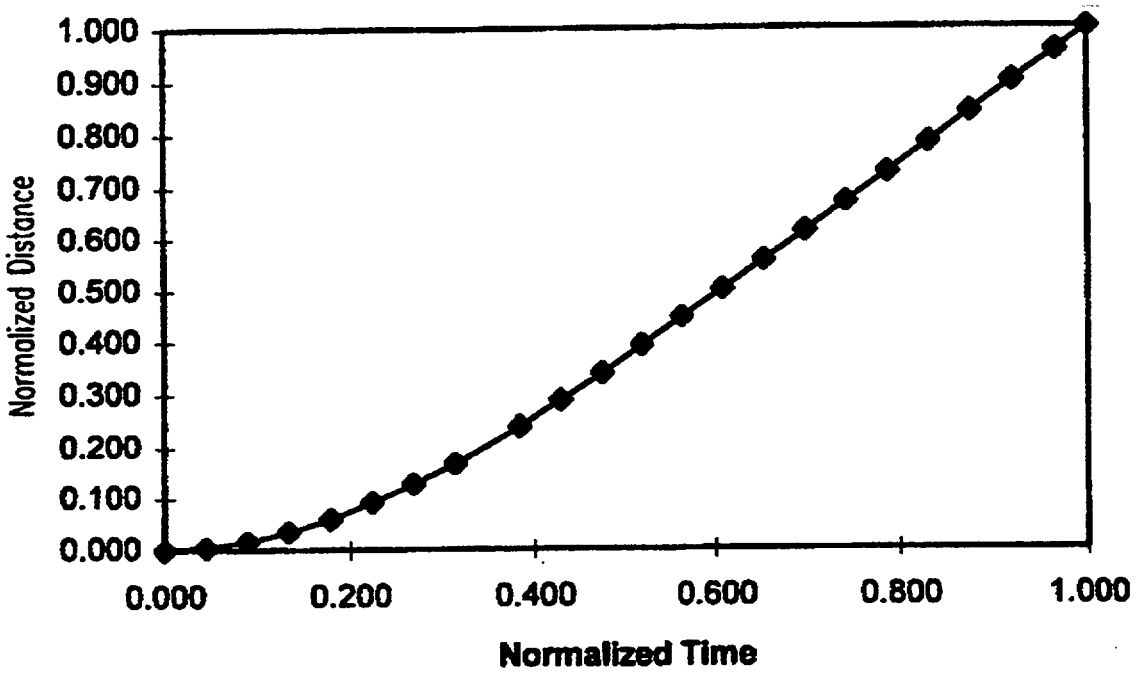
SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Integrated Doppler Power Coefficient Assumed in Analyses	
	REV. 07	FIGURE 15.0-2



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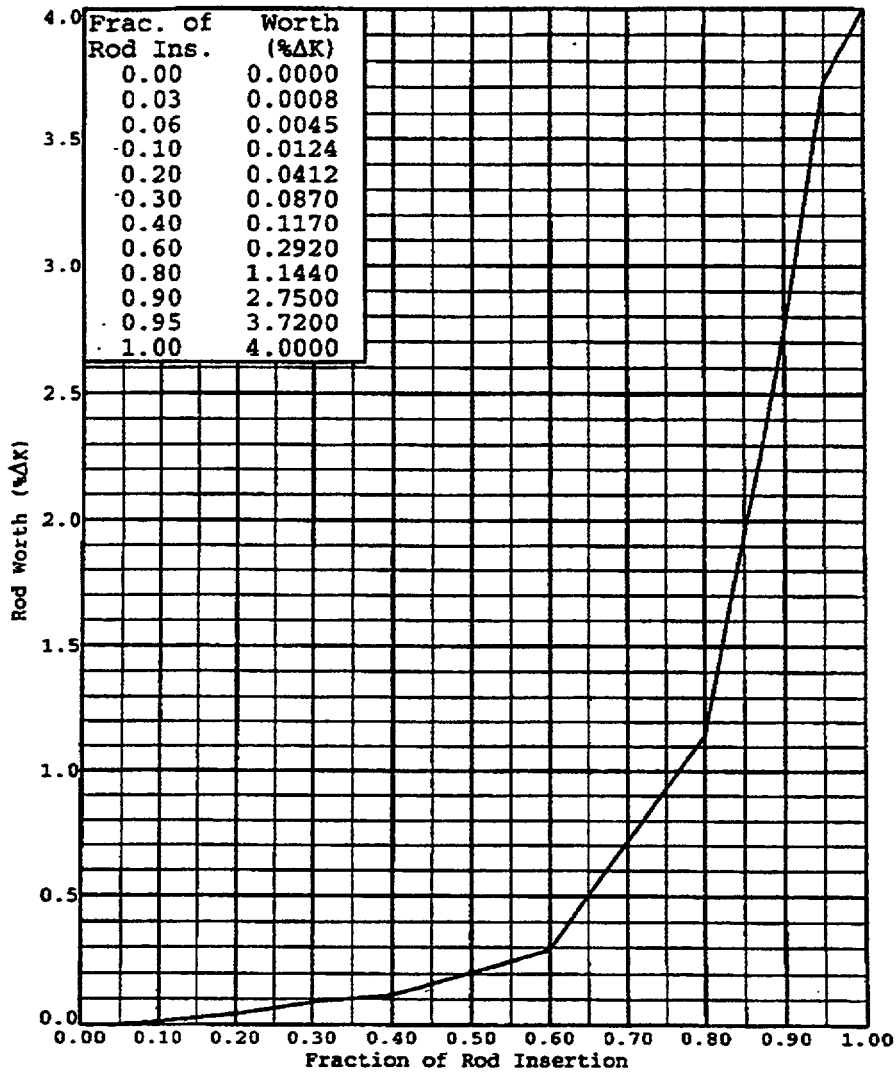
SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Normalized RCCA Position (fraction insertion) vs. Normalized RCCA Drop Time After Release	
	REV. 07	FIGURE 15.0-4



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SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	RCCA Reactivity Worth vs. Rod Insertion (fraction)	
	REV. 07	FIGURE 15.0-5

in Subsection 15.1.3 for an excessive load increase incident, which evaluates the consequences of a 10 percent step load increase. Therefore, the transient results of this analysis are not presented.

15.1.1.3 Radiological Consequences

No radioactivity releases are anticipated as a direct result of this malfunction. Consequently, no radiological consequences are predicted.

15.1.1.4 Conclusions

The decrease in feedwater temperature transient is less severe than the increase in feedwater flow event (Subsection 15.1.2), and the increase in secondary stem flow event (Subsection 15.1.3). Based on results presented in Subsections 15.1.2 and 15.1.3, the applicable acceptance criteria for the decrease in feedwater temperature event have been met.

15.1.2 Feedwater System Malfunctions Causing an Increase in Feedwater Flow

15.1.2.1 Identification of Causes and Accident Description

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

An example of excessive feedwater flow would be a full opening of a feedwater control valve due to a feedwater control system malfunction or an operator error. At power this excess flow causes a greater load demand on the RCS due to increased subcooling in the steam generator. With the plant at no-load conditions, the addition of an excess of 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 activates the feedwater isolation. Pre-trip alarm of high steam generator level is available in the control room.

An increase in normal feedwater flow is classified as an ANS Condition II event, a fault of moderate frequency. See Subsection 15.0.1 for a discussion of ANS Condition II events.

Plant systems and equipment which are available to mitigate the effects of the accident are discussed in Subsection 15.0.8 and listed in Table 15.0-5.

15.1.2.2 Analysis of Effects and Consequences

a. Method of Analysis

The excessive heat removal due to a feedwater system malfunction transient is analyzed by using the detailed digital computer LOFTRAN[1] code. This code simulates the neutron kinetics of the 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 system is analyzed to demonstrate acceptable consequences in the event of an excessive feedwater addition, due to a control system malfunction or operator error which allows a feedwater control valve to open fully. Four cases are analyzed as follows:

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

2) Accidental opening of one feedwater control valve with the reactor in manual control at full power.

3) Accidental opening of one feedwater control valve with the reactor in automatic control at zero load, with the reactor just critical.

4) Accidental opening of one feedwater control valve with the reactor in manual control at zero load, with the reactor just critical.

This accident is analyzed with the revised thermal design procedure as described in WCAP-11397[5]. The reactivity insertion rate following a feedwater system malfunction is calculated with the following assumptions:

a. For the feedwater control valve accident at full power, one feedwater control valve is assumed to malfunction resulting in a step increase to 187 percent of nominal feedwater flow to one steam generator.

b. For the feedwater control valve accident at zero load condition, a feedwater control valve malfunction occurs which results in an increase in flow to one steam generator from zero to 200 percent of the nominal full load value for one steam generator.

c. For the zero load condition, feedwater temperature is at a conservatively low value of 32°F.

d. No credit is taken for the heat capacity of the RCS and steam generator thick metal in attenuating the resulting plant cooldown.

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

Normal reactor control systems and engineered safety systems are not required to function. The reactor protection system may function to trip the reactor due to overpower or high-high steam generator water level conditions. No single active failure will prevent operation of the reactor protection system.

b. Results

The calculated sequence of events for this accident is shown in Table 5.1-1.

The full power cases with maximum reactivity feedback coefficients give the largest reactivity feedback and result in the highest peak power. The manual and automatic rod control cases give similar results (although the automatic control case has a slightly higher peak power and a slightly lower minimum DNBR value). The rod control system is not required to function for an excessive feedwater flow event.

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

Transients results for the full power cases are provided in Figures 5.1-1a and 5.1-1b and for the zero power cases in Figures 5.1-1c and 5.1-1d. The DNBR does not fall below the limit value. Following reactor trip (full power cases), the plant approaches a stabilized condition; standard plant shutdown procedures may then be followed to further cool down the plant.

Since the power level rises during the excessive feedwater flow incident, the fuel temperatures will also rise until after reactor trip occurs. The core heat flux lags behind the neutron flux response due to the fuel rod thermal time constant, hence the peak value does not exceed 118 percent of its nominal value (i.e., the assumed high neutron flux trip setpoint). The peak fuel temperature will thus remain well below the fuel melting temperature.

The transient results have shown that DNB does not occur at any time during the excessive feedwater flow incident; thus, the ability of

the primary coolant to remove heat from the fuel rod is not reduced.

15.1.2.3 Radiological Consequences

No fuel failure and radioactivity releases are anticipated as a direct result of this malfunction. Consequently, no radiological consequences are predicted.

15.1.2.4 Conclusions

The results of the analysis show that the DNB ratio encountered for an excessive feedwater addition at power is above the limit value; hence, no fuel or clad damage is predicted.

15.1.3 Excessive Increase in Secondary Steam Flow

15.1.3.1 Identification of Causes and Accident Description

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

Steam flow increases greater than 10 percent are analyzed in Subsections 15.1.4 and 15.1.5.

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

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

Protection against an excessive load increase accident is provided by the following Reactor Protection System (RPS) signals:

- Overpower ΔT
- Overtemperature ΔT
- Power range high neutron flux
- Low pressurizer pressure

An excessive load increase incident is considered to be an ANS Condition II event, a fault of moderate frequency. See Subsection 15.0.1 for a discussion of Condition II events.

15.1.3.2 Analysis of Effects and Consequences

a. Method of Analysis

This accident is analyzed using the LOFTRAN Code (Reference 1). The code simulates the neutron kinetics, Reactor Coolant System, pressurizer, pressurizer relief and safety valves, pressurizer spray, steam generator, steam generator safety valves, and Feedwater System. The code computes pertinent plant variables including temperatures, pressures, and power level.

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

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

A conservative limit on the turbine valve opening is assumed, and all cases are studied without credit being taken for pressurizer heaters.

This accident is analyzed with the revised thermal design procedure as described in WCAP-11397[5]. Initial reactor power, RCS pressure and temperature are assumed to be at their nominal values.

Normal reactor control systems and engineered safety systems are not required to function. The reactor protection system 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 will prevent the reactor protection system from performing its intended function.

The cases which assume automatic rod control are analyzed to ensure that the worst case is presented. The automatic function is not required for core protection.

b. Results

Figures 15.1-2, Sh.1 and 15.1-2, sh.2 illustrate the transient with minimum reactivity coefficients and the reactor in the manual rod control mode. Figures 15.1-2, sh.3 and 15.1-2, sh.4 illustrate the transient with minimum reactivity coefficients and the reactor in the automatic rod control mode. Figures 15.1-2, sh.5 and 15.1-2, sh.6 illustrate the transient with maximum reactivity coefficients and the reactor in the manual rod control mode. Figures 15.1-2, sh.7 and 15.1-2, sh.8 illustrate the transient with maximum reactivity coefficients and the reactor in the automatic rod control mode.

The case with minimum reactivity coefficients an automatic rod control reaches the highest calculated power and has the lowest DNBR. All cases show that the core power increases, thereby reducing the rate of decrease in average coolant temperature and pressurizer pressure. For all cases, the DNBR remains above the limit value.

For all cases, the plant rapidly reaches a stabilized condition at the higher power level. Normal plant operating procedures would then be followed to reduce power. The calculated sequence of events for the excessive load increase incident is shown on Table 15.1-1.

15.1.3.3 Radiological Consequences

No fuel failure and no radioactivity releases are anticipated as a direct result of this malfunction. Consequently, no radiological consequences are predicted.

15.1.3.4 Conclusions

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

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

15.1.4 Inadvertent Opening of a Steam Generator Relief or Safety Valve

15.1.4.1 Identification of Causes and Accident Description

The most severe core conditions resulting from an accidental depressurization of the Main Steam System are associated with an inadvertent opening of a

single steam dump, relief, or safety valve. The analyses performed assuming a rupture of a main steam line are given in Subsection 15.1.5.

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

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

> Assuming a stuck Rod Cluster Control Assembly, with offsite power available and assuming a single failure in the Engineered Safety Features System, there will be no consequential damage to the core or Reactor Coolant System after reactor trip for a steam release equivalent to the spurious opening, with failure to close of the largest of any single steam dump, relief, or safety valve.

Accidental depressurization of the secondary system is classified as an ANS Condition II event. See Subsection 15.0.1 for a discussion of Condition II events.

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

a. Safety injection system actuation from any of the following:

   1. Two out of four pressurizer pressure signals

   2. Two out of three high-1 containment pressure signals

   3. Two out of three low steam line pressure signals in any one loop.

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

c. Redundant isolation of the main feedwater lines.

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

d. Trip of the fast-acting steam line stop valves (designed to close in less than 5 seconds) on:

1. High-2 containment pressure

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

3. Two out of three high negative steam line pressure rate in any one loop (below Permissive P-11).

Plant systems and equipment which are available to mitigate the effects of the accident are also discussed in Subsection 15.0.8 and listed in Table 15.0-5.

15.1.4.2 Analysis of Effects and Consequences

a. Method of Analysis

The consequences of an inadvertent opening of a steam generator relief or safety valve are bounded by the zero power steam line rupture discussed in Section 15.1.5. The opening of a steam generator relief or safety valve causes a slower steam generator blowdown and RCS cooldown than the steam line rupture event. This would result in a lower power level if a return to power were to occur as predicted for the zero power steam line rupture. The minimum DNBR for the zero power steam line rupture, which remains above the safety analysis limit, would be lower than that for the opening of a steam generator relief or safety valve.

b. Results

Since the minimum DNBR for the zero power steam line rupture (Subsection 15.1.5) remains above the safety analysis limit, there would be no fuel failure predicted for an inadvertent opening of a steam generator relief or safety valve.

15.1.4.3 Radiological Consequences

The radiological consequences resulting from this malfunction are considerably less than those calculated for a main steam line rupture. The analyses performed assuming a rupture of a main steam line are given in Subsection 15.1.5.3.

15.1.4.4 Conclusions

The analysis shows that the criteria stated earlier in this section are satisfied. The DNBR in maintained above the safety analysis limit value.

15.1.5 Steam System Piping Failure

15.1.5.1 Identification of Causes and Accident Description

The steam release arising from a rupture of a main steam line would result in an initial increase in steam flow which decreases during the accident as the steam pressure falls. The energy removal from the RCS causes a reduction of coolant temperature and pressure. In the presence of a negative moderator temperature coefficient, the cooldown results in an insertion of positive reactivity. If the most reactive Rod Cluster Control Assembly (RCCA) is assumed stuck in its fully withdrawn position after reactor trip, there is an increased possibility that the core will become critical and return to power. The core is ultimately shut down by the boric acid delivered by the Safety Injection System.

The analysis of a main steam line rupture is performed to demonstrate that the following criteria are satisfied:

a. Assuming a stuck RCCA with or without offsite power, and assuming a single failure in the Engineered Safety Features, the core remains in place and intact. Radiation doses do not exceed the guidelines of 10 CFR 100.

b. Although DNB and possible clad perforation following a steam pipe rupture are not necessarily unacceptable, the following analysis, in fact, shows that no DNB occurs for any rupture assuming the most reactive assembly stuck in its fully withdrawn position.

A major steam line rupture is classified as an ANS Condition IV event. A minor steam line rupture is classified as an ANS Condition III event.

The major rupture of a steam line is the most limiting cooldown transient and is analyzed at zero power with no decay heat. Decay heat would retard the cooldown thereby reducing the return to power. A detailed analysis of this transient with the most limiting break size, a double-ended rupture, is presented here.

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

a. Safety injection system actuation from any of the following:

   1. Two out of four low pressurizer pressure signals
   2. Two out of three high-1 containment pressure signals
   3. Two out of three low steam line pressure signals in any one loop.

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

c. 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, a safety injection signal will rapidly close all feedwater isolation valves and backup feedwater control valves and trip the main feedwater pumps.

d. Trip of the fast-acting Main Steam Isolation Valves (MSIVs) which are designed to close in less than 5 seconds after receipt of a signal on:

1. High-2 containment pressure
2. Safety injection system actuation derived from two out of three low steam line pressure signals in any one loop (above Permissive P-11)
3. Two out of three high negative steam pressure rate in any one loop (below Permissive P-11).

For breaks downstream of the isolation valves, closure of all valves would completely terminate the blowdown. For any break, in any location, no more than one steam generator would experience an uncontrolled blowdown even if one of the isolation valves fails to close.

Flow restrictors are installed in the steam generator outlet nozzle, an integral part of the steam generator. The effective throat area of the nozzles is 1.4 square feet, which is considerably less than the main steam pipe area; thus, the nozzles also serve to limit the maximum steam flow for a break at any location. Also, the main steam isolation valve seat area limits the reverse blowdown from the intact steam generators.

15.1.5.2 Analysis of Effects and Consequences

a. Method of Analysis

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

a. The core heat flux and RCS temperature and pressure transients resulting from the cooldown following the steam line break. The LOFTRAN[1] code has been used.

b. The thermal and hydraulic behavior of the core following a steam line break. A detailed thermal and hydraulic digital computer code, VIPRE[4], has been used to determine if DNB occurs for the core conditions computed in item a above.

Studies have been performed to determine the sensitivity of steam line break results to various assumptions (reference 6). Based upon this study, the following conditions were assumed to exist at the time of a main steam line break accident:

1. End-of-life shutdown margin at no load, equilibrium xenon conditions, and the most reactive RCCA stuck in its fully withdrawn position: operation of the control banks during core burnup is restricted in such a way that the addition of positive reactivity in a steam line break accident will not lead to a more adverse condition than the case analyzed.

2. A negative moderator coefficient corresponding to the end-of-life rodded core with the most reactive RCCA in the fully withdrawn position. The effect of power generation in the core on overall reactivity is shown in Figure 15.1-6.

   The core properties associated with the sector nearest the affected steam generator and those associated with the remaining sector were conservatively combined to obtain average core properties for reactivity feedback calculation. Further, it was conservatively assumed that the core power distribution was uniform. These two conditions cause underprediction of the reactivity feedback in the high power region near the stuck rod.

   To verify the conservatism of this method, the reactivity, as well as the power distribution, was checked for the limiting conditions for the cases analyzed. This core analysis considered the Doppler reactivity from the high fuel temperature near the stuck RCCA, moderator feedback from the high water enthalpy near the stuck RCCA, power redistribution and non-uniform core inlet temperature effects. For cases in which steam generation occurs in the high flux regions of the core, the effect of void formation was also included. It was determined that the reactivity employed in the kinetics analysis was always larger than the reactivity calculated, including the above local effects for the statepoints. These results verify conservatism; i.e., underprediction of negative reactivity feedback from power generation.

3. Minimum safety injection flow capability corresponding to the most restrictive single failure in the safety injection system. The Emergency Core Cooling System (ECCS) consists of three systems: a) the passive accumulators, b) the low head safety injection (residual heat removal) system, and c) the high head safety injection (charging) system. Only the safety injection system and the passive accumulators are modeled for the steam line break accident analysis.

   The modeling of the safety injection system in LOFTRAN is described in Reference 1. The flow corresponds to that delivered by one charging pump delivering its full flow to the cold leg header. No credit has been taken for the low concentration borated water, which must be swept from the lines downstream prior to the delivery of high concentration boric acid to the

reactor coolant loops.

For the cases where offsite power is assumed, the sequence of events in the safety injection system is the following: After the generation of the safety injection signal (appropriate delays for instrumentation, logic, and signal transport included), the appropriate valves begin to operate and the high head safety injection pump starts. In 22 seconds, the valves are assumed to be in their final position and the pump is assumed to be at full speed. The volume containing the low concentration borated water is swept into core before the 2,700 ppm borated water from the refueling water storage tank reaches the core. This delay, described above, is inherently included in the modeling.

4. Design value of the steam generator heat transfer coefficient, with no allowance for fouling factor, to maximize the cooldown.

5. Since the steam generators are provided with integral flow restrictors with a 1.4 square foot throat area, any rupture with a break area greater than 1.4 ft^2, regardless of location, would have the same effect on the NSSS as the 1.4 ft^2 break. The following cases have been considered in determining the core power and RCS transients:

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

   b. Case (a) with loss of offsite power simultaneous with the steam line break and initiation of the SIS. Loss of offsite power results in reactor coolant pump coastdown.

6. The limiting steam line break with return to power corresponds to the case with offsite power available. The offsite power available case results in the greatest challenge to DNB.

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

   The core parameters used for each of the two cases correspond to values determined from the respective transient analysis.

Both the cases above assume initial hot shutdown conditions at time zero since this represents the most pessimistic initial condition. Should the reactor be just critical or operating at power at the time of a steam line break, the reactor will be tripped by the normal overpower protection system when power level reaches a trip point. Following a trip at power, the reactor coolant system contains more stored energy than at no-load, the average coolant temperature is higher than at no-load, and there is appreciable energy stored in the fuel. Thus, the additional stored energy is removed via the cooldown caused by the steam line break before the no-load conditions of RCS 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.

In addition, since the initial steam generator water inventory is greatest at no-load, the magnitude and duration of RCS cooldown are more severe than steam line breaks occurring at power.

8. In computing the steam flow during a steam line break, the Moody curve[2] for $fL/D = 0$ is used.

9. Emergency Feedwater flow is limited by passive flow restrictors to protect the pumps against a runout condition during main steam line rupture.

b. Results

The calculated sequence of events is shown on Table 15.1-1. The results presented are a conservative indication of the events which would occur assuming a steam line rupture since it is postulated that all of the conditions described above occur simultaneously.

1. Core Power and Reactor Coolant System Transient

Figures 15.1-6, sh.2-sh.6 show the RCS transient and core heat flux following a main steam line rupture (complete severance of a pipe) at initial no-load condition (case a).

Offsite power is assumed available so that full reactor coolant flow exists. The transient shown assumes an uncontrolled steam release from only one steam generator. Should the core be critical at near zero power when the rupture occurs, the initiation of safety injection by low steam line pressure will trip the reactor. Steam release from more than one steam generator will be prevented by automatic trip of the fast-acting isolation valves in the steam lines, by high containment pressure signals, or low steam line pressure. Even with the failure of

one valve, release is limited to no more than 10 seconds for the other steam generators while the one generator blows down. The steam line stop valves are designed to be fully closed in less than 5 seconds from receipt of a closure signal.

As shown in Figure 15.1-1, the core attains criticality with the RCCAs inserted (with the design shutdown assuming one stuck RCCA) before boron solution at 2,700 ppm enters the RCS. A peak core power less than the nominal full power value is attained.

The calculation assumes the boric acid is mixed with and diluted by the water flowing in the RCS prior to entering the reactor core. The concentration after mixing depends upon the relative flow rates in the RCS and in the safety injection system. The variation of mass flow rate in the RCS due to water density changes is included in the calculation as is the variation of flow rate in the safety injection system due to changes in the RCS pressure. The safety injection system flow calculation includes the line losses in the system as well as the pump head curve.

Once the pressure in the RCS falls below the pressure in the accumulators, boron solution at 2,600 ppm also enters the RCS from the accumulators.

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

2. Margin to Critical Heat Flux

A DNB analysis was performed. It was found that all cases had a minimum DNBR greater than the limit value.

15.1.5.3 Radiological Consequences

a. Assumptions and Parameters

A realistic case and a conservative case are considered. The conservative analysis employs more pessimistic assumptions regarding fission product release and transport. The assumptions and parameters for the two analyses are summarized in Table 15.1-2. Detailed assumptions not stated in this table are discussed in this section.

For both cases, it is assumed that a rupture occurs in an nonisolable portion of a main steam line, resulting in the blowdown of one steam generator. Because the blowdown would occur rather rapidly, the one hour χ/Q values of Appendix 15B are used in calculating the dose contributed from this phase of the accident. The dose contribution from subsequent leakage is calculated using the values for the "standard" time periods listed in Appendix 15B. For the conservative case, it is also assumed that there is a coincident loss of AC power, and that during the cooldown, there is leakage of primary coolant into the steam generator with the nonisolable break. Increased primary coolant radioactive iodine levels due to iodine spiking are also considered in the conservative and realistic analysis.

1. Conservative Analysis

(a) The steam line which ruptures is attached to a steam generator containing secondary coolant water with activity concentrations at the Technical Specification limit of 0.1 μCi/gm dose equivalent I-131. Noble gases are continuously removed from the secondary side by the Main Steam Condenser Evacuation System; therefore, the secondary side noble gas activity concentration at the time of the rupture is considered negligible. Blowdown is assumed to continue over a half hour period, releasing the total iodine inventory of one steam generator and in addition the activity resulting from leakage of 0.347 gpm of primary coolant into the defective steam generator. Primary coolant leakage activity concentrations are based on 60 μCi/gm dose equivalent I-131 for the pre-existing iodine spike case and 1.0 μCi/gm dose equivalent I-131 with an increase of 500 in the rate of release of iodine into the primary coolant, for the coincident iodine case. Noble gas activity concentration of the primary coolant is conservatively based on assuming that the Technical Specification limit of $100/\bar{E}$ μCi/gm is due solely to noble gases. These values are presented in Appendix 15B. Release to the atmosphere during this 30 minute blowdown period is presented in Tables 15.1-3 and 15.1-4.

(b) The subsequent cooldown of the plant by releasing steam from the three unaffected steam generators results in the release of 1.0 percent of the iodine in those generators over an eight-hour period. Leakage at the rate of 940 gallons per day for the three intact steam generators continues for an eight-hour period. Table 15.1-4 lists the activity released from this source.

(c) Leakage at the rate of 500 gallons per day through the affected steam generator continues for a period of eight hours. All the noble gases and 10 percent of the iodines which leak through during this period are released to the atmosphere. Table 15.1-4 lists the activity released from this source.

2. Realistic Analysis

(a) Iodine activity in each steam generator is based upon 0.12 percent fuel failure and 100 lbs. per day primary to secondary leakage distributed evenly among the four steam generators. Table 11.1-4 lists the concentration of the iodine isotopes.

(b) One hundred percent of the iodine and noble gases in the affected steam generator are released to the environment. Table 15.1-5 lists the activity released.

(c) Leakage continues at the rate of 100 lbs/day used above (65.3 lbs. per day through the three intact steam generators, and 34.7 lbs. per day through the defective steam generator) for a period of 8 hours. All the noble gases from the four steam generators are released during this eight hour period. Ten percent of the iodine from the affected steam generator and 1.0 percent of the iodine in the 3 unaffected steam generators is also released during this period. The main condenser is available for releases from the unaffected steam generators, providing an iodine decontamination factor of 100. Table 15.1-5 lists the activity released by isotope. These quantities are based on the reactor coolant noble gas concentrations listed in Table 11.1-1. Reactor coolant iodine levels are based on maximum expected spiking levels listed in Appendix 15B.

b. Results

The doses resulting from this accident for both the realistic and conservative cases are shown in Table 15.1-6.

15.1.5.4 Conclusions

The analysis has shown that the criteria stated in Subsection 15.1.5.1 are satisfied.

Although DNB and possible cladding perforation following a steam pipe rupture are not necessarily unacceptable and not precluded by the criteria, the above analysis, in fact, shows that the DNB design basis is met for any rupture assuming the most restrictive RCCA stuck in its fully withdrawn position.

The doses which have been calculated for the major secondary system pipe rupture are below the values specified in the applicable regulation, 10 CFR 100, "Reactor Site Criteria."

15.1.6 References

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

3. WCAP-7979-A, "TWINKLE - A Multi-Dimensional Neutron Kinetics Computer Code," D. H. Risher, and R. F. Barry, January 1975

4. WCAP-14565, "Vipre-01 Modeling and Qualification for Pressurized Water Reactor Non-LOCA Thermal-Hydraulic Safety Analysis," Y. X. Sung, et al., April 1997

5. WCAP-11397-P-A, "Revised Thermal Design Procedure," A. J. Friedland and S. Ray, April 1984

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7. WCAP-9230, "Report on the Consequences of a Postulated Main Feedline Rupture," G. E. Lang and J. P. Cunningham, January 1978

8. WCAP-11394-P-A, "Methodology for the Analysis of the Dropped Rod Event," R. L. Haessler, et al., January 1990

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

10. WCAP-7588, Revision 1-A, "An Evaluation of the Rod Ejection Accident in Westinghouse Pressurized Water Reactors Using Spatial Kinetics Methods," D. H. Risher, January 1975

11. IN-1370, "Annual Report - SPERT Project, October 1968 - September 1969 Edition (Idaho Nuclear Corporation)," T. G. Taxelius, June 1970

12. ANL-7225, "Studies in TREAT of Zircaloy-2-Clad, UO_2-Core Simulated Fuel Elements," R. C. Liimantaininen and F. J. Testa, p. 177, November 1966

13. Letter from W. J. Johnson (Westinghouse) to R. C. Jones (USNRC), "Use of 2700°F PCT Acceptance Limit in Non-LOCA Accidents," NS-NRC-893466, October 1989

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

TABLE 15.1-1
(Sheet 1 of 2)

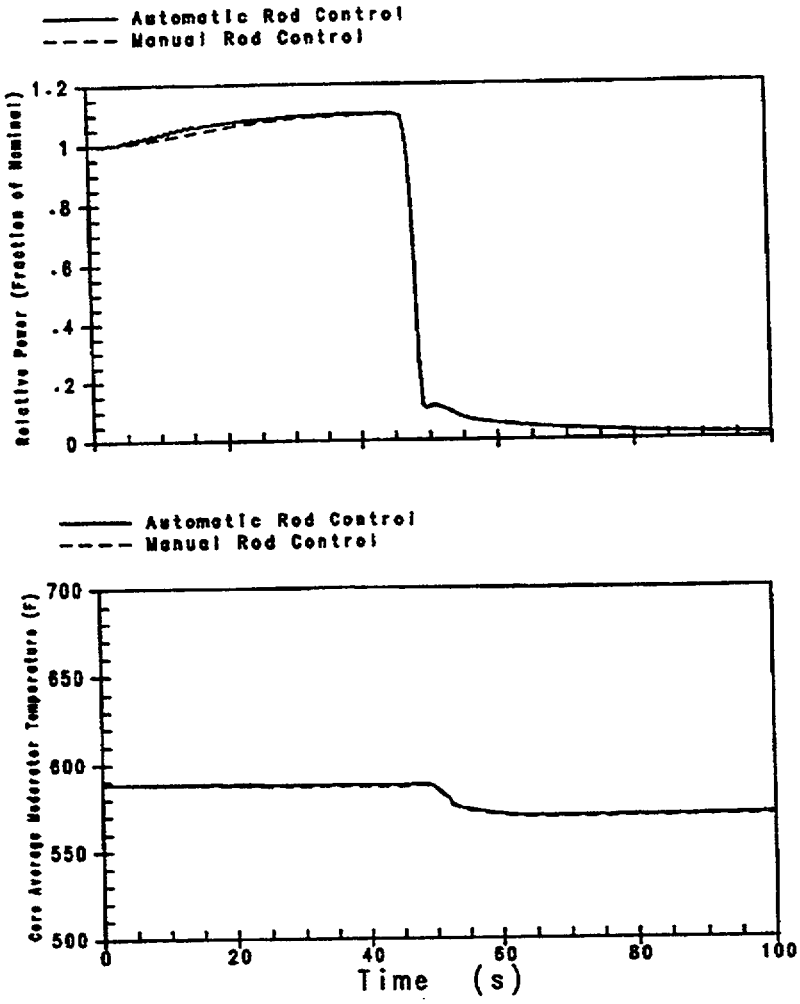
TIME SEQUENCE OF EVENTS FOR INCIDENTS THAT CAUSE AN INCREASE IN HEAT REMOVAL BY THE SECONDARY SYSTEM

Accident	Event	Time (seconds)
Excessive Feedwater Flow Malfunction at Full Power		
	One main feedwater control valve fails open fully	0.0
	High-high steam generator water level setpoint reached	42.8
	Minimum DNBR occurs	45.0
	Feedwater isolation valves fully closed	54.8
Excessive Load Increase		
1) Manual rod control (minimum reactivity coefficients)	10% step load increase	0.0
	Equilibrium condition reached (approximate time)	200.0
2) Automatic rod control (minimum reactivity coefficients)	10% step load increase	0.0
	Equilibrium condition reached (approximate time)	200.0
3) Manual rod control (minimum reactivity coefficients)	10% step load increase	0.0
	Equilibrium condition reached (approximate time)	200.0
4) Automatic rod control (minimum reactivity coefficients)	10% step load increase	0.0
	Equilibrium condition reached (approximate time)	200.0

TABLE 15.1-1
(Sheet 2 of 2)

TIME SEQUENCE OF EVENTS FOR INCIDENTS THAT CAUSE AN INCREASE IN HEAT REMOVAL BY THE SECONDARY SYSTEM

Accident	Event	Time (seconds)
Steam System piping failure (with offsite power available)		
	Steam line ruptures	0.0
	Criticality attained	21.8
	SI flow begins	28.0
	Accumulators actuate	102.6
	Minimum DNBR occurs	104.2



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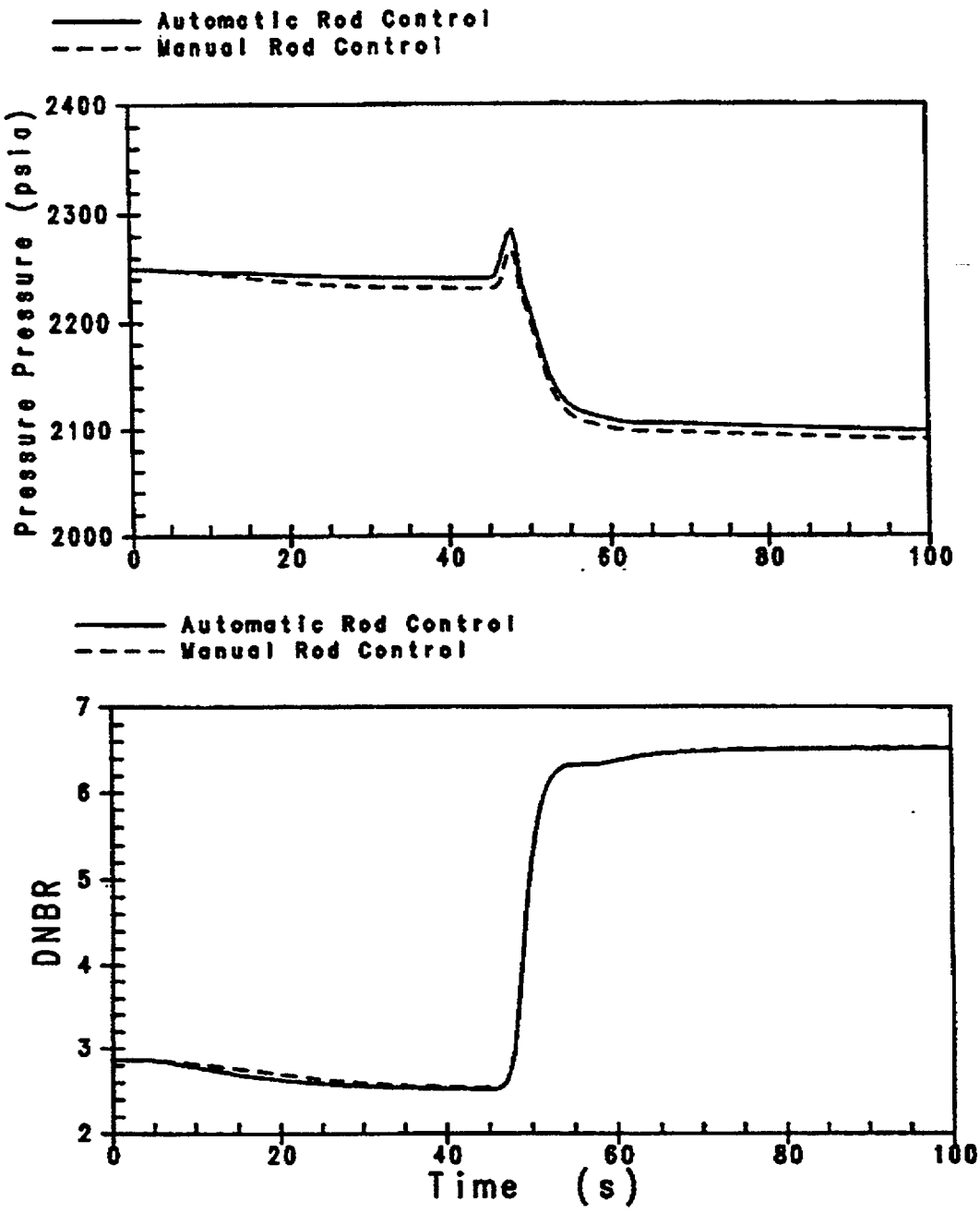
SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Relative Power and Core Average Moderator Temperature Transients for a Feedwater Control Valve Malfunction at Full Power	
	Rev. 07	Figure 15.1-1, Sh 1



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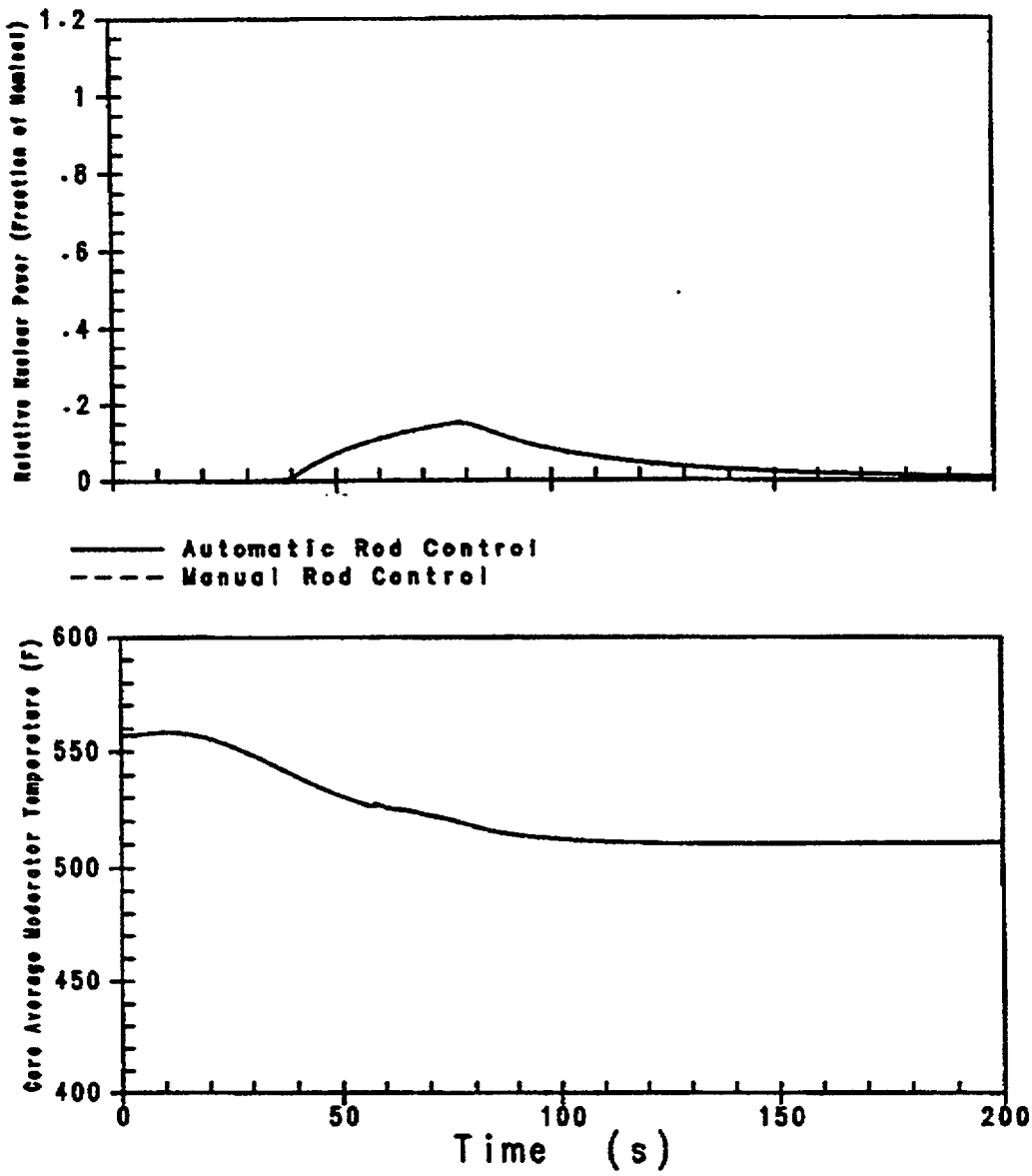
SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Pressurizer Pressure and DNBR Transients for a Feedwater Control Valve Malfunction at Full Power	
	REV. 07	FIGURE 15.1-1, Sh 2



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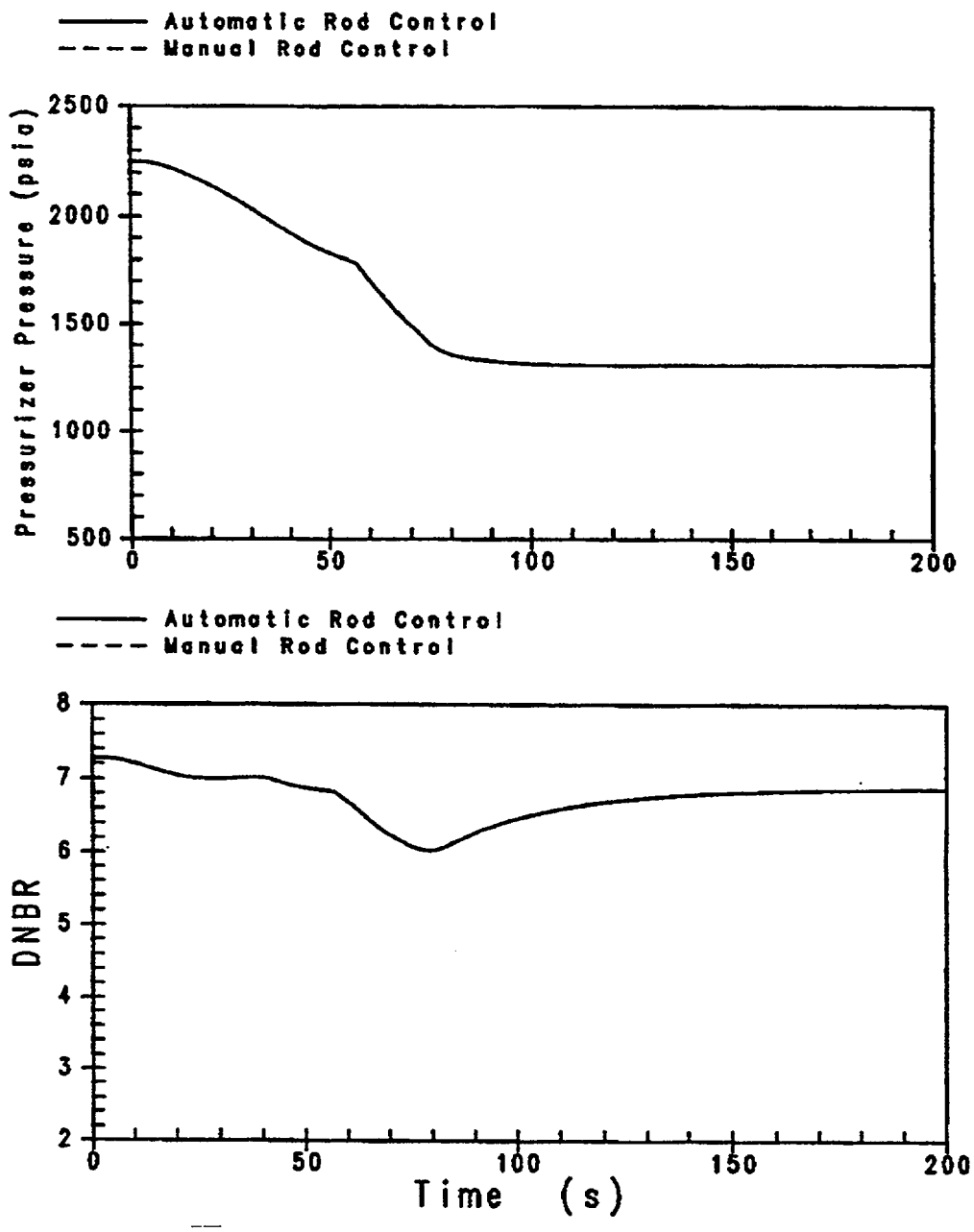
SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Relative Power and Core Average Moderator Temperature Transients for a Feedwater Control Valve Malfunction at Zero Power	
	REV. 07	FIGURE 15.1-1, Sh 3



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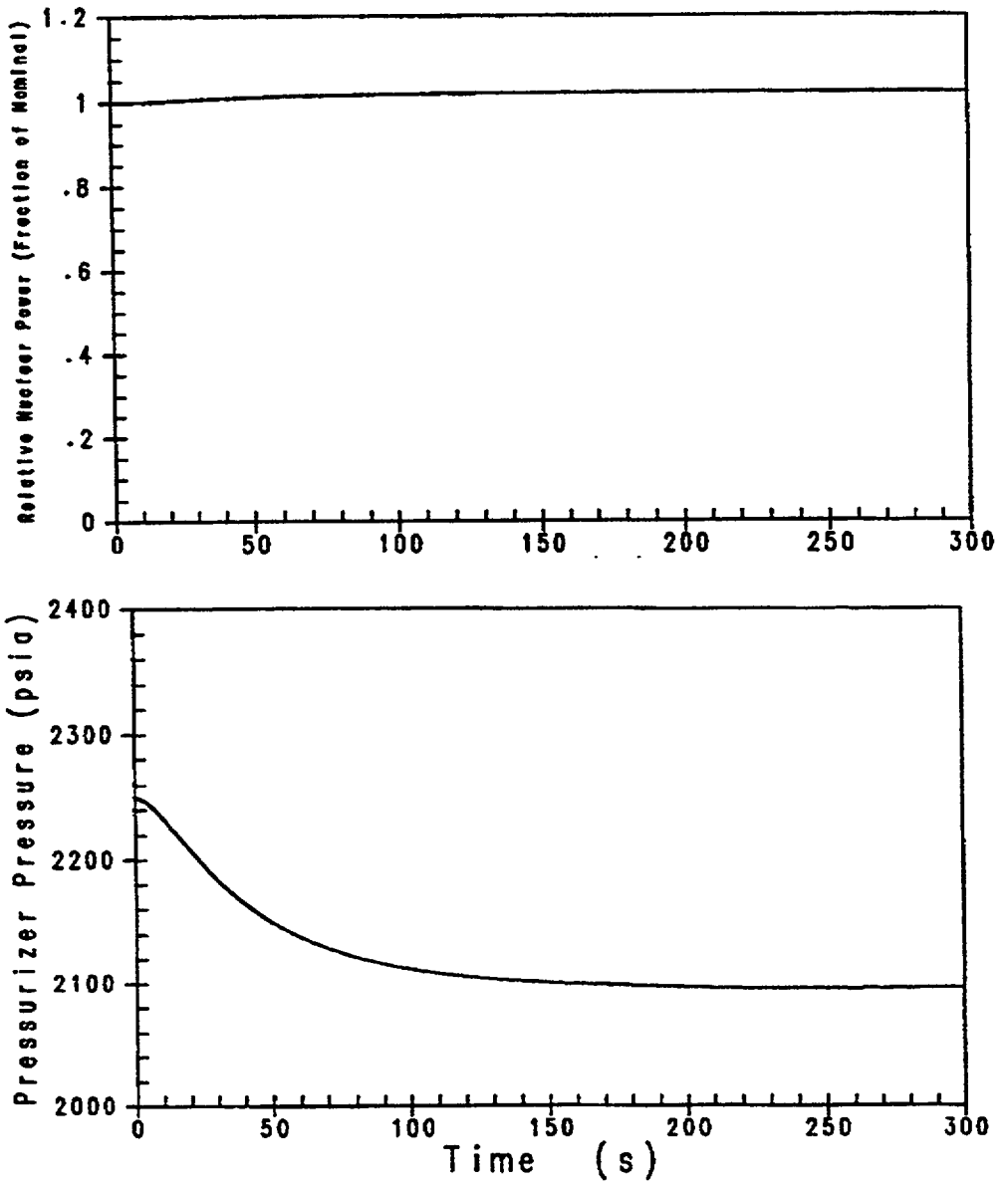
SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Pressurizer Pressure and DNBR Transients for a Feedwater Control Valve Malfunction at Zero Power	
	REV. 07	FIGURE 15.1-1, Sh 4



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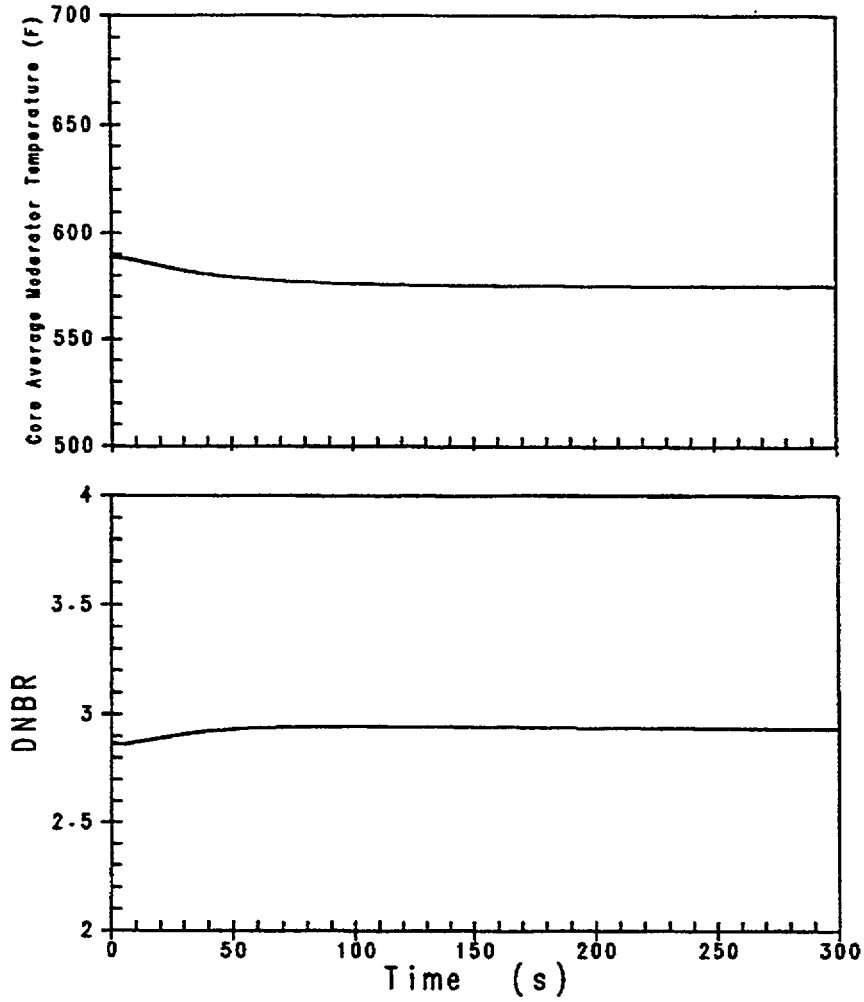
SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Relative Power and Pressurizer Pressure Transients for an Excessive Load Increase Event (manual rod control; minimum reactivity coefficients)	
	REV. 07	FIGURE 15.1-2, Sh 1



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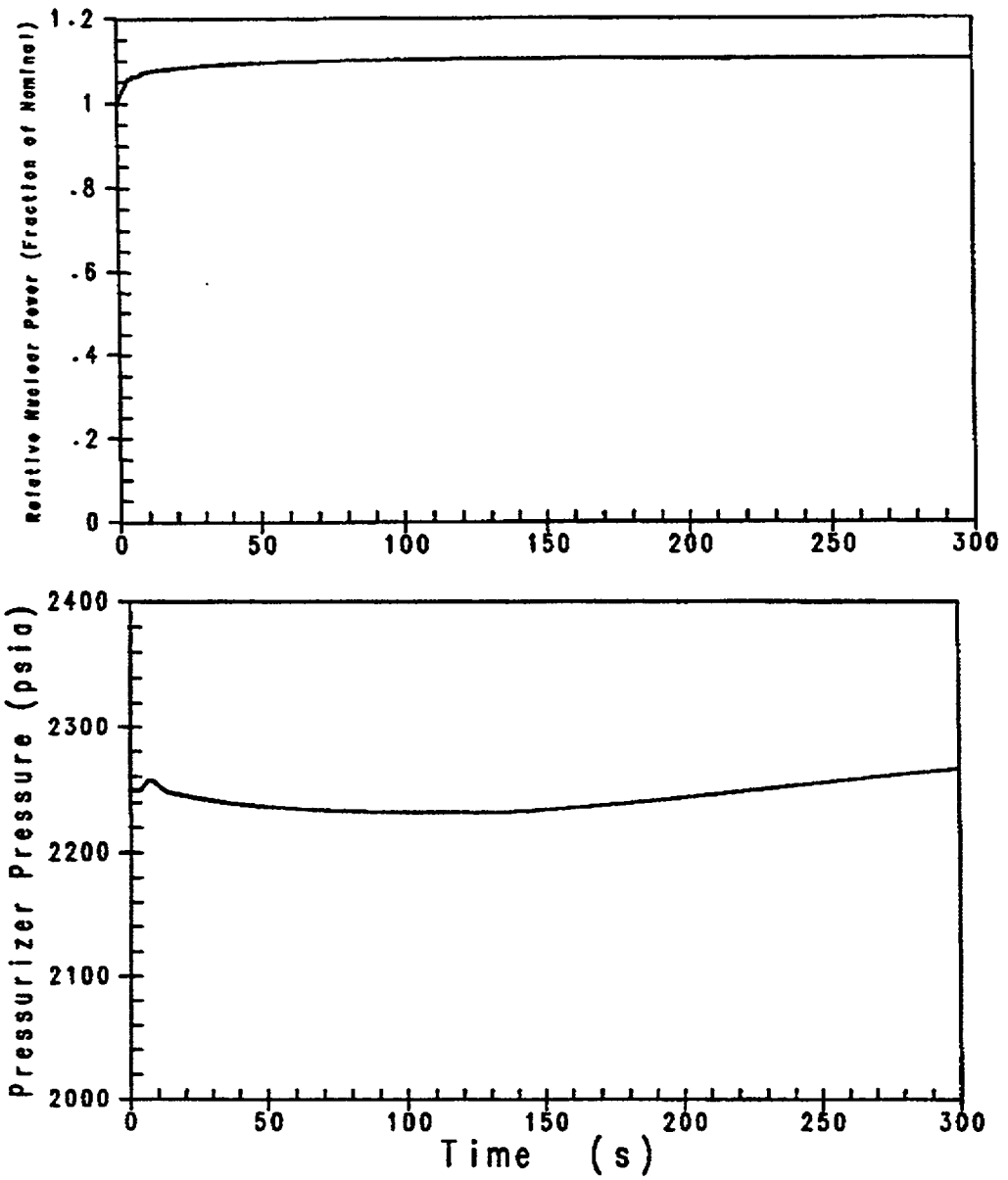
SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Core Average Moderator Temperature and DNBR Transients for an Excessive Load Increase Event (manual rod control; minimum reactivity coefficients)	
	REV. 07	FIGURE 15.1-2, Sh 2



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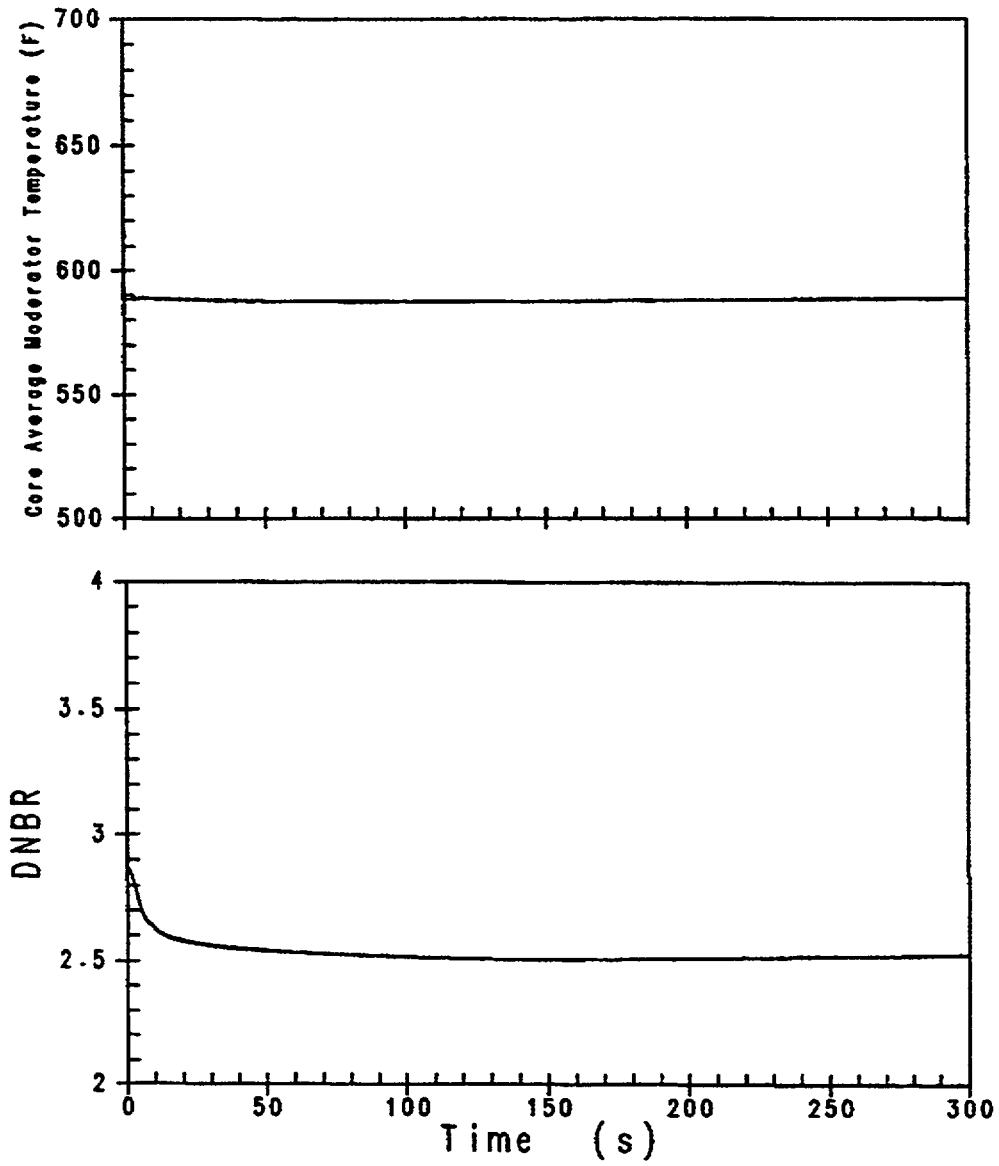
1SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Relative Power and Pressurizer Pressure Transients for an Excessive Load Increase Event (automatic rod control; minimum reactivity coefficients)	
	REV. 07	FIGURE 15.1-3, Sh 1



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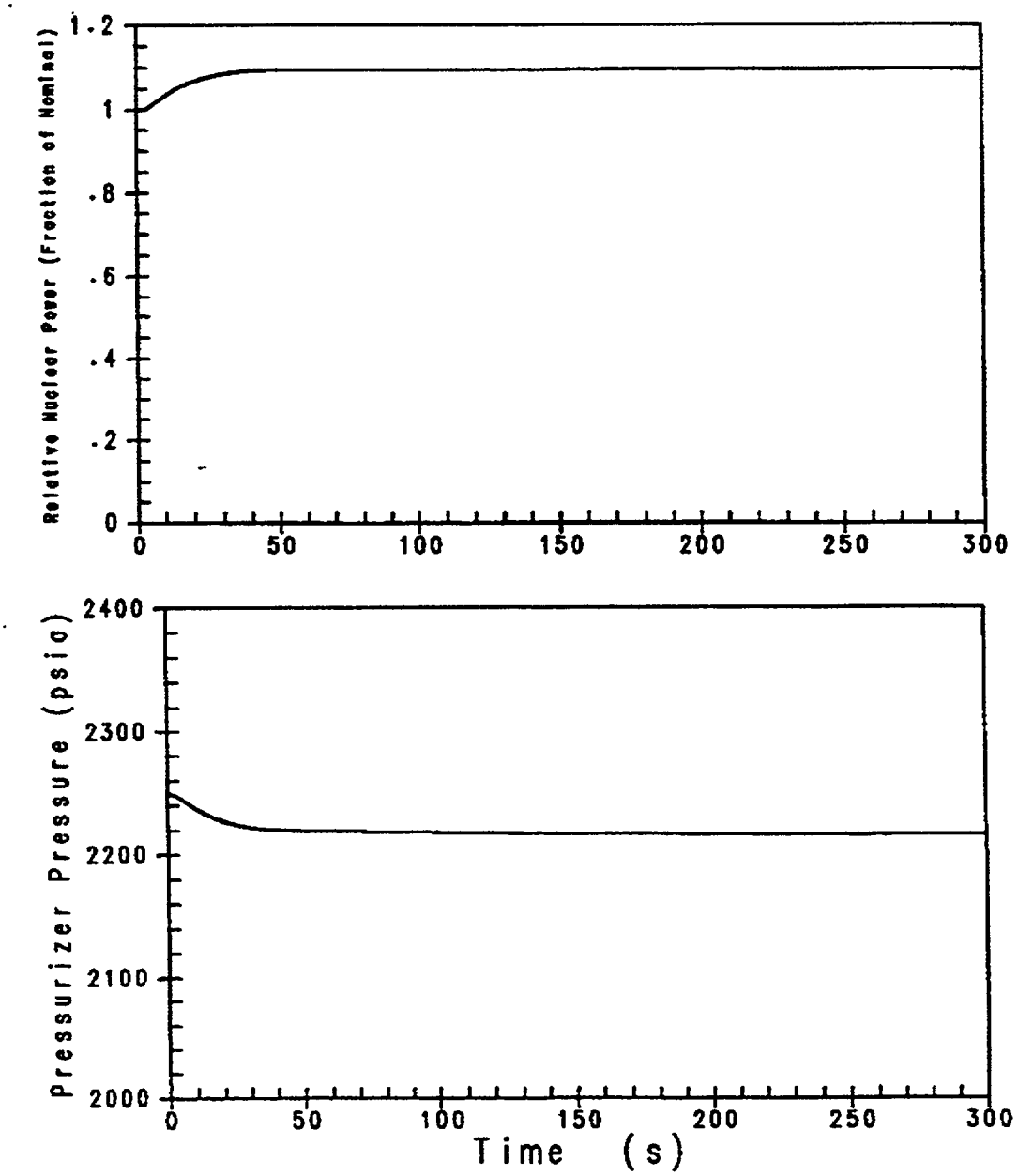
SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Core Average Moderator Temperature and DNBR Transients for an Excessive Load Increase Event (automatic rod control; minimum reactivity coefficients	
	REV. 07	FIGURE 15.1-3, Sh 2



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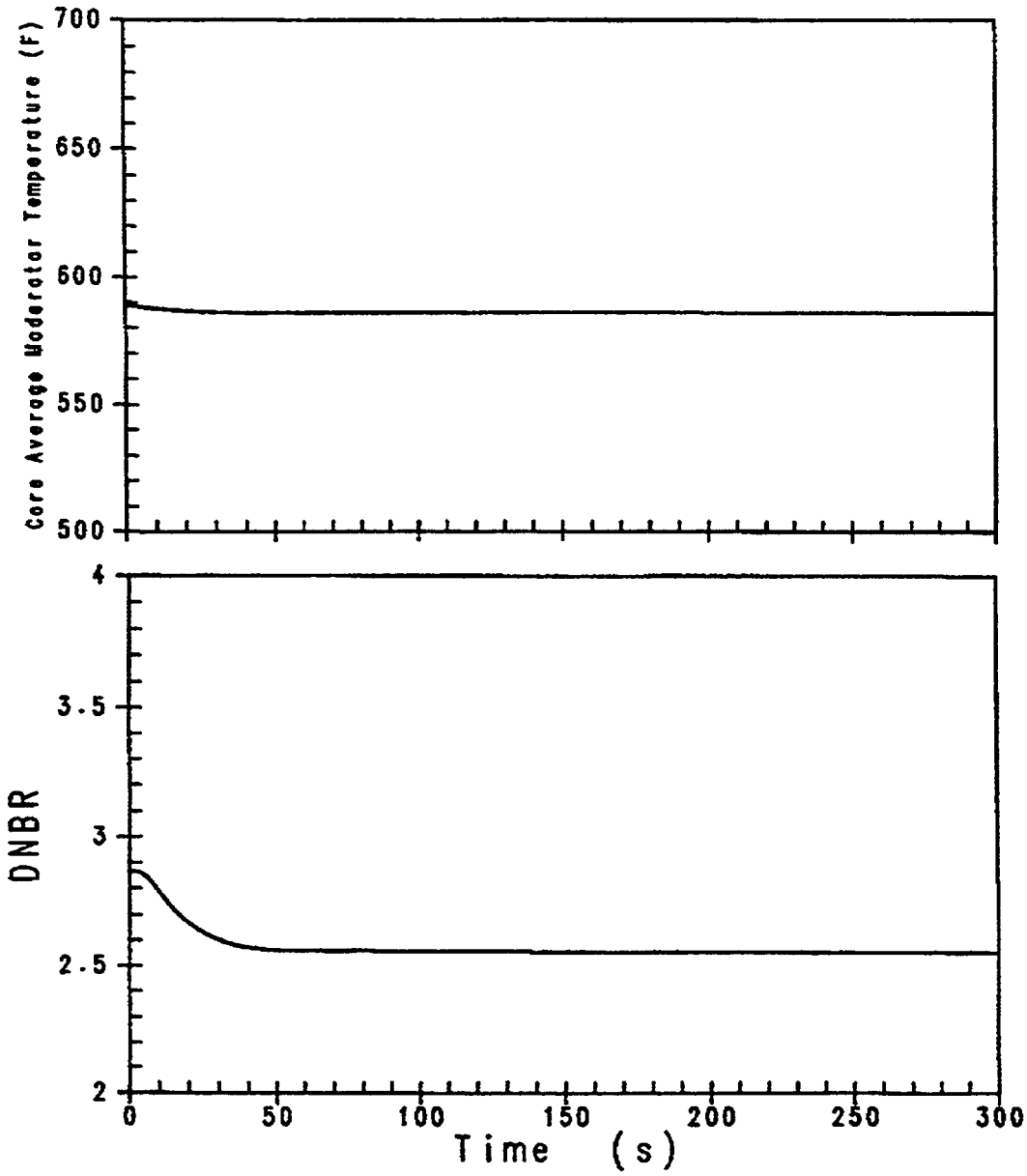
SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Relative Power and Pressurizer Pressure Transients for an Excessive Load Increase Event (manual rod control; maximum reactivity coefficients)	
	REV. 07	FIGURE 15.1-4, Sh 1



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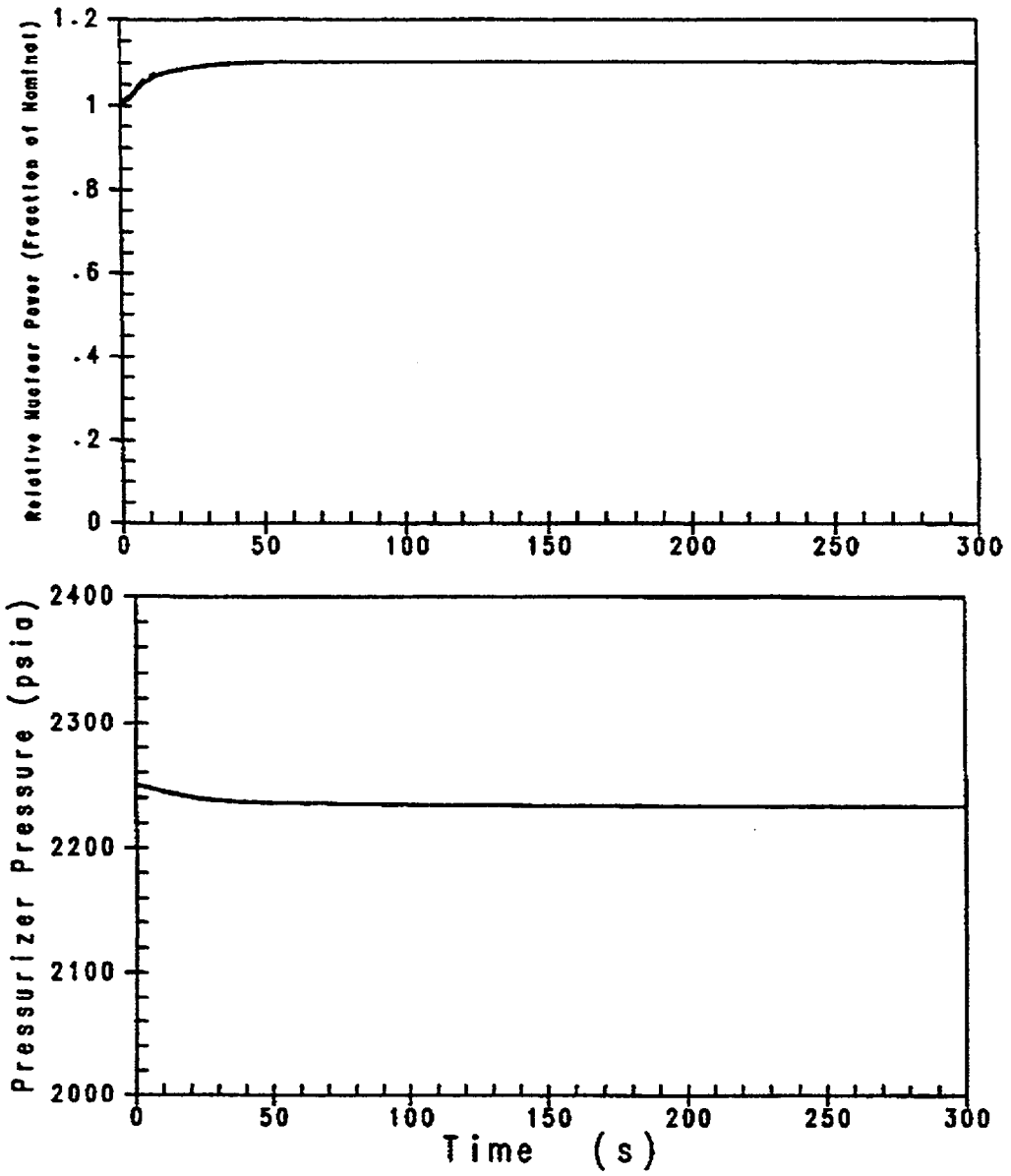
SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Core Average Moderator Temperature and DNBR Transients for an Excessive Load Increase Event (manual rod control; maximum reactivity coefficients)	
	REV. 07	FIGURE 15.1-4, Sh 2



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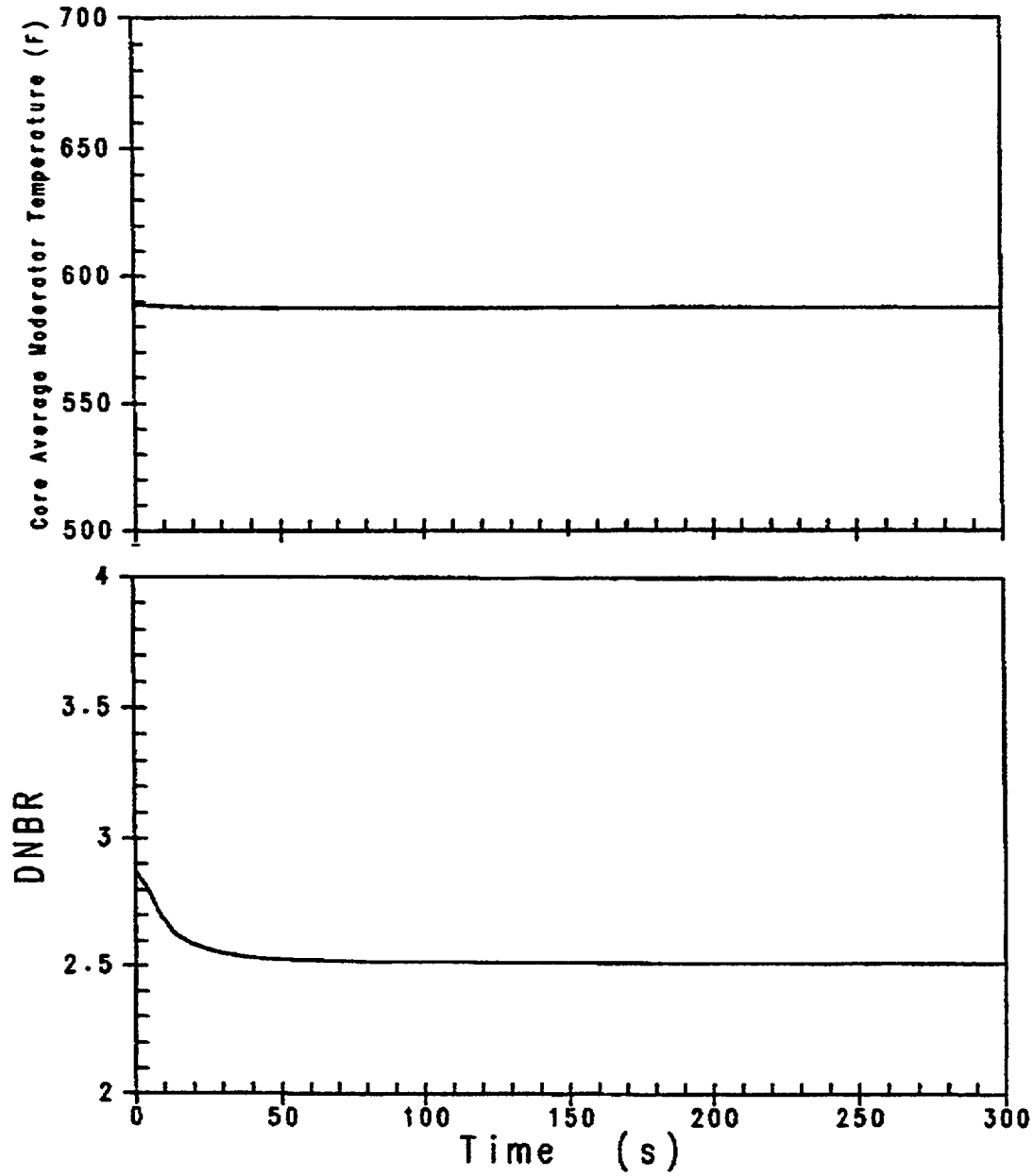
SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Relative Power and Pressurizer Pressure Transients for an Excessive Load Increase Event (automatic rod control; maximum reactivity coefficients)	
	REV. 07	FIGURE 15.1-5, Sh 1



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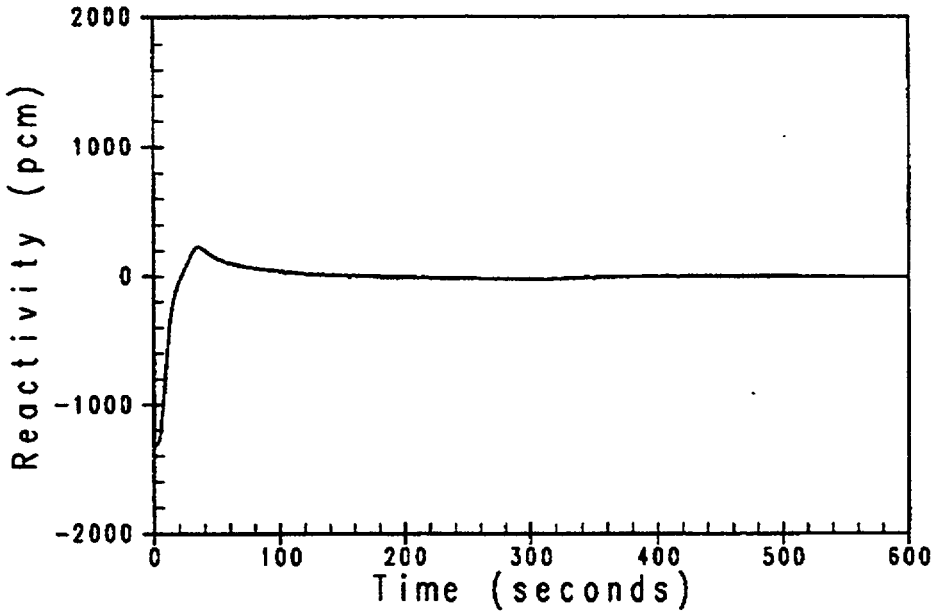
SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Core Average Moderator Temperature and DNBR Transients for an Excessive Load Increase Event (automatic rod control; minimum reactivity coefficients	
	REV. 07	FIGURE 15.1-5, Sh 2



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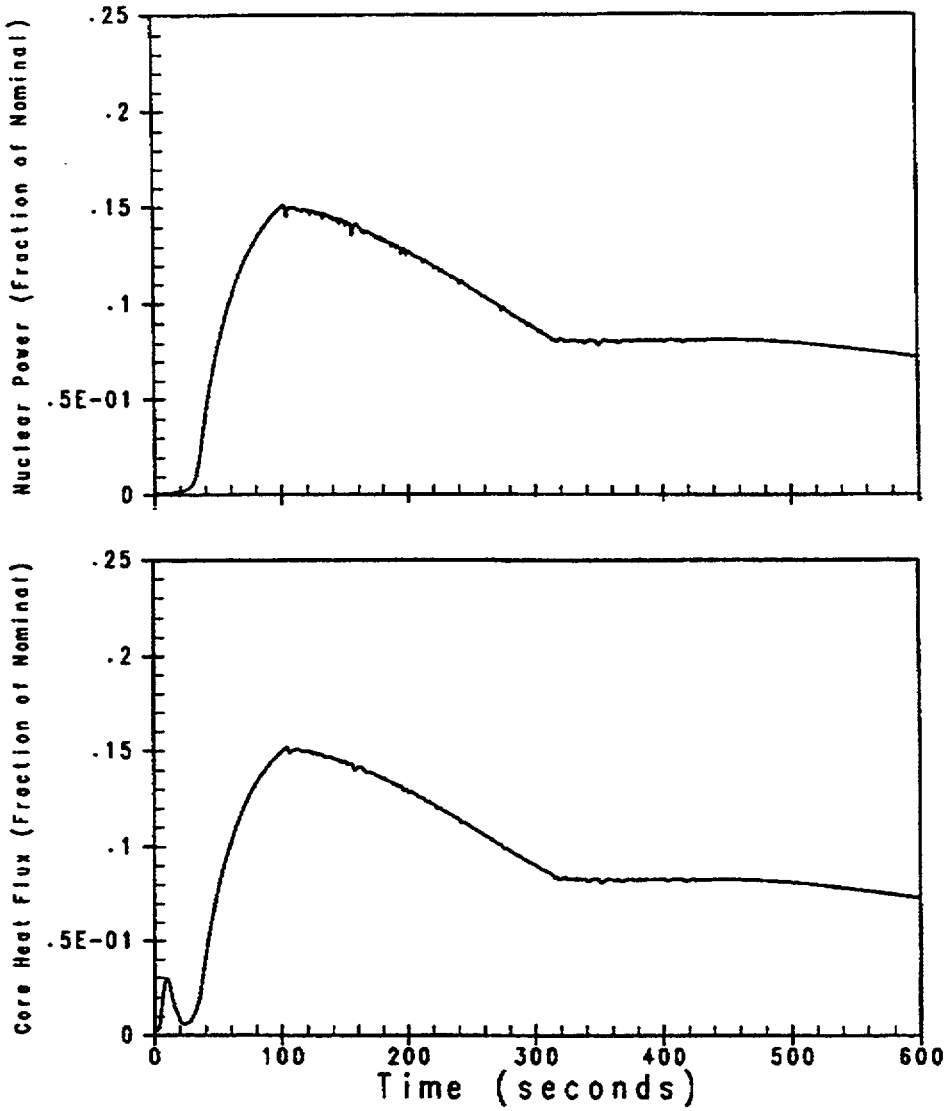
SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Reactivity Transient for a Steam Rupture at Zero Power with Offsite Power Available	
	REV. 07	FIGURE 15.1-6, Sh 1



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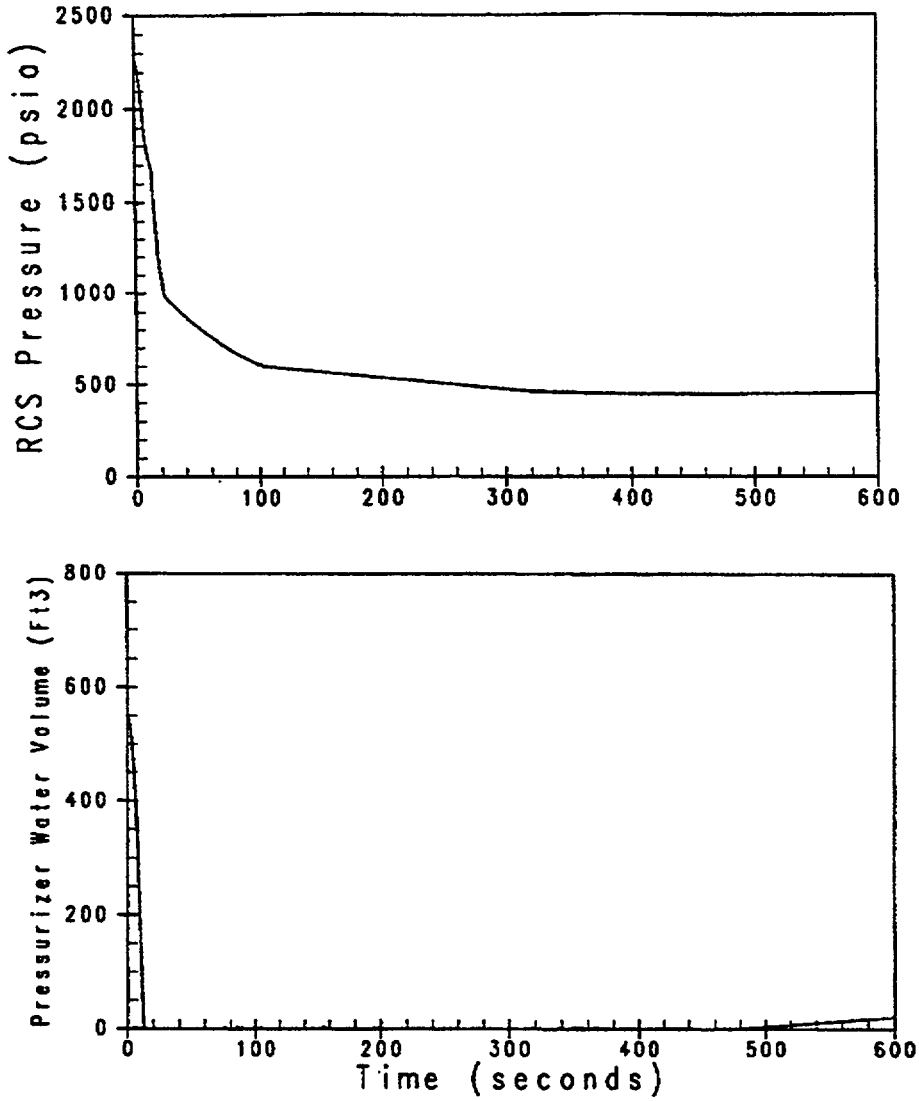
SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Nuclear Power and Core Heat Flux Transients for a Steam Line Rupture at Zero Power with Offsite Power Available	
	REV. 07	FIGURE 15.1-6, Sh 2



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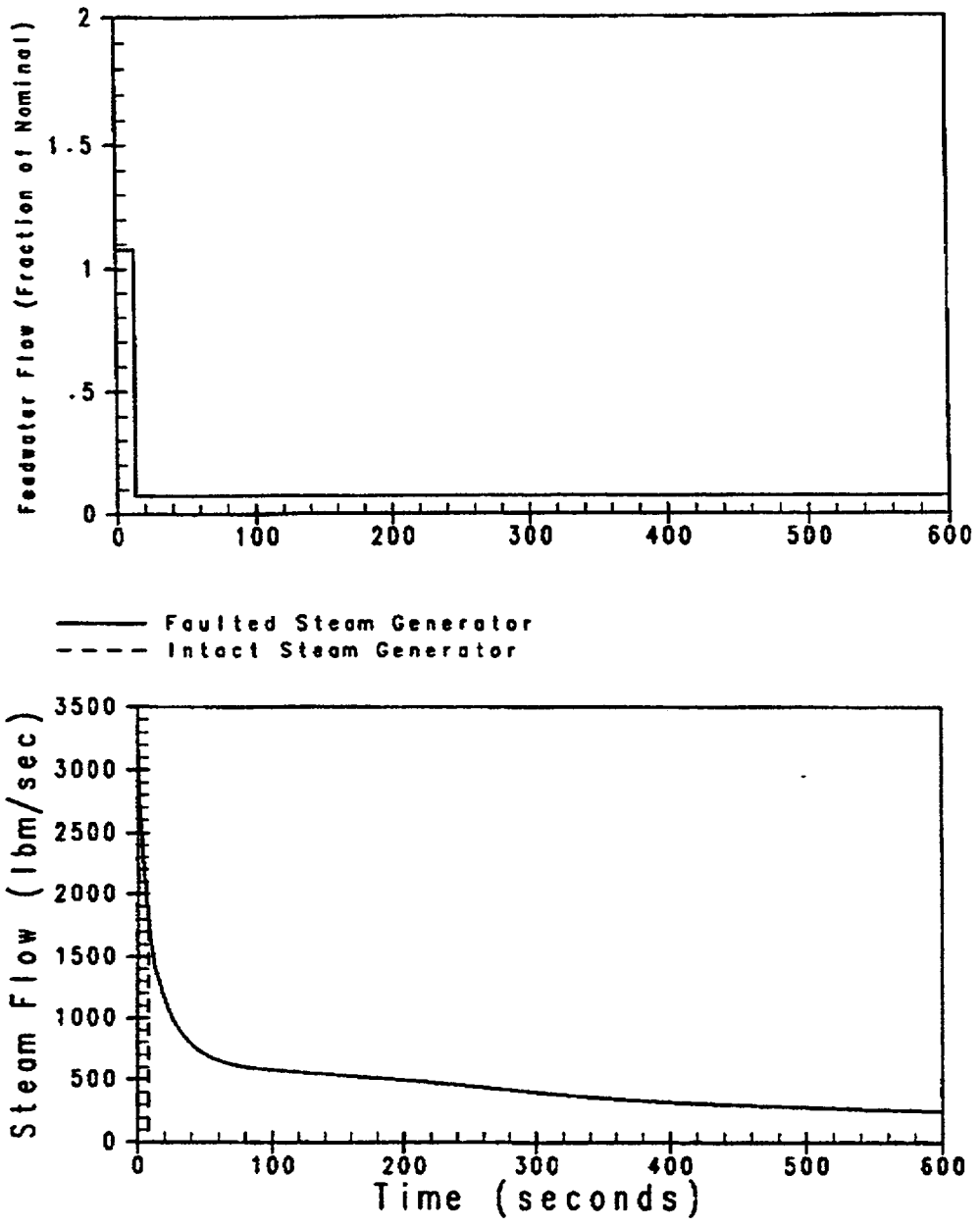
SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	RCS Pressure and Pressurizer Water Volume Transients for a Steam Line Rupture at Zero Power with Offsite Power Available	
	REV. 07	FIGURE 15.1-6, Sh 3



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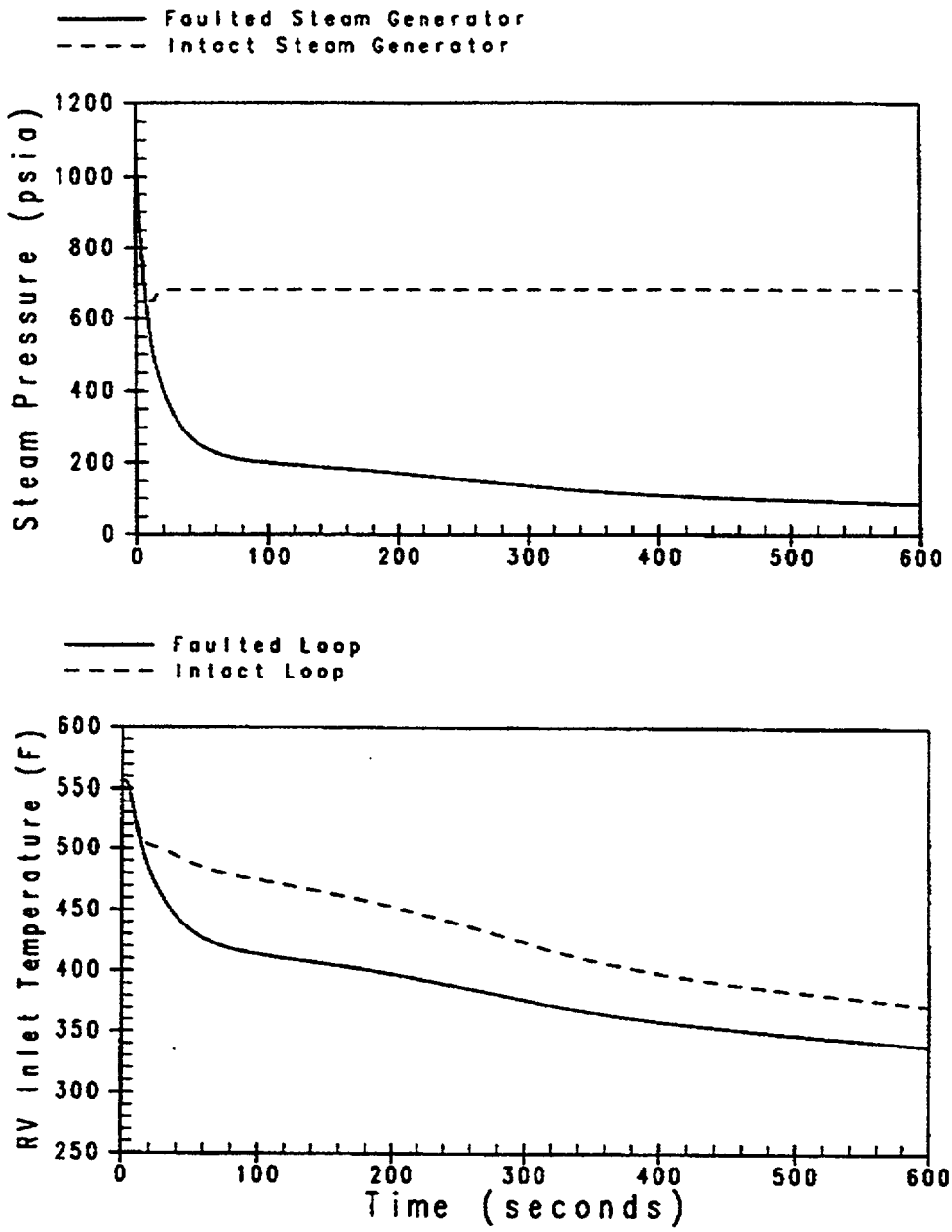
SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Feedwater Flow and Steam Flow Transients for a Steam Line Rupture at Zero Power with Offsite Power Available	
	REV. 07	FIGURE 15.1-6, Sh 4



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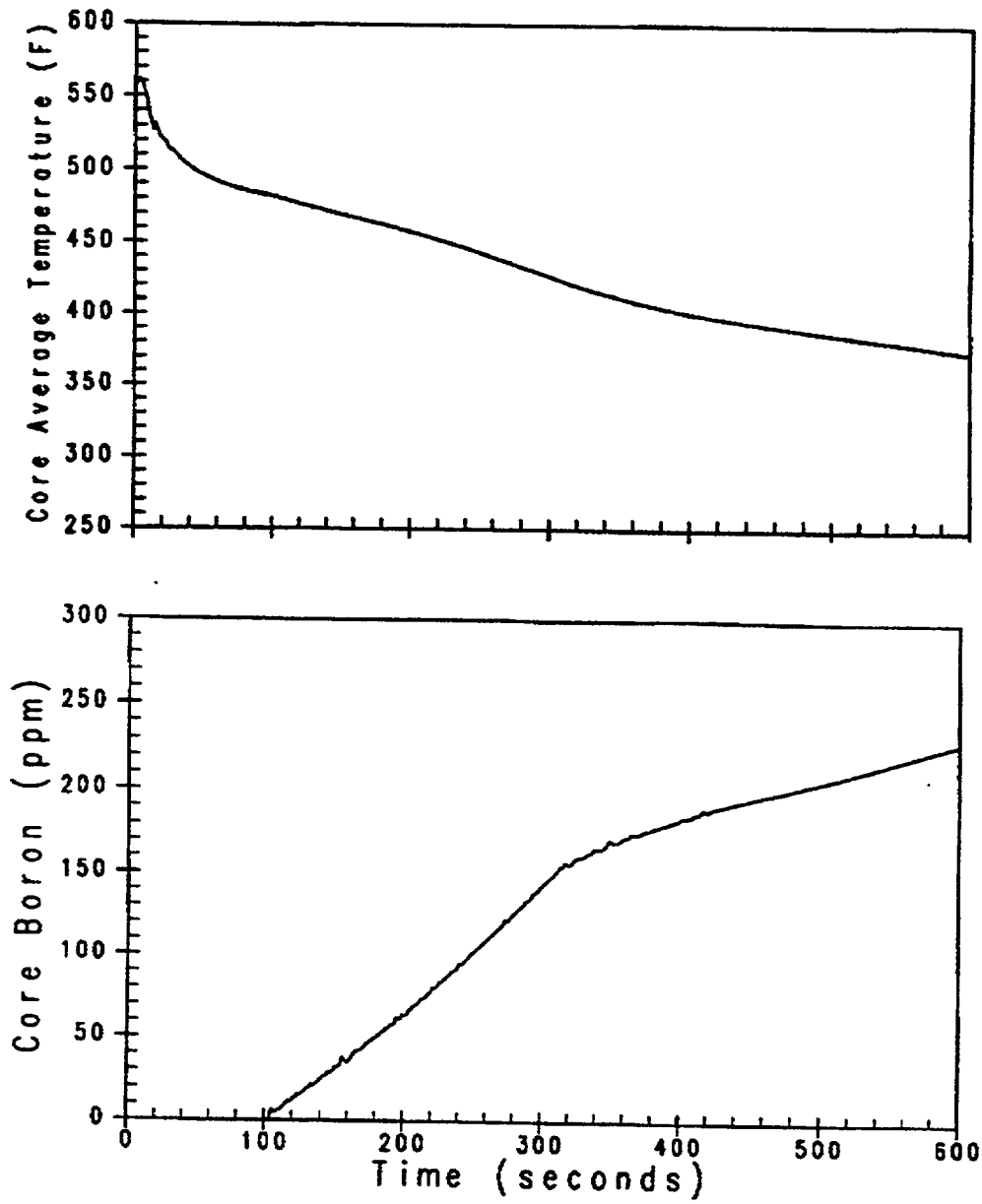
SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Steam Pressure and RV Inlet Temperature Transients for a Steam Line Rupture at Zero Power with Offsite Power Available	
	REV. 07	FIGURE 15.1-6, Sh 5



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SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Core Average Temperature and Core Boron Transients for a Steam Line Rupture at Zero Power with Offsite Power Available	
	REV. 07	FIGURE 15.1-6, Sh 6

A more complete discussion of overpressure protection can be found in Reference 1.

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

A loss of external load event results in a nuclear steam supply system transient that is less severe than a turbine trip event (see Subsection 15.2.3). Therefore, a detailed transient analysis is not presented for the loss of external load.

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

The protection available to mitigate the consequences of a loss of external load is the same as that for a turbine trip, as listed in Table 15.0-5.

15.2.2.2 Analysis of Effects and Consequences

a. Method of Analysis

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

Normal Reactor Control Systems and Engineered Safety Systems are not required to function. The Emergency Feedwater System may, however, be automatically actuated following a loss of main feedwater, which will further mitigate the effects of the transient.

The Reactor Protection System may be required to function following a complete loss of external load to terminate core heat input and prevent Departure from Nucleate Boiling (DNB). Depending on the magnitude of the load loss, pressurizer safety valves and/or steam generator safety valves may be required to open to maintain system pressure below allowable limits. No single active failure will

prevent operation of any system required to function.

15.2.2.3 Radiological Consequences

The radiological consequences resulting from this malfunction are considerably less than those calculated for a main steam line rupture. The analyses performed assuming a rupture of a main steam line are given in Subsection 15.1.5.3.

15.2.2.4 Conclusions

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

15.2.3 Turbine Trip

15.2.3.1 Identification of Causes and Accident Description

For a turbine trip event, the reactor would be tripped directly (unless below the P-9 setpoint) from a signal derived from the turbine emergency trip fluid pressure and turbine stop valves. The turbine stop valves close rapidly (typically 0.1 seconds) on loss of trip fluid pressure actuated by one of a number of possible turbine trip signals. Turbine trip initiation signals include:

a. Electrical faults associated with the generator or transformers

b. Low condenser vacuum

c. Loss of lubricating oil

d. Turbine thrust bearing failure

e. Turbine overspeed

f. Main steam reheat high level

g. Manual trip.

Upon initiation of stop valve closure, steam flow to the turbine stops abruptly. Sensors on the stop valves detect the turbine trip and initiate steam dump and, if above the P-9 setpoint, a reactor trip. The loss of steam flow results in a rapid rise in secondary system temperature and pressure. The turbine trip event is analyzed because it results in the most rapid reduction in steam flow.

The automatic Steam Dump System would normally accommodate the excess steam generation when the unit is operating below the P-9 setpoint. 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 were not available, the excess steam generation would be dumped to the atmosphere and main feedwater flow would be lost. For this situation, feedwater flow would be maintained by the Auxiliary Feedwater System to ensure adequate residual and decay heat removal capability. Should the Steam Dump System fail to operate, the steam generator safety valves may lift to provide pressure control. See Subsection 15.2.2.1 for a further discussion of the transient.

A turbine trip is classified as an ANS Condition II event, a fault of moderate frequency. See Subsection 15.0.1 for a discussion of Condition II events.

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

The plant systems and equipment available to mitigate the consequences of a turbine trip are discussed in Subsection 15.0.8 and listed in Table 15.0-5.

15.2.3.2 Analysis of Effects and Consequences

a. Method of Analysis

In this analysis, two cases are analyzed. In one case, the behavior of the unit is evaluated for a complete loss of steam load from 102 percent or rated thermal power without direct reactor trip, primarily to show the adequacy of the pressure relieving devices to limit the maximum RCS pressure and steam generator pressures to 110 percent of their design values. The second case analyzes the accident with respect to determining the minimum DNBR. In both cases the turbine trip is assumed to trip without actuating any of the sensors for reactor trip on the turbine stop valves. This assumption delays reactor trip until conditions in the RCS result in a trip due to other signals. Thus, the analysis assumes a worst case transient.

In addition, no credit is taken for steam dump. Main feedwater flow is terminated at the time of turbine trip, with no credit taken for auxiliary feedwater to mitigate the consequences of the transient.

The turbine trip transients are analyzed by employing the detailed digital computer program LOFTRAN[2]. The program simulates the neutron kinetics, RCS, pressurizer, pressurizer relief and safety valves, pressurizer spray, steam generator, and steam generator safety valves. The program computes pertinent plant variables

including temperatures, pressures, and power level.

The following turbine trip cases are analyzed:

A. Minimum reactivity feedback, with RCS pressure control

B. Minimum reactivity feedback, with no RCS pressure control

Case A is performed to calculate a conservative minimum DNBR, and is analyzed using the revised thermal design procedure as described in WCAP-11397[3]. Case B is analyzed to calculate a conservative maximum RCS and steam generator pressure.

Major assumptions are summarized below:

1. Initial Operating Conditions - For case A, the initial core power, reactor coolant temperature, and pressurizer pressure are assumed to be at their nominal full power values. Uncertainties in initial conditions are included in the limit Departure from Nucleate Boiling Ratio (DNBR) as described in Reference 3. For case B, initial conditions of core power, reactor coolant temperature, and pressurizer pressure are obtained by applying maximum errors to the nominal full-power values in the conservative direction.

2. Moderator and Doppler Coefficients of Reactivity - The turbine trip is analyzed with minimum reactivity feedback, which assumes a positive moderator temperature coefficient (negative moderator density coefficient) and a least negative Doppler coefficient.

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

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

5. Pressurizer Spray and Power-Operated Relief Valves - Two cases are analyzed:

   a. For evaluating the minimum DNBR, full credit is taken for the effect of pressurizer spray and power-operated relief valves in reducing or limiting the coolant pressure. Pressurizer safety valves are also available.

b. For evaluating maximum RCS and steam generator pressures, no credit is taken for the effect of pressurizer spray and power-operated relief valves in reducing or limiting the coolant pressure. Pressurizer safety valves are operable.

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

7. Steam Flow - Steam flow is assumed to be lost at the time of turbine trip.

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

Except as discussed above, normal reactor control system and engineered safety systems are not required to function. A case is presented in which pressurizer spray and power-operated relief valves are assumed, but the more limiting case where these functions are not assumed is also presented.

The reactor protection system may be required to function following a turbine trip. Pressurizer safety valves and/or steam generator safety valves may be required to open to maintain system pressures below allowable limits. No single active failure will prevent operation of any system required to function.

b. Results

The transient responses for a turbine trip from full power operation are shown for two cases: 1) for minimum DNBR and 2) for maximum RCS and steam generator pressure. The calculated sequence of events for the accident is shown in Table 15.2-1.

Figures 15.2-1, sh.1 and 15.2-1, sh.2 show the transient responses for the turbine trip with minimum reactivity feedback assuming full credit for the pressurizer spray and pressurizer power-operated relief valves. No credit is taken for the steam dump. The reactor is tripped by the high pressurizer pressure trip signal. The

minimum DNBR remains well above the limit value. The steam generator safety valves limit the secondary steam conditions to saturation at the safety valve set pressure.

The turbine trip accident was also studied assuming the plant to be initially operating at 102 percent of rated thermal power 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 15.2-1, sh.3 and 15.2-1, sh.4 show the transients with minimum reactivity feedback. In this case, the pressurizer safety valves are actuated, and maintain system pressure below 110 percent of the design value.

15.2.3.3 Radiological Consequences

The radiological consequences resulting from this malfunction are considerably less than those calculated for a main steam line rupture. The analyses performed assuming a rupture of a main steam line are given in Subsection 15.1.5.3.

15.2.3.4 Conclusions

Results of the analyses show that the plant design is such that a turbine trip 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 safety analysis limit value. The above analysis demonstrates the ability of the Nuclear Steam Supply System to safely withstand a full load rejection.

15.2.4 Inadvertent Closure of Main Steam Isolation Valves

Inadvertent closure of the main steam isolation valves would result in a turbine trip. Turbine trips are discussed in Subsection 15.2.3.

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

Malfunction of the condenser vacuum pumps, improper valve positioning or excessive air leakage may result in loss of condenser vacuum.

The loss of condenser vacuum is one of the events that will cause a turbine trip. Other turbine trip initiating events are described in Section 10.2 and Subsection 15.2.3. In case of loss of condenser vacuum, the Condenser Steam Dump System cannot be used and the excess steam generated is discharged to the atmosphere through the relief and/or safety valves. On loss of condenser vacuum, an alarm will activate at 5.0"HgA, and at 7.5"HgA, turbine trip will occur.

A turbine trip due to loss of condenser vacuum does not entail more adverse effects than the general turbine trip accident analyzed in detail in Subsection 15.2.3, because in that analysis no credit is taken for condenser steam dump. Therefore, the analysis results and conclusions of Subsection 15.2.3 apply to the loss of condenser vacuum.

15.2.6 Loss of Nonemergency AC Power to The Plant Auxiliaries (Loss of Offsite Power)

15.2.6.1 Identification of Causes and Accident Description

A complete loss of nonemergency AC power may result in the loss of all power to the station auxiliaries, i.e., the reactor coolant pumps, condensate pumps, etc. The loss of power may be caused by a complete loss of the offsite grid accompanied by a turbine generator trip at the plant, or by a loss of the onsite AC distribution system.

For this event the decrease in heat removal by the secondary system is accompanied by a flow coastdown which further reduces the capacity of the primary coolant to remove heat from the core. The reactor will trip: (1) due to turbine trip; (2) upon reaching one of the trip setpoints in the primary and secondary systems as a result of the flow coastdown and decrease in secondary heat removal; or (3) due to loss of power to the control rod drive mechanisms as a result of the loss of power to the plant.

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

a. Plant vital instruments are supplied from emergency DC power sources.

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

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

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

The Emergency Feedwater System is started automatically as described below.

Both the motor-driven emergency feedwater pump and the turbine-driven emergency feedwater pump are started on any of the following:

a. Low-low level in any steam generator

b. Any safety injection signal (SIS)

c. Manual actuation.

Refer to Section 6.8 for a discussion of the Emergency Feedwater System.

The motor-driven emergency feedwater pump is supplied power from the ESF buses. The turbine-driven emergency feedwater pump is driven by steam from the secondary system and exhausts to the atmosphere. Both types of pumps will start and supply rated flow within 75 seconds of the initiating signal. The emergency pumps take suction from the condensate storage tank for delivery to the steam generators.

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

A loss of nonemergency AC power to the station auxiliaries is classified as an ANS Condition II event, a fault of moderate frequency. See Subsection 15.0.1 for a discussion of Condition II events.

A loss of AC power event is a more limiting event than the turbine trip initiated decrease in secondary heat removal without loss of AC power, which was analyzed in Subsection 15.2.3. A loss of AC power to the station auxiliaries as postulated above could also result in a loss of normal feedwater if the condensate pumps lose their power supply.

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

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

The above assumptions are more conservative than an actual loss of nonemergency AC because the reactor power is maintained following the loss of AC and loss of feedwater. This minimizes the steam generator heat transfer

capability and increases the amount of RCS stored energy at the time of reactor trip and loss of primary coolant flow.

The plant systems and equipment available to mitigate the consequences of a loss of AC power event are discussed in Subsection 15.0.8 and listed in Table 15.0-5.

15.2.6.2 Analysis of Effects and Consequences

a. Method of Analysis

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

The assumptions used in the analysis are as follows:

1. The plant is initially operating at 102 percent of the ESF design rating.

2. Core residual heat is based on the 1979 version of ANS 5.1[4]. ANSI/ANS-5.1-1979 is a conservative representation of the decay energy release rates. Long term operation at the initial power level preceding the reactor trip is assumed.

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

4. Emergency feedwater is delivered by one emergency feedwater pump. A total flow of 650 gpm is assumed to be delivered equally to all four steam generators.

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

6. The initial pressurizer pressure is assumed to be 50 psi higher and lower than the nominal value to determine the limiting case.

7. The initial reactor coolant average temperature is assumed to be 5.8°F higher and lower than the nominal value assumed for the ESF design rating to determine the limiting case.

8. The most negative moderator density coefficient, the least negative Doppler temperature coefficient, and the most negative Doppler-only power were assumed for conservatism.

b. Results

The transient responses of the RCS and the secondary side following a loss of nonemergency AC power are shown in Figure 15.2-5.

The first few seconds after the loss of power to the reactor coolant pumps will closely resemble a simulation of the complete loss of flow incident (see Subsection 15.3.2), i.e., core damage due to rapidly increasing core temperatures is prevented by promptly tripping the reactor. After the reactor trip, stored and residual decay heat must be removed to prevent damage to either the RCS or the core.

Natural circulation flow is available and is sufficient to provide adequate core decay heat removal following reactor trip and reactor coolant pump coastdown.

Since DNBR does not fall below the limit value, no fuel or clad damage occurs. The calculated sequence of events for this accident is listed in Table 15.2-1.

15.2.6.3 Radiological Consequences

The radiological consequences resulting from this malfunction are considerably less than those calculated for a main steam line rupture. The analyses performed assuming a rupture of a main steam line are given in Subsection 15.1.5.3.

15.2.6.4 Conclusions

Analysis of the natural circulation capability of the RCS has demonstrated that sufficient heat removal capability exists following reactor coolant pump coastdown to prevent fuel or clad damage. In addition, no water is lost from the RCS through the pressurizer relief or safety valves.

15.2.7 Loss of Normal Feedwater Flow

15.2.7.1 Identification of Causes and Accident Description

A loss of normal feedwater (from pump failures, valve malfunctions, or loss of offsite AC power) results in a reduction in capability of the secondary system to remove the heat generated in the reactor core. If an alternative supply of feedwater were not supplied to the plant, core residual heat following reactor trip would heat the primary system water to the point where water relief from the pressurizer would occur, resulting in a substantial loss of water from the RCS. 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 events occur upon loss of normal feedwater (assuming main feedwater pump failures or valve malfunctions):

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

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

A loss of normal feedwater is classified as an ANS Condition II event, a fault of moderate frequency. See Subsection 15.0.1 for a discussion of Condition II events.

A reactor trip and emergency feedwater actuation on low-low water level in any steam generator provides protection for a loss of normal feedwater.

The Emergency Feedwater System is started automatically as discussed in Subsection 15.2.6.1. The motor-driven emergency feedwater pump is supplied power from the ESF buses. The turbine-driven emergency feedwater pump is driven by steam from the secondary system and exhausts to the atmosphere. The pumps take suction directly from the condensate storage tank for delivery to the steam generators.

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

15.2.7.2 Analysis of Effects and Consequences

a. Method of Analysis

A detailed analysis using the LOFTRAN Code, Reference 2, is performed in order to obtain the plant transient following a loss of normal feedwater. The simulation describes the plant thermal kinetics, RCS, 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.

Assumptions made in the analysis are:

1. The plant is initially operating at 102 percent of the ESF design rating.

2. Core residual heat is based on the 1979 version of ANS 5.1[4]. ANSI/ANS-5.1-1979 is a conservative representation of decay energy release rates. Long-term operation at the initial power level preceding the trip is assumed.

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

4. Emergency feedwater is delivered by one emergency feed pump. A total flow of 650 gpm is assumed to be delivered equally to all four steam generators.

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

6. The initial pressurizer pressure is assumed to be 50 psi higher and lower than the nominal value to determine the limiting case.

7. The initial reactor coolant average temperature is assumed to be 5.8°F higher and lower than the nominal value assumed for ESF design rating to determine the limiting case.

8. The most negative moderator density coefficient, the least negative Doppler temperature coefficient, and the most negative Doppler-only power were assumed for conservatism.

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

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

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

Plant systems and equipment which are available to mitigate the effects of a loss of normal feedwater accident are discussed in Subsection 15.0.8 and listed in Table 15.0-5. Normal reactor control systems are not required to function. The Emergency Feedwater System is required to deliver a minimum emergency feedwater flow rate. No single active failure will prevent

operation of any system required to function.

b. Results

Figure 15.2-6, sh.1-4 shows the significant plant parameters following a loss of normal feedwater.

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. Within 75 seconds following the initiation of the low-low level trip, the emergency feedwater pumps are automatically started, reducing the rate of water level decrease. The capacity of the emergency feedwater pumps is such that the water level in the steam generators being fed 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 pressurizer safety valves. Figure 15.2-6 shows that at no time is there water relief from the pressurizer.

The calculated sequence of events for this accident is listed in Table 15.2-1.

As shown in Figures 15.2-6, sh. 1-3 the plant approaches a stabilized condition following reactor trip and emergency feedwater initiation at hot standby with the emergency feedwater removing decay heat. The plant may be maintained at hot standby or further cooled through manual control of the emergency feed flow. The operating procedures would 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 Emergency 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.

15.2.7.3 Radiological Consequences

The radiological consequences resulting from this malfunction are considerably less than those calculated for a main steam line rupture. The analyses performed assuming a rupture of a main steam line are given in Subsection 15.1.5.3.

15.2.7.4 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 emergency feedwater capacity is such that sufficient core heat removal is maintained, the RCS does not overpressurize, and reactor coolant water is not relieved from the pressurizer relief or safety valves.

15.2.8 Feedwater System Pipe Break

15.2.8.1 Identification of Causes and Accident Description

A major feedwater line rupture is defined as a break in a feedwater line large enough to prevent the addition of sufficient feedwater to the steam generators to maintain shell-side fluid inventory in the steam generators. If the break is postulated in a feedwater line between the check valve and the steam generator, fluid from the steam generator may also be discharged through the break. Further, a break in this location could preclude the subsequent addition of emergency feedwater to the affected steam generator. Also, the Emergency Feedwater (EFW) flow is assumed to be lost through the break prior to isolation of EFW flow to the faulted steam generator. A break upstream of the feedwater line check valve would affect the NSSS only as a loss of feedwater, which is covered by the analyses in Sections 15.2.6 and 15.2.7.

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

This event is analyzed in order to evaluate the capacity of the emergency feedwater system to remove core decay heat, to prevent overpressurization of the RCS and secondary systems and to prevent reactor core uncovery. A more limiting criterion is that the maximum hot leg temperature remains below the saturation temperature until the EFW heat removal capability exceeds the RCS heat generation.

A major feedwater line rupture is classified as an ANS Condition IV event. See Subsection 15.0.1 for a discussion of Condition IV events.

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

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

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

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

An emergency feedwater system is provided to assure that adequate feedwater will be available such that:

a. No substantial overpressurization of the RCS shall occur; and

b. Sufficient liquid in the RCS shall be maintained in order to provide adequate decay heat removal.

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

The severity of the feedwater line rupture transient depends on a number of system parameters including break size, initial reactor power, and credit taken for the functioning of various control and safety systems. Sensitivity studies presented in WCAP-9230[5] illustrate that the most limiting feedwater line rupture is a double-ended rupture of the largest feedwater line. Analyses were performed at full power with and without loss of offsite power and no credit taken for pressurizer power-operated relief valves.

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

a. A reactor trip on any of the following conditions:

1. High pressurizer pressure

2. Overtemperature ΔT

3. Low-low steam generator water level in any steam generator

4. Safety injection signals from any of the following:

(a) Two out of three low steam line pressure in any one loop,
(b) Two out of three high containment pressure (hi-1), or
(c) Low pressurizer pressure.

Refer to Chapter 7 for a discussion of the actuation system.

b. An Emergency Feedwater System to provide an assured source of feedwater to the steam generators for decay heat removal. Refer to Section 6.8 for a description of the Emergency Feedwater System.

15.2.8.2 Analysis of Effects and Consequences

a. Method of Analysis

A detailed analysis using the LOFTRAN[2] code is performed in order to determine the plant transient following a feedwater line rupture. The code describes the plant thermal kinetics, RCS including natural circulation, pressurizer, steam generators, and feedwater system, and computes pertinent variables including the pressurizer pressure, pressurizer water level, and reactor coolant average temperature.

The cases analyzed assume a double-ended rupture of the largest feedwater pipe at full power. Major assumptions made in the analyses are as follows:

1. The plant is initially operating at 102% of engineered safeguards power (104.5% of NSSS power).

2. Initial reactor coolant average temperature is 5.8 degrees F above the nominal value, and the initial pressurizer pressure is 50 psi above its nominal value.

3. No credit is taken for the pressurizer power-operated relief valves or pressurizer spray.

4. Initial pressurizer level is at the nominal programmed value; initial steam generator water level is at the nominal value plus 5% in the faulted steam generator, and at the nominal value minus 5 percent in the intact steam generators; steam generator masses in both the faulted and intact loops are reduced by 10% to account for modeling uncertainties.

5. The worst case assumes maximum reactivity feedback - most positive moderator density coefficients, most negative Doppler temperature coefficients, most negative Doppler-only power coefficients and minimum delayed neutron beta-effective values.

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

7. The worst possible break area, a double-ended break downstream of the EFW connection, is assumed. This maximizes the blowdown discharge rate following the time of trip, which maximizes the resultant heatup of the reactor coolant.

8. Choked flow is assumed at the break.

9. The analysis assumes a conservatively low value of 0% NRS for the steam generator low-low level setpoint, which actuated the EFW system.

10. EFW pump performance is based on loss of one train (single failure) and minimum flow versus steam generator back pressure injected to the three intact steam generators by the operational pump. Cold EFW is assumed to not reach the steam generators until the three feedwater branch lines have been swept clear of hot feedwater.

11. Turbine trip is assumed to occur 0.5 seconds after break initiation and no credit is taken for the atmospheric steam dump valves.

12. Safety Injection Actuation is credited on low pressurizer pressure.

13. Minimum high head ECCS pump performance and maximum ECCS temperature (98°F) are assumed.

14. No credit is taken for heat energy deposited in RCS metal during the RCS heatup.

15. No credit is taken for charging or letdown.

16. Steam generator heat transfer area is assumed to decrease as the shell side liquid inventory decreases.

17. Conservative core residual heat generation is assumed based upon long-term operation at the initial power level preceding the reactor trip is assumed.

18. One of the redundant EFW flow control valves leading to the faulted steam generator is assumed to close on a high flow rate signal with a bounding stroke time of 25 seconds to terminate EFW flow through the break. This stroke time is conservatively modeled as an additional delay over and above the EFW signal delay (2 seconds) and the delay for EFW pump start (75 seconds).

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

    (a) High pressurizer pressure

    (b) Overtemperature ΔT

    (c) High pressurizer level.

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

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

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

The Engineered Safety Systems assumed to function are the Emergency Feedwater System and the Safety Injection System.

Two Emergency Feedwater System configurations were considered. In the first configuration, both emergency feedwater pumps were assumed to operate; however, the emergency feedwater flow control valve to one intact steam generator was assumed to fail closed (single failure). As a result, only two intact steam generators receive emergency feedwater following the break. The flow restrictor and control valves on the faulted loop limit the flow spilling out the break to 750 gpm prior to control valve closure. The flow through the open control valves to the remaining two intact loops is at least 235 gpm each, ensuring the minimum required flow of 470 gpm. The second configuration considered operation of only one of the two emergency feedwater pumps (single failure), providing flow to all three intact steam generators. Flow from the operating emergency feedwater pump will spill out of the break in the faulted loop prior to automatic closure of one of the redundant flow control valves. With the control valve closed the intact steam generators in combination will receive the minimum required flow of 470 gpm. The analysis presented was performed using the second configuration. This configuration is slightly more conservative because it maximizes the time elapsed prior to cold emergency feedwater reaching the intact steam generators.

A detailed description and analysis of the Safety Injection System is provided in Section 6.3. The Emergency Feedwater System is described in Section 6.8.

b. Results

Calculated plant parameters following a major feedwater line rupture are shown in Figure 15.2-7. The calculated sequence of events is listed in Table 15.2-1.

The RCS heatup prior to reactor trip is due to loss of subcooling as a result of MFW spillage through the break and the increased secondary temperature and pressure following the turbine trip. Reactor power increases prior to the trip due to the RCS heatup. The primary and secondary systems were calculated to remain below 110 percent of their respective design pressures.

Following the reactor trip, steam flow out the break cools the RCS and eventually causes the pressurizer to empty. However, the core remains covered with water. Low main steam line pressure causes closure of the MSIV's, ends the cooldown period, and starts safety injection. Addition of safety injection flow aids in cooling down the primary and ensures that sufficient fluid exists to keep the core covered with water.

The MSIV closure and resulting increase in steam generator pressure and temperature cause the second RCS heatup. As a result, the rising primary system pressure exceeds the shutoff head of the ECCS pumps and then increases to the pressurizer safety valve setpoint. The heatup ends when the intact steam generators reach their main steam safety valve (MSSV) setpoint and the combination of steam relief through the MSSV's and EFW injection match core decay heat plus RCP heat.

The maximum hot leg temperature remains below the saturation temperature throughout the transient. Therefore, no fuel damage will occur.

15.2.8.3 Radiological Consequences

No fuel failures are predicted for this event. The radiological consequences resulting from this malfunction are considerably less than those calculated for a main steam line rupture. The analyses performed assuming a rupture of a main steam line are given in Subsection 15.1.5.

15.2.8.4 Conclusions

Results of the analyses show that for the postulated feedwater line rupture, the Emergency Feedwater System capacity is adequate to remove decay heat, to prevent overpressurizing the RCS, and to prevent uncovering the reactor core. The maximum hot leg temperature remains below the saturation temperature. Therefore, no fuel damage will occur.

15.2.9 References

1. Mangan, M.A., "Overpressure Protection for Westinghouse Pressurized Water Reactors," WCAP-7769, October 1971.

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

3. WCAP-11397-P-A, "Revised Thermal Design Procedure," A. J. Friedland and S. Ray, April 1984

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

5. WCAP-9230, "Report on the Consequences of a Postulated Main Feedline Rupture," G. E. Lang and J. P. Cunningham, January 1978

The RCS heatup prior to reactor trip is due to loss of subcooling as a result of MFW spillage through the break; and the increased secondary temperature and pressure following the turbine trip. Reactor power increases prior to the trip due to the RCS heatup and the assumed positive MTC. The primary and secondary system pressures quickly reached their respective safety valve setpoints. However, the peak pressures in the RCS and secondary systems were calculated to remain below 110 percent of their respective design pressures.

Following the reactor trip, steam flow out the break cools the RCS and eventually causes the pressurizer to empty. However, the core remains covered with water. Low main steam line pressure causes closure of the MSIV's, ends the cooldown period, and starts safety injection. Addition of safety injection flow aids in cooling down the primary and ensures that sufficient fluid exists to keep the core covered with water.

The MSIV closure and resulting increase in steam generator pressure and temperature cause the second RCS heatup. As a result, the rising primary system pressure exceeds the shutoff head of the ECCS pumps and then increases to the pressurizer safety valve setpoint. The heatup ends when the intact steam generators reach their main steam safety valve (MSSV) setpoint and the combination of steam relief through the MSSV's and EFW injection match core decay heat plus RCP heat.

The minimum DNBR throughout the transient remains above the safety analysis limit value. Therefore, no fuel damage will occur.

15.2.8.3 Radiological Consequences

No fuel failures are predicted for this event. The radiological consequences resulting from this malfunction are considerably less than those calculated for a main steam line rupture. The analyses performed assuming a rupture of a main steam line are given in Subsection 15.1.5.

15.2.8.4 Conclusions

Results of the analyses show that for the postulated feedwater line break, the Emergency Feedwater System capacity is adequate to remove decay heat, to prevent overpressurizing the RCS, and to prevent uncovering the reactor core. The minimum DNBR remains above the safety analysis limit. Therefore, no fuel damage will occur. The radiological consequences from the postulated feedwater line rupture are less than those presented for the postulated steam line break. All applicable acceptance criteria are met.

TABLE 15.2-1
(Sheet 1 of 3)

TIME SEQUENCE OF EVENTS FOR INCIDENTS THAT CAUSE A DECREASE IN HEAT REMOVAL BY THE SECONDARY SYSTEM

Accident	Event	Time (sec)
a. Turbine Trip / Loss of Load		
A. With Pressure Control	Turbine trip, loss of main feedwater flow	0.0
	Initiation of steam release from pressurizer relief valves	5.0
	High pressurizer pressure reactor setpoint reached	6.1
	Rods begin to drop	8.1
	Minimum DNBR occurs	9.5
B. Without Pressure Control	Turbine trip, loss of main feedwater flow	0.0
	High pressurizer pressure reactor setpoint reached	5.0
	Initiation of steam release from pressurizer safety valves	6.7
	Rods begin to drop	7.0
	Maximum steam generator pressure occurs; steam generator safety valves actuate	7.4
	Peak RCS pressure occurs	7.7

TABLE 15.2-1
(Sheet 2 of 3)

TIME SEQUENCE OF EVENTS FOR INCIDENTS THAT CAUSE A DECREASE IN HEAT REMOVAL BY THE SECONDARY SYSTEM

Accident	Event	Time (sec)
b. Loss of Non-emergency AC Power to the Station Auxiliaries		
	AC power is list; main feedwater flow stops	10
	Low-low steam generator water level setpoint reached	62
	Rods begin to drop	64
	RCPs begin to coast down	66
	Peak pressurizer water level occurs	74
	4 SGs begin to receive emergency feedwater flow from one emergency feedwater pump	139
	Minimum SG inventory occurs	555
c. Loss of Normal Feedwater		
	Main feedwater flow stops	10
	Low-low steam generator water level setpoint reached	62
	Rods begin to drop	64
	Peak pressurizer water level occurs	68
	4 SGs begin to receive emergency feedwater flow from one emergency feedwater pump	139
	Minimum SG inventory occurs	980

TABLE 15.2-1
(Sheet 3 of 3)

TIME SEQUENCE OF EVENTS FOR INCIDENTS THAT CAUSE A DECREASE IN HEAT REMOVAL BY THE SECONDARY SYSTEM

Accident	Event	Time (sec)
d. Feedwater System Pipe Break		
	Main feedwater line rupture occurs	10.0
	Low-low steam generator water level reactor trip setpoint reached in broken loop	16.5
	Rods begin to drop	18.5
	Low pressurizer pressure reached for SIS injection	94.6
	Emergency feedwater flow is started	118.5
	Safety injection flow is started	121.6
	Low steam line isolation setpoint reached	181.0
	All main steam line isolation valves are closed	188.0
	Steam generator inventory in broken loop completely discharged through break	212.0
	Pressurizer safety valve setpoint reached	434.0
	Relatively cold emergency feedwater is delivered to the steam generators of intact loops	550.0
	Steam generator safety valves open in steam generators of intact loops	645.2
	Core decay heat plus pump heat decrease to emergency feedwater heat removal capacity	~3252

TABLE 15.2-2
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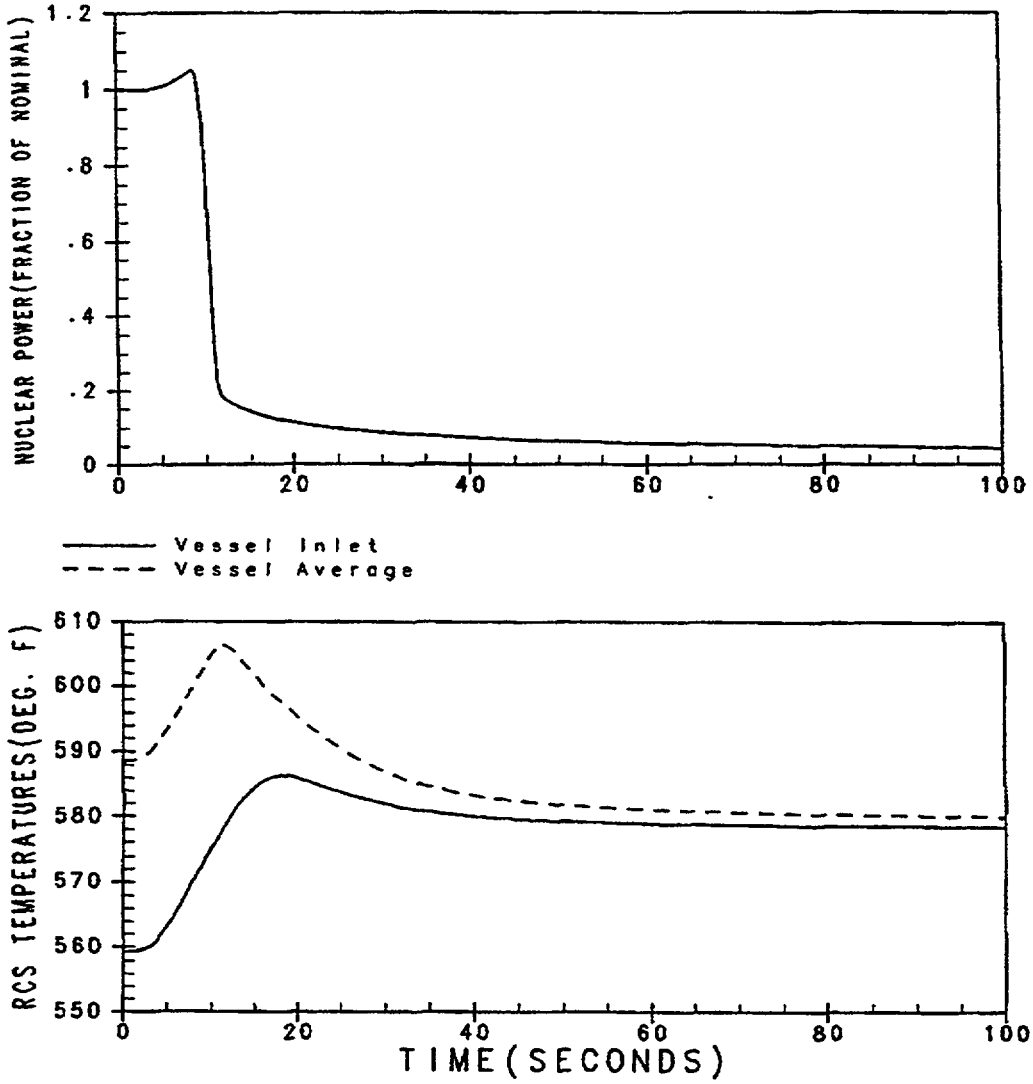

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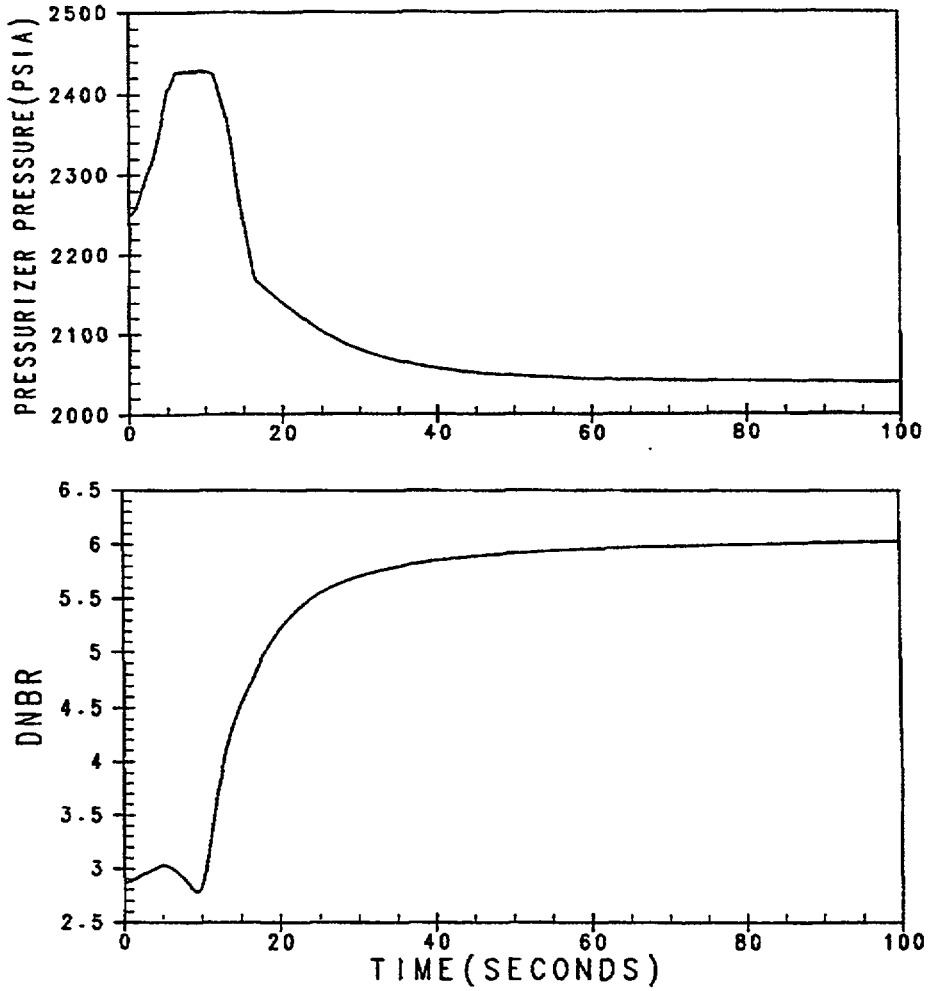
SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Nuclear Power and RCS Temperature Transients for a Turbine Trip Event With Pressure Control	
	REV. 07	FIGURE 15.2-1, Sh 1



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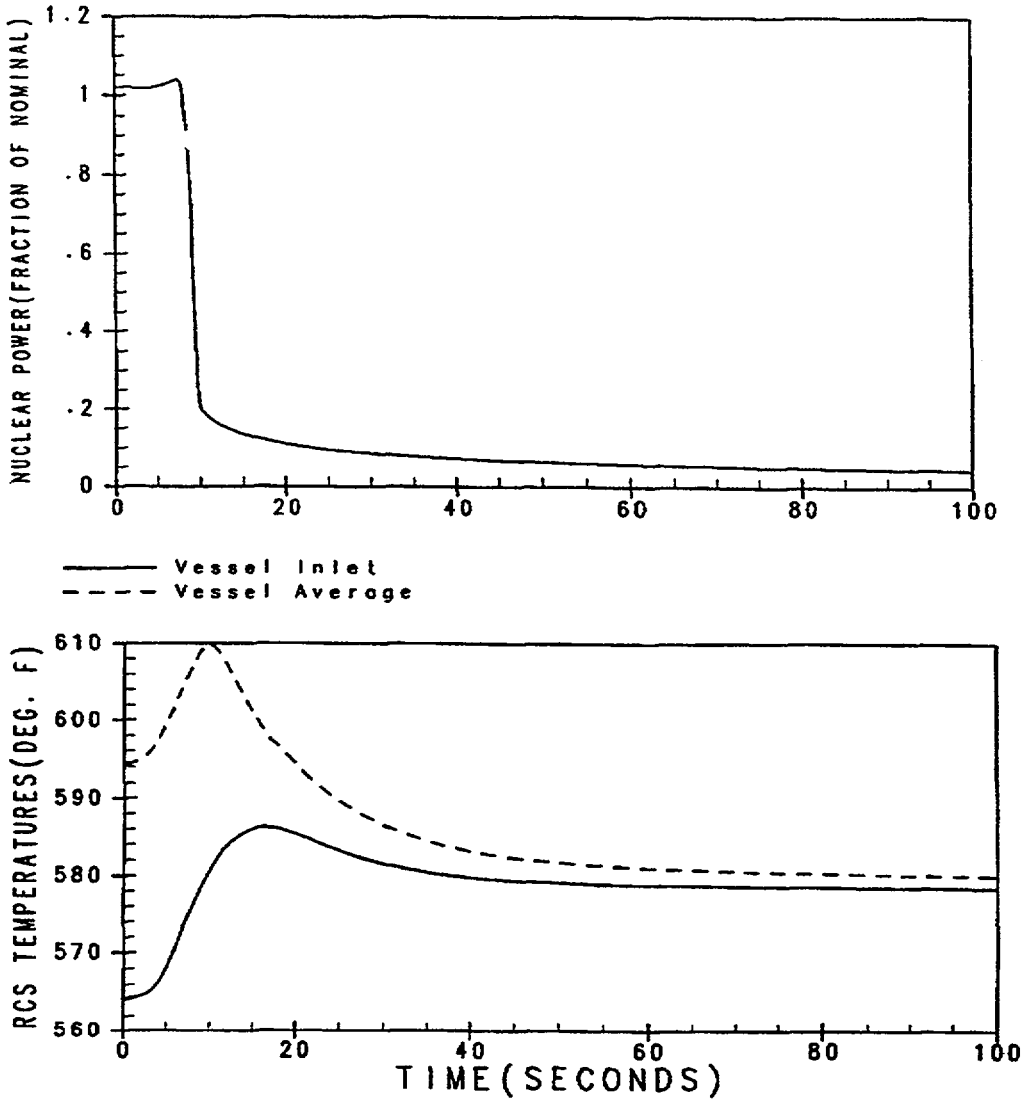
SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Pressurizer Pressure and DNBR Transients for a Turbine Trip Event With Pressure Control	
	REV. 07	FIGURE 15.2-1, Sh 2



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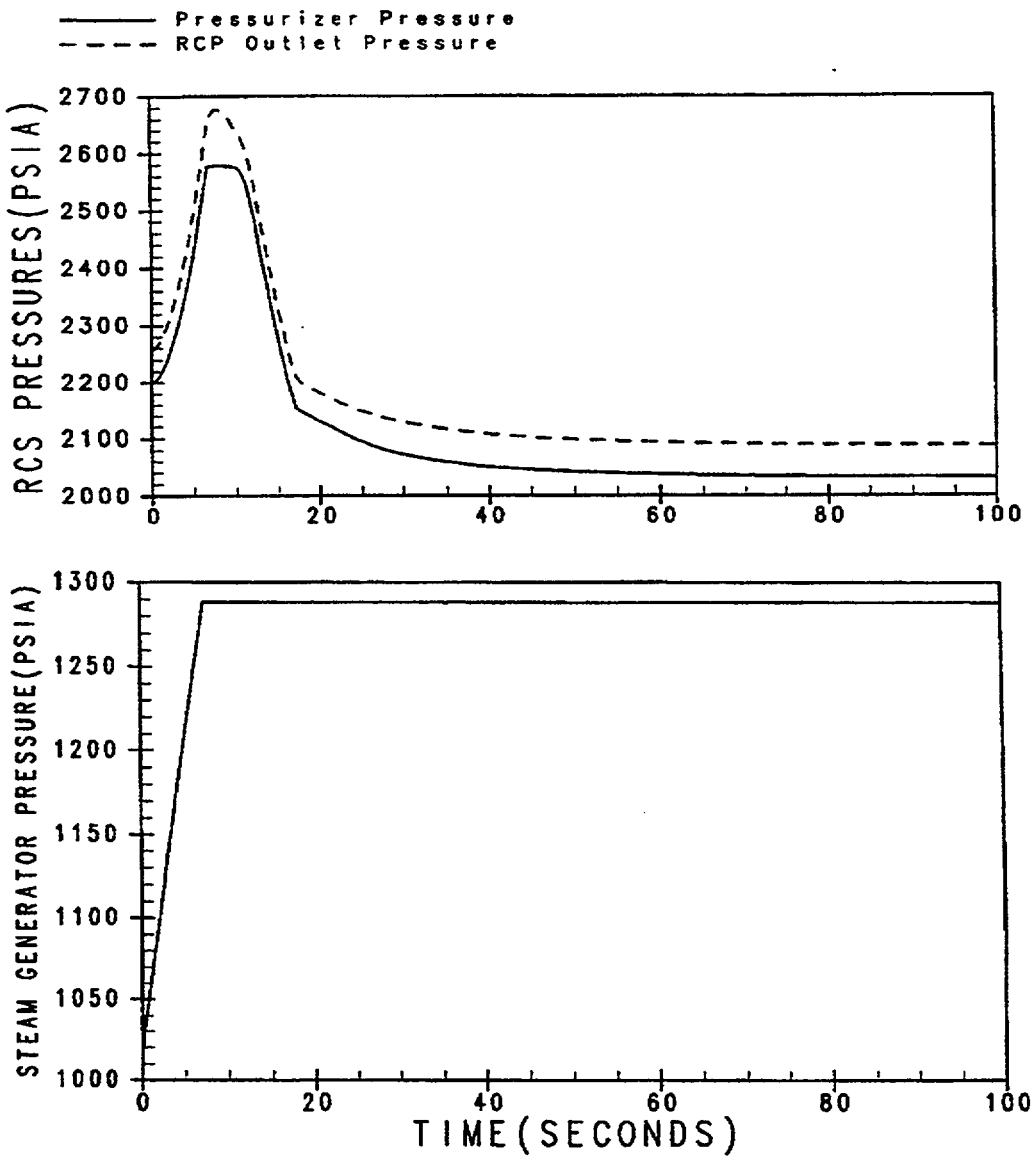
SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Nuclear Power and RCS Temperature Transients for a Turbine Trip Event Without Pressure Control	
	REV. 07	FIGURE 15.2-1, Sh 3



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SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	RCS Pressure and Steam Generator Pressure Transients for a Turbine Trip Event Without Pressure Control	
	REV. 07	FIGURE 15.2-1, Sh 4

FIGURE 15.2-2

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FIGURE 15.2-3

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FIGURE 15.2-4

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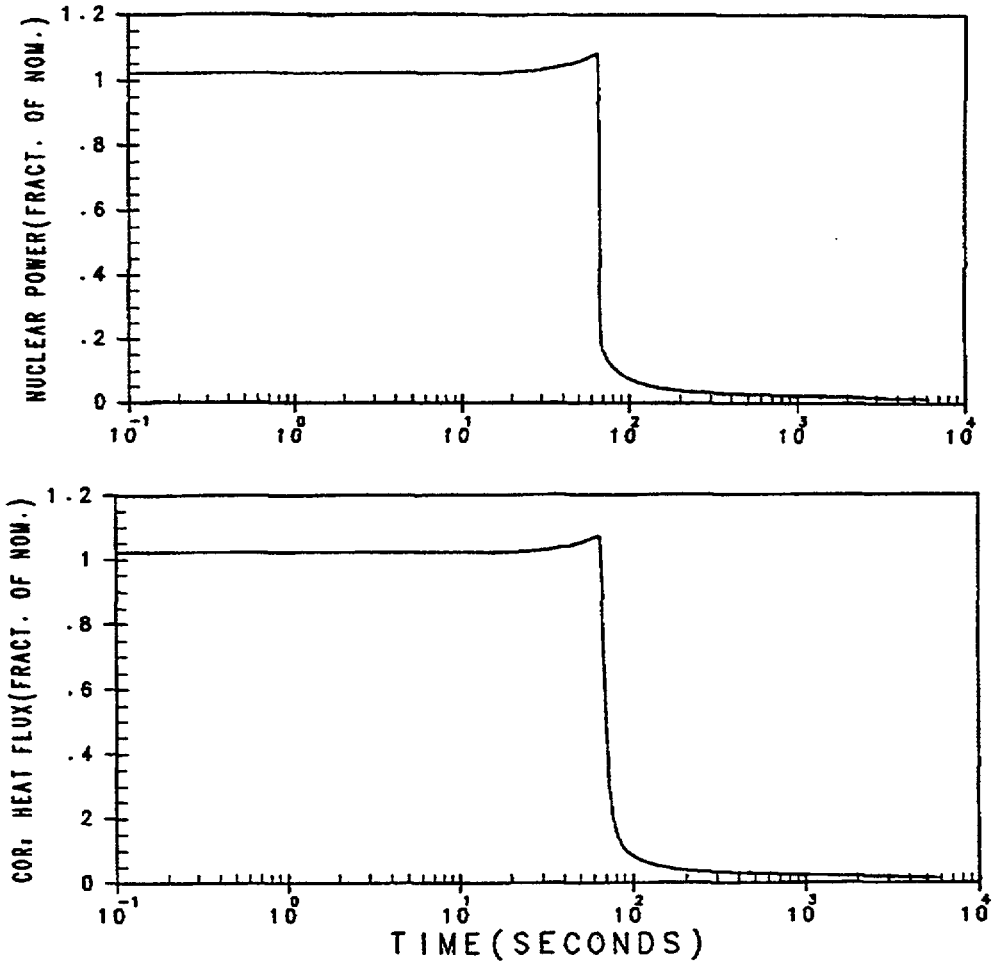

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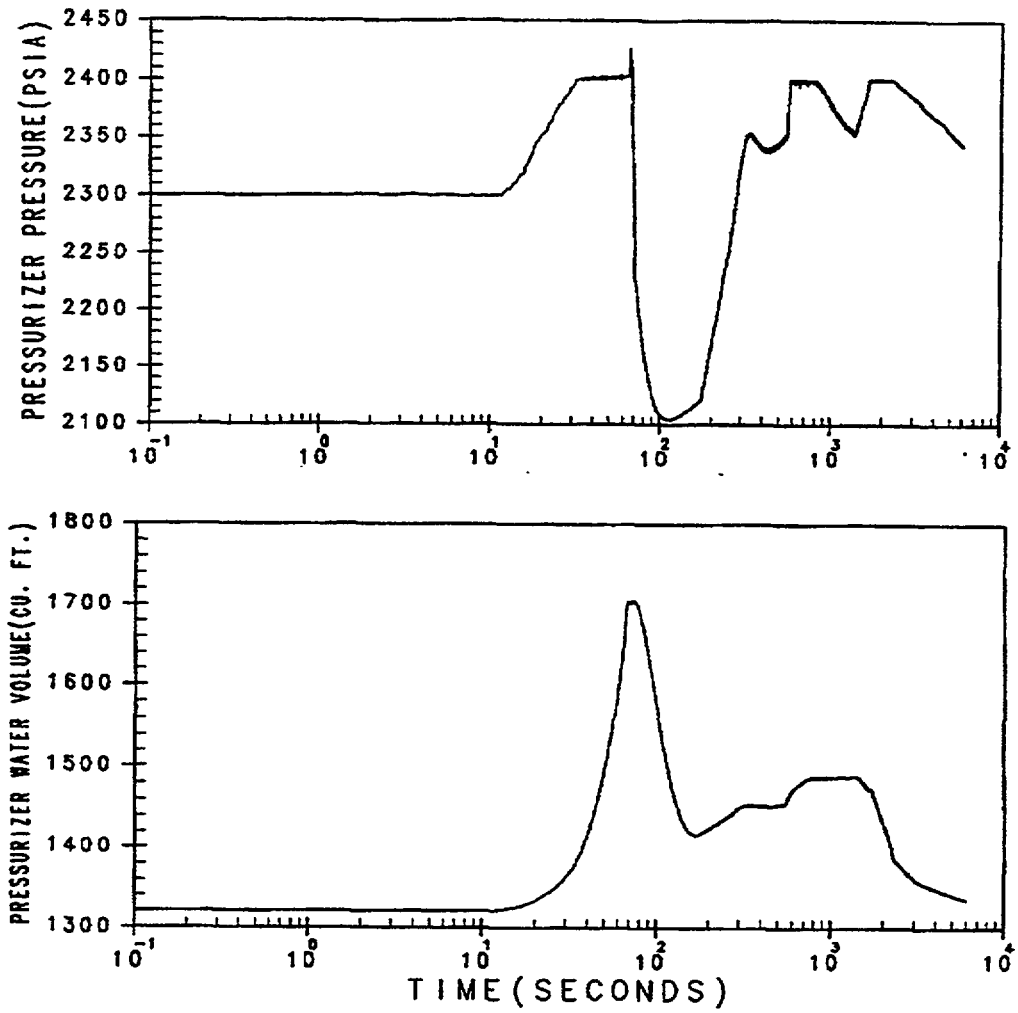
SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Nuclear Power and Core Heat Flux Transients for a Loss of Non-emergency AC to the Station Auxiliaries	
	REV. 07	FIGURE 15.2-5, Sh 1



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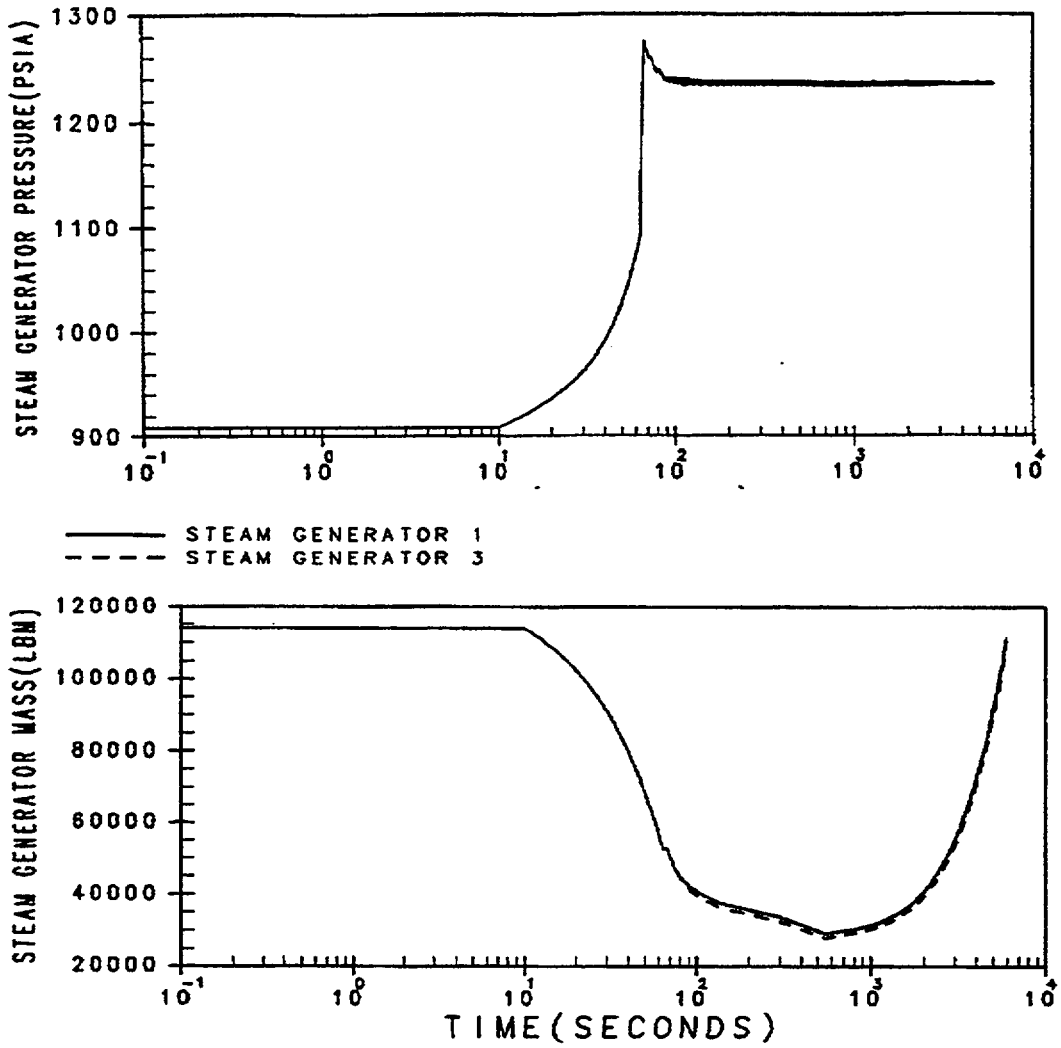
SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Pressureizer Pressure and Pressurizer Water Volume Transients for a Loss of Non-emergency AC to the Station Auxiliaries	
	REV. 07	FIGURE 15.2-5, Sh 2



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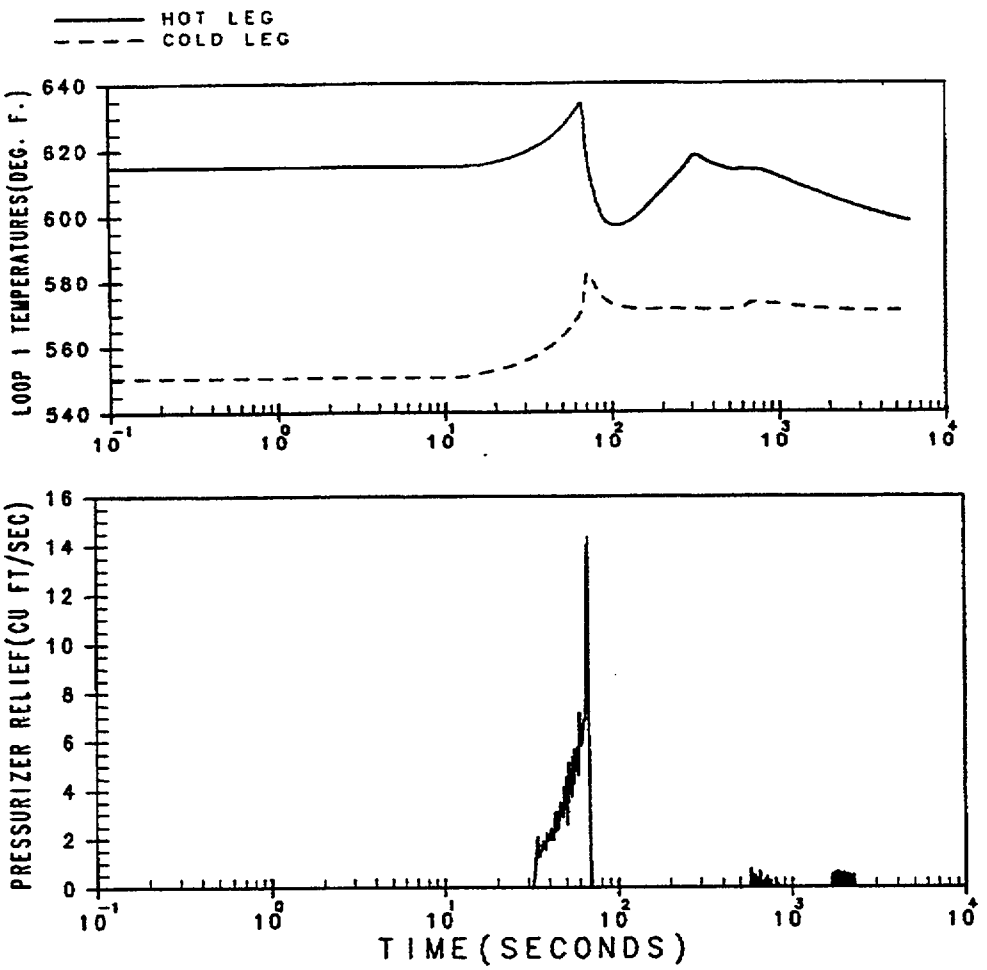
SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	SG Pressure and SG Mass Transients for a Loss of Non-emergency AC to the Station Auxiliaries	
	REV. 07	FIGURE 15.2-5, Sh 3



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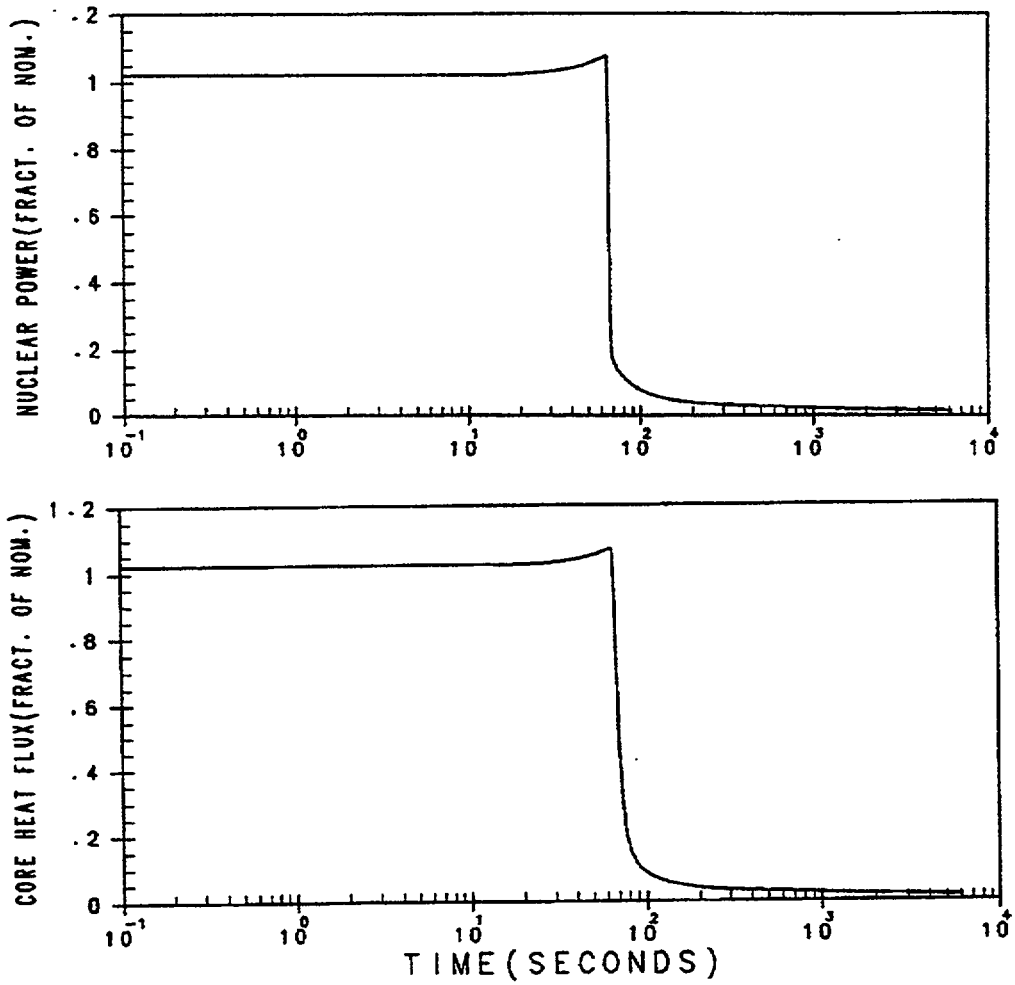
SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Loop 1 Temperature and Pressure Relief Transients for a Loss of Non-emergency AC to the Station Auxiliaries	
	REV. 07	FIGURE 15.2-5, Sh 4



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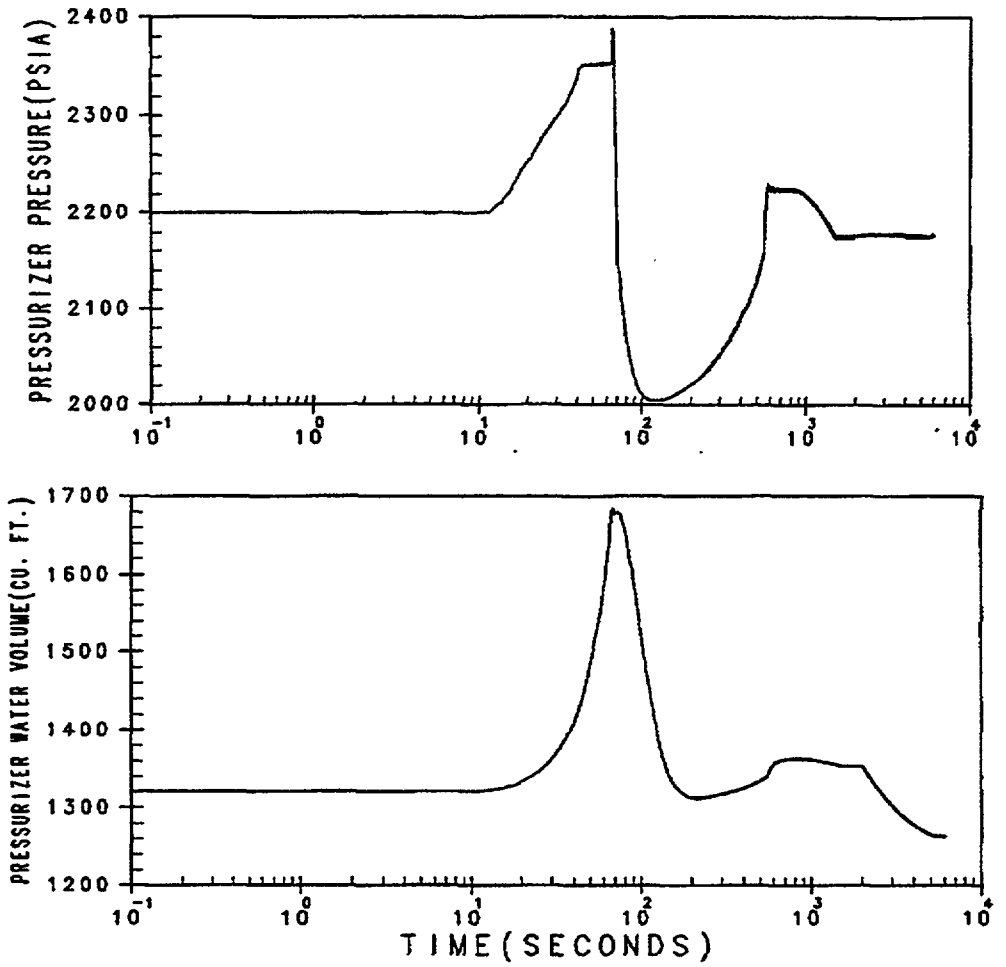
SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Nuclear Power and Core Heat Flux Transients for a Loss of Normal Feedwater	
	REV. 07	FIGURE 15.2-6, Sh 1



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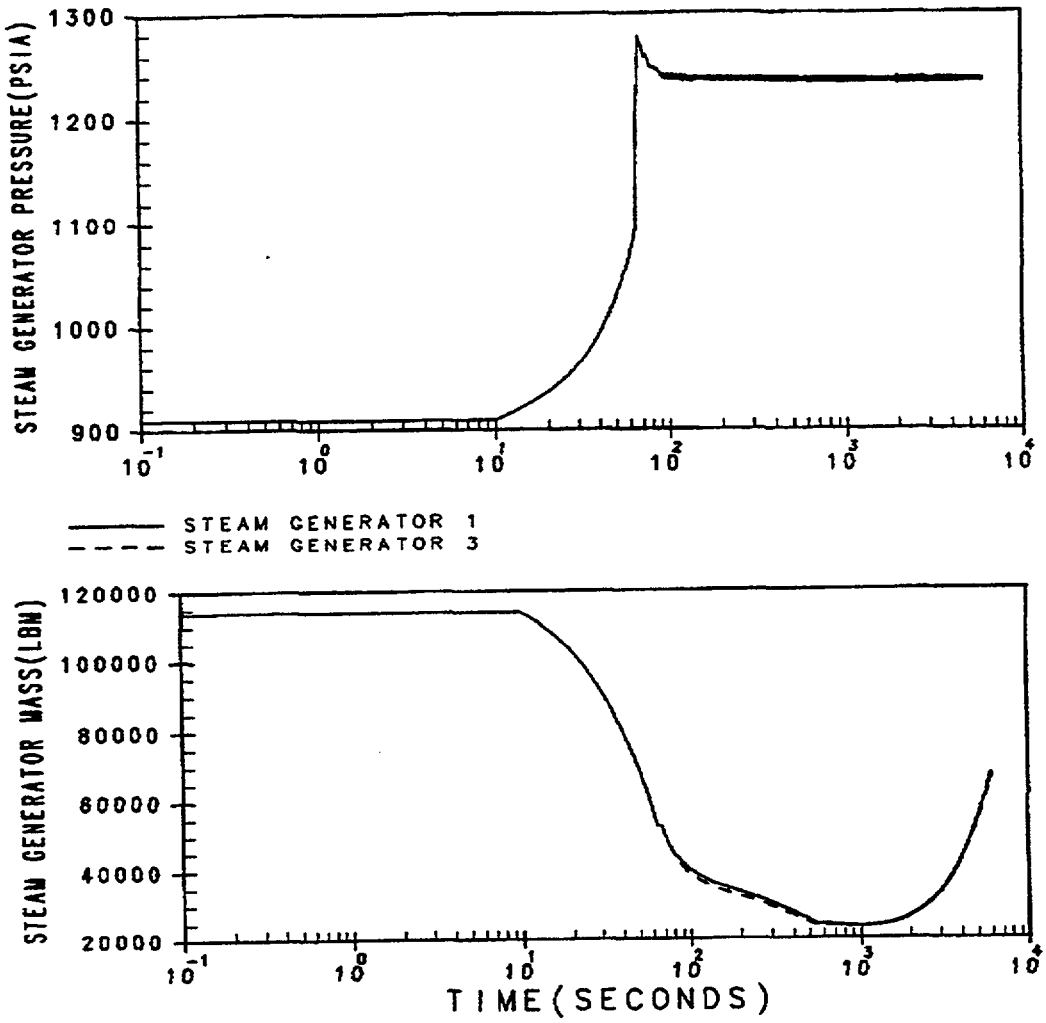
SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Pressurizer Pressure and Pressurizer Water Volume Transients for a Loss of Normal Feedwater	
	REV. 07	FIGURE 15.2-6, Sh 2



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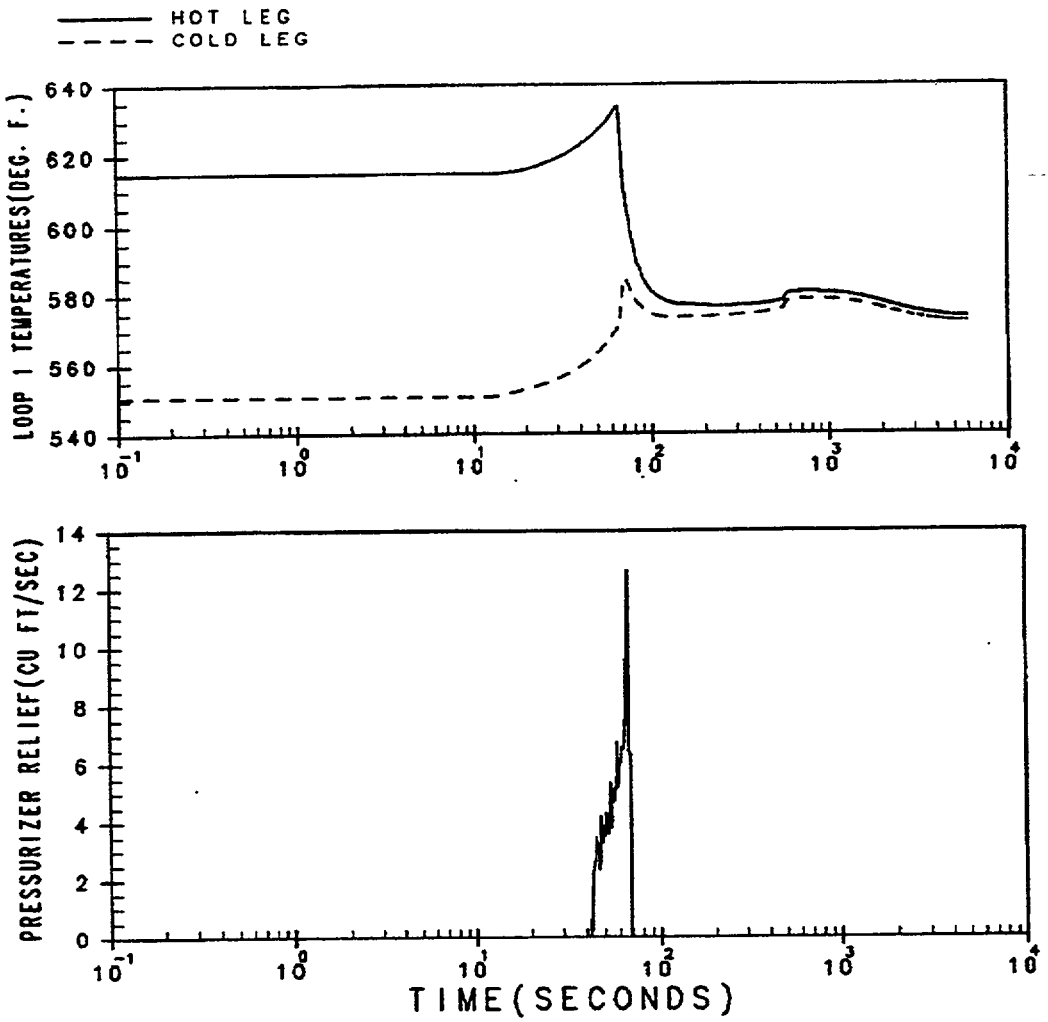
SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	SG Pressure and SG Mass Transients for a Loss of Normal Feedwater	
	REV. 07	FIGURE 15.2-6, Sh 3



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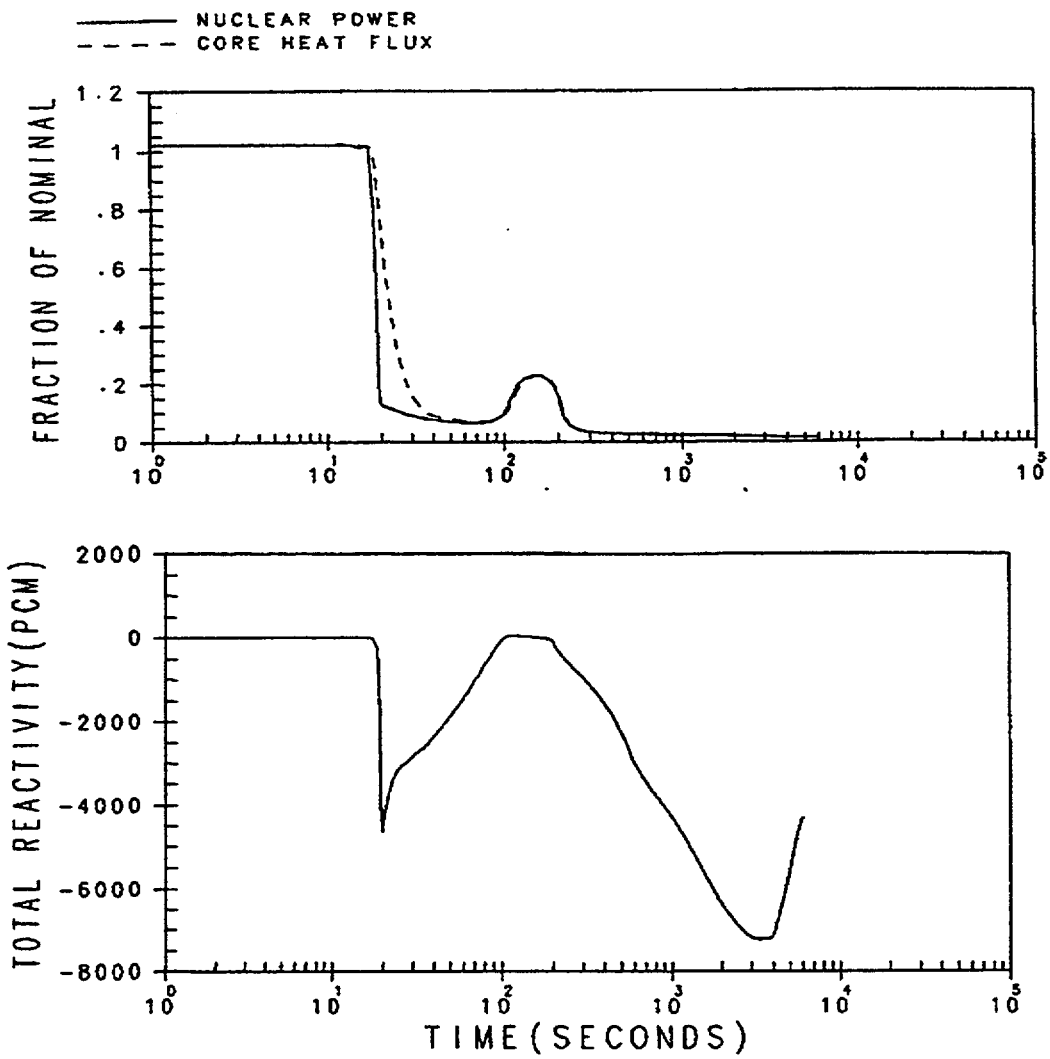
SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Loop 1 Temperature and Pressurizer Relief Transients for a Loss of Normal Feedwater	
	REV. 07	FIGURE 15.2-6, Sh 4



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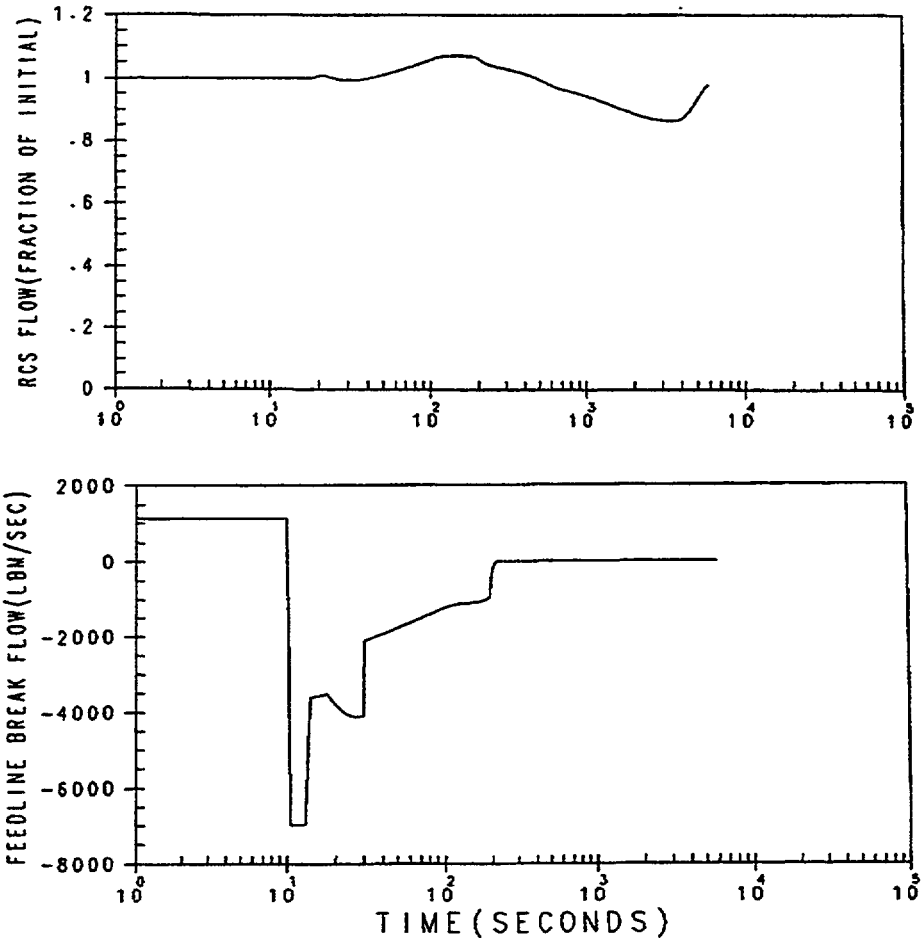
SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Nuclear Power, Core Heat Flux and Total Reactivity Transients for a Feedwater System Pipe Break (offsite power available)	
	REV. 07	FIGURE 15.2-7, Sh 1



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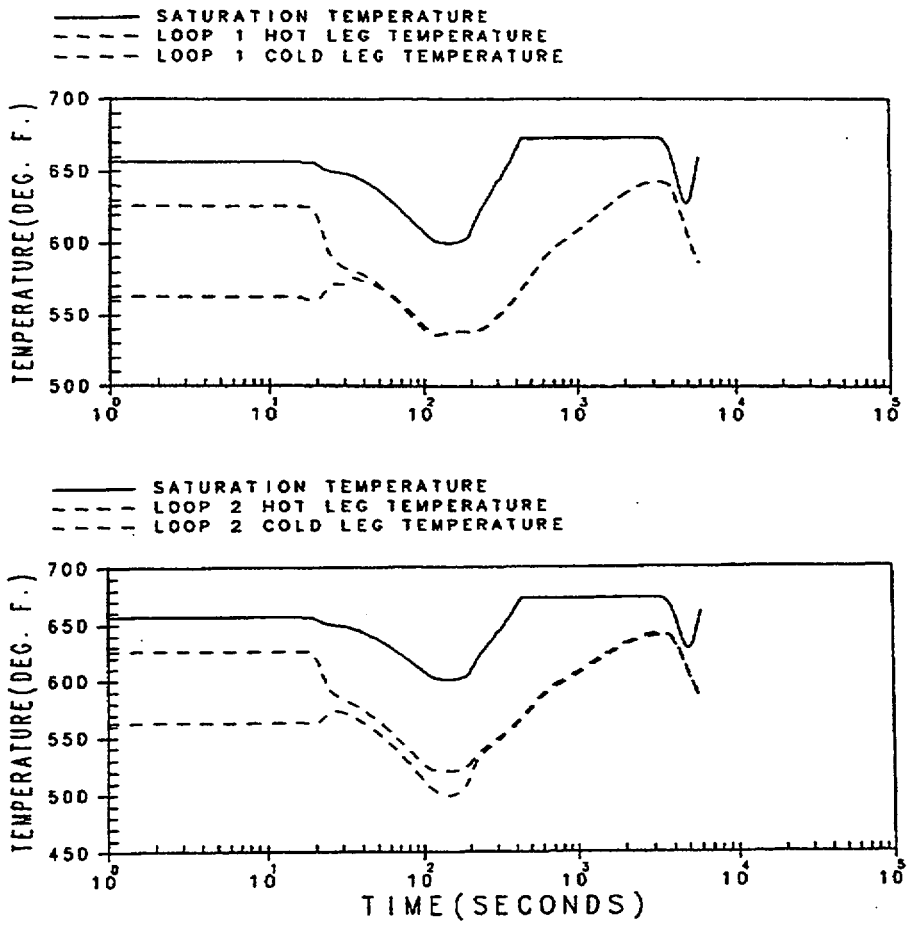
SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	RCS Flow and Feedline Break Flow Transients for a Feedwater System Pipe Break (offsite power available)	
	REV. 07	FIGURE 15.2-7, Sh 2



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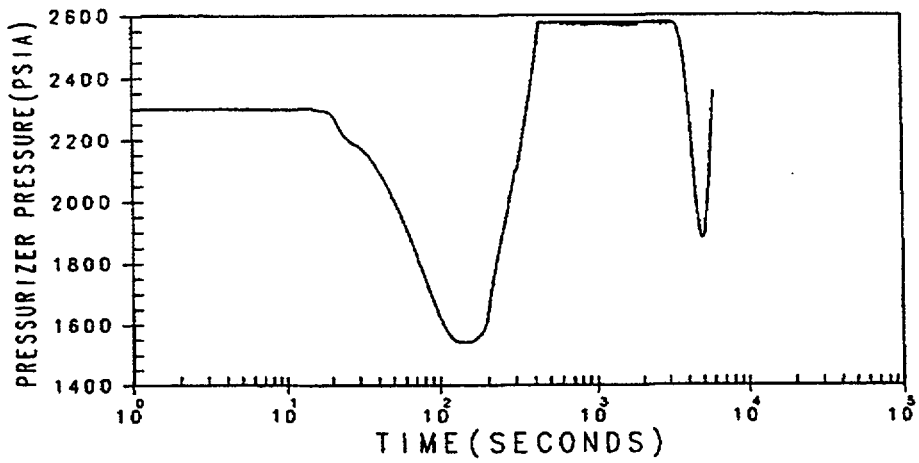
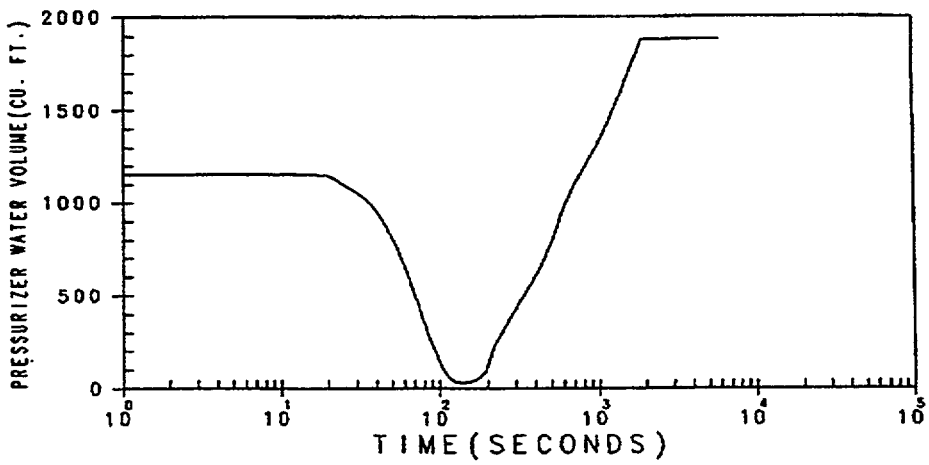
SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Loop 1 (faulted) and Loop 2 (unfaulted) Temperature Transients for a Feedwater System Pipe Break (offsite power available)	
	REV. 07	FIGURE 15.2-7, Sh 3





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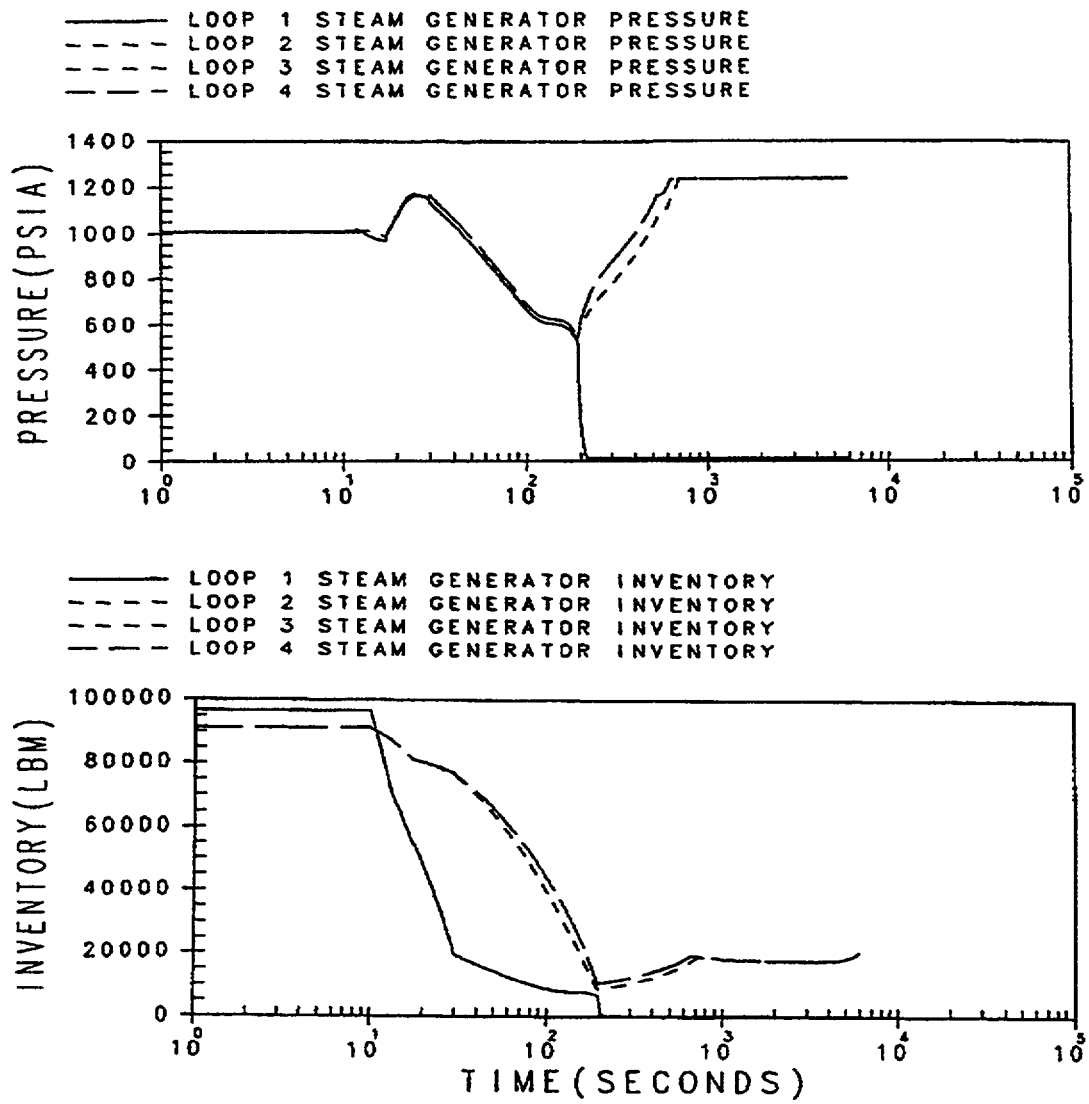
SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Pressurizer Pressure and Pressurizer Water Volume Transients for a Feedwater System Pipe Break (offiste power available)	
	REV. 07	FIGURE 15.2-7, Sh 4



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SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	SG Pressure and Inventory Transients for a Feedwater System Pipe Break (offsite power available)	
	REV. 07	FIGURE 15.2-7, Sh 5

15.3 DECREASE IN REACTOR COOLANT SYSTEM FLOW RATE

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

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

a. Partial Loss of Forced Reactor Coolant Flow

b. Complete Loss of Forced Reactor Coolant Flow

c. Reactor Coolant Pump Shaft Seizure (Locked Rotor)

d. Reactor Coolant Pump Shaft Break.

Item a above is considered to be an ANS Condition II event, item b an ANS Condition III event, and items c and d ANS Condition IV events. Subsection 15.0.1 contains a discussion of ANS classifications.

15.3.1 Partial Loss of Forced Reactor Coolant Flow

15.3.1.1 Identification of Causes and Accident Description

A partial loss-of-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.

The plant design is such that the four reactor coolant pumps are supplied through two buses, two pumps per bus, connected to the generator. When a generator trip occurs, the generator breaker is tripped open, the buses are automatically transferred to an offsite power source, and the pumps will continue to supply coolant flow to the core. Following any turbine trip, there is immediate generator trip and automatic transfer of the buses to offsite power.

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

The necessary protection against a partial loss-of-coolant flow accident is provided by the low primary coolant flow reactor trip signal which is actuated in any reactor coolant loop by two out of three low flow signals, in any reactor coolant loop. Above Permissive 8, low flow in any loop will actuate a reactor trip. Between approximately 10 percent power (Permissive 7) and the power level

corresponding to Permissive 8, low flow in any two loops will actuate a reactor trip. Above Permissive 7, either power supply low voltage on both buses or opening of one reactor coolant pump breaker on each bus will actuate the corresponding undervoltage relays, resulting in a reactor trip. Additionally, underfrequency on the two buses will actuate a reactor trip above P-7. These trips serve as a backup to the low flow trip.

15.3.1.2 Analysis of Effects and Consequences

a. Method of Analysis

Partial loss of flow involving loss of two pumps with four loops in operation has been analyzed.

This transient is analyzed by three digital computer codes. First the LOFTRAN[1] code is used to calculate the loop and core flow during the transient, the time of reactor trip based on the calculated flows, the nuclear power transient, and the primary system pressure and temperature transients. The FACTRAN[2] code is then used to calculate the heat flux transient based on the nuclear power and flow from LOFTRAN. Finally, the VIPRE[4] code is used to calculate the Departure from Nucleate Boiling Ratio (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.

This accident is analyzed with the revised thermal design procedure described in WCAP-11397[5].

1. Initial Conditions

Initial reactor power, pressure, and RCS temperature are assumed to be at their nominal values. Uncertainties in initial conditions are included in the limit DNBR as described in reference 4.

2. Reactivity Coefficients

A conservatively large absolute value of the Doppler-only power coefficient is used. A positive moderator temperature coefficient of +5 pcm/°F is assumed. This results in the maximum core power during the initial part of the transient when the minimum DNBR is reached.

3. 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.

Plant systems and equipment which are necessary to mitigate the effects of the accident are discussed in Subsection 15.0.8 and listed in Table 15.0-5. No single active failure in any of these systems or equipment will adversely affect the consequences of the accident.

b. Results

Figures 15.3-1 and 15.3-2 show the transient response for the loss of two reactor coolant pumps with four loops in operation. Figure 15.3-2 shows the DNBR to be always greater than the limit value. Since DNB does not occur, the ability of the primary coolant to remove heat from the fuel rod is not reduced. Thus, the average fuel and clad temperatures do not increase significantly above their respective initial values.

The calculated sequence of events is shown on Table 15.3-1. The affected reactor coolant pumps will continue to coast down, and the core flow will reach a new equilibrium value corresponding to the two pumps still in operation. With the reactor tripped, a stable plant condition will eventually be attained. Normal plant shutdown may then proceed.

15.3.1.3 Radiological Consequences

The radiological consequences of this malfunction are bounded by the results presented in Subsection 15.3.2 (Complete Loss of Forced Reactor Coolant Flow).

15.3.1.4 Conclusions

The analysis shows that the DNBR will not decrease below the safety analysis limit value at any time during the transient. Thus, no fuel or clad damage is predicted, and all applicable acceptance criteria are met.

15.3.2 Complete Loss of Forced Reactor Coolant Flow

15.3.2.1 Identification of Causes and Accident Description

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

Normal power for the reactor coolant pumps is supplied through buses from a transformer connected to the generator. When a generator trip occurs, the generator breaker is tripped open, the buses are automatically transferred to an offsite power source, and the pumps will continue to supply coolant flow to the core. Following any turbine trip there is immediate generator trip and automatic transfer of the buses to offsite power.

This event is classified as an ANS Condition III incident (an infrequent incident) as defined in Subsection 15.0.1.

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

a. Reactor coolant pump power supply undervoltage or underfrequency

b. 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 offsite power. Channel response time includes consideration of the bus voltage decay time due to generated Electro-Motive Force (EMF) from motors connected to the bus as the motors coast down. This function is blocked below approximately 10 percent power (Permissive 7).

The reactor trip on reactor coolant pump underfrequency is provided to trip the reactor for an underfrequency condition, resulting from frequency disturbances on the power grid. Reference 9 provides analyses of grid frequency disturbances and the resulting nuclear steam supply system protection requirements which are generally applicable.

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 Permissive 8, low flow in any loop will actuate a reactor trip. Between approximately 10 percent power (Permissive 7) and the power level corresponding to Permissive 8, low flow in any two loops will actuate a reactor trip. If the maximum grid frequency decay rate is less than approximately 2.5 Hz/second the low flow trip function will protect the core from underfrequency events. This effect is fully described in Reference 9.

15.3.2.2 Analysis of Effects and Consequences

a. Method of Analysis

The complete loss of flow transient has been analyzed for a loss of four pumps with four loops in operation.

This transient is analyzed by three digital computer codes. First, the LOFTRAN (Reference 1) Code is used to calculate the loop and core flow during the transient, the time of reactor trip based on the calculated flows, the nuclear power transient, and the primary system pressure and temperature transients. The FACTRAN (Reference 2) Code is then used to calculate the heat flux transient based on the nuclear power and flow from FACTRAN. Finally, the VIPRE Code (see Section 4.4) 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 Subsection 15.3.1, except that following the loss of power supply to all pumps at power, a reactor trip is actuated by either reactor coolant pump power supply undervoltage or underfrequency.

b. Results

Figures 15.3-3 through 15.3-5 show the transient response for the loss of power to all reactor coolant pumps. The reactor is assumed to be tripped on an undervoltage signal. Figure 15.3-5 shows the DNBR to be always greater than the limit value. Since DNB does not occur, the ability of the primary coolant to remove heat from the fuel rod is not greatly reduced. Thus, the average fuel and clad temperatures do not increase significantly above their respective initial values.

The calculated sequence of events for the case analyzed is shown on Table 15.3-1. The reactor coolant pumps will continue to coast down, and natural circulation flow will eventually be established, as demonstrated in Subsection 15.2.6.

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

15.3.2.3 Radiological Consequences

No radiological consequences have been calculated for this postulated accident since no fuel or clad damage is predicted.

15.3.2.4 Conclusions

The analysis performed has demonstrated that for the complete loss of forced reactor coolant flow, the DNBR does not decrease below the safety analysis limit at any time during the transient; thus, no fuel or clad damage is predicted, and all applicable acceptance criteria are met.

15.3.3 Reactor Coolant Pump Shaft Seizure (Locked Rotor)

15.3.3.1 Identification of Causes and Accident Description

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

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

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

15.3.3.2 Analysis of Effects and Consequences

a. Method of Analysis

Two digital computer codes are used to analyze this transient. The LOFTRAN Code (Reference 1) is used to calculate the resulting loop and core flow transients following the pump seizure, the time of reactor trip based on the loop flow transients, the nuclear power following reactor trip, and to determine the peak pressure. The thermal behavior of the fuel located at the core hot spot are investigated using the FACTRAN Code (Reference 2), using core flow and nuclear power calculated by LOFTRAN. The FACTRAN code includes the use of a film boiling heat transfer coefficient.

At the beginning of the postulated locked rotor accident (i.e., at the time the shaft in one of the reactor coolant pumps is assumed to seize) the plant is assumed to be in operation under the most adverse steady-state operating conditions (i.e., maximum steady-state power level, maximum steady-state pressure, and maximum steady-state coolant average temperature).

1. Evaluation of the Pressure Transient

After pump seizure, the neutron flux is rapidly reduced by control rod insertion. Rod motion begins one second after the flow in the affected loop reaches 87 percent of nominal flow. No credit is taken for the pressure reducing effect of the pressurizer relief valves, pressurizer spray, steam dump or controlled feedwater flow after plant trip. Although these systems are expected to function and would result in a lower peak RCS pressure, an additional degree

of conservatism is provided by ignoring their effect.

2. Evaluation of DNB in the Core During the Accident

For this accident, DNB is assumed to occur in the core, and therefore, an evaluation of the consequences with respect to fuel rod thermal transients is performed. Results obtained from analysis of this "hot spot" condition represent the upper limit with respect to clad temperature and zirconium-water reaction.

In the evaluation, the rod power at the hot spot is assumed to be 2.5 times the average rod power (i.e., $F_Q = 2.5$) at the initial core power level.

An additional analysis is performed using the VIPRE[4] code to determine the extent of DNB in the core.

Film Boiling Coefficient

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

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

Fuel Clad Gap Coefficient

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

Zirconium Steam Reaction

The zirconium-steam reaction can become significant above 1800°F (clad temperature). The Baker-Just parabolic rate equation shown below is used to define the rate of the zirconium steam reaction.

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

where:

w = amount reacted (mg/cm^2)

t = time (seconds)

T = temperature (Kelvin)

The reaction heat is 1,510 cal/g.

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

No single active failure in any of these systems or equipment will adversely affect the consequences of the accident.

b. Results

The transient results for the most limiting conditions of the locked rotor accident are shown in Figures 15.3-6, sh. 1-4. The results of these calculations are also summarized in Table 15.3-1. The peak RCS pressure reached during the transient is less than that which would cause stresses to exceed the faulted condition stress limits. Also, the peak clad surface temperature is considerably less than 2700°F. It should be noted that the clad temperature was conservatively calculated assuming that DNB occurs at the initiation of the transient.

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

15.3.3.3 Radiological Consequences

a. Assumptions and Parameters

A conservative analysis and a realistic analysis are considered. The conservative analysis employs more pessimistic assumptions regarding fission product release and transport. The assumptions and parameters of the two analyses are presented in Table 15.3-3. Detailed assumptions which are not stated in Table 15.3-3 are discussed in this section.

1. Conservative Analysis

(a) Initial primary coolant activity prior to the accident corresponds to the Technical Specification Limit with the pre-existing iodine spike (see Appendix 15B).

(b) Eight percent of the fuel rods are assumed to fail following the accident, resulting in release of the activity from the cladding gaps of the failed fuel rods to the reactor coolant. The gap activity of noble gases and iodines is shown in Table 15.0-6.

The above release from the damaged fuel rods, plus the activity initially in the primary coolant, represents the conservative source term for this accident and is shown in Table 15.3-4.

(c) Turbine trip and coincident loss of offsite power are assumed. Atmospheric steam releases from the steam generators are required to cool down the plant. The time period for plant cooldown is assumed to be 8 hours.

(d) The conservative source term in Table 15.3-4 is uniformly mixed with the mass of primary coolant. Table 15.3-5 lists the activity concentration of primary coolant after the accident.

(e) Primary to secondary coolant system leakage rate is 1 gpm (at an average primary coolant liquid density of 0.71 gm/cc) for the 8-hour cool down period. Steam releases to the atmosphere may be required during this time period. One hundred percent of the noble gases and 1 percent of the iodines in the Secondary Coolant System, are assumed to be released with the steam to the atmosphere. Table 15.3-5 lists the activity released to the environment via the steam generator safety/relief valves.

2. Realistic Analysis

(a) Initial primary coolant activity prior to the accident corresponds to the equilibrium concentration at 0.12 percent clad defects (see Table 11.1-1).

(b) As a result of the accident, 0.02 percent of the iodines and noble gases of the core activity are released into the Primary Coolant System.

This release from the damaged fuel rods, plus the primary coolant activity prior to the accident, represents the source term for the realistic analysis of this accident and is listed in Table 15.3-6.

(c) For the realistic analysis, offsite power is assumed to be available. The condenser and the Secondary Coolant System are used to cool down the plant. The time period of plant cooldown is assumed to be 8 hours.

(d) The realistic source term in Table 15.3-6 is uniformly mixed with the mass of primary coolant. Table 15.3-7 lists the activity concentration of primary coolant after the accident.

(e) Primary to secondary coolant system leakage rate is 0.009 gpm for the 8-hour cooldown period. Releases from the secondary side are via the condenser air evacuation pump. Decontamination factors of 0.01 and 1.0 for iodine and noble gases, respectively, are assumed for the steam generator. Decontamination factors of 0.001 and 1.0 for iodines and noble gases, respectively, are assumed for the main condenser. During this 8-hour period of time, 100 percent of the noble gases and 0.15 percent of the iodines in the Secondary Coolant System are assumed to be released with the steam to the atmosphere. Table 15.3-7 lists the activity released to the environment via the condenser air evacuation pump.

b. Results

The doses resulting from this accident for both the conservative and realistic analyses are shown in Table 15.3-8.

15.3.3.4 Conclusions

a. Since the peak reactor coolant system pressure reached during the transient is less than that which would cause stresses to exceed the faulted condition stress limits, the integrity of the Primary Coolant System is not endangered.

b. Since the peak clad surface temperature calculated for the hot spot during the worst transient remains considerably less than 2700°F, the core will remain in place and intact with no loss of core cooling capability.

c. The doses which have been calculated for the locked rotor accident are small fractions of 10 CFR Part 100 guideline values.

15.3.4 Reactor Coolant Pump Shaft Seizure (Locked Rotor) Including Loss of Offsite Power

15.3.4.1 Identification of Causes and Accident Description

The transient presented here is the most limiting for the locked rotor and shaft break accidents in terms of radiological consequences.

In the event of a locked rotor/shaft break of a reactor coolant pump (RCP), the remaining three RCPs will continue to run. Analysis of the breaker coordination shows the following: under all postulated operating conditions, including maximum load of one of the 13.8 kV buses (2 RCPS, 2 circulating water pumps and the 13.8 kV substations) and minimum bus voltage, failure of one RCP (with incipient locked rotor amps) will not result in tripping of the incoming breaker to the 13.8 kV bus. Because of the separate power supply to the other 13.8 kV bus (see Figure 8.3-1), this event will have no effect on the power supply of this bus.

Offsite power will not be lost as a consequence of the event. Subsection 8.2.2.3 provides the results of stability studies showing that the loss of Seabrook Station will not cause a loss of offsite power. Figure 8.3-1 is a one-line diagram of the Electrical Distribution System showing the generator circuit breaker used for isolating the generator without affecting the normal supply to the 13.8 kV bus.

Nevertheless, a bounding radiological evaluation of a locked rotor, including a loss of offsite power, is provided below. The transient is postulated to occur in the following manner:

a. RCP rotor locks (or shears) and flow in that loop begins to coastdown.

b. The reactor is tripped on low RCS flow in one loop.

c. Turbine-generator trips.

d. Offsite power is lost even though grid stability analyses show it will not be lost.

e. The loss of offsite power causes the three remaining RCPs to coast down.

15.3.4.2 Analysis of Effects and Components

Method of Analysis

The method of analysis used is the same as presented in Subsection 15.3.3. A bounding value of maximum reactor coolant pressure is calculated by assuming offsite power is lost one second after turbine trip. This assumption is conservative because grid stability analyses

show offsite power will not be lost.

15.3.4.3 Radiological Consequences

Assuming that all fuel rods with a minimum transient DNBR of less than the safety analysis limit become failed rods, then the fraction of failed fuel is predicted to be less than 8 percent for this bounding event.

In the event of a failed open atmospheric steam dump valve, the offsite doses are well within the design limits specified in 10 CFR Part 100. The offsite thyroid doses, in the event of a failed open valve concurrent with the locked rotor event plus the assumed 8 percent fuel clad failure and the assumed loss of offsite power, are a maximum of 100 times the values presented in Updated FSAR Subsection 15.3.3.3, or 230 rems and 240 rems at the Exclusion Area Boundary (EAB) and Low Population Zone (LPZ), respectively. The offsite whole body gamma doses for this event are 1.4 rem and 1.9 rem at the EAB and LPZ, respectively.

With the failure of a steam dump valve in the open position, plant procedure would call for isolation of the affected steam generator's feedwater flow with subsequent drying out of the steam generator and loss of any iodine partitioning afforded by the water volume contained above the point of primary to secondary leakage. Although some finite period of time is required for the considerable amount of water contained within a typical Westinghouse steam generator to boil off, it has been conservatively assumed that this time is small and no partitioning of iodine occurs within the steam generator.

The failure of an atmospheric steam dump valve, discussed above, is considered to be the limiting single failure for radiological dose evaluations associated with the locked rotor event. In addition to the sequence of events mentioned above, it is also conservatively assumed that the entire primary to secondary allowable leakage of 1 gpm occurs in the steam generator that is venting through the stuck open valve.

15.3.4.4 Conclusion

The transient presented here is the most limiting for the locked rotor and shaft break accidents in terms of peak clad temperature and maximum reactor coolant pressure. Since the peak RCS pressure reached is less than that which would cause stresses to exceed the faulted condition stress limits, the integrity of the Primary Coolant System is not endangered. Also, the maximum peak clad surface temperature calculated for the hot spot remains considerably less than 2700°F; this ensures that the core will remain in place and intact with no loss of core cooling capability. As in the locked rotor analysis discussed in Subsection 15.3.3, this analysis conservatively assumes that DNB occurs at the beginning of the transient.

Grid stability analyses show that offsite power will not be lost following a turbine trip. However, a conservative radiological dose calculation was performed assuming offsite power is lost at the time of turbine trip and

assuming the entire primary to secondary allowable leakage of 1 gpm occurs in a steam generator that is venting through a stuck-open atmospheric dump valve.

15.3.5 Reactor Coolant Pump Shaft Break

15.3.5.1 Identification of Causes and Accident Description

The accident is postulated as an instantaneous failure of an RCP shaft. Flow through the affected reactor coolant loop is rapidly reduced, though the initial rate of reduction of coolant flow is greater for the RCP rotor seizure event (Sections 15.3.3). With a failed shaft the pump impeller could conceivably be free to spin in the reverse direction instead of being in a fixed position. The effect of such reverse spinning is a slight decrease in the end point (steady-state) core flow.

The analysis presented in Sections 15.3.3 represents the limiting condition, assuming a locked rotor for forward flow but a free-spinning shaft for reverse flow in the affected loop.

This event is classified as an ANS Condition IV incident (a limiting fault).

15.3.5.2 Radiological Consequences

The radiological consequences of this malfunction are no worse than those calculated for the locked rotor incident (see Subsection 15.3.3).

15.3.5.3 Conclusion

The conclusions of Section 15.3.3 apply for a reactor coolant pump shaft break accident.

15.3.6 References

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

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

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

4) WCAP-14565, "VIPRE-01 Modeling and Qualification for Pressurized Water Reactor Non-LOCA Thermal-Hydraulic Safety Analysis," Y. X. Sung, et al., April 1997

5) WCAP-11397-P-A, "Revised Thermal Design Procedure," A. J. Friedland and S. Ray, April 1984

6) WCAP-9226-P-A. "Reactor Core Response to Excessive Secondary Steam Releases," S. D. Hollingsworth, et al., January 1998

7) WCAP-9230, "Report on the Consequences of a Postulated Main Feedline Rupture," G. E. Lang and J. P. Cunningham, January 1978

8) WCAP-11394-P-A, "Methodology for the Analysis of the Dropped Rod Event," R. L. Haessler, et al., January 1990

9) WCAP-8424, Revision 1, "An Evaluation of Loss of Flow Accidents Caused by Power System Frequency Transients in Westinghouse PWRs," M. S. Baldwin, et al., June 1975

TABLE 15.3-1

TIME SEQUENCE OF EVENTS FOR INCIDENTS THAT RESULT IN A DECREASE IN REACTOR COOLANT SYSTEM FLOW

Accident	Event	Time (seconds)
Partial Loss of Forced Reactor Coolant Flow	Two of four operating RCPs begin coasting down	0.0
	Low flow reactor trip setpoint reached	1.4
	Rods begin to drop	2.4
	Minimum DNBR occurs	3.6
Complete Loss of Forced Reactor Coolant Flow	All four operating RCPs lose power and begin coasting down, bus undervoltage begins to decay; RCP undervoltage reactor trip setpoint reached	0.0
	Rods begin to drop	1.5
	Minimum DNBR occurs	3.2
Reactor Coolant Pump Shaft Seizure (Locked Rotor)	Rotor on one RCP locks	0.0
	Low flow reactor trip setpoint reached in the affected loop	0.02
	Rods begin to fall into core, Turbine trip, Loss of offsite power, unaffected reactor coolant pumps begin to coast down	1.02
	Maximum clad temperature occurs	2.8
	Maximum RCS pressure occurs	3.3

TABLE 15.3-2

SUMMARY OF RESULTS FOR LOCKED ROTOR TRANSIENT

Maximum Reactor Coolant System Pressure (psia)	2635
Maximum Cladding Temperature (°F) Core Hot Spot	1709
Zr-H_2O Reaction At Core Hot Spot (% by weight)	0.21

TABLE 15.3-8

OFFSITE DOSES AT GIVEN SITE DUE TO LOCKED ROTOR ACCIDENT

Site*	Thyroid (rem)	Whole Body (rem)	Skin (rem)
Conservative Analysis			
EAB (0-2 Hours)	2.3E+00**	4.4E-01	7.4E-01
LPZ (0-8 Hours)	2.4E+00	2.1E-01	3.6E-01
Realistic Analysis			
EAB (0-2 Hours)	1.0E-07	1.9E-05	3.3E-05
LPZ (0-8 Hours)	1.1E-07	2.1E-05	3.6E-05

* EAB: Exclusion Area Boundary; LPZ: Low Population Zone

** 2.3E+00 = 2.3×10^{0}

TABLE 15.3-9

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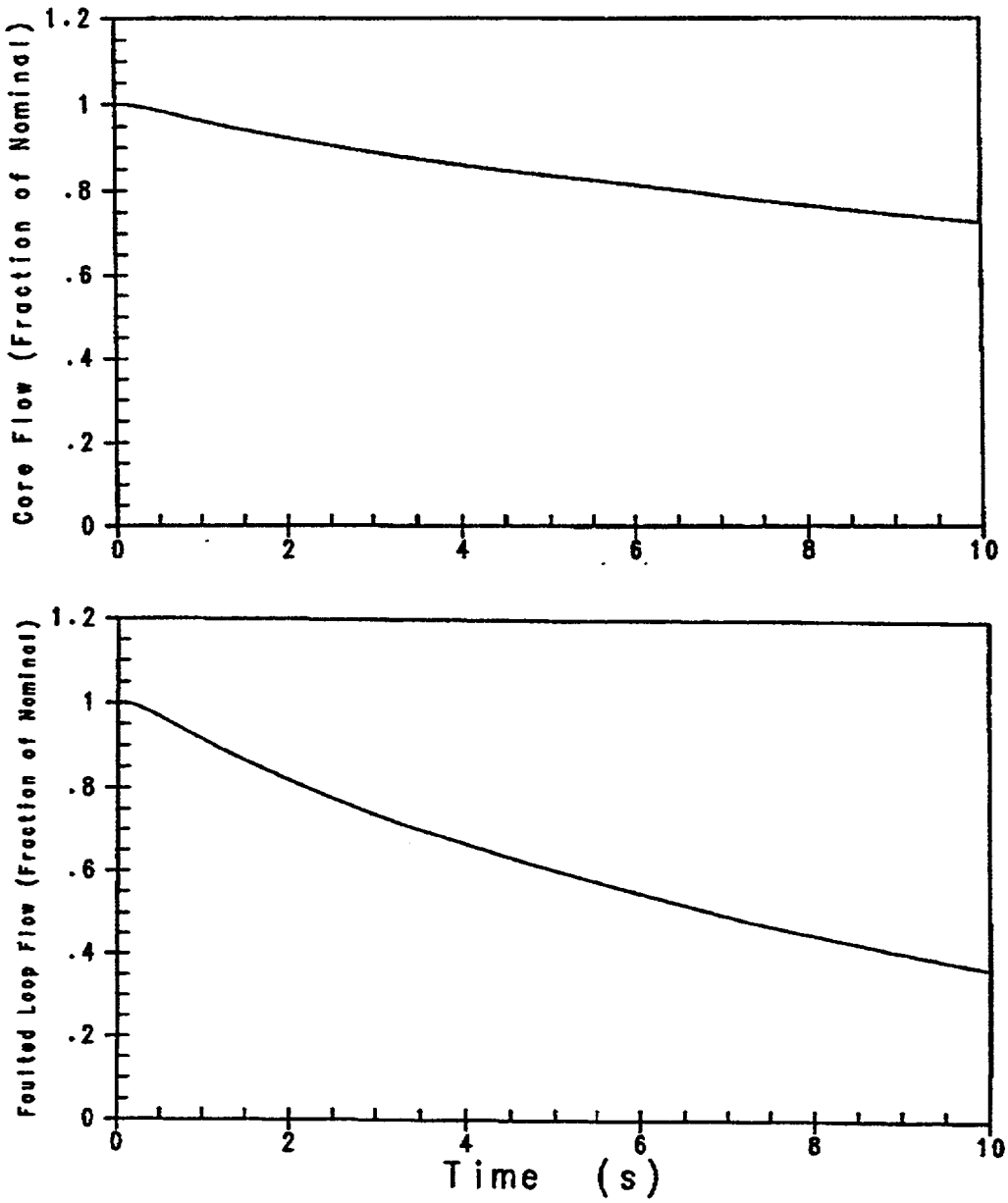

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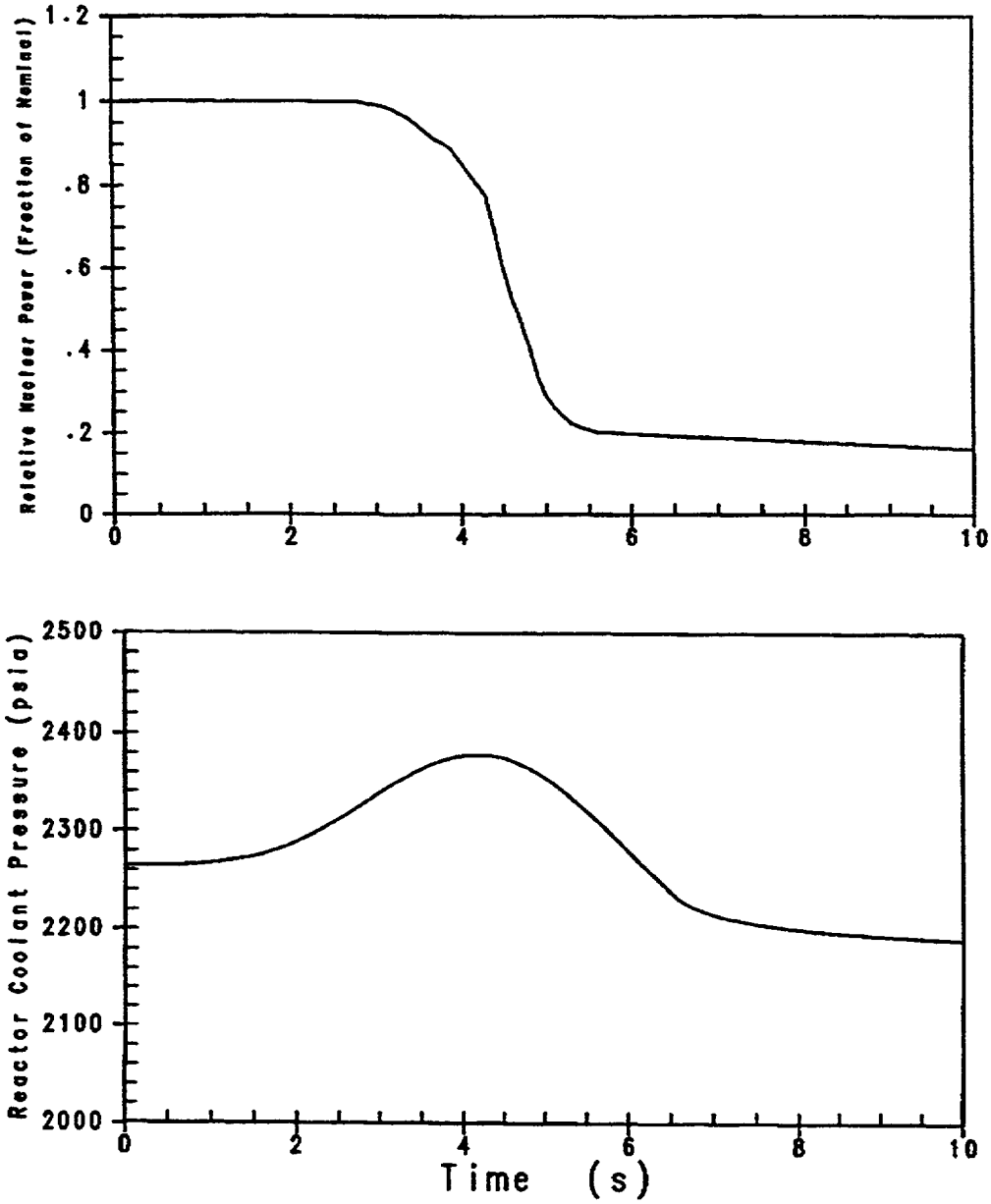
SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Core Flow and Faulted Loop Flow Transients for a Partial Loss of Forced Reactor Coolant Flow (4 loops in operation, 2 RCPs coasting down)	
	REV. 07	FIGURE 15.3-1, Sh 1



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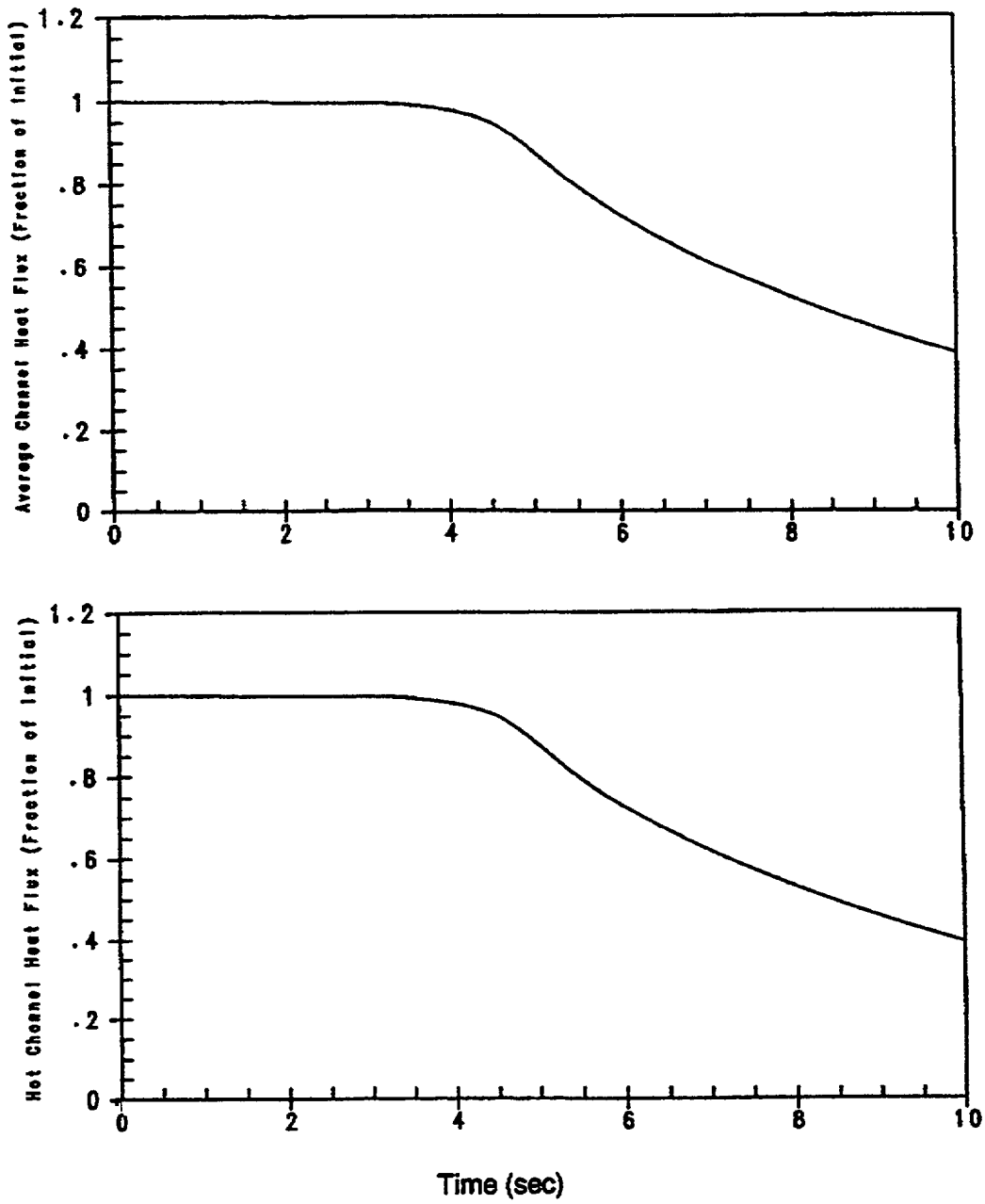
SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Nuclear Power and Pressurizer Transients for a Partial Loss of Forced Reactor Coolant Flow (4 loops in operation, 2 RCPs coasting down)	
	REV. 07	FIGURE 15.3-1, Sh 2



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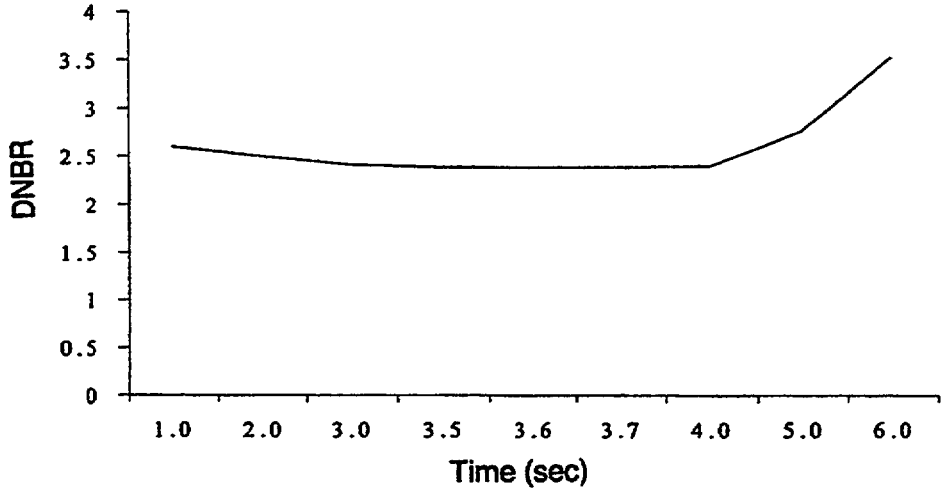
SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Average and Hot Channel Heat Flux Transients for a Partial Loss of Forced Reactor Coolant Flow (4 loops in operation, 2 RCPs coasting down)	
	REV. 07	FIGURE 15.3-1, Sh 3



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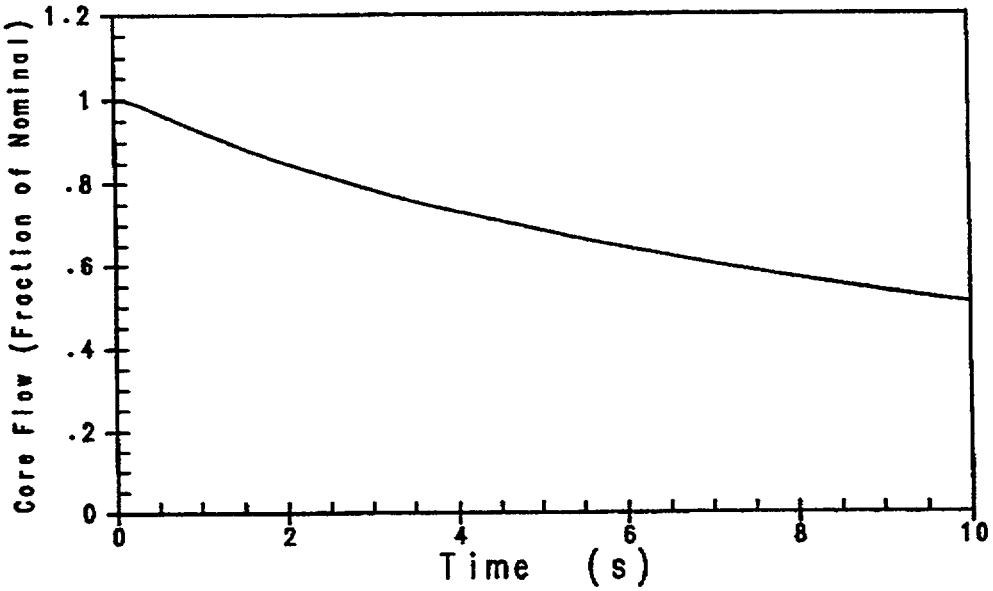
SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	DNBR Transient for a Partial Loss of Forced Reactor Coolant Flow (4 loops in operation, 2 RCPs coasting down)	
	REV. 07	FIGURE 15.3-2



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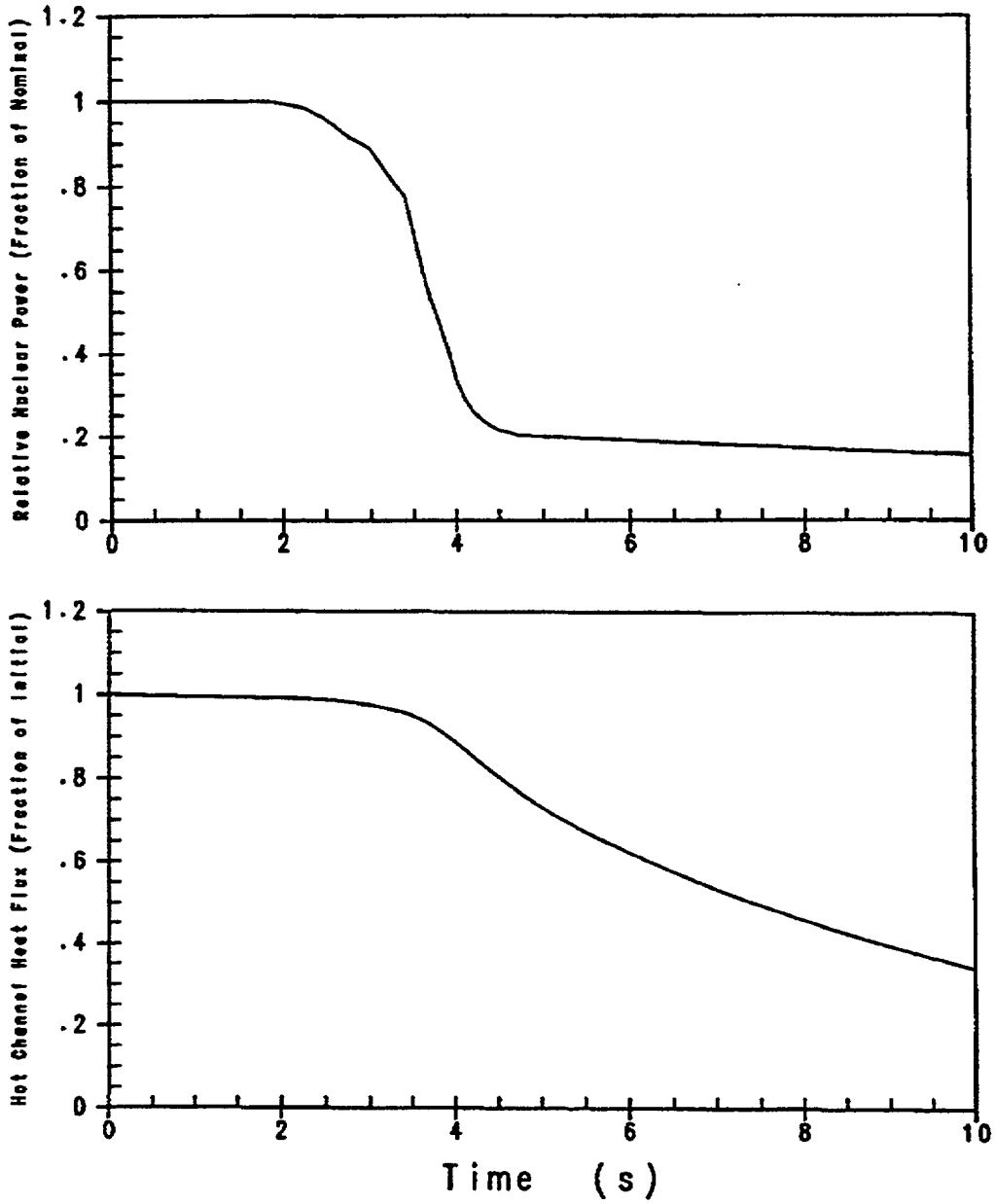
SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Core Flow Transient for a Complete Loss of Forced Reactor Coolant Flow (4 loops in operation, 4 RCPs coasting down)	
	REV. 07	FIGURE 15.3-3



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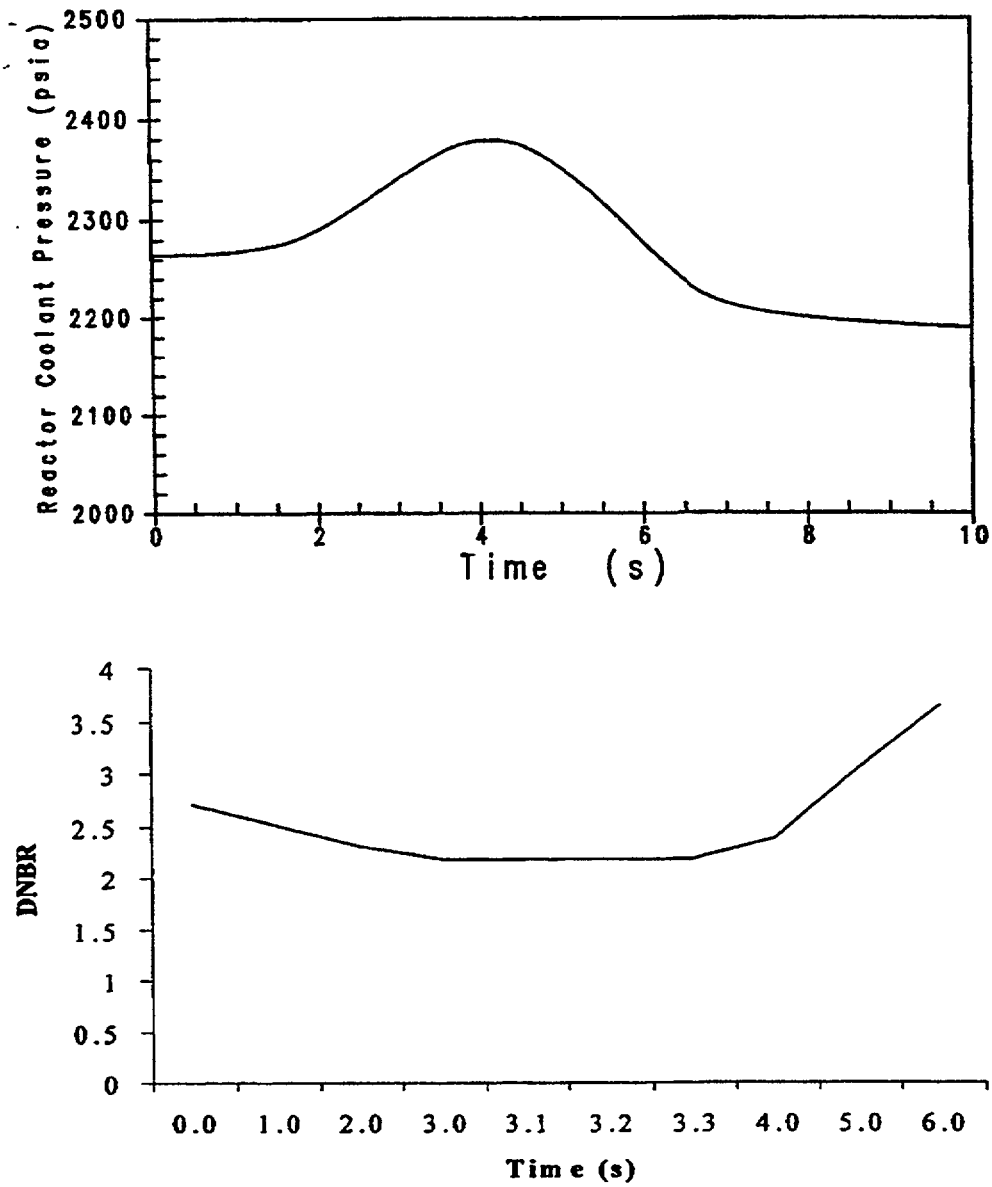
SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Nuclear Power and Hot Channel Hot Flux Transients for a Complete Loss of Forced Reactor Coolant Flow (4 loops in operation, 4 RCPs coasting down)	
	REV. 07	FIGURE 15.3-4



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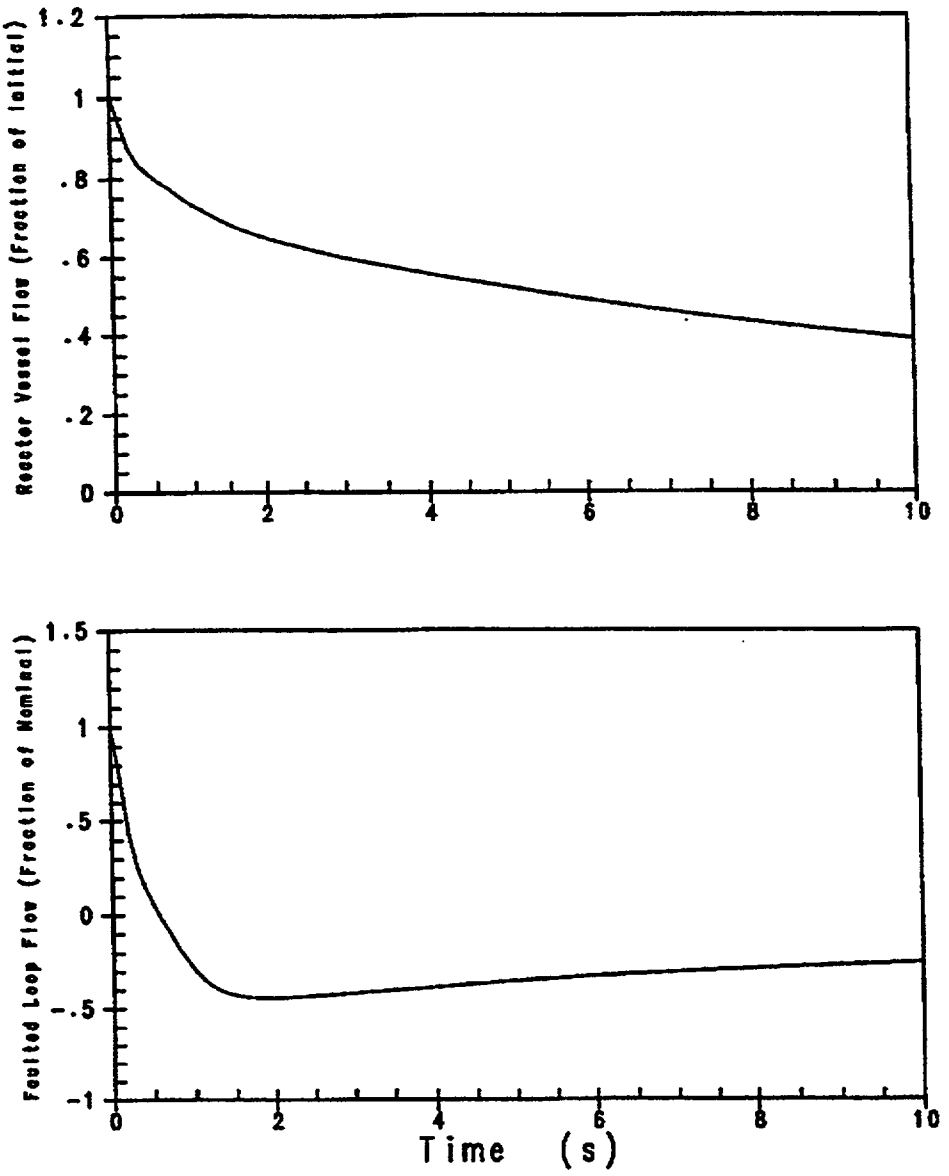
SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	RCS Pressure and DNBR Transients for a Complete Loss of Forced Reactor Coolant Flow (4 loops in operation, 4 RCPs coasting down)	
	REV. 07	FIGURE 15.3-5



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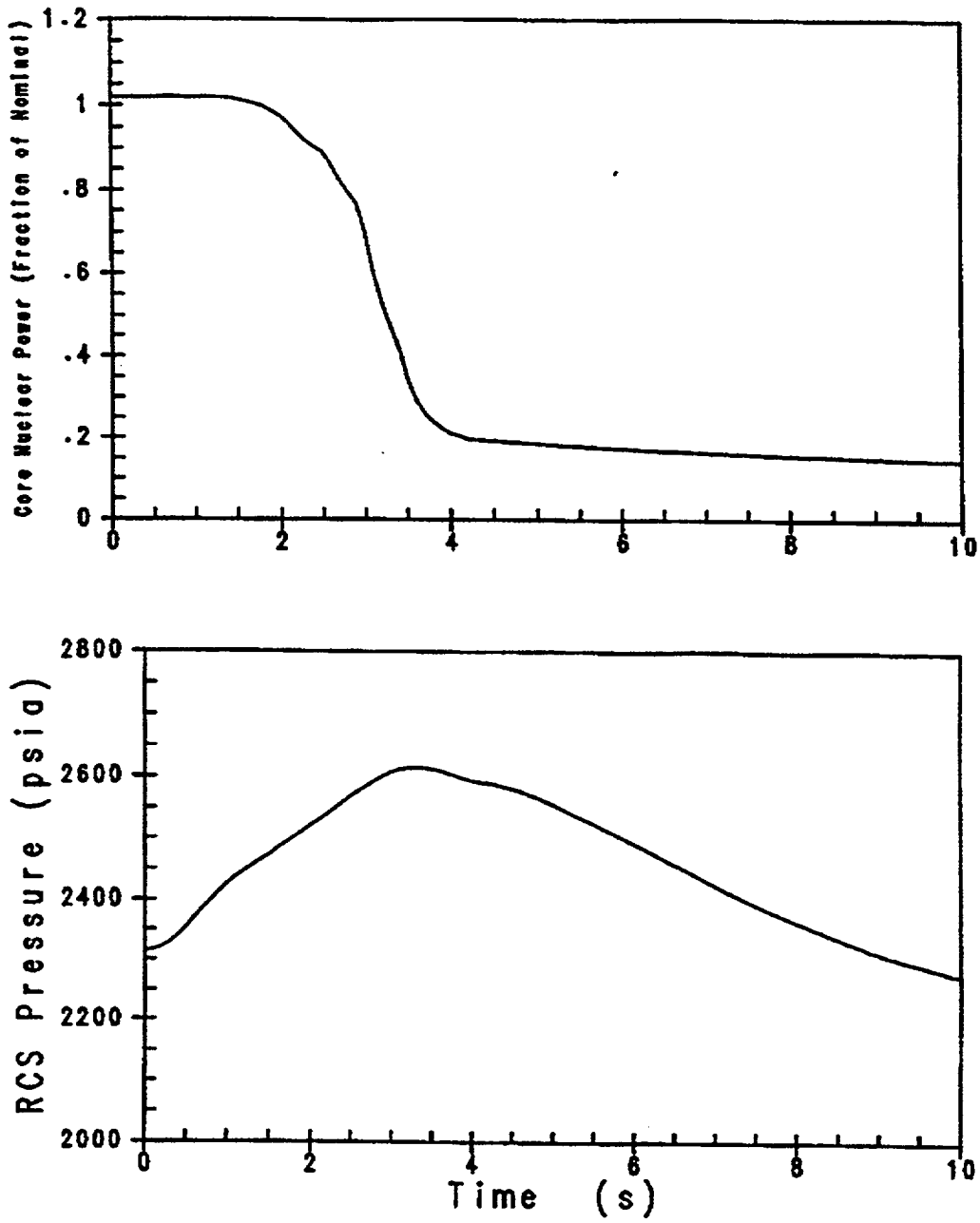
SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Reactor Vessel Flow and Faulted Loop Flow Transients for a RCP Rotor Seizure	
	REV. 07	FIGURE 15.3-6, Sh 1



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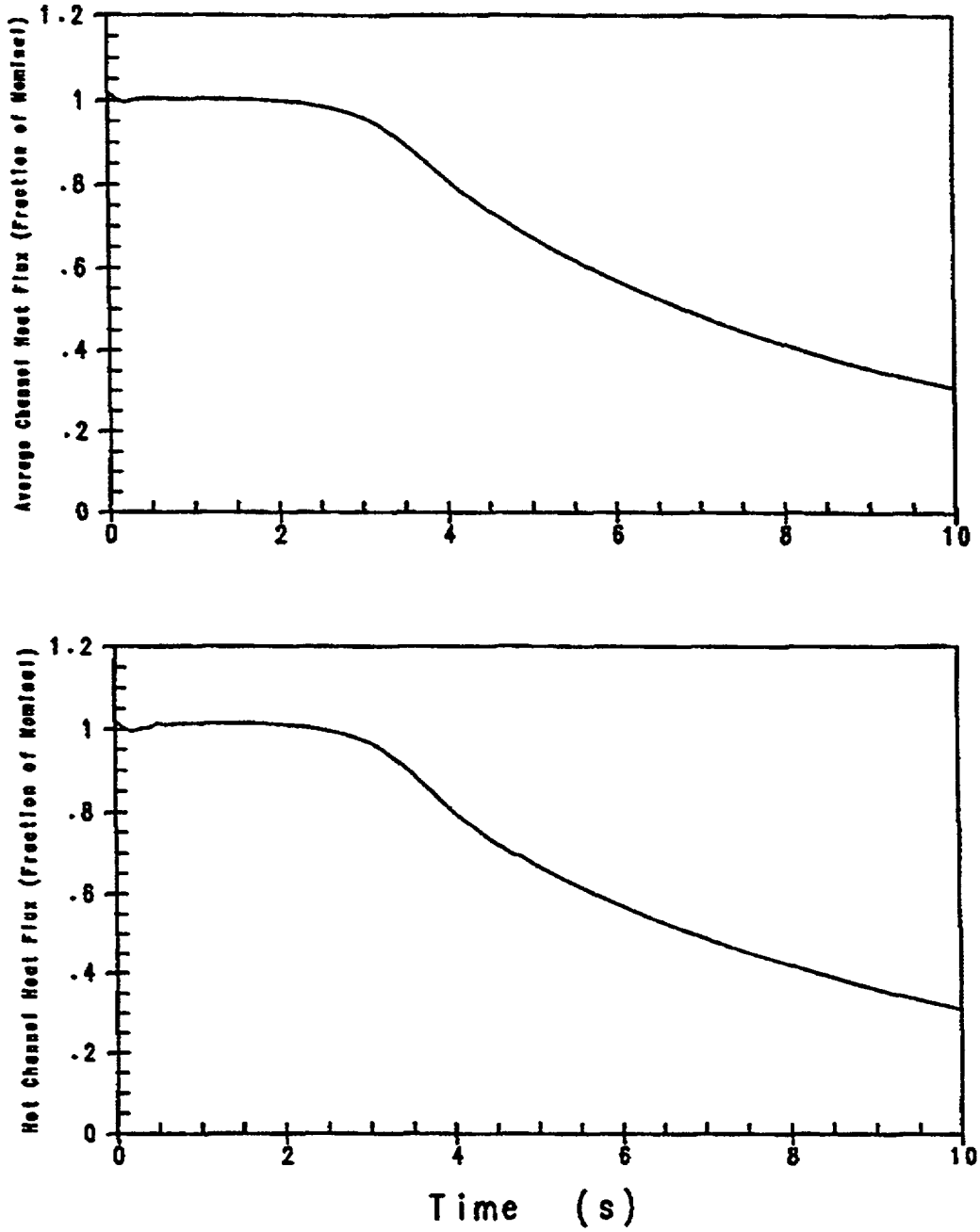
SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Nuclear Power and RCS Pressure Transients for a RCP Rotor Seizure	
	REV. 07	FIGURE 15.3-6, Sh 2



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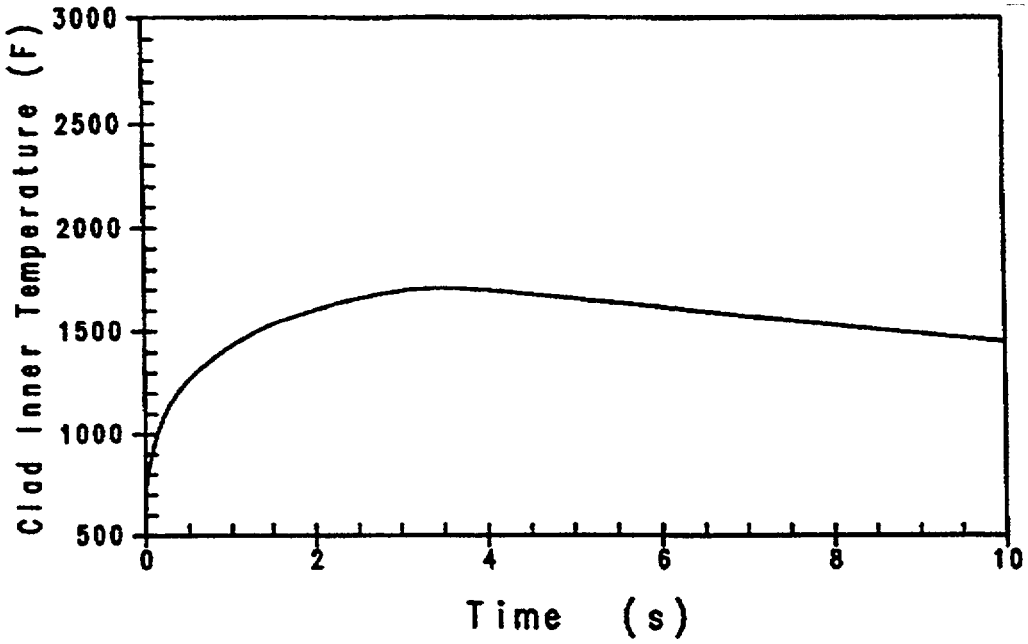
SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Average Channel and Hot Channel Heat Flux Transients for a RCP Rotor Seizure	
	REV. 07	FIGURE 15.3-6, Sh 3



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SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Clad Inner Temperature Transients for a RCP Rotor Seizure	
	REV. 07	FIGURE 15.3-6, Sh 4

FIGURE 15.3-7

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15.4 REACTIVITY AND POWER DISTRIBUTION ANOMALIES

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

Discussions of the following incidents are presented:

a. Uncontrolled Rod Cluster Control Assembly bank withdrawal from a subcritical or low power startup condition

b. Uncontrolled Rod Cluster Control Assembly bank withdrawal at power

c. Rod Cluster Control Assembly misalignment

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

e. Chemical and Volume Control System malfunction that results in a decrease in the boron concentration in the reactor coolant

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

g. Spectrum of Rod Cluster Control Assembly ejection accidents.

Items a, b, d, and e above are considered to be ANS Condition II events, item f an ANS Condition III event, and item g an ANS Condition IV event. Item c entails both Condition II and III events. Item d is precluded by technical specifications which prohibit 3-loop operation. Item f is precluded by being detectable without consequence during refueling/startup physics tests. Subsection 15.0.1 contains a discussion of ANS classifications.

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

15.4.1.1 Identification of Causes and Accident Description

A Rod Cluster Control Assembly (RCCA) withdrawal accident is defined as an uncontrolled addition of reactivity to the reactor core caused by withdrawal of 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 either the reactor subcritical, at Hot Zero Power, or at power. The "at power" case is discussed in Subsection 15.4.2.

Although the reactor is normally brought to power from a subcritical condition by means of RCCA withdrawal, initial startup procedures with a clean core call for boron dilution on RCCA withdrawal. The maximum rate of reactivity increase in the case of boron dilution is less than that assumed in this analysis (see Subsection 15.4.6, "Chemical and Volume Control System Malfunction that Results in a Decrease in Boron Concentration in the Reactor Coolant").

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

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

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:

a. 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 adjusted 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.

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

c. Power Range High Neutron Flux Reactor Trip (Low Setting) - Actuated when two out of the four power range channels indicate a power level above approximately 25 percent of full power. This trip function may be manually bypassed when two out of the four power range channels indicate a power level above approximately 10 percent of full power, and is automatically reinstated only after three of the four channels indicate a power level below this value.

d. 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.

e. High Nuclear Flux Rate Reactor Trip - Actuated when the positive rate of change of neutron flux on two out of four nuclear power range channels indicate a rate above the preset setpoint. This trip function is always active.

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

15.4.1.2 Analysis of Effects and Consequences

a. Method of Analysis

The analysis of the uncontrolled RCCA bank withdrawal from subcritical accident is performed in three stages: (1) an average core nuclear power transient calculation, (2) an average core heat transfer calculation, and (3) the DNBR calculation. The average core nuclear calculation is performed using spatial neutron kinetics methods, TWINKLE (Reference 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 (Reference 2). The average heat flux is next used in VIPRE (described in Reference 4) for the transient DNBR calculation. Plant characteristics and initial conditions are discussed in Subsection 15.0.3.

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

1. Since the magnitude of the power peak reached during the initial part of the transient for any given rate of reactivity insertion is strongly dependent on the Doppler coefficient, a conservatively low Doppler power defect of -900 pcm was used. See Subsection 15.0.4 and Table 15.0-3.

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. The analysis assumes a moderator temperature coefficient of at least +5 pcm/°F at the zero power nominal temperature.

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 RCCA release, is taken into account. A 10 percent increase is assumed for the power range flux trip setpoint, raising it from the nominal value of 25 percent to 35 percent. Since the rise in the neutron flux is so rapid, the effect of errors in the trip setpoint on the actual time at which the rods are released is negligible. In addition, the reactor trip insertion characteristic is based on the assumption that the highest worth RCCA is stuck in its fully withdrawn position. See Subsection 15.0.5 for RCCA insertion characteristics.

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

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

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

8. Two reactor coolant pumps are assumed to be in operation consistent with plant operating Mode 3 technical specification requirements. This is conservative with respect to DNB.

No single active failure in any of these systems or equipment will adversely affect the consequences of the accident.

b. Results

The nuclear power, core heat flux, hot spot fuel average and clad temperature transient results are shown in Figures 15.4-1, 15.4-2 and 15.4-3. The DNB analysis demonstrates that the DNBR remains above the applicable safety analysis limit value at all times.

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

15.4.1.3 Radiological Consequences

No radiological consequences have been calculated for this postulated accident since no fuel or clad damage is predicted.

15.4.1.4 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 DNBR is greater than the limit value for all regions of the core. Thus, no fuel or clad damage is predicted as a result of DNB.

15.4.2 Uncontrolled Rod Cluster Control Assembly Bank Withdrawal at Power

15.4.2.1 Identification of Causes and Accident Description

Uncontrolled 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 could eventually result in DNB. Therefore, in order to avert damage to the fuel clad, the Reactor Protection System is designed to terminate any such transient before the DNBR falls below the limit value.

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

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

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

b. 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.

c. Reactor trip is actuated if any two out of four ΔT channels exceed an Overpower ΔT setpoint. This setpoint is automatically varied with coolant temperature to protect against centerline melting.

d. 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.

e. A high pressurizer water level reactor trip actuated from any two out of three level channels when the reactor power is above approximately 10 percent (Permissive P7).

Figure 15.0-1 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 that the limit imposed by any given DNBR can be represented as a line. The DNB lines represent the locus of conditions for which the DNBR equals the safety analysis limit value. All points below and to the left of a DNB line for a given pressure have a DNBR greater than the safety analysis limit value. The diagram shows that DNB is prevented for all cases if the area enclosed with the maximum protection lines is not traversed by the applicable DNBR line at any point.

The area of permissible operation (power, pressure and temperature) is bounded by the combination of the following reactor trips: high neutron flux (fixed setpoint); high pressurizer pressure (fixed setpoint); low pressurizer pressure (fixed setpoint); Overpower and Overtemperature ΔT (variable setpoints), and the opening of the steam generator safety valves.

15.4.2.2 Analysis of Effects and Consequences

a. Method of Analysis

This transient is analyzed by the LOFTRAN Code (Reference 1). 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, pressure, and power level. The core limits as illustrated in Figure 15.0-1 are used as input to LOFTRAN to determine the minimum DNBR during the transient.

This accident is analyzed with the revised thermal design procedure, as described in WCAP-11397[5]. In order to obtain conservative results for an uncontrolled rod withdrawal at power accident, the following assumptions are made:

1. Initial reactor power, pressure, and RCS temperatures are assumed to be at their nominal values. Uncertainties in initial conditions are included in the limit DNBR as described in Reference 5.

2. Reactivity coefficients - Two cases are analyzed:

   (a) Minimum reactivity feedback

   A positive moderator coefficient of reactivity is assumed corresponding to the beginning of core life. A least negative Doppler power coefficient is assumed.

   (b) Maximum reactivity feedback

   A conservatively large positive moderator density coefficient and a most negative Doppler power coefficient are assumed, corresponding to the end of core life.

3. The reactor trip on high neutron flux is assumed to be actuated at a conservative value of 118 percent of nominal full power. The overtemperature ΔT trip includes all adverse instrumentation and setpoint errors with maximum delays for trip actuation.

4. The RCCA trip insertion characteristic is 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 the $f(\Delta I)$ penalty function, which decreases the overtemperature ΔT setpoint proportional to the decrease in margin to DNB.

No single active failure in any of these systems or equipment will adversely offset the consequences of the accident.

b. Results

Figures 15.4-4, sh. 1-3 the transient response for a rapid RCCA withdrawal incident starting from full power. Reactor trip on high neutron flux occurs shortly after the start of the accident. Since the neutron flux increase is rapid with respect to the thermal time constant, small changes in coolant temperature and pressure result and margin to DNB is maintained.

The transient response for a slow RCCA withdrawal from full power is shown in Figures 15.4-5, sh. 1-3. Reactor trip on overtemperature ΔT occurs after a longer period and the rise in temperature and pressure is consequently larger than for rapid RCCA withdrawal. Again, the minimum DNBR is greater than the limit value.

Figure 15.4-6 shows the minimum DNBR as a function of reactivity insertion rate from initial full power operation for minimum and maximum reactivity feedback. It can be seen that two reactor trip channels provide protection over the whole range of reactivity insertion rates. These are the high neutron flux and overtemperature ΔT channels. The minimum DNBR is never less than the limit value.

Figures 15.4-7 and 15.4-8 show the minimum DNBR as a function of reactivity insertion rate for RCCA withdrawal incidents starting at 60 and 10 percent power respectively. The results are similar to the 100 percent power case, except as the initial power is decreased, the range over which the overtemperature ΔT trip is effective is increased. In both cases the DNBR remains above the limit value.

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

Referring to Figure 15.4-7, the 60 percent power minimum feedback case, it is noted that:

1. For high reactivity insertion rates (i.e., between ~ 20 pcm/sec and 100 pcm/sec) reactor trip is initiated by the high neutron flux trip for the minimum reactivity feedback cases. 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 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 channels initiate a reactor trip when measured coolant ΔT exceeds a setpoint based on measured reactor coolant system average temperature and pressure. It is important in this context to note that the average temperature contribution to the circuit as well as the measured ΔT that is compared to the setpoint are 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 (i.e. at ~ 20 pcm/sec reactivity insertion rate).

   For reactivity insertion rates less than ~ 20 pcm/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.

Figures 15.4-6, 15.4-7, and 15.4-8 illustrate the minimum DNBR calculated for minimum and maximum reactivity feedback at 100, 60, and 10 percent power, respectively.

Since the RCCA withdrawal at power incident is an overpower transient, the fuel temperatures rise during the transient until after reactor trip occurs. For high reactivity insertion rates, the overpower transient is fast with respect to the fuel rod thermal time constant, and the core heat flux lags behind the neutron flux response. Due to this lag, the peak core heat flux does not exceed 118 percent of its nominal value (i.e., the high neutron flux trip setpoint assumed in the analysis). Taking into account the effect of the RCCA withdrawal on the axial core power distribution, the peak fuel temperature will still remain below the fuel melting temperature.

For slow reactivity insertion rates, the core heat flux remains more nearly in equilibrium with the neutron flux. The overpower transient is terminated by the Overtemperature ΔT reactor trip before a DNB condition is reached. The peak heat flux again is maintained below 118 percent of its nominal value. Taking into account the effect of the RCCA withdrawal on the axial core power distribution, the peak fuel centerline temperature will remain below the fuel melting temperature.

Since DNB does not occur at any time during the RCCA withdrawal at power transient, the ability of the primary coolant to remove heat from the fuel rod is not reduced. Thus, the fuel cladding temperature does not rise significantly above its initial value during the transient.

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

15.4.2.3 Radiological Consequences

No radiological consequences have been calculated for this postulated accident since no fuel or clad damage is predicted.

15.4.2.4 Conclusions

The high neutron flux and overtemperature ΔT trip channels provide adequate protection over the entire range of possible reactivity insertion rates, such that the minimum value of DNBR remains above the limit value.

15.4.3 Rod Cluster Control Assembly Misoperation (System Malfunction or Operator Error)

15.4.3.1 Identification of Causes and Accident Description

Rod Cluster Control Assembly (RCCA) misalignment accidents include:

a. One or more dropped RCCAs within the same group

b. A dropped RCCA bank

c. Statically misaligned RCCA

d. Withdrawal of a single RCCA.

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

Full length RCCAs are always moved in preselected banks, and the banks are always moved in the same preselected sequence. Each bank of RCCAs is divided into two groups. The two groups in a bank move sequentially such that the first group is always within one step of the second group in the bank. A definite schedule of actuation (or deactuation of the stationary gripper, movable gripper, and lift coils of a mechanism) is required to withdraw the RCCA attached to the mechanism. Since the stationary gripper, movable

gripper, and lift coils associated with the four 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.

The dropped RCCA, dropped RCCA bank, and statically misaligned assembly events are classified as ANS Condition II incidents (incidents of moderate frequency) as defined in Subsection 15.0.1. However the single RCCA withdrawal incident is classified as an ANS Condition III event, as discussed below.

No single electrical or mechanical failure in the Rod Control System could cause the accidental withdrawal of a single RCCA from the inserted bank at full power operation. The operator could withdraw a single RCCA in the control bank since this feature is necessary to retrieve an assembly should one be accidentally dropped. The event analyzed must result from multiple wiring failures or multiple significant operator errors and subsequent and repeated operator, disregard of event indication. The probability of such a combination of conditions is so low that the limiting consequences may include slight fuel damage.

Thus, consistent with the philosophy and format of ANSI N18.2, the event is classified as a Condition III event. By definition, "Condition III occurrences include incidents, any one of which may occur during the lifetime of a particular plant," and "shall not cause more than a small fraction of fuel elements in the reactor to be damaged..."

This selection of criterion is in accordance with General Design Criterion (GDC) 25 which states, "The protection system shall be designed to assure that specified acceptable fuel design limits are not exceeded for any <u>single</u> malfunction of the reactivity control systems, such as accidental withdrawal (not ejection or dropout) of control rods." (Emphasis has been added.) It has been shown that single failures resulting in RCCA bank withdrawals do not violate specified fuel design limits. Moreover, no single malfunction can result in the withdrawal of a single RCCA. Thus, it is concluded that the criterion established for the single rod withdrawal at power is appropriate and in accordance with GDC 25.

A dropped RCCA or RCCA bank is detected by:

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

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

c. Rod at bottom signal

d. Rod deviation alarm or

e. Rod position indication.

Misaligned RCCAs are detected by:

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

b. Rod deviation alarm or

c. Rod position indicators.

The resolution of the rod position indicator channel is ± 1.7 percent of span (± 2.5 inches). Deviation of any RCCA from its group by twice this distance (3.4 percent of span, or 5 inches) will not cause power distributions worse than the design limits. The deviation alarm alerts the operator to rod deviation with respect to the group position in excess of 5.1 percent of span. 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 operating instructions shall be followed to assure the alignment of the nonindicated RCCA. The operator is also required to take action as outlined by the Technical Specifications.

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

No single active failure in any of these systems or equipment will adversely affect the consequences of the accident.

15.4.3.2 Analysis of Effects and Consequences

a. Dropped RCCAs, Dropped RCCA Bank, and Statically Misaligned RCCA

1. Method of Analysis

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

The LOFTRAN[1] is used to calculate the transient system response for the evaluation of the dropped RCCA

event. The code simulates the neutron kinetics, RCS, 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.

Transient statepoints (temperature, pressure and power) are calculated by LOFTRAN 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 analysis and the hot channel factor from the nuclear analysis, the DNB design basis is shown to be met using the VIPRE code. The transient response analysis, nuclear peaking factor analysis, and performance of the DNB design basis confirmation are performed in accordance with the methodology described in WCAP-11394[8]. Note that the analysis does not take credit for the negative flux rate reactor trip.

(b) Dropped RCCA Bank

A dropped RCCA bank results in a symmetric power change in the core. As discussed in reference 8, assumptions made for the dropped RCCA(s) analysis provide a bounding analysis for the dropped RCCA bank.

(c) Statically Misaligned RCCA

Steady-state power distributions are analyzed using appropriate nuclear physics computer codes. The peaking factors are then used as input to the VIPRE code to calculate the DNBR. The analysis examines the following cases: the worst rod withdrawn with bank D inserted at the insertion limit, the worst rod dropped with bank D inserted at the insertion limit, and the worst rod dropped with all other rods out, all with the reactor initially at full power. The analysis assumes this incident to occur at beginning of life since this results in the minimum value of the moderator temperature coefficient (least negative). This assumption maximizes the power rise and minimizes the tendency of the large moderator temperature coefficient (most negative) to flatten the power distribution.

2. Results

(a) One or More Dropped RCCAs

Single or multiple dropped RCCAs within the same group result in a negative reactivity insertion. The core is not adversely affected during this period, since power is decreasing rapidly. Either reactivity feedback or control bank withdrawal will reestablish power. Partially dropped RCCA results are bounded by the fully dropped RCCA results. The operator may manually retrieve the RCCA by following approved operating procedures.

Following a dropped rod event in manual rod control, the plant will establish a new equilibrium condition. Without control system interaction, a new equilibrium is achieved at a reduced power level and reduced primary temperature. Thus, the automatic rod control mode of operation is the limiting case.

For a dropped RCCA event in the automatic rod control mode, the rod control system detects the drop in power and initiates control bank withdrawal. Power overshoot may occur due to this action by the automatic rod controller after which the control system will insert the control bank to restore nominal power. Figures 15.4-9, sh. 1-2 show a typical transient response to a dropped RCCA (or RCCAs) in the automatic rod control mode. In all cases, the minimum DNBR remains above the limit value.

Following plant stabilization, the operator may manually retrieve the RCCA by following approved operating procedures.

(b) Dropped RCCA Bank

A dropped RCCA bank results in a negative reactivity insertion greater than 500 pcm. The core is not adversely affected during the insertion period, since power is decreasing rapidly. The transient will proceed as described for a dropped RCCA above, however, the power increase will be less due to the greater worth of the entire bank. The power transient for a dropped RCCA bank is symmetric. Following plant stabilization, normal procedures are followed.

(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 inserted to its insertion limit 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. 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 the incident.

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.

The RCCA misalignment cases are analyzed using the revised thermal design procedure as described in WCAP-11397[5]. The initial reactor power, pressure, and RCS temperatures are at their nominal values, but with the increased radial peaking factor associated with the misaligned RCCA. Uncertainties in the initial conditions are included in the limit DNBR value.

For the RCCA misalignment case with bank D inserted to its full power insertion limit and one RCCA fully withdrawn, DNBR does not fall below the limit value.

DNB calculations have not been performed specifically for RCCAs missing from other banks; however, power shape calculations have been performed 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 limit value.

DNB does not occur for the RCCA misalignment incident and thus the ability of the primary coolant to remove heat from the fuel rod is not reduced. The peak fuel temperature 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.

Following the identification of an RCCA misalignment condition by the operator, the operator is required to take action as required by the plant technical specifications and operating instructions.

b. Single RCCA Withdrawal

1. Method of Analysis

Power distributions within the core are calculated. The peaking factors are then used by VIPRE to calculate the 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 assumption maximizes the power rise and minimizes the tendency of increased moderator temperature to flatten the power distribution.

2. Results

For the single rod withdrawal event, two cases have been considered as follows:

(a) If the reactor is in the manual control mode, continuous withdrawal of a single RCCA results in both an increase in core power and coolant temperature, and an increase in the local hot channel factor in the area of the withdrawing RCCA. In terms of the overall system response, this case is similar to those presented in Subsection 15.4.2; however, the increased local power peaking in the area of the withdrawn RCCA may result 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 limit 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 limit value is 5 percent.

(b) If the reactor is in the automatic control mode, the multiple failures that result in the withdrawal of a single RCCA will result in the immobility of the other RCCAs in the controlling bank. The transient will then proceed in the same manner as Case (a) described above.

For such cases as above, a reactor trip will ultimately ensue, although not sufficiently fast in all instances to prevent a minimum DNBR in the core of less than the limit value. Following reactor trip, normal shutdown procedures are followed.

15.4.3.3 Radiological Consequences

No radiological consequences have been calculated for these postulated accidents since no significant fuel or clad damage is predicted. The case of the accidental withdrawal of a single RCCA has an upper limit potential of some clad damage; however, the radiological releases and offsite doses are bounded by the results of Subsection 15.4.8.3 (radiological consequences for the spectrum of rod ejection accidents).

15.4.3.4 Conclusions

For cases of dropped RCCAs (including partially dropped RCCAs) or dropped banks, the DNBR remains above the limit value and core damage does not occur.

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

For the case of the accidental withdrawal of a single RCCA, with the reactor in the automatic or manual control mode and initially operating at full power with bank D at the insertion limit, an upper bound of the number of fuel rods experiencing DNB is 5 percent of the total fuel rods in the core.

15.4.4 Startup of an Inactive Reactor Coolant Pump at an Incorrect Temperature

15.4.4.1 Identification of Causes and Accident Description

If the plant were allowed 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, and assuming the

secondary side of the steam generator in the inactive loop is not isolated, there is a temperature drop across the steam generator in the inactive loop and, with the reverse flow, the hot leg temperature of the inactive loop is lower than the reactor core inlet temperature.

Starting of an idle reactor coolant pump without bringing the inactive loop hot leg temperature close to the core inlet temperature would result in the injection of cold water into the core, which would cause a reactivity insertion and subsequent power increase.

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

Should the startup of an inactive reactor coolant pump accident occur, the transient will be terminated automatically by a reactor trip on low coolant loop flow when the power range neutron flux (two out of four channels) exceeds the P-8 setpoint, which has been previously reset for three loop operation.

15.4.4.2 Analysis of Effects and Consequences

Three loop operation at Seabrook Station is prohibited by technical specifications. Therefore this event was not analyzed.

15.4.5 A Malfunction or Failure of the Flow Controller in a BWR Loop That Results in an Increased Reactor Coolant Flow Rate

Not applicable to Seabrook.

15.4.6 Chemical and Volume Control System Malfunction that Results in a Decrease in the Boron Concentration in the Reactor Coolant

15.4.6.1 Identification of Causes and Accident Description

Reactivity can be added to the core by feeding primary grade water into the RCS via the CVCS. Boron dilution is a manual operation under strict administrative controls with procedures calling for a limit on the rate and duration of dilution. A boric acid blend system is provided to permit the operator to match the boron concentration of reactor coolant makeup water during normal charging to that in the RCS. The 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 the boric acid and primary grade water on the control board. The CVCS is designed to limit, even under various postulated failure modes, the potential rate of dilution to a value which, after indication through alarms and instrumentation, provides the operator sufficient time to correct the situation in a safe and orderly manner.

The inadvertent opening of the Reactor Makeup Water (RMW) control valve in conjunction with a failure in the blend system permitting 0 ppm water to flow from the discharge of a single RMW pump to the charging pump suction is considered the limiting ANS Condition II boron dilution event for all modes of

operation. In order for makeup water to be added to the RCS at pressure, at least one charging pump must be running in addition to an RMW pump.

Information on the status of the RMW is continuously available to the operator. Lights are provided on the control board to indicate the operating condition of the pumps in the CVCS. Alarms are actuated to warn the operator if boric acid or demineralized water flowrates deviate from preset values as a result of system malfunction.

The inadvertent dilution from this source can readily be terminated by closing the reactor makeup control valve or stopping the RMW pump.

The rate of unborated makeup water addition to the RCS for this worst-case scenario is limited to the discharge flow capacity of a single RMW pump to the CVCS boric acid blender (150 gpm).

An additional source of unborated water which can dilute the reactor coolant is the Boron Thermal Regeneration System (BTRS). Borated RCS water is depleted of boron as it passes through the BTRS.

The BTRS is capable of supplying diluent at a rate comparable to that of one RMW pump. However, water from the BTRS is passed to the CVCS Volume Control Tank (VCT) where it mixes with water maintained at or nearly equal to the RCS boron concentration. Because of the size of the VCT and the mixing of BTRS diluent with water in the VCT, inadvertent operation of the BTRS is capable of creating only a mild boron dilution transient, which is bounded by the limiting scenario discussed above. The BTRS is excluded as a source of unborated water during refueling, cold shutdown, and hot shutdown since Technical Specifications require the BTRS be rendered inoperable in these modes.

Regardless of the cause of a dilution event, numerous alarms and indications including a shutdown monitor system alarm will alert the operator to a potential loss of shutdown margin. The Shutdown Monitor System augments the source range nuclear instrumentation by monitoring for statistically significant increases in the excore neutron flux, as an indication of a potential return to criticality. Specifically, when the neutron count rate increases by more than a preset ratio an alarm is generated. Further description of the Shutdown Monitor System is provided in Section 7.6.11.

The boron dilution event is classified as an ANS Condition II incident (a fault of moderate frequency) as defined in Subsection 15.0.1.

15.4.6.2 Method of Analysis

The method of calculating the time to lose all shutdown margin is briefly described below. If a boron mass balance is formed, the following relationship can be derived:

$$C(t) = C_o e^{-\alpha t}$$

$$\alpha = Q\ \rho_{DIL}/(7.481\ V\rho_{RCS})$$

where:

$C(t)$ = time-dependent core boron concentration, ppm

C_o = initial core boron concentration, ppm

t = time after event initiation, minutes

Q = constant diluent volumetric flow rate, gpm

ρ_{DIL} = diluent density at 70°F, 62.3 lb/ft^3

ρ_{RCS} = density of RCS fluid, lb/ft^3

V = effective mixing volume within the RCS, ft^3

In this equation, the diluent (injection flow) is assumed to have zero boron content.

The above equation may be rearranged to calculate the time to loss of shutdown margin.

$$tc = \frac{1}{\alpha} \ln (C_o/C_c)$$

where:

C_c = critical boron concentration, ppm

The initial core boron concentrations assumed in the analysis are the minimum concentrations required to meet Technical Specification requirements for boron concentration or minimum shutdown margin applicable to each mode of operation.

Uncertainties are applied so as to increase the value of C_c and conservatively minimize the ratio C_o/C_c.

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

a. Dilution during Refueling

The following conditions are assumed for an uncontrolled boron dilution during refueling:

1. The maximum boron concentration to lose all shutdown margin conservatively bounds the condition of All Rods In (ARI) less the highest worth assembly, most reactive time in life, no xenon, cold (70°F ≤ T ≤ 140°F).

2. The boron concentration of all filled portions of the Reactor Coolant System and the refueling canal shall be maintained uniform and sufficient to ensure that the more restrictive of the following reactivity conditions is met:

   a. A K_{eff} of 0.95 or less; or

   b. A boron concentration of greater than or equal to 2,000 ppm.

3. Dilution flow is assumed to be 150 gpm.

4. Mixing of the reactor coolant is accomplished by the operation of at least one residual heat removal pump.

5. A minimum water volume (3,395 ft^3) in the Reactor Coolant System is used. This is the minimum volume of the RCS for residual heat removal system operation. The water in the reactor vessel is assumed to be drained so that the nozzles are half-filled. The total volume includes the reactor vessel up to the nozzle centerline, one hot leg half filled up to the RHR connection, two cold legs half filled up to the RHR connections, and the active volume of one RHR loop.

6. The density of RCS fluid is assumed to be 62.3 lb/ft^3 (cold).

b. <u>Dilution during Cold Shutdown (with Filled Loops)</u>

1. The maximum boron concentration required to lose all shutdown margin conservatively bounds the condition of All Rods In (ARI) less the highest worth assembly, most reactive time in life, no xenon, and cold ($70°F \leq T \leq 200°F$).

2. Technical Specifications and the Core Operating Limits Report require a minimum shutdown margin. The assumed initial boron concentration is the minimum boron concentration required to meet this shutdown margin.

3. The assumed dilution flowrate is 150 gpm.

4. Mixing of the reactor coolant is accomplished by the operation of one residual heat removal pump.

5. A minimum water volume of 3,992 ft^3 in the Reactor Coolant System is used. The total volume includes the reactor vessel excluding the upper head region, one hot leg up to the RHR connection, two cold legs up to the RHR connections, and the active volume of the smaller RHR loop.

6. The density of RCS fluid is assumed to be 62.3 lb/ft^3 (cold).

c. Dilution during Cold Shutdown (with Drained Loops)

1. Technical Specifications require that 2000 ppm be maintained in this condition. The initial boron concentration is assumed to be 2000 ppm.

2. The maximum boron concentration to lose all shutdown margin is identical to the case with filled loops.

3. The assumed dilution flowrate is 150 gpm.

4. Mixing of the reactor coolant is accomplished by operation of at least one residual heat removal pump.

5. A minimum water volume of 3,395 ft^3 in the Reactor Coolant System is used. This is the minimum volume of the RCS for Residual Heat Removal System operation as described under "Dilution During Refueling."

6. The density of RCS fluid is assumed to be 62.3 lb/ft^3 (cold).

d. Dilution during Hot Shutdown

1. The maximum boron concentration to lose all shutdown margin conservatively bounds the condition of zero power, ARI less the highest worth rod, most reactive time in life, no xenon, hot ($200°F \leq T \leq 350°F$).

2. Technical Specifications and the Core Operating Limits Report require a minimum shutdown margin. The assumed initial boron concentration is the minimum boron concentration required to meet this shutdown margin.

3. The assumed dilution flowrate is 150 gpm.

4. A minimum water volume of 3,992 ft^3 in the RCS is used. This is the minimum volume of the RCS for RHR operation as described under "Dilution During Cold Shutdown (with Filled Loops)."

5. The density of RCS fluid is assumed to be 60.1 lb/ft^3 (200°F, saturated condition conservatively used. RCS is maintained 50°F subcooled).

e. Dilution during Hot Standby

For the bounding case in this operational mode, the reactor is assumed to be initially subcritical with all rods in less the highest worth rod and with the Technical Specification and the Core Operating Limits Report requirement for shutdown margin met using soluble boron.

1. The maximum boron concentration to lose all shutdown margin conservatively bounds the condition of zero power, ARI less the highest worth rod, most reactive time in life, no xenon, hot ($350°F \leq T \leq 557°F$).

2. Technical Specifications and the Core Operating Limits Report require a minimum shutdown margin. The assumed initial boron concentration is the minimum boron concentration required to meet this shutdown margin.

3. Dilution flow is assumed to be limited to the capacity of one RMW pump (150 gpm).

4. A minimum water volume (8,855 ft^3) in the Reactor Coolant System is used. This volume corresponds to the active volume of the Reactor Coolant System, minus the pressurizer and surge line volumes.

5. Mixing of the reactor coolant is accomplished by operation of the reactor coolant pumps.

6. The density of the RCS fluid is assumed to be 55.6 lb/ft^3 (350°F and 400 psia).

f. Dilution During Startup (Mode 2)

The following conditions are assumed for an uncontrolled boron dilution during startup.

1. The dilution flow rate is assumed to be limited to the capacity of one RMW pump with the reactor coolant system at pressure (approximately 150 gpm).

2. A minimum water volume (9873.5 ft^3) in the RCS is used. This is a conservative estimate of the active volume of the RCS minus the pressurizer volume, and accounts for 8% steam generator tube plugging.

3. The initial condition in the analysis is assumed to be during the dilution, corresponding to a critical, hot zero power condition with the control rods at the rod insertion limits. A reactor trip on source range high neutron flux is assumed to occur at this condition, alerting the operator to the dilution in progress. The maximum boron concentration at which the reactor will again attain criticality with all rods inserted less the most reactive RCCA stuck out of the core is taken as 1750 ppm. The minimum change from this condition to the initial condition at the rod insertion limits is taken as 500 ppm.

g. Dilution During Power Operation (Mode 1)

The following conditions are assumed for an uncontrolled boron dilution during power operation.

1. During power operation, the plant may be operated two ways: under manual operator control or under automatic rod control. While in manual or automatic rod control, the dilution flow rate is assumed to be the maximum flow capacity of a single RMW pump or 150 gpm.

2. A minimum water volume (9873.5 ft^3) in the RCS is used. This is a conservative estimate of the active volume of the RCS minus the pressurizer volume, and accounts for 8% steam generator tube plugging.

3. For the case of manual reactor control, the initial condition in the analysis is assumed to correspond to a critical, hot full power condition with the control rods at the rod insertion limits. Dilution causes the power and RCS temperature to rise, resulting in a reactor trip on overtemperature ΔT, alerting the operator to the dilution in progress. The maximum boron concentration at which the reactor will again attain criticality at hot zero power with all rods inserted less the most reactive RCCA stuck out of the core is taken as 1750 ppm. The minimum change from this condition to the initial condition of hot full power at the rod insertion limits is taken as 500 ppm.

   For the case of automatic reactor control, the initial condition in the analysis is assumed to correspond to a critical, hot full power condition with the control rods at the rod insertion limits. The operator will be alerted to the dilution in progress by the low-low rod insertion limit alarm. The maximum boron concentration at which the reactor will attain criticality at hot zero power with all rods inserted less the most reactive RCCA stuck out of the core

is taken as 1750 ppm. The minimum change from this condition to the initial condition of hot full power at the rod insertion limits is taken as 500 ppm.

15.4.6.3 Results of Analysis

a. Dilution during Refueling

For dilution during refueling, the minimum time required for the shutdown margin to be lost after a Shutdown Monitor System alarm or other alarm will allow at least a half hour for the operator to prevent a loss of all shutdown margin.

b. Dilution during Cold Shutdown (with Filled Loops)

For dilution during cold shutdown with filled loops, the minimum time required for the shutdown margin to be lost after a Shutdown Monitor System alarm or other alarm will allow at least 15 minutes for the operator to prevent a loss of all shutdown margin.

c. Dilution during Cold Shutdown (with Drained Loops)

For dilution during cold shutdown with drained loops, the minimum time required for the shutdown margin to be lost after a Shutdown Monitor System alarm or other alarm will allow at least 15 minutes for the operator to prevent a loss of all shutdown margin.

d. Dilution during Hot Shutdown

For dilution during hot shutdown, the minimum time required to lose all shutdown margin after a Shutdown Monitor System alarm or other alarm will allow at least 15 minutes for the operator to prevent a loss of all shutdown margin.

e. Dilution during Hot Standby

For dilution during hot standby, the minimum time required to lose all shutdown margin after a Shutdown Monitor System alarm or other alarm will allow at least 15 minutes for the operator to prevent a loss of all shutdown margin.

f. Dilution During Startup (Mode 2)

In the event of an unplanned approach to criticality or dilution during power escalation while in the startup mode, the operator has at least 15 minutes following reactor trip on source range high neutron flux until the loss of shutdown margin.

g. Dilution During Power Operation (Mode 1)

During full power operation with the reactor in manual control, the operator has at least 15 minutes following reactor trip on overtemperature ΔT until the loss of shutdown margin. The maximum reactivity insertion rate resulting from the boron dilution is 1.7 pcm/sec.

During full power operation with the reactor in automatic control, the operator has at least 15 minutes following the low-low rod-insertion limit alarm until the loss of shutdown margin.

15.4.6.4 Radiological Consequences

No radiological consequences have been calculated for this postulated event since no fuel or clad damage is predicted.

15.4.6.5 Conclusions

The results presented above show that for all the operating modes, there is adequate time for the operator to terminate an unplanned boron dilution event prior to loss of all shutdown margin. Following termination of the dilution flow, the reactor will be in a stable condition with no fuel damage. The calculated sequence of events for the limiting cases described above is shown in Table 15.4-1.

15.4.7 Inadvertent Loading and Operation of a Fuel Assembly in an Improper Position

15.4.7.1 Identification of Causes and Accident Description

Fuel and core loading errors, such as can arise from the inadvertent loading of one or more fuel assemblies into improper positions, loading a fuel rod during manufacture with one or more pellets of the wrong enrichment or the loading of a full fuel assembly during manufacture with pellets of the wrong enrichment, will lead to increased heat fluxes if the error results in placing fuel in core positions calling for fuel of lesser enrichment.

Any error in enrichment, beyond the normal manufacturing tolerances, can cause power shapes which are more peaked than those calculated with the correct enrichments. There is a 5 percent uncertainty margin included in the design value of power peaking factor assumed in the analysis of Condition I and Condition II transients. Successful completion of the reload startup physics tests provides assurance that the plant can be operated as designed.

To reduce the probability of core loading errors, strict administrative controls are placed on the entire core loading sequence. Then, using the core loading patterns from the just completed cycle and the ensuing cycle along with the spent fuel pool map, a core loading sequence is developed. The core loading sequence provides the step-by-step instructions necessary to end up

with the desired core configuration. Lastly, as part of the reload process, fuel assembly identification numbers and component types are verified during the reload process. This positive identification along with the multiple independent verification programs utilized during the placement in the reactor vessel provides additional assurance that the fuel is loaded in accordance with the core loading pattern.

The power distortion due to any combination of misplaced fuel assemblies would significantly raise peaking factors and would be readily observable with fixed incore detectors located in about one third of the fuel assemblies in the core. Each fixed incore detector also includes a core exit thermocouple which would also indicate any abnormally high coolant enthalpy rise. Incore flux measurements are taken during the startup subsequent to every refueling operation.

This event is classified as an ANS Condition III incident (an infrequent incident) as defined in Subsection 15.0.1.

15.4.7.2 Radiological Consequences

Any localized fuel or clad damage that may result for this postulated accident or from enrichment errors is assumed to result in radiological consequences which are less severe than those presented in Subsection 15.4.8.3 (radiological consequences for the spectrum of rod ejection accidents).

15.4.7.3 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 and perhaps the immediately adjacent pins.

Fuel assembly loading errors are prevented by administrative procedures implemented during core loading. In the unlikely event that a loading error occurs, the resulting power distribution effects will either be readily detected by the reload startup test program or will cause a sufficiently small perturbation to be acceptable within the uncertainties allowed between nominal and design power shapes.

15.4.8 Spectrum of Rod Cluster Control Assembly Ejection Accidents

15.4.8.1 Identification of Causes and Accident Description

This accident is defined as the mechanical failure of a control rod mechanism pressure housing resulting in the ejection of a RCCA and drive shaft. The consequence of this mechanical failure is a rapid positive reactivity insertion together with an adverse core power distribution, possibly leading

to localized fuel rod damage.

a. Design Precautions and Protection

Certain features in the Seabrook pressurized water reactor are intended to preclude the possibility of a rod ejection accident, or to limit the consequences if the accident were to occur. These include a sound, conservative mechanical design of the rod housings, together with a thorough quality control (testing) program during assembly, and a nuclear design which lessens the potential ejection worth of RCCAs, and minimizes the number of assemblies inserted at high power levels.

1. Mechanical Design

The mechanical design is discussed in Section 4.6. Mechanical design and quality control procedures intended to preclude the possibility of a RCCA drive mechanism housing failure are listed below:

(a) Each full length control rod drive mechanism housing is completely assembled and shop tested at 4100 psi.

(b) The mechanism housings are individually hydrotested after they are attached to the head adapters in the reactor vessel head, and checked during the hydrotest of the completed Reactor Coolant System.

(c) Stress levels in the mechanism are not affected by anticipated system transients at power, or by the thermal movement of the coolant loops. Moments induced by the design earthquake can be accepted within the allowable primary working stress range specified by the American Society of Mechanical Engineers (ASME) Code, Section III, for Class I components.

(d) The latch mechanism housing and rod travel housing are each a single length of forged Type 304 stainless steel. This material exhibits excellent notch toughness at all temperatures which will be encountered.

A significant margin of strength in the elastic range together with the large energy absorption capability in the plastic range gives additional assurance that gross failure of the housing will not occur. The joints between the latch mechanism housing and head adapter, and between the latch mechanism housing and rod travel housing, are threaded joints reinforced by canopy-type rod welds. Administrative

regulations require periodic inspections of these (and other) welds.

2. Nuclear Design

Even if a rupture of a RCCA drive mechanism housing is postulated, the operation of a plant utilizing chemical shim is such that the severity of an ejected RCCA is inherently limited. In general, the reactor is operated with the RCCAs inserted only far enough to permit load follow. Reactivity changes caused by core depletion and xenon transients are compensated by boron concentration changes. Further, the location and grouping of control RCCA banks are selected during the nuclear design to lessen the severity of an RCCA ejection accident. Therefore, should an RCCA be ejected from its normal position during full power operation, only a minor reactivity excursion, at worst, could be expected to occur.

However, it may be occasionally desirable to operate with larger than normal insertions. For this reason, a rod insertion limit is defined as a function of power level. Operation with the RCCAs above this limit guarantees adequate shutdown capability and acceptable power distribution. The position of all RCCAs is continuously indicated in the control room. An alarm will occur if a bank of RCCAs approaches its insertion limit or if one RCCA deviates from its bank.

Operating instructions require boration at the low insertion limit level alarm and emergency boration at the low-low level alarm.

3. Reactor Protection

The reactor protection in the event of a rod ejection accident has been described in Reference 10. The protection for this accident is provided by high neutron flux trip (high and low setting) and high rate of neutron flux increase trip. These protection functions are described in detail in Section 7.2.

4. Effects on Adjacent Housings

Disregarding the remote possibility of the occurrence of an RCCA mechanism housing failure, investigations have shown that failure of a housing due to either longitudinal or circumferential cracking would not cause damage to adjacent housings.

5. Effects of Rod Travel Housing Longitudinal Failures

If a longitudinal failure of the rod travel housing should occur, the region of the position indicator assembly opposite the break would be stressed by the reactor coolant pressure of 2250 psia. The most probable leakage path would be provided by the radial deformation of the position indicator coil assembly, resulting in the growth of axial flow passages between the rod travel housing and the hollow tube along which the coil assemblies are mounted.

If failure of the position indicator coil assembly should occur, the resulting free radial jet from the failed housing could cause it to bend and contact adjacent rod housings. If the adjacent housings were on the periphery, they might bend outward from their bases. The housing material is quite ductile; plastic hinging without cracking would be expected. Housings adjacent to a failed housing, in locations other than the periphery, would not bend because of the rigidity of multiple adjacent housings.

6. Effects of Rod Travel Housing Circumferential Failures

If circumferential failure of a rod travel housing should occur, the broken-off section of the housing would be ejected vertically because the driving force is vertical and the position indicator coil assembly and the drive shaft would tend to guide the broken-off piece upwards during its travel. Travel is limited by the missile shield, thereby limiting the projectile acceleration. When the projectile reached the missile shield, it would partially penetrate the shield and dissipate its kinetic energy. The water jet from the break would continue to push the broken-off piece against the missile shield.

If the broken-off piece of the rod travel housing was short enough to clear the break when fully ejected, it would rebound after impact with the missile shield. The top end plates of the position indicator coil assemblies would prevent the broken piece from directly hitting the rod travel housing of a second drive mechanism. Even if a direct hit by the rebounding piece was to occur, the low kinetic energy of the rebounding projectile would not cause significant damage.

7. Possible Consequences

From the above discussion, the probability of damage to an adjacent housing must be considered remote. However, even if damage is postulated, it would not be expected to lead to

a more severe transient since RCCAs are inserted in the core in symmetric patterns, and control rods immediately adjacent to worst ejected rods are not in the core when the reactor is critical. Damage to an adjacent housing could, at worst, cause the RCCA not to fall on receiving a trip signal; however, this is already taken into account in the analysis by assuming a stuck rod adjacent to the ejected rod.

8. Summary

The considerations given above lead to the conclusion that failure of a control rod housing, due either to longitudinal or circumferential cracking, would not cause damage to adjacent housings that would increase severity of the initial accident.

b. Limiting Criteria

This event is classified as an ANS Condition IV incident. See Subsection 15.0.1 for a discussion of ANS classifications. Due to the extremely low probability of a RCCA ejection accident, some fuel damage could be considered an acceptable consequence.

Comprehensive studies of the threshold of fuel failure and of the threshold of significant conversion of the fuel thermal energy to mechanical energy, have been carried out as part of the SPERT project by the Idaho Nuclear Corporation (Reference 11).

Extensive tests of UO_2 zirconium clad fuel rods representative of those in pressurized water reactor-type cores have demonstrated failure thresholds in the range of 240 to 257 cal/gm. However, other rods of a slightly different design have exhibited failures as low as 225 cal/gm.

These results differ significantly from the TREAT (Reference 12) results, which indicated a failure threshold of 280 cal/gm. Limited results have indicated that this threshold decreases by about 10 percent with fuel burnup. The clad failure mechanism appears to be melting for zero burnup rods and brittle fracture for irradiated rods. Also important is the conversion ratio of thermal to mechanical energy. This ratio becomes marginally detectable above 300 cal/gm for unirradiated rods and 200 cal/gm for irradiated rods; catastrophic failure, (large fuel dispersal, large pressure rise) even for irradiated rods, did not occur below 300 cal/gm.

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

1. Average fuel pellet enthalpy at the hot spot below 225 cal/gm for unirradiated fuel and 200 cal/gm for irradiated fuel.

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

3. Fuel melting will be limited to less than 10 percent of the fuel volume at the hot spot even if the average fuel pellet enthalpy is below the limits of criterion 1 above.

It should be noted that the original FSAR included an additional criterion that the average clad temperature at the hot spot must remain below 2700°F. The elimination of this criterion as a basis for evaluating the RCCA ejection accident results is consistent with the revised Westinghouse acceptance criteria for this event[13].

15.4.8.2 Analysis of Effects and Consequences

a. Method of Analysis

The calculation of the RCCA ejection transient is performed in two stages, first an average core channel calculation and then a hot region calculation. The average core calculation is performed using spatial neutron kinetics methods to determine the average power generation with time including the various total core feedback effects; i.e., Doppler reactivity and moderator reactivity. Enthalpy and temperature transients in the hot spot are then determined by multiplying the average core energy generation by the hot channel factor and performing a fuel rod transient heat transfer calculation. The power distribution calculated without feedback is pessimistically assumed to persist throughout the transient. A detailed discussion of the method of analysis can be found in Reference 10.

b. Average Core Analysis

The spatial kinetics computer code, TWINKLE[3], is used for the average core transient analysis. This code solves the two-group neutron diffusion theory kinetic equation in one, two, or three spatial dimensions (rectangular coordinates) for six delayed neutron groups and up to 2000 spatial points. The computer code includes a detailed multiregion, transient fuel clad coolant heat transfer model for calculation of pointwise Doppler and moderator feedback effects. In this analysis, the code is used as a one-dimensional axial kinetics code since it allows a more

realistic representation of the spatial effects of axial moderator feedback and RCCA movement. However, since the radial dimension is missing, it is still necessary to employ very conservative methods (described below) of calculating the ejected rod worth and hot channel factor.

c. Hot Spot Analysis

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

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

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

d. System Overpressure Analysis

Because safety limits for fuel damage specified earlier are not exceeded, there is little likelihood of fuel dispersal into the coolant. The pressure surge may, therefore, be calculated on the basis of conventional heat transfer from the fuel and prompt heat generation in the coolant.

The pressure surge is calculated by first performing the fuel heat transfer calculation to determine the average and hot spot heat flux versus time. Using this heat flux data, a VIPRE calculation is conducted to determine the volume surge. Finally, the volume surge is simulated in the LOFTRAN[1] computer code. This code calculates the pressure transient taking into account fluid transport in the RCS and heat transfer to the steam generators. No credit is taken for the possible pressure reduction caused by the assumed failure of the control rod pressure housing.

Calculation of Basic Parameters

Input parameters for the analysis are conservatively selected on the basis of values calculated for this type of core. The more important parameters are discussed below. Table 15.4-2 presents the parameters used in this analysis

a. Ejected Rod Worths and Hot Channel Factors

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

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

Power distribution before and after ejection for a "worst case" can be found in Reference 10. During plant startup physics testing, ejected rod worths and power distributions are measured in the zero and full power configurations and compared to values used in the analysis. Experience has shown that the ejected rod worth and power peaking factors are consistently overpredicted in the analysis.

b. Reactivity Feedback Weighting Factors

The largest temperature rises, and hence the largest reactivity feedbacks, occur in channels where the power is higher than average. Since the weight of a region is dependent on flux, these regions have high weights. This means that the reactivity feedback is larger than that indicated by a simple channel analysis. Physics calculations have been carried out for

temperature changes with a flat temperature distribution, and with a large number of axial and radial temperature distributions. Reactivity changes were compared and effective weighting factors determined. These weighting factors take the form of multipliers which, when applied to single channel feedbacks, correct them to effective whole core feedbacks for the appropriate flux shape. In this analysis, since a one dimensional (axial) spatial kinetics method is employed, axial weighting is not necessary if the initial condition is made to match the ejected rod configuration. In addition, no weighting is applied to the moderator feedback. A conservative radial weighting factor is applied to the transient fuel temperature to obtain an effective fuel temperature as a function of time accounting for the missing spatial dimension. These weighting factors have also been shown to be conservative compared to three-dimensional analysis.

c. Moderator and Doppler Coefficient

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

The Doppler reactivity defect is determined as a function of power level using a one-dimensional steady-state computer code with a Doppler weighting factor of 1.0. The Doppler weighting factor will increase under accident conditions, as discussed above.8

d. Delayed Neutron Fraction, βeff

Calculations of the effective delayed neutron fraction (βeff) typically yield values no less than 0.70% at beginning of life and 0.50% at end of life for the first cycle. The accident is sensitive to βeff if the ejected rod worth is equal to or greater than βeff as in zero power transients. In order to allow for future cycles, pessimistic estimates of 0.54% at beginning of cycle and 0.44% at end of cycle were used in the analysis.

e. Trip Reactivity Insertion

The trip reactivity insertion assumed is given in Table 15.4-2 and includes the effect of one stuck RCCA adjacent to the ejected rod. These values are reduced by the ejected rod reactivity. The shutdown reactivity was simulated by dropping a rod of the required worth into the core. The start of rod motion occurred 0.5 second after the high neutron flux trip point is reached. This delay is assumed to consist of 0.2 second for the instrument channel to produce a signal, 0.15 second for the trip breaker to open, and 0.15 second for the coil to release the rods. A curve

of trip rod insertion versus time was used which assumed that insertion to the dashpot does not occur until 2.4 seconds after the start of fall. The choice of such a conservative insertion rate means that there is over one second after the trip point is reached before significant shutdown reactivity is inserted into the core. This is a particularly important conservatism for hot full power accidents.

The minimum design shutdown margin available for this plant at hot zero power (HZP) may be reached only at end of life in the equilibrium cycle. This value includes an allowance for the worst stuck rod, an adverse xenon distribution, conservative Doppler and moderator defects, and an allowance for calculational uncertainties. Physics calculations have shown that the effect of two stuck RCCAs (one of which is the worst ejected rod) is to reduce the shutdown by about an additional 1%Δk. Therefore, following a reactor trip resulting from an RCCA ejection accident, the reactor will be subcritical when the core returns to HZP.

Depressurization calculations have been performed assuming the maximum possible size break (2.75 inch diameter) located in the reactor pressure vessel head. The results show a rapid pressure drop and a decrease in system water mass due to the break. The emergency core cooling system (ECCS) is actuated on low pressurizer pressure within 1 minute after the break. The RCS pressure continues to drop and reaches saturation (1100 to 1300 psi depending on the system temperature) in about 2 to 3 minutes. Due to the large thermal inertia of primary and secondary system, there has been no significant decrease in the RCS temperature below no-load by this time, and the depressurization itself has caused an increase in shutdown margin by about 0.2%Δk due to the pressure coefficient. The cooldown transient could not absorb the available shutdown margin until more than 10 minutes after the break. The addition of borated safety injection flow starting one minute after the break is much more than sufficient to ensure that the core remains subcritical during the cooldown.

f. Reactor Protection

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

g. Results

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

(1) Beginning of Cycle, Full Power

Control bank D was assumed to be inserted to its insertion limit. The worst ejected rod worth and hot channel factor were conservatively calculated to be 0.25%$\Delta\rho$ and 6.0 respectively. The maximum fuel stored energy was 167 cal/gm. The peak hot spot fuel center temperature reached melting, conservatively assumed at 4900°F. However, melting was restricted to less than 10% of the fuel pellet.

(2) Beginning of Cycle, Zero Power

For this condition, control bank D was assumed to be fully inserted and banks B and C were at their insertion limits. The worst ejected rod is located in control bank D and has a worth of 0.78%$\Delta\rho$ and a hot channel factor of 11.5. The maximum fuel stored energy was 137 cal/gm. The peak fuel center temperature was 3750°F.

(3) End of Cycle, Full Power

Control bank D was assumed to be inserted to its insertion limit. The ejected rod worth and hot channel factors were conservatively calculated to be 0.25%$\Delta\rho$ and 7.0 respectively. The maximum fuel stored energy was 143 cal/gm. The peak fuel center temperature was 4575°F.

(4) End of Cycle, Zero Power

The ejected rod worth and hot channel factor for this case were obtained assuming control bank D to be fully inserted and banks B and C at their insertion limits. The results were 0.85%$\Delta\rho$ and 26.0 respectively. The maximum fuel stored energy was 147 cal/gm. The peak fuel center temperature was 3878°F. The Doppler weighting factor for this case is significantly higher than that of the other cases due to the very large transient hot channel factor.

A summary of the cases presented above is given in Table 15.4-2. The nuclear power and hot spot fuel and clad temperature transients for the worst cases (beginning of life full power and end of life zero power) are presented in Figures 15.4-10 through 15.4-11. The calculated sequence of events for these worst case rod ejection accidents is presented in Table 15.4-1. For all cases, reactor trip occurs very early in the transient, after

which the nuclear power excursion is terminated. As discussed previously, the reactor will remain subcritical following reactor trip.

The ejection of an RCCA constitutes a break in the RCS, located in the reactor pressure vessel head. Following the RCCA ejection, the operator would follow the same emergency instructions as for any other LOCA to recover from the event.

h. Fission Product Release

It is assumed that fission products are released from the gaps of all rods entering DNB. In all cases considered, less than 10% of the rods entered DNB based on a generic analysis. Although limited fuel melting at the hot spot was predicted for the beginning-of-life full power case, melting is not expected since the analysis conservatively assumed that the hot spots before and after ejection were coincident.

i. Pressure Surge

A detailed calculation of the pressure surge for an ejection worth of one dollar at beginning of life, hot full power, indicates that the peak pressure does not exceed that which would cause stress to exceed the faulted condition stress limits[10]. Since the severity of the present analysis does not exceed the "worst case" analysis, the accident for this plant will not result in an excessive pressure rise or further damage to the RCS.

j. Lattice Deformations

A large temperature gradient will exist in the region of the hot spot. Since the fuel rods are free to move in the vertical direction, differential expansion between separate rods cannot produce distortion. However, the temperature gradients across individual rods may produce a differential expansion tending to bow the midpoint of the rods toward the hotter side of the rod. Calculations have indicated that this bowing would result in a negative reactivity effect at the hot spot since Westinghouse cores are under-moderated, and bowing will tend to increase the under moderation at the hot spot. Since the 17x17 fuel design is also under-moderated, the same effect would be observed. In practice, no significant bowing is anticipated, since the structural rigidity of the core is more than sufficient to withstand the forces produced. Boiling in the hot spot region would produce a net flow away from that region. However, the heat from the fuel is released to the water relatively slowly, and it is considered inconceivable that cross flow will be sufficient to produce significant lattice forces. Even if

massive and rapid boiling, sufficient to distort the lattice, is hypothetically postulated, the large void fraction in the hot spot region would produce a reduction in the total core moderator to fuel ratio, and a large reduction in this ratio at the hot spot. The net effect would, therefore, be a negative feedback.

It can be concluded that no conceivable mechanism exists for a net positive feedback resulting from lattice deformation. In fact, a small negative feedback may result. The effect is conservatively ignored in the analysis.

15.4.8.3 Radiological Consequences

a. Assumptions and Parameters

A realistic analysis and a conservative analysis are considered. The conservative analysis employs more pessimistic assumptions regarding fission product release and transport. The assumptions and parameters of the two analyses are presented in Table 15.4-3. Detailed assumptions which are not stated in Table 15.4-3 are discussed in this section. No credit for iodine removal by the Containment Spray System is assumed for either analysis.

1. Conservative Analysis

(a) Initial primary coolant activity prior to the accident corresponds to the Technical Specification limit with the pre-existing iodine spike (see Appendix 15B).

(b) Ten percent of the fuel rods are assumed to fail due to the accident, resulting in release of the radioactivity from the cladding gaps to the reactor coolant. The gap activity is shown by radionuclide in Table 15.0-6. A small fraction of the fuel rods are assumed to melt due to the accident, resulting in release of 0.25 percent of the noble gases and 0.125 percent of the halogens of the core activity into the primary coolant.

The above two releases from the damaged fuel, plus the activity initially in the primary coolant, represent the conservative source term of this accident and are presented in Table 15.4-4.

(c) For the containment leakage releases, the conservative source term in Table 15.4-4 is assumed to be released to the Containment. Fifty percent of the iodines released to the Containment are assumed to be

immediately removed by natural effects, including plate-out and settling. Table 15.4-5 lists the airborne activity inside the Containment. This airborne activity is assumed to be mixed instantaneously with the containment free air volume and available for release to the environment from containment leakage.

(d) Leakage from the containment structure passes directly to the containment enclosure emergency exhaust filters, without credit for mixing in the annulus between the primary and secondary containments. Table 15.4-6 lists the activity released to the environment by radionuclide from the containment enclosure emergency exhaust filters.

(e) For the secondary side releases, offsite power is assumed to be unavailable at the time of the accident. Atmospheric steam release is therefore required for plant cooldown. This condition persists for 8 hours. After this time, the reactor coolant system pressure would be reduced to less than the relief valve settings on the steam generator, and pressure and temperature would be further reduced by the Emergency Core Cooling System.

(f) For the release through the Secondary Coolant System, the conservative source term in Table 15.4-4 is assumed to be mixed instantaneously with the mass of primary coolant. Table 15.4-7 lists the activity concentration of the primary coolant available for leakage to the secondary system.

(g) Primary to secondary leakage is one gpm for the 8-hour period during which steam release to the atmosphere may be required. For this 8-hour period, 100 percent of the noble gases and 1 percent of the iodines in the Secondary Coolant System, from the primary coolant leakage, are assumed to be released with the steam to the atmosphere. Table 15.4-7 lists the activity released to the environment via the Secondary Coolant System.

2. Realistic Analysis

(a) Initial primary coolant activity prior to the accident corresponds to the equilibrium concentration at 0.12 percent clad defects. Table 11.1-1 lists the activity concentration by radionuclide at 0.12 percent clad defects.

(b) Ten percent of the realistic gap activity is assumed to be released to the coolant as a result of the accident. The realistic gap activity in the fuel cladding gaps is shown in Table 15.4-8.

This release from the damaged fuel, plus the primary coolant activity prior to the accident, represents the source term for the realistic analysis of this accident and is listed in Table 15.4-9.

(c) The realistic source term in Table 15.4-9 is released into the containment atmosphere. Fifty percent of the iodine activity released to the Containment is assumed to be immediately removed by natural effects, including plate-out and settling. Table 15.4-10 shows the airborne radioactivity inside the Containment by radionuclide.

(d) Leakage from the containment structure mixes uniformly with 50 percent of containment enclosure air volume. The containment enclosure air change rate of 2000 cfm is passed through particulate and charcoal filters before release to the environment. Table 15.4-11 lists the activity released to the environment from the containment structure leakage.

b. Results

The doses resulting from this accident for both the realistic and conservative analyses are shown in Table 15.4-12.

15.4.8.4 Conclusions

Even on a pessimistic basis, the analyses indicate that the described fuel and clad limits are not exceeded. It is concluded that there is no danger of sudden fuel dispersal into the coolant. Since the peak pressure does not exceed that which would cause stresses to exceed the faulted condition stress limits, it is concluded that there is no danger of further consequential damage to the RCS. The analyses have demonstrated the fission product release as a result of fuel rods entering DNB is limited to less than 10% of the fuel rods in the core.

15.4.9 Spectrum of Rod Drop Accidents in a BWR

Not applicable to Seabrook.

15.4.10 References

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9) WCAP-8424, Revision 1, "An Evaluation of Loss of Flow Accidents Caused by Power System Frequency Transients in Westinghouse PWRs," M. S. Baldwin, et al., June 1975

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